International Conference on Occupational Radiation Protection:

Enhancing the Protection of Workers — Gaps, Challenges and Developments

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Session 1: International recommendations and standards on occupational radiation protection: Recent changes and the challenges in their practical implementation

OCCUPATIONAL RADIATION PROTECTION IN INDIAN SCENARIO - REGULATORY INSTRUMENTS AND ENFORCEMENTS

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Abstract

A multi-pronged strategy is adopted by Atomic Energy Regulatory Board (AERB) to enforce the Instruments available to it from Legislative (Atomic Energy Act), Legal (Rules under the Act) and Regulatory (AERB Safety Codes, Standards, Guides, Safety Directives etc) frame work and also through various Apex Safety Review and Advisory Committees, for regulating the nuclear and radiation facilities under its purview. AERB while emphasizing optimization of protection at the work place also lays down safety standards, prescribes dose limits, dose constraints and investigation levels for better exposure control of occupational workers at an early stage. With these, AERB has a strong institutional approach for an effective occupational radiation protection towards a lower exposure regime in nuclear and radiation facilities.

1. INTRODUCTION

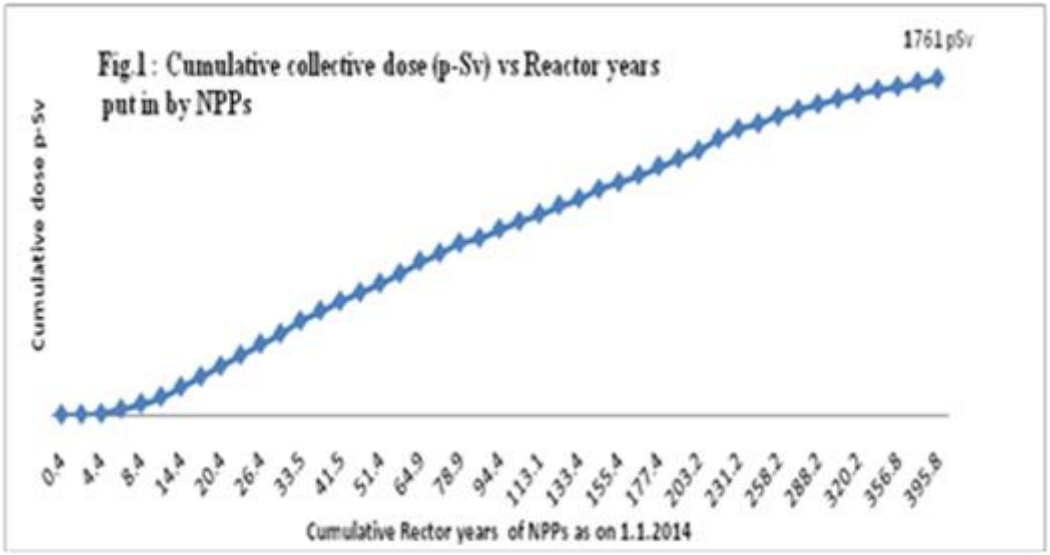
Atomic Energy Regulatory Board (AERB), India- established in 1983- derives legal powers from relevant Rules under the Atomic Energy Act, 1962 to carry out regulatory and safety functions related to nuclear power generation, nuclear fuel cycle facilities, medical, industrial and research applications of radiation in India. AERB emphasizes optimization of protection at the work place [1, 2]; prescribes annual effective dose limits and constraints; issues Safety codes, guides and directives [3, 4] and also prescribes Investigation Levels for exposure control at an early stage. AERB is supported by two Apex Safety Review Committees, namely SARCOP (Safety Review Committee for Operating Plants) for nuclear fuel cycle facilities and SARCAR (Safety Review Committee for Application of Radiation) for radiation facilities and other advisory and safety review committees, supplemented by periodic inspections by AERB teams. With these instruments, AERB has a strong institutional approach for an effective occupational radiation protection in facilities under its purview- as elaborated in this paper.

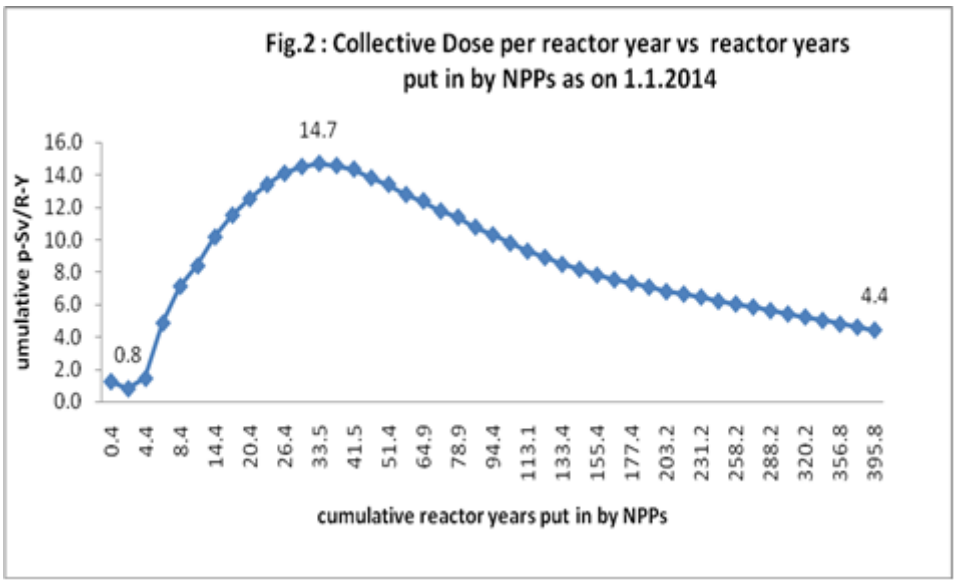
2. NUCLEAR POWER PLANTS (NPP)

In each NPP, a Radiological Safety Officer (RSO) - who is basically the Station Health physicist - is appointed by the Station Director, based on qualifications and experience as stipulated by AERB and his designation as RSO being approved by Chairman, AERB, to entrust with the responsibility of providing radiological surveillance and safety support functions of the Station. AERB on its part makes the Radiation Protection Review of each NPP through, (1) Multi-tier Safety review (Plant Level, AERB Safety committee and SARCOP), (2) Review of Station annual dose budget by SARCOP, (3) Review of Excessive Exposure Cases by a Standing Committee, (4) Periodic Regulatory Inspections (5) Special inspections by a task group constituted by SARCOP.

2.1. Towards lower occupational dose regime

At NPP, all jobs consuming 0.1person-mSv or more are accounted. Due to various regulatory enforcement efforts by AERB and implementation by NPPs, the collective dose and average dose in NPPs show a declining trend over the years [5]. As on 1.1.2014, the Indian Nuclear Power Programme had logged around 390 Reactor-years of commercial operating experience. A plot of cumulative collective dose (p-Sv) against the cumulative reactor years shows a near linear response (Fig.1). A normalized plot of collective dose per reactor year against reactor year (p-Sv/R-y) gives a positively skewed smooth curve (Fig. 2) expressing clear contours of decline in occupational doses**.** The curve shows ascension with pSv/R-y peaking at 14.7 around the year 1981 (with all along when TAPS 1, 2, and RAPS 1, 2 – the only NPPs existing at that time - engaged in gaining experience in addressing teething trouble- resulting in more collective dose- in operation and maintenance of India's early stage nuclear power generation) after which, the value declined steadily and substantially. Currently, the typical annual collective dose per Unit of Indian PHWRs stands at around 1. 0 p-Sv.

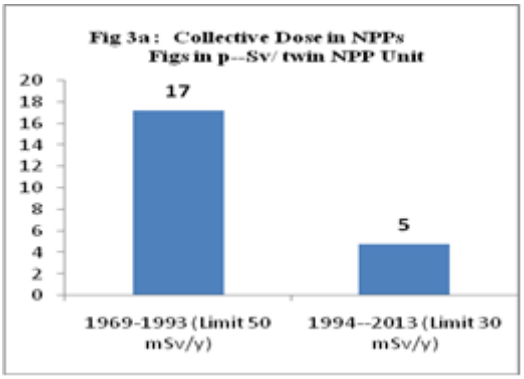


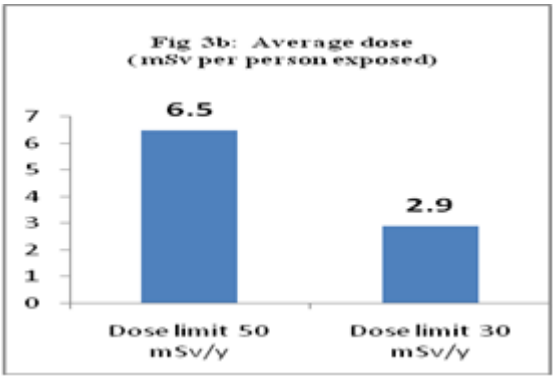


2.2. Effect of AERB Safety Directive on collective dose and average dose in NPP

AERB issued a Safety Directive by which the effective dose limit for occupational worker was brought down from 50 mSv to 30 mSv in any single year with a dose constraint of 20 mSv/y averaged over 5 years. This came in to full implementation in 1994. To be able to comply with the revised dose limits, each NPP took several efforts such as, (1) design modification wherever possible (2) Shielding or removal of system hot spots (3) Refining/ modifying work practices (4) Reinforcing radiation protection awareness through sustained training of radiation workers, and (5) ALARA review on daily basis etc.

The effect of these, are shown in Figs 3a and 3b, to reveal that, (A) The p-Sv/twin NPP unit came down from 17 (period 1969-1993) to 5 (period : 1994 to 2013) or, expressed differently, from 9 p-Sv/R-y to 3 p-Sv/R-y during the period under consideration and (B) The average dose, for the same reference period came down from 6.5 to 1.9 mSv per person per twin unit NPP



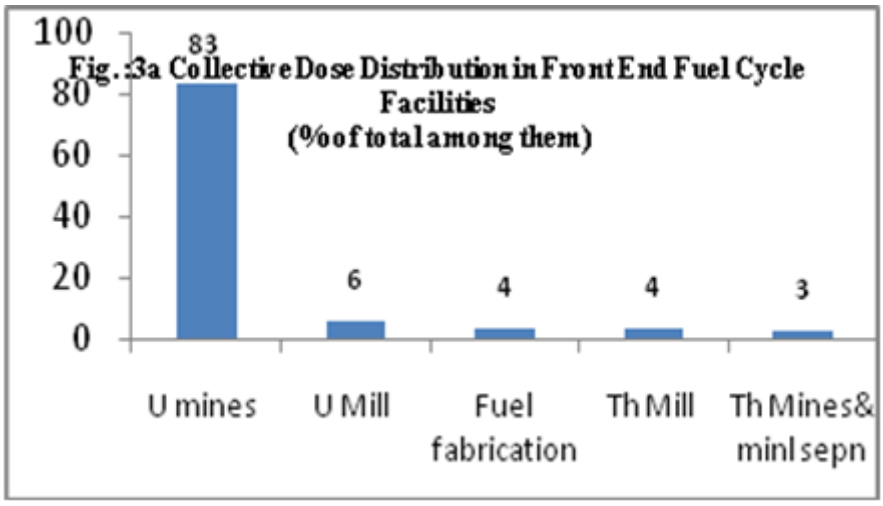


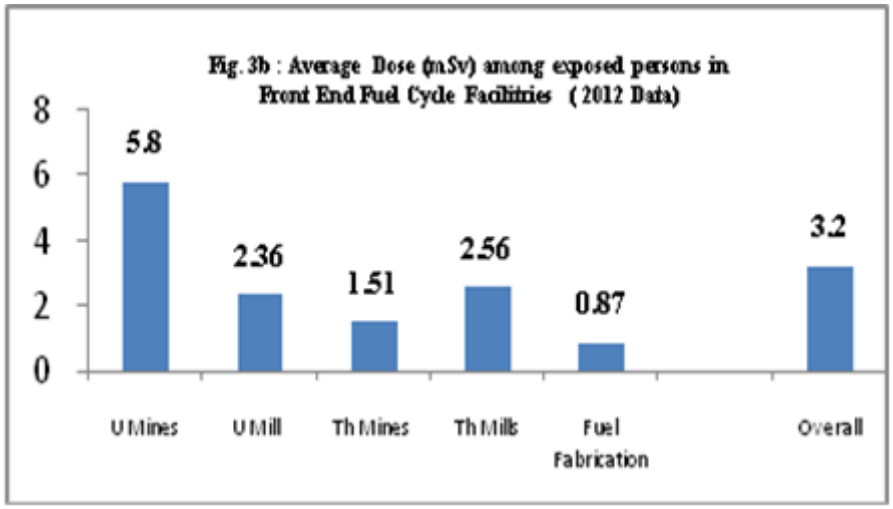
2.3. Challenges/ Issues for NPPs

(1) Tracking the dose history of a large number of temporary workers (constituting 50-60 % of dose share of the NPP) migrating from plant to plant; or carrying out work under different names. (2) Build up of hot spots in the system piping leading to enhanced exposures and (3) Collective dose during Bi annual shut down (BSD) consumes about 60-80 % of the station budget. Some of the jobs performed in BSD are repetitive in nature.

3. FRONT END FUEL CYCLE FACILITIES

The regulatory requirements of Radiological Safety in the Front End Fuel Cycle Facilities (Mining, Milling, Fuel fabrication) are stipulated by [7, 8]. SARCOP reviews the radiological safety issues of FEFCFs. External and Internal dose estimations are done by ambient dosimetry and computational methodology. Based on 2012 data, the % distributionof collective dose and the average dose among exposed persons in different units of FEFCF are shown in Fig 3a and 3b respectively.





3.1. Challenges/Issues in FEFCFs

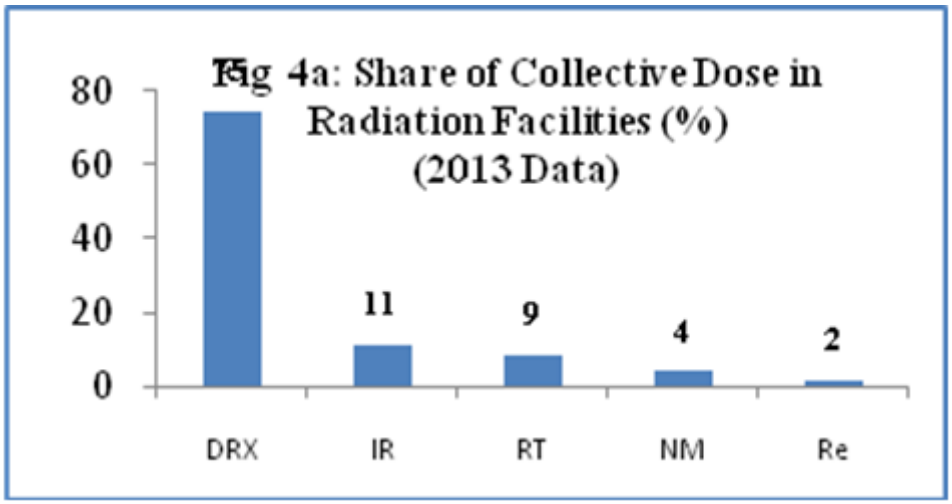
(1) Inadequacy of ventilation provision in Uranium mines leading to significant air activity levels. (2) Internal dose estimation in mines involving lot of uncertainties, (3) Maintaining air activity levels below 0.1 DAC at Fuel Fabrication facility and (4) Computation of occupational doses in Thorium mining and milling plants where radiation level at some work places is observed to be lower than the natural radiation background around the plant.

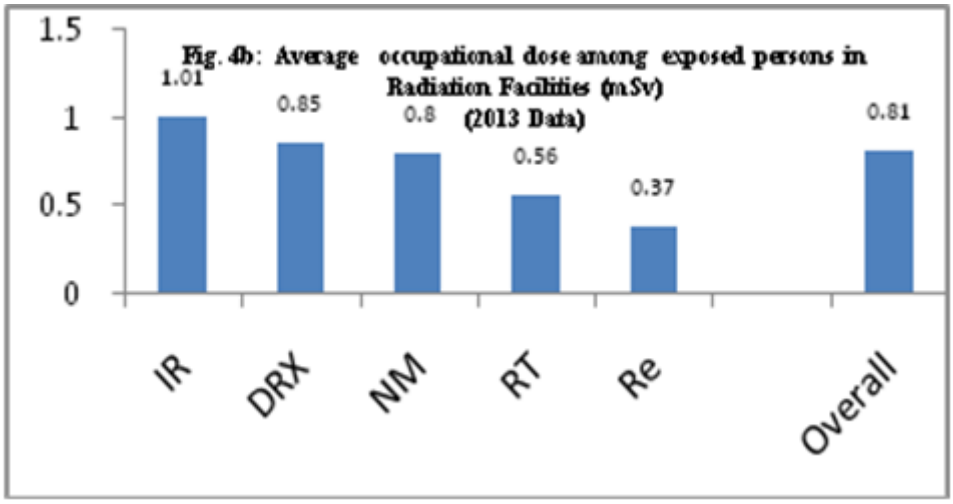
4. RADIATION FACILITIES (RFs)

The radiation facilities are broadly classified as Medical, Industrial and Research. Regulation of RFs is done through a graded approach to consenting as described by [8]. They are also put under specific inspection and surveillance programme depending upon the hazard potential of the facility put under the classification of IAEA Categorization of Sources [9].

4.1. Occupational Radiation Protection in RFs

The actual implementation of radiation safety in Radiation Facilities is carried out by Radiological Safety Officer (RSO) employed by the facility and approved by AERB. Based on 2013 data, the share of collective dose (%) and the average occupational dose among exposed persons are shown in Fig 4a and Fig.4b respectively. (Legend: DRX – Diagnostic radiology, RT- Radio therapy, NM—Nuclear Medicine, IR- Industrial Radiography).





4.2. Issues and challenges in radiation facilities

A large number of diagnostic radiology facilities still need to be registered with AERB with new facilities coming up at around 10-20% per year. (2) Awareness level on radiation protection among workers is low. (3) High average dose among personnel working

in interventional radiology. (4) Improper use of personnel monitoring devices leading to over-exposure due to non-technical factors and (5) Frequent instances of non-availability of RSOs in the radiation facilities.

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Optimización de la protección radiológica en el control de la exposición ocupacional, Colección de informes de seguridad N^{o} 21, IAEA, Vienna (2004).

[2] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Optimization and decision making in radiological protection, Publication 55, Ann. ICRP 20 (1), Pergamon Press, Oxford (1990).

OVERVIEW OF THE REGULATION ON OCCUPATIONAL RADIATION PROTECTION IN INDONESIA

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Abstract

Occupational radiation protection in Indonesia is regulated in general in a government regulation of 2007 on radiation safety and source security, which further elaborates in a chairman of BAPETEN regulation (CBR) of 2013 on radiation protection and safety. As explained in its elucidation, the government regulation was drawn up and harmonized with the BSS, IAEA Safety Series No. 115, 1996. The CBR, however, was issued after the IAEA revised the BSS by GSR Part 3 (Interim), 2011. While the CBR, as implementing regulation, should not deviate from government regulation, it apparently also accommodates contents from GSR Part 3 (Interim) as the revised publication of the BSS. This paper discusses how this situation results in a few shortcomings in this particular Chairman of BAPETEN regulation, including those related to the occupational radiation protection. Nevertheless, the CBR is basically well structured, and shall be implemented to ensure the safe and secure utilization of nuclear energy in Indonesia.

1. INTRODUCTION

The activities applying nuclear material and radiation sources in Indonesia has been carried out since more than fifty years ago. In 1958, the government of Indonesia established the Institute of Atomic Energy, which in 1964 became the National Atomic Energy Agency (BATAN) with the tasks to implement and regulate the utilization of nuclear material and radiation sources in Indonesia. To avoid conflict of interest between the functions of promoting and controlling, the Act No. 10 of 1997 on Nuclear Energy established an execution body (BATAN, but now became National Nuclear Energy Agency) and a regulation body (BAPETEN, Nuclear Energy Regulatory Agency). Since 1997, therefore, all nuclear related implementing regulations in Indonesia, including inspection and licensing, are stipulated and implemented by BAPETEN.

As one of the International Atomic Energy Agency (IAEA) member states, Indonesia has to apply the safety standards developed by the IAEA, including the standards regarding radiation protection and safety. This has been the case for all radiation protection and safety related regulations in Indonesia. The Government Regulation (GR) No. 33 of 2007 on the Safety of Ionizing Radiation and the Security of Radioactive Sources [1], the second grade of regulation after the Act No. 10 of 1997, applies the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (Safety Series No.115, 1996) [2] as its reference.

Chairman of BAPETEN Regulation (CBR) No. 4 of 2013 on Radiation Protection and Safety in the Utilization of Nuclear Energy [3] is the implementing regulation of the above mentioned government regulation. This CBR replaces the Chairman of BAPETEN Decree No. 01/Ka-BAPETEN/V/99 of 1999 which is revoked and is no longer valid. During the 14 years since 1999, certainly many have been progressed in radiation protection and safety issues, so that the content of the new CBR is much different from the previous one.

While BAPETEN regulation should not deviate from of its parent government regulation, there is a need to accommodate the contents of GSR Part 3 (Interim) of 2011 [4], which is a revised document for the BSS. This paper will review the contents of CBR regarding occupational radiation protection from the technical point of view, and compares them with the BSS as the referenced document for the GR No. 33 of 2007, and also with GSR Part 3 (Interim) as the revision document for BSS.

2. THE BODY OF CBR

CBR contains six chapters; Chapter I gives a range of definition of terms used, Chapter II lists the person in-charge in radiation safety, Chapter III describes the implementation of radiation protection requirements, Chapter IV on radiation protection and safety programme, Chapter V the transitional provisions, and Chapter VI, the concluding remarks. This regulation quite thick, around 178 pages, but as GSR Part 3 (Interim), most of the pages are only annexes containing the dose coefficients for intakes of radionuclides by workers and members of the public.

From the name of the chapters mentioned above, it can be seen that the main provision of the regulation is in chapter III, implementation of radiation protection requirements. This chapter describes in detail the provision for radiation protection – consists of justification, dose limitation and optimisation of protection and safety.

In the justification section, it describes that justification is based on the principle that benefit gain from practice outweigh the harm that it might cause on exposure. The factors to be taken into account in justification are other technologies whose risks are less than the risk from radiation exposure – economic and social, health and safety, and management of radioactive waste and decommissioning.

Dose limitation section outlines the dose limits that should not be exceeded and the activities that must be performed in keeping doses received below the limits. The latter consists of classification of areas, monitoring of radiation exposures and/or radioactive contamination in workplace, monitoring of environmental radioactivity outside the facility or installation, and monitoring of occupational doses.

In the section of optimisation of protection and safety, it describes that the optimisation shall be performed through the establishment of dose constraints and/or guidance levels for medical exposure. It further provides provision that dose constraints for workers are determined by the licensee with approval from the chairman of BAPETEN, and shall be reviewed during the lifetime operation of facility or installation.

3. DISCUSSIONS

The enactment of CBR is a major advancement for the nuclear regulatory framework in Indonesia, particularly in the field of radiation protection and safety. Previous regulation governing radiation protection and safety was a Decree of 1999, so it appears that it took 14 years for BAPETEN to revise regulation that is essential for the safety of workers, public and the environment.

From the contents, CBR seems to adopt as much as possible the major points of GSR Part 3 (Interim) which was published in 2011. The dose limit for the lens of the eye, for example, has been reduced to 20 mSv per year. The CBR has also adopted the dose coefficients for intakes of radionuclides by workers and members of the public contained in the GSR Part 3 (Interim). However, since this CBR is an implementing regulation of GR No. 33 of 2007, which refers to the 1996 BSS, it appears that the content of CBR is inconsistent in

some places.

Obvious example is in the sequence of the radiation protection requirements. Although GSR Part 3 (Interim) put the sequence as JOL (justification, optimization and dose limitation), CBR still put it as JLO (justification, dose limitation and optimization), the sequence used by the BSS. As there are philosophical reasons behind the sequence, the CBR seems lost this philosophical point in this particular requirement.

Another inconsistency of the CBR is the dose limit to the lens of the eye for occupational exposure of apprentices. Although GSR Part 3 (Interim) has reduced it to 20 mSv per year, but CBR still uses 50 mSv per year as recommended in the BSS. It is not clear the reasons of CBR to stick to the old limit, since the international community through the IAEA has agreed that the danger of cataracts which can occur in the eye lens was greater than previously thought, so the dose limit to this particular organ is reduced in the GSR Part 3 (Interim).

The CBR has also a different way in classifying working areas. Since the publication of the BSS, the licensee has some difficulties in defining the controlled and supervised areas in their facilities, as the quantitative value had been removed from the definition of both areas. The new CBR then comes with an idea to reinsert this quantitative value, so that controlled area is now defined based on the criteria of potential doses exceeding 3/10 of dose limit for occupational exposure and potential of contamination. Similarly, supervised area is defined by taking into account the potential doses exceeding the dose limit for public exposure but less than 3/10 of dose limit for occupational exposure, and free from contamination. This criterion, while recommended in the IAEA Safety Series No. 9 (1982 Edition) [5], no longer adopted by both the BSS and GSR Part 3 (Interim).

These inconsistencies, some parts remain referred to the BSS and some others referred to the new GSR Part 3 (Interim), can be regarded as shortcomings of this current Chairman of BAPETEN regulation on radiation protection and safety.

The hierarchy in the IAEA Safety Standards Series could perhaps be modelled in the preparation of radiation protection and safety regulation in Indonesia. The IAEA Safety Fundamentals can be regarded as the Act No. 10 of 1997 on Nuclear Energy, General Safety Requirements can be adopted to be Government Regulations, and Specific Safety Requirements, as well as Safety Guides, can then be adopted as Chairman of BAPETEN regulations.

As regard occupational radiation protection, however, by referring to occupational exposure section in the GSR Part 3 (Interim), some points can be made:

a. CBR has made regulatory body to establish and enforce requirements to ensure optimisation of protection and safety, and to enforce compliance with dose limits for occupational exposure;
b. CBR has made provision for monitoring and recording of occupational exposure,
c. CBR has drawn up responsibilities of the licensee, radiation worker and radiation protection officer to the radiation protection and safety as a whole, specifically to occupational radiation protection, in the facility;
d. Under radiation protection and safety program, CBR has made the licensee to designate the controlled and supervised areas, establish local rules, procedures and personal protective equipment, restrict access to working area, monitor occupational exposure and radioactive contamination, monitor environmental radioactivity in off-site facility or installation, monitor individual exposure of workers, health examination of workers, and the arrangement for emergency situation.
e. As to special arrangement, CBR does not regulate the need to inform female workers regarding the risk to the embryo and fetus due to exposure of pregnant woman, the risk of health effects for a breast-fed infant due to ingestion of radioactive substances and the

adaptation of working conditions for pregnant workers. However, CBR does forbid person under the age of 16 years to enter controlled area, and access to controlled area for person of the age of 16-18 years is only for the job training purposes.

4. CONCLUSIONS

Chairman of BAPETEN Regulation (CBR) No. 4 of 2013 is an implementing regulation of the Government Regulation (GR) No. 33 of 2007 which has been awaited for a long time. Apart from the few shortcomings presented in this paper, CBR is basically well structured, and remains a positive law which shall be met in implementing radiation protection and safety principles in the utilization of nuclear energy in Indonesia. To overcome the shortcomings, it is suggested that the contents of GSR Part 3 is used and adopted to revise GR No.33 of 2007, while BAPETEN only adopt IAEA Specific Safety Requirements and Safety Guides series to be its chairman regulations.

REFERENCES

[1] Government Regulation No. 33 of 2007 on the safety of ionizing radiation and the security of radioactive sources (In Indonesian).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, International Basic Safety Standards for protection against ionizing radiation and for the safety of radiation sources, Safety Series No.115. IAEA, Vienna (1996).
[3] Chairman of BAPETEN Regulation No. 4 of 2013 on the Radiation protection and safety in the utilization of nuclear energy (In Indonesian).
[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety of radiation sources: International Basic Safety Standards. Interim Edition. General Safety Requirements Part 3. No. GSR Part 3 (Interim). IAEA, Vienna (2011).
[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Basic safety standards for radiation protection (1982 Edition), Safety Series No.9. IAEA, Vienna (1982).

OCCUPATIONAL RADIATION PROTECTION IN MEDICINE IN MALI

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Abstract

In Mali, there is no private Service provider in the field of occupational dosimetry. Actually, it is the IAEA which, in 2005, provided AMARAP (National Regulatory Authority) with it's first TLD reader (HARSHAW 4500) and some TLD cards to restart this important activity which was interrupted since several years. In 2010, the Government of MALI, provided AMARAP with a new TLD Reader (HARSHAW 6600plus) and more TLD cards. About 600 radiation workers are (or were) monitored either individually or collectivly (including a few workers from Guinea, Benin and Burkina Faso).

1. INTRODUCTION

In Mali, a Decree has been adopted by the government (Decree N°06-488 /P-RM of 23rd November 2006) which covers the subject of occupational radiation protection. And, there are some provisions about how regulations require the implementation of radiation protection programs by end-users as well as provisions about requirements for the authorization of technical services (service providers) related to occupational exposure.

2. METHODS

For the moment, workers are being monitored only for external exposure using TLD cards. Since 2006, AMARAP is the unique service provider for external occupational dosimetry in the country.

A TLD system for external monitoring has been provided to AMARAP by the IAEA in 2005 under Project RAF/9/027. The system consists of a manual Harshaw 4500 TLD reader with Nitrogen generator and other accessories including some 300 TLD cards. In 2008, AMARAP purchased 500 TLD cards. In 2010, the Government of Mali provided AMARAP with a new TLD Reader which is an automatic HARSHAW 6600plus. During the same year (2010), the IAEA provided to AMARAP 500 TLD cards.

The monitoring period is 3 months and travel dosimeter is used with automatic background substraction. During Expert Mission held in July 2011, procedures have been set up, related to quality management and operational aspects of the Dosimetry Department. Dose record keeping system has been set up in electronic and hard copy formats.

There is no monitoring for intake of radionuclides in place yet. But AMARAP is considering the possibility of doing extremity dosimetry using its new HARSHAW 6600plus. AMARAP is also intending to use for this purpose its radiological analysis laboratory set up

in 2007 with Government funds. This laboratory is consisted of a Gamma spectrometer Chain and an Alpha-Beta Counter.

According to regulations, workplaces must be classified and monitored and the control shall be done by the RPO. Most of the practices in the country are using X-rays or gauges (apart from Nuclear Medicine, using only ^{99m}Tc. However, environment monitoring using TLD cards is used for some workplaces.

AMARAP is the unique service provider to-date in the field of occupational exposure monitoring. A new service provider called "RAYTEC –MALI" has been licensed by AMARAP in 2008 for maintenance and quality control of radiological equipment.

Since 2010, the Government of Mali funded a project on "Natural radioactivity measurements" in some regions of the country. Ambient gamma radiation is being measured and sampes (water, soil) are being analyzed. A Laboratory with HPGe and NaI detectors and also alpha-beta measuring equipment have been set up and personnel were trained. However, no investigations regarding natural sources of radiation that have to be considered as occupational have been made yet.

3. RESULTS

About 600 radiation workers are (or were) monitored by AMARAP (individually or collectivly) in 30 facilities including a few workers from some West African Countries (Benin, Burkina Faso and Guinea). Efforts are underway to increase the numbers by sensitization so that all radiation workers could be monitored. Five AMARAP staff members are dealing with monitoring activities (covering medical and non-medical sectors).

SECTOR	Deep doses (Hp(10)) in µSv	Shallow doses (Hp(0.07)) in µSv)
Dental X-ray	435.21	519.57
Container scanning	537.38	589.56
Road construction (gauges)	230.17	246.95
Banks (Luggage scanning)	281.09	238.83
Conventional radiology	1 398.33	1 516.65
Gold Mines (Nuclear gauges)	4460.37	5544.72

TABLE 1. AVERAGE ANNUAL EXTERNAL DOSES (monitoring with TLD

From these average doses, the following observations are made:

(a) Except for Hp(10) for Conventional radiology, all the other workers are bellow public exposure level (1 mSv per year);
(b) For Conventional radiology, the doses are bellow 3/10 of dose limits;
(c) For all sectors, shallow doses (Hp(0.07)) are higher than deep doses (Hp(10));
(d) for dental X-ray machines, doses are of the same order as for the other sectors except for conventional radiology; which means that dental machines must not be neglected.

4. DISCUSSIONS

Some people are against a Regulatory Body offering Occupational Dosimetry services, fearing that it will result in a conflict of interest issue. The thought may not be competely out of place. But in many countries, there is no private company to do the job. In this case, the choice has to be made either to leave workers not monitored or to have the work done by the Regulatory Authority.

In the case of Mali, the IAEA decided that the Regulatory Authority can do the dosimetry rather than leave the radiation workers unmonitored. In this regard, in 2005, IAEA provided AMARAP with it's first TLD Reader and some TLD cards to start with. A one week training also was conducted in Bamako for AMARAP staff.

The major problem that is facing in the laboratory is about calibration of the the dosimetry/monitoring systems. Since the department do not have a Secondary Standard Dosimetry Laboratory (SSDL), calibration dosimeters are reqired to be sent abroad for irradiation. Traing of personel is also a challenge.

5. Conclusion

To-date, occupational radiation monitoring is done by the Regulatory Authority. But as soon as a reliable service provider is available, the monitoring job will be handed over to the service provider, and inspected by AMARAP from time to time. By that time, some measures have to be taken to:

(a) Monitor intakes of radionuclides (for Nuclear Medicine Department),
(b) Start extremity dosimetry (for some research institutions and C-arm users),
(c) Continue AMARAP staff training in this field,
(d) Make sure that all radiation workers are monitored, and
(e) Fix the broken TLD 4500 reader.

OCCUPATIONAL RADIATION PROTECTION IN NEPAL: CURRENT STATUS, ISSUES & CHALLENGES

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Abstract

Nepal has a long history of use of radiation in medical field since1923. But the country still does not have any Occupational Radiation Protection infrastructure to control the use of ionizing radiation in the various related fields. This study is an assessment of the occupational radiation protection in Radiology, Radiotherapy and Nuclear Medicine departments as well as different institutions having radioactive materials, carried out at different hospitals/institutions having either radiation facility or radioactive material in different regions of the country. Sampling was done to make the selection more representatives for the different regions in the country. Seventy three different hospitals/institutions and poly clinics from different parts of Nepal were studied. During the study, it was found that Occupational Radiation Protection was problematic due to lack of personnel monitoring service for radiation workers in Nepal. The result shows that around 67% of radiation workers are not monitored for radiation exposures. The surveyed hospitals with medical physicists have TLDs for personnel monitoring. There is a great need for Radiation Protection Regulation in the country. In 2008, Nepal became a member of the IAEA which will hopefully support and speed up the creation of appropriate conditions.

1. INTRODUCTION

Nepal is one of the world's least developed countries [1] with a population of nearly 26.6 million [2]. Radioactive materials and radiation are mainly used in the medical fields in Nepal. Small quantities of low-level radioactive materials are also being used in research and education at other institutions. Radiation is also being used for the calibration of equipment at some research centers and in the fields of mining, agriculture, food and drug administration. But there is neither legislative body nor any regulatory body to set standards for radiation used in medical and other fields. Reliable records of the number of radiological facilities in operation were also lacking, until a few years ago, when the Ministry of Science & Technology and Environment (MoSTE) provided an opportunity to a few professionals including this scribe to review and record the current status of radioactive materials in Nepal.

Ionizing radiation is being used at diagnostic imaging, external beam radiation therapy, brachytherapy, and nuclear medicine to diagnose and treat a number of common conditions. To ensure the safety of patients, providers, and surrounding staff members, it is important that the health care community become familiar with the terminology, common equipment, and standard practices used in radiation safety and monitoring [3].

2. CURRENT STATUS

Nepal has a long history of medical radiology (1923) [4]. Spectacular development has taken place since then, especially in the field of diagnostic radiology and radiotherapy. Radiotherapy was first introduced in 1976 with the use of a radium needle in the Maternity Hospital. CT scans and Nuclear Medicine procedures were then introduced in 1988 at the National Academy of Medical Sciences (NAMS), Bir Hospital. In 1991, NAMS, Bir hospital

also introduced their first Radiotherapy unit with a Telecobalt-60 machine [4]. Today, there are various facilities that provide radiotherapy services and also tremendous development has taken place in the field of diagnostic radiology in Nepal. The latest radiological equipment have been acquired by various hospitals and this has had a positive impact on the general health service. But the lack of control could cause a significant problem such as radiation hazards to radiation workers as well as the general public. Besides, the quality of service being delivered can by no means be overlooked, both in terms of effectiveness of diagnoses and treatments and especially since it entails very high-intensity radiation. The following tables show the status of distribution of radiation emanating equipment and radiation workers especially working in medical field.

TABLE I. (a) TOTAL NO. OF RADIOLOGICAL EQUIPMENT PER MILLION INHABITANTS (b) TOTAL NO. OF RADIOLOGICAL WORKERS, PER MILLION INHABITANTS

(a)

Diagnostic& Therapeutic Equipment	Total number of Equipment	No.per million inhabitants
Cobalt- 60	3	0.112
Linear Accelerators	3	0.112
HDR Brachytherapy	4	0.150
MRI	11	0.413
Gamma Camera	3	0.113
CT Scan	34	1.278
Mammography	14	0.526
X-ray/Fluoroscopy	1000 (Approx.)	37.59

(b)

Profession	Total number	No. per million* inhabitants
Radiologists[b]	150	5.639
Radiation Oncologists	15	0.564
NM Physician	2	0.113
Radiographers[c]	365	13.721
Medical physicists	9	0.338

[a]Population 26.6 million based on November, 2012 (source: Central Bureau of Statics)

[b]Nepal Radiologist Association website (www.nranepal.org, March 2014)

[c]Nepal Health Professional Council registration data. (December 2012)

As the above-mentioned account indicates, radiotherapy, diagnostic radiology and nuclear medicine has made substantial progress in Nepal.

3. METHODS

In Nepal, Radioactive materials and radiation are mainly used in the medical field. Hence the main focus of the study is on medical field. This study was carried out in Government Hospitals in the Kathmandu valley, Zonal Hospitals and Regional / Sub Regional Hospitals in different regions of the country under the Ministry of Health & Population (MoHP). This study was also conducted in some private hospitals, Teaching hospitals, small poly clinics with X ray facility and other institutions having radioactive material. Sampling was done to make the selection more representative for the different regions in the country. Selection was done to cover government hospitals, teaching hospitals, private hospitals and poly clinic with radiation facility. The study was done with site visit and questionnaire. The questionnaire for radiation workers consisted of questions seeking information regarding professional responsibility, protection training, and availability of personal radiation dosimetry service etc.

4. RESULT & DISCUSSION

Altogether, 73 different hospitals/institutions and poly clinic (medical test facility mainly for recruiting for foreign labor job seekers) with X-ray facility were studied. Evaluation of study shows that Occupational Radiation Protection is a big problem due to lack of personnel radiation dose monitoring system for radiation workers in Nepal. The result shows that around 67% of radiation workers are not monitored for radiation which is 2% higher than earlier study [4]. The dose limit of the radiation workers as recommended by ICRP [5] is virtually unknown in many cases. The surveyed hospitals with medical physicists have TLDs for personnel monitoring. The TLDs used in Nepal, are form Bhabha Atomic Research Center (BARC), India. There is a great need for rules, regulation and Radiation Protection Regulation in the field of radiation applications in medical field.

Study also shows that there is no Quality Assurance (QA) program in almost all diagnostic imaging facility. There must be regular quality control parallel to maintenance program for the X-ray equipment at regular intervals. The basic radiation protection principles of Justification and Optimization should be taken into consideration, in this period of rapid increase of investigation following the availability of new equipment. Through proper radiation education and training and regularly organized seminars, conferences people are becoming more and more aware about the benefits of radiation, its uses in medicine, etc. By establishing basic safety standard and radiation control authority, rules and regulations can be enforced in the country effectively and efficiently.

The resources in terms of qualified radiological workers are limited in Nepal. Except few qualified medical physicists and radiological technologists, others do not have any formal education and training in radiation protection [4]. Medical physicists or Radiation Protection Officers, who are responsible for an overall radiation safety issues, are very limited in Nepal. World Health Organization provided radiation monitoring film badges to radiologists and radiographers in 1978; however, dose monitoring was not done routinely [6]. One study showed that only 149 personnel were monitored for their occupational doses and non-monitored personnel performed 76 % of the X-ray procedures like angiography, catheterization producers and intestinal barium procedures covered in that study for the year 2007 [6] .

4.1. Issues & Challenges

Though the history of radiation practice is long in Nepal, there is as yet no Radiation Act, nor any legal standards for occupational radiation protection. There are no centralized official records on radiological facilities in operation, and the number and types of units, radiation workers and their qualifications, safety measures and conditions of workplace were virtually unknown. Due to lack of the regulatory body, some clinics are operating the X-ray in a room with a window and door without lead shielding; one disused source of ^{60}Co is lying at the premises of the B.P. Koirala Memorial Cancer Hospital, Bharatpur. In 1971 & 1975, Radium source in the form of tubes which was donated to Nepal Government by the American Institute was buried at the premises of the Department of Health Services in Kathmandu.

The IAEA, nuclear watchdog of the United Nations will certainly support and speed up the creation of appropriate conditions. And the Ministry of Science, Technology & Environment (MoSTE) is the line agency responsible for official contact with the IAEA. Now is the time for the establishment of a Radiation Regulatory Body for developing and monitoring of essential radiation safety and radiation control infrastructure in the country. The most essential introduction of Nuclear Act is long overdue, not to mention its subsequent enforcement for providing licenses, establishment of other concomitant radiation rules and regulations, code of radiological practice, supervision of quality assurance and radiation

protection program, training of manpower and conducting required research to sustain and maintain quality assurance and radiation protection, establishment of personnel radiation monitoring system along with proper management and disposal of radioactive waste .

The achievement and maintenance of a high level of safety in the use of radiation sources and in the management of radioactive waste depends on a sound legal and governmental infrastructure, including a regulatory body with well-defined responsibilities and functions. An appropriately organized and staffed regulatory body with access to adequate resources is a key element of such an infrastructure. The IAEA publication Fundamental Safety Principles [8] sets out safety principles that provide the bases for the IAEA Safety Standards. The International Basic Safety Standards for Radiation Protection and Safety of Radiation Sources, GSR Part 3 [9] establishes requirements for the protection of people and the environment from harmful effects of ionizing radiation and for the safety of radiation sources

5. CONCLUSION

Though the history of radiation practice is long in Nepal, there is as yet no Radiation Act, nor any legal standards for occupational radiation protection. Seventy three different hospitals/institutions and poly clinics from different parts of Nepal were studied. During the study, it was observed that Occupational Radiation Protection is a big problem due to lack of Personnel Monitoring services for radiation workers in Nepal. The result shows that around 67% of radiation workers are not monitored to determine radiation exposure. The surveyed hospitals with medical physicists have TLDs for personnel monitoring. There is a great need for rules, regulation and Radiation Protection Regulation in the field.

The Way Forward

In 2008, Nepal became a member of the IAEA and this will certainly support and speed up the creation of appropriate conditions. Despite all these issues and challenges, the author remains optimistic about the eventual promulgation of the Nuclear Law and the formation of an effective Regulatory Body. From 2012 onwards, Nepal has also been involved in various projects associated with the IAEA including those concerned with the establishment of a radiation regulatory framework. In 2007, a National Nuclear Policy document was produced which mandated MoSTE as its promoter, regulator and facilitator for implementation. At present, IAEA-linked nuclear related activities are being carried out by MoSTE. One IAEA Technical Cooperation project entitled “Developing and Establishing National Infrastructure for Radiation Safety” was started in 2012 to develop radiation protection infrastructure relating to rules and regulations on radiation safety in all thematic areas in Nepal.

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REFERENCES

[1] THE WORLD BANK WEBSITE: http://www.worldbank.org/en/country/nepal/overview Retrieved on 23 June 2014.
[2] CENTRAL BUREAU OF STATISTICS, Nation Population & Housing Census 2011 National Report, Central Bureau of Statistics, Kathmandu, Nepal (2012).

[3] CUARON, J.J., HIRSCH, A.E., MEDICH, D.C. et al., Introduction to radiation safety and monitoring, J. Am. Coll. Radiol. (2011) 259-64.
[4] ADHIKARI et.al., Status of radiation protection at different Hospitals in Nepal, Journal of Medical Physics, Vol. 37, No. 4, (2012) 240-4.
[5] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, The 2007 recommendations of the International Commission on Radiological Protection, ICRP, (2007) 37.
[6] BHATT C.R, WIDMARK A, SHRESTHA S.L, KHANAL T, WEEN B., Occupational radiation exposure in health care facilities, Kathmandu Univ. Med. J., 39(3) (2012) 53.
[7] ADHIKARI K.P., Radiation Regulation in Nepal: A Challenging Journey, Radiation Regulator, The International Journal Supporting Continuous Professional Development, Vol.1, No.1 (2012) 6-10.
[8] EUROPEAN ATOMIC ENERGY COMMUNITY, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Fundamental Safety Principles, Safety Standards Series No. SF-1, IAEA, Vienna (2006).
[9] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Standards: Radiation Protection and Safety of Radiation Sources: International Basic Safety Standard, GSR Part 3 (Interim), IAEA, Vienna (2011).

PROPOSAL FOR THE CONSTRUCTION OF ALPHA CURVES

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Abstract

When a cost benefit analysis is applied to the optimization of practices involving radiation protection, the Alpha Value (α) is used to determine the amount of money required to be invested in a practice to minimize radiation doses to acceptable levels. The Alpha Value is often linked to the GDP per Capita, so the monetary reference value of Person-Sievert can often be different in each country. This is not a good measure since the risks associated with a practice are the same in all countries. Usually, a cost benefit analysis only compares the costs associated with a practice versus collective doses. The distribution of the maximum individual doses is not directly taken into account in the analysis. This paper proposes that a better approach to the optimization of practices in radiation protection is to work with an "Alpha Curve " or a set of "Alpha Curves " rather than a single value for α . These curves can provide values of Alpha as a function of the highest individual doses evaluated for each option considered in the optimization process. The Alpha Curves are independent of the GDP per Capita, being only a function of the amount currently paid to the highest individual doses that are used in each country.

1. INTRODUCTION

When a cost benefit analysis is applied to the optimization of practices involving radiation protection, an Alpha Value (α) is used to determine the amount of money required to be invested in a practice to minimize radiation doses to acceptable levels. The Alpha Value is often linked to the GDP per Capita of the country where the practice takes place and therefore the person-Sievert reference value can be different in each country. As a consequence there is no global or regional uniformity in the choice of these monetary reference values, which leads to unsatisfactory outcomes since the risks associated with a practice are the same in all countries.

To address this anomaly, this paper presents a methodology that uses alpha value curves based on global or regional statistical distribution of the highest individual doses to workers in different practices involving ionizing radiation. This alpha curve approach should be used in a decision-making technique so that the highest individual dose for each option is effectively considered in an optimization problem. Thus, this paper proposes a change in the current technique of cost-benefit to one in which Y is the biological detriment directly related to the highest individual doses of each option.

Although we have emphasized that the values for the construction of α curves should be a function of the distribution of maximum individual doses currently practiced in different countries, we do not have such data to construct these global or regional curves. So, in this paper we construct a hypothetical curve based on monetary value of the person-Sv used by different utilities in Germany, Belgium and France, in accordance with the reference [1], and shown in Table I.

In order to present the changes in the techniques of cost-benefit, we used the hypothetical example of the ventilation system of a uranium mine which was developed previously in reference [2].

From this example we used the distribution of individual doses, for which we assumed that the average dose for each group is equal to the highest individual dose received for each group, the collective doses and the protection costs of each option.

The protection costs of each option have been updated from those used for the year 2004 when reference [1] was published. An update rate of one percent per month has been used for this paper.

2. THE ALPHA CURVE

$$\alpha = f\left(H_{\max}\right)$$

To build this hypothetical curve, we used the monetary values of the person-Sv used by different utilities in Germany, Belgium and France in accordance with reference [1], and shown in Table I.

TABLE I. MONETARY VALUE OF A PERSON-SV USED BY DIFFERENT UTILITIES IN GERMANY, BELGIUM AND FRANCE

Country	Utility	Monetary value of person-Sv in U.S. dollars
Germany	Proposal of the VGB[a] under trial by the utilities	<1 mSv : no value 1-10mSv: 170 000 10 – 20mSv: value growing linearly to reach 1 695 000 at 20mSv
Belgium	SCK-CEN[b]	<1 mSv : 27 000 1-2mSv: 67 000 2-5mSv: 267 000 5-10mSv: 667 000 10-20mSv :1 333 000 20-50 mSv : 5 333 000
France	Eletrecité de France	0-1 mSv: 17 000 1-5mSv: 83 000 5-15 mSv:383 000 15 – 30 mSv:1 117 000 30-50 mSv: 2 500 000

[a] **VGB**:Technische Vereinigung der Grosskraftwerkbetreiber

[b] **SCK-CEN**: Studiecentrum voor Kernenergie/Centre d'étude de l'énergie nucléaire.

Table I shows the values of the person-Sv at 5, 10, 20 and 30 mSv. With these values we calculated the average amount paid by the different utilities in the three countries and adjusted an alpha curve depending on the maximum individual dose, $\alpha = f\left(H_{\max}\right)$, given by:

$$\alpha = 129{,}9H_{Max}^{3} - 1160H_{Max}^{2} + 41326H_{Max} - 20533 \quad (1)$$

3. CHANGES IN THE TECHNIQUES OF COST-BENEFIT ANALYSIS

The proposed change in the technique of cost-benefit analysis will take into account the highest individual dose of each option using the curve constructed for the alpha values. Unlike the original technique in which alpha was a constant value, in this revised technique alpha is a function that can take infinite values and as a consequence we have been able to create a modified detriment Z, defined by:

$$Z = f(H_{\max})S \qquad (2)$$

Where,

- $f(H_{\max}) = \alpha$ is the function that assigns monetary values to a unit collective dose, i.e., 1 person-Sv according to the maximum individual dose observed in each radioprotection option.
- S is the collective dose for each radioprotection option.

The optimal solution of radioprotection is the point where:

$$(X + Z)_{\min} \text{ or } (X + f(H_{\max})S)_{\min} \qquad (3)$$

4. APPLICATION OF THE TECHNIQUE TO THE EXAMPLE OF URANIUM MINE

Consider the problem of Uranium Mine proposed by ICRP 55[2], as shown in Tables 2 and 3.

TABLE 2. DOSE DISTRIBUTION IN THE EXAMPLE OF URANIUM MINE

Option	1	2	3	4	5
Collective dose S (person-Sv)	0.561	0.357	0.335	0.196	0.178
Average Individual Dose $_{\mathrm{H}}$ ($mSva^{-1}$)					
Group I - (4 employees)	40.8	28.4	26.0	17.5	15.8
Group II - (4 employees)	34.5	22.3	21.0	12.6	11.3
Group III - (9 employees)	28.9	17.1	16.3	8.4	7.8

The protection costs of each option have been updated for the year 2004, when the reference [1] was published. Update rate used was one percent per month.

TABLE 3. VALUES OF THE COST OF RADIOLOGICAL PROTECTION (X) PER OPTION AS IN REFERENCE [2] AND THE UPGRADED COST OF RADIOLOGICAL PROTECTION FOR 2004 (X_{2004}) OF EACH OPTION

Option	S (person-Sv)	X (US$)	X_{2004} (US$)
1	0.561	10 400.00	62 356.34
2	0.357	17 200.00	103 127.79
3	0.335	18 500.00	110 922.33
4	0.196	32 200.00	193 064.82
5	0.178	35 500.00	212 850.97

The adjusted alpha curve, Eq.(1), provides the alpha values for the highest individual doses per option, as shown in Table 4. rom the uranium mine example, we used the distribution of individual doses and we assumed that the average individual dose for each group is equal to the highest individual dose received for each group.

TABLE 4. ALPHA VALUES FOR THE HIGHEST INDIVIDUAL DOSES PER OPTION

Option	H_{max} (mSv/y)	α (US$/ person-Sv)
1	40.800	8 557 044.23
2	28.400	3 193 044.69
3	26.000	2 552 905.40
4	17.500	1 043 604.81
5	15.800	855 201.53

With obtained alpha values for each option, one can calculate the modified detriment, Z, as shown in Table 5.

TABLE 5. VALUES OF THE MODIFIED DETRIMENT, Z, FOR THE 5 OPTIONS OF URANIUM MINE

Option	H_{max} (mSv/y)	S (person-Sv)	$Z = \alpha S$ (US$)
1	40.800	0.561	4 800 501.81
2	28.400	0.357	1 139 916.95
3	26.000	0.335	855 223.31
4	17.500	0.196	204 546.54
5	15.800	0.178	152 5.87

5. RESULTS

Table 6 shows the result of the analysis. The optimal solution, in bold is where $(X+Z)_{min}$.

TABLE 6. ANALYTICAL SOLUTION USING THE MODIFIED DETRIMENT, Z

Option	S (person-Sv)	X_{2004} (US$)	$Z = \alpha S$ (US$)	X+Z (US$)
1	0.561	62 356.34	4 800 501.81	4 862 858.15
2	0.357	103 127.79	1 139 916.95	1 243 044.74
3	0.335	110 922.33	855 223.31	966 145.64
4	0.196	193 064.82	204 546.54	397 611.36
5	0.178	212 850.97	152 225.87	**365 076.84**

6. DISCUSSION OF THE RESULTS

Table 7 shows the results obtained using the methodology of ICRP 55, where the alpha value is fixed based upon the GDP of the region, the result of cost benefit analysis for the example of uranium mine indicates option 1 to be the optimum result.

TABLE 7. RESULTS ACCORDING ICRP55 FOR COST-BENEFIT ANALYSIS

Option	H_{max} (mSv/y)	S (person-Sv)	α ICRP55 (US$/ person-Sv)	Y (US$)	X (US$)	X+Y (US$)
1	40.800	0.561	20 000.00	11 220.00	10 400.00	**21 620.00**
2	28.400	0.357	20 000.00	7 140.00	17 200.00	24 340.00
3	26.000	0.335	20 000.00	6 700.00	18 500.00	25 200.00
4	17.500	0.196	20 000.00	3 920.00	32 200.00	36 120.00
5	15.800	0.178	20 000.00	3 560.00	35 500.00	39 060.00

The above results were obtained using an alpha value fixed, according to ICRP 55, at US$ 20,000,00, and with each option considering only the collective dose S, that is the sum of the mean doses for each option. Comparing the results from ICRP 55 with the alpha curve methodology presented in this paper, the results obtained given in Table 6 show option 5 to be the optimised result. This result was derived by prioritizing both the collective dose and the maximum individual dose of each option. For each maximum individual dose of each option of protection, a different alpha values is assigned and not a fixed alpha value as the ICRP 55 method. These different alpha values were obtained by means of a fitted curve considering the values alpha practiced in Germany, Belgium and France as shown in Tables 1 and 4. It is noted that only the values of Germany, Belgium and France were used as an example of how such values used in countries can affect the outcome of the cost benefit analysis.

7. CONCLUSIONS

Using the hypothetical curve constructed here as an example, the values used to illustrates how the adoption of an alpha-curve can produce more realistic results than using a fixed alpha value based on GDP in accordance with ICRP 55. In addition, these curves can consider the maximum individual doses for each option in a project or operation involving different applications of ionizing radiation. If statistical data is available for that actual investment in the person-Sv in the different countries or regions, then it would be possible to create more representative curves of the monetary value of the person-Sv for these regions. A further refinement to the alpha curve approach would be to replace curves based on the U.S. dollar to curves based on economic indicators such as, the purchasing power parity (PPP). The creation of these curves and the use of a modified detriment technique would provide even greater equality between workers of different countries or regions.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Optimización de la protección radiológica en el control de la exposición ocupacional, Colección de informes de seguridad N^{o} 21, IAEA, Vienna (2004).

[2] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Optimization and decision making in radiological protection, Publication 55, Ann. ICRP 20 (1), Pergamon Press, Oxford (1990).

OCCUPATIONAL RADIATION EXPOSURES IN SRI LANKA

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Abstract

An occupational exposure assessment program is a key component in Occupational Radiation Protection in any institution when management of radiation safety of radiation workers and minimizing radiation risk are concerned. A Personal Monitoring Service Laboratory (PMSL) with a Quality Management System (QMS) in compliance with ISO/ IEC 17025:2005 [1] for monitoring of occupational exposure of radiation workers in the country employing thermoluminescence dosimeters (TLDs) has been established by the Atomic Energy Authority (AEA) of Sri Lanka which is recently accredited. This paper describes status of effective dose of radiation workers in industry, research and educational institutions in the country and experience in management of over-exposures reported during the period from 2012- 2013.

1. INTRODUCTION

Measurement of personal dose equivalent, Hp(10) [2], is made using a Harshaw TLD-100 dosimeter worn on the chest by each worker involved in radiation work for estimation of effective dose of the whole body from external radiation during their work. Verification of results is done through the accredited Secondary Standard Dosimetry Laboratory (SSDL) facility of AEA and participation of an international inter-comparison program.

2. MATERIAL AND METHOD

TL dosimeter worn by each worker is read using Harshaw TLD Readers (models: 4500 or 6600-plus) at the end of a two-month monitoring period. Personal dose equivalent, $H_p(10)$, is calculated by subtracting natural background dose of each TL dosimeter worn by individuals. Element Correction Coefficient (ECC) for each TL element is also applied for correction of non homogeneity of TL response of dosimeters [3, 4].

A set of TL dosimeters with ECC values less than ± 10% (Hereafter referred to as Quality Dosimeters) mounted on a 30 cm x 30 cm x 15 cm slab phantom of polymethylmethacrylate (PMMA) [2] are exposed to personal dose equivalent of 5 mSv for calibration of TLD Readers which is done once in six months. After calibration of the Reader, verification of results is done using TLDs exposed to a range of known values of reference doses. Quality Dosimeters exposed to known reference doses are read at the beginning of reading of the TLDs, at the middle of the reading and after reading of TL dosimeters to ensure the accuracy of the results. Light readings and noise readings are also regularly monitored to verify the stability of the Reader. All TL dosimeters exposed to known reference doses for calibration / Quality Assurance (QA) procedure are left for 48 hours for fading away the low temperature peaks.

The performance of TL dosimeters (non-linearity, reproducibility) is also tested. TLDs having their performance within the acceptance limits specified in ISO/IEC 62387-1:2007 [5] are used for occupational exposure measurements.

3. RESULTS

Table 1 shows the occupational exposures of staff working in different facilities. Highest expanded uncertainty of dose estimation at 95 % confidence limit with coverage factor, k = 2 is ± 24% for photon radiation.

TABLE 1. OCCUPATIONAL EXPOSURES OF RADIATION WORKERS, 2012 - 2013

Facility	Effective Dose (mSv)[c] (2-month period)	Effective Dose (mSv)[e] (Maximum Annual dose)
Industrial facilities:		
All industrial workers	< 0.15	0.90
Radiography workers[d]	0.15 - 2.10	3.10
All workers exposed to NORM	0.30 - 0.50	3.00
Research facilities:		
All categories of workers	< 0.15	0.90
Educational facilities:		
All categories of workers	< 0.15	0.90

c - Effective dose in each two-month monitoring period and /or its variation.
d - Records of only 5 workers in an institute show these high values in some monitoring periods.
e –Estimated maximum annual effective dose.

4. DISCUSSION AND CONCLUSION

Occupational exposure data of radiation workers from Jan. 2012 - Dec. 2013, in few industrial, research and educational facilities, including an industry dealing with Natural Occurring Radioactive Material (NORM) were considered for analysis. Results are reported in Table 1. Radiation workers in all categories, except workers exposed to NORM received doses below recording limit, 0.15 mSv. A few radiography workers in an institute have received relatively high doses in certain monitoring periods.

The PMSL participated in an inter-comparison exercise conducted by the Nuclear Research Centre of Algeria to ensure the accuracy of radiation dose measurements using Thermoluminescence dosimetry and is firmly committed to achieve global harmonization wherever possible. As such, the QMS assures the quality and accuracy of the services provided to institutions, for the safety of their radiation workers.

The QA procedure assures the accuracy of the results of dose record on TLDs during the period of work of each worker. Reference doses used by the accredited SSDL verify the traceability of results to international system of units of measure. Parameters associated with

the measurement of uncertainty (Reader calibration factor, reproducibility, fading, non-linearity, reference dose used for calibration) are considered for estimation of uncertainty.

Accuracy of estimation of occupational dose is affected by a number of factors that influence the amount of radiation reaching the dosimeter from direct beam, and scattered radiation from the wearer's body when the worker facing backside to the radiation beam. Therefore, additional uncertainties arise as dosimeter calibration condition in the SSDL and the radiation monitoring environment in workplaces are not identical.

There are possibilities of recording a lesser dose due to inappropriate wearing of the dosimeters or wearing in the wrong location on the body or forgetting to wear the dosimeter during work. All these factors prevent an estimation of true value of effective dose received from occupational exposure.

Occupational doses of radiation workers who deal with X-rays are normally overestimated by about 25% due to higher value of energy responses of TLDs over the X-ray region with respect to ^{137}Cs [6]. Occupational exposure of four workers in an industry exposing to NORM has been monitored and their effective doses are somewhat high when compared to other workers in industry.

Investigations reported on overexposures revealed that the reason for most of the overexposure incidents in the industrial radiography has occurred as a result of not following the instructions given due to negligence of the workers.

However, according to the evaluation of occupational exposure results over the period of last two years, it is revealed that annual dose limit recommended by the ICRP for occupational exposures [7] are not exceeded.

REFERENCES

[1] ISO/IEC 17025:2005, General requirements for the competence of testing and calibration laboratories, (2005).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Calibration of radiation protection monitoring Instruments, Safety Report Series No. 16, IAEA, Vienna (2000).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of Occupational Exposure due to External Sources of Radiation, Safety Standards Series No. RS-G-1.3, IAEA, Vienna (1999).

[4] Publication No. ALGM-W-C-0801-001- Harshaw Dosimetry System (TLD Reader, WinREMS, and Dose Computation Algorithm), System calibration procedure, Publication No. ALGM-W-C-0801-001, Thermo Scientific, Erlangen (August 3, 2001).

[5] ISO/IEC 62387-1:2007- Passive integrating dosimeter systems for environmental and personal monitoring, First edition, 2007-07.

[6] C.HRANITZKY, H.STADTMANN, P.OLKO, Determination of LiF:Mg,Ti and LiF:Mg,Cu,P TL efficiency for X-rays and their application to Monte Carlo simulations of dosemeter response, Radiation Protection Dosimetry, Vol. 119 (1-4) (2006) 483-486.

[7] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, General principles for the radiation protection of workers, ICRP Publication 75. Ann. ICRP (27) 1. Pergamon Press, Oxford (1997).

REVIEW OF 10-YEAR WHOLE-BODY OCCUPATIONAL RADIATION EXPOSURE FOR RADIATION WORKERS IN GHANA

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Abstract

Occupational exposure to radiation workers in Ghana has been analyzed for a 10-y period between 2000 and 2009. Monitored dose data in medical, education, research and industrial institutions were extracted from the database of RPI of Ghana. 180 medical facilities were monitored. Highest annual collective dose of 601.2 man-mSv was recorded in 2002 and the least of 142.6 man-mSv was recorded in 2009, in the medical sector. Average dose per exposed worker for the medical sector was least in radiotherapy and highest in diagnostic radiology, with values 0.14 and 1.05 mSv, respectively. Range of average effective doses in diagnostic radiology, radiotherapy and nuclear medicine facilities were 0.328–2.614, 0.383–0.728 and 0.448–0.695 mSv respectively. In the education, research and industrial institutions, a total of 34 facilities were monitored. Annual collective doses received by exposed workers in education/research and industrial sectors reduced by ~39 and ~62% respectively, between 2000 and 2009. Average dose per exposed worker for the study period was least in industrial sector and highest in education/research sector with values 0.6 and 3.7 mSv, respectively. Range of institutional average effective doses within the education/research and industrial sectors were 0.059–6.029, and 0.110–2.945 mSv, respectively.

1. INTRODUCTION

The Radiation Protection Institute (RPI) in Ghana is responsible for the operational functions of the Radiation Protection Board (RPB), the body mandated by law as the national competent authority in the country with regulatory, monitoring and advisory responsibilities in matters pertaining to ionizing radiation [1, 2, 3, 4]. The functions of the RPB can be likened to that of the National Radiological Protection Board [5] in the UK. RPI through the occupational radiation protection sector monitors occupationally exposed workers (OEWs) in Ghana.

The RPB of GAEC has adopted, tried and tested, and is successfully using the IAEA's developed Dose Management System (DMS) as a tool to improve personnel and area monitoring in Ghana [8, 9, 10, 11]. For all justified practices that could involve occupational exposure, dose limits are imposed so that no exposed worker will be subject to an unacceptable risk attributable to the radiation exposure [6]. The dose limits are set and specified by the RPB with the backing of the LI 1559 to prevent the occurrence of deterministic effects and limit the probability of stochastic effects [7]. This study evaluates whole-body occupational exposure to ionizing radiation for radiation workers in Ghana during the 2000–09 period.

2. MATERIALS AND METHODS

The RPI employs an automated TLD processing service with manual data transfer system. Harshaw 6600 Plus Automated TLD Reader system [12] was used by RPI for whole body, extremity and environmental monitoring. The system offers 'one dosimetry solution' by its ability to monitor whole body (beta, photon and neutron), extremity and environmental exposure, with a single dosemeter. It is connected to an external personal computer (PC), and is operated through installed menu-driven WinREMS software.

LiF-100 TLDs were used in monitoring whole-body occupational exposure by RPI. The TLDs were calibrated against ^{137}Cs source. Skin and deep exposure values (R_{skin} and R_{deep}) were generated by the TLD reader and manually entered into a Microsoft Excel spreadsheet to estimate the corresponding personal dose equivalent values Hp(0.07) and Hp(10) [8, 12, 14]. The estimated dose data were then transferred manually into the DMS, where they are stored.

The skin and deep doses are calculated from the personal dose equivalent expressions (1) and (2).

$$\text{Skin dose: } Hp(0.07) = \left[(1.2958 \cdot R_{skin}) + 0.0097\right] \text{ mSv} \quad (1)$$

$$\text{Deep dose: } Hp(10) = \left[(1.3772 \cdot R_{deep}) + 0.0566\right] \text{ mSv} \quad (2)$$

3. RESULTS AND DISCUSSIONS

The annual collective doses for the sectors considered under this study are presented in Figure 1. Annual estimates of the 'average dose per exposed worker' and 'average dose facility/institution' were performed separately for each of the 10-y period and analyzed.

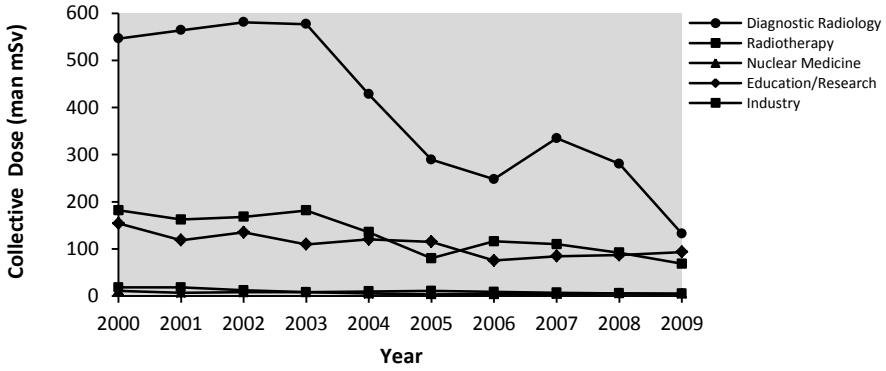

FIG. 1. Annual collective doses in sectors

3.1. Occupational dosimetry in medical practice

The annual collective dose received by exposed workers in the medical institution in Ghana reduced by a factor of 4 between 2000 and 2009. Maximum annual collective dose of 601.2 man mSv for the 10-y study period was recorded in 2002 and a minimum of 142.6 man-mSv was recorded in 2009. Annual average dose per medical institution decreased by 79 % from a value of 5.7 in 2000 to 1.19 in 2009. The annual average dose per exposed worker in the medical institution also followed a similar trend with a 67.6 % reduction in value from 2000 to 2009. The dose per exposed worker and dose per facility for the three categories are shown in Figs. 2 and 3, respectively. Annual collective dose in the diagnostic radiology decreased from 546.2 man-mSv in 2000 to 132.4 man-mSv in 2009, which shows ~76 % fall

in annual collective dose during the 10-y study period. The highest dose of 580.9 man-mSv was recorded in 2002. The radiotherapy and nuclear medicine facilities also showed reduction in collective doses by ~72 and ~55 %, respectively from 2000 to 2009. Highest annual collective dose in the two categories were however recorded in the year 2000.

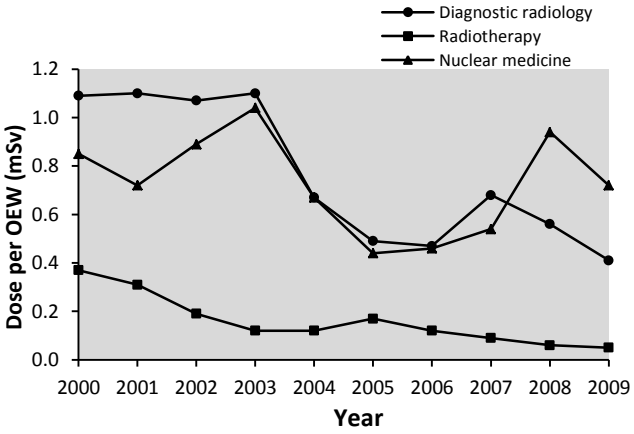

FIG. 2. Average dose per exposed worker in medical sector

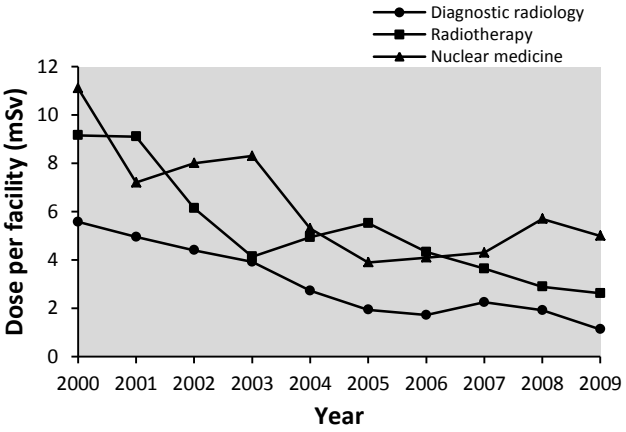

FIG. 3. Average dose per medical facility

Average dose per exposed worker in all medical facilities showed reduction over the study period. This observation may be the result of decreased workload or observation of proper radiation protection protocols [15]. Fig. 2 shows that the average dose per exposed worker was consistently low in radiotherapy when compared with the other facilities in each year. This observation is confirmed in Table 1, which shows average dose per worker values (for the 10-y period) of 1.05, 0.14 and 0.72 mSv in the diagnostic radiology, radiotherapy and nuclear medicine facilities, respectively. Average effective dose within the diagnostic radiology, radiotherapy and nuclear medicine facilities varied in the range 0.328–2.614, 0.383–0.728 and 0.448–0.695 mSv, respectively.

Nuclear medicine recorded the highest ‘dose per facility’ throughout the study period, except for the years 2001, 2005 and 2006, whilst diagnostic radiology consistently recorded the least in the study period. In the first year, ‘dose per facility’ value of 11.1 mSv was recorded in nuclear medicine as against 5.57 mSv in diagnostic radiology, while the last year of the study period recorded ‘dose per facility’ value of 5.0 mSv in nuclear medicine as against 1.13 mSv in the diagnostic radiology.

Diagnostic radiology practice recorded most of the individual doses > 1 mSv. For all individual doses > 1 mSv, ~97 % were in diagnostic radiology. The highest individual dose in radiology was 31.76 mSv, recorded in a period of 17 months. In radiotherapy, an individual

dose of 59.5 mSv was also recorded in a period of 5 months in the year 2001. No reasons were given for this observation but it is believed the TLD of the personnel might have been left in a treatment room for a long period. Subsequently, monthly dose records of the personnel were observed to have shown continuous reduction to levels below 1 mSv.

TABLE 1. SUMMARY OF DOSE DATA FOR OCCUPATIONAL RADIATION EXPOSURE IN MEDICAL PRACTICE (2000-09)

Medical facility	Average effective dose (mSv)		Total collective dose (man mSv)	Monitored medical facilities	Workers receiving doses	Average dose per medical institution (mSv)	Average dose per exposed worker (mSv)
	Min	Max					
Diagnostic radiology	0.328	2.614	3981.0	1353	5152	2.94	1.05
Radiotherapy	0.383	0.728	105.0	20	747	5.24	0.14
Nuclear medicine	0.448	0.695	62.9	10	87	6.30	0.72
All categories	0.328	2.614	4148.9	1383	5986	3.00	0.69

3.2 Occupational dosimetry in educational, research and industrial sectors

Annual collective dose in the education/research sector decreased from 154.1 man mSv in 2000 to 93.5 man mSv in 2009, which shows ~39 % fall in annual collective dose during the 10-y study period. The highest annual collective dose of 154.1 man mSv was recorded in 2000. The industrial sector also showed reduction in collective dose by ~62 % from 2000 to 2009. This observation could highly be attributed to a general improvement in radiation protection measures in the sectors [15]. Trend of 'dose per exposed worker' and 'dose per institution' in the education/research and industrial sectors are shown in Figure 4 and Figure 5 respectively.

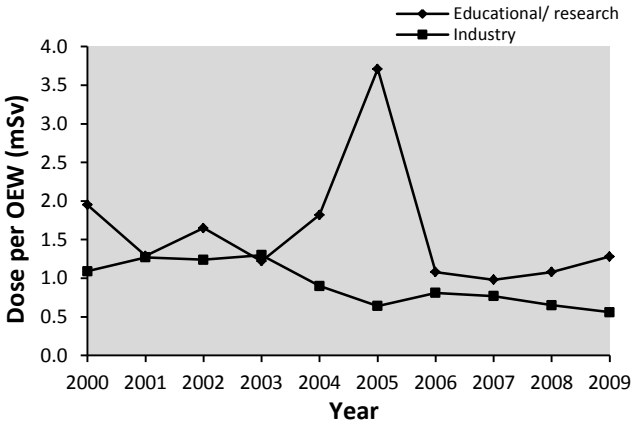

FIG. 4: Average dose per exposed worker in education, research and industry

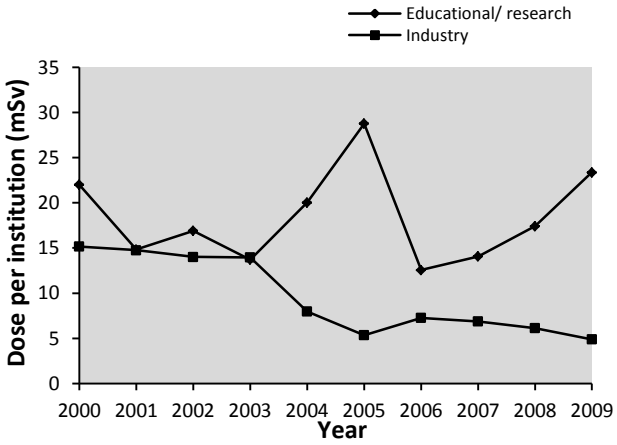

FIG. 5: Average dose per institution in education/research and industry

Average dose per exposed worker in the education/research and industrial sectors failed to follow a particular pattern between the 10-y period. Dose per exposed worker was consistently lower in the industrial sector than in the education/research sector in each year except for 2003. The observation may be due to decrease in workload or adherence to proper radiation protection protocols in the industrial sector. The sudden rise in annual dose per exposed worker value in 2005 in the education/research sector could be due to improper radiation protection measures resulting in unintended overexposure of certain TLDs [16].

Institutional average effective dose within the education/research and industrial sectors varied in the range 0.059–6.029 and 0.110–2.945 mSv, respectively as shown in Table 2. Between 2000 and 2009, average dose per institution values for the education/research sector were found to be higher than those in the industrial sector except for 2003. In the first year of the survey period, 'dose per institution' value in the education/research sector was ~45 % more than 'dose per institution' value in the industrial sector, while in the last year of the study period, 'dose per institution' value in the education/research sector was ~380 % more than 'dose per institution' value in the industrial sector.

Most prominent individual annual doses which were recorded in the study period were in the education/research sector between 2004 and 2005. Doses > 1 mSv were received primarily by exposed workers in the education/research sector, representing ~66 % of all doses > 1 mSv. The highest individual dose of 27.48 mSv was recorded in a period of 9 months. After this period, there was observed reduction in the levels of individual doses, possibly due to improved radiation protection measures.

TABLE 2. SUMMARY OF DOSE DATA FOR OCCUPATIONAL RADIATION EXPOSURE IN EDUCATION/RESEARCH AND INDUSTRIAL SECTORS (2000-09)

Institution	Average effective dose (mSv)		Total collective dose (man - mSv)	Monitored medical facilities	Workers receiving doses	Average dose per medical institution (mSv)	Average dose per exposed worker (mSv)
	Min	Max					
Education/ research	0.059	6.029	1113.5	62	750	17.96	1.48
Industry	0.110	2.945	1296.7	141	1397	9.20	0.93
All sectors	0.059	6.029	2410.2	203	2147	11.87	1.12

4. CONCLUSION

The overall collective dose of occupational exposure in medical practice for the study period was estimated to be 4148.9 man mSv, with corresponding average dose per exposed worker and average dose per medical institution values of 0.69 and 3 mSv, respectively. The total collective dose of occupational exposure in the educational, research and industrial sectors for the study period is 2410.2 man mSv, with corresponding average dose per exposed worker and average dose per sector values of 1.12 mSv and 11.87mSv, respectively. Collective doses in each of the studied categories reduced between 2000 and 2009, an indication that there could be further reduction in subsequent years. This observation could be a result of improvement in radiation protection protocols in the respective facilities. Generally, the individual doses also showed reduction with time.

REFERENCES

[1] ATOMIC ENERGY COMMISSION ACT, Act 2004 (1963).
[2] ATOMIC ENERGY COMMISSION (AMENDMENT) LAW, P.N.D.C.L 308 (1993).
[3] ATOMIC ENERGY COMMISSION ACT, Act 588 (2000).
[4] RADIATION PROTECTION INSTRUMENT, L.I 1559 (1993).
[5] O'RIORDAN, M. C., The NRPB era. J. Radiol. Prot. 30, 85 (2010).
[6] COLGAN, P. A., CURRIVAN, L. AND FENTON, D., An assessment of annual whole-body occupational radiation exposure in Ireland (1996–2005). Radiat. Prot. Dosim. 128, (2008) 12–20.
[7] Occupational Radiation Protection, Radiation protection and safety guide. GRPB-G3:2000. RPB (2002).
[8] HASFORD, F., AMOAKO, J. K., DARKO, E. O., EMI-REYNOLDS, G., SOSU, E. K., OTOO, F. AND ASIEDU, G. O., Implementation of dose management system at RPB of GAEC. Radiat. Prot. Dosim. 148(1) (2012) 3–8.
[9] IAEA, Dose management system (version 1.0) user manual, NSRW/PPSS, IAEA, Vienna (2004).
[10] IAEA, Dose management system (version 2.0) user manual. NSRW/PPSS, IAEA, Vienna, (2006).
[11] ICRP, 1990 Recommendations of the ICRP, ICRP Publication 60, Pergamon Press (1990).
[12] Harshaw TLD Model 6600 Plus Automated TLD reader user manual. Thermo Fisher Scientific Inc. (2007).
[13] ICRU, 1993 Quantities and units in radiation protection dosimetry, ICRU Report 51, ICRU (1993).
[14] HART D., WALL, B. F., Radiation exposure to the UK population from medical and dental X-ray examinations. NRPB-W4, National Radiological Protection Board (2004).
[15] BOADU, M., SCHANDORF, C., EMI-REYNOLDS, G., FAANU, A., INKOOM, S., GYEKYE, P. AND MENSAH, C. K., Systematic approach to training occupationally exposed workers in Ghana and the rest of Africa, Health Phys. J. 101(Suppl. 2) (2011) S116–S120
[16] HART, D. AND WALL, B. F., Radiation exposure to the UK population from medical and dental X-ray examinations, National Radiological Protection Board, NRPB-W4 (2004).

IMPLEMENTATION OF THE INTERNATIONAL RECOMMENDATIONS AND REVISED STANDARDS ON RADIATION PROTECTION FOR NUCLEAR FACILITIES IN BULGARIAN NATIONAL LEGISLATION
ADOPTED APPROACH AND CHALLENGES

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Abstract

Implementation of relevant IAEA international safety standards in nuclear field is considered last years as a proper established practice in Bulgaria as IAEA Member State. In particular, implementation of GSR Part 3 [1] as a basic standard in radiation protection area forms the highest priority for the Bulgarian regulatory body. The accepted iterative approach, the current status and some challenges are discussed here. Concideration is given only in respect to occupational exposure in nuclear facilities.

1. INTRODUCTION

The IAEA Safety Standards Series are aimed [2] to improve coherence and integration of standards in all areas, including radiation safety. As restructured in 2003 on three levels, now the IAEA safety standards present a complete base for maintaining the national legislation.

Bulgarian national policy on radiation protection is established by the national nuclear legislation. In the Act on the safe use of nuclear energy (ASUNE) [3] are incorporated the fundamental safety objective and ten safety fundamental principles as stated in the IAEA Fundamental Safety Principles [4]. In accordance with Article 26 of ASUNE, in the secondary legislation (the Regulation on the Basic Norms for Radiation Protection [5] are established the requirements on radiation protection. Mainly, regulation is subject of updating for implementing the IAEA revised standards, respectively GSR Part 3 [1].

2. ADOPTED APROACH

An iterative step-by step approach in regulations updating was accepted taking into account the external and internal circumstances, mainly:

(a) National specifics of the governmental infrastructure, legislation, interested parties;
(b) Maintain the proactive behaviour of the competent national authorities;
(c) Obligations under Euratom treaty;
(d) International cooperation as Member State of IAEA;
(e) Opportunities to implement effective feedback.

Bulgarian Nuclear Regulatory Agency (BNRA) accepted the policy of implementing the IAEA safety standards as a basis for developing the national nuclear legislation, and incorporated it in the internal quality management procedures for maintenance of the legislative documents.

According to Article 26 of ASUNE, the basic standards for radiation protection shall be established by a regulation adopted by the Council of Ministers on a motion by the BNRA Chairman and the Minister of Health. Besides BNRA, the Ministry of Health, Ministry of Interiors and Ministry of Environment are the competent authorities for the respective areas in radiation protection for nuclear facilities and activities. This requires an effective coordination of all national competent authorities as highest priority in conducting of the process.

Originating from GSR Part 1 [6] as Requirement #7, a real challenge for BNRA is to avoid potential overlap of regulatory responsibilities, and duplication of activities under different authorities. Omissions or undue duplications of requirements were avoided in permanent consultancy process between the relevant authorities starting at very early stage of drafting the regulation revision.

Important contribution to the legislation updating process is the participation of BNRA representatives in the IAEA safety standards committees, as well in working groups and committees, including: ENSREG, WENRA, HERCA, Scientific and Technical Committee under the EURATOM Treaty.

3. PERFORMED ACTIVITIES

Updated Regulation on basic norms for radiation protection [5] has been developed and adopted in September 2012 to reflect the IAEA GSR Part 3 requirements immediately after its publishing in 2011 and was linked also to the Draft “European basic safety standards" published by the European Commission in 2012 (adopted later on as Directive 2013/59/EURATOM) [7]. The most important changes in occupational radiation protection requirements are:

(i) Justification is required only for new activities which had been not proven yet. Existing activities have to be re-justified if new safety significant circumstances appear;
(ii) Occupational exposure dose limit of 20 mSv for each separate year, whereupon the limits of annual equivalent dose are: 20 mSv for eye lens; 500 mSv for skin; 500 mSv for limbs (palms, arms below elbows, foots, ankles);
(iii) Entities accredited according to ISO/IEC 17020:2012 for personal monitoring has to present the monitoring results within 15 days to the employer, as well to the the National Dose Registry;
(iv) In case of radiation incident, the exposure dose (external and iternal) has to be assessed by qualified expert on radiation protection;
(v) The licensee shall provide for the external personnel in the controlled area protection means and personal monitoring identical to its own personnel;
(vi) Radiological and tissue weight factors for assessment of external and internal exposure are up-dated;
(vii) Graded approach is required as related to regulatory regime;
(viii) A mechanism for application of the concept of exemption and clearance of material in accordance with publications “GSR Part 3” and “RS-G-1.7- is introduced.

Significant update in the Regulation [5] is the annual effective dose limitation to 20 mSv for each separate year. It was a challenge, because that limit initially proposed in the Draft “European BSS" couldn’t find consent.

The decision was elaborated based on the operational feedback. Last year, the NPP operator in Bulgaria implemented comprehensive Operational radiation protection program, covering all the topics as presented in IAEA safety guide NS-G-2.7 [8]. The legislative requirements for qualification and training of personnel are addressed, and for all radiation

workers is provided appropriate training in radiation protection ensuring the necessary level of competence. Practical implementation of the ALARA principle is central issue in that program. So called "ALARA Council" was established aimed to strengthen the involvement in planning and feedback from performed activities. As a result, significant reduction of the occupational exposure at sustainable level is reached, as presented in Table I.

TABLE I. OCCUPATIONAL EXPOSURE AT KOZLODUY NPP SITE FOR THE PERIOD 2006-2013 [9]

Annual effective dose	2006	2007	2008	2009	2010	2011	2012	2013
Collective [man.Sv]	1.65	1.06	0.66	0.68	0.90	0.59	0.40	0.46
Average individual [mSv]	0.38	0.33	0.21	0.20	0.28	0.20	0.18	0.34
Maximal individual [mSv]	13.0	8.6	9.3	7.3	10.6	6.9	6.5	8.2

4. CONCLUSION

Proper coordination between the national competent authorities is essential in planning and conducting the process of implementation of the international recommendations and revised standards on radiation protection. The accepted iterative approach seems to be appropriate in updating the Bulgarian national legislation on radiation protection.

The international cooperation in the safety standards elaboration and in their adaptation at national level is the common benefit.

REFERENCES

[1] IAEA SAFETY STANDARDS SERIES No. GSR Part 3 (Interim), Radiation protection and safety of radiation sources : international basic safety standards : general safety
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, WASTE SAFETY STANDARDS COMMITTEE, SIXTH THREE YEAR REPORT, period 2011-2013.
[3] ACT on the Safe Use of Nuclear Energy, S.G. 63 (2002), amended S.G. 68 (2010).
[4] IAEA SAFETY STANDARDS SERIES No. SF-1, Fundamental Safety Principles, IAEA, Vienna (2006).
[5] Regulation on the basic norms for radiation protection, S.G. 76 (2012).
[6] IAEA SAFETY STANDARDS SERIES No. GSR Part 1, Governmental, legal and regulatory framework for safety, IAEA, Vienna (2010).
[7] COUNCIL DIRECTIVE 2013/59/EURATOM of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, (2013).
[8] IAEA SAFETY STANDARDS SERIES No. NS-G-2.7, Radiation protection and radioactive waste management in the operation of nuclear power plants : safety guide, IAEA, Vienna (2002).
[9] BNRA, National Report on the Radiation Protection Status for the period from 2007 to 2012 Sofia (2013).

COLLECTIVE DOSE ASSESSMENT FOR OCCUPATIONALLY EXPOSED WORKERS IN MADAGASCAR DURING THE LAST 20 YEARS

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Abstract

The application of the Radiation Protection of workers through the individual monitoring is part of the conditions for the use of emitting Ionizing Radiations Devices. In Madagascar, the use of such techniques covers several fields dominated mainly by the Industry and Medicine fields. The National Law requires that all people working under Ionizing Radiations should be monitored by an individual dosimetric program. The assessment of the workers individual dose is quarterly done. In Madagascar, a HARSHAW 6600 Reader is used for individual monitoring using Thermoluminescent Dosimeter (TLD). The present work shows the evolution of the collective dose for each activity field from 1992 to 2013. Currently, the Industrial field presents the highest collective dose because it represents the largest number of exposed workers. In fact, during the last two decades, the collective dose in the industrial field increased from 22% to 73%, whereas that of the medical field decreased from 69% to 23%.

I. INTRODUCTION

The exposition of workers from ionizing radiation comes from several practices. In Madagascar, the use of ionizing radiation sources steadily increases especially in medical and industrial fields. To control the exposure of workers, it is necessary to monitor each worker involved in the practice. For personal monitoring, Madagascar uses Thermoluminescent dosimeters (TLDs) to measure the workers received dose. For that an HARSHAW 6600 reader is used. The "Institut National des Sciences et Techniques Nucléaires" (INSTN-Madagascar) is the national organization in charge of the regulatory personal monitoring in Madagascar.

The objective of the present work is to determine and to analyze the trend of the workers collective dose from 1992 to 2013. The collective dose is the product of the average workers received dose and the total number of workers which reflects the total dose received by the workers under ionizing radiations at national level.

2. MATERIALS AND METHODS

Before using the TLD reader and the dispatching of TLDs, the system is calibrated at the SSDL of INSTN-Madagascar; the calibration is performed at 1mSv dose.

After the calibration is done, the TLDs are dispatched to the workers occupationally exposed. Since 1992, the TLDs are read each quarter and the results of the reading are registered in a national database including the identification of the worker and the type of

work done by the worker. Each worker is classified into Medical or Industrial field. The average annual dose received by the batch of worker is calculated. Then the corresponding collective dose is determinate using the equation (1). The collective doses are expressed in term of $H_P(10)$.

$$S = \sum_i E_i \times N_i \qquad (1)$$

Where, *S* represents the worker collective dose, *Ei* represents the average of the received dose of the workers in a defined field (Industry or Medical), *Ni* represents the total number of the batch of workers.

3. RESULTS AND DISCUSSION

The collective dose for each batch of workers (Industrial and Medical) is listed in Tables 1 and 2. The following Figs. 1 to 4 illustrate the evolution of the number of workers involved in Medical and Industrial field and the trend of the collective dose for each sector from 1992 to 2013.

TABLE 1. TREND OF THE COLLECTIVE DOSE FROM 1992 TO 2013 FOR INDUSTRIAL FIELD

Year	Number (N)	Average Dose (E) (mSv)	Collective Dose (S) (p-mSv)
1993	37	0.80	29.7
1994	47	3.70	174.1
1995	63	3.28	206.9
1996	44	1.97	86.6
1997	55	2.05	112.7
1998	61	3.24	197.4
1999	59	3.59	212.1
2000	19	0.93	17.6
2001	37	0.73	26.9
2002	7	1.11	7.8
2003	31	0.94	29
2004	17	1.56	26.6
2005	39	6.51	254
2006	67	3.05	204.4
2007	139	1.12	156.2
2008	221	2.19	484
2009	217	2.87	621.9
2010	222	3.81	845.6
2011	325	3.37	1094.4
2012	380	3.06	1164.3
2013	409	3.15	1286

TABLE 2. TREND OF THE COLLECTIVE DOSE FROM 1992 TO 2013 FOR MEDICAL FIELD

Year	Number (N)	Average Dose (E) (mSv)	Collective Dose (S) (p-mSv)
1992	103	3.13	322
1993	106	3.19	337.9
1994	69	4.85	334.9
1995	43	4.76	204.6
1996	22	5.50	121.1
1997	28	3.09	86.6
1998	59	2.49	147.2
1999	75	2.54	190.5
2000	71	2.78	197.2
2001	46	2.33	107.3
2002	57	0.93	53.1
2003	73	2.41	175.7
2004	77	3.86	297.2
2005	73	3.96	289
2006	65	4.18	271.6
2007	75	2.47	185.6
2008	81	3.60	291.2
2009	71	3.72	263.8
2010	101	5.84	589.4
2011	114	4.44	506.7
2012	99	3.05	302.4
2013	74	4.02	297.5

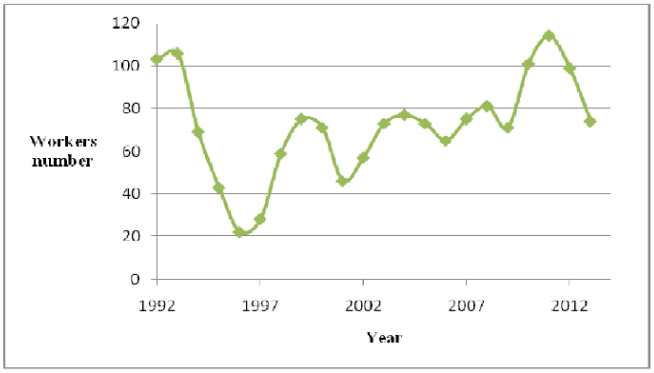

FIG. 1. Number of workers involved in Medical field

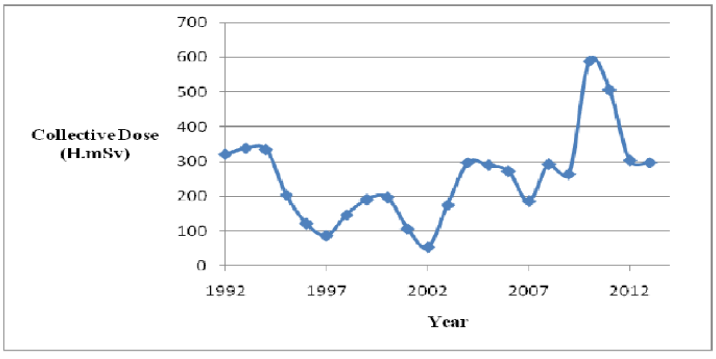

FIG. 2. Representative curve of the variation of collective dose in Medical field

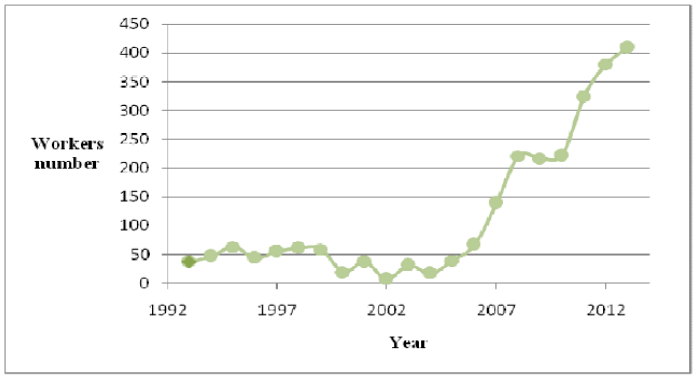

FIG. 3. Number of workers involved in Industrial field

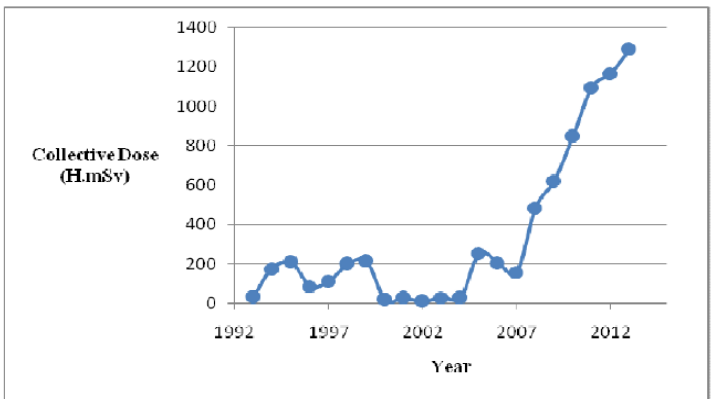

FIG. 4. Representative curve of the variation of collective dose in Industrial field

The observations are:

(1) The number of workers involved in the industrial field increases widely compared to medical field.
(2) The collective dose in industries increases significantly and steadily since 1992.
(3) The collective dose from medical field increases slowly through the years.
(4) The two main sectors contributing to the significant collective dose still the medical and industrial fields. In the beginning, the medical sector was the first contributor to the collective dose but the actual trend on the use of radioactive sources on industrial field grows speedily the dose received from this sector may increase as compared to the medical field.
(5) It is to underline that until now, the highest annual record dose was 25.8 mSv.
(6) This record dose was established in 2005 from the use of portable gauge. An investigation was performed to determine the cause. It was stated that the operator kept his TLD badge near of the source. An informative and training program was then conducted for all operators of the company about the use of TLD, following which the recorded doses became normal, i.e., under the annual limit.
(7) Some efforts were being made to reduce the collective dose: safety and security information and training program, good practice guide developing and implementation of the radiation protection program.

4. CONCLUSION

The increase in the collective dose is significant: about 22 % in 1992 against 73% in 2013 in the industrial field, and about 69% in 1992 against 23 % in 2013 in the medical field. It is stated that the industrial sector represent the main part of the workers exposures in Madagascar. The increase of the collective dose in the industries is due not only to the increase of the number of workers but also to the practice itself. In fact, during these last years the number of gammagraphy practices increased. Finally, it can be said that the analysis of collective dose from individual monitoring shows a good trend of the practices using ionizing radiation in the Country.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety of radiation sources: International Basic Safety Standards - Interim Edition. General Safety Requirements Part 3. IAEA Safety Standards Series GSR Part 3 (Interim), IAEA, Vienna (2011).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Occupational radiation protection. IAEA Safety Standards Series No. RS-G-1.1, IAEA, Vienna (1999).
[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of occupationale exposure due to intakes of radionuclides, IAEA Safety Standards Series No. RS-G-1.2 , IAEA, Vienna (1999).
[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of occupational exposure due to external sources of radiation, IAEA Safety Standards Series No. RS-G-1.3, IAEA, Vienna (1999).
[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Occupational radiation protection in the mining and processing of raw materials, IAEA Safety Standards Series No. RS-G-1.6, IAEA, Vienna (2004).
[6] United Nations Scientific Committee on the Effects of Atomic Radiation UNSCEAR 2000 Report to the General Assembly, with Scientific Annexes, Volume I: Sources and effects of ioizing radiation, United Nations, New York (2000).
[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Optimization of radiation protection in the control of occupational exposure, Safety Reports Series No. 21, IAEA, Vienna (2002).
[8] INTERNATIONAL ATOMIC ENERGY AGENCY. Intercomparison of radiation dosimeters for individual monitoring. Final report of a co-ordinated research programme 1988–1992). TECDOC-704 (Vienna: IAEA) (1993).
[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Intercomparison for individual monitoring of external exposure from photon radiation. Results of a co-ordinated research project 1996-1998. TECDOC-1126, IAEA, Vienna (1999).

DOSE TRENDS IN OCCUPATIONAL RADIATION EXPOSURE IN REPUBLIC OF MACEDONIA

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Abstract

Aim of the individual monitoring is the evaluation of the equivalent dose and the effective dose. The monitoring results are useful for the reduction of radiation dose to monitoring workers, the improvement of the practices and the methods used at different sectors and the compliance with the radiation protection system. In this paper are presented the annual avarege effective dose trends data for different category of application and occupational category in Republic of Macedonia.

1. INTRODUCTION

There are no nuclear applications in the Republic of Macedonia. The ionizing radiation sources are mainly used in medicine as well as in industry, veterinary medicine, education and science etc. Ionizing radiation sources in use are ^{60}Co in teletherapy unit, ^{192}Ir in brachytherapy and industrial gamma radiography, ^{90}Sr in industrial gauges, ^{131}I and other open radioactive sources in nuclear medicine, ^{152}Eu and ^{60}Co in radioactive lightning rods, linear accelerators in external radiotherapy, simulators and CT simulators, general radiography and fluoroscopy units, mammography units, dental X-ray equipment, CT units, kilo voltage unit in medicine, mobile accelerators in controlling the goods to be imported or exported etc.

2. INDIVIDUAL MONITORING

The Law on radiation protection and safety [1] and the Regulation on the way and measurement of the exposure of occupational exposed workers, keeping records and submitting reports [2] requires monitoring of occupational radiation exposure.

The workplace shall be classified as controlled area if employees may be exposed to radiation doses above 6 mSv per year or if the dose to the hands may exceed 150 mSv per year.

The workplace shall be classified as a supervised area if employees may be exposed to radiation doses in excess of 1 mSv per year or if the dose to the hands may exceed 50 mSv per year. Persons working within a controlled or a supervised area are considered to be occupational exposed workers classified as Category A and Category B, respectively.

The wearing period of personal dosimeters is one month for Category A occupational exposed persons and three months for Category B occupational exposed persons. The numbers of monitored occupational exposed workers within the past five years are in the range 1200 to 1500.

The dose limits for the whole body effective dose and equivalent dose for the lens of the eye, skin and extremities for occupational exposed persons are prescribed with the Regulation [3] in line with the Council Directive 96/26 [4] as follows:

(a) the dose limit for the effective dose is 100 mSv in a consecutive five-year period or 20 mSv per year averaged over five consecutive years but no more than 50 mSv in any single year;
(b) the dose limit for the equivalent dose to the lens of the eye is 150 mSv in a year and for the equivalent dose to the extremities or the skin is 500 mSv in a year

Workers are monitored only for external radiation exposure and the results are in in personal dose equivalent H_p (10) or H_p (0.07).

For the assessment of the whole body exposure and exposure to extremities of occupational exposed persons in Institute of public Health's laboratory, following equipment and dosimeters are in use:

i. TLD Reader Thermo 6600Plus DXTRAD;
ii. TLD cards model HARSHAW 0110 (LiF) and
iii. TLD cassettes with filters model HARSHAW 8814.

The laboratory is accredited against ISO 17025, and the measurements are based on:

1) ISO/IEC 1066
2) ICRP 75, General principles for the radiation protection of workers"
3) IAEA RS-G-1.3, Assessment of occupational exposure due to external sources of radiation
4) RP 160, Technical recommendations for monitoring individuals occupationally exposed to external radiation

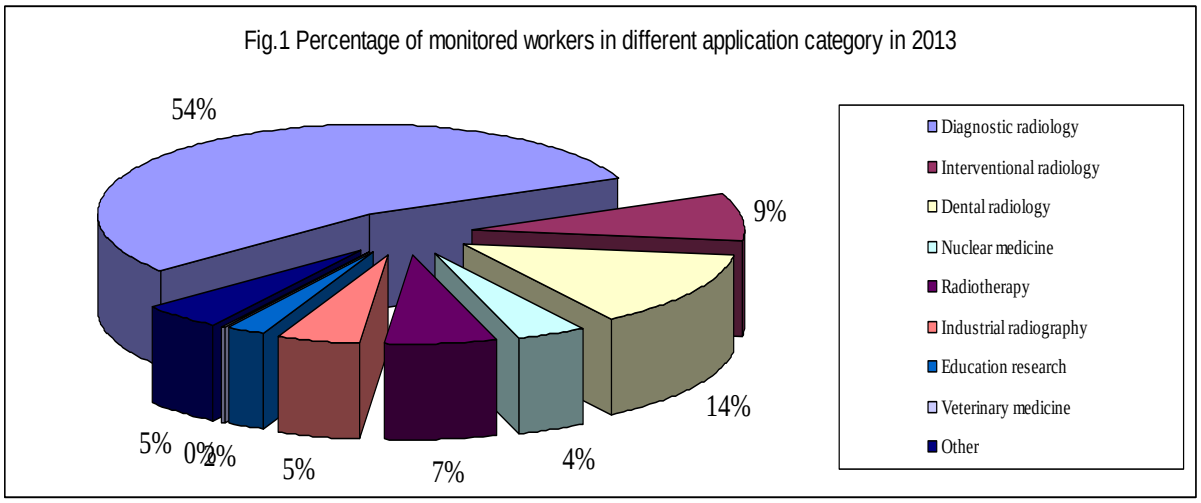
All dose results are stored in data base at Institute of Public Health together with information of the occupational exposed workers. Data in the data base are used to present dose statistics for occupational exposure in Republic of Macedonia.

The dosimetry laboratory at Institute of Public Health reports the dose results to the employer/undertaking with doses to individuals for the present monitoring period and accumulated doses the present year. In case of unusual or unexpected dose results or doses exceeding the dose limits, Institute of Public Health notifies the employer, monitored worker and regulatory body.

The total number of monitored workers in 2013 is 1249. The average annual dose for occupational exposure to external radiation is 1.03 mSv. The total number of monitored workers in 2013 and number of monitored workers in different dose ranges are presented in Table 1. The highest percentage, 54% of monitored workers (Fig. 1), is in diagnostic radiology applications.

TABLE 1. THE TOTAL NUMBER OF MONITORED WORKERS IN 2013, AND THE WORKERS IN DIFFERENT DOSE RANGES BY APPLICATION CATEGORY

Application category	Total number of monitored workers	Number of monitored workers E< 2 mSv	Number of monitored workers E from 2 - 5 mSv	Number of monitored workers E from 5 - 10 mSv
Diagnostic radiology	680	658	1	0
Interventional radiology	108	108	1	1
Dental radiology	172	172	0	0
Nuclear medicine	52	52	0	0
Radiotherapy	84	84	0	0
Industrial radiography	64	64	1	0
Education research	25	25	0	0
Veterinary medicine	5	5	0	0
Other	59	59	0	0



The total number of monitored workers in 2013 and number of monitored workers in different dose ranges by occupational category, are presented in Table 2. The ranges of average annual doses for different occupational categories are within 0.5 mSv for medical physicist to 2.01 mSv for industrial radiographers.

TABLE 2. THE TOTAL NUMBER OF MONITORED WORKERS IN 2013 AND NUMBER OF MONITORED WORKERS IN DIFFERENT DOSE RANGES BY OCCUPATIONAL CATEGORY

Occupational category	Number of monitored workers	Eavg. (mSv)	Number of monitored workers E< 2 mSv	Number of monitored workers E from 2 - 5 mSv	Number of monitored workers E from 5 - 10 mSv
Army stuff	20	0.87	20	0	0
Cardiologist	22	1.88	22	0	0
Dental assistant	67	0.97	67	0	0
dentist	105	0.93	105	0	0
Dosimetrist	4	1.08	4	0	0
Education/research	25	0.99	25	0	0
Health physicist	23	0.88	23	0	0
Ind. operator	20	0.93	20	0	0
Ind. Radiographer	128	2.1	128	1	0
Inspector	2	1.08	2	0	0
Interventional radiology Drs.	17	0.95	15	1	1
Med. Assistants	46	0.86	46	0	0
Med. Nursing stuff	101	0.99	101	0	0
Med. Radiology Drs.	192	0.93	191	1	0
Med. Technician	315	1.14	315	0	0
Medical Drs.	108	1.01	108	0	0
Medical physicist	15	0.50	15	0	0
Nuc. Med. Laboratory	8	0.96	8	0	0
Other	19	0.98	19	0	0
RPO	6	0.84	6	0	0
Veterinary medicine	5	0.90	5	0	0

2. CONCLUSIONS

Annual effective doses from occupational exposures in different applications in Republic of Macedonia are below the dose limits. The average annual dose for occupational exposure to external radiation is 1.03 mSv. The highest percentage, 54% of monitored workers (Fig. 1), is in diagnostic radiology applications. The range of average annual doses for different occupational categories are within 0.5 mSv for medical physicist to 2.01 mSv for industrial radiographers.

REFERENCES

[1] OFFICIAL GAZETTE OF THE REPUBLIC OF MACEDONIA No. 48/02, 135/07), Law on radiation protection and safety.

[2] OFFICIAL GAZETTE OF THE REPUBLIC OF MACEDONIA No. 29/10), Regulation on the way and measurement of the exposure of occupational exposed workers, keeping records and submitting reports.

[3] OFFICIAL GAZETTE OF THE REPUBLIC OF MACEDONIA, No. 29/10, Regulation on the limits of exposure to ionizing radiation and the conditions for exposure in special circumstances and in a case of emergency.

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[4] EUROPEAN COMMISSION, COUNCIL DIRECTIVE 96/29/EURATOM of 13 May 1996 laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiation, Official Journal L 15, 29/06 (1996).
[5] IAEA SAFETY STANDARDS SERIES, Assessment of occupational exposure due to external sources of radiation, Safety Guide, No. RS-G-1.3, IAEA, Vienna, (1999).
[6] EUROPEAN COMMISSION, Radiation Protection No. 160, Technical recommendations for monitoring individuals occupationally exposed to external radiation (2009).

REVIEW ON RADIATION WORKERS STATUS IN MALAYSIAN NUCLEAR AGENCY

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Abstract

When dealing with ionizing radiation, there is an obligatory requirement to monitor and assess the radiation exposure of the workers. Currently, there are 445 workers in Malaysian Nuclear Agency (Nuclear Malaysia), working in different fields and who are monitored using Thermoluminescence Dosimeter (TLD). They are also registered with the regulatory body as radiation workers. According to Basic Safety Radiation Protection 2010, only those working in a controlled area shall be monitored for their occupational dose and have to be registered as the radiation worker. Therefore, there is a need to review their status to fulfil the regulation requirements. This paper will discuss the review process which involves an assessment of their occupational exposure for 1 year. The occupational radiation exposure of monitored workers in 2013 will be presented. The radiation protection aspects correlate with the requirement of the National Law, the Atomic Energy Licensing Act 1984 (Act 304).

1. INTRODUCTION

Malaysian Nuclear Agency (Nuclear Malaysia) is the leading agency in introducing and promoting the application of nuclear science technology in Malaysia. The agency provides several major nuclear facilities purposely for research and commercialisation such as reactor, irradiation plants and radioisotope production laboratory. There is a national responsibility in regulating safety [1]. In Malaysia, the radiation protection aspect was drawn under Atomic Energy Licensing Act 1984 (Act 304). To support the act, Basic Safety Radiation Protection 2010 (BSRP 2010) was gazetted.

Currently, there are 445 workers in Malaysian Nuclear Agency (Nuclear Malaysia) who are monitored using Thermoluminescence Dosimeter (TLD) and working in different fields: medical/preparation of radiopharmaceuticals, industry/non-destructive training/plant assessment, agriculture, processing technology, Environment (radiochemistry lab for assessing environment's sample, radioactive waste management), Secondary Standard Dosimetry Laboratory (SSDL) for assessing the TLD and calibration of survey meter, research reactor, and irradiation plant (Gamma Greenhouse, Gamma-ray irradiation plant, Electron Beam Plant). At this time, all of them have been registered with the regulatory body as radiation workers regardless of area classification (controlled and supervised). Nuclear Malaysia as an employer have to monitor their occupational exposure, manage their health surveillance and keep the related records [2]. According to BSRP 2010, only those who are working in a controlled area shall be monitored, their occupational dose have to be registered as a radiation worker. Therefore, there is a need to review their status as radiation workers to fulfil the regulation requirements.

2. METHOD

At the early stages, the review process involves an assessment of their occupational exposure for 1 year. In this study, the annual occupational dose of 445 TLD users in Nuclear Malaysia in 2013 were assessed. These TLDs were processed and evaluated monthly by SSDL, Nuclear Malaysia. The TLD result could be obtained from e-SSDL [3]. The minimum detectable level (MDL) for these TLD is 0.1 mSv and the doses were recorded as zero when the doses are less than the MDL [4]. Nuclear Malaysia follows United Nations Scientific Committee on the Effects of Atomic Radiations (UNSCEAR) to classify the individual dose interval [4].

3. RESULTS AND DISCUSSION

Table 1 shows the number of radiation workers monitored using TLDs and the total /collective dose in the different division/unit in 2013. The workers in SSDL, those assessing the TLD and also film badge from other company show the highest total dose of 3.13 mSv/y while in the agriculture and processing technology their doses are less than the MDL. The SSDL workers are believed to be doing counting everyday compared to other lab. The agriculture and processing technology workers might be dealing with small amount of radioactive source or good in practicing the radiation protection principles, and the dose received is less than the MDL. The irradiation plants operated based on request from customer and should have a good shielding and operating system to protect the operators. The average annual dose received by the radiation workers in Nuclear Malaysia in 2013 is 0.01 mSv which is considered to be very small.

TABLE 1. THE NUMBER OF RADIATION WORKERS MONITORED USING TLD AND TOTAL/COLLECTIVE DOSE IN DIFFERENT DIVISION/UNIT

Bil	Division/unit	Number of TLD users	Total dose received (mSv/y)
1.	Secondary Standard Dosimetry Laboratory (SSDL)	28	3.13
2.	Irradiation Plant	44	0.36
3.	Industry	85	0.44
4.	Agriculture	28	0.0
5.	Processing technology	25	0.0
6.	Environment	59	0.21
7.	Research reactor	43	0.70
8.	Maintenance and radiation protection	99	0.36
9.	Medical Facilty	34	0.31
	Total	445	5.51
	Average:		0.01

As indicated in Fig. 1, 95.3% of radiation workers in Nuclear Malaysia received dose less than the MDL. Only 1 person received dose in the ranges: 1.0 – 4.99 mSv/y, and 20 workers obtained the MDL – 0.99 mSv/y. Nobody was overexposed and received dose more than the annual limit.

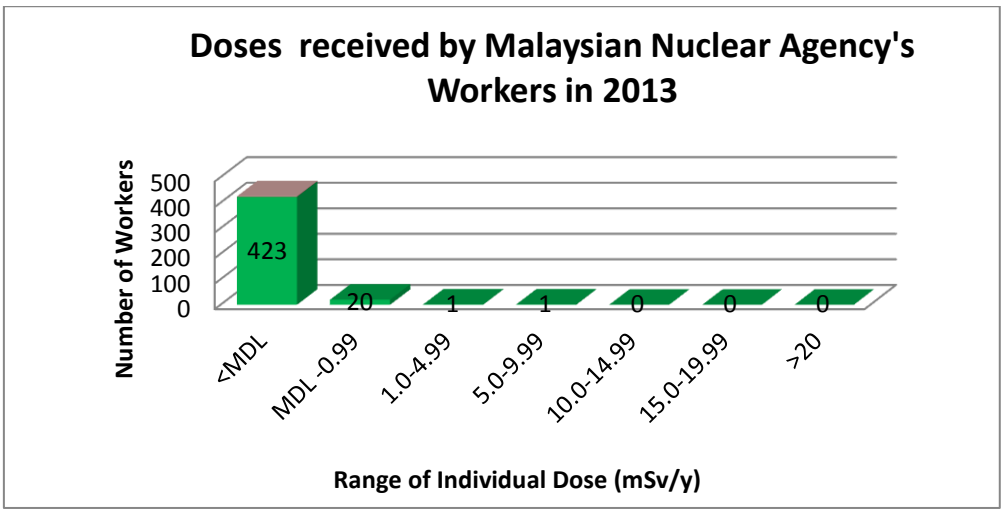

FIG.1: The distribution of exposed radiation workers in different dose ranges.

The results show that the dose received by Nuclear Malaysia's workers does not exceed the annual dose limit, which is 20 mSv/y [2]. The working area will be classified as controlled area where the workers are likely to received more than 6 mSv/y, and as supervised area when the workers are likely to receive dose of 1mSv – 6mSv/y [2]. Based on occupational exposure, most of the workers in Nuclear Malaysia are not required to be monitored and registered as radiation workers. However, there are other factors to be considered before the termination of radiation workers status, for instance, an area monitoring result. Some workers are dealing with radiation equipment and some of them are handling sealed and unsealed sources. There might be also internal contamination. Hence, further study needs to be carried out. After all the studies, which include occupational dose, area monitoring and internal dose assessment, the decision to terminate the radiation workers status could be made.

4. CONCLUSION

The occupational exposures of the workers who are dealing with ionizing radiation in Nuclear Malaysia were monitored by using TLD. The BSRP 2010 stipulates that only those working in the controlled area shall be registered as radiation worker, with the regulatory body. The annual dose record of 2013 shows that no one received the dose of 6mSvy-1, which is the guideline for classifying the area as controlled area. It is concluded that based on the annual dose assessment, the existing workers are not required to be registered as radiation workers. Their status have to be reviewed in order to fulfil the requirement of the national law. It could reduce the number of radiation workers in Nuclear Malaysia, and save time and energy of the divisions/units from doing unnecessary work in monitoring and managing the other requirements of radiation workers such as: medical surveillance, record keeping and provide training and radiation safety awareness [5]. Nevertheless, to complete the review process, further studies such as an assessment based on the area monitoring and internal dose assessment has to be carried out.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Standards for Protecting People and The Environment, General Safety Requirements, Part 3, No. GSR Part 3 (Interim), (2011).

[2] ATOMIC ENERGY LICENSING BOARD, Basic Safety Radiation Protection (2010).
[3] NUCLEAR MALAYSIA, Website of e-SSDL: http://essdl.nuclearmalaysia.gov.my
[4] A.B.A.KADIR et. al., Occupational radiation exposure in Malaysia for the year 2010-2013.
[5] Malaysian Nuclear Agency Radiation Protection Programme (2013).

A REVIEW OF OCCUPATIONAL DOSES IN CANADA OVER THE PERIOD OF 2008 TO 2012, REPORTED THROUGH THE CNSC ANNUAL SAFETY PERFORMANCE REPORTS FOR NUCLEAR SUBSTANCES AND PRESCRIBED EQUIPMENT

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Abstract

The Canadian Nuclear Safety Commission (CNSC), the federal authority for regulating the use of nuclear substances and prescribed equipment in Canada, publishes an annual safety performance report for licensees who use these regulated materials. There are four main sectors encompassed in this report; medical, industrial, commercial and academic research with each sector comprised of various different sub-sectors. Information from the most recently published report, covering the year 2012, demonstrates that whole body effective doses from the occupational use of nuclear substances and prescribed equipment in Canada remain very low, well below any regulatory limits. This paper provides a brief summary of the information available over the period of 2008 to 2012 to demonstrate that Canada has an effective program of licensing and compliance verification to ensure that the CNSC can meet its mandate to provide for the health and safety of persons. More comprehensive information may be found from the various annual reports that have been published on the CNSC website, starting with the first report covering the years 2008-2009.

1. INTRODUCTION

The Canadian Nuclear Safety Commission (CNSC) is the federal authority in Canada responsible the safe use of nuclear substances and prescribed equipment and this is accomplished thorugh a comprehensive program for authorisation and compliance verification. The CNSC issues licences and conducts compliance verification for the approximately 2600 licences issued for industrial, medical, commercial and academic research sectors. Approximately 70% of the licences are issued for industrial applications, such as fixed and portable nuclear gauges.

Beginning in 2011, the CNSC started to publish an annual report on the safety performance of licensees with respect to compliance with regulatory requirements. Currently, the following reports are available on the CNSC website: 2008-2009, 2010, 2011 and 2012. The CNSC evaluates the safety performance of each sector through an evaluation of regulatory compliance, worker doses and reported events. The medical sector, including diagnosis and treatment of patients in hospitals and medical clinics, represents 22% of all licences issued. The industrial sector, such as fixed and portable nuclear gauges and industrial radiography, accounts for the majority of the issued licences, approximately 58% of all licences were issued for these uses. The commercial sector, covering service providers and distributors, and the academic research sector, covering teaching and research, each account for 10% of the total number of licences issued. These reports do not contain information on occupational doses for uranium mines & mills, waste facilities, nuclear power plants or research reactors.

This paper reviews some of the results of the worker dose monitoring program in Canada as a basis for the effectiveness of the radiation protection programme for licensees of

the industrial, medical, commercial and academic research sectors. It also presents an overview of the CNSC annual industry safety performance report.

2. METHODS

In Canada, all licensees are required to submit an Annual Compliance Report (ACR) which provides a summary of their activities and performance during the preceeding 12 months. As part of the ACR, licensees must provide information regarding the doses received by workers. In Canada, persons who are expected to receive more than the public dose limit of 1 mSv per year are classified as nuclear energy workers (NEWs). Occupationally exposed persons who are not NEWs are required to adhere to the annual dose limit for members of the public (1 mSv/y).

In order to obtain the information for the reports on safety performance, CNSC staff sampled a representative number of dose records from each sector, arising from the approximately 2600 ACRs that are submitted each year. For example, in 2012, approximately 23% of the submitted ACRs were evaluated for the purposes of obtaining data for this report.

3. RESULTS

Fig. 1 shows the percentage of all nuclear energy workers and other workers in Canada who received an effective dose of less than 1 mSv/y resulting from the use of nuclear substances and prescribed equipment. The information is broken down by reported sector and includes information from 2008 to 2012.

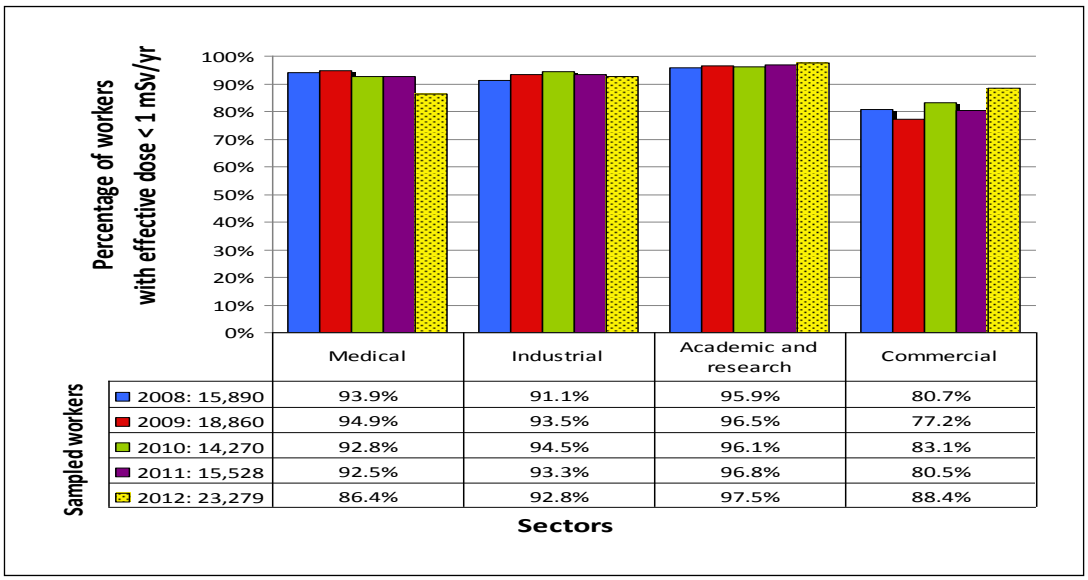
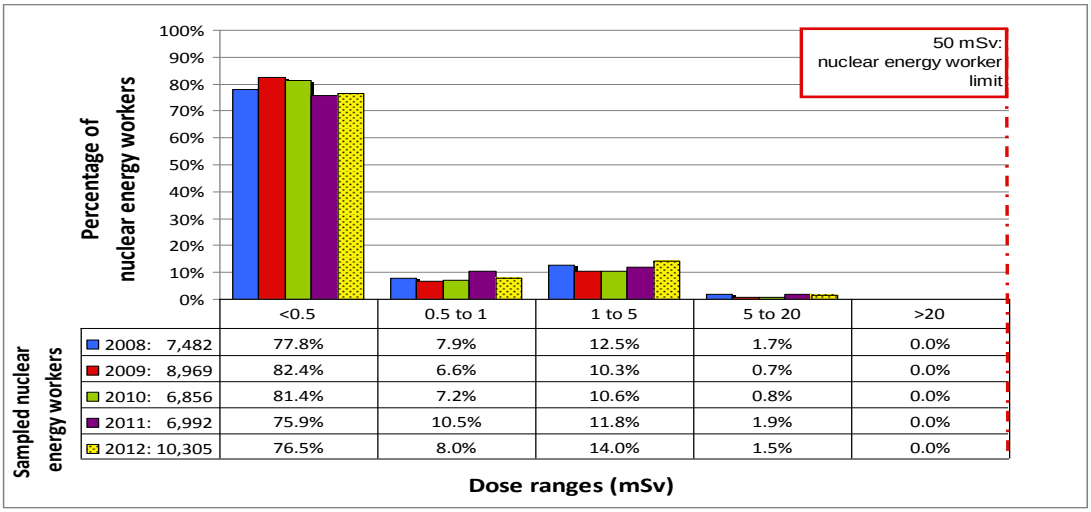


FIG. 1. Sector-to-sector comparison – Percentage of nuclear energy workers and other workers who received an effective dose of less than 1 mSv/y

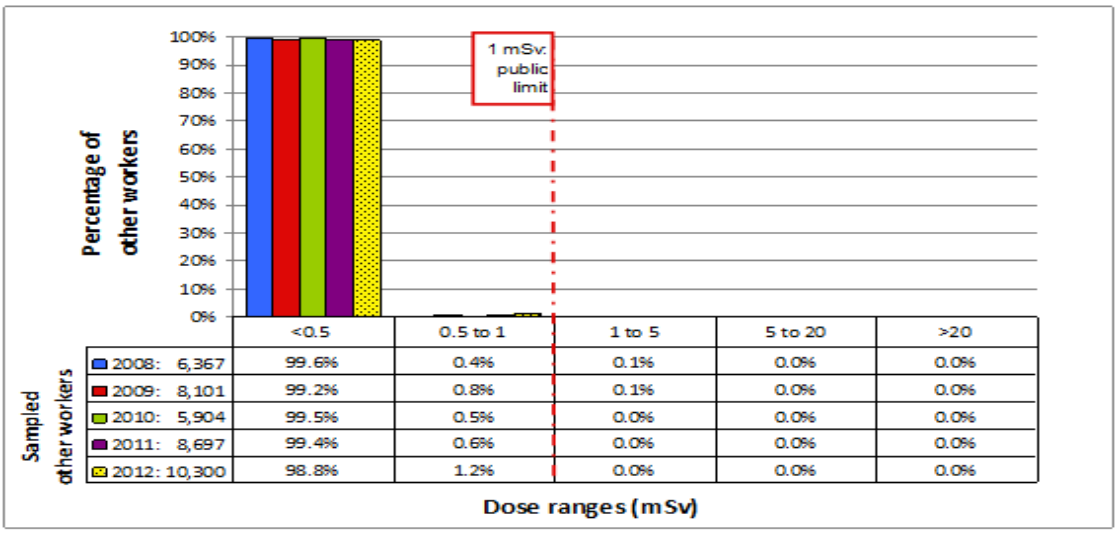
Fig. 2 shows the distribution of doses for NEWs in Canada with all sectors combined for the period of 2008 to 2012. In Canada, the external whole body dose limit for NEWs is 50 mSv per year or 100 mSv average over five years.



Sampled nuclear energy workers	<0.5	0.5 to 1	1 to 5	5 to 20	>20
2008: 7,482	77.8%	7.9%	12.5%	1.7%	0.0%
2009: 8,969	82.4%	6.6%	10.3%	0.7%	0.0%
2010: 6,856	81.4%	7.2%	10.6%	0.8%	0.0%
2011: 6,992	75.9%	10.5%	11.8%	1.9%	0.0%
2012: 10,305	76.5%	8.0%	14.0%	1.5%	0.0%

FIG. 2. All sectors combined – annual effective doses to nuclear energy workers in Canada

Fig. 3 illustrates the distribution of doses for persons who are occupationally exposed but are not considered as nuclear energy workers (NEWs), known as "other workers". In Canada, the external whole body dose limit for other workers is 1 mSv/y, the same as the public dose limit.



Sampled other workers	<0.5	0.5 to 1	1 to 5	5 to 20	>20
2008: 6,367	99.6%	0.4%	0.1%	0.0%	0.0%
2009: 8,101	99.2%	0.8%	0.1%	0.0%	0.0%
2010: 5,904	99.5%	0.5%	0.0%	0.0%	0.0%
2011: 8,697	99.4%	0.6%	0.0%	0.0%	0.0%
2012: 10,300	98.8%	1.2%	0.0%	0.0%	0.0%

FIG. 3. All sectors combined – annual effective doses to other workers in Canada

4. DISCUSSION

The information presented in these three figures clearly demonstrates that the use of nuclear substances and prescribed equipment in Canada is carried out in a safe manner and that licensees are taking adequate measures to protect occupationally exposed workers in this country.

From Fig. 1, it can be seen that, on average, approximately 90% of all workers in Canada who work with nuclear substances and prescribed equipment (which includes devices such as portable and fixed nuclear gauges, radiography cameras, self-shielded irradiators and cancer therapy machines) are reporting an annual whole body exposure of less than 1 mSv/a. There is little variation of this value over the period of 2008 to 2012. In the medical sector, there was a slight decrease in the number of workers reporting less than 1 mSv/y and this is attributed to the larger number of NEWs who were sampled in the 2012 report from the diagnostic and therapeutic sub-sector as compared to previous years (1963 workers in 2012 as compared to an average of approximately 650 workers in years prior). The number of workers in the commercial sector reporting less than 1 mSv/y increased slightly over the previous year and continues a slow trend upward, indicating increased attention to radiation safety, although more time will be required to demonstrate that this is an ongoing trend.

The information provided in Fig. 2 shows that the annual effective doses to NEWs, those persons who are expected to receive an exposure greater than 1 mSv/y as a result of their work, are well below the CNSC regulatory limit of 50 mSv/y. On average, close to 80% of workers for the period 2008 to 2012 have recorded an exposure of less than 0.5 mSv/y. Generally, only 1-2% of all monitored workers are reporting an effective whole body dose in the range of 5 to 20 mSv/y. This demonstrates that workers who are expected to receive a significant radiation exposure as a result of their occupation are adequately protected.

A review of Figure 3 clearly demonstrates that the effective doses recorded by other workers are well below the CNSC public dose limit of 1 mSv/a. The majority of workers in this category recorded effective exposures of less than 0.5 mSv/y, with only about 1% recording effective whole body exposures in the range of 0.5 to 1 mSv/y.

This limited review of information on the management of radiation exposures as a result of the use of nuclear substances and prescribed equipment demonstrates that Canada has an effective regulatory program that encourages licensees to maintain exposures consistent with the ALARA principle. Through a comprehensive licensing process and a system of compliance verification inspections, the CNSC has been able to effectively discharge its responsibilities under Canadian legislation, which include the protection of health and safety of people..

The annual safety performance reports, available on the CNSC website, provide a wealth of additional information regarding the performance of nuclear substance and prescribed equipment licensees in Canada. Information on dose results, compliance inspections and events reported to the CNSC are analysed for each sector (medical, industrial, commercial, academic) as well as by various sub-sectors. For example, the medical sector is further divided into the following sub-sectors: diagnostic and therapeutic nuclear medicine, radiation therapy and veterinary nuclear medicine.

The annual safety performance reports represent a comprehensive evaluation of a disparate collection of licensees in Canada for whom the use of nuclear substances and prescribed equipment is, in most cases, incidental to the work being performed by the licensee. The CNSC is not aware of any other regulatory authority in the world that produces a similarly all-inclusive report on the performance of these licensed activities. For completeness, the CNSC website also contains additional annual reports covering the operating performance of nuclear power plants and annual reports for other nuclear facilities in Canada, as well as reports on the application of controls for sealed sources.

5. CONCLUSION

The use of nuclear substances and prescribed equipment in Canada is well controlled and safe, as demonstrated by high safety performance by the nuclear industry against regulatory requirements. The annual safety performance reports for the use of nuclear substances and prescribed equipment in Canada are comprehensive and detailed in their analysis of the performance of these licensees.

The CNSC properly exercises its regulatory authority and achieves its mandate to protect the health and safety of Canadians. As a result, Canadians can have a high degree of confidence that the application of nuclear substances and prescribed equipment in various locations across Canada is carried out in a safe manner.

REFERENCES

[1] CANADIAN NUCLEAR SAFETY COMMISSION, Nuclear Substances in Canada: A Safety Performance Report for 2012. Ottawa, Canada, March (2014).
[2] CANADIAN NUCLEAR SAFETY COMMISSION, Nuclear Substances in Canada: A Safety Performance Report for 2011. Ottawa, Canada, March (2013).
[3] CANADIAN NUCLEAR SAFETY COMMISSION, Nuclear Substances in Canada: A Safety Performance Report for 2010. INFO-0826. Ottawa, Canada, March (2012).
[4] CANADIAN NUCLEAR SAFETY COMMISSION, Nuclear Substances in Canada: A Safety Performance Report for 2008-2009. INFO-0821. Ottawa, Canada, September (2011).

OCCUPATIONAL RADIATION PROTECTION STRUCTURE IMPLEMENTED AT BRAZILIAN RESEARCH INSTITUTE OF RADIATION PROTECTION AND DOSIMETRY – IRD/CNEN

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Abstract

The Institute of Radiation Protection and Dosimetry (IRD/CNEN) is one Institute of the National Commission of Nuclear Energy (CNEN-Brazil) with a staff of about 300 professionals and is considered to be one of the most important bodies in Latin America dedicated to radiation protection, dosimetry and metrology of ionizing radiation. The IRD´s mission is to act with excellence in these areas generating and disseminating knowledge and technology for the safe use of the ionizing radiation and nuclear technology, in order to improve the quality of life in the Country. With the CNEN restructuring and the new edition of CNEN/DRS Resolution 112/2011, all CNEN Research Institutes must undergo a new nuclear licensing process. In this context, IRD/CNEN performed an update in its Radiation Protection Policy to meet regulatory requirements and implement best practices in routine radiation protection in its activities. This paper shows the occupational radiation protection structure implemented at Brazilian Research Institute of Radiation Protection and Dosimetry for the protection of radiation workers, general worker, contractors, and professional fellows, as well as employees, students, visitors, members of the general public, and the environment.

1. INTRODUCTION

The Institute of Radiation Protection and Dosimetry (IRD/CNEN), that is one Institute of the National Commission of Nuclear Energy (CNEN-Brazil), is considered to be one of the most important bodies in Latin America dedicated to radiation protection, dosimetry and metrology of ionizing radiation. The IRD´s mission is to act with excellence in these areas generating and disseminating knowledge and technology for the safe use of the ionizing radiation and nuclear technology, in order to improve the quality of life in the Country.

With the CNEN restructuring and the new edition of CNEN/DRS Resolution 112/2011, all CNEN Research Institutes must undergo with a new nuclear licensing process. A Radiation Protection Plan to meet regulatory requirements and implement best practices in routine radiation protection in its activities was performed in accord to new regulations and international recommendations.

The occupational radiation protection structure implemented at IRD/CNEN is basically the General Plans with general guidelines and Occupational Radiation Protection Specific Plans for the Divisions with their laboratories. With this structure the IRD/CNEN is enabled to achieve state of the art in nuclear and environmental licensing process and follow the radiation operation according to international recommendations too. Additionally, the methodology developed may be deployed by other CNEN Institute.

1.1. The Institute of Radiation Protection and Dosimetry – IRD/CNEN

The Institute of Radiation Protection and Dosimetry (IRD), that is one Institute of the National Commission of Nuclear Energy (CNEN-Brazil), was officially created in 1972 with the aim of environmental monitoring, calibration of area monitors and training of

professionals in the medical area in radiation protection and clinical dosimetry. Nowadays, with a staff of about 300 professionals, it is considered to be one of the most important bodies in Latin America dedicated to radiation protection, dosimetry and metrology of ionizing radiation. The IRD´s mission is to act with excellence in the areas of radiation protection, dosimetry and metrology, generating and disseminating knowledge and technology for the safe use of the ionizing radiation and nuclear technology, in order to improve the quality of life in the Country. The basic research activities carried out by the IRD have contributed to the development of knowledge, new technologies and solutions of problems of radiological protection, dosimetry and metrology of ionizing radiation. Due to the specificity and quality of the work developed, the IRD is considered to be a national reference center and a source of knowledge, strengthening the activities of education and qualification. The IRD has also a significant participation in normative and technical committees in Brazil and abroad.

The IRD´s main objectives are:

(a) To carry out scientific research and to develop technologies in the areas of radiation protection and metrology of ionizing radiation specifically on medical physics, dosimetry, radioecology, radiation biophysics and metrology.
(b) To assure the traceability of measurements for the units related to the ionizing radiation of the International Systems to the National and International Metrology Networks. The IRD is recognized by the International Atomic Energy Agency (1976, IAEA) as a Secondary Standard Dosimetry Laboratory (SSDL). Through its Laboratory for Metrology of Ionizing Radiation, the IRD is assigned by the National Institute of Metrology, Normalization and Industrial Quality (1989, INMETRO) as the Brazilian representative of the International System of Metrology of Ionizing Radiation.
(c) To maintain a team trained and integrated to respond to radiological and nuclear emergency situations in Brazil. The IRD is the coordinator of the World Health Organization Collaborating Center for Radiation Protection and Medical Preparedness for Radiation Victims (1999, WHO). The IRD is also nominated by the IAEA the "National Warning Point" and the "National Competent Authority for Accidents Abroad" (2000, IAEA).
(d) To promote the qualification of human resources in the areas of radiation protection and metrology of ionizing radiation. The education and training activities developed at IRD are mainly the Short Courses and Post-Graduate Program of Master degree in Radiation Protection and Dosimetry that is implemented in the areas of Radiation Biophysics, Medical Physics, Radioecology and Metrology.
(e) To offer services of calibration, dosimetry and assays. The IRD offers services of photographic, thermo luminescent, and biological dosimetry. Methods of bio-analysis (radiochemistry and whole body counting) are still offered to verify the internal contamination due to radionuclide in specific parts of the human body. The IRD also offers assay services for radionuclide determination in food and other environmental samples. The IRD belongs to the international measurements system associated to the Comprehensive Nuclear Test Ban Treaty (1996, CTBT).

To meet its activities, the technical organization of the IRD/CNEN is basically based on seven technical Divisions: Metrology, Radiation Protection, Dosimetry, Medical Physics, Emergency, Radiation Industry and Safeguard and their 68 laboratories. These laboratories are considered as supervised and controlled areas where radioactive materials are handled in the form of sealed and unsealed radioactive sources and radiation emission equipment. The works carried out in laboratories involve exposure to radionuclides emitting α, β, γ, neutron sources and generators of ionizing radiation.

2. OCCUPATIONAL RADIATION PROTECTION STRUCTURE

For purposes of the Licensing each one of the seven Divisions of the IRD/CNEN, with its laboratories, was classified as a Radiative Installation in accordance with the statements of the CNEN/DRS Resolution 112/2011. The classification was done based on the inventory of radioactive sources and the risk of these sources. The Metrology Division with the calibration and unsealed laboratories was classified as highest risk installation.

The occupational radiation protection structure is basically divided in two main Plans: General Plans with general guidelines and Occupational Radiation Protection Specific Plans for the Divisions with their laboratories as shown in Fig. 1.

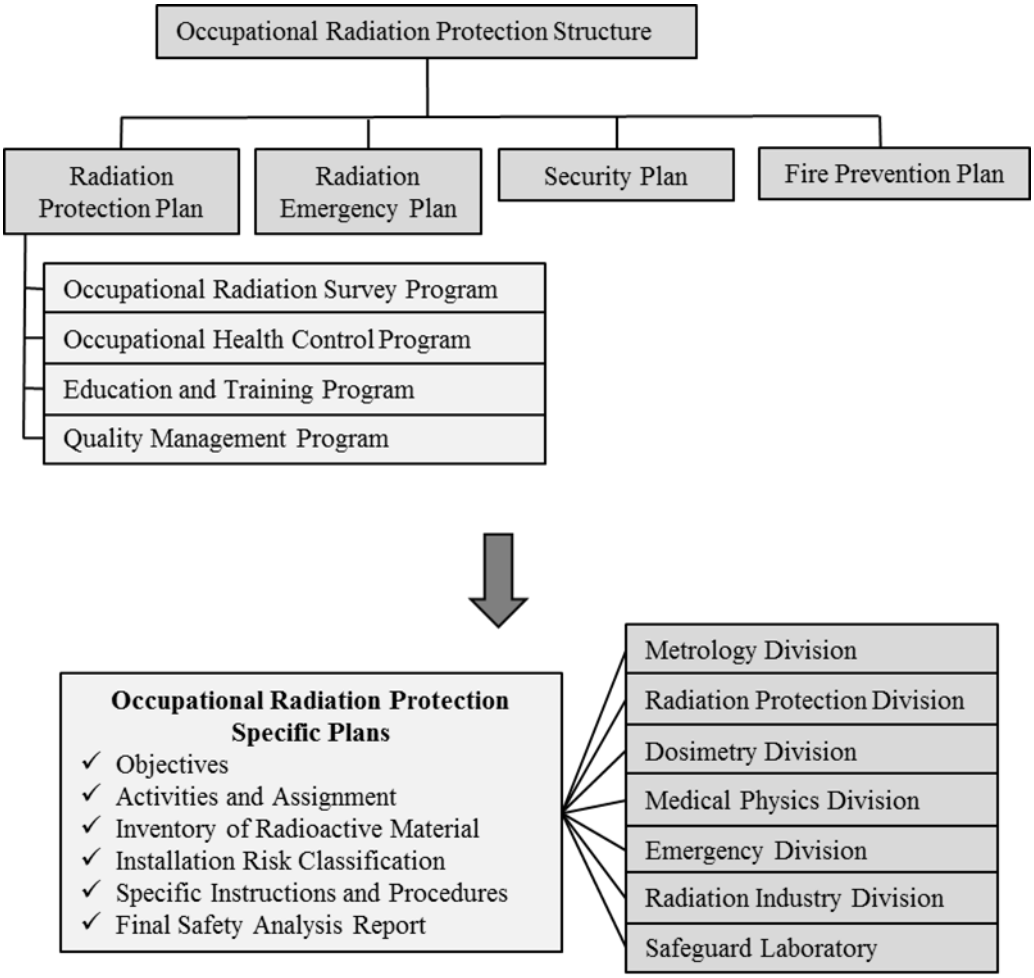

FIG. 1. IRD/CNEN Occupational Radiation Protection Structure

2.1 General Plans

The General Plans are formed by four Plans that give the general guidelines:

(a) Radiation Protection Plan that gives the main guidelines to safety and radiation protection and has the following Programs: Occupational Radiation Survey Program; Occupational Health Control Program; Education and Training Program; Quality Management Program;

(b) Radiation Emergency Plan that gives the main guidelines to preparedness and response of the possible emergencies;
(c) Security Plan that gives the main guidelines to security of radioactive sources inside the Institute; and
(d) Fire Prevention Plan that gives the main guidelines to prevention and control fire.

The Radiation Protection Plan has the following main guidelines:

i. Guidelines to Occupational Radiation Control: consists of aspect external and internal individual monitoring, area classification and monitoring, external contamination and individual dose control and evaluation.
ii. Guidelines to Occupational Health Control: every radiation worker must be submitted to an annual health control, based on the occupational health principles.
iii. Guidelines to Education and Training: every radiation worker must be submitted to an annual radiation protection and safety training. This training must have two steps: a general training with three levels: basic, intermediate and complete, depending on the type of work; and a specific training related to the type and radioactive source used by the worker.
iv. Guidelines to Radiation Emergency Response and Preparedness Situation: the abnormal and emergency events that could be happened inside the IRD laboratories must be postulated; and the procedures to response which one must be written and trained by the radiation workers.
v. Guidelines to Control of Radiation Protection Records: a specific control of records must be established and annually updated.
vi. Guidelines to Quality Management: the IRD/CNEN radiation protection system must be inside the IRD/CNEN Quality Management

2.2 Occupational Radiation Protection Specific Plans

These specific plans cover the objectives, activities and assignment, inventory of radioactive material, the risk classification, specific instructions and procedures and the final safety analysis report for its seven technical Divisions: Metrology, Radiation Protection, Dosimetry, Medical Physics, Emergency, Radiation Industry and Safeguard.

The Occupational Radiation Protection Specific Plans have the objective to establish the radiation protection system for each Division. The Specifics Plans also define the basic requirements that must be followed by the Division's radiation workers to perform their radiation laboratory activities safely.

3. CONCLUSIONS

The IRD/CNEN radiation protection policy with the Structure, the General Plans and Specific Plans showed very useful to attend the CNEN/DRS Resolution 112/2011 for licence the radioactive installations. The documents were approved by the CNEN Regulatory Authority and now necessary action to implement this structure has been performed by the IRD Radiation Protection Officer.

REFERENCES

[1] IRD/CNEN, Conceptual Structure to IRD/CNEN Licencing, IRD/CNEN (2013).

CHALLENGES IN IMPLEMENTING OCCUPATIONAL RADIATION PROTECTION PROGRAM IN TANZANIA

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Abstract

The Atomic Energy Act, 2003, provides the regulatory framework for occupational radiation protection in Tanzania. Attempts to fulfill the requirements of the International safety standards have been ongoing for past 30 years with notable progress achieved. However, the national occupation radiation (ORP) infrastructure is still inadequate due to some challenges. These include some existing gaps in regulatory framework, limited human and financial resources as well as limited awareness among various stakeholders. Other challenges include limited competency as well as the absence of active radiation protection association. More national efforts are needed to improve the status of ORP in the country. As a starting point, the occupational radiation appraisal service mission has been requested and approved by IAEA. This should be followed by a national forum of key stakeholders to formulate strategies and streamline related actions. There is also a need to strengthening education and training programs in radiation protection as well as identifying a national financing strategy for the desirable sustainability. The commitments of the government, regulatory authority, employers and licensees as well as that of the employees appear to be critical towards an improved ORP status

1. INTRODUCTION

The Atomic Energy Act, 2003, provides the regulatory framework for occupational radiation protection (ORP) in the United Republic of Tanzania. Under this law, the Tanzania Atomic Energy Commission (TAEC), is the sole regulatory authority for practices involving ionizing radiation [1]. Currently there are 472 authorized centres involved with ionizing radiation practices in medicine, industry, research and teaching. Figure 1 summarizes typical distribution of these centres. A total of 1400 workers employed in 280 centres, have the potential to incur measurable radiation exposure and are therefore participating in individual monitoring of external radiation as shown in Figure 2. Over nearly 30 years, the country has been attempting to implement the ORP requirements in order to make the national ORP infrastructure compatible with International requirements [2].

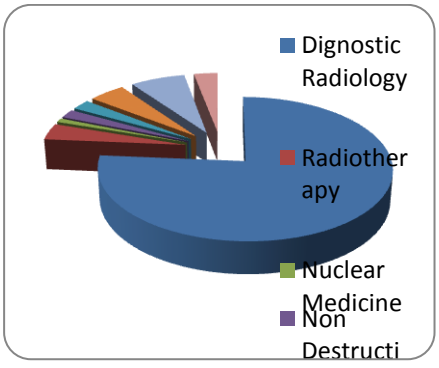

Fig. 1. Distribution of authorized practices

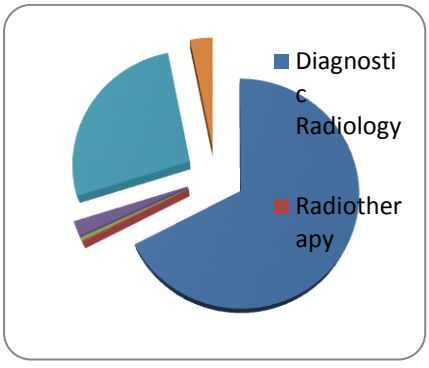

Fig. 2. Distribution of radiation monitored persons

As a result of this implementation, some notable achievements have been recorded. However, full compliance to the international requirements is not yet been fulfilled. The aim of this paper is to describe the challenges facing the implementation of ORP programme in Tanzania. Potential solutions are identified for improvement.

2. OVERVIEW OF THE NATIONAL STATUS OF OCCUPATIONAL RADIATION PROTECTION

According to IAEA, a desirable ORP programme should include seven main elements in order to function effectively. The elements include the regulatory infrastructure for occupational exposure, individual monitoring for external radiation sources (IMERS), and individual monitoring for intakes of radionuclides (IMIR) [2]. Other elements are workplace monitoring (WM), requirements for service providers, the implementation of the requirements by the end users and ORP of natural radiation sources. Apart from the availability of legislation and regulatory authority, other major achievements of the National ORP programme include the availability of key trained technical staff, basic equipment as well as basic technical services such as dosimetry, work place monitoring, radioactive analysis, repair and maintenance and training at regulatory authority. IMERS has also recorded achievements in personnel dosimetry comparisons [3, 4]. At the level of end users, major achievements include availability of authorization details, radiation safety officer (RSO) and protective gears as well as satisfactory administrative controls i.e. designated areas, local rules and supervision.

However, there are still some limitations in each of seven ORP elements. First, is inadequate legal framework to enforce ORP requirements. For instance, there are no provisions in the Act and regulations that requires the approval of technical services, certification or their accreditation. In some cases, the Act requires the regulatory authority to provide technical services, which can potentially result to conflict of interest. The safety guides that could assist regulatory compliance by end users are also still in draft form. Second, is the non availability of regulations on the control of radiation exposure from natural sources. Third, is inadequate infrastructure for individual monitoring and work place monitoring. The coverage of individual monitoring service of external radiation, which is provided by regulatory authority, is still not yet full due the insufficient number of personal dosimeters. The service is also not formally approved; performance testing is partially done and type testing has not yet been started. In addition, there is no quality system in place according to ISO 17025 standard, although regulations emphasize on quality assurance. Currently, there is no internal dosimetry services in the country despite some existing internal exposure potentials in some practices for instance in I-131 therapy and mines. On work place monitoring, most relevant licensees do not perform this activity due to absence of equipment or/and commitment. In places where the equipment is available, the measurements lack traceability with inadequate quality management systems. WP services are mainly performed by the regulatory authority during radiation safety inspections. The fourth and major limitation is inadequate fulfillment of the requirements for end users. Despite the fact that Regulations assign prime responsibility of radiation safety to end users, few licensees fulfill these requirements. For instance there is no formal cooperation between employers although some workers are part time staff in more than one facility. There is also inadequate implementation of RPP according to the international standards and often the management structure for radiation safety is inadequate. Furthermore, there no formal intervention plans in emergency as well as non active health surveillance and quality assurance programmes.

3. CHALLENGES OF IMPLEMENTING ORP PROGRAMME

The main challenges facing the implementation of ORP in Tanzania can be described as follows:

3.1. **Limited trained and qualified human capacity**

There are very few universities that provide education related to radiation protection and therefore sound radiation protection foundation is inadequate. Except few workers that have received education abroad, many others at regulatory authority and end users received general science or engineering education. Afterwards, some joined postgraduate diploma course in radiation protection offered by IAEA and the majority participated in IAEA fellowship and short training programmes. This challenge extends also to the regulatory side as there is inadequate capability to regulate some of nuclear technology applications e.g. in oil and gas industry, radiotherapy and anticipated uranium mining and milling activities. Recently, there has been an initiative to establish the Tanzania National Network for Nuclear Education, Science and Technology (TAN-NEST), which stands at a better chance to improve the national education status in nuclear science and technology (including radiation protection).

This challenge leads to limited competency both at regulatory body and at end users. Considering the support from the international community to the country, the progress in implementing ORP programmes appear to be at low pace. There is a core of few trained persons in the country but such personnel tend to be overwhelmed by many types of radiation protection tasks that require different specific specializations. Because the number of trained workforce is small, engagement in multiple fields tends to be the course with consequence of limited competency, which leads to ineffective implementation of ORP plans. Even for those implemented plans, the quality of implementation becomes less satisfactory. This can be due to changes in staff interest following internal movement, change of career position, commitment, socio-economic pressures which lead to laxity or inertia to implement ORP programme as required. Consequently, the rate of transformation of the gained knowledge and skills during training courses to the desirable outputs tends to be low.

3.2. Inadequate financial support

Like other programs, ORP need adequate funding to enable the implementation of some related work programs. Regulatory authority requires adequate funding in order to enforce regulations while the end users also require the same to comply with regulations. Inadequate funding due to developing economy is a limiting factor to procure and maintain equipment hence less sustainability of the radiation protection infrastructure. Due to limited funding, there is a tendency to assign low priority on ORP issues in comparison to other issues and consequently ORP programs suffer.

3.3. Low awareness on ORP issues

Low awareness on ORP issues apparently exists at some end users where there is reluctance to comply with the relevant regulations. This is seen in implementing ORP issues that even do not require financial resources. Issues like cooperation among employers, classification of work places, policy and procedures writing, records maintenance as well as that of management structure for radiation safety are not implemented at some workplaces because of low awareness. Consequently this has resulted to assigning low priorities on radiation protection issues in comparison to other issues especially when financial issues come in. Long term low awareness eventually leads to weak safety culture which is a kill to the implementation of ORP programs.

3.4. Absence of active radiation protection association

The absence of radiation protection professional society also bears a negative consequence on the implementation of ORP programs. The role of government, regulatory authority and that of end users on radiation protection are clearly defined. The role of professional society would be to provide independent professional advice or guidance in the light of any conflicting radiation protection decisions or operations of the regulatory authority or end users. As a result sometimes, lack of mutual trust tends to lead to laxity/inertia in implementing ORP requirements. Some personnel in the country have joined the Eastern Africa Association for Radiation Protection (EAARP), which is affiliated to the International Radiation Protection Association (IRPA). However, the activeness of this association in the country is still at initial stage.

4. DISCUSSION

The implementation of standard ORP requirements is a shared responsibility between various stakeholders [2,6]. The identified challenges suggest that the related solutions can be initiated locally for sustainable ORP infrastructure. Key stakeholders should hold discussions on these challenges and a way forward agreed upon. As seen, there is a need to improve the provision of financial resources and the national training programs in radiation protection issues. There is also a need to improve the awareness and commitments of key stakeholders i.e. government, regulatory authority, end users and workers. In addition, the engagement of professional associations on ORP issues might expedite the implementation of ORP and hence comply with international requirements. Recognizing the difficulties that may be encountered in prioritize actions, the occupational radiation appraisal service (ORPAS) mission has been requested and approved by IAEA. It is suggested that after this mission, a national forum of key stakeholders be held to have open discussions on the need to formulate strategies and streamline related actions Such approach has proved to be successful during the initial stakeholders' efforts to establish TAN-NEST (Tanzania Network for Nuclear Education, Science and Technology), which is potential to improve the situation assuming commitment at government and top managements.

5. CONCLUSION

The challenges facing the implementation of ORP program has been presented. As a starting point, an ORPAS mission is to be conducted to prioritize actions. This would be followed by discussions among key stakeholders in the country in strategizing and streamline ORP actions. In our view, although the participation of various stakeholders is necessary to achieve positive results, the commitments of the governments, regulatory authority, employers and workers appear to be central to the quick results.

REFERENCES

[1] THE UNITED REPUBLIC OF TANZANIA. The Atomic Energy Act No. 7,2003 Tanzania Government Printer, Dares Salaam, 2003.

[2] INTERNATIONAL ATOMIC ENERGY AGENCY. Radiation Protection and safety of Radiation sources: International Basic Safety Standards. Interim Edition. General Safety Requirements, Part 3 No. GSR Part 3 (Interim), IAEA, Vienna,2012.

[3] W.E. MUHOGORA, E. BYORUSHENGO, U.S. LEMA, *et al.* Occupational radiation exposure in Tanzania (1996-2010): Status and Trends.*Radiat.Prot.Dosim*. 2013; 153(4):403-10.

[4] Z.MSIMANG, R.CRUZ-SUAREZ AND J. ZEGER. Intercomparison of measurements of quantity personal dose equivalent, H*p*(10) in photon fields in the African region. *Radiat. Prot. Dosim.* 2011; 90(2):143-148

[5]. INTERNATIONAL ATOMIC ENERGY AGENCY. Government, Legal and Regulatory Framework for Safety. General safety Requirements Part 1.No. GSR Part 1, IAEA, Vienna, 2010

Session 2: Dose assessment of occupational radiation exposures

A COMPARATIVE STUDY BETWEEN MONDAL SOFTWARE AND A CONSTRUCTED MODEL FOR CALCULATING INTERNAL EXPOSURE OF SOME RADIONUCLIDES

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Abstract

The present work provides a computer program with C-language based on a series of exponential functions for nine radionuclides. The radionuclides under investigation are: cobalt, iodine, cesium, strontium, ruthenium, radium, thorium, plutonium and uranium. The committed effective dose has been calculated by our model so as to obtain the urinary and faecal excretion rates for each radionuclide. The said model is further validated by a comparison with the widely spread Mondal software and a simulation program. The results obtained show a harmony between the Mondal package and the model we have constructed. There were a good agreement between the software and the constructed model. It has shown a goodness-of-fit (R^2 above 0.97).

1. INTRODUCTION

Occupational exposure to ionizing radiation can occur in a range of industries, such as mining and milling; medical institutions; educational and research establishments; and nuclear fuel facilities. Adequate radiation protection of workers is essential for the safe and acceptable use of radiation, radioactive materials and nuclear energy [1]. Internal exposures occur when radionuclides have been inhaled, ingested or otherwise taken into the body through wounds and intact skin. A proportion of inhaled material will eventually be swallowed. Radionuclides inside the body are called internal emitters [2]. Individual monitoring for internal exposure is based on the direct measurement of radionuclides in excreta. Removal of deposited material from the body occurs principally by urinary and faecal excretion [3, 4]. The biological samples used for the estimation of intake and the assessment of internal exposure are most commonly urine and faeces, although breath and blood or other samples can be used in special cases [5, 6]. For any radionuclides, the fraction excreted each day changes rapidly with time after intake. For excreta monitoring in particular it is recommended that the daily excretion graphs should be inspected for evidence of a rapid decrease of daily excretion in the few days after intake. If the activity of the second samples is very much less than that of the first, this suggests that either the first sample was contaminated with un-metabolized material or that the first sample was collected shortly after intake [7]. The present work aimed to use the obtained daily excretion graphs from constructed program to assess intake of radionuclides and committed effective doses. From the daily excretion graphs one obtains the excretion rate "M" at time "t" after intake. Assessment of the intake of radionuclides is by applying the following relation [8]:

$$Intake = \frac{M}{m(t)}$$

Where m(t) - represents the measured quantity at time t days after intake of one Bq of isotope. The committed effective dose from an estimated intake may then be calculated using the dose coefficient (committed effective dose per unit intake) of the radionuclide of interest specified for inhalation or ingestion as appropriate. Dose coefficients have been calculated for hundreds of different radionuclides. For occupational exposure, the committed effective dose to the worker is integrated over the fifty years following the intake (E_{50}), irrespective of the age of the adult at time of intake. This assessment of exposure may then be compared against relevant effective dose equivalent limits.

2. MONDAL SOFTWARE

Mondal software is distributed by National Institute of Radiological Sciences (NIRS) Anagawa, Japan. The personal computer based software: MONDAL used to provide a useful tool for dosimetrists involved in radiation protection to assess intakes of radionuclides and the resulting tissue equivalent and effective doses from bioassay measurements for both workers and all members of the public [9].

3. CONSTRUCTED PROGRAM

Mathematical representation of the transfers within the body is required in order to establish the relationship between intake and excretion rate. The clearance of inhaled material from compartment is described by a set of interlinked first order differential equations, used to assess the urinary and faecal excretion along interval of time [10, 11]. This differential equations used in this program to draw graphs of activity in daily urinary excretion as a function of time after intake of radionuclides. Data are provided for periods up to 1000 days, the resulted data are applied for assessment of intake and dose calculations. Graphical data are not provided for times less than 1 day after intake.

4. RESULTS AND DISCUSSION

The next figures show the daily excretion graphs of Mondal software and the corresponding graphs of the constructed program. For excreta monitoring in particular it is recommended that the graphs of daily excretion should be inspected for evidence of a rapid decrease in the few days after intake.

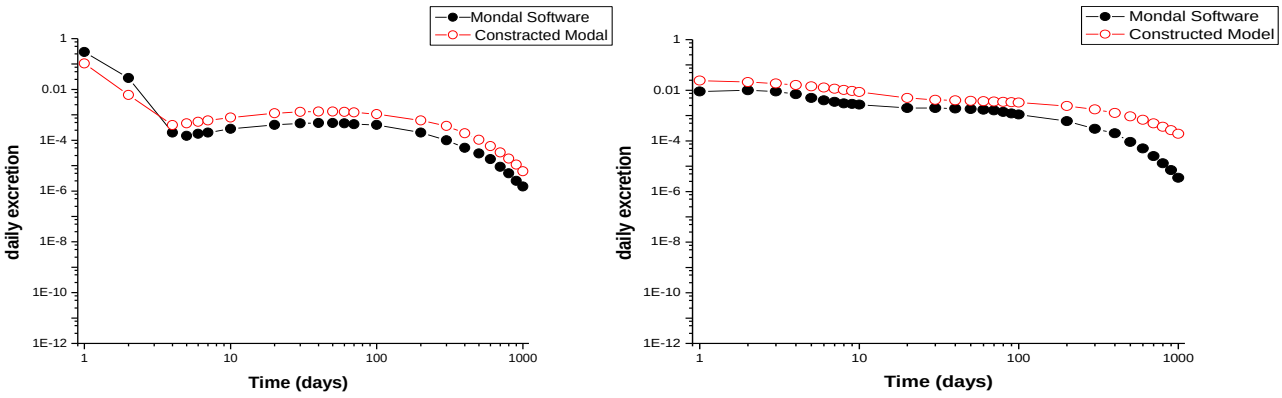

FIG. 1. Iodine excretion in urine

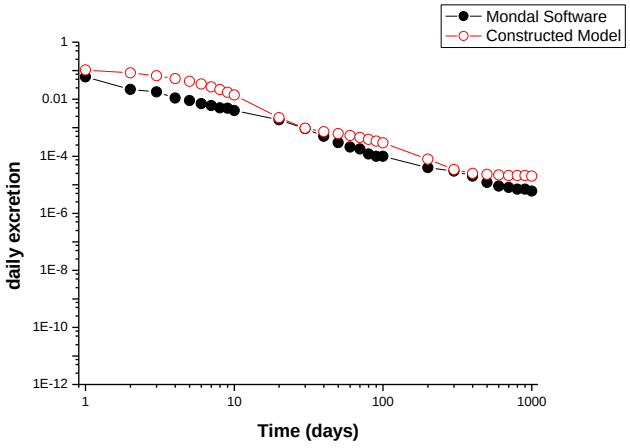

FIG. 2. Cesium excretion in urine

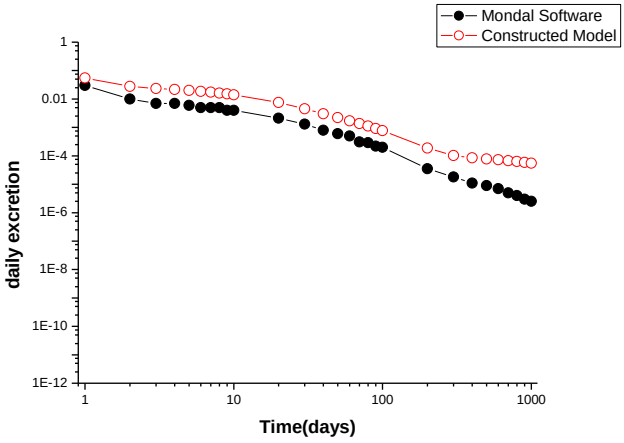

FIG. 3. Strontium excretion in urine

FIG. 4. Ruthenium excretion in urine

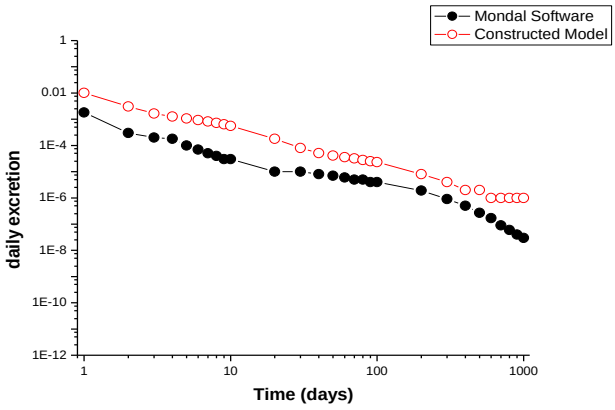

FIG. 5. Radium excretion in urine

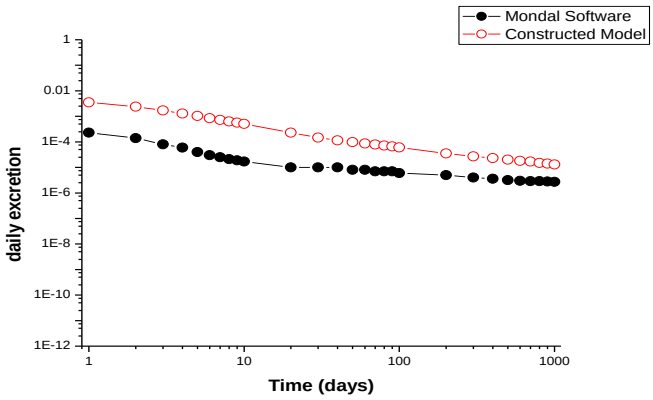

FIG. 6. Plutonium excretion in urine

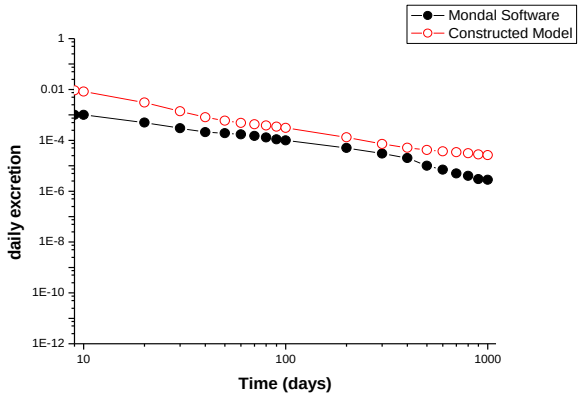

FIG. 7. Cobalt excretion in urine

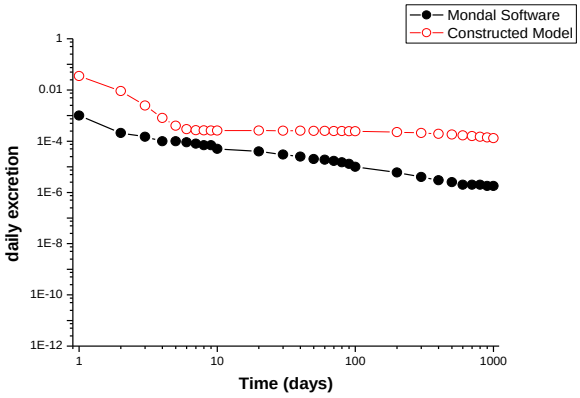

FIG. 8. Thorium excretion in urine

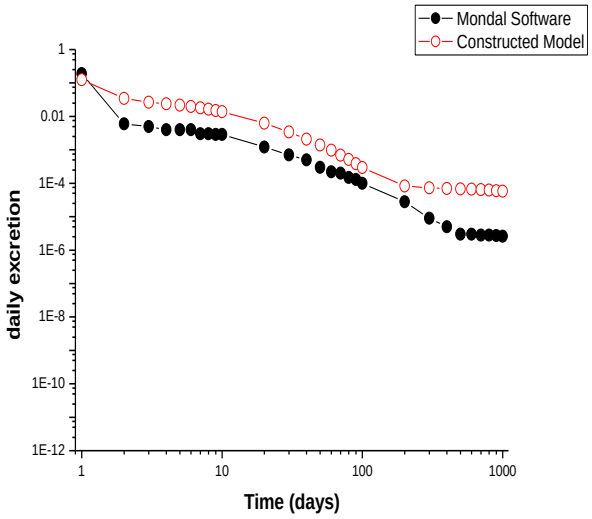

FIG. 9. Uranium excretion in urine

The obtained data of the software and the constructed model were fitted using t-test. The R^2 coefficients and the sums of squares (which represent the goodness of fit) were done. The within- case standard deviation was used to estimate the magnitude of the difference between the models [12]. A ρ value lower than 0.05 was considered significant. It has been shown that: iodine provided the best fit (R^2 0.99 [SD 0.00288]), followed by cobalt (R^2 0.98 [SD 4.13×10^{-4}]), radium (R^2 0.97 [SD 5.58×10^{-5}]), thorium (R^2 0.96 [SD 3.45×10^{-5}]), cesium, radium (R^2 0.95 [SD 6.58×10^{-4}]), plutonium (R^2 0.95 [SD 1.04×10^{-5}]), ruthenium (R^2 0.91 [SD 0.0018]), uranium (R^2 0.86 [SD 0.0134]), strontium (R^2 0.81 [SD 0.00527]). These differences were significant in the ANOVA test and $\rho < 0.01$ in all tests.

5. CONCLUSION

The results show that the goodness of fit of the urine excretion of the nine radionuclides under investigation is remarkably high. The first order differential

equations allow fitting from exposed persons. The constructed model can be applied to obtain faecal excretion curves too.

REFERENCE

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of occupational exposure due to intakes of radionuclides, Safety Standards Series, No. RS-G 1.2, IAEA, Vienna (1999).
[2] KARPAS, Z ; PAZ-TAL, O ; LORBER, A ; SALONEN, L; KOMULAINEN, H; AUVINEN, A; SAHA, H., KURTTIO, P., Urine, hair, and nails as indicators for ingestion of uranium in drinking Water, Health Phys. 88 (2005) 229-242.
[3] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Individual monitoring for intakes of radionuclides by workers: Design and interpretation, Publication No. 54 (1988).
[4] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Limits for intakes of radionuclides by workers, Publication No. 30, Part 1 (1979).
[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of occupational exposure due to intakes of radionuclides, Safety Standards Series No. RS-G 1.2, IAEA (1999).
[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Methods for measuring radionuclides in the human body, Safety Series No. 114, IAEA (1996).
[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Indirect methods for radionuclides causing occupational exposure, Safety Reports Series No. 18, IAEA (2000).
[8] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Individual monitoring for intakes of radionuclides by workers: Design and interpretation, Publication No. 54 (19988).
[9] http://www.nirs.go.jp.8080//anzendb/RPD/gpnd.php
[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Methods for assessing occupational radiation doses due to intakes of radionuclides, Safety Reports Series No. 37, IAEA, Vienna, (2004).
[11] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, "Individual monitoring for internal exposure of workers, Replacement of ICRP Publication 54, ICRP Publication No. 78 (1997).
[12] GUILLERMO M ALBAICETA, ESTEBAN GARCIA AND FRANCISCO TABOADA, Comparative study of four Sigmoid models of pressure-Volume curve in acute lung Injury. BioMedcal Engineering, OnLine, 6:7 (2007).

EXTREMITIES AND EYE LENS DOSIMETRY IN ROMANIA: CHALLENGES AND DEVELOPMENTS

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Abstract

This article presents the means by which requests and recommendations regarding the determination of the operational quantities Hp(0.07) and Hp(3) were implemented within Dozimed Laboratory. Additionally, it deals with the challenges encountered during the implementation of these new services.

1. INTRODUCTION

The prospect of developing methodologies necessary to evaluate extremities and eye lens doses was taken into consideration ever since the launch of the EC RP-160 [1] technical recommendations. Nevertheless, the deciding factors in developing the methodologies for determining Hp(3) and Hp(0.07) were the interesting results obtained during the ORAMED project [2, 3]. This motivated a modernization process for the evaluation of extremities and eye lens doses for the IC/IR medical staff in Romania. The first step in completing this process was the approval in 2011 for Hp(0.07) evaluation by the national regulatory body. This process was finalized in 2013 with the approval for Hp(3) evaluation.

2. CURRENT SERVICES PROVIDED BY THE LABORATORY

Dozimed Laboratory was founded in 1999 as a response to the need of improving the quality of Romanian dosimetric services. Until 2010, the photographic dosimeter was the only one in use. Starting with 2009 a modernization process emerged firstly through the introduction of thermoluminescent dosimetry, which was followed by gradually announcing new services such as neutron dosimetry, extremities and eye lens dosimetry.

An important role in completing this modernization process was played by the lessons learned from European laboratories with experience in this field. Our constant participation in the intercomparison exercises organized by EURADOS, as well as our collaboration with its experts in both photon and neutron dosimetry lead to a continuous improvement in the quality of the services provided by our laboratory.

Today, Dozimed laboratory offers the following types of services:

(a) Hp(10) evaluation for photon exposure with the Panasonic thermoluminescent dosimetric system

(b) Hp(10) evaluation for photon exposure with the photographic dosimetric system (Foma film and PTW badge)

(c) Hp(10) evaluation for neutron exposure with thermoluminescent albedo dosimeters (Harshaw 6776 - 4 element card and 8806 holder)

(d) Hp(0.07) evaluation for extremities monitoring with Harshaw DXTRAD

thermoluminescent detectors

(e) Hp(3) evaluation for eye lens monitoring with Harshaw EXTRAD thermoluminescent detectors.

The laboratory monitors approximately 11.000 exposed workers per month from all fields: medicine, industry, nuclear power, security, uranium mining. The distribution of the exposed workers is as follows: 74% are working in medicine, 13% in industry, 7% in uranium mining, 4% in nuclear power generation and 2% in education.

In evaluating Hp(10) due to photon exposure, Panasonic thermoluminescent dosimeters are gradually replacing photographic dosimeters. This process began in June 2013 and is now approaching 50% replacement of the monitored workers. It is planned that the photographic dosimeter will be further used for several years due to its advantages, such as: low cost, information regarding contamination, unique exposure or partial exposure of the dosimeter.
Out of the total monthly dose determinations, 96% are Hp(10) determinations due to photon exposure, 3% are Hp(10) determinations due to neutron exposure and only 1% represent Hp(0.07) and Hp(3) determinations.

2.1. Extremities monitoring services

Services for extremities monitoring with Harshaw thermoluminescent detectors were introduced in 2011. Out of the total number of exposed workers monitored by our laboratory, 9% represents the number of medical staff members involved in the IC/IR and an additional 2% are in nuclear medicine. The number of determinations was, however, limited to a hundred per month. Initially, Harshaw EXTRAD LiF:Mg, Ti [4] dosimeters were used, yet the majority of monitored workers informed us that the ring was uncomfortable and likely to break the surgical glove, leading to difficulties in wearing it. In 2013, extremities dosimeters were replaced with Harshaw DXTRAD LiF:Mg, Ti [5].

Nevertheless, some pro-bono experimental studies were performed annually in order to evaluate the doses received by the medical staff exposed in nuclear medicine and radiology/interventional cardiology. Not even under these conditions are there more than a hundred monthly determinations.

Table I. RESULTS OF THE EXTREMITIES MONITORING AND PRO-BONO EXPERIMENTAL STUDIES
[* MDL = Minimum Detection Limit]

Dose interval [mSv]	IR/IC (%)	Nuclear Medicine (%)	Industry (%)
< MDL*	62.5	43.6	66.9
[MDL, 1]	26.8	8.6	18.4
(1, 5]	5.9	22.9	3.8
(5, 10]	2.5	10.9	1.5
(10, 20]	1.6	6.7	6.0
(20, 42]	0.3	4.6	2.6
(42, 100]	0.3	2.3	0.4
(100, 500]	0.0	0.6	0.4
> 500	0.0	0.0	0.0

The results from the monthly monitoring and from the studies are presented in Tabel I as percentages from the total number of doses evaluated for each category. Most of the results are below the derived monthly limit. The derived montly limit of 42 mSv is calculated as 1/12 from the annual dose limit (500 mSv). The doses exceeding the derived monthly limit were analyzed with the radiation protection officer of each departement and most of them were caused by carelessness of the personnel (dosimeters fell in the radioactive solution, remained inside the glove for a long period of time or were worn for a longer period of time).

2.2 Eye lens monitoring services

Due to the fact that the number of Hp(3) determinations will be limited to a narrow spectrum of professionally exposed medical staff within the IC/IR, the evaluation of these doses is done by using EXTRAD LiF:Mg, Ti detectors, with a 1.5 mm PTFE filter. This assures a filtering equivalent to 3.3 mm of tissue. The detectors are placed on a plastic head band of adjustable size [6].

Irradiations for the type tests were carried out in the secondary standards laboratory of Czech Metrology Institute, Inspectorate for Ionizing Radiation, at photon radiation qualities specified in ISO 4037 [7]. The irradiations were performed on a cylindrical PMMA phantom for Hp(3), water-filled, 20 cm diameter with a wall thickness of 3.5 mm [8]. Conversion coefficients published by the LNHB laboratory were used [9]. For angle dependence of response, irradiations were perfomed at ±30°, ±45°, ±60°, ±90°, ±120°, on horizontal axis from left to right of subject through geometric centre at the folowing photon radiation qualities: N-10, N-15, N-20, N-30, N-40, N-60, N-80, N-100, N-150, N-200, N-250 and N-300. No tests were done for beta exposure, this will be subject of a future work.

The authors also participated in the intercomparison organized by EURADOS WG12 in 2014 for eye lens dosimetry and are looking forward to the results.

The measurements currently performed for eye lens monitoring are limited to twenty monthly determinations of Hp(3) requested by several medical institutions and a number of experimental studies. In performing the studies, head band dosemeters with three detectors each were used (left eye, right eye, central) for the identification of the area registering the maximum dose. At higher doses, the studies highlighted significant differences between the three detectors, as seen in Table 2. The results are shown as percentages from the total number of doses evaluated for each detector position.

TABLE 2. RESULTS OF THE EYE LENS MONITORING AND PRO-BONO EXPERIMENTAL STUDIES

[* MDL = Minimum Detection Limit]

Dose interval [mSv]	Left Eye (%)	Center (%)	Right Eye (%)
< MDL	52.5	55.9	59.3
[MDL - 1]	22.0	16.9	16.9
(1 - 1,7]	1.7	3.4	3.4
(1,7 - 20]	1.7	3.4	0.0
> 20	0.0	0.0	0.0

3. CONCLUSION

Studies dealing with eye lens doses were first available in Romania starting with 2013. The reduction of eye lens dose limit will be introduced in Romanian legislation in the next four years, as stated by the 2013/59/EURATOM directive [10]. Until then, we intend to promote the necessity of eye lens and extremities monitoring in radiology/interventional cardiology.

This aspect shows the need for thorough training of the medical staff with regard to work procedures or revision of the work procedures which can only be achieved with the full implication of all the stakeholders in setting the radiation protection programme.

In the future, the experimental studies will continue in the scope of acknowledging and understanding the doses received by the exposed workers in nuclear medicine and IC/IR in order to optimize radiation protection.

Although, extremities and eye lens monitoring contributes significantly to the improvement of the radiation protection programme, the development of new services at reasonable costs poses a serious challenge since the Romanian medical institutions have gone through tough economic situations.

REFERENCES

[1] EUROPEAN COMMSSION, Radiation Protection No. 160 - Technical recommendations for monitoring individuals occupationally exposed to external radiation, RP 160 (2009).

[2] CARNICER A., BAECHLER, S., BARTH, I., DONADILLE, L., FERRARI, P., FULOP, M., GINJAUME, M., GUALDRINI, G., KRIM, S., ORTEGA, X., RIMPLER, A., RUIZ-LOPEZ, N., SANS MERCE, M., VANHAVERE, F., Extremity exposure in diagnostic NM with ^{18}F and ^{99m}Tc labelled radiopharmaceuticals, ORAMED Presentations, http://www.oramed-fp7.eu/en/Presentations (last accessed on June, 2014).

[3] VANHAVERE, F., CARINOU E., DOMIENIK, J., DONADILLE, L., GINJAUME, M., GUALDRINI, G., KOUKORAVA, C., KRIM, S., NIKODEMOVA, D., RUIZ LOPEZ, N., SANS-MERCÉ, M., STRUELENS, L., Measurements of eye lens doses in interventional radiology and cardiology: Final results of the ORAMED Project, ORAMED Presentations, http://www.oramed-fp7.eu/en/Presentations (last accessed on June, 2014).

[4] Thermoscientific.com website – see http://www.thermoscientific.com/en/product/single-double-chipstrate-ext-rad-extremity-dosimeters-elements.html (last accessed on June, 2014).

[5] Thermoscientific.com website – see http://www.thermoscientific.com/en/product/dxt-rad-extremity-dosimeters-accessories.html (last accessed on June, 2014).

[6] GILVIN, P. J., BAKER, S. T., GIBBENS, N. J., ROBERTS, G. H., TANNER, R. J., EAKINS, J. S., HAGER, L. G., DANIELS, T. J., Type testing of a head band dosemeter for measuring eye-lens dose in terms of $H_p(3)$, Radiation Protection Dosimetry, 157 (3) (2013) 430 – 436.

[7] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, ISO 4073-3:1999 X and gamma reference radiation for calibrating dosemeters and dose rate meters and determining their response as a function of photon energy, part 3, ISO (1999).

[8] BORDY, J., M., GUALDRINI, G., DAURES, J., MARIOTTI, F., Principles for the design and calibration of radiation protection dosemeters for operational and protection quantities for eye lens dosimetry, Radiation Protection Dosimetry 144 (2011) 257 – 261.

[9] DAURES, J., GOURIOU, J., BORDY, J.M., Conversion coefficients from air kerma to personnal dose equivalent Hp (3) for eye-lens dosimetry, Report CEA – R – 6235, (2009).

[10] EUROPEAN COMMISSION, Council Directive 2013/59/Euratom of 5 December 2013", L13 OJEU 57 (2014).

SKIN EXPOSURE: A SPECIFIC PROBLEM IN OCCUPATIONAL MONITORING

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Abstract

The paper deals with some problems related to the quantification of the skin exposure including some aspects associated with personal monitoring of extremities with special emphasis on the skin of hands and compliance with the relevant regulatory requirements and international standards. This is especially pertinent to the assessment of the equivalent dose of the skin of workers in nuclear medicine where handling of radiopharmaceutical during their preparation and administration may result in relatively high local doses characterized by considerable inhomogeneous distribution over the surface of the exposed skin. The use of current dose limits is scrutinized and a new approach aimed at the modification of the current limitation approach in order to ensure realistic protection of workers against both stochastic and deterministic effects is suggested.

1. INTRODUCTION

It is well known that the main purpose of radiation protection is to guarantee the adequate protection of persons from harmful effects of ionizing radiation (further only radiation). A prerequisite for the assessment, limitation and minimisation of these effects is the quantification of exposure to the radiation involved. The ICRU[(1,2)] and the ICRP[(3,4)] have tried very hard over the last decades to develop a comprehensive system of quantities and units for controlling radiation exposure of professionals, patients and members of the public. The system went through several stages where existing quantities were constantly modified and refined, and, at the same time, some new quantities were introduced as well. This resulted in a rather complicated and intricate set of too many quantities, the interpretation and use of which have not always been unambiguously and unequivocally understood by all users.

Many radiation workers, and even some specialists responsible for radiation protection at workplaces, have become confused by the definitions and specific features of so many quantities, where their understanding requires a sound knowledge of details of interaction processes of radiation with matter and a familiarity with basic dosimetry. While quantities characterizing radiation sources, radiation fields, interactions of radiation with matter, and consequently quantities used in dosimetry are defined on a physical basis, the quantities in radiation protection serve as an assessment of biological effects and thus cannot be considered to be physical quantities but rather bio-physical quantities.

In medical applications of radiation and radionuclides, the personnel always receive some exposure which has to be strictly controlled in line with international standards and national regulations. This is especially important in such fields as nuclear medicine, where the distribution of doses throughout the human body is usually inhomogeneous. In handling some high-activity radiopharmaceuticals the assessment of the skin dose may be very important since it can reach and even exceed the annual equivalent dose limit of 500 mSv.

2. METHODS IN QUANTIFYING THE WHOLE BODY AND SKIN EXPOSURE

Two important quantities used for the estimation of stochastic biological effects (at low doses) are based on the dose, taking into account the type of radiation R and the organ or tissue T exposed. These quantities – the equivalent dose and effective dose – cannot serve as a measure of deterministic biological effects.

The definitions of equivalent dose H_T and effective dose E are as follows

$$H_T = \sum_R w_R \, D_{T,R}, \quad E = \sum_T w_T \; H_T = \sum_T w_T \sum_R w_R \; D_{T,R}$$

where w_R is the radiation weighting factor, w_T is the tissue weighting factor and $D_{T,R}$ is the average dose in the organ or tissue T produced by the radiation of type R. The basic unit of quantities is Gy (for the dose) and Sv (for radiation protection quantities).

In order to keep stochastic effects below the level which is considered adequately safe, dose limits have been introduced [4]. Under normal circumstances, they are not supposed to be exceeded. The current limit is related to the equivalent dose to the skin avearaged over 1 cm^2 or 10 cm^2 (as recommended by the ICRP [4] and NCRP [5], respectively).

The present concept of the effective dose includes the skin as one of organs or tissues considered with the tissue weighting factor of 0.01. In this case it is presumed that the equivalent dose to the skin averaged over the entire surface of the body is taken into account. Assuming that the average adult skin area is about 1.8 m^2, a local skin dose of 500 mSv (over a relatively small exposed area – 1 cm^2) may result in an effective dose of only about 0.3 μSv. Even if the surface of the entire body was exposed to an equivalent dose of 500 mSv, this would contribute to the effective dose of only 5 mSv. Such kind of exposure is rather rare and occurs usually when the whole body is exposed to penetrating radiation where, however, other organs will be much more exposed and their contribution to the effective dose will be much more significant and the contribution from the skin can be neglected.

The quantification of stochastic effects (the exposure to lens of the eye is an exception) relies on the use of the effective dose and equivalent dose. There is no unified alternative system of similar quantities for the assessment of deterministic effects, although for organs or tissues sometimes the so-called RBE-weighted dose, which uses the unit Gy-Eq (gray-equivalent), was proposed. While for stochastic effects the radiation weighting factor w_R is used, for deterministic effects, as a weighting factor RBE (relative biological effectiveness) is applied instead.

There are some specific features regarding the assessment of the equivalent dose to the skin which may cause some ambuiguities and confusion since compliance with the relevant limit is related to a reference area of the most exposed part of the skin.

The radiation effects on skin at low exposure include the induction of skin cancer. If a large area is more or less uniformly exposed, the probability of skin cancer is obviously proportional to the product of the area irradiated and the average skin equivalent dose. In the case of a relatively large exposed area (10-100 cm^2), the use of the current tissue weighting factor of 0.01 may result in conservative assessment of stochastic effects. When the skin is exposed non-uniformly or the area is much less than about 100 cm^2, the risk of stochastic effects is significantly reduced. However, for very small areas (at the level of 1 cm^2 or less), stochastic effects are not so important and the dose control should be based on the prevention of deterministic effects.

3. MONITORING OF THE EQUIVALENT DOSE TO THE SKIN

Strictly speaking, the equivalent dose to the skin is not rigorously introduced by the current definition which defines this quantity as a product of the organ (skin) dose and the radiation weighting factor. In this case, however, it is not clear how the skin as an organ is

specified (e.g., in terms of its mass or volume). For monitoring purposes the equivalent dose to the skin is approximated by the operational quantity of the personal dose equivalent at the reference depth of 0.07 mm. This reference depth may not always be representative for the reflection of the effects in the skin since its upper dead layer on some parts of the body may significantly exceed the depth of 0.07 mm.

The ideal dosimeter to assess the personal dose equivalent to skin would be a sensor as thin as possible, sufficiently sensitive to charged particles (primary and secondary) and covered by the tissue equivalent layer of the thickness of 0.07 mm. The appropriately thin thermoluminiscent dosimeters (TLDs) seem to be the best choice for the monitoring of $H_p(0.07)$. For practical reasons, the dosimeter for measurement of the skin dose should show a response which is directly a measure of the energy absorbed in a layer of skin at a depth in the range between 5 and 10 μm. Normal routinely used TLDs (much thicker than 0.07 mm) considerably underestimate the skin dose, especially in the case of mixed gamma-positrons radiation.

Another peculiarity regarding skin monitoring for regulatory purposes consists in satisfying the requirements with respect to the so-called equivalent dose to the skin averaged over a certain highly exposed area. In this case, the personal dose equivalent $H_p(0.07)$ as essentially a point quantity is used instead. One can formally write the following relationships for H_{skin}, $H_p(0.07)$ and its average value related to the surface skin area selected

$$H_{skin} = \sum_R w_R D_{skin,R}(0.07)$$

Where, the skin dose has to be taken as an average dose over a specified mass or volume.

It is obvious that the equivalent dose to the skin is only a fiction which cannot be used even theoretically for its averaging over the reference surface. It has to be simulated by an operational quantity – the personal dose equivalent $H_p(0.07)$ – where, in principle, one may introduce the averaging using the formula

$$\overline{H_p(0.07)} = \frac{1}{S_r}\int_0^{S_r} H_p(0.07,S)dS \cong H_{skin}$$

Where, S_r is the area over which the personal dose equivalent $H_p(0.07)$ is averaged.

A question arises whether the substitution of the recognized dose limit quantity – the equivalent dose to skin (related to the mean skin dose multiplied by the appropriate weighting factor w_R) – can adequately be substituted by the personal dose equivalent, which relies on a dose at a point of interest in the skin at the depth of 0.07 mm. Apparently, the quantity used in the present system of imitating the skin exposure based on the equivalent dose to skin is not fully consistent with the definitions of dose limit quantities, since obviously $\overline{H_p(0.07)}$ is principally not the same as virtual H_{skin}.

In practical applications, however, it may be acceptable to monitor $H_{T,skin}$ by means of standard finger dosimeters or appropriately calibrated sufficiently thin thermoluminiscent dosimeters which are placed in the positions where the highest exposure is expected. For routine dosimetry, finger dosimeters are commonly used although their readings have to be corrected in order to obtain the maximum value of the equivalent dose to skin, which is usually in a place other than the position of the finger dosimeter and thus appropriate conversion factors have to be used.

The results of the recent studies, carried out under the ORAMED project [6], have shown that the exposure of about 20% of workers preparing and administrating radiopharmaceuticals at some nuclear medicine clinics have exceeded the relevant dose limit. Their maximum local skin doses related to the area of 1 cm^2 were higher than the dose limit of 500 mSv per year.

Similar independent monitoring at two nuclear medicine departments in the Czech Republic has resulted in a conclusion consistent with the ORAMED findings [7].

4. AN ALTERNATIVE APPROACH IN SETTING THE SKIN DOSE LIMITS

At present, it is expected that with the increasing number of examinations in nuclear medicine, the equivalent dose of the skin will also go up and the annual dose limit with respect to the skin may be exceeded in more than 20% of workers, which is a rather alarming prognosis. Exceeding the dose to the skin over the limit is nothing by a violation of regulatory requirements and the situation has to be urgently addressed.

From the above discussion, the following conclusions in terms of the skin dose limits can be derived:

a. Adequate protection of the skin against the stochastic effects can be sufficiently ensured by setting the equivalent dose limit to the present level of 500 mSv/y; however, this skin dose will correspond to its mean value averaged over the area of 10 cm^2, and
b. Another dose limit at the level of 1-2 Sv per 1 cm^2 may be introduced in order to minimize the induction of deterministic effects.

The proposed approach will reasonably solve the present problem of exceeding the current ICRP skin dose limit since the equivalent dose is now related to a rather small area (1 cm^2). The adoption of our proposal will not increase by any means the danger due to local skin exposure; it will only contribute to a more realistic assessment of stochastic and deterministic affects without radically changing the existing system of dose limitation and optimization. The only additional requirement will consist in a more accurate mapping of skin doses encountered in some specific operations characterized by potentially higher local skin doses.

5. CONCLUSION

Since obviously the current conception of the equivalent dose to the skin has some shortcomings and ambiguities as far as its interpretation and monitoring is concerned, it would be most appropriate to revisit the issue and to adopt a sufficiently clear definition of this quantity. The main reason behind our proposal is aimed primarily at the unification of definitions of all quantities, which should be rigorous so that they can be assessed by computational or monitoring methods. Apparently, in the case of the exposure of the skin, and presumably also some other organs or tissues where their parameters have not been identified and agreed on, the equivalent dose becomes a quantity where one cannot strictly follow the official definition because its mass (or volume) has not been specified. Here, we have to rely on a point quantity – the dose equivalent at a given depth below the surface of the skin, i.e. personal dose equivalent for weakly penetrating radiation, rather than to refer to the relevant dose limit quantity recommended for this purpose (the equivalent dose). Presumably, it would be a more rigorous approach to express the dose limit for the skin in terms of the personal dose equivalent $H_p(0.05)$ average over a specified area of the skin. It looks like two areas may be proposed: one, equal to 10 cm^2, aimed at minimizing stochastic effects, and another, 1 cm^2, to ensure adequate protection against deterministic effects. While in the first case the dose limit may stay as it is, namely 500 mSv a year, in the second case the limit may be something like 1-2 Sv (or rather 2 Gy-Eq when RBE-weighted dose is used instead of the equivalent dose).

REFERENCES

[1] INTERNATIONAL COMMISSION ON RADIATION UNITS AND MEASUREMENTS, On line: www.icru.org, (accessed 12 April 2014)

[2] INTERNATIONAL COMMISSION ON RADIATION UNITS AND MEASUREMENTS, Fundamental quantities and units for ionizing radiation, ICRP Report 85a-Revised), Bethesda, USA (MD), (2011).

[3] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Online: www.icrp.org/page.asp?id=3, (accessed 12 April 2014).

[4] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, 2007 Recommendations of the International Commission on Radiological Protection. ICRP Publication 103, Annals of the ICRP 1(3) (2009).

[5] US NRC , Occupational dose limits, Reviewed/Updated Thursday, July 25, 2013. On line: http://www.nrc.gov/reading-rm/doc-collections/cfr/part020/part020-1201.html (accessed 12 April 2014).

[6] Extremity Doses of Nuclear Medicine Personnel: A Concern. Eur. J. Nucl. Med. Mol. Imaging, (2012); 39(3): 529–532, Published online Nov 15, 2011. Doi: 10.1007/s00259- 011-1973-z, Online: http://www.hse.gov.uk/ radiation/ionising/doses/ (accessed 12 April 2014).

[7] FULOP, M., HUDZIETZOVÁ, J., SABOL, J. et. al, A Method for the identification of hand exposure by annihilation photons and positrons during the handling of ^{18}F, Collection of Abstracts, Annual Congress of European Assoc. of Nucl. Medicine. Lyon, France, 19-23 Oct. 2013. On line: http://www.hse.gov.uk/ radiation/ionising/doses/(accessed 12 April 2014).

COMPARISON OF THE RESPONSE AND BEHAVIOR OF TL NEUTRON-GAMMA DOSIMETERS USED IN INDIVIDUAL DOSIMETRY SYSTEM FOR ^{241}Am-Be AND ^{252}Cf SOURCES

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Abstract

Neutron Individual dosimetry is challenging subject in dosimetry field. Thermoluminescent dosimeter or TLD is one of the types of personal dosimeters that are extensively used to determine the doses of neutron sources. Typically, this type of dosimeters is used in different centers with a variety of applications after calibration with a standard neutron radiation sources. Usually, materials of TL crystal such as (LiF), used in personal dosimetry purposes, have the effective atomic number of soft tissue. However, the heavy dependence of the neutron spectrum to geometry of the source location makes the estimation of doses particularly the dose equivalent very difficult. This research involves studies on TLD600-700, used in individual neutron - gamma dosimetry services in Iran. These dosimeters are irradiated with standard sources ^{241}Am-Be and ^{252}Cf at low doses to high doses and their behavior has been studied and analyzed. Also, the important information about relative response of these dosimeters, irradiated by ^{241}Am-Be and ^{252}Cf sources is presented.

