International Conference on

RADIATION SAFETY

Improving Radiation Protection in Practice

EXTENDED ABSTRACTS

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Please note that this is a compilation of the extended abstracts which were accepted for oral and poster presentation. They have not been edited.

7-THE EFFECT OF THE AIR CONDITIONING SYSTEM OPERATION ON RADON CONCENTRATION AT THE RADIOACTIVE WASTE INSTALLATION IN INDONESIA

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ABSTRACT

Most people spend their time indoors where the level of harmful pollutants is often much higher than in their homes. Radon (222Rn) is radioactive pollutants and the main source of ionizing radiation. The purpose of this study is to determine the impact of air conditioning systems on Radon concentration levels that contribute to the effective dose. The average value of long-term tests (1 year) is normally used in Radon assessment studies. However, the results of short-term tests can be used for initial screening or to perform follow-up test when the results of long-term test exceed the limit values. Measurements were carried out at Radioactive Waste Installation (RWI) in BATAN on 2019/02/01. Radon measurements were carried out with indoor air parameters of two operating the air conditioning system modes e.g operation and not operation. The levels of air supply and air recirculation are considered. The effect of the air conditioning system turned off, the Radon concentration is very large with a graph that tends to increase ranging from 21.25 Bq/m^3 - 30 Bq/m^3. The effect of the air conditioning system in operation, the Radon concentration is very small and tends to decrease in the range of 16.71 Bq/m^3 - 1.05 Bq/m^3. The results of the effective dose estimation of indoor Radon airborne contamination levels use the maximum Radon concentration when air conditioning system is not operated 0.432 mSv/years. These results indicate that the concentration of Radon indoor at RWI-BATAN is below 300 Bq/m^3 and less than dose limit, so that it is still safe to work. Radon is one of the contributors to a dose of natural radiation large enough to reduce the risk of lung cancer. The most likely effort is to regulate the ventilation or air conditioning system. This is needed to minimize the concentration of Radon in the room.

BACKGROUND

In order to increase safety control, the management of radioactive waste is done by monitoring the level of air concentration before release to environment. The parameter of monitoring consists of level of concentration of alpha, beta, radon and thoron. This paper discusses the effect of the air ventilation system on the levels of Radon concentrations that contribute to effective doses of workers. The effect of Radon concentration levels during air ventilation when operated or when turned off. In addition, it discusses the comparison of permissible radon concentration limits based on international regulations.

THEORY AND METHODS

In order to monitor air contamination, BATAN uses the Canberra iCAM Alpha Beta Air Monitor which is equipped with a detector to measure the level of contamination or the concentration of radon in Bq/m^3 by real-time system. If the result of measuring the concentration of radon in air in the form of a concentration is known, then the effective dose is calculated using the following equation [1]:

$$DR_n = F_k R_n \times FR_n \times T \times CR_n \quad \text{[mSv/years]} \qquad (1)$$

with:

DR_n = Effective dose due to inhalation of Radon and/or deceased daughter [mSv / year]

F_kR_n = Radon equilibrium factor with its whole daughter (0.4)Natural circulation
FR_n = Radon dose conversion factor $[9\frac{\text{nSv}}{\text{Bq}}.\frac{\text{hours}}{m^3}]$
T = Length of stay in the room $[hours/years]$, total effective working hours of 2000 hours in 1 years
CR_n = Radon gas concentration in the room $[Bq/m^3]$

RESULTS AND DISCUSSION

The results of the estimation of the effective dose using the maximum Radon concentration when air conditioning is not operated at 06:00:00 as follows:

$$DR_{n\,max} = 0.4 \times 9\,\frac{\text{nSv}}{\text{Bq}}.\frac{\text{hours}}{m^3} \times \frac{8\,hours}{2000\,hours} \times 30\left[\frac{Bq}{m^3}\right] = 0.432\ [\text{mSv/years}]$$

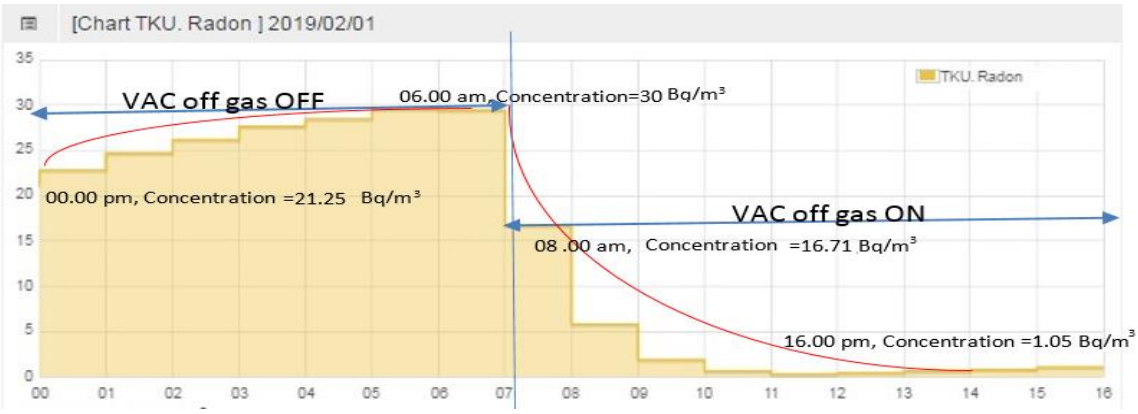

FIG. 1. Effect of Air Conditioning System (VAC off gas) on Radon Concentrations

From the measurement results of Radon based on the graph in FIG. 1, the effect of VAC air ventilation off gas when not operated on the concentration of Radon is very large with a graph that tends to increase around 21.25 Bq/m^3 - 30 Bq/m^3, while the effect of Air Conditioning System (VAC off gas) when operated on Radon concentrations is very small with a graph that tends to decrease in the range of 16.71 Bq/m^3 - 1.05 Bq/m^3 [2].

Indonesia through the Nuclear Energy Regulatory Agency (BAPETEN) has issued the highest permissible concentration limit for radon gas in the air (environment) of 400 Bq/m^3, but BAPETEN have been no regulations regarding the concentration limits of Radon indoor. For this reason, BATAN follows the latest regulations IAEA GSR Part 3 (2014) indoor 300 Bq/m^3 and outdoor 1000 Bq/m^3 and ICRP 103 (2007) with an indoor radon limit of 300 Bq/m^3, outdoor 1000 Bq/m^3 [3].

CONCLUTIONS

From the results of Radon measurements, the effect of air condition (VAC off gas) when not operated on the concentration of Radon is very large with a graph that tends to increase around 21.25 Bq/m^3 - 30 Bq/m^3, while the effect VAC air vents off gas when operated against Radon concentrations are very small and tend to decrease in the range of 16.71 Bq/m^3 - 1.05 Bq/m^3. The results of the estimated effective dose of workers using maximum Radon concentration when VAC off gas is not operated 0.432 mSv/years. These results are still far below the limit so it is said to be safe for workers. Radon is one of the contributors to the dose of natural radiation which is large enough to reduce the risk of lung cancer, it is necessary to minimize the concentration of radon in the room [4]. The most feasible effort that can be done is to operate and regulate the ventilation system or ventilation VAC off air gas as possible.

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8-ESTABLISHMENT AND USE OF DIAGNOSTIC REFERENCE LEVELS (DRLS) IN MEDICAL IMAGING IN UGANDA

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Objective

To proper establish and use Diagnostic Reference levels concept in diagnostic, interventional radiology and nuclear medicine in the imaging of both Pediatric and adult population.

Background

Various patient groups provided the stimulus for monitoring practice to promote improvement in patient protection. Different imaging modalities highlighted the substantial variations in dose between some health care, facilities indicated the need for standardization of dose and reduction in dose without comprising the clinical purpose of each examination or procedures.

DRLs will provide opportunity to provide feedback to ensure good practice in Medical exposures is maintained. Image quality consistent with the medical imaging task will be ensured. DRLs will provide a benchmark for comparison not to defence maximum and minimum dose. DRLs will be applied with flexibility to allow higher doses when indicated and justified be the radiologists

Methodology

Patient and radiation doses were retrospectively collected from August 2017- August 2019 for 40 consecutive patient treated with a system using a state of the art image processing and reference acquisition chain. Radiation dose was quantified using Dose area product (DAP) while procedure complexity using fluoroscopy time, procedure duration, fluoroscopy time, procedure duration, dose. Calculation for each type of examination the median (mean) values of dose quantities was obtained.

Results

Graphs of mean fluoroscopy time, dose area product and effective dose per type of intervention and diagnostic procedures, mean cine frames aided the analysis in the study.

Quantification at 50% ,40 frames ,10 exposure, (750,1000) mGy,10-20minutes, (1000-1800)micro Gycm^2 was observed for percutaneous (PCI) and coronary angiography respectively. **Quantification at 70%** for PDA for Peaditric was 200cGycm^2 for <20kg.

Conclusion

DRLs have intended to promote improvement in patient protection by allowing comparison of current practice, national and local DRLs. When set for each examination or procedures, for each clinical indication and each patient group will represent the most frequent examinations performed in Uganda for which dose assessment will be practicable and will have approved a useful tool in support of dose audit and practice review for promoting improvement in patient protection.

10-ARCHITECTURAL STANDARDS AND RADIATION PROTECTION CRITERIA AS A BASIS OF ENVIRONMENTAL DESIGN THE NUCLEAR MEDICINE UNITS TO REDUCE UNJUSTIFIED DOSES

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This study aimed is to implement the principle ALARA, and Achieve quality of design in the field of radiation protection, by improving a draft for radiation protection in nuclear medicine that meets the current international recommendations. The draft includes the principles of safe design of nuclear medicine departments, and methods of radiation protection for workers, patients. The practical part, what is stated are the corresponding checklist in the inspection file. The conclusion of this study expose that reviewed the proposal correction are given that could improve the goodness of radiation protection in the nuclear medicine department

The IAEA guidelines describing the buildings for the essential equipment of a basic radiotherapy clinic recommend are suggested here. Shielding requirements are met with walls of thickness equivalent to 230 mm of solid brick or concrete, and lead-lined sliding entrance doors, which is standard for diagnostic facilities. windows for the operators should be lead glass and embedded into the wall structure. The inner room dimensions should be the same as for the EBRT bunkers (structurally 7 m × 7 m × 4 m high)

ORIGINALITY PUBLICATION

* IAEA Human Health Reports No. 10, Radiotherapy Facilities Master Planning And Concept Design Considerations International Atomic Energy Agency, Vienna, 2014

THE BASICS OF SAFE DESIGN OF NUCLEAR MEDICINE UNITS

Nuclear medicine particular plan have areas performance where the equipment will be installed in the surrounding areas and they're objective including floors above and below. Surface finishes and ventilation also to be included lighting, ventilation, plumbing and drainage.

CONSIDERATIONS

- Location and Relationships, and the room must have radiation shielding to walls, Lead lining and glazing as advised.
- The minimum ceiling height shall be 2.39 meters (7 feet 10 inches), wall finish treatments shall not create ledges or crevices that can harbor dust and dirt, plastic, break-resistant material.
- All doors between corridors, rooms shall be of the swing type or shall be sliding doors.
- Each patient shall have access to a toilet room without having to enter a corridor.
- The floors and walls should be anti-static heat resistant anti-bacterial, anti-infection.
- Adequate ventilation and air exchange (with at least 25 air changes per hour as per American Society of Heating, Refrigerating and Air-Conditioning Engineers.
- High-efficiency particulate air nominate delivered at or near the ceiling, exhaust outlets located near floor level, bottom exhaust outlets should be at least 75mm above the floor. Room temperature shall be maintained 18-22 °C with room humidity 35-70% .
- Toilets air pressure should be kept negative pressure with respect to any adjoining areas and should have minimum 10 air changes per hour. Sterilizing area air pressure should be kept negative pressure with respect to any adjoining areas and should have minimum 10 air changes per hour.

RADIATION PROTECTION METHODS

LOCATION

A ground floor site is preferred but if this cannot be achieved, consideration should be given to units above, below and adjoining the proposed location with regard to radiation shielding requirements.

UNIT LAYOUT

layout staff and patient flows in the department are critical to include, that patients staff and visitors are not exposed to radiation, in areas occupied by dosed patients. An efficient plan can also lessen the need for costly radiation protection and separation of areas particularly patient and staff corridors and entry areas.

PATIENT WAITING

Waiting areas must be let separation of dose and undoes patients, in particular as some patients might need to wait for 45 minutes after dosing for uptake. It is also preferable to separate dosed patients from relatives and visitors to the unit which may include young adults, pregnant women, and children. Dosed patients should have access to drinking water and toilet facilities without having to access general waiting areas.

RADIATION PROTECTION METHODS REQUIREMENTS

- Distance is an important means of protection and area should be considered. In general, a minimum distance of 2 meters between the patient and the operator should be achieved.
- Structural shielding must restrict the potential radiation doses to radiation workers to not more than 10% of the effective dose limit. (For the public 0.5 mSv per .
- Sheet lead is used, the sheets must be overlapped by a minimum of10 mm (or the joins covered with strips of the same thickness or lead).
- Layout and relative proximity of imaging, injection and laboratory rooms must be as directly as possible and avoiding traffic paths.

RECOMMENDATIONS

• Establishing and enhancing the available environmental design capacity regarding education and training of radiation workers on radiation protection.
• Reviewing this draft of code of environmental design practice by the regulatory authority and radiation protection experts as the first step towards issuance of the code.
• Carrying out more inspection mission by the regulatory authority followed by taking the correct enforcement actions.
• The SAEC has to establish an information system aiming to raising the awareness of the public and professional about radiation hazards and protection.

CONCLUSION

Draft of a new code of practice for radiation protection in nuclear medicine has been developed. The current status of radiation protection in such department is not satisfactorily as many of the essential items of a radiation protection issues are not applied. This situation is likely will lead to exposing patient, workers and public to unnecessary radiation doses.

Preconstruction Design Issues Dose constraints for staff and public must be adopted in designing the facility, Layout of department should be considered. Direct lines of sight between resting areas and staff areas should be eliminated.

Post-construction Design Issues after construction, if actual measured exposure levels are too high, shielding must be increased or other corrective measuring taken.

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11-MEASUREMENT OF RADON AND RADON PROGENY IN ABU-TARTOUR OPEN-PIT PHOSPHATE MINE IN EGYPT

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ABSTRACT

In this work, Abu Tartour open-pit Phosphate mine and underground tunnels have been studied from radiation safety point of view. The plateau is situated in the Southwestern sector of Egypt in the Western desert at 300 km west of Assiut city at River Nile, 650 km south of Cairo city and 700 km west of Safaga port at the Red Sea coast. Radon, Radon daughters and Thoron daughters were measured for 31 air samples collected from 5 main sites that represent the process of phosphate mining. Radon, Radon daughters, and Thoron daughters concentration results were ranged from 1.377 to 131.520 Bq/m, (0.000099 to 0.001138) WL, and 0 to 0.92 Bq/m, respectively in case of open-pit mining. For postpone tunnels, Radon, Radon daughters and Thoron daughters concentration results ranged from 411.545 to 2539.274 Bq/m, 0.056 to 0.37 WL, and 1.26 to 3.89 Bq/m, respectively. Furthermore, 6 soil samples were collected from each site to measure the natural radioactivity that were ranged for U-238 from 153$\pm$9 to 222$\pm$11 Bq/kg. From 8$\pm$1 to 20$\pm$2 Bq/kg for Th-232 and from 30$\pm$3 to 80$\pm$4 Bq/kg for K-40. Also, assessment of radioactive surface contamination together with gamma dose-rate had carried out. The effective annual dose expected for the workers in the mine was calculated and ranged from 0.033 to 0.153 mSv/y for open-pit and 4.948 mSv/y for underground mine. It can be concluded that, open-pit mining method is much safer from the occupational radiation protection point of view.

INTRODUCTION

Phosphate mines in Egypt are distributed through the eastern and western desert where El Hamraween, Al Quser and Safaga mines are located in eastern desert 500 km south of Cairo. Abu-Tartour mine is located in the western desert 650 km south Cairo on El-Dakhla Oases road, 50 km west of El-Kharga city, capital of New Valley Governorate, Egypt. The plateau is situated in the South-western sector of Egypt in the Western desert at 300 km west of Assiut city at River Nile, 650 km south of Cairo city and 700 km west of Safaga port at the Red Sea coast. Abu-Tartour plateau forms a part of the rugged stretch that separates Dakhla and Kharga Oases in the Western Desert of Egypt [1], it is a close cast mine. The current reserve estimate in the exhaustively investigated area is in the order of a billion tons of Phosphate ore. The planned ore rock annual production is 4 million tons which processed to produce 2.2 million tons per year of wet rocks. Some studies discussed the radon concentration in some Phosphate mines. A typical concentration of ^{238}U in sedimentary Phosphate deposits is 121 mg/kg (1500 Bq/kg) with a range of 30–260 mg/kg (372–3224 Bq/kg) [2]. The uranium contents of some Egyptian Phosphate rocks in the Red Sea coast and several Nile valley sites are in the ranges of 19–142 mg/kg (235–1761 Bq/kg) and 48–185 mg/kg (595–2294 Bq/kg), respectively [3]. The average ^{238}U content in Abu-Tartour Phosphate rock is about 32.9 mg/kg (408 Bq/kg) [4].

EXPERIMENTAL WORK

RDA-200 is used for measuring the ambient levels of Radon and Radon daughter concentrations. The alpha particles register on the ZnS (Ag) phosphor coating of the scintillator cell or tray in the form of light flashes. Each flash of light, is transformed into an electrical impulse. These impulses are accumulated, counted and then digitally displayed by the scaler circuitry, after a present counting time has been completed. A small battery-powered air pump was used for the collection of Radon, Radon daughters, Thoron daughter's samples, which has a flow rate ranges from 1-10 L.min^{-1}. Lucas cells (PYLON type RN-150A) with a volume of 160 mL were used in this work. The cell is equipped with two valves, one for a small pump connection to draw the air sample through the cell (outlet valve) and the other for replacing the air in the cell with air from the environment (inlet valve). Scintillator trays were used to measure the concentration of radon daughters by counting the deposited particles on the filters using EDA-200. A Millipore type 25 mm diameter with 0.8 micron Millipore filter disc were used to filter out long lived radon daughters and allows only for gas to get into Lucas cell. Surface contamination monitor to detect the surface contamination was used in addition to Eberline survey meter was used to detect the gamma dose rate [5].

RESULTS AND DISCUSSION

Table 1 shows the average concentration of radon, radon daughters and thoron daughters respectively, for each site on logarithmic scale. Also standard deviation and standard errors were calculated and recorded.

TABLE 1: MEAN, STANDARD DEVIATION AND STANDARD ERROR OF RADON, RADON DAUGHTERS AND TORON DAUGHTERS FOR ALL SITES

	Radon			Radon Daughters			Thoron Daughters		
	Mean	SD	SE	Mean	SD	SE	Mean	SD	SE
Sec2	42.40	59.55	34.38	0.000359	0.000199	0.000115	0.2468	0.2113	0.1057
Sec5	22.34	12.12	6.99	0.000247	0.000163	0.000094	0.1206	0.1393	0.0697
Cr3	16.01	7.14	7.26	0.00028	0.000083	0.000048	0.2296	0.4592	0.2296
Cr7	9.09	7.36	3.68	0.000326	0.000090	0.000045	0.1081	0.1481	0.0662
St3	20.54	10.87	10.87	0.000264	0	0	0	0	0
St29	15.15	0	0	0.00033	0.000093	0.000093	0.1494	0.2113	0.1494
SB	25.16	12.42	6.209	0.00091	0.000898	0.000449	0.0458	0.1024	0.0458
UM	1374.51	934.88	467.44	0.175099	0.129704	0.064852	2.3743	1.1562	0.5171

The detected radon ranged from 1.377 to 131.52Bqm^{-3}, the highest radon concentration in the mining area (131.52Bq m^{-3}) was the nearest point to the ore. Measurements in underground tunnels vary from 411.55 to 2539.27Bq m^{-3}due to the variation in ventilation and air flow rate where (UM-1) and (UM-2) were collected from the western site of the underground mine where there was good ventilation. But on the other hand; the eastern site the ventilation system was broke down which cause the highest level of radon concentration (2539.27Bqm^{-3}) [7]. Radon daughters concentration ranged from 0.000099 to 0.00237 WL. Measurements in postpone tunnel vary from 0.0555 to 0.3695WL. The highest concentration was inside the eastern site (UM-3) due to the bad ventilation. Thoron daughters concentration ranged from 0 to 0.918 Bq.m^{-3}. Measurements in postpone tunnel vary from 1.264 to 3.891Bq.m^{-3}. The highest concentration was inside the eastern site (UM-3) due to the bad ventilation. The effective annual dose expected for the workers in the mine was calculated and recorded in table 5, and ranged from 0.033to 0.153mSv/y for open-pit and 4.948 mSv/y for underground mine with average annual dose 0.687 mSv/y.

TABLE 2. EFFECTIVE ANNUAL DOSE IN EACH SITE IN mSv/y

Site	Mean Radon Concentration (Bq/m^3)	Effective annual dose (mSv/y)
Sec2	42.405	0.153
Sec5	22.341	0.080
Cr3	16.010	0.058
Cr7	9.089	0.033
St3	20.543	0.074
St29	15.149	0.055
UM	1374.509	4.948
SB	25.156	0.091
Average	190.6502179	0.687

CONCLUSION

All the sites except underground tunnels have Radon gas, Radon daughters' concentrations and annual doses within permissible limits recommended by UNSCEAR, 2010 [3]. For underground tunnels, most of the radon gas concentrations, radon daughter concentrations, thoron daughters concentration and effective annual dose exceeds the permissible limits. So the open-pit technique is much safer from the occupational radiation protection point of view.

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12-RADIATION PROTECTION OPTIMIZATION IN FIXED INDUSTRIAL RADIOGRAPHY BASED PHITS MONTE CARLO CODE SIMULATION

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Nondestructive test (NDT) has been widely used for defects detection, well joints inspection, and material integrity verification in recent decades. Gamma radiography's applications in industry are known as high risk-related techniques among the nuclear-related industries nowadays. For example, the IAEA safety report about the lesson learned from accidents in industrial radiography reported that around 45 % of accident in the nuclear industry accounted for industrial radiography in both developed and developing countries [1]. This demonstrates the need to optimize safety and protection measures around gamma sources used in fixed industrial radiography. The present study focuses on the optimization of engineering barriers during sitting and construction phases of such facilities. Monte Carlo methods-based PHITS are used to optimize the shielding design of enclosures of a radiographic facility to create a safe working environment for both radiographers and the public. The Particle and Heavy Ion Transport code System [2, 3] was used to determine the appropriate concrete wall and dose estimation to keep radiation As Low As Reasonably Achievable (ALARA). As shown in Fig.1, there are many concerns in old built facilities and few in new ones, but the primary objective is to provide radiological protection regulation or to update and optimize the existing regulation, taking into consideration the shielding design of the facility [4].

FIG. 1. The Real situation encountered in an industrial radiography installation (mainly the proximity of radiation and non-radiation workers and facilities).

As the radiation exposure to any given material depends on the thickness of the shielding, the quantity, and the energy of the radiation, both Co-60 and Ir-192 sources were used in our study as they are the most used high-energetic radionuclides used in industrial radiography. The MC code used for computation, PHITS, is a general-purpose Monte Carlo Particle and Heavy Ion Transport code System developed by a collaboration between Japanese institutions and Europe. The version used, PHITS 3.10 with several changes, allows the simulation of photon and other particles of interest transport over a wide range of energy. The design principle here is based on providing enough shielded enclosure to keep the dose rates out of the facility (in the closest adjacent areas to the facility) lower than 2.5 µSv/h, in adherence to the ALARA principle for the public exposure. If not, a large exclusive area should be set, but this part is considered as administrative controls, which are discussed differently. The facility design was based on IAEA safety report series [7] and the photon flux distribution simulated is presented in Fig. 2 along with the plot of the dose conversion factor use for computation.

TABLE 1. CHARACTERISTICS OF CO-60 AND IR-192 GAMMA EMITTERS USED IN FIXED INDUSTRIAL RADIOGRAPHY CONSIDERED HERE.

Gamma Sources	**Co-60**	**Ir-192**
Half-life	5.26 years	74 days
Source Classification	High	Medium
Primary energy	Dual (1.173 & 1.333) MeV	Multi (0.206 – 1.4) MeV

Maximum activity	100 Ci	50 Ci

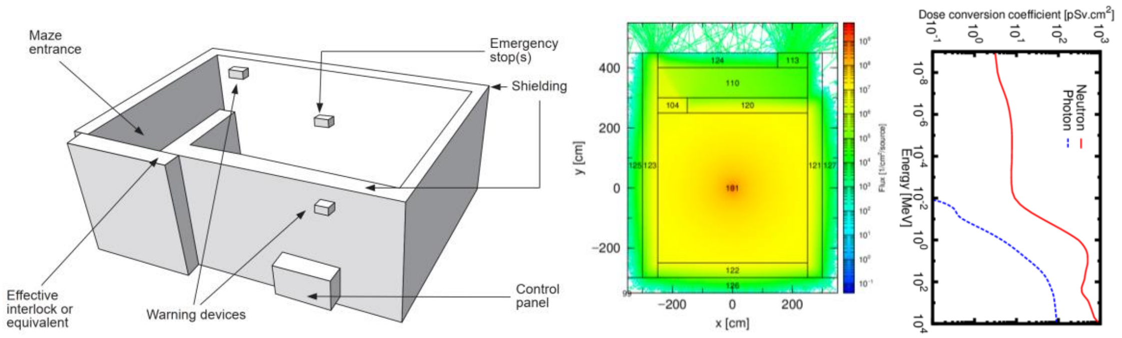

FIG. 2. A simplified view of 3D shielded enclosure for our room (left). XY projection of the gamma flux around the radiographic installation (Center) and plot of the dose conversion coefficients from ICRP116 (flux to effective dose) [5] *(left)*

Appropriate concrete thickness to shield radiation from isotropic sources was computed prior to the radiographic testing room design. The minimum concrete wall thickness for facility using Co-60 source described previously was found to be 120 cm while it was found to be 70 cm for Ir-192 related facility as shown in TABLE 2. To ensure that the radiation dose falls under the recommended limits set either by international organizations as IAEA, or by the regulatory authority of Cameroon, the necessary and appropriate shielding walls and shielding materials shall be installed and regulated by laws.

It is recommended to the government, conjointly with the NRPA of Cameroon, to make a law project in this regard that will be passed in view to facilitate the regulation of the wide spreading practice of industrial radiography using radioactive sources, especially gamma imaging. The present code could be really helpful to the Government of Cameroon in developing database for sitting and construction of radiographic testing rooms depending on the radionuclide type, its energy and intensity, and its activity. Appropriate design-based maze technology was developed by Guembou in his thesis [6]. There is a real need for the implementation of the international rules and the adoption of clear and specific national guides for the application of industrial radiography in Cameroon as well as in different other countries. Different IAEA safety standards series, safety reports series, and technical documents were provided to help governments, institutions, and individuals involved in the use of radioactive sources for industrial radiography to develop an appropriate safety culture. In this regard, enclosures of a radiographic room should be properly designed and used for the sources for which they were designed, considering the maximum activity, the type of radioisotope, their energy, and intensity. Developing countries as Cameroon could use MC simulation for national regulation improvement according to their socioeconomic statute and technology-based considerations.

TABLE 2. SUMMARY OF THE EFFECTIVE DOSE RATE CALCULATION RELATED TO CONCRETE WALL THICKNESS IN THE CLOSEST PUBLIC AREA IN THE CASE OF CO-60 AND IR-192 SOURCES.

Co-60 source (100 Ci max)			**Ir-192 source (50 Ci max)**		
Thickness	Dose(µSv/h)	Relative error (%)	Thickness	Dose(µSv/h)	Relative error (%)
100cm	1.398E+01	2.164E-01	60 cm	9.74E+00	3.49E-01
110cm	4.175E+00	2.759E-01	65 cm	3.95E+00	8.97E-01
120cm	***1.193E+00***	***2.791E+00***	***70 cm***	***1.83E+00***	***3.63E+00***

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14-AWARENESS OF COSMIC RADIATION AND ITS HAZARDS AMONGST AIR TRAVELERS AND AVIATION WORKERS IN MALLAM AMINU INTERNATIONAL AIRPORT, KANO, NIGERIA

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OBJECTIVES

To determine the awareness of cosmic radiation and its hazards amongst air travelers and aviation workers in Mallam Aminu Kano International Airport, Kano. The study also sought to determine the respondents' level of education and their awareness of the health hazards associated with cosmic radiation.

METHOD

A 12-item self-completion questionnaire was used to elicit response from air travelers and airline crews. The questionnaire has three sections: Demographics; Awareness on cosmic radiation; health effects of cosmic radiation. Eight questions were used to elicit response on awareness of cosmic radiation while seven questions elicit response on health effect. The questions on awareness have three options: agreed; disagreed; and I don't know. Disagreed and I don't know implied unawareness while agreed implied awareness of the respondent on cosmic radiation and its health effects.

Taro Yemeni's formula [1] was used in sample size estimation and a total of 233 respondent participated in the study, of which 173 (74.2%) were males and 60 (25.8%) were females. Summary statistics of frequency was generated using Statistical Package for Social Sciences version 20. The study was conducted between June and November, 2019.

RESULT

Result from the study revealed that 142(60.99%) and 124(53.10%) of the respondent didn't know what cosmic radiation is and the health implication of its increased dose on human respectively. Also 24(10.48%) of the respondents disagree with the health implication of cosmic rays on human. Finally, it was discovered that those of the respondents who knew about cosmic radiation and the health implication of its increased dose on human are 91(39.01%) and 85(36.42%) respectively. This also indicated that 102(43.78%) of the respondents got the information from books, while 131(56.22%) got their information from other source.

Table 1 Awareness On Cosmic Radiation

	Agreed (Aware)	*Disagree*	*I Don't Know*	*Unaware*

Cosmic Radiation is found in the atmosphere	***132(56.7%)***	***10(4.3%)***	***91(39.1%)***	***101(43.4%)***
Cosmic radiation increases with increase in altitude	***36(15.5%)***	***67(28.8%)***	***130(55.8%)***	***164(70.4%)***
Cosmic radiation is a form of ionizing radiation	***82(35.2%)***	***21(9.0%)***	***130(55.8%)***	***151(64.8%)***
The more time spent at high altitude the higher the exposure to cosmic radiation	***98(42.1%)***	***10(4.3%)***	***125(53.6%)***	***135(57.9%)***
Only atomic weapons and atomic reactors emits ionizing radiation	***37(15.9%)***	***60(25.8%)***	***136(58.4%)***	***196(84.2%)***

Table 2 Awareness On Hazard of Cosmic Radiation

	Agree	Disagree	I don't know	Total
All radiation can cause cancer	115(49.4%)	35(15.0%)	83(35.6%)	233(100%)
Our bodies absorb cosmic radiation in the environment	101(43.3%)	14(6.0%)	118(50.6%)	233(100%)
The radiation received during a single chest x-ray is almost equivalent to the radiation received during transatlantic flight	12(5.2%)	29(12.4%)	192(82.4%)	233(100%)
Radiation may cause damage to DNA	99(42.5%)	17(7.3%)	117(50.2%)	233(100%)
Radiation may induce birth defect on fetus in pregnant women	125(53.6%)	6(2.6%)	102(43.8%)	233(100%)
When individual is exposed to ionizing radiation some symptoms don't show till after 10 to 15 years of exposure	77(33.0%)	14(6.0%)	142(60.9%)	233(100%)
I am not aware of any safety measure against radiation	65(27.9%)	56(24.0%)	112(48.1%)	233(100%)

CONCLUSION

This study reveals the awareness level of cosmic radiation and its hazard amongst passengers and aviation workers in Mallam Aminu International Airport Kano is poor.

Keyword: *Cosmic radiation, Hazards, Air travelers, Aviation workers, Altitude*

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15-SHIFTING PEOPLE'S MIND AND INTRODUCE RADIATION SAFETY CULTURE

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Radiation is widely used throughout the world in different applications to improve human activities easy for better life. Though the need for radiation technologies are increasing the main challenge is facing its consequence when one is unnecessarily exposed which might affect the health condition of citizens (from minor injuries to death). The applications of radiations have been introduced in Ethiopia since early 1970's. Accordingly almost all Ethiopians do have little know-how/knowledge on how to be protected from radiation hazards if exposed unnecessarily. The paper focused on the achievements of one of the awareness activities that was started and continue for about 10 years. During these 10years time more awareness creations were given specially for workers in the area of medical practices using x-rays and radiotherapy centre and nuclear density gauges. It has also been given even in high schools and universities by including the topics of basic concepts, applications, biological impacts, emergencies and response for radiation.

Though the number of radiation workers in the country is large in contrast to the radiation protection experts, in a planned manner and based on the request of the facilities the authority has given awareness through face to face, electronic and print media on the safety of radiation for particular groups radiographers, female radiographers, working in industries, construction sites, medical and associated facilities

According to [1] IAEA safety requirement GSR part 3 paragraph 1.12 Safety culture includes individual and collective commitment to safety on the part of the leadership, the management and personnel at all levels. Paragraph 2.51 also states that the principal parties shall promote and maintain safety culture by (a) promoting individual and collective commitment to protection and safety at all levels.

Considering the status of knowledge of radiation and protection in the people in general the Ethiopia Radiation Protection Authority responsible for had prepared a team under the radiation sources and activities notification and authorization (NAD) department. The author working in the team based on the graded approach principle had prepared its plan for activities to reach people to get aware through electronic and print media as well as face to face training. The result of the study regarding the effect of trainings has become promising to develop radiation safety culture throughout the country and helping the authority to achieve one of its main objectives '**Introducing Radiation safety culture to protect the public from radiation hazards**'. As it is mentioned above the higher officials were not considered and included in the yearly activity plan.

But such activities were not including at higher levels. Therefore in order to actively implement the safety culture the awareness shall be considered as it is stated Basic legal documents enacted by the national legislative body according to [2] Ethiopian Proclamation No. 1025/2017 on Radiation and Nuclear Protection WHEREAS, it is found essential to strengthen radiation sources and nuclear material protection system which comply with international standards by enacting laws to make persons, who are responsible for safety and security of radiation sources and nuclear materials acutely aware of their legal responsibilities and to take appropriate measure commensurate with the radiation hazard against anyone who causes radiation exposure in excess of the acceptable level;

From the experience the author believes that parliament members shall be given first the awareness prior to the public and special emphasis should be given to female radiation workers.

During evaluating the authority's yearly performance by the science standing committee of the parliament members the author requested all the committee members how much did they know about radiation and ways of protection and found only 2 out of the 17 the science standing committee members having to some extent a little knowledge that was shocking on how to inspect and took the responsibility to aware them. To get them focused the training was arranged out of the capital

Finally after being equipped with the appropriate knowledge on radiation safety culture they all agreed that the same awareness creation to be delivered to whole parliament members so that the radiation safety culture to be in place at all level. Hence without making aware the higher officials and then step by step to the public it becomes difficult to inherit safety culture on radiation protection.

After consistent and continual effort people's minds have been shifted and these days' people are requesting from different sectors in the country to give them the awareness training. Outreaching awareness creation includes not only parliament members but also higher officials, facility owners, medical practitioners, administrators, university students & teachers etc. The author believed that continual improvement on radiation safety culture will be achieved when radiation is given as training and shall be included in the school curriculum.

It can be concluded that in order to protect individuals, the society and the environment in current and future generation against the harmful effect of radiation, safety culture should not be left to only radiation workers and shall be implement at all levels.

A table and free sketch diagram showing before, during and after the awareness creation will be included in the body of the poster.

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[1] Ethiopian Radiation and Nuclear Protection Proclamation No. 1025/2017
[2] IAEA safety requirement GSR part 3

16-CHALLENGES OF RADIATION INFORMATION COMMUNICATION IN PARTNERSHIP WITH INTEREST GROUPS: THE CASE OF ETHIOPIAN RADIATION PROTECTION AUTHORITY (ERPA)

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INTRODUCTION

Establishing clear evidence on challenges of communicating radiation information in partnership with interest groups can significantly help mitigate the problems and enhance the communication and consultation performance of the Authority. Bringing interest groups on board and making them active participants in the collaborative communication process requires identification of their needs, interests, stake and roles, areas of engagement, and related challenges in the collaborative effort which this essentially determines the prospective partnership to be fomed, its sustenance and public awareness to be created.

AIMS AND OBJECTIVES

The study aimed at examining factors that hamper collaborative communication, the degree to which interest groups do involve in collaborative communication, impacts of inappropriate media use, communication gaps caused by radiation technical knowledge difference (asymmetry) between information provider and interest groups as target audiences.

METHODOLOGY

The qualitative approach was employed for this study. The sample population comprises six focus group discussants (FGD) experts from ERPA and interviewees: two from government media, one from the parliament and another from public who are unconstructively affected by setbacks of collaborative communication. The data collection instruments employed were FGD and semi-structured in-depth interview methods. The data analysis formally went through six steps which included generation of raw data, reading through raw data and making notes, identifying themes attitudes and behaviors, amalgamating themes into core concepts and interpretation of data as a result of which data was labeled into interpretative themes.

RESULTS / FINDINGS

The findings established that working in partnership with interest groups can bring a concerted effort and capability for members. The result further revealed critical projected challenges that demand strategic intervention. The study found out that identification of relevant interest groups with their clear roles for the partnership should be a prime task. The study proved absence of applicable binding legal and working systems ineffectually impact the initiative. It was also indicated that failure to establish regular multi-channel media relations aligned with segmented audience tailored messages significantly affects the partnership outcomes.

DISCUSSION

Informants presented diverse views that were eventually grouped into themes. The themes that emerged in the course of data analysis include: potential interest groups in frontline, roles of interest groups, benefits of collaborative work, factors hindering collaborative communication and impacts of inappropriate media use detailed below.

Potential Interest Groups: Informants provided with important interest groups that could fruitfully collaborate with the Authority. Government bodies: The parliament, Executive and Judiciary could play pivotal roles in the partnership. The mining, drink, transport and construction industries were also noted to have stake. Private firms such as operators, service providers, suppliers and professional associations were credited as potential arms.

The roles of interest groups: They participate in plan development, implementation and evaluation. They co-fund programs, mobilize resources, share knowledge skill experience and relevant practices, co-create messages, lead and coordinate activities, share risks, bring innovative ideas, support media relations, contribute to system development, help improve regulatory decisions and complement the Authority's regulatory gap.

Benefits of collaborative work: It was found that collaborative communication could fetch enormous benefits for partners: Helps develop an enriched work plan and improve performance, promotes resource mobilization, with varieties of knowledge skills and experiences can create an all-encompassing capability and adds the potential to mitigate obstacles. It creates harmony from diverse backgrounds, brings innovativeness, facilitates complimentary role, information exchange and builds self-reliance.

Factors hindering collaborative communication: It was pointed out that there were also multi-dimensional challenges that could obstruct collaborative communication efforts. Differences in organizational working culture, failure to see positive outcomes of the partnership, conflict of interest, weak executive capacity of partners, considering partnership activities a side issue, leaders' poor integrity and bureaucratic challenges were the most critical challenges observed. Impacts of corrupt minds, political interference, lack of sense of ownership and accountability were also noted as lingering problems. Resource constraints, low understanding of technical knowledge and miscarriage to identify interest groups in terms of relevance clearly were also serious encounters underlined. Lack of commitment and binding laws that govern partners, absence of branch offices in regional states were main impediments noted by the finding. The private interest groups' focus being profit making was also underlined as a challenge that impact partnership's effort.

Impacts of inappropriate media use: Weak multi-media relations, lack of audience tailored messages and uncontrolled and unethical media use were pronounced as predictable problems of the collaborative effort. These challenges could lead to limited accessibility to information in time and coverage, misunderstanding about the information communicated, poor media budget expenditure, bad image formation and loss of confidentiality.

CONCLUSION

Having an established evidence of pivotal challenges that encounter collaborative communication efforts can bring success for partners in their collective endeavor. Identification of potential interest groups in line with their relevance, interests, roles and areas of engagement is essential to better coordinate activities and ensure effectiveness. Developing working systems and binding laws that govern actors can also fetch them incredible benefits.

RECOMMENDATIONS

- The critical collaborative communication challenges identified in the study need to be carefully and knowledgeably considered before the collaborative initiative is entered and strategically intervened in the implementation process
- Collaborative radiation information communication should be a social responsibility for interest groups at all levels
- Creating clear understanding about the collaborative initiative itself and anticipated outcomes among members remains crucial
- Assigning capable coordinators and leaders for the collaborative work on agreed up on arrangements while respecting differences in culture, background and knowledge of members can help bring concrete gains

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17-STEPPING-UP RADIATION SAFETY CULTURE IN GHANA - WHAT THE RADIATION WORKER AND THE GENERAL PUBLIC SHOULD KNOW

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Due to harmful effects of exposure to ionizing radiation, the Radiation Protection Institute (RPI) of the Ghana Atomic Energy Commission (GAEC) has among other objectives provides radiation protection consultancy; public exposure control; nuclear safety and security and radioactive waste management. In line with these objectives, the Radioactive Waste Management Center, of the RPI conducts on-site radiation safety assessment for sectors of the economy where radiation sources are applied. The Center also retrieves and transports disused sealed radioactive sources (DSRSs) from end users for further management. Through the Radiation Protection Training and Consultancy Centre, the RPI conducts a robust training program for occupationally exposed workers and a sensitization of the general public on radiation safety.

1. INTRODUCTION

In Ghana, radiation sources are applied mainly at medical diagnostic Centres for cancer treatment and x-ray imaging; in industry and civil constructions for moisture and level gauge determination; and in agriculture for pest control and extension of shelf life of foods. Other sectors where radiation sources are used in the country include the gold mines and oil and gas exploration fields. The use of these sources are strictly under institutional control and regulatory monitoring to ensure safety of users/workers (occupationally exposed people), the public and the environment from the harmful effects of ionizing radiation emanating from the sources.

When radiation sources become deficient, breaks down or do no longer serve the purpose for which they were manufactured and imported into the country, they are termed disused sources or radioactive waste. Despite being disused, their activity are still high enough to cause radiation injury when mishandled and/or overly exposed to them [1]. In this regard, the Radioactive Waste Management Centre (RWMC) of the Radiation Protection Institute (RPI) of Ghana has a mandatory function of collecting all radioactive waste generated in the country for further management at its licensed Centralised Radioactive Waste Management Facility (CRWMF). Other functions of the RPI towards radiation safety include safety assessment of radioactive sources in use; training of occupationally exposed people and environmental monitoring. The article therefore presents a summary of the role of RPI of Ghana in building radiation safety culture, provide facts and figures on DSRSs and their management. Participation in the International Conference on Radiation Safety is an opportunity to learn from best practices elsewhere; form or be part of a network of experts in radiation protection and ultimately improve radiation safety program in Ghana.

2. MATTERIALS AND METHOD

All operations involving radiation sources and in line with the International Atomic Energy Agency's (IAEA) safety principles [1] require the use of personal protective equipment (PPEs) such as vinyl gloves, woolen gloves, safety boots, lead apron, overall coat, helmet, goggles etc. Radiation monitoring equipment required include: radiation dose meter also referred to as radiation survey/radiation monitoring meter; personal dosimeter badge (thermal luminescent dosimeter, TLD), HPGe neutron and gamma detector; alpha and beta detectors.

Safety assessment:

As required by the Nuclear Regulatory Authority of Ghana (NRA), on-site safety assessment are conducted bi-annually for radiotherapy and gamma irradiation facilities and annually for industries including the mines. Upon request by radiation source end users, a technical team of experts are constituted to carry out this task.

Disused source retrieval and transport:

Retrieval and transport of disused radioactive source is a process that begins with the end user of the source. The ultimate goal is safety of people and security of the source(s) [2].

i. The end user first applies for a permission from the NRA to decommission the source and transfer same to the RWMC for further management,

ii. RWMC seeks permission from NRA to transport the source from the end user to its facility for further management,

iii. Prior to retrieval of the source, RWMC conducts safety assessment of both source and site to ascertain their actual conditions,

iv. Source is retrieved by RWMC at a predetermined date. This is done by measuring and recording dose rates before and after source is securely placed on the transporting trick as follows:

a) measure and record background dose rate at ~50 m away from the source
b) measure and record dose rate at source surface
c) measure and record dose rate at 1 m radius of the source
d) determine transport index

Training of occupationally exposed people:

As a regulatory requirement, occupationally exposed staff need some level of training on safe handling of radiation sources and general radiation protection training. Upon request by end users and transporters of radiation sources, the RPI through its Radiation Protection Training and Consultancy Centre (RPTCC) delivers a robust radiation protection training designed for low, intermediate and high-level staff of the various organisations.

3. RESULTS AND DISCUSSION

Projection of the RWMC indicates that, an average of 4 disused sealed sources are retrieved and transported to the CRWMF for further management. The current outlook of the source inventory is presented in Figure 1 which illustrates the number of sources; their current activity and contribution as gamma, alpha or neutron emitters to the inventory.

As the number of radioactive sources being used in Ghana increases, it presupposes the risk to radiation exposure will increase if proper care is not taken. It has been gathered through the radiation training and sensitization programs that, significant number of industry players including transporters and end users have very little knowledge of radiation effects, and safety measures during operation. Emergency response during radiological accident and/or incidence at site or on the road during transport is another area that scored very low and therefore requires much attention. In view of these and in line with its strategic plan, the RPI, seeks to intensify its training, sensitization and advocacy on radiation safety and compliance to safety standards. This include a survey that will simultaneously test and educate members of the public on one hand and radiation workers on the other hand; on radiation safety, radioactive sources security and protective measures. Ultimately, radiation safety culture will be built on the basis of robust quality management system.

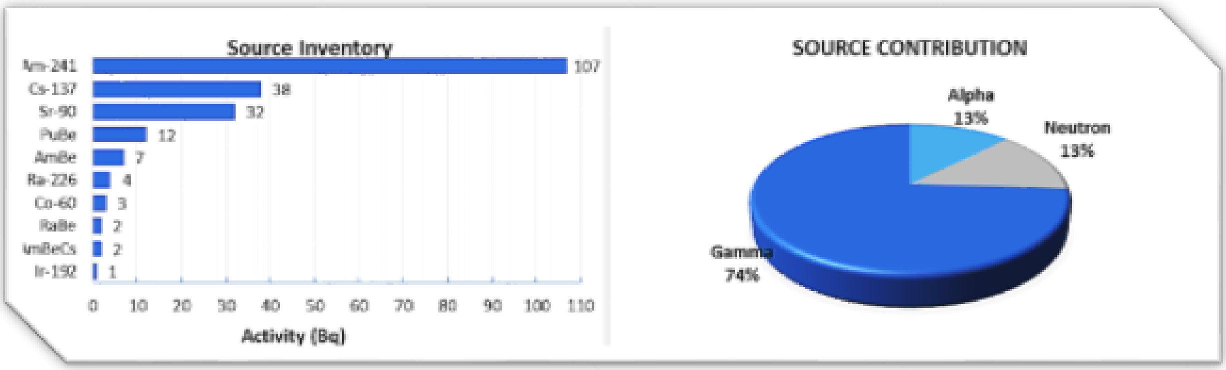

FIG. 1. Ghana's radioactive source inventory

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20-EVALUATION OF THE PROTECTIVE EFFECT OF LEAD GLASSES ON OCCUPATIONAL EYE DOSE IN INTERVENTIONAL CARDIOLOGIC PROCEDURES; DOES ITS BENEFIT THE SAME FOR ALL GROUPS OF RADIATION WORKERS?

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BACKGROUND

Increasing incidence of radiation-induced cataract caused ICRP (International Commission on Radiological Protection) in its 2011 report to reduce the threshold dose for cataract from 2-5 Gy to 0.5 Gy regardless of the duration of the exposure, and decrease annual equivalent dose of lens from 150 mSv to 20 mSv in radiation workers [1-4]. The assessment of lens dose using collar dose is a routine but inaccurate method which can cause significant errors in estimation of the eye dose [5-7]. Considering the increased use of interventional cardiologic procedures and concern about irradiation to the eyes, it is necessary to measure eye dose in radiation workers in catheterization laboratories. So, the study was designed to measure eye dose of the Cath lab staff for each procedures and evaluate the protective effect of lead glasses in various cardiologic procedures using thermoluminescent dosimeters (TLDs).

METHODS

This was a cross-sectional study on radiation workers in angiography ward of Afshar hospital performed between April 2017 and April 2019. Eye dose was measured by high sensitive GR-200 TLDs for left and right eyes in three groups of radiation workers (cardiologists, nurses and radio-technologists) in 100 different procedures, including: cardiac angiography (CA), angioplasty (PCI) and angiography + angioplasty (adhoc). Three TLDs in plastic bags were located in each point and the mean of them was used as the final dose of that point. Before measurements, TLDs were calibrated by Unfors dose meter type 6001 which was calibrated by a secondary standard dosimetry lab. Plastic bags containing TLDs were attached to the right and left arms of the goggles, in close proximity to the eye, and were read after each procedure by TLD reader model 7103 (Raman Safety Development of Kish, Tehran, Iran).

RESULTS

Totally 100 procedures were assessed in the study. Mean annual equivalent dose in left eye was 4.2 mSv for physicians. Mean annual dose of the left eye was 1.5 mSv and 1.2 mSv for nurses and radio-technologists, respectively; which was significantly lower than the physicians' dose (P=0.01 and 0.005, respectively). The highest correlation was observed between KAP and physicians' left eye dose in all three procedures. Among nurses, the significant correlation was seen between KAP and left eye dose in PCI (P=0.006) and CA (P=0.001), but not in adhoc procedures (P=0.02); and among radio-technologists, the significant correlation was observed only in PCI (P=0.01). Figure 1 shows the correlation between KAP and eye dose in different staff groups without lead goggles.

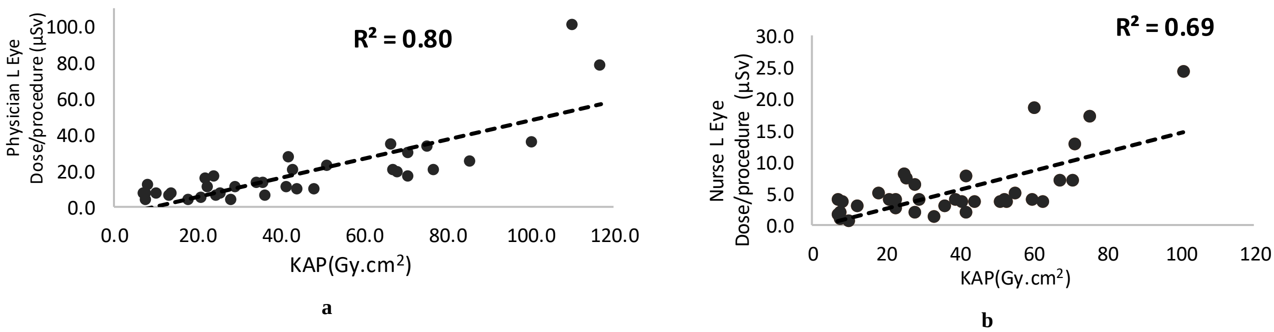

FIG1. Correlation o left eye dose/procedure (μSv) with KAP (Gy.cm2) in procedures without Lead goggles for a) physicians; b) nurses

Some radiation workers didn't routinely wore lead glasses during the procedures; so the eye dose was measured with and without wearing goggles as well (table 1).

TABLE 1. The median (range) equivalent dose per procedure (μSv) and annual dose (mSv) for staff with and w/o lead goggles.

Staff	Goggles	Eye	Procedure*		
			PCI	**Adhoc**	**CA**
Physician	Yes	Right	2 (0-39)	1 (0.9-5)	0.1 (0-3)
		Left	5 (0-36)	2 (1-19)	1 (0-15)
	No	Right	3 (0-11)	3 (1-14)	2 (0-8)
		Left	14 (2-60)	24 (7-100)	6 (2-26)
Nurse	Yes	Right	1 (0-34)	1 (0-6)	1 (0-5)
		Left	2 (0-80)	3 (0-14)	2 (0-6)
	No	Right	3 (1-16)	3 (0-10)	1 (0-3)
		Left	4 (0-24)	4 (2-18)	3 (0-7)
Radio-technologist	Yes	Right	1 (0-7)	0.6 (0-4)	1 (0-3)
		Left	1 (0-9)	0.7 (0-3)	3 (0-7)
	No	Right	2 (0-9)	1 (0-3)	1 (0-4)
		Left	2 (1-8)	1 (0-6)	1 (0-4)

* PCI: angioplasty, Adhoc: angiography + angioplasty, CA: angiography

DISCUSSION

The results of the study showed that one of the cardiologists didn't use goggles, so his dose was significantly higher than others. his annual dose was 2.5 times the dose of other cardiologists. Comparing the eye dose in different job groups shows that it was higher in physicians than others, which is probably due to the different distances of staff from the radiation tube and the patient. Also in some instances physicians bent their head for better observation which has caused their eye dose to be higher than others. The eye dose was significantly lower when the staff wore lead goggles ($p<0.01$), specially for cardiologists. The protective effect of the goggles were lower in radio-technologists who were in the farthest distance from the tube. The results showed a high correlation between eye dose and KAP when the radiation workers didn't wear goggles. The highest correlation was observed in physicians probably due to the closeness to the patient. The difference in the occupational eye dose in different studies [8-13] is probably due to different KAP, IRP, fluoroscopy time and using protective devices such as floor installed shield and lead goggles.

CONCLUSION

Eye dose of cardiologists was significantly higher than nurses and radio-technologists, so as their annual eye dose was more than one third of standard annual dose. Therefore it is recommended that cardiologists should be monitored periodically for their eye dose, but this is not necessary for nurses and radio-technologists. This study also showed that lead goggles can significantly reduce the eye dose, so it is recommended for the cardiologists to use lead goggles, lead screens or floor installed shields in Cath labs.

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21-EVALUATION OF BRAIN CT FINDINGS FOR ASSESSMENT OF UNNECESSARY CT EXAMINATIONS; A MULTI-CENTER STUDY IN IRAN

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BACKGROUND

The use of CT Scans grew quickly, about 600%[1], rising from fewer than 3 million per year in 1980 to more than 81 million now. In overall, CT scans have the single most important role to estimate the worldwide collective effective dose from diagnostic imaging, approximately 4 million person-Sv/year[2]. It is estimated that about 2% of all cancers in the USA is due to CT and fluoroscopy [3]. Moreover, eye lens and thyroid, may receive high doses in brain CT scans [4]. Clinically unjustified CT scans, even though the individual radiation risk will still be very small, but considering a large number of individuals the potential exists to generate a significant long-term public health concern. The aim of the study, therefore, is the assessment of normal and abnormal reports of brain CT examinations in the imaging facilities of Yazd province hospitals. Results can potentially improve the evaluation of clinically unnecessary brain CT scans.

METHODS

The radiologist reports and medical records were retrospectively reviewed from 3950 (1610 women, 2341 men) brain CT examinations requested from primary care clinics from all hospitals in Yazd, Iran. Systematic randomized method used to select patient reports at these hospitals. Images falling in these categories were extracted from the patient electronic files of these hospitals during May 2015 to May 2016.

Since the most unjustified CT scans are from non-pathologic reports, named as normal reports, so samples were rated as either normal or abnormal reports, based on the available reports of the expert radiologists. As a final point, the abnormal/normal ratios of reports for different age groups and gender was the most important question of the study. Abnormal and normal reports were analyzed by age, gender, imaging department type (public and private hospitals). The distributions of normal and abnormal reports were presented by frequency and percentages in hospitals.

RESULTS

The distribution of normal and abnormal reports in private and governmental hospitals in terms of 3 different age groups are shown in Figure 1. Figure also showed that, normal reports for patients under and above 40 years old were about 9.4% more and approximately 11.7% less than mean (68.8%), respectively. In private centers, the difference of normal reports and total mean ones (68.8%) were about -0.7%, +17.3% and -13.6% for <15, 15-40 and >40 years old, respectively. For governmental centers this difference was +12.2%, +9.4% and -10.6% for those 3 age groups.

Table 1, also, summarizes the details of the normality and abnormality of all hospitals based on gender type. From this table it can be stated that, the male normal reports are about 10.7% less than the female ones. This difference in private centers (17.2%) was more than governmental centers (9.4%). But the analysis of total normal reports in all hospitals showed that the females were comprises 11.6% lower than the male ones.

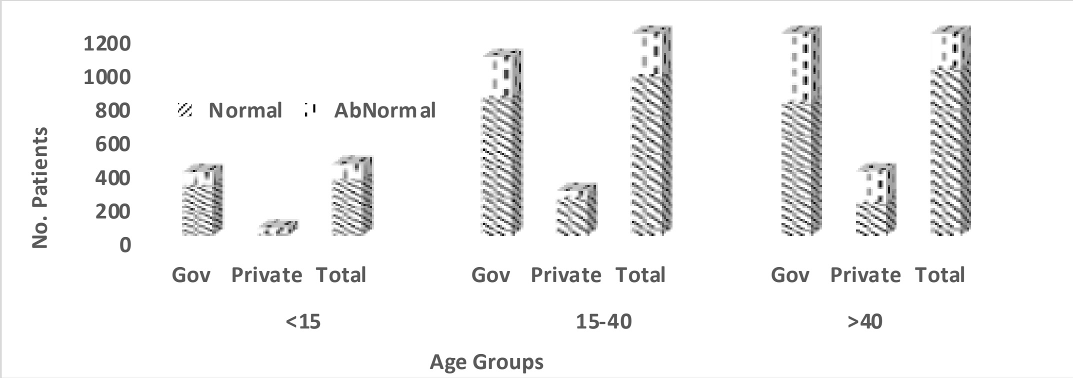

FIG 1: Distribution of Normal and Abnormal Brain CT in Private and Governmental Hospitals in Three Age Groups.

TABLE1- reports of the 6 hospitals with the number and percentage of normal, abnormal and gender type (male/female)

Hospital type	# of patients	Female(%) Normal (%) Abnormal(%)	Male(%) Normal(%) Abnormal(%)	Normal(%) Female(%) Male(%)	Abnormal(%) Female(%) Male(%)
Governmental centers	**3231**	**38.1**	**61.9**	**69.3**	**30.7**
		76.0	66.6	41.3	30.7
		24.0	33.4	58.7	69.3
Private centers	**719**	**51.0**	**49.0**	**66.8**	**33.2**
		75.1	57.9	57.9	38.5
		24.9	42.1	42.1	61.5
Total(all hospitals)	**3950**	***41.1***	***58.9***	***68.8***	***31.2***
		75.7	***65.0***	***44.2***	***32.6***
		24.3	***35.0***	***55.8***	***67.3***

DISCUSSION

Although the children contribute about 12% of all brain CT, but 80% of these pediatric CT studies were normal. Besides, though the female contribute 41% of all brain CT, but 75.7% of their reports were normal and, therefore, it contributes more than 44% of normal reports. Because of high radiation sensitivity of children and females, the high frequency of normal reports that most of them maybe unjustified or unnecessary, can wastefully lead to extra cancer risk of population. About 43% of all patients, in the study, had no symptoms or only the headache with no important neurologic signs. On the other hand, routine CT examination for most of the patients with a chronic headache without the substantial neurological signs or any unusual clinical symptoms is more likely to have a normal reports, close to 80%, [5, 6] [7]. It is important to note that more than one-third for head CT were unjustified [8] and could practically be replaced by other radiation free imaging modalities, especially in pediatric and young patients. So, it is concluded that more than 32% of CT scans in the study may be unjustified. However, reducing CT scan or replacing by other modalities is a very hard task, in that physicians are subject to significant "non-medical" force [9-11]. clinically unnecessary CT scans may result in a significant unnecessary collective dose and consequently increase the population risk in the future and increased the costs and wasted the time of patient and hospital.

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22-IS THERE A CONCERN ABOUT EXTREMITIES RADIATION DOSE FOR INTERVENTIONAL CARDIOLOGISTS IN CATHETERIZATION LAB?

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BACKGROUND

Angiography is increasingly used to help diagnose cardiovascular diseases, so as its use in the US has been increased from 2.45 million procedures in 1993 to 4.9 million procedures in 2010. In Iran about 400,000 cardiovascular angiographies have been performed annually [1]. These procedures pose cardiologists to a high level of X-ray comparing to other radiographic methods [2]. So measurement of absorbed dose in interventional cardiologists is important to plan for reducing the carcinogenic and probabilistic effects of radiation [3]. Considering that most studies were based on experimental measurements on human phantoms, the results may have under-estimated or over-estimated the real exposure. Some studies have used TLDs in humans, but their data was not obtained in each procedure individually, so their results are probably not exact. Since, some physicians do not routinely use radiation protection devices and personal dosemeters during interventional procedures, especially for extremities, this may significantly affects their absorbed dose. Therefore, the study was designed to measure the absorbed dose in some important organs and extremities in cardiologists during different procedures in catheterizations lab.

METHODS

This was a cross-sectional study conducted in a specialized cardiologic hospital, equipped with one angiography device, from Aug 2018-Aug 2019. The entrance skin dose for extremity absorbed dose of the physicians were measured by TLD GR-200 chips. The participants were three cardiologists with different work histories. Measurements were performed for different CA, PCI and Adhoc (CA+PCI) procedures for each physician. In each measurement, 21 TLDs were used for each cardiologist during the procedure. The number of frames/second in all procedures was constant (15 frames/s). The exposure parameters such as Dose Area Product (DAP, in $mGy.m^2$), and the IRP dose (dose in reference point, mGy) were also recorded separately in each procedure. TLD badges were attached to the pre-determined points: thyroid, right and left chest, and wrists, and left leg. Also the correlation of dose to exposure parameters performed using spearman correlation test.

RESULTS

From all procedures, 105 measurements were performed for PCI, PCI+CA, CA. On average the maximum organ absorbed dose was for PCI+CA procedures. Table 1 shows the characteristics of radiation exposure and entrance skin dose among cardiologists in 6 different points in these procedures. There was a relatively high correlation between left leg dose and left wrist dose with total DAP for all three physicians. The correlation, also, was relatively high between left leg dose and IRP but relatively low for left wrist dose and IRP. Mean ±SD for entrance dose /procedure /DAP for PCI procedures were 1.8±1.3, 5.5±4.2, and 24.4±14.4, for right wrist, left wrist and left leg, respectively. These measures for CA were 1.7±0.9, 9.1±5.5, and 22.5±13, respectively for right wrist, left wrist and left leg; and 1.7±1.1, 6.0±5.2, and 24.8±11.0, for CA + PCI. Figure 1 (a, b, and c) shows dose per procedure/DAP for 6 organs in PCI+CA procedures.

TABLE 1. Summary of entrance skin dose of three cardiologists in 6 different points and radiation exposure parameters due to angioplasty +angiography procedures. All data presented as (median, SD and range)

	Median ± SD (Range)		
Variable	**Physician 1**	**Physician 2**	**Physician 3**
DAP ($mGy.m^2$)	**4.3**±1.8 (1.8-8.7)	**6.4**±6.4 (2.6-24.7)	**3.9**±1.2 (2.5-24.7)
IRP dose (mGy)	**721**±318.6 (286-1498)	**999**±1258 (370-4685)	**731**±240 (375-936)
Thyroid dose (µGy)	**0.44**(0-3.01)	**1.71**(0.67-5.92)	**1.77**(0.59-2.61)
Rt chest dose (µGy)	**0.04**(0-1.17)	**0.58**(0-1.88)	**0.32**(0-1.44)

Lt chest dose (μGy)	**0.0**(0-15.44)	**0.90**(0.23-4.41)	**0.92**(0-3.33)
Rt wrist dose (μGy)	**4.88**(1.79-16.52)	**17.94**(3.14-35.10)	**7.61**(4.26-12.04)
Lt wrist dose (μGy)	**16.59**(4.39-90.41)	**33.05**(5.91-170.36)	**21.25**(16.14-68.08)
Lt leg dose (μGy)	**88.38**(36.87-329.77)	**174.57**(29.96-514.47)	**99.92**(65.14-105.55)

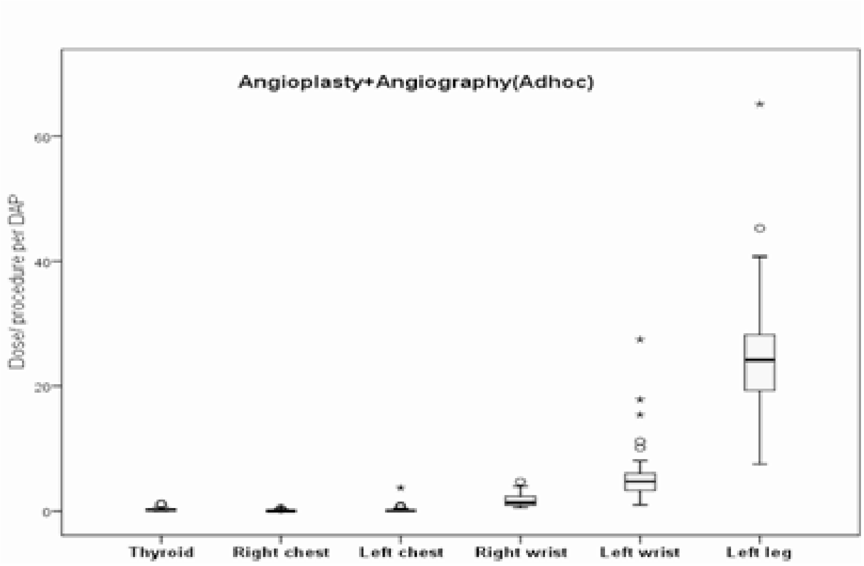

FIG1. Distribution of Dose per procedure/DAP for 6 points in CA+ PCI procedures.

DISCUSSION

The results showed that there was a large difference in variables in different procedures and even in the same procedure between the physicians, so as the maximum of DAP was 12 times the minimum in PCI procedures. After normalizing the entrance skin dose to DAP for physicians with the highest absorbed dose, the diversities in different situations were reduced and relatively similar results for each organ in different procedures have been achieved, in consistent with [4, 5]. It means that, regardless of the type of the procedure, the characteristics of device output and especially DAP have a direct role in the absorbed dose of the organs, especially those outside the shield.

For all organs, the dose in left side was higher than right one ($p<0.05$), specifically for wrists due to proximity to the tube and patient, which was consistent with the results of [5-7]. In the interventional procedures, the complexity of the procedure, technique which the physician uses and physician's workload have a considerable effect in wrist dose [8]. The highest dose for all procedures was observed in left leg, left wrist and right wrist ($p<0.05$), respectively, which is probably due to the position of the tube (below the patients' bed) and physician's position. The physician1, at most of the times, was further away from the patient's bed, so, his wrist dose in physician 1 was significantly lower than others ($p<0.05$). The annual absorbed dose of the physicians extremities were lower than ICRP recommended limits [9]. Lower than standard dose may protect against deterministic events, but there is no guarantee to prevent stochastic events such as cancerogenicity, which do not have any threshold dose. The probability of these events is possible in low dose long-term radiation, therefore it is recommended to use protection methods to reduce the dose of extremities.

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23-ESTIMATED DOSE OF STAFF IN NUCLEAR MEDICINE BY POST STUDY AND WASTE MANAGEMENT

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BACKGROUND

The post study consists of evaluating, under normal working conditions, the doses that can be delivered to staff, following external and internal exposures to sources of ionizing radiation. It allows identifying and estimating the risk in order to implement the appropriate prevention actions and to provide elements for the management of possible incidents.

The nuclear medicine department is also concerned with waste management (solid and liquid ...) and to ensure that only the background noise remains, it is necessary to manage the waste well

METHOD

You have to list the various positions of professions potentially exposed to ionizing radiation in connection with nuclear medicine activity. In our service, we only use technetium 99 and iodine 131. For the post study, we always place ourselves in the most penalizing conditions to estimate the effective and equivalent dose.

You have place trash cans with bags for solids in the concerned premises while the liquids (iodine 131) go directly into the tanks and we always place ourselves under the penalizing conditions.

RESULTS

The results are summarized in table 1

Table 1 Exposure of workers in our service

Radio-nuclide	Activity	Personnel	Exposure (mSv/year)	
			Whole body	**Extremities**
Ba 133	Quality Control	Physicist	0.015	0.25
Cs 137	Quality Control	Physicist	0.006	0.54
I 131	Therapy, administration	Physician	5	/
I 131	Therapy, post-treatment scan	Technologist	1	/
I 131	Therapy, consultation	physician	3	/
Tc 99m	Diagnosis, preparation	Technologist	0.05	44
Tc 99m	Diagnosis, scans	Technologist	1	/

For technetium 99m waste management is not a problem because the ten period rule is almost valid except for the first elution if they are not used while for liquids (iodine 131) the RPO apply the 30 period rule from 3.7 GBq

CONCLUSION

The nuclear medicine department is particularly concerned with post studies. It poses a relatively
high risk of exposure to ionizing radiation for workers since they handle radioactive sources daily.
For occupational exposure of workers, the dose limits are strictly regulated and checked in our service.

With the methods we use we conclude that the waste is well managed and it becomes ordinary

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25-ORGANIZATIONAL LEARNING PROMOTIONS TO ENHANCE SAFETY CULTURE USING GROUNDED THEORY TO DISCOVER ORGANIZATIONAL FACTORS IN DEVELOPING COUNTRIES' CONTEXT

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The author was first searching for a relationship between knowledge management (KM) system and safety culture. He came to know that the established KM processes have a significant direct effect on safety culture in long periods. Kiyantaj and colleagues interpreted this effect through a Practice-based perspective to knowledge, knowing and learning in organizations [1]. Subsequently, the investigation revealed that despite some criticism about conflicts of power-politics relations and the maximum involvement of employees, the concept of the learning organization that was dominant in the 1990s still extant in the format of KM. Nevertheless, almost all researchers recognize the concept of organizational learning and its impacts on organizational culture.

Considering this interplay, the main research question arises: What are the factors that improve organizational learning, which subsequently enhances safety culture? In other words, what are the factors that make organizations become fast learners?

The study used a qualitative methodology, precisely the grounded theory approach. The research question is relevant to those who work in Iran's nuclear and radiological organizations dealing with major radioactive hazards.

Theoretical and purposive sampling was used in the study, and the data gathering was continued until it reached a diminishing return point; that is to say, the research was saturated with various data on a specific subject. The research is based on a collection of over 20 semi-structured and in-depth interviews of Iran's nuclear, radiological and regulatory employees over an extended period. In addition to this, some personnel from Iran's Nuclear Regulatory Authority (INRA) were interviewed. The INRA is not a high-reliability organization (HRO), and it is not directly subject to safety culture. However, because of their daily effects on safety issues on such organizations, they were asked to take part in the study [2].

The interviews' inputs were categorized based on the frequency of their occurrence by ATLAS.TI software. The data was analyzed and coded using a three-step process, i.e. open coding, axial coding, and selective coding [3]. Based on the findings from the empirical study and grounded theory, 34 concepts revealed through coding the quotations of interviewers, which affect individuals' learning process in organizations. These concepts were categorized into six co-axial elements according to their causes, contexts and consequences. All the elements were integrated to generate a domestic model of organizational learning to promote safety culture in Iran's nuclear and radiological industries. The concepts and co-axial elements identified are as follows:

— Appropriate policy-making and commitment to it
  - Commitment to being goal-oriented rather than task-oriented
  - Commitment to regulations and transparent bureaucracy
  - Commitment to continuous improvement and sustainable development
  - Commitment to behaviour according to justice

— Strategic orientation and process-driven approach
  - Being proactive and long-term or mid-term planning
  - Seeing opportunities, threats and future needs associated with the organization's weakness and strengths
  - Avoiding the allocation of resources according to the power of informal and formal groups
  - Effective training courses
  - Systemic thinking and having a process-driven approach
  - Evaluating organizational productivity and performance of individuals according to the main goals of the organization

— Using effective and modern management information systems

— A democratic and transparent bureaucracy in the organization
  - Establishing an integrated management system
  - Effective information flow and knowledge sharing
  - Bureaucratic power in the organization
  - Solving the dualism of transparency and confidentiality
  - Restricting political activity of formal and informal groups to absorb resources and opportunities
  - Effective involvement of staff and stakeholders

— Leadership
  - Having technical and managerial competence
  - Coaching
  - Having an inspirational vision and motivation
  - Having the ability to Obtain maximum employees engagement
  - Helping people to discover their actual
  - Aligning the personal interests of employees with the organization's development goals

— Organizational Culture (negative correlation)
  - Belief in that managers and officials are perfect and rightful anyway
  - Blame and gossip environment in the organization and not realizing the importance of learning from mistakes and failures
  - The sense of being a simple worker rather than being the system owner
  - Belief in that any conflict in an organization is a disaster
  - Recognition of age and academic degree rather than experience and knowledge for real competency
  - Belief in the inefficiency of diversity
  - Belief in the uselessness and inefficiency of employees and stakeholders involved in decision making
  - Belief in the formality and ineffectiveness of government supervision
  - Job promotion of knowledge hoarders
  - Distrusting the importance of safety for the excellence of the organization
  - The despair of growth and change, and the belief that "The organization will never be excellent."
  - Distrusting the whole organization's goals

All the four criteria (credibility, transferability, dependability, and confirmability) introduced by Lincoln and Guba (1985) for the trustworthiness of qualitative research have been considered and met through appropriate actions, in the beginning, during and at the end of the research [4].

As mentioned before, organizational learning can have a significant impact on organizational culture. Therefore, managers, especially in developing countries, are advised to use these findings, not only for the sake of upgrading their organization's abilities in learning but also for improving their organization's safety culture. Furthermore, the findings of this study emphasize greater insight into practical applications and considerations necessary for enhancing organizational learning and its interactions with organizational culture, particularly the safety culture aspect of organizational culture.

Also, as a suggestion for future research, since the outcomes of the research are based on the grounded theory approach, which is an inductive method, researchers can use these qualitative research findings to draw out independent factors and discover their relationships with quantitative research methods.

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26-NATURAL AND ARTIFICIAL RADIONUCLIDES PRESENT IN CUBAN FOODSTUFFS.

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The radioactivity to which we are exposed in our environment is a combination of natural radioactivity (coming from the cosmos, from the earth's crust or from our body) and the artificial radioactivity that results from human activities. Both radiations, natural and artificial, behave in the same way. The natural radioactive materials in the earth's crust are absorbed by plants and animals and dissolve in water. Therefore the food and beverages we eat contain varying amounts of radioactive isotopes. Among the natural radionuclides present in food, K-40, Ra-226, Ra-228 and the descendants of the latter two predominate. On the other hand, food can be contaminated with artificial radioactive materials as a result of nuclear weapons tests, nuclear or radiological accidents, among others.

For this study, those foods of national production with the highest consumption in the country were selected, and representative of the following food groups: roots and tubers, grains, cereals, vegetables, fruits, meat and milk. The samples were collected directly in the markets, coming from different localities of the country.

The treatment of the samples was carried out using a laboratory procedure [1]. Specifically, the treatment performed on the matrices consisted of separating the clean edible part (eliminating adhering soil, shell, bones, etc.). The selected parts were washed in running water and allowed to dry on filter paper or absorbent towels. The edible portion of the sample was weighed and cut into fragments whose size was given by the measurement geometry (cylindrical bottle of 250mL volume).

Natural radionuclides (K-40, Pb-210, Ra-226 and Th-232) and Cs-137 as the artificial component of environmental radioactivity were selected to be analyzed. The samples were analyzed by high resolution gamma spectrometry to determine those environmental radionuclides of interest [2]. A spectrometric gamma system from BSI (Baltic Scientific Instruments) with a 40% efficiency hyperpure Germanium detector was used. For the detector calibration, the Mixed Method was used, where the efficiency of the photopic is determined from a Monte Carlo simulation with the help of an energy-dependent transfer factor that is obtained from the measurement of an experimental curve. The simulation was carried out with the MonteCarlo simulation software DETEFF, validated for this purpose and the quantitative analysis was carried out with the help of the software SpectraLineGP, version 1.6.8315.

The most significant radionuclide among those analyzed was K-40, which corresponds to practically 100% of the concentration of activity detected in the food under study. For the rest of the radionuclides of natural origin, belonging to the radioactive series of uranium and thorium, the activity concentrations detected are close to the detection limits of the spectrometric gamma system used. These values are in correspondence with other studies carried out in the country [3, 4] and those reported by international literature for areas of normal environmental radioactive background [5]. The presence of Cs-137 is due to the influence of global radioactive receptions, not exceeding the expected values, if we take into account the results of the environmental monitoring of the national territory [6].

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27-IDENTIFICATION OF STAKEHOLDERS' INVOLVEMENT TO OPTIMIZE PROTECTION STRATEGIES FOR EMERGENCY MANAGEMENT

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INTRODUCTION

Identification of the right stakeholders and their potential role and position for each phase of emergency is crucial to effectively and efficiently use scares resources, achieve and take a sustainable urgent/timely response. In order to engage the right stakeholders, it is important to understand the factors that mold and constrain the situations.

DISCUSSION

The factors that mold and constrain the situations are scale and magnitude of the emergency, demography and geography of the emergency site, type of emergency and legal framework of a country. Understanding these enables us to clearly identify the synergies and conflicts of interest among stakeholders.

Scale and magnitude of the emergency: The scale and magnitude of an emergency/ies can be treated with respect to temporal, spatial coverage and its/their impact/s (severity of the event).

Rate of Speed	Temporal …….. duration, frequency, rate of transmission/expansion
Impact	Spatial ………intensity, radius of the emergency site, depth, height, and boundary
Per unit area	Impact……… minor/major, deterministic, stochastic, somatic, and genetic

Demography and geography of the emergency site: Geographic features of nature are important parameters to manage and identify stakeholders' participation in emergency preparedness and response. The variations in geographic features such as landscape, urban, rural, island, accessibility of the features to transport, etc. and human interaction with the physical environment determine the emergency preparedness and response plan. These geographical conditions should be considered while formulating stakeholders' involvement.

Geography
- Human geography ……. Social, Cultural, Economic and Political Geography
- Physical geography ……. Atmosphere, Hydrosphere, Biosphere

Spatial distribution of people (settlement of a population), norms, and cultural values of a society and mental and physical readiness of people to emergency and educational level, previous emergency experiences all affect emergency preparedness and response. So it is important to consider the demographic factors to identify the right stakeholders to be engaged and assign their roles and responsibilities.

Type of Emergency: Knowing the type of emergency make things easy to identify the stakeholders and let them discharge their duties completely and successfully. Understanding the causes and nature of the emergency enable us to configure and predict the type of emergency and its actual and poetical consequences.

- Nature of the emergency ….. detectability, radioactivity, combustibility, toxicity, fissionable, explosive, dispersible, diffuse, poisonous, transmissibility, localized, trans-boundary, contaminant, dynamics, static, fatal, predictability, adaptability,
- Cause of the emergency ……. human, natural or technological

Legal Framework: The legal framework of a country should focus on emergency management by charging and empowering of competent regulators to create enabling conditions to promote and engage stakeholders on the provisions of emergency preparedness and response. The legal framework should be strong, effective and integrated which is compatible with the international protocols and conventions that the country signed. It should also consider international cooperation i.e. bilateral and multilateral agreements. Since some emergencies are trans-boundary, the legal framework should consider the involvement of international organizations and governments to harmonize and cooperate in the response activities. The emergency preparedness and response policy should clearly define the roles and responsibilities of stakeholders involving in emergency management.

TABLE 1 MAJOR STAKEHOLDERS AND THEIR ROLES

No.	Type of emergency	Phase/s of emergency	Stakeholders involved	Roles
1	All types of emergencies	All phases	National emergency preparedness and response office	Organize, coordinate ad prepare action plan
2	All types of emergencies	All phases	Local Regional & National Authorities	Provision of instruction for actions to be taken
3	Major emergencies such as Radiological chemical, weather , biological	All phases	Statistical agency	Giving information about the size of the population and their distribution
4	All types of emergencies	1st and 2nd phases	First responders (Comprises range of stakeholders)	
5	All types of emergencies	1st and 2nd phases	Media Institutions	Creating awareness, notifying and disseminate information
6	Major emergencies such as Radiological chemical, weather , biological	1st and 3rd phases	Research Institutes	Conduct research ad provide recommendations for policy and decision makers
7	All types of emergencies	1st phase	Technical Experts	Train First Responders
8	Major emergencies such as radiological chemical, weather , biological	Depends on e request of a country	International Organizations and NGOs	Assist and donate
9	All types of emergencies	All phases	Public	Participate in emergency management and decision

Challenges

- Identifying stakeholders within the framework of the factors mentioned above is multidisciplinary so that different experts should participate. This creates a challenge in resources (human, financial and time) management.
- Lack of optimization mechanism to balance the level of stakeholders involvement in decision making
- The dynamic nature of stakeholder involvement
- Complex nature of emergency situations

Opportunities

- National and international best practices
- Lessons learned from previous emergency management systems

CONCLUSION

Planning and implementing effective and sustainable emergency preparedness and response management system requires an understanding and preferences of a wide range of stakeholders according to their poetical and skate in the service delivery, cost, and corresponding environmental and social impacts.

RECOMMENDATIONS

- Prepare a stakeholder map in order of priority based on the factors that determine stakeholders' identification.

Since stakeholders engagement is dynamic, revise

28-ENHANCING RADIATION SAFETY THROUGH OCCUPATIONAL RADIATION PROTECTION IN NIGERIA

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INTRODUCTION

Occupational exposure is refers to a cumulative exposure to workers in ionizing radiation facilities during their daily work routines, according to the IAEA safety standards. The United Nations Scientific Committee on the Effects of Atomic Radiation, has outlined techniques of occupational dose assessment which entails categorizing type of practices and radiation environment employed [1,2]. Individuals working in both radiological and nuclear facilities are often exposed to sources of ionizing radiation resulting in some level of occupational dose, in which depending on amount of doses incurred has likelihood for radiological hazards. However, having an effective radiation protection in place enhances dose reduction to the barest minimum.

Radiation workers in facilities such as fuel cycle facilities, diagnostic radiology, nuclear medicine, industrial radiography, and nuclear power plants etc, are always exposed to radiation both internal and external dose in varying amounts of radiation, depending on their jobs and sources with involved. The human body interacts with radiation particles either internally or externally and thus resulting in biological damaging effect. It is pertinent to note that these radiation particles ionize living cells of the body either directly or indirectly thereby breaking chemical bonds of DNA biological molecule. The resultant effect of this may likely cause impairment, permanent alteration and death of the cell [3,4].

The evaluation of annual average effective exposure doses data reported in this paper relates to dose received by the occupationally exposed workers in various practices in Nigeria. In this regard, the Nigerian Nuclear Regulatory Authority (NNRA) has in place strong regulatory framework to ensure occupational exposure for all practices involving the use of ionizing radiation is safe; Exposure is kept As Low As Reasonably Achievable (ALARA Principle) and dose limits specified for individuals does not exceed. To ensure the dose limits are not exceeded and in line with IAEA safety standard, the NNRA has recommended dose limits of occupational exposures to radiation workers was stipulated as 20 mSv per year (averaged over a period of 5 consecutive years) [5,6,7]. The purpose of occupational radiation dose assessment program has been to provide facts on the capability of protection measures, considered as key input for operative assessments related to principle of optimisation and also to validate compliance with relevant recommended international standards [8].

METHOD

Employers of radiation workers engage the services of an accredited Dosimetry Service Provider (DSP) who monitors and report doses of radiation workers to the NNRA in quarterly and summary report in annual basis. Different dosimeter products like calibrated Thermos-Luminescent Dosimeters (TLDs), Optimum Stimulated Dosimeters (OSLs), Ion Exchange Dosimeters (Instadose), for occupational dose monitoring were used to monitor and report doses of radiation workers to the NNRA. A yearly Proficiency Test is carried out by Secondary

Standard Dosimetry Laboratory (SSDL) to ascertain the reproducibility, linearity etc of dosimeter used credibility of dose reports by the DSPs [9].

RESULT

The annual average effective doses ($Hp_{[10]}$ and $Hp_{[0.07]}$) collation of occupational radiation workers from Industrial Uses, Medical Uses, Education and Research and other Activities and practices submitted from 2012 to 2016 were collated and analyzed. Range of average doses were found to be (0.31–2.48)mSv/yr for 2012, (0.30 – 1.44)mSv/yr for 2013, (0.14 – 2.34)mSv/yr for 2014, (0.23 – 2.42)mSv/yr for 2015 and (0.04 – 2.07)mSv/yr for 2016 of facilities within an occupational groups, which are presented in Table 1. Also, a collective doses of 0.56manSv for 2012, 0.50manSv for 2013, 0.71manSv for 2014, 1.52manSv and 0.72manSv 2016 of all occupational groups. This shows that the risk due to radiation exposures low and also below the regulatory limits [5,6,7,10].

TABLE 1: OCCUPATIONAL DOSE ASSESSMENT FOR NIGERIAN RADIATION WORKERS

Occupational Groups	Effective Dose 2012 (mSv/yr)	Effective Dose 2013 (mSv/yr)	Effective Dose 2014 (mSv/yr)	Effective Dose 2015 (mSv/yr)	Effective Dose 2016 (mSv/yr)
IR	0.57	0.54	0.50	1.10	0.79
DR	1.1	0.47	1.64	1.41	0.58
WL	0.34	0.30	0.14	2.42	0.85
RR	2.48	1.44	2.34	1.11	1.86
GIF	0.95	0.86	-	1.45	2.07
RT	1.22	0.35	0.55	0.92	0.41
NM	-	-	-	1.84	0.04
Ac	0.31	0.40	0.36	0.23	0.15
NG	1.26	0.71	0.25	0.64	0.87
Average	**0.56**	**0.50**	**0.71**	**1.52**	**0.72**

Evaluation of the dose records submitted from 2012 – 2016 in cumulative and collective for each occupational groups were below regulatory limits.

CONCLUSION

Exposure from sources of radiation in radiological and nuclear facilities delivers occupational doses in Nigeria. Different products (dosimeters) for occupational dose monitoring were used to monitors and report doses to the NNRA. The regulations requires that employers of radiation workers engage the services of an accredited DSP who monitors and report doses of radiation workers and submits report in to the NNRA in quarterly and annual basis. Also, an Occupational Radiation Protection Appraisal Service (ORPAS) mission conducted on Nigeria's legislative, regulatory infrastructure as regards to Occupational Radiation Protection and practical implementation by technical services and facilities against the international safety standards shows the presence of an effective regulatory oversight, strong awareness level on radiation protection and compliance with the national and international requirements. It is however recommended that the present standards of Occupational Radiation Protection in Nigeria be maintained and constantly improved upon.

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30-DESIGN, INSTALLATION AND OPERATION OF AUTOMATED DRAINAGE SYSTEM FOR PET-CT FACILITY: AN OPTIMIZED APPROACH TO PROTECT PERSONNEL AND ENVIRONMENT IN NUCLEAR MEDICINE FACILITIES

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INTRODUCTION

Protecting people and environment is the main objective of the Fundamental Safety Principles [1] that should be implemented and verified by all relevant parties in all facilities and activities. The prime responsibility for ensuring safety implementation and verification rest with the licensee through applying safety requirements pre-defined by the regulatory body. The licensee shall use the most recent technological advances for this objective whenever possible and applicable. Nuclear medicine units are considered as one of the most important practices associated with possible radiation risks due to considerable amount of solid and liquid radioactive waste. Thus, rigorous radiation protection program is a must to protect personnel and environment from such waste and to assure safe discharge of radioactive sewage. This contribution is considered to be an experimental verification of the fifth principle of the ten safety principles namely "optimization of protection" in the field of nuclear medicine. Although the safety report series no. 58 issued by IAEA [2] encompasses all relevant issues of safety and radiation protection in PET-CT facilities, but unfortunately such important safety aspect (sewage system in PET-CT) is missed. This paper addresses design, commissioning and operation of programmable drainage system for collecting and discharging sewage contaminated with radioactive isotope 18F in a PET-CT facility unit under construction in Egypt. The system is working automatically without human intervention..

SYSTEM DESCRIPTION

The system consists of 3 tanks as shown in Fig. 1. The sewage waste is collected from the hot lab and patients' rooms in two tanks of them namely Tank1 and Tank2. These two tanks are used mutually such that each is filled one after another and then discharged in the same order to the 3rd tank (Tank3) provided that the sewage will not be discharged from any of such two tanks before 20 h form its complete filling. The 3rd tank plays an important role in verifying the defence in depth principle. If, for some reason, the swage should be discharged from one of tanks (Tank1 or Tank2), then Tank3 is used to store the sewage to complete the required 20 h. The system is controlled by means of Programmable Logic Controller (PLC) connected with six levelling sensors (three in each tank) and 4 solenoid valves (two in each tank). Filling levels and storage time for each tank is transferred to monitoring unit in hot lab; the system could be fully controlled and reprogrammed (if necessary) without need to get inside the tanks' room. The tanks' room is equipped with surveillance camera to monitor the room all the time.

SYSTEM MECHANISM

Tank 1 and Tank 2 are mutually filled and discharged along the following steps:

1. The system starts at S1 opened while S2, S3 and S4 are closed
2. When Tank1 is full a signal is sent from L1 to PLC to close both S1 and open S3 simultaneously.
3. When Tank2 is full a signal sent from L4 to PLC to open S2 to empty Tank1.
4. When Tank1 is empty a signal from L3 is sent to PLC to close S2, and after 30 seconds S1 is opened and S3 is closed simultaneously.
5. When Tank1 is full a signal from L1 is sent to PLC to open S4 to empty T2
6. When Tank2 is empty a signal form L6 is sent to PLC to close S4, and after 30seconds S3 is opened and S1 is closed simultaneously.
7. When Tank2 is full go to the step no. 3.

The previous steps are repeated provided that Tank1 (Tank2) should spend at least 20 hours (slightly more than 10 times the half life time of F18 ≈ 110 min) from closing its input valve S1 (S3) before opening its drain valve S2 (S4). Valves S2 and S4 are programmed such that never opened before such 20 hours.

In case of unexpected failure that may led to Tank1 (Tank2) is filled and should be discharged before spending 20 hours, a signal from L2 (L5) is sent to PLC to open S2 (S4) and an alarm is sent to the monitoring unit in hot lab notifying that the tank is forced to be discharged into Tank3. In such case, if L2 and L5 are absent, both S1 and S3 are closed which will force the sewage to return to patients rooms and hot lab.

According to expected workload (16 patients/day), tanks' room is characterized with radiation levels and thus is classified as "controlled area". The location of room has been chosen to be located adjacent to very low occupancy area. Two walls of the room are adjacent to private parking lot of the building dedicated for employees while the other two walls are adjacent to air conditioning cooling units. Taking in consideration the previous factors, the shield of the tanks' room was calculated according to [3] and it is found that 20 cm of ordinary concrete will be very conservative.

The system is examined using clean water and found to be working efficiently according to the design's parameters. This concept of the proposed system could be applied in other nuclear medicine facilities after scaling it according the estimated radioactive waste and workload

CONCLUSION AND RECOMMENDATION

The proposed automated drainage system for sewage in a PET-CT facility showed efficient performance and get the radiation risk to the personnel and environment to its possible lower limit. The system could be adopted as a standard of practice for sewage system in PET-CT facilities. The same concept could be scaled and applied to other nuclear medicine facilities working with other radioisotopes. Regulatory bodies are encouraged to adopt such approach as a safety requirement to help the licensee in complying with the safety standards

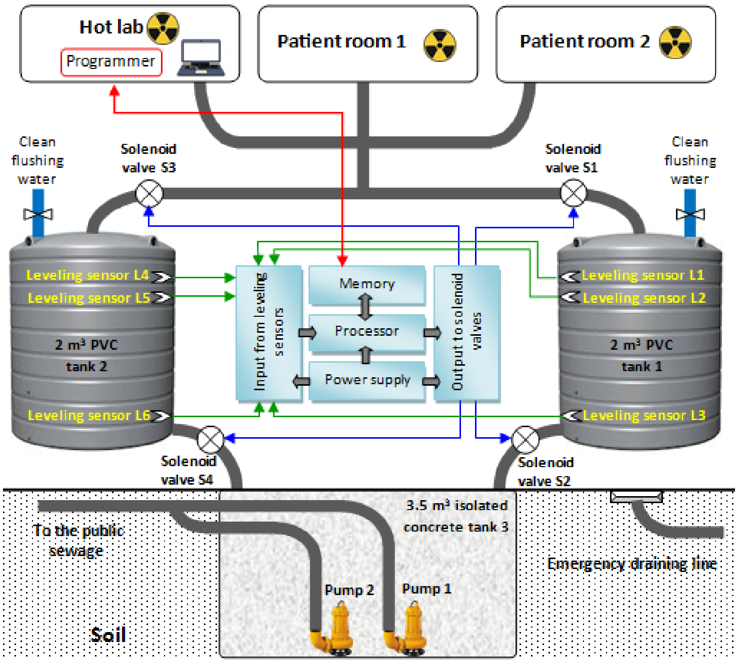

FIG. 1. Automated drainage system for PET-CT facility programmed and controlled using Programmable Logic Controller (PLC) unit.

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31-COMPETENCE IN RADIATION PROTECTION: ANALYSIS OF A MODULE ON RADIATION SAFETY AND KNOWLEDGE MANAGEMENT OFFERED TO A POSTGRADUATE EDUCATIONAL COURSE (PGEC)

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The abstract aims to analyze a module on Safety Culture (SC) and Knowledge Management (KM) offered to a Postgraduate Educational Course (PGEC) in Radiation Protection and the Safety, at the Regional Training Center (RTC) of the International Atomic Energy Agency (IAEA) in Brazil. The module lasted 3 days and was offered for the first time in 2019. After a positive evaluation by the students, the module may last more days, for a better immersion in the content and deepening of the debates on the theme of thematic security culture and knowledge management. Thus, it is understood the relevance of the two themes for the course and for the training of professionals who will work with radiation protection.

INTRODUCTION – THE PGEC

The Postgraduate Course in Radiation Protection and Security of Radioactive Sources (PGEC) in Brazil, is offered by the Institute of Radiation Protection and Dosimetry (IRD). The PGEC was designed to meet the needs of professionals with higher education at university level in different areas, such as physics, chemistry, health and earth sciences or engineering, and who are working in the field of radiation protection and radiation source safety in their countries. IRD is the IAEA's RTC for portuguese-speaking countries. The course, created in 2011, provides the foundations necessary for those who will become instructors in their area, forming qualified experts that will act as multipliers of the knowledge in the area. The course has a workload of 472 hours and is divided into 17 modules, with theoretical classes and practical training (such as demonstrations, laboratory exercises, case studies, technical visits, simulation exercises and workshops). Some theoretical topics and exercises are developed online using the courses virtual classroom [1-3].

THE SAFETY CULTURE AND KNOWLEDGE MANAGEMENT MODULE AT PGEC

The module lasted 3 days and was carried out as follows: on the first day the class on SC was given; on the second day the KM class; and on the last day a joint activity on the two themes with the students. 3 teachers and 18 students participated in the module. This was the last module of the course. In the section on SC, contents were addressed on: Fundamentals of SC and its evolution over time; SC attributes; SC levels and indicators; Evaluation and continuous improvement; Promotion and development actions; SC and management systems; SC in the regulatory bodies. In the KM section, content on: Knowledge characteristics; Difference between data x information x knowledge; Types of Knowledge; Knowledge processes; KM practices. At the end of the class there was also a dynamic in the classroom, where the students were divided into groups for a case study. In this study, students should: Describe the scenario and point out problems; Indicate the wrong strategies or errors that the company, in the study, could have made in the past and other potentials; What actions could mitigate the problem? Would KM practices (or processes) be recommended? What if there was a threat to the security of the company's operations? In the joint activity, on the third day, students were divided into groups and students were asked to analyze 5 situations in which it was possible to apply SC and aggregate learning with KM.

STUDENT ASSESSMENT OF THE MODULE

Joint activities were used to assess students. The general average of the class was 9.3 (Table 1). In addition, to check student satisfaction, a questionnaire with 9 questions was applied, in which students answered as yes or no (Table 1). Students were also asked to rate the module (Table 1). At the end of the questionnaire, students could critique the modules. In this sense, for student A., "*There is a greater need to increase the module's workload, so that there would be more time for debates and case studies*." For student P., "*The module calls for special attention to knowledge and the importance of individual security, with this module being more important in the initial phase*.". For student C., "*The study material could be translated into Portuguese, since it is in Spanish*.". For student J., "*From my professional experience, I observed that most services do not have this SC, and with this there is no commitment from employees to their activities. I suggest that this module be applied at the end of the course*.". For student G., "*The module broadly addresses the concept of security in facilities that have radioactive sources in order to improve everyone's view of the concept of radiological protection*." For student S., "*The module pointed out issues related to the concept of SC and how we should apply it according to the scenarios found. The KM theme reminded me of the importance of disseminating knowledge, adding values to the process*.".

TABLE 1. MDULE EVALUATION

Issues Evaluated	Grade
1) Did the Program achieve its objectives?	10
2) Was there a logical organization following the content?	10
3) Will the course contribute to a better performance of your activities?	10
4) Is the knowledge transmitted adequate to my work needs?	9
5) Did students' motivation during classes remain high?	10
6) Was the time taken to deal with program issues adequate?	7
7) Did the material provided cover the entire content of the program?	9
8) Did I, student, receive all the information necessary for my participation in this program on time?	9
9) Were the teaching facilities adequate?	10
10) Assign a grade to the module	9

FINAL CONSIDERATIONS

As expressed by the students, the module proved to be important for understanding the importance of SC in installations that use radioactive materials, and that this culture should be part of the work routine. Regarding KM, students understood that the dissemination of knowledge is an important tool for radiological protection. Also from the students' evaluation, it is understood that the course was very interesting, and the biggest criticism is in relation to class time. Another observation is regarding the moment that the module should be offered, whether at the end or beginning of the course. So, after a positive evaluation by the students, the module may last more days, for a better immersion in the content and deepening of the debates on the theme of thematic security culture and knowledge management. Thus, it is understood the relevance of the two themes for the course and for the training of professionals who will work with radiation protection.

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32-REGULATORY SYSTEM FOR SETTING OF RADIOACTIVE LIQUID AND AIRBORNE DISCHARGE LIMITS

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Limiting the radioactive discharges into envinronment is the accepted practice and key element of safety regulation in the States that use atomic energy [1], including Russian Federation [2]. Since airborne and liquid discharges, if they occur due to facilities' normal operation, are the main contributors to radiological environmental impact, in accordance with best international practice [1], [2] national regulatory bodies are set discharge limits. In Russian Federation the Federal Environmental, Industrial and Nuclear Supervision Service (Rostechnadzor) has a mandate for setting the radioactive discharge limits. This practice comply with recommendations of point 3.132 [3]. Evaluation of the discharge limits is based on the regulatory documents (methodologies) developed by SEC NRS and enacted by Rostechnadzor [4] and [5].

To follow recommendation of 3.132 [3] to carry out discharge limits authorization with account of radiological environmental impact, Rostechnadzor empowered its two technical and scientific support organizations [6], including SECNRS, to fulfill review of materials of discharge limits assessment. This practice is widely accepted in other States that use atomic energy [7]. In accordance with [8] review of materials of discharge limits assessment is necessary for discharge limits authorization.

The methodologies [4] and [5] are establish:

– criteria for exemption of source of discharge of radioactive substances from regulatory control (i.e. criteria based on which a decisions are made that there is no need to establish discharge limit for a specific source);

– criteria for the selection of radionuclides, for which it is not necessary to establish discharge limits (radionuclide exemption criteria);

– criteria on the basis of which the discharge limits for those discharge sources, that are subject to regulatory control, are defined;

– exposure pathways that must be taken into account when setting discharge limits (cloudshine, watershine and groundshine external exposure; contaminated foodstuff and inhalation internal exposure);

– identification of environmental and dosimetric models' parameters (e.g. wind characteristics (velocity, direction), atmospheric stability classes, Kd factors, food consumption rates), which are crucial for discharge limit evaluation.

According to the Russian approach [4] and [5] the criteria for selection of discharge sources and radionuclides, for which discharge limits shall be set, are:

– if annual effective dose due to the source exceeds 10 μSv (without dilution), the discharge limits shall be set for this source (i.e. the non-dilution approach recommended in [9] and [10] is implemented), so both point and diffuse radioactive discharge sources shall be regulated;

– for those radionuclides discharged, which contribute 99% in annual effective dose due to specific discharge source, setting discharge limits is required (radionuclides which contribute less than 1 % are exempted from discharge setting).

According to Russian legislation the list of radionuclides, which are subject to regulatory control and from which the radionuclides, that are subject for discharge limits setting, are selected, is rather prescriptive and approved by the Government of the Russian Federation [11]. The list consists of 94 radionuclides, regulated in airborne effluents and 81 radionuclides regulated in liquid effluents. The list also includes global radioactive pollutants ^{3}H and ^{14}C. The list is compiled based on the Russian and foreign experience (for example, [12]), taking into account the radionuclides pertain for operation of nuclear power plants and spent nuclear fuel reprocessing.

It should be noted that a particularity of the documents [4] and [5] is that they establish only overarching principles and criteria for discharge limits evaluation and minimize the requirements for methods for calculating the model parameters such as dilution factors, water to plant and soil to plant transfer factors, dose coefficients etc. Thus, requirements to the models, input data and their conservatism, are guided by the availability of the results of field observations, which carried out at the location of discharge source. These requirement are quite flexible as far as the models and input data are selected and justified by the developer of discharge limits. However, in order to facilitate compliance with the requirements of [4] and [5], SEC NRS has developed and Rostechnadzor has approved safety guides with recommendations on preferred models and input data. So the structure of Rostechnadzor documents, establishing requirements and recommendations for discharge limits evaluation, is shown in the Fig. 1.

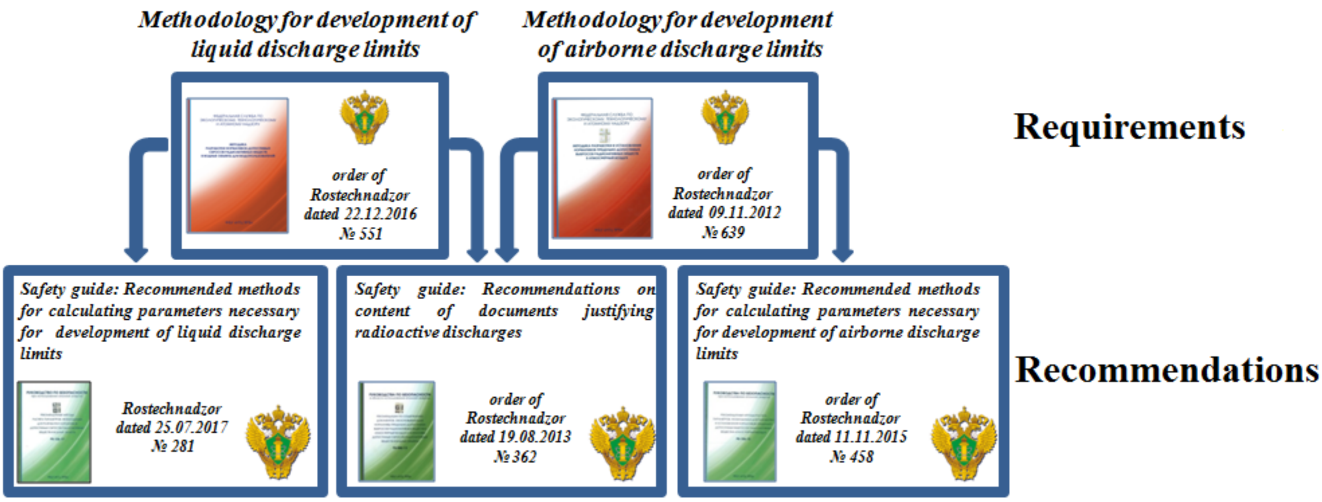

FIG. 1. Legal and regulatory basis for limiting of radioactive airborne and liquid effluents

It is important to note that the provisions of more than 20 documents of international organizations such as the IAEA and ICRP and experience of other countries with highly developed nuclear industry are taken into account within the framework of development of the legal and regulatory basis for limiting of radioactive discharges.

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35-NUMBER OF SEGMENTS AND MU EFFECT ON OUT-OF-FIELD DOSE IN DIFFERENT INTENSITY MODULATED RADIOTHERAPY CASES

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BACKGROUND

The protection of normal tissues can be performed by the collimation of radiation beam. For irregular radiation fields, either Cerrobend blocks[1] or Multi-Leaf Collimators (MLCs) can be used [2].

The main advantages of MLCs are: shorting of treatment time, less time consuming in the preparation, This eliminating the need to handle the Wood's alloy which is toxic. The possibility of treatment information transfer and providing a clean environment [3].

Conformal radiation therapy that confined the radiation dose to Planning Target Volume (PTV) but IMRT has additional advantage of sparing Organ at Risks (OARs) which is difficult to spare with 3D-CRT [4].

Dose escalation strategies through the 3D-CRT are superseding conventional techniques resulting in a better therapeutic outcome profile but some points like the optimal number of beams, field arrangements and shaping methods to spare the adjacent normal tissues still remains controversial [5].

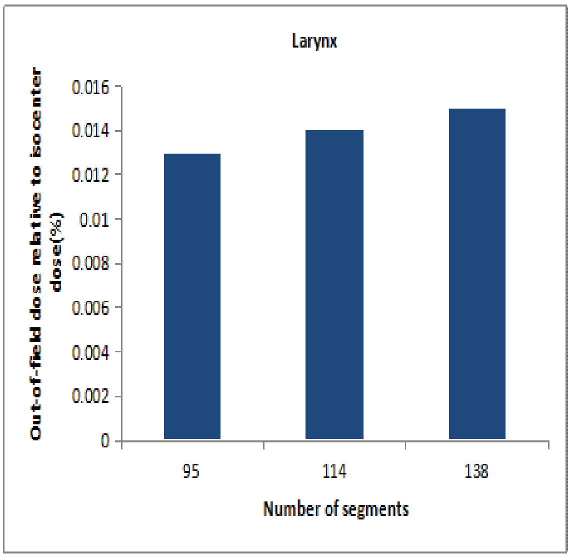
In many times with 3D-CRT, it is difficult to spare the OARs without compromising the PTV coverage where IMRT has the ability to produce the desired dose distributions shaped to PTV with sparing of OARs [6].

METHODS

This work aims at providing estimation for out-of-field dose in different intensity modulated radiotherapy cases included larynx, shwanoma, oropharynx and tongue cases. This is carried out through the employment of Pinpoint ionization chamber detector in an estimation of the change in out-of-field dose that reaches the organ at risk at different distance from field edge (as left eye where this organ use in this work as organ at risk) with different number of segments and different MU in these cases.

RESULTS

Results show that there is almost no increase in the out-of-field dose that reaches the left eye with increase in number of segments in all of the investigated cases as shown in fig.1. At small distance from central axis (3 cm) and small MU (411), the mean out-of-field dose relative to isocenter dose that reaches the left eye is large (0.48%) as shown in fig.2. In this case it is recommended to use an eye shield to protect the eye from the out-of-field dose. At large MU (912) and large distance from central axis (14 cm), the out-field dose relative to isocenter dose that reaches the left eye is negligible (0.014%) where the out-of-field dose decreases with distance from central axis.