1. INTRODUCTION

Application of ionizing radiation in medicine and industry is highly developed and has been increasing. So, in order to protect the staff and people, it is important that best methods are used for evaluation of radiation exposure. Thermoluminescent dosimeter or TLD is one of personal dosimetry systems used to measure the dose received from neutron - gamma sources.

Neutron dosimetry is extremely complex due to the strong dependence on the geometry of neutron exposure and heterogeneous weighting factor for various neutron energies (spectral components) [1-3].

Some semiconductor materials or insulation if they are first irradiated and then heated; release the visible light that the phenomenon is called thermoluminescent. So thermoluminescent is visible light emission from the irradiated material (semiconductor or insulator) due to thermal excitation. Materials that have this property are called thermoluminescent materials.

Crystals used in Thermoluminescent phenomena have a regular crystal lattice such as LiF that have been doped by impurities and defects. For personal dosimetry, Lithium (LiF:Mn) based TLDs are used as they are tissue-equivalent [1-4].

In this work, the empirical dependence thermoluminecent neutron-gamma dosimeters response to dose and the neutron energy spectrum is studied. Based on the data obtained an appropriate factor to correct the response of neutron dosimeters
calibrated in different neutron sources will be presented.

2. MATERIALS AND METHODS

The dosimeters used for the measurements were TLD 600-700 cards . These cards have 4 crystals (Fig.1). Two crystals are TLD 600 and two crystals are TLD 700.
TLD 600 is LiF crystal with 95.12% ^{7}Li and 4.38% ^{6} Li. TLD 700 is LiF crystal with 99.47% ^{6}Li and 0.03% ^{7}Li. Li has a large cross section for low energy neutrons via the n capture reaction given in Equ.1 [1, 2, 4].

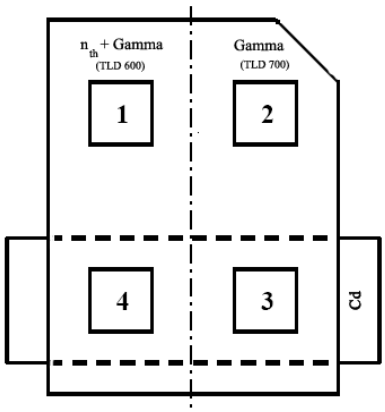
$$^{6}\mathrm{Li} + \mathrm{n} \rightarrow {}^{3}\mathrm{H} + \alpha \qquad (1)$$

Thus, TLD 600 is sensitive to gammas and neutrons while TLD 700 is sensitive to gammas only [4].

The crystals have dimensions of about 1/3 × 1/3 mm and a thickness of 4/0 mm. TLD cards on the field and ^{252}Cf and ^{241}Am-Be neutron sources have been irradiated in air. The process of reading and interpretation of the data was conducted to evaluate the response and also the response of the dosimeter for ^{241}Am-Be and ^{252}Cf sources is provided.

2.1. TLD600/700 response in241 Am-Be and 252 Cf for different doses

Four TLD 600-700 cards (with makeup 6776) have been irradiated in 20 cm from the ^{241}Am-Be and ^{252}Cf sources in air. ^{241}Am-Be source is located in the laboratory calibration of radiation monitoring of activity within 370GBq stainless steel cylinder with a diameter of 30 mm and height 60 mm. Exposure Schematic of TLD crystals used in this study is shown in Fig. 1.



TL_1(TLD 600) : n_{th} + n_{albedo} + gamma

TL_2 (TLD 700): gamma

TL_3(TLD 700) : gamma

TL_4(TLD 600) : n_{th} + gamma

FIG 1. Schematic of radiation TLD600-700 with neutron – gamma source

3. RESULTS

Data obtained from the dosimeter readings of the TLD 600-700 that irradiated at low and high doses with ^{241}Am and ^{252}Cf sources are shown in Figs. 2 and 3. As has been observed in the behavior response from low doses to high doses can meet the linearity [5].

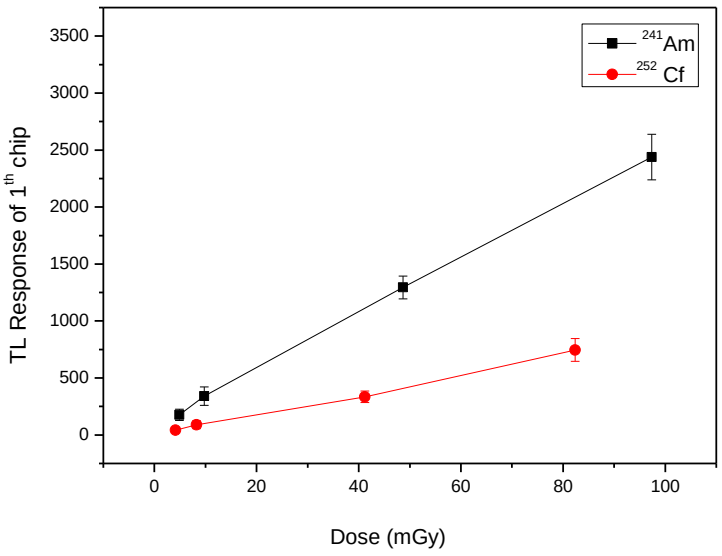


FIG. 2. Response of 1st TL 600 chip (as shown in Fig. 1) for different doses of ^{241}Am- Be and ^{252}Cf sources.

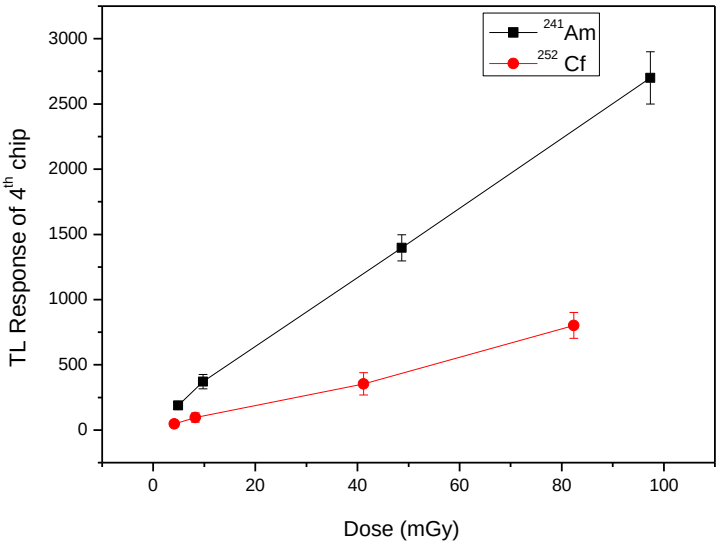


FIG. 3. Response of 4th TL 600 chip (as shown in Fig. 1) for different doses of ^{241}Am- Be and ^{252}Cf sources.

3.1 Calculation of relative response for 241Am-Be source to 252Cf source

TLD 600-700 cards are used to calculate the relative response for ^{241}Am-Be and ^{252}Cf sources. The TLD cards in badges are irradiated on the phantom, at a distance of 50 cm from the ^{241}Am-Be and ^{252}Cf source for 10 mSv.

Relative response for thermal neutrons (TL'_4-TL'_3) on irradiation with ^{252}Cf source to ^{241}Am source is equal to:

$$\frac{67.9\pm1.68}{64.62\pm3.65} = 1.05$$

Relative response for albedo neutrons (TL'_1-TL'_4) on irradiation with ^{252}Cf source to ^{241}Am source is equal to

$$\frac{54.5 \pm 4.91}{41.80 \pm 1.38} = 1.30$$

TL formula corrected by applying RL and ECC and Background factors as follows (Eq. 2):

$$(corrected TL_i = TL_i^{'} = (TLi \times \frac{RL0}{RLi} \times ECC_i - TL_{iBG} \times \frac{RL0}{RL_{iBG}} \times ECC_{iBG}) \qquad (2)$$

The response/dose ratio of each TL crystals which irradiated to ^{252}Cf and ^{241}Am-Be neutron sources are shown in Table 1.

TABLE 1. RELATIVE RESPONSE / DOSE OF TL 600-700 IRRADIATED TO ^{252} CF AND ^{241}Am-Be IN 10 mSv

Response/Dose (nC/mSv)	TL600 0-1	TL700 0-2	TL700 0-3	TL600 0-4
^{241}Am-Be	15.3 ±0.7	0.408 ±0.57	3. 07 ±0.51	9. 84 ±0.28
^{252}Cf	11.63 ±0.67	0.5 ±0.3	0.987 ±0.17	7. 44 ±0.54

4. CONCLUSION

Today, the application of thermoluminecent dosimeter ^{7}LiF and ^{6}LiF in the mixed neutron – gamma fields is widespread. Using the combination of these crystals with cadmium filter, it is possible to evaluate the dose and analyse the neutron spectrum.

Besides these properties, the other important characteristics such as response dependency to dose and energy and also the effects of energy remaining in the crystal after irradiation with neutron high doses have not been fully discussed quantitatively.

For this reason, the study of quantitative assessment of these features is very important. Empirical research indicated that the used TLD in a neutron - gamma dosimetry service have the linear trend response in a wide range of doses. The third important properties of neutron-gamma TLD response dependency to energy were quantitatively studied and shown that the dependency of energy is very strong.

The estimated relative response for 2 widely used sources, ^{241}Am-Be and ^{252}Cf, was studied. The relative correction factor to convert the recorded dose to real dose according to a used source for calibrating the dosimeters was presented.

REFERENCES

[1] MANOUCHEHRI, F., TORKZADEH, F., MIRZAJANI, N., JALILIAN, SH., Thermal Neutron Fluence Measurements Using Thermoluminescence Dosimeter-TLD-600, J. of Nuclear Sci. and Tech. 42 (2008) 1-6.

[2] MENDEZ, R., INIGUEZ, M.P., BARQUERO, R., MANANES, A. et.al., Response

components of LiF: Mg, Ti around a moderated Am-Be neutron source. Radiation Protection Dosimetry, Vol 98 (2) (2002) 173-178; PhD Thesis,Austria; 2002.

[4] TANNER, R.J., THOMAS, D.J., BARTLLET, D.T., HAGER, L.G., HORWOOD, N., TAYLOR, G.C., Effect of the energy dependence of response of neutron personal dosemeters routinely used in the UK on accuracy of dose estimation, NRPB – W25; (2002).

[5] YOO, Y.S., KIM, P.S. AND MOON, P.S., A study on the neutron dosimetry with LiF themoluminescent, Journal of the Korean Nuclear Society, Vol. 7, (3) 1975.

A NEW PERSONAL DOSIMETRY BADGE BASED ON COMBINED LUMINESCENCE TECHNIQUES (TL AND OSL)

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Abstract

TL and OSL dosimeters have been widely employed in the individual monitoring of occupational doses. While personal dosimetry systems based on TL technique make use of a variety of materials and commercial readers available, OSL dosimetry systems explore the optical nature and speed of the readout process, in addition to the possibility of successive readings of the detectors without significant signal loss. In this work, we proposed a new personal dosimetry badge based on different combinations of TL and OSL detectors employing the advantages of both combined techniques for assessment of occupational photon radiation exposures. The badge was made of a material with low density through a 3D printer and it was designed so that one pair of detectors could be read using OSL and TL commercial readers available. Two combinations of TL and OSL detectors ($CaSO_4$/BeO and Al_2O_3/LiF) were used in developing this badge, performing a series of tests in accordance with international standards/criteria. Preliminary results showed that corrections with respect to energy can be carried out through ratio between OSL and TL responses from detectors employed in the badge with reduced angular dependence, without the use of attenuation filters. Moreover, it was possible evaluate simultaneously single and accumulated dose over time and to improve the doses estimation combining the techniques.

1. INTRODUCTION

The assessment of doses in individual and area monitoring is a requirement to radiological protection and it is necessary to demonstrate compliance with dose limits and meet regulatory requirements. There are several dosimetry systems commercially available and a variety of luminescence materials that can be used into a passive dosimeter to record personal exposure to radiation. The systems most employed actually to individual and area monitoring are those based on thermoluminesce (TL) and optically stimulated luminescence (OSL) techniques. Both techniques are analogous, except that the stimulation is carried out optically (light) for OSL, and thermally for TL [1,2].

Works published in the last years highlight the main characteristics and advantages of both techniques[2-4]. For example, by using the OSL technique, materials can be read several times with a negligible degradation of signal and thermal annealing steps are not required; when using the TL technique, it is possible to use TL glow curve as a quality control for

radiation dosimetry as well as minimizing light-induced fading.

In this work we proposed a new personal dosimetry badge based on different combinations of both TL and OSL detectors for assessment of occupational photon radiation exposures. The developed OSL/TL dosimeter consists in a badge made of a low-density material through a 3D printer and a pair of OSL and TL materials. The developed OSL/TL dosimeter allowed to evaluate accumulated and single dose simultaneously combining OSL and TL and to apply dose corrections with respect to energy, without the use of attenuation filters in the badge, exploring the intrinsic characteristics (high luminescence efficiency, effective atomic number) from different detector materials. In addition, the performance of the new personal OSL/TL dosimeter was evaluated according international standards [5, 6].

2. MATERIALS AND METHODS

2.1.OSL and TL detector materials

The four detector materials used were Al_2O_3:C (*Landauer Luxel Tape*) and BeO (*Thermalox 995*) with OSL technique and LiF:Mg,Ti (*Bicron TLD100*) and $CaSO_4$:Dy (Brazilian IPEN/CNEN) with TL technique. All detector materials were submitted to a sensitivity selection process which consisted of separating them into groups with similar luminescent responses when exposed to the same energy beam and radiation dose. After the sensitivity selected process, the OSL and TL detectors suffered a pre-irradiation bleaching/annealing treatment as recommended in the literature [1].

2.2.Badge Project

There were designed two types of badge, one to each combination of pairs of TL/OSL materials (LiF/ Al_2O_3 and $CaSO_4$/BeO), considering OSL and TL available commercial readers. The dosimeter was constructed to be symmetrical with respect to change in the direction according IEC standard [6]. The badge was made of a material with low-density (*Fullcure 870 Veroblack* with density about 1g/cm³ after it was sintered) through a 3D printer (*Stratasys Objet connex 350*). Figure 1 shows the design of the two developed badges.

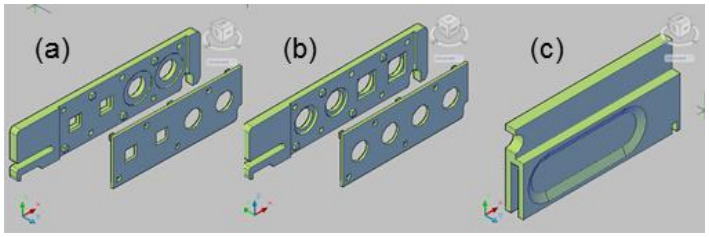

Figure 1. Design of the badge project for the pairs OSL/TL detectors (a) Al2O3/ LiF, (b) CaSO4/BeO and (c) the cover of the badge.

2.3.Preliminary Tests and Algorithm to Evaluate Doses using the Ratio between OSL/TL

Preliminary tests with the four materials were performed and an algorithm to evaluate dose through OSL and TL response from detectors was developed. It was done corrections with respect to energy in a blind test according to the ratio between OSL/TL responses of the detectors.

First the relative energy response to ^{60}Co for the four material detectors employed in the dosimeter was evaluated. The detectors were irradiated with photon beams of radiation qualities implemented at the Medical Physics and Radiation Dosimetry Laboratory of the

Institute of Physics of University of São Paulo (IFUSP), ranging from 20 to 1250 keV (RQR beams [7, 8], ISO beams [9] and gamma rays from ^{137}Cs and ^{60}Co). Through this, it was possible to evaluate the ratio between OSL and TL relative energy response to ^{60}Co. As usually done in dosimetry, it was possible apply a correction factor according the ratio result between OSL/TL response for an OSL/TL pair exposed to unknown dose and radiation beam.

To evaluate the blind dose values from a pair of OSL/TL, an algorithm were composed with the following sequence of steps: (i) detector readouts are converted into dose (M) using a reference calibration curve evaluated from ^{60}Co photon energy; (ii) the ratio R between M_{OSL} and M_{TL} is evaluated; (iii) the R value is compared with the ranges of ratio between OSL and TL detectors normalized energy responses ($R_{OSL/TL}$) previously evaluated for each energy range; (iv) the apparent doses (M_{OSL} and M_{TL}) are corrected using the respective S factor evaluated from the relative sensitivity to ^{60}Co to each OSL and TL detector (S_{OSL} and S_{TL}), corresponding to the energy range found in the previous step; (v) through the product between S and M, we have the corrected dose with respect to energy to OSL and TL detectors, D_{OSL} and D_{TL}, respectively; and (vi) blind dose to pair OSL/TL (D) can be evaluated from a linear combination of D_{OSL} and D_{TL}.

2.4.Performance Tests and Criteria

Each combination of TL/OSL in the new dosimeter was submitted to the performance tests and criteria described in the international standards [5,6]. Those standards present recommendation and tests to dosimetry system that includes relative response due to the non-linearity, relative response due to mean photon energy and angular response [for $H_P(10)$] and environment, mechanical, electromagnetic and physical issues. In this work, we focus just in the requirements, tests and criteria related to dosimeter as presented in Table I.

TABLE I. PERFORMANCE TESTS FOR PERSONAL AND ENVIRONMENTS DOSIMETERS APPLIED IN THE TL/OSL DOSIMETER ACCORDING INTERNATIONAL STANDARDS AND CRITERIA [5,6].

Aspects	**Characteristics under test**	**Main characteristics or influence quantity to be considered**
Physical and detector material proprieties	Coefficient of variation	Statistical fluctuations and variation of the response due to a change of the dose equivalent
	Non-linearity	
	Overload	Response to high doses shall not be less than an upper limit dose.
	After Efects	Effects on subsequent measurements after the dosimeter have been exposed to upper limit dose.
	Reusability	
	Radiation Energy and angle of incidence	Variation of relative response due to changes in energy and angle of incidence.
	Over Response	Over response due to irradiation incidence from the side of dosimeter.
Environment Conditions	Light exposure	Relative response of dosimeter and deviation after light exposure within of range 0 to 1000W/m².
	Dose build-up	
	Fading	Variation of relative response of dosimeter over time.
	Self-irradiation	
	Natural radiation	Relative response due to dosimeter exposure to a natural radiation during a storage time.
	Sealing	Effectiveness of the sealing to prevent the ingress of moisture.
Mechanical	Drop	Resistance and variation in the relative response of dosimeter due to drops from a height of 1.0m.

The variation of the relative response due to change of radiation energy and incidence angle was evaluated according IEC62387-1 using four mandatory radiation qualities (N-30, N-80, N-200 and S-Co) and two radiation incidence angle, 0° and 60°. Those angles are chosen because badge is symmetrical with respect to a change of direction. The test was performed with the dosimeters placed on an appropriated phantom (ISO phantom).

3. RESULTS AND DISCUSSIONS

3.1. **Blind Test with the new OSL/TL dosimeters**

The ratios of energy responses for the two pairs of detectors present a flat behavior for higher energies, with a constant value near to 1, to energies between 50-300 keV the ratios for both pairs significantly decrease with the increase of the energy (Figure 2).

The ratios of relative energy responses calculated to both detector combinations agree with the actual values of 29 and 90 keV, respectively (Table II). Although the coefficients of variation (CV) of the estimated blind doses range up to 21%, these values are compatible to the reference values (actual values evaluated with ionization chamber), with differences smaller than 6%, demonstrating that this algorithm can be useful to correct the energy response and dose of a personal dosimeter using a pair OSL/TL.

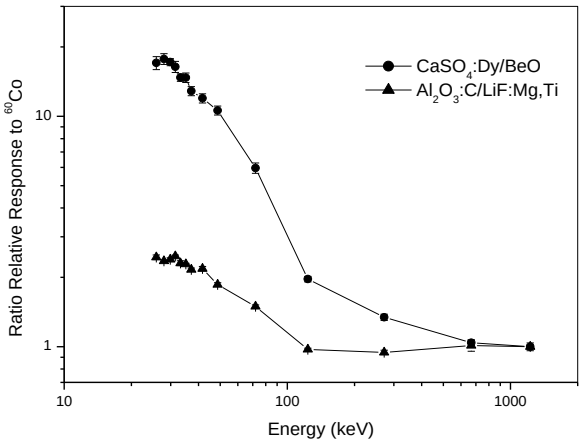

Figure 2. Ratio between TL/OSL or OSL/TL relative energy responses to energy 60Co for pairs of CaSO4:Dy TL/ BeO OSL and Al2O3:C OSL/ LiF:Mg,Ti TL detectors respectively.

Table II. Blind test results: energy range and dose estimated from OSLD/TLD detector combination, expected reference values and relative difference between actual and estimated dose values.

	Blind Test			**Actual Values**		
Detector Combination	**Ratio of Relative Response**	**Energy Range (keV)**	**Dose (mGy)**	**Energy (keV)**	**Dose (mGy)**	**Difference (%)**
Al_2O_3/LiF	2.4 ± 0.1	≤50	28 ± 6	29	27	3.7
	1.35 ± 0.02	50 - 100	84 ± 3	90	85	1.2
$CaSO_4$/BeO	14 ± 1	≤50	28 ± 3	29	27	3.6
	3.8 ± 0.2	50 - 100	90 ± 8	90	85	5.9

3.2. **Results of Performance Tests**

3.3. **Relative response due to radiation quality and angle of incidence**

The relative response to ^{60}Co for both combinations used in TL/ OSL dosimeters presents a small

variation with respect to incidence angle (<0.2%) for all energies used in this work (Figure 3). Those results indicate that angular dependence can be reduced using the developed TL/ OSL dosimeter that explore the intrinsic proprieties of TL and OSL materials as different energy responses to the same energy range.

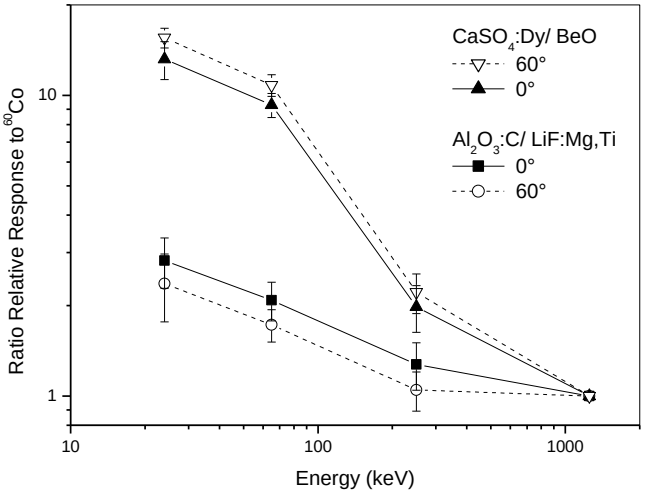

Figure 3. Ratio relative energy responses to ^{60}Co for $CaSO_4$:Dy TL/ BeO OSL and Al_2O_3:C OSL/ LiF:Mg,Ti TL dosimeters for two angle of incidence (0° and 60°).

3.4. **Others physical and detector material proprieties**

The response of the new dosimeter did not show differences with respect to irradiation posterior-anterior because the badge was project to be symmetrical. The response of dosimeters was not affected to changes in the dose equivalent and present a variation coefficient <5%. Moreover the dosimeter present a dose response linear in the mandatory range of energy and equivalent dose for $H_p(10)$ dosimeters according international standards.

3.5. **Environment and mechanical conditions**

The response of dosimeter to both combinations was unaffected due to changes in environment conditions like light exposure and natural radiation. The badge was project to avoid contaminations and contact of the detectors with external environment. The sealing of dosimeter was considered efficient and the new badge presented mechanical resistance to drops from 1m.

4. CONCLUSIONS

We evaluated doses in a blind test using two combinations of TL/OSL detectors (Al_2O_3:C/ LiF:Mg, Ti and $CaSO_4$:Dy/ BeO) exposure to two unknown radiation beams, in order to exemplify the applicability of ratio of photon energy response between TL/OSL dosimeters. After this, we developed a new TL/OSL dosimeter based on different dosimetric materials and stimulation techniques for readouts, without using attenuation filters. The developed TL/OSL dosimeters were submitted to several performance test and criteria based on international standards to personal monitoring. In all test performed with the both combinations, to both combinations TL/OSL used in this work, the new dosimeter fulfill the requirements of standard, demonstrating its applicability for assessment of occupational photon radiation exposures.

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REFERENCES

[1] Yukihara, E., McKeever, S.W.S., 2011. Optically Stimulated Luminescence: Fundamentals and Applications. John Wiley and Sons Ltda.

[2] McKeever SWS, Moscovitch M. On the advantages and disadvantages of optically stimulated luminescence dosimetry and thermoluminescent dosimetry. Radiation Protection Dosimetry 2003; 104:263-270.

[3] Olko, P., 2010. Advantages and disadvantages of luminescence dosimetry. Radiat. Meas. 45, 506–511.

[4] Malthez, A.L.M.C., Freitas, M.B., Yoshimura, E.M., Button, V.L.S.N., 2013. Application of optically stimulated luminescence technique to evaluate simultaneously accumulated and single doses with the same dosimeter. Radiat. Phys. Chem. DOI: 10.1016/j.radphyschem.2013.02.026.

[5] ANSI/HPS N13.11-2001. Personnel Dosimetry Performance – Criteria for Testing. July 2001. Health Physics Society, McLean, VA.

[6] BS EN 62387-1:2012. Radiation protection instrumentation – Passive integrating dosimetry system for environmental and personal monitoring. Part 1: General characteristics and performance requirements (IEC 62387-1:2007, modified). February 2012. CENELEC.

[7] Reina, T.R., Vieira, D.V., Costa, P.R., 2013. Determinação e Validação das Qualidades de Radiação RQR e RQA da IAEA. In: Congresso Brasileiro de Física Médica, 2013, São Pedro.

[8] TRS 457 (Technical Reports Series No. 475), 2007. Dosimetry in diagnostic radiology: an international code of practice. International Atomic Energy Agency.

[9] Guimarães, C.C., 2005. Monitoração Individual Externa: Experimentos e Simulações. Ph.D Thesis. University of São Paulo, São Paulo, Brazil.

TOTAL REFERENCE AIR KERMA AS A DOSIMETRIC PARAMETER FOR ASSESSING OCCUPATIONAL RADIATION PROTECTION IN BRACHYTHERAPY

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Abstract

High dose rate (HDR) brachytherapy (BT) has some benefits over the low dose rate technique. It becomes more significant in countries such as Nigeria where the incidence of cervix cancer is considerably high. Most of the afterloading HDR units offer dose rates above 300 Gy/h, making the need for radiation protection of personnel and patients necessary. This study therefore assesses occupational radiation protection in brachytherapy at the pioneer HDR center in Nigeria by evaluating total reference air kerma (TRAK). Two groups of 30 patients, each given point A fraction dose of 5 Gy from July 2008 to March 2009 and November 2010 to December 2011 within the first half-life of Bebig's ^{60}Co radionuclide at the Department of Radiotherapy, University College Hospital (UCH), Ibadan, Nigeria were entered into this study. Analysis and comparisons of the source air kerma strength (AKS) and treatment TRAK at both periods were carried out using the SPSS software. Unlike source strength (p=0.000), the observed difference in TRAK is not statistically significant (p = 0.409), indicating comparable values in both groups. For quality assurance (QA) among HDR users utilizing ^{192}Ir or ^{60}Co radionuclides, assessment of TRAK should be given consideration, in addition to routine radiation survey.

1. INTRODUCTION

The American Brachytherapy Society (ABS) strongly recommends that radiation treatment for carcinoma of cervix should include brachytherapy. Intracavitary brachytherapy (ICBT) is therefore essential in Nigeria given the high incidence of cervix cancer in the teeming population. The most common nuclide used in modern HDR afterloading machines is ^{192}Ir, however, the use of ^{60}Co is increasing. Dose distributions for both radioisotopes are nearly identical [1] and this makes the latter an attractive alternative being economical (longer half-life) for developing countries. ^{60}Co is therefore the choice of radionuclide currently used at UCH, Ibadan which is the first HDR centre in Nigeria. However, it requires increased radiation shielding due to the higher air kerma rate constant, Γ_{δ} (around 0.306 μGyh^{-1} m^2/MBq for ^{60}Co sources in comparison to 0.099-0.11 μGyh^{-1} m^2/MBq for the commercially available ^{192}Ir sources. The need arises therefore for QA assessment in respect of radiation protection of personnel involved in HDR BT at the centre above. An important way of performing such evaluation is the use of TRAK.

The International Commission on Radiological Units and measurements (ICRU) in its reports 38 and 58 gave some recommendations regarding dose specification in gynaecological brachytherapy [2, 3]. One of its guidelines is the need to report TRAK at the basic level. It is the successor to mgRaeq-h, commonest way of dose specification for a long time, proportional to integral dose to the patient and can serve as a useful index for radiation protection of personnel. This study is therefore aimed at assessing occupational radiation protection by evaluating TRAK at two different time periods in the use of the ^{60}Co radionuclide at the pioneer HDR centre in Nigeria.

2. METHODS

A total of sixty patients who received HDR BT for cervix cancer with Bebig's Co-60 Gynesource Afterloader at the Department of Radiotherapy, UCH, Ibadan, Nigeria were entered into this study. The first group of 30 patients were treated from July 2008 to March 2009 with a dose prescription of 15 Gy in 3 fractions at point A using ring applicator. The 2nd group has same number of patients, treatment dose and applicator type as the initial, but treated from November 2010 to December 2011. However, across the groups, the distribution in applicators' (particularly the tandem) dimensions are different (Fig. 3). Standard planning approach was adopted for all patients on HDR basic treatment planning system (TPS) using tandem-ring dwell time ratio of 1:1 (Fig. 2). The Activity of the ^{60}Co radionuclide ranges from 56.27 to 60.96 GBq and from 38.66 to 45.15 GBq for the earlier and latter groups respectively. Source strengths at these periods were re-expressed as air kerma strength, AKS using equation (2) while TRAK was calculated for individual standard plan on the TPS using equation (3).

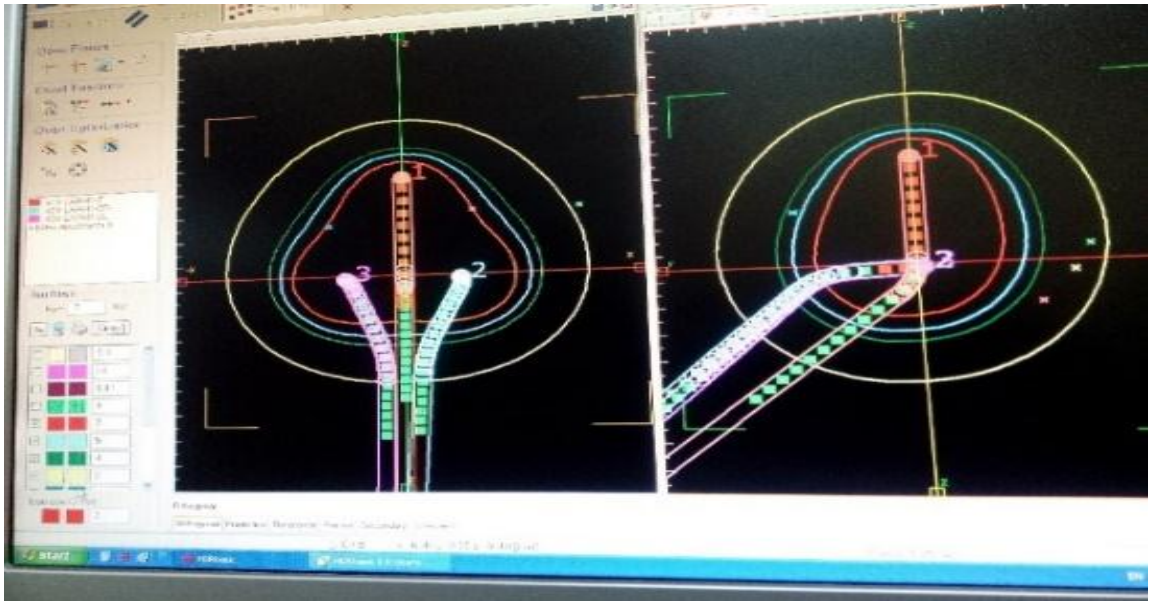
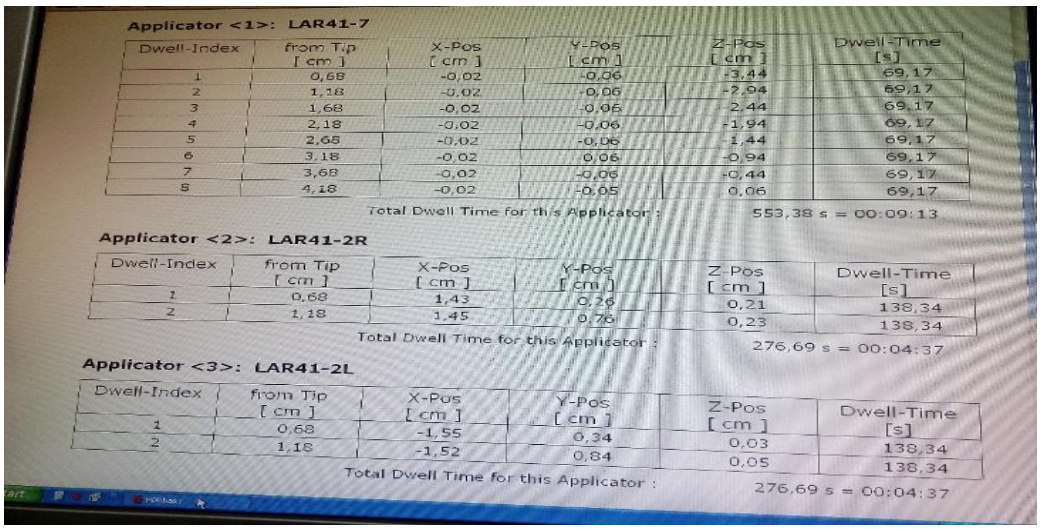


FIG. 1. Standard treatment plan with use of intra-uterine 4 cm tandem and 30 mm diameter ring



Applicator <1>: LAR41-7

Dwell-Index	from Tip [cm]	X-Pos [cm]	Y-Pos [cm]	Z-Pos [cm]	Dwell-Time [s]
1	0,68	-0,02	-0,06	-3,44	69,17
2	1,18	-0,02	-0,06	-2,94	69,17
3	1,68	-0,02	-0,06	-2,44	69,17
4	2,18	-0,02	-0,06	-1,94	69,17
5	2,68	-0,02	-0,06	-1,44	69,17
6	3,18	-0,02	0,06	-0,94	69,17
7	3,68	-0,02	-0,06	-0,44	69,17
8	4,18	-0,02	-0,05	0,06	69,17

Total Dwell Time for this Applicator : 553,38 s = 00:09:13

Applicator <2>: LAR41-2R

Dwell-Index	from Tip [cm]	X-Pos [cm]	Y-Pos [cm]	Z-Pos [cm]	Dwell-Time [s]
1	0,68	1,43	0,26	0,21	138,34
2	1,18	1,45	0,76	0,23	138,34

Total Dwell Time for this Applicator : 276,69 s = 00:04:37

Applicator <3>: LAR41-2L

Dwell-Index	from Tip [cm]	X-Pos [cm]	Y-Pos [cm]	Z-Pos [cm]	Dwell-Time [s]
1	0,68	-1,55	0,34	0,03	138,34
2	1,18	-1,52	0,84	0,05	138,34

Total Dwell Time for this Applicator : 276,69 s = 00:04:37

FIG. 2: Dwell time pattern: Dwell times are equal within each applicator component but differ across the ring and tandem (Tandem time, 9.22 min and Ring time, 9.22 min are in ratio 1:1)

$$1\ \text{MBq} = 0.306\ \mu\text{Gyh}^{-1}\ \text{m}^2\ (\Gamma_\partial \text{ for Co-60 radionuclide}) \qquad (1)$$

$$1\ \text{GBq} = 306\ \text{cGyh}^{-1}\ \text{cm}^2 \qquad (2)$$

$$\text{TRAK} = \text{AKS}_{\text{plan}}\ (\text{cGyh}^{-1}\ \text{cm}^2) \times \text{time}_{\text{plan}}\ (\text{hr}) \qquad (3)$$

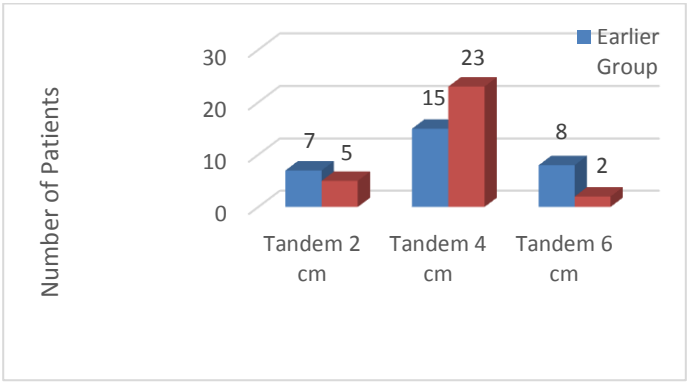


FIG. 3. Distribution of tandem lengths across patients in both groups

Using SPSS, tests of significance was carried out for both dosimetric paramters, AKS and TRAK to make comparisons between both time periods.

3. RESULTS

The results obtained are presented in Tables 1 and Table 2.

TABLE 1: COMPARISON OF AIR KERMA STRENGTH (MGY/HR.) BETWEEN BOTH GROUPS TREATED AT DIFFERENT PERIODS

AKS	Earlier Group	Later Group
Min.	17218.59	11829.94
Max.	18653.73	13815.88
Median	17980.53	13442.56
Mean	17857.50	12801.96
p-value	0.000	

TABLE 2. COMPARISON OF TOTAL REFERENCE AIR KERMA (CGYH-1 CM2) BETWEEN BOTH GROUP

TRAK	Earlier Group	Later Group
Min.	0.218	0.240
Max.	0.456	0.492
Median	0.321	0.321
Mean	0.319 ± 0.044	0.322 ± 0.053
p-value	0.409	

4. DISCUSSION

Results (Table 2) show that observed difference in mean TRAK is not statistically significant ($p = 0.409$), indicating comparable values in both groups. The different trend (Table 1) in the case of source AKS ($p = 0.000$) which implies there is significant difference in source strength is justified by the considerable radioactive decay of the ^{60}Co radionuclide over the time intervals. The survey of Pötter et al. 2001, [4] indicates that reporting TRAK is not widespread, especially among high dose rate (HDR) users. The outcomes of TRAK evaluation in this study show there is variation in its values across patients treated with same fractional dose (5 Gy) to the reference point A. This is warranted by the differences in dose distributions due to different dimensions of applicator components, particularly the tandem (Fig. 3), which influences the number of dwell positions involved in treatment planning. In view of the comparable mean TRAK between the two periods under study, its relevance as a radioprotective index for personnel is therefore established. Personnel radiation exposure records for staff involved in both external beam radiotherapy and HDR BT at our centre ranged from 1.54 – 2.72 mSv (mean 2.11 ± 0.01 mSv) and 1.98 – 3.25 mSv (mean 2.64 ± 0.02 mSv) for the earlier and subsequent periods respectively. The increase observed can be

attributed to longer duration in the 2nd monitoring period. Therefore, higher AKS of the ^{60}Co during the initial study period did not have serious impact on dose to staff in brachytherapy and the connection to occupational expsoures of medical staff is negligible. Periodic assessment of total reference air kerma as a vital dosimetric parameter will help in keeping personnel radiation exposure during brachytherapy in check in addition to controlling dose delivery to patients.

5. CONCLUSION

The relevance of TRAK to occupational radiation protection in HDR brachytherapy has been evaluated at the University College Hospital, Ibadan, Nigeria. Between the two time periods under the study, the observed difference in mean TRAK is not statistically significant ($p = 0.409$), indicating comparable values in both groups. For continuous quality assurance and inter-comparisons among HDR users utilizing ^{192}Ir or ^{60}Co radionuclides, TRAK should therefore be regularly assessed. In addition to routine radiation survey and other daily QA checks, giving consideration to this dosimetric parameter will engender uniformity in dose delivery and enhance safety of personnel in the practice of the HDR technique.

REFERENCES

[1] GRUNDEI B, ANDRASSY M., Eingangsdosimetrie an einer Co60-Afterloading quelle. Poster, DGMP-Tagung Regensburg, (2006).

[2] ORTON CG, SEYEDSADR M, SOMNAY A., Comparison of high and low dose rate remote afterloading for cervix cancer and the importance of fractionation. Int. J. Radiat. Oncol.Biol.Phys. 21 (1991) 1425–34.

[3] PETEREIT DG, PEARCEY R., Literature analysis of high dose rate brachytherapy fractionation schedules in the treatment of cervical cancer (Is there an optimal fractionation schedule?). Int. J. Radiat. Oncol. Biol. Phys. 43 (1999) 359–366.

[4] PO¨TTER R, LIMBERGEN EV, GERSTNER N, WAMBERSIE A., Survey of the use of the ICRU 38, In recording and reporting cervical cancer brachytherapy, RadiotherOncol. 58 (2001) 11–18.

ASSESSMENT OF THE CONFORMITY OF DOSIMETERS USED TO MEASURE DOSE TO THE LENS OF THE EYE *A REGULATORY APPROACH*

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Abstract

The International Atomic Energy Agency adopted the reduced dose limit for the eye lens in Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, General Safety Requirements Part 3 No. GSR Part 3 (Interim) in September 2011. The reduction of the dose limit for the eye lens followed the recommendation of the International Commission on Radiological Protection in its Statement on Tissue Reactions in April 2011. The Council Directive 2013/59/EURATOM of 5 December 2013 laying down basic safety standards for the protection of health of workers and the general public against the dangers arising from ionizing radiation reducing the dose limit is to be complied with by all Member States by 6 February 2018. In the process of preparing Implications for Occupational Radiation Protection of the New Dose Limit for Lens of the Eye it became obvious that there was a lack of available information concerning eye lens doses at nuclear facilities. The Swedish Radiation Safety Authority initiated the assessment of the conformity of dosimeters used to measure dose to the eye lens at the Swedish nuclear facilities in April 2013. The assessment methodology was developed in cooperation between the facilities concerned by this assessment.

1. INTRODUCTION

In its Statement 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 in April 2011. 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.

The International Atomic Energy Agency (IAEA) adopted the reduced dose limit for the lens of the eye in Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, General Safety Requirements Part 3 No. GSR Part 3 (Interim) [2] in September 2011. The reduction of the dose limit for the eye lens followed the recommendation of the ICRP. The Council Directive 2013/59/EURATOM of 5 December 2013 laying down basic safety standards for the protection of health of workers and the general public against the dangers arising from ionizing radiation [3], including the reduced dose limit for the eye lens, was adopted in December 2013. The new directive is to be complied with by all Member States by 6 February 2018.

In the process of preparing IAEA´s TECDOC Implications for Occupational Radiation Protection of the New Dose Limit for Lens of the Eye [4] it became obvious that there was a lack of available information concerning the existing eye lens doses at nuclear facilities as well as the risk for such.

The Swedish Radiation Safety Authority (SSM) is currently examinating the assessment reports delivered by Vattenfall Research and Development (VRD) and the Swedish nuclear facilities. The aim of the examination is to ensure the correctness of the results when it comes to dosimetric methodology, measurements and conclusions drawn. The aim is also to ensure compliance with the reduced dose limit for the lens of the eye at the Swedish nuclear facilities on a national level. A system and a programme for monitoring the eye lens dose is to be in place, if and/or when needed. 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 will change, needs to exist as well.

2. DESCRIPTION OF THE ASSESSMENT

SSM is currently revising the national regulations concerning radiation safety in Sweden. The reduced dose limit for the lens of the eye, so as the whole directive [3], is to be complied with by all Member States by 6 February 2018.

According to Radiological Protection – Procedures for Monitoring the Dose to the Lens of the Eye, the Skin and the Extremities (DRAFT) [1], monitoring of the eye lens dose shall be undertaken if a dose in a single year above 15 mSv is liable to be received, or if in consecutive years more than 6 mSv per year is liable to be received. For dose levels expected to be lower than these values, a survey shall be sufficient to demonstrate that the above levels are not exceeded.

In the process of preparing IAEA´s TECDOC Implications for Occupational Radiation Protection of the New Dose Limit for Lens of the Eye [4] it became obvious that there was a lack of available information concerning the risk for as well as the existing eye lens doses at nuclear facilities.

There are eight nuclear facilities in Sweden, concerned by this assessment:

(a) Four nuclear power stations, three of which in operation, one to be decommissioned.
(b) A facility dealing decommissioning services and waste management.
(c) A facility managing and disposing of all radioactive waste from Swedish nuclear power plants.
(d) A facility dealing with advanced technical services to the international nuclear power industry in such areas as waste treatment, consultancy services as well as nuclear fuel and materials technology.
(e) A facility manufacturing nuclear fuel and dealing with nuclear services and automation.

The area and/or extent of operations varies greatly between the facilities.

SSM initiated an assessment of the conformity of dosimeters used to measure dose to the lens of the eye at the Swedish nuclear facilities in April 2013.

Each facility was to assess:

i. radiation fields present at the facility,
ii. critical work tasks where considerable dose to the eye lens can be obtained,
iii. critical work groups which can obtain considerable dose to the eye lens when executing their work tasks and
iv. the effectiveness of personal protective equipment (PPE) for the lens of the eye in the radiation fields present at the facility.

3. CONDUCTING THE ASSESSMENT

The Swedish nuclear facilities decided to cooperate in developing the methodology for the assessment with VRD acting as the coordinator. The assessment methodology and results have been presented at ISOE European Symposium in Bern in April 2014 by L. Bäckström [6].

The most accurate method for monitoring the equivalent dose to the lens of the eye, according to [4], is to measure the personal dose equivalent at 3 mm depth, $H_p(3)$, with a dosimeter worn as close as possible to the eye and calibrated on a phantom representative of the head. Further, in order to ensure an appropriate individual monitoring, the monitors and/or dosimeters should comply with internationally agreed performance requirements. At present, dosimeters designed for $H_p(3)$ are not yet widely available and monitors and dosimeters for other quantities may be used according to [4]. The use of other quantities, though, might be questionable according to Behrens et al [7].

The dosimeter chosen by the facilities to be used when conducting the assessment was a $H_p(3)$ calibrated thermoluminescent dosimeter from Public Health England (PHE) Personal Dosimetry Service (PDS), EXTRAD^{TM}, together with a polytetrafluoroethylene (PTFE) filter. The dosimeter has been type tested by PHE, as accounted for in Type testing of a head band dosemeter for measuring eye lens dose in terms of $H_p(3)$ [8].

A number of tests were also conducted by the Swedish nuclear facilities to confirm the compliance with Swedish Radiation Safety Authority´s regulations concerning basic provisions for the protection of workers and the general public in practices involving ionising radiation [2]. As in many countries, though, $H_p(3)$ has n**ot yet** been implemented in legal metrology in Sweden, and only $H_p(0.07)$ and $H_p(10)$ are currently used for monitoring the radiation exposure of workers in accordance to Radiation Protection 73, Technical recommendations for monitoring individuals occupationally exposed to external radiation [3]. These recommendations have since been replaced by revised Radiation Protection No 160, Technical Recommendations for Monitoring Individuals Occupationally Exposed to External Radiation [4]1, which is to be considered in the on-going revision of Swedish regulations for the protection of workers and the general public in practices involving ionising radiation.

Irradiations for the tests conducted by the facilities in question were carried out at the Physikalisch-Technische Bundesanstalt (PTB) in Germany. Both the photon and the beta radiation qualities used were chosen to be those representative for the energies most common at the facilities.

4. RESULTS

4.1. Calculations

As a part of the assessment a series of calculations based on the facilities´ activity inventory was executed for estimation of dose to the eye lens from both photon and beta radiation fields.

The estimates of the dose to the eye lens for photon radiation were calculated with the help of MicroShield software. For the calculation of the estimates of the dose to the eye lens for beta radiation the MCNP5 software was used. A number of cases involving ion exchange resins and surface contamination (CRUD[2]) were calculated. The calculations resulted in dose estimates and a number of recommendations for in situ measurements [6].

For personnel whose trunk may be shielded, but not the head, the eye lens dose might not be correctly assessed by a trunk dosimeter. When executing tasks on open systems,

[1] Available at: http://ec.europa.eu/energy/nuclear/radiation_protection/doc/publication/160.pdf
[2] Chalk River Unidentified Deposits

dealing with heavily surface contaminated objects and/or ion exchange resins the personnel is most often being exposed to mixed radiation fields.

4.2. In-situ measurements

The test groups at the facilities consisted of radiation protection technicians, mechanics, welders, control rod drive mechanisms workers, cleaners/decontamination workers, waste management workers, insulation workers, contaminated metal melting facility workers, laboratory engineers, cleaners and operators preparing uranium powder. Some of these groups of personnel had been identified as most likely to receive a considerable dose to the eye lens.

A reference group was also chosen by the facilities.

The highest equivalent dose to the eye lens during this 2-month period was 2.9 mSv. The dose from the trunk dosimeter for the same period was 2.3 mSv. The person in question was executing decontamination work at a nuclear power plant.

In beta radiation fields present at the facilities, the PPE provides good protection. In photon fields the material of PPE can, in same cases, slightly enhance the dose to the eye lens [5].

5. DISCUSSION AND CONCLUSIONS

According to Radiological Protection – Procedures for Monitoring the Dose to the Lens of the Eye, the Skin and the Extremities (DRAFT) [1]**,** monitoring of the eye lens dose shall be undertaken if a dose in a single year above 15 mSv is liable to be received, or if in consecutive years more than 6 mSv per year is liable to be received. For dose levels expected to be lower than these values, a survey shall be sufficient to demonstrate that the above levels are not exceeded.

The results from the assessment show that at the nuclear power plants in operation, at the waste treatment facility and at the nuclear fuel facility there is a risk that the eye lens dose exceeds 6 mSv per year for certain work groups. A test period of eye lens monitoring will be implemented at these facilities in order to assess the eye lens dose to the possible risk groups more closely.

The time frame originally defined by SSM was too narrow, the assessment to be completed and reported to the regulator by the 31st of January 2014. At present, as mentioned earlier, dosimeters designed for $H_p(3)$ are not yet widely available and it took some time before the in-situ measurements could be carried out at the facilities.

The in-situ measurements will continue during the outage period 2014 at one of the nuclear power plants in operation in order to achieve a wider picture of the possible risk groups who might need regular monitoring of the dose to the lens of the eye.

In a number of cases the trunk dosimeter does provide a good enough estimate for the eye lens dose at the Swedish nuclear facilities. In some cases a dosimeter calibrated for $H_p(10)$ and even $H_p(0.07)$, complemented with a filter, provide a good enough estimate for the assessment of the eye lens dose.

There shall be guidelines at the facilities concerning the correct use of PPE. For beta radiation fields present at the facilities the PPE provides good protection.

At some of the facilities in question the area and extent of the operations is to widen in the future. Nuclide vectors were developed in order to assess the work tasks and work groups to undergo a test period of eye lens monitoring when introducing new areas of operation.

REFERENCES

[1] *ISO/CD 15382 "Radiological Protection - Procedures for Monitoring the Dose to the Lens of the Eye, the Skin and the Extremities", DRAFT, ISO/TC 85/SC 2.*

[2] *Swedish Radiation Safety Authority, SSMFS 2008:51, Swedish Radiation Safety Authority's regulations concerning basic provisions for the protection of workers and the general public in practices involving ionising radiation, Stockholm (2008).*

[3] *EUROPEAN COMMISSION, Radiation Protection 73, Technical recommendations for monitoring individuals occupationally exposed to external radiation, Report EUR 14852 EN, EC, Luxembourg (1994).*

[4] *EUROPEAN COMMISSION, Radiation Protection 160, Technical recommendations for monitoring individuals occupationally exposed to external radiation, Final Report of Contract TREN/07/NUCL/S07.70121, EC, Luxembourg (2009).*

[5] *Andgren, K., Bäckström, L., Equivalent dose to the lens of the eye at nuclear facilities and shielding factors for protective eye wear – measurements and calculations, ISOE Symposium, Bern (2014).*

[6] *INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Statement on Tissue Reactions, ICRP (2011).*

[7] *INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, General Safety Requirements Part 3 No. GSR Part 3 (Interim), IAEA, Vienna (2911).*

[8] *EUROPEAN COMMISSION, Council Directive 2013/59/EURATOM of 5 December 2013 laying down basic safety standards for the protection of health of workers and the general public against the dangers arising from ionizing radiation, EC, Brussels (2013).*

[9] *INTERNATIONAL ATOMIC ENERGY AGENCY, Implications for Occupational Radiation Protection of the New Dose Limit for Lens of the Eye, IAEA TECDOC 1731, IAEA, Vienna (2013).*

[10] *Gilvin, P. J. et al., Type testing of a head band dosemeter for measuring eye lens dose in terms of Hp(3), Radiation Protection Dosimetry (2013), pp. 1-7.*

[11] *Behrens, R.; Dietze, G. Monitoring the eye lens: which dose quantity is adequate? Physics in Medicine and Biology 55 (2010), pp. 4047-4062.*

COMMERCIAL TL AND OSL AL_2O_3:C DETECTORS FOR USE IN BETA OCCUPATIONAL MONITORING

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Abstract

Thermoluminescence (TL) and optically stimulated luminescence (OSL) have been applied in different areas of science, and these results have been related and discussed in several papers. In this work, the Al_2O_3:C material, studied in relation to its TL signal, and nowadays also applied as an OSL material, was characterized in relation to its properties. Al_2O_3:C commercial samples (TLD-500) were exposed to the beta secondary standard sources ($^{90}Sr+^{90}Y$, ^{85}Kr and ^{147}Pm) to verify their response for use in beta radiation dosimetry. The tests performed demonstrated that the detectors present potential use as TL and OSL beta dosimeters, because they showed good results and high sensitivity to beta radiation.

1. INTRODUCTION

Luminescent techniques as thermoluminescence (TL) e optically stimulated luminescence (OSL) in radiation dosimetry have been discussed by different authors in relation to their applications, including a comparison between their advantages and disadvantages [1-2]. Although the OSL technique is nowadays so utilized as the TL technique, the first one presents some advantages in relation to the TL technique as: it requires no heating of the samples, the detectors may be evaluated several times, and OSL is a relatively cheaper method than TL [3].

The Al_2O_3:C material, initially developed as TL dosimeter, is a detector which has become the main studied OSL material, because it presents good TL response, excellent OSL dosimetric characteristics [4], and high sensitivity [5]. This kind of material has already been studied in several radiation beams, through the TL and OSL phenomena, showing good behaviour [6-7].

Al_2O_3:C dosimeters, commercialized as TL materials [8], were previously studied [9] in relation to their TL characteristics. The objective of this work was to characterize and to evaluate the performance of Al_2O_3:C Rexon commercial detectors, using both TL and OSL techniques, in standard beta radiation beams, to verify their application for beta radiation occupational monitoring.

2. MATERIALS AND METHODS

Carbon doped luminum oxide (Al_2O_3:C) pellets (5.0 mm in diameter and 1.0 mm in thickness) were studied in this work, acquired as commercial dosimeters from Rexon TLD Systems.

At the Calibration Laboratory of IPEN (LCI) there are two beta radiation secondary standard systems: BSS1, Buchler GmbH, & Co., Germany, and BSS2, Isotrak, Germany. In Table I, the main characteristics of these sources are shown. The sources have calibration certificates issued by the primary standard laboratory Physikalisch – Technische Bundesanstalt (PTB), Germany. During the irradiations, the samples were positioned in a polymethylmetacrilate (PMMA) support, kept in dark and covered by a Mylar foil (superficial density of 0.71 mg/cm^2) to avoid the TL and OSL signal fading.

TABLE I. CHARACTERISTICS OF THE BETA STANDARD SOURCES USED FOR THE IRRADIATION OF THE AL_2O_3:C DOSIMETERS

Beta Secondary Standard System	Radiation Source	Field Flattening Filter	Absorbed Dose Rate (μGy/s)	Calibration Distance (cm)	Reference Date
BSS1	^{90}Sr+^{90}Y	No	518.4 ± 5.180	11	04.02.81
BSS2	^{90}Sr+^{90}Y	No	16.46 ± 0.220	30	12.01.05
	^{85}Kr	Yes	39.70 ± 0.500	30	30.11.04
	^{147}Pm	Yes	2.350 ± 0.050	20	19.11.04

The TL and OSL response of the samples were determined using the Risø, model TL/OSL-DA-200 reader system. In the case of TL, a sample heating rate of 10°C/s and a final temperature of 400°C were used. For the OSL measurements, blue LEDs with optical power of 90% were used; in this case, a stimulation time of 50 s was adopted. A filter basket Hoya U-340 was used in front of the photomultiplier. For the measurements with both techniques, a black mask with a central orifice of 5.0 mm in diameter was used between the photomultiplier and the filter. This accessory was important to avoid the saturation of the photomultiplier. After each cycle of irradiation and measurement, the Al_2O_3:C samples were thermally treated at 400°C during 1 h, for posterior reutilization.

3. RESULTS

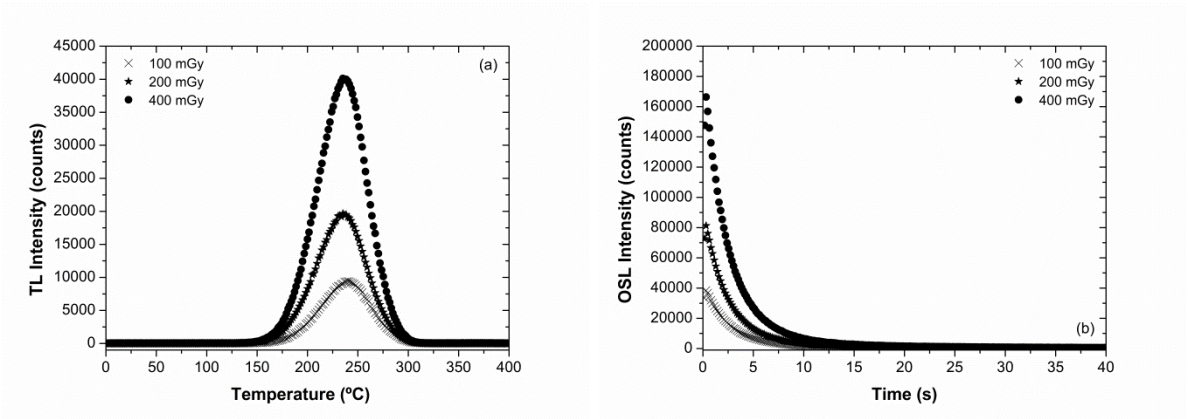
The TL and OSL responses of the Al_2O_3:C detectors were verified by means of their main characteristics.

3.1. Reproducibility of Response

The reproducibility of the TL and OSL response of Al_2O_3:C detectors was obtained using the beta radiation source of ^{90}Sr+^{90}Y from the BSS1 system, at a source-detector distance of 11 cm, and an absorbed dose of 10 mGy (in five cycles of irradiation, measurement and thermal treatment). In the case of TL response, the reproducibility obtained was 3.2% (final standard deviation of the mean value of five cycles using 10 samples), and the maximum standard deviation of all measurements was 3.9% (maximum deviation obtained from all deviations considering each sample and all the five cycles). For the OSL technique, the reproducibility obtained was 4.1%, and the maximum standard deviation was 4.8%.

3.2. TL Emission Curve and OSL Signal Decay

The TL emission curve and OSL signal decay (Fig. 1) were obtained exposing the Al_2O_3:C samples to the ^{90}Sr+^{90}Y radiation source (BSS1 system), at a source-detector distance of 11 cm, and in a dose interval from 100 mGy to 400 mGy.



(a) *(b)*

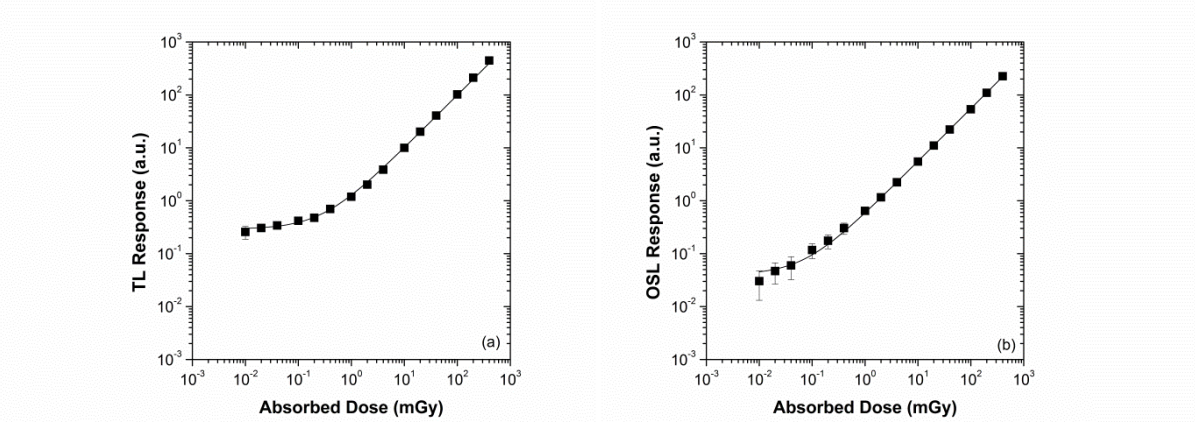
FIG. 1. TL and OSL signal of the Al_2O_3:C detectors: (a) TL emission curve; and (b) OSL signal decay.

Fig. 1(a) shows the TL signal emission. A dosimetric peak can be observed at about 235°C, differently of that informed at the technical specifications of the material [8], which was 185°C, although they used a maximum temperature of 270°C during the measurements, and in this work 400°C was reached. The recommended maximum temperature was tested, but the dosimetric peak was not possible to observe. Therefore, a higher temperature (400ºC) was used. Fig. 1(b) presents the OSL signal decay after the detector irradiation with the same absorbed doses as in the case of the TL signal.

3.3. Dose-Response Curves

In this study, the detectors were irradiated with absorbed doses between 1 mGy and 400 mGy ($^{90}Sr+^{90}Y$, BSS1), and 0.01 mGy and 0.4 mGy ($^{90}Sr+^{90}Y$, BSS2), at the source-detector distance of 30 cm. The maximum standard deviations obtained were 6.1% (10 mGy, TL), and 7.1% (400 mGy, OSL). Fig. 2 shows the TL and OSL dose-response curves obtained in this work. In both cases, linear behaviors can be observed in the interval from 1 mGy to 400 mGy, with correlation coefficients greater than 0.9997.

This study presented results comparable to those of other's works. The linearity was already observed at the TL response of Rexon Al_2O_3:C samples irradiated with beta radiation [9], but for high doses and in different parameters (heating rate of 5°C/s, and maximum temperature of 500°C). Other Al_2O_3:C samples were studied by Akselrod et al. [6]: they observed linearity in the same dose interval as in this work for TL response (beta radiation), from 1 mGy to 400 mGy.



(a) *(b)*

FIG. 2. Dose-response curves: (a) TL and (b) OSL.

3.4. Dependence of the TL and OSL Response with the Radiation Energy

The Al_2O_3:C detectors were exposed to the beta sources of the BSS2 system at the calibration distances of 30 cm ($^{90}Sr+^{90}Y$ and ^{85}Kr sources), and 20 cm (^{147}Pm source). The maximum standard deviation obtained at the TL measurements was 5.4% ($^{90}Sr+^{90}Y$ source), and at the OSL readings it was 6.0% (^{147}Pm). In Table II the results obtained for the energy dependence study can be observed.

These results show the high energy dependence of both TL and OSL responses in beta radiation, which is very similar for both techniques. In the case of the OSL technique, the results presented similar behavior but slightly higher than those of another work [7], where the energy dependence of Landauer Al_2O_3:C detectors was studied using the same sources ($^{90}Sr+^{90}Y$, ^{85}Kr, ^{147}Pm, BSS2).

3.5. Minimum Detectable Dose

The values of the minimum detectable dose (also called lower detection limit) using the TL and OSL techniques were obtained irradiating the Al_2O_3:C detectors with the $^{90}Sr+^{90}Y$ sources of the BSS1 and BSS2 systems, without any field flattening filters. In this characterization test, the detectors were irradiated in the same conditions described in item 3.3. The minimum detectable doses were determined graphically. For both cases (TL and OSL response), the result obtained was 0.04 mGy.

TABLE 2. ENERGY DEPENDENCE STUDY OF THE AL_2O_3:C DETECTORS

Radiation Source	Field Flattening Filter	Absorbed Dose (mGy)	Beta Mean Energy (MeV)	Normalized Response to $^{90}Sr+^{90}Y$	
				TL	OSL
$^{90}Sr+^{90}Y$	No	10	0.80	1.0000 ± 0.0482	1.0000 ± 0.0214
^{85}Kr	Yes	10	0.14	0.1929 ± 0.0092	0.1713 ± 0.0110
^{147}Pm	Yes	615	0.06	0.0120 ± 0.0004	0.0125 ± 0.0001

4. CONCLUSIONS

The characterization tests performed show that the Rexon Al_2O_3:C (TLD-500) samples can be used as TL and OSL detectors. They present good reproducibility of the TL and OSL responses. The dose-response curves show a linear behavior of the TL and OSL responses in the dose interval from 1 mGy to 400 mGy. The minimum detectable doses obtained show the usefulness of this material for low doses. Despite the high dependence of the TL and OSL response with beta radiation energy, the TL and OSL responses presented high sensitivity to beta radiation. Therefore, Al_2O_3:C detectors present potential use for beta radiation dosimetry, and for occupational monitoring, especially when the beta source is known.

ACKNOWLEDGMENTS

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REFERENCES

[1] McKEEVER, S.W.S., MOSCOVITCH, M., On the advantages and disadvantages of optically stimulated luminescence dosimetry and thermoluminescence dosimetry, Radiat. Prot. Dosim. 104 (3) (2003) 263.

[2] OLKO, P., Advantages and disadvantages of luminescence dosimetry, Radiat. Meas. 45 (2010) 506.

[3] BØTTER-JENSEN, L., MCKEEVER, S.W.S., WINTLE, A.G., Optically Stimulated Luminescence Dosimetry, Elsevier Science, Amsterdam (2003).

[4] AKSELROD, M.S., McKEEVER, S.W.S., Radiation dosimetry method using pulsed optically stimulated luminescence, Radiat. Prot. Dosim. 81 (1999) 167.

[5] YUKIHARA, E.G., MCKEEVER, S.W.S., Optically stimulated luminescence (OSL) dosimetry in medicine, Phys. Med. Biol. 53 (2008) 351.

[6] AKSELROD, M.S., MCKEEVER, S.W.S., MOSCOVITCH, M., EMFIETZOGLOU, D., DURHAM, J.R., SOARES, C.G., A thin-layer Al_2O_3 beta TL detector, Radiat. Prot. Dosim. 66 1-4 (1996) 105.

[7] PINTO, T.N.O., CECATTI, S.G.P., GRONCHI, C.C., CALDAS, L.V.E., Application of the OSL technique for beta dosimetry, Radiat. Meas. 43 (2008) 332.

[8] REXON, TLD Systems, Inc. TLD-500 Technical Specifications – Aluminum Oxide (Al_2O_3:C), 2009.

[9] CHITHAMBO, M.L., Concerning secondary thermoluminescence peaks in α-Al_2O_3:C, South Afr. J. Sci. 100 (2004) 524.

CAN A MEDICAL LINAC BE USED FOR TESTING RADIATION PROTECTION DOSEMETERS?

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Abstract

Measurements in the scope of radiation protection in the radiation fields of medical linear accelerators have become more and more important. Since the photon energy and dose rate in the primary beam of these accelerators are above the rated ranges of most radiation protection personal dosemeters, the idea is to use a defined scattered field for testing these devices. Different scattering objects in the primary beam should act as the source of the scattered field. The dose rate in the scattered field could then be tuned by the amount of material in the object, i.e. by its volume. The photon energy will be reduced in the scattered field according to the Compton scattering process. Tests have been made with three different objects in the primary beam and measuring the scattered field using a secondary standard ionization chamber.

1. INTRODUCTION

In radiation therapy for tumours, commercially available medical linear accelerators (linacs) are used. The photon energies used for therapy are in the MeV range and above the rated ranges of most radiation protection dosemeters. Additionally, the radiation is delivered in short pulses with durations in the microsecond range. This results in a high dose rate in the radiation pulse and also in a high (desired) mean dose rate for high pulse repetition frequencies *f*. The dose rate needed for therapy is unfortunately also above the limits of most radiation protection personal dosemeters. Especially electronic dosemeters based on detector pulse counting principles have severe difficulties in such pulsed radiation fields [1, 2, 3]. Therefore, testing such dosemeters in the primary beam is not useful.

However, it is important to test radiation protection dosemeters in such fields. The idea to enable the testing of current dosemeters is to reduce both the dose rate and the energy of the photon radiation through the use of a scattering object and not measuring in the primary beam. In the scattered field, the dose rate should only depend on the volume of the scattering objects, i.e. a large object will yield a higher dose rate in the scattered field than a small one. This is only true if the object is fully irradiated and the attenuation in the object can be neglected. The energy in the scattered field will depend on the primary beam energy and on the scattering angle according to the Compton scattering rule. To determine the conventional quantity value for the radiation protection quantities in the radiation field, measurements have been performed using secondary standard ionization chambers. The focus in this study is on the photon dose only, because the evaluation of the neutron and electron components needs additional careful inspection.

2. DESCRIPTION OF THE SETUP

As a first step, the ambient dose equivalent rate $\dot{H}^*(10)$ at a position well out of the primary beam was determined in the accelerator room of one Elekta Precise clinical linear accelerator of the Physikalisch-Technische Bundesanstalt (PTB). The accelerator room is built similarly to a medical treatment room, but no patient treatments are possible or allowed. The radiation field size is optimized by a multi-leaf collimator and a set of additional jaws for each scattering object.

The ambient dose equivalent rate $\dot{H}^*(10)$ was determined with the HS01 secondary standard ionization chamber provided by Seibersdorf with a 1000 cm³ volume (in the

following: *H**(10) chamber) [4]. The energy response of the chamber was optimized using a Makrolon® shell [5].

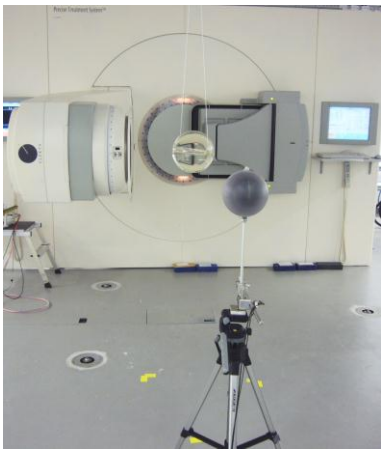

FIG. 1 (left): Setup: H(10) chamber at the Elekta Precise medical linear accelerator at PTB. In the foreground the H*(10) chamber is mounted on a tripod, outside the primary beam. The accelerator head is on the left and the PMMA sphere is hanging in the centre of the primary beam which is towards the right*

FIG. 2 (right): PMMA scattering objects: sphere with a diameter of 300 mm; cylinder with a diameter of 100 mm and a height of 109 mm and a cylinder with a diameter of 16 mm and a height of 16 mm

The chamber has been calibrated traceable to German primary standards according to ISO 4037-1 [6]. The energy dependence of the response of the chamber is, as regards its respective measurement quantity, much less than ±10 % from 100 keV up to several MeV. The expected mean photon energy in the scattered radiation field is much less than in the primary beam (up to 25 MeV) due to the energy loss in the scattering processes. The scattered field from the scattering object (see below) is measured under a scattering angle of 90° with respect to the primary beam axis (see Fig. 1). In this direction photons with energies around 500 keV are expected due to Compton scattering processes in the object.

For scattering the primary beam, three different objects have been used (see Fig. 2):(a) a sphere with a diameter of 300 mm and a volume of about 14,100 cm³, (b) a cylinder with a diameter of 100 mm, a height of 109 mm and a volume of 870 cm³, and (c) a cylinder with a diameter of 16 mm, a height of 16 mm and a volume of 3.1 cm³. All three objects are made of PMMA and the geometrical centre of the object was brought to the isocentre of the accelerator setup. The ionization chamber was aligned with the centre at the same height as the isocentre but at a distance of one metre away from the isocentre perpendicular to the primary beam (Fig. 1). It is expected that the dose rate measured at the position of the ionization chamber depends on the volume of the scattering object, i.e. the large sphere produces more scattered radiation than the small cylinder.

The accelerator produces radiation pulses with a fixed duration of about 3 µs and a fixed dose per pulse of approximately 0.3 mGy absorbed dose to water at a depth of 10 cm depending on the acceleration high voltage. The time averaged mean dose rate can be altered by variation of the pulse repetition frequency *f*.

3. MEASUREMENTS AND THEIR RESULTS

3.1. Dependence on the pulse repetition frequency *f*

The first test of the method is to check whether the ionization current, *I*, in the *H**(10) chamber which is proportional to the dose rate, has a linear dependence on the pulse repetition frequency *f*. This is checked for the 100 mm cylinder in the primary beam and different *f* and high voltage (HV) values. The results are given in Table 1. The measured ionization currents are almost perfectly linear to the pulse repetition frequency. Only at the low repetition frequency of 6.25 Hz, a small deviation from the linear behaviour can be seen. This result may be explained by the badly defined pulse rate of the accelerator at the low end of the pulse repetition frequency range.

TABLE 1. MEASURED IONIZATION CURRENT, *I*, IN THE *H**(10) CHAMBER AT A DISTANCE OF 1m FROM THE ISOCENTRE USING DIFFERENT ACCELERATION HIGH VOLTAGES (HV) AND PULSE REPETITION FREQUENCIES *F*. F_0 = 100 Hz WAS CHOSEN AS THE NORMALIZATION CONDITION

HV in MV	*f* in Hz	f/f_0	I/I_0
6	6.25	0.0625	0.04
6	50	0.5	0.50
6	100	1	1.00
6	200	2	1.99
10	6.25	0.0625	0.05
10	100	1	1.00
10	200	2	2.00
15	6.25	0.064	0.04
15	49	0.5	0.50
15	98	1	1.00
15	196	2	2.01

3.2. Check for sufficient build-up

To test whether the wall of the *H**(10) chamber yields a sufficient dose build-up, an additional PMMA plate of 300 mm × 300 mm × 3 mm was positioned directly in front of the ionization chamber, facing the scattering object. Using the 100 mm scattering object, the effect of the plate at an acceleration voltage of 6 MV was: $I_{\text{with plate}} / I_{\text{without plate}} = 0.99$ and at 15 MV $I_{\text{with plate}} / I_{\text{without plate}} = 0.94$. Therefore, the plate reduces the measured current in the ionization chamber in both cases, and no build-up was found. The plate reduces the ionization current due to photons and may be due to electrons from the radiation field.

3.3. Dependence on the scattering object's volume

The main aim of this investigation was the determination of the effect of the scattering object's volume on the ionization current. First the 300 mm PMMA sphere was installed in the primary beam and the ionization current was measured. Then the sphere was replaced by the 100 mm PMMA cylinder and again the ionization current was measured. The ratio of the

volumes is: $V_{sphere} / V_{100\,mm} = 16$ and it is expected that the ionization currents will have the same ratio. The result is: $I_{sphere} / I_{100\,mm} = 4.1$. This ratio is the same for acceleration voltages of 10 MV and 15 MV. Repeating the measurement using the 16 mm cylinder instead of the sphere: $V_{100\,mm} / V_{16\,mm} = 277$ yields an ionization current ratio of only $I_{100\,mm} / I_{16\,mm} = 1.4$ at an acceleration voltage of 15 MV.

According to these results, the ionization current does not scale linearly with the volume of the scattering object. The scaling assumption has the prerequisite, that the scattering object is the only radiation source, which does not seem to be fulfilled.

3.4. Source of the radiation

To check the validity of the assumption that the scattering object is the main source of radiation, is to test if the dose rate decreases according to the distance r from the scattering object at the isocentre. If the assumption of a point source is correct, the dose rate should decrease with an increasing distance r following $1/r^2$.

Therefore, the $H^*(10)$ chamber was positioned at distances r of 1 m, 2 m and 3 m from the scattering object perpendicular to the primary beam axis. The results are summarized in Tab. 2. The results in Tab. 2 do not follow the $1/r^2$ law, i.e. $I \times r^2$ is not constant. This indicates that the scattering object at the isocentre is not the only source of scattered radiation in the accelerator room.

TABLE 2. MEASURED IONIZATION CURRENT I IN THE H*(10) CHAMBER AT A DISTANCE r FROM THE CENTRE OF THE SCATTERING OBJECT (300 mm SPHERE) USING DIFFERENT ACCELERATION HIGH VOLTAGES (HV) AND FREQUENCIES f

HV in MV	f in Hz	distance r in m	I in A	$I \times r^2$ nA·m^2
6	400	1.0	$3.16 \cdot 10^{-9}$	3.16
6	400	2.0	$8.82 \cdot 10^{-10}$	3.53
6	400	3.0	$4.63 \cdot 10^{-10}$	4.16
10	200	1.0	$2.24 \cdot 10^{-9}$	2.24
10	200	2.0	$5.82 \cdot 10^{-10}$	2.33
10	200	3.0	$3.14 \cdot 10^{-10}$	2.82

3.5. First improvements

The previous results indicate a strong influence of scattered or stray radiation not originating from the scattering object. The main source is assumed to be inside the accelerator head. Therefore, an additional shielding wall consisting of 5 cm thick lead bricks is positioned in the line of sight between the accelerator head and the ionization chamber. The current in the ionization chamber is reduced by a factor of 2.4 if the wall is in place and the 16 mm PMMA object is in the primary beam. Removing this small object from the primary beam yields only a reduction of 1 %. Inserting the 300 mm sphere into the primary beam yields an increase of the ionization current of a factor of 8.7 as compared to $I_{sphere} / I_{100\,mm} \times I_{100\,mm} / I_{16\,mm} = 5.7$ according to section 3.3.

4. DISCUSSION

The proposed setup cannot be used for testing radiation protection dosemeters without

further modifications. The results shown above demonstrate that there are many sources of scattered radiation in the scattered field of the Elekta Precise. The influence of a PMMA object in the primary beam is almost negligible for small objects and the amount of scattered radiation in the considered direction perpendicular to the primary beam does not scale linearly with the volume of the object. The almost perfect scaling with the repetition frequency *f* (see 3.1) can be understood since all scattered radiation depends on the primary beam dose rate. If a suitable collimator is installed around the dosemeter the scattered field of the 300 mm sphere might be useful for testing dosemeters, since the difference with and without a sphere is almost a factor of 10.

Further investigations of the scattered radiation field, especially with a focus on the dose rates in the maze of the accelerator room have been performed and can be found in reference [7].

REFERENCES

[1] ANKERHOLD, U., HUPE, O. AND AMBROSI, P., Deficiencies of active electronic radiation protection dosemeters in pulsed fields, Radiat. Prot. Dosim. 135, (2009) 149-153.

[2] AMBROSI, P., BOROWSKI, M. AND IWATSCHENKO, M., Considerations concerning the use of counting active personal dosemeters in pulsed radiation fields of ionising radiation, Radiat. Prot. Dosim. 139 (2010) 483-493.

[3] ZUTZ, H., HUPE, O., AMBROSI, P. AND KLAMMER, J., Determination of relevant parameters for the use of electronic dosemeters in pulsed fields of ionizing radiation, Radiat. Prot. Dosim. 151(3) (2012) 403-410.

[4] DUFTSCHMID, K., HIZO, J. AND STRACHOTINSKY, C., A Secondary Standard Ionisation Chamber for the Direct Measurement of Ambient Dose Equivalent H*(10), Radiat. Prot. Dosim. 40 (1992) 35-38.

[5] ANKERHOLD, U., Optimisation of a secondary standard chamber for the measurement of the ambient dose equivalent, H*(10), for low photon energies, Radiat. Prot. Dosim. 118 (2006) 16-21.

[6] ISO, X and gamma reference radiation for calibrating dosemeters and dose rate meters and for determining their response as a function of photon energy -- Part 1: Radiation characteristics and production methods, ISO 4037-1:1996.

[7] ZUTZ, H. AND HUPE, O., Ambient dose and dose rate measurements in the vicinity of elekta precise accelerators for radiation therapy, Radiat. Prot. Dosim. (2013) doi: 10.1093/rpd/nct356.

PULSED RADIATION FACILITY WITH ABOUT 115 NS PULSE DURATIONS
CHARACTERIZATION AND QUALITY ASSURANCE

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Abstract

The Physikalisch-Technische Bundesanstalt (PTB) uses a novel reference field for type testing radiation protection dosemeters in pulsed radiation fields [1]. This facility covers most of the parameters of typical workplace fields in the medical sector. There are, however, some special workplace fields with very short pulse durations where this test facility is not adequate for testing a dosemeter, due to its minimum pulse duration of 0.2 ms. Thus, PTB combined a commercially available mobile X-ray flash unit with PTB standard measuring equipment. By doing this, there is a chance to expand the test possibilities of the reference field to pulsed radiation with pulse durations of a few hundred nanoseconds.

1. INTRODUCTION

Most personal and area radiation protection dosemeters are used in pulsed radiation fields. To be sure that a dose indication of a dosemeter is correct, it has to be tested in appropriate radiation fields. As a consequence there is a need for a test facility which produces pulsed radiation. PTB has installed such an X-ray unit [1] as a reference field for pulsed radiation. This X-ray unit can produce pulses with a duration of about 0.2 ms up to continuous radiation. This covers a large range of pulsed radiation fields such as the medical diagnostic sector. But there are other application areas, e.g. electron synchrotron radiation units, that produce radiation pulses much shorter than 0.2 ms. Therefore, PTB set up a test facility which can produce radiation pulses of a few hundred nanoseconds.

This paper describes a method to characterize and to ensure the quality assurance and traceability of such a test facility, which can be used for type tests of dosemeters in pulsed radiation fields with pulse durations of about 115 ns.

2. SETUP OF THE TEST FACILITY

The test facility consists of an X-ray flash unit from Golden Engineering, a monitor ionization chamber from PTW (TM786) [2], two line lasers for marking the beam axis and a mobile measuring system which includes the high-voltage supply for the monitor chamber, the electrometer, sensors for environmental parameters and the computer interface [3], see Fig. 1. The monitor ionization chamber is permanently installed in the facility setup and has to be calibrated by means of a well characterized transfer ionization chamber for K_a, e. g. a cylinder stem ionization chamber from PTW (TM23361). This setup is similar to the standard reference X-ray facilities at PTB. The main difference, however, between this setup and all other reference fields is that no parameter of the XR200 flash generator, e.g. tube voltage, tube current, repetition frequency or pulse durations, can be adjusted or measured directly. Only the number of pulses can be varied. According to the manufacturer's manual [4] the high voltage is about 150 keV.

Consequently, there is a need to find methods to characterize such an unknown radiation source. As a first step, the spectra, the pulse duration and the pulse shape have to be measured. This has already been done by Behrens [5] and Zutz et. al. [6]. As a result of both publications, for the XR200, the mean energy is about 54 keV and the pulse duration is about 115 ns with a double-peak structure.

As a next step the conversion coefficient from air kerma free-in-air, K_a, to ambient dose equivalent, $H^*(10)$, and personal dose equivalent, $H_p(10,0°)$ should be determined [7], in order to derive $H^*(10)$ and $H_p(10,0°)$ from measurements of K_a with the monitor ionization chamber.

FIG. 1. Pulsed radiation facility with a pulse duration of about 115 ns. XR200 flash unit on the left and monitor chamber TM786 on the right

3. DETERMINATION OF THE CONVERSION COEFFICIENTS $H^*_K(10,XR200)$ AND $H_{PK}(10,XR200,0°)$

If only a standard instrument for the measurand K_a is used for dosimetric measurements, e.g. a monitor ionization chamber, then appropriate conversion coefficients shall be applied to the measured K_a values for the phantom related measurands $H^*(10)$ and $H_p(10)$. These conversion coefficients shall, in principle, be determined by spectrometry for any reference field. But there is also a second method to determine the conversion coefficients. This method uses dosimetric measurements and is described in the following.

The measurement of the conventional quantity values has to be undertaken for the measurands K_a, $H_p(10)$ and $H^*(10)$, each with the appropriate ionization chamber at the same irradiation conditions, e.g. at a distance of 1 m. As a result the conversion coefficients for the radiation source XR200 are given by

$$h_{pK}(10,XR200,0°) = H_p(10)/K_a \quad (1)$$

and

$$h^*_K(10,XR200) = H^*(10)/K_a \quad (2)$$

Where,

$H_p(10,0°)$ is the conventional quantity value of the personal dose equivalent,
K_a is the conventional quantity value of the air kerma free-in-air and
$H^*(10)$ is the conventional quantity value of the ambient dose equivalent.

A prerequisite for the experimental determination of these values is the use of nearly energy-independent secondary standard ionization chambers for the respective measurands, see

Table I and Figs. 2 and 3. In addition, these ionization chambers are capable of measuring short radiation pulses in contrast to most electronic personal and area radiation protection dosemeters. [8] All three chambers should be calibrated for the same radiation quality which is similar to the spectra produced by the XR200, e.g. C150 [9] or RQR9 [10]. Due to the unknown filtration of the XR200 the uncertainty for the calibration factor of each ionization chamber is estimated to be about 3 %. To measure simultaneously with the monitor chamber and the secondary standard chamber, a second electronic system, identical to the mobile measuring system for the monitor chamber, is used.

TABLE I. USED SECONDARY STANDARD IONIZATION CHAMBERS

Ionization chamber	**Measurand**	**Detecting volume**	**Variation of response[+]**
Cylinder stem ionization chamber from PTW (TM23361) [2]	K_a	30 cm³	3.5 %
$H_p(10)$ secondary standard chamber from PTW (T34035) [2, 11]	$H_p(10)$	10 cm³	2.5 %
HS01 ionization chamber from Seibersdorf [12, 13]	$H^*(10)$	1000 cm³	2.0 %

[+] Variation of response to the measurand in the range from about 50 keV to about 80 keV.

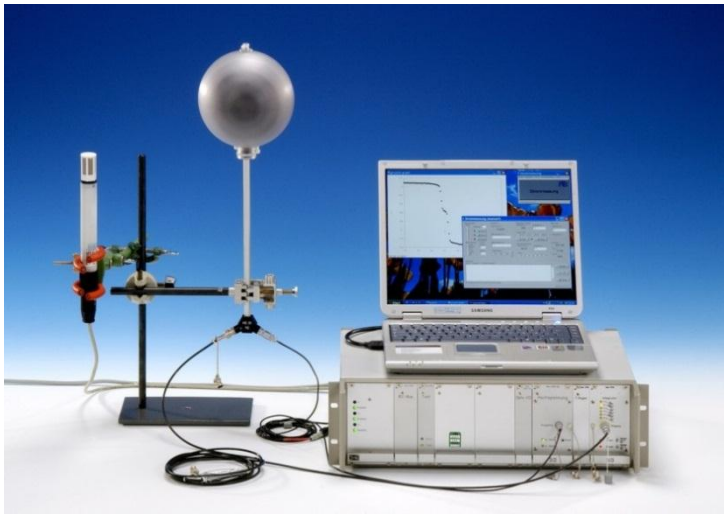

FIG. 2. Measuring setup with the H(10) ionization chamber and the corresponding electronic system [3].*

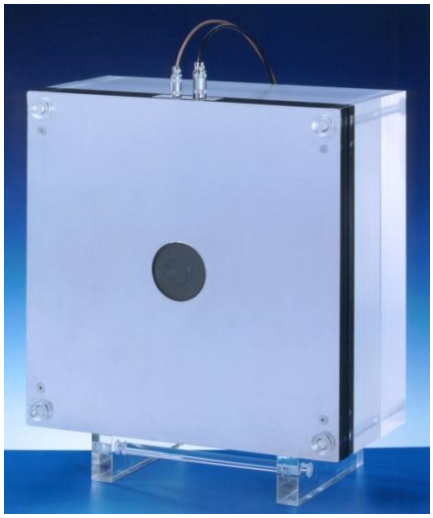

FIG. 3. Secondary standard chamber for the measurand $H_p(10)$.

4. RESULTS

In order to receive a significantly high signal (charge), each chamber, see Table I, was irradiated with a sequence of 20 radiation pulses. The measurements were carried out at two distances: 0.5 m and 1 m between the focus of the XR200 and the point of test of each chamber. Each measurement was repeated five times; the resulting coefficient of variation was below 3 %. The conversion coefficients were calculated according to equation (1) and (2) by using the mean value of the doses which were previously normalized to the corresponding measured charge of the monitor chamber. The results are summarized in Table 2 including the expanded measurement uncertainty.

TABLE 2. CONVERSION COEFFICIENTS FOR THE XR200 MEASURED AT 0.5m AND 1m

Number of pulses	Distance cm	$K_{a,mean}$ µGy	$H^*(10)_{mean}$ µSv	$h^*_K(10)$ Sv/Gy	U_h *(k = 2)*	$H_p(10,0°)_{mean}$ µSv	$h_{pK}(10,0°)$ Sv/Gy	U_h *(k = 2)*
20	50	177.4	212.2	1.21	9 %	215.6	1.24	11 %
20	100	43.9	57.4	1.36	7 %	57.6	1.34	10 %

5. DISCUSSION OF THE RESULTS

Normally, the conversion coefficient is independent of the irradiation distance. This assumes that the radiation source is a point source with a parallel radiation beam. But this is not the case for the XR200 flash unit. The divergent and non-collimated beam of the XR200 leads to an inhomogeneous irradiation of the secondary standard ionization chambers. This could be the reason for the difference between the conversion coefficient values at a distance of 0.5 m and 1 m.

Another indication of the inhomogeneous irradiation is the non-fulfilment of the inverse square law. The deviation from the inverse square law is about 1 % for the cylinder stem chamber whereas the deviation for the $H^*(10)$ ionization chamber is about 8 %, and for the $H_p(10)$ ionization chamber it is about 7 %.

6. SUMMARY

It is possible to determine the conversion coefficient for any radiation source with an expanded measurement uncertainty of about 10 %, if nearly energy-independent ionization chambers are used. Knowledge about the spectra and the resulting mean energy of the XR200 is necessary to choose the nearest reference quality [14] and the corresponding calibration factor for the respective ionization chamber. Through this, the uncertainty for the calculated conversion coefficient can be reduced.

The characterization of the presented test facility has been completed by the determination of the conversion coefficients. Following the monitor chamber was calibrated for the measurand K_a traceable to the primary standard of PTB. From now on dosemeters can also be tested with radiation pulse durations of a few hundred nanoseconds for the measurand $H^*(10)$ or $H_p(10)$ by using the determined conversion coefficients.

REFERENCES

[1] KLAMMER, J., et. al., Novel reference radiation fields for pulsed photon radiation installed at PTB, Radiation Protection Dosimetry 151(3) (2012) 478-482.
[2] PTW, Detectors, Product Catalog (2014).
[3] HUPE, O., et. al., Determination of ambient and personal dose equivalent for personnel and cargo security screening, Radiation Protection Dosimetry 121(4) (2006) 429-437.
[4] GOLDEN ENGINEERING INC., XR200 X-ray source operator's manual, Golden Engineering Inc.
[5] BEHRENS, R., A spectrometer for pulsed and continuous photon radiation, Jinst (2009).
[6] ZUTZ, H., et. al., Recent results of irradiations of DIS-1 dosemeters with an XR200 X-ray flash unit, Radiation Protection Dosimetry 154(4) (2013) 401-404.
[7] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, X and gamma reference radiations for calibrating dosemeters and doserate meters and for determining their response as a function of photon energy – Part 3: Calibration of area and personal dosemeters and the measurement of their response as a function of energy and angle of incidence, ISO 4037-3 (1999).
[8] ANKERHOLD, U., et. al., Deficiencies of active electronic radiation protection dosemeters in pulsed fields, Radiation Protection Dosimetry 135(3) (2009) 149-153.
[9] DEUTSCHES INSTITUT FÜR NORMUNG, Strahlenschutzdosimeter – Teil 1: Allgemeine Regeln, DIN 6818-1 (2004).

[10] INTERNATIONAL ELECTROTECHNICAL COMMISSION, Medical diagnostic X-ray equipment - Radiation conditions for use in the determination of characteristics, CEI/IEC 61267 (2005).
[11] ANKERHOLD, U., et. al., A prototype ionisation chamber as a secondary standard for the measurement of personal dose equivalent, $H_p(10)$, on a slab phantom, Radiation Protection Dosimetry 86(3), 167-173 (1999).
[12] DUFTSCHMIDT, K. E., et. al., A secondary standard ionisation chamber for the direct measurement of ambient dose equivalent H*(10), Radiation Protection Dosimetry 40(1), 35–38 (1992).
[13] ANKERHOLD, U., Optimisation of a secondary standard chamber for the measurement of the ambient dose equivalent, H*(10), for low photon energies, Radiation Protection Dosimetry 118(1), 16-21 (2006).
[14] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, X and gamma reference radiations for calibrating dosemeters and doserate meters and for determining their response as a function of photon energy – Part 1: Radiation characteristics production methods, ISO 4037-1 (1996).

TYPE TEST REQUIREMENTS AND REFERENCE FIELDS FOR RADIATION PROTECTION DOSIMETRY IN PULSED RADIATION FIELDS

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Abstract

It has been known for some years that electronic dosemeters may have deficiencies in pulsed fields. But for the testing of active area and personal dosemeters, the necessary type test requirements and reference pulsed fields have been missing for years. Now, the IEC/TS 62473 is the first standard defining type test requirements for dosemeters using the counting technique. The necessary reference pulsed radiation is specified in ISO/TS 18090-1.

1. INTRODUCTION

The application of pulsed radiation fields for research, security screening and medical investigations has increased remarkably in the past few years. The testing of active area and personal dosemeters for operation in pulsed radiation fields is a necessity to judge the suitability of the dosemeter. Up to now, nearly all radiation protection dosemeters have only been tested in continuous fields, although they are used for measurements in pulsed radiation fields as well.

2. MEASUREMENTS

When carrying out measurements in pulsed radiation fields which have already been used for a long time in medicine, it turned out, that the measurements of electronic radiation protection dosemeters are reliable only to a limited extent [1-3]. This is due to the method of measurement applied, which is usually of the "counting" type. Modern electronic area and personal dosemeters are mostly equipped with one or several detectors with complex measuring electronics. Consequently, a dosemeter does not only count single events through the interaction of the photons with the detector. The measuring electronics should rather also compensate for known deficits such as, for example, the dependence of the detector on the energy of the radiation or the dead time. The dead time is the time in which a dosemeter – after a counted event – can no longer detect further events. The dead time is responsible for the fact that very short X-ray pulses with a high dose rate can hardly be measured. Thus, it does not matter whether the dead time of the dosemeter is caused by the detector, by the electronic system, or by both.

As an example, measurements with two widely used electronic personal dosemeters, EPD MK2 and DMC 2000 S, at the reference pulsed field of the Physikalisch-Technische Bundesanstalt (PTB) [4, 5] are shown in figure 1. The response of the dosemeters, normalized to the response value at 0.4 Sv/h, is plotted vs. the increasing pulse dose rate. The dose of each single pulse of $H_p(10)$ = 1 mSv was kept constant. It can be seen that both dosemeters measure well as long as the pulse dose rate is only a few sieverts per hour. At higher pulse dose rates, the response drops very sharply. Therefore, a correction factor for special

workplace conditions cannot be applied, as it would, e.g., depend on the distance from the source.

For comparison, in medical X-ray diagnostics, the dose rate in the scattered radiation field is in the order of a few millisieverts per hour, but in the direct beam the dose rate can be up to 400 Sv/h, where the response of the electronic dosemeters is very low. In case of an accident, e.g. exposure in the direct beam, the dosemeter would give wrong results or would even fail completely for dose rates above 200 Sv/h.

In contrast to electronic dosemeters, for passive dosemeters with, e.g., film, TLD or the direct ion storage [6] detectors, it can be expected that they will measure correctly in pulsed radiation fields due to their measuring principle.

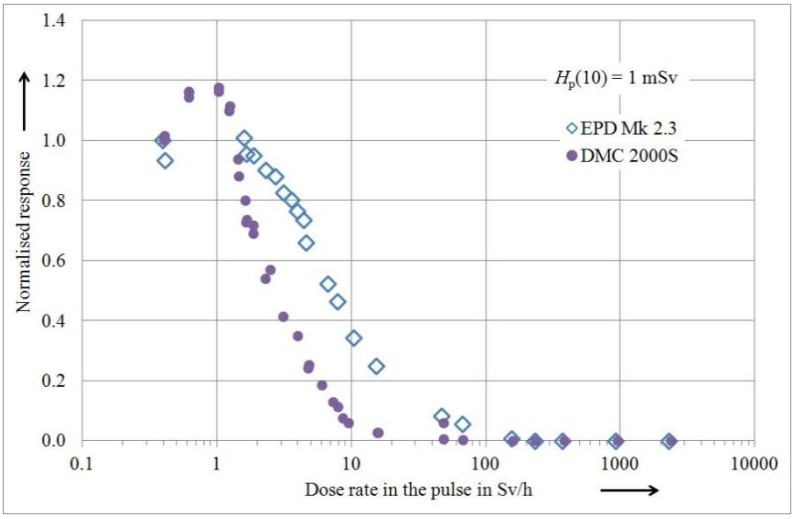

FIG. 4. Response of an EPD Mk2 and a DMC 2000 S for a constant dose per pulse of 1 mSv (RQR8) at decreasing pulse duration and therefore increasing pulse dose rate. The response values are normalized to the value at 0.4 Sv/h and the relative expanded measurement uncertainty (k = 2) is in the order of about 7 %

3. TYPE TEST REQUIREMENTS AND REFERENCE PULSED FIELDS

Up to now, type tests have been performed only in continuous radiation fields, which are equivalent to pulse durations longer than 10000 ms with maximum dose rates of 10 Sv/h. The results of a test in continuous radiation can hardly be transferred to the situation in pulsed fields, due to the above- mentioned measuring electronics. By performing measurements at constant pulse dose but different pulse duration (pulse dose rate), the usability of a dosemeter for measurements in pulsed fields can be estimated.

For active direct reading dosemeters, which use pulse counting techniques, a concept of requirements and testing procedures has already been developed as a technical specification (IEC/TS 62743 Ed.1:2012-09 “Radiation protection instrumentation — electronic counting dosemeters for pulsed fields of ionizing radiation” [7]).

The determination of the relevant parameters according to this standard is shown for an electronic personal dosemeter of type EPD Mk2, in [3]. For instruments not based on counting techniques, performing a set of measurements similar to those shown in figure 1 could be used to gain information about the ability of the dosemeter to measure in pulsed fields. Such a concept is planned to be included in a new standard in the near future.

The determination of reference pulsed fields is a basic requirement for the development, testing and calibration of radiation protection dosemeters as well as for the further development of radiation sources for the range of pulsed radiation. Therefore, a new

international ISO standard for pulsed reference fields dealing with the requirements for such reference fields is in preparation (ISO/DTS 18090-1 "Radiological protection — Characteristics of reference pulsed radiation — Part 1: Photon radiation", date 2014-01-30 [8]). This technical specification is approved by the ISO P-members and will be published at the end of 2014.

In this standard, the pulse parameters are determined by using an equivalent trapezoidal pulse, which is equivalent with respect to dose and dose rate, see Fig. 2. The determination of the equivalent trapezoidal pulse is described in annex A of the technical specification [8].

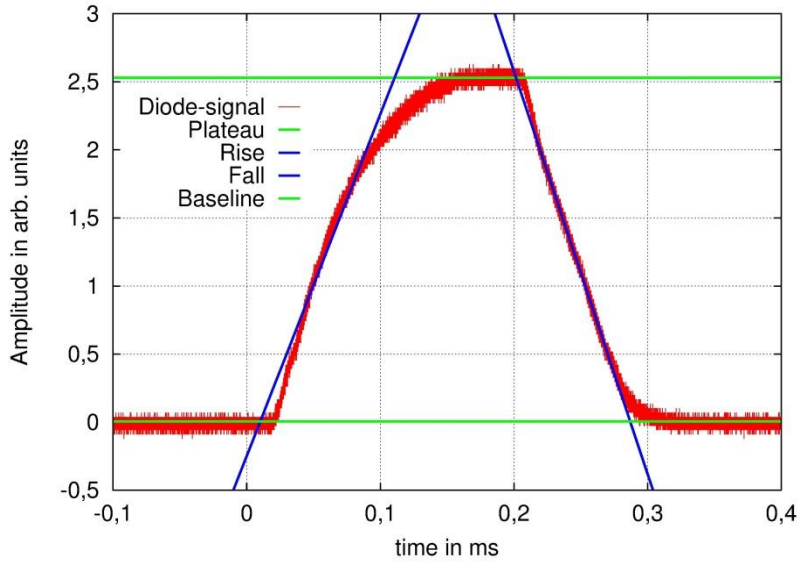

FIG. 5. Result of a time-resolved measurement of the air kerma rate during the pulse with a semiconductor diode and a scope. The measured air kerma rate is then transferred into an equivalent trapezoidal radiation pulse. This procedure is described in detail in ISO/DTS 18090-1 [8].

The relevant parameters of the radiation fields, for example, pulse duration t_{pulse}, pulse rise time $t_{pulse,rise}$, pulse fall time $t_{pulse,fall}$, pulse dose rate and dose per pulse are then defined on this equivalent trapezoidal pulse, see figure 3. As a requirement, the quotient of the integral over the indicated dose rate values of the trapezoidal pulse and the integral over the indicated dose rate values of the measured radiation pulse shall not deviate from unity by more than 0.03.

It is necessary to determine the dose per pulse by independent measurements with an ionisation chamber. The measurement of the pulse shape and the determination of the pulse duration using a diode detector and an associated amplifier are described in annex B of ISO/DTS 18090-1 [8].

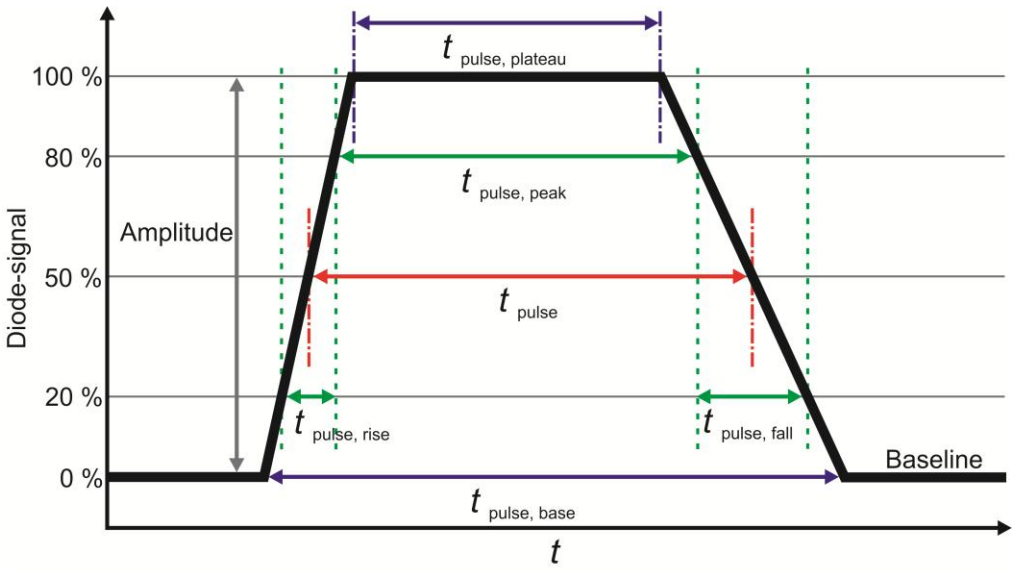

FIG. 3. Equivalent trapezoidal radiation pulse with the relevant parameters.

Currently, two reference pulsed X-ray fields are in operation; one at the French national metrology institute Laboratoire National Henri Becquerel (LNHB) and the other one at the German national metrology institute [5], the latter with pulse durations tunable from 0.2 ms up to continuous radiation.

4. CONCLUSIONS

Reference radiation fields and type test requirements are available, but adequate suitable radiation protection dosemeters are still lacking. Although diagnostic dosemeters work correctly in pulsed radiation, this technique was not transferred to radiation protection dosemeters. Therefore, especially for area dosimetry, the problems of correct measurements in pulsed radiation fields remain unsolved. Here, the experts must, for the time being, rely on their expertise to detect a possible mis-measurement of the dosemeter used. Measurements in pulsed radiation fields can, for example, be checked for plausibility by variation of the distance and application of the $1/r^2$ law. New radiation protection dosemeters suitable for pulsed radiation are urgently needed.

REFERENCES

[1] CLAIRAND, I. et al. Active personal dosemeters in interventional radiology: tests in laboratory conditions and in hospitals, Radiation Protection Dosimetry, 144 (2011) 453–458.

[2] ANKERHOLD, U., HUPE, O. AND AMBROSI, P., Deficiencies of active electronic radiation protection dosemeters in pulsed fields, Radiation Protection Dosimetry, 135(3) (2009) 149–153.

[3] ZUTZ, H., HUPE, O., AMBROSI, P. AND KLAMMER, J., Determination of relevant parameters for the use of electronic dosemeters in pulsed fields of ionising radiation, Radiation Protection Dosimetry, doi:10.1093/rpd/ncs027, online (20 March 2012).

[4] HUPE, O., ZUTZ, H. AND KLAMMER, J., Radiation protection dosimetry in pulsed radiation fields, TS2f.3, www.irpa13glasgow.com, IRPA conference 2013, Glasgow

[5] KLAMMER, J., ROTH, J. AND HUPE, O., Novel reference fields for pulsed photon radiation installed at PTB. Radiation Protection Dosimetry, 151(3) (2012) 478–482, doi:10.1093/rpd/ncs043 (2012).

[6] ZUTZ, H. AND HUPE, O., Recent results of irradiations of DIS-1 dosemeters with an XR200 X-ray flash unit. Radiation Protection Dosimetry, 154(4) (2013) 401–404.
[7] INTERNATIONAL ELECTROTECHNICAL COMMISSION, Radiation protection instrumentation — electronic counting dosemeters for pulsed fields of ionizing radiation, IEC/TS 62743 Ed.1:2012-09.
[8] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Radiological protection — Characteristics of reference pulsed radiation — Part 1: Photon radiation, ISO/DTS 18090-1 date 2014-01-30.

DOSES RECORDED THROUGH OCCUPATIONAL EXPOSURES USING TWO ENSHRINED PASSIVE DOSIMETRY METHODS AND PERSONAL MONITORING OPTION

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Abstract

In this paper, are presented the occupational exposure monitoring results from different nuclear areas, such as: research, industry, medicine, safety and inspection nuclear units recorded during the period 2009 – 2013, using two passive dosimetry methods. A part from the personnel occupational exposure was double monitored using both halide film dosimeter HFD and thermoluminescent dosimeter TLD. The statistical analysis on the number of workers and the collective dose values on dose ranges were performed. The customer preferences regarding the dosimetric method were especially highlighted. In the case of the TLD monitoring, the results show that the most workers are situated in the dose range of 1.0 and 2.0 mSv because the environmental dose was not subtracted from the occupational doses; above these limits too few workers have recorded doses. The most number of workers with doses above HFD limit of detection, almost 30 in 271 workers was recorded in 2011 in nuclear industry area. In 2009, after the TLD method accrediting, approximately 360 persons were radiological monitored by this relatively new method for the nuclear industry from Romania, and in 2013 the number of persons increased to 954; the number of workers monitored by HFD decrease from 798 in 2009 to 619 in 2013.

1. INTRODUCTION

The personal monitoring is assured by the Personnel and Environmental Dosimetry Laboratory, LDPM, from Horia Hulubei, National Institute for R&D in Physics and Nuclear Engineering, IFIN-HH, by two enshrined dosimetry methods: thermoluminescence and halide film detectors. The halide film dosimetry is performed by the IFIN – HH since ’60. In 2009, the LDPM laboratory, following the ISO 17025 recommendations among other nuclear legislations regarding radioprotection and occupational monitoring [1-5], obtained approval issued by National Commission for Nuclear Activity Control - CNCAN, nuclear Romania authority, and RENAR accreditation to perform personal monitoring not only by the HFD method but also by the TLD method. So, starting with 2009, some LDPM customers chose HFD or TLD method and other have opted for double radiological monitoring. The TLD method is used in Romania since ‘80 only in some nuclear places such as Cernavoda power plant. The Cernavoda personal dosimetry uses TLD only locally and not provides services for other nuclear units. The most occupationally exposed workers being monitored with HFD. Generally, implementation of a new method for dosimetry into the radioprotection routine rules requires effort not only by the personal monitoring service provider but the client part also. So, the dosimetry laboratory has to adopt a new dosimetry method, to supply services according with the quality and testing rules, to elaborate new strategy of marketing, to obtain the low cost services and the customers have to change the rules of the personal dosimeter handling, to perform the training of the workers.

The statistical data regarding the number of workers and the dose values on dose ranges were reported by Photodosimetry laboratory, IFIN-HH to the ESOREX program via CNCAN nuclear authority beginning with 2005. Since 2009 the personal doses recorded by TLD method and only a part of data recorded by HFD method have been reported. So, regarding the persons double monitored, only the doses measured by TLD method were reported to the CNCAN nuclear authority in order to be centralized in ESOREX program. Likewise, others papers report this kind of statistic taking into consideration different criterions regarding dosimetry method and nuclear area features, [6-8]. This paper is especially focused on the doses recorded by the same persons that were monitored both HFD and TLD and the "new dosimetry method" impact on the customers from the nuclear area. In this way, the statistic on number of workers on dose range, the collective doses on the dose ranges, the doses recorded by the person double radiological monitored and the evolution of the customer number regarding the personal dosimetry option starting with 2009, the year when the TLD method was accredited, are presented.

2. MATERIALS, METHODS AND RESULTS

The halide film and thermoluminescent dosimeter systems, types, methods and performances are described extensively in scientific papers developed by the LDPM laboratory [9,10]. The HFD consists of the Agfa personal monitoring film with two types of emulsions (D2 low speed and D10 very sensitive) and the FB-III-D dosimetric badge made by Nuclear & Vacuum, Romanian which contains different filters: Al, Cu, Pb and plastic. The TLD contains GR-200A type detectors introduced into the DIT–MF dosimetric badge with aluminum and plastic filters. The measurement uncertainty is below the 15% limit permitted by regulations in force [3]. The HFD method has a good accuracy on the 0.1 mSv - 1000 mSv dose range depending on the radiation energy and over the (0.001-100) mSv dose range in the case of TLD method. Both methods allow the X, γ and β radiation measurements over the 30 keV – 3 MeV energy range.

3. RESULTS AND DISCUSSIONS

The personal monitored by the LDPM are from different nuclear fields namely: research area, industry, medicine, safety and inspection. In 2009 the number of the worker monitored by LDPM laboratory was about 1000 which add almost 100 of double monitored workers from nuclear research area. In 2013 the number of the clients increased to 1400 but the number of the persons double monitored has remained the same. The number of monitored workers is approximated due to fluctuation phenomena.

The TLD system development according with the testing and quality control standards led to the customer trusting increase into TLD method. Many nuclear laboratories have chosen the TLD device for personal monitoring taking into consideration the working conditions with different radioactive sources and the TLD performances. The major changelings occurred in the nuclear research area where the customers have optioned for TLD instead of HFD. Later, the clients from industry, safety and inspection fields have shown interest for the personal monitoring with TLD. The Table 1 presents the number of workers involved in nuclear activities which use gamma and X-ray radiation sources and that were monitored by LDPM with HFD and TLD during the period 2009 – 2013. About 100 workers from research area were double monitored. During the studied period, ten persons have recorded doses over the HFD minimum detection limit (MDL) so, only for these subjects the doses measured by halide film and TL dosimetry methods have been compared. The dose values were comparable only for five persons and ranged with about 0.55 mSv; for other five persons the

difference was more significant, until to 4.00 mSv. The dosimetry investigations have shown that persons in question wore only the film dosimeters during the special nuclear activities.

TABLE 1. THE NUMBER OF WORKERS MONITORED BY TLD AND HFD DURING THE PERIOD 2009 - 2013

year	Dosimeter Type	Industry	Medicine	Research	Safety and inspection	Total clients (workers)
2009	TLD	-	-	233	127	360
	HFD	358	18	155	267	798
2010	TLD	-	-	268	132	400
	HFD	253	106	129	263	777
2011	TLD	380	-	261	139	780
	HFD	291	93	141	151	685
2012	TLD	510	-	253	138	901
	HFD	256	121	143	206	738
2013	TLD	550	-	272	132	954
	HFD	268	59	141	137	619

In research area about 95-100 workers were monitored both TLD and HFD.

In Table 2 is presented the number of workers from different nuclear activities on dose ranges.

The environmental dose rate during the period 2009 – 2013 was about 85 ÷ 120 nSv/h. So, in the case of dose measuring with TLD the environmental dose was not subtracted and from this reason almost the workers have doses between 1.0 mSv and 2.0 mSv and too few persons have recorded doses over this limit.

TABLE 2. THE NUMBER OF WORKERS ON DOSE RANGES MONITORED BY FILM AND TL DOSIMETERS DURING THE PERIOD 2009 – 2013

Year	Activity	Number of workers								Dosimeter type
		< MDL	(0.1-0.2) mSv	(0.2-0.5) mSv	(0.5-1.0) mSv	(1.0-2.0) mSv	(2.0-5.0) mSv	(5.0-10.0) mSv	(10.0-15.0) mSv	
2009						231	2			TLD
	Research	148	-	-	2	4	1	-	-	HFD
		-	-	-	-	124	3	-	-	TLD
	Security	267	-	-	-	-	-	-	-	HFD
	Industry	339	-	6	2	3	5	2	1	HFD
	Medicine	18	-	-	-	-	-	-	-	HFD
2010		-	-	-	-	265	2	1	-	TLD
	Research	124	-	2	-	1	2	-	-	HFD
		-	-	-	-	132	-	-	-	TLD
	Security	263	-	-	-	-	-	-	-	HFD
	Industry	236	2	6	3	3	3	-	-	HFD
	Medicine	106	-	-	-	-	-	-	-	HFD
2011		-	-	-	-	259	2	-	-	TLD
	Research	141	-	-	-	-	-	-	-	HFD
		-	-	-	-	139	-	-	-	TLD
	Security	151	-	-	-	-	-	-	-	HFD
		-	-	-	-	380	-	-	-	TLD
	Industry	271	9	9	-	3	9	-	-	HFD
	Medicine	81	6	6	-	-	-	-	-	HFD
2012		-	-	-	-	251	2	-	-	TLD
	Research	143	-	-	-	-	-	-	-	HFD
		-	-	-	-	137	1	-	-	TLD
	Security	206	-	-	-	-	-	-	-	HFD
		-	-	-	-	510	-	-	-	TLD
	Industry	226	2	3	6	2	2	2	-	HFD
	Medicine	115	5	-	1	-	-	-	-	HFD

Table 2. Continued

2013		-	-	-	-	271	1	-	-	TLD
	Research	141	-	-	-	-	-	-	-	HFD
		-	-	-	-	131	1	-	-	TLD
	Security	135	-	-	2	-	-	-	-	HFD
		-	-	-	-	550	-	-	-	TLD
	Industry	261	-	-	2	4	1	-	-	HFD
	Medicine	59	-	-	-	-	-	-	-	HFD

In 2011 about 42 from 653 workers monitored with HFD record doses over HFD limit of detection, and collective dose was 39.52 mSv, Table 3. Generally, few workers records doses over the detection limits and the collective doses are low, too.

TABLE 3. THE COLLECTIVE DOSE ON DOSE RANGES FOR OCCUPATIONAL EXPOSURE MONITORED BY FILM AND TL DOSIMETERS DURING THE PERIOD 2009 – 2013

Year	Activity	Collective dose								Dosimeter type
		< MDL	(0.1-0.2) mSv	(0.2-0.5) mSv	(0.5-1.0) mSv	(1.0-2.0) mSv	(2.0-5.0) mSv	(5.0-10.0) mSv	(10.0-15.0) mSv	
2009		-	-	-	-	332,6	5.73	-	-	TLD
	Research	148	-	-	1.3	5.4	3.12	-	-	HFD
		-	-	-	-	163.7	6.8	-	-	TLD
	Security	267	-	-	-	-	-	-	-	HFD
	Industry	339	-	1.89	1.37	4.07	15.83	12.1	10.0	HFD
	Medicine	18	-	-	-	-	-	-	-	HFD
2010		-	-	-	-	328.3	6.5	7.3	-	TLD
	Research	124	-	0.9	-	1.2	4.1	-	-	HFD
		-	-	-	-	95.3	-	-	-	TLD
	Security	263	-	-	-	-	-	-	-	HFD
	Industry	236	0.28	2.28	1.93	4.01	8.42	-	-	HFD
	Medicine	106	-	-	-	-	-	-	-	HFD

Table 3. Continued

2011		-	-	-	-	285.5	5.7	-	-	TLD
	Research	141	-	-	-	-	-	-	-	HFD
		-	-	-	-	188.1	-	-	-	TLD
	Security	151	-	-	-	-	-	-	-	HFD
		-	-	-	-	501.6	-	-	-	TLD
	Industry	271	1.24	2.51	-	5.16	27.88	-	-	HFD
	Medicine	81	0.83	1.9	-	-	-	-	-	HFD
2012		-	-	-	-	324.5	6.2	-	-	TLD
	Research	143	-	-	-	-	-	-	-	HFD
		-	-	-	-	179.3	2.8	-	-	TLD
	Security	206	-	-	-	-	-	-	-	HFD
		-	-	-	-	673.2	-	-	-	TLD
	Industry	226	0.25	0.72	3.6	3.2	12.97	11.67	-	HFD
	Medicine	115	5	-	1	-	-	-	-	HFD
2013		-	-	-	-	350.7	2.9	-	-	TLD
	Research	141	-	-	-	-	-	-	-	HFD
		-	-	-	-	175.2	2.2	-	-	TLD
	Security	135	-	-	1.1	-	-	-	-	HFD
		-	-	-	-	693.4	-	-	-	TLD
	Industry	261	-	-	1.48	5.33	2.72	-	-	HFD
	Medicine	59	-	-	-	-	-	-	-	HFD

4. CONCLUSIONS

Few workers have recorded doses over HFD limit of detection during the studied period. Generally, these are from industry and most of the cases have been recorded in 2011 when about 10% from monitored persons have had doses above 0.1 mSv HFD detection limit. The most persons from those that record doses above HFD detection limit are situated under 5 mSv value (annual dose for population allowed by nuclear safety norms). The TLD monitoring shows that almost all the occupation exposures are situated under 2 mSv, and in these cases, the environmental radiation was taken into account.

Almost all persons double monitored from research field are under detection limit of the HFD and between 1.0 to 2.0 mSv in case of TLD monitoring. Maximum two persons have recorded dose from 2.0 to 5.0 mSv dose range in the both cases of personal dosimeter monitoring.

The results show that occupationally exposed workers have not recorded doses above the annual limit of dose allowed by radiation protection norms.

After TLD method accreditation in accordance with nuclear safety norms and quality assurance standards, many nuclear laboratories especially from research and industry fields have opted for TLD dosimeters. So, the number of workers radiological monitored by TLD method increased from 360 persons in 2009 to 954 persons in 2013.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, International basic safety standard for protection against ionizing radiation and for the safety of radiation sources. Safety Series No. 115-1, IAEA, Vienna (1994).

[2] INATIONAL COMMISSION FOR NUCLEAR ACTIVITIES CONTROL, Fundamental Norms of Radiological Safety, NSR-01, CNCAN, Bucharest (2002).

[3] NATIONAL COMMISSION FOR NUCLEAR ACTIVITIES CONTROL, Norms on Radiological Safety, Norms of individual dosimetry, NSR-06, CNCAN, Bucharest (2002).

[4] COMMISSION ELECTRONIC INTERNATIONAL/ INTERNATIONAL ELECTRONICAL COMMISSION, Thermoluminescence dosimetry systems for personal and environmental Monitoring, Geneva, Elvetia. CEI/IEC 61066, (2006).

[5] INTERNATIONAL STANDARD ORGANIZATION/ INTERNATIONAL ELECTRONICAL COMMISSION, General requirements for the competence of testing and calibration laboratories, ISO/IEC 17025, (2005).

[6] JABEEN, A., MUNIR, M., et al., Occupational exposure from external radiation used in medical practices in Pakistan by film badge dosimetry, Radiat. Prot. Dosim. 140 (2010) 396–401.

[7] PETROVA, K., PROUZA, Z., The system of personal monitoring and the evaluation of the occupational exposure in the Czech republic, Radiat. Prot. Dosim. 96 (2001) 63–66.

[8] MIHAI, F.; BERCEA, S.; STOCHIOIU, A., et al., Low-dose monitoring for occupational exposure and the influence of measurement accuracy", International Journal of Low Radiation 6 (2009) 37-42.

[9] STOCHIOIU, A., MIHAI, F. AND SCARLAT, F., A new passive dosimetric system with thermoluminiscent LiF:Mg, Cu, P detectors applied in individual radiation monitoring, J. Optoelectron. Adv. M **8** (2006) 1545-1551.

[10] MIHAI, F., STOCHIOIU, A., Expected performances and practical aspects in personal doses measured by halide films Photodosimeters, Romanian Reports in Physics, 65 (2013) 1535–1546.

ANGULAR DEPENDENCE OF TWO DIFFERENT (LIF BASED) EYE LENS TL DOSIMETERS

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Abstract

The dose limit for the eye lens has been lowered, which is the reason why new ways to quantify the irradiation of eye lens more precisely are developed. A new model of eye lens dosimeter designed in ORAMED project was considered in this work. The dosimeter was attached on the right eye of the antropomorphic Rando phantom and irradiated under laboratory conditions. Its angular dependence was investigated for X-ray spectra N40 and N60. The obtained results were compared with data of theoretical angular dependence of personal dose equivalent at the depth of 3 mm, $H_p(3)$. The measurements were performed for two types of TL (thermoluminescent) detectors, LiF:Mg,Ti and LiF:Mg,Cu,P. Their angular response seems to be very similar in a large range of angles, and both the materials can be used for $H_p(3)$ measurements in interventional radiology.

1. INTRODUCTION

The ICRP [1] has recommended to lower the limit of the dose to the eye lens for occupationally exposed persons to a mean value of 20 mSv per year (averaged over 5 years, with a maximum of 50 mSv per year). The reason for this change was that epidemiological studies had shown that the threshold dose for the induction of cataract is much lower than it was supposed. As the new limit value can be exceeded in common work conditions, e.g. in interventional radiology, it is advisable to measure $H_p(3)$, which represents the quantity used to estimate the dose to the eye lens.

In cooperation with RADCARD company, as a task of EU project ORAMED (Optimization of RAdiation protection for MEDical staff), a new type of TLD (thermoluminescent dosimeter/detector) EYE-D^{TM} was produced to measure $H_p(3)$ in close proximity of the eye [2]. The used TL detector was LiF:Mg,Cu,P. It was inserted in a special holder providing 3 mm of tissue equivalent polyamide. A compatible headband was utilized for fixing the dosimeter on the temple in close proximity of the eye. As a part of the project, a cylindrical phantom, imitating more real conditions of human head, was developed for calibration purposes, and applicable conversion coefficients $h_{pK}(3)$ (Sv/Gy) were derived [3, 4].

We have tested this design of eye dosimeter for two LiF based materials: LiF:Mg,Ti with natural lithium proportion (92.5% ^{7}Li + 7.5% ^{6}Li) and LiF:Mg,Cu,P (99.93% ^{7}Li + 0.07% ^{6}Li). The dosimeters were irradiated on right eye of the anthropomorphic Rando phantom. The aim was to investigate angular dependence of such dosimeters in conditions corresponding to interventional radiology.

2. METHODS OF MEASUREMENT

The eye dosimeters were placed on the right eye of the anthropomorphic Rando phantom head, which simulated human head more realistic the cylindrical phantom. Three dosimeters of each type were irradiated one by one for every set angle.

Two radiation qualities, N40 and N60, were employed. These X-ray beams from narrow-spectrum series were chosen to represent the scattered radiation field, where interventionalists usually work. Air kerma K_a= 5 mGy was applied. A highly stabilised X-ray unit Isovolt (GE Inspection Technologies, USA) with its 160 kV tube MXR 160/0.4-3.0 was employed. The required beam qualities were adjusted by means of proper Al and Cu filters, and X-ray tube voltage 40 and 60 kV for N40 and N60, respectively. The focus speciment distance was 1842 mm (reference distance).

To determine the angular dependence of EYE-D dosimeters, the head phantom was successively turned around the reference point (the right eye), from 0° with 30° spacing up to 360°. First, AP geometry (0°) was set up, i.e. the head phantom faced the X-ray tube and radiation entered the dosimeter from the front side. Irradiation of the right side of the head followed. Angles above 180° represent the situation when the dosimeter was on the side of head reversed of the radiation source.

For TL measurements, a manual TLD reader model Harshaw 3500 was utilized. The applied parameters of the TL reading for the particular TL materials are given in Table I. Calibration in terms of $H_p(3)$ was done via conversion coefficients $h_{pK}(3)$ for AP geometry for the used X-ray spectra [3]. The obtained TLD results in terms of $H_p(3,\alpha)$ were expressed in form of ratio $\frac{Hp(3,\alpha)}{Ka}$, which enabled direct comparison with the theoretical values derived for the cylindrical phantom [3]:

$$h_{pK}(3,\alpha)= \frac{Hp(3,\alpha)}{Ka}.$$

Conversion coefficients values [3] were derived for the range from 0° to 180°. There was no need to analyze the other side for the cylindrical phantom. The coefficients values are shown in Fig. 1. Those for higher angles (180-360°) correspond to the symmetrical ones for angles from 0° to 180°.

3. ANGULAR DEPENDENCE RESULTS

The results obtained for the used radiation qualities N40 and N60 and particular TLD materials are shown in Fig. 1. Each point represents the average value of three dosimeters. To draw a comparison, the data from Ref. [3] are included.

TABLE I. PARAMETERS OF TL READING

	LiF:Mg,Cu,P	LiF:Mg,Ti
Preheating	165°C for 10 s	120°C for 8 s
Temperature rate	15°C/s	15°C/s
Maximum temperature	240°C for $13^1/_3$ s	260°C for $26^2/_3$ s
Annealing	240°C for 10 s	260°C for 10 s

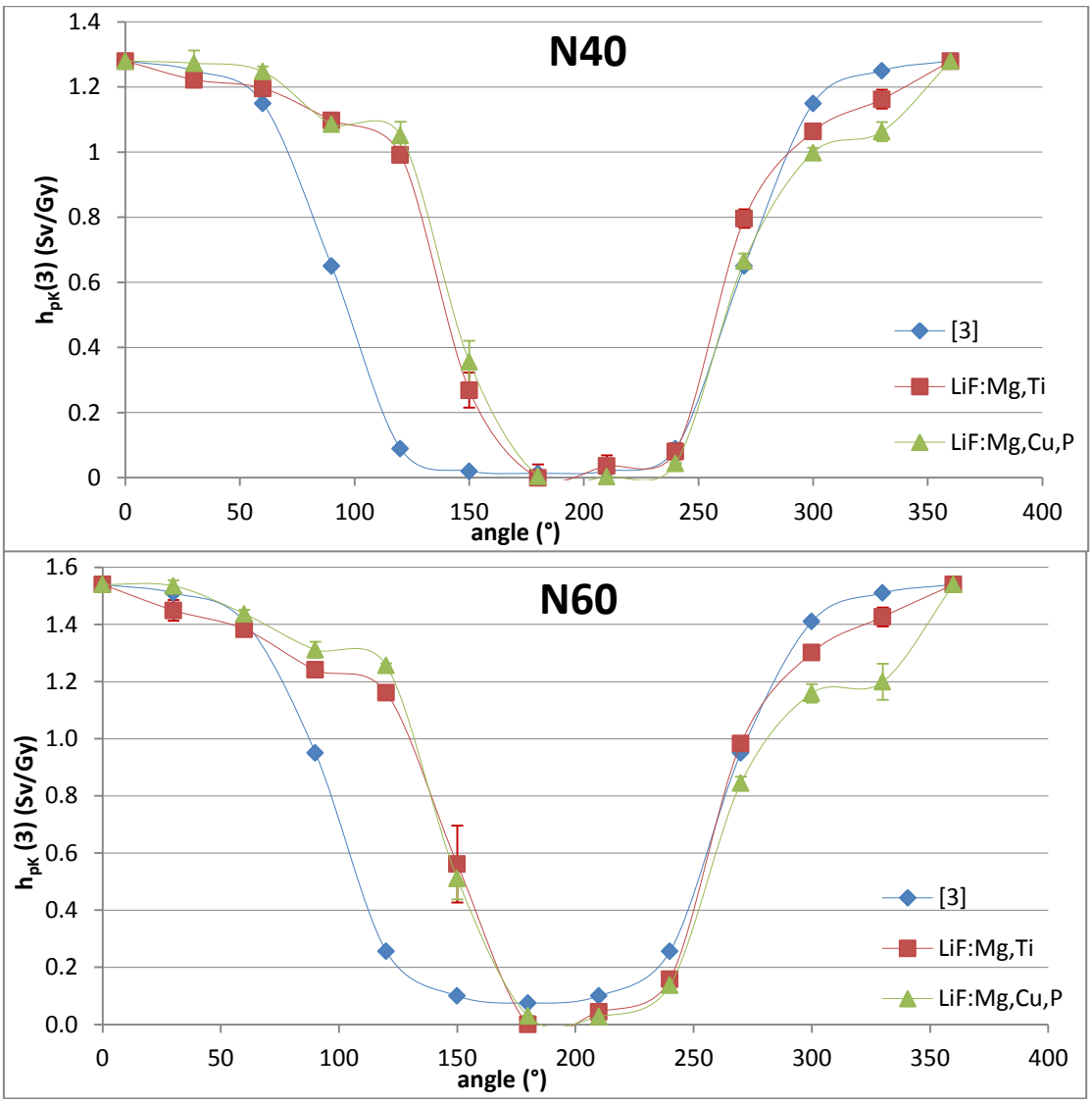

FIG. 1. The dependence of $h_{pK}(3)$ on the angle of incidence for N40 and N60 qualities. Experimental values of both tested TL materials are compared to conversion coefficients published in Ref. [3]. The standard deviation of three signals is shown as error bars.

4. DISCUSSION

The results in Fig. 1 show that no significant $H_p(3)$ underestimations were found for both the LiF based dosimeters. For angles from 60° to 150° the dosimeters tend to overestimate the theoretical $H_p(3)$ values. The overestimation is maximally by factor of 5, which was observed for angle of 120°.

Angular dependence of the used TL materials is similar for the particular X-ray qualities. LiF:Mg,Ti slightly underestimates at low angles but it evinces a better response at large angles. However, the dosimeter based on LiF:Mg,Cu,P can be more favorable for its higher sensivity and negligible fading [5].

Taking into consideration the fact that workers in interventional radiology may change their position towards the source, the new eye dosimeter can measure $H_p(3)$ reliably, even in cases when it is worn on the reverse side of head. However, if the worker is not equipped with an additional dosimeter for the second eye, this situation means that dose to the second eye lens, which can be irradiated significantly, is not measured. This fact represents the main disadvantage of dosimeters worn on the temple in close proximity of the eye.

5. CONCLUSION

The results obtained affirm that eye doses measured with this type of dosimeter are in acceptable correspondence with theoretical values of $H_p(3)$ derived for the cylindrical phantom. Both tested TLD materials, LiF:Mg,Cu,P and LiF:Mg,Ti can be used for the eye dosimeter.

REFERENCES

[1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Statement on Tissue Reactions (2011), http://www.icrp.org//page.asp?id=123.
[2] BILSKI, P., et al., The new EYE-D^{TM} dosemeter for measurements of $H_p(3)$ for medical staff, Radiation Measurements 46 (2011) 1239- 1242
[3] BEHRENS, R., Air kerma to $H_p(3)$ conversion coefficients for a new cylinder phantom for photon reference radiation qualities, Radiation Protection Dosimetry **151** (2012) 450- 455.
[4] GUALDRINI, G., et al., A new cylindrical phantom for eye lens dosimetry development, Radiation Measurements 46 (2011) 1231- 1234.
[5] MOSCOVITCH, M., Personnel dosimetry using LiF:Mg,Cu,P, Radiation Protection Dosimetry **85** (1999) 49- 56.

EQUIVALENT DOSE ESTIMATION OF EYE LENS ON PLANNED EXPOSURE SITUATION OF INDUSTRIAL GAMMA RADIOGRAPHY USING THE VISUAL MONTE CARLO BRAZILIAN SOFTWARE

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Abstract

The International Commission on Radiological Protection, ICRP publication 103, reviewed recent epidemiological evidence to suggest that, for the eye lens, the absorbed dose threshold for induction of late detriments is about 0.5 Gy. On this basis, on April 21, 2011, the ICRP has recommended changes to the occupational dose limit in planned exposure situations, reducing the eye lens dose equivalent limit of 150 mSv to 20 mSv per year, on average, during the period of 5 years, with exposure not exceeding 50 mSv in a single year. In face of this recommendation, the Brazilian Commission of Nuclear Energy (CNEN) adopted the new limit to the eyes lens through resolution CNEN 114/2011 published on September 1, 2011. Recent studies have pointed to the need for a more systematic monitoring of doses to the eye lens of radiation workers involved in planned exposure situations. However, the recent studies were done only in medical area, mainly in interventional medicine. Studies with planned exposure on nuclear and industrial areas are really needed and will be very important due to the new ICRP dose limit to the eye lens. This work proposes to estimate the dose to eye lens in planned exposure situations, notably in industrial gamma radiography which is one of the industrial applications of radiation that has a higher radiation risk and is considered as a Category 2 of the IAEA Categorization System. The estimated annual doses in eye lens, the estimated annual effective doses (whole body dose) and the percentage between the estimated annual equivalent doses to eye lens and the annual limit on equivalent dose to the lens are showed. The results show that during planned exposure situations, the operators of industrial gammagraphy could be exposed to annual doses in eye lens from 16.9 to 66.9 mGy/year, based on the investigated scenarios. It means that the new annual limit on equivalent dose to the lens (20 mSv/year) can directly impact the activities of industrial gammagraphy, mainly radioactive facilities with high number of exposures per year.

1. INTRODUCTION

Industrial Radiography is one of the nondestructive inspection methods more used in the inspection of ferrous and non-ferrous materials welds, castings and forgings, where the quality requirements of the industrial sector are necessary to avoid discontinuities in parts, components, equipment, etc [1].

Industrial Radiography is a planned exposure situation that involves high occupational doses and high number of radiation accidents. Until now , 80 radiological accidents involving 120 radiation workers, 110 members of the public and 12 deaths were recorded by IAEA.

Finding a correlation between the effect caused by radiation and the dose of radiation received, has always troubled researches and doctors. They have longed to achieve through a two-way relationship, a more precise diagnosis and primarily a more humane treatment for the victims of radiological accidents. With this intention, various researchers have tried through their scientific work to estimate doses of radiation using the technology available at

the time. The most certain data was related to high doses, to radiological accidents and to observing victims of Hiroshima and Nagasaki or experiments with guinea pigs [2].

The tissues most sensitive to radiation are the parts of the reproductive organs, bone marrow and the lenses of the eyes. This last effect is generating most attention on the part of researchers and doctors.

Cataracts caused by radiation have been proved amongst those involved with procedures using X-ray equipment. A series of studies has suggested that there can be a significant risk of opacity to the lens of the eye in people exposed to low doses of ionising radiation. Evidence for this is from: those having taken C.T. examinations, astronauts, radiology technicians, radiotherapists, as well as from data on survivors of the atomic bomb and those exposed in the Chernobyl accident [3].

The main objective of this paper was to provide an estimate of the eye lens doses, that were received by industrial gammagraphy operators in planned exposure situations. A Brazilian computer program called "Visual Monte Carlo Dose Calculation", specially developed for external dosimetry was used. A comparision between the results of the annual dose limit in the lens eyes and the investigation level established by the regulatory body was also performed.

With this estimation it will be possible to evaluate the impact of this new dosage limit and foresee adequately current scenarios and as well as this to make recommendations so that the professionals that work in radiological areas do not receive doses above those established by the regulating authority.

2. INDUSTRIAL GAMMARADIOGRAPHY SCENARIOS

The goal is to define the real and most critical scenarios, used in industrial gammagraphy, especially considering the time and distance of the operator in relation to the radioactive source during planned exhibitions.

Industrial gammagraphy uses ^{60}Co, ^{75}Se, and primarily ^{192}Ir radioactive sources. The ^{192}Ir radioactive source has been selected for this simulation because it is widely used and has greater values. To facilitate the doses extrapolation by the calculation method it was defined the activity of 1 TBq (27 Ci) as reference to estimate the computational program. Conservatively, will not be considered any kind of additional shielding, such as plates, rails, collimators and shields, that are commonly found in the radiographic scenarios. Possible scenarios depend on many factors, especially the type and location of the equipment to be inspected. A steel piece welded, for example, can be placed on the ground during the test and a pipe to be inspected can have your many height above the ground, depending on the construction and industrial purpose.

Based on field observation and interviews with operators gammagraphy, some scenario models most commonly used in industrial gammagraphy was defined. Thus, the scenarios were established three different heights of the exposed source from the ground. (Fig. 1)

The critical distance between the operator and the source is when the operator performs the procedure to exposure and retract the radioactive source. In this scenario, the operator keeps at a variable distance depending on the length and positioning of the gammagraphy equipment.

Generally, the total length of the gammagraphy equipment is almost the same for every model device. But the position angle may vary depending on the radiographic arrangement, especially in services in height. In this case, the operator keeps at a distance of 5 to 10 meters from the radioactive source considering the most critical condition, when he uses a smallest guide tube. The displacement of the source was not considered relevant in this scenario, since

the variation of the distance between the radioactive source and the simulator is contained in the range given above.

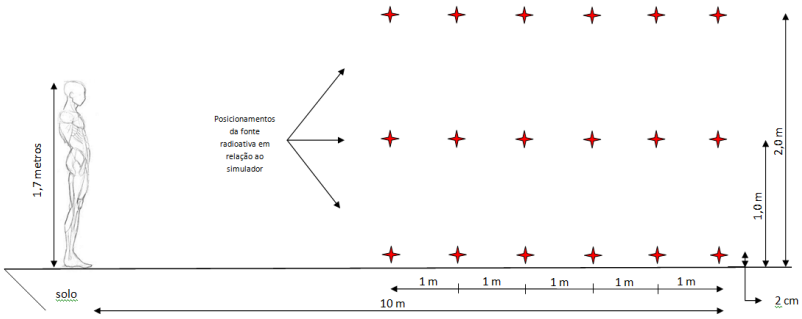

FIG. 1. Scenarios during the exposure and retract of the radioactive source

The time that the operator stay exposed during a radiography is divided into two different times: the time of exposure (the radioactive source route from the irradiator to the exposure terminal) and time of retract (the radioactive source route from the end to the device) of the radioactive source. There is no shielding during these time. The operator performs the exposure of the radioactive source (Fig. 2) after preparation of the arrangement and placement of radiographic film. The retractind is after the radiography film is irradiated. Both times depend on the length of the guide tube used in the gammagraphy equipment. The radioactive source moves in the guide tube with an average speed of 1.0 m/s. As in this scenario the length of the guide tube is 2 meters long, the radioactive source will move in an average time of 2 seconds to exposure and retract. It is necessary to take into account the time to exposure the source and the time to retract the source. Both time is approximately 8 seconds.

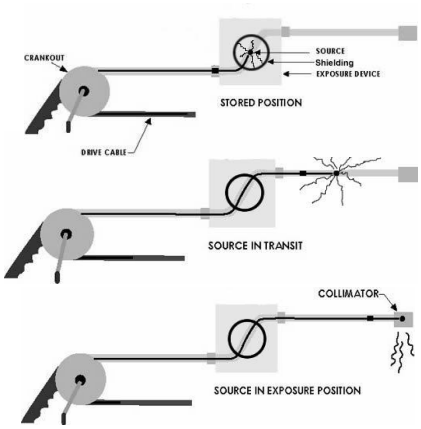

FIG. 2. Gamma radioagraphy source exposure

The behavior of the operator, mainly regarding to source retracting, can change the time to perform the radiography. The operator can perform the radiography so quickly to avoid possible overexposure in the radiography film. In this scenario, it was estimated 12 seconds to perform the exposure and retrac of the source by radiograpy/film. Then during a working day, an operator can perform 12 radiographies that means that the operator will exposure during 144 seconds on average per day [4].

3. THE COMPUTATIONAL CODE VISUAL MONTE CARLO

The modern version of the Monte Carlo method was invented in the late 1940s and since then, after seven decades, many researchers developed software to improve the estimative of radiation doses received by the workers.

A Brazilian software named “Visual Monte Carlo –VMC” was also developed using the

Monte Carlo Method and a human body simulator. The Visual Monte Carlo (VMC) transports photons, protons and alpha particles through inhomogeneous geometries, mostly through voxel geometries. The VMC photon transport algorithms have been applied to dose calculations (VMC dose calculation) and to simulating the counting geometry as applied to in-vivo measurements (VMC in-vivo). This software VMC enables to calculate the absorbed dose received by each organ and tissue relevant to the calculation of effective dose, as defined in ICRP Publication 103.

The VMC code has been effective to estimate quickly the doses of radioactive sources in planned or accidental exposures situations, especially for cases of handling these radioactive sources. The code can be used with the source near the surface of the body or inside the pocket with a very good dose estimation.

This computer program uses a simulator to permit the calculation of dose from external irradiation. The program uses the Monte Carlo technique to simulate the irradiation of organs, the transport of photons in the energy range from 0.035 to 2 MeV through the tissues and detection of radiation [5].

The human body simulator is composed of voxels. Each voxel has a cuboid shape, and its physical composition depends on the location in the body. The interaction of photons, by photoelectric effect or Compton process, there is the probability of each photon depositing energy in a specific voxel. With the amount of energy deposited can calculate the dose to the set of voxels representing each organ or body part.

In this work, the medium of interaction in the VMC program for the whole body, consisted of a simulator called ICRP Male (Figure 3). The program allows the use of simulators ICRP Male and ICRP Female, representing models of adult male and female adult reference reference.

The organs and tissues represented in VMC simulators meet the recommendations in ICRP 103 [2007]. The ICRP 103 weighting factors for organ and tissue are used in simulators and the values are established by calculating the average by gender and age for all organs and tissues, including for men and women breasts, testes, and ovaries. [6]

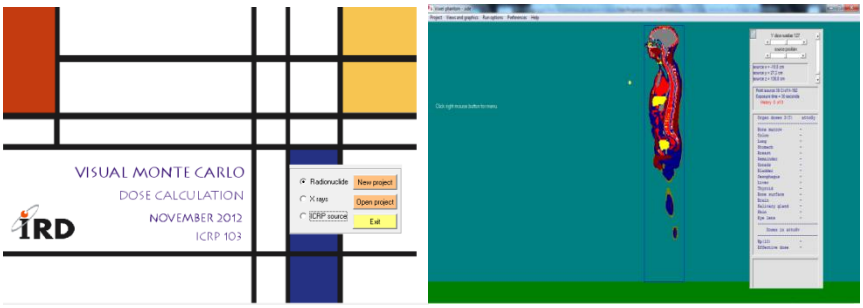

FIG. 3. VMC and ICRP Male Simulator

4. RESULTS AND CONCLUSIONS

The VMC program used to estimate dose of scenarios showed consistent results with expectations. As the analysis was performed focusing mainly on the organs of interest in radiation protection, ie, eye lens, thyroid, gonads, and the effective doses and Hp(10), the VMC program used to estimate dose of scenarios showed consistent results. The estimated annual doses in eye lens, the estimated annual effective doses (whole body dose) and the percentage between the estimated annual equivalent doses to eye lens and the annual limit to equivalent dose to the lens are showed in Table 1.

The results show that during planned exposure situations the operators of industrial gammagraphy could be exposed to annual doses in eye lens from 16.9 to 66.9 mGy/year, based on the investigated scenarios. It means that the new annual limit to equivalent dose to the lens (20 mSv/year) can directly impact the activities of industrial gammagraphy, mainly radioactive facilities with high number of exposures per year.

The use of ALARA principle during the radiography is extremely necessary, especially in planned exposure of onsite industrial gammagraphy. The use of collimators and adicionals shieldings, the division of duties for exposure and retrait the source between the operators and to stay less time near the source during the exposure is strongly recommeded to optimize the equivalent doses in the eye lens during industrial radiography testing.

Table 1 gives results of dose estimated to eye lens using Monte Carlo Methos (VCM) extended to annual doses taking into account the following planned exposure situation: ^{192}Ir radioactive source with 1 TBq of activity, 12 seconds by source exposure, 12 radiographic film by day, 250 days of work by year.

TABLE 1. RESULTS OF DOSE ESTIMATION TO EYE LENS AND EFFECTIVE DOSE USING THE MONTE CARLO METHOD

Distance Source / Operator (meter)	Estimated Annual Dose to Eye Lens (mGy)	Estimated Annual Effective Dose (mSv)	Percentage between Estimated Annual Dose to Eye Lens and Annual Equivalent Dose Limit to Eye Lens* (%)
5 (a)	62,4	52,7	312
5 (b)	66,9	54,9	334
5 (c)	63,8	52,4	319
6 (a)	45,5	36,9	228
6 (b)	46,4	38,4	232
6 (c)	44,9	37,4	224
7 (a)	34,3	27,5	171
7 (b)	35,1	28,6	176
7 (c)	36,3	27,8	182
8 (a)	26,8	21,0	134
8 (b)	26,5	22,1	132
8 (c)	26,2	21,7	131
9 (a)	19,9	16,8	99
9 (b)	22,5	17,5	112
9 (c)	20,8	17,3	104
10 (a)	17,3	13,7	87
10 (b)	16,9	14,3	84
10 (c)	17,7	14,2	89

Source height related to the ground: (a) 0.02 m (b) 1 m and (c) 2 m

*Annual Equivalent Dose Limit to Eye Lens: 20 mSv/year

REFERENCES

[1] C.M ARAUJO LIMA, "Análise de Acidentes e Incidentes Radiológicos em Radiografia Industrial", Monografia Pós-Graduação IRD, (2011).
[2] F.C.A. DA SILVA "Dosimetria Reconstrutiva de Acidentes Radiológicos usando o Método de Monte Carlo", Tese Doutorado, COPPE/UFRJ, (2003).
[3] RELID (Retrospective Evaluation of Lens Injuries and Dose). Available on: http://rpop.iaea.org/RPOP/RPoP/Content/News/relid-cataract-study.htm(last accessed on 12 March 2013).
[4] J.C. LEOCADIO, " Estimativa do Risco de Exposição Potencial em Instalações Industriais", Tese Doutorado, COPPE/UFRJ, (2007).
[5] J.G. HUNT, "Calibração de um Sistema de Medida in vivo através da Simulação da Fonte de Radiação e do Detector utilizando-se a Técnica de Monte Carlo", Tese de Doutorado, UERJ, (1998).
[6] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, 2007 Recommendations of the International Commission on Radiological Protection ICRP Report 103; Ann. ICRP 37 (2007).
[7] COMISSÃO NACIONAL DE ENERGIA NUCLEAR. NN – 3.01. Diretrizes Básicas de Proteção Radiológica, Resolução nº 114, (2011).

OPSCI: SOFTWARE TO OPTIMIZE INDIVIDUAL ROUTINE MONITORING PROGRAMME OF INTERNAL CONTAMINATION

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Abstract

Occupational exposure to internal contamination can be monitored by periodic bioassay. The frequency and the type of bioassay depend on the exposure conditions. A monitoring programme is considered to be adapted if it allows the detection of any event leading to dose higher than 1 mSv. It is therefore important to quantify the minimum dose detectable by monitoring. The OPSCI software was developped to calculate the minimum detectable dose (MDD) for a monitoring programme given the uncertainty on the exposure conditions, the bioassay frequency and the uncertainty on the biossay. Using OPSCI allows to determine the MDDs for different possible monitoring programmes and to choose the best one by compromising the MDD with the costs of the programme. OPSCI was validated and used for the evaluation and optimization of monitoring programmes of a reprocessing plant and of a MOX preparation facility.

1. INTRODUCTION

In case of potential occupational exposure, a routine monitoring programme is designed to verify and document that the worker is protected adequately against risks from radionuclide intakes and that the protection complies with regulatory dose limits and dose constraints. It is implemented by periodic bioassay measurement interpreted in terms of intake *i* and committed effective dose *d* through biokinetic and dosimetric models. After a prospective evaluation of exposure at a workplace, a monitoring programme consistent with the dose constraint can be defined by choosing *in vivo* or *in vitro* measurement technique and frequency of measurement. The International Organization for Standardization (ISO) [1] recommends criteria to specify a routine monitoring program:

(1) The consequences resulting from an unknown interval between intake and measurement shall be limited so that, on average over many intervals, doses are not underestimated and the maximum underestimation of the dose resulting from a single intake does not exceed a factor of three;
(2) The detection of all annual exposures that can exceed 1 mSv shall be ensured; and
(3) At least two measurements shall be performed annually.

On the other hand, to evaluate the quality of a program, Carbaugh [2] presented the concept of minimum detectable dose (MDD). The MDD is calculated from the minimum

detectable amount (MDA) which is "the smallest amount (activity or mass) of an analyte in a sample that will be detected with a probability ß of non-detection (Type II error) while accepting a probability of erroneously deciding that a positive (non-zero) quantity of analyte is present in an appropriate blank sample (Type I error)" as defined by the American National Standard [3]. The MDD is:

$$\boldsymbol{MDD = \frac{MDA}{B(T)} \times E_{50}}$$

Where, T is the monitoring period, B the excretion or retention of activity as a fraction of intake, and E_{50} the committed effective dose per unit of intake. From the standard, MDD has to be less than 1 mSv.

However, MDD is uncertain because the determination of MDA, E_{50} and B is subject to uncertainties. Firstly, the exposure conditions which affect E_{50} and B are usually not well defined: from a routine measurement, the intake rate (acute or chronic) and the contamination time are difficult to assess; the aerosol size distribution and its solubility are rarely known. Secondly, the activity counting is subject to Poisson variability and calibration errors, and, in case of in vitro measurement, uncertainty on the result is increased by the sampling of excreta. Therefore, uncertainty should be considered in MDD evaluation.

In this study, the uncertainties on the different parameters are modelled by probability distributions allowing the calculation of the dose probability distribution knowing the value of the MDA. The MDD is determined as the 95th percentile of this distribution. Using this approach, it can be assured that the MDD is not exceeded under a 95% level of confidence when no activity is measured in the sample.

This article presents software developed to evaluate the MDD for a monitoring programme given the uncertainty on the exposure conditions and on activity analysis, in order to help in the choice of the best monitoring programme.

2. METHODS

To calculate the Minimum Detectable Dose (MDD), previously published methods[4] are implemented in new software developed during this study. This software named OPSCI (French acronym of optimization of monitoring programmes of internal contamination) is coded in IDL® (Interactive Data Language, Exelis, McLean, VA, USA).

3. SOFTWARE DESCRIPTION

OPSCI presents 4 different modules of increasing complexity:

(1) calculation of the dose from one measurement using ICRP reference parameter values;
(2) evaluation of the dose from several measurements using ICRP reference parameter values;
(3) evaluation of the MDD for one monitoring programme taking uncertainty into account and,
(4) optimization by calculating the MDDs of different monitoring programmes, taking uncertainty into account.

The first module calculates the intake and committed effective dose from the bioassay function and the dose coefficient calculated by DCAL software [5]. It was validated by comparison with the published results [6] for an acute or chronic contamination by inhalation

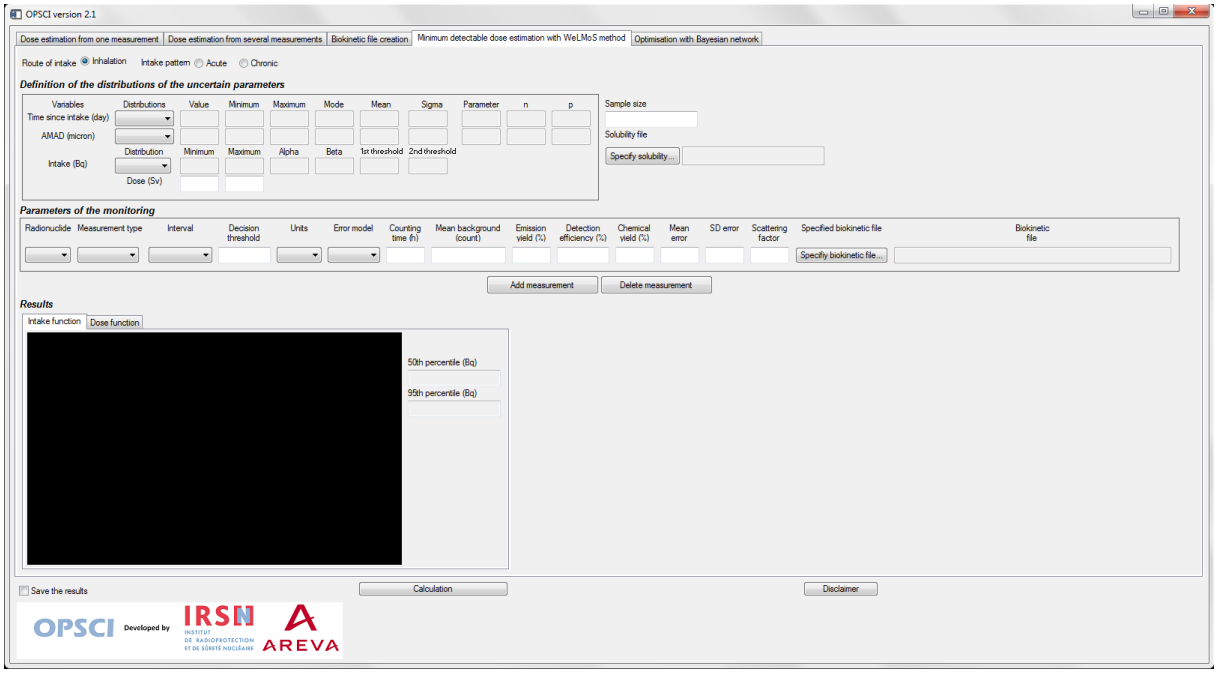

FIG. 1. Module 3 of OPSCI software

of an aerosol of ^{239}Pu with an AMAD of 5 µm with either absorption Type M or Type S. The differences between the doses published by Stradling et al and OPSCI are below 5%.

The second module calculates the committed effective dose following a contamination by several radionuclides and monitored by several measurements. The intake of each radionuclide is estimated as the geometric mean of the intakes determined from each measurement of this radionuclide. Intake is then multiplied by the corresponding dose coefficient. The total dose is the sum of the doses due to the different radionuclides. To validate this module, some of the measurement results from a real contamination case were used. The results obtained with OPSCI were compared by calculations done in Excel®: the two sets of data were identical.

The third module (Fig. 1) calculates the MDD of a monitoring programme taking account of uncertainties on measurement and conditions of exposure. The WeLMoS method [7], slighlty modified to improve convergence [8], is used. The Monte-Carlo sampling of the uncertain parameters is carried out by SUNSET (IRSN, Fontenay-aux-Roses, France). The calculation of the bioassay function and of the dose coefficient for each sampled sets is done by DCAL.

In the fourth module, the MDD is calculated using a Bayesian network[4], for all the monitoring programmes defined by OPSCI user. By comparing the MDDs and the measurement frequency and cost, the user can choose which monitoring programme is the best for the considered exposure conditions.

The third and fourth module of OPSCI were validated by comparison with the WeLMoS method [4, 8].

4. CONCLUSION

OPSCI software presented here is a new tool to calculate the committed effective dose from one or several bioassay using the reference models of ICRP implemented within DCAL and a deterministic approach. But its innovation lies in the evaluation of the MDD of a monitoring programme using the WeLMoS method and furthermore in its module for the optimisation of a monitoring programme where the user can evaluate the sensitivity of several

programmes and then choose the best one, compromising the cost and the sensitivity. All OPSCI modules are validated by comparison with published methods or through step-by-step validation.

OPSCI was already used to evaluate the sensitivity of monitoring programmes in a reprocessing plant [4] and in a MOX fuel fabrication facility [9]. OPSCI is currently extended to help in the definition of bioassay protocols following acute contamination by using the uncertainty on the dose estimate as a criteria to define the follow-up protocols.

ACKNOWLEDGEMENT

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REFERENCES

[1] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Radiation protection – Monitoring of workers occupationally exposed to a risk of internal contamination with radioactive material, ISO 20553, ISO, Geneva (2006).

[2] CARBAUGH, E.H., Minimum detectable dose as a measure of bioassay programme capability.
Radiat. Prot. Dosim. 105 1-4 (2003) 391.

[3] HEALTH PHYSICS SOCIETY, An American national standard: performance criteria for radiobioassay. Rep. HPS N13.30-1996, McLean, VA: HPS, (1996)

[4] DAVESNE, E., CASANOVA, P., CHOJNACKI, E., PAQUET, F., BLANCHARDON, E. Optimisation of internal contamination monitoring programme by integration of uncertainties, Radiat. Prot. Dosim. 144 (2011) 361.

[5] ECKERMAN, K.F., LEGGETT, R.W., CRISTY, M., NELSON, C.B., RYMAN, J.C., SJOREEN A.L., et al., User's Guide to the DCAL System. Rep. ORNL/TM-2001/190, Oak Ridge National Laboratory, USA, (2006)

[6] STRADLING, G.N., HODGSON, A., FELL, T.P., ANSOBORLO, E., BÉRARD, P., ETHERINGTON, G., et al., Plutonium compounds: exposure limits and assessment of intake and dose after inhalation. Rep. NRPB-W52, National Radiological Protection Board, Chilton, (2003).

[7] PUNCHER, M., BIRCHALL, A., A Monte Carlo method for calculating Bayesian uncertainties
in internal dosimetry. Radiat. Prot. Dosim., 132 (1) (2008)1

[8] DAVESNE, E., Optimization of monitoring programme of internal contamination by the study of uncertainty related to the dose assessment, PhD thesis, Université Paris XI, France, (2010)

[9] DAVESNE, E., QUESNE, B., DE VITA, A., CHOJNACKI, E., BLANCHARDON, E., FRANCK, D., Integration of workplace monitoring in optimization of medical follow-up, Poster Presentation in the « 6èmes journées sur l'optimisation en radioprotection dans les domaines nucléaires, industriel et médical » organized by SFRP in Saint-Malo, France, 11-12 June 2014. (2014) (in French)

DEVELOPMENT OF A NEW TOOL OF EXPERTISE FOR INTERNAL CONTAMINATION ASSESSMENT OF NUCLEAR MEDICINE WORKERS

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Abstract

To ensure the medical surveillance of workers exposed to a risk of internal contamination, IRSN, the French institute for radiological protection and nuclear safety, has developed two laboratories, unique in Europe for on-site monitoring campaigns and designed to meet the new radiation protection requirements in nuclear medicine departments and for radiological emergency response. A general description of these systems is presented in a first part, for different types of organs measurements (thyroid, lung and whole body). The feedback of campaigns is presented in a second part showing the help that could bring this kind of tool for the occupational physicist and the radiation protection officer to improve the risk management of internal exposure and also to better communicate about the risk to workers.

1. INTRODUCTION

The medical surveillance of workers exposed to a risk of internal contamination by radionuclides is mainly based on the interpretation of the results of their dosage in excreta (radiotoxicology) and\or the measurement of their retention by direct measurement (in vivo monitoring). The choice of the most relevant examination to be implemented depends particularly on the physical period of the radioelement, on the nature of its emissions, and on its biokinetics in the body.

As a matter of fact, the *in vivo* monitoring will be generally preferred for X and gamma emitters of relatively short period while the analysis of 24 hours urines or the 72 hours faeces will often represent the examination of choice for alpha and beta emitters. Nevertheless, because the in vivo measurements requires the patient to be physically present during the measure, the analysis of urines is often preferred by the professionals of nuclear medicine, although it is not always adapted to exposures by short live emitters such as Tc-99m.

To answer to these this challenge, the IRSN, the French institute for radiological protection and nuclear safety, has developed a new laboratory,: a whole body counting mobile facility, for on-site monitoring of personnel exposed to radiological contamination risks. Thanks to the unique features not found on others present in Europe (no reliance on liquid nitrogen, their improved communication facilities) it appears at the forefront of person monitoring in the event of a radiological incident.

This paper describes in the first part, the development of these 2 in vivo mobile laboratories and performances of these systems. In a second part some feedback of campaigns will be presented.

2. GENERAL PRESENTATION OF THE MOBILE LABORATORY

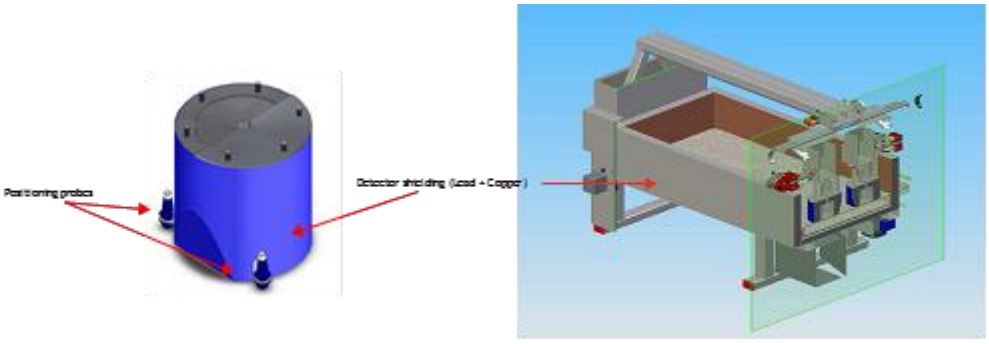
The unit consists of a 13 tons entirely self-contained truck (fig. 1), equipped with the most recent equipment for monitoring internal contamination and for processing results. Technical details are given in [1]. Isolated in a chamber constituted by a lead bathtub covered with copper, the person to be controlled for internal contamination is placed under two high resolution germanium detectors. For professional monitoring, measurement time of contamination thyroid is 15 minutes and 20 minutes for the whole body counting.

Fig 1: General presentation of the mobile laboratory

ABACOS-GPC, Canberra software of GENIE-PC®, provides the operating procedures for the analysis of spectra as well as performing calibration functions and quality assurance operations.

As the system is designed to be readily transported it is not possible to use the massive shielding often used for fixed whole body counters [2] [3] [4]. Moreover, in order to keep the truck easy to drive it was decided when designing the system to keep the weight to a maximum of 13 tonnes. As a matter of fact, the thickness of the shielding has been optimised by considering a shadow shield arrangement composed with 5 cm thick low background lead and copper placed where the shielding is most effective:

- Cylinders are placed around the detectors so as to collimate the field of view and reduce the background radiation when monitoring is done (Fig. 2a). To avoid any accident when positioning the detectors before the measurements, the shielding is equipped with probes preventing the system from being moved when its distance get closer than 5cm to the subject.
- A bathtub equipped with an electric door has been especially constructed to isolate, the person to be controlled (Fig. 2b).



a b

Fig. 2: Shielding of the detectors: (a) cylinders placed around the detectors; (b) bathtub equipped with an electric door

Besides, all the materials used for the construction of the truck have been checked for background radiation (even the truck structure) before the development of the mobile unit.

In order to determine the ideal position for the detectors in each counting geometries (whole body, thyroid or lung), the detectors are fixed in a mechanical arch moving up and down electrically. Furthermore, both detectors can be independently adjusted in six degrees in all directions

Figure 3 shows the positioning of the detectors for the 3 geometries: whole body, thyroid and lung:

- Whole body detectors are positioned so that the 2 detector centres are 25 cm above the subject's chest and thigh. This position can be used for all adults and for children over the age of four (i.e. taller than 100 cm). For smaller children this vertical position cannot be maintained and the detectors are placed as low as possible. The counting time is fixed at 20 min;
- Only one detector is used for the thyroid measurement. It is positioned over the lower neck and can be accurately placed for all adults and children. The counting time is fixed at 15 min;
- For lung monitoring the 2 detectors are placed over the subject's lungs and in uniform contact with the subject's upper chest. The counting time is fixed at 30 min.

a b c

Fig. 3: Different geometries of measurement: (a) whole body, (b) thyroid and (c) lung

3. FEEDBACK OF IN SITU MEASUREMENT CAMPAIGNS

Since 2009, the IRSN has put in place in situ measurement campaigns, in Nuclear Medicine Department in France for workers at risk of internal contamination such as the personal assistant of contaminated patients or handling radiopharmaceuticals for diagnostic purposes therapeutic or research. These campaigns are carried out using two of these mobile laboratories and specially focused on the detection of short half-life radionuclides such as Tc-99m, I-131 or F-18.

In 2013 during 8 campaigns, more than 500 workers were controlled with 186 in nuclear medicine services. The results put in evidence 36 contaminated workers: 17 % of internal contamination of workers including 6.5 % due to Tc-99m, 7.5 % due to I-131 and 3 % due to others such as F-18, Tl-201 and In-111 (fig. 4)

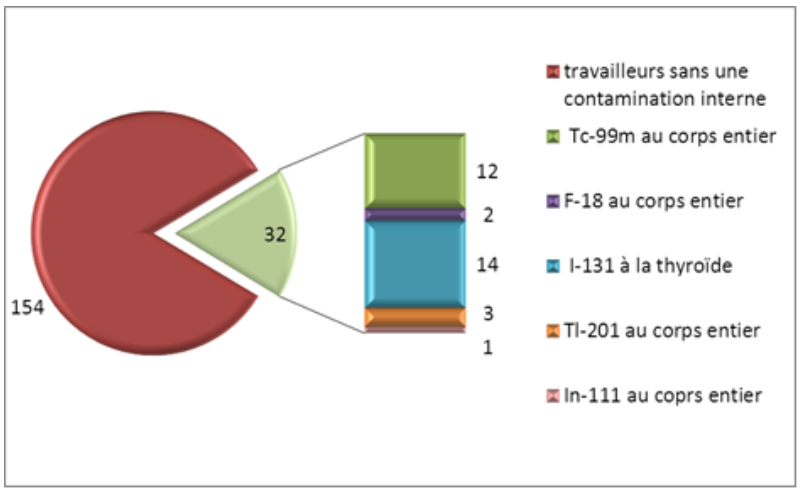


Fig 4: Type of contamination found for workers in nuclear medicine service in 2013

These campaigns show that *in vivo* measurements carried out close to the workplace are very adequate for detecting internal radioactive contamination for short half-life radionuclides. In all institutions, these results helped the occupational physicist and the radiation protection officer to improve the risk management of internal exposure and also to better communicate about the risk to workers.

REFERENCES

Franck, D., Bernière, J., Viltard, D., Parre, F., Challeton-de Vathaire, C., Agarande, M., 2012. Development of two mobile laboratories for Routine and Accident Monitoring of Internal Contamination. Appl. Radiat. Isot. 70 1095–1099

Lopez Pontez, M.A., Bravo, T. N.A, 2000. Low energy germanium detector system for lung counting at the WBC facility of CIEMAT. Radiat. Prot. Dosim. 89, 221–227.

Marsh, J., Shutt, A., Smith, J., Etherington, G., Youngman, M., 1996. Calibration and evaluation of a germanium detector whole body monitor. Chilton, NRPB-M672 Report. ISBN 0 85951 483 8

Schmitzer, C., Fischer, H., Andrasi, A., Bouvier, C., Carlan, L., Franck, D., Hollriegl, V., Li, W.B., Oeh, U., Ritt, J., Roth, P., Wahl, W., Zombori, P. , 2007. Improvements in routine internal monitoring - an overview of the idea project. Radiat. Prot. Dosim. 125, 472-476.

ESTIMATED RADIATION DOSE TO THE EYE LENS WITH PHOTOLUMINISCENCE DOSIMETERS
ANALYSIS OF THE UNCERTAINTY

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Abstract

This paper presents a method to estimate eye lens radiation doses received by workers at interventional radiology and cardiology environment, using photoluminescence dosimeters. The uncertainty of the most commonly reported in the literature eye lens doses per procedure is estimated. In addition, the uncertainty of the eye lens cumulative dose that could imply surpassing the new annual dose limit recommended by ICRP is estimated for a month, if approximately the same workload is maintained. Acceptable uncertainties were obtained for doses greater than 500 μSv in the usual energy range of interventional radiology.

1. INTRODUCTION

The new directive 2013/59, published on December 5, 2013, which incorporates the April 2011 ICRP statement, reduces the limit for the equivalent crystalline lens dose to 20 mSv per year for occupational exposure in planned exposure situations. Due to the high radiation field gradients in interventional procedures, the best way to accurately evaluate the equivalent dose received by the crystalline lens in exposed workers is by specifically monitoring the eye with point dosimeters placed near them. In recent years the use of optically stimulated luminescence dosimetry (OSL) as a rapid and inexpensive method to perform dose measurements in diagnostic radiology has been applied. The OSL also enables successive readings of absorbed doses [3], reducing uncertainty. However, the photo luminescent dosimeters (OSLD) are little equivalent to tissue, thus being very energy dependent in the energy ranges used in radiology. This complication can be partially avoided by calibrating the reader properly or correcting the readings for beam quality [4, 5]. For estimates of crystalline lens doses received in interventionism, care should be taken because the accuracy of measurements with these dosimeters can be compromised due to the particularities of the irradiation, even though the reading is corrected by a certain quality of beam or a specific calibration. In this paper the uncertainty of the crystalline lens dose equivalent $H_p(0.07)$ in interventional procedures for different dose ranges and beam qualities have been estimated.

2. MATERIALS AND METHODS

2.1. Equipment used

The dosimetry equipment used in this study consists of a set of photoluminescent glass dosimeters (dosimeters Nanodots screened, Landauer Inc.), a reader (MicroStar, Landauer Inc.) and an automatic erase device (Pocket Inlight Annealer, Landauer Inc.). The physical characteristics of the Nanodot dosimeters have been extensively described in the literature [3-4]. In all statistical analyses performed in this study, a p value less than 0.05 has been considered as statistically significant.

2.2 Validation of the dosimetry system and calibration

Prior to the analysis of the uncertainty in the measurement process, different tests have been performed in order to validate the dosimetry system used, by methods described in the literature [4], with similar results. The obtained signal loss by repeated readings or "depletion" was 0.5%. The stability of the reader system has been evaluated through incorporated equipment quality control, which analyzes the PMT response to different stimuli, considering the system is stable if none of the responses of the PMT exceed the corresponding calculated mean and variance control limits [6]. The angular dependence of the dosimeters in a homogeneous scattered radiation field has been assessed, obtaining a maximum variation of 17% for non-filtered beam. Since in interventional procedures, the greatest contribution to the dose is expected to come from the acquisition [5] with unfiltered beams and high rates, a calibration in terms of $H_p(0.07)$ and beam quality RQR-6 [7] has been used, with an uncertainty in the irradiation of 5%. The Nanodot dosimeters screened used show an individualized sensitivity, S, with a nominal uncertainty of 2%. According to the manufacturer, signal losses occur over time or "fading" approximately 2% every 6 months

2.3 Reading method and uncertainties estimation

The reading procedure that has been established is to read each dosimeter five times and take the average, C_m, of the last four readings, corrected by the factor of "depletion" as the best estimator of the cumulative dose in the dosimeter after irradiation. The error associated with the average of the readings is given by the quadratic sum of the type A and B uncertainties of C_m [8], assuming an equally likely distribution for the errors associated with stability (the ends of the distribution are the action limits of the control charts obtained for each of the stimuli), and the limited reader resolution (±1 count). Before irradiation the dosimeters are deleted for 20s using the Pocket Annealer eraser. It is considered that the best estimate of the residual counts after deletion, C_{20}, is the average of the maximum and minimum residual counts obtained from a dose range similar to the expected range in clinical practice, and its associated error is calculated assuming that they are equiprobable. Accumulated counts due to environmental background, C_f, are obtained with the slope of the linear fit of the average of the counts read by three dosimeters during 5 weeks, depending on the elapsed days (correlation $r^2 > 0.99$). The C_f error is calculated in each case as the maximum difference between the central value of the C_f ordered distribution. The C_f ordered distribution is generated from normal distributions obtained randomly, varying the adjusted slope and the number of days (where the standard deviation of each distribution is taken as the error of the slope and associated with a variation of ±1 days, respectively), minus the value corresponding to event 16000 and 84000 (1SD) of a total of 100000 events (Monte Carlo method [9]). The uncertainty of the calibration coefficient, N_Q, was obtained by uncertainties propagation from its relationship with the average counts of three calibration dosimeters irradiated at the same nominal dose, and the calibration Nanodot sensitivity used in each case. Finally, due to the

energy and angular dependence of OSL dosimeters [3-5], correction factors corresponding to the beam quality, k_Q, and the expected angle of irradiation, k_a, should be applied to the dose. Therefore, the uncertainty of the estimated eye lens personal dose equivalent is calculated by the uncertainty propagation in the expression (1)

$$H_p(0,07) = \left(C_m - C_{20} - C_f\right) \cdot N_Q \cdot S \cdot k_Q \cdot k_a \qquad (1)$$

Considering that the eye lens dosimeter is located on the external left side of the interventional radiologist's goggles, it is very complicated to ensure what the most probable angle of the dosimeter is, with respect to a field of scattered radiation whose geometry changes with time. Therefore, it was decided not to correct for angularity (i.e., k_a = 1), although it will be taken into account in the calculation of the uncertainty of the dose, considering a rectangular distribution given by the maximum and minimum response of the dosimeter depending on the relative angle. As a large variability in the scattered radiation field energy in interventional procedures is expected (changes in kV, added filtering, use of the ceiling suspended screen), it is not possible to define exactly the most probable beam quality and apply a correction factor k_Q. Therefore, similar to the angular dependence, it was decided not to correct for energy (i.e., k_Q = 1), although this dependence will also be taken into account in the calculation of uncertainty. For this purpose, the different correction factors obtained by the manufacturer [10] in a wide range of energy will be considered equiprobable, ranging from mean energies of 39 keV, which may correspond to low kV acquisitions used in pediatric patients, up to 118 keV, typical of heavily filtered beams transmitted by protection ceiling suspended screens [5]. Since the energy range considered is crucial in estimating the uncertainty, its behavior is also studied for the correction factors in a small range of mean energies containing the beam quality RQR-6, and ranging from 39 keV to 48 keV, which is equivalent to assume that most of the radiation is due to the acquisition. Likewise, the uncertainty for an energy range containing the RQR-6 and RQR-9 qualities has been estimated, considering the latter corresponds to low-dose fluoroscopy [5]. All results have coverage K= 2.

3. RESULTS

Table 1 shows the absolute and relative uncertainties of the dose equivalent for a wide range of doses estimated by procedure, which includes a minimum dose of 60 μSv and a maximum dose of 4.5 mSv, and three different ranges of beam qualities. It was considered that the dosimeter reading is performed one day after irradiation.

TABLE 1. ABSOLUTE AND RELATIVE UNCERTAINTIES OF Hp (0.07) FOR A WIDE RANGE OF DOSES ESTIMATED BY PROCEDURE AND BEAM QUALITY (K= 2)

	$\Delta H_p(0,07)$ (mSv)			Δ_r(%)		
$H_p(0,07)$ (mSv)	39-118 (keV)	39-65 (keV)	39-48 (keV)	39-118 (keV)	39-65 (keV)	39-48 (keV)
0,06	0,05	0,04	0,04	88,37	61,47	60,95
0,19	0,13	0,04	0,04	67,51	22,93	21,51
0,48	0,31	0,07	0,06	65,40	15,66	13,49
0,75	0,49	0,11	0,09	65,15	14,59	12,25
1,11	0,73	0,16	0,13	65,09	14,32	11,91
1,82	1,18	0,26	0,21	65,05	14,13	11,70
2,52	1,64	0,35	0,29	65,03	14,04	11,58
4,52	2,94	0,63	0,52	65,03	14,02	11,56

Reading is done one day after irradiation. The range of mean energies 39-118 keV takes into account the use of leaded screens and low kV acquisitions (thin or pediatric patients). The 39-65 keV energy range includes RQR-6 and RQR-9 qualities, corresponding to the acquisition and the low and high dose fluoroscopy. The most restricted energy range 39-48 keV includes beam quality RQR-6, corresponding to the acquisition.

The cumulative dose uncertainty in a month of 1.8 mSv has been also estimated, for the corresponding RQR-6 and RQR-9 energy range, which would imply the exceeding of the annual eye lens dose limit recommended by ICRP in the case of maintaining approximately the same workload for the rest of the year. In order to assess the influence of the counts due to environmental background radiation, C_f, the uncertainty for the same cumulative dose has been also calculated, as if it had been received in a single procedure and had been read the next day. The results are presented in Table 2.

TABLE 2. DOSE UNCERTAINTY (K= 2) ACCUMULATED DURING A MONTH THAT COULD EXCEED THE ANNUAL DOSE OF 20 MSV IN THE CASE OF MAINTAINING APPROXIMATELY THE SAME WORKLOAD. COMPARED WITH THE UNCERTAINTY THAT WOULD RESULT IF THE DOSIMETER IS IRRADIATED IN A SINGLE PROCEDURE, AND READ THE NEXT DAY

$H_p(0.07)$ (mSv)	$\Delta H_p(0,07)$ (mSv)	Δ_r(%)	days elapsed
1,82	0,26	14,13	1
1,79	0,25	14,14	30

4. DISCUSSION

For a dose of 60 μSv per procedure, a minimum uncertainty of 61% for a range of restricted beam qualities for the acquisition is observed. For estimated doses about 500 μSv and higher, relative uncertainties less than 16% are obtained when qualities typical of the acquisition and the high and low fluoroscopy are assumed. The reading of dose per procedure lower than 190 μSv presents an uncertainty greater than 20%. In all the analyzed energy ranges, an increase in relative uncertainty regarding lower cumulative dose is observed, more pronounced at low doses, due to fact that the influence of the uncertainty type B of C_m (which increases slightly with the dose because C_m depends on depletion) is more important. The use of ceiling suspended screens can increase the measurement uncertainty up to 88%, depending on the dose range considered. If the dosimeter reading is performed a month after the irradiation, the influence of accumulated background radiation in the estimation of the uncertainty is negligible. Longer times of irradiation are not advisable, because in such cases the effects of fading may be significant.

5. CONCLUSION

The OSLD allow estimating $H_p(0.07)$ in eye lens with acceptable uncertainty for doses greater than 500 μSv in the regular energy range of interventional radiology. Since the contribution to the uncertainty of the environmental background is negligible, the continued irradiation of the dosimeter corresponding to a month's work proceedings in one session is recommended. The influence of highly filtered radiation fields produced by ceiling suspended screens and other elements of radiation protection significantly increase the measurement uncertainty.

REFERENCES

[1] COUNCIL DIRECTIVE 2013/59/EURATOM of 5 December 2013.
[2] VANO, E., et al. Lens injuries induced by occupational exposure in non-optimized interventional radiology laboratories, The British Journal of Radiology 71 (1988) 728-733.
[3] E. G. YUKIHARA et al. Optically stimulated luminescence (OSL) dosimetry in medicine. Physics in Medicine and Biology, Vol. 53 (2008) R351-R359.
[4] RANI M. AL-SENAN et al. Characteristics of an OSLD in the diagnostic energy range. Med. Phys. 38 (7), (2011).
[5] R. M. SANCHEZ et al. Measurements of eye lens doses in interventional cardiology using OSL and electronic dosemeters, Radiation Protection Dosimetry (2014) 1–8.
[6] JOHN S. OAKLAND. Statistical Process Control, fifth edition. Butterworth Heinemann. 2003.
[7] INTERNATIONAL ELECTROTECHNICAL COMMISION (IEC). Medical diagnostic X-ray equipment—radiation conditions for use in the determination of characteristics. 61267 Ed. 2.0. IEC (2005).
[8] INTERNATIONAL STANDARDIZATION ORGANIZATION (Geneve, Switzerland), guide to the expression of uncertainty in measurement, 1st Ed. 1995.
[9] W. JANKE, Statistical Analysis of Simulations: Data Correlations and Error Estimation. Quantum Simulations of Complex Many-Body Systems: From Theory to Algorithms, Lecture Notes, J. Grotendorst, D. Marx, A. Muramatsu (Eds.), John von Neumann Institute for Computing, ISBN 3-00-009057-6 (2002)
[10] C. J. YAHNKE et. al.,Microstar calibration conversión factors for Dot's. Landauer, Inc. 2 Science Road, Glenwood, IL 60425, (2009). http://www.landauer.com/uploadedFiles/Resource_Center

THE POSSIBILITY OF DETERMINING THE DOSE IN THE LENS OF THE EYE FOR RADIATION WORKERS

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Abstract

In 2011, the International Commission on Radiological Protection (ICRP) issued a declaration based on recent studies of the effects of ionizing radiation on the lens of the eye and reconsidered the attitude to the threshold for the occurrence of radiation cataracts and subsequently recommended to reduce the limit for the lens of the eye exposure for radiation workers for 20 mSv per year within 5 years that should not exceed 50 mSv in any year. This new limit is already included in the newly proposed European Directive which determines basic safety standards for the protection against the dangers resulting from the exposure to ionizing radiation. The State Office for Nuclear Safety (SONS) assumed this development, and in 2012 entered two small contracts which should have mapped possibilities for routine monitoring of doses to the lens of the eye. Each of the studies focused on the different part of this issue so all the related aspects have been covered. Now SONS have an exact idea of the circumstances under which it will be necessary to determine the exposure of the lens of the eye from a separate dosimeter placed in the eye, the position of the reference point, the possibilities of dosimetry services to perform this standard measurement and the method of monitoring the exposure of the lens of the eye using protective goggles.

1. INTRODUCTION

Recently, there have appeared a number of studies showing that the lens of the eye are far more sensitive to the radiation than it was previously considered [1]. The originally estimated dose threshold of 2Gy seems to be significantly overvalued. Apparently doses as 0,25Gy may be cataractogenic and may not be excluded no-threshold effects of ionizing radiation on the lens of the eye [2, 3]. It also shows an increased incidence of cataracts among physicians performing interventions under fluoroscopic control [4, 5]. ICRP recommendation stated in paragraph 103 (249) states that: the dose limits for the lens of the eye remain unchanged, but new data on radio sensitivity of the eye are expected.

2. DESCRIPTION OF THE STUDY

2.1. Objectives

Determine, whether in practice this limit is exceeded, try to answer the question of whether a dose of the eye in interventional radiologists and cardiologists should be routinely monitored; Verify the accuracy of the determination of the equivalent dose to the lens of the eye energy X-ray; 25 keV to 85 keV (range energy X-rays used in interventional cardiology and radiology – the riskiest category of workers at risk cataract of the lens of the eye.

Identify the differences between equivalent doses of the lens of the eye for each eye and the point midway between two eyes and the angle of incidence of X rays 0°, 15°, 30°, 45°, 60° and 75°. While comparing the measured values in the middle of the forehead and eyes to the value of the dosimeter placed on the side as in the case of the dosimeter RadCard

s.c. EYE-D^{TM}. Determine the degree of attenuation, which the protective goggles provide, depending on the energy X-ray spectrum.

2.2. Procedure

On the basis of the reference measurements was examined to what extent are the values determined from the "whole-body" OSL dosimeters[6] consistent with directly measured values of commercially available TLD EYE-D^{TM} [7].

The survey covered seven departments of five medical facilities. Sample of 22 physicians (intervention radiologists and cardiologists) who are exposed to a relatively high radiation exposure, so as to ensure that, when performed procedures will result in dose which is sufficient to compare. These workers have used, along with their own personal OSL dosimeters also eye-dosimeters EYE-D^{TM} for four to five monitoring period. At the end of each month dosimeters were evaluated [8-11] and compared.

3. CONCLUSIONS

3.1. Compared to Hp(3)

The results show that the values of Hp(3) measured by the eye-dosimeter EYE-D^{TM} can be up to 10 times lower than the value of Hp(3) calculated from the Hp(10). The ratio of these two values between different doctors and monitoring periods between individual physicians varies significantly. It ranges from 0.92 to 11.00. Due to the large variance ratio values, it is not possible to clearly determine its actual size.

Given to the fact that during the examination, changes occur in positions between the radiation source and the whole body dosimeter, the radiation source and the eye dosimeter and the two dosimeters with each other, it is logical that the value of the ratio Hp(3) measured and calculated from Hp(10) will change during the examination.

It is evident that the eye dosimeter can better evaluate the real radiation dose to the lens of the eye, since it is located close to the eye, and is able, unlike the whole body dosimeter, to reflect changes in positions of the eyes and the radiation source.

3.2. Receipt doses of dosimeter EYE-DTM

The measured results show that the doctor working on angiographic line gets an average dose of 1.6 mSv per month, which annually makes approximately 19 mSv. This value is certainly well below the current annual limit for the lens of the eye 150 mSv, but almost reaches the intended new annual limit of 20 mSv. Furthermore, it can be said that at least 9 out of 22 monitored doctors would be probably exceeded the new limit.

The maximum monthly measured dose was 5.4 mSv. At the same practice, the physician would annually receive up to 65 mSv. This is a value that exceeds more than 3 times the new thinking limit and reaches about 43% of the current limit. The average month dose of cardiologists is 1.4 mSv. Radiation doses of intervention radiologists are on average 36% higher, namely around 1.9 mSv.

3.3. Attenuation coefficient of protective glasses

From the experiments it is seen that the front glass of protective glasses with shielding equivalent of 0.75 mm Pb is able to shield up to 94% of the scattered radiation. The side glass to shield the equivalent of 0.5 mm Pb can reduce the dose by up to 89% then. In both cases, the shielding ability of glasses depends on the energy of the scattered radiation.

From this, it follows that if the angiographic workplace is equipped with protective goggles and if the surgeons use them each time of surgery, radiation load to the lens of the eye will drop to approximately one-tenth.

Average annual Hp(3) would be moved about 1.9 mSv, which would be well below the recommended limit of 20 mSv. The maximum annual value of Hp(3) would amount to about 6.5mSv by using of the protective glasses. And these values would meet the new limit for the eye lens. Their consistent use would ensure that none of the doctors in normal practice would exceed the new limit to the lens of the eye.

REFERENCES

[1] BROWN N.P., The lens is more sensitive to radiation than we had believed, British Journal of Ophthalmology, 81 (1997) 257–259.
[2] CHUMAK V.V., WORGUL B.V. et al., Dosimetry for a study of low-dose radiation cataracts Among Chernobyl clean-up workers., Radiat. Res. 167(5) (2007) 606-14.
[3] NAKASHIMA E, NERIISHI K, MINAMOTO A. A reanalysis of atomic-bomb cataract data, 2000-2002: a threshold analysis. Health Phys., 90 (2006) 154-60.
[4] VANO E., GONZALEZ L., BENEYTEZ F., MORENO F., Lens injuries induced by occupational exposure in non-optimized interventional radiology laboratories, BR. J. Radiobiol., 71(847) (1998) 728-33.
[5] JUNK A.K., HASKAL Z., WORGUL B.V., Cataract in Interventional Radiology – an Occupational Hazard?, In-vest Ophthalmol Vis. Sci. 45 (2004) 388.
[6] INLIGHT TECHNICAL SPECIFICATIONS, Prepared by Stanford Dosimetry, LLC, July 14, (2006).
[7] MANUFACTURER'S DATASHEET RADCARD, EYE-DTM
[8] P. BILSKI at. al. (ORAMED), A new dosemeter for measurements of Hp(3) for medical staff (http://www.oramed-fp7.eu).
[9] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Individual thermoluminescence dosemeters for extremities and eyes, ISO 12794, ISO, Geneva, (2000).
[10] Daures J., Gouriou J., Bordy J. M., Conversion coefficients from air kerma to personal dose equivalent Hp(3) for eye-lens dosimetry, ORAMED project, Rapport CEA, CEA, LIST, Laboratoire National Henri Becquerel (LNE LNHB), F-91191 Gif-sur-Yvette, France.
[11] MARIOTTI, FANTUZZI, GUALDRINI, BOTTA, ULERI, BORDY, DONEZIERE, ENEA Extremity dosemeter based on LiF(Mg, Cu, P) to evaluate Hp(3,α), Radiat. prot. Dosim., Vol. 144 (1-4) (2011) 187-191.

GAPS AND CHALLENGES IN NEUTRON PERSONAL DOSIMETRY: INTERCOMPARISONS AND APPLICABLE CRITERIA FOR DOSIMETRIC PERFORMANCE

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Abstract

Dosimetry for neutron radiation, which is inevitably accompanied by a photon component, still presents challenges despite many years of development of neutron personal dosemeters. Neutron dosimetry is still a very challenging task in routine because of the variability of the characteristics of the workplace radiation fields and, last but not least, the actual routine measured $H_p(10)$ values lie below 300 microSv, where accuracy is even more difficult to be achieved. Regular intercomparisons, either in standard laboratory conditions and "in-field", provide an essential tool to guide development in neutron dosimetry. Such intercomparisons are usually not achievable in only one country and therefore an international effort in designing and planning such testing sessions is needed. The

EURADOS Intercomparison for whole-body neutron dosimeters (EURADOS IC2012n) was an important action in the field and it showed that performance criteria for neutron dosimetry should be agreed internationally.

1. INTRODUCTION

Neutron sources are intentionally used and/or incidentally created in various scientific areas and technical applications. Routine neutron dosimetry is still very challenging task as neutrons are: present in mixed-fields, are indirectly ionizing particles and pose more problems for their detection than other types of radiation, their energies may cover extremely large ranges from 9 (nuclear industry) to 12 (particle accelerators, flight altitudes) orders of magnitude, and their "quality" and hence their conversion coefficients from fluence to dose vary by a factor of 50 over the entire energy range. Last but not least, the actual routinely measured $H_p(10)$ values usually lie below 300 microSv, where accuracy is even more difficult to achieve. Workplace monitoring is often a prerequisite to achieve sufficient accuracy, i.e. by evaluating a spectrum correction factor to be applied. However, workplace monitoring is performed mainly with multi-sphere spectrometers or simply by area monitors, which do not provide information on the direction distribution of neutrons and therefore results are not sufficient to determine personal dose equivalent and to provide workplace field characterisation in terms of $H_p(d)$: the simultaneous measurement of energy and direction distribution is still a matter of research.

Calibration of neutron personal dosemeters requires specific attention. In standard laboratories it is not possible to reproduce the variety of conditions (mixed-fields and wide energy and angle of incidence ranges) in which dosemeters are then used in workplaces. Various types of neutron dosemeters are used routinely and they have to be tested for their performance.

Therefore, there are various factors that make it both difficult and expensive to conduct a neutron personal dosemeter intercomparison. These challenges need to be addressed to avoid skewing the intercomparison in favour of one type of dosemeter, whilst ensuring that it provides an adequate test and does not become prohibitively expensive.

The EURADOS Intercomparison for whole-body neutron dosimeters (EURADOS IC2012n) [1] was an important action in the field of regular performance tests in neutron dosimetry, for which intercomparisons at international level have been performed only every 8-10 years.

2. GAPS AND CHALLENGES FOR INTERCOMPARISON IN NEUTRON PERSONAL DOSIMETRY

Reference neutron fields are detailed in ISO 8529: Reference neutron radiations, this is divided into three parts [2, 3, 4], and in ISO 12789: Reference radiation fields - Simulated workplace neutron fields" [5, 6]. These are a mixture of radionuclide source and accelerator generated fields. Ideally, the intercomparison would have been restricted to ISO recommended radiation fields from these standards, but field availability and dose rate had to be considered.

Generation of fields using accelerators is more difficult for neutrons than photons, because the accelerators are larger, and more expensive, but also because the fluence/dose rates that can be generated are limited due to technical constraints. The problem for neutrons, however, is exacerbated when simulated workplace fields are generated, because the down-scattering in energy lowers the dose rate and there is inevitable capture that further lowers the fluence rate.

When high-energy accelerator facilities are excluded, terrestrial workplaces have neutrons that range in energy from 10^{-8} eV to 100 MeV; over 10 orders of magnitude.

Workers are rarely exposed directly to bare sources; instead the neutrons have lost energy via several or many scatters, so they have a very broad range of energies. Typically the energy distribution features a thermalized peak (E_n < 0.4 eV), a smaller intermediate energy component (0.4 eV < E_n < 10 keV) and a residual fast peak (E_n > 10 keV). Ideally an intercomparison would test dosemeters across this range of energies, though the intermediate energy range is less dosimetrically important.

The fluence to personal dose equivalent conversion coefficients vary strongly with neutron energy and they also fall, in general, with increasing angle of incidence so irradiations performed at higher angles will need to be longer to ensure that the dose is high enough.
The choice of the radiation fields is problematic not only because of the contrasting characteristics of neutron workplace and reference fields but also the deficiencies of different detector types.

The main types of neutron personal dosemeter in use in the workplace are etched-track and TLD-albedo dosimeters. In some cases electronic dosimeters are in use. These different dosemeters have very different deficiencies in their responses, which will make different fields tougher for them in the intercomparison. For example, the inclusion of angles of incidence other than normal to the reference direction of the dosemeter causes different problems for different types of dosemeters. The best designs of albedo dosemeter should have good angle dependence of response for forward angles, though 90° can be problematic. Above the fast neutron threshold the angle dependence of response is generally not so good for track detectors and electronic devices.

Consequently, it is necessary to balance the rigour and fairness of the test against the cost.

3. APPLICABLE PERFORMANCE CRITERIA TO INTERCOMPARISON RESULTS

To perform a fair and accurate analysis of the results of an intercomparison it is more appropriate to conduct it on the basis of procedures and criteria agreed by the scientific community. However, in practice there is not an internationally agreed document answering precisely the question “which procedure and criterion should be used for overall dosimetric performances and comparison between different kind of personal dosemeters?”. Therefore, there is no “internationally agreed” criterion for the performance of neutron dosemeters in individual monitoring.