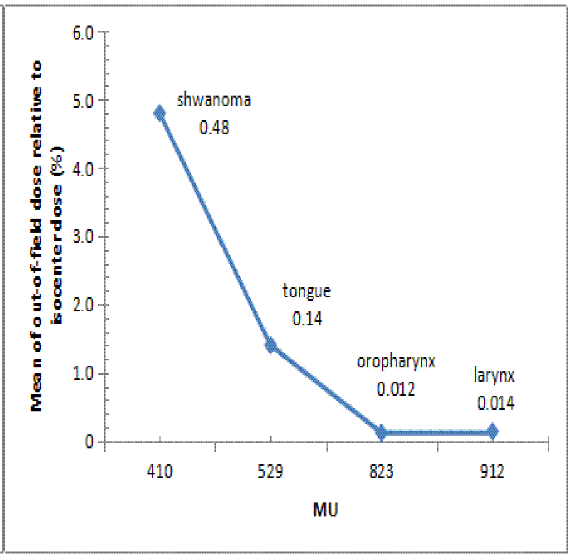


Fig .1 Variation with the number of segments of Out-of-field dose.

Fig. 2 Variation with MU of mean out-of-field dose

CONCLUSION

at small distance from central axis (3 cm) and small MU (411), the mean out-of-field dose relative to isocenter dose that reaches the left eye is large (0.48%). In this case it is recommended to use an eye shield to protect the eye from the out-of-field dose. At large MU (912) and large distance from central axis (14 cm), the out-field dose relative to isocenter dose that reaches the left eye is negligible (0.014%) where the out-of-field dose decreases with distance from central axis

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38-DETERMINATION OF THE RADIATION SHIELDING PROPERTIES OF MAGNETITE AGGREGATE CONCRETES

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ABSTRACT

The radiation is widely used in many areas in daily life such as; in medical treatments as well as the industrial application in thousands of usage throughout developed societies as a resul of technological improvements, it becomes one of the facts in our life. The major sources of radiological dose to the public are natural and medical sources of radiation. Any source of radiation, as with most hospital radiation sources and some industrial processes can be entirely shielded to protect workers and the member of the public.

The principle and methods of radiation protection have gained great importance in terms of radiological workers and members of the public as well as the bad impact of our living environment. The time, distance, and shielding properties are three basic rules of the methods of radiation protection. The most important and more effective of these three rules of methods of radiation protection is the shielding properties. In this study,the linear attenuation coefficient of magnetite aggregate concretes in many different rates has been tested practically by using a gamma spectrometer against real gamma isotropic sources. The Cobalt (^{60}Co) and Sodium (^{22}Na) emitted four energies; 0.511, 1.173, 1.275, and 1.332 MeV the results were supported with XCOM online soft program calculation. The measured linear attenuation coefficients are displayed, where it can be concluded from this work that the linear attenuation coefficients decreased with the increasing photon energy..

39-RADIATION PROTECTION ASSESSMENT OF IONIZATION INSPECTION SYSTEM KIND (RAPISCAN-GARDS)

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The international production of the ionization inspection system has been increasing year-by-year, especially after the events of September 11th, 2001. Since that, many countries needed high-efficiency systems to detect to conserve both time and effort. These systems entered and started development in Iraq and other countries, especially after 2003, a lot of Iraqi and world sectors became dependent on these systems in the inspection and detection process. However, the using of these systems mainly, nowadays, as an optimized solution in any country or organization to save the security situation. The workers of these systems are border security. Therefore, their knowledge in the radiation protection field is very limited, and poor, and their exposure to radiation is highly possible.

The aim of this research is to measure the accumulated equivalent dose to the radiological workers (operator) and the under inspection vehicles drivers and member of public and then compare with constrains of radiation exposure limits used by other countries which use these systems as a first step to put the local limitations. The published version of IAEA (GSR. Part no. 3) in 2011 mentioned that using these systems are forbidden only after prevented the getting the justifications of the regulatory body for each country.

Some dosimetries are need to be give to the number of truck drivers which passing through the border or number of truck drivers and instruct them to pass through the scanning system with different speeds to measure the accumulated equivalent dose to the truck drivers, assistants, operators, and general people. The results need to compare with the theory calculations. So, all these values should submit within the commitment limitation which documented in the operation manual of the system.

The radiation sources are widely using in our modern life via it can penetrate matter, they have many use in medicine, agriculture, industry, mining, oil exploration, and research [1]. To avoid overexposure, radiation protection methods should apply on the equipment design and the uses of procedures. The goal is to ensure that exposures are as low as reasonably achievable (ALARA) and within acceptable limits [2-3]. Three basic physics criteria are used to measure the effectiveness of any imaging system: penetration, contrast, and resolution, it related to the level of energy, hence the number of photons sent through whatever is being scanned [4].

- Penetration is probably the foremost of the three criteria for cargo screening, the inspection fails if the imaging photons lack the energy to punch through a container's thick steel walls, the key to penetration is photon energy– the more energetic, the deeper the photons penetrate a material.
- Contrast sensitivity, the second important criterion for cargo screening, is important for distinguishing between items inside a container. Imaging experts say that the higher the contrast sensitivity, the greater the chances for detecting contraband. Ionization inspection based systems have proven to be ten times more contrast sensitive than other systems "The aim of nonintrusive screening is to image the contents of a cargo container with enough clarity to decide about the contents".
- The third criterion, resolution, is a measurement of the ability to see spatial details in an image. If you are looking for hundreds of pounds of drugs, just knowing there's something large and unexpected inside the container is enough. However, if you're looking for nuclear weapons components, which can be small, you want the best resolution you can get.

The International dose limitation:-

- European Committee:- The proposed European approach determined in the total effective dose to which the general population that exposed to (1mSv / yr) from all radiation sources, according to the recommendations issued by the International Commission for Radiation Protection (ICRP), and set the European Commission radiation dose from the examination organizations using ionizing radiation with only 25% of this limit, equivalent to 250µSv/ year. [5].

- The American Standard: - The American approach to screening systems using the ionizing radiation used in the security offender determines the radiation dose to which the general public from ionizing radiation screening systems is subjected to 25% of the recommendations issued by the International Committee for Radiation Protection (ICRP), equivalent to 250µSv / year [5].
- It is mentioned in the international basic safety standards BSS for atomic energy that there are three types of radiation exposures. As for the exposures resulting from the use of radioactive sources on the security side, they are not authorized except with the authorization of the supervisory authority for each country [6-7].

CONCLUSION

According to the field visit, calculations, and interviewing the operator of the system, the team noted that the equivalent dose depending on the screening time; the system screening time takes in a single examination process (3-7 seconds) the equivalent dose as shown below.

- The annual equivalent dose of the operators inside the Rapiscan truck compartment are within the limits of the radioactive workers because the irradiation window of the radioactive source arranged in the form of a narrow two-dimensional window and only towards the sense pad, where the readings at the driver's compartment were close to the natural radiological background of the area.
- The annual equivalent dose for the vehicle drivers (checked trucks) that passing through the inspection system, the annual dose were within the limits set for the public (less than 1mSv / year) with regular operation of the system so that the traffic of the vehicles examined is the opposite of the direction of the Rapiscan truck, so it is seating position the driver is 4m away from the source and when the radioactive source is at a height of 0.5m from the ground and when the speed of the truck is within 1.5 km / hour (0.4m/ s) because the parts of the truck have become as a shield it protects the driver from exposure to radiation. As the equivalent dose, for each examination process was within (0.185 µSv/ inspection) and this shows that the equivalent dose remains within the internationally permitted limit, it uses even the system for (5000) examination process.
- For the truck driver's assistances or anyone other than the driver, they forbid it to be inside the vehicle during the examination, and to be outside the control area to prevent unjustified radiation exposure to them.
- For small vehicle drivers, bus drivers, and other types of vehicles, the examination process using the second method of inspection approved in the operator’s guide will stop the vehicle that examined and all people from inside it, including the driver, be sent. And the RapiScan truck will pass over it to avoid unjustified exposure to ionizing radiation.

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40-DOSE ASSESSMENT IN NORMAL OPERATION AND ACCIDENTAL SITUATION AS PART OF THE SAFETY ASSESSMENT OF A CENTRALIZED WASTE STORAGE FACILITYAN EXAMPLE IN PRACTICE OF HOW TO APPLY SAFRAN TOOL FOR SAFETY ASSESSMENT PURPOSES

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This paper shows a practical exercise using SAFRAN tool as part of the Safety Assessment of a centralized storage facility through the application of the IAEA recommendations.

CASE STUDY

To carry out the dose assessment in both, normal operation and accidental situation, a generic facility was created and called “Waste Storage Facility” (WSF). The “Regulatory Framework” was completed using data corresponding to the dose limits and constraints established for workers and public by the Regulatory Authority.

GENERAL CONSIDERATIONS MADE FOR THE CALCULATION

The following was calculated using SAFRAN (according to the scenarios set out below):

- Dose received by a worker who performs certain tasks corresponding to normal operation;
- Dose received by a worker in case of the fall of a container during handling;
- Dose received by a member of the public in case of a fire in the WSF.

Other considerations taken:

- Calculations in the software were made only for Co-60 sources already conditioned in containers, coming from disused radiotherapy cobalt units and industrial irradiators.
- Two types of waste components were loaded: a 200 L container holding capsules with Co-60 sources and capsules itself (0.0082 m^3).

The following table shows data from the storage of sources that were introduced in the software.

TABLE 1. ASPECTS OF THE WASTE COMPONENTS INTRODUCED IN THE SOFTWARE FOR DOSE CALCULATION

Radionuclide	Source Activity (Bq)	Number of Sources	Total activity per capsule (Bq)	Total Activity (Bq)	Sources per container	Number of containers
Co-60	7,4E+09	702	3,70E+10	5,19E+12	120	6

Normal Operation

The following activities were identified as those which contribute the most to occupational exposure:

- Reception and location of DRS (already conditioned inside the containers) in the facility. Estimated time for running the activity: 45 min / operation * 30 operations / year.
- Routine radiological controls performed in the facility. Estimated time for running the activity (12 controls / year * 30 min / control).

Accidental Situation

Table 2 reflects aspects introduced in the software for dose assessment in accidental scenarios.

TABLE 2. CRITERIA USED FOR CALCULATIONS IN ACCIDENTAL SCENARIOS

Accidental Scenario	Prob. of occurrence	Person affected	Distance	Other interesting aspects
"Fall of a container during handling"	Medium	Worker	1m (between worker and the container)	- The exposure time considered was 30 minutes. - The geometry considered for the container was an axial container geometry, 89 cm high and 28 cm radius, without shielding.
“Fire in the WSF”	Low	Public	30 meters from the emission in the direction of the pen.	- It was considered a partial affectation of the WSF and a Co-60 retention fraction of 99%. -Other data needed for calculations were obtained from the software.

RESULTS

Finally, the calculations were carried out selecting the analysis criteria and parameters and then, running the SAFCALC model. The results obtained are shown below in Figure 1.

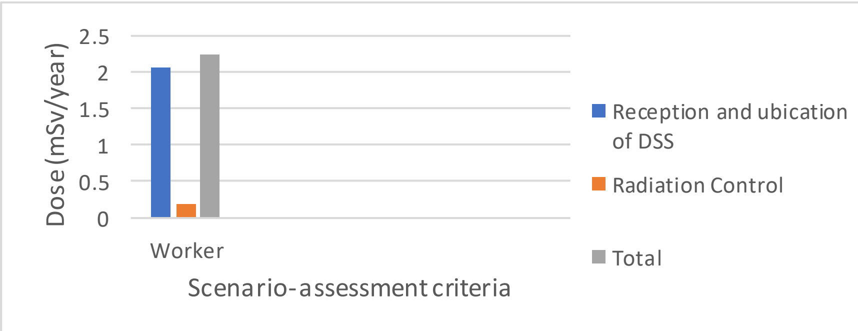

FIG. 2. Doses received by the worker due to normal operation

According to the information presented in Figure 1, the total dose received by the worker who performs both activities is 2.25 mSv / year. This value remains under the constraint established for this practice by the Regulatory Authority.

TABLE 3. DOSE CALCULATED FOR BOTH POSTULATED ACCIDENTAL SCENARIOS

Accidental Scenario	Person affected	Dose received (mSv/year)
"Fall of a container during handling"	Worker	4.23
“Fire in the WSF”	Public	2.8

The doses calculated for both, normal operation and accidental situations, were considered reasonable for the scenarios postulated and are consistent with the considerations made. The exercise carried out showed that, although SAFRAN is still a modern tool that is currently under development, it can be adapted to various situations and diverse characteristics in the Safety Assessment for different kind of facilities.

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41-SAFETY IN RADIATION ONCOLOGY (SAFRON) DATABASE ON REPORTING OF RADIOTHERAPY INCIDENTS AND NEAR MISSES: A SINGLE INSTITUTION EXPERIENCE ON RADIOTHERAPY EVENTS REPORTING AND LEARNING SYSTEM

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OBJECTIVES: SAFRON is a database of error reporting and incident learning system (ILS) on radiotherapy developed by International Atomic Energy Agency (IAEA) in December 2012. Identifying and analyzing safety-related events is a proven way to enhance the quality of cancer care and radiation delivery. Adapting SAFRON will help to assist our institution in promoting safety culture and to improve patient safety through analysis of incidents and formulate action plans to minimize errors in the future. This study will present a descriptive analysis about the radiotherapy events and near misses in our institution utilizing the SAFRON database.

METHODS: This study utilized SAFRON database of incidents (July 2017 – June 2019) reported by the Department of Radiotherapy, Jose R. Reyes Memorial Medical Center, Manila, Philippines. Data were analyzed based on the following: 1.) who discovered the incidents; 2.) how the incident was discovered; 3.) reachability to patients; 4.) process phase; 5.) clinical severity, 6.) failure of safety barriers. The results were analyzed using descriptive methods.

RESULTS

There were 167 reported incidents to the SAFRON database by our institution, majority of which were reported by radiation therapist (74%). Most of the incidents were discovered during chart checks (81%). More than two-thirds of the incidents occurred during the treatment phase while based on clinical incident severity, 80% were minor incidents and around 18% were classified as near miss with potential serious consequences.

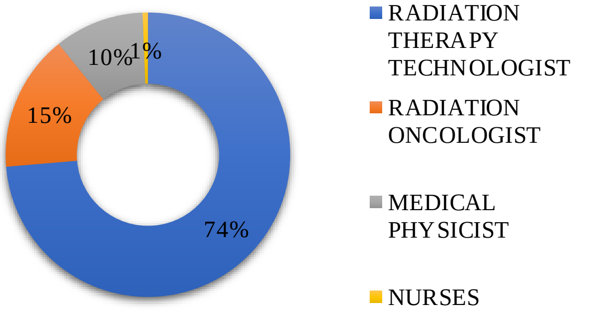

FIG. 1. Distribution of Radiotherapy Staff in Reporting

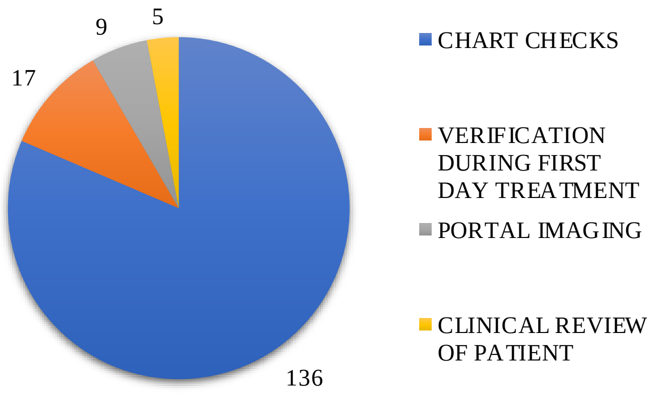

FIG. 2. Distribution of Events by Method of Detection

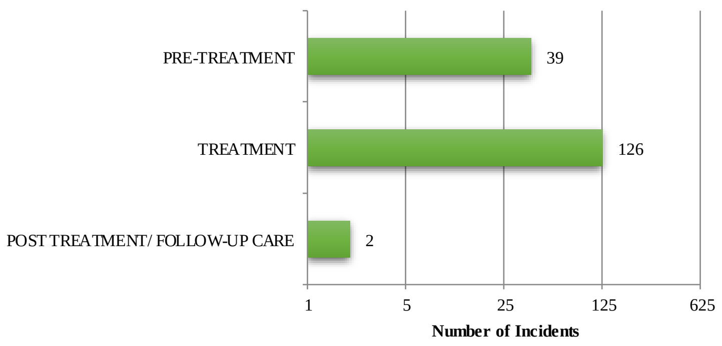

FIG. 3. Distribution of Events by Process Phase

We further investigated into what safety barriers were breached which includes: failure to do time-outs, failure to review treatment plan and inability to do independent confirmation of dose, and absence of record verifying system.

CONCLUSION: SAFRON demonstrated that ILS will promote and reinforce the safety and quality of radiotherapy. Our institution's participation to SAFRON have led to increase awareness of safety culture and reporting of treatment errors through regular chart checks and clinical review of treatment plans; strengthening the identification of errors and near misses especially during the treatment phase, detection of it's clinical severity and reachability to our patients; and furthermore identifying our safety barriers that need improvement and modifications.

ACKNOWLEDGEMENTS

The authors would like to thank the staff of the Department of Radiotherapy, Jose R. Reyes Memorial Medical Center, Manila, Philippines. This paper is dedicated to all cancer patients who deserve quality and safe radiotherapy and care.

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42-ANALYSIS OF DEMAND AND SUPPLY OF TECHNICAL SERVICES FOR NON-POWER NUCLEAR SCIENCE AND TECHNOLOGY APPLICATIONS IN TANZANIA

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ABSTRACT.

Tanzania has advanced in nuclear science technology applications through the Government support and valuable assistance provided by international organizations for nuclear technology applications development. This advancement has created demand for technical services in radiation protection, dosimetry services, calibration services, quality controls services, packaging and transport services, waste management services, technical consultant on nuclear technology applications, training and research. It is evident that availability of effective technical services in nuclear science and technology applications enables operators to comply with regulatory requirements. Therefore the increasing nuclear science and technology applications in the country and the inadequacy in regulatory infrastructure call for public, private, local and international stakeholder's participation in the industry so as to improve compliance so as to explore the potential of the industry.

This paper is intended to show the general overview of the trend of the development of the nuclear science and technology applications in Tanzania, shortage of the technical services, and the inadequacy of the regulatory infrastructures in Tanzania.

Key words: Technical Services, Effective Regulatory Control

45-PERCEIVED VS. ACTUAL - THE CHALLENGE OF COMMUNICATING RADIATION RISK IN THE MODERN WORLD

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

Recognising this challenge the Society for Radiological Protection, (UK Chartered Professional Body for the field of Radiation Protection), has launched a number of collaborative work streams to tackle this issue. This has included:

- Running a workshop as part of their 2019 Annual Conference, including invited talks from science communicators, journalists and practitioners involved in the communication of radiation risk post the accident at Fukushima Daiichi. The output was captured in a paper published in the Journal for Radiological Protection [1].
- Launching a workstream aimed at developing a series of short (~ 10 pages) specific user guides to Communication Radiation Risk in certain scenarios such as in support of Outreach, Emergency Preparedness or Medical Exposures, targeted at practitioners. This ongoing workstream draws on the experience of a variety of contributors, including UK Government, the Regulators, Media and Social Media Specialists, Radiation Protection Practitioners and Local Authorities.
- Actively supporting the International Radiation Protection Association (IRPA) in developing its guiding principles for engagement with the public on radiation and risk.
- Contributing to various international work streams including as an invited speaker and panellist at the NEA Workshop on Stakeholder Involvement: Risk Communication, and Japanese Health Physics Society 2019 Conference.

The proposed talk will explore the origins of the negative public perception of radiation and how the learning from the above work streams can be applied to help tackle this challenge.

To assist in exploring this perception an example provided by EDF will be used to provide an indication of the scale of the challenge. This example was encountered as part of the construction of the cooling water system at Hinkley Point C in Somerset. Here, there was a need to dredge sediment from the seabed and deposit it in a licensed disposal area. Due to this sediment being located near an existing operational Nuclear Power Station concerns were raised by the public on the potential for the sediment to be radioactive, posing a perceived risk to human health and the environment. Concerns were still raised despite extensive independent analysis that showed the sediment was not radioactive under UK law.

Due to the scale of the social media campaigns against the dredging operation, two petitions to stop the operation attracted national and international media coverage and resulted in a number of high profile protests.

The talk will explore the strategies, tools and techniques used successfully to build trust with stakeholders whilst also considering the wider communications techniques that supported relationships with the general public and media, and the lessons learnt which have fed into SRPs ongoing work streams.

It is hoped the talk will provide a useful perspective and guidance to those involved in the communication of radiation risk.

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SRP would like to acknowledge the contributions to the various external parties involved in the production of the guidance and previous workshops including: UK Government, The Environment Agency, EDF Energy, The Office for Nuclear Regulation, RadSafe, Japanese Health Physics Society and International Radiation Protection Association.

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46-USING THE DIELECTRIC CONSTANT AS A FUNTION FOR NEUTRON FLUX IN EXTREME ENVIRONMENT: TO DETERMEINE APPROACHED DOSES

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In this work, the dielectric constant of selective material using MAS-Li-B system to measure the neutron flux was used as an indirect function to estimate the approximated dose of the neutron exposure in the extreme environment. The MAS-Li-B system was consisted from MgO-Al_2O_3-SiO_2-ZnO-Li-B. This system has good properties that stand in high temperature because of its good mechanical properties, good cross section for neutron interactions (absorption) and its electrical properties that changed hardly in such conditions. The environment was used with high flux neutron source, both Cf-252 and Am-241/Be in addition to high temperature in the range 100-500 °C. The dielectric constant was measured using LCH meter. The neutron flux was ranged by using appropriate shielding in each step of increasing the neutron flux. The change in the neutron flux was used to study how the dielectric constant effected by the neutron interaction. It found that the dielectric constant changed consequently with neutron flux. The dielectric constant in specific frequency decreases with increasing of neutron flux by indirect interactions that affect the dielectric properties. This system showed good features that stand in hard conditions when other kinds or traditional neutron detector failed. And finally, the dose from the neutron radiation was measured.

The certainty of the measurement was compared with other procedure that done for same source. It was used the inverse beta decay by measurement of antineutrino by specific detector using GEANT4 Code through Monte Carlo Neutral Particle transition. Through this process, neutron flux estimated and consequently the neutron dose. This method provided approached results that strength the first method to measure the neutron flux.

This untested method before for measure the neutron dose gives new line and invented method in radiation protection to ensure safe environment specially in the nuclear accidents or in the core of nuclear reactor where it hard to reach such places.

47-INFLUENCES OF CRISIS IN SYRIA ONOCCUPATIONAL EXPOSURE MONITORING

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A large group of workers in many fields are exposed to ionizing radiation, for them it is important to monitor this exposure periodically and take the necessary protection procedures to minimize this exposure to as low as reasonably achievable (ALARA principle).

The Atomic Energy Commission of Syria (AECS) is the sole provider for personal monitoring service in Syria. The work presents a detailed analysis of occupational exposure data during the period from 1990 to 2017 for the monitored workers in Syria [1]. The types of work for monitored workers have been classified into five main categories as in UNSCEAR report [2]. Sub-classification for monitored workers has been also carried out according to the practices. Using the national dose record, the numbers of monitored workers were determined each year and the collective doses and average annual doses were calculated for each practice.

In addition, explanations for causes ofsome high exposures were given with suggestions andrecommendation to reduce these types of exposure. However, rare real high exposures were noticed. Most of cases were for forgotten films in the imaging rooms and intended / non-intended direct exposure on the dosemeter from X-ray tube or contamination of the holder by a radionuclide.

Indeed, the crisis in Syria has affected the personal monitoring service. The number of monitored workers increased from 924 workers in 1990 to 5657 workers in 2011, the first year of crisis; after that the number decreased to 1288 in 2017 (Fig. 1).

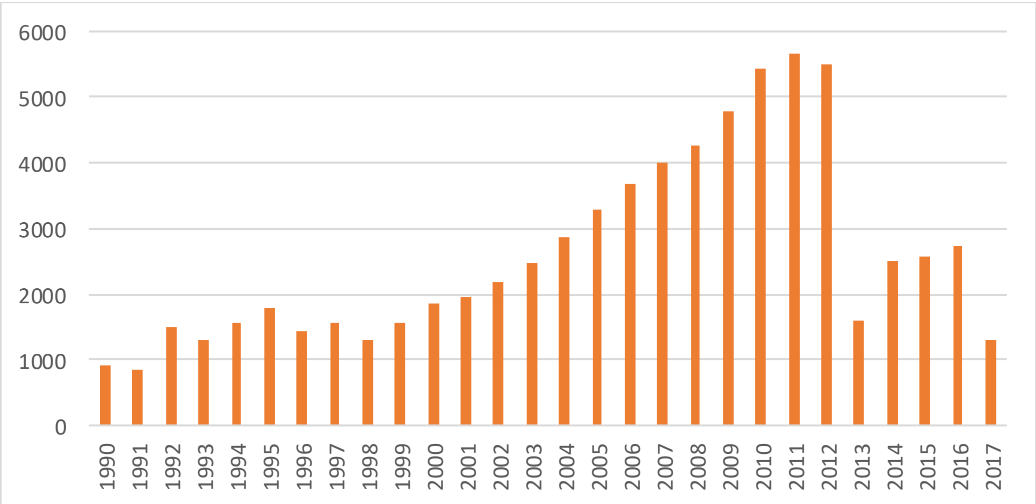

FIG.3. Total monitored workers from 1990 to 2017.

Many logistic and technical modifications were made to overcome some difficulties. Before 2013, KODAK personal monitoring films type II were in use. Because of economic boycott, obtaining of such dosemeters was not possible. Personal monitoring films from FOMA BOHEMIA (Czech Republic) represented a good alternative as they have size and characteristics close to those of KODAK films.

Due to the troubles in different areas in the country, the delivery of films to the monitored establishments was a challenge. On the one hand, the monitoring period was changed from two months tothree months. On the other hand, methods of delivery, other than the Post office, were discussed with the establishments' owners. Solutions were arranged on, such sending films by AECS workers living in the same area of the monitored establishment, or collecting films by hand in AECS centre.

Human resources posed also big problems. Many qualified workers left their jobs. Some of the new-nominated liaison officers, with no knowledge or low knowledge in radiation protection principles, were incapable to understand how to distribute the films to monitored workers according to the provided list. Others were ignorant that the film needs to be put into its holder before being used and that there is a specific way to put the film in the holder. After receiving the new films, some liaison officers were keeping used films for two or three periods instead of sending them periodically to AECS for processing and issuing dose report.AECS organized many training courses, during the last period,oriented to qualify radiation protection officers. Also, a circular,with detailed information about the service, was sent to all relevant establishments in order to provide AECS with updated lists of monitored workers and to specify their needs of holders instead of those missed or destroyed.

The lessons from AECS experience in this regard can be utilized for better management of radiations does in such crises.

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49-ESTABLISHING LOCAL DIAGNOSTIC REFERENCE LEVELS (LDRL) FOR A TYPICAL FLUOROSCOPIC EXAMINATION IN TWO RADIOLOGICAL IMAGING INSTITUTIONS IN GHANA

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BACKGROUND

The primary aim of the research work was to propose the establishment of local diagnostic reference levels for typical fluoroscopic examinations in two radiological imaging institutions in Ghana. It also aimed to investigate the distribution levels of patient radiation dose received during fluoroscopic examinations for subsequent improvement of optimization.

METHODS

Prior to the starting of this research a series of Quality Control Tests were performed using the Piranha kit to assess the whether the machine output is exactly what is expected. The research was done at two public hospital, located in the Greater Accra Region, between December 2017 and June 2018. A prospective quantitative research method was adopted to obtain frequency of fluoroscopic examinations in the Radiology Department [1]. In use, the KAP meter with all associated electronics was placed perpendicular to the central beam axis and in position to completely intercept the entire area of the X-ray beam as shown in Fig 1 [2].

KAP data values of the fluoroscopy procedure performed were obtained from the machine's console after each patient's examination. However, no additional adjustments or scan protocols were used for this research, to ensure the study reflected the actual normal practices in all the centres. The data obtained was statistically analysed using Microsoft Excel and results presented in descriptive statistics. DRLs do not represent a boundary between good and bad medical practice. However, DRLs do assist in investigating unusual high or low doses in facilities. Hence DRLs are tools for promoting the process of optimization in clinical settings [3].

FIG. 1. *KERMA X-plus iba dosimeter (KAP meter) on an X-ray tube*

RESULTS

A total of one hundred and thirty-six (136) patient dose data was collected for this study. DRL was established for the most frequently performed procedure which is hysterosalpingogram (HSG) examination and the results compared with studies done in Kenya [4] which is shown in table 1.

TABLE 1. Comparison of Diagnostic reference levels (DRLs) of KAP values for Hysterosalpingography for this study with a study done in Kenya

Examination	Facility A	Facility B	Wambani et al, 2014 Kenya Study
HSG	6.0 $Gy.cm^2$	4.1 $Gy.cm^2$	3.0 $Gy.cm^2$

CONCLUSIONS

The research work provided the frequency of fluoroscopy procedure for HSG examinations and the typical values of the related dose quantities, surveyed in two facilities in the Greater Accra of Ghana. The fluoroscopy examinations are description of the current practice in these facilities, hence the proposed diagnostic reference levels estimated could serve as a guideline before the establishment of a regional or national DRL. Local DRLs were established in terms of KAP, screening for HSG examination based on the 75th percentile dose values from the survey. Appropriate inter-comparison studies were done with International values to confirm whether these facilities in Ghana were meeting international standards or internationally proposed DRLs.

Generally, the KAP values estimated as the DRL in the study were not deviating much from other studies. The Kenyatta National Hospital [3] had a higher value than that of Facilities A and B by a factor of 3.5 and 4.2 respectively. Due to observed variations in KAP values, the work suggested standardization of protocols across facilities as a means to increase optimization of doses.

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51-TOWARDS ESTABLISHING NATIONAL DIAGNOSTIC REFERENCE LEVET FROM COMPUTED TOMOGRAPHY IN SYRIA - A PILOT STUDY

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Computed Tomography (CT) represents an important and powerful tool for diagnosis and treatment planning nowadays. Its ability to produce high resolution tomographic 2D images with three dimensional representations permits to practicing physicians to obtain more accurate and fast medical decisions which explain the vast increase in CT procedures performed yearly. However, the associated radiation dose with this imaging modality was relatively important which incorporate with an increased carcinogenesis risk to patients [1]. Therefore, the radiation protection rules in terms of justification and optimization should be respected in order to minimize the relevant radiation hazards. The International Commission on Radiation Protection (ICRP) introduced the Diagnostic reference levels (DRLs) in 1996 as an optimization process to identify the abnormalities in the radiation doses delivered to patients in radiological diagnostic modalities through the comparison with a dose threshold balancing between the diagnostically desired image quality and its related radiation dose. Moreover, the DRLs are recommended to be determined on a national or regional data so it could meet the national requirements which falls in with ALARA principle (as low as reasonably achievable) [2].

In Syria, the number of CT scanned patients and the range of CT examinations were also increased. The aim of the present study is to investigate the current radiation dose for several adult CT procedures using two dosimetric quantities: dose length product (DLP) which quantify the total radiation dose delivered to patient during CT scan and the CT volume index ($CTDI_{vol}$) as an indicator to the radiation output of CT scanner. These CT dose quantities are provided by the CT scanners at the end of each scan and they are recommended by the ICRP to establish CT diagnostic reference level [3]. A pilot study was conducted in two major teaching hospitals in Damascus for two months in 2019: ALBAYROUNI University Hospital and ALASSAD University hospital, where two multi-slice CT scanners are available: Siemens SOMATOM Perspective 64 slices and Toshiba Activation 16 slices.

A survey of 453 randomized CT scanned patients in the selected hospitals was established. For each patient the following parameters were recorded: Patient weight, CT protocol, peak tube potential, tube current, $CTDI_{vol}$ and DLP. The CT dose data was collected manually from the CT console whereas the accuracy of these values are guaranteed by the regular CT scanner quality control tests effectuated by Atomic Energy Commission of Syria. The targeted CT examinations were: Sinuses, Chest-Abdomen-Pelvis (CAP), Abdomen-Pelvis (AP), Lumbar Column, Chest, High-Resolution Chest and Head. The average patient size was considered when patient's weight is between 60 and 80 kg. The statistical analysis was performed using Microsoft Excel Data Analysis tool. The quantitative variables are presented as mean ± standard deviation. Only the CT dose quantities DLP (mGy.cm) and $CTDI_{vol}$ (mGy) for the average patient size were taken into account so their values are averaged and the values of the third percentile are calculated as the suggested preliminary DRLs values for the selected CT examinations for each hospital. Table 1 shows the descriptive statistical values of the $CTDI_{vol}$ and DLP from the both participated hospitals.

The calculated DRLs were communicated to each participated hospitals in order to be taken into account during the patient CT scan optimization process. Moreover, the DRLs values were compared with selection of DRLs from different countries as France [4], UK [5], Norway [6], Switzerland [7], Germany [8] and Ireland [9]. The comparisons of DRLs values, presented in Table 2, permitted to evaluate the local status of CT scan procedures level in the scope of optimization of patient radiation protection. The variations between the obtained DRLs values compared to other countries were revealed for many CT examinations which are principally due to the differences in clinical practices in terms of scan parameters or techniques. A follow-up study integrating more hospitals national-wide are planned to performed in order to optimize the patient doses in CT towards establishing a national DRLs values.

TABLE 1. DESCRIPTIVE STATISTICAL OF THE CT DOSE QUANTITIES FOR CT EXAMINATIONS FROM BOTH HOSPITALS

Examination	Number of patients	$CTDI_{vol}$ (mGy)			DLP (mGy.cm)		
		Mean	Standard Deviation	3rd Percentile	Mean	Standard Deviation	3rd Percentile
Sinuses	23	23.25	0.29	23.30	281.08	34.21	289.95
Chest-Abdomen-Pelvis (CAP)	69	18.65	3.44	20.39	1083.78	205.19	1160.70
Abdomen-Pelvis (AP)	24	21.11	5.16	21.20	948.64	141.55	987.60
Lumbar Column	21	27.32	7.02	28.30	775.69	151.75	840.40
Chest	18	18.67	3.49	21.20	673.46	136.46	772.55
High-Resolution Chest	17	13.01	2.11	12.50	510.65	45.75	522.30
Head	38	35.94	12.25	30.08	679.75	232.58	630.15

TABLE 2. COMPARISON OF DRLS FROM DIFFERENT COUNTRIES, $CTDI_{vol}$ (mGy), DLP (mGy.cm)

Examination	France (2019)		Norway (2004)		Ireland (2010)		Switzerland (2004)		Germany (2010)		Present study	
	$CTDI_{vol}$	DLP	$CTDI_{vol}$	DLP	$CTDI_{vol}$	DLP	$CTDI_{vol}$	DLP	$CTDI_{vol}$	DLP	$CTDI_{vol}$	DLP
Head	46	850	75	1000	66	990	65	1000	65	1000	30	630
Sinuses	14	250	-	-	16	210	9	350	9	100	23	290
Chest-Abdomen-Pelvis (CAP)	11	750	-	-	12	850	15	1000	-	-	20	1161
Chest	9.5	350	15	400	11	390	10	400	12	400	21	773
Abdomen-Pelvis (AP)	13	625	-	-	12	600	-	-	-	-	18	1140
Lumbar Column	28	725	-	-	19	420	-	-	-	-	28	840
High-Resolution Chest	-	-	35	280	7	280	-	-	-	-	12	522

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53-ANNUAL AND SEASONAL VARIATIONS OF INDOOR RADON CONCENTRATIONS IN ACCRA, GHANA

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Indoor radon concentration in buildings with respect to weather conditions such as rainy and dry seasons, have not yet been evaluated in Ghana. At the present time, Ghana does not have national guidelines specifying the acceptable radon levels in both workplaces and residences but have been using international action levels and limits. Most studies on radon concentration in Ghana have focused few studies on water, soil, residences but no offices and laboratories [1-9]. Two hundred and twenty eight (228) CR-39 detectors were used for indoor radon measurements of which 76 offices, 26 laboratories and 126 residences (bedroom, sittings and kitchen) for annual and seasonal studies from April - September as rainy and October-March as dry seasons respectively. Indoor radon concentrations for the studied buildings were randomly selected within the Accra, Ghana. In each building, a detector was placed either in a bedroom, kitchen, sitting room, office or laboratory, at a height of 1 to 1.5 m above the floor, at a distance greater than 0.5 m from each wall, and at a minimum of 15 cm from any other objects for period of 12 months and 6 months in dry and rainy seasons respectively. Exposed, radon detectors were sent to Centro Regionale di Radioprotezione, Agenzia Regionale per la Protezione dell' Ambiente del Friuli Venezia Giulia (ARPA FVG)\, Udine, Italy for analysis. The detectors were etched and latent tracks formed were counted in 144 fields using an optical microscope of 40 × magnification objective lens. The tracks density left on track films were then used to evaluate the indoor radon concentration. Normality associated with the result was determined. The distribution associated with the radon gas exposure in the buildings were done using statistical analysis such as cumulative frequency distribution, normalizing Q-Q plots, Kolmogorov-Smirnov and Shapiro-Wilk statistical test.

The analysis for both workplaces and residences showed that indoor radon gas exposed are not normally distributed. The strong positive correlation between the two seasons occurred at 95 % confidence level with 2 tailed. The rainy season recorded highest coefficient variation of $r^2 = 0.98$ with the dry season recorded r^2=0.97. Statistical analysis of median (39.3), AM (103.4), GM (57.9) and GSD (3.2) for rainy season were greater than that of the dry season of median (26.9), AM (88.2), GM (49.2) and GSD (2.8) respectively. Rainy season was found to contain high radon concentrations than the dry season for all the studied locations. Average radon exposure in the offices and the laboratories were found to be 157.6 Bq/m^3 and 167.6 Bq/m^3 higher than the concentrations of the bedroom, kitchen and sitting rooms which were also found to be as 31.3 Bq/m^3, 23.4 Bq/m^3 and 26.8 Bq/m^3 respectively. This can be attributed to the fact that most of the offices and laboratories in the workplace were in dense environment, resulting in poorer ventilation than in most residential locations that had access to air flow during the daytime through open doors and windows. In general, workplace had radon concentration far greater than residences.

The studied result ranged 13.6 - 533.7 Bq/m^3 of which 25.5 % were found to be greater than levels proposed by WHO, BSS, EC, ICRP [10-12], indicating that not all the buildings in the studied area is negligible. The studied also compared with similar studies done in some areas within the country. The result from the studies will be very important in formulating guidelines for radon exposure map and strategy for the control of radon exposure in buildings within greater Accra region of Ghana.

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54-ETHICS AND ETHICAL TRAINING SCENARIOS IN RADIOLOGICAL PROTECTION FOR MEDICAL DIAGNOSIS AND TREATMENT

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ABSTRACT

The International Commission on Radiological Protection (ICRP) set up a task group (TG 109) to advise medical professionals, patients, families, carers, the public and authorities about the ethical aspects of radiological protection of patients in the diagnostic and therapeutic use of radiation [1]. Occupational exposures and research-related exposures are not in the TG 109 scope. The task group is comprised of ICRP members from both committees 3 and 4 as well as a WHO patient advocate, and experts in ethics. The starting point of TG 109 is ICRP Publication 138, which identifies four core values (beneficence/non-maleficence, prudence, justice and dignity) and three procedural values (accountability, transparency, and inclusiveness) associated with the system of radiological protection. One important goal of this TG is to explain these radiation protection ethical values in the context of diagnostic radiology and radiation therapy and put them into perspective with the principles and values of biomedical ethics (typically: beneficence/non-maleficence, justice and autonomy).

The patient's volition is an essential part of the acceptance of risk in planned medical exposures. This comes directly from the values of dignity and autonomy and is deeply associated with the principle of justification in radiological protection [2]. The decision of accepting a risk involves the benefit/risk assessment of dignity and prudence, and usually based on the application of the beneficence/non maleficence values, which require careful assessment of risks and benefits. In many cases this is over-simplifying or impracticable, and other core and procedural ethical values need to be taken into account. Depending on the situation, the values of solidarity or common good, sustainability or the good of future generations, as well as honesty and empathy could also be useful in this process. Honesty toward the patient and family in disclosing what is known and what is not known for example; and empathy is valuable in gaining an understanding of what is important for the patient in making a decision when evidence is lacking. Compliance with dignity and autonomy also means that when the evidence base is not conclusive, the uncertainties involved have to be disclosed, both to allow the patient to make a good decision and to give real informed consent. The 'right to accept the risk voluntarily' and 'an equal right to refuse to accept' such a risk, was already mentioned in ICRP Publication 62 [2]. Together with the concept of the right to know, informed consent was clearly established in ICRP Publication 84 on pregnancy and medical radiation [3].

Once the theoretical framework has been presented, the approach taken by TG 109 takes a practical and pragmatic path. An evaluation method is proposed to analyse a specific situation, to develop criteria or for education and training purposes. A wide range of clinical diagnostic radiology and radiation oncology situations (e.g. pregnancy, elderly, paediatric, end of life) are considered in two steps: first within a realistic scenario on which the evaluation method is applied; and the second within a more general context. Scenarios are presented and discussed, with attention to specific patient circumstances, and on how and which reflections on ethical values

can be of help in the decision-making process. The table below provides an example of how a student might learn reflect on a situation from an ethical perspective. While there exists minimal training guidance for radiation protection [4], and many other guidelines for justification and optimisation, there is a lack of guidance for what radiation protection workers should do when faced with ethical dilemmas. The TG 109 is working to fill this gap and welcomes feedback on its work to date.

TABLE 1. EXAMPLE OF SCENARIO: ELDERLY EXPOSURE

During his annual health check, Mr Michael, 66 years old, is given an abdominal ultrasound scan for liver and gallbladder analysis. His doctor prescribes him a CT scan with contrast agent due to the suspicion of a potential kidney problem. The patient is not given any specific information about the risk and benefit of the CT examination, which is considered as a routine examination to verify the status of his kidney. The CT clearly confirms the presence of an early stage tumour, in the right kidney. This is followed, within three days, by an interview with the urologic surgeon, who carefully and completely explains the tumour context and suggests that a tumour removal surgery be performed as soon as possible. Thanks to his private insurance, Mr Michael can skip the typical two-month waiting list, and his surgery is performed within a week after the meeting with the surgeon. After the surgery, the patient is informed that the tumour was small and well located. It is therefore decided not to carry out radiotherapy or other radiological treatments. However, in order to follow the local guidelines, a follow-up CT scan is prescribed every 4 months, for 2 years.

After his release from the hospital, Mr Michael wants to understand more about his health situation and starts to surf the web. He cannot find any information about his follow-up CTs on the hospital website. However, after consulting the website of the national society of radiology of his country, he is especially concerned about the follow-up CT scans because they are carried out with ionising radiation and that in general, ionising radiation is associated with possible risk to his health, such as the induction of tumours. At this point Mr Michael becomes anxious, realizing that he had just been successfully treated for a tumour and that, in the recommendations that are indicated for the next 2 years, he will be exposed to the radiation that could potentially cause another cancer.

The table below shows the proposed (paired) values for radiation protection in medicine to be applied in all scenarios, graded in terms of first compliance and then non-compliance for this patient's needs. The TG 109 uses one or two smiley faces or sad faces, or a dash line as not applicable, in the example below. Following completion of each table, there is an explanation for how each value may be evaluated based on the scenario.

	beneficence/ non-maleficence	prudence/ precaution	justice	dignity/ autonomy	transparency/ accountability	inclusivity/ empathy
Compliance	☺☺	☺☺	☺☺	☺	-	-
Noncompliance	☹	-	☹	☹☹	☹	☹☹

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55- RISK MITIGATION FOR THERAPEUTIC NUCLEAR MEDICINE PATIENTS UNEXPECTEDLY RETURNING FOR MEDICAL CARE

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Therapeutic Iodine-131 administration up to 250 mCi has increasingly been delivered on an outpatient basis in recent years, with radiation exposure to others managed by at-home instructions rather than hospital isolation. This is permitted following a thorough risk evaluation and assessment of patient's ability to comply with directions however it does not account for situations in which the patient has a medical emergency and is unable to communicate their hazard to care professionals. The Ottawa Hospital (TOH) has implemented a simple bracelet system to ensure the risk is made apparent without reliance on the patient.

In December of 2018, a patient at TOH was treated with 200 mCi of I-131 via oral administration on a standard outpatient protocol basis. This is a routine activity in which patients are assessed for medical and self-care capacity and provided with education and documentation describing precautions they must take at home to limit exposure to others, for a duration of approximately five days.

Approximately 24 hours after leaving TOH, the patient experienced a medical emergency. Although the patient explained to paramedics that she had recently received radiation therapy, she was unable to provide the letter of explanation, and it was not understood that she remained radioactive. She was transported to the TOH Emergency Department (ED), where she spent approximately 12 hours in triage, waiting rooms, and a bed, before being admitted to a ward where she remained for three days. On the fourth day, the TOH Nuclear Medicine department attempted to contact the patient at home for a routine follow-up, it was then discovered that the patient had been admitted to a ward and had remained there since.

TOH Radiation Safety staff were deployed to recover the details of the patient's admission, produce exposure estimates for staff and members of the public, including thyroid monitoring where deemed applicable, and direct decontamination activities where it was discovered that the patient had become incontinent. All exposure estimates were below public exposure limits, and the patient's room was cleared for routine use after three days post-decontamination to allow for remaining activity to decay to background levels. Staff were notified of possible exposure. This caused unnecessary concern to pregnant personnel, although it was later determined that overall exposures were below levels where staff safety concerns would be warranted.

In response to this event, TOH sought to develop a system that would ensure that the hazards associated with an I-131 outpatient who either a) had a change in health immediately prior to or following treatment that led them to be less capable of following RP instructions (i.e. a medical emergency), or b) was erroneously assessed as being capable to follow RP instructions, would be evident to any individual providing them with medical or first response care.

TOH launched a task force to evaluate and implement corrective and preventative actions, in accordance with TOH's Just Culture [1] to address situations such as these.

TOH has since implemented an alert system with both electronic and physical means to indicate that a patient may require special precautions during the five days following their I-131 administration. This is accomplished via:

(a) An electronic flag in TOH's admitting/administration system which generates a message if that patient is admitted to any TOH facility during this 5-day period. The message indicates that the patient has recently received an I-131 treatment, that precautions need to be taken, and that the nuclear medicine on-call physician must be called.
(b) Following administration, a yellow bracelet is put on the patient's wrist, which indicates the same information as the electronic flag. Existing TOH protocol has the patient return for a follow-up appoint after five days, so the removal of the bracelet has now been made a standard part of this follow-up procedure.

To support this practice, TOH disseminated a communication to all staff in emergency departments, housekeeping, nursing professional practice, and logistical staff. Awareness of both the electronic flag and bracelet is to be added into the standard radiation awareness training required by all employees upon hiring and at refresher frequencies for certain staff.

TOH also disseminated this information among regional first responders through its Emergency Management team and Hospital Emergency Preparedness Committee of Ottawa to ensure that paramedics and other first responders are aware of this practice and understand that this does not restrict them from providing life saving care using routine precautions.

This practice is expected to help protect TOH and regional workers and facilitators who may be required to provide care for patients who have undergone nuclear medicine procedures of sufficient activity to require handing and/or isolation requirements. Regional implementation will be pursued to improve the compatibility of electronic alerts, awareness of using the bracelet as a hazard indicator, and seek ubiquity among other hospitals offering these treatments. In the absence of regionally coordinated health information systems, we recommend that hospitals who deliver therapeutic nuclear medicine administrations on an outpatient basis implement a similar system with similar regional integration in order to protect emergency workers and the public and maintain compliance with regulatory obligations. This practice is being considered in other health centres in the region and now throughout Canada.

ACKNOWLEDGEMENTS

Special thanks to Ines Lonz, Sheila Dowell, and Nadia Zaid from The Ottawa Hospital for their roles in the initial event investigation and solution development.

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56-EFFECTS OF HIGH-DOSE RADIATION ON LUMINESCENCE PROPERTIES OF COSMIC RAYS' DETECTORS BASED ON $(Y_{0.7}GD_{0.3})_2O_3$: EU^{3+} NANOPARTICLES

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Cosmic rays consist of high-energy particles: hydrogen and helium nuclei, electrons, and specific particles – muons. They originate from the sun, from outside of the solar system, and from distant galaxies. Every second, an average of 34 particles of cosmic radiation passes through our bodies. That is a total of 3 million particles a day and can cause DNA mutations in the body. Aircrew and space crew are much more exposed to this radiation. For example, astronauts on Apollo mission received on average 1.2 mSv/day. Most radiation workers on Earth receive less than 10 mSv per year. Cosmic rays' detectors are based on the fundamental principle: the transfer of part or all the energy to the detector mass where it is converted into some other form more accessible to human "perception". The form in which the converted energy appears to depend on the detector and its design. Scintillation detectors are often used to detect cosmic rays. A scintillator is a material that exhibits scintillation when excited by ionizing radiation. A scintillator, when struck by an incoming particle, absorb its energy, and re-emit the absorbed energy in the form of light.

One of the scintillators used for cosmic rays' detection is $(Y_{1-x}Gd_x)_2O_3$ oxide doped with Eu^{3+} ions [1]. As these samples are used as a sensor for cosmic rays' detection it is rather important to better understand the actual conditions that will happen during the use of the material. Although they detect relatively small doses of radiation, these materials absorb significant doses of radiation over their lifetime. In the paper, the effects of high-dose radiation on luminescence properties of $(Y_{0.7}Gd_{0.3})_2O_3$: Eu^{3+} powders with different particle sizes are analysed [2]. Powders were prepared by polymer complex solution route, followed by annealing [3]. Nanoparticles were obtained with an average size according to annealing conditions. Samples were exposed to gamma-irradiation (doses up to 8 MGy) on an industrial gamma-irradiation facility. The effect of irradiation on different nanoparticle sizes is followed by XRD, SEM, TEM and monitoring their luminescence properties [2]. XRD analysis show pure crystal phase for all samples, without additional peaks from impurities, as can be seen from Fig. 1

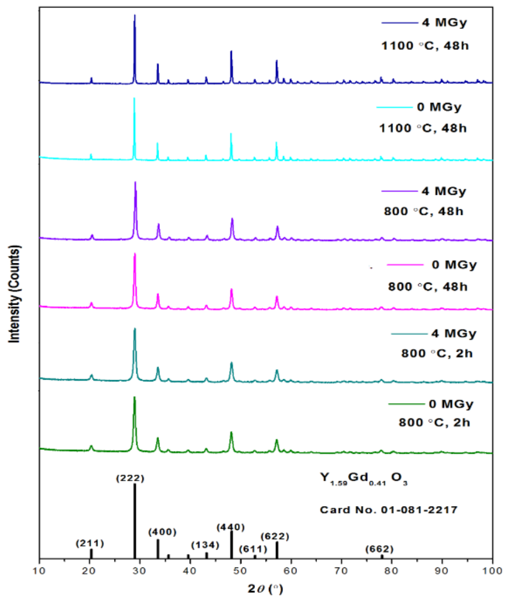

FIG. 4. XRD pattern of $(Y_{0.7}Gd_{0.3})_2O_3$: 5at%Eu calcined on 800ºC for 2h, calcined 800°C for 48h and calcinated on 1100°C for 48h; non-irradiated and exposed to gamma radiation dose of 4 MGy.

Morphology of particles (Y, Gd)$_2$O$_3$: Eu^{3+} of different sizes as well as changes in morphology due to exposure to different doses of ionizing radiation were examined using transmission and scanning electron microscopy. Based on these analyzes, no influence on the morphology of (Y, Gd)$_2$O$_3$: Eu^{3+} powders can be observed when exposed to ionizing radiation up to 4 MGy, regardless of the particle size.

To examine the impact of radiation on the luminescent properties of Eu^{3+} doped (YGd)$_2$O$_3$ powders, photoluminescence under UV excitation, and radioluminescence under X-ray excitation were measured. Also, the quantum efficiency of the samples was calculated. From Fig. 2, one can see that UV and X-ray excitation leads to different luminescence properties. These observations can be explained by the fact that X-rays penetrate deeper into the sample, while UV rays are mainly absorbed over the sample surface [4].

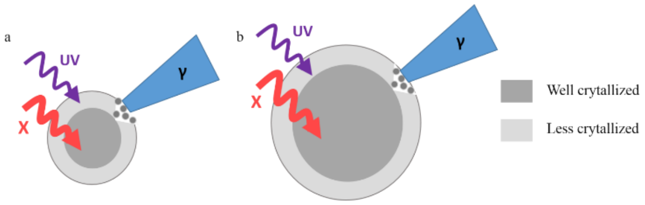

FIG. 2. Schematic representation of particles under several irradiation: (a) small nanoparticles (b) bigger particles.

After a detailed analysis of these materials used as Cosmic rays' detectors, no change was observed in structure, morphology, and steady-state emission. On the opposite, after irradiation, excited-state lifetimes and quantum efficiency values are particle size-dependent.

The obtained results of structural, morphological and optical analyzes showed that the examined powders doped with ions of europium show potentials for wider application in industrial production of scintillation radiation detectors, taking into consideration the ease of preparation, good yields of materials and high structural and morphological stability to the impact of high-energy radiation. The optical characteristics of the materials are changed when the materials are exposed to high doses of ionizing radiation, but not to the extent that would limit their use in scintillation radiation detectors.

ACKNOWLEDGEMENTS

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57-A MONTE CARLO STUDY ON THE SHIELDING MATERIALS FOR IR-192 GAMMA SOURCE

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Radiation therapy requires the use of particle accelerators or high dose rate (HDR) gamma sources such as Ir-192 gamma source for treating tumours using ionising radiation. Photon dosimetry is indispensable in designing irradiation facilities shielding as radiation dose must be controlled properly to prevent hazards caused by exposure to radiation. Clinical (patient monitoring) and occupational dosimetry (staff monitoring) are the cornerstones of any dose optimisation effort [1]. Therefore radiation shielding is important to reduce the number of transmitted photons to protect people outside of the irradiation area [2]. As an example, the brachytherapy room is usually made of normal concrete with extremely thick walls to reduce the primary and scattered radiation since the gamma sources are unshielded in all directions.

While considering the photon attenuation coefficient (μ) and its related parameters for photons shielding to protect people outside of the treatment area (as shown in Fig. 1), it is necessary to account for its reflected photons energy and dose from the concrete walls or floor [2]. The calculation of how radiation incident on a surface is reemitted through the surface of concrete toward some point of interest (patient location) is a frequently encountered problem in radiation shielding [3]. A transport technique by using Monte Carlo (EGS5 code [4]) will be used for estimation of reflected doses component. Due to the source condition used, the incident photon fluence is significantly altered while passing through the shielding material and contributing to the scattered photons. The study found that the walls closest to the gamma source would likely be the largest contributors to the scattered radiation at the patient location as shown in Fig. 2. From a dose standpoint, there was a need to take measures to reduce the reflection dose to the patient that was already receiving the prescribed dose from Ir-192. It is necessary to reduce the reflected photons in order to reduce the effective dose to the patient.

The study would propose an approach with respect to cost and convenience to reduce the reflected dose component. The presentation demonstrated two approaches in order trying to reduce it; by adding some lead to the inner walls to reduce reflection and by using a higher density concrete in the walls that incorporates higher atomic number materials such as barium in barytes concrete. The study found that the latter approach is not as effective as adding lead sheets to the inner surface. It is due to photons scattered near the inner surface may not encounter the higher atomic number material before reemerging. Detail results such as shielding materials, walls design, the transmitted and reflected photon energy spectra and dose rates are going to be discussed in the presentation. Such study is beneficial to bring scientific support to improve our regulations of radiation protection for patients and medical personnel.

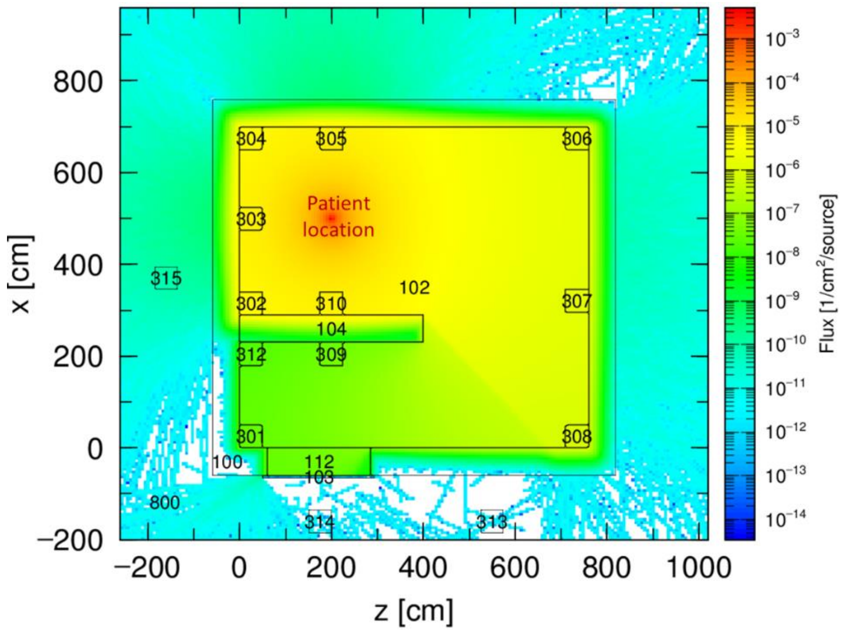

FIG.1. The top view of brachytherapy facility shielding at Advanced Medical and Dental Institute (AMDI), Universiti Sains Malaysia (USM), Malaysia. The number indicated the scoring region of the calculations.

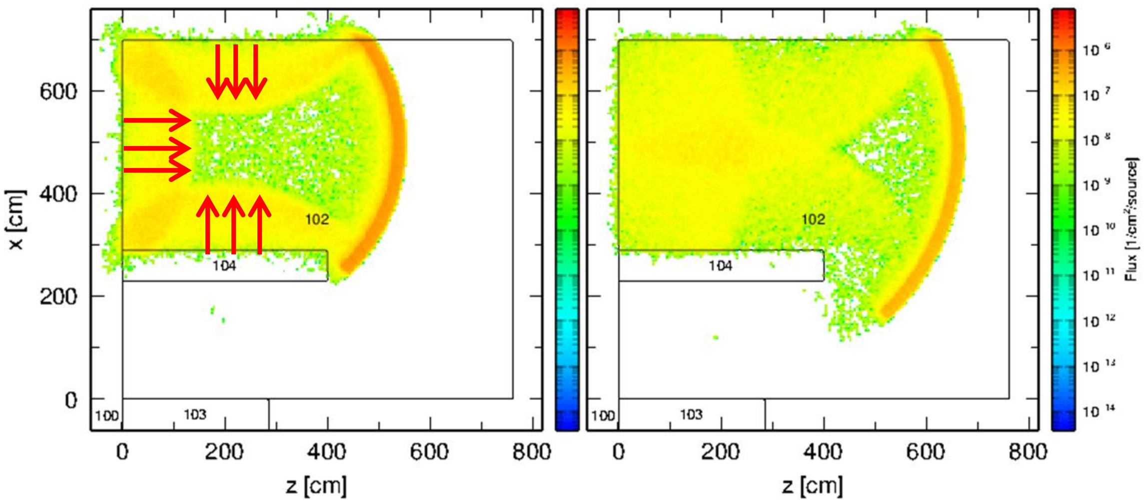

FIG. 2. The arrows in the left figure indicated the photons trajectory that reflected from the current concrete walls. The walls closest to the source would likely be the largest contributors to the scattered radiation at the patient location as shown in the right figure.

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59-IMPORT OF RADIOACTIVE SOURCE CO-60 TO SERBIA FROM THE EUROPEAN UNION COUNTRIES: HARMONIZATION OF LEGISLATION AND STRENGTHENING INTERNATIONAL COOPERATION

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Import and transport of radioactive materials are subject to numerous legal regulations in Serbia. Regulations for the circulation of radioactive materials in Serbia sometimes do not comply with EU regulations. Serbia is a candidate country for EU membership and is currently in the process of harmonizing all legal regulations with the EU legislation.

On August 21, 2018, a radioactive source of Co-60 with the total activity of 1858,3 TBq (50 kCi) was imported to be used in Radiation Unit for Industrial Sterilization of the Vinca Institute of Nuclear Sciences. Radioactive material was purchased from the Hungarian company „Izotop“. Figure 1 shows a container with a radioactive source delivered to the Vinca Institute.

Figure 1. Container with a radioactive source

During this trade, it was necessary to harmonize numerous legal requirements of Serbia and Hungary. Serbian Radiation and Nuclear Safety and Security Directorate (earlier: Serbian Radiation Protection and Nuclear Safety Agency) is a regulatory body that issues licenses for performing radiation and nuclear activities in Serbia. Before starting the process of ordering radioactive materials, the manufacturer was requested to provide detailed information on the type of radioactive source, activities of individual sources, total activity, dimensions, and to send the Emergency response and action plan. Table 1 shows the source type and dimensions. After reviewing the documentation, the Directorate issues approval for the start of procurement of radioactive material.

Table 1. Source type and dimensions

Type Code	***Inactive Dimensions***		***Active Dimensions***		***Maximum Equivalent Activity***		***ISO Classification***
	D [mm]	L [mm]	d [mm]	h [mm]	TBq	Ci	
CoS-43 HH	11	451	7	437	300	8110	E 64434

After harmonization of the documentation, the radioactive source was imported and loaded into the storage pool (Figure 2).

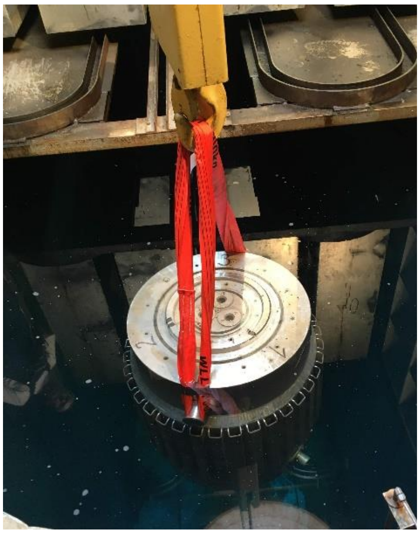

Figure 2. Loading a radioactive source in a storage pool

The paper describes in detail all procedures for importing radioactive material into Serbia from the EU countries. Similarities and differences in legal frameworks between Serbia [1] and the EU countries are described. It has been established that harmonization of legislation and strengthening of international cooperation is a necessary condition for the enhancement of radiation and nuclear safety and security in Serbia and the region.

ACKNOWLEDGEMENTS

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60-MODELLING OF ATMOSPHERIC DISPERSION AND RADIATION DOSE FOR A HYPOTHETICAL ACCIDENTIN RADIOISOTOPE PRODUCTION FACILITY

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Abstract

Atmospheric dispersion modelling and radiological safety analysis is perfomed for public outside radioisotope production facility (RPF) in case of hypothetical radioactive Iodine spilling and leakage from hot cell. Potential human error is expected and column that occupies Iodine may be broken causing it to spill on the hot cell floor. Ventilation exhaust system is dedicated to extract dispersed material through dedicated filters before gases expelled outside the facility. Two scenarios are performed in this paper, the first one is prediction the dispersion with good filtration from extract ventilation system, while the other with loss efficiency of filtration components. The spilled radioiodine is the source term, and the HotSpot 3.1, Health Physics code was designed by LLNL [1] is used to provide required calculation tool to assess and evaluate emergency situation including radioactive nuclides release to illustrate the transport modelling which is then applied to estimate the total effective dose equivalent (TEDE) in different Pasquill stability classes, and how it would be transferred to human body depending on downwind distance and radionuclide activity. The adopted methodology uses dominant site-general meteorological information and theories of dispersion models to study the effect of hypothetical dispersion and release to the environment from the selected radionuclide and assess how such dispersion may have radiological worse impact on public.

TABLE 1. SOURCE TERM IN NORMAL OPERATION

Isotope	Activity (Ci)	Isotope	Activity
Kr-83m	3.93E-02	Xe-133	2.66E+01
Kr-85m	2.51E-03	Xe-135m	5.40E-01
Kr-85	2.77E-01	I135	2.19E-02
I131	5.66E-02	Xe-135	6.75E+00
I132	1.23E-01	Xe-131m	2.70E-02
I133	1.11E-01		

RESULTS

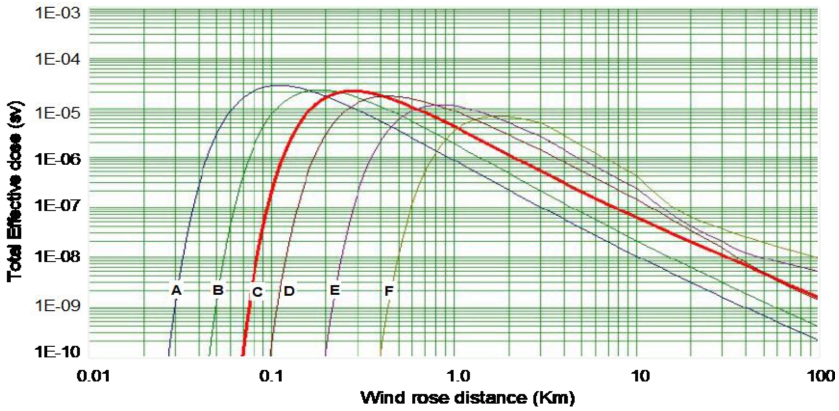

FIG. 1 TEDE variation from the release point

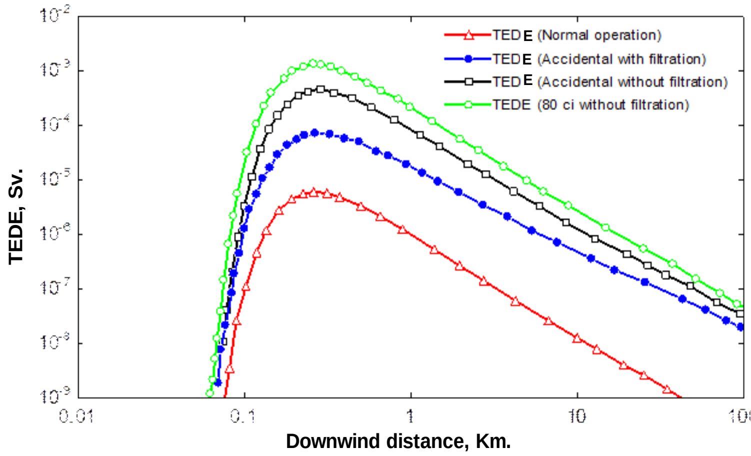

FIG. 2 TEDE of different RPF statues for class B

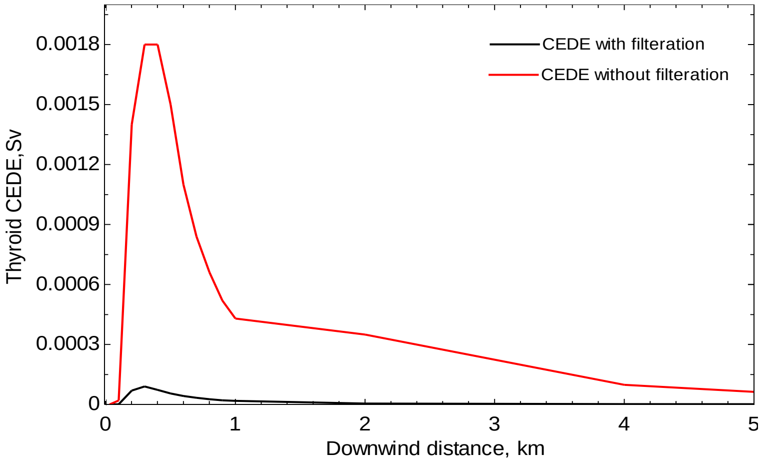

FIG. 3 Target Thyroid committed equivalent dose as a function of downwind location

This study introduced different statues of hypothetical radiological release to atmosphere due to expected human error considering atmosphere function of a Pasquill class and wind speed. The evaluation of filtration is studied and compared with no filtration case to show the annual dose for public. The radioactive material for evaluation was selected based on data from the process sector. Gaussian plume Hotspot code is used to model the release in all stability classes and available metrological data. The predicted results show that:

1. In case of normal operation of filtration system the maximum public total effective dose offsite is far from dose limit for all metrological stability classes.
2. In case of loss of filtration system the maximum public total effective dose is higher but still less than the limit.
3. Intension should be considered if the facility decides to increase the production from Iodine to double activity as the TEDE exceeds the limit to be 1.2 msv.
4. The committed equivalent doses of thyroid organs are predicted for the two cases and the results show that they are under permissible committed dose limits.

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61-DEVELOPMENT OF OPERATIONAL RADIATION PROTECTION INFRASTRUCTURE: GHANA'S EXPERIENCE

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This paper gives perspective on the graded approach in the development of effective radiation protection infrastructure in Ghana since 1993. Effective radiation protection infrastructure development in Ghana started with the promulgation of the Radiation Protection Instrument, LI 1559 in 1993 which established the Radiation Protection Board (RPB) under the Atomic Energy Commission by PNDC Law 308 in 1993 [1, 2]. Initially the RPB functioned as both regulatory authority and service provider which amounted to conflict of interest. In the year 2000, the Radiation Protection Institute (RPI) was established to provide technical services to the RPB in line with implementation of the LI 1559 [3, 4]. In the year 2015 a new Act, the Nuclear Regulatory Authority Act 895, 2015 was promulgated. The Act established the Nuclear Regulatory Authority (NRA), Ghana as the statutory nuclear regulatory body in Ghana, thus separating the regulatory functions from the operational functions of the Ghana Atomic Energy Commission [5]. The functions of the Radiation Protection Institute among others are to undertake research and provide technical services. With the establishment of the NRA, the RPI of the GAEC also assumed the position of a Technical Support Organization (TSO) for the successful implementation of the Act 895 of 2015. The technical services provided by the RPI include:

- Personal monitoring of individual workers exposure to radiation and Workplace Radiation Monitoring
- Radiation instrument calibration and performance testing
- The analysis of food and environmental materials, including; air, soil, water, etc
- Safe and secure management of radioactive Waste
- Radiation Protection and Safety Training programs for Radiation Protection Officers and Qualified Operators.

The Ghana Atomic Energy Commission (GAEC) through the Radiation Protection Institute (RPI) has acquire and operating a Radosys RadoMeter 2000 system for measurement of radon concentration levels. The acquisition of the new radon equipment complements the effort of RPI to cover all areas in Radiation Protection.

The School of Nuclear and Allied Sciences (SNAS) was established in 2006 by Ghana Atomic Energy Commission in collaboration with the University of Ghana and the International Atomic Energy Agency (IAEA) to develop and maintain human resource in nuclear and allied sciences. The Department of Nuclear Safety and Security of the SNAS is responsible for the development of human resource capacity in Radiation Protection. The Department offers MPhil and PhD Degree programmes in Radiation Protection. The School (SNAS) in October 2011 was recognized and endorsed by the IAEA as a Regional Designated Centre (RDC) of Excellence in Radiation Protection for education and training in Radiation Protection. The school (SNAS) in collaboration with the RPI, GAEC has been hosting of Post Graduate Education Certificate (PGEC) in Radiation Protection since 2011.

The objective of this paper is to describe Ghana's Experience in the Development of Operational, Radiation Protection Infrastructure, Challenges and Way Forward. The paper will also highlight the relevant provisions of nuclear regulation as enshrined in the NRA Act 895, Regulations, Radiation Protection Infrastructure and Services in Ghana.

ACKNOWLEDGEMENTS

The authors are grateful to the RPI of GAEC for their support.

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62-CREATING AN EFFECTIVE ASSESSMENT OF NUCLEAR SAFETY CULTURE

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The objective of a Safety Culture Policy Statement (SCPS) is to specify the mission and vision of an organization with respect to the safety culture in the design, construction and operation of nuclear facilities. The SCPS exposes the structure to manage the development, implementation, sustainability, growth and practices of the safety culture, and from it are derived instructions, procedures and forms.

An organization should consider the evaluation of safety culture as a proactive way of working in order to improve the effectiveness of management. Therefore, evaluations are periodically performed to analyse the state of safety culture management. The presentation introduces elements required for assessment to ensure an effective safety culture environment. Fig. 1 shows the elements for effective safety culture assessment which are described below.

SELF-ASSESSMENT

The planned self-evaluation process for the directorate, managers and individuals should be implemented, with the aim of analyzing the execution of the work carried out and the continuous improvement of the safety culture. The processes should be performed systematically taking into account regular periods of evaluation time, which show important achievements in the safety aspects. The results should be reviewed and evaluated periodically so that they can be duly communicated to affected people or groups with the actions to take.

The nuclear organization should ensure that safety can be monitored and evaluated enduringly, in which the perception of all employees regarding the safety culture is analyzed and thus establish an order of priorities on the actions to be carried out in the framework to improve the efficiency of the management of the safety culture. Indicators are available to determine the effectiveness of the self-assessment process such as the following:

(c) General criticisms.
(d) Comparison of results with other independent internal or external self-assessments.
(e) Feedback on benefits with project managers.

The organization should adhere to the Principles of Conduct (PoC), of international acceptance that allows the implementation of best practices in nuclear safety and the development of task lists for self-assessments.

Several self-assessment techniques should be used, such as observations of operations, inspections of techniques or equipment, questionnaires and process evaluation. They will be developed in an organized and planned way, in teams or individually. They can also be performed as a result of situations that need to be reviewed or developed (events or new regulatory requirements).

Guidance must be available to provide a framework and criteria necessary to audit the organization safety culture so that it can be systematically evaluated. A system for internal independent evaluation of the performance of the safety culture should be performed, ensuring that its results are sustainable over time, with internal audits and personnel surveys being two of the tools used for this purpose. Independent internal and external evaluations of the important aspects related to safety are carried out in order to contrast its operation with international best practices.

SAFETY CULTURE INDICATORS

The indicators considered by an organization should allow measuring the effectiveness of the safety culture. They are recorded, evaluated and allowed to identify in time, if the developed attributes do not decline, and thus take appropriate measures. A usual practice is to evaluate the result of analysis of lessons learned, which is used to identify improvements in safety performance and learn from success in previous projects.

The indicators come from the internal and periodic analysis of results of self-assessments, interviews and audits, supported by external reviews. The indicators will be analyzed and classified according to origin, repetition frequency, expiration of deadlines, regulatory compliance, relationship with human error, etc.

MEASUREMENT AND ANALYSIS

From the directorate of the organization, actions that are necessary to promote measurement activities, lessons learned and indicators of measurement results and monitoring of the safety culture must be promoted.

The effectiveness of the safety culture and the impression of how employees follow it within the organization is an important tool for monitoring observation by management. A safety culture survey should be sent to all members of project management as a tool to improve and optimize the management carried out in the aforementioned management. This process is continuously evaluated to optimize it in subsequent iterations.

The measurement of the effectiveness of the safety culture encompasses all the organization employees has the objective of fostering a culture of safety, and above all, achieving an active participation of each employee.

COMMUNICATION OF RESULTS

The results must be communicated to the management, and then to the other employees, through internal communication, or area meetings, so that each participant receives a feedback. The objective of the analysis of evaluations is to understand trends that could cause dangerous behaviors for safety. The results of these evaluations influence the decision making by the directorate and for the development of future areas of interest.

Fig. 1. Elements for effective assessment of nuclear safety culture

ACKNOWLEDGEMENTS

The author would like to thank King Abdulaziz City for Science and Technology (KACST) and King Abdullah City for Atomic and Renewable Energy (K.A.CARE) for supporting the participation with the current presentation.

63-BARITE PROPORTION EFFECT ON NEUTRON SHIELDING BY FLUKA SIMULATION CODE

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Abstract

Concrete is the most generally used as neutron shield material as it is inexpensive and adjustable for any construction design. Radiation shielding properties of concrete may vary depending on the concrete composites. Different types of special concrete have been developed by the addition of shielding material. The neutron shielding properties of concrete containing different proportions of barite as an aggregate have been investigated. The macroscopic neutron removal cross sections have been calculated. The calculation of neutron removal cross section has been done by using FLUKA simulation code. The transmission of neutrons has been obtained as a function of thickness of concrete for all concrete types. According to the simulations results, it is clear that increasing barite proportion in the concrete decreased the neutron removal cross section.

Keywords: Neutron shielding, FLUKA simulation code, macroscopic neutron removal cross sections

1. INTRODUCTION

Concrete is a composite construction material composed primarily of aggregate, cement and water [1]. FLUKA is a Monte Carlo simulation package for a variety of models of particle transport and interaction with matter. [2]. The neutron is a neutral particle with a mass approximately equal to proton mass [4].

2. MATERIAL AND METHOD

Three different types of concrete samples (BC 0), (BC 50) and (BC 100) produced with barite proportions of 0%, 50% and 100% respectively were used.

3. RESULTS AND DISCUSSION

The neutron removal cross-section for three types of concrete have been calculated.

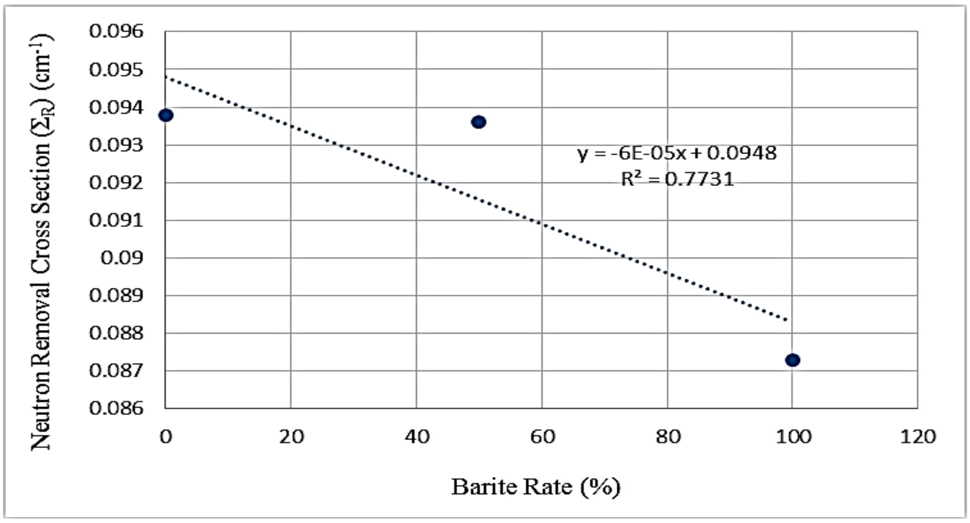

Fig. 1. Variation of neutron removal cross section versus barite rate.

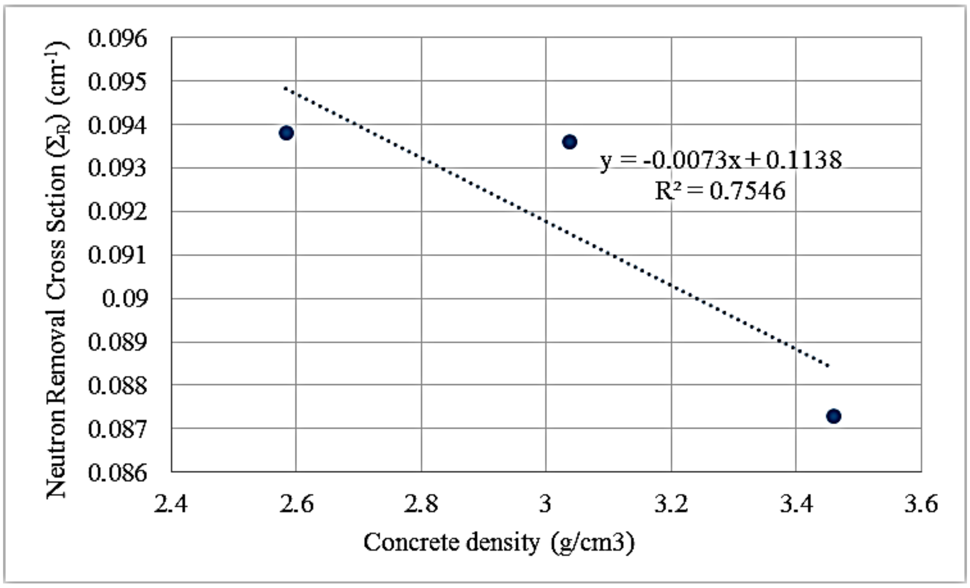

Fig. 2. Variation *of neutron* removal cross section *versus concrete density*.

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64-IMPLEMENTATION OF RADIATION SAFETY MEASURES AND NUCLEAR SECURITY STANDARDS FOR EMERGENCY PREPAREDNESS AND RESPONSE FOR HIGH DOSE RATE BRACHYTHERAPY SEALED RADIOACTIVE SOURCE USED IN A MEDICAL RADIATION FACILITY

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The radioactive material is an integral tool in Global healthcare system, manufacturing, research and quality control industries. The aim of present study is to provide guidance on basic concepts and elements for radiation safety measures for a high dose rate (HDR) brachytherapy sealed radioactive source implemented to enhance nuclear security culture in a medical radiation facility. These sources are Category 2 (High risk-very dangerous) radioactive source and Security Level B applied to cover the range of security measures needed for the associated risk with Remote After-Loading (RAL) HDR brachytherapy sources.

Security and protection systems to deter, detection, delay and response applied to attempted any unauthorized removal of source. These security measures were implemented based on a risk informed graded approach applies to minimize the likelihood of loss of control, theft, sabotage or other malicious act. The security measures installed incorporating the basic concepts of defense in depth, fail safe mechanism and redundancy principles. The facility was situated in lower Basement which is not easily accessible to general public and this area is accessible for authorized personnel only. The control access to the source was ensured efficiently by identification and verification by magnetic card reader and finger print impression in the biometric machine at controlled entry point. The keys kept under control with Radiation Protection Officer (RPO)/ Licensee. RAL operating and data storage devices was "protected" via passwords or encryption, and they were only accessible to authorized users. The equipment and premises were secured by surveillance cameras and access-controlled doors. The facility is continuously monitored by the security personnel and surveillance cameras for detection, assessment, and communication with response personnel in case of security event. Unauthorized personnel and visitors are under escort in this controlled secured area. The detection of an attempted intrusion of unauthorized person performed using visual observation and electronic surveillance in the facility. There is a continuous video surveillance using Closed Circuit Television (CCTV) cameras and authorized personnel/ security guards for detection of any unauthorized entry in the secured area. The CCTV cameras were in operation during power off because they connected to uninterrupted power supply (UPS) and automatic generator of institute which automatically starts when a fluctuation in power levels is detected, in order to provide redundancy. CCTVs has a backup of one months and monitored from a remote location by operator/ response personnel. There is a provision that these CCTVs online footage can be monitored in cellular phones of authorized personnel as well with valid password protected authentication for immediate assessment for detection. The gamma zone monitors with auto reset type were installed at entry point of brachytherapy facility, as well as in RAL brachytherapy machine to provide in-build radiation safety of equipment. The last man out (LMO) switch was installed in the facility and functioning on a separate circuit in combination with door interlock (which serves as contactor) and timer to enable source transitions from machine in safe manner during treatment of patients (i.e. planned exposures). The authorized personnel were familiar with the responsibilities and functions, line of authority to be contacted at occurrence of an incidence. The key contact details of personnel including office and personal numbers displayed at appropriate places such as control panel of the facility in case of emergency. They are also provided with most direct and alternate lines of communication. All the authorized personnel provided Closed Universal Group (CUG) postpaid SIM and cellular phones for reliable, secure, rapid communication and response in the shortest possible time. The landline phone network was also provided at appropriate places within the radiation facility for fast effective communication considering the graded approach. There was multiple barrier system installed in order to provide delay function to prevent illegal act and enable adequate reaction of the security personnel. The security system of two layers of barriers includes walls, door which were resistant to physical attack using handheld tools and provide sufficient delay to enable response personnel to interdict. The door made of solid core wood fitted with non-removable pinned hinges. It has an automatic door closer system. Sufficient arrangement made for maintaining security regime in case of malfunctioning security equipment, regular maintenance, repair or testing. The machine has two physically separate solid barriers a lock and machine with padlock to secure the source against unauthorized removal when mobile machine unit is not monitored. The physical verification of source performed periodically using measurement of radiation and temper detection by visual inspection to ensure condition of source to detect potential loss of source. The contingency plan with security objectives for incidents and procedures for action were prepared in order to implement response function to fulfil goal of adequate security of source. The overall security plan for the source security were prepared for protection and response to theft, emergency during transport, flood, fire, earthquake and other natural calamities. Any incidence reporting and documentation mechanism has been established. The medical facility equipped with adequate protection equipment such as Survey meter, Digital contamination monitor, Gamma zone monitor, Digital radiation monitor, Personnel dosimeter, Pocket dosimeter, long handled forceps, shielding material and Emergency source storage container etc. to deal with minor incidences. Minor incidences are planned to resolved locally, with the help of regional protection agencies if necessary and detailed report sent to national regulatory authority. However, local RPO limits his role as first responder only to major incidence in absence of enough protection equipment to handle a radiation emergency. The national regulatory authority promptly reported upon determination of loss of control of source. The trustworthiness and reliability of personnel prior authorization access to the source ensured through background checks with police authority, identity check, review of previous employments and verification of references. The authorized personnel receive appropriate radiation safety training commensurate with the radiation hazards posed by the source and the precautions to be observed in order to ensure restrictions of their exposures and protection of other by their actions. Radioactive source transport security achieved by minimizing total time duration of transport, limiting number and duration of transport interruptions, Transport Radiation Emergency (TREM) Card documents which includes name, address and contact details of all parties involved in the transport. A valid photographic identity document checked for every person engaged in the transport of radioactive source. The approval authorization for transport in the conveyance/ vehicle checked prior transportation. Placards appropriateness checked on transport container and radiation levels were monitored. A visual security inspection performed of the conveyance prior to the commencement of transport to detect any addition objects on the conveyance that could affect transport. The gate pass is mandatory for taking the source out of the radiation facility and this requires multiple authorizations from different personnel/ agencies. It is a mandatory regulatory practice for users that the no-objection certificate (NOC) has to be obtained from national regulatory authority prior transport for the import and export of radioactive source. All sensitive and confidential information pertaining to source kept under custody of RPO in lock and key. After use, the source related documents destroyed as per protocol.

The emergency response plan prepared and implemented to mitigate the radiological consequences and source recovery measures can be implemented without delay. The emergency response plan documented against emergency scenarios during transport, loss of radioactive source, theft, sabotage, natural calamities/ disasters such as Fire, Flood and Earthquake. When theses high activity sources are safely managed and securely protected under control, the risk to occupational staff and the public will be minimized and the benefits will overweigh the associated radiological hazards. This study provided guidance to effectively mitigate the consequences of an emergency for human health and safety, quality of life, property and the environment. Transport of radioactive source is the responsibility of consignors, consignee and carriers under the legislation, supervision and control of regulatory authority.

66-A CONTRIBUTION TO STRENGTH SAFETY CULTURE IN THE RADIONUCLIDE THERAPY WITH INTEGRATED RISK ANALYSIS IN CUBA

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For the management of quality and safety in medicine with ionizing radiation and the continuous improvement of the processes, the performance of the risk analysis is required for the evaluation of safety of the practices in Cuba. There are different methodologies for risk analysis. The state-of-the-art shows risk matrix (RM), the analysis of failure modes and effects (FMEA) and the incident learning system (ILS) have a limited combined use and uncommon and without application in radionuclide therapy (RT).

The objective of this work is to analyse radiological risks in RT in Cuba, by establishing synergy between prospective and reactive methods focused in root causes for patients, workers and public.

To undertake this investigation the following procedure was followed:

(a) Review and adaptation the developed risk matrix generic models [1-2] and taking into account the publication TG-100 of AAPM [3] for each case of study (the first five cases for conventional therapy and the six case for personalized treatment);
(b) Selection of initiating events (IE) and defences (frequency reducers (FR), barriers (B) and consequences reducers (CR)), applicable to the practices that were carried out in Cuba at the moment of the investigation. These practices are radiosynoviorthesis (RSV) and the myelosupressor treatment of Polycythemia Vera (PV) with Phosphorous 32, the treatment of cancer and benign thyroid diseases and personalized treatment of thyroid illness. Determination by case in study of the inherent risk and the residual risk;
(c) Identification of the main stages of process and defences, both due to their percentage participation and their impact on the level of risk when they are eliminated and determination of consequences of incidents behaviour for patients, workers and public.
(d) Main sub-processes and root causes from FMEA are in correspondence with the values of the risk priority number (RPN) and severity (S), but on the basis of equivalence between RM and FMEA with the previously adaptation of FMEA scale for patients, workers and public. Also, a conversion of each defence in a root cause is applied by expert´s criteria and a deployment of these. The used selection criterions are RPN≥100 and ISev≥7 and the 20% of highest value of RPN.
(e) Creation of an ILS with information from articles and published reports, mainly from Australia and the United States. This includes around 30 years of published events and near misses (for the last case it was used an adapted Nyflot´s five level scale [4]), a wide standard list of root causes and adapted SAFRON´s severity scale. Realization of the FMEA-ILS synergies, to determine the most important sub- process by FMEA and the predominant basic causes.
(f) Validation of the risk matrix generic model with RM-ILS, with the better matching between initiating events and records, determination of the most reported root causes in the ILS;
(g) Use of the Cuban code SECURE MR-FMEA version 3.0, developed in the Higher Institute of Advanced Technologies and Sciences (INSTEC) and
(h) Creation as a training tool of an informational compendium made with Macromedia Dreamweaver 8.
(i) Recommendations of actions for strengthening of safety culture in an organization and recommendations to the Cuban regulatory bodies taking into account the ACCIRAD project ENER/D4/160-2011 [5] and adapting these.

The clinical prescription of treatment, preparation and administration of radiopharmaceutical are the stages with the most important contribution to risk. This can be observed in Figure 1. Besides, Table 1 showing the most contributing root causes for five services of RT. Similar behaviour was obtained for personalized treatment with the inclusion of the maintenance and repair of equipment and systems and same main basic causes. This study reflects the necessity of safety culture building and contributes to strengthen it with all staff and leadership.

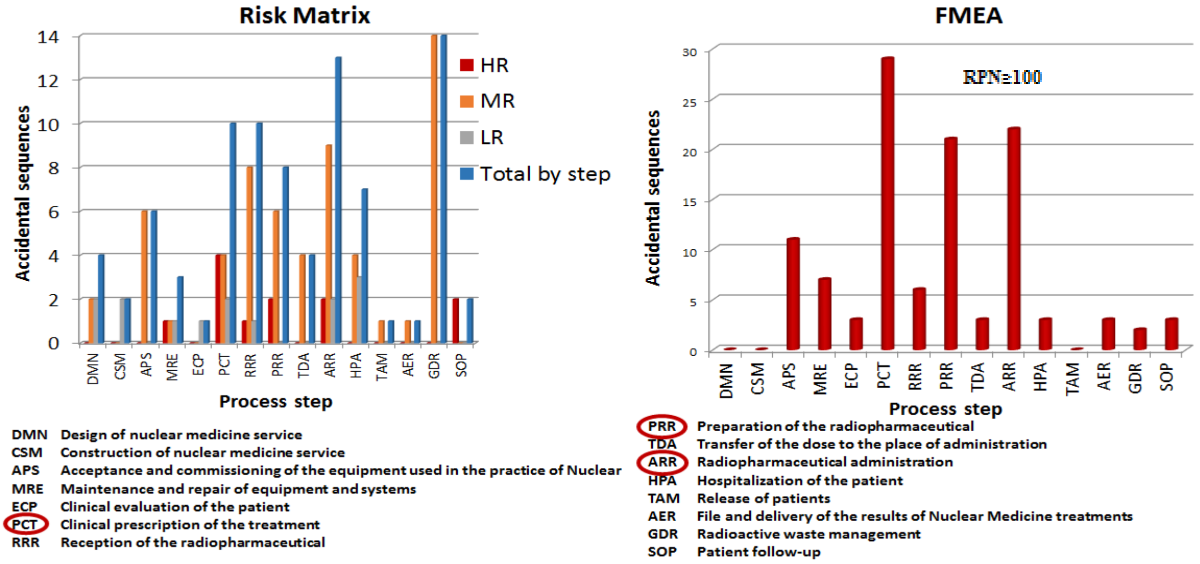

FIG. 5. Number of accidental sequences by process stage with high risk (HR), medium risk (MR) and low risk (LR) and total of these by stage from risk matrix (left side) and accidental sequences from FMEA with RPN≥100 (right side), for studied case No. 5

TABLE 1. ORDER OF IMPORTANCE OF THE MOST CONTRIBUTING ROOT CAUSES FROM FMEA FOR EACH STUDIED CASE IN THE CONVENTIONAL NUCLIDE THERAPY (1-5) AND PERSONALIZED TREATMENT (6)

Root case	Studied case number					
	1	2	3	4	5	6
	Order of importance					
1.3 Practices, protocols, procedures or standards- non-compliance	1	1	1	1	1	1
6.1 Development of skills and knowledge- inadequate training/orientation (lack or inadequacy)	3	2	3	2	2	2
8.4 Worker's perception-Fatigue of staff	2	3	2	3	3	3

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68-OCCUPATIONAL DOSE MONITORING FOR RADIOLOGY STAFF AT INSTITUT KANSER NEGARA: FIVE YEARS EVALUATION

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Introduction

Interventional radiology increased by more than 10% per year on average between 1992 to 2001 [1], and increased 60% for all imaging from 2000 to 2008 [2]. All the medical staffs experienced with ionising radiation represents the major part of occupational dose monitoring attributable to low radiation doses for long term exposure. The main reason to monitored occupational dose is to get the findings on the competency of radiation protection provided. In spite of that, it will be a core input for optimising the medical protocol and compliance with recommended dose limit, below 20 mSv averaged over five consecutive years and within 100 mSv in five years [3]. The aim of this study is to evaluate the annual effective dose of the staff at Radiology Department, Institut Kanser Negara, Putrajaya using optically stimulated luminescent dosimeter. The radiology staff are involved in four types of medical imaging modalities includes computed tomography (CT) scan, fluoroscopy, general radiography and mobile radiography.

Material and methods

The effective dose were measured using an optically stimulated luminescence dosimeter (OSLD) placed at the chest level because the highest radiation exposure is expected in this part of the body. All OSLD were readout using the microStar reader system (Landauer, Inc.). In this study, four quantities were used to analyse occupational doses received by radiology staff from 2014 to 2018: (i) collective dose; (ii) annual average collective dose; (iii) annual effective dose; and (iv) individual cumulative effective dose.

Result

Overall, 351 annual doses records were analysed. The interventional radiologist (IR) received the highest maximum annual effective dose (19.36 mSv). Fig. 1 demonstrate that annual effective dose distribution skewed to the left with 26.21% (N=92) of radiology staff received no measurable doses, while 0.28% received doses more than 10 mSv. Doses received between 0 to 1 mSv is the highest percentage of radiology staff recorded with 58.40% (N=205). Annual effective dose more than 10 mSv (0.28%; N=1) recorded as 19.36 mSv and it was a dose received by IR in 2016.

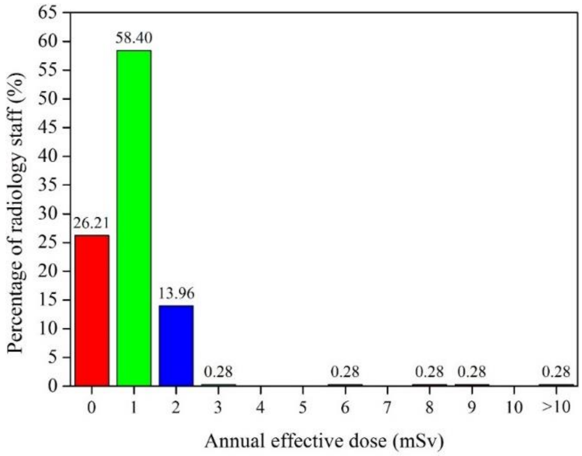

FIG. 1 Distribution annual effective dose records from 2014-2018

Cumulative effective doses in 5 years received by IR, radiologist, medical officer (MO), radiographer, nurse, medical physicist (MP) and healthcare assistant (HCA) are ranged from 5.53-28.57, 0.18-3.26, 0-8.92, 0-4.10, 0-4.09, 0.05-2.30, and 0.55-2.30 mSv, respectively as shown in Fig. 2. The results were well below the international recommended annual dose limit of 20 mSv and 100 mSv over the five-years period.

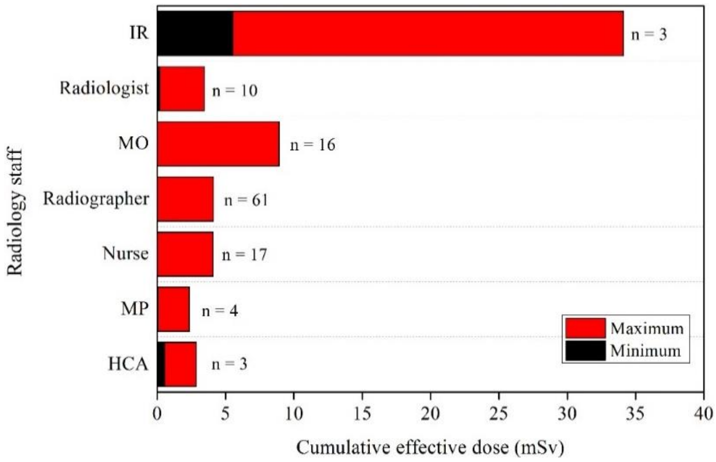

FIG. 2 Minimum and maximum individual cumulative effective dose records in 5 years (2014-2018)

Conclusion

Needless to say, IR will be received higher effective dose because they were exposed to x-rays when performing their routine works. Although average annual collective dose for radiology staff is 0.58 mSv and this value still below recommended limit (20 mSv), all radiology staff should be continuously provided with an appropriate medical education (CME) and training to enlighten the best practices working with irradiation facilities. In fact, suitable shielding selections for personnel should make available for radiology staff to ensure that the dose receive is kept as low as reasonably achievable. In the nutshell, the three key words in principles of radiation protections; (i) justification, (ii) optimisation, and (iii) dose limit, ought to understand and implements.

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70-ESTIMATING EXCESS CANCER RISK DUE TO DIGITAL DIAGNOSTIC RADIOGRAPHY EXAMINATION IN NORTHERN IRAN

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BACKGROUND

Ionizing radiation has an important role in diagnostic radiology. Exposure to diagnostic X-ray, due to its ionizing nature, can be a harmful source of radiation and has its risks [1, 2]. Therefore, the purpose of the current study was to estimate the Entrance Surface Dose (ESD) and Effective Dose (ED) in digital radiography and estimating excess cancer risk due to digital radiography examination in Northern parts of Iran.

METHODS

The data in this study were collected from 15 digital radiology centres for two age groups (10-15 and adults). For the purpose of this study the ESD and ED for the following radiographs were estimated: Skull (PA and lateral views), cervical spine (AP and lateral views), chest (PA and lateral views), thoracic spine (PA and lateral views), lumbar spine (AP and lateral views), pelvic (AP view), and abdomen (AP view). In order to obtain the organ dose, effective dose and radiation risk estimation of the above examinations, PCXMC (v2.0) was used.

RESULTS

The summary of minimum, maximum, the ratio of maximum and minimum, and average dose (mGy) for each X-ray examination presented in Table 1.

TABLE 1. MINIMUM, MAXIMUM, THE RATIO OF MAXIMUM AND MINIMUM, AND AVERAGE DOSE FOR EACH EXAMINATION

X-ray examination	Age	Entrance Surface dose Value (mGy)			
		Mean±SD	Minimum	Maximum	Ratio Max/Min
Skull (PA)	10>15 year	1.09 ± 0.42	0.51	1.82	3.57
	adult	1.19± 0.41	0.52	2.31	4.53
Skull(LAT)	10>15 year	1.08± 0.38	0.71	1.89	3.71
	adult	1.16± 0.40	0.54	2.09	4.10
Cervical spine(AP)	10>15 year	0.97± 0.21	0.65	1.15	2.25
	adult	1.17± 0.24	0.49	1.88	3.69
Cervical spine(LAT)	10>15 year	0.99± 0.30	0.73	1.18	2.31
	adult	1.20± 0.27	0.69	2.71	5.31
Chest(PA)	10>15 year	0.99± 0.36	0.68	1.21	2.37
	adult	1.05± 0.31	0.58	2.11	4.14
Chest(LAT)	10>15 year	1.32± 0.41	1.19	1.49	2.92
	adult	1.56± 0.43	1.11	3.01	5.90
Thoracic spine(AP)	10>15 year	1.38± 0.49	1.23	1.51	2.96
	adult	1.62± 0.51	1.50	2	3.92
	10>15 year	1.90± 0.69	1.56	2.26	4.43

Thoracic spine(LAT)	adult	2.36± 0.67	1.79	2.89	5.67
Lumbar spine(AP)	10>15 year	2.32± 1.21	1.73	3.09	6.06
	adult	2.59± 1.27	1.86	6.05	11.86
Lumbar spine(LAT)	10>15 year	3.62± 1.38	2.78	4.99	9.78
	adult	3.85± 1.44	2.42	6.25	12.25
Abdomen	10>15 year	1.76± 0.65	1.20	2.37	4.65
	adult	1.99± 0.67	1.49	2.78	5.45
Pelvis	10>15 year	1.63± 0.87	1.24	2.11	4.14
	adult	1.74± 0.89	1.39	3.25	6.37

ED and the number of excess cancers due to common radiographic examinations for the total population are shown Table 2.

TABLE 2. NUMBER OF EXCESS CANCERS DUE TO RADIOGRAPHIC EXAMIATIONS

X-ray examination	Effective dose (10-15 years) µSv	excess cancers for the total population in one year	Adult Effective dose µSv	excess cancers for the total population in one year
Skull (PA)	0/0104	0/897	0/01335	4/60575
Skull(LAT)	0/0188	0/897	0/02094	4/60575
Cervical spine(AP)	0/03388	1/339	0/03691	6/87525
Cervical spine(LAT)	0/01504	1/339	0/01831	6/87525
Chest(PA)	0/07436	7/7051	0/08087	39/56273
Chest(LAT)	0/06222	0/715	0/06779	3/67125
Thoracic spine(AP)	0/11664	0/2652	0/13145	1/3617
Thoracic spine(LAT)	0/11253	0/2652	0/12806	1/3617
Lumbar spine(AP)	0/24679	1/7953	0/29802	9/218175
Lumbar spine(LAT)	0/15968	1/7953	0/20111	9/218175
Abdomen	0/33175	2/587	0/37846	13/28325
Pelvis	0/17723	2/4518	0/20768	12/58905

CONCLUSION

Results reveal that in both groups the increased risk of cancer and fatality due to cancer was associated with three techniques: lumbar spine (AP), abdomen (AP) and pelvic (AP). Furthermore, this risk higher for the 10-15 age range. Therefore, choosing the proper technique is a matter of great care and sensitivity for the patients aged between 10 and 15. Besides, the Quality Assurance (QA) plan, emphasizing the quality of detectors and investigating other radiographic equipment, can be a logical step towards reducing the patient dose.

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71-STRATEGY OF COMPUTED TOMOGRAPHY LIVER IMAGE OPTIMISATION IN ABDOMINAL CT SCAN

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BACKGROUND

CT scan examinations have markedly increased in recent decades and the annual per caput effective dose has doubled in the recent decade [1]. This has increased the risk of cancer due to CT scan examinations [2]. Therefore, one of the most important aims in CT scans examinations is achieving is doing it with the lowest dose and suitable quality [3]. There are different strategies for reducing the dose in CT scans. The purpose of this study was to optimize the liver imaging parameters (tube potential, tube current time and pitch) in abdominal CT scan.

METHODS

The subjects of the study were 20 patients with normal Body Mass Index (BMI) who needed an abdominal CT scan in order for their livers to be examined. All the examinations were non-contrast. The patients were divided into 4 groups. Group A were patients scanned using the conventional parameters and group B, C and D patients were scanned using optimized (new) parameters (table 1). Dose indexes (DLp, CTDIvol and CTDIw) and Image noise for each group was measured ad then organ dose and ED were calculated using ImpactDose software.
The objective image quality was based on the amount of noise and was obtained by measuring the standard deviation of Ct number for one standard pixel in a one-centimeter ROI. Measuring points included anatomical parenchyma of liver, port vein and IVC. Subjective assessment was done through scoring system. In this method three radiology specialists assessed the images and the scores given by specialists were put into SPSS software to be investigated.

TABLE 1. CT SCAN PARAMETERS AND DEFAULT AND OPTIMIZED DOSES IN ABDOMINAL CT FOR ANALYZING LIVER

Protocols	Group A	Group B	Group C	Group D
Voltage (kVp)	120	120	120	100
Current (mA)	265	235	135	145
Rotation time (s)	0.8	1	1	0.8
Collimation (mm)	2*2.5	2*5	2*5	2*7
Pitch	1.35	1.35	1.5	0.95
Window width/ level Hounsfield unit (HU)	350.40	350.40	350.40	350.40
Scan Length (mm)	162	162	162	162
mAs	212	235	135	116.0
Scan Time	20.8	14.0	13.0	11.3
CTDIw (mGy)	18.7	15.0	10.0	3.9
CTDIvol (mGy)	13.9	11.12	6.9	4.17
DLP (mGy × cm)	243.4	209.7	107.9	76.4

RESULTS

The value ratios of dose indexes for group 3 to group 1 in liver CT scan examination were CTDIw=87%, CTDIvol=101% and DLP=125.5%. Reduction in these indexes resulted in the reduction of patient dose and organ dose. The obtained Effective Dose was 2.2 mSv for group A, 4 mSv for group B, 2 mSv for groups C and 7.1 mSv for group D. variation range and max-min signal-to-noise ratios for the four groups are shown in table2. The results of subjective image quality assessment which was done by 3 radiology specialists using 0-5 scoring system are shown in table 3.

TABLE 2. VARIATION RANGE AND MAX-MIN SIGNAL-NOISE RATIO VALUES IN 4 GROUPS

Groups	Liver parenchyma		Portal vein		IVC	
		Min	Max	Min	Max	Min
Group A	31.8	21.5	40.3	23.4	22.8	21.7
Group B	20.6	14.2	24.7	13.6	16.2	11.9
Group C	12.8	10.8	16.9	9.8	13.0	10.2
Group D	10.7	6.8	12.5	7.4	11.5	6.2

TABLE 3. SUBJECTIVE EVALUATION SCORES

Groups	Patient number	Liver parenchyma			Portal vein			IVC		
		Radiologist 1	Radiologist 2	Radiologist 3	Radiologist 1	Radiologist 2	Radiologist 3	Radiologist 1	Radiologist 2	Radiologist 3
Group A	1	4	3	3	4	3	3	4	3	3
	2	3	3	4	3	3	3	3	3	3
	3	3	3	3	3	3	4	3	3	4
	4	3	3	3	3	3	3	3	3	3
	5	4	3	4	3	3	3	3	3	3
Group B	6	3	3	2	3	3	2	3	3	3
	7	3	3	3	3	3	3	3	1	3
	8	2	1	1	3	1	1	2	2	1
	9	3	1	1	3	3	2	3	1	2
	10	3	1	2	3	3	3	3	1	1
Group C	11	3	1	1	3	3	1	3	1	1
	12	3	1	2	3	3	3	3	3	3
	13	4	1	2	4	3	3	4	3	2
	14	3	1	2	4	3	2	3	3	2
	15	3	2	2	4	3	3	3	1	2
Group D	16	3	2	2	4	3	2	3	1	2
	17	3	1	2	4	3	1	3	2	1
	18	2	1	2	2	2	2	2	1	1
	19	2	1	1	2	2	1	2	1	1
	20	3	1	2	3	2	1	3	1	2

CONCLUSION

According to the results, group C had the lowest dose and more efficient quality compared to other groups and its parameters can be used for liver CT scan examinations.

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72-COMPARATIVE ANALYSIS OF THE EXTERNAL EXPOSURE OF DIFFERENT PROFESSIONAL GROUPS OF MEDICAL STAFF OF DUSHANBE CITY OF THE REPUBLIC OF TAJIKISTAN

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The work provides a comparative analysis of average annual individual radiation doses based on the obtained values of individual dose equivalent Hp (10), which had been obtained during 5 years of investigations (2014-2018) by using the thermo-luminescent dosimetry method for 70 employees working computed tomography, radioscopy, fluorography and radiography from 15 medical institutions in Dushanbe city. The ratio of medical personnel of the different professional groups showed that 63% of them were engaged in radiography, 19% x-ray, 10% computed tomography and 8% radioscopy. Analysis of the average annual radiation doses of every occupational group showed that the specialists in charge of fluorography have a high dose (max 1.74 mSv) and the personal of computed tomography have the lowest dose (max 1.34 mSv), and over time there is a tendency to equalize the values of average annual doses all professional groups in the area close to the value of 1.5 mSv. The obtained data of the effective annual dose for all occupational categories had not exceeded the permissible dose limits values required by the "Radiation Safety Standards" (NRB-06 SP 2.6.1.001-06). Calculations are given without subtracting background values. The analysis data allows to identify the most exposed professional groups and to address the issues of ensuring radiation safety in a targeted manner.

73-RADON MONITORING IN THE TERRITORY OF TAJIKISTAN

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The article presents the radon monitoring results in the territory of Northern Tajikistan. The Soghd region territory is characterized by the presence of a number of radioactive tailing dumps, caused by technogenic manifestations of increased radon hazard, the main of which are tailing dumps in the Istiklol city, the village of Digmay, the village of Adrasman and others.

Radon equivalent equilibrium volume activity (EEVA) in indoor air is an important characteristic for radon hazard of territories, considered when designing and operating buildings for various purposes. This is primarily due to the fact that EEVA is a complex characteristic that takes into account the volume activities (VA) of decay products of radon, such as ^{218}Po, ^{214}Pb and ^{214}Bi, which two of last are gamma-ray emitters. Radon inflow to indoor air is regulated by the presence of ^{226}Ra in the underlying surface, in building materials that make up the structure of a building. The radon EEVA determined on the basis of characteristic of a shift in radioactive equilibrium between radon and its decay products. The annual regime of radon EEVA is directly dependent on temperature and atmospheric pressure fluctuations.

Radon monitoring in the region had been carried out using both the integral measurement method and the instantaneous method and the higher value of radon VA is observed in the area of the *Digmai* radioactive tailing dump and the nearest settlement – Ghoziyon township, showed that the average ambient dose rate on the surface of the *Digmai* tailing dump reaches 20 μSv/h, the radon flux density from the surface of the tailing dump reaches 65 Bq/(m^2·s), while radon VA in the atmospheric air above the tailing dump in different areas varies from 200 to 1000 Bq / m3. It has been established that the specific activity of ^{238}U in the tailing material reaches 980 Bq/kg, and ^{226}Ra up to 7620 Bq/kg.

The surface of the *Gafurov* tailing dump is covered with loess-like loam up to 2.5-3.0 m thick, the exhalation does not exceed 0.1 Bq/m^{2}s with average values of 0.04-0.06 Bq/m^{2}s. These values are 10-50 times lower than the norms limit established for buried tailings.

The average value of radon VA in indoor air on the territory of Buston city is up to 47 Bq/m^3. Taking into account the instrumental measurement uncertainty, an array of EEVA values for Radon-222 has been obtained, and the distribution of occurrence frequency of EEVA values in indoor air in Buston city has been defined. The average values of radon VA in rooms varied in the range from 20 to 193 Bq/m^3.

Based on the results of radon monitoring, it has been found that the specific activity of uranium and radium radionuclides in the tailing material varies: 1405-2140 Bq/kg, 5935-9843 Bq/kg, respectively, and the average value of radon VA in the air is from 20-45 Bq/m3. It has been shown that the concentration of radon in the air above the surface of the tailing dumps is low. However, the exhalation of radon from the soil surface of all surveyed tailings is higher than the standards (1 Bq/(m^{2}s)).

The results of radon VA measurements in the air of residential buildings and facilities located around the Istiklol city territory showed that the radon VA value is in the range of 44-195 Bq/m^3. Higher value of radon VA is noted at the former dilapidated plant (1319 Bq/m^3).

The performed work analysis shows that at present, the negative impact of radiation hazardous objects on the environment on the territory of the Sughd region of the Republic of Tajikistan is mainly localized within the existing radioactive tailing dumps.

The results of radon monitoring indicate that uranium legacy sites are still sources for potential environmental pollution with radionuclides. The facilities of uranium waste in Istiklol city and the *Digmay* tailing dump are the most radon-hazardous areas and demand a set of measures to remediate the adjacent areas.

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74-THE CONTRIBUTION OF THE ROMANIAN SOCIETY FOR RADIOLOGICAL PROTECTION TO THE IMPROVEMENT OF THE SAFETY CULTURE IN ROMANIA

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The RSRP is an Associate Society to the International Radiation Protection Association since 1992. The paper intend to make a presentation of the Romanian Society for Radiological Protection with the occasion of the 30-th anniversary of its official establishment - especially to show the activity of developing the safety culture in Romania in correlation with the IAEA's Radiation Protection and Safety of Radiation Sources – Basic Safety Standards GSR Part 3 and the Council Directive No. 2013/59/EURATOM.

The society is an independent, non-profit, non-governmental organization of Romanian specialists in radiological protection involved in nuclear activities from industry, hospitals, research, waste management, radioactive source production, nuclear security, Cernavoda nuclear power plant operation and maintenance, research reactor operation and decommissioning. RSRP is permanently engaged for improving radiological protection culture among its professionals following a fundamental desiderate to protect the population and radiation workers against the harmful effects of ionizing radiation, to develop and make known the scientific, technical, medical and legal aspects of radiological protection on a nationwide scale and to imply the civil society in the benefit/risk analyses for the ionizing radiation uses.

INTERNATIONAL COOPERATION

The specialists from RSRP have had a good individual collaboration with IAEA since the early time of our organization and before. Beside the fact that first two authors of the presentation were in the roster of IAEA experts, the second author, after an IAEA fellowship at Lawrence Livermore National Laboratory, was implied, aside some IAEA personalities as A. J. Gonzales, R. V. Griffith, M. Gustafsson, G. A. M. Webb in developing computing programs for IAEA, including IAEA's BSS-115, English version, on diskette (1997, ISBN 92-0-100997-6) and on CD (2003, ISBN 92-0-106003-3 [1], IAEA's BSS-115, Spanish version (2004) and in IAEA's ORPGUIDE (2000, ISBN 92-0-103100-9). Although outdated now, these programs have played a significant role in spreading and deepening the notions contained in the old BSS and even in the exhaustive approach to the changes that led to the new BSS (GSR Part 3-Pub1578-2014). Also, directly or through the Romanian National Regulatory Authority, the Romanian National Commission for Nuclear Activities Control (CNCAN), there were made several observations regarding the transposition in Romania of the new regulations contained in the new international and European BSSs issued by IAEA and EURATOM in 2014, respectively.

RSRP is member of the Central European Association of IRPA Associate Societies since 2000 and has a memorandum of cooperation from 2001 with American Nuclear Society (ANS). Also it has a good collaboration with about all European Radiological Protection societies, Canadian Radiation Protection (Joint Association) and Japanese Society for Radiological Protection. Members of RSRP participated at all IRPA Congresses beginning with the 9-th, held in Vienna in 1996, at many European IRPA Congresses and some National Conferences (Croatia, Slovenia, Hungary).

Particularly appreciated was the organization by RSRP in Brasov on 24-28 September 2007 of the IRPA Regional Congress for Central and Eastern Europe, with the theme "Regional and Global Aspects of Radiological

Protection" [2]. New issues related to exposure to ionizing radiation, regulations and policies in radiological protection, from international requirements to regional issues were addressed at this congress.

Two images from the Conference and the cover of its book are showed in Fig.1

Fig.1. IRPA Regional Congress for Central and Eastern Europe, 24-28 September 2007, Brasov, Romania

RSRP ACTIVITIES

Also, the specialists from RSRP published few books, as well as numerous articles in environmental and radiological protection journals and also participated in numerous radios or television shows where they presented and debated radiological protection issues. Special attention was paid to the effects of the Chernobyl Accident (1986) in Romania, problems related to the Fukushima Nuclear Accident – Japan in 2011, as well as the operation of nuclear energy or research reactors in Europe. The 1st author visited Belarus after Chernobyl accident and then Fukushima, publishing his impressions in the ANS Globe no. 21 from June 2014.

On the occasion of the annual conferences organized by RSRP, there is a good exchange of information between its specialists, as well as with other invited persons/personalities from Romania and abroad.

RSRP publishes all presentations from the annual conferences in proceedings. From 2014 until 2017, the topics of the National Conferences were related to the Council Directive 2013/59/EURATOM. RSRP works intensively on organizing various scientific manifestations or popularizing knowledge related to the use of nuclear energy for various peaceful purposes to ensure radiological protection, including high school students from Bucharest schools and organizing joint scientific events within International Symposium for Nuclear Energy (SIEN) and with the Society Romanian of Labor Medicine, Woman in Nuclear (WIN), CNCAN and the Romanian Association for Nuclear Energy (AREN).

More about RSRP activities can be seen in www.srrp.ro. Several covers of the books issued and published by RSRP, as “Natural Radioactivity in Romania”, “Artificial Radioactivity in Romania”, “National Conference regarding the Concept on Culture of Radiological Protection and his role in the protection of population and environment”, National Conference with the occasion of Romanian Centenary “Evolution of Radioprotection in Romania” and National Conference in 2019 on ”Application of the ALARA principle in optimizing radiological protection – new implementations” are showed in Fig.2.

Fig.2. Few Romanian books and RSRP national conferences – covers

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75-CHALLENGES FACED BY OPERATORS IN DEVELOPING COUNTRIES TO ACHIEVERADIATION SAFETY IN MEDICAL FACILITIES: *CASE STUDY OF UGANDA*

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The Atomic Energy Council (AEC) was established by Atomic Energy Act no. 24 of 2008 (AEA, 2008), an Act of Parliament to regulate the peaceful application of ionizing radiation in Uganda [1]. AEC sets the standards, guidelines and monitors compliance to ensure safety and security of people and the environment from the dangers arising from ionizing radiation [2]. In exercise of the powers conferred upon the AEC by section 73 of the AEA, 2008, the Atomic Energy Regulation (AER, 2012) came into effect. A number of medical facilities were already in possession and/ using radiation sources yet not regulated. AEC has realized a number of achievements but the challenges still remain especially in enforcing radiation safety in the medical facilities [3]. In this paper, a discussion of the challenges faced by operators in order to achieve radiation safety in medical facilities is presented.

a) The relatively high cost of the Quality Control (QC) kits required by some facilities in order to perform quality controls to ascertain the performance of the machines and perform safety assessments in their facilities is increasingly becoming a challenge. These facilities are not able to perform these QC tests and thus a compromise to radiation protection and safety for the workers and the patients.
b) There is a shortage of equipped and competent qualified experts like Clinically Qualified Medical Physicists (CQMPs) and Biomedical Engineers. Uganda currently has Six (6) CQMPs, three of which work in the Uganda's only radiotherapy Centre, the Uganda Cancer Institute. Two (2) work in Academia and One (1) in the regulatory body. This has left the facilities wanting since the available CQMPs are already absorbed in government and are not able to give qualified expert guidance to a country that has over 450 active facilities. In addition, Medical Physics is not yet recognized as a profession in Uganda.
c) The lack of National Diagnostic reference levels (NDRLs) for all practices in Uganda is still a challenge due to the financial constraint and skilled human resource required. Establishment of these NDRLs by the Ministry responsible for Health and professional bodies in Uganda in conjunction with AEC would strengthen the optimization of protection in the medical exposure of patients as required by Regulation 51 of the Atomic Energy Regulation, 2012.
d) The unsuitable room designs by some facilities that operate in rented premises cause a huge risk to ensuring radiation safety. Most of these rooms are not designed to house radiation sources and as a result high radiation levels might be realized in the surrounding controlled areas.
e) There is a low and poor safety culture by the administrators and operators in the facilities. Some facilities wait to be enforced by the regulatory body in order for them to implement the various radiation safety aspects that are highlighted in the inspection reports, and this reduces radiation safety.
f) The current Atomic Energy Regulation (AER, 2012) has not been reviewed to be in line with GSR Part 3 of 2014. This reduces on the levels of radiation safety that would be attained in the facilities since the available provisions for medical exposure protection in the AER, 2012 are not exhaustive.
g) There are a few available qualified and skilled radiographers especially in "hard to reach" districts where one radiographer part-times in a number of hospitals located in different districts. With the poor road network in these districts, transport is not easy. These facilities end up working on a schedule since the radiographer would be rushing to get to their next work-station and this compromises the radiation safety levels in these facilities.
h) Uganda lacks accredited technical support organizations (TSOs) to provide technical advisory and support services in radiation safety and health surveillance to facilities that possess radiation sources.

These challenges could be overcome through; Continuous sensitization of facilities (licensees and radiation workers) on the need of maintaining a good strong safety culture in the workplace. More stringent provisions on the performance testing of the radiation machines imported into the country. Fast tracking the recognition of Medical Physics as a profession in Uganda through the Ministry of Health and the Uganda Medical Physicists Association. Inclusion of Medical Physics as a course unit and or standalone courses at Bachelors level in tertiary institutions of learning so that students appreciate the role of Medical Physicists in the society, and thus becoming an option in their career path. National trainings could be organized e.g. on how to establish National Diagnostic Reference Levels for different practices.

In conclusion, operators in facilities could create a network that would help them co-ordinate and work together in ensuring radiation protection and safety in their facilities. In addition, operators working closely with the regulatory body would go a long way in realizing radiation safety in facilities.

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76-APPROPRIATENESS OF COMPUTED TOMOGRAPHY REQUISITIONS IN SELECTED HEALTH FACILITIES: *A case study of sub-Saharan Africa*

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INTRODUCTION

The rapid advances in diagnostic imaging technology such as the multi-detector computed tomography (MDCT) have revolutionized the practice of medicine by improving patients' care outcomes. However, this has also prompted unjustified and inappropriate imaging requisitions in ranges of 20%-50% globally (1, 2). Such inappropriate use wastes resources and exposes patients to unnecessary radiation, increasing the life time risk for radiation induced cancers especially in children (2).

Unfortunately, sub-Saharan Africa has limited capabilities to acquire innovations in imaging like ultrasound and magnetic resonance imaging (MRI) which use non ionizing radiation. Because of limited access to MRI, computed tomography (CT) scans are performed even when MRI would be the most appropriate investigation. MDCT is associated with high doses of radiation with 2% of all future cancers being contributed to previous CT scans exposure (3, 4).

Although the level of appropriateness of imaging requisitions has not been determined in many of sub-Saharan African countries, the situation is likely to be worse due to absent or weak functioning policies, regulations, standards and guidelines for radiation safety practices. This is a radiation safety and public health concern given the young population in this region where 75% are 35 years and below with a median age of 15.8 years(5).

There is however, paucity of information concerning the level of appropriateness of CT scan requisitions in this region. Unfortunately, the findings from the high resource settings cannot be extrapolated to the low resource countries due to differences in infrastructure, costs, disease patterns and clinical practice settings.

Therefore, this study aims at determining the level of appropriateness of CT imaging requisition in six selected hospitals in sub-Saharan Africa.

METHODS

A total of 842 CT request forms for patients ≤ 35 years seen between 1st July to 31st December 2018 from Mbarara Regional Hospital, Kiruddu general Hospital, Kampala Hospital limited, Mount Elgon, St. Francis Nsambya and Mengo Hospital were reviewed for appropriateness.

Appropriateness of the CT procedure was determined using the I-Guide (ESR clinical imaging guidelines), as the difference between the score on the most ideal exam and the exam requested by a referrer using the stated clinical indications on the referral form. Frequencies and Proportions were used to present the level of appropriateness for the different health facilities

RESULTS

The level of inappropriateness of CT requisitions in six selected hospitals was variable ranging from 31% to 59%. Figure.1. shows the level of inappropriateness of CT requisitions for the six hospitals Mbarara (47%), Kiruddu (44%), Kampala (59%), Mbale (31%), Nsambya (59%) and Mengo hospitals (53%).

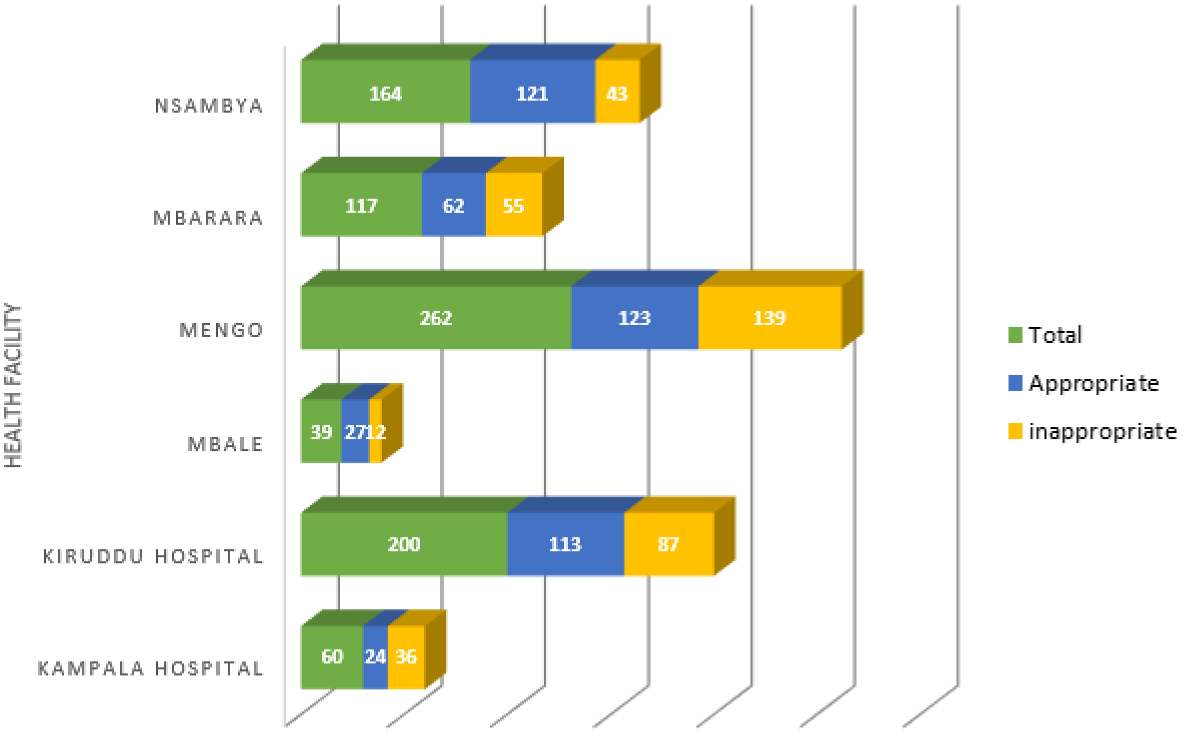

FIG. 1. Apappropriatness CT requistions in six health facilities

CONCLUSION

Such levels of inappropriateness indicate that many patients get unjustified radiation which poses a health risk to them.
Generalizability of these results to sub-Saharan hospitals is limited, since the data was collected from six hospitals in Uganda. Therefore, further research is needed to expand appropriateness evaluation in this type of health care setting, to investigate the causes of inappropriate use of CT imaging examinations and to evaluate the effectiveness of some strategies such as the use of evidence-based interventions like clinical imaging guidelines in order to reduce unnecessary imaging.

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77-OPERATIONAL RADIATION SAFETY FOR THE MANAGEMENT OF LEGACY DISUSED SEALED RADIOACTIVE SOURCES IN CAMEROON

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During the twentieth century, the use of ionizing radiation was not made in the framework of the regulatory control in Cameroon since the first general manager of the National Radiation Protection Agency (NRPA) has been appointed in 2007. Legacy sources which are those that pre-date effective regulatory requirements have been found in the country.

A lack of appropriate controls can lead to radiological incidents including lost, theft, unauthorized sale, environmental contamination, public exposure and recycling in scrap yard. Some of these incidents can have serious consequences, including the death of several exposed persons as well as environmental impacts and serious economic consequences. According to this, measures have been taken to recover legacy and orphan radioactive sources in Cameroon. Search and secured of orphan radioactive sources operation implemented by the National Radiation Protection Agency (NRPA) which is the regulatory body in the country according to the decree 2002/250 of 31st October 2002 on its creation, organization and functioning [1], is conducted once per year in scrap yards, municipal waste storages and time to time in suspected locations.

Searches for sources can primarily be divided into administrative and physical aspects. The difference is that, physical searches will involve the use of radiation detectors to determine the presence of sources. According to unpublished NRPA guide on search and secured orphan radioactive sources [2], and the IAEA TECDOC 1388 on strengthening control over radioactive sources in authorized use and regaining control over orphan sources [3], administrative search is applied when they are any means that do not involve the use of radiation detectors to gather information about sources that are unknown, lost, missing, stolen or found.

Search operation has been coupled to International Atomic Energy Agency (IAEA) fact finding mission to determine origin of some legacy sources, which technical information as well as special form certificates were not found. These sources including, 01 ^{60}Co source with a serial number of 3860, and an initial activity of 203.6 TBq on 06/02/1995 located in Radiotherapy Department of the Yaounde General Hospital, 01 category 1 ^{60}Co source with a serial number of 3271 and initial activity of 60.2 TBq on 13 September 1985 found in Polyclinic Bonanjo, 01 category 3 ^{137}Cs source with a serial number of 1934 and initial activity 6.03 GBq on 27/03/1969 found in Department of Physics of the University of Yaounde I, 01 ^{60}Co source with unknown characteristics contained in the dredge “GAROUA” which has been packaged in concrete blocks located in Douala Sea Port, 08 moisture gauges containing 08 ^{137}Cs and 08 ^{241}Am-Be sources stored in the public area in Labogenie which is national civil engineering laboratory.

According to the graded approach, 02 category 1 ^{60}Co sources located in Yaounde General Hospital and Polyclic Bonanjo and 01 ^{137}Cs located in Department of Physics of the University of Yaounde have been repatriated in France. Operational measures have been taken during this specific operation which had involved sealed high activities radioactive sources (SHARS) [4]. The mature draft radioactive waste management national policy and strategy, which present responsibilities of the involved administration and technology options planned, management of category 3 to 5 low activity disused sealed sources and the return to the suppliers for category 1 to 2 disused sealed sources, has been finalized [5]. Management of the low activity disused sealed radioactive sources is conducted within the newly constructed Cameroon interim storage facility presented in the following figures.

FIG.1. Interim storage facility.

FIG.2. Inside the container

From a regulatory viewpoint, all authorized sources need to be adequately controlled. However, because resources are often limited, the priority from a safety and security perspective must lie with the dangerous, higher category sources according to the graded approach. Therefore, the efforts associated with national strategies should concentrate on these high activity sources. For improving control over radioactive sources, Consideration needs to be given to the flow of radioactive material into and out of the country and the status of the regulatory control of sources in neighbouring and trading countries. Ideally, the goal is to regain control over all orphan sources and increase control over vulnerable sources. A desirable objective in this regard would be full compliance with the international basic safety standards, the requirements for legal and governmental infrastructure for Radioactive Waste and transport safety and full implementation of the Code of Conduct on the Safety and Security of Radioactive Sources [6; 7].

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78-MONITORING ON OCCUPATIONAL EXPOSURE OF RADIATION WORKERS FOR RADIATION PROTECTION IN MYANMAR (OSLD, MYANMAR)

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BACKGROUND

External monitoring of radiation workers from various sectors was carried out by using personal dosimetry badges, i.e. Optically Stimulated Luminescent dosimeters (OSLDs). Use of OSL as personnel dosimetry techniques introduced since 2014, at the Division of Atomic Energy, Radiation Protection Section, using ICRP Limit in services work for the measurement of radiation dose delivered to personnel (individual) during their occupational exposure. This service has started since 2000 with Thermoluminescence Dosimetry (TLD) before OSL measurement and the number of badges using by radiation workers gradually increases up to 1600 workers till now. The performance of OSLDs, Al_2O_3: C was evaluated in terms of the operational quantity of Co-60 external beam. The reproducibility, signal depletion, and dose linearity of each dosimeter was investigated by 2016, Office of Atoms for Peace (OAP), coordinate with Thailand institute of Nuclear Technology (TINT) established " Intercomparison of Personal Dose Equivalent for Hp (10) and Hp (0.07) for OSL Dosimetry " project. As a result, for Hp (10), and Hp (0.07), the uncertainty was 2.7 and 2.1 with 95% confidence level. In March 2019, "Intercomparison of Personal Dose Equivalent Hp (10) for Photon field for Individual Monitoring Services Laboratory in Southeast and Asia Region" project, Myanmar got Intercomparison Certificate No.IC-003. Understanding the health impacts of low-level chronic public exposure is critical, about 1700 Myanmar workers from various areas in artificial (man-made) sources of radiation from medical uses : private, institutes, government hospital, company, military hospital were compiled the date for personal radiation monitoring services, and the received total annual collective dose was 9.3 mSv in 2019. This paper will endeavor to highlight all of the radiation services in the field of external personnel dosimetry in Myanmar.

Keywords: Thermoluminescence Dosimetry (TLD), Optically Stimulated Luminescent (OSL), personal monitoring, medical facility.

METHOD AND PROCEDURE

Before measurements, OSL dosimeters were prepared its procedures as marking bar code on the dosimeters, putting into the holder and repack into the plastic packages to reduce damages. The absorbed doses in the coded badges were measured by OSL dosimetry method at Occupational and Medical Exposure Control Lab. The badges were coded with serial number 1, 2, 3, 4, and 5,…put in the different areas and were sent to monitor in different medical facilities for 2 months. After two months using, OSL badges were sent back to DAE. Other new badges were given to the customers for further personal monitoring. After storage interval, the absorbed radiations (radiation exposure in OSL dosimeter) were measured with OSLD technique. The whole process is applied ICRP (International Commission on Radiological Protection) limits to prevent radiation workers of different areas, with personal dosimetry services. OSL system (In Light 2T Automatic reader and Annealer Modal Auto 50A) was kept in a cool room, at temperature 25°C, and humidity RH = 40% - 80%.

RESULTS

The observed values of personal radiation monitoring services from various sectors are depicted in Fig1. It can be seen that, the use of radiation workers gradually increases up to 1700 at the end of 2019. In addition, in Fig 2, the average annual collective dose in different areas (private, institute, government hospitals, companies, military hospitals) are shown respectively.

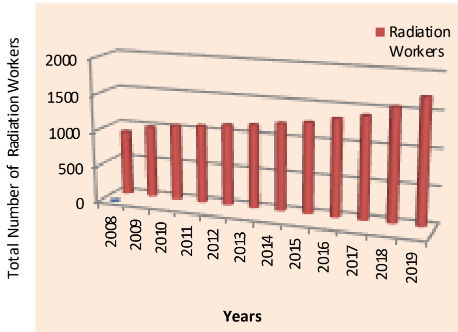

FIG. 6. Yearly increases of radiation workers.

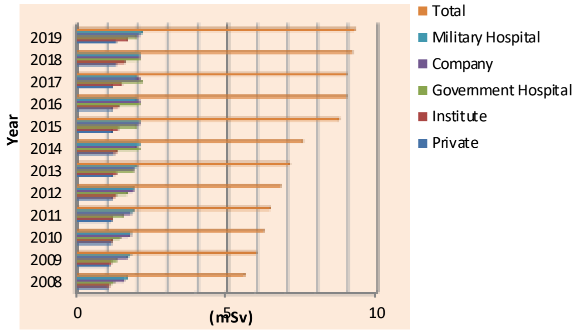

FIG.2. Average annual collective dose in different areas.

CONCLUSIONS

The absorbed doses in the coded badges were measured with OSLD technique at Radiation Protection Section, Occupational and Medical Exposure Control Lab, in Myanmar. It is observed that the maximum annual collective dose values from different area are not more than 10 mSv, which is significantly lower than 20 mSv/year according to ICRP Dose Guidance Value of Occupation Exposure limit for radiation workers. Therefore, it is ensure that uses of radioactive sources and radiation apparatus is reliable and safety for radiation workers. From the above study, it can be said that personal individual monitoring services under regulatory control gives safety aspects of external radiation hazard for radiation workers in Myanmar.

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79-CAPACITY DEVELOPMENT OF RADIATION METROLOGY IN MYANMAR (ESTABLISHING A NATIONAL SSDL IN MYANMAR)

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INTRODUCTION

Although Myanmar has no nuclear installation, requirement of radiation metrology in view of range and accuracy is greater with the increase in use of ionizing radiations in various fields of life. Annually, approximately more than 7500 new cancer patients are registered in all radiotherapy centres, in Myanmar. Radiotherapy treatment is offered to cancer patients for free of charge in the available facilities which will be extended in the near future for several numbers. Myanmar strongly needs the Agency's assistance to establish a National Secondary Standard Dosimetry Laboratory in order to provide calibration services, to improve dosimetry accuracy and training in radiation measurement and calibration techniques. So, Myanmar is trying to establish the Secondary Standard Dosimetry Laboratory (SSDL) as national development plan which may cover all levels of radiation in order to reduce undesirable risks of radiation exposure to radiation-induced activities.

Keywords: Secondary Standard Dosimetry Laboratory (SSDL), dosimetric accuracy, calibration of dosimeters, irradiation facilities

INCREASES OF USING RADIATION SOURCES AND APPARATUS AND RADIATION WORKERS

Number of users of radioactive sources and radiation apparatus who hold licence from Division of Atomic Energy (DAE) and adhere to statutory requirements laid down in the Atomic Energy Law, increases yearly. It can be also seen that, the number of facilities and radiation workers who uses OSL dosimeter for (Personnel Radiation Monitoring) gradually increases up to 1700 at the year of 2019 as shown in Figure 1.

There are 18 linear accelerators and 4 Co-60 units, 7 brachytherapy sources for radiotherapy and 3 therapeutic for nuclear medicine. Number of facilities for general diagnostic radiology, dental radiology and veterinary radiology for diagnostic is over 1800. All facilities, nearly 1900 included medical, industrial, research and education, need to be assessed dosimetric accuracy for corresponding level of radiation such as radiation protection, diagnostic and radiotherapy. Percentages of facilities of radiation level in Myanmar are as shown in Figure 2.

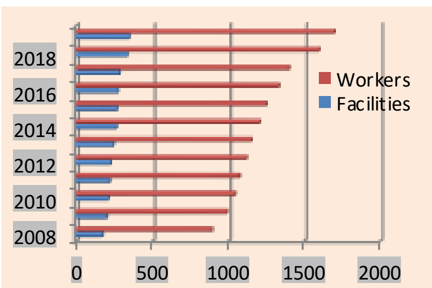

FIG.1. Increases of radiation workers and facilities

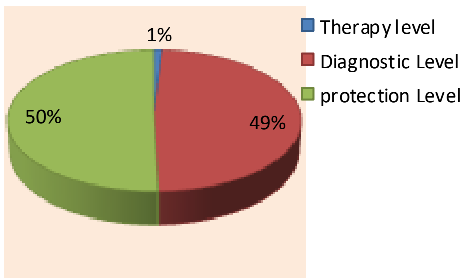

FIG.2. Percentages of facilities in radiation level

IRRADIATION FACILITIES AND HUMAN RESOURCE DEVELOPED CAPACITY

Standard X and gamma fields, in wide range of doses, to enable calibration of dosimeters that are used in both radiation protection and radiotherapy were considered before construction of the laboratory. The laboratory is based on the concept that it may be used for realization of X and gamma primary standards for exposure and absorbed dose, in the future. The design of the calibration facilities has been constructed in accordance with the relevant national and international safety regulations and also taken into account the International Basic Safety Standards (Safety Series No 115). It is located in a separated one - story building and consists of three irradiation rooms and one compound control room. Complete set of X-ray system including monitor chamber and dosimetry equipment supported by IAEA have already been received and installation may be done at the end of 2020. To receive Gamma Beam Irradiation system is under processing and IAEA will provide in accordance with relevant IAEA Safety Standards and Guidance.
SSDL Team leader has visited Thailand for two weeks to experience SSDL's activities and two members have attended to get two- month Fellowship training course in 2016. In addition, one of the SSDL members also has visited Malaysia for one week of Scientific Visit and the other three members have attended two-month Fellowship course in 2017. One of the SSDL team members has also got the training Activity on the Establishment of a Secondary Standards Dosimetry Laboratory and a Quality Management System at Vienna, Austria in 2018.

FURTHER GOALS

All calibration works and procedures will be undertaken according to methods and procedures of nationally or internationally recognized. Statistical techniques will be applied for reviewing of the results. The Laboratory will be participated with inter-laboratory proficiency-testing programs organized by the IAEA or other standardization laboratories in order to ensure the quality of the calibration results. Then, the SSDL will be upgraded to radiotherapy level and also participated in measurement comparisons within the IAEA/WHO SSDL Network, to exchange information and improvement of measurement techniques.

CONCLUSION

Nuclear technology and application of radiation is widely spread in all fields and nowadays, medical utilization is the highest using field of radiation so that the higher dosimetric accuracy is important and needed for our country, Myanmar. As SSDL has been established in Myanmar, which may cover for all levels of radiation and so that undesirable risks of radiation exposure can be reduced by improving dosimetric accuracy and the safe use of every application of radiation technology in Myanmar can be fostered in the future.

ACKNOWLEDGEMENTS

The authors would like to acknowledge the Director General of Department of Technology Promotion and Coordination under the Ministry of Education, Myanmar, for his kind permission and guidance. The authors are also grateful to the Project Management Officers (PMO) and Technical Officers (TO) from IAEA, Radiation Protection Section, OSLD Laboratory (OSLD and the project team of SSDL) under Division of Atomic Energy for their supports and guidance.

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80-ESTABLISHMENT OF DIAGNOSTIC REFERENCE LEVELS (DRLS) IN DIAGNOSTIC RADIOLOGY IN INDIA

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BACKGROUND

Optimization of medical exposures without compromising on the diagnostic information available for the medical imaging procedures is a goal that all countries are trying to achieve to reduce population exposures and radiation risk to the population to optimal levels [1]. Setting up of Diagnostic Reference Levels (DRLs) for common diagnostic examinations in various imaging modalities is the first step in eliminating unnecessary patient exposures in diagnostic radiology. To establish DRLs, at National level, there is a need to survey across the country and measure patient doses for sample populations [2]. For want of data, the regulatory body in India has so far not come up with any NDRL recommendations in medical imaging procedures. To help the regulatory body in this area, PSG Hospitals, situated at Coimbatore, a city in the southern part of India, initiated a DRL program in the year 2014, with the support and a liberal grant from the national regulatory body. The purpose of this paper is to summarize the work so far done by PSG research group in this field.

METHODS

(a) In CT imaging, the first study was to measure adult CT radiation dose for head, chest and abdomen procedures in South India. The study was performed on 110 CT scanners - Data collected with regard to 16,500 examinations in patients - installed in major cities (from four chosen states) of South India. For adult CT DRL, CTDI PMMA head and body phantoms were placed one after the other on the couch as per FDA's recommendation and the ion chamber was inserted into the phantom along the central axis and the doses received by the head and body phantoms at the centre ($CTDI_{100,\,c}$) and periphery ($CTDI_{100,\,p}$) were measured respectively [3]. Using these dose values, the other CT dose indices viz, $CTDI_w$, $CTDI_v$, and DLP were calculated. From the overall calculated $CTDI_v$ and DLP value, the 75th percentile of $CTDI_v$ and DLP for head, chest and abdomen procedures of adult thus calculated was calculated as the respective third quartile values (Table 1).

(b) In the second study, the regional peadiatric CT dose indices for selected procedures viz., head, chest and abdomen were carried out in Tamilnadu state towards establishing the peadiatric CT DRL. This work was performed for 30 CT scanners installed at various parts of the Tamilnadu region. Patient cohort was divided into two age groups: < 1 year and 1 to 5 years [4]. Data for at least 20 patients of each age group, for each examination (a total of 120x30 = 3600 procedures) were collected using survey. The peadiatric CT phantom dose studies were carried out and third quartile values were arrived (Table 2).

(c) In third study, 67 panoramic dental X-ray machines including 22 different models were randomly selected based on workload from the Tamil Nadu region and measurements were performed. The product of measured dose and the measured radiation field area gave the DAP values [5]. After calculating DAP from 67 panoramic dental scanners, 3rd quartile values were calculated using Microsoft Excel (Table 3).

RESULTS

TABLE 1: COMPARISON BETWEEN SOUTH INDIAN AND OTHER COUNTRY ADULT CT DRL

Study region	Proposed South Indian Adult CT DRL 2017		Other country CT DRLs							
			EC 1999		Germany 2010		Switzerland 2010		Norway 2009	
	$CTDI_v$ (mGy)	DLP (mGy.cm)	$CTDI_v$ (mGy)	DLP (mGy.cm)	$CTDI_v$ (mGy)	DLP (mGy.cm)	$CTDI_v$ (mGy)	DLP (mGy.cm)	$CTDI_v$ (mGy)	DLP (mGy.cm)
Head	47	1041	60	1050	60	1050	60	800	75	1000
Chest	10	445	12	650	12	400	10	400	15	400
Abdomen	12	550	35	900	20	770	15	710	15	710

TABLE 2: COMPARISON BETWEEN TAMIL NADU AND OTHER COUNTRY PEADIATRIC CT DRL

Study region	Age group in year	Peadiatric Tamilnadu DRL 2017		Peadiatric European DRL 2015	
		$CTDI_v$ (mGy)	DLP (mGy.cm)	$CTDI_v$ (mGy)	DLP (mGy.cm)
Head	**< 1**	20	352	25	300
	1 to 5	20	360	25	505
Chest	**< 1**	7	120	3.3	80
	1 to 5	8	132	5.6	115
Abdomen	**< 1**	12	252	5.7	160
	1 to 5	14	270	5.7	170

TABLE 3: COMPARISON BETWEEN TAMIL NADU AND OTHER COUNTRY ADULT OPG DRL

Country	Year	Peadiatric Panoramic DRL ($mGycm^2$)	Adult Panoramic DRL ($mGycm^2$)
Greece	2004	77	117
Germany	2007	75.4	101.4
Great Britain	2009	82	-
UK	2010	67	-
Korea	2011	-	120.3
Korea	2014	95.9	-
Kosovo	2019	73	93
Tamil Nadu, India	2019	81.8	114

CONCLUSIONS

Based on these studies, DRLs, established by us, can be used by other hospitals, as reference values, to reduce the spread in doses amongst the adult and peadiatric patients in CT and Dental procedures. In future, the study will be extended to the other parts of India so that national DRLs can be set for different imaging procedures. This will enable the Atomic Energy Regulatory Board to recommend national DRLs for various imaging modalities.

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81-IMPROVEMENT IN THE MONITORING OF TRITIUM IN HYDROSPHERE OF UKRAINE

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At present, the problem of tritium contamination of aquatic ecosystems in the areas of location of nuclear fuel cycle facilities can be considered as one of the key issues in radioecology. This is due to the relatively stable increase of its concentrations in the natural water within recent decades, the high migratory capacity of tritium as well as the absence of reliable systems of this radionuclide containment and retention.

Tritium is a radioactive isotope of hydrogen with a physical half-life of 12.3 years. Tritium enters the natural environment irregularly and rapidly migrates from its primary pollution sites, so its single and unsystematic measurements do not allow identifying the real scale of contamination of water and biological systems. Due to these reasons, tritium has become a global polluter of the Earth's natural water ecosystems.

The initial phase of the formation of tritium background of the surface water bodies in Ukraine was formed as a result of tritium emissions from nuclear weapons tests. The development of the second stage of tritium background is connected with releases of nuclear power plants and the accident on Chernobyl NPP [1]. According to the experimental data, the concentration of tritium in natural water and atmospheric vapour just after the nuclear weapons testing has been set at about 5 to 40 Bq/L [2]. This value can be conventionally taken as the level of regional technogenic background. The figure exceeding this value will indicate an additional inflow of radionuclide into aquatic ecosystems from a certain local source.

The permissible concentration of tritium in drinking water in the official standardisation of our country at different times was varied for some period of time. For comparison, acceptable concentrations are often considerably lower than in Ukraine: the USA safe drinking water standard is 740 Bq/L; in Canada, the guideline reference level is 7,000 Bq/L; Member States in the European Union are required to decide whether remedial actions are needed if the concentration is above 100 Bq/L [4]. Obviously, this is due to the fact that the scientific community of these countries is aware of the radiation-and-ecological significance of tritium and puts it on a par with such a dangerous long lived radionuclide as plutonium.

It is common knowledge that Ukraine is heavily dependent on nuclear energy. In 2019, the Country's electricity production from nuclear power amounted to 83.0 TWh, representing more than half (53.9%) of the total electricity generation [5]. As of 1 January 2020, in Ukraine there are four operating nuclear power plants (NPP) with 15 nuclear energy units with the installed capacity of 13835 MW [6].

As is known, nuclear power plants are significant consumers of water for generating electricity and cooling their units. At the same time, Ukraine has a relatively low availability of internal water resources as compared to other countries in Europe. It ranks the 124th among 181 countries as to the amount of internal renewable water resources available per capita in 2014 [7]. Among 20 European countries, Ukraine is the 17th by the amount of total renewable water resources available per capita. A significant part of the country's territory has a low and very low water availability ($1.98 - 0.12 \times 10^3$ m^3 per capita). In dry years, the volume of surface water decreases to 1.2×10^3 m^3 per capita, which characterizes Ukraine, according to the UNESCO classification, as a water-insecure country.

Surface waters of the Dnieper river basin cool 12 nuclear energy units (Zaporizhzhya NPP, Khmelnytsky NPP and Rivne NPP), and 3 nuclear energy units (South-Ukraine NPP) are cooled by surface waters of the South Bug river basin.

Water resources of the Dnipro River basin provide approximately 65% of the drinking water supply (over 20 million people) [8]. The total area of the irrigated land at the edge of the Dnipro River Basin is 1960 km^2. The Southern Bug River basin provides about 8% of the population (4.2 million people) with water [9]. The total number of reservoirs within the Southern Bug river basin is 187, and there are almost 9877 of ponds with a total area of 854 km^2. Within the Dnipro River Basin, there are 504 reservoirs and 24634 of ponds (9201 km^2). The increased regulation of river basins where NPPs are located has led to a decrease in water exchange activity and, as a result, has increased the sensitivity to chemical constituents and the temperature of flow used to fill the cooling ponds of NPPs.

An additional factor influencing deterioration of tritium water indicators is the impact of water volume reduction including its evaporation from the surface of water bodies (Table 1).

TABLE 1. POTENTIALLY PREDICTED CONCENTRATION OF TRITIUM IN THE SURFACE WATER OF THE DNIPRO AND SOUTHERN BUG RIVER BASINS

NPPs/ number of nuclear energy units	NPPs electricity generation in 2019 (TWh)	River Basins	The average annual runoff (10^9 m^3/y)	Pond and reservoir area (km^2)	Potentially predicted concentration of tritium (Bq/L)	Possible increase in tritium concentration due to water evaporation, (%)
Rivne NPP/4 Zaporizhzhya NPP/ 6 Khmelnytsky NPP/ 2	19,1 38,4 7,5	Dnipro River basin	53,3	9201	40000	14
South-Ukraine NPP / 3	18	Southern Bug River basin	3,2	864	19000	23

Summing up, it should be emphasized that the problem of studying tritium migration in natural environment, especially in water bodies and in the areas of location of nuclear fuel cycle enterprises is not sufficiently studied. The long term biological effect of tritium small concentrations on a human body has also been poorly studied. In addition, the inflow of tritium into groundwater, primarily into the groundwater stratum, can be attributed to the factor of additional deterioration of environmental parameters of domestic drinking water supplies, given the interaction of groundwater and even confined aquifers.

A widespread distribution of land flooding sites within Ukraine (up to 25-30% of the area) may also, in our opinion, contribute to the vulnerability of the underground hydrosphere to tritium radionuclides. It should also be noted that information on the dynamics of tritium content in the water ecosystems throughout Ukraine is extremely limited.

Therefore there is a need in a systematic research of the impact of tritium realized by means of studying the tritium migration into the surface and ground waters as well as by monitoring the state environmental system of Ukraine not only in the areas of direct location of nuclear cycle facilities, but also all over Ukraine.

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84-THE EDUCATION AND CULTURE ABOUT RADIATION PROTECTION

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This paper takes the example of the radiation protection training of the ABC research institute. Explaining the contents and characteristics of the radiation protection in training planning, training courses designing, training practice, effects feedback and culture. Raising some suggestions for creating a good cultural environment to enhance the employee's radiation protection awareness and technology.

(a)Classifying training level of radiation protection and making the plan.

Generally, based on the staff's work experience and skills, radiation protection training is divided into three levels: elementary, middle and advanced. Every level clarifies the boundaries of training staff, training duration, training characteristics, and develops targeted training programs.

(b)Setting courses for each training level.

1. Elementary level focuses on basic knowledge, necessary protection skills, radiation zones and controlling, etc.

2. Middle level focuses on more specialized training.

3. In addition to enhancing the training of skills, advanced level is more inclining to the exchange of experience.

(c)The model of feedback and effects.

Every training should establish the training effect feedback mechanism, that will be better improve the training works. We set up a feedback table from four aspects: teachers' ability, teaching content, teaching emotion and the combination with practice.

(d)Safety culture in the radiation protection training.

Safety culture has great influence on radiation protection, the content of safety culture should be added to the courses of all levels of radiation protection training.

(e)Suggestions.

According the passage above, we put forward suggestions about course setting, training content and training frequency.

85-OPERATIONAL RADIATION PROTECTION IN NUCLEAR FUEL CYCLE FACILITIES: INNOVATION AND ADVANCEMENTS IN SAFETY ASPECTS

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Radiological safety in nuclear fuel cycle facilities is mainly ensured by area monitoring, detection and management of hot-spots, air activity monitoring, personal monitoring and monitoring of environmental discharges (aquatic as well as atmospheric) by the operational health physicists. Regulatory requirement and its adherence is the benchmark for radiation safety requirements in the operational radiation protection of nuclear fuel cycle facilities. The adoption and application of international recommendations towards safety (IAEA Safety Standards, GSR-Part3) and relevant safety publications of ICRP has helped in the implementation of safety standards and greater consistency between the arrangements for protection and safety globally. In India the requirements brought forward by the BSS is being adopted in all the safety and regulatory documents. In a global and long term perspective, the impact of the environment associated with the operation of nuclear facilities may persist for long periods of time. The need is to demonstrate that the environment is being protected against deleterious effects of such activities. For this, regulatory body shall determine which practices or sources within practices are to be exempted from regulations and approve which sources, including materials and objects, within practices may fall under regulatory control. Health Physics Instruments have undergone a paradigm shift in terms of availability of new detection systems and methodologies. The advent of various passive and active detector systems, optically stimulated luminescence (OSL) dosimetry systems, plastic scintillator based monitoring systems and advanced neutron spectrometry systems has helped in reduction of detection limits. The monitoring techniques using state-of-the art detectors are being employed in nuclear fuel cycle facilities in front and back end facilities in India for workplace monitoring to ensure compliance with regulatory limits and constraints. The paper discusses indigenous approach in developing few such monitoring systems / techniques with special emphasis on detection methodologies.

EXTERNAL AND INTERNAL DOSIMETRY:

The external and internal monitoring of workers for operation and maintenance activities are well designed in processing laboratories. The purpose of individual monitoring in general is to verify and document that each worker is protected adequately and therefore, it forms part of the overall radiation protection programme. In the external radiation dosimetry scenario, thermoluminescence dosimeter (TLD) is being practiced over several decades as an established method for personnel and environmental monitoring. However, in the recent past, OSL technique is being increasingly adopted for radiation dosimetry which includes personnel and environmental monitoring, medical dosimetry, space dosimetry, homeland security etc. In view of this, BARC has developed a cost effective alternate method for the large scale preparation of highly sensitive dosimetric grade α-Al_2O_3:C phosphor and OSL based dosimetry system [1] covering a useful dose range of 50 μGy to 10 Gy with minimum detection limit of ~ 50 μGy and a reproducibility of ±5 % for the OSLD badges. ICRP, in April 2011 issued a white paper on tissue reactions in which the equivalent dose limit for the lens of the eye has been reduced from 150 mSv to 20 mSv in a year, averaged over defined periods of 5 years, with the dose in no single year exceeding 50 mSv for all the occupational workers. Many sensitive eye lens dosimeters have been designed and reported [2] to ensure this dose limit for the eye lens. Normally, the eye lens dosimeter consists of a natural LiF based TLD sealed in a plastic holder. BARC has developed a new two element Eye Lens dosimeter [2] based on OSLD for the occupational workers in medical, industrial & nuclear facilities. This Eye lens dosimeter is capable of measuring doses from x-ray, beta and gamma radiation to the lens of eye in terms of Hp(3) in the energy range of 10 keV to 1.25 MeV for photons and 0.7 MeV to 3.54 MeV for beta radiation. In the field of internal dosimetry, BARC has developed a number of systems like portable whole body monitor (PWBM), portable thyroid monitors

(PTM) to cater to the requirement of emergency response, vehicle mountable gamma detection system for onsite screening of contaminated environmental, dietary and biological samples, quick scan whole body monitor for scanning of workers & members of the public for any radiation emergency. A family of BOMAB phantoms (adults and children) have been fabricated for age-dependent efficiency calibration of all these systems. Efforts were also made to scale down the ICRP Voxel phantom to Indian reference man dimensions for calculation of absorbed fractions. Implementation of the latest ICRP occupational intakes of radionuclide recommendations is also being worked out.

OPERATIONAL RADIATION PROTECTION:

Radiation protection aspects of front end nuclear fuel cycle facilities are associated with safe handling of large quantity of open low active sources such as natural U and Th ores. For nuclear fuel cycle facilities, front end refers to exploration, mining, processing of radioactive minerals and fuel fabrication facilities. Radioactive mineral exploration/ mining, processing and waste management radiation protection focuses on internal hazards due to inhalation of short-lived decay products of radon (^{222}Rn) like ^{218}Po, ^{214}Pb, ^{214}Bi, ^{214}Po and long lived alpha activity associated with respirable dust containing radionuclides U, ^{226}Ra. ^{230}Th, ^{210}Po. Gamma emitters (^{226}Ra, ^{214}Bi, ^{214}Pb etc) originating from various sources forms part of the external hazard in the facilities. Both active and passive techniques are used for monitoring and surveillance in the above facilities. In India, indigenously developed ZnS(Ag) based scintillation cell, personal radon dosimeters, TLD, personal air samplers, gamma survey meters and ALSCIN contamination monitors are used. Conventional and LED based Fluorimeters (for uranium) and emanometric set for ^{226}Ra/^{222}Rn are used for analysis of environmental samples.

In back end fuel cycle, processing of spent fuel poses challenge due to the handling of fission products. Radiation safety challenges during fuel processing are due to the dispersible forms of actinides and fission products with restrictive values of ALI and DACs. Radioactive waste management involves handling, treatment and disposal of various categories of wastes in solid and liquid forms. In waste management facilities the high active liquid waste handling requires special efforts especially shielding, assessing dose rate at full occupancy areas and also air contamination control. Radiological measurements such as radiation survey for gamma and neutron, air activity concentration and contamination levels are carried out at every stage of operation to assess overall radiological safety. Gaseous effluents generated from the reprocessing plant are discharged into the environment through stack after monitoring for particulate activity for gross alpha, gross beta and gaseous isotope ^{85}Kr as per the design intent. To address the issue of sampling and measurement of ^{3}H, ^{14}C and ^{129}I in gaseous effluents [3] which also may be released into the environment through stack, a comprehensive off gas sampling methodology is developed and installed in reprocessing plants for the estimation of these radionuclides in gaseous effluents. Multiple alpha counting systems has been developed for simultaneous counting of 9 filter paper samples for a given preset time. It has helped in reducing counting time drastically and thus saving man-hours.

Low Level Liquid Waste generated at Reprocessing Plants is discharged into the environment through authorized aquatic route. ^{99}Tc, present in liquid effluents, is a beta emitting long lived fission product and considered as one of the most important radionuclide from waste management point of view considering its environmental impact. Therefore a study has been undertaken to quantify ^{99}Tc in liquid effluent stream of reprocessing plants using a solvent extraction method followed by Liquid Scintillation counting technique [4]. Various liquid waste streams from reprocessing plant to waste management facility are transferred through pipe-in-pipe lines along the shielded concrete trench. Hence an Annulus Sampling System has been developed and installed to detect the leakage in waste transfer lines along the waste transfer trench.

To demonstrate that materials which are released from a nuclear facility meets the clearance criteria , a 3-AXIS mechanical manipulator based on computed tomography waste assay system is in operation. This is expected to give valuable inputs thus creating an extensive database for the proposed decommissioning facilities.

Protection factor (PF) of a respirator is a number that describes the effectiveness of various classes of respirators in providing protection against exposure to airborne contaminants including particulates, gases and vapours. The PF is derived from the ratio of the concentration of an airborne contaminant outside the respirator (Co) to the concentration inside the respirator (Ci). PF Test Facility for the estimation of PF for various respiratory protective equipment is developed and installed [5].

Advanced techniques and systems mentioned above and many more developed over the years for nuclear fuel cycle facilities have helped immensely in the effective implementation of radiation protection practices as well as in achieving ALARA in exposure control.

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86-IMPROVING RADIATION PROTECTION THROUGH ESTABLISHING A NATIONAL SECONDARY STANDARD DOSIMETRY LABORATORY IN MYANMAR

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BACKGROUND

The use of ionizing radiation and radiation sources in various fields of science and technology has existed since late 1950s in Myanmar. Accurate radiation dosimetry is required for different applications such as the optimization of radiation imaging for diagnostic radiology and nuclear medicine, and the radiation protection of staff, patients and the public.While currently some dosimeters are being sent to abroad for calibration and the rest have to be used without calibration, Division of Atomic Energy (DAE) is going to establish its own Secondary Standard Dosimetry Laboratory (SSDL) with the aid of IAEA to improve national capability for sustainable radiation safety infrastructure in the country.

CURRENT STATUS OF ESTABLISHING A NATIONAL SECONDARY STANDARD DOSIMETRY LABORATORY

As an establishment of a calibration facility for radiation measuring equipment is essential for ensuring confidence in the radiation dose for radiation protection, especially given to patients in radiotherapy and radio-diagnostics, a National SSDL under Division of Atomic Energy (DAE) has already been set up with the strong support by IAEA through the Technical Cooperation Projects since 2014. The SSDL Bunker has been designed with the help of Technical Officers from IAEA and technical expert from Department of Medical Sciences (DMSC), Thailand. It has already been constructed by Myanmar side and technical and required instruments have been provided by IAEA.

The SSDL occupies three separate calibration rooms, two of each are 11 m long and 6 m wide and the rest is 8 m long and 6 m wide and each with proper air conditioning. Their walls are concrete and the entrance doors are plated with lead to protect the control rooms and the surroundings against radiation. The first calibration room, will accommodate an X-ray unit (20-225 kV, 3200 Watt) with a 4 m long calibration bench, aperture wheel assembly designed to modify the spectral characteristics of the x-ray beam to meet the various beam codes requires for calibrating instruments, a set of filter assemblies to control beam definition according to ISO 4037, and a half-value layer kit for diagnostic radiology calibration purposes.

The second calibration room, which is adjacent the first, will accommodate a Cs-137 source (approximate activity of 740 GBq) with an ancillary equipment which will be provided by the Agency coming soon for radiation protection level calibration purposes. The last calibration room, which is adjacent the second, will accommodate a Co-60 (approximate activity of 300 TBq) calibration system for radio-therapy level calibration purposes which will also be provided during the IAEA TC project cycle 2020-2021.
Currently, Myanmar has already received an X-ray unit with ancillary equipment. Unfortunately, the installation time for the X-ray unit will be postponed at the convenient time because COVID-19 virus is spreading across the globe. In addition, IAEA will also provide a Cs-137 source with ancillary equipment soon. After installation of the equipment and proper training for operators by supplier, it is expected that the radiation protection and diagnostic radiology calibration services will be provided during this year. Moreover, the existing SSDL will be

upgraded to a radiation therapy level calibration facility during 2020-2021 with help of IAEA and the related calibration services will be provided accordingly.

EXPECTING ACTIVITY OF SSDL IN FUTURE

The SSDL, Myanmar together with regulatory authority must expand its activity to assure the safe handling of radiation sources. As the correct reading of radiation monitoring instruments through calibration is highly important, the SSDL will provide the following radiation dosimetry services:

- Providing the calibration services for radiation dosimetry in X and Gamma radiations and issuing calibration certificates with all necessary information, including the estimated uncertainties
- Improving the accuracy of radiation dosimetry in provided calibration services to reduce unwanted risks of radiation
- Checking the quality of Radiation equipment and suggestions can be given about these instrument
- Developing human resources of radiation physicist of SSDL
- Saving time and cost for customers
- Improving income by dosimetry services

Moreover, Myanmar will be trying to implement a proper quality management system to get ISO/IEC 17025(2017) accreditation and participate in the IAEA/WHO Network, ASEAN-SSDL Network and with other radiation metrological laboratories in the exchange of information and improvement of measurement instruments and techniques.

CONCLUSION

As accurate dose measurement of the Radiation Monitoring Instruments is essential for the radiation protection of workers, people and the environment, these instruments will be properly calibrated by SSDL to provide the necessary confidence to the users of radiation sources in assessing the hazard potential. Hence it will improve radiation protection for the people within the country, Myanmar.

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88-EASY PROCEDURES TO REDUCE ORGAN DOSES IN HEAD CT SCANS

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INTRODUCTION

Computed Tomography (CT) has been one of the most used exam for radiologic diagnostic in medicine. The increase of CT is a global concern due to high doses of radiation [1]. The head CT scans are commonly used for diagnosis of traumatic head injuries, infections and other diseases with instability, so it can be associated with a high radiation dose to organs such as lenses, parotid gland, and hypophysis, when compared with conventional radiology [2]. The lens is one of the organs located in the skull with relative high radiosensitivity and received the X ray primary beam during the head CT scans [3]. The main objective of this study was to analyse the reduction of the absorbed dose in head CT scan with the use of bismuth shielding and with the head tilted.

METHODS

The experiments were conducted using a GE CT scanner LightSpeed VCT model, with 64 channels. An anthropomorphic male phantom model CIRS ATOM 701 was used to perform head scans from the cervical vertebra C1 to the top of skull. The scan parameters used in this experiment were 120 kV, 175 mA, 0.8 s and 0.984 of pitch. The distance used for head CT scans with and without bismuth shielding were 150 mm and with the head tilted 120 mm. Three head CT scans were conducted with the phantom in supine position as are shown in the Fig. 1.

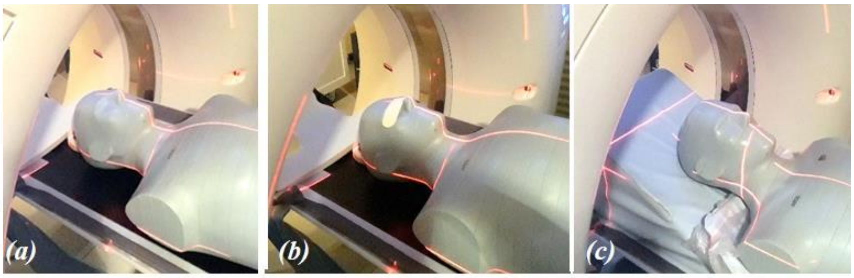

FIG. 7. Positioning of the phantom in the gantry: without bismuth shield (a), with bismuth shield (b) and with the head tilted (c).

Punctual dose measurements have been performed using GAFCHROMIC XR-CT radiochromic film strips to record the individual doses in the organs of interest such as lenses, thyroid, pharynx, hypophysis, spinal cord, breasts, parotids and salivary gland. To validate the quality of the image by CT, it was done the noise analysis from the image of the central slice of the phantom head in each of the experiments to determine the influence of the secondary radiation in the quality of the images.

RESULTS

The results of this experiment show absorbed doses from 0.62 to 24.06 mGy and they are shown in Table 1. The highest recorded dose occurred in the eye lens that stressed the situation of unnecessary radiation exposure. The influence of the bismuth shielding and the tilted positioning of the head led to a decrease in the absorbed dose in eye lens of 15.06 mGy and 12.41 mGy, respectively. The lenses absorbed doses were 37.40% smaller with the use of bismuth shielding and 48.42% with the head tilted. In organs such as lenses, pharynx, parotid glands, and hypophysis, the influence of the head tilted in absorbed dose rates is better compared with the use of bismuth shielding. The analysis of noise in the image of the head central slice presented acceptable values for soft tissues, less than 1%.

TABLE 1. Average absorbed dose in organs during head CT scans with and without eye bismuth shielding and head tilted.

Organ position	Average absorbed dose (mGy)		
	Without bismuth shielding	With bismuth shielding	Head tilted
Eye Lens	24.06 ± 0.69*	15.06 ± 0.71	12.41 ± 0.67
Hypophysis	13.54 ± 0.0.71	11.47 ± 0.57	10.41 ± 0,77
Pharynx	6.22 ± 0.58	4.86 ± 0,85	4.12 ± 0,64
Spinal Cord	2.52 ± 0.42	2.11 ± 0.44	2.03 ± 0.38
Parotid Gland	20.94 ± 0,64	14.05 ± 0,65	8.16 ± 0.61
Salivary Gland	7.58 ± 0.66	4.89 ± 0.93	2.30 ± 0.49
Thyroid	2.66 ± 0.45	2.05 ± 0.43	1.57 ± 0.41
Breast	1.15 ± 0.34	0.89 ± 0.37	0.62 ± 0,43

*Standard deviation.

CONCLUSIONS

Absorbed doses were evaluated during head CT scans with and without eye bismuth shielding and with the head tilted. Dose values were reduced in all organs studied, suggesting that the use of bismuth shielding or the head tilted promotes reduction in organ dose, but with the head tilted it is a better method since you don't need to use any device just need to change the head position of the patient.

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89-EMERGENCY EXPOSURE OF THE PUBLIC AND WORKERS DUE TO A HYPOTHETICAL PARTIAL CORE MELT ACCIDENT IN TEHRAN RESEARCH REACTOR

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BACKGROUND

Determining the precautionary action and urgent protective zones (PAZ and UPZ) in nuclear facilities is of great important in nuclear and radiological accidents. The worst possible case of nuclear accident for a facility may be assumed to determine the boundaries of these areas using simulation methods. Performing this simulation during the emergency preparedness phase can be effective in speeding up decision making when actual accidents occur and taking required actions such as evacuating, sheltering, restricting food and water consumption, and distributing iodine tablets.

According to the IAEA guidelines, the PAZ is not defined, and the UPZ is typically set around 500 meters for 2-10 MWt research reactors [1]. Moreover, these areas should be determined more precisely for each reactor depending on its design and other special conditions. In this study, the total effective doses as well as the dose equivalent of radio- iodine uptake in thyroid are investigated after a hypothetical accident of 10% core melting of the Tehran Research Reactor (TRR).

METHODS

The Tehran research reactor with a power of 5 MWt is located in the centre of the capital of Iran. According to the Safety Analysis Report (SAR) of this reactor [2], the worst case scenario for a probable nuclear accident is 10% core melting due to a reactivity shock about 2.5$ after a fuel rod loading accident. The Hot-Spot software along with the total effective dose equivalent (TEDE) quantity were used for the dose calculation [3]. The weather conditions were postulated to be night time, low rainfall and wind speed of 0.5 m/s. Other simulation assumptions are presented in Table 1. The current reactor source-term along with the assumption of a one month period that the reactor was in on-operation mode were considered in the simulation. The external dose difference of the thymus and thyroid glands was used to estimate the dose due to the absorption of radioiodine in the thyroid.

RESULTS

Table 2 presents the results of total effective doses in the early hours as well as on the 7th and 30th days after the accident. As can be seen, at the 300 meters distance of the reactor and assuming public permanent residence, the integrated TEDE was estimated to be 68 mSv within 7 days after the accident which is less than the corresponding level of 100 mSv for any urgent protective actions recommended by the IAEA [4]. Also, the dose of radio-iodine uptake at this distance and time is estimated to be 10 mSv, which is well below the corresponding 50 mSv level for iodine prophylaxis recommended by the IAEA (table 3).

It is not possible to estimate the annual dose by the software. On the other hand, the half-life of the majority of the released decay products often are not so long, and limited to several days. Consequently, given the 30-days effective dose of 86 mSv at the distance of 300 meters, the annual effective dose is expected to be less than the100 mSv (the recommended level by IAEA for the protective actions) at the distance. Also, given the 7-day dose data, it is reasonable to predict a 200-meter radius to evacuate staff around the reactor, either in exercises or actual accident

CONCLUSION

According to the obtained results, assuming the worst-case scenario in the Tehran research reactor, the total effective and thyroid dose of inhabitants around the reactor are below the levels for any protective actions recommended by the IAEA. The reactor is located in the centre of the city and the drinking water sources are also far away from the accident area, thus there is no need to restrict the use of agricultural products, food and drinking water in the related nuclear accident.

TABLE 1. SOME BASIC ASSUMPTIONS IN NUCLEAR ACCIDENT SIMULATION FOR TRR

Assumption	value
Reactor Power	5 MWt
Stack diameter & height	2.5 & 57.0 m
Average air temperature in summer	30 ºC
Wind speed	0.5 m/s
Flow of main exhaust stack	572 m^3/min
Breathing rate (adult, child, infant)	7300, 5500, 1900 m^3/year
Leakage rate of iodine and noble gases from cladding fuel to the reactor pool	27%, 100%
Leakage rate of iodine and noble gases from reactor pool to the dome	5.0%, 40%
Leakage rate of cesium and tellurium from cladding fuel to the reactor pool	27%, 0.01%
Leakage rate of rubidium, strontium and barium from cladding fuel to the reactor pool and from pool to the dome	3%, 0.01%
Leakage rate of other radionuclides from cladding fuel to the reactor pool and from pool to the dome	0.1%, 0.01%
Leakage rate of iodine, noble gases and fission products from reactor dome to the outside	1.5%, 100%, 1%

TABLE 2. ESTIMATION OF TEDE AFTER A 10% CORE MELTING ACCIDENT AT TRR

Distance(m)	TEDE (Sv) in various exposure duration			
	1h	2h	7 days	30 days
10	1.6E-1	2.8E-1	NA	NA
30	5.1E-2	9.0E-2		
50	3.0E-2	5.4E-2		
80	1.9E-2	3.3E-2		
100	1.5E-2	2.6E-2		
150	9.7E-3	1.7E-2		
200	7.1E-3	1.3E-2	1.0E-2	1.4E-1
300	4.6E-3	8.2E-3	6.8E-2	8.6E-2

NA: Not applicable because there is no possible for permanent residency within these zones

TABLE 3. DOSE ASSESSMENT OF SOME CRITICAL ORGANS AT 300 m DISTANCE FROM THE TRR AFTER 7 DAYS OF A 10% CORE MELTING ACCIDENT

Organ	Red Bone	Stomach	Thyroid	Thymus	Lung	Skin	Breast
Dose(Sv)	5.2E-2	5.3E-2	6.4E-2	5.6E-2	5.8E-2	2.1E-1	7.9E-2

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91-OPTIMIZATION OF PROTECTION IN VARIOUS NUCLEAR/RADIOLOGICAL RESEARCH PRACTICES IN IRAN

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BACKGROUND

Optimization of radiation protection in all radiological/nuclear activities is one of the important issues to consider [1]. Nuclear and radiological research covers a wide range of activities in industry and medicine. In this study, the occupational exposure of number of staff who works in a variety of radiological/nuclear research activities along with the abnormal radiation exposure from 2015 to 2019 in Iran are investigated.

METHODS

Occupational exposures were investigated in radiological/nuclear practices included the research reactors, nuclear materials (uranium ore processing and condensation), radiation application (high dose irradiation, production of radiopharmaceuticals, and radioisotopes, neutron activation analysis, calibration and dosimetry), accelerator facilities (charge particle accelerators, ion implantation, and neutron physics), plasma and nuclear fusion (magnetic confinement and plasma focus), and nuclear agriculture.
Personal monitoring was performed on bimonthly dosimetry periods using film dosemeters. The solid state nuclear track dosimeters (SSNTDs) or TLDs were used for the staffs who may be exposed by the thermal and fast neutron fields. The annual and 5-year dose limits recommended by ICRP-103 [2] as well as bimonthly investigation level of 1.1 mSv are considered in the personal monitoring program.

RESULTS

Table 1 presents the results of the individual dose of the workers. As can be seen, among the types of research practices, the highest occupational exposure was primarily for the workers in research reactors, and then for the others particularly those of dealing with the open radiation sources in some way. Occupational exposures of the workers in plasma and nuclear fusion and nuclear agriculture practices were calculated around zero. The data in the table also show that there has been a decreasing trend in occupational dose over the last five years. This decrease can be due to a gradual improvement in optimization in radiation protection, the need for halt in some activities for several months, such as equipment upgrades or overhauls at the facility, and work-to-stop policies in some case of research activities. Eight cases of unusual exposures (radiation exposure exceeding the investigation level) in the last five years were related to work in research reactors (Table 2). Reasons for the unusual exposures were somehow similar to those of in other radiation practices [3].

CONCLUSION

According to the results obtained, optimization of radiation protection in activities related to nuclear and radiological research is desirable. This optimization is due to establishing the; control system of work-place condition of the practices, radiation protection trainings, regular personal monitoring, regular cooperation with the national regulatory body, regular medical tests and periodic examinations of the workers, coherent network of health physicists, and quality management system in documentation.

TABLE 1. OCCUPATIONAL EXPOSURE IN NUCLEAR/RADIOLOGICAL RESEARCH ACTIVITIES IN IRAN

No.	Type of Practice	Number of workers	Averaged annual dose per year (mSv)					Averaged of annual dose (mSv)	Averaged 5-year dose (mSv)
			2015	2016	2017	2018	2019		
1	Research reactors	145	0.28	0.17	0.39	0.11	0.06	0.20	1.00
2	Radiological and medical application	166	0.12	0.16	0.09	0.05	0.05	0.09	0.46
3	Nuclear materials	190	0.08	0.03	0.02	0.04	0.00	0.03	0.17
4	Research accelerators & neutron physics	50	0.17	0.12	0.05	0.06	0.02	0.08	0.40
5	Plasma physics & nuclear fusion	65	0.00	0.00	0.00	0.00	0.02	0.00	0.04
6	Nuclear agriculture	10	0.00	0.00	0.00	0.00	0.00	0.00	0.00

TABLE 2. UNUSUAL EXPOSURES IN RESEARCH ACTIVITIES IN IRAN

Worker	Type of work	Bimonthly dose (mSv)	Reason
#1	Operator of waste in reactor	1.48	Waste displacement
#2	Reactor operator	1.15	High workload
#3	Reactor technician	2.21	High workload
#4	Reactor health physicist	1.36	Reactor overhauls
#5	Reactor technician of health physics	1.37	Decontamination
#6	Reactor operation manager	1.17	Fuel displacement
#7	Operator of waste in reactor	2.39	Decontamination
#8	Reactor shift supervisor	2.76	High workload

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92-ASSESSMENT OF THE EYE LENS DOSE FOR CLINICAL STAFF IN INTERVENTIONAL RADIOLOGY IN IRAN

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4. BAKGROUND

Staff at interventional radiology and cardiovascular angiography centers may receive significant doses because of their proximity to the X-ray source during treatment. Although the lead aprons cover most of the body trunk, the radiation protection of eye lens and exposed area of the body are of great important in this situation. The ICRP has recently lowered the annual dose limit of eye lens of workers to 20 mSv [1].

5. MATERIAL & METHODS

In this study, the eye lens, skin and total effective doses of 38 cardiac angiography staff from three selected Tehran University of Medical Sciences hospitals were estimated within four months using personal dosimeters. The angiography system model of Artis Zee (Siemens, Germany), 80 kVp, are used in all the three hospitals. Two TLD pellets (GR-200, China) next to the left and right eye lens (not covered by lead glasses) as well as one at the middle point of forehead were used to estimate the eye lens dose. Moreover, a TLD card [contains two TLD-100 chips (Harshaw, USA), one for depth and the other for skin dose] over the apron, and a film dosimeter under the apron, were used for measuring the total effective dose (Fig. 1). The double-dosimetry method was used for calculation of total effective dose, E, as follows [2]:

For physicians and residents: $$E = 2.25\, H_P(10)_{under} + 0.075 H_P(10)_{over} \quad (1)$$

For nurses and radiotechnologists: $$E = 2.25\, H_P(10)_{under} + 0.12 H_P(10)_{over} \quad (2)$$

3. RESULTS

Table 1 presents the monthly value of eye lens, skin and total effective dose of physicians, residents, nurses and radiologists of the selected hospitals. As it is shown, all their corresponding annual dose values are calculated to be less than the dose limits recommended by ICRP 107 and 118 [1][3]. The occupational exposures of the physicians and nurses are greater than those of residents and radiologists because of more proximity of their locations to radiation sources and/or higher workloads.

4. CONCLUSION

Since all the selected hospitals have a high daily workload, by the obtained results, it is expected that the occupational exposure in interventional radiology practice would be acceptable in Iran. However, more extent researches must be conducted to ensure the optimization of protection are met in this practice.

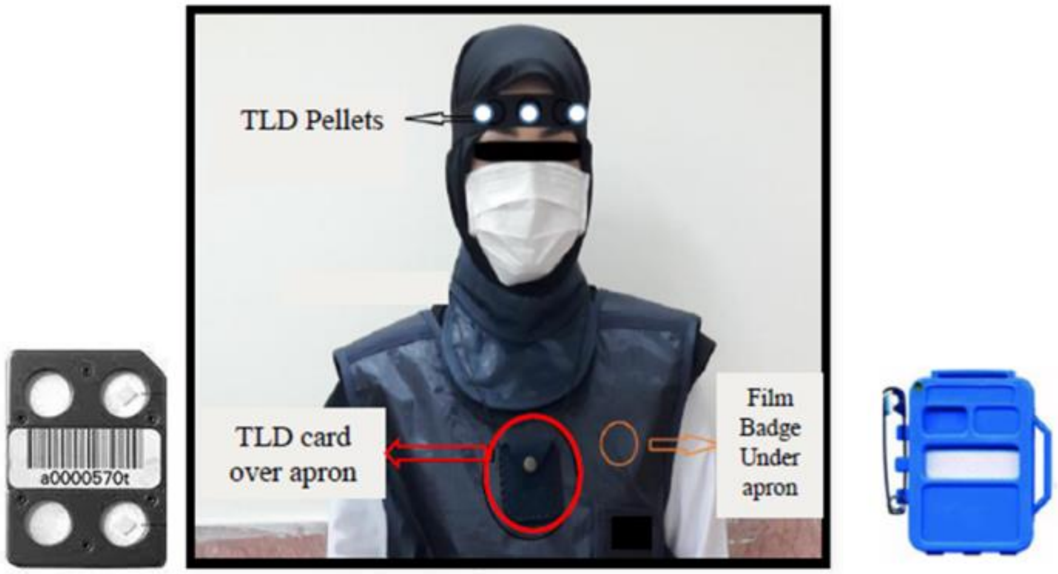

FIG. 8. A worker of interventional radiology equipped with several dosimeters

TABLE 1. OCCUPATIONAL EXPOSURE OF STAFF IN INTERVENTIONAL RADIOLOGY

Type of work	number of staff	Forehead dose, $H_p(0.07)$ (mSv/month)			Skin dose, $H_p(0.07)$ (mSv/month)	Dose over apron	Dose under apron	Total dose, $H_p(10)$ (mSv/month)
		Left lens	Middle point	Right lens				
Physician	11	0.28	0.21	0.26	0.19	0.21	0.07	0.173
Resident	11	0.15	0.16	0.18	0.18	0.18	0.01	0.036
Nurse	12	0.35	0.41	0.38	0.31	0.31	0.05	0.150
Radiotechnologist	4	0.09	0.07	0.08	0.03	0.02	0.01	0.025

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94- ANALYTICAL APPROACH TO ACCOMPANY THE DEVELOPMENT OF NATIONAL DIAGNOSTIC REFERENCE LEVELS

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At the request of the Government of the Republic of Cameroon, an international team of senior safety experts met representatives of the National Radiation Protection Agency (NRPA) from 12th to 21st October 2014 to conduct an Integrated Regulatory Review Service (IRRS) mission. The review compared the Cameroon regulatory framework for radiation safety against the IAEA Safety Standards as the international benchmark for safety. The mission was also used as an opportunity to exchange information and experiences between the IRRS review team and the National Radiation Protection Agency (NRPA) counterparts in the areas covered by the IRRS. One of the recommendations was that the government of Cameroon should revise ministerial order number 1152 and replace the maximal skin entry doses with diagnostic reference levels. NRPA should also define how to use these values as optimization tools and control the implementation of this requirement [1].

NRPA decided to support health professionals in establishing National Diagnostic Reference Levels (NDRL) of Cameroon with the aim of implementing the requirements of IAEA publication GSR-part 3 paragraphs 3.148 and 3.169 [2]. This was established with the help of regulatory inspections findings by NRPA of non-systematic determination of patient doses (audit) and lack of quality assurance at imaging centres. This was also to help avoid excessive radiation dose to the patient that does not contribute additional clinical information value to the medical imaging task from the most frequently performed radiological examinations. The role of NRPA was to assist in the collect, control and analysis of national data which has to be published periodical in the Ministry of Health Order [3]. A national ad hoc committee composed of ten staff from NRPA, Professional Radiological Associations and Ministry of Public Health was established in 2015 [4]. A policy document with various protocols was produced by this committee to investigate different radiological examinations. This policy defines the target population; the beneficiaries of the project, the objective of the activities to be carried out, the work methodology and the action plan of activities for carrying out NDRLs. Surveys were focused on specific diagnostic practices with graded approach. For example, in the case of Computed Tomography (CT), the methodology adopted by the working committee consisted of five (5) main phases: Preparation phase, Training and Data Collection phase, Analysis phase, Adoption phase and Vulgarization phase. The 75th percentile method was adopted for the calculation of NDRLs. The sample data of over 5000 patients for each protocol included a minimum of 10 patients of average body weight (mean [± SD], 70 ± 20 kg) were collected from 20 facilities. These were used to determine the NDRLs of the 11 most common and / or most irradiating examinations selected for each protocol. Training was done at the 20 selected centres on how data was to be collected on the spot with distributed forms. Quality control compliance tests were carried out on each identified machine before the collection of data was done. Statistical Package for Social Science (SPSS) Software Version 21.0 and Microsoft Excel Software were both used to do the calculations and comparative analysis. For each examination, the NDRL with 75th percentile was determined. SPSS data analysis software identified and eliminated devices providing constant dose values and exams with a small number of patients who could not express variability. Also, special sample of some specific examinations were randomized and blinded for review by experienced radiologists who graded diagnostic image quality.

Each Centre was entitled to an Excel Sheet file with her results as how NDRLs were calculated as the 75th percentile of patient dose distributions from their imaging Centres. An enacted revised national regulation was then adopted for all radiological centres to respect these NDRLs. This respected the recommendations of the IAEA IRRS mission recommendation. Many centres have changed their machines to digital due to the fact that NDRLs exposed some of their analogue machines' quality. This has enhanced patient protections. Clinical guidance and procedures have been revised to make sure that Local DRLs are in compliance with NDRLs.

The values obtained did not differ significantly from the values used in many developed countries. The use of the specialized data analysis software SPSS made it possible to control and improve the results obtained by the Excel spreadsheet which is much easier to use, the results thus obtained are more reliable and precise. The differences between the mean values of the dose distributions from each practice as per the policy were statistically significant ($p < 0.05$) for all examinations. Analysis of image quality revealed some statistically significant differences in scoring categories by some radiologists.

Establishing NDRLs allows an effective optimization (dose audit which could go hand in hand with image quality audit) in patient dose without resulting in degradation of image quality and saving machine live-spans. Implementation of national DRLs is to be followed closely by the competent authorities (NRPA and Ministry of Public Health) using the enacted regulations. Dose audit is being encouraged to be used in the centres in MS Excel in order to meet up with the compliance imposed by the regulatory requirements.

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95-THE IMPLEMENTATION OF SAFETY- SECURITY – SAFEGUARD INSPECTORS'S GUIDE IN CAMEROON: THE BENEFITS OF THE INTEGRATED APPROACH FOR RADIATION PROTECTION

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The aspect of Regulatory inspections in line with international standards are cited in Sections 30, 78 and 82 of Law N° 2019/012 of 19th July 2019 to Lay Down the General Framework for Radiological and Nuclear Safety, Nuclear Security, Civil Liability and Safeguards Enforcement. An integrated, national approach to Capacity Building and Human Resource Development is essential for establishing Regulatory Bodies in Non-Nuclear Countries. The mission of a Regulatory Body, as National Radiation Protection Agency (NRPA), is to Protect People, Society and the Environment which has no boundary whether in a nuclear state or not. There are about 500 nuclear and ionizing radiation technology applications in Cameroon. The role of the government is important as without its support organizational capacity will be affected. We carried out yearly Self-Assessment of capacity building activities in order to determine the gaps in capacity building by establishing a knowledge management plan for our inspectors. We were inspired through a National Training Course that was done by IAEA to encourage us to undertake Systematic Assessment of Regulatory Competence Needs (SARCoN) [1]. At NRPA, Knowledge management has been a journey since 2015 to serve at National, organizational, individual and products levels.

An Inspector's Regulatory Guide was enacted to detail inspection method, handling of inspection item, inspection record and report for effectiveness in meeting the requirements in regulations and International Atomic Energy Agency (IAEA) standards, especially the International Basic Safety Standards [2]. Inspection is carried out by examination, observation, measurement, or performance test undertaken by or on behalf of NRPA to assess structures, systems, components, and materials, as well as operational activities, processes, procedures and personnel competence by the inspectors during any stage of the regulatory process to ensure compliance of a facility or activity to regulatory requirements [3]. Inspections are combined with a capability to enforce the results of the inspection effort. The guide has content for stating two categories of inspections, prescriptive based and performance based, describe the approach taken when conducting the inspection. It illustrates on the focus area of the inspection (safeguards, security, and safety) and the scope of the inspection, which is a specific element of the safeguards and security program, commonly referred to as limited scope, or the entire program, commonly referred to as full scope. A balanced inspection approach in conducting inspections is the goal of all inspection programs at NRPA. In the guide, each type of inspection is illustrated as:

- Initial/preoperational – prior to granting the license or receiving ionizing radiation onsite.
- Periodic – generally defined as routine inspections that are based on the category of radioactive/nuclear sources and complexity of the site/facility Safety, safeguards and security program.
- For cause – due to a previously identified Safety deficiency or safeguards/security event.
- Termination – due to a request by the licensee to terminate the license after all ionizing radiation sources has been removed from the site/facility.

Three phases of an inspection are described in the inspector's guide. The first is the planning phase, which is important to verify if there are any special requirements that must be addressed prior to the inspection to ensure access to the facility. This includes access to necessary records and reports, availability of personnel, safety and security training required to access the areas of the site/facility, impact on operational activities, and ability to support performance tests, if needed. The second phase of the inspection involves collecting the data using four primary data collection techniques: document review, interviews, observation and performance testing. Validation of the information collected is also important to be sure that all parties understand the deficiency revealed and the basis of the deficiency. This gives both the inspection team and the facility operator the opportunity to be sure all information collected by the inspectors is accurate and understood. The output from this phase is a preliminary report that can be shared within the inspection team for concurrence of the initial inspection findings. The final phase of the inspection described in the guide involves developing a final draft of the inspection report. This phase also includes formally communicating the inspection results to site/facility management. This phase includes the formal notification regarding the sanctions that may be enforced. It also includes a request for information regarding the cause of the deficiency and plan of action to correct the problem (commonly referred to as a corrective action plan). This phase of the inspection is not completed until NRPA can verify all corrective actions have resolved the cause of the problem. It may include a final follow-up inspection to close out the inspection.

At NRPA, the training of new inspectors is defined in internal processes. The training of the future inspectors starts with the inspector's guide which includes the following steps: Training of the legal bases; Training of the inspection process (planning, preparation, execution, reporting); Professional training; Training of safety and security culture in the nuclear field (integrated safety and security culture); Radiation protection for self-protection; Negotiation and interviewing techniques; Organization of a nuclear facility; Technology of nuclear and ionizing radiation technologies locally available; Plant tours and visits of the safety and security -relevant systems; Participation in inspections as a silent observer; Supervised leading of inspections and Examination inspections.

Through integrated training, the nuclear security inspectors also gained a good and comprehensive understanding of the systems important to safety in a facility. These systems are the ones that nuclear security is protecting against sabotage, but to which even the nuclear security inspectors rarely have access. This knowledge was judged to be important not only for inspections, but also for all other supervisory activities, in particular for assessments and the issuing of authorizations for facility modifications. An important aspect is the training of the integrated safety and security culture. It is also important for the supervising authority to consider the sometimes-conflicting requirements in the areas of nuclear safety and nuclear security when assessing inspection items.

Development of Inspector's Guide and putting in place a knowledge management training program for inspectors for safety- security -safeguards interface is a decision where both safety and security issues are taken into consideration for the radiation protection of the population and its environment. This is an advantage for a young regulatory body to focus on the training of inspectors of both nuclear safety and nuclear security. Important prerequisites are unified inspection processes for both nuclear safety and nuclear security. Experienced inspectors support new inspectors and accompany them during on-site inspections. Both established concepts of guide and training using IAEA international standards benefit NRPA to harmonize effective inspection coverage in Regional Offices. Management change and improvement requires good communication and the willingness to understand each other through guide/procedures and trainings.

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96-STRENGTHENING THE RADIATION PROTECTION AND SAFETY OPTIMIZATION FOR WORKER OF HEALTH FACILITIES IN INDONESIA TROUGH IMPLEMENTATION OF DOSE CONSTRAINT AT OPERATIONAL STAGE

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The operational stage of dose constraint for radiation workers is the dose value that can indicate the implementation level for radiation protection optimization of occupational exposure in a facility. BAPETEN has regulated that the facility should implement the optimization of radiation protection and safety through the determination of dose constraint for radiation workers, which are determined at the construction stage and/or operational stage [1]. The dose constraint for the construction stage has been set by regulation, while the dose constraint for the operational stage must be determined by each facility. However, based on the results of surveys for health facilities in 2015-2017 it was found that almost 90% of respondents have not determined and implemented the concept of dose constraint for radiation workers at the operational stage, which is caused by the lack of awareness of the importance of dose constraint's role as an indicator of action for optimization efforts and the unavailability of a method for determining dose constraint that can be applied for various situations and characteristics of health facilities [2], [3], [4].

A descriptive-analytical study was carried out to provide recommendations on the method of determining dose constraints in various health facility conditions and the methods for the implementation. The study was conducted on the technical aspects of the annual dose of the radiation worker based on the operating period of the facility, and the optimization elements that could be subjected to a dose constraint review.

The method for determining the dose constraint will be done through a conservative approach which is based on the annual dose of radiation workers. Facilities are required to monitor the dose of radiation workers regularly every 3 months, thus it is assumed that each facility has a complete record of worker dose data since the facility has been operating for a minimum of one year. With this assumption, it is proposed 3 scenarios of determining dose constraint based on the facility's operating period category, as presented in Table 1. The category of an operating period is determined since the facility was operating with the first modality used. [5].

TABLE 2. SCENARIOS OF DETERMINING THE DOSE CONSTRAINT BASED ON THE OPERATING PERIOD OF A FACILITY

	Dose constraint determination
Operating period 1 (0 – 2 year)	the dose constraint is determined according to the dose constraint at the facility construction stage which is equal to ½ the dose limit value.
Operating period 2 (2 – 4 year)	It is assumed that the facility has at least 2-3 groups of annual dose data for workers so that it does not yet represent the trend of receiving doses statistically. The dose constraint can be determined based on the maximum annual dose value received by each type of radiation worker profession.
Operating period 3 (> 4 year)	It is assumed that the facility has at least 4 groups of annual dose data for workers so that it is sufficient to represent the statistical trend of receiving doses. The dose constraint for each profession can use the 3rd quartile calculation method of the radiation worker dose distribution.

However, several conditions need to be considered, for example, related to differences for annual doses of workers between professions and data on annual doses of workers which almost all show zero or less than 1 mSv/year. If the data shows that the difference of annual dose between any types of professions is not significant then the dose constraint can be uniformly determined for all types of professions in the facility. If there is a condition that almost all worker's annual doses show a value of zero or less than 1 mSv/year, it is recommended that the facility set a dose constraint value at 1 mSv.

In the facilities that have determined the numerical value of dose constraint, the implementation of the dose constraint should be monitored and reviewed periodically by considering the significance level of occupational exposure risk and the trend of receiving radiation dose from year to year. The monitoring is recommended to be carried annually by comparing the annual worker dose with the dose constraint value. The dose constraint review is recommended to be carried every 5 years, by analyzing trends of dose distribution in the last 5 years and the last 10 years. By comparing these data, it can be suggested to change the dose constraint value by considering several aspects.

If the monitoring and review results show that the worker's dose tends to exceed the dose constraint which has been set, then it needs to analyze the root cause. The analysis carried out, in this case, will reflect the review of optimization implementation of occupational exposure radiation protection, therefore it can be approached from a management perspective and specific technical operational perspective [5]–[7].

Management perspective is focused on evaluating the commitment, policy, and management's role in terms of fulfilling the principle of radiation protection optimization, which include elements such as establishing the Radiation Protection and Safety Program, providing competent and qualified radiation workers, implementation of effective communication, improvement of radiation workers awareness and involvement, providing reliable radiation protection equipment. Specific operational technical perspectives are focused on evaluating the fulfillment of radiation protection optimization principles in routine operational practices at the facility, which include elements such as quality control of radiation protection equipment, arrangement for work scheduling/assignment system, reduction of the dose rate received by workers, controlling the duration of work time in the radiation area, monitoring radiation levels in the work area [7]–[10].

Thus it can be concluded that the determination of the dose constraint for the operational stage can be done with a conservative approach based on the annual dose of radiation workers. The method of determining the dose constraint can use a scenario that considers the operating period of the facility. Data on the annual dose of workers can provide dynamic results depending on the situation and condition of the facility, in certain conditions a cut-off method is needed to determine the dose constraint as 1 mSv/year. Implementation of the dose constraint for the operational stage must be regularly monitored annually and reviewed every 5 years. The parameter used in the dose constraint review can be approached by management perspectives and specific technical operational perspectives. Through the review results using those approaches, it will be easily identified the parts that need improvement so that radiation protection and safety for workers will be optimized.

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97-AN OVERVIEW OF RADIATION PROTECTION PRACTICES IN MEDICINE IN BANGLADESH

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History of Using Radiation in Medicine in Bangladesh

The uses of radiation in medicine were originated in Bangladesh at 1954 when the first X-Ray machine was established in Kumudini hospital. The first Radiotherapy department was set up at 1957 in the same hospital and in 1960 the first Nuclear Medicine Centre was established in Dhaka Medical College Hospital (DMCH). First Gamma Camera was installed at National Institute of Nuclear Medicine & Allied Sciences (NINMAS) in 1983. The first PET-CT was installed in the United hospital, Dhaka in 2011. At present fifteen nuclear medicine establishments expanded all over the country which are administrated by Bangladesh Atomic Energy Commission (BAEC). Another five nuclear medicine department was operated by full faced hospital in private sector. Under these facilities, 60 Gamma Camera/SPECTs and 8 PET/CT were installed. Nuclear medicine provides both diagnostic and therapeutic services in Bangladesh. There are 28 Radiotherapy centres in our country where 22 linear accelerators, 10 CT-Simulators, 10 Tele-Cobalt machines and 14 Brachytherapy units have been equipped. Around 5500 X-ray unit and 200 CT have been installed in our country so far. The first medical cyclotron was installed in 2011 at the United Hospital, Dhaka. This cyclotron produced PET tracers such as F-18. Another one cyclotron is now under commissioning in NINMAS campus and it is expected that the cyclotron will be going for production very soon.

Background of Radiation Protection Practices in Medicine

According to the Bangladesh Atomic Energy Regulatory (BAER) Act-2012 and Nuclear Safety and Radiation Control (NSRC) Rules-1997 the responsibility of implementing and enforcing the Act and Rules is bestowed upon Bangladesh Atomic Energy Regulatory Authority (BAERA) since 12th February 2013 as per section 4 of the Act. Before establishing BAERA, Nuclear Safety and Radiation Control Division (NSRCD) under Bangladesh Atomic Energy Commission (BAEC) performed all regulatory activities in the country. As requirement of issuing or renew license, all departments have to appoint a licensed Radiation Control Officer (RCO). RCO should be a medical physicist.

Major Responsibilities of RCO

RCO is the main responsible person for radiation safety of the patients, staff and public. RCO have to monitor radiation dose level at all area of the work place. He also monitor the dose receiving of personnel, process for issuance of TLD badges for the radiation worker and record all dose data, maintain calibration data of radiation motoring equipment, maintain contamination monitoring data. RCO is responsible for supervise the radioactive waste management procedures, quality control and maintenance of imaging equipment. All SOP's related to radiation have to be prepared by RCO. He performs the shielding calculation of the facility and record all shielding assessment data. He also takes necessary steps to provide appropriate protective devices.

Activities of BAERA in the field of Medicine

Radiation Control Division (RCD) of Bangladesh Atomic Energy Regulatory Authority (BAERA) is responsible for regulating the Nuclear Medicine, Diagnostic Radiology and Radiotherapy facilities in Bangladesh.

As per the BAER Act-2012 and NSRC Rules-1997, user licenses for all facilities are issued by RCD, BAERA. RCD also issue the all import/export license, permit and NOC for importing of radioactive materials and equipment, transport license for ensuring the safe transport of radioactive materials. To create awareness among the radiation workers and to motivate them to perform their duties in compliance with the needs of BAER Act 2012 and NSRC Rules-1997, training programs were conducted by BAERA for the policy makers, Radiation Control Officers (RCO) and other relevant persons regularly.

Radiation Protection Practices in Nuclear Medicine

All Nuclear Medicine (NM) facilities including PET-CT follow the all rules and regulations of BAERA. All facilities ensure the proper shielding for safe practice of radiation. The shielding calculations of the facility have to be submitted to BAERA at the time of issuance license. The license of the NM facility is issued for one year by BAERA and it will be renewed every year with proper submission of all radiation protection related information and RCO report. RCO is the main key person for radiation protection of the facility. RCO has submitted all radiation related activities data in his report such as radiation dose level of the workplace, area monitoring record, radiation dose data for radiation workers, radioactive contamination records, Medical surveillance record of radiation workers etc at the time of issuance renewal license.

Radiation Protection Practices in Diagnostic Radiology

As per the BAERA regulations, all radiology departments of hospitals in Bangladesh have to appoint a RCO for radiation safety of the department. He ensures the safe practice of radiation during radiography work. Radiographer stands behind the radiation protective barrier during radiography work. The lead aprons, lead glasses, thyroid shield, etc. have been used while doing interventional radiology. Radiographers operate the mobile x-ray and dental x-ray machine from a distance using extendable control cables and wear lead protective devices. The radiation warning sign have been displayed on the entrance door of the diagnostic x-ray equipment room. The red warning light is available at the entrance of the x-ray room. It has been ensured that nobody enter or stand at the entrance door of the x-room when the red warning light is ON. Time-Distance-Shielding (TDS) principle is always applied to reduce the radiation exposure. In view of ALARA (As Low As Reasonable Achievable) principle the exposure to the patient must be kept to the low without losing diagnostic information.

Radiation Protection Practices in Radiotherapy

In radiotherapy department, sealed radioactive sources and linear accelerator have been used for treatment of oncology patients. The radiation hazards to patients, staff and public in a radiotherapy department have been effectively minimized because the regulatory requirements stipulated by BAERA have been strictly implemented. Area monitoring, personnel monitoring, medical surveillance of radiation workers, wearing the appropriate radiation protective devices etc are the key activities of the safety program. The radiation symbol has been visibly displayed at the entrance door of the radiation room. Radiation indicator lights have been presented visibly at the radiation room entrance. RCO ensure the radiation safety of the department. The Senior Medical Physicist has been performed as RCO in the radiotherapy department.

Conclusion

It is very important to safe practice of radiation in medicine for overall radiation safety for the patients, staff and public. The strong regulation of BAERA, implementation of radiation protection practices is improving significantly day by day in Bangladesh.

98-EYE LENS DOSES OF OCCUPATIONALLY EXPOSED STAFF IN INTERVENTIONAL CARDIOLOGY: A COMPARATIVE STUDY FOR OPTIMIZATION OF PROTECTION

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ABSTRACT

A study on the assessment of effective doses of cardiologists who perform PCI, coronary angiogram and angiogram with angioplasty using the electronic pocket dosimeter (EPD) was conducted in 2011. A follow-up study is performed for assessing the eye lens dose of the cardiologists, nurses and radiologic technologists for the same procedures using the Hp(10) dose records and nanoDot OSL dosimeters. The aim of the study is to determine the category of staff that obtains the highest eyes lens dose and the possibility of exceeding the new dose limit. It also aims to determine the current causes of high doses and practices for optimization of protection.

INTRODUCTION

Interventional procedures in cardiology can result to high patient doses depending on the complexity of the procedures. As patient doses increase, staff who are inside the room during the procedure can also be exposed to high doses [1,2]. Concerns on high staff doses increase as the number of interventional cardiology procedures increase by about two-fold every year. In the year 2011, a study on the assessment of the effective doses of cardiologists from electronic pocket dosimeter placed at the collar level was undertaken [3]. In 2007, the International Commission on Radiological Protection (ICRP) recommended that a review of the dose limit for the lens be undertaken because of the sensitivity to radiation [4]. A new dose limit for the lens of the eyes was recommended to be 20 mSv averaged over 5 years but not to exceed 50 mSv in a single year [5,6,7]. The large decrease in the dose limit necessitates that eye lens doses of radiation workers who could be exposed to high doses be monitored and radiation protection practices be optimized so as not to exceed the new eye lens dose limit [8].

METHODS

The mean eye lens dose of the cardiologists included in the 2011 study was assessed from the EPD readings placed as the collar level during the PCI, coronary angiogram and coronary angiogram with angioplasty. The algorithm of A. Omar $Dose_{eye} = 2$ EPD was used.

A direct measurement of the eye lens doses of three categories of staff in the cardiac suites is performed on cardiologists, nurses and radiologic technologists in the same procedures. Estimation of the eye lens dose from Hp (10) dose records for one year is also performed. The obtained eye lens doses of the cardiologists in the 2011 study and the current study are compared and evaluated to determine improvement in radiation protection practices in terms of the impact of the new eye lens dose limit. Investigation on the possibility of approaching and exceeding the new eye lens dose limit is made for the other two categories. Recommendations for dose reduction practices are made for the three categories of workers so as not to exceed the new limit.

RESULTS

Results of the 2011 study showed that angiogram + angioplasty yielded the highest whole body dose of 7 μSv per procedure (Table1). The annual dose approaches 2mSv if the cardiologist has 1 patient daily. Factors for the high doses of cardiologists are the complexity of the procedure, longer beam on time and use of different lead equivalent thicknesses of the personal protective apparel. The lack of lead goggles was also a factor that contributed to the eye lens dose.

Initial result of the eye lens dose assessment from read out of OSL dosimeters is in agreement with the 2011 study that coronary angiogram with angioplasty is a high dose procedure compared to the other two procedures. The obtained mean dose from the OSL is higher by a factor of 2 compared to the previous study. The procedure constitutes approximately 60% of the total procedures for adult patients. Readings of the OSL show that the eye lens dose of the consultant cardiologist who is closest to the patient is higher by a factor of about 2 to 4 compared to the dose of the assisting cardiologist. The assisting nurse eye lens dose is lesser than the cardiologist by a factor of 5 while the radiologic technologist dose is the least. Increased patient workload can lead to an eye lens dose that will approach the new dose limit for the consultant cardiologists.

CONCLUSION

Both studies show that eye lens doses of cardiologists do not exceed the new limit. This is attributed to the training in radiation protection and awareness on the new eye lens dose limit. Since the patient workload increases every year and complex procedures are being introduced due to the acquisition of sophisticated and advanced technologies, monitoring of eye lens doses should be made mandatory if the dose approaches or exceeds 5 mSv in a single monitoring period. The use of lead goggles is at times not followed. To reduce exposure of the lens of the eye the use of appropriate lead goggles should be strictly imposed. Awareness of the radiation protection practices through education and training programs should be strengthened. A policy on rotation of cardiologists for high dose procedures and proper head posture with reference to the axis of the primary bean should be developed and implemented.

TABLE 1. EYE LENS DOSES OF CARDIOLOGISTS ASSESSED FROM EPD MEASUREMENTS.

Procedure	No. of staff monitored	Mean EPD reading per procedure (μSv)	Estimated eye lens dose/procedure (μSv)	Estimated yearly annual eye lens dose (mSv)
PCI	14	0.23	0.46	0.13
Coronary angiogram	40	0.65	1.30	0.40
Coronary angiogram with angioplasty	10	3.5	7.0	2.02

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99-QUALITATIVE AND QUANTITATIVE IMPACT OF IONIZING RADIATION VERSUS TO DRUGS INDUCED DIFFERENT IMMUNOLOGICAL CONDITIONS

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The study aims to classify and evaluate the potential immunomodulatory response of several doses of ionizing radiation (IR), assessing its effects against various immunological conditions; this can propose an innovative medical application that can be promising particularly in low-doses. Forty albino male rats were divided into eight groups, governed by all ethical protocols for animal care. Group-I was control and Groups-II to VI exposed to 0.25, 0.5, 1, 2, and 5 Gray (Gy) IR respectively, Group-VII received orally 500mg/kg *Echinacea purpura* (immune-stimulant), Group-VIII received IP 50mg/kg Cyclophosphamide (immune-suppressant), daily treatments for 7 days. The animals were irradiated using the Cesium $_{137}$ gamma source at a dose rate of 0.66 cGy/Sec. in the National Center for Radiation Research and Technology, Egypt. Hemoglobin content, Hematocrit percentage, White, Red, and Platelet cell counts were counted using an automated hematology analyzer, ELISA performed for Interleukin-10 (a potent anti-inflammatory cytokine that plays an essential role in preventing inflammatory and autoimmune pathologies), Tumor necrosis factor-alfa (an important pro-inflammatory cytokine), Colorimetric determination for Nitric oxides and Malondialdehyde (oxidative stress biomarkers in many health problems). Data were Statistical analyzed using the Statistical Package for the Social Sciences version 25. Kruskal-Wallis H tests, which were carried out to measure a specific difference between the means for Groups I-VI, Mann-Whitney test used to compare Groups VII, VIII, to all the studied groups. The Spearman correlation coefficient was applied to fit the relationships between the different studied variables.

TABLE 1. THE LEVELS OF INTERLEUKIN-10, TUMOR NECROSIS FACTOR-ALFA (IL-10, TNF-α, PG/ML), NITRIC OXIDE AND MALONDIALDEHYDE (MDA, NMOL/ML AND NO, MMOL/ML) IN THE SERUM OF CONTROL (GROUP-I) RATS AND THOSE GIVEN 0.25 (GROUP-II), 0.5 (GROUP-III), 1 (GROUP-IV), 2 (GROUP-V), 5 (GROUP-VI) GY ABSORBED DOSES OF GAMMA RADIATION AT 24 HOURS POST-EXPOSURE, AND THOSE GIVEN 500 MG/KG ORALLY *ECHINACEA PURPURA* (GROUP-VII) AND 50 MG/KG INTRAPERITONEAL CYCLOPHOSPHAMIDE (GROUP-VIII) AT DAY 7 AFTER DRUGS ADMINISTRATION

Parameters	Group-I	Group-II	Group-III	Group-IV	Group-V	Group-VI	Group-VII	Group-VIII
IL-10	111±1.6	117±1.5*	117±4.3*¤	93±0.8*¤#	110±2.6*¤#	71±2.9*¤#•@	195±9.1*¤#•@¥	72±3.4*¤#•@ə
% of change[1]	-	5%	5%	-16%	-0.9%	-36%	76%	-35%
TNF-α	26±1.5	77±4.8*	119±9.9*¤	110±4.0*¤	129±4.0*¤@•	139±7.4*¤•	73±5.9*#•@¥	17±0.8*¤#•@¥ə
% of change[1]	-	196%	357%	323%	396%	434%	181%	-35%
NO	13±1.0	74±6.0*	94±5.2*	98±4.1*	90±7.9*	115±3.7*¤#@	21±1.8¤#•@¥	78±5.9*¥ə
% ofchange[1]	-	469%	623%	653%	592%	754%	61%	500%
MDA	16±1.2	33±1.3*	59±1.4*¤	81±3.8*¤#	94±0.7*¤#	97±1.4*¤•#	14±1.9¤#•@¥	100±4.9*¤#ə
% of change[1]	-	106%	268%	406%	487%	506%	-12%	525%

Data represented as a Mean± SEM, % of change[1] compared to Group-I.
*, ¤, #, •, @, ¥ and ə indicate a significant change between Group-I, II, III, IV, V, VI and VII respectively at α=0.05 (P<0.05).

The Kruskal-Wallis H test confirmed that there was an association in the serum levels of IL-10, TNF-α, NO, and MDA between Group-I and all the irradiated Groups at $\chi2_{(cv,5)}$= 23, 24, 21 and 27 respectively at α=0.05 ($P < 0.001$).Also the serum levels of IL-10, TNF-α, NO and MDA showed a correlation coefficient with IR doses as being valued, r = -0.69, r = +0.88, r = +0.79 and r =+0.97 respectively at α= 0.05 ($P < 0.01$).Mann-Whitney test confirmed a significant change between all groups at α=0.05 ($P < 0.05$), except with Group-VII and Group-I in MDA, NO and Group-II in TNF-α at $\chi2_{(cv,5)}$ =25, 16 and 27 respectively, the same as in Group-VIII and Groups-II to V in the serum levels of NO at $\chi2_{(cv,5)}$ =24,23,19 and 21, Group-VIII and Group-VI in IL-10 and MDA at $\chi2_{(cv,5)}$= 27 and 21 and also MDA of Group-V $\chi2_{(cv,5)}$= 20 at α= 0.05 ($P \geq 0.05$).