ICRP Publication 75 [7], when dealing with accuracy recommendations for individual dose assessment, states at §251 that in workplace fields, where the energy spectrum and orientation of the radiation fields are usually not known, “the overall uncertainty at the 95% confidence level in the estimation of effective dose around the relevant dose limit may well be a factor of 1.5 in either directions for photons and may be substantially greater for neutrons of uncertain energy and for electrons. Greater uncertainties are also inevitable at low levels of effective dose for all qualities of radiation”. The use of the factor 1.5, mentioned in ISO 14146 [8] for photon dosimetry performance requirement, is followed for the EURADOS photon intercomparisons. However, this would probably be too restrictive for neutron dosimetry.

3.1. Overview of the international standards and guidelines related to personal neutron dosimetry

At an international level, the standards which are relevant for personal dosimetry are of two kinds. There are standards related to the realization and the use of reference radiation fields and standards giving the requirements and recommendations for testing the performances of personal dosemeters.

For neutrons, besides *ISO 8529* and *ISO 12789* series very few standards are available.The standard ISO 21909 [9] is the document establishing type tests and the requirements for passive neutron personal dosemeters. This standard has been under revision since 2011 with the objective of rectifying the weaknesses of the present document. Indeed, this present version defines tests and criteria which differ for the different techniques (nuclear tracks emulsions dosemeters, solid state nuclear track dosemeters, thermoluminescence albedo dosemeters, superheated emulsion dosemeters, ion chamber dosemeters with direct ion storage). The new version defines criteria undependent from dosemeter type and may have less constraining criteria at low doses to assure the quality of the dosimetry without being unachievable.

IEC standard 61526 [10] is the international document establishing the type tests and requirements for all active personal dosemeters for gamma, neutron and beta radiations. Other standards exist at a national level.

3.2. Criteria for the performance of personal neutron dosimeters at intercomparisons

The basic principle of a dosimetry intercomparison is to expose dosemeters to accurately known doses in reference fields, i.e. reference neutron sources, monoenergetic neutron fields or a thermal field and simulated workplace fields, and to compare the reported $H_p(10)$ values, $H_{\text{participant}}$, with the reference values, H_{ref}, given by the Irradiation Laboratories, by evaluating the responses, R:

$$R = \frac{H_{participant}}{H_{ref}}$$

To evaluate the intrinsic quality of the response of a dosimetric system and to quantify the difference between systems, criteria are needed to appreciate what can be considered in terms of an acceptable under-response or an acceptable over-response.

IEC 61526 and ISO 21909 provide such criteria applied to the response, but only in specific conditions: none of them provide a detailed guidance applicable for all types of dosemeters at an intercomparison test with various kind of radiation fields and dosimeters as in EURADOS IC2012n.

IEC 61526, covering active devices, gives different criteria for a combined energy and angle dependence of response for three neutron energy ranges (below 100 keV, between 100 keV and 10 MeV and above 10 MeV) and angles of incidence from 0° to 60°.

The ISO 21909 for passive dosimeters provides a series of test and performance requirements for specific issues (e.g. linearity, detection threshold, energy and angle dependence of response, etc.) for various types of dosimeters. For example, for the energy dependence of response, it does not provide any criterium for albedo dosimeters and it sets a limit to 50% for under- or over-response for track detectors, to be measured at a dose level of at least 1 mSv and for 4 chosen energy fields.

Considering this lack of international consensus for criteria adapted to neutron intercomparisons, different criteria were used at previous international intercomparisons: the EURADOS 1999 Performance test [11] and the IAEA Intercomparison on measurements of the quantity personal dose equivalent $H_p(d)$ in mixed (neutron-gamma) fields (2003-2004) [12]:

- EURADOS 1999

$$\frac{1}{1.5}\left(1 - \frac{2H_0}{H_0 + H_{ref}}\right) \leq \frac{H_{participant}}{H_{ref}} \leq 1.5 \left(1 + \frac{H_0}{2H_0 + H_{ref}}\right) \qquad with\ H_0 = 0.085\ mS$$

- IAEA 2003-2004

$$\frac{1}{2}\left(1-\frac{2H_0}{H_0+H_{ref}}\right)\leq\frac{H_{participant}}{H_{ref}}\leq 2 \qquad with\ H_0 = 0.1\ mSv$$

Where, H_0 is the detection limit of the system.

Considering the variety of approaches and criteria and the results of previous intercomparison, the Organization Group decided to use a factor of 2 as a general criterion for the response, *R*, for all dose values. Therefore this criterion for an "acceptability good" response eventually used for the 2012 EURADOS neutron intercomparison was:

$$\frac{1}{2}\leq\frac{H_{participant}}{H_{ref}}\leq 2$$

It should be clear from the above discussion that this criterion has to be considered only as a guideline to the performance of the personal dosimetry system.

4. CONCLUSIONS

The availability of reference radiation fields appropriate to test the performance of personal neutron dosimeters and the different characteristics of the actual personal dosimeters themselves make the design and realization of a large scale intercomparison at an international level a difficult task to be achieved for either technical and cost issues. Moreover, there is no "internationally agreed" criterion for the performance of neutron dosemeters in individual monitoring. The standard ISO14146 is not applicable to neutrons and no other international standard provides guidance on: how to perform an intercomparison among neutron personal dosimetry systems nor on the criteria to be applied to the results.

It is recommended that international organizations include such a topic in their new project item lists in order to achieve at least a common and scientific agreed procedure to perform an intercomparison and to evaluate its results.

REFERENCES

[1] FANTUZZI E., M-A CHEVALLIER M-A. , CRUZ-SUAREZ R. , LUSZIK-BHADRA M., MAYER S., THOMAS D. J., R. TANNER R., F. VANHAVERE F., EURADOS IC2012n Intercomparison for whole-body neutron dosemeters. Radiat. Prot. Dosim. Published online Feb2014 (2014), doi:10.1093/rpd/nct295.

[2] ISO 8529-1:2001, Reference neutron radiations - Part 1: Characteristics and methods of production (2001).

[3] ISO 8529-2:2000, Reference neutron radiations - Part 2: Calibration fundamentals of radiation protection devices related to the basic quantities characterizing the radiation field (2000).

[4] ISO 8529-3:1998, Reference neutron radiations - Part 3: Calibration of area and personal dosemeters and determination of response as a function of energy and angle of incidence (1998).

[5] ISO 12789-1:2008, Reference radiation fields, Simulated workplace neutron fields - Part 1: Characteristics and methods of production (2008).

[6] ISO 12789-2:2008, Reference radiation fields, Simulated workplace neutron fields - Part 2: Calibration fundamentals related to the basic quantities (2008).

[7] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, General principles for the radiation protection of workers. ICRP publication 75. Ann. ICRP 27 (1) (1997).

[8] INTERNATIONAL ORGANIZATION FOR STANDARDISATION, Radiation protection-Criteria and performance limits for the periodic evaluation of processors of personal dosemeters

for X and gamma radiation, ISO 14146 (2000).

[9] ISO 21909, Radiation protection - Passive personal neutron dosemeters - Performance and test requirements; 2005-06 and Corrigenda 1 as of 2007-10

[10] IEC 61526 Ed.3, Radiation protection instrumentation – Measurement of personal dose equivalents Hp(10) and Hp(0,07) for X, gamma, neutron and beta radiations – Direct reading personal dose equivalent meters and monitors; 2010-07

[11] BORDY, J. M., STADTMANN, H., AMBROSI, P., BARTLETT, D. T., CHRISTENSEN, P., COLGAN, T. AND HYVÖNEN, H., Performance Test of dosimetric services in the EU Member States and Switzerland for the routine assessment of individual doses (Photon, Beta and Neutron). Radiat. Prot. Dosim. 89(1−2) (2000) 107−154.

[12] CRUZ SUÁREZ, R., SCHUHMACHER, H., POCHAT, J. L., ZIMBAL, A. AND MRABIT, K., Intercomparison on Measurements of the Quantity Personal Dose Equivalent $H_p(d)$ in Miexd (Neutron-Gamma) Fields—An IAEA Project, Radiat. Prot. Dosim. 125(1−4) (2007) 61−68.

MONTE CARLO INVESTIGATION INTO SCATTER RADIATION FROM CT FLUOROSCOPY GANTRY: EFFECT ON STAFF DOSE *A CONTRIBUTION TO OCCUPATIONAL RADIATION PROTECTION IN MEDICINE*

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Abstract

Investigation into scatter radiation from CT fluroscopy gantry using Monte Carlo N-Particle eXtended (MCNPX) has been done. Using already proven method of modelling point source, bowtie-filter and collimator, the CT compartment of the PET-CT equipment at Karlsruhe Klinikum was modelled. Additions to the model were CT gantry block, patient table, CT room, standing and supine Male Adult meSH (MASH) phantom. Varying the gantry block material with air, zinc, iron, tungsten, aluminium, magnesium, barium, tin, gadolinium oxysulfide, caesium iodide and xenon, the scatter radiation at the isocenter of the CT and to the medical staff was investigated. Scatter radiation from the gantry block had an influence on measurements with inonisation chamber at the isocenter. Radiation exposure to the eye lens and thyroid of medical staff was independent of the presence of a gantry block or material. The whole body exposure of medical staff was however dependent on the presence of a gantry block but independent of gantry block material of zinc or tin.

1. INTRODUCTION

Introduction of Computed Tomography Fluoroscopy (CTF) has tremendously improved medical interventional procedures [1]. This is because it has a wide dynamic range to image air, soft tissues and bones, and further provides an acceptable image quality which is less affected by patient breathing and motion [2]. The benefits and safety associated with CTF to the patient comes with detrimental scatter radiation to the medical staff, since the presence of the medical staff in the room is needed for an effective interventional procedure [3, 4]. Compton scattering is very prominent in the radiation energy range of CTF, therefore exposing the medical staff to scatter radiation from the patient, CT gantry and patient table [2].

Reports from other studies on CTF have suggested high radiation exposure to the body of the medical staff during a procedure [5-7]. These doses are high compared with conventional fluoroscopy due to high current applications in CTF [2]. Since the medical staff is expected to protect themselves with lead aprons, goggles, thyroid shield and gloves where applicable and necessary, the scatter radiation to the sensitive organs such as the eye lens and the thyroid is expected to be minimal.

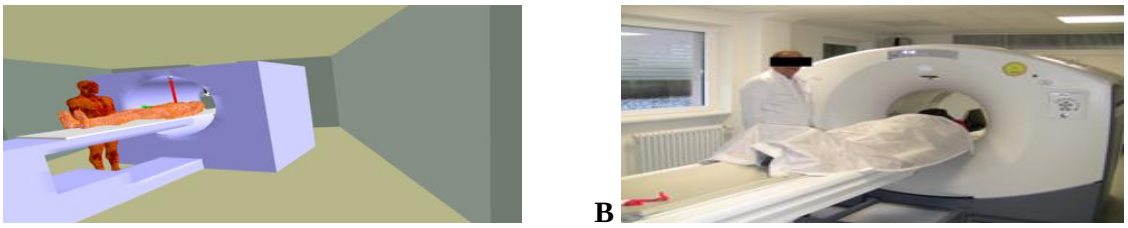
This paper presents Monte Carlo investigation into the radiation energy to the medical staff's eye lens, thyroid and whole body when protective gadgets are not worn. Emphasis is laid on the contribution of scatter radiation from the gantry.

2. METHODOLOGY

Using an already proven and tested method of modeling a CT point source (36 point source at 10^{o} intervals), bowtie-filter, pencil ionization chamber and collimator as reported by Figueira *et al* [8] and Gu *et al* [9], the CT compartment of a General Electric Discovery PET-CT 710, at Karlsruhe Klinikum, Germany was modeled. In addition to existing models as given in literature [8, 9], the CT gantry block, patient table, CT room, and standing and supine (MASH) phantom [10] were modeled as shown in Figure 1 (A). Figure 1(B) shows the position of patient and staff during a CTF procedure. All the models except the phantoms were achieved through the use of SimpleGeo [11] (software for drawing solid objects). The obtained models were put together on one platform as they are spatially located by using a Karlsruhe Institute of Technology (KIT) in-house built software called VOXEL2MCNP [12] and transferred to MCNPX input-files.

The photon energy spectrum of a 120 kVp, 12 degree angle of tungsten anode with 2.5 mm thickness of aluminium filter was generated using SpekCalc [13]. The mean energy of the spectrum was 54.45 keV. MCNPX [14] radiation transport code was used for the study. One billion number of particles were tracked in order to have a good compromise between relative error and reasonable computational time.

In order to demonstrate the contributing effect of scatter radiation from the gantry block to measurements made at the isocenter and to medical staff during Monte Carlo (MC) simulations, probable cheap, strong and readily available metals (zinc, iron, tungsten, aluminium, magnesium, barium and tin) that could be part of the gantry and detector materials (gadolinium oxysulfide, caesium iodide and xenon) were assumed to be component part of the gantry block. Previous studies have been silent on the gantry block and this may be due to the unavailability of material composition in literature for MC studies or irrelevant for their study. MC studies were done for varying the CT gantry material to determine the energy measured with a pencil ionisation chamber at the isocenter and to the medical staff (radiation to eye lens, thyroid and whole body) for 360^{o} rotation of point source.



A **B**

FIG. 1. Pictures of medical staff performing a CTF procedure (A) 3D model generated with VOXEL2MCNP and (B) photo at the Klinikum Karlsruhe.

3. RESULTS AND DISCUSSION

Table 1shows the simulated energy deposition (F6 standard tally of MCNPX) per rotation, in a pencil ionization chamber free-in-air at the isocenter of the CT device with respect to different material composition of the CT gantry block. Air was used as the reference material to define “no scatter radiation” from the gantry to the isocenter. Therefore, the difference of deposited energy in a material and that of air in percentage (percentage scatter radiation), and the ratio between the deposited energy in a material to air (scatter factor) at the isocenter for air is 0.0 and 1.00 respectively. The gantry material with the lowest and highest scatter radiation percentage at the isocenter is zinc and tin, respectively. Gantry materials of aluminium, magnesium, gadolinium oxysulfide, barium, caesium iodide, xenon and tin contributed scatter radiation percentages above 80 at the isocenter.

TABLE 1. ENERGY DEPOSITED IN IONIZATION CHAMBER AT ISOCENTER FREE-IN-AIR OF CT FLUOROSCOPY GANTRY FOR ONE ROTATION VARYING GANTRY MATERIAL (MCNPX F6 TALLY)

Gantry Material	Deposited Energy (MeV/g/particle)	Percentage Scatter Radiation (%)	Scatter Factor
Air	2.38×10^{-9}	0	1.00
Zinc	2.58×10^{-9}	7.7	1.08
Iron (Steel)	2.80×10^{-9}	14.8	1.17
Tungsten	4.96×10^{-9}	51.9	2.08
Gadolinium Oxysulfide	1.31 x 10-8	81.9	5.51
Aluminium	1.90×10^{-8}	87.4	7.96
Magnesium	2.33×10^{-8}	89.8	9.77
Barium	2.36×10^{-8}	89.9	9.92
Xenon	2.69×10^{-8}	91.1	11.30
Caesium Iodide	2.83×10^{-8}	91.6	11.90
Tin	3.36×10^{-8}	92.9	14.10

Energy deposited in the 10 cm long three concentric cylindrical sections of the ionization chamber (IC) at the isocenter free-in-air for varying gantry material is presented in Table 2. The active volume of the IC is air-filled inner cylinder with a diameter of 6.7 mm. The second cylinder had a diameter of 10.2 mm. The space between the first and second cylinder represents the electrode wall which was simulated as C552 air-equivalent plastic with a density of 1.76 g cm^{-3}. The third cylinder had a diameter of 13.7 mm. The space between the second and third cylinder represents the ion chamber build-up cap, simulated as polyacetal plastic with a density of 1.43 g cm^{-3} [9]. The reference gantry material of air was observed to decrease in energy absorbed by the different sections of the IC as one moves from the air-filled cylinder to the build-up cap. The same could be said for gantry materials of aluminium, magnesium, gadolinium oxysulfide, barium, caesium iodide, xenon and tin. Gantry materials of zinc, iron and tungsten behavior was on the contrary i.e. increase in the energy absorbed by the different sections of the IC as one moves from the air-filled cylinder to the build-up cap.

TABLE 2. ENERGY DEPOSITED IN THE THREE SECTIONS OF THE IC

Gantry Material	Energy deposited in IC per Rotation (MeV/g/particle)		
	Air-filled Cylinder	Electrode Wall	IC Build-up Cap
Air	2.38×10^{-9}	2.36×10^{-9}	1.99×10^{-9}
Zinc	2.58×10^{-9}	3.06×10^{-9}	8.61×10^{-9}
Iron (Steel)	2.80×10^{-9}	2.80×10^{-9}	3.70×10^{-9}
Tungsten	4.96×10^{-9}	5.18×10^{-9}	7.31×10^{-9}
Gadolinium Oxysulfide	1.31×10^{-8}	1.31×10^{-8}	1.13×10^{-8}
Aluminum	1.90×10^{-8}	1.89×10^{-8}	1.63×10^{-8}
Magnesium	2.33×10^{-8}	2.32×10^{-8}	1.99×10^{-8}
Barium	2.36×10^{-8}	2.34×10^{-8}	1.86×10^{-8}
Xenon	2.69×10^{-8}	2.65×10^{-8}	2.11×10^{-8}
Caesium Iodide	2.83×10^{-8}	2.79×10^{-8}	2.23×10^{-8}
Tin	3.36×10^{-8}	3.29×10^{-8}	2.65×10^{-8}

Varying the gantry block material with zinc and tin respectively belonging to the two sets of groups identified in Table 1 and 2, the energy deposited in the staff's eye lens, thyroid and whole body is presented in Table 3. There is no significant difference in the radiation to the eye lens and thyroid irrespective of the presence of gantry block or not and the type of gantry block material. The radiation to the whole body without a gantry block (air) is more

than a factor of 2.3 higher than with a gantry block. However there is no difference in the whole body of staff when the type of gantry block material is varied.

TABLE 3. RADIATION ENERGY DEPOSITED IN THE EYE LENS AND THYROID, AND WHOLE BODY EXPOSURE PER ROTATION OF STAFF VARYING CT FLUOROSCOPY GANTRY MATERIAL

	Energy Deposited (MeV/g/particle/rotation)		
Gantry Material	Eye Lens	Thyroid	Whole Body
Air	1.05×10^{-9}	4.18×10^{-10}	6.41×10^{-9}
Zinc	1.04×10^{-9}	4.12×10^{-10}	2.81×10^{-9}
Tin	1.04×10^{-9}	4.11×10^{-10}	$2.80\ 10^{-9}$

4. CONCLUSION

The relevance of a CT gantry in MC studies for medical staff exposure has been illustrated. Gantry block materials of aluminium, magnesium, gadolinium oxysulfide, barium, caesium iodide, xenon and tin with scatter radiation contribution of more than 80 % at the isocenter were observed to have similar characteristics in the sections of IC as compared with no gantry (air). Gantry block materials of zinc, iron and tungsten were observed to have a scatter radiation contribution of approximatley less than 50 % at the isocenter but had opposite characteristics as compared with air in the sections of IC.

The radiation deposition to the staff's eye lens and thyroid standing close to the patient in a CT room is independent of the presence of a gantry block or varying gantry block material. The whole body radiation exposure of medical staff was however dependent on the presence of a gantry block but independent on the type of gantry block material of zinc or tin.

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REFERENCES

[1] CARLSON, S. K. et al., Benefits and safety of CT fluoroscopy in interventional procedures, Radiology, 219 (2001) 515-520.

[2] TACK, D., et al. (eds), Radiation Dose from Multi-detector CT, Medical Radiology Diagnostic Imaging. Book Chapter: BULS, B., VANDENBROUCKE, F., DE MEY, J., "Dose reduction in CT fluoroscopy", DOI: 10.1007/174_2011_453, ©Springer-Verlag Berlin Heidelberg, (2012).

[3] KATO, R., et al., Interventional radiology of the abdomen and pelvis with real-time CT fluoroscopy, Radiology, 201 (1996) 478.

[4] SEIBEL, R., et al., Interventional procedures with real time CT, Radiology, 205 (1997) 383.

[5] NAWFEL, R., et al., Patient and personnel exposure during CT fluoroscopy-guided interventional procedures, Radiology, 216 (2000) 180-184.

[6] AMERICAN ASSOCIATION OF PHYSICISTS IN MEDICINE, The measurement, reporting, and management of radiation dose in CT", Report of AAPM Task Group 23 of the Diagnostic Imaging Council CT Committee, Report No. 96. ISBN: 978-1-888340-73-0 (2008).

[7] DEMARCO, J.J. et al., Estimating radiation doses from multidetector CT using Monte Carlo simulations: effects of different size voxelized patient models on magnitudes of organ and effective dose, Phys. Med. Biol. 52 (2007) 2583–2597.

[8] FIGUEIRA, C., et al. Medical staff extremity dosimetry in CT fluoroscopy: an anthropomorphic hand voxel phantom study, Phys. Med. Biol. 58 (2013) 5433-5448.

[9] GU, J. et al., The development, validation and application of a multi-detector CT (MDCT) scanner model for assessing organ doses to the pregnant patient and the fetus using Monte Carlo simulations, Phys. Med. Biol. 54 (2009) 2699-2717.
[10] CASSOLA, V. F., et al., FASH and MASH: female and male adult human phantoms based on polygon mesh surfaces: I. Development of the anatomy, Phys. Med. Biol. 55 (2010) 133-162.
[11] THEIS, C. et al., Interactive three-dimensional visualization and creation of geometries for Monte Carlo calculations, Nuclear Instruments and Methods in Physics Research, A 562 (2006) 827-829.
[12] PÖLZ, S. et al., Voxel2MCNP: a framework for modeling, simulation and evaluation of radiation transport scenarios for Monte Carlo codes, Phys. Med. Biol. 58 (2013) 5381-5400.
[13] POLUDNIOWSKI, G., et al., SpekCalc: a program to calculate photon spectra from tungsten anode x-ray tubes, Phys. Med. Biol. 54 (2009) N433-N438.
[14] PELOWITZ, D. B., MCNPXTM USER'S MANUAL, Version 2.6.0, Los Alamos National Laboratory Report LA-CP-07-1473 (2008).

MEASUREMENT OF PERSONAL DOSE EQUIVALENT HP(10) IN PHOTON FIELDS IN THE AFRICA REGION : ORGANIZATION, RESULTS AND RECOMMENDATIONS OF THE INTERCOMPARISON 2013

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Abstract.

In the framework of RAF9043 Technical Cooperation project, an intercomparison on measurement of Personal Dose Equivalent Hp(10) was organized in 2013. Four sets of personal dosemeters, mainly TL and OSL dosemeters, were irradiated at the Secondary Standard Dosimetry laboratory of Algiers, aiming at verifying the performance of the Individual Monitoring Services (IMS) of the participants in order to assess their capabilities to measure the quantity Hp(10) in photon (gamma and X ray) fields. The main goal of the intercomparison being to help the participants to comply with internationally recognized dose limitation requirements. The organization of the intercomparison is described in details. For each participating country, the results of the linearity, energy and angular responses as well as well as the blind test are presented. The discrepancies observed are analyzed and, for some countries, correction factors are applied in order to improve the results. Finally, recommendations are drawn in order to improve the quality of the services provided by IMS in Africa.

1. INTRODUCTION

Personal dosemeters, mainly Thermoluminescence and Optically Simulated Luminescence Dosemeters, are widely used in many countries where occupational exposure to ionizing radiation can occur, e.g. medical applications, education and research institutes, nuclear plants etc. Adequate radiation protection of workers is essential for the safe and acceptable use of any ionizing radiation.

Pursuant to General Conference resolution GC(43)/RES/13 [1], the International Atomic Energy Agency (IAEA) organized since the early 1980s intercomparisons for personnel dosimetry focusing on the performance of personal dosimetry services for photon beams [2,3,4,5,6,7,8]. In the framework of the IAEA TC Regional Project RAF/9/043: Strengthening the Transfer of Experience Related to Occupational Radiation Protection of the Nuclear Industry and other Applications involving ionizing radiation, an intercomparison on measurements of the quantity personal dose equivalent Hp(10) in photon fields in the Africa Region was organized in 2013 in cooperation with the Algerian Secondary Standard Dosimetry Laboratory [9]. Twenty four (24) countries from the Africa region and three (03) countries from outside Africa region participated in the intercomparison. One country participated with two systems. The confidentiality of the intercomparison results was preserved by randomly generating code numbers from C1 to C28.

The overall objective was to verify performance and improve the Individual Monitoring Services (IMS) for the participating countries. This was planned to be achieved with specific objectives of assessing the capabilities of the dosimetry services to measure the quantity Hp(10) in photon (gamma and X ray) fields, helping the participating Member States in achieving sufficiently accurate dosimetry service and, if necessary, providing guidelines for improving the performance of the existing dosimetry services.

2. MATERIAL AND METHODS

The scope of the intercomparison was to compare the measurements of personal dose equivalent Hp(10) in photon radiation fields. For this purpose, the following dosemeters, which are commonly implemented in the region, were used (Fig. 1): Thermoluminescence dosemeter (TLD) and Optically Simulated Dosemeters (OSL).

Fig. 1. Dosemeters used for the intercomparison

The response of the dosemeters is evaluated in terms of ratio H_{pm}/H_{pt}, Where H_{pt} is the conventional true value stated by the irradiating laboratory, H_{pm} is the value measured by the participant. This ratio should meet the criteria as given in the IAEA safety standards series RS-G-1.3 "Assessment of Occupational Exposure Due to External Sources of Radiation" [10] and represented by a so called "Trumpet Curve".

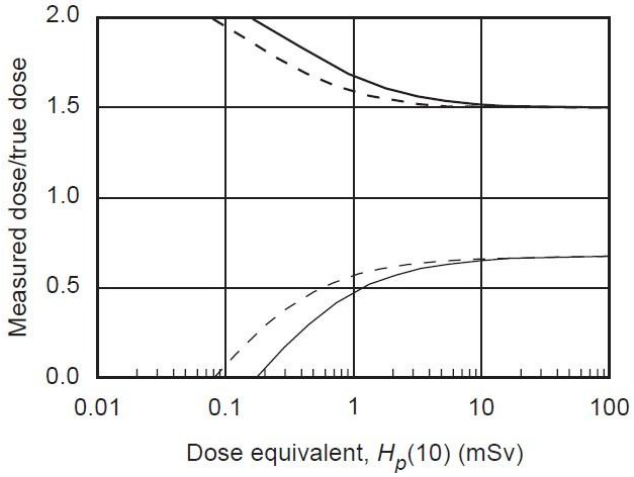


FIG. 2. Acceptable upper and lower limits for the ratio measured dose/conventional true dose as a function of dose for Hp(10)(reproduced from the reference 9)

The performances of the dosemeters are evaluated by irradiating them using ISO 4037 beam qualities [11, 12, 13] and aiming at checking:

1. The Linearity response: Irradiations with doses lying between 0.5 mSv to 10 mSv, with normal incidence in a Cesium-137 gamma source. Three sets of dosemeters were used for each dose.
2. The energy response: Irradiations to 2 mSv using three ISO 4037 X-ray qualities (N-60, N-80 and N-150). Three sets of dosemeters were used for each quality.
3. Irradiations to 2 mSv in Cs-137 with angular incidences of 0°, 45° and 60°. Two dosemeters were used for each angle

In addition, Irradiations were performed in mixed photon qualities using different irradiation conditions (ISO N and S series qualities) including rotational field, simulating real workplace conditions. This blind test was aimed at checking the ability of the participants to measure doses in real working conditions.

Dosemeters used for evaluating the transportation and background doses were added to the package. These dosemeters were useful for getting information about the doses received especially during the screening at the airport before and after irradiations.

The dosemeters were irradiated using the facilities belonging to the Algerian Secondary standard dosimetry Laboratory: a ^{137}Cs gamma irradiator of type OB6, a Philips X-ray unit and an ELDORADO 78 ^{60}Co therapy level irradiation unit, using the geometrical irradiation conditions illustrated by the figure 3 in case of x-rays.

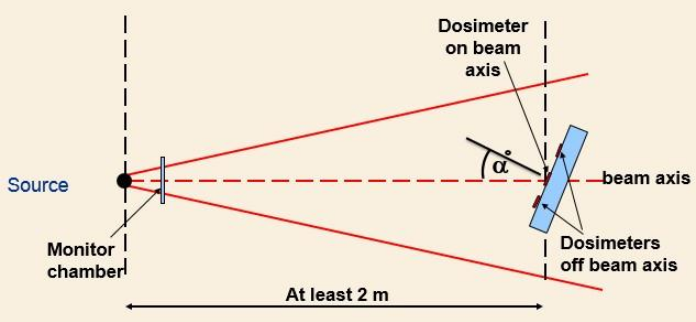


Fig. 3. Example of irradiation geometry of dosemeters in X-rays qualities

3. RESULTS OF THE INTERCOMPARISON

3.1. Background and Transport dose

As part of the intercomparison process, up to twelve (12) additional dosemeters were added aiming at evaluating the background and transportation doses, BGBT, received by the dosemeters before and after their irradiation (environmental irradiations, scanning process at the airports,…). As can be seen in the figure 4, the mean value of BGBT was (0.25 ±0.14) mSv which is not negligible compared to the lowest dose value used in the intercomparison. For those participants who did not subtract the BG value from the evaluated doses, the results transmitted by them were recalculated using this mean value.

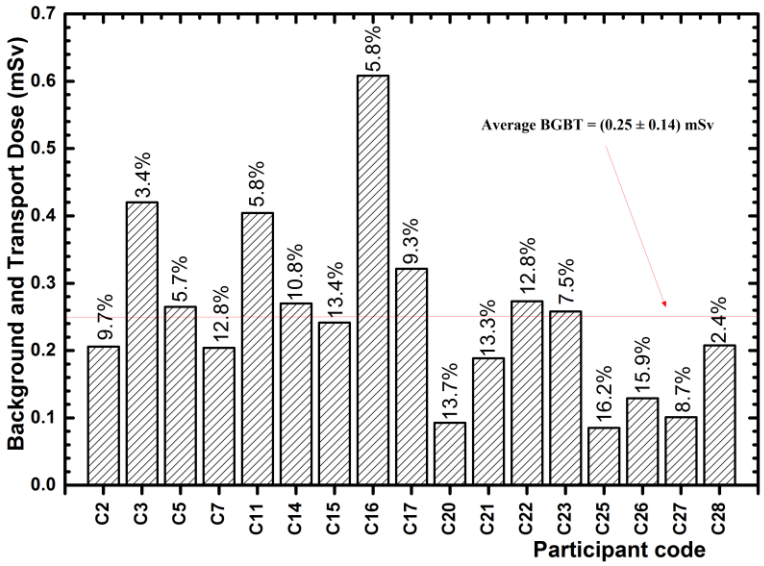


Fig. 4: Background and transport dose (BG) as measured by some participants. The labels represent the relative standard deviations (reproduced from the reference 14).

3.2. Results of the intercomparison

The overall results of the intercomparison are given for each participant by the figures 5 (linearity response), Figure 6 (energy response), Figure 7 (angular response) and Figure 8 (blind test, mixed field). The figure 9 illustrates the percentage of results outside the trumpet curve, given as overestimation and underestimation of doses.

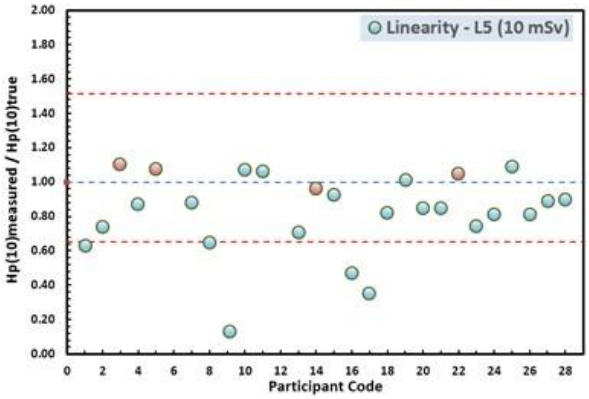


Fig. 5: Results of linearity response

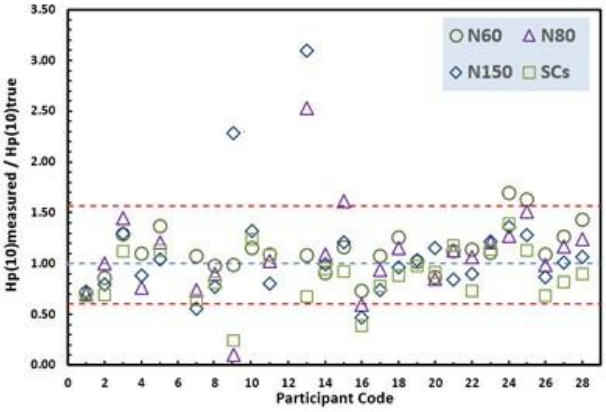


Fig. 6: Results of Energy response

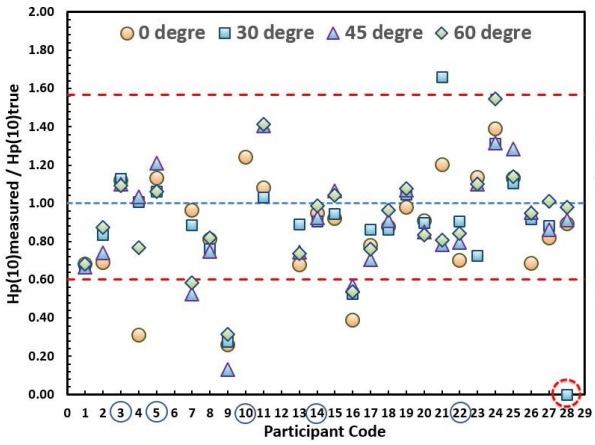


Fig. 7: Results of Angular response field)

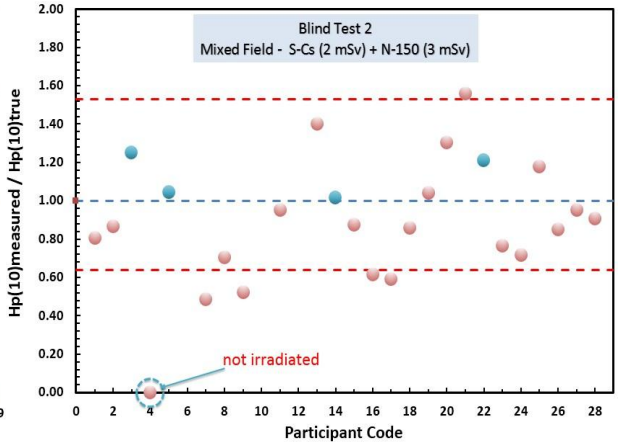


Fig. 6: Results of Blind test (mixed field)

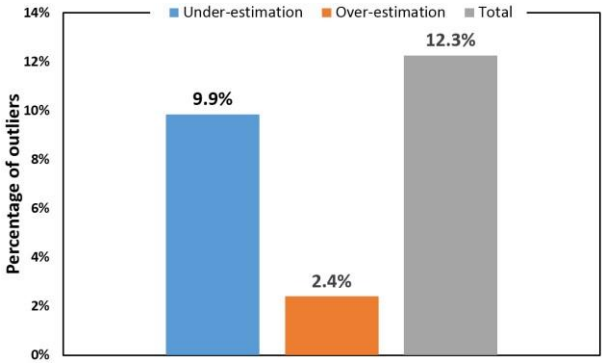
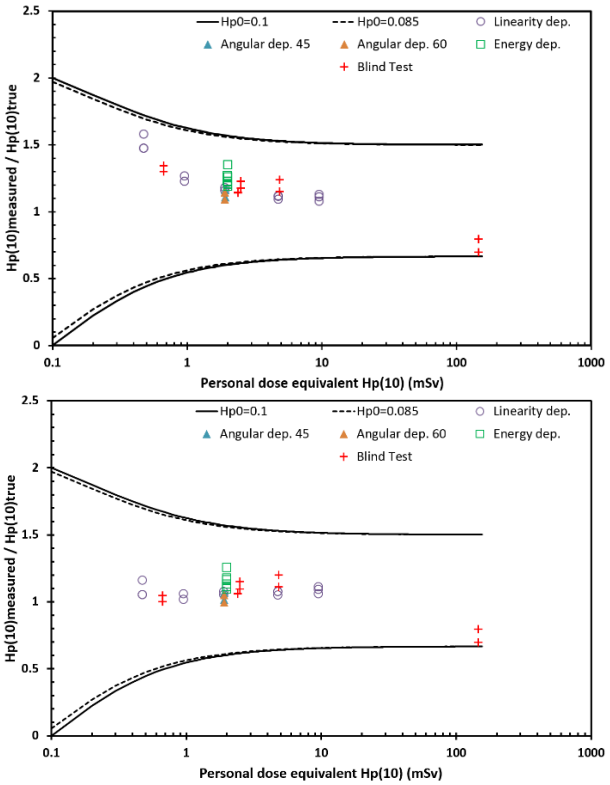


Figure 9. Overall results: Percentage values of outliers (under and over-estimation of dose)

4. DISCUSSIONS

As can be seen on figure 4, the doses received by the dosemeters from the background and transport (airport screening) irradiations are not negligible compared to the lowest dose communicated to the dosemeters. These extra irradiations were not taken into account by many participants leading to faulty results. Indeed as can be seen in figure 10, taken from one participant, the results are improved when correction for background and transport dose is applied.



(a) (b)

Figure 10: Example of results before (a) and after (b) correction for background and transport dose. (reproduced from reference 14)

After application of BG correction, the following results can be observed:

1- The mean ratio of the $Hp(10)_{measured}$ over $Hp(10)_{true}$ is 0.999 ± 0.34 for all the results. This ratio is 0.997 ± 0.62 for linearity (1.00 ± 0.52 for TLD and 1.07 ± 0.25 for OSL), 1.07 ± 0.39 for energy response and 0.95 ± 0.25 for angular response.

2- For linearity, there are 10.8% of outliers among which 10% underestimate the dose and 0.8% over estimate it

3- For Energy response, we have 15.4% of outliers with 7.7% of under estimation and 7.7% of overestimation respectively

4- For angular response we have 15.4% of outliers with 12.8% under estimation and 2.6% over estimation of dose

5- The 10.0% outliers for blind test gave 9.2% and 0.8% under estimation of dose

5. CONCLUSIONS

As can be seen on figure 9, most of the 12.3% outliers which are below the lower acceptance limit lead to an underestimation of dose (9.9%). The rest (2.4%) are overestimation of dose. It should be recognized that an underestimation of dose is more dramatic and could have grave impact on the monitored workers. Therefore, the dosimetry systems of the participants concerned by these outliers, need urgently to be recalibrated by sending the golden cards to any of the regional SSDLs for reference irradiations.

As main recommendation to the IAEA, a second intercomparison should be organized after recalibration of the dosimetry systems by the countries. This intercomparison should include $H_P(10)$ quantities and well as Hp(0.07) for ISO 4037 qualities and, if necessary, Beta radiations for extremity dosimetry. For this purpose, the SSDL of Algeria is willing to organize again this intercomparison.

REFERENCES

[1]. INTERNATIONAL ATOMIC ENERGY AGENCY, General Conference. MEASURES TO STRENGTHEN INTERNATIONAL CO-OPERATION. Resolution adopted on 1 October 1999 during the ninth plenary meeting. http://www.iaea.org/About/Policy/GC/GC43/GC43Resolutions/English/gc43res-13_en.pdf

[2]. IAEA. Intercomparison of radiation dosemeters for individual monitoring (final report of a coordinated research programme 1988 – 1992). IAEA TECDOC Series No. 704, IAEA (1993).

[3]. INTERNATIONAL ATOMIC ENERGY AGENCY, Intercomparison for individual monitoring of external exposure from photon radiation. Results of a co-ordinated research project 1996-1998. Vienna (1999)

[4]. INTERNATIONAL ATOMIC ENERGY AGENCY, Intercomparison of radiation dosemeters for individual monitoring. IAEA TECDOC-704, Vienna (1993).

[5]. INTERNATIONAL ATOMIC ENERGY AGENCY, Intercomparison of Measurements of Personal Dose Equivalent Hp(10) in Photon Fields in the West Asia Region. IAEA-TECDOCCD-1567, Vienna (2007)

[6]. INTERNATIONAL ATOMIC ENERGY AGENCY, Intercomparison of Personal Dose Equivalent Measurements by Active Personal Dosemeters Final Report of a joint IAEAEURADOS Project. IAEA-TECDOC-1564, Vienna (2007)

[7]. Z. Msimang, R. Cruz-Suárez and J. Zeger. Intercomparison on measurements of the quantity personal dose quivalent h p (10) in photon fields in the african region. Radiation Protection Dosimetry (2011), Vol. 144, No. 1 – 4, pp. 282–286

[8]. M. Saravı, A. Zaretzky, C. Lindner, J.Dıaz, G. Walwyn, R. Amorim, D. De Souza, B. Gregori, S. Papadopulos, A. Meghzifene, P. Ferruz and R. Cruz Suarez Results of the regional intercomparison exercise for the determination of operational quantity Hp(10) in latin america Radiation Protection Dosimetry (2006), 1 of 4

[9]. M. Arib, A. Herrati, F. Dari, J. Ma and Z. LounisMokrani. Intercomparison 2013 on measurements of the Personal dose equivalent $H_P(10)$ in photon fields In the African region. Radiation Protection Dosimetry (2014), pp. 1–8 doi:10.1093/rpd/ncu202.

[10]. INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of occupational exposure due to external sources of radiation. Safety Standards Series, No. RS-G-1.3, IAEA (1999).

[11] ISO (International Organization for Standardization). X and gamma reference radiation for calibrating dosemeters and doserate meters and for determining their response as a function of photon energy—Part 1: Radiation characteristics and production methods. ISO 4037-1, ISO (1996).

[12]. ISO (International Organization for Standardization). X and gamma reference radiation for calibrating dosemeters and doserate meters and for determining their response as a function of photon energy—Part 2: Dosimetry for radiation protection over the energy ranges 8 keV to 1,3 MeV and 4 MeV to 9 MeV. ISO 4037-2, ISO (1997).

[13]. ISO (International Organization for Standardization). X and gamma reference radiation for calibrating dosemeters and doserate meters and for determining their response as a function of photon energy—Part 3: Calibration of area and personal dosemeters and the measurement of their response as a function of energy and angle of incidence. ISO 4037-3, ISO (1999).

[14] M. Arib, A. Herrati, F. Dari and Z. Lounis-Mokran. Effect of background and transport dose on the results of the personal dose equivalent Hp(10) measurements in photon fields obtained during the intercomparison 2013 of the African region. Final proof in Radiation protection Dosimetry (November 2014).

EURADOS WG02 ACTIONS: HARMONIZATION OF INDIVIDUAL MONITORING IN EUROPE

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Abstract

A summary of the actions of EURADOS WG02 performed since 1996 until today are briefly described. Special attention is given to the preparation of the Technical Recommendations for Monitoring Individuals Occupationally Exposed to External Radiation (EC publication Radiation Protection 160), the regular organization of inter-comparison exercises, training course actions and surveys for the assessment of QA & QC as well as collaboration to the organization of conferences fostering Harmonization of Individual Monitoring in Europe.

1. INTRODUCTION

The European Radiation Dosimetry Group (EURADOS) comprises a self-sustainable network of more than 60 European institutions and 300 scientists active in the field of radiation dosimetry. EURADOS e.V. is registered in the German Register of Societies as a non-profit association with the aim to promote research and development and European cooperation in the field of dosimetry of ionizing radiation. For this, EURADOS has established Working Groups (WGs): WG02 on harmonization of individual monitoring, WG03 on environmental dosimetry, WG06 on computational dosimetry, WG07 on internal dosimetry, WG09 on radiation protection dosimetry in medicine, WG10 on retrospective dosimetry, WG11 on dosimetry in high energy radiation fields, and WG12 on dosimetry in medical imaging. EURADOS is governed by a Council composed of not more than twelve members elected by the representatives of the institutions (voting members) for three-year periods. The Council ordinarily meets twice every year with all the WG chairpersons to analyze the work performed and decide on future work plans. EURADOS has recently prepared a Strategic Research Agenda available at www.eurados.org . More information on EURADOS and its WGs can be found at this web site.

2. SUMMARY OF WG02 ACTIVITY SINCE 1996

EURADOS WG02 on Harmonisation of Individual Monitoring in Europe was established in 1996 by the EURADOS Council. Its activity is briefly summarized below:

1996–2000 Main task: Evaluation of the implementation of Council Directive 96/29 EURATOM of 13 May 1996 with respect to monitoring persons occupationally exposed to ionizing radiation in terms of the then new quantities personal dose equivalent, also in view of the then recently published Technical Recommendations EUR14852. In this period the WG was supported by the EC DG Science, Research and Development (SRD) under contract F14P CT96-0061, and chaired by D. Bartlett [1].

2001–2005 Main tasks: Harmonization activities addressing: The implementation of standards to individual monitoring; Integration of dosimetry for internal and external occupational exposure; On the use of active (electronic) dosemeters; Quality Assurance and Quality

Control (QA&QC) and reliability of dosimetric systems. The WG was also supported by EC DG SRD under contract FIR1-CT-2000-20104, and chaired by J.W.E. van Dijk [2].

2005–2009 The EURADOS Council decided to support WG02 chaired by V. Kamenopoulou [3] with two tasks: 1. On the need to revise EUR 14852 Technical recommendations for monitoring individuals occupationally exposed to external radiation; 2. A feasibility study for a self-sustained programme of intercomparisons (ICs) for passive devices. These tasks evolved into EU-Trimer, IC2008 and IC2009.

EU-Trimer 2007-2009 In the end of 2006, EC issued a call for a tender for the preparation of the new European technical recommendations for monitoring individuals occupationally exposed to external radiation that would replace EUR 14852. A Consortium agreement was established between GAEC (leader) and EURADOS involving PHE (formerly HPA), NRG, PTB, ENEA, RPII and IST (formerly ITN). The proposal was successful and a task group composed by D. Bartlett, J.W.E. van Dijk, P. Ambrosi, E. Fantuzzi, L. Currivan, J.G. Alves and V. Kamenopoulou was set. As an output of this project, the European Commission's Radiation Protection 160 (RP 160) was published [4]. RP 160 is a wide consensus document (EU Member States' IM services, national radiation protection bodies, national metrology laboratories, authorities, standardization bodies (ISO, IEC), ICRP, ICRU and IAEA were all consulted at different stages). RP 160 was published in Nov-2009 and in Mar-2010 it was presented to the wider community at the European IM conference IM2010 [5,6].

IC2008 In 2008 the EURADOS Council decided to perform the first self-sustained intercomparison exercise for whole-body dosemeters for photon fields and appointed an Organizing Group (OG) composed by T. Grimbergen (coord), M. Figel, A. McWhan, A. Romero and H. Stadtmann. The action ended with the participant's session and presentation of certificates at the EURADOS Annual Meeting 2009. IC2008 was a success and encouraged the Council to carry on with this activity [7].

IC2009 The same OG was appointed to prepare a second IC exercise this time for extremity dosemeters in photon and beta fields. The action was also successful and ended with the participants' session at the European Individual Monitoring conference IM2010 in Athens [8].

2010–present Main tasks: Organization of regular and self-sustained IC exercises; Quality assurance and quality control, dose recording and dose reporting; Dissemination of activity including the organization of Training Courses on the Implementation of RP160. The WG is supported by the EURADOS Council and chaired by J.G. Alves.

2.1. NETWORK OF CONTACTS

For the development of the above mentioned tasks, regular updating was done by a network of contact persons, consisting of one person per country, usually the attendant to the WG02 meeting. Depending on the type of necessary information, this person gets in touch with the IM services and/or national radiation protection authorities in his country and neighbouring countries. At present, the network includes contacts from all European nations and a few neighbour countries.

2.2. Self-sustained actions

The Intercomparison (IC) exercises and Training Courses (TC) are organized as self-sustained actions. That is, the revenue from the attendants' fees covers expenses and preferably generates a positive balance. In general, actions are carried out by an organizing group (OG) suggested by the WG and appointed by the Council following the analysis of a calendar and the approval of a preliminary budget. The budget includes manpower costs for

the coordinator and OG members, travel and subsistence and other costs, e.g. irradiation costs for IC. Travel and subsistence are covered at real costs. Yet, EURADOS counts on the collaboration of the home institutes, that is, manpower is not charged at the real cost of dedicated amount of time and/or work. On the other hand, the institutes also recognise the importance of the activity and the increased visibility for their institution within the dosimetric community [9].

3. INTERCOMPARISON EXERCISES

WG02 developed a system for self-sustained IC exercises for IM services for external radiation. The first IC exercise (IC2008) took place in 2008-2009 and was run by OG mentioned above, who shared all necessary tasks: preparation of a calendar, set-up of the irradiation plan (final plan is only know by the co-ordinator), contacts with accredited metrology laboratories for irradiations, receiving dosemeters from participants and sending to the irradiation laboratory, receiving dosemeters following irradiation and distribution back to the IMS for evaluation; collection and analysis of results declared by participants, preparation and distribution of certificates to participants). In general, at EURADOS Annual Meetings a special session is held for the participants to the IC exercise and generally takes place at EURADOS Annual Meetings or at IM conferences (see 5. below). This session includes presentations describing the IC set-up, irradiations at the irradiation laboratory, problems, if any, are also addressed and a preliminary assessment of the results obtained is presented and briefly discussed. EURADOS reports with all data are published. Presentations to conferences attended by the IMS community and summaries to scientific journals are also prepared, presented and published. From 2008 and until 2014 the following IC have already taken place:

TABLE I. IC EXERCISES PERFORMED IN THE PERIOD 2008 – 2014

Scope	IC	IMS	Systems	Participants	Certificates
Whole-body dosemeters for photon fields	2008	52	62	Europe, IAEA	AM2009
	2010	70	84	Europe, IAEA	AM2011
	2012	74	88	Europe, Argentina, Japan, USA	AM2013
	2014	96	112	As above, India, Israel, Lebanon	IM2015
Extremity, ph/β fields	2009	44	59	Europe, IAEA	IM2010
Neutron dosemeters	2012	31	34	Europe, IAEA, Japan, USA	Neudos12

Four whole-body photon IC exercises have been organized on a two-year interval, meeting the IMS needs to comply with ISO/IEC 17025 requirements for accreditation. Special IC exercises were organized in 2009 and 2012, respectively, for extremity dosemeters for photon and beta fields and for neutron dosemeters The OG composition of this last IC was E. Fantuzzi, S. Mayer, M. Luszik-Bahdra, F. Vanhavere, R. Tanner, D. Thomas, M.-A. Chevallier and R.C. Suárez [10].

4. TRAINING COURSES

WG02 organized Training Courses on the Implementation of RP 160 and lessons learned from Intercomparison exercises. The course covers the implementation of Radiation Protection 160, as well as the implications of the accreditation standard (EN ISO/IEC 17025) to IMS. An organizing committee composed by J.W.E. van Dijk, O. Hupe, J.G. Alves, P. Gilvin, M. Figel and R. Kopec (previously also P. Ambrosi, D. Bartlett and H. Stadtmann) prepared and imparted most of the lectures. So far two TC have taken place, the first in Krakow (Poland) at the Institute of Nuclear Physics, November 2012 with 41 participants

from EU, Japan and Ukraine and the second at the Rudjer Boskovic Institute in Zagreb (Croatia), November 2013 with 32 participants from EU and Turkey. A third TC is planned for 2014 at Instituto Superior Técnico in Bobadela, close to Lisboa (Portugal).

5. EURADOS QA & QC SURVEYS

The collection of information from the network of contacts is often performed by use of surveys. The most recent one took place in 2012-2014. The analysis of the 2012 survey indicates that the profile of QA is high amongst the responding IMS and that most are following good practice. The majority of services are certified (around 70%) or declared themselves compliant to quality standards, mostly in accordance with EN ISO/IEC 17025 or with ISO 9001. Accreditation is gradually becoming important for European IMS. All respondents quoted a traceability route, and most check it at least annually. The majority undergo proficiency testing and around 60% take part in IC exercises. Uncertainty assessment is of concern and about 40% declare having an internal procedure for its assessment. Most are in the range 11-15%, although some reported values seem unrealistically low suggesting an incomplete assessment.

6. COLLABORATION IN THE ORGANIZATION OF INDIVIDUAL MONITORING CONFERENCES

Following the first WG on harmonization, an individual monitoring conference was organized with direct and/or indirect support of EURADOS (IM2000, Helsinki). The EURADOS network, e.g. members of WG02, WG06, WG07, WG12 mainly, are involved in the organization of the IM conferences as members of the scientific committees, invited lecturers, session chairs, co-chair and rapporteurs, referees for the preparation of proceedings, etc. The preparation work roughly starts 2y in advance and ends approximately one year after with the publications of presented papers. So far, the following IM conferences were organized every 5y: IM2000 in Helsinki, organized by STUK, IM2005 in Vienna, organized by Seibersdorf Laboratories and the IAEA, IM2010 in Athens organized by GAEC and to forthcoming IM2015, in Bruges, organized by SCK-CEN and Controlatom.

7. FURTHER STEPS FOR WG02

EURADOS WG02 will continue to regularly organize IC and TC actions, with the following plans: IC for whole-body dosemeters for photon fields every 2y; IC for extremity dosemeters for photon and beta fields every 5-6y; IC for neutron dosemeters every 5-6y; TC on Implementation of RP160 and special topics of radiation protection dosimetry every year, depending on demand.

For IM of external radiation the following is to be considered: The evolution of ICRP and ICRU concepts and recommendations; The publication of EU Council Directive 2013/59 incorporating ICRP/ICRU recommendations, their implications on measurement quantities, phantoms and the new revised annual limits; Recommendations issued from the analysis of the results obtained from surveys and deviations observed in IC exercises; Implications due to the change of dosimetry systems based on passive (film, TLD, OSL, track-etch, etc) detectors to active devices (APDs) and other novel methods; Identification of causes for deviations and seeking for improvement of quality; Identification of eventual needs for the revision of reference documentation. A strong collaboration with the IMS community is needed in order to meet their needs and contribute to Harmonization of Individual Monitoring in Europe.

ACKNOWLEDGEMENTS

This paper is co-authored by the present WG02 members and former chairpersons. Thanks are due to all previous members as well as to the network of contact persons who contribute to this work and made it possible.

REFERENCES

[1] D.T. BARTLETT, P. AMBROSI, J.M. BORDY, J.W.E. VAN DIJK, Harmonization and Dosimetric Quality Assurance in Individual Monitoring for External. Radiation, Radiat. Prot. Dosim. 89 (2000) 1-2.

[2] J.W.E. VAN DIJK, T. BOLOGNESE-MILSZTAJN, E. FANTUZZI, M.A. LOPEZ PONTE, H. STADTMANN, Harmonization of Individual Monitoring in Europe, Radiat. Prot. Dosim. 112, (2004) 1.

[3] V. KAMENOPOULOU et al., Aspects of Harmonization of IM for External Radiation in Europe: Conclusions of a EURADOS Action, Radiat. Prot. Dosim. 118 (2) (2006) 139-143.

[4] Radiation Protection 160 European Technical Recommendations for Monitoring Individuals Occupationally Exposed to External Radiation. DG Energy, Luxembourg, (2009).

[5] J.G. ALVES et al., The new EC technical recommendations for monitoring individuals occupationally exposed to external radiation, Radiat. Prot. Dosim. 114 (1-4) (2011) 17-25.

[6] J.W.E. VAN DIJK et al., A Review of New European Technical Recommendations for Monitoring Individuals Occupationally Exposed to External Radiation, Rad. Meas. 55 (2013) 30-33.

[7] EURADOS Report 2012-01, EURADOS IC2008, Braunschweig, January (2012).

[8] EURADOS Report 2013-03, EURADOS IC2009, Braunschweig, October (2013).

[9] T.W.M. GRIMBERGEN, M. FIGEL, A.M. ROMERO, H. STADTMANN, A.F. MCWHAN, EURADOS Self-sustained Programme of IC for IM services, Radiat. Prot. Dosim. 144 (1-4) (2011) 266-274.

[10] E. FANTUZZI et al., EURADOS IC2012n Intercomparison for whole-body neutron dosimetry, Radiat. Prot. Dosim. Published online Feb2014 (2014).

PERSONAL DOSEMETERS PERFORMANCE TESTING FOR SIX SERVICE PROVIDERS IN FIVE DIFFERENT COUNTRIES IN WESTERN ASIA REGION

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Abstract

The overall objective was to verify performance of dosemeters used by six Individual Monitoring Services (IMS) operating in western Asia region. Thermoluminance dosemeter TLDs were used by most involved IMS but other types of radiation detectors such as films and photo-luminescent detectors were also studied. All used dosemeters have been tested for their ability to determine the whole body personal equivalent dose quantity H_p (10) for penetrating radiation. The results were studied without background correction and dosemeters response analysis was based on the trumpet curves methodology. All delivered results by IMS, except one, were within accepted trumpet curve recommended limits. The relative deviation observed in the dosemeter response at used photon energy 662 keV were within the required ± 40% for most investigated dosimetry system; however, two IMS show higher level especially at high H_p (10) values.

1. INTRODUCTION

The primary objectives of personnel dosimetry programs are the monitoring of radiation doses received by radiation workers during routine occupational exposure. Verifying the compliance of personal dosimetry system with the performance requirements can be achieved through regular calibration, performance testing of the used detectors as well as participating in dosimetry inter-comparison. Simple photon beam and more complex mixed photon qualities with different irradiation conditions including rotational field, simulating real work place conditions are required. Usually, the term 'performance test' means a routine performance or proficiency testing. Such tests have the purpose of determining the reliability of the routinely applied measurement procedures. Performance tests can examine the consistency of measurements (i.e. reliability) as well as spot testing of the overall accuracy. There are three types of proficiency testing in general, the 'blind' test, the 'surprise' test and the 'announced' regular test. The overall accuracy of the dose assessment given by a dosimetry method depends on the characteristics of the dosemeter, its suitability for the purpose, and on the dosimetry services quality assurance system [1]. These determine also the reliability and consistency for the measurement method application. The dosimetric properties of the radiation dosimetry system are achieved by type-testing, calibration and performance exercises, which may include energy, angular dependence response of the dosemeter type in use. Repeatability, reproducibility, effect of influence quantities, and checking other factors linked to the measurement method are also important.

The ICRP recommendations [2] are applied in order to find the magnitude of the quotient H_m/H_c of the measured dose value, H_m, and the conventional true value, H_c, (i.e. value given be the standard dosimetry laboratory). It can be used to determine H_p (10) for whole body dosemeters as follows:

(a) When the dose value equal to or approaching the annual dose limit, that is 20 mSv, acceptable performance is described by the relation $1.5 \geq H_m/H_c \geq 1/1.5$ at the 95 % confidence level.
(b) For a dose value at about the recording level for a monitoring period, the corresponding relation is taken as $2.0 \geq H_m/H_c \geq 0$, with no confidence levels are given in this case. These two criteria have been joined together by a smooth curve, the so-called "trumpet curve" methodology and applied generally to dosemeters routine performance testing.

These ICRP recommendations [3] have become the basis criteria for type testing, calibration and performance testing of personal dosimetry systems. When applied to the determination of the measurement quantity, the criterion of a factor of 1.5 on overall accuracy at or near dose limits is reasonably achievable for photon and electron beams. Inter-comparisons carried out by some national or international organisation such as the EURADOS and by IAEA [4] have demonstrated that it is possible for most personal dosimetry system to achieve these criteria.

Performance tests are intended to assess the capability of the dosimetry service in making the measurements and using a specific dosimetric system to comply with the specific performance criteria. They can be used to obtain an estimation of the overall accuracy of a dosimetry service, as in the tests against the 'trumpet curve'. Performance testing is used to assess relative deviation "or bias" and standard deviation, for groups of dosemeters irradiated to a range of radiation doses. A consideration of the acceptable accuracy to be expected of a personal dosimetry system will involve applying a magnitude of the doses exist in reality, and may take account of the fact that the quantity measured, $H_p(10)$, differs from the quantities limits based on other radiation protection quantities [5].

2. METHODOLOGY

Basic performance of six personal dosimetry service providers for external radiation occupational exposure was investigated. This study was organised on a voluntary bases, hence participating IMS can assess their ability in achieving sufficiently accurate results. Ten dosemeters from each IMS were irradiated at the Syrian secondary standard dosimetry laboratory. The reference radiation beams were calibrated using reliable dosimetry system calibrated at the IAEA dosimetry laboratory. Dosemeters were irradiated on an ISO polymethil methacrylate PMMA slab phantom with dimension 30×30×15 cm^3 using Cs-137 horizontal collimated photon beam at 2 m distance from the source [6]. The main criterion for stating compliance with the performance requirements was the acceptability of the dosemeters responses results through the "trumpet curve" as established in equation (1) [1].

$$\frac{1}{F}\left(1-\frac{2H_0}{H_0+H_c}\right) \leq R \leq F\left(1+\frac{H_0}{2H_0+H_c}\right) \qquad (1)$$

Where:

$R = \frac{H_m}{H_c}$, is the dosemeter response,

F=1.5: for general case,

H_c is the conventional true value given by the standard dosimetry laboratory,

H_m is measured value of the personal equivalent dose,

H_0 is the measuring range limit; H_0=0.1 mSv for whole-body dosemeters used for measuring H_p (10).

3. RESULTS AND DISCUSSIONS

Fig. 1 shows all IMS response values as a function of the reference doses using a logarithmic response scale. The solid lines represent the trumpet curves limits. The outliers count for about 15% of the total numbers of reported values considering a strict criteria factor F=1.5. Although the detailed study has considered other irradiation options, the reported results are only for ^{137}Cs photons radiation quality and 0-degree incidence angle. Results delivered by participated dosimetry service providers were generally in good agreement with the standards over a range of applied doses, however some discrepancy and outliers can be found. Lower measuring range limits in the trumpet curve methodology are considered as the smallest stated values H_0. Relative deviation, of the measured personal equivalent dose H_m, up to ±100% from the conventional true value H_c is allowed for doses in the range close to H_0. However, with increasing the personal equivalent dose values, the maximum permissible relative deviation decreases according to equation (1). Therefore, with high H_c values the ratio H_m/H_c should be within the range 0.67 to 1.5. Any results, not contained within the acceptability band of the trumpet curve were regarded as performance indicators "outliers" [7]. The dose assessment method or the used dosemeter calibration need to be check in these cases for system reliability improvement. When applying doses higher than 5 mSv, which was mainly used for personal dosimetry inter-comparison; the limits of the trumpet curve have been set to the figures 0.67 as lower limit and 1.5 as upper limit [8].

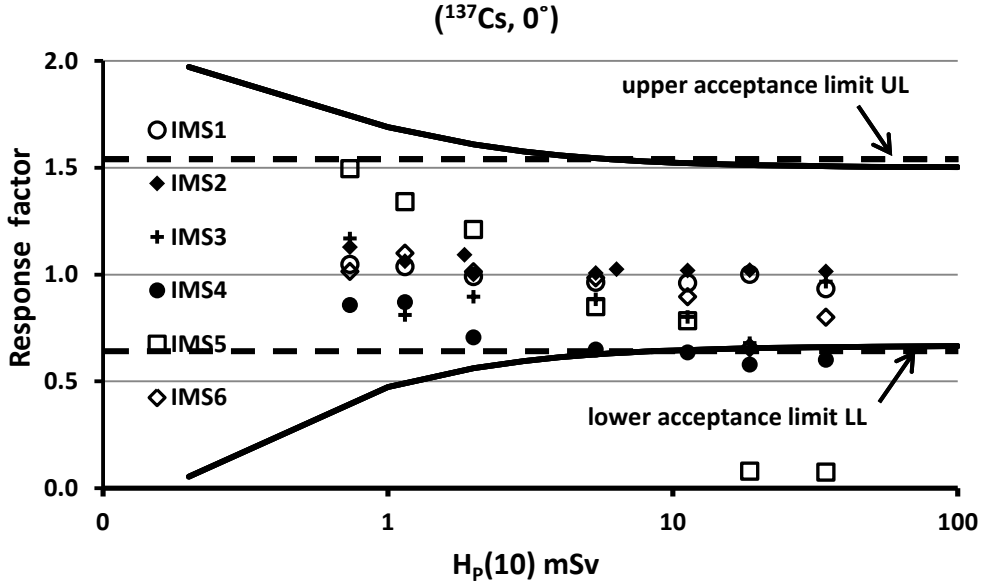

FIG. 1. Personal dosemeter response over a wide range of doses

For the assessment of measurement accuracy, which is the closeness of agreement between the IMS measured results and the conventional true values of the personal equivalent dose H_p (10). The overall measurement accuracy of H_p (10) depends generally on the dosemeter type, quality assurance and quality management systems applied by the dosimetry service. These determine the reliability and consistency of the applied measurement method as well as the characteristics of the used dosemeter.

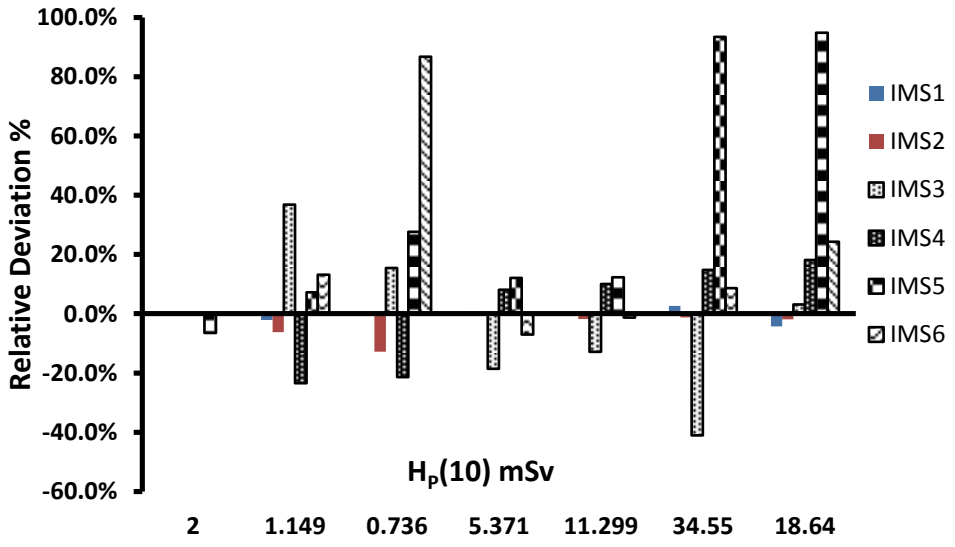

FIG. 2: Relative deviation for the studied dosimetry systems

In most studied samples, H_p (10) has provided a good estimate of the whole-body doses, see Fig. 2. The ICRP Publication 75, stated that: "In practice, it is usually possible to achieve an accuracy of about 10% at the 95% confidence level for measurements of radiation fields in good laboratory conditions. However, in the workplace, where the energy spectrum and orientation of the radiation field are generally not defined, the uncertainties in the measurement made with an individual dosimeter will be significantly greater" [3]. The overall uncertainty at the 95% confidence level in the estimation of equivalent dose around the lower dose limit may well be a factor of 1.5 in either direction for photon beam. The recording level has been interpreted as: allowing 100 % relative deviation near a true dose value equal to the lower recording level and 40% for higher range.

4. CONCLUSIONS

Regular verifications of the personal dosimetry systems are important procedures for keeping the service to the required acceptable standard. Single experimental test on six different IMS shows some outliers, which can be minimised by accurate calibration at competent standard dosimetry laboratory, and by applying good practices in this field. Regular performance checks are strongly recommended; as well as the participation in inter-comparison will help harmonising personal dosimetry services on the national and regional levels. Implementing quality assurance system, which is required for accreditation process and controlling significant parameters affecting the results are very important for improving of the dosimetry system selected for individual monitoring regardless of the type of detector in use.

REFERENCES

[1] INTERNATIONAL ELECTROTECHNICAL COMMISSION, Radiation protection instrumentation – Measurement of personal dose equivalents $H_p(10)$ and $H_p(0,07)$ for X, gamma, neutron and beta radiations – Direct reading personal dose equivalent meters and monitors, IEC 61526-02, Geneva (2005)

[2] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Recommendations of the International Commission on Radiological Protection, ICRP Publication 60, Ann ICRP 21, (1991).

[3] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, General principles for the radiation protection of workers, ICRP Publication 75, Pergamon Press, Oxford, (1997).

[4] R CRUZ SUÁREZ, H SCHUHMACHER, J L POCHAT, ANDREAS ZIMBAL, K MRABIT, , Inter-comparison on measurements of the quantity personal dose equivalent Hp(d) in mixed (neutron-gamma) fields--an IAEA project, Radiat. Prot. Dosim., 125 (1-4) (2007) 61.

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Inter-comparison of Measurements of personal dose equivalent Hp(10) in photon fields in the West Asia Region, TECDOC-CD-1567, IAEA, Vienna (2007).

[6] INTERNATIONAL ORGANISATION FOR STANDARISATION, X and gamma reference radiation for calibrating dosemeters and dose rate meters and for determining their response as a function of photon energy: Part 3: Calibration of area and personal dosimeters and the measurement of their response as a function of energy and angle of incidence, ISO 4037-3, Geneva (1999).

[7] INTERNATIONAL ORGANISATION FOR STANDARISATION, Radiation protection-criteria and performance limits for the periodic evaluation of processors of personal dosemeters for X and gamma radiation, ISO 14146, Geneva (2000)

[8] INTERNATIONAL COMMISSION ON RADIATION UNITS AND MEASUREMENTS, Conversion coefficients for use in radiological protection against external radiation, ICRU Report 57, Bethesda (1998).

PRELIMINARY INVESTIGATIONS OF AN OCCUPATIONAL ^{241}Am INCORPORATION EVENT

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Abstract

Aspects of an occupational ^{241}Am incorporation event happened in a radioactive waste treatment and storage facility, are presented. Three workers inhaled ^{241}Am radionuclide in kBq range of activity causing accident level of committed effective dose. The personal incorporations were assessed by whole body counting and urine analysis. Intensive survey was performed to determine the physical and chemical form of the contaminant and these results were utilized in planning the decorporation procedure and in the decontamination of the laboratory and other devices.

1. INTRODUCTION

Internal contamination of three workers was discovered in the course of a routine whole body counting measurement. It turned out that serious ^{241}Am incorporation event happened in December, 2013 at the Radioactive Waste Treatment and Disposal Facility (RWTDF at Püspökszilágy, Hungary). In the course of classification and processing of waste drums, three workers became contaminated by ^{241}Am both on their body surface and by inhalation.

The radioactive waste originated from a laboratory-scale research experiment performed few years ago. In the course of this experiment, ^{241}Am was electro-deposited on stainless steel plates. After finishing the project, the contaminated laboratory devices, filter papers used to sponge up the solutions and the deposited plates were handled as radioactive waste and packed in steel drums which were transported to the RWTDF in summer 2013.

The observed internal contamination is attributed to a waste classification procedure which is the initial step in routine waste processing at RWTDF. Further, airborne contamination was probably generated by the dispersion of compressible waste components, primarily pieces of dry filter paper. These were not observed during the process itself. A part of the airborne contamination was inhaled by the workers and also deposited to their body and surfaces in the laboratory.

2. RECOGNITION OF THE CONTAMINATION

The personal contaminations were observed in the Whole Body Counter (WBC) laboratory of the Centre for Energy Research of the Hungarian Academy of Sciences. The yearly routine screening involves whole body counting and tritium measurement from urine sample. Routine whole body counting is performed using “scanning end stop” geometry, which provides nearly homogeneous sensitivity along the whole body and the device is calibrated for uniform activity distribution. Since in case of ^{241}Am homogeneous distribution cannot be supposed, the activity can this way be determined only with large uncertainty.

Furthermore, the presence of suspected surface contamination could harshly influence the calculation of the intake so it was not estimated that time.

The employer institution of the workers was informed immediately about the result. They answered that the affected workers had handled ^{241}Am containing waste two days before their WBC measurement. Due to this rather short time between the assumed event and the measurement, not only internal but external contamination on body surfaces can also be present. Because of the high level of measured activity and the indicated uncertainties a repeated and more detailed measurement has been decided.

Surface contamination of ^{241}Am on the hands was measured by hand-held surface contamination monitor (up to 8 cps/cm^2), faces, hair, chest and back of the workers as well as on their personal belongings. The whole body counting in the scanning-end-stop geometry was repeated, but the count rate vs. position data were also recorded from which a rough longitudinal activity distribution could be plotted (Fig. 1). This indicates that the activity is concentrated in the chest region, probably in the lungs and the route of intake was most probably inhalation which was confirmed by the circumstances of the waste handling procedure reported by the workers.

The alpha-decaying ^{241}Am emits the characteristic X-rays of its daughter element Np in the 13-21 keV range and it has another gamma-line at 26.34 keV beside its 59.54 keV main line. Due to the significant self absorption of low energy photons in the human body, these lines cannot come out from the deeper layers of the body. Fig. 2 illustrates that the presence of low energy lines in the first spectrum corresponds to surface contamination, which disappeared already by the time of the second measurement.

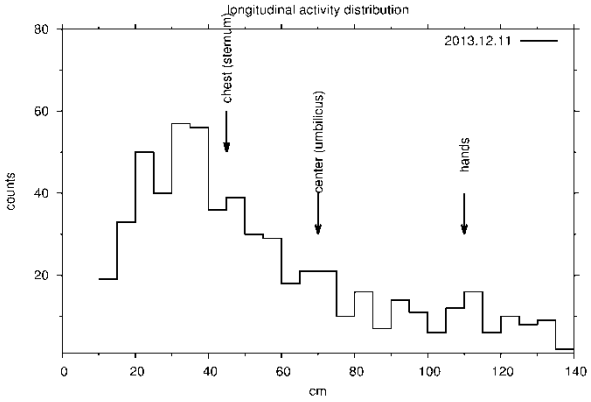


FIG. 1. Longitudinal count rate distribution obtained by whole body scanning

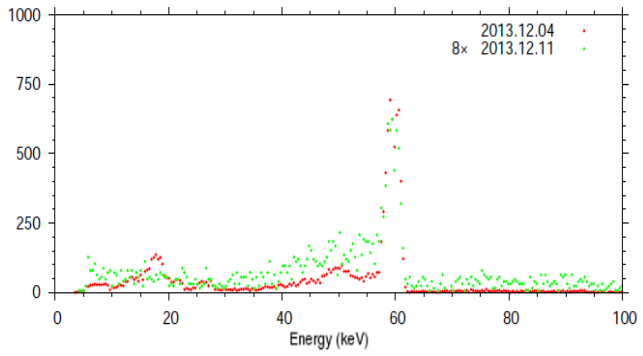


FIG. 2. Comparison of whole body spectra recorded 2 and 9 days after the contamination The two spectra were normalized to the same vertical maximum height

Because the scanning measurement indicated that the activity was basically located in the respiratory tract, direct lung measurements were performed with fixed detector. Since the device was not calibrated in this geometry for ^{241}Am earlier, the calibration was performed after the human measurements. A home-made MIXD chest phantom containing human skeleton and simulated lungs was used for calibration, in which ^{241}Am point source was located in different positions in the lung area and an averaged value was calculated.

Parallel to the whole body counting, urine samples were also collected. International guidelines recommend 24 hours urine collection, however, only momentary (spot) samples were available. The ^{241}Am content of the urine samples were below the sensitivity of gamma-spectrometry (~ 1 Bq/l), therefore the samples were analysed by alpha-spectrometry and ICP Mass Spectrometry. The results are in good agreement with each other. The results of the lung measurements and urine analyses are listed in Table 1.

TABLE 1. MEASURED ACTIVITIES IN THE LUNG REGION AND URINE-ACTIVITIES 9 DAYS AFTER THE CONTAMINATION

	Lung region	Urine – alpha-spectr.	Urine – ICP-MS
Person A	3.3 kBq	0.87 Bq/l	0.77 Bq/l
Person B	0.9 kBq	0.15 Bq/l	0.16 Bq/l
Person C	0.3 kBq	0.062 Bq/l	0.083 Bq/l

The dose consequences of the incorporation were calculated by using dose coefficients from ICRP publications [1] and the MONDAL3 [2] computer code was also used. According to these calculations the committed effective dose in the case of the most contaminated Person "A" is in the range of 1-2 Sv. The relatively large uncertainty is caused by several unknown – still significant – factors. The estimated committed effective dose from urine measurement considering also its uncertainty resulted in a value of around 1 Sv, which is in good agreement with that estimated by lung counting.

According to the recommendations of IDEAS guidelines [3] in case of the reported level of intake, the dose estimation should be improved by using individual biokinetic parameters pertaining to the contaminant. This requires detailed follow-up investigations for longer time period. For more accurate dose estimation some physical and chemical parameters of the incorporated radionuclide are also necessary: absorption type, particle size, etc. To obtain these parameters detailed physical and chemical characterisation of the contaminant is needed. Moreover, these properties are needed also for an efficient decontamination of the contaminated laboratory devices and personal objects.

According to the Hungarian legal regulations, in case of a radiological accident, the coordination of the measurements and the necessary countermeasures are the tasks of the "Frédéric Joliot-Curie" National Research Institute for Radiobiology and Radiohygiene (NRIRR – OSSKI) and therefore, the whole body and excreta measurements were continued at NRIRR. The physical and chemical characterisation of the contaminant was done by our institute.

3. PHYSICAL AND CHEMICAL CHARACTERISATION OF THE CONTAMINANT

Taking into consideration that the way of intake was inhalation the physical and chemical form of airborne Am at the RWTDF location should be determined. Since no direct air sampling device was operated in the laboratory, the necessary properties should be determined in a different way. The breathing masks worn by the workers during the work were available for analysis. The form of the contamination of different surfaces of the laboratory and on other objects was determined by the analysis of smear samples.

3.1 Solubility of the contaminant

As it was known, ^{241}Am was delivered originally in oxide form for the experiments. Taking into consideration the chemical procedures performed on the material there were 3 possible chemical forms of the Am dispersed in course of the waste processing:

(a) Electrolitically deposited Am on steel surface that reacted to form $Am(OH)_3$
(b) $Am(OH)_3$ transformed to Am_2O_3 by annealing
(c) Dry, partially mouldered filter paper waste containing $Am_2(SO_4)_3$ (ionic, soluble Am^{3+}).

In order to determine the chemical form(s) the electroplating procedure of the Am covered steel plates was re-enacted. Then dissolution experiments were performed using these plates and Am-sulphate dried onto filter paper surface for selecting an appropriate solvent of the chemical form(s) present on the surface. The tested solvents and the possible forms of Am are listed in Table 2.

TABLE 2: DISSOLUTION EXPERIMENTS WITH Am

Chemical form of Am	Applied solvent	
	0.01 M HCl	0.1 M HCl
Am_2O_3	insoluble	insoluble
$Am(OH)_3$	insoluble	soluble
Am^{3+}	soluble	soluble

Pieces of the breathing mask were processed by the described dissolution procedure. The activity of both the insoluble ^{241}Am fraction (remained in the sample) and the soluble fraction was measured by gamma-spectrometry. Solubility of the test materials was compared to that of smear samples taken in the contaminated workplace of RWTDF. According to the results the chemical form of the majority (>80%, depending on the recovery) of ^{241}Am was soluble Am^{3+}.

Corresponding to the determined chemical form of the contaminant, a decontamination procedure was also suggested for the contaminated surfaces of RWTDF. The optimal decontamination solvent contained 5% citric acid as complexing agent in 0.01 M HNO_3 solution.

3.2. Autoradiography by solid state track detectors

The aim of the autoradiography experiments performed on the breathing masks was to determine the type of particles, their distribution on the surface and – if it is possible – the size of the hot spots.

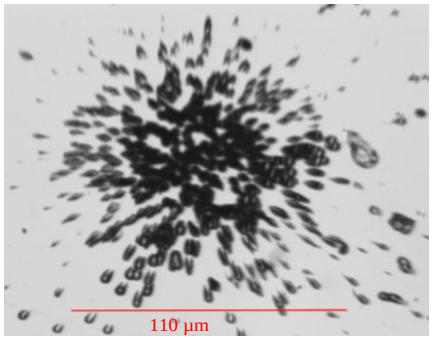


FIG. 3(a). Autoradiographic image of a characteristic section of a breathing mask

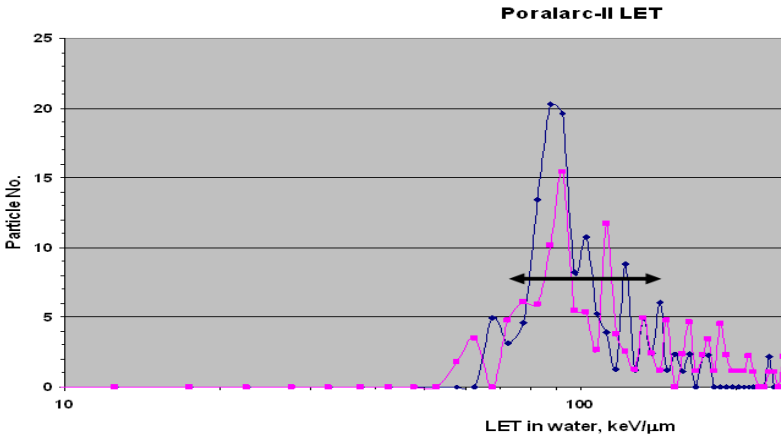


FIG. 3(b). LET distribution of alpha tracks of a characteristic section of a breathing mask

Track groups with different sizes containing up to thousand tracks and single tracks are both visible on the detectors (Fig. 3(a)). The Linear Energy Transfer (LET) distribution shown on Fig. 3(b) confirms that most of tracks must have come from ^{241}Am.

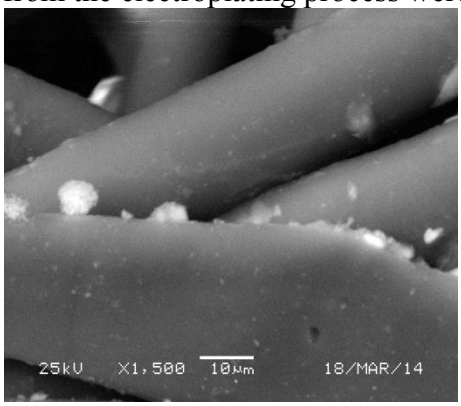
The number of tracks corresponds to the activity on the surface. This can be compared with the bulk activity of the sample measured by gamma spectrometry thus the surface/bulk ^{241}Am activity concentration ratio can be obtained. The estimated activity of one single hot spot was less than 0.2 Bq determined by the most exposed track detector. This means that Am was well dispersed in the inactive carriers and could not be found in the form of Am-oxide grains.

3.3. Scanning electron microscope experiments

The purpose of the scanning electron microscope (SEM) analysis was the assessment of the relation between the measured ^{241}Am activity of the breath mask and the captured grains on the filter texture. This assessment is very helpful in distinguishing between the two possible ways of origin of airborne contamination. Particulates of Am-oxide could have been generated by the dispersion of the steel plates covering or Am-activity could be attributed to mouldering of the filter paper that contained Am-cations from aqueous-acidic solution.

The SEM assessments were performed on the same slices of breathing masks on which the solid state track detector exposures were taken. ^{241}Am containing spots were located by consecutive masking of the surface followed by measurement by surface contamination monitor. The locations of selected parts of the mask were checked by optical microscope on the respective track detector, where massive track groups were visible. In a few selected locations dark grains were visible even by naked eye on the surface of the masks.

The texture of the mask and the attached grains were examined in secondary electron image mode. Then grains with higher average atomic number were searched in backscattered electron image mode for the identification of Am (Z=95). A set of selected grains were analysed by energy dispersive X-ray spectrometry. Due to the limits of the excitation and line overlaps with other elements the X-ray spectrometry analysis of Am is not so simple. All in all, Am was not detected in any of the selected grains. Taking into consideration that the detection limit of the EDX analyser is in the few percent range for the very high atomic number of elements, the Am content of the examined grains should be lower than this value so grains originating directly from the electroplating process were not identified.



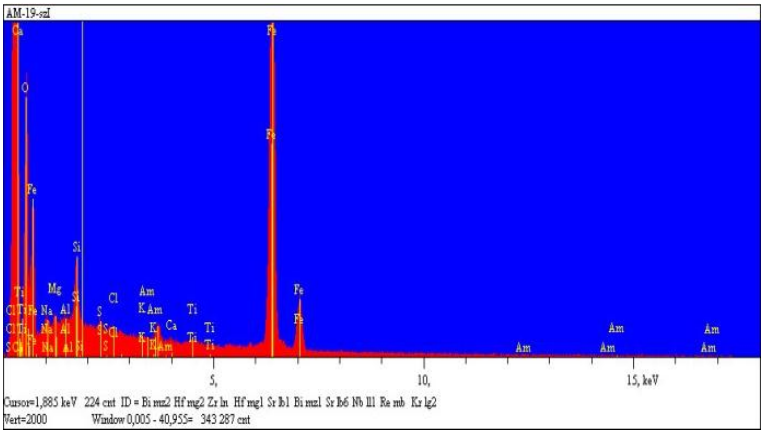


FIG. 4. SEM image of "suspected" grains on the breathing mask and the corresponding X-ray spectrum

4. RESULTS

After the adequate calibration of the whole body counter, the actual personal activities were measured, from which preliminary committed effective dose estimation was done based on default values of biokinetic and physical/chemical parameters of the contaminant. The survey on the physical and chemical characterisation of the contaminant focussed primarily on the chemical composition and particle size distribution of the contaminant. The chemical composition was unambiguously determined as highly soluble ionic form (Am^{3+}). Although the particle size distribution could not be determined quantitatively, from various approaches, a well-established assumption was made that the Am is dispersed in sub-micrometer size and adhered to inactive particulates.

5. CONCLUSION

The main goals of the physical and chemical characterisation of the contaminant were the determination of the chemical composition and the determination of the particle size distribution. These characteristics – together with appropriate calibration of the direct monitoring systems – are the necessary tools to determine the individual biokinetic parameters, which are fundamental for more accurate estimation of dose consequences and then also for the assignment of the appropriate decorporation technique, and the decontamination of different contaminated objects related to this event.

ACKNOWLEDGEMENTS

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REFERENCES

[1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Compendium of dose coefficients based on ICRP Publication 60, ICRP Publication 119 (2012).
[2] MONDAL 3, NOBUHITO ISHIGURE, MASAKI MATSUMOTO, TAKASHI NAKANO, HIROKO ENOMOTO, Support System for Internal Dosimetry, NIRS, Japan.
[3] H. DOERFEL, A. ANDRASI, M. BAILEY, E. BLANCHARDON, R. CRUZ-SUAREZ, V. BERKOVSKI, C.-M. CASTELLANI, C. HURTGEN, B. LEGUEN, I. MALATOVA, J. MARSH, J. STATHER, AND J. ZEGER, General guidelines for the assessment of internal dose from monitoring data: progress of the IDEAS project, Radiat. Prot. Dosim. 125 (1-4) (2007) 19-22.

OVERVIEW OF THE ACTIVITIES ON OCCUPATIONAL DOSIMETRY WITHIN EURADOS WG 12, DOSIMETRY IN MEDICAL IMAGING

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Abstract

The European Radiation Dosimetry Group, EURADOS is a network of scientists specialized in the field of dosimetry. One of the main topics of EURADOS working group 12 (WG 12) is dosimetry of medical staff, in particular eye lens dosimetry. There is an increased interest amongst the scientific community in this topic for occupationally exposed medical workers due to the new eye lens dose limit and there are many open questions and topics which need to be addressed within this field. An overview of the main tasks undertaken by the working group is presented: a literature review on eye lens dose data and on available procedures for estimating or measuring eye lens doses, the distribution of a questionnaire to European hospitals to collect feedback about the current status of eye lens monitoring in hospitals and the organization of an intercomparison between European individual dosimetry services to verify the performance of available eye lens dosemeters.

1. INTRODUCTION

The European Radiation Dosimetry Group, EURADOS (wwww.eurados.org) is a network of more than 60 European institutions and 300 scientists specialized in the field of dosimetry. The aim of the network is to promote research and development and European cooperation in this field. Its activities are developed through several Working groups (WG) which focus on different topics of interest. In particular, EURADOS WG12 deals with dosimetry in medical imaging and more specifically, among other topics, it is interested in eye lens dosimetry for medical workers. Most of the work of EURADOS WG12 is under progress at the moment.This paper presents the main work undertaken by the Working group in the last two years.

The new ICRP recommendation (1) of reducing the eye lens dose limit for workers from 150 mSv to 20 mSv per year averaged over defined periods of 5 years, with no single year exceeding 50 mSv, has been adopted by the IAEA (IAEA Basic Safety Standards(2) and also by the European Commission (EURATOM Directive(3)). However, eye lens monitoring is not well established, there are only a few dosemeters developed in the market and there is a lack of data of eye lens doses at various workplaces. Moreover, several studies (4,5,6,7) have highlighted situations where the new limit could be exceeded and thus additional radiation protection tools are required. Within this framework, EURADOS WG12 has initiated 3 main tasks related to occupational eye lens dosimetry:

(1) A literature review on eye lens dose data and on available procedures for estimating or measuring eye lens doses.
(2) The distribution of a questionnaire to European hospitals to collect feedback about the current status of eye lens monitoring in hospitals.
(3) The organization of an intercomparison between European individual dosimetry services which provide eye lens monitoring.

2. METHODS

The implication of the new eye lens dose limit is of great concern, especially regarding the radiation protection of workers in interventional radiology and interventional cardiology (IR/IC). To identify the areas where major developments are needed a thorough literature review has been undertaken. 118 papers published from 2006 and 2013 are analysed. The main aim of this is to describe the current situation on the implementation of the eye lens monitoring in the medical field related to the eye lens dose assessment, the dosemeters used and their calibration. The ultimate scope is to provide the scientific community with some recommendations on the hot issues of eye lens monitoring, calibration and eye lens dosimetry.

A questionnaire regarding the knowledge on the proposed eye lens dose limit, monitoring and dosimetry issues and training and radiation protection means in IR/IC and nuclear medicine departments was prepared and sent to radiation protection officers and medical physicists in European hospitals. For this task the main aim is to map the current status of eye lens radiation dose monitoring in hospitals around Europe.

An intercomparison to test and compare different eye lens dosemeters has been organized in 2014. Twenty services were invited to participate. Irradiations were performed at three secondary standard calibration laboratories: IRSN (France), SCK (Belgium), UPC (Spain) and the French National Laboratory LMRI (France). Reference photon radiation beams together with typical scattered beams found in radiology are used. The operational quantity Hp(3) is used in the intercomparison.

3. RESULTS AND DISCUSSION

3.1 Literature review

The literature review has highlighted an increased interest of researchers on eye lens dosimetry, especially since the publication of ICRP recommendation (1) with the reduction of eye lens dose limit. The 118 analysed papers have been classified in 7 topics which correspond to the main concerns in the field. Figure 1 shows the distribution of topics, almost 40 % are related to clinical issues. The main questions raised in the papers are: how to be consistent with the new eye dose limit and what is the monitoring programme that should be developed especially in IR/IC workplaces. Another critical point is the calibration methodologies of the newly developed eye lens dosemeters (phantoms, quantities and conversion coefficients). Harmonisation on these issues and practical reccommendations for all the relevant stakeholders are needed.

The literature review has highlighted, that in spite of the large number of recent studies on eye lens dosimetry (8), the comparison of results is often difficult because measurement protocols are not the same. In addition because of the recent increase in the use of fluoroscopy guided procedures performed in a wide range of medical areas both for diagnostic or therapeutic purposes (9), there is still a lack of realistic data for some specific procedures. To overcome this problem, a list of clinical areas and procedures where additional information is required and a common protocol has been prepared to be used by all EURADOS WG12 partners. The protocol provides guidance about the position of the dosemeters and the number of procedures or Kerma Area Product (KAP) levels needed to obtain an appropriate signal in the eye dosemeter. Together with the eye lens dose measurement, the protocol aims at collecting other parameters such as the whole body dose at the collar level (outside the lead protection) and the KAP.

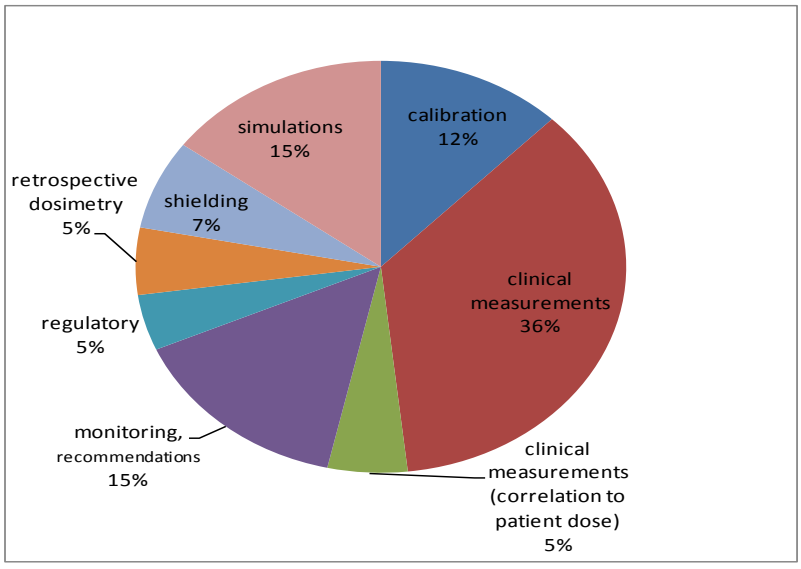

FIG. 1. Distribution of topics of the various papers on eye lens issues

3.2 Eye lens questionnaire

One hundred and ninety-five (195) responses from 23 European countries were received from the questionnaire distributed from November 2013 to February 2014 among medical physicists and radiation protection officers in European hospitals. Most of the responses

(93%) stated that they were familiar about the change in the eye lens limit for the occupationally exposed personnel and about half of them had already started some specific studies in monitoring practice. Although most of the hospitals are not performing eye lens monitoring regularly, this practice is starting to be introduced for some IR/IC services (25% of the answers). In Table 1 it is shown the number of responses per country for IR/IC workplaces, who replied that they know about the reduction of the eye lens limit. The two last columns show the number of hospitals where eye lens monitoring is performed within the framework of specific studies and those who have never performed eye lens monitoring. The analysis of the answers to the questionnaire has been published by Carinou et al. (10).

TABLE 1. NUMBER OF ANSWERS PER COUNTRY FOR THE IR/IC WORKPLACES WHERE SPECIFIC MEASUREMENT HAVE BEEN PERFORMED OR NEVER HAVE BEEN PERFORMED

Country	Number of answers for IR/IC	Number of answers who replied positively for the new eye lens limit	Number of answers where test measurements for eye lens doses have been performed	Number of answers where measurements have never been performed
Austria	3		3	1
Belgium	6	4	6	
Croatia	5		5	2
Cyprus	1	1	1	
Czech Republic	5	1	5	3
Finland	4	2	4	1
France	12	9	12	2
Germany	30	6	30	16
Greece	15	4	15	4
Hungary	1	1	1	
Ireland	15	11	15	2
Italy	18	12	18	
Luxembourg	1	1	1	
Norway	8	1	8	3
Poland	8	6	8	4
Portugal	5	1	5	
Serbia	3	2	3	
Slovakia	3	2	3	
Spain	13	9	13	3
Sweden	8	7	8	
Switzerland	5	2	5	2
UK	21	16	21	
Ukraine	1	1	1	

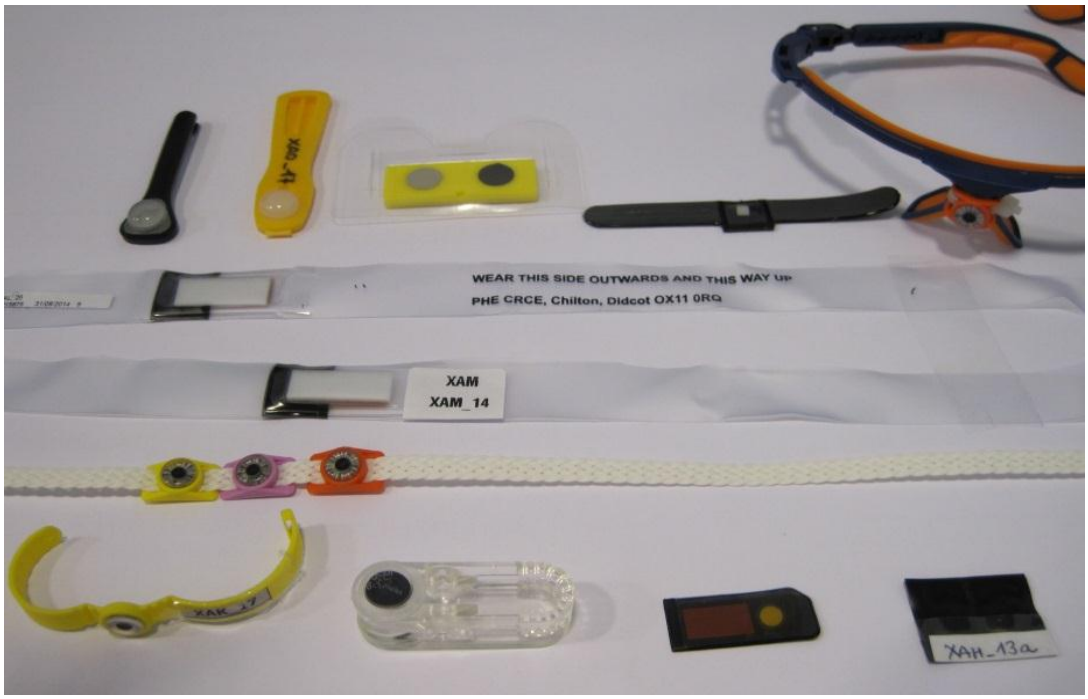

FIG. 2. Different types of eye lens dosemeters tested in the intercomparison

3.3 Intercomparison for eye lens dosemeters

Fig. 2 shows the different types of dosemeters that have been used in the intercomparison for eye lens dosemeters. Nine services are using the EYE D dosemeter whereas the other services use some other designs which usually are extremity dosemeters in the adapted holders. The irradiations are being performed in July-August and the results will be available in December.

4. CONCLUSIONS

EURADOS is a unique scientific community for the promotion and co-ordination of research activities in several fields. In particular, the work initiated by EURADOS WG12 for the improvement of eye lens dose measurements of medical staff is crucial. The fact that the group consists of researchers from universities, research centers, hospitals, metrology labs, technical services and regulatory bodies, with experience in radiation protection of medical staff and in dosimetry is useful to find global and practical solutions to the issues identified.

The key element is to give answers and draft recommendations for the tasks described above. This will be of great help towards a harmonized implementation of eye lens dose monitoring and for improving radiation protection of workers.

REFERENCES

[1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, ICRP Statement on tissue reactions/early and late effects of radiation in normal tissues and organs—Threshold doses for tissue reactions in a radiation protection context, ICRP Publication 118; Ann. ICRP 41 (1/2), (2012).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety of radiation sources: International basic safety standards, IAEA Safety Standards Series No. GSR Part 3 (Interim), IAEA, Vienna (2011).

[3] EUROPEAN COMMISSION, Council Directive 2013/59/EURATOM of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionizing radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom Official Journal, 13 (2014) 1-73.

[4] VANHAVERE et al., Measurements of eye lens doses in interventional radiology and cardiology: Final results of the ORAMED project, Radiation Measurements 46 (2011) 1243-1247.

[5] JACOB, S, et al., Interventional cardiologists and risk of radiation-induced cataract: Results of a French multicenter observational study, Int. J. Cardiol. doi:10.1016/j.ijcard.2012.04.124 (2012)

[6] REHANI M., et al., Radiation and Cataract, Radiat. Prot. Dosim. Vol. 147 (1–2) (2011) 300–304.

[7] VANO E, et al., Radiation-associated lens opacities in catheterization personnel: results of a survey and direct assessments, J. Vasc Interv Radiol., 24(2) (2013) 197-204.

[8] CIRAJ-BJELAC, O, REHANI M.M., Eye dosimetry in interventional radiology and cardiology: Current challenges and practical considerations, Radiat. Prot. Dosim. Nov 20. [Epub ahead of print] (2013).

[9] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Radiological protection in fluoroscopically guided procedures outside the Imaging Department, ICRP Publication 117, Ann. ICRP 40(6) (2010).

[10] CARINOU et al., Status of eye lens radiation dose monitoring in European hospitals, submitted to Journal of Radiological Protection (2014).

DEVELOPMENT OF A STANDARD FOR THE MONITORING AND INTERNAL DOSIMETRY OF EXPOSED WORKERS OF NUCLEAR MEDICINE

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Abstract

A new ISO standard is under development to provide guidance for the practical application to the staff involved in the diagnostic or therapeutic use of radionuclides in medicine of the three ISO standards dedicated to internal contamination. It takes into account the special aspects in nuclear medicine resulting from the short effective half-times of the nuclides in use and the distances between department of nuclear medicine and whole body and thyroid counting facilities or laboratories undertaking spectrometry on urine samples. This presentation enhanced two important points addressed in this future standard: the need for a monitoring programme and the design of the different proposed monitoring programmes**.**

1. INTRODUCTION

In the course of employment, individuals might work with radioactive materials that, under certain circumstances, could be taken into the body. Protecting workers against risks of incorporated radionuclides requires the monitoring of potential intakes and/or the quantification of actual intakes and exposures. For these reasons, three ISO standards for the monitoring programmes (20553:2006) [1], for the laboratory requirements (28218:2010) [2], and for the dose assessment (27048:2011) [3] have been developed and can be applied in a straightforward manner to many workplaces where internal contamination may occur. However, their application for the staff involved in the diagnostic or therapeutic use of radionuclides in medicine requires account to be taken of special aspects resulting from the short effective half-times of the nuclides in use and from the distances between department of nuclear medicine and whole body and thyroid counting facilities or laboratories undertaking spectrometry on urine samples. Consequently, a new ISO standard (ISO/DIS 16637 - Radiological protection — Monitoring and internal dosimetry for staff exposed to medical radionuclides as unsealed sources) is under development by the working group 13 (TC85/SC2) to provide guidance for the practical application of the three standards cited above to the nuclear medicine staff.

2. PURPOSE AND NEED FOR MONITORING PROGRAMMES IN NUCLEAR MEDICAL DIAGNOSIS AND THERAPY

The purpose of monitoring, in general, is to verify and document that the worker is protected adequately against risks from radionuclide intakes. This protection must comply with legal requirements. Therefore, it is part of the overall radiation protection programme,

which should starts with an assessment to identify work situations showing a risk of radionuclide intake by workers, and to quantify the annual likely intake of radioactive material and the resulting committed effective dose. Decisions about the need for monitoring and the design of the monitoring programme should be made in the light of such a risk assessment. A monitoring programme for internal contamination is required if the worker is occupationally exposed and the assessed dose contribution from intakes of radionuclides is likely to be significant. The recommended level of the likely annual committed effective dose to initiate monitoring is 1 mSv.

3. MONITORING PROGRAMMES

3.1. General

Individual monitoring gives information on the exposure of a single worker by measuring individual body activities, excretion rates or activity inhaled (using personal air samplers). Workplace monitoring, either by air monitoring or by measurements of the surface contamination, helps to assess the internal exposure of workers through inhalation and provide information on the risk of contamination for setting up individual monitoring programmes for workers. In nuclear medicine, workers can be contaminated by inhalation of volatile compounds (mainly radioiodine), or aerosols. As a result, individual monitoring for internal contamination may be necessary for those workers who regularly work with large activities of volatile radioactive materials [4].

3.2. Categories of monitoring programme

3.2.1. Confirmatory monitoring programmes

Confirmatory monitoring, which consists of workplace and/or individual monitoring performed at regular intervals (by example every month for workplace measurements or every six months for individual measurements) enables to check the assumptions about exposure conditions underlying the procedures selected, e.g. the effectiveness of protection measures.

3.2.2. Triage monitoring programmes

Triage monitoring programmes rely on frequent individual screening measurements performed at the workplace to the whole staff at risk to detect whether potential intake has occurred. If the screening threshold is exceeded, in vivo or in vitro radiobioassays are performed in order to confirm internal contamination and to quantify the incorporated activity for dose assessment.

3.2.3 . Routine monitoring programmes

Routine monitoring programmes are performed to quantify exposures where there is the possibility either of undetected accidental intakes or of chronic intakes.

In nuclear medicine, routine monitoring based on individual measurements can be performed to monitor the risk of iodine 131 inhalation when significant activities in volatile forms are manipulated. The recommended method is *in vivo* thyroid measurement with a maximum time interval of 15 days between two measurements. When thyroid measurements cannot be performed, an alternative is to precede urine *in vitro* analyses with the same maximum time interval i.e.15 days.

3.2.4. Special monitoring programmes

Special monitoring programmes are performed to quantify significant exposures following actual or suspected abnormal events (by example the spill of a radiopharmeutical solution) or in case of a positive screening during triage monitoring. The purposes of dose assessment in such cases include assisting in decisions about countermeasures (e.g. decorporation therapy), compliance with legal regulations and aiding decisions for the improvement of conditions at the workplace. Table I summarizes recommended methods for special individual monitoring.

TABLE 1. RECOMMENDED METHODS FOR SPECIAL MONITORING PROGRAMMES AFTER INHALATION

Radionuclide	Urine *in vitro* analyses		*In vivo* measurements	
	Spot sample	24 h	WB	Thyroid
F-18	+		++	
Ga-67	+		++	
Sr-89		++		
Y-90		++		
Tc-99m		+	++	
In-111			++	
I-123		+		++
I-131		+		++
Sm-153		+	++	
Er-169		++	+	
Lu-177		+	++	
Re-186		+	++	
Re-188		+	++	
Tl-201		+	++	
Ra-223		++		

++ = Recommended, += Supplementary (helpful but not mandatory) WB =Whole Body

3.2.5. Task-related monitoring programmes

Task-related monitoring programmes apply to a specific operation. The purpose and the dose criteria for carrying out task-related monitoring programmes are identical to those for routine monitoring programmes.

4. CONCLUSION

The ISO standard under development for the monitoring and internal dosimetry of exposed workers of nuclear medicine provides guidance for the decision whether a monitoring is required and how it should be designed. The DIS was approved in May 2014 and the FDIS is in preparation for a ballot in 2015. The programmes proposed for the

detection of potential intakes of radionuclides by nuclear medicine workers enable an acceptable monitoring while taking into account practical and economic considerations.

ACKNOWLEDGEMENTS

The authors wish to thank the members of ISO TC85/SC2 WG13 (Internal dosimetry) who contributed to this work**.**

REFERENCES

ISO 20553:2006, Radiation protection—Monitoring of workers occupationally exposed to a risk of internal contamination with radioactive material (2006).

ISO 28218:2010, Radiation protection—Performance criteria for radiobioassay (2010).

ISO 27048:2011, Radiation protection—Dose assessment for the monitoring of workers for internal radiation exposure (2011).

INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION (ICRP), Radiological protection of the worker in medicine and dentistry, ICRP Report 57. Pergamon Press, New York (1989).

UPGRADING THE NATIONAL RADIATION STANDARDS FOR PROTECTION LEVEL CALIBRATION AT THE SECONDARY STANDARDS DOSIMETRY LABORATORY (SSDL) IN THE PHILIPPINES

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Abstract

The programmes for establishing the national radiation standards for diagnostic & protection Level at the Secondary Standards Dosimetry Laboratory (SSDL) in the Philippines are presented. Improved standards, new equipment & systems upgrades are discussed. With the upgrades, the SSDL-PNRI is geared towards a strengthened capacity in radiation dosimetry and metrology and thereby ensuring safety of the users of ionzing radiation.

I. INTRODUCTION

The national standards for ionizing radiation are established and maintained by Secondary Standards Dosimetry Laboratories (SSDL). The SSDLs provide a means to ensure that radiation exposures and measurements are consistent, accurate and traceable to primary radiation standards. In the Philippines, the SSDL is operated by Philippine Nuclear Research Institute (SSDL-PNRI) & the Center for Device Regulation, Radiation Health & Research (CDRRHR-DOH). The SSDL-PNRI maintains the national standard for protection level qualities while the SSDL of the CDRRHR-DOH maintains the therapy level radiation standards.

With the country's continuing efforts to continuously improve its metrology standards, under its National Metrology Act 2003, the SSDL-PNRI has embarked on several projects to upgrade its national standards for protection level calibrations. Two 2 irradiation bunkers were constructed for the Cs-137 and the 225 kVp / 30 mA Constant Potential X-ray (CPX) Unit. Through an IAEA Technical Cooperation Project, personnel training, expert assistance and acquisition of reference instruments were achieved.

In this paper, we present the upgrades of the SSDL-PNRI in addressing the need to improve the national radiation standards. Standardization of high and low dose rate Cs-137 radiation quality is discussed. Establishment of the narrow spectrum series X-ray radiation quality is also shown. In addition, upgrades on safety, security and calibration alignment systems are also presented.

2. IMPROVING THE PROTECTION LEVEL RADIATION STANDARDS

Following the IAEA TC and other national grants-in-aid projects, new reference dosimeters were acquired and calibrated. A higher activity ^{137}Cs source, 111 GBq as of December 1975, was installed in addition to the 17.4 GBq (March 1979) source. To perform the standardization measurements, an NE 2575 600 cc cylindrical ionization chamber traceable to the IAEA-SSDL was used. The calibration coefficients of the reference dosimeter cover the protection level radiation qualities S-Cs, S-Co, and the X-ray Narrow Spectrum Series (ISO 4037). A PTW LS-01 32002 working instrument was also available for the

measurements. Fig. 1 below shows the set-up for the standardization measurements. Air kerma output measurements of the sources were conducted. To achieve a low kerma-rate output, i.e. less than 20 uGy/h, a lead absorber is used.

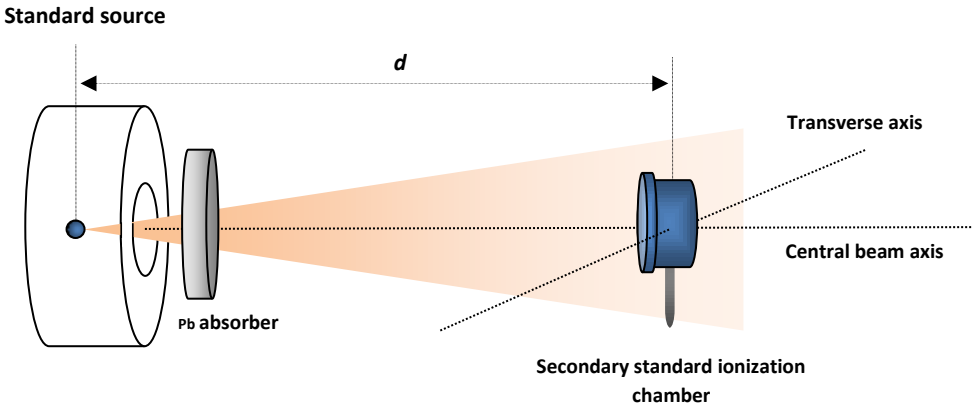

FIG. 1. Set-up for the standardization measurement of source output

The quantity used in the calibration of radiation monitoring instruments is in terms of equivalent dose. Before, the standard output used is limited to the range 50 - 800 µSv/h only. Table 1 below shows the standard output of the two ^{137}Cs sources, after new standardization measurements were conducted. This standard is now used for calibration. It now allows the calibration for dose rates upto 5 mSv/h. This range, as prescribed by the Philippine Regulations, is suitable for monitoring in high exposure practices such as industrial radiography and radiotherapy. The high dose rate source also made the calibration of personal dosimeters much faster. Dose rates upto 10 µSv/h were also standardized to enable calibration of instruments used in low-dose rate measurements.

Recently, the PNRI-SSDL participated to the IAEA TLD Postal Audit for radiation protection calibrations held last 2013. Results of the audit show that air kerma output of the PNRI-SSDL is within the acceptable criteria.

TABLE 1. STANDARD DOSE RATE OUTPUT OF THE ^{137}Cs SOURCES USED IN THE CALIBRATION OF RADIATION MONITORING INSTRUMENTS

	Distance (cm)	Dose Rate (uSv/h)	Collimator	Absorber
Cs-137 (JL Shepherd (NPC))	89.4	5000	1	No Pb Absorber
	110.0	3000	1	
	125.6	2100	1	
	175.1	1000	2	
Cs-137 (NEN C5-492)	89.4	800	1	
	95.4	700		
	102.6	600		
	125.0	400		
	175.0	200	2	
	249.6	100		
	102.6	60	1	With Pb Absorber
	126.6	40		
	171.0	20	2	
	236.0	10	1	

To establish the narrow spectrum radiation qualities for low energy calibrations, the setup is depicted below. Initial measurements show that for the N-80 radiation quality, the half-value layer measurement is within the prescribed ISO4037 standard. Table 2 shows the result. Details of other qualities will be discussed.

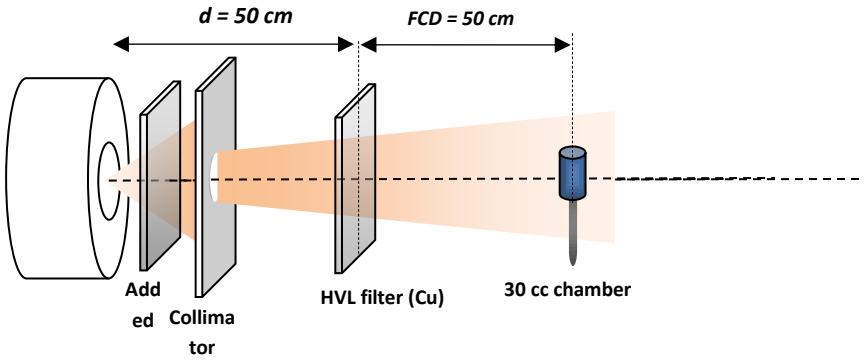

FIG. 2. Setup for the establishment of narrow spectrum protection level radiation quality

TABLE 2. HVL OF THE N-80 RADIATION QUALITY AT SSDL

Beam Quality	N-80
Nominal kV setting	80 kV
Current Rating	20 mA
FCD / Exposure time	100 cm / 20s
Added Filter	2 mm Cu + 4 mm Al
Air Kerma at 200cm FCD	214.7 μGy/min
HVL_{SSDL}	0.59 mm Cu
$HVL_{ISO4037}$	0.58 mm Cu
% difference	-1.5%

3. INFRASTRUCTURE UPGRADES

As a whole, in radiation dosimetry and metrology, several systems upgrades were put in place. The safety systems of the irradiation bunkers have been upgrade to ensure the safety of personnel during work. Area monitors have been installed in each bunker to measure radiation levels inside during and after exposure. Shielded housing of the higher activity Cs-137 source has been reinforced. A remote shutter system for the source was also put in place. Cameras were installed to allow the remote viewing of instruments during calibration. A laser system is now used in aligning the instrument along the central beam axis whereas the calibration bench system has been modified. Reference instruments, which are mostly ionization detectors, are stored in a dry cabinet with controllable temperature and humidity levels to help maintain its quality.

A quality management system is also now in place for the whole Institute ensuring further, the overall quality and personnel competence of the facilities.

4. SUMMARY

The programs for establishing the national radiation standards for diagnostic & protection Level at the Secondary Standards Dosimetry Laboratory (SSDL) in the Philippines

are presented. Ongoing projects, training programs, new equipment & infrastructure and initial results are discussed. Based on initial measurements results, the X-ray tube output and the radiation quality of the irradiation sources are within the international standards. With these steps, the SSDL-PNRI is geared towards a strengthened capacity in radiation dosimetry and metrology and thereby safety of the users of ionizing radiation.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Calibration of Radiation Protection Monitoring Instruments, Safety Report Series No. 16, IAEA, Vienna (2000).

INTERNAL CONTAMINATION MONITORING FOR WORKERS FROM NUCLEAR FACILITIES IN IFIN-HH ROMANIA – CURRENT AND FUTURE PRACTICES

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Abstract

Internal contamination, routine and special measurements and dose evaluation are performed by the Whole Body Monitoring Laboratory (WBML) for workers from Horia Hulubei National Institute of Physics and Nuclear Engineering (IFIN-HH) large scale facilities involved in nuclear activities with potential risk of internal exposure due to gamma and beta-emitting radionuclides. WBML has updated its Quality Assurance system in accordance with the development of the IFIN-HH infrastructure for frontier research in nuclear physics and related fields. It is the only national laboratory, notified and nominated by the national competent authority in the field, the National Commission for Nuclear Activities Control (CNCAN), as Individual Dosimetry Body for in vivo whole body and thyroid measurements, in vitro monitoring of excretion samples and internal dose assessment. Detailed statistics of the measurement results in the period 2000-2013 and trends of internal exposure monitoring in IFIN-HH are presented.

1. INTRODUCTION

The Romanian legislation into force for monitoring the radiation occupational exposure of workers is represented by the document NSR-01: "Fundamental norms of radiation safety"[1], where there are expressed the general demands to assuring the radioprotection for the exposed workers, public and environment, and by the more specific document, NSR-06:"Norms of occupational personal dosimetry"[2]. The documents were issued by CNCAN, in accordance with the guidelines in the field of the International Commission of Radiation Protection (ICRP) and International Atomic Energy Agency (IAEA).

In order to perform radioactive internal contamination monitoring, in notified conditions, the Whole Body Monitoring Laboratory from IFIN-HH has implemented since the year 2000, its own Quality Assurance system [3], based on CNCAN legislation, developing an annual monitoring program for workers involved in nuclear activities from IFIN-HH long-term large scale facilities as: Radioisotope Production Center, Nuclear Waste Processing Center, National Radioactive Waste Repository, Multipurpose Irradiation Facility, U120 Cyclotron, VVRS Reactor (under decommissioning). New research units from IFIN-HH as Tritium Laboratory, Radiopharmaceutical Research Center and Carbon Laboratory (to be authorized) are necessitating the development of internal contamination monitoring programmes – both in vivo and in vitro measurements.

2. MATERIALS AND METHODS

The WBML equipment for *in vivo* internal contamination monitoring, approved by CNCAN, consist in two Body Counters, for whole body and thyroid measurements.

The Whole Body Counters are gamma spectrometric systems based on shadow shield chair geometries, equipped with lead shielded detectors of NaI(Tl) scintillation crystal (120 mm diameter and 100mm length) and of Carbon window HPGe detector (85mm diameter and

32mm length). These are able to detect gamma rays of incorporated radionuclides, in the energy range tens of keV to 2500 keV, with excellent resolutions and good efficiencies.

The Thyroid Counters are, also, gamma spectrometric systems, equipped with lead shielded NaI(Tl) scintillation detector of 40mm/50mm diameter, and 50mm thickness, being used for the detection of ^{131}I, in thyroid. Their efficiency calibrations were performed with a BOMAB phantom and a Plexiglas thyroid phantom, simulating the standard adult whole body and thyroid anatomical shapes and volumes, filled with certified radioactive solutions of known activity of ^{60}Co, ^{65}Zn, ^{137}Cs, ^{152}Eu and ^{131}I, respectively. The associated electronics of detectors consists on state-of-art analog and digital ORTEC equipments and for spectra acquisition is used the dedicated software ORTEC Renaissance-32.

The in vitro measurements of tritium in urine were performed using a beta Analyser, aTri-Carb 1600 TR model, of Packard Company dedicated for high performance liquid scintillation counting, using Insta Gel Plus scintillation cocktail.

3. RESULTS AND DISCUSSION

The radionuclides identified during the measurement of the subjects were ^{22}Na, ^{60}Co, ^{99m}Tc, ^{65}Zn, ^{58}Ga, ^{131}I and ^{192}Ir, specific for every type of nuclear activity, as follows:

(a) Production of radiopharmaceuticals for nuclear medicine purpose: ^{131}I and ^{99m}Tc
(b) Production of sealed sources for industry : ^{60}Co and ^{192}Ir
(c) Cyclotron maintenance : ^{65}Zn
(d) Research activities : ^{22}Na, ^{58}Ga, ^{3}H

There were estimated the committed effective doses received by the workers and registered them in a database according to the dose ranges established by CNCAN, for annual statistical reporting. The data were represented in Figs.1- 4 being considered the collective doses for in vivo thyroid and whole body monitoring and the associated cumulative values of the contaminated workers, in the interval 2000-2013. The dose ranges from their associated legends are expressed in mSv.

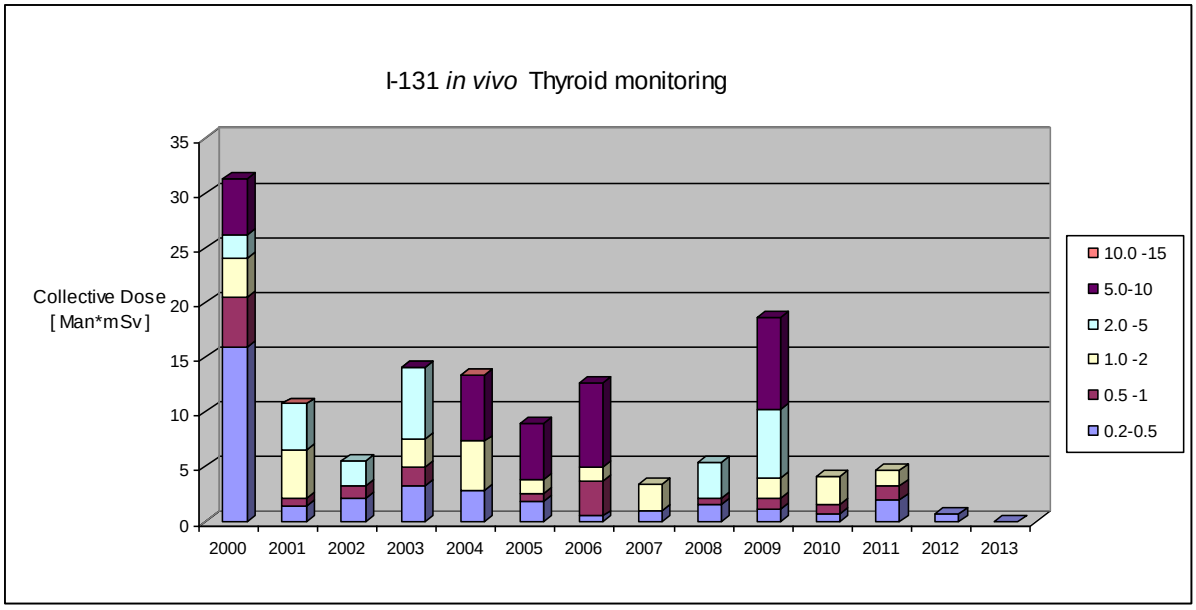

FIG.1. In vivo thyroid monitoring - The Collective Dose

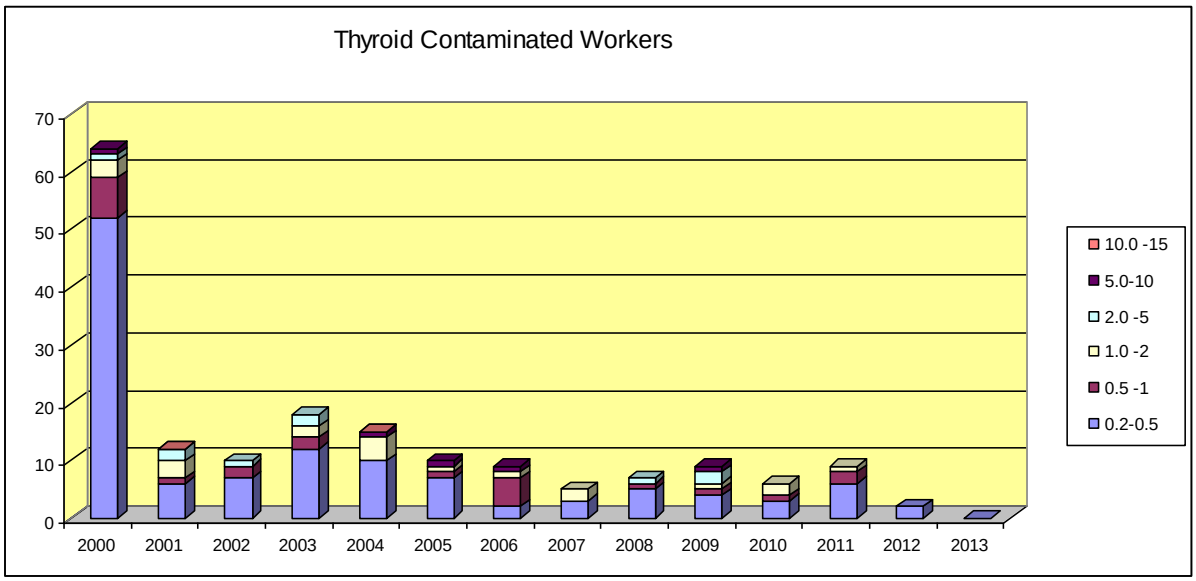

FIG. 2. In vivo thyroid monitoring - The annual cumulative number of ^{131}I contaminated workers grouped into dose ranges (mSv)

All thyroid measurements have detected internal contamination with ^{131}I, due to multiple intakes through inhalation occurred during regular production of radiopharmaceuticals. The improvement of radioprotection during the time had a direct impact on the decrease of occupational doses received by the workers.

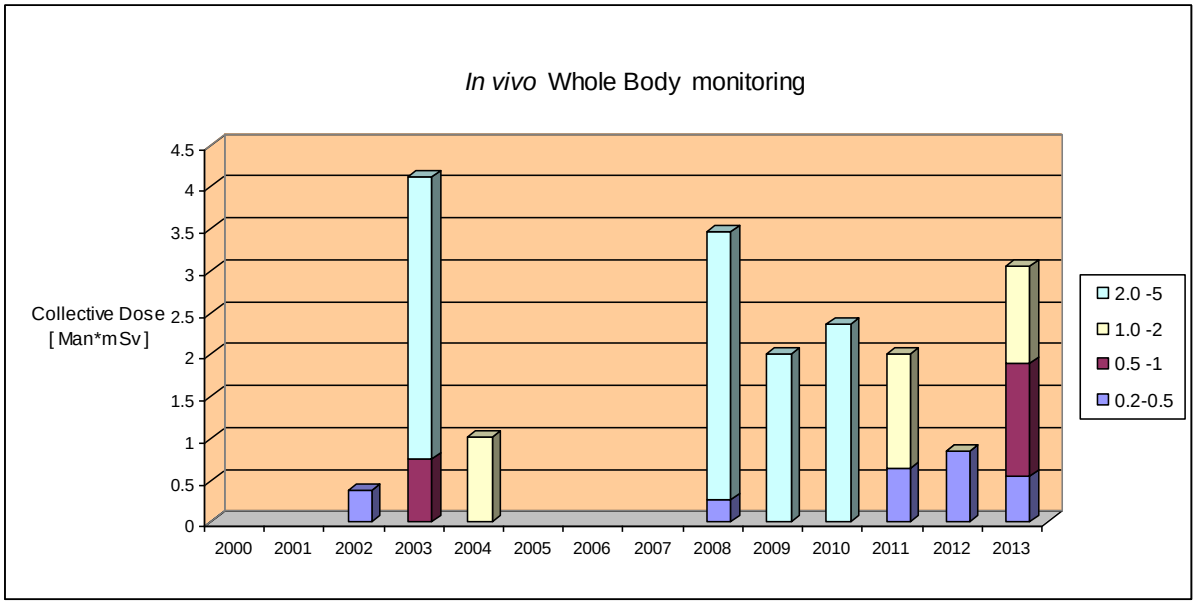

FIG. 3. In vivo whole body monitoring - The Collective Dose

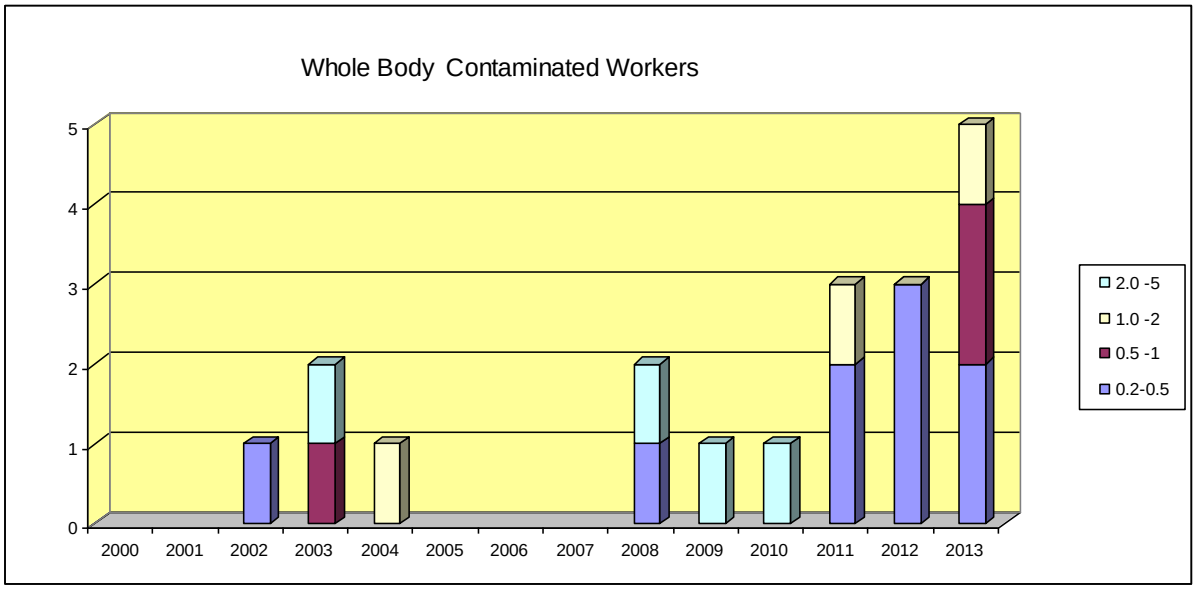

FIG. 4. In vivo whole body monitoring -The annual cumulative number of contaminated workers grouped into dose ranges (mSv)

The whole body internal contamination detected in some cases was resulted from research activities, production of radioactive sources for industry and Cyclotron maintenance, being treated as acute intakes.

In vitro tritium monitoring in certified conditions, is a new activity in the WBM Laboratory. It was performed monthly, during the year 2013, for a target group of two workers (A and B in Fig. 5), involved in research activities of Tritium laboratory from IFIN-HH. The variation of the tritium concentration in the urine, plotted in Fig. 5, shows values of

tritium in urine in the 50-35000 Bq/l range to which correspond committed effective doses less than 0.1 mSv.

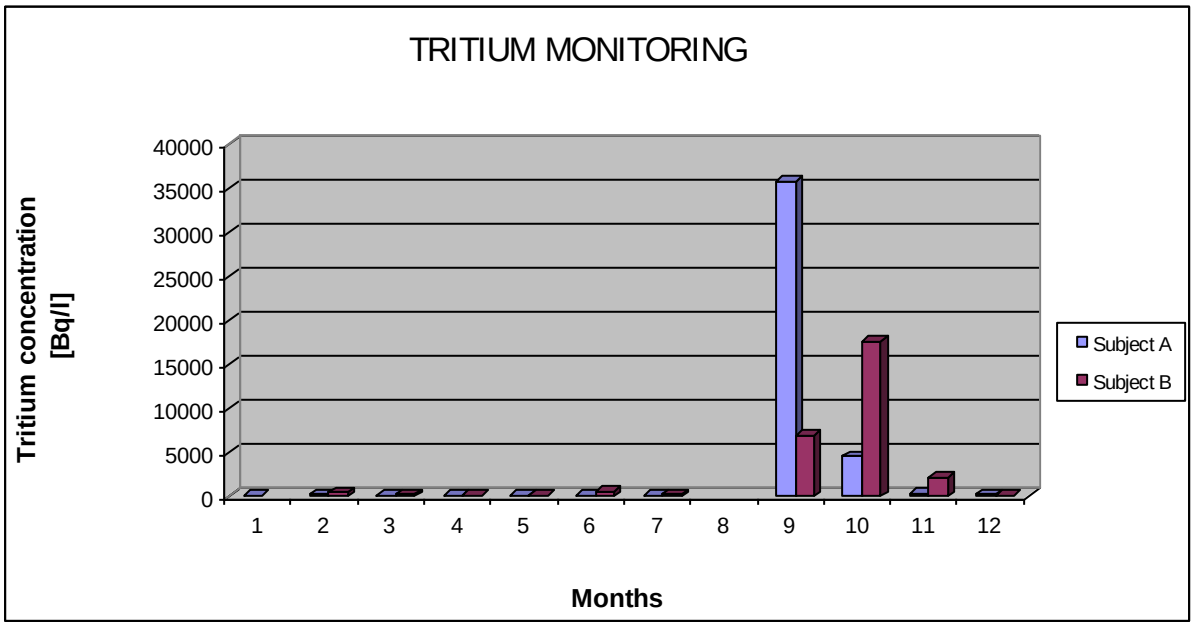

FIG. 5. In vitro tritium monitoring

Considering the different contributions to the committed effective dose, Fig. 6 shows the variation of total radioactive internal contamination of workers from IFIN-HH in the period 2000-2013, in terms of collective dose and number of workers.

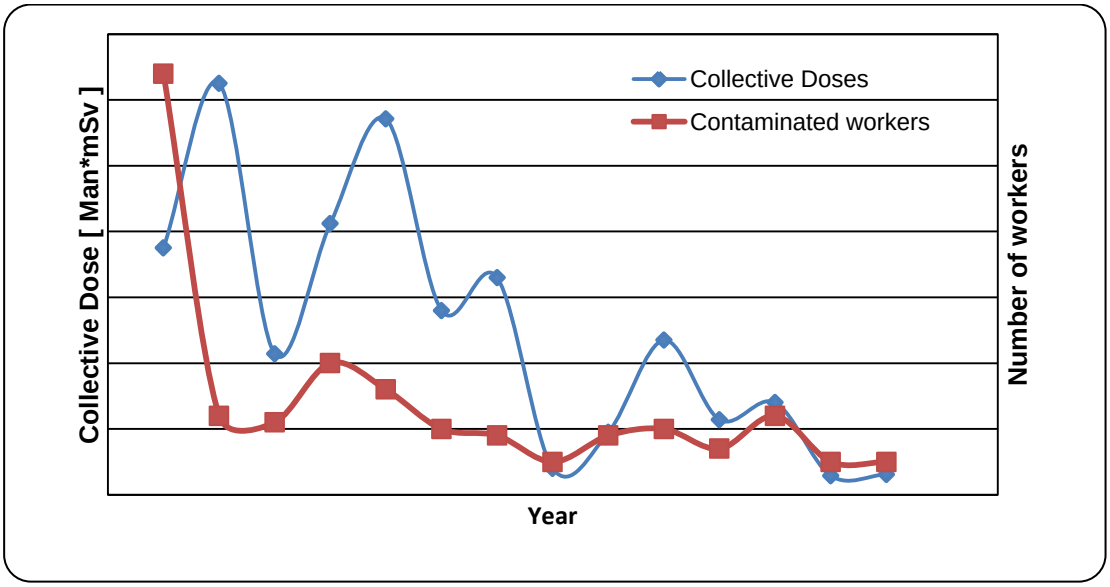

FIG. 6. Internal contamination in IFIN –HH facilities since 2000

The gradual decrease of the collective doses and of the number of contaminated workers is a success of the radioprotection optimization and of a better implementation of the monitoring programme**.**

4. TRENDS OF INTERNAL EXPOSURE MONITORING IN IFIN-HH

The new research units from IFIN-HH, namely: Tritium Laboratory, Radiopharmaceutical Research Center for PET medical radioisotopes production and Carbon Laboratory (in process of authorization) require the development of internal contamination monitoring practices in WBML according to the characteristics of biokinetic models of radionuclides to be measured as ^{3}H, ^{18}F and ^{14}C. In addition to in vivo measurements, in vitro monitoring will be performed routinely on urine samples by beta spectrometry using the liquid scintillation counting.

5. CONCLUSION

The Whole Body Monitoring Laboratory from IFIN-HH Romania has an important role at the national level for enhancing the protection of workers handling radioactive materials. It has state-of-art equipment and an updated quality assurance system that is permitting the development of reliable internal contamination, in vivo and in vitro, monitoring practices for a larger number of radionuclides produced or used in all facilities and specialized research laboratories of IFIN-HH.

REFERENCES

[1] NATIONAL COMMISSION FOR NUCLEAR ACTIVITIES CONTROL, Fundamental norms of radiation Safety, NSR-01 (2000).
[2] NATIONAL COMMISSION FOR NUCLEAR ACTIVITIES CONTROL**,** Norms of occupational personal dosimetry, NSR-06 (2002).
[3] SAIZU M.A., Quality Assurance Programme Implemented in the Whole Body Monitoring Laboratory of IFIN-HH Romania, Radiation Protection Dosimetry, Vol.144 (1-4) (2011**)** 137-139,

INVENTION OF UNIQUE AND DEVELOPEMENT OF ROUTINE RADIATION MONITORING TECHNIQUES FOR POLISH NUCLEAR PROGRAMME, INDUSTRY AND MEDICINE

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Abstract

According to Polish Nuclear Programme announced by Polish Government, the first nuclear power plant in Poland will start operating in 2030. The development of nuclear industry means the need of improvement already used dosimetric techniques and implementation of the new ones. Radiation monitoring techniques are also necessary in medicine. Increase in the number of medical procedures performed using ionizing radiation is also the cause for developing radiation monitoring methods. This paper presents the achievements of the dosimetry service of National Center for Nuclear Research – Radiation Protection Measurement Laboratory, in the field of environmental, internal and mixed radiation dosimetry. Presented techniques may be used for nuclear industry purposes as well as for medicine and others sectors of industry. They will contribute to increasing safety of occupationally exposed workers.

1. INTRODUCTION

The Polish Government decided early in 2005 that for energy diversification and to reduce carbon and sulfur emissions the country should move immediately to introduce nuclear power. A resolution by the Council of Ministers then called for the construction of at least two plants in Poland. The revised program was announced in 2014. According to it the country first nuclear power plant will start to operate by 2030 and the second one by 2035. This program presents a new challenge to Polish dosimetric service in the field of environmental and occupational exposure monitoring for nuclear power plant purposes.

Currently, there is only one nuclear reactor in Poland. Research reactor MARIA is operated by National Centre of Nuclear Researches (NCBJ). Thus it is the only Polish institution providing the radiological monitoring of nuclear facility. Radiation Protection Measurement Laboratory (LPD) is responsible for this monitoring. LPD provides the routine environmental monitoring of the NCBJ area and vicinity, and also individual internal exposure monitoring. The one of LPD division is Mixed Radiation Field Laboratory which is engaged in development of new measurements techniques in the field of mixed radiation dosimetry.

The radiation measurements techniques used by LPD are sufficient for current needs. The development of nuclear industry in Poland brings the need to develop already used techniques and to implement of the new measurements methods. This will allow to provide the occupationally exposure routine monitoring for nuclear power plant purposes.

On the other hand Poland is still developing the radiation application in medicine and industry. Radiation Protection Measurement Laboratory participates in some projects concerning dosimetry for medical and industry purposes in the field of occupationally and patient exposure routine monitoring.

The NCBJ and LPD activities are very important in the face of Polish Nuclear Programme. LPD realizes several projects related to this programme in the field of radiation dosimetry which include invention of unique and development of routine radiation monitoring

techniques. One of these projects was Task 6 “Development of nuclear safety and radiological protection methods for the nuclear power engineering’s current and future needs” of The National Centre for Research and Development Strategic Programme “Technologies Supporting Development of Safe Nuclear Power Engineering”. NCBJ is a member of the network which realized it. The leader of the network is Central Laboratory for Radiological Protection, other members are Institute of Nuclear Physics PAN and Institute of Nuclear Chemistry and Technology. The project was realized since September 2011 until August 2014.

LPD, as the only Polish laboratory responsible for radiological monitoring of nuclear object, realized a part of the project in the field of environmental, internal and mixed radiation dosimetry.

2. ENVIRONMENTAL AND INTERNAL EXPOSURE MONITORING TECHNIQUES

The aim of environmental monitoring task was to develop assumptions for model dosimetric laboratory for nuclear purposes. The model laboratory carries out two types of measurement– the environmental samples measurements and internal dosimetry. LPD elaborated the guideline for active and passive radiological monitoring of the nuclear power plant and its vicinity during normal work and in the emergency situation. The guideline includes project of environmental and individual monitoring system. This system must work during the building phase and then during starting, repairing and decommissioning of nuclear object. The system is based on active and passive monitoring. The guideline is one of the products of “Development of nuclear safety and radiological protection methods for the nuclear power engineering’s current and future needs” project.

Other topic realized by LPD was to improve the measurements capabilities in the field of internal dosimetry. The second aim was to improve *in vivo* (whole body counter and thyroid counter) and *in vitro* (urine samples) measurements provides by LPD. The new situation regarding the nuclear power plant building creates the need for develop and implement alpha spectrometry and emergency measurements techniques. Earlier LPD monitored the internal exposure to alpha emitters only by total alpha activity measurements. Now the procedure of plutonium isotopes determination is validated and accredited. In parallel to urine samples measurements the studies on alpha emitters determination in environmental samples are carried out.

The next internal contamination measurement technique which should improve is the implementation of emergency measurement methods. LPD has an experience in medical and nuclear industrial post-accident contamination measurements and analyzing [1] but general its capabilities let measure activity and assess internal doses at low levels, typical for routine exposure. In case of accidental exposure and higher levels of contamination the routine techniques are inappropriate. The first study concerned the measurements of high iodine ^{131}I activities in thyroid gland. Iodine activity in thyroid of female patient was measured with different radiation meters in order to estimate a possibility to use them in case of radiation accident. The results showed that most of the measuring devices used in this work can be used for initial estimation of internal contamination with iodine ^{131}I. These measurements can be performed outside the laboratory without additional shielding but the simple dose rate meters may only serve for identification and selection of contaminated persons who should be later subjected to the measurements with specially dedicated equipment [2]. The possibility of using some detectors was also checked by MCNP numerical models [3].

Realization of the topics mentioned above allowed to increase the laboratory capabilities in the field of environmental and internal dosimetry. LPD may provide now the full range of internal exposure monitoring (routine and emergency) for nuclear power plant purposes.

3. RECOMBINATION CHAMBERS FOR INDUSTRY AND MEDICINE

Mixed Radiation Field Laboratory team designs recombination chambers which are suitable for measurements in various mixed radiation fields using adequate recombination methods, eg. high pressure recombination method [3]. They may be used as detectors for medical and nuclear industry. The new products are ring-shape recombination chamber, micro gap ionization chamber and ionization chambers containing boron.

The innovative ring-shape recombination chamber KP-1 (Fig. 1) has been designed for estimation of stray radiation doses and quality factors in hadron therapy. The chamber allows for determination of absorbed dose and recombination index of radiation quality in phantoms at the distances close to the organs at risk near the primary radiation fields. The chamber body and electrodes are ring-shaped so the beam can be directed through the empty center of the ring. The ionization of the filling gas is caused by secondary or scattered radiation and can be related to the dose absorbed in the tissues close to the irradiated target volume [4].

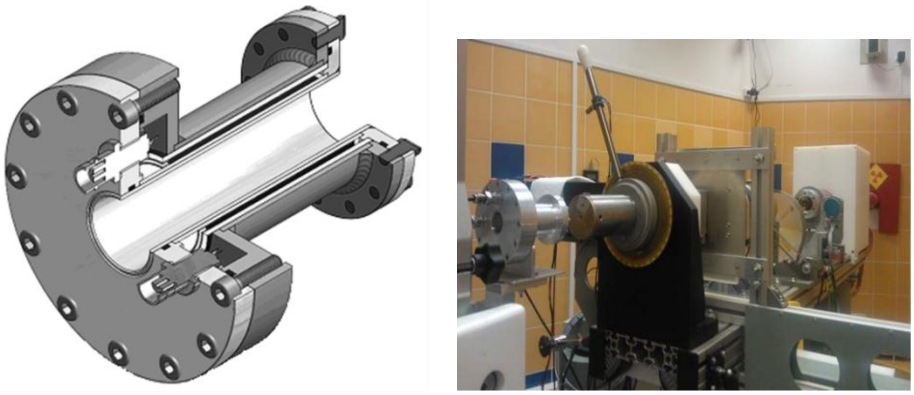

FIG. 1. 3D cross-section of the ring-shape recombination chamber KP-1 on the left and on measuring position in Institute of Nuclear Physics in Cracow (Poland) on the right

A micro-gap air-filled ionization chamber (Fig. 2) was designed for criticality dosimetry. The special feature of the chamber is its very small gap between electrodes of only 0.3 mm. This prevents ion recombination at high dose rates and minimizes the influence of gas on secondary particles spectrum. The electrodes are made of polypropylene because of higher content of hydrogen in this material, when compared with soft tissue. The difference between neutron and gamma sensitivity in such chamber becomes practically negligible. The chamber's envelope contains two specially connected capacitors, one for polarizing the electrodes and the other for collecting the ionization charge [5].

The ionization chambers containing boron (Fig. 3), operated in the initial recombination regime are the chambers were either filled with BF3 or the chamber electrodes were covered with B4C. The chambers can be placed in paraffin moderators. The sensitivity of the chambers was investigated depending on gas pressure, moderator thickness and polarizing voltage. The results showed that it was possible to obtain nearly the same sensitivity of the chamber to H*(10) for photons and neutrons in restricted energy range, however further investigations are needed to make an optimum design. The chambers containing boron can be used for dosimetric measurements in mixed radiation fields near medical linear accelerator and in the vicinity of high-energy proton accelerator [6].

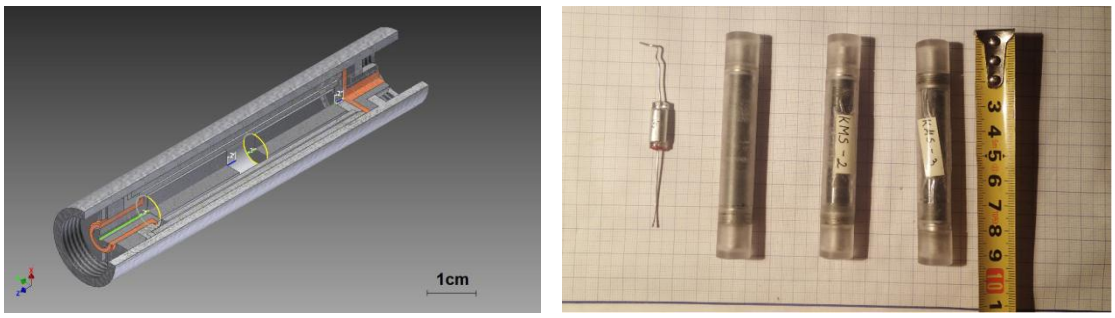

FIG. 2. Cross-sectional drawing with parts list of the micro-gap chamber on the left and physical model of the chamber

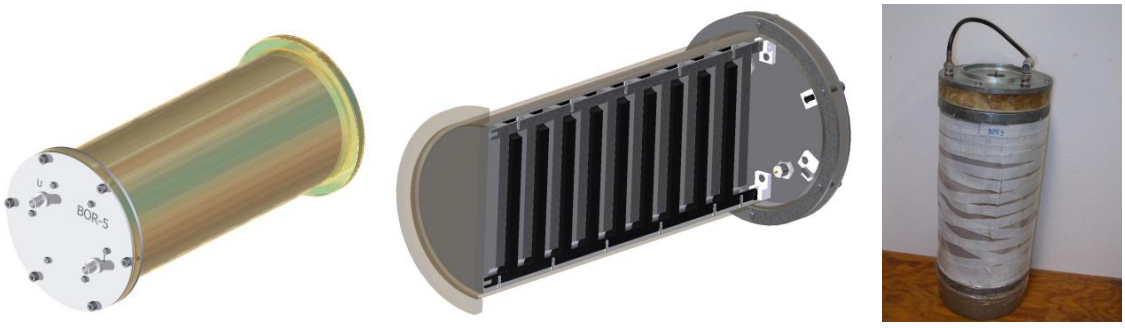

FIG. 3. Virtual model of novel chamber denoted BOR-5 on the left and its cross-section in the middle. The base of construction is the chamber BOR3 on the right tested in various radiation fields

4. CONCLUSION

The Polish Government decision nuclear power plant construction is a challenge for ionizing radiation dosimetry and also a chance to develop radiation monitoring techniques in Poland. The developed methods will be used not only in nuclear industry but also for medicine and others sectors of industry. They will contribute to raising the level of radiological protection and occupational safety in all areas.

ACKNOWLEDGMENTS

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REFERENCES

[1] OŚKO J,. GOLNIK, N., PLISZCZYŃSKI, T., Cases of post-accident contamination with iodine ^{131}i, registered in the institute of Atomic Energy POLATOM in Świerk, Poland, Radiat. Prot. Dosim. (2011) 144 (1-4) 560-563.
[2] OŚKO J,. et al., Measurements of high level of iodine activity in thyroid with different radiation meters, Pol J Med Phys. Eng. 18(1) (2013) 1-6.
[3] OŚKO, J., TYMIŃSKA, K., Numerical model of detector for in vivo internal dosimetry in radiological incidents, Pol. J.Med. Phys. Eng. 18(2) (2013) 29-39.
[4] ZIELCZYŃSKI, M., GRYZIŃSKI, M.A., GOLNIK, N., Method for determination of gamma and neutron dose components in mixed radiation fields using a high-pressure recombination chamber, Radiat.Prot. Dosim. 126(1-4) (2007) 306-309.
[5] JAKUBOWSKA, E., ZIELCZYŃSKI, M., GRYZIŃSKI, M.A., KRZEMIŃSKI, Ł., A ring-shape

recombination chamber for hadron therapy dosimetry, Radiat. Prot. Dosim. (2014), doi:10.1093/rpd/nct355.

[6] ZIELCZYŃSKI, M., GOLNIK, N., MURAWSKI, Ł., GRYZIŃSKI, M.A., A micro-gap, air filled ionization chamber as a detector for criticality accident dosimetry, Radiat. Prot. Dosim. (2013) doi:10.1093/rpd/nct323.

[7] ZIELCZYŃSKI, M., GRYZIŃSKI, M.A., GOLNIK, N., TULIK, P., Ionization chamber containing boron as a neutron detector in medical accelerator fields, Radiat. Prot. Dosim. 126(1-4) (2007) 274-277.

ARMIR: THE SYSTEM FOR ESTIMATION OF RADIOLOGICAL RISK FROM OCCUPATIONAL EXPOSURE

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Abstract

At present more then 90% of the personnel of the State Corporation Rosatom monitored for individual radiation exposure have reports on radiation risks assessed with the computed system ARMIR. For several years average value of the risk in the Russian nuclear industry does not exceed 0.00008. As of today radiation dose received by the nuclear workers corresponds to a permissible levels of health risks. The percentage of the workers in the high risk group (risk value is above 0.001) is 1.25% of the total number of the personnel registered in the ARMIR system. The most part of the high risk group are old stagers.

1. INTRODUCTION

The State Corporation Rosatom jointly with the National Radiation Epidemiological Registry and the Russian Scientific Commission on Radiological Protection developed information analytical system ARMIR for evaluation of occupational risk [1]. The ARMIR system is based on principles and methods of calculating radiation risks recommended by IAEA and ICRP [2-4]. The requirement of the IAEA Internationl Basic Safety Standards "3.110. Employers, in cooperation with registrants and licensees: (a) Shall provide all workers with adequate information on health risks due to their occupational exposure in normal operation, anticipated operational occurrences and accident conditions" [5] was taken into account in the system.

Risk estimates calculated with the use of ARMIR are published in «Public annual report» and special mass media [6-7] and available to professionals and public.

2. PROFESSIONAL RADIATION EXPOSURE AND RADIATION RISKS FOR PERSONNEL

The system ARMIR was first used in the "Mayak Production Accociation" in 2006. At present the system is used in 50 radiation and nuclear hazardous sites of the Rosatom. In 27 plants Internet service is used for the work with the system (Fig. 1). Radiation safety units in 23 plants use a stand-alone software.

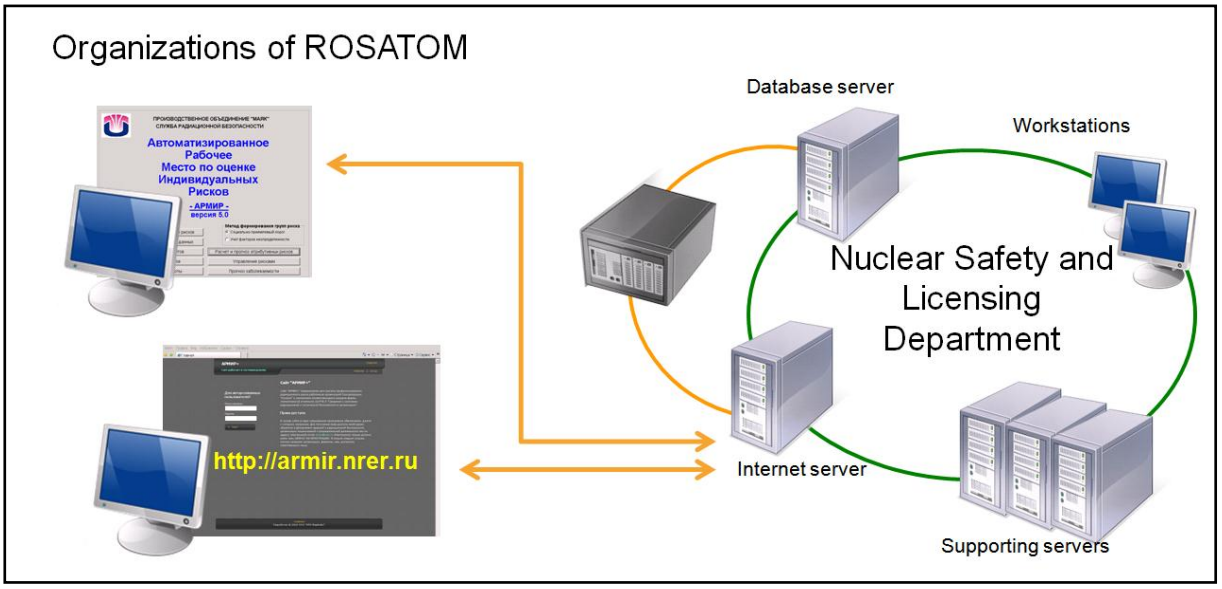

FIG. 1. Information structure of the ARMIR system

The ARMIR system control is carried out at the Department of Nuclear and Radiation Safety. The essential function of the control is to assure quality and completeness of initial data. The ARMIR data bases consist personal information, such as identifier, last name, first name, patronymic name, gender, date of birth, and annual radiation doses of monitored workers. As of today, the ARMIR databases comprise personal information of 61304 workers, it is 90.8% of all monitored workers.

Summary data on occupational radiation doses to workers put into the ARMIR system are given in Table 1. Data are given for different divisions of the Rosatom.

Average value of radiation risk does not practically change as compared with that in the previous year, it is $0.76*10^{-4}$. At the same time, the value of the maximum risk decreases significantly from 0.012 in the last year to 0.007 in the reporting year (Table 2).

TABLE 1. BASIC CHARACTERISTIC OF OCCUPATIONAL RADIATION EXPOSURE IN MAIN DIVISIONS OF THE ROSATOM

Division	Number of workers	Average age, years	Average cumulative dose, mSv	Average annual dose, mSv	Average length of work, years
Electric power division	25560	42.2	51.30	1.70	11.0
Fuel division	9960	43.2	20.60	1.11	10.6
Nuclear weapons complex	14522	42.6	33.20	1.92	11.6
Mining division	3626	37.2	35.45	3.58	6.7
Nuclear and radiation safety complex	2706	44.6	38.76	0.79	15.7
Innovation managment unit	4550	47.7	44.57	2.12	15.7
ROSATOM	61304	42.7	39.86	1.75	11.4

TABLE 2. AVERAGE AND MAXIMUM VALUES OF RADIATION RISK IN MAIN DIVISIONS

Division	Average risk	Maximum risk
Electric power division	1.0E-04	6.1E-03
Fuel division	3.4E-05	6.9E-03
Nuclear weapons complex	5.9E-05	6.7E-03
Mining division	2.1E-05	2.2E-04
Nuclear and radiation safety complex	1.0E-04	4.4E-03
Innovation managment unit	1.1E-04	3.2E-03
ROSATOM	7.6E-05	6.9E-03

Both in previous year and in reporting year the most part of the staff have worked under conditions of acceptable risk from occupational exposure.

In 769 workers the personal risk exceeded 10^{-3}. The relative number of the workers with high risk decreased from 1.30% to 1.25% of those put into the ARMIR system. The average age of workers at high risk is 60 years, no workers of the age younger than 45 years are at high risk. Average accumulated dose to those at high risk is 449 mSv, average dose received in 2013 is 3.5 mSv. Though the average length of work with radiation sources in the group at thigh risk exceeds 36 years, the people continue their operative activity.

3. CONCLUSIONS

At present, more than 90% of the personnel of the State Corporation Rosatom monitored for individual radiation exposure have reports on radiation risks assessed with the computed system ARMIR. During several years, average radiation risk in the industry does not exceed 0.00008. At present, radiation dose to the staff of the State Corporation Rosatom corresponds to acceptable level of health risk.

The percentage of workers at high radiation risk is 1.25% of all workers put into the ARMIR system. The major part of the high risk group are old stagers, the average age in the group is 60 years. Average length of work exceeds 3.5 times, and average accumulated dose exceeds 12 time the average length of work and average dose characteristic of all monitored workers.

Radiation risk estimates and their temporal changes should be used for planning radiation protection programmes, and for optimization of radiological protection in order to prevent increasing the number of staff members at high risk from occupational exposure.

REFERENCES

[1] IVANOV. V.K.. et al.. Estimation of individualized radiation risk from chronic occupational exposure in Russia, Health Physics, 97 (2009) 107.

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Methods for estimating the probability of cancer from occupational radiation exposure, IAEA-TECDOC-870, IAEA, Vienna (1996).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE,

WORLD HEALTH ORGANIZATION., Approaches to attribution of detrimental health effects to occupational ionizing radiation exposure and their application in compensation programmes for cancer, Occupational Safety and Health Series No. 73, ILO, Geneva (2010).

[4] INTERNATIONAL COMMISION ON RADIOLOGICAL PROTECTION, The 2007 Recommendations of the International Commission on Radiological Protection, Publication 103, Ann. ICRP 37 (2-4).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY. Radiation protection and safety of radiation sources, International basic safety standards, General safety requirements Part 3 (Interim), IAEA, Vienna (2011).

[6] http://www.rosatom.ru/en/investors/anreps/.

[7] http://www.rosatom.ru/journalist/news/9c164f0042229bddbeafbf00bbc6a657.

THIRTY FIVE YEARS OF OCCUPATIONAL INDIVIDUAL MONITORING AT UNIVERSITY OF SÃO PAULO

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Abstract

The External Dosimetry Service of Institute of Physics of the University of S. Paulo monitors workers occupationally exposed to ionizing radiation since the end of seventies. In this paper, methodology used for monitoring and assessment of dose equivalent is described. The data of dose equivalent, obtained with monitors worn on the trunk and on the wrist, collected from 1995 to 2013 are presented.

1. INTRODUCTION

External Dosimetry Service of Dosimetry Laboratory of Institute of Physics of University of S. Paulo performs individual and area monitoring since the end of seventies. Currently, around 800 trunk and 50 wrist monitors are routinely processed, consisting of detectors based on thermoluminescent dosimetry for external radiation. In the case of individual monitoring, the workers wear the monitor on the trunk, to represent the whole body dose, and, in some cases, they wear a second one on the wrist. The main users belong to the Institutes of Chemistry, Physics and Biomedicine, the Veterinary Hospital and the University Hospital. The Service is accredited by the Comitê de Avaliação de Serviços de Ensaio e Calibração (CASEC - Committee on Evaluation of Essay and Calibration Services), especially designated for this purpose by National Nuclear Energy Commission.

2. METHODS

Thermoluminescent (TL) dosimetry is the technique used by the Dosimetry Laboratory of Institute of Physics of S. Paulo University to monitor workers occupationally exposed to radiation. Two types of detectors are used: TLD-100 and natural green CaF_2 pellets [1] shown in Fig. 1. Brazilian natural green fluoride as-from-the-mine, illustrated in Fig. 1, is ground and sieved to obtain homogeneous powder with grain size from 85 to 185 μm. CaF_2 pellets are produced by cold pressing the fluorite in the mass proportion of 60% with reagent grade NaCl, as agglutinant, with 40%.

FIG. 1. Brazilian natural green fluoride as-from-the-mine and TL detectors: TLD-100 LiF and CaF_2 pellets

FIG. 2. Individual monitoring badge used by External Dosimetry System

The badge to accommodate the TL detector, shown in Fig. 2, is made of black plastic with 0.587 kg·m^{-2} thickness, and contains two TLD-100 and two natural CaF_2:NaCl detectors - herein after referred as CaF_2. One detector of each type is kept within 0.5 mm Pb filters. Inside the badge there is also a card of 0.184 kg·m^{-2} thickness, for the user identification. TLD-100 and CaF_2 detectors are pre-heated before use at 400°C/1 h followed by 100°C/2 h and 400°C/20 min, respectively. A TL reader based on a photon counting system is used to register the glow curves.

Dosimeters were type tested and are routinely calibrated by irradiating monitors free-in-air with gamma rays of a calibrated ^{60}Co source. The value of air kerma free-in-air is obtained by multiplying the dosimetric peak area of each detector by the respective calibration factor. As the response of both TL detectors depends on the photon energy, a correction is performed according to the mean photon energy incident on the monitor [2]. For this, the curves of TL response normalized by air kerma as a function of photon energy from 25 to 1250 keV were experimentally obtained for both detectors, with and without the Pb filter. The operational quantity $H_p(10)$, personal dose equivalent is used for the assessment of the effective dose.

The dose calculation is performed with an algorithm especially developed in the laboratory. The workers use the monitors for periods of one or three months, according to their activities: the three months is chosen for workers whose occupational dose in a month is of the same order of magnitude of the background radiation dose. The lower limit of detection is 0.1 mSv.

2.1 Performance testing of individual monitors

The Dosimetry Laboratory sends monthly four monitors to CASEC, which exposes them to ^{60}Co or ^{137}Cs gamma radiation with different doses. The results of testing during 2011 and 2012 can be seen in the graph of Fig. 3, together with the trumpet curve that defines smooth upper and lower dose equivalent limit curves of the permissible interval around the conventional true value.

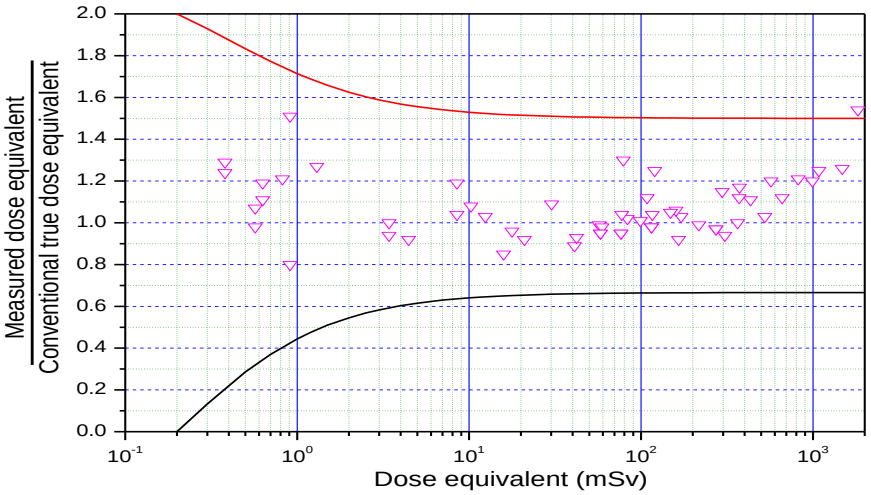

FIG. 3. Results of performance testing of thermoluminescence dosimeters for photon radiation

3. RESULTS

The data of annual personal dose equivalent collected from 1995 to 2013 with trunk and extremity monitors are illustrated in Figs. 4 and 5 respectively. These data correspond to the effective dose (whole body dose) and extremity dose (hands equivalent dose), respectively. According to ICRP Publication 103 [3], for monitoring occupational exposures to external radiation, individual dosimeters to be used to measure the personal dose equivalent $H_p(10)$. This measured value is taken as an assessment of the effective dose under the assumption of a uniform whole body exposure. Only the one specialized professional in endoscopy at the University Hospital received annual dose as high as 340 mSv.

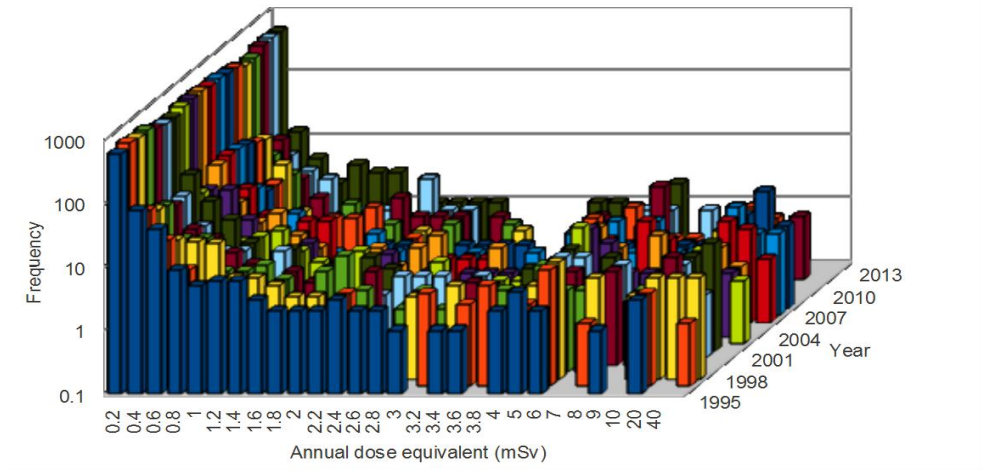

FIG. 4. Data of annual dose equivalent assessed by monitors worn in the trunk from 1995 to 2013

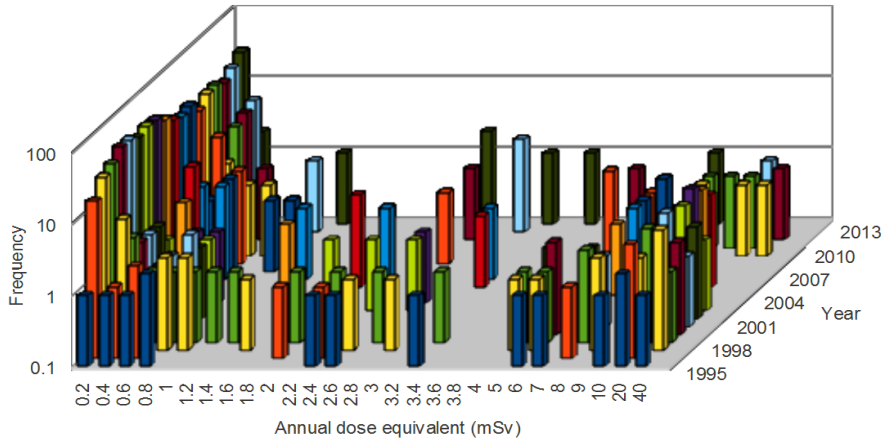

FIG. 5. Data of annual dose equivalent assessed by monitors worn in the wrist from 1995 to 2013

4. CONCLUSIONS

The results obtained by the laboratory in the verifications promoted by CASEC are usually in the interval of 20% around the conventional true-value, and practically always inside the interval of the trumpet curves, showing a good performance. As far as the doses to workers of the university are concerned, in the period from 1995 to 2013, whole body doses exceeding 20 mSv per year or extremity doses higher than 40 mSv per year, are rarely found. Workers from the hospitals are the ones who receive the highest doses, due to their activities in interventional radiology.

REFERENCES

[1] TRZESNIAK, P., YOSHIMURA, E.M., CRUZ, M.T. and OKUNO, E., Brazilian fluorite-based dosimetric pellets: history and post-use review", Radiat. Prot. Dosim. 34 (1990) 167.

[2] GUIMARÃES, C.C., OKUNO, E., Effective energy determination of radiation incident on a monitor. Proceedings of Fifth Congress on Radiation Protection and Safety (an IRPA Regional Congress), Recife, Brasil (April, 2001).

[3] Internatinal Commission on Radlogical Protection, 2007. The 2007 Recommendations of the International Commission on Radiological Protection. ICRP Publication 103. Ann. ICRP 37 (2-4) (2007).

NEW ISO STANDARDS FOR THE RADIATION PROTECTION OF AIRCRAFT CREW

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Abstract

The dose assessment for aircaft crew exposed to an elevated level of radiation exposure owing to the cosmic radiation field in the atmosphere is based on numerical calculations of the dose distribution at altitudes up to about 12 km. The respective software codes are based on numerical simulations of the entire interaction processes of the primary cosmic radiation entering the atmosphere from outer space or from sets of previous measurements performed on board aircraft at flight altitudes. The validation of such codes can only be done by comparing the results with measurements in terms of the operational quantity ambient dose equivalent, $H^*(10)$. Since the radiation field is very complex in its composition (the main components are neutrons, protons, electrons and positrons, muons and photons) and particle energies (from keV to GeV), new ISO standards were required. The ISO 20785 series describes the dosimetry methods applicable in such a radiation field (part 1), the calibration of the instruments used (part 2), and the special circumstances to be considered when measuring on board an aircraft (part 3). Part 4, which will deal with the approval of software codes used to evaluate the effective doses received by aircraft crew, is in preparation.

1. INTRODUCTION

In 1996, the radiation protection of aircraft crew based in the European Union became a legal issue by the publication of the directive 96/29/EURATOM [1] in which Article 42 states that each member state "shall make arrangements for undertakings operating aircraft to take account of exposure to cosmic radiation of air crew who are liable to be subject to exposure to more than 1 mSv per year and that the undertakings shall take appropriate measures, in particular to assess the exposure of the crew concerned. This directive was the basis for a period with extensive research activities to investigate the world wide dose distributions at aviation altitudes, i.e. at altitudes up to 12 km and above [2, 3, 4]. At cruising altitudes between 8 km and 12 km, a typical range for passenger and cargo aircraft, the radiation field of cosmic radiation is composed of neutrons, protons, electrons/positrons, photons, pions and muons. The contribution to the radiation exposure of aircraft crew is dominated by neutrons (about 40 % to 60 %) which cover an wide energy range from thermal up to 1 GeV [5, 6]. Compared to other regulated radiation protection areas, this complex field requires sophisticated techniques to determine the ambient dose equivalent, or its rate, and the subsequent estimation of the effective dose as the protection quantity.

Since the source of the radiation field at aviation altitudes, i.e. the galactic comsic radiation (GCR), is rather constant, it is recommended [7] and accepted by national radiation protection authorities that the effective doses of aircraft crew can be calculated using the flight route data. Only the software codes which can be used for that purpose need to be verified by measurements. It must be emphasized that the radiation protection quantity, the effective dose, is not a measureable quantity. Thus, the software codes can only be verified by comparing the calculated and measured ambient dose equivalent, $H^*(10)$, or its rate. From the experience gained during the research activities mentioned above, a group of experts initiated the formulation of an international standard concerning the dose measurements in aircraft at altitudes.

2. ISO 20785: DOSIMETRY FOR EXPOSURES TO COSMIC RADIATION IN CIVILIAN AIRCRAFT

In the year 2000, a new work item proposal was submitted to the ISO secretariat to formulate a basic standard for measuring the ambient dose equivalent at aviation altitudes, ISO 20785 "Dosimetry for exposures to cosmic radiation in civilian aircraft". Within the following process, it became clear that the dosimetry inside aircraft at altitudes requires a very detailed description of steps to be undertaken in order to achieve traceability to national primary standards. It was also recognized that a single document would be too complex to be finalized within a few years' time. Therefore, it was decided in 2004 that the ISO standard would be split into three parts describing common recommendations and requirements concerning the dosimetry methods which are applicable (part 1), the calibration of the instruments used (part 2), and the special circumstance to be considered when measuring on board an aircraft (part 3).

2.1. ISO 20785-1: Conceptual basis for measurements

After providing definitions of the relevant terms and a general description of the cosmic radiation field in the atmosphere, part one of the standard describes concepts, instruments, and methods which are applicable for dosimetry of the cosmic radiation field at aviation altitudes. Representative particle fluence rate energy distributions for the six major particles of the cosmic radiation field at flight altitudes are provided in an annex. Based on how the various particles interact, the cosmic radiation field can be divided into neutron and non-neutron components. The neutrons plus the neutron-like interactions of protons comprise the neutron component. Directly ionizing particles and the secondary electrons from indirectly ionizing photons comprise the non-neutron component. Based on the dependence of the quality factor on linear energy transfer (LET), the field can be divided into low LET (< 10 keV/μm) and high LET (≥ 10 keV/μm) components. The relative contributions to the total ambient dose equivalent of the low LET and non-neutron components and the high LET and neutron and neutron-like components are not necessarily the same, but are generally similar in magnitude.

The ambient dose equivalent is reasonably approximated, assuming suitable calibration and normalisation, by the response of a tissue equivalent proportional counter (TEPC), recombination ionisation chamber or semiconductor spectrometer. The low LET or non-neutron energy deposition can be determined using an ionisation chamber, silicon-based detector, or scintillation detector; or a passive luminescence or ion storage detector. The high LET or neutron component can be measured using an extended range neutron survey meter, multi-sphere spectrometer, passive etched track detector, bubble detector, or fission foil with damage track detector. The summed components, low LET plus high LET or non-neutron plus neutron and neutron-like, with suitable calibration and normalization, give the total ambient dose equivalent. It is essential for the measurement of the complex radiation fields

that the instruments used are fully characterized at national standards laboratories where possible so that full traceability is established.

2.2. ISO 20785-2: Characterization of instrument response

Part two of ISO 20785 deals with the calibration of the instruments used. The procedure described here is similar to the one used for the calibration of area monitoring instruments. The calibration of instruments has been defined in ISO 29661, which was developed in parallel to ISO 20785-2. This led to the same terms and definitions; therefore, comparable procedures for both standards can be used. This is a very important issue since both dosimetry practices, i.e. area monitoring using instruments and calculation of the radiation exposure, apply to the radiation protection of occupational exposed persons.

Because of the number of different particles and the wide particle energy range of the radiation field at aviation altitudes, the measurement of ambient dose equivalent, or its rate, requires detailed characterization of instruments under well-defined conditions. The energy and angle dependence of the response of instruments need to be measured, and a calibration or response function (more usually a matrix) established. Response measurements will need to be made in reference fields and can, in addition, be made in radiation fields representative of cosmic radiation. The influences of environmental parameters, such as air pressure and vibration, have to be examined. In a radiation field that is representative of a component of the cosmic radiation field, a field-specific calibration factor or calibration coefficient can also be established directly. Response measurements should be performed in conditions similar to those at aviation altitudes or the instrument indication corrected for all significant influence quantities. The single field-specific correction factor, or calibration factor or coefficient, can be applied for the assumed energy and angle distributions of the given particle type or types. In some instances, this same factor or coefficient can be applied for the range of flight altitude, geomagnetic latitude and solar modulation over which measurements are to be made.

Because of the several types of particle contributing to the measurement quantity and their large energy ranges, more than one measuring device is often needed or one measuring device can be operated in several modes; for example, a tissue-equivalent proportional counter where different ranges of energy deposition event sizes can be treated differently. In principle, for all such cases, the combination of measuring devices, or operating modes, together with any algorithm, shall be treated as one measuring assembly and calibrated as such. In practice, based on prior knowledge of the response characteristics, it might be acceptable to calibrate components of the measuring assembly separately, but then attention shall be paid to the effect of other radiation types and the algorithm.

2.3. ISO 20785-3: Measurements at aviation altitudes

This part of 20785, deals with the measurements on board aircraft to determine the ambient dose equivalent or its rate to monitor the exposure of aircraft crew. These measurements may also be used to validate codes (section 3 below) and to study the contribution of solar activity. To obtain exposure information representative of aircrew members, the chosen instruments are typically located inside the aircraft in the flight deck or cabin. Since the aircraft structure may influence the dose rate, a correction factor, K_l, should be applied. In practice, factors such as fuel, passenger, baggage, and cargo loads may vary and make this determination very difficult. In addition, for instruments with an energy dependence to different radiations, a correction factor, $K_{E,R}$, and, for expected anisotropic response, a correction factor, K_{Ω}, should be applied. In practice, in place of evaluating these factors, an estimate of the associated uncertainty must be made.

Co-operation from the aircraft crew is usually needed, including the flight plan and the actual route, barometric altitudes, and geographical coordinates during the flight. Although solar activity will be relatively constant during a flight, a solar particle event may greatly influence the results. It is recommended that the instruments be non-powered or self-powered to avoid the complications of connections to the aircraft. It is likely that all equipment, including portable computers, will have to be checked for electromagnetic interferences. For passive equipment, security screening may mean that any dose obtained may need to be subtracted.

3. CODE VALIDATION

Program codes used for the evaluation of the effective doses received by aircraft crew need to be validated because these calculated values enter into the personal dose record of each crew member and, in some countries, enter the national radiation dose register. Since the effective dose E is not measureable, codes can only be validated by comparing the calculated ambient dose equivalent with measured data which should be obtained in accordance with the ISO 20785 as described above.

A code can also be validated by comparing the calculated dose rates with reference data published by ICRU which are based on more than 20000 data points of the ambient dose equivalent rate measured by different groups with different instruments [8]. From these data set, Bayesian analysis methods have been used to evaluate dose rates for the reference conditions which are given for three times during a solar cycle (January 1998, January 2000 and January 2002) and at three altitudes (FL 310, FL 350, FL 390) as a function of the vertical cut off rigidity from 0 GV to 17 GV.

Another possibility for code validation is a comparison of different codes which have been used to calculate the dose rate profile along a predefined flight route and/or the total route dose. In this case, the ambient dose equivalent $H^*(10)$, or its rate, as well as the effective dose E, or its rate, can be compared. The disadvantage is that a code comparison gives only a measure of the deviation of the different codes from their mean value. A validation can only be made by using the ICRU reference data mentioned before or by including codes which are based on measured data which are traceable to a national primary standard. An example of such a code comparison has been published by the EURADOS group [9]. The main result from this action was that all eleven codes agreed within ±20 % from the median.

Both approaches are only functional tests of the codes, i.e. testing whether the numerical results are correct. Since legal radiation protection requires codes which are certified, the software code itself must be tested using standard software testing procedures.

4. SUMMARY

The ISO 20785 series describing the dosimetry methods for cosmic radiation measurements on-board aircraft were necessary to harmonize procedures for measuring the ambient dose equivalent and to take into account the fact that classical procedures for calibration using reference fields cannot always be applied because of the specificity of a complex and high energy field.

REFERENCES

[1] COUNCIL DIRECTIVE 96/29/EURATOM of May 13, (1996). Official Journal of the European Communities N° L 159 of 29.6.96. (1996).
[2] L. LINDBORG, D. T. BARTLETT, P. BECK, I. R. MCAULAY, K. SCHNUER, H. SCHRAUBE AND F. SPURNÝ, Cosmic radiation exposure of aircraft crew. Compilation of measured and calculated data. Final Report of the EURADOS WG 5. European Commission, Directorate-General for Energy and Transport, Radiation Protection Issue No. 140, Luxembourg, ISBN 92-894-8448-9, (2004).
[3] M. KELLY, H.-G. MENZEL, T. RYAN, K. SCHNUER (edts.), Cosmic radiation and aircrew Exposure, Proceedings of an International Conference, Dublin, 1999. Radiation Protection Dosimetry, 86 (1999) 243-353.
[4] J.-F. BOTTOLLIER-DEPOIS, G. REITZ, W. RÜHM, F. WISSMANN (edts.), Cosmic radiation and aircrew exposure, Proceedings of the EURADOS workshop Braunschweig, 2009. Radiation Protection Dosimetry, 136 (2009) 231-324.
[5] D. T. BARTLETT, Radiation protection aspects of the cosmic radiation exposure of crew, Radiation Protection Dosimetry, 109 (2004) 349-355.
[6] B. WIEGEL, A. V. ALEVRA, M. MATZKE, U. J. SCHREWE, J. WITTSTOCK, Spectrometry using the PTB neutron multisphere spectrometer (NEMUS) at flight altitudes and at ground level. Nuclear Instruments and Methods in Physics Research A 476 (2002) 52-57.
[7] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37, Elsevier, (2007).
[8] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Reference data for the validation of doses from cosmic radiation exposure of aircraft crew, ICRU Publication 84, Journal of the ICRU 10 Oxford University Press (2010).
[9] J.F. BOTTOLLIER-DEPOIS, P. BECK, M. LATOCHA, V. MARES, D. MATTHIÄ, W. RÜHM, F. WISSMANN, Comparison of Codes assessing radiation exposure of aircraft crew due to Galactic Cosmic Radiation, EURADOS Report 2012-03, European Radiation Dosimetry, e.V., Braunschweig, ISBN 978-3-943701-02-9 (2012).

IMPLICATIONS OF THE NEW LENS DOSE LIMIT FOR DOSIMETRY AND RADIATION PROTECTION PROGRAMS AT NUCLEAR POWER PLANTS

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Abstract

The ICRP recently recommended the eye lens dose limit be equal to the effective dose limit. This triggered significant advances in lens dosimetry, but further advances and better dosimetry are required in NPPs, where non-uniform beta and photon fields occur. Significant levels of these have the effect of reducing the allowable effective dose to ensure that the eye lens dose limit is not exceeded. Significant changes to dosimetry, dose records, workplace characterization, radiation instrumentation, protective equipment, and RP programs and procedures will be required. The extent of these changes are less if low effective doses are maintained. The extent of the changes would be less, without sacrificing protection of the lens, if the ICRP had recommended a somewhat higher dose limit for eye lens keeping in view the risk of cataract and risk of fatal cancer in perspective. .

1. INTRODUCTION

In April 2011, the International Commission on Radiological Protection (ICRP) published a new recommended dose limit for the lens of the eye [1]. Based on new radiobiological evidence, the ICRP considered the threshold for the lens to be 0.5 Gy. Based on this, the ICRP recommended an equivalent dose limit f of 20 mSv/year, averaged over defined periods of 5 years, with no single year exceeding 50 mSv. This new limit equals the current limit for effective dose [2]. The annual average limit of 20 mSv is 7.5 times lower than the previous limit of 150 mSv [3]. New radiobiology supports reducing the limit. However, making the lens and effective dose limits equal has significant implications for the dosimetry and radiation protection programs for nuclear power plants (NPPs).

2. DOSIMETRY

Because it is generally not possible to measure the dose limiting quantities, personal dose equivalents are used: for effective dose, $H_p(10)$, measured at a depth of 10 mm (1000 mg cm^{-2}); for the lens, $H_p(3)$, measured at a depth of 3 mm (300 mg cm^{-2}); and for the skin, $H_p(0.07)$ measured at a depth of 0.07 mm (7 mg cm^{-2}). In NPPs, most dose is received from photon and beta sources, with neutrons ususally contributing little annual dose. Most photons have energies from about 10 keV to 1.3 MeV, with a smaller contribution from photons up to 6 – 7 MeV. Only betas with energies > 800 keV can penetrate to 3 mm depth of the lens and contribute to $H_p(3)$. Many radionuclides present in NPPs have maximum beta energies > 800 keV, but the high ratio of the old lens dose limit to the effective dose limit meant that meeting the effective dose limit ensured the lens dose limit would not be exceeded. This is no longer the case, and triggered a significant amount of recent research on lens dosimetry.

A study of 8 sets of $H_p(0.07)$ photon dosimeters concluded that all were suitable for measuring $H_p(3)$ from photons, provided they were on the head and close to an eye [4]. A new cylindrical phantom that mimics a human head with both a height and diameter of 20 cm has been developed that has a better angular response than that from the slab or rod phantoms [5,

6]. Two European groups have calculated air kerma to $H_p(3)$ conversion coefficients using Monte Carlo simulations according to the methods specified in ISO-4037 [7, 8, 9]. Behrens has subsequently calculated $H_p(3)$ dose conversion coefficients for mono-energetic electron exposures using a realistic model of the eye embedded in a mathematical human phantom [10]. Eakins et al. have calculated $H_p(3)$ dose conversion coefficients for the poly-energetic electron spectra coming from $^{90}Sr/^{90}Y$ beta fields [11].

Gilvin et al. type-tested a Harshaw EXTRAD thermoluminscent dosimeter (TLD), a thin high-sensitivity, nearly tissue equivalent thin layer of LiF:MCP as the dosimeter element for measuring $H_p(3)$ [12]. There was a reasonably flat photon energy (range of about ± 20%) and angle (range of about ± 10%) response, but for $^{90}Sr/Y$, a significant under-response from -60° to +30°, with an over-response of > 2 for +60°, due to the asymmetric design of the dosimeter. The under-response may be due to the extra shielding (about 0.5 mm, or 50 mg cm^{-2}) beyond the 300 mg cm^{-2} over the lens. This under-response contrasts to an over-response of 20% to $^{90}Sr/^{90}Y$ for the same dosimeter as an extremity dosimeter measuring $H_p(0.07)$ [13]. We are not aware of any useful measurements of the beta energy response for $H_p(3)$ of LiF:MCP other than to $^{90}Sr/Y$, and it is thus very problematic to use a lens dosimeter which has only been tested with a single beta-emitting radionuclide combination.

Ontario Power generation (OPG) has developed a four-element TLD badge using different shielding on each element to provide accurate measurments of $H_p(10)$ and $H_p(0.07)$ for both photons and betas (response within 5% of 1.00 from 0 to 40°for photons and for ^{85}Kr and $^{90}Sr/^{90}Y$ at 0° [14]. A review of its performance indicates that it could readily provide an accurate measurement of $H_p(3)$ for photons, but not for betas. A Monte Carlo study shows that by increasing the shielding of one of the two elements used for beta dosimetry, an accurate measurement of $H_p(3)$ at 0º can be made, although the $H_p(3)$ response falls off significantly with angle [15]. Because of its size, the OPG TLD badge (or a modification of it) could not readily be used as a lens dosimeter, and a single-element dosimeter will likely be required for lens dosimetry. However, the results from the less accurate lens dosimeter need to be compared to a more accurate measurment of $H_p(10)$ for the head to determine if there is a net lens dose. This may require that the allowable effective dose be reduced to ensure that the eye dose limit is not inadvertently exceeded because of the poorer accuracy of the lens dosimeter.

3. LIMITING EFFECT OF LENS DOSE LIMIT ON EFFECTIVE DOSE

In uniform photon fields, the dose to the eye, $H_p(3)$, will generally lie between $H_p(0.07)$ and $H_p(10)$. In the photon energy range found in NPPs, the measured $H_p(3)$ dose will be very close to $H_p(10)$, and the effect of making the lens dose limit equal to the effective dose limit is small. If betas are present, $H_p(3)$ will be greater than $H_p(10)$, as long as there is a significant amount of betas with energies above 800 keV. Thus under certain circumstances, the effective dose could be limited by beta dose to the lens. Use of protective goggles with a density thickness of 700 mg cm^{-2} would make the lens dose equal to the measured $H_p(10)$ dose, even if some high energy betas were present.

When radiation fields are non-uniform, it is common in NPPs for workers to wear multiple dosimetry. With beta dose to the lens eliminated or very significantly reduced by sufficiently thick goggles, the dose is received from photons. Very often, pairs of dosimeter badges and electronic personal dosimeters (EPDs) are worn on the head and trunk. The effective dose, E, is usually calculated by combining the two results. At OPG, the effective dose is calculated from the head (H) and trunk (T) TLD badge results as shown in equation (1).

$$\mathbf{E = 0.11 * H + 0.89 * T} \qquad (1)$$

If it is assumed that the photon component of the lens dose, L, is equal to the $H_p(10)$ reading from the H dosimeter, equation (1) can be rewritten as equation (2), with R being the ratio of the H to T doses.

$$\mathbf{E/L = 0.11 + 0.89/R} \qquad (2)$$

About 1200 pairs of H and T TLD badge readings for the Darlington (DN) station and 1950 pairs for the Pickering (PN) station were analyzed, with the selection criteria that both the H and T readings had to be ≥ 0.1 mSv. The statistical analysis is shown in Fig. 1.

The average R value for the DN sample was 1.2 ± 0.2 (1 σ), while for PN the average R value was 1.3 ± 0.3. For the DN sample, the 99.5% confidence level interval (coverage factor k=3) yields an R value of about 2, while for the PN sample the corresponding R value is around 2.5. By substituting the most conservative R value found (2.5) into equation 2, the annual effective dose must stay below 9.3 mSv in order for the lens annual dose to comply with the recommended dose limit of 20 mSv. At some point a sufficiently high dose to the head or lens should limit E. However, it is not necessarily appropriate that this should occur to such an extent at relatively low ratios of H to T dose.

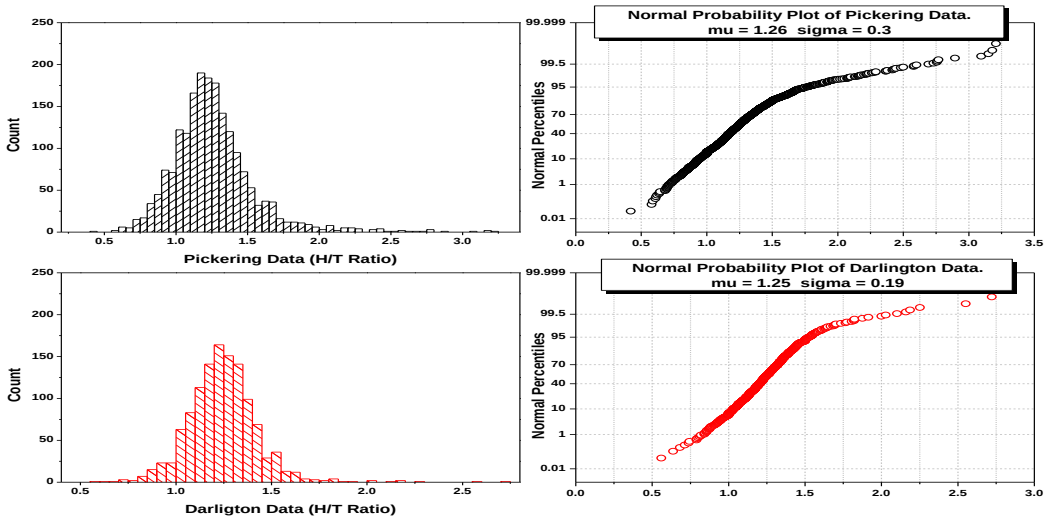

Fig. 1. OPG DN and PN stations: statistical analysis of head and trunk reading ratios (R value)

4. WHEN TO MEASURE OR ESTIMATE $H_P(3)$

With the lower lens dose limit, it will be necessary to determine the lens dose, either by estimation or the use of a lens dosimeter. For most exposure situations, it is typical to specify that dosimetry is worn when the dose exceeds some fraction of the annual dose limit. In Canada the Canadian Nuclear Safety Commission (CNSC) Radiation Protection Regulations require the use of a licensed dosimetry service to estimate doses for workers who have a "reasonable probability" of effective and committed effective doses of 5 mSv or more. For equivalent doses, it is OPG's practice to ensure that equivalent doses that are 10% of the annual dose limit are measured. With a 500 mSv annual extremity dose limit and relatively low extremity doses, it is routine to just add measured extremity doses to the whole body skin doses to obtain the total extremity dose. With the same dose limit for the lens and for effective dose, the double counting of the effective dose received while a lens dosiemter is worn is not accepatable. However, requiring separate lens dosimetry in uniform photon fields is also not appropriate, especially as the lens photon dose from a properly designed four-element

dosimeter will almost certainly be more accurate than that from a single element lens dosimeter. Unless sufficiently thick goggles are worn, a lens dosimeter will likely be required in uniform high energy beta fields, and also in non-uniform photon and beta fields. Because of the problem of double counting, this will require that a non-regular dosimeter badge or H and T pair of dosimeters must be worn whenever the lens dosimeter is worn.

5. REVISIONS TO RADIATION PROTECTION (RP) PROGRAMMES

In addition to the technical dosimetry issues outlined above, the following list comprises some of the changes that will be required to implement the requirements of the new lens dose limits:

i. Revision of facility and federal radiation dose record systems to include lens dose,
ii. Changes to federal or state/provincial requirements for testing and qualifying dosimeters,
iii. Workplace characterization for lens dose hazards,
iv. Development and testing of radiation survey instruments to measure $H_p(3)$ from betas,
v. Development and testing of thicker protective goggles, and
vi. Revision to RP procedures, Radiation Exposure Permits, ALARA programs, adminstrative dose levels, etc.

Some of these effects can be mitigated by effectively monitoring and controlling hazards to the lens such that lens dosimetry will seldom be required, and by maintaining effective doses sufficiently low that there is sufficient margin for the head to receive a higher dose than the trunk without exceeding the dose limit.

6. COMPARISON OF ICRP RISK DATA FOR CANCER AND CATARACT

The accepted risk for cancer that is the basis for the effective dose limit is about 4%/Sv [2]. For an effective dose of 20 mSv/y for 25 years = 500 mSv, the risk is 2%. But the risk of cataract for a 500 mSv dose is 1%. Given that in the developed world cataract is readily treatable, about 95% of the time, by a simple operation, the risk does not seem commensurate to that from a potentially fatal cancer. If the lens limit had been made 50 to 100% larger than the new recommended limit, it would have made it possible to reduce lens doses in NPPs through protective equipment without frequently requiring the use of lens dosimetry, along with the significant burden and costs this will require.

7. CONCLUSIONS

The new recommended eye lens dose limit is equal to the current effective dose limit. Meeting the new recommendations will require the ongoing advances in lens dosimetry to continue until it becomes a mature discipline, as well as requiring significant changes to RP programs in NPPs. A modestly higher lens limit would lower lens doses with a reduced cost.

REFERENCES

[1] INTERNATION COMMISSION ON RADIOLOGICAL PROTECTION, ICRP Statement on Tissue Reactions, ICRP Publication 118, PART 1, 21 (2011).