It is extremely important to examine the impacts of different IR levels and determine their immune stimulant or suppressive effects. In this context, an innovative medical application that can make IR doses especially at low-levels very appealing. More particularly, the immunological effects of various doses should not only be evaluated and compared to the un-irradiated group, but also evaluated by comparing to the Induced immune stimulant or suppressive responses, as seen in the study. The null hypothesis is represented when there is no significant difference between groups, the null hypothesis was accepted in some tested parameters between Immune-stimulant drug and the low-dose, high-dose, and immunosuppressive drug.

In comparison with Group-I, IL-10 showed a significant elevation of 5% in the Group-II and III, although it showed a significant negative correlation coefficient with IR doses, It suggests a positive impact on IL-10 at low-levels. The potent, anti-inflammatory activity of IL-10 has been long established [1], in regulatory T-cells, it has an induction function as well and it seems to exist in a dynamic equilibrium with Th-17 cells, which involved in the pathogenesis of many chronic inflammatories and autoimmune diseases [2]. However, the timing and site of IL-10 production influence its impacts, the various behaviors that include both immune-suppressive and stimulating, suggest the concept of optimum IL-10 doses, IL_{10} acts to antagonize the pro-inflammatory cytokines [1], Group-II showed a minimum rise of TNF-α, while IL-10 displayed the highest response in all the groups irradiated. Pro-inflammatory cytokines are the main elements of early gene programs and can be stimulated quickly after irradiation [3], and this was confirmed in the study with the strong correlation coefficient with radiation doses ($P<0.05$). The elevation of TNF-α supported the assumption about its role in organ-specific autoimmune damage that TNF-α reacts mainly to produce IFNγ and TNFα and cooperates with CD8+ T cells and M1 macrophages to eliminate intracellular viruses, bacteria, and tumors [3]. In Group-II showed the lowest increase of oxidative stress levels for all irradiated groups. Group II also showed the least significant change of hemoglobin content and RBC counts by -7% and -8%, whereas group III showed -14% and -38% respectively, and all in comparison to Group-I, while WBCs accepting null hypotheses in the same groups.

The null hypothesis was rejected in all of the irradiated groups for all measured parameters with Group-VII except TNF-α corresponding to Group-II at $\alpha= 0.05$ ($P \geq 0.05$), although both of them were significantly different with Group-I, Group-II were the dose showing the least differed significantly from Group-VII. In Group-VII and Group-I, II, the null hypothesis were not rejected by hemoglobin content, RBCs and platelets counts at $\alpha= 0.05$($P \geq 0.05$), whereas in Group-VII, WBCs reject null hypotheses with 15% elevation compared to Group-II $\alpha= 0.05$ ($P<0.05$).

In Group-VI, IR effects converge with pro-inflammatory cytokines, The effect of secondary ROS released from pro-inflammatory cytokines post-radiation exposure can be profound [4], Increased MDA and NO in the higher doses with 506 and 754% compared to Group-I, with a strong correlation with IR doses ($P<0.05$), Therefore, their levels increased to further release of ROS. TNF-α Increased by high IR level, and this was confirmed by the highest elevation of 434 % in Group-VI, whereas in Group-VIII it decreases significantly by -35%, both of them in comparison with Group-I, Indicating a high-level of IR cannot be addressed as an absolute immune-suppressant dose. Groups-VIII and VI failed to reject the null hypothesis in IL-10 and MDA responses at $\alpha= 0.05$ ($P \geq 0.05$), While in all other tested parameters rejection of the null hypothesis occurred at $\alpha= 0.05$ ($P<0.05$). Finally, the high level of IR leads in excessively proinflammatory cytokine, whereas low dose anti- and pro-inflammatory cytokines may trigger, additional findings for new approaches to identify a potential therapeutic possibilities under IR.

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100-RADIATION SAFETY KNOWLEDGE AND PERCEPTIONS AMONG RESIDENTS AND PHYSICIANS IN A BRAZILIAN MEDICAL UNIVERSITY

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INTRODUCTION

The introduction of new technologies in the area of radiodiagnosis, associated with the development of new X-ray equipment, has contributed to the increasing use of ionizing radiation in medical applications. All this technological development resulted in an increase in the frequency of exams and in the diversification of procedures, requiring special attention to the doses received by the patients and the quality of the image. In fact, despite the great advantages, radiological examinations represent the main source of human exposure to artificial sources of radiation and, therefore, they must be justified, considering the benefits that may be produced and the detriment that may be associated with given exposure. The justification of the exams, dose optimization and the training of health professionals involved in these procedures has become an important challenge [1]. Itt is observed that the physicians in general are not aware about the biological effects of radiation and its risk and they request high number of radiological exams, which are not always justified. Universities and residency courses have a great responsibility to contribute to the training of physicians and to the dissemination of concepts of radiological protection. In this sense, the aim of this presentation is to evaluate the radiation safety knowledge and perceptions among residents and physicians in a Brazilian medical university.

MATERIALS AND METHODS

This cross-sectional study was conducted in the Medical Faculty of the Pontificia University Catholic of São Paulo, Brazil, during the period of September to November 2019. Choice questionnaire was distributed to the academic medical students, comprising 1st to 6th year residents and multispecialty physicians (N=311).

The questionnaire tried to test the participants' knowledge regarding diagnostic procedures, such number of exams performed, justification of these exams, and importance of radiation protection course in medical curriculum. In addition, it was included questions about the radiation dose received by the patient in CT and lumbar x-ray exams, imaging procedures that result in high dose for the patient and pediatric risk in CT procedures.

RESULTS

The results show that 75.7% of the participants, around 75.7%, independent of their academic level, identified that there are a high number of requested radiographic exams, which in most cases are not justified. Figure 1shows the reasons reported by the participants for this high request of radiographic examinations, and Figure 2 shows distribution of the responses according the level of academic training of the participants. It is observed that for the students of the first residence level (1st and 2nd) the financial interest is one of the motivations to request the radiographic exam. Defensive medicine was indicated, by all group of participants, as the highest cause to ask for radiographic exams for the patient. It is important to observe that these results indicate that the concept of justification and dose risk are not well understanding for the medical students and for the physicians of different specialities, that participate in this study. The practice of justification is an important safeguard for patients against the adverse health effects due to the use of ionising radiation.

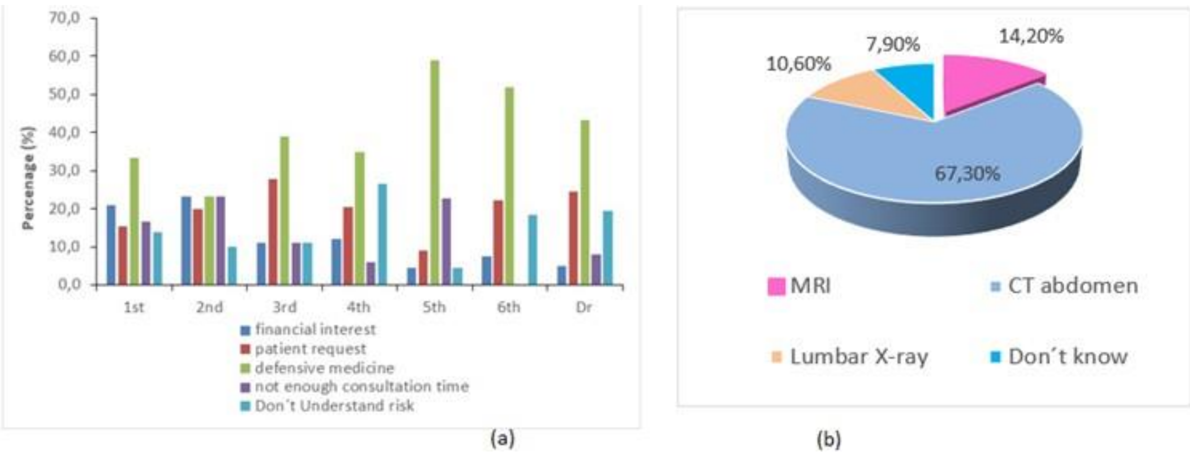

Figure 1- (a) Distribution of the reasons reported by the participants to explain the large order for radiographic examinations according their academic level. (b) Percentage response of the participants about the imaging examination that produced high radiation dose for the patient

Most of the participants (96,3%) considered important to have an information access system about patient's previous exams, but they identify that rarely it is available. When asked about in which type of radiographic exam the patient receive high radiation dose, 7.9% of the participants declare that do not know, 10.7% answered that it is the lumbar x-ray and 67.3% the abdomen CT, as shown in Figure 2. The surprise was that 14.2% of the participants answered that Magnetic Resonance Imaging (MRI) produced high ionizing dose radiation. These results indicate that the operation principle of x-ray and of MRI equipment are not understood. On the other hand, 94.4% of the participant considered important to have radioprotection course in the curriculum.

CONCLUSION

It is possible to conclude that that medical students and physicians participating of this study need more orientations about radiological exams. Curriculum reformulation to medical students, residents and educators must be done. The assessment instruments contributed to the educational planning and as a pilot study to the members of radioprotection commission of Brazilian College of Radiology. Radioprotection activities will be incorporated into the pedagogical curriculum as well as permanent education to multi-professional students, residents, health professionals, physicians, and educators in the Medical Faculty of Pontificia University of Sao Paulo

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101-REAL-TIME VISUAL AND AUDITORY RADIATION DETECTION AND ALARM SYSTEM IN CLINIC TO PROTECT STAFF AND PUBLIC

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With growing application of radiopharmacy in clinic, increasing patients benefit from radionuclides for diagnostic and therapeutic purposes. However, it is non-trivial that radionuclide administered patients ignore instructions of keeping necessary *social-distancing*, remaining active in non-planned area. These situations obviously expose the public to non-medical radiation, among whom clinical workers are at high risks. This has become a serious and challenging issue.

To effectively identify and alarm *radioactive* patients to protect clinic staff and the public, a real-time visual and auditory radiation detection and alarm system is developed and deployed in Dept. Of Radiation Oncology, Beijing Cancer Hospital (also known as *Peking University Cancer Hospital & Institute*).

The system was initially designed for homeland security inspection, and has been well-established in the Customs. The key components of the system are: a CsI(Tl) based gamma detector, a spatially coded aperture (in Fig.1a), a general-purpose visual camera, and a data-processing workstation. The system can i) measure count rate and equivalent dose rate, ii) locate gamma sources within FOV, and ii) combine the gamma source location maps with the visual images. In system configuration, count/dose rate thresholds and user group information can be defined. Once over-threshold signals are received, the system starts to notify users with hotspots-identified images, and the audio warning signal will be alarmed simultaneously.

The demo system (in Fig1b) was installed in the waiting area of Department of Radiation Oncology, and focused towards the public entrance which is close to the patient exit of Department of Nuclear Medicine. Since the system commissioning in August 2019, alarming events have been reported from time to time (in Fig1c), and monthly-summary reports have been feedbacked to Department of Nuclear Medicine for information.

In spite of some performance improvements to expect, the clinic feedback is very positive. The system is sensitive to radiations, easy to deploy and user-friendly to operate. What's important, since alarming events have been effectively reported and shared with Department of Nuclear Medicine, the radioactive patients can be well instructed, which leads to a significant decrease in case numbers. To conclude, the system is highly helpful in managing radioactive patients and protecting clinic public in non-planned exposure area.

(a) (b) (c)

FIG. 1. Visual&auditory radiation detection and alarm system: (a) coded aperture imaging principle; (b) detector appearance and ceiling installation in Beijing Cancer Hospital; (c) an alarming case of a radioactive patient

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103-ESTIMATION OF THE TIME-DEPENDENT ACTIVITY OF THE RADIONUCLIDES AND MANAGEMENT OF THE RADIOLOGICAL EXPOSURE FOR FUKUSHIMA-DAIICHI UNIT 1 ACCIDENT

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On 11th March 2011, Fukushima Daiichi Unit 1 (FD-U1) was generating electricity when the earthquake occurred and shutdown automatically and the resulted tsunami halted the emergency core cooling. Therefore, water levels in the reactor vessel dropped below the top of the hot fuel, and steam began reacting with the zirconium fuel cladding to produce large amounts of hydrogen and core was melted. Following the accident at FD-U1, the highest radiation exposures for the public occurred in the first months due to the external irradiation from deposited material, inhalation and, possibly, ingestion of foods. At later times, radiation exposures decreased significantly and external radiation from the deposited material became the most significant exposure pathway.

In the presentation, the radionuclides inventory and time dependent activities for the fission product of the burned fuel of FD-U1 accident were calculated using MCNPX 2.7 -code. These results are used for the support of the accident management program and in planning the emergency actions to be taken for mitigation of the environmental releases and protection of workers and public.

Emergency Response Procedures (ERP) were initiated for both on-site and off-site responses. The maximum radiation dose and the activity levels that would necessitate implementation of protective measures were defined.

A Severe Accident Management (SAM) program, with appropriate implementing procedures, should be established to address mitigating actions after an accident that results in widespread physical damage to fuel and core structures accompanied by a large release of fission products into the facility and potentially to the environment.

Methods, systems, and equipment for assessing and monitoring actual or potential off-site radiological consequences were used. Specific to airborne effluents, these methods should include processes to track and estimate exposures from all types of airborne releases of radioactive materials, including noble gases, radioiodine, and particulates. In addition to tracking airborne radiological plumes, methods should be in place to estimate radiological doses from the inhalation and deposition of radionuclides.

Also, the occupational emergency plan was presented, which provides valuable information for the local community, helping to ensure that any radiological health effects are detected quickly, and that appropriate actions are taken to protect the health of the population.

105-RADIATION RISK REDUCTION - A GRADED AND PRAGMATIC OPERATIONAL APPROACH ADAPTING THE UK NPP PHILOSOPHY TO HELP CONTROL RADIATION RISK IN PLANNED EXPOSURE SITUATIONS.

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INTRODUCTION

In the United Kingdom (UK) the Nuclear Power Plants (NPP) use a graded approach to designating radiation areas to facilitate control of access which is commensurate with the risk. The UK philosophy is summarised and the method adapted to produce a generic approach to radiation control. It is hoped this simplified approach will help inform and guide applications and situations where the depth of experience of NPP is not available to assist the development of radiological safety culture

SYNOPSIS OF UK NPP PHILOSOPHY

Current UK radiation protection legislation [1] is based on the Euratom BSSD [2], and UK Health and Safety legislation puts the onus on the owner of the risk to reduce that risk to a level which is As Low As Reasonably Practicable (ALARP). The use of ALARP, rather than As Low As Reasonably Achievable (ALARA), comes from ‘Reasonably Practicable’ having been tested in the UK law courts [3], noting this was prior to the first UK NPP being built. The radiation employer must also restrict or control access to areas where radiation exposure could exceed certain levels, namely 1 mSv/y and 6 mSv/y whole body dose[1]. The approach to how UK NPP design and operation meets these requirements is the focus of the first part of the presentation.

The starting point is to translate the annual dose limits to dose rates, as this is what is measured in operation and used in the design phase. The UK sub-divides the controlled area into further zones to give an indication of the increasing radiation hazard, as shown in Table 1.

TABLE 1. TYPICAL UK NPP ZONING DOSE RATES

‘R’ Level	R1	R2	R3	R4	R5
Colour	Blue	Green	Yellow	Orange	Red
Start doserate	0.5 µSv/h	3 µSv/h	25 µSv/h	0.5 mSv/h	100 mSv/h
End doserate	3 µSv/h	25 µSv/h	500 µSv/h	100 mSv/h	-

The increase in zoning also indicates an increase in control. The type of control can give an improvement in effectiveness, often referred to as the hierarchy of controls, with passive engineered controls being at the higher end, active engineered controls in the middle and written or administrative controls at the lower end.

For UK NPP the whole site is controlled as part of the security of nuclear material and this also gives the first level of access control to R1 areas. For R2 areas, workers require electronic dosimetry and this allows a level of control as the database will have details of the workers training and authorisation to enter. R3 zones do not have physical controls but worker training informs the workers only to access those areas if required as part of their work. The purpose of R3 zones is to highlight a higher radiological risk. R4 zones are locked and a specific permit to work is required to get access which includes radiological monitoring. R5 zones are prohibited access; there would have to be a specific work package to reduce the radiological hazard to an acceptable level before accessing the area.

In practical terms most individual areas or rooms do not have uniform dose rates; the room is zoned according to the highest dose rate and the whole room is treated at this zoning due to the practicalities of controlling access to thousands of rooms on a large industrial site. Some areas have transitory higher dose rates and the zoning can either be kept at the higher level to restrict access or the area is kept at the lower zoning and the work activity which creates the higher dose rate is only allowed once the area is zoned, and thus controlled, at the higher level.

[1] There are also limits for the lens of the eye and extremities, but these are not considered in this presentation

ADAPTATION TO OTHER SITUATIONS

The starting point for adapting the UK philosophy is knowledge of the local legislation, radiological rules and the dose where the regulatory authority wants the radiation exposure controlled. It is also important to know if it will only be radiation workers accessing the area as other workers and members of the public will have much less training and familiarisation of radiological hazards and are likely to have much lower dose limits.

The next step is to understand the radiological hazards within the main work areas. It is also important to understand non-radiological risks, such as chemical or toxicological hazards, as these could require another type of control. In these cases, it is pragmatic to combine the radiological and non-radiological controls if possible.

In practical terms the graded approach starts with an outer boundary of control, the first zone. This is chosen to only allow persons who have a need to interact with the radiation source to access the whole area and prevent non-authorised persons receiving a radiation exposure which would warrant regulatory attention. For low hazard radiation situations this may be the only control required.

For medium hazard situations a second, inner zone, is required. This should be chosen when workers have to be approved to access and/or work in that radiation environment. These workers will need their radiation exposure assessed and thus using the required controls for issuing dosimetry is the pragmatic way to ensure the workers have approval. An important point is to be consistent in the approach.

For high hazard situations there should be a high level of control. Typically, this will involve the controls for both low and medium hazard situations, plus a higher level of control. The further inner zone could also need passive and active engineered controls as well as administrative control. Passive engineered control includes shielding and active control includes interlock arrangements. If the high hazard situation is an occasional exposure, then the higher level of control may not be necessary when the situation is not present. However, the setting up of higher controls from zero should be assessed as, for instance, installing and removing temporary shielding could incur significant radiation dose to workers doing that task.

A schematic view of three zoning areas is shown in Fig. 1 below.

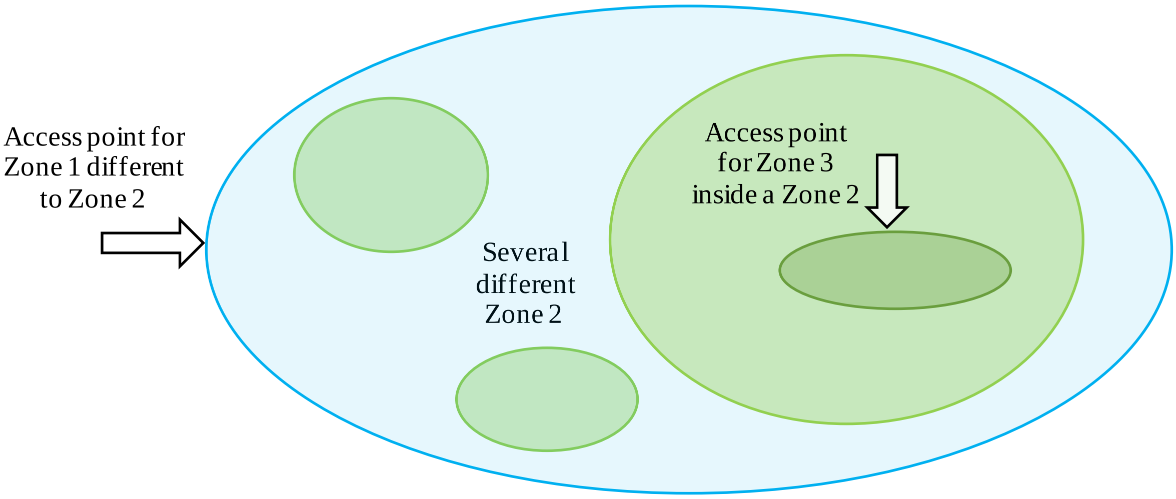

FIG. 9. Schematic layout of low, medium and high radiation zones.

CONCLUSIONS

The UK arrangements of radiation zoning at NPP can form the basis for a graded approach to controlling the access to radiation risk areas in situations different from NPP. Local circumstances can adapt the specific measures and zoning values as appropriate, following the general advice of increasing the controls on access as the radiation hazard increases.

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106-ENHANCING CAMS PERFORMANCES BY MEANS OF A DEDICATED ALPHA COLLIMATOR

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The task of the modern Continuous Air Monitors (CAMs) is to provide for the continuous monitoring of the radioactive particulates collected on a filter media from air, using spectrometric techniques. They usually perform the on-line alpha and beta measurement from the filter involving appropriate and advanced algorithms to evaluate the presence of long lived alpha (f.i.: 239Pu) and beta emitters. The main results are the associated concentrations in air expressed in Bq/m3 or DAC. Aerosol detection is carried out using an on-line solid state (usually ion-implanted silicon) detector connected to a measuring chain including a relatively fast MCA section for 1024/2048 channels. The spectra are analyzed by the algorithms to separate the contributing components of alpha and beta emitters, and determine whether they are ‘long lived’ or ‘naturally’ (specifically radon and thoron short lived daughters) occurring. [1] [2].

The measuring chain is the typical one for charged particles detection and analysis, but with the presence of specific situations, some not favorable with respect analogous operations made in laboratory, represented by:

- the system works in air and not in vacuum; the charged particles have a path in air before reaching the detector window; some residual light may reach the detector window.
- The track of sampled particles (dust) on filter media is characterized by shape and dimensions properly selected in order to enhance the matching of the circular emitting area in front of the circular sensitive detection window.
- Due to the above two conditions the path of the charged particles from the filter to the detector is characterized by a spread dependent on the dimensional/geometrical characteristics; this is particularly negative for the alpha particles.
- The system works while particulates are sampled (and accumulated) on the filter media; therefore the distance between the filter and the detector window shall allow a suitable air path for representative sampling, and the air flow (usually moved by a motor pump) shall generate no disturbances.

R&D Laboratory (RTS Instruments R&D Lab) has approached the above issues in order to define the best possible set-up for its families of Continuous Air Monitors (CCAM series - compact, small, wall/skid or cart mounted - and RAM series - usually cabinet rack or multi-skid mounted), with particular regard to the study, design, test and employment of an alpha collimator optimized for this application. [3] [4] [5] [6].

The following Fig. 1 shows the spectrum of Am-241 with the use of the final developed collimator in the used geometry (the collimator width is 3 mm and the overall distance is 5.9 mm, 450 mm2 detector, disc source with 28 mm diameter, MCA with 1024 ch.), for a ‘M’ type (metallized) silicon detector. The very satisfying results are indicated below:

- FWHM = 4.60 % that is 252 keV for the Am-241 peak (5.486 MeV / 85.2% + 5.443 MeV / 12.8%).
- Region ‘A’ = 23.60 % of total, Region ‘B’ = 0.52 % of total.
- Alpha detection efficiency ≈ 9.5 %. Beta detection efficiency ≈ 12.3 %.

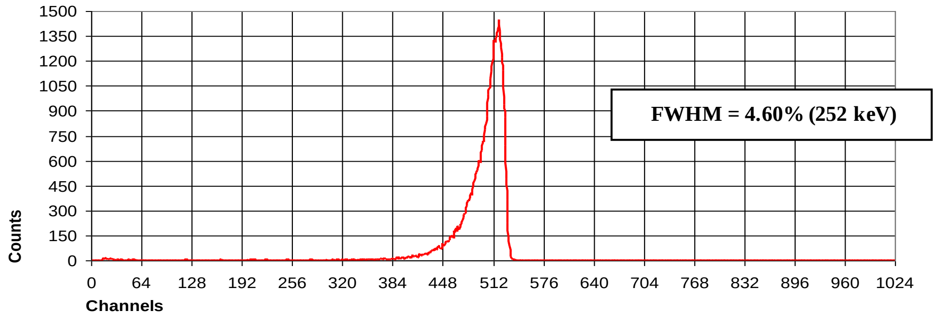

FIG. 10. Chart showing the Am-241 spectrum 'M' at 5.9 mm air collimator.

Regarding alpha efficiency it is to be noted that for the typical distances without collimator (3.3 - 3.6 mm) it is in the range 23 - 24 %. But for 5.9 mm it would be 18 - 19 %. Therefore the efficiency reduction due to the collimator is about 50%. But the associated advantages greatly overcome this in terms of sensitivity and reliability (Fig. 2).

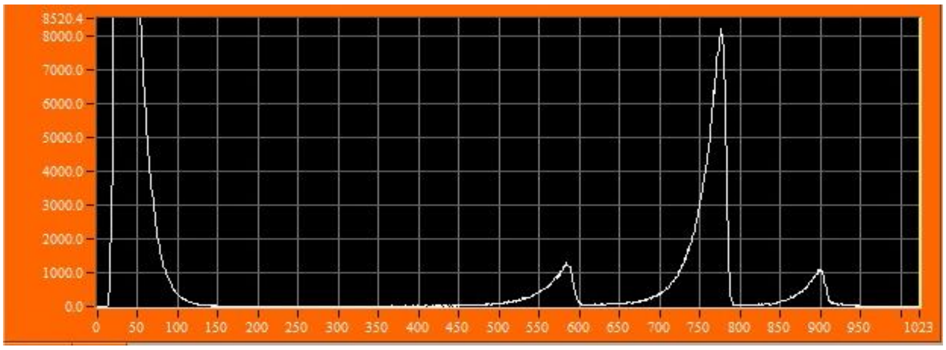

FIG. 2. Chart showing the result of the collimator applied to a real CAM that samples the Radon/Thoron solid daughters on filter, with a 'P' type (passivated) 450 mm2 silicon detector.

The average values of a production batch of ten CCAM-TF units (Compact Continuous Air Monitors with Tape Filter) in terms of alpha resolution (FWHM %), alpha efficiency (Ea %) and beta efficiency (Eb %) are the following:
FWHMm% = 4.64 ± 0.39 (254.5 ± 21.3 keV), Eam% = 9.51 ± 0.21, Ebm% = 12.33 ± 0.24.

This paper shows that modern CAMs can be equipped with suitable alpha collimators in order to facilitate alpha spectrometry from filter media thanks to better energy resolution (FWHM around 250 keV or even better) of the alpha peaks. Appropriate design may ensure acceptable alpha peak efficiency (9.2 to 9.9 %) and alpha spectra characterized by good shapes and linearity and none or negligible, or in any case not increased, spurious peaks and unwanted coincidences. Beta efficiency acceptable (12.1 to 12.6 %). Enhanced alpha resolution allows the Radon/Thoron compensating algorithms to work with better reliability due to much lower 'background' components. This, combined with alpha efficiencies that are just around 40-45% of the typical ones without collimator, allows to reach lower alpha MDLs with usual sampling/measuring times and sampling flow-rates. And to decrease the false alarm probability for alpha contamination. All the above, together with an appropriate selection/design of the associated parameters/devices coming from a long time and continuous work by our R&D Lab - ensures a significant step forward in the field of in-air radioactive particulate monitoring.

Further investigations are also planned for what concerns the detector (a firm topic for too many years), the filter media and the compensation/identification algorithms.

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107-ESTABLISHING AND RUNNING THE NATIONAL RADON PLAN FOR CONTROLLING PUBLIC EXPOSURE DUE TO RADON INDOORS IN CAMEROON

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INTRODUCTION

The requirements to put in place a national radon action plan are well defined in the IAEA Safety Standards Series Nº. GSR Part 3: Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [1]. The government shall provide information on levels of radon indoors and the associated health risks and, if appropriate, shall establish and implement an action plan for controlling public exposure due to radon indoors. Where activity concentrations of radon that are of concern for public health are identified, the government shall ensure that an action plan is established comprising coordinated actions to reduce radon levels for existing buildings and for future buildings. The regulatory body or other relevant authority shall establish a strategy for protection against exposure due to radon in workplaces, including the establishment of an appropriate reference level for radon.

Having acquired preliminary data on radon, thoron and progeny in 450 dwellings thanks to international scientific collaboration [2-11], an IAEA Technical Cooperation project (CMR9009) on "Establishing a National Radon Plan for Controlling Public Exposure due to Radon Indoors" was initiated and accepted to run for the IAEA TC cycle 2018–2019. The accomplishments of the project, the lessons learned and the next challenges will be presented.

ACCOMPLISHMENTS

Within the framework of TC cycle 2018-2019 the national project CMR9009 was implemented with the following results:

- 03 persons trained in two European laboratories (Spain and Hungary) in March 2019;
- 30 persons trained during a national training course on Regulatory Control of Public Exposure to Radon in Dwellings and in Workplaces held in July 2019;
- 1500 radon track detectors delivered and deployed in dwellings of the whole country;

- 1400 detectors analysed and the results obtained (see Table 1);
- The arithmetic mean at the national level is 108 $Bq.m^{-3}$;
- 45%, 9%, and 2% of houses have respectively radon concentration above 100 $Bq.m^{-3}$, 200 $Bq.m^{-3}$, and 300 $Bq.m^{-3}$;
- Radon calibration system provided to allow for independent and reliable radon measurements;
- Equipment for radon measurements in soil and *in-situ* gamma spectrometer provided for radon-risk mapping;
- 20 integral radon monitors provided to allow for long term radon measurements at workplaces and in dwellings;
- Regulation on radon drafted and being validated through standard process by 2020;
- National radon action plan validated during a national workshop in September 2020.

To efficiently develop and prepare for the implementation of the young national radon action plan and mitigate high radon levels in houses and workplaces, capacity building and development in the area of radon risk communication, radon mitigation and prevention is needed additionally to measuring capacity and regulations, developed in the framework of CMR9009. A new project on *Strengthening National Radon Action Plan to Mitigate Public Exposure to Radon in Dwellings and at Workplaces in Cameroon* is under review for 2022-2023 TC Cycle.

TABLE 1. Number of RADTRAK detectors deployed per region in Cameroon and corresponding indoor radon concentrations.

Region	**Number of detectors deployed and collected**	**Range**	**Arithmetic mean**
Adamawa	306	27-538	133
Centre	150	2-140	41
East	100	21-304	65
Far North	282	62-310	157
Littoral	100	2-77	24
North	150	15-2264	39
North West	40	-	-
South	200	23-2620	58
South West	40	-	-
West	120	41-530	140
Total	**1488**		

ACKNOWLEDGEMENTS

CMR9009 TC project is funded by the International Atomic Energy Agency. The Institute of Geological and Mining Research and the National Radiation Protection Agency fund field works and the organization of local events related to the project.

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108-CLASSIFICATION OF THE AREA AROUND THE X-RAY ROOM AT GULU REGIONAL REFERRAL HOSPITAL IN UGANDA

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ABSTRACT

The study used the commonest parameters for the lumber-spine and skull examinations.
It was found that positions 4 (control room) and 2 (left of the X-ray machine) had the highest radiation doses at 30cm and 60cm respectively from the wall. The highest radiation dose recorded was 7.52×$1O^{-5}$mSv/year at position 2, followed by 7.20×$1O^{-5}$ mSv/year at position 4 (control room).
The lowest radiation dose was 2.93×$1O^{-5}$mSv/year and 3.3×$1O^{-5}$mSv/year at 30cm and 60 cm respectively from the wall at position. The results of the study were all below the annual dose limit (1mSv/year) and were therefore classified as public areas.
It was also observed that the radiation warning lights were faulty and could stay the same during an examination. Basing on the results above, it was advised that the positions where averagely higher values were seen should be kept out of bounds to people with no knowledge of the radiation safety principles so that the absorbed doses are kept As Low As Reasonably Achievable (ALARA).

INTRODUCTION

An estimated 3.6 billion diagnostic medical examinations such as those performed by X-rays are done worldwide every year. The use of radiation for medical diagnostic examinations contributes over 95% of manmade radiation exposure [1]. X-rays are a form of electromagnetic radiation, similar to visible light. Unlike light, however, X-rays have higher energy and can pass through most objects, including the body.

Medical exposures to radiation are intended to provide a direct benefit to the exposed individuals, it is however possible that some members of the public and the radiation workers are exposed to higher than the recommended doses [2]. This is because the area the X-ray bunkers is not classified or clearly demarcated.

A controlled area is one in which specific measures for protection and safety are or could be required for controlling exposures in normal operations and preventing the likelihood of exposures in anticipated operational occurrences and accident conditions while a Supervised area is any area not already designated as a controlled area but for which occupational exposure conditions need to be kept under review, even though specific measures for protection and safety are not normally needed [3]. Public areas are all other areas in the hospital or clinic and the surrounding environment.

In many parts of Uganda today, protecting patients, the hospital staff and the public from radiation risks can be a challenge, radiation guidelines are often inadequate and rarely adhered to.

METHODS AND MATERIALS

A Pressurized μR ion chamber survey meter (451P-RYR, serial number 6542 containing isotope Cs-137) manufactured by Fluke Biomedical shown in Fig. 1.0 below, a tape measure and a bucket of water were used. Four points were marked on the X-ray bunker at which dose measurements were to be made. For consistency of results, a bucket of water was placed on the couch in normal patient position and the collimator adjusted to have a sizeable field.

FIG 1.0. *Survey meter in position 1*

The commonly used X-ray parameters for lumber-spine or skull were selected for they provide the highest patient doses. The Survey meter was set at 30 cm from the wall, 90 cm from the ground at position 1 and when an exposure is taken, a reading is made. The procedure was repeated at 60 and 90 cm from the wall at the same position. The readings were recorded and the procedures above repeated for positions 2, 3 and 4. The annual exposure dose was computed as a product of the number of patients received per day, the exposure time per exam, the Survey meter reading and the workload. The area classification was based on the following area:

(a) Controlled area: > 6 mSv/year
(b) Supervised area: 1 mSv/year ≤ S ≤ 6mSv
(c) Public area: 1mSv/year

RESULTS AND DISCUSSION

TABLE 1. MEASURED LEAKAGE RADIATION

Position	Distance from the wall (cm)	Annual workload (hrs)	Annual dose X10 $^{-5}$	Classification
1	30	0.135	3.65	Public area
2	30	0.135	7.52	Public area
3	30	0.135	2.93	Public area
4	30	0.135	5.13	Public area

From Table 1 above, at 30cm from the wall, the radiation varies in all positions. The highest radiation dose of 7.52×10^{-5}mSv/year, is seen at position 2 and the lowest, 2.93×10^{-5}mSv/year, at position 3. The high value at position 2 is because of the small distance between the machine and the wall in this direction, this means that a lot of radiation is scattered in this direction. The low value of position 3 is due to the presence of the lead door which attenuates the radiation in its direction.

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109-ESTABLISHMENT OF CT DIAGNOSTIC REFERENCE LEVELS IN SELECT PROCEDURES IN NORTH OF IRAN

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BACKGROUND

During the last decade, much research has been done in the field of patient dose reduction [1]. Diagnostic Reference Levels (DRLs) were introduced as a tool for identifying facilities with usually high doses and for promoting the optimization process [2, 3]. Computed tomography is a device with higher patient dose in comparison with other conventional radiation procedures. Therefore, the goal of the present study was to evaluate the current levels of patient radiation dose for some routine computed Tomography procedures Toward establishing diagnostic reference levels in north of Iran.

METHODS

A questionnaire was developed and patient related data, the specifications of the CT scan machine and protocol were recorded. Four CT examinations including brain, sinus, chest, and abdomen and pelvic were examined. In this study dose measurement was performed with pencil ionization chamber connected to Xray multimeter and CT dosimetry phantom. DRLs are proposed based on two primary dosimetry metrics: CT dose index volume (CTDIvol) and dose length product (DLP). So, for dose measurement, phantom was placed in iso-center where its axis was paralleled to the gantry rotation axis in the center of scan plane. Then ionization chamber was placed in dosimetry hole and other holes were filled with PMMA plugs. Measurements have been done 3 times. According to questionnaire, a single axial scan was performed. This procedure was repeated for all phantom holes and then CT dose quantities were calculated.
The mean value, third quartile, standard deviation (SD) and p-values of data were calculated using SPSS software version 18. Finally, for each examination, the third quartile of CTDIw was considered as DRL.

RESULTS

The third quartile of CTDIw obtained for brain, chest, sinus, and abdomen and pelvic examinations were 59.5Gy, 7.8Gy, 17mGy and 11mGy respectively. These values were suggested as DRLs. Mean, range and standard deviation of CTDIw and DLP are shown in Table1. The obtained doses for identical examinations in the seven centers in this study varied from each other. This difference can be due to using different protocols, different radiation parameters or the difference in CT machine manufacturers.

TABLE 1. MEAN, RANGE AND STANDARD DEVIATION OF CTDIw AND DLP

Examination	CTDIw (mGy)			DLP (mGy)		
	Mean	Range	SD	Mean	Range	SD
Brain	42.16	15.8-73	21.2	535.6	194.4-981	291
Sinus	11.19	3.8-25.8	7.68	124	41.8-167.7	49.8
Chest	7.94	4.51-16.3	3.8	193	131-342	72.8
Abdomen and Pelvic	10	7-16.3	3.12	393.3	283.6-486	67.9

CONCLUSION

This study meant to contribute to determining the national DRL of CT examinations in Iran. The results for all examinations were lower than international reference levels.
Wide variation in CT doses between centres for identical examination has a large potential for optimization of examinations. So, the survey is ongoing, allowing practices to optimize dose delivery.

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110-ESTABLISHING A DIAGNOSTIC REFERENCE LEVEL FOR MAMMOGRAPHIC EXAMINATION IN NORTHERN IRAN

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BACKGROUND

Due to the increased prevalence of breast cancer in northern Iran, it is recommended to use mammography screening method to diagnose breast cancer in the early stages [1]. However, there is no diagnostic reference level in mammography examinations for effective management and reaching optimal dose for the patient (according to ALARA Principle), along with preserving the quality of diagnostic images in Iran [2]. So this study aimed to determine the diagnostic reference level for optimal dose in mammography examination in Northern Iran and to compare it with international standards.

METHODS

In this study, mammography centers were selected in four Referral Hospitals in Northern Iran to study two common mammography views. Questionnaires which included specification of devices, radiation parameters and patient information were sent to the centers and their data were collected. In the next step, the RTI dosimeter of the Piranha Barracuda model was used to perform dosimetry of different centers devices. The obtained data were then analyzed to determine the diagnostic reference level, and the quantities of Entrance Surface Dose (ESD) and Mean glandular dose (MGD) were calculated.

RESULTS

The average value, third quartile and 95% frequency distribution of ESD in the four centers in the MLO view were 16.81, 2.81, and 37.73, and 15.84, 18.94 and 36.88 in the CC view, and totally calculated as 16.71, 20.65 and 36.92 respectively. Also, the average value, third quartile and 95% frequency distribution of MGD in the four centers in the MLO view were 3.14, 4.12 and 7.06 mGy and 2.97, 3.98, and 6.91in the CC view and totally calculated as 3.12, 4.02 and 6.95 mGy.

CONCLUSION

One of the centres in this study had a higher MGD value than the international standard values and other centres had values somewhere around the standard level. Effective use of digital system software and implementing Quality Assurance tests can have an effective role in reducing the dose of patients.

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111-RADIATION PROTECTION ON THE RADIO-ACTIVE MINERALS PROCESSING IN VIET NAM - THE DIFFICULTIES AND CHALLENGES

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As you know, Vietnam is a developing country so the exploitation of ore mines is a very important issue in economic development. During the mining process, a large amount of radioactive waste containing natural radioisotopes (NORMs) is generated and thus will seriously affect the health of the people and the environment. In Vietnam, now we have a few legal documents to manage this Radiation protection on the minerals processing but it is still not comprehensive. In my presentation, I will present some contents of the following:

1. Regulations related to Radiation protection in Viet Nam contain:
   a) Atomic energy law 2008
   b) Decision 452/ QĐ- TTg on April 12/2017
   c) Environmental protection Law 2014
   d) Circular 19/2012/TT-BKHCN
   e) Circular 22/2014/TT-BKHCN
2. Management of radioactive waste in research and processing of Uranium ore in Viet Nam
3. Radiation protection in mining and processing of coastal placer minerals in Viet Nam
4. Radiation protection residues in Viet Nam
5. Radiation protection in some other industry which Viet Nam company are implementing as ZOC production, DAP fertilizer production
6. The difficult and Challenger on Radiation protection in Viet Nam

112-LONG-RANGE STRATEGY OF IRAN NATIONAL PLAN FOR INDOOR RADON CONCENTRATION SURVEY

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ABSTRACT

This paper introduces the framework of the Iran national plan for survey of radon concentration in homes which is commenced in 2015 by National Radiation Protection Department (NRPD) of Nuclear Regulatory Authority (INRA) and cooperation of Ministry of Health and Medical Education (MHME). The aims of this study are to obtain an estimate of the proportion of the Iranian population living in homes with radon gas levels above the ICRP guideline of 300 Bq/m^3 (as well as >200 Bq/m^3 and >100 Bq/m^3), to identify previously unknown areas where radon gas exposure may constitute a health risk, and to build, over time, a map of indoor radon gas exposure levels though the country. Participants for the study are selected and recruited via a random selection method using the postal codes. Homes are going to be sampled across Iran and a long-term (six-month) radon test is going to be conducted by participants during spring and summer or fall and winter. By sampling in all rural and urban areas as opposed to only few large cities, an estimate of the geographic distribution of radon levels across the country is planned to be obtained. By completion of the project across Iran, it would be practical to define national (and/or provincial) reference level(s) based on the reliable collected data. Furthermore, the survey will help to define potential radon prone areas that are still unknown. Another outcome of the study can be searching for any possible correlations between radon levels and home characteristics based on reviewing and analyzing the questionnaire data from the full study.

Keywords: National Plan, Indoor Radon Concentration, Reference Level, Radon Map.

INTRODUCTION

Radon is a noble radioactive gas that is produced by the decay of uranium. Since uranium is found in all rocks and soil radon gas can be found naturally throughout our environment. Radon does not pose a health risk in outdoor air because it is greatly diluted in atmosphere. In contrast, radon can concentrate in closed spaces like homes and workplaces exposing inhabitants to high dose levels [1]. It is proven that prolonged exposure to high levels of radon increases the risk of developing lung cancer. In 2015, NRPD began to collaborate with MHME to plan and conduct a national radon survey in the country. The National Indoor Radon Survey, consists of five components:

a)A National Radon Laboratory (NRL) in the NRPD to support radon testing projects and provide calibration services to companies and organization who have active or passive radon measuring devices; b) Radon testing projects to assess the levels of radon in residential levels across the country;c) A cooperation framework to divide the duties of the survey to planning, evaluating, and analytical and laboratorial tasks (NRPD), and field operations (MHME); d) Research projects on the health issues of radon and on optimized methods (for Iran) to reduce public exposure to indoor radon; e) A program to improve awareness of homeowners as well as public health practitioners about radon health issues and methods of mitigation of exposure to radon.

Sampling Methodology

The sampling strategy was designed based on stratifying the territory of Iran into provincial units (31 units). To reach to minimum (and comparable) statistical power in every unit, the minimum number of samples was independently defined for each province. The principals that the strategy follows include:

a) only residential dwellings are included in the survey;
b) all homes of a province are put in the same pool;
c) postal codes are used for random sampling;
d) there is no limitation for floor number or location of dwellings;
e) The state of being a tenant inhabitants is difficult to be defined from the codes. They are not excluded from the study.

The total number of homes of the country and the ones of every province is according to the 2011 national census. The guidelines that were taken into account in defining the number of samples for every province included:

a) The number of samples to be taken from each province should be relatively (not absolutely) proportional to its number of homes to be a good representative of its population (say 0.1% of homes). Imposing an upper limit like 2500 samples for provinces with a very dense urbanized population like Tehran (~90% live in cities) seems rational;
b) The statistical strength obtained for every province should meet a minimum acceptable degree. To this end the standard formula (1) was adopted for calculating the minimum number for samples that are needed for every province. The presumptions were:
    (i) indoor radon concentration follows a log-normal distribution, thus, the sample number is greater; the accuracy of the results is greater;
    (ii) practically, not more than 5%-15% of the dwellings in a country/province is recommended to be above the national/provincial reference level (RL);
    (iii) not only the frequency distribution of radon concentration but also the potentially high-radon zones must be defined (this was not focused on in the pilot phase);

    Based on above considerations, p=0.1 and d=0.02 (degree of precision) were set for defining the rate of >RL in (1).
c) because of the possible loss of samples due to distribution, returning, physical damages, and laboratory failures, an excess rate of 20% was applied on the minimum sample number;
d) considering the number of rural districts in every province, an extra number of samples were considered to assure that each such units receives a minimum of, say, 5 samples. In other words, low population areas may need to be oversampled (disproportionate random sampling) to avoid missing such areas due to a sampling strategy that only focuses on population;
e) a rate of 5% was considered for duplicate dosimeters (for QA purpose);
f) Some control detectors to be used to verify if 6-month and 12-month radon concentrations are too different in a stratum. 2-3% of samples are enough and better to be installed in dwellings that show a high interest for cooperating with the survey (this was not done in the pilot phase).

$$n = \left[\frac{z^2{}_{1-\frac{\alpha}{2}} \times p(1-p)}{d^2}\right] \qquad (1)$$

for a confidence level of 95% ($z1-\frac{\alpha}{2}$ = 1/96), n = 864. Such a sample size from each province, will define the top 10% high-radon dwellings with a 95% confidence level and -+2% error (that falls in the range of 8% - 12%).

Quality Assurance [2]

The QAP of the plan consists of the following elements:

a) All procedures are defined and written (standard operating procedures);
b) The minimum detectable concentration (MDC) of the technique is defined and controlled whenever new batches of Lexan films, chambers, air filters, etc. are used;
c) INRA is equipped with a 'System for Test Atmospheres with Radon (STAR)' that took part in an international intercomparison exercise in 2011 and obtained good marks [3];
d) Performance tests and blind spikes: INRA acts as both the licensing body and service laboratory for the plan, thus, relies only on the results that were obtained by its STAR in 2011 intercomparison[3];
e) Calibration of detectors and continuous radon monitors through the STAR and related standard procedures;
f) Control of the background of passive (Lexan) detectors as well as the continuous monitors are routinely done;
g) Detectors are sealed from radon from the time they become ready to use until they are opened at houses (inside radon resistance sealed bags);
h) Detectors are sealed from radon from the time they are collected from houses until they are opened at the laboratory for reading (inside radon resistance sealed bags);
i) Field background control measurements: 2-3% of radon-sealed detectors are kept unused as 'blank detectors' during every distribution. They must undergo the same handling and storage conditions except that they are not distributed among houses;
j) Duplicate measurements (5% of samples are duplicated);

The criteria for rejection of returned detectors are: a) Any signs of unusual application of detectors, e.g. if the candidate declares that the detector is put inside the cabinet; b) Unusual condition of returned detectors, e.g. muddy detectors, manipulated detectors,… c) Raptured filters; d) Physical damage to detectors.

CONCLUSION

The purpose of the plan is gathering long-term (six months or longer) indoor radon test results from across Iran in order to:

a) To obtain the frequency distribution of radon concentration in homes of Iran and estimating the proportion of the Iranian population living in homes with radon gas levels above 100, 200, and 300 Bq/m3;
b) To establish national reference level for radon concentration
c) To provide provincial radon data of enough statistical strength so that decision making (setting independent reference and/or action levels) for every single province is possible, if necessary;
d) To identify previously unknown radon prone areas (some of such areas have already been identified in the north of Iran); and
e) To develop a map of indoor radon concentration.

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113-DOSE ASSESSMENT OF OCCPUPATIONAL EXPOSURE INDUCED RADON PROGENY INHALATION IN ALISADR TOURIST CAVE

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ABSTRACT

The concerns over occupational exposure of workers from radon in tourist caves has been carried out in the most famous cave of Iran. This paper presents a survey performed to quantify the annual dose of exposure in this cave environment. The measuring plan was arranged such that to measure the radon levels at representative sites in each season over a period of one year to estimate the radiation doses. The measurements were carried out by use of integrating radon and gamma dosimeters based on lexan polycarbonate (PC) solid state nuclear track detector (SSNTD) and TLD (GR-200) respectively. In Alisadr cave average annual values of radon concentration were measured to 1499 Bq.m^{-3} and average annual radon dose and gamma dose were 37.6 and 1.38 mSv respectively. Discussion and interpretation on the level of radon which could lead to occupational exposure were made considering the recommendation of ICRP 103 (2014) as well as working hours and the employment period of workers.

Keywords: Radon levels, PC detectors, Cave, Annual dose, Health hazard.

INTRODUCTION

Health hazard with special consideration to the inhalation of radon and its progeny have been a serious concern in recent years[1-4]. The International Commission on Radiological Protection in its publication 65, 103, 137 and also BSS have provided recommendations on reference level of radon concentration in workplace to be 1000 Bq m^{-3}. Radon concentration varies widely in caves due to mainly the different ventilation conditions and geological structures [5-7]. Alisadr is the world's largest water cave in Iran which attracts thousands of visitors every year. The walls can extend up to 40 meters high, and it contains several large and deep lakes. It is a highly recommended destination for tourists from all corners of the world.

MATERIALS and METHODS

In this study radon measurement was performed by use of Radon Diffusion Chambers using electrochemically etched track detectors (Lexan polycarbonate). The counting of registered alpha tracks was carried out with an automated image analysis system [8-9]. The counting unit consisted of nuclear track counting software (NTCS) and related hardware (e.g. a high resolution scanner with transparency adapter). Gamma dose measurement was carried out by use of LiF: Mg, Cu, P (GR-200) TL dosimeter. A piece of Lexan film with dimension of 3×3 cm^2 inside the radon diffusion chamber and TLD card with two pellets in a holder were put in the representative sites to measure radon concentration and gamma dose level at each site. Preliminary measurements were performed by using active radon monitor (Pylon AB-5) to select representative sites. Calibration of the radon detector was carried out by passing known concentration of radon from a Pylon radon source model Rn-1025 through the cell.

RESULTS

During a year, the seasonal average radon concentrations in Alisadr cave ranges between 490 and 2500 Bq/m^3 (in spring, summer, fall and winter are 2442, 1883, 1179 and 490 Bq/m^3 respectively), associated with reasons cave architecture and meteorology particular cave topography, season-related cave ventilation, and complex tectonic and geological settings surrounding each location. The seasonal average Radon concentration were shown in figure 1. According to ICRP 137 part 3, the effective dose (mSv/year) of tour guides based on the average radon concentration inside the cave air (Bq/ m^3), the average radon equilibrium factor between radon and the decay products (F=0.4), the time spent inside cave (h/yr) for part-time cave workers (1,760 h yr^{-1}), and full-time cave workers (2,000 h yr^{-1}) and DCF = dose conversion factor for workers (mSv/WLM) were calculated 15.33, 11.82, 7.40 and 3.07 mSv/season for spring, summer, fall and winter respectively.

CONCLUSION

It is normally expected in workplaces with radon levels above 1000 Bq m^{-3} to reduce the radon concentration, usually by improving the ventilation. Increasing the ventilation in tourist caves is not possible since it affects the delicate balance of carbon dioxide, partial pressure, temperature and humidity, with the risk of damage to cave decoration. The alternative is to designate the workers in such areas as radiation workers and to maintain detailed radiation exposure records in order to demonstrate compliance with regulatory radiation dose limit. At present, there are no regulations covering workplace exposure to natural radiation but the policy of the cave management to restrict individual hours spent in the Alisadr cave to less than 500 hours per year has constrained the radiation doses to less than 10 mSv per year.

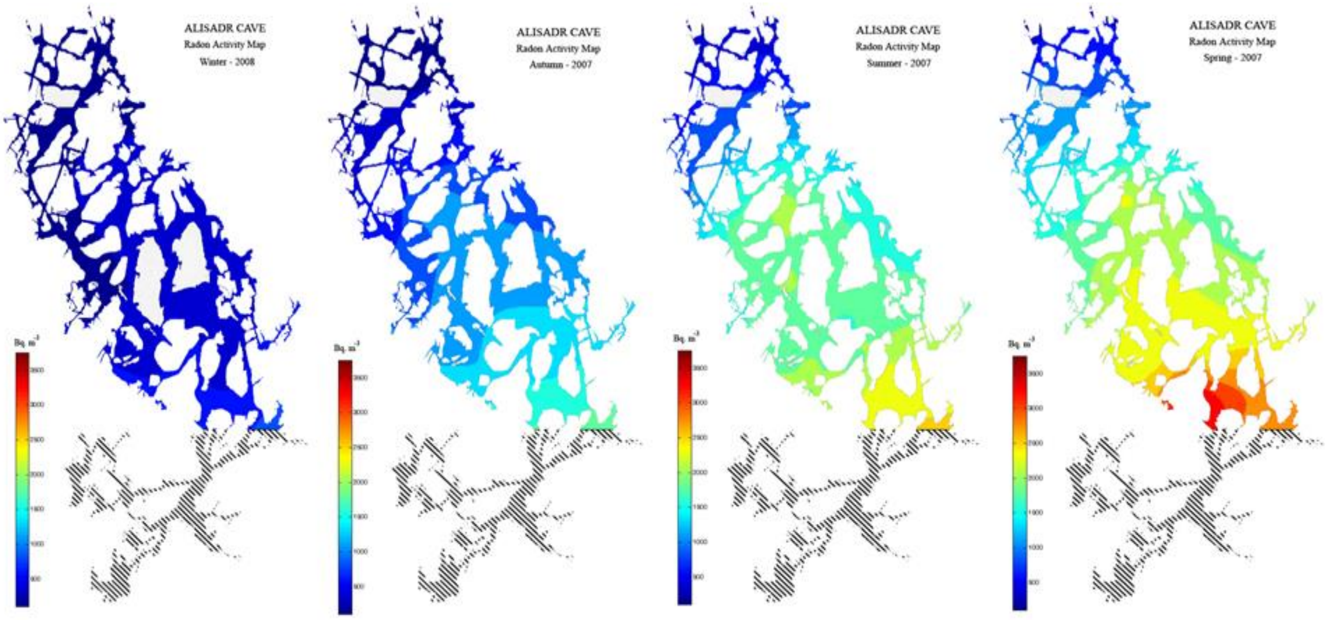

Fig.1. Radon concentration levels map for Alisadr cave

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114- AN ASSESSMENT OF THYROID DOSE REDUCTION IN INTRAOAL RADIOLOGY BY CHANGING IN CILLIMATION GEOMETRY

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ABSTRACT

Radiology imaging as a diagnostic solution for tooth disorders and decays can have irreversible effects in case of noncompliance with radiation protection issues. radiation protection of thyroid as a sensitive and vulnerable organ especially for children, which have more time to develop the symptoms enhanced by radiation, have become a concern. our demand among this study is thyroid absorb dose reduction in intraoral radiography via changing at collimator geometry. This study has done by changing in collimation of the intraoral radiography device from circular to rectangular mode and applying TLD for dosimetry in 32 individual patients with intraoral radiograph demands. Patient dose measured after dividing patient into two equal groups (Intervention and Control). Three TLD employed per person and their thyroids three- main parts absorb dose were calculated. Data were calculated and analyzed. After TLD irradiation and readout, result reported over 60 % declination in thyroid absorb dose. On the other hand, this issue could improve image quality by decreasing radiation field aiming To reduce scattered radiation and limiting it to sensitive area of radiographic sensors.

Keywords: Thyroid Dose, Dental Radiography, TLD, Collimation Geometry.

INTRODUCTION

Today with the advancement of radiology and with its importance in the diagnosis of diseases and also the convenience of it, we are seeing an increase in the number of refer to imaging centers. But that's could be a problem if you do not follow the principles of protection against ionizing radiation and can contain irreparable effects [1-5]. The purpose of this study reduce the risk of thyroid cancer especially in children and women as the More vulnerable population against ionizing radiation.also this research checking effect of changing common circular collimators to the rectangular collimator to reduce the absorbed dose of the thyroid gland in dental radiography [5-9].

MATERIALS AND METHODS

In this study, the absorbed dose of thyroid was evaluated by placing three thermoluminescence dosimeters (TLD chips) on the three main parts of the thyroid (Left and right lobes and istemus). Usual dental radiographs were done with common circular collimators. After irradiation dosimeters, they were taken to the relevant center for reading. This procedure was performed for 32 patients referred for maxillary teeth imaging. Because in routine imaging techniques of the mandibular teeth, the thyroid will not be exposed to direct radiation. Therefore, only the imaging of maxillary teeth was examined. This process was repeated by changing the collimation to a rectangle. 16 patients with circular and 16 patients with rectangular collimator, were irradiated. Eventually doses received were calculated and compared. Figure 1 shows the circular and rectangular collimator of dental Radiology device which used.

RESULTS

After calculation the TLD chips, doses were compared and the results showed that after changing The collimation from circular to rectangular, a reduction of about 60% in the amount of absorbed thyroid dose can be achieved. The average dose absorbed by the three main parts of the thyroid in the rectangular and circular collimation was compared in figure 2. The main reason was the reduction of direct radiation to thyroid because of its limitation to sensitive surface of the intraoral radiographic sensors.

CONCLUSION

In this study, the efficiency and performance of rectangular collimators with conventional circular collimators were compared, and the results reported a 60% reduction in the absorbed dose in the thyroid .This findings demonstrate the effectiveness of this method in line with the objectives of the ALARA principle. We hope that the results of this study and other similar research, also Sensitivity and importance of the thyroid gland and an increasing trend towards diagnostic imaging techniques, convert the propensity of rectangular collimator usage to a scientific and ethical principle due Tto the high sensitivity of the thyroid gland and frequent referral for dental radiographyduring. It seems changing the collimator from circular to rectangular could to be necessary. The effect of this action is too practical that many scholars and experts consider its effects similar to using of lead collar.

FIG. 1. Circular and rectangular collimator of dental Radiology device.

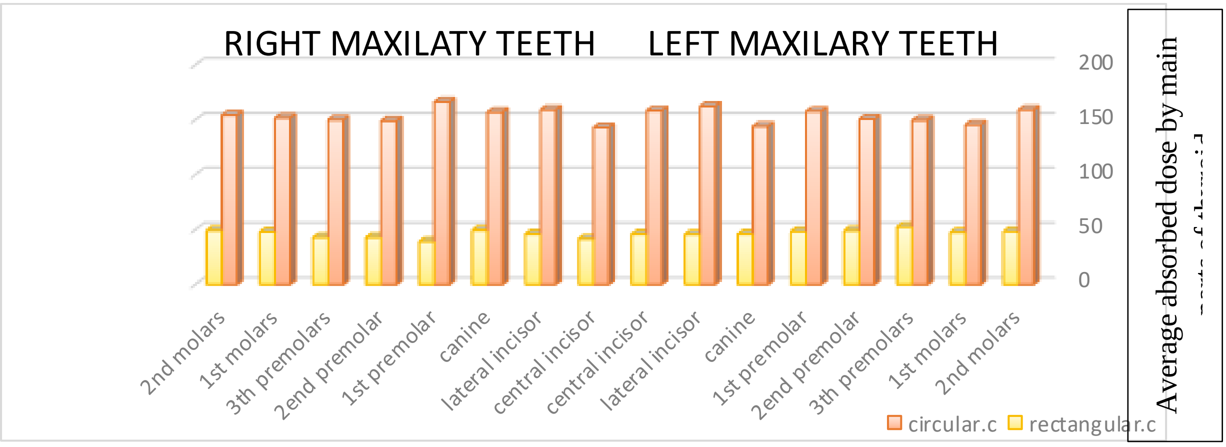

*FIG. 2. Comparison of thyroid dose by using circular and rectangular collimator **in intraoral radiology.***

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116-TYPICAL RADIATION DOSES FOR CHILDREN UNDERGOING COMMON COMPUTED TOMOGRAPHY EXAMINATIONS IN SUBSAHARAN AFRICA: A CASE STUDY OF UGANDA CANCER INSTITUTE

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Computed tomography (CT) provides disproportionally higher ionizing radiation in comparison with other imaging modalities. (1). However, the shortage of advanced medical equipment that uses less or no radiation such as ultrasound and MRI in sub-Saharan Africa has fueled the increased use of CT. This is radiation safety concern especially for children with cancer who not only have 10 times higher chances of developing ionizing radiation-induced cancers and cellular mutations, but are likely to undergo several CT scans as a part of the process for management of cancer (2-6). Therefore, optimization of paediatric imaging is of particular importance due to their increased radiation sensitivity, higher effective dose, and longer life expectancy compared to their adults' counterparts (2, 7).

The diagnostic reference level (DRL) has been proven to be an effective tool for dose optimization during medical exposure of patients. Unfortunately, many countries in sub-Saharan Africa have neither established local nor National DRLs.

The purpose of this study was to determine the typical radiation doses for most common CT examinations for children at the Uganda cancer institute (UCI), and compare them with the published DRL data

Methods: This was a cross sectional descriptive study at UCI. Data were collected for 109 children who underwent head, chest and abdomen, and abdomen CT scan examinations using a 16 multislice Phillips big bore CT scanner between November 2018 and February 2019. Dose data (CT volume index (CTDIvol) and dose–length product (DLP) was recorded to calculate a median CTDIvol and DLP value. The rounded 75th percentile was used to calculate typical DRLs for the facility. Results were compared with published DRL data.

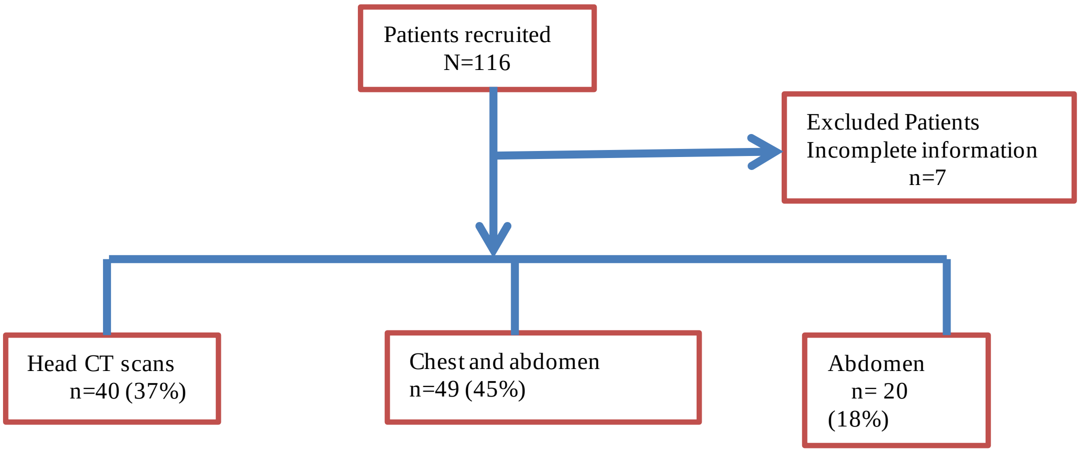

Fig.1: Study profile flow diagram and anatomical sites scanned

Results: The commonest scanned anatomic site was chest and abdomen (45%) followed by head (37%), see Fig1. The median age of the participants was 6 years, with a slight predominance of the male gender accounting 51%. The typical local CTDIvol (mGy) and DLP (mGy cm) for CT head across all age groups (42.5 and 1879.9 -2359.6 respectively), chest and abdomen across all weight groups (7.1 and 600.6mGy.cm to 935 respectively), and abdomen (9.45, 11.8, 10.15 and 577.85 -1241.5 respectively) in children with body weight (kg) of 5-14, 15-29, and 30-49 respectively.
The median DLP and scan length of the head increase as the age increases. However, the median CTDIvol values across age groups remain constant, see Table 1.

TABLE 1: MEDIAN VALUES OF CTDIvol, DLP, AND SCAN LENGTH OF THE HEAD AS A FUNCTION OF AGE

Age groups	Head		
	CTDIvol (mGy)	**DLP (mGy.cm)**	**Scan length (cm)**
< 1 year	42.5	1879.9 (1610 - 2777)	21.8(14.5-33.5)
1-4 years	42.5	2007.3 (664 – 2314)	23.250(18.3_58.7)
5-9 years	42.5	2211.1 (1024 - 2578)	26.3 (22.3-54.2)
10-14 years	42.5	2321.55 (1159 - 2449)	25.5 (22.0-28)
15-17	42.5	2359.6 (1150 - 2483)	26.5 (24.6-62.2)

The median values of CTDIvol, DLP, and scan length of chest and abdomen, and abdomen during all phases increased with increasing body weight, see Table 2.
The75th percentiles of CTDIvol and DLP and the can length at UCI were higher when compared with published values European values (8), Table 3 and Table 4 (9).respectively.

TABLE 2: TYPICAL RADIATION DOSES FOR CHILDREN UNDERGOING CT EXAMINATIONS AT UCI AS A FUNCTION OF BODY WEIGHT AND ANATOMICAL SITE

Weight groups (kg)	Anatomical sites and CT dosimetry					
	Chest and Abdomen			Abdomen		
	CTDIvol (mGy)	DLP (mGy.cm)	Scan length (cm)	CTDIvol (mGy)	DLP (mGy.cm)	Scan length (cm)
5-14	14.2 (14.2-21.3)	600.6 (513-830)	35.769 (25-70.2)	18.9 (14.2-23.6)	577.85 (424-732)	27.375 (26-28.4)
15-29	21.3 (14.2-21.3)	828 (526-1258)	44.7 (24.3-67.0)	23.6 (11.8-141.6)	1114.15 (410 -1693)	42.475 (32.5-51.7)
30-49	17.75 (14.2-21.3)	935 (742-1163)	57.2 (22.2-62.1)	26.2 (17-35.4)	1241.5 (865-1618)	40.775 (37.8 – 43.8)

TABLE 3: TYPICAL CT DOSES FOR CHILDREN AT UCI COMPARED WITH EUROPEAN DRLS

Anatomic site	Age group or weight group	UCI		EDRL*	
		CTDIvol (mGy)	DLP (mGy.cm)	CTDIvol (mGy)	DLP (mGy.cm)
Head	1-6 years	42.5	2200.68	40	504
	>6 years	42.5	2359.6	50	650
Abdomen	5-<15 kg	9.45	577.85	3.5	45
	15-<30 kg	11.8	1114.15	5.35	118
	30-<50 kg	10.15	1241.5	7.3	151

***** European Diagnostic Reference Levels (8)

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TABLE 4: COMPARING SCAN LENGTH OF CT EXAMINATIONS AT UCI WITH AUSTRALIAN HOSPITAL

Weight group (kg)	Chest and abdomen		Abdomen	
	UCI	WCH*	UCI	WCH*
Scan length(cm)				
>7-11	37.7	21	-	-
>11-22	44.7	25	44.325	21
>22-40	57.425	30	48.525	21

*Women and Children Hospital in Adelaide, Australia (9).

Conclusion: The radiation doses in terms of CTDIvol, and DLP as well as the scan length for common CT examinations at UCI were higher compared to other similar published values, which is radiation safety concern.

Recommendations:

-Establishing pediatric CT scan examination protocols based on body weight and age at UCI.
-Studies with ample sample size including all anatomic sites and setting DRL based on clinical indication.
-Strengthening optimization of medical exposures during pediatric CT examinations

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117-DEVELOPMENT OF IN-HOUSE REFERENCE ACTIVATED CHARCOAL CARTRIDGE STANDARD FOR AIRBORNE RADIOIODINE MONITORING IN NUCLEAR MEDICINE

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Airborne radioiodine (^{131}I) can be founded in working areas of nuclear medicine in several areas for example radionuclide laboratory, treated patient room and waste storage area. This airborne radioiodine can cause a potential health hazards to radiation worker from internal exposure via inhalation process. Consequently, monitoring of airborne radioiodine is recommended in the IAEA (International Atomic Energy Agency) safety standards [1]. The monitoring process is included the sampling of air from the breathing zone with adsorption media using charcoal cartridge and filter paper. In nuclear medicine, the main component of airborne radioiodine is in the gaseous form which can be adsorbed by activated charcoal cartridge [2]. To obtain accurate measurement, the efficiency calibration of the detector is needed to determine using either a commercial standard cartridge or a known activity cartridge with same source geometry. Therefore, the aim of this work was to develop and evaluate in-house reference activated charcoal cartridge standard using scintillation detector.

The in-house standards were fabricated in 4 patterns (as illustrated in Fig. 1) using standardized ^{131}I in form of sodium iodide solution. This standardized ^{131}I solution was measured with a 4π well-type ionizing chamber (Centronic model IG12/N20, United Kingdom) at the National Standard Radioactivity Laboratory (NSRL) of the Office of Atoms for Peace (OAP) of Thailand.

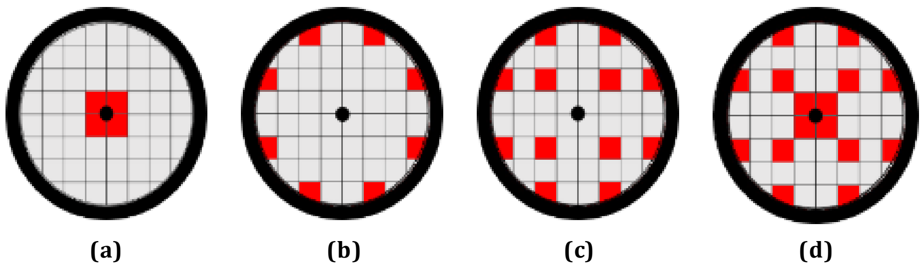

FIG. 1. In-house reference activated charcoal cartridge standard in 4 patterns.

The volume and activity (192 μL, 10 kBq) of standardized ^{131}I solution was similar for each pattern. Then, the standardized radioactive iodine solution was dropped on the activated charcoal cartridges (F&J model TE3C, FL, USA) using micropipette. Then, the counting efficiency was determined using a gamma spectroscopy system with NaI(Tl) scintillation detector (Alpha Spectra Inc, CO, USA) at the Division of Nuclear Medicine, Faculty of Medicine Ramathibodi Hospital. The counting efficiency shows in Table 1.

TABLE 1. COUNTING EFFICIENCY OF IN-HOUSE REFERENCE ACTIVATED CHARCOAL CARTRIDGE STANDARD WITH SCINTILATION DETECTOR

Counting Efficiency (% ± SD)			
Pattern (a)	Pattern (b)	Pattern (c)	Pattern (d)
21.00 ± 0.13	18.80 ± 0.20	18.71 ± 0.30	21.26 ± 0.42

As expected, there were difference in the counting efficiencies between each pattern. The highest efficiency was found in pattern (d) which radionuclide was distributed all over the surface of the cartridge. In addition, the counting efficiency of pattern (a) was also high. This could be explained by the assumption that the solid angle of pattern (a) was less than other patterns, hence, more gamma photons could be collected. In practical, it is still questionable how is the actual distribution of airborne radioiodine in the hospital, however, many researchers believed that it should be homogeneous distribution.

In conclusion, this work demonstrated that the in-house development of a reference activated charcoal cartridge standard was possible. Patterns (a) and (d) could be produced for using as the reference standard for calibration of scintillation detector for airborne radioiodine monitoring. However, other radionuclides which are longer half-life and gamma energy close to ^{131}I might be applied with the proposed method for long-term used for example ^{133}Ba (half-life of 10.5 years, 356 keV with 62 % abundance).

ACKNOWLEDGEMENTS

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118-PREDICTING LIVER TRANSAMINASES ALTERATIONS FROM LOW-DOSE MEDICAL RADIATION EXPOSURE THROUGH MACHINE LEARNING (ML) APPROACH

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BACKGROUND & OBJECTIVES

Ionizing radiation (IR) is a cancer causing agent and can alter several biological effects via. oxidative stress [1]. The presence of an oxidative stress in the body can develop liver diseases [2]. The liver is a radiosensitive organ [3], and the oxidation in liver enzymes can occur [4]. The study examined the hepatic function in medical radiation workers of INMOL (Institute of Nuclear Medicine and Oncology) hospital, who are chronically exposed from low IR doses. We applied machine learning (ML) algorithm Random Forest (RF) to predict alterations in liver enzymes from annual average effective doses (AAED) and other cofactors. Machine learning is a method of artificial Intelligence (AI), in which the system learns the given patterns from the data via. learning and training to make a model, which is useful in deciding diagnosis and treatments in medical science [5].

MATERIALS & METHODS

90 radiation personnel from Radiology and Nuclear Medicine departments were included. A high-capacity thermoluminescent dosimeter was used to assess the whole-body AAED (mSv). The liver function tests (LFT) were conducted through informed consents in the biochemistry lab of INMOL hospital. A supervised learning tree model Random Forest (RF), was trained and cross-validated (5 folds) on the given data (AAED, liver enzymes, age, gender). The model was bagged with 100 iterations and base learner with Seed=1. The basic RF model [6] is a non-parametric general purpose ensemble machine learning (ML) algorithm [7] and a classifier tree for constructing a forest of random trees [8]. Two classes of liver enzymes alanine transaminase (ALT) and aspartate transaminase (ALT) were made according to 'above normal range' values.

RESULTS

The mean age of the radiation exposed workers was 43 years. The AAED in the range 0.07 – 1.15 mSv during 2014-2019. Half of the radiation workers were found with high levels of ALT (mean: 61.6 U/L). High levels of AST (mean: 38.83 U/L) were observed in 20% exposed workers. Random Forest, achieved 90% and 96.6% accuracies in ALT and AST prediction models, respectively. RF also achieved reduced number of errors and good kappa statistics (i.e., 79% & 89%) (TABLE 1).

DISCUSSION

Chronic exposure of IR can induce late health effects. National Research Council's BEIR-V Committee has mentioned that a long-term exposure to radiation can induce a liver cancer [9]. Although, the AAEDs (mean: 0.255 mSv) was within the limit (<20 mSv), but high levels of ALT can be observed in 50% of the radiation exposed workers. There are reports in which assessment of Radiation-induced liver disease (RILD) in patients receiving radiotherapy are mentioned [10]. There is also need to regularly monitor the hepatic function in occupational radiation exposed personnel. The application of machine learning based (ML) models can provide us fast monitoring of biochemistry to point out an earliest assessment in case of any alterations. We used ML based random decision forest algorithm which can predict up to 90% accurate results of alterations in liver transaminases. The chronic use of radiation can induce some changes in liver metabolism in many occupational-radiation exposed groups [11]. A total body irritation can impact other body organs including liver. When a liver receives radiation doses, the "upregulation in the genes of main proinflammatory chemokines occurs from the activity of proinflammatory cytokines" [12]. Hepatic metabolic alterations occurred with the radiations (8.5 Gy), which also lead to the radiation-induced carbonylation of associated liver enzymes [4]. Some studies reported that high levels of liver enzymes from thorium exposures in occupational workers [13]. A prevalence of liver cancer was found by among the medical radiation workers from low-level radiation doses [14].

TABLE 1. ACCURACIES & ERRORS FOR RANDOM TREE MODEL

ALT					
Correctly Classified Instances	**Kappa Statistics**	**Mean Absolute Error**	**Root Mean Squared Error**	**Relative Absolute Error**	**Root Relative Squared Error**
90%	0.7982	0.1696	0.2526	34.0503 %	50.5894 %
AST					
96.6667 %	0.898	0.0908	0.1618	27.9296 %	40.3842 %

ALT								
Class	**TP Rate**	**FP Rate**	**Precision**	**Recall**	**F-Measure**	**MCC**	**ROC**	**PRC**
A	0.857	0.063	0.923	0.857	0.889	0.800	0.976	0.974
B	0.938	0.143	0.882	0.938	0.909	0.800	0.976	0.980
AST								
A	0.972	0.056	0.986	0.972	0.979	0.898	0.997	0.999
B	0.944	0.028	0.895	0.944	0.919	0.898	0.997	0.983

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119-STRENGHTENING THE RADIATION EMERGENCY PREPAREDNESS PROGRAM IN SAUDI ARABIA: STRATEGIC APPROACH

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ABSTRACT

Saudi Arabia's 20 year development plan includes the national upgrading of medical facilities in terms of radiation uses in medicine. As the uses of radiation increase, there is a need for a strong regulatory infrastructure to control the uses and provide the means to address radiological emergencies. The aim of the presentation is to share the Ministry of Health strategic approach to planning and responding from small to large scale radiological emergencies.

INTRODUCTION

The Saudi Council of Ministers issued Resolution number 406 dated 10April 2018 approving the Law of Nuclear and Radiological Control to be implemented by the new Nuclear and Radiological Regulatory Commission (NRRC). The law includes the Emergency Preparedness and Response plan.

The Ministry of Health (MOH) is the largest user of ionizing radiation in which there is a total of 21 radiotherapy, 21 PET/CT, 55 SPECT/CT, 35 SPECT and Gamma camera and 77 DEXA units being used by government and private hospitals. A total of 313 for industrial radiography and nuclear gauge units.. The MOH national plan for equipment upgrade envisions the increase of more radiation emitting and producing facilities and in addition industrial x-ray units which necessitates the development of a relevant and achievable emergency response plan [1,2]. Threats of radiation emergencies are not only internal but also from nearby regions outside Saudi Arabia.

MOH INITIATIVES FOR THE IMPLEMENTATION OF THE NEW LAW

The signing of the Law of Nuclear and Radiological Control addresses the issues on adopting the IAEA Basic Safety Standards on safety during emergency exposure situations and provision of effective responses during radiological emergencies[3]. The MOH is proactively linking radiological emergencies to all applications of ionizing radiation in a large scale. It has developed two new guidelines for radiological emergencies relevant to medical management namely: "A Guide to Medical Response during Radiation Emergencies" and "A Guide to Medical Management during Radiation Emergencies". The first Guide discusses the technical requirements for response to emergencies and management of persons and the environment while the other Guide is focused on the clinical management of injured individuals.

GUIDE TO MEDICAL RESPONSE DURING RADIATION EMERGENCIES

The Guide is highlighted with the strict requirement on identification of the facility that will respond to radiological emergency, assignment of a radiation emergency physician and a radiation emergency room. MOH introduces the classifications of injuries during radiation emergencies, the key persons and organizations and their responsibilities and the preparations of hospital facilities to be able to adequately respond (Fig.1), equipment needed and systems and procedures. A well-structured standard reporting system is enforced for all MOH response hospitals and teams from receiving to final reporting and conclusion of the emergency.

GUIDE TO MEDICAL MANAGEMENT DURING RADIATION EMERGENCIES

The Guide provides the basic information on the different biological effects of ionizing radiation that could possibly occur on persons exposed to radiation and could have injuries. Requirements for immediate and accurate medical provision and management is emphasized with proper handling of persons externally and internally exposed to radiation during emergencies [4]. The Guide includes management of patients a with life-threatening condition from the transfer location to receiving facility with the control of contamination.

The Guide provides the requirements for handling gross contamination of the injured person with radiation protection practices to be performed by responders. Specified actions for declaring the conclusion of the emergency procedure with assessment of the response in terms of timeliness and appropriateness of the medical action for patient management and the chain of commands for effective execution of the response are included in the Guide.

CONCLUSION

The issuance of the two Guides on medical response and management of injured and contaminated patients are the main strategies of the Ministry of Health in strengthening capabilities to respond to radiation emergencies. Proper dissemination of the Guides through drills and training are tools for their implementation and success for enforcing the Law of Nuclear and Radiological Control. These two Guides were published in English (Fig. 2) and the Arabic translation is in the pipeline.

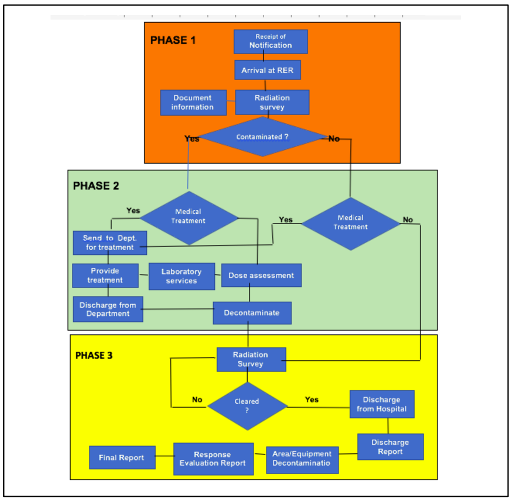

FIG. 1 Chart showing the flow of actions during response to radiation emergency.

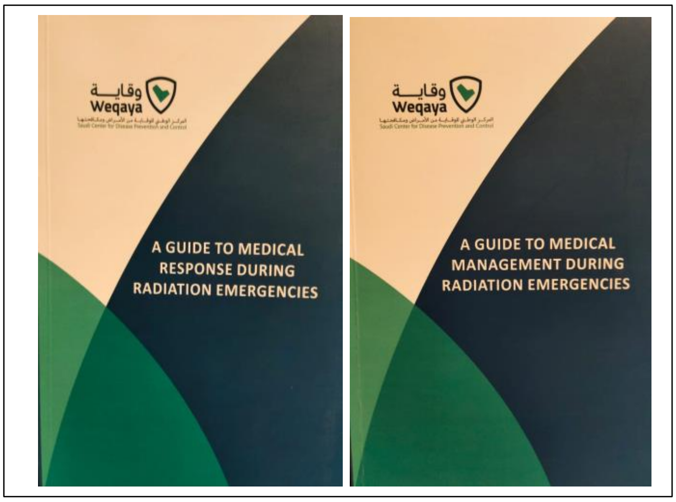

FIG. 2 Published Guides in English version.

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120-LOCAL DIAGNOSTIC REFERENCE LEVELS FOR PAEDIATRIC COMPUTED TOMOGRAPHY IN MOROCCO

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INTRODUCTION

The frequency of use the computed tomography (CT) scans in African countries increased in the last decade. However, a published study had reported that the frequency of CT examinations performed on children in Africa (20%) was higher than in Asia (16%) and Eastern Europe (5%) continent [1].

The International Commission on Radiological Protection (ICRP) implemented the concept 'diagnostic reference levels DRLs' in 1996, as a crucial tool for optimizing the radiation dose received by the patients during the imaging examinations [2]. Thus, the DRLs are classified as local DRLs which are used to identify the dose levels at one facility or group of hospitals, while the national DRLs must be based on national surveys.

Currently, there is not national DRLs established in Morocco for any imaging procedure. Therefore, the purpose of the study is to establish the local diagnostic reference levels for paediatric head computed tomography.

METHODS AND MATERIALS

The data collection was conducted at three Moroccan university children's hospitals from 1 January to 30 April 2019.The dose information (Age(years), Volume Computed Tomography Dose Index CTDIvol(mGy), Dose Length Product DLP(mGy.cm) and Phantom-type (Head-16cm, Body-32cm) were provided by CT systems and collected retrospectively.

The age was used as a parameter of grouping patient for head CT, according to the recommendations of the European guidelines on diagnostic reference levels for paediatric imaging [3]. Therefore, the sample was classified per age groups as follows: <1 year, >1-5 years, >5-10 years and >10-15 years.

The LDRLs proposed in the present work are treated statistically and defined such as mean, median, minimum-maximum, standard deviation(SD) and the third quartile of the distribution for both CTDIvol(mGy) and DLP(mGy.cm) values per age group. Whereas, the LDRLs were setted only for head CT which is the most common CT examination undergoing on children at the participated hospitals.

RESULTS AND DISCUSSION

The number of paediatric patients collected is 792, at three Moroccan university children’s hospitals over 4 months period. All CT examinations presented in the study were based on single-phase scans (with or without venous contrast) and performed within the helical mode scanning.

The proposed local DRLs for paediatric head CT in Morocco are summarized in table 1 and compared with those of other countries. The 75th percentile values were determined for the total distributions of both CTDIvol(mGy) and DLP(mGy.cm) for each age group. Furthermore, the comparison was based on the 75th percentile parameter and took into consideration the same age grouping patients (as mentioned above) as an appropriate method to assure the accuracy of the comparison. Thus, two published studies were selected, which reported the international DRLs for paediatric CT and Thailand’s survey [4-5].

Regarding the comparison showed in table 1, the derived local DRLs in terms of CTDIvol(mGy) were higher than the international DRLs established by the IAEA and those of Thailand only for the <1 year age

TABLE 1. THE PROPOSED LOCAL DRLS FOR PAEDIATRIC HEAD CT

Dosimetric Quantities		**CTDIvol (mGy)**						**DLP (mGy.cm)**					
Age group (years)	**N.of patients**	**Mean($\pm$SD)**	**Median**	**Min-Max**	**75th***	**IAEA[4]**	**Th**[5]**	**Mean($\pm$SD)**	**Median**	**Min-Max**	**75th**	**IAEA[4]**	**Th[5]**
< 1	198	23.83(5.93)	24.19	9.32-57.74	27.14	26	25	401.39(128.74)	388.66	136.65-931.99	469.54	440	400
$> 1 - 5$	231	25.75(5.86)	24.33	9.29-41.88	28.73	36	30	465.89(138.30)	447.08	33.60-1020	530.64	540	570
$> 5 - 10$	206	28.67(7.36)	26.18	6.88-46.41	35.73	43	40	545.56(156.5)	499.36	65.49-1545.20	636.83	690	610
$> 10 - 15$	157	30.76(7.64)	27.24	18.78-43.44	39.80	53	45	593.32(154.5)	534.38	319-1108.1	731.40	840	800

*$75th$: 75th percentile **Th : Thailand [5]

group and lower for the other age groups. It is important to note that the CTDIvol values setted in the international DRLs were based on multiphase scans and obtained by the averaging of values per number of phase scans performed. Moreover, those values were proposed without adjusting phantom size. While in the present study, several CTDIvol values recorded were based on 32cm PMMA phantom which is used to represent the adult body. Therefore, an adjustment of CTDIvol values (multiplying by a factor of 2) was necessary to be representative of the paediatric head CT (presented usually by the 16cm head phantom).

On the other hand, the comparison in terms of DLP values reported that the local DRLs for <1 year age group, were higher than those of selected studies and lower for the other age groups except for >5-10 years age group, their DLP value was higher than those of Thailand. The DLP values defined as LDRLs in the study were based on one phase scan, but those derived in the IAEA survey were determined by using the total DLP which is displayed for all multiphase scans (whole exam). Another parameter can affect the DLP value is the scan length, it is observed that the scan lengths were setted without a restriction to the region required. Further, the children were scanned with the adult protocols at one of the children’s hospitals participated.

CONCLUSION

Even though the results of the present investigation, the concepts of radiation safety for patients are not considered in the Moroccan’s hospitals practices because of the following limitations:

- The frequently request of the examinations which are based on ionizing radiation without justification.
- The ignorance of patients about the risks of ionizing radiation.
- Using the adult protocols to scan the children.
- The national DRLs are not established in Morocco.
- The lack of continuous education in radiation protection for radiologists, radiographers and prescribers.
- None recruitment of medical physicists in the diagnostic radiology department.
- The absence of image quality and dose controls in the diagnostic imaging.

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121-DOSE REFERENCE LEVELS INDIGITAL MAMMOGRAPHY: A CLINICAL RETROSPECTIVE STUDY AND SPECTRA OPTIMIZATION

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INTRODUCTION

The optimization of dose in mammography is an ongoing subject related to lowering the patient dose while increasing imaging quality [1]. The dose is quantified by the Mean Glandular Dose (MGD), which is indirect calculated by the product between measured air kerma and conversion factors [2]. Since it is not possible to determine exactly the breast dose from a mammography exam, standard breast models are used to estimate the dose levels: usually a homogeneous breast composed of 50%/50% glandular and adipose mixture surrounded by a 4 mm skin layer or 5 mm adipose layer [2]. However, newer studies determined that the glandular content of the breast is lower than 50% [3], and the skin thickness is in the order of 1.45 mm thick [2]. The presentation compares the MGD levels between different standard breast models from a retrospectively set of data obtained from DICOM header images of real mammography exams to stablish dose reference levels (DRL). Moreover, the clinical anode/filter combinations and tube potential values were compared from the optimized ones by Monte Carlo simulations and experimental studies to quantify dose reduction.

METHODS

The optimization of the x-ray spectra was performed using the Figure of Merit (FOM) as the main metric, defined by: $FOM = CNR^2/MGD$ [1]. Where CNR is the contrast-to-noise ratio and MGD is the Mean Glandular Dose. The optimization study is divided in two parts: Monte Carlo simulations and experimental results. For the MC simulations, the PENELOPE (2014) [4] + penEasy (2015) [5] Monte Carlo code was employed. Breast phantoms were generated with compressed thickness from 2, 5 and 8 cm, modelled as a semi-cylinder of 10 cm radius. The phantoms were composed of a homogeneous mixture of glandular/adipose tissues with different proportions: 1/99, 20/80, 50/50 and 70/30, surrounded by a 1.45 mm skin thickness. Two type of lesions were included tumour and calcification [1]. The experimental evaluation was performed in mammography unit Selenia Dimensions (Hologic, Bedford, USA), using breast phantoms from Model 012A (CIRS, Norfolk, USA). Afterwards, the optimal spectra were compared from those employed in clinical examination retrospectively. The exposure parameters and breast thickness were extracted from DICOM header of 4449 images. The MGD was estimated using Artificial Neural Networks [6], with the information extracted from DICOM images. Three breast models were considered: (a) 1.45 mm skin, 20:80 composition; (b) 5 mm adipose tissue, 50:50 composition; (c) 4 mm skin 50:50 composition. The possible dose reduction was calculated using $r = (1 - FOM_{ref}/FOM_{opt})$, where ref/opt are the reference/optimized spectra. Therefore, the dose levels were compared retrospectively from the reference DICOM data with the new calculated doses with optimized spectra.

RESULTS AND DISCUSSION

Figure 1 shows the retrospective MGD for different breast models categorized by breast thickness. The 1.45 mm skin 20:80 models present an average systematically increase of MGD (third quartile) of (a) 6%/27%, (b) 13%/32%, (c) 17%/34%, (d) 18%33% compared to models 5 mm adipose 50:50 and 4 mm skin 50:50, respectively. Those values show the importance to considerate which breast model is used to estimate the DRL. Moreover, it becomes evident how the populational characteristics could affect the DRL: in this case the average (standard deviation) of the compered breast thickness was 5.7(1.5) cm. However, the DRL could be altered significantly for different breast characteristics.

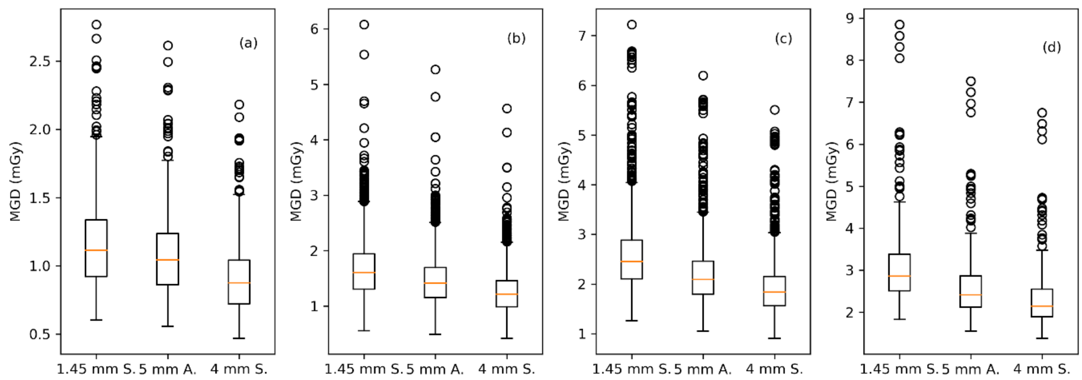

FIG. 11. MGD distribution for different breast models and compressed thickness(t, cm) range: (a) 2≤t<4, (b) 4≤t<6, (c) 6≤t<8, (d) 8≤t<11. Centerline: median, box: first and third quartiles, whiskers: 1%-99% interval. Circles: outliers.

The optimization results indicate that, for mammographic system evaluated, the W/Ag anode/filter combination for breasts thicker than 6 cm could be used instead of the W/Rh, with a potential of dose reduction up to 6%. In other cases, the automatic exposure control selected the same parameters found as optimal in this study. MC results showed that for thick and dense breasts (above 8 cm), the W/Al combination could be an option, with a similar performance compared to W/Ag.

CONCLUSION

According to the results, the standard breast model has a significant impact on the estimation of the MGD values for a mammographic examination, and consequently to the dose reference levels. These results point that the DRL can vary on the mammographic equipment used, since the breast model for dose estimation in DICOM reader is different for each vendor. It is important to clarify that this is a trend estimation and could not be used to determine the patient-specific dose. Finally, from optimization studies, it was observed that the doses could be reduced by employing the optimal anode/filter combinations.

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122-PAEDIATRIC RADIOLOGY: OPTIMIZED X-RAY SPECTRA VERSUS CLINICAL DATA AND THE IMPACT ON DOSE REDUCTION

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INTRODUCTION

Paediatric radiology is a peculiar specialty and with different characteristics from adult patients' radiology [1] and excellent image quality is necessary for lesion detectability. However, the use of ionizing radiation carries risks, with the largest one being the occurrence of carcinogenesis [1]. In addition, children have a higher radiosensitivity than adults and a longer life expectancy [1]. Therefore, the exposure parameters optimization is necessary to have good image quality and adequation of the dose deposited in the patient. The presentation compares the optimal x-ray spectra obtained using Figure of Merit (FOM) with the spectra that are applied in clinical settings for paediatric patients and verify its influence on dose reference levels reduction. The optimal spectra for different patient's ages were obtained via Figure of Merit (FOM), computed using results of image quality and dose obtained through Monte Carlo (MC) simulations [2]. The exposure parameters used in clinical settings were obtained from DICOM reader in a retrospective study in a reference hospital.

MATERIALS AND METHODS

The exposure parameters used in clinical settings were obtained in the Institute of Hospital Radiology (INRAD) of the Faculty of Medicine, University of São Paulo (USP). Two x-ray imaging rooms were evaluated, both have a Philips Digital Diagnost (Philips, Amsterdam, Netherlands) x-ray tube with digital radiography (DR) detection technology, in two different rooms. Both systems are used with automatic control exposure, in room A the examinations are performed without additional filters and in room B additional filters of 0.1 mm Cu plus 1 mm Al are used. A retrospective study was implemented to analyse the anonymized data of chest X-ray examinations in the postero-anterior projection (PA) for 170 patients, with ages between 0 and 5 years, collected between 2014 and 2017, with due approval from the local ethics committee (CAAE:55420616.3.0000.0068). The relevant information for this study (filtration and tube potential) were extracted from the DICOM header. The optimum spectra were determined via Monte Carlo (MC) simulation using the code PENELOPE [3] (v.2014) + penEasy (v.2015) [4] with homogenous acrylic (PMMA) phantoms with thickness varying and representing two ages: 0 and 5 years [2]. The optimal exposure parameters were achieved by the maximization of the Figure of Merit (FOM), which balances image quality and dose independently of exposure. The simulation results show that the use of additional filtration increases the highest FOM values, among the filters, 1 mm Cu achieves the highest FOM values; therefore, it was the most indicated [2]. However, an increase in filtration can be related to an exposure time increase, which could both increase the motion blurring and tube heating. Considering both FOM maximization and equipment limitations an additional filtration of 0.2 mm Cu + 2 mm Al was indicated.

RESULTS AND DISCUSSIONS

Fig. 1 (a) shows a boxplot of the tube potential distribution in the INRAD for different ages groups compared optimal potential range of the tube determined based on FOM results obtained using MC simulation. There are no clinical data for ages 0-1 year in room B. The red horizontal lines represent the medians of the distributions and the vertical lines in blue and green represent the MC simulation results [2]. Fig.1 (b) shows the kerma-area product (KAP) distribution for each room.

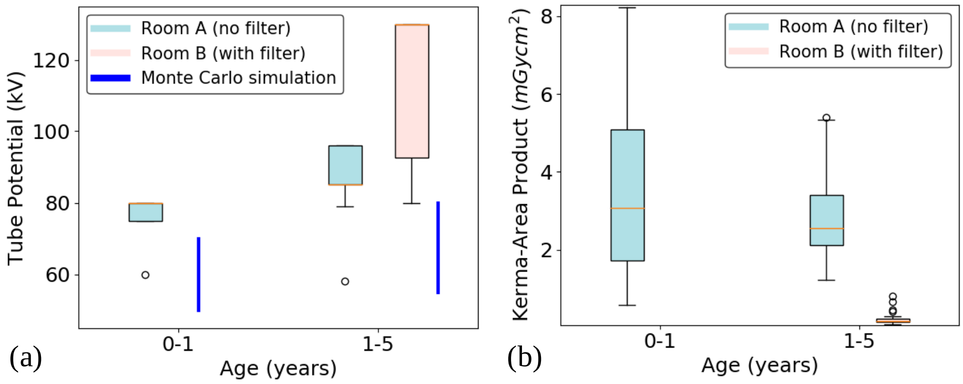

FIG. 12. (a) Boxplot of the tube potential distribution used in INRAD for different age groups and the optimal intervals of tube potential obtained in the Monte Carlo simulation [2]. (b) (a) Boxplot of the kerma-area product distribution in INRAD

Fig.1 (a) shows that the tube potential increases with age. In the INRAD, the tube potential values increase with the inclusion of additional filtration. However, the optimization results show that the additional filtration inclusion is responsible for a decrease in the optimal tube potential value [2]. Fig.1 (b) shows a decrease in KAP values as the filters are added, the filter addition increases the spectra energy, yielding in a lower KAP value. Considering the boxplot's third quarter in Fig.1 (b) as the KAP reference level, the tube potential that yields in this value was considered the reference for the room also. Therefore, using the FOM to compare the reference spectrum with the optimum, is possible to estimate the dose reduction maintaining the image quality by using the optimum spectra. Table 1 shows the dose reduction for each room and age.

TABLE 1. DOSE REDUCTION ESTIMATIVE BY USING THE OPTIMAL SPECTRA

Ages (years)	ROOM A	ROOM B
0-1	42.2 ± 0.9 %	-
1-5	38.2 ± 0.8 %	17.1 ± 0.6 %

CONCLUSIONS

Through the presentation, it was possible to compare the optimal spectra obtained by Monte Carlo simulation [2] with the spectra in clinical environments and verify its influence on dose reference levels reduction. For the case using additional filtration, there is no agreement between the clinical data and the results of the optimization study. Then, there is a possibility of dose reduction, maintaining the image quality, by using the exposure parameters obtained in the optimization study, and, consequently, lower health risk of paediatric patients.

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124-SLOVENIA: RADON IN HOMES AND WORKPLACES NATIONAL RADON PROGRAMME IMPLEMENTATION

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Slovenia lies at the intersection of the Alpine, Mediterranean, Pannonian and Dinaric terrain and covers an area of 20,273 square kilometers with a population of 2,080 million [1]. The average annual effective dose from natural sources received by an average member of the public in Slovenia is approximately 2.5 to 2.8 mSv. It is estimated that the largest contribution, approximately 50%, is from inhaling indoor radon and its progeny (1.2–1.5 mSv per year) in residential buildings. Radon, classified as carcinogenic to humans by IARC, induces approximately 10 % of all lung cancers [2]. This is the reason IAEA in GSR Part 3 [3] and European Commission in EU Directive EURATOM 2013/59 define stricter rules for radon programmes, which are aimed to reduce this share [4].

In 2018 Slovenia transposed requirements from EU Directive EURATOM 2013/59 to the country's national legal system. It is considered that with adoption of the EURATOM directive 2013/59 requirements, IAEA GSR Part 3 requirements are also met. In line with EU Directive EURATOM 2013/59 a Decree on National Radon Program [5] was adopted in 2018. Together with the Ionizing Radiation Protection and Nuclear Safety Act [6], the Decree sets the legislative framework for systematic measurement of radon. Reference level for annual average radon concentration indoor of 400 Bq/m³ has been reduced to 300 Bq/m³ [5] and financial resources dedicated to Radon concentration measurements were increased. The program for taking measurements in schools and kindergartens was expanded. In 2018 the Register of Radon Measurements was developed. Companies performing radon measurements must report all measured results to the Register, which will help to assess radon exposure in Slovenia in the future.

In 2018, 480 measurements in dwellings in areas where higher radon concentration is expected were financed for the first time [2]. Radon measurements are taken in 1 to 3-month period during the heating (winter) season so the worst-case scenario is evaluated. If the first measurement exceeds reference level of 300 Bq/m³ then another measurement is done during summer to estimate the long-term (annual) average radon concentration [5]. All radon measurements required by legislation must be done by an approved organisation. The approval to perform radon measurements is issued to a competent organization by the SRPA. At the moment two companies have been issued the approval and are performing the official measurements in Slovenia. Based on the existing measurement data on radon concentration in the ground, geological structure and indoor radon measurements a radon map of Slovenia was created and areas with higher radon concentration were defined. Out of 212 municipalities, 24 were declared as areas with increased radon concentration. Additional 27 municipalities were defined where systematic measurement program should be carried out [5]. Data from previous years indicate that about 22 % of Slovenia is classified as an area with higher radon concentration. Approximately 30% of measurements in areas with higher radon concentration exceed 300 Bq/m³. Regarding occupational exposure to radon, tourist guides working in Postojna cave are the most occupationally exposed group of workers in Slovenia. With average dose of 7,1 mSv per year (dose coefficient 10 mSv/WLM), overall approximately 10% of workers in Karst caves receive an annual dose higher than 10 mSv. Individual doses due to radon exposure (tourist guides in Karst caves) are included in the Central record of personal doses since 2000.

To prevent high radon exposures in new buildings the SRPA in 2017 financed preparation of "Guidelines for building a Radon safe new Buildings" that were prepared by Slovenian National Building and Civil Engineering Institute. Ministry of Education, Science and Sport in 2017 financed "Guidelines for remediation of buildings in case of increased indoor radon concentration in public educational institutions". However, revised building legislation that would address new requirements for radon protection in existing and newly build objects has not been prepared yet. By the year 2021 an employer shall ensure workplace radon measurements on ground levels and in basements in areas with higher radon concentration and at locations where increased radon concentration can be expected, such as in spas, pools, caves, mines, etc. If annual average radon concentration exceeds 300 Bq/m³, dose assessment shall be made. If effective dose is lower than 6 mSv per year, advisement on how to reduce concentrations of indoor radon shall be provided, conditions affecting the dose shall be monitored and when changes happen dose shall be reassessed. Effective dose higher than 6 mSv shall result in measures to reduce the exposure, e.g. ventilation, relocation of people, reorganization of tasks and working hours and/or building reconstruction. After the application of appropriate measures verification measurements and dose reassessment shall be made. If the annual effective dose still exceeds 6 mSv radiation protection measures which apply to the exposed workers carrying out radiation practice shall be applied. If children, patients or other sensitive group of people receive annual effective dose higher than 6 mSv in public buildings used for the childcare, education, cultural or health care programmes, the resources for implementing measures for reducing exposure shall be provided by the State [5].

To build public awareness and encourage action on radon the SRPA actively cooperates with schools, local communities and media to provide information on radon associated health risks and recommend appropriate action on reducing indoor radon. The SRPA also actively participates in campaigns of the Association of Slovenian Cancer Societies. Due to the high interest, the SRPA bought additional measurement devices for unofficial preliminary radon measurements. They are available for use by interested individuals from the wider public who can borrow the devices free of charge to measure radon concentrations in their dwellings. At the same time concerned members of the public have an opportunity to talk directly with an expert.

With transposition of requirements from EURATOM directive 2013/59 and IAEA GSR Part 3 Slovenia has built a protection strategy for the efficient management of existing exposure to radon in dwellings and workplaces. National Radon Programme consists of actions aimed to reduce exposure in existing and future buildings thus leading to improved protection of public and workers from harmful radiation due to indoor radon. It is expected that remediation and protective actions together with consistent radon communication will yield sufficient benefits to outweigh the cost associated with taking them.

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125-DNA DOUBLE-STRAND BREAKS AND INDIVIDUAL RESPONSE IN RADIODIAGNOSIS

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The evaluation of the potential risks of radiation low doses is an important societal issue. Low doses are generally delivered in very different conditions with various physical characteristics, making this evaluation more complex. Dose values, dose rate, number of irradiations and time between exposures may significantly impact the biological and clinical response. To date, radiodiagnosis procedures, notably CT scan exams, represent the largest cause of medical exposure to ionizing radiation (IR) [1]. In parallel, epidemiological studies are showing an increase incidence of cancers after CT scan exposure, especially for children [2], [3]. However, this studies are limited due to retrospective assessment of radiation dose from CT scans, indication bias, and lack of statistical power [4], [5]. Futhermore, although it is still not included in the International Radiation Protection Recommendations yet, individual radiosensitivity appears to be an unavoidable factor to consider in any radiation-induced risk evaluation [6]. This is why the contribution of radiobiology is essential to complete our knowledge on low doses in clinical situations by considering the individual factor [7].

Since 2016, a unified model of the individual response to IR relevant to both high and low doses and based on the radiation-induced nucleoshuttling of the ATM protein kinase (RIANS) has been proposed [8]. In the frame of the RIANS model, IR trigger the monomerization of the ATM dimeric forms in cytoplasm. The resulting ATM monomers diffuse in nucleus and phosphorylate the H2AX histone variant molecular (γH2AX) at DSB sites, which triggers the formation of nuclear γH2AX foci, quantifiable by immunofluorescence. The ATM-dependent formation of γH2AX foci is one of the earliest recognition step of DSB managed by the non-homologous end-joining (NHEJ), the major DSB repair pathway in humans[9]. During the DSB joining process, two ATM monomers reassociate on the DSB site and form the autophophorylated ATM (pATM) foci in nuclear, also visible by immunofluorescence. The RIANS model permits to predict radiosensitivity and radiosusceptibility with one of the highest statistical robustness [8].

In order to better quantify the contribution of the individual sensitivity in the response to radiodiagnosis exposures, we exposed 20 human primary fibroblasts, astrocytes, and mammary epithelial cell lines from different patients in various clinical exposure conditions of head and chest CT scan on appropriated phantom. Absorbed dose was measured with a dosimeter based on scintillating fiber developed by the Fibermetrix company [10]. The purpose was to study the functionality of the DNA double-strand breaks (DSB) repair and signaling using both γH2AX and pATM biomarkers in exact clinical conditions.

When comparing the results of the patients and the different tissues with each other's there was statistically significant differences in their capacity to recognize and repair radio-induced DNA DSB. Physical parameters as well as irradiation protocol play a major role in the biological response. It is noteworthy that if, for a given physical dose, the number of recognized and/or residual DSB significantly differ from a patient to another, it should be sufficient to integrate these findings in the discussion about the justification of the CT scan exam. Our results show that reliable biological tools can be developed at large scales to take individual radiosensitivity into account to quantify the risk for each patient exposed to CT scan exams and to adapt medical practices. This study is innovative in terms of the cell types and exposure models used, the biological markers studied, the monitoring of the absorbed dose achieved and the consideration of doses repetitions effects. Our data provides new clues that, even for low-dose exposure applied in CT scan exams, the cells from some patients may show very different biological response to IR.

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126-SHIELDING EVALUATION OF CONSTOR® RBMK-1500/M2 STORAGE CASK

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Ignalina Nuclear Power Plant (INPP) is the only nuclear power plant in Lithuania. Having two Units with RBMK-1500 reactors, INPP operated from 1983 y to 2009 y and during this period generated ~ 22 000 pieces of spent nuclear fuel (SNF) assemblies. Currently, there are two storage facilities (old and new spent fuel storage facilities – SFSFs) at INPP for dry interim storage of all generated SNF. The old SFSF was put in operation in 1999 y and was fully loaded in 2010 y when last of the 98 CONSTOR® RBMK-1500 and 20 CASTOR® RBMK-1500 casks (containing 6016 SNF assemblies in total) was placed in it. For the storage of the remaining ~16 000 SNF assemblies, construction of the new SFSF was initiated and the first cask was placed to the dedicated storage position in 2016 y under the "hot trials" programme. The new SFSF is on-site, dry-type SNF storage facility having concrete foundation slab, walls and roof, with the capacity to store up to 202 CONSTOR® RBMK-1500/M2 type casks, see *FIG. 13*. These CONSTOR® RBMK-1500/M2 casks are of a different design compared to the older CONSTOR® RBMK-1500 casks and have an increased space for housing up to 91 SNF assemblies (instead of 51 SNF assemblies in the older CONSTOR® RBMK-1500 cask). Furthermore, the CONSTOR® RBMK-1500/M2 cask is designed to be loaded with SNF of any initial enrichment that was used at INPP, while the older one was licenced only for the storage of SNF having 2.0% U-235 initial enrichment.

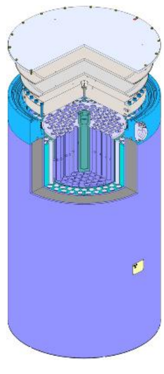

FIG. 13. General view of CONSTOR® RBMK-1500/M2 cask.

After "hot trials", CONSTOR® RBMK-1500/M2 casks loaded with SNF were routinely transferred from Reactor Units to the dedicated storage places in the new SFSF. Having the actual data of the specific SNF assemblies that were placed in the particular CONSTOR® RBMK-1500/M2 cask, the dose rate modelling for that particular cask was performed. Therefore, this study presents the analysis of the dose rates around the one of the casks that was transferred to the new SFSF at INPP. The RBMK-1500 SNF characteristics were modelled using computer code TRITON from SCALE 6.1.3 computer codes system and then, employing the modelled characteristics of SNF, neutron and gamma transport in the particular CONSTOR® RBMK-1500/M2 cask was modelled and shielding capabilities of the cask were evaluated by the help of MCNP 5 ver. 1.6 code.

127-ENHANCING RADIATION PROTECTION IN NORTH AFRICA: A COMMON STRATEGY FOR AFROSAFE RAD AND ARABSAFE CAMPAIGNS

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INTRODUCTION

Africa and the Arab world are linked by a pattern of political, economic and cultural connections particularly with several Arabic countries from north Africa. (Algeria, Libya, Mauritania, Morocco, Tunisia…)

The Afrosafe Rad and Arabsafe campaigns activities focus on enhancing the compliance with the safety standards, policies, procedures and activities that supports the implementation of justification and optimization principles in medical imaging. The program of the campaigns is closely linked to the actions of the “Bonn Call for Action” adopted by the IAEA and WHO with the mission of ensuring a favourable benefit-risk ratio in the use of ionizing radiation in medicine through encouraging adherence to international safety standards and guidelines (1,2,8,9,10).

DESCRIPTION

The triple burden of disease are rampant , whereby communicable diseases are still prevalent, non-communicable diseases are increasing, with high mortality from maternal, perinatal and nutritional causes in several countries from Africa and the Arabic countries. This rising burden, including the risks in radiation has not received the full attention it deserves due to many issues, including low awareness of the burden and other conflicting health demands and limited resources.

The Covid19 is spreading to Africa, because many of the health care systems on the continent are inadequate, having problems such as lack of equipment, lack of funding, insufficient training of healthcare workers, and inefficient data transmission .

Although the initial wave of the Covid 19 pandemic has abated in many countries, healthcare providers are still looking to identify as many Covid 19 patients as possible and contain the disease. Fast and accurate diagnosis is especially important when unsuspecting patients with a coronavirus infection come to the hospital with health complaints but don’t yet show symptoms of COVID-19.

The RT-PCR is considered the first line test and currently recommended for the diagnosis of COVID-19, however the and supply shortages the wait time for results, and the false negative rate as screening tools are still an handicap in the daily practice..

X-ray imaging is a relevant diagnostic modality for patients with COVID-19 disease, especially if the availability of other diagnostic methods becomes limited due to a rapid increase of infected patients. In many locations isolated areas are set up for the diagnosis and therapy of COVID-19 disease to manage the increased number of patients.

The use of chest imaging in acute care of adult patients with suspected, probable or confirmed COVID-19 with the use of imaging modalities (radiography, computed tomography and ultrasound) is well established to describe the circumstances under which each recommendation would benefit patients (10).

Concrete actions in line with the workplan were the focus on radiation protection and the role of imaging in the global public health agenda with planned involvement of WHO and IAEA accounting the variations in the benefits and harms of chest imaging in different situations particularly for the use of the chest CT exams. CT radiation doses questions draw international attention and need to be controlled.
To deal with the growing demand for the safety policy, radiation protection measures needs to be better integrated with other healthcare services, helping to improve ordering behaviors, manage demand on services and ensure the results lead to better patient care.

The actions and initiatives of the Afrosafe Rad and Arabsafe camapigns carried out to date have yielded palpable results concerning justification, optimization, medical exposures and especially the training initiated during the various scientific congresses at the national, regional or continental level.

The results of the Afrosafe Rad and Arabsafe campaigns with these actions developed on the basis of initiatives and investment were led by the radiological community and the international organizations.

If conclusive results in the implementation of Afrosafe Rad and Arabsafe campaigns are recorded, consideration should be taken to quantitative and qualitative issues registered, focusing on characteristics related to the local ,regional ,continental efficiency linked to the international cooperation and partnerships.

This, requires an identification of critical barriers for quality improvement and capacity expansion related to differences between the north and the south of Africa , but with a weighting of the data collected, and the basic indicators (resources at the local, national and regional scale).

The Afrosafe RAD and Arabsafe campaigns will be achieved with a logical framework approach and its implementation will be carried out at a national and regional level record keeping, gathering of specifics of individual approach related to mission, geographic representation, leadership, strategic plan, business model, relationship to health authorities, and communication strategies.

CONCLUSION

A strategy for the implementation of Afrosafe Rad and Arabsafe campaigns should be graded as part of a needs-based improvement program. It should be tailored and selected to sub- regions and countries based on the resources and needs but also the culture and the languages; changing the way of the professionals operate is vital to meet current safety health challenges, considering that people, and therefore healthcare organizations, are inherently resistant to change.

Political will, scaling up and empowering human resource ,professional collaborations are essential. The success depends on taking into consideration the peculiarities and specificities of each country as the culture and the language in order to ensure a good dissemination with a balanced distribution of burdens between countries at the regional and sub-regional levels.

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128-INDONESIA DIAGNOSTIC REFERENCE LEVEL FOR CT-SCAN: 2019 REVIEW

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ABSTRACT – Diagnostic Reference Level (DRL) is an effective tool that can be used to improving the optimization of radiation protection in medical exposure. The purpose of this research is to analyze the Indonesian Diagnostic Reference Level (IDRL) of adult patients for CT-Scan procedures in Indonesia. To analyze IDRL for CT-Scan procedures, quantity Volume CT Dose Index (CTDIvol) and Dose Length Product (DLP) are used. In addition, the analysis of standard patient weight in Indonesia is done by average analysis. The set of IDRL is based on an analysis of the 75th percentile (Q3) of hospitals DRL distribution. The hospital DRL analysis is done by the analysis of median (Q2) of patient dose data distribution in the hospital. For standard patient weight analysis, it was obtained that the average weight of radiology patients in Indonesia is 60 kg with a standard deviation of 10 kg. Analysis of IDRL was done for Head, Abdopelvis and Chest procedures. For CTDIvol quantity, it was obtained that the IDRL for Head, Abdopelvis and Chest procedures were 59 mGy, 14 mGy, and 14 mGy consecutively. For DLP quantity, the IDRL for Head, Abdopelvis and Chest procedure consecutively were 1300 mGy.cm, 850 mGy.cm and 600 mGy.cm.

INTRODUCTION

Diagnostic Reference Level (DRL) is a tool introduced by the International Commission on Radiological Protection (ICRP) for improving the optimization of radiation protection in medical exposure [1]. A survey to evaluate National DRLs in Indonesia has been done by a web-based application. This research is proposed to analyze the Indonesian Diagnostic Reference Level (IDRL) of the adult patients for CT-Scan procedures. The result of this research can be used by BAPETEN as the Regulatory Body in Indonesia to set National DRL. The value will be discussed by BAPETEN with relevant parties before it is set as National DRL. The selection of CT-Scan procedures is done because the frequency of examinations performed using these modalities is quite high and the doses given to patients are quite high.

METHOD

The survey was conducted using a web-based application called the Radiation Dose of Patient Information Management System (*Sistem Informasi Data Dosis Pasien*, Si-INTAN). The data used in this study were collected in 2019. The hospital reported patient dose data with a minimum data of 20 patients per procedure. IDRL analysis for CT-Scan procedure was performed on the quantities of CTDIvol and DLP. The IDRL analysis was performed based on an analysis of 75 percentiles (Q3) of the distribution of Hospital DRLs that participated in the survey. To analyze IDRL, there was a minimum of 15 hospitals that participated in the survey for a specific procedure. Whereas Hospital DRLs were obtained from median analysis (Q2) of data distribution of patient doses at standard weight for a certain procedure. Determination of standard patient weight in Indonesia was based on an average analysis of the patient's weight collected in the Si-INTAN application for CT-Scan procedure.

RESULTS AND DISCUSSIONS

Based on the survey conducted in 2019, the analysis of the standard weight of patients in Indonesia for CT- Scan procedure is shown in Figure 1. The results of the analysis showed that the average weight of patients in Indonesia is 60 kg with a standard deviation of 10 kg. Based on these results, the analysis of IDRL and Hospital DRL was conducted on data of patients with weight in the range of 50-70 kg except for head examination.

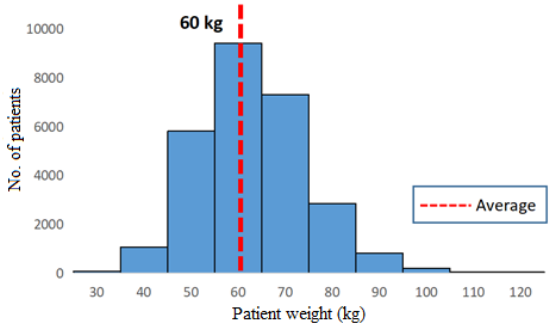

FIG. 1. Profil of patient weight distribution in Indonesia for CT-Scan modality

The IDRLs analysis for CT-Scan modality was performed on Head, Abdomen, Abdopelvis, and Chest procedure. The IDRL on the CT-Scan procedures is shown in Table 1. Based on the comparison of the IDRL with the National DRL from several countries, it was found that the IDRL value for the CTDIvol quantity has approached the comparative country National DRL. But the IDRL for DLP is still quite high so investigation should be done and optimization should be improved. Two factors that affect the value of the DLP is the length of the examination and the number of scanning phases per examination. Therefore, it is necessary to evaluate the procedure especially in selecting the scan length and the number of scanning phases per examination.

TABLE 1. RESULT OF IDRL IN 2019 AND COMPARISON WITH DRL FROM OTHER COUNTRIES

Procedure	No. of Hospitals	CTDIvol (mGy)		DLP (mGy.cm)		National DRL					
		75th percentile (Q3)	**IDRL**	75th percentile (Q3)	**IDRL**	AUS (2012) [2]		GBR (2011) [3]		FRA (2013) [1]	
						CTDIvol	DLP	CTDIvol	DLP	CTDIvol	DLP
Head	122	58.6	**59**	1309	**1300**	60	1000	60	970	65	1050
Abdopelvis	15	13.7	**14**	834	**850**	15	700	15	745	17	800
Chest	15	14.3	**14**	604	**600**	15	450	12	610	15	475

CONCLUSION

National DRLs analysis for CT-Scan procedures in Indonesia has been conducted based on the 2019 survey. The survey process was carried out with the Si-INTAN application. The IDRL analysis was performed on patient dose data in the range of weight 50-70 kg except for head examination. The IDRL analysis for CT-Scan procedures have been done for Head, Abdopelvis and Chest procedures and compared with National DRL of the comparative country. The results of the analysis showed that the IDRL for CTDIvol approaches the National DRL of the comparative country. While the IDRL for the quantity of DLP is still quite high, it is necessary to increase optimization. The IDRL in this research can be established as National DRL in Indonesia.

ACKNOWLEDGMENTS

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129-INVESTIGATION OF RADIOACTIVITY DISTRIBUTION INSIDE AND OUTSIDE HOUSES SUFFERED THE FUKUSIMA DAI-ICHI NUCLEAR POWER PLANT ACCIDENT

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Enormous amounts of radionuclides were released into the air by the Fukushima Dai-ichi Nuclear Power Plant accident, and some of them penetrated to houses. Understanding the radioactivity distribution inside and outside a house is useful for dose assessment from indoor and outdoor radionuclides. The radioactivity distribution was investigated on two houses in Okuma town, which is located in the difficult-to-return area in Fukushima Prefecture. In October, 2019, the house material samples, such as floor (carpet, tatami mat, wooden flooring), indoor and outdoor wall, ceiling, glass wool covering the attic floor, and roof (clay tile, slate), were collected. The radioactivity in the samples were measured using a high-purity germanium detector. Before the measurement, the top surface area of the samples was also measured. The surface density (Bq m^{-2}) was calculated by dividing the radioactivity in the sample by the top surface area. The surface density in thick samples, such as the floor materials, glass wool, roof materials, and soil, could include the radioactivity penetrating into the material as well as one on the surface of the material.

The ratios of the surface density in the floor material, indoor wall, ceiling, glass wool, and air conditioner filter to that in the soil (depth: 0–5 cm) were 0.01–0.09, 1×10^{-4}–5×10^{-4}, 1×10^{-4}–4×10^{-4}, 0.003–0.02, and 0.01–0.02, respectively. The surface density in the floor material tended to decrease in the order of carpet, tatami mat, and wooden flooring. The ratios of the surface density of the roof material and outdoor wall to that in the soil were 0.004–0.2 and 0.002–0.01, respectively. The ratios varied depending on the material and the location of the material. For example, the surface density in the clay tile, one of the Japanese traditional roof materials, was higher than that in the slate. The surface density in the clay tile was higher on the upper side, while that in the slate was higher on the lower side. The surface density in the outdoor wall materials was higher on the lower side.

ACKNOWLEDGEMENTS

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130-JOINT ACTIONS OF THE REPUBLIC OF TAJIKISTAN AND USA IN THE FIELD OF ENSURING ON SAFETY AND SECURITY OF IONIZING RADIATION SOURCES

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Tajikistan is not a nuclear country, but the achievements of nuclear science are widely applied in various areas of the national economy - in medicine, industry, agriculture, scientific research, and educational processes. During the years of Soviet Union in the republic, more than 1000 radioactive sources of various activities were used. A large uranium ore processing plant operated in the north of Tajikistan, which was preserved after the collapse of the Soviet Union due to the lack of raw materials.

Tajikistan in 2001 became a member of the IAEA. In December 2002, the Atomic Energy Agency was established at the Academy of Sciences of the Republic of Tajikistan, which in 2003 was transformed into the Nuclear and Radiation Safety Agency (NRSA). According to the Law of the Republic of Tajikistan "On Radiation Safety" adopted in 2003, the Government gave to the NRSA the authority to create infrastructure to ensure nuclear and radiation safety of the country.

In November 2004, our Government signed the Safeguards Agreement and the Additional Protocol to the Safeguards Agreement, which were ratified by the Parliament of the Republic.

In this regard, the issue of strengthening and improving the regulatory framework and infrastructure in the field of radiation safety, as well as the organization of the state system of accounting, control, storage of radioactive sources and materials in accordance with international requirements, began to take a special place in the activities of NRSA. A particularly important factor is the provision of reliable physical protection of ionizing radiation sources (IRS) in the places of their storage. These issues were central topic in the discussion about security issues of IRS during a visit of the US Ambassador to the Republic of Tajikistan to the Academy of Sciences of the Republic of Tajikistan by and a group of experts from the US Department of Energy on December 14, 2002, as a result of which the Basic Agreement was signed between the Academy of Sciences of the Republic of Tajikistan and the Pacific Northwest National Laboratory.

From that day, cooperation began between the Republic of Tajikistan and the United States to assist our republic on specific tasks in the field of accounting, control of radiation sources, physical protection of radioactive materials, improving the regulatory framework, strengthening the material and technical base and advisory assistance in the field of radiation safety and licensing the safe use of ionizing radiation sources.

In 2006, a similar agreement was signed between the NRSA and the US Nuclear Regulatory Commission.

In the area of these agreements, our republic has received and continues to receive tangible assistance from the US Embassy, Department of Energy and the US Nuclear Regulatory Commission. With the help and assistance of the U.S. Embassy, new building was built for the NRSA Calibration Laboratory, which was further equipped with modern measuring and research equipment, also Customs Committee's, Border Guard's, and NRSA staff were trained in the search and detection of radioactive materials at the PNNL Richland Training Center. Various institutions were provided with detecting and measuring equipment.

With the help of the US Department of Energy, starting in 2003, security alarms were installed at the State Institution "Radioactive Waste Disposal Site" (SI RWDS, Fayzabad), the Oncology Scientific Center of the Ministry of Health, and the Gamma Laboratory of the Tajik National University. Full maintenance of the building No. 20 in SI RWDS was carried out, where unused radioactive sources with a total activity of more than 80 kCi were stored, as well as the Gamma laboratory building of TNU. A fence was built in controlled area around the SI RWDS. All of these facilities have physical protection in accordance with international requirements. These are: installation of video cameras, continuous recording in the register of all events in monitored rooms and facilities, motion sensors to notify security when unplanned crossing the borders of protected facilities, doors with double locks, according to the rules for using "two keys", etc.

I would especially like to note the assistance of the US Department of Energy in the search for orphan sources. According to the search results, more than 700 sources were discovered, which were subsequently transported and stored in the SI RWDS (Fayzabad).

Collaboration with the US Nuclear Regulatory Commission (NRC) is also successful. According to the agreement between the US NRC and NRSA, more than 10 projects were carried out to conduct an inventory of

IRS, introducing amendments and additions to a number of laws and preparing new regulatory documents. In recent years, the NRSA special group, with financial support of US NRC through Qi Tech., LLC, has held consultative seminars for users of ionizing radiation sources to familiarize operators with the basic concepts of authorization, notification and registration, provide practical knowledge for the licensing process and provide information on the latest developments and amendments to the national and international requirements on inspection and enforcement. More than 10 seminars were held throughout the republic, in which more than 300 participants attended.

We especially want to note the implementation of US NRC project in 2007 on creating the National Register of IRS in the Republic of Tajikistan. The creation of a national register meant checking the available information about the presence of IRS in organizations by conducting inventory inspections and entering the verified information into RASOD information system, which was also developed with the support of the US NRC and regularly updated with friendly use browser.

As a result, 250 organizations were inspected, 1976 IRS were found, most of which were stored in warehouses and then transported to RWDS for long-term storage .

In addition, in recent years, some sources have been withdrawn from use and transferred to the RWDS in Fayzabad district. Some of this information is known to the NRSA and is already registered in the Register, but other sources and equipment still need to be tracked with the help of local authorities and other authorized bodies in the field of accounting, control and ensuring radiation safety and security.

Therefore, at the beginning of last year, we discussed with the US NRC the issues of further modernization the register of IRS, updating and preparing some regulatory documents, strengthening the material and technical base of the NRSA, and found good mutual understanding.

US NRC, considering the relatively accelerated development in both industry and medicine, associated with the modernization of equipment and new technologies in Tajikistan, as well as the opening of new public and private enterprises, hospitals, diagnostic centers and medical institutions using radiation sources in their daily work, supported our initiative on updating the State Register of IRS.

Currently, with the financial support of the US NRC, the inventory of IRS in the Sughd region of Tajikistan has been completed. Over 100 new unregistered sources have been registered. In the near future, it is planned to continue the inventory of IRS in the southern part of Tajikistan.

All this allows NRSA to improve the state system of accounting, control, storage and movement of radioactive sources and constantly monitors the "fate" of the sources.

Currently the “Joint Action Plan between the Government of the United States of America and the Government of the Republic of Tajikistan on Combating Smuggling of Nuclear and Radioactive Materials” was signed. With this agreement, the U.S. and Tajik governments are committing to a partnership to combat that threat. Tajikistan will implement independently the steps that it can take on its own, and the United States will help to identify sources of funding within either the U.S. government or the international community to support the steps that require assistance.

The cooperation of the Governments of Tajikistan and the United States together allows us to organize the country’s appropriate conditions for the safe management of ionizing radiation sources, which contributes to their safety and guarantees their safety.

We hope that the recognition by our republic of international treaties and agreements under IAEA framework, as well as fruitful cooperation with the IAEA and the Regulators of other countries, especially the countries of the European Union, USA, Kazakhstan, Russian Federation, Armenia, Georgia and Uzbekistan will allow the international community to see us as reliable partners, and we will be allowed to use atomic energy more effectively for peaceful purposes and to raise the radiation protection system of the environment and population at a higher level

131-NATURAL RADIOACTIVITY LEVEL MEASUREMNET IN SOME ROOT VEGETABLE MOSTLY USED IN MYANMAR

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In the present study, the activity concentration of ^{226}Ra, ^{232}Th and ^{40}K were measured in the root vegetable in Myanmar. Samples of potatoes, onions, radishes, carrots and sweet potatoes were collected from different locations of Myanmar. The activity concentration levels of Ra-226, Th-232, and K-40 were measured by direct gamma ray spectrometry. The average activity concentration of ^{226}Ra, ^{232}Th and ^{40}K in root vegetable samples were found as 14.23 ± 3.77, 13.99 ± 3.74 and 282.92 ± 16.82 Bq kg^{-1}, respectively. The activity results indicated that all root vegetable was similar activity in ^{226}Ra, ^{232}Th and the highest in ^{40}K activity concentration was measured in radish 436.70 Bq kg^{-1}.

The annual effective dose due to intake of these radionuclides through ingestion of root vegetable samples was estimated to assess radiological risks. The average annual effective dose was found as 61.96 μ Sv y^{-1}. It is less than the recommended values of the World Health Organization (WHO).

Keywords: Natural Radioactivity, Root Vegetable, Gamma ray spectrometry

INTRODUCTION

Myanmar is trying to implement the Myanmar Nuclear Law and related regulation of nuclear safety, security and safeguards. The one of the objectives of this law is to lay down and carry out measures for protection of people and the environment against harmful nuclear and radiation effects.

People are all exposed daily to small amounts of radiation from natural sources of radioactive material. Radioactive substances and radiation in the environment may irradiate our body from the outside externally. Or people may inhale the substances in air, swallow them in food and water or absorb them through skin and wounds, and then they irradiate us from inside-internally.

In the present work, radioactivity levels of Ra-226, Th-232, and K-40 in daily diets were determined in selected root vegetable from Myanmar. The obtained results were used for the estimation of annual intake of radionuclides and it is also essential for the development of database of national radionuclide in environment of Myanmar.

MATERIAL AND METHODS

25 root vegetable samples include potatoes, onions, radishes, carrots and sweet potatoes were collected from different location of Myanmar for the measurement of the specific radioactivity of ^{226}Ra, ^{232}Th and ^{40}K. The root vegetable samples were dried at 105º-110ºC and then crushed and sieved through a 200 mesh. The dry weight of samples were between 60 to 200 grams thus prepared were packed in plastic containers and sealed for four weeks to reach ^{222}Rn and its short-lived daughter products into equilibrium with ^{226}Ra.

Radioactivity from the root vegetable samples was detected using with High Purity Germanium (HPGe) and NaI (Tl) detector and counting time is 18,000 sec.

RESULTS AND DISCUSSION

Radioactivity content in root vegetable

The measured activity concentrations of Radium, Thorium and Potassium of root vegetable results are shown in Table 1 and Fig.1. The average activity concentration of ^{226}Ra, ^{232}Th and ^{40}K in root vegetable samples were found as 14.23, 13.99 and 282.92 Bq kg^{-1}, respectively.

TABLE 1. MEAN ACTIVITY CONCENTRATIONS (in Bq/Kg dry weight) IN ROOT VEGETABLE SAMPLES

Root Vegetables	Number of samples	226Ra (Bq/Kg)	232Th (Bq/Kg)	40K (Bq/Kg)
Onion	5	15.45	14.33	200.95
Carrot	5	11.47	12.53	301.82
Radish	5	14.93	15.76	436.70
Sweet Potato	5	13.85	12.88	283.68
Potato	5	15.44	14.47	191.46
Average		14.23	13.99	282.92

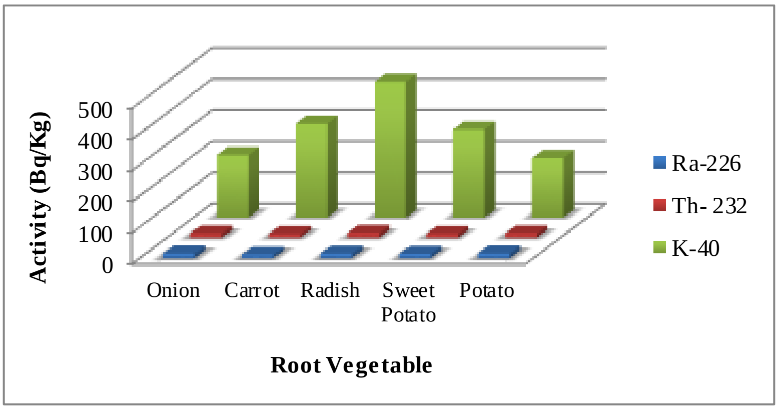

FIG. 14. Mean activity concentrations of 226Ra, 232Th and 40K in root vegetable samples.

Daily intake of natural radionuclides

The daily intake of radionuclide was calculated from the measured concentration of radionuclide in root vegetables, the daily consumption rate and the number of days of the vegetable consumed in a week. The highest value of natural radionuclide ingested is from the consumption of onion because it is daily used in every curries of Myanmar. Table 2 presents the daily intake of natural radionuclide in root vegetable.

TABLE 2. DAILY INTAKE OF NATURAL RADIONUCLIDE IN ROOT VEGETABLE SAMPLES

Vegetable	Bq/day			Total (Bq/day)
	226 Ra	232 Th	40K	
Onion	0.0315	0.0029	4.0994	4.1338
Carrot	0.0033	0.0004	0.8796	0.8833
Radish	0.0044	0.0005	1.2727	1.2775
Sweet Potato	0.0040	0.0004	0.8267	0.8311
Potato	0.0045	0.0004	0.5580	0.5629

Annual effective dose (AED) of root vegetable samples

The Annual effective dose (AED) due to ingestion of the radionuclides in the root vegetable was calculated based on the metabolic model developed by the International Commission of Radiological Protection (ICRP). The effective dose from radionuclides in vegetable is determined by using the measured concentration of radionuclide in root vegetable, annual consumption rate and dose coefficient for intake by ingestion of radionuclides. In this study, total annual effective dose is obtained 61.96 µSv/y due to daily intake of 226Ra, 232Th and 40K via those root vegetable. Table 3 shows the AED from root vegetable.

TABLE 3. ANNUAL EFFECTIVE DOSE (AED) BY INGESTION OF ROOT VEGETABLE SAMPLES

Vegetable	AED (µSv/y)			Total Annual effective dose from each type of vegetable (µSv/y)
	226 Ra	232 Th	40 K	
Onion	3.03	2.31	8.72	14.06
Carrot	1.61	1.44	9.36	12.41
Radish	2.09	1.81	9.64	13.54
Sweet Potato	1.94	1.48	8.80	12.22
Potato	2.16	1.66	5.90	9.73
Total annual effective dose from each radionuclide	10.83	8.71	42.42	61.96

ACKNOWLEDGEMENTS

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132-EVALUATION OF EDUCATION AND TRAINING ACTIVITIES ON RADIATION SAFETY IN TURKEY FOR THE FIVE YEARS PERIOD (2013-2017)

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Education and training (E&T) activities of radiation workers for radiation protection is a basic aspect of the optimization of all exposures to radiation. Turkish Atomic Energy Authority (TAEA) has established radiation training programs to protect radiation workers, patients and the public against the health risks of radiation exposure under planned, existing and emergency exposure situations at the national level. This is a case study of the application of protection and safety training to persons in various categories working with ionizing radiation in Turkey.

INTRODUCTION

Ionizing radiation has many beneficial applications, including uses in medicine, industry, energy production, agriculture and research when it is used safely. But the use of ionizing radiation sources can be harmful effect to the health of human, and environment if it not properly used or not controlled. Occupational exposure due to radiation sources and devices in Turkey arises from mainly three different fields of activity, namely: 1- Medicine 2- Conventional and 3- Research and education. The national legislation requires specific training in radiation Protection. The TAEA organizes and/or coordinates courses to implement training for radiation protection in the country arises from different fields of radiation application [1]. In this study, general overview of Turkey's radioactive sources and equipment, E&T objectives and methods of TAEA, types of national courses, overview of TAEA's syllabus, the course materials and equipment, a central examination system of TAEA and implementation and evaluation of national course activities on occupational radiation protection between 2013 and 2017 have been presented.

METHODS AND MATERIALS

Education, training and continuing professional development is essential to improve safety and quality in the uses of ionizing radiation. The IAEA presents information on the structure and content for training courses in radiation protection for radiation workers [2]. According to the BSS's definition, RPO is a person technically competent in radiation protection issues related to a given type of practice who is designated by the registrant or licensee to oversee the application of relevant requirements. A similar definition is included in the Turkish "National Regulation of Radiation Safety, in Article 4-i, (No: 23999) [3]. TAEA is mainly interested in the area of radiation protection training and routinely provides training courses which are named Radiation Protection Course for Industrial Applications (RPC-IA), Radiation Protection Course for Industrial Radiography (RPC-IR), Radiation Protection Course For Diagnostic Radiology (RPC-DR), Radiation Measurement and Radiation Protection Course (RPC-RM), Course for Safe Transportation for Radioactive Materials (C-STRM) and One Day Training Course for Radiation Protection (RPC-1day) at the national level. The materials and equipment such as survey meters, personnel monitoring equipment, standard small calibration radioactive sources, sets of absorbers, a demo industrial radiography device with source changer and associated equipment, conventional X-ray system, C-Arm fluoroscopy system, personal protective equipment and course documents are commonly used for classroom demonstrations and practical exercises in our training courses in radiological protection. The radiation protection courses are delivered as five-day seminar (included exam) by physicists, radiation biologists, radiation protection experts, medical physicist. A final test is given to participants at during the periods specified in the year by TAEA. All participants should take this approximately one-hour test for measuring their achievement and certification. A Certificate for Achievement on the test is issued according to the test results by TAEA. According to this test score, two different certificates are issued. If the examination score is 70-over 100 points, a “Certificate of Achievement” is awarded to the successful participants at the end of the course. For lower scores, a “Certificate of Attendance” is given to the participants.

RESULTS AND CONCLUSION

Total 190 radiation protection courses were organized and 4393 participants trained between 2013 and 2017. Figure 1 presents the number of course participants for each radiation protection courses for the period. According to the total evaluation of 3756 participants, 3393 of the participants were granted with “Certificate of Achievement” for the year 2013 – 2017. In other words, the rate of ‘Certificate of Achievement’ given within 5 years is % 90.34. In the five-day radiation protection courses (RPC-IA, RPC-IR, RPC-DR, RPC-RM and C-STRM) for the year 2013 – 2017, this ratio is %95.67, %83.08, %96.29, %96,73, %90.98 and %90.34, respectively. Figure 2 presents the percentage of given ‘Certificate of Achievement’ devoted to the five-day radiation protection courses for the year 2013 – 2017. The training provided by the TAEA is primarily aimed at regulators, professionals working in radiation protection and those responsible for the development of training programmes in Turkey.

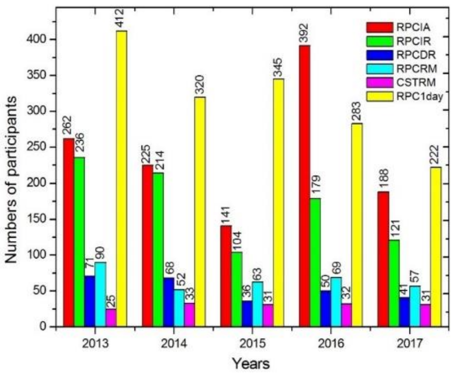

FIG. 15. Number of course participants which is trained by TAEA in the radiation protection courses for the period of 2013– 2017..

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134-INSPECTIONS OF RADIATION SOURCES USED IN AUTOMATION SYSTEMS OF INDUSTRIAL APPLICATIONS AND THE REASONS FOR FALLING INTO ORPHAN SOURCES

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The aim of this study is to provide evaluation and investigation of the nonconformities in the inspections of radiation sources used in automation systems of industrial applications and the reasons for falling into orphan sources.

INTRODUCTION

Measurement systems containing ionizing radiation sources are used in order to control the quality and production method of the products quickly and reliably and to ensure continuity in the production in the industrial field. Such systems are called Nuclear Measurement Systems. In this system, radiation generating devices (such as X-ray devices) and devices containing radioactive sources are used as ionizing radiation sources. Thickness, weight, density, moisture and level measurements of the products are made by using radiation devices and sources. The material does not come into contact with the radiation source or device.

DEFINATIONS AND EVALUTIONS

The nuclear measurement systems are categorized into two types as fixed and portable systems. (1) Fixed measurement systems: It can be called as automation systems. The sources can be fixed or mobile in these systems depending on the application area. (2) Mobile measurement systems: They are systems that can be carried by one person and can be used in different places according to the purpose. The sources of automation systems have long half-life (Co-60, Cs-137, Po-210, Am-241, Sr-90, Cf-252, Kr-85, Tl-201, Pm-147). Their long half-life has an advantage in terms of cost of use, while a disadvantage in terms of long-term safety and security of the source.

Orphan Source: Radiation sources that are not under the control of the regulatory authority, which are abandoned, stolen, forgotten, misplaced, lost, not properly transported without authorization or permission, or never taken under control [1]. *Vulnerable Source*: They are under the control of the regulatory authority whereas they are insufficient sources to ensure their safety and security for a long time. It can easily become an orphaned source [2]. Although under the control of the regulatory authority, *Disused Sources* and *Spent Sources*, which are defined below, are also *Vulnerable Sources*. These sources can easily be lost. *Disused Source*: They are unused radiation sources and not planned for future use [1]. *Spent Source*: They are radiation sources that do not have sufficient activity for the purpose of implementation [2]. The use of the sources is used in automation systems is common in Turkey. In accordance with the legislation of the country, it is necessary to obtain a license from the regulatory authority for the use of these sources. A license is issued if it is found appropriate by conducting radiation measurement and inspection before license. After the license is obtained, it is subject to inspection by the authority for certain periods. During the audits; it has been seen that there are sources that comply with the above definitions of *Spent Source*, *Disused Source* and *Vulnerable Source*. It has been understood that it is not possible to detect these situations in the offices before coming to the inspection. These sources are potential *Orhan Sources*. The reasons for falling into *Orphan Source* status can be examined in two steps:

(1) *Reasons arising from the user organization:* Radiation sources used in automation systems generally do not directly enter the field of activity of the organization, but are classified as equipment or an element used as an auxiliary parameter in its production. the main purpose is to produce the product and to provide appropriate personnel and equipment in the organization. These organizations operating in the production sector do not have sufficient information about radiation. As a result of the audits, the nonconformities determined as the reason for being *orphan source* can be listed as follows:
   (a) Organizations do not want to avoid costs and not want to spend the expired source for the processes of sending to waste and country of origin.
   (b) Not notifying the authority that the Radiation Protection Officer has left and not hiring new personnel.
   (c) No radiation measuring device or failure of the failed device to supply new equipment.
   (d) High probability of the source being scrapped when the business is closed or dismantled.
   (e) Radiation warning labels of sources from dust, dirt, oil, etc. fall for reasons used in automation systems. Forgetting their location and suspending of the sources. If there are no radiation warning labels on the sources, as a result of encountering natural disasters and disasters such as earthquakes, floods, sources can be evaluation as normal equipment, components, parts.
   (f) The training of people working in automation systems is generally far from radiation and radioactivity issues. This situation creates weakness in protection from radiation and in ensuring the safety of radioactive source.
   (g) Not being prepared for emergencies and lack of emergency plans due to insufficient trained and qualified personnel. Lack of internal audits within the organization.
   (h) The system becomes inoperable as a result of not performing the maintenance on the system on time, the system becomes inoperable as a result of failure to fix it, and the source will be scrapped as a result of removing the source from its place and leaving it there.
   (i) Leaving the source where it was left during the processes to be carried out for waste or export, transportation performed by unauthorized and unauthorized persons, leaving the source in the wrong place as a result of forgotten in the vehicle, theft of the vehicle, abandonment of the source.
   (j) As a result of the closure of the organization due to bankruptcy, the work with radioactive sources was stopped and the sources were stored for a long time (especially in textile factories).
   (k) It is not known how many sources are available in the enterprise. The radioactive sources are not kept and kept regularly.
   (l) Failure to transfer the changes to the authority due to the fact that the existing legal regulations are not dominated or avoided.

(2) *Reasons arising from the competent authority:* General reasons arising from authority; as a result of changes in the system and personnel, the decrease in the command of the employees on the subject, the lack of field experience and knowledge of the employees, the insufficient knowledge and training, the lack of command of the legislation, the lack of routine inventory control and desk control, the failure to determine the priorities in radiation practices, ineffective planning and control.

RECOMMENDATION AND CONCLUSION

Especially, the competent authority should work to eliminate the gaps in the control and inspection mechanisms in automation systems of industrial applications and the reasons for falling into *orphan sources*. It is necessary to cooperate with other people and organizations in this regard and to benefit from people who have knowledge and experience on the subject. The authority and user organization should have full knowledge of all stages from the import of the radiation source to waste treatment. In this regard, they should cooperate with each other in all administrative and technical aspects within the framework of the legislation provisions and ethical rules.

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135-STUDY ON APPLYING GRADED APPROACH IN THE COUNTRY LIMITED USED OF RADIATION SOURCES

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Since regulating radiation safety is a national responsibility, every country including the developing countries limited using radiation sources must try to meet their responsibility. One of the most challenges in implementing effective regulatory control for radiation safety in the developing countries is applying graded approach to the regulations. As Graded Approach means applying resources and systems to the requirements based on different levels of possible radiation risks associated facilities and activities and harmful consequences of exposure to ionizing radiation. Such approach helps regulators and operators to determine what their policies and objectives should be; it also helps them understand the most appropriate safety enhancements to apply. And it is important to set the right level of requirements and guidance to enable flexibility without compromising safety. However it is difficult to apply graded approach to the regulations for the small country with the lack of resources. Most of the regulatory bodies in the developing countries have limited financial, human and physical resources, and inadequate legislations. This study aim to suggest application of Graded approach in accordance with the relevant requirements for ensuring Radiation Safety in the country limited used of Radiation Sources.

GRADED APPROACH IN REGULATION

The important feature of the graded approach is the provision for exemption and clearance and for notification, authorization by registration and authorization by licensing. In addition, the graded approach should be applied to the products and activities of the competence management processes. For the developing countries only using Radiation Sources, the requirements of X-ray facilities and facilities using Radioactive Sources should be in accordance with graded approach. This means that the more safety need the more costs should be focused and less money may need to be spent where the risks are lower. However, it should be remembered that even sources categorised as "exempt" can cause disruption, create liabilities and damage reputations. The competency of regulators in dealing with regulatory process should be graded according to their expertise.

Every state has an obligation of diligence and duty of care, and is expected to fulfil their national and international undertakings and obligations. National Policy and Strategy for Radiation Safety should be implemented in accordance with the principle of graded approach according to the national circumstances and radiation risks associated with radiation activities. To achieve the objective, a common action is needed by the operators, regulators and other involved institutions. The graded approach is a method in which analyses applied are commensurate with the level of risk posed by facility. In applying a graded approach, consideration should be given to the utilization and the type of facilities and dangerous level of the radioactive sources. Therefore, to apply graded approach to regulations, the regulatory body shall employ a sufficient number of personnel with the necessary qualifications, experience and expertise to undertake its functions and responsibilities.

EFFECTIVE REGULATORY FRAMEWORK

To achieve effective regulatory framework, the national policy for radiation safety shall be appropriate to the facilities and activities for the regulatory body to implement the regulations. And management system should be established with goals, strategies, plans and objectives that are consistent with the policies of the regulatory. According to the level of risk the regulator should be assigned based on their knowledge. It is also needed to make sure that they are adequately knowledgeable about safety policies, procedures, responsibilities and requirements. However it is difficult to apply requirements and guidance to the right level of the risk if the resources are inadequate, the regulators should try according to the national circumstances to achieve the effective regulatory framework as in Fig.

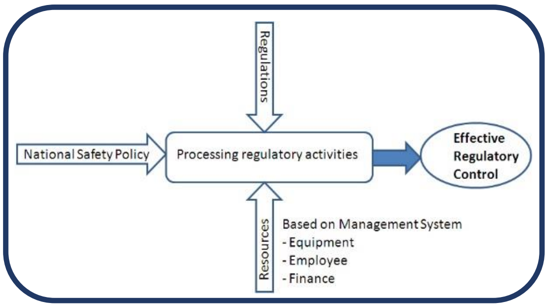

Fig – Applying graded approach to achieve effective regulatory framework

CHALLENGES

As Graded Approach is part of Regulatory Body's management systems, human resource plan is essential to match the right people with the right skills at the right level of risk. In developing country, human resource development in Nuclear engineering are not attracted to young generation. Since there is no Nuclear Power Plant in the country, there is no job opportunity for this field. And due to the lack of legislations, radiation safety officer is not essential to be assigned for the user of radiation sources. Therefore it is suggested that even the country using limited radiation sources, Radiation safety law and regulations should be developed. The regulatory body should foster and support a strong safety culture through the development and reinforcement of good safety attitudes and behaviour in individuals and teams. Safety performance should be improved through the planning, control and supervision of safety related activities in normal or emergency's situations.

CONCLUSION

It is concluded that the graded approach should be applied in establishing the regulatory body and determining aspects of its organizational framework based on the potential hazards of all of the facilities and activities under its supervision or oversight. The regulatory body shall ensure to provide technical basis for emergency planning and a preliminary hazard categorization based on an inventory of Hazardous Radioactive Materials. The Hazard Category can be based on an assessment of the consequences resulting from postulated accident scenarios. It is recommended that the regulatory body is required to be provided with sufficient authority and power, and a sufficient number of experienced staff and financial resources to discharge its assigned responsibilities in applying Graded Approach to regulations.

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137-DOSIMETRY AUDIT IN RADIOTHERAPY KEY ROLE IN SAFETY CULTURE

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During the International Conference on Radiation Protection in Medicine held in Bonn in 2012, several areas for improvement were identified and one of them is ***radiation safety culture***. In general the "***culture is how we do things around***". International Atomic Energy Agency (IAEA) has a special definition of term "***culture***". According to the IAEA definition organizational culture is a mixture of an organization's traditions, values, attitudes and behaviour [1]. Since the nuclear activities and knowledges are unique in many ways in the nuclear applications, culture is associated with term "***safety culture***" [2]. It was adopted by the IAEA in recognition of the fact that nuclear safety is strongly dependent on the actions and thoughts of the people within organization. The nuclear organizations are fully responsible to create a safety culture, working in manner of professional cooperation and communications with staff in high degree of transparency. Maintaining and improving safety culture requires continuous action and commitment. There is a need to establish a program to measure, review and audit all activities, health and safety performance against predetermined standards. The effectiveness of the safety culture can be improved if radiation ***safety audit*** is performed on the system.

Modern radiotherapy is one of most rapidly developing nuclear applications in medicine and today it is a safe and highly effective cancer treatment modality. Radiotherapy is a multi-disciplinary speciality involving complex equipment and procedures. It is recommended as part of treatment for more than 50% of cancer patients [3, 4]. It can be used on its own, or to complement and enhance the effects of other treatments, for example to shrink or control a cancer before and after surgery [5, 6]. Precise radiation dosimetry measurements are used to keep radiotherapy safe and effective. The need of dosimetric and geometric accuracy in radiotherapy is a re well defined [7, 8]. Recommendations of the International Commission of Radiation Units and Measurements (ICRU) given as early as in 1976, state that the dose delivery to the primary target should be within ±5% of the prescribed value (but in some special circumstances it should comply within ±2% to the prescribed dose to the target) [9].

The physical and technical aspects of Quality Assurance (QA) programmes in radiotherapy include regular quality control of equipment, dosimetry of radiotherapy beams, treatment planning procedures and treatment delivery. A fundamental step in any dosimetry QA programme is the ***audit*** performed by an independent external body. The dosimetry audit is organized and conducted in order to increase the confidence in accuracy of clinical dosimetry and to ensure the adequacy of all performed procedures, as well as the quality of the radiotherapy equipment used for the treatment. The ultimate goal of external dosimetry quality radiotherapy audits in radiotherapy offered by the IAEA is to improve the accuracy and consistency of clinical dosimetry, in radiotherapy centres worldwide and to prevent or reduce the likelihood of errors and incidents [10]. Over the years, the audits have contributed to dosimetry practice and accuracy of dose measurements [11, 12].

Dosimetry audit in reference and non-reference conditions on and off axis was successfully introduced in some leading Bulgarian radiotherapy centers in the period 2006 – 2010 using temoluminescent dosimetry system (TLDs). The results are shown on the Fig.1. [10].

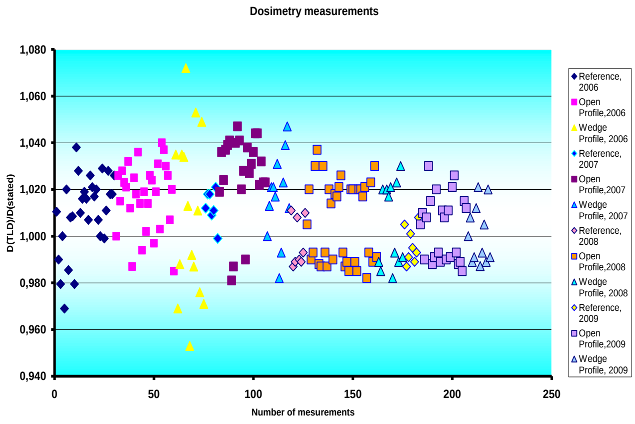

Fig.1. *The results of national TLD audit 2006 – 2009.*
Ratios of TLD measured dose to the participant stated dose

Radiotherapy equipment in the country has been subsequently replaced with new state-of-the-art linear accelerators since 2011. In the period 2017-2019 12 /twelve/ Bulgarian radiotherapy centres participated in the IAEA/WHO Postal Dose Audit Service with radiophotoluminescent glass dosimeters (RPLD). Number of checked beams was 34. The results in terms Diaea/Dstated have shown that 33 beams were within 5% acceptance level.
The results are given on Fig. 2.

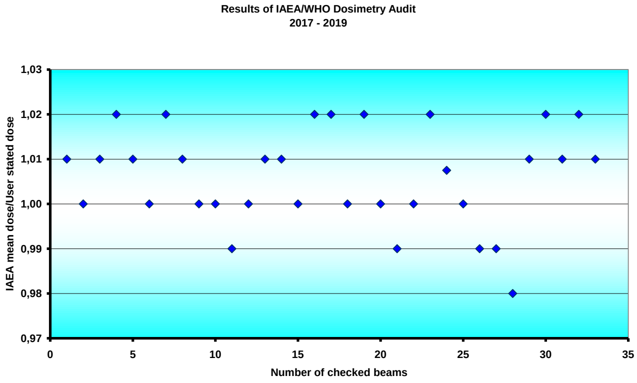

Fig.2. *The results of IAEA/WHO RPLD audit 2017 – 2019.*
Ratios of IAEA mean dose/Stated dose

The ***national dosimetry audit*** with radiophotoluminescent glass dosimeters (RPLD) is under preparation in frame of IAEA Technical Cooperation programme Project, Concept Number: BUL2016005 "Establishing a National Dosimetry Audit System and Dosimetry Quality Audit Programme in Radiation Therapy" and is expected to start in the beginning of next year [13**]**. In 2007 such type of audit was initiated in Japan. RPLD is silver activated phosphate glass with the density of 2.61 g/cm3 and its characteristics such as repeatable readout and negligible fading effect is suitable for the postal dose audit [14].

Dosimetry audit ensure, that the correct therapeutic dose is delivered to the patients undergoing radiotherapy and play a key role in activities to create a ***safety culture***. One important component of safety culture, particularly in the nuclear applications is radiation safety for employees and local communities, while in radiotherapy means safety of the patients and hospital staff. The good ***safety culture*** creates an environment for success and allow us to treat more patients in efficient, effective and safely manner.

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138-ANALYSIS OF THE DOSE GENERATED BY CT SCAN IN A PET/CT EXAMINATION USING TWO DIFFERENT PROTOCOLS

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INTRODUCTION

Following the growing use of hybrid imaging modalities, the demand for PET/CT examinations has increased in recent years, due to their high accuracy, sensitivity and specificity, making possible accurate localization and definition of pathologies. These devices allow the functional images overlapping obtained from the radiopharmaceuticals administration and anatomical images generated by X-ray beam attenuation.

The PET/CT clinical applications have been expanding, mainly in oncologic diagnosis and management. However, hybrid systems, especially those that include diagnostic CT, inevitably result in an increase in patient radiation exposure, compared with a single CT or PET scans [1]. In this study, absorbed doses from CT scan were evaluated in 21 organs using radiochromic film strips, positioned into ATOM anatomical male phantom, using two distinct protocols.

METHODS

The experiments were performed in a PET/CT Discovery 690 (D-690) from the General Electric (GE) manufacturer. To evaluate the absorbed doses from the CT scan, ATOM anatomical male phantom was used. It consists of a whole-body phantom, built with tissue-equivalent epoxy resins using the ICRP 23 and ICRU 48 as reference [2]. Dose measurements from CT scan were done with radiochromic film strips. These strips were cut from a Gafchromic XR-QA2 film sheet [3] in dimensions of 1.5 cm by 0.5 cm. The film strips were placed inside the phantom at points corresponding to the desired organs, all in the same direction, perpendicular to the CT beam. The film strips are 0.5 cm wide and the beam cuts them widthwise and aims to record the punctual dose. The film strips were scanned before irradiation to record the Background (BG) radiation. The phantom position into PET/CT table is shown in Figure 1. A scout frontal and another lateral were performed on the skull-caudal axis, using 120 kV and 10 mA, scanning the whole phantom.

FIG. 16. ATOM phantom into PET/CT table.

Two computed tomographic protocol were performed, one only for anatomical mapping and the second one with diagnostic quality images. The main parameters used in the anatomical mapping acquisition were: 120 kV, auto mA, mA range of 10-120, being the mA.s average value 7.7 and noise index of 25. In the diagnostic protocol were: 120 kV, auto mA, mA range of 50-400, being the mA.s average value 35.7 and noise index of 16.05. One day after the CT scanning, digital images of the film strips were generated to obtain the darkness intensity and after the corresponding absorbed dose values in the organs. The experiment was repeated three times for each protocol. The punctual absorbed dose calculation was performed according to the intensity calibration curve versus absorbed dose for the X-ray beam of 120 kV [4].

RESULTS

The results of absorbed dose for each organ, as the standard deviation, are shown in Table 1.

TABLE 1. ABSORBED DOSES FROM CT SCANS

Organ	D_T (mGy)		Organ	D_T (mGy)	
	Standard	**Diagnostic**		**Standard**	**Diagnostic**
Brain	4.12 ± 0.21	7.81 ± 0.27	Gallbladder	5.11 ± 0.17	12.19 ± 0.25
Crystalline	3.81 ± 0.15	6.18 ± 0.21	Stomach	5.78 ± 0.10	13.63 ± 0.21
Salivary Glands	3.69 ± 0.15	7.39 ± 0.32	Colon	5.81 ± 0.24	12.85 ± 0.21
Thyroid	8.58 ± 0.19	22.60 ± 0.23	Pancreas	4.06 ± 0.14	12.34 ± 0.17
Bone Marrow	4.52 ± 0.21	12.12 ± 0.27	Small intestine	6.15 ± 0.18	14.76 ± 0.30
Esophagus	6.80 ± 0.12	15.61 ± 0.23	Kidney	4.70 ± 0.18	12.51 ± 0.21
Lung	5.28 ± 0.15	12.66 ± 0.24	Sigmoid	6.22 ± 0.27	16.00 ± 0.21
Hearth	5.41 ± 0.14	12.15 ± 0.15	Bladder	5.70 ± 0.15	14.91 ± 0.40
Breast	5.42 ± 0.23	12.23 ± 0.17	Testicles	10.08 ± 0.28	28.85 ± 0.40
Liver	5.12 ± 0.19	12.36 ± 0.20	Skin	5.20 ± 0.23	17.84 ± 0.17
Spleen	5.18 ± 0.16	11.91 ± 0.18	Bone Surface	5.63 ± 0.23	12.36 ± 0.66

The organs that had the highest absorbed dose were testicles and thyroid in both protocols. It happened because the measurements of these organs were done on the phantom skin, as they are very superficial organs. The results show that the absorbed doses by the organs are on average 145% higher for the diagnostic protocol. When organ and tissue radiosensitivity is analyzed, the anatomical protocol provides 6.86 mSv of effective dose, while the diagnostic protocol provides 17.48 mSv. It is important to emphasize that is necessary a careful evaluation in the protocol choice to be performed, using the protocol with high imaging quality and consequently a higher dose for all patients only in cases in which this protocol is essential, not as routine for all patients.

ACKNOWLEDGEMENTS

The authors are thankful to CAPES, FAPEMIG and CNPq for the financial support and to Luxemburgo Hospital for the phantom loan.

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140-MEASUREMENT OF THE LINEAR ATTENUATION COEFFICIENT OF RESIN-BASED WATER-EQUIVALENT MATERIALS

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Phantoms are devices that represent the human body both in shape and composition. Phantoms materials are designed to simulate different human tissues such as bone, lung, soft tissue, muscle and others [1]. Liquid water is also a widely simulated material because water comprehends a significant fraction of the human body. To represent human tissues, phantom materials must simulate its physical and radiological properties. Mass density ρ and effective atomic number Z_{eff} can describe the physical quantities, but radiological characterization requires knowledge of the attenuation properties because such properties are highly dependent on energy. Radiological properties can be well-represented by the linear attenuation coefficient $\mu(E)$. In recent publications, measurements of the linear attenuation coefficients were done for characterization of breast tissues [2] and bone tissue substitutes [3].

In order to characterize water-substitute materials, four resin-based water-equivalent samples developed by a mathematical algorithm [4] were analysed and had their linear attenuation coefficient determined within the energy range from 10 to 150 keV. A poly-energetic narrow beam was adopted using a tungsten-target x-ray tube (model MG 450, Philips Medical Systems, 3000 Minuteman Road, Andover, MA) attached to a constant-potential generator. The tube was operated at constant-potential of 150 kV and had a cylindrical collimator with internal diameter of 1.5 mm placed 14 cm from the x-ray tube focal spot. An additional filtration of 4.6 mm of aluminium was also used after the collimator.

The detection system consisted of a Cadmium Telluride (CdTe) spectrometer (model XR-100T, Amptek, inc., Bedford, MA) and associated electronics. For the narrow beam geometry, the detector was placed 298 cm from the x-ray tube focal spot and two cylindrical collimators were additionally used with internal diameter of 2 mm and 0.04 mm, respectively placed farther and closer from the CdTe crystal. These collimators were placed inside the stainless steel collimator housing separated by a brass spacer of 36 mm from Amptek's EXVC X-Ray Collimator Kit.

Every sample had six slices of approximately 1 cm thickness each. In six steps, one for each slice, samples thickness was gradually increased from approximately 1 cm to 6 cm. The first slice was placed 18 cm from the x-ray tube source and successively the other slices were added. X-ray spectra acquisition was done using 300 s and the measured spectra was corrected using a stripping algorithm [5,6].

The linear attenuation coefficient was experimentally determined based in the Lambert-Beer's law, which is valid for the narrow beam geometry. The incident $I_0(E)$ and transmitted $I_i(E)$ beam intensities of photons for discrete energies E were recorded for six different sample thickness x_i. Hence, the mean linear attenuation coefficient $\bar{\mu}(E)$ of the samples was calculated by Eq. (1).

$$\bar{\mu}(E) = -\frac{1}{6}\sum_{i}^{6}\frac{1}{x_i}\ln\left(\frac{I_i(E)}{I_0(E)}\right) \tag{1}$$

Fig. 1 shows the mean values of the linear attenuation coefficient for all samples using Eq. (1) and the linear attenuation coefficients of liquid water NIST XCOM database [7]. To compare samples, the relative root mean square error (rRMSE) was used. rRMSE for samples A, B, C and D are respectively 15.6%, 14.5%, 15.2% and 13.8% within energy range from 10 to 150 keV.

Comparison test showed that linear attenuation coefficients of all four resin-based samples are statistically in accordance with the linear attenuation coefficients of liquid water. Overall, sample D presented the closer result as a water-equivalent material.

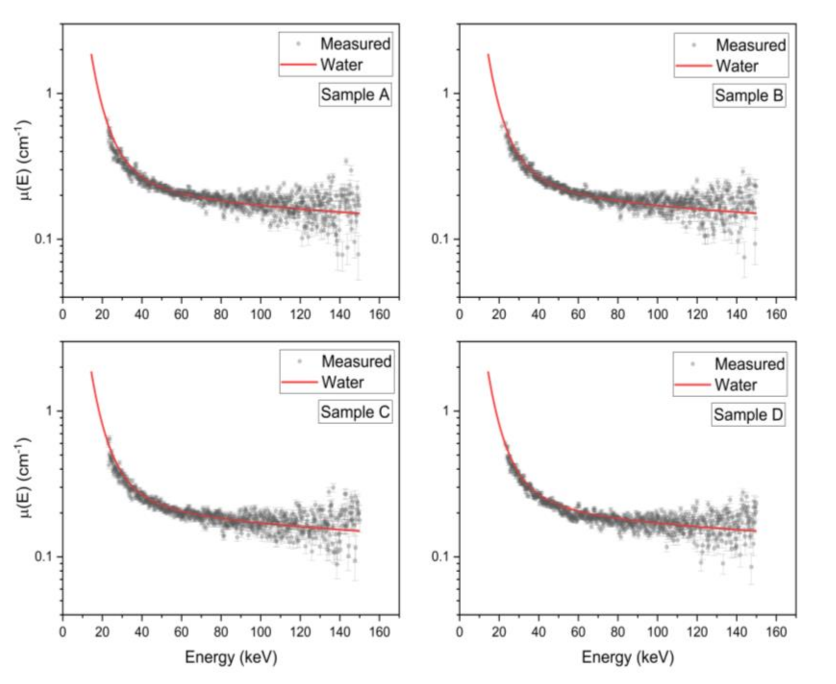

FIG. 1. Mean linear attenuation coefficients (gray points) using Eq. (1) for samples A, B, C and D measured using 150 kV. The red line represents the coefficients of liquid water obtained using NIST XCOM database [7].

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141-USING MODERN RISK-INFORMED, PERFORMANCE BASED APPROACHES TO ENHANCE THE NRC'S MATERIALS INSPECTION PROGRAM

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The U.S. Nuclear Regulatory Commission (NRC) conducts materials inspections to verify that licensees authorized to possess, use, store, transfer, and dispose of radioactive material for various types of use properly conduct their activities to protect public health and safety and security of licensed material. The various types of use include industrial, academic, research and development, manufacturing, distribution, irradiators, well logging, industrial radiography, medical programs, and services (e.g., leak testing of sealed sources, calibration of instruments, servicing of devices, transportation). Although these inspections are currently being performed using a risk-informed, performance-based approach, the NRC is revising the materials inspection program to further risk-inform and modernize the program in a way that makes the program more efficient and effective but does not diminish public health and safety or security. This initiative involves updating each of the Inspection Procedures (IPs) for the various types of use of licensed material, reviewing and revising materials Incident Response inspection documents, evaluating potential changes to inspection frequencies and priorities, providing additional guidance on dispositioning low safety significant inspection items, and evaluating opportunities to provide inspectors with upgraded Information Technology to support field inspections.

The presentation will highlight recent changes to the NRC's materials inspection program and provide details regarding on-going activities to address the proposed changes to the inspection program noted above.

142-RADIOLOGICAL CHARACTERIZATION AND SAFETY DURING DECOMMISSIONING OF A NUCLEAR REACTOR

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Radiological characterization for spent nuclear fuel and structure materials of a nuclear reactor plays an important role in the decommissioning plan. It is the basis for radiation protection and safety of the workers, public and environment. It can support the identification of contamination and assessment of potential risk. The important role and the significance of radiological characterization become clear when the following objectives are considered:

- Determination of the type, isotopic composition and extent of contamination in spent nuclear fuel, structures, systems and components;
- Identification of the nature and extent of remedial actions and decontamination methods;
- Supporting decommissioning planning;
- Cost estimation of decommissioning

Radiological characterization is relevant in all stages of the life cycle of a nuclear reactor, with different levels of detail and with different types. Basically, the following characterization stages can be distinguished:

- Pre-operational characterization;
- Characterization during operation;
- Characterization during the transition stage (after final reactor shutdown and before decontamination and initiation of dismantling);
- Characterization during dismantling (including remediation and decontamination);
- Characterization to support the final status survey for site release.

The most comprehensive characterization activities are usually carried out during the transition stage in preparation for implementation of dismantling activities, or during the dismantling stage where systems, structures, components, and buildings have to be characterized for decisions regarding the extent of decontamination.

Graded approach using comprehensive accurate software can be used for the calculation of the spent nuclear fuel inventory and activation products in the structure materials of the reactor. The information can be used for selection of proper fuel transport and in case of accidental conditions for the spent fuel and in the choice of decontamination methodology. Also, measurements, smear tests and sampling should be available as a complementary method for successful radiological characterization.

In this paper, examples of sophisticated software for characterization and calculation of source term, the radionuclide inventory, amount of fissile materials (U-235, Pu-239 and Pu-241), burn-up, activity of each nuclide against cooling time and activation products will be presented. The calculations depend on the actual irradiation history, initial enrichment, discharge burn-up, fuel cell array, dimensions, cooling medium and neutron cross section data library. The isotopic composition and activation products can be changed during cooling time as different decay schemes lead to production and destruction of some new nuclides. Also, radiation protection planning, and safety management will be discussed in case of emergency exposure situations during decontamination and dismantling stages based on international basic safety standards.

143-A SURVEY ON RADIATION RISK COMMUNICATION -THE CASE OF THE CONVERSION OF THE NIGERIAN RESEARCH REACTOR

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In the year 2018, the Nigerian Research Reactor-1 (NIRR-1) core was successfully converted from the use of Highly Enriched Uranium (HEU) fuel to the use of Low Enriched Uranium (LEU) fuel. The conversion, which is a radiation high-risk activity, involved removal of the HEU reactor core and repatriating it back to China- its country of origin. The period of the conversion exercise was characterized by unusual increase in vehicular movement and security presence in and out of the premises of NIRR-1. The premises of NIRR-1 are separated from a community of people by just a few meters. A pilot survey conducted by the authors of the study revealed that the immediate community to the premises of NIRR-1 had been living with anxiety and uncertainty about the safety of its residents during the conversion exercise. The residents' safety concern was occasioned not by the knowledge of the conversion exercise going on but by the unusual increase in security presence and movement within their locality. As members of the public usually have incomplete knowledge and a great deal of uncertainty when it comes to issues of nuclear and radiation safety, it was the responsibility of the regulatory body to appropriately inform and educate the community about the radiation risk associated with the conversion activities. In the event of nuclear accidents or radiological attacks, knowledge and information about radiation risk are very important for radiation protection for the public. The aim of the survey was to establish the status of radiation risk communication among the neighbouring community of NIRR-1. The specific objectives were to determine: the proportion of the community's residents who were knowledgeable about the concept of radiation; the proportion of the residents who were aware of the NIRR-1 facility and the conversion activities; and the proportion of the residents who felt they needed enlightenment on any future activities or operations of the reactor facility that might engender a feeling of anxiety or fear among them. Between 2nd to 31st October, 2019, anonymous survey questionnaires developed by the authors and validated by a panel of experts were delivered by hand to heads of households in residential quarters immediately neighbouring the premises of NIRR-1. Estimated number of 200 households was contacted in Area A and Jama'a residential quarters. All 200 responses were returned. Survey questions relating to the three thematic areas, namely; radiation knowledge, awareness status and enlightenment needs were tested against the educational level of the respondents. Analysis of the obtained data was done using simple percentages. Consequently, recommendations for future risk communication to the community were made. From the results of the study, it was found that, 100% of the respondents had at least secondary school level education, indicating that the NIRR-1 facility is sited in an area where most of the neighbouring community residents have at least a basic level of education. A significant 93.5% of the respondents had knowledge about the term 'radiation' in general, which term they probably learn from today's widespread usage of computers and modern communication gadgets. But only 3% of the respondents knew what ionizing radiation was about, indicating that the majority of them did not take science subjects in secondary school. But of those who knew what ionizing radiation was, only 16.7% had information about its harmful effect, and none knew what appropriate actions they would take to protect themselves in the event of accidents or incidents. Expectedly, a significant 88.5% of the respondents were aware of the increased security presence and movement in and out of the premises of NIRR-1 during the period of the conversion exercise, as these activities took place in their neighbourhood, but none had official information about what was going on. On the enlightenment need of the respondents about any future activity concerning NIRR-1 that might engender anxiety or fear among the residents, 84% answered in the affirmative, 4.5% answered in the negative, while 11.5% could not say whether they needed enlightenment or not. Of those who needed enlightenment, 46.4% wanted the enlightenment to take place in lecture halls, 22% via radio broadcast, while 31.6% through printed materials. The study was the first of its kind to assess the status of radiation risk communication among the neighbouring community of NIRR-1 since after the conversion exercise. In conclusion, the main findings of the study were: significant lack of understanding among the community residents of what ionizing radiation entails despite most of them having at least secondary school level education; absence of radiation risk communication in the community; and gross lack of knowledge among the residents of the essence and purpose of NIRR-1. Based on the analysis of the results, all the respondents had a minimum of secondary school education which is a basic education needed for effective communication, and so, this would make it easier for the responsible authority to take radiation education easily to the community. The study did not address what the content of the radiation education would be, but the authors recommend that, the responsible authorities like the Nigerian Nuclear Regulatory Authority should launch an educational programme for the community on the knowledge of radiation and activities and operations of NIRR-1. This would help to allay fears and uncertainties arising from the activities involving continuous operation or modification of the NIRR-1 facility. It would also prepare the community for easy access and handling should any emergency arise concerning the NIRR-1.

144-ASSESSMENT OF RADIATION SAFETY IN OPHTHALMIC BRACHYTHERAPY USING MONTE CARLO SIMULATION

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Ophthalmic brachytherapy with ruthenium-106/rhodium-106 (^{106}Ru/^{106}Rh) eye plaque has been increasingly used to treat eye cancer due to effectiveness outcomes. ^{106}Ru/^{106}Rh emits several beta particles with the maximum energy of 3.54 MeV as well as various energies of gamma rays [1] resulting in increasing radiation dose from both gamma rays and generated bremsstrahlung radiation to the workers. In the treatment procedure, the medical staff is required to work closely to ^{106}Ru/^{106}Rh eye plaque for a period of time in order to place the eye plaque adjacent to the lesion. The radiation dose to the staff especially to the eye lens should be considered to ensure the radiation safety of the procedure. Therefore, the study was aimed to determine the radiation doses to the eye lens, fingers and whole body using Monte Carlo stimulation.

The main experiments were separated into radiation measurement and Monte Carlo simulation. The first part, high purity germanium (HPGe) detector was used to measure ^{106}Ru/^{106}Rh eye plaque in order to compare with Monte Carlo simulation. The HPGe detector was pre-calibrated with the standard sources to cover the energy between 59.4 keV and 1,331 keV. The counting rate and photon spectrum of the ^{106}Ru/^{106}Rh eye plaque were measured by a calibrated HPGe detector by attaching convex side of the eye plaque at the eye area of the RANDO phantom.

For simulation part, Monte Carlo N-Particle version 5 (MCNP5) was chosen to simulate radiation doses from the ophthalmic brachytherapy procedure. Validation of MCNP code was primarily performed by comparing photon spectrum of the standard source, ^{137}Cs, and ^{106}Ru/^{106}Rh eye plaque on the RANDO phantom with the HPGe measurement using similar condition to that of experimental measurement. Then validated MCNP simulation was used to calculate radiation dose rates (microsievert per hour) from 10 MBq ^{106}Ru/^{106}Rh plaque (nominal activity) to the eye lens, fingers and whole body of the staff performed the eye plaque implantation. The MCNP simulation geometry of the plaque implantation procedure was illustrated in figure 1. The distance from the eyes of staff to patient was approximately 40 cm and 10 cm was the distance between staff's fingers and the eye plaque in the patient's left eyeball area.

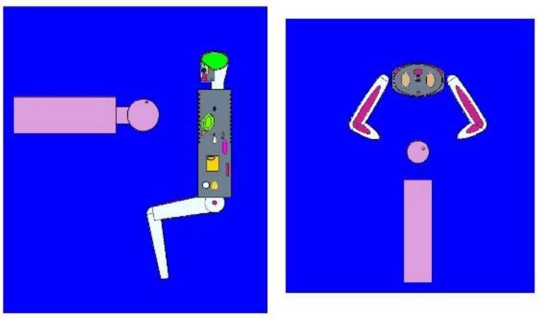

FIG. 17. Geometry of ^{106}Ru/^{106}Rh plaque implantation procedure side view (left) top view (right).

The result of the MCNP5 code validation with ^{137}Cs standard source was shown 0.3 % different between two approaches and the similar trend of ^{106}Ru/^{106}Rh spectrum between HPGe measurement and simulation was revealed as illustrated in figure 2.

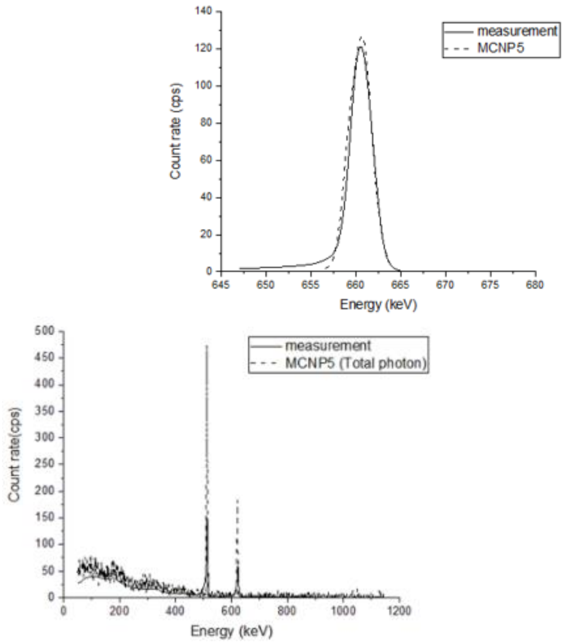

FIG. 2. Comparison between HPGe measurement and MCNP5 simulation of ^{137}Cs standard source (left) $^{106}Ru/^{106}Rh$ plaque (right).

The radiation dose rates to the eye lens, fingers and whole body of the medical staff performed $^{106}Ru/^{106}Rh$ eye plaque procedure were demonstrated in Table 1.

TABLE 1. RADIATION DOSE RATES OF MEDICAL STAFF PERFORMED EYE PLAQUE IMPLANTATION

	Radiation dose rate $\pm$ SD (microsievert per hour)
Equivalent dose to the eye lens	7.02 $\pm$ 0.01
Equivalent dose to the fingers	89.90 $\pm$ 0.02
Whole body dose (effective dose)	24.10 $\pm$ 0.03

In general, duration of the eye plaque implantation is about one hour so the radiation doses to the eye lens, fingers and whole body of the medical staff per one treatment procedure should be about 7.02, 89.90 and 24.10 microsievert, respectively. Regarding International Commission on Radiological Protection (ICRP) 118, recommendation for the annual occupational dose limits for the eye lens and fingers are 20 millisievert and 500 millisievert, respectively [2]. Then medical staff could perform $^{106}Ru/^{106}Rh$ ophthalmic brachytherapy up to approximately 2,849 procedures and 5,562 procedures based on eye lens dose and fingers dose, respectively. Meanwhile annual effective dose limit is 20 millisievert so the medical staff could perform 830 procedures per year to keep radiation dose lower than the limit.

In conclusion, the estimated radiation doses to the eye lens, fingers and whole body of the medical staff using Monte Carlo simulation were successfully determined. In addition, results from the study could suggest that ophthalmic brachytherapy with $^{106}Ru/^{106}Rh$ eye plaque was a safe procedure according to ICRP 118 recommendation.

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145-RADIATION SAFETY TO THE PUBLIC AND WASTE MANAGEMENT OF RADIOACTIVE IODINE IN PATEINTS TREATED FOR HYPERTHYROIDISM AND THYROID CANCER

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I-131 is the common radioisotopes used for ablation of functioning thyroid for hyperthyroidism management and for the elimination of post-surgical residual thyroid tissues in differentiated carcinoma of thyroid [1]. In 2019, MaharatNakhonratchasima Hospital,Thailand were radioiodine treatment 267 differentiated thyroid cancer (193 undergone as inpatient and 74 as outpatient) and 586 hyperthyroidism(undergone as outpatient). After administration, radioiodine will be excreted from the patient primarily by the kidneys, the next most significant pathway is saliva and lesser pathways are sweat and faeces [2]. As we know that the secretion from the patient who treated with radioiodine will contain radioisotope as waste unsealed source, its potential adverse effect in both occupational and environmental fields. To protection of people and the environment, all radioiodine patient treatment before discharge from the hospital were measured dose rates at 1 meter immediately after radioiodine administration for outpatient and sequentially 24,48,72,96 h intervals for inpatient. To ensure the radiation exposure to an individual (public, radiation worker, and patient) and the environment does not exceed the prescribed safe limits by according to NRC regulatory to release patient from isolation, the radioactivity should be remaining in the body less than 32.43 mCi or dose rate at 1 m less than 0.07 mSv/h. The mean measured dose rates at 1 m in µSv/h unit from the patient who treatment with a typical activity as show in Table 1.

TABLE 1. MEASURED DOSE RATES AT 1 M FROM PATIENTS POST ADMINISTRATION OF I-131

Patients	0 h	24h	48h	72h	96h
			Dose rates in µSv/h		
Thyroid cancer					
100 mCi	143.02±23.42	30.03±8.49	11.39±6.43	6.05±4.71	4.59±4.48
150 mCi	191.74±26.24	42.37±14.43	14.74±7.6	7.32±4.52	4.38±3.28
200 mCi	250.12±28.67	60.98±23.20	27.14±21.6	15.92±19.70	16.64±25.73
Hyperthyroid					
7 mCi	10.86±3.14				
10 mCi	14.95±3.72				
15 mCi	20.73±4.7				
20 mCi	25.60±5.73				
30 mCi	39.69±9.51				

Thyroid cancer patient who treatment with activity over 30 mCi or measured dose rate at 1 m is more than 0.07 mSv/h require isolation in a lead shielded room for a certain period of time (usually 2-4 days). The most of the administered activity in thyroid cancer will appear in the urine because the lack of thyroid tissue. The fraction will largely be determined by the amount of remnant and metastatic thyroid tissue. In most cases, 50-60% of the administered activity is excreted in the first 24 hours, and around 85% over a stay of 4-5 days [2]. This represents a significant potential for radioactive contamination. Therefore, to management the radioactive waste can be done by two stages with collection and disposal [3]. After the patient getting flushed, the excreta and urine of patients will passes the PVC pipes through the route into "I-131 waste management system" to collection as show in Fig.1

a: system diagram b:tanks and pump c:tanks in concreat container

d:lead shield door e:manual control f:tele-automatic control

FIG. 1. Showing the semi- automatic I-131 waste management system.

The I-131 waste management system compose of 1 septic tank of 4000 liters,1 holding tank of 4000 liters, 3 delay tanks of 4000 liters and 1 sampling tank of 1000 liters and all of tanks put into container made with concrete wall thickness 20 cm. The system work as a semi-automatic by the first tanks to receive the excreta and urine is septic tank; heavier solids sink to the bottom and undergo bacterial digestion. This reduces the quantity of solids and also changes its composition to sludge, which builds up in the bottom of the tank [4]. The remaining liquid, called effluent, flows from the tanks through the holding tank, when the holding tank full, the pump will do automatic by pumping the waste to the delay tank no.1, then delay tank no.1 will start day counting. When the delay tank no.1 is full, the holding tank will pump the waste to the delay tank no.2 and no.3 with a sequentially. At the end, all tank will be full, the waste system will automatic pump the waste from the tank no. 1(longest day) through the sampling tank to measurement dose rate by AMP50 (Waterproof GM-tube) 3 times before disposal to the sewerage system of the hospital as show in Fig.2.

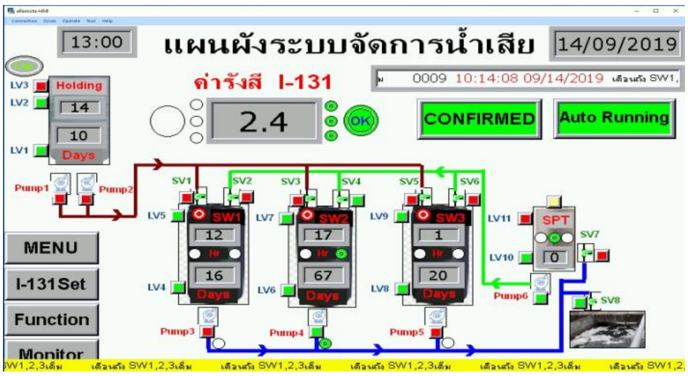

FIG. 2. Showing the work sequentially of I-131 waste management system

In addition to the management of radioactive hospital waste on scientific lines, the basic principles for radiation protection to be adopted are justification of practice (use radiation only if benefits outweigh the risks), optimization of practice (keep magnitude of individual dose and number of people exposed as low as reasonably achievable, ALARA) and dose limitation [3].

ACKNOWLEDGEMENTS

The authors thank to all staff worker, especially radiological technologist team in the Nuclear medicine department, MaharatNakhonratchasima hospital for their kindness and hard work.

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147-THE NUCLEAR SCIENCE MUSEUM: AN INTERDISCIPLINARY LEARNING ENVIRONMENT TO IMPROVE CITIZENS' COMPREHENSION OF RADIATION RISKS AND BENEFITS

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Risk perception and risk acceptance are a matter of education and properly communication. Although the important role of ionizing radiation to industry, medicine and power generation, nuclear technology remains a controversial issue among a great fraction of Brazilian population [1]. Misinformation and common wrong beliefs confuse public's perceptions of risks and benefits [2]. Most often, people fear the harmful effects of ionizing radiation to human health and the environment. Sensitive to the need of new approaches between scientific community and society, the Nuclear Science Museum of Nuclear Energy Department – UFPE, Brazil, has developed a range of education programs to improve communication about nuclear technology.

INTRODUCTION

Controversies on the biological effects of radiation distort and confuse public's perceptions of radiation risks and benefits [3 - 4]. Internet, the most used source to gather information, most often presents a great amount of apparently contradictory information. Regarding the biological effects of low-dose radiation, Internet reports the exposure to indoor radon as a risk factor for lung cancer [5]. Internet also reports that the monazitic sand, radioactive due to the presence of thorium, brings benefits to humans´ health [6].

It is not easy for the population to understand contradictions or to identify reliable sources. Individuals of the public do not understand highly specialized papers. Sensationalistic material and anti-nuclear information, on the other hand, seem to be easily understood [7]. It is therefore questionable whether the scientific community must develop effective education actions to bridge the gap between nuclear science and the public. The objective of this paper is to communicate strategies and actions developed by the Nuclear Science Museum of Nuclear Energy Department – UFPE to improve nuclear science education in Brazil.

METHODOLOGY

The Nuclear Science Museum, created in 2010 in Pernambuco, northeast Brazil, is the first and the only Brazilian museum fully dedicated to nuclear science. It is an interactive space to teach adults and children about the beneficial applications of nuclear technology. Although traditionally museums' role is to preserve and present artefacts of cultural and historic importance, over the past decade, this museum has experienced new missions and great challenges, providing different programs to educate teachers and students, beyond the school limits. Indoors, the museum presents a wide range of nuclear techniques as well as their beneficial applications in industry, medicine and nuclear power generation. Outdoors, there are travelling exhibitions with itinerant expositions in shopping malls and schools. Furthermore, the museum provides specific one-week courses and workshops for high-school science teachers. The Summer Course, for instance, is a program of the Brazilian National Network of Education and Science to improve science education throughout the country. Also, the Nuclear Science Museum participates every year in national thematic weeks, developing education activities, lectures, workshops and guided tours for groups.

Finally, Brazil is a country with continental dimensions and in order to reach teachers and students all over the country, the Nuclear Science Museum has developed a vast range of free multimedia materials, available online to educate teachers and students through interactive activities and games [8].

Every program for the public is designed to provide a wider perspective on the benefits and risks resulting from ionizing radiation in everyday life, making nuclear science education effective and significant.

RESULTS AND DISCUSSION

Other than communicating the growing impact of the beneficial applications of nuclear technology in medicine, industry, agriculture and electric power generation, the museum presents and discusses about radiological protection, occupational radiation exposure and basic elements of radiation protection, shielding, protective equipment and dosimeters. This is fundamental strategy regarding public acceptance of nuclear technology.

People fear what they cannot understand, and the scientific community is expected to communicate about the impacts of ionizing radiation in daily life. Teachers themselves are most often unaware of the issue and it is a must and a challenge to increase teachers' knowledge, providing reconstructive questioning and critical thinking. Experience proved that schoolteachers who attended lectures, workshops and training courses were more likely to return to the museum with their students.

The results support the idea that the museum is an effective interdisciplinary learning space. From November 2018 to August 2019, the Nuclear Science Museum received about 4300 visitors, mostly school groups from public and private schools. On a 10-point scale, where 10 means a great experience, 88,6% of the visitors rated 9 to 10, proving the high-quality experience in learning nuclear science at the Museum.

FINAL CONSIDERATIONS

Although nuclear science is a prevalent issue of discussion in Knowledge Society, the predominance of misinformation, unfounded prejudices and wrong mass media information distort and confuse public's perceptions of radiation risks and benefits.

The balance between risk perception and risk acceptance depends on effective, trustworthy and understandable information. It is essential to promote a fair comprehension about the biological effects of radiation, as well as the role of nuclear technology and its impacts on the world's economic and social progress.

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148-TREND OF RADIATION SAFETY TRAINING FOR HAMAD MEDICAL CORPORATION RADIATION WORKERS

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Radiation safety training is generally provided to promote a radiologically safe work environment, and to help ensure that doses are kept As Low As Reasonably Achievable (ALARA). Radiation safety training in Hamad Medical Corporation started on 2002 and has been developed from small lectures into full training programs in addition to training for different specialities. In 2019, the total number of trainees were 906 including Physicians, Radiologists, Medical Physicist, Technologists as well as Nurses. The aim of the training course is to ensure that the basic instruction, information, training and supervision of radiation safety practices is being observed and implemented in the workplace.

INTRODUCTION

As per the Qatari Radiation Protection Law No. 31 of year 2002, the radiation protection training is mandatory for each individual who works with ionizing radiation in the medical field as requirement to hold a valid personal radiation license [1]. The radiation safety support services in the Radiation Safety Section - Occupational Health and Safety (OHS) Department should provide all workers with the appropriate training on radiation safety including information on the health risk due to their occupational exposure [2].

METHODOLOGY

This study was divided into two phases as follows:

Phase1: 2007-2016: The Radiation Safety Training program was held 8 to 10 sessions per year including the practical exercises using tools and equipment needed and preparing a handbook which is 30 pages (prepared by Radiation Safety Section) that summarized the whole program lecture topics and was prepared according to the International Basic Safety Standard (BSS) as issued by the IAEA (International Atomic Energy Agency) [3]. This workshop included both HMC and non-HMC staff participants.

Phase 2: During this phase, the training became more specialized and increased due to the huge number of workers recruited during the last 3 years and additional hospitals were opened during that time for a total of 13 hospitals under HMC. The training courses were organized for HMC staff only while specific training were done for dedicated audience such as training course for CT, Nuclear medicine, breast imaging, RPO and Physicians.

RESULTS

As shown in Table 1, the total number of trainees increased from 258 for the year 2007 (for both HMC and Non-HMC staff) to 906 in the year 2019 (for HMC staff only). This number involved all radiation and non-radiation worker categories. The total number was increased by almost 71% in the year 2019 than that for 2007. The table also illustrates the drop of the radiation worker participants in the year of 2016 due to the shift from phase 1 to phase 2 then started to increase at the following years to register the highest number in 2019. This high number in 2017-2019 was for HMC staff only. Details of the trainee's total number during the year 2019 is illustrated in FIG 1. The number of courses in some months was more than four and the reason behind is that the course is mandatory for all radiation workers to renew their personal radiation license. That's why some courses were repeated during the year.

TABLE 1. DISTRIBUTION OF THE NUMBER OF TRAINEES FOR HMC AND NON-HMC During The 13 Years

Year/number of trainees	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017	2018	2019
HMC staff	258	160	113	120	282	120	312	426	304	31	349	623	906
Non-HMC staff	-	-	-	178	201	73	170	316	305	66	-	-	-
Total number	258	160	113	298	483	193	482	742	609	97	349	623	906

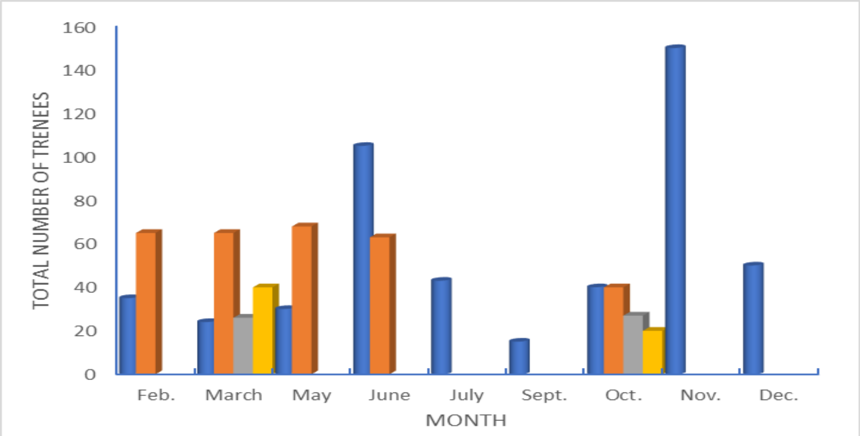

FIG 1: Chart showing the number of trainees during year 2019

The trainings provided serves not just for the sole purpose of fulfilling requirement for the Personal Radiation License Application. These can also be used for the application or renewal of QCHP License because of the approved CPD/CME points along with these courses which were also printed on the certificates. Certain other countries even recognized these certificates as a valued CPD points which can be used for the renewal of the local license of the country. To improve and enhance the future courses, evaluation forms will be distributed to all attendees after completion of the course.

CONCLUSION

Radiation awareness sessions should include other referring doctors and administrators within HMC. Specific sessions for these categories should be considered. The awareness of radiation protection issues has been increased and a lot of staff in the hospitals are taking precautionary measures while working in radiation area.

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151-RADIOLOGICAL SAFETY ASSESSMENT DUE TO SEVERE ACCIDENTAL CONDITIONS OF VVER-1200 REACTOR OF BANGLADESH

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It was reported in the past that nuclear reactors do not release a substantial amount of radioactivity to the environment throughout their regular operation. However, in extreme conditions, such as a nuclear accident, adequate radioactivity can be produced and released into the atmosphere. This influences a threat to human health and must be dealt with a regulatory requirement [1]. Results of such investigations are useful for emergency preparedness or mitigation plans if such incidents or accidents occur at the reactor site. At present, two nuclear power plants of VVER-1200 technology, a Generation III+ reactor, type AES-2006/V-392M (1200 MWe) is under construction that is expected to start operation in 2023 at Rooppur, Paksey Union, Ishwardi Upazilla, Pabna district, Bangladesh [2]. From the two units of Rooppur Nuclear Power Plant (RNPP), it is planned to generate a total power of 2.4 GWe. It will be the country's first nuclear power plant that is being constructed by the Rosatom Russian State Atomic Energy Corporation. In case of an accident of NPP, the released radioactive material then undergoes disperse into the air and deposit on the ground, which ultimately cause an impact on the environment. Human being has become the utmost concern after the Chernobyl accident in the former USSR in 1986 and most recently, after the Fukushima Di-ichi accident in Japan, 2011. Also, one of the major topics in nuclear safety is the quantitative evaluation of the radionuclides source-term in nuclear reactors under postulated accidental conditions. Therefore, the study of source-term evaluation, atmospheric dispersion of radioactive material in the environment, and radiological dose assessment are of utmost importance for the people in and around the RNPP site and regulatory requirement for updating the safety analysis report. The source-term and radiological doses were evaluated due to the release of radioactive material from the VVER-1200 type nuclear power plant that is under construction at Rooppur, Pabna district, Bangladesh. The scenario of the accident was developed considering the Station Black Out (SBO) accident due to loss of off-site electric power system simultaneous to turbine trip and absence of the onsite emergency ac power system and Loss of Coolant (LOCA) accident caused due to rupture of the reactor coolant system. The accident proceeds from the reactor core damage to the containment failure and finally releases of radioactive materials to the atmosphere. The analysis was performed by using the radiological assessment system for consequence analysis code RASCAL3.0.5 developed by the Office of Nuclear Security and Incident Response, US Nuclear Regulatory Commission [3-5]. The input parameters used in the RASCAL3.0.5 code for the model evaluation are dependent on the accident scenario development, the technical design parameters of the VVER-1200 type nuclear power plant in addition to the local meteorological parameters. In both dry and rainy seasons, the accidents were evaluated, and the assessment was performed until 48 hours period from the release initiation. Fifty-one radionuclides with activity were evaluated as a source-term that was used as a source of released radioactive materials to the environment. Radiological doses were assessed within 25 miles (40.2 km) radius from the reactor.

Based on the released source-term, the level of the accident was evaluated as level 6: Severe accident of the International Nuclear and Radiological Event Scale (INES) [6]. The results of maximum dose distribution in dry and rainy seasons indicate that the local meteorological conditions have a strong influence on the maximum dose distribution. Maximum dose distributions were significantly higher in rainy season than that of the dry season due to the occurrence of substantial wet deposition than the dry season. In the case of the dry season, the maximum doses are far below the acceptable limit of IAEA Basic Safety Standards and local regulatory guide. But in the case of the rainy season, the total EDE values are found to be 1.4 to 15 times higher for the general public than the acceptable limit in between the distance 4.8 to 11.3 km from the reactor. It is required to inform and coordinate earnestly with the Emergency Response Authority of the RNPP area to monitor the accident progression and take appropriate protective measures. It is recommended to stop right away from consuming food, meat, milk, and vegetable produce in the stated area until obtaining further sample testing results of the region. In case of emergency preparedness and public evacuation, the communication procedures of the general people around the site must be carried out according to the direction of the local regulatory authority.

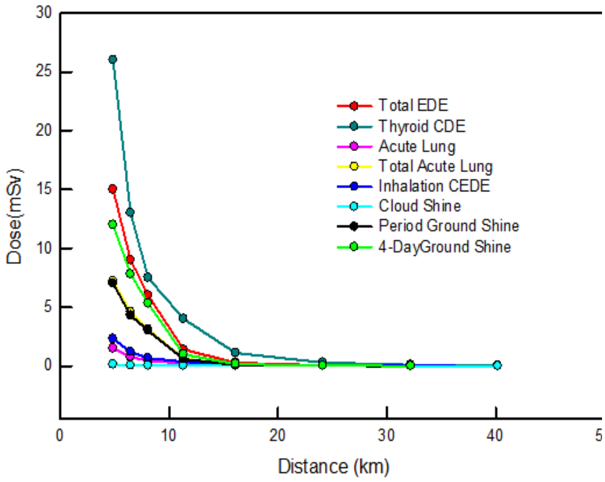

FIG. 1: *Maximum dose (mSv) as a function of distance (km) for the rainy season at dispersion time 42 hrs.*

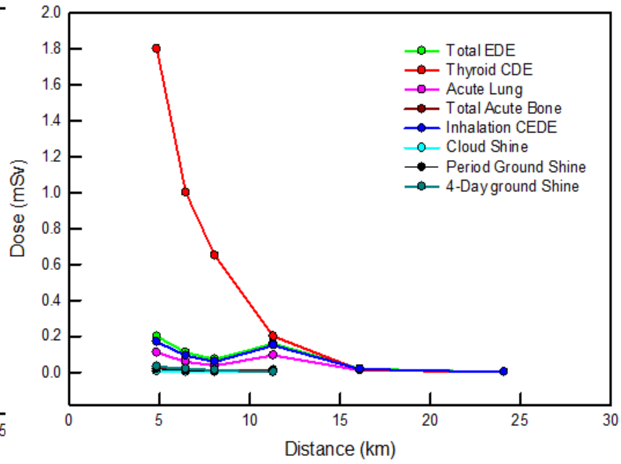

FIG. 2: *Maximum dose (mSv) as a function of distance (km) for the dry season at dispersion time 42 hrs.*

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152-OCCUPATIONAL RADIATION EXPOSURE FOR RADIATION WORKERS IN HAMAD MEDICAL CORPORATION

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ABSTRACT

A total of 1250 radiation workers from Hamad Medical Corporation (HMC) were monitored during the year 2019. Thermoluminescent dosimeters (TLD) allocated by Occupational Health and Safety Department are being used for radiation dose measurements for the staff working in radiation areas at HMC hospitals. Bi-monthly dose measurements were regularly collected for a period of 14 years (2006 to 2019). The selected number of Radiation Workers (RW) for this study was 991 as per 2019 statistics for the staff working in the following departments; Radiology, Oncology, Nuclear Medicine, Cath-Lab, Urology and Operational Theater. The maximum annual effective dose for all monitored workers was 3.594 mSv which was registered for the Cath Lab nurse category during the year 2012. The average individual annual dose in different categories is less than 1 mSv/y and no individual approached the dose limit of 20 mSv as recommended by the ICRP. The aim of this study is to evaluate the radiation dose received by RW in various HMC hospitals.

INTRODUCTION

The HMC Occupational Dose Monitoring Program was considered the essential indicator to ensure the efficiency of radiation safety condition in all applications. This is in accordance with HMC policy formulated for all Hamad Medical Corporation (HMC) health care providers working with or near radiation sources to ensure their occupational exposures to radiation are maintained as low as reasonably achievable and are within the occupational dose limits in line with the national regulations and with the requirements of the international safety standard [1-3]. This study aims to assess the RW annual effective dose over the last fourteen years in different medical practices within HMC hospitals.

METHODOLOGY

Thirteen hospitals belong to HMC were included in this study. Two TLD readers type; Harshaw Model 6600 plus and model 8800 plus with their TLD cards were used by the Personal Monitoring Service in HMC. According to IAEA guidelines, the dosimeter should be worn at chest level for estimating personnel dose equivalents Hp (10) and Hp (0.07). Hp (10) were used as the best estimate for effective dose. Workers used one individual dosimeter to be carried continuously while at work.

RESULTS

FIG 1. Illustrates the distribution number of monitored worker during the last fourteen years; 2006 to 2019. The occupational dose of five categories; Doctors, Technologists, Nurses, Medical Physicist including Assistants (diagnostic, nuclear medicine and radiotherapy), and others (admin, security, bioengineers and speech therapy) were evaluated. As shown in FIG.1, the number of TLD monitored workers increased from 251 in the year 2006 to 991 in 2019 which is four times than the initial number. This is primary due to the opening of new hospitals and recruiting new radiation workers accordingly.

Table 1 indicates the maximum annual dose. The current occupational dose limit is 20 mSv per year averaged over defined periods of 5 years and according to the HMC policy No. SA 1065, the investigation levels for intervention, Nuclear medicine and general practice are 6 mSv, 4.2 mSv and 2.4 mSv per year, respectively [3]. The highest annual doses for different categories that has been recorded during this period was 3.594 mSv: received by Cath-Lab nurse, 3.468 mSv: received by anesthesia technologist, 3.365 mSv: received by Radiologist, 1.033 mSv: registered for oncology Medical Physicist and 0.62 mSv for Speech Therapist. For all the monitored categories, the doses were well below the national dose limit of 20 mSv per year. All annual doses were below 6 mSv which is the investigation level for workers in Interventional Radiology. Therefore, the staff monitoring effort make radiation workers more aware, and helped them to improve their radiation protection practices. The maximum annual effective dose was registered for the Cath-lab individuals because of the high doses associated with such interventional procedures.

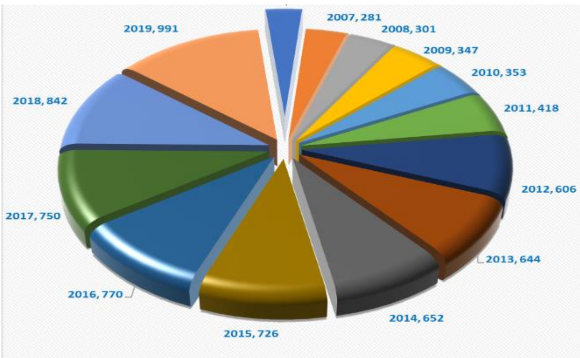

FIG. 1: Distribution number of monitored workers during the last fourteen years: 2006 to 2019.

TABLE 1: MAXIMUM ANNUAL EFFECTIVE DOSES FOR All CATEGORIES DURING THE LAST 14 YEARS

Category	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017	2018	2019
Doctors	1.297	1.943	1.750	2.096	2.726	2.207	2.209	2.664	3.365	1.808	1.709	1.560	1.662	1.895
Nurses	1.429	2.219	2.964	1.173	1.913	2.360	3.594	1.810	2.378	2.127	1.150	1.423	1.432	2.404
Techs	1.459	2.193	2.362	2.655	1.914	3.237	3.468	3.014	2.974	2.581	1.453	3.064	2.960	2.500
Medical Physicist	0.364	0.699	0.846	0.690	0.604	0.732	1.023	1.033	0.562	0.564	0.830	0.705	0.690	0.820
Others	-	-	-	-	-	-	-	-	-	0.518	0.536	0.582	0.620	0.614

CONCLUSION

The average individual annual doses for different workers were less than 1 mSv per year. The maximum doses of all occupationally exposed workers were well below the national dose limit of 20 mSv per year. Furthermore, all doses during the last fourteen years were below the annual investigation level for workers in interventional radiology which is equal to 6 mSv.

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153-RELEASE OF PATIENTS CONTAINING RADIOACTIVE MATERIAL FOLLOWING MEDICAL PROCEDURES

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The U.S. Nuclear Regulatory Commission (NRC) authorizes the release of patients containing radioactive material following medical procedures when the radiation dose to members of the public from the patient is not likely to exceed 5 mSv. In addition, written safety instructions must be given to the patient if the radiation dose to others could exceed 1 mSv. Patient release has a prodigious and controversial history, including whether the release limits are per treatment or per year; the quality of patient safety instruction; and adherence to instructions. In January 2018, the NRC completed an evaluation of its regulatory requirements for patient release. The NRC believes its current patient release regulations are protective of public health and safety, the release criteria appropriately balances public safety with patient access to medical treatment, and that changes to the release criteria are not warranted. Furthermore, the NRC considers that radiation exposure to other individuals from released patients can be safely controlled by the current patient release criteria, patient dose calculations, and patient adherence to the safety instructions. However, the NRC has determined that a comprehensive update to its patient release guidance, including the equations and methodologies described in guidance to calculate dose to members of the public from released patients, would improve the patient release program. The presentation will provide an overview of the NRC's patient release evaluation and an update on guidance development.

154-RELEASE OF ANIMALS CONTAINING RADIOACTIVE MATERIAL FOLLOWING VETERINARY PROCEDURES

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Over the last few decades, the U.S. Nuclear Regulatory Commission (NRC) has authorized the release of animals containing radioactive material following veterinary procedures when the exposure to members of the public from the animal is expected to be well below public dose limits, which are 1 mSv and 20 µSv any one hour. It has been the NRC's practice to allow veterinarians to give animal caregivers instructions to ensure these public limits are met. Historically, the most common procedures the NRC has authorized have involved cats who are treated with iodine-131, which are able to be released within a few days following treatment. To release cats, caregivers are given instructions of short duration (a few days to a week) and the instructions are not the primary mechanism to ensure public dose limits are met. More recently, the NRC has been requested to authorize several new veterinary procedures for pets, including use of tin-117m colloid to treat osteoarthritis in dogs. Due to the activity used during the treatment, the long half life of tin-117m, and the retention of the colloid, there is a higher potential that individuals who have daily interactions with a dog will exceed the public dose limit following the dog's release. In this instance, instructions would need to be the primary method to ensure public dose limits are met. This presentation will discuss the NRC's past practices in releasing animals containing radioactive material following veterinary procedures and highlight recent risk-informed initiatives to update these practices to evaluate new treatments.

157-OPTIMIZATION OF RADIATION PROTECTION FOR THE CONTROL OF OCCUPATIONAL EXPOSURE

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It is often stated that the main emphasis in radiation protection is on the optimization of protection. This phrase, however, does not readily bring to mind what actually needs to be done in the workplace to implement optimization.

In the presentation, an attempt has been made to demystify the concept by describing in direct terms what is to be done to carry out an optimization process and to free the way of thinking that is the foundation of optimization from excessive reliance on analytical techniques such as cost-benefit analysis, as these techniques are merely tools. In order to do this, the acronym ALARA has been used in the presentation as it brings to mind the twin concepts of dose reduction and reasonableness.

In describing a general approach to optimization, considerable attention has been paid to the full and systematic evaluation of the radiological conditions in a workplace. This analysis is crucial as it forms the basis for understanding what needs to be done, what can be done and what are the available approaches to getting it done. It also documents the starting conditions so that the effectiveness of the implementation of an ALARA plan can be monitored.

The other main component of the presentation is a general review of the means that are likely to be available in most workplaces to reduce exposure. These are divided into global means, which can be applied throughout an organization and those that are more jobs specific. Some of these global means are no more than would be expected in any well managed organization, such as an application of effective and efficient procedures for the management of work and provision for the education and training of workers. A well-managed and effective organization that pays due I regard to the safety of its workers will recognize the benefits of these means without the - application of a complex decision analysis. There are, however, situations in which the optimization of protection with respect to particular jobs is needed. In many of these cases it will still be clear that measures to reduce doses can be taken with little cost or even with savings through increased efficiency, or conversely that in other cases the necessary allocation ofresources would be disproportionate to the dose reductions. Nevertheless, there will be some cases in which it will not be obvious how much it is appropriate to do to reduce doses in a cost effective manner; some form of decision aiding technique can be helpful in such cases.

The outcome of an evaluation and analysis of options for improvement results in what has been called in this project work an ALARA plan. This is a combination of short term and long term orcontinuing actions. The effectiveness of an ALARA plan depends on commitment on the part of the management and workforce, which is fostered by the participation of both groups in the ALARA plan's formulation.

Monitoring the effectiveness of an ALARA plan provides the necessary feedback for sustaining appropriate attitudes to ALARA throughout an organization in the longer term.

The approach described in this project work is intended to be general and has therefore been expressed in broad terms. The examples given are intended to show how the approach can be and has been applied in different circumstances. The application will be at a different level of detail for a large facility or a small company, but in all cases the general approach set out can be adopted and applied for the benefit of radiation workers, managers and their organizations.

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158-ARTIFICIAL RADIONUCLIDES IN AIR, SOIL, GRASS MILK AFTER THE FUKUSHIMA ACCIDENT AND THE EFFECTIVE DOSES DUE TO THE DIFFERENT PATH-WAYS

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ABSTRACT

The concentrations of the ^{131}I, ^{137}Cs and ^{134}Cs radionuclides in air and water, vegetables and food chain, were determined in the region of Milan, Italy immediately after the nuclear accident in Fukushima, Japan. Grass and soil samples and samples from goat and cow milk were also analyzed. The detected activities in all environmental samples were very low. Moreover, an evaluation of the Effective Dose for population was done, based on the determination of the effective dose for inhalation and ingestion of ^{131}I, ^{134}Cs and ^{137}Cs and clearly showed that the effective doses were very far below levels of concern.

INTRODUCTION

Immediately after the Fukushima reactor accident, and as soon as the radioactive plume arrived in central Europe, systematic collection and analysis of environmental samples was undertaken in order to detect and quantify the radionuclides due to the Fukushima fallout in Milan Italy, by LASA Laboratory of the INFN Sez. of Milano and Università deli Studi di Milano.

The main focus was the detection of the radioactive nuclides of ^{131}I, ^{134}Cs and ^{137}Cs which easily enter the human body through inhalation and ingestion. Therefore, the concentrations of the above radionuclides in air and water, vegetables and food chain, were determined. Grass and soil samples and samples from goat and cow milk were also analysed. Moreover, an evaluation of the Effective Dose for population was done, based on the determination of the effective dose for inhalation and ingestion for ^{131}I, ^{134}Cs and ^{137}Cs measured.

RESULTS AND DISCUSION

During April 2011 the maximum ^{131}I, ^{137}Cs and ^{134}Cs were observed in Milan, Italy. Very similar results reported in Thessaloniki, Greece [1].The maximum observed ^{131}I activity was 467 μBq m^{-3}. The maximum observed activities of ^{137}Cs and ^{134}Cs in air were 63 μBq m^{-3} and 61 μBq m^{-3}, respectively. At the end of April 2011 all the observed values were below the MDA.

Iodine-131 in grass samples in Milan, Italy ranged between 37 and 135 mBq kg^{-1} and in soil samples between 0.57-1.99 Bq kg^{-1}. In fresh goat and cow milk the highest observed ^{131}I values were 0.25 and 0.21 BqL^{-1} respectively. Not any ^{131}I observed in soil samples collected from inside greenhouses.

The ^{137}Cs in grass samples ranged between 41 and 89 mBq kg^{-1} and in soil between 9.62-85.17 Bq kg^{-1}. The ratio ^{134}Cs/^{137}Cs in all soil, grass and milk samples was much lower than 1 that was observed in atmospheric samples and this was an indication of a contribution of “older” ^{137}Cs in the grass, soil and milk samples, and more specifically from Chernobyl accident and not the Fukushima accident.