[2] INTERNATION COMMISSION ON RADIOLOGICAL PROTECTION, ICRP Publication 103, The 2007 Recommendations of the International Commssion on Radiological Protection, (2007).
[3] INTERNATION COMMISSION ON RADIOLOGICAL PROTECTION, 1980 Statement and Recommendations of the International Commission on Radiological Protection from Its 1980 Meeting (1980 Brighton Meeting). In: Annals of the ICRP 4 (1/2).
[4] BEHRENS, R., et al., $H_p(0.07)$ Photon Dosemeters for Eye Lens Dosimetry, Calibration on a Rod vs. a Slab Phantom, Radiation Protection Dosimetry, 148(2) (2011) 139-142.
[5] GUALDRINI, G., et al., Eye Lens Dosimetry: Task 2 Within the ORAMED Project, Radiation Protection Dosimetry, 144(1-4) (2011) 473-477.
[6] BORDY, J.M., et al., Principles for the design and calibration of radiation protection dosemeters for operational and protection quantities for eye lens dosimetry, Radiation Protection Dosimetry, 144(1-4) (2011) 257-261.
[7] ISO 4037, parts 1-3, X and gamma reference radiation for calibrating dosemeters and doserate meters and for determining their response as a function of photon energy, ISO, Geneva.
[8] BEHRENS, R., Air kerma to $H_p(3)$ conversion coefficients for a new Cylinder Phantom for Photon Reference Radiation Qualities, Radiation Protection Dosimetry, 151(3) (2012) 450-455.
[9] GUALDRINI, G., et al., Air kerma to $H_p(3)$ conversion coefficients for photons from 10 keV to 10 MeV, calculated in a Cylindrical Phantom, Radiation Protection Dosimetry, 154(4) (2013) 517-521.
[10] BEHRENS, R., Dose conversion coefficients for electron exposure of the human eye lens: Calculations including a Whole Body Phantom, Radiation Protection Dosimetry, 155(2) (2013) 224-235,
[11] EAKINS, J.S., et al., Monte Carlo modelling of $^{90}Sr/^{90}Y$ and ^{85}Kr beta fields for $H_p(3)$ measurements, Radiation Protection Dosimetry, 158(1) (2013) 115-121.
[12] GILVIN, P.J., et al., Type testing of a head band dosemeter for measuring eye lens dose in terms of $H_p(3)$, Radiation Protection Dosimetry, 157(3) (2013) 430-436.
[13] ISO 12794, Nuclear energy - radiation protection – individual thermoluminescence dosemeter for the extremities and eyes, ISO, (2000).
[14] CHASE, W.J., HIRNING, C.R., Application of radiation physics in the design of the Harshaw 8828 beta-gamma badge.
[15] WITHARANA, S.S.H., DUBEAU, J., HEINMILLER, B, Response of OPG/Bruce Power dosimeter elements determined from MCNP modelling of various beta spectra, COG TN-13-3011 (2014).

THE FIRST NATIONAL INTERCOMPARISON OF WHOLE BODY DOSEMETERS IN PHOTON FIELDS IN UKRAINE: PRELIMINARY RESULTS

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Abstract

Blind tests, specifically intercomparisons, proved to be extremely efficient quality assurance mechanism for personal dosimetry services and laboratories. The variety of broad national, international and bilateral intercomparisons is being practiced for decades leading to significant improvement of the accuracy and reliability of occupational dosimetry. Unfortunately, although about 80 laboratories in Ukraine are authorized for personal dosimetry measurements, there were no national intercomparisons so far. Only three Ukrainian laboratories ever took part in the intercomparisons organized by IAEA and EURADOS. Therefore, the First National Intercomparison of personal dosimetry labs was initiated by the State Nuclear Regulatory Inspeciton of Ukraine and organized and performed by institutes from Ukraine and Belgium in order to expose problematic areas in occupational dosimetry in Ukraine. Irradiation of dosimeters was performed in terms of $H_p(10)$ and $H_p(0.07)$ in photon fields using standard X-ray and gamma series in a range of doses and incidence angles compatible with real life situations. Irradiation of dosimeters was conducted in compliance with ISO-4037 standard. All participants are coded in order to secure unbiased treatment of dosimeters and judgement of results. However, actual names of dosimetry services will be available in the confidential version of final report available to regulators.

1. INTRODUCTION

A practice of international intercomparisons, established by IAEA [1] and EURADOS [2] proved to be an efficient tool for control and enhancement of the quality measurements of occupational exposure doses, which are performed by individual monitoring services and laboratories. Despite fulfilment of in-house quality procedures, very often laboratories demonstrate unsatisfactory or dubious results in course of intercomparisons, calling thus for modification of the measurement and data interpretations protocols or implementation of other corrective actions. Among of up to 80 laboratories, which are authorized for monitoring

of individual doses of occupational exposure in Ukraine, only three laboratories ever took part in the intercomparisons organized by IAEA (1996-2005) or EURADOS (2008-2014) and only one laboratory participated in all intercomparisons of these series. All other laboratories had never participated in neither international nor national intercomparisons.

Therefore, in order to provide unbiased evaluation of the quality of individual dosimetric monitoring measurements provided by laboratories in Ukraine, the First National Intercomparison of individual monitoring services was organized and carried out in Ukraine under auspices of the State Social Program of Enhancement of Safety, Occupational Hygiene and Industrial Environment adopted by the State Law in 2013. Coordination of the intercomparison was performed by the National Research Centre for Radiation Medicine (National Academy of Medical Sciences of Ukraine) with administrative and technical support from the State Nuclear Regulatory Inspection of Ukraine as well as by Belgian Nuclear Research Centre SCK-CEN (Mol, Belgium) and Kyivoblstandartmetrologiya metrological SSDL (Byela Tserkva, Ukraine).

2. METHODS

The design of the First National Intercomparison was elaborated in line with well proven principles, which are usually applied to this kind of tests [2]. The laboratories were invited to provide 30 dosimeters each to be irradiated in unannounced designated metrology labs to doses and radiation qualities unknown to participants. Upon irradiation and return to the dosimetry labs, the dosimeters were to be read out following the regular measurement protocols and results to be reported to coordinator. In order to secure blindness of the test and avoid any kind of bias, dosimeters were coded and submitted for irradiations in anonymous way (e.g. metrology labs were unaware of affiliation of particular dosimeters or their batches). All dosimeters were labeled by coordinator with bar-coded lebels in addition to regular marking used by the participating labs (Fig. 1). According to the intercomparison design, 24 dosimeters were used for calibrated irradiations (22 in Mol and 2 in Byela Tserkva), while remaining 6 dosimeters were not irradiated and were used as transport/background dosimeters (5 – Mol, 1 – Byela Tserkva). In order to minimize unwanted background dose, the time scale of intercomparioson was set very tight, so time between annealing/preparation of dosimeters and their readout after irradiation did not exceed 49 days (on average – about 37 days). In addition, in order to avoid x-ray inspection at post/customs/airport security, the dosimeters were land transported to the metrology labs for irradiations – by car to Byela Tserkva and by train-car-bus to Mol. Transporations to and from coordinator in Kyiv, where possible, was done by personal delivery or, if not feasible (9 labs), by express post delivery with ‘no x-rays’ marking of the parcels.

Irradiations were performed in attested metrology labs following ISO 4037 standard in photon fields according to the schedule presented in Table 1. Nominal doses (conventional true values) were listed in terms of $H_p(10)$ and $H_p(0.07)$, the participants were invited to evaluate deep and shallow personal dose equivalents using their dosimetry systems.

FIG.1. Dosimeter types tested in the First National Intercomparison and code labeling of the dosimeters

In view of limited number of dosimeters available for intercomparison and with respect to allocation of reasonable amount of time for irradiations, relatively few doses and radiation qualities were chosen in order to evaluate the following aspects: calibration of dosimetry systems, energy and angular response of dosimeters, ability to measure doses in practical range of occupational doses (simulation of emergency exposure was not an issue), scatter of results for identical irradiation conditions. More detailed study of particular aspects of dosimeter system performance can be conducted in course of the following national intercomparisons. Judgment of the results was conducted according to criteria of ISO 14146 using trumpet curves with H_o equal to 0.2 mSv.

It should be noted that nominal doses and radiation qualities applied to particular dosimeters were known only for metrology labs until the moment of receiving measurement protocols from all participating laboratories. Upon fixation of the incoming results, the coordinator ordered metrology labs to send coded lists of dosimeters with indication of radiation qualities and delivered doses. After that, the irradiation data was communicated to the participating dosimetry labs. Such rigorous attention allocated to coding and anonymization of data had secured absolute prevention of data leak and abuse.

TABLE 1. RADIATION QUALITIES AND PROVISIONAL DOSE LEVELS APPLIED TO DOSIMETERS WITHIN THE INTERCOMPARISON

Item #	Radiation quality	Incident angle	Provisional nominal $H_p(10)$, mSv	Number of dosimeters
1	S-Cs	0^o	0.5	2
2	S-Cs	0^o	2	3[a]
3	S-Cs	0^o	5	4
4	S-Cs	0^o	20	2
5	S-Cs	0^o	60	1
6	S-Co	0^o	5	3
7	N-60	0^o	5	3
8	N-60	45^o	5	2
9	N-60	60^o	5	2
10	N-40	0^o	5	2

[a] irradiated in two different calibration labs

3. RESULTS

According to the data by SNRI, in 2009 about 80 laboratories were authorized to perform individual dosimetric monitoring of occupationally exposed workers. This number includes both large dosimetry services (i.e. personal dosimetry labs of the nuclear power plants) and tiny in-house facilities operating several EPDs. After annexation of Crimea and unveiling of war in Eastern Ukraine, 30 laboratories became unavailable for engagement into intercomparison. The announcement of intercomparison was sent through local SNRI inspectorates and collected primary feedback from 33 dosimetry services and laboratories. Since the intercomparison was aimed at laboratories using passive personal dosimeters, several laboratories were not qualified for participation in the intercomparison, some other labs refused to participate due to problems with instrumentation or concerns regarding ability to match quality criteria. As a result, 19 dosimetry services with 20 different dosimetry systems took part in the intercomparison. Since one lab was late with sending dosimeters for calibration, at the end the dosimeters from 18 services/19 systems were irradiated and yielded into the results of intercomparison.

Participating dosimetry services provide dosimetric monitoring in various application areas (Fig.2) and are representative for the whole dosimetric monitoring domain. The total number of workers monitored by these services is 37,777 persons, which covers about 75% of estimated number of occupationally exposed personnel in Ukraine. The performance of dosimetry labs and ability to match ISO 14146 criteria varied by the type of the service. Breakdown of this number by the type/affiliation of the services, as well as success within the intercomparison are presented in Table 2.

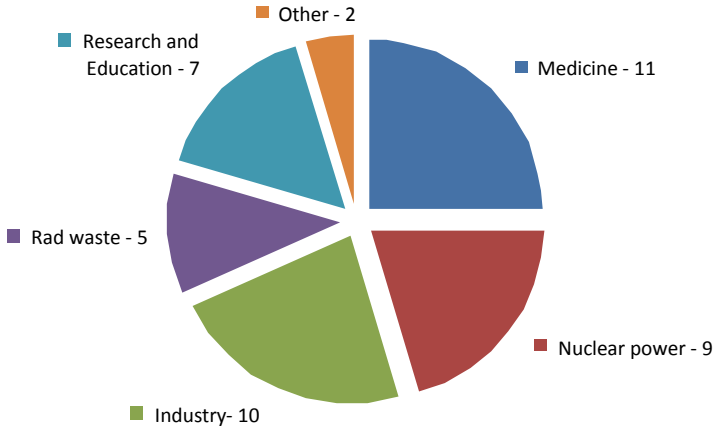

FIG.2. Areas of use of ionizing radiation covered by participating laboratories

All laboratories are qualified for measurement of whole body dose from photons in terms of $H_p(10)$, only eight laboratories can measure $H_p(0.07)$. Instrumentation used for dosimetric monitoring vary: 17 laboratories use TLD systems (7 – automated TLD readers, 10 – obsolete manual TLD readers of two different types) and one laboratory uses film dosimeters. Altogether, 7 different dosimeter types are used by participating laboratories (Fig.1). Eight of 19 dosimetry systems had matched ISO 14146 criteria. Results, demonstrated by the participants who did not match ISO 14146 criteria vary by the nature: seven laboratories demonstrated marginal performance (failure rate 15-20%) due to slight problems with energy/angular dependence of dosimeters and non-optimal calibration, four other laboratories demonstrate drastically wrong results. The results show good traceability of calibrations, acceptable type A uncertainty and, with exception of film dosimeters, good linearity of dose response.

TABLE 2. CHARACTERISTICS OF DOSIMETRY SERVICES PARTICIPATING IN THE FIRST NATIONAL INTERCOMPARISON

Affiliation of the service	Number of services	Services passed ISO 14146 critera	Number of monitored workers:			Total number of customers/ detached divisions	Maximum number of personnel within single customer	Minimum number of personnel within single customer
			Total	Quarterly monitoring	Monthly monitoring			
NPPs	4	3	13732	8074	5658	156	4100	1
Radiation facilities[1]	3	2	12686	514	12172	183	2041	2
Sanitary surveillance service	4	1	2353	2353	0	305	98	1
Hospitals	2	-	326	326	0	2	200	126
Research institutes	4	1	7751	7728	23	783	175	1
Other	1	-	623	623	0	74	114	1

[1]Including ChNPP and Object ‘Shelter’, Chornobyl restriction zone, Emergency Technical Center

4. CONCLUSIONS

The First National Intercomparison, which was carried out in Ukraine in 2014 marks a quantum leap in application of quality assurance of occupational dose measurements in this state. It was found that 8 critical dosimetry services match stringent international criteria, some more services could match the criteria after application of realistic corrective actions. Unfortunately, 4 laboratories (2 from oncology hospitals) demonstrated unacceptable results. It is anticipated that discussion of the results of the First National Intercomparison would lead to spotting the problems and correction of the weak elements in the occupational dose measurement process. There is an intention to conduct similar intercomparisons on a regular basis in a future.

REFERENCES

[1] IAEA TECDOC-1126, Intercomparison for individual monitoring of external exposure from photon radiation: Results of a co-ordinated research project 1996-1998. ISSN 1011-4289 (1999). http://www-pub.iaea.org/MTCD/publications/PDF/te_1126_web.pdf

[2] EURADOS Intercomparison 2008 for Whole Body Dosemeters in Photon Fields ISBN 978-3-943701-00-5.
http://www.eurados.org/~/media/Files/Eurados/documents/EURADOS_Report_201201.pdf

INACCURACY OF PERSONAL OSL DOSIMETERS IN INTERVENTIONAL RADIOLOGY

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Abstract

After a long period of film personal dosimetry in the Czech Republic, in 2008 there started also the OSL personal dosimetry. After some time many of the interventional radiologists found out that their personal doses were visibly higher after this change without changing of their practices. Since 2010 when the problem occurred first time, State Office for Nuclear Safety (SÚJB) made several researches to find the reason. It was a big angle dependence of the OSL personal dosimeters in the low energies and high angles (above 60°). The interventional radiologists have a significant probability that their personal dosimeters are exposed under a high and approximately constant angle (with low energies). This is caused by the geometry of the dosimeter, where the OSL elements are very close to each other, so at the high angles some radiation that went through one filter is detected in a different OSL element, or the element can be partly irradiated without going through the expected filter. Because a specific accuracy (to fit in the "trumpet curve") is internationally required only within angles +- 60°, the OSL personal dosimeters were calibrated only within these angles. But the real practice have shown that in interventional radiology, this angle is often exceeded (that led to an important overestimation of the doses, up to 5 times). So, the Czech personal dosimetry put in the calibration new data including the high angles in low energies. This solved the problem and finally it was ensured that the calibration didn't change within lower angles and higher energies.

1. INTRODUCTION

After a long period of film dosimetry in The Czech Republic, in 2008 there started also the OSL personal dosimetry. After some time many of the interventional radiologists found out that their personal doses are visibly higher after this change without changing of their practice. Since 2010 when the problem occurred first time, SÚJB made several researches including the laboratory measurements to find the reason.

2. METHODS

The dosimeters were irradiated to known doses under energies of scattered radiation in interventional radiology (IR) using several angles. There was a significant angle dependence of overestimating with higher angles, so the next laboratory measurement focused on extreme angles (but those which could practically appear in IR). The measurement confirmed that the dosimeters really strongly overestimate Hp(0.07) and Hp(10) in X-Ray energies and angles higher than 60° (as it is shown in Tables 1 and 2 and Fig. 1).

TABLE 1. OVERESTIMATING OF THE OSL PERSONAL DOSIMETERS DUE TO THE ORIGINAL CALIBRATION. $H_P(0.07)_R$ IS THE REAL $H_P(0.07)$ AND $HP(0.07)_M$ IS $H_P(0.07)$ MEASURED BY THE PERSONAL DOSIMETER.

Beam quality	60° from bellow			75° from bellow		
	$H_P(0.07)_R$ [mSv]	$H_P(0.07)_M$ [mSv]	$H_P(0.07)_M$/ $H_P(0.07)_R$	$H_P(0.07)_R$ [mSv]	$H_P(0.07)_M$ [mSv]	$H_P(0.07)_M$/ $H_P(0.07)_R$
N40	6.53	9.3	1.43	25	10.18	2.49
N40	6.53	6.6	1.01	31	10.18	3.07
N60	5.30	4.8	0.91	18	7.17	2.47
N60	5.30	4.8	0.90	16	7.17	2.21
N80	5.22	4.7	0.90			
N80	5.22	4.4	0.85			
N100	5.29	5.1	0.96	8.8	6.66	1.31
N100	5.29	4.6	0.86	8.4	6.66	1.27

TABLE 2. OVERESTIMATING OF THE OSL PERSONAL DOSIMETERS DUE TO THE ORIGINAL CALIBRATION. $H_P(10)_R$ IS THE REAL $H_P(10)$ AND $HP(10)_M$ IS $H_P(10)$ MEASURED BY THE PERSONAL DOSIMETER

Beam quality	60° from bellow			75° from bellow		
	$Hp(10)_R$ [mSv]	$Hp(10)_M$ [mSv]	$Hp(10)_M$/ $Hp\ (10)_R$	$Hp(10)_R$ [mSv]	$Hp(10)_M$ [mSv]	$Hp(10)_M$/ $Hp\ (10)_R$
N40	4.80	8.7	1.82	4.75	27	5.58
N40	4.80	6.2	1.29	4.75	31	6.51
N60	4.84	5.1	1.04	4.79	17	3.65
N60	4.84	5.0	1.02	4.79	16	3.39
N80	5.00	5.0	0.99			
N80	5.00	4.7	0.93			
N100	5.07	5.3	1.05	5.07	8.9	1.75
N100	5.07	4.8	0.95	5.07	8.7	1.71

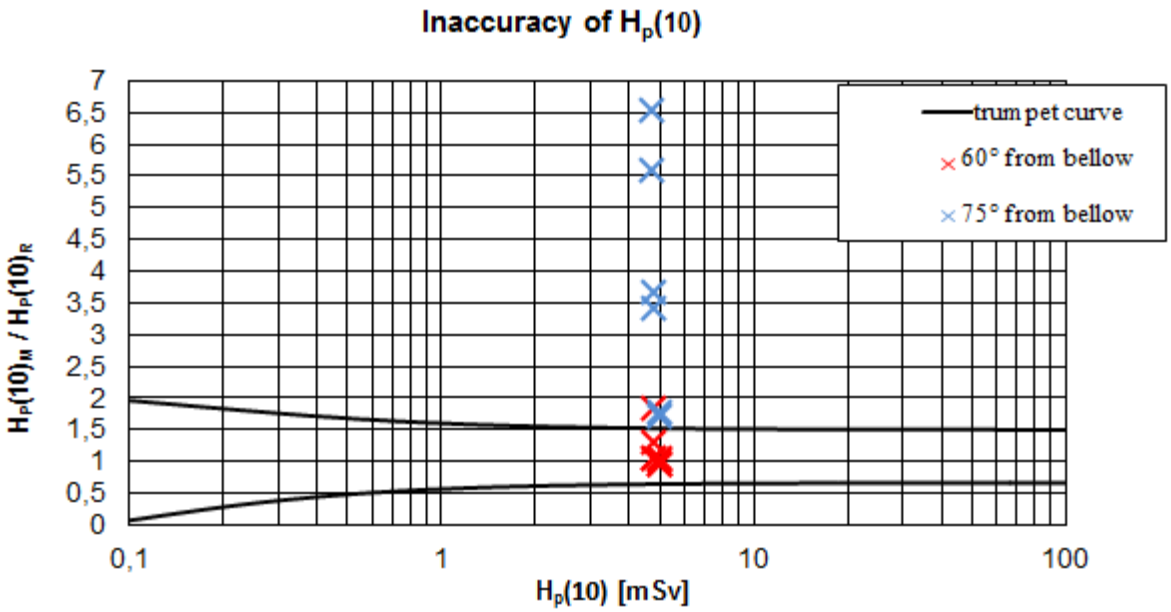

FIG. 1. Overestimating of the OSL personal dosimeters due to the original calibration

The OSL dosimeters are constructed for exposing within angles ±60°, but the use of the dosimeters is within higher angles and especially the interventional radiologists have a significant probability that their personal dosimeters would be exposed under a high and approximately constant angle (with low energies).

SÚJB made an inspection in the dosimetry service which uses the OSL dosimeters and the result was that the service must have changed the methodology of evaluating the doses and that they must have found a way how to compensate this high angle dependence.

The high angle dependence in low energy is caused by the design of the dosimeter: the four OSL elements are very close to each other and the different material filters are only above and under the elements. Within a high angle from up or below this, leads to that part of radiation which went through one filter is detected in a different OSL element (as it is shown in Fig. 2). And when it is irradiated from high angle from side, it causes that a part of the OSL element is irradiated without going through the expected filter. Both of these effects lead to wrong calculation of energy of the radiation that leads to wrong energy calibration. This caused the multiple overestimating of the doses (in some cases 5 times).

The original calibration of these dosimeters didn‘t expect irradiation in such angles, so it didn‘t contain the calibration data with low energy and high angles.

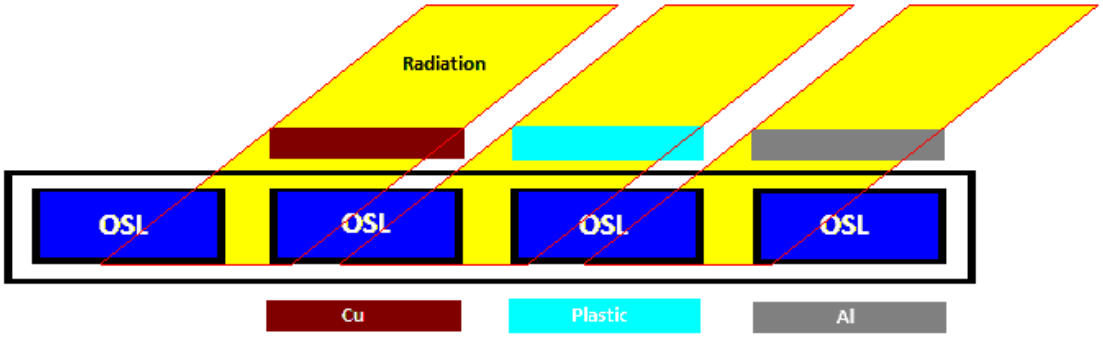

FIG. 2. Schematic illustration of the reason of the inaccuracy of the OSL personal dosimeters.

3. RESULTS

A new evaluating program was created using these calibration data and checked the same in laboratory measurement. This new calibration led to precise evaluation of Hp(0,07) and Hp(10) in the normal conditions (high energies, low angles), but also in these specific conditions which are in IR (all the measured doses were within the „trumpet curve“) – as it is shown in Table 3 and 4. and Fig. 3. So, the original evaluation was abandoned and started using their new one.

TABLE 3. ACCURACY OF THE OSL PERSONAL DOSIMETERS AFTER RECALIBRATION. $H_P(0.07)_R$ IS THE REAL $H_P(0.07)$ AND $H_P(0.07)_M$ IS $H_P(0.07)$ MEASURED BY THE PERSONAL DOSIMETER

Beam quality	Angle [°]	from	$H_P(0.07)_M$ [mSv]	$H_P(0.07)_R$ [mSv]	$H_P(0.07)_M$ / $H_P(0.07)_R$
N40	0	0	5.34	3.90	1.37
N40	30	bellow	4.32	4.06	1.06
N40	30	side	5.64	4.08	1.38
N40	50	bellow	5.54	4.51	1.23
N40	50	side	6.42	4.52	1.42
N40	60	bellow	4.92	6.52	0.75
N40	60	side	8.96	6.53	1.37
N40	70	bellow	9.24	6.37	1.45
N40	70	side	7.70	6.40	1.20
N40	75	bellow	9.89	10.19	0.97
N40	75	side	18.11	10.18	1.78
N60	0	0	5.37	4.51	1.19
N60	60	bellow	5.58	5.31	1.05
N60	60	side	5.98	5.30	1.13
N60	75	bellow	8.42	7.16	1.18
N60	75	side	11.15	7.17	1.56
N80	0	0	3.41	3.63	0.94
N80	30	bellow	3.58	3.69	0.97
N80	30	side	3.30	3.69	0.89
N80	50	bellow	3.32	3.90	0.85

N80	50	side	4.18	3.90	1.07
N80	60	bellow	6.67	5.22	1.28
N80	60	side	4.29	5.22	0.82
N80	70	bellow	7.04	4.76	1.48
N80	70	side	4.84	4.76	1.02
N100	60	bellow	5.61	5.30	1.06
N100	60	side	4.15	5.29	0.78
N100	75	bellow	7.04	6.68	1.05
N100	75	side	8.66	6.66	1.30

TABLE 4. ACCURACY OF THE OSL PERSONAL DOSIMETERS AFTER RECALIBRATION. $HP(10)_R$ IS THE REAL $H_P(10)$ AND $H_P(10)_M$ IS $H_P(10)$ MEASURED BY THE PERSONAL DOSIMETER

Beam quality	Angle [°]	from	$H_P(10)_M$ [mSv]	$H_P(10)_R$ [mSv]	$H_P(10)_M$ / $H_P(10)_R$
N100	60	bellow	5.40	5.08	1.06
N100	60	side	3.97	5.07	0.78
N100	75	bellow	5.35	5.08	1.05
N100	75	side	7.01	5.07	1.38
N120	0	0	1.97	1.99	0.99
N120	50	bellow	1.92	2.00	0.96
N120	50	side	2.19	2.00	1.10
N120	60	bellow	2.00	2.00	1.00
N120	60	side	2.00	2.00	1.00
N120	70	bellow	1.90	1.99	0.95
N120	70	side	2.06	1.99	1.04
N40	0	0	3.10	2.89	1.07
N40	30	bellow	3.69	3.70	1.00
N40	30	side	3.58	3.72	0.96
N40	50	bellow	3.90	4.03	0.97
N40	50	side	4.18	4.03	1.04
N40	60	bellow	4.10	4.01	1.02
N40	60	side	5.22	4.85	1.08
N40	70	bellow	3.91	4.01	0.97
N40	70	side	3.57	4.01	0.89
N40	75	bellow	4.89	4.75	1.03
N40	75	side	7.13	4.75	1.50
N60	0	0	4.01	4.83	0.83
N60	60	bellow	5.31	4.86	1.09
N60	60	side	4.03	4.95	0.81
N60	75	bellow	5.48	4.79	1.14
N60	75	side	7.01	4.79	1.46
N80	0	0	3.20	2.98	1.07
N80	30	bellow	3.21	3.99	0.80
N80	30	side	3.09	3.99	0.77
N80	50	bellow	3.25	2.99	1.09
N80	50	side	3.06	2.99	1.02
N80	60	bellow	3.20	2.99	1.07
N80	60	side	3.37	3.00	1.12
N80	70	bellow	3.05	2.98	1.02
N80	70	side	3.05	2.98	1.02

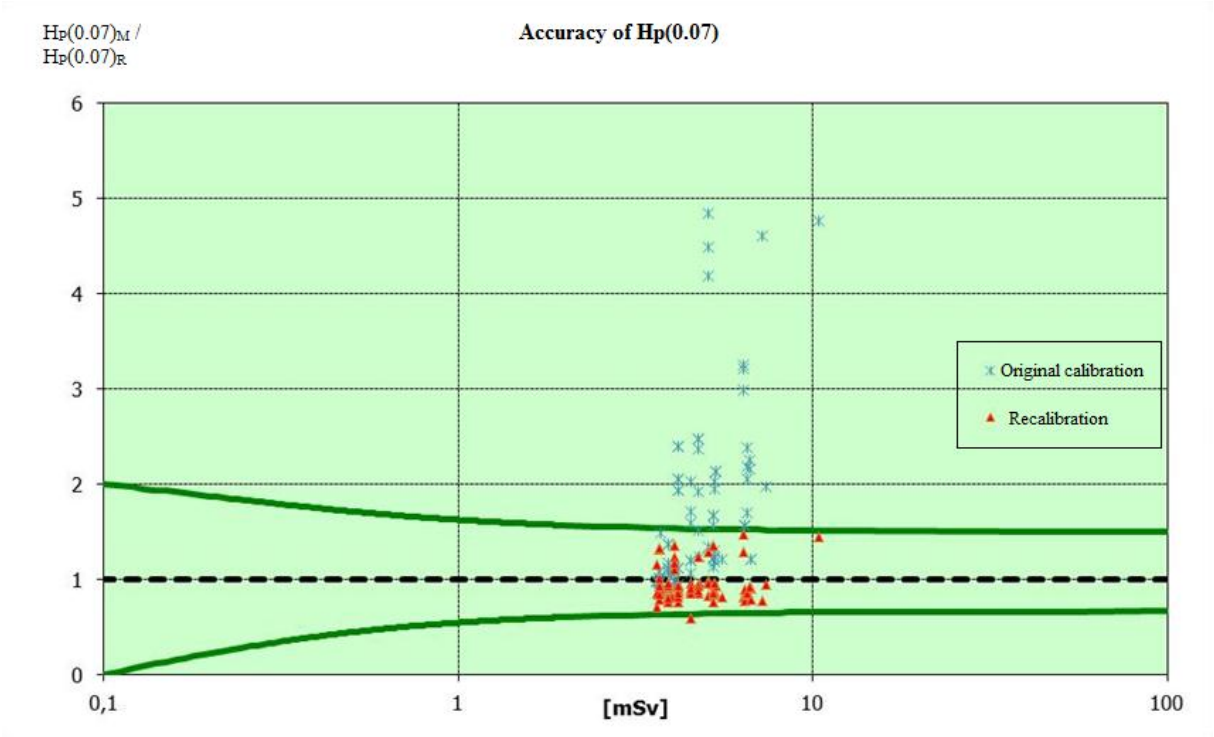

FIG. 3. Accuracy of the OSL personal dosimeters after recalibration

4. DISCUSSIONS

The same OSL dosimeters are used in many countries, but it seems, that no-one found this inaccuracy in low energies and high angles. The reason could be that The Czech Republic is quite specific in two facts – the OSL dosimetry started there quite recently but the film dosimetry has kept working – this has led to high concurrency when the competitors have tried to find mistakes of the other ones. There has been also a possibility of comparing the doses of one worker measured first using film and second OSL, while the worker hadn‘t changed his practice.

And the second specific fact is that personal body dosimeters should be worn on the protective apron in The Czech Republic without using second dosimeter under it – the personal dose is then recalculated using the attenuation coefficient of the apron for the effective energy that the dosimeter measured. Most of the departments have the investigation levels set to 20 mSv per year – when it is exceeded they make the recalculation. The inaccuracy of the OSL dosimeters led to a need of recalculation more often even for the doctors which hadn‘t needed it before.

These both probably led to noticing that there can be some problem with accuracy of these dosimeters. Although in the other countries the inaccuracy is probably the same for the high angles, but the personal doses under the apron changed from e.g. 0.2 mSv to 0.8 mSv - still so low under the investigation levels, so no-one noticed it and no-one started the investigation.

5. CONCLUSIONS

The international requirements for the accuracy of the personal dosimeters are limited only for the angles within ±60°, but this investigation showed that at least in IR/IC this is not enough and that it can lead to an important inaccuracy.

Session 3: Radiation effects and health risks from radiation exposure at the workplace

ENDOTHELIAL PROGENITOR CELLS IN PERIPHERAL BLOOD OF CARDIAC CATHETERIZATION PERSONNEL

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Abstract

The aim of the present study was to evaluate damage and rejuvenation capacity among cardiac catheterization (CC) staff occupationally exposed to ionizing radiation (IR) during fluoroscopic procedures. Subjects and Methods: Venous blood samples were obtained from 70 cardiac catheterization staff exposed to x-ray during fluoroscopy procedures at three busy hospitals in Cairo – Egypt vs. 40 controls. Blood was assayed for the frequency of micronuclei (FMN), plasma stromal growth factor (SDF-1) and cell phenotype of circulating endothelial progenitor cells (EPCs), whose surface markers were identified as the CD34, CD133 and kinase domain receptors (KDR). The individual annual collective dose information, as measured by thermoluminscent personal dosimeters (TLD), ranged between 2.16 and 14.9 mSv/y. Results: SDF-1α and FMN were significantly higher among CC staff compared to controls. CD34, CD133 and KDR were also significantly higher among CC staff compared to the controls. Smoking seemed to have a positive effect on the FMN and SDF-1, while negative on EPCs. Conclusion: It is found that among CC staff, the numbers of EPCs had increased indicating an increased capacity for tissue repair. This regenerative process is hindered by smoking, evidenced by increased levels of SDF-1 and decreased numbers of EPCs. Further studies are required to prove whether changes in of EPCs' levels can offer a reliable detection marker for radiation exposure.

1. INTRODUCTION

The complex and prolonged coronary interventional procedures has increased levels of IR exposure among CC staff, who due to close contact with patients, have the highest potential risk of receiving long-term exposures to low level IR [1, 2]. Published epidemiological literature suggests that there is significant association between high dose IR exposure and circulatory diseases [3, 4]. Vascular injury is recognized as the cause for late radiation-therapy morbidity, which is manifested as atherosclerosis in large vessels [5].

EPCs are circulating bone marrow-derived cell populations that appear to participate in vasculogenesis, vascular homeostasis and vascular repair [6, 7]. Levels of circulating EPCs have been correlated with endothelial function and atherogenic risk factors [8, 9]. It is suggested that endothelial injury in the absence of sufficient circulating EPCs induces the progression of cardiovascular disease [10, 11] and that the amount of circulating EPCs, measured in terms of circulating mononuclear cells that express CD34, CD133 and KDR surface markers, offer the ideal markers for assessing environmental stresses [12]. Recruitment of EPCs from remote locations such as the bone marrow into ischemic areas is promoted by the chemokine SDF-1 [13], which has been shown to be up regulated in many damaged tissues as part of the injury response [14] and subsequently contributes to ischemic neovascularization *in vivo* by augmenting EPCs recruitment to ischemic sites [15].

Little is known about the effects of IR on the levels of circulating EPCs among the occuationally exposed workers. Accordingly, the aim of the present study is to identify effect of IR on circulating EPCs and whether changes in EPCs levels can offer a sesitive measure for IR exposure. FMN was also measured since it is an established biological dosimeter for radiation exposure [16].

2. SUBJECTS AND METHODS

Venous blood samples were obtained from 70 CC staff exposed to IR during fluoroscopy procedures at Al Azhar, Ain Shams Universities and Heart National Institute, Cairo - Egypt. Mean age was (42.8 ± 5.2) years and the period of occupational exposure ranged 17.8 ± 6.5 years. Controls included 40 persons (mean of age 42 ± 4.8 years) not exposed to IR and socio-economic matching. They were non-smokers with no past history of exposure to ionizing radiations or chemicals. All participants were subjected to medical examination to evaluate their state of health. Those with chronic diseases were excluded together with those who have had infections during the last three months before the study.

2.1. IR Dose Measurements

Annual IR dose for CC staff members were measured using TLD cards, which were placed underneath the lead apron.

2.2. Biochemical Investigations

2.2.1. Flow cytometry for Circulating Progenitor cells

Peripheral mononuclear cells were isolated from blood by Ficoll density-gradient centrifugation. The isolated cells were labeled with PE (R-phycoenythrin)-conjugated CD133 antibody (MACS Milteny Biotech), CD34– FITC (MACS Milteny Biotech) and fluorescein conjugated KDR (R & D Systems). EPC numbers were determined by fluorescence-activated cell sorting (FACS) analysis [17]. Data expressed number of cells per 10^5 mononuclear cells.

2.2.2. SDF-1 by ELISA

Plasma levels of SDF-1 was measured using ELISA (R&D Systems). The results were expressed in pg/ml [18].

2.2.3. Frequency of Micronucleii

Peripheral blood lymphoocytes from patients and controls were cultured and binucleated cells were prepared according to the conventional technique of Fenech, 1985 [16]. 500 binucleated cells were scored for each person.

3. RESULTS

3.1. Subjects and controls baseline characteristics are summarized in Table 1. Mean and SD of estimated annual dose in every hospital are shown in Table 2.

TABLE 1. BASELINE FOR CHARACTERISTICS OF CARDIAC CATHETERIZATION STAFF AND CONTROLS

Characteristic	CC Staff	Controls	P Value
n	70	40	
Age (years)	42.8 ± 5.2	42 ± 4.8	NS
During of occupational exposure (years)	17.8 ± 6.5	–	S
Smoking habits	34 smokers	–	
	46 non-smokers	–	

TABLE 2. MEASUREMENT OF ANNUAL COLLECTIVE RADIATION DOSE IN THE STUDIED GROUP

	Embaba Heart Institute	Ain Shams University	Al Azhar University
Dose Range	2.16 – 8.44 m Sv/year	4.2-9.4 m Sv/year	4.4 ± 14.9 m Sv/year

3.2. Biochemical tests

SDF-1α (563.8 ± 70.9 *vs.* 494 ± 38.7 pg/ml) and FMN (19.9 ± 5.5 *vs.* 2.8 ± 1.4) were significantly higher among CC staff compared to controls. Similarly, EPCs per 10^5 mononuclear cells: CD34 (50.8 ± 7.5 *vs.* 44 ± 8.3), CD133 (52.25 ± 9.2 *vs.* 45.1 ± 9.4) KDR (44.6 ± 11.6 *vs.* 38.8 ± 7.4) were also significantly higher among CC staff compared to controls (table 3). Smoking induced a significant decrease among smokers compared to non-smokers in CD34 (37.8 ± 1.2 *vs.* 58.2 ± 6.4), CD133 (49.7 ± 3.8 *vs.* 54.5 ± 5.7) KDR (41.5 ± 8.8 *vs.* 53.4 ± 12.4) as shown in Fig. 1.

TABLE 3. ENDOTHELIAL PROGENITOR CELL SURFACE MARKERS PER 10^5 MONONUCLEAR CELL AND PLASMA SDF-1A IN BLOOD CARDIAC CATHETERIZATION STAFF COMPARED TO CONTROLS

	CD34	CD133	KDR	SDF-1 pg/ml	FMN/500 cell
CC staff	50.8 ± 7.5	52.25 ± 9.2	44.6 ± 11.6	563.8 ± 70.9	24.8 ± 4.5
Control	44 ± 8.3	45.1 ± 9.4	38.8 ± 7.4	494.4 ± 38.7	13.4 ± 6.4
	p<0.001	p<0.01	p<0.001	p<0.05	p<0.0001

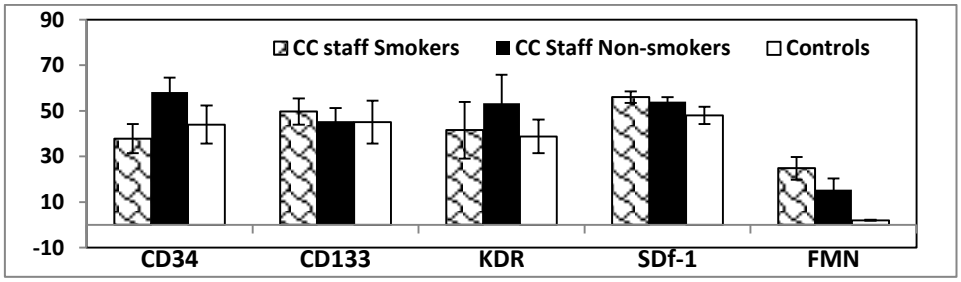

FIG. 4. Endothelial progenitor cell Surface markers per 10^5 mononuclear cell, SDF-1, FMN in blood of cardiac catheterization smoker male staff compared to non-smoker staff

4. DISCUSSION

The intention of the study was to find out whether, in spite of the required protection measures being observed during CC procedures, there is an alteration in endothelial progenitor mononuclear cells and chemokines in the CC staff occupationally exposed to IR in three hospitals in Cairo - Egypt. The physical exposure doses were within the range of accepted occupational dose limits of exposure to IR, which is 100 mSv every 5 years (i.e., 20 mSv per year), with a maximum of 50 mSv allowed in any given year [19]. However, these measurements are limited due to the fact that the TLD was placed beneath the lead apron on the chest of CC staff with a high workload. Additionally, CC staffs are exposed to scattered radiation, which results in non-uniform exposure doses. CC staffs receive high doses to the head and extremities that may be unshielded, which may increase the cumulative risk [20]. A dosimetry evaluation with multiple badges would be more accurate but less practical [21]. Biological dosimetry or biodosimetry, is mainly performed in addition to physical dosimetry, with the aim of individual dose assessment [22].

Results of the present study showed a significant increase in the FMN among CC staff compared to controls. Several previous *in vivo* studies indicated that chronic low doses of ionizing radiation can lead to significant somatic DNA damage in professionally exposed technicians. Italian studies carried out on intervention CC staff in Italy showed significantly increased levels of MN compared to controls [23, 24]. These studies related multiple risk alleles of DNA repair genes to inter-individual differences in radiation sensitivity and genetic susceptibility [24]. Similar results were indicated by Iranian, Korean, Egyptian and Japanese studies [25-27]. A biological dosimeter that measures true cellular injury resulting from radiation could be a more accurate indicator of cancer risk than a physical dosimeter [28].

The present study aimed at measuring circulating EPCs numbers and SDF-1 levels, which have recently been established as a specific and sensitive marker of endothelial activation and damage in a variety of vascular disorders [10]. SDF-1 in the present study was significantly higher among CC staff compared to controls. SDF-1 is considered as a part of host defense processes that protect stem cells from DNA-damaging agents including IR [10]. Human data regarding effect of IR levels of SDF-1 are not available. However animal studies showed that SDF-1α has been significantly increased in normal brain tissues of C57BL/6 mice after 15 Gy whole brain irradiation [29]. Sublethal IR has been shown to increase both the mRNA as well as the protein levels of SDF-1 in the bone marrow of infant C3H/He mice [30]. *In vitro* studies showed that a sub lethal dose of IR icreases the expression levels of SDF-1 mRNA significantly 24 and 48 hours after irradiating immature human osteoblasts and endothelial cells exposed to 5 Gy [31, 32]. It is believed that SDF-1 promoter region, in endothelial cells, has radiation-responsive sites [33].

The level of EPCs in the present study was significantly higher among CC staff compared to controls. Animal studies have shown that IR increases stem cell–active mobilization factors stimulating EPCs migration directly through the expression of SDF-1 [33, 34]. Overexpressiuon of SDF1 in the peripheral circulation results in the mobilization of subpopulations of hematopoietic cells with repopulating capacity such as progenitor cells and precursor cells [35]. Taking into consideration the previous information and the results of the present study that show increased plasma levels of SDF-1 in the study subjects, it could be concluded that those EPCs are therefore significantly increased.

The result in the present study that EPCs were significantly decreased among CC smokers is contradictory to another study [12], that observed a sustained increase in the number of EPC in male individuals, which was dependent on a cessation of smoking. Another study showed the correlation between blood levels of EPC and smoking status [10]. However,

in the present study there is a dual effect of IR and smoking. IR and smoking induce their effects by increasing both oxidative and nitrosative stress [36]. Data concerning levels of SDF-1 and smoking are scarce, however it is believed that DNA damaging agents such as IR induce the expression of SDF-1 [37]. Cigarette smoke is documented to be a DNA damaging agent [38]. The significant increase in SDF-1 among smoker CC staff indicates that exposure to both smoking and IR induce addetive damage. However this damage is not repaired due to a decrease in circulating EPCs.

In conclusion, the present work shows that exposure to IR even within permissible levels, stimulates regenerative processes as indicated by the increase in EPCs numbers and SDF-1 levels. This regenerative process is decreased by smoking as evident by increased levels of SDF-1 and decreased levels of EPCs. Since an increase in the number of EPCs in the peripheral blood is associated with vascular injury, repair and neovascularization [38, 39], further studies are required to investigate whether changes in of EPCs' levels can offer a reliable detection marker for radiation exposure.

REFERENCES

[1] DELICAS, M., et al., Radiation exposure to cardiac catheterization technicians performing interventional cardiology procedures. European journal of radiology, 48(3) (2003) 268–273.
[2] KIM K., et al., Occupational radiation doses to operators performing cardiac catheterization procedures. Health Physics, 94(3) (2008) 211–227.
[3] BORGHINI, A., et al., Ionizing radiation and atherosclerosis: current knowledge and future challenges, Atherosclerosis, 230(1) (2013) 40-7.
[4] METZ-FLAMANT, C., et al., Low doses of ionizing radiation and risk of cardiovascular disease: a review of epidemiological studies, Rev. Epidemiol. Sante. Publique 57 (2009) 347–359.
[5] DARBY, S., et al., Radiation-related heart disease: current knowledge and future prospects, Int. J. Radiat. Oncol. Biol. Phys. 76 (2010) 656–665.
[6] KHAKOO A., FINKEL T., Endothelial progenitor cells, Annu. Rev. Med. 56 (2005) 79-101. Review.
[7] FOLKMAN, G., Role of angiogenesis in tumor growth and metastasis, Semin Oncol. 6 (Suppl 16) (2002) 2915- 2918.
[8] HUANG, P., et al., Vascular endothelial function and circulating progenitor cells in patients with cardiac syndrome X, Heart 93 (2007) 1064–1070.
[9] SCHMIDT-LUCKE, C., et al., Reduced number of circulating endothelial progenitor cells predicts future cardiovascular events: proof of concept for the clinical importance of endogenous vascular repair, Circulation. 111 (2005) 2981–2987.
[10] HILL, J., et al., Circulating endothelial progenitor cells, vascular function and cardiovascular risk, N. Engl. J. Med. 348 (2003) 593–600.
[11] VASA, M., et al., Number and migratory activity of circulating endothelial progenitor cells inversely correlate with risk factors for coronary artery disease, Circ. Res. 89 (2001) E1–7.
[12] KONDO, T., Smoking cessation rapidly increases circulating progenitor cells in peripheral blood in chronic smokers, Arterioscler. Thromb. Vasc. Biol. 24(8) (2004) 1442-1447.
[13] HATTORI, K., et al., The regulation of hematopoietic stem cell and progenitor mobilization by chemokine SDF-1, Leuk. Lymphoma 44 (2003) 575–582.
[14] ASKARI, A., et al., Effect of stromal-cell-derived factor 1 on stem-cell homing and tissue regeneration in ischaemic cardiomyopathy, Lancet 362 (2003) 697–703.
[15] YAMAGUCHI, J., et al., Stromal cell-derived factor-1 effects on ex vivo expanded endothelial progenitor cell recruitment for ischemic neovascularization, Circulation 107 (2003) 1322–1328.
[16] FENECH, M., Measurement of micronuclei in lymphocytes, Mut. Res. 147 (1985) 29-36.
[17] LO, K., et al., Automated gating of flow cytometry data via robust model-based clustering, Cytometry Part A 73A (4) (2008) 321–332.
[18] LENG, J., et al., Elisa multiplex technologies for cytokine measurement in inflammation and ageing research, J.Gerontol a Biol Sci Med Sci 63 (8) (2008) 879–884.

[19] DELICHAS, M., et al., Radiation exposure to cardiac catheterization technicians performing interventional cardiology procedures. European journal of radiology 48(3) (2003) 268–273.
[20] LITTLE, M., A model of cardiovascular disease giving a plausible mechanism for the effect of fractionated low-dose ionizing radiation exposure. PLoS Comput Biol. 5(10) (2009)e1000539.
[21] CHIDA, K., et al., Radiation dose and radiation protection for patients and physicians during interventional Procedure, J. Radiat. Res. (Tokyo).51(2) (2010) 97-105, Review.
[22] MCNAMEE, J., et al., Validation of the cytokinesis-block micronucleus (CBMN) assay for use as a triage biological dosimetry tool. Radiat. Prot. Dosim. 135(4) (2009) 232-42.
[23] ANDREASSI, M., et al., Genetic polymorphisms in XRCC1, OGG1, APE1 and XRCC3 DNA repair genes, ionizing radiation exposure and chromosomal DNA damage in interventional cardiac catheterization technicians. Mutat. Res. 666(1-2) (2009) 57-63.
[24] ANDREASSI, M., et al., Somatic DNA damage in interventional cardiac catheterization technicians: a case- control study, FASEB J., 19(8) (2005) 989-999.
[25] ZAKERI, F., and Assaei, R., Cytogenetic monitoring of personnel working in angiocardiography laboratories in Iran hospitals. Mutat. Res. 562 (2004) 1-9.
[26] AN, M., and KIM, T., Frequencies of micronuclei in peripheral lymphocytes in Korean populations after chronic low-dose radiation exposure. J. Vet. Sci., 3(3) (2002) 213-218.
[27] KORRAA, S., et al., Markers of Apoptosis and Lymphocytes phenotypes among persons occupationally exposed to IR. In Rapid Diagnosis in Populations at risk to ionizing radiation and Chemicals, Cebulska-Wasilewska eds. (2010), IOS Press.
[28] MURIHEAD, C., et al., Mortality and cancer incidence following occupational radiation exposure: third analysis of the National Registry for Radiation technicians, Br. J. Cancer 100 (2009) 206–212.
[29] BOYACI, B., et al., http://www.ncbi.nlm.nih.gov/pmc/articles/PMC2634664/Evaluation of DNA damage in lymphocytes of cardiac catheterization technicians exposed to radiation during cardiac catheterization by the COMET ASSAY, Jpn. Heart J. 45(5) (2004) 845-853.
[30] ZHOU, W. Irradiated normal brain promotes invasion of glioblastoma through vascular endothelial growth and stromal cell-derived factor 1α. Neuroreport. 24(13) (2013) 730-734.
[31] ARIYOSHI, K., et al., Age dependence of hematopoietic progenitor survival and chemokine family gene induction after gamma irradiation in bone marrow tissue in C3H/He mice. Radiat. Res. 181(3) (2014) 302-13.
[32] GARCIA-MOROJA , C., et al., Functional characterization of SDF-1 proximal promoter. J Mol Biol 348(1) (2005) 43-62.
[33] PONOMARYOVA, T., et al., (2000) Induction of the chemokine stromal-derived factor-1 following DNA damage improves human stem cell function. J. Clin. Invest. 106(11) (2000) 1331-1339.
[34] LERMAN, O., et al., Low-dose radiation augments vasculogenesis signaling through HIF-1-dependent and –
independent SDF-1 induction, Blood, 116(18) (2010) 3669-3676.
[35] CERADINI, D., Progenitor cell trafficking is regulated by hypoxic gradients through HIF-1 induction of SDF-1,
Nat. Med. 10 (2004) 858–864.
[36] CHANG, C., Dose dependent effect of radiation on angiogenic and angiostatic CXC chemokine expression in
human endothelial cells. Cytokine 48(3) (2009) 295-302.
[37] PRYOR, W., et al., Fractionation of aqueous cigarette tar extracts: fractions that contain the tar radical cause DNA damage, Chem. Res. Toxicol. 11(5)(1998) 441-448.
[38] BIJIL, M., et al., Effect of smoking on activation markers, Fas expression and apoptosis of peripheral blood Lymphocytes, Eur. J. Clin. Invest. 31(6) (2001) 550-553.
[39] HUNTING, C., Circulating endothelial (progenitor) cells reflect the state of the endothelium: vascular injury, repair and neovascularization, Vox Sang. 88(1) (2005) 1-9, Abstract

OCCUPATIONAL RADIATION EXPOSURE, DNA DAMAGE AND GENETIC POLYMORPHISMS IN DNA REPAIR GENES

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Abstract

The aim of this study was to determine the relationship between genetic polymorphisms in genes coding DNA repair enzymes in different pathways; XRCC1 (Arg194Trp and Arg399Gln), OGG1 (Ser326Cys), APE1 (Asp148Glu), XRCC3 (Thr241Met) and XPG (Asp1104His) and the levels of DNA damage investigated by micronucleous (MN) assay in interventional cardiology staff. MN frequency was significantly higher in the exposed group (n=90) than in the unexposed control group (n=60); (27.7±17.2 ‰ vs. 8.0±2.6‰, $p < 0.0005$). Significant difference in MN frequency was also observed between occupational radiation doses of less than and more than 3 mSv/y ($p = 0.002$). Age was associated with increased MN frequency in the both groups. Within the exposed group, individuals carrying a XRCC3 Met241 allele with exposure >10 years had higher MN frequency in comparison to those exposed <10 years (39.5±17.5‰ vs. 24.2±12.3‰, $p = 0.001$, $F=5.2$). An interactive increasing effect on the MN frequency was found in the exposed group carrying both XRCC3 Met241 and XPG His1104 alleles than those carrying wild alleles ($p = 0.005$). Within the control group, individuals carrying a homozygous OGG1 Ser326 allele (wild type) had lower MN frequency than variant of OGG1 Cys326 (7.2±2.8‰ vs. 8.7±2.3‰, $p= 0.04$). These polymorphisms may be important in determining an individual's ability to repair cellular DNA after ionizing radiation exposure, and therefore, to modulate the toxicological outcome.

1. INTRODUCTION

Radiation exposure causes a variety of DNA damages and misrepair of DNA damages can lead to chromosomal aberrations, mutagenesis and carcinogenesis. Inherited Single-nucleotide polymorphisms (SNPs) in DNA repair genes are thought to modify the effects of low-dose radiation exposure on DNA damage and may account for the inter-individual differences to radiation sensitivity doses and cellular response to ionizing radiation. Radiation exposure is a significant concern for medical staff in cardiac catheterization laboratories not only because workloads and the complexity of procedures have increased over the past few years, but also many interventional procedures could have long screening times and multiple image acquisition that involve considerable exposure to ionizing radiation and enhance the risk of deterministic effects in both staff and patients [1].

DNA is a huge molecule that can be damaged by radiation (or the free radicals it produces). However the damage can be repaired through several ways such as base excision repair (BER), homologous recombination repair (HRR), nucleotide excision repair (NER) and mismatch repair (MMR) systems. Different kinds of gene products play roles in these DNA repair systems [2].

In literature it has been well documented that defects in DNA repair are associated with human disorders and an increased risk of developing various kinds of cancer. Despite the lack of a pathological phenotype, humans bearing variant alleles of DNA repair genes could show a different individual response to DNA damage. The principal source of inter-individual variability is represented by genetic polymorphisms. The presence of polymorphic alleles in DNA repair genes may alter the repair capacity modifying the biological responses to exogenous and endogenous DNA insults, both at cellular and tissue level, and the individual susceptibility in developing different kind of disease, such as cancer [3].

As MN can arise from exposure to various clastogenic agents in the form of acentric chromosome fragments, as well as to aneugenic agents as whole chromosomes, they are not radiation specific and also show inter-individual variability. However, because ionizing radiation is a strong clastogenic agent, and thus a potent inducer of MN, the cytokinesis blocked micronucleus (CBMN) assay has been validated as an appropriate biological dosimetry tool to evaluate in vivo radiation exposure of occupational, medical and accidentally exposed individuals and to assess in vitro radiosensitivity and cancer susceptibility [4]. On the other hand, several association studies have recently addressed the link between DNA repair polymorphisms and MN induction.

The involvement of hOGG1, XRCC1and XRCC3 gene products in the repair of oxidized bases, single strand breaks (SSBs) and double strand breaks (DSBs), respectively, is well documented. Moreover, despite some controversial results, genetic variants in hOGG1, XRCC1 and XRCC3 genes have been associated with cancer risk [5]. The XPG gene product has a role in nucleotide excision repair. It has been previously reported that the polymorphisms of this gene are related to lung and prostate cancers [2, 6]. Therefore in the present study, we investigated the potential links between genetic polymorphisms in genes coding DNA repair enzymes in different pathways and the levels of DNA damage investigated by micronucleous assay in interventional cardiology staff.

2. METHODS

2.1. Study population

The study population comprised 150 subjects:90 interventional cardiologists, technicians and nurses (59 male and 31 female, mean age 41.5±7.6 years) who operate in high volume cardiac catheterization laboratories, and 60 individuals (38 male and 22 female, mean age 41.4±9.1 years) working in the same hospitals without radiation exposure as the control group. Exclusion criteria for both exposed and control subjects were personal medical history of disease, cancer, or recent infectious state. All subjects gave their informed consent before entering the study. The Ethical Committee approved the study. All participants were asked to fill in a standardized questionnaire. Staff dosimetry for the last 1 and 5 years was obtained by a monthly film bage dosimeter. Dosimeter was located under lead apron over the chest.

2.2. Cytokinesis-block micronucleus test

Two separate cultures from each sample were set up by mixing 0.5mL of whole blood with 4.5mL of RPMI 1640 medium: the cultures were incubated at 37◦C for 72 h. Cytochalasin B (6µg/ml) was added 44 h after culture initiation. Cells were then harvested and fixed according to the standard method [7]. For each culture, 1000 binucleated cells were scored under optical microscope (final magnification 400×) for micronucleus analysis, following the criteria for micronucleus acceptance.

2.3. PCR-RFLP genotyping assays

Genomic DNA was extracted from peripheral blood leukocytes. Genetic polymorphisms were analyzed by PCR combined with restriction fragment length polymorphism (RFLP) .PCR products were digested with specific restriction enzymes that recognized and cut either the wild-type or variant sequence site. Details of primers sequence, annealing temperature, restriction pattern, and restriction enzymes used for each genotyping assay are listed in Table 2. The digested PCR products were analyzed on 10% polyacrylamide gels and 4% ultra pure agarose gel and stained with Gel red (Biotium, USA). Genotype results were regularly confirmed by random repetition of the samples. Allele frequencies obtained for the analyzed genes were consistent with literature data obtained for Caucasian population.

2.4. Statistical analysis

Statistical analyses of the data were conducted with SPSS package, version 17. Data are expressed as mean (±SD). Because of the skewness of the distributions of MN values, analyses were performed by using logarithmic transformation of data. Differences between the means of the two continuous variables were evaluated by the Student's t-test. The data for different groups were analyzed by ANOVA, the level of significance was set at $p < 0.05$ was considered for all statistical analyses. Combinations of the various genotypes were tested to check for interactions amongst themselves or with exposure, Age groups, years of employment and sex.

3. RESULTS

The main characteristics of the study subjects are shown in Table 1. There were no statistically significant differences in the genotype distribution between exposed group and control. Over the last year, and the last 5 years, the mean cumulative radiation doses recorded by monthly dosimeters under apron were 3.5±2.7mSv and 11.2±10.5 mSv for the exposed group, respectively.

TABLE 1. GENERAL CHARACTERISTICS OF THE EXPOSED AND CONTROL GROUPS.

	Interventional Cardiology staff	Control	*p* value
male	59	38	0.48
female	31	22	
Mean age, years± SD	41.5±7.6	41.4±9.1	0.93
Mean years of employment, years± SD	9.5±6.7	13.8±7.7	
Mean last year exposure mSv± SD	3.5±2.7	-	
Mean last five years exposure mSv± SD	11.2±10.5	-	

MN frequency was significantly higher in the exposed group than in controls (27.7±17.2 ‰ vs. 8.0±2.6‰, $p<0.0005$). Significant difference in MN frequency was also observed between occupational radiation doses of less than and more than 3 mSv/y ($p = 0.002$). Age was associated with increased MN frequency in the both groups. Also MN values were higher in the exposed group with exposure >10 years in comparison to exposed ≤10 years ($p = 0.04$). Within the exposed group, individuals carrying a XRCC3 Met241 allele with exposure >10 years had higher MN frequency in comparison to those exposed <10 years (39.5±17.5‰ vs. 24.2±12.3‰, $p = 0.001$, $F=5.2$). An interactive increasing effect on the MN frequency was found in the exposed group carrying both XRCC3 Met241 and XPG His1104 alleles than those carrying wild alleles ($p = 0.005$). Within the control group, individuals carrying a

homozygous OGG1 Ser326 allele (wild type) had lower MN frequency than variant of OGG1 Cys326 (7.2±2.8‰ vs. 8.7±2.3‰, p= 0.04).

TABLE 2. DETAILS OF PCR AND RFLP PROCEDURES

Gene	SNPs	Primers 5'-3'	Annealing Temp.	Restriction Enzymes	Fragment sizes (bp)
XRCC1	R194W	F: GCC AGG GCC CCT CCT TCA A R:TAC CCT CAG ACC CAC GAG T	62°C	Pvu II	485(R/R),396+89(W/W) 485+396+89(R/W)
	R399Q	F:AGT AGT CTG CTG GCT CTG G R:TCT CCC TTG GTC TCC AAC CT	56°C	Msp I	269+133(R/R), 402(Q/Q) 402+269+133 (R/Q)
APE1	D148E	F:CTG TTT CAT TTC TAT AGG CTA R:AGG AAC TTG CGA AAG GCT TC	54°C	FspB I	164(D/D), 144+20(E/E) 164+144+20(D/E)
OGG1	S326C	F:ACT GTC ACT AGT CTC ACC AG R:CCT TCC GGC CCT TTG GAA C	58°C	Sat I	156(S/S), 100+56(C/C) 156+100+56(S/C)
XRCC3	T241M	F:GGT CGA GTG ACA GTC CAA AC R:TGC AAC GGC TGA GGG TCT T	60°C	Hin1 II	315+140(T/T) 210+140+105(M/M) 315+210+140+105(T/M)
XPG	D1104H	F:TGG ATT TTT GGG GGA GAC CT R:CGG GAG CTT CCT TCA CTG AGT	62°C	Hin1 II	159(D/D),100+59(H/H) 159+100+59(D/H)

4. DISCUSSION AND CONCLUSION

In this study, as reported by others, occupational exposure to ionizing radiation in interventional cardiologists, technicians and nurses is associated to an increased DNA damage expressed as MN in human lymphocytes, that is early predictor of cancer [1, 8]. Large-scale biomonitoring studies of radiation workers, radiation accidents and patient studies have shown a clear dependence of MN formation on the accumulated dose and MN-derived dose estimates were in striking agreement with dose values obtained from dicentric studies [4]. These many applications of the CBMN assay highlight the important role of this test in assessing radiation exposure but also its shortcomings. We studied polymorphisms of the DNA repair genes XRCC1, OGG1, APE1 involved in base excision repair, XRCC3 involved in recombination repair and in maintaining chromosomal stability and XPG involved in nucleotide excision repair pathways. In agreement with our results it has been reported that repair capacity is slower in individuals with Ser/Cys or Cys/Cys OGG1 genotypes compared to those with the Ser/Ser OGG1 genotype, MN frequencies increased with age and the cumulative dose of radiation, genetic polymorphisms in XRCC1 resulted in higher residual DNA values and the Met/Met variant of XRCC3 resulted in an increased frequency of micronuclei [9]. Andreassi et al. [10] also reported that the MN frequency was significantly higher in interventional cardiologists than in clinical physicians and within the exposed group, individuals carrying a XRCC3 Met241 allele had higher frequency than homozygous XRCC3 Thr241 . Angelini et al. [11] has reported that radiological workers with variant alleles for XRCC1 or XRCC3 polymorphisms or wild-type alleles for XPD exon 23 or 10 polymorphisms showed a significantly higher MN frequency than controls with the same genotypes.

Recently, polymorphisms in DNA repair genes, such as XPD and XPG, have been reported for association with lung, prostate and breast cancer and besides acting as the excision nuclease in nucleotide excision repair, XPG also stimulates base excision repair of oxidative DNA damage and could play a role in transcription [2].

Results of these studies show that these polymorphisms may be important in determining an individual’s ability to repair cellular DNA after ionizing radiation exposure, and therefore, to modulate the toxicological outcome. And individuals with XRCC3, XPG and OGG1 polymorphisms who are exposed to ionising radiation represent a specific population requiring closer medical surveillance because of their increased mutagenic/carcinogenic risk.

In view of the importance of DNA damage repair genes and their involvement in cell and tissue responses to radiation, SNPs occurring in such genes affect health, disease and can cause an individual to have exceptionally high risk for the development of cancer. Therefore apart from radiation protection tools and advices to reduce or minimize the occupational radiation dose, it may be very useful to apply new approaches which would allow the screening of radiation workers for radiosensitivity. An approach is to use biomarkers like MN to elucidate the biological effects and to relate such effects with the SNPs in genes involved in DNA damage repair in order to predict an individual's probable radio-response and to direct the positive cases to more relevant screening for cancer susceptibility. This will help in developing better occupational radiation protection program.

REFERENCES

[1] ZAKERI, F. et al., Biological effects of low-dose ionizing radiation exposure on interventional cardiologists, Occup. Med. (Lond), 60 (6) (2010) 464-9.
[2] NATERI, K.F. et al., Bladder Cancer and Polymorphisms of DNA Repair Genes (XRCC1, XRCC3, XPD, XPG, APE1, hOGG1), Anticancer Researh 29 (2009)1389-1394.
[3] STERPONE, S., COZZI, R., Influence of XRCC1 Genetic Polymorphisms on Ionizing Radiation-Induced DNA Damage and Repair, J. Nucleic Acids (2010) 6 pages.
[4] THIERENS, H., VRAL, A., The micronucleus assay in radiation accidents, Ann Ist Super Sanità 45, 3 (2009) 260-264.
[5] RALUCA, A., et al., hOGG1326, XRCC1399 and XRCC3241 polymorphisms influence micronucleus frequencies in human lymphocytes in vivo, Mutagenesis 23, 1 (2008) 35–41.
[6] BERHANE, N. et al., Repair genes polymorphism (XPG and XRCC1) and association of prostate cancer in a north Indian population, Mol. Biol. Rep. 39,3 (2012) 2471-9.
[7] FENECH, M., The cytokinesis-block micronucleus assay: a detailed description of the method and its application to genotoxicity studies in human populations, Mutat. Res. 285 (1993) 35–44.
[8] ANDREASSI, M.G., et al. Somatic DNA damage in interventional cardiologists: a case-control study, FASEB J 19 (2005) 998–999.
[9] AKA, P. et al., Are genetic polymorphisms in OGG1, XRCC1 and XRCC3 genes predictive for the DNA strand break repair phenotype and genotoxicity in workers exposed to low dose ionising radiations? Mutat. Res. 556 (2004) 169–181.
[10] ANDREASSI, M.G., et al. Genetic polymorphisms in XRCC1, OGG1, APE1 and XRCC3 DNA repair genes, ionizing radiation exposure and chromosomal DNA damage in interventional cardiologists, Mutat. Res. 666 (2009) 57–63.
[11] ANGELINI, S. et al., Micronuclei in humans induced by exposure to low level of ionizing radiation: influence of polymorphisms in DNA repair genes, Mutat. Res. 570 (2005) 105–117.

Session 4: Dose record management of occupational radiation exposures

FIFTEEN YEARS OF OCCUPATIONAL EXPOSURE MONITORING IN FEDERATION OF BOSNIA AND HERZEGOVINA

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Abstract

Monitoring of exposed workers in Bosnia and Herzegovina started in 1960s. After brief interruption in 1990s, the dosimetry service resumed in 1999 after International Atomic Energy Agency (IAEA) provided first TLD reader in Bosnia and Herzegovina. Until 2013, Radiation Protection Centre of the Institute of Public Health of Federation of Bosnia and Herzegovina (RPC) was the only institution in the country that was capable of providing this service. In December 2013 RPC covered 1,485 of exposed workers with personal dosimetry, which is more than 70% of all radiation workers in the country. Most of the TLD users work in medical institutions – 1,417. Other occupations include industry, veterinary medicine etc. Total number of TLD users who were evaluated for annual doses is approximately 15,000. Majority of the annual doses received were less than 0.99 mSv y^{-1} (96%), some users received doses 1.00–1.99 mSv y^{-1} (3.3%), and very few doses between 2.00 and 2.99 mSv y^{-1} (0.6%). In isolated cases, TLD users received doses higher than 3 mSv y^{-1}. There are no registered cases of exceeding the annual limit (20 mSv y^{-1}). Exposed workers performing interventional procedures in radiology, cardiology, cardiac surgery and gastroenterology (cca.90 persons) are provided with 2 TLDs that are worn below and above the lead apron. In such cases, Niklason's methodology is used for the estimation of effective dose. Results of analysis show improvement in radiation protection in the last 5 years, which is most likely due to active involvement of the State Regulatory Agency for Radiation and Nuclear Safety (SRARNS).

1. INTRODUCTION

There is a wide variety of situations in which people at work are exposed to ionizing radiation. The conventional definition of occupational exposure to any hazardous agent includes all exposures incurred at work, regardless of source [1]. However, to distinguish the exposure that should be subject to control by the operating management from the exposure arising from the general radiation environment, the term `occupational radiation exposure' is taken to mean those exposures that incurred by workers in the course of their work and that can reasonably be regarded as the responsibility of the operating management [2, 3]. Such exposures are normally subject to regulatory control. Usually, the exposures are determined

by individual monitoring, and an important objective is to provide information on the adequacy of protection measures, as they serve as key inputs to operational decision to the optimization principle.

Individual monitoring of exposed workers is a requirement of international standards and national regulation of Bosnia and Herzegovina (BiH) [2, 4]. Personal dosimetry in BiH started in 1960s. It was interrupted in 1990s, and continued in 1999 after International Atomic Energy Agency (IAEA) donated a thermoluminescent dosimetry reader and a set of thermoluminescent dosemeters (TLD) [5]. Until recently, the only institution that provided personal dosimetry service in the country was the Radiation Protection Centre (RPC) of the Institute of Public Health of Federation of Bosnia and Herzegovina (FBiH). In covers approximately 70% of occupationally exposed workers in the country.

This paper summarized the dosimetry results for the past 15 years, starting from 1999 when thermoluminescent dosimetry was introduced, until December 2013. Data is compiled to match UNSCEAR 2008 Report [6].

2. MATERIALS AND METHODS

RPC is equipped with one thermoluminescent reader (Thermo Scientific^{TM} Harshaw TLD^{TM} Model 4500 Manual Reader, Waltham, MA, USA) with more than 3000 TLDs (TLD-100, LiF:Mg,Ti). Individual sensitivity of dosimeters is determined using ^{90}Sr source while absolute calibration in the radiation field of ^{137}Cs source. Annealing was performed using the procedure recommended in the IAEA safety guide [7].

Readout of the dosimeters is performed on a monthly basis. However, in 2013 some users were categorized to the category “B” of professionally exposed workers, making them eligible for quarterly readout of the dosimeters.

All users are advised to wear dosimeters under the apron, if present. Special attention is given to the workers in interventional radiology (interventional cardiology and gastroenterology included) whom RPC issued two dosimeters, one to wear under and other to wear above the apron, on the neckline. Finger and eye dosimeters have not been introduced yet. Meanwhile, introduction of new diagnostic and therapy procedures (PET/CT, iodine therapy, increase of number of patients in interventional cardiology, etc.) emphasised the need for finger and eye lens dosemeters.

If a worker is using one dosimeter then the effective dose is approximated by personal dose equivalent, Hp(10). In case of two TLD users for dosimetry, occupational effective dose is estimated using methodology described by Niklason et al [8]. Minimum reportable level (MRL) is 0.08 mSv [4].

3. RESULTS

In the past fifteen years, number of TLD users increased steadily, with 73 users per year on average (Fig. 1). In 2013, RPC covered 1416 users with personal dosimetry.

Workers are classified into groups according to their profession. Most of them work in medical practice (diagnostic and interventional radiology, nuclear medicine, radiotherapy, dentistry, veterinary medicine) while smaller number are industrial workers. On a 5-year average, none of them received annual effective dose greater than 10 mSv per year.

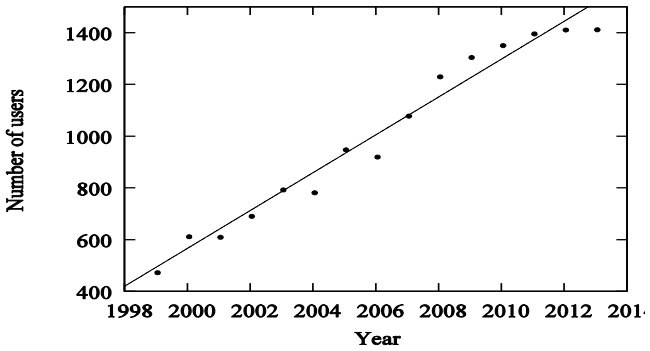

FIG. 1. Increase of number of TLD users in FBiH. The number of occupationally exposed workers covered by TL dosimetry increased by 73 users per year (linear regression curve $y=73.16x-1.457\times10^5$)

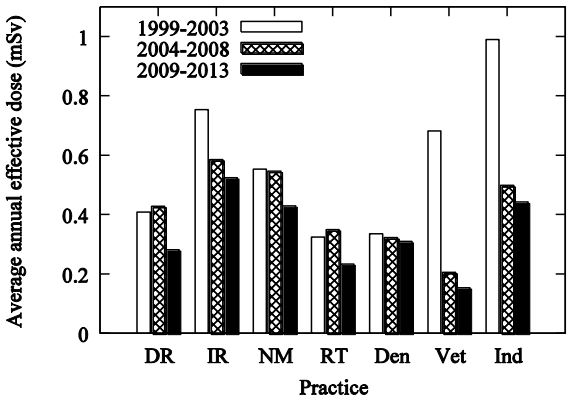

FIG. 2. Average doses in FBiH in past 15 years in diagnostic radiology (DR), interventional radiology (IR), nuclear medicine (NM), radiotherapy (RT), dentistry (Den), veterinary medicine (Vet) and industry (Ind).

TABLE 1. AVERAGE ANNUAL EFFECTIVE DOSES (mSv) IN COMPARISON TO OTHER COUNTRIES [6, 9]

	In mSv			Roma-nia[1]	Slova-kia[1]	UK[1]	Eastern Europe	World[2]
Practices	99–03	04–08	09–13					
Diagnostic radiology	0.408	0.428	0.281	0.54	1.79	0.07		
Interventional radiology	0.753	0.585	0.524	3.58	3.72	0.21		
Nuclear medicine	0.553	0.546	0.429			0.59	0.68	0.79
Radiotherapy	0.324	0.349	0.233				1	0.55
Dental practice	0.335	0.322	0.309				0.16	0.06
Veterinary practice	0.681	0.205	0.153					
Industry	0.989	0.498	0.442	2.75	1.60	0.76		

[1]Data from UNSCEAR Report 2008 [6]
[2]Data from UNSCEAR Report 2000 [9]

4. DISCUSSION

During the 15-year period of individual monitoring, no doses above the limit (20 mSv/y) were reported. Most of the professionals received doses less than 1 mSv/y. It is evident from Fig. 2 that the highest personal doses are associated with professionals in interventional radiology, nuclear medicine and industry. It is interesting to see how doses in industry have decreased over the years. This is most likely due to the slowing down of certain branches of economy in FBiH.

Diagnostic radiology plays a significant role in collective dose only, whereas other exposures are low. Obtained data is similar in comparison to results from other countries (Table 1).

5. CONCLUSION

All professionally exposed workers receive doses below the annual dose limit. This attributes to good working procedures and the use of radiation protection means, such as aprons, protective barriers, etc. New practices in medicine emphasise the need for more personal dosemeters, as well as specialized dosemeters for extremities, eye lenses, etc.

REFERENCES

[1] INTERNATIONAL LABOUR OFFICE, Radiation protection of workers (ionising radiations), ILO, Geneva, Switzerland (1997).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety of radiation sources: International Basic Safety Standards (interim edition), IAEA, Vienna, Austria (2011).
[3] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Annals of the ICRP publication 103, The 2007 Recommendations of the International Commission on Radiological Protection, ICRP (2007).
[4] STATE REGULATORY AGENCY FOR RADIATION AND NUCLEAR SAFETY, Regulation on radiation protection for occupational and public exposure, Official Gazette of Bosnia and Herzegovina, Sarajevo (2011).
[5] BAŠIĆ, B., BEGANOVIĆ, A., SAMEK, D., SKOPLJAK-BEGANOVIĆ, A., & GAZDIĆ-ŠANTIĆ, M., Ten years of monitoring the occupational radiation exposure in Bosnia and Herzegovina, Radiation protection dosimetry, 139(1-3) (2010) 400-402.
[6] UNITED NATIONS SCIENTIFIC COMMITTEE ON THE EFFECTS OF ATOMIC RADIATION, UNSCEAR 2008 report: Sources and effects of ionizing radiation, United Nations, New York (2010).
[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment of occupational exposure due to external sources of radiation: Safety Guide, IAEA, Vienna (1999).
[8] NIKLASON, L. T., MARX, M. V., & CHAN, H. P., The estimation of occupational effective dose in diagnostic radiology with two dosimeters, Health physics, 67(6) (1994) 611-615.
[9] UNITED NATIONS SCIENTIFIC COMMITTEE ON THE EFFECTS OF ATOMIC RADIATION, UNSCEAR 2000 report: Sources and effects of ionizing radiation, United Nations, New York (2001).

Session 5: Occupational radiation protection in industrial and research & education facilities

EXPOSURE OF INDUSTRIAL RADIOGRAPHY OPERATORS *A CASE STUDY*

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Abstract

National Radiation Protection Agency (NRPA) in Cameroon detected exposure of NDT workers from their dose monitoring performed on monthly basis. Investigation conducted by the NRPA demonstrated that the the work procedures are inappropriate. Recommendations were provided to the company to avoid or to minimize such hasards during gamma radiography practices .

1. INTRODUCTION

Industrial radiography means an examination of the structure of materials by the nondestructive method of utilizing ionizing radiation to make radiographic images. The industries in which this technique is used are widespread including chemicals, petroleum, oil and gas, automobiles, aerospace, power generation (both nuclear and non-nuclear), civil engineering, welding, general engineering fabrication plants, and maintenance operations in many industrial processing plants.

"The radiography examination is the only one that can clearly detect a lack of penetration at the root of the welds… This capability of identifying the nature of defects, make radiography the mandatory test for the qualification of welding procedure and welder qualification… Another large application of Radiography is the examination of castings… No other test may be used in this case" [1].

Industrial radiography is a potentially hazardous activity. Harmful radiation doses have been received in the past by both radiographers and members of the public from a variety of accidents, most of which could have been avoided had appropriate safe working practices been followed. Personal monitoring records show that doses received by operators of industrial radiography equipment are amongst the highest of any group of radiation workers [2].

In Cameroon, several companies have been using radiography techniques for many years. Workers dose monitoring was performed abroad up to 2011. From this year, NRPA started the operation of its professional dose monitoring service which provides personal dosimeter reading on a monthly basis. In 2012, NRPA noted an unusual exposure of radiography operators of NDT Company. The present communication describes the NRPA's efforts to clarify this incident.

2. METHODS

2.1 Worker dosimeter reading and analysis

The following table shows the monthly doses of three workers in August 2012.

TABLE 1. AUGUST 2012 DOSE RECORDS

n°	Workers	Monthly Dose for August 2012 Hp(10) (mSv)
01	W1	13.46
02	W2	10.38
03	W3	6.8

These values are quite high in comparison to what could be expected, taking into account the requirements for compliance with 20 mSv per year for this category (A) of radiation workers. By a written letter NRPA requested explanations from the RPO and received no reply. Then NRPA's staff visited the company on 18th January 2013 to investigate for the causes of the high level of exposure.

3. RESULTS

Meeting in the company was attended by three NRPA's staff, the company RPO, one NNDT operator, and a company management responsible.

RPO reported that he requested written explanation from the workers when he received the dose report on August 2012. The radiographer who conducted the experiment replied as following:

1. The incident occurred on 9th August 2012 at 02:30 am during a gamma radiography operation to control an oil distillation column, because workers went close to the projector which was not in secure position
2. Workers concerned didn't make good use of available radiation detectors that they brought in the field, such as: dose rate meter, individual beeper, security system of the source projector with three lighting colors;
3. They also mentioned difficulties encountered while operating on the top of pillar and the hard conditions in the night time, 2:30 am, which didn't allow to comply with radiation protection requirements.

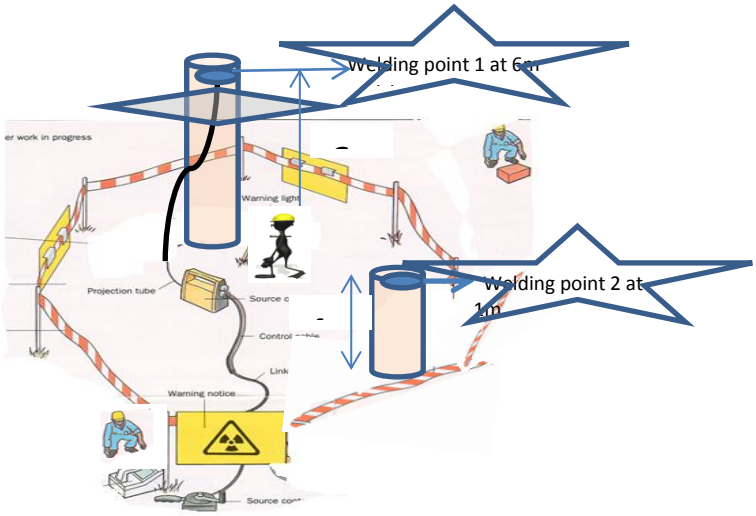

FIG. 1. Site of operation

3.1. Causes of the incident

Causes of incident might be one or combinaison of the following:

(a) Misuse of projector
(b) Dysfonctioning of the projector, and
(c) Failure or misuse of radiation detection devices.

3.2. Analysis

From the workers report it was clear that:

a. the state of equipment used by the workers was not controlled before the staring of the operation;
b. Worker safety culture is weak;
c. Workers were equiped with electronic dosimeters they were not accustomed to;
d. Working procedures are not effective.

NRPA provided the company specific recommendations which are commonly known and always put forward by regulatory bodies to avoid such incident [3, 4]. They include:

i. Review of the management system: radiation user's organization and devices that meet all relevant safety standards;
ii. Practical instructions and appropriate working methods are necessary in order to keep the radiation doses of the workers and other individuals as low as possible. Owing to the risk of accidents involved in industrial radiography, it is essential to anticipate abnormal events, to prevent safety-threatening events as effectively as possible, and to provide instructions for the case that any such should occur**;**
iii. The exposure container shall have a locking device; it shall be possible to lock the container without a key and to open it with a key only. Locking must be prevented if the radiation source is not in the storage position. The locking device shall be clearly marked to show whether the exposure container is locked or not;
iv. Before starting work, radiographers must ensure that the radiography devices are in proper condition and that the persons engaged in radiography are provided with a functioning dose rate meter and personal dosimeters and radiation alarm devices.

v. Accidental exposures are prevented by using the radiation survey meter when approaching the exposure device and by surveying the exposure device following every radiographic exposure.

4. CONCLUSION

Mostly with site radiography, incidents occur because of poor job planning, misuse of radiation detection equipment, failure to use adequate local source shielding, or inadequate systems of work. In addition, results of WGIR questionnaire in 2012 revealed that about one-half of the radiographers and the NDT companies reported that on-site radiography was being performed without the presence of the radiation protection officer (RPO), and hence without the benefit of the specific radiation protection expertise. [WGIR Report on the Questionnaires on Occupational Exposure in Industrial Radiography, Executive Summary, October 2012)].

The absence of RPO during experience with incident reported in Cameroon highlighted this concern. This incident provides the importance of worker dose monitoring and report, as well as analysis jointly performed by the radiography user and the regulatory authority which shall cooperate to protect workers life.

REFERENCES

[1] SUMMARY AND RECOMMENDATIONS OF THE 5TH EUROPEAN ALARA WORKSHOP on "Industrial radiography – Improvements in radiation protection" Rome, Italy, October (2001).

[2] AUSTRALIA NATIONAL HEALTH AND MEDICAL RESEARCH COUNCIL, Code of Practice for safe use of industrial radiography equipment, Radiation Health Series N°31, December (1989).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety in industrial radiography, IAEA Safety Series Report No. 13, IAEA, Vienna (1999).

[4] FINLAND RADIATION AND NUCLEAR SAFETY AUTHORITY, GUIDE ST 5.6 / 9 March (2012).

THE OCCUPATIONAL RADIATION PROTECTION OF THE TRAINING REACTOR OF BUDAPEST UNIVERSITY OF TECHNOLOGY

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Abstract

The Institute of Nuclear Techniques of the Budapest University of Technology and Economics is part of the Faculty of Natural Sciences. The Institute has two units: the Department of Nuclear Techniques organizes the educational tasks, whereas the Department of Nuclear Energy operates the nuclear training reactor. This facility plays a central role in the educational and research activities of the Institute. Furthermore, the training reactor is visited every year by approximately 1500 students from secondary schools and universities. In terms of radiation protection, these activities require careful consideration and planning. Therefore, the Department of Nuclear Energy has a dosimetry group to ensure radiation protection of the employees, students and visitors. The dosimetry group has the appropriate devices to ensure the personal dosimetry. Furthermore, we have number of special rules to prevent the possible emergency. Due to the advanced practice of radiation protection in the training reactor, the annual personal radiation dose of the employees and students is kept at a very low level.

1. INTRODUCTION

The Institute of Nuclear Techniques (NTI) of the Budapest University of Technology and Economics (BME) is part of the Faculty of Natural Sciences. The Institute has two units: the Department of Nuclear Techniques is responsible for the educational tasks of the Institute, whereas the Department of Nuclear Energy operates the nuclear training reactor (

FIG. 6). This reactor is a pool type nuclear reactor with a maximum power of 100 kW. . In terms of training, the main task of NTI BME is to educate the undergraduate, graduate and PhD students of BME and other Hungarian universities and higher education institutions in the field of nuclear engineering, nuclear and reactor physics and radiation protection. The educational and research activities of the Institute cover various fields, such as research of Generation IV reactor types, thermal hydraulics, radiochemistry, fusion research and medical physics. An accredited radiochemistry laboratory is operated in the reactor building for radiochemistry research/education and also in order to ensure the professional environment for the nuclear medicine student laboratory exercises.

Furthermore, the training reactor is visited every year by approximately 1500 students from secondary schools and universities. The above mentioned activities require advanced supervision in terms of radiation protection. Therefore, the Department of Nuclear Energy has a dosimetry group to ensure the radiation protection of the employees, students and visitors. The radiation protection in the training reactor is based on the relevant international recommendations [1] [2], Hungarian laws and specific rules. In this paper we introduce the devices, methods and operational rules applied in the training reactor.

2. METHODS

The dosimetry group of the Department of Nuclear Energy is well equipped with radiation measuring devices to ensure personal dosimetry. In addition, a number of special rules are applied to prevent potential emergency situations.

The reactor building has two zones. The first zone, which is the entrance area and the reception, has a lower security level. . The second zone (reactor hall, labs and offices), which is called controlled area, has a high security level. Between these two separated areas, a radiation monitoring gate is installed, which also serves security purposes. The radiation monitoring gate must be used for all employees, students and visitors upon entering and exiting the controlled zone of the training reactor to prevent carrying contamination outside the controlled area. Only certificated people (ie. staff members and PhD students) are allowed to enter the controlled area alone. Furthermore, the certification levels are indicated by use of a color code of the lab cape. The white cape indicates the highest permissions: and the wearer can move alone in the building (normally the employees wear white). The green lab cape denotes lower permissions (typically students and visitors should wear this color). Green wearers cannot move without an accompanying person (being in white cape). Finally, the brown lab cape marks the maintenance people also without free movement permissions (**Error! Reference source not found.**). Visitors and students receive adequate training on adiation protection rules before entering the controlled area. The level of training is determined by the activities they are supposed to perform (eg. visitors only receive a briefing on the most important rules while students performing measurement exercises are given a detailed set of rules they must obey).

FIG. 6. The training reactor

FIG. 7. The color code of the lab capes indicates the permissions of the wearer.

In the reactor building, there is a radiochemistry lab where potential contamination cannot be fully excluded. Therefore, a second radiation monitoring gate has been installed at the entrance of this laboratory to monitor the contamination separately. Special rules for this lab are prescribed so as to practice operations using inactive components before performing the real exercise. Measurements of the subject “Laboratory Practice in Medical Physics” are also related to radiation protection.

Personnel radiation monitoring is carried out in the training reactor with the help of the official personal dosimeters and electronic personal dosimeters (

FIG. 8). Conventional film based official dosimeters were used before March 2013. Since then, we have been using thermoluminescent dosimeters (TLD) for personnel monitoring. The official TLDs are evaluated every two months (

FIG. 8). Personal monitoring is supplemented by use of electrical dosimeters (gamma, beta and neutron sensitive). TABLE 1 shows the threshold doses and rates of the electrical dosimeters.

TABLE 1. THE THRESHOLD DOSES AND RATES OF THE ELECTRICAL DOSIMETERS

Dose rate		**Dose**	
Warning	Alarm	Warning	Alarm
20 μSv/h	100 μSv/h	100 μSv	300 μSv

FIG. 8. *The personal monitoring devices used in the training reactor*

The potential incorporation would be estimated taking into account the type of the isotope, activity and the type of the operation which involves the application of open radiation sources [3]. Should the estimated incorporated dose be higher than 1 mSv/y, the employee would be sent for an official investigation. We must note here that during the 43 years of operation of the training reactor along with the radiochemistry laboratories no incorporation has ever occurred.

The dose rates in the building are also monitored using a number of digital gamma survey meters and GM-tubes (the latter instruments are installed to certain points of the building). The area, where the measured dose rate is higher than 20 μSv/h, must be marked using a special board. limits in the training reactor.

TABLE 2 shows the typical dose limits in the training reactor.

TABLE 2. THE TYPICAL DOSE LIMITS IN THE TRAINING REACTOR

Dose/ Dose rate	Note
> 20 μSv/h (digital gamma survey meter)	Must be marked using special board
> 2 mSv/year (personal TLD)	Official investigation
> 200 μSv/h (digital gamma survey meter)	The experiment cannot be continued without special permission

In order to ensure the safety of working in terms of radiation protection, special radiation protection education is required for staff members. On the base of the Order of the Minister of Health No. 16/2000 we have tree levels of training: basic (8 hours), advanced

(min. 26 hours) and comprehensive (min. 40 hours). Each certificate is valid for 5 years, after which they should be renewed by attending proper refreshment courses [4].

3. RESULTS

The management of the Institute of Nuclear Techniques considers it essential to maintain and further improve the radiation protection knowledge of the employees. Therefore, 10 %, 20 % and 70 % of the employees have comprehensive, advanced, and basic level certificates, respectively.

TLDs are more sensitive than the film dosimeters applied before 2013 so they can measure lower doses. Therefore, after the transition to the TLD, in some cases the measured doses seem to be higher than the detection limit. According to the officially measured data 6 TLDs measured higher dose than the detection limit (>0.2 mSv/2 months) in 2013. Personal dose values measured using the electronic dose meters was lower than 0.5 mSv/2 months at each case. In 2014 we only have data from January to April and in this time period 4 TLDs measured higher dose than 0.2 mSv/2 months but the maximum dose was lower than 0.4 mSv/2 months.

Dose received by the visitors and the students is also monitored using electrical personal dosimeters. However, the measured values have been found to be under the detection limit in all these years since the commissioning of the training reactor.

This result clearly shows that the occupational radiation protection in the training reactor successfully minimizes the dose of the employees, visitors and students.

4. DISCUSSIONS

Occupational radiation protection is absolutely necessary in the nuclear techniques. The international recommendation for the radiation protection provides a base to decrease the dose of the employees. In the case of the education, the radiation protection of the students is one of the most important issues. At the Institute Nuclear Techniques, during the required practical tasks/exercises, the students get acquainted with the basic principles of radiation protection such as the justification, optimization and dose limitation. Students become familiar with the legal aspects of radiation protection, theory and practice of nuclear physics and design issues of workplaces to ensure radiological safety.

Therefore, the radiation protection in the training reactor is in very advanced level, and this is the reason for the very low annual individual dose of the employees.

REFERENCES

[1] INTERNATIONAL COMMISSION ON RADIATION PROTECTION, 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4) (2007).

[2] INTERNATIONAL COMMISSION ON RADIATION PROTECTION,. 1990 Recommendations of International Commission on Radiological Protection, ICRP Publication 60, (1991).

[3] INTERNATION ATOMIC ENERGY AGENCY, Assessment of occupational exposure due to intakes of radionuclides, IAEA Safety Guide, No. RS-G-1.2, Vienna (1999).

[4] R. ELEK et al, Curriculum of radiation protection educations and trainings in Hungary, ETRAP 2013, 12 – 15, Vienna, Austria (2013) 168-173.

CHALLENGES OF OCCUPATIONAL RADIATION PROTECTION AT HIGH POWER LASER FACILITIES

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Abstract

Ionizing radiation protection has never been a concern at laser facilities; however, this is not valid any more, as the rapid devolopment of laser systems during the last decade resulted in ability of the lasers to generate not only X-rays but also high energy charged particles. This paper summarizes the methods for designing and implementing occupational radiation protection program, with emphasis on challenges raised by specifics of ionizing radation sources generated by lasers. Experience in designing the system for a newly built laser research centrum ELI Beamlines, and experiments performed primarily using laser systems located within PALS Research Cetre in Prague are described.

1. INTRODUCTION

Ionizing radiation has never been an issue at laser facilities; however, the rapid development of laser technologies during the last decade has resulted in systems that are able to generate beams of ionizing radiation by focusing ultra-short high intensity pulses onto targets. Therefore, it is necessary to assess the radiological situation and to ensure adequate radiation protection of both personnel and public.

In general, it is sufficient to follow the primary radioprotection principles, i.e. protect the personnel by shielding, time, and distance. To implement these principles into a functional occupational radiation protection system several aspects have to be covered, namely shielding design, personal dosimetry, workplace monitoring, and facility operational regime.

However, laser facilities require some special considerations. This work summarizes the challenges raised by these specific sources and describes problems encountered and lessons learnt mainly at laser research centre PALS. These are implemented into the design of occupational radiation protection at laser research centrum ELI Beamlines, which is under construction in Dolní Břežany, Czech Republic.

2. DESCRIPTION OF INVOLVED FACILITIES

2.1. PALS

The PALS Research centre in Prague accommodates two laser systems:

(a) 3 TW iodine high-power laser system of fundamental wavelength of 1.315 μm that operates in a single pulse regime, producing 2 shots per hour. The pulse duration is 300 ps.
(b) 25 TW Ti:sapphire laser system of central wavelength of 810 nm, delivering laser pulses of 40 fs duration.

Both systems use roughly spherical interaction chambers made of 1 cm thick steel and 100 cm (PALS) or 80 cm (Ti:sapphire) in diameter.

2.2. ELI Beamlines

The project Extreme Light Infrastructure (ELI) is a large scale effort of the European laser community. When fully implemented, it will consist of three new laser research centres with different focus. While ELI ALPS in Hungary will concentrate on applications of attosecond pulses in material sciences and biology, the Romanian ELI NP will primarily study laser induced nuclear physics. ELI Beamlines shall focus on development a new generation of secondary sources of ionizing radiation (high brightness sources of X-rays and high energy charged particles) for various interdisciplinary applications in physics, biology, medicine and material sciences [1].

The facility is to host four laser systems with power ranging from 0.5 to 10 PW. In full operation, the facility will accommodate 13 beamlines in six experimental halls, dedicated to different kind of applications. Typically, the laser beam interacting with a target inside specially developed end-stations will generate pulsed beams of X rays, high energy electrons, or protons. Although the source term is beamline and application dependent, a brief overview can be provided [2]:

(a) Pulse length 10 – 30 fs
(b) Primary particles per shot 10^8 - 10^{12}
(c) X-rays from few eV
(d) Repetition rate 0.1 - 1 kHz
(e) Electron energy up to 50 GeV
(f) Proton energy up to 3 GeV

3. METHODS AND CHALLENGES

Although general principles of radiation protection are valid and applicable for designing and implementing occupational radiation protection system at laser facilities, it is necessary to be aware of certain challenges specific for these installations.

3.1. Shielding design (bulk and local)

General principles for shielding design are well known and are fully applicable for radiation fields generated by laser. Design of the civil structure needs to provide sufficient shielding to minimize dose accrual of personnel. Suitable design of beam dumps further reduces the dose rates. Calculations were performed using Monte Carlo transport code FLUKA [3] and the FLAIR interface [3], together with discrete ordinates code ATTILA [5]. International recommendations [6, 7] implemented into national legislation [8] were taken as a guideline to design the facility according to the ALARA principles.

Challenges

(a) Upgrade of existing facilities: Existing laser sites were not designed to accommodate sources of ionizing radiation. Therefore, when the acceleration regime is implemented or new generation system is installed, shielding capabilities of a surrounding civil structure needs to be examined and shielding adequacy confirmed, or improved.
(b) Penetrations: Complexity of the laser technology and demanding requirements on the building parameters result in a large number of penetrations for various services. In a newly built facility, most of them can be juggled or moved to a more favourable location. However, many penetrations have to be straight – characteristic examples are penetrations dedicated to laser distribution, where each elbow not only affects the beam quality but is also extremely expensive. Moreover, the penetrations are often large, with minimal dimensions of 1 m×1 m, and are effectively empty, as they accommodate vacuum or air-conditioning tubes.
(c) Dumps: Each beam-line operating in the acceleration regime requires appropriate beam dump. However, neither the source term nor the exact geometry of the beamline is known, as these are either the subject of the laser research or a part of a specific experimental setup. Therefore, the beam dump design has to be variable and their basic design can reflect only a model or estimated maximal achievable source term.

3.2. Monitoring (personnel and workplace)

Obtaining relevant and ideally on-line information on the dose rate is a crucial part of a well-functioning occupation radiation protection system, therefore a careful choice of measuring devices is vital. As commercially available instruments are designed for detection in continuous field, the main difficulty is the unique time characteristics of the fields – very short pulses with a low repetition rate.

Challenges

(a) The requirements on measuring instruments are rather severe, as they have to have:

i. A reliable performance in the mixed radiation field of a wide range of energies,
ii. Ability to reliably detect pulses of ~ 20 fs length
iii. Good efficiency to the prompt radiation, not susceptible to saturation in high dose rate environment
iv. Resistance to electromagnetic pulses of few hundreds kV,
v. Sensitivity to detect small fractions of legal limits,
vi. Ability to provide on-line data.

(b) Active versus passive dosimetry: Active dosimeters based on silicon diodes are a popular choice for personal monitoring. However, their functionality in pulsed fields generated by lasers is questionable [9]. To verify their reliability, response of active dosimeters was compared to the response of passive systems (films, TLDs, bubble detectors, CR39) in several experimental settings, especially at PALS Research Centre in Prague [10, 11].

3.3. Facility operation

Integral part of the radioprotection system is a system of facility operation and management. To comply with the legal requirements [6-8], radiation zoning was introduced. Supervised and controlled areas need to be delineated based on expected levels of potential

hazard, occupancy, prompt and residual radiation. Typical operation regime needs to be defined and analysed. The dose uptake assessment can then be performed for various groups of personnel and measures taken if necessary. Only qualified and trained personnel have access to the hazardous areas. Access restrictions represent a powerful tool to minimize an unavoidable dose accrual.

Challenges

(a) Combined hazard management: Laser facilities can be rather complex, therefore a combined hazards have to be managed. Beside ionizing radiation, other hazards have to be considered and their joint effect assessed, such as laser radiation, high voltage, chemicals, nanoparticles, biohazards, cryogenics, etc. [12]. For example, ionizing radiation can function as an ignition source. Safety critical infrastructure has to be well designed and implemented.

(b) Access system: Due to the facility complexity, the access to the experimental halls has to be controlled not only by radiation monitoring system but also by security access, gas monitors, fire safety, laser safety, and laser control systems. Therefore, an utmost caution is required when integrating all these independent systems.

3.4. RESULTS AND DISCUSSION

Facilities operating high power lasers have to consider both prompt and residual radiation. Prompt radiation, including primary particle beam and generated secondaries, is strongly experiment dependent and exists only while the laser is being fired at the target. Occupational protection is ensured primarily by a proper shielding design and construction of the civil structure and dumps [13], accompanied by strict access restrictions during the laser operation. To minimize problems during construction, the civil structure of ELI Beamlines was designed in ordinary concrete, whose quality is much easier to maintain. Development of a functional occupational radiation protection system requires an active and close cooperation of radioprotection specialists with architects and future facility users already during building design phase.

Residual radiation is a continuing source with relatively low dose rate and decreasing with time. Further, the source is localized, primarily within the materials close to the target area. Nevertheless, the exposure of the personnel working in the vicinity can be high. Therefore, materials less susceptible to activation have to be chosen, access has to be temporarily restricted in needed (lockout period after the laser shutdown to allow for short-lived radionuclides to decay), or supply suitable local shielding.

Although wide variety of detectors is available on the market, choosing a device suitable for detection in laser generated fields is challenging. The measurements performed at PALS during several various experiments suggest that passive dosimetry is more reliable solution. The response of active personal is unstable - either reasonable, lower than expected or null. Further experiments with other systems are planned (e.g. OSL, direct storage). In general, more research in detection techniques suitable for detection in mixed high energy pulsed fields is needed.

Considering the envisioned regime of operation and the planned source term, ELI Beamlines is characterized as an important source, similar to a particle accelerator and thus a category III radiation workplace. Even though the design of the building targeted at the maximal dose accrual of a typical worker lower than 1 mSv/year, at least some of the staff will be category A exposed workers. A careful design of the personal interlock system integrating all the monitoring systems and reflecting all the possible hazards is crucial and essential prerequisite of a safe facility operation.

5. CONCLUSIONS

To conclude, designing a functional occupational radiation protection system is a complex and time consuming procedure that has to be performed with utmost caution in cooperation with a number of specialists in different fields. This fact emphasises the necessity of finding a common language, as different terminology is a common root of possible fatal misunderstanding. Also, development of a healthy "safety culture" at all levels of management as well as among staff and visitors is a basic prerequisite for an accident free operation.

REFERENCES

[1] MOUROU, G.A., et al, The ELI Whitebook. THOSS Media, GmbH, Berlin, Germany (2011).
[2] OLŠOVCOVÁ, V., et al, Bulk shielding for laser research centre ELI Beamlines, Progress in Nuclear Science and Technology 4 (2014) 247-251.
[3] FASSO, A., et al., FLUKA-2008: a multi-particle transport code, CERN-2005-10, INFN/TC_05/11, SLAC-R-773 (2005).
[4] VLACHOUDIS, V., FLAIR: A powerful but user friendly graphical user interface for FLUKA., Proc. Int. Conf. on Mathematics, Computational Methods & Reactor Physics (M&C 2009), Saratoga Springs, New York (2009).
[5] ATTILA 7.0.1, Transpire Inc. Attila Radiation Transport Software Support Tips: Anisotropic Point Source File, (2007), www.transpireinc.com.
[6] 96/29/EURATOM, Council directive, Basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiation, (1996).
[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety of radiation sources: International Basic Safety Standards – Interim Edition", General Safety Requirements Part 3, No. GSR Part 3 (Interim), IAEA, Vienna (2011).
[8] CZECH REPUBLIC, Atomic Law (Act 18/1997 Coll.), implementation in several Decrees.
[9] BORNE, F., et al., Radiation protection for an ultra-high intensity laser, Radiat. Prot. Dosim. 102 (1) (2002) 61–70.
[10] OLŠOVCOVÁ, V., et al, Radiation protection aspects in the vicinity of TW class laser systems, Progress in Nuclear Science and Technology, (2014).
[11] OLŠOVCOVÁ, V., et al, Response of dosemeters in the radiation field generated by a TW class laser system, Rad. Prot. Dosim., in press, doi:10.1093/rpd/ncu024.
[12] PROCHÁZKA, P., et al, High hazards in laser experimental work: Challenges for afety management of laser research facilities, (Proc. of the 2nd Int. e-Conference on Optimization, Education and Data Mining in Science, Engineering and Risk Management, (2012)).
[13] FERRARI, A., et al, Radiation field characterization and shielding studies for the ELI Beamlines facility, Appl. Surf. Sci. 272 (2013) 138-144.
[14] ESPOSITO, A., Radiation protection issues for laser-based accelerators, Radiat. Prot. Dosim. 146 (4) (2011) 403-406.

STATISTICAL ANALYSIS OF THE DOSES RECEIVED BY THE AIRCREW AND THE MEDICAL PERSONNEL: COMPARISON AND TRENDS

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Abstract

The present study attempts a first analysis of the aircrew doses and a comparison with the respective values of the medical personnel. The national radition proetection database developed in EEAE has been proved a powerful tool that enabled such a procedure. The dose distributions, the mean annual doses and their respective trends are used as evaluation criteria of the national radiation protection programme. From the analysis, it was shown that the doses of the aircrew never exceeded the 6 mSv. The shape of the distribution of the two working categories and the two sexes are different which reflects the differences in the respective working experience and duties. Finally, it is shown that the mean annual doses of the exposed medical personnel are lower than the respective values of the aircrew.

1. INTRODUCTION

Greek Atomic Energy Commission (EEAE) is the national competent authority for radiation protection, radiological and nuclear safety. As such, EEAE has developed and operates an integrated central information system, the National Radiation Protection Database (NRPD), where all information related to occupationally exposed workers is kept. Moreover, it contains information about the institutions and laboratories using ionising radiation and their equipment. The NRPD has been designed in the 90's and then amended according to the requirments of the Technical Recommendations for Monitoring Individuals Occupationally Exposed Workers [1]. The NRPD contains all the relevant data since 1989.

In Greece, the vast majority of the occupationally exposed workers belong in the medical sector and more specifically, in radiology departments. In 2013, 11000 workers, out of a total of 13000, are registered in around 1300 medical institutions. From these a small percent (almost 12%) receive doses higher than the reporting level.

According to the the Greek radiation protection regulations [2] the air companies shall be equipped with a software, approved by EEAE, in order to assess the exposure of the personnel concerned. The results of the dosimetric assessment of the air crew are reported to EEAE only for those exceeding 1 mSv/y. Since 2011, EEAE's database was adjusted in a way that the aircrew doses are incorporated into the existing database. Nowadays 7 airline companies with 1300 workers submit calculated effective doses of the aircrew. In practice, the doses are calculated on monthly basis and all of them (not only the ones more than 1 mSv/y) are introduced to the NRPD twice per year.

In the present study a first analysis of the data related to the aircrew and a comparison with the respective data of the medical personnel is attempted. Parameters such as the dose distributions, the mean annual doses and their trends can serve as valuable evaluation criteria for the optimisation of the radiation protection programme implemented in the country.

2. METHODS

Specific queries have been designed in order to perform statistical analysis of the data kept in the NRPD. The data used in the present study are related to the working categories of the medical and aviation personnel. The rest of the categories (industry, research and education) consist only a small percent of the collective dose (1,4 and 0,1 % respectively) as well as of the total number of the exposed workers registered in the NRPD (less than 0,1 %). The analysed data contain all doses greater than or equal to 0,1 mSv. It should be mentioned that the whole body dosemeter is worn above the radioprotective apron in Greece; however the data used for the analysis refer to the effective dose which are calculated by the whole body measurements by applying appriate correction factors [3]. The analysis is performed for the 3 year period-2011 to 2013.

3. RESULTS

In Table 1 the collective dose, the total number of workers and the mean annual dose for workers who received doses greater than or equal to 0,1 mSv are shown.

TABLE 1. COLLECTIVE DOSE, TOTAL NUMBER OF WORKERS AND MEAN ANNUAL DOSE FOR THE MEDICAL AND AVIATION PERSONNEL WITH DOSES MORE THAN 0.1 mSv.

	collective dose		
Person-Sv	2011	2012	2013
medical personnel	2,01	1,56	1,31
aircrew	1,07	1,40	1,17
	number of workers with E>=0,1 mSv		
	2011	2012	2013
medical personnel	1872	1704	1561
aircrew	1387	1362	1167
	mean annual dose of workers with E>=0,1 mSv		
mSv	2011	2012	2013
medical personnel	1,36	0,91	0,84
aircrew	0,77	1,03	1,01

As it is can be seen from the above Table, in 2011 the collective dose for the aircrew was almost half the respective value of the medical staff. However, for the years 2012 and 2013, the values of the collective dose for the two working categories are very close. The lower value of the collective dose in 2011 for the aircrew was due to the fact that not all of the Greek airline companies were contacted at that time so the data are incomplete. About the number of workers in the two sectors who have received doses more than 0,1 mSv, it seems that there is a decrease in the this years period which is due to the general economic instability of the country. Many people have been retired and new personnel is not being hired at the moment. It is of special inerest to point out that, though in 2011, the mean annual dose of the medical personnel was almost double the respective value of the aircrew, the following years (2012 and 2013) the latter is higher than the first.

This is partly due to the fact mentioned above (incomplete set data gathered in 2011) and it is also due to an increase of the doses in medical staff encountered in some very specific cases. When these cases were identified, stricter radiation protection measures were

imposed and the doses reached a lower level, shown by the respective decrease in the mean annual doses. It should be noted that the values of the collective doses of medical staff in Greece are higher than in many European countries [4]; probably due to the fact that the whole body personal dosemeter is worn above the radioprotective apron.

In Fig. 1, the sum of the effective dose for the medical and aircrew personnel for specific ranges is shown. From the figure we observe that there are no doses for the aircrew in the range > 6mSv. This is due to the fact that according to the legislation the airline companies shall organise the flight schedules of the personnel in such a way that the aircrew doses do not exceed 6 mSv/y. Moreover, we observe that the total dose in the range 1 to 6 mSv for the aircrew is higher than for the medical personnel.

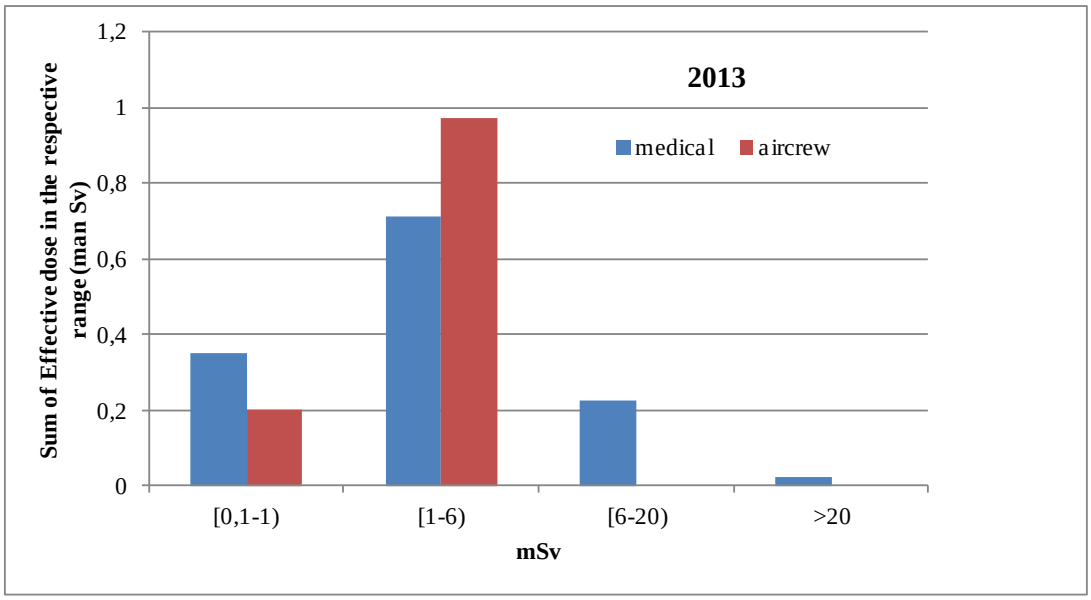

FIG. 1. Sum of the effective dose for the medical personnel and the aircrew for the year 2013 per range as shown in the horizontal axis

The analysis was also performed using the parameter of sex for the two working categories. The results are shown in Fig. 2. There is no difference in the mean annual doses between the two sexes for the aircrew. However, there is a difference for the medical personnel. From the figure it was observed that, in general, men in the medical sector are more exposed than women.

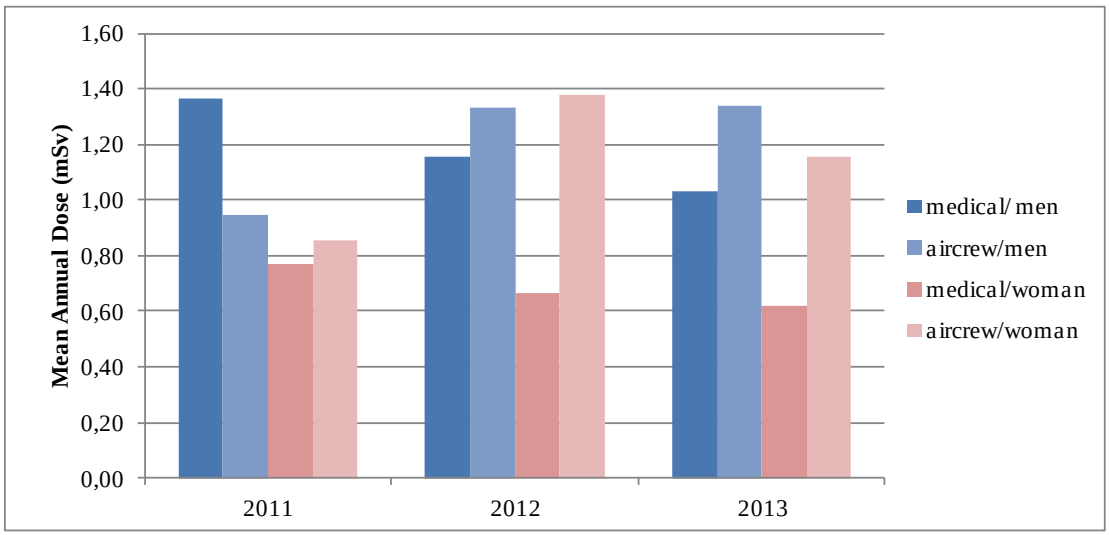

FIG. 2. Mean Annual Dose (mSv) for both sexes for the medical and aviation personnel

This obervation was analysed to explore in more detail the distribution of doses for the two sexes in the two sectors studied. By doing this we get the results in the following Fig. 3.

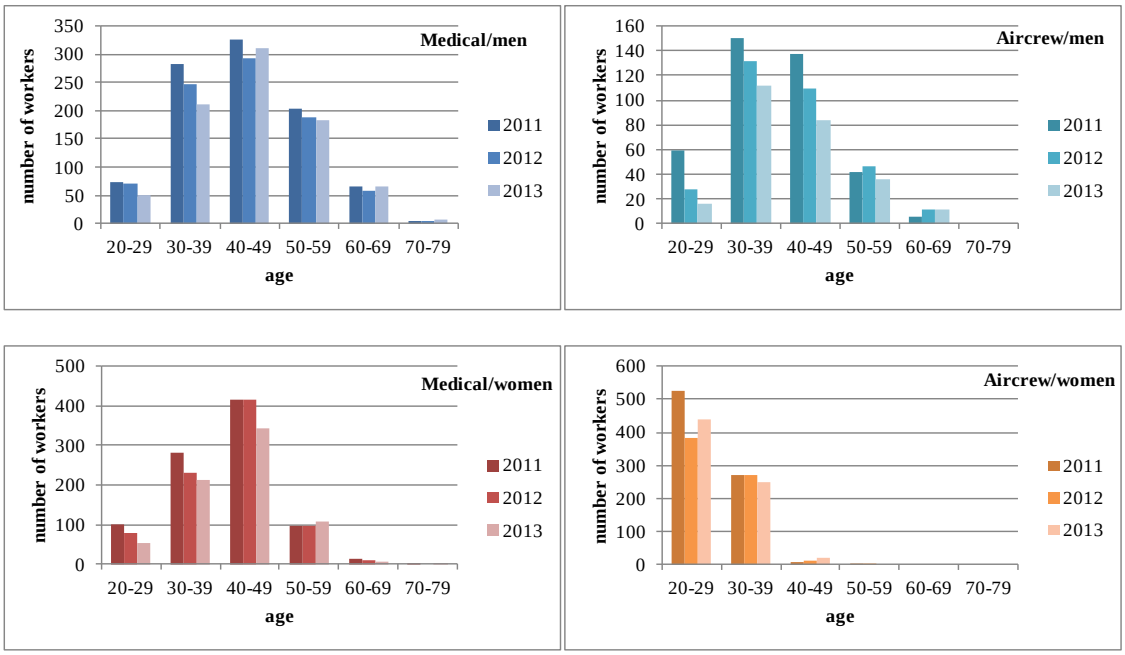

FIG. 3. Distribution of the number of workers for medical personnel and aircrew according to the age and sex.

The shape of the distribution of the number of workers in the medical sector for both sexes is more or less the same. The differences are focused on the facts that: in the decade of 30-39, less women are registered than men and secondly women stop being occupationally exposed at the age of 60, while men at the age of 70. The shape of the distribution of the number of workers for the aircrew for the two sexes is completely different. For men, the maximum of the distribution is located at the decades 30-49 while for women the maximum is very early 20-29 years old. This is explained by the fact that most of men are pilots and the flight hours depend on the experience they gain. For the air hostesses no such experience is

required. Moreover, the maximum age for women is 50 years old while for men is the decade of 60.

4. CONCLUSIONS

A national central register is a powerful tool for continuously monitoring and evaluation the implementation of the national radiation protection programme. Such situations were identified in 2011 in the medical sector and treated accordingly. Moreover, the NRPD that is developed in EEAE is a relational database, allowing thus the performance of statistical analysis. In the present study, a comparison of the doses and the respective distribution between the medical personnel and the aircrew was performed. It was shown that the doses of the aircrew never exceeded the 6 mSv in a year. Finally, it is shown that the mean annual doses of the exposed medical personnel are lower than the respective values of the aircrew.

REFERENCES

[1] EUROPEAN COMMISSION, RP 160, Technical Recommendations for monitoring individuals occupationally exposed to external radiation, Directorate-General for Energy and Transport., Luxembourg (2009).

[2] REGULATION ON RADIOLOGICAL PROTECTION, JOINT MINISTERIAL ORDER 1014 (FOR) 94, Official Gazette No 216/B/6-03-2001, Radiation Protection Regulations, Athens (2001).

[3] NATIONAL COUNCIL ON RADIATION PROTECTION AND MEASUREMENTS, NCRP Report No. 122, Use of personal monitors to estimate effective dose to workers for external exposure to low-LET radiation, Bethesda (1995).

[4] G.FRASCH AND K.PETROVA., Dose trends in occupational radiation exposure in Europe: Results from the ESOREX project, Radiation Protection Dosimetry, Vol. 125 (2007) 121-126.

RADIATION PROTECTION DURING THE NEUTRON GUIDE MODIFICATION PROJECT AT THE OPAL RESEARCH REACTOR (AUSTRALIA)
DOSE ASSESSMENT AND MANAGEMENT OF OCCUPATIONAL RADIATION EXPOSURES

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Abstract

In early 2012, ANSTO commenced a project to modify the Cold Neutron Guide at its 20MW OPAL Research Reactor. The actual modification work started in November 2012. The purpose of this work was to deliver a third cold neutron guide in addition to the the two guides that were installed initially giving the neutron research divison increased flexibility in terms of their research capabilities. Radiological exposures were anticipated to be received by staff during this project and the exposures were considered to be justified on the basis that once successfully completed the operation of the facility with the additional neutron beam lines would provide a benefit to the Australian public, industry and the international research community. The radiation protection team at ANSTO utilised the dose rate modelling data from our Nuclear Analysis Services team (NAS) to derive a dose estimate for the project. As the project progressed, actual measurement were taken and the data was used to further refine our dose estimates. This paper describes the methodology used to arrive at the dose estimates, the actual dose recieved and the radiation protection arrangements in place to achieve this and keep doses ALARP.

1. INTRODUCTION

As part of the radiation protection arrangements, the radiation protection team carried out a dose assessment to create a dose estimate for the proposed project of installing a new neutron guide into the OPAL research reactor. The role of the radiation protection team was also to:

(a) Prepare a radiation protection plan for the project describing the hazards, controls, area classifications and monitoring programs
(b) Providing dose updates/trends to staff and advising the project team of their dose budget on a daily basis
(c) Carry out routine, task monitoring and individual monitoring
(d) Provide specific radiation safety training to the staff carrying out the work.
(e) Review the radiation protection arrangements daily and provide the project team with updates, and
(f) Provide with a report detailing the individual and collectives doses following the completion of the work.

2. DESCRIPTION

The Neutron Guide modification work was primarily carried out in the inner bunker of the Neutron Guide Bunker (NGB) over an estimated period of six weeks. The NGB is divided

into two sections; the inner bunker (closest to the reactor face) and outer bunker which is before entering the inner bunker. Entry into the inner bunker is via the outer bunker. Entry into the inner bunker was via swipe access and through a temporary barrier in the outer bunker.

The NGB and all the areas within it were provided with hazard classifications according to the assumed risks of radiological contamination levels, external radiation levels and exposure times. Three levels of classifications were used; red, blue and white with red being the most hazardous and white being the least hazardous. A temporary change barrier was set up near the exit of the Neutron Guide Hall (NGH). A frisking contamination monitor was placed at this barrier and a final portal monitor was in place for exit out of the NGH. Area radiation monitors with alarm settings and visual displays were placed in strategic locations to alert workers in the case of a sudden increase in dose rates in the inner and outer bunker.

2.1. Dose Rate Modelling and Dose Estimates

A detailed assessment was performed by NAS to estimate the gamma dose rates during the installation of the new in-pile, primary shutter and front cover for the Cold Neutron Guide system. The gamma radiation was due to the activation of the internal components of the OPAL Research Reactor. The first step in the assessment was to calculate the neutron flux along the main internal components of the OPAL Reactor. A 3D model of the internal components was made using MCNP/2/. The number of neutrons born in the core per second was estimated assuming 20MW thermal power (the maximum power of the OPAL Research Reactor).

Using the neutron flux the total activity induced in the components was then estimated. The irradiation time was conservatively assumed to be 10 years (the actual irradiation time was approximately (four years) and the decay time after the irradiation corresponded to 1 week. Given the chemical compositions of the different components it was determined that the most significant isotopes produced were ^{95}Zr, ^{95}Nb, ^{59}Fe, ^{60}Co, ^{122}Sb and ^{124}Sb. This information was used to produce a dose rate profile of the immediate area around the work area and of the components that were to be removed.

Radiation protection in consultation with the project team used the dose rate modelling data to carry out dose assessments and arrive at dose estimates per each task and for the entire project. A detailed instruction for the work was prepared by the project team which identified the steps, time required to complete, number of staff involved and any specific actions required such as hold points during critical tasks. This information from the project team plus the dose rate profiles provided by NAS was used to prepare a dose estimate and a dose spread sheet by the radiation protection team to monitor the dose to staff (individual and collective) during each phase/task of the project.

EPD (Electronic personal dosimeter) data was used to constantly monitor for individual and collective doses. Task ID's were created for all the major task and dose alarms and dose rate alarms were set accordingly. Individuals were required to log on using their personal ID and the specific task ID for the day. The maximum effective individual dose target was 5mSv with an individual dose constraint of 10mSv. The maximum estimated effective collective dose for the project was 15 person-mSv.

2.2. Radiation Protection Arrangements

Radiological monitoring was carried out by the radiation protection team in accordance with the project radiation protection plan.

Radiological monitoring was carried out in two parts:

2.2.1. Workplace and Area Monitoring

Routine area monitoring was carried out during the neutron guide modification project to confirm effective control of dose rates and contamination levels in classified areas as per the radiation protection plan and ANSTO standards and guides. This consisted of carrying out routine radiological surveys of the areas prior to start of shift and also confirming that the fixed gamma area radiation monitors were working and within acceptable range.Task related radiological monitoring was also performed for each stage of the project to confirm dose rates as per the theoretical calculations and data obtained from this were used to refine dose estimates. A dose rate constraint of 10mSv/hr was applied to the general work area, where the effect of this increased dose rates were to be analysed and the impact on individual and collective dose assessed prior to proceeding with any work.

2.2.2. Individual Monitoring

Individual monitoring was carried out to measure, assess and evaluate the radiological exposure to all individuals directly involved in the neutron guide installation project. Routine external exposure were monitored using TLD (Thermo luminescent dosimeter) for effective (whole body) and wrist TLD for (extremity) dose. Individual monitoring using EPD for effective dose assessments was carried out and these were used to monitor daily dose exposure.

Frisking contamination monitors and walk through monitors were available for final contamination checks prior to exit from the NGB into a white area. All staff went through the Whole Body Monitoring for monitoring internal uptake of contaminants prior to start of work and after the completion of the project.

2.2.3. Area Classification

The area classification of the NGB during the project was red-radiation and white-contamination for the inner bunker area and blue-radiation and white-contamination for the outer bunker area.The inner bunker area was reclassified during the project to reflect the potential contamination hazards during the core drilling phase onto the reactor face to mount the shielding doors. During this task the inner bunker area was reclassified to red-radiation and blue contamination. These changes were updated on the safety hazard notice boards accordingly as an administrative control.

3. RESULTS

The following tables show the estimated and actual dose rates measured and the estimated and actual doses received by staff.

TABLE 1. MEASURED GAMMA DOSE RATES VS. ESTIMATED GAMMA DOSE RATES

Tasks	Measured Maximum Gamma Dose rates (μSv/h)	Estimated Gamma Dose rates (μSv/h) – *(modelling)*
Removal of the Primary shutter	9000	5000
Removal of the In-Pile	2000	5000
Storing of the In-Pile	50	2000
Installation of the In-Pile	10000	5000
Installation of the new shutter	100	5000

There was some variation in the estimated gamma dose rates and actual measured maximum gamma dose rates as shown in Table 1. However through implementation of the controls and ongoing monitoring of dose trends against dose budgets, individual doses and collective doses were below the estimated dose as shown in Tables 2 and 3.

TABLE 2. DOSIMETRY EFFECTIVE – COLLECTIVE DOSE DATA

	Estimated collective dose(person mSv)	Actual collective dose(person mSv)
EPD Collective dose data (person- mSv)	15	9.6
TLD Collective dose data (person- mSv)	15	11.2

It was observed that there is slight variations in the data obtained from EPD's and TLD's with the TLD data approximately twenty percent higher than the EPD data. This was attributed due to several reasons; where the dosimetry units were worn, how efficient the units were in measuring, etc. As the TLD is the legal dosimetry in Australia, this data was used for reporting to the regulator. Overall, the variance was within limits and the collective and individual dosimetry values were within our dose targets and constraints.

TABLE 3. DOSIMETRY EFFECTIVE – MAXIMUM INDIVIDUAL DOSE DATA

	Estimated Maximum individual dose (mSv)	Actual Maximum individual dose (mSv)
	5	0.72
EPD Individual (mSv)		
TLD Individual (mSv)	5	0.91

TABLE 4. DOSIMETRY EQUIVALENT (EXTREMITY) – MAXIMUM INDIVIDUAL DOSE DATA

	Estimated Maximum individual dose (mSv)	Actual Maximum individual dose (mSv)
TLD Individual (mSv)	100	4.6

3. DISCUSSIONS/ LESSONS LEARNT

The benefits of detailed planning and including all divisions from the start of the project are essential in the success of a project. This is clearly evident from the neutron guide modification project. There was a clear line of communication and hierachy and information was clearly disemminated via planning meetings and weekly and daily briefs during the actual project work.

The working out of dose estimates with the theoretical data and having a step by step instruction with time estimates were extremely benefical in creating a realistic dose estimate. The continuous monitoring of daily and collective dose for staff (actual versus estimated) using a dose spread sheet and effectively applying the planned radiation protection arrangement assisted the radiation protection team to advise and control exposures to staff and maintain them below the agreed dose constraints.

Use of fixed and mobile shielding during the project were found to be beneficial in reducing exposures particularly during tasks where individuals had to perform work in areas with potential high dose rate beams on in close proximity to radiation sources.

4. CONCLUSION

It can be concluded that the neutron guide modification project by ANSTO was conducted safely and successfully from radiation protection point of view. All personal dosimetry and radiological monitoring was carried out as per the projected radiation protection plan. Whole body monitoring was carried out for all staff involved in the work.

This was performed prior to start and post finishing the project. No internal contamination was registered for anyone.

Workplace and area monitoring were conducted successfully and no airborne or surface contamination were detected. Individual monitoring using both EPD's and TLD's indicated that all individual doses received were significantly less than the set individual dose target of 5 mSv, and the collective dose was less than the estimated value of 15 person-mSv.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Guide, Occupational radiation protection, RS-G-1.1, IAEA, Vienna, (1999).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Guide, Assessment of occupational exposure due to external sources of radiation, RS-G-1.3, IAEA, Vienna (1999).
[3] International Commission on Radiological Protection, ICRP Publication 103, 2007 Reccomendations of the International Commission on Rradiological Protection, Elsevier, (2007).
[4] ARPANSA RADIATION PROTECTION SERIES (RPS) 1. (2002).
[5] ANSTO RADIATION SAFETY STANDARD AS2310, Radiation Safety (2009).
[6] MAHARAJ.P, POLEWSKI.M, Ansto CG2 in pile installation radiation protection plan.
[7] MAHARAJ.P, Ansto radiation protection report cold guide installation works, November-December (2012).
[8] NUCLEAR ANALYSIS GROUP (NAS) ANSTO, Dingo and CG2 dose rate assessment.

Session 6: Occupational radiation protection in emergencies and existing (post-accident) exposure situations

PERSONAL PROTECTIVE EQUIPEMENT (PPE) TO FACE NUCLEAR RISKS

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Abstract

Before considering to operate in a nuclear environment, it is obviously important to assess both specific risks: irridiation and contamination. For the last 60 years, various strategies have been worldwide implemented including policies like ALARP, ALARA, TEDE … in view of reducing the impact of these risks on the worker.

In the meantine, various PPE have been developed against these risks for all working configurations (Fuel cycle industry, Electricity production, Post Nuclear accident…), combining a higher whole body protection efficiency comparing to the old full face mask combined with a non ventilated suit and a more comfortable way to perform the task with the associated impact on the dose reduction. Starting from the EC 89/686 European directive, we will try to offer a way to compare the PPE performance.

1. INTRODUCTION

1989 and to implement the same level of protection for any worker in the European Union, two important European Directives have been issued :

(a) The EC89/686 related to personal protective equipment and council directive of 21 December 1989 on the approximation of the laws of the member states relating to personal protective equipment (89/686/EEC) [1].
(b) The EC 89/656 related to the directive 89/656/EEC - use of personal protective equipment of 30 November 1989 on the minimum health and safety requirements for the use by workers of personal protective equipment at the workplace (third individual directive within the meaning of Article 16 (1) of Directive 89/391/EEC) [2].

This presentation describes how the Personal Protective Equipment in the Nuclear environment are used for the Occupational Radiation Protection of workers - Gaps, Challenges and Developments.

2. PPE BASICS REQUIREMENT

2.1. EC 89/686 Basics Health and Safety Requirements and how it applies to the Nuclear PPE

PPE must provide adequate protection against all risks encountered:

1. Highest level of protection possible against the risk (2-3)
2. Innocuousness of PPE: the PPE has to be designed in a way to minimize as much as possible the extra risk due to the PPE itself (2-4)
3. Comfort and efficiency: PPE has to be designed to offer the maximum comfort and efficiency (2-4)

2.2. Main risks to face in the nuclear environment

2.2.1. Irradiation by direct exposure to a radioactive source.

How to minimize it:

i. Increase distance,
ii. Decrease exposure time
iii. Protective screen (lead blanket, apron, etc)

It is to be noted here that a new device is availbale for personnel protection - post nuclear accident, based on the bone marrow protection in the pelvic area. Whole-body shielding solutions are made using only thin layers of inherently heavy radiation-attenuating material. This type of shielding may be effective for blocking of alpha and beta radiation, yet is ineffective in blocking highly penetrating gamma radiation. Whole-body shielding is therefore incapable of preventing the acute health effects of exposure to gamma radiation in acidental situations.

An alternative to whole-body shielding is to selectively shield tissues of increased radiosensitivity with substantial amounts of shielding material in an approach coined "partial shielding". The most radiation sensitive tissue in the human body is bone marrow. Significantly, it is sufficient to effectively shield only a minor fraction of bone marrow to enjoy systemic benefits such as increased blood counts, immune resistance and even survival. Indeed, in the common procedure of bone marrow transplantation, the quantity of hematopoietic stem cells extracted from a single active marrow site is sufficient to support the complete reconstitution of the hematopoietic stem cell compartment in a lethally irradiated recipient [25, 26]. In contrast to mature blood cells, which are dispersed throughout the body, hematopoietic stem cells are confined to the bones, allowing for effective targeted shielding. Thus, as has been confirmed in several animal models, blocking entry of ionizing radiation into a single active marrow site of a subject receiving otherwise full-body irradiation is sufficient to spare hematopoietic functions and allow survival [27-30].

2.2.2 Contamination Protection

Three main ways of contamination:

I. External skin contamination,
II. Internal by inhalation or ingestion .
III. Absorption through cuts and wounds.

And how to protect against it:

(a) Layers as protective suit, gloves for skin protection
(b) Respiratory protective devices as mask, hood,
(c) Encapsulated Ventilated pressurized full suit.

Given below some information concerning some contaminant commonly found with some Uranium isotopes:

a. The allowable amount of Pu/m^3 in the air in C2 working areas is: 8×10^{-9} $\mu g/m^3$, or if converted in terms of particle concentration: about10^9 times less than the atmospheric dust around us.
b. Sub micron airborne particle are invisible but measurable
c. ^{239}Pu: Half life is – 2.41×10^4 years

d. ^{238}U Half life - $4{,}46x10^{9}$ (4.5 billion years)
e. ^{232}Th Half life - $1{,}4\ x10^{10}$ (14 billion years)

2.3. Basic requirement no. 1

2.3.1. Highest level of protection possible against the risk to face ➔ evaluation of the protection against radioactive contamination

Important developments have been made in the last 30 years in this field of nuclear comtamination protection. And standard as EN 1073-1; EN 1073-2 have been written for the evaluation of the performances of those equipment allowing to characterize protection factor (PF) and inward leakage up to 10E5 (100 000) for a solid 1/3µ NaCl particle.

Definition: Inward Leakage (IL): $T.I.L = \frac{C_1}{C_2} x100[\%]$

Where: C_1 = concentration in enclosure (as test chamber)
C_2 = mean concentration at the sampling point for each exercise during practical performance test

The same test agent is used to evaluate the inward leakage for a respiratory protective device, but also for a non ventilated protective suit or a pressurized encapsulated suit. The location of the probe varies and the test protocol for practical performance test is defined by the appropriated following standard:

- EN 136 Respiratory protective device. Full face mask testing marking.
- EN 14325 Protective clothing against chemicals.
- EN 1073-2 Protective clothing against radioactive contamination non Ventilated protective suit
- EN 1073-1 Protective clothing against radioactive contamination ventilated protective suit.

According to basic requirements no. 1, here are the inward leakage class respectively for:

- Masks ➔ EN 136. The total inward leakage during practical performance test shall not exceed an average value of 0,05% (i.e.PF> 2000) and the associate Credited Fit factor = 1/10 PF
- Non-ventilated suits ➔ EN 1073-2. Here after are the inward leakage :
- Ventilated suits ➔EN 1073-1. Here after are the inward leakage.

Table 2 - Total inward leakage

Class	Mean value of inward leakage at the three sampling positions inside the suit during exercise of		Nominal protection factor[a]
	one activity (TIL_E) %	all activities (TIL_A) %	
3	0,3	0,2	500
2	3	2	50
1	30	20	5
a Nominal protection factor = 100 :TIL_A			

EN 1073-1 : 1998

Table 2: Leakage

Class	Maximum value of mean inward leakage into the hood during exercise of		Nominal protection factor
	One activity %	All activities %	
5	0,004	0,002	50000
4	0,01	0,005	20000
3	0,02	0,01	10000
2	0,04	0,02	5000
1	0,10	0,05	2000
NOTE 1: Maximum value is calculated as the average performance over all test sequences. NOTE 2: Nominal protection factor is the reciprocal of the IL obtained during all activities (100 : IL)			

So as a partial conclusion regarding the protection factor against solid contamination depending on the PPE you can consider the following:

By wearing a full face mask associated with a non ventilated protective suit, we can expect a PF for 2000 for respiratory tracts, and 50 for body. However, if we choose to wear a ventilated pressurized protective suit, we will easily get a PF higher than 50 000 (i.e class 5) for whole body including respiratory tracts.

2.4. Basic requirement no. 2 and no. 3

No. 2 The PPE has to be designed in a way to minimize as much as possible the Extra-risk due to the PPE itself

No. 3 Comfort and efficiency: PPE has to be designed to offer maximum comfort and efficiency .

A CEPN evaluation has been conducted 1991 to 1994 (Report n°226/CEPN) pointing and evaluating the impact of various PPE on the exposed time. Also a NEA compilation OECD 2009/NEA N°6399 Work Management to Optimize Occupational Radiological Protection at Nuclear Power Plants [8].

And as a partial conclusion, the use of an inappropriate PPE can increase up to more than 80% the time to perform the same task. Which also means in nuclear surroundings 80% more dose intake.

Main reason: Ergonomics and Generated Heat stress.

Most of the time, the use of a non-ventilated PPE for body protection generates heat

stress and cardiac risks. This induces both discomfort and painful job. Moreover it has a dramatic impact on the time increase to perform the same task.

Regarding painful: French CEA does not allow to work a two time a day - 45min, inclusive dressing and undressing. This is established if the worker is only equipped with a full face mask and non-ventilated suit!

3. CONCUSIONS

Based on the risk evaluation and the progress being made in the field of PPE which is intended to protect against nuclear risks, a major development has been made in the ventilated pressurized protective concept. The main benefits are:

(a) Full body contamination protection increased by a factor of 1000
(b) Heat stress removal that allows to perform the same task much faster
(c) Respiratory tracts protection by a factor of more than 100.

Those equipment have been developed in various shape depending on:

- Working place (reactor outage, fuel cycle industry, and decommissioning, post nuclear accident.)
- Various raw material
- Various air-fed system: air-fed thanks to a breathable network or self-fed thanks to a blower equipped with filtration cartridges also thanks to the lesson learned after Chernobyl or Fukushima accident.

The use of this European Directive as a tool is highly helpful to conduct a systematic approach regarding PPE design and progress. It also allows to accurately evaluate the efficiency against the contamination risk to face and to select the most appropriate Personal Protective Equipment for the work that has to be performed.

In line with EC89/686 Basics requirements, huge progress has been made during the last decade regarding PPE, in terms of contamination protection, extra risk reduction as well as comfort and efficiency.

REFERENCES

[1] COUNCIL DIRECTIVE of 21December 1989 on the approximation of the laws of the Member States relating to personal protective equipment (89/686/EEC)EC 89/686, dated 10/1996 (1996).
[2] COUNCIL DIRECTIVE of 30 November 1989 on the minimum health and safety requirements for the use by workers of personal protective equipment at the workplace 89/656/EEC (Employers' obligations).
[3] NF EN 529, Respiratory Protective device - Recommendations for selection, use, maintenance and Care, (2006).
[4] NF EN 136, Respiratory Protective device- Full face mask: Requirement, Testing, Marking.
[5] EN 1073-1, Protective clothing against solid airborne particles including radioactive contamination- Part 1: Requirements and test methods for compressed air line ventilated protective clothing, incorporating integral respiratory protection.
[6] EN 1073-2 : Protective clothing against radioactive Contamination .Part 2: Requirements and test methods for non-ventilated protective clothing against particulate radioactive contamination.
[7] EN 14325:2004, Protective clothing against chemicals – Test methods and performance classification of chemical protective materials, seams, joins and assemblages (2004).
[8] OECD 2009/NEA N°6399 Work Management to Optimise Occupational Radiological Protection

at nuclear Power Plants (2009).

RECONSTRUCTIVE DOSIMETRY AND THE CLINICAL OBSERVATION BASED ON LOCALIZED RADIATION INJURIES IN RADIOLOGICAL ACCIDENTS

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Abstract

According to the International Atomic Energy Agency (IAEA), a significant number of radiological accidents have happened in the last years. These accidents have occurred mainly on practices referred to as potentially high-risk activities, such as radiotherapy, large irradiators and industrial radiography. These radiological accidents have caused severe injuries in exposed persons due to high radiation doses. In the industry´s area, especially in industrial radiography, 80 cases involving 120 radiation workers, 110 members of the public and 12 deaths were recorded by IAEA and UNSCEAR. In Brazil, relevant radiological accidents have only occurred in industrial practices (excluding the Goiania radiological accident in 1987) resulting in the development of the Cutaneous Radiation Syndrome in hands and fingers. Brazilian data include 5 serious cases related to industrial radiography (gammagraphy), affecting 7 radiation workers and 19 members of the public, none of them fatal. Some methods of reconstructive dosimetry have been used to estimate the radiation dose to assist medical treatment to be prescribed. Moreover, the clinical observation of the effects of radiation in the exposed body areas is the first estimation that one can achieve. This paper presents the state of the art of reconstructive dosimetry used to estimate the radiation doses and a comparison between the results of dose calculation using the software Visual Monte Carlo and the clinical observation based on localized radiation effects.

1. INTRODUCTION

According to the International Atomic Energy Agency (IAEA), a significant number of radiological accidents have happened in the last years. These accidents have occurred mainly on practices referred to as potentially high-risk activities, such as radiotherapy, large irradiators and industrial radiography.

These radiological accidents have caused severe injuries in exposed persons due to high radiation doses. In the industry´s area, especially in industrial radiography, 80 cases involving 120 radiation workers, 110 members of the public and 12 deaths were recorded by IAEA and UNSCEAR. Brazilian data include 5 serious radiological accidents affecting 7 radiation workers and 19 members of the public, resulting in the development of the Cutaneous Radiation Syndrome in hands and fingers. One of the latest Brazilian cases (2000) is about an operator that was performing routine exposures with a ^{60}Co apparatus containing a 2.11 TBq source. He suffered a partial-body exposure after keeping his left hand very close to the radioactive source for, approximately, 30 seconds. [1,2]

Over the past 10 years, several techniques have been used to perform the reconstructive

dosimetry and to evaluate accidents radiation doses. These radiation doses can be estimated by physical, computational and biological dosimetry methods and clinical parameters. These methods have been used, also, to assist the medical staff in the evaluation and prescription of suitable medical procedures for the patient's treatment and follow-up [3]. Moreover, the clinical observation of the effects of radiation in the exposed body areas is the first estimation that one can achieve.

2. OBJECTIVE

This paper presents the state of the art of reconstructive dosimetry used to estimate the radiation doses and a comparison among the results of dose calculation using the dosimetry methods and the clinical observation based on localized radiation effects.

3. STATE OF THE ART OF RECONSTRUCTIVE DOSIMETRY

A bibliography review of the state of the art of reconstructive dosimetry in the last 10 years was done mainly about the physical, biological and computational methods and clinical parameters.

The most important physical dosimetry methods are the luminescence methods including thermoluminescence and optical stimulated luminescence and the use of electron paramagnetic resonance. They have their strengths in determining absorbed dose in bricks and porcelain or in the electronic components of mobile phones.

The most common biological method used for biodosimetry is the cytogenetic technique that provides the analysis of the cytogenetic damage in peripheral blood lymphocytes induced by ionizing radiation. Other biological methods are used, such as, genetic techniques; haematological techniques and protein biomarkers.

The computational dosimetry methods are based, generally, on Monte Carlo method, such as, the dosimetry by numerical computer code MCNPX and the computational program based on voxel anthropomorphic phantom associated with Monte Carlo Methods. The numerical code without Monte Carlo method is still used by the Geant4 code.

The Brazilian Monte Carlo calculation code, "Visual Monte Carlo –VMC, is used as computational dosimetry method. VMC can be used as for whole-body as for partial-body dose calculation. The hand voxel simulator from the voxel simulator NORMAN was also used. This simulator is based on information from a whole-body magnetic resonance image scan of a real man, adjusted to make the simulator the same height (1.76 m) and mass (73 kg) as the reference man as defined in ICRP publication 2003. The size of each voxel is 2.08 mm x 2.08 mm x 2.02 mm. The tissue type (e.g. bone, muscle) of each voxel is defined.

The clinical parameters to evaluate radiation doses are based on observation of the clinical signs and symptoms of the radiation effects (Acute Radiation Syndrome) or the localized radiation injuries (Cutaneous Radiation Syndrome) relating to radiation dose [4,5]. The absorbed doses to hands were high, enough to cause deterministic effects, notably serious localized injuries to the fingers of the left hand, called as Cutaneous Radiation Syndrome. Summarizing the literature, the following symptoms can be expected: erythema for doses between 3 and 10 Gy; dry desquamation between 10 and 15 Gy; wet desquamation between 15 and 25 Gy; and necrotic lesions for doses higher than 25 Gy.

4. RECONSTRUCTIVE DOSIMETRY AND THE CLINICAL OBSERVATION OF THE ACCIDENT

The Brazilian industrial gamma radiography operator, that was performing routine exposures with a ^{60}Co apparatus containing a 2.11 TBq source, in May 2000, suffered a

radiological accident after keeping his left hand very close to the radioactive source, for approximately 30 seconds, on Day 1. In order to estimate the absorbed doses received by the operator's left hand, the ranges of radiation doses estimated by physical and computational dosimetry methods and their corresponding clinical effects were taken into account [6].

The physical dosimetry method was performed in two steps: first to estimate the effective dose (whole-body) and second to estimate the absorbed dose to the hand (partial-body). The effective dose to the operator was estimated by the film badge individual monitor that recorded a dose of 88.1 mSv. The absorbed dose estimated for the operator's hand was done by physical simulation through the simulator of a left hand with thermoluminescence dosimeters. The highest doses estimated were in the thumb and index finger with injuries that were recorded 7.41 Gy and 17.56 Gy, respectively.

The biological dosimetry method used was the cytogenetic analysis that was done with the operator on Day 15. Dose assessment through this biological indicator was performed and no dicentrics were observed at one thousand cells scored. Therefore the average whole-body dose, estimated by biological dosimetry, was lower than 60 mGy.

The computational dosimetry methods used was based on the Brazilian Monte Carlo calculation code, named "Visual Monte Carlo –VMC, with human body voxel simulator, to calculate the absorbed dose received by each organ and tissue relevant to the calculation of effective dose and to simulate the accident while taking into account the specific morphology of the irradiated individual, as well as the source characteristics. This computacional code also allowed to estimate the dose on localized radiation injury. The highest doses estimated in the thumb and index finger with injuries were 7.80 Gy and 15.90 Gy, respectively.

The clinical parameters, corresponding clinical effects, were taken into account to estimate the radiation dose. The operator presented on Day 8 oedema and erythema in his hands. The appearance of blisters on his left hand, mainly on the thumb and index finger (Day 21), and dry desquamation on both hands (Day 23), lead the medical staff, mainly based on the onset and extent of the wet desquamation, to assess an absorbed dose between 10 and 20 Gy to the most affected fingers.

5. CONCLUSIONS

The distribution of absorbed doses on the operator's left hand, as estimated through physical dosimetry methods and Visual Monte Carlo dose calculation program, is in accordance with the clinical observation based on localized radiation effects manifested at the operator's hand.

Comparing the results obtained through physical reconstruction to those obtained by VMC, about half of the pairs of results obtained through both methods were similar within a range of uncertainty of 20%. Thus, the VMC software can be considered suitable for estimating the distribution of doses to the hands. Using this VMC code it is possible to estimate quickly radiation dose at localized radiation injury.

This initial dose estimate through clinical indicators served as a reference, making easier the retrospective dose reconstruction.

REFERENCES

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Lessons learned from accidents in industrial radiography, Safety Reports Series, No. 7, IAEA, Vienna (1998).
[3] DA SILVA, F.C.A. A methodology for analysing radiological accident in industrial radiography, M.Sc. Thesis, IME, Rio de Janeiro, Brasil (1990).
[4] DA SILVA, F.C.A.. Reconstructive dosimetry of a radiological accident using the Monte Carlo method. D.Sc. Thesis – COPPE/UFRJ, Rio de Janeiro, Brasil (2003).

[5] AINSBURY, E. A., et al. Review of retrospective dosimetry techniques for external ionizing radiation exposures, Radiation Protection Dosimetry, Vol. 147, No. 4 (2011) 573–592.
[6] HEALTH PROTECTION AGENCY, High dose radiation effects and tissue injury. Report of the Independent Advisory Group on Ionising Radiation, (2009).
[7] DA SILVA, F.C.A.. Dose reconstruction of a Brazilian industrial gamma radiography partial-body overexposure case, J. Radiol. Prot. 25 (2005) 289–298.

A NOVEL DEVICE FOR PREVENTING ACUTE RADIATION SYNDROME AND REDUCING CUMULATIVE MARROW DOSE

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Abstract

In order to shield as much of the body as possible, existing personal shielding solutions use only thin layers of inherently heavy radiation-attenuating materials. These types of solutions are ineffective in blocking energetic gamma radiation. Receiving a high dose of gamma radiation over a short period of time may result in Acute Radiation Syndrome. Protracted exposures to gamma radiation may result in malignancies such as leukemia. In the case of high-dose exposure, the survival-limiting factor at doses upto 10 Gy is irreversible bone marrow (BM) damage. Notably, doses in past nuclear catastrophes were largely under 10 Gy. Thus, numerous fatalities in a catastrophe may be avoided by preserving Bone Marrow (BM). Remarkably, due to its extraordinary regenerative potential, it is enough to protect only a small volume of BM to preserve its viability. In the case of protracted exposure, BM is also very susceptible to carcinogenic effects of radiation. Thus, exposure of large areas of BM to radiation significantly increases the risk of leukemia. Approximately 50% of the of the body's active BM is contained within the pelvis. As such, shielding this region holds great promise. In this study, a first-of-its-kind device providing potentially life-saving protection from gamma radiation is presented.

1. INTRODUCTION

At present, there is no effective personal protection from gamma radiation. While no efforts are spared to prevent exposure to doses of radiation that may lead to detrministic effects, such exposures unfortunately cannot be excluded due to the volatile nature of emergency events. In Chernobyl, first-responders entered the nuclear meltdown site equipped with ineffective makeshift lead sheeting for protection from gamma radiation [1]. Many of those brave individuals suffered from ARS due to high dose exposures. Only recently, in Fukushima, emergency personnel undertook critical disaster mitigating activities without protection from potentially lethal gamma radiation [2]. The long-term effects of this gamma exposure are yet to be known.

Even on a routine basis, radiation professionals run a small but palpable risk of being exposed to potentially harmful gamma radiation either due to accidental exposure or on a cumulative basis. Such professionals include select employees of the nuclear industry or of any other occupation involving the use of gamma sources (e.g. non-destructive testing,

industrial sterilization, etc.)

There is thus an urgent need for personal protective equipment that is able to shield against both high dose and cumulative exposures to gamma radiation.

2. METHODS

2.1. The making of a Bone Marrow Shield

An exhaustive anatomical study of active BM distribution in the human skeleton using the Visible Human data set was performed [3]. Based on this, and in combination with Monte Carlo simulations, we aimed to develop a device to selectively shield a critical volume of BM in a realistic setting. As it is enough to preserve the viabilty of only 5% of bodily BM to allow for hematopoitic reconstitution [4], this was defined as the absolute minimal amount that should be rescued even at 10 Gy radiation. A belt-like radiation protection device for the pelvic bone marrow was developed and named the "StemRad 360$^{\gamma}$" (Fig. 1A). The belt contains a radiation-attenuating component comprised of numerous layers of uniquely cut lead sheet layers that when compiled form a topography of a thickness inversely related to the thickness and density of the tissue present between the device and the protected bone marrow the (iliac crest, Fig. 1B,C). By uniquely bringing into account the natural shielding properties of human tissue, this shield configuration guarantees that only the minimal amount of radiation attenuating material needed is used. This shield is designed to provide a substantially uniform dose to 240 cm^3 of active marrow in the posterior pelvis and lesser degrees of protection to an additional ~700 cm^3 of marrow (assuming adult male).

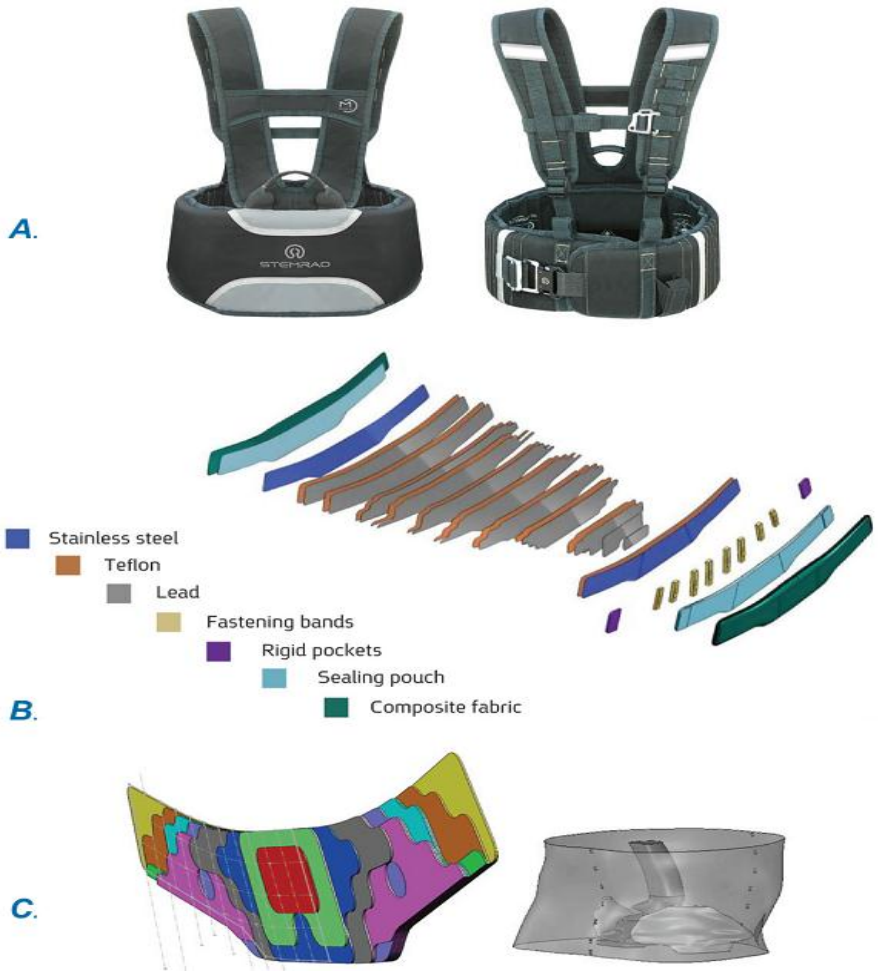

FIG. 1. *Structure of the StemRad 360$^{\gamma}$. A. Front and back views of exterior. B. An exploded view of the radiation attenuating component. The different elements are color-coded to match descriptions in the legend. C. The topography of the attenuating component. The component is made of uniquely shaped lead layers which when compiled create a topography (left) reflecting the unique anatomy of the pelvic marrow (right).*

2.2 Bone Marrow Shield Testing

To test the StemRad 360$^{\gamma}$, a life size phantom model of a human body was created by placing anatomically accurate human skeletons in enclosures having the approximate contours of the human body. Sixty TLD dosimeters were embedded evenly throughout the pelvic marrow space of the skeletons (Fig. 2). Enclosures were filled with tap water, which served as a tissue equivalent. The BM-shielding device was placed on the protected phantom in a manner akin to what is intended for humans. Phantoms were irradiated with a configuration of Cs-137 presented as fallout, ensuring isotropic exposure (Fig. 2). BM doses in the presence vs. in the absence of the BM-shielding device were determined by harvesting the TLDs and reading them on a TLD reader (Harshaw 3500).

3. RESULTS AND DISSCUSSION

To test the StemRad 360$^{\gamma}$, life size phantom models of the human body were created and sixty TLD dosimeters were embedded evenly throughout the pelvic marrow space of the skeletons. Phantoms were irradiated with a configuration of ^{137}Cs presented as fallout, ensuring isotropic exposure (Fig. 2).

BM doses in the presence vs. in the absence of the StemRad 360$^{\gamma}$ were determined. The shielding provided by the StemRad 360$^{\gamma}$ was significant throughout the pelvis and was especially evident in the posterior iliac crest. We expect that effectively protecting this area (the site from which marrow is harvested for transplantation) should translate into a dramatic difference in survival, as demonstrated below.

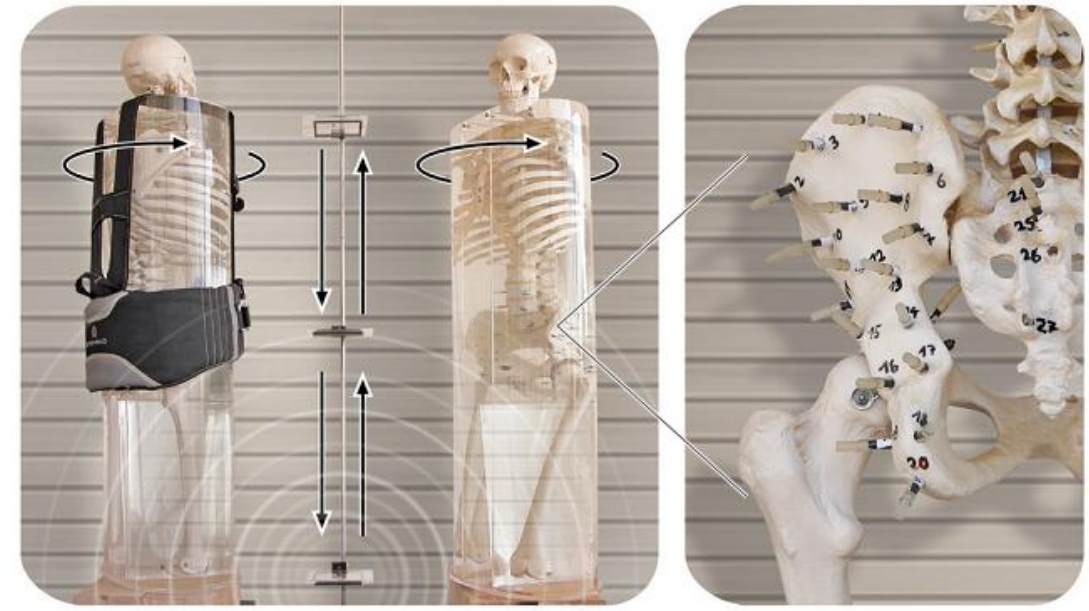

FIG. 2. Experimental Set-Up: Human skeleton replicas with TLDs embedded throughout pelvic BM cavities were submerged in water to create phantom models. Protected or unprotected phantoms were irradiated with Cs-137. To create a realistic setting, the source was shifted in relation to the phantoms along the Z-axis while the phantoms were rotated along the X-Y plane. Water-proof glass tubes containing a TLD each are shown (inset).

Knowing the dose attenuation conferred by the StemRad 360$^{\gamma}$ to specific marrow volumes, we were able to determine the dose-to-volume histogram in its presence or absence (Fig. 3, blue bars). This allowed us to determine, based on the human BM radiosensitivity curve, the amount of live BM that would remain following a Chernobyl-like 9 Gy whole-body exposure in the absence vs. in the presence of the StemRad 360$^{\gamma}$ (Fig. 3, red bars). Adding together the amounts of live marrow remaining gave us a total of 27 grams of live marrow for an individual exposed without protection and a total of 127 grams of live marrow for an individual equipped with the device. The significance of this difference cannot be overstated; the minimum quantity of live marrow necessary for the reconstitution of a lethally irradiated average sized adult is approximately 50 grams (National Marrow Donor Program).

The ability of the BM-shielding device to protect the pelvic marrow from a cloud-like source was corroborated by employing MCNP Monte Carlo simulations. Here too, attenuation was especially pronounced in the posterior iliac crest.

It is well documented that the likelihood of hematological malignancies increases following exposure of pelvic marrow to ionizing radiation [5, 6]. As such, the significant shift in dose to volume provided by the BM shielding device is beneficial in protecting the pelvic marrow not only from the acute health effects of radiation but also from emergence of malignancies such as leukemia due to cumulative exposure.

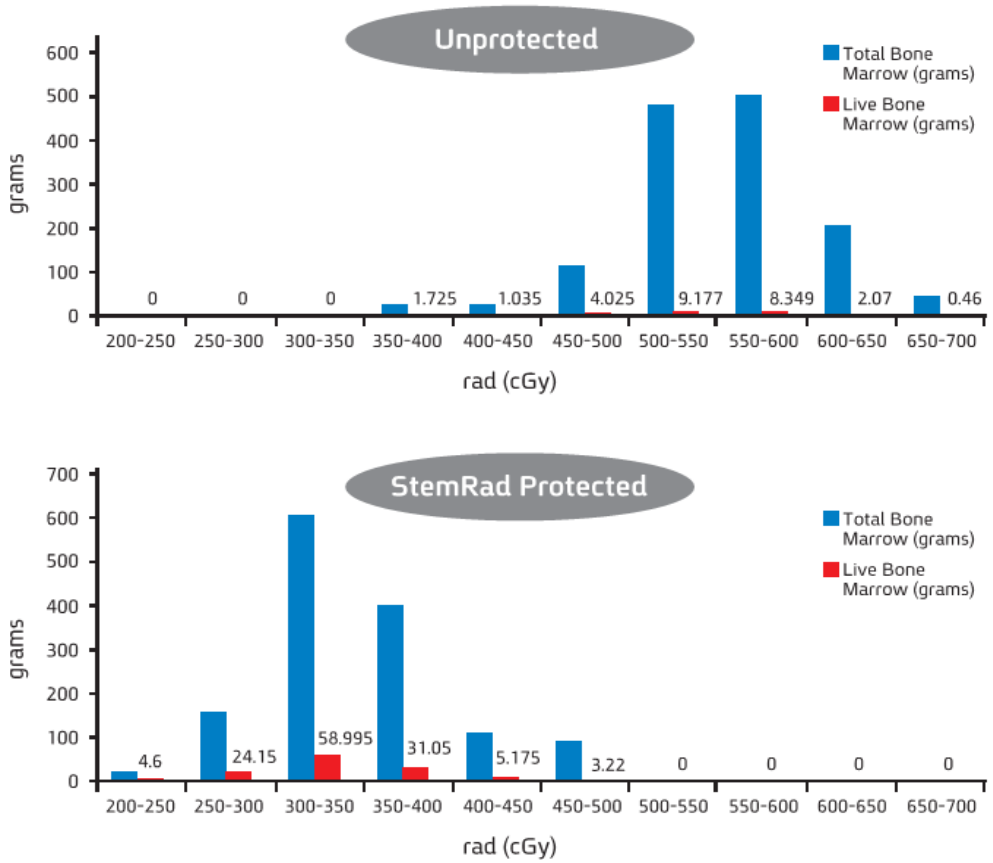

FIG. 3. Dose to bone marrow volume at 9 Gy ^{137}Cs and resulting live marrow quantities. The distribution of red marrow volumes into 50 cGy dose bins and corresponding live marrow quantities is shown for an unprotected individual and for an individual protected with the bone marrow shield. Viability of marrow was determined based on documented marrow radiosensitivity.

4. CONCLUSION

A device was developed that even at radiation doses above 9 Gy, is able to protect a volume of BM which is sufficient for hematopoietic reconstitution thereby preventing lethal ARS. Equipping responders with this novel device should offer dramatic improvements to survivability even under extreme radiological scenarios in which prevention of exposure to high dose radiation fails.

This device may also aid in reducing the cumulative marrow dose over the lifetime of the employee if used while performing tasks with potentially abnormal gamma exposures. As such, it may be used to protect professionals in a variety of occupations ranging from gamma radiography to agricultural sterilization.

REFERENCES

[1] UNSCEAR. Sources and Effects of Ionizing Radiation- Report to the General Assembly, (2008).
[2] WADA, K., et al., Emergency response technical work at Fukushima Dai-ichi nuclear power plant: occupational health challenges posed by the nuclear disaster, Occup. Environ. Med, 69(8) (2012) 599-602.
[3] ACKERMAN, M.J., et al., The Visible human data set: an image resource for anatomical visualization, Medinfo., 8 Pt 2 (1995) 1195-8.
[4] MACVITTIE, T.J., et al., The prolonged gastrointestinal syndrome in rhesus macaques: the relationship between gastrointestinal, hematopoietic, and delayed multi-organ sequelae following acute, potentially lethal, partial-body irradiation, Health Phys., 103(4) (2012) 427-53.
[5] INSKIP, P.D., et al., Leukemia, lymphoma, and multiple myeloma after pelvic radiotherapy for benign disease, Radiat. Res., 135(1) (1993) 108-24.
[6] WRIGHT, J.D., et al., Pelvic radiotherapy and the risk of secondary leukemia and multiple myeloma. Cancer, 116(10) (2010) 2486-92.

THE EVALUATION OF AIRBORNE RADIOACTIVITY FOR OCCUPATIONAL EXPOSURE IN ACCIDENT CONDITIONS

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ABSTRACT

According to Chinese nuclear safety regulation HAF102 “Safety Code on Nuclear Power Plant Design” and the standard GB18871-2002 “Basic Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources”, comprehensive safety assessment for nuclear power plant is required to estimate the potential exposure under operation conditions and accident conditions. This evaluation is required during review of the design, and to ensure that the occupational exposure will meet the requirements in the regulations and standards. Compared with the dose estimation from sealed source, doses resulting from the airborne radioactivity to workers are not comprehensively considered before. This paper is devoted to describe the relevant analysis of airborne radioactivity for workers exposure under accident conditions.

1. INTRODUCTION

The estimation of the potential exposure depends on post-accident operation procedure which determines the local working area. In addition to the estimations of sealed source exposure in the operation area and passages, The exposure of airborne radioactivity for workers should be also included. In some cases, the main dose contribution is from airborne radioactivity to workers compared with the exposure from sealed sources in NPP buildings.

According to the Chinese safety guide "The emergency preparation and emergency resp onse of Nuclear Power Plant", the habitability of the main control room in design should be c onsidered to satisfy the requirements of habitability criteria. Therefore the estimation of airbor ne radioactivity includes both in the local operation area and in the main control room under accident conditions.

2. METHOD

2.1 The assumptions of source term in containment

The radioactivity fractions released to containment under accident conditions are adopted the level 2 source term according to the Chinese standard NB/T 20194-2012 "The design Criterion of radiation Shield in PWR nuclear power plant ".The released fractions of nuclides from the core are listed in Table 1.

TABLE 1. THE RELEASED FRACTION OF NUCLIDES FROM THE CORE UNDER ACCIDENT CONDITIONS

Nuclides	Fraction(%)
Kr、Xe	100
I、Br	75
Cs、Rb	75
Te、Se、Sb	30.5
Sr、Ba	12
Ru、Rh、Pd、Mo、Tc、Co	0.5
Ce、 Np、Pu	0.55
La、Zr、Nd、Eu、Nb、Pm、Pr、Sm、Y、Cm、Am	0.52

2.1 The assumptions of airborne radioactivity

Two aspects of airborne radioactivity are considered. The first is the concentration of airborne radioactivity in building where it is operation area after accident, the second is the concentration of airborne radioactivity in the main control room resulted from the air intakes of emergency habitability system and leakage of the main control room pressure boundary.

2.2.1 The airborne activity in safety building

According to post-accident operation procedures, there are some operation positions are located outside the reactor building designed in Chinese NPP, for example in safety building. The concentration of radioactivity in safety building is determined by the decay of nuclides, the leakage of double containments and the function of filter in annular area. For the simplified purpose, the direct leakage fraction from the inner containment is considered without weaken. It is assumed that the concentrations of safety building and annular area of double containments is homogeneous .The radioactivity concentration of main nuclides in safety building is shown in Table 2.

TABLE 2. THE AIRBORNE RADIOACTIVITY CONCENTRATION OF MAIN NUCLIDES IN SAFETY BUILDING AT DIFFERENT TIME IN ACCIDENT CONDITION

Nuclides	The concentration of activity(Bq/m^3)				
	1h	7h	24h	7d	15d
Kr-85	1.54×10^{3}	1.76×10^{4}	5.86×10^{4}	2.24×10^{5}	2.62×10^{5}
Xe-131m	1.90×10^{3}	2.14×10^{4}	6.84×10^{4}	1.85×10^{5}	1.36×10^{5}
Xe-133	2.93×10^{5}	3.24×10^{6}	9.83×10^{6}	1.70×10^{7}	6.93×10^{6}
Xe-133m	9.21×10^{3}	9.73×10^{4}	2.59×10^{5}	1.49×10^{5}	1.39×10^{4}
Xe-135	7.43×10^{4}	5.38×10^{5}	4.91×10^{5}	3.24×10^{1}	1.69×10^{-5}
I-131	1.12×10^{4}	1.25×10^{5}	3.92×10^{5}	8.94×10^{5}	5.24×10^{5}
I-133	2.23×10^{4}	2.09×10^{5}	3.96×10^{5}	1.25×10^{4}	2.44×10^{1}

2.2.2. The airborne radioactivity in main control room habitability area

The radioactivity of main control room habitability area is resulted from the air intakes

of emergency habitability system and the leakage of pressure boundary of habitability area .The reduction factors include nuclides decay, the exhaust air of emergency habitability system and the inner close loop ventilation system. The concentration of activity in main control room is described as below:

$$\frac{\mathrm{dA}}{\mathrm{dt}} = \left[Q_f(1-f)\cdot\left(\frac{\chi}{Q}\right)_1\cdot R - Q_u\cdot\left(\frac{\chi}{Q}\right)_2\cdot R\right] - \left(\lambda_i + \frac{Q_r}{V_r}\cdot f_r + Q_e/V_r\right)\cdot A \quad (1)$$

foot note 1 is the air intake of ventilation system
foot note 2 is the location of leakage of pressure boundary

Where,

A is the radioactivity in control room habitability area,

R is the releasing rate of nuclides from containment,

χ/Q is the atmosphere dispersion factor,

$Q_{f,}Q_u, Q_e$ is the air intake flow rate through filter, the leakage flow rate and the exhaust air flow rate of main control room separately

$\frac{Q_r}{V_r}\cdot f_r$ is the reduction of nuclides by the inner close loop ventilation system,

3. DOSES ESTIMATION RESULTS

The dose estimation includes external exposure and inhalation committed effective dose. The concentrations of activity in safety building and occupational doses are estimated on basis of the assumptions above. The results are listed in Table 3 and Table 4. For main control room ,the exposure pathways also include the exposure resulted from the airborne activity outside the main control room and the direct exposure from the reactor building as well .It is demonstrated that the main dose contribution to workers is from the airborne activities inside the main control room for workers.

TABLE 3. THE DOSE RATES IN SAFETY BUILDING AFTER ACCIDENT

	Dose rate level at different time（Sv/h）				
	1hour	7 hours	24 hours	7 hours	15 hours
external exposure	6.55×10^{-5}	1.61×10^{-4}	8.05×10^{-5}	2.03×10^{-5}	8.16×10^{-6}
Inner exposure	2.73×10^{-4}	2.23×10^{-3}	3.57×10^{-3}	1.97×10^{-3}	9.62×10^{-4}

TABLE 4. THE CUMULATED DOSES OF OPERATOR IN MAIN CONTROL ROOM AFTER ACCIDENT (Sv）

Time(h)	Equivalent dose of hypothyroid	Equivalent dose of inner exposure	Equivalent dose of Immersion exposure	Total effective dose
2	1.63×10^{-2}	9.32×10^{-4}	1.12×10^{-3}	2.05×10^{-3}
8	4.41×10^{-2}	2.45×10^{-3}	5.02×10^{-3}	7.46×10^{-3}
24	6.98×10^{-2}	3.81×10^{-3}	7.16×10^{-3}	1.10×10^{-2}
96	9.40×10^{-2}	5.10×10^{-3}	7.88×10^{-3}	1.30×10^{-2}
720	1.34×10^{-1}	7.50×10^{-3}	8.61×10^{-3}	1.61×10^{-2}

4. DISCUSSION

4.1 The on-off time for double air intakes of emergency habitability system

The emergency habitability system adopts double air intakes. The cumulated distribution frequency (CDF) to average atmosphere dispersion factors of double air intakes are shown in Fig. 1. The curves in Figure 2 standard for variation of CDF to average atmosphere dispersion factors with different on-off time between two air intakes.

The on-off time between double air intakes is determined by the sensibility of monitoring system and consideration of relative concentration stability. It is shown that the shorter of the on-off time, the lower of the average atmosphere dispersion factors. Under the condition of the 95% CDF and 10 minutes on-off time, the average dispersion factors of double air intakes is nearly 8 times lower than that of single air intake.

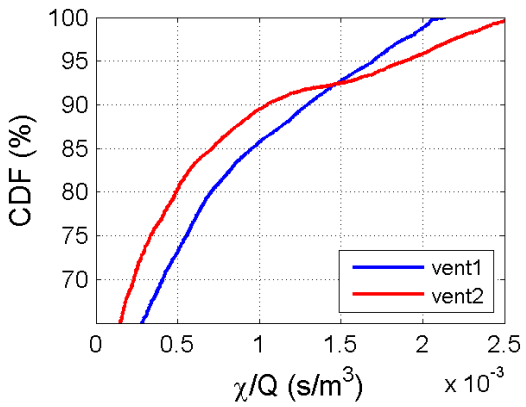

FIG. 1. The variation of CDF to average dispersion factors

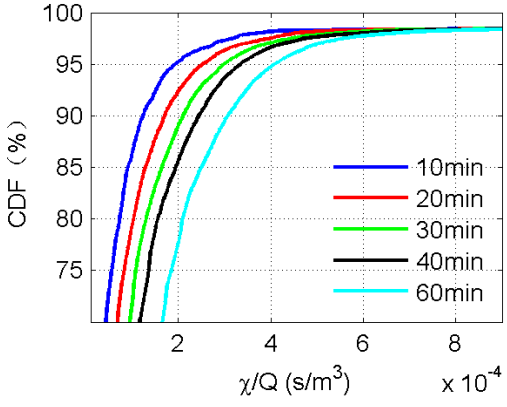

FIG. 2. The variation of CDF to average atmosphere dispersion factors with different on-off time

4.2 The effect of inner close loop ventilation system.

In design, we consider there is some leakage pathways to main control room habitability area without passing through the filter after accident .It will increase the radioactive concentrations in habitability area. But the design of the inner close loop ventilation system will reduce the concentrations of iodine and aerosol. The activity of ^{131}I that varies with the recirculation flow rate passing through the filter in accident is shown in Figure 3.The efficiency of filter is 99%. With the filtered recirculation air flow rate of 1000m^3/h(RF), it is shown that the concentration of ^{131}I from 0 to 24 hours will be half of those without filter. With the increasing of the flow rate of inner close loop ventilation system, the reduction of concentrations is more obvious. The concentration of ^{131}I with the filtered recirculation air

flow rate of 6800 m^3/h will be nearly one to seventh of that without filter.

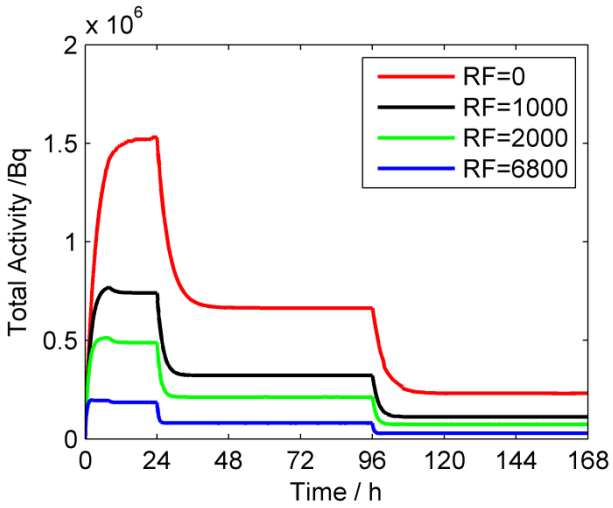

FIG. 3. The variation of activity in main control room habitability area along with time

5. CONCLUSION

In general, the dose of airborne radioactivity has less contribution to the workers compared with the dose resulted from sealed sources. But in the main control room habitability area, the main dose contribution is from airborne radioactivity. The evaluation results can meet the Chinese regulation (the effective dose of occupational exposure is less than 50mSv and the equivalent dose of hypothyroid is less than 500mGy during the period of emergency response(normally 30 days)). In order to reduce the doses from airborne radioactivity, the leak tightness of the penetrations of safety building and the main control room habitability area should be enhanced. In order to ensure the habitability of the main control room , the double air intakes can be adopted for emergency habitability system and inner close loop ventilation system is considered in design to reduce the concentration of radioactivity .

REFERENCES

[1] CHINESE NUCLEAR SAFETY REGULATION, Safety Code on Nuclear Power Plant Design, HAF102-2004.
[2] CHINESE NUCLEAR SAFETY STANDARD, Basic standards for protection against Ionizing radiation and for the safety of radiation sources, GB18871-2002.
[3] CHINESE STANDARD, The design criterion of radiation shield in PWR nuclear power plant, NB/T 20194-2012.
[4] CHINESE NUCLEAR SAFETY GUIDE: The emergency preparation and emergency response of nuclear power plant, HAD002/01-2010.

THE STRATEGY ENSURING RADIATION SAFETY OF THE LIQIDATORS IN A RADIATION ACCIDENT

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Abstract

This paper is about the accident on the solid radioactive waste storage No. 5, 6, 7 of State specialised enterprise «Kiev state interregional specialised enterprise» (hereinafter-KSISE), associated with radioactive water leakage from this storages to environment, including underground water.

1. INTRODUCTION

Near-surface storages, Nos. 5, 6, and 7 were commissioned in 1962. Storages represent themselves a rectangular containers, inset in soil with size 10m x 15m x 3m, divided on length into two sections. Bottom and walls of the containers were made of a monolithic concrete. Covered storage section with reinforced concrete slabs, over which arranged the waterproofing layer (one layer of asphalt concrete and two layers of roofing material on the asphalt mastic). The volume of each storage - 400 m^3 (volume of the section - 200 m^3). Project storage availability was not provided by covering over them.

The storage areas were filled with radioactive waste from the 60s to the 90s of the last century. According to the autopsy of the storage made in in 1995, it was found that the storage sections were almost completely filled. Waste volume in three storages is about 1000 m^3. According to the passport date, total tritium waste activity is 250 000 Ku ($3.25 \cdot 10^{15}$ Bq).

Status of radiation accident in storages No. 5, 6, 7 was established in 1995.Under the Act, based on the results of work of the State Commission of Ukraine, it was concluded that the leakage of tritium into the air and groundwater is due to the driver violating the integrity of the protective barriers, which are storage structures, that led to the spread of tritium within the first industrial site, and then in the sanitary protection zone. In storage No. 5, water level in 1995 was - 1 m above the surface of the waste. In storage No. 6 and 7, the water above the surface of the waste was not found, but streaks on the walls of storage pointed to its recent presence and the possibility of finding it in the bottom.

Since 1997, in order to stop falling precipitation to the emergency storages, the shed was built over them. Since then, until the mid-2011, no measures to eliminate accident have been conducted.

2. PRECAUTIONARY MEASURES

In 2011, based on the Governmental assignment and potential risks of tritium groundwater contamination in the south-west borders of Kiev, works were begun on the elimination of radiation accident.

It was decided:

(a) In the first stage: conduct work on the survey to clarify the amount of water in storages, its radioactive contamination, as well as perform the work on pumping the maximum possible water amount;

(b) The second stage: remove total waste from storages, move them into containers, build a new storage for this containers where they should be kept until to agree on moving them on "Vector" in the Chernobyl exclusion zone;
(c) The third step: carry containers with radioactive waste from emergency storage to "Vector".

During the period from 2011 to 2013, the first stage, when the water was pumped from the storages No. 5 and 6 was conducted. In the storage No. 7, water was not observed.

By a combination of factors influence the radiation situation at the working site, on the surface of storages Nos. 5, 6, 7 could qualify as hazardous:

i. The equivalent dose rate on the surface enclosed workplaces slabs ranged from 0.16 to 3.8 mSv/h and with open plates - from 3.8 to 17.0 mSv/h;
ii. The concentration of tritium in the working area when working, DK exceed PC_A^{inhal} (permitted concentrations) 3.5 - 12.0 times;
iii. Contamination of surface slabs, critical equipment and clothing should not exceed, respectively, 10,000 particles • min^{-1} • cm^{-2}, 11 particles • min^{-1} • cm^{-2} and 0.4 particles • min^{-1} • cm^{-2};
iv. The concentration of tritium in the water samples withdrawn from the storage No 5 and 6 was $n•10^{11}Bk/m^3$ that exceeds the allowable concentration of tritium in water for a population of $n•10^3$ times, this water can be attributed to the intermediate-level liquid radioactive waste, taking into account the concentrations of other radionuclides studied.

As a result of almost deserted the water pumping technology, the staff, who participated in the studies, received during the 3 year works total effective radiation dose does not exceed 1 mSv, the contribution of internal dose due to tritium does not exceed 11%. About 38 m^3 of water were recovered from storages, although a residual amount of the water in the form of sludge is left at the bottom and removing it in the presence of radioactive waste in containers is impossible.

The next stage of radiation accident elimination involves the waste extraction from storages. For radiation safety of these operations was necessary to consider the following:

a) in storages, there are the low-, medium-and high-level radioactive waste and contaminated surface radioactive materials;
b) radionuclide composition of waste presented: ^{137}Cs, ^{3}H, $^{90}Sr + ^{90}Y$, ^{60}Co, ^{14}C, ^{226}Ra, ^{232}Th, ^{239}Pu, ^{238}U and other short-and long-lived radionuclides;
c) waste were taken in packages which created equivalent dose rate more than 100 mkSv/h at 1 m from the surface, as well as high-level IRS in biosecurity; waste containing tritium were taken in disposable packaging;
d) wastes were dumped in storage in bulk-form without sorting;
e) most of the waste is wet.

Lack of experience in the elimination of such radiation accidents in Ukraine and other countries, the existing legal framework of Ukraine do not conclusively determine the principles and approaches to the treatment of such radioactive waste, the ultimate aim should be to rebury them. Moreover, the main issue that cannot be resolved within the framework of the current legislation of Ukraine, is to ensure the application of principles of radiation

protection of personnel NRBU-97 (non-exceedance, optimization and justification) subject to the established acceptance criteria.

Given the impossibility of any way affect the reduction of radioactive waste in the normative state - in accordance with the acceptance criteria of the complex "Vector", following decisions were taken, which are reflected in the developed by specialists KSISE "Concept (ideology and basic principles) of extraction from waste storage №5,6,7 KSISE".

3. CONCEPT (IDEOLOGY AND BASIC PRINCIPLES) OF EXTRACTION FROM WASTE STORAGE №5,6,7 KSISE

This concept in subsequently formed the basis of design documentation relating to the development of extraction technologies RW, necessary equipment and planning works):

1) Submit to governmental authority the proposals to introduce a legislation of Ukraine concept of "historical radioactive waste" (i.e. those radioactive waste which are placed in the near-surface repository special combine with 60s to 90s the last century, and in the disposal of which is not supposed to use them further, recovery, identification, sorting and recycling);
2) The issues of destroyed and ruined the waste packages, the lack of markings on the packages, the complexity of qualitative and quantitative determination of the radionuclide composition of the waste exclude identification operation, sorting and recycling of waste when removed from storage;
3) Tritium, decay products of the organic waste, unknown medical and laboratory preparations, mercury and other chemicals complicate the manufacturing operations, that require the presence of staff, making them cumbersome and dangerous, having elevated radiation risks in terms of impact on humans, so the presence of such operations in extraction technology should be minimized as much as possible;
4) The "historical radioactive waste" do not meet the acceptance criteria in the existing repository "Vector", so you need to develop specific criteria for acceptance of waste and reconcilea them with the regulatory body;
5) Minimizing of radiation accident and subsequent liquidation of the accident consequences (removal of storage structures, land rehabilitation) are possible only when the extraction of all waste containers and transfer them to a temporary storage container for disposal facilities before deciding on the reburial of "historical radioactive waste".

4. TECHNOLOGICAL SOLUTIONS TO EXTRACT THE WASTE

Based on these basic principles of "historical radioactive waste" management that take into account the real problems, risks and threats, following technological solutions to extract waste have been proposed:

- mechanization of technological operations for extracting and loading waste to containers, remote control equipment shall avoid direct contact the staff with radioactive waste in storage sections, and with individually derived packages;
- in the waste extraction process and loading them into containers identification operation and sorting packages of radioactive waste should be excluded;
- establishes a simplified requirements for the placement of recoverable waste in the container, which are defined remotely without the presence of personnel in the area of work;

- minimizing manual labor and residence of staff on the surface of storage (under the covering on the surface of storage №5,6,7) during the waste execution;
- required the use of modern means of individual protection of the body and respiratory system, excluding tritium and radon entry into the body of personnel;
- works performance should be in autumn and spring at temperatures above 10^0C, during this period of time the concentration of tritium in the working area should be significantly less than in summer;
- the protection of personnel, radiation safety measures must comply with the rules, regulations and standards established by normative acts of Ukraine, as well as internal documents KSISE.

The main technological solutions to extract radioactive waste from emergency storages comprises the following complex of works:

- the construction of the planned temporary covering for container storage of radioactive waste recovered from emergency storage;
- placement of the necessary equipment and the gradual opening of storage;
- removing waste from storage sections and loading in certified containers;
- transportation, receipt and emplacement of containers in a temporary covering for container storage of radioactive waste;
- certification of containers;
- decontamination of equipment used;
- radiation monitoring at all stages of the work.

5. CONCUSIONS

The "historical radioactive waste" extraction technology is planned to be performed at the store №5 as it is the safest in terms of its low capacity of liquid radioactive waste and harmful chemicals.

After the waste extraction from storage №5, the experience obtained are analyzed; the radiation safety measures are reviewed; the operations should be made perfect, the volume of work is adjusted and the need of equipment / containers for further use when working on the storage №6 and 7 is reviewed. Changes and additions are made to technical solution with subsequent amendments to the project documentation.

Given the application of all the above mentioned principles, the potential individual effective dose of emergency personnel was calculated. This dose should not exceed 12 mSv/year, which corresponds to a reference level of exposure to personnel KSISE during practice.

Session 7: Occupational radiation protection in the workplaces involving exposure to naturally occurring radioactive materials and cosmic rays

ENVIRONMENTAL IMPACT STUDIES OF HIGH RISK INDUSTRIES

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Abstract

A preliminary safety assessment was performed to estimate occupational and public exposures. Based on the scenarios considered, the radiation dose of the worker present during the dumping of the slag waste is lower than the Egyptian regulation. Iron & Steel factory, at El-Gamaza El-Cobra region, dumped the wastes generated during steel manufacturing as disposal piles outside the factory, in open area without engineered structures. These wastes are considered technologically enhanced naturally occurring radioactive materials (TENORM wastes).The nearby individual who lives at 100 m away from the disposal area is exposed to a higher dose than the regulation. In order to reduce dose to the nearby individuals, shallow ground disposal in the form of trench design is proposed for temporary disposal. Additionally, the ground disposal of the slag waste may contaminate the groundwater from domestic uses. Therefore, a second preliminary safety assessment was carried out to evaluate the radiological impact of the TENORM which could be disposed in trench in term of ingestion dose received by individuals' drinking water from a domestic well at 100 m away from this trench.

1. INTRODUCTION

Naturally-occurring radioactive materials (NORM) are found in rocks, fertilizers, coal fly-ash, coke slag, fire bricks, soil and others. Some sources of natural radiation have been enhanced by human technological activities (TENORM), [1]. Exposures to radiation originating from TE-NORM containing residues can threat and impact human health during industrial activities. These impacts depend on mode of disposal, the local environment, and demographics of the population in the region. Meanwhile as a general rule in some civilized countries, waste containing mixture of hazardous chemical, radioactive and biological agents is treated as radioactive waste [2].The study area represent an industrial zone of different manufactories: iron & steel, coke, chemicals, metal industries. A number of these factories are not connected to the sewage network and discharged their liquid effluents directly to the agriculture drains [3, 4] in addition of direct exposure to individuals. Therefore with the absence of specific national TE-NORM regulations, the radiological impacts from the disposal piles of TENORM should be studied through a safety assessment analysis and safety assessment studies of TENORM waste trench disposal

1. RADIOLOGICAL IMPACT OF SLAG DISPOSAL PILES

Slag material is a byproduct granular material obtained from Iron & Steel industry. It has particles ranging from coarse grit (1-2 mm) down to fine dust (<0.05 mm) and known by high content of heavy metals which are available to be absorbed into the bloodstream by ingestion or if finer particles are inhaled [5, 6].

2.1 Potential radiological hazard of slag waste

Radioactive materials, either particles or gases, may be transported great distances by local and large-scale air movements. Air releases are by far the major source of radiological exposures of the

public from the disposal piles of slag waste.. Therefore, the inhalation is the primary route of exposure to the TE-NORM slag dust. Since slag waste contain many radionuclides, including ^{232}Th, ^{238}U, and decay products of these two radionuclides, exposures resulting from the inhalation of ^{232}Th, ^{238}U and their decay products from inhalation of slag dust lead to bronchogenic lung cancer [7].

2.2 Specific activity of TE-NORM source term

The specific activity of TE-NORM in the slag waste is the only source term considered. Based on the time frame of scenarios developed, radionuclides with half life lower than one year are eliminated and dose estimated from ^{222}Rn (radon gas) is excluded and will be evaluated in separate studies [8, 9].

1 tone of Coke + 2 tones of Iron +0.5 tone of Limestone = 1 tone Pig Iron + 0.5 tone Slag

The specific activity of ^{238}U, ^{232}Th, ^{226}Ra and ^{40}K in the raw materials are analyzed in laboratory from site-sampling [10, 11]. Further, daughters nuclides from decay chains are calculated based on; 1) isotopes of uranium are estimated from their natural abundance [12] and (2) assuming secular equilibrium between parents and daughter(1)(13). According to these assumptions, twelve radionuclides are studied; ^{238}U, ^{235}U, ^{234}U, ^{232}Th, ^{230}Th, ^{228}Th, ^{226}Ra, ^{227}Ac, ^{210}Pb, ^{231}Pa, and ^{40}K; the specific activities for these radionuclides are presented in Table 1.

TABLE 1. SPECIFIC ACTIVITY OF TE-NORM IN RAW AND SLAG MATERIALS

Radionuclide	Half-life (y)	Specific activity of NORM in Coke Bq/kg	Specific activity of NORM in iron ore Bq/kg	Specific activity of NORM in limestone Bq/kg	Specific activity of TE-NORM in Slag (source term) Bq/kg
U-238*	4.50E+09	1.01E+01	9.12E+01	2.50E+01	9.75E+01
U-235	7.00E+08	1.15E-01	1.04E+01	2.86E+00	1.11E+01
U-234	2.50E+05	1.01E+01	9.12E+01	2.50E+01	9.75E+01
Th-232*	1.40E+10	7.50E+00	9.90E+00	2.60E+00	1.06E+01
Th-230	7.50E+04	1.01E+01	9.12E+01	2.50E+01	9.75E+01
Th-228	1.90E+00	7.50E+00	9.90E+00	2.60E+00	1.06E+01
Ra-228	5.80E+00	7.50E+00	9.90E+00	2.60E+00	1.06E+01
Ra-226*	1.60E+03	1.01E+01	9.12E+01	2.50E+01	5.32E+02
Ac-227	2.20E+01	1.15E+00	1.04E+01	2.86E+00	1.11E+01
Pb-210	1.90E+01	1.01E+01	9.12E+01	2.50E+01	9.75E+01
Pa-231	3.30E+04	1.15E-01	1.04E+01	2.86E+00	1.11E+01
K-40*	1.30E+09	3.18E+01	8.40E+01	1.90E+01	8.88E+01

* Wafaa Fawzy, 1999, and N. M. Ibrahim et al, 1999

2.3 Exposure scenarios

Based on reasonable generic data [14, 15]. Two normal evolution scenarios are developed to evaluate the radiological impact resulted from TENORM slag piles. The first one is "worker scenario", and the second is "individual scenario". Radiological assessment in both scenarios is evaluated based on of inhalation, submersion and external doses and compared with the standard dose limit of 1 mSv/y [16]. Due to the insufficient data required, ingestion dose is not considered for public scenario. Additionally, the external radiation dose is not considered in worker scenario due to the limited time of worker exposure. For both scenarios, it is assumed that radionuclides are uniform distributed in the disposal piles and continuously released in the dust at the disposal area by resuspension by the effect of wind [9]. Pasquill stability category D is assumed for atmosphere condition (neutral conditions) [17]; this is a conservative assumption because Egypt is subjected to seasonable high windy weather. Also, it is assumed that the radinonuclides in the dust plume are not depleted by the natural decay process during all the calculations of dose assessment for the workers and members of public in 1 year.

2.4 Worker Scenario

At the area of waste dumping, wind erosion and the process of mechanical dumping of waste are the main mechanism of dust generation. The dust loaded radionuclides is floated in the dumping area and worker has inhalant the very fine particles, and submersed in this dust particles in different occasions. The total dose "Dose" in Sv/y received by a worker spends one hour daily at the area of waste disposal. The total dose is the sum of inhalation and submersion doses and is calculated by:

$$Dose = [C_{dust} \cdot t_{out,\,dust} \cdot b_{r,\,dust} \cdot DF_{inh} \cdot Occ_{dust}]_{inh} + [C_{dust} \cdot t_{out,\,dust} \cdot DF_{sub} \cdot Occ_{dust}]_{sub} \quad (1)$$

$$\text{Hence, } Dose = C_{dust} \cdot t_{out,\,dust} \cdot Occ_{dust} \cdot (b_{r,\,dust} \cdot DF_{inh} + DF_{sub})$$

Where: C_{dust} is the air concentration of the radionuclide in the dust (Bq/m^3), $t_{out,\,dust}$ is the time spent exposed to the dust (h) Occ_{dust} is the number of dust releases per year (y^{-1}), $b_{r,\,dust}$ is the breathing rate of worker (m^3/h), DF_{inh} is the dose factor for inhalation (Sv/Bq), DF_{sub} is the dose factor for external irradiation from submersion in the dust ($Sv.h^{-1}/Bq.m^{-3}$)

$$C_{dust} = A_m \cdot Dust \quad (2)$$

Where: A_m is the specific activity of the radionuclide in the waste (Bq/kg), dust is the dust level in the air breathed by the workers (kg/m^3).