The radioactive plume that reached European countries had only small amounts of radioactive isotopes, which were deposited by wet and dry deposition and have contaminated the land, and as a consequence the whole food chain. Based on the observed values the total doses have been calculated. The effective dose is calculated by the following relation:

$$E = E_{est} + \sum_j h(g)_{j,ing} J_{j,ing} + \sum_j h(g)_{j,inh} J_{j,inh} < 1mSv\ a^{-1}$$

where E_{est} is the effective dose for exposure; $J_{j,ing}$ and $J_{j,inh}$ are the intake activity (Bq) by ingestion and by inhalation of radionuclide j respectively; $h(g)_{j,\ ing}$, $h(g)_{j,\ inh}$ (Sv Bq^{-1}) are the coefficients of committed dose for unit of intake by ingestion and/or by inhalation for the population of age group g, due to radionuclide j.

The effective doses were calculated by using the maximum concentrations observed in the environment for ^{131}I, ^{134}Cs and ^{137}Cs in a hypothetical extrem scenario of 1 year constant radionuclide intakes and for two different population ages (a) ages younger than 1 year old and (b) ages older than 17 year old [2]. Calculations were done using the annual individual usage factors for inhalation, food consumption and external exposure reported in NCRP-123 [3]. Even in this scenario, the obtained effective doses were at least one order of magnitude less than 1mSv a^{-1}.

TABLE 1. EFFECTIVE DOSES IN MILAN, ITALY AFTER FUKUSHIMA ACCIDENT

Different Pathways	Age <1	Age >17a
Water	0.15 mSv a^{-1}	25.9 μSv a^{-1}
Cow Milk	18.7 μSv a^{-1}	5.0 μSv a^{-1}
Goat Milk	15.9 μSv a^{-1}	4.4 μSv a^{-1}
Air	0.30 μSv a^{-1}	35.5 10^{-3} μSv a^{-1}

CONCLUSIONS

The estimated effective doses for Italian population related to the contribution of Fukushima fallout due to different pathways were at least one order of magnitude less of the limit of 1mSv a^{-1}, even with the most conservative assumptions. Radioisotopes of iodine and cesium were found above their detection limits in all environmental samples, but far below of levels of concern.

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160-INSIGHT INTO URANIUM SORPTION SPECIATION ON GRANITE: EVIDENCES FROM TRLFS, EXAFS AND EPMA.

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Hexavalent uranium is a prominent radioactive contaminant in both sediments and aquifers around nuclear activity sites, and poses a potential health and environmental risk to the biosphere. Understanding on the detailed interaction at the solid-liquid interface between uranium and granite is crucial for the safety assessment of high level radioactive waste geological repository in granitic terrain, as well as prediction of uranium's geochemical fate in the environment. However, the interaction between uranium and granite is complicated by the ubiquitous ligands in natural media such as phosphate, humic substances et al, thus the detailed insight into U-granite interface is critical in governing the subsurface mobility of uranium in disposal environment. Full understand on complicated interaction mechanism between uranium and ligands at the granite-water interface requires full identification of surface species with the aid of sensitive spectroscopic techniques.

In this work, the adsorption of uranium on granite in both absence and presence of phosphate was investigated by a combination of batch measurements and spectroscopic techniques, including cryogenic time-resolved laser induced fluorescence spectroscopy (TRLFS) and extended x-ray absorption fine structure (EXAFS). Results showed that phosphate is beneficial for uranium immobilization, the spectroscopic confirmation revealed that multiple surface species including inner-sphere complexes and surface precipitates were formed on granite surface with their abundances varying as a function of acidity. The EPMA results showed that uranium mainly located on mica mineral, thus the detailed interaction between uranium and mica was further investigated. The results showed that uranium sorption on phlogopite mica was strongly dependent on pH while minimally affected by the ionic strength, multiple inner-sphere surface species (including $\equiv SOUO_2^+$, $\equiv SO(UO_2)_2(OH)_2CO_3$- and $\equiv SOUO_2(CO_3)_x^{1-2x}$) were formed with their abundance varying as a function of pH, and a portion of uranium precipitated as uranyl oxyhydroxides at pH > 9. The presence of HA made significant difference on uranium sorption behavior as well as surface species. The finding in this work is helpful for understanding on the geochemical fate of uranium in granitic environment as well as setting a reliable reference for surface complex models.

161-OPTIMIZATION OF PEDIATRIC CT PROCEDURES: COMPARATIVE RESULTS

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INTRODUCTION

In recent years, dose estimates in pediatric CT have been an area of great interest because of increased awareness of the radiation risks associated with the exposure from this modality in childhood [1]. Optimizing dose in CT scans is imperative, especially for pediatric patients, who have longer life expectancy [2]. Additionally, the optimization of the protection intending to provide the highest level of safety that can reasonably be achieved is one of the five Fundamental Safety Principles described on the IAEA Basic Safety Standards [3]. The work aims to use an experimental methodology for estimating organ dose due to paediatric CT scans protocols and to compare it with optimized protocols, adopted after optimization process in local facility. In addition, $CTDI_{vol}$ and DLP of pediatric brain and thorax CT scans were collected and then compared to DRLs proposed by three previous works [1,4,5], to monitor protocols optimization

MATERIALS AND METHODS

Organ dose estimates were performed with OSL dosimeters (Landauer Luxel Al_2O_3:C), positioned inside a CIRS ATOM model 705 pediatric phantom, equivalent to a five-year-old patient. Four brain CT protocols and thorax CT protocols were evaluated. CT scans were performed on a Philips Brilliance 64 CT scanner and the techniques for the evaluated CT protocols are shown in Table 1.

TABLE 1. ACQUISITION PARAMETERS OF THE EVALUATED CT PROTOCOLS

	Non-Optimized Protocols			Optimized Protocols		
	Helical Brain	Axial Brain	Thorax	Helical Brain	Axial Brain	Thorax
Tube Voltage (kV)	120	120	120	100	100	100
Charge (mAs)	299	200	55	149	110	110
$CTDI_{vol}$ (mGy)	38.7	31.7	7.0[2]	11.8	10.6	4.3
DLP (mGy.cm)	1100.8	730.1	202.5	334.6	244.9	126.1

RESULTS

Comparative results of organ dose from the evaluated CT scans with percentage differences between non-optimized and optimized protocols are shown in Table 2 for thorax exams and Table 3 for brain exams.

[2] $CTDI_{vol}$ displayed for a 16-cm phantom.

TABLE 2. COMPARATIVE RESULTS FOR THORAX PROTOCOLS

	Non-Optimized Protocol	Optimized Protocol	
Organ	Dose (mGy)	Dose (mGy)	Difference (%)
Lungs	6.9 ± 0.4	8.2 ± 0.5	+18
Thyroid	7.8 ± 0.2	9.6 ± 0.4	+24

TABLE 3. COMPARATIVE RESULTS FOR BRAIN PROTOCOLS

	Helical Protocols			Axial Protocols		
	Non-Optimized	Optimized		Non-Optimized	Optimized	
Organ	Dose (mGy)	Dose (mGy)	Difference (%)	Dose (mGy)	Dose (mGy)	Difference (%)
Eye Lenses	32.5 ± 1.4	9.2 ± 0.8	-72	24.9 ± 1.5	7.5 ± 0.9	-70
Thyroid	4.4 ± 1.4	13.1 ± 0.12	-70	31.7 ± 1.0	12.6 ± 0.6	-60

CONCLUSIONS

It was found that brain protocol optimization for helical exams reduced organ dose up to 72% and for axial exams reduced organ dose up to 70%. Regarding the thorax protocols, a 20% dose increase was verified with the optimized protocol. The results found are in good agreement with results available in the literature.

ACKNOWLEDGEMENTS

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162-RELATION AMONG INDOOR RADON CONCENTRATION, CLIMATE, AND SMOKING HABIT TO LUNG CANCER

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Radon was initially considered to be the direct cause of the lung cancer in the miners. Later, it was found that it was the decay products of radon, and not radon, that delivered the pertinent dose to lung cells. The fact that indoor radon is higher than outdoor radon has increased people concern on radon, since most people time was spent indoor (about 80% as researched by EAP). Epidemiologic studies have been conducted to assess the general population's risk of lung cancer associated with indoor radon and complementary animal and laboratory studies have been carried out to address uncertainties in assessment of the risks associated with indoor radon. Study on geogenic parameters, moisture, humidity, grain size, etc. to indoor radon level have been done by many researchers. However, in a bigger extent, climate and smoking habit were still not studied much. This study will try to see the relation among climate, indoor radon, and smoking habit to lung cancer. The result showed that cold climate has higher rate of invasive cancer incidence, which agreed previous research done by Shah as the first study on the field. In addition, some direct and indirect relation among the parameters seems to be found.

RESULT SUMMARY

Correlation between radon isotopes and their progeny with an increase of lung cancer risk has been studied by many researchers [1]. The International Commission on Radiological Protection had issued recommendations on ionizing radiation from man-made source of 20 $mSv.y^{-1}$ for occupational exposure and 1 $mSv.y^{-1}$ for common people, excluding their radiation exposure applied for medical patient [2].

Shah et al., did the first study on the connection between climate and lung cancer on 2019 and claimed that cold climate has higher rate of invasive cancer incidence [3].This study will attempt to find out the relation among climate, indoor radon concentration, and smoking habit to lung cancer.

This study will try to see and find relation between indoor radon concentration and some other parameters including climate, population, and smoking prevalence to lung cancer. To conduct this study, secondary data from trusted sources were extracted, collected, and analysed. Seventeen countries which represent all climate zones including tropical, subtropical, temperate, and cold climate zone were chosen.

Relation of climate and indoor radon concentration

Based on mean value of each climate groups, indoor radon concentration seems to increase from tropical, subtropical, temperate, to cold climate. The indoor radon concentration average was found to be 60.93, 65.40, 65.87, and 74.32 $Bq.m^{-3}$, respectively. All indoor concentration of country groups and its average value is presented in the following block diagram, shown in Fig 1.

The underlying reason for the use of average value in each climate zone is that there is a gradual change in climate conditions at each latitude shift. Countries located at the edge of every climate zone will tend to have a bit similar condition.

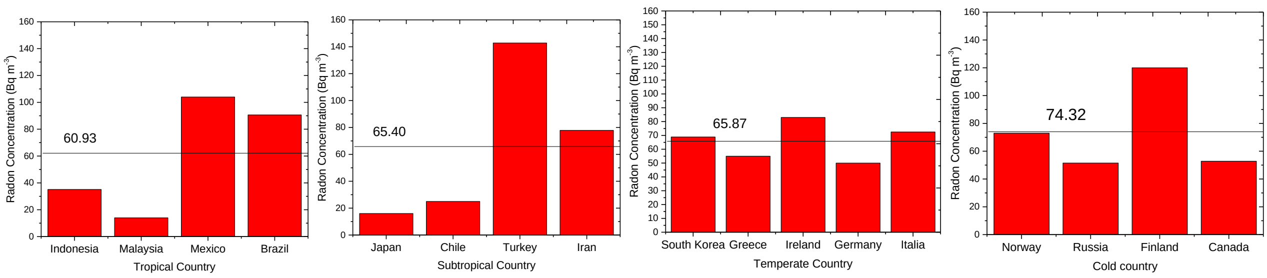

FIG1 indoor radon concentration of countries in climate groups (note: mean concentration value presented in straight horizontal line).

Public behaviour may take a significant role in the higher indoor radon concentration in cold climate countries. Since cold temperature tend to occur longer throughout the year in cold climate, people will strongly have a tendency to stay indoors.

Reviewing indoor radon and lung cancer death data, it seems that indoor radon value is similar to lung cancer number in most of the countries. This can be estimated that radon concentration may have an effect on the lung cancer risk. The different trend in few countries, however, indicates that radon may give insignificant influence on the risk. In other words, radon is not the main contributor to induce lung cancer.

Relation of lung cancer and climate

Table 1 present lung cancer data and average lung cancer fraction in country climate groups, respectively. If we connect the lung cancer in a population and show them as a fraction of population, lung cancer fraction seems to increase from tropical to cold climate countries. From Table 1, we can see the average lung cancer fraction of 0.01, 0.04, 0.06, and 0.06 from tropical, subtropical, temperate, and cold climate zone, respectively. The lung cancer fraction seems to increase gradually from tropical to temperate zone and fall into saturation from temperate to cold climate zone.

This finding agrees Shah et al. result which first conducted this type of study in 2019. The same typical results of these studies have strengthened our confidence on the hypothesis. However, more researches should be done to proof the hypothesis and to give more benefit in the future.

TABLE 1. LUNG CANCER FRACTION FOR EVERY CLIMATE ZONE.

Country	Lung cancer fraction* (% of population)	Lung cancer (mean, %)
Indonesia; Malaysia; Mexico; Brazil	0.011; 0.014; 0.003; 0.016	0.01
Japan; Chile; Turkey; Iran	0.094; 0.021; 0.042; 0.008	0.04
South Korea; Greece; Ireland; Germany; Italia	0.056; 0.090; 0.061; 0.081; 0.068	0.06
Norway; Russia; Finland; Canada	0.062; 0.043; 0.050; 0.068	0.06

***source**: IARC [154]

Environmental parameters such as rainfall, temperature, and water content may not be a direct cancer contributor. From tropical to cold climate, precipitation is decreasing while lung cancer fraction is increasing. There may be an indirect relation between them since precipitation has an effect to radon level.

Relation of lung cancer and smoking

About 80-90% lung cancer mortalities were predicted to be caused by smoking [88]. In this study, lung cancer number was compared to smoker number from all climate zone to see whether there is relation between both. Comparation between both parameters showed an identical trend between smoker number and lung cancer number. It can be concluded that smoking is definitely one of the most responsible parameters to induce lung cancer.

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163-INVESTIGATION OF OCCUPATIONAL RADIATION EXPOSURE FROM C-ARM FLUOROSCOPY GUIDED PROCEDURES

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The use of fluoroscopy as a medical imaging technique is one of the main sources of occupational radiation exposure. Often fluoroscopic procedures are lengthy and staff dose is considered to be high due to prolonged exposure time and the fact that operating staffs are often next to the patient while x-ray procedures is undergoing [5,6]. The study's aim was to measure and analyze the scattered dose rate around a fluoroscopy room, to determine staff exposure during C-arm fluoroscopy guided procedure and to compare the dose measurements obtained with occupational doses in the NRPA occupational monitoring data base.

The scattered radiation dose was measured at 12 positions across the operating theatre using a radiation survey meter (FIG. 1). A Perspex phantom was placed on a C-arm x-ray machine and thereby irradiated to stimulate a patient undergoing fluoroscopy guided examination. Two experiments were conducted with the C-arm tube under couch in both cases. However, an anteroposterior (AP) and a cranial (CRA 15°) projection was performed, respectively. Staff radiation dose measurement were conducted in the same operating theatre where three medical staff (Doctor, nurse, and radiographer) were monitored using thermoluminescent dosimeters (TLDs) over a period of three months. Both staffs wore TLDs at the neck, chest, and waist over the lead apron.

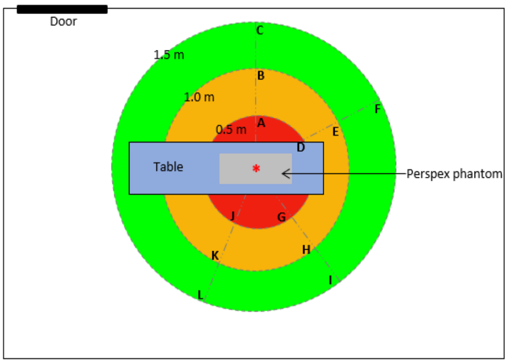

FIG. 1. The points marked A to L represents where scatter radiation doses at 0.5, 1.0 and 1.5 m from the central point and 1.0 m above the floor were measured. The center of the Perspex phantom is marked with a red asterisk.

The scatter radiation dose decreases with distance from the source (FIG. 2 and FIG. 3). This is very important on reducing scatter radiation to staff during fluoroscopic procedures along with the use of proper shielding and less exposure time. The doctor received the highest annual effective dose of 1.68 mSv (deep dose) and 1.835 mSv (skin dose) due to proximity to the patient when conducting procedures (TABLE 1) and also possible due to the orientation of the tube which is mostly under couch [3]. The radiation dose to the doctor and the nurse obtained from the study are comparable those in the NRPA occupational monitoring database and lower than the annual occupational radiation exposure limit of 20 mSv [4].

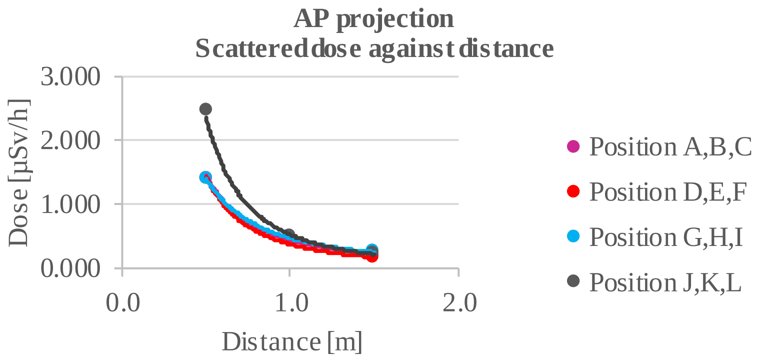

FIG. 2. Scatter dose for an anteroposterior (AP) projection.

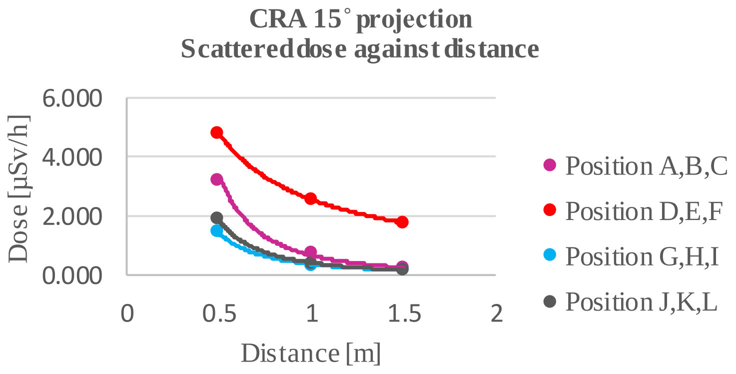

FIG. 3. Scattered dose for a CRA projection.

TABLE 1. THE DOSE RECEIVED BY STAFF PER YEAR

	Body part	Doctor		Radiographer		Nurse	
		deep dose	skin dose	deep dose	skin dose	deep dose	skin dose
Effective dose/year (mSv)	Chest	1.688	1.896	1.172	1.288	1.064	1.244
	Waist	1.804	1.972	1.080	1.148	0.912	1.040
	Neck	1.548	1.636	0.964	1.124	1.136	1.236
Average dose/year (mSv)		1.680	1.835	1.072	1.187	1.037	1.173
NRPA Total dose/year for 2017 (mSv)		2.100	1.880	-	-	0.940	0.860

ACKNOWLEDGEMENTS

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164-THE AARST CONSORTIUM STANDARD ON PROTOCOL FOR THE COLLECTION, TRANSFER AND MEASUREMENT OF RADON IN WATER— AN UPDATE

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The American Association of Radon Scientists and Technologists (AARST) Consortium on National Radon Standards is a non-profit organization owned by AARST. Accredited by the American National Standards Institute (ANSI), the AARST consortium has been serving as an ANSI-approved developer of standards on various aspects of radon such as sampling, measurement, mitigation, and others. The standards are developed and maintained following ANSI's established consensus process to ensure openness, balance and due process through 1) Open Participation, 2) Stakeholder Balance, 3) Standard Development, 4) Consensus Vote, 5) Public Comment and Appeals, 6) Standards Publication, and 7) Standards Maintenance. When testing for radon in water, laboratories in North America use different methods of sample collection, sample preparations and analysis, each of which has some advantages and limitations. Given these discrepancies, in 2017 the AARST Consortium on National Radon Standards tasked a committee for developing a standard on the "Protocol for the Collection, Transfer and Measurement of Radon in Water." From fall 2017 to spring 2020, the committee drafted, reviewed, received and addressed public comments and appeals, and almost finalized the standard. The developed standard contains procedures, minimum requirements and guidance for measuring radon in water that enters a building through groundwater supplies for determining if mitigation is necessary to protect current and future occupants of family dwellings and commercial buildings. This standard of practice specifies the minimum requirements and procedures for the collection and transport of water samples, as well as protocols for the quantitative transfer of the sample to a measurement device to determine radon concentrations in water. This standard includes the United States Environmental Protection Agency-accepted analytical methodologies, liquid scintillation and alpha scintillation cells, along with essential information on use of electret ion chambers and acknowledgement of continuous radon monitors. A structure for defining a national reference for calibration and quality control, in lieu of a federally reference, is provided, as well as recommended action levels and acknowledgement of current state-recommended action levels. The presentation will bring to the international audience the process followed for developing this standard and its salient features along with some original research results utilized by the committee during the process. It will also compare and contrast the AARST consortium standard with other international standards like ISO13164-1,2,3, 4 and others (if any).

165-REASONABLENESS AND TOLERABILITY IN THE SYSTEM OF RADIOLOGICAL PROTECTION: ICRP ON-GOING REFLECTIONS

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The model of reasonableness and tolerability of radiological risk is a conceptual framework for the implementation of the ICRP principles of optimisation (guided by constraints and reference levels) and limitation, based mainly on the level of exposure, and closely related to the level of risk. Discussions of reasonableness and tolerability have been part of ICRP publications for many years, including the introduction of a model of risk tolerability in Publication 60 [1]. More recently, Publication 101 [2] developed the approach to address the implementation of the optimisation process and the way to elucidate what is reasonably achievable. Further considerations have been addressed while examining the ethic, identifying four core values underpinning the system of radiological protection. In 2019, ICRP has set up a dedicated task group to review the historical and current perspectives on reasonableness and tolerability in order to consolidate and clarify Publication 103 [3], and to prepare the considerations and basis needed for development of future recommendations.

This model provides guidance for the implementation of the principles in planned, emergency and existing exposure situations. Reasonableness plays a key role in the decisions regarding appropriate levels of protection and is the core of the implementation of the optimisation principle taking into account societal and economic aspects, and also protection of the natural environment, as well as organisms that are part of, and adds value to, people's environment and livelihood. Reasonableness applies to any exposure situation as far as an optimisation process is implemented, although what may be considered reasonable will depend on the specifics of the situation and involvement of stakeholders in an inclusive process.

The ICRP on-going reflections on reasonableness and tolerability in the system of radiological protection have already identified a wide number of questions to be addressed, including, for example: What is the link between tolerable and reasonable? Tolerable and reasonable for whom? What are the considerations and criteria on which the concepts of tolerability and reasonableness are based? What is the trade-off between level of exposure and benefit? How do we account for sensitive or vulnerable populations? How are interests reflected and integrated for human and nonhuman exposure? How do we best support reasonable value for society? What are some strategies to assist in balancing competing values in determining what is tolerable and/or reasonable? Are the same criteria applicable for tolerability in all exposure situations? How do different paradigms used for non-radiological risks inform and contribute to radiological protection?

For addressing these questions, the model of reasonableness and tolerability of radiological risk is revisited with the following objectives:

- Better articulate the link between tolerability and reasonableness in the process of implementation the radiological protection system, with clarification on the criteria to be considered for defining "where we don't want to go above" and which process could be put in place for evaluating "what is reasonable".

- Refine the criteria to be considered and their link with dose limits and reference levels, relying on the radiological detriment as benchmark for tolerability and reasonableness as well as using risk comparison but without limiting to numerical criteria.
- Emphasize the importance of the application of the model for the different exposure situations, including the deliberative process with the stakeholders for the implementation of the optimisation principle, referring to good judgement, fairness, practicability and moderateness.

This paper will present the current reflections of the ICRP Task Group with the aim to get feedback from the participants and national and international organisations on their views on these issues.

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169-GRADED APPROACH IN THE APPLICATION OF SAFETY REQUIREMENTS FOR ON-SITE LOADING AND ACCEPTANCE TESTING OF THE LEKSELL GAMMA KNIFE

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The gamma knife was invented in 1968 by Prof. Dr. Lars Leksell, and Prof. Dr. Börje Larsson in Stockholm, Sweden. The device uses 192 highly focused beams Co-60 which provides a single, high-dose radiation with pinpoint accuracy to the brain tumors with minimal effect to surrounding tissues that known as stereotactic radiosurgery technique. The device is highly benefiting in treating patients with multiple brain tumors. In general, Malaysia require 13 Gamma Knife machines nationwide, to meet ratio of 1 gamma knife to 2.4 million of its population. Gamma knife installed in Pantai Hospital Kuala Lumpur are among earlier units that the Co-60 sources were loaded on-site. The objective of this studies is to identify the compliancy of the safety requirements during site preparation, delivery and loading of the unit through graded approach. The same approach also has been used during the installation and usage of gamma knife to minimize the disruption of hospital activities while maintaining acceptable levels of radiation exposure. The studies were assessed through analysis and grading activities as an important safety measure by including elements such as regulatory supervision, management and verification of safety, site evaluation, design and operation. The implementation of graded approach is able to identify several key areas of the assessment by providing overview on the highest contribution to doses and risk. However, without graded approach there will be wasting effort at irrelevant areas and overlooking critical exposure pathways and scenarios. In conclusion, Pantai Hospital Kuala Lumpur are complied with all relevant safety requirements during on-site loading and acceptance of the Leksell gamma knife by applying graded approach and stipulated legal provisions.

OBJECTIVE

The objective of this studies is to identify the compliancy of the safety requirements during site preparation, delivery and loading of the unit through graded approach. The same approach also has been used during the installation and usage of gamma knife to minimize the disruption of hospital activities while maintaining acceptable levels of radiation exposure.

MATERIALS AND METHODS

The studies were assessed through analysis and grading activities as an important safety measure by including elements such as regulatory supervision, management and verification of safety, site evaluation, design and operation [1].

RESULTS

Based on the studies conducted, graded approach is applied for the several safety assessment elements such as regulatory supervision, management and verification of safety together with site evaluation, design and operation while still commensurate with complexity and hazard potential of the facility and work to be performed[3]. Below are result based on graded approach in safety assessments as per Table 1.

TABLE 1. ANALYSIS AND GRADING OF SAFETY ACTIVITIES

NO.	GRADING	ANALYSIS	DESCRIPTION
1.	Regulatory supervision	i. Enforce legal infrastructure. ii. The regulatory body that enforces the law. iii. Establishment of several licensing process and procedures[6]. iv. Execution of programme for inspection and enforcement.	1. Atomic Energy Licensing Act 1984 and Strategic Trade Act 2010. 2. Ministry of Health and Atomic Energy Licensing Board. 3. Permission to install and permission to use. 4. Inspection during testing and commissioning gamma knife.
2.	Management and verification of safety	i. Establish and implement related instructions and procedures. ii. Comprehensive testing, surveillance, maintenance and inspection activities.	1. Standard operating procedures in handling/use of gamma knife[5]. 2. RPP involves testing, surveillance, maintenance and radiation surveys.
3.	Site evaluation	i. Potential external events due to natural origin, such as seismic[1]. ii. Population density and distribution[1].	1. Malaysia has minimal seismic impact with reference to Malaysian Meteorological Department. 2. Hospital are located at high density location; centre of Kuala Lumpur City.
4.	Design	i. Execution of safety functions. ii. Design for emergency planning.	1. Basic safety functions; system shut down and confining radioactive material[4]. 2. Inclusive alarm systems, communication and designated assembly areas.
5.	Operation	i. Radiation protection programme (RPP)[7]. ii. Training and qualification programmes[2].	1. Documented and implementation RPP for facility. 2. Organizing annual related training/course.

DISCUSSION

The implementation of graded approach is able to identify several key areas of the assessment by providing overview on the highest contribution to doses and risk[3]. However, without graded approach there will be wasting effort at irrelevant areas and overlooking critical exposure pathways and scenarios.

CONCLUSION

In conclusion, Pantai Hospital Kuala Lumpur are complied with all relevant safety requirements during on-site loading and acceptance of the Leksell gamma knife by applying graded approach and stipulated legal provisions.

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170-MEASURING OCCUPATIONAL RADIATION DOSE TO THE RADIOGRAPHERS PERFORMING CARDIAC CATHETERIZATION LABORATORY PROCEDURES AT SRI JAYAWARDENEPURA GENERAL HOSPITAL, SRI LANKA

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BACKGROUND

In the past three decades, the fluoroscopy guided procedures dramatically increased due to the various applications in treating injuries and growing range of diseases. The fluoroscopy guided procedures are the real time radiological images (fluoroscopy), used for diagnostic and therapeutic purposes. In present, there are different types of imaging methods available for the evaluation of cardiac anatomy and pathology such as Plain radiography (Chest X ray), Computed Tomography (CT), Magnetic Resonance Imaging (MRI), Echocardiography, Radionuclide imaging and cardiac catheterization with the use of C arm imaging equipment [1]. Among them, the cardiac catheterization enables us to perform both diagnostic and therapeutic procedures where the cardiac chambers, the anatomy and pathology of coronary vessels are accessed using a catheter inserted through the blood vessels. The cardiac catheterization plays a major role in diagnostic and therapeutic purposes. Meantime, concern has been expressed regarding the radiation exposure to medical staff and patient [2-5]. The aim of this study is to measure the occupational dose to the radiographers who work at Cardiac catheterization laboratory-B, Sri Jayewardenepura General Hospital, Sri Lanka.

METHODS

This study was conducted at Cardiac Catheterization Laboratory (Cath lab) – B, Sri Jayewardenepura General Hospital, Sri Lanka. The data was collected from all three radiographers who work at cardiac catheterization laboratory B during three months from July 24, 2018 to October 24, 2018. 189 procedures were selected for this study and among them 123 and 66 procedures were Coronary Angiogram (CA) and Percutaneous Coronary Intervention (PCI) respectively. Two personal dosimeters were placed, one dosimeter attached on the anterior aspect of the chest inside the Lead apron, and the other dosimeter attached on the neck outside to the thyroid protective collar. The measured readings were used to determine the effective dose using the formula which recommended by National council on radiation protection [6].

RESULTS

Table 1 shows the details of mean effective dose per procedure for the three radiographers and Table 2 shows the mean procedure per month for the three radiographers.

TABLE 1. MEAN EFFECTIVE DOSE PER PROCEDURE

Procedure	Radiographer 1	Radiographer 2	Radiographer 3
CAG(D_{CAG})(μSv)	0.057 ± 0.006	0.053 ± 0.006	0.061 ± 0.008
PCI (D_{PCI}) (μSv)	0.216 ± 0.026	0.113 ± 0.017	0.217 ± 0.023

TABLE 2. MEAN PROCEDURES PER MONTH

Procedure	Radiographer 1	Radiographer 2	Radiographer 3
CAG (N_{CAG})	41/3	44/3	38/3
PCI (N_{PCI})	15/3	25/3	26/3

CONCLUSIONS

The measured annual effective doses for three radiographers were (0.0223 ± 0.0010), (0.0206 ± 0.0020), (0.0318 ± 0.0027) mSv per year respectively. Effective doses to the radiographer depend on the procedure type either CAG or PCI. The occupational radiation dose to radiographers in the studied place is well below than the IRCP recommended value of 20 mSv.

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171-OPTIMIZATION OF DOSE TO SKIN OF FINGERS FOR NUCLEAR MEDICINE WORKERS

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Monitoring of occupationally exposed personnel gives a quantitative perspective on achieving individual and workplace safety. The goal is optimization of the radiation doses and adopting necessary measures to ensure that it is ALARA (as low as reasonably achievable) for radiation workers. Effective doses estimated from Hp(10) whole body dosimeters give an underestimation of the total radiation dose absorbed [1] in anisotropic irradiation cases such as in nuclear medicine where the hands receive the highest dose due to direct contact to unshielded syringes and vials which contain radiopharmaceuticals. In this study, finger thermoluminescent dosimeters (TLD) were used to determine equivalent dose to the fingers for two workers who prepare and dispense (WP1 and WP2) and two workers who administer (WA1 and WA2) radiopharmaceuticals at our local nuclear medicine centre. The radiopharmaceutical in use was Tc-99m. Optically stimulated dosimeters (OSLD) were also used to determine the dose to both wrists for the same workers. Fig.1 shows how the dosimeters were worn on the hand, whilst Fig.2 and Fig. 3 show a comparison of the results of measurement for workers performing preparation duties and those performing administration duties respectively.

FIG. 1. Dosimeter distribution on the hand

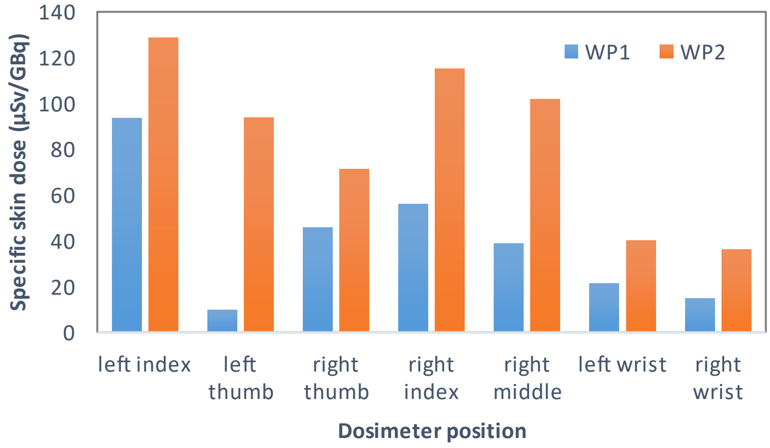

FIG. 2. Comparison of specific skin dose between the two workers responsible for preparation of radiopharmaceuticals

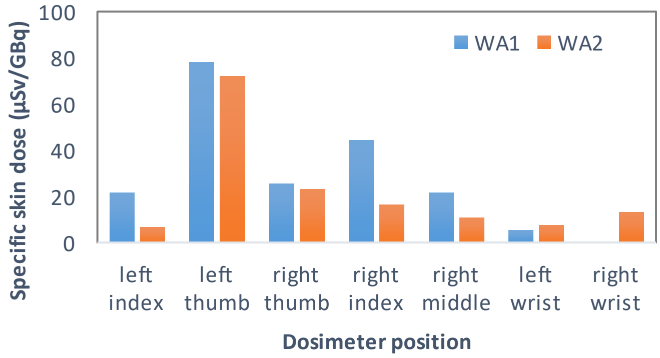

FIG. 3. Comparison of specific skin dose between two workers responsible for administration

For our local setting, the results show that preparation duties lead to more radiation exposure to the skin as compared to administration duties and consistent with literature [2]. Table 1 showing the range of personal dose equivalent between the two groups of workers as well as the mean dose.

TABLE 1. PERSONAL DOSE EQUIVALENT RANGE AND MEAN

Worker duties	Dose equivalent range (µSv/GBq)		Mean dose equivalent ±SD (µSv/GBq)	
	Fingers	Wrist	Fingers	Wrist
Preparation	9.84 – 128.62	14.70 – 40.07	75.47 ±37.54	28.07 ± 12.00
Administration	6.58 – 77.98	0 – 13.16	31.97 ± 24.81	6.51 ± 5.44

Further, estimation of dose to the fingertips was done using ORAMED dose estimation tool [1]. This is because dose distribution across the hand is not uniform [3]. The tool gave a factor of 2.5 increase in dose estimates at tip of index finger as compared to that measured at base of index finger, for workers involved in preparation of radiopharmaceuticals.

We can conclude that:

a) there is a possibility of dose reduction to extremities for workers especially when a shielded syringe is used or proper procedures and radiation protection techniques are followed for radiation dose optimization. [2]
b) the index finger of the non-dominant hand is best finger to wear finger dosimeter for preparation duties.
c) the thumb of the non-dominant hand is the best finger to wear finger dosimeter for administration duties.

d) Estimates of annual dose for each worker indicates that the effective skin dose limit of 500 mSv [4] is not exceeded.

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172-DEVELOPING RADIATION SAFETY INFRASTRUCTURE IN NEPAL

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Nepal, a landlocked, developing country situated in South Asia with 26.6 million people [1]. Thanks to the rapid and miraculous developments in the field of science and technology, human life has become much more comfortable and easier than in the past. Similarly, modern technology has made the use of radiation and radioactive materials possible in ameliorating great many ailments considerably limiting at the same time it’s inevitable risks. Radiation and radioactive substances have many beneficial applications, but it's even as important that the radiation risks to workers, general public and therefore the environment that arise from these applications are properly assessed and controlled.

The paper seeks to evaluate the present status, activities and improvement of the radiation protection infrastructure in Nepal. In Nepal, radiation sources are mainly used in the medical field since 1923. In 1976, radiation therapy services were started by using radioactive materials (Radium) [2]. Eye popping development has taken place especially in the field of diagnostic radiology, radiotherapy and nuclear medicine where the use of ionizing radiation is maximum [3]. In fact, it is estimated that around ninety-five percent (95%) of radioactive sources are used in the medical sector. In addition to the medical field, the use of radiation and radioactive materials is widely used in agriculture, mining, energy, industry, archeology and other fields, but Nepal has not been able to get much benefit from it in these sectors.

In 2008, Nepal became a member country of the International Atomic Energy Agency (IAEA) with the main objective of developing safe and peaceful use of nuclear technology. Since obtaining IAEA membership, Nepal has made several strides in the implementation and development of an infrastructure for radiation protection. The most important “Nuclear & Radioactive Materials Protection and Security Act Bill 2019”, has already been introduced in the Parliament in order to protect the lives of the common people and protect the environment from the adverse effects of ionizing radiation, safe and peaceful use of nuclear technology and ionizing radiation, security of radioactive sources, protection from unauthorized use of nuclear materials, radioactive materials and plants [4]. The main accomplishments include the approval of Nuclear & Radioactive Materials Protection and Security Bill form lower House and Upper house of the parliament of Nepal and it is at the final stage of endorsement.

Nepal has yet to constitute rules and regulation as well as regulatory body in the field of radiation safety. But, Ministry of Education, Science & Technology (MoEST), line ministry to IAEA, has issued “Nuclear Materials Regulatory Directive on 2015” to manage radiation sources in the country. Under this Directive there is one recommendation committee to recommend MoEST to manage all nuclear related activities in the country [4]. This happens to be an only valid legal document on regulation of radiation sources in Nepal till date. Committee usually follows IAEA BSS [5]. Also, at the same time, the finalized draft for minimum standards required for operating Diagnostic Radiology and Nuclear Medicine facility has already been completed through MoEST. Similarly, committee constituted for drafting of standards for transportation of radiation materials and Physical Protection Security Management System of Radioactive materials is going on. In the meantime, Ministry of Health and Population (MoHP) has already started practicing a requirement of pre-approval before importing any kind of radiation emanating equipment in Nepal.

Since 2012, Nepal has been involved in various Technical Cooperation (TC) projects associated with the IAEA. The national project entitled “Developing and Establishing National Infrastructure for Radiation Safety (NEP: 9001)”, which main objective was to finalize nuclear law and to establish radiation regulatory system in Nepal. Under that project Nepal was able to finalized nuclear law, started personal radiation dosimetry system, received radiation measuring devices & quality control equipment which also include fellowships/training as well as expert mission mainly focus on radiation safety. There is also ongoing national project entitled “Establishing a National Regulatory Infrastructure for the proper use of ionizing radiation (NEP 9003)”. Under this project, IAEA has been providing radiation measuring instruments, fellowships and training to staffs on regulatory related matters. There is also one more national project entitled “Strengthening Radiation Safety Through Regulatory Infrastructure, Occupational and Medical Exposure Control (NEP 9005)” focused on radiation safety related matters. In the meantime, Nepal has been participating in various Regional Cooperative Agreement (RCA) projects on radiation protection. Different professionals working at different institutions have been trained through various IAEA/RCA projects focused on radiation safety. According to Rozental, the main reason for the deficiency of sources control and dose limitation are related to the lack of an appropriate legal and regulatory framework [6]. MoEST has completed radiation awareness program throughout the country since 2018. Under that, several radiation awareness workshops have been conducted to all seven province of Nepal. The main target participants were health workers mainly working in the field of ionizing radiation.

Radiation protection principles include the use of well- established justification, optimization and dose limitation standards, but also involves the development of competence through training and education. Therefore, the future of safely use of radiation in Nepal mainly depends on the infrastructure of a strong regulatory system and sustainable safety culture of radiation users. Despite all the impending challenges, the "Radioactive Materials Uses and Regulatory Act has materialized effective as of July, 2020. This has paved the way to the formation of a Regulatory Agency necessary for effective supervision and implementation of radioactive regulations for safety. The need for establishing a radiation protection culture on a national level, and improving radiation protection culture at an organizational level is therefore evident. But safety standards are effective only if they are properly applied in practice. Hence, regulating nuclear and radiation safety should be our priority and should be our national responsibility.

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173-DEVELOPMENT OF A METHODOLOGY BASED ON GRADED APPROACH TO DEFINE THE FREQUENCY OF INSPECTIONS AT RADIOTHERAPY FACILITIES

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Recently the IAEA fundamental objective and associated fundamental safety principles incorporate the requirement for a graded approach, particularly in respect to the assessment of safety and the assessment of radiation risks. The Brazilian National Nuclear Authority is responsible for the licensing and regulation of the radiotherapy facilities in Brazil. So far, the method applied for organizing the RF inspection program have been time elapsed since the last inspection. This method is difficult to be applied when there are low human resources, and in a very heterogeneous country, as Brazil. This work suggests a new methodology to be used by the Brazilian Nuclear Authority to organize the inspection program, using the graded approach criteria. This new methodology can optimize the number of inspections realizes each year, and focus specially on the facilities that present higher risks.

INTRODUCTION

The Brazilian National Nuclear Authority (CNEN) is the government institution responsible for licensing and regulation of all kinds of facilities that use radiation sources or equipment that produce ionizing radiation, except the ones for medical diagnosis and dental uses. The licensing process of Radiotherapy Facilities (RF) involves a complex number of activities performed today, by only 7 inspectors. The number of RF has been growing in the last decade and, today, there are 241 RF with 350 linear accelerators (LINACs) distributed, not equally, at 27 Brazilian states. According to Brazilian law, all RF must undergo through a licensing process that involves documentation analysis as well as *in loco* inspections. The inspections are divided into three groups: first-inspection (for all the new LINACs before they are used to treat patients), routine (according a program scheduled for the whole year) and reactive (if an abnormal occurrence warrants an immediate investigation).

The routine inspections are planned in January of each year, and all RF are supposed to undergo routine inspections every two years. Since CNEN have only 6 inspectors for 241 RF, it’s been very difficult to follow this procedure, and as a result, some facilities could be working without the routine inspections for more than two years. CNEN makes a constant effort to keep updated with international standards and national needs to strength the radiological protection status of the country.

The IAEA has recently published the TECDOC 1740 [1], related to the use of graded approach to the application of the management system requirements, and this document discusses some of the factors to be considered in the grading of an inspection program and the associated corrective actions. IAEA TECDOC 1740 emphasizes that the scope and frequency of inspection programs should be determined based on many different aspects. One of the most important considerations is to promote the potential safety, occupational health, environmental and security risk posed by the nuclear facility or activity - and during particular situations such as organizational changes or personnel turnover, and this could also be applied to a RF.

At GSR Part 1 (Rev. 1) [2] it is stated that the government shall establish laws and regulations to make provision for the inspection of facilities and activities, and for the enforcement of the regulations, in accordance with a graded approach. Also, that inspections of facilities and activities shall be balanced-with the radiation risks associated within the facility or activity, in accordance with a graded approach. And according GSR Part 1 (Rev. 1) [2] the regulatory body shall develop and implement an inspection program of facilities and activities, to guarantee compliance with regulatory requirements and with any conditions specified in the authorization. In this program, it shall specify the types of regulatory inspections, and shall stipulate the frequency of inspections and the areas and programs to be inspected, in accordance with a graded approach.

The main criteria used by CNEN for the RF inspection program have been time elapsed since the last inspection. Facilities that have not been inspected for a long time should have a higher priority. And in order to be consistent with IAEA recommendations, CNEN is studying how to implement the graded approach as part of the RF inspection program. The main goal of this study is to implement a new methodology, based on graded approach, to define the frequency of radiotherapy facilities inspections. This would save resources, optimize inspections, and in the future could be used in other regulatory areas controlled by CNEN.

METHODS

This study is divided into four main steps:

(1) Identify the main non-compliances historically observed during the RF inspections, according to CNEN regulatory guides [3,4].

(2) Apply the AAPM TG100 [5,6] risk analysis methods to define a risk priority number (RPN) for each non-compliance.

(3) Use the determined RPN at the inspection guide after each inspection, which would result in Quality Factor of each inspected RF.

(4) Create a criteria of frequency of inspections based on the Quality Factors of each RF, low quality factor would implies in more frequent inspections.

RESULTS AND DISCUSSION

The first difficulty found to organize the non-compliances was that although they were at the regulatory guides, each inspector used to write them using different words, and they were not standardized. This way, it was created a non-compliance spreadsheet, and it is now mandatory that all the inspectors use the same spreadsheet to write the non-compliances at all the regulatory documents.

The first step was performed, based on the last 2 years of performed inspections at RF, years 2018 and 2019, a total of 173 inspections. Authors will include in this list also the inspections done in 2015, 2016 and 2017, totalizing 415 inspections. The five most frequent non-compliances observed during the inspections, and its frequency are shown on table 1.

TABLE 1. THE MOST FREQUENT NON-COMPLIANCES OBSERVED DURING THE RF INSPECTIONS

Noncompliance	Frequency
The occupational health certificates of occupational exposed individuals are not up to date	70%
The Radiation Safety and Protection Plan doesn't contain all the mandatory information	55%
The RF doesn't perform risk analysis for radiation therapy quality management	48%
The constancy checks of ionization chamber are not being performed every three months, the recommended frequency.	37%

This study was programmed to be concluded in 2020. But due to the pandemic situation installed in the whole word, the deadlines had to be changed. The next step will be to apply the RPN concept to all identified non-compliances, assigning numerical values to three parameters: O (occurrence), S (severity), and D (lack of detectability). These three parameters will be multiplied together to generate a single quantitative metric called RPN, which will help the regulators to identify what features of the failure mode contribute most to the overall risk associated with it. The Occurrence was established, using the frequency of occurrence of each non-compliance. And authors will establish grades for Severity, which describes the severity of the effect on the final process outcome resulting from the failure mode if it is not detected or corrected, and also for D (lack of detectability), which describes the likelihood that the failure will not be detected in time to prevent an event.

Steps 3 and 4 will be applied in the next months, and this study will be ready by July 2021, to be applied in the inspection program starting in January 2022~~,~~ using then the developed grades approach.

CONCLUSION

At the end of the study, all RF will have a quality factor based on this developed methodology, and a criteria of frequency of inspections will be established according to the quality factor. RF with high quality factors will receive a more sparse routine inspection, and CNEN will intensify the frequency of inspections at RF with low quality factors. Using this method, based on graded approach, the Brazilian Authority expects to improve the quality of RF, being more present at the ones that have more difficulties to follow regulations.

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175-REGULATORY ENFORCEMENT AS A TOOL TO ENHANCE RADIATION SAFETY IN MEDICAL EXPOSURE

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INTRODUCTION

Medical imaging is used in the diagnosis and treatment of numerous medical conditions in both children and adults and several types or modalities of medical imaging procedures, each using different technologies and techniques are available [1]. Worldwide, an estimated 3.6 billion diagnostic medical examinations, such as X-rays, are performed every year and this number continues to grow as more people access medical care [2]. Today, electromagnetic radiation from medical procedures constitutes the single largest manmade means by which people encounter radiation exposure [3]. Protection against the medical use of radiation is therefore even more important than protection against any other source of radiation. Radiation protection and radiation safety are crucial in medical exposure in order to ensure safety of both the patient and the healthcare giver.

The IAEA requires governments to establish and sustain an effective legal and governmental framework for safety, including an independent regulatory body and places prime responsibility for safety on the upon the person or organization responsible for facilities and activities that give rise to radiation risks [4].

Radiation safety is based on a whole set of regulations and standards, design solutions, operating instructions, safety culture, training, being good leadership, commitment to safety and other factors, all of which contribute to the control of the occupational exposures and prevention of the release of radioactive materials into the environment [5,6]. Regulators also plays a crucial part in ensuring radiation safety and their role cannot be overlooked.

Weak radioactive source regulation and enforcement by the regulator may lead to radiation accidents, radiation injuries, contaminated land, water, air and property that can cause social disruption, financial costs, adverse political impact for the regulator and government and greater potential for illicit trafficking [7]. Cases have been recorded where weak enforcement and lack of regulatory oversight has led to severe accidents leading to injuries and death [8, 9, 10, 11, 12].

The presentation outlines the advantages and challenges of using regulatory enforcement as a tool to ensuring radiation safety in medical exposure and assesses its effectiveness in that regard.

LEGAL FRAMEWORK FOR RADIATION SAFETY IN ZIMBABWE

In Zimbabwe, the Radiation Protection Authority of Zimbabwe was established through the Radiation Protection Act [Chapter 15:15] of 2004 as the regulatory body responsible for protecting people and the environment from the harmful effects of radiation [13].

The regulator conducts audit inspections at facilities to verify compliance and to ensure there are no deviations or deficiencies from the set standards. The frequency of the inspections follow a graded approached based on the safety risks associated with the facilities.

The Radiation Protection Act empowers the regulatory body to take enforcement action in the event of non-compliance being identified or in case of a breach of the licensing conditions. To determine the enforcement action to be taken, the regulator considers the safety and security risk posed by the violation; the complexity of the corrective action that is needed; whether it is a repeat violation; whether there has been wilful violation of the limits and conditions specified in the authorization or in regulations; or the licensee's compliance history among other things [14].

According to RPAZ Enforcement Policy, enforcement actions should properly reflect the safety or security significance of the violations and should be carried out on a risk graded approach. As per the Act and the regulations, enforcement actions that RPAZ is empowered to take include Verbal notification; Written notices or warnings; Orders; Imposition of additional regulatory requirements and conditions; Seizure of substances or equipment as stated in the Act; Prosecution and Closure of facilities [13, 14].

BENEFITS THAT WERE OBSERVED IN ZIMBABWE AS A RESULT OF ENFORCEMENT

In Zimbabwe, the regulatory body started operations in 2010 when facilities were already in existence and were used to doing things in their own way without anybody checking on them. During the period 2010 to 2015 when the regulatory body was still in its infancy, high numbers of violations were recorded and the non-compliances ranged from administrative issues of not displaying warning notices, no record keeping system in place, not paying license fees to violations with significant potential on safety such as inadequately trained personnel, no workplace monitoring, X-ray machines with no collimation and faulty machines with no calibrations for long periods of time emitting high radiation doses. The following are some of the notable benefits that the regulatory body realised as a result of applying constant enforcement actions:

(f) The regulatory body managed to correct violations by applying graded approach enforcements and through trainings and stakeholder engagements, educating operators on the importance of compliance.

(g) Enforcements encouraged self-regulation by licensees and compliance rates significantly increased as can be seen below in Fig. 1.

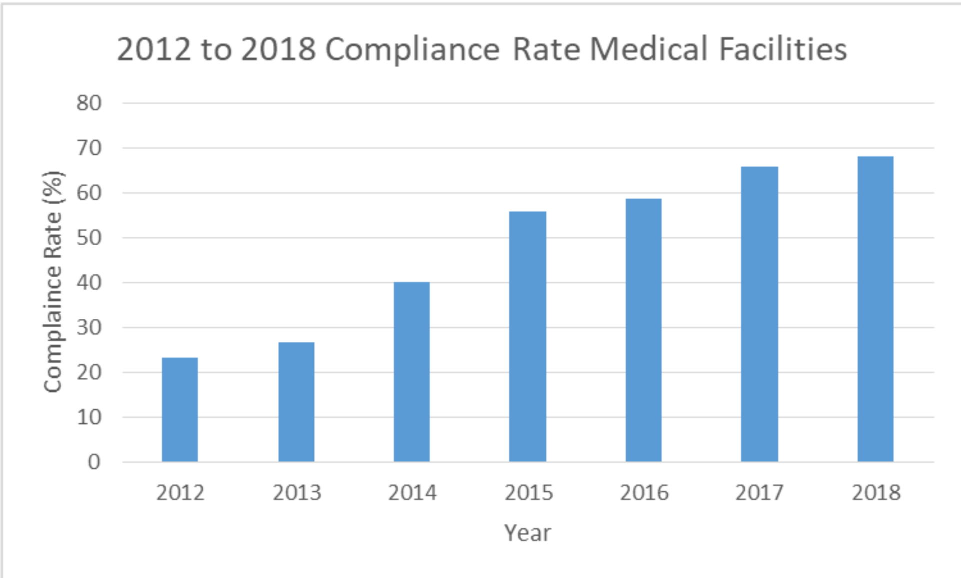

FIG. 1. Chart showing compliance rates for medical facilities from year 2012 to 2018.

(h) The regulatory body reduced the frequency of inspecting other facilities, still considering the risk associated with the practice and the compliance reputation of the facility which in turn led to a significant reduction in the inspections and enforcement budget

(i) Enforcements deterred other facilities from committing similar offenses.

CHALLENGES INCURRED IN USING ENFORCEMENT AS A TOOL TO ENSURE RADIATION SAFETY IN MEDICINE

Although there were benefits that were obtained using enforcements, the regulator faced a number of challenges (some of which still needs to be overcome) which include the following:

(a) Socio-economic impacts- Zimbabwe is a developing country in which there are certain provinces with few healthcare facilities some of which have deteriorating equipment. . In Zimbabwe, there were cases where the only public hospital serving the whole province had to be closed down for non-compliance.

However, issues would arise that the whole community was being deprived of their constitutional right to healthcare and the government would then be engaged for the order to be lifted.

(b) Difficulties in maintaining consistency- For enforcement to be effective, the regulatory body needs to be consistent in its decisions [16]. Because in certain cases the regulator need to use discretion to come up with an enforcement action, it always appeared to the private hospitals, which are usually in compliance as if the regulatory body is favouring government or public hospitals.

(c) In some cases enforcement led to strained relationship between the regulatory body stuff and other stakeholders especially Ministry of Health which used to be the former parent ministry of the regulator.

(d) Financial constraints- the regulatory body had limited financial resources and in some cases it was not possible for inspectors to go and verify implementation of corrective actions where it was warranted.

(e) Human resource constraints- the regulator had limited stuff, and certain cases were withdrawn from the court as the process was time consuming and the inspectors had to be doing other duties.

(f) Lack of training on the part of the inspectors- at times the inspectors would not follow the enforcement procedure or the Orders were written incorrectly and failed to suffice as evidence in court.

(g) The Radiation Protection Act has limited enforcement actions for example there is no provision for fines and no enforcement actions against individuals, which could potentially reduce certain violations.

CONCLUSION

Enforcement can be used by regulatory bodies as a tool to ensure radiation safety in medical imaging. However, there are a lot of challenges involved by using this tool especially in the medical sector considering negative impacts that might arise after an enforcement action is taken against a facility. Maintaining and improving safety culture requires continuous action and commitment. Regulators should encourage licensees to establish a safety culture and employ non-punitive environments and systems for reporting errors and accidents within their organizations. Just as important, they should develop and maintain ongoing training programs for healthcare givers to ensure they remain abreast with new technologies and procedures, and radiation protection methods.

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176-STRATEGIES FOR RADIATION SAFETY EDUCATION AND TRAINING FOR GHANA ATOMIC ENERGY COMMISSION'S WORKERS

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In recent times, global uses of radiation technologies and radioactive sources have increased widely, mostly in medicine as well as in industry, agriculture, and research. In line with the IAEA's mandate to build competence in the area of radiation protection, it offers a range of education and training services and activities to its Member States (MS). The strategic objective was to support MS as they develop sustainable infrastructure and capability to train key personnel to ensure the safe use of ionizing radiation.The Ghana Atomic Energy Commission (GAEC) drawing lessons from such IAEA events over the years have been implementing strategic measures aimed at reducing radiation hazard and radiation doses for its workers and other stakeholders' institutions through education and training in radiation protection and nuclear safety. These set of measures include provision of a safe working environment, promotion of knowledge of radiation risks, minimization of unsafe practices, control radiation risks, sharing responsibility among workers and improving the quality of existing radiation protection programmes. GAEC was established in 1963 by Act (204) of the parliament of Ghana as the sole nuclear authority in Ghana responsible for matters related to peaceful uses of atomic energy, thus fulfilling the state's obligation on nuclear safety, security and environmental care. Initially, the Commission was regulated internally by the Radiation Protection Board (RPB), established by Atomic Energy Amendment Law (PNDCL 308) in 1993 as the national nuclear regulator. In 2015, the RPB was modified into an independent Nuclear Regulatory Authority (NRA) by Act 895 of Parliament.

Generally, it is important to recognize that GAEC operates a wide diversity of nuclear facilities which includes: a 30-kW research reactor; Gamma Irradiator; linear accelerator and Radioactive Waste Management Centre among others (Fig. 1 & 2).

Figure 19: Some Nuclear Facilities in GAEC (a) Modern State of the Art Waste Management Centre (b) Ghana Research Reactor-1 (GHARR-1)

These facilities make GAEC unique as the only organization that renders some specific services in the nuclear industry in Ghana. These services ranged from food preservation, elemental analysis, isotope production, and environmental studies to non-destructive testing among others. For nuclear safety reasons, some of these facilities run Operational Radiation Protection Programme (ORPP) which ensures the effective control of external and internal doses to workers, the public, and of releases to the environment, to ensure conformance with all regulatory requirements and to enable further optimization of operational practices. One important component of the ORPP is Education and Training (E&T) for all workers who may be occupationally exposed to radiation. Therefore, periodic updates of all employee training and education are well planned and executed appropriately.

GAEC being a large and diverse organization employs not less than one thousand categories of workers. These categories of workers are scientific staffs, technical and non-technical staffs. The E&T programmes for individual category differs. For instance, human resource development aspect of the E&T hinges primary on the nuclear scientists. In the area of radiation protection, the commission depends on highly trained workforces, who are produced from the School of Nuclear and Allied Sciences (SNAS). SNAS was established by GAEC in partnership with the University of Ghana, Legon, with the support of the IAEA in 2006. The School was designated as AFRA/IAEA Regional Centre of Excellence for Professional and Higher Education in Nuclear Science and Technology in 2009 and Radiation Protection in 2011. Thus, it offers IAEA Post-Graduate Education Course in Radiation Protection for other MS in Africa. SNAS is currently accredited for programmes leading to MPHIL and PhD in nuclear safety. Presently, a significant percentage of GAEC scientific workforce are SNAS graduates. To further enhance and sustain knowledge, these trainees participate in a number of international and local professional development courses, training workshops and conferences in order to build their competence and capacity in radiation protection. In addition, on the job training arrangements are also an important area for capacity building for these scientists.

As a result of the strategic nature of GAEC, most Ghanaians envisage it as an enterprise where nuclear weapons and other destructive explosives are manufactured making it look very apprehensive in the eyes of the public. This public perception of GAEC is also shared by most staff who works outside the laboratories/facilities. Thus, for staffs in the non-scientific and non-technical staff category, specialized programmes are arranged and executed for them in radiation safety. Given their large number and varied educational backgrounds, GAEC has an elaborated programme/project to educate and train them on the basics of radioactivity, radiation and its uses and effects. The training sessions are done such that, local languages are used where possible for those with minimal formal education. These innovative arrangements have sought to demystify some of the technical concepts regarding nuclear science and technology. It has also foster greater acceptance and collective ownership by staff of nuclear science and security on policies laid down by management. Besides training of its workers, GAEC also through coordination and cooperation liaise, educate and train personnel from other state institutions like, the police, Immigration and other frontline officers. The objective of these engagements is to help control the export and import trade in nuclear materials as well as employing radiation protection in their work.

In conclusion, it is important to state that E&T in radiation protection provided by GAEC over the year is implemented according to international standards. As a result, the Commission can provide IAEA Fellowship and Scientific Visit programmes for African Member States

177-MULTICRITERIA BUDGET ALLOCATION MODELING FOR RADIOACTIVE WASTE DISPOSAL PROJECTS

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ABSTRACT

Radioactive waste disposal has raised greater public concern among policymakers worldwide; having long term safety and budgetary implications. In this study, repository costs and management costs of radioactive waste categories. Using hypothetical cases, three projects are considered under different levels of radioactive waste. The main objective of the study is to develop a multi-criteria budgetary allocation model that determines the combination of repository and management costs required to finance radioactive waste disposal projects. A goal programming method is proposed where budget priorities, goals for expenditure and deviation variables are established in order to define the objective function and goal constraints. Using the simplex method, a feasible solution is obtained that determines the level of goal attainment for allocated expenditure to low level, intermediate level and high-level radioactive waste categories. Results indicate variations in terms of goal attainment for budgetary expenditure based on the cost structures governing a specific type of radioactive waste disposal project. The goal attainment-based approach is therefore an effective tool for allocating budgetary expenditure to radioactive waste disposal projects where relevant cost categories can be priotized if necessary.
Keywords: *Budget; expenditure; goals; radioactive; waste*

1.INTRODUCTION

The allocation of expenditure to radioactive waste disposal budgets is a critical concern among policy makers worldwide. As time has progressed, the population and the amount of waste per person has increased Dormuth [1]; leading to greater public concern over waste disposal problems. Countries with small and medium size nuclear programs may miss the full-range expertise and necessary financial resources; Ferrero [2] to launch their own repository programs. Despite the complexity of radioactive waste disposition with long term safety and budgetary implications, disposal systems of high radioactive waste must satisfy the radioactive safety precaution imposed by regulatory agencies. Radioactive waste is however inevitable. This is because use of radioactive materials from industry and medicine can facilitate generation of electricity. To support management and disposal of radioactive waste management Shimizu [3], development of new technologies of radioactive waste management and reliable planning methods are of special interest to support short term / long term decisions on radioactive waste disposal. According to World Nuclear Association [4], radioactive waste can be categorized as: low level, intermediate level or high level. The major categories of costs incurred during radioactive waste disposal include: repository costs and management costs.

Repository costs comprise of: planning and licensing, design and construction, operation and closure; while management cost components include: interim storage, waste treatment, conditioning, packaging. and incentives.

2. METHODOLOGY

The presentation considers budgetary expenditure using hypothetical cases of three radioactive waste disposal projects with low level, intermediate level and high-level radioactive waste respectively. A goal programming method is proposed where budget priorities, goals of budgetary expenditure and deviation variables are established in order to define the objective function. The goal constraints are then formulated as the criteria for meeting budgetary requirements for low level, intermediate level and high-level radioactive waste disposal projects. The sum of weighted deviations is minimized from the goal values set for budgeted costs of projects. The goal with the highest budgetary cost priority is achieved first and an attempt is made to achieve the next highest ranked goal so that a satisfactory solution is achieved. Overachievement or underachievement of allocating targets for expenditure is determined by using the simplex method for linear programming; that require solution of a minimization problem. Resource leveling is achieved across a hierarchy set of priorities for allocating expenditure in order to accomplish radioactive waste disposal projects.

3. RESULTS AND DISCUSSION

Results from the numerical examples presented indicate that goal attainments for repository costs and management costs can be fully or partially achieved depending on the priority levels and targets for expenditure that govern a specific category of radioactive waste disposal project.

The solution provides feasible results; taking into consideration the contradictory nature of the priorities and criteria involved in executing radioactive waste disposal project activities. The application of this solution approach allows analysts to identify satisfactory allocation of expenditure based on the priority levels set for achieving goals of the budget.

4.CONCLUSION

The goal-attainment based methodology for allocating expenditure to radioactive waste disposal projects can be effective; where relevant cost categories can be priotized if necessary. This can ensure that a mix of balance in project expenditure meets expectations of analysts or policy makers for implementing radioactive waste disposal projects.

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178-COMPUTERIZED RADIOGRAPHY IN RADIOLOGICAL SAFETY

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Due to the need to have a Zonal Surveillance evaluation and control system, which guarantees safety for technicians and the "public", during the generation of ionizing radiation in the quality control of welds; our Organization has implemented the use of computerized radiography, together with the technique of radiation emission by X-rays generators.
The computerized radiography technology obtains images directly in digital format (Fig. 1) by means of the operation of an optical scanner (Fig. 2) that performs the development of photosensitive phosphoric image plates (Fig. 3), which can be reused hundreds of times.

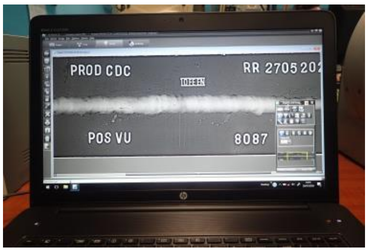

FIG. 1. Digital image

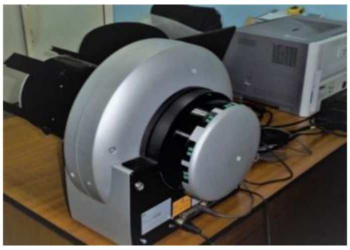

FIG. 2. Optical scanner

FIG. 3. Phosphor imaging plate

Advantages of Computerized Radiography

- Decreases the intensity in Kv by 20% and use only half the exposure time during radiation emission, compared to the same parameters defined by the Exposure Chart for analog radiography.
- The scanning process of the phosphor imaging plate is not affected by the incidence of natural or artificial light. This makes it possible to eliminate the chemical processing of X-ray films, as well as their biological effect on the human organism.
- The equipment is easy to transport due to its small dimensions; facilitating its operation in the field and making decisions quickly.
- The image is converted into a file in the memory of a computer, and can be sent through a service network for storage and subsequent use.
- The software does not allow manipulations of the digital image obtained. It has a package of computer tools for its processing, which enable a better visualization of the detected defects.

Radiation is emitted through air-cooled high-vacuum tube-type X-rays generators. Fig. 4.

FIG. 4. Rx generator

Advantages of X-rays Generators

- Small size, robust and long service life.
- Great portability.
- Stable and controlled operation.
- Remote control with a range of up to 100 meters.
- High quality of radiographic image. 2x2 mm^2 focal spot.

Dosage Rate Record

Based on the results obtained with the computerized radiography, regarding "Good Practices", in the period 2015-2019 no radiological incidents have occurred due to increased safety during emissions. In this time, it was achieved that the general average reading of the Dose Rate represented 5% compared to the regulated values (20 microSv/hour Control Zone), which demonstrates the effectiveness of this technology, in accordance with international standards of control to situations of over-exposure. Table 1.

TABLE 1. DOSAGE RATE RECORD

	Year 2015	Year 2016	Year 2017	Year 2018	Year 2019
			Dosage Rate Average (microSv/hour)		
Ships Repair	0.349	0.286	0.975	0.091	0.391
Fuel Tank Repair	0.600	0.290	0.527	-	-
Homologation of Welders	0.059	0.073	0.916	0.050	0.040
Total	1.008	0.649	2.418	0.141	0.431
General average			0.929		

Conclusions and Achievements

The use of computerized radiography has allowed us to improve the safety at work of technicians and more effectively protect the "public" during practices.
The main achievements were the following:

- The safety levels of all the personnel involved were increased, due to the fact that it was possible to reducethe intensity of the radiation by 20% and the exposure time by 50%, compared to analog radiography.
- The general average Dose Rate reading (0.929 microSv/hour), during radiographic practices, represented5% compared to the regulated value (20 microSv/hour. Control Zone).
- The Personal Equivalent Dose Hp(10) values reported by the external dosimetry laboratory were keptbelow the lower detection limit (0.1 mSv).

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179-STUDY OF GENOME DAMAGE BY GAMMA-H2AX AND COMET ASSAY IN MEDICAL RADIATION WORKERS

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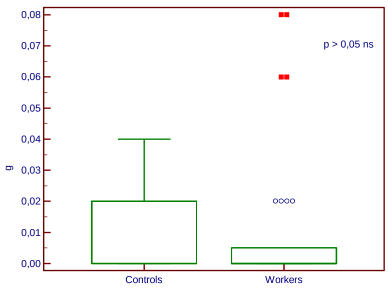
A population study is reported in which the DNA damage and the kinetics of the subsequent repair were estimated by and comet assays in isolated lymphocytes of 41 volunteer radiation medical worker and administrative worker as control from Fatmawati Hospital, Jakarta, Indonesia. The parameters occupational radiation exposure-induced DNA damage, the rate of repair, and residual non-repaired damage were measured by γ-H2AX index and comet assay parallel. From this study, there were no statistical different between γ-H2AX and total length of migration DNA (TL) by comet assay as a parameter of genome damage between medical radiation worker and administrative staff (control) ($p>0.05$), it found tend of positive correlation between γ-H2AX and total length of migration DNA workers compare in controls. In conclusion, there were any change of DNA in workers that can be detected by γ-H2AX and comet assay

Keywords: γ-H2AX, Comet Assay DNA damage

Ionizing radiation exposure in several condition as accidental, occupational or medical, may leads to dangerous biological effect to death or tumorigenesis. Significant doses of exposure are clearly known to initiate acute and chronic effects in humans, while the possibility risk for detrimental effects related with low doses of radiation exposure is still a matter of discussion [1]. In this current study total, 41 intravenous blood sample was collected in heparinized from 33 workers and 8 control, can be seen in Table 1. In the Fig 1 (a,b)) can be seen gamma-H2AX Index, TL DNA (μm) in control and workers, the relationship between gamma-H2AX Index and TL DNA (μm) in control and workers in Fig 2 (a and b).

TABLE 1. CHARACTERISTIC WORKERS AND CONTROLS

Subject	Gender (W/M)	Ages (mean±SD, years)	Gamma-H2AX Index	TL DNA (μm)
Controls n=8	W=2, M=6	33-57	0,00-0,04	13-23
Workers	W=16, M=15	27-64	0,00-0,08	12-28



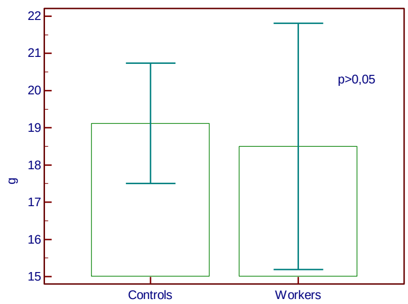


Fig 1. Gamma-H2AX Index, TL DNA (μm) in control and workers

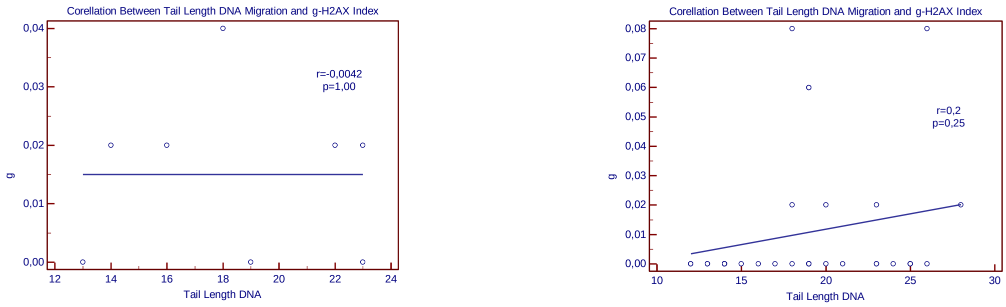


Fig 2. Corellation between Gamma-H2AX Index and TL DNA (μm) in control and workers in Fig 2 (a and b).

There were no significant different both gamma H2AX foci index and TL DNA between workers and control. This result almost identic with our publication before [2]. In Fig 2a, and tendency correlation between gamma H2AX foci index to and TL DNA (b), it still need more discussion to know the using gamma H2AX assay and comet assay to detect any change in DNA workers that accepted ionizing radiation exposure in range occupational dose.

ACKNOWLEDGEMENTS

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180-EXPRESSION OF FREQUENCIES OF MICRONUCLEI AND ΓH2AX IN LYMPHOCYTE OCCUPATIONAL MEDICAL WORKERS

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Occupational radiation workers in medical area have risks to be exposed to ionizing radiation that also potentially cause DNA damage. An important effect of radiation exposure is the formation of DNA double-strand breaks (DSBs), considered to be one of the most damaging DNA lesions. It also has potential to influence cell mitosis and DNA damage both double-strand break (DNA DSB) and single strand break . Cell damage at this level of which can be observed in forme d of micronulei with biomarkers γ-H2AX [1,2]. This study aims to determine the relationship between micronuclei and γ-H2AX expression in individuals who worked on chronically exposed by low dose Ionizing radiation. The peripheral lymphocytes taken from 30 blood samples were taken from radiation workers and 21 from administrative staff served as control. The study showed that the frequency of micronuclei and expression of γH2AX both in exposed and control group still in normal range and there were no significance statistically significance both in γH2AX ($p = 0.1$) and micronuclei ($p=0.8$) to control. There were no significance statistically significant correlation or association between the foci γH2AX and micronuclei ($p = - 0.76$). The mean for duration of work were 15.75±10.217 . The mean γ-H2AX foci calculated from the number of foci in 50 lymphocytes while the micronucleus 1000 in binucleat cell lymphocytes were showed in table I.

Keywords: γ-H2AX, Micronuclei, Assay ,DNA damage

Table .1The mean value of MN and γ-H2AX foci and control groups.

Group	Σ of sample	Duration work (Years)	Σcells (BNC)	Range and Mean MN	Range and Mean pociH2AX
Expose group	30	1 – 34 18.26.± 9.7	30.000	1-33 17.6 ±9.47	0.0 - 0.9 0.15± 0.2
Control	23	-	23.000	2-29 17.08 ± 6.9	0-0.36 0.05 ± 0.07
All samples	53		53.000		

* Students t-test: P >0.05 (not different from control group)

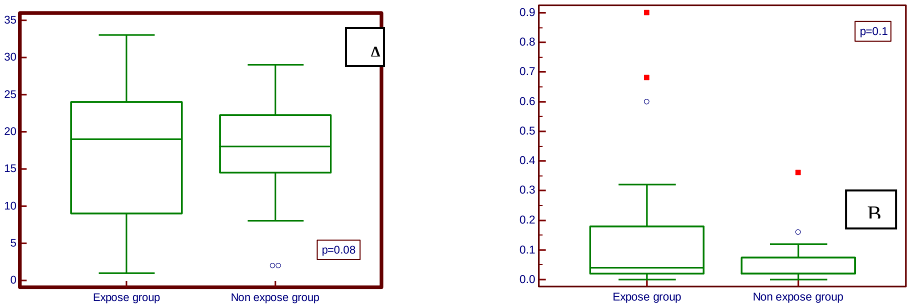

Figure 1. The mean of ϒ-H2AX foci in expose subject and control (A) and The mean frequencies of MNCB

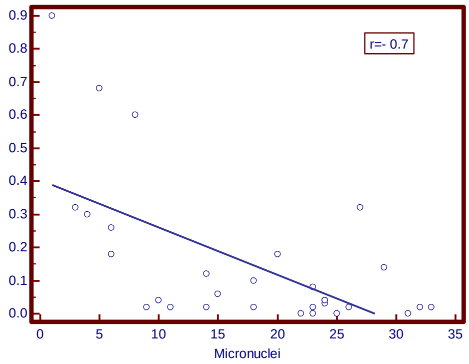

Figure 2 : Correlation between micronuclei and H2AX in medical workers

The Study showed that the frequency of MN and γ-H2AX both in exposed and control group still in normal range and there were no significance statistically significance both in γH2AX (p = 0.1) and MN (p=0.08) to control. There were no relationship correlation between micronuclei and γ-H2AX foci and (r = -0.76) Figure 2. According to several researcher reporting that the correlation between MN and γ-H2AX lies in the fact that the rapid dephosphorylation of γ H2AX, which is the signal for the correct recruitment of DNA repair proteins, can prevent the progression of DSB to chromosomal fragments and thus to MN [3]. The existence of DSBs in MN are the products of DNA damage, but the initiation of DNA Damage Response is controversial as MN do not contain DNA repair machinery. Nevertheles in this study its can be state that there were no different low doses of radiation exposure that can be make the potential of DNA dsb both in expresi MN and γ H2AX foci. A number of studies have suggested H2AX expression will increase with age, and the number of foci in men is greater than in women. [4].It can be concluded that chronic low radiation dose exposure did not affect the expression both of micronulei and γ H2AX.

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181-THE IMPACT OF OPTIMIZATION IN CT PEADIATRIC IMAGE QUALITY WITH REGARD TO FIGURE-OF-MERIT: AN ANTHROPOMORPHIC PHANTOM STUDY

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Computed tomography (CT) scanner has been acknowledged as one of the most important imaging modalities for diagnostic purposes as it produces high contrast sectional images that helps physicians diagnosed the lesion precisely. Despite its advantages, CT scan is associated with significant amount of ionizing radiation. Radiation dose from CT is much of concern especially for paediatrics as there are vulnerable to radiation and are expected to have longer a lifespan to develop long-term radiation-induced health effects [1]. Hence, there are numbers of optimizations technique developed by manufacturers and researcher to reduce the impact of radiation to the patients [2], [3]. However, the impact of the optimization usually related to the diagnostic value of CT images. Adjusting tube potential and tube current are the most common technique used to reduce radiation dose, but it has influence towards the image quality of the CT images where the noise may significant. Henceforth. the only practical way to determine the effectiveness of optimization, balance between reducing dose and image quality, is by quantifying the value of signal-to-noise ratio (SNR) and contrast-to-noise ratio (CNR) of the images.

Adjusting tube current for optimization remains limited by the use of the standard filtered back projection (FBP) reconstruction technique, because the FBP technique significantly increases the image noise when the dose reduction is too great [4]. With the introduction of new iterative reconstruction (IR) algorithm in CT allowed user for a significant noise reduction in CT images and yet it is clinically relevant [5]. Thus, selecting the appropriate optimization technique is an important task but the combination of either IR or FBP and determining how much radiation dose reduction still depends on the region and the diagnostic task being performed. The aim of this study is to evaluate the impact of optimization on the SNR and CNR and to establish the Figure-of-Merit that represent the relationship between image quality and radiation dose.

METHODOLOGY

In this study, a physical anthropomorphic phantom (ATOM Phantoms; CIRS, Norfolk, VA) was used representing the paediatric person as 1-year-old (Figure 1). The phantom is 75 cm height and 10 kg weight and made of radiologically tissue-equivalent material. The phantom was scanned by using fixed tube voltages, 100 kVp of a 64-slice multidetector row CT scanner (LightSpeed VCT; GE Healthcare, Inc, Milwaukee, WI). Seven different protocols include in this study with the fix configuration of the nominal beam width (64 x 0.625 mm), the rotation time of the gantry (0.5 seconds), the pitch factor (1.2), and the reconstructed slice thickness (5 mm) (Table 1). Some of the protocols involves dynamic change of the tube current in low attenuating region and is called as an automatic tube current modulation (ATCM). ATCM allows technologist to pick a desired level of noise or reference to mAs.

Table 1: Parameter involves in the study

Parameter	Protocols						
	P1	P2	P3	P4	P5	P6	P7
Tube current (mAs)	210	180	150	210	180	150	210
Tube current modulation (TCM)	On	On	On	Off	Off	Off	On

The image quality of the 7 protocols produced have been analysed by placing 6 circular shape of Region of Interest (ROI) on mid-thorax region. From the ROI, the mean, the standard deviation (SD), minimum and maximum signals can be observed within ellipse. Both SNR and CNR were calculated from the infomation. Figure of merit (FOM) were established to indicate the optimization index and can be calculated using formula below:

$$FOM = \frac{\left(\frac{(HU_a - HU_b)}{SD_b}\right)^2}{CTDI_{vol}} = \frac{CNR^2}{CTDI_{vol}}$$

where HU_a and HU_b are respectively the object's HU and background, and the SD_b is the background's noise.

RESULTS AND DISCUSSION

As the tube current increases, the SNR and CNR are also increases. The protocol P4 gives higher values of SNR as protocols switch off the TCM function and use higher effective mAs. The image quality of the CT thorax is being determine by the performance of FOM. As the values of SNR and CNR is higher, the FOM is also higher. Figure 1 gives insight on the effect of scanning parameter (tube current) towards FOM of SNR.

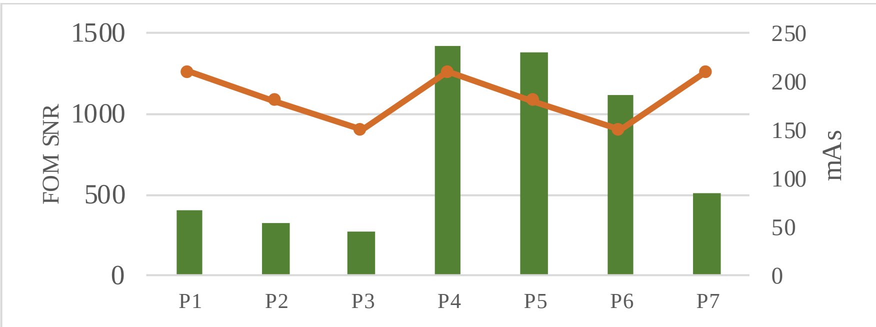

Figure 1: Figure of merit calculated from difference protocols

The FOM is analysed by calculating the ratio of SNR and CNR with $CTDI_{vol}$. The highest FOM of SNR is at P4 (1423.72) while the smallest value is at P3 which is 271.76. Protocol P4 is also the one that contribute highest radiation dose which $CTDI_{vol}$ is 13.7 mGy. Since there is no significant finding between doses, the increase value of FOM indicate the ability of a system to deliver better performance of image quality in terms of SNR and CNR at a relevant used dose. To conclude, the FOM is relevant for optimization purposes and should be done periodically to identify the ideal performance of CT image under a certain condition and parameters

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182-MULTICENTRIC COMPARATIVE STUDY OF DOSE INDEXES USING AN "IN VIVO" OPTICAL FIBER DETECTION SYSTEM DURING CT EXAMS

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Aims:

Determine the discrepancies between measured and estimated dose indexes. Provide a better estimation of the patient dose and monitor the CT performances during examinations.

Material and method :

The multicentric study is conducted on a period of 2 years in France on scanners from 3 Hospital centers (CHU de Rangueil - Toulouse, Centre Léon Bérard - Lyon and Centre Paul Strauss - Strasbourg) equipped respectively with a Canon Aquilion One Genesis, a Siemens Somatom Definition Edge and a GE LightSpeed VCT64. An innovative detector, based on the use of a scintillating optical fiber (IVIscan®, FIBERMETRIX, France) allows to measure the delivered dose and the CT dose indexes (CTDI and DLP) with a temporal resolution of 1 ms during CT exams.

The Relative Dose Difference (RDD) was calculated between CT estimated and measured CTDI and DLP for CT exams in the following anatomical regions : skull, thorax and thorax-abdomen-pelvis (TAP). Large fluctuations of the RDD was used to identify outlier exams. The IVIscan® system was also used to determine the dose recovery and distribution along the axial direction for examinations involving several acquisitions.

Results:

An average of 30% RDD was observed for CTDI and DLP between the CT estimations and IVIscan measurements estimated for the different examinations, regardless of the center studied. Variations of RDD up to +400% were detected for cardiac examinations with prospective ECG gating in a CT model. The real-time measurement allowed the detection of a malfunction in the intensity modulation that was not visible through the CT dose estimation. In addition, for TAP examinations often requiring multiple acquisitions with overlapping areas, the assessment of the dose recovery showed a local dose exceeding national diagnostic reference levels (DRLs).

Conclusion:

The study shows significant differences between measured and estimated doses indexes regardless of the examination performed, the CT device or the practices of the center. The mean overestimated CT dose indexes is in part the result of morphological issues. Once these differences are known and documented for a given examination, large variations in these differences can be used to monitor any practice shift or the state of a scanning device. The use of the IVIscan® system makes it possible to obtain a real-time assessment of the patient measured and estimated dose, and therefore to identify possible malfunctions of the scanning device or unusual practices. Moreover, the dose recovery information given by the optical fiber dosimetry system helps to improve the determination of the absorbed dose and organ dose during a CT exam.

183-ASSESSMENT OF DNA DAMAGE IN MEDICAL RADIATION WORKERS USING THE ALKALINE COMET ASSAY AND THE CHROMOSOME ABERRATION

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This study aims to provide preliminary data on the genotoxic implications resulting from the work of chronic low-dose ionizing radiation exposure. The assessment of primary DNA damage in peripheral blood leukocytes from medical staff was carried out using a comet alkaline test and the data obtained were compared with conventional cytogenetic results using a chromosome aberration (CA) test. Evaluation of genetic damage was then performed on 54 subjects (30 exposed and 24 controls). From the results comet assay, there were significantly increase the tail length value (TL) of the exposed group ($p = 0.0047$) but the value of the tail moment (TM) was not significantly different between medical radiation workers and administrative staff (control) ($p > 0.05$). From the results aberration chromosome, there were significant differences between the two groups although the values were still within normal limits where the range of background values was 1-3. The conclusions from this study indicate the possible genotoxic implications of nuclear medicine in employees exposed to low-dose ionizing radiation.

Keywords: Abberation chromosome, Comet Assay, DNA damage

Irradiation exposure at work has become a concern because of ionizing radiation deposits energy in the cell, causing cell damage [1,2]. Medical radiation exposure is usually a low dose, but for a long period. There are still several levels of risk for the development of genetic damage after radiation [3,4]. Our Research is evaluating the damage caused by medical radiation exposure by single-cell electrophoretic gel (comet test) and chromosomal aberration analysis. The evaluation was performed on 54 subjects (30 exposed and 24 controls). The characteristics of the sample are presented in table 1.

TABLE 1. CHARACTERISTIC OF THE STUDY POPULATIONS (N=54) GROUPED ACCORDING TO EXPOSURE STATUS.

	Exposed subject		Control subject	
	Number	Age (years)	Number	Age (years)
Male	16	44,62±7,88	11	40,54±11,48
Female	14	42,42±9,75	13	40,43±10,93

In primary DNA damage there was a significant increase in TL value of the exposed group ($p < 0.05$) but the TM value was similar to the control group. There was an increase in CA frequency compared to the observed controls. In the exposed group, the value of dysenteric chromosomes (Table 2) was obtained 1.13 per 250 cells and fragments 1.15 per 250 cells while the value of the two parameters in the control group was 0 (Table 3). There were significant differences in the two groups although the values were still within normal limits where the range of background values was 1-3.

TABLE 2. COMET ASSAYS, END-POINT IN THE CONTROL AND EXPOSED STUDY GROUPS.

Study Group	Tail Length (µm)	Tail Moment (µm)	Tail Length distribution (µm)	Tail Moment distribution (µm)
Control	14,11±2,09	0,97±0,28	10,14-17,78	0,44-1,49
Exposed	16,28±3,01	0,90±0,47	11-23,16	0,29-2,54

TABLE 3. STRUCTURAL ABERRATIONS IN THE CONTROL AND EXPOSED STUDY GROUP.

Study Group	Age (Years)	Dicentric Chromosom	Acentrik Fragmen
Exposed	44,46 ± 10,38	1,13 ±0,34	1,15 ±0,37
Control	45,23 ± 10,87	0	0

The conclusions from this study indicate the possible genotoxic implications of nuclear medicine in employees exposed to low-dose ionizing radiation. Therefore, medical personnel must implement radiation protection procedures carefully and must minimize radiation exposure as low as possible, to avoid the possibility of genotoxic effects [4]. The advantage of biomarkers is that damage to individual radiation can be measured which includes the variability of individual radiosensitivity. According to our results, Comet assays and CA test data are sensitive biomarkers that can be used as additional supplements for physical dosimetry in the regular health surveillance of occupational-exposed radiation workers. It still needs more study to know the using comet assay and CA test to detect any change in DNA workers that accepted ionizing radiation exposure in range occupational dose.

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184-RADIATION PROTECTION REQUIRMENTS FOR MEDIKAL APLICATION OF IONIZING RADIATION IN THE REPUBLIC OF NORTH MACEDONIA

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Abstract – In this paper, the regulatory infrastructure in radiation protectionin the Republic of North Macedonia is presented. The national radiation protection requirements for the medical application of ionizing radiation are reviewed for both occupational exposed persons and patients undergoing medical treatment with ionizing radiation and their compliance with the international standards is considered.
The gaps identified on the national level are presented and steps for overcoming such gaps are analyzed.

RSD ROLE IN RADIATION PROTECTION

Radiation Safety Directorate (RSD), as independent state administrative body with the capacity of legal entity The regulatory body is conferred upon the legal authority and is provided with the competence and resources necessary to fulfill its statutory obligations for the regulatory control of facilities and activitiesThe principle functions and activities of Directorate determined by the Law are: establishing radiation protection and safety requirements through development of regulations, guides and other acts, inspection, enforcement.licensing practicesinvolving ionizing radiation sources, performing inspections and enforsing the regulatory requirements, maintaining national register of ionisin radiation sources, occupationaly exposed person and nuclear material, establishing intervention levels and undertaking interventions in case of emergency, establishing institutional and international cooperation on matters within the competence of the RSD etc. Therefore, the radiation protection and safety requirements in different applications of ionizing radiation sources, including the medical applications, are to be established and enfoced by the RSD.

2. RADIATION PROTECTION IN MEDICAL APPLICATION OF IONISING RADIATION

The national legislation regulates the system of control of all ionizing radiation sources, as well as the protection of population and environment against the exposure or potential exposure to ionizing radiation. The main objectives of the Law are adequate protection of population, society and environment against harmful effects of ionizing radiation and safety of ionizing radiation sources and radioactive waste and the safety and the security of radioactive sources.

2.1 Basic principles of radiation protection

The law promotes the three basic principles of radiation protection – justification, optimization and limitation of doses .Therefor e:

- No license shal be issued by the RSD according article 9 in the Law for practice involving ionizzing radiation that is not justified taking in account all social, economic and all other relevant factors,
- Each legal entity performing practices involving ionizing radiationshal ensure that the occupational protection and safety is to be optimized in accordance with ALARA principle, and
- The total dose recived by any person shall not exceed the dose limits prescribed in { 4} taking into account all the exposure pathways and all the sources of exposure with the exception of the doses incurried within medical exposure..

2.2. Occupational radiation protection

The Law on ionizing radiation protection and radiation safety prescribes general principles of radiation protection. In accordance with the Law, the general requirements for radiation protection and ensuring security of the ionizing radiation sources should provide normal exposure within the limits of the doses prescribed by the Directorate and in accordance with the following basic principles of radiation protection:

1.Principle of justification,

2. Principle of optimization

3. Principle of limitation of dose

.2.3. Radiation Protection in Medical Exposure

Registrants and licensees shall ensure that no person incurs a medical exposure unless there has been an appropriate referral, responsibility has been assumed for ensuring protection and safety, and the person subject to exposure has been informed as appropriate of the expected benefits and risks.

3. CONCLUSION

The RSD continuously follows the updates of the international and EU standards and implement them in the national legislation by preparing and issuing amendments. The practice, the experience and reports from the Unit on inspection, as well as the implementation of the regulations during the licensing process are important part of the updating of the regulations. In the process of review and revision of the regulations are taken into account the problems and situations that are identified in the process of analysis of the field that is in the competence of the RSD performed each year in the process of the strategic planning of the RSD.

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185-MONTE CARLO CALCULATIONS OF EXPOSURES IN MEDICAL LINEARR ACCELERATOR TREATMENT HEAD USING ADVANTG

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An application of the ADVANTG (AutomateD VAriaNce reducTion Generator) code, which combines the Denovo deterministic transport solver with the MCNP (Monte Carlo N-Particle) Monte Carlo code, to the medical linear accelerator treatment head dose rate estimation in the surrounding environment is presented in this paper [1].

In 1953 at Hammersmith Hospital (London, UK) first Linear accelerator (LINAC) was applied. It was used to cure tumor in a 2-year-old boy eye [2]. After more than sixty years and more than 50 million patients, medical linear accelerators have become the crucial device in radiation therapy for cancer worldwide.

In LINACS electron beams are generated by electron gun. The first ones used diode type guns. In London first device could generate electron beam which energy was only 8 MeV, but nowadays modern LINACS are using Pierce-type electron gun [3]. Differences are not just in gun type but also in electron beams energies as well. In LINACS beam energies differences depends of equipment provider, in particular Varian 21iX LINAC can operate at 6, 9, 12, 16, and 20 MeV, a Varian TrueBeam: 6, 9, 12, 16, and 20 MeV, and an Elekta Versa HD: 6, 9, 12, and 15 MeV. [4]. To ensure safe operation and safety of medical personal or patients it is important to know distribution of electron/photon radiation fields in treatment head and their interaction process.

Neutral particles and electron transfer calculations have fundamentally differences. Neutral particle interactions are characterized by relatively uncommon collisions and long mean free path, while electrons interact with matter by the Coulomb laws, which cause a lot small interactions. Suppose what photons and electron interacts with an aluminium plate. When energy decreases from 0.5 MeV to 0.0625 MeV, photons will have less than 10 interactions and electrons have more than 10^5 [4].

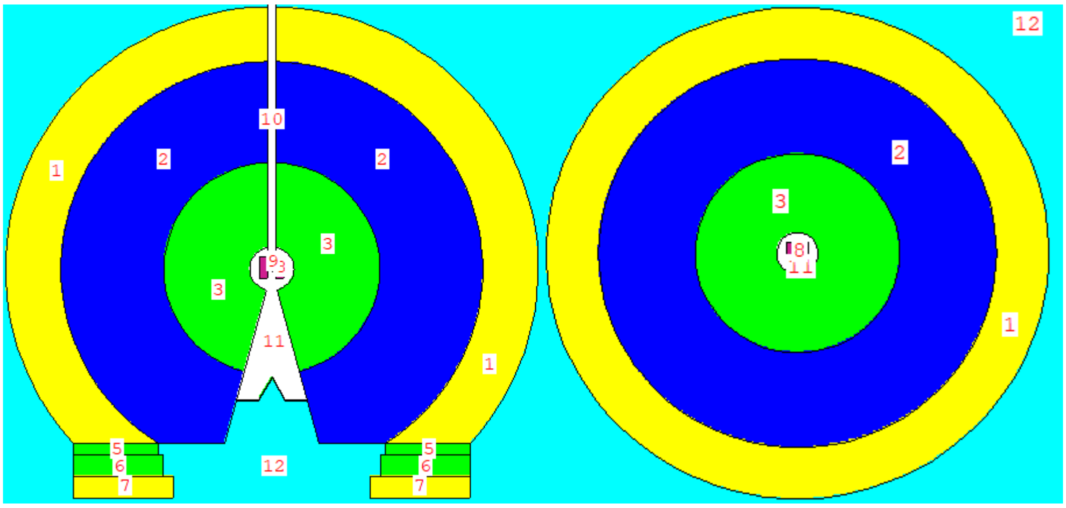

FIG. 1. MCNP model of linear accelerator treatment head in the Y-Z (left) and Y-X (right) plane.
(Cells materials: 1,7-Lead; 2-Stainless Steel (SS-316L); 3,5,6-Tungsten; 9: Copper, 12-Air; 11,12-vacuum)

In this work, the MCNP [5] model was created for medical linear treatment head. It was used for studies of interactions process induced by electrons, which energies are 9, 16 and 20 MeV. Attention is also drawn to the resulting photons and their interaction processes. Firstly model of medical linear accelerator treatment head was created using MCNP-VIS software (see figure 1). Subsequently 10^8 particles histories was calculated using MCNP6 to obtain gamma dose radiation fields around the treatment head with coupled ADVANT code.

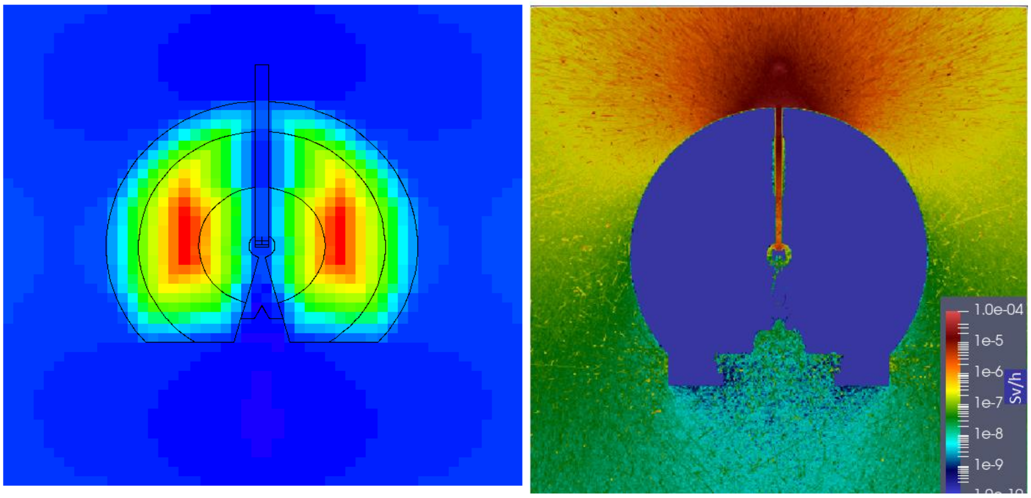

FIG. 2. WW generated using ADVANTG for the model (left) and gamma dose rate field (right) for the model of linear accelerator treatment head.

After calculations, one can observe the tendency stating that as the electron energy increases, the electron migration between the cells decreases. However, the particle population is increasing in almost all parts of treatment head. Deriving from this result, it can be concluded that after increasing the energy, the decrease of the electron free path could be observed and the number of interaction processes in each shell is increasing in the elaborated model. In contrast to electrons, gamma migration in treatment head cells and their occurrence in certain cells increases, resulting the increase of initial electron energy. In addition, calculations show that gamma formation processes increased approximately twice by changing the initial electron energy from 9 to 20MeV. Moreover, the number of electron interaction processes increases approximately by 13 times. The number of photon interaction processes increased more than by the order of 2.

The most common interactions are: secondary electron creation, X-ray formation, and photon capture. The 1.2 time increase of photon interaction process as the electron energy increases from 9 to 16 MeV increases the dose rate from 0.54 mSv/h to 0.81 mSv/h. Further increasing the energy from 16 to 20 MeV, although the reaction increased 1.2 times too, but the dose rate changed only by 0.08 mSv/h (0.81-0.89 mSv/h). More details can be observed from Figure 2, where weight windows, created using ADVANTG, and gamma radiation field maps are presented.

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186-DOSIMETRIC STUDY OF CT CHEST SCANS USING ADULT PHANTOMS

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6. INTRODUCTION

The spread of Computed Tomography (CT) exams in clinical applications has been expanding in the diagnosis of traumas, staging and screening of tumors, brain diseases etc. These uses promoted a significant increase in the absorbed dose by patients and, consequently, in the population's doses.

The patient's dose values on Computed Tomography depend on the scan acquisition protocols. Today, CT scans are a very fast, painless and non-invasive tests, performed with high quality images, widely accessed by the general population.

Therefore, it is essential to improve protocols, seeking lower doses, without impairing the image diagnostic quality. The doses received by the people are related to risks of stochastic effects, and these risks increase with the dose value and a long life expectancy.

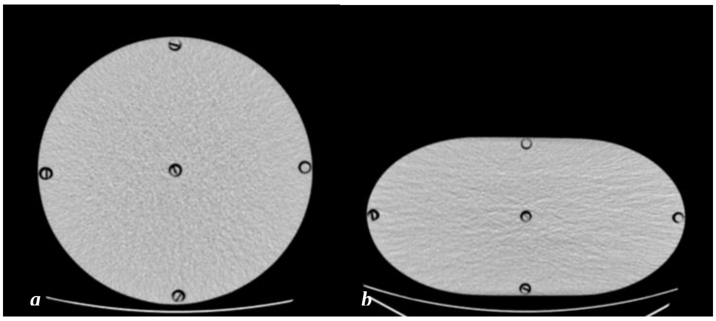
7. METHODS

The experiments were performed on a Toshiba CT scanner, Aquilion Prime model, with 80 channels. In this study were used two chest phantoms made of polymethylmethacrylate (PMMA). The cylindrical chest phantom is the standard used to obtain Computed Tomography Dose Index (CTDI) to the body. This chest phantom has a diameter of 32 cm and 15 cm in length.

An oblong chest phantom was developed, based on the chest size of an adult human body, in the axillary region. The oblong phantom has 43 cm in width, 22 cm in height and 15 cm in length. The axial cutting area of the oblong phantom is defined by two semi ellipses generated from an ellipse of 30 cm by 22 cm and a rectangle of 22 cm by 13 cm. Fig. 1 shows a sectional image of the central slice of the two chest phantoms.

The chest phantoms have five openings with a diameter of 12.67 mm for the placement of the pencil chamber, one central and four peripheral lagged of 90º. The center of the peripheral openings is positioned 10 mm from the phantom edge. In analogy to the display of an analog clock, peripheral openings were named 3, 6, 9 and 12, according to position within the gantry during the scanning of the central slice. The chest phantom have the same axial cutting area, the same length and, consequently, the same volume. CT sectional images of the central slice of the chest phantoms are in the Figure 1.

FIG. 1. Chest phantoms, standard (a) oblong (b).



Each phantom was placed in the CT scanner isocenter, and the central slice (10 mm) of both were irradiated successively using two different voltages (120 and 135 kV) and 100 mA.s. Measurements of air kerma values (C_{k100}) in each opening was done using a pencil chamber. It was realized five measurements in each opening.

The scanning of 10 cm of the phantom central region was done in helical mode using each voltage (120 and 135 kV), mA.s automatic with a maximum value of 250 mA.s, tube time of 0.5 s, pitch of 1.388, X-ray beam of 4 cm (80x0.5), image reconstruction 2 mm. Based on this protocol and the Air kerma values were calculated the $CTDI_{vol}$ values. The PMMA/air conversion values used to obtain absorbed dose in PMMA were 1.042 and 1.045 to 120 and 135 kV, respectively.

8. RESULTS

The volumetric CT dose indexes obtained are showed in the Table 1. The average value of mA.s using 135 kV was 150 to the standard and 170 to the oblong phantom and using 120 kV were 200 to the standard and 190 to the oblong phantom. Observing the results, the use of 120 kV the highest average current value occurred for the standard phantom and the for 135 kV the highest current value occurred for the oblong phantom.

Comparing the protocols, the use of 135 kV voltage was better for the standard phantom, with the lowest dose index (9.95 mGy) and the use of 120 kV voltage was better for the oblongo phantom (10.42 mGy).

TABLE 1. Volumetric dose indexes for chest phantoms.

Voltage	Chest Phantom	Average Current (mA)	Average Charge (mA.s)	CTDIvol (mGy)
135	Standard	300	150	9.95 ± 0.13*
135	Oblong	340	170	11.83 ± 0,25
120	Standard	400	200	10.42 ± 0.19
120	Oblong	380	190	10.04 ± 0.23

Standard deviation

9. CONCLUSIONS

The results obtained in this study demonstrate the $CTDI_{vol}$ value in chest scans, which is formed by radiosensitive structure such as mediastinum, heart, blood and lymphatic vases.

The protocols purposed for this CT scanner showed that the shape of the phantoms have influence in the dose deposition and on the adjustment of current automatic control. The use of the acquisition protocol of 120 kV has presented the lowest dose index for the oblong phantom, with a shape similar to the chest. This work allowed to observe the dose variation absorbed the chest phantom with the variation of the value of the tube voltage and adjust of the other parameters.

Nowadays, even though the technological equipment's manufacturers already provide the acquisition protocols for the clinical exams, which also includes automatic tools to improve image quality and reduce the patient's doses, complementary tests can help considerably in the optimization of CT image acquisition processes.

ACKNOWLEDGEMENTS

This study was financed in part by the Coordenação de Aperfeiçoamento de Pessoal de Nível Superior - Brasil (CAPES) - Finance Code 001. Also, this work was supported by the FAPEMIG.

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189-DEVELOPMENT OF INDONESIA'S REGULATION ON GRADED APPROACH OF DOSE MONITORING PERIOD FOR RADIATION WORKERS ON PLANNED EXPOSURE SITUATION.

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SYNOPSIS

The purpose of this paper is intended to describe Indonesia's action plan to follow up on the findings of the IAEA mission through Integrated Regulatory Review Services (IRRS) in 2018. In the first place, according to the IAEA findings, Indonesia should consider requiring the individual monitoring periods that are commensurate with the exposure condition. This is due to Indonesian BAPETEN Chairman Regulation No. 4 of 2013 on "Radiation Protection and Safety in the Utilization of Nuclear Energy" requires that a licensee have to monitor the radiation worker doses once every three months and this provisions apply to all planned exposures and existing exposure conditions. This provision is not in line with the IAEA recommendation on GSR part 3 which provides the basis for the graded approach to the control of exposure that "The regulatory body shall adopt a graded approach to the implementation of the system of protection and safety, such that the application of regulatory requirements is commensurate with the radiation risks associated with the exposure situation."

To implement GSR part 3 recommendations and to meet the findings of the IRRS mission 2018 related to monitoring individual doses, a follow-up by BAPETEN as the regulatory body is to amend BCR No.4 of 2013 regarding individual dose monitoring. Amendment of BCR No.4 of 2013 must be in line with the draft of Government Regulation on Licensing for the Utilization of Ionizing Radiation Sources and Radioactive Material where in this draft stated that the activities must have a license from Nuclear Regulatory Body. Activities that must have a license in the draft are categorized based on the level of risk, level of complexity in the operation and historical of accidents. According to the categorization of activities, IAEA finding about monitoring dose for worker will be part on the amendment of BCR No.4 of 2013. Requirement about monitoring dose for worker will be established with the graded approach on the implementation depending on the activities. At the moment the draft of Government Regulation has been completed by the Ministry of Law and ready to be signed by the President.

TABLE 1. BELOW IS CATEGORIZATION OF ACTIVITIES BASED ON PLANNED EXPOSURE SITUATION ON DRAFT OF GOVERNMENT REGULATION.

Category A	Category B	Category C
1. High risk for safety 2. Complicated operation procedure 3. Needed complex safety training 4. There is a history of problems with safety in operations.	1. Middle risk for safety 2. Need more one operation procedure 3. Needed safety training 4. There is a history of few problems with safety in operations.	1. Low risk for safety 2. Simple operation procedure 3. Needed simple safety training 4. There is a no history of problems with safety in operations.

Types of activities based on Table of Category A, Category B and Category C on draft of Government Regulation is:

1. Activities on the Category A consist of: production of radioisotope, Radiotherapy, Gamma Irradiator, Electron Beam Irradiator, radioactive waste facility;
2. Activities on the Category B consist of: export and import of radioactive material, well logging, industrial radiography, gauging, non-medical human imaging equipment, equipment for inspection purposes, diagnostic and interventional for medical purposes;
3. Activities on the Category C consist of: export and import for radiation generator, gauging, analytical equipment.

IAEA on RS-G-1.9 about Categorization of Radioactive Sources introduce 5 categories for Radioactive, there are:

TABLE 2: CATEGORIZATION OF RADIOACTIVE SOURCES

Category	Activity ratio (A/D)	Risk
1	$A/D \geq 1000$	Extremely dangerous to the person
2	$1000 > A/D \geq 10$	Very dangerous to the person
3	$10 > A/D \geq 1$	Dangerous to the person
4	$1 > A/D \geq 0.01$	Unlikely to be dangerous to the person
5	$0.01 > A/D$ and $A >$ exempt[d]	Most unlikely to be dangerous to the person

Indonesia has proposed graded approach justification for activities on draft of Government Regulation based as IAEA recommendation on Table 2.
Table 3 is cross reference between Table 1 and Table 2, where table 3 will be used to determining safety and security requirements in radiation protection and safety practices as well as security of radioactive material.