As shown in Table 3, the inhalation dose received by a worker is three times higher than the submersion dose, which it is expected because of the interior higher impact of radionuclide inside the respiratory system of human. ^{230}Th, ^{226}Ra, ^{40}K are the most contributed radionuclides in the inhalation dose. Nevertheless, the total dose of both the submersion and the inhalation for worker of 4.8E-04 mSv/y is lower than the dose limit of 1 mSv/y.

TABLE 2. GENERIC DATA CONSIDERED (IAEA, 2000)

Parameter	Value
Duration of dust release	1 h
Time exposed to dust	1 h
Breathing rate of worker	1.2 m^3/h
The dust level in the breathed by the worker	1.10E-03 kg/m^3
Number of dust releases per year	1 y^{-1}

TABLE 3. EXPOSURE DOSES RECEIVED BY THE WORKER

Radionuclide	C_{dust} (Bq/m^3)	Dose Factor of Submersion Sv.m^3/h.Bq	Submersion Dose (Sv/y)	Dose Factor of Inhalation Sv/Bq	Inhalation Dose Sv/y	Total Dose Sv/y
U-238	9.75E-04	4.90E-12	3.58E-10	8.00E-06	9.36E-09	9.71E-09
U-235	1.11E-04	2.80E-11	2.34E-10	8.50E-06	1.14E-09	1.37E-09
U-234	9.75E-04	2.70E-14	1.97E-12	9.40E-06	1.10E-08	1.10E-08
Th-232	1.06E-04	3.10E-14	2.45E-13	1.10E-04	1.39E-08	1.39E-08
Th-230	9.75E-04	6.30E-14	4.60E-12	1.00E-04	1.17E-07	1.17E-07
Th-228	1.06E-04	2.90E-10	2.29E-09	4.40E-05	5.57E-09	7.87E-09
Ra-228	1.06E-04	1.70E-10	1.35E-09	1.60E-05	2.03E-09	3.37E-09
Ra-226	5.32E-03	3.20E-10	1.28E-07	9.50E-06	6.07E-08	1.88E-07
Ac-227	1.11E-04	6.70E-11	5.60E-10	5.70E-04	7.62E-08	7.67E-08
Pb-210	9.75E-04	3.20E-13	2.34E-11	5.70E-06	6.67E-09	6.69E-09
Pa-231	1.11E-04	6.20E-12	5.18E-11	1.40E-04	1.87E-08	1.88E-08
K-40	9.75E-04	1.70E-10	1.13E-08	1.60E-05	1.70E-08	2.84E-08
			1.44E-07		3.39E-07	4.83E-07

This dose is obtained as a result of the limited time spent by the worker in front of the dumping area. However, this dose can be higher with certain magnitude according to the worker role in the area, place of working, and the time spent per year near the slug waste. In the present work, the 4.8E-04 mSv/y reflects the dose received only from the exposure of a worker spent very limited time at the disposal area.

2.5 Individual Scenario

The dust plume, by the action of wind speed, was transferred to atmosphere. It is assumed the presence of resident individual living at 100 m far from the area of slag waste disposal.

2.5.1 Distribution of the source term in the environment

During the formation of dust plume, radionuclides in the slag waste (source term) are not totally released in this plume. A fraction of 0.0001 from the specific activity of each radionuclide was assumed to be released in the dust. and the release rate of a radionuclide in dust, R_{dust} (Bq/h) is calculated by:

$$R_{dust} = f_{rel,\ dust} \cdot V_{dust} \cdot A_m \cdot \rho_{bd} / t_{dust} \quad (3)$$

Where: $f_{rel,\ dust}$ is the release fraction for the radionuclide, V_{dust} is the volume of the waste from which the dust is released [m^3] and A_m is the specific activity of the radionuclides in the waste [Bq/kg], ρ_{bd} is the bulk density of the waste [$kg.m^3$], t_{dust} is the duration of the dust release.

The movement of soil particles in the direction of prevailing wind causes the horizontal transport of these particles. Once the dust particles from the ground surface, large particles fall back to the ground. Smaller particles tend to remain suspended in the air and easily disperse away by turbulent motion in the atmosphere. The air concentration of radionuclide at ground level $C_{air,\ dust}$ [Bq/m^3] at a known distance is given by:

$$C_{air,\ dust} = R_{dust} \cdot C_{integ.dust} \quad (4)$$

Where: $C_{integ.dust}$ is the time integrated air concentration at ground level at the given distance in [$Bq.h/(m^3.Bq)$]

The surface concentration of a radionuclide $C_{surf,\ dust}$ (Sv/y) from one release from deposited activity both during and after the passing of the plume is equal to:

$$C_{surf,\ dust} = C_{air,\ dust} \cdot t_{dep,dust} \cdot (V_{g,\ dust} + W_{out,\ dust} \cdot h_{dust}) \quad (5)$$

Where: $t_{dep,dust}$ is the time over which deposition occurs [s], $V_{g,\ dust}$ is the dry deposition velocity [m/s], $W_{out,\ dust}$ is the washout coefficient [s], h_{dust} is the plume height [m].

The calculation of radionuclides distribution and the results obtained are given in Table 4.

TABLE 4. GENERIC DATA CONSIDERED (IAEA, 2000)

Parameter	Value
Exposure time during the passage of dust plume	1 h
Breathing rate of any member	1 m^3/h
Time integrated air concentration at ground Level under Pasquill stability catego Value given from 100 m from groun release	3.24 Bq.h/(m^3
Indoor shielding factor	0.1
Time spent indoors	6575 h/y
Time spent outdoors	2192 h/y
time over which deposition occurs	3600 s
dry deposition velocity	0.002 m/s
washout coefficient	3.00E-04
plume height	10 m
Indoor shielding factor	0.1

TABLE 5. DISTRIBUTION OF TE-NORM SOURCE TERM

Radionuclide	Release rate of radionuclide in dust (R_{dust}) Bq/h	Air concentration of radionuclide at ground level ($C_{air,dust}$) (Bq/m^3)	Concentration of radionuclide in dust from wet and dry deposition ($C_{surf,\ dust}$) (Bq/m^3)
U-238	9.75E-01	3.16E+00	5.68E+01
U-235	1.11E-01	3.61E-01	6.50E+00
U-234	9.75E-01	3.16E+00	5.68E+01
Th-232	1.06E-01	3.42E-01	6.15E+00
Th-230	9.75E-01	3.16E+00	5.68E+01
Th-228	1.06E-01	3.42E-01	6.15E+00
Ra-228	1.06E-01	3.42E-01	6.15E+00
Ra-226	5.32E+00	1.72E+01	3.10E+02
Ac-227	1.11E-01	3.61E-01	6.50E+00
Pb-210	9.75E-01	3.16E+00	5.68E+01
Pa-231	1.11E-01	3.61E-01	6.50E+00
K-40	8.88E-01	2.88E+00	5.18E+01

The total dose "Dose" received by person living around the area resulted from dust plume are expressed as (Sv/y):

$$\text{If} \quad Dose = Dose_{sub} + Dose_{inh} + Dose_{ext} + Dose_{ing} \qquad (6)$$

Where, $Dose_{sub}$ is the dose due to the external irradiation from submersion in the dust plume [Sv/y]; $Dose_{inh}$ is the dose from inhalation of the dust of radionuclides in the plume [Sv/y], $Dose_{ext}$ is the dose from external exposure from deposited activity both during and after the passing of the plume [Sv/y], $Dose_{ing}$ is the dose from the ingestion of activity deposited on leafy green vegetables [Sv/y].

The dose from the ingestion of activity required a number and variety of specific data. Therefore, it will be neglected in this study and the total dose will be equal to:

$$Dose = Dose_{sub} + Dose_{inh} + Dose_{ext}$$

$$Dose = [C_{air,\ dust} \cdot t_{out,\ dust} \cdot b_{r,\ dust} \cdot DF_{inh} \cdot Occ_{dust}]_{inh} + [C_{dust} \cdot t_{out,\ dust} \cdot DF_{sub} \cdot Occ_{dust}]_{sub} + [C_{surf,\ dust} \cdot ((1- e^{-\lambda t})/\lambda t) \cdot (sf. t_{in} + t_{out}) \cdot DFext_{,\ surf}]_{ext}$$

Hence,

$$Dose = t_{out,\ dust} \cdot Occ_{dust} (C_{air,\ dust} \cdot b_{r,\ dust} \cdot DF_{inh} + C_{dust} \cdot DF_{sub}) + [C_{surf,\ dust} \cdot ((1- e^{-\lambda t})/\lambda t) \cdot (sf. t_{in} + t_{out}) \cdot DF_{ext,}]_{ext}$$

Where, $t_{out,\ dust}$ is the time spent outside during the passage of the dust plume (h), Occ_{dust} is the number of dust releases per year (y^{-1}), $C_{air,\ dust}$ is the air concentration of the radionuclide at ground level (Bq/m^3), $b_{r,\ dust}$ is the breathing rate of member of the public (m^3/h), $C_{surf,\ dust}$ is the surface concentration of a radionuclide from one release (Bq/m^2), λ is the radionuclide decay constant (y^{-1}) , t is the exposure duration (y), sf is the indoor shielding factor (dimensionless), t_{in} is the time spent indoors (h/y), t_{out} is the time spent outdoors (h/y), DF_{inh} is the dose factor for inhalation (Sv/Bq), DF_{sub} is the

dose factor from submersion in the plume ($Sv.h^{-1}/Bq.m^{-3}$) and DF_{ext} is the external exposure dose factor ($Sv.h^{-1}/Bq.m^{-2}$).

The total dose calculated for resident individual is 1.1 mSv/y, which is crossed the standard limit as a result mainly from the inhalation dose. This dose is expected and can be explained by the short distance between the residence individual home and the disposal piles in this area in the studied area. Additionally, the exposure dose calculated for the individual and for the worker shows a great distinction which are also expected and appeared in cases of different slag waste from various industries [15].Consequently, the waste should be isolated within simple economical design. Different options are available in order to decrease the impact of TENORM waste piles on public living in the area; fixation and covering the piles in situ or burial the slag in simple underground structure design as trench. Several studies should be performed before reaching a final decision for waste isolation options. Concerning the shallow disposal of slag waste in trench, geosciences investigations; geological & geophysical are normally carried out prior to construct disposal site and to identify the engineered characteristics of the soil foundation [18, 19, 20].

TABLE 6. TOTAL DOSE CALCULATED AND RECEIVED BY INDIVIDUAL

Radionuclide	Decay constant (y^{-1})	Dose Factor for external exposure ($Sv.h^{-1}/Bq.m^{-2}$)	Submersion Dose Sv/y	Inhalation Dose Sv/y	External dose Sv/y	Total Dose Sv/y
U-238	1.55E-10	1.10E-13	1.55E-11	3.03E-05	1.78E-08	3.03E-05
U-235	9.85E-10	6.00E-13	1.01E-11	3.68E-06	1.11E-08	3.69E-06
U-234	2.83E-06	2.70E-15	8.52E-14	3.56E-05	4.37E-10	3.56E-05
Th-232	4.95E-11	2.00E-15	1.06E-14	4.51E-05	3.51E-11	4.51E-05
Th-230	9.00E-06	2.70E-15	1.99E-13	3.79E-04	4.37E-10	3.79E-04
Th-228	3.63E-01	5.10E-12	9.91E-11	1.80E-05	7.50E-08	1.81E-05
Ra-228	1.21E-01	3.30E-12	5.81E-11	6.56E-06	5.45E-08	6.62E-06
Ra-226	4.33E-04	6.00E-12	5.52E-09	1.97E-04	5.31E-06	2.02E-04
Ac-227	3.18E-02	1.40E-12	2.42E-11	2.47E-04	2.55E-08	2.47E-04
Pb-210	3.11E-02	1.30E-14	1.01E-12	2.16E-05	2.07E-09	2.16E-05
Pa-231	2.11E-05	1.50E-13	2.24E-12	6.06E-05	2.78E-09	6.06E-05
K-40	5.33E-10	1.00E-13	4.89E-10	5.52E-05	1.47E-08	5.52E-05
			6.22E-09	**1.10E-03**	**5.51E-06**	**1.10E-03**

2.6 Preliminary Safety Assessment of the Shallow Waste Disposal

For the disposal of the slag in a trench covered with layer of soil, the gradual migration of radionuclides from the slag to groundwater is the main concern pathway scenario in which individual exposure could occur. People can be exposed to these radionuclides via different other paths; however, the ingestion of drinking water is considered one from the most important path. In the present study, the drinking water scenario was assessed through generic study. This scenario analyzes the impact of trench disposal (trench design). The natural release fraction of TE-NORM waste from the trench disposal to a well at 100 m far from the steel factory is calculated and dose from drinking water received by an individual [21]. The calculations are determined according to the following equations:

2.6.1. Release fraction of radionuclide

The release fraction R in Bq/y that flows in the ground water from distance x=0 and time t=0 to x= 100 m and t = 1 y and expressed in y^{-1}. R is calculated as follows:

$$F_i = A \exp(Vx/2D - Ct - A^2/4t)/2(\pi t^3)^{1/2} \quad (7)$$

$$A = X (R_i/D)^{1/2}$$

Where: R_i is the retardation factor of a radionuclide (dimensionless) and equal $1+\rho K_d/\theta$, where: ρ is the density of the soil (kg/m^3), K_d is the distribution coefficient of radionuclide within the soil (m^3/kg), θ is the moisture content (dimensionless) D is dispersion coefficient (m^2/y) and equal αV, where:

α is the longitudinal dispersivity in porous media (m) V is the ground water velocity (m/y)

$$C= \lambda_i + V^2/(4DK_i)$$

λ_i is the radioactive decay rate of a radionuclide (y^{-1}) and equal $\ln(2)/t_{1/2}$, where: ln(2) is the natural logarithm of 2, $t_{1/2}$ is the half-life of radionuclide.

2.6.2. Dose received from drinking water

The ingestion dose from drinking water D_r (Sv/y) received by a member of the public used the well for 1 year (Matthew Kozak, private communication).

$$D_r = C_r \cdot D_{ing} \cdot A_f \cdot e^{-\lambda t}/Vol \quad (8)$$

Where: C_r is the consumption rate of drinking water per year (m^3/y), D_{ing} is the dose conversion factor of the radionuclide (Sv/y), A_F is the inventory of radionuclide release, Vol is the volume of waste trench.

1. RESULTS AND DISCUSSIONS

The selection of the northeast side of the factory is considered as a perfect recommended area for constructing a trench disposal. Additionally, the presence of fault displaces the dry limestone layers beside the clay formation which prevents the upward water movement. The primary route for exposure to individual is direct ingestion of ground water used as drinking water because ground water contamination has the potential to impact the greatest number of individuals. Therefore, the presence of a domestic well at 100 m far from the area was assumed for the study. The annual release fraction of each radionuclide from the trench disposal was calculated and presented in Table 10.

In the body, ^{40}K poses a health hazard from both the beta particles and gamma rays. The health hazard of ^{40}K is associated with cell damage caused by the ionizing radiation that emitted from the radioactive decay, with a general potential for subsequent cancer induction .The total ingestion dose from drinking water, of 6.17E-04 mSv/y, represent lower value than the limit. The results show the possibility of decreasing the impact of TENORM waste by the construction of shallow disposal. The idea is to prevent the dispersion of slag dust to the environment.

TABLE 10. RELEASED FRACTION OF RADIONUCLIDES

Radionuclide	Half-Life (y)	Kd* m^3/kg	Fraction
U-238	4.50E+09	3.50E-02	6.30E-04
U-235	7.00E+08	3.50E-02	6.30E-04
U-234	2.50E+05	3.50E-02	6.30E-04
Th-232	1.40E+10	3.20E+00	0.00E+00
Th-230	7.50E+04	3.20E+00	0.00E+00
Th-228	1.90E+00	3.20E+00	0.00E+00
Ra-228	5.80E+00	5.00E-01	3.04E-75
Ra-226	1.60E+03	5.00E-01	3.43E-75
Ac-227	2.20E+01	4.50E-01	1.75E-67
Pb-210	1.90E+01	2.70E-01	1.01E-39
Pa-231	3.30E+04	5.50E-01	6.47E-83
K-40	1.30E+09	1.50E-02	2.71E-01

* coefficient K_d are given in Goodwin, B. W., et al,1994

1. CONCLUSIONS

The objective of this study is to evaluate the impact of TENORM waste accumulated from steel factory and disposed in open area. Additionally, the study assessed an alternative option of shallow disposal which prevent the exposure of individuals to TENORM dust and provided primary site selected for this shallow design. From the detailed study, it was concluded that:

(i) The disposal of TENORM waste in both cases of piles or shallow disposal does not reflect any hazard for the workers who spent limited time at the disposal area. That has resulted from the low exposure time of workers outside the factory.
(ii) The disposal in piles is a cause of concern to the health of public living around the site
(iii) ^{40}K, ^{226}Ra, and ^{230}Th are the radionuclides of concern to the health of public.
(iv) The impact of TENORM buried in a well-designed trench is safe from health considerations of workers and the members of the public and,
(v) Finally, it is concluded from the above study that the area chosen (north-east side of the factory) to construct a disposal site is very appropriate.

REFERENCES

[1] ENVIRONMENTAL PROTECTION AGENCY (EPA), Washington, D. C., (1999).
[2] UNIVERSITY OF TORONTO, Laboratory Hazardous Waste Management and Disposal Manual, Section 5, (2002).
[3] ADEL EL DANAF, Case Studies from West Asia, the Mediterranean, and the Arab Region (2000)
[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Technical Report Series No. 419, IAEA, Vienna (2003).
[5] TITAN AMERICA, Material Safety Data Sheet for Iron Blast Furnace Slag, Ground, August, (2001).
[6] LAKE MACQUARIE CITY COUNCIL, Fact Sheet, Lake Macquarie City Council web page, (2004).
[7] PAUL LINSALATA, Radat.. Phys. Vol. 34, No 2 (1989) 241-250
[8] SAMUEL M. ALLEN, S.M., Online Encyclopedia (2004).
[9] CROCKETT G.M ,SMITH K.R ,OATWAY,W.B.AND MOBBS,S.F. , part 2 the Steel Production Industry (2003)
[10 WAFAA FAWZY EL-SAYED AHMED BAKR, Thesis for the Master Degree, Helwan University, For Faculty of Science, Chemistry Department, Cairo, Egypt, (1999).
[11] IBRAHIEM, N. M. NADA A.,ABD EL MAKSOUD T. M, EL EZABY B., Applied Radiation Isotopes (1999).
[12] NATIONAL NUCLEAR DATA CENTER, Brookhaven National Laboratory (2004).
[13] PRINCE J.R., J. Nuc. Med., Vol/Issue 20:2, 1 February (1979).
[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Division of Radiation and Waste Safety, IAEA, Vienna (2000).
[15] HOFMAN J., LEICHT R., WIINGENDER H. J., WORNER J., Report EUR 19264, August 2000.
[16] INTERNATIONAL COMMISSION OF RADIOLOGICAL PROTECTION, ICRP Publication 60, (1991).
[17] GHULAM RASUL ATHAR, IAEA Workshop, Trieste, Italy, May 12-23, (2003).
[18] MCGINNIS, L.D., AND KEMPTON, J.P, Illininois State Geol. Surv., Circular 323, 23 P (1961).
[19] STOETZNER, U., FREIB.FORSCH., HEFT C,128, Leipzig, (1965) 6- 85.
[20] DUTTA, N.P., Geophys, Prosp., 32 (1984) 1103-1110.
[21] GOODWIN B. W., KISHOR K. MEHTA, Internal Report, AECL-10901, COG-93-265, Atomic Energy Authority of Canada Limited (1994).

OCCUPATIONAL RADIATION PROTECTION ASPECTS IN A MONAZITE BASED RARE EARTH PRODUCTION FACILITY

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Abstract

One of the largest reserves of monazite in the world is present in the Indian subcontinent. Monazite ore has around 8-9% thorium oxide and nearly 60% Rare earth oxides. Selective acid extraction is used to separate the composite rare earths. The main radiological hazard arises from the presence of thorium and it daughter products. The occupational radiation protection aspects for such a facility is different from uranium mining and milling plants due to the presence of thoron and high energy gamma radiation from ^{208}Tl of thorium series. Radiological aspect for this extraction of rare earths was studied. The general radiation field in the rare earth production plant was 0.1-10 $\mu Gy \cdot h^{-1}$ and the average short-lived air activity was 40 ± 9 mWL. Studies were also done to estimate the residual radioactivity in the separated rare earth compounds using gamma spectrometry. The results show presence of ^{227}Ac arising due to the protactinium fraction in the thorium concentrate. The occupational radiation exposure by the Rare Earths production plant is only 6% of the total institutional dose, and the average individual dose is 1.6 mSv per year.

1. INTRODUCTION

Monazite is an orthophosphate of thorium and rare earths and contains about 8-9% of Th as ThO_2 and 60% Rare Earth oxides. The mineral also contains about 0.35% of uranium as U_3O_8. M/s Indian Rare Earths Limited, an undertaking of the Department of Atomic Energy, at Udyogamandal in Kerala was engaged in the separation of rare earths from monazite by chemical processing for almost 50 years. The separation of rare earths from Thorium concentrate separated from Monazite is now being done at this facility. Radiological hazards in the operating plants are mainly due to thorium chain radionuclides [1]. Rare earths carbonate and cerium nitrate are the main products that are produced in this facility. The process involves retrieval of thorium concentrate from silo followed by acid dissolution. The chloride solution which contains Thorium, Uranium and rare earths is pumped to a solvent extraction plant where Uranium is separated. Further the Thorium is separated and converted to thorium oxalate and the Rare earth chloride solution is separated by a series of solvent extraction processes after deactivation. In the Rare Earths production plant (REP), $RECl_3$ is converted to RE Carbonate. Cerium nitrate is also produced in this plant. The block diagram of the operations in the facility is given in FIG. 9. The paper gives the occupational radiation data associated with the production of Rare Earth compounds.

2. MATERIALS AND METHODS

The measurements of different radiological parameters was done and the data studied to optimise the radiation exposure to the occupational workers. The external gamma exposure rate at different locations in REP were carried using different radiaton monitoring systems. The continuous logging of data was done to study the variations with different plant conditions. The data was compared with data generated over the the years to analyse for any trends due to operations for a long period of time.

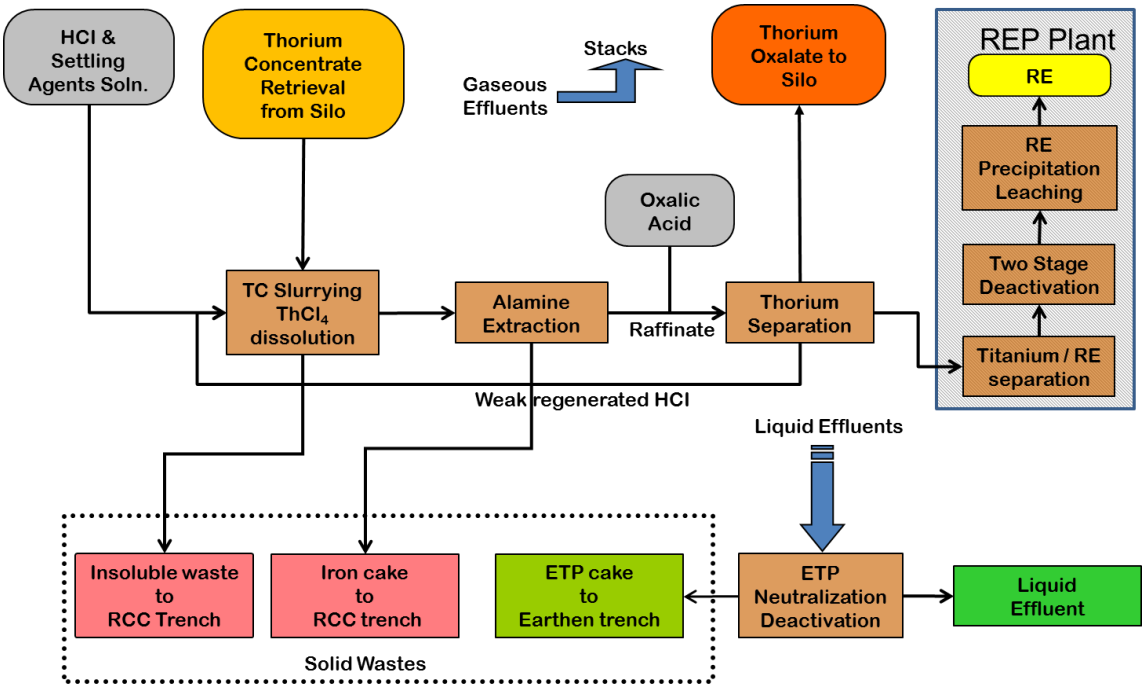

FIG. 9. The block diagram of the Rare Earths production facility

The range and mean values for the different locations of the plant was studied. Air samples were collected from different locations of the plant using vacuum pump with flow rate of 70 lpm. The samples were collected using GFA filter paper of area 5 cm^2 and analysed for ^{232}Th and Thoron daughter activity using a ZnS(Ag) based gross alpha counting system. The samples were collected at a height of around 1.5 m from ground level to make it representative of breathing zone. The results are expressed in Working Level (WL) or mWL (10^{-3}·WL) which is the potential alpha energy concentration (PAEC) or potential alpha energy per unit volume of air from thoron progeny equal to 2.1 x 10^{-5} J m^{-3} and corresponding to 275 Bq m^{-3} equilibrium equivalent concentration of thoron. The counting was done after pre-determined delays to estimate the PAEC. A windows based software application was developed to calculate the working level, generate the dose data and also store the data along with the measurement parameters. A database in the software stores the data for summarizing the data over different time intervals and estimation of internal dose to workers by taking into account time spent in the plant, mean air activity and breathing rate on a quarterly basis. The work occupancy for an year was taken as 2000 h for occupational dose estimation and for thoron progeny, the inhalation dose was estimated by using the dose conversion factor 1.67 mSv per working level month [2]. Floor contamination measurements were carried out using a ZnS(Ag) based gross alpha contamination monitor with effective area of detector around 100 cm^2. The typical values for different locations in the plant were estimated and the range

and average values for the last two years was taken to study the contamination levels in the plant.

Samples of rare earth products were also analysed by gamma spectrometry system [HPGe and NaI(Tl)] to study the residual activity and build up in semi purified samples. The high resolution spectrometer was used to generate secondary standards and use it for routine analysis using NaI(Tl) gamma spectrometry systems that are available in the facility. The analysis of low levels of residual activity is used to devise suitable methods for deactivation for the production of high purity rare earths.

3. RESULTS AND DISCUSSION

The radiation fields at different locations of the plant was studied for more than five years of operation and is summerised in TABLE 3. The general background is in the range 0.1 – 10 $\mu Gy \cdot h^{-1}$. A maximum radiation field of 125 $\mu Gy \cdot h^{-1}$ was observed at the RE Chloride feed tank mainly due to sludge accumulation and was reduced by periodic deactivation by descaling and sludge removal from the tank. Generally there is no occupancy near these storage tanks and the consequence to the institutional dose is mainly due to maintenance of the tanks. The average general background in the worker-occupied areas was 0.4 – 2 $\mu Gy \cdot h^{-1}$. The average external dose for an occupancy period of 2000 h in an year is estimated to be 1.6 mSv.

TABLE 3. RADIATION LEVELS AT DIFFERENT LOCATIONS IN THE PLANT

Location	Radiation field-(μGyh^{-1})
General	0.1 – 10
Working areas	0.4 – 2
RE Chloride tanks	0.3 – 15
Feed tanks	2 – 125

The typical floor contamination levels in the Rare Earths production plant is given in TABLE 4. The contamination levels ranged from 0.2 - 1.8 $Bq \cdot cm^{-2}$. Nearly 50 % of the contamination is transferable.

TABLE 4. TYPICAL FLOOR CONTAMINATION LEVELS

Location	Alpha Contamination ($Bq \cdot cm^{-2}$)
Ground Floor	0.43 - 1.78
First Floor	0.22 - 0.90
Top Floor	0.25 - 0.95

The PAEC due to thoron progeny and airborne long-lived alpha activity due to ^{232}Th mainly contribute to the internal exposure in the plant. Measurements were carried out at different locations for the period 2009–2013 and the average and range are shown in TABLE 5.

TABLE 5. TYPICAL AIR ACTIVITY LEVELS IN THE PLANT

Location	Thoron daughter (mWL)		^{232}Th ($Bq \cdot m^{-3}$)	
	Range	Mean	Range	Mean
Ground Floor	1 - 97	27	0.001 - 0.038	0.004
First Floor	5 - 194	45	0.001 - 0.034	0.004
Top Floor	1 - 120	42	0.001 - 0.052	0.005
Deactivation Tanks	5 - 376	46	0.001 - 0.066	0.008

The annual average PAECs were in the range 27–46 mWL, whereas individual samples showed PAEC ranging from 1 to 376 mWL. The plant average for the 5-year period is 40 mWL for a total of 907 samples . The derived air concentration limit for the occupational settings applicable to the plant is 1000 mWL and the current levels are only 4 % of the limit. The estimated annual inhalation dose to occupational workers due the intake of thoron progeny is 0.8 mSv, assuming an occupancy time of 2000 h in a year. Airborne ^{232}Th activity at various locations in the plant ranged between 0.001 and 0.066 Bq·m^{-3}. The mean activity during the period was 0.005 Bq·m^{-3}. Assuming a breathing rate of 1.2 m^3 h^{-1} for 2000 working hours in a year, the likely inhalation dose works out to 0.625 mSv. The yearly total inhalation dose due to the intake of thoron progeny and long-lived alpha activity is estimated to be 1.425 mSv for occupational workers. The likely total annual dose to occupational worker is estimated to be 3.025 mSv, including the external dose component of 1.6 mSv. A study of the dose apportionment of different plants to the total dose was done based on internal and external dose data of occupational workers. The contribution from REP to the the facility was found to be 6 %.

4. CONCLUSION

The general radiation field in the rare earth production plant was 0.1 - 10 μGy·h^{-1}. The average general background in the worker-occupied areas was 0.8 μGy·h^{-1}. The external dose for an occupancy period of 2000 h in a year is estimated to be 1.6 mSv. The average short-lived air activity due to thoron progeny was 40 ± 9 mWL and that for long lived ^{232}Th was 0.005 Bq·m^{-3}. The internal dose estimated using these values is 1.425 mSv and the avergage individual dose for REP is 3.025 mSv for 2000 hrs occupancy time. The actual measurement of total dose for occupational radiation workers of REP was found to be only 1.6 mSv with the internal dose contribution of 26%. This is mainly due to the actual occupancy time being only 50% of the estimated time. The floor contamination levels are well within acceptable levels, but significant reduction has to be achieved for high purity RE production. High resolution gamma spectrometric analysis showed presence of ^{231}Pa (^{235}U series) [3] and its daughter products, particularly ^{227}Ac in the $RECl_3$ samples. The levels are below the internationally acceptable value of 1000 Bq·kg^{-1}. A typical spectrum is shown in FIG. 10 and typical activity of RE chloride sample.

TABLE 6. ACTIVITY IN THE RARE EARTH CHLORIDE SAMPLES

Radionuclide	Activity ($Bq \cdot kg^{-1}$)
^{227}Ac	950 ± 9
^{28}Th	118 ± 3
^{228}Ra	96 ± 3
^{138}La	63 ± 2
^{226}Ra	36 ± 5

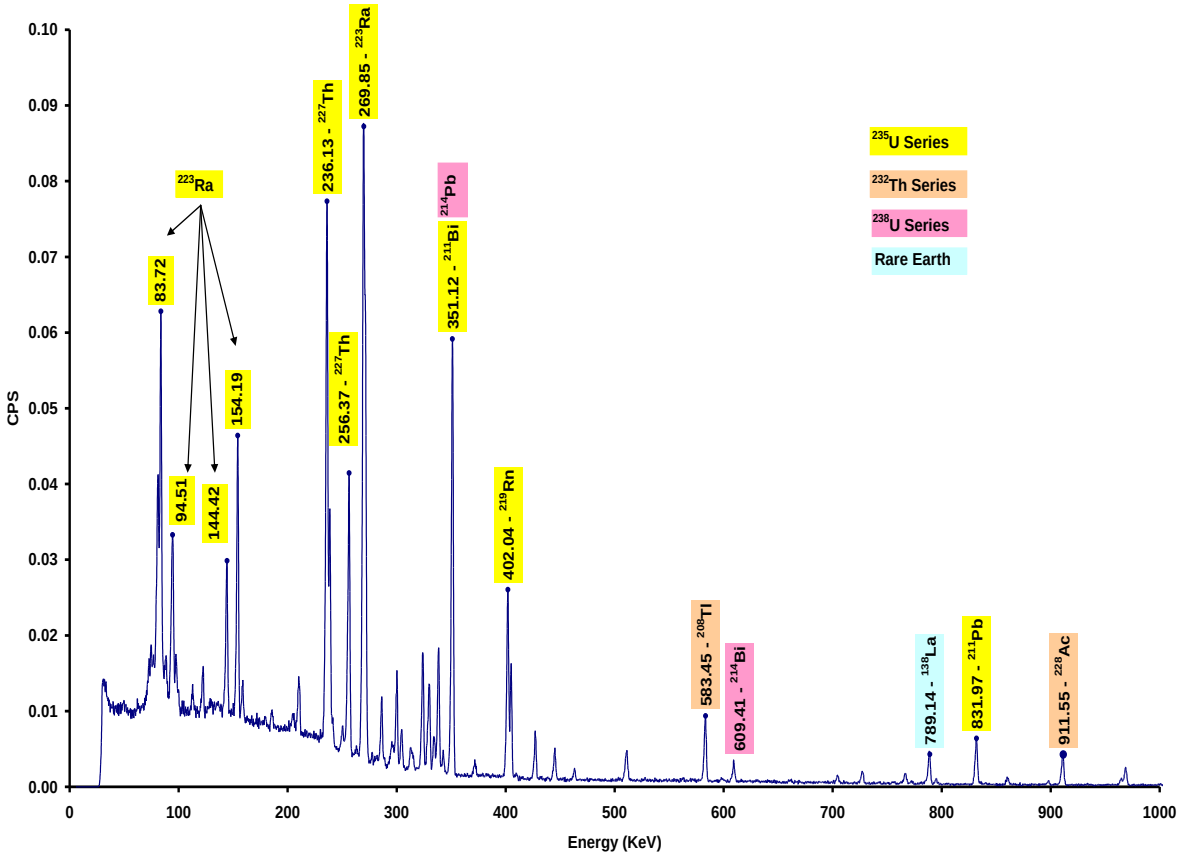

FIG. 10. Gamma Ray Spectrum of Rare Earth Chloride Sample

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REFERENCES

[1] HARIDASAN P. P. et.al., Occupational radiation exposure due to NORM in rare earth compounds production facility, Radiation Protection Dosimetry (2008) 1-5.
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection against radon in work places other than mines, IAEA Safety Report No.33, Vienna (2003).
[3] R. PARTHASARTHY et.al., Determination of ^{231}Pa in monazite and other streams of thorium production cycle, Journal of Radioanalytical and Nuclear Chemistry, Vol. 246, No. 1 (2000) 239-242.

BRIDGING THE GAP BETWEEN NORM CHARACTERIZATION OF ORES IN DIFFERENT INDUSTRIES AND ITS IMPACT ON PUBLIC HEALTH

THE HATCH APPROACH

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Abstract

Radiation protection in the mining and metallurgical industries involving Naturally Occurring Radioactive Material (NORM) feed need to be strengthened; better identification of their activities and output materials causing the radiation exposure. The use of NORM raw material presents a considerable challenge to promoters of new projects due to uncertainty of the radiation risks, potential health effects in the workplace and on the public, possible eco-toxicity as a result of their activities and public perception of these possible risks. Although the NORM regulations, guidelines and characterization have developed substantially and the awareness regarding their impacts on the public health has been recently increasing, the gap still exists in the NORM prediction behaviour and risks when developing a project that has not been tested or operated elsewhere. Hatch is proposing a chemical/metallurgical approach to radiation exposure estimation based on the physiochemical properties of the radio-elements. The results of these calculations will serve as a first step to identify potential NORM hazards and to develop management plans to be integrated into the project's design and prevent economical liabilities and social unacceptability of the projects. This paper illustrates the Hatch approach for evaluating the radio-elements behaviour through the mining/processing activities handling NORM-containing ores.

1. INTRODUCTION

More and more, engineering teams and industrial-plant promoters face the challenge of identifying and quantifying the risks of radiation exposure in the workplace and radiation dispersion in the environment in early phases of project definition, when the process is not fully developed but intends to use NORM-containing ore as feed. Workers in the metallurgical industry, as well as the general population of the communities living in the neighbourhood of mining and metallurgical facilities, are more sensitive to hazards and risks related to radioactivity [1]. However, promoters as well as the general population often confuse radioactivity related to artificially (nuclear) generated radioactivity, such as power generation, with naturally occurring radioactivity from the earth crust/geology. Divided between the concerns of the promoters to see their projects slowed down or their financial viability jeopardized by the risk mitigation measures and the necessity to rigorously identify the risks or hazards associated with a new project, Hatch process and environmental specialists have been involved in several mining and metallurgical plant projects intending to use raw material containing naturally occurring radioactivity.

Based on its experience to date, Hatch acknowledges the difficulty in quantifying the radio-elements content of products and/or waste streams before their production. At the same time, through reviewing several case studies in the literature, Hatch realizes that the prediction of radioactivity from NORM in process output streams depends on assumptions [2, 3] that are

not necessarily justified or not function of the process operating conditions. As a result, Hatch is tapping into its expertise in metallurgical and chemical fundamental principles to establish the most probable path of the feed-material containing radio-elements in a process, and at the same time, to identify, in the early scoping and pre-feasibility studies, the critical areas for radioactivity exposure to the workers, and to the social and natural environment. This analysis will allow to include, right from the start in the project definition, the control of NORM exposure, its reduction at the source and related mitigation measures, which will render the project socially and environmentally safe while remaining economically viable. The present paper provides an introduction to this approach as developed by Hatch and applied on recent projects, handling NORM-raw feed in various fields of the mining and metallurgical industry.

2. METHOD

The approach used by Hatch relies upon a thorough understanding of the properties of the NORM feed. Radio-elements of the radioactive-chains are analyzed in ore feed samples to quantify their concentration and activity, and to assess the level of equilibrium of the degradation chain in the material. The behaviour of the detected radio-elements, in the proposed mining or metallurgical plant unit operations, is then established based on the physiochemical properties of these elements, including boiling or melting temperature, density and/or solubility [4] and on the planned operating conditions. When laboratory tests work is accessible, the evaluation of the partition factors for the main process steps can be performed. A mass balance around the process or unit operation is then performed for each radio-element with a significant long half-life (more than 24-hours). These calculations allow the determination of the most probable end destination for each of the radio-elements, in the product, by-products or waste streams of the plant. The potential range of radiation emissions from each of the outlet streams can then be estimated and compared to the health and safety exposure limits and guidelines. As a result, exposure mitigation measures and proper personal protection equipment can be identified and included in the project scope; the risks of emissions to the environment of radon, radio-active dust, solid residues and/or effluents are estimated. Hatch has validated this method by applying it to existing facilities processing NORM feed; predicted levels of radioactive elements and their activities in products, waste residues and even off gas streams were compared to real-time measurements in process streams to validate the estimated values.

3. RESULTS

Hatch has been involved in several projects presently being developed throughout the world that intend to use NORM-containing ores such as phosphate rock, Ilmenite ore, coal and Rare Earths ores; these ores exhibit low levels of naturally occurring radioactivity. Most promoters are reluctant to publish these aspects of their foreseen operations due to the general public perception regarding the potential risks or hazards related to NORM. Hatch is therefore not authorized to publish details regarding their plans, projects in development or existing operational performance.

However, Hatch can show a comparison between the results obtained by applying its proposed method regarding the radio-elements mass balance approach and the actual observations made of existing plants in publicly available literature as presented in TABLE 7.

TABLE 7. CHARACTERIZATION OF NORM-FEED MATERIAL USED IN TWO MINING/METALLURGICAL FACILITIES

	Case I [5] Acid Mine Drainage Treatment S. Africa	Case II [6] Pulverized-Coal Power Plant Spain
Raw feed Characteristics	Gold mine water	Bituminous Coal
$^{238}U^{++}$ content	280 µg/L	1.47E-06 ppm
$^{232}Th^{++}$ content	18 µg/L	4.9 ppm
$^{40}K^{++}$ content	--	0.35 ppm

Similar to the typical process mass and energy balances, performed in the engineering industry (using software such as METSIM, LIMS or HSC), to model the process recoveries and to determine the preliminary design basis of these facilities, the radio-nuclides mass balances were calculated to estimate the mass of radio-nuclides that would most probably end up in the products, by-products, effluents, off gas and waste residues streams. The partition factor of each radioelement in a specific unit operation was established based on the physical and chemical/metallurgical properties of this element, such as those presented in TABLE 8 below, and on the operating conditions prevailing in the equipment.

TABLE 8. EXAMPLES OF METALLURGICAL, CHEMICAL OR PHYSICAL PROPERTIES USED TO ESTIMATE THE PARTITION COEFFICIENT IN MAIN PROCESS-UNIT OPERATIONS

Radionuclide	Temperature (°C)		Solubility (in)	
	Melting	Boiling	H_2O	Other solutions
Lead (Pb)	328	1,750	insoluble	HCl, H_2SO_4
Polonium (Po)	254	962	insoluble	HCl, H_2SO_4, HNO_3
Thorium (Th)	1,755	4,788	insoluble	HCl
Uranium (U)	1,133	3,800	insoluble	HCl, HNO_3
Radium (Ra)	699.8	1,737	soluble	insoluble

Through these process simulations, Hatch obtained the following characteristics of the main output streams of the two cases studied. Considering the quantities of the radio-nuclides calculated in the outlet streams, the resulting activity of the outlet steams was estimated based on the activity of each nuclide. The results are presented in TABLE 9 and

TABLE 10, and compared to the actual stream characteristics measured in-situ.

TABLE 9. AMD TREATMENT CASE STUDY COMPARING SIMULATED AND MEASURED TENORM STREAM*

Case I [5] Acid Mine Drainage (AMD) Treatment, S. Africa					
Mass conc. (mg/kg)	Simulated	Measured	Activity (Bq/kg)	Simulated	Measured
Solid Residue (U, Th, Pb)	64.4	66.7	Solid Residue (U, Th, Pb)	0.0534	0.0514

*Assumptions: 1) Assume gypsum in fly ash stays gypsum; the sulfate from AMD becoming ettringite and gypsum
2) Partition coefficients for U, Th and Pb based on laboratory test work results.

TABLE 10. COAL POWER PLANT CASE STUDY COMPARING SIMULATED AND MEASURED TENORM STREAM*

Case II [6] Pulverized-Coal Power Plant, Spain					
Fly Ash – Activity (Bq/kg)	Simulated	Measured	Bottom Ash – Activity (Bq/kg)	Simulated	Measured
^{40}K	359	306	^{40}K	2	235
^{226}Ra	243	191	^{226}Ra	306	149
^{232}Th	90	74	^{232}Th	113	66
^{210}Po	344	257	^{210}Po	60	57
Overall	1,035	828	Overall	480	507

*Assumptions: 1) Complete combustion, at temperature between 1,300 and 1,400 °C
2) Due to their boiling points, majority of K and Po will vaporize and end up in the fly ash

4. DISCUSSION

In the different cases, the simulated results in the identified output streams activities fall within 20 to 25% of the measured values. This difference can easily be attributed to the following uncertainties:

(a) The limits of accuracy of the actual activity measurements;
(b) The difficulties to collect representative samples of certain output streams such as solid metals products or dust contaminated off gas streams;
(c) Differences between actual and expected process conditions which may affect the evaluation of in the partition factors;
(d) The disruption of the radiological secular equilibrium, when certain radio-elements split from the degradation series due to their specific metallurgical, chemical or physical properties.

The results presented above indicate, however, that the proposed Hatch approach can provide good indications on the potential contamination with NORM of the output streams, which would otherwise be considered free from any contamination.

5. CONCLUSIONS

As socially responsible stakeholders, industrial promoters want to be informed early in their project development of the potential risks of exposure to their workers and the plant surrounding environment. A rigorous assessment of the potential risks allows them to include in their project scope the following:

i. Proper process reduction measures, such as efficient dust collection units,
ii. Workplace exposure control systems, such as ventilation of enclosed areas, and,
iii. Workers exposure mitigation protocols, including exposure time limitation, personal dosimeters and specific personal protection equipment.

The proposed Hatch approach allows industrial promoters to demonstrate to the surrounding communities and environmental authorities with the full picture, back-up with scientific facts, of their upcoming operation, which often results in radiation emissions not exceeding the internationally agreed safe-exposure limits.

REFERENCES

[1] GODFREY MADZIVIRE, PEANE P. MALEKA, VISWANATH R.K. VADAPALLI, WILSON M. GITARI, ROBERT LINDSAY AND LESLIE F. PETRIK., Fate of the naturally occuring radioactive paterials during the treatment of acid mine drainage with coal fly ash and aluminium hydroxide. 133, Cape Town, South Africa : Journal of Environmental Management, (2014).

[2] H.R. VON GUNTEN, H. SURBECK AND E. ROSSLER., Uranium series disequilibrium and high thorium and radium enrichments in Karst formations, Environmental Science and Technology, Vol. 30 (1996) 1268-1274.

[3] HORVATH, PHILIPPE DUPORT, Practical aspects of monitoring and dosimetry of long-lived dust in uranium mines and mills- Determination of the Annual Limit on Intake for uranium/thorium ore dust, Radiation Protection Dosimetry, Vol. 26 (1989) 43-48.

[4] A. BEAZA, M. DEL RIO. A. JIMENEZ, C. MIRO AND J. PANIAGUA, Influence of geology and soils particle size on the surface area/volume activity ratio for natural radinuclides, Journal of Radioanalytical and Nuclear Chemistry, Vol. 189 (1995) 289-299.

[5] LINDA, E. R., Naturally occuring radionuclides from industrial sources: characteristics and fate in the environment, Radioactivity in the Environment, Vol. 10 (2007) 211.

[6] JUAN C. MORA, JOSE A. CORBACHO, BEATRIZ ROBLES , ANTONIO BAEZA, DAVID CANCIO AND ANA M SUAIIEZ, Methodology used in the radiological assessment of a coal-fired
power plant, AIP Conference Proceedings American Institute of Physics, (2008), 978-0-7354-0559-2/08.

NATURAL RADIOACTIVITY, RADIOLOGICAL HAZARDS AND ANNUAL EFFECTIVE DOSE ASSESSMENT IN INDIAN FLY ASH SAMPLES

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Abstract

Fly ash, by product of burnt coal contains many radioactive elements such as ^{238}U (uranium), ^{232}Th (thorium) and ^{40}K (potassium), and exposure to the radiations coming out may have deleterious effects on the health of the workers and the residents. Natural radioactivity, radiological hazards and annual effective dose assessment was carried out in fly ash samples collected from different thermal power stations, other fly ash handling facilities and National Council for Cement and Building Materials (NCB) using Gamma spectrometry. The measurements indicated that the hazard indices, the minimum and maximum values of absorbed dose and indoor and outdoor annual effective doses were found to be within the recommended limits.

1. INTRODUCTION

Natural radioactivity in fly ash comes from ^{222}Ra and ^{232}Th series and natural ^{40}K. It has been reported that the combustion of coal in various thermal power plants results in the release of only some natural radioactivity to the environment and the largest part of the coal radioactivity remains with the ashes [1]. When used in building construction materials, land filling and other purposes, it becomes potentially hazardous to humans and workers. Measurement of radioactivity in fly ash samples in terms of activity concentration of ^{222}Ra, ^{232}Th and ^{40}K is due to its health hazards posed to humans and environmental pollution. Radon concentration has been reported higher in a house with fly ash than in a house built without it [1-3]. Thus measurement of radioactivity in fly ash is very important from radiation protection point of view and in the present work we have reported the radioactivity in fly ash samples.

2. MATERIALS AND METHODS

2.1 Sample collection and preparation

In the present investigation, fly ash samples were collected, powdered and shaken in a sieve of 250 micron-mesh size and particles of size ≤ 250 microns were obtained. The samples were dried for 10–15 h at 110°C in an electric oven (hot air oven) to obtain a constant dry weight. Sieved samples were packed and sealed in 300 ml air tight PVC container and kept for about four weeks period to allow radioactive equilibrium among the radon (^{222}Rn), thoron (^{220}Rn), and their short lived progenies. On an average 0.3 kg of ash was taken for each sample.

2.2 Measurement technique

HPGe detector of high-resolution gamma spectrometry system was used to count the activity of samples. The detector was a co-axial n-type high purity germanium detector (Make EG&G, ORTEC, Oak Ridge, USA) having a resolution of 2.0 keV at 1332 keV and relative efficiency of 20%. For calibration of the low background counting system, a secondary standard was obtained, calibrated with the primary standard obtained from the International Atomic Energy Agency (details discussed elsewhere) [4]. The samples were counted for a period of 72000 seconds and the spectra were analyzed of the photo peak of uranium, thorium daughter products and ^{40}K. The net count rate under the most prominent photo peaks of radium and thorium daughter peaks are calculated by subtracting the respective count rate from the background spectrum obtained for the same counting time. Then the activity of the radionuclides is calculated from the background subtracted area prominent gamma ray energies.

3. FORMULAS USED IN MEASUREMENTS

3.1 Measuring Activity concentration of Uranium, Thorium and Potassium

The concentrations of Uranium, Thorium and Potassium were calculated using the following equation:

(1)

Where, CPS - Net count rate per second, B.I. - Branching Intensity, Eff – Efficiency of the detector

3.2 Radium Equivalent Activity

The widely used radiation hazard index Ra_{eq} is called the radium equivalent activity and was calculated using the following equation

$$Ra_{eq} = C_U + 1.43C_{Th} + 0.077C_k \quad (2)$$

Where, C_u, C_{Th} and C_k are the specific activities of ^{238}U, ^{232}Th and ^{40}K in Bq/ kg respectively.

3.3 Assessment of Radiological Hazards

Radiological hazards in terms of external hazard index (Hex), Internal hazard index (H_{in}), Gamma index (I_γ) and Alpha index (I_α) have been calculated using equations 3, 4, 5 and 6 respectively [5-7].

$$H_{ex} = \frac{A_{Ra}}{370} + \frac{A_{Th}}{259} + \frac{A_K}{4810} \quad (3)$$

$$H_{in} = \frac{A_{Ra}}{185} + \frac{A_{Th}}{259} + \frac{A_K}{4810} \quad (4)$$

$$I_\gamma = \frac{A_{Ra}}{300} + \frac{A_{Th}}{200} + \frac{A_K}{3000} \quad (5)$$

$$I_\alpha = \frac{A_{Ra}}{200} \quad (6)$$

To limit the external gamma radiation dose from materials below 1.5 mGy/y, the external hazard index, H_{ex} should obey the following relation $H_{ex} \leq 1$ [8].

3.4 Estimation of Absorbed and Annual Effective Dose

The measured activity concentrations of ^{238}U, ^{232}Th and ^{40}K were converted into doses ($nGyh^{-1}$ per $Bqkg^{-1}$) and the total absorbed gamma dose rate in air at one meter above the ground level was calculated using the following equation [9].

$$D\ (nGyh^{-1}) = (0.462\ C_U + 0.604\ C_{Th} + 0.0417\ C_K) \quad (7)$$

Where, C_U, C_{Th} and C_K are the activity concentrations (Bq/kg) of uranium, thorium and potassium in the samples. To estimate annual effective doses, account must be taken of (a) the conversion coefficient from absorbed dose in air to effective dose and (b) the indoor occupancy factor.

Annual estimated average effective dose equivalent received by a member is calculated using a conversion factor of 0.7 Sv/Gy, which is used to convert the absorbed dose rate to annual effective dose with an outdoor occupancy of 20% and 80% for indoors [10].

The annual effective doses are determined as follows:

$$\text{Indoor (mSv)} = (\text{Absorbed Dose}) \text{ nGy/h} \times 8760\text{h} \times 0.8 \times 0.7\ SvGy^{-1} \quad (8)$$

$$\text{Outdoor (mSv)} = (\text{Absorbed Dose}) \text{ nGy/h} \times 8760\text{h} \times 0.2 \times 0.7\ SvGy^{-1} \quad (9)$$

4. RESULTS AND DISCUSSION

The concentration of the radionuclides, ^{238}U, ^{232}Th and ^{40}K, and radium equivalent activity calculated using equations 1 and 2 in the fly ash samples studied in the present investigation are given in Table 1.

TABLE 1. ACTIVITY CONCENTRATION OF URANIUM, THORIUM AND POTASSIUM IN FLY ASH SAMPLES

Sample Code	Activity Concentration (Bq/kg)			Radium Equivalent Activity (Ra_{eq}) (Bq/kg)
	^{238}U	^{232}Th	^{40}K	
FA-1	89±3	103±6	344±1	263
FA-2	76±3	123±2	154±2	264
FA-3	23±6	197±3	342±5	331
FA-4	89±4	212±1	786±6	453
FA-5	221±2	154±4	698±4	495
FA-6	89±1	123±5	BDL	453
FA-7	230±2	67±3	455±4	361
FA-8	98±5	103±2	654±4	296
FA-9	56±3	99±5	412±5	229
FA-10	112±4	45±1	BDL	243
FA-11	56±3	109±2	543±6	254
FA-12	69±1	125±2	876±5	315
FA-13	114±5	167±1	345±3	379
FA-14	109±6	67±4	453±2	240
FA-15	234±6	54±6	BDL	365
FA-16	169±3	BDL	BDL	432
FA-17	154±2	76±5	BDL	342
FA-18	64±4	123±3	659±5	291
FA-19	89±4	107±4	352±3	269
FA-20	106±5	BDL	230±2	321

MADL (Minimum Activity Detection Limit) = 2Bq/kg, 2Bq/kg and 4Bq/kg for ^{226}Ra, ^{232}Th and ^{40}K respectively.

The radiological hazards in terms of external hazard index (Hex), internal hazard index (H_{in}), Gamma index (I_γ) and Alpha index (I_α) for the samples calculated using equations 3, 4, 5 and 6 are given in Table 2.

The radiation absorbed dose and annual effective dose from fly ash calculated using equations 7, 8 and 9 are given in Table 3. Our findings are in good agreement with the findings of other researchers reported in the literature [9, 11-12] The concentration of the radionuclide ^{40}K is higher in some of the samples but for all the samples analyzed, the radium equivalent activity value is well within the permissible limits of 370Bq/kg [9, 13].

TABLE 2. EXTERNAL HAZARD INDEX, INTERNAL HAZARD INDEX, GAMMA INDEX AND ALPHA INDEX IN FLY ASH SAMPLES

Sample Code	Hex	H_{in}	I_{γ}	I_{α}
FA-1	0.71	0.95	0.93	0.45
FA-2	0.71	0.92	0.92	0.38
FA-3	0.89	0.96	1.18	0.12
FA-4	1.22	1.46	1.62	0.45
FA-5	1.34	1.93	1.74	1.11
FA-6	0.72	0.96	0.91	0.45
FA-7	0.97	1.60	1.25	1.15
FA-8	0.80	1.06	1.06	0.49
FA-9	0.62	0.77	0.82	0.28
FA-10	0.48	0.78	0.60	0.56
FA-11	0.69	0.84	0.91	0.28
FA-12	0.85	1.04	1.15	0.35
FA-13	1.02	1.33	1.33	0.57
FA-14	0.65	0.94	0.85	0.55
FA-15	0.84	1.47	1.05	1.17
FA-16	0.46	0.91	0.56	0.85
FA-17	0.71	1.13	0.89	0.77
FA-18	0.78	0.96	1.05	0.32
FA-19	0.73	0.97	0.95	0.45
FA-20	0.33	0.62	0.43	0.53

TABLE 3. RADIATION ABSORBED DOSE AND ANNUAL EFFECTIVE DOSE FROM FLY ASH SAMPLES

Sample Code	Absorbed Dose Rate ($nGyh^{-1}$)	Annual Effective Dose (mSv)	
		Indoor	Outdoor
FA-1	118	0.58	0.14
FA-2	116	0.57	0.14
FA-3	144	0.71	0.18
FA-4	202	0.99	0.25
FA-5	224	1.10	0.27
FA-6	115	0.57	0.14
FA-7	166	0.81	0.20
FA-8	135	0.66	0.16
FA-9	103	0.50	0.13
FA-10	79	0.39	0.09
FA-11	114	0.56	0.14
FA-12	144	0.71	0.18
FA-13	168	0.82	0.21
FA-14	110	0.54	0.13
FA-15	141	0.69	0.17
FA-16	78	0.38	0.09
FA-17	117	0.57	0.14
FA-18	131	0.64	0.16
FA-19	120	0.59	0.15
FA-20	59	0.29	0.07

The H_{ex} and H_{in} for the studied samples are less than unity and therefore these samples are safe from health point of view. Value of $I_\gamma \leq 2$ corresponds to a dose rate criterion of 0.3 mSv/y, whereas $2 \leq I_\gamma \leq 6$ corresponds to a criterion of 1 mSv/y [6]. Thus, the samples with $I_\gamma > 6$ should be avoided to be used, since these values correspond to the dose rates higher than 1 mSv/y which is higher than the recommended safe limit values [9]. All the current 'I_γ' values of the studied samples follow the criterion ($I_\gamma \leq 2$) therefore it may be concluded that the samples are safe from health and hygiene point of view and don't pose any significant health hazards to the consumers/workers. The recommended limit for concentration of ^{226}Ra is 200 Bq/kg, for which $I_\alpha = 1$ [14]. The observed values are less than unity except one sample showing that the most of the samples are safe from the health and hygiene point of view and don't pose any environmental radiation hazards.

In all the samples, the indoor annual effective dose was less than the recommended limit of 1 mSy/y for general public [6].

5. CONCLUSIONS

The present investigations showed the inborn radioactivity in coal samples was modified during technological enhancement (converting coal into ash) but all the samples were found to satisfy the safety criteria. Efforts should be made at national and international level to reduce ^{226}Ra activity in the fly ash as the ash is frequently being used as a building construction material and for some other purposes like land filling, cement manufacturing etc.

REFERENCES

[1] AKERBLOM, G., WILSON, C., Radon gas: radiation hazard from radioactive bedrock and building materials, Bulletin of the Int. Asso. Engineering Geology, 23 (1981) 51.
[2] ABU-JARAD, F., et al., A study of radon emitted from building materials using plastic a-track detectors, Phy. Med. Biol. 25 (4) (1980) 683.
[3] AL-JARALLAH, M.N., Radon exhalation from granite used in Saudi Arabia, J. Environ. Radioact. 53 (2001) 91.
[4] SONKAWADE, R.G., et al., Natural radioactivity in common building construction and radiation shielding materials, Atmos. Environ. 42 (2008) 2254.
[5] BERETKA, J., MATHEW, P.J., Natural radioactivity of Australian building materials, industrial wastes and by products, Health Phys. 48 (1985) 87.
[6] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, 1990-91 Recommendations of the. ICRP Publication 60, Annals of the ICRP, Pergamon Press, Oxford, UK (1991).
[7] RIGHI, S., BRUZZI, L., Natural radioactivity and radon exhalation in building materials used in Italian dwellings, J. Environ. Radioact. 88 (2006) 158.
[8] KRIEGER, V.R., Radioactivity of construction materials, Betonwerk Fertigteil Tech. 47 (1981) 468.
[9] UNITED NATIONS SCIENTIFIC COMMITTEE ON THE EFFECTS OF ATOMIC RADIATION, Sources and Effects of Ionizing Radiation, United Nations Publication, New York, USA (2000).
[10] UNITED NATIONS SCIENTIFIC COMMITTEE ON THE EFFECTS OF ATOMIC RADIATION, Report to the General Assembly, United Nations, New York (1993).
[11] BOUKHENFOUF, W., BOUCENNA. A., The radioactivity measurements in soils and fertilizers using gamma spectrometry technique, J. Environ. Radioact, 102 (2011) 336.
[12] RAFAT, M.A., et al., Assessment of natural radioactivity and radon exhalation rate in Sannur cave,
eastern desert of Egypt, J. Radiol. Prot. 28 (2008) 213.
[13] OECD, Exposure to radiation from natural radioactivity in building materials, Report by a group of
experts of the OECD Nuclear Energy Agency, Paris, France (1979).
[14] The Radiation Protection in Denmark, Iceland, Norway and Sweden, Naturally occurring radiation in the Nordic Countries- recommendations, The Flag-Book Series (Reykjavik) (2000).

EXTERNAL DOSE RATE AT THE ORE TREATMENT UNIT, A DEACTIVATED MINING AND MILLING URANIUM INSTALLATION

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Abstract

The Ore Treatment Unit (UTM), formerly known as the Industrial Mining Complex of Poços de Caldas (CIPC), was the first uranium mine in Brazil and got its name in 2005 when it was used in the treatment of monazite for the extraction of rare earths. Along the period 2002-2014, sampling of exposition rates (in $mrem \cdot h^{-1}$) were realized using a Geiger-Müller equipment, and the lectures converted to $mSv \cdot h^{-1}$, using a conversion factor. In the present work, a descriptive statistical analysis and a histogram of frequencies were performed together with a radioprotection analysis. A set of 22,252 measurements of external dose rate was analyzed. The average value was 0.0068 $mSv \cdot h^{-1}$ with a standard deviation of 0.03 $mSv \cdot h^{-1}$. This high standard deviation associated with high kurtosis (66) and high skewness (7.68) indicate that the data fits the log-normal distribution rather than the Gaussian distribution. The maximum value was 0.42 $mSv \cdot h^{-1}$, equal to 4 times the limit of dose rate for the occupationally exposed individual. More than 92% of the data were below the derived limit per hour for the dose rate which is 0.01 $mSv \cdot h^{-1}$, and less than 8% of the data (1,699 readings) were higher than this limit.These data indicate that the processes of uranium mining and extraction of rare earths from monazite were performed under efficient supervision of the radiation protection service that maintain the process under control from the point of view of Radiation protection, and optimized doses.

1. INTRODUCTION

The Ore Treatment Unit (UTM) is the successor of the Poços de Caldas Industrial Complex (CIPC) which was the first facility for uranium mining and processing in Brazil. It is located in the municipality of Caldas, plateau of Poços de Caldas, Minas Gerais, Brazil. In the mid-1990s, with the end of the economic viability of uranium exploration, CIPC was disabled. In 2005, with the operation of 400 tons of monazite for rare earth extraction, CIPC was renamed UTM [1, 2, 3].

The former facilities of CIPC and those used by UTM represent areas with possible radiological risk. To ensure the safety conditions of workers, operations and areas a program of occupational monitoring was maintained aiming to understand the radiological conditions in the area and, consequently, to be able to plan the entry of employees, of personal and

collective protective equipment, the entry and exit of areas, the optimization of received doses and, finally, the optimization of radiological conditions in the area [4, 5, 6].

One of the ways of worker exposure is the external exposure to dose rates, which should be kept as low as reasonably achievable [6]. The present work aims to analyze statistically the behavior of the areas of CIPC/UTM regarding external exposure in the period 2002-2012.

2. METHODOLOGY

The first step was to identify the areas as controlled, supervised and free, through modeling and confirmation with monitoring devices [4, 5]. After areas definition, monitoring frequencies were defined; these ranged from daily, weekly, monthly to semiannual. Locations were sampled with frequencies set between 2002 and 2012 [4, 5]. A descriptive statistical analysis of data was conducted and a frequency histogram built up to evaluate the distribution of doses at CIPC/UTM [7]. The sampling was performed by trained technicians, using Geiger-Müller equipment, calibrated in $mrem \cdot h^{-1}$. Data were converted to $mSv \cdot h^{-1}$ by a conversion factor, tabulated by date of collection, sampling location, maximum, minimum and average values of each measure [4, 5]. For the analysis, the average values were used, making a total of 22,252 samples.

3. RESULTS

The data demonstrated a wide range (0-420 $\mu Sv \cdot h^{-1}$), with a low average of 6.8 $\mu Sv \cdot h^{-1}$, below the derived limit for workers (10 $\mu Sv \cdot h^{-1}$). The descriptive statistics, classes and frequencies of the classes are shown in Tables 1 and 2 respectively, and the histogram appears in Fig. 1. High kurtosis and skewness (66 and 8 respectively), besides the high standard deviation (29 $\mu Sv \cdot h^{-1}$) point to a distribution nearest a log-normal one, rather than to a Gaussian distribution. 92% of the sampled values were below the derived limit per hour (10 $\mu Sv \cdot h^{-1}$), slightly more than 78% of samples were below the derived limit per hour of investigation derived limit per hour and 87% below the derived limit per hour of intervention.

TABLE 1. DESCRIPTIVE STATISTICS OF EXTERNAL DOSE RATE MEASURES ($\mu Sv \cdot h^{-1}$) AT CIPC/UTM COMPLEX

External dose rate	(□Sv·h^{-1})
Median	6.75
Standard deviation	28.97
Kurtosis	66.03
Skewness	7.68
Range	420
Minimum	0
Maximum	420
Sum	150165.8
Counts	22,252

TABLE 2. CLASSES, CLASSES FREQUENCIES OF EXTERNAL DOSE RATE AT THE CIPC/UTM COMPLEX

Classes	*Frequency*	*Cumulative %*
1	9370	42.11
3	8127	78.63
6	2018	87.70
8	584	90.32
10	454	92.36
25	1045	97.06
30	128	97.64
40	35	97.79
Higher	491	100.00

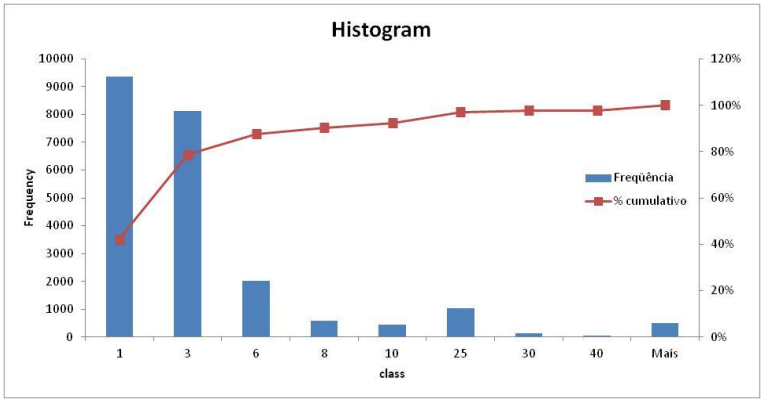

FIG. 1. Histogram of frequencies of external dose rates due to gama exposition of workers of the CIPC/UTM complex for the period 2002 -2014

4. CONCLUSION

The maximum values of dose rate found (420 $\mu Sv \cdot h^{-1}$) show unequivocally areas associated with high radiological risk (42 times the derived limit per hour). The average dose rate values demonstrated to be higher than the investigation levels, but reflect the weight of only 8% of the measures whose higher values are above the derived limit of 10 $\mu Sv \cdot h^{-1}$ [6]. This statement is based on the fact that almost 80% of the values are below the derived limit of dose rate per hour, with a large number of low dose values, are easily manageable. With 92% of the dose rates below the derived limit per hour for workers, the process of optimizing doses [6] need to be strengthened so that no value should exceed this limit, in order to be able to keep them below the derived limit per hour of intervention, in the next step. It is the aim, afterwards, to maintain all the values below the derived limit of dose rate per hour for investigation; optimize the exposures, and cultivate safety culture among the workers to improve further the radiation protection sytems [6].

REFERENCES

[1] WAGNER DE S. PEREIRA, ALPHONSE GAC KELECOM, JULIANA R. DE S. PEREIRA,DELCY DE A. PY JÚNIOR: Release of Uranium by an Ore Treatment Unit at Caldas, MG, Brazil. Journal of Environmental Protection, 06/2013, 4(6) (2013) 570-574.

[2] WAGNER DE SOUZA PEREIRA, SERGIO QUINET DE OLIVEIRA, DELCY DE AZEVEDO PY JÚNIOR, MARCELINO V A DANTAS, ANA CLAUDIA ANTUNES SILVA, OSWALDO GARCIA FILHO, JULIANA RODRIGUES DE SOUZA PEREIRA, avaliação de onze anos de monitoração de área para taxa de dose externa em um depósito de rejeitos radioativos na unidade de tratamento de minério – *UTM*.. XV Congresso Brasileiro de Energia, Rio de Janeiro, (2013)

[3] FERNANDES, H.M., VEIGA, L.H.S., FRANKLIN, M.R., PRADO, V.C.S. AND TADDEI, J.F. Environmental impact assessment of uranium mining and milling facilities: a study case at the Poços de Caldas uranium mining and milling site, Brazil. Journal of Geochemical Exploration, 52 (1994) 161-173.

[4] Industrias Nucleares do Brasil, Programa de proteção radiológica do Complexo minero-industrial do Planalto de poços de caldas (CIPC). (1998).

[5] Industrias Nucleares do Brasil, Programa de proteção radiológica da Unidade de Tratamento de Minério. (2004).

[6] Comissão Nacional de Energia Nuclear, Norma CNEN-NN-3.01, Diretrizes Básicas de Proteção Radiológica, (2005) 34.

[7] CETENO, A. J., Curso de estatística aplicada a biologia. Goiás: UFG, (1981) 188.

CONTAMINATION BY LONG HALF-LIFE ALPHA EMITTERS AT THE ORE TREATMENT UNIT, A DEACTIVATED MINING AND MILLING URANIUM INSTALLATION

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Abstract

The first uranium mine in Brazil known as the Industrial Mining Complex of Poços de Caldas (CIPC) received in 2005 the name Ore Treatment Unit (UTM) when it was used in the treatment of monazite for the extraction of rare earths. Samples of particulate matter in the air were performed between 2002 and 2012, using high volume sampling pumps (HIVOL) and cellulose acetate filters which were subsequently counted in a total alpha proportional counter to estimate the concentration in air of long half-life total alpha emitters (in $Bq \cdot m^3$). We here report a descriptive statistical analysis and a data histogram carried out on the 2,956 counts obtained. Among these, 21 samples (0.71%) had scores lower than the background; 99.26% of samples were below the derived limit for concentration in air of alpha emitters of long half-life (DL = 0.37 $Bq \cdot m^{-3}$) and 22 were above the derived limit (0.71%). The average value was one order of magnitude below the DL (0.015 $Bq \cdot m^{-3}$) but with a standard deviation ten times higher and with a maximum value of 3.7 $Bq \cdot m^3$. The high kurtosis (284) and the strong skewness (3.7) indicated a log-normal distribution of data. These data show a proactive action of the radioprotection service, keeping the concentration in air of long half-life alpha emitters within allowed limits, and optimizing the doses received in a way making them as low as reasonably achievable.

1. INTRODUCTION

The Ore Treatment Unit (UTM) is the successor unit of the Poços de Caldas Industrial Complex (CIPC) which was the first unit of uranium mining and milling in Brazil. It is located in the municipality of Caldas, plateau of Poços de Caldas, Minas Gerais, Brazil. In the mid-1990s, with the end of the economic viability of uranium exploration, CIPC has been disabled. In 2005, starting the operation of 400 tons monazite for rare earths extraction, CIPC was renamed UTM. [1, 2, 3]

The former facilities of CIPC and those used by UTM represent areas with possible radiation risk. In order to ensure the safety conditions of workers, operations and areas a program of occupational monitoring was created aiming to understand the radiological conditions in the area, and with this to plan the entry of employees, personal and collective protective equipment, the procedures of entry and exit of areas, the optimization of received doses and the optimization of radiological conditions in the area [4, 5, 6].

One of the forms of worker exposure is by inhalation of long half-life alpha emitters, existing in the air. The present work aims to analyze statistically the behavior of the areas of CIPC/UTM in relation to air contamination by long half-life alpha emitters, in the period 2002-2012.

2. METHODOLOGY

The first step was to identify the areas as controlled, supervised and free, through modeling and confirmation with monitoring devices [4, 5]. After definition of the areas, the monitoring frequencies were defined and ranged from daily, weekly, monthly to semiannual. Locations with the frequencies set between 2002 and 2012 were sampled [4, 5]. A descriptive statistical analysis of the data was carried out and a frequency histogram obtained, in order to evaluate the distribution of doses at the CIPC/UTM complex [7]. The sampling was performed by trained technicians using high sampling air volume equipment (HIVOL), and subsequent analysis on a proportional counter; results were expressed in $Bq \cdot m^{-3}$. Knowing the composition of radionuclides present in the air, the derived limit of alpha emitters concentration in air (DL) was evaluated as being 0.37 $Bq \cdot m^{-3}$ [8] at a dose of 20 $mSv \cdot y^{-1}$. Data were tabulated by date of collection, sampling location, value of each measure, for a total of 2,956 samples.

3. RESULTS

The average value obtained (0.0015 $Bq \cdot m^{-3}$) was one order of magnitude lower than that of the DL concentration of alpha emitters in air (0.37 $Bq \cdot m^{-3}$), for the DL to the worker (20 $mSv \cdot y^{-1}$). The maximum value was ten times the DL (37 $Bq \cdot m^{-3}$). The high kurtosis and skewness (284 and 16, respectively), together with the high standard deviation (0.15 $Bq \cdot m^{-3}$) indicated a frequencies distribution close to the log-normal, distant from a Gaussian distribution. The values of the descriptive statistics appear in Table 1; the classes of frequencies and their respective frequencies can be seen in Table 2, and the frequency histogram is shown in Fig. 1.

TABLE 1. DESCRIPTIVE STATISTICS OF THE CONCENTRATIONS OF LONG HALF-LIFE TOTAL ALPHA EMITTERS IN THE AIR AT THE CIPC/UTM COMPLEX ($Bq \cdot m^{-3}$)

LONG HALF-LIVE TOTAL ALPHA ($Bq \cdot m^{-3}$)	
Average	0.015
Mode	0.000001
Standard deviation	0.15
Sample variance	0.02
Kurtosis	284.28
Skewness	15.81
Range	3.74
Minimum	0
Maximum	3.74
Sum	45.63
Counts	2.956
Confidence level (95,0%)	0.005372

TABLE 2. CLASSES, FREQUENCIES AND CUMULATIVE FREQUENCIES OF LONG HALF-LIFE TOTAL ALPHA EMITTERS IN THE AIR AT THE CIPC/UTM COMPLEX ($Bq \cdot m^{-3}$)

Classes	Frequency	Cumulative %
0	21	0.71
0.37	2913	99.26
0.74	7	99.49
1.11	2	99.56
1.48	1	99.59
1.85	5	99.76
2.22	3	99.86
2.59	2	99.93
2.96	1	99.97
3.33	0	99.97
3.7	0	99.97
higher	1	100.00

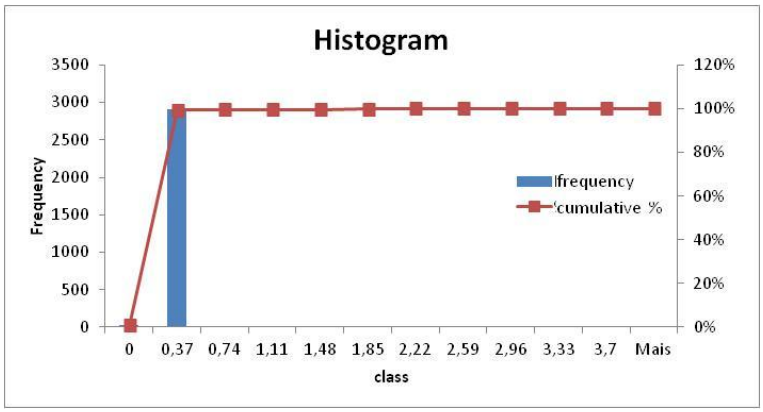

FIG. 1. Histogram of frequencies and cumulative frequencies of long half-live total alpha emitters in the air at the CIPC/UTM complex.

4. CONCLUSIONS

More than 99% of the sample values were below the DL. Of the 2,956 collected samples, only 22 were above the DL, showing that even working within legal limits, in a few cases, values above the legal values occured. In this case, the control of the time of permanency of the worker, the use of PPE's, and optimized procedures are essential. More rigid controls of the source terms must be applied in order to bring all samples within the legal limit. Thus, the optimization process should be strengthened and further efforts must be applied to control the source term, avoiding the dispersion of radionuclides in the air, in order to optimize control of residence time and the use of PPE's. In addition, the safety culture must be developed to reinforce the process of optimization. Training must be increased to improve the awareness of employees and, ultimately, to take care of their own safety.

REFERENCES

[1] WAGNER DE S. PEREIRA, ALPHONSE GAC KELECOM, JULIANA R. DE S. PEREIRA, DELCY DE A. PY JÚNIOR: *Release of Uranium by an Ore Treatment Unit at Caldas, MG, Brazil*. Journal of Environmental Protection, 4(6) 2013) 570-574.

[2] WAGNER DE SOUZA PEREIRA, SERGIO QUINET DE OLIVEIRA, DELCY DE AZEVEDO PY JÚNIOR, MARCELINO V A DANTAS, ANA CLAUDIA ANTUNES SILVA, OSWALDO GARCIA FILHO, JULIANA RODRIGUES DE SOUZA PEREIRA: avaliação de onze anos de monitoração de área para taxa de dose externa em um depósito de rejeitos radioativos na unidade de tratamento de minério – UTM.. XV Congresso Brasileiro de Energia, Rio de Janeiro, (2013)

[3] FERNANDES, H.M., VEIGA, L.H.S., FRANKLIN, M.R., PRADO, V.C.S. AND TADDEI, J.F., Environmental impact assessment of uranium mining and milling facilities: a study case at the Poços de Caldas uranium mining and milling site, Brazil. Journal of Geochemical Exploration, 52 (1994) 161-173.

[4] INDUSTRIAS NUCLEARES DO BRASIL, Programa de proteção radiológica do Complexo minero-industrial do Planalto de poços de caldas (CIPC), (1998)

[5] INDUSTRIAS NUCLEARES DO BRASIL, Programa de proteção radiológica da Unidade de Tratamento de Minério. (2004)

[6] Comissão Nacional de Energia Nuclear, Norma CNEN-NN-3.01, Diretrizes Básicas de Proteção Radiológica, 34 (2005) 34.
[7] CETENO, A. J., Curso de estatística aplicada a biologia. Goiás: UFG, (1981) 188.
[8] WAGNER DE SOUZA PEREIRA, A. KELECOM, J. R. S. PEREIRA, D. A. PY JÚNIOR, A C. A. SILVA, O. GARCIA FILHO, Radiological control in a mine with a naturally occurring radioactive material - NORM: II assessment of external exposition in a pilot plant.. 2013 International Nuclear Atlantic Conference - INAC 2013 Recife, PE, Brazil, November 24-29, 2013, Recife, PE, Brazil, November 24-29, 2013, 11(2013).

Session 8: Occupational radiation protection in the workplaces involving exposure to radon

THE IMPLICATIONS OF THE PROPOSED CHANGES IN RADON DOSIMETRY TO THE URANIUM INDUSTRY

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Abstract

In 2009, the International Commission on Radiological Protection published a statement on radon stating some significant departures from the approach currently in existence. The changes related to the most recent results of epidemiological studies which suggested a higher risk per unit exposure and also a move to the use of dosimetric rather than epidemiological approach to the calculation of dose conversion factors. For the uranium industry, the changes in radon dosimetry could be very important to the viability of some uranium operations and in particularly operations with a higher potential for radon exposure such as underground operations. Under the World Nuclear Association banner, a number of uranium mining companies formed a working group to examine the implications of the proposed changes and also the practical approaches required to ensure a high standard for occupational radiation protection. These studies included examining the existing radon and radon decay product doses, the characterisation of the physical properties of the radon decay products, and also the effectiveness of corrective measures such as increased ventilation, the use of purification of supplied air and the effectiveness of protective equipment. The overall aim of the uranium industry is to remain at the forward edge of the science of radiation protection to ensure the safety and protection of the workforce in a practical and appropriate manner.

1. BACKGROUND

Historically, the dosimetry of radon and more specifically Radon Decay Products (RDP) has been handled differently than other internal radionuclides. In 1995, the International Commission on Radiological Protection (ICRP) published ICRP-65 [1] which provided dose conversion factors (DCF) based on epidemiological studies conducted at this time. This was different from other internal emitters because radon and RDP was the only radionuclides with direct epidemiological evidence. The difference between the DCF calculated by reference biokinetic and dosimetric parameters and the epidemiological approach was of the order of two.

In November 2009, ICRP published a statement on radon [2] which described the current state of epidemiological studies on radon. This indicated that the risk per unit intake was higher than previous studies had indicated and the increase was of the order of two. The statement also indicated that as there was now a closer line-up between the dosimetric and epidemiological DCF than it would be recommended the use of a reference biokinetic and dosimetric models. This would be consistent with the methodology used for other radionuclides and would also allow consideration of different physical attributes of the RDP. In particular, it could allow for consideration of different aerosol parameters relating to RDP particle sizing which is critical in calculating the size specific DCF.

Since 2009, the ICRP has published additional information on Lung Cancer Risk from Radon and Progeny (ICRP-115) [3], but at the time of writing this paper had not published revised DCF for radon (and recommended the continued use of DCF from ICRP-65). ICRP-115 also raised the relationship between lung cancer risk due to radon with the confounder of smoking. Evaluation of the epidemiological data indicated the risk relationship between radon

and smoking was not additive but multiplicative in nature. Although ICRP indicates the use of a single combined group for determination of dose conversion factors this does raise questions around how this could be managed in occupational exposure situations.

2. URANIUM INDUSTRY APPROACH

Radiation protection from radon and RDP is critical for the uranium industry. Depending on the nature of the operation, radon can be the major contributor to occupational exposure and also is critical in determining exposure to the public both during operations and post closure. Industry also believes that it is critical to be at the forefront of the science to ensure the best possible protection for its workforce. Due in part to the significant change in the approach to radon, six uranium companies formed a working group within the World Nuclear Association (WNA), to address the changes occurring with potential to affect the industry. The changes to radon dosimetry were one of the focus areas for the resulting WNA Uranium Mining Standardisation Working Group. The group focused on the science behind the changes and also on the practical implications this would mean in terms of all aspects of occupational radiation protection.

The first focus concentrated on the chance in risk factors by of the order of two. Under the banner of the WNA working group, the uranium industry reviewed how a doubling of the DFC for RDP would affect worker doses and also what measures could be implemented to remediate against any change in doses. Included in this was consideration of how these changes could best be communicated to the workforce and the need for communication with regulatory agencies on implications of this change. This was complicated by the lack of any definitive values being available from the ICRP and also the recommendation to continue using the pre-existing DCF.

The second focus was on the physical properties of RDP within the mining environment. Since the early 1990's little research had been performed on characterising RDP within mining environments. In particular, RDP particle sizing data had no recent information and there was no available equipment or methodology to determine these factors within a modern mine. Using existing biokinetic and dosimetric models, the DCF for RDP is heavily dependent on the particle sizing. Variations of over an order of magnitude are possible depending on the particle size distribution (particularly in the nanometre range) and this information is critical for determining doses should the dosimetric approach be recommended. There was also concern that the measures being implemented in modern mines, for occupational hygiene reasons, could actually increase dose due to reduction in the availability of condensation nuclei in the mine atmosphere. This led to some concern that by reducing other important parameters (such as reduced diesel particulates due to better emission control), radiation doses could actually be increased. The lack of modern data is therefore of high importance to industry and also to provide feedback to ICRP on implications of the change in approach.

3. CHANGE IN RISK FACTOR AND MITIGATION MEASURES

The majority of workers in modern uranium mines have occupational exposures well below the recommended limits. This has been due to a firm commitment to the optimisation of radiation doses and in particular to having a strong focus on the principle of keeping doses As Low As Reasonably Achievable societal and economic factors being taken into account (ALARA). Also, radon and RDP is only a small contributor to most workers with the exception of underground miners and some speciality roles such as maintenance in confined spaces. This means that a change in the radon DCF by of the order of two could be absorbed into current operation methodologies with little need for definitive changes in occupational

roles. However, there would be a need for more focus on a small number of individuals and similar exposure groups for a small number of mines. However, the gap between doses and the recommended limits would be reduced and hence industry is examining a range of options to improve practices to further reduce exposure and hence dose.

In calculating radon and RDP dose, there is generally a significant amount of conservatism in how the dose is calculated. It has always been the approach of industry that where there is uncertainty than the conservative approach is taken and this has led to some overestimation of dose. Examples include the inclusion of background radon doses in the occupational contribution and the use of time and area monitoring to determine doses. By reducing this conservatism, a more accurate, and lower, measure of exposure can be determined and this can reduce the implication of the change in risk factors. It will involve more stringent monitoring practices and will probably result in the use of more personal dosimeters for monitoring purposes. It will also require more communication with both regulators and the workforce on both the change in approach and the reasons behind the overall change in risk factor. Difficulties in this communication should not be underestimated as any change of this significance can be very difficult to understand and convey.

There are also a range of mitigation measures being considered by industry which directly reduce the exposure to radon and RDP. The concentration of radon, and its equilibrium relationship with RDP is heavily dependent on such factors as ventilation and residency times. In underground uranium operations this is already a priority focus and further improvements are possible. However, there is appoint at which further improvements are not practical (due to a range of economic, engineering and hygiene related reasons). There is also potential for purifying the air workers breathe in the course of their duties. Although removal of radon is difficult and not practical in most cases, RDP being particulates can be removed using conventional filtration processes. For example, a significant proportion of underground workers operate within an air-conditioned work environment (for safety and heat stress reasons). By using filtration on the air-conditioner, it is possible to reduce the RDP concentration that a worker breaths. In this case there are a number of important factors to consider (such as residency time, one pass ventilation and no air recirculation) but significant reductions to RDP can be realised. Similarly, worker are often required or chose to use some form of respiratory protection, such as Powered Air Purifying Respirators, which offer protection from a range of potential hazards (such as dust, rock-fall, eye damage and heat stress). These items give high levels of reduction in RDP and due to the extremely short residency time, can greatly reduce RDP exposure.

4. DETERMINATION OF THE PHYSICAL PROPERTIES OF RDP IN MINE ATMOSPHERES

Prior to the early 1990’s there was considerable scientific research undertaken on determining the physical parameters of RDP in a range of atmospheres including uranium mines. This utilised a range of technologies ranging from simple wire screen RDP monitoring through to cascade or multiple impactors to provide a full characterisation of the RDP size distribution. However, following the publication of ICRP65, interest in this specialised of field of particle science was drastically reduced and in effect no significant work has been performed in the last twenty years. However, mine environments have substantially changed in this period, generally with the emphasis on providing better environments to the workers. Focus on such aspects as the removal of diesel particulates from the mine atmosphere and reduction in dust and fumes arising from blasting has reduced the number of condensation nuclei in the mine atmosphere. This in turn has the potential to significantly alter the size distribution of the RDP and increase the proportion of RDP in the nanometre size range for a certain residency time.

Since the 2009 announcement of the ICRP, there has been a concerted effort to better understand the modern mine atmosphere. However, this has been very difficult due to both the absence of current scientific study on this specialised area and the lack of equipment capable of monitoring the size distribution of ‘RDP. In Australia a joint Industry, Commonwealth and State task group was set up to resurrect the old equipment and science and commence a fresh round of atmospheric characterisation. Luckily some twenty plus year old equipment was still available and with the help of some technological wizardry was able to be made operational again. This equipment has been in active use to re-characterise the atmosphere in an underground mine that had previously been examined in the 1990s [4]. However, this equipment remains unsuitable for routine use in an operating uranium mine so further development of robust operational equipment is required to allow routine determination of RDP size distribution. This remains a priority for industry and this work is also important to the scientific community and the ICRP as it will be of high importance for future re-examination of epidemiologic studies. For non-mine workplaces there is even less information and the revision of radon dose conversion factors may greatly increase the significance of these workplaces. It will also be of high important for other non-occupational exposures to radon and RDP as environmental characterisation of RDP size distribution is very poor and this may be critical for future implementation of dosimetric models.

5. THE SMOKING CONFOUNDER

In ICRP-115, a summary of epidemiological studies on radon gave strong evidence that the risk relationship between radon exposure and the smoking confounder was multiplicative. This shows the strong dominance of smoking with respect to lung cancer risk but it does raise the question on how radon and RDP exposure should be handled with respect to a mixed worker population of smokers and non-smokers. The ICRP appears to be pursuing the use of a combined population for the determination of dose conversion factors. However, how this will apply when we have a well-known and defined group of workers who smoke is still of concern. The uranium industry is awaiting direction from ICRP on this issue.

6. CONCLUSION

Radon and the inhalation of RDP is one of the significant pathways for exposure within uranium operations. This is particularly important for underground operations and also where an operation has areas of restricted ventilation associated with ore or process material. The ICRP proposed changes in radon risk and the move to a biokinetic and dosimetric approach has the potential for significant impact on occupational exposures. Industry is already working to both extend the current baseline knowledge on radon and RDP, and is also developing mitigation approaches to ensure workers remain well protected at levels below the limit.

REFERENCES

[1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Protection against Radon-222 at Home and at Work, ICRP65, (1993).

[2] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Statement on Radon, ICRP, (2009).

[3] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Lung Cancer Risk from Radon and Progeny and Statement on Radon, ICRP-115, (2010).

Session 9: Occupational radiation protection in medicine

IMPACT OF CAMEROON REGULATORY TECHNICAL CONTROLS ON OCCUPATIONALLY EXPOSED WORKERS IN MEDICINE

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Abstract

The number of occupationally exposed workers in medicine has been increasing rapidly over the years in Cameroon, and individual occupational exposure varies widely among those involved in medical care. There are certain medical procedures that might give substantial doses to medical staff, and the education of medical professionals in radiological protection aspects is rquired to be done and it is a continuing problem. National Radiation Protection Agency (NRPA) was created on 31st October 2002 under Decree n° 2002/250 and Law n° 95/08 of 30th January 1995 on radiation protection. NRPA is mandated to regulate all ionizing radiation sources as well as the protection of people and environment against ionizing radiation hazards. NRPA activities started with inventory program in 2009, which was completed in July 2010. Over 500 occupationally exposed workers were recorded with about 12 % being monitored. There was no national dosimetry service provider, all were monitored using the service provider abroad. Regulatory control started in 2011. There were lots of discrepancies recorded at the level of occupational monitoring and state of equipement used. NRPA acquired robust dosimetric monitoring kits in 2011, which are widely used in radiation protection, offer a number of potential advantages for monitong occpational exposures in medical applications of radoisotopes in diagnosis and therapy.

1. INTRODUCTION

Created in 2002 in application of Law n° 95/08 of 30th January 1995 on radiation protection , the NRPA started functioning in 2007 with the appointment of a management team. The following regulations reinforced the decree of application of the law:

i. Arrêté N° 1150 /A/MINSANTE on Work Place Monitoring of 11 June 2013 laying down the conditions and signaling of supervised and controlled areas and especially restricted and prohibited zones as well as imposed hygiene, safety and maintenance rules.
ii. Arrêté N° 1151 /A/MINSANTE on Dosimetry Monitoring of 11 June 2013 laying down procedures of medical and occupational monitoring of workers and patients exposed to ionizing radiation.
iii. Arrêté N° 1152 /A/MINSANTE on Licensing and Practices Modalities for X Ray Generators of 11 June 2013 laying down procedures for possession, usage and handling of devices emitting X-rays in hospital.

Data on radiation sources were collected in 2009 and 2010 nation-wide. Three (03) ^{60}Co Category 1 radioactive sources are used for radiotherapy. Fifty seven Category 4 radioactive sources are available for brachytherapy and Nuclear Medicine. About 415 X-ray machines are used in medicine. The distribution is shown in Table 1 and Fig. 1.

TABLE 1. X-RAY MACHINES USED IN MEDICINES

Practice	Number
CT Scanner	24
Mammography	36
Dental	90
Conventional/Interventional Radiography	265
Total	415

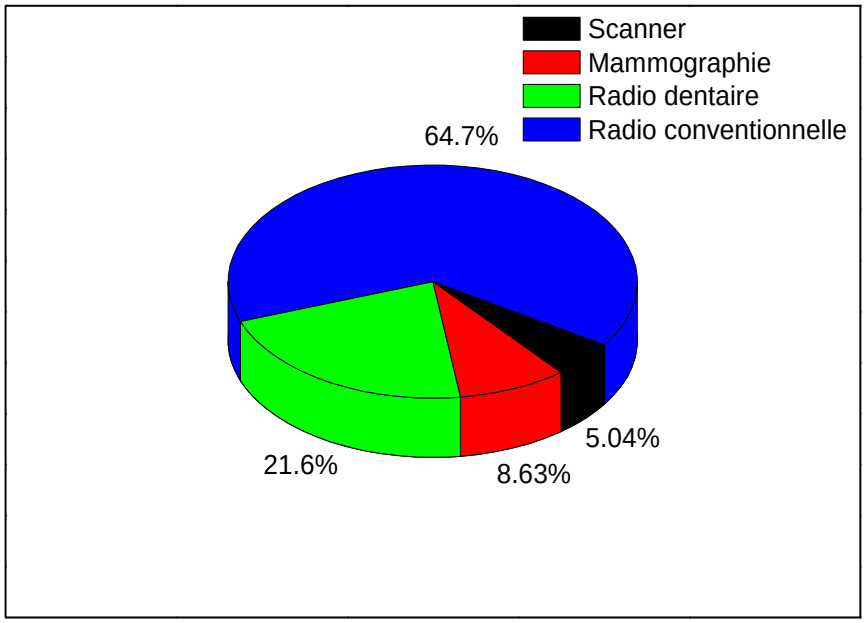

FIG. 1. Distribution of the medical procedures

TABLE 2. NUMBER OF RADIATION WORKERS IN MEDICINE AND NUMBER MONITORED IN 2011

Pratice	Number of Workers	Number of Workers Monitored
Diagnostic and Interventional Radiology	502	45
Radiotherapy	18	10
Médecine nucléaire	10	10
Total	530	65

As of 2011, out of the 530 occupationally exposed workers, only 12 % are being monitored. This required adequate and enhanced regulatory control masures.

Two ratdiotherapy centres and one nulear medicine with over 100 medical facilities have been programmed and visited so far. Regulatory controls involved inspection and some quality control tests carried out on X-ray machines, and to verify their performances.

NRPA acquired dosimetric monitoring kits with 500 individual dosimeters, 50 patient dosimeters and 50 environmental dosimeters with a TLD reader in April 2011. Individual monitoring started in June 2011. Before this time, all individaul monitoring were serviced from abroad, Europe, South Africa and USA.

2. METHODOLOGY: REGULATORY TECHNICAL CONTROLS

2.1 Technical Specifications of Machines which may lead to leakages of radiation

Equipment of diagnostic radiology should conform to applicable national or international standards such as the International Electro technical Commission (IEC) or International Standards Organization (ISO). Such equipment should satisfy the technical requirements of verification of machine parameters, and Quality Assurance tests at large. Under all circumstances, the equipment should be accompanied the necessary documents including the service and operating manuals, results of acceptance tests, calibration certificate for the required machine parameters of the X-ray machine.

For all radiographic equipment the following technical requirements have to be met simultaneously:

i. All radiographic equipment control panels should be fitted with a clear light signal during exposure. It can alternatively be an audible signal.
ii. Exposure (or irradiation) should be initiated only when pressing the exposure switch. Termination should be effected by releasing the pressure on the switch or automatic interruption at the end of the set time.
iii. Adjustable beam limiting devices (Example skull units, collimator, cones) that help to keep the radiation beam within the limits of the film has to be provided.
iv. The inherent filtration of every X-ray tube assembly should be marked permanently and clearly on the housing.
v. Every filter should be marked permanently and clearly with its filtration in millimeters of Aluminum equivalent.
vi. Every X-ray source assembly should be marked to identify the nominal focal spot position, tube type/model, tube serial number, manufacturer, date of manufacture, etc.

In addition to the requirements mentioned above:

- A conventional radiography X-ray machine should be equipped with working grid, collimator and beam alignment devices,
- The total filtration of the beam should be equivalent to not less than 2.5mm of Al of which 1.5mm should be permanent, and
- The department should have a quality assurance program to perform QC tests periodically to make sure that the machine's quality conforms to the quality parameters set by NRPA [3].

The following checks were done with respect to these:

a) KVp accuracy $< \pm 10\%$
b) Timer accuracy $< \pm 10\%$

c) Collimator accuracy < ±2%
d) Beam alignment accuracy < ±2%
e) Out put consistency < ±5%
f) HVL (Half Value Layer) acceptable for a given kVp setting.
g) The leakage should be less than 100 mR per hour at maximum rating
h) Grid alignment should be acceptable

In order to avoid undue deviations of the machine parameters, it is advised that institutions arrange for an adequate power supply from the main line by providing appropriate thickness of wire and ensuring short distance from the transformer.

2.2 Control Methods for Workers

i. Placards containing international radiation hazard sign (Tri–foil symbol) and notices in English and French were checked to be available and posted at suitable locations.
ii. A warning red light should be available synchronized with the machine power to show when radiation is on.
iii. Local rules should be presented describing working procedures and safety rules.
iv. Protective panels having a protective equivalent of not less than 0.55 mm lead or lead aprons that provide equivalent shielding for operator protection should be provided.
v. Lead apron should be used for pregnant women and for comforters.
vi. Protective barrier for operator protection should be constructed. The barrier should adequately protect the operator. It should be provided with protection window made of lead glass to ensure operator protection as well as to have good patient view. The lead glass may be sufficient to have a nominal value of 2 mm lead or its equivalent.
vii. Lead aprons of 0.55 mm of lead are indispensable for the examiner and the radiographer as necessary during fluoroscopic procedures. Comforters, children, and pregnant women should be strictly provided with lead aprons under any exposure circumstances.
viii. Gonad shields of different sizes for lying and standing positions should be made available to be used by patients and pregnant women.
ix. Lead gloves should be provided for operators during fluoroscopy procedures and by comforters in supporting the patient.
x. Lead glass goggles have to be provided for protection of the eye of the patient, helpers or radiation workers [3].

2.3. Control of Technical Records

2.3.1. Periodic training of all Radiation Workers

The following monitored are classified as radiation workers and the management of the institution authorized to practice radiological services is required to arrange periodic training and retraining of the workers :

(a) Specialised physicians, medical physicists, technologists, Radiographers, nurses and radiopharmacists who are normally exposed to radiation in controlled areas.
(b) The users of radioisotope sources, such as clinical specialists, research staff and ancillary workers who frequently work in controlled areas [4].

2.3.2. Individual Monitoring

(i) All members of the radiological team should obtain personal monitoring services rendered by NRPA or any other accredited institution;
(ii) The management of the institution should arrange a systematic record of the doses of the workers and provide access to the information;
(iii) The management of the institution should make sure that no radiological service is conducted in the absence of such a monitoring service;
(iv) It is the responsibility of the worker and the management at large to take all the necessary steps to make sure that the dose received by workers is within the dose constraints but under all circumstances dose limits and dose constraints are not exceeded [1].

2.3.3. Dose limit and investigation level

(a) An effective dose of 20 mSv, averaged over five-year period, is adopted as the annual limit for occupational exposure. Radiological personnel should not, however, exceed a dose constraint of 12 mSv/y, set by the NRPA. A dose of 1 mSv in one month is, therefore calls for a further investigation in to the causes of the incident by the NRPA [2].
(b) Radiological Departments could also set an even lower dose constraint as their own investigation level and to initiate remedial actions.

3. RESULTS

The Decree of 2002 assigns responsibility to NRPA for Radiation Workers monitoring in Article 4. NRPA decided to start a Dosimetry monitoring of workers as service provider since there was non in the country.

The discrepancies noticed (e.g. non compliance of quality control tests, wrong dispositions of individual dosimeters and controlled dosimeters) during the technical controls as a cause of increased in risk of occupational exposure were reported and follow-ups made according to the recommendations. These helped to implement safety culture and therefore reduced exposure. The number of workers monitored and doses of monitored workers reduced considerably as illustrated in the table below:

TABLE 3. NUMBER OF RADIATION WORKERS IN MEDICINE AND NUMBER MONITORED FROM 2011 TO 2014

Year	Number of Facilities Registered	Number of Workers Monitored	Average Annual Cumulated Whole Body Dose
2011	05	64	1 mSv
2012	11	117	0.5 mSv
2013	22	203	0.4 mSv
2014	25	250	0.3 mSv

4. DISCUSSIONS

Occupational radiation protection is effective in medicine only if and only if there is a radiation protection culture put in place. This can only be adequately manifested through

routine regulatory controls to check compliances of equipement and the attitudes of the radiation workers with respect to their workplaces.

5. CONCLUSIONS

From the operational radiation protection aspects, it was noticed that most of the workers in medical facilities were not trained in the proper use of the Radiation Dosimeter, including how to properly wear the individual dosimeters. However, the safety culture of the workers has improved, and this lead to a remarkable reduction in the individual effective doses in most of the facilities. It should be noted that more than 30% (150 Medical and 100 industrial) of the workers are now monitored as compared to 12 % when there was no regulatory control mesures in place. The ultimte aim is to implement the necessary measures by putting in place a more comprehensive legal and regulatory framework so that 100 % of the workers can be monitored within the next two years.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety of radiation sources: International basic safety standards: general safety requirements, IAEA Safety Standards Series, ISSN 1020-525X ; no. GRS Part 3 (Interim)), STI/PUB/1531. ISBN 978–92–0–120910–8. IAEA, Vienna (2011).

[2] INTERNATION COMMISSION ON RADIOLOGICA PROTECTION, The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Annals of the ICRP 37(2-4) (2007) 1-332.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA-TECDOC-1526, Inspection of radiation sources and regulatory enforcement, ISBN:92-0-113106-2, IAEA, Vienna (2007).

[4] https://rpop.iaea.org/RPOP/RPoP/

EYE DOSIMETRY ASSESSMENT FOR INTERVENTIONAL RADIOLOGY AND CARDIOLOGY STAFF

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Abstract

By 2018, the new eye dose radiation dose limit of 20mSv is required to be implemented into law in the Republic of Ireland. It is expected that this will have implications for the classification of interventional radiation workers in radiology and cardiology. A study was undertaken to measure eye doses for a number of staff in these two departments. In addition, staff compliance with hospital policy on the use and wearing of personal protective equipment and apparel was audited. Findings which are relevant to the implementation of the new eye dose limit in our institution, and measures that can be employed by staff to reduce their personal radiation eye dose will be explored.

1. INTRODUCTION

It is now well accepted that radiologists and cardiologists may potentially receive significant radiation doses to their eyes while conducting interventional procedures [1, 2]. In recent years, the International Commission on Radiological Protection (ICRP) have reviewed epidemiological evidence on tissue reactions and concluded that the threshold dose for deterministic effects in the lens of the eye is 0.5 Gy [3]. Arising from these findings the ICRP proposed that the occupational dose limit for the eye be reduced from 150 mSv to 20 mSv, averaged over five years, with no single year allowed to exceed 50 mSv. The European Community (EC) in publishing the recently revised Basic Safety Directive [4] has included the ICRP recommendation. The EC have directed all member states to implement the new directive into National law by 2018 [4]. In the Republic of Ireland, it is likely when these new eye dose regulations are implemented, a classified radiation worker, will be defined as those staff members who are likely to receive greater than 15mSv to their eye arising from their work.

University Hospital Galway is one of the largest university affiliated tertiary referral centres in Ireland. The hospitals size and available "high end" interventional radiology/cardiology equipment has increased significantly over the past 15 years. Almost 3200 interventional cardiology (IC) procedures and 2169 interventional radiology (IR) procedures have been conducted at the hospital during 2013.

In view of the above, and recognising the increasing body of literature which highlights the issue of the impending reduction of the eye dose limit, and its consequences for , the increased need for personal dose monitoring and radiation workers classification [5], it was decided to conduct an initial six week study and estimate the scale of the potential problem. Armed with this knowledge, the hospital would be better positioned on how to address and comply with the impending legislation. The objectives of the study were firstly, to measure the eye dose for a range of IR and IC workers over a six week period, and secondly to observe

practice in the interventional laboratories with a view to determine how compliant staff were with hospital policy on employing the available personal protective equipment (PPE) and apparel.

2. METHODOLOGY

A total of seven staff, two cardiologists, three interventional radiologists, and two nurses were recruited to participate in the study for a six week period. They were all assigned an optically stimulated luminescence (OSL) radiation dosimeter (Nanodot, Laundauer, UK) [6] and requested to wear it for all interventional procedures for which they were present for the duration of the study. The dosimeter was attached to an elastic headband which was worn on the eyebrow above the eye which was in closest proximity to the source of the radiation scatter for each procedure. The OSL dosimeter (based on Al_2O_3) has a dynamic range of 10μGy to 100Gy. Laundauer provided a calibration to $H_p(0.07)$. $H_p(3)$ calibration was not available at the time of the study. However, ICRP 103 [7] states that, in practice the $H_p(0.07)$ can be used for monitoring eye lens dose in the absence of $H_p(3)$ data.

An image intensifier based GE Advantx system was used for all IR procedures, while a flat panel based GE Innova 2000 performed all IC procedures. Over the six week period, a range of procedures were conducted in each laboratory, including; coronary angiograms, percutaneous transluminal coronary angioplasties, percutaneous coronary interventions etc, while in radiology, procedures such as peripherally inserted central catheterisations, venograms, hickman insertions, nephrostomies among others were conducted. For each procedure, the dose area product and total fluoroscopy time was recorded. By availing of the radiology and cardiology information systems, the total number of procedures per annum performed by each of the Interventionists can be determined and was used to estimate annual eye doses for the seven staff.

Available in both laboratories are a range of personal protective equipment including, Pb aprons, thyroid shields, ceiling suspended Pb glass shields (Cardiology only) and Pb glasses. Observations were made over the course of the study to determine compliance of staff with good radiation protection practice.

3. RESULTS

Table 1 presented for each staff member the total number of procedures conducted over the period of the survey, the total fluoroscopy time, associated DAP, measured eye radiation dose and estimated total eye dose per annum. The number of procedures over the six week period ranged from 16 to 23, giving rise to DAP values from 274.6 to 2041Gycm^2 respectively. Corresponding fluoroscopy times ranged from 38 to 224.5 mins. Measured eye doses arising from this number of procedures ranged from 0.74μSv to 31.7μSv per procedure. Extrapolation of this data over a 12 month period using data from Cardiology/Radiology information systems, results in estimated annual eye doses from 0.8 to 13.7mSv.

Table 2 summarises the findings of the survey to determine the compliance of staff members with the hospital policy on the wearing/use of personal protective equipment / apparel. Compliance is good on the wearing of Pb aprons, thyroid shields, whole body and eye dosimeters. However, only one of the three cardiology staff and one of the four radiology staff were wearing their Pb glasses while interventional procedures were in progress during the period of the study. Universal use is made of the ceiling suspended Pb glass shield in the cardiac catheterization laboratory while no such shield is available to staff in the interventional radiology laboratory.

TABLE 1. NUMBER OF PROCEDURES PERFORMED BY EACH STAFF MEMBER, TOGETHER WITH THE TOTAL FLUOROSCOPY TIME, DAP (Gy cm^2), MEASURED EYE DOSE (µSv), EYE DOSE PER PROCEDURE (µSv), AND ESTIMATED ANNUAL EYE DOSE (mSv).

Staff Member	Number of procedures performed	Fluoroscopy time (min.)	DAP ($Gycm^2$)	Eye Dose (µSv)	Eye dose per procedure (µSv)	Eye dose per year (mSv)
Cardiologist A	23	224.5	2041.6	410	17.8	13.7
Cardiologist B	27	202.9	1985.9	20	0.74	4.6
Cardiology Nurse	16	105.7	1043.8	210	13	41.6*
Interventional Radiologist A	14	50.6	347.6	120	8.6	1.0
Interventional Radiologist B	12	40.1	274.6	380	31.7	3.3
Interventional Radiologist C	16	38	876.3	90	5.6	0.8
Radiology Nurse	23	81.5	401.9	110	4.8	10.4*

*Eye dose per year for nurses is calculated as “worst case”, it is assumed the same nurse was in close proximity to the patient for each procedure in either the IR/IC laboratory

TABLE 2. A SURVEY OF STAFF COMPLIANCE WITH HOSPITAL POLICY ON THE WEARING, AND USE OF, PERSONAL PROTECTIVE EQUIPMENT/CLOTHING

Personal Protective Equipment	Interventional Radiology Laboratory	Cardiology Catheterisation Laboratory
Pb aprons	Yes	Yes
Thyroid shields	Yes	Yes
Pb glasses	No	One staff member compliant, remainder no.
Whole body radiation dosemeter	Yes	Yes
Eye dosemeter	Yes	Yes
Ceiling suspended Pb shield	No	Yes

4. DISCUSSION

Over the past number of years, the subject of interventional radiology/cardiology eye dosimetry has received considerable attention in the literature [1, 2, 5]. This issue will be continue to be debated over the coming years as all European countries enact the new BSS, and the revision downward of the eye dose limit, into national legislation. In Ireland, it is likely that occupational eye dose monitoring will be required if a radiation worker is likely to receive an eye dose greater than 15mSv. The short study outlined in this paper is an initial attempt to determine the nature and scale of the potential problem for our institution.

Consideration of the data presented in Tables 1 and 2 raises a number of interesting points. Cardiologists A and B performed a similar number of procedures over the course of the study and corresponding DAP and fluoroscopy time data are also similar. However their respective eye doses per procedure are 0.74 and 17.8μSv, and estimated annual eye doses are 4.6 and 13.7mSv respectively. This finding clearly demonstrates the importance of wearing Pb glasses during interventional procedures. The benefit of using the ceiling suspended Pb glass shield is also clear from the measured eye doses received by Cardiologist A (410 μSv) and Interventional Radiologist B (380 μSv). Both received similar eye doses, however a factor of 7.4 existed between their respective total DAP's. Correctly positioned ceiling suspended Pb glass shields must be made available in all laboratories where interventional procedures are conducted.

An initially surprising result was the eye doses received by both Cardiology and Radiology Nurses who are present during interventional procedures. Nurses' eye dose per procedure approached that received by Cardiologists/Radiologists. These Nurses stand near the patients head during the procedures and hence receive significant amounts of radiation scattered dose. Based on the annual number of procedures conducted in the IR and IC laboratories in 2013, a single Radiology/Cardiology Nurse could potentially receive radiation doses of 10.4 and 41.6mSv respectively to their eyes. To reduce this dose, it is very important that the Nurses closest to the patient during interventional procedures rotates this task with colleagues, and is advised strongly to wear the Pb glasses provided. Across all grades of staff in this study there is a large variation in the measured eye doses per procedure. This variation is a reflection of many factors which include the following; numbers of and types of procedures, complexity of procedure, operator technique, and wearing and use of PPE and apparel. Eyes doses measured in the study however agree in the main with other similar studies cited in the recent literature [5, 8].

A number of recent studies have suggested that a reasonably strong relationship exists between received eye dose and corresponding DAP data for the procedure [9]. Data in Table I does not support those findings. However, this may be explained by the factors mentioned above, i.e. procedure complexity, operator technique, wearing/use of PPE etc. Further work by this group will explore this point in more detail.

Finally, the estimated annual eye doses received by IR and IC radiation workers at our institution are clearly a matter of concern. Our findings suggest that some staff will reach over the course of a year the "classified" dose limit of 15mSv, and therefore will need to be monitored. Further risk assessments by medical physics staff will need to be undertaken in all areas including general radiology, operating theatres, endoscopy etc where fluoroscopy equipment is used to determine which staff will require eye dose monitoring when this legislation is implemented.

5. CONCLUSION

A six week investigation to measure the eye dose of a range of interventional radiation workers in our hospital has yielded important data which will be used by the hospitals scientific staff to provide advice on how to address the implementation of new legislation on eye dose monitoring. Based on the eye doses measured for both IC and IR workers, annual eye doses in our institution will for some workers approach the likely classification level of 15 mSv. Hence, eye dose monitoring shall be implemented across all laboratories for interventional radiologists and cardiologists. In addition, Nurses who work in close proximity to the patient during interventional procedures receive significant radiation doses to their eye and therefore will also require monitoring. Our findings also demand that the same Nurse is not standing beside the patient during the clinical procedures and therefore it is very important that this Nursing role is rotated among a number of staff.

Finally, in the case of the limited number of staff who participated in this study, very poor compliance with hospital policy was observed on wearing and use of PPE and apparel. This poor compliance was evident in the levels of radiation dose delivered to their eyes over the period of the investigation. This clearly highlights the need for medical physics and clinical specialist and radiography staff to provide adequate education and advice on the importance of employing available PPE as an aid to reduce personal radiation doses.

REFERENCES

[1] VANO E et al., Occupational radiation doses in 8527 interventional cardiology: a 15 year follow up, Br. J. Radiol. 79 (2006) 383-388.
[2] KIM KP et al, Occupational radiation doses to operators performing cardiac catheterization procedures, Health Phys. 94 (2008) 211-227.
[3] INTERNATIONAL COMMISSION RADIOLOGICAL PROTECTION, Statement on tissue reactions. ICRP 4825-3093-1464, (2011).
[4] EUROPEAN COUNCIL DIRECTIVE 2013/59 EURATOM of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionizing radiation, and repealing Directives 89/618, 90/641, 96/29, 97/43 and 2003/122 Euratom. Official J European Union, 57, 17 (Jan, 2014).
[5] MARTIN CJ., Personal dosimetry for interventional operators: when and how should monitoring be done? Br. J. Radiol., 84 (2011) 639-648.
[6] NANODOT, see http://www.landauer.com/uploadedFiles/InLight_nanoDot_FN.pdf
[7] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION. ICRP publication No 103. Ann ICRP 37:1-332, (2007).
[8] LIE OO., et al., Assessment of effective dose and dose to the lens of the eye for the interventional cardiologist, Radiat. Prot. Dosim., 132 (3) (2008) 313-318.
[9] SCHUELER BA et al., An investigation of operator exposure in interventional radiology, Radiographics, 26 (2006) 1533-1541.

EVALUATION OF RADIATION RISK COMING FROM DENTAL RADIOGRAPHY, SOME RECOMMENDATION ISSUED FOR OCCUPATIONAL STAFF IN DENTAL PRACTICES IN ALBANIA

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Abstract

Diagnostic medical and dental radiography comprise 82% of all man-made radiation exposure of population in Albania. The monitoring of occupational and patients exposures is obligatory for all licensed radiation users category A and B of workers. A rough estimation has shown about 800 radiation workers in public, private institution in Albania. The majority of these workers are engaged in medicine by X-ray devices for examinations. Dental radiography does not make a major contribution to annual effective dose for occupational staff of dental clinics in Albania. The annual effective dose of an exposed worker in dentistry may not exceed l mSv. The mean annual effective dose of occupational staff in dentistry Clinics in Albania has been evaluated to be about 0.8-1.0 mSv for the last five successive 2009-2013 years [1]. Some recommendations are given for patient and occupational staff in diagnostic and dental radiography for dose rate reduction. The Radiation Act No. 8025 dated 09.11.1995 and Code of Practices for Radiation Protection by Irradiation in Rontgen - Diagnostic stipulate that the undertaking shall arrange monitoring of radiation exposure for exposed workers in X-ray diagnostics applications, including dentistry.

1. INTRODUCTION

1.1. Involvement of Radiation Protection Legislation for Radiology Examinations

Radiation safety in radiology examinations, as well as in dental radiology is regulated through the Ionizing Radiation Regulation. The Radiation Act No. 8025 dated 09.11.1995 stipulates that the undertaking shall arrange monitoring of radiation exposure for exposed workers in X-ray diagnostics applications by X-ray devices, as well in dentistry examination. The former regulations has had included all radiation protection matters that apply to occupational exposure and protection of the public, while the latter apply exclusively to radiation protection of the patient including the training of staff exposing patients at ionizing radiation fields.

In addition to the Regulations, there was approved Code of Practice in Roentgen-diagnostic, which supports the Radiation Act 8025, and there are various forms of guidance, which provide additional documentation including practical methods of implementation. The guidance documents of principal relevance to dental radiology are the Guidance Notes for the protection of persons against Ionizing Radiation arising from medical and dental use. All radiation protection legislation provides a framework intended to ensure that doses to the staff, patients and visitors are as low as reasonably practicable (ALARP).

In these Regulations and Code of Practice including dental radiography the Albanian Radiation Protection Commission - Board (ARPC) has foreseen and issued a special statement on diagnostic medical exposures during pregnancy time to the female patients. From this statement it can be concluded that the normal selection

criteria for dental radiography do not need to be influenced by the possibility of the female patients at any stage of a pregnancy. The statements are foreseen for adults and kid less than 18 years old.

2. RADIATION RISK FROM RADIOGRAPHY EXAMINATIONS

2.1. Health effects of radiation

The effects of exposure to low levels of irradiation may be the appearance, at long times after exposure of a small excess of cancers in any irradiated population and, one or more generations. Later, of a small excess of hereditary disorders. The more important for setting protection standards are the risk of radiation inducted cancer. It is not possible, at present, to distinguish a radiation-induced cancer from one arising from other causes, so that any estimate of the risk has to be made from statistical analysis of the long-term health of irradiated populations. Theoretical consideration and experimental results have to be considered in order to estimate risk at low doses and low doses rates [2, 3].

2.2. Patient dose reduction in diagnostic radiography

Diagnostic medical and dental radiography comprise 82% of all man-made radiation exposure of the population in Albania. The need to reduce patient doses depends on the level risk, to both populations and individuals, associated with the X-ray examinations. Ultimately in a resource-limited health service, the need will be met only if the methods and benefits of dose reduction can complete cost effectively with other forms of health care. Patient dose reduction is achieved by adherence to the following general principle.

(a) All diagnostic practices should be justified at a broad level and the expected clinical benefits demonstrated to be sufficient to offset the radiation detriment.
(b) There should be a valid clinical indication for all medical exposures at the level of individual procedures.
(c) There should be a commitment to optimization of radiological protection at the levels.

In this context, optimization of radiological protection means that patient doses should be kept as low as reasonably practicable (ALARP), consistent with achieving diagnostic objectives.

3. METHODS

3.1. Monitoring of occupational staff exposures in dental radiography

The monitoring of occupational exposures is obligatory for all licensed radiation users. A rough estimation has shown that there are around 800 radiation workers in public and private institutions and organizations in Albania and majority of them are engaged in medicine - X-ray, brachytherapy and tele-therapy examinations. Others are in research, industry and agriculture activities. Therefore, the monitoring of occupational exposures is implemented for the radiation workers of the principal cities and their number for the moment is around 500.

The official dosimeters used for occupational exposures monitoring for moment

are TLD-110 cards supplied from Thermo-Fisher Company. The dosimeters are distributed bimonthly basis and their evaluation is carried out through standard procedures. The superficial dose H_p (0,7) and depth dose H_p (10) are evaluated and registered in fundamental register in IANP. The process of dosimeters calibration is performed through known radiation field of radioactive standard sources ^{137}Cs and X-ray beam machine of secondary calibrating laboratory.

4. RESULTS

The Table 1 shows that for the majority of radiation workers annual doses are below 5 mSv. Concerning the dose above investigation level the follow-up studies showed they are related with intervention radiology and brachytherapy and for doses above 20 mSv the dosimeters are irradiated in primary beam, it was false information and the workers were penalized by RPO staff. The mean annual effective dose of occupational staff in dentistry Clinics of Albania has been about 0.8 -1.0 mSv, for five last successive (2009-2013) years. Table 1 shows the results of evaluation of the annual effective dose in Dentistry Clinics [1].

TABLE 1. ANNUAL DOSE RANGE DISTRIBUTION DURING 2009 – 2013 (FIRST TRIMESTER)

E, mSv	2009	Number 2010	of 2011	Personnel 2012	(I^{st}-trimester) 2013
E < 5	372	380	400	403	395
5 < E < 10	4	3	12	5	11
10 < E < 20	-	3	3	5	7
E > 20	7	1	4	2	5
Total	383	387	419	415	418

TABLE 2. ANNUAL EFFECTIVE DOSE IN DENTISTRY CLINICS OF TIRANA CITY DURING 2009-2013

Years	2009	2010	2011	2012	2013 first semester
Mean annual dose (mSv)	0, 96	1,00	0, 84	0, 93	1,04

5. DISCUSSIONS

5.1. Recommendations to improve the standards of dental radiography

Diagnostic interpretation is the final stage of the radiographic process, the outcome of which must be of benefit to the patient. It involves the ability to make a valid diagnostic judgment from the image on the radiograph. Given a good quality radiograph with adequate viewing facilities, the accuracy and validity of the diagnostic interpretation will depend on an effective combination of basic education, training and experience. In this context, the Albanian legislation, regulations and Code of Practices in Radiography examination-diagnostic recommend:

An attention should be paid to improving the standards of diagnostic interpretation and in particular that a mechanism for the clinical audit and peer review of written

radiographic reports.

In general, the standards for equipment used in dental radiology are well established and, the minimum potential recommended for dental radiography is 50 kV. As the value of operating potential is increased, the patient entrance dose required to produce an acceptable radiographic density is reduced, but at the same time the radiographic contrast decreased [3]. So, for these purposes need to be recommended:

X-ray equipment for intra-oral dental radiography should operate at a potential not less than
50 kV, and for new equipment (panoramic device) a range 60 - 80 kV, and that it should operate within 5 kV, of the indicated value for different procedures of examinations.

A range of exposure times must be available that allows for the correct exposure of the chosen dental film for all radiographic subjects, taking into account such factors as patient movement and tube filament warm-up time. The Table 3 shows the requirements that may best be met to the following optimum output rates for self-rectified tube heads at a focus to skin distance (fsd) of 200 mm.

TABLE 3. NOMINAL X-RAY OUTPUT RATES AT CONE TIP
(200 mm FSD IN MGYS-1)

Operating Potential (kV)	Nominal X-ray output rate ($mGys^{-1}$)
50	6
60	4
80	3

The exposure time required for a dental radiograph should be chosen in relation to the following parameters: 1) operating potential; 2) total beam filtration; 3) focus to skin distance; 4) film speed group; 5) patient type; 6) radiograph view. From these considerations, the specialists of dental applications have recommended that on new equipment, there should be a range of X-ray output and exposure time settings are available such that dental films of speed group D and faster can correctly and consistently exposed. A rectangular collimation to be adopted for bitewing and periapical radiography [2, 3]. To take advantage of the increased sensitivity of rare-earth intensifying screens, equipment should have provision for the selection of a range of tube currents to a minimum value of 4 mA. Many older sets require modification to allow the selection of the lower output required for these faster film/screen combinations. If the tube current can't be adjusted, it may be possible to achieve a suitable reduced output by the addition of extra beam filtration or by reducing the beam width.

The trend towards narrower beams and shorter cycle times demands the provision of tube-heads having an effectively constant potential (DC) output to eliminate the effects of mains frequently banding on radiographs.

Concerning with above-mentioned problem, the specialists have recommended that new panoramic X-ray equipment should incorporate adequate provision for varying radiograph exposure and advises the choice of tube-heads with an effectively constant potential (DC) output.

There is no mandatory requirement for the routine use of lead aprons for patients in dental radiography. Lead aprons do not protect against radiation scattered internally within the body and only provide a practicable degree of protection in the case of the infrequently used vertex occlusal projection. In the ease of panoramic radiography, a lead apron may physically interfere with the procedure and can often be detrimental to

the diagnostic image. In the few cases where the thyroid gland is in the primary beam there is value in providing protection in the form of a thyroid collar, if one is available [4, 5].

There is no justification for the routine use of lead aprons for patients in dental radiography. Their use during panoramic radiography is positively discouraged. The introduction of rectangular collimation and paralleling techniques will make thyroid shielding unnecessary.

For purpose of QA- quality assurance, in dental radiography is to ensure consistently adequate diagnostic information, whilst radiation doses to patients and staff are controlled to be as low as reasonably practicable (ALARP). The QA program will need to take account of relevant statutory requirements and this will determine many of the operational objectives.

6. CONCLUSIONS

Based on these guidelines, and recommendations issued by specialists of dentistry application made to radiology standards in primary dental care, we conclude that:

(1) Individual doses, and therefore individual risks, from dental radiology are low, especially in comparison with those from most other forms of diagnostic radiology.
(2) There is the potential for significant reduction in the collective dose to patients, and consequently for a significant benefit to society as a whole.
(3) There is scope for a marked improvement in the diagnostic standards of the radiology, including the dental devices entered and used during five last years in Albania.

REFERENCES

[1] FUNDAMENTAL REGISTRY (INP) to dose rate of occupational staff during 2009-2013 years,
Tirana, Albania.
[2] HMSO, SI, The Ionizing radiation protection of persons undergoes medical examination or Treatment, (1988) 778.
[3] HEWITT, J. M., Monitoring radiation protection in dental practice, Radiology Protection Bulletin,
No. 151 (1994) 13-16.
[4] SMITH, N J., Assessment of the quality of panoramic radiographs taken in general dental practice, Journal Dental Resume No. 72 (1993) 204-210.
[5] STRENSTROM, B., Radiation shielding in dental radiography, Sweden. Dental Journal, No. 7 (1983)
85-91.

QUALITATIVE AND QUANTITATIVE ANALYSIS FOR OCCUPATIONAL RADIATION DOSES IN DIFFERENT MEDICAL PRACTICES AT HAMAD MEDICAL CORPORATION IN QATAR

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Abstract

In this work, the occupational dose of Radiation Workers (RW) in Hamad Medical Corporation (HMC) Hospitals in State of Qatar for different departments such as Radiology, Oncology, Nuclear Medicine, Cath-Lab, and Urology and Operational titer are evaluated. Bi-monthly dose measurements were regularly carried out for a period of 5 years (2009 to 2013). Out of a total of around 1000 medical radiation workers, 755 are selected for this study. The maximum annual effective dose for all monitored workers was 5.79 mSv per year.

1. INTRODUCTION

The occupational dose monitoring program is considered the essential indicator to ensure the efficiency of radiation safety condition in all applications under any exposure levels (routine dose level up to emergency and incident level). In spite of the complexities of such program especially for that organizations which working on national level and covering all of medical exposure types IAEA regulations and Qatar radiation protection law engage licensees to provide RW with the individual dose monitor which require a certified lab with qualified staff and high running cost [1, 2].

This study aims to assess the RW annual effective dose over the last five years in different medical practices within HMC hospitals, also to compare dose levels with limits permitted locally and internationally. In addition, the study is to verify the efficiency of radiation protection conditions and requirements at HMC, aiming to reduce Occupational Radiation Doses (ROD) on individual and institutional level.

Analysis of occupational doses is an important component of institutional radiation protection programs. Appropriation of radiation protection resources should take into account dose variation among various occupational groups. Highly exposed groups should be appropriated more resources in terms of training, provision of protective devices and implementation of dose reduction schemes. Trends in the mean annual dose, collective dose and dose distribution can be used as indicators of good institutional radiation safety practices. In the safety optimization process, dose investigation levels are set by institutional management for early detection of any conditions that might lead to deterioration in radiation safety practices [3, 4].

2. MATERIAL AND METHODS

Eight hospitals, belongs HMC were included in this study. As shown in Table 1, all RW provided with individual TLD. According to IAEA guidelines, the dosimeter should be worn

under the lead apron (at waist level) for estimating the effective dose. Personnel dose equivalents Hp (10) and Hp (0.07) were evaluated by employing one individual dosimeter to be carried continuously by the occupational exposed person while at work.

TLDs readings are available online in HMC intranet for each individual and department. Annual dose records of RW from 2009-2013 were taken from the HIS (Health Information System) and exported to Excel 2010 (Microsoft Corp., Redmond, WA, USA) for analysis. The numbers of monitored and measurably exposed workers for each category are presented in Fig. 1.

3. RESULTS AND DISCUSSION

Fig. 1 illustrate the percentage distribution of the radiation workers, it is noticed the increase of physicians number in all hospitals, on the contrary the lowest number for other category (bio-engineering, radiology office staff and aids).

As Shown in Table 1 and Fig. 2, the number of TLD monitored workers increased from 417 in the year 2009 to 775 in 2013 approximately 86% from the initial number; this is due to the opening of three new hospitals during the last three years. Table 1 indicates that, the total number of monitored workers in HGH is grater than the other hospital because it is the biggest hospital and this is the main centre in HMC and in Doha the capital of Qatar. The slightly reduction for HGH monitored RW in 2012 due to opening new hospitals.

TABLE 1. NUMBER OF WORKERS DURING THE LAST FIVE YEARS IN HMC HOSPITALS HGH, AKH, NCCCR, RH, WH, HH, AWH and CH.

Hospital /year	2009	2010	2011	2012	2013
HGH	299	319	362	391	337
AKH	51	51	54	57	57
NCCCR	47	47	52	73	105
RH	13	16	14	24	27
WH	7	8	11	13	14
HH			66	135	147
AWH			7	26	41
CH			19	23	27
HMC	417.0	441.0	585.0	742.0	755.0

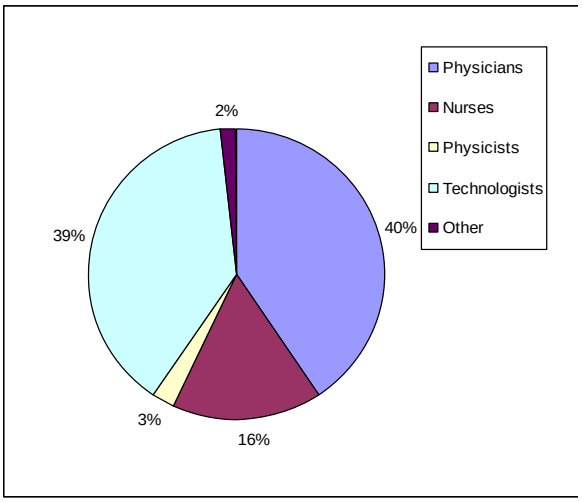


FIG. 1. The percentage distribution of the RW numbers during the period 2009 to 2013 in

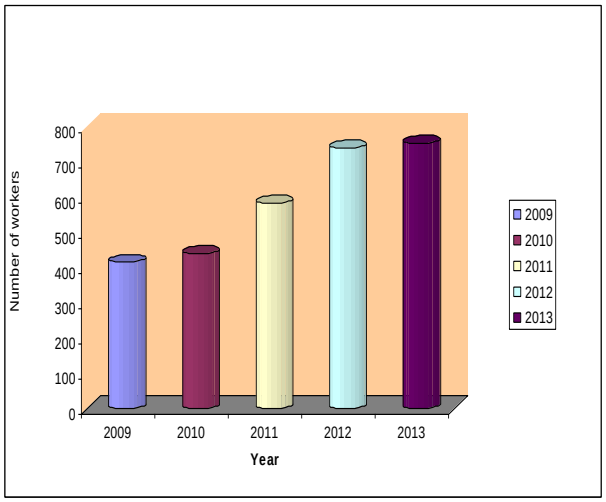
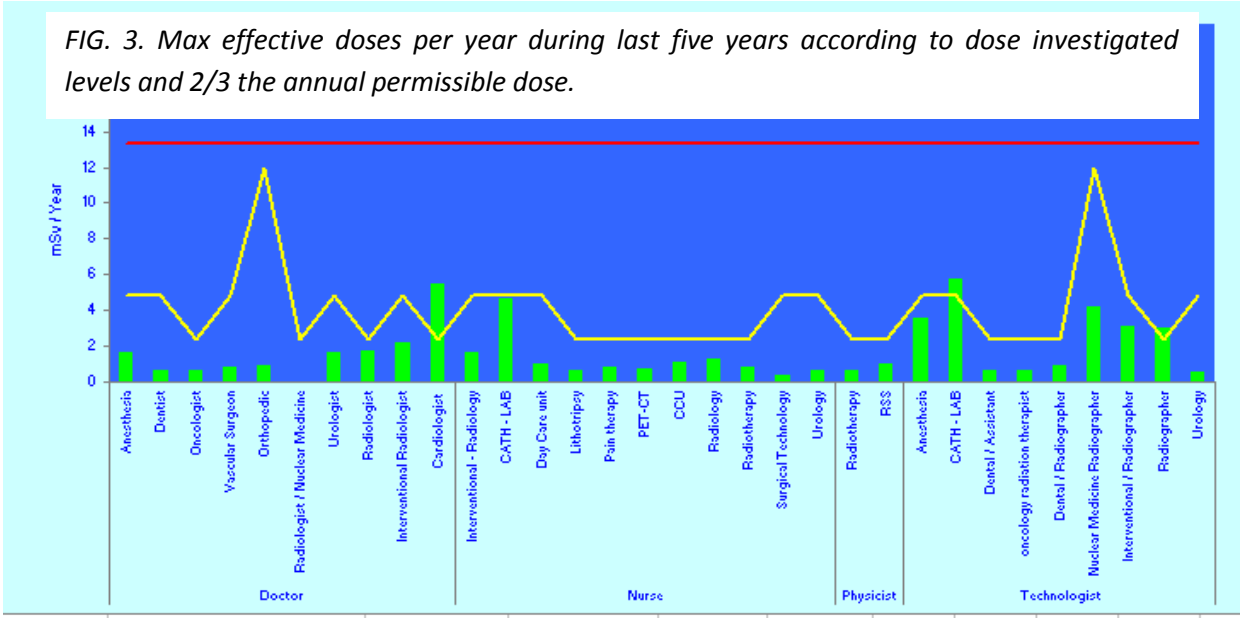


FIG. 2. Total number of monitored worker during the last five years 2009 to 2013

Fig. 3 indicates the maximum annual dose for each category compared with the Investigation level per year, and 2/3 maximum permissible dose (MPD), The current occupational dose limits is 20 msv per year averaged over defined periods of 5 years and the investigation level as 12.0, 2.4 and 4.8 per year for nuclear medicine, general radiology and interventional respectively. The highest annual individual dose has been recorded during this 5 years was 5.79 msv received by Cath-Lab technologist and 5.46 mSv received by interventional cardiology while 3.01 mSv received by general x-ray technologist. For all the categories monitored, the doses were well below the internationally recommended limit of 20 mSv per year. 99% of annual dose below 5 mSv, greater than that reported for Portugal (1986-1988), which showed that 97.8% [5], that demonstrate the adequacy of structural radiation shielding in these facilities.

The maximum annual occupational dose for doctors category was 5.46 mSv for interventional cardiologist and 0.61 mSv for dentist and for nurse category was 4.68mSv cath-lab nurse and 0.37 mSv for surgical nurse while for technologist category was 5.79 mSv for cath-lab and 0.56 mSv for urology. Therefore the personnel monitoring effort make radiation worker more aware, and led to improve their radiation protection practices. The maximum annual effective dose was registered for the cath-lab individuals, doctors' technologist and nurse because of the increasing the use of ionizing radiation in PCI procedures.

FIG. 3. Max effective doses per year during last five years according to dose investigated levels and 2/3 the annual permissible dose.



4. CONCLUSION

Analysis of occupational doses is an important component of institutional radiation protection programs. The study found that the maximum annual effective dose 5.79 mSv, and for all the categories monitored, the doses were well below the internationally recommended limit of 20 mSv per year. The low occupational doses also demonstrate the adequacy of structural radiation shielding in these categories. The maximum annual effective dose was registered for the cath-lab RW, doctors' technologist and nurse because of the increasing number of procedures per year.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety of radiation Sources: International Basic Safety Standards, Interim Edition, General Safety Requirements, Vienna (2011).

[2] DECREE-LAW NO.31 OF (2002), Protection from Radiation, Qatar.

[3] CALDER J., The history of radiology in Scotland 1896-2000., Edinburgh, Dunedin Academic Press, (2001).

[4] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, General principles for the protection of workers, ICRP Publication 75, Pergamon Press, Oxford (1997).

[5] KORIR,G.K., WAMBANI,LS., KORIR,I. K., Estimation of annual occupational effective doses from external ionizing radiation at medical institutions in Kenya, S.A. Journal of radiology (2011) 116-119.

RADIATION PROTECTION OF STAFF IN PET CENTRE ACTIVITIES – HOW TO CHANGE FROM 'YES BUT' TO 'WILL DO' IN STAFF RADIATION PROTECTION

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Abstract

The radiation protection of staff is of concern when PET (positron emission tomography) imaging is applied. Several sites of handling radionuclides during synthesis of radiopharmaceuticals, preparation of vials and syringes, injecting patients and setting them up for imaging require awareness, protective actions and monitoring to minimize staff exposure. This project was designed to collect accurate data on doses of the staff in a PETcentre, to identify sites in work processes where additional protective actions or changes of work habits are needed. The staff members of a major PET centre in Finland are involved in the synthesis of ^{18}F, ^{11}C and ^{68}Ga labelled tracemolecules, preparation of syringes injecting and setting up the patients. Within this project they are monitored for eye lens, finger and whole body exposure more thoroughly than with routine radiation dose monitoring, The dosimeters are prepared, calibrated and evaluated at the Federal Office for Radiation Protection (BfS), Germany.Work task and site specific data will be collected for analyses of staff exposure. Staff education and training will be tailored on the basis of this information to improve the radiationprotection and minimize the exposure as necessary.

1. INTRODUCTION

The radiation protection of staff is of concern when PET (positron emission tomography) imaging is applied. Several sites of handling radioactive isotopes, preparation of vials and syringes, injecting patients and setting them up for imaging require awareness, require actions and monitoring to minimize staff exposure.

ORAMED (Optimization of radiation protection of medical staff) was a collaborative project funded in 2008 within the 7th EU Framework programme, EURATOM Programme for nuclear research and training. The project revealed that the annual limit of the local skin dose on hands is often exceeded in nuclear medicine. The actual skin dose may remain underestimated because the measured radiation exposure to hands of the staff depends on the placement and characteristics of the ring dosimeter.

European Commission has given a new limit for eye lens dose, which was lowered from150 mSv/year to 20 mSv/year. The eye lens dose data in PET workplaces is rarely monitored. Thus, more information is needed on management and optimization of radiation protection in PET environment.

The regulations concerning radiation protection of staff apply ALARA principle, and strict respect to as low as possible exposure within international safety standard requirements. This study will provide data which may be applicable to improve the protection of the staff in a PET centre. The study plan has been approved by the Turku University Hospital ethical committee. Written consent is obtained from staff to participate in the study.

2. AIMS OF THE STUDY

The aims of the study were

(a) To collect accurate data of the whole body, eye lens and skin dose during different actions in a PET-centre,
(b) To identify the steps in the processes where additional radiation protection actions or change of work habits are needed, and
(c) To use the data received in education and training in radiation protection.

3. METHODS

The staff members (class A workers) of the PET centre at Turku University synthesising the ^{18}F, ^{11}C and ^{68}Ga labelled tracers, preparing the syringes, injecting and setting up the patients for imaging are being monitored with the following dosimeters during three measuring periods:

I. Electronic personal dosimeters EPD Mk2^{+} (Thermo scientific) to wear on the trunk for measurement of the personal deep dose equivalent $H_p(10)$ to estimate the whole body dose,
II. New eye lens dosimeters Eye-D (RadPro) to detect $H_p(3)$ on the forehead to estimate the eye lens dose,
III. Thin-layer thermoluminescent dosimeters (TLD) of the type MCP Ns to detect the personal surface dose equivalent $H_p(0.07)$ on fingertips to estimate the maximal local skin dose of the hands and
IV. Routine ring dosimeters will be worn on both hands (index finger)

Each measuring period will take four weeks, in April 2014, September 2014 and February 2015, except the finger tips dose measurements which last only one week per period. The dosimeters will be prepared, calibrated and evaluated by the staff members of the Federal Office for Radiation Protection (BfS)

3.1 Estimation of the whole body and the eye lens dose

These measurements were not individual measurements but workplace/work activity/performance/site related. The three different work sites selected for the measurements were synthesising the radiopharmaceuticals, preparing the syringes and injecting patients and setting them up for imaging. This means that different persons may wear the same dosimeters during one working day at the certain workplace.

The background radiation level including that from the mailing is detected by means of additional dosimeters, which are kept at a place with "a normal background radiation". (e.g. in the office of the radiation protection officer). The reading of these background dosimeters has to be subtracted from gross reading of the personal dosimeters.

The EPD at the trunk and the eye lens dosimeter at the forehead or on the safety goggles must be worn simultaneously. There are three separate protocols for the three workplaces. The following data must be recorded for each workplace:

(a) Date, abbreviation of the name of the staff member
(b) Usage of shielding (Yes or No)
(c) Description of shielding: thickness and material
(d) Number of syringes prepared or patients examined
(e) Sum of the activity, separated for the radionuclides (e.g. ^{11}C, ^{18}F and ^{68}Ga)
(f) Reading of the EPD at the beginning and at the end of the action, every day
(g) Remarks (e.g. contaminations, intensive care for a patient, actions without shielding, amount of synthesis made)

3.2 Estimation of the maximal local skin dose

To measure the maximum local skin dose on hands, the tapes with thin-layer TLD are fixed at the inner side of the tips of the thumb, index and middle finger of both hands. In previous studies these fingers and positions were identified to receive the highest dose due to the close contact to vessels and syringes containing activity.

Moreover, the authorized ring dosimeters worn on the usual spot and two additional authorized ring-dosimeters were worn on the base of both index fingers simultaneously because the monitoring period does not agree with the usual measuring period. These dosimeters are evaluated by the Finnish dosimetry service (Doseco oy). The measurements are repeated twice after a break of 3–4 months.

4. RESULTS

It is of international interest to accomplish data on the exposure of the eye lens in processing and using radiopharmaceuticals because of the current ICRP recommendation on reduction of the limit of the eye lens dose. Data to clarify the degree of eye lens protection from using protective eye wear during synthesis, preparation and administration activities of staff in the PET environment will be obtained by November 2014 and can be presented in the meeting. The degree of protection with VWR-standard protective glasses is verified during synthesis of the radiopharmaceuticals. Exposure to both hands and different fingers will be clarified in relation to the activities used in syntheses of different radiopharmaceuticals, preparation of the syringes and administration of injections.

In the first 4 weeks' measurements, exposure differences related to different tasks and differences in the individual exposure level among workers on the same tasks were observed. The second measurement period is ongoing in September-October 2014 and the repeated results will be presented in the poster.

Even though similar protective eyewear was used, we observed substantially different attenuation with similar protective glasses during ^{18}F and ^{11}C synthesis (12% vs. 21%). This needs confirmation from the new measurements.

Substantial differences observed in finger doses although similar levels of activities were handled, indicate importance of assessment of individual working habits. The processes need to be discussed and a protocol for compliance to be developed in the educational session on 24th November.

5. CONCLUSION

The results indicate those sites and steps in the processes, which will need to be targeted for improving radiation protection measures like using new protective equipment e.g. automatic dispensing of radiopharmaceuticals. The collected exposure data and knowledge gained on related work performance will be applied in optimization of the radiation protection of the staff through education and training session based on the above assessments of radiation exposure at PET workplaces. The study will enlighten the question, whether a routine eye lens dose monitoring is necessary in various work steps the PET environment.

REFERENCES

[1] BEHRENS, R. Dose conversion coefficients for electron exposure of the human eye lens: Calculations including a whole body phantom, Radiation Protection Dosimetry, Vol. 155, No. 2 (2013) 224–235.

[2] CARNICER A, SANS-MERCE M, BAECHLER S, BARTH I, DONADILLE L, FERRARI P, FULOP M, GINJAUME M, GUALDRINI G, KRIM S, MARIOTTI M, ORTEGA X, RIMPLER A, RUIZ N, VANHAVERE F, Radia. Meas., Hand exposure in diagnostic nuclear medicine with ^{18}F- and ^{99m}Tc-labelled radiopharmaceuticals, Results of the ORAMED project, 46 (2011)1277-1283.

[3] COPEC, R., BUDZANOWSKI, M., BUDZYNSKA, A .et al., On the relationship between whole body, extremity and eye lens doses for medical staff in the preparation and application of radiopharmaceuticals in nuclear medicine, Radiat. Meas. 46 (2011) 1295-1298.

[4] COVENS P, BERUS D, VANHAVERE F, CAVELIERS V., The introduction of automatic dispensing and injection during PET procedures: A step in the optimisation of extremity doses and whole body doses of nuclear medicine staff, Radiat. Prot. Dosim.,140 (2010) 250-258.

[5] EC DIRECTIVE 2013/59/EURATOM, Basic Safety Standard (BSS), EURADOS Report 2012-02:ORAMED: Optimization of radiation protection of medical staff, ISBN 978-3-943701-01-2, Braunschweig (2012).

[6] GINJAUME,M., PEREZ, S., DUCH, MA, ORTEGA, X., Comparison of TLD-100 and MCP-Ns as extremity dosemeter for PET nuclear medicine staff, 15th International Conference on Solid State Dosimetry, July 8-13 (2007).

[7] LEIDE- SVEGBORN, S., Radiation exposure of patients and personnel from a PET/CT procedure with ^{18}F-FDG, Radiat. Prot. Dosim. 139 (2010) 208-213.

[8] MATTSSON, S:, SÖDERBERG, M., Radiation dose management in CT, SPECT/CT and PET/C Techniques, Radiat. Prot. Dosim. 147 (2011) 13-21.

[9] SEIERSTAD, T., STRANDEN, E., BJERING, K. et al., Doses to nuclear technicians in a dedicted PET/CT centre utilising ^{18}F fluorodeoxyglucose (FDG), Radiat. Prot. Dosim. 123 (2007) 246-249.

OPERATIONAL IMPROVEMENTS TO ENHANCE OCCUPATIONAL RADIATION PROTECTION FOR AN I-131 ABLATION TREATMENT SERVICE OVER A DECADE OF CLINICAL USE AT CORK UNIVERSITY HOSPITAL

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Abstract

Iodine ablation treatments in support of thyroid cancer treatment have been delivered at Cork University Hospital (CUH) for many years. A program of operational improvements has been progressed following a minor occupational exposure incident a decade ago. A significant increase in clinical demand for these treatments brought new problems and additional focus to the program. This work describes briefly the nature of the original incident and the practical steps taken to improve the institution's policies, procedures and infrastructure within a busy but cost constrained environment. The principal aims of this work were to reduce the risk of an occupational exposure incident occurring and to improve the quality and quantity of data acquired in support of the service to better ensure adherence to national legislation and also licence conditions laid down by the national regulator. Improvements in the overall efficiency of the service and ability to provide detailed information to the national regulator as part of its program of periodic inspections were realised.

1. INTRODUCTION

A facility to provide iodine ablation therapy has existed at CUH for many years. The room consists of a standard double bedded hospital ward with a suite which has been converted to a single bedded unit. The room used was originally constructed and shielded for Low Dose Rate brachytherapy patients treated using manually afterloaded ^{137}Cs source trains. The bulk of the shielding is 20mm Pb and up to 380mm of mass concrete. However, these treatments ceased over a decade ago and the room is now used exclusively for thyroid cancer treatments. The entrance to the room comprises a short corridor with a door at either end. A transfer hatch is provided on the inner door for communication with the patient and promotes good practice with respect to the ALARA principle. Neither door is shielded; rather, protection relies on sufficient distance from the ward corridor to reduce the radiation exposure.

An event in 2005 triggered a series of steps to address perceived deficiencies in the infrastructure and procedures used to deliver ^{131}I therapy for thyroid cancer. Contamination was arising as a result of sub-optimal plumbing for the treatment room which resulted in contaminated sewage reflux into the en-suite facility for the room. Moreover, blockages were forcing open a manhole cover external to the room and leading to areas of ground contamination. After several such episodes, an employee from the hospital's Maintenance Department accessed the system without following written procedures and was subject to a low (<10 μSv) but unnecessary exposure. Regardless of its minor nature, the incident served as a catalyst to address the need for infrastructure improvements and to undertake a thorough review of procedures to limit the possibility of further and potentially more serious incidents. In addition, work was carried out to better understand how the treatment facilities were being used.

2. METHODS

Following the occupational exposure event the entire approach to delivering these treatments was assessed since it was clear the same problems could recur and possibly with a less benign outcome. The immediate problem required remedial work to be carried out on the plumbing infrastructure servicing the treatment room. Despite careful pre-treatment induction of patients, inappropropriate materials were flushed down the toilet e.g. bandages, paper handtowels. These materials do not degrade as readily as toilet tissue and caused blockages in our facility with consequent backflow contamination reaching the en suite shower tray as a consequence of the shower and toilet waste pipes merging to a common outflow. The cost of significant changes to the plumbing system was prohibitive and required an alternative to be found.

Although written procedures existed for work involving ^{131}I, they were limited in scope and communication of them to relevant stakeholders was sub-optimal. Consequently the review involved all procedures being re-written, extended and communicated effectively to all stakeholders. Where required, new procedures were introduced.

In our facility a new Holding Tank was installed about 12 years ago to contain waste from ablation treatments. It has an internal capacity of 1000 litres and is constructed from 300mm thick precast concrete below ground level. The tank is controlled by three valves: a pair of opposing valves for tank use and bypass into the main sewer and one valve to empty the tank into the main sewer. It was not known how much water was used by patients undergoing treatment, so flow meters were deployed to make an assessment. The meters were installed to ensure that only relevant pipe flows were considered and calibrated to ensure that the volume of water in the holding tank was accurately determined.

Other waste associated with each ablation patient consists of general domestic waste (e.g. contaminated food waste, disposable ware, patient clothing), bed linen/towels and occasionally small amounts of clinical waste. Given the half-life of ^{131}I, contaminated waste has to be stored for a significant amount of time before it decays to background, which puts pressure on available storage space. The store room used formerly housed facilities for the preparation of ^{137}Cs Brachytherapy treatments and provides more than adequate shielding for the waste. The main issue with the storage of general domestic waste was related to food waste and proximity of the store room to a heavily used corridor. Despite multiple bagging of this waste, the odour of rotting food presented a significant problem. Various options for dealing with food waste were considered. These included use of a macerator, use of a well-ventilated outside storage facility that is also shielded and safe against vermin, use of fridge freezer units to freeze food waste, or use of air tight containers.

The number of patients treated per year at the hospital was in the range 5 to 10 until about 2006. In following years there was a rapid increase which appears to have reached a plateau in the range 40 to 50 patients per year. These figures include both first time and follow-up ablation treatments. The marked increase in the number of patients created additional contamination problems that required careful consideration.

3. RESULTS & DISCUSSION

A cost effective solution to the plumbing problems involved the introduction of an ‘inspection chamber’ outside the treatment room building. The inspection chamber incorporates a polymethylmethacrylate (PMMA) window under a manhole cover which allows a quick visual assessment of the condition of the system channelling the radioactive waste to be made. A mandatory quality control check was introduced prior to every patient commencing treatment which involves observing the flow at the inspection chamber to ensure

there are no blockages in the sewage system leading to the holding tank. The PMMA window removes any risk of splashing that may occur and staff are instructed not to remove it without appropriate personal protective equipment and a Medical Physics presence. During these works the opportunity was also taken to separate the flow from the shower and toilet in order to eliminate the risk of further instances of backflow of effluent into the shower tray. Waste water from the shower is now diverted directly to the main sewer and does not contribute to flow meter measurements. All access ports to the sewage system for the treatment room now also incorporate radiation warning signs and alert staff to the possibility of contamination. Staff from the hospital's Maintenance Department are always supervised by someone from the Medical Physics Department when carrying out this work.

In years when low numbers of patients were being treated, clinical indicators combined with bed management issues had resulted in longer patient admissions (usually 6 nights). Some confusion among staff groups developed as a result of a perceived link between this length of stay and radiation protection concerns related to the higher activity of administered ^{131}I for these patients. Following the conclusion of treatments, the room could also be used by other patients assuming satisfactory contamination levels applied. However, with increased use of the room for ablation treatments, the efficiency of bed use became a more important consideration as did more pro-active decontamination procedures. Consequently, patients are now admitted for an average stay of 65 hours.

Ambient dose rates at 1 metre are measured before discharging a patient. It is expected that dose rates at discharge will not exceed $40\mu Svh^{-1}$. However, given the current average patient stay, dose rates are normally comfortably within 25 μSvh^{-1}. A dose rate in the range 25 to 40 μSvh^{-1} corresponds to a residual activity level of approximately 600 MBq [1]. The time at which a patient is ready for treatment coupled with our hospital's normal discharge time (mid-morning) also exerts some influence over the length of stay, since in theory many patients could be discharged after 48 hours with appropriate instructions to follow for defined periods [2]. Acquisition of dose rate measurements at one metre was simplified by incorporating a suitable floor mark one metre from the inner door hatch during a room maintenance project which involved resurfacing the floor. Also, with increased use of the room for ablation treatments, it was decided to designate the immediate treatment room a radiation controlled area at all times whether a patient was present or not. The hospital's licence conditions require designation of an area as controlled if contamination levels exceed 40 $Bqcm^{-2}$ and supervised if levels exceed 4 $Bqcm^{-2}$. Contamination levels rarely decrease sufficiently to allow designation of the room to be completely relaxed before the next ablation patient requires treatment. The link corridor designation is relaxed to a public area when no patient is present provided also that contamination does not exceed 0.4 $Bqcm^{-2}$. This permits free access by nursing staff to a small store room off the short corridor linking the main ward corridor to the treatment room. The room itself utilises a separate ventilation system to the rest of the hospital. Fans were available to assist with dispersal of airborne ^{131}I but it was noted that they were not always used by clinical staff. To overcome this problem the fans were fitted with timers and operate automatically for several periods during the day. They are not operated continuously due to noise levels. To further regulate entry to the area, a key pad entry system is used to prevent unauthorized entrance. The door can be opened as normal from the inside in the event of an emergency.

A coincidental advantage of diverting waste water from the shower directly to the sewer was to focus the use of the holding tank on patient excreta which would contain the highest concentration of radioiodine. Nevertheless, the average amount of water used by a patient for an average stay of 65 hours was found to be 630 ± 280 litres at one standard deviation. Thus, the main limitation of the holding tank was found to be its limited size. This finding led to much shorter waste retention times before discharging into the main sewer. Previously a crude estimate that the tank would hold the waste of 2-3 patients before requiring release had been

used. The new data allowed more informed use of the tank and resulted in more accurate annual activity discharge figures being submitted to the regulatory body. With the small size of the holding tank and the marked increase in the number of patients treated annually the holding tank capacity is now constrained to a maximum of one patient's waste. This often means that the maximum decay time possible is less than 1 week. To further reduce the quantity of waste water, the volume of each toilet flush was reduced to 6 litres.

More recently any relevant pipe work discharging into shafts accessible by staff, under manhole covers and at shallow depths (e.g. holding tank overflow pipe), have been extended to ensure discharge close to the base of the shaft. This limits the possibility of contamination on the walls of shafts used by staff who may require access for operational issues arising elsewhere on the hospital site. It also reduces risks arising from unauthorised access to the system. Notwithstanding the improvements made to the operation of the holding tank, the risks and cost/benefit of continued use remain uncertain. However, within this jurisdiction the balance appears to favour continuing operation of in-situ holding tanks [3].

The apparently simple solution of using a macerator to address other waste could not be progressed due to a hospital policy to gradually withdraw their use in order to reduce the hospital's waste water Biological Oxygen Demand and also infection control risks. To date, cost has prevented the provision of an outside storage facility and similar factors together with space limitations have ruled out the use of freezers. A simple solution to the handling of waste has been found which involves the use of air tight waste bins. These containers employ an adhesive resin in the lid and have been extremely successful in containing the odour problem. The main drawback with the container type is that they cannot be opened once sealed and so the entire container must be disposed of to landfill. Each patient would typically result in one waste bag (maximum 90 litres) of bed linen/towels, one general domestic airtight waste bin (~60 litres) and occasionally a clinical waste bag (maximum 90 litres). The activity levels vary significantly from item to item and so the length of time required for storage to decay varies accordingly. The mean storage time of waste to decay recently measured over 1 year was 57 days. Variation for different 'types' of waste was small: Bed linen/towels: 53 days; General domestic: 61 days; Clinical: 60 days. An in-house software database was created using Microsoft Access and Visual Basic for Applications to log the waste and contaminated items. Records of the isotope, waste type, description, date, counts per second, contamination monitor and storage location are maintained. A unique ID number is also assigned to each waste item. The waste details are automatically printed on a label (Fig. 1) that includes the calculated disposal date i.e. the date after which the item should have decayed to background. The units of Bq/cm^2 are derived from the contamination monitor's calibration factor for ^{131}I and the measured counts per second. The calibration factor is stored in the database. After this date the waste is checked with a contamination monitor to confirm that background levels have been reached and the label is removed. The waste is disposed of as normal waste or returned to laundry in the case of bed linen/towels. The database is then updated with these results. The barcode refers to the unique ID of the waste item and allows the relevant database details to be accessed quickly and easily with a barcode reader.

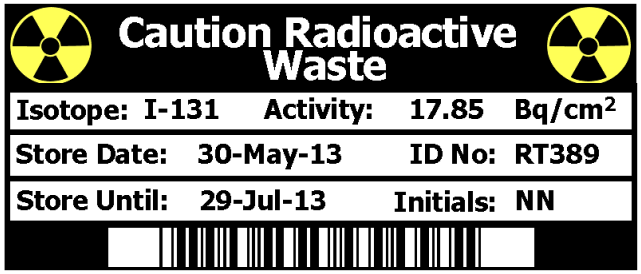

Fig. 1. The caution board on display

Detailed reports can be generated to provide details of current storage items and also records of disposed items. This allows for easy management of waste and can demonstrate transparent and detailed records of waste management to the regulatory body.

Areas of contamination in the room are minimised by covering with Benchkote® to minimize the amount of contamination that can occur. Nevertheless, there can be significant areas of contamination in the room following a treatment, with amounts being highly variable from patient to patient. Following the removal of all contaminated items for storage and decay, the room is checked for contamination using a hand held large area contamination monitor. Larger areas of contamination are typical around the toilet and sinks.

There are a number of radioactive decontamination agents available; Radiacwash®, Bindit® and Decongel®. Bind-it® is specifically marketed for decontamination of iodine isotopes and is used most frequently. Surfaces are sprayed and wiped down until no more contamination is removed. If the remaining contamination exceeds the licence limits, wipe tests using alcohol wipes are taken from the area to ensure the amount of removable contamination is negligible. Radiacwash® has quite a strong odour and so is not the preferred choice although it has also been found to be effective. Decongel® has proved very successful at removing contamination – particularly from rough surfaces. Decongel® is a thick solution that is spread over the area of contamination using a brush or scraper. Over a period of 24 hours it spreads into the pores of the surface and becomes hard. It can then be peeled off the surface removing some of the contamination. In one test case where Bindit® was used more than 10 times, with resulting wipe tests being negligible, the subsequent use of Decongel® removed > 60% of the remaining contamination. The long wait time between application and removal means that is cannot be used in circumstances where time is a factor. It is also awkward to apply.

Further improvements are envisaged in the near future as a result of approval to renew the treatment room's ensuite facilities. The opportunity will be taken to ensure that the floor of the en suite shower does not result in the 'pooling' of waste water which can occur currently due to poor floor gradient.

4. CONCLUSIONS

Following an unintended exposure incident, a wide range of steps were taken to limit the risk of an inadvertent exposure from the operation of an iodine ablation therapy service. Limited scope to fund infrastructural improvements required long term application to develop a more resilient process and leverage meaningful risk reduction.

REFERENCES

[1] AUSTRALIAN RADIATION PROTECTION AND NUCLEAR SAFETY AGENCY, Recommendations: Discharge of patients undergoing treatment with radioactive substances. Radiation Protection Series No.4. (2002).

[2] EUROPEAN COMMISSION. RADIATION PROTECTION following Iodine-131 therapy (exposures due to out-patients or discharged in-patient), Directorate-General Environment, Nuclear Safety and Civil Protection. Radiation Protection 97, Elsevier Science Inc. (1998).

[3] ENVIROS CONSULTING LTD. (2010), Review of Best Practice in relation to Iodine-131Ablation Discharges to Sewer. Available from http://www.rpii.ie/RPII/files/cf/cfaed62e-2feb-475a-88b9-3fc022acd296.pdf.

IMPLEMENTATION OF PROCEDURES FOR RADIOLOGICAL SURVEILLANCE IN PREGNANT WORKERS IN THE RADIOTHERAPY DEPARTMENT AT THE ONCOLOGY CENTER OF CAMPECHE, MEXICO

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Abstract

The increasing number of exposed workers in our Oncology Center require to improve and develop procedures to ensure the radiological protection.The surveillance is an important mechanism to ensure the safety of employees, and pregnant workers require an specific procedure of surveillance. So, the aim of this work is to establish a procedure to assure the health proteccion of the fetus and mother at work based on international guidelines, the needs of our department and measurements done at the radiotherapy service.

1. INTRODUCTION

Last two years, the demand of oncology services have increased in our region. Acording to that, the number of exposed workers in the radiotherapy department increased, being women in reproductive ages the third part of the department. Anticipating the possibility of pregnancy in our workers, the aim of this work is to stablish a procedure to assure the health proteccion of the pregnant woman and the fetus.

There are many investigations about radioproteccion during pregnancy, developing international guidelines to assure the safety of the fetus. That is the case of the the International Commission on Radiological Protection (ICRP) which recommends that the fetus of a pregnant worker should not exceed 1 mSv after the notification of pregnancy[1].

Also we know that the dose obtained from a personal dosimeter of a radiotherapy worker is not the same to the fetal dose[2] being the abdominal surface dose of the worker closer to fetus dose.

Many countries and institutions have established protocols to assure the radiological proteccion to the mother and fetus, some times it include a signed agreement where the surveillance methodology is described. The protocol resulting of this work has to guarantee to the pregnant worker the health proteccion of both, mother and fetus, even when a fetal dosimeter is not available.

2. METHODOLOGY

To design this procedure we based on international recommendations and some measurements at our radiotherapy department.

2.1 Historical Review

To ensure the safety of our building and evaluate the practices of our personal, we reviewed the historical deep equivalent dose (H_{10mm}) during the last twelve months using a

Optically Stimulated Luminescense (OSL) dosimeter at thorax. It will be presented in averages by groups of work using:

$$\bar{H} = \frac{1}{n}\sum_{i=1}^{n} a_i \qquad (1)$$

Where a represents the individual equivalent dose of the each person of the group in that month. Finally the groups presented are: Oncologist, Technicians, Nurses and Physicist.

2.2. Mesurements

At this time, we consider the technicians as the group with more probabilities of exposure in our department. So, we decided to measure the equivalent dose in these workers to complete the surveillance proposed. Taking in count that they work in periods of fifteen days at brachytherapy service or teletherapy service, we established two periods of measurements. The first period was based in case of missing staff, for example in case of a technician has abandoned the job, the remaining staff of that group cooperate to complete the work of the missing staff while the new staff is hired, increasing the probabilities of work in both services at the same period of fifteen days. The second period of measurements will be a month, it is based on the established time to exchange the personal dosimeter and the schedule described before, that means that the technician works the first fifteen days at teletherapy and the next fifteen days at brachytherapy with technicians staff complete.

In both periods, a worker will wear two dosimeters during the journal, one at Thorax(D-001) and the other at Abdomen(D-002). The thoracic dosimeter measures the deep equivalent dose and the abdominal dosimeter mesures the skin equivalent dose, in both cases we used a OSL dosimeter.

3. RESULTS

3.1. Historical review

After calculating average dose per month in each group with equation 1, we present the Fig. 1 with the graph of the historical equivalent dose achieved in each group of work. Approximately a technician achieved 1.089 mSv during a year, a nurse 0.94 mSv, an oncologist 1.03 mSv and a physicist 0.98 mSv and it is presented at the Fig. 2.

As additional data the standard derivation of the dose in the last year is 0.1mSv and the worker with the highest equivalent dose was a technician with 13% over the media.

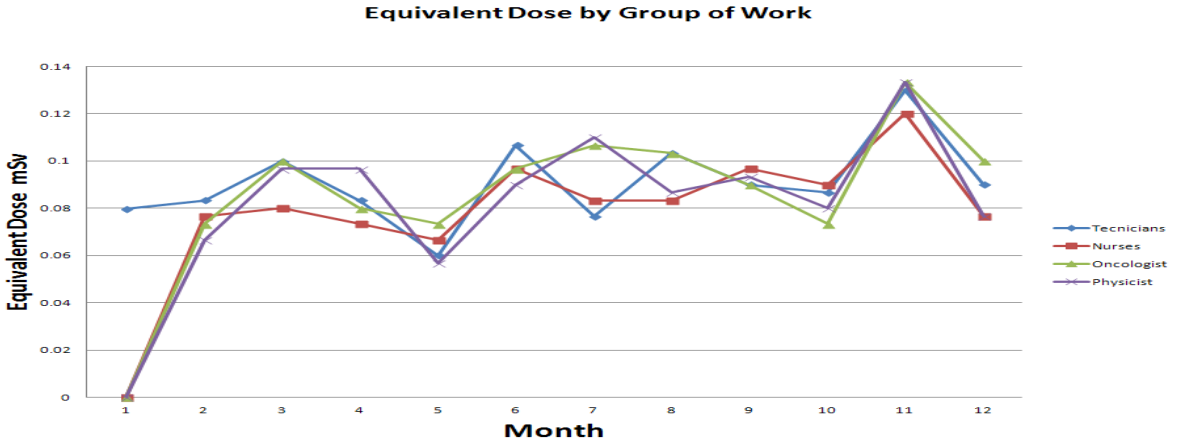

FIG. 1. Historical equivalent dose per group of work

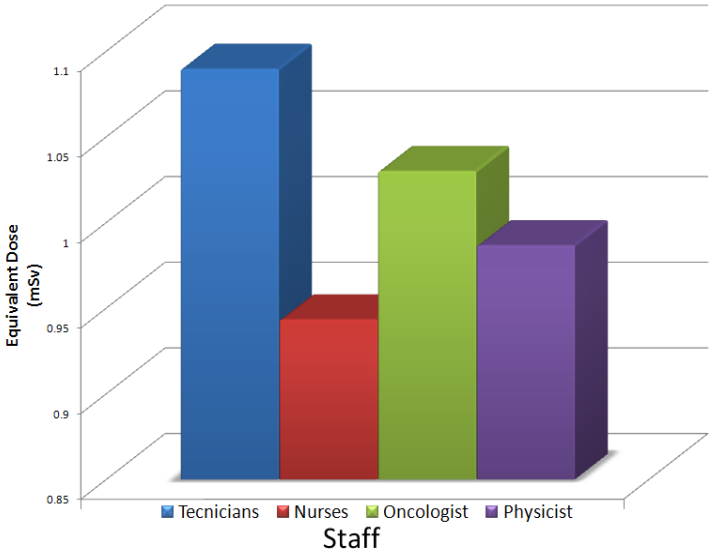

FIG. 2. Equivalent dose in last twelve months by group

3.2. Measurements

The Table 1 presentd the results of the measurements of the periods proposed. As we can see the equivalent dose of the abdomen in the first period is closer to the thoracic dose than in the second period of measurement .

4. ANALYSIS

The historical reviews show us that in a year a worker has an equivalent dose near of 1 mSv in twelve months. Thats is the recommended dose for public in a year, showing that the practices of our workers are safe.

We know that the fetal dose is the 25% of the depth dose of the gestant woman[2] and supposing that the mother has a equivalent dose of 1 mSv during the pregnancy we can estimate a fetal dose of about 0.25mSv. In our measurements we find that in the second period of measurements the equivalent dose at the abdomen is the 31 % of the dose measured at the thorax. Also we note that the abdominal dose increased during the first period, where the worker contributed to two services to cover the vacancy.

Using the results of the second period of measurements we can estimate the dose limits to ensure the health protection of the unborn child. To keep the limits from the safe side we divide 1 mSv into 9 months which means that the fetus should not exceed 0.11 mSv per month, and using the relationship of 31% from the second period of measurements we can say that the equivalent deep dose of the mother should not exceed of 0.35 mSv. Based on that we can establish 0.35 mSv per month measured at the deep dose dosimeter of the mother as the limit to ensure health protection.

Comparing the limit of 0.35 mSv measured at the thorax of the mother with the historical equivalent dose we find that it is less probably achieved than the dose at the routine work of the occupational personal. That shows that during the gestation the fetus does not have radiological risks if we set the limit of 2mSv during this period at the deep dose equivalent of the mother. However we have to consider the physical risks, for example the lifting of heavy objects during the first three months or the probability of be under scatter radiation using portable radiation generators.

In case of wearing a fetal dosimeter at the abdomen, the fetal dose could be obtained from the half of this measure and to confirm it we need to measure with phantoms at different depths to estimate the abdominal dose limit at our radiotherapy department [3].

TABLE 1. EQUIVALENT DOSE MESUAREMENT

Period of Measurement	Period 1	Period 2
Thoracic Dosimeter	0.22 mSv	0.29 mSv
Abdominal Dosimeter	0.25 mSv	0.09 mSv

Also, the Spanish protocol for pregnant woman [4] establishes that the pregnant woman should wear a deep dose dosimeter at the abdomen which could estimate fetal dose from the half of the equivalent dose of the abdominal dosimeter.

5. CONCLUSIONS

The surveillance will include the use of a abdominal dosimeter, and in absence of it we can use the results of the measurements at the second period, where we established a dose limit of 0.35 mSv deep dose equivalent for the mother to ensure the health proteccion of the unborn child.

In the protocol, it is established that the worker will receive partial record of dose per month after the notification of pregnancy. That record will include the estimated dose for the fetus and equivalent dose of the mother, and the limit of dose will not achieve 2 mSv in the dosimeter of the mother. Also, as part of the health protection the mother will not work with portable radiation generators.

Finally, we would like to continue the measurements to ensure the radiological proteccion of the fetus.

REFERENCES

[1] INTERNATIONAL COMISSION ON RADIOLOGICAL PROTECCION, Radiological protecction on medicine, ICRP Publication 105, 37 (6), Oxford (2007).

[2] INTERNATIONAL COMISSION ON RADIOLOGICAL PROTECCION, Pregnancy and medical radiation, ICRP Publication 84,30 (1), Oxford (2000).

[3] MOUNTFORD, P.J., STEEL, H.R., Fetal dose estimates and the ICRP abdominal dose limit for occupational esposure of pregnant staff to Technetium-99m and Iodine-131 patients, European Journal of Nuclear Medicine (1995).

[4] FERNANDEZ, B. et al, La proteccion de las trabajadoras gestantes expuestas a radiaciones ionizantes en el ambito hospitalario, Consejo de Seguridad Nuclear, Madrid, España (2003).

EFFECTIVENESS OF THE SHIELDING MECHANISM IN ROOMS HOUSING X-RAY DIAGNOSTIC EQUIPMENTS
A CASE STUDY OF MULAGO HOSPITAL

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Abstract

This purpose of this study was to provide useful data needed to protect occupationally exposed workers, patients and the public from exposure to scattered and leakage X-ray radiation which could pose health risks. The exposure doses of the occupational workers were determined. The availability and effectiveness of the protective gears was also investigated. TLDs were installed at selected points for a period of four weeks. Radiation leakages through the walls and doors to the members of the public were measured using a survey meter, scattered X-ray radiation received by radiation staff was measured using TLD badges. The protective gears were checked visually and suspicious ones exposed for verification. Results of scattered radiation in the imaging rooms varied between 1.20 mSv/month from the Computed Tomography (CT) Room and 0.44 mSv/month from the Casualty Center. Results of scattered radiation doses received by radiation workers were highest from Room 4 (plain radiography) of 6.0 mSv/y and lowest in the Casualty Center at 1.4 mSv/y. Radiation leakages through selected doors were found to be 0.010 mSv/h at the Uganda Cancer Institute and 0.012 mSv/h from Room 4. Overall, the values are low considering the recommended limits.

1. INTRODUCTION

The application of X-rays in medicine has greatly improved human health through diagnosis and treatment of diseases [1]. The use of ionising radiation in medicine although advantageous for diagnostic and therapeutic purposes and has been justified, accidental exposures to patients, radiation workers and members of the public may lead to deterministic effects such as radiation burns or death when the threshold dose is exceeded or stochastic effects of which cancer induction is an example even at lowest dose levels of exposure to ionising radiation [2]. Medical exposures to radiation are intended to provide a direct benefit to the exposed individuals [3] and yet it is possible that some members of the public and the radiation workers are exposed to higher doses than the recommended limits due to the ineffectiveness of the shielding by the imaging rooms and defects in lead aprons which may result in higher exposures to staff and the public. Therefore there is a need to investigate the exposure doses of occupational staff in Mulago Hospital and members of the public in waiting areas and places adjacent to the imaging room who may be exposed to higher doses of ionising radiation.

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2. MATERIALS AND METHODS

(1) Four imaging units in Mulago Hospital namely; the Uganda Cancer Institute, Room 4 and CT room at the Second Floor and the Casualty Center were chosen.

Measurements were done on plain X-ray machines at all the above rooms including the CT Room which houses a 16-slice multi CT scanner. The measurements were taken at patient waiting areas, areas surrounding the bunkers, operator's consoles and inside the imaging rooms.

(2). LiF Thermoluminiscent Dosimeters (TLDs) were given to two radiation workers in each of the four selected imaging rooms to wear around their chest or waist areas when at work in the imaging unit for four weeks. The TLD badges were collected and read out using the Harshaw TLD 4500 Reader. The two radiation staffs chosen in the imaging unit were assigned codes e.g. X and Y in each imaging unit. The crystals were then annealed after the data acquisition procedure and then re-used [4].

(3). Radiation leakages through the walls and doors reaching the patients' waiting areas were monitored using a survey meter. Similar maximum exposure factors commonly used for each imaging modality were used. A graph of the intensity of radiation leaking through the doors was plotted against the corresponding distance. Leakage to the offices within the selected units was monitored using the TLD badges that were placed on the office walls.

(4). Scattered X-ray radiation in the Operator's Console and in the imaging room was also monitored using TLDs. The crystals were sealed in polythene and plastic packets and stuck in the rooms at different levels. The couch was used as a reference point for measuring distances to where TLDs were placed. The crystals were then exposed for a period of 4 weeks and then taken for a data acquisition procedure as in (2).

(5). Physical counting of the existing lead aprons in each imaging unit in Mulago hospital Radiology department was also done. Visual and physical inspection of the status of each apron was carried out and the suspected defective lead aprons were exposed using plain radiography with the film directly underneath the lead apron. Films were processed to reveal the defects.

3. RESULTS AND DISCUSSION

3.1. Scattered radiation

From the Uganda Cancer Institute, the average quantity of scattered radiation within the imaging room was 0.56 mSv/month. This is high in a room of dimensions 5.03 m × 4.2 m when compared with the recommended value of 0.4 mSv/month for a medium sized room of 4 m×4 m [5]. This could be attributed to the high congestion in the imaging room leading to multiple scatter. The quantity of radiation transmitted to the control room and adjacent office was 0.34 mSv/month and 0.244 mSv/month respectively. This is low and within the recommended dose limits.

From Room 4, the average quantity of the scattered radiation within the imaging room was 0.88 mSv/month. This is high within the imaging room of dimensions 4.72 m × 5.83 m when compared with the set limits[5]. This is attributed to the congested space and human factors that cause a lot of scattered radiation. The quantity of radiation transmitted to the control room was 0.63 mSv/month. This value is low and within the recommended limits [6]. Of great concern is the changing room within the imaging room where comforters remain and patients dress from after examinations. There is a huge amount of scattered radiation (1.52 mSv/month) reaching it hence not safe at all.

The Casualty Center indicates that the average value of scattered radiation within the imaging room was 0.39 mSv/month. This is a room of dimensions 5.52 m × 2.82 m.

Therefore the quantity of scattered radiation is higher than the acceptable limit [5]. The size of the room coupled with the number of examinations carried out in a month could account for this high value of scattered radiation. From the waiting corridor within the imaging room, results indicate that quite an appreciable amount of radiation exists when the X-ray machine is in operation. The annual derived values for these locations are 3.41 mSv and 3.79 mSv respectively which are acceptable when compared with the annual limit for a single year (50 mSv). The quantity of radiation transmitted through the secondary barriers per month was 0.31 mSv. This is low and within the recommended dose limits [6].

From the CT, the average quantity of scattered radiation within the imaging room was 1.19 mSv/month. This is higher than the recommended maximum limits [5]. Regarding size, the CT imaging room is big enough (5.76m×4.72m) to control the scattered radiation but congested with faulty and some non-functional equipments which could cause multiple scatter. Also the CT machine operates in the energy range that permits scattered radiation due to Compton scatter. In the control room, an average of 0.56 mSv/month was recorded. This is low and within the set limits [6].

3.2. Occupational exposure The radiation doses received by staff are smaller than the maximum recommended value of 20 mSv/y for occupationally exposed staff [7] members are given in Table 1.

TABLE 1. RESULTS OF OCCUPATIONAL EXPOSURE DOSES FOR ALL IMAGING ROOMS STUDIED

Imaging Room	Radiation Worker Code	Radiation Dose (mSv/y)
Uganda Cancer Institute	X	6.6
	Y	2.8
Room 4	P	5.2
	Q	6.8
Casualty Center	R	1.35
	Z	3.67
CT Room	M	6.1
	N	4.4

Taking the CT as an example, in the operator's room, the estimated average dose per month is 0.56 mSv. This is equivalent to 6.72 mSv per year which is higher than Staff M and Staff N doses. The actual dose accumulated by staff depends on how they operate within the dose rate distribution spectrum in the X-ray room to make their doses ALARA hence Staff M and Staff N occupied different positions. Therefore if the average of 1.19 mSv per month is used (average scattered radiation), this will be equivalent to 14.3 mSv per year which is lower than the maximum permissible limit of 50 mSv in any single year. A comparison with other imaging units shows that the radiation doses in the CT room is within the same range like those of Room 4, Casualty and the Uganda Cancer Institute.

3.3. Shielding Gadgets

The number of protective gadgets physically present in the imaging rooms was not proportional to the number of staff per room. It was noted that the main shielding mechanism used by the majority of staff is the Operator's console (lead glass) with all of them running behind it at the time of examination. The lead aprons were left to the comforters and rarely used by the staff. Suspicious lead aprons from the study centers were carefully analyzed using the image viewer. However there were no cracks or faults leading to leakages found from the exposures taken on them.

3.4. Attenuated radiation

The figure below shows the variation of attenuated radiation with distance from the door of one of the selected imaging rooms in Mulago hospital.

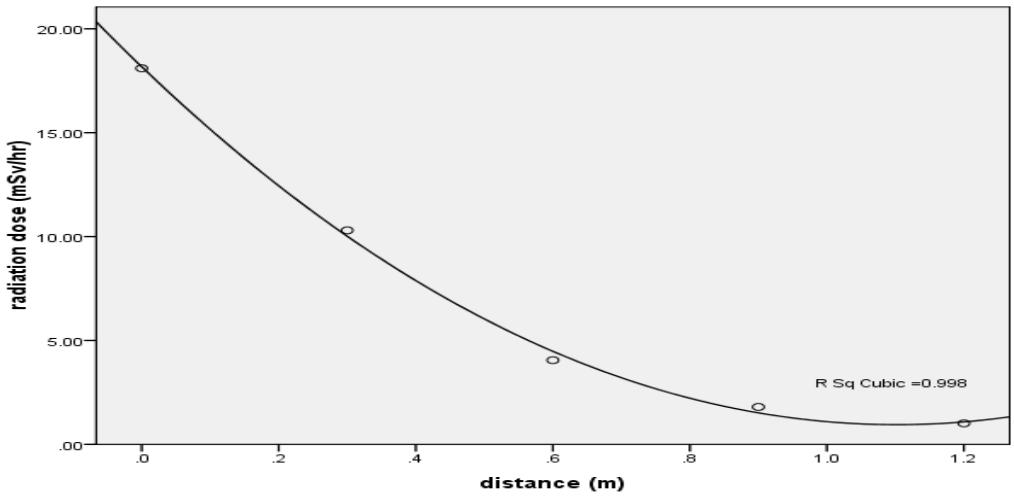

FIG. 1. Variation of radiation doses with distance from the entrance at the Uganda Cancer Institute, Mulago

The graph indicates that closer to the door, the radiation doses registered are very high. For similar exposure factors, the quantity of radiation is inversely proportional to the square of the distance from the reference point (door), that is, it follows the inverse square law. For exposures near the door, the intensity of the radiation beam is high but decreases with distance as the inverse of the square of the distance from the source, that is, a person at a distance of 1.2 m from the door (Fig. 1) receives a lower dose (0.001 mSv h^{-1}) than a person at the door (0.018 mSv h^{-1}). The dose rate value of 18.1μSv/h implies that the door needs some lead lining to reduce the dose rate to an acceptable value..

4. CONCLUSIONS

a) From the results, there is a high scattered radiation above the recommended limit of 0.4 mSv/month for a medium sized room of 4 m× 4 m for most of the imaging rooms studied

(b) Throughout the rooms studied, an average of 0.46 mSv/month leaks to the control console. This value is low considering the recommended limits for leakage radiation [6].

(c) On an average, across all rooms studied, an occupationally exposed radiation worker in Mulago hospital receives 4.31 mSv /y. This value lies below the recommended dose

limit of 20 mSv per year averaged over five consecutive years as set by the World Health Organization, International Commission on Radiological Protection (ICRP) and IAEA [5].

(d) The available lead shielding provides adequate protection of the occupationally exposed workers.

REFERENCES

[1] HENDEE, R.W. et al (1987): Radiological Physics, Equipment and Quality Control Year Book. Medical Publishers, Inc.

[2] HALL, E. J, GIACCIA, A. J., Radiobiology for the Radiologist. Lippincott Williams and Wilkins Philadelphia, PA 19106 USA, (2006).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Standards Series, Radiological potection for medical exposure to ionising radiation, Austria: No. RS-G-1.5, IAEA, Vienna (2002).

[4] ATTIX, F. H., Introduction to Radiological Physics and Radiation Dosimetry. New York, Wiley, (1986).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Basic Safety Standards Series (BSS), International basic safety standards for the protection against ionising radiation and safety of radiation sources. Series No.115, IAEA, Vienna (1996).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Training for regulators on authorization and inspection of radiation sources in diagnostic and interventional radiology, IAEA-AIRS-MED2, Austria (2004).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection in diagnostic and interventional radiology, L4 International system of radiation protection, IAEA, Vienna (2004).

SKIN AND EYE LENS DOSIMETRY IN THE TEAM OF NUCLEAR MEDICINE DIAGNOSTIC AND THERAPEUTIC WITH RADIOISOTOPES: ^{18}F, ^{90}Y, ^{99M}TC ETC: ANALYSIS OF A MEASUREMENT CAMPAIGN

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Abstract

The preparation and administration of radiopharmaceuticals in nuclear medicine diagnostic and therapeutic procedures requires the use of high-active unsealed beta-gamma or pure beta emitting radionuclides. For the evaluation of the dose to the skin, 10-20 dosimeters were prepared to be placed on the fingers of each operators; in addition, 1 dosimeter to the eyes lens also was prepared. The relative combined standard uncertainty is found to be of the order of 15%. The measured values confirmed the strong inhomogeneity of Hp(0.07) between the various positions for the individual operator and great diversity between the maximum doses (range 1.22 - 31 mSv) in the various operators. For photon fields, for the evaluation of Hp(3), the adequacy of the value adopted until now as the maximum value of Hp(10) and Hp (0.07) dosimeter worn on the trunk, was confirmed; with ^{90}Y, the values of Hp(3) measured in each therapeutic session, are not negligible (range 0.02 – 0.67 mSv). The measurement campaign has been a process of training and optimization of the method of working: individual results were discussed with each operator and with the team, to improve the radiation protection level in the operational phases of the work.

1. INTRODUCTION

The preparation and administration of radiopharmaceuticals in nuclear medicine diagnostics and therapy requires the use of high-activity radioisotope unsealed beta- gamma or pure beta emitter (every week 35 GBq of ^{18}F , 25-50 GBq of ^{99m}Tc, 15 GBq of ^{90}Y) in high-risk processes for dose skin: within the cell manipulation and during the administration phase, operators' hands are a short distance from very intense fields and inhomogeneous: the irradiation of the extremities is not uniform and therefore the use of conventional dosimeter bracelet or ring fails to provide an adequate assessment of the dose skin. Also the dose to the lens has been studied in order to verify if the value attributed for photon radiation on the basis of the dosimeter worn on the trunk is able to adequately estimate the equivalent of absorbed dose to the lens: a dosemeter was positioned close to the eyes in the preparation and administration with ^{99m} Tc and ^{18}F and in the phase of administration with ^{90}Y.

2. MATERIALS AND METHODS

For the evaluation of the dose to the skin with ^{99m}Tc or ^{18}F, for each operator 20 dosimeters were prepared: 10 chips are placed on each hand positions (similarly to the project Oramed) [1] - 1 for the thumb, 9 in the fingertip, nail and phalanx of the index, middle and ring fingers) to wear for a week dedicated to the activity together with 1 dosimeters bracelet and 1 ring worn on the dominant hand. In addition, 1 dosimeter was for the lens prepared. During the therapy session with ^{90}Y, dosimeters (in number of 3-5 for each hand) to the fingertips were worn as a routine checkup and in the administration phase, the operator also wears a dosimeter near the lens. The radiation fields in Nuclear Medicine are made of electrons, and

photons and methods of measurement of the operational quantities Hp(0.07) and Hp(3) have been developed for such fields.

For the measurement of Hp(0.07), the dosimeters used were calibrated on the road phantom in photonic fields and ^{90}Sr/Y source .TLDs (LiF : Mg , Cu, P) GR 200A (0.8 mm thick) and MCP -Ns thin TLD (LiF : Mg , Cu, P) (effective thickness 8.5 mg / cm^2) were worn on the fingers without containers but only protected by a thin polyethylene film for the measurement of Hp(0.07) and for the measurement of Hp(3) for ^{90}Y. The dosimetric system used in this mode on the fields mixed beta-gamma with has been studied in particular for the energy dependence and angular (angles from 0° to 60°) ; the compliance of the same with IEC 62387-2 [4] has been verified and was carried out the evaluation of uncertainty combined with Monte Carlo Method [3].

The importance of the electrons with the beta-emitters in the medium-low-energy from radiation protection considerations, was tested with MCP -Ns TLD thin dosimeters (dosimeters with a curve of constant dependence on energy beta emitters) for comparison with the GR200A (to medium-low energy beta emitters the GR200A dosemeters underestimates over 40-50%): the analysis was carried out both on operators (which have been distributed in the same positions dosimeters GR200A and MCP -Ns thin) both on laboratory tests: in the laboratory, the analysis of the relative response of the TLD GR200A and MCP-Ns has been carried out in some conditions such as to simulate the working mode and maximize the presence of the eventual electronic field at low energy (^{18}F syringe under 1 layer of vinyl glove and under 1 layer of vinyl glove + 10mm plexi-glass; ^{90}Y vial under 1 layer of vinyl glove and under 1 layer vinyl plus + 0.3mm Pb + 1 layer vinyl.

For the meaurement of Hp(3), the dosimeters used are calibrated on phantom slab with photonic fields and with source ^{90}Sr/^{90}Y using the conversion coefficients reported in 62387-2; for fields with ^{90}Y a factor equal to 1 has adopted conservatively [5, 6, 7, 8].

3. RESULTS AND DISCUSSION

The evaluation of uncertainty on the measurement of Hp (0:07) was made with numerical Monte Carlo method [3] by simulating 100000 events with possible combinations of uncertainty in dependence:

i. The variability of response of the dosimeter (already corrected the intrinsic sensitivity factor)
ii. The variability of the response of the reader (already corrected with dosimeters control)
iii. Energy and angular dependence of the detectors (we have assumed energy distributions similar to the Compton and photoelectric fields ^{18}F and ^{99m}Tc; we have assumed Gaussian distribution of angles with sigma= 20°)
iv. The variability of background (we have assumed Gaussian distribution with sigma = 5 µSv.

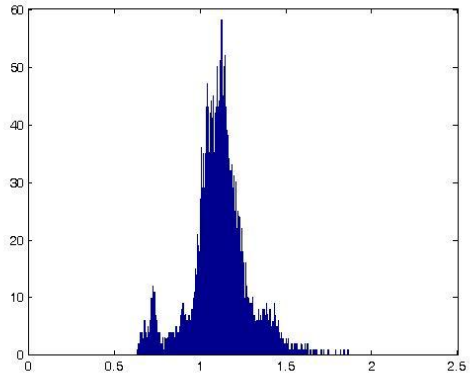

FIG.1. Response: $Hp(0.07)_{Measured}/Hp(0.07)_{true}$

TABLE 1.UNCERTAINTY RESULTS

	^{18}F and ^{99m}Tc Hp(0.07)rod
Coverage interval 95% Relative standard uncertainty combined at 1 mSv	0.84 – 1.36 15%
	^{90}Y and Photon Hp(0.07)rod
Coverage interval 95% Relative standard uncertainty combined at 1 mSv	0.78 – 1.3 15%

TABLE 2. RESULTS OF COMPARISON TESTS BETWEEN DOSIMETERS GR200A AND MCP-NS

Analysis of the relative response of TLD GR200A and MCP-Ns in simulation experimental work mode	^{90}Y vial under a) vinyl glove b) vinyl gloves vinyl +0.3 mm PB+ vinyl glove	^{18}F syringe under vinyl glove a) without plexigass b) with 10mm plexigass
related ratio GR200A / MCP-Ns	a) 0.97 b) 0.96	a) +1.08 b) +1.14
Analysis of the relative response of TLD GR200A and MCP-Ns worn by workers in the same positions	Worker at the synthesis with ^{90}Y	Worker assigned to the fractionation with the ^{18}F
related ratio GR200A / MCP-Ns	0,96 (10 dosemeters mean)	+1,05 (40 dosemeter mean)

Occurred with both the ^{18}F that with the ^{90}Y that dosimeters thin do not add further information with respect to GR200A: likely radiation fields do not contain sufficient numbers of electrons into the range of the low energy to make them more efficient in response to GR200A more sensitive, repetitive and safe: ttherefore, the thin dosimeters were not used in the measurement campaign nor in the activity routine.

3.1. Measurement results in the use of ^{18}F and ^{99m}Tc

The absorbed dose varies greatly between the different actors for the same work: (1) for operators who prepare the radiopharmaceutical - the maximum value is between 1.5mSv and 24.2 mSv median 2,85 mSv standard deviation 8.mSv; the fingers with the maximum value are mainly the index and middle fingers of the non-dominant hand on the fingertips or fingernails; the ratio between the maximum and the dosimeter worn on the ring phalanx (of dominant hand) varies from 5 to 20, ratio between the maximum and the dosimeter worn on the wrist (of dominant hand) varies from 17 to 96; (2) for operators who administer the drug, the maximum value varies from 1.22 mSv to 31 mSv and the median 3,05 mSv, standard deviation 11,75 mSv: the fingers with the maximum value are mainly the index and middle fingers of the non-dominant hand on the fingertips or fingernails, the ratio between the maximum and the dosimeter worn on the ring phalanx (of dominant hand) varies from 5 to

22, ratio between the maximum and the dosimeter worn on the wrist (of dominant hand) varies from 10 to 71.
The qualified expert has decided to give the value of dose with respect to the ring to a ratio personalized for the individual operator. The highest values correspond to the staff still in training or for any complications that occurred in the method (for example, a contamination of the glove and / or the dosimeter.