TABLE 3. CATEGORIZATION ACTIVITIES BASED ON INDONESIAN GOVERNMENT REGULATION

Activities Category Base On Government Regulation	Categorization of Radioactive Material (RS-G-1.9)
A	1
B	2 and 3
C	4 and 5

To meet the IAEA's recommendations IRRS mission 2018, Indonesia must amend regulations regarding the evaluation period for personnel dose monitoring. To change this provision, Indonesia will amend BCR No. 4 of 2013. Based on table 3 and current data, almost monitoring dose tools that are used by the workers in Indonesia are TLD, this is because most dosimetry laboratories are used TLD readers. TLDs is a convenient means of monitoring whole body exposure to beta, X and gamma radiations. TLD appropriate for wearing periods of up to three months. This is according to the IAEA recommendation on Practical Radiation Technical Manual Individual Monitoring on 2004. Therefore, the proposed amendment to BCR No. 4 of 2013 in accordance with the categorization of activities justified in the Government of Indonesia Regulation is:

TABLE 4. PERIOD OF DOSE MONITORING FOR WORKER VS ACTIVITIES CATEGORY BASE ON GOVERNMENT REGULATION

Activities Category Base On Government Regulation	Period of Dose Monitoring for Worker
A	Once every 1 months
B	Once every 2 months
C	Once every 3 months

Key Words: radioactive material, radiation generator, and licensing system.

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190-CONSCIOUSNESS ANALYSIS ON SAFETY CULTURE IMPROVEMENT IN RADIATION FACILITIES IN JAPAN

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Recently, many of the accidents and incidents are caused not by the failures of the device or hardware used but by the failures of users. It is not easy for us to effectively and specifically countermeasure these causes because the target of the countermeasure is human. Understanding the current situation on radiation facilities is indispensable for developing effective measures. This study is mainly targeting radiation facilities in Japan such as universities, research institutes, and hospitals. This research on fostering a safety culture is divided into three steps. The first step aims to understand current opinions on the safety culture of radiation facilities in Japan. The second is to identify each characteristic for a group of facilities and the third is to suggest effective countermeasures for radiation facilities in the Japanese research and education sector.

In the first step, we conducted questionnaires on safety culture mainly to radiation safety managers at radiation facilities. This has two objectives: (a) Extracting keywords for fostering a safety culture at radiation facilities mainly for radiation safety managers; (b) Organize and analyse trends in opinions of radiation safety managers and radiation workers. A deeper level of information is needed for Objective (a). Therefore, an open-ended questionnaire was adopted. Objective (b) requires a lot of datas to analyse it. For this reason, a selective questionnaire was adopted.

Referring to the description of Nuclear Safety White Paper of Japan (2005), which summarized the status of nuclear research and development in Japan and overseas at that time by the Japan Nuclear Safety Commission, the open-ended questionnaire was constructed based on four items for fostering a safety culture. Four items are: creating an organizational culture that allows persons on front lines to work with pride and a sense of responsibility, need for top management commitment, need for dialogue between different groups and organizations through adequate communication, and maintaining a “questioning attitude” which an organization and an individual who belongs to it take. For each item, opinions were sought from three viewpoints: (1) current problems, (2) medium- to long-term improvement measures, and (3) long-term improvement measures. The results showed that 100% of radiation safety managers agreed with the "need for top management commitment " and "need for dialogue between different groups and organizations by adequate communication" . On the other hand, “creating an organizational culture that allows persons on front lines to work with pride and a sense of responsibility” and “keeping a questioning attitude within the organization and individuals belonging to it” need to be reconsidered for fostering a safety culture.

The first step of the questionnaire also showed trends based mainly on differences in the size and purpose of radiation facilities. There was a big difference in the tendency to recognize "accidents, incidents, troubles and failure cases" between small- and large-scale facilities. Many respondents from large-scale facilities answered that they could easily share the latest related information in their facilities. This is probably because large-scale facilities usually have existing systems and chances to share information relating accidents in contrast to small-scale facilities.

There are many types of facilities in various fields, and it is difficult to identify each characteristic for each radiation facility such as hospitals, universities, and pharmaceutical companies. This fact led us to the second step of the research. The data of the free-form questionnaire above was analysed again, focusing on the types of differences among various facilities. Co-occurrence Network Analysis by the text mining method was applied by linking the respondents' affiliation information to the answer. As a results, universities and hospitals tended to call for new systems to foster a safety culture such as evaluation methods for their management and organization of a professional group on specialized safety management and education. Sterilization and Pharmaceuticals would like to make improvements to the existing systems and organization currently in operation for safety management and education. A research laboratory responded differently comparing with another research institutes, and research laboratories tended to show all aspects of other radiation facilities.

In these previous two step analyses, questionnaires were focused on the opinions of radiation safety managers. The goal of the study is to grasp the latest radiation safety situation of Japan's facilities from the viewpoint of culture, and an additional questionnaire has been started. As the third step, we are exploring effective countermeasures for radiation facilities in the Japanese research and education sector. In addition, comparison of Japanese analysis results with overseas situations to clarify each feature and to understand the differences. This questionnaire was created based on the activity of IRPA TG HERT regarding the fostering of a safety culture in the field of research and education. The questions are to rank the effective approaches to fostering a safety culture. The 10 key items have been shown from the result of a survey in UK. The following are the representative keys for effective approaches to fostering safety culture listed in the questionnaire.

(1) Engagement of Management
(2) Appropriate Training
(3) Regular audit/inspection of radiation safety procedures/practices
(4) Appropriate management of radioactive materials and radiation generating equipment
(5) Appropriate appointment & use of Recognised Experts & Officers
(6) Management of staff doses
(7) Appropriate Incident handling
(8) Effective Communication
(9) Resources
(10) Professional Societies

The target of this questionnaire was set broadly from radiation safety managers to radiation workers and clerical staffs because safety culture is a problem that should be widely related to all members of the facility. Japan Society of Health Physics and The University of Tokyo have cooperated in this questionnaire. Comparing the results of these analyses with those from overseas could characterize the differences in thinking between Japan and other countries.

ACKNOWLEDGEMENTS

A part of the research is based on the project of Nuclear Regulation Authority of Japan in 2016. We received generous support from the following experts in this study; Dr. S. Higaki and Mr. N. Kosaka from Isotope Science Center, The University of Tokyo, Dr S. Kato from Yokohama University of Pharmacy, Dr K. Takahashi from Osaka University, Dr S. Futatsugawa from Japan Radioisotope Association. We would like to express our gratitude also to Task Group (2018-2019) for Fostering Radiation Safety Culture in Japan Health Physics Society.

192-CHARACTERIZATION OF MTS-6 AND MTS-7 TLD CRYSTALS FOR PERSONAL DOSIMETRY IN 241AMBE AND 137CS FOR NUCLEAR GAUGE OPERATORS

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Personal dosimetry in neutron fields is complicated because of the need to differentiate the gamma and neutron components of the dose. Combinations of ^{6}Li and ^{7}Li enriched crystals can be paired with an albedo holder to estimate both gamma and neutron personal equivalent dose, given that ^{6}Li is sensitive to slow neutrons, while ^{7}Li only to gammas [1]. An example of such crystals are MTS-6 and MTS-7, LiF doped with (Mg,Ti) -manufactured by TLD Poland-, and enriched in ^{6}Li and ^{7}Li, respectively.

A batch of MTS-6 and MTS-7 thermoluminescent crystals were characterized and type tested to be used for personal dose monitoring in mixed gamma and neutron fields. The whole neutron dosimeters consist of 4 crystals: 2 MTS-6 in positions 1 and 4, and 2 MTS-7 in positions 2 and 3, in a boron enriched plastic holder. Crystals in positions 1 and 2 are behind a thin window and can be used to measure Hp(0.07), while positions 3 and 4 are used for Hp(10). The TLD reader used was RADOS RE-2000, with a reading temperature of 350° during 15 seconds. They had been previously calibrated with ^{137}Cs and ^{241}AmBe sources to measure Hp(10) [3]. Additional type tests are presented. The variations in individual crystal characteristics (zero counts, individual sensitivity and crystal reproducibility) are presented in FIG. 1. The reproducibility of most crystals is below 3%, but most of the uncertainty comes from the zero counts determination, and special care should be taken with annealing procedures.

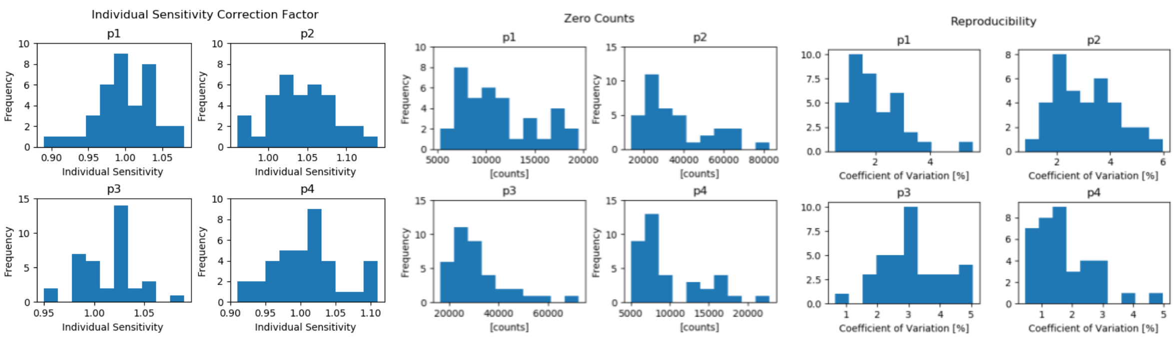

FIG. 20. Distribution of the individual crystal sensitivities, zero readings and coefficient of variation for all dosimeter positions (MTS-6: p1, p4, MTS-7: p2, p3). Individual sensitivities correct for variation in sensitivity of the crystals with respect to the batch used for reader calibration, and a lower value means the crystal has a higher response. The zero counts are the readings of the dosimeters after the signal has been erased by 2 successive readings. The reproducibility is the coefficient of variation of 3 successive readings after irradiation at 1 mSv of ^{137}Cs.

To determine the annealing procedure, the dosimeters were irradiated then read multiply times (FIG .2.). It was seen that 2 readings were enough to erase all of the origina signal. Because of the high temperature used for reading the crystals, reading them more than 2 increased the background counts and is not recommended. New crystals also tended to decrease the zero counts after a couple of irradiations before reaching stability. Angular dependence was tested with the crystals inside the holder, irradiating on a water slab phantom (30 cm x 30 cm x 15 cm) with a ^{137}Cs source (FIG. 2). There doesn't seem to be any trend in the variations observed, and average angular response is within recommended tolerances in IAEA No. RS-G-1.3. Fading was tested irradiating a batch of dosimeters to a dose of 10 mSv with the ^{137}Cs at reading them at different times after the irradiation. We can see the response quickly goes down after 30 days of storage at ambient temperature, and then starts to increase by an amount that is bigger than would be caused by the natural background. Monthly cumulated dose due to background was measured with a batch of 15 dosimeters and the average doses were found to be 0.13 mSv for MTS-6 and 0.20 mSv for MTS-7. Fading from a ^{241}AmBe irradiation, was found to be 20% and 7% (decrease) for positions 2 and 3, and an increase of the response of 19% and 26% was found for MTS-6 in positions 1 and 4, in a period of 30 days.

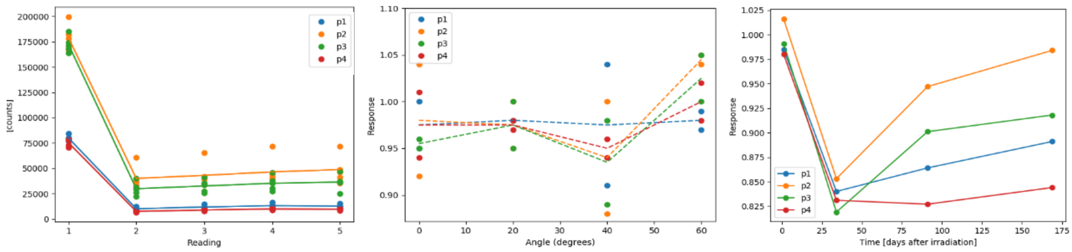

FIG. 2. Stability of the zero for successive readings, angular dependence, and fading with ^{137}Cs.

The linearity of the response with increasing dose was checked with the ^{137}Cs, between 0.10 mSv and 100 mSv. The response was found to be close to 1 for all cases, but the uncertainty for the smallest doses (0.10 -0.30 mSv) is very big (FIG. 3). The linearity was also tested with a ^{90}Sr irradiator (β), and the dose per irradiator unit was measured with 3, 10 and 30 irradiator units. The response decreases with increasing doses, although the irradiator units were found to be linear with dose response for tests with other LiF crystals. FIG. 3 (c) shows the constancy of reader calibration factor over a year.

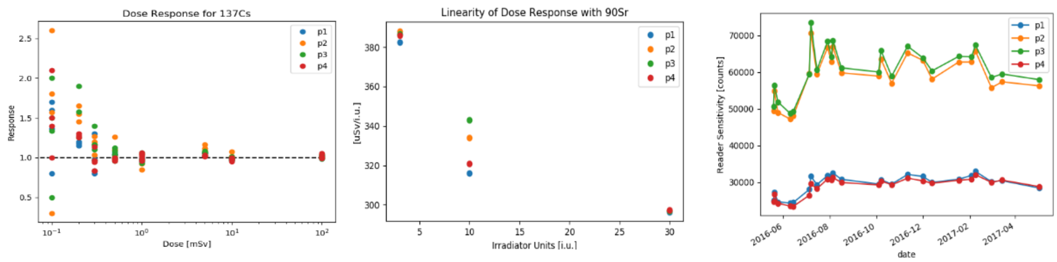

FIG. 3. Linearity of the response with dose for a 137Cs source, a 90Sr irradiator, and constancy of reader calibration factor over a year.

The methodology described by Hirning C.R. was used to calculate the lower limit of detection of the system. A value of 0.40 mSv was used for MTS-7 and 0.09 mSv for MTS-6, for Hp(10), and a value of 0.30 mSv for MTS-7 for Hp(0.07). Operators of nuclear gauges using ^{241}AmBe and ^{137}Cs were monitored for periods up to a year, the result is shown in table 1. It can be seen that the neutron dose could be almost as high as the gamma, as already shown in [2], even though in this case it is also mostly below the limit of detection.

TABLE 1. PERSONAL DOSE MONITORING FOR NUCLEAR GAUGE OPERATORS

Worker Dose	γ Hp(10) [mSv]	n Hp(10) [mSv]	γ Hp(0.07) [mSv]
Average	0.84	<LD	1.38
Maximum	1.76	1.49	2.49

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193-ASSESSMENT OF THE RADIOLOGICAL RISKS ASSOCIATED WITH THE PRODUCTION OF IODINE-131 BY THE TRIGA MARK II REACTOR, DURING THE TRANSITION FROM 80G OF TEO2 TO 150G

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AIMS

The objective of this post study is to assess, under normal and incidental working conditions, when the mass of irradiated target of TeO_2 used for the production of iodine 131 will be increased, the doses likely to be delivered to the workers under external exposure due to ionizing radiation. It will also make it possible to identify the hazards and to estimate the potential risks, for the implementation of appropriate prevention actions and to provide elements for the possible management of incidents.

This evaluation of external doses aims to provide the elements necessary for:

- Set up collective protection equipment and safety instructions;
- Delimit restricted areas (supervised, controlled, specially restricted areas prohibited);
- Define personal protective equipment;
- Determine the classification of personnel (A, B, unexposed);
- Choose the dosimetric techniques adapted to the exposure conditions and define the modalities of individual and ambient dosimetric monitoring.

More generally, this study will make it possible to verify compliance with regulatory dose limits and the application of the principle of optimization of radiation protection (ALARA) in the case of an increase in the mass of the target used for the production of iodine 131.

APPLICATION AND METHODS

The evaluation of the doses that operators could receive in this study is based on a numerical calculation using the following calculation codes:

- **ORIGEN2** (version 2.2): for calculating the activity of the target.
- **MICROSHIELD** (version 6.2): for calculating the flow of individual dose equivalent Hp(10) received by operators.

The doses related to external exposure delivered to workers are evaluated on the basis of knowledge of the characteristics of the radiation fields emitted at each work station (nature, flow, energy, spatial and temporal characteristics) as well as their work tasks.

THE SOURCE TERM

The source term obtained using ORIGEN2 version 2.2, is based on input data supplied by the UAS, the UCR and the UPR during the preparation of the safety file for the production experience of iodine 131 using the CNESTEN TRIGA reactor. We took into account all the major elements present in our studied source. The iodine activity is estimated at **6.35 Ci** using a target with 150g of TeO_2.

CALCULATION RESULTS IN NORMAL WORKING SITUATIONS

The risk assessment would be based on the different radiations emitted by all of the radioelements present in the TeO_2 target which contained 150g.The results below represent the virtual exposure levels obtained using

MICROSHIELD software (version 6.2), relating to the different operations carried out in the iodine 131 production process. For a practical assessment, 2 phases are considered:

— The irradiation phase;
— The transfer phase from the reactor to the production laboratory.

(a) The irradiation phase

Depending on the procedure adopted (wet) and at the end of irradiation and cooling, the dose rate at the level of the pool water surface is evaluated at: $D_{top\text{-}pool}$ (8m of water) = 1,176E-15 mSv/h.

(b) The transfer phase

This part concerns the sending of the target from the reactor module to the production laboratory module, the use of an armored container With 10cm of thick makes it possible to reduce the dose rate significantly. We can go from a flow rate of **8 Sv/h** to **0.67 mSv/h**.

CALCULATION RESULTS IN INCIDENTAL WORK SITUATION

In this case, we will prevent the blocking of the target during its passage through the sheath connecting the central sock to the re-entry of the transport container. The use of protective screens is inevitable in this case, to minimize the exposure of the operator concerned.

The unblocking intervention is carried out in accordance with the procedure pre-established by the reactor control unit for this purpose. For a good control of the operations, the distance from the operator must not exceed 1m to intervene quickly. The time factor in such situations is very important to reduce as much as possible the doses likely to be received by the operator responsible for this intervention. The calculation was carried out to estimate the equivalent doses taking into account the two variables (time and screens).

CONCLUSION

The work is a continuation of the initiated work at CNESTEN which aimed to increase the activity produced in Iodine 131 from a target of 150 g of TeO2. This work contributes in this perspective by assessing the radiological risks associated with this increase.

The study carried out within the framework of this work focused on the evaluation of the radiological risks, associated with the increase in mass of the target of TeO2 used for the production of iodine 131. The major risk which was taken into consideration in this study was the estimation of effective doses through the fair contribution of external exposure

The results obtained by the MicroShield calculation code show that the current protections implemented during the various production phases meet the normative and regulatory requirements in terms of radiation protection in the event of transition to the use of 150g of TeO2. A critical phase to take into account when removing the target during its passage through the sheath, when it is bare outside the reactor pool before its introduction into the armored transfer container. The equivalent dose rate during this phase is reduced by a factor of 3 after using an 8mm lead coating on the transfer sheath.

ABBREVIATIONS

UAS : Spectroscopic Analysis Unit.
UCR : Reactor Control Unit.
UPR : Radiopharmaceutical Production Unit.

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194-SPECIFICATION FOR RADIOLOGICAL PROTECTION OF NUCLEAR MEDICINE PRACTITIONERS

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BACKGROUND AND OBJECTIVE

The purpose of the study is to establish a recognized radiation protection standard. We recommend that this standard applies to radiological protection for practitioners of nuclear medicine in medical and health institutions.

METHODS

The relevant terms and definition have been defined. We have developed a series of the system and management requirements, and protection requirements for radioactive workplaces referenced to Chinese National GB 18871 basic standards for protection against ionizing radiation and for the safety of radiation sources.

RESULT

Medical and health institutions should be qualified to carry out nuclear medicine diagnosis and treatment. Management documents including the radiation work management system, post responsibilities, work procedures, operating procedures, protection specifications and emergency plans should be established. Nuclear medicine practitioners should have corresponding qualifications and receive training as required. Medical and health institutions should conduct personal dose monitoring and occupational physical examination and establish health management files for nuclear medicine practitioners. Protection requirements for radioactive workplaces were seen table 1 for radioactive workplaces category. Controlled area and supervised area should be set up in radioactive workplace category I and II. The channels for nuclear medicine practitioners and patients should be set up separately, and the entrance and exit of nuclear medicine diagnosis and treatment should be separated. Radioactive workplaces can be divided into diagnostic, therapeutic, and research workplaces in terms of functional settings independent ventilation system should be set up in the controlled area of radioactive workplace. Radioactive workplaces should be equipped with corresponding radiation protection supplies (see table 2) and first aid articles for radioactive contamination; instruments with functions to monitor external exposure doses and surface contamination. Relevant instruments should be regularly verified or calibrated according to regulations. A sanitation passage should be set up in front of the controlled area of the radioactive workplace, so that practitioners can wear protective equipments before entering the controlled area, and monitor and decontaminate before leaving the controlled area. The rooms, injection tables, fume hoods (packing cabinets) or biosafety cabinets of radioactive clinical sites and research sites shall adopt corresponding shielding schemes in accordance with the radionuclides type, energy and maximum used amount as required by GB 18871. After the end of daily radioactive operations (or before the start of the next day's work), radioactive workplaces and items should be monitored for radiation and surface contamination. Daily radiological monitoring records should be completed and the monitoring results should be analyzed in time. The removal of surface radioactive contamination should be performed in time. Protection requirements for radiopharmaceutical operations were showed as follow. Before radioactive operation, the practitioners should wear corresponding radiation protective equipment (see table 2) and a personal pen-type dosimeter. All the operating procedures and protection specifications should be strictly followed. Radiolabeling and packing operations of radiopharmaceuticals should be carried out in a fume hood (packing cabinet), biosafety cabinets or hot cell. If the needle stick injury occurs during the administration of radiopharmaceuticals, decontamination and medical treatment should be given immediately. Personal protective equipment is required when performing radiological diagnostic operations such as instrument quality control and calibration, patient positioning, and radioactive counts measurement. The disposal of radioactive waste should be implemented in accordance with GB 18871 and related regulations. Radiation waste container should be placed in the radioactive operation area and be classified and processed according to radionuclide half-life and type.

Table 1. Radioactive workplaces category

Category	Weighted activity of daily operating maximum radionuclide（MBq）
I	＞50000
II	50～50000

III	＜50

Table 2. Protective measures and equipment according to radiation energy and packing and administration of radiopharmaceuticals

Name	Radiation energy (radiation category)	Packing and administration of radiopharmaceuticals	Protective measures and equipment
^{99m}Tc	140 keV, Gamma	Intravenous injection administration	● Lead shields for injection table or car should be used. ● Lead syringe shield is advisable to use. ● Lead shielding (thyroid shields, lead apron) and lead glasses should be worn.
		Oral administration	● Lead shielding (thyroid shields, lead apron) and lead glasses should be worn.
		Inhaled administration	● Lead shielding (thyroid shields, lead apron) and lead glasses should be worn. ● Filtration protective face mask should be worn. ● Disposable medical mask should be worn by the patient.
^{131}I	610 keV, Beta; 364 keV, Gamma	Pharmaceuticals packing	● Shielding of at least 30mm lead equivalence should be used in packing system. ● Lead shielding (thyroid shields, lead apron), lead glasses and masks should be worn.
		Intravenous injection administration	● Lead shields for injection table or car should be used. ● Lead syringe shield is advisable to use. ● Lead shielding (thyroid shields, lead apron), lead glasses and masks should be worn.
		Oral administration	● Lead shielding (thyroid shields, lead apron), lead glasses and masks should be worn.
^{18}F and other positron-emitting nuclides	511keV, Gamma	Pharmaceuticals packing	● Shielding of at least 50mm lead equivalence should be used in packing system. ● Time protection should be paid attention to.
		Intravenous injection administration	● Lead shields for injection table or car should be used. ● Lead syringe shield is advisable to use. ● Time and distance protection should be paid attention to.
^{89}Sr	1.463 MeV, Beta	Intravenous injection administration	● Poly (methyl methacrylate) is advisable to use.
^{32}P	1.709 MeV, Beta	External application or interventional administration	● Poly (methyl methacrylate) is advisable to use.
^{186}Re	1.076 MeV, Beta; 0.939 MeV, Beta; 137 keV, Gamma	Intravenous injection administration	● Lead shielding (thyroid shields, lead apron) and lead glasses should be worn. ● Lead syringe shield is advisable to use.
^{188}Re	2.13 MeV, Beta; 1.98 MeV, Beta; 155 keV, Gamma	Intravenous injection administration	● Lead shielding (thyroid shields, lead apron) and lead glasses should be worn. ● Lead syringe shield is advisable to use.
^{177}Lu	497 keV, Beta; 208 keV, Gamma; 113 keV, Gamma	Intravenous injection administration	● Lead shielding (thyroid shields, lead apron) and lead glasses should be worn. ● Lead syringe shield is advisable to use.
^{125}I	35 keV, Gamma	Implantation in local administration	● Lead shielding (thyroid shields, lead apron) and lead glasses should be worn.

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195-NUCLEAR SCIENCE TECHNOLOGY AND RADIATION SAFETY IN MEDICAL APPLICATIONS-A MINI REVIEW

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This mini review aims to strengthen and pay attention to scientific knowledge of nuclear science technology and radiation safety in medical applications.

Since roentgen discovered X-ray in 1895 and Becquerel discovered radioactive uranium in 1896, it has opened the door to the peaceful use of human atomic energy. Atomic energy industry in China has experienced more than half a century of development, and it has been widely used in industry, agriculture, military, medicine, environment and other aspects. In recent years, nuclear technology is more and more widely used in medicine and accepted by people. It can be said that the diagnosis and treatment of clinical medical diseases cannot do without nuclear technology. At present, PET/CT, the most representative molecular imaging technology, has important clinical application value in the early diagnosis and differential diagnosis, staging and restaging, monitoring treatment response, implementation of biological target conformal precise radiotherapy plan and prognosis judgment, and it also has unique advantages in the diagnosis of cardiovascular and cerebrovascular diseases. Nuclear medicine is a subject of clinical diagnosis, treatment and biomedical research using nuclide and its labeled compounds. With the rapid development of modern medicine, nuclear medicine is also rapidly changing, especially the potential application of nuclide tracing technology in molecular imaging and molecular medicine has shown an important prospect. The molecular functional imaging of nuclear medicine can realize the early diagnosis of disease at the molecular level, and at the same time, it can make use of the spontaneous radiation of radionuclide and the biological effect of its marker to achieve the unique role of "cross fire" targeted treatment of pathological tissue. Undoubtedly, it will help Chinese medicine to adapt to the current medical model from the original diagnosis and treatment to prevention and health care, and to use modern molecular functional imaging technology to early-warning, monitoring and truly achieve early diagnosis and treatment of disease.

Using safe and non-invasive radionuclide tracer technology to target tumor therapy has been widely used in clinical, and more and more attention has been paid. Radionuclide iodine-131 [^{131}I] molecular targeting therapy for hyperthyroidism and differentiated thyroid cancer postoperative recurrence and metastasis, Strontium [^{89}Sr] therapy for bone metastasis and bone pain of malignant tumor, iodine-125 [^{125}I] particle source therapy for refractory malignant tumor have become routine clinical treatment methods and clinical diagnosis and treatment guidelines.

Nuclear technology is a "double-edged sword", which can bring great impetus to human progress. For example, the application of nuclear technology in medicine can benefit human health, but it can't be ignored that the radiation harm caused by nuclear radiation or ionizing radiation lead to nuclear radiation cannot be ignored. Therefore, we should balance the advantages and disadvantages, make a right choice, keep pace with the times, and conform to the long-term interests of human society. According to a report published by the United Nations Scientific Committee on the effects of atomic radiation, medical radiation is the main source of human exposure to ionizing radiation. Of all kinds of radiation caused by natural or artificial factors, 20% of the world's population is from medical radiation, while the proportion of medical radiation caused by all man-made factors is as high as 98%. In some developed countries, the large increase of medical radiation means that it has replaced the natural radiation source and become the most important way for human body to contact with radiation. X-ray diagnosis accounts for the largest share in all kinds of medical exposure. Recently, ICRP pointed out that the risk of cancer will increase with the increase of the number of radiation diagnosis. It is suggested that the radiation sensitive organs of the examinees should be shielded during the X-ray examination, and all unnecessary radiation should be avoided at the same time. Although the share of nuclear medicine in medical radiation using radionuclides and their labeled compounds for clinical medical disease diagnosis and basic medical research is far less than that of X-ray diagnosis, in recent years with the continuous development of nuclear medicine diagnostic drugs and instruments (SPECT/CT, PET/CT), its application is more and more available.

The widespread application of radionuclides has brought great benefits to human beings, such as targeted diagnosis and treatment of cardiovascular and cerebrovascular diseases and tumors, which benefits human health. However, due to improper use and management, the loss of radioactive sources will also cause radiation accidents, resulting in damage to property and life. Radioactive substances can be inhaled by breathing, absorbed into the body through skin wounds and digestive tract, causing internal radiation. γ-radiation could penetrate a certain distance and can be absorbed with external radiation injury to personnel. This is why it is necessary to standardize the diagnosis and treatment of radionuclide in nuclear medicine, especially the treatment of large dose of radionuclide iodine [^{131}I] for recurrence or metastasis of differentiated thyroid cancer patients after operation, which must be hospitalized in medical institutions with special protective conditions that meet the national requirements or standards.

At present, the main problems of medical institutions are as follows: (1) some hospitals do not pay enough attention to radiation protection work, and also do not provide enough financial and human support. Many patients who need high dose radionuclide treatment cannot be hospitalized for treatment without special protection ward, and the implementation of radiation protection rules and regulations is not strict enough. (2) Some hospital bulletin boards of radiation protection are only in form, so it is difficult for the examinee to understand the radiation hazards and how to use the protective equipment through the bulletin boards; the use of the protective equipment for the examinee is not optimistic. The patients who need to be treated do not know enough about nuclear radiation, and they are afraid or just hold "indifferent" attitudes.

Therefore, in order to strengthen the construction, reconstruction and expansion of radioactive workplaces, it is necessary to carry out environmental assessment (to ensure environmental safety), occupational disease preassessment and final effect assessment (to ensure the safety of occupational personnel), and at the same time do a good job in the use of nuclear technology and nuclear radiation safety and other popular science publicity. It's well known for us to ensure the application and development of nuclear technology in the field of medicine for the benefit of mankind.

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196-IMPLEMENTATION OF EFFECTIVE INCIDENT REPORT VIA QR CODE IN RADIOTHERAPY DEPARTMENT

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1. BACKGROUND

The incidence of radiotherapy errors may involve in every process, although the QA procedure and verification sheet are performed before and during the treatment. In addition, human error is also the main source of incidents [1-3]. The document's hard copy of the incident report is considered as the tool for reporting the event to the head of the department. However, it is no longer seen as an effective method because the documents are often lost, and the problem is not solved immediately. The risk management (RM) committee in our department was established in 2017 and has been organizing our department until now. The incident report has changed from hard copy to QR code since 2019 which has been reported directly to the RM committee and showed no loss of the data anymore. This study aims to observe the effectiveness of QR code for incident reports compared with the hard copy through the questionnaire.

2. METHODS

The RM committee member consists of a radiation oncologist, medical physicist, radiation technologist, nurse, and RT assistance. The roles of the RM committee are to review and analyze all incident errors which occurred in the department. The weakness and drawbacks of the radiotherapy process were subject to be collected and discussed further before it was launched as the standard operation procedure (SOP) for the staff to prevent the same incident occurs again. The regular meeting for the team was held every three months, except for the incident effect on the patients, where the meeting would have been held immediately to provide a quick solution. The QR code was posted in staff working areas. The incident was reported directly to the committee once the submission report via the QR code was completed. The summary of SOP was announced to all staff in the regular meeting. To observe the satisfaction level for both QR code and hard copy report, the questionnaire was distributed to 10 staff randomly. The questionnaire was categorized into five levels representing the range from extremely satisfied to dissatisfied (score 5 to 1). The scored parameters were easy reporting, the time duration of report submission to the RM team, time duration of RM team response, patient's data credential, and data storage and management.

3. RESULTS

The satisfaction of staff in QR code and the hard copy is explored in Fig. 1. The QR code showed a higher score than a hard copy with a significant difference, student pair t-test analysis, p-value < 0.05 for all parameters. Most of the parameters in QR code were filled into extremely satisfied, except the time duration of the RM team response. The meeting time between the regular meeting of the RM committee and the radiotherapy department team was different, where the RM committee meeting was held every three months and the radiotherapy department team meeting was held every two months. Therefore, the solution to this improper time was returning the SOP to each group of staff either via the sub-group meeting or official line group messages, before regular staff meeting was held. On the other hand, the use of QR code may also prevent misunderstanding of the incident due to personal handwriting. All incidents were encouraged to be reported by QR code which was directly shown to the RM team. This procedure is very useful in the complexity of modern radiotherapy planning and delivery systems.

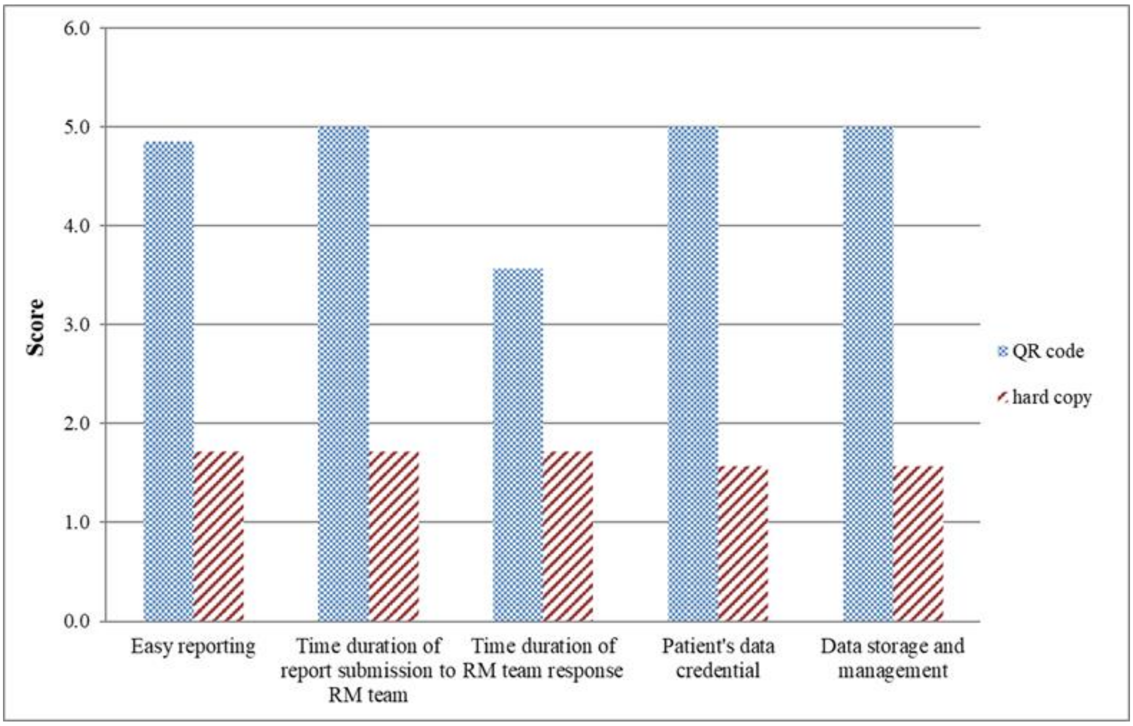

Fig. 1 The satisfaction score between QR code and hard copy incident reporting

4. CONCLUSION

The QR code is the appropriate method to report the incidents. The setup team that responds to work out the errors, however, is also challenging. Great teamwork is obviously important to succeed in the outcomes.

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197-PRACTICAL APPROACH FOR RADIOLOGICAL PROTECTION IN AVIATION-HOW CAN WE ASSESS THE COSMIC RADIATION EXPOSURES OF FLYING INDIVIDUALS?

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IAEA has pointed out the need of assessment of aviation doses received by aircraft crew as a general safety requirement. Also, ICRP recently recommended that frequent flyers as members of the public should know self-assessment individual doses and, as appropriate, adjust their flight rosters. However, it is hard for some of the flyers to get their accurate aviation doses through repeated calculations using the detailed information of flight profiles. The method for individual monitoring should be easy to perform, requiring no professional skills and elaborative work. This task could be achieved by either or mixture of two approaches by using 1) an easy-to-handle program for route dose calculation and 2) a portable electronic dosimeter having long operational time. In this report, possible practical approaches for aviation dose monitoring are discussed and some promising results related to a dosimeter-based approach are presented.

BACKGROUND

As the number of long-haul flights increases, exposures of both aircraft crew and passengers to cosmic radiation has also continued to increase. With this concern, IAEA has stated that a framework including a reference level and a methodology for the assessment and recording of aviation doses received by aircraft crew should be established where it is deemed to be warranted by the regulatory body or other relevant authority [1]. Also, ICRP recently published a strategy for radiological protection in aviation [2] which recommended that some groups of the passengers who fly frequently, so-called “frequent flyers”, should be managed in a manner similar to those occupationally exposed on a case-by-case basis, while their exposures be considered as the public exposure. As concrete measures, those flyers are encouraged to assess their individual doses on a yearly basis and, as possible, adjust the frequency of their flights. It is hard, however, for some of the flyers to adequately implement those measures without full support of their employers, since they have limited time and knowledge/interest that are needed to follow the radiological protection issue.

POSSIBLE APPROACHES FOR AVIATION DOSE MONITORING

For the busy and indifferent flyers, the following two approaches would work for management of their individual doses: 1) to use an easy-to-handle program for route dose calculations and 2) to use a robust, portable electronic dosimeter having a long operational time up to 1 year. In regard to the former approach, it is desirable that the program is simple and easy to use for anyone, ideally running on a smart phone, so that a flyer could calculate the aviation doses of own flights anytime. For example, it is preferable that he/she could get the route dose of a certain flight by just selecting the date and the names of departing and arriving airports. The concern of this approach is that any passengers have no detailed information on their flight profiles (i.e., changes of latitudes, longitudes & altitudes with time) and they would get a representative dose value of a certain route. As result, the annual individual dose obtained as the sum of several route doses would have a considerable uncertainty.

The latter approach using a personal dosimeter could overcome the limitation of the former one, if it could monitor the annual aviation dose of a flyer without additional efforts. For this purpose, the dosimeter should have a function of automatic dose-rate recordings for at least one year and also have a good portability, i.e., robustness, small size and light weight. In addition, it should not induce electromagnetic wave interference which is strictly prohibited to happen onboard aircraft. Needless to say, low costs for purchasing and maintenance are desirable.

VERIFICATION OF THE DOSIMETER-BASED APPROACH

With the thought above, the author has investigated possible applications of a recently developed electronic personal dosimeter, called “D-shuttle” (Chiyoda Technol Cooperation) which satisfy the above conditions required for the 2nd approach. The D-shuttle is coupled with a silicon PIN diode detector and lithium coin battery and has a good portability with the small size (H68 × W32 × D14 mm^3) and light weight (23 g). It records automatically both hourly dose rates and cumulative dose as $H^*(10)$ for a long period up to 400 days without exchange of the internal battery. The recorded data are occasionally read out with a dedicated reader.

As example, the results obtained by using several pieces of D-shuttle in round-trip flights from Tokyo/Haneda, Japan to Munich, Germany on 23rd and 27th June 2019 are shown in Table 1 [3]. During both the outward and inward flights, 3 dosimeters were placed each at following locations: (*a*) in a hand luggage under the passenger seat, (*b*) on the chest of the author and (*c*) in a suitcase placed in the cargo area. Though the location b (on the chest) is the most appropriate for any personal dosimeter, it is expected that most of the flyers would keep the dosimeter in location *a* (in a hand luggage) or location *c* (in a suitcase). The hourly $H^*(10)$ rates measured by D-shuttle were compared with the calculations obtained by JISCARD EX [4].

TABLE 1. Comparisons of the $H^*(10)$ value obtained by D-shuttle [3] and the total $H^*(10)$ values calculated with a route-dose calculation program JISCARD EX [4] for four long-haul flights from Japan to Germany; three dosimeters were used at each location.

Flight route	Departure date (yyyy-mm-dd, JST)	Flying time [h]	Calculated $H^*(10)$ [μSv]	Measured $H^*(10)$: average ± std. dev. [μSv]		
				In hand luggage (*a*)	On chest (*b*)	In suitcase (*c*)
Haneda to Munich	2019-06-23	11.1	79.0	23.1±0.1	24.9±0.5	17.9±0.3
Munich to Haneda	2019-06-27	10.6	72.1	19.7±1.1	18.7±0.7	18.2±0.3

The response variation among the dosimeters at a specific location was below 6%. While, as expected, the ratios of the measured $H^*(10)$ values in a hand luggage to the calculated doses ranged from 25% to 29%. The $H^*(10)$ values obtained with D-shuttle on the chest were nearly the same with those in a carry bag placed under the seat in front of the passenger. While, it was found that the values obtained in a suitcase placed in the cargo area were significantly different from those on the chest, which was attributable to the different shielding conditions between the cargo area and the passenger cabin. Accordingly, D-shuttle placed in the passenger cabin could be used to monitor the aviation dose of a flyer, if a precise correction factor to convert the measurement to the total $H^*(10)$ or effective dose value was well defined for the specific flight.

ACKNOWLEDGEMENTS

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198-JUSTIFICATION PROCESS OF NON-MEDICAL HUMAN IMAGING FOR DETECTION PURPOSE IN CORRECTIONAL INSTITUTION FOR DRUGS CRIME IN INDONESIA

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Justification of facilities and activities that give rise to radiation risks must yield an overall benefit [1]. Human imaging using radiation for the detection of concealed objects that can be used for criminal acts that pose a national security threat shall be justified only by the government [2]. Cases in Indonesia, the use of radiation examination for non-medical human imaging to fight against drug crime in the correctional institution needs to be justified. Drugs crime is regulated in Act No. 35 of 2009 [3] whereas correctional institution is regulated in Act No. 12 of 1995 [4]. Justification process the use of Body Scanner for detection purposes in correctional institutions related to drug crimes in Indonesia will be the main focus. Discussion on drug crime in the correctional institution and the government's effort to improve the detection and examination system in the correctional institutions using x-ray for non-medical human imaging. The results of National Survey in 2017 conducted by National Narcotics Board (BNN) Data and Information Research Center in collaboration with the Health Research Center University of Indonesia obtained the number of socio-economic cost losses due to drugs abuse in Indonesia during 2017 reached to eighty-four trillion Rupiah (around six billion and three hundred million Dollar US) [5]. The following is the prevalence of drug abusers in Indonesia in 2011, 2014, 2017, and 2019.

TABLE 1. THE PREVALENCE OF DRUGS ABUSERS IN INDONESIA [6]

Year	2011	2014	2017	2019
Prevalence	2.23 %	2.18 %	1.77 %	1.80 %

The drug inmates and prisoners in Indonesia are around 63% of the total inmates and prisoners [7]. This leads to the problem of illicit drug trafficking in correctional institutions. Two problems identification of drug crime in the correctional institution: illicit drug trafficking in correctional institutions and prisoners who control illicit drug trafficking in the outside correctional institution [8]. Based on this situation, the Ministry of Law and Human Rights, who in charge of the correctional institution, takes action to tighten access in correctional institutions by adding smuggling detection equipment using Body Scanner as non-medical human imaging equipment. In 2017, the Ministry of Law and Human Rights requested licensing from BAPETEN to use the Body Scanner in correctional institutions. The Body Scanner will be installed in correctional institutions and detention centers. BAPETEN responded to the request by formed an assessment team from the Regulatory Assessment Center on Radiation Facilities and Radioactive Sources to conduct a radiation safety assessment on the use of Body Scanner because the Body Scanner has never been used in Indonesia. The internal report [9] BAPETEN by the Regulatory Assessment Center published as an internal report in late 2017 to support Directorate of Licensing BAPETEN taking decisions.

In March 2018, BAPETEN held a coordination meeting with the Ministry of Law and Human Rights in the discussion of follow-up requests for Body Scanner Licensing. In the same year, the President issued Executive Order No.6 of 2018 concerning the National Action Plan for the Prevention and Eradication of Abuse and Illicit Traffic in Drugs and Drugs Precursors which is one of the instruction point 'B.1.h' is the Supervision of Drugs Correctional Institution [10]. Ministry of Law and Human Rights also issued a Ministerial Regulation No. 35 of 2018 on Revitalization of Correctional Institution stipulated drugs correctional institution classified as medium security correctional institution [11].

Refer to GSG-5 Justification of Practices Including Non-Medical Human Imaging [12], BAPETEN considers; Benefit: Officer and suspecting visitors (except pregnant woman, children or people who can be checked by manual examination) checked by Body Scanner from smuggling communication devices and drugs can prevent illicit drugs trafficking; Detriments: Subject examined by Body Scanner will get exposure radiation; Evaluation: Radiation dose from Body Scanner examination for suspecting visitors is relatively low; Decision: BAPETEN decided to justify the use of the Body Scanner and give conditional license to be used in correctional institutions. The conditional license requires the licensee must establish an examination procedure with an exception for pregnant women, children, or people who can be checked by manual examination. The Body Scanner license has been deciding to be classified into Category B according to Government Regulation No.29 of 2008 on Licensing of the Use Ionizing Radiation Sources and Nuclear Material [13]. Category B Licensing Technical Requirements: Operational Procedure, Specification Technical, Radiation Protection Tool (e.g. TLD for Operator, Surveimeter), Radiation Protection Programme, Worker's Health Surveillance, and Radiation Protection Officer.

BAPETEN justification use of Body Scanner as smuggling detection equipment used in correctional institutions by taking into account technical and non-technical issues, in this case, is a fight against drugs. In the future, BAPETEN plans to draft BAPETEN Chairman Regulation (BCR) on Justification to be the reference in conducting justification and review existing justifications[14] including reviewing non-medical human imaging justification refer to the document SSG-55 Radiation Safety of X-Ray Generators and Other Radiation Sources Used for Inspection Purposes and for Non-medical Human Imaging.

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199-DELIBERATE BIAS IN SAFETY ASSESSMENTS HELPING OR HURTING STAKEHOLDER CONFIDENCE?

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Safety assessments provide valuable input to support regulatory decisions regarding options costing as much as hundreds of millions of dollars or more. Performance of natural and engineered systems is considered far into the future. Such calculations involve contextual and technical uncertainties that must be managed. One approach to managing uncertainty involves the use of intentional pessimistic (cautious) bias in assumptions intended to err on the side of overpredicting consequences. However, excessive bias can damage stakeholder confidence by producing results that imply greater consequences than would reasonably be expected and can pose challenges to determine what results are or are not used as the basis for determining compliance with a regulatory requirement.

Including some pessimistic bias is an unavoidable reality for efficient and defensible implementation of a safety assessment. Table 1 includes a number of examples of biases that can and are often included in a safety assessment. However, any inherent pessimistic bias that is included in assumptions related to regulatory and technical aspects of a safety assessment needs to be acknowledged and communicated to properly risk-inform such decisions.

Policy-related biases are deliberately included in the context for safety assessments (e.g., dose limits, dose response assumptions, land use, exposure pathways). The intent of these biases is to provide safety margins that build confidence that reasonable assurance can be achieved. Such safety margins are often not specifically addressed or identified in the quantitative analysis or when making decisions. A primary example is the safety margin associated with dose constraints for disposal that are set at a fraction of the annual dose limit of 1 mSv, which is established based on the linear no threshold hypothesis. The annual dose limit is also well below the average annual dose in the United States (6.2 mSv). Although such biases are commonly included, without proper communication they can significantly impact decision making and public confidence, especially when interpreting "what-if" analyses and uncertainty analyses that are routinely expected for modern safety assessments.

Technical capabilities of safety assessment models and overall implementation are continually improving. This has resulted in more realistic modeling approaches and substantial improvements to sensitivity and uncertainty analysis. Nevertheless, pessimistic bias is still used for specific assumptions for the models as a means of managing some of the more challenging uncertainties. Common areas for pessimistic bias include assumptions for the performance of engineered barriers and evolution of the performance over time. "What-if", sensitivity and formal uncertainty analyses often explore a wide range of potential assumptions, including ignoring specific barriers, to explore the influence on the safety assessment results (e.g., dose or concentrations in different media). When pessimistic bias is introduced in such calculations, it can pose challenges for interpretation relative to decisions regarding compliance. For example, if a dose calculated for an exploratory simulation using pessimistic assumptions is greater than the applicable constraint, does that imply non-compliance? Such calculations are essential to improve the understanding of system behavior, but can yield some extreme results. Such results can be especially challenging when communicating with stakeholders.

Further guidance may be beneficial to clarify how to account for pessimistic assumptions when interpreting results from "what-if" simulations, sensitivity analyses or the tails of uncertainty analysis distributions, especially in the context of comparing them with a regulatory standard for compliance. There is a need for proper perspective to avoid making very costly decisions that are biased by pessimistic assumptions in a safety assessment.

TABLE I. EXAMPLES OF BIASES BUILT-IN TO SAFETY ASSESSMENTS (ADAPTED FROM [1])

Performance Objectives and Measures	– Performance objectives are a small fraction of annual average background exposures – Performance objectives for disposal facilities are a fraction of the 1 mSv/yr annual public dose limit recommended by ICRP and IAEA – Linear no threshold model used to establish the public dose limit assuming impacts observed at high doses are extrapolated to lower doses – Inadvertent human intrusion is assumed to occur after a hypothetical loss of memory and institutional controls
Protective Barriers	– Assumptions about effectiveness of engineered barriers – Assumptions about timing for changes in effectiveness of engineered barriers – Assumptions about rate of changes in effectiveness of engineered barriers – Not accounting for other barriers to exposures (records, memory, land use controls, government ownership, etc.)
Receptors and Habits	– Assumptions about loss of institutional controls and memory of facility – Assumptions about the timing and location of public exposure (peak concentration) – Assumption about habits (residential, subsistence farmer) – Inadvertent intruder ignores indications of waste
PA Methods and Models	– Likelihood of exposure and scenarios typically not considered – Often opt to not include potentially beneficial processes rather than defend assumptions – Limited quantification of uncertainties and use for decision-making – Pressure to consider "What-If" cases that represent extremes

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200-ARRANGMENTS FOR TERMINATION OF A NUCLEAR OR RADIOLOGICAL EMERGENCY

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The International Atomic Energy Agency (IAEA) Safety Requirements publication Preparedness and Response for a Nuclear or Radiological Emergency (IAEA Safety Standards Series No. GSR Part 7) [1] requires governments to ensure that arrangements are made for the termination of a nuclear or radiological emergency, taking into account the need for a resumption of social and economic activity. This was an important lesson learned from response to Fukushima Daiichi Accident. In order to support Member States in their initiative for preparing such arrangements IAEA developed a new General Safety Guide "Arrangements for the Termination of a Nuclear or Radiological Emergency" (IAEA Safety Standards Series No. GSG-11) [2], co-sponsored by ten international organizations and published in 2018, which provides guidance and recommendations on this subject matter.

GSG-11 provides guidance on what is the primary objective of the termination of an emergency. It elaborates on the prerequisites that should be met to facilitate official declaration of emergency as ended and to define activities and criteria applicable to the transition from emergency exposure situation to either an existing exposure situation or a planned exposure situation. The general prerequisites, that should be achieved, include but are not limited to the following: source is under control and exposure situation is well understood and stable; radiological situation is well characterised; exposure pathways are identified and doses to the public are evaluated; reassessment of hazards is performed; existing emergency arrangements are reviewed and new ones are established to the extent possible for timely termination of emergency; consideration is given to the management of radioactive waste; interested parties are consulted; non-radiological consequences are taken into account; assessment against established generic criteria and operational criteria is performed and compliance between estimated residual doses and reference level is demonstrated. The primary objective and identified prerequisites guide the development and implementation of protection strategy and therefore prescribe the arrangements required for the termination of a nuclear or radiological emergency.

Fig. 1 schematically presents the logic behind the process of transition from emergency exposure situation to the planned or existing exposure situation and subsequent termination of an emergency.

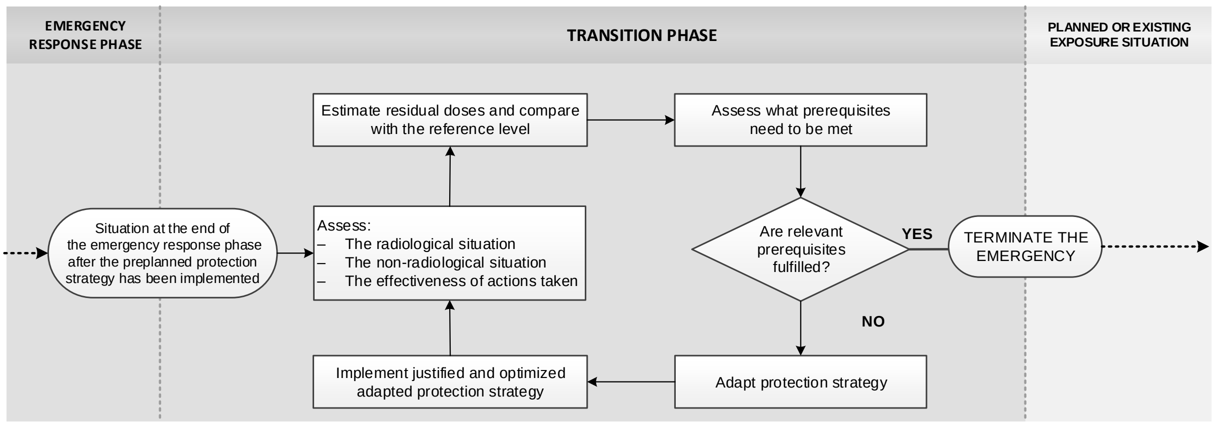

FIG. 21. Transition from emergency exposure situation to planned or existing exposure situation

As required by GSR Part 7 [1] and elaborated in GSG-11 [2] arrangements required for the transition phase of a nuclear or radiological emergency should be put in place during the preparedness stage and include the following:

(a) Establishment of legal and regulatory framework and clear assignment of authority, roles and responsibilities.
(b) Hazard assessment that is aimed to identify facilities and activities, areas and locations where (1) implementation of protective and other response actions may be warranted in the event of emergency and (2) for which actions may be needed to enable the termination of emergency.
(c) Establishment of protection strategy to achieve all the goals of emergency response provided in [1] and arrangements for adaptation of preplaned protection strategy to a new one suited to real conditions.
(d) Arrangements for characterisation of the exposure situation that includes development of a comprehensive monitoring strategy and means for assessing radiological consequences.
(e) Arrangements for protection of emergency workers and helpers, including identification of emergency workers who will be involved in the transition phase; establishment of guidance values for restricting their exposure and arrangements for dose assessment and recordings and provision of medical support.
(f) Establishment of arrangements for long term medical follow-up and provision of mental health and psychological support in the aftermath of a nuclear or radiological emergency for reducing adverse psychological and societal consequences of emergency.
(g) Arrangements for managing radioactive waste and other conventional waste generated by emergency itself and from the emergency response actions.
(h) Arrangements for communication and consultation with the public and other interested parties. As part of such arrangements interested parties should be identified, mechanism and means for their involvement and consulting at different phases of emergency should be established.
(i) Compensation of victims for damage. Strategy for compensations should be developed in advance.

All the prerequisites listed above should be supported by establishment of the adequate infrastructural arrangements, as indicated in requirements 20 to 26 of [1].

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201-COMMUNICATING RADIATION RISKS: EXPERIENCES AND BEST PRACTICES *EFFECTIVE PUBLIC COMMUNICATION IN NUCLEAR OR RADIOLOGICAL EMERGENCIES*

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The public's and the media's heightened perception of the risks posed by radiation can be effectively clarified by providing plain language responses to the question "Am I safe" based upon current IAEA Safety Standards, IAEA-EPR guidance and IAEA plain language materials, as well as IAEA public communication training, including social media simulation. The plain language definition of "safe" and supporting materials are provided in IAEA guidance [1] to support effective public communication for emergency preparedness and response. Further, in emergency response, easy to understand graphics as included below, and other public communication arrangements, help to place radiological hazards – if any – in context [2].

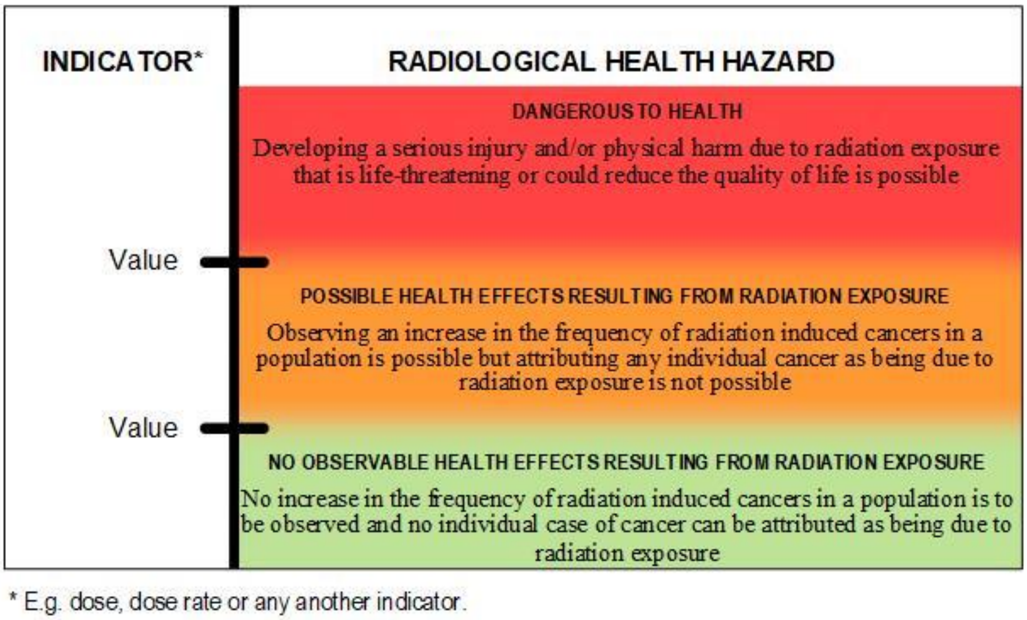

FIG. 1. System for putting radiological health hazards in perspective

The vast majority of the world's population are unlikely to be exposed to radiation levels that present a health concern. Rather they are exposed to so-called low doses of radiation falling within the range of doses resulting from artificial and natural background radiation. After many decades of research, UNSCEAR notes that epidemiological studies have not shown that people who are exposed to doses below 100 mSv delivered over a period of many years have an increased frequency of disease or adverse health consequences [4,5].

Following a nuclear or radiological emergency, the vast majority of the global population will likely be exposed to no measurable or only minimal increases in radiation dose levels. Although the radiological risks remain unchanged, and are at these levels of no safety significance, millions of people will nonetheless experience increased anxiety. Increased anxiety is recognized as a public health safety risk [2]. Empathetic and easily understandable communication that helps people place radiological health hazards in an accurate context is essential in reducing anxiety. Communicators therefore need to be prepared to swiftly communicate understandably and empathetically in the event of any elevation in the average radiation dose levels, no matter how slight. If the relevant national radiation protection authorities and/or emergency response authorities do not consider additional protective actions to be necessary, then these communities can be described as "safe" [1].

In specific areas, where radiation is dispersed and deposed following an accident or malicious act, annual dose levels can be temporarily increased. In keeping with international best practice [2], the relevant authorities will undertake preparedness measures in advance of such an event to proactively engage the affected citizens to learn their concerns. In the event of an accident or malicious act, the authorities' preparedness will support their delivery of easily understood information that helps place the health consequences in context and to provide protective instructions to prudently reduce radiation exposure without disrupting daily life. During and following an emergency, some communities may be exposed to higher radiation levels, above the peak dose levels associated with everyday radiation exposure. These communities will require a focused and intensive communication on radiological safety and the projected risks [2]. Communicators will need to deliver frequent, accurate and easily understandable advice on prudent personal protective actions. Residents in affected areas will be urged to follow the safety and protective instructions issued by the relevant authorities and maintain routine medical monitoring. Specific communication guidance is provided in [1], including charts designed for plain language explanations of health consequences and protective actions for vulnerable populations and the general public. In addition, [2] provides a simplified chart to swiftly, easily and unambiguously communicate the risk level in any given location.

The generic criteria for protective actions in emergencies foresee urgent protective actions if the thyroid or foetal effective doses are expected to exceed 50 and 100 mSv respectively in the first 7 days after an emergency commences. Similarly, these urgent actions will be ordered if the effective dose is expected to exceed 100 mSv in the same time period. If the equivalent dose for the thyroid or foetus is expected to exceed 100 mSv over the course of one to 12 months, protective actions are also foreseen. The same guidance applies if the effective annual dose is expected to exceed 100 mSv. [3]

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202-ENSURING NATIONAL COMPATIBILITY ACROSS MULTIPLE REGULATORY AUTHORITIES

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The primary regulatory authority for the safety and security of radioactive materials in the United States is the Nuclear Regulatory Commission (NRC). However, due to increased state interest in nuclear energy and technologies in the 1950s, the Atomic Energy Act was amended in 1959 to allow the states to enter into agreements with the Atomic Energy Commission (now the NRC) where the federal government discontinues and individual states assume regulatory authority over certain radioactive and nuclear materials and their uses within the state's border. As of 2020, there are 39 Agreement States who, along with the NRC, co-regulate the civilian sector and oversee the National Materials Program (NMP). Although the NRC has discontinued its authority within these individual states, the NRC maintains oversight authority to ensure that each state maintains program elements[3] that are adequate to protect public health and safety and that are compatible with NRC requirements. The NRC retains the authority over activities designated vital to protect the common defense and security of the United States and may only allow Agreement States to conduct some of these activities by special agreement. The framework for the NMP is described in the Commission's "Agreement State Program Policy Statement."

An Agreement State radiation control program is compatible with the NRC's regulatory program when the State program does not create conflicts, duplications, gaps, or other conditions that jeopardize an orderly pattern in the regulation[4] of radioactive material included in the agreement (e.g., agreement material), on a nationwide basis. Compatibility focuses primarily on the potential effects of a State's action or inaction either on the regulation of agreement material on a nationwide basis or on other jurisdictions. The concept of compatibility does not directly address matters of health and safety within a particular Agreement State; such matters are addressed directly under adequacy. However, many program elements for compatibility may affect public health and safety; therefore, they also may be considered program elements for adequacy. Further, basic radiation protection standards and other program elements that address or cross jurisdictional boundaries (between two Agreement States or NRC and an Agreement State), although important for health and safety within the State, should ensure uniformity of regulation nationwide for compatibility purposes.

The characterization process for individual program elements is categorized into one of six criteria. The process used to assign a health and safety or compatibility category to an individual program element is described in NRC's Management Directive 5.9, "Adequacy and Compatibility of Program Elements for Agreement State Programs," and is based upon a series of questions to identify a particular compatibility or health and safety significance. Each program element is tested by asking the series of questions below in the order given. The answers to these questions determine the compatibility category for each NRC program element or identify it as having particular health and safety significance.

- Question (1): Do the essential objectives of the program element address a regulatory area reserved solely

[3] Program elements are any component or function of a radiation control regulatory program, including regulations, staffing, licensing and inspection procedures, incident response, and/or other legally binding requirements, imposed on regulated persons that contributes to implementation of that program.

[4] The physical protection of radioactive materials is included in the regulatory structure for all agreement materials. That is, safety and physical protection are fully integrated.

to the authority of the NRC? If the response to the question is "yes," the category designation is "NRC." If the response to the question is "no," then proceed to Question (2).

- Question (2): Do the essential objectives of the program element address or define a basic radiation protection standard as defined by the policy statement or is it a definition, term, sign, or symbol needed for a common understanding of radiation protection principles? If the response to this question is "yes," the compatibility category designation is "A." If the response to the question is "no," then proceed to Question (3).
- Question (3): Do the essential objectives of the program element address or define an issue that crosses jurisdictional boundaries? If the response to this question is "yes," the compatibility category designation is "B." If the response to the question is "no," then proceed to Question (4).
- Question (4): Would the absence of the essential objectives of the program element from an Agreement State program create a conflict, gap, or other condition which impacts the orderly regulatory pattern? If the response to this question is "yes," the compatibility category designation is "C." If the response to the question is "no," then proceed to Question (5) to determine whether the program element should be identified as having particular health and safety significance.
- Question (5): Would the absence of the essential objectives of the program element from an Agreement State program create a situation that could directly result in exposure to an individual in excess of the radiation protection standards found in compatibility category A? If the response to this question is "yes," the program element is not required for purposes of compatibility, but is identified as having particular health and safety significance, then category H&S applies. If the response to the question is "no," then the program element must be identified as compatibility category "D."

Under this process Agreement States will need to adopt some program elements that are nearly identical to those established by the NRC but may have flexibility to be more restrictive for other program elements. With regard to Compatibility Category A and B, the Agreement State program elements have to be essentially identical to those of the NRC. The term essentially identical means the interpretation of the text must be the same regardless of the version (NRC or Agreement State) and often means the wording is the same. For Compatibility Category C and Category H&S, Agreement State program elements have to meet the essential objective of the program element. The essential objective is the action that is to be achieved, modified, or prevented by implementing and following the regulation or other program element. In some instances, the essential objective may be a numerical value (e.g., restriction of radiation exposures to a maximum value) or it may be a more general goal (e.g., the conduct of a radiation survey). For example, the dose limits for radiation workers are identical across all NMP jurisdictions and designated as Compatibility Category A; however, the dose limit allowed for license termination are designated as Compatibility Category C which allows Agreement States to be more restrictive than NRC's limit of 250 microSieverts per year. Consequently, there is no variation in the dose limits for radiation workers, but the license termination dose limits vary across the NMP.

The determination of the adequacy or compatibility criteria for program elements, particularly regulations, begins during their development. Agreement States are given multiple opportunities during the rulemaking process to comment on the regulations and the compatibility designations; they work with NRC in the development of the proposed regulations. Licensees and the public can also comment on the proposed compatibility designations. Whereas the Agreement States often want flexibility in the designations (Compatibility Category C or Category H&S) to allow the state to take into account local requirements imposed on them (but not the NRC), licensees often desire more uniformity nationally (Compatibility Category B) to minimize the differences in requirements as licenses move from one jurisdiction to another. Licensees that operate at field locations or have offices across the country in several states face the challenge of meeting similar but not identical regulatory requirements because of compatibility. In recent years, the tension between flexibility and uniformity in compatibility designations were seen during the development of the current training and educational requirements for medical providers and for certain reporting requirements. Agreement States, licensees and professional organizations were active in pursuing their positions on compatibility. It is not until the NRC Commission approves the final regulations are the compatibility designations finalized.

For the NMP to work effectively, the 40 regulatory agencies that comprise the NMP need to have their regulations compatible. Once the NRC adopts amendments to its regulations, the Agreement States have 3 years to adopt compatible requirements. This is a challenge for many Agreement States given the wide range of processes and levels of government that approve state regulations. The time periods for Agreement States to adopt compatible regulations range from 6 months to 3 years with a one-year period being typical. Agreement States will often wait 1 to 2 years and bundle the NRC amendments into one rulemaking package. Over the last 15 years, more Agreement States have made the decision to adopt part or all of their radioactive materials regulations by reference to the NRC regulations to overcome their lengthy internal processes for adopting the required NRC regulations. The other approach frequently taken by Agreement States is the adoption of legally binding requirements through the amendment of the impacted licenses. This is a particularly effective method when the NRC amendment only impacts a class of licensees. The NRC is required by the Atomic Energy Act to periodically review each Agreement State program to ensure that they maintain an adequate and compatible program. This is accomplished through the Integrated Materials Performance Evaluation Program (IMPEP). IMPEP reviews are conducted every 4 to 5 years and has found that nearly all Agreement States maintain compatibility with NRC requirements. In cases where an Agreement State has not updated their regulatory requirements in a timely manner, the impact on the NMP is minimal.

203-THE APPLICATION OF THE GRADED APPROACH TO THE REGULATION OF RADIOACTIVE SOURCES IN THE UNITED STATES

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Development and oversight of regulations that account for the overall risk posed by radioactive materials requires integration of safety and security programs. Implementing a graded approach to both safety and security allows the U.S. Nuclear Regulatory Commission (NRC), along with our Agreement State partners, to ensure adequate protection without unnecessary burden. This graded approach allows the United States (U.S.) to realize and take full advantage of the benefits of the various. uses of radioactive materials, while ensuring consistent and sustainable safety, security, and control of radioactive sources – from allowing exemptions from regulations for specific items, through sources and uses for which prudent management practices or existing safety requirements are sufficient, to the highest activity sources and practices that deserve the tightest control.

The U.S. framework relies on various. safety - including operational experience - and security analyses - including threat, vulnerability, and consequence - to determine the appropriate framework and requirements for each circumstance. These analyses form the basis for the graded requirements applicable to all civilian radioactive and nuclear material in the U.S. For example, sources that present minimal to negligible hazard, such as smoke detectors and gunsights, are exempt from licensing entirely for the end-user, while still other sources, such as those in robust devices, are subject to registration and notifications of transfers to the regulatory agency.

Although the basic radiation protection standards and licensing standards for all licensee types can be found in 10 CFR Part 20 and Part 30, respectively, additional requirements have been developed and are enforced for users based upon both use type and the activity of the radioactive source or sources. As an example, for the majority of licensed civilian radioactive and nuclear material in the U.S. the physical protection measures can be found in 10 CFR Part 20, in paragraphs 1801 and 1802. Although, in short, these paragraphs state only that licensees must secure their material while in storage and that it must be under constant surveillance while in use – they do not specify exactly what means a licensee must use to accomplish those objectives. Similarly, while the radiation protection and licensing requirements provide dose limits and activity levels as a basis for things such as transport labeling and financial assurance for decommissioning, the regulations are, again, generally performance-based and do not specific exactly what means a licensee must use to accomplish these objectives. Thus, licensees must develop processes and procedures that are subject to inspection, to meet all the applicable objectives and regulations.

For other licensed material, and for specific modalities of use, additional (not replacement) requirements apply. An example of this is a well logging licensee who possesses a category 3 americium-241/beryllium source. This licensee must comply with the safety and security requirements in 10 CFR Part 20 that includes the establishment and maintenance of a radiation protection program that will limit occupational and public exposures to radiation to below regulatory limits. Additionally, the licensee must comply with requirements in 10 CFR Part 30 to maintain their license in good standing including allowing for regulatory approval to amend a license. For this licensee type compliance with further requirements in 10 CFR Part 39 that are specific to well logging operations also apply. These requirements include additional labeling, security, and transport requirements as well as operational and administrative requirements including for operating from field stations or temporary jobsites.

Another example is that of a radiopharmacy who also must comply with the same safety, security, and licensing requirements in 10 CFR Parts 20 and 30, but due to their special operational methods, must also comply with the additional requirements for a distributor of radioactive materials in 10 CFR Part 32 as well as comply with the training requirements for a nuclear pharmacist in 10 CFR Part 35. Notable in this example is that the nuclear pharmacist, in order to be designated on the radioactive materials license must have successfully completed an accredited pharmacy educational program and be a licensed pharmacist – neither the accreditation nor the pharmacy license is overseen by the NRC, but are examples of cooperation across professional specialties and regulatory programs.

For sources or aggregated quantities of radioactive material that the U.S. has determined to be risk-significant, that is that meet or exceed the category 2 threshold, further requirements of 10 CFR Part 37 must be implemented by the licensee to ensure additional physical protection. An example of this situation is an industrial radiography licensee who possesses a camera (or multiple cameras) containing a category 2 iridium-192 source who must comply with 10 CFR Parts 20 and 30 (like the previous examples), the additional specific safety requirements for radiography in 10 CFR Part 34 such as personnel wearing alarming dosimetry, and further security requirements in 10 CFR Part 37 such as providing extra barriers for their mobile source(s).

Radioactive sources also pose extremely varied potential detriment to human health. The NRC's graded approach also takes this into consideration by requiring more extensive, and sometimes more prescriptive, requirements for radioactive sources or activities that pose higher risk of injury. At extreme ends of this spectrum are the case of a residential smoke detector that contains a very low activity americium-241 source and a panoramic irradiator that can utilize a very high activity of cobalt-60 for sterilization purposes. The smoke detector, while it's manufactured and distributed by a licensed entity, is destined for possession, use, and ultimately disposal, by a member of the public. This person has no requirements imposed on them because the possession and use is of a product that is exempt from licensing. Contrasting that situation with the panoramic irradiator that has robust safety and security measures applicable to its operation, including for construction and functioning that will prevent a person from entering the irradiation vault but will also fail-safe and shield the radioactive sources if a person does manage to enter during an irradiation cycle.

These examples demonstrate the commitment to maintaining adequate protection of workers, the public, the environment, and the security of the U.S., but also demonstrate a recognition of the differences among the large population of users of radioactive and nuclear material within the U.S.

The NRC has conducted multiple efforts to evaluate this framework in the past 3 years and continues to develop guidance and revise policies to ensure that modern technology is accounted for in the regulatory framework. To ensure that security measures are adequate the NRC has performed a review of the security framework for Category 1 and 2 quantities of radioactive materials[5], and later performed a reevaluation of the security and licensing framework for Category 3 radioactive sources[6]. Technology and use-specific evaluations with a safety focus have also been conducted to ensure that requirements that were often drafted decades ago can accommodate current technology and current industry and public need. For example, in 2016 the NRC evaluated the use of combination dosimetry devices for use in radiographic operations. The evaluation and decision to allow the use of such devices in the U.S. to comply with existing requirements was provided to licensees.[7] Similarly, concern with release of patients who have undergone treatment with radiopharmaceuticals, considering both healthcare practices in the U.S. as well as doses that may be received by members of the public or the patient's family and friends, has also generated an evaluation by the NRC into the patient release criteria. This activity was submitted to the NRC Commission for final direction after receiving a high degree of interest from a wide variety of sectors including the healthcare profession, patient advocacy groups, and insurance/Federal health agencies.[8]

The U.S. regulatory framework, designed to rely on a graded and integrated safety and security approach, changes deliberately with careful consideration and allows the population of the U.S. to realize the full potential positive uses of the radioactive sources, while also protecting the public health and safety, common defense and security, and the environment from the potential harmful effects of radiation.

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[5] See "10 CFR Part 37 Program Review" on NRC's public website at https://www.nrc.gov/security/byproduct/10-cfr-part-37-program-review.html

[6] See "Category 3 Source Security and Accountability Re-Evaluation" on NRC's public website at https://www.nrc.gov/security/byproduct/category-3-source-security-accountability-reevaluation.html

[7] See "NRC Regulatory Issue Summary 2017-06: NRC Policy on Use of Combination Dosimetry Devices During Industrial Radiographic Operations"

[8] See "SECY-18-0015: Staff Evaluation of the U.S. Nuclear Regulatory Commission's Program Regulating Patient Release After Radioisotope Therapy"

205-RADIOLOGICAL EMERGENCY PREPAREDNESS TRAINING AND CAPABILITY DEVELOPMENT IN THAILAND

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Abstract

In the year 2000 an unsecured category I radioactive source has been stolen from the temporary storage area caused serious injuries and three deaths in Thailand. The radiological emergency team was secured a medical radiotherapy Cobalt-60 source within 24 hours after notification. Thailand Institute of Nuclear Technology (TINT) is the major utilization of nuclear technology in Thailand and had been successfully upgraded the physical protection systems for a 2 MW nuclear research reactor and category I of radioactive sources based on performance objectives found in IAEA document INFCIRC/225/Rev.5 and had a radiation emergency response plan. Every year the TINT's radiological emergency response team has been trained to increase the capability and interoperability between radiological assessors and response local agencies team in the nation to response quickly and effectively when an emergency happens. The training objective was ensure adequate and coordinated among local agencies with appropriate procedures, people and equipment to deal effectively and tested the effectiveness of manuals and lessons learned incorporated. The training subjects were for, and conducting, first aid, decontamination monitoring of people and equipment, incident command system and interoperability between radiological assessors and response local agencies and performed the table top and field exercises. The topics of training has been included in emergency responder training procedures from simple awareness and recognition to technical knowledge of the materials, detection and identification capabilities, self-protection, medical effects, and countermeasures to overall public and environmental safety. In 2019, the five days training course there were 223 personnel from many local agencies participated. The lessons learned incorporated to ensure continuous improvement of local emergency agencies capabilities and testing of the effectiveness of radiation emergency plan manuals and procedures of TINT and enhanced the relevant local agencies' relationships and managing radiation exposure to workers and the public. Finally the training had been improved integration of the radiological response into the incident command system and related inter-agency interoperability and the next radiological emergency training need to be included and communicated to members of the public and responders in all media.

10. INTRODUCTION

In the year 2000, there was a radioactive accident in Thailand, an unsecured category I radioactive source has been stolen from the temporary storage area. The 15.7 TBq (425 Ci) of Co-60 medical teletherapy was removed from the shielding and left unattended in the scrap metal yard for almost one month caused ten serious injuries and three deaths as shown in Fig. 1. The unshielded source was retrieved by the radiological emergency team and now storing at the waste storage facility at the Thailand Institute of Nuclear Technology (TINT). Details of this radiological accident which occurred in 2000 are described in the IAEA document as The Radiological Accident in Samut Prakarn, Thailand, IAEA, 2002. There were many lessons learned and improved from this accident such as laws, regulations of radiation safety, physical protection system (PPS) and radiological emergency preparedness and response plan in the nation. The objectives of the upgrading and strengthening the performance of the PPS for the safety of radioactive sources that will endanger human health and provide the preventive maintenance and extended warranty for the installed equipment. The security enhancements are based on performance objectives found in IAEA Nuclear Security Series No. 11, Security of Radioactive Sources, and *Nuclear Security Series No. 13,* Recommendations on Physical Protection of *Nuclear* Material and *Nuclear*

Facilities (INFCIRC/225/Revision 5), as well as the laws and regulations for radiation safety from the Office of Atoms for Peace (OAP) as a national regulator. The tasks were to implement security enhancement at the category I of radioactive sources in the waste storage facility and agriculture product irradiation facility as well as the category III of a 2 MW nuclear research reactor.

11. POLICY OF RADIOLOGICAL EMERGENGY PREPAREDNESS AND RESPONSE

For national level, Thailand has developed national nuclear and radiological emergency plan and integral operational disaster plan for nuclear and radiological disaster and the action plan is obligation for licensees to establish their own facility emergency plan – to comply with Nuclear Energy for Peace Act 2016. Licensee has to initially respond following radiation protection plan, and has to immediately notify competent officer and cooperate with the officer. In the case of the emergency situation expand to be public damage, local and national competent officers under the national prevention and mitigation act 2007 will have authority to respond to the situation. TINT has a radiation emergency response team every year this team has been trained to increase the capability and interoperability between radiological assessors and response local agencies team in the nation such as regulator, firefighter, police, military, local provincial etc.

The training objective was ensure adequate and coordinated among local agencies with appropriate procedures, people and equipment to deal effectively and tested the effectiveness of manuals and lessons learned incorporated. The subjects of the five days training were for, and conducting, first aid, decontamination monitoring of people and equipment, incident command system and interoperability between radiological assessors and response local agencies and performed the table top and field exercises.

12. RADIOLOGIAL EMERGENCY TRAINING, EXERCISE AND CAPACIBILITY DEVELOPMENT

The 2019 TINT’s radiological emergency response training was for, and conducting, first aid, decontamination monitoring of people and equipment, incident command system and interoperability between radiological assessors and response local agencies and performed the table top and field exercises. The topics of training has been included in emergency responder training procedures from simple awareness and recognition to technical knowledge of the materials, detection and identification capabilities, self-protection, medical effects, and countermeasures to overall public and environmental safety. There were 223 personnel from many local agencies participated during five days training course.

13. CONCLUSIONS

The lessons learned incorporated to ensure continuous improvement of local emergency agencies capabilities and testing of the effectiveness of radiation emergency plan manuals and procedures of TINT and enhanced the relevant local agencies’ relationships and managing radiation exposure to workers and the public. Finally the training had been improved integration of the radiological response into the incident command system and related inter-agency interoperability and the next radiological emergency training need to be included and communicated to members of the public and responders in all media.

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206-DETERMINATION OF DOSE CONSTRAINTS FOR IODINE-131 PRODUCTION AT THAILAND INSTITUTE OF NUCLEAR TECHNOLOGY

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ABSTRACT

Thailand Institute of Nuclear Technology (Public Organization), or TINT, has a core mission related to research and the provision of nuclear and radiation technology. One of TINT's services is Iodine-131 production for thyroid diagnosis and therapy. In the production process, the workers have potential risk of both internal and external exposure. Therefore, these exposures must be assessed and controlled for the safety of workers. This study aims to determine the dose constraints for workers in Iodine-131 production process. The total effective dose (TED) of workers collected during 2017 to 2019 from two sources of exposure (1) External exposure using optically stimulated luminescence (OSL) badge (2) Internal exposure calculated from airborne monitoring. The dose constraints in accordance with history of exposure and an established criteria are proposed.

RESULT AND DISCUSSION

The analysed data of external exposure ($H_p(10)$), internal exposure (Committed Effective Dose, CED) and total effective dose (TED) for Iodine-131 radioisotope production workers recorded during 2017 to 2019 are illustrated in Table 1. The highest exposure observed were 6.86 mSv/y, 2.30 mSv/y and 8.89 mSv/y for $H_p(10)$, CED and TED, respectively. However, the value of total effective dose was far below the legal annual dose limit of 20 mSv. The dose constraints were determined at the 80th percentiles of the data which is a reasonable criteria. The proposed dose constraints for $H_p(10)$, CED and TED are 2.95 mSv/y, 2.30 mSv/y and 5.04 mSv/y, respectively.

TABLE 1. IODINE-131 ANNUAL DOSE RECEIVED BY RADIOISOTOPE PRODUCTION WORKERS DURING 2017 TO 2019

	Number of dose records	Average annual dose (mSv)	Minimum annual dose (mSv)	Maximum annual dose (mSv)	Proposed annual dose constraint (mSv)
$H_p(10)$	15 -18*	2.10	0.00	6.68	2.95
CED	15 -18*	1.44	1.06	2.30	2.30
TED	15 -18*	3.54	1.06	8.89	5.04

**15, 17 and 18 workers in the year 2017, 2018 and 2019, respectively.*

In 2014, H. Piwowarska-Bilska, et al. (H. Piwowarska-Bilska, M. Nowak, M.H. Listewnik, P. Zorga, B. Birkenfeld, 2014) studied the dose constraints for diagnostic medical departments' worker from 2008 to 2011 and 2009 to 2011. The exposure data were collected using chest badges equipped with Kodak film type 2 in the DNM, and TLD badges in the DDIIR. The group of workers were nurse, technician and medical doctor. The result showed that, the dose constraints for this group of workers were between 1 mSv/y to 2.5 mSv/y depend on the position of the worker. Moreover, there also has the similar studied of V. Kamenopoulou, et al. (V. Kamenopoulou, G. Drikos and P. Dimitriou, 2001) in 2001. The authors studied the dose constraints in the medical sector by collecting dose data from 1996 to 1999 using film badge (Film Kodak Type 2, Holder NRPB type). The group of workers were radiology, cardiology, surgery, nuclear medicine and tele-brachytherapy in different department and specialty. The result showed that, the dose constraints for this group of workers were between 1.5 mSv/y to 6.7 mSv/y depend on their department and specialty. However, only data of external exposure were considered in these two studies.

The concept of dose-constraint-like-instruments also applied in nuclear power plant for example, Krško NPP in Slovenia, the dose constraint due to external radiation is 15 mSv/y for category A workers and 6 mSv/y for category B workers and the dose constraint due to internal exposure is 0.2 mSv/y. (Dose constraints - Dose constraints in optimisation of Occupational Radiation Protection and implementation of the Dose constraint concept into Radiation Protection regulations and its use in operators' practices, 2011) Forsmark NPP in Sweden the planned annual dose shall not exceed 10 mSv and no internal contamination exceeding 0.3 mSv. (Dose constraints - Dose constraints in optimisation of Occupational Radiation Protection and implementation of the Dose constraint concept into Radiation Protection regulations and its use in operators' practices, 2011)

From the above researches mentioned, preliminary discussions can be made about determining the dose constraint mainly depends on the type of workplace and the type of work. The dose constraint must be consistent with the exposure pathway of the worker and this should be the value that can be used as a precaution in order to prevent radiation exposure beyond the legal dose limit.

CONCLUSION

Based on this study, the proposed dose constraints for Iodine-131 radioisotope production worker were 2.95, 2.30 and 5.04 for $H_p(10)$, CED and TED, respectively. It is a useful tool to controlling the Iodine-131 radioisotope production process to be safe for workers. It is also beneficial for investigation process when there is an occurrence of abnormal exposure.

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207-THE MEASUREMENT OF LEAD-210 ACTIVITY IN HUMAN SKULL WITH A HPGE DETECTION SYSTEM

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Studies have shown that the high incidence of lung cancer in miners is related to the Radon-222 progenies cumulative exposure. For miners have decades of service under mines, it is not easy to obtain the Radon-222 progenies cumulative exposure they have received during their decades of work, while the method of measuring Lead-210 activity in skull with HPGe detection system is a feasible way.

The objective of this paper is to study the feasibility of using HPGe detection system to measure Lead-210 activity in skull of miners, and to establish a method to estimate cumulative exposure to Radon-222 progenies base on Lead-210 activity in skull. 7 miners from coal mines in the southwest of Beijing had been studied with a HPGe detection system consist of double detectors and a low background shielding room. Before the measurement of miners, the HPGe detection system were calibrated by two different methods: Firstly, it was calibrated with a skull phantom with Lead-210 radioactive source, the calibration results are shown in TABLE 1;Then it was calibrated with MCNP, the calibration results are shown in TABLE 2. The former method was used to obtain the calibration factor with Lead-210 distributed only on the bone surfaces, and the latter method was used to obtain the calibration factor with Lead-210 evenly distributed in the bone volume. The relationship between calibration factors and facial tissue thickness was considered in both calibration methods.

TABLE 1 CALIBRATION FACTORS OBTAINED BY A SKULL PHANTOM WITH LEAD-210

facial tissue thickness， cm	calibration factor， cps/Bq
0	$(1.89\pm0.40)\times10^{-2}$
0.4	$(1.43\pm0.28)\times10^{-2}$
0.8	$(1.17\pm0.24)\times10^{-2}$
1.2	$(1.09\pm0.27)\times10^{-2}$
1.6	$(1.00\pm0.28)\times10^{-2}$

TABLE 2 CALIBRATION FACTORS OBTAINED BY MCNP DIGITAL PHANTOM

facial tissue thickness， cm	calibration factor， cps/Bq
0	(1.61±0.30)×10-2
0.4	(1.49±0.34)×10-2
0.8	(1.39±0.37)×10-2
1.2	(1.31±0.30)×10-2
1.6	(1.25±0.38)×10-2

7 miners'measurement results and calculated Lead-210 retention in skull are shown in TABLE 3, the calibration factors of miners when calculating were different because of their different physical characteristics.

TABLE 3 MINERS'MEASUREMENT RESULTS AND CALCULATED LEAD-210 RETENTION IN SKULL

miners	Net counts under 46.5 keV's peak， cps	calibration factor,cps/Bq	Lead-210 Retention in skull， Bq
No.1	$(1.18\pm0.03)\times10^{-2}$	$(1.39\pm0.37)\times10^{-2}$	$(2.10\pm0.02)\times10^{1}$
No.2	$(1.59\pm0.03)\times10^{-2}$		$(2.83\pm0.03)\times10^{1}$

No.3	$(1.48\pm0.03)\times10^{-2}$	$(1.31\pm0.30)\times10^{-2}$	$(2.79\pm0.03)\times10^{1}$
No.4	$(6.60\pm0.3)\times10^{-3}$		$(1.25\pm0.01)\times10^{1}$
No.5	$(9.30\pm0.3)\times10^{-3}$		$(1.83\pm0.02)\times10^{1}$
No.6	$(7.90\pm0.3)\times10^{-3}$	$(1.25\pm0.38)\times10^{-2}$	$(1.56\pm0.02)\times10^{1}$
No.7	$(1.35\pm0.03)\times10^{-2}$		$(2.66\pm0.03)\times10^{1}$

To verify the feasibility of the HPGe detection system, the same measurements were conducted for 4 public people as control group with no history of exposuring in sites with high level Radon-222 concentrations. The control group's measurement results and calculated Lead-210 retention in skull are shown in TABLE 4.

TABLE 4 CONTROL GROUP'S MEASUREMENT RESULTS AND CALCULATED LEAD-210 RETENTION IN SKULL

Control person	Net counts under 46.5 keV's peak, cps	Lead-210 Retention in skull, Bq
C1	(0.02±0.3)×10-3	(0.01±0.01)×101
C2	(9.72±0.3)×10-3	(1.73±0.02)×101
C3	(8.78±0.0)×10-3	(1.66±0.02)×101
C4	(5.96±0.3)×10-3	(1.18±0.01)×101

The measurement result turned out to be that the Lead-210 activity in the skulls of all 7 miners was higher than MSA(minimum significant activity), and 4 of which were higher than MDA(minimum detectable activity). Lead-210 activity in the skulls of 2 public people's were higher than MSA, and all of which were lower than MDA.

The conclusion was that it is feasible to use the HPGe detection system to measure Lead-210 activity in the skull of miner. With the measurement result, the Radon-222 progenies cumulative exposure of miners were retrospective studied according to the Lead-210's biokinetic model recommended by ICRP 130, ICRP 67 and others necessary parameters,which was described in another paper.

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209-MIODOSE: USER-FRIENDLY SOFTWARE TO ASSESS DOSES FOR INTERNAL CONTAMINATION AND TO OPTIMIZE INDIVIDUAL MONITORING

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

MIODOSE software, developed in collaboration between IRSN and Orano, allows the estimation of internal dose from a single or a series bioassay measurements. It also helps to choose the most appropriate individual monitoring programme to the exposure conditions by comparing the minimum doses detectable by different programmes.

MIODOSE offers the following functionalities (Figure 1):

- Managing intakes through different pathway (inhalation, ingestion, wound),
- Managing intakes from several radionuclides,
- Managing different types of bioassay (activity excreted in urine or faeces, activity retained in the whole body or lungs),
- Integrating DTPA chelation treatment in dose assessment,
- Integrating ICRP biokinetic and dosimetric models,
- Advising user to choose the most suitable biokinetic parameter through a database of radionuclide chemical forms referenced by ICRP,
- Assessing internal doses by fitting the model predictions to the activity measurement results,
- Checking the fit quality,
- Presenting a user interface guiding the user step by step,
- Integrating the new biokinetic and dosimetric models of the ICRP (OIR series) as soon as they become regulatory obligation.

Most of these functionalities can be combined, *e.g.* one can consider multiple pathways, multiple radionuclides, different kind of measurement in a single analysis.

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

This software allows, by integrating uncertainty:

- estimating the minimum dose detectable by a routine monitoring program from available information on physico-chemical forms of the handled material, on the level of activity at the workplace, and on the detection limits of the techniques available to measure incorporated radionuclides (Figures 2 and 3);

— assessing the committed effective dose following an intake incident, along with its associated uncertainty, from measured retained and/or excreted activities.

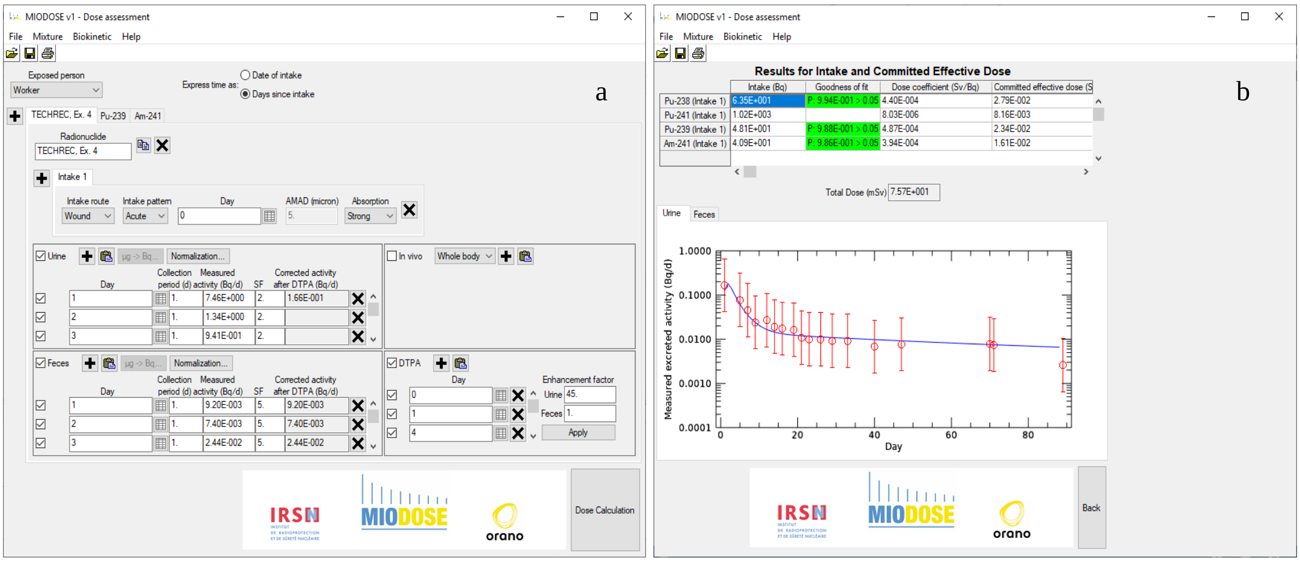

FIG. 22. Interfaces a) to collect measurement values and exposure conditions; b) to present intake and dose assessment results

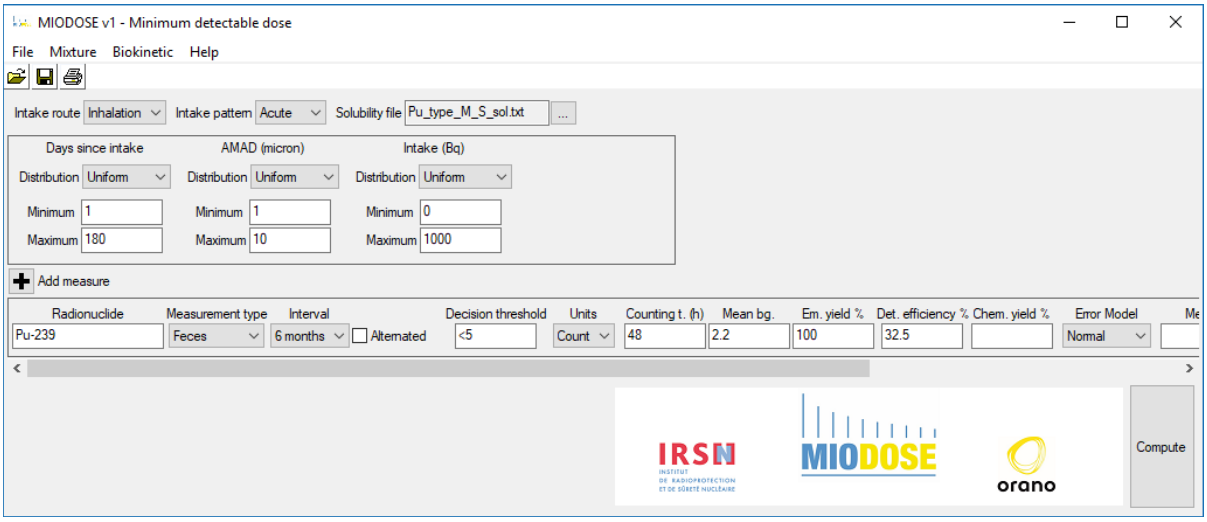

FIG. 23. Interface to collect uncertainty on exposure conditions, activity measurements and monitoring parameters

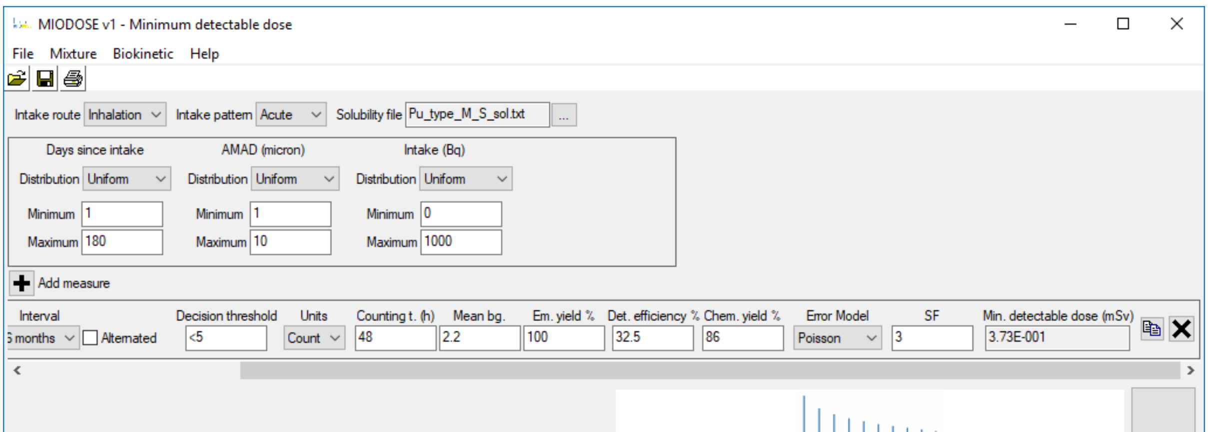

FIG. 24. Interface presenting the minimum dose detectable for the specified monitoring programme

213-NEUTRON DOSE ASSESSMENT OF THE PHILIPPINE NUCLEAR RESEARCH INSTITUTE SSDL - NEUTRON LABORATORY USING ALBEDO OSL DOSIMETERS

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The Department of Science and Technology-Philippine Nuclear Research Institute (DOST-PNRI), through the Radiation Protection Services Section (RPSS), provides calibration services of radiation monitoring instruments using the Secondary Standards Dosimetry Laboratory (SSDL) [1]. Currently, the PNRI-SSDL is extending its capabilities to calibrate neutron monitoring instruments with the recently established Neutron Laboratory (NL). The laboratory was setup by refurbishing an old facility which has an existing structural shielding. In 2019, characterization and standardization were done for a neutron calibration field of a bare Californium-252 source in PNRI-SSDL [2]. Personnel that will be involved in the calibration of neutron monitoring instruments are at risk to occupational neutron exposure. At the same time, the space where the NL is now housed was originally designed for a different purpose. Thus, it is necessary to assess the workplace conditions of the NL to help ensure the safety of personnel during conduct of calibrations [3].

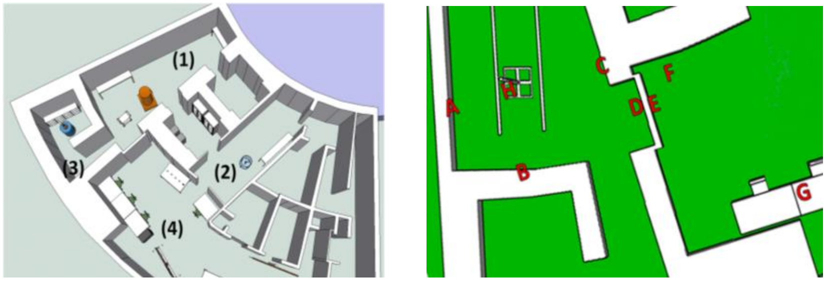
Currently, albedo optically stimulated luminescence dosimeters (OSLD) are readily available in the Philippines and are being offered as part of the individual monitoring services of PNRI-RPSS. Further studies on capabilities of albedo OSLD for neutron detection are also being done [4]. Although OSLD cannot replace area monitors, they can be tools in confirming the different radiation levels in a facility where neutron radiation is present [3]. The presentation aims to show the results of the neutron dose assessment and the derived dose schedule; for both members of the public and RPSS personnel, using albedo OSLD measurements through ambient dose monitoring in the NL's irradiation room and control room.

The NL has two standard bare sources: Am-241 with an activity of 1.85 GBq as of March 2019 and Cf-252 with an activity of 200 MBq as of April 2018. The neutron irradiation parameters on the experiment are listed in *Table 1*. The neutron source used is the bare Cf-252 source from Frontier Technology Corporation (FTC) model 100s with an initial mass of 10.3 μg and an initial source strength of 2.37×10^7 $neutrons/sec$.

TABLE 1. NEUTRON IRRADIATION PARAMETERS FOR ALBEDO OSL DOSIMETERS

Irradiation Conditions	**Date: August 28, 2019**
Source	Bare Cf-252
Emission Rate	1.65×10^7 cm^{-2} s^{-1}
Expected Dose	1 mSv
Irradiation Time	4 hours 47 mins
Albedo OSLD	Landauer's InLight Basic – OSLN dosimeter

Albedo OSLD, labelled "A" - "G", were positioned around the walls of the NL. Figure 1a shows a diagram of the NL, while Figure 1b shows the locations of albedo OSLD. Albedo OSLD A-C were located inside the irradiation room, while D was pinned to the door. OSLD E-G were in the control room. Locations for F and G are important because these are where RPSS personnel stay when irradiation is on-going.



A) Diagram of the Neutron Lab in PNRI-SSDL *B) Diagram showing the location of albedo OSLD*

FIG. 25. Schematic diagram of the PNRI-SSDL's Neutron Laboratory where (1) Irradiation Room, (4) Control Room.

Based on the assessment result in Table 2, both members of the public and radiation workers are not safe within the irradiation room when the Cf-252 source is 'ON', because the annual accumulated doses computed for 1 hour each working day are higher than regulatory limits; which are 1 mSv/yr and 20mSv/yr for the public and radiation workers respectively [1, 3]. However, in locations F and G where RPSS personnel stay during 'ON' position of the Cf-252 source, the maximum annual neutron dose is 17.6 mSv, which is lower than the annual dose limit for occupational exposure [1, 3]. This can be considered background radiation level, thus these locations are safe whether the Cf-252 source is 'ON' or 'OFF'. The dose assessment shows that the NL is safe to use for conducting calibration of neutron monitoring instruments given that proper radiation protection measures such as proper shielding, lessening time spent in the irradiation room when the source is 'ON', and keeping the doses ALARA are continuously enforced. The results may serve as support to the radiation protection program of neutron facilities in the country. The study is limited to neutron dose assessment only since the expected hazards are mostly from neutron sources, as it would give greater dose to personnel. Although no gamma dose measurements were done, gamma doses from neutron sources may be estimated as stated in Table XXI V of the IAEA's Safety Report Series No. 16: Calibration of Radiation Protection Monitoring Instruments.

TABLE 2. SUMMARY OF THE ALBEDO OSL DOSIMETER MEASUREMENTS FOR NEUTRON DOSE ASSESSMENT

Albedo OSLD Location	**(OSLD) Net** *H*(10)*	**Dose Rate**	***Number of hours allowable per day**	
Units	*mSv*	*mSv/hr*	**Public**	**Radiation Worker**
A	1.72	0.36	0.01	0.25
B	1.69	0.35	0.01	0.26
C	1.70	0.35	0.01	0.26
D	0.80	0.17	0.03	0.53
E	0.89	0.18	0.03	0.51
F	0.02	0.00	> 8	> 8
G	0.04	0.01	0.45	> 8

ACKNOWLEDGEMENTS

The authors would like to acknowledge DOST – PCIEERD and DOST-GIA OneLab Project for funding. This facility's neutron reference detector system was traceable to primary standards laboratory through International Atomic Energy Agency - Technical Cooperation (IAEA-TC) project PHI0015.

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214-THE REGULATION DEVELOPMENT PROCESS ON JUSTIFICATION OF ASYMPTOMATIC EXPOSURE IN INDONESIA

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One of the recommendation of Integrated Regulatory Review Service (IRRS) Follow-Up Mission in 2019 was BAPETEN shall regulate justification of asymptomatic exposure. The IRRS team carried out a review of the progress made on each recommendation and suggestion that is documented in the 2015 IRRS Mission report. The current condition since the first IRRS mission in 2015, most of the revisions of the relevant legislation are still under development such as the amendment to Act No. 10 of 1997 on "Nuclear Energy" and the associated Government regulation and BAPETEN Chairman Regulations.

The objectives of the presentation was to conduct the action on recommendation of IRRS Follow-Up Mission regarding asymptomatic exposure to be in line with GSR Part 3. The analysis of the presentation was guided by literature review of identifying and interpreting applicable rules.

The presentation will examine the report of IRRS Follow Up Mission and the current national legal frameworks of Indonesia in context of medical exposure and its justification. BAPETEN is the national nuclear regulator for Indonesia, and is responsible for all aspects of regulating radiation safety, nuclear safety and security.

Based on IRRS Follow Up Mission Report, asymptomatic exposure are not explicitly covered by the regulations. Meanwhile, riview of the current regulation on medical facilities shows that there are no requirements on justification for asymptomatic exposure. Because of the report, the IRRS team gave recommendation for government and BAPETEN to ensure that the legal and regulatory framework shall kept up to date and corresponds to the current IAEA standards.

When the IRRS mission was conducted, BAPETEN were in process of revision government regulation (GR) No. 33 year 2007. The draft revision of GR No. 33 year 2007 specifies provisions for general and individual justification, for consultation between radiological medical practitioner and referring medical practitioner, for the referrals. However, there are no provisions related to asymptomatic exposure and self-referred patients exposure or to the expected diagnostic or therapeutic benefits of the radiological procedure as well as the radiation risks. Based on the recommendation, BAPETEN will take action to review all the draft revision of regulations in order to be in line with GSR Part 3.

BAPETEN will provide the legal basis for developing specific regulations for medical practices, the new draft of GR No 33 of 2007 on "Safety of Ionizing Radiation and the Security of Radioactive Sources" and other regulations will include relevant aspects of medical exposure such as:

- Justification of medical exposure, including for self-referred and asymptomatic patients, and the use of referral guidelines
- Responsibilities and competence requirements of relevant parties
- Optimization of protection and safety taking into account all operational aspects
- Prevention of unintended or accidental medical exposure, investigation and follow-up of such exposures
- Criteria and guidelines for the release of patients
- Radiological reviews and their records

Therefore BAPETEN hope the action to the recommendation of follow up IRRS mission will answer problems that's not explicitly covered by the regulations, such as asymptomatic exposure. Ministry of Health and BAPETEN shall establish safety requirements for control of medical exposures, including requirements for responsibilities, justification, optimization and accidental exposures.

Key words: *justification, asymptomatic exposure, IRRS mission, regulation*

ACKNOWLEDGEMENTS

I would like to express my very great appreciation to colleagues at BAPETEN especially at Directorate for Regulation Development for Radiation Facility and Radioactive Material, and also IRRS team for their valuable and constructive suggestions during the mission, and development of the reguation. Their willingness to give their time so generously has been very much appreciated.