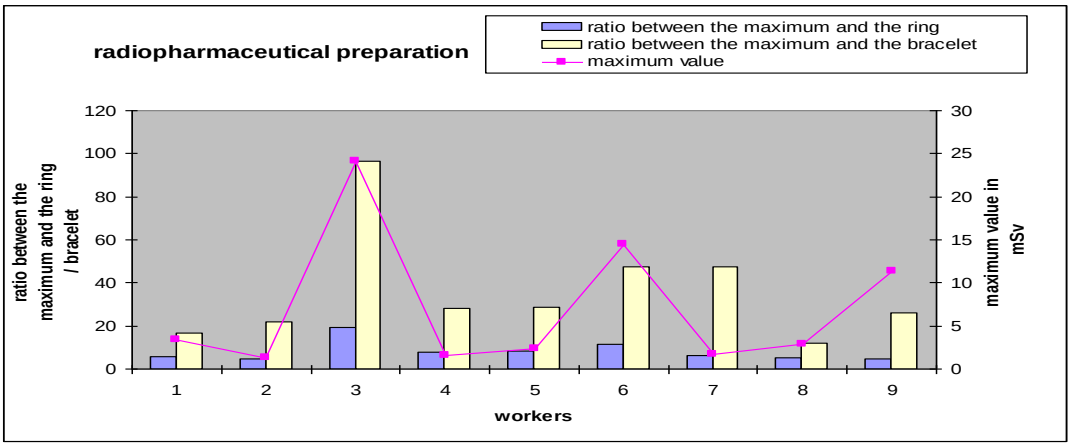

FIG. 2. Radiopharmaceutical preparation

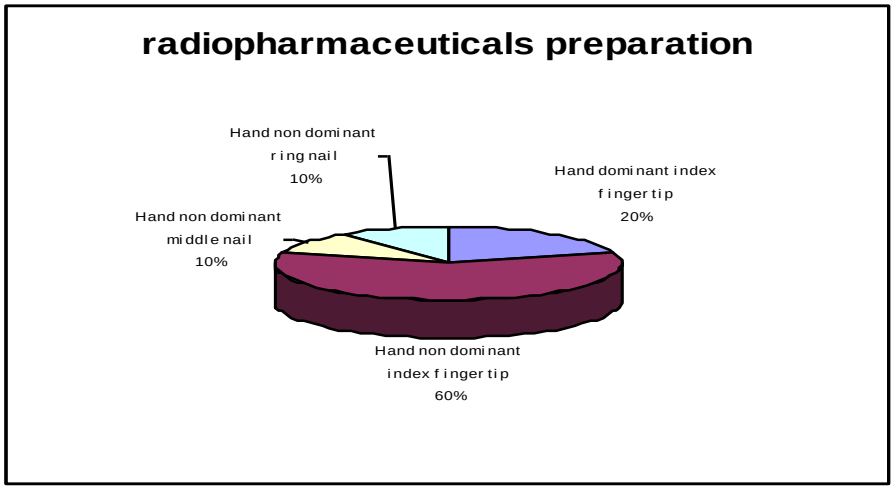

FIG.2b. Pie Chart showing the frequency of the location of the maximum

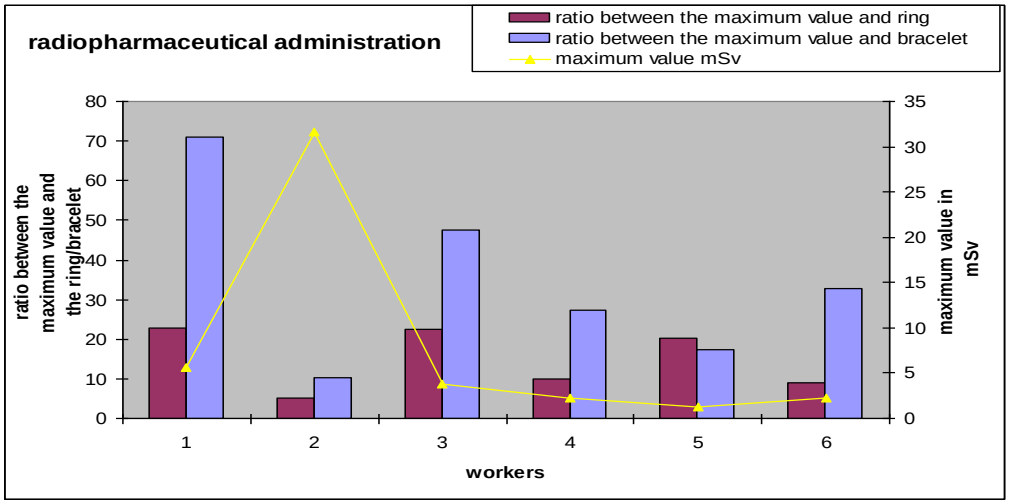

FIG. 3a radiopharmaceutical administration

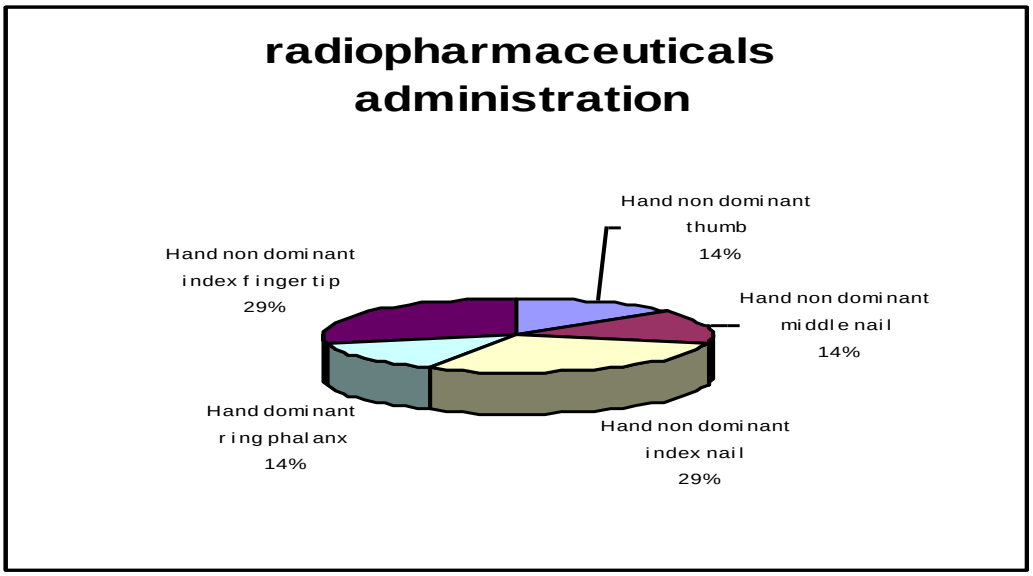

FIG. 3b. Pie Chart showing the frequency of the location of the maximum

The dose to the lens was found to be negligible in the preparation of the radiopharmaceutical and less than 0.1 mSv in the phase of administration of the radiopharmaceutical: the measured values confirmed the adequacy of the approach taken so far to take the maximum value of Hp(10) and Hp(0:07) measured at the operator's trunk during the period when the total body dosimeter is worn.

3.2 Measurement results in the use of ^{90}Y

The absorbed dose varies greatly between the various actors in the various operations in which it was divided and controlled by the method: labelling, quality control phase, fractionation phase administration. The labelling step is subjected to greater risk as all the activity of ^{90}Y at high concentration using semi-automatic procedure, is synthesized with peptide. Higher values correspond to possible complications occurred in the method. The fractionation step with automatic fractionation system gives very low dose values per session. In the administration step, the physician can absorb high values if there are difficulties in the system of inoculum in the vein of the patient, while for the operator that assists the measured

values were almost all equivalent to the detection threshold of the system (0.02 mSv). We report the results of a year (2013) of measures (such measures are made routinely on all operators those involved in the therapy sessions)

TABLE. 4. READINGS OF DOSIMETERS WORN ON THE FINGERS AND CLOSE TO EYES (mSv)

(2013 year : 20 session 15.5 GBq ^{90}Y per session)					
Step	type of dosimeter	Median	SD	Range	
Labelling	finger dosimeters $Hp(0.07)_{rod}$	4.94	3.87	0.49	16.65
Qualiy Control	finger dosimeters $Hp(0.07)_{rod}$	0.63	1.61	0.07	7.22
Fractionaction	finger dosimeters $Hp(0.07)_{rod}$	0.18	0.1	0.06	0.4
Administration	finger dosimeters $Hp(0.07)_{rod}$	0.63	2.2	0.11	2.7
Administration	eye lens dosimeters $Hp(3)_{slab}$	0.11	0.6	0.02	0.67

The values have been progressively reduced compared to previous years through both experience and communication of doses immediately after the therapy session. The measured values to the lens of the eye during the administration correlate with the values measured in the fingers, although not negligible measured values are such that they provide to not exceed the new limits.[9, 2].

4. CONCLUSIONS

The measurement campaign has been a process of training and optimization of the method of individual work. Individual results were discussed with each operator and with the team, and the methods of work has been improved for all the staff members. The preparation and administration of radiopharmaceuticals are at high risk of skin dose, underestimated by the ring dosimeter; respect for the individual limits is safeguarded by the knowledge of the maximum dose value to the hands. The value can be estimated by a multiplicative factor with respect to the ring reading, customized to individual operator.
The measured dose to the lens was found to be conservatively estimated by the dosimeter worn on the trunk in diagnostic, while in the step of therapeutic administration with ^{90}Y, the routine monitoring of this quantity with appropriate dosimeter for beta is important even if the values measured until now have maintained very low even with respect to the new limits.

REFERENCE

[1] EURADOS REPORT 2012-02, Braunschweig, April 2012 ORAMED: Optimization of radiation protection of medical staff, 920120, (2012).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety of radiation sources, International Basic Safety Standards, IAEA, Vienna (2011).
[3] BIPM, IEC,IFCC,ISO IUPAC,IUPAP AND OIML Joint Committee for Guides in Metrology Evaluation
of measurement data – Supplementb1 tom the "Guide to the expression of uncertainty in measurement" – Propagation of distributions using a Monte Carlo method.(BIPM:Sèvres) (2008).
[4] J.W.E.VAN DIJK, Evaluating the uncertainty in measurement of occupational exposure with personal
dosemeters. Rad. Prot. Dosim.125 (2007) 387-394.
[5] INTERNATIONAL ELECTROTECHNICAL COMMISSION RADIATION PROTECTION INSTRUMENTATION, Radiation protection instrumentation passive integrating dosimetry systems for

environmental and personal monitoring for external photon and beta radiation using electronic devices
or data evaluation, IEC 62387-2, Final Draft International Standard (Geneva IEC), 2012-12
[6] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Conversion coefficients
for use in radiological protection quantities for external radiation, ICRP Publication 116, Oxford
Pergamon Press (2011).
[7] EUROPEAN COMMISSION .Technical recommendations for monitoring individuals occupationally
exposed to external radiation, Radiation Protection n.160 (2009).
[8] Beherens et al, Dose conversion coefficient for electron exposure of the human eye lens Phys.Med..Biol.
55 (2010) 3937-3945.
[9] COUNCIL DIRECTIVE 2013/59/EURATOM of 5 December 2013, Laying down basic safety standards
for protection against the dangers arising from exposure to ionising radiation, and repealing Directives
89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom (2003).

DOSE MEASUREMENTS TO THE LENS IN THE TEAM INTERVENTIONAL RADIOLOGICAL GUIDANCE: ASSESSMENT OF THE EFFECTIVENESS OF PROTECTIVE EYEWEAR ANTI-X AND ACCURACY OF MEASUREMENT IN TERMS OF Hp(3)

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Abstract

The new limit of 20mSv to eye lens raises the need for further assessment of the equivalent dose to the lens in the workers performing interventional procedures under radiological guidance, particularly: a) the effectiveness of protective eye-wear anti-X, b) the accurate measurement of Hp(3) and c) an accurate assessment of the relationship between Hp(0.07) to the trunk above the apron and the equivalent dose to the lens. Measurements were made with TLD GR200A dosimeters in the workplace: an anthropomorphic phantom was used to represent the operator and a phantom of plexi-glass was employed, to simulate the presence of a patient.
The measurements carried out have shown the mean attenuation factor of the glasses anti-X is equal to ~ 4 (range 3.3-5.2) with the condition of the head of the operator is realistically rotated with respect to the source of radiation and in the diffuse-field conditions. The accuracy of the measurement in terms of Hp(3) with the numerical method of Monte-Carlo simulation, is evaluated to be ~ 12%. A conservative estimate of the dose to the lens can then be done using the reading in Hp(0.07) recorded by the dosimeter WB and applying a correction factor if eye-wear is worn**.**

1. INTRODUCTION

The new limit of 20 mSv to the lens recommended by publications ICRP, IAEA and the European Directive 2013/59 [1, 2] raises the need for further assessment of the equivalent dose to the lens in the workers performing interventional procedures under radiological guidance and in particular:

(a) The effectiveness of protective eyewear anti-X
(b) The accurate measurement of Hp(3)
(c) An accurate assessment of the relationship between Hp (0.07) to the trunk above the apron and the equivalent dose to the lens

2. MATERIAL AND METHODS

It was evaluated the effectiveness of protective eyewear anti-X, in realistic conditions more unfavorable in which the operator is located in the diagnostic angiography in which the component of the scattered radiation is able to circumvent the protective elements contained in the eyewear to obtain an estimate attenuation that the eyewear used by us can guarantee.
To represent the operator was employed anthropomorphic phantom (Rando) tissue equivalent of height equal to 90 cm above a support (70 cm), in order to simulate the presence of a patient and re-create the corresponding conditions of the scattered radiation was used a cylindrical phantom of plexi-glass 30 x 30 cm by 23 cm thick+ plexi-glass slab 7.5. A eye-wear model anti-X used by operators with front protection equal to 0.75 mm of Pb and side protection equal to 0.5 mm of Pb.

GR200A TLD dosimeters (chips) that were placed at the eye of the phantom under glasses and outside glasses; simultaneously were positioned the dosimeter close to eyes lens and a dosimeter whole body (WB) on the trunk above the apron.

	location and orientation of the X-ray tube Source detector distance 110 cm	X-ray tube potenzial	DAP Gy·cm^2
Scenario 1	Under couch - Right oblique 38° plane	90-110 kv	88/266
Scenario 2	Under couch - Right oblique 15° caudal 25°-32°	87-110 kv	123/594

FIG.1. Measurement conditions: angiography diagnostic, operator-phantom

The shielding effectiveness of protective eyewear has been evaluated in two different scenarios, i.e. the operator phantom was positioned laterally at the X-ray tube; the operator's-phantom's eyes in this geometry were placed at 70 cm from the base of the patient table and are therefore oriented at 20° to the patient-phantom. For each scenario, 5/15 sequences of images were acquired to obtain a significant irradiation of the TLD. The protective efficacy of the eyewear was then assessed by calculating the attenuation factor of the eyewear as the ratio between the reading of the TLDs placed outside the glasses (right eye chip, left eye chip, lens dosemeter EYE-D^{TM}) and those placed on the eyes protected by eyewear.
To the measure Hp (3), calibration and type testing of the slab with the conversion factors set out in IEC 62387-2 was carried out [3] and was assessed the uncertainty of the operational quantity Hp(3) with the numerical method of Monte-Carlo simulation on the fields of energy used in angiography
The evaluation of uncertainty on the measurement of Hp(3) was made [5, 6] by simulating 100000 events with possible combinations of uncertainty in dependence:

(a) The variability of response of the dosimeter (already corrected the intrinsic sensitivity factor)
(b) The variability of the response of the reader (already corrected with dosimeters control)
(c) Energy and angular dependence of the detectors
(d) The variability of background.

The reading of the dosimeters placed at chest level instead was used to evaluate the relationship between dose to the whole body on the trunk above the apron and the dose at eye

level. The same relationships are also analyzed retrospectively operators in a year of measurements (2013).

3. RESULTS AND DISCUSSION

For each scenario, Table 1 shows the ratios between between related reading outside and under the glasses separately for the two eyes, then the ratios between the Hp(3) of lens dosemeter EYE-D^{TM} worn by operator and the Hp(3) of chips placed on the eyes (left and right) on the phantom-operator.

TABLE 1. PROTECTIVE EFFICACY OF GLASSES

right eye ratio chips reading outside and under glasses	left eye ratio chips reading outside and under glasses	right eye ratio reading of lens dosemeter EYE-D^{TM} and chips under glasses	left eye ratio reading of lens dosemeter EYE-D^{TM} and chips under glasses
4.22	4.08	3,3	3,6
5,1	4	5,2	4,6

X-rays in the presence of a diffuse field can pass, not mitigated through the spaces between the unprotected face and glasses thereby increasing the dose to the eye. This is probably due to the reduction of surface shielding eyewear in a geometry in which the operator has the face oriented with an angle different from zero with respect to the diffuser means. In practice, in this case, the protection is ensured primarily by the lateral shielding less extensive and less high (0.5 mm Pb-eq) and therefore less effective. Measurements in a phantom indicate a value of dose to the lens with the use of protective eyewear approximately 0.7 (range 0.4 -1.5) for a kerma-area-product of 1 Gycm^2: these values are consistent with the reported work [7].

TABLE 2. THE EVALUATION OF UNCERTAINTY

 eye-lens dosemeter EYE-D^{TM}	parameters adopted: 20-150 kev photon field 0°-60° angles 100.000 events
Coverage interval 95% :	0.9 - 1.25
Relative standard uncertainty combined at 1 mSv	12%

The response obtained in terms of type testing united to the correction for the sensitivity factor intrinsic relative etc. allow to obtain a good evaluation of measurement uncertainty (relative standard uncertainty combined at 1 mSv equal to 12%): this is important to remain within the 20% as recommended by the ICRP 60 as the measured values are close to the limit.

TABLE 3. RATIO Hp(0.07) OF DOSIMETER WB ABOVE THE APRON AND Hp(3) OF THE EYE LENS DOSEMETER

	Ratio Hp(0.07) trunk/Hp(3) eye lens dosemeter EYE-D™	Right eye Ratio Hp(0.07) trunk/chips under glasses	Left eye Ratio Hp(0.07) trunk/chips under glasses
Scenario 1	1,73	5,6	6,2
Scenario 2	1,5	7,6	6,9

From Table 3, it is possible to verify the ratio between the dose to the trunk above the apron in Hp (0.07) and the reading of the dosimeter worn close to eye lens, as well as the ratio, if you are wearing glasses, between the dose to the trunk above the apron in Hp(0.07) and to the actual dose to the lens of the eye.

TABLE 4. ANALYSIS OF DOSIMETRIC READINGS OF CARDIOLOGISTS IN 2013

	Ratio Hp(0.07) trunk/Hp(3) lens dosemeter	Median Hp(3)	Range Hp(3)
Operator 1	4,09 ± 1,4	3,38	2,92 - 4,9
Operator 2	1,59 ± 0,5	3,22	2,17 - 6,5
Operator 3	1,3 ± 0,3	2,04	0,94 - 2,7
Operator 4	1,28 ± 0,1	3,7	2,01 – 5,05
Operator 5	1,3 ± 0,35	3,37	2,96-3,49

Note: The readings listed are for a period of worn of the dosimeter equal to 45 days.

Another aspect highlighted by the study is the ratio between the level of the reading dosimetric of the trunk and the reading level of the eyes. In practical terms, from Table 4 it is possible verify that a conservative estimate of the dose to the lens can then be done using the reading in Hp(0.07) recorded by the dosimeter Whole Body and applying a correction factor. The measured ratios (column 2) are for the most part of the operators slightly higher than 1, ever smaller: therefore, in the absence of the measure for the lens can be used the value of Hp(0.07) measured on the trunk above apron; the measured data (columns 3 and 4) lead to the conclusion that without the anti-X-glasses attenuation, Hp(3) values exceed the new annual limit [2].

4. CONCLUSIONS

The measurements carried out have shown the average attenuation factor of the eyewear anti-X is equal to ~ 4 (range 3.3-5.2) with the condition of the head of the operator is realistically rotated with respect to the source of radiation and in the diffuse-field conditions, because the screening surface of the eyewear decreases and consequently, the attenuation is less than nominal value of shielding, since the scattered radiation penetrates more easily through the spaces between the face and eyes. In addition to nominal value of shielding, it is necessary to assess the extent of the screening surface laterally and fit that should ensure good adherence to the face to reduce the unprotected areas of the face and eyewear. This result is in agreement with data reported in the literature [1, 2].
The accuracy of the measurement in terms of Hp(3) is thoroughly evaluated (relative standard uncertainty combined at 1 mSv equal to 12%) within the limit recommended by the European guideline [8]. Under the conditions of irradiation and measurement taken, it is confirmed that Hp(0.07) recorded by the dosimeter TB on the trunk above the apron, can be used as in the

past, as a conservative indicator of the dose to the lens and applying a correction factor if eyewear was worn

REFERENCE

.

[1] EURADOS REPORT 2012-02 Braunschweig, April 2012 ORAMED: Optimization of radiation protection of medical staff (2012).

[2] COUNCIL DIRECTIVE 2013/59/EURATOM of 5 December 2013 – laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom.

[3] INTERNATIONAL ELECTROTECHNICAL COMMISSION RADIATION PROTECTION INSTRUMENTATION, Radiation protection instrumentation passive integrating dosimetry systems for environmental and personal monitoring for external photon and beta radiation using electronic devices for data evaluation, IEC 62387-2, Final Draft International Standard (Geneva IEC) 2012-12.

[4] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Conversion coefficients for use in radiological protection quantities for external radiation, ICRP Publication 116 (Oxford Pergamon Press) (2011).

[5] BIPM,IEC, IFCC,ISO IUPAC,IUPAP AND OIML, Joint Committee for Guides in Metrology Evaluation of measurement data – Supplementb1 tom the "Guide to the expression of uncertainty in Measurement, – Propagation of distributions using a Monte Carlo method.(BIPM:Sèvres) (2008).

[6] J.W.E.VAN DIJK, Evaluating the uncertainty in measurement of occupational exposure with personal Dosemeters, Radiat. Prot. Dosim., 125 (2007) 387-394.

[7] THERESE GEBER, MIKAELN GUNNARSSON, SŐREN MATTSSON, Eye lens dosimetry for interventional procedures - Relation between the absorbed dose to the lens and dose at measurement positions, Radiation Measurements, 46,1248-1251 (2011).

[8] EUROPEAN COMMISSION. Technical recommendations for monitoring individuals occupationally exposed to external radiation protection, no. 160 (2009).

ASSESSMENT OF RADIATION EXPOSURES OF THE STAFF DURING HSG PROCEDURE AND EFFECTS OF CURRENT PRACTICE AMONG CLINICS IN NIGERIA

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Abstract

This is a research study of staff exposure values during Hysterosalpingogram (HSG) procedures in different major X-ray imaging clinics around Federal Capital Territory of Nigeria. Hysterosalpingogram (HSG) involves the investigation of the female endometrial system through the use of Fluoroscopy or conventional X-ray machine by radiologist [1]. A calibrated Thermo fisher scientific area survey meter (Model: RADEYE G20-10) was used to measure dose which is in the range of 8 ± 0.6 – 115.1 ± 4.5 nSv for a single exposure during the procedure per staff across different centres. The average annual estimated dose per staff member ranged from 7.71 ± 0.57 - µSv to 110 ± 4.5 µSv which indicates that the exposures of staff members involved during this practices are within the acceptable range of doses, and within the occupational dose limits.

1. INTRODUCTION

X-ray imaging procedure used in the assessment of the fallopian tubes and that of the uterine cavity towards its physiological functions is referred to as Hysterosalpingogram (HSG)[2]. The procedure involves the involvement of radiologists, nurses and radiographers staying close to the scattered beam during exposures which is of concern to the workers. Unfortunately, the ineffectiveness of the thermoluminescent dosimetry system and services within the country in recent times has increased the fear amongst medical practitioners to participate effectively during HSG procedures. Clinical diagnostic centres across Nigeria perform HSG with an average of four patients per day with five exposures per patients. The blind process of HSG involves taking five films which includes preview, Anterior- posterior, delayed, right and left lateral obliques. Close proximity of staffs during the procedure are of general concerns, though personal monitoring devices in these X-ray centres are lacking, these studies offers the opportunity to have a real time studies and also evaluate the practices across different centres in the country.

2. MATERIALS AND METHODS

A Thermo fisher scientific gamma ray area survey meter (Model: RADEYE G20-10) donated by the International Atomic Energy Agency (IAEA) is used for the purpose of these study placed to the closest position obtainable during the HSG procedure. The maximum value is obtained by placing the survey meter at the closest possible range of approach by the staff during examination; dose observed is calculated as follows:

$$\text{Dose per procedure} = (\ Max.Value\ .\ Exposure\ time\ in\ Seconds) \quad (1)$$

Since the whole procedure requires five exposures, it implies that the equation (1) will be multiplied by a factor of five except for the fluoroscopy units that will be evaluated separately for the screening period.

A daily average over several patients is obtained for each centre by multiplying the whole procedure dose by a factor of 4. Machine exposure settings vary across different machines but the parameters remains the same for each examination. Varying machine settings, filtrations, collimations, patient sizes and so on are taken into consideration to establish a standard basis for comparison among the results of study.

3. RESULTS

Table 1 presents the results from the clinical centres evaluated for the HSG procedure. Nine centres were evaluated with the maximum dose value for each X-ray exposure; corresponding time was used in dose calculation per procedure. Dose for each procedure is multiplied by 5 (for each patient), daily value is also estimated by multiplying by a factor 4 (Average per day). Average annual dose is estimated by multiplying with a factor of 240 (This implies the working days for Medical Practitioners in the country).

TABLE 1. RESULTS FROM THE CLINICAL CENTRES EVALUATED FOR HSG PROCEDURES

Clinical Centres	Max./Patient (µSv/h)	Time of Exposure(s)	Dose/ Exp.(nSv)	Dose/ Procedure(nSv)	Daily Exp.(µSv/day)	Annual Exposure(µSv/y)
1	148.3±8.3	0.1460	6.01	30.1	0.1203	28.9 ± 1.6
2	85.6±1.2	0.0850	2.02	10.1	0.0404	9.7 ± 0.14
3	1980±9.5	0.0173	9.52	47.6	0.1903	45.7 ± 0.2
4	262±10.7	0.3163	23.02	115.1	0.4604	110.5 ± 4.5
5	338.1±6.7	0.2296	21.56	107.8	0.4313	103.5 ± 2.1
6	226 ± 5.8	0.1601	10.05	50.3	0.2010	48.24 ± 1.24
7	52.5±3.9	0.1101	1.61	8.0	0.0321	7.71 ±0.57
8	380 ±4.4	0.1067	11.26	56.3	0.2253	54.1 ±0.6
9	354 ±8.8	0.1884	18.53	92.6	0.3705	88.9 ±2.2

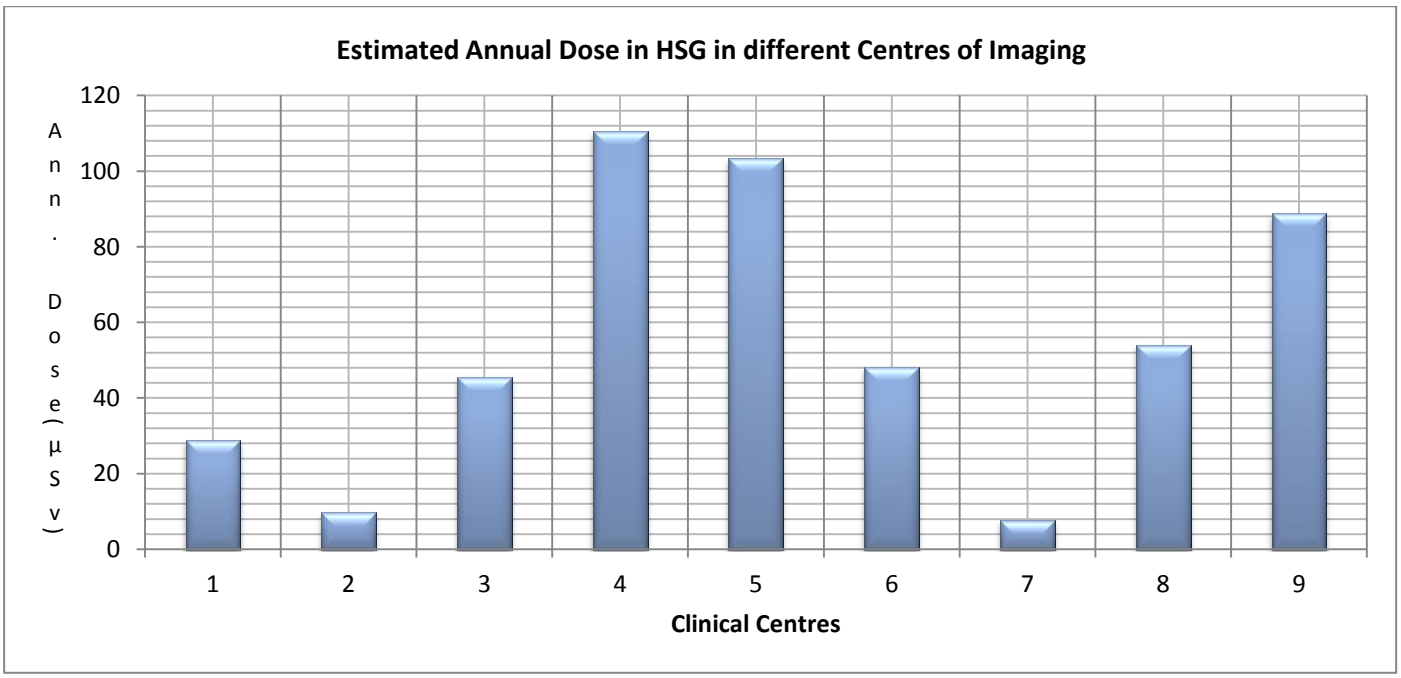

FIG.1. A graphical illustration of annual estimated dose for each centre across the Federal Capital Territory
in Nigeria for the Blind set up procedure used in HSG examination

Centre 3 presents the observed and calculated results from Table 1 without the inclusion of the dose estimated from the screening effect during fluoroscopy. Table 2 gives the results for the screening effects (dose) per procedure, daily and annual dose evaluation.

TABLE 2. RESULTS OF THE SCREENING EFFECTS PER PROCEDURE, DAILY AND ANNUAL DOSE

Clinical Centres	Max./patient (μSv/h)	Time of exposure(s)	Dose/ Procedure (μSv/h)	Daily Exp.(μSv/h) (x 4)	Annual Exposure(mSv/y) (x 240)
3	504 ± 10.7	80.4	11.26	45.02	10.81 ± 0.23

4. DISCUSSIONS

The result from Table 1indicates that the dose evaluated per procedure and extrapolated annual dose based on the blind methods without the addition of the fluoroscopy screening effect (Row 3). This shows a value which is far below the acceptable dose limits during the HSG examination across all centres in the Federal Capital Territory. Table 1 clearly indicates that irrespective of the machine types used, filtration, collimation and patient size the expected estimate of annual dose to staff remains within the acceptable dose limit expected for occupational workers. Fig. 1 is the graphical representation of the estimated average dose annually across the 9 centres investigated for the study.

Centres 4, 5 and 9 are observed to have the highest values of 110.5 ± 4.5, 103.5 ± 2.1,

88.9 ±2.2 µSv/h amongst other centres which still falls within the annual dose limit for occupational workers. Clinical centres 1, 2 and 7 shows low values of annual dose to staff members. The fluoroscopy unit (3) presents the safest centre of practice with the exclusion of the screening effect result which falls within the limits for the whole procedure. Table 2 presents the screening effect of the fluoroscopy unit, which indicates an annual estimated value of 10.81 ± 0.23 mSv/y.

5. CONCLUSIONS

The above results during the HSG procedures in different centres across different imaging centres in the Federal capital territory of Nigeria indicate the procedures followed are safe considering the blind set up procedure followed. Estimated annual staff individual exposures range between 9.7 ± 0.14 - 110.5 ± 4.5 µSv, which falls far below the acceptable dose limit for occupational workers. Though the result presented above indicates safe practices, the need for effectiveness and efficiency during HSG procedures amongst staff members makes fluoroscopy to be more preferable than the conventional X-ray evaluated above.

The Fluoroscopy screening effect evaluated shows a dose of 10.81 ± 0.23 mSv/y which exceeds the annual public exposure but lies within the occupational limits. The results show a high value which can be reduced by cautioning staff members in terms of number of patient procedures, shielding enhancement and increase in the number of staff involved. The later will help in following rotational duty amongst staff to reduce the individual radiation exposure. The need for improvement in staff monitoring is also essential for the fluoroscopy unit. Though during fluoroscopy, staff members are meant to have proper and clearer understanding of the investigation for HSG to perform with possibility of less film wastage per procedure.

REFERENCES

[1] BULS, N., OSTEAUX, M., Patient and Staff dose during Hysterosalpingography, Radiological protection of patients in diagnostic and interventional radiology, Nuclear Medicine and Radiotherapy, IAEA- CN -85 - 73.

[2] AMERICAN SOCIETY FOR REPRODUCTIVE MEDICINE, Patient's fact sheet: Hysterosalpingogram (HSG), URL: www.asrm.org.

[3] PODGOSARK .E. B., Radiation Oncology physics: a handbook for teachers and students, IAEA (2005).

[4] KHOURY, H.J., MAIA, A., OLIVEIRA, M., KRAMER, R., Patient Dosimetry in Histerosalpingography, IAEA International conference on radiological protection of patients in diagnostic and interventional radiology, Nuclear Medicine and Radiotherapy. Malaga, Spain, (2001).

[5] FIFE, I. A . J.; WILSON, D .J AND LEWIS, C. A., Entrance surface and ovarian doses in hysterosalpingography. Br. J. Radiol. 67 (1994) 860-863.

[6] FERNANDEZ, J.M., VANO, E. AND GUIBELALDE, E., Patient doses in hysterosalpingography. Br. J. Radiol. 69 (1996) 751-754.

RADIATION DOSE IMPACT ON THE WORKERS FROM THE RADIOPHARMACEUTICAL FACILITY

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Abstract

Some important indicators useful in the evaluation of occupational exposure to radiation were determined for the workers in a Brazilian radiopharmaceutical facility during the last three years. These indicators were analysed to identify and to correlate the main parameters that had an impact on the radiation exposure of workers. The data analysis took into account the number of monitored workers, the distribution of dose, the annual effective doses (above record levels), and the collective effective dose, the radiation monitoring at the workplace and environment. The conclusions from this paper were used to optimize the radiation protection procedures at this installation. The results obtained from the monitoring practice, over the last three years, are discussed and they are in agreement with the limits recommended by national and international regulatory authorities**.**

1. INTRODUCTION

Occupational exposure is the irradiation of workers during their work, regardless of the exposure situation. For the purpose of establishing practical requirements for protection and safety of workers, two different types of exposure situation cover the situations of occupational exposure: planned exposure situations and emergency exposure situations [1]. The dose limits are applied only in situations of planned exposure (normal exposure or potential exposure). In such situations, the exposure of individuals shall be restricted neither the total effective dose nor the equivalent dose to relevant organs or tissues, due to the possible combination of exposures from authorized practices that exceeds any established dose limits.

A radiological assessment should identify all aspects presented by the facility operation related to the sources from normal exposures and potential exposures, which result from the surface contamination, air contamination and sources of external and internal radiations.

The nature and magnitude of exposure, and its probability of occurrence, may be associated with any combined or isolated events of system, structure, component, radiological protection and safety procedures such as human failures. In the occurrence of any failure, improvements shall be implemented. A radiological protection program in any facility contains several indicators that may be used to control the workplace and to reduce the radiation exposure. The purpose of the monitoring of radiation levels is to characterize the workplace conditions, area classification, to give support to activities involving radiation exposure and to provide information about the external radiation sources. These monitoring procedures are also performed to identify areas that require additional shielding or application of other techniques for dose reduction.

The objective of this work is to present the main indicators that had impact on the radiation dose of workers in a Brazilian radiopharmaceutical facility in Brazil, during the years 2011 to 2013. A radiological protection programme has been well established in compliance with the national regulatory authorities [2]. This programme includes the workplace monitoring and individual monitoring, which has contributed to control of occupational exposure. In general, the monitored workers are involved in radioisotope production, labelling, encapsulation, packaging and distribution of about 95% of the radiopharmaceutical material in Brazil. Furthermore, the monitoring programme includes also a working group engaged with new radiopharmaceuticals development and quality control procedures [3].

2. METHODOLOGY

2.1. Radiation level monitoring

The workplace monitoring included both measures of general areas as those where there is contact with radiation sources. The external irradiation was detected with thermoluminescent dosimeters, placed at different points that may indicate exposures or detect abnormal situations. The monitoring of surface contamination was made using indirect methods by smearing test and counting with a high purity germanium semiconductor detector.

2.2. The environmental monitoring

The monitoring of the airborne radioactivity allowed the detection and the quantification of the concentrations of radioactive material in the air. The purpose is to limit and prevent internal exposure, as well as to provide an indication of the effectiveness of appropriate engineering controls and work practices to prevent the spread of contamination, and to support the choice of appropriate personal protective equipment. Routine monitoring at different points was undertaken, mainly in the cells for the production of ^{131}I and ^{99m}Tc-generator.

2.3. Occupational doses

For occupational exposure, the dose limits were taken into account in compliance with national standards [2]:

(a) An effective dose of 20 mSv per year averaged over five consecutive years (100 mSv in 5 years), and 50 mSv in any single year;
(b) An equivalent dose to the lens of the eye of 20 mSv per year;
(c) An equivalent dose to the extremities (hands and feet) or the skin of 500 mSv in a year.

The concepts of dose constraint and reference level are used in conjunction with the optimization of protection to restrict individual doses. A level of individual dose, either as a dose constraint or a reference level, always needs to be defined [1, 4].

2.4. Dose constraint

In the optimization protection process, the ICRP 75 [5] recommends the establishment of the dose constraint to reflect the maximum level of individual exposure that is achievable in a well-designed and managed workplace. In order to ensure an adequate level of protection

for workers, the value of 10 mSv/year was adopted as dose constraint to the effective dose in relation to any source or to practices developed in the radiopharmaceutical facility.

2.5. Reference levels

The recording level value is 0.20 mSv per month, according to national regulatory authorities. The doses equal to or greater than this value have to be recorded. For occupational exposure the investigation levels established according to national standards are:

(a) An effective dose of 6 mSv/year or 1 mSv in any month
(b) An equivalent dose to the lens of the eye of 20 mSv/year
(c) An equivalent dose to the extremities (hands and feet) or the skin of 150 mSv in a year or 20 mSv in any month.

3. RESULTS AND DISCUSSION

The main indicators describing the radiation dose distribution of workers at the radiopharmaceutical facility for 2011-2013 are presented. Radionuclides detected by air monitoring in the workplace during the radioisotope production were ^{131}I, ^{99m}Tc and ^{99}Mo. Measured emission concentrations (Bq/m^3) were below the maximum permissible values, as set out in the national standard [2].

3.1 Occupational doses assessment

Table 1 presents the number of monitored workers in function of the dose range in the radiopharmaceutical facility over the years.

TABLE 1. NUMBER OF MONITORED WORKERS PER YEAR ACCORDING TO THE DOSE RANGE

Dose range(mSv)	2011	2012	2013
$0< E \leq 2.4$	148	177	151
$2.4< E \leq 5.0$	30	22	45
$5.0< E \leq 10.0$	17	15	16
10.0< E 15.0	09	04	07
$15.0< E \leq 20.0$	01	01	01
$20.0< E \leq 25.0$	00	00	01

The main indicators useful in the assessment of occupational exposure at radiopharmaceutical facility for 2011-2013 periods are shown in Table 2.

TABLE 2. MAIN INDICATORS USED IN THE EVALUATION OF RADIATION DOSE FOR WORKERS

Main indicators	Monitoring period (year)		
	2011	2012	2013
Total number of monitored workers	205	219	221
Collective dose, S(person-mSv)	602.99	469.91	636.67
Average annual effective dose, E(mSv)	2.93±2.67	2.14±2.43	2.88±2.76
Measurable collective dose, S(person-mSv)	342.17	253.29	379.97
Number of measurably exposed workers	57	42	70
Average measurable effective dose, E(mSv)	6.00±3.34	6.03±3.30	5.43±3.66

In summary, according to Table 2 the following characteristics of dose were considered useful for assessment of occupational workers:

3.1.1. The total number of monitored workers

The number represents the whole workforce of the radiopharmaceutical facility, regardless of the task performed. This data is an indication of the monitoring programme dimension. Nevertheless, it is not necessarily an indicator of the measurably exposed workers. This fact is due to a conservative practice at radiopharmaceutical facility, because the management does not expect that workers exceed the dose standards established. Several workers are monitored for reasons of safety in compliance with the radiological programme established for this facility and in accordance with the national standards.

3.1.2. Measurable dose

The number of workers that received measurable doses (Table II) represents the exposed population to radiation. The number of workers with measurable doses included any individual with a reported dose equal to or greater than the value 2.4 mSv/year, the recording level.

3.1.3. External radiation dose

The external radiation was considered the main source exposure of workers at radiopharmaceutical facility. The contribution of this component to the total dose to monitored workers was almost 100 %, with average annual values of 2.92 mSv, 2.13 mSv, and 2.88 mSv in 2011, 2012 and 2013 respectively. The individual monitoring was performed with personal thermoluminescent dosimeters, TLDs, worn on the surface of the body. To evaluate the doses to hands, wrist dosimeters (TLDs) were used. However, no worker received more than three-tenths of the dose limit in the studied period.

3.1.4. Committed effective dose

The internal contamination dose was estimated from measurements performed by area monitoring and *in vivo* measurements using a whole-body counter and thyroid monitoring. The data reported show that in the year 2011, four workers received committed effective dose 0.26 mSv, 0.39 mSv, 0.83 mSv and 1.77 mSv. In 2012, two workers received doses of 0.55 mSv and 0.64 mSv. In 2013 two workers received doses of 0.24 mSv and 0.35 mSv. However, this internal component was not significant when compared with those doses from external irradiation.

3.1.5. Annual effective dose

The highest annual effective doses were received by three different workers (Table 1) 15.02 mSv**,** 19.80 mSv and 21.12 mSv in the years 2011, 2012, and 2013 respectively. The last dose value is above the arithmetic mean value recommended by the national regulatory authority for a year's work, but the values of average effective dose are relatively constant, as given in Table 2.

4. CONCLUSIONS

According to the results obtained from the monitoring of occupational exposures at the radiopharmaceutical facility of IPEN during 2011-2013, it was indicated that the percentage of measurably exposed workers was about 30% of the total number of monitored workers, and their contribution to the collective dose was about 60%. However, no workers exceeded 50 mSv, maximum value for a worker in a single year.

Considerable care should be taken to ensure that the filter systems are appropriate to check the lifetime of the filters and the cells because the doses increase with the time (over the years).

The optimization included operational measures such as modernization of the production lines, modernization of hot cells and improvement in the packaging system. The continuous training of workers in safety principles and good practices should be reinforced, independent of the amount of activity handled.

REFERENCES

[1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, The 2007 Recommendations of the International Commission on Radiological Protection, Annals of the ICRP, Publication 103, Elsevier Ltd. (2007).

[2] COMISSÃO NACIONAL DE ENERGIA NUCLEAR. Diretrizes Básicas de Proteção Radiológica, CNEN-NN-3.01, (2005).

[3] SANCHES M.P.; CARNEIRO, J.C.G.G; SORDI, G. M.A.A, Social-demographic profile and dose evaluation of the radiopharmaceutical facility workers**,** Int. J. Low Radiation, Vol. 7, No. 6, 2010 467, Inderscience Enterprises Ltd. (2010).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety of radiation sources: International Basic Safety Standards Interim edition, No. GRS part 3, Vienna (2011).

[5] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, General principles for the protection of workers, Publication 75, Pergamon Press, Oxford (1997).

OCCUPATIONAL RADIATION PROTECTION IN NUCLEAR MEDICINE DEPARTMENT KUWAIT CANCER CONTROL CENTER

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Abstract

The objective of this paper is to evaluate occupational radiation exposure and the current radiation protection practices in Kuwait Cancer Control Center nuclear medicine department to ensure compliance with the international and local regulations. Nuclear medicine has a key role in the management of patients' treatment by early disease diagnosis. The increased use of CT in hybrid imaging and the availability of in-house cyclotron has raised many radiation safety issues. The study addressed the sources of radiation exposure to nuclear medicine staff during their routine work. The radiation dose received by the physicians, technologists, nurses, physicists, and radiochemists were analysed and presented. In addition, the dose to pregnant worker during her entire period of pregnancy was also discussed. The level of radiation exposure varies according to the task and the work environment. Following the basic rules of radiation protection (time, shield, distance) and rotation of staff to different types of work within the department is an effective way to reduce the radiation dose received by the staff. Technologist and radio-chemist working in cyclotron and hot-lab were the most exposed groups, with significant individual differences in effective (whole-body) doses. The average occupational radiation dose for all workers at the department, under study, was within the annual occupational dose limit. Implementing radiation protection principles had a significant impact on the reduction of doses to the most exposed employees.

1. INTRODUCTION

The Kuwait Cancer Control Center (KCCC) is a comprehensive center dedicated entirely to the purpose of providing cancer care across the State of Kuwait. Founded in 1968, KCCC is a governmental center affiliated to the Kuwait Ministry of Health. It is primarily made up of the Hussain Makki Juma building for specialized surgery, the Sheikha Badriya Al Sabah medical oncology building, and the Faisal Sultan Ibn Issa building which contain radiology, nuclear medicine, and radiotherapy departments.

Nuclear Medicine is a very important branch of medical imaging which utilizes gamma-ray emitting radionuclide to obtain information about the physiology of the organs within the human body as well as the detection of tumor growth sites. This is done by introducing the radioactive labelled substance into the patient body and placing the patient around external radiation detectors. These radioactive labelled substances, which are called radiotracers, can then be used to track physiological processes in vivo typically imaged using a gamma camera [1]. The nuclear medicine department in KCCC is one of the largest department in Kuwait provided with 11 MeV Cyclotron and two Positron emission tomography–computed

tomography (PET/CT) scanners, two Single-photon emission computed tomography (SPECT/CT) scanner, four gamma camera scanners. The KCCC nuclear medicine department diagnose over of 7,000 patients each year. There are 6 physicians, 20 technologists, 14 nurses, 4 physicist, 2 radiopharmacist and 3 cyclotron staff working in the department. There is always concern about the radiation exposure and the use of unsealed radioactive materials in nuclear medicine departments. The aim of this paper is to evaluate the occupational radiation exposure and the current radiation protection practices followed by the staff working in KCCC nuclear medicine department and compare them with the recommended limit of (ICRP-103) and IAEA basic safety standard publications.

2. CURRENT RADIATION PROTECTION PRACTICES

2.1. Protection of personnel from radioactive unsealed source

In nuclear medicine department, staff member may be exposed to radiation due to their daily routine work. In KCCC nuclear medicine department the Staff member may be exposed to radioactive source during transfer/receiving of packages, preparation of radiopharmaceuticals, and in the administration of radiopharmaceutical to the patients. The radiation level from the radioactive packages is usually measured before transportation of any radioactive material (e.g. ^{99}Mo / ^{99m}Tc generators) using survey meter. The radiation level should fall within the transport index (TI) limits. All members of staff are provided with thermoluminescent dosimeter (TLD- LiF), calibrated and measured (Hp10, Hp0.07) whole body and skin dose respectively, on a monthly basis by the Kuwait radiation protection department, dosimetry Lab (See section 3). The exposure should be kept as low as reasonably achievable (ALARA) and within the ICRP-103 recommended dose limits (Table 1).

TABLE 1. SUMMARY OF ICRP - 103 RECOMMENDED DOSE LIMITS [2]

EXPOSURE TYPE	DOSE LIMITS (annual)
Whole Body Effective dose	20 mSv per year averaged over defined periods of 5 years
Effective dose to the embryo or fetus	1 mSv
the lens of the eye the skin the hands and feet	150 mSv 500 mSv 500 mSv

Every effort is be made to keep these limits as low as reasonably achievable by implanting the basics radiation protection rules (Time, shield, distance). Shielding of the radioactive source is an effective way to reduce the staff exposure during preparation and administration of radioactive materials. To study this effect; a comparison of the physicians TLDs Hp(10) whole body effective dose reading while administering fluorodeoxyglucose (^{18}F-FDG) and (Na^{18}F) sodium fluoride dose to the patients (with and without) using auto-injector (MEDRAD Intego PET Infusion System) was made. The department Started using

the auto injector in 2012, before that time, the injection of ^{18}F-FDG was administered manually. The result in Fig.1 shows that the whole body (effective dose) Hp(10) was reduced by 40% using the automatic injector.

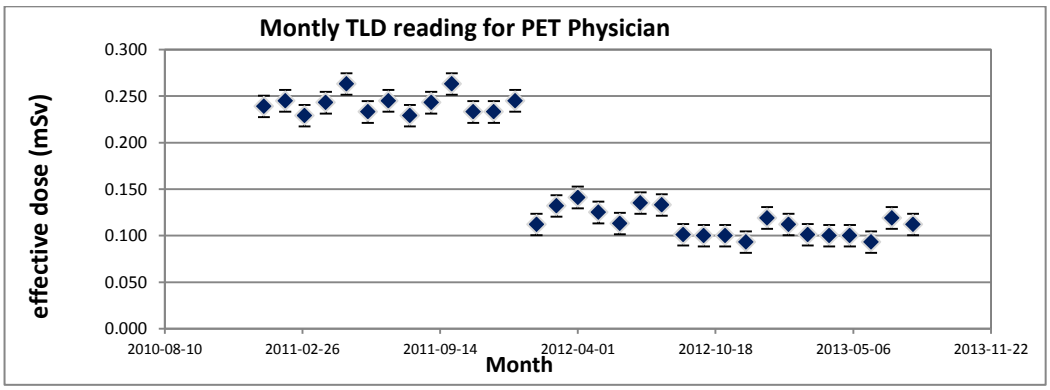

FIG. 1. shows a comparison of the whole body effective dose (Hp10) for the physicians working in PET unit mainly administering ^{18}F- FDG, Na^{18}F before and after using the auto-injector

2.2 PROTECTION OF PERSONNEL FROM RADIOACTIVE PATIENT

As mentioned earlier, in nuclear medicine procedures the patients will be injected with radiopharmaceuticals and hence, the patient will be a radioactive source for a period of time depend on the type and half-life of the radioactive source administered. As radiation workers, the staff should follow the radiation protection rules and try to minimize their exposure from all sources of radiation. Therefore, the KCCC nuclear medicine department is renovated to fulfill this approach with separate waiting area and toilets for patients injected with radioactivity, i.e., especially designed imaging room with 4mm leaded walls, and control rooms with shielded glass for SPECT/CT and PET/CT scanners. Explaining the procedure and answering all patient questions before the injection of the radioactivity is an essential practice to minimize the time spent with the patient post-injection. Also, keeping enough distance between the staff and the injected patient is an effective way to reduce the staff exposure in gamma camera rooms according to the inverse square law [1]. To study this Law, the dose rate from the patient in gamma camera room while scanning cardiac patients injected with 25 mCi (^{99m}Tc), was measured versus different distances from the patient (1, 2, 3 and 4 m). Fig. 2 shows that the dose rate is reduced by factor of 4 if the distance increased from 1 meter to 2 meter from the patient. Hence, keeping at least 2 m from the injected patients in gamma camera room during scanning time will help to minimize the staff exposure.

2.3. Protection of pregnant worker

If a member of staff is, or believes she might be, pregnant she must inform the RPO in writing as soon as practicable. The RPO will review her recent dose record, and ensure that her occupational exposure is minimised for the remainder of her pregnancy. According to the International Basic Safety Standards [3] “notification of Pregnancy shall not considered a reason to exclude female worker from work”. However, if it is thought likely that her exposure will exceed 1mSv, during the remaining period of her pregnancy, and then changes to her duties will be considered. She will also be kept away from any potential sources of major contamination and sources of radioiodine. To evaluate the radiation exposure for a pregnant worker in our

department during the entire pregnancy period, a pregnant worker used an electronic dosimeter (DoseRAE2) in the abdomen area to measure the abdominal dose equivalent, and TLD in the Trunk area to measure the whole body dose equivalent. Fig. 3 shows that the total whole body dose equivalent for the pregnant worker does not exceed 1 mSv for the entire period of pregnancy.

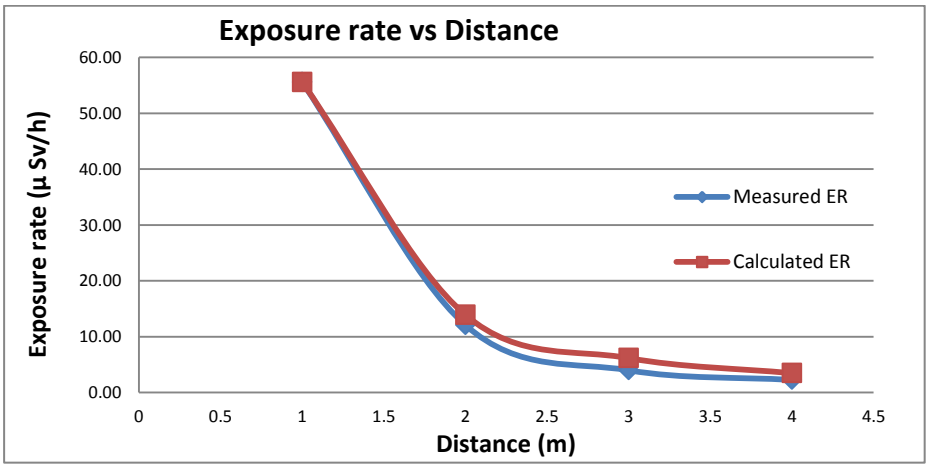

FIG. 2. The average exposure rate from radioactive patients injected with 25mCi ^{99m}Tc cardiac scan in gamma camera room versus different distances according to the inverse square law. The blue line shows the measured exposure rate, the red line shows the calculated exposure rate

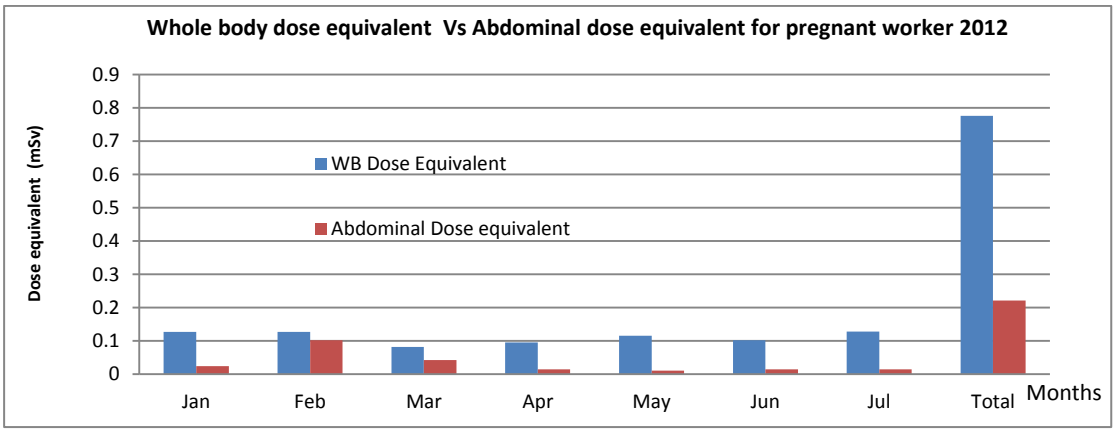

FIG.3. Comparison between the Whole body dose equivalent and the abdominal dose equivalent for pregnant worker in KCCC nuclear medicine department.

2. INDIVIDUAL MONITORING Hp(d)

For individual monitoring the operational quantity is the personal dose equivalent, Hp(d). The personal dose equivalent, Hp(d), is the dose equivalent in ICRU tissue at depth of d mm in human body below the position where an individual dosimeter is worn. For monitoring of the Skin dose d=0.07mm is recommended and for whole body effective dose d=10mm [2]. As mentioned earlier, TLDs are used in the department by all members of staff as a personal dosimeter, doses measured on monthly basis by the national dosimetry lab. The TLDs reading represent Hp(10) and Hp(0.07) values. To investigate the radiation exposure among the staff working in KCCC nuclear medicine department, the whole body effective dose Hp(10) and the skin dose Hp(0.07) values were analyzed from 2011 till 2013 (Figs. 4 and 5). The Hp(10) values shows that cyclotron staff received average dose value of 13 mSv in 2011. An investigation was done by the radiation protection officer in the department to

know the reason why the cyclotron staff were received this high reading. Many issues were raised in the investigation process.

In the production (chemical-synthesis) unit, there is only one dispensing unit, and the department produces ($Na^{18}F$) in the same day. The dispensing unit lead shielding was designed for activity in the range of (50-120 mCi), however, the work load require activity in the range of (300-350 mCi). While the Quality Control (Q.C) tests of NaF products are carried out, one of the tests must be done with undiluted NaF sample and injected to the HPLC device for 20 minute run. There are three parts in the HPLC device which should be covered with lead shield; the main body of the device, the lines which carries the radioactive waste to the waste vial, and the waste vial. Another problem was in the shielding design of the elevator which transfer the radioactive doses to the PET unit. These problems were solved eventually and, hence, the annual whole body dose for the cyclotron staff was reduced to 7 mSv in 2012 and 6 mSv in 2013. However, increasing the number of cyclotron staff and distributing the work between them will reduce the exposure time and hence, the total effective dose.

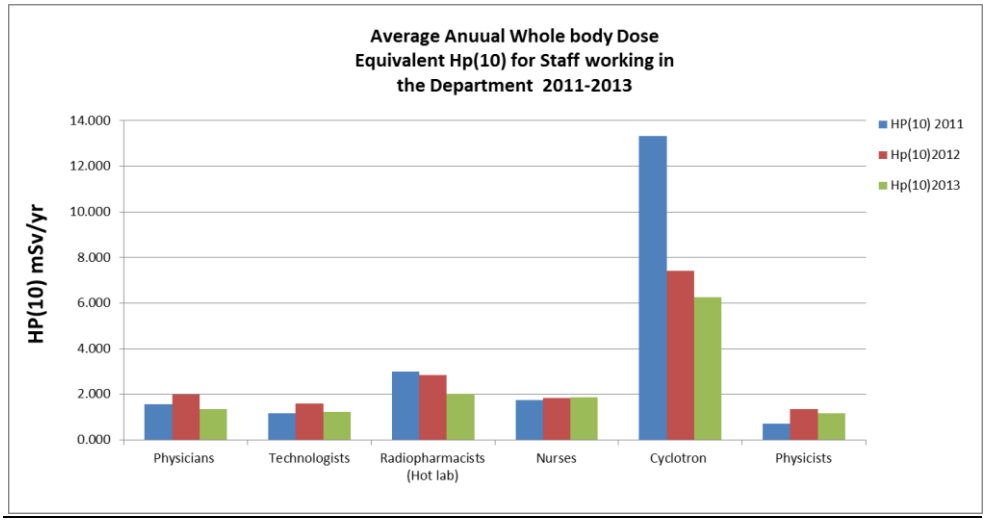

FIG. 4. Annual whole body effective dose for staff working in KCCC nuclear medicine department from 2011-2013.

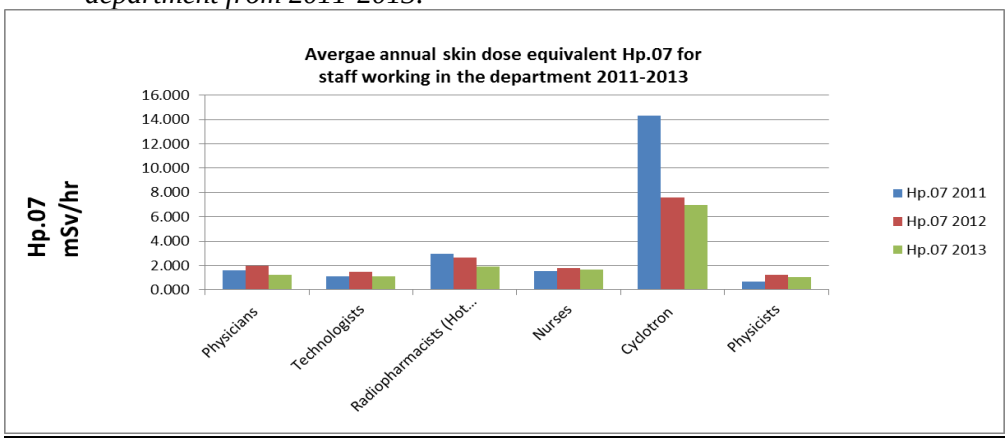

FIG.5. shows the annual Skin dose for staff working in the KCCC nuclear medicine department from 2011-2013.

4. CONCLUSION

Nuclear medicine plays an important role in patient's diagnosis and treatment. The level of radiation exposure differs significantly depending on the types of work and the roles played by the different personnel. Following the basic rules of radiation protection (time, shield, distance), and rotation of staff to different types of work within the department, are the effective ways to reduce the radiation dose received by the staff. The KCCC nuclear medicine department is following the IAEA basic safety standards and the dose limits for the staff working in the department, and maintains the staff exposures within the ICRP-103 recommended limits.

REFERENCES

]1] CHERRY, SIMON R., JAMES A. SORENSON, AND MICHAEL E. PHELPS, Physics in nuclear medicine. Elsevier Health Sciences, (2012).

[2] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103. Ann. ICRP 37 (2-4), (2007).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Reports Series No. 40, Applying radiation safety standards in nuclear medicine, (2005).

PERSONAL PROTECTIVE EQUIPMENT FOR IONIZING RADIATION PROTECTION IN HEALTH SECTOR

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Abstract

Studies on the effects of radiation which is often used to diagnose and to treat in health sector, particularly focus on used methods, devices, and health of patients. On the other hand, occupational safety and health (OSH) for the employees which are exposed to radiation risk is an important concern which should be considered. Therefore, eliminating the radiation risk completely is impossible. To cure and to diagnose the health problems, radiation has to be used in some procedures. Hence, PPE usage is essential to protect the employees conducting these practices. Personal protective equipment (PPEs) which are used for radiation protection in the health sector have to be CE (Conformity European) marked; have a user guide in official language of the country and also have to meet the requirements of PPE Directive of EU (89/686 EEC).

In the present study, PPEs used for radiation protection in health sector on placed market in Turkey, especially the protective clothes, are investigated in the aspect of complying with the basic health and safety requirements and in the light of related standards.

1. INTRODUCTION

Studies on the effects of radiation, which is often used to diagnose and to treat in health sector, particularly focus on used methods, devices and the health of patients. n the aspect of occupational health and safety, the employees must be focused on, too. The main occupational safety and health problems faced by employees are exposure to cytotoxic drugs, adverse effects of shift work, physical and psycological violence, ergonomic problems and burnout syndrome [1-5].

On the other hand, studies on occupational health and safety problems of the employees is limited. This is an important concern, especially for employees working in radiologic monitoring and diagnostics (X-ray, magnetic resonance, PET, CT, etc.) and interventional treatment (angiography, radiothreapy etc.) [6].

Therefore, as radiation is used in the above mentioned procedures to treat and to diagnose the health problems, eliminating the radiation risk completely is impossible. To prevent an adverse effect of radiation, it is stated in the international legislation that technical, medical and organizational measures should be taken with a proactive approach. Technical measures have a hierarchical structure. The structure gives priority to eliminate the risk at the source itself. In case, it is not possible to reduce the risk at the source, and other technical measures are not sufficient to ensure requied level of protection, PPE usage is inevitable [8, 9].

PPEs which are used for radiation protection have to be CE (Conformity European) marked, have to have user guide in official language of the country and also have to meet the requirements of PPE Directive of EU (89/686 EEC). As these products have a complex design and provide protection against a high risk which cannot be recognised by employees in a reasonable time to avoid, they are in the Categori III according to Directive and they must be certified by an independent conformity assessment body which is called as “notified body”. These products have a number which is unique for each notified body [7-11].

2. METHODS

In this study, radiation protection is investigated in workplace practices and product safety aspects.

For workplace practices, oncology services in diagnostic and interventional treatment departments are visited in a hospital in Ankara. This hospital is selected as it has the highest capacity and patient circulation in Turkey. Data was collected by focus group interview methods. This group consists of department manager (the most experienced manager), purchasing manager, radiology technicians and experts. Working conditions and applications which lead to a risk of exposure to radiation are identified, as well as the PPEs.

For product safety, PPEs produced for radiation protection in health sector are evaluated during market surveillance activities planned especially for this study in Ankara, Turkey. CE marks, user guides and technical documentation are investigated and analyzed in terms of conformity to basic health and safety requirements stated in the PPE Directive.

3. RESULTS AND DISCUSSIONS

In an analysis conducted in oncology services at the hospital, Interventional Radiology, Radiology, CT, Simulation devices and Nuclear Medicine units, it was determined that there is a possibility of direct exposure to radiation for employees. It has been observed that the radiology technicians and medical staff including physicians who perform nuclear medicine applications are in the main risk group. It was also observed that in these units for radiology technicians, protection against radiation exposure is provided by means of lead glass and by the application of preventive working methods to eliminate direct exposure. It has been seen that there are some types of monitoring devices which are surrounded by lead curtain for protection. Thus, more effective protection method has been used by limiting the risk at the source. Provision of protective equipment provided as well as technical measures are taken. PPEs suplied for radiation protection are; lead apron, lead vest and skirt, thyroid protectors, gonad protective, radiation protective eyewear and lead glove, as shown in Fig. 1.

FIG. 1. Existing PPEs in the market

In the interview, despite the use of radiation dosimetry measurements by physcian at the nuclear medicine application, they find difficult to wear aprons for long-term use and for this reason, they stated that they do not use them in some cases. The weight of lead aprons are reported as main reason for this. It is an interesting finding that the sensitivity in the measurement of radiation exposure is not displayed. The reason of this cannot be thought as a result of lack of information.

Personal protective equipments are examined in the hospital. It is observed that the old ones do not have CE marks. Nevertheless, new ones have CE marks and they were in

accordance with technical regulation. In this equipment, the users do not pay attention to product's due date, but the effectiveness of the apron's lead layer is checked by X-ray monitoring.

Also, during the investigation in the market, the CE marking was present on the products, but it has been found that 30% of the products are not in accordance with the technical regulation. It is detected that the technical files are not processed/regulated according to PPE Directive, but to Medical Device Directive.

For non-conforming products examined during the market surveillance; distributors, producers and importers were informed that certification should be in accordance to the technical regulations.

4. CONCLUSIONS

i. PPEs for radiation protection should be safe because they are being used as personal and designed for protection. Thus, products should be in line with PPE Directive and should be selected accordingly instead of Medical Device Directive.
ii. In the use of this type of equipment which has layer of lead, life time of the product specified by the manufacturer must be taken into account.
iii. Manufacturers should be informed about that PPE for radiation protection should be certified according to PPE Directive.
iv. For raising the usage level of lead apron, management of hospital should make it compulsory, like dosimeters, and select more ergonomic and lightweight products. In order to improve self-protection awereness of physicians, programmes aiming change in attitude and behaviour should be carried out.
v. It is recommended to carry out further research to study on the test of products according to standards and evaluate the efficiency of the technical measures.

REFERENCES

[1] REVISTA PANAMERICANA DE SALUD PUBLICA, Physical and psychological violence in Jamaica's health sector, Vol.18 No.2,Washington (2005).
[2] CURATIONIS JOURNAL OF THE DEMOCRATIC NURSING ORGANISATION OF SOUTH AFRICA, The lived experience by psychiatric nurses of aggression and violence from patients in a Gauteng psychiatric institution, Vol.32 No.3, South Africa (2009).
[3] OCCUPATIONAL & ENVIRONMENTAL MEDICINE, Occupational exposure to cytotoxic drugs in two UK oncology wards, No.59, UK(2002).
[4] CLINICAL JORUNAL OF ONCOLOGY NURSING, Impact of shift work on the health and and safety of nurses and patients,Vol.10 No.4,USA (2007).
[5] EUROPE PUBMED CENTRAL, Stress and burnout in oncology, New York (2000).
[6] EUROPEAN COMMISSION, Occupational health and safety risks in the healthcare sector (2010).
[7] EUROPEAN COMMISSION, Directive 89/686/EEC on personal protective equipment (1989).
[8] EUROPEAN AGENCY FOR SAFETY AND HEALTH WORK, Directive 89/391/EEC – OSH Framework Directive, (1989).
[9] OCCUPATONAL SAFETY AND HEALTH LAW (6331), O.G.No.28339, Turkey (2012).
[10] PERSONAL PROTECTIVE EQUIPMENT REGULATION, O.G.No.26361, Turkey (2006).
[11] REGULATION ON CATEGORIZATION OF PERSONAL PROTECTIVE EQUIPMENT, O.G.No.28230, Turkey (2012).

ASSESSMENT OF OCCUPATIONAL RADIATION EXPOSURE IN MEDICAL PRACTICE

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Abstract

For the purpose of radiation protection, dose of occupational exposure in medicine comprising nuclear medicine, interventional cardiology, radiotherapy and radiology were analyzed. Dose measurements were performed using two-element TLD cards. The range of dose distribution pattern showed left skewed towards low doses in accordance with the distribution pattern described by UNSCEAR, meaning that the most occupationally exposed workers receive very low doses with only a small number receiving high doses. The highest maximum dose received in 2013 was 49.37 mSv, and the lowest dose was 10.32 mSv received in 2010. Among the exposed workers, it was observed that the highest average dose was received by the workers in interventional cardiology department.

1. INTRODUCTION

Ionizing radiation has been increasingly applied in medicine over more than last few decades and is now confidently recognized as a vital tool for diagnosis and therapy. The overwhelming benefits accruing to patients from properly conducted procedures have fostered the widespread practice of medical radiology [1], with the result that medical radiation exposures have become an important component of the total radiation exposure of populations. The application of ionizing radiation in medicine for diagnosis and treatment takes place within the framework of several specialities such as X-ray diagnostics, interventional radiology, nuclear medicine, radiation therapy etc. The benefits to human health from these practices are extremely valuable when indications are justified and performance of the respective procedures is qualitatively satisfactory.

First deterministic effects with skin injuries have been reported in medicine a short time after the introduction of X-rays [2]. In the succeeding years erythemas, skin-ulcers, cancers and even fatal outcomes were observed [3]. Exposure of workers who perform diagnostic and therapeutic procedures is an unavoidable part of the risk that accompanies the health benefits to the patients. These workers form the largest, or one of the largest groups of people in the world exposed occupationally to ionizing radiation. Unfortunately physicians and technicians are unaware of radiation protection, dose, long-term risks and population health impact caused by the use of medical ionizing radiation.

Thus, use of radiation for medical diagnosis and treatment implies acceptance of certain level of risk by the medical staff members and hence, it is required to ensure appropriate protection. The occupational radiation workers should be aware of potential exposure to radiation and its health effects. The medical use of ionizng radiation, while offering great benefit to patients, also contributes significantly to radiation exposure of individuals and populations [4-6].

The aim of this study is to analyze the occupational radiation dose during diagnostic procedures and treatments in nuclear medicine, interventional cardiology, radiotherapy and radiology.

2. METHODS

In this study, two-chip TLD cards kept in a holder are issued for quarterly (3 months) basis to the occupational workers working in different organizations. The worker wears the TLD on torso at the working time. After using the cards for the stipulated time, organizations send back those used TLDs to the Health Physics Division (HPD). The doses received by the TLDs are measured using the TLD Reader. In the reader, the gas heating system uses a stream of hot nitrogen at precisely controlled, linearly ramped temperatures to a maximum of 300°C. The hot gas heating under closed loop feedback control and the superior electronic design produces consistent and repeatable glow curves. The annealed TLD again issue along with the dose report to the relevant worker for use of next quarter cycle.

The operational dose quantity used for the estimation of doses from external radiation is the personal dose equivalent $H_p(10)$. To measure the personal dose equivalent, manual Harshaw TLD Reader Model 4500 is used [7]. The typical TL card used consists of LiF:Mg,Ti (TLD-100); phosphor has the effective atomic number of 8.2, approximately equivalent to that of the soft tissue of a human body, TL chips 3 mm (1/8 inch) square, encapsulated between two sheets of Teflon 0.003 inches (10 mg/cm^2) thick and mounted on an aluminum substrate [8].

Prior to use, each TLD is exposed with 2mSv dose from Secondary Standard Dosimetry Laboratory (SSDL) of BAEC with respect to $H_p(10)$, using a ^{137}Cs beam incident on a slab phantom of PMMA for measurement of elemental correction coefficient (ECC). The BAEC dosimetry service (SSDL) participated in intercomparison of International Atomic Energy Agency (IAEA) for individual monitoring of radiological measurement for monitoring purposes in 2001 [9]. By following the IEC publication, the performance of the TLD systems also periodically tested [10].

In this study, the annual effective doses measured from 2010 to 2013 have been analyzed. The dosimetry service at HPD uses personal dosemeter with minimum detectable dose limit (MDL) of 0.05 mSv after deducting the background radiation dose (320 μSv/quarter) for a three-month monitoring period. The workers who received effective doses less than MDL are considered as non-exposed. Therefore, the dose less than MDL is recorded as zero. All values of $H_p(10)$ are recorded and reported as the effective dose.

3. RESULTS AND DISCUSSION

In Bangladesh, monitoring of occupational dose has only been performed by HPD under Bangladesh Atomic Energy Commission since 2000. Number of monitored physician and non-physician (medical physicist, scientific officer, radiochemist, technician and nurse) radiation workers in four occupational categories namely nuclear medicine, interventional cardiology, radiotherapy and radiology and their total dose were summarized in Table 1. In this study, total 49 organizations were included; out of this, number of nuclear medicine department was 18, number of interventional cardiology department was 11, number of radiotherapy (only included those from bigger hospitals) department was 10 and number of radiology (only included those from bigger hospitals which has more than five radiation workers) department was 10.

TABLE 1. NUMBER OF YEARLY MONITORED WORKERS IN DIFFERENT OCCUPATIONAL CATEGORIES AND THEIR TOTAL DOSE FROM 2010-2013

Year			2010	2011	2012	2013
Occupational category	Nuclear medicine	Physicians	57	61	68	70
		Non-physicians	207	243	252	224
		Total worker	264	304	320	294
		Total dose (mSv)	60.16	80.69	156.54	103.72
	Interventional cardiology	Physicians	46	45	46	42
		Non-physicians	58	50	49	49
		Total worker	104	95	95	91
		Total dose (mSv)	108.29	127.50	208.03	150.15
	Radiotherapy	Physicians	78	70	64	67
		Non-physicians	123	112	115	104
		Total worker	201	182	179	171
		Total dose (mSv)	52.49	31.67	36.32	52.14
	Radiology	Physicians	42	32	56	55
		Non-physicians	151	124	174	167
		Total worker	193	156	230	222
		Total dose (mSv)	18.91	42.05	20.38	62.28

It is found from Fig. 1 that the average dose is received by the radiation worker working in radiology department was the lowest (0.18 mSv) on the other hand the highest average dose was received by the radiation worker working in interventional cardiology department (1.54 mSv).

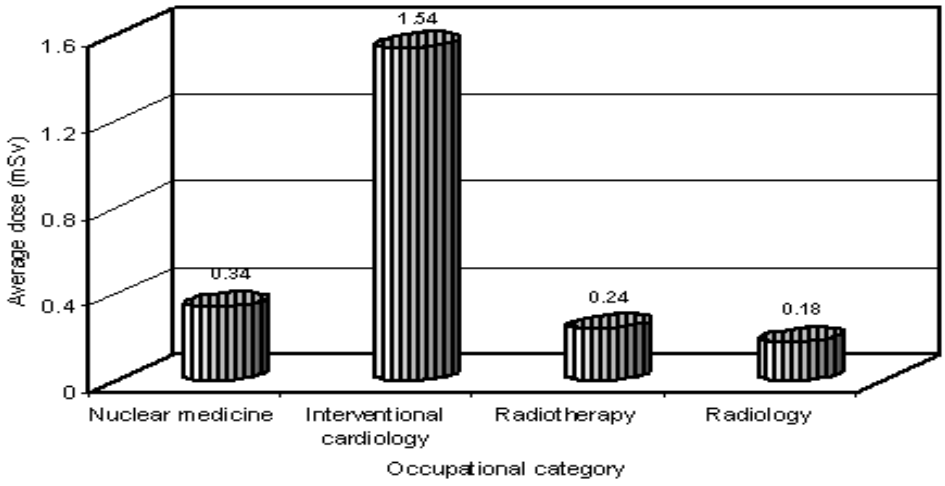

FIG. 1. Average dose received in four occupational categories, from 2010 o 2013 inclusive.

Fig. 2 shows the range of dose distribution pattern for all workers during the period 2010–2013. Doses less than 0.05 mSv after subtracting the background radiation doses, 320 μSv, in three months period is recorded as less than MDL. The distribution is left skewed towards low doses in accordance with the distribution pattern described by UNSCEAR [4] and similar to the findings by L.Currivan et al [11], the conclusion of which is that most occupationally exposed workers receive very low doses with only a small number receiving high doses.

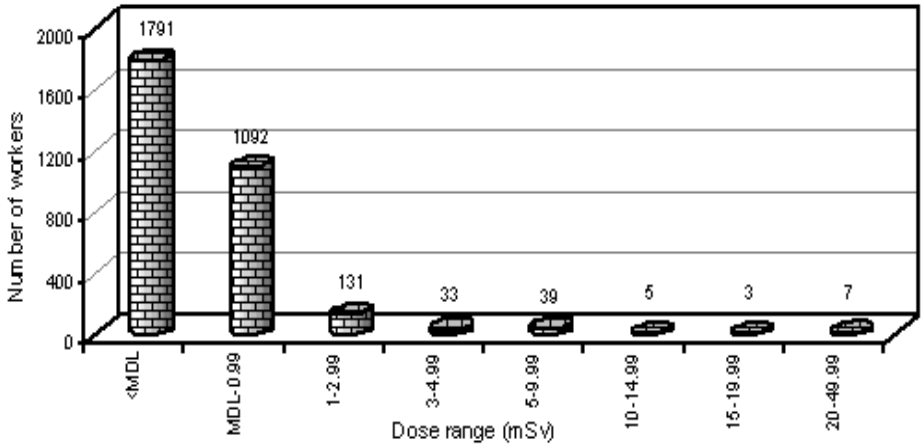

FIG. 2. Distribution of workers on different dose range between 2010 and 2013 inclusive.

Average dose distribution of physician and non-physician radiation workers was summarized in Fig. 3. It is observed that the average dose received by non-physician was higher in consecutive three years from 2011 to 2013 except in 2010 when physician received higher average dose. The maximum average dose was 0.50 mSv in 2012 and the minimum average dose was 0.06 mSv in 2011.

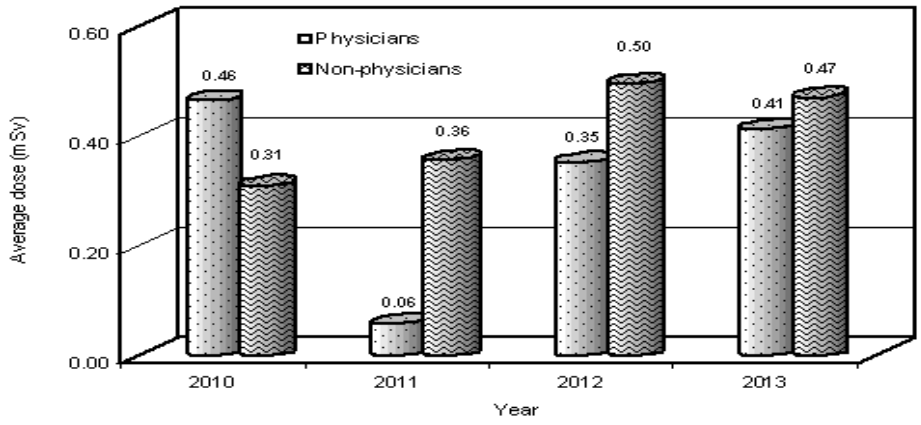

FIG. 3. Yearly average dose distribution of physician and non-physician.

In the study period from 2010 to 2013, maximum dose received in each year was mentioned in the Fig. 4. The highest maximum dose received in 2013 was 49.37 mSv and the lowest maximum dose was 10.32 mSv received in 2010. It is observed that the first two years (2010 and 2011) physicians received maximum dose and in the last two years (2012 and 2013) non-physicians received maximum dose.

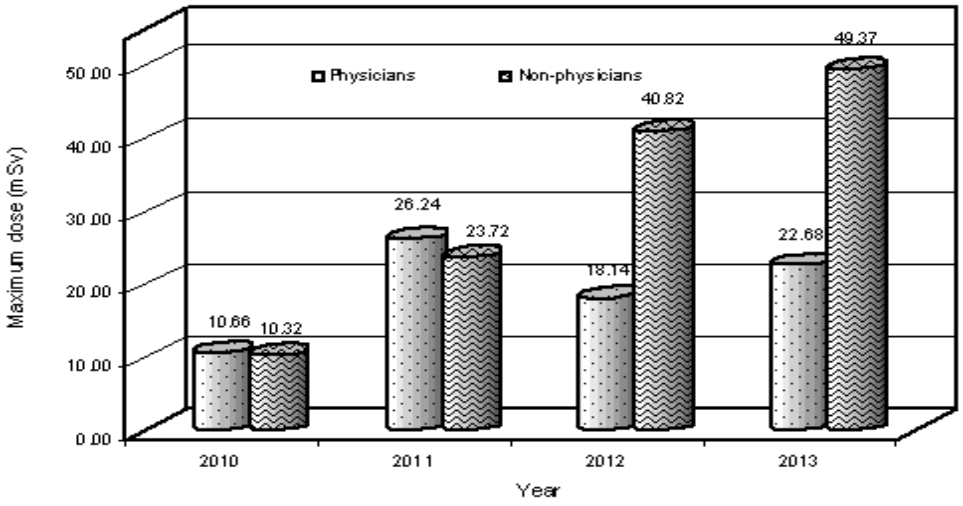

FIG. 4. Maximum dose received in each year from 2010 to 2013.

Within the study period, total of seven radiation workers (two physicians and five non-physicians) were exceeded the average annual dose limit of 20 mSv, but none of them had crossed the maximum annual dose limit of 50 mSv.

4. CONCLUSIONS

Radiation doses were evaluated for the occupational workers in forty nine organizations in four occupational categories namely nuclear medicine, interventional cardiology, radiotherapy and radiology during the period 2010-2013. Most occupationally exposed workers received very low doses with only a small number receiving high doses. Among the exposed workers it was observed that the highest average dose was received by the workers in interventional cardiology department. Even though seven radiation workers crossed average annual dose limit and the highest recorded dose was 49.37 mSv, no worker exceeded the maximum annual dose limit. Finally, it can be seen that though the dose is within the maximum annual dose limit, the workers should pay more attention to radiation protection procedures and practices to keep radiation dose as low as possible, below the average annual dose limit.

ACKNOWLEDGEMENT

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REFERENCES

[1] GARD, E.T., Healthful radiation, Health Phys. 72(1) (1997) 97-99.
[2] GROEDEL F.M., LINIGER H., LOSSEN H., Materialsammlung der Unfälle und Schäden in Röntgenbetrieben, Heft 1, Lucas Gräfe & Sillem, Hamburg, (1925).
[3] KOENIG T.R., METTLER F.A., WAGNER L.K., Skin injuries from fluoroscopically guided procedures: part 2, review of 73 cases and recommendations for minimizing dose delivered to patient
AJR, Am. J. Roentgenol., 177(1) (2001) 13-20).
[4] UNSCEAR, Sources and Effects of Ionizing Radiation, 2000 report to the General Assembly

with scientific annexes, United Nations, New York (2000).
[5] EURATOM, COUNCIL DIRECTIVE 97/43, On health protection of individuals against the dangers
of ionizing radiation in relation to medical exposure, and repealing Directive 84/466/Euratom, Official
Journal of the European Communities. L 180/22. (1997).
[6] UNSCEAR, Sources and effects of ionizing radiation. Report to the General Assembly, with scientific
annexes, New York, UN. (1993).
[7] HARSHAW MODEL 4500 Manual TLD Workstation Operator's Manual (Publication No. 4500-0-
O-0598-002, 6801 Cochran Road, Solon, OH 44139) (1998).
[8] A.F. MCKINLAY, Thermoluminescent Dosimetry (Bristol: Hilger) (1981).
[9] IAEA TLD postal quality audit for SSDL, August (2001).
[10] IEC REPORT 1066, Geneva: IEC (1991).
[11] L. CURRIVAN, H. DONNELLY, D. DAWSON, D. SPAIN, AND P. A. COLGAN, Analysis of whole-body doses received by occupationally exposed workers in Ireland (1996-1999), Radiat.
Prot. Dosim. 96 (No. 1-3) (2001) 53-56.

PILOT STUDY ON EYE LENS DOSES TO INTERVENTIONAL CARDIOLOGISTS DURING CA/PTCA PROCEDURES IN SLOVAKIA

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Abstract

According to the recommened new occupational dose limit for the lens of the eye we focused our attection to the identification of high dose fluoroscopic areas where the eye lens doses of the medical staff may potentialy exceed the recommended limits. In the pilot study was estimated the radiation load of eye lenses among interventional cardiologists performing CA/PTCA procedures in the Department of Interventional Cardiology in Slovakia where the frequency of the chosen interventional procedure is prevailingly used. The exposures of eye lens and the whole body doses were measured for three interventional cardiologists by TLD`s and RaySafe dosemeters during two measuring periods. Presented results have shown that although the whole-body annual doses obtained by the followed cardiologists does not exceed the annual limit of effective dose, the equivalent doses to the lens of the eye obtained from TLD`s reached the overflow value. The results also confirmed the need of the eye lens dose monitoring and optimisation of the used protection tools, as well as of the performance and time reduction of the interventional procedure.

1. INTRODUCTION

The recent studies on health effects of ionizing radiations concluded that radiosensitivity of eye lens is higher than expected until now [1].As a consequence, the current limit of 150 mSv was lowered to 20 mSv per year with possibility to average it over 5 years [2]. It is suspected that this value will be exceeded mainly in the interventional cardiology. In the presented study, an attempt was made to investigate the radiation load of lens of the eye in a group of cardiologists with prevailing treatment of patients by CA/PTCA interventional procedure. The use of TLD dosimetry was chosen to evaluate H_p (0,07) placed close to the eye on protection glasses. Simultaneously, real time whole body doses were individualy meassured by RaySafe dosemeter, located above the apron. The dosemeters were used during one month period. The results have shown that highest doses of eye lens were observed on the left side and there are large differences between the eye lens doses even when the workload is similar. It is possible significantly to exceed the proposed new limit for lens of eye even when the individual whole body dose does not reach 3/10 th of the limit

2. METHODS

For assessment of radiation exposure of the eye lens, the data were collected from the Department of Interventional Cardiology in the particular Cardiologic Healthcare Facility in Slovakia (working name Cardiologic Healthcare Facility C - CHF C), mainly executing

percutaneous transluminal coronary angioplasty (PTCA). The eye lens doses have been measured since12.04.2013 (08:00 h) to 02.01.2014 (16:30 h), preceeded by laboratory preparation of thermoluminiscent dosimeters (TLD`s) (annealing, packaging, indication). It included 242 interventional procedures. After annealing, 3 pieces of TLD were packed into the translucent capsule, colour-marked, separately for the right and left eye. The type of used TLD was LiF: Mg, Ti (TLD 100 Harshaw) with dimensions of 3.1 x 3.1 x 0.8 mm. Consequently, we stuck TLD dosimeters to the left and right side of the protective lead glasses for three (out of five) interventional cardiologists in CHF C. The types of glasses and location of TLD`s can be seen at the Fig. 1.

FIG. 1. The location of the TLD`s on the protective glasses

Evaluation of dosimeters was performed by HARSHAW TLD 3500. Collected parameters of each individual examination were expressed in the quantity - personal dose equivalent Hp(0,07). Simultaneously during the measurement, the following parameters were registered continuously, i.e., Dose Area Product (DAP) values, the fluoro time, the air KERMA values and the number of series.

The whole body doses during the CA/PTCA procedures were controlled by RaySafe i2 dosimetry system, allowing the collection of real time radiation exposure of the medical staff.

3. RESULTS

Values of personal dose equivalent Hp(0,07) on the left eye lens (where we observed higher dose values), were extrapolated to annual doses and compared with the new eye lens limit, recommended by BSS IAEA [2] and 2013/59/EURATOM Directive [3]. In the Fig. 2, the results of this comparison is shown, by calculating the average annual dose from the gathered annual workload of each cardiologist. The results obtained indicate that 67% of interventional cardiologists would exceed the new proposed limit for eye lens doses (20 mSv /year).

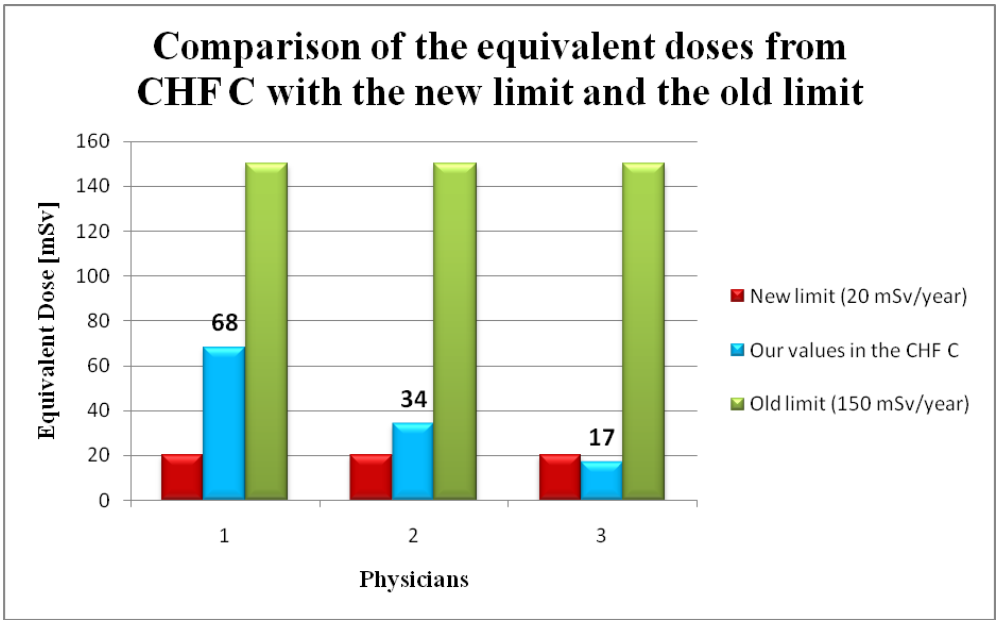

FIG. 2. Comparison of equivalent doses of lens of the eye for interventional Cardiologists in CHF C with the proposed new limit

Important information comes from the results of RaySafe measurements, which refer to the fact, that although the whole-body annual doses obtained by the cardiologists do not exceed the annual limit on effective dose, the equivalent doses to the lens of the eye obtained from TLD reached the limit value (Fig. 3).

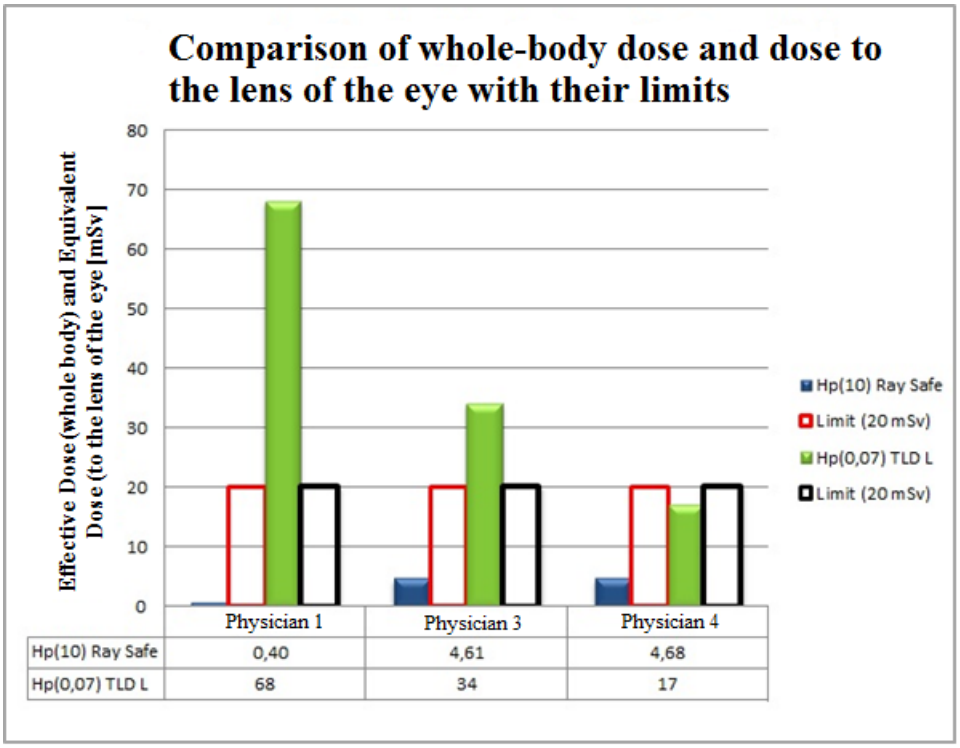

FIG. 3. Comparison of the annual whole body effective dose and the equivalent dose to the eye lens with the annual limit for interventional cardiologists in CHF C during CA/PTCA procedures

4. DISCUSSIONS

During our investigation, it was observed that there were broad range differences in the measured values as a function of used equipment, of applicated dose area product values, of fluoroscopic time, of available protection tools and also of BMI of patients. These differences were observed also when only one type of interventional precedure was applied. This indicates the significant role of training and education of the medical staff as well as the optimization [4]. Sine different models of lead glasses have different dose reduction abilities, they should be individualy tested and used to ensure optimal reduction of the eye lens dose.

5. CONCLUSIONS

The pilot study of estimation of radiation dose to the eye lens of interventional cardiologists demonstrated big variations in the doses, probably due to individual working methods and lack in radiation protection knowledge and training. Further impovement in precise eye lens dose assesment should be made taking into account all exposure conditions, importance of optimization and consistent use of radiation protection tools. Further specification of interventional procedure for monitoring of eye lens doses is necessary, and should be provided in the near future.

REFERENCES

[1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Statement on tissue reactions / early and late effects of radiation in normal tissue and organs – Threshold doses for tissue reactions in a radiation protection context, (2012).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety of radiation sources: International Basic Safety Standards, IAEA, Vienna (2011).
[3] COUNCIL DIRECTIVE 2013/59/EURATOM, Basic safety standards for protection against the dangers arising from exposure to ionizing radiation, Luxembourg (2013).
[4] VANHAVERE, F. et al. International Workshop on Optimalization of Radiation Protection of Medical Staff, ORAMED - Bracelona (2011).

FIRST REPORT OF RADIATION ABSORBED DOSE ASSESSMENT OF RADIATION WORKERS ARISING FROM VARIOUS IMAGING MODALITIES IN DENTAL FACULTY- MASHHAD- IRAN

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Abstract

X- ray imaging modalities are used widely. Dental radiography is an essential tool for accurate diagnosis of dental disease. Although staff & patient doses arising from conventional X-ray imaging are relatively low, but ever increasing application of the new techniques would increase, staff and patient dose. In this study, for the first time, the radiation dose experienced by patients and staffs from different X-ray imaging modalities in use in dental faculty of Mashhad university of medical sciences have been measured. But only staff doses are presented in this paper. To measure an organ dose, TLD-100 chips and standard method of TL dosimetry were employed. Four individual radiographers, responsible for normal interaoral& pre-apical, digital OPG and CBCT imagings were monitored for one month. Altogether 60 TLD-100 chips in plastic sachets were placed on the surface of their laboratory coat (chest, abdomen, and pelvic regions), thyroid shield and their hands. At the conclusion of one working month absorbed dose by TLD were measured. Annual doses of individual radiographers were calculated. The highest annual dose was attributed to the CBCT and the lowest to the pre-aprical technique. Nevertheless The highest annual dose incurred by the staff's organs is less than the annual limit set by ICRP.

1. INTRODUCTION

Various X- ray imaging modalities are now essential tools in medicine and dentistry for fast and accurate diagnosis of diseases and disorders. A high perecentage of exposure to ionizing radiation from man-made sources is attributed to medical and dental applications. In recent time, higher incidence of genetic and carcinogensis effects of ionizing radiation are belived to be intiated following to dental, and maxillofacial radiographies [1, 2]. Although conventional dental radiographies are low dose technique [1, 4] but ever increasing application of X-rays in dental diagnosis in one hand, and introduction of new higher dose techniques, on the other hand have increased staffs and patients dose. Most studies have concentrated their efforts on patient dose monitoring, however the dental workers are exposed to X-ray on a daily basis, which would cause higher potential of harmful effects of radiation

for them. This would be particularly worrying, as no dose thereshold is assumed for stochastic effects, such as cancer and genetic disorders. Childeren are more sensitive to health consequences of ionizing radiation, e.g. thyroid cancer, on the other hand dentists may preseribe more dental radiograph for children and young adults [3].

Although generally, occupational expouser from conventional radiographies (medical & in particular dental) is very low but digital OPG and CBCT techniques may not be considered as very low dose technique. All national and international committees and institutions responsible for radiation protection do emphesise on following ALARA principle for public, patients and radiation workers. Implication of ALARA principle dictates continuous monitoring and assessment of occupational dose of all radiation workers, however it may be thought that their incurred dose is negligible.

2. METHODS

Thermoluminsence dosimetry (TLD) is a technique widely used for radiation monitoring. Different compounds of TLD material, have been utilized, but LiF:Mg; Ti commercially known as TLD-100 is mostly used for medical dosimetry particulary in diagnostic departments.

In this study TLD-100 chips and standard method (including caliberation and annealing) of TL dosimetry commonly used in our center and elsewhere, were employed [4-6]. Four individual radiographers, who were responsible for: normal interaoral & pre-apical, digital OPG and CBCT were selected and monitored for one working month. For each radiographer 15 TLD-100 chips enclosed in plastic sachets were attached to the surface of their body (Labcoat) in the following manner: three at chest region (right, center and left); three at abdominal region (right, centre and left); two at gonadal (right & left) region; three on the surface of theroid shield and finaly two on the wrist region of their right and left hands.

At the end of one working month TLD chips were collected and safely transferred to our dosimetry unit and the absorbed dose were obtained. A Harshaw 3500 mannual TLD reader was used to read the exposed TLDs.

The following imaging equipments were in use while this work was carried out:

Planmeca Intra (2009), Trophy (2003), Proline (PM 2002 CC) and Planmeca Promax 3D. To estimate annual dose of (organs – regions) of interest of the radiographers, mean absorbed dose of the relevant TLDs were obtained. To obtain dose per examination the dose value was divided by the number of examinations performed during the study priod, the average yearly number of various examinations were extracted from the exiciting statistics in radiography department. To estimate annual dose, the dose per examination was multiplied by the average annual number of examinations. And finally the estimated annual doses were compared with ICRP recommended annual limits for occupationally exposed persons. ANOVA, Tukey HSD and SPSS ver. 16 were used for statistical tests of our data.

3. RESULTS

Altogether 60 TLD -100 were used to measure organ doses of the four radiographers monitored for one working month. The estimated annual doses are presented in tables 1, 2, 3 and 4.

TABLE I. AVERAGE (ORGAN/ BODY REGION) OCCUPATIONAL DOSE RECEIVED FOLLOWING INTERAORAL RADIOGRAPHY

Organ/ body region	One working month dose (μSv)			Estimated annual dose (mSv)		
	Right	Center	Left	Per/examination	Per/980 examinations	Annual dose
Thyroid	345.5	393.3	395.8	3.86×10^{-4}	0.378	0.940
Chest	296	350.2	290	3.18×10^{-4}	0.312	0.775
Abdomen	329.3	325.7	301.5	3.25×10^{-4}	0.319	0.792
Gonads	341.6		318.65	3.37×10^{-4}	0.330	0.820
Hands	307.7		306.5	3.13×10^{-4}	0.307	0.763

TABLE 2. AVERAGE ANNUAL (ORGAN/BODY REGION) OCCUPATIONAL DOSE RECEIVED FOLLOWING ANALOGUE PRE-APICAL RADIOGRAPHY

Organ/ body region	One working month dose (μSv)			Estimated annual dose (mSv)		
	Right	Center	Left	Per/examination	Per/260 examinations	Annual dose
Thyroid	301.6	320.5	331.6	12.23×10^{-4}	0.318	0.440
Chest	374.8	409.4	380.2	14.93×10^{-4}	0.388	0.537
Abdomen	362.6	354.3	402.3	14.35×10^{-4}	0.374	0.517
Gonads	380.7		324.7	13.57×10^{-4}	0.353	0.488
Hands	311.8		360.6	13.77×10^{-4}	0.358	0.496

TABLE 3. AVERAGE ANNUAL (ORGAN/BODY REGION) ACCUPATIONAL DOSE FOLLOWING DIGITAL PANORAMIC EXAMINATIONS

Organ/ body region	One working month dose (μSv)			Estimated annual dose (mSv)		
	Right	Center	Left	Per/examination	Per/248 examinations	Annual dose
Thyroid	385.2	301.2	322.1	13.56×10^{-4}	0.336	1.254
Chest	436.3	400.9	359.3	16.08×10^{-4}	0.399	1.488
Abdomen	413.5	377.1	389.4	15.86×10^{-4}	0.393	1.467
Gonads	422.4		339.4	15.76×10^{-4}	0.382	1.458
Hands	379.6		325.1	14.21×10^{-4}	0.380	1.417

TABLE 4. AVERAGE ANNUAL (ORGAN/BODY REGION) ACCUPATIONAL DOSE FOLLOWING CBCT EXAMINATION

Organ/ body region	One working month dose (μSv)			Estimated annual dose (mSv)		
	Right	Center	Left	Per/examination	Per/ 50 examinations	Annual dose
Thyroid	334.5	385.3	316.2	69.07×10^{-4}	0.345	1.658
Chest	338.5	312.2	295.2	63.06×10^{-4}	0.315	1.513
Abdomen	315.2	343.5	368.6	68.49×10^{-4}	0.342	1.644
Gonads	391.2		310.35	70.16×10^{-4}	0.351	1.684
Hands	312.2		288.5	60.07×10^{-4}	0.300	1.442

4. DISCUSSION

Our results in Tables 1 to 4 are evident that:

(a) Highest and lowest organ dose per examination is received from CBCT and intraoral examinations respectively Table 1 and 4.

(b) The difference between thyroid doses arising from the four different modalities are significant (p-value <0.001). This conclusion is true for all examind organs/body regions Table 1, 2, 3 and 4.

(c) Thyroid dose from CBCT technique was significantly higher than the corresponding values obtained from applications of other three techniques Table 1, 2, 3 and 4.

(d) Highest and lowest annual organ/body regions dose is received from CBCT and analogue pre-apical radiographies respectively Table 2 and 4.

(e) In this study the highest annual organ dose arising from digital panoramic imaging was delivered to chest regon while thyroid received the lowest dose (Table 3), but Gibels et al in their study reported that annual thyroid dose per technician radiographer was less than the gonadal dose of the same person [1].

(f) The results of Goren et al for finger, chest, eyes and gonads doses of dental radiographers were less than the annual dose limits recommended by ICRP and are in compliance with our data [7, 8].

5. CONCLUSION

Although digital panaromice and CBCT deliver higher organ doses to radiation workers, these values are by far much lower than the annual limits set by the ICRP. This is in compliance with other studies carried out elsewhere [1].

REFERENCES

[1] F. GIJBELS et al, Dosimetry of digital panaromic imaging, Part II: Accupational exposure DMFR, Vol. 34, Issue 3, May (2005).

[2] K. HORNE, P.N. HIRSCHMAN, Dose Reduction in Dental Radiography, Journal of Dentistry,18(4) (1990) 171-184.

[3] F. A. PICKETT, Reduce radiation risk, The Journal of Professional Excellence; Dimension of Dental Hygiene, April (2013).

[4] F. AKBARI, Evaluation of Local LDRL for panaromic radiography in the Greater Khorasan Province, M. Sc. Thesis, Mashhad university of Medical Sciences, Mashhad, Iran (2009).

[5] M. ASADINEZHAD, national Survey: of Patien doses from routine diagnostic X-ray examinations in Iran, Ph.D. Thesis, Mashhad University of Medical Sciences, Mashhad, Iran, (2007).

[6] M. MOJIRI, An assessment of organ doses in some conventional radiographies by the means of TLD-rando Phantom, DAP Meter and ODS-60 Software, Mashhad University of Medical Sciences, Mashhad-Iran (2004).

[7] A.D. GOREN et al, Radiation exposure with the NOMAD portable X-ray system, Dentomaxillofacial Radiology, 37(2) (2008) 109-12.

[8] VALENTIN J., The 2007 recommendations of the international commission on radiological Protection, Published for the International Commission on Radiological Protection by Elsevier, (2007).

MAFRAQ HOSPITAL EXPERIENCE IN OCCUPATIONAL RADIATION MONITORING

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ABSTRACT

Staff radiation Dosimetry and monitoring is mandatory for all radiation occupants in medical, industrial and other radiation facilities. Eight to ten thousands staff members in Abu Dhabi Emirate are considered to be radiation workers, covering most of private and governmental facilities.

In 2009, the plan arose to establish an in-house radiation Dosimetry service (RDS) for Mafraq Hospital radiation workers, where time and money is saved as well as necessary confidentiality of the radiation records is created.

In 2010, Abu Dhabi Health authority (HAAD) issued a license for Mafraq Hospital to practice RDS for all radiation occupants for Abu Dhabi Emirate, and to provide personal radiation monitoring service for all personnel associated with radiation environment of industrial and healthcare facilities.

The RDS Lab. equipped with high technology readers, renders service for more than 5000 radiation occupants in Abu Dhabi Emirate. After 5 years of experience we succeeded to create an awareness of radiation protection and safety among radiation workers and theirs administrators; facilitate and lighten the licensing procedures for radiation facilities; achieve an on-time badges delivery and report; overexposure alert and emergency situations. RDS is facing challenges and working hard to be lined with the rapid development in nuclear energy in UAE.

1. INTRODUCTION

In 2010, there were numerous improvement projects running in parallel as part of transforming the hospital to international standards. For Radiology Department, various types of data were gathered and different quality tools were used to assess the gaps and opportunities for improvement in order to transform it to international standards.

Radiation safety is the ultimate goal for the multidisciplinary team which deals with radiation. At the radiology department level the suggestion of establishing "In house Dosimetry Service" was strongly agreed upon and submitted to the hospital executive team. The Chairman of Clinical Imaging Department (CID) was tasked by the executive team to develop a strategy and action plan to execute the project considering SEHA approval.

This fact gave a sense of priority to this project as it addresses staff radiation safety issues that should not be compromised and it highlights the importance of introducing accurate and reliable "In house" Radiation Dosimetry (TLD) service.

2. CRITICAL ELEMENTS THAT CONTRIBUTED TO THE PROJECT SELECTION

The critical elements that contributed to the project selection are:

(a) Ensure staff radiation safety: Over exposure cases are required to be investigated to determine the root cause and corrective measures to prevent reoccurrence.
(b) Unsatisfactory service from third party Dosimetry Services such as delay in sending reports, using old technology and standards, concerns about reliability/accuracy of results.
(c) Lack of standardized Dosimetry service across SEHA entities: Different SEHA entities deal with different Dosimetry service providers thus the radiation Dosimetry "standards" are not unified.
(d) Optimum radiation safety standard: best practices in radiation safety (Federal Authority of Nuclear Regulation (FANR)) set the threshold for investigating radiation exposure at 0.6mSv while service provided by third party set the threshold at 1.5mSv, which indicates the risk of receiving "false negative" results and missing overexposed staff..
(e) The Government and SEHA direction toward establishing Abu Dhabi as world class level healthcare provider encourages the hospital aim of establishing a trusted "In house Dosimetry service".
(f) The project team conducted a brain storming session to identify the project's potential stakeholders and link it to organization stakeholder analysis as per the strategic plan.

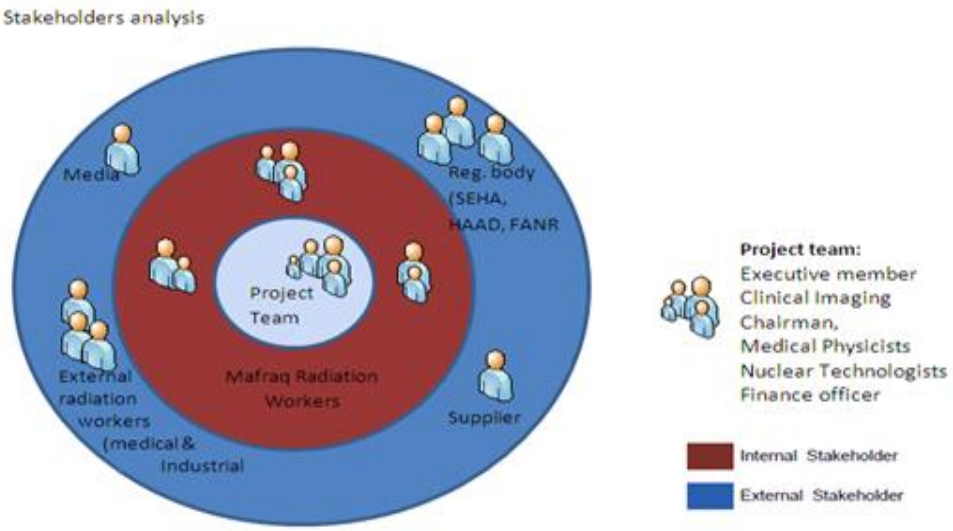
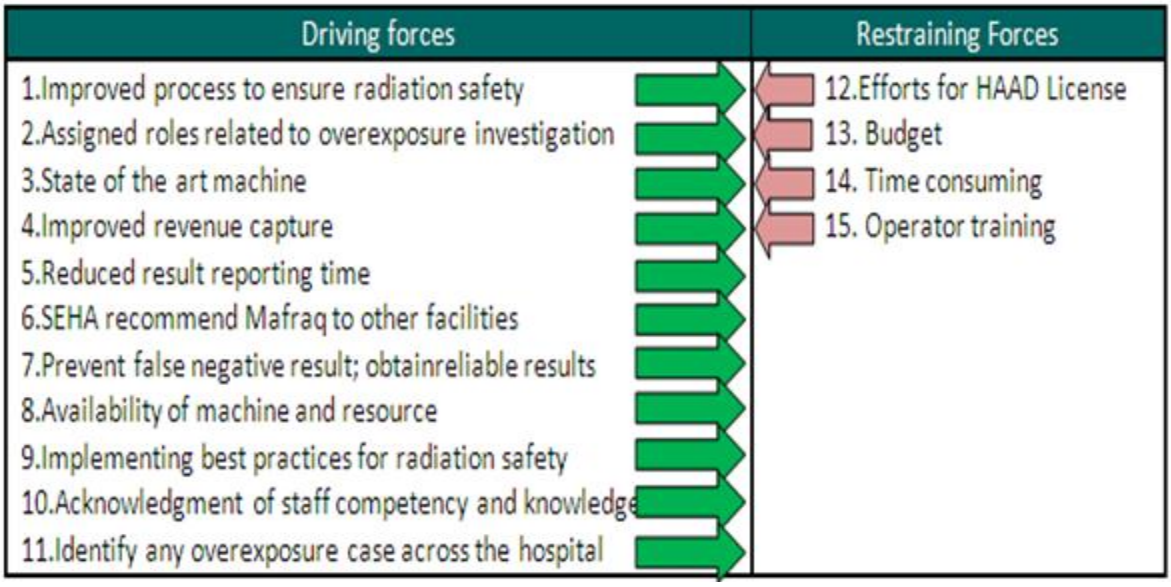


FIG. 1. Stakeholders Analysis

The impact of the project varied for the different stakeholders, and the project team decided after a brainstorming session to use a simple force field analysis of the project, then to link identified driving and restraining forces on each stakeholder and assess if the impact is direct, Indirect, positive, or negative (refer to 1Cc). The analysis recognized many driving forces (positives) that can overcome the restraining forces.

TABLE 1. FORCE FIELD ANALYSIS



3. METHOD

The management team led by the executive member conducted an initial assessment & gap analysis from January till September 2010 to identify the current state of the Department and areas for improvement. The project team, through brainstorming sessions, conducted a SWOT analysis to evaluate the strengths, weaknesses, opportunities and threats of implementing a Radiation Dosimetry service in order to identify the key internal and external factors that are important to achieving the project.

The team brainstormed and identified potential causes and listed them onto a Fish Bone Diagram to illustrate the cause and effects of all issues. The potential causes were classified into five main categories (People, Process, Policy/Procedures, Training/ Systems).

4. DISCUSSION

The information gathered was discussed by the team members to reach consensus. The team focused on weaknesses and threats from the **SWOT** analysis and looked into strengths and opportunities to overcome the challenges and have a sustained, positive impact on the outcomes of the project.

TABLE. 2. SWOT ANALYSIS

Strengths	Weaknesses
• Management support • Knowledgeable and competent radiology team • Availability of personnel dosimeter system (**Harshaw 4500 TLD** Reader, dosimeters cards, irradiator) • Commitment towards safety (for patients and employees) • Government stresses on safety and upgrading the standards to international best practices. • Compliance and exceeding HAAD, JCI, FNAR, regulations.	• Depending on external company to cover dosimetry services for Mafraq • Delay in getting results reports • Annual fees for the contracted service • Poor utilization on internal resources and equipment (**Harshaw 4500 TLD** Reader classified as condemned item after closing radiotherapy service) • Number of cases reflected overexposed staff • External company use 1.5mSv as threshold for overexposure investigation while FNAR use 0.6mSv as threshold for investigation. • The process does not involve medical physicist in receiving and investigating overexposure related to staff working outside radiology department.
Opportunities	**Threats**
• Establish in house lab to provide the service (Phase 1: cover hospital staff need) • Establish Mafraq as center to provide this service (Phase 2: cover SEHA facilities and different community needs) • Provide educational activities to increase staff awareness regarding radiation safety • Establish Mafraq as center for radiation safety excellence • Competitive value: only one local facility in Dubai is providing dosimetry service, different facilities are depending on external labs outside UAE.	• Poor management of radiation exposure threatens staff and patient safety • Relying on third party contradicts with direction to cost cutting and proper efficient way of resource utilization. • Budget constraints • Lack of trust from staff regarding the efforts to ensure their safety

The team developed an action plan to address areas for improvement identified from the SWOT Analysis and Cause & Effect Diagram. Simultaneously, benchmarking with best practice and standards was used as one method proposing and choosing solutions. The team membership included a representative from finance to advise the team regarding all financial aspects of the project and proposed solutions.

The project team conducted simple market study to understand who provides dosimetry service and what service other institutions are receiving.

TABLE 3. MARKET ANALYSIS FOR TLD SERVICE PROVIDERS

Change of time	Clarification
1983 – Sep.2010	Since Hospital establishment MOH TLD service was used,
Jan.2010 - Sep.2010	Still with MOH, but a second TLD badge was given to selected Mafraq radiation workers(e.g. Cath. Lab. Staff), And used available Harshaw reader.
Oct.2010 to Now	100%relialance on Mafraq TLD service (established Radiation Dosimetry Lab.)
March 2011 to Now	Introduce Panasonic TLD multi readers (latest technology)
Sep. 2011	Start to render service to other facilities(private and governmental, medical , industrial and others)
Nov. 2011	Cover most of SEHA health facilities

5. RESULTS

There are tangible and intangible realized results, shown below:

5.1. Tangible Results

(a) Radiation workers Satisfaction:
(b) Increase in revenue captured:
(c) Decrease in reporting timeframe
(d) Enhanced the accuracy of the personnel dosimeter
(e) Standardized the Dosimetry service for other SEHA radiation workers.
(f) Utilize the service to cover (governmental & private) medical and industrial radiation workers.
(g) Promote opportunity for Research and Education (Table 4).
(h) Enhance the reputation of SEHA and the hospital as employer of choice, and increase sense of loyalty.
(i) Establish competitive value in the market for Mafraq Dosimetry service.

5.2. Intangible Results

a) Improved the team knowledge/motivation
b) Heightened sense of team spirit
c) Created a new sense of collaboration between SEHA and other facilities
d) Prestigious status to be the first licensed center in Abu Dhabi
e) The project team shared the results with senior management team and leadership council during regular meetings and while reviewing financial statements.
f) The results which are related to external stakeholders and customers are communicated through Dosimetry Service team.
g) The external survey result carried numerous positive comments from the customers regarding their satisfaction from the support and professionalism of Mafraq team.

TABLE 4. PROMOTING RADIATION EDUCATION AND RESEARCH

Research	**Experimental**	Research on measuring the whole body effective dose with two dosimeters (one above the lead apron and the other one underneath) and compare it to the effective dose from one dosimeter (below the lead apron). o *Outcome of this experience :* Under process
	Investigational	As an action taken after finding one overexposure case the employee claimed that to have accidently left the dosimeter hanging on the coat in the x-ray procedure room for one week. Two TLD badges kept in the same place for the same period and the result was analyzed. o *Outcome of this experience:* -The scenario that the employee gave was incorrect. It was later discovered that the employee had a CT procedure with her TLD badge on. -Other staff working at the same area were assured after knowing that scatter radiation in the room is within normal range and they are not in danger if they spent long time in the room.
Education	**Radiation Safety conference**	Conference on patient radiation safety with local and international speakers. The conference was held in conjunction with The Federal Authority for Nuclear Regulation (FANR), International Atomic Energy Agency (IAEA) and the Dubai Health Authority (DHA). The 2 day conference covered radiation safety education and training; measures to minimize radiation exposure and evaluation to ensure regulations are applied. Further details available in Appendix.

6. CONCLUSION

The project team members were selected based on their knowledge and experience related to the topic. The project required administrative skills, technical knowledge/skills and financial expertise. Thus, the team included Chief Operating officer and CID Chairman as Administrative experts. The medical radiation technologists and medical physicist were the technical experts while the finance officer helped in the financial issues.

The project team members were involved in all meetings/discussions to ensure that all members are aware of the objectives and the strategy that will be followed to implement the service.

The team designed SMART action plans with clear roles/time-frames to keep the project moving forward. Regular meetings conducted to ensure proper execution of the plan. The team was assessing the progress of the project as every successful step leads to the next step.

REFERENCES

[1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, ICRP, 1990 Recommendations of the International Commission on Radiological Protection, ICRP Publication 60. Ann. ICRP 21(1-3) (1991).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Optimization of radiation protection in the control of occupational exposure, Safety Reports Series 21, IAEA, Vienna (2002).
[3] FEDERAL AUTHORITY FOR NUCLEAR REGULATION, Radiation dose limits and optimization of radiation protection for nuclear facilities - the Federal Law by Decree No 6 of 2009, FANR-REG-04, Concerning the Peaceful Uses of Nuclear Energy (2009).

OCCUPATIONAL EYE DOSES FROM DIAGNOSTIC & INTERVENTIONAL PROCEDURES IN AN IRISH HOSPITAL SETTING
A REVIEW OF FINDINGS FROM 2011 – 2014

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Abstract

Radiation protection of the eye lens has received considerable attention since the International Commission on Radiological Protection (ICRP) recommended a significantly reduced eye dose limit in April 2011. In February 2011, a series of planned investigations were commenced to establish eye lens doses in four key clinical areas in two large Irish acute hospitals. Over the last three years, eye doses have been assessed in (i) Gastroenterology (specifically ERCP procedures), (ii) Interventional Radiology, (iii) Interventional Cardiology and (iv) Positron Emission Tomography (PET). In total, eye lens doses to 30 staff members from more than 1,900 X-ray and PET procedures were monitored using a dedicated eye lens dosimeter. Results are presented in terms of the personal dose equivalent $H_p(3)$. All doses were monitored above the lead glasses (where worn) and represent dose to the unprotected eye lens. The results show that eye lens doses in PET are relatively low and well within the new ICRP limit of 20mSv/y, while eye lens doses from Gastroenterology X-ray procedures (ERCP) can be significant if the X-ray tube is used overcouch. Eye lens doses to Interventional Radiologists and Interventional Cardiologists may exceed 20mSv per year, if adequate eye protection is not worn. The data will contribute to the growing body of measured eye lens doses.

1. INTRODUCTION

Interventional Radiology (IR) and Interventional Cardiology (IC) procedures can result in occupational radiation doses which are sufficiently high to warrant concern [1-4]. PET is considered to be a high dose diagnostic modality and staff doses are typically higher than in other nuclear medicine services [5]. There has been a significant focus in recent years on occupational eye doses, due to the 2011 ICRP statement on tissue reactions [6]. The ICRP recommended an equivalent dose limit for the lens of the eye of 20 mSv/y, and this is a considerable reduction from the previous dose limit of 150mSv/y. Since this statement was released, awareness in the scientific literature has grown at a rapid pace and the need for improved eye lens dosimetry has been acknowledged [7, 8]. Although the rationale for reducing the eye lens limit has been debated in the literature [9, 10], the limit is now on a firm footing within Europe as it has been adopted into the new EU basic safety standards Directive [11]. The Directive must be transposed into national legislation by EU member states within a period of four years. Efforts to establish reliable estimates of eye lens doses prior to legislative changes will assist with the transition to this significantly lower limit. The goal of this study was to obtain reliable estimates of eye lens doses in terms of $H_p(3)$ to PET and interventional staff in two Irish hospitals.

2. METHODS

A newly designed eye dosimeter was used [12 – 14] to measure eye dose in terms of

$H_p(3)$ in four different diagnostic / interventional specialties across two Irish hospitals as shown in Table 1 below.

TABLE 1. DESCRIPTION OF CLINICAL SPECIALTIES INCLUDED IN THIS EYE DOSE MONITORING STUDY

Clinical Speciality using Ionising Radiation	Brief description of clinical work assessed	Staff monitored
Gastroenterology	Endoscopic retrograde cholangiopancreatography (ERCP) (Diagnostic and Therapeutic procedures)	— Gastroenterologists — Nurse assisting patient — Nurse assising Doctor
Interventional Radiology (IR)	Angiography / embolization, biliary and genitourinary procedures, drainages and line placements	— Interventional Radiologists
Interventional Cardiology (IC)	Coronary angiography and angioplasty, pacemaker insertion, electro-physiology (EP) studies and Transcatheter Aortic Valve Implantations (TAVI)	— Interventional Cardiologists — Cardiology Clinical Nurse Specialist
PET	FDG F-18 oncology imaging	— Radiographers — PET Nurse — Healthcare Assistant — Physicist

A dedicated eye lens dosimeter (EYE-D™, RADCARD, Krakow, Poland) designed and calibrated to measure $H_p(3)$ was issued to staff participating in this study. The dosimeter consists of a TLD pellet (MCP-N, LiF:Mg,Cu,P) in a plastic (polymide) capsule. Following each measurement period, all dosimeters (including small batches of unirradiated transport dosimeters) were returned to RADCARD in Poland to be read out.

Eye dosimeters were worn using two different methods for attachment. For the Gastroenterology and PET studies, the dosimeter was worn on an adjustable headband that is provided by the supplier, with the dosimeter positioned adjacent to the left eye. Lead glasses were not worn in either of these two clincial areas. For the IR and IC studies, we received feedback that wearing the TLDs on a headband was difficult to tolerate therefore the TLDs were affixed to the outside arm of a pair of glasses to overcome some of the issues. In most cases, lead glasses were used, however, one IR staff member wore the TLD affixed to ordinary prescription spectacles.

The focus of this study was occupational eye lens dosimetry. Nonetheless as staff exposure is strongly correlated to patient exposure, a record of Kerma-Area Product (KAP) (Gy cm^2) and fluoroscopy screening time was recorded for each X-ray examination. For PET staff, relevant data on patient workload was recorded.

3. RESULTS

A summary of results from the four specialities is shown in Table 2 below. Eye dosimeters were worn for periods from 6 weeks (Gastroenterology) up to 13 weeks (IR, IC and PET). Results for the monitoring period were then extrapolated to estimate annual eye doses based on typical workloads. Further analysis was also carried out to determine eye dose per procedure and eye dose per unit KAP. In total, eye doses to 30 staff from more than 1,900 X-ray and PET procedures were monitored. This comprised of approximately 1,000 PET

patients injected with FDG ^{18}F and 900 interventional X-ray procedures resulting in a total patient KAP of more than 40,000Gy cm^2.

TABLE 2. ANNUAL EQUIVALENT DOSE THE LENS OF THE EYE FROM INTERVENTIONAL AND PET PROCEDURES IN TWO IRISH HOSPITALS

Clinical Speciality using Ionising Radiation	Staff Monitored	Annual equivalent dose to the eye lens, $H_p(3)$ (mSv)
Gastroenterology[a]	Gastroenterologists	Mean: 11.7 (overcouch X-ray, left eye) Mean: 1.3 (undercouch X-ray, left eye)
	Nurse assisting Doctor	Mean: 2.6 (overcouch X-ray, left eye) Mean: <1.3 (undercouch X-ray, left eye)
	Nurse assising patient	Mean: 3.9 (overcouch X-ray, left eye) Mean: <1.3 (undercouch X-ray, left eye)
IR	Interventional Radiologists	Mean: 24.7 (Left eye) Range: 7.1 – 44.9 (Left eye) Mean: 14.6 (Right eye) Range: 4.1 – 29 (Right eye)
IC	Interventional Cardiologists Clinical Nurse Specialist	Mean: 12.5 (Left eye) Range: 4.2 – 33.3 (Left eye) Mean: 7.5 (Right eye) Range: 2.4 – 18.9 (Right eye)
PET	Radiographers Physicist Healthcare Assistant PET Nurse	Mean: 0.87 (Left eye) Range: 0.04 – 2.01 (Left eye)

a. For further details see [15]

4. DISCUSSION

In this study we have estimated lens dose to PET and interventional staff in terms of $H_p(3)$ for the first time in an Irish hospital setting. Staff performing interventional procedures should be aware that key factors in reducing eye doses are (i) positioning the X-ray tube under the patient table, (ii) habitual use of ceiling-mounted screens and (iii) consistent use of appropriately fitting lead-protective spectacles. These practices are well established, yet the operators who participated in our study agreed that the attention given to eye dosimetry has encouraged them to revisit their own procedures. We found that where left and right eye dose was monitored (IR and IC), the left eye dose was always higher, as expected based on staff position. In PET, the highest doses were attributed to radiographers.

The eye doses quoted in this study are based on measurements taken over lead glasses, where they were worn. This was decided for pragmatic reasons largely because it is difficult to get the chosen eye dosimeter to fit securely under lead glasses. Furthermore, the exact placement for the dosimeter under lead glasses to measure protection factors in a clinical setting is an area of debate. As such, until protection factors can be accurately verified, we have taken the approach that measurements over lead glasses are repeatable and reliable in terms of positioning, they are unobtrusive and the lens dose below the glasses will be much lower.

Some staff in our study work at more than one location and the eye doses measured here were for one employer only. This means that the actual annual eye lens doses from all workplaces may further exceed the new ICRP limit. It is recommended that there is greater

communication and sharing of personal dosimetry data amongst employers in order to protect the employee from cumulative doses exceeding annual limits.

5. CONCLUSION

Occupational eye doses from IR and IC procedures have the potential to exceed the new ICRP equivalent dose limit of 20mSv per annum, if adequate eye protection is not worn and/or if the X-ray tube is overcouch. An interventional radiologist / cardiologist / gastroenterologist may also exceed the new EU threshold for Category A workers of 15mSv to the lens of the eye. Lead glasses should be considered as an absolute requirement for operators carrying out interventional procedures. The results show that eye doses in PET are relatively low and well within the new ICRP limit. The data will contribute to the growing body of measured eye lens doses. Eye doses to staff performing any fluoroscopically-guided procedures should be measured and kept under review, particularly in light of the reduced ICRP eye lens dose limit.

REFERENCES

[1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Avoidance of radiation injuries from medical interventional procedures, ICRP Publication 85. Ann. ICRP 30(2) (2000).

[2] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Radiological protection in fluoroscopically guided procedures performed outside the Imaging Department, ICRP Publication 117. Ann. ICRP 40(6) (2010).

[3] MILLER DL, VAÑÓ E, BARTAL G et al., Cardiovascular and Interventional Radiology Society of Europe (CIRSE); Society of Interventional Radiology (SIR). Occupational radiation protection in interventional radiology: a joint guideline of the Cardiovascular and Interventional Radiology Society of Europe and the Society of Interventional Radiology, Cardiovasc Intervent Radiol. 33(2) (2010) 230.

[4] REHANI M., VANO E., CIRAJ-BJELAC O., KLEIMAN N. J., Radiation and Cataract, Radiation Protection Dosimetry, Vol. 147 (No. 1–2) (2011) 300–304.

[5] INTERNTIONAL ATOMIC ENERGY AGENCY, IAEA Report 58, Radiation protection in newer medical imaging techniques: PET/CT., IAEA, Vienna (2008).

[6] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Statement on tissue reactions, ICRP 4825-3093-1464, April (2011).

[7] HASKAL Z., Get Protected: The eyes have it, J. Vasc. Interv. Radiol. 24 (2013) 205–206.

[8] INTERNATIONAL RADIATION PROTECTION ASSOCIATION, Report of Task Group on the Implementation of the ICRP recommendations for a revised dose limit to the lens of the eye, IRPA, May (2013).

[9] MARTIN C.J., What are the implications of the proposed revision of the eye dose limit for interventional operators? Br. J. Radiol., 84 (2011) 961–962.

[10] ENGLEFIELD C., Is the new ICRP eye dose limit justified? J. Radiol. Prot., 31(2011) 499-500.

[11] COUNCIL DIRECTIVE 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom. Official Journal of the European Union.; Volume 57. ISSN 1977-0677. Luxembourg (2014).

[12] BILSKI P. BORDY J-M., DAURES J. et al., The new EYE-D™ dosemeter for measurements of Hp(3) for medical staff, Rad. Meas. 46 (2011) 1239-1242.

[13] GUALDRINI G, MARIOTTI F, WACH S. et al., Eye lens dosimetry: task 2 within the ORAMED project. Rad. Prot. Dosim. 144 (2011) 473–7.

[14] BORDY JM, GUALDRINI G, DAURES J, MARIOTTI F., Principles for the design and calibration of radiation protection dosemeters for operational and protection quantities for eye lens dosimetry, Radiat. Prot. Dosim. 144(1-4) (2011) 257-61.

[15] O'CONNOR U, GALLAGHER A, MALONE L, O'REILLY G., Occupational radiation dose to eyes from endoscopic retrograde cholangiopancreatography procedures in light of the revised eye lens dose limit from the ICRP, B. J. Radiol., 86:20120289 (2013).

OCCUPATIONAL EXPOSURE OF MEDICAL STAFF IN UKRAINE: RESULTS OF PERSONAL MONITORING in 1991-2013

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Abstract

The experience of centralized individual monitoring of external exposure for more than 6500 occupational workers in radiation therapy, nuclear medicine, diagnostic radiology and interventional radiology from all regions of Ukraine in 1991-2013 is presented. The dynamics of annual collective and average effective doses for main professional groups of medical staff was analyzed. The groups of medical staff with highest doses of occupational exposure were determined. Radiation risk of stochastic effects of occupational exposure was estimated and possible methods for optimization of radiation protection were identified

1. INTRODUCTION

For fulfill the basic principles of radiation safety and protection of the National Radiation Safety Standard of Ukraine (NRSSU-97) - 'Do not exceed the established dose limits' for staff Category A and for ensuring of optimization of radiation protection, it is necessary to organize the radiation control of occupational exposure, namely the workplaces and personal dose monitoring of radiation workers [1].

Monitoring of working places allows to study a radiation safety conditions during the work with the radiation sources, to identify the technological operations which give the main contribution to the dose of personnel, however, only through individual monitoring one can measure the actual doses to each employee. Therefore, the organization of individual monitoring is a key objective of all programs for radiation monitoring of occupational exposure. For occupationally exposed workers of Category A, the individual monitoring should be based on systematic dose measurements, which must be made by ‘appropriate dosimetry service under an adequate quality assurance programme [1, 2].

Radiation monitoring of occupational exposure of medical staff in Ukraine is carried out from 1979 by Central Laboratory of Radiation Hygiene of Medical Staff (Center Laboratory, Grigorev Institute for Medical Radiology, Kharkiv). Up to now, the Central Laboratory is the only approved dosimetry service for personal monitoring in Ukraine that carries out the measurements, assessment and analysis of medical occupational doses on systematic basis for a long time. In this paper, the results of personnel occupational monitoring of medical staff which working in radiation therapy, nuclear medicine, diagnostic radiology and interventional radiology departments during 1991-2013 are presented.

2. MATERIALS AND METHODS

At present, more than 6500 persons from 670 hospitals and institutions of all regions of Ukraine covered by centralized individual dose monitoring which carried out by Central Laboratory. There are practically all medical workers of radiation therapy and nuclear medicine departments and about 35 % of medical staff involved into the X-ray diagnostics and interventional radiology.

For personal dose monitoring, the Central Laboratory use the quantity of dose equivalent HP(10), mSv. The individual dose monitoring is performed on a quarterly basis. The measurements of personal doses are performed using TL-readers DTU-01 (Latvia) and DTG-01 (Russia) and TL-detectors LiF (Mg, Ti), DTG-04.

The results of personal monitoring are analyzed using the special software and stored in Database IDAIS. This system allows to collect, accumulate, save the individual doses data during all period of monitoring. Personal data in DB for each medical radiation worker has the unique identification, his employment history, employer information, monitoring information, data about quarterly, annual and accumulated doses of each person. Database is updated on a quarterly. This system also provides the opportunity to create different annual reports for sending out to all monitored hospitals, annual total report - to the Ministry of Health of Ukraine and State Nuclear Regulatory Inspectorate of Ukraine. All cases of exceeding the dose limits are analyzed separately.

For characterization and classification of different kinds of work with radiation sources in medicine, 8 categories of medical procedures were identified, i.e., the brachytherapy and manual radiotherapy with sealed sources; teleradiotherapy, X-ray therapy, nuclear medicine; diagnostic radiology; interventional radiology, etc. There are 35 occupational groups of medical staff with unique database codes for preparation of different types of reports with the annual dose results: physicians-radiologists; nurses; medical physicist; engineer; radiation protection specialists, technician and others. The information about the doses received is used by the radiation protection officers in the hospitals and the inspectors of Regulatory Authority of Ukraine for optimization of the radiation protection at different workplaces.

3. RESULTS AND DISCUSSION

For analysis and presentation of the results of personal monitoring all medical personnel were separated into several professional groups:

i. Sum – all medical staff working with gamma-radiation sources–medical radiologists;
ii. A – medical staff carrying out the brachtherapy and manual gamma-therapy;
iii. B - medical staff carrying out the teleradiotherapy;
iv. C – medical staff of nuclear medicine departments;
v. RS-keepers – medical staff working as nurses-handler of radioactive sources;
vi. RPP – radiomanipulation paramedical personnel;
vii. X-rays – medical staff in diagnostic radiology and interventional radiology.

The summary results about collective and mean annual doses and its dynamics during observation years (1991-2013) for different types of work with radiation sources in medicine and for some occupational groups of medical staff of Ukraine are shown at Fig.1 and Fig. 2.

During 1991-2013, the collective dose of group Sum was changed within the range of 1600- 4200 man-mSv. The collective dose of group A consists 55.1-73.9 % in total collective dose of all medical radiologists, groups B and C - 12.3-24.0 % and 7.6-18.7 % respectively (Fig. 1).

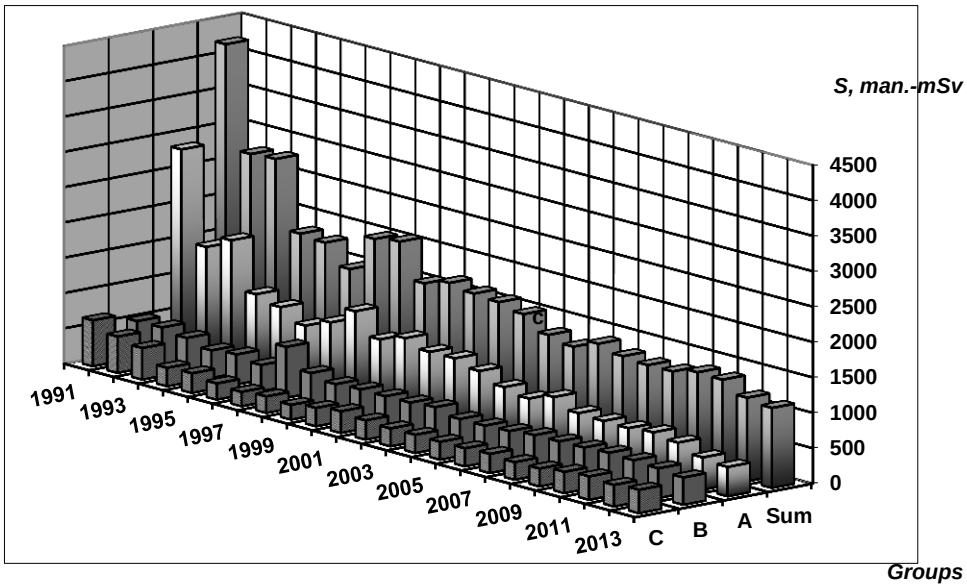

FIG.1. Dynamics of collective doses of medical radiologists of Ukraine in 1991-2013

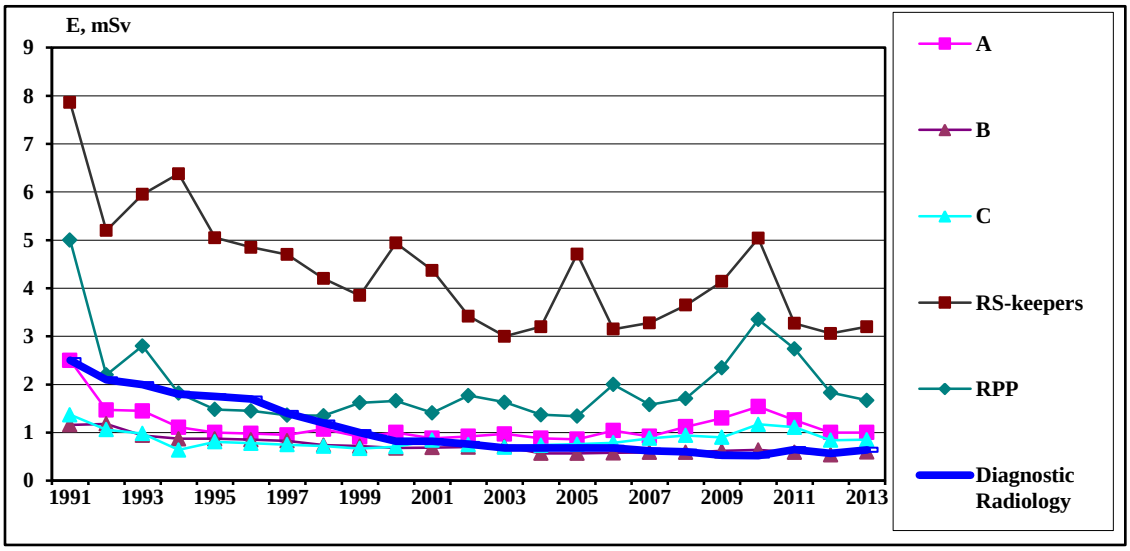

FIG. 2. Dynamics of average annual doses of medical staff in 1991-2013

The annual average doses received were in the ranges: for group A - 1.0-2.5 mSv/y, group B - 0.55-1.18 mSv/y, group C – 0.64-1.37 mSv/y, group X-ray diagnostics - 1.0-4.1 mSv/y. The annual doses of some groups of medical staff in brachytherapy (RS-keepers of radioactive substances and RPP - radiomanipulation paramedical nurses were exceeded or approached to 0.1 MDL (3.0-8.5 mSv/y and 1.5-5.0 mSv/y) (Fig. 2).

The dynamic of the average annual doses of radiologists which carry out the fluoroscopy procedures (conventional diagnostic radiology) and interventional radiology are presented in Fig. 3. The average doses of radiologists and specialists which involved to interventional radiology higher the doses of radiologists working in fluoroscopy studies 1.5-2.7 times.

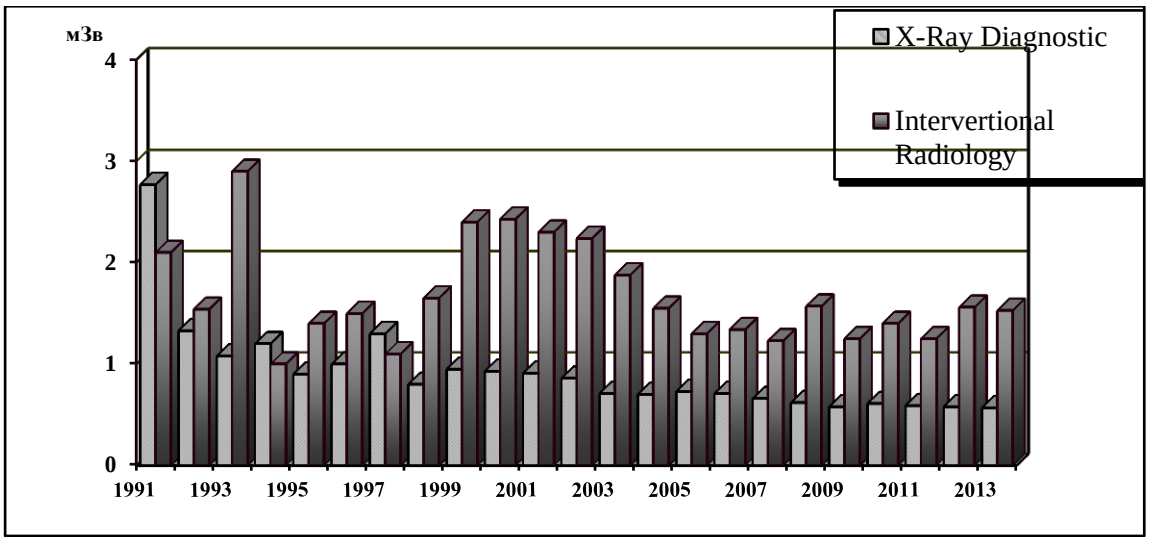

FIG.3. Average annual doses of radiologists in Fluoroscopy and Interventional Radiology

The analysis of empirical distribution of annual doses for different groups of medical staff connected with radiation sources shows that for most of them (up to 92-98 %) obtained annual doses are below 0.1 - 2 mSv/y, the 20-40 % of personnel group of keeper of RS, RPP and specialist involved to interventional radiology have the annual doses in range 2-10 mSv/y, there are some cases when the individual doses of personnel of these group are higher than dose limit for category A – 2-5 % (Fig. 4).

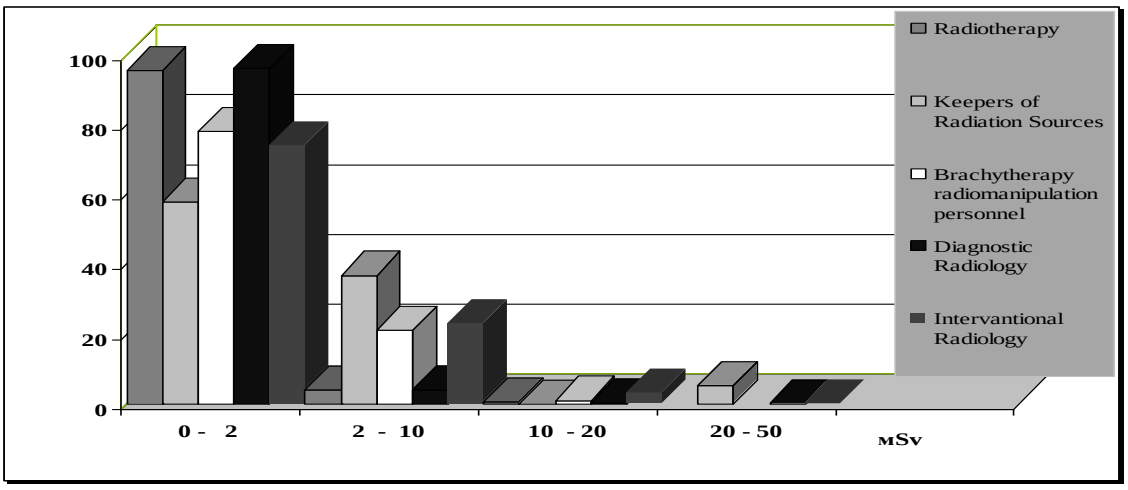

FIG. 4. The typical distribution of personal annual doses of some groups of medical staff of Ukraine

Radiation risk of stochastic effects of occupational medical exposure for all medical staff is 0.5 % additional deaths to natural level, for groups of highest doses it is 0.7-2.0%.

4. CONCLUSIONS

The annual doses for most of medical staff (up to 92-98 %) are less than 5 mSv. The most of exposed groups of medical staff are RS-keepers of radioactive substances and RPP-nurses in brachytherapy and personnel involved to the interventional radiology. Further optimization of radiation protection personnel in medical radiology should be aimed at the reduction of dose of professional groups in brachytherapy and interventional radiology.

REFERENCES

[1] NATIONAL RADIATION SAFETY STANDARD OF UKRAINE (NRSSU-97), Kiev (1998) 135.
[2] INTERNATIIONAL ATOMIC ENERGY AGENCY, International Basic Safety Standards for protection against ionizing radiation and for the safety of radiation sources, Safety Series No. 115, IAEA, Vienna (1996) 382.

BENEFITS OF PERSONAL REAL TIME DOSEMETERS IN FLUOROSCOPY GUIDED PRACTICES

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Abstract

The experience with a solid state dosemeter for real time personal dosimetry in fluoroscopy guided procedures is presented. The dosemeter was worn over the apron to estimate eye lens dose in more than 600 individual procedures. Depending on the medical speciality and procedure, median/average over apron Hp(10) per procedure was 21/65 µSv in interventional cardiology, 19/46 µSv in interventional neuroradiology and 24/57 µSv in interventional radiology. To keep eye lens doses below the new recommended limits, it is essential in complex procedures or for professionals with high workloads to use the ceiling suspended screen or goggles. The information recorded with this active dosimetry system proved invaluable in providing radiation protection training to medical specialists and in promoting optimization actions.

1. INTRODUCTION

In April 2011 [1], the International Commission on Radiological Protection (ICRP) issued a statement on tissue reactions (deterministic effects). Since then, several regulatory initiatives and scientific activities have been launched to improve patient and staff radiation safety and to foster optimization actions in interventional radiology. For occupational exposure, the ICRP recommended an equivalent dose limit for the lens of the eye of 20 mSv/y, averaged over defined periods of 5 years, with the dose in a single year not exceeding 50 mSv. The immediate consequence was a change in the international basic safety standards (BSS) [2] and in the European BSS to adopt this new limit [3].

Recent studies have reported relevant doses on eye lenses in interventional radiology and cardiology [4, 5] concluding that, if radiation protection tools like ceiling suspended screens or goggles were not used properly, the new recommended limit could be exceeded even with moderate workloads. Surprisingly, there was scarce interest in some medical specialists from some countries for radiological protection. In a multicentre study in Spain [6], only half of the interventional cardiologists in the study were reported to use their personal dosemeter correctly. The ISEMIR working group has reached similar conclusions in a worldwide survey [7]. With the new occupational dose limit, eye lens monitoring is needed and, in many cases, to reduce eye lens doses optimization strategies need to be achieved and radiological protection training to interventional operators reinforced.

In this paper, an experience with a real time solid state dosemeter conducted over a four-year period is presented. $H_p(10)$ measurements over the apron in more than 600 individual interventional procedures at a university hospital are presented. The usefulness of the real time solid state dosemeter as an educational tool is also discussed.

2. METHODS

The active solid state dosemeters Dose Aware (Philips Healthcare) are designed to measure $H_p(10)$ in interventional practices and initial results have already been reported in previous papers [8, 11]. They are equipped with a wireless connection that sends the scatter dose rate and cumulative scatter dose readings to a base station every second whenever a certain level of radiation (from few μSv/h) is detected. To save energy and increase battery life, whenever the dosimeter does not detect any significant radiation level, no information is transmitted to the base station. The base station shows the dose rate received by all dosemeters on a 10" screen in real time. Each dosemeter has the capability of storing 3600 records, i.e. one hour of radiation. All the records of instant dose rate and cumulative dose are stored in a database and can be viewed through the "Dose Manager" software (Philips Heathcare). These dosemeters were compared with TLD passive dosemeters and were found accurate enough to measure personal doses in interventional laboratories [12].

Operators wore dosemeters over the apron at chest level. Occupational dose rate and cumulative dose relative to single procedures were extracted from the information provided by the X-ray unit and included in the patient dose report, using the procedure date and time and correlations with patient dose values were subsequently established. It was also possible to identify the dose received by each operator during specific imaging and fluoroscopy series. This proves particularly useful when the DoseAware system is used for educational purposes.

An additional dosemeter was located at the C-arm 45º down with respect to the isocenter horizontal plane. With this dosemeter, an estimation of the scatter radiation dose at a reference position near the patient can be obtained.

3. RESULTS

Table 1 presents statistic results for Hp(10) measured in different interventional specialities.

TABLE 1: MEASURED Hp(10) VALUES IN DIFFERENT INTERVENTIONAL PROCEDURES

	C-arm (μSv/procedure)			Physician (μSv/procedure)		
	Average	Median	3rd quartile	Average	Median	3rd quartile
Cardiology (204)	982	682	1298	65	21	67
Neuroradiology (274)	1103	646	1470	46	19	44
Interventional Radiology (220)	764	449	1120	57	24	54

4. DISCUSSIONS

Relevant doses have been recorded for a large sample of fluoroscopy-guided procedures. All professionals involved in this survey had the second-level accreditation on radiological protection as required by national regulations. Depending on the medical speciality, median/average doses over the apron ranged from (21-24)/to (46-65) μSv/procedure. In the case of interventional cardiology, interventional neuroradiology and interventional radiology, these results are in accordance with the average dose/procedure published by ORAMED [5]: 50 μSv/procedure for cardiology, neuroradiology and lower limb procedures. Assuming that this was the average dose received by a professional per procedure, and that the ceiling suspended screen or protection glasses were not used, the eye lens dose limit would be exceeded with 400-500 procedures/year.

The dosemeter located at C-arm gave mean values of cumulative dose/procedure from 13 to 24 times higher than those of the interventionalists, thus showing that the ceiling suspended screen had been regularly used in our survey. This C-arm dosemeter could be used to conservatively estimate professional doses when the personal dosemeter was not used [13].

Part of the information, i.e. dose rates and cumulative doses, recorded by the DoseAware system during different situations and operation modes was shown to interventionalists during clinical sessions. The comparisons of the situation with and without the ceiling suspended screens, distance effect, fluoroscopy and acquisition modes, etc. were very informative and instructive to optimize the medical staff practice.

5. CONCLUSIONS

The relevant information about staff doses in interventional radiology has been compiled using an active electronic dosimetry system. Working safely in interventional laboratories is possible, but the professionals that do not use the ceiling suspended screen or goggles may exceed the new occupational eye lens dose limit. Even, in case of high workloads (>400 procedures/year), the use of both ceiling suspended screens and goggles may help keep eye lens dose under limits. The information recorded for single procedures with the real time dosimetry system could be used as an educational tool for medical interventionalists and as such help achieve real optimization actions in practices to reduce occupational doses, and also help estimate the level of risk attached to procedures performed outside the imaging department.

ACKNOWLEDGEMENTS

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REFERENCES

[1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, ICRP publication 118, ICRP statement on tissue reactions. Early and late effects of radiation in normal tissues and organs, Threshold doses for tissue reactions in a radiation protection context, Ann ICRP 41(1-2):1-322 (2012).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety of radiation sources. International basic safety standards (BSS), IAEA Safety Standards Series GSR part 3 (interim), IAEA, Vienna (2011).

[3] EUROPEAN DIRECTIVE 2013/59/Euratom on basic safety standards for protection against the dangers arising from exposure to ionising radiation and repealing Directives: 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom. Off J Eur Commun, 17 Jan 2014; L13; 57: 1-73.

[4] DOMIENIK J., BRODECKI M., CARINOU E., DONADILLE L., JANKOWSKI J., KOUKORAVA C., KRIM S., NIKODEMOVA D., RUIZ-LOPEZ N., SANS-MERCE M., STRUELENS L., VANHAVERE F., Extremity and eye lens doses in interventional radiology and cardiology procedures, First results of the ORAMED project. Radiat. Prot. Dosim. 144 (1–4) (2011) 442–7.

[5] VANHAVERE F., CARINOU E., DOMIENIK J., DONADILLE L., GINJAUME M., GUALDRINI G., KOUKORAVA C., KRIM S., NIKODEMOVA D., RUIZ-LOPEZ N., SANS-MERCE M., STRUELENS L., Measurements of eye lens doses in interventional radiology and cardiology: Final results of the ORAMED project, Radiat. Meas. 46(11) (2011) 1243–47.

[6] SANCHEZ RM, VANO E, FERNÁNDEZ JM, SOTIL J, CARRERA F, HERNÁNDEZ-ARMAS J, et al., A national programme for patient and staff dose monitoring in interventional cardiology, Radiat. Prot. Dosim., 147 (2011) 57-61.

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, ISEMIR working group on interventional cardiology, Report on the Pilot Survey on obtaining occupational exposure data in interventional cardiology, (2013). Available at http://www-ns.iaea.org/tech-areas/communication-networks/orpnet/documents/wgic-second-survey-full-report.pdf.

[8] SANCHEZ R.M., VANO E., FERNANDEZ J.M., Staff radiation doses in a real-time display inside the angiography room, Cardiovasc. Intervent. Radiol. 33 (2010) 1210-1214.

[9] VANO E, FERNANDEZ JM, SANCHEZ R., Occupational dosimetry in real time - Benefits for interventional radiology, Radiat. Meas. 46(11) (2011) 1262–5.

[10] VANO E, FERNANDEZ JM, SANCHEZ RM AND DAUER L., Realistic approach to estimate lens doses and cataract radiation risk in cardiology when personal dosimeters have not been regularly used, Heath Physics 105 (2013) 330-339.

[11] RACADIO J, NACHABE R., CARELSEN B, et al., Effect of real-time radiation dose feedback on pediatric interventional radiology staff radiation exposure, J. Vasc. Interv. Radiol. 25 (2014) 119-126.

[12] CHIRIOTTI S., GINJAUME M., VANO E., SANCHEZ R., FERNANDEZ J.M., DUCH M.A., SEMPAU J., Performance of several active personal dosemeters in interventional radiology and cardiology, Radiat. Meas., 46(11) (2011) 1266–70.

[13] VANO E., FERNÁNDEZ J.M., SÁNCHEZ R.M., DAUER L.T., Realistic approach to estimate lens doses and cataract radiation risk in cardiology when personal dosimeters have not been regularly used, Health Physics, 105 (2013) 330-339.

RELATIONSHIP OF DOSIMETRY IN PERSONNEL OCCUPATIONALLY EXPOSED RADIOIMMUNOASSAY LABORATORY AND RADIOPHARMACY

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ABSTRACT

The objective of this study was to know the variation in the five-year radiation exposure data. A retrospective analysis was carried out of the results of occupational exposures during the years 2009-2013, of the personnel in the Radioimmunoassay and radiopharmacy labs of the Institute of Nuclear Medicine. The results which were submitted by the Bolivian Institute of Nuclear Science (Laboratory dosimetry personal). The readings of personnel dosimeters (TLD) of Radioimmunoassay and radiopharmacy are reported on quarterly basis. The aim of this study was to see correlation for the variation of the results of exposure during the last 5 years.

It was noted that in the laboratory of radioinmunoanalis that it had a value of 0.7 mSv in relation to the laboratory for radiopharmacy which was 1.9 mSv, there is a variation of 1.0, because in radiopharmacy working with ^{99m}Tc, with energy of 140 keV, in comparison to the laboratory of Radioimmunoassay which works with ^{125}I, with 35 keV energy. In both laboratories of the Institute of Nuclear Medicine, radiation protection standards, and proper use of individual dosimeter are made as documentation indicates institutional licensing. The continuous refreshing / updating of knowledge of radiation protection is important for the workers occupationally exposed, to reduce the risks of exposure to radiation.

1. INTRODUCTION

Regarding the health surveillance of workers exposed to radiation, it is important to consider the basic principles of radiation protection. In normal working conditions, with dose below the established limits, there are no biological indicators of exposure, hence to know the exposures of the personnel is importance by personal radiation monitoring, using dosimeters.

1.1. Background

The Universidad Mayor Real & Pontificia de San Francisco Xavier of Chuquisaca, was founded March 27, 1624. In Sucre, capital of the Republic of Bolivia was created the center of Nuclear Medicine at the Hospital Santa Barbara on 23 October 1966, dependent of the Universidad Mayor Real and Pontificia de San Francisco Xavier de Chuquisaca, by Dr. Antonio Pardo. Radiopharmacy laboratory was conducted by Dr. Ricela Guardia, performing different tests with application of radionuclides, including^{131}I, Hg^{197}, Hg^{203}; later by Dr. Hortencia Carrasco, Professor at the Faculty of Pharmacy and biochemistry, in the subjects of Radiochemistry and radiopharmacy.

On January 14, 1960 was created the Bolivian Commission of Nuclear energy (COBOEN), by Supreme Decree No. 5389, in the city of La Paz, under the authority of the National Planning Board, with the consideration that the country should not remain marginalized in the scientific and technical progress achieved by other Nations, the creation of that body being suitable for promoting research in the field of nuclear energy and its practical applications in the country peaceful, he subsequently changed name for IBTEN. The area of in vitro diagnostics: Radioimmunoassay began in Sucre in 1987 through projects of the National Institute of Nuclear Medicine of La Paz and support of the IAEA.

The use of generator ^{99}Mo-^{99m}Tc for marking molecules in the Institute of Nuclear Medicine, of the University, started in date 11 November 1993, under the direction of Dr. Emma Echalar Kawano.

Currently, the Institute of Nuclear Medicine-Sucre comprises: area of in Vitro Diagnostics, Clinical laboratory, Radioimmunoassay; the area of diagnostic in vivo, Gamma cameras, stable isotopes laboratory and electromedical.

1.2. Radiopharmacy-Radioimmunoassay

1.2.1. Radiofarmacia

Specialty which studies the pharmaceutical, chemical, biological and physical aspects of radiopharmaceuticals.The specialist should: prepare, break up, control and administer radiopharmaceuticals. The radiopharmaceutical in the laboratory of the Institute of Nuclear Medicine is: ^{99m}Tc, emitting gamma with an energy of 140 keV.

1.2.2. Radioimmunoassay

This technique was developed by Solomon. A. Berson and Rosalyn Yalow in 1960 to determine the concentration of insulin in the blood plasma. For this reason, R. Yalow received the Nobel Prize for medicine in 1977. Esta technique is used to detect and quantify substances that are found in very small and mixed with many other quantities. It is therefore, a very sensitive and very specific technique. Using high affinity antibodies can be detected to pico-grams of Antigen. (1 pg = 10^{-12}g). The radioisotope that is used in the laboratory of Institute Radioimmunoassay is ^{125}I emitter of gamma-ray energy of 35 keV.

Personnel working in these two areas are exposed to ionizing radiation, constituting a potential risk for the personnel who handle/administer the activity to those who benefit from its use, being necessary to regulate procedures to control exposures of the occupational workers.

As previously mentioned, the objective of this study, was to examine the variation of the five-year exposure data of the of Radioimmunoassay and radiopharmacy area staff members after the renewal of individual licences for handling radioactive material.

2. METHODOLOGY

A retrospective analysis of 60 readings of thermoluminescence dosimeters (TLDs) referred by the Bolivian Institute of science and Nuclear Technology (personal Dosimetry Laboratory), in the years 2009-2013, of occupational exposures to personnel working in the Institute of Nuclear Medicine Radiopharmacy and Radioimmunoassay area (after renovation de individual licenses). The readings of dosimeters (TLD) of the workers in Radiopharmacy and Radioimmunoassay are on a quarterly basis.

Conditions of the use of dosimeters were: the TLDs are non-transferable, wearing at the height of the thorax, for use exclusively in the work area, stored and returned after the period of use. It should not be damaged or manipulated

3. RESULTS

It was observed that in Radioimmunoassay laboratory a value of 0.7 mSv was taken in relation to the laboratory for radiopharmacy which was 1.9 mSv there is a variation of 1.0, because in radiopharmacy working with ^{99m}Tc, has an energy of 140 keV and activities handled are 10 to 30 MBq as compared to the Radioimmunoassay laboratory, which works with ^{125}I whose energy is 35 keV and kBq levels of activity are handled.

4. CONCLUSIONS

In areas Radioimmunoassay, Radiopharmacy and Nuclear Medicine Institute for Radiation Protection Standards are met, it is indicated that proper use of personal dosimeter is made as documentation indicates institutional licensing, noting that there is a variation in the results in both areas due to the use of radioisotopes for diagnostic radiology with different energy and activities.

Constant updating on radiological protection standards and biosafety, is justified to reduce the risks of workers occupationally exposed at the Institute of Nuclear Medicine.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, International Basic Safety Standards for protection against ionizing radiation and for the security of radiation sources, IAEA publication 115, Collection of security N ° 115, IAEA, Vienna (1996).

[2] Radiation Protection 119. Multimedia and audovisiual radiation protection training in interventional radiology. MARTYR. martir@med.ucm.es

[3] Unit Ionizing Radiation dosimetry. Report results. IBTEN

[4] CHOEDER HR, HINES CM, OSBORN, et al, The inmunochemiluminometry assay for hepatitis B surface antigen. Clin. Chem. (1981) 1378-1384.

[5] HUANCA SARDINES, et al. (April 2012). Exhibition: Radiological dosimetry in occupationally exposed workers in Nuclear Medicine and Radioimmunoassay of the Institute of Nuclear Medicine, International Protection Symposium, Cusco, Peru (2012) 28-29

[6] IAEA, Radioinrnunoassay and Related Procedures in Medicine, IAEA, Vienna (1982)

CHALLENGES IN OCCUPATIONAL RADIATION PROTECTION AT ADVANCED RADIOTHERAPY FACILITIES IN PAKISTAN

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Abstract

The medical sector in Pakistan is progressing day by day with the inclusion of new technologies. During the last few years, a number of advanced modalities have been introduced in medical radiation sector. Pakistan Nuclear Regulatory Authority (PNRA) being the sole regulatory body in the country, has the mandate to protect workers, public and environment from the harmful effects of radiation. However, the advance medical facilities like stereotactic radio-surgery and high energy radiotherapy etc. have evolved new challenges in the protection of workers in Pakistan. In this paper, the challeges have been presented at radiotherapy facilities like, shielding calculations for complex techniques like IMRT, Cyberknife, availability of neutron dosimetry, emergency preparedness etc. and their remedial actions with local resources**.**

1. INTRODUCTION

The useful applications of radiation are continually increasing in Pakistan with the passage of time. During the last decade, the application of radiation in the field of medicine has been increased considerably. The medical centers of Pakistan Atomic Energy Commission (PAEC) were the major service provider in nuclear medicine and radiotherapy in the country. It has been observed from the last few years that private sector hospitals are also involved in providing these services. As a result, technologically advanced modalities have also been established in the country which includes Cyberknife, Gammaknife, Intensity Modulated Radiation Therapy (IMRT), Image-guided radiation therapy (IGRT), etc.

2. CHALLENGES FACED AND NECESSARY STEPS TAKEN BY PNRA IN REGULATING ADVANCED MODALITIES

The advanced modalities provide flexibility and ease in use during treatment of patients. These modalities are complex in nature and delicate in design. Therefore, particular attention is needed related to the matters of radiation protection by the radiation workers dealing with these modalities, licensee and the regulatory body [3, 4, 5, 6]. The increasingly use of advanced technologies in radiotherapy has led to many challenges in occupational exposure protection. These challenges include the unavailability of Technical Support Organization (TSO) in the regulatory body, non-existence of formal professional bodies of radiation protection, lacking of formally qualified workforce, identification and solution of technical matters in local environment keeping in view the available resources etc. PNRA has realized these challenges and started to take necessary steps to encounter them. The detailed description of these challenges and their remedial actions taken by PNRA are given as under:

2.1. Education and Training

The education system related to radiation related areas is yet not well established in the country like professional certifications and postgraduate diplomas for medical physicists, radiation protection officers (RPO), qualified radiotherapy technologist etc. Pakistan Institute of Engineering & Applied Sciences (PIEAS) is the only university in the country offers masters level degree in medical physics. All the graduates of medical physics program join medical centers of PAEC after their graduation. The private sector and other medical centers of government departments (like health department) have to rely mainly on regular physics graduates. These graduates becomes familiar with the operation and daily business of the medical centers in few months but still they do not find any urgency in learning the other technical and specialized matters which are not included in routine jobs like shielding calculations, area survey (particularly for high energy LINAC, Gamma knife, etc.). The same situation has been observed for RPOs. PNRA Regulations on Radiation Protection (PAK/904) [15] requires that the licensee will designate a radiation protection officer to supervise all matters of radiation protection.

The PRO and medical physicist play primary role in the radiation protection of patients, workers and general public, also prescribes the qualification criteria and their jobs are largely overlapping in Pakistan. The qualification criteria of medical physicists for radiotherapy facilities and their comparison with some countries is given in Table 1 [16, 17].

TABLE 1. COMPARISON OF QUALIFICATION CRITERIA OF MEDICAL PHYSICISTS FOR RADIOTHERAPY FACILITIES

Pakistan	USA	India
• Medical physicists having MS medical physics degree or MSc Physics with 6 months on the job training. * no mechanism of certification	• Has earned a master's and/or doctoral degree in physics, medical physics, biophysics, radiological physics, medical health physics, or equivalent disciplines from an accredited college or university; and • Has been granted certification* in the specific subfield(s) of medical physics with its associated medical health physics aspects by an appropriate national certifying body and abides by the certifying body's requirements for continuing education. * Certification by designated bodies is mandatory	• M.Sc in Medical Physics from a recognized university with two years experience immediately prior to appearing for the membership examination. • M.Sc in Physics/Bio-Physics with one-year Postgraduate diploma in Radiological Physics from Bhabha Atomic Research Centre or equivalent thereto and minimum of two years of experience in medical physics immediately prior to appearing for the membership examination. • B.Sc in Physics with one-year Postgraduate diploma in Radiological Physics from Bhabha Atomic Research Centre or equivalent thereto and minimum of four years of experience in medical physics immediately prior to appearing for the membership examination.

There appears to be a large gap of clinical experience of the medical physicists of Pakistan and other countries and the situation is same for RPOs as well [1, 2]. In addition to

this, formal certification of examination system for medical physicists or RPO is yet to be established.

In order to encounter this gap, PNRA has established National Institute of Safety and Security (NISAS). PNRA has arranged many professional courses on various topics of radiation protection for medical radiation facilities from the platform of NISAS in the recent years. These courses were majorly on radiation protection, regulatory requirements and shielding calculations etc. Some specialized courses were also conducted on quality assurance (QA) in radiotherapy not only focusing on the patient protection but also including occupational exposure. The events provides to the professionals a useful forum of discussions, experience sharing and feedback. Further, PNRA conducts courses for their licensees to elaborate the measures of workers safety at the licensee's premises. PNRA is also planning to offer professional diplomas in specific areas of radiation protection from the platform of NISAS.

2.2. Expert Advice

The IAEA Education and Training Appraisal (EduTa) mission has visited Pakistan in 2013 and observed that the "Qualified Experts" which may provide expert advice in radiation protection are not available in the country. Further, IAEA Integrated Regulatory Review Services (IRRS) in 2014 observed the non existence of TSO for the regulatory body in the technical matters of radiation protection. In current scenario, all the activities related to radiation protection are solely carried out by the licensee and reviewed / assessed by the PNRA.

As discussed in section 1 some of the advanced modalities have already been established in Pakistan but the unavailability of TSO, qualified experts and professional bodies is in contrast with the system of the advance countries like USA, UK, Canada, etc. These entities are helpful in providing technical advice, detailed review of the radiation protection programs, emergency plans, performing shielding calculations, etc as required in PAK/904 [15].

In the absence of TSO, professional body on radiation protection / medical physics and university education etc., PNRA has managed to fill this gap by increasing the coordination among the licensees as well as its own inspectors. In this setup, expert advice is not fully based on the licensee's RPOs and not based on the inspectors of the regulatory body. PNRA inspectors are also encouraged to learn from external sources like Health Physics Society of USA, National Council on Radiation Protection and Measurements (NCRP), The American Association Of Physicists In Medicine (AAPM) etc. and this knowledge is also shared with licensees. PNRA also participate in locally arranged conferences and international events (like IAEA courses) to enhance the competency of its officers.

2.3. Shielding Calculations

Many private and public sector medical centers in Pakistan have high energy LINACs. At initial stages, these centers usually perform conventional radiotherapy and the shielding of the bunkers were performed accordingly. These shielding calculations are verified by the PNRA. However, with the passage of time, the staff gets more competency and they tend to perform IMRT techniques. More shielding is required for IMRT technique than conventional radiotherapy even at the same energy. If LINACs are operated without considering this factor, it may cause additional exposure to workers on regular basis. Furthermore, the shielding calculations of advance modalities like Cyberknife, Gammaknife, LINAC (trilogy systems), etc. needs specialized expertise and qualification which is, at the moment, available on personal interest basis [6, 12 -15].

Shielding calculations for radiotherapy machines is a cumbersome and delicate job. Most of the time, the shielding calculation is performed by the vendors and submitted to PNRA for review and assessment. PNRA has covered this gap by developing expertise considerably in shielding calculations. PNRA has developed its own software for shielding calculations and arranged a number of workshops on this topic. By using this software, all types of shielding calculations for LINAC, ^{60}Co, Brachytherapy etc., can be performed at PNRA and licensee's submissions can be verified. PNRA mostly uses relatively more conservative parameters and reduces dose limits for workers in order to accommodate any increase in the workload and change in techniques e.g. IMRT etc.

2.4. Area Survey

Being an essential regulatory requirement the area survey is considered an important factor for ensuring the occupational radiation safety. Specifically for area survey related to LINAC bunkers, well qualified medical physicists also needs specialized expertise. Further, the high energy LINACs tend to produce secondary neutrons for which a neutron survey meter is required, which is very rare in medical centers [9, 11].

Area survey and interpretation of results at a radiotherapy facility is needs training and expertise. As discussed in section 2.3, PNRA has conducted many workshops for its licensees on shielding calculations. The performance of area survey is an important part of these workshops. Furthermore, PNRA inspectors perform area survey of radiotherapy facilities in accordance with the procedures given in IAEA documents [12]. However, challenges are realized for neutron area monitoring because not many licensees have neutron survey meters. For the time being, the results of area monitoring performed by PNRA inspectors are used by the licensee. However, PNRA is suggesting its licensees to have better cooperation among them and utilize and share the resources collectively.

2.5. Personal Dosimetry

It has been observed that some qualified workers are doing jobs at more than one place. In these situations, personal dosimetry and record keeping becomes more challenging as most of the workers are availing personal dosimetry services from their primary employer. In high energy radiotherapy when neutrons are generated, it is more realistic to use personal dosimeters which are also capable of measuring doses from neutrons as well. It has been reported by various licensees that the personal dosimeters which are capable of measuring doses from neutrons are not readily available in the local market at the moment. Therefore, commonly used personal dosimeters in the country do not take into account the neutron doses.

Personal dosimetry is the ultimate indicator of the effectiveness of all radiation protection measures. PNRA has been taking several steps for radiotherapy workers who are doing jobs at more than one place and recommends more forcefully to use separate dosimeters for each station. In order to counter the situation of neutron doses, conservative shielding calculations are performed to minimize the chances of undue neutron exposure and the same is verified at the time of inspections, review & assessment.

2. CONCLUSIONS

PNRA is promoting their licensees to use the advanced modalities by considering all the necessary radiation safety measures. PNRA has taken several steps in collaboration with other stakeholders to manage the challenges being faced due to technology advancements and rapid increase in radiotherapy centers. In order to meet the upcoming requirements, there is a need to increase the competent work force of all the stakeholders involved in this business. The

most important factor is the frequent training and retraining of all the stakeholders which is already established and PNRA is planning to expand its scope by offering more specific and professional courses. Ultimately, these arrangements will results in the improved safety culture at licensee's premises.

REFERENCES

[1] AMERICAN ACADEMY OF HEALTH PHYSICS, Medical Health Physics Section of the Health Physics Society Standards of Qualification and Practice (SQ/P) Qualifications for Healthcare Facility Radiation Safety Officer, AAHP.

[2] THE AMERICAN ASSOCIATION OF PHYSICISTS IN MEDICINE, Radiation safety officer qualifications for medical facilities, AAPM, Report of AAPM Task Group 160 (November 2010).

[3] WORLD HEALTH ORGANIZATION, Global Initiative on Radiation Safety in Healthcare Settings Technical Meeting Report 15-17 December, 2008 WHO Headquarters, Geneva.

[4] NATIONAL COUNCIL ON RADIATION PROTECTION AND MEASUREMENTS, Emerging issues in radiation protection in medicine, Emergency response, and the nuclear fuel cycle, NCRP (March 12–13, 2012).

[5] R. HALLARD, J. BROUGHTON, R. COATES, J. CROFT, K.DAVIES, I. DEVINE, C. LEWIS, J. Radiol., Developing the radiation protection safety culture in the UK P Cole, 34 (2014) 469–484.

[6] THE AMERICAN ASSOCIATION OF PHYSICISTS IN MEDICINE, Neutron measurements around high energy X-Ray radiotherapy machines, Report Of Task Group 27 Radiation Therapy Committee American Association Of Physicists - In Medicine, AAPM.

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Emerging Challenges In The management of occupational exposures, 12th congress of the international radiation protection association (IRPA12): strengthening radiation protection worldwide- highlights, global perspective and future trends, Geneva, IAEA (2010).

[8] ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, Scientific issues and emerging challenges for radiological protection, Report of the Expert Group on the Implications of Radiological Protection Science NEA No. 6167, OECD (2007).

[9] LUND UNIVERSITY, Estimation of neutron dose contributions to personnel working around high-energy medical linear accelerators for radiation therapy, Drita Muca, Medical Radiation Physics Clinical Sciences (2006).

[10] UNIVERSITY OF TSUKUBA, Effects of Secondary Neutron Beam Generated in Radiotherapy on Electronic Medical Devices Tomonori ISOBE, Hiroaki KUMADA, Kenta TAKADA, Graduate School of Comprehensive Human Sciences, Tennodai, Tsukuba, Ibaraki, 305-8575 Japan.

[11] DIVISION OF RADIATION ONCOLOGY, SHIZUOKA CANCER CENTER HOSPITAL, 1007 Shimonagakubo, Nagaizumi- cho, Sunto-gun, Shizuoka, 411 -8777 Japan, Progress in NUCLEAR SCIENCE and TECHNOLOGY, Vol. 2, (2011) 524-529.

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Applying radiation safety standards in radiotherapy, Safety Reports Series No.38, IAEA, Vienna (2006).

[13] NATIONAL COUNCIL ON RADIATION PROTECTION AND MEASUREMENTS, radiation Protection For Particle Accelerator Facilities, NCRP REPORT No. 144, Revised March 4, (2005).

[14] NATIONAL COUNCIL ON RADIATION PROTECTION AND MEASUREMENTS, Structural Shielding Design and evaluation for maga voltage X-ray and gamma ray radiotherapy facilities, NCRP Report No 151, December 31, (2005).

[15] PAKISTAN NUCLEAR REGULATORY AUTHORITY, Regulations on Radiation Protection (PAK/904), PNRA, Islamabad (2012).

[16] THE AMERICAN ASSOCIATION OF PHYSICISTS IN MEDICINE, Definition of a Qualified Medical Physicist, http://www.aapm.org/medical_physicist/fields.asp.

[17] COLLEGE OF MEDICAL PHYSICS OF INDIA, Eligibility to appear for CMPI Examination, http://cmpi.org.in/?p=191.

OPERATIONAL EXPERIENCES IN RADIATION PROTECTION IN MEDICINE

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1. INTRODUCTION

Radiation protection aims at eliminating any ionizing radiation exposure risk to a third party during a medical procedure. While there are some common principles of protection that apply to all X ray imaging, and therapeutic procedures, significant issues and actions are related to the methods, equipment type and protocols being applied. The Radiation Protection Board of Kenya has de-linked the Radiation Safety Services from the board and delegated to approved service providers through a memorandum of understanding (MOU).

2. SITUATION ANALYSIS

Technical Support Organizations (TSOs) are empowered to offer Radiation Safety and Quality Control services to radiation operating facilities. Radiation exposures are under strict regulatory control and mechanisms are in place to protect the personnel, patient, and public. However, some of the TSOs may be using under trained personnel and QA/QC equipment with limitations, compounding to added risks. Results from sampling of operational reports have shown that there are shortcomings leading to secondary risks that should be addressed.

2.1. Aims of the paper

a) Demonstrate increased surveillance coverage attained through the introduction of A certified TSOs by the Regulatory Body.
b) Demonstrate practical shortcomings, challenges and steps taken to minimize risks.
c) Explicitly display gaps in routine radiation protection processes.

2.2. Expected outcomes

i. Ensure that all Radiation professionals and workers are involved in radiation protection and safety matters. .
ii. Establish professional TSOs practice protocols through rigorous assessment, monitoring and Evaluation for compliance, therefore it must be entrench into the law/regulations
iii. Propose and implement practical remedies in partnership with the International and National Authorities such as the IAEA, WHO, KEBS and RPB.
iv. Embrace and promote new technologies while entrenching safety culture at work.
v. Develop accessible and affordable continuous professional education/ development for Radiology professionals', engineers and other cadres involved in radiation work.

3. EXPERIENCES OF RADIATION SAFETY SERVICES BY THE TECHNICAL SUPPORT ORGANIZATION (TSOS) IN KENYA

TABLE 1. 2014 DISTRIBUTION OF RADIOLOGY EQUIPMENT IN KENYA

	Equipment Type	Number
Ionizing Radiation	CT scanners	38
	Cath Labs	9
	Screening Fluoroscopy	82
	General Radiography	551
	Mobile Units	266
	C-arms	74
	OPG	43
	Intraoral Dental	332
Non-ionizing Radiations	MRI	21
	Ultrasound	329

TABLE 2. 2014 NUMBER OF MEDICAL RADIOLOGY STAFF IN KENYA

Medical doctors	5050
Radiologists	150
Medical physicists	15
Radiographers	800
Biomedical Engineers	680

TABLE 3. ANNUAL RADIATION EXPOSURE DOSE OCCUPATIONALLY EXPOSED PERSONNEL IN KENYA

Occupational Classification	Average Annual Dose (mSv)
Radiologists	3.33
Oncologists	2.75
Dentists	3.24
Medical Physicist	3.53
Technologists	3.55
Nurses	2.97
Film Processors	2.46
Ancillary Staff	2.39
Radiology Office Staff	2.41

3. CURRENT STATUS

The regulatory authority plays a key role in the implementation of effective quality assurance and control programs within the country.

Kenya chose TSOs Sub contracting to complement Government's efforts in promoting competence and involving a wide base of health and safety professionals' in order to identify inadequacies and oversights. The arising issues of either new factors or techniques to improve the services are therefore addressed by the regulatory body or during stakeholder's

meeting. The RBP approved certified Technical Support Organizations to provide services in the following categories;

3.1 Quality Assurance Services

- Verification of the appropriate physical building and shielding
- Verification of physical parameters of the radiation generators and imaging devices at the time of commissioning and thereafter periodically
- Availability of code of practice
- Records of calibration, maintenance and assessment reports are well kept and traceable.
- Dosimetry and monitoring equipment database.
- Quality Assurance program personnel
- Regular and independent quality audit reviews of the QA program
- Emergency preparedness plan.

3.2. Quality Control Services

- Administrative procedures or quality management systems.
- Verification that the equipment operates satisfactorily.
- The quality control techniques are performed properly and according to the facility QA manual.
- QC results are evaluated promptly, accurately and records maintained.
- The necessary corrective measures are taken in response to above results.

3.3. Dosimetry and personal dose monitoring services

- Film badges read and exchanged monthly.
- TLD's are read and renewed monthly.

3.4. Radio Analytical Services

- Safety assessment protocol and report format is being standardized and now in the final stages, so as to harmonize the reading for faster channeling of the results to expedite remedial actions.

4. CHALLENGES AND GAPS

The TSO's takeover of the services has lead to realization of challenges and gaps in radiation protection and safety in health. This calls for a review by the Radiation Protection Board and establishment of safety standards.

4.1. Equipment and facility

i. Country level legislation which has greatly improved is still having the growth challenges especially in the devolved county healthcare.
ii. Inadequate number of trained human resource, radiation survey equipment, testing and calibration especially in the new devolved health set-ups.

iii. Inconsistent shielding designs and patterns, leading to inadequate shielding integrity assessments and dose calculations thus false radiation *protection impressions*.
iv. Over/under protection of premises resulting in unnecessary extra costs incurred or inadequate radiation protection. "Often times, the NZ principle is applied thus: - If all else fails, 2 mm lead for primary, 1mm lead for scatter!!"
v. Financial and business plan to enable TSOs to serve the low resource facilities.
vi. Inadequate tools and equipment at the SSDL calibration centers, lack of identification and traceability of the TSOs**.**
vii. Experiences in the field during assessment for leakage and scatter survey using designated meters, revealed detection of radiation leakage in the protective booth joints, overlap, between glass and the shielding sheets, and sometimes old overused lead aprons
viii. New or support staff not used to warning signs or lights, not knowing when to enter or leave a control area.
ix. Support staff are often ignored or overlooked during acquisition of radiation monitoring badges, so no radiation monitoring is given to them. (Effective radiation dose exposure is ignored)**.**
x. Conflict of interests with respect to RPB staff. Some of the personnel at the regulatory body still offer their skilled services through the TSO's and this is against existing MOU and the law.

4.2. Protocol

i. Available QC, tests and maintenance reports are brief and only appropriate for billing purposes
ii. General ignorance of safety cautions, or restricted access areas enclosed purposely for limiting radiation exposure, i.e. the image intensifying and fluoroscopy where non-RT's must have to operate the equipment.
iii. Inappropriate use of protection gear/ gown by the worker for special procedures in consideration of different organ sensitivities. For example; one sided aprons in an orthopeadic room.
iv. Support staff is often ignored or overlooked when acquisition of occupational radiation exposure dosimeters and so they are not monitored.
v. Personnel not qualified and certified as Medical Physicists or Biomedical Engineers doing the TSOs assessments.
vi. Wrong or non-familiar warning signs of malfunctioning of equipment or undergoing maintenance, on when it is certified for patient applications.

4.3. Personnel

i. The overall performance in this category is fairly good**.**
ii. Non-adherence to health and safety culture at work
iii. A small number of facilities don't have radiation warning signs posted conspicuously or in the right place.
iv. Not much can be concluded from the workers monthly radiation dose readings (there is no analysis of the risk).
v. Some of the facilities do not recognize RSO role or refresher radiation safety courses.
vi. The personnel lack proper record keeping of equipment technical performances and routine maintenance including action taken after maintenance and repair work of the medical devices.

vii. Engineers' equipment performance reports may look similar but may not mean the same things as those done by a qualified and certified medical physicist in verification of equipment performance.

5. CONCLUSIONS AND RECOMMENDATIONS

i. An urgent tasking of the Government to provide SSDL centre that is affordable to all service providers.
ii. Radiation protection inspection measures to be ascertained during and after construction but prior to commissioning radiology services.
iii. Empowering the TSOs by continuous professional development and awareness campaigns, establishment of professional certification for Doctors/ Radiologists dealing with radiation medicine.
iv. Collective bargain for purchasing typical calibration and testing for the multiple TSOs under private professional partnership with the support from IAEA to acquire equipment on friendly financial terms.
v. Involving all the stakeholders for example KAR, RPB, SORK and all radiology professionals in the establishment of certification standards and regulations for TSO's safety services.
vi. Vendors for fluoroscopy and image guided procedures to incorporate a mandatory complete protection gear instead of leaving it for the Health provider to procure by themselves hence getting the wrong or inadequate protective gear.
vii. Avail and strengthen the institutionalized and tailor made short courses, to address the occupationally exposed radiology support personnel and related professionals, on radiation risks and safety. The UNES department at the Nairobi University began this initiative and could not be a more perfect start, and is bringing a ray of hope on such a course.
viii. Take measures to increase medical Physicists in the country, and to educate and empower RSOs to articulate daily safety measures and monitoring of the radiation protection services.
ix. Select and adopt a particular colour coding for protective gear, for the different uses and departments using different modes of radiation applications. i.e. For the diagnostic, blue, for the IGPs, red etc.
x. Promote KAP/DAP meters availability and incorporation of patient dose records in the patient file
xi. Dose monitoring and measurements with live recording meters to keep on actively recording and displaying/sending warnings of real-time doses

REFERENCES

[1] KORIR G.K, WAMBANI J.S., et al. Quality Management Systems in Radiology, SAfri. J. Rad. 17 (3) (2013) 84-88.
[2] KORIR G.K, WAMBANI J.S, KORIR I.K., Estimation of annual occupational effective doses from external ionizing radiation at medical institutions in Kenya, SAfri. J. Rad. 88(11) (2011) 116-119.

Radiation safety assessment as part of QC, at MTRH Ampath, Eldoret.

IAEA, Conference in ORP

MAIN RESULTS OF THE ASSESSMENT OF RADIATION WORKER'S RISK IN RADIOTHERAPY MODALITIES OF SOME MEXICAN RADIOTHERAPY FACILITIES, WITH THE HELP OF SEVRRA SOFTWARE

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Abstract

The present work describes the main results of the risk assessment of radiotherapy radiation workers due to preforming their work in the radiotherapy modalities: Linear Accelerators (LINAC), ^{60}Co, High dose rate brachytherapy and Low dose rate Brachytherapy in Mexico. These evaluations were made applying the risk matrices method throughout SEVRRA (risk evaluation software tool), developed at the Mexican regulatory body, National Commission for Nuclear Safety and Safeguards (CNSNS), in cooperation with the Iberoamerican Regulators Forum (FORO). The methodology used is supported by risk matrices method, which is a mathematical tool for risk assessment, focused in this paper on the assessment of risk to radiation workers from mechanical failures, miscalibrations, human mistakes, and so on. The initiating events are defined as those undesirable events that, together with other failures, can produce an over-dose to a radiation worker. Initiating events frequency and reducer of its frequency (actions intended to avoid the accident) are estimated as well as robustness of barriers to those actions, such as mechanical switches, which detect and prevent the accident from occurring.

1. THE SEVRRA SOFTWARE TOOL

SEVRRA is software developed by the CNSNS, derived from the outcome got through the risk matrix for radiotherapy facilities performed by the FORO [1] and [2]. Every SEVRRA user has the possibility to choose between each of the actual four different modalities uploaded in SEVRRA. It includes ^{60}Co, Linear Accelerator, High Dose Rate Brachytherapy (HDRB), and Low Dose Rate Brachytherapy (LDRB) practices in order to get the resulting risk assessment using risk matrices. SEVRRA' works as a sorted set of summarized scenarios that covers all the different stages needed to deliver a treatment, from the post-commissioning and setup, up to the treatment execution and unit maintenance. Some stages involve accident initiator events for patients, public, or radiation workers. Risk is computed as a function of the frequency of the accident initiator event, their consequence magnitude, and the robustness of safety barriers to the accident initiator events.

Until now, SEVRRA has being used in the following countries, with wide acceptance but, undisclosed results: Argentina, Brazil, Cuba, Chile, México, Perú, Spain and Uruguay. Mexican radiotherapy licensees can access to SEVRRA software through the regulatory body web page [3].

1.1. Reference radiotherapy facilities

By applying SEVRRA's analysis, the first goal in this risk assessment is to know how the existing radiotherapy facilities suit/match with the corresponding reference installation [1], by correcting the weakness of the real practices. This is expected to be done by the user by his own, under the regulatory instances supervision.

2. INITIATOR EVENTS IMPACTING RADIATION WORKERS

According to the FORO study [1], there are some accident initiator events that could impact radiation workers, if there are not set the necessary safety barriers. For each radiation therapy practice, the amounts of accident initiator events detected by the FORO [2] for radiation workers are as indicated in Table 1.

TABLE I. AMOUNTS OF POSSIBLE ACCIDENT INITIATOR EVENTS DISCOVERED BY FORO STUDIES, INVOLVING RADIATION WORKERS BY MODALITY

Modality	Accident initiator events involving radiation workers	Percentage from the total of accident initiator events detected by FORO studies in each modality.
Linear Accelerator	5	3.5%
^{60}Co	7	5.0%
High-Dose rate Brachytherapy	16	14.0%
Low-Dose rate Brachytherapy	10	13.0%

In the following tables (Tables 2 to 5) below it is shown the accident initiator events for radiation workers in the Linear Accelerator, ^{60}Co, HDRB and LDRB modalities detected by FORO studies [1], some of them could achieve a high risk level if no proper safety barriers are set to face them out:

TABLE 2. ACCIDENT INITIATOR EVENTS FOR RADIATION WORKERS IN RADIOTHERAPY TREATMENTS WITH LINEAR ACCELERATOR MODALITY

1	Weaknesses in the treatment room shielding.
2	Start the irradiation when a worker is inside the treatment room while performing maintenance tasks on the Linear Accelerator while this in service mode.
3	The worker tries to enter in the treatment room when the device is radiating.
4	Start the treatment with a radiation worker inside of the treatment room (undetected by the radiation workers at control panel).
5	Start irradiation with an inadvertent radiotherapy technician inside the engine room.

TABLE 3. ACCIDENT INITIATOR EVENTS FOR RADIATION WORKERS IN RADIOTHERAPY TREATMENTS WITH COBALT-60 MODALITY

1	Radioactive source jamming during the charge or change of it.
2	Abnormal leakage of radiation through the head (fissure, or unacceptable activity source, other deficiencies).
3	Shortcomings in the treatment room shielding.
4	Start the irradiation when a worker is held inside the treatment room while performing tasks of maintenance of the equipment.
5	The worker tries to enter the treatment room with the source exposed.
6	Jams of the drawer with the source exposed at the end of the treatment.
7	Start the treatment with a radiation worker inside of the treatment room (undetected by the radiation workers at control panel).

TABLE 4. ACCIDENT INITIATOR EVENTS FOR RADIATION WORKERS IN RADIOTHERAPY TREATMENTS WITH HIGH DOSE RATE BRACHYTHERAPY MODALITY

1	Anomalous jam of the radioactive source during the charge or change of it.
2	Shortcomings in the device container shield
3	Shortcomings in the shielding of the room.
4	Start the irradiation when a worker is held inside the treatment room while performing tasks of maintenance of the equipment.
5	Jams of the radioactive source during source switching.
6	Failure in the shielding of the sources shipping containers.
7	Unplugging of the transfer guide tube with the source inside, during treatment.
8	Loss of power that prevents the return of the source to the container.
9	Attempt to entry of a radiation worker in to the treatment room during irradiation of a patient.
10	Attempt to entry of a radiation worker in to the treatment room, with the source out of its armored housing and out of the patient too.
11	Jams of the source, remaining within intracavitary implants or surface implants after treatment ending.
12	Jams of the source remaining inside of interstitial implants, after treatment.
13	Jams of the source after the treatment, staying out of the patient and the shield.
14	Undock of the supplying cord, remaining within implants intracavitary or surface after treatment.
15	Undock of the supplying cord, remaining within implants interstitial, after treatment.
16	Undock of the source of the supplying cord, falling out side its armored housing and inside of the treatment room.

TABLE 5. ACCIDENT INITIATOR EVENTS FOR RADIATION WORKERS IN RADIOTHERAPY TREATMENTS WITH LOW DOSE RATE BRACHYTHERAPY MODALITY

1	Inappropriate shielding of the storage room for sources.
2	Improper shielding of the treatment room.
3	Shielding deficiencies or lack of means for the handling of sources in the preparation room.
4	Shielding deficiencies of container or lack of sources conveyors trolleys.
5	Use of leaking sources, due to problems of handling and storage, for the treatment of patients.
6	Loss of control of a source during the preparation.
7	The patient tries to withdraw or eject the implant during the administration of the treatment.
8	During the withdrawal of the sources there is control loss of one or more of them remaining inside the patient.
9	During the withdrawal of the sources there is control loss of one or more of the sources staying in the room outside of the shielded container.
10	Loss of control of one or more seeds during implantation.

For every one of the previous accident initiator events listed above, FORO studies have found safety barriers to face them up [1, 2] and [3]. For Linear Accelerator modality the safety barriers for radiation workers are listed in Table 6.

TABLE 6. SAFETY BARRIERS FOUND TO PROTECT RADIATION WORKERS IN RADIOTHERAPY TREATMENTS WITH LINEAR ACCELERATOR MODALITY

1	Initial evaluation of treatment room shielding.
2	Door interlock. Electric switch in the door entrance of the treatment room that interrupts radiation of the device.
3	Light signal on the entrance door indicating that the treatment device is radiating.
4	Equipment's light signal in the head of the treatment device when radiating, observed by the closed circuit of TV.
5	Beep sound that indicates that the device is radiating.
6	Timer. Device that will not allow irradiation to start in the treatment room at an interval of time set after the radiotherapy technician press the "abandonment of the room" button.

For ^{60}Co, HDRB and LDRB modalities the safety barriers found are listed in [1] and [2]. Not all of the barriers apply for every accident initiator event. The proper correspondence of each one is described in references [1- 3].

3. RISK ASSESMENTS WITH SEVRRA SOFTWARE TOOL IN MEXICAN FACILITES

Majority of the Mexican licensees reported to the regulatory body in 2013 that they only have one accident initiator event having high risk level concerning radiation workers, for which they have no safety barrier set in order to avoid an accident from occurring: "Start the treatment with a radiation worker inside of the treatment room (undetected by the radiation workers at control panel)". This initiator event was found having high risk level in majority of both Linear Accelerator and ^{60}Co modalities of the Mexican radiotherapy licensees. But it was not reported to be present in none of the Mexican licensees of brachytherapy modalities. For this accident initiator event, SEVRRA software (FORO studies [1] suggests to the radiotherapy users to set the safety barrier listed (6) in Table 6. This safety barrier is considered strong enough to reduce the risk level from "High" to "Medium" in the aim of the FORO studies. Some of the barriers are not required yet to the licensees in the actual regulation; the licensees install them as part of their quality assurance program. However, the Mexican regulatory body is proposing to modify some parts of the regulation to include the missing safety barriers that could reduce risk levels to radiation workers as suggested by FORO studies. A paper describing the risk assessment results achieved by Ibero American licensees for patient, public, and radiation workers was presented in Bonn in 2012 [4].

4. CONCLUSIONS

Assessing risk level of the radiotherapy radiation workers with SEVRRA software has let Mexican regulatory body to have a global scope on how safety is carried out in the radiotherapy practices in Mexico, with the chance to recommend governmental plans that lead to the best regulations and laws in radiation safety matters, promoting the setting up of new installations, accurate decision making, and improving the cost-profit of inspections and licensing.

REFERENCES

[1] FORO IBEROAMERICANO DE ORGANISMOS REGULADORES RADIOLÓGICOS Y NUCLEARES, 2010, Aplicación del método de la matriz de riesgo a la radioterapia, Vols. 1 and 2, downloable at FORO's home page (in Spanish) http://www.foroiberam.org.
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Aplicación del método de la matriz de riesgo a la radioterapia, IAEA TECDOC 1685/S, IAEA, Vienna (2012).
[3] SEVRRA Home Page 2012, in CNSNS home page at: http://www.cnsns.gob.mx.
[4] INTERNATIONAL ATOMIC ENERGY AGENCY - at International Conference on Radiation Protection in Medicine - Setting the Scene for the Next Decade, Main results of the risk assessments to some Ibero American radiotherapy facilities using SEVRRA software, 3 to 7 December (2012).

OVERVIEW OF OCCUPATIONAL DOSE IN INTERVENTIONAL RADIOLOGY IN ESTONIA 2009 -2013

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Abstract

Interventional radiology involves long procedure and fluoroscopic time. As interventional radiology develops rapidly, radiation protection of exposed workers and patients is an important issue. This paper presents data on occupational exposure to ionizing radiation in interventional radiology in Estonia for the period 2009 - 2013 and provides the trends about annual effective occupational doses. The survey is based on the data from the registry of exposed workers and sources of ionizing radiation, and the data from health statistics database of National Institute for Health Development. This paper is intended to share the information on trends of occupational doses of interventional radiology staff and to indicate a need to improve the safety and comfort of personnel.

1. INTRODUCTION

After Estonia became a member of the EU in 2004, the new Radiation Act was adopted that determined the categories of occupational workers. The occupational dose limits are established by the government regulation No 193 of 17 May 2004: "Effective Dose and Equivalent Dose Limits for the lens of the Eyes, Skin and Extremities for Exposed Workers and Members of the Public". The dose limits are exactly the same as provided in the Council Directive 96/29/Euratom. According to the new Council Directive 2013/59/Euratom of 5 December 2013, which has to be adopted by EU countries in 2018, the equivalent dose for eyes must be reduced from 150 mSv/y to 20 mSv/y.

The Environmental Board is responsible for maintaining the State Dose Register of Exposed Workers, Register of Sources of Ionizing Radiation, and Register of Radiation Practice Licences with the purpose to collect, process and provide data on licence, sources and workers.

RadPRo International GmBH (former RADOS) TLD system with LiF tablets is used for the personal dose measurements.

The following algorithm is used for the workers who have two dosimeters:

$$E = 0.5\ Hp(10) + 0.025\ Hp(0,07)$$

Where,

E	Effective dose
Hp(10)	Personal dose equivalent at waist or chest, under the apron.
Hp(0,07)	Personal dose equivalent at neck, outside the apron.

2. OVERVIEW OF OCCUPATIONAL EFFECTIVE DOSES FOR INTERVENTIONAL STAFF

The number of interventional procedures performed annually throughout Estonia has increased in the past years and is shown in Table I.

TABLE I. INTERVENTIONAL RADIOLOGY EXAMS, 2009-2012*

	Number of exams			
	2009	2010	2011	2012
Angiography studies	13035	5601	5312	6010
Coronary angiography	6751	6943	7474	7380

*Sources of data: National Institute for Health Development, Health Statistics

The number of facilities available for interventional radiology increased from 4 to 5 in 2012 (source of data: Register of Radiation Practice Licences). According to the Register of Sources of Ionizing Radiation the number of angiography systems used in interventional radiology is 8. The years of manufacture range from 1995 to 2012. However, 7 angiography systems used for interventional and diagnostic procedures are completed as fully digital systems and part of them have integrated flat detector technology. There are also 3 digital C-arm systems used for interventional and diagnostic procedures manufactured by Siemens and Philips, one of them with flat detector technology.

The trends and distribution of doses to the interventional radiology staff in the period 2009 to 2013 has been studied using the data of State Dose register of Exposed Workers. During this period the average annual doses of monitored workers by the type of medical radiation practice varied a little. However, some workers involved in interventional radiology procedures had high effective doses. The highest annual dose recorded for this period reached 21.8 mSv.

The trends in average annual occupational effective doses in health care services are shown in Fig. 1 and average annual effective doses in interventional radiology staff using one dosimeter is shown in Fig. 2. During the period 2009 – 2012 the average annual doses of monitored interventional radiology staff varied a little. In 2013 the average effective doses of interventional physicians and nurses increased by a factor of two. This change could have been caused by several things, for example involvement of specialists with insufficient training in the interventional diagnostic and treatment procedures.

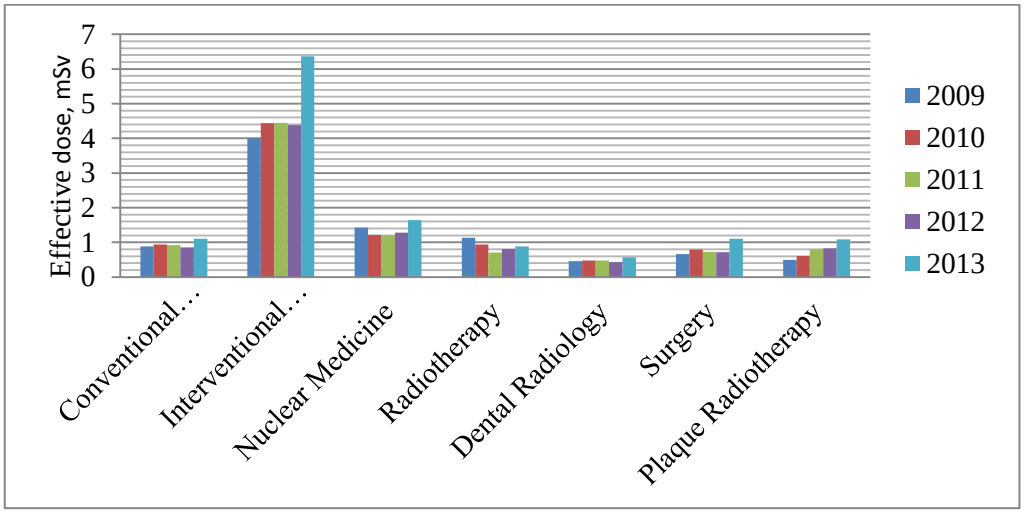

FIG 1. Average annual occupational effective doses in the health care service, 2009-2013

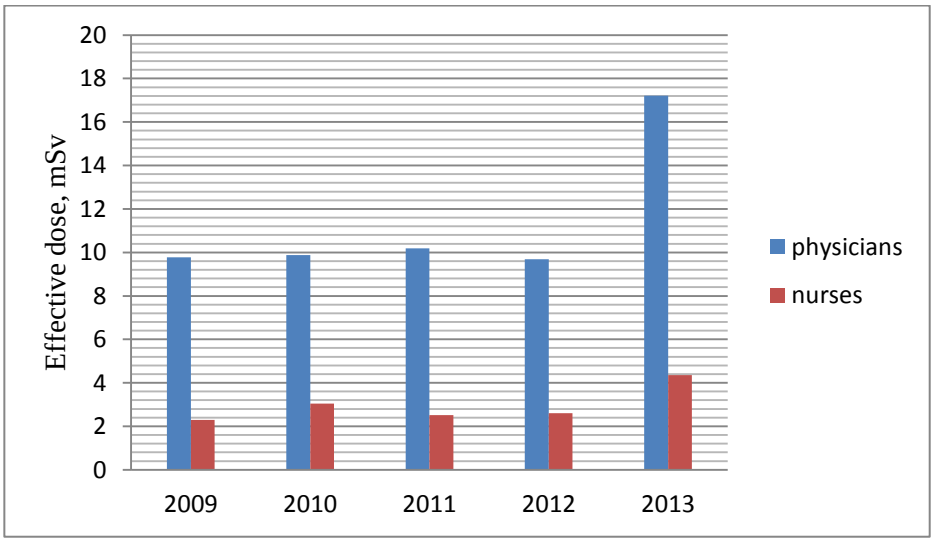

FIG. 2. Average annual occupational effective doses of interventional radiology staff using one dosimeter, 2009-2013

Interventional radiology tends to involve long procedure and long fluoroscopic time. Therefore, radiation protection and dose evaluation for interventional radiology staff is an important issue. Interventional radiology physicians currently wear two TLD-s, one under the personal lead apron or vest-skirt combination protective equipment (PPE) at waist level and one at neck level (collar) outside the personal lead apron. With the two dosimeters method, the effective dose was determined by the calculation algorithm from the NCRP report no 122. Number of physicians using two dosimeters has increased from 6 in 2009 to 16 in 2013. Interventional radiology nurses use single dosimeter outside the apron. Therefore, the evaluation of effective doses for nurses was mostly overestimated. The trends in average annual effective dose of interventional physicians using two dosimeters are shown in Table 2.

Radiation protection technology for effective dose reduction is increasingly important not only for interventional radiology staff, but also for a patient. The development of interventional radiology equipment and techniques and their intelligent use is to improve the safety of both sides, workers and patients. Interventional radiology staff uses for radiation protection a combination of tableside drapes (0,5 mm lead equivalent) and ceiling lead acrylic shields (0.5 mm lead equivalent). The interventional physicians and nurses wear a protective vest-skirt combination of PPE or apron (0.25-0.5 mm lead equivalent) and protective collar (0,5 mm lead equivalent). Some physicians use protective glasses (0.7 mm lead equivalent).

TABLE 2. AVERAGE ANNUAL EFFECTIVE DOSES FOR INTERVENTIONAL RADIOLOGY PHYSICIANS USING TWO DOSIMETERS, 2009-2013

Staff	Effective dose (mSv)				
	2009	2010	2011	2012	2013
Physicians	1,27	1,11	1,12	0,87	1,23

3. CONCLUSION

The overview shows that the occupational effective dose for interventional radiology staff using two or one dosimeters is continuously increasing. The growth of effective dose for interventional radiology physicians and nurses last year (2013) requires regular overview of radiation protection measures and upgrading them accordingly due to the development of technology.

REFERENCES

[1] RADIATION ACT
[2] REGULATION OF GOVERNMENT NO. 193, Effective dose and equivalent dose limits for the lens of the eyes, skin and extremities for exposed workers and members of the public, (2004).
[3] REGULATION OF GOVERNMENT NO 244, Statutes for maintenance of the State Dose Register of Exposed Workers (2004).
[4] EUROPEAN COMMISSION, COUNCIL DIRECTIVE 2013/59/EURATOM of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom.
[5] NATIONAL COUNCIL ON RADIATION PROTECTION AND MEASUREMENTS, Use of personal monitors to estimate effective dose equivalent and effective dose to workers for external exposure to low-LET radiation, NCRP Report No. 122, Bethesda MD: National Council on Radiation Protection and Measurements (1995).

EVALUATION OF OCCUPATIONAL DOSE IN ENDOVASCULAR PROCEDURES FOR HEPATIC CHEMOEMBOLIZATION

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Abstract

There is limited database on the occupational doses in chemoembolization for hepatocellular carcinoma treatment. The aim of this study is to evaluate the medical staff's doses in an attempt to optimize radiation protection. Methodology: Active dosimeters, by Mirion Technologies, Inc.(Instadose^{TM}) were used. Twenty patients were undergone to fluoroscopy and cineradiography series from up to 30s, using 2, 1 or 0.5 fps for each 10 s. The radiation doses to the left temporal region, chest and left ankle were registered, as well as in the anterior and posterior regions of nurses' thorax. Kerma-area product (KAP in Gy.cm²) for fluoroscopy and cineradiography were recorded. Results: The 3rd quartile (median) values for KAP obtained were 858.5(404) Gy.cm² for total dose, 574.5(295.6) Gy.cm² for Cine dose and 284(127.5) Gy.cm² for Fluoro dose. The occupational radiation doses were 0.28±0.17mSv (crystalline), 0.22±0.17 mSv (chest) and 1.6±1.7mSv (ankle). For nursing staff the occupational doses were: 0.063±0.063 mSv (anterior thoracic region) and 0.015±0.027mSv (posterior thoracic region). Discussion and Conclusion: The values obtained in the ankle region is about 10 times higher than in other regions suggesting the importance of installing table-mounted lead drapes for the physicians. The exposure values of nursing staff obtained justify wearing front and back shield apron during the procedures.

1. INTRODUCTION

The digital subtraction technique development allows physicians to study blood vessels through images with better resolution and minimum artifacts.[1] Diseases such as atherosclerosis, aneurysms, stenosis, dissections and arteriovenous malformation, are often diagnosed by angiography procedures, in which vascular structures are visualized after contrast injection. This technique is based on the vessel dissection, passing a guide wire through it where a catheter is introduced and will be positioned at the site to be treated [2]. One benefit of this technique is to decrease morbidity (risk of disability) and mortality, lower costs, as well as decrease the period of hospital stay[3]. Embolizations are therapeutic procedures that consist in obstructing a vessel with normal or abnormal blood flow. In this procedure catheter is inserted with a guide wire into the patient's artery, guided by fluoroscopy until it reaches the interest region. The liquid embolic is injected through the catheter occluding the artery. It is possible to make chemotherapy more effectively by introducing the drug directly into the tumor in procedures where blood flow into the tumor is blocked through embolization [4].

The hepatocellular carcinoma chemo-embolization (HCC) is the most commonly used treatment in patients with unresectable tumors or liver transplant. It is a procedure where tumor blood flow is blocked by microspheres, once chemotherapy has been administered [4]. In this procedure, patient and medical staff are exposed to high levels of radiation, through fluoroscopy and cinefluoroscopy, since the number of images processed and the procedure

time are high. Studies have shown that in this kind of complex procedures high doses are required which may result in damage to the patient's skin. Patients exposed to radiation dose above the threshold for deterministic effects in cerebral embolization procedures are reported in the scientific literature [5].

There is no reference dose level established for interventional procedures. However, the determination of these values contributes to optimize actions of radiation protection, minimizing the risks for professionals and patients [6].

During the procedure, the medical staff is positioned beside the patient and is exposed to secondary radiation scattered by the patient. The effective dose estimative is important to guide actions for protection of all medical staff involved in procedure. There are studies that report high radiation doses registered in medical staff [7].

Vanõ and its colaborators reported two cases of vascular interventional physicians, who perform their tasks without personal protective equipment (PPE), in which they were detected opacity in the lens due to exposure to secondary radiation. It was estimated that the equivalent doses in the lens of radiologists were between 450-900 mSv/year [8].

The aim of this study is to estimate the occupational doses of medical staff in real time during chemoembolization**,** in order to optimize the conditions for radiation protection during these procedures, minimize risks and extend the culture of safety.

2. METHODOLOGY

The study is being conducted at the university hospital, Escola Paulista de Medicina-UNIFESP using angiography equipment Philips INTEGRIS V5000, with an image intensifier with different diameters of fields. The equipment has been tested for performance verification of the adequacy of the quality parameters national standards (Portaria 453/98) [9]. Some of the features of the equipment are described below:

(a) Air kerma rate measured for diameter 31cm and 20 cm thickness of scattering material (nylon): 16.9mGy/min.
(b) Maximum rate at the entrance of the image intensifier: (Low: 41.24 mGy/min; Normal: 50.08mGy/min; High: 86.16mG/min)
(c) Layer semirredutora to 80 kVp: 5.5mmAl
(d) The images acquisitions are defined by pre-set standards by the manufacturer and in agreement with the medical staff.
(e) Patients were undergone a series of cinefluorography of up to 60s, depending on the complexity using 2, 1 and 0.5 fps every 20s.
(f) The console of the equipment has a display with the fluoroscopy time registered and kerma-area product (KAP) measured in Gycm² for fluoroscopy (Fluoro dose), cinefluorography (Cine dose) and total (Total dose).

Radiation doses received by the medical staffs were recorded in 20 therapeutic procedures using active dosimeters manufactured by Mirion Technologies, Inc. (Instadose^{TM} Dosimeter). The radiation doses were received by the left temporal, thorax and left ankle regions of physicians, as well as the nursing staff effective dose on the anterior and posterior regions of thorax. In each procedure the total time of exposure was registered. The KAP value (in Gy.cm²) for fluoroscopy (Fluoro dose) cineradiography (Cine dose) and total (Total dose) were also recorded.

Protective shields available such as, under-table and ceiling lead drapes, were not used due to operational difficulties imposed on projections / angles arc "C" during the procedure. However, the personal shielding (apron and thyroid shield) was used by all professionals exposed. The physicians who were responsible by the procedures also were using lead

glasses.

3. RESULTS

The 3rd quartile values for KAP of 20 patients who underwent chemoembolization ranged from 858.5 (404) Gy.cm² for Total dose, 574.5(295.6) Gy.cm² for Cine dose and 284(127.5) Gy.cm² for Fluoro dose. Cinefluoroscopy doses, ranged from 1 to 87% relative to the Total dose. The total exposure time varied from 6 to 41 minutes and the number of images taken in cineradiography mode ranged from 5 to 434 images.

Occupational doses recorded were: 0.28 ± 0.17 mSv (lens), 0.22 ± 0.17 mSv (thorax) and 1.6 ± 1.7 mSv (ankle). For nursing staff the following occupational doses were obtained: 0.063 ± 0.063 mSv (anterior thorax) and 0.015 ± 0.027 mSv (posterior thoracic region).

4. DISCUSSION

Unlike the thermoluminescent dosimeters (TLD), the use of an active dosimeter (Instadose™ Dosimeter) provided speedy response and efficiency in data collection. The data obtained in the ankle region of the physicians suggested that is important to use table lead drapes as collective protection.

Occupational doses received by the staff members are varying depending on technical/operational conditions and complexity of the procedures.

5. CONCLUSION

The values obtained in the ankle region is about 10 times higher than in other regions suggesting the importance of installing table-mounted lead drapes for the physicians. As expected, the nursing staff doses were lower than physician's doses, however, the levels of doses received justify wearing front and back shielded aprons.

REFERENCES

[1] BUSHONG, S.C. Ciências Radiológica para Tecnólogos. Elsevier Editora Ltda., 9ª ed., Rio de Janeiro, (2010) 709.
[2] LUNELLI, NEURI ANTONIO., Estimativa da dose no paciente e na equipe medica em procedimentos intervencionistas de neurorradiologia, (2012). Tese de Doutorado, Recife / Pernambuco / UFPE.
[3] MOONEY R.B., MCKINSTRY C.S., KAMEL H.A., Absorbed dose and deterministic effects to patients from interventional neuroradiology. Br. J. Radiol., 73 (2000) 745–51.
[4] HASHEM B.EL-SERAG.M.D.,M.P.H., Hepatocellular carcinoma. N. Engl. J. Med. 365 (2011) 1118-27.
[5] FLETCHER DW, MILLER DL, BALTER S, et al., Comparison of four techniques to estimate radiation dose to skin during angiographic and interventional radiology procedures. J. Vasc. Interv. Radiol.,13 (2002) 391–97.
[6] ALEXANDER M.D., OLIFF M.C., OLORUNSOLA O.G., et al., Patient radiation exposure during Diagnostic and therapeutic interventional neuroradiology proce- dures, J. NeuroIntervent. Surg., 2 (2010) 6–10.
[7] VAÑÓ E, GONZÁLEZ L, BENEYTEZ F, MORENO F., Department of Radiology, Complutense University of Madrid, Spain, Br. J. Radiol., 71(847) (1998) 728-33 (Lens injuries induced by occupational exposure in non-optimized interventional radiology laboratories).
[8] VAÑÓ E, GONZÁLEZ L, GUIBELALDE E, et al. Radiation exposure to medical staff in interventional and cardiac radiology, Br. J. Radiol., 71 (1998) 954–60.

[9] Agência Nacional de vigilância Sanitaria. Portaria/ MS/SMS n° 453 de 01 de junho 1998 (Ministério da Saúde – Brasil).

APPLICATION OF PROTOCOL FOR PREGNAT EMPLOYEES ACCORDDING TO THE IAEA SAFETY STANDARDS RECOMMENDATIONS IN RADIOTHERAPY DEPARTMENT AT HOSPITAL MEXICO

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Abstract

The application of International safety recommendations regarding pregnant workers, will be discussed. In the Radiotherapy Department at Hospital Mexico, the high number of women in different positions, implies the validation of a protocol to define actions that may reduce the level of radiation to which they are exposed. In the case of radiation technologist, the most of the circumstances will lead to reassignment, where the pregnant employee will be expected to perform all duties as assigned in the new position. The preceding procedures should be implemented in the new work area as well, in order to minimize potential radiation exposure to the fetus.

1. INTRODUCTION

Radiation protection is important for all medical workers, but it takes on a new meaning when a technologist or physician becomes pregnant. It is incumbent upon the pregnant worker who may be exposed to radiation to protect herself and her fetus during pregnancy. Because of its power to mutate DNA or cause cell death, radiation can trigger an array of conditions in an unborn child. This occurs at levels of radiation exposure that are typically not reached when proper occupational protection is in place to reduce occupational radiation.

Declaration of Pregnancy: When a medical worker of the radiotherapy department confirms that she is pregnant, the first step in protecting herself and her fetus is to declare her pregnancy to the coordinator of the area. In the eyes of some institutions, the institution is not liable for proper precautions to protect a pregnant worker from radiation unless she has officially acknowledged her pregnancy. Once a declaration has been filed, the fetus is treated like a member of the general population [1].

The National Council on Radiation Protection and Measurements (NCRP) [2] of U.S. recommends an occupational radiation fetal dose limit of 5.0 mSv during an entire pregnancy (with a daily limit of 0.025 mSv), and less than 0.5 mSv per month. The International Commission on Radiological Protection (ICRP) recommends less than 1.0 mSv total fetal exposure during an entire pregnancy. In general, these limits are achievable with the proper precautions in place.

Employer Obligations: In an ideal situation, a worker's declaration of pregnancy should trigger the following actions by her employer:

(a) Careful evaluation of the woman's environment to determine whether there are any risks of radiation exposure that could exceed the limit of exposure to her fetus.
(b) A full explanation provided to the pregnant employee about the potential risks of fetal radiation exposure, local policies, and dose limits.

(c) A review of what methods and extra protection the woman can use to reduce her exposure.

The most dangerous time for a fetus to suffer the negative effects of radiation are weeks 8 through 15 of gestation, when its organs and nervous system are forming. Severe issues have a higher probability of occurring after 1.0 to 2.0 Sv of radiation exposure, but defects can occur at levels of 100 to 200 mSv. The risk of developing childhood cancer is highest if a fetus is exposed to 200 to 250 mSv between weeks 2 through 15 of gestation, while more than 100 mSv of radiation can increase the frequency of childhood cancer and cause small head size, seizures, and reduced IQ. The risk of developing childhood cancer is highest if a fetus is exposed to 200 to 250 mSv between weeks 2 through 15 of gestation, while more than 100 mSv of radiation can increase the frequency of childhood cancer and cause small head size, seizures, and reduced IQ.

2. METHODS OF APPLICATION AND DISCUSSION

Workers can employ the radiation protection principles of time, distance, and shielding to reduce their exposure to radiation, and consequently that of their fetus. Some researchers also suggest that, because of the possibility that a learning curve will extend exposure time, a pregnant woman may want to postpone learning new radiologic techniques until after she gives birth.

2.1. Fetal Dose Monitoring

To determine how much radiation her fetus has received, a good recommendation for the pregnant worker is to wear a monitor at her waistline. Passive monitors give the dose reading at the end of the day or month, while active monitors measure in real-time and sound an alarm if the exposure limit is exceeded. An active monitor may provide more peace of mind because a woman will know whether she is receiving too much radiation and immediately adjust her position to reduce it.

2.2. Policy

Regulations of the Health Ministry of Costa Rica [3], governing the occupational exposure to ionizing radiation requires that the radiation dose to the fetus of occupationally exposed pregnant women be held to 5 mSv or less during the pregnancy. All radiation employees, especially women of childbearing age, are encouraged to carefully monitor their dosimeter badge readings and become familiar with their potential sources of exposure and means of minimizing it.

2.3. Procedure

When an employee informs her coordinator in writing, that she believes she is pregnant, the coordinator shall review the worker’s employment duties, and modify the duties or activities, where reasonable to do so, to ensure that the worker’s effective dose of ionizing radiation does not exceed the applicable maximum effective dose limits

i. If the worker cannot be assigned to other duties, the Radiation Protection Officer will issued a second dosimeter to be worn during the gestation period at waist level to serve as a measure of fetal exposure. The supervisor will take all reasonable steps to maintain the radiation exposure for this employee (as recorded by the waist level badge) to less than 0.50 mSv/month.

ii. Pregnant radiation employees should be particularly diligent in avoiding unnecessary exposure during their regular work assignment. Accordingly, employees should minimize their time of exposure, maximize their distance from the radiation source.
iii. In keeping with IAEA recommendations to hold embryo/fetal exposures ALARA (As Low As Reasonably Achievable), if the pregnant employee is currently assigned to duties whereby her potential for exposure is significantly increased (e.g. Operator of High energy machines), she may be reassigned to duties involving lower potential for exposure for the duration of her pregnancy, if such temporary reassignment is deemed administratively practical.
iv. After reassignment, if practical, the pregnant employee will be expected to perform all duties as assigned. The preceding procedures should be implemented in the new work area as well, in order to minimize potential radiation exposure to the fetus.
v. A declared pregnant employee who is exposed to workplace radiation may be eligible to utilize leave time consistent with applicable Human Resources policies.
vi. A copy of this policy will be given to all female radiation employees at the beginning of their employment. A second copy will be provided whenever a pregnant employee notifies her coordinator of her pregnancy, in writing.

In the past, several modification in duties or reassignment according to the previous procedure have been done in the Radiotherapy department, some of them are presented in the Table 1.

TABLE 1. GENERAL REASSIGNMENT OF SOME EMPLOYEES

Position	Reassignment
Radiation Therapy Physician	Same duties except working with patients under Cobalt Unit treatment
Nurse	Same duties except working with patients under Cobalt Unit treatment
Radiation Therapy Technician	Administrative tasks, management of patient lists, patient appointments.

3. CONCLUSION

Ionizing radiation may affect any living tissue through which it passes, potentially leaving damage in its wake. Many times radiation has lifesaving effects, but for a Radiotherapy technologist Physicist or Physicians who uses it as part of his or her occupation, radiation could have detrimental consequences. It is possible for medical imaging and radiation therapy professionals to have long, safe careers when they monitor their radiation exposure and employ the 3 techniques for radiation protection: time, distance, and shielding. Pregnant women who work in settings that could expose them to radiation should take additional steps to protect a developing fetus, such as wearing a fetal radiation monitor.

REFERENCES

[1] OIEA, Normas básicas internacionales de seguridad para la protección contra la radiación ionizante y
para la seguridad de las fuentes de radiación, OIEA colección de seguridad n° 115, Vienna (2004).
[2] NCRP, Report No. 174, Preconception and prenatal radiation exposure: Health Effects and Protective

Guidance, (2013).
[3] DECRETO EJECUTIVO N° 24037-S, Reglamento sobre protección contra las radiaciones ionizantes,
(2002).

OPTIMIZING FDG DISPENSING: AN ENVIRONMENTAL AND WORKERS EVALUATION

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Abstract

Optimization of radiation protection is a continuous challenge in any production facility. This work shows three main exposure situations in a PET centre. The first one is associated to the original layout of the production laboratory. An important increase in dose rate was detected during FDG transport from the hot cell to the laminar flow, where manual dispensing was performed. As a consequence, there was a modification in the laboratory design and the laminar flow was changed to a nearer position in the hot cell. Values of dose rate during FDG preparation vs. time decreased by 40% of the dose detected in a centralised radioprotection system. The second situation was related to the increase of 243.5% in FDG production in the centre, since the second semester of 2013. Dose rate with the new layout only raised 20.3%, showing the effectiveness of the modification. Finally, hand doses of workers involved in the task were analysed in order to co-relate them to the increase of produced activity. The raise in hand doses in this case was of 35.8%.

1. INTRODUCTION

The Uruguayan Centre of Molecular Imaging (CUDIM) is the first PET centre in the country, inaugurated in 2010. The institution is dedicated to develop research, training and applications in health sciences, where diagnosis and biomedical research activities are promoted. Clinical examinations for patients primarily in the fields of oncology and neurology are performed.

In this frame the aim of this work was to optimise the exposition of workers to radiation during FDG dispensing in the Radiopharmacy Production area.

The original laboratory layout resulted in an unnecessary exposure of the staff during FDG production. The situation was analysed and a new layout of the laminar flow was proposed, the influence of this modification in the total dose of the laboratory environment is presented in this paper.

2. METHODS

The centre operates with a MediSmart centralised system, particularly the FDG laboratory has a GM detector, that records the Dose and Dose Rate every second in a 24/7 regime. Supported by these data and TLD dosimetries of staff members dedicated to production, three situations were analised.

The first one was associated to an important increase in dose rate that was detected during FDG manual transport in a lead shield from the hot cell to the laminar flow. The need of change in the laboratory layout was observed and suggested by workers involved in the

FDG production. As a result, there was a modification in the laboratory arragment and the laminar flow was switched to a nearer position to the hot cell and a shielded tubing was installed to transport the FDG solution to the shielded laminar flow. Values of dose rate during FDG preparation vs. time were plotted for productions before and after layout modification. Special attention was paid to compare productions of the same magnitude. Areas under the curve (AUC) profiles were calculated and total dose was determined for each production.

The second situation was related to the increase of FDG production in the centre, an increment higher than 240% has occurred since the second semester of 2013 till date. Thus, an evaluation of the environmental dose was compared with the previous layout of the laboratory.

In third place, hand doses of 3 workers involved in this task were analysed in order to relate them to the increase of produced activity.

3. RESULTS AND DISCUSSION

The displacement of the laminar flow to a new position lead to a drastic decrease in dose rate during FDG production as can be seen in Fig. 1 where AUC vs activity production were plotted; each series corresponds to similar amounts of total activity.

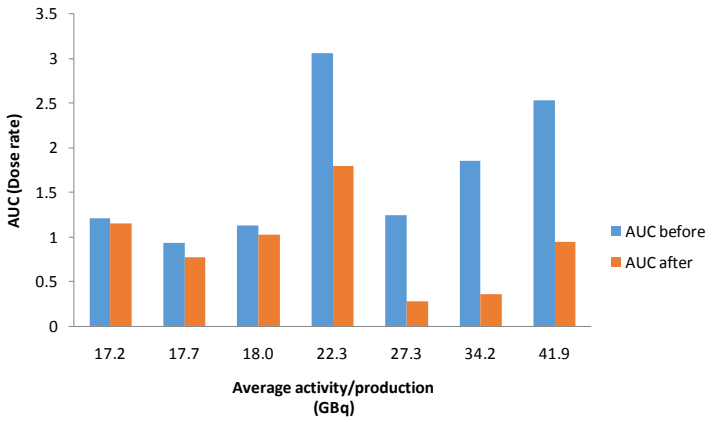

FIG. 1. AUC behaviour before and after hot cell layout modification

Fig. 2, a decrement in dose rate percentage is observed, where the highest decrease is associated with the higher amounts of produced activity.

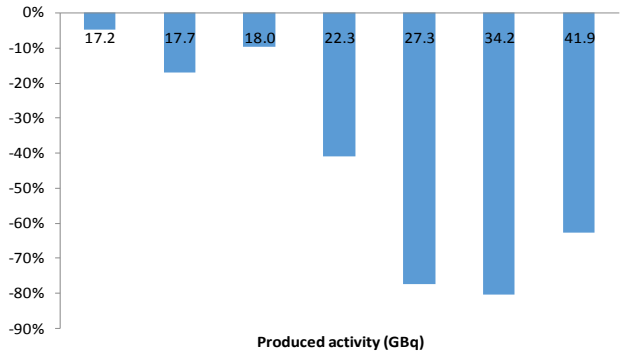

FIG. 2. Decrement profile in dose rate percentage

Before the layout change, the Dose rate/GBq ratio was 0.066, but it decreased to 0.02 with the same amount of FDG production, after laminar flow new situation, that means 71.2% less dose rate per produced Bq. Notwithstanding the important increase in FDG production of 243.5% during the last 9 months, the Dose rate/GBq ratio raised to 0.08. The total dose rate was only 20.3% higher than the previous conditions as shown in Table 1. The mean dose rate in all situations was always into the legal limits of 10 µSv/h.

TABLE 1. DOSE RATE/GBq RATIOS AND THEIR MODIFICATION WITH ACTIVITY PRODUCTION

Situation	Dose rate (µSv/h)	µSv/MBq	Decrement in Dose/GBq	Increment in Dose /GBq after 243.5% if FDG activity
Before layout change	2.72	66		--------
After layout change	1.01	20	71.2%	--------
After production increase	10	80	-----	20.3 %

The following analysis comprises doses in hands from January to December 2013 of workers involved in FDG production. Average hand doses profile are compared with the total produced activity in Table 2. An increase of almost 207% in FDG activity leads to a 14% increment in hand doses. A similar situation occurs when total activity growths 245% hand doses raise up to 48% from the initial situation.

TABLE 2. DOSE PROFILE TO THE HANDS COMPARED WITH TOTAL FDG ACTIVITY

Period of analysis	Average hand doses (mSv)	% increment in hand doses	Average activity of FDG (GBq)	% increment in activity of FDG
01-2013 to 05-2013	9± 3	14	30.2	207
06-2013 to 09-2013	11 ± 1		62.4	
10-2013 to 12-2013	14 ± 3	48	74.4	245

4. CONCLUSIONS

The change in layout lead to a decrease in environmental dose. Moreover, with the important augment in FDG demand 2.4 times higher; the total dose rate was of the same order of magnitude after the modifications. This is a clear example of continuous improvement in radiation protection optimisation. Despite hand doses are far below the allowed limits, it is obvious that if the centre continues increasing FDG production, it will be necessary to pay attention to dispensing process in order to keep under control these values.

REFERENCES

[1] GUÍA DE SEGURIDAD RS-G-1.1
[2] SAFETY REPORTS SERIES No. 64 Programmes and Systems for Source and Enviromenmental Radiation Monitoring.
[3] REAL DECRETO 783/2001, Reglamento sobre protección sanitaria contra radiaciones Ionizantes, (2001).

HAND DOSE LEVELS IN FLUORO-CT GUIDED PROCEDURES – USE OF A NEEDLE HOLDER

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Abstract

In this work the dose received by the dominant hand of the interventional radiologist was assessed during fluoro CT guided procedures while using a needle holder. The results show that the needle holder significantly reduced the per procedure dose levels and hence the total accumulated dose. The measurements were performed with thermoluminescence extremity dosemeters inserted in special gloves prepared in-house. $H_p(0.07)$ on the five finger tips and bases of the dominant hand was measured. Although the dose values show a large variation, in general, the middle, ring and little fingers are the most exposed. Maximum dose values per procedure in the range 5.7 to 8.1 mSv were obtained. The base of thumb and little fingers are the less exposed with a maximum dose value around 1 mSv. The values are nearly 10 times lower compared with previous results where this tool was not considered. One

concludes that the use of needle holders is strongly recommended in order to increase the distance between the hand and the primary beam and hence reducing the dose received by the dominant hand.

1. INTRODUCTION

Optimizing individual monitoring in CTF-guided procedures requires a reasonable understanding of the typical dose distributions which may be difficult to attain as they are very different from those of conventional fluoroscopy. High radiation dose to the radiologist´s hands is expected [1] Earlier papers report that the dose to the radiologist's hands in the direct beam could reach 120 mSv per procedure [2]. Since the hands of the IR are very close to the imaging/radiation plane, high hand exposures are expected. The use of biopsy needle holders may reduce the dose to the hand by increasing the distance to the scan plane [3, 4, 5] although this tool may decrease tactile feedback and may lead to longer fluoroscopy times [6]. The results presented by Stoeckelhuber shown that a long needle holder decreased the dose rates by 30% [6]. Opinions differ regarding how easy it is to use needle holders. Some authors report no difficulty [2, 3], while others argue that needle holders decrease tactile feedback and grip [4, 7, 8]. Dedicated needle holders have been developed [2, 3, 6, 9] but many authors prefer metallic sponge forceps or towel clamps due to their widespread availability, lightweight, strength, ease of sterilization and relatively low cost [4, 5, 7].

The aim of this work is to characterize the dose distribution to the dominant hand of the interventional radiologist (IR) in CTF-guided biopsies with the use of a needle holder to increase the distance to the radiation plane in real-life clinical conditions. Per procedure dose values to the dominant hand were obtained during 34 procedures when needle holders was used, in conditions as similar as possible to those of previous measurements where the needle was directly gripped [10-12] (same radiologist, CT-scanner, type of procedures, method of random sample selection and sample size) to allow for the comparison of results.

2. MATERIAL AND METHODS

CTF procedures were all performed at IPO-Porto by a single experienced interventional radiologist. The CT-scanner was a Toshiba Asteion 4-slice with the following parameters: tube voltage of 120 kV, current 40 mA or 50mA, rotation of 0.75s and 8mm of beam collimation. The typical biopsy procedure has been described in detail elsewhere [13]. The intermittent imaging method proposed by Silverman is always used, so that the hands of the interventionist are kept away from the beam during irradiation [7]. However, in some situations the quick-check method is combined with periods of needle manipulation during real-time imaging and irradiation. To prevent direct irradiation of the hand, an improvised needle holder (towel clamp) was used. This technique is different to that previously reported, where the needle was held by the side handle [13].

For the assessment of the dose to the hands, thin plastic gloves were developed in-house with special casings for the insertion of high-sensitivity thermoluminescence extremity detectors used on a per procedure basis [10]. A total of 10 detectors of LiF:Mg,Cu,P (TLD-100H) of the EXT-RAD type were placed on the casings at the tip and base of all fingers as shown in Fig. 1. A sterylized glove was used on top. The dosemeters were calibrated in terms of $H_p(0.07)$ at the Secondary Standard Dosimetry Laboratory of IST-LPSR using a N120 X-ray beam incident on a ISO rod phantom. The readouts were performed on a Harshaw 6600 reader using predefined cycles, the day after irradiations [10]. In this case study, the hand dose assessment was performed in 34 biopsy procedures where CTF-guidance was necessary, mainly to the lungs (27), but also abdomen (5) and bone (2).

FIG. 1. Dominant hand with ten extremity dosemeters placed at the tip and base of each finger

3. RESULTS AND DISCUSSION

Fig. 2 shows the distribution of per procedure $H_p(0.07)$ dose measurements organized by dose intervals. Almost all procedures showed dose levels below 1 mSv, particularly in the case of the detectors placed at the base of the fingers. The detectors placed on the tip showed a higher variation, althought more than 80% of the values are also below 1mSv.

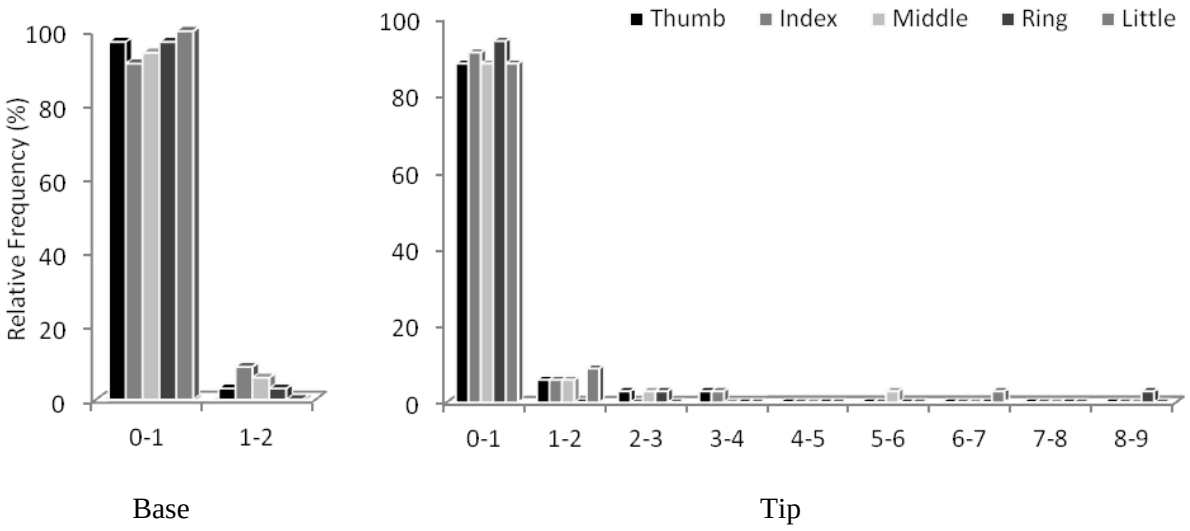

FIG. 2. Distribution of the per procedure $H_p(0.07)$ values measured on the base (left) and tip (right) of each finger organized by dose intervals (in mSv).

The maximum and mean (per procedure) dose values obtained on the tip and base of each finger are presented in Table I. It can be observed that the dose levels received on the tip of the fingers is higher than on the base. The tip of the middle, ring and little fingers are the most exposed regions with maximum values of 5.68, 8.09 and 6.05 mSv, respectively. The results also show that the base of thumb and little fingers are the less exposed regions with mean values of 0.23 and 0.33 mSv and maximum values of 1.09 mSv and 0.99 mSv, respectively.

TABLE I. MAXIMUM AND MEAN $H_P(0.07)$ VALUES (PER PROCEDURE) MEASURED AT THE TIP AND BASE OF EACH FINGER, EXPRESSED IN mSv (ALL 34 PROCEDURES CONSIDERED)

	Thumb		Index		Middle		Ring		Little	
	Tip	Base	Tip	Base	Tip	Base	Tip	Base	Tip	Base
Maximum	3.89	1.09	3.10	1.25	5.68	1.28	8.09	1.09	6.05	0.99
Mean	0.51	0.23	0.49	0.37	0.55	0.39	0.57	0.36	0.53	0.33

The total accumulated dose integrated in the 34 CTF procedures in all measurement positions is shown in Table II. The results obtained in previous work without the use of the needle holder [12] are also included for comparison. A higher number of procedures was analised in Ref. [12], however, the results presented herein were interpolated so that both situations could be compared. The results obtained when the needle holder is used show that in general the tip of the fingers is more exposed than the base. This statement is valid in both situations, with and without the use of the needle holder. When the needle holder is used, the dose values at finger tips are very similar in the range 17-19 mSv. Taking into account the annual dose limit to the extremities of 500 mSv the results suggest the IR could perform approx. 850 procedures every year.

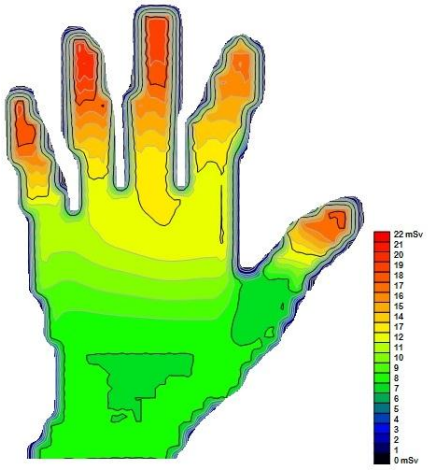

FIG. 3. Total dose to the dominant hand of the IR considering all 34 procedures. (colour grading to guide the eye).

TABLE 2. INTEGRATED $H_P(0.07)$ TO THE TIP AND BASE OF EACH FINGER: WITH NEEDLE HOLDER (THIS WORK) AND WITHOUT [12]

	Tip		Base	
Finger	With (this work)	Without [12]	With (this work)	Without [12]
Thumb	17,3	118	7,8	26
Index	16,7	186	12,6	89
Middle	18,7	179	13,3	101
Ring	19,4	216	12,2	87
Little	18,0	133	11,2	71

Compared with previous results [10-12] obtained by this team and particularly in Ref. [12] the use of the needle holder significatly reduced the dose levels on the hands of the IR, almost by a factor of 10. Whithout the needle holder the dose to the tips is much higher than the dose to the base, in some cases by a factor of 2; the tip of the index, middle and ring fingers is also more exposed than the thumb and little fingers.

With the needle holder tool, the number of procedures with $H_p(0.07)$ values below 1 mSv increased and the maximum dose values considerably decreased. The dose values to the tips is again higher that the dose to the base, but all five fingers are more homogeneously irradiated and at the same time the difference between the dose to the tips and to the bases is not so large.

4. CONCLUSIONS

In this work the dose received by the IR on the dominant hand was studied in 34 CTF-guided procedures with the use of a needle holder. The results obtained suggests a significant dose redution on the exposure to the hand of the IR, highlighting the importance of using needle holders as a protective tool for optimization of radiological protection in CTF procedures. On the other hand, when the needle holder is used the similarity of the dose values on all fingers, irrespective of the position (tip or base) minimizes the uncertainty on the selection of the dosemeter position.

In the absence of the needle holder tool the results suggest that the dosemeter should be worn on the tip of the index, middle or ring fingers. But if the needle holder is used the usual (and comfortable) ring type dosemeter can be safely used.

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REFERENCES

[1] D.L. MILLER et al. Occupational radiation protection in interventional radiology: A Joint Guideline of the Cardiovascular and Interventional Radiology Society of Europe and the Society of Interventional Radiology, Cardiovasc. Interv. Radiol. 33 (2010) 230-239.

[2] R. KATO et al. Radiation dosimetry at CT Fluoroscopy: Physician's hand dose and development of needle holders, Radiology 201 (1996) 576-578.

[3] IRIE, T., KAJITANI, M. AND ITAI, Y. C.T., Fuoroscopy-guided intervention: marked reduction of scattered radiation dose to the physician's hand by use of a lead plate and an improved I-I device. J. Vasc. Interv. Radiol., 12 (2001a) 1417–1421.

[4] B. DALY et al., Evaluation of biopsy needles and prototypic needle guide devices for percutaneous biopsy with CT Fluoroscopic Guidance in simulated organ tissue, Radiology, 209 (1998) 850-855.

[5] S.K. CARLSON et al., Benefits and safety of CT-Fluoroscopy in interventional radiological procedures, Radiol. 219 (2001) 515-520.

[6] B.M. STOECKELHUBER et al., Radiation dose to the radiologist's hand during continuous CT fluoroscopy-guided interventions, Cardiovasc. Intervent. Radiol. 28 (2005) 589-94.

[7] S. G. SILVERMAN et al., CT-Fluoroscopy guided abdominal interventions techniques, Results and radiation exposure, Radiology, 212 (1999) 673-681.

[8] S. K. CARLSON et al., Benefits and safety of CT-Fluoroscopy in interventional radiological procedures, Radiology, 219 (2001) 515-520.

[9] M. IRIE et al., Biopsy of lung nodules with use of I-I device under intermittent CT-Fluoroscopic guidance: preliminary clinical study. J. Vasc. Interv. Radiol., 12 (2) (2001) 215-219.

[10] M. PEREIRA et al., Preliminary assessment of the dose to the interventional radiologist in Fluoro CT guided procedures, Radiat. Prot. Dosim. 144 (2011) 448-452.

[11] J.G. ALVES et al., Assessment of the dose to the interventional radiologist in Fluoro-CT guided procedures, presented at the 8th International Workshop on Individual Monitoring of Ionizing Radiation, Oarai, Japan, 1st-2nd December (2012).

[12] J. PEREIRA et al., Hand monitoring in Fluoro-CT guided procedures. Poster presented at the 20th Int. Conference on Medical Physics and Biomedical Engineering, Brighton, September (2013).

[13] N. BULS, J. PAGES, J.DE MEY AND M. OSTEAUX, Evaluation of patient and staff doses during various CT Fluoroscopy guided interventions, Health Phys., 85 (2) (2003) 165-173.

RADIATION EXPOSURE OF OCCUPATIONALLY EXPOSED PROFESSIONALS IN THE AREA - MEDICAL

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ABSTRACT

The aim of this study was to determine which medical professionals are subjected to greater exposure to ionizing radiation. To do a retrospective analysis of the results of individual dosimeters, thirty users from different medical specialties, sent by the Bolivian Institute of Nuclear Science and Technology for a period of 30 months. The specialities from state hospital included the services of Nuclear Medicine, X-ray, CT and where you have the service of Interventional Cardiology. Comparison was made of radiation exposures among different services. The review of readings of the dosimeters indicated that the highest exposure dose corresponding to two medical professionals of interventional cardiology with doses of 43.7 mSv and 23.1 mSv, respectively, compared to 11.2 mSv and and 29,4 4 mSv of the technicians who accompany the intervention procedures. In terms of radiation exposures of the workers occupationally exposed in nuclear medicine, the dose received by the technologist - 8 mSv, is higher as compared to that of nuclear physician - 3.2 mSv. It is obsrved that larger exposures to ionizing radiation in the hospital under investigation corresponds to the occupationally exposed professionals who spend more time with the patient, i.e., medical interventional cardiologist and medical technologist nuclear.

1. INTRODUCTION

Worker exposure to ionizing radiation in Bolivia has increased in different fields such as medicine, industry, agriculture, etc. In this paper radiation exposures of workers in the area of medicine are considered. In recent times, before installing short-term medical equipment such as CT scanners, nuclear magnetic resonance tomography SPECT single photon emission, cyclotron installation projects - positron emission tomography, or building long-term projects such as nuclear reactors, certain requirements should be met such as having institutional license, and workers having individual licenses for performing complex procedures such as interventional radiology techniques - comprising diagnostic and therapeutic techniques guided flouroscopia.

2. BACKGROUND

Interventional Radiology originated in diagnostic radiology as a subspecialty of invasive diagnosis. Interventional radiology is now a therapeutic and diagnostic specialty that encompasses a broad range of therapeutic image-guided procedures, minimally invasive as well as invasive diagnostic images. The range of diseases and possible bodies undergo therapeutic image-guided procedures is broad, constantly evolving and includes structures and vascular diseases, gastrointestinal, hepatobiliary, genitourinary, lung, skeletal muscle and the

central nervous system. Complicated surgeries are now being replaced by these comparatively simple procedures, benefiting both the patients and the health care system, thus reducing the risk involved, the duration of hospital stay for the patient, and the total cost of the procedure.

However, interventional radiology is a medical specialty of radiology that provides higher doses to occupationally exposed professionals. Occupational doses are high, having been identified deterministic effects, such as cataracts and depilation in interventional medicine. The high doses are caused by the proximity of some team members, especially the interventional physician to the patient.

To reduce exposure to X-rays, the doctor must use leaded apron and patient shielding for protection. The main materials protection for occupationally exposed are: leaded apron, thyroid necklace, leaded glasses and gloves.

It should be noted that professionals working in interventional radiology rooms are cardiologists, orthopedic surgeons, vascular surgeons, neurologists, etc., who have no training in radiation protection. To minimize exposures, following are the considerations:

(a) X-ray spectrum emitted by the pipe may be modified to reduce patient dose.
(b) Collimation is the best way to reduce the dose to the patient and operating personnel, while producing improved image quality.
(c) The use of grid to reduce radiation scattered implies an increase in the radiation dose to the patient. Therefore, whenever possible, the grid should be removable during interventional radiology procedures.
(d) The use of carbon fiber material between the patient and the image receptor, which combines high strength and low loss, minimizing attenuation caused by the materials and results in a reduction of patient dose.
(e) The use of iris and automatic brightness control, digital storage and pulsed fluoroscopy associated with the process of image acquisition reduced doses to patients during interventional procedures involving fluoroscopy.
(f) Digital radiology has considerable potential for reducing patient doses in interventional radiology.
(g) Changing the direction of the projection, the approach to the patient enhancer and increased focus-skin distance of the patient are excellent methods to reduce patient dose.
(h) The use of sizes smaller intensifier (magnifiers) implies an increase in patient entrance dose and the image intensifier dose rates greater demand for smaller fields.

3. METHODOLOGY

For the preparation of this work, A retrospective analysis was made of the results of individual dosimeters sent by the Bolivian Institute of Nuclear Science and Technology of a 30-month period corresponding to thirty users in different medical specialties where they work with ionizing radiation. The reference was a state hospital with the services of Nuclear Medicine, X-ray, CT and where has the interventional Cardiology service. A comparison of radiation exposures between services was also made.

The dosimeters are read on a quarterly basis, and dosimeter user had to pass a course on radiation protection, and should have individual license. The Radiation Protection Act and corresponding regulations states that: "The effective dose limit for workers is 20 mSv per year, averaged over five consecutive years, and must not exceed 50 mSv per year (Cap.III, Article 18).

4. RESULTS

The review and analysis of 30 months' data consisting of the results of dosimeter readings indicated that increased exposure in occupationally exposed workers corresponding to two medical professionals interventional cardiologists are 43.7 mSv and 23.1 mSv respectively in relation to technical professionals who received doses of 11.2 and 29, 4 mSv respectively.

It also indicated that the radiation exposure of occupationally exposed worker - nuclear medicine technologist - is higher at 8 mSv in comparison with the nuclear physician who received a dose of 3.2 mSv.

5. DISCUSSION

Courses have been organized with the objective of updating the contemporary scope of interventional cardiology and the vascular area, emphasizing interventional aspects in the management of cardiovascular diseases. Many of these centers have assisted numerous cardiologists, interventional cardiologists, interventional, cardiovascular surgeons, other disciplines physicians, residents, students, technicians and nurses. Still- images are taken of each of the proceedings, live, from the catheterization laboratory to the auditorium, where more than 100 participants who followed step-by-step the actions and dynamics of interventional work sick gathered.

Particularly striking, is the lack of content of radiological protection in these centers that do not include these protection issues, probably due to lack of knowledge or the lack of importance given to it, which is of concern to the exposed worker and the patient.

6. CONCLUSION

Increased exposure to ionizing radiation in the hospital under analysis correspond to the occupationally exposed professionals who spend more time with the patient, generally, medical interventional cardiologist and nuclear medical technologist. More studies may be required to confirm these findings.

This group of personnel would be encouraged through courses, trainings, conferences of different medical specialties to address the issues of both patient and the personnel safety who are occupationally exposed to radiation.

REFERENCES

[1] BUSSHONG S, Manual de Radiología para Técnicos, Física, Biología y Protección radiología 5ta. Ed. Mosby Dayane, Libros Madrid.

[2] PROTECCION RADIOLGICA Parte I: Conceptos Generales. Colección Sanidad Ambiental. Madrid, (1988) 11-13.

[3] BEDOYA RODRIGUEZ R E, Evaluación de la dosis recibida por personal ocupacionalmente expuesto en radiología intervencionista. Panamá (2006).

[4] PROTECCION RADIOLOGICA 119. Multimedia and audovisiual radiation protection training in interventional radiology. MARTIR. martir@med.ucm.es

[5] SOCIEDAD BOLIVIANA DE CARDIOLOGIA I Jornada de Cardiologia. intervencionista http://www2.bago.com.bo/sbc/latido/Vol6_n7/html/jornada1.html

[6]HUANCA Sardinas, et al., Exposición: Dosimetría en trabajadores ocupacionalmente expuestos en Medicina Nuclear y Radioinmunoanálisis del Instituto de Medicina Nuclear, Simposio Internacional de Protección radiológica, Cusco, Peru; (2012) 28-29.

PRELIMINARY DOSIMETRY STUDY IN OCCUPATIONAL RADIOLOGY IN MOROCCAN HOSPITALS

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Abstract

The objective of this study, conducted in two public hospitals and two private clinics, is to estimate the equivalent dose received by the doctors during procedures requiring the use of X-rays in Moroccan radiology departments. Medical interventions considered are involved in coronary, stent implantation, embolization and dental radiology. The measurement procedure is based on the estimation of the effective and equivalent doses in organs or tissues most exposed, using thermo luminescent dosimeters (TLD). The study gives preliminary results of a survey that should be supplemented by measurements permitting to establish a radiation protection system. This work carried out by the research team is in progress. The measurements show that the effective (bellow the lead aprons) is less than 3 mSv and the dose on the front (corresponding to crystalline dose) is of the order of 3.8 mSv.

1. INTRODUCTION

Interventional radiology, bringing an undeniable benefit to the patient, appears as the most radiological risk for operators especially for extremities and the lens. Indeed, the personnel working close to the patient in a field of scattered radiation can therefore receive, during a certain period relatively high doses of radiation. The values depend on several parameters, which we mention the experience of the doctor, the nature of the act, the difficulty of the case and the urgency of the operation. In this context, radiation safety training of personnel and dosimetric studies undertaken periodically are necessary for optimizing radiological procedures as well as the dose to patients and workers. Particular attention was paid to the evaluation of the dose to the lens by the International Commission on Radiological Protection ICRP [1] has recently lowered the threshold equivalent dose to the lens of 2 to 0.5 Gy.

2. MATERIALS AND METHODS

The present work concerns estimation of annual exposition to RX of doctors in three cardiology, dental radiology and angiography departments:

(a) A department of Cardiology at the University Hospital of Rabat where the study involved four doctors for a period of three months,
(b) A private cardiology clinic where one doctor was involved for a period of 28 days,
(c) A private dental radiology department for a period of two weeks.

The measurements were carried out using thermo luminescent dosimeters. Doses are identified Hp (10) and Hp (0.07). For the lens, the measurement corresponds to a dosimeter placed on the left side of doctor front.

Dosimeters were placed at the chest and gonads levels below and above the lead apron, at shoulders, on front thyroid (one case) and wrist (one case). For reasons of confidentiality, we note the department by A, B, C and D and we call D1, D2, D3, D4, D5, D6 and D7 the doctors who participated in this study.

3. RESULTS AND DISCUSSIONS

In the following table we group the results of measurements for doctors D1, D2, D3 and D4 in department A. We calculated the average annual dose based on observations taken by our staff during the period of data collection.

TABLE I. ANNUAL DOSE IN EXTREMITY IN CARDIOLOGY (mSv)

Doctor	Chest 1 (ext)		Chest 2 (int)		Gonad 1 (ext)		Gonad 2 (int)		Right Shoulder	Left Shoulder
	Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)
D1	8.12	8.47	1.91	1.84	7.00	8.35	1.58	1.54	1.76	1.97
D2	8.47	9.09	1.92	2.03	6.73	7.46	1.88	1.84	2.56	2.56
D3	5.33	5.15	2.15	2.14	3.34	2.53	1.79	1.75	6.78	7.93
D4	18.19	19.63	2.29	3.17	16.9	18.8	13.41	12.1	13.01	14.80

Results show that the dose depends on the intervention type and the doctor's experience as the four doctors work in the same conditions. Note that the dose received by doctor, D4, is high and close to the dose limit value recommended by international regulations. This doctor ensures mainly stent implantation while the others are primarily involved in coronary angiography. The doctor C was known for his rapid actions during the intervention and the expertise acquired through his professional experience.

A previous study [2] showed that in the case of catheterization, the annual dose is 3.8 mSv on the apron is 1.4 mSv on the front and 0.2 mSv under the protective apron. In our case, the dose is equal to 3.8 mSv and 1.45mSv on the front, under the protective apron.

The difference in values may be explained by the skill and experience of the practitioner, the quality of equipment and protection materials. The present study should be completed taking into account the different parameters affecting dose.

Table 2 gives dose values received by the doctor D5 for a period of 28 days in a department of angiography. The data represent annual values of Hp (10) and Hp (0.07) received by the left and right shoulders and ankles, thyroid, left wrist, left forehead and the chest below lead apron. The results show that values are between 1.3 and 5.8 mSv.

Previous work shown [3] that the median value received by the lens is 6.0 mSv for cardiologists indicating an urgent need to educate radiation protection professionals to reduce the risk of cataracts due to the use of RX. In the present work, the annual dose corresponding to the lens (on front) is 3.8 mSv.

An occupational dosimetry survey carried out in 12 European radiology centers proposed an annual effective dose Hp (0.07) on the protective apron of 14 mSv for optimizing the exposure of the most exposed [4] operators.

TABLE 2. ANNUAL DOSE VALUES (mSv) RECEIVED BY DOCTOR D5 (EMBOLISATION)

Left Ankle		Right Ankle		Left Shoulder		Right Shoulder	
Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)
2.81	3.70	2.00	2.07	2.16	2.14	1.65	1.65
Chest (int)		Thyoid (ext)		Left Wrist		On Front	
Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)
1.45	1.33	4.67	5.84	3.09	2.99	3.85	3.83

TABLE 3. SUMMARIZES THE MEASUREMENT RESULTS IN A DENTAL RADIOLOGY CENTER DURING TWO WEEKS (mSv)

Chest (int)		Chest (ext)	
Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)
5.82	5.98	8.40	7.75

TABLE 4. WEEKLY DOSE VALUES FOR DOCTOR D6 (μSv)

First week

Chest (int)		Chest (ext)		Wrist	
Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)
208.09	203.69	238.86	227.19	304.83	313.62

Second week

Chest (int)		Chest (ext)	
Hp(10)	Hp(0.07)	Hp(10)	Hp(0.07)
28.21	34.07	71.55	62.53

After discussion with doctor D6, it was concluded that the first week corresponds to the case where the doctor takes the maximum of radio and the second week to the minimum (Tables 3 and 4). Based on the average annual values at the chest, Hp (10) is equal to 5.82 mSv below the protective apron, and 8.40 mSv above the apron.

TABLE 5. ANNUAL DOSE VALUES RECEIVED BY DOCTOR D7 (CATHETERISM DEPARTMENT)

Chest (INT)		Chest (ext)		Gonad (int)		Gonad (ext)		Left Shoulder		Right Shoulder	
Hp(10)	Hp(0,07)	Hp(10)	Hp(0,07)	Hp(10)	Hp(0,07)	Hp(10)	Hp(0,07)	Hp(10)	Hp(0,07)	Hp(10)	Hp(0,07)
1,94	2,05	6,12	6,88	1,29	1,41	9,54	12,01	17,13	18,82	1,24	1,28

Dose values received by doctor D7 (Table 5) indicate a good attenuation of RX by the protective aprons used in this department. Furthermore, doses on the shoulders and the front are quite important. Particular care must be conducted to optimize procedures and use all types of radiation protection materials is necessary.

4. CONCLUSIONS

The risk of exposure of the extremities, the eye, thyroid and shoulders of occupational staff should be seriously considered during radiological procedures. Dose measurements represent a quantification of the risk.

In this work, the measurements were performed in four departments using ionizing radiation. The results show a dispersion of dose values received by practitioners. The annual whole body dose is usually within the international regulation values except for doctor D4 who received a dose a high dose nearing the recommended dose limit. Dispersion of the dose values is essentially due to doctors' experience and quality of equipment and means of protection used.

This study should be followed up taking into account these differences in the objective of optimizing the dose to workers using the RX in Moroccan hospitals.

REFERENCES

[1] CLEMENT C.H. (Ed.) ICPR Statement on tissue reactions, ICRP publication 118 part1. Ann ICRP, 41 (1-3) (2012) 9-14.

[2] BARDIA FARMAN (PSRPM)-Marc BOURRELLY (PCR), Radiologie interventionnell Dosimétrie extrémités des médecins, Assistance Publique Hôpitaux de Marseille Unité de Radioprotection et de Physique Médicale Journée de rencontre ASN du 17 juin 2011 Professionnels de radiologie interventionnelle et de blocs opératoires.

[3] VANO. E, KLEIMAN. NJ, DURAN. A, REHANI M.M., ECHEVERRI. D, CABRERA M, Radiation cataract risk in interventional cardiology personnel, Radiat. Res. 174(4) (2010) 490-5, doi: 10.1667/RR2207.1.

[4] C. FOTI, R. PADOVANI1, A. TRIANNI, Staff dosimetry in interventional cardiology: survey on methods and level of exposure, Oxford Journals Mathematics & Physical Sciences, Radiation Protection Dosimetry, 129 (1-3) (2008) 100-103.

COMPARISON OF DEEP DOSE EQUIVALENT (Hp(10)) BETWEEN PRIVATE AND PUBLIC HOSPITALS IN ZIMBABWE

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Abstract

This paper reports on the dose assessment for occupationally exposed workers in Zimbabwe. The research was carried out over a period of two and a half years for different medical facilities in the country. A comparison was made between doses for occupationally exposed workers in the private hospitals and public hospitals. This was done to check how each sector was responding to improvement in the worker safety concerning personal monitoring. The country has one centre for personnel monitoring that uses the Harshaw 6600 PLUS Thermolunimiscent dosimetry monitoring system which was used in acquisition of doses. A total number of 50 TLD dosimeters were used and were procured from Thermofisher (Europe).

1. INTRODUCTION

The Dosimetry Section of the Radiation Protection Authority of Zimbabwe has the responsibility of providing personal dosimetry monitoring to individuals occupationally exposed to ionizing radiation. Its laboratory is capable of carrying out dose assessment for radiation workers in the country. Radiation workers are exposed to beta, gamma, X-rays and neutron radiation over a wide range of energies. In this research the concentration was mainly on X-rays from the diagnostic X-ray departments. The TLD dosimeter contains Lithiun fluoride doped with Magnesium (LiFl:Mg). The dosimeters were read using the thermolunimescent reader with the results evaluated using a dose equivalent conversion algorithm. This paper describes comparison of deep dose equivalent, Hp(10) between selected Private and Public hospitals in Zimbabwe. Dose assessment is essential to help ensure that radiation workers' exposure is properly controlled and dose limits are not exceeded so that harmful effects of radiation are reduced.

2. JUSTIFICATION OF THE MONITORING AND ASSESSMENT PROGRAMME

The primary justification for individual monitoring is to help achieve and demonstrate adequate protection, including implementation of optimization of protection. Monitoring should also be done in order to;

(a) Provide information about conditions in the workplace and means of establishing whether these are under satisfactory control and whether operational changes have improved or worsened the radiological working conditions;
(b) demonstrate compliance with limits and the application of the principle of 'As Low As Reasonably Achievable', economic and societal factors being taken into account as part of legislative or regulatory systems;
(c) inform workers of their radiation exposure, where doses are low this may be for reassurance;

(d) Evaluation and development of operating procedures from review of collected monitoring data for individuals and groups (such data may be used to identify both good and bad features of operating procedures and design characteristics, and thereby contribute to the development of safer radiation working practices)

3. METHODOLOGY

Radiation workers have a 30 day wearing period for TLD doismeters. The TLD dosimeters are submitted to the Radiation Protection Authority of Zimbabwe for dose assessment on a monthly basis. For this research dosimeters submitted between January 2012 and May 2014 were used. Individual doses were recorded each month from two Public hospitals and two private hospitals. An average of the dose from each facility was recorded. A Harshaw Thermoluminescent Dosimetry system was used in acquisition of data. The system uses a lithium-fluoride chip doped with magnesium (TLD 100). The TLD card measures deep dose equivalent, Hp(10) and equivalent dose to the skin, Hp(0.07). For the purpose of this research, only deep dose equivalent, Hp(10) was considered. For each hospital, the values of Hp(10) were comparable and an average value was used for data analysis in this research.

Throughout the period in question, inspections were conducted on the facilities to check compliance with the requirements given in the Radiation Protection Act and associated regulations and to promote implementation of a safety culture. The requirements include;

i. Provision of occupational exposure monitoring to all workers who are occupationally exposed to radiation.
ii. Ensuring that occupationally exposed workers do not exceed dose limits specified in the regulations and international standards. Occupationally exposed workers should not exceed an effective dose of 20mSv per year averaged over five consecutive years without exceeding 50mSv in any single year.
iii. To practice a safety culture and optimize protection such that workers do not only aim to be at dose limits but to keep doses ‘As Low As Reasonably Achievable’

4. RESULTS

This research focused on the dose assessment for occupationally exposed workers in the medical field. The data was averaged from readings obtained per each facility on a monthly basis and recorded in Table 1. A graph was also plotted so as to show the trend of the deep dose equivalent from January 2012 to May 2014. From the trends public hospitals started with doses that are higher than private hospitals. These doses decreased gradually and 2014 shows comparable doses in both sectors.

These changes could be attributed to a number of reasons which might include:

(a) The regulatory body in the country effectively training radiation users on radiation safety principles
(b) Radiation users adopting a safety culture, good working practices (e.g. the adequacy of supervision and training) and engineering standards.
(c) The governments playing a big role in availing resources to public hospitals in terms of equipment and training of personnel.

TABLE 1.COMPARISON OF MONITORING BETWEEN PUBLIC AND PRIVATE HOSPITALS IN ZIMBABWE

month	Private 1(mSv)	Private 2(mSv)	Public 1(mSv)	Public 2(mSv)
Jan I	0.289	0.383	0.834	0.785
Feb I	0.323	0.411	0.649	0.958
Mar I	0.205	0.365	0.901	0.951
Apr I	0.236	0.53	0.722	0.758
May I	0.408	0.283	0.567	0.961
June I	0.5	0.37	0.569	0.836
July I	0.265	0.26	0.602	0.816
Aug I	0.29	0.284	0.745	0.749
Sept I	0.538	0.418	0.689	0.699
Oct I	0.259	0.399	0.497	0.886
Nov I	0.244	0.418	0.45	0.785
Dec I	0.306	0.253	0.735	0.819
Jan II	0.283	0.266	0.621	0.968
Feb II	0.409	0.344	0.5	0.659
Mar II	0.392	0.297	0.653	0.756
Apr II	0.508	0.333	0.403	0.659
May II	0.276	0.274	0.485	0.824
June II	0.288	0.243	0.367	0.524
July II	0.398	0.4	0.295	0.684
Aug II	0.367	0.299	0.325	0.493
Sept II	0.561	0.315	0.324	0.584
Oct II	0.422	0.201	0.401	0.491
Nov II	0.286	0.426	0.378	0.685
Dec II	0.428	0.52	0.354	0.521
Jan III	0.249	0.254	0.263	0.496
Feb III	0.355	0.391	0.305	0.485
Mar III	0.273	0.3	0.318	0.5
Apr III	0.292	0.47	0.439	0.588
May III	0.411	0.243	0.471	0.452

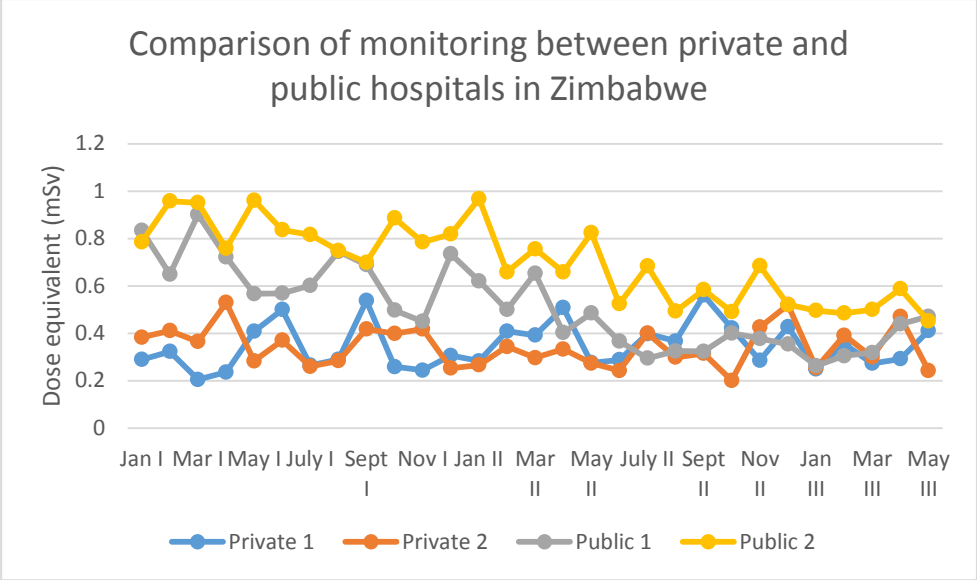

FIG. 1. Comparison of monitoring between public and private hospitals in Zimbabwe

Key: I-Year 2012 II- Year 2013 II- Year 2014

5. CONCLUSION

The comparison has provided a robust way to assess the dosimetry laboratory capabilities and also ways to improve personal monitoring in the country. This has shown the necessity to improve safety culture at workplace as a way to reduce doses to radiation workers in the medical sector. The research has given more avenues to keep the doses As Low As Reasonably Achievable (ALARA), thereby minimizing unnecessary doses to workers.

REFERENCES

[1] EUROPEAN COMMISSION, Radiation Protection No 160, Technical Recommendations for monitoring individuals occupationally exposed to external radiation, Directorate-General for Energy and Transport, Luxembourg (2009).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Occupational radiation protection, Safety Guide No. RS-G-1.1 (1999).
[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection and safety of radiation sources: International Basic Safety Standards, GSR Part 3 (Interim) (2011).
[4] RADIATION PROTECTION AUTHORITY OF ZIMBABWE, Radiation Protection Act Chapter [15:15] of 2004, (2004).
[5] RADIATION PROTECTION AUTHORITY OF ZIMBABWE, Radiation Protection (Safety and Security of Radiation Sources) Regulations, 2011(Statutory Instrument 62 of 2011)

RISK DUE TO THE EXPOSURE OF SKIN ON HANDS OF WORKERS HANDLING RADIOPHARMACEUTICALS LABELLED WITH ^{18}F: PRELIMINARY RESULTS FROM THE CZECH REPUBLIC

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Abstract

This preliminary study summarizes the results of radiation burden related to the skin of hands of workers at three different nuclear medicine departments in the Czech Republic. Attention was focused especially on the assessment of maximum local exposure received by hands during the preparation and administration of ^{18}F-labelled radiopharmaceuticals.

1. INTRODUCTION

Positron emission tomography (PET) is a very important imaging method, where the number of installations in the world and as well as in the Czech Republic (CR) is continually increasing. At present, there are 10 nuclear medicine departments operating PET or PET/CT systems. Important role in these examinations play radiopharmaceuticals labeled with ^{18}F where in the CR is mostly used the type ^{18}F-FDG (in about 97 % of all cases in PET) [1]. The nuclear medicine personnel coming into contact with these radiopharmaceuticals may receive a significant exposure of the skin of hands where even relevant dose limits may be exceeded.

This has recently been confirmed by the results under the ORAMED (Optimization of Radiation protection for Medical staff) [2], which was carried out from 2008-2011. Seven European countries participated in the ORAMED project, however, the CR was not among them. The ORAMED survey documented that there were a number of cases where nuclear medicine staff's annual exposure to skin of the hands was higher than the relevant equivalent dose limit of 500 mSv.

Since there was in CR no systematic study similar to that of ORAMED, an attempt in terms of the monitoring of workers for their specific extremities exposure was undertaken. The relevant monitoring has been going on since November 2012 at selected departments. The preliminary results from these departments also confirmed that in some cases the annual

reference level as well as equivalent dose limits related to the skin of hand may be exceeded [3-5]. Consequently, it would be appropriate to take appropriate steps in order to reduce this excessive exposure of workers.

The paper presents some results and comparisons of the maximum local exposure of the skin of hands of workers at three selected nuclear medicine departments the technological facilities of which are not identical and differ to certain extent as to equipment used for handling radiopharmaceuticals. Obviously, the local conditions may be one of many other factors affecting the exposure.

2. MATERIAL AND METHOD

The survey summarizes the results of the monitoring of some workers at three PET/CT departments who were coming into a contact during the preparation and administration of positron radiopharmaceuticals. The workers were wearing gloves with thermoluminiscent dosimeters (TLDs), the positions of which were similar to those used in the ORAMED study. However, in our case, it was used an additional position L (the second phalanx of the finger on the palm side of the hand) which would be the most appropriate position for the finger dosimeter. In each of the A-L locations, a pair of TLDs (namely MCP-7 and MCP-Ns), was placed. The sensitivities of these types of TLDs differed as to their response to photons and positrons. Uncertainties of the measurement were estimated to be about 20 %-30%.

Altogether 27 workers from three departments which differed as to the number of examined patients (Table 1). It has to be noted that in the case of the department 2 and 3 not all workers were monitored. At the Departments 1 and 2 the workers were monitored in three cycles where each cycle consisted of 25 dispensing and 25 application operations. At the Department 3, each worker has been up to now (June 2014) monitored twice but the monitoring is still going in order to complete the third cycle by the end of October 2014.

TABLE 1. THE NUMBER OF PERSONS MONITORED AT INDIVIDUAL DEPARTMENTS

Department	Number of examine patients per year	Number of monitored workers (number of all workers)	
		Preparation	Administration
Department 1	7 090	10 (10)	5 (6)
Department 2	3 000	2 (3)	4 (7)
Department 3	1 800	4 (4)	2 (7)

Individual departments use technological equipment which is not of the same provenience. All departments were equipped with semi-automatic dispensing stations for acquiring of radiopharmaceuticals. The specific parameters of equipment as well as the individual approach applied has obviously some impact on the exposed of workers.

With respect to the preparation of radiopharmaceuticals, some differences could be noted:

(a) Department 1 – The preparation of radiopharmaceuticals was performed by means of a semi-automatic dispensing feeder from the company 1. The administration was carried out either using a cannula or directly to the vena.
(b) Department 2 – The semi-automatic feeder from the company 2 where the workers used also rubber gloves, which were part of semi-automatic feeder. The administration was carried out entirely through the cannula.
(c) Department 3 – The preparation of radiopharmaceuticals was performed by means of a semi-automatic dispensing feeder from the company 2. The administration of radiopharmaceuticals was carried out using a semi-automatic equipment where the

worker was not in direct contact with the source. The worker only supervised the whole process making sure that the activity of radiopharmaceuticals applied was correct.

A fair detailed protocol was maintained reflecting all essential information about each workers monitored. Such a protocol included the following data: workplace, ID, duration of experience working in the field, present assignment, main activities (preparation, administration), position of finger dosimeter provided by an authorized personal dosimetry service, dominant hand, the date and time of monitoring, the time during which a worker was in contact with a radiopharmaceutical, radioactive contamination (if any), the handled activity of a radiopharmaceutical. Each cycle of the monitoring took place during the time needed for 25 preparations or administrations where the total handled activity was usually in the range 7-12 GBq.

At the end of every cycle of the monitoring, the response of TLDs were read and the personal dose equivalent $H_p(0,07)$ normalized to the activity of 1 GBq evaluated. There was made an attempt to assess the maximum annual local exposure of the skin of hands of every worker where the estimation was based on the assumption about the total activity a worker could handle during the year. The resulting value was then compared against the investigation level (3/10 of the annual limit for H_{skin}) and the annual dose limit in terms of the annual equivalent dose to the skin (500 mSv/y).

3. RESULTS

The local exposure of hands of workers showed quite significant differences. Table 2 presents the results of the measurements at individual departments in terms of the values of the equivalent dose to the skin approximated by the relevant personal dose equivalent measured using TLDs.

TABLE 2. AVERAGE VALUES OF THE MAXIMUM LOCAL EQUIVALENT DOSE TO THE SKIN OF HANDS

Department	**Preparation [mSv/GBq]**	**Administration [mSv/GBq]**
Department 1	0,51 ± 0,37	0,80 ± 0,23*
Department 2	0,23 ± 0,02	0,37 ± 0,08
Department 3	0,27 ± 0,23	0,13 ± 0,15

*The values when not considering two workers whose approach in handling radiopharmaceuticals was somewhat different from the approach of their colleagues

The most frequent occurrence of the local skin exposure was found in the position F (fingertip). In 16 workers (preparation) and 11 workers (administration) this maximum was found on the right or left hands in 10 cases attributed to the preparation and 14 cases related to the administration. Table 3 summarizes the number of cases corresponding to the situation when the investigation level or dose limit for the skin of hands could be surpassed.

TABLE 3. ESTIMATION OF THE NUMBER OF CASES AT INDIVIDUAL DEPARTMENTS WHERE THE INVESTIGATION LEVEL AND DOSE LIMIT MAY BE EXCEEDED

Range of exposure	Department 1		Department 2		Department 3	
	Preparation	Administration	Preparation	Administration	Preparation	Administration
< 150 mSv	4	0	2	4	2	2
150 –500mSv	6	3	0	0	2	0
> 500 mSv	0	0	0	0	0	0

4. DISCUSSION

At the Department 1, a group of medical doctors administrating radiopharmaceuticals was found to have received systematically higher exposure to the skin of hands than professionals who were engaged in the preparation of radiopharmaceuticals. This is most probably due to the fact that the workers preparing radiopharmaceuticals were using the semi-automatic feeders and thus, they were relatively shorter time in contact with the source. The doctors administered radiopharmaceuticals using cannulas and also directly to vena where the exposure was visibly higher than in the case of using cannulas. When administrating directly to the vena, the workers worked under the conditions characterized by high dose rate which explains why the maximum local exposure at the Department 1 was higher than the exposure of workers at the Departments 2 and 3.

As to the doctors from the Department 1, the highest exposure to the skin of hands was identified in the right hand which is usually used to grasp the syringe during the administration process (4 out of 5 doctors use the right hand as a dominant hand). There was also the case of a worker preparing radiopharmaceuticals the exposure of which was almost three time higher than the exposure of his colleagues.

In some other cases, the maximum exposure of the skin of the right hand of three workers at one department was significantly higher that the maximum dose of other co-workers. One of these workers received the dose to the skin as high as about five times the average dose of other workers engaged in preparing radiopharmaceuticals. Although all these workers follow more or less the same procedures, their exposures considerably differed in some cases which can be explained by their different skills and experience. If the exposure of workers at Departments 1 and 3 is compared (they used the same type of dispensing feeder), it has been noticed that their maximum local exposure of the skin of hands was lower at the Department 2. It looks like the technology applied at Depaertment 2 was more efficient.

The results produced by ORAMED survey [2] documented that the exposure of hands of workers engaged in the preparation and administration of radiopharmaceuticals were in the range of 0.10-4.43 mSv/GBq and 0,14 – 4,11 mSv/GBq, respectively. At the nuclear medicine departments in the CR, where this study was carried out, the maximum local exposures were comparable or slightly less than those reported by the ORAMED. The reason was presumably due to the larger number of workplaces monitored under the ORAMED study which were not equipped by semi-automatic dispensing equipment.

5. CONCLUSIONS

From the overview of the results of the exposure of the skin of hands of nuclear medicine workers handling radiopharmaceuticals at three selected departments in the CR, it

can be concluded that there are quite large discrepancies in local doses at various workplaces. The difference can mainly be attributed to such factors as the individual approach of the workers performing specific operations which can be carried out using slightly different technique, the technological equipment and conditions at each department, and at last but not least, on the use of shielding and protective tools. The use of semi-automatic dispensing feeders proved to result in reduction of the exposure of workers. In addition, the exposure of the skin of hands reflects the level of general radiation protection situation at the workplace where compliance with relevant regulatory requirements depends largely on the skills and experience of workers and their professional competence which has to be time to time refreshed and updated in specialized training courses.

ACKNOWLEDGMENTS

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REFERENCES

[1] BĚLOHLÁVEK, O., The needs of PET/CT: Number of applications of individual radiopharmaceuticals for PET in 2013, [onlin e]. [2014-06-29], Accessible from: http://www.csnm.cz/article/show/aboutDomain/ pet-ct-need.

[2] EUROPEAN RADIATION DOSIMETRY E. V., Optimization of radiation protection of medical staff, Extremity dosimetry in nuclear medicine, EURODAS Report 2012 - 2. Braunschweig, (2012) 133 – 177, ISBN 978-3-943701-01-2.

[3] HUDZIETZOVÁ, J., FÜLÖP, M., SABOL, J., Assessment of local exposure of the skin of hands of workers during the manipulation with radiopharmaceuticals-^{18}F. In: Nuclear Medicine. Prague: Czech Medical Society of J. E. Purkyně, (2013) 7, ISSN 1805-1146 (in Czech).

[4] HUDZIETZOVÁ, J., FÜLÖP, M., SABOL, J., Assessment of the radiation burden of extremities of workers handling radiopharmaceuticals labelled with ^{18}F, In: AWHP 2013. Prague: Manus, (2013) 164-174. ISBN 978-80-86571-18-8 (in Czech).

[5] HUDZIETZOVÁ, J., FÜLÖP, M., SABOL, J., The impact of the exposure of extremities to the total effective dose due to the operations with ^{18}F-Labelled radiopharmaceuticals at some PET departments, In XXXV, Day of Radiation Protection – Proceedings. Prague: Czech Technology – CTU Publishing House, (2013) 109, ISBN 978-80-01-05356-0 (in Czech).

A NEW METHOD FOR EXPERIMENTAL EVALUATION OF OPERATIONS WHEN HANDLING RADIOPHARMACEUTICALS

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Abstract

A new method for experimental evaluation of operations when handling radiopharmaceuticals based on an unfolding of integral distribution of skin doses measured in 10 different locations on the hands was introduced. A matrix for skin dose maps for unfolding of integral skin dose map has been measured by using of tissue equivalent hand phantoms simulating various irradiation geometries that occur during handling radiopharmaceuticals. The proposed method was used for evaluation of operations when handling FDG in three nuclear medicine departments and for evaluation of physician operations with the high exposures of his hands during the FDG administrations to patients. Preliminary results have pointed out selectivity of the method that depends on careful selection of physician operations with FDG for construction of matrix of skin dose maps.

1. INTRODUCTION

When preparing radiopharmaceuticals administered to patients, the hands of staff of nuclear medicine departments are irradiated in non-uniform fields resulting in non-homogeneous local exposure of the skin (in terms of the personal dose equivalent $H_p(0,07)$). This local exposure may reach quite high levels and is often a limiting factor in the work with radiopharmaceuticals. The parts on the hand with maximum exposure are usually investigated by mapping of the local skin dose on hands during handling radiopharmaceuticals. The locations on hands for dose mapping measurements are shown in Fig. 1.

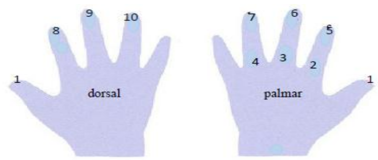

FIG.1 Positions of skin dose monitors

The skin dose map contains integral information about operations that contribute to the skin dose when handling the radiopharmaceutical. By unfolding of the integral dose map one can obtain the assumption about particular contributions of these operations to the hand irradiation. Based on the analysis of the hand operations the following measures for radiation protection optimization can be proposed:

(a) New dosimeters for hand dose monitoring,
(b) Optimization of ring dosimeter position on hand,
(c) Improvements of hand shielding by special gloves or thimbles on finger tips,
(d) Development of manipulators for handling radiopharmaceuticals,
(e) Optimization of procedures during handling radiopharmaceuticals,
(f) Investigation of radiation safety problems, such as overexposures, accidents, spills and other deviations from approved radiation safety practice and implementation of corrective actions as necessary.

2. METHOD

A way to analysis the operations when handling radiopharmaceuticals consists of several steps:

i. Recognition of single operations that could significantly contribute to the value of maximum skin dose on hands $H_{pmax}(0,07)$ when handling radiopharmaceuticals.
ii. By hand phantoms simulating the single j operations (that has been specified in previous item) to perform measurement of skin gamma and beta (electron or positron) dose maps $G_{i,j}$ in all i localities (shown in Fig. 1) .
iii. Construction of dose map matrix from $G_{i,j}$, i=1,2,…,10, j=1,2,3,… normalized to equal handled activity of 1 GBq.
iv. Measurement of integral skin gamma and beta (electron or positron) dose map D_i in i=1,2,..,10 localities on hands of nuclear medicine staff during routine handling radiopharmaceuticals. The integral dose map D_i represents numerous operations by hands carried out during routine handling radiopharmaceuticals.
v. To perform an unfolding of the measured integral hand dose distribution D_i by the multiple linear regression method as a superposition of G dose maps – elements of the dose map matrix that represents specific operations (in the dose map matrix signed under numbers k, l, m,…) when handling the radiopharmaceutical:

$$D_i = a + b_{i,1}.G_{i,k} + b_{i,2}.G_{i,l} + b_{i,3}.G_{i,m} \ldots \quad i=1,2,3,\ldots,10 \qquad (1)$$

$$f(x) = (_{0} + \sum_{(j=1)}^{(j=n)} [(_{j} \cdot G_{1}(i,j)]$$

vi. Importance of the operations k, l, m,… inductive of hand irradiation can be assumed by quantification of contributions of products $b_{i,1}.G_{i,j}$ for the value i that belongs to the number of place (see Fig.1), where the maximum value in dose map D_i was found.

3. RESULTS AND DISCUSSION

Examples of operation with FDG during syringe administration to patient by a physician and simulations of irradiation geometry by hand phantoms are shown in Fig. 2 and Fig. 3.

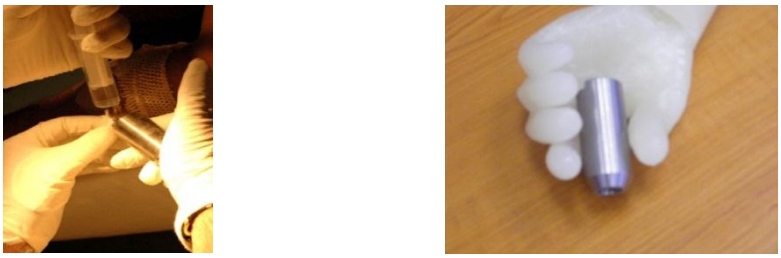

FIG. 2. Operation by shielded syringe with FDG among fingers and thumb in physician hand (left) and in hand phantoms (right)

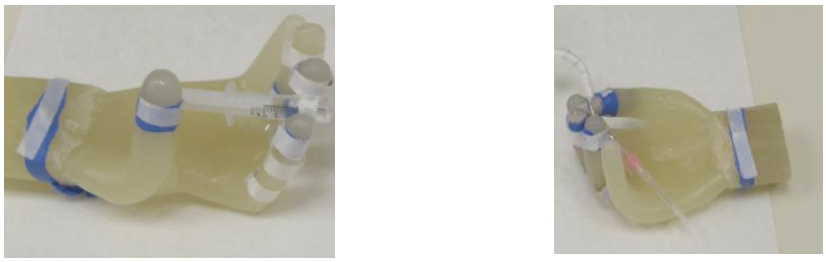

FIG. 3. Operation by unshielded syringe with FDG between index and middle fingers of hand phantom (left) and by infusion tube hold by fingers and thumb of hand phantom (right)

The matrix $G_{i,j}$ of 17 dose maps measured in 10 localities on the hand phantoms simulating 17 operations frequently used when handling FDG at nuclear medicine department is shown in Fig. 4.

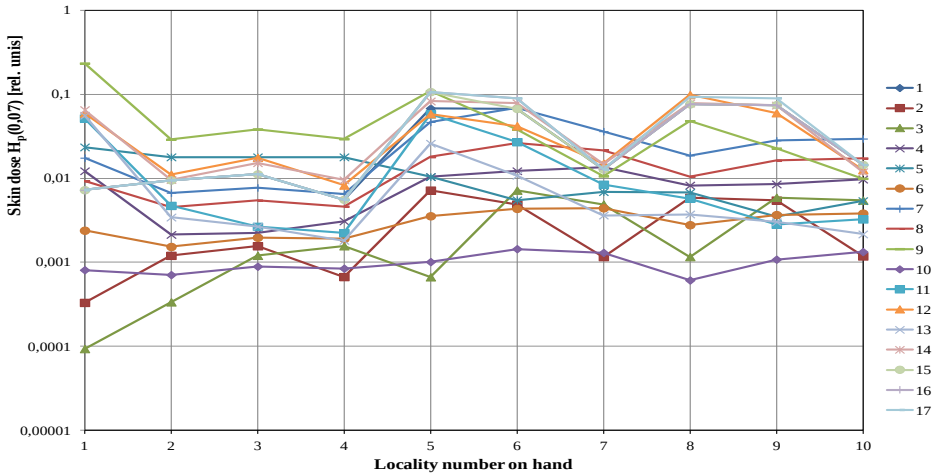

FIG. 4. Matrix $G_{i,j}$ of 17 skin dose maps measured in 10 localities (see in Fig.1) on hand phantoms simulating 17 operations when handling FDG radiopharmaceuticals

3.1. Evaluation of operations when FDG application at three nuclear medicine departments

The proposed evaluation method of hand operations during FDG administration to patients was used at three hospitals with nuclear medicine departments (hereinafter NMD1, NMD2 and NMD3) in Slovakia. In each department two physicians were 4 or 5 times measured. One measurement period lasted 5 working days (about 50 applications of FDG to patients).

Integral skin dose maps measured in 10 localities (see Fig.1) on hands of the physicians of NMD1, NMD2 and NMD3 are in Fig.5. For better illustration, the skin dose maps in Fig.5 are normalized to maximum skin doses that were measured in locality No.5 (heel of index finger) in all NMDs.

The integral skin dose maps measured in nuclear medicine departments NMD1, NMD2 and NMD3 were approximated by superposition of up to four operations. The superposition for all possible combinations (more than 3000) of 17 operations (represented by dose maps) from which was composed dose maps matrix were calculated by a multiple linear regression.

The integral skin dose distribution (average of all 28 measurements from NMD1, NMD2 and NMD3) approximated by superposition of dependences No.2, No.11 and No.17 (see Fig. 4) is shown in Fig. 6.

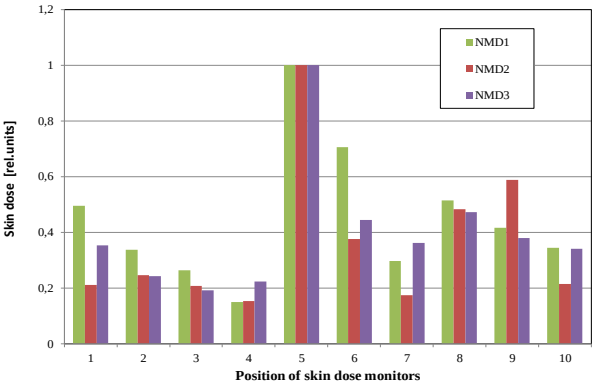

FIG. 5. Integral skin dose maps measured dose in 10 localities (see Fig. 1) on hands of the physicians of NMD1, NMD2 and NMD3

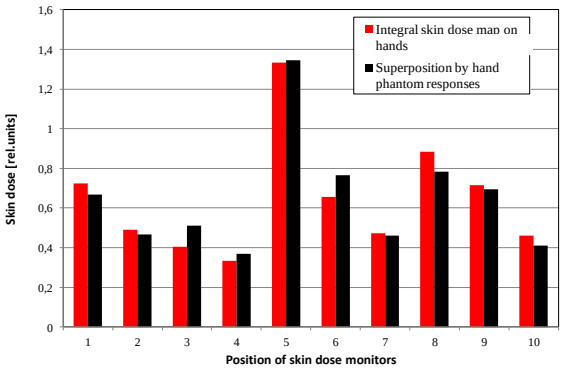

FIG. 6. Approximation of the integral skin distribution (average across 28 measurements at all three NMDs) by superposition

Based on the evaluation of coefficients of the superposition equations it was found that FDG with unshielded syringe was administered mainly to patients of NMD1 and to lower number of patients of NMD2 and NMD3.

3.2. Evaluation of hand operations of worker with a high hand exposure

Properties of the proposed method have been verified for the evaluation of physician operations with the high exposure of hands during the FDG administrations to patients. The distributions of integral skin doses measured in 10 locations on right and left hands of the physician are shown in Fig. 7.

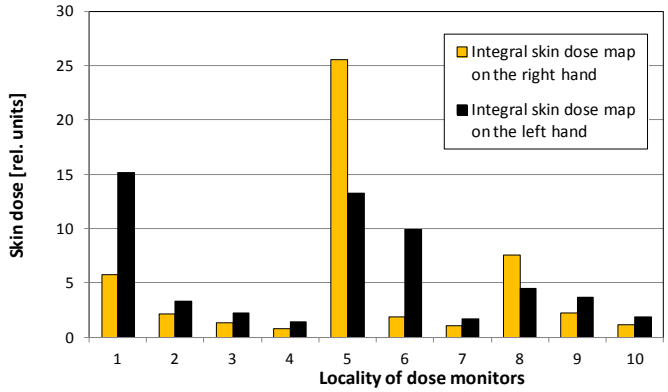

FIG. 7. Distributions of integral skin doses of the physician hands

Based on the analyses of superposition equations of integral skin dose distributions, the right hand of the physician was mainly irradiated by FDG applications with shielded syringe holding among fingers and thumb (see Fig. 2). Irradiation of the left hand was mainly due to holding of infusion tube with FDG (see Fig. 3).

4. CONCLUSIONS

A method for experimental evaluation of operations when handling radiopharmaceuticals was introduced. The method is based on unfolding of integral distribution of skin doses measured in 10 different locations on the hand. Results of the unfolding procedure have been expressed by a superposition of single skin dose maps measured on hand phantoms simulating various operations when handling radiopharmaceuticals.

The proposed method can be used for searching of operations with dominant contribution to local irradiation of skin on hands during manipulation with radiopharmaceuticals. Applying this approach, it was possible to design more targeted measures of radiation protection of nuclear medicine personnel.

ACKNOWLEDGMENTS

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AN ANALYSIS ON MODIFYING TECHNIQUES TO REDUCE STAFF DOSES IN PET/CT

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Abstract

Adult and pediatric patients for PET/CT in Saudi Arabia increases every year by almost two-fold. Doses to staff have been determined using different methodologies and efforts to reduce staff doses have been recommended. The study showed that the estimated staff dose per procedure could be about 9.6 μSv. Staff doses during preparation, injection and patient positioning could vary by a factor of 3 to 7. This study aims to determine how techniques can be modified to reduce staff doses from Fluorine-FDG (^{18}F-FDG). Staff doses are measured using a survey meter and electronic dosimeter. Rotation of staff, use of 3-way stopcock extension line during injection, use of syringe carrying box with lead shielding for Fluorine and use of a transfer station for ^{18}F-FDG from preparation bench to injection area when giving instructions are recommended for staff dose reduction.

1. INTRODUCTION

The use of PET/CT for cancer therapy and management of adults and pediatric patients in Saudi Arabia increases every year by almost two-fold. PET/CT imaging is also utilized in radiotherapy treatment plans [1]. Due to the high energy of Fluorine there is a concern on staff whole body and wrist doses [2, 3]. There are centers by which doses to staff are being monitored [4]. This study is aimed to investigate the factors that cause high staff doses and determine the techniques to be modified for dose reduction.

2. METHODS

Two PET/CT centers were included in the study one center has an automatic injector and the other with manual injector. A total of 6 staff were included in the study. Staff doses were detemined at diffferent phases of the work namely: preparation of radiopharmaceutical including opening and/or loading of vial, injection and patient positioning. A survey meter set in the integrated mode and an Aloka electronic dosimeter were used. Both measuring instruments were calibrated at King Faisal Specialist Hospital and Research Centre SSDL. The measured doses measured were considered as the whole body doses and the total cumulative dose for each staff per day was calculated. The average whole body dose was determined for each staff. The variation of measured whole body doses was evaluated and the causes for high values were investigated. A comparison of the doses of staff who use automatic injector with those who use manual injector was made. The correlation of age with staff whole body dose during positioning was investigated using the Pearson correlation equation. Techniques that can be modified were recommended.

3. RESULTS

The use of automatic injector decreased staff dose by a factor of 4. With the use of automatic injector, loading the vial with the radiopharmaceutical increased staff doses to about 3 μSv. Staff doses during preparation, injection and patient positioning varied largely with a coeffcient of variation of 0.80. Staff doses during preparation decreased by a factor of 4 when an automatic injector was used.The highest whole body dose was 9.75 μSv (Table 1) and this was due to the use of lead container for ^{99m}Tc instead of lead container for ^{18}F – FDG). The dose of the technologist is higher than the other technologists who used the correct lead container by a factor of 5. Training provided a dose reduction by a factor of 10. One staff had difficulty of removing the canulla and the obtained dose during injection was 5.6 μSv. The study of K. Dalianis et al showed that the staff (nurse) dose during injection is in the range of 2.7 to 4 μSv per procedure and the dose to the technologist during positioning is in the range of 3.5 to 5 μSv per procedure.

There is a tendency for the staff dose during positioning to increase when dealing with older patients (Fig. 1). The Pearson correlation (r) between patient age and staff dose during positioning was 0.42.

4. DISCUSSION

Increase in staff dose was due to the use of wrong lead container, having the syringe in the injection trolley while giving instruction for manual injection and removal of canulla. These techniques should be modified. Removal of canulla should be part of staff training. Staff dose during patient positioning can be about 40% of the total cumulated dose per procedure. The time of contact with patients during positioning should be limited. During the injection phase, one staff had the syringe with the wrong lead container beside him and this contributed to additional dose. A transfer station can be used while instruction is being given to patients. After the injection one staff had to fix the patient chair and therefore additional unnecessary dose was obtained.

Practice of patient positioning should be part of staff training. One cause of increased staff dose is the longer time needed to give instructions to older patients during positioning. This is shown with the Pearson correlation value which is expected to increase with a large sample size. It is recommended to categorize patients in terms of difficulty to handle such easy, difficult and very difficult with the difficult ones being assigned to more experienced technologists. Another strategy for dose reduction is the possibility of assigning a different staff to perform the injection. The staff should have the skill for injection and thus the time of contact with the patient will be reduced.

5. CONCLUSION

Staff doses vary from person to person due to techniques. Techniques to be evaluated and modified to reduce staff whole body doses are: rotation of staff and distribution of tasks; use of 3-way stopcock extension line during injection; use of syringe carrying box with lead shielding for ^{18}F, and use of a transfer station for ^{18}F-FDG from preparation bench to injection area when giving instructions. Routine check of canulla, patient injection chair and the availability of appropriate lead containers should be done before the procedure starts. Standard protocol for staff protection such giving instructions, loading vials in automatic injector and removing canulla should be implemented. The use of automatic injector is recommended.

TABLE 1. STAFF DOSES AT DIFFERENT PHASES OF WORK AND THE CUMULATED DOSES FOR WHOLE BODY SCAN DURING PET/CT PROCEDURE USING ^{18}F-FDG

Code	Injector used	Average dose per procedure (μSv)			
		Preparation	Injection	Positioning	Cumulated
A	Auto	0.71	0.59	1.32	2.55
B	Auto	0.06	0.56	0.59	1.29
C	Manual	0.26	4.01	2.43	6.7
D	Manual	1.6	2	3.2	6.8
E	Manual	0.6	1.95	0.7	3.25
F	Manual	2.75	4.15	2.85	9.75

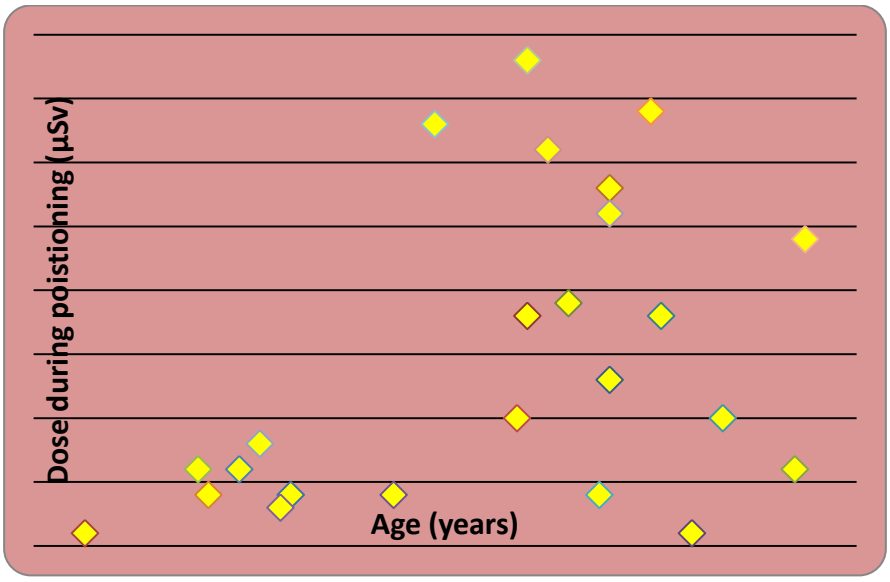

FIG. 1. Scatter graph of the positioning dose against the patient age for whole body PET/CT procedure using Fluorine-FDG. Pearson coreelation (r) is 0.42

REFERENCES

[1] CARSON, K.J., YOUNG, V.A., COSGROVE, V.P., JARRITT, P.H., HOUNSELL, A.R. Personnel radiation dose considerations in the use of an Integrated PET-CT Scanner for radiotherapy treatment planning, BJR, Vol 82 (2009) 946-49.

[2] PANT, G.S., SENTHAMIZHCHELVAN, S. Radiation exposure to staff in a PET/CT facility. JNM, 21(4) (2006) 100-103.

[3] DALIANIS, K., MALAMITSI, J., GOGOU, L., PAGOU, M., EFHIMIADOU, R., ANDREOU, J., LOUIZI, A., GEORGOIU, E., Dosimetric evaluation of the staff working in a PE, T/CT department. Dosimetry Nuclear Instruments & Methods in Physics Research, A569 (2006) 548-550.

[4] AL-HAJ, A. . LOBRIGUITO, A., ARAFAH A., PARKER, R., Deriving Staff and Public Doses in a PET/CT Facility from Measured Radiation Levels using Thermoluminescent Dosimetry, published in the book of abstracts of the European Conference on Individual Monitoring of Ionizing Radiation, Athens, Greece, held on 8-12 March 2010. Published in Radiation Protection Dosimetry, 452 (2010) 1-5.

IN SITU PHANTOM MEASUREMENTS AT THE TIME OF INTERVENTIONAL CARDIOLOGY PRACTICE AS A TOOL FOR DEVELOPMENT AND VALIDATION OF DOUBLE DOSIMETRY ALGORITHMS

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Abstract

Irradiation of occupationally exposed personnel occurs under permanently changing conditions – due to movement of a person and variation of a radiation field. Therefore, measurements in interventional cardiology (IC) operation room, which are usually conducted under static conditions (voltage, projection etc), might not be informative in terms of interpretation of actual exposure of a personnel. In order to approximate real life situation, longitudal *in situ* phantom measurements were conducted in course of regular IC practice with the aim of validation of double dosimetry method, used for assessment of effective dose of interventional cardiologists. RANDO-Alderson type anthropomorphic heterogeneous phantom of was exposed in IC operation room in two measurement sessions when normal surgical practice was going on. Conventionally true effective dose values derived from phantom measurements were compared with the estimates of *E* obtained by application of NRCRM double dosimetry algorithm, which uses readouts of various pairs of individual dosimeters as an input. Algorithm data agreed with the results of phantom measurements, observed deviation was within –18% to +34%. Cross-application of the algorithm weighting factors in both measurement sessions show robustmess of the algorithm.

1. INTRODUCTION

Interventional cardiology (IC) is one of the fast growing areas of interventional radiology in Ukraine and worldwide, which, however, is associated with high exposure of medical staff. Use of protective gear (lead aprons, collars, local screens) makes individual dosimetry and assessment of effective dose quite difficult. The most common approach is based on double dosimetry when two personal dosimeters – under and over apron – are used and the effective dose *E* is estimated as:

$$\tilde{E} = \alpha \cdot H^{u} + \beta \cdot H^{o} \qquad (1)$$

Where, H^{u} is the dose value read from the dosimeter worn under the apron and H^{o} over the apron, α and β·are respective weighting factors. However, until today there is no consensus either about a suitable algorithm (the way of α and β assessment) or about location of the dosimeters on the cardiologist's body [1].

The purpose of this study was to validate experimentally a new generic approach to algorithm developing, which takes into account group-specific conditions of exposure during

IC procedures. The idea of the algorithm development, as described elsewhere [2], is that knowing empirical frequency distributions of typical characteristics of cardiac procedures (like a source beam energy spectrum, a field of vision (FOV), C-arm angulations and x-ray scattering/shielding objects) one can use the results of computer simulations performed for given conditions to estimate effective dose, and thus adapt the double dosimetry algorithm for some general classes of situations: for certain types of cardiac procedures, for certain types of equipment, for certain types of protective clothes etc.

2. METHODS

To get information about the most typical conditions of a cardiology surgeon's irradiation and to account for variability of parameters of X-ray procedures, we conducted measurements in a real operating room of the endovascular surgery and angiography department of the A.A.Shalimov National Institute of Surgery and Transplantology (NIST), National Academy of Medical Sciences of Ukraine, where an angiographic system Toshiba Infinix CS (model INFX-8000F) is used.

RANDO-Alderson type phantom [3] loaded with 270 LiF dosimetry pellets was used to simulate the surgeon. It was placed on a movable table with bottles of water replacing phantom's legs. The phantom was covered with a regular 0.35 mm Pb protective apron and a collar commonly used in the hospital. To measure doses in all plausible positions for personal dosimeters, 22 Harshaw TLDs were placed under and over the apron (Fig. 1) at three levels on frontal and rear surfaces of the phantom.

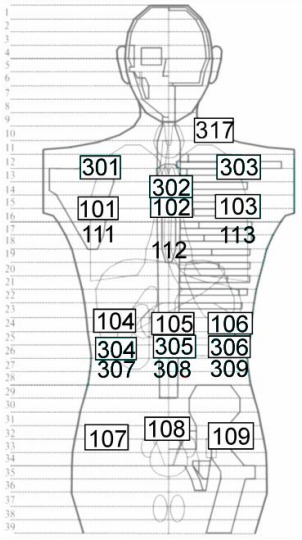

FIG.1. Illustration of positions of the dosimeters on the phantom. Dosimeters with numbers 3## – placement over protective clothes, 1## – placement under protective clothes. Dosimeters with framed number were placed in front of the phantom, others from the backside.

Two in situ measurement sessions were undertaken within operating room for 14 and 12 working days, respectively, in course of regular surgical practice. The phantom was placed symmetrically to the position of a surgeon at opposite side of the table to be exposed under similar irradiation conditions not compomising surgeon's actions. When non-IC operations (abdomenal or other) were performed, the phantom was temporarily removed from the operating room to a shielded location.

Irradiation of an IC surgeon was also studied by Monte Carlo simulations using a detailed anthropomorphic phantom ADAM [4] modified by adding a wrap-around lead apron and a collar of 0.35 mm Pb equivalent similarly to [5]. All respective Monte Carlo

simulations were conducted for this mirror position of a surgeon, representing actual position of a phantom. Partial doses to organs and to individual dosimeters received by a surgeon dressed in lead apron were calculated for each particular irradiation situation (a unique combination of energy – angulation – FOV). Total organ doses were estimated from partial dose values as the weighted sum, with weights proportional to frequencies of experimental irradiation situation and their contribution to dose (measured patient DAP was used as a dose index for this purpose).

3. RESULTS

All dosimetrically significant parameters of the X-ray machine collected during two sets of measurements were retrieved and summarized. Since the X-ray machine records parameters only in radiography mode and just briefly reflects current parameters on a monitor in course of radioscopic visualization, it was not possible to rely on the machine log for reconstruction of irradiation situation history. In order to overcome this limitation, video registration of the monitor display had been performed. Later, the video files were analyzed using home-made image processing program and semi-automated recognition of graphical data concerning operation parameters of the X-ray system and then all retrieved data was stored in the database. This was by far the most challenging part of this study – in total 2010 and 2146 single irradiation situations were recognized and recorded for two measurement sessions, respectively. As a result, for further computer simulations we selected 12 values of photon energy (from 30 to 110 keV), 9 C-arm angulations, 4 different FOV (15x15, 20x20, 25x25, 30x30 cm^2). The energy spectrum of photons corresponded to a typical tube with the maximum tube voltage in a range of 80-120 kV, the anode angle of 12° and 0% ripple and was modified by 3.5mm aluminum and 0.3mm copper filters.

At the stage of algorithm development [2], six optimal pairs of dosimeters were selected based on the following criteria:

1. A pair of dosimeters should have the lowest sum of deviations between the estimated and the simulated effective dose;
2. Contributions of components of equation (1) related to dosimeters under and over an apron should be comparable;
3. A pair of dosimeters should have comfortable location for a physician.

In situ phantom measurements used for validation of the algorithm had added one more selection criterion, speficically: satisfactory conformity of the estimation with the experimental results obtained during *in situ* irradiation of a RANDO-Alderson phantom.

Six pairs of dosimeters which meet criteria 1-3 were chosen at the stage of algorithm development. The pair ranked #1 includes a dosimeter at hip (side pocket) level from the side of the source as under-apron location and a dosimeter at chest level in the middle of the chest as over-apron location. It could be seen that more commonly located chest level over-apron dosimeter has an under-apron pair at the hip (side pocket) level and shows quite acceptable agreement as well. It can be seen that traditional dosimeter combination “chest level from the side of the source (under) - collar level (over)” does not sit the top position by goodness of *E* approximation.

4. DISCUSSIONS

It may be seen from Table 1 that for presumably optimal dosimeter pairs the deviation of Ẽ from conventional true value of E, obtained by in situ phantom measurements does not exceed 127%. After exclusion of the pair #6, for remaining five pairs, algorithm overestimates

E by not more than 34% and underestimates by not more than 18%. These results are well within the required limits for individual dosimetry with Hp(10) dosimeters [6], which, in fact, does not consider compliance between operational quantity Hp(10) and protection quantity E. For all other considered dosimeters pairs observed deviation was far larger - within -99% to +520%. The worst effective dose estimation for some of dosimeter pairs can be explained by inappropriate dosimeters positions for effective dose estimation (e.g. both dosimeters for double dosimetry algorithm are placed on the belt level for the side opposite to the source).

5. CONCLUSIONS

In general, experimental in situ validation of NRCRM double dosimetry algorithm under real life exposure conditions has demonstrated excellent accuracy and robustness of dose estimates provided by the algorithm. In fact, straightforward application of popular algorithms [1] gives results, which significantly overestimate actual *E* values – see Table 1. This difference is caused by the approach used for development of the NRCRM algorithm. Testing of several algorithm options (various dosimeter pairs and respective weighting factors) had proven that discrepancy between of estimated and measured (conventional true) effective dose values did not exceed 34%.

TABLE 1. COMPARISON OF THE NRCRM ALGORITHM VALIDATION DATA WITH SIMILAR DOUBLE DOSIMETRY ALGORITHMS BASED ON THE SAME PHANTOM EXPERIMENTAL DATA. DOSIMETER CODING SEE AT FIG. 1

#	Authors	α_{opt}	β_{opt}	$\tilde{E}_{algorithm}/E_{exp}$ session1	$\tilde{E}_{algorithm}/E_{exp}$ session2
	This work (for specific dosimeter pair)				
1	107/302	0.24	0.016	1.34	0.90
2	101/303	0.36	0.020	1.17	0.88
3	107/301	0.27	0.008	1.21	0.82
4	101/302	0.39	0.014	1.12	0.88
5	107/303	0.16	0.029	1.22	0.97
6	101/317	0.21	0.056	2.27	1.11
	Other algorithms				
1	von Boetticher et al., 2008 [7]	0.65	0.017	2.0	1.3
2	NCRP 122 (NCRP, 1995) [8]	0.5	0.025	2.1	4.4
3	Niklason et al., 1994 [9]	0.98	0.02	2.8	1.4
4	Franken & Huyskens, 2002 [10]	1	~0.033	3.4	2.3
5	McEwan, 2000 [11]	0.71	0.05	3.7	2.5
6	Swiss ordinance, 1999 [12]	1	0.05	4.2	2.8
7	Sherbini & DeCicco, 2002 [13]	1	0.07	5.1	3.4
8	Wambersie & Delhove, 1993 [14]	1	0.1	6.5	4.4
9	Clerinx et al., 2008 [15]	1.64	0.075	6.6	4.4

REFERENCES

[1] JARVINEN, H., et al., Overview of the double dosimetry procedures for the determination of the effective dose to the interventional radiology staff, Radiat. Prot. Dosim. 129 (1-3) (2008) 333.

[2] CHUMAK, V. et al., Optimization of the double dosimetry algorithm for interventional cardiologists, Rad. Phys. Chem. (2014) DOI: 10.1016/j.radphyschem.2014.01.014.

[3] ALDERSON, S. W. et al., An instrumented phantom system for analog computation of treatment plans, Am. J. Roentg., 87 (1962) 185.

[4] KRAMER, R., et al., The Calculation of Dose from External Photon Exposures Using Reference Human Phantoms and Monte Carlo Methods. Part 1: The Male (Adam) and Female (Eva) Adult mathematical Phantoms. GSF-Bericht S-885, GSF, Neuherberg (1999).
[5] SCHULTZ, F.W. J. ZOETELIEF, Dosemeter readings and effective dose to the cardiologist with protective clothing in a simulated interventional procedure. Radiat. Prot. Dosim. 129 1-3, (2008) 311.
[6] INTERNATIONAL COMISSION ON RADIOLOGOCAL PROTECTION, General Principles for the Radiation Protection of Workers, Publication 75, Pergamon Press, Oxford and New York (1997).
[7] von Boetticher et al., Effective dose estimation in diagnostic radiology with two dosimeters: impact of the 2007 recommendations of the ICRP. Health Phys. **95** 3 (2008) 337.
[8] NATIONAL COUNCIL ON RADIATION PROTECTION AND MEASUREMENT, Report122, Use of personal monitors to estimate effective dose equivalent to workers for external exposure to low-LET radiation, Bethesda, MD: NCRP, 64 (1995).
[9] NIKLASON, L. T., The estimation of occupational effective dose in diagnostic radiology with two dosimeters, Health Phys., 67 (1994) 611.
[10] FRANKEN, Y, HUYSKENS, CJ., Guidance on the use of protective lead aprons in medical radiology: protection efficiency and correction factors for personal dosimetry, Proc.of the 6th European ALARA Network Workshop, Madrid, Spain. (2002). http://ean.cepn.asso.fr/program6.html
[11]MCEWAN, A.C., Assessment of occupational exposure in New Zealand from personal monitoring records. Radiat. Prot. Australas. 17(2) (2000) 60.
[12] Swiss Ordinance for Personal Dosimetry Ed. by the Federal Chancellery. – Bern (1999).
[13] SHERBINI, S., DECICCO, J., Estimation of the effective dose when protective aprons are used in medical procedures: a theoretical evaluation of several methods. Health Phys. 83 (6) (2002) 861.
[14] WAMBERSIE, A., DELHOVE, J., Radiation protection in diagnostic radiology, a debated practice: how to wear the individual dosemeters? J. Belge Radiol. 76 (6) (1993) 382.
[15] CLERINX, P., et al., Double-dosimetry algorithm for workers in interventional radiology, Radiat. Prot. Dosimetry, 129 **(**1-3) (2008) 321.

RADIATION PROTECTION OF WORKERS THROUGH THE ANALYSIS OF THE MANAGEMENT AND QUANTIFICATION OF RADIOACTIVE WASTE GENERATED IN A NUCLEAR MEDICINE SERVICE

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Abstract

Depending on how the radioactive waste is managed, it can become an additional source of exposure to the worker. In order to minimize this exposure, the amount of radioactive waste generated by a nuclear medicine center was analyzed, quantified, qualified and compared with the doses obtained by personnel dosimetry in the Nuclear Medicine Service of InRad HCFMUSP. The largest quantity of radioactive waste produced was of ^{99m}Tc. Approximately 90% of the solid waste was compactable (gauze, gloves and other) and the remaining 10% were of non-compactable waste, such as needles and others. Over the 2 years, there has been a significant variation in the amount of waste, which is directly connected with the number of procedures performed. The medium dose value observed was of 0.6 mSv per month for all the workers (dosimeters positioned in thorax) and 1.6 mSv in wrist dosimeters. It was observed that months with greater amount of waste coincided with months of higher doses. However, this increase was not significant and was not proportional due to optimization in handling the waste.

1. INTRODUCTION

There is a major concern with the protection of the worker especially when the radioactive material arrives in a nuclear medicine service since it usually has high activity. It is also important to have similar concern when treating the radioactive waste or it can become an additional source of exposure to the worker. The management of radioactive waste is a set of administrative and technical activities involved in segregation, processing, packaging, transportation, storage, control and disposal of the waste. The main objective of this management is to protect human health and the environment, both now and in the future from any deleterious effects caused by radioactive materials considered as having no more use [1].

As defined by IAEA [2], the management of radioactive waste must be subject to standards of safety. The Nuclear Medicine Service of InRad HCFMUSP follows strict standards in order to guarantee the best working conditions and the safety of the exposed worker. When analyzing the significance of the assimilation of radiopharmaceuticals in diagnostic and therapy procedures, the proper management and implementation of technical standards and radiation protection and safety in the nuclear medicine service must be a priority. However, as the number of exams and therapy using radiopharmaceuticals increases, there was not so far (in the studied clinic) an analysis of how the increase or decrease of radioactive waste relates to the doses of these workers.

The generation of radioactive waste should be as far as possible minimized [3], which can be achieved through the adoption of appropriate operating procedures, so as to avoid contamination, to reduce exposure and in order to reduce the volume of waste to be managed. In order to minimize the exposure of the worker the amount of radioactive waste generated by a nuclear medicine center was analyzed, quantified and qualified with the additional benefit of optimizing the management of that waste.

2. METHODS

A retrospective analysis was made of the records of storage and disposal of radioactive waste from January 2010 to December 2012 in the Nuclear Medicine Service of InRad HCFMUSP. Using the data of personal dosimeters used by 12 workers (including radiopharmacists, nurses and physicists) of the above mentioned nuclear medicine service in the same interval a comparison between the equivalent doses and the quantities of radioactive waste produced was made.

3. RESULTS

The greatest quantity of radioactive waste produced was of 99mTc-99m, representing 75%. In terms of mass, this corresponds to approximately 781 kg. Other nuclides used in the clinic of this study generated the following quantities of waste: 186 kg of ^{51}Cr, 52 kg of ^{131}I and 20 kg of ^{67}Ga.

Approximately 90% of the solid waste was compactable (gauze, gloves and other) and the remaining 10% were of non-compactable waste, such as needles and others. Liquid radioactive waste was not included in this study due to alterations in the handling of that waste in the given interval.

The doses obtained from the personal dosimeters resulted in a range from background values to 1.4 mSv in a month (the latter was observed only once in one nurse, being the second and more common value of 0.9 mSv observed in more than one worker). Wrist dosimeters showed a range from background values to 8.5 mSv (this dose was observed only once in a radiophamarcist, being the second and more common value of 2 mSv observed in more than one worker).

The medium dose value observed was of 0.6 mSv per month for all the workers (dosimeters positioned in thorax) and 1.6 mSv in wrist dosimeters. However, it's important to emphasize that some workers are more exposed than others and that a slight increase in the doses was observed in months with more procedures conducted, and consequently generated more radioactive waste.

4. DISCUSSION AND CONCLUSIONS

The majority of procedures performed for diagnostic purposes in the Nuclear Medicine Service of InRad HCFMUSP uses radiopharmaceuticals labeled with ^{99m}Tc. Thus, it is expected that larger quantity of the ^{99m}Tc waste is generated.

There was an increase throughout the 2 years, in the number of exams performed in the service, which is directly connected with the quantity of waste generated. Hence, an increase in the doses received was expected. It was observed that months with greater amount of waste coincided with months of higher doses. However, this increase was not significant and was not proportional due to optimization in handling the waste. All the doses received were within acceptable levels and well bellow the limits stipulated by CNEN [4].

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Management of waste from the use of radioactive material in medicine, industry, agriculture, research and education, Safety Standards Series No. WS-G-2.7, IAEA, Vienna (2005).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Disposal of Radioactive Waste, Safety Standards Series No. SSR-5, IAEA, Vienna (2011).
[3] COMISSÃO NACIONAL DE ENERGIA NUCLEAR, Gerência de rejeitos radioativos de baixo e médio níveis de radiação, CNEN NN-8.01, Brasil (2014).
[4] COMISSÃO NACIONAL DE ENERGIA NUCLEAR, Diretrizes básicas de proteção radiológica, CNEN NN-3.01, Brasil (2014).

OCCUPATIONAL RADIATION PROTECTION ISSUES ENCOUNTERED IN THE WEST OF IRELAND

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Abstract

The Department of Medical Physics and Bioengineering provides a regional diagnostic physics imaging support service to hospitals and public health facilities utilising ionizing radiation located in the West / Northwest of Ireland. Radiation protection of staff members comes under the remit of this department. A series of incidents involving deficiencies in the shielding of boundaries and protective screens of new and existing general X-ray facilities have occurred in the recent past. These include non-continuity of lead between adjoining sections of a fixed protective screen, inadequate protection provided by the operator's lead glass viewing window and the complete absence of lead lining of a section of wall adjacent to a wall stand bucky in a general X-ray room. The findings of the group highlight the importance of the verification of the efficacy of shielding barriers and the ongoing vigilance in this area to ensure that the protection of staff is not compromised and that occupational dose is maintained as low as reasonably achievable.

1. INTRODUCTION

The Department of Medical Physics and Bioengineering (Dept. of MPBE) consists of three sections, Radiotherapy, Clinical Engineering and Diagnostic Imaging. The Diagnostic Imaging (DI) section provides a regional support service for the hospitals of the West / North West Hospitals Group. The DI section also provides support to the public dental clinics utilising ionizing radiation in the West of Ireland. Radiation protection of staff, patients and public comes under the remit of this group with advice provided in the main by the Radiation Protection Adviser (RPA). There are approximately two hundred X-ray tubes across all modalities in clinical use in this region. The assessment of the shielding provided by all barriers for new facilities is undertaken as part of the commissioning process for all X-ray based equipment. In addition, assessments of existing facilities and replacement protective screens have been carried out as and when required. It is imperative that these assessments are performed to verify that the shielding provided by the barriers is acceptable and meets the required specifications. This ensures that the applicable public or staff dose constraints of 0.3mSv or 1mSv [1, 2] respectively are met. A selection of issues encountered in the recent past shall be presented in this paper.

2. METHODOLOGY

The determination of the lead equivalence of boundaries was carried out using an X-ray tube in line with the methodology as outlined by Sutton, et. al. [3]. A Barracuda device (RTI Electronics, Sweden) with attached R100B solid-state detector was used to measure the air kerma. Technical application notes [4] from RTI indicate that the R100B can be used for low dose measurements in lieu of an ion chamber due to its flat energy response and air-kerma-rate independence. The availability of a high sensitivity mode suitable for low dose

applications made the R100B suitable for the detection of low levels of air kerma post barrier. The dose sensitivity of the R100B detector was 0.1nGy [4].

A mobile X-ray unit was used to carry out the transmission measurements. The focus to detector distance (FFD) was set to 1 metre. Exposure parameters of 125kVp and 50mAs were used. Initial measurements of air kerma were carried out without the barrier in place. The measurements were then repeated at multiple locations with the X-ray tube and R100b detector positioned on either side of the barrier under investigation whilst maintaining the 1 metre FDD.

A transmission factor B through a barrier of thickness x was determined using Eq. 1, for all barriers under investigation. K(x) was the measured air kerma after the barrier, K(0) was the measured air kerma without the barrier in place.

$$B(x) = \frac{K(x)}{K(0)} \tag{1}$$

By use of Eq. 2, a reconfiguration of the equations developed by Archer et al. [5] as part of the empirical model to describe the attenuation of X-rays through a given material, the lead equivalent thickness of the barrier in mm was determined at a particular X-ray energy.

$$x = \frac{1}{\alpha\gamma}\ln\left[\frac{B^{-\gamma} + \left(\frac{\beta}{\alpha}\right)}{1 + \left(\frac{\beta}{\alpha}\right)}\right] \tag{2}$$

The appropriate α, β and γ coefficients for the relevant kVp as outlined in Table I. [3] were used to complete the calculations.

TABLE I. COEFFICIENTS FOR GENERATING PRIMARY TRANSMISSION CURVES FOR LEAD @ 125 kVp

Material	kVp	α	β	γ
Lead	125	2.219	7.923	0.539

2.1. The barriers assessed included the following

2.1.1. Staff protective screen

Staff protective screen for a new general X-ray room. It was fabricated in sections by a local company specialising in furniture making, see Fig 1. The lead glass was acquired from a specialist manufacturer for installation by the screen fabricator.

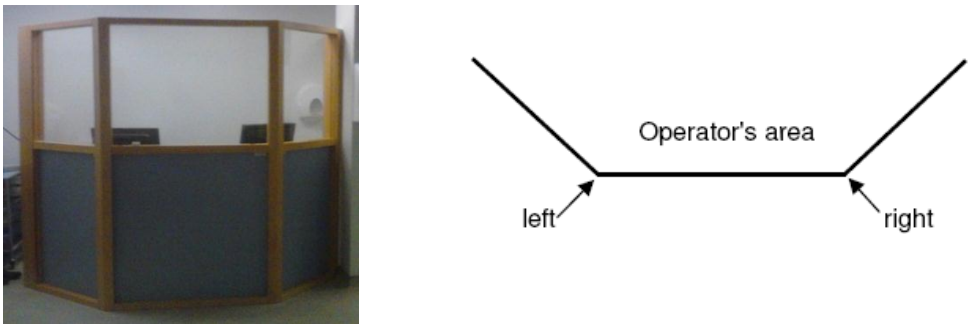

FIG. 1. Picture of Operator's Screen with associated plan view schematic

Transmission measurements were carried out at numerous locations on the glass and solid sections of the screen. The arrows in Fig 1 indicates areas of potential shielding deficiencies at the corner joints. Visual assessment of these locations (Fig. 2.) from the top of the screen revealed no evidence of lead overlap. Extensive transmission measurements were carried out in these areas.

FIG. 2. Plan view of corner joints of protective screen

2.1.2 Replacement Protective Screen

A replacement staff protective screen for an existing general X-ray room to allow for a reconfiguration of the operator's console area.

2.1.3 X-ray Room Boundary Wall

A boundary wall between a high use general X-ray room and a storeroom. This was carried out due to the change of use of the storeroom to an occasionally occupied staff area.

3. RESULTS

3.1. Staff Protective Screen

The lead equivalence (L.E.) of the glass sections were confirmed to be acceptable and as per specification. Subsequent measurement confirmed the absence of lead at the corner joint locations confirming initial visual observations. A selection of the initial results for the solid portions of the screen and subsequent data post remediation work are outlined in Table 2.

TABLE 2. DETERMINED LEAD EQUIVALENCE OF SOLID PORTION OF SCREEN PRE AND POST REMEDIAL WORK

Screen Measurement location	Initial L.E. in mm @ 125kVp	Post L.E in mm @ 125kVp
Front	2.10	2.10
Left corner	0.14	2.20
Right corner	0.15	2.15

3.2. Replacement Protective Screen

The glass portion of the screen was labelled as 0.5mm lead equivalent at 150kVp. Table 3: outlines the measured L.E. of the screen before and after the replacement of the glass section.

TABLE 3. DETERMINED LEAD EQUIVALENCE OF SCREEN PRE AND POST REMEDIAL WORK

Screen Measurement location	Initial L.E. in mm @ 125kVp	Post L.E in mm @ 125kVp
Front Glass	0.57	2.37
Front Shield	2.23	2.23

3.3. X-ray Room Boundary Wall

A selection of the results of the L.E. determinations before and after remediation work are outlined in Table 4.

TABLE 4. DETERMINED LEAD EQUIVALENCE OF BOUNDARY WALL PRE AND POST REMEDIAL WORK

Wall Measurement location	Initial L.E. in mm @ 125kVp	Post L.E in mm @ 125kVp
Left of wall stand bucky	1.25	3.50
Right of wall stand bucky	2.05	4.30
Unshielded section	0.20	2.42

4. DISCUSSIONS

The deficiencies in the new staff protective screen at the corner sections were due to the inexperience and lack of appreciation by the manufacturer of the importance of adequate overlapping of lead at screen junctions. The remedial work carried out involved the fixation of a suitably wide section of Code 5 (2.24mm thick) lead [6] along the complete height of the exterior of the joints. Subsequent measurement confirmed that this was carried out satisfactorily.

The issue with the replacement glass having lower than expected lead equivalence was due to a lack of consultation with the RPA for this facility. It occurred at a remote satellite hospital where the Radiographer ordered a complete replacement screen to allow for a reconfiguration of the operator's console area.

There was no specification provided with the order and a presumption was made by the supplier that a glass rating of 0.5mm L.E. @ 150kVp was adequate. However, this was not an acceptable level of protection for the operator based on the shielding calculations carried out by the RPA. Subsequent remediation work involved the replacement of the glass for 2.1mm L.E @ 110kVp. This rating was confirmed through measurement and deemed adequate for this facility.

The deficiencies observed for the wall were due to two factors. The first factor was the use of barium plaster which had been applied in a non-homogenous manner across the inside surface of the wall. Secondly, a reconfiguration of a section of the wall between the X-ray room and the store room adjacent to the vertical wall stand bucky occurred without consultation with the RPA and without due consideration for the radiation protection implications of such an action. It transpired that the wall composition in that section consisted

of standard gypsum plasterboard. It was recommended that a layer of Code 5 lead be applied to the entire internal surface of the wall to ensure that sufficient protection was in place at every location. Subsequent measurements carried out confirmed that the wall was adequately shielded.

5. CONCLUSIONS

The issues outlined highlight the importance of the use of reputable suppliers and installers of lead protective equipment. If new suppliers or installers are to be engaged, it is imperative that they are apprised of the importance of continuous and contiguous layers of shielding with adequate overlapping where lead layers meet [2].

The requirement to consult with the RPA in advance of the acquisition of protective equipment has been highlighted. It is a legal requirement to inform and consult with the RPA in advance of any changes to a facility where ionizing radiation is in use [1, 2]. This is of particular importance where the RPA is providing a service to a dispersed geographical area. The specification of any shielding barrier should be provided in writing to the supplier or manufacturer of the same, to ensure that the protection is so designed as to meet the requirement and to maintain staff dose as low as reasonably achievable (ALARA), and within the prescribed dose constraints [2].

It is imperative that all protective equipment and barriers are assessed before ionizing X-ray equipment is put into clinical use. It is essential that any deficiencies are highlighted and discussed with the supplier / installer as soon as possible in order to achieve a satisfactory outcome in an expeditious manner. It is also worth reviewing existing facilities for areas of possible deficiencies. This may be particularly relevant if one has recently taken over the radiation protection responsibilities for a facility, and are uncertain of the fabric and associated protection offered by shields and barriers where ionizing radiation is being utilised. All barriers and boundaries should be labelled with their L.E for future reference [2]. A comprehensive report outlining the specified and verified shielding for each and every location where X-rays are utilised should be generated and the record is maintained.

REFERENCES

[1] STATIONERY OFFICE, 2000. Radiological Protection Act. 1991 (Ionizing Radiation) Order, 2000, S.I No. 125 of 2000. Dublin: Stationery Office.

[2] RADIOLOGICAL PROTECTION INSTITUTE OF IRELAND, The design of medical facilities where ionizing radiation is used, Code of Practice, RPIL, (2009).

[3] BRITISH INSTITUTE OF RADIOLOGY, Radiation shielding for diagnostic Radiology, 2nd Edition, Report of a BIR Working Party. D.G. Sutton, C.J. Martin, J.R. Williams, D.J. Peet. London, BIR, (2012).

[4] RTI ELECTRONICS, Application Note 1-AN-52020-6, Measurements of scattered radiation, (2007). http://www.rti.se/api/downloads?id=1-AN-520206_Measurements_of_scattered_radiation_v2007.12A.pdf. Accessed June 2014

[5] ARCHER, B.R., THORNY, J.I., BUSHONG, S.C., Diagnostic X-ray shielding design based on an empirical model of photon attenuation, Health Phys. 44 **(**1983) 507-17.

[6] BSI, BS EN 12588:2006, Lead and lead alloys. Rolled lead sheet for building purposes, BSI British Standards (2006).

OCCUPATIONAL RADIATION PROTECTION IN THE PRODUCTION OF POSITRON EMITTERS AND IN THE SYNTHESIS OF PET RADIOPHARMACEUTICALS

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Abstract

The positron emission tomography (PET) has become an important type of diagnosis through images in Nuclear Medicine (NM) providing more precise information and per se a better tool for an exact and useful diagnostic. Based on the previous statement, it can be said that a new era in Nuclear Medicine started with the development and vertiginous diffusion of medical cyclotrons. These are intended to produce positron emitters on a large scale such as: ^{18}F, ^{13}N and ^{11}C, which are the starting point to synthesize radiopharmaceuticals that are used in PET. The cyclotron and the radiopharmacy facilities, where the chemical synthesis of the PET radiopharmaceuticals is carried out pose a significant radiation risk on account of the high dose rate and the high energy of the radiation as well as the amount of unsealed radioactive material that is involved. For these reasons, safety systems, interlocks, warnings and a high degree of automation are required for achieving a safe operation. This paper presents an overview of the safety systems (interlocks, warning systems) that are commonly used to ensure an appropriate exploitation of this technology, the training program for workers and procedures to apply in order to reduce the occupational doses.

1. INTRODUCTION

The Positron Emission Tomography is the clearest example of the evolution of NM and its importance in oncology. This technology is used to study physiological and biochemical processes in tissues and organs, improving the staging and volume delineation of tumors, providing assistance to the physician to decide on changes in the treatment and also an improved patient management.

The positron emission tomography has become a better tool for an exact and useful diagnosis therefore a vertiginous diffusion of medical cyclotrons has taken place worldwide.

A cyclotron is a compact electromagnetic device that can accelerate charged particles to high energies and produce positron emitting radionuclides for use in diagnosis and nuclear medicine investigation. They could be either self-shielded or not.

Facilities for positron emitting isotopes production consist basically of a cyclotron vault in which the targets are placed, a control room where the cyclotron operation is controlled, a technical room, a laboratory with hot cells where the chemical synthesis of the radiopharmaceuticals is carried out and a laboratory for pharmaceutical quality control.

There is a high ionizing radiation risk in facilities with both cyclotron and labs for synthesis of radiopharmaceuticals, as a result of the manipulation of large amounts of unsealed radioactive material and the operation of the cyclotron itself. The need for shielding a high energy beam and protecting people from the external radiation and the contamination with radioactive material make these facilities rather complex.

2. REQUIREMENTS OF RADIOLOGICAL PROTECTION AND REDUCTION OF OCCUPATIONAL DOSE.

The first step to achieve an adequate control of the occupational dose is a good design of the facility due to its high complexity, therefore different interlocks, alarms and security systems have to be considered.

The facility must fulfill certain basic requirements: be safe, functional and comply with regulations on radiation safety and drug manufacturing by ensuring:

(a) The radiation safety and security of unsealed radioactive materials produced at all times.
(b) Minimizing occupational exposure.
(c) Preventing the spread of radioactive contamination.

Once designed and built; the facility must establish an appropriate occupational dose reduction program, which should be based on an effective institutional commitment to safety, a consistent financial support to this program and a clear policy in this regard.
The bases for structuring this program are:

a) The establishment, implementation and improvement of a management system.
b) Promoting a safety culture to stimulate a questioning and learning approach of the workers toward the safety and security of the facility and discourage complacency attitudes.

As a result of the activation process in several parts of the equipment, operators are exposed to this source of radiation frequently as part of the day-to-day operations including the placement of targets, routine repairs and maintenance works, and also the revision of several accessories and systems.

Due to the above mentioned aspects, an effective maintenance program needs to be planned, implemented, regularly reviewed and adjusted. This plan should consider the requirement of competence of the technical staff and the necessity of this staff to respond in case of contingencies during the operation. Specialized technical staff should be kept solely for the like maintenance and emergency response operations; since these are the operations that cause the highest radiation doses. Other aspects to be considered are to perform drills simulating risky operations and review the scope of these operations, and prior to any contingency interventions relevant plans should be précised.

Another factor is the role of the radiation safety staff which must maintain strict monitoring of the facility to ensure that all operations and activities are carried out keeping the dose as low as reasonably achievable.

In order to limit the occupational dose; automation, safety systems, interlocks (both electrical and software-based) and alarms are provided to ensure the protection of the workers at all times, for example they can be triggered/activated to stop the operation of the cyclotron or prevent the personnel from entering areas with high dose rates or high level of activity, etc. Some examples of safety systems, interlocks and alarms are as follow:

a) Safety interlock and access control systems of the cyclotron room and the hot cells.
b) Emergency stop button.
c) Safety delay timer to allow the decay of activation products.
d) Last person interlock.
e) Ventilation system with pressures cascade effect.
f) System for transferring radioactive materials from the cyclotron to the hot cell.
g) Remote real time reading dosimeter

h) Shielding
i) Intercoms
j) Visible and audible warning signals and alarms that are triggered when deviations of normal operating conditions or failure of a safety system or procedure are detected.
k) Automatic synthesis modules.

All these safety systems and alarms should be subjected to a verification process as well as periodic reviews and inspections so that these systems remain capable of meeting their design requirements for which they were installed throughout their lifetime.

Other provisions that should be available at the facility to ensure an appropriate dose reduction program during the repairing and maintenance activities are: special tools, mobile shielding, continued monitoring of the dose to the workers and the use of personal protecting equipment.

Another important topic is the initial and systematic training of the staff. Usually, there is few staff employed; therefore, it is often carried out by cross-training activities; however, a specific training program in radiation protection must be implemented and continuously enhanced. An additional point to consider is the clear definition of the responsibilities assigned to each worker. So a complete training program for individuals associated to the operation of an accelerator facility should include a general radiation safety training, facility-specific safety training (the accelerator facility's safety rules and procedures), task-specific training (the knowledge and skills required to perform specific tasks), a system for recording training results and a mechanism for formally confirming that an individual has been judged qualified to perform the duties that were the objective of the training.

If there are individuals, among the staff, able to perform different tasks and functions, a rotational policy of the work positions may be implemented in order to reduce individual dose. Nevertheless, attention has to be paid to assign clearly the task to be carried out avoiding conflict of interests.

3. CONCLUSIONS

i. The implementation of the technical requirements identified internationally for accelerator facilities, contributes positively and directly in the occupational radiation protection.
ii. Staff training can not be underestimated, considering the risks involved and the complexity of the technology.
iii. The existence of operational and routine maintenance, procedures, contribute significantly to reduce the occupational dose, as well as the implementation of procedures for verifying and testing all safety systems, alarms and interlocks.

REFERENCES

[1] Autorización de Instalaciones de Tomografía por emisión de positrones.ExperienciaAdquirida. C. Álvarez y Mª L. Ramírez. Revista de la sociedad Española de protección radiológica No36. Vol X. (2003).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Cyclotron produced radionuclides: Principles and practice, Technical Reports Series no. 465, IAEA, Viena (2008).
[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Cyclotron produced radionuclides: Guidelines for setting up a facility, Technical Reports Series no. 471, IAEA, Vienna, (2009).
[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Planning a Clinical PET Centre, Human Health Series no.11, IAEA, Vienna (2010).

[5] Foro Iberoamericano, Criterios para el licenciamiento y requisitos de inspección de instalaciones con ciclotrones para producción de radioisótopos utilizados en aplicaciones e investigaciones médicas. (2013).

EVALUATION OF SCATTERED DOSE REDUCTION OF OCCUPATIONAL EXPOSURE IN INTERVENTIONAL RADIOLOGY USING LEAD-FREE PROTECTION SHEET

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Abstract

Scattered radiation in interventional radiology has become a concern due to increasing complexity of the procedures and prolonged fluoroscopic time. Because of the increasing number of interventional procedures, additional protecting measures are desirable. A disposable, sterile and lead free protection drape (RadPad, WIT, USA) has recently been introduced to significantly reduce scattered radiation dose to medical staff by 50-95% during the interventional procedures. The purpose of this study was to evaluate the efficacy of RadPad protection drape in reducing scattered dose to the medical staff during interventional procedures.

1. INTRODUCTION

Interventional radiology has been one of the major growth areas in radiology due to the improvements in imaging performance of X-ray equipment and refinements in catheter design in recent years [1]. It utilizes minimally invasive image-guided method to diagnose and treat diseases which is often a complex and lengthy procedure. The prolonged fluoroscopic time often lead to high occupational doses to the working personnel especially the interventionalist involved in the procedure [2, 3]. The radiation dose received by the workers is mainly come from the scattered radiation from the patient's body and it is associated with the patient dose [4]. Therefore, adequate physical and protective devices can minimize radiation exposure to the radiation workers [5].

Lead or Pb-82 has been used as primary shielding material in X-ray environment for many decades. Recently, a new radiation protection drape (RadPad, Worldwide Innovations & Technologies, Overland Park, Kansas) has been developed using lead-free material, primarily Bi-83 and Sb-51 [6-11] to provide additional protection to the working personnel. The RadPad protection drape is sterile, lightweight, non-vinyl, repositionable and disposable. The drape is individually designed for different interventional procedures and each type of drapes is available in five categories which provide different levels of protection vary from 50 to 95% attenuation at 90 kVp X-ray energy.

2. METHODS

2.1. Physical characterization of RadPad protection drape

2.1.1. Attenuation properties

The study was conducted using an under couch digital radio-fluoro system (Philips Easy Diagnost Eleva, Philips Healthcare, Netherland) and 20 cm PMMA phantom simulating a

typical patient's body thickness. The RadPad protection drape was placed on the phantom. Two calibrated semiconductor detectors (Unfors Xi, Raysafe, Sweden) were used; one was placed between the PMMA phantom and the protection drape, another one was placed above the drape. The phantom was then exposed with a series X-ray energies, ranged from 60 to 125 kVp at fixed 25 mAs and 100 SSD. Two types of RadPad protection drapes were studied; one with 90% attenuation (known as RadPad Orange), another with 75% attenuation (known as RadPad Yellow) at 90 kVp. The RadPad was then replaced with the Pb-equiv. apron and the measurement was repeated. A graph of percentage attenuation versus energies was plotted.

2.1.2. *Evaluation of backscatter radiation properties*

For the study of backscattered radiation, the detector was placed on the 20 cm PMMA phantom. The dose reading was taken without any shielding above the detector. Then, without removing the detector, the RadPad was placed above the detector and the dose reading was recorded. The differences of the reading between with and without the RadPad was calculated.

2.1.3. *Dose profiles before and after attenuation by the RadPad*

Next, in order to study dose distribution before and after the RadPad protection drape, two radiochromic films (Gafchromic XR-QA2, Ashland Inc., USA) were placed on 5 cm PMMA phantom; one before the RadPad, another one after the Radpad. The phantom was then exposed with X-ray of 90 kVp and 200 mAs. The radiochromic films were then scanned with a flatbed scanner (Epson Expression 10000XL, Epson America Inc., Canada) and analyzed using the ImageJ software.

2.2. Study of scattered radiation dose during fluoroscopy-guided procedure

An anthropomorphic phantom (Alderson RANDO Phantom, RSD Inc., USA) was placed on the couch of an angiography system (Axiom Artis dFA, Siemens Healthcare, Germany) to simulate an angiography procedure. A RadPad protection drape (femoral, orange type) was placed on the phantom. The RadPad should be placed outside the primary beam and on the side of where the radiologist should stand. Another anthropomorphic phantom (Atom, CIRS Inc., USA) simulating the radiologist was placed at 0.3 to 0.9 m, at 0.1 m increasing step, away from the RANDO phantom. Three calibrated electronic personal dosimeters (Unfors EDD-30, Raysafe, Sweden) were placed at the brain, thyroid and chest level, respectively of the Atom phantom. X-ray exposure was made using routine fluoroscopy settings (66 kVp, 79 mA, AEC mode) for 5, 10 and 15 min. The scattered radiation dose detected at the brain, thyroid and chest was recorded, and each measurement was repeated twice. The same measurement was then repeated by removing the Radpad on the RANDO (patient) phantom.

3. RESULTS

3.1. Physical characterization of RadPad

3.1.1. Attenuation properties

Fig. 1 shows the percentage of attenuation between RadPad Yellow, RadPad Orange and 0.25 mm Pb-equiv. shield. At 90 kVp, highest attenuation was achieved by the RadPad

Orange (85.8 ± 0.1%), followed by 0.25 mm Pb-equiv. shield (84.8 ± 0.1%), %) and RadPad Yellow (71.6 ± 0.1%). Fig. 2 shows the linear attenuation coefficient, μ calculated for RadPad Yellow, Orange and 0.25 mm Pb-equiv. shield. The Pb-equiv. shield has the highest μ, followed by RadPad Orange and Yellow. There were strong correlations between μ and X-ray energy (kVp).

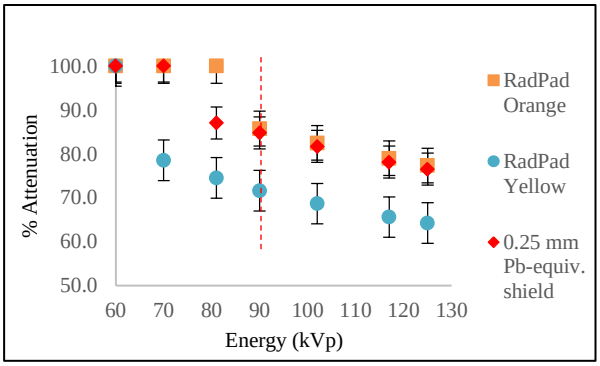


FiG. 1

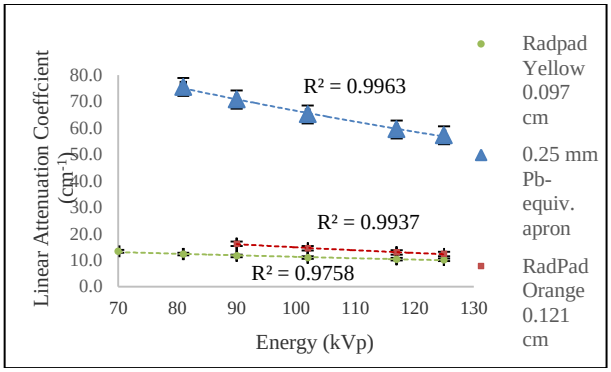


FIG. 2

FIG. 1. Comparison of percentage attenuation versus X-ray energies between RadPad Orange, Yellow and 0.25 mm Pb-equiv. shield.

FIG. 2. Comparison of linear attenuation coeficient, μ versus X-ray energies between RadPad Orange, Yellow and 0.25 mm Pb-equiv. shield.

3.1.2. Evaluation of backscatter radiation properties

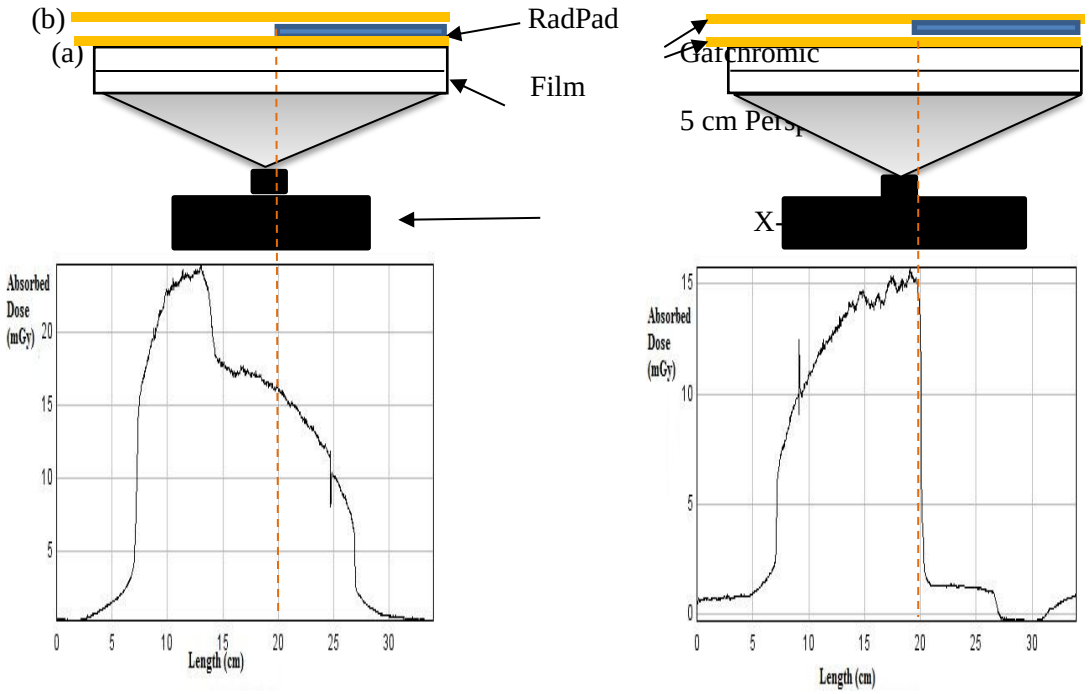
Table 1 shows the results of backscattered radiation measurement from the RadPad Orange and Yellow. The "+" sign indicates that the radiatin dose increased when RadPad was applied on the phantom; whereas the "-" sign indicates that no increment of radiation dose when RadPad was applied.

TABLE 1. DIFFERENCES BETWEEN RADIATION DOSE MEASURED AT THE SAME POINT WITH AND WITHOUT RADPAD APPLIED ON THE PHANTOM

Energy (kVp)	RadPad Orange (μGy)	RadPad Yellow (μGy)
60	-0.093 ± 0.091	+5.120 ± 0.070
70	-0.162 ± 0.074	+7.573 ± 0.232
81	-0.474 ± 0.201	+9.968 ± 0.114
90	-0.534 ± 0.358	+11.103 ± 0.100
102	-0.580 ± 0.259	+13050 ± 0.100
117	-1.240 ± 0.251	+12.975 ± 0.330
125	-1.440 ± 0.329	+12.575 ± 0.275

3.1.3. Dose profiles before and after attenuation by the RadPad

Fig. 3 shows the dose distribution before and after attenuation by the RadPad Orange. There was a sharp dose fall-off at the edge of the RadPad protection drape, as shown in Fig. 3b.



3(a) Before RadPad *3 (b) After RadPad*

FIG. 3. Dose profiles before and after attenuation by the RadPad

3.2. Study of scattered radiation dose during fluoroscopy-guided procedure

The scattered radiation dose measured at different distances at the brain, thyroid and chest of the ATOM (radiologist) phantom. The scattered radiation dose for brain was not detected during the study and no scattered radiation was detected at all the organs above 0.6 m away from the RANDO (patient) phantom. The thyroid was reduced to 100% with the RadPad protection drape applied onto RANDO phantom for all distance measured. For the chest during 15 min fluoroscopic time, the percentage reduction was reduced to 82.6 ± 0.1% at 0.3 m, 57.0 ± 0.2% at 0.4 m and 56.3 ± 0.1% at 0.5 m when the RadPad was applied onto the RANDO.

4. DISCUSSION

4.1. Physical characterization of RadPad

The attenuation of a X-ray beam is dependent on the X-ray energies, types of material, density and thickness of the shielding material. Bi-83 has an X-ray absorption edge at 90.8 keV, therefore it is used primarily in the construction of RadPad material. The other material used is the Sb-51 which has an X-ray absorption edge at 30.5 keV. The mixture ratio of Bi-83 to Sb-51 is not given by the manufacturer. There is no different in term of the material used in the RadPad Orange and Yellow except their thickness (0.21 versus 0.97 mm).

From the study, the percentages attenuation of RadPad Orange and Yellow were found to be 85.8 ± 0.1% and 71.6 ± 0.1%, respectively at 90 kVp. These values are slightly below the percentages as claimed by the manufacturer, which are 90% and 75% for RadPad Orange

and Yellow, respectively. However the RadPad Orange has a slighly higher attenuation percentage than the 0.25 mm Pb-equiv. shield. The probability of the photoelectric absorption per unit mass is proportional to Z^3/E^4 where E is the energy of the incident photon and Z is the atomic number. This means more electrons are available for interactions for the material with higher atomic number [12]. In this example, Bismuth has a Z of 83 and Plumbum has a Z of 82.

An important findings from this study is regarding the backscattered properties from the RadPad. It was found that the RadPad Yellow provided higher dose to the phantom (patient) when the drape was applied on the phantom due to backscatter mechanism, however this dose increment did not show when using the RadPad Orange. Therefore it is important to choose the correct type of protection drape to suit the need. Further studies need to be carried out to verify this findings.

4.2. Study of scattered radiation during fluoroscopy-guided procedure

The scattered radiation dose was found to be the highest at the left side of the operator's chest. Generally the closer the body's part to the field of view, the higher the scattered dose received. No scattered radiation was detected at the head even at the closest distance of 0.3 m from the patient. By applying the RadPad on the patient's body, the scattered radiation dose to the chest and thyroid can be efficiently reduced by 82.6 ± 0.1% and 100%, respectively for a prolonged 15 min fluoroscopy-guided procedure.

5. CONCLUSION

The RadPad protection drape can significantly reduce the scattered radiation dose to the staff during a prolonged fluoroscopy-guided procedure. Other advantages of RadPad include light-weight, lead-free, sterile and easy to use. The RadPad Orange provided slightly higher (about 1.2%) attenuation than 0.25 mm Pb-equiv. lead at 90 kVp, however the percentages of attenuation for both RadPad Orange and Yellow were slighly below the specification mentioned by the manufacturer by 4.5 to 4.7%. Backscattered radiation was detected from the RadPad Yellow and it indicates increased radiation dose to the patient when the RadPad Yellow is in place. However, this property was not seen in the RadPad Orange. Further studies are recommended to verify this findings.

REFERENCES

[1] FAULKNER, K., et al. Practical aspects of radiation protection in interventional radiology. in The 10th International Congress of the International Radiation Protection Association (2000).

[2] VANO, E., et al., Radiation exposure to medical staff in interventional and cardiac radiology. The British journal of radiology, 71(849) (1998) 954-960.

[3] MILLER, D.L., et al., Radiation doses in interventional radiology procedures: The RAD-IR study Part II: Skin dose. Journal of vascular and interventional radiology, 14(8) (2003) 977-990.

[4] BUSHBERG, J.T. AND J.M. BOONE, The essential physics of medical imaging, Lippincott Williams & Wilkins (2011).

[5] DA SILVA, E.A., et al. Measurement of doses in interventional procedures at Hospital São Paulo -. in Proceedings of IRPA12: 12. Congress of the International Radiation Protection Association: Strengthening Radiation Protection Worldwide-Highlights, Global Perspective and Future Trends (2010).

[6] KING, J.N., et al., Using a sterile disposable protective surgical drape for reduction of radiation exposure to interventionalists, American Journal of Roentgenology, 178(1) (2002) 153-157.
[7] SIMONS, G.R. AND W.W. ORRISON, Use of a sterile, disposable, radiation-absorbing shield reduces occupational exposure to scatter radiation during Pectoral Device Implantation, pacing and clinical electrophysiology, 27(6p1) (2004) 726-729.
[8] DROMI, S., ET AL., Heavy metal pad shielding during fluoroscopic interventions. Journal of vascular and interventional radiology, 17(7) (2006) 1201-1206.
[9] JEAN-PIERRE, A., et al., A novel protection drape reduces scattered radiation for interventional radiology. Physica medica, 28 (2012) S6.
[10] POLITI, L., et al., Reduction of scatter radiation during transradial percutaneous coronary angiography, A randomized tria l using a lead-free radiation shield. Catheterization and Cardiovascular Interventions, 79(1) (2012) 97-102.
[11] KLOEZE, C., ET AL., Use of disposable radiation-absorbing surgical drapes results in significant dose reduction during EVAR procedures, European Journal of Vascular and Endovascular Surgery (2014).
[12] SPRAWLS, P., Physical Principles of Medical Imaging, Aspen Publishers (1987).

MONITORING EXPOSURE OF PATIENTS AND PERSONNEL IN INTERVENTION RADIOLOGY USED IN MOTHER TERESA HOSPITAL, TIRANA, ALBANIA

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Abstract

Ionizing radiation has become a powerful tool in diagnostic, radiotherapy and nuclear medicine. It is increasing significantly the population's contribution to medical exposures. As a result, a large number of people always need training for the above purposes. The immediate task is the optimization of these exposures, so as to avoid possible unnecessary detrimental effects of radiation, especially for those procedures that are related to high doses in intervention radiology.

Occupational exposures in medicine have to follow the three principles of radiation protection requirements: justification, optimization and dose limits.

Many institutions and companies, aim to reduce the doses received by patients. In addition, there is a significant reduction of doses received by persons exposed professionally. In diagnostic radiology, three is periodical check of the physical and geometrical characteristics of X-ray beam, and if needed, shielding screens are used to divide areas which are not related to the examination. In general, periodic quality control checks are needed of all the equipment required or related with use of ionizing radiation in medical use. The requirements of IAEA International Basic Safety Standards related with classifications of areas, dose limits etc. have to be complied with. Finally, TLD dosimeters for monitoring of personnel exposure are used by all the medical staff which are working in different cabinets of "Mother Teresa" Hospital

1. INTRODUCTION

The aims of this presentation are in medical exposure situations are: Describe and understand the basic elements of the methods and techniques used to perform Quality Control (QC) and Quality Assurance (QA), and measurement of radiation doses in intervention radiology (IR) in medicine; optimization of medical exposures in IR which is closely related to the improvement of the information received from these exposures (QC) using the smallest possible radiation doses to patients; increasing radiation protection measures for personnel performing IR procedures to minimize their occupational exposures.

The project was implemented in IR labs, hemodynamic and angiography at University Hospital Center "Mother Teresa" in Tirana. The cabinets are equipped respectively with a "Coroskop Top" from the manufacturing firm Siemens, which is used for heart catheterization, and an Angiography also manufactured from Siemens, which is used to examine the brain.

For measuring radiation doses in hemodynamic, a DAP device, supplied by the International Atomic Energy Agency in the framework of the University Hospital Center "Mother Teresa" with this agency, was used.

2. MATERIALS AND METHODS

A Siemens Angiograph is used a for cardiology procedures produced in 1995, presented in Fig.1, and a Siemens Angiograph for neurology procedures device produced in 2007 featured on Fig. 2. The equipment were used for the procedures of heart, brain and peripheral parts catheterization performed under a continuous fluoroscopic monitoring. This device provides automatic adjustment of the voltage and current of the fluorescent tube (automatic brightness control system), depending on the examined anatomical area and the thickness of the patient.

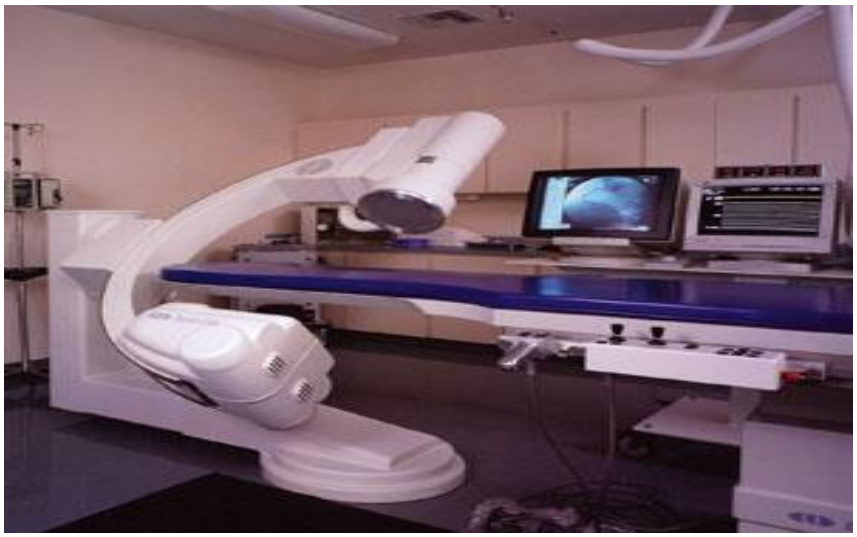

FIG. 1. Coroskop Top for cardiology procedures.

During a routine procedure of QC, of radiation output (O / P) by X device was measured at 75 cm distance to the whole range of possible fluoroscopic voltages of the tube.

FIG. 2. Angiograph used in neurology procedures

In 80 kVp voltage, the output O/P was 0.107 mGy /min. As and Half-Value Layer (HVL) was 3 mm aluminum (Al), for an average sized patient (equivalent to 4 cm Al), the voltage and current of the fluoroscopic tube were: 67 kVp and 1.9 mA, and the respective dose power at the entrance of the fluoroscopic grid (Scattered) was measured 0.43 ± 0.1 μGy / s. All measurements were performed using a digital multimeter (PMX-III R / CT RTI Electronics, Molndal, Sweden) with a detector (R25), and having a calibration done in a standard laboratory.

The device "DAP" consists of a room ionization meters and a Cp type " DIAMENTOR E2", where are displayed the data of the radiation dose rate and the dose-area product given by the ionization chamber.

FIG. 3. DAP device view

Exposure data were recorded for 100 patients who had different diagnostic procedures with fluoroscopic lookout. For each patient, age, weight, type of procedure, fluoroscopic time,

higher values of voltage and current, the dose rate and dose-area product values obtained by DAP were recorded. When used, a lead apron and thyroid collar with 0.5 mm lead equivalent, the thyroid, chest and gonad doses will be less than 3% of these values.

3. RESULTS AND DISCUSSION

The annual limit on effective dose for occupational personnel is 20 mSv and the respective limits for equivalent dose to the skin (as well as the hands and feet) and the eyes are respectively are 500 mSv and 150 mSv. As per the personal dosimeter readings, for compliance with the limits, only 320 procedures in a year can be carried out (for personal dosimeters located in the thoracic level and outside protective lead apron). There is operational need to perform more than 320 procedures in a year.

When aprons and protective collar are used, the actual effective dose will be only a small fraction of the limit, and hence more than 3000 such procedures may be performed before the surgeon working will receive an annual effective dose exceeding the limit. For the hands, feet and eyes more than 1900, 4440 and 17800 procedures in a year should be carried out in order to reach the dose equivalent limits to the skin and eyes

Surgical Procedure	No off Patients	Exposure Time (min)		kVp	mA	DAP μGym^2		Max. DAP rate $\mu Gy\ m^2/s$	
		Limit min, max	Average ±SD	Average ±SD	Average ±SD	Limit	Average± SD	Limit	Average ±SD
Pacemaker	10	0.4 1.7	0.9±0.5	70 ±5	4.4±2.3	1942.3 2349.7	2056.90± 123	54 82	73±0.8
PTCA+STENT	17	6.4 25	14.58±7	180 ±7	12.3±3.9	6028.4 23048.9	15079.2± 428	75 354	210±45
SAK II APP	32	1.2 3.9	2.7±0.6	90 ±3	14.3±4.7	2350.49 6473.24	5854.95± 162	64 159	128.±32
SAK II APQ	27	1.1 3.5	2.5±0.4	90 ±5	13.9±5.3	1937.65 5698.13	4758.82± 157	47 138	110±28
Celebral Hemorrhage	14	7.9 18	16.47±2	190 ±8	10.7±4.5	8025.46 21045.9	19433.2± 423	180 392	357.±52

From Table 2, it appears that IR does not pose significant risk to the operator and his assistants. The scattered dose at a distance of 2 m for an average procedure with fluoroscopic time and exposure factors recorded in this material is only 0.01 mSv, that is equal to the daily dose limit in uncontrolled areas. Therefore, a sanitary or any other assistant who stands at a distance of 2 m or more during fluoroscopi can participate in 1 procedure in a week without exceeding the current effective dose limit of 1mSv in a year available for non-professionally exposed personnel and the members of public. The data on the doses of surgeon during

various types of surgical in cardiology and neurology procedures in the literature are presented in Table 1.

TABLE 2. OCCUPATIONAL DOSE VALUES*

Profession	Period of 2 months Average values (mSv)	Annual total dose mSv /year	dose limit mSv /month	dose limits mSv /year
Cardiologist 1	0.53	3.5	1.4	20
Cardiologist 2	0.42	2.7	1.4	20
Neurologist	10.47	3.2	1.4	20
Nurse (Cardiology)	0.35	2	1.4	20
Nurse (Neurology)	0.27	1.5	1.4	20

*TLD measured during cardiac and neurological procedures in a 1-year

4. CONCLUSIONS

In the past 25 years, there has been an increase of the IR. Age distribution of patients who undergo IR is between 40 - 80 years. But many children undergo these procedures, especially in our country where "mutations" in the heart are very prevalent.

IR is characterized by high doses of the patient as compared to those in diagnostic X-ray examinations. The doses come from a combination of long fluoroscopi time, great power during these procedures, and the required number of radiographic images.

In some cases, dose levels exceeding the threshold for deterministic effects. Stochastic effects should be taken into consideration for IR procedures in specific groups such as children. Some ways to optimize IR procedures include accepting some image noise in order to reduce patient dose. A great effort is made to ensure that the equipment on the market to optimize for the quality of the image /dose and not maximize dose and image.

This paper shows that by using the results of QC procedures to evaluate Entrance Skin Dose (ESD), one can have a reliable method for monitoring patient dose. Staff radiation dose level should be monitored and regularly seen in order to be sure that the doses are lower than the prescribed limits.

REFERENCES

[1] FAULKNER K. et al., Radiation doses and somatic risk to patients during cardiac radiological procedures, British J. Radiology, 59 (1986) 359-363.
[2] MACCIA C. et al., Patient Doses in Interventional Radiology, Radiation protection in interventional radiology, London, British Institute of Radiology (1995).
[3] MARSHALL, N. W., NOBLE, J. DHE FAULKNER, K., Patient and staff dosimetry in Neuroradiological procedures, Br. J. Radiol. 68 (1995) 495-501.
[4] VANO, E. GUIBELALDE, E., FERNANDES, J. M. AND GONZALES, L. Patient dosimetry in Interventional Radiology using Slow Films, Br. J. Radiol. 70 (1999) 95-200.
[4] VANO, E., GONZALES, L., GUIBELALDE, E.,FERNANDES, J. M AND TEN, J.I., Radiation exposure to medical staff in interventional and cardiac radiology, Br. J. Radiol. 71 (1998) 954-960.
[5] WAGNER, L., EIFEL, P. J. AND GEISE, R.A., Potential biological effects following high X-ray dose interventional procedures, J. Vasc. Interv. Radiol. 5 (1) (1994) 71- 84.
[6] SHOPE, T. B., Radiation induced skin injuries from Fluoroscopy, Radiographics 16 (5) (1996) 1195 – 1199

[7] PUBLIC HEALTH ADVISORY, Center for devices and radiological health, Food and Drug Administration, Avoidance of serious X-ray – induced skin injuries to patients during fluoroscopicallyguided procedures, FDA (30 Sep 1994)
[8] GRAY, J. E. et al., Quality control in diagnostic imaging: a quality control handbook, Baltimore, MD, University Park Press, (1993).
[9] Measurement of Performance Characteristics of Diagnostic X-Ray Systems used in Medicine, York, Institute of Physical Sciences in Medicine (IPEM Report 32), Part II, 1996, X-ray Tubes and Generators, 2nd edition Medical Exposures: guidance on the 1990 Recommendations of ICRP, Documents of the National Radiological Protection Board, 4(2) (1993) 43 – 74.
[10] Transmision, absorption, scatter and attenuation based on sites http : // www.ndt-ed.org/ and http: //www.life.rmit.edu.au/
[11] CONFERENCE OF RADIATION CONTROL PROGRAM DIRECTORS, INC., Q.A .Collectible,
kVp measurements, Aailable from http://www.crcpd.org/Pubs/QAC/kVpMeasurements.pdf.

Session 10: Occupational radiation protection in nuclear/fuel cycle facilities

INTRODUCTION OF KHNP'S DOSE REDUCTION PROGRAMS

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Abstract

The Korea Hydro and Nuclear Power Co. (KHNP) has maintained a phased dose reduction plan every ten years. The phase-1 dose reduction plan started in 1991 and was applied to 9 plants at Kori, Hanbit, Hanul, and Wolsong sites. The phase-2 plan started in 2001 for 16 plants at the same sites, but it ended in 2006 because the reduction goal was achieved early. The phase-3 plan started in 2007 and will be in effect until 2016. Statistical analysis shows that the overall dose reduction will be about 30% as a result of the plans. The effective methods of dose reduction were steam generator replacement, removal of RTD bypass lines, installation of tritium removal facilities, and zinc injection. KHNP is now preparing a new dose reduction plan based on the results of EPRI ALARA assessment and several on-going R&D projects.

1. KHNP'S PHASED DOSE REDUCTION PLANS

The phase-1 dose reduction plan ran from 1991 to 2000 and included the improvement of operating practices, facilities, maintenance equipment, and radiation management practices. The phase-2 plan ran from 2001 to 2010 and included the installation of tritium removal facilities and improvement of the internal dose assessment system. However, the phase-2 plan ended in 2006 because the dose reduction goal was achieved early. The phase-3 plan, which aims to upgrade systems one step of WANO PI, started from 2007 and will run until 2016; it includes steam generator replacement and zinc injection. Table 1 provides outlines of the dose reduction plans.

TABLE 1. KHNP'S PHASED DOSE REDUCTION PLANS

	Phase-1 Plan ('91~'00)	Phase-2 Plan ('01~'10)	Phase-3 Plan ('07~'16)
Main Reduction Methods	-Removal of RTD bypass lines -Brand new ECT equipment -Improvement of radiation safety services -Adoption of new nozzle dams -Improvement of RCS pump shafts	-Installation of tritium removal facilities -Application of chemical decontamination -Improvement of internal dose assessment systems -Initiation of ALARA workshops	-Steam generator replacement -Zinc injection -Ultrasonic cleaning of fuel rods -Simplification of Rx heads
Reduction Goal of Final Year (man·Sv)	1.2	0.78	0.43

2. Result of the Dose Reduction Plans

Fig. 1 shows the trend of radiation exposure of KHNP from 1991 to 2010. Over 80% of the total dose was found to occur during plant outages. Fig. 2 shows the results of the phase-1 plan; in this figure we can see that the annual doses are in the range of 0.91~1.42 man-Sv. These values denote an approximately 40% dose reduction compared to the values from the 1980s. The effective methods of the first plan were the removal of RTD bypass lines and the

adoption of new nozzle dams. For instance, the collective dose was found to decrease 46% before and after using the new nozzle dams.

Fig. 3 shows the results of the phase-2 plan; in this figure we can see that the annual doses are in the range of 0.59~0.77 man-Sv. These values denote an approximately 43% dose reduction compared to the values from the 1990s. The effective methods of the second plan were the installation of tritium removal facilities and the application of chemical decontamination to RCS.

Fig. 4 shows the results of the phase-3 plan; in this figure we can find that the annual doses are in the range of 0.46~0.82 man-Sv. These values denote an approximately 8% dose reduction compared to the values from the 2000s. At this point, dose reduction considerably slowed; as a result, it can be inferred that the main dose reduction actions were completed. The effective methods of the third plan were ultrasonic washing of the fuel rods and zinc injection.

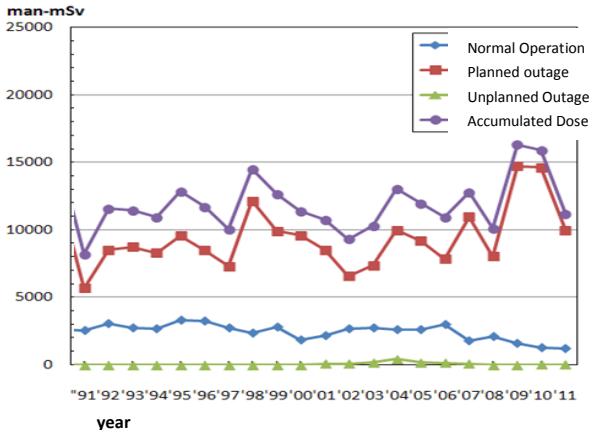

FIG. 1. Trend of Radiation Exposure

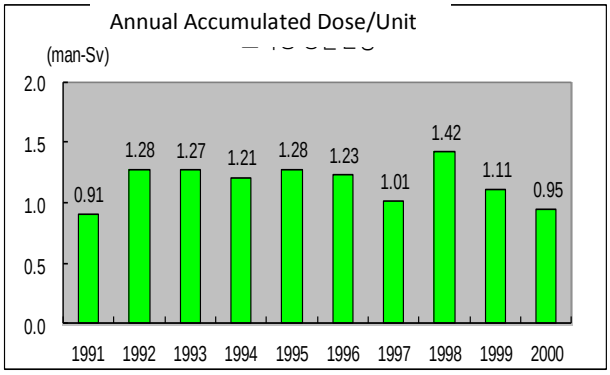

FIG. 2. Results of Phase-1 Plan

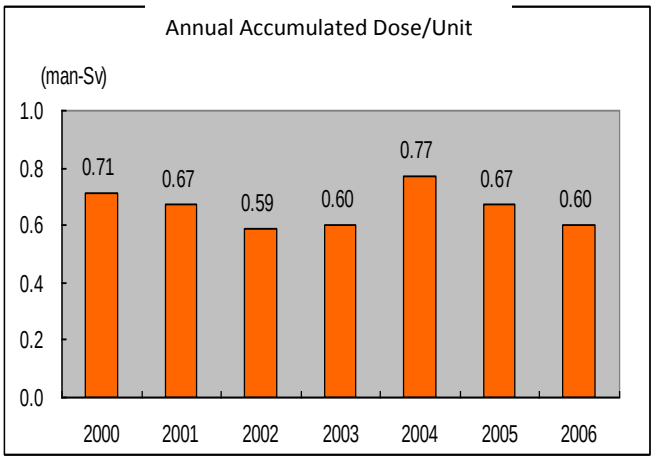

FIG. 3. Results of Phase-2 Plan

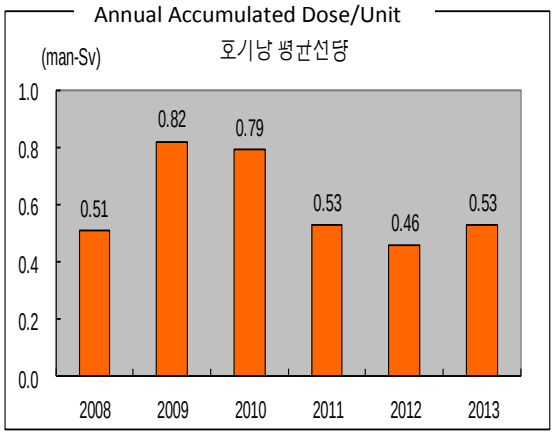

FIG. 4. Results of Phase-3 Plan

3. CONCURRENT ISSUES FOR FUTURE PLAN

Using the assessment methodology of EPRI, an independent analysis of one PWR plant was performed during the period of May 6-10, 2013. This assessment identified a number of good ALARA practices and was able to make recommendations including lower thresholds for ALARA reviews and post-job reviews.

There are several on-going studies related to radiation protection in KHNP; details are shown in Table 2. Fig. 5 shows one of the research outputs, denoted ‘Real-time Personal Dose

Monitoring System (RPDMS)'; this indicator has been applied to Hanbit unit-3 and Hanul unit-4.

Concurrent issues for future dose reduction planning at KHNP are:

(a) Preparation of radiation protection for decommissioning
(b) Paradigm shift of ALARA culture and micro-Sievert
(c) Application of up-to-date ITs to radiation protection management
(d) Maintaining the measurement authority

TABLE 2. CURRENT KHNP R&D TOPICS

No	Title	Period	Remarks
1	Research on countermeasures to reduce the occupational radiation exposure for long-term operated nuclear power plant	12.3~14.2	Completed
2	Construction of standard radiation field and process improvement for tests of personal dosimeters	11.4~14.3	Completed
3	Development of assessment technology and reduction measures for expected airborne tritium effluents at Hanul NPP units 1 & 2	12.4~14.3	Completed
4	Development of an ALARA Type Radiation Monitoring System	11.4~13.8	Completed
5	*Development of purification system to reduce residual radiation in SFP*	*'15.5~'17.11*	*Planning*
6	*Development based technologies for radiation protection during decommissioning*	*'15.9~'18.8*	*Planning*

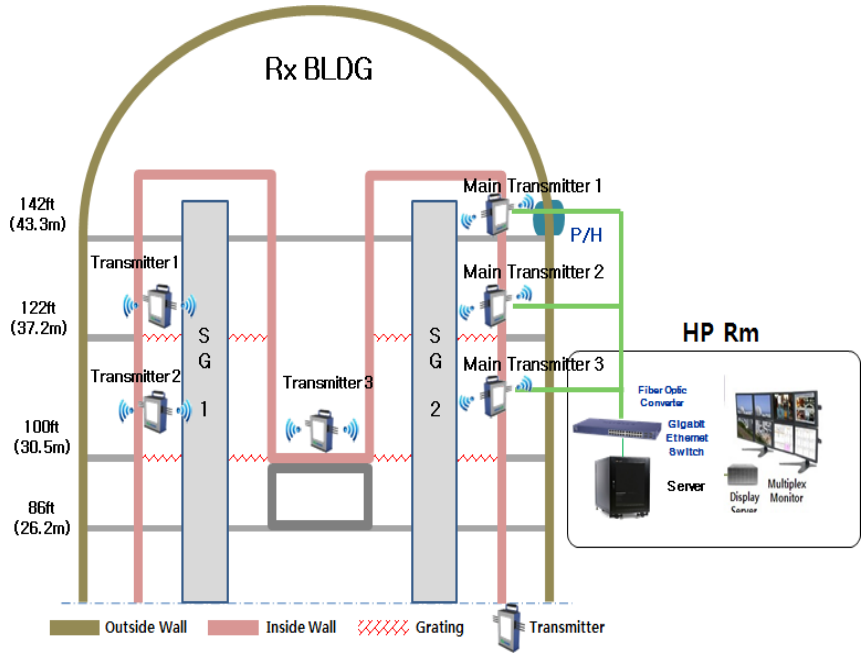

FIG. 5. Schematic diagram of KRMS in Hanul unit-4

4. CONCLUSION

KHNP's dose reduction plan shows that an approximately 30% dose reduction has been achieved in each phase. The effective methods of dose reduction include steam generator replacement, removal of RTD bypass lines, adoption of new nozzle dams, installation of tritium removal facilities, ultrasonic washing of fuel rods, and zinc injection. Using the assessment methodology of EPRI, an independent analysis of one PWR plant was performed during the period of May 6-10, 2013. This ALARA assessment gave us a chance to review our RP practices, to derive many good practices, and to formulate recommendations including lower thresholds for ALARA reviews and post-job reviews. KHNP is preparing the phase-4 plan for the preparation of radiation protection during decommissioning, the change of the ALARA culture and micro-Sievert, and the application of up-to-date ITs to radiation protection management.

REFERENCES

[1] KHNP T.R., Research on the countermeasures to reduce the occupational radiation exposure for long-term operated nuclear power plant (2014).

OCCUPATIONAL RADIATION EXPOSURE AND RADIOACTIVE SOURCE TERMS CONTROL OF PWRS IN CHINA

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Abstract

The improvement of radiological safety status and reduction of collective radiation exposure are important objectives for all NPPs. More than eighty percent of the occupational collective doses are received during the outage for PWRs. Activated corrosion products deposited on the surface of reactor coolant system and other equipments, are the main causes of these doses. A measurement program called "the occupational exposure source term characterization and dose assessment during the outage of NPPs" was launched in China by CNNC for Qinshan phase II and other NPPs. The objective of this project is to improve knowledge of radiological source term and the related occupational exposures in nuclear power plants.

1. INTRODUCTION

China National Nuclear Safety Administration (NNSA) has published the data of annual average Collective Radioactive Exposure (CRE) dose data each year. A good industry perfomance of occupational dose received by workers can be proved by the comparison between CRE dose and the index median/advanced values provided by WANO as shown in Fig.1.

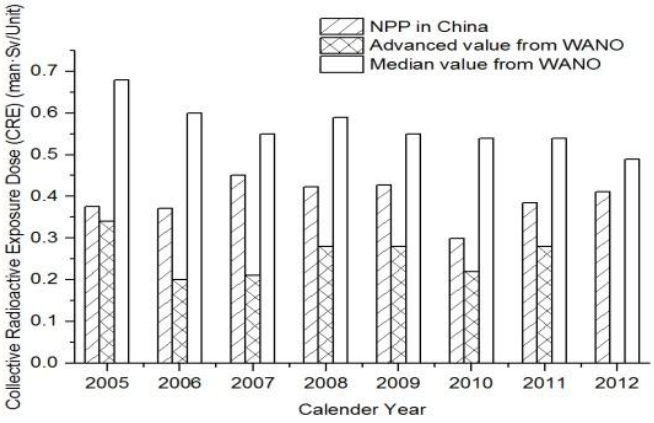

FIG.1. Comparison between average annual CRE dose in China and index from WANO(2005~2012)

The fundmental occupational radioacitve requirments for operating and constructing NPPs in China is based on a comparatively integrated national regulation and standard

system, which is built under the "radiation protection basis frame" established by internationally authorized institutions IAEA and ICRP. After ICRP 2007 recommendation(ICRP Publication 103) published, IAEA issued the GSR Part3 in 2011, according to relevant changes, NNSA revised the major items of occupational exposure control as follows:

1) Under normal operating condition, the personal dose limit and dose constraint are respectively 20mSv (average value over 5 consecutive years; the value should be no more than 50mSv in any single year) and 15mSv/y.

2) According to ALARA principle, target value in design and management during operating, which should be an appropriate portion of dose limit and reflect the concept of dose constraint, is required to be illustrated.

3) Optimization value of CRE was required as one of regulatory requirements in form of design target value which is 1man.Sv/GWe.a (upper limit of a single year).

For the purpose of ALARA and in order to achieve these enhanced objectives, NPP licencees, engineering, design and research institutes in China have started a series of research works about dose assessment, radioacitive field analysis, source term measurement and reduction control. CIRP,CNPE and CNNP in CNNC developed a program called "the occupational exposure source term characterization and dose assessment during the outage of NPPs" initiated to promote radiation dose reduction in operating NPPs (Qinshan phase-I, II, III, and Tianwan NPPs.) and apply an experience feedback procedure in new NPP design. This paper gives a brief introduction about the relative work in Qinshan Phase-II NPP.

2. TECHNICAL METHODS

2.1 Dose Assessment on Qinshan Phase II NPPs' Occpational Radioactive Exposure

Four 650MWe 2 loops PWR units in Qinshan Phase II NPP have been safely operated over 30 unit-years with 19 times outage before 2013, without any unplanned exposure or collective and individual dose of over exposure, no event of personnel radioactive contamination occured. The CRE and maximum individual dose are much lower than the control target. Analyzing the dose monitoring database, an obviously result can be found, which is over 90% CRE dose was receieved during the outage period as shown in Fig 2.

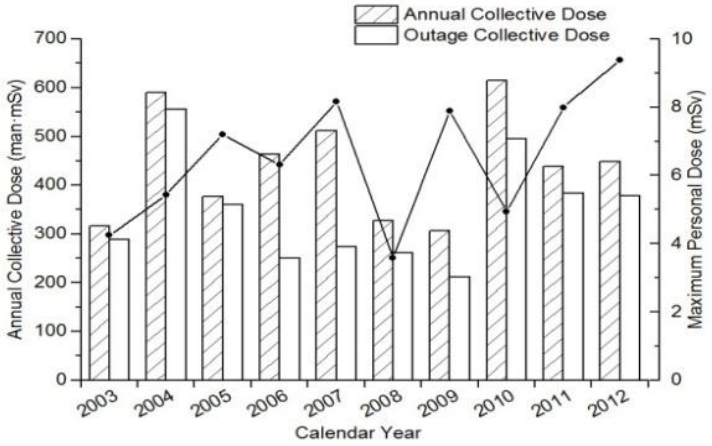

FIG. 2 Collective and individual Dose data of Qinshan Phase II NPP

Referring to the recommended dose assessment process in RG8.l9(NRC,1979) and considering the ISOE statistical category, each potentially significant dose-causing activity at Qinshan Phase II NPP was evaluated by using the histroy recording data. The CRE dose data distribution of unit 1 for a typical annual outage and for a decade outage are shown in Fig.3.

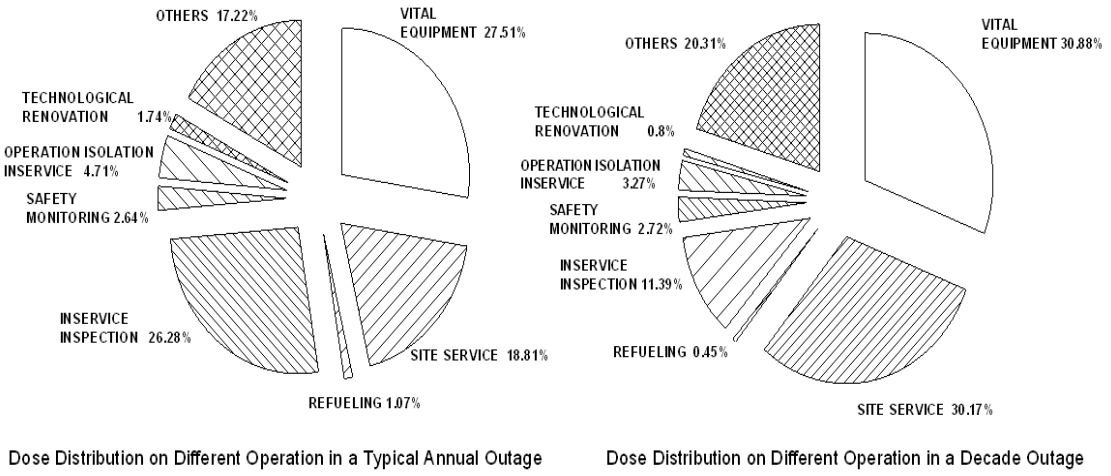

FIG.3 Outage CRE dose distribution of Qinshan Phase II NPP unit 1

Clearly the same as other PWR NPPs, four special activities on main equipments（reactor vessel operation,steam generator and pump maintenance,etc.）, site service, inservice inspection, and others（waste solidification, electrics, instrumentation and control, etc.）are the dominant CRE dose contributor to annual and decade outages (over more than 80% and 90% in total of the entire dose respectively). The dose can be confirmed to be mostly caused by the activited corrosion products deposited on primary coolant system, known as the "crud". The phenomena associated with crud is very complex, the parameters of effective operation days, steam generator area and tubing materiality, primary coolant chemistry control（pH value, shutdown oxygenation, etc.）, decontaminate flow rate, etc, can all impact the behaviour of crud strongly, it is difficult to make an accurate analysis by using current computer codes. An essential method to evaluate the source term due to corrosion products is by collecting the most accurate operating experience involving making regular measurements at exactly the same locations throughout the lifetime of the plants.

2.2 Activated Corrosion Products Measurement

The measurement campaigns were performed during the NPP's outage (i.e., after shutdown). There are typically15-20 measurement pointsfor each measurement campaign. As an example, a typical measurement program is presented in table 1.

TABLE 1. EXAMPLE OF THE MEASUREMENT LOCATIONS

Measurement point	System	Location
1~6	RCP (RCS)	Hot leg-Loop, Cold leg-Loop, Crossover leg-Loop 1&2
7	RCP (RCS)	Pressurizer surge line
8~13	RCV (CVCS)	Letdown pipe,before demineralized pre-filter, After demineralizer, After demineralized after-filter, After volume control tank, After seal water pump
14~16	RRA (RHRS)	Upstream pipe, RRA-to-PTR pipe ,Downstream pipe
Remarks: RCP (Reactor Coolant System), RRA (Resident Heat Removal System), CVCS (Chemical and Volume Control System), REP (Boron recycle system)		

Both of the dose rate and gamma spectra were measured for each measurement point. The contamination characterization was achieved by using two in-situ gamma spectroscopy systems, called Sterm-HPGe and Sterm-CZT respectively, which were developed by CIRP in 2005. Sterm-HPGe (Source Term-HPGe) was developed based on a HPGe detector, which has a high energy resolution, but with a shortcoming of heavy shielding and low accessibility. Therefore, Sterm-CZT was also developed based on a CZT detector, which has an acceptable energy resolution (2% @ 662keV), light shielding and high accessibility.

The main procedure of data analysis is shown in fig4. The detection efficiencies of Sterm-HPGe/CZT are calculated by using Sterm-MC software. After the processes of spectrum analysis, detection efficiency calculation, and activity calculation, the surface activity was obtained for each radionuclide deposited on the inner surface of measured pipe. The comparison of calculated and measured dose rate would give an evident that the deposited activity was measured properly, meanwhile, the dose-rate contribution of each radionuclide was also obtained, which was a very useful information for assessment of radiation field and occupational exposure in NPPs.

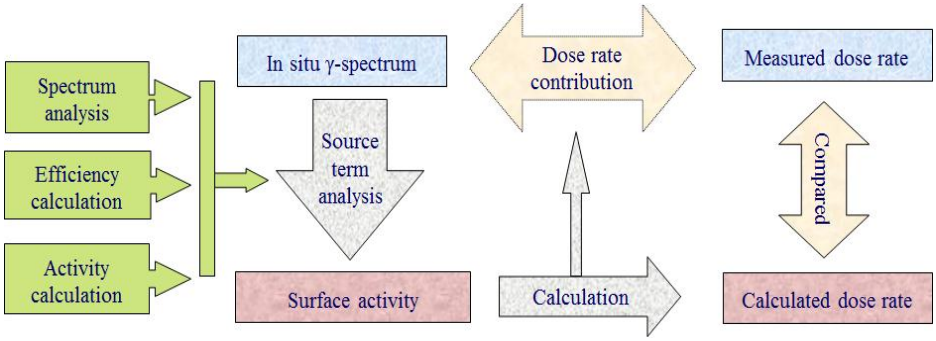

FIG. 4. Flow chart of data analysis

3. RESULTS AND ANALYSIS

Since 2005, it has performed 14 measurement campaigns for several NPPs in China. It can be seen that most activated corrosion products are be recognized in CZT-spectrum, though which has poorer energy resolution if compared to the HPGe-spectrum, as exampled in Fig 5.

The activated corrosion products measured in Qinshan-II are presented in table 2. Almost all activated radionuclides can be measured, and it is also found that, ^{110m}Ag becomes as a main contributor for dose-rate for RRA, RCV and REP besides ^{60}Co and ^{58}Co, when the NPP's operation ages growing, while the main contributors are only ^{60}Co and ^{58}Co for RCP system.

The dose-rate contribution of ^{60}Co and ^{58}Co for RCP system during the outage of 301(unit3,the first time outage), 302, 207 and 108 is shown in Fig. 6. It shows that ^{58}Co plays as the main contributor to dose rate in the first outage, which can reach up to more than 90%. However, it will decrease quickly down to approximately 20% in the 8th outage. In contrast, ^{60}Co will become more and more important with the growing of operation ages, which can reach up to 70% in the 8th outage. An interesting phenomenon was observed that the ratio of ^{58}Co/^{60}Co started with a high value (nearly 100) at the 1st outage, and decreased quickly in the following outages, then approximately seemed to be a constant value of 2 from 5th to 8th outages. Simply reasson can be assumed associate with the decay of radionuclide, the oxygen processing after shutdown, further work of data analysis is ongoing.

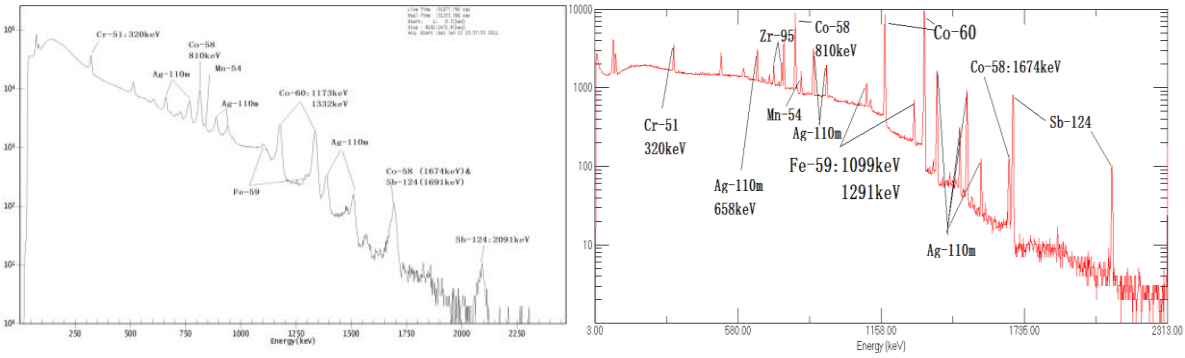

FIG.5. Example of gamma spectra measured by CZT and HPGe detectors

TABLE 2. THE ACTIVATED CORROSION PRODUCTS MEASURED IN QINSHAN-II

Radionuclides	Co-60、Co-58、Ag-110^{m}、Fe-59、Mn-54、Zr-95、Zn-65、Nb-95、Cr-51、Sb-124
Main dose-rate contributors for different systems	RCP: Co-60、Co-58 RRA: Co-60、Co-58、Ag-110^{m} (in later operation time) RCV: Co-60、Co-58、Ag-110^{m} (in later operation time) REP : Co-60、Co-58、Ag-110^{m} (in later operation time)

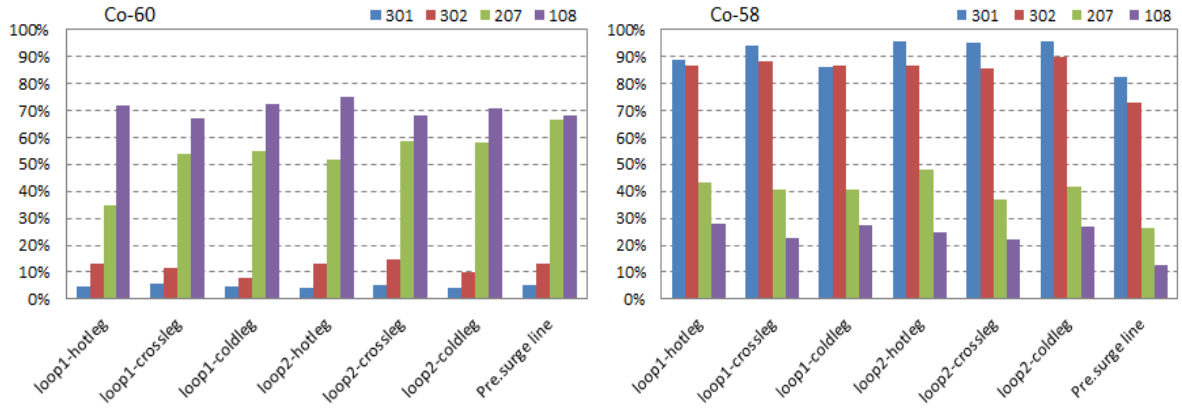

FIG. 6. The dose-rate contribution of ^{60}Co and ^{58}Co for RCP system

4. CONCLUSIONS

As discussed above, efforts of the project “the occupational exposure source term characterization and dose assessment during the outage of NPPs” were focused on assessing the dominanted dose-caused activities and representative radioactive nuclides in PWR NPPs in China. Dose evluation shows that over 80% occupational radioactive exposure dose is contributed from unit outages, mostly associate with the primary coolant works. An essential regular standard dose measurement method was established to evaluate the source term due to corrosion products in Qinshan phase-II NPP. Measurement results can provide a basic data for in-depth assessing the radiological state and occupational dose of NPPs. It gives a tool for studying the impact of chemistry, operation procedure and plant design parameters on radiation fields. It is very helpful for understanding the radioactive contamination mechanisms and can also provide useful information for the future source term reduction program.

REFERENCES

[1] NATIONAL NUCLEAR SAFETY ADMINISTRATION (NNSA), Annual Report, (2005~2012).
[2] U.S.NRC, Occupational radiation dose assessment in light-water reactor power plants design stage man-rem estimates, RG8.19 (1979).
[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation protection aspects of design for nuclear power plants, IAEA Safety Guide No. NS-G-1.13, IAEA, Vienna (2005).
[4] GB18871-2002, Basic Standards for protection against ionizing radiation and for the safety of radiation sources (2002).

OCCUPATIONAL RADIOLOGICAL MONITORING EXPERIENCE IN URANIUM FUEL FABRICATION FACILITY OF INDIA

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Abstract

Nuclear fuel complex (NFC), Hyderabad, India manufactures and supplies 19/37 element Natural Uranium Fuel bundles for all the Pressurised Heavy water reactors (PHWR) in India. The process of fuel fabrication involves different chemical, metallurgical and mechanical operations. Radiological surveillance is an integral part of the process. India has vast experience in providing radiological surveillance in each step of the fuel fabrication operation for the protection of workers and the work environment. Radiation protection procedures are formulated taking into account the absorption class of the airborne activity and the mode of radiation hazards associated with the processes. Work place monitoring and Personnel monitoring procedures and methodologies adopted at various stages of fuel fabrication process are elaborated in this paper. Engineering and administrative measures adopted at various stages have controlled the average air activity level in the work area and occupational exposure to plant personnel. The average uranium air activity in the work area, and occupational exposures to all plant personnel are well below the regulatory limits. The external exposures are well under control and the average external dose to the individual has progressively come down to 0.41 mSv from 1.22 mSv in last ten years. The total average annual individual dose has reduced from 1.74 mSv to 0.87 mSv in the last ten years against the regulatory limit of 20 mSv/y.

1. INTRODUCTION

Nuclear fuel complex (NFC) manufactures and supplies 19/37 element Natural Uranium Fuel bundles for all the Pressurised Heavy water reactors (PHWRs) in India. Hazards associated in handling of Natural Uranium includes both due to its radioactive properties and chemical toxicity. The process of milling of ores for extraction of uranium aims to suppress extraction of its daughter products including radium. Thus freshly separated natural Uranium emits mainly alpha radiation while the immediate daughters of the series are neither high energy gamma emitters nor have high yield. The external radiation comprising of Beta and gamma is emitted by two immediate daughters ^{234}Th and ^{234m}Pa which attains equilibrium in

about 300days. The Fuel fabrication process consists of purification and UO_2 production plant, pelletizing plant, and assembly plant. In the UO_2 production area, the raw materials [(Magnesium Di-uranate (MDU) / Uranium Ore Concentrate (UOC)/Heat treated Uranium Peroxide (HTUP)/ Sodium Diurnate (SDU)] are dissolved in nitric acid to get uranyl nitrate solution. It is further purified and precipitated as ammonium diuranate (ADU). The ADU is then subjected to calcination and reduction operation to obtain UO_2 powder. The UO_2 powder is converted to green pellets of required size and density. The green pellets are sintered at high temperature in reducing atmosphere to get the desired density. The sintered pellets are sent to Assembly plant where they are loaded in zircoloy tubes. The Zircoloy cladded UO_2 pellets are made into fuel elements which are further converted to fuel bundles by arranging 19 or 37 elements in definite geometry and welding them together to end plate at both the ends.

A well-defined occupational radiological monitoring program is in place at Uranium Fuel Fabrication facility operational at Nuclear Fuel Complex (NFC), Hyderabad, India. The basic purpose of radiological monitoring is to estimate the actual exposure of workers and to demonstrate compliance with regulatory limits. In addition, it is also helpful in demonstrating the integrity of engineering systems, confirmation of good working practices (e.g. the adequacy of supervision and training) and engineering standards and to test the efficacy of newly introduced engineered practices. It is also useful in risk assessment studies, to supplement medical records and also for epidemiological studies of the exposed population. This paper describes results of radiological monitoring of work areas at NFC, Hyderabad and the resulting conclusions.

2. MATERIALS AND METHODS

The occupational radiological monitoring program includes ambient dose assessment, air activity assessment and personnel monitoring.

Ambient dose assessment in the work area is carried out using a GM based survey meter which has a range of 0 to 200 µSv/h. Air sampling in work area, using continuous air samplers, is carried out by drawing air through glass fiber filter paper by vacuum line / vacuum pump at the rate of 20 liter per minute. Long-lived alpha activity is estimated using a ZnS (Ag) Alpha Counting System, after allowing 5 days decay to eliminate short-lived radon /thoron progenies. Personnel exposures comprise of internal and external exposures. External monitoring is carried out using thermoluminescence dosimeter in which $CaSO_4$:Dy is used as luminescence phosphor, embedded in Teflon disc [2]. Internal contamination monitoring is carried out for uranium lung burden to all the persons working in areas of oxide and pelletizing plant, once in a year. Graded steel room lung counter is used at NFC for uranium lung burden measurement. It has 20 cm mild steel all around the compartment accommodating a NaI(Tl) (12.7 cm dia. x 1.27 cm thick) detector. Gamma ray photons of 63 keV and 93 keV with abundance of 3.9% and 5.6% respectively from ^{234}Th are used for measurements. Correction for counts due to natural ^{40}K in body is done based on the weight to height ratio of the individual. The detector is calibrated by a realistic anthropomorphic phantom for thorax, developed by Japan Atomic Energy Research Institute (JAERI) phantom for reference Asian Man. Bioassay monitoring is conducted for persons handling Type-M compounds at an annual frequency for U(nat.) in urine. Overnight urine samples are collected in polythene bottle. Uranium in urine is estimated by chemical separation followed by UV fluorimetry [14, 15]. Factors given by ICRP-78 [6] are being used for both bioassay and lung monitoring for estimating the intake. Special internal monitoring is done in case of unusual occurrence of airborne activity.

3. RESULTS AND DISCUSSION

3.1. Ambient air activity assessment and internal dose evaluation

The air activity in work area depends upon the type of operation of the plant. The extent of inhalation hazard depends upon the chemical nature of the radionuclide and the particle size distribution in the aerosols present in the plant. It has been reported that the average Activity Median Aerodynamic diameter (AMAD) for different locations at NFC varied from 5.8 to 7.7 μm [3]. The chemical form of uranium handled in oxide and pelletizing plants of NFC and the corresponding Derived Air Concentration (DAC) are given in Table 1. Continuous air activity monitoring is carried out in these areas and the average air activity of a typical year observed is presented. It can be seen from the table that the average uranium air activity varied from 0.09 to 0.3 DAC. The administrative and engineered controls helped in curtailing the air activity in the plants [1]. This is further substantiated by lung counting and bioassay monitoring of individual workers. All the individuals were found to have internal contamination below the regulatory limits in the last decade of operation.

3.2. Ambient dose assessment and external dose evaluation

External dose received by individuals depends upon the ambient radiation levels and occupancy factor. The ambient dose rate for a typical year varied from 0.25 to 4.5 μSv/h in various plants of NFC (Table 2). Personnel monitoring using thermoluminescent dosimeters indicated that the annual individual external doses varied from 0.23 to 0.67 mSv.

3.3. Total dose to the radiation worker

Fig. 1 shows the dose distribution pattern of radiation workers of NFC for a typical year. All the radiation workers have received dose below the annual limit as per the guidelines given by ICRP [7]. It is observed that around 99% of the individuals had received exposures below 5 mSv in a year which is far below the annual exposure regulatory limit i.e. 20 mSv.

The engineering and administrative controls [8] and imparting regular training to workers helped in controlling the exposure levels. Regular monitoring, feedback and corrective measures which were implemented wherever necessary, helped in controlling the exposure levels.

TABLE I. DIFFERENT AREAS IN THE FACILITY AND THE EXISTING AIR ACTIVITIES

Sl. no	Area	Uranium Compound	Physical form of Uranium	Class	DAC	Annual average air activity(Bq/m^3)
1	Dissolution	MDU/HTUP/UOC	Powder	Class M/S	4.5/1.3	0.42
2	ADU Handling area	ADU	Slurry and Powder	Class M	4.5	0.45
3	Furnace Area	Oxides	Powder	Class S	1.3	0.17
4	Pre Compaction	UO_2	Powder	Class S	1.3	0.39
5	Final Compaction	UO_2	Powder and pallet	Class S	1.3	0.36
6	Centre less Grinding	UO_2	Pallet and Slurry	Class S	1.3	0.35

TABLE 2. DIFFERENT PLANTS IN THE PLANT WITH OPERATIONS AND THE RADIATION LEVELS OBSERVED

Sl. No.	Plant	Operations/areas in the Plant	Range of Radiation levels observed(µSv/h)[#]	Annual average External dose(mSv)*
1	Oxide plant	Dissolution, ADU precipitation, Furnace area	0.3 – 2.5	0.47(247)
2	Pelletization Plant	Precompation, Final compaction, Sintering and Centreless Grinding	0.3 – 4.5	0.67(253)
3	Assembling plant	Tube and Bundle assembling	0.25 – 2.0	0.23(193)

*The number given in parenthesis gives the number of individuals monitored.

[#] Radiation levels monitored at 1 m distance from the container/equipment

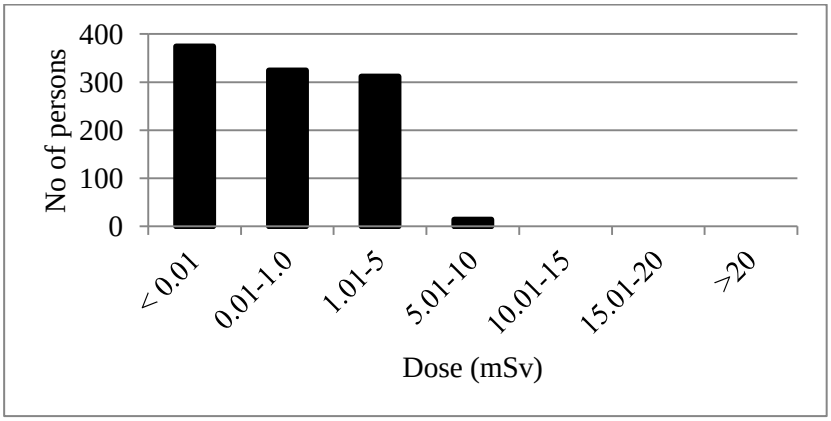

FIG. 1. Total dose distribution pattern of NFC radiation workers for a typical year

4. CONCLUSION

The operational radiation protection procedures adopted at Nuclear Fuel Complex, India are adequate to meet the regulatory requirement. The radiation protection measures taken had reflected in the average annual dose which is far below the annual limit of 20 mSv. No individual was found to cross the action level both in lung counting and bioassay. The decrease in the average air activity over the last decade shows the improvement in administrative and engineered safety features adopted in the facility.

REFERENCES

[1] ATOMIC ENERGY RGULATORY BOARD, Uranium Oxide Fuel Fabrication Facilities, Guidelines No. AERB/FE-FCF/SG-3 (2009).

[2] SNEHA. C., et al, Study of minimum detection limit of TLD personnel monitoring system in India, Radiat. Prot. Dosim. 141 (2010) 168-172.

[3] VISHWA PRASAD K., et al, Aerosol size distribution in a uranium processing and fuel fabrication facility, Radiat. Prot. Dosim. 140 (2010) 350-61.

[4] OAK RIDGE NATIONAL LABORATOTY, Review of uranium bioassay techniques, Bogard, J.S., ORNL-6857 (1996).

[5] SAWANTH PRAMILLA D., et al, Quality assurance exercise for estimating low – levels of alpha emitters in urine samples: Performance of Trombay's Bioassay Laboratory, Radiat. Prot. Environ, Vol.26, No. 1–2 (2003) 134–138.

[6] INERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Individual Monitoring for internal exposures of workers, ICRP Publication 78 (1997).

[7] INERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Recommendations of the International Commission on Radiation Protection. ICRP Publication 60, Ann ICRP 21(1-3) (1990).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of uranium fuel fabrication facilities, IAEA Safety Standards, Specific Safety Guide No. SSG-6 (2010).

ASSESSING THE POSSIBLE RADIOLOGICAL IMPACT OF ROUTINE DISCHARGES FROM PROPOSED NUCLEAR POWER STATION

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Abstract

The aim of this work is to assess the radiological impact on the population around a nuclear power plant in the case of normal operation, where the concentration at different distances from the plant of some radioactive materials may be released from the stack of the plant, especially iodine. The radiation doses to all organs of the body of the workers (at the exclusion zone) and population (low population zone) were calculated. The pathways through which this radiation transports to humans, such as ingestion and inhalation, were investigated.

Doses and risks were tabulated as a function of radionuclide, pathway, location and organ. The effective dose equivalent at the site boundary and annual individual and collective dose values from immersion, inhalation, and ingestion pathways were estimated, also the location of the maximum exposed individual (MEI), of the system of dose limitation recommended by the International Commission on Radiological Protection (ICRP). Lifetime fatal cancer risk to the population living around the proposed site due to the atmospheric pathway was calculated using the CAP88-Pc dose modeling program.

1. INTRODUCTION

The Radiological impact from Nuclear Power Plants on the environment could be evaluated though the calculation of the external whole body dose due to immersion in a radioactive plume , exposure to ground surface contaminated with radioactivity , inhalation of food stuffs due air contaminated with radioactivity and ingestion of contaminated food stuffs due to releases of radioactive effluents.

Fig. 1. illustrates the major radioactive pathways to human. Three steps in evaluation of nuclear radioactive releases and health consequences are:

a) Evaluation of the releases (source term)
b) Evaluation of the dispersion of releases in the environment (transport)
c) Evaluation of the health consequences(doses and risks)

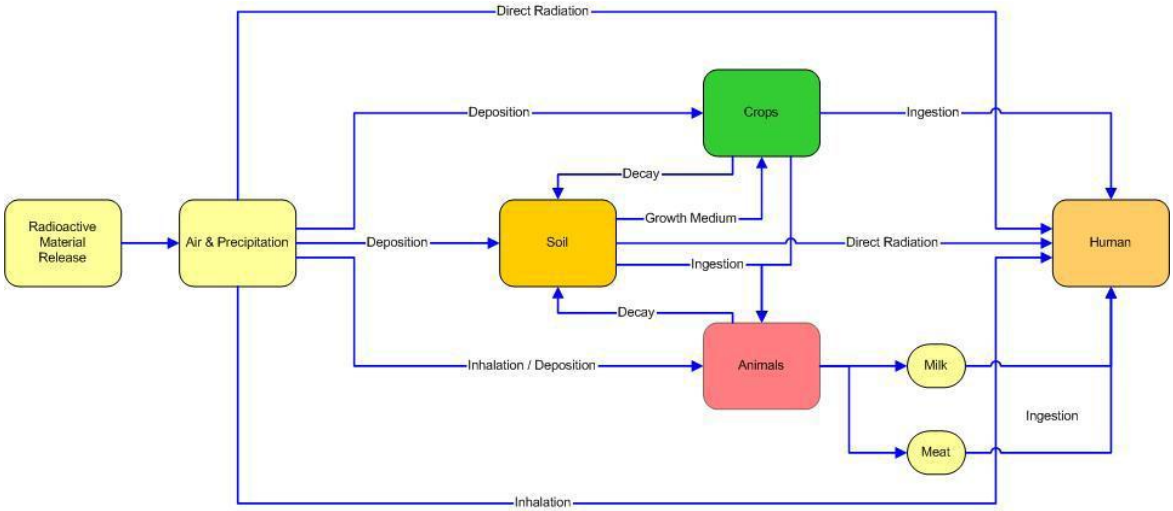

FIG. 1. Atmospheric releases of radionuclides and its major pathways to human

Assessment of Radiation exposure to individuals at various distances from the nuclear reactor has always been a very important consideration.

1. MATERIALS AND METHODS

The atmosphere is an important pathway for the transport of radioactive releases from a nuclear power plant to the environment and thereby to man. The potential health effects arising from routine releases from nuclear facilities are estimated using the computer code CAP88-PC (Clean Air Act Assessment Package-1988 for personal computers) [1, 2].

The following is the typical data and information required as inputs for the CAP88-PC code. Site meteorological data such as wind speed, direction and population data in 16 sectors and for different radii are provided as inputs. Facility-specific radionuclide release rates (Bq/y) were used for continuously monitor the facility (Fig. 2). The environmental factors, such as humidity of the reactor site and specifications such as stack height and diameter are used in the dispersion model are also provided as inputs. Organ dose and related weighting factors follow the Federal Guidance Report No.13 (FGR13; Eckerman et al.1999) method.

2. CALCULATION OF RADIATION DOSES

A person immersed in the plume would inhale an amount of radioactive material proportional to the time of residence in the plume; the person's breathing rate, and the concentration of radioactive material at his or her location. The inhalation dose depends on the radionuclide concentrations at ground level and on the breathing rate it is calculated as follows: Individual dose for air immersion:

$$E_{immersion} = C \times CF_{immersion} \times t$$

where: $E_{immersion}$ = Effective dose from external exposure due to immersion in contaminated air [mSv], C = Average concentration of radionuclide in air [kBq/m^3], $CF_{immersion}$ = Conversion factor for radionuclide,
t = Exposure duration
[h].

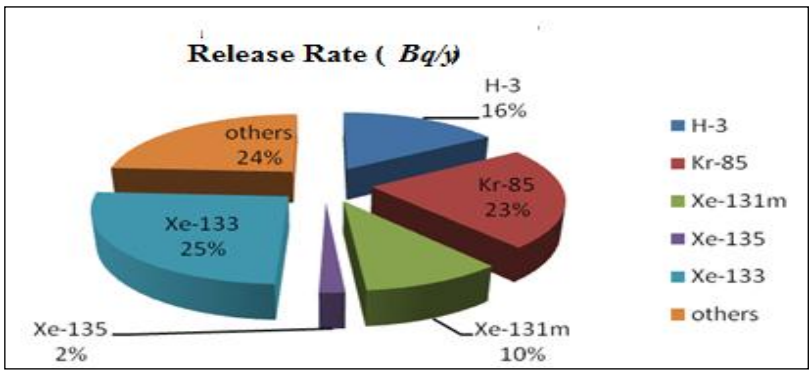

FIG. 2. Radionuclides average release rate (Bq/y) for PWR 1000

The very useful feature of CAP88-PC code is its capability of dose calculation for individual and population and for various radioactive materials. Also, its capability to provide fatal cancer risk in different body organs for both individual and total population around the reactor site is another strong reason of using this code.

3. RESULTS AND DISCUSSIONS

All the above data are inputted to CAP88-PC and the outputs are presented and summarized as follows:

3.1 Ground level (CHI/Q) values for ^{131}I

During normal operation, a nuclear power plant releases radioactive materials which could be transported downwind and dispersed by normal atmospheric mixing process. Iodine-131 (^{131}I), radioactive releases from Nuclear power plant 1000 MWe at normal operation, has an 8.03 day half-life, and emits beta and gamma radiation. The annual-average air concentration is represented by concentration isopleths (lines of constant concentration), also the annual effective dose equivalent (EDE) for the highest exposed individual around the site due to atmospheric pathway were assessed by the Computer code CAP88-PC[2]

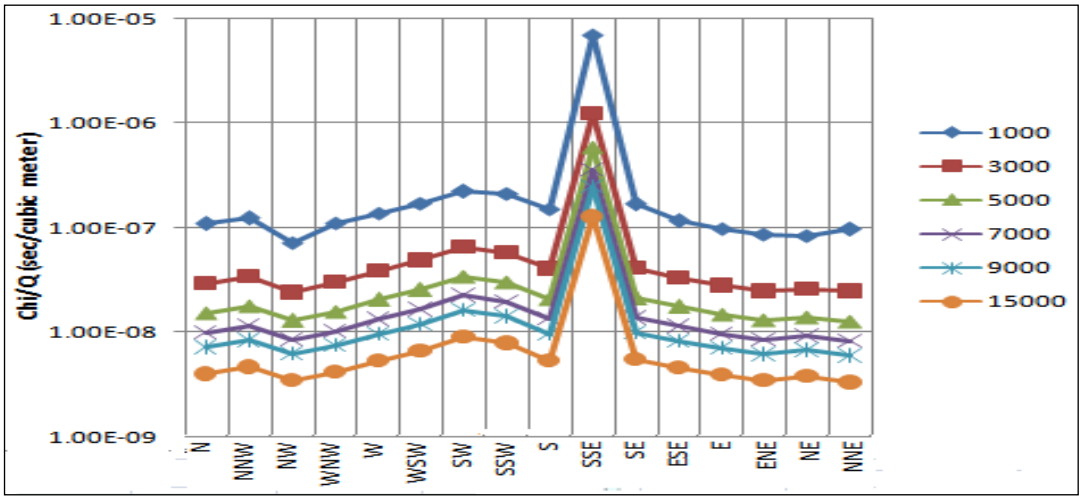

FIG. 3. (Chi/Q) for ^{131}I

3.2. Ground-Level CHI/Q values for ^{3}H

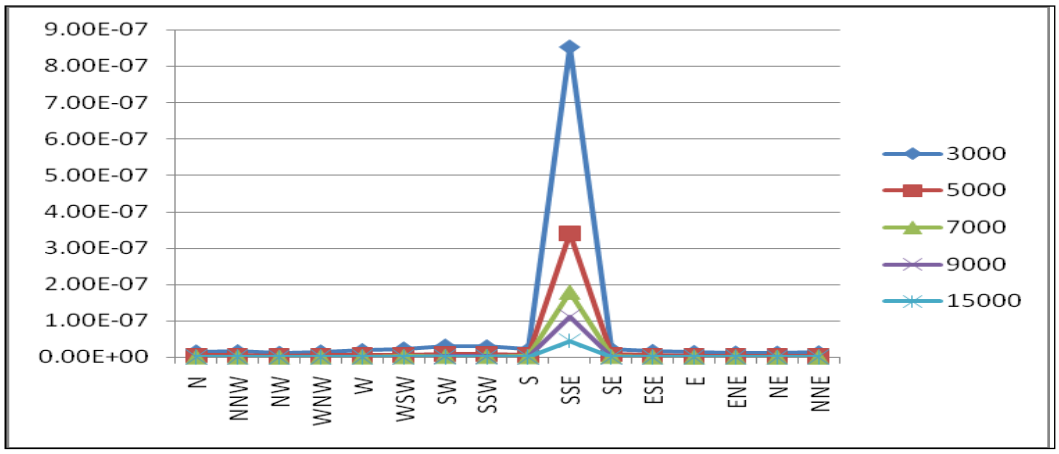

FIG. 4. (Chi/Q) for ^{3}H

Figs. 3 and 4 represent the dispersion factor (Chi/Q) for ^{131}I and ^{3}H for every sector at different distances, South South East (SSE) is the dominant one with highest (Chi/Q), 8.99E-07and 5.99E-06 (sec/cubic meter) respectively.

3.3. Maximum Individual Exposure

The highest concentration that would be expected at 3000 meters radial distance from the release point was towards the south southeast with an estimated effective dose equivalent 0.0261mSv/year (Table 1) as well below the 1mSv/year -annual limits, as shown in Fig. 5.

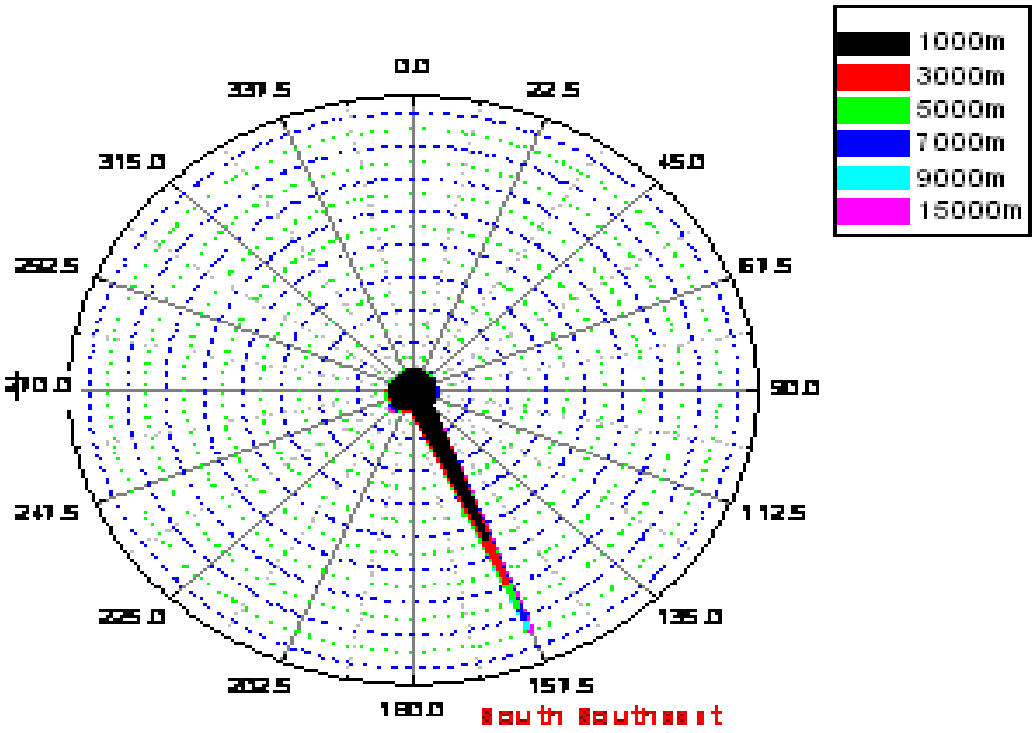

FIG. 5. Maximum Exposed individual sector

3.4. Pathways effective dose equivalent

The different modes of exposure, exposure pathways and the critical (target organs) are shown in the Fig. 6 (a) and the Table 1.

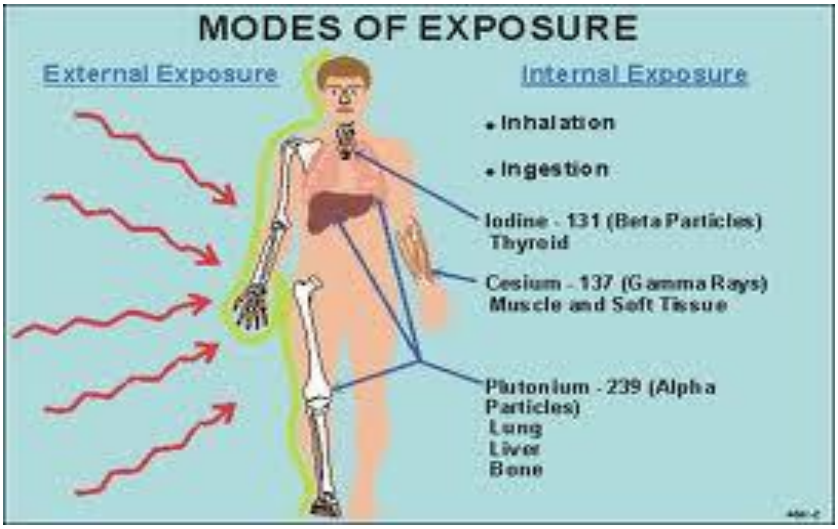

FIG. 6(a). Modes of exposure

TABLE 1. EXPOSURE PATHWAYS AND THE CRITICAL ORGANS (TARGET ORGANS) FOR DIFFERENT RADIONUCLIDES RELEASED THROUGH EFFLUENTS

Radionuclide	Effluent	Exposure Pathways	Critical Organ
Iodine	Airborne	Ground deposition (external) Air inhalation Grass →cow→ milk Leafy vegetable	Whole body Thyroid Thyroid Thyroid
Tritium	Liquid	Drinking water Fish (and shellfish) consumption	Thyroid Thyroid
	Airborne	Submersion(external) Air inhalation	Skin Whole body
	Liquid	Drinking water Food consumption	Whole body Whole body
Cesium	Airborne	Ground deposition(external) Grass→ Cow→ milk Grass→ cattle→ meat Inhalation	Whole body Whole body Whole body Whole body
	Liquid	Sediments(external) Drinking water Food consumption	Whole body Whole body Whole body
Metal (Fe,Co,Ni,Zn,Mn)	Liquid	Drinking water Food consumption	GI tract GI tract
Direct radiation from plant		External	Whole body

The effective dose equivalents to the highest exposed individual resulting from both external and internal irradiation pathways around the NCE nuclear power plant site is given in Table 1 (b) and Fig. 6(b).

TABLE 1(b). ANNUAL EFFECTIVE DOSE EQUIVALENTS

Pathway	(mSv/y)	(person-Sv/y)
Ingestion	1.58E-03	3.66E-03
Inhalation	6.29E-04	9.48E-04
Air Immersion	3.67E-04	5.44E-04
Ground Surface	3.9E-05	5.69E-05
Total	2.61E-03	5.22E-03

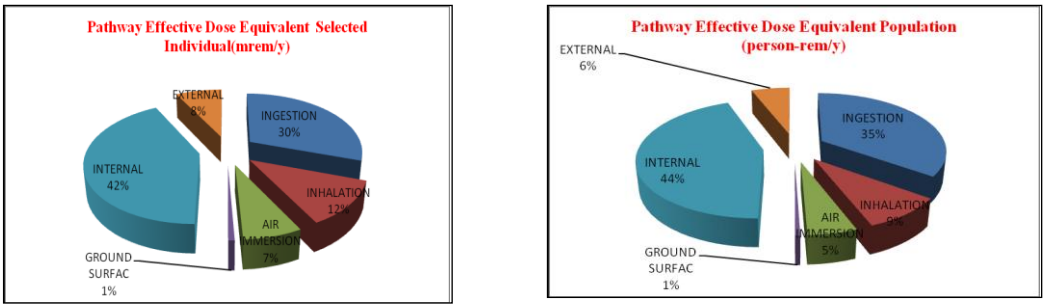

FIG. 6(b). Pathways effective dose equivalent

Calculated Effective Dose Equivalent rate values at all sectors around the hypothetical Nuclear Power Plant for various distances is illustrated in Fig. 7 which shows that the highest prediction Chi/Q values will be on the SSE.

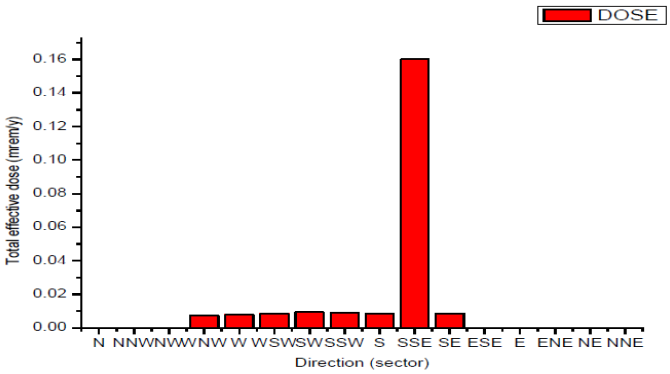

FIG. 7. Effective dose Equivalent for various distances

The highest total effective dose equivalent in populated sectors around the proposed reactor site is determined to be (7.09E-05 mSv) at distance 3000 at SW sector; however, it doesn't exceed the accepted limits (1mSv) for populated area as shown in Table 2. The contribution to individual doses from various nuclides is shown in Table 2.

TABLE 2. NUCLIDE EFFECTIVE DOSE EQUIVALENT SUMMARY

Nuclide	(msv/y)	Nuclide	(msv/y)
Ar-41	7.09E-05	Rb-88	1.3E-06
C-14	1.34E-04	Sb-125	1.29E-7
Co-57	3.71E-10	Sr-89	6.44E-7
Co-58	4.8E-06	Sr-90	1.02E-05
Co-60	1.46E-05	H-3	7.9E-05
Cr-51	1.95e-11	Xe-138	5.8E-06
Cs-134	2.5E-06	Cs-138	2.6E-06
Cs-135	9.53E-9	Zr-95	2.04E-7
Cs-137	1.67E-05	I-131	4.74E-05
Ba-137m	8.78E-7	Nb-95	2.19E-7
Xe-133	1.52E-04	Xe-135m	2.9E-06
Kr-85m	7.7E-06	Xe-135	1.15E-04
Kr-85	3.14E-05	Cs-135	8.73E-15
Kr-87	1.77E-05	I-133	1.32E-05
Rb-87	1.01E-06	Xe-133m	3.6E-06
Kr-88	1.38E-04	Xe-133	1.95E-04

4. DOSE AND RISK EQUIVALENT SUMMARIES

4.1. Organ dose equivalent summary

Thyroid and skin dose equivalents have the highest value 3.31E-04mSv/y, and 6.16E-05mSv/y respectively, and represents 42% and 8% of the total organ dose, as shown in Fig. 8.

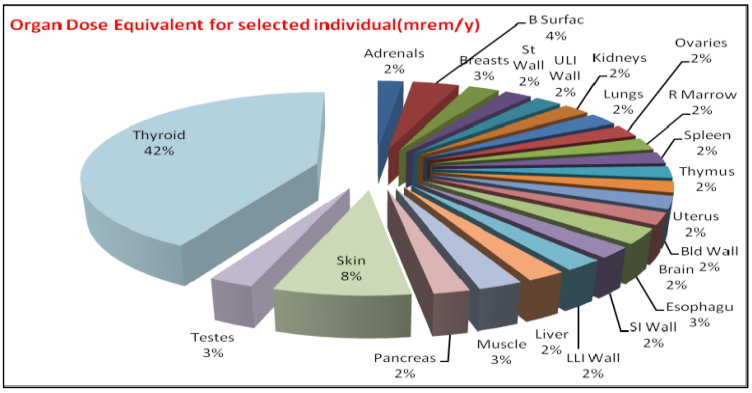

Fig. 8. Organ dose equivalent summary

4.2. Cancer Risk Summary

Results of CAP88-PC as total lifetime fatal cancer for selected individual and total collective population for various organs are obtained, are presented Table 3. And the lifetime fatal cancer risk is shown in Fig. 9.

Lifetime Fatal Cancer Risk is 9.15E-08 as shown in Fig. 9. It can be concluded that the risk of cancer for the personnel working and moving in the reactor exclusion area is very low for routine releases. According to most national and international regulatory guides, dose limits for worker and population are 20 mSv/y and 1 mSv/y respectively. As shown in this presentation, the calculated dose levels are within the acceptable dose limits (ICRP 103).

TABLE 3. CANCER AND FATAL RISKS FOR VARIOUS ORGAN

Cancer	Fatal Cancer Risk	(Deaths/y)
Esophagus	1.52E-11	3.47E-10
Stomach	7.24E-11	1.37E-09
Colon	1.81E-10	3.44E-09
Liver	2.67E-11	5.05E-10
LUNG	1.84E-10	3.48E-09
Bone	2.83E-12	5.36E-11
Skin	6.13E-12	1.16E-10
Breast	9.58E-11	1.81E-09
Ovary	2.57E-11	4.87E-10
Bladder	4.65E-11	8.86E-10
Kidneys	9.17E-12	1.74E-10
Thyroid	1.36E-10	3.13E-09
Leukemia	1.06E-10	2.00E-09
Residual	2.62E-10	4.96E-09
Total	1.17E-09	2.28E-08

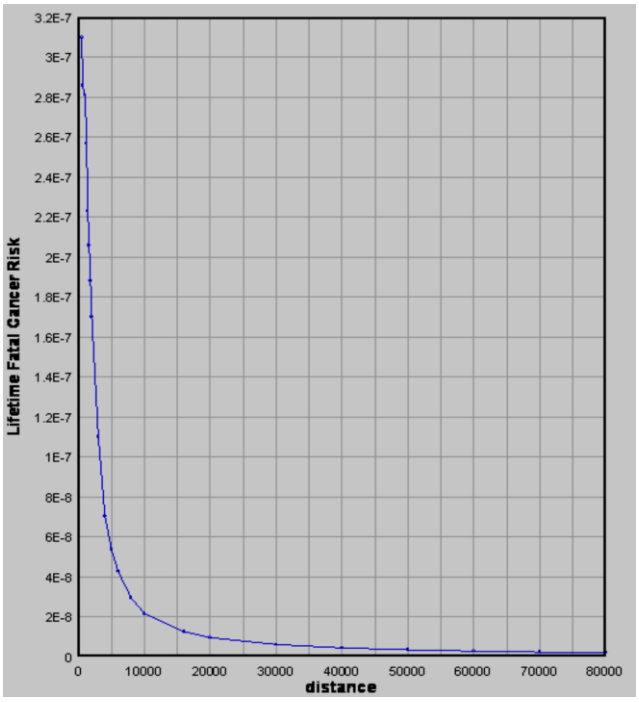

FIG. 9. Lifetime Fatal cancer Risk vs distance

5. CONCLUSION

The following conclusions may be drawn from the above study:

a) Studies of the Radiological assessment are essential in the selection of a nuclear power plant site and in the evaluation of the hazards of nuclear operations.
b) For a maximally exposed individual at a location of the individual is 3000 meters, SSE with an estimated effective dose equivalent 2.61E-03 mSv/year, and this value is below the annual dose limit of 1 mSv for members of the public, as recommended by the ICRP-103[6].
c) Safety criteria and projected dose values should be estimated in accordance with National Standards of Radiation Safety, which are based on the ICRP recommendations and in accordance with IAEA recommendations.

(d) The most affected area with the pollution is the South southeast direction. The results show that the radiation doses are in the range of internationally acceptable levels. Siting factors and safety criteria are important in assuring that radiation doses from normal operation and postulated accidents will be acceptably low.

REFERENCES

[1] http://www.epa.gov/rpdweb00/assessment/CAP88/index.html.
[2] CAP88-PC Version 3.0 User Guide, December 09 2007. Trinity Engineering associates, Inc. 8832 Falmouth Dr., Cincinnati, OH 45231-5011 (2007).
[3] F.PASQUILL, The estimation of the dispersion of windborne material, Meteorological Magazine, 90, pp. (1961) 33–49.
[4] N.NASSAR., "Assessment and modeling of radiation level over North Western Coast of Egypt, Ph.D. Thesis; Nuclear Engineering Department, Alexandria University, (2012).
[5] http://cfr.vlex.com/source/code-federal-regulations-protection-environment-108940
[6] INTERNATIONAL COMMISSION ON RADIOLOGICAL POTECTION, 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Elsevier, (2007).

POLARIS-H IMAGING SPECTROMETER SOURCE-TERM MEASUREMENTS

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Abstract

Understanding the primary source term is critical in limiting occupational exposure in nuclear power plants and similar facilities. One wishes to learn the identity of radioactive materials and their locations and intensities. A new commercial instrument, Polaris-H, is introduced which has energy resolution near that of HPGe and the ability to image spatial distributions of each isotope in all 4π. Additionally, it does not require cryogenic cooling, so can start in 2 minutes. Two sample measurements from a measurement campaign are shown which are important for limiting exposure. In the first measurement, a previously unknown source is discovered under the floor, while confirming the presence of a known source from a pipe. The other measurement confirms the adequacy of shield material and shows contamination along a pipe. Other applications are also mentioned.

1. INTRODUCTION

In order to limit occupational exposure in nuclear power plants and related facilities, special attention must be paid to source-term management. Management of the source term can be done through chemistry changes in the plant, plant design, proper upkeep of parts, and other initiatives. However, in order to understand the source term and learn how to best reduce the source term, it must be accurately measured. Current measurements of the source term are done with electronic dosimeters, high-purity germanium (HPGe) detectors, and sometimes NaI(Tl) scintillators or other detectors. These units can be either non-directional, like most survey meters, or directional, by collimating the detector. In recent years, the new semiconductor CdZnTe (CZT) has also been used [1].

Survey meters can quickly give the dose rate at different locations, and, in the hands of a skilled operator, can be used to locate the primary source terms for dose. However, to fully characterize the source, a detector with high energy resolution and some location specificity, like collimated HPGe, must be used to identify the isotope(s) and activities in each hot spot. Unfortunately, HPGe in not an ideal mobile spectrometer because it must be cooled to about 100 K. HPGe requires either repeated filling with liquid nitrogen or use of a mechanical cooler, both of which add weight and bulk. Additionally, cooling must begin at least several hours before the device will be used.

This paper describes a new commercially available instrument, Polaris-H, which aims to be a practical detector for most source-term measurements. In addition to providing competitive energy resolution to HPGe without the need for cryogenic cooling, Polaris-H is able to give gamma-ray directional information, potentially eliminating the need for heavy collimator shields. In the Methods section, we describe the system and its performance. Then, in the Results and Discussion section, we describe a few measurements performed in recent measurement campaigns at nuclear power plants in Europe and North America. Finally, in the Conclusion, we summarize and point to future applications of this system.

2. METHODS

The underlying technology in Polaris-H is a 20 mm × 20 mm × 15 mm CZT crystal with 11 × 11 pixelated anode. The pixelation allows both position sensitivity for multiple simultaneous interactions, which enables Compton imaging [2], and voxel-by-voxel calibration, which enables high energy resolution [3]. Energy resolution as a function of energy for a representative detector is show in FIG. 1. Because all charge is collected by a single anode pixel when the gamma ray interacts only once, the resolution of events with only one interaction can be better than that of events where there are multiple interactions and multiple pixels that collect charge. Energy resolution is better than simply reading out the entire bulk because of a voxel-by-voxel calibration and other corrections.

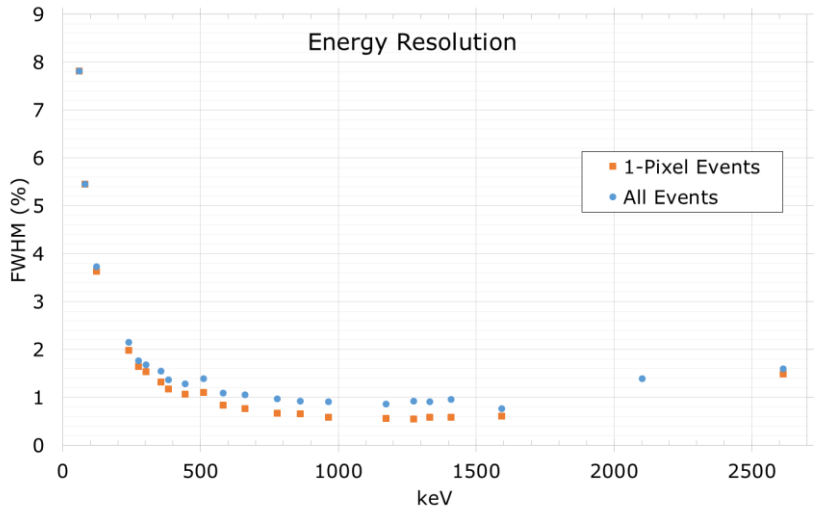

FIG. 11. Energy resolution as a function of energy for a representative Polaris-H detector.

Whenever more than one interaction occurs from a single gamma ray, the event can be used to estimate the emission distribution around the detector. When two or more interactions occur at moderate energies, Compton scattering has occurred. One can use the measured interaction locations and energies to localize the source to somewhere on a surface of a cone, assuming full-energy deposition in the crystal. With enough of these events, the spatial distribution of the emitters can be found. Further, by only imaging events from within the full-energy peak(s) of each particular isotope, an isotope-specific image is produced. Therefore, one can view the primary direction(s) of emissions due to each isotope's un-attenuated gamma rays.

This CZT crystal is packaged into a water-tight enclosure. The enclosure contains the CZT crystal, readout, embedded computer, high-voltage power generation, and an embedded battery. An almost-2π camera is mounted in the front of the enclosure, which allows the radiation image to be matched to features in the optical field of view. Data (list-mode data, optical images, and ANSI N42.42 spectra) are stored to an externally mounted USB flash drive. All together, the system weighs under 4.1 kg and has dimensions of 21 cm × 19 cm × 13 cm. A 7-inch (17.8-cm) tablet is used to display the real-time spectrum and isotope-specific radiation image. It connects to the main system through Wi Fi or a cable. Finally, since the crystal operates at room temperatures, it can start up in 2 minutes, without need to cool first.

3. RESULTS AND DISCUSSION

In this section, we review a number of field measurements at nuclear power plants which highlight some of the applications of Polaris-H.

Fig. 2 shows the spectrum and images from a 2.6-minute measurement of a previously-known ^{60}Co source. The spectrum in Fig. 2a shows that ^{60}Co, ^{58}Co, ^{54}Mn, and ^{137}Cs are all present in this environment. The large continuum at lower energies is primarily continuum from the environment and is not all the result of the system response. Imaging just the ^{60}Co lines in the spectrum produces the image in FIG. 2c. This view shows all direction in the forward hemisphere, projected in equi-rectangular coordinates. Several hotspots are observed. The expected location has a source, but there is also a stronger source just to the left and another previously unknown source below the floor. The image of ^{58}Co (Fig. 2d) shows a hotspot in a similar location below the floor, suggesting that this last source is more recent than the ones near the ceiling in which the ^{58}Co component has already decayed. Note that the image quality will improve significantly in a longer measurement as more statistics are gathered.

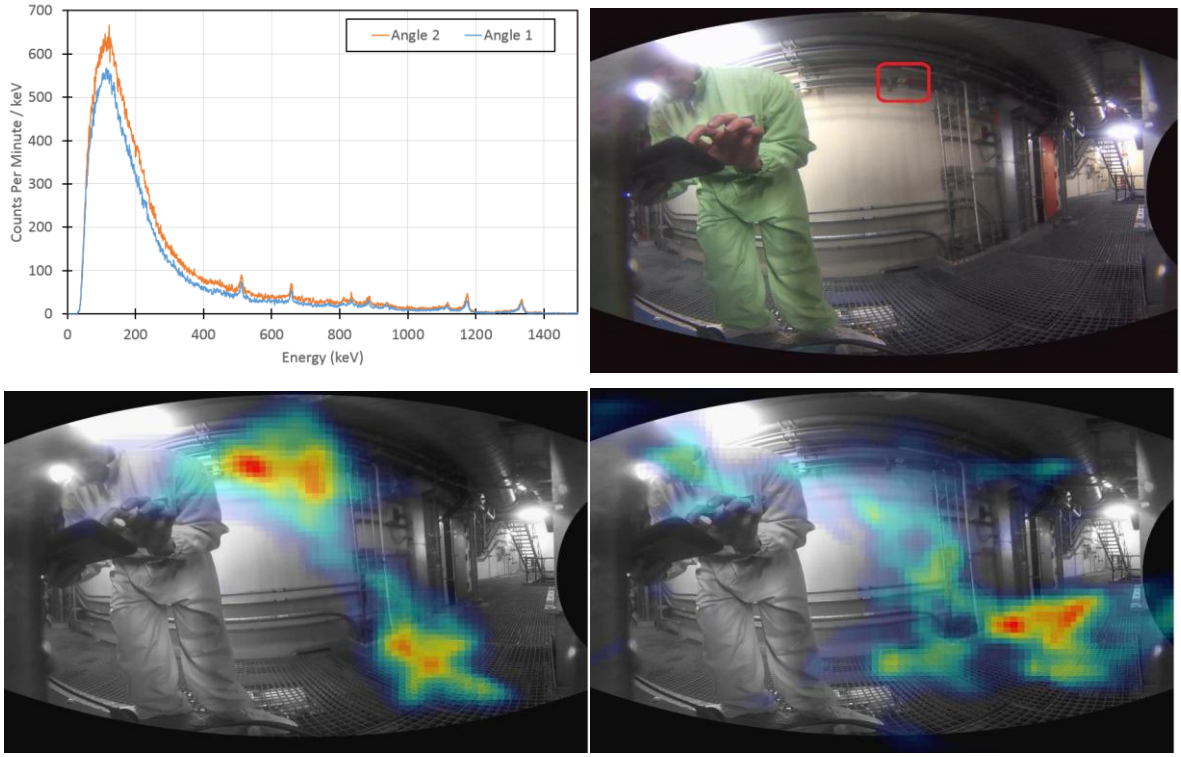

FIG. 12. From left to right and top to bottom: (a) The spectrum from two measurements at slightly different angles in this environment. (b) The optical image with the expected ^{60}Co source boxed in red. (c) The ^{60}Co gamma-ray image. (d) The ^{58}Co gamma-ray image.

This measurement is important from an occupational dose perspective because it was able to identify additional primary source terms which would have otherwise gone undetected. Indeed, survey meters were able to verify that the newly discovered source did have higher dose rate at the surface than the originally known source. In several of our measurements, we have discovered unknown sources like this, most often below the floor or above the ceiling. With this knowledge, better decisions can be made about taking preventive measures to reduce dose.

Another 2.6-minute measurement is shown in Fig. 3. Radioactive barrels were shielded at one end of the hall and this measurement was performed to verify that the shielding was

effective without any streaming paths in the direction of the detector. The image shows all directions (both hemispheres) again in an equi-rectangular projection. The spectrum in FIG. 3a has full-energy peaks from ^{60}Co and ^{54}Mn and a large continuum, possibly due to scatters in the barrel shield material. However, the image in Fig. 3c of the un-attenuated gamma rays shows that they are primarily coming from along a pipe along the ceiling. The pipe appears hotter when it is closer because more of those gamma rays reach the detector. Very few full-energy gamma rays appear to be coming from the direction of the shield, showing that it is fully covering the material well. This measurement shows the capability of Polaris-H to verify shielding is working as intended.

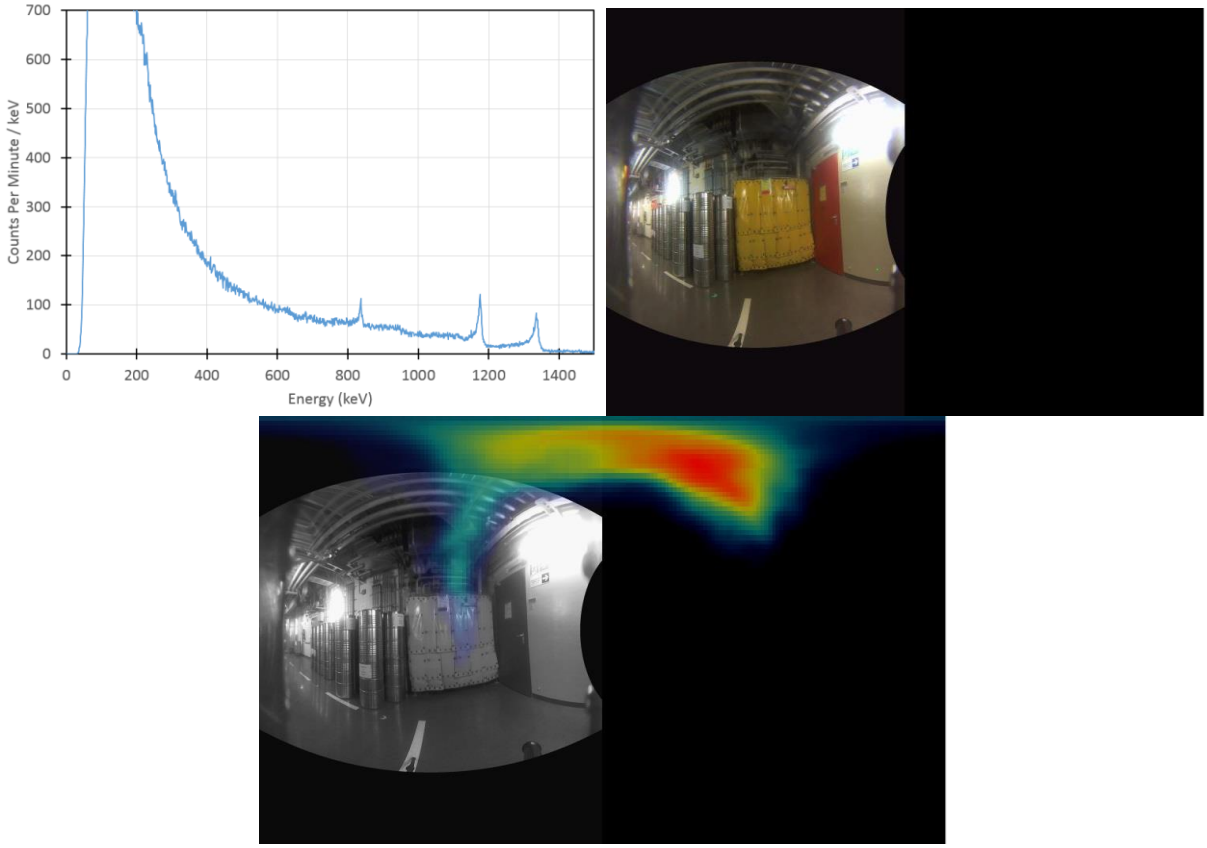

FIG. 13. From left to right and top to bottom: (a) The spectrum from a shielding-verification measurement. (b) The optical image. (c) An image of un-attenuated ^{60}Co and ^{54}Mn gamma rays

4. CONCLUSIONS

The measurements shown here are just a small sample of the applications of Polaris-H. The ability to achieve near 1% FWHM at 662 keV make it competitive with HPGe spectroscopy measurements, but in a much more portable form factor. Measurements which once required a collimated HPGe detector may be possible using less or no collimation due to the directional capability of this system, though this still needs to be tested experimentally. Because it can create an image of the hottest spots in all directions simultaneously, this detector can replace hand surveys of rooms, reducing dose to radiological-protection technicians.

Other plants have used Polaris-H to find discrete particle in clean areas, to locate isotopes in shipping containers as they leave or enter the facility, to locate and track crud in

pipes and valves, to verify clean-up progress, to determine the spatial extent of contamination, to characterize the isotopes in certain areas or parts, to identify sources in high-continuum environments, and to monitor the evolution of contamination from one cycle to the next. Other uses include emergency response and decommissioning. Future improvements and tests of Polaris-H are planned and ongoing.

REFERENCES

[1] NUCLEAR ENERGY AGENCY, ORGANIZATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, Radiation protection aspects of primary water chemistry and source-term management (2014).
[2] XU, D., HE, Z., LEHNER, C.E., ZHANG, F., 4π Compton imaging with single 3D position sensitive CdZnTe detector, Proc. of SPIE , 5540 (2004) 144.
[3] ZHANG, F., HE, Z., XU, D., Analysis of detector response using 3-D Position-Sensitive CZT Gamma-Ray spectrometers, IEEE Trans, Nuc. Sci. 51 (2004) 3098.

DEMONSTRATING ALARA ACROSS THE UK NUCLEAR INDUSTRY
IMPLEMENTATION OF THE CONTROL OF OCCUPATIONAL RADIATION EXPOSURE (CORE) REGULATORY INSPECTION PROJECT

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Abstract

In October 2013, the Office for Nuclear Regulation (ONR) initiated the Control of Occupational Radiation Exposure (CORE) project in order to assure itself and its stakeholders that doses incurred by workers in the UK nuclear industry are ALARA. The project requires the inspection of 37 nuclear sites across the UK within the three-year duration of the project, with the first year of the project nearing completion. A novel and consistent approach has been developed which is proving to be effective and efficient and is proportionate to the full range of nuclear facilities that require inspection. In addition to assessing the compliance of nuclear sites with relevant UK legislation, the project is also intended to identify any industry-wide themes that could be considered to be areas for improvement. A further significant objective of the project is to identify examples of good practice which will be communicated to site operators for them to consider adopting in their own arrangements.

1. INTRODUCTION

The UK's principal piece of legislation that sets out the requirements on nuclear operators with regard to the restriction of occupational exposure to ionising radiations is the Ionising Radiations Regulations 1999 [1]. The Regulations and its supporting Approved Code of Practice (ACOP) and guidance [2] establish a framework for ensuring that exposure to ionising radiation arising from work activities is kept as low as reasonably achievable (ALARA) and does not exceed dose limits specified for individuals. Additional guidance for nuclear sites is available in ONR's Nuclear Safety Technical Assessment Guide 038 [3].

As the UK's enforcing authority for the Regulations on nuclear sites, ONR's mission is to provide efficient and effective regulation of the nuclear industry, holding it to account on behalf of the public. ONR must assure itself and its stakeholders that nuclear operators are maintaining occupational exposures ALARA and so developed the CORE project in order to inspect arrangements across the whole industry.

2. APPROACH

The CORE project has three main objectives:

i) To provide assurance that licensees are broadly complying with legal requirements.

ii) To identify any industry wide themes that could be considered to be areas for improvement.

iii) To identify examples of good practice and communicate them back to nuclear operators.

The challenge faced by ONR was to develop an approach that could be effectively applied across the whole range of sites that it regulates, including power producing civil reactors, non power producing civil sites (such as those that produce and reprocess nuclear fuel), and defence sites (nuclear propulsion and weapons). This required an approach which is sufficiently high-level and generic that it can be applied at any site, but be sufficiently flexible in order to allow inspectors to focus in on specific topics of interest. It also needed to align with ONR's wider strategy for regulating nuclear sites and avoid causing a disproportionate burden on either ONR's resources or the resources of nuclear operators. The outcome of the project needed to be in the form of ratings and findings that will be simple to interpret and consolidate into a final summary report which will be published by ONR.

After a period of consultation within ONR and with external stakeholders, ONR finalized its CORE methodology. Eight CORE criteria were identified which ONR can use to rate the site operators' ALARA performance using a colour-coded and numbered rating system (as summarised in Table 1). The CORE criteria are described in Section 3.

TABLE 1. CORE RATING SYSTEM

	Rating
Purple	1. Exemplar
Blue	2. Good Standard
Green	3. Adequate
Yellow	4. Below Standard
Orange	5. Significantly Below Standard
Red	6. Unacceptable

Each inspection is divided into two elements: the first element uses a standard questionnaire to assess the general arrangements for ensuring that worker exposures are ALARA and the second element involve the inspection of a specific work area or activity in order to ensure that the arrangements are being applied at the point of work.

In order to ensure consistency between each inspection, a standard questionnaire was developed that requires information on the 8 CORE criteria and is provided to each site approximately one month before the site visit. The site is expected to return the completed questionnaire in advance of the site visit so that the inspector can review the responses. The inspector discusses the responses in the questionnaire on the first day of the site visit and will request clarification and supporting evidence on any relevant topic. The remainder of the site visit is then used to sample a particular work activity or work area in order to assess the adequacy of the implementation of the arrangements. The selection of the specific work activity or work area to be visited during the inspection is informed by each site operator's responses in the questionnaire and by ONR's knowledge of the site.

At the closing meeting for the inspection, the inspector assigns a rating for each of the criteria and informs the operator of the outcome, in addition to any good practices or areas for improvement that have been identified.

At the end of the CORE project, ratings for each site against the CORE criteria will be

consolidated into a single table. The report will highlight any good practices that have been identified, any industry wide themes that could be considered to be areas for improvement, and any regulatory actions that have been taken following the identification of a rating below adequate (i.e. 4-6).

3. CORE CRITERIA FOR ASSESSING ALARA PERFORMANCE

3.1. Criterion 1 – ALARA strategy

Nuclear site operators should have a clear ALARA policy and strategy set out in documentation. Consultation with the workforce and radiological protection specialists should take place when developing the strategy and it should have been communciated to employees in documentation, via training courses, etc.

3.2. Criterion 2 - Dose limits, dose targets / budgets / objectives / action levels

Nuclear site operators should have a range of numerical indicators that they use to assure themselves that worker doses are ALARA and that legal dose limits will not be exceeded. These indicators could take the form of action levels or investigation levels and may be associated with specific facilities, work activities, or groups of workers and they should prompt the operators to take action if exceeded. The operators should be able to explain their rationale for the setting of any dose action levels, since they are unlikely to be effective if they are set too high so that ALARA investigations are unlikely to be required.

3.3. Criterion 3 – Trending and analysis

Nuclear site operators should trend doses over time in order to demonstrate that they are ALARA and to identify any potential adverse patterns that require action. The number of data sets that are trended depends on the types and complexity of work activities that are undertaken on the site. The data should be regularly reviewed by competent experts and the findings from the reviews used to inform ALARA arrangements.

Nuclear site operators should review, and where appropriate trend, the findings from radiological surveys in order to identify any significant changes in the radiological conditions of work areas that might impact doses. The level of attention provided to this topic will depend upon the radiological hazard associated with the facility and the potential for the hazard to change.

3.4. Criterion 4 - Learning from experience/radiological incidents and near misses

Nuclear site operators should have arrangements for monitoring and recording radiological incidents and near misses and there should be a demonstrably positive culture amongst employees for reporting incidents. Incidents should be reviewed in order to identify potential common issues and actions required to address them. Where appropriate, operators should carry out reviews at the end of work activities in order to identify good practice and areas for improvement and there should be a process for applying the lessons learnt to future work. There should be robust arrangements for communicating good practices and areas for improvement to workers, preferably using a range of media.

3.5. Criterion 5 - Targeting of ALARA measures

Nuclear site operators should have identified work activities or groups of workers with a

higher radiological risk and targeted efforts at reducing that risk. This process should have been informed by adequate risk assessments. Operators should be able to provide examples of the control measures that have been employed to restrict the exposure associated with the highest risk activities, whilst avoiding placing undue reliance on dose sharing.

3.6. Criterion 6 – Work scheduling

Nuclear site operators should schedule work to ensure that doses are ALARA, including organising work during periods when radiological conditions are most favourable. The operators should also have a robust planning process in order to ensure that tasks that might impact on each other and present an increased radiological risk are scheduled in such a way as to reduce that risk.

3.7 Criterion 7 – Provision of information, instruction and training to workers on radiological Protection

Nuclear site operators should provide information, instruction and training to workers so that they understand the risks associated with work with ionising radiation and understand what measures are required to restrict their exposure. The operators should target this training based on the radiological risk of different workers and should record and track training requirements to ensure that refresher training is carried out at appropriate intervals.

3.8 Criterion 8 - Benchmarking and sharing of relevant good practice

Nuclear site operators should regularly benchmark their ALARA performance against other facilities in order to assess its adequacy and to share relevant good practice.

4. PRELIMINARY FINDINGS OF THE CORE PROJECT

ONR considers the first year of the CORE project to have been a success. So far, it has demonstrated that nuclear sites across the UK are broadly complying with relevant legislation, with generally high standards noted across the industry. The nuclear site operators themselves have been supportive of the initiative, welcoming the opportunity to have a full review of their arrangements to assure themselves that workers doses remain ALARA.

Some specific areas of good practice ONR has observed include:

i) The increasing use of ‘mock-up’ work environments to rehearse high dose tasks before they take place in the real work locations, thus ensuring that exposures are minimised.

ii) Effectively using electronic personal dosemetry systems to assess, trend, review and restrict exposure. It is increasingly common for these systems to be linked to access arrangements to radiation controlled areas in order restrict entry to those personnel who are authorized to enter.

iii) Utilising computer software to record and trend radiological survey data.

iv) The increasing use of innovative remote technologies, such as Remotely Operated Vehicles in fuel storage pools in order minimise exposure.

Most areas for improvement that have been identified are facility-specific, but generic topics include:

a. Ensuring that dose action/investigation levels are set appropriately. ONR considers that some operators have set them too high to be effective or they are not applied consistently.
b. In general, ONR considers that operators could improve their arrangements for performing post-task reviews of work activities in order to identify learning points.

ONR will continue inspections for the remaining two years of the CORE project. The ratings and findings will then be consolidated into a final report and communicated to the industry and stakeholders via a range of methods.

REFERENCES

[1] S.I. 1999 No. 3232, Ionising radiations regulations (1999).
[2] APPROVED CODE OF PRACTICE AND GUIDANCE, Work with Ionising Radiation, Ionising Radiations Regulations 1999, L121, HSE Books (2000).
[3] NS-TAST-GD-038, Issue 5, ONR, Nuclear Safety Technical Assessment Guide – Radiological Protection, (March 2014).

NUCLEAR POWER PLANT SOURCE TERM CHARACTERIZATION WITH CZT TECHNOLOGY

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Abstract

The Cook Nuclear Plant has extensively tested and used a new detector technology over the past several years to improve understanding of source terms and plant operations. In this paper, several measurements performed with this instrument to improve source-term understanding are reported. Several measurements performed with this instrument to improve source-term understanding are discussed. In addition, useful modes of operation with this technology are also reported. The technology is now available commercially.

1. INTRODUCTION

A thorough understanding of the source terms at nuclear power plants is critical in limiting occupational exposure. Better understanding of source terms most directly impacts shielding, work planning, and quick area clean-up to reduce dose imparted to workers. Less directly, but no less importantly, thorough understanding of source terms can engender changes in reactor operation, chemistry, and procedures, which can significantly reduce source terms systematically over longer time periods.

The Cook Nuclear Plant has extensively tested and used a new detector technology over the past several years to improve understanding of source terms and plant operations. In this paper, several measurements performed with this instrument to improve source-term understanding are reported. In addition, useful modes of operation with this technology are also reported.

Through detailed collaboration with the manufacturer, Cook has provided suggestions and feedback leading to packaging of this technology more amenable to use in nuclear facilities. Some of these lessons learned are also shared. The technology is now available commercially.

2. EQUIPMENT AND METHODS

Measurements were performed with prototype and commercial versions of Polaris-H, a CdZnTe (CZT) gamma-ray detector manufactured by H3D, Inc. Like other CZT detectors, Polaris-H operates at room temperature. That is, the detector does not need to be cooled before operation, so it is ready within a few minutes of turning it on. The H3D instrument uses 6 cm^3 of CZT for greater efficiency than other CZT modules. Most importantly, the readout method from H3D gives the detector better than 1.1% FWHM energy resolution at 662 keV and the ability to image the origin of gamma rays with energy above 250 keV over

all directions. Combined together, these features allow isotope-specific imaging. Further details of the detector are reported in reference [1].

3. EXAMPLE MEASUREMENTS

In this section, we present example measurements performed with the H3D detector. The measurement shown in FIG. 1 was taken in a hallway on the 587' level. This is the only radiation area on this level that is not located in a room. The hot spot on a valve in the waste disposal system (in the centre of the image) was known, and measured at 60 mR/h (0.6 mSv/h) on contact and at 5 mR/h (0.05 mSv/h) at 30 cm. The goal of the measurement was to better understand the source terms in this area. The detector was placed on a tripod in this area and collected data for 11.5 minutes.

FIG. 1 shows the spectrum recorded. Immediately we see ^{137}Cs and ^{60}Co are present from their characteristic gamma-ray energies. There is also a significant amount of scattered gamma rays (more than expected from the system response). The images in FIG. 1b and Fig. 1c represent all directions around the detector, like a map view of the world. An optical camera in the front shows reference points for sources in front of the detector. Looking only at (un-scattered) ^{60}Co gamma rays in Fig. 1b, we see a single hot spot in the direction of the known hot spot on the valve. Other sources of ^{60}Co must be significantly weaker than this spot, as viewed from the detector location. Looking only at (un-scattered) ^{137}Cs gamma rays, we see several hot spots. First, there is a localized spot in the same valve as the ^{60}Co. But, there is also a hot spot of comparable intensity on the floor slightly behind the detector (red spot on right side of image). Further investigation showed that this was ^{137}Cs contamination painted into the floor. The light blue spots on the floor also hint of hot particles painted into the floor, but without more statistics in the image (a longer measurement), it is inconclusive. Nevertheless, follow up with a dose probe located previously unknown hot particles of ^{137}Cs in the floor.

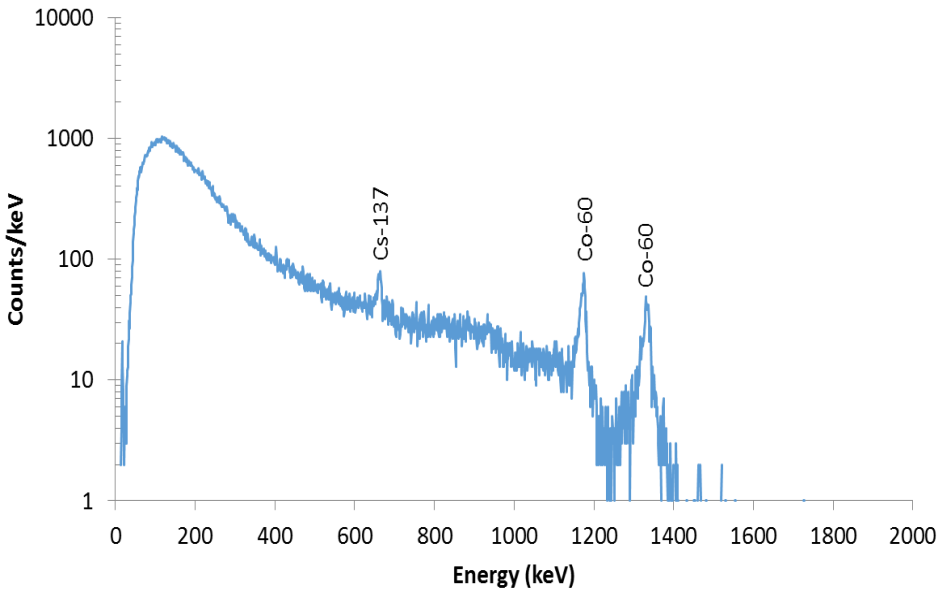

FIG. 1(a). *The spectrum from this measurement of a radiation area in a hallway;*

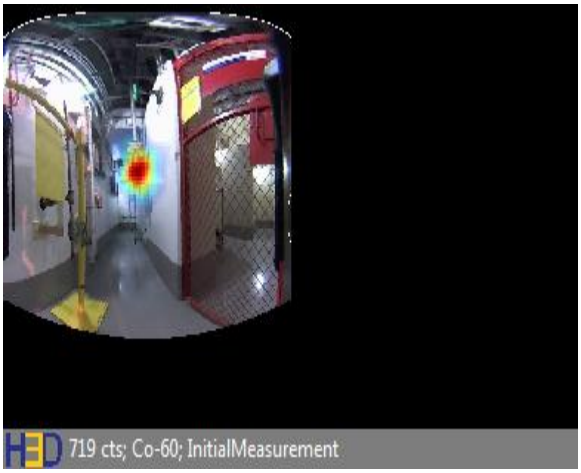

FIG. 1(b).The image of ^{60}Co gamma rays

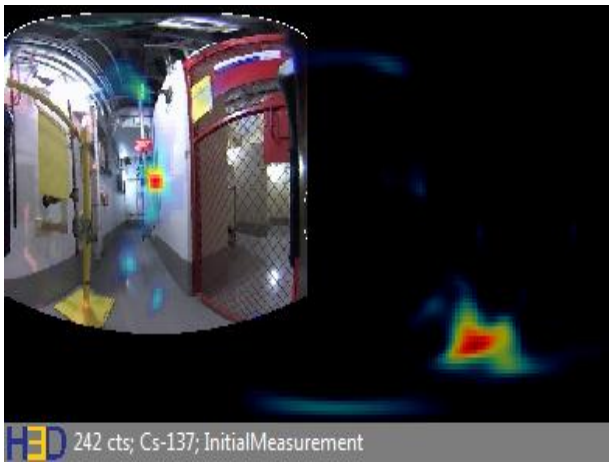

FIG. 1(c). The image from ^{137}Cs gamma rays

A simpler measurement is shown in FIG. 2. A stairway was slightly above background and a 15-minute measurement was made with the H3D detector to determine the primary source terms. The spectrum in FIG. 2(a) shows weak ^{60}Co present in the stairway along with background continuum. The high energy resolution of the spectrum helps make the ^{60}Co peaks detectable. The image of ^{60}Co, Fig. 2(b), shows an obvious hot spot near some pipe bends. The image has 115 counts, which is close to the minimum number required to form a trustworthy image, but in this case, the image quality is already quite good because the source is concentrated. To further identify the source of the ^{60}Co, a computational method supplied by H3D is used to produce the high-resolution image in Fig. 2(c). Here it is easy to see exactly which pipe is the cause of hot spot.

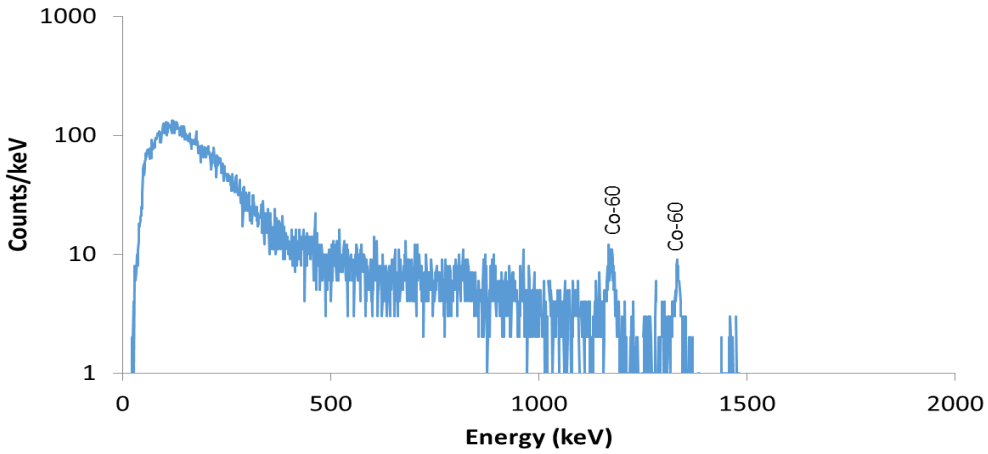

Fig. 2 (a). Shows weak ^{60}Co present in the stairway along with background continuum.

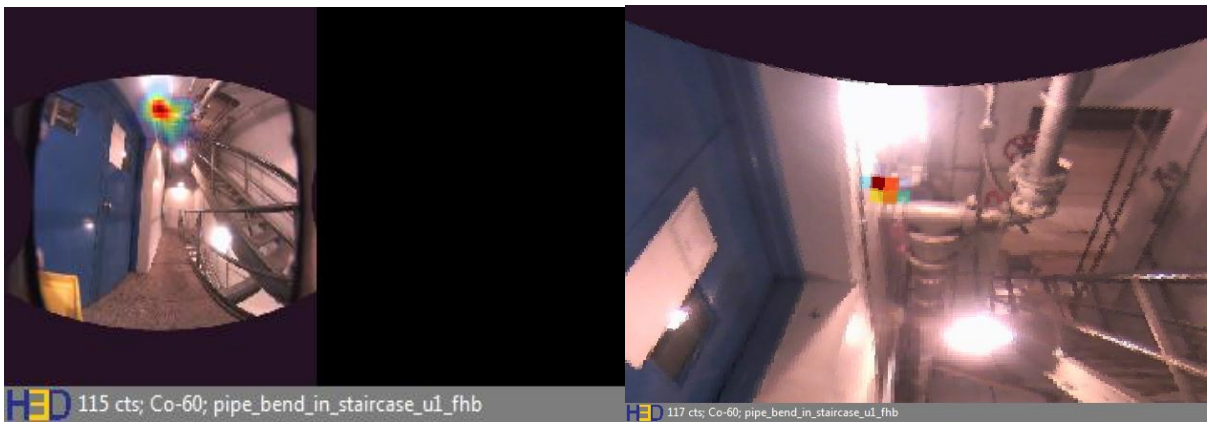

FIG. 2 (b) and 2(c). From left to right: (b) The image of ^{60}Co gamma rays; (c) The high-resolution image of ^{60}Co gamma-rays zoomed into the pipe area.

Finally, a third measurement is shown in (Fig. 3). In this measurement, the East Residual Heat Removal Heat Exchanger was imaged pre-outage to determine source terms for shielding requirements during outage work. It was also used in a measurement campaign to understand gamma-emitter transients in the cooling system during outages. The spectrum recorded over 10.5 minutes with the H3D detector is shown in FIG. 3a. Sb-124, Cs-137, and Co-60 peaks are observed in the spectrum. It was originally expected that the hottest spots would be in the pipes and valves in this room, but the image in FIG. 3b shows that the hottest spot (for un-attenuated gamma rays) is under the floor. This changed how shielding was employed, as only the floor needed to be shielded and not pipes and valves for the greatest reduction in dose. Later in the outage, the pattern of contamination was observed to change, and the hottest spot was from a nearby pipe.

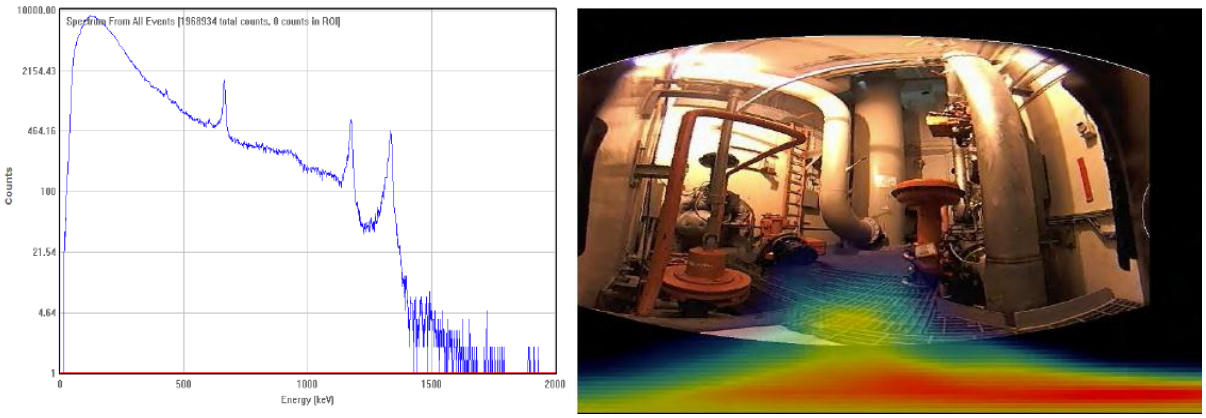

FIG. 3. From left to right: (a) The spectrum of this measurement of the residual heat removal heat exchanger; (b) The image of ^{60}Co gamma rays. Images of ^{137}Cs and ^{124}Sb look similar.

4. DISCUSSION AND MODES OF OPERATION

The measurements in the previous section point to several useful modes of operation for detectors that can image on an isotope-by-isotope basis.

First, such detectors are useful for simply locating the direction or object that is the hottest from the detector's location so it can be cleaned up or shielded. As we have seen in the example measurements, we may sometimes be surprised at the objects that are producing the most un-attenuated gamma rays. By pin-pointing the location of maximum contamination, it can be shielded with minimal shield material. For instance, instead of shielding the entire pipe in Fig. 2, only the hot elbow needs to be covered.

The shield can also be verified using the H3D imaging detector. Imaging with the shield in place will tell if the hottest direction is still the direction of the shielded source or another direction from a weaker source. Though scattered gamma rays can still be coming from the shield, when un-attenuated gamma rays from another source are stronger than from the shielded source, further shielding the shielded source produces diminishing returns.

Another useful way of thinking about the output of the H3D imager is as a way to fill in the gaps of traditional radiological surveys. Traditional surveys map the dose rate at specific locations in the room so that in work planning, lower-dose regions can be identified. But, because the surveys only emphasize frequently used area, there are inherent gaps in the survey areas. By replacing survey locations with images of all directions from selected areas, the source of the dose is more apparent, allowing health physicists and radiation workers to visualize the origins of the radiation. This fills in the gaps of traditional surveys as users intuitively understand which areas need to be avoided since they understand the source of the dose.

Although beyond the scope of this paper, another way of using an imaging spectrometer like the H3D instrument is in interpreting the contamination distribution in terms of plant chemistry and operations. By measuring the flow of each isotope in the last example measurement during an outage, insights into plant chemistry can be gained that are valuable in reducing dose on a larger scale throughout the plant.

5. TECHNOLOGY PACKAGING

Through the process of testing the prototype system and giving feedback on the packaging of this technology, the packaging of this CZT detector was improved for use in nuclear facilities. While the technology of isotope-specific imaging is important, it also must have other features to make it realistic to use in our plant.

Most significantly through the development process, the system became water and air tight. The original prototype unit could only be used in locations without loose contamination because a fan blew air though the unit. In the final commercial unit, the detector is enclosed in an air-tight housing with smooth aluminium sides. A fan blows over outside fins, but even when bringing it into contaminated areas, contamination did not stick to the unit.

A second feature developed over the technology-development processes was to have a wireless connection to the control tablet. On the original unit, the output was viewed on a screen attached to the unit, however in high-dose environments, it did not satisfy ALARA to stand by the unit during measurements. The commercial product has a tablet to view real-time images and spectra that connects through bluetooth or Wi-fi with the unit, allowing the operator to stand behind a wall and still monitor the measurement progress.

Other developments in the technology package include attachment methods for operation above pools, a tripod mount, and more user-friendly control interface.

6. CONCLUSIONS

A new technology capable of imaging isotope-by-isotope with a portable detector has been tested extensively at the Cook Nuclear Plant. Several representative measurements were shown to demonstrate some of the capabilities and limitations of this technology. The system was able to locate and identify previously unknown sources of gamma rays and to pin-point the location and isotopes present in previously known hot spots. These measurements point to suggest modes of operation to characterize sources, find primary source terms, verify shielding, fill in traditional survey maps, and understand plant chemistry. Finally, the significant detector features are stated to be more useful in realistic situations, including being air-tight and being able to monitor measurements from behind walls.

ACKNOWLEDGEMENT

The authors are thankful to a collaborator, Bennett Williams, PHD Candidate, Nuclear Engineering Department, University of Michigan, Ann Arbor, Michigan, USA.

REFERENCES

[1] WAHL, C.G., KAYE, W.R., WANG, W., ZHANG, F., JAWORSKI, J., KING, A., BOUCHER, Y.A. HE, Z., The Polaris-H Imaging Spectrometer, Nuc. Instrum. Meth. A, Proc. SORMA XV (2014), In Press.

Session 11: Education and training in occupational radiation protection

BUILDING COMPETENCE OF RADIATION PROTECTION OFFICERS IN AN INDUSTRY USING GAUGING SYSTEMS: CASE OF BURKINA FASO

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Abstract

In Burkina Faso, radioactive sources are used in several fields such as industry, health, mining, research, agriculture and civil engineering. The National Radiation Protection and Nuclear Safety Authority (ARSN) has established a legislative framework for protecting people and the environment against radiation risks. So, it has implemented an annual programme of training of radiation protection officers to meet the compliance of the law 032-2012/AN on nuclear safety and security and safeguards, and to build competence in radiation protection and the safe use of radioactive sources.

1. INTRODUCTION

In Burkina Faso, there is not power or research nuclear reactor and not other nuclear facility. But many kind of radioactive sources are used such as:

(a) one irradiator with four category I sources of ^{137}Cs;
(b) more than 46 nuclear gauges of category II, III and IV;
(c) a new LINAC irradiator of research is ready to use;
(d) two category I sources of ^{60}Co (555TBq) for irradiation are expected for November 2014;
(e) two radiotherapy services including LINAC and Brachytherapy are currently in process of construction.

An educational and training framework of Radiation protection officers is established by ARSN in response of the needs of radiation protection and technical requirements for practices and to fill the lack of national infrastructure of protection and nuclear safety training.

2. LEGISLATIVE FRAMEWORK

The National Radiation Protection and Nuclear Safety Authority is the competent authority for safety and security of nuclear and other radioactive material. Under the law N°032-2012/AN, ARSN is responsible for regulation, licensing and inspection at national level of all activities involving nuclear energy. It's also responsible of drawing and implementing policies and strategies of safety culture [1].

Through the above law, registrants and licensees and employers of workers who are engaged in activities involving normal exposures shall identify a Radiation protection officer who is responsible of operational activities of radiological safety and must have a higher competence and capabilities in radiation protection.

3. DESIGN OF THE TRAININIG OF RADIATION PROTECTION OFFICERS PROGRAMME

3.1. Purpose of the training

The purpose of the training of radioactive officers is to meet the compliance of radiation protection and technical requirements for activities involving normal radioactive exposures [2]. This training intended also to build competence in radiation protection and the safe use of radioactive sources for responsibles of operational activities of radiological safety.

3.2. Public target

The target audience is:

(a) Personnel with specific responsibilities or functions in radiation protection in industrial or mining facilities;
(b) Operators who have the responsibility for the day-to-day use of radiation sources;
(c) Emergency response personnel, radiation sources maintenance personnel.

The participants should have secondary education with a technical or scientific background.

3.3. Training team

ARSN has implemented a staff focused for managing the training programme which is responsible for defining the training needs and the topics of the course, establishing the terms of reference, the schedule and the programme, establishing the tools of assessment of the training and the selection of trainers according to their skills and the topics. This team should produce the training material and all arrangement for the training, which might include theoretical and practical training. [3]. The team is managed by a coordinator who is under the responsibility of the National Director of ARSN.

3.4. Topics of the course

The topics of the course include: Basic nuclear physics; Quantities and measurements; Sources of radiation exposure; Hazards and biological effects of ionizing radiation; Radiation protection and nuclear safety; Radiation protection officer: role and duties; Legislative and regulatory framework; Emergency preparedness and response.

3.5. Approach for selection of participants

he coordinator of the training team informs all registrants and licensees and employers of workers who are engaged in activities involving radioactive sources, the due and the programme of the up coming training of Radiation protection officers. It's always an opportunity to call for participation and to sensitize the licensees about its responsibilities for establishing and maintaining the necessary competences with regard to safety [4].

3.6. Approach for the selection of trainers

The selection of trainers is decided within the staff focused for managing the training programmes. Trainers are chosen according the topic, their experience and their technical and educational competences. Thus, some teachers from the Departement of Physics of the University of Ouagadougou are chosen to teach the physics of ionizing radiation and the trainers of biological effects are from the Departement of Medicine and Biology. So, the topics related to protection and safety, the use of radiation detectors, the role of a Radiation protection officer and regulatory framework are provided by the experts of ARSN. The

Emergency preparedness and response topic is ensured by the experts of Protection civile and the regulatory authority body.

3.7. Nature of training

The training consists of lectures, demonstrations and practical exercises, working groups on elaboration and assessment of radiation protection programme, technical visits in the field of safe use of radiation sources and a part of assessment of trainees. The duration of the course is five days from monday to friday through eight hour per day. At the end of the course, all the participant who will succeed the final assessment received a certificate valid for two years. Those who don't succeed, a completion of the training is formally recognized for them to perform the weakness.

3.8. Role of Radiation protection officers on site

A Radiation protection officer is "An individual technically competent in radiation protection matters relevant for a given type of practice who is designated by the registrant or licensee to oversee the application of the requirements of the Standards" (Ref. [2], Glossary). On the site, he is responsible for establishing and assessment of the programme of radiation safety within radiological facilities and to ensure that work is carried out safely and in accordance with the relevant national requirements. He ensures the links between the workplace, the registrant or licensee and the regulatory body, and makes sure that operations involving radiation are in compliance with established regulations.

The radiation protection officer is also responsible for organizing training of workers on hazards of ionizing radiation, radiation protection and the safety use of radioactive sources [5]. He has to be officially designated by the licensee for his function and the licensee must send this letter of nomination to ARSN.

4. RESULTS OF THE TRAINING

Since 2010, 57 Radiation protection officers have been trained for 10 institutes of industry, research and mining. The training is highly appreciated by the licensee and its strategies are currently to provide this training to all the users and managers of radioactive sources, the responsibles of emergency preparedness programme and radiation sources maintenance personnel. Every year, about dozen of stakeholders are trained either for initial or for refresher training.

5. KEY LESSONS LEARNED TO THE TRAINING AND PERSPECTIVES

5.1. Key lessons

The training of Radiation protection officers programme has provided skills on protection and safety on the site for stakeholders. An assessment of the effectiveness of training is carried out by the annual inspection programme of ARSN. There is a better organization of the radiation protection programme in radiological facilities through: classification and signalization of areas, dose rate monitoring, health surveillance, establishing of local rules and local emergency response plan, responsibilities of radiation protection, implementation of training and sensibilization programmes for stakeholders on hazards and protection against ionizing radiation, safety and security use of storage sources, improving the safety culture on the day-to-day use of radiation sources.

This training programme has also provided a better collaboration between licensee and the regulatory authority body about authorization, inspection and notification of useful information.

5.2. perspectives

In collaboration with the International Atomic Energy Agency (IAEA), Burkina Faso is in process to establish a national strategy of educational and training programme in radioprotection and safety. A national committee which includes representatives of the national stakeholders has been created to analysis the training needs and design a training programme. In this case, the University of Ouagadougou will implement soon, a Master degree programme of training on Radiation protection. According to the responsibilities of the regulatory body (Ref [3], par. 2.8), ARSN will support this programme for building practical competence and skills on protection, safety and security of radioactive sources.

6. CONCLUSION

Radioactive sources are more and more used in many fields of application in Burkina Faso. In order to protect people against the risks associated with exposure to ionizing radiation, strengthening competence on protection and safety is strongly important.

The training programmes of Radiation protection officers implemented by the regulatory authority body come over the needs of the compliance of radiation protection and the reglementary requirements. The purpose of these training is highly appreciated.

However, in order to assume fully its responsibilities of regulatory body, ARSN is in process to recognize and to accredit some national training centers and courses for the strengthening of capabilities in radiation protection and the safety of radiation sources. It must be in a reglementary framework and through the results of a pluridisciplinary committee working groups assigned for that.

REFERENCES

[1] LAW N°032-2012/AN of 8 June 2012 on nuclear safety and security and safeguards (2012).
[2] INTERNATIONAL ATOMIC ENERGY AGNECY, International Basic Safety Standards for protection against ionizing radiation and for the safety of radiation sources, Safety Series No. 115, IAEA, Vienna (1996).
[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Standards Series, Building competence in radiation protection and the safe use of radiation sources, RS-G-1.4, IAEA, Vienna (2001).
[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Fundamental Safety Principles, SF-1, IAEA, Vienna (2006).

THE SYLLABUS ON FUNDAMENTAL RADIATION PROTECTION COURSES FOR RADIOLOGISTS IN TURKEY, (2011-2013)

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Abstract

The aim of this paper is to describe the Turkish experience in radiation protection (RP) and communicate the experience in the organization of RP training courses for radiologists. The courses provided theoretical and practical knowledge on RP on the basis of scientific and technical recommendations from The International Commission on Radiological Protection (ICRP), The Council Directive 97/43/EURATOM and the International Atomic Energy Agency (IAEA). Courses were organized by the Ankara Nuclear Research and Training Center (ANAEM), a branch of Turkish Atomic Energy Agency aimed at providing specialized RP education for radiologists, medical doctors, technicians, physicians, and radiation workers. The courses were presented as five-day seminars by a group of instructors composed of a physicist, radiation biologists, RP experts, and medical physicists. The implemented RP training programs were effective and efficient according to the overall assessments obtained by the ANAEM. Similar training programs can be organized for cardiologists and other medical practitioners conducting interventional procedures by institutions and organizations in accordance with the Turkish RP regulations. To this end, it is suggested that RP training courses should be accredited and sustained at the national level.

1. INTRODUCTION

Training in radiological protection (RP) is widely recognized as one of the basic components of radiological optimization programs for reducing medical exposures. Occupational and patient radiological risks during procedures used in interventional radiology can be quite high. Consequently, continuous education programs regarding RP are of paramount importance to ensure that the radiation doses to which patients and staff exposed are minimized as much as possible.

Radiology utilizes minimally invasive image-guided procedures to diagnose and treat some diseases in the organ system. The focus of radiology is to diagnose and treat patients while using the least invasive techniques currently available to minimize patient risk and obtain favorable health outcomes. Fluoroscopy is extensively used to conduct and document interventional radiology procedures. Professionals and patients in this medical field can encounter significant radiological risks. Radiological protection, therefore, is an important issue in the design of X-ray rooms, the selection and purchase of equipment, and in routine practice and training.

The International Commission on Radiological Protection (ICRP) published general recommendations for RP and radiation safety for interventional radiology [1]. The ICRP recommends that interventional radiologists receive training in RP for their patients, their staff, and themselves and that this training should include guidance on how to properly use personal protective equipment and monitor exposure.

The International Atomic Energy Agency (IAEA) and the World Health Organization (WHO) have also made relevant efforts to promote radiation safety in radiology. Their main publications on this topic include recommendations for areas of training in both medical aspects and RP [2-5]. The WHO published one of the first and most complete publications with detailed orientations on the different levels of training requirements (basic, intermediate, advanced) for different topics and personnel involved in radiology procedures [2].The IAEA also published the Standard Syllabus Training Course Series, which serves as a guide for standardized curricula with postgraduate educational courses in radiation protection and the safe use of radiation sources [5].

European Council Directive 97/43/EURATOM [6] established in Article 7 that "Member States shall ensure that practitioners and those individuals mentioned in Articles 5 [3] and 6 [3] [medical physics experts and other staff members involved in radiological practices] have adequate theoretical and practical training for the purpose of radiological practices, as well as relevant competence in RP. For this purpose, Member States shall ensure that appropriate curricula are established and shall recognize the corresponding diplomas, certificates or formal qualifications." In addition, according to Article 9.2 in the Council Directive, appropriate training should be given to practitioners and other staff members performing radiological practices for the medical exposure involving high doses to the patient such as interventional radiology, computed tomography or radiotherapy. The European Commission's published guideline commends education and training in radiation protection for medical exposures and includes specific recommendations for interventional radiology]7]. Additionally, the European Guideline emphasizes 20 to 30 hours of training as well as the accreditation process. The European Association of Radiology (EAR) further defines the "core of knowledge for general radiology" syllabus in its recommendations [8].

In Turkey, national legislation requires specific training in RP (Official Journal of Turkish Government, 2000), and the ANAEM affiliated with the Turkish Atomic Energy Agency (TAEA) decided to organize and coordinate a pilot course to implement training for RP in the country. The duties and responsibilities of the ANAEM include national education, training, and research on topics of RP, such as radiation safety, nuclear power, nuclear safety, and the application of nuclear energy and radiation.

In this report, we present the Turkish experience in RP training for radiology. We describe how RP training for interventional radiologists is organized and identify the primary components of a syllabus for Turkish RP courses. The general learning objectives, main content, and duration of RP courses are summarized.

2. METHODS

2.1. Legislation and regulatory requirements in Turkey

The TAEA regulates and supervises all topics in RP. The TAEA was reformed by the new Act (No: 2690, year 1982); with its main objective to set policy for nuclear energy, nuclear regulation, licensing, inspection, as well as for nuclear research, training and development. The TAEA is the established authority for radiation protection and nuclear safety [10]. The Department of Radiation Health and Safety (RSGD) is affiliated with the TAEA and has duties and responsibilities in implementing tasks [established by the Turkish Atomic Energy Authority Law (No: 2690, Article: 8-b)] concerning the services related to the licensing, regulation, transportation and storage of radioactive materials, inspection of instruments and systems connected with radiation, and other related duties [9]. The protection and safety objectives of the TAEA contained in the National Regulations of Radiation Security are consistent with IAEA-The Basic Safety Standards (BSS) and ICRP recommendations [1,3].

In accordance with Turkish Legislation, every public and private foundation/establishment that uses exposure devices or sources for all practices and activities must be registered and licensed by the TAEA-RSGD [9]. According to the National Regulations of Radiation Security [9], training and experience are key elements in the first phase of issuing a license to any public or private foundation/establishment that uses exposure devices/sources in their activities, and employers, registrants, licensees, and workers should have suitable appropriate and periodic education and training in RP and radiation safety.

The National Regulation of Radiation Safety, in Article 4-i, (No: 23999) [9] defines an RP officer and his/her required qualifications. An RP officer must be a qualified person trained in the basic safety standards for RP and who implements these standards according to the characteristic requirements of his/her job. This individual must also have a proven track of record of accomplishment in this field that has been approved by the TAEA.

To apply for a license, one must first be a qualified RP officer who has successfully completed the appropriate training courses in his/her field. This requirement is spelled out in Regulation Article 73 of the National Regulations of Radiation Security, which states that any facility dealing with radiation, including radiation sources, shall have an RP officer certified by the TAEA.

Recently, the TAEA issued "The Procedures and Principles for Implementation of Trainings in Radiation Protection" on May 3, 2013 [11]. According to this new regulation, public and private enterprises can also conduct radiation protection training/courses provided that they first be approved and licensed by the TAEA. Articles pertaining to training objectives, training topics, selection criteria for lecturers, and technical infrastructures were also defined in this regulation [11].

2.2. Ankara Nuclear Research and Training Center (ANAEM)

Education and training in RP for radiology must meet the needs of professionals at the graduate level or the equivalent. Initial training should establish a basis in RP and the safe uses of radiation sources and equipment. The training courses are designed to provide the necessary basic tools for those who plan to become RP trainers in Turkey. In course planning, recommendations from The ICRP, The Council Directive 97/43/EURATOM and standards on radiation protection and their implementations [2,6] are taken into account so that both theoretical and practical sessions in scientific and technical bases are allocated. Council Directive 97/43/EURATOM on medical exposures highlighted these recommendations in Europe. This directive considers interventional radiology (Article 9) to be a special practice that involves high doses to the patient.

Interventional practices imply additional risks, due to more complex procedures and the requirement for more operator time near the patient during the fluoroscopic screening, and also carry the risk of deterministic effects on the patient [8]. For this reason, RP training of interventional radiologists is considered important to help minimize these risks.

The planning and implementing of these courses requires oversight by a group formed by experts working in the field, called the Ankara Nuclear Research and Training Center (ANAEM). The ANAEM was established on 18.08.2010 upon the Council of Ministers' Decision and published on 30.09.2009 in the Official Gazette with the legislation number [12]. The ANAEM conducts various radiation protection and pilot apprenticeship training courses focusing on the specialized training of radiologists, medical doctors, technicians, physicians and radiation workers and rates the outcomes of these activities by known teaching evaluation methods.

These specialized courses are tailored to the needs of the public and private sectors and are thus called "sector-specific courses." RP training for interventional radiologists represents an example of one of these courses that is periodically held by the ANAEM during the year.

The course programs are announced before the beginning of the year on the TAEA website. The training courses, held by the ANAEM itself or in collaboration with another appropriate training center affiliated with TAEA, are conducted by qualified instructors working in the field. These instructors are selected from the TAEA's own instructor repository or from universities that have been working in parallel with the targeted activities.

The main duty of the ANAEM is to meet the qualified manpower needs for all practices and activities involving the use of ionizing radiation in Turkey. The divisions and duties of the ANAEM are as follows:

(i) The Training Administration Division is responsible for establishing efficient, productive, standardized and systematic national and international trainings and to ensure that these trainings are conducted by qualified trainers. In addition, this division develops teaching materials and methods, maintaining and updating them according to current needs. This division cooperates with other national institutions as needed.

(ii) The Training Application Division organizes trainings on radiation protection, radiation safety, nuclear power, nuclear safety, nuclear assurance, nuclear technology, and related subjects. The division is responsible for implementing the Annual National Education Program approved by the TAEA administration.

(iii) The Public Information Division arranges seminars, workshops, and formal meetings, preparing and updating written and visual materials, to disseminate effective and transparent public information for people or groups in the range of interest.

2.3. Training objectives and course syllabus in radiation protection for interventional radiologists

The EAR has presented a syllabus for radiation protection on "diagnostic imaging-physical and biological aspects," prepared by its working group [8]. This syllabus contains 40 hours of theoretical education, supplemented by demonstrations to fulfill the demands. The standard syllabus along with the postgraduate educational course syllabus in radiation protection and the safe use of radiation sources were also provided by the Standard Syllabus Training Course Series of the IAEA [5].

From the point of view of the ANAEM, the syllabus for radiation protection and the safety of radiation sources and RP equipment is to be integrated into the curricula of educational institutions in Turkey to promote consistency and create a standardized level of technical content in such courses. The focus of the courses on technical and administrative frameworks is necessary for regulatory and operational controls to protect against ionizing radiation in radiological applications. The course syllabus on radiological protection for radiologists in Turkey has been implemented according to the following content in the final course programme of the ANAEM:

Table 1 summarizes the general learning objectives and durations of the RP courses. Table 2 shows the percentage of total course time devoted to each module in the course syllabus on radiological protection for radiologists implemented by the final course program of the ANAEM.

2.4. Implementation of radiation protection courses for radiologists

The course programs were announced on the TAEA website at the beginning of each year. The courses were presented to the participants as five-day seminars (total 40 hours) by a

group of instructors comprising physicists, radiation protection experts, radiation biologists, and medical physicists.

3. RESULTS

To date, over fourteen RP courses for radiologists have been organized by the ANAEM, and 314 participants participated between 2011 and 2013. About 100 participants were interventional radiologists. This activity continued into 2014. The appropriate training material and arrangements were used in both theoretical and applied sessions. Applied sessions primarily took place in a private hospital's radiology and fluoroscopy laboratory and in the medical physics laboratory of an institute that was participating under an agreement with the TAEA. This facility was informed and provided with official letters requesting appropriate time availability in advance prior to the beginning of each course. In the applied sessions, analyses of patient and staff doses for the fluoroscopy and cine acquisition modes were conducted, and differences in image quality and dose values were highlighted.

The participants were monitored with thermoluminescent dosimeters (TLDs) during on-the-job training by the National Dosimetry Service [13,14]. The TLDs were based on the Panasonic model UD-802 dosimeter. The UD-802 model TLD consists of two different phosphor types ($Li_2B_4O_7$:Cu and $CaSO_4$:Tm), which when using the appropriate holder can measure the dosimetry of photons and beta radiation. The minimum detectable dose level (MDL) was 0.1 mSv per our national dosimetry service. Radiation workers who had effective doses less than 0.1 mSv (MDL) were considered as non-exposed. Therefore, doses falling below the MDL were recorded as zero; the effective dose exposure for all participants was less than 1 mSv during the course periods.

During each course, revised training/course notes were prepared and disseminated to the course participants for future reference. At end of the each training course, participants completed a test. Based on this test score, one of two different certificates was issued. If the examination score was 70 or above out of 100, a "Certificate of Achievement" was issued; for lower scores, a "Certificate of Attendance" was given. 312 participants completed the examination with a high score and received a "Certificate of Achievement" from the TAEA between 2011 and 2013.

Course evaluation questionnaires were completed by the participants at the end of each course. Table 3 contains participants' ratings of the training courses, evaluating their instructors within their own assigned syllabus content on a five-point scale. The questionnaires primarily documented the best- and worst-rated aspects of the training courses for the years 2011 and 2013 (Table 4). Participant feedback on the quality and effectiveness of the training was taken into account in the improvement of the next set of courses. At the same time, feedback from the inspectors of the RSGD affiliated with the TAEA was also incorporated.

4. DISCUSSION

In this report we briefly described the current regulations and requirements for RP in Turkey and shared our experience in the organization of RP training courses for radiologists.

The current RP training programs appear quite beneficial and useful in fulfilling the intended purpose of promoting the protection of patients and staff from the harmful effects of radiation. Educational programs in radiation physics, radiobiology, and RP are currently well defined. The guidelines for establishing the content in educational courses have been designated through collaboration among organizations such as the EAR and the European Federation of Organizations of Medical Physics. Combined training programs as well as on-the-job training programs were found to be effective and efficient. Overall, our findings

suggest that RP education and training for diagnostic and interventional radiology should be implemented at all levels of education, including continuous training for practitioners with professional experience. This training is usually welcomed as long as it is seen as fostering better long-term implementation of RP rules.

5. CONCLUSION

In conclusion, in Turkey, the relevant authorities have been meeting the radiological training needs for RP. Similar training courses for cardiologists and other medical practitioners conducting interventional procedures can also be organized by institutions and organizations in accordance with the related Turkish regulations. To that end, RP training programs should be accredited and sustained at the national level. This process should be undertaken by the regulatory authority with the help of the ANAEM. Indeed, the TAEA has already begun this process, which is expected to conclude in a year. Therefore, it is hoped that other scientific or professional societies and academic institutions (universities), those that have similar experiences in their field of activity, can organize and implement RP education and training courses providing that they sufficiently meet the criteria determined by regulatory authorities.

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REFERENCES

[1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Avoidance of Radiation Injures From Medical Interventional Procedures, Publication 85, Ann. ICRP 30(2) (2000).
[2] WORLD HEALT ORGANISAZION, Efficacy and Radiation Safety in Interventional Radiology, Geneva (2000).
[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards--Interim Edition. General safety requirements No. GSR Part 3, Vienna (2011).
[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Training in Radiation Protection and the Safe Use of Radiation Sources. Safety report series No. 20, Vienna (2001).
[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Postgraduate Educational Course in Radiation Protection and the Safe Use of Radiation Sources. Standard Syllabus. Training course series No. 18, , Vienna (2002).
[6] THE COUNCIL OF THE EUROPEAN UNION, Council Directive 97/43 Euratom, On Health Protection of Individuals Against The Dangers of Ionizing Radiation in Relation to Medical Exposure, And Repealing Directive 84/466 Euratom, Official Journal of the European Communities No L 180, July 9 (1997) 22–27.
[7] EUROPEAN COMMISSION, Guidelines on Education, Training In Radiation Protection for Medical Exposures, Radiation Protection 116, Directorate General Environment, Nuclear Safety and Civil protection (2000).http://ec.europa.eu/energy/nuclear/radiation_protection/medical/publications_en.htm. Accessed on December 28, 2012.
[8] EUROPEAN ASSOCIATION OF RADIOLOGY, European Training Charter for Clinical Radiology, (2005).
[9] OFFICIAL JOURNAL OF THE TURKISH GOVERNMENT, Ministerial Order, National Regulations of Radiation Security, No 23999 (2000).

[10] OFFICIAL JOURNAL OF THE TURKISH GOVERNMENT, Ministerial Order, Turkish Atomic Energy Authority Law; No 2690 (1982).
[11] TURKISH ATOMIC ENERGY AUTHORTIY, The Procedures and Principles for Implementation of Trainings in Radiation Protection, http://www.taek.gov.tr/en/belgeler-formlar/documents/. Accessed on May 3, (2013).
[12] OFFICIAL JOURNAL OF THE TURKISH GOVERNMENT, Ministerial Order, Foundation of ANAEM; No 27715 (2009).
[13] ZEYREK C. T., GÜNDÜZ H., Occupational exposure to ionising radiation with thermoluminescence dosimetry system in Turkey, in 2003, Radiat. Prot. Dosim. 113 (2005) 374–380.
[14] ZEYREK C. T., AKBIYIK H., Education and training activities on personal dosimetry service in Turkey, Radiat. Prot. Dosim. 156 (2013) 451–454

TABLE 1. OVERVIEW OF THE ANAEM SYLLABUS FOR RADIOLOGICAL PROTECTION COURSES

Module	**Objective**	**Main content**	**Duration (hours)**
Fundamental concept of ionizing radiation	To become familiar with the basic knowledge of ionizing radiation.	Atomic structure; radiation sources; radiation dose; external and internal exposure; X-ray production.	4
Units and conversions	To understand measurement units and dosimetric quantities and to perform related calculations.	Dosimetric quantities; RP quantities; application of dosimetric calculations and conversions.	3
Radiation measurement and detection	To be familiar with operating principles and characteristics and to be able to choose the appropriate detectors.	Interaction of X-rays with matter; different types of radiation detectors; practical exercise for measurement.	2
Biological effects of radiation	To become familiar with the mechanisms of different types of biological effects following exposure to X-rays; radiological risks.	Effects of radiation on cells; effects of whole-body irradiation; effects of partial-body (skin, thyroid, eye lens; gonads, etc.) irradiation; stochastic effects; radiation detriment.	3
Principles of radiation protection	To become aware of the ICRP's conceptual framework and international recommendations in RP; to become acquainted with the role played by international organizations; to discuss the optimization of radiology procedures.	Recommendations of the ICRP; optimization of procedures; reference levels; role of international organizations in RP.	2
Personal monitoring, dose limits, typical patient and staff dose levels	To be able to use monitoring programs (whole-body, extremity and skin dosimetry); to be able to apply RP principles to radiology; to understand the concepts used for calculating doses to patients.	Monitoring programs; importance of the suitable location of personal dosimeters; entry dose and dose rate; influence of equipment positioning on occupational doses; effects of using different fluoroscopy modes and personal protection.	3
Radiation shielding for radiology	To become familiar with the shielding for radiology.	General and specialized radiology; safety related to equipment; shielding and calculation of shielding.	2

National regulatory agency for radiation security	To become acquainted with the elements of the regulatory infrastructure for RP and safety.	Scope of National Regulation of Radiation Safety (9); Safety requirements and guides; System of notification; registration; licensing and control of radiation sources; training requirements.	1
X-ray systems and imaging modalities for IR	To be able to safely operate IR equipment (X-ray and imaging systems, etc.)	X-ray systems; parameters of irradiation; the implications of generated power and heat loading of X-ray tubes; the effect of high additional filtration on conventional X-ray beams; usual focus sizes; operation of continuous and pulsed modes and their benefits in terms of image quality; the benefits of the grid-controlled X-ray tube when using pulsed beams; mapping; temporal integration; new imaging modalities; digital sensors, etc.; practical exercises in the laboratory (X-ray rooms).	5
Dose reduction techniques for radiologists	To become familiar with the operating principles and characteristics for reducing occupational doses.	The most important factors that influence staff doses in IR; relationship of X-ray beam filtration and kV setting to occupational dose; effects of using different fluoroscopy modes on occupational doses; influence of C-arm positioning; influence of radiation field size and the importance of the collimation; effects of personal protection equipment (e.g., leaded aprons, gloves, eyeglasses, thyroid protectors, etc.); impact of distance between the staff and patient; analyses of beam direction (scattering angles, incident beam); the importance of equipment ergonomics in staff protection.	3
Dose reduction techniques for patients	To become familiar with operating principles and characteristics for the reduction of patient doses.	Analysis of correlation between the number of images taken in a procedure and the dose received by patients; effects of focus-to-skin distance and placement of the image intensifier; the effect of using an anti-scatter grid on patient dose; typical values of patient entry doses in high- and low-dose fluoroscopy modes; modifying the image rate in cine or digital acquisition for dose reduction; the importance of using different C-arm protections in some high-dose procedures; the effect of using different magnifications on patient dose; the cost-benefit relationship between concentration and type of radiological	2

		procedure and image quality.	
QA programs	To become familiar with QA programs and their importance in dose reduction.	QA procedure; periodical control of irradiation and image parameters; importance in QA programs of periodical control of patient dose and its comparison with reference dose levels.	2
On-the-job-training for dose reduction	To be able to use knowledge obtained in the RP course in practice.	Application of dose reduction techniques for interventional radiologists and patients in the laboratories (X-ray rooms).	8

ICRP, International Commission on Radiological Protection; RP, radiation protection; QA, quality assurance

TABLE 2. THE PERCENTAGE OF TOTAL COURSE TIME DEVOTED TO EACH MODULE IN THE COURSE SYLLABUS

Modules	Percentage of total course time
Fundamental concept of ionizing radiation	9.3%
Units and conversions	7.0%
Biological effects of radiation	4.7%
Radiation measurement and detection	7.0%
Principles of radiation protection	4.7%
Personal monitoring, dose limits, typical patient and staff dose levels	7.0%
Radiation shielding for radiology	4.7%
National regulatory for radiation security	2.3%
X-ray systems and imaging modalities	12.0%
Dose reduction techniques for interventional radiologists	12.0%
Dose reduction techniques for patients	7.0%
Quality assurance programs	4.7%
On-the job-training for dose reduction	19.0%
Total	100%

TABLE 3. PARTICIPANTS' RATINGS OF RADIATION PROTECTION COURSES/INSTRUCTORS

Modules	Participants' rating
Fundamental concept of ionizing radiation	4.5
Units and conversions	4.4
Biological effects of radiation	4.0
Radiation measurement and detection	4.2
Principles of radiation protection	4.2
Personal monitoring, dose limits, typical patient and staff dose levels	4.1
Radiation shielding for radiology	4.0
National regulatory for radiation security	4.2
X-ray systems and imaging modalities	4.1
Dose reduction techniques for radiologists	4.3
Dose reduction techniques for patients	4.3
Quality assurance programs	4.2
On-the-job training for dose reduction	4.0

Ratings were performed on a five-point scale

TABLE 4. PARTICIPANTS' ASSESSMENT OF THE RADIATION PROTECTION TRAINING COURSES FOR THE YEARS 2011–2013

	Participants' assessment	
Class location	Unsuitable	3%
	Suitable	97%
On-the job-training	Insufficient	4%
	Sufficient	21%
	Satisfactory	75%
Scope of the educational content	Disagree	2%
	Partly agree	19%
	Fully agree	79%
Provided handouts	Insufficient	24%
	Sufficient	76%
Course duration	Long	10%
	Suitable	90%

EDUCATION AND TRAINING IN OCCUPATIONAL RADIATION PROTECTION IN LITHUANIA

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Abstract

Radiation Protection Centre (RPC) is a regulatory authority that plays very important role not only in Radiation Protection Supervision and Control, but also in creation of Radiation Protection Training (RPT) system in Lithuania. The Law on Radiation Protection is one of the main legal documents in Lithuania, regulating the requirements for persons, who have to be educated and trained in radiation protection. Regarding to the mentioned Law, on 22 November 2011 there was adopted an Order of the Minister of Health (Order No. 1001 On the Approval of Compulsory Radiation Protection Training and Instruction Procedure). On this Order there are determined the requirements for: persons, who have to be trained in radiation protection; for persons, who want to become lecturers; for training programmes and for the institutions, which want to provide the radiation protection training. According to this Order, the persons, working with ionizing radiation sources, also the persons, who might deal with the ionizing radiation sources on their work and the persons, responsible for radiation protection at their working facilities, have to be trained by initial training programmes before they start work and have to be retrained every five years by the refresher training programmes to renew their knowledge.

1. INTRODUCTION

Education and training of radiation protection is one of the basic instruments to form responsible workers' dealing with ionizing radiation sources position of safe behaviour at their work. Skills and knowledge gained during radiation protection education and training courses guarantee proper and effective use of radiation protection principles to protect public, patients and workers, dealing with sources of ionizing radiation from harmful ionizing radiation effects for health and environment. Several years of experience in radiation protection education and training has proved that the quality of the training courses depends on the collaboration between Technical Support Organizations (TSO), regulatory authorities and different ministries. Therefore, following the IAEA Safety Reports Series No. 20 (Training in Radiation Protection and the Safe Use of Radiation Sources) and European Union requirements and recommendations, RPC prepared the legal acts on compulsory radiation protection training.

2. LEGISLATION OF OCCUPATIONAL RADIATION PROTECTION EDUCATION AND TRAINING IN LITHUANIA

The legislation of occupational radiation protection education and training in Lithuania consists of these legal acts:

i. Law on Radiation Protection,

ii. Order of the Minister of Health (Order No. 1001 On the Approval of Compulsory Radiation Protection Training and Instruction Procedure),
iii. Hygiene standard (HN 73:2001 "Basic Standard of Radiation Protection"),
iv. Order of the Director of Radiation Protection Centre (Order No. V-2 on Establishment of Attestation Commission for Persons, Training Persons Responsible for Radiation Protection and Workers Conducting Practices with Radiation Sources and on Approval of Its Statute).

3. OCCUPATIONAL RADIATION PROTECTION EDUCATION IN LITHUANIA

There are several Universities in Lithuania, which are able to provide the study programs which include the specific topics of radiation protection that cover the need of knowledge of occupational radiation protection. Radiation Protection Centre, as regulatory body, is interested in close cooperation with the Universities' communities and the other organizations related to professional staff education and training. As an example of a good practice – RPC collaboration with Lithuanian Dental Chamber which is a non-profit public association working for the improvement of dental practice and oral care in the Republic of Lithuania. RPC and Lithuanian Dental Chamber time to time are organizing an open discussion with the Universities and providing recommendations for dentistry study programs improvement. Also RPC collaborates with Universities' divisions responsible for medicine physics study programs improvement and this collaboration already gave the result – regarding to RPC recommendations more attention have been draw on subjects, related to patients radiation protection during various diagnostic and treatment procedures, also it was included more topics of occupational radiation protection of professional staff.

RPC is interested to ensure that staff, especially the ones, who are working or might deal at their work with ionizing radiation sources of I-III risk categories, would be qualified and ready to react appropriately in different situations (especially at unexpected cases). According to this RPC collaborates with Kaunas University of Technology (KTU), which organizes the special radiation protection trainings for staff who will be responsible for work with ionizing radiation sources of I-III risk categories at medicine or industry area. These training courses consist of 270 hours of theory and practice and the topics of the training program was created by RPC in collaboration with KTU (the program is approved by the Order of the Minister of Health).

RPC expects to continue successful cooperation with the Universities and in a future to pay more attention on internship and residency study programs of physicians' radiologists.

It is necessary to note, that at the moment regarding to the Order of the Minister of Health all workers, dealing with ionizing radiation sources, must be trained in radiation protection, only except the persons who not earlier than five years ago completed the professional programs related to occupational radiation protection, but these programs have to comply with the duration and consist requirements mentioned on the Order of the Minister of Health.

4. OCCUPATIONAL RADIATION PROTECTION TRAINING IN LITHUANIA

The main legal act, which determines the radiation protection training in Lithuania, is the Law on Radiation Protection. The detailed requirements for radiation protection training are described on Order of the Minister of Health.

In Lithuania there are the main following groups, who have to be trained in radiation protection:

a) Workers, dealing with ionizing radiation sources;
b) Government officials (Customs officers, State Border Guard Service officers, Police officers and fire fighters) and other employees and persons (as workers of metal scrap yards) whose work (activities) is associated with orphan sources of ionizing radiation and detection of materials contaminated with radionuclides;
c) Staff responding to emergency situations (firemen, police officers, workers of medical emergency service).

Under the Order of the Minister of Health, 14 modules of radiation protection training have been drawn, which are a guide for developing radiation protection training programmes for various groups of specialists (RPOs, workers dealing with ionizing radiation sources, government officials, etc.). Each group of such specialists works with ionizing radiation sources of different risk categories (I to V), and programmes are also developed taking into account the risk category of ionizing radiation sources. For more effective training, there are determined the minimum requirements of education levels for persons, RPOs and Workers, dealing with ionizing radiation sources on their work:

TABLE 1. MINIMUM REQUIREMENTS OF EDUCATION TRAINING DURATION LEVELS FOR RPOS – MEDICAL AREA

RS risk category	Minimum education	Initial training duration	Refresher training (every 5 years)
I	University degree in biomedicine, physics sciences or technological sciences	270 hours	20 hours
II, III	University degree in biomedicine, physics sciences or technological sciences	270 hours	20 hours
IV, V	University degree in biomedicine, physics sciences or technological sciences	60 hours	20 hours
Dental X-ray machines	University degree in biomedicine, physics sciences or technological sciences	20 hours	8 hours

TABLE 2: THE MINIMUM REQUIREMENTS FOR LEVEL OF EDUCATION AND TRAINING DURATION FOR RPOS – INDUSTRIAL AREA

RS risk category	Minimum education	Initial training duration	Refresher training (every 5 years)
I	University degree in biomedicine, physics sciences or technological sciences	270 hours	20 hours
II, III	General education in biomedicine, technological or physics sciences, or specialized secondary school education for graduates up to 1995	270 hours	20 hours
IV, V	High school education, or specialized secondary school education for graduates up to 1995, and acquired professional qualifications equivalent to the type of work	60 hours	20 hours

TABLE 3: THE MINIMUM REQUIREMENTS FOR LEVEL OF EDUCATION AND TRAINING DURATION FOR WORKERS DEALING WITH IONIZING RADIATION - MEDICAL AREA

RS risk category	Minimum education	Initial training duration	Refresher training (every 5 years)
I - V	General education in biomedicine, physics sciences or technological sciences, or specialized secondary school education for graduates up to 1995, and acquired professional qualifications equivalent to the type of work with ionizing radiation sources	30 hours	20 hours
Dental X-ray machines	General education in biomedicine, physical sciences or technological sciences, or specialized secondary school education for graduates up to 1995, and acquired professional qualifications equivalent to the type of work with ionizing radiation sources	14 hours	8 hours

TABLE 4: THE MINIMUM REQUIREMENTS FOR LEVEL OF EDUCATION AND TRAINING DURATION FOR WORKERS DEALING WITH IONIZING RADIATION SOURCES - INDUSTRIAL AREA

RS risk category	Minimum education	Initial training duration	Refresher training (every 5 years)
I - V	Secondary school education and acquired professional qualifications equivalent to the type of work with ionizing radiation sources	30 hours	20 hours

Legal entities and enterprises must at their own expense organize compulsory training of their workers, who are engaged in activities with ionizing radiation sources, or who can be exposed to radiation. RPC inspectors carrying out supervision and control can evaluate radiation protection knowledge of the staff during inspections.

There are some institutions (TSOs) providing RPT in Lithuania. Head of TSO is responsible for the quality assurance of the training and issued certificates.

Knowledge of course participants is evaluated after every RPT course. There are some requirements for their effective knowledge assessment:

(a) An Evaluation Commission should be established to assess knowledge of course participants;
(b) The Chairman of Evaluation Commission should be a representative of RPC;
(c) At least one member of the Evaluation Commission has to be a qualified lecturer;
(d) Evaluation is divided into two parts - theory and practice: a test (30 questions for theoretical knowledge assessment) or 3 open questions, which requires oral answers and demonstration for practical knowledge assessment;
(e) Evaluation results must be recorded in an examination protocol;
(f) If participants pass the examination, they get certificates.

Also for the effective training it is necessary to have high qualified lecturers, who would be able to share their knowledge with the participants of the courses. Persons wishing to be radiation protection lecturers have to pass an examination of the Attestation Commission and get a certificate. The certificate is issued for specified topics. The Attestation Commission is consisted under an order of the Director of RPC. Examination is divided into two parts - theory and practice. A certified person must have a university degree in technology or physics, or biomedical sciences.

RPC is interested in effective implementation of radiation protection training, so once per year (or if it is necessary – more than once) is organizing the verification of institutions (TSO), providing radiation protection training. Also, RPC organises qualification improvement courses – seminars, radiation protection trainings for RPOs and workers dealing with ionizing radiation sources, where radiation protection specialists presents the changes in radiation protection system. In cooperation with IAEA and other international organisations as European Commission organises trainings for trainees, trainings and practices for specialists. RPC cooperates with various Governmental institutions and organises compulsory radiation protection training courses for Government officials (Customs officials, State Border Guard Service, Police officers, fire fighters, municipal civil protection specialists).

5. CONCLUSION

Lithuania has created a radiation protection training system based on Lithuanian legislation and EU and IAEA recommendations. Legal requirements for radiation protection training are based on the Order of the Minister of Health and met in practice. The created system ensures that persons, who work and deal with ionizing radiation sources or are responsible for radiation protection at working objects, get the main information and skills, required for their effective work and safety.

However, Radiation Protection centre is interested in acknowledgment of occupational radiation protection programs in Universities, which could ensure the proper level of knowledge of graduated persons.

REFERENCES

[1] LAW ON RADIATION PROTECTION.
[2] ORDER OF THE MINISTER OF HEALTH, Order No. 1001 on the: Approval of Compulsory Radiation Protection Training and Instruction Procedure.
[3] HYGIENE STANDARD (Hygiene Standard HN 73:2001, Basic Standard of Radiation Protection) (2001).
[4] ORDER OF THE DIRECTOR OF RADIATION PROTECTION CENTRE (Order No. V-2 on Establishment of Attestation Commission for Persons, Training Persons Responsible for Radiation Protection and Workers Conducting Practices with Radiation Sources and on Approval of Its Statute).

CARGO SCAN: IMPROVING EDUCATION AND TRAINING IN OCCUPATIONAL RADIATION PROTECTION OF BRAZILIAN WORKERS

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Abstract

For more than a decade, the non-invasive inspection systems for cargo, vehicles and containers has been used worldwide to find illegal products, such as weapons, narcotics, explosives, contraband and even human trafficking through the image of large objects inside cargo containers, unoccupied vehicles, trains, trucks or ships. This technology uses conventional low X-ray generators and linear electron accelerators with high energy between 1.5 and 9 MeV. The linear accelerators must be submitted to a regulatory licensing process and only occupationally exposed workers with radiation protection training can operate them in Brazil. There are, in Brazil, 26 non-invasive inspection equipment in operation and a large number of equipment are still in regulatory licensing and commissioning process. As these non-invasive inspection systems work in continuous operation, that is, 24 hours a day, 7 days a week, 365 days a year, it is necessary to have a large number of well-trained radiation workers, mainly in radiation protection, to operate the systems, according to the Brazilian legislation. This paper shows the methodology used to train the radiation workers using the classroom learning and e-learning making a total of 140 hours course taught by Industrial Radiation Protection Officer accredited by the Regulatory Authority.

5. INTRODUCTION

After the September 11, 2001 USA attack, the security actions have been increasingly stringent in ports all over the world. The use of ionizing radiation technologies to carry out screening of cargo, vehicles and containers was expanded considerably in recent years, mainly as a result of laws established in the United States that required, since 2013, the digital scanning of all containers arriving at American ports [1].

For more than a decade, the non-invasive inspection systems of cargo loads, vehicles and containers have been used to find illegal products, such as weapons, narcotics, explosives, contraband and even human trafficking through large image objects [2].

As this technology uses conventional low X-ray generators and linear electron accelerators with high energy between 1.5 and 9 MeV, the linear accelerators must be submitted to a regulatory licensing process and only occupational exposed workers with radiation protection training can operate them in Brazil.

6. BRAZILIAN LEGISLATIONS

In 2009 and 2010 the Brazilian Federal Revenue Authority regulated the technical and operational requirements for the use of non-invasive inspection equipment of cargo and vehicles in ports and Brazilian borders. [3]

The Brazilian Regulatory Authority - National Nuclear Energy Commission (CNEN) - regulated the safe use of the equipment establishing the requirements for licensing the

installations that use X-ray devices for non-invasive inspection with energy between 0.60 MeV to 50 MeV [4]. It was also regulated the training program to radiation workers to operate these X-ray devices [5], as well as the requirements for certifying the qualifications of Radiation Protection Officer [6].

To assure safety and radiation protection for the radiation workers and members of the public, the installation must be authorized to operate and perform the maintenance of the X-ray non-invasive inspection devices. The installation must submit a Radiation Protection Plan with the operational procedures and training program.

7. NON-INVASIVE SCANNING TECHNOLOGY

The non-invasive inspection technology utilizes both conventional X-ray generators of low-energy and linear accelerators with high-energy between 1.5 and 9 MeV. The type of X-ray source device and the energy are selected based on the volume and density of the load to be inspected. There are two types of non-invasive inspection equipment: the fixed and mobile. In the fixed system the loads are conducted to be inspected on previously installed non-invasive equipment. In the mobile system the non-invasive equipment is taken to inspect the loads. Both inspection systems have infrared sensors to prevent any exposure of member of the public and mainly the truck driver during the scanning.

The Brazilian port terminals and national and international borders are using fixed and mobile non-invasive inspection equipment.

8. WORKER'S ACADEMIC FORMATION

In Brazil, there are special graduate professional named "radio-technologist" and technician professional named "x-ray technician" that are responsible to operate x-ray devices. Radiation physics, monitoring equipment, radiobiology, radiation protection, as well as anatomy, physiology, pathology, ethics, and psychology, among others, are the scope of the academic formation. But this scope is restricted to medical applications, such as radiology, CT, radiotherapy, nuclear medicine, x-ray diagnostic, among others [7- 8].

Those professionals must have an extra training to work with any industrial application including the non-invasive inspection systems. This training must be taught by an Industrial Radiation Protection Officer accredited by Regulatory Authority (CNEN).

A radioactive installation with non-invasive inspection equipment must have two Industrial Radiation Protection Officer to supervise the safe operation of the equipment. One Industrial RPO must be accredited in linear accelerator and another in industrial x-ray device with energy higher than 600 kV [6]. Nowadays there are only 62 Industrial Radiation Protection Officer accredited in linear accelerators in the country [9]. To operate these equipment the installation must have, at least, three operators by turn with radiation protection training.

9. EDUCATION AND TRAINING METHODOLOGY IN RADIATION PROTECTION

In order to supply the demand for well-trained workers to operate safely the high-energy radiation equipment, the following methodology of classroom learning and e-learning is being used:

a) The radiation protection training is subdivided into three modules: basic, specific and practical, and they can be made by classroom learning or e-learning, making a total of

140 hours course. Optionally, the basic and specific modules can be done using e-learning methodology, that the professional receives the theoretical training via internet. This distance learning allows to access the online classes, interactive exercises, simulations, interactive forum, virtual library and other online resources. The practical training module must be conducted only in classroom, allowing access to technical laboratories and visits in radioactive facilities.

b) The first module, the basic training, has the following scope: fundamentals of atomic and nuclear physics, biological effects of ionizing radiation, radiation quantities, radiation protection concepts and nuclear instruments.
c) A theoretical evaluation must be done after the ending of the basic module to evaluate the professional performance.
d) After approved, the professional begins the specific training module. This module provides theoretical training in x-ray equipment, operating principles and applications in industry. The program is complemented with aspects of security, radiological accidents, response to radiological emergencies, safety procedures and guidelines and an overview of regulations in industrial radiation protection.
e) The practical module includes activities using radiation monitoring equipment to learn how to survey and make interpretations of results. The module also provides practical application of operational procedures for testing, inspection and maintenance of safety and radiation protection equipment, and emergency drills based on radiological accidents.
f) During the practical module, the professional visits radioactive facilities to improve the training received and to contact experts in industrial radiation protection.
g) At the end of the course the final evaluation is performed based on general and specific examinations, evaluation of practical activities reports and a group seminar about radiation protection.

10. RESULTS AND CONCLUSIONS

Brazil has a large territory that borders with almost every country in South America. Moreover, it has numerous ports an extensive area that contributes to the construction and expansion of the Brazilian port system to increase their exports and imports, which are considered of utmost importance to the economic development of the country.

In 5 years 1200 professionals received radiation protection training by Maxim Industrial being 98% through classroom training [10]. These classroom training were taught in different cities of the country, which allows training professionals nationwide, as shown in Figure 1. Moreover, it is possible to select and hire these well-trained professionals by the owners of non-invasive inspection equipment companies, not only in large urban centres and capital but also within the country.

FIG. 1. Distribution of professionals trained by Brazilian States

The geographical distribution of the trained professionals shows that they are from the major urban centres of the country because the location of the ports. As the border regions are located on the opposite side of the large urban centres there are few trained professionals.

The need of well-trained professionals is eminent, especially in the interior of the country. The use of distance learning technology (e-learning) can provide greater geographic accessibility, allowing empower, improve and enhance the knowledge of professionals from distant locations and preparing them for the job market.

REFERENCES

[8] M.A. DESCALLE, K. VETTER, A. HANSEN, J. DANIELS, S.G. PRUSSIN, "Detector design for high-resolution MeV photon imaging of cargo containers using spectral information", Lawrence Livermore National Laboratory, LLNL-JRNL-423524 (2010).

[9] ROGÉRIO S. GOMES, JOANA D'ARC R. L. GOMES, MARA LUCIA L. COSTA AND MÁRCIA VALÉRIA F. E. S. MIRANDA, "Dose to drivers during drive-through cargo scanning using geant4 Monte Carlo simulation", INAC (2013)

[10] COANA. Ato Declaratório Executivo Coana nº 27, de 22 de dezembro de 2010 http://www.receita.fazenda.gov.br/Legislacao/AtosExecutivos/2010/COANA/ADCoana027.htm

[11] CNEN. Resolução CNEN Nº 166/14 – Licenciamento de Instalações Radiativas (2014).

[12] CNEN. Resolução CNEN Nº 145/13 – Requisitos de Segurança e Proteção Radiológica para Serviços de Radiografia Industrial (2013).

[13] CNEN. Resolução CNEN Nº 146/13 – Certificação da Qualificação de Supervisores de Proteção Radiológica (2013).

[14] LEI nº 7394, de 29 de outubro de 1985. Regula o Exercício da Profissão de Técnico em Radiologia e dá outras providências (1985).

[15] DECRETO nº 92790, de 17 de junho de 1986, Regulamenta a Lei nº 7.394, de 29 de outubro de 1985, que regula o exercício da profissão de Técnico em Radiologia e dá outras providências (1986).

[16] CNEN Homepage da CNEN - http://www.cnen.gov.br/ (2014).

[17] MAXIM INDUSTRIAL, Homepage da empresa Maxim Industrial – http://www.maximindustrial.com.br/ (2014).

TRAINING IN RADIATION PROTECTION FOR WORKERS OCCUPATIONALLY EXPOSED IN PERU

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Abstract

According to the IAEA GSR Part 3 Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, (BSS), a worker is "Any person who works, whether full time, part time or temporarily, for an employer and who has recognized rights and duties in relation to occupational radiation protection". These persons will need to be provided with the appropriate information, instruction and training on protection and safety. In Peru, the current regulations provide that these workers must have an authorization (Individual License), which is granted by the Technical Office of the National Authority. This is the technical arm of the Peruvian Institute of Nuclear Energy (IPEN), responsible for the control of ionizing radiation in the country. The Individual License is granted when the applicant meets the requirements and one of them is have knowledge of safety and radiation protection.

Since 1972, the Superior Center of Nuclear Studies (CSEN) of IPEN has conducted various training courses to enable people to work safely with ionizing radiation in medicine; industry and research, so much so that until 2013 has organized 2231 courses which enabled the training of 26213 people. In this paper the historical evolution of radiation protection courses and their importance is shown to work safely in the country.

1. INTRODUCTION

Radiation protection courses have been almost always related to regulations that require compliance with appropriate training and approval of the proficiency exams so that users can have the appropriate authorization to work with ionizing radiation. This is one of the reasons why training programs have been modified over time considering in turn the international recommendations.

The Peruvian Institute of Nuclear Energy created in 1975, has as one of its main functions to plan and implement training activities, developing and coordinating programs and development expertise in the nuclear field, and also in the regulatory aspect, issues rules on Radiological Protection and controls compliance nationwide. Since its establishment the Superior Center for Nuclear Studies of IPEN, is the main center for training for nuclear issues in the country and especially in radiation protection.

2. SUPERIOR CENTER OF NUCLEAR STUDIES (CSEN)

Between 1955 and 1975 there was the Board of Control of Atomic Energy (JCEA) was the agency responsible for developing control activities in all matters relating to atomic energy and the production of raw materials that generate. The JCEA preceded the Peruvian Institute of Nuclear Energy.

In the 134th meeting of the Board of JCEA that occurred on November 23, 1972, the Superior Center for Nuclear Studies (CSEN) is created to replace the Superior Institute for

Nuclear Studies, in order to train personnel in the fields related to nuclear science by delivering courses, seminars and conferences to expert staff that the country needs to develop nuclear energy and its applications.

Since its establishment the CSEN has developed numerous academic activities highlighting the Masters in Nuclear Energy (6) and Medical Physics (11) with the National Engineering University (UNI), also one Master in Nuclear Physics was organized with the National University of San Marcos and one Master in Nuclear Chemistry with the Pontifical Catholic University of Peru. UNI also has been developing the 3rd Program of the Second Professional Specialization in Radiation Protection. 2 Nuclear Medicine Diploma and 4 Long-term courses for Nuclear Energy Technicians were performed. There have been numerous courses for secondary school teachers (nuclear energy, nuclear physics, radiochemistry and radiobiology). Radiation protection issues were incorporated into all courses. In addition there have been numerous courses on applications of ionizing radiation in industry, medicine and research and courses on radiation protection.

3. DEVELOPMENT OF EDUCATIONAL PROGRAMS FOR RADIATION PROTECTION

Programs for radiation protection courses have evolved in content and specificity in regard to the guidance on whom it is addressed, and these programs also have been modified according to regulatory requirements. For this reason we have identified four clearly defined stages as detailed below.

3.1. `Stage 1: 1972 – 1980

The CSEN opens in the JCEA until 1975 when the IPEN is created. In its beginnings CSEN organized courses for personnel radiation protection IPEN counting with the help of some foreign teachers visiting their facilities. The course duration was varied from a few days (approximately 20 hours of classes) until about 3 weeks with 30 hours of classes. At this stage the CSEN organized 8 courses and trained 143 people.

3.2. Stage 2: 1981 – 1989

From this stage, courses radiation protection are more needed by people working in radioactive facilities due to regulations such as the "Regulation on Radiological Protection" and "Rules of Facilities with sources Ionizing Radiation" which were promulgated by IPEN a resolution on 27 October 1980. These provisions required the staff of nuclear, radioactive and ionizing radiation generating equipment installations will have a license that are specific and apply only to a specific and pre-defined installation. This should meet one of the requirements was training in radiation protection. This regulation is mainly applied to staff research reactors and radioactive facilities where it was decided that it was the Supervisor who was able to direct the activities in a particular installation.

For this reason, the first radiation protection courses were aimed at training the supervisors of these facilities. These courses are called: Radiation Protection Supervisor and had a duration of 2 weeks, ie 54 hours of classes: 33 hours of theory, 15 hours of practice and 6 hours for 2 exams. The topics were: General concepts, radioactivity, interaction of radiation with matter, radiation measurement, dosimetry, radiation risks, shields, biological effects of radiation, internal contamination, waste management and transportation laws and regulations. Four courses of these types are conducted.

Subsequently, these courses were aimed at the application of radiation in industry and medicine and why 2 types of courses are offered: Radiation protection Radiation in medical application and radiation protection in industrial application, which lasts for about 20 hours. In addition to the issues mentioned in courses for Supervisors, the uses of ionizing radiation are included and give details on the production of X-rays, characteristics, uses and radiation protection, measures. Seven such courses are held.

Subsequently, the course content is modified because the programs are made in relation to the type of radiation being used. The courses are called: Radiation protection in the use of radioactive sources and Radiation Protection in the use of X-ray all with a duration of about 20 hours. Only 3 courses radioactive sources because the attendees, but were mostly in the industrial field, they had to take classes on the uses of radiation and radiation protection aspects in industry and medicine were made. For that reason, these courses have an amendment at this stage and are directed only to industrial field, while courses on X-rays were more oriented to the medical field, but remains so until the end of this period. By these modalities, 13 courses are arranged.

At this stage, all 24 courses and trained 443 people are developed. People who passed the examinations could process the corresponding Individual License.

3.3. Stage 3: 1990 – 1997

On 29 September 1989, the "Regulation on Radiological Protection" was approved by Supreme Decree provides that where handling ionizing radiation sources or perform work involving exposure to ionizing radiation shall be permitted only to persons holding license Individual. This will need to demonstrate knowledge of radiation protection in the specific area where you work.

In CSEN continues with the issuance of certain courses of Radiation Protection in the use of radioactive sources and X-ray until you decide to change the content of them and make them more specific according to the specific application of radiation, for example: courses of Radiation Protection in the medical use of X-ray. At this stage 17 courses are taught and trained 354 people. As in the previous stage, people who pass the courses continue their paperwork for their respective Individual License.

3.4. Stage 4: Desde 1998

As in the previous case, this stage began with the enactment of the "Regulations on Radiological Safety" which was approved on May 20, 1997 through a Supreme Decree, where it is mentioned that the handling, operation or work with sources ionizing radiation will be allowed only to persons authorized by an Individual License, provided you meet the requirements. Regulation granted within one year so that all people are suited to the provisions thereof. This stage is characterized by the radiation protection courses were more specific and had greater demand due to regulatory requirements.

The CSEN organizes and develops courses but tests the participants performed the Technical Office of the National Authority (OTAN). Passing the exam is required to process the Individual License.

The courses provide some common themes but according to the type of course specific issues are incorporated, for example, X-ray production and characteristics. Team characteristics, Radiation Protection in the use of X-rays etc..

Most of the courses are developed in a week and last for 20 hours of practical classes where classes are held, in several cases, radioactive facilities or X-ray duration of the courses include depends on the relevance of practice.

The current name of some courses is: Radiation Protection in Medical Radiology, Radiation Protection in Dental Radiology, Radiation Safety in Nuclear Medicine, Radiation Safety in Radiotherapy, Radiation Safety in the use of Nuclear Gauges, Radiation Safety in Industrial Radiography and Radiation Safety in the use of radioactive sources.

This stage is also characterized by the 17 July 2003 Law 28028 approved "An Act to regulate the use of sources of ionizing radiation," which regulates the practices that result in exposure or potential exposure to ionizing radiation in order to prevent and protect, their harmful effects, the health of people, environment and property. The competent authority to implement the provisions of this regulation is the Peruvian Institute of Nuclear Energy, hereinafter the Authority; and in accordance with its Charter approved by Decree Law No. 21875, as amended by Legislative Decree No. 158, will be responsible for regulatory functions, authorization, control and supervision of the use of ionizing radiation sources relating to radiation safety and nuclear safeguards and physical protection of nuclear material in the country.

This determined the increase courses even begin to dictate the refresher training on radiation safety, which have a duration of 5 hours and are aimed at people who need to renew the Individual License and which must necessarily attend the course order to update their knowledge because it is a regulatory requirement.

Subsequently, specific rules where the presence of the Radiation Protection Officers required in radiotherapy facilities, industrial radiography, nuclear medicine and X-ray medical diagnosis why the CSEN began dictating courses for the training of these specialists are held. This step has been made where 2182 courses have trained 25273 people. The Table 1 shows the evolution of the courses and the trained people.

TABLE 1. THE EVOLUTION OF COURSES IN RECENT YEARS

STAGE	COURSES	TRAINED PEOPLE
1972 - 1980	8	143
1981 - 1989	24	443
1990 - 1997	17	354

YEAR	COURSES	TRAINED PEOPLE
1998	19	350
1999	14	163
2000	9	104
2001	13	200
2002	41	522
2003	82	653
2004	93	1136
2005	138	1804
2006	155	1899
2007	196	1913
2008	272	3025
2009	224	2379
2010	220	2350
2011	249	2953
2012	224	2831
2013	233	2991

4. CONCLUSIONS

Since the creation of CSEN, 2231, courses have been conducted on safety and radiation protection which has allowed the training of 26213 people. 90% of the courses were conducted in the period 2004-2013, which has resulted in training of 89% of all participants in the first 41 years of the CSEN. The regulatory requirement to have the Individual License was important for the development of radiation protection courses as the highest percentage of trainees are working with ionizing radiation.

The courses have evolved positively in terms of content and in terms of agreement to be specific to each application. In addition, they have incorporated the concepts emanating from international organizations and has been taken as reference information from the IAEA and has begun to include presentations on specific courses developed by the Radiation Safety - International Atomic Energy Agency.

Most of the teachers of the courses run by the IPEN CSEN are professionals who have at least graduation in radiation protection and extensive professional experience in various subjects. This is complemented by the use of equipment and infrastructure of IPEN.

Since 1972, number of courses have been increasing and also the number of trained man power. In step 1, 7 courses to 143 people were issued in step 2, 24 courses to 443 people were issued in step 3, 17 courses to 354 people were issued, and the last stage had 2182 courses that allowed training of 25,273 people.

All this activity contributes to the improvement of safety and radiation protection in the country.

REFERENCES

[1] MEDINA GIRONZINI, EDUARDO, Reglamentación y normativa sobre protección radiológica en el Perú., Primer Congreso Regional sobre Seguridad Radiológica y Nuclear, Buenos Aires, 21-25 octubre, (1991).

[2] MEDINA GIRONZINI, EDUARDO, Training on Radiological Protection in Peru". 11th International Congress of the International Radiation Protection Association. IRPA – 11. Madrid, 23 – 28 mayo (2004).

[3] MEDINA GIRONZINI, EDUARDO, Second Professional Specialization in Radiological Protection in Peru 11 th International Congress of the International Radiation Protection Association. IRPA – 11. Madrid, 23 – 28 mayo (2004).

[4] RAMÍREZ Q., R. Y MEDINA G., E., Desarrollo histórico y tendencias de las actividades regulatorias del Instituto Peruano de Energía Nuclear-Autoridad Nacional". Tercer Congreso Regional sobre Seguridad Radiológica y Nuclear, Cusco, 23-27 octubre (1995).

[5] Centro Superior de Estudios Nucleares. Actas de cursos (1972-2013). Lima.

[6] Ley 28028. Ley de Regulación del uso de fuentes de radiación ionizante". Publicada el 18 de julio de 2003 en el diario oficial El Peruano.

[7] Decreto Supremo Nro. 041-2003-EM. Reglamento de la Ley 28028. Publicado el 12 de diciembre del 2003 en el diario oficial El Peruano.

METHODOLOGY FOR DEVELOPING A NATIONAL STRATEGY FOR EDUCATION AND TRAINING IN OCCUPATIONAL RADIATION PROTECTION: OVERVIEW OF BRAZILIAN ACTUAL STATUS AND PLANNING TO MEET THE TRAINING NEEDS OF PROFESSIONALS

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Abstract

The Institute of Radiation Protection and Dosimetry (IRD) it is a research institute established in 1972, with the aim of carrying out environmental assessments and occupational assessments, internal and external dosimetry, metrology of radiations, physical medical and training of professionals in this areas. The IRD, provides regular education and training courses, collaborating with IAEA in education and training activities for Latin American and Caribbean countries through various Projects of the Latin: American Region – RLA: 9/066, 9/070 and 9/075, IAEA in order to strengthen this area.

This paper will show the results obtained to identify the radiation practices in Brazil with the numbers of radioprotection supervisors, the graphics (practices x supervisors) for Brazilian states and regions and the need for radiation protection supervisors for each practice. The information obtained about the education suppliers, the number of professionals in the country and number of practices running in Brazil will be used to perform the strategy for providing training and education. Several planned actions, will consolidate the application of national strategy methodology and, in particular, efforts to engage in this process the Brazilians radiation protection stakeholders, mainly the Brazilian Regulatory Body, CNEN.

1. INTRODUCTION

This work began at the meeting of the regional meeting of Coordination and Planning Project RLA 9/065 [1], having continuity in the following years through RLA 9/070 [2], consolidating a lot of information and documents prepared during the several years of its implementation. Continuing the work started in 2010 we are currently working on a new project the RLA 9/075 [3] which aims to unify the areas in a single project:

(a) Implementation of the radiation protection program requirements by end users at facilities/activities including optimization and safety culture;
(b) Technical support services available and authorized, related to individual and workplace monitoring, calibration and advisory services including operations of the National Dose Registry;
(c) Radiation Protection and optimization programmes in digital radiography, Computed Tomography, PET/CT, interventional procedures (Cardiology and non-Cardiology) with emphasis in special groups as children and Pregnant, are in place, through appropriate training to physicians, medical physicists and technicians e;
(d) National strategy in education and training established and programme designed, implemented and evaluated, based on assessed needs and identified existing resources.

For this work a methodology was used in job steps: Step 1: Identification of the practices used in Brazil; Step 2: Mapping the relation of the number of practices with practice; Step 3: Mapping of practices with the number of radiation protection office in the country; Step 4: Mapping of individualized practices by Brazilian states; Step 5: Mapping of individual practices by states x-radiation protection office and Step 6: Mapping of training courses by the states.

All these steps have provided and will provide information for the preparation of a national strategy for education and training in radiation protection in accordance with the demands of practice x supervisors x states.

2. METHODS AND RESULTS

The steps involved assessment of the information obtained through the National Commission of Nuclear Energy site - CNEN, the Brazilian regulatory body: Professional certificates, certifications and authorized facilities. Practical identified in Brazil as well as the number of practices are shown in Fig. 1.

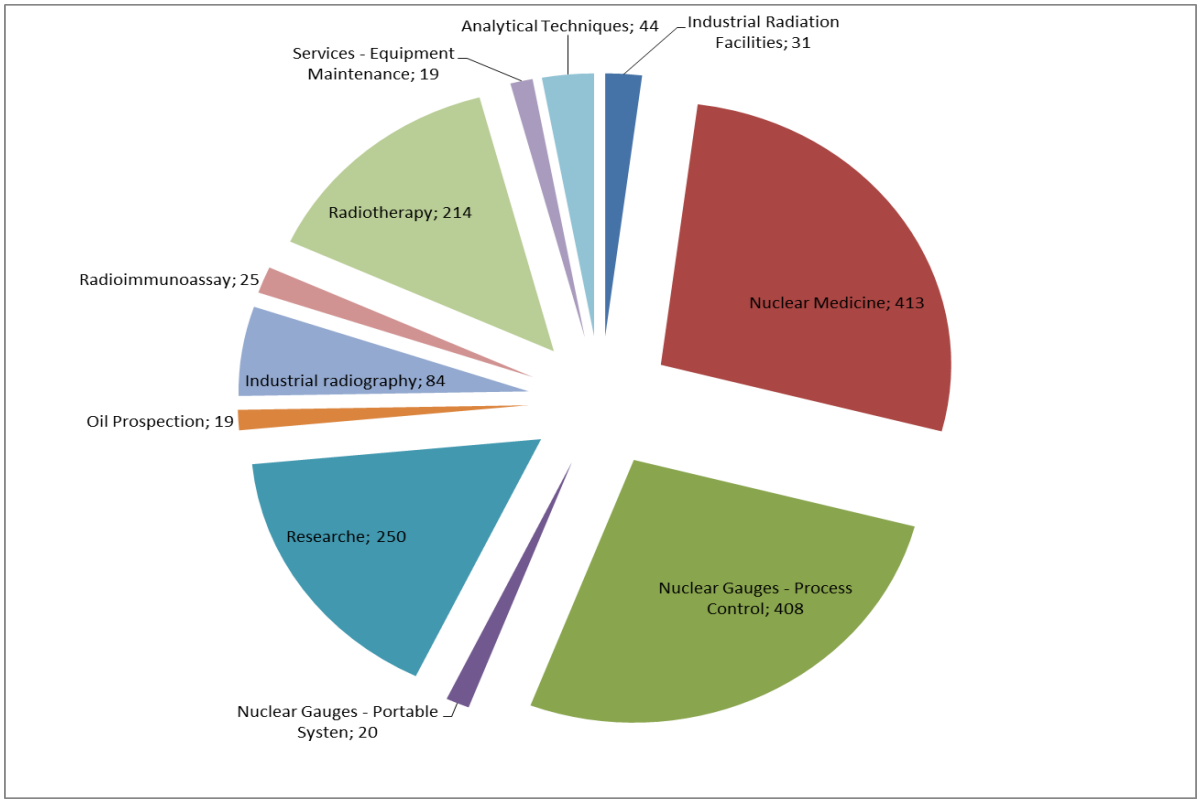

FIG. 1. Distribution of practice and their quantities

The relation of the number of practices with radiation protection supervisors are shown in Fig. 2.

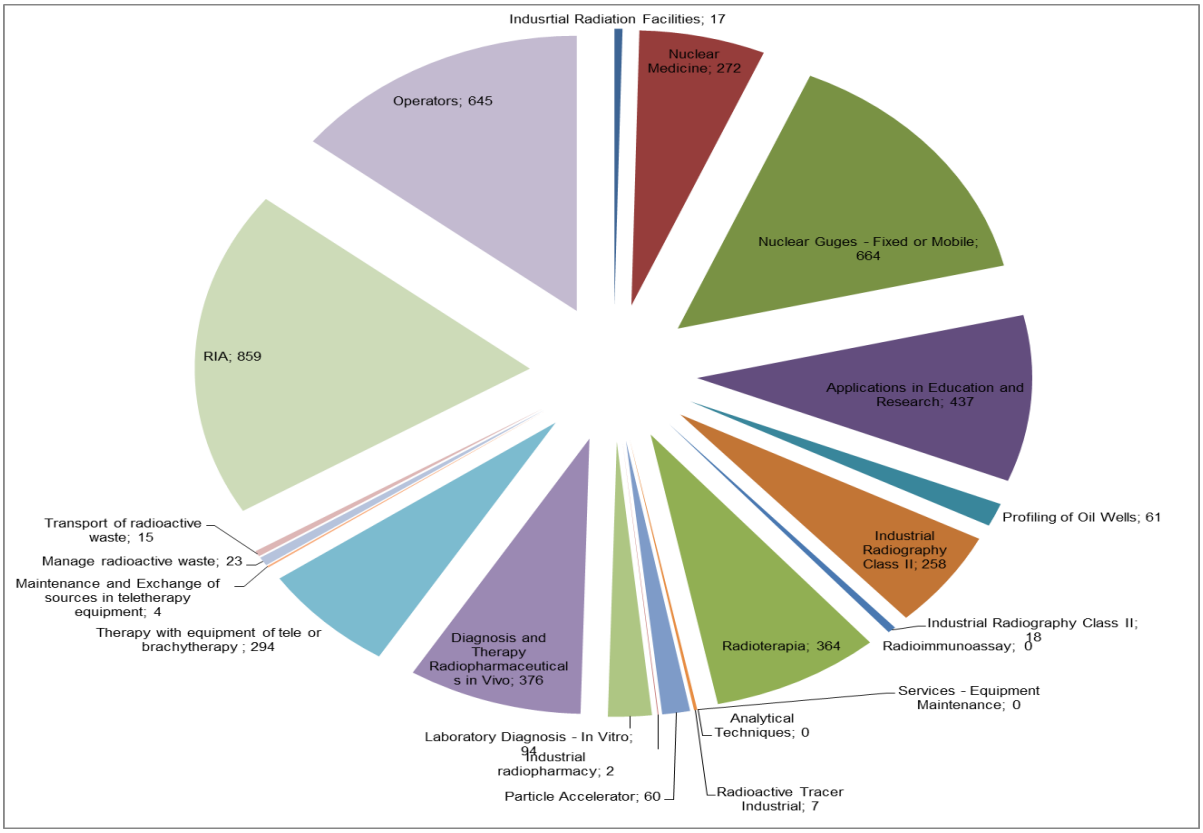

FIG. 2. Identification of Practices related to the number of Radiation Protection Office

In Table 1 and the charts below show an example of the efficiency of this work and how it can guide the development of a national strategy for education and training:

TABLE 1. REGIONS AND BRAZIL STATES

Regions	States	Symbol	Regions	States	Symbol
North	Acre	AC		Sergipe	SE
	Rondônia	RO		Bahia	BA
	Amazonas	AM	Midwest	Mato Grosso	MT
	Pará	PA		Distrito Federal	DF
	Tocantins	TO		Goiás	GO
	Amapá	AP		Mato Grosso do Sul	MS
	Roraima	PR	Southeast	Minas Gerais	MG
Northeast	Maranhão	MA		Espirito Santo	ES
	Piauí	PI		Rio de Janeiro	RJ
	Ceará	CE		São Paulo	SP
	Rio Grande do Norte	RN	South	Paraná	PR
	Paraíba	PB		Santa Catarina	SC
	Pernambuco	PE		Rio Grande do Sul	RS
	Alagoas	AL			

1) Distribution of Industrial Radiation Facilities vs Distribution of the Brazilian States Fig. 3.

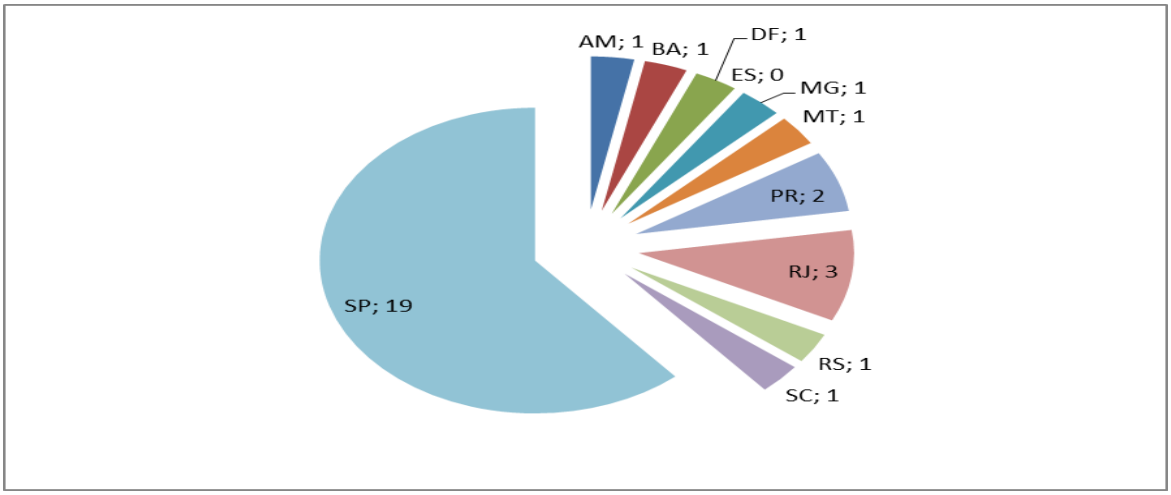

FIG. 3. Industrial Radiation Facilities vs Brazilian States

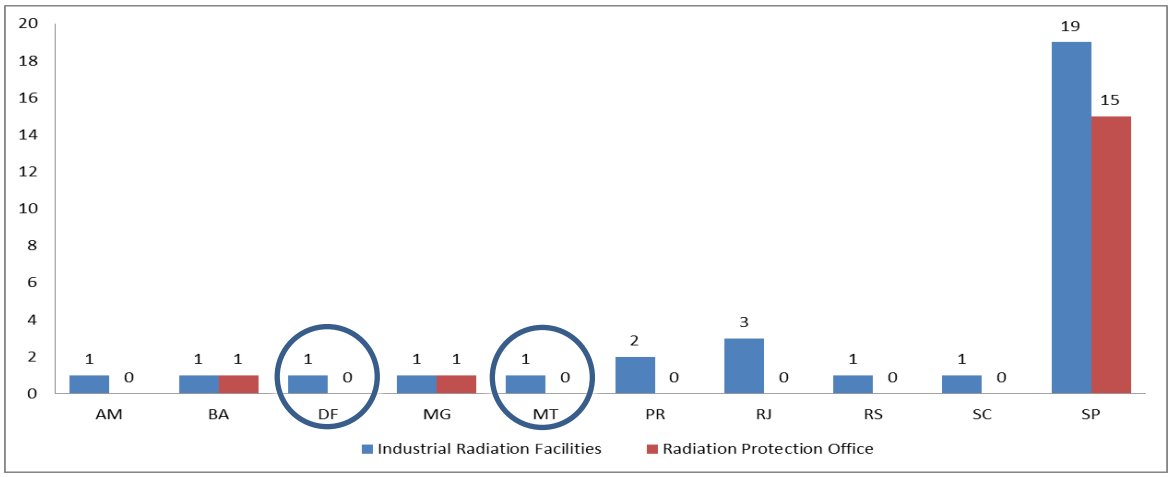

FIG. 4. Industrial Radiation Facilities vs Radiation Protection Office

The results in Figs. 3 and 4 show the practice of Industrial Radiation Facilities distributed by states and the number of Radiation Protection Office - RPO. In this graph, it can be idenified that in the state of Mato Grosso / MT and the Federal District / DF that both have a practice but has no RPO, it should consider installing this region a training center to meet demand this practice and other practices.

2) Distribution of the practice of nuclear medicine x Distribution by Brazilian States.

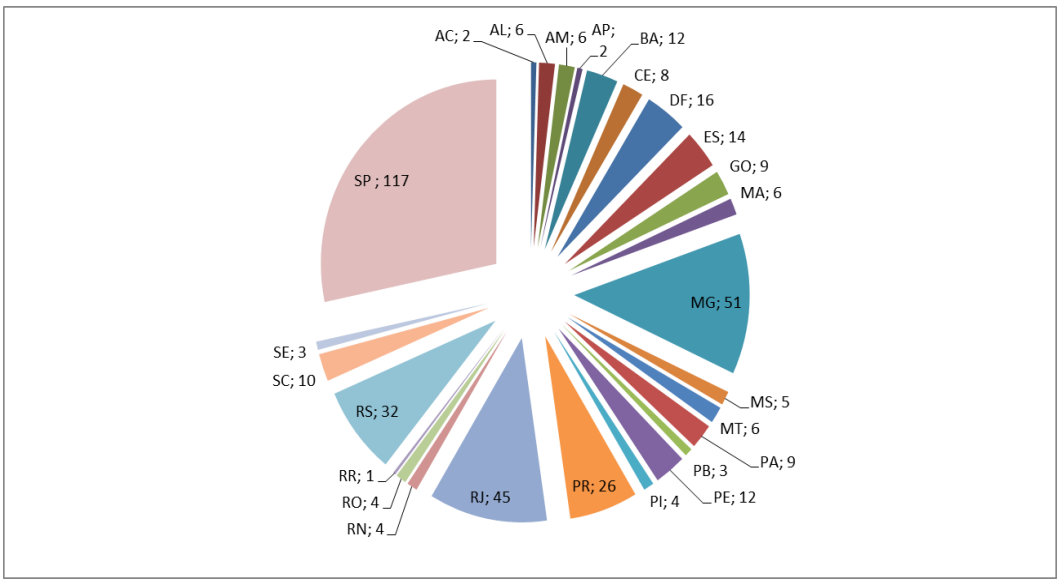

FIG. 5. Practice of Nuclear Medicine vs Brazilian States

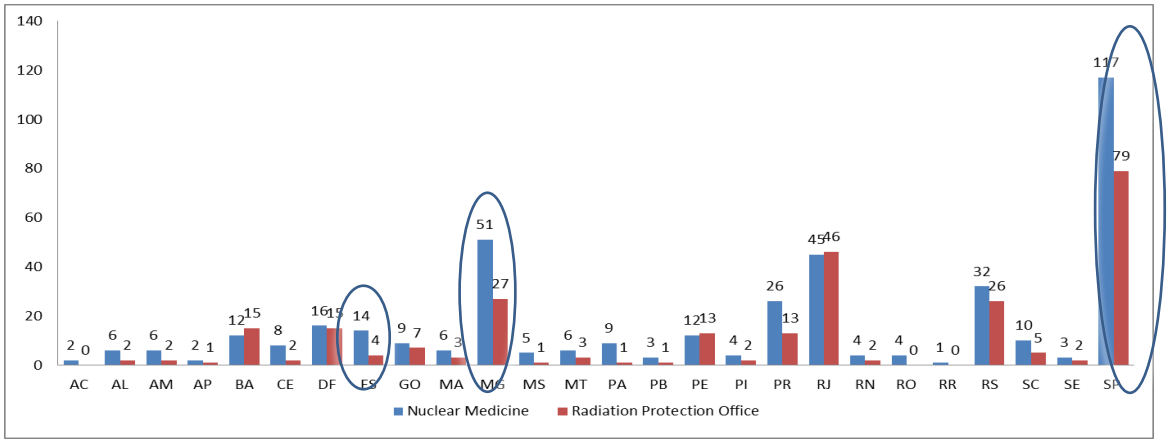

FIG. 6. Practice Nuclear Medicine vs Radiation Protection Office

The data in Figs. 5 and 6 show the practice of Nuclear Medicine distributed by states and the number of Radiation Protection Office/RPO. In this graph, one can identify that, in the state of Espirito Santo/ES, Minas Gerais/MG e São Paulo/SP have the practice but do not have the appropriate number of RPOs, it should consider installing in this region a training center to meet demand of this practice and other practices.

Regarding the number of training courses and training for RPO have not been identified in its entirety as well as their distribution among Brazilian states as an example we will present the data relacionadaos the practice of Irradaiadores Large Size in only two states.

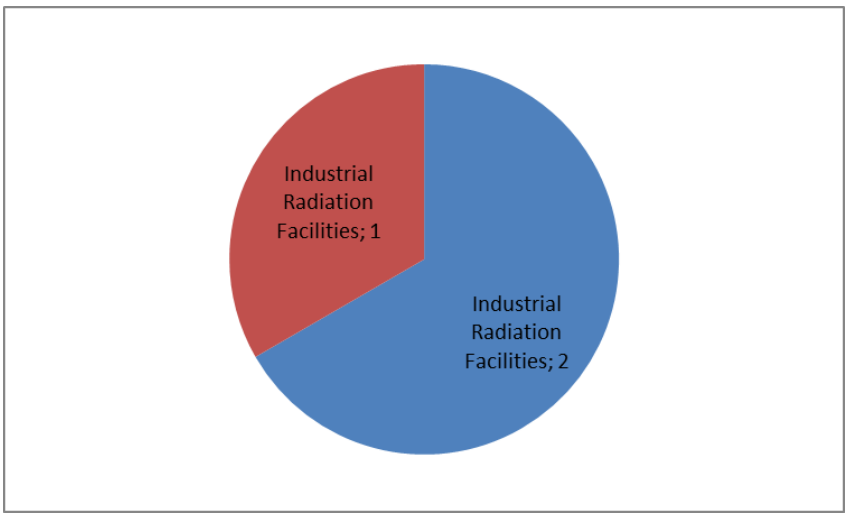

FIG. 7. Practice Industrial Radiation Facilities vs Educational Institutions / Training

3. CONCLUSION

The compilation of the results of all practices relating them to the Radiation Protection Office - RPO's and educational institutions and training and identification of the needs for the training of professionals that meet the demand of the Brazilian states / regions centralizing efforts in real need of RPO's by practices.

This work will continue in a new Projects of the Latin American Region - RLA for the next four years, the RLA 9/075 - Strengthening the Regional Nuclear Sector and the Application of Nuclear Science and Technology for Development through Training and Facilitating Strategic Activitie, this project is still in the stage of analysis of the the compilation of data related to the identification of practices by states, regions and identifying demanada of the RPO's practices as well as the phase of identification of educational institutions and training in Brazil.

Following the compilation of data and the final analysis of the information related to educational institutions it is hoped to develop an integrated national strategy that meets the demand in the area of education and training in radiation protection.

REFERENCES

[1] RLA 9/065 – Supporting infrastructure for radiological protection through education and training.
[2] RLA 9/070 - Strengthening the education and training infrastructure and building competence in radiation safety.
[3] RLA 9/075 - Strengthening the regional nuclear sector and the application of nuclear science and technology for development through training and facilitating strategic activities.
[4] Brazilian Nuclear Energy Commission/CNEN site – The Brasilian Regulatory Body.

25 YEARS EDUCATION AND TRAINING IN OCCUPATIONAL RADIATION PROTECTION AND HOSPITAL SAFETY IN THE BIOMEDICAL ENGINEERING DEPARTMENT OF TEI OF ATHENS: EXPERIENCES AND PERSPECTIVES

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Abstract

This paper reports on the accumulated experience, during a quarter-of-a-Century, in training of BME-graduating students of the BME Department of TEI of Athens, in the fields of occupational Radiation Protection (RP) and Hospital Safety (HS), The educational approach begins with a "virtual walk" around the Hospital Departments, focusing on RP-issues and other HS-technical-managerial Standards, as well as, medical-managerial Guidelines and Protocols. The trainees are acquainted with the procedures and other cardinal components and aspects of the Quality Management System (QMS), in each major Department and the associated Radiation hazards, Chemical and Biological risks, Electromechanical Safety aspects etc. Beyond Lectures and digital Demos, hands-on Practicals related to RP-HS have been offered. More training was available, during the compulsory six-month on-the-job training in the BME-Department, in Hospitals, R&D Facilities and Companies. Further, numerous students have been involved, during their Thesis-preparation, in experimental routine or R&D projects, concerning Dosimetry, Quality Assurance, Acceptance testing, periodic Quality Control (QC) measurements, documentation and other additional advanced RP-HS monitoring of Locations, Patients and Staff. Finally, we have always defined the state-of-the-art, followed the RP-HS innovation-trail and predicted the Technology- and Market-trends, by evaluating numerous Industrial Property (IP) Documents filed, during the last 25 years.

1. INTRODUCTION

This paper reports on the accumulated experience, during a quarter-of-a-Century, in training of BME-graduating students of the BME Department of TEI of Athens, in the fields of occupational Radiation Protection (RP) and Hospital Safety (HS). The educational approach begins with a "virtual walk" around the Hospital Departments:

(a) The Outpatient Department.
(b) The Accident – Emergency Department.
(c) The in vitro Diagnostic Laboratories.
(d) The Medical Imaging Department.
(e) Surgery Department and the Operating Rooms.
(f) The Intensive Care Units.
(g) The Inpatient Wards.
(h) Rehabilitation Unit.
(i) Radiotherapy.
(j) Blood bank and Transfusion Medicine Units.
(k) The Supporting Facilities.

This "tour" describes the functional design of the Departments, focusing specifically on RP-issues and other HS-technical-managerial Standards, as well as, medical-managerial Guidelines and Protocols. The students are acquainted with the procedures and other cardinal components and aspects of the Quality Management System (QMS), in each major Department and the associated Radiation hazards, Chemical and Biological risks, Electromechanical Safety aspects etc.

Beyond Lectures and digital Demos, hands-on Practicals related to RP-HS have been offered. More training was available, during the compulsory six-month on-the-job training in the BME-Department, in Hospitals, R&D Facilities and Companies. Further, numerous students have been involved, during their Thesis-preparation, in experimental routine or R&D projects, concerning Dosimetry, Quality Assurance, Acceptance testing, periodic Quality Control (QC) measurements, documentation and other additional advanced RP-HS monitoring of Locations, Patients and Staff.

Finally, we have always defined the state-of-the-art, followed the RP-HS innovation-trail and predicted the Technology- and Market-trends, by evaluating numerous Industrial Property (IP) Documents filed, during the last 25 years.

2. RADIATION PROTECTION (RP) AND HOSPITAL SAFETY (HS) IN MAJOR HOSPITAL DEPARTMENTS

Radiation Protection is obviously extremely important in the Medical Imaging and the Radiotherapy (RT) Departments. However, the Surgery Department the Emergency Department, the Intensive Care Units and other Units, are also involved in occupational RP procedures.

Concerning Radiation Protection we offered training focused on:

i. Methodology for calculating the required shielding: Primary, scattered and leakage radiation.
ii. Architectural and constructional comments for shielding, adapted to the specific needs of the installed System, e.g. CT, Angiography, LINAC etc.
iii. Dosimetry, Spectrometry, Auxiliary Detectors and other RP-equipment.
iv. RP in Nuclear Medicine facilities: Functional design, RP shielding calculations (e.g. Hot-Lab).
v. Radiopharmaceuticals and other active Waste Management.
vi. Concerning Hospital Safety we offered training focused on:
vii. Chemical risks: Gaseous and air-borne pollutants, liquid and solid chemical waste Management.
viii. Biological risks: Nosocomial Infections, Disinfection, Sterilization etc. Adverse Events reporting.
ix. Electrical Safety: Earth and Enclosure Leakages, Defibrillation Energy, Grounding, E/M Field etc.
x. Mechanical Safety: Medical Gases, Ventilation, Air Conditioning,

3. TECHNICAL STANDARDS AND MEDICAL GUIDELINES

An essential part of the training aims to acquaint the students with the existing common Technical Standards and Medical Guidelines, Clinical Terminologies, Vocabularies and other forms of structuring and combining Technical and Medical-managerial knowledge [1]-[3], in

everyday work-routine, synopsized in Tables 1 and 2.

TABLE 1. SOME INDICATIVE AUTHORIZED ORGANIZATIONS AND WIDELY ACCEPTED TECHNICAL STANDARDS AND CLINICAL TERMINOLOGIES AND VOCABULARIES FREQUENTLY TAUGHT AND EMPLOYED IN OUR EDUCATION AND TRAINING PROGRAM

Medical – Managerial Standard and Guidelines setting Organizations	Examples of common Standards Guidelines Clinical Terminologies and Vocabularies
World Health Organization (WHO)	ICD-9/10 (other: SNOMED, LOINC, ABC, CCC etc.)
Intern. Commission on Radiological Protection (ICRP)	Relevant Publications (Radiology, Radiotherapy etc.)
International Society of Blood Transfusion (ISBT)	ISBT 128 Standard Technical Specification 4.1.0
American National Standards Institute (ANSI)	ASTM E2369-05 Continuity of Care Record
International Organization for Standardization (ISO)	ISO 13606-1:2008 EHR communication ISO 22870 Point-of-Care Testing ISO 13485 Design & manufacture of medical devices
European Committee for Standardization (CEN) and European Committee for Electrotechnical Standardization (CENELEC)	prEN-13940 European Standard for Continuity of Care and other mainly hardware-related Standards
HL7 (Health-Level 7)	CDA (Clinical Document Architecture)
ACR/NEMA300 (1985) today Digital Imaging and Communications in Medicine (DICOM)	Standard for handling, storing, printing, and transmitting information in Medical Imaging.
International Electrotechnical Commission (IEC)	IEC 60601-1 Medical Design Standards
Inst. of Electrical and Electronics Engineers (IEEE)	Numerous mainly hardware-related Standards

TABLE 2. FOCUSING ON SOME MAJOR ORGANIZATIONS INVOLVED IN THE STANDARD-SETTING PROCESS

Indicative Major Standard Setting Organizations (SSO)	Main Field of Activity
International Atomic Energy Agency (IAEA)	Scientific-technical cooperation in Nuclear Science
International Telecommunication Union (ITU)	Telecommunications
International Electrotechnical Commission (IEC)	Electro-science & Technology
International Organization for Standardization (ISO)	General except ITU/IEC subject-matter
Institute of Electrical and Electronics Engineers (IEEE)	ICT, Power, Energy, Nanotechnology etc.
European Telecommunications Standards Institute (ETSI)	Telecommunications
American National Standards Institute (ANSI)	Accreditation

A Technical Standard is an established norm or requirement in regard to technical systems. It is usually a formal document that establishes uniform engineering or technical criteria, methods, processes and practices. On the other hand, a Medical Guideline is a document with the aim of guiding decisions and criteria regarding diagnosis, management, and treatment in specific areas of healthcare. Their combined employment is a necessary prerequisite to achieve high-quality Health-Care.

4. RP AND HS COMPONENTS AND ASPECTS OF A QMS

A Quality Management System (QMS) in e.g. a Hospital is a collection of Processes/Procedures, focused on achieving its Quality policy and Quality objectives. The Quality Manual (QM) is an official document produced by e.g. a Hospital or Department that details how its QMS operates. Thus, RP and HS constitute important components and aspects of a QMS [4]. The starting-point for training on QMS is the relevant Standards & Guidelines, as mentioned previously (WHO, IAEA, IEC, ISO etc.). An overview of our RP and HS

related QMS-Training-Outline is presented in Table 3.

TABLE 3. OVERVIEW OF OUR RP AND HS RELATED QUALITY MANAGEMENT SYSTEM TRAINING-OUTLINE

Quality Assurance (QA) procedures - Quality Manual	Acceptance tests & routine Quality Control (QC) Data-Documentation	Additional Safety monitoring of Persons and Locations
Responsibilities for RP and HS related Quality Assurance	Major Biomedical Equipment acceptance and periodic testing.	Persons/Locations Dosimetry, E/M, Chem. & Biol. Exposure
Documentation: Imaging & RT Equipment, Test-objects etc.	Periodic Quality Control of each Unit's Systems and Components	X-ray/γ Energy Spectra experimental determination
Tests modalities and procedures in the case of abnormal results	Accuracy of image registration, visual display, copy printing etc. if applicable	Experimental verification of the RP X-ray/γ dose calculations
Records of all tests, calibrations and corrective actions performed	Adverse events Documentation and Reporting to the competent Authorities	Magnetic-field and REMF compliance measurements
Staff-training documentation (Equipment, QC) procedures etc)	Setting Quality Objectives and Quality Indicators in the considered facility	Software for QMS technical-managerial data-handling

5. ON-THE-JOB TRAINING AND THESIS-PREPARATION RELATED TO RP AND HS.

Hundreds of on-the-job training projects and final-project Thesis have been carried-out in our Laboratory, during the last 25 years. A lot of their results have been published in Journals and in International Conference Proceedings with Referral System.

Among totally over 350 contributions (236 in English and 116 in Greek), about 38% of them (in English) are projects, mostly experimental, related to RP and HS.

This close relation between Education, Training and "real world" job-assignments (cf. Table 4) was the main reason that the graduates of Biomedical Engineering Department of TEI of Athens, experienced until the great crisis (2010), no unemployment at all.

TABLE 4. SOME OF THE MEASURING DEVICES EMPLOYED DURING THE PERFORMED RP RELATED TRAINING

kV meter/ timer	High Voltage Tank Blinder	mA/mAs meter, up to 2000 mA, up to 3 sec	Multimeters Oscilloscope	CdTe Xray Spectral Analyzer
Ionization Chambers	Oscilloscope A/D Autoset	Mammography Dose-meter up to 40 kV (up to 20 mGy)	Photo-sensitive plates	Solid-State Dose-meter

6. TRENDS IN OCCUPATIONAL RP AND HS AS DEPICTED ON IP-DOCUMENTS.

We have always defined the state-of-the-art, followed the RP-HS innovation-trail and predicted the Technology- and Market-trends [5], by evaluating numerous Industrial Property (IP) Documents filed, during the last 25 years. Some indicative results are:

TABLE 5. SOME INDICATIVE AND RELEVANT PATENT DOCUMENTS DISCLOSING VARIOUS DETECTORS GROUPED ACCORDING TO THEIR PHYSICAL PRINCIPLE OF OPERATION	
Detector Type	***Selected recent indicative Patent Documents***
Semiconductor based avalanche photodiode (APD)	US20090094180, US20090093710, US20090134334 US20090242773 US20080214927, US20080230704
Silicon photomultiplier (SiPM)	US20100010343, US20080121806, US20080203309
Indirect Flat Panel Detectors (FPD) Amorphous Si	US20090095914, US20090108209, US20090127470 US20090014659 US20090202038, US20090202044 US20080240366, US20080104835
Direct Flat Panel Detectors (FPD) Amorphous Se or Ge	US20090159806, US20090084966, US20080006787
Charge coupled device (CCD)	US20080198969, US20080035852, US20070230659
Complementary metal-oxide Semiconductor (CMOS)	US20090181491, US20090108311, US20080217545

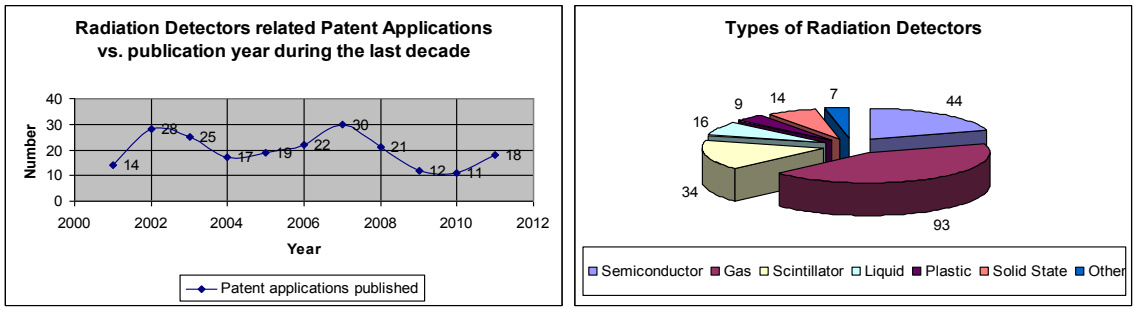

FIG. 1. Patent Applications filed per year concerning Radiation Detectors (left) and their distribution in the most important categories according to the main physical principles (right).

7. CONCLUSIONS

High-quality RP and HS demand Technical Standards, Medical Guidelines and Experimentation, all combined in a QMS, tracing of Innovation through Patent-searching and systematic hands-on-training.

REFERENCES

[1]. B. SPYROPOULOS, Combining ICT-Standards Essential-Patents and Medical-Managerial Guidelines towards sustainable Assisted Living and Home-care, ITU-Kaleidoscope 2014, St. Petersburg, Russian Federation (2014).

[2]. B. SPYROPOULOS, Biomedical device standards vs. industrial property rights? Towards a fair solution of the health-care related standard essential Patents' licensing problem 2nd WHO Global Forum on Medical Devices, 22-24 November 2013, Geneva, Switzerland (2013).

[3]. B. SPYROPOULOS, Combining Medical-Managerial Information and Communications Technology Standards to promote interoperability in health-Care: A Status Report, ITU-WHO e-Health Workshop, Geneva (2012).

[4]. B. SPYROPOULOS et al. A web-based system supporting the Certification of a Surgical Department in a general hospital, Proc Conf IFMB 2011 Budapest, Springer (2011).

[5]. B. SPYROPOULOS, Trends in Patients' Radiation Protection: Manufacturers' research strategy as outlined in Industrial Property Documents, IAEA Int Conf RP Med, Bonn, Germany, 3–7 December (2012).

EST-RAD: A REGIONAL NETWORK OF PROFESSIONALS DEDICATED TO THE OPTIMISATION OF OCCUPATIONAL RADIATION PROTECTION

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Abstract

When optimising radiation protection for any area of work, meeting the challenge of coordinating a radiation safety program is a far from easy task. Due to the wide variety of procedures that use radioactive sources or radiation-producing equipment, a significant amount of education and professional experience is required to ensure a continued safe working environment. In whatever field; radiation medicine, industry or research, relevant competence is a core concept at the heart of both professional development and the legal status of radiation protection officers (RPO). Certainly their commitment to lifelong learning is the key to their recognition by national authorities. The challenge for professional organisations is therefore to offer RPOs educational opportunities that can meet these learning requirements, thus providing them with the specific professional skills required to continue to work efficiently and knowledgeably. EST-RAD network was founded in 2013 as a non-profit association for radiation protection to provide the field with a multi-sector structure aware of RPO's specific concerns. EST-RAD aims at keeping professional knowledge up-to-date by disseminating information and sharing experiences as well as by promoting radiation protection and developing a multidisciplinary team approach to incident prevention in order to deal with the changes required by legislation.

1. INTRODUCTION

Council Directive 96/29 EURATOM (1) requires a "qualified expert" (QE) to be assigned technical responsibility for the radiation protection (RP) of workers and members of the public. Requirements for QE training, experience and recognition are also specified. Nevertheless, RP experts' curricula and their ability to perform certain RP tasks vary greatly from one member state to the other. In France, the legal framework was updated in 2003. Certification was introduced at that time that established a nationally recognized minimum standard of relevant knowledge and practical training and with the requirement for periodic renewal every five years. Those passing the examination are entitled to call themselves "Personne Compétente en Radioprotection "(PCR) - designated here as radiation protection officer (RPO). They can then be appointed to perform certain tasks, including supervision of practices and appropriate training of staff in radiation protection, and will play a key role in its organisation in their institution.

2. ABOUT EST-RAD : AIMS & ONGOING PROJECTS

Nationally, about 20,000 individuals from various specialties - radiographers, medical physicists, medical practitioners, educators, radiation protection technologists, etc…- are involved in the optimisation of occupational radiation protection. Wherever they work, many

of them will feel isolated with their own problems and having no obvious means by which to update their knowledge in a timely manner is one of their main difficulties. Joining a RP regional network is therefore an ideal opportunity to facilitate communication between members having very similar concerns. The association EST-RAD was created in 2013 by twelve RPOs working in the medical field. Its main goals are to promote radiation protection in all work areas, to facilitate colleagues sharing experiences so that they feel less isolated and to develop high standard educational opportunities at low cost for RPOs on a regional basis. Benefits are in the greater interchange of experiences and in encouraging a better status for professionals within the national RP community. EST-RAD was the fourteenth network joining the national Coordination of RP network coordinators (CoRPAR) in October 2013.

In less than one year, membership has grown to 60 individual members from various sectors – medicine, industry, research, and professional education - all of them willing to share their differing experiences to the benefit of their institutions.

80 % work in the medical field. Colleagues from industry who support educational activities are now accepted as "associate members" so that registration fees can be kept at a minimum. EST-RAD is governed by five volunteer officers elected for 2 years. All projects and activities are organised by a group of members in close collaboration with local competent authorities. Some members with extensive experience in their field may also take part in working groups at national level as experts.

Multidisciplinary teamwork and open and prompt communication are the keys of EST-RAD success. E-newsletters are sent quarterly to keep members aware of the latest information related to their working environment. For the moment, educational activities consist of two annual continuing education sessions:

a) A one-day programme including an overview of latest regulatory issues and new publications & guidelines. This day may also include developments in radiation protection overall and/or individual equipment presented by industry leaders. Case studies of particular investigations or risky situations are presented with sufficient time for open discussions ;
b) A one-day seminar dedicated to a single topic of interest including lectures by experts and practical "How to do sessions" in small groups. These are to train the trainers who will then be able to replicate these sessions in their own institutions. All invited speakers and workshop supervisors are acknowledged as specialists in radiation protection.

For the past year, more than 100 people have been using the EST-RAD network to keep in touch. Their feedback confirms that the services on offer meet their needs. Dealing with the protection of the health of workers is an iterative process that can be influenced by every member of a multi-faceted team working together.

REFERENCE

[1] COUNCIL DIRECTIVE, 96/29 EURATOM of 13 May 1996 laying down the Basic Safety Standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiation (1996).

REPROLAM NETWORK
SHARING KNOWLEDGE AND LESSONS LEARNED IN LATIN AMERICA

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Abstract

At the first International Conference on Occupational Radiation Protection: Protecting Workers against Exposure to Ionizing Radiation which took place in Geneva, Switzerland in August 2012, it was decided that an action plan with 14 actions, should be developed to improve the occupational radiation protection in Member States. Because the creation of various forums and networks in the free world of the internet, was specified an action, Action 7, with the theme: To provide a focal point for exchange of information through networking, since due to the ease of access to the internet disseminating untrue or even inaccurate information could cause great distrust of the workforce in the authorities and employers. So the IAEA and ILO, in order to meet the Action 7 of the Action Plan, has created and maintains the site ORPNET and encouraged the creation of regional networks for optimizing occupational radiation protection. Several regional networks were created, such as RECAN, ARAN, REPROLAM (Red de Optimización de Protección Radiológica Ocupacional en LatinoAmérica). This paper presents the activities proposed and conducted by REPROLAM, their difficulties and successes.

1. INTRODUCTION

The first International Conference on Occupational Radiation Protection was held in Geneva from 26 to 30 August 2002. It was organized by the International Atomic Energy Agency (IAEA), which convened the Conference jointly with the International Labour Office

(ILO). It was co-sponsored by the European Commission (EC) and held in cooperation with the World Health Organization (WHO) and OECD Nuclear Energy Agency (NEA) and a number of other international organizations. The Conference produced a number of important findings and recommendations. These were considered in September 2002 by the IAEA General Conference, which requested the IAEA's Director General, in cooperation with the ILO and other relevant bodies, to formulate and implement an action plan.

The IAEA and ILO prepared a draft that was reviewed by the organizations and key participants involved in the Geneva Conference as well as by the International Confederation of Free Trade Unions (ICFTU) and the International Organisation of Employers (IOE). The Action Plan was approved by the IAEA Board of Governors on 8 September 2003. In order to ensure the successful implementation of the Action Plan, the IAEA and ILO agreed to establish a Steering Committee (SC) with the overall remit to advise on, monitor and assist in the practical implementation of the International Action Plan (IAP). The First, Second and Third Meetings of the Steering Committee were held in Vienna on 4-6 February 2004, 25-27 January 2006, and 28 – 30 January 2008 respectively (Report on the Fourth Meeting of the Steering Committee for the International Action Plan on Occupational Radiation Protection - IAPORP).

2. ACTION PLAN

The Action plan, with 14 actions, was implemented in 2004 and had the following chronology (IAEA, Action Plan for Occupational Radiation Protection, 2002):

August 2002 (26th/30th): the IAEA/ILO Geneva international conference on occupational exposures set up the scene, pointed out problems and made recommendations for further improving the situation

September 2003 (1st/5th): the IAEA international conference on national infrastructure highlighted again in Rabat some of the Geneva recommendations

September 2003 (8th) the IAEA Board of Governors approved the 14 actions of the Action plan

February 2004: first IAPORP Steering Committee for precizing who do what and how
2004- 2011 : IAPORP implementation and follow up (Steering Committee Jan 2006, Jan 2008, Feb 2010) Closure June 2011

The Action 7

The action 7 is "The IAEA to provide a focal point, on a website, where networks may be established for exchanges of information, experience and lessons learned between interested parties" (IAEA, IAORP Fifth Steering Committee Meeting, 2007).

A result of this action was created a website called Occupational Radiation Protection Network (ORPNET) (IAEA, IAEA ORPNET) where is posible to access all networks related to occupational radiation protection.

The ORPNET website have information to access to many regional and woldwide websites linked to occupational radiation protection between them REPROLAM website and meny other like Asia Regional ALARA Network (ARAN), European ALARA Network

(AEN), Regional European and Central Asian ALARA Network (RECAN) and other local and topic specific networks.

REPROLAM – Red de Optimización de Protección Radiológica Ocupacional en LatinoAmérica

REPROLAM is the regional Network for Latin America, which was born under the Technical Cooperation Project RLA/9/066 "Strengthening and upgrading the technical skills to protect the health and safety of workers occupationally exposed to ionizing radiation", where the participants countries were agreed to study the feasibility of creating a regional network for optimization of occupational exposure in Latin America.

This activity was sponsored by the IAEA in the implementation of the International Plan of Action on Occupational Radiation Protection.

A coordination and promoting group was created with participation of representant of Argentina, Brazil, Costa Rica, Peru and Uruguay to coordinate the preparatory actions. This group met in the *Instituto de Radioproteção e Dosimetria* (IRD), Rio de Janeiro, Brazil, in June 2011, where the optimization Red Occupational Radiation Protection in LatinAmerica (REPROLAM) was founded. During this meeting was elaborated the REPROLAM Statute and an action plan for the next years.

After that, two more meeting were held to evaluate the previous plans and correct the routes.

The main activities developed by REPROLAM are (REPROLAM, Estatuto de la Red de Optimización de Protección Radiológica Ocupacional en Latinoamérica, 2011):

i. Facilitate the exchange of information and integrated into the practical aproach of the principle of optimization of occupational radiation protection.
ii. Contribute to the harmonization of policies and practices of occupational radiation protection, particularly in relation to the principle of optimization in the different components of the national infrastructure: users of radiation sources, scientific and technical support services and regulatory authorities.
iii. Maintain, improve and develop levels of competence in radiation protection with special emphasis on the application of the optimization principle in cases of occupational exposure, normal exposure situations and emergencies.
iv. Contribute to the integration and cooperation in relation to knowledge and specialized services in occupational radiation protection.
v. Identify and investigate significant topics of common interest to implement in processes of optimization of occupational radiation protection.

3. RESULTS

REPROLAM have been working to establish a group of national representantative for each country of the region, actually REPROLAM have eleven representants (Argentina, Brasil, Bolivia, Chile, Costa Rica, Honduras, México, Nicaragua, Peru, Uruguay and Venezuela), which have a task to promote the dissemination of the information generated by REPROLAM. It is neccesary to ensure that the network grows up and incorporate representants from another countries of the region.

In the year 2013, REPROLAM published in its website a translations to spanish and portuguese of seven poster (REPROLAM, Posters, 2012) developed by ORPNET (IAEA, Radiation Protection of Workers: Approved Posters, 2011), this poster's were publised to disseminate basic knowledge about optimization of occupational radiation protection of

target areas like Diagnostic Radiology, Industrial Radiography, Nuclear Medicine, Radiotherapy, Industrial Irradiators, Nuclear Gauges and Radioactive Tracers. The use of these poster's have been to promote by mail, list of professionals of radiation protection, in conferences and workshops of ocupational radiation protection.

An important contribution from REPROLAM is the translations to spanish and portuguese of the frequently asked questions on ALARA published by ORPNET (IAEA, OPRNET: Frequently asked questions on ALARA, 2010), this tranlation will be published in the REPROLAM webpage in the next months.

The activities of REPROLAM in the last year have been to disseminate the observation performed in the Technical Cooperation Project RLA/9/066, with the objective to make known an state of art of the occupational radiation protection in Latin America, this information will be useful for end users of radiation sources, scientific and technical support services and regulatory authorities, to stablish focal points for attending and to know the actual state in each country of the region, actually we are working in the edition of report and we are going to this information to public access in the end of this year.

Among the tasks to be performed in the future are: the organization of specific workshops for eac target group (end user of radiation sources, scientific and technical support services and regulatory authorities), with the intention to focus the action in the region, and improve the radiation protection condition of the end user of radiation soucer especially.

4. CONCLUSION

In accordance with the Action Plan for Occupational Radiation Protection of the IAEA, Latin America created its own network to interchange experience and information in the area of occupational radiation protection. Since 2011, many action had been developing with the intention to strengthen the structure of the organization and to disseminate very useful information in spanish and portuguez about topics of interest.

It is neccesary to promote the participation of all countries of the region, and to have more participation of experts and end users of radition sources, all of these things are going to be developed through the organization of workshops and new publications.

BIBLIOGRAPHY

IAEA, Action Plan for Occupational Radiation Protection. Geneva: IAEA, Vienna (2002).

IAEA, IAORP Fifth Steering Committee Meeting, (2007), Retrieved 06 01, 2014, from http://www-ns.iaea.org/tech-areas/rw-ppss/iaporp-fifth-meeting.asp

IAEA, International Action Plan on Occupational Radiation Protection. Progress Report 2007 IAEA, Vienna, (2007) 1-5.

IAEA, OPRNET: Frequently asked questions on ALARA, (2010, 03), Retrieved 06 01, 2014, from http://www-ns.iaea.org/tech-areas/communication-networks/orpnet/faqlist.asp

IAEA, Radiation Protection of Workers : Approved Posters, (2011, 05), Retrieved 06 01, 2014, from http://www-ns.iaea.org/tech-areas/communication-networks/orpnet/radiation-protection-posters.asp IAEA, IAEA ORPNET. (n.d.), Retrieved 06 01, 2014, from http://www-ns.iaea.org/tech-areas/communication-networks/orpnet/default.asp

REPROLAM, Estatuto de la Red de Optimización de Protección Radiológica Ocupacional en Latinoamérica. Rio de Janerio (2011).

REPROLAM, Posters, (2012), Retrieved 06 01, 2014, from http://reprolam.ird.gov.br/index.php/documentos/Posters/Spanish/

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