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215-ESTABLISHMENT OF A NEUTRON DOSIMETRY FACILITY FOR PROTECTION LEVEL CALIBRATION IN THE PHILIPPINES

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A neutron dosimetry facility for protection level calibration in the Philippines was established to bridge the gap in occupational radiation protection of workers from neutron exposures in the Philippines. The presentation shows the considerations of the facility, method used for standardization, traceability of the field, and human resource development that were funded by the IAEA Technical Cooperation Project. This facility ensures the safety of radiation workers all over the country and fulfils the recommendations of IAEA GSR Part 3.

INTRODUCTION

Many facilities in the Philippines are already utilizing neutron radioisotopes and high-energy radiation fields that produce photo-neutrons. With the advent of the increasing number of radiation facilities using neutron radiation, there is now a growing need to ensure the safety of Filipino radiation workers from neutron exposures. Calibrated radiation protection instruments for neutrons need to be available for use to assess and confirm safe workplace conditions. The International Atomic Energy Agency (IAEA) General Safety Guide Part 3, discusses that the government must ensure that technical facilities are available to calibrate monitoring instruments. The Philippine Nuclear Research Institute Secondary Standard Dosimetry Laboratory (PNRI-SSDL) therefore initiated the establishment of a neutron dosimetry facility to develop the capability of the country to have national standards for neutrons and thereby calibration of protection level monitoring instruments.

The neutron facility was established by utilizing an unused building that has existing structural shielding. The neutron calibration field was standardized according to international standards [1, 2] via the shadow-cone method using a reference instrument traceable to a PSDL. The results were then compared to numerical calculations. Human resource capabilities on computational and measurement methods were also advanced through training of staff various neutron facilities in Japan and Czech Republic which was primarily funded through the IAEA Technical Cooperation Project. The presentation shows how the facility was established at a relatively lower cost, standardization methods used, traceability of instruments, and laboratory capabilities to sustain the radiation protection programs all over the Philippines.

METHODS AND ANALYSIS

The current facility is from a refurbished building structure connected to the Philippine Research Reactor-1. The facility structure has existing 1-meter thick walls with high density concrete, especially in the current irradiation room. This existing structural shielded lowered the overall cost of refurbishing activities. Correspondingly this also allowed bare sources to be turned 'on' without any leakage radiation. The facility was primarily built for nuclear fuel processing; hence existing physical security was considered adequate for the projected use which is for calibration and dosimetry purposes.

The current main irradiation room houses a bare Californium-252 neutron source and a bare Americium-241/Beryllium neutron source, both with calibrated source emission rates. The irradiation room has a minimum room length of 3 m x 3 m x 3 m in accordance with the ISO 8529 standard for 40% room return for calibration radiation protection instruments [1]. The calibration bench and tabletop are made of aluminium and has minimal scatter during calibration due to its low neutron cross section; and can be moved in the x-, y-, and z-axes. A secondary standard reference instrument with a Helium-3 proportional counter embedded in a 10" Bonner sphere traceable to the National Physical Laboratory, U.K. is used for standardization and characterization of neutron fields in the laboratory. The instrument is calibrated for fluence response for Cf-252 and Am241/Be. The AmBe bare source and reference instrument were procured with the support of the IAEA TC Project (PHI0015).

The international standard ISO 8529 on reference neutron radiations recommends four methods to characterize a neutron beam, however, of the four methods, the shadow cone method is most recommended [3] because it uses experimental methods to determine the scattered component in the field. Other available methods use analytical and computational methods; however these methods may not fully realize the scattered component of the neutron fields, which may lead to errors and uncertainties in the calculations.

A shadow cone according to ISO 8529 was therefore fabricated and was used for characterizing the neutron field. To reduce costs, spliced slabs instead of a solid polyethylene was used. Analytical and computational methods using MCNP5 were also conducted and compared with the measurements. The measured field is then used for calibration of neutron radiation monitoring instruments such as dose rate meters and personal dosimeters.

Calibration points were established in the irradiation room at a height of 130 cm. Calibration were performed at source-to-detector distances of 100 cm, 130 cm, and 150 cm. The reference instrument was placed at the three calibration points, and the shadow cone method was applied to determine the fluence rate of the primary neutron field of the Cf-252 bare source. The measured fluence rates at the calibration points were then compared with analytical methods and MCNP simulations [4].

The training of staff was conducted in 2018 and the neutron facility was fully completed in early 2019. Standardization activities were conducted, and the established fluence rate of the Cf-252 bare source is agreeable to within 11% with analytical and simulated results [4]. This result is found to be consistent with other studies conducted [5]. The national standard for Cf-252 protection level neutron field was therefore established. Following the standardization, calibration of a neutron dose rate meter and neutron passive dosimeters such as OSLD and TLD instruments were performed. This then allowed individual monitoring for neutron exposures of personnel. The results show that the facility is now ready for use for protection level calibrations in the Philippines and will therefore lead to provision of new radiation protection services in the country.

CONCLUSION

To ensure safe radiation workplaces and fulfil the recommendations of IAEA GSR Part 3, a secondary standard neutron dosimetry facility was established for calibration of radiation protection instruments. The facility was established at relatively lower cost by utilizing in an existing facility with sufficient shielding for the irradiation room, and adequate physical safety and security features. The neutron calibration field was established according to ISO 8529 using shadow cone method. Field measurements have shown to be agreeable up to 11% when compared with analytical and computational methods. With the support of an IAEA Technical Cooperation project, a neutron dosimetry facility for calibration of radiation protection instruments was established to promote safety in radiation facilities and ensure effective radiation protection programs in the Philippines.

ACKNOWLEDGEMENT

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216-IMPROVEMENT OF RADIATION SAFETY USING PERSONAL RADIATION DOSIMETER PLUG IN SMARTPHONE

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In this research, improvement of nuclear safety and nuclear security using a personal radiation dosimeter (PRD) that plug into the smartphone is considered. The members of the general public and peoples in various occupational levels in nuclear medicine and nuclear industries are the stakeholders of this object. In both cases of official personnel and public population, a small but effective PRD can be connected wirelessly by blue-tooth or plugged into a smartphone according to the phone designing. New generations of PRDs that can be connected to smartphones have been developed by a number of companies. It is possible that national nuclear security command and emergency centers could greatly enhance their radiological mapping and analysis capabilities using personal smartphones and existing national communication systems. The essential elements for social and physical infrastructure for application of personal detectors and smartphones in the establishment of received dose collection data a real-time radiological monitoring is described.

The radiation risks to workers, public and to the environment that may arise from these applications have to be assessed and, if necessary, controlled [1]. Activities such as the medical uses of radiation, the operation of nuclear installations, the production, transport and use of radioactive material, and the management of radioactive waste must therefore be subject to standards of safety. Exposure of any part of the human body to X-rays, Gamma, Neutron, Alpha and Beta can be highly injurious to health. Wherever radioactive sources and X-ray equipment are in use, appropriate legal requirements must be applied. Time, distance, and shielding actions minimize human exposure to radiation risk. The Universal Declaration of Human Rights (UDHR) is a milestone document in the history of human rights. UDHR was drafted by representatives with different legal and cultural backgrounds from all regions of the world. The Declaration was proclaimed by the United Nations General Assembly as a common standard of achievements for all peoples and all nations. According to the UDHR, article 3 “The right to life. We all have the right to life, and to live in freedom and safety and personal security”, people have the right to protect themselves against radiation hazards. Humans have five basic senses: touch, sight, hearing, smell and taste. The sensing organs associated with each sense send information to the brain to help people understand and perceive the world around them. But, these five senses have not ability to sense the radiation. This is the significant and basic problem of people against radiation hazards. The IAEA is an international organization that seeks to promote the peaceful use of nuclear energy, and to protect people from radiation hazards. Due to the IAEA mission, IAEA have global right to treat an international program for member states to provide a cheap, accessible and user friendly radiation dosimeter for their people [2].

The recent developments in real time sensing, mobile, and embedded devices have attracted considerable attention toward mobile health monitoring applications. However, the existing architectures aimed at facilitating the realization of these mobile applications have shown to be not suitable to address some challenging issues such as the seamless integration of heterogeneous devices and the estimation of vital parameters not measurable directly or measurable with a low accuracy. A personal radiation monitoring applications for different scenarios, by exploiting commercial detectors and mobile devices with commercial platforms such as Android as well as knowledge-based technologies must be assessed for radiation safety. Highlighting the capabilities of for smartphones that being rapidly customized, personalized or eventually modified by software developers with confirming its effectiveness with respect to a real scenario is one essential step of this idea [3].

For radiation protection, the first step is identification of radiation and its source. A user friendly radiation dosimeter can help people to sense the radiation and act next steps for protection against radiation. The second step is considering the time, distance and shielding against radiation. Time: Minimizing the exposure time reduces the dose from the radiation source. Distance: When people move further away, the dose of radiation decreases dramatically as people increase his distance from the radiation source. Shielding: Barriers of lead, concrete, or water provide protection from penetrating (Fig. 1) [4].

FIG. 26. Images of commercial Personal Radiation Detector plugged into smartphone as example.

It is clear that the first step for protection of people against radiation hazards is identification of radiation and radiation sources. Also, it is possible to identify the radiation just using radiation dosimeters. Hence, people have the right to provide a personal radiation dosimeter for their own life. According to the IAEA Safety standard Principle 7: "Protection of present and future generations people and the environment, present and future, must be protected against radiation risks" and the safety standard principle 10: "Protective actions to reduce existing or unregulated radiation risks, Protective actions to reduce existing or unregulated radiation" [1], States shall treat the necessary infrastructure in their country to help their people to have their own personal radiation dosimeter. Due to development of technology, most of people of around the world have Smartphone. As an alternative, it is possible for most of radiation detector manufactures to provide a cheap and user-friendly personal radiation dosimeter that attached to the Smartphone or as separate device [5].

Whenever people always have their own radiation dosimeter with themselves as like as smartphone, people will be sure, when they receive radiation, they will aware on time. Hence, they can start suitable reaction against radiation source. This ability helps people to have very less fear against radiation, nuclear activities and nuclear Technology and hence, help the IAEA and their states to increase the peaceful applications of nuclear energy and nuclear technology. Additionally, during a nuclear and radiological accident, people can help the states to find radiation sources and accelerate the emergency plan. People with a pocket or handheld radiation dosimeter can record their received radiation dose during daily life from any kind of sources [6].

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217-OVERVIEW OF THE RADIATION PROTECTION SYSTEM AND RADIATION MONITORING AT URANIUM MINING ENTERPRISES USING ISL METHOD

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KAZATOMPROM RADIATION SAFETY SYSTEM

As of the end of 2018, Kazakhstan is a leader at uranium mining and takes second place in terms of explored uranium reserves [1]. Uranium mining in Kazakhstan is entirely carried out by ISL method, the final product is chemical concentrates of natural uranium (U_3O_8, UO_4) [2]. The radiation protection system at Kazatomprom enterprises is based on a wide range of legal requirements, IAEA standards and various best practices.

Kazatomprom (the Company) and its subsidiaries and affiliates operate on 26 sites on the territory of the Republic of Kazakhstan that united in 13 mining enterprises.

Kazatomprom has developed a standard for the radiation safety management system at Kazatomprom enterprises, which contains a risk-oriented approach and applies to all stages of the natural uranium mining and processing process and contains requirements for the following stages:

- Training of workers;
- Equipment;
- Instructions and procedures;
- Contractors;
- Radiation monitoring;
- Waste management;
- Emergency response;
- Analysis and continuous improvement.

RADIATION MONITORING

Radiation monitoring programs have been developed at enterprises in accordance with legal requirements. Preliminary assessment of radiation risks associated with radiation factors at each stage of the mining and processing of natural uranium has been done, as well as possible ways of their impact on workers [2]. External gamma radiation, radon daughters, and long-lived alpha isotopes in the air of the working zone affect the personnel at various stages of uranium production. The programs include types and routes of exposure, personal protective equipment used, monitoring of workplaces, production area, boundaries of the sanitary protection zone, residential territory and personnel monitoring. Work with radiation sources is carried out in specially designated places. The task of radiation monitoring is to preserve and retain the source in these areas, to prevent the transfer of radioactive substances from the controlled area. This is observed by conducting routine radiation monitoring, and operational radiation monitoring when moving equipment and vehicles from the controlled area. Personnel of the working area, when leaving it, passes through specially designed bathing and delousing centers that contain the necessary conditions for replacing personal protective clothing, taking a shower, and conducting radiation monitoring of skin and clothing [3]. The frequency of scheduled monitoring depends on the risk assessment, and the current state of radiation safety. Individual monitoring is carried out using Thermoluminescence dosimeters. Individual monitoring is carried out daily, reading of the results is carried out on a quarterly basis. Determination of the dose for internal exposure is carried out by calculation. In uranium mining activities, the doses are relatively not high, and are comparable with background values. Nevertheless, the task of the radiation protection system is aimed at ensuring the assessment, regulation and monitoring of exposure in order to reduce radiation risks to the achievable ones [4].

EMERGENCY RESPONSE

In accordance with the legislative requirements of the Republic of Kazakhstan, an emergency response team has been formed at each enterprise, emergency response plans have been developed, emergency response exercises are being conducted according to schedules. Particular attention is paid to the emergency response process during the transportation of finished products. In order to minimize accidents during the transportation of radioactive substances, vehicles are equipped with GPS trackers and movement in columns with an escort vehicle is organized.

ANALYSIS AND CONTINUOUS IMPROVEMENT

Single-control of enterprises work allows to use all possible communications to analyze the current state, identify and respond as soon as possible to newly identified risks, and organize improvement of radiation protection. These communications include both communications within the Group of Companies and interaction with government bodies, as well as with international organizations such as the IAEA.

PRESENT AND FUTURE CHALLENGES

The current radiation safety situation is based on the currently assessed radiological risks and takes into account the stages involved in the mining and production of uranium by the ISL method and the current state of legislation in the field of atomic energy use. Work processes are being studied to reduce operations with radioactive substances, the automation and the optimization of technological processes.

Further development of the deposits will lead to large-scale development of ISL blocks, subsequent remediation will require the development of new approaches to ensuring safety in the management of radioactive waste, and the development of the necessary competencies for personnel.

In addition, the systematization and step-by-step approach to the analysis of personnel radiation doses allows optimizing individual work processes in order to reduce the average personnel radiation doses for a group of companies.

Therefore, the further development of the radiation protection system at Kazatomprom will take into account the risks associated with the planned and developing stages of the mining and processing of natural uranium, new knowledge and experience of other international organizations and companies, and will be aimed at automating the processes of obtaining data on the state of radiation safety, as well as the automation of technological processes to reduce the contact of workers with radioactive substances.

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220-FEASIBILITY OF USING HIGH-DENSITY 3D-PRINTED FILAMENTS FOR RADIATION SHIELDING

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The most common radiation shielding material for Gamma and X-rays is lead due to its high atomic number and high density (11.34 g/cc). For an alternative and more flexible shielding material, this study proposes utilizing 3D printing filaments infused with high density elements in the shape of nanoparticles for radiation shielding. Two different types of high-density filaments were used to 3D-print slabs for Gamma and X-ray attenuation analysis. Both types consist primarily of Acrylonitrile Butadiene Styrene (ABS) while one is infused with Bismuth, and the other is infused with Tungsten. They will be referred to as ABS-Bi and ABS-W, respectively. Furthermore, a pure Polylactic Acid (PLA) 3D-printed slabs were used for comparison. This study aims to evaluate the feasibility of using high density 3D printing filaments for radiation shielding against low and high energy Gamma and X-rays. Moreover, the study includes the mass attenuation coefficient values measured for all the filaments used at four different energies.

A desktop type of Fused Deposition Modelling (FDM) 3D printer was used to print six 50x50x5 mm^3 slabs of each filament type to be used as radiations attenuators. The infill for each slab was set to 100%. The measured density for PLA is 1.1 g/cc, while it is 2.2 g/cc for ABS-Bi and 3.6 g/cc for ABS-W. Field Emission Scanning Electron Microscope (JEOL JSM-7600F) was used to determine the chemical composition of the infused filaments with detection capability up to 100 PPM (particle per million) of concentration. The amount of Tungsten in ABS-W is 6300 PPM, while Bismuth present in the ABS-Bi sample is less than 100 PPM.

The radiation attenuation analysis experiment was set by placing a source at 10 cm away from a Sodium-Iodide (NaI-doped with Tl) detector while a slab-holder was placed in the space between the source and the detector. The uncertainty of the measurements ranged from 0.45% to 3.25%.

Gamma-ray sources used in this study were Am-241 (59.6 keV), Cs-137 (662 keV), and Co-60 (1337 keV). Moreover, the slabs (3-cm-thick) were scanned at 35 mA and 120 kV by a medical CT scan machine (SOMATOM definition flash model, manufactured by Siemens). ABS-Bi, ABS-W, and PLA slabs were used as attenuators, but while experimenting with Cs-137 and Co-60, PLA slabs were replaced by lead slabs.

The following four figures present the attenuation results achieved. FIG. 27. shows the attenuation effect against Am-241 radiation. Intensity decrement was significant for the infused filament slabs compares to the pure PLA slab. Less than one cm-thick of ABS-Bi and ABS-W reduced the intensity to below 10%. FIG. 28. shows the attenuation results for the CT scan. ABS-Bi and ABS-W slabs attenuated nearly 75% of the intensity while PLA slabs attenuated about 50%. FIG. 29. shows the attenuation of Cs-137 radiation. Approximately a thickness of 1 cm of the high-dense slab is sufficient to attain the half value layer (HVL) compared to lead, which only required about 5 mm. FIG. 30. shows the attenuation of Co-60 radiation source. The required thickness of the high-dense slab to reach the HVL is calculated to be approximately 4 cm compared to the HVL of lead, which is 1.5 cm.

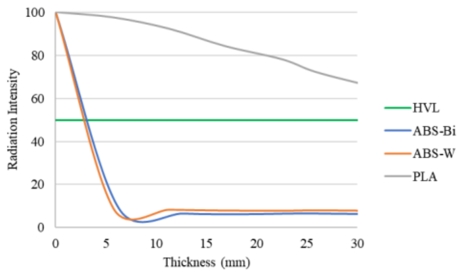

FIG. 27. Radiation intensity attenuation for Am-241 source (59.6 keV).

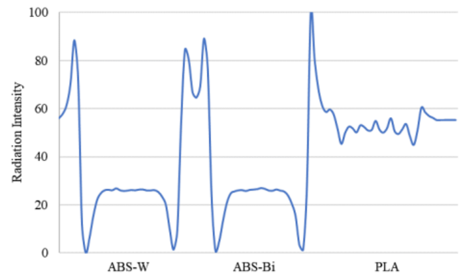

FIG. 28. CT-scan intensity attenuation (120 KV).

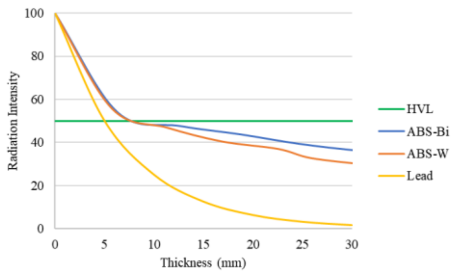

FIG. 29. Radiation intensity attenuation for Cs-137 source (662 keV).

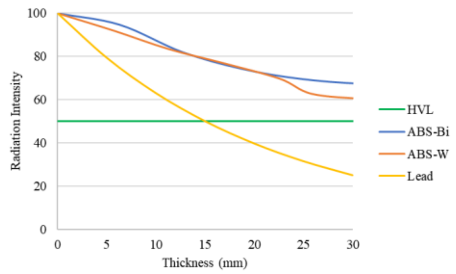

FIG. 30. Radiation intensity attenuation for Co-60 source (1337 keV).

TABLE 3. The mass attenuation coefficient (μ/ρ) for each high-density filament at different energies. Coefficient values were calculated using the law of exponential attenuation for Gamma-rays, $I = I_0 e^{-\mu x}$, to compute (μ/ρ) from the measured values of incident intensity, transmitted intensity, material density, and attenuator thickness. [1]

TABLE 3. MASS ATTENUATION COEFFICIENT FOR EACH FILAMENTS AT DIFFERENT ENERGIES (cm^2/g)

Material	26 keV	60 keV	662 keV	1337 keV
ABS-Bi	5.08	19.63	0.80	0.31
ABS-W	8.91	13.94	1.59	0.67
PLA	0.17	0.16	0.10	0.08

In conclusion, this study demonstrates the efficiency of utilizing 3D printing technique with high-density filaments for shielding against radiation with energies between the range of 20 and 800 keV. This energy range covers medical radiography applications, part of industrial radiography applications, and orthovoltage radiation therapy.

ACKNOWLEDGMENTS

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221-IAEA SAFETY STANDARDS IN THE AREA OF EMERGENCY PREPAREDNESS AND RESPONSE: REFERENCE LEVELS, GENERIC CRITERIA AND OPERATIONAL CRITERIA

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The IAEA's Statute authorizes the IAEA to establish or adopt — in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and the specialized agencies concerned — standards of safety for the protection of health and the minimization of danger to life and property, and to provide for the application of these standards. The IAEA commenced its safety standards programme in 1958.

The first Safety Standard in the area of emergency preparedness and response (EPR), Safety Requirement No. GS-R-2 (Preparedness and Response for a Nuclear or Radiological Emergency) [1] was issued in 2002 and co-sponsored by seven international organizations. This Safety Requirements publication established the requirements for an adequate level of preparedness and response for a nuclear or radiological emergency in any State. In 2015, it was superseded by Safety Requirement No. GSR Part 7 (Preparedness and Response for a Nuclear or Radiological Emergency) [2], which was co-sponsored by 13 international organizations (FAO, IAEA, ICAO, ILO, IMO, INTERPOL, OECD/NEA, PAHO, CTBTO, UNEP, OCHA, WHO and WMO). Currently, the set of Safety Standards in the area of EPR includes Safety Requirement No. GSR Part 7 and three Safety Guides [3-5], while two additional Safety Guides are at the final stage of preparation or publication.

The safety standards in the area of EPR are based on the scientific considerations derived from the findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). They take into account the recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP), and provide for their practical application.

One of the examples of a link between the fundamental science of UNSCEAR, the recommendations of the ICRP and the requirements of the IAEA safety standards in EPR is related to the protection strategy for the public in case of a nuclear or radiological emergency. GSR Part 7 requires that "*protection strategies are developed, justified and optimized at the preparedness stage for taking protective actions and other response actions effectively in a nuclear or radiological emergency to achieve the goals of emergency response.*" These Requirements explain the importance of the application of criteria and differences of criteria established for different purposes. Specifically, they explain the application of reference levels (that are applicable to the overall protection strategy); the generic criteria (that are applicable for individual protective actions); and the operational criteria (that are applicable for initiating the different actions within an emergency response plan). The requirements outline the process of developing the strategy, from setting up a reference level (from ICRP recommendations [6]), establishing a set of generic criteria, and then to developing default operational criteria (such as, conditions on the scene, operational intervention levels and emergency action levels).

A reference level expressed in terms of residual dose needs to be set, typically as an effective dose in the range 20–100 mSv, acute or annual, that includes dose contributions via all exposure pathways. Generic criteria are based on the current knowledge of deterministic and stochastic effects, as outlined by UNSCEAR. They address both external and internal exposure that could be directly related to the full range of important radionuclides. Generic criteria cover extended list of protective actions and other response actions, including restrictions on food, milk and drinking water; restrictions on commodities other than food; contamination control; decontamination and health screening. There are also generic criteria for the implementation of protective actions and other actions aimed at enabling the termination of a nuclear or radiological emergency and the subsequent transition to an existing exposure situation. Operational criteria are associated with directly measurable quantities or observable conditions and are based on the generic criteria.

It is important to highlight that choosing a reference level from the range 20–100 mSv at the numerical level, which is lower than 100 mSv annual residual effective dose, does not require scaling down of applied IAEA generic criteria [2, 5]. This is because the reference level is expressed in terms of residual dose and is used for implementation of the protection strategy and its planned adaptation in an emergency, while generic criteria are expressed in terms of projected dose without implementation of any protective actions. What will be the resulting residual dose, after implementation of the protective actions based on the projected dose, will depend on the efficiency of these protective actions. If all these actions are 100 percent efficient, then hypothetically protection strategy will result in 0 mSv residual annual doses for affected population, regardless of the level of projected doses at which the generic criteria are established. In reality, it is difficult to achieve 100 percent efficiency of the protection strategy, as it depends on the type of the implemented strategy, available resources and other reasons. All these reasons will influence the efficiency to some extent, but there is a high chance that the resulting residual dose would be even lower than 20 mSv (lower level of the 20-100 mSv range of reference levels for an emergency). Therefore, depending on the efficiency, taking actions at levels of projected doses, at which IAEA generic criteria are established, may result in various residual doses to be achieved. Thus, IAEA generic criteria [2, 5] are compatible with the reference level selected at any numerical level of residual dose up to and at 100 mSv. These considerations need to be carefully analyzed at the preparedness stage during the hazard assessment and development of the protection strategy before the national reference level is selected. Such analysis also helps authorities to identify ways of increasing efficiency of protective actions.

GSR Part 7 and GSG-11 [2, 5] address also protection to be provided at doses lower than the above-discussed internationally agreed generic criteria, and emphasize the need for thorough justification and optimization to ensure that (1) implemented actions do more good than harm, considering social and economic factors; and (2) best protection under the prevailing circumstances is achieved (which is not necessarily accomplished through the implemented action based on criteria that is established at a lowest level of dose).

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222-RADON: ICRP ADVICE AND DOSE COEFFICIENTS

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The International Commission on Radiological Protection (ICRP) recommends a system of radiological protection and provides guidance and information to facilitate the implementation of practical protection measures. ICRP has developed the protection quantities, equivalent and effective dose, and publishes dose coefficients for use in the assessment and control of exposures of workers and members of the public, considering external sources and intakes of radionuclides. The sets of dose coefficients provided for radionuclide intakes include values for the inhalation of radon isotopes, ^{222}Rn (radon), ^{220}Rn (thoron) and ^{219}Rn (actinon), together with their radioactive progeny.

The inhalation of radon and progeny is unique among internal exposures in that there is strong quantitative evidence of the relationship between exposures and lung cancer induction from studies of underground miners and people in their homes [1-3]. In general, control of radon exposures can be based directly on measurements of concentrations in Bq m^{-3} and the setting of reference levels for homes and workplaces [4,5]. However, dose estimates and hence dose coefficients are required for circumstances where, from the outset, the exposure is considered to be occupational, and in circumstances where exposures in workplaces remain persistently above the reference level despite remediation. Dose estimates may also be required in assessing public exposures in some circumstances, for example when considering doses resulting from past contamination of buildings with radium isotopes. Exposures to thoron, and particularly actinon, are substantially less important in most cases but can be significant in particular circumstances.

ICRP Publication 65 [6] used the so-called epidemiological approach or dose conversion convention to calculate effective dose coefficients for radon, in which estimates of lifetime lung cancer risk from radon are divided by values of total detriment from cancer and hereditary effects. Dose coefficients of 5 mSv per WLM for workers and 4 mSv per WLM for the whole population were calculated using a lifetime lung cancer risk of 2.83 x 10^{-4} per WLM and ICRP Publication 60 [7] detriment values (1 WLM = 3.54 mJ h m^{-3}).

ICRP Publication 115 [3] provided an updated review of epidemiological data and, focussing on more recent data and lower levels of exposure, a revised value for lifetime lung cancer risk of 5 x 10^{-4} per WLM was proposed. Consequently, the upper reference level for radon in homes was lowered to 300 Bq m^{-3} from the value of 600 Bq m^{-3} given in the 2007 Recommendations [4]. Taking account of the different lengths of time spent in homes and workplaces, a level above which the requirements of occupational protection would apply was set at 1000 Bq m^{-3}.

ICRP Publication 126 [5] updated advice on protection of the public and workers against radon exposures. The advice centres around the optimisation of protection, to maintain or reduce exposures to levels that are as low as reasonably achievable, taking economic and social circumstances into account. The objective is to reduce both the overall risk of lung cancer in the general population and the individual risk to the most highly exposed individuals. The upper reference level of 300 Bq m^{-3} was confirmed, with the advice that national authorities should set a reference level in the range of 100 to 300 Bq m^{-3} depending on their particular circumstances. This is consistent with advice given by the World Health Organisation [8] that the reference level should be set at 100 Bq m^{-3} if possible but otherwise at a level not exceeding 300 Bq m^{-3}.

For protection of workers, Publication 126 goes further than Publication 115 and recommends that the upper reference level of 300 Bq m^{-3} should apply generally to all buildings and hence to workplaces such as offices as well as mixed-use settings such as shops, restaurants and schools. A graded approach is recommended in which protection is first optimised below the national reference level. If remediation is unsuccessful in reducing exposures to below this level, a second step will be a realistic estimation of effective dose, taking account of exposure conditions including occupancy. If, despite all reasonable efforts to reduce radon exposures, the doses remain persistently above 10 mSv, workers should be considered to be occupationally exposed. For some work environments, such as thermal spas, show caves, and underground mines, the radon exposure will be considered from the outset to be the responsibility of the operating management and hence categorised as occupational exposure.

ICRP Publication 137 [9] is the third part in a series of reports providing dose coefficients and associated data for occupational exposures to radionuclides, and includes radioisotopes of radon. Dose coefficients are calculated using biokinetic and dosimetric models and those for radon include values for exposures in mines, indoor workplaces and tourist caves. However, noting that inhaled ^{222}Rn and progeny is a special case for which there is good epidemiology as well as dosimetry, and taking account of the two methods of calculation of dose coefficients with their associated uncertainties, ICRP recommend a single rounded value for use in most circumstances of occupational exposure of 3 mSv per mJ h m^{-3} (approximately 10 mSv per WLM), equivalent to 6.7 nSv per Bq h m^{-3} applying an equilibrium factor between radon and progeny of 0.4. ICRP has also indicated that this value is applicable to exposures in homes. Using this dose coefficient, the reference level of 300 Bq m^{-3} corresponds to 14 mSv per year for homes (7000 hours) and 4 mSv per year for workplaces (2000 hours).

For occupational exposures to radon in which conditions such as aerosol characteristics are significantly different from the reference conditions, where estimated doses warrant more detailed consideration, and reliable data are available, it is possible to calculate site-specific dose coefficients using data provided by ICRP. A second higher value of 6 mSv per mJ h m^{-3} (approximately 20 mSv per WLM) was referred to in ICRP Publication 137 but this may be seen as an example of requirements for more specific calculations when warranted.

UNSCEAR [2] has recently reviewed epidemiological studies of lung cancer in underground miners. Estimates of lifetime excess absolute risk from radon ranged from 2.4 to 7.5 x 10^{-4} per WLM. Using the most recent detriment values from the 2007 Recommendations of the ICRP [4] of 4.2 x 10^{-2} per Sv for workers and 5.7 x 10^{-2} per Sv for the whole population, the ranges in lifetime risk of lung cancer derived by UNSCEAR correspond to ranges in dose coefficient of 5.7 – 17.9 mSv per WLM for workers and 4.2 – 13.2 mSv per WLM for the whole population. UNSCEAR [2] also reviewed recent published dosimetry assessments for exposures in homes, indoor workplaces and mines. Applying an equilibrium factor of 0.4, these values correspond to a range of 3 to 14 nSv per (h Bq m^{-3}) with means of 6 - 7 nSv per (h Bq m^{-3}). Thus, the available data as reviewed by UNSCEAR support the use of ICRP dose coefficients as central values.

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223-ICRP DOSE COEFFICIENTS AND CONSERVATISM

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A common misconception is that the dose coefficients provided by the International Commission on Radiological Protection (ICRP) are conservative values that tend to result in overestimates of dose, particularly for inhaled and ingested radionuclides. In reality, the biokinetic and dosimetric models used in the calculation of dose coefficients have been improved following each set of general recommendations with the intention of providing central estimates of organ doses per exposure. However, the formulation of effective dose and committed effective dose includes simplifications which potentially could impede the optimisation process because the inferred risk per Sv varies according to the age and sex of the exposed individuals.

The 2007 Recommendations of ICRP [1] introduced the use of reference anatomical models of the human body for the calculation of dose coefficients. The reference adult male and female phantoms being used in current calculations are based on medical imaging data, with the volumes of organs and tissues constituted using voxels [2]. As it is now possible to perform radiation transport calculations without voxelization, ICRP is also developing mesh-type reference phantoms with even greater spatial resolution, enabling the calculation of doses in very small tissue volumes, including single cell layers [3]. The biokinetics of inhaled and ingested radionuclides are increasingly being modelled to include absorption to blood and the dynamics of recirculation to and from organs and tissues, as well as loss from the body by urinary and faecal excretion [4,5]. These physiologically realistic models can be used for the interpretation of bioassay measurements as well as the calculation of organ retention and doses.

Having calculated values of absorbed dose (Gy) to organs and tissues per exposure as best estimates using sophisticated modelling approaches, effective dose (Sv) coefficients are calculated as a doubly weighted quantity, applying simplified weighting factors relating to stochastic risks from radiation [1]. Radiation weighting factors, w_R, are used to adjust for the relative effectiveness per Gy of different radiation types in causing cancer. However, a single value of 20 is used for alpha particles relative to low LET radiations (using 1 for all), despite recognised differences in values for different cancer types [6]. Tissue weighting factors, w_T, adjust for the relative contributions of organs / tissues to overall stochastic risks, expressed as detriment [1]. However, a simplified set of values are used that relate approximately to relative detriment values, considering life-time risk averaged over all ages, both sexes, and seven Asian and Euro-American populations.

There is arguably an inconsistency of approach between the rigorous application of science in the calculation of absorbed doses to organs and tissues and the simplifications included in making risk adjustments in the calculation of effective dose. The rationale has been that effective dose is used in the control and optimisation of exposures below constraints and reference levels that apply to all workers or all members of the public and that greater scientific accuracy would not improve practical protection. While effective dose in its current formulation is central to the well-established system of protection applied around the world, it is instructive to consider the potential benefits of changes such that effective dose coefficients would more precisely reflect the scientific evidence.

Values of w_R could be refined to consider alpha particle effectiveness per Gy for different cancer types, with a value of 10 for lung and liver cancer and a value of perhaps 2 for leukaemia [6]. Inclusion of a low value for leukaemia would reduce or eliminate the anomaly that the dose coefficient for plutonium-239 suggests that excess leukaemia might be seen in the Mayak workers, whereas cancers relating to plutonium-239 doses are seen for lung, liver and bone cancers but not leukaemia [7]. A change could also be made to take account of the greater effectiveness of low energy, low LET radiations in causing cancer in comparison with cobalt-60 gamma rays. A value of 2 might be used, for example, for tritium beta particles [8].

Values of w_T are based on relative detriment values but are rounded so that only four values (0.01, 0.04, 0.08, 0.12) are used for all organs / tissues that are assigned risk estimates [1]. For example, colon, lung, red bone marrow and stomach are each assigned a w_T of 0.12, based on relative detriment values of 0.08, 0.16, 0.11 and 0.12, respectively. Detriment is calculated largely from cancer incidence data, adjusted for fatality, years of life lost and quality of life, with a small addition for hereditary effects. The primary source of information on cancer risks is the follow-up studies of the Japanese A bomb survivors. These data show differences between males and females in risk of cancer incidence for the various cancer types, and also a marked effect of age at exposure. Overall risks per Gy for females are greater than for males, by approaching a factor of two, and risks are substantially greater for exposures at younger ages, with a difference of a factor of 2 – 3, for example, for exposures at 0 – 9 years compared with 30 – 39 years [9]. These differences are substantially greater when considering the example of thyroid cancer, with higher risks in young females (0 – 9 years) compared to males by approaching an order of magnitude and a similar factor of ten difference between these groups and the corresponding adult groups (30 – 39 years). Although such differences are recognised, they are not considered explicitly in the current system which instead relies on two averaged nominal risk coefficients of 7.3×10^{-2} Sv^{-1} for the whole population and 5.6×10^{-2} Sv^{-1} for the working age population [1].

It would be possible to calculate cancer risks and detriment (or some similar measure) separately for males and females of different ages. These calculations could correspond to the set of dosimetric phantoms of the human body used in calculations of dose coefficients; males and females of ages: newborn, 1 year, 5 years, 10 years, 15 years and 20 years, but also consider older age groups. Values would be derived for absolute and relative detriment for males and females separately at each age at exposure. It might then be most appropriate and informative to calculate effective dose separately for males and females of the various ages. Such changes would represent best use of the available scientific evidence and avoid the criticism that women and children are not adequately protected. It would be clear that the inferred risk associated with, for example, a 5 mSv reference level would be different depending on the age and sex of the exposed individuals. The corollary should then be that optimisation is applied with a clear understanding of possible risks in the situation being considered. An important caveat, of course, is that the doses being considered are often below the range of direct epidemiological observations of excess risk and so all risk estimates rely on the use of the linear non-threshold dose-response model.

A further factor to consider in the case of inhaled and ingested radionuclides is that doses are integrated to age 70 years to take account of the long retention times and long physical half-lives of some radionuclides; this is a 50 year period for workers who are all assumed to be 20 years of age. The committed effective dose, the total integrated dose to age 70 years, is assigned to the year of intake. Dose commitment is not an important consideration for many radionuclides that deliver dose within weeks or months of intake (eg. iodine-131) but can be important for others (eg. plutonium-239) that deliver dose throughout an individual's lifespan. Clearly, the inferred risk associated with a continuing dose will change with age and reduce to low levels. A more sophisticated system would take account of the changing risks per Sv.

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225-INTERNATIONAL RADIATION MONITORING INFORMATION SYSTEM (IRMIS) – ADDING THE DECISION SUPPORT IN NUCLEAR OR RADIOLOGICAL EMERGENCIES

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The International Atomic Energy Agency (IAEA) International Radiation Monitoring Information System (IRMIS) [1] provides useful information for the decision support process performed by Member States and for the protection of the people and the environment in nuclear or radiological emergencies.

DESCRIPTION

IRMIS is the web-based tool for sharing, aggregating and visualizing large quantities of radiation monitoring data during a nuclear or radiological emergency. The routine provision, on a voluntary basis, of the monitoring data from fixed monitoring stations in non-emergency situations is intended to ensure that the data are reported effectively during an emergency.

During an emergency, additional data (for example: data from handheld measurement devices or mobile monitoring systems such as backpack, vehicle or aerial systems) can be provided by the Member State and is aggregated and displayed for visualization on IRMIS. All data reported in IRMIS remains under the ownership of the reporting Member State. The Member States are responsible for operating and maintaining their monitoring network.

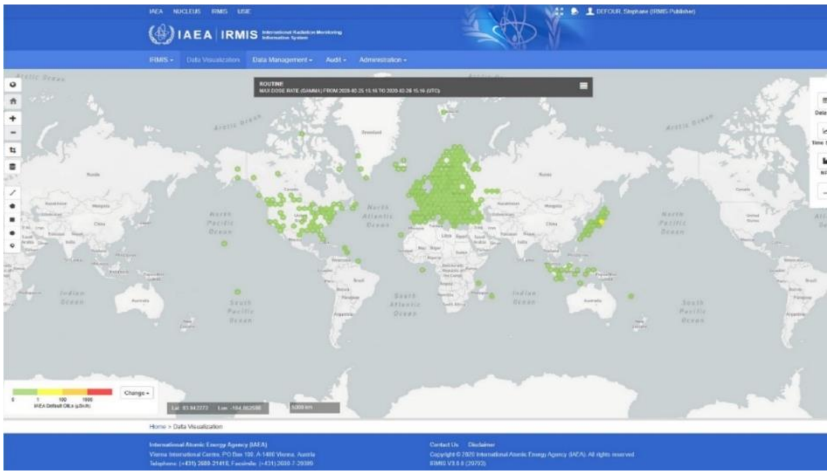

FIG. 31. IRMIS landing page.

IRMIS IN THE DECISION SUPPORT PROCESS

In a nuclear or radiological emergency, IRMIS contributes to the decision support by consolidating the radiological assessment and prognosis conducted by the Accident State and the IAEA. While IRMIS is collecting and displaying fixed point monitoring data and data collected by field deployed emergency monitoring teams and devices, IRMIS also allows the user to compare the monitoring data with the pre-established operational intervention levels (OILs). Also, IRMIS displays planned and/or executed public protective actions (PPAs) communicated by the Accident State [2].

The ambient dose equivalent rate $H_p^*(10)$ is shared and visualized via IRMIS. The ambient dose equivalent rate is an appropriate and effective indicator in an emergency involving a release of radioactivity to the environment. All ambient gamma dose rate values are visualized on a map with reference to the relevant OIL, an ambient dose equivalent rate that corresponds to a generic criterion for taking protective actions [3]. A default OIL is used immediately and directly (without further assessment) to determine the appropriate protective actions on the basis of an environmental monitoring measurement (a measurement by instruments in the field or determined by laboratory analysis). In its current configuration, IRMIS data are compatible with OIL 1, 2 and 3 for the gamma dose rates [2]. To support decision making and inform PPAs beyond those supported by OIL 1, 2 and 3 for gamma dose rates, it is planned that IRMIS will incorporate other data types (such as radionuclide specific air concentration) that will be supported by and used in conjunction with other OILs.

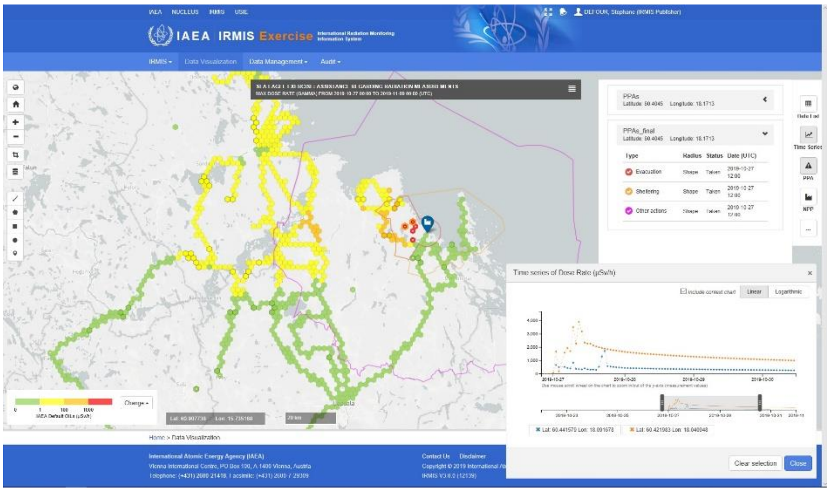

FIG. 2. Simulated data on IRMIS during the Sea Eagle Exercise, Sweden, 2019

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226-THE IAEA ASSESSMENT AND PROGNOSIS IN RESPONSE TO A NUCLEAR OR RADIOLOGICAL EMERGENCY

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The roles of the IAEA in response to a nuclear or radiological emergency are:
- Provision of notification and exchanging and sharing of official information with States and international organizations;
- Performing an assessment of potential consequences and a prognosis of the likely emergency progression;
- Timely coordination and provision of assistance or advice, upon request, to States/international organizations;
- Provision of timely, clear, factually correct, objective, consistent and easily understandable public information;
- Coordination of the inter-agency response.

The purpose of the assessment and prognosis role is for the IAEA's Secretariat to provide Member States, international organizations and the general public with timely, clear, factually correct, objective and easily understandable information during a nuclear or radiological emergency on its potential consequences, including analysis of the available information, and prognoses of possible scenarios based on evidence, scientific knowledge and the capabilities of Member States. In order to provide Member States with details on the IAEA assessment and prognosis process, including its technical basis, the Operations Manual for IAEA Assessment and Prognosis during a Nuclear or Radiological Emergency (EPR-A&P 2019) [1] was published in February 2020. To test the IAEA assessment and prognosis process with Member States, the ConvEx-2e type of emergency exercise was established in 2014. As of April 2020, 41 ConvEx-2e exercises were conducted.

To support its assessment and prognosis process and activities in response to a nuclear or radiological emergency, the IAEA has developed response flowcharts and web-based assessment and prognosis tools. The IAEA response flowcharts aim at providing guidance to the Technical Team members of the IAEA Incident and Emergency System (IES) on the actions to be conducted and on what tools are to be used in different types of emergencies. The scope of these response flowcharts covers various emergency scenarios including:
- A dangerous source, possibly lost or stolen, damaged or disconnected;
- Contamination;
- Radiation exposure.

The application of a response flowchart for a given situation is completed with the use of one or several IAEA assessment and prognosis tools by the IAEA IES Technical Team. This set of tools include:
- The IAEA Radiological Source Assessment Tool, which has been designed to assist in the process of capturing essential information during an emergency involving one or more radioactive sources. This tool aims at helping an expert user categorize the sources involved, as well as assess the potential hazards associated with the category of the sources involved. The IAEA Radiological Source Assessment Tool implements the methodology outlined in the IAEA Safety Guide RS-G-1.9 [2] and in the EPR Series Publication, EPR-D-values 2006 [3].
- The IAEA Dose Assessment Tool, which has been designed to assist in the process of evaluating radiological consequences during an emergency involving potential exposure to one or more radiation sources. This tool aims at helping an expert user estimate the radiation dose to an individual in the case of external or internal exposure to radioactive material in predefined scenarios. The IAEA Dose Assessment Tool implements the methodology outlined in the IAEA-TECDOC-1162 [4].

The IAEA assessment and prognosis tools, as well as related training on their use, are made available to experts in Member States on request. The IAEA also encourages Member States to exercise the use of these tools during ConvEx-2e exercises.

There are various views in different countries on the use of atmospheric dispersion and dose calculation tools. To gather experts views on this area of expertise from various Member States and to participate in building an international consensus on this topic, the IAEA started a new Coordinated Research Project (CRP) on the Effective Use of Dose Projection Tools in the Preparedness and Response to Nuclear and Radiological Emergencies (CRP J15002). In January 2020, the IAEA hold the first Research Coordination Meeting of this CRP [5]. Most activities, which were identified to be conducted with the participating institutes, will focus on the use of dose projection tools in the urgent response phase. The urgent response phase is defined as the period of time, within the emergency response phase, from the detection of conditions warranting emergency response actions that must be taken promptly in order to be effective until the completion of all such actions. The urgent response phase may last from hours to days depending on the nature and scale of the nuclear or radiological emergency [6]. Establishing good practises, identifying limitations and uncertainties for the use of dose projection tools will better support the use of decision support systems and therefore, help manage exposure of workers and the public during an emergency.

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227-CONDUCT OF CONVENTION EXERCISE, CONVEX-2B DURING COVID-19 PANDEMIC

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The IAEA Response and Assistance Network (RANET) provides a compatible and integrated system for the provision of international assistance to minimize the actual or potential radiological consequences of a nuclear or radiological incident or emergency for health, environment and property. It also allows for advice and assistance to be passed on to a requesting State on response activities undertaken on the scene of an emergency, to mitigate its impact [1].

RANET includes State Parties to the Convention on Assistance in Case of a Nuclear Accident or Radiological Emergency who have identified national assistance capabilities that consist of qualified experts, equipment and materials, which could be made available to assist another State. These States are capable and willing to provide, upon request, specialized assistance by appropriately trained, equipped and qualified personnel with the ability to respond in a timely and effective manner to nuclear or radiological emergencies irrespective of their causes.

The IAEA has developed a set of exercises of various levels of complexity — called ConvEx (Convention Exercise) — to practise with Member States and relevant international organizations different arrangements within the international emergency preparedness and response framework. The ConvEx-2b exercise tests the arrangements for a request for, and the provision of, assistance. This type of exercise is conducted once a year on an announced date. The IAEA Incident and Emergency Centre (IEC) invites Competent Authorities (CAs) to participate and to coordinate the participation of relevant national capabilities in the exercise. The exercise specifically addresses the CAs that have registered RANET capabilities but also promotes the participation of other CAs to encourage them to join RANET [1, 2].

In March 2020, the IAEA held a ConvEx-2b exercise with the participation of 35 Member States and two Regional Specialized Meteorological Centres (RSMCS) of the World Meteorological Organization. The three-day exercise, from 24 to 26 March, was conducted while the responders in many Member States and in the IEC in Vienna worked remotely, according to measures taken to prevent the spread of the novel coronavirus COVID-19.

The IAEA Director General participated to the ConvEx-2b exercise and noted that "*We need to be prepared for the possibility that nuclear and radiological emergencies resulting from a safety or security event could be accompanied by natural disasters, pandemics or other crises. Conducting this exercise at a time when all of our lives are being seriously disrupted by the coronavirus crisis demonstrates our determination to maintain our emergency response capability, regardless of the causes and circumstances of any crisis, the IAEA will act quickly to coordinate an effective international response.*". This exercise provided participants a unique opportunity to test operational arrangements during an on-going pandemic situation such as COVID-19, where first responders might operate in an even more challenging environment.

The exercise tested the participant's efficiency and effectiveness in offering or requesting assistance. "Assistance Action Plans", prepared during the exercise, included immediate COVID-19 testing for the Field Assistance Teams on arrival and providing them with pandemic-specific personal protective equipment and on-going medical assessment while they deliver assistance.

Lessons learned from this exercise will be shared with Member States and International Organizations on the IAEA's Unified System for Information Exchange in Incidents and Emergencies (USIE) website.

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228-INTUITIVE ROBOT TASK PLANNING FOR ROUTINE MONITORING AND SURVEILLANCE IN NUCLEAR FACILITIES TO INCREASE RADIATION SAFETY

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Abstract. Today robots are widely used in the nuclear industry. Their main application is to perform automated and repetitive work or to execute hazardous tasks that are dangerous to human beings. Typical examples are manipulators used for maintenance in PWRs or mobile robots for post-accidental tasks. It may be of interest to determine the influence of such equipment on the availability of safety systems/functions and implications for nuclear safety in general (workers and public). Robots are used special purposes, where human factor problems are minimized. In this research, an intuitive Semi-Intelligent Mobile Observing Robot is proposed. The proposed system aims are to measure temperature and to retransmit televised views of the area then we can develop a contamination map of nuclear facilities. Equipment used includes radiation detectors, temperature sensors and a camera mounted on a telescoping mast. The proposed robot is programmed to be controlled remotely for safety reasons. The safety goals are two-fold; reduction of dose for the operator, and improved integrity of plant operation in such nuclear facilities.

1- INTRODUCTION.

The next third generation of robots are robots with intelligent control from computer. They are distinguished from second generation robots through a complexity of functions and perfectness of automatic control system including some or other element of artificial intellect. The intelligent robots are intended not only for imitation of physical action of the Man but rather for automatization of his intellectual activity, i.e. for solving production problems of intellectual character. Now in wide use are first and second generation robots. Third generation robots are in the stage of creation [1, 2].

Nuclear power engineering is historically one of the first industries where robots were introduced. Robot are used for fuel reprocessing in plants, during nuclear power plants (NPPs) refueling, equipment diagnostics, repair, in post-accident activity, during NPPs decommissioning. The use of robots is dictated by the necessity and reason to replace the Man where high radiation fields are present, aggressive media are used, compact layout of equipment, during fulfilment of actions which are not inherent to Man in his normal activity. All of the technologies require an interaction to accomplish a task, even if only to turn the equipment on and off. While in almost all situations, hands-on operation by the worker is acceptable, in certain conditions it is not. For example, when a work area contains a hazardous environment, such as a high radiation field or a chemically contaminated atmosphere, human presence should be limited to maintain safe operating conditions. For a manually operated system, limiting human presence means limiting operating time and thus productivity. Therefore, it is often desirable to provide equipment that can be operated from outside the hazardous environment to overcome these limits. This is the primary reason for using remotely operated equipment.

2- ENHANCED SAFETY, COST REDUCTION AND ACCESSIBILITY.

Because the areas are highly radioactive, workers who enter these fields can receive their legal dose limit in only a few minutes of work on the equipment, which means more workers to complete each task and an increase in overall dose burden. Remotely operated technology can minimize manual intervention time in these areas, thus enhancing safety and productivity. The use of remotely operated equipment can also result in cost reductions. In other words, since the remote equipment operating in the high radiation field replaces many human workers, the employment costs for those human workers are saved. Another aspect of remote operation for cost reduction is accessibility.

3- THEORETICAL WORK.

In order to assure a systematic approach, a program for application of robotics should be established and implemented to insure that this application retains or even enhances the safe operation of the plant. Application of robotics may vary significantly between cases. Safety and non-safety related applications are taken into account and should be clearly classified [3-5]. An intuitive interface for HRI based on AR has been developed in this research. Figure 1 shows an overview of the AR-based interface. HRI starts with the operator who deals with the image of the working environment displayed by the PC monitor. The operator defines the object, destination, and the tracking path by defining interesting points using the mouse/joystick on the screen monitor which displays the working environment including the robot. The robot is remotely located from the operator. The task planner system receives the operator's intention from the interaction with the image and outputs information and commands to AR (Augmented Reality) device and microcontroller respectively. AR device converts the information from the task planner to virtual graphics superimposed on the image from the stereo-camera. The operator obtains AR feed-back of the planned task. The operator is kept in a loop of task modification and virtual feedback until no modification is needed. The task is never executed until the operator confirms the task plan. The operator's commands are translated into control code which is sent to the microcontroller. The microcontroller sends signals to the robot to perform the planned task successfully. The task is performed under the supervision of the operator which monitors the working environment through stereo-camera. The research algorithms were fully implemented and programmed using Python with OpenCv and MATLAB/SIMULINK with computer vision toolbox.

4- SYSTEM DESCRIPTION.

Experimental work has been conducted to validate the research algorithms using test-rig shown in Fig. 1. The system consists of mobile robot system with stereo camera and the proposed detectors. The robot motions are achieved by using DC motors. The overall structure allows moving with detectors through the target nuclear environment and collecting the required data using well calibrated detectors. A stereo camera is used to capture the workspace. The life streams are sent to the PC which is capable of image processing and running of vision algorithms. All estimated information is sent to a microcontroller type Arduino board based on ATmega2560 microprocessor. The microcontroller uses the robot inverse kinematic equations to generate the required signals to move the robot joints for task performance.

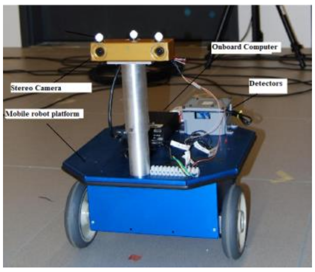

Fig.1 Overview of the proposed system

5- CONCLUSION.

To increase radiation safety, the remotely controlled robotic system is proposed in the handling of highly radioactive materials, the deployment of tools and sensors and the dismantling of components, labor-intensive processes that have the potential for high exposure rates, heat stress and injury to personnel. The proposed system provides solutions to these hazards. Such remote surveilling, detecting and handling systems are required to perform tasks within budget and on schedule while justifying the expense by saving in cumulative doses received by workers. To reach this goal, the following are additional factors that need to be evaluated when preparing a project:

- The operating and control system should be user-friendly. Controls should be well laid out, with ergonomics suitable for numerous personnel with differing levels of experience, and normal operations should be logical and easy to execute. System parameters and alarm indicators must be accessible and easy to evaluate and respond to.
- The equipment must be able to perform all tasks within its capabilities safely, effectively, and efficiently with little downtime and no failures that would jeopardize personnel safety or place the system or task in a non-recoverable position.
- The system must be flexible and easily adapted to changing conditions, tooling requirements, and operational needs.
- The system must truly be remotely operated. Adequate distance or shielding must be available to operators such that exposures to radiation, hazardous materials, and conditions are minimized
- The systems, if possible, should be able to perform remote tasks nearly as rapidly as conventional practices would allow or have the ability to perform tasks that would otherwise be difficult, impossible, or impractical to perform.

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230-CYTOGENETIC AND GENE EXPRESSION STUDY IN RADIATION WORKERS OCCUPATIONALLY EXPOSED TO LOW LEVELS OF IONIZING RADIATION

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SUMMARY

BACKGROUND: Exposure to ionizing radiation (IR) causes damage to living cells, especially to DNA, the degree of cellular damage depends on the radiation amount. Humans are naturally exposed to ionizing radiation from cosmic rays, and artificially through diagnostic procedures, medical treatments or occupationally during work shifts. Cytogenetic assay of peripheral blood lymphocytes can provide a biological estimation of the dose received in ionizing radiation exposures. The assay of cytokinesis-blocked micronucleus (CBMN) is widely used, since it represents a reliable test to detect radiation-induced chromosome aberration and it is a valuable biomarker in many biomonitoring studies on human populations occupationally exposed to ionizing radiation[1].The nuclear division index (NDI) is a cell proliferation marker in cultures which is considered a measure of general cytotoxicity, the relative frequencies of the cells may be used to define cell cycles progression of the lymphocyte after mitogenic stimulation. .Nevertheless the assay is frequently employed as a useful research tool for understanding the cell cycling kinetics of the cultures [2]. Many gene expression studies demonstrated an up-regulation of genes involved in the processes of signal transduction, control of cell cycle, DNA repair and apoptosis after exposure of ionizing radiation in different mammalian cell types [3].Altered expression of a few genes playing specific roles in DNA repair/cell cycle control such as *CDKN1A*, , *XPC*, *GADD45A*, *DDB2* and *PCNA* and cell cycle regulation/proliferation such as *IL16*, *CABLES2*, *TGFB2* and *RHOA*[3,4].

OBJECTIVE: The present study aims to use the MN, NDI and gene expression analysis as biomarkers for investigation of the effects of ionizing radiation exposure in some radiation workers occupationally exposed to low ionizing radiation in Al-Tuwaitha site, Baghdad/ Iraq. Also, assess the effect of ionizing radiation on the expression of some marker of genes such as: *RHOA, CDKN1A, GADD45A and RAD52*.

METHODS: This study was carried out on thirty Iraqi male radiation workers occupationally exposed to low levels of ionizing radiation at Al-Tuwaitha site, Baghdad, non-smokers and non- alcoholic, aged (30-59 year), as well as thirty apparently healthy individuals males collected randomly from population living Baghdad, aged (30-59 year) which are non-smokers, non- alcoholic as control group. Three molecular genetic parameter employed such as CBMN, NDI and gene expression assay. The MN and NDI assay were performed according to the description by IAEA, 2001[5].Gene expression assessments were performed by real-time according to the description by Livak and Schmittgen [6].Total RNA was isolated from blood for radiation worker and control groups.The RNA concentration was determined by measuring their absorbance that dependent on the ratio A_{260}/A_{280} of the wavelength, which leads to the determination of RNA purity, which ranged from 1.79-2.1 in two

groups. Using RT-PCR for study gene expression, four types of specialized primer genes were selected for the mRNA genes *RHOA, CDKN1A, GADD45A* and *RAD52* which have a relation with ionizing radiation in addition to the primers for internal control *Housekeeping gene* (*(β-actin*) was used as a reference gene to normalize the quantity of the target genes.

RESULTS: The results of cytogenetic analysis showed that MN frequencies was significantly higher ($p<0.01$) in the radiation workers group in Al-Tuwaitha site, as compared with the control group, and showed significant decrease ($p> 0.01$) in the NDI was observed in radiation workers in Al-Tuwaitha site, compared to the controls (Table1). Micronuclei are formed from lagging chromosomal aberration type fragments or whole chromosomes at anaphase which are not included in the nuclei of daughter cells [7,8].Table1 shows distribution MN in peripheral lymphocytes for some radiation workers occupationally exposed to low levels of ionizing radiation and control group .CB cells lymphocytes having one, two and three micronuclei are rendered evident to all radiation workers and some control .The nuclear division index as biomarker of cell proliferation in cultures which is considered a measure of general cytotoxicity and the relative frequencies of the cells may be used to define cell cycles progression of the lymphocyte after mitogenic stimulation and how this has been affected by the exposure of mitogenic material like ionizing radiation exposure [8, 9].

TABLE 1.FREQUENCY MN AND NDI (MEAN ± SE) IN PERIPHERAL LYMPHOCYTES FOR RADIATION WORKERS AT AL-TUWAITHA NUCLEAR SITE AND CONTROL GROUP

Study groups	No. of samples	Micronuclei frequency								NDI / 100 cell (Mean ± SE)
		No. of BN cells	Total of MN	MN/cell (Mean ± SE)	MN distribution in BN cells				Cells with MN	
					0	1	2	3		
Radiation Workers	30	15000	344	).0230 ± 0.0010 [a]	14683	294	19	4	317	.235 ± 0.0261 [a]
Control	30	15000	207	0.0138 ± .00009	14808	177	15	0	192	1.352± 0.0310

[a] *Compared with the control group,* $p < 0.01$.

Real-time PCR was used to confirm gene expression profiles for four mRNA genes RHOA, *CDKN1A, GADD45A* and *RAD52*, using the same RNA samples that were used for gene expression experiments. The expression analysis showed an up-regulation of three genes *RHOA, CDKN1A, GADD45A*, and a down-regulation of *RAD52* gene, relatively to control levels (Fig.1).These aspects need to be considered for assessing the expression of gene profiles showed in radiation workers. Several genes commonly associated with radiation response belong to many processes of DNA repair, stress response, signaling transduction and cell cycle/ proliferation after ionizing radiation. Altered expression of a some genes playing specific roles in DNA repair/cell cycle control such as *CDKN1A, DDB2, XPC, GADD45A* and *PCNA* and cell cycle regulation/proliferation such as *RHOA, CABLES2* , *TGFB2* and *IL16* [3,4].Several genes involved in cell cycle regulation and DNA repair were found to be significantly induced by radiation exposed to low levels of ionizing radiation in human.

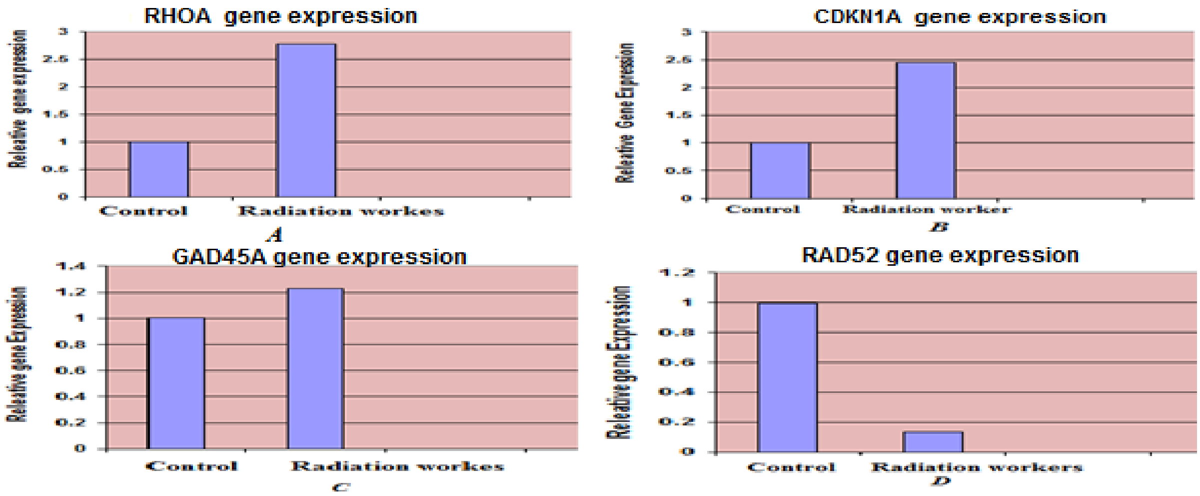

FIG. 1 Rt-PCR graphs showing the relative fold expression levels for RHOA(A),CDKN1A(B), GAD45A(C) and RAD52 (D) genes in peripheral lymphocytes for radiation worker and control groups.

CONCLUSIONS

The results indicated that there is a possibility of using the changes in the MN, NDI and genes expression such as *RHOA*, *CDKN1A*, *GADD45A* and *RAD52* as useful biomarkers for detection of radiation exposure in radiation workers occupationally exposed to low levels of IR.

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231-ASSESSMENT OF MEDICAL EXPOSURE OF PATIENTS IN INTERVENTIONAL CARDIOLOGY

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Medical exposure of patients varies for the same type of interventional cardiology procedure in different hospitals depending on various factors i.e. physician training, experience, type of angiography machine, patient size, complexity of the case etc. There may be chance of high radiation exposure to patients due to prolonged use of radiation during complex interventional procedures. To avoid unnecessary radiation exposure to patients in interventional cardiology procedures, diagnostic reference levels (DRLs) are needed to be established and used [1, 2]. Hospitals should perform periodic studies to determine reference levels in terms of dose-area product (DAP), number of images acquired and fluoroscopy time per procedure for patients and compare the same with the relevant DRLs. The procedures at hospitals may be improved by performing such periodic assessments [2, 3].

To assess the medical exposure of patients in interventional cardiology, a study was conducted by collection and analysis of patients' exposure data from hospitals. Data of 259 adult patients who underwent interventional cardiology procedures i.e. coronary angiography (CA) and percutaneous coronary intervention (PCI) was collected from five hospitals.

In this study, hospitals' reference levels and local DRLs have been determined in terms of dose-area product (DAP)/ air kerma-area product (P_{KA}), number of images acquired and fluoroscopy time per procedure. Furthermore, locally determined DRLs are compared with the DRLs suggested by IAEA [4, 5].

The locally determined DRL values in terms of dose-area product (DAP)/ air kerma-area product (P_{KA}), number of images acquired and fluoroscopy time per procedure are 48 Gy-cm^2, 515 and 5 minutes respectively for CA procedure; and 138 Gy-cm^2, 1361 and 12 minutes respectively for PCI procedure. Comparison of local DRLs with those suggested by IAEA is shown in Fig. 1.

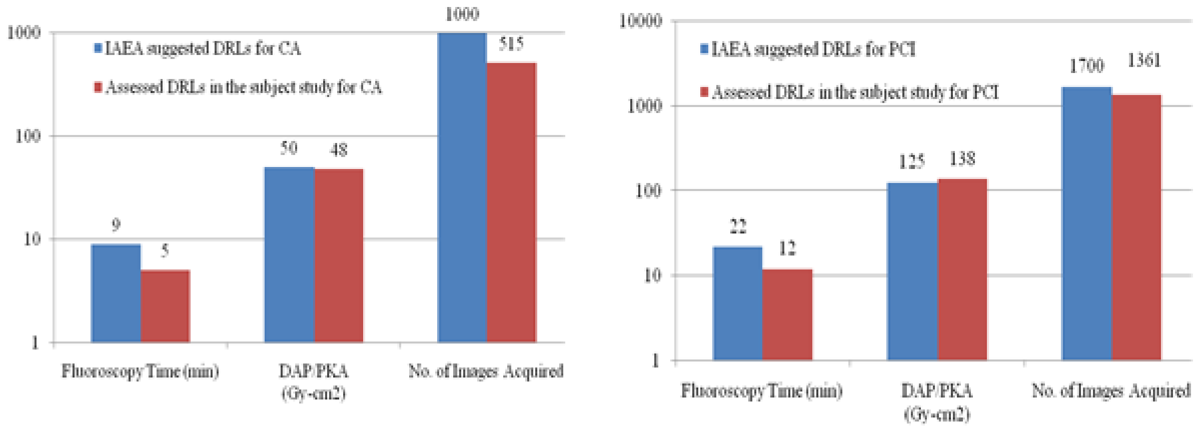

FIG. 1. Comparison of DRLs for CA and PCI Procedures.

In this study, it is found that the locally determined DRL values for CA and PCI procedures are lower than those suggested by IAEA except DAP/P_{KA} value for PCI procedure. CA procedure is frequently performed procedure (i.e. 66% of total procedures). It is necessary to collect large scale patients' data from hospitals to establish DRLs at national level.

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232-STUDY OF RADIATION RISK IN MEDICAL CYCLOTRON FACILITIES AND GENERAL APPROACH TOWARDS OPERATIONAL RADIATION PROTECTION

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ABSTRACT

The use of charge particle accelerators is well-known in medical, industry, research and education. Medical cyclotrons are widely used for diagnostic and therapeutic purpose [1]. Currently, there are around 1300 big and small medical cyclotron radiation facilities in the world with beam energy ranging from few MeVs to several MeVs [2]. Due to diverse applications, the demand of cyclotrons is annually increasing. However, there are many regulatory challenges related to shielding, operation, QC, repair & maintenance and management of radioactive waste [1]. Therefore, this study was conducted to make necessary measurements, analyze the data and recommend such way forward for improvement of regulatory practices that may help in the enhancement of radiation safety at medical cyclotron facilities.

This research comprises of experimental studies carried out on shielded and unshielded cyclotrons installed in Pakistan. The study includes assessment of neutron field inside and outside medical cyclotron vault [3], identification of radioactive byproducts by medical cyclotrons used for ^{18}F production [4], evaluation of radiation shielding [5], design of medical cyclotrons and assessment of radioactive impurities in the production of [18F]FDG (Fluorodeoxyglucose) [6]. TLDs 600 & 700 were used for neutron dose while HPGe detector was used for gamma spectrum analysis. Theoretical calculations were carried out using different computer codes and applications. The measured results indicated the neutron dose inside the cyclotron vault and indirect activation of cyclotron components with ejected neutrons (2-4 MeV) [3]. The present results are in line with the earlier investigations. Based on the results, solutions to regulate radiation safety practices in shielding, operation, QC, repair & maintenance and management of radioactive waste at medical cyclotron facilities were proposed [7].

As limited literature is available related to activation of cyclotron components, decommissioning strategies, and lessons learned, therefore, this experimental investigation may be useful for the scientists and researchers engaged in nuclear industry, particle accelerators, long-lived radioactive waste management and shielding & design. Based on the recommendations, radiation exposures and personnel doses can be optimized with the careful selection of energy, projectile, target, and components.

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235-THE IMPORTANCE OF INCLUDING THE HUMAN FAILURE AND RISK MANAGEMENT IN TRAININGAND EDUCATION PROGRAMS TO AVOID RADIOLOGICAL INCIDENTS OR ACCIDENTS

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INTRODUCTION

In the last decades there have been numerous efforts and contributions to improve and integrate various issues related to radiation safety training programs and education of occupationally exposed workers. The implementation of education and training activities in radiation safety needs to continue and facilitate the inclusion of new approaches and developing and implementing different mechanism and strategies to strengthen capacity building.

International information concerning the occurrence of radiological events, show that even after this effort, new incidents still occur reported by countries. As examples, we have the International Nuclear Event Scale (INES) [1], as a communication tool about safety significance and Safety in Radiation Oncology (SAFRON) for medical purposes.

For this reason, is necessary to implement a strategy to update and change training programs effectively, incorporating models that include deep concepts of communication and perception of radiological risk. Among them, vulnerability analysis and risk management must be taken into account and other educative tools for the assessment of training needs. [2]

TRAINING PROGRAMS

The current training programs develop in their content, a diverse variety of topics that include an extensive agenda on radiation protection and physical security, which allows educating users of radioactive material.

However, through international experience and lessons learned from radiological incidents and accidents, this training should be increased, extending the concepts towards the comprehensive management of radiological risk. Among them:

— *Risk management;*
— *Prevention ethics;*
— *Risk communication;*
— *Risk analysis; Self-evaluation;*
— *Mechanisms or methodologies for detecting and correcting anomalies or deficiencies;*
— *Audits. Evolution towards continuous improvement.*

This group of preventive concepts, in general, should be implemented to complement the content flow of radiation protection programs.

SWOT analysis (Strengths, Weaknesses, Opportunities and Threats) can offer a methodology to help in the detection of anomalies or deficiencies and contribute to apply risk perception concept and implement corrective actions.

THE RISK DETECTION PROCESSES

The risk detection mechanism must involve, within the training programs, at least two different aspects, which can lead to a radiological risk situation:

Internal Context

The vulnerability of an installation, as well as the accident factors, is detectable thanks to the evaluation carried out through safety programs, generating defense mechanisms that help to detect and correct deviations in a preventive manner. Lists of personal aspects that allow us to evaluate the causes or factors that originate a low level of risk perception are: *cultural issues, human factor, social and personal values, changes in attitude, evolution resistance, frequency of deviations or anomalies detected by the management system, personality, commitment level.*

External Context

These factors are related to the organization of the company and their detection and correction will directly impact the main objective of risk management: *prevention.* [3], [4]. Among them we can mention: *lack of specific training, poor conditions of safety during operation, work pressure, complex work environment, vulnerability of the devices that contain radioactive sources, decisions based on economic aspects and not on safety aspects.*

METHODOLOGIES SUGGESTED FOR RISK REDUCTION

By introducing into the training programs, the inclusion of the radiological risk reduction system, as a primary concept, [4] is including a set of measures that can be defined as:

— **Preventive measures** *(risk communication, risk management, prevention ethics, risk analysis and errors detection and correction mechanisms, Self-evaluation)*
— **Personal measures** *(aptitude, attitude, commitment, risk perception, training, human factor)*
— **Organizational measures** *(procedures, work guides, performance evaluations, audits, evolution towards continuous improvement)* [2], [3], [4].

CONCLUSIONS

The use of educational methodologies on reducing radiological risk could help to avoid failures or errors as follows:

— External agents that can cause incidents (lack of attention) are minimized;
— Lack of application or follow-up of procedures during the operation;
— Deviation, anomalies and underestimated errors during practices;
— Daily routine events not reported;
— Improper handling of devices that contain radioactive material;
— Failure to not use suitable radioprotection instruments or their absence during the job;
— Lack of motivation;
— Ignorance of the risks of the task;
— Absence of good practices during the job;
— Leadership failures ;
— Lack of staff trained in radiological emergency response;
— Lack of attitude in their usual task;
— Decisions that take into account only economic issues minimizing radiation protection and safety;
— Work under extreme pressure.

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237-PRELIMINARY SURVEY FOR THE ESTABLISHMENT OF REGIONAL DIAGNOSTIC REFERENCE LEVELS IN ADULT COMPUTED TOMOGRAPHY FOR FOUR AFRICAN COUNTRIES

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BACKGROUND

This study reports nationwide survey embarked on by four countries: Ghana, Kenya, Namibia and Senegal, to establish National DRLs in their respective countries and provide current data for the establishment of regional diagnostic references levels for the most commonly performed adult CT examinations. Governments through regulatory bodies, professional bodies and health authorities have the responsibility to ensure that DRLs are established for the country [1]. Frequency of CT examinations has increased significantly [2, 3] and is known to be a significant contributor to individual and population dose [3, 4]. Egypt is one of the few countries in Africa that established and implemented national DRLs [5].

MATERIALS AND METHODS

Computed tomography head (16-cm diameter) and body (32-cm diameter) PMMA dosimetry phantoms were used in the study to represent an adult head and body respectively. In all, data was collected from fifty-four (54) CT facilities in the four countries using a structured questionnaire provided by the International

Atomic Energy Agency (IAEA), recording for each patient examination parameters displayed on the console of the CT scanner. Dose assessment was undertaken in terms of $CTDI_{vol}$ and DLP for head CT, chest CT and abdomen CT examinations using standard methods, and values compared with console values. Experienced radiologists graded the diagnostic image quality of the radiographs using a stratified random sampling. Median DLP and $CTDI_{vol}$ data from each facility were calculated to estimate the typical dose in each country. The national DRLs were set based on the third quartile values of distributions of median for the $CTDI_{vol}$ and DLP values from contributing facilities. The results were compared with international published data. Average of the national DRLs was used to establish regional DRLs for Africa from the four African countries.

RESULTS

Comparison of measured $CTDI_{vol}$ with console values of all facilities in all four countries were within 20% as recommended. Using the 75th percentile of the distribution of the four countries, the range of the median values of $CTDI_{vol}$ were (47.1-56.5 mGy), (9.85-13.5 mGy) and (10.5-12.6 mGy), for head CT, chest CT and abdomen CT respectively. Similarly, the range of the median values of DLP were (277-1394 mGy.cm), (277-575 mGy.cm) and (495-875 mGy.cm) for head CT, chest CT and abdomen CT respectively. The established $CTDI_{vol}$ DRLs for head CT, chest CT and abdomen CT were 63.9, 14.7 and 13.9 mGy respectively. Similarly, that of DLP DRLs were 1251, 672 and 795 mGy.cm respectively for head CT, chest CT and abdomen CT.

CONCLUSION

Baseline regional diagnostic references levels for the most commonly performed adult CT examinations in four African countries has been established. The DRLs were comparable to other DRLs from other countries, although there were some variations. This DRL data would serve as baseline data for adult CT DRLs in African countries.

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239-EFFECTIVITY EVALUATION OF THE CURRENT PERSONAL PROTECTIVE EQUIPMENT IN INTERVENTIONAL RADIOLOGY

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INTRODUCTION

X-Ray techniques are key tools for interventional radiologist at diagnostic and therapeutic procedures. It is linked with negative effects of ionizing radiation. It is well-known that already a small amount of ionizing radiation could cause molecular changes which can lead to tissue damage [1]. There are certain health risks related to the interventional radiologist`s work, such as cataracts, left-sided brain tumours or cardiovascular diseases [2]. It is therefore necessary to follow ALARA principles at interventional radiology departments. The ALARA principles state to keep the exposures to radiation as low as reasonably achievable [1]. Shielding plays an important role in decreasing the exposure to ionizing radiation. This depends on the architecture of the workplace, installed protection devices and personal protective equipment (PPE). Lead apron (LA) is one of the most common PPE [3]. The robust construction of these LAs causes an increased load for the musculoskeletal system, what can lead to an unexpected career end of an interventional radiologist [4]. The Zero-Gravity suspended Radiation Protection System (ZG), from Biotronik company, appears to be an adequate alternative to the commonly used LA. This system prevents the operator from the orthopedic injuries and provides more effective protection against the ionizing radiation. The advantage of the system is the fact that it was designed to provide a protection for the complete head part in all working positions, without visibility or movement limitations while in use. ZG utilizes a suspended 1.0 mm lead body shield that attaches magnetically to the vest worn by the operator [5]. It is possible to achieve a reduction of the overall operator´s radiation exposure by 87-100% if the ZG is used correctly [4]. The aim of the study is to summarize and evaluate the ZG effectivity in comparison with the commonly used LAs.

METHODS

Prospective pilot study was performed on the selected interventional radiology department in Slovak republic. Based on the pilot study, 3-year multicenter cohort study is running currently. An interventional radiologist has been chosen for the study, who performed 32 endovascular procedures: 18% percutaneous transluminal angioplasty (N=6), 41% digital subtraction angiography (N=13), and 41% peripheral vascular intervention (N=13), while his radiation exposure was monitored. Half of the procedures was realised using two-piece LA (vest and skirt), collar and glasses with lead equivalent. The other half of the procedures was realised using ZG, collar and glasses with lead equivalent. Monitoring of the operator radiation exposure was done by two personal dosimetry devices, type RaySafe i3. Both dosimeters were placed on both sides of the reference point, on the left-hand side of the chest. One of the dosimeters was placed on LA/ZG and the other one underneath LA/ZG. The patients` radiation exposure data were collected during the interventional procedure from the picture archiving and communication system. The research was done in compliance with the General Data Protection Regulation (GDPR). Wilcoxon test was used for data analysis, while the level of significance was set to 5% (0.05%).

RESULTS

Total patient entrance dose median was at DSA 421.3 mGy (range 17.6 – 1575 mGy), at PVI 146.5 mGy (range 46.5 – 934.7 mGy) and at PTA 194.25 mGy (range 12.5 – 2768 mGy). Cumulative dose median of the radiologist, measured on the PPE, was at DSA 0.05 mSv (range 0.01 – 0.31 mSv), at PVI 0.03 mSv (range 0 – 0.13 mSv) and at PTA 0.055 mSv (range 0.01 – 11.75 mSv). Cumulative dose median of the radiologist, measured underneath the PPE, was at DSA 0 mSv (range 0 – 0.07 mSv), at PVI 0 mSv (range 0 – 0.03 mSv) and at PTA 0 mSv (0 – 0.64 mSv).

Due to various time and technical requirements of the individual procedures, the operator was exposed to a higher cumulative dose when he used LA compared to ZG. The 75th quartile of cumulative dose of the operator in single interventional procedures was 0.05 mSv on ZG and 0 mSv was underneath the ZG. This has proved 100% radiation capture. The 75th quartile of cumulative dose of the operator in single interventional procedures was 0.11 mSv on LA and 0.03 was underneath the LA. This has proved 73% radiation capture. (Table 1, Fig. 1.)

TABLE 1. RADIATION CAPTURE COMPARISON IN SINGLE INTERVENTIONAL PROCEDURES: ZG VS. LA

	Zero-Gravity system (ZG)		**Lead apron (LA)**	
Cumulative dose (mSv)	Median±IQR	The 75th quartile	Median±IQR	The 75th quartile
Dosimeter placed on PPE	0.03±0.04	0.05	0.05±0.09	0.11
Dosimeter placed underneath PPE	0±0	0	0±0.03	0.03
Fluoroscopy time	9m 11s ± 3m 9s	10m 52s	7m 23s ± 5m 48s	10m 25s
Total patient entrace dose (mGy)	153.1 ± 162.83	240.8	388.5 ± 740.93	967.78
Dose area product (μGy.m^2)	3234.75 ± 1919.65	4084.38	6846 ± 16046.23	20249.75

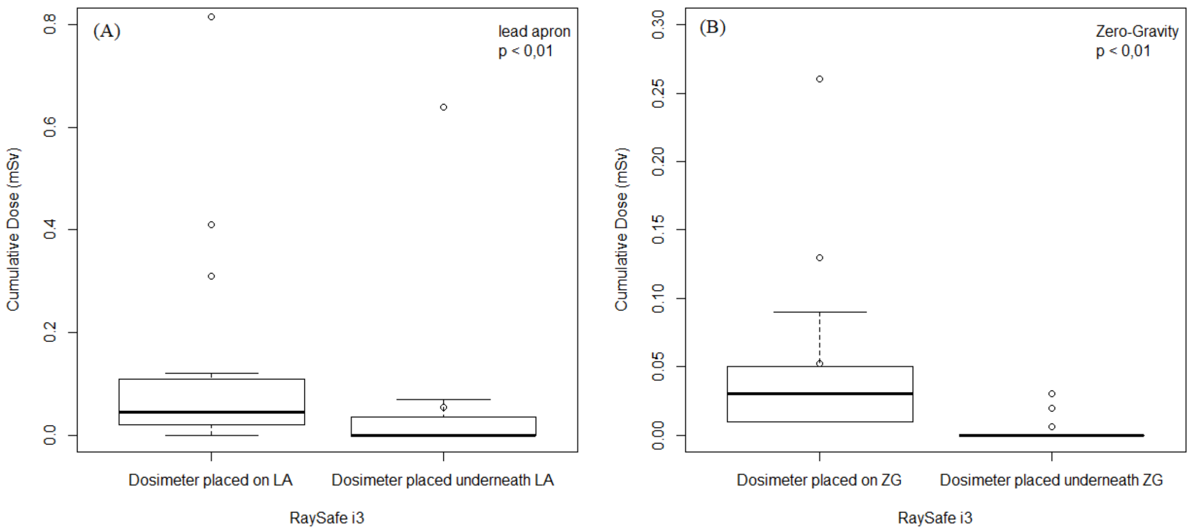

FIG. 32. Radiation capture effectivity comparison of LA vs. ZG in single interventional procedures

CONCLUSION

Based on the preliminary results of the study and scientific literature, it may be assumed that if ZG is used, radiation exposure of the operator is significantly lower compared to LA. ZG represents an adequate alternative to PPE in department of interventional radiology. In addition, it is important to notice that all our conclusions are preliminary only and the study is in the data-collection phase. Results of the study will be considered as a contribution to the radiation protection and orthopedic conditions improvement of operators.

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240-ENHANCING PREPAREDNESS IS KEY FOR POST-ACCIDENT RECOVERY FIRST CONCLUSIONS FROM DEDICATED OECD/NEA EXPERT GROUPS

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Large nuclear accidents cause major changes that are accident-specific, environment-specific and land-specific and are evolving in space and time. These changes impact the lives of a large number of residents of the affected territories by impacting the environment and disrupting the socio-economic fabric of the society, making recovery management an extremely complex process of a multidisciplinary dimension. A key aspect that has been identified to improve the recovery process after a nuclear accident by the Nuclear Energy Agency Committee of Radiological Protection and Public Health (CRPPH) is to advance preparedness, focusing on a holistic multidimensional approach, that will incorporate functional cross-sectoral links between various aspects of emergency impact on a society (*e.g.*, health, environment, economy, social and cultural aspects).

This topic is addressed by the Expert Group on Recovery Management (EGRM) created in 2019, by producing guidance on how to develop a nuclear or radiological post-accident recovery management framework which can be adapted to national conditions and as such, can contribute to assist NEA member countries to plan and improve their preparedness for response and for medium to long-term recovery.

At the same time, the Expert Group on Non-radiological public health aspects of Radiation emergency planning and response (EGNR) is focusing on how to consider mental health and psychosocial impact of the emergency and protective actions, and develop tools for mitigation of these aspects. It has been reported in numerous studies that Chernobyl and Fukushima NPP accidents have resulted in non-radiological health risks associated with sheltering, evacuation and/or relocation of vulnerable groups (*e.g.*, among patients evacuated from health care facilities and nursing homes), including a wide range of public health and social issues [1][2]. A multi-disciplinary approach with a broader stakeholder involvement is essential to shift from a radiological protection strategy to a more holistic view of health protection, including mental health and psychosocial support [3]. This should benefit from an advanced preparedness, based on the World Health Organisation (WHO) recommendations elaborated in “A framework for mental health and psychosocial support in radiological and nuclear emergencies” [4].

The presentation will build on the main findings of EGRM and EGNR’s work and the outcomes of international events as follows:

— An international workshop held in February 2020, dealing with "Preparedness for post-accident recovery process: lessons from experience", jointly organised by OECD/NEA and the Japanese Nuclear Regulatory Authority [5], and;
— Two web-based workshops held in June and July 2020, on "Non-Radiological Public Health Aspects of Protection Strategies in Radiation Emergencies", jointly organised by OECD/NEA with BfS (Germany) and WHO [6].

Among the shared lessons from these events, the presentation will illustrate how thinking in advance globally (*i.e.* in a holistic and multi-sectorial manner, balancing health, social, cultural, economic, environmental impacts) will aim to ensure that emergency response strategy will tackle the emergency situation and will not delay or impede the recovery process. Preparedness strategy should include actions targeting the resilience of societies and engaging local communities, as described in the recent WHO framework [4] and in the up-coming ICRP publication dealing with radiological protection of people and the environment in the event of a large nuclear accident [7]. Preparedness for post-accident recovery would benefit from adopting a comprehensive and operational generic framework covering key aspects such as *e.g.*, public health, radiological monitoring and dose assessment, risk communication, decommissioning and environmental decontamination (both strongly associated with waste management), food and drinking water management, business continuity, well-being of concerned people and communities. Finally, the new idea of exercising Post-Accident Recovery Management to practice and evaluate the effectiveness and efficiency of stakeholder involvement, and/or of any other issues at stake for recovery, will be discussed.

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241-OPTIMIZATION AFTER MINIMIZATION: SQUARING THE CIRCLE IN RADIATION PROTECTION

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Optimization is one of the pillars of the ICRP Radiation Protection System. It is often seen as the synonym for ALARA, the famous acronym of “as low as reasonably achievable”. Since its introduction with the ICRP Publication 26 in 1977 [1] the original text was modified several times by adding further side conditions in ICRP 60 [2], in the IAEA-BSS [3] and in the EU-BSS [4]. All these modifications did not “do more good than harm” to the radiation protection community. Unfortunately, not only the optimization principle was complicated, also other recent publications of ICRP made the system of protection more stringent and unnecessary complicate, e.g. the new environmental radiation protection concept and the new limit for the lens of the eye.

In reality when talking about optimization the focus is too often laid to the “low” in ALARA and the result is minimization and not optimization. This was neither the intention of ICRP, nor is it a wise use of resources.

In a recent report of UNEP [5] some figures are given about the worldwide situation in radiation protection. These figures reveal that the exposure of occupationally exposed workers is very low, a tiny piece of the limits, and the exposure for people of the public is a very small percentage of the limit as well as of the natural background.

TABLE 1. EXPOSURE DATA acc. to [5]

workers	average dose [mSv/a]
medicine	0,5
nuclear industry	1,0
industrial use	0,3

It was mentioned in this report too, that for doses below 100 mGy no radiation effect was observed. So, summing up this situation one can say that radiation protection is minimized and any further reduction of doses makes simply no sense. It would be a waste of resources without any benefit. As we are minimized with respect to doses it will not be possible to optimize. The result of a true optimization would be that higher doses might come out. But this is inacceptable for authorities and the public. Thus optimization after minimization is just like “squaring the circle” and simply impossible.

So, it is time to stop further minimizing of exposure and define a clear cut-off for what is called optimization, declaring a radiation protection situation as optimized when

- the dose for occupationally exposed persons is less than 1 mSv/a (measured does),
- the dose for members of the public is less than 100 µSv/a (calculated dose).

In this way we could stop the waste of resources and come back to a reasonable radiation protection.

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242-HOW TO IMPROVE THE COMMUNICATION OF THE RADIATION PROTECTION SYSTEM TO STAKEHOLDERS AND THE PUBLIC

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The radiation protection system presently in use has been developed, improved and refined over time. It has been gradually adapted to different exposure situations for medical patients, the public and professionally exposed persons. People working in the field of radiation protection are increasingly confronted with criticism from the public, stating that the radiation protection system is hard if not impossible to understand. Even specialists, when trying to explain the radiation protection system, realise and experience themselves the complicated nature of the regulations.

The Swiss Nuclear Safety Inspectorate (ENSI), as Switzerland's competent authority in the field of radiation protection and emergency preparedness and response in nuclear power plants, is also playing a major role in the communication of actions in radiation protection, of radiological risk and protective measures. Over the years, this communication between ENSI, stakeholders and the public became increasingly difficult and challenging. What appears as a comprehensible and scientifically sound communication for one side proved impossible to accept or understand by the other side.

In order to bridge the gap in understanding between the public and the specialists and try to clarify the issues, ENSI organized an international conference "on emergency preparedness and lessons learned from Fukushima and Chernobyl" in March 2020. Due to the Sars-Cov-2 pandemic, this meeting had unfortunately to be cancelled at the last minute. Based on the preparational work for the cancelled conference, the present paper aims at the precise identification of the communication problems mentioned above and proposes possible solutions.

As it turns out, it is extremely difficult for the "general public" to understand some of the fundamental principles of radiation protection:

Within the basic principle of limitation, the difference between dose limits, dose constraints, reference values or dose criteria for design basis or for example activity limitation is hard to understand and hard to put into a practical context. Values denominating upper dose or activity limits are easily misinterpreted as a distinct border between harmless and harmful. Given this public perception, dose "limits" for different time intervals, e.g. 1 mSv per year in normal, planned exposure situations or 100mSv in the year following an accident in an emergency exposure situation, raise the question as to why the accident limit can be 100 times higher than the one for normal circumstances.

Every planned practice and also every measure taken during an emergency shall do more good than harm (the principle of justification), which seems easy to understand, but its practical implementation is barely comprehensible for a layperson and even controversial among experts. As an example for this discussion, the question whether the evacuation after the Fukushima accident retrospectively did more harm than good is an ongoing controversy.

How could one describe the difference between the significance of a dose *attributed* to an individual (mostly retrospectively) by means of dosimetry or a well-established individually calculated dose assessment on one side and a prospective generic dose *estimate* for the most affected person based on conservative worst-case scenarios on the other? This misunderstanding is especially problematic, since the conservative generic dose estimates for the most affected person can easily (but misleadingly) be projected to huge numbers of people in order to create fearmongering scenarios with thousands of cancer deaths as a potential consequence of a nuclear accident.

There is also the everlasting and always current issue of calculating, applying and interpreting risk. Here too, the public is not aware of the different significance and purpose of an *attributed* risk to an individual based on a significant dose and an *inferred* risk to an individual in the low-dose region below 100 mSv, where the risk is estimated by a theoretical, extrapolated dose-risk model.

What could be the causes for these frictions?

The ICRP radiation protection system is based on a sound scientific and coherent foundation, well elaborated over the years, but the terminology used *is difficult to understand* for the general public (and sometimes even for experts) and only partially suitable for communication.

The knowledge about radiation protection and its scientific foundation is limited in the general public because it is nearly inexistent in general education. In the occurrence of an event triggering a sudden growing interest and a surge for knowledge, people seek information from every possible source whether trustworthy or not; this promotes the creation of half-knowledge, confusion and anxiety.

Various scientific disciplines (medicine, radiation biology, physics, communication etc.), authorities and experts are involved in radiation protection and they do not always speak the same language. Additionally, they represent their own opinion formed on their different approaches. This makes a uniform, coherent communication difficult and destroys the trust of the general public in these authorities and experts.

The impact of the loss of trust in experts and authorities is especially obvious during times of high media interest or pressure, e.g. after an event; in such situations, claims and myths are made or suggested in the classic media and, nowadays, above all via the social media. The authorities and experts are lagging behind and are forced into reacting instead of acting. Some organisations and "self-proclaimed experts" play themselves as advocates of the population. With the lack of a clear, prepared strategy of the relevant authorities, this situation leads at worst to a total loss of trust, confusion and fear in the public, which will cause more harm than the radiation itself.

So, how to avoid such a scenario? What could be possible solutions? In our view, this requires three closely interrelated measures, namely:

– Increase the understanding of the public for the radiation protection system also in everyday life and not just under the pressure of an event;
– Discuss the emergency preparedness measures with relevant authorities and stakeholders by strictly applying the justification principle (doing more good than harm, *example: evacuation vs. sheltering*) now and on an ongoing basis, to create a fundament for a uniform and coherent point of view for any occurring event;
– Build and maintain the trust of the public and the stakeholders now and on an ongoing basis, as waiting for an event to occur for starting it is leaving it to late.

And first of all, this requires a review of the international radiation protection system - not primarily with regard to the scientific basis and its implementation, which seems to be quite sound, but with regard to an improved communicability and comprehensibility of the existing concepts and principles for laypeople. This includes the use of a terminology that is as simple, unambiguous and generally understandable as possible without degrading the scientific background. Above all, the concerns and fears of stakeholders and the general public must always be taken seriously.

Based on this, ENSI intends to initiate a national multidisciplinary discussion between experts from different authorities to obtain a comprehensible, uniform strategy for the information and communication, which is adapted to the targeted audience.

243-A STATE-OF-THE-ART DOSE CALCULATION TOOL FOR THE CONSEQUENCE ANALYSIS

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The European Spallation Source (ESS) is one of the largest science infrastructure projects being built in Europe today. Based in Lund, Sweden, its purpose is to generate neutron beams for science. Lloyd's Register is supporting ESS to perform consequence analyses and to estimate activity transport and doses to the public. This is one of the requirements imposed by the Swedish Radiation Safety Authority (SSM) for the ESS facility. To demonstrate compliance with acceptance criteria issued by SSM, a state-of-the-art tool for dose calculations –ESS Doctor – has been developed by Lloyd's Register. Based on various accident sequences, ESS Doctor was developed for calculating activity concentrations and doses to workers and to the public, and to produce lists of the nuclides that dominate in contributing to the doses. This extended abstract presents the methodology adopted for the dose calculations in ESS Doctor. An overview is given of the development and the analysis steps; for example, accident sequence characterization, modelling, calculation of activities and doses, identification of dominating nuclides, and contributions in terms of inhalation, external cloud, external ground, and ingestion. ESS Doctor can also be utilized in Emergency preparedness by calculating doses during an emergency.

INTRODUCTION

The radioactive material that may be available for release and emission to the environment, following an accident is located in systems and rooms surrounded by other rooms and spaces. The release path to the outside through the facility may thus be fairly complex, and depending on the volumes and flows, the transport and emission of the activity may take some time. Most of the analysed accident scenarios have very short delays between the initiation of the event and the start of the emission of activity to the environment, and thus also a very short time to actuate any mitigating actions for the public. Doses are calculated for a representative person of the public, which is an adult and a one-year old child living on a farm approximately 300 m from the main stack of the facility. The exposure paths include inhalation, external radiation from the radioactive cloud passing the representative person (cloud shine), external radiation from the activity deposited on the ground when the radioactive cloud passes (ground shine), and ingestion of food and water contaminated by the emitted radioactivity. This is comprehensively described in an ESS report [Spanier L., 2019]. For the doses to the workers, the time-averaged activity concentration during the exposure time is also calculated by considering the activity and half-life of the nuclide released from the source; and performing transport and decay calculations as the nuclide passes the different rooms in the facility. Doses are calculated for four different emission heights e.g. 10 m, 20 m, 30 m and 45 m (main stack height) and two different weather situations i.e. median weather, P50%, and the 95 % percentile weather, P95%. The doses calculated with the P95% weather is used when comparing with the acceptance criteria for the different accident scenarios. The P50% weather is given for comparison with earlier results.

DEVELOPMENT OBJECTIVES

The development of the new dose calculation tool for ESS was based on the methodology adopted in the dose calculations currently performed for ESS [Spanier L., 2019]. The methodology rests on a solid theoretical foundation, has been accepted by SSM, and is therefore used as a roadmap for the development. The calculations have traditionally been performed using multiple applications – Mathcad for transport and decay calculations, and several Excel workbooks for calculations of doses and dominating nuclides; requiring detailed knowledge of Mathcad modelling and data processing in and between Excel sheets. The objectives for developing ESS Doctor were to automate many steps and reduce the large number of manual actions needed to obtain the final results, reduce the susceptibility to manual errors, increase user-friendliness using an intuitive interface, reduce the time needed to obtain results, facilitate calculations by ESS, and facilitate traceability.

ANALYSIS PRINCIPLES

The initial work aimed at creating a beta version of the program that included all calculation steps needed to obtain the doses, and a graphical user interface that guides the user through the different steps:
Step 1: Entering or reading volumes, flows, activity source parameters, and other input. Presentation of the input as perceived by the program.
Step 2: Performing transport and decay calculations – solving a system of differential equations for numbers of nuclei and activities in different volumes, and emitted to the environment, as functions of nuclear half-lives.
Check of calculation stabilities.
Step 3: Establishing functions for activities as results of nuclide half-lives.
Step 4: Establishing data needed for the dose calculations – scaling of inner source terms with given activity source data. Calculation of activities in rooms, and released to the environment, for the final dose calculations.
Step 5: Calculation of doses to workers and/or the public. Sorting for dominating nuclides. Compiling and exporting results and applied inputs.
The program also provided the possibility to change other inputs like dose coefficients, nuclides to calculate the dose from, and the volatility/chemical state of different chemical elements (typically Gas, Volatiles,
Aerosols, Others). Such data can be placed in external files.

REALIZATION OF THE TECHNICAL SPECIFICATION – ANALYSIS EXAMPLE

A number of different hypothetical accident scenarios were considered to formulate a detailed technical specification for the development of the tool, and to verify that the outcome, e.g. ESS Doctor, would produce results as expected for these scenarios [Klug J., 2019]. This section accounts for the actual outcome of the development, by showing how the tool appears to the user and by elaborating in more detail on the five analysis steps. A hypothetical accident scenario is used as an example; with the leak path LP-101B, i.e. leak through the beam pipe, the accelerator tunnel and the klystron galleries, ending in a release to the environment via the stack; mitigated after 40 seconds. The accident includes four different activity sources with different amounts of radioactive material and release dynamics. Material from the helium coolant and the filter particles are released during the first ten seconds, From the moderator water the duration of the release is 20 000 s, and from the tungsten target it is 1000 s. The activity release dynamics are different enough for the four sources to be handled in different calculations. However, since the helium coolant and the filter particles have the same release time, three separate transport calculations have to be done (instead of four).

BENCHMARKING AND CONCLUSION

For five accident scenarios, benchmarking calculations [Klug, J., 2019] have been performed with ESS Doctor version 1.0 against calculations performed previously, utilizing Mathcad 15.0 and Microsoft Excel. The five tests represent a range of different analysis prerequisites that ESS Doctor must be able to handle. The major findings in the benchmarking come from a very complex accident and leak path with eight different activity sources releasing radioactive isotopes, and where one of the sources becomes active relatively late after the beginning of the accident due to deflagration of hydrogen.
The final results in all benchmark tests for public doses, worker doses, dominating nuclides and committed effective and equivalent doses from iodine to the thyroid gland for a one-year old child have differences in the sub-percent range when compared to previously obtained results. The conclusion is that ESS Doctor adopts the established methodology and reproduces the results much faster, and as perfectly as can be expected. The same approach can be utilised in Emergency dose calculations.

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244-CONTINUOUS DRINKING WATER MONITOR FOR KEY GAMMA NUCLIDE

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Many facilities discharge low levels radioactive nuclide to environmental such as rivers, lakes, etc., which would become drinking water. Reasonable continuous monitoring of these radioactive nuclides are useful for drinking water safe. A prototype of drinking water monitor base on gamma nuclide monitoring was developed. It is designed and optimized. It is lighter weight compare with traditional continues radioactive water monitor. The monitor had been tested hundreds hours, results shows it is stable and reliable, has the ability of radionuclide identification

245-IDENTIFICATION AND ASSESSMENT OF THE HAZARDS IN A NUCLEAR FUEL FABRICATION FACILITY (NFFF)

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In uranium fuel fabrication facilities, large amounts of radioactive material are present in a dispersible form. This is particularly so in the early stages of the fuel fabrication process. In addition, the radioactive material encountered exists in diverse chemical and physical forms and is used in conjunction with flammable or chemically reactive substances as part of the process. Thus, in these facilities, the main hazards are potential criticality and releases of uranium hexafluoride (UF_6) and (U_3O_8), from which workers, public and the environment should be protected. In nuclear fuel fabrication facility, the process for the obtainment of U_3O_8 for fuel elements fabrication for research reactors starting from UF_6 comprises two well defined stages characterized by the risks involved in the raw materials and intermediate products. The first stage is the wet process (conversion process) includes the hydrolysis of UF_6 to UO_2F_2 and posterior precipitation to ammonium diuranate (ADU); the second stage is dry process to obtain the U_3O_8 powder from ADU at high temperature.
This work will be show the analysis of the events in nuclear fuel fabrication facility that have as a consequence the stated risks, their detection and prevention to protect the workers, public and environment from the hazards of radiological and toxicity material.

1- INTRODUCTION AND BACKGROUND

The radiation risks to workers and the public and to the environment that may arise from these applications have to be assessed and, if necessary, controlled. In nuclear fuel fabrication plant, nuclear material (uranium enriched 20%) are used, stored and disposed of, in quantities or concentrations that pose potential hazards to workers, the public and the environment. The physical and chemical forms of the uranium compounds may also vary within a nuclear fuel fabrication plant. Some of the processes use large quantities of hazards chemical substances and gases, which may be toxic, corrosive (UF_6), combustible, or explosive, and consequently may give rise to the need for specific safety requirements in addition to requirements for nuclear safety. In uranium fuel fabrication plant, the main hazards are potential criticality and releases of uranium hexafluoride (UF_6) and UO_2, from which workers, the public and the environment must be protected by means of adequate design and construction, by safe operation and physical protection system. The chemical toxicity of uranium in a soluble form such as UF_6 is more significant than its radiotoxicity [1]. However, the radiological consequences of an accidental release of reprocessed uranium would be likely to be greater. Along with UF_6, large quantities of hazardous chemicals such as hydrogen fluoride (HF) are also present. In addition, when UF_6 is released it reacts with the moisture in the air to produce HF and soluble uranyl fluoride (UO_2F_2), which present additional safety hazards [1]. Therefore, safety analyses for uranium fuel fabrication facility should also address the potential hazard resulting from these chemicals. In this paper we study the identification and assessment of the hazards in a nuclear fuel fabrication plant uses the low enriched uranium (LEU) 20% to be ensure that in all operational states, exposures to radiation are kept below prescribed limits and as low as reasonable corresponding (ALARA) principle and to ensure mitigation of the radiological consequences of accidents [2,3].

2. WET PROCESS (UF_6-ADU CONVERSION PROCESS)

In the wet process or ammonium diuranate (ADU) process, the UF_6 with enriched Uranium 20% is vaporized and transferred to reaction vessel, hydrolyzed with water, and neutralized with NH_4OH to form a slurry of ADU in an aqueous solution of ammonium fluoride and ammonium hydroxide. The ADU is recovered by centrifuging and then is clarified, dried, and calcined to form UO_2 or U_3O_8 powder. The 4 steps of process are:
1- Vaporization process – conversion of a UF_6 solid into a gaseous state by adding heat for UF_6 Cylinder.
2- Hydrolysis process – a chemical process by which the oxygen or hydrogen in water combines with an element, or some element of a compound, to form a new compound.
3- Precipitation – formation of finely divided solids in a chemical reaction.

4- Separation – remove or separate solid particles ADU from the liquid effluent.

3. RISK ANALYSIS IN NFFF

A major concern in nuclear fuel fabrication facility NFFF is the potential for accidental release of uranium hexafluoride. The UF_6 is a reactive substance which reacts with water forming HF and UO_2F_2. The HF is a highly corrosive substance and the UO_2F_2 is very toxic. A sudden release of UF_6 inside a building or to the atmosphere could conceivably cause undesirable health effects to workers and the public in general.
Risk is the possibility of a hazard having adverse consequences under a defined set of conditions. Safety, the inverse of risk, is the probability that harm will not occur under specified conditions. Substances that are extremely toxic can be used safely if the environment is controlled to prevent the absorption of the toxic substance [3]. Any accidental or malicious act by an employee can potentially lead to catastrophic incidents that threaten the environment and the reputation of facility. For this reasons, we should to be study the hazards during the operation of NFFF.

4. TYPES OF HAZARDS IN NFFF

Hazard refers to the potential that a chemical or physical characteristic of a material, system, process, or plant will cause harm or produce adverse consequences. Hazards from nuclear fuel fabrication facility can be dominated by the toxic rather than by the direct radiological effects of the nuclear material.
Increasing in UF_6 mass transferred into hydrolyser without control. It is assumed that the whole contents of UF_6 cylinder ~ 25 kg of UF_6 are transferred into the hydrolyser tank due to operators. [4].
1- Heating at temperature higher than 120 ºC would lead to hydraulic rupture of a full UF_6 cylinder.
To prevent this event, the redundant, independent controls of temperature linked to automatic stopping of heating for above setting temperature [4].
2- Contact of UF_6 with hydrocarbons generates explosive mixture.
To prevent this forbidden use any hydrocarbons in the plant [4].
3- Blocking in the piping or valves are events produce to pressure increase in the gas transfer system.
To prevent the explosion transfer the UF_6 gas to expansion tank system [4].
4- Criticality accident may be take place during the process especially in the wet conversion area such as, Tokaimura nuclear criticality accident in Japan 1999 [5].
Factors that must be controlled to prevent criticality include the following: mass and volume, enrichment, geometry, interaction and separation, moderation, reflection, concentration and density, neutron absorber or poisons, and heterogeneity.
The prevention of criticality is given operatively by the mass control of fissile material at the wet process, and the mass and moderator control at the dry process. Always units of less than 2.4 kg of uranium (20% U^{235}) are handled [4].
5- Fire
6- Theft of nuclear material or unauthorized of removal of fissile material.
To prevent this hazard should be use the NMAC program.

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246-AMSSNUR IMS: CURRRENT STATUS AND PERSPECTIVES

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Abstract

1- Since its creation in 2016 by the law 142-12 on nuclear and radiological safety and security, AMSSNuR has established a vision to become by 2021 an independent credible and effective Regulatory authority at regional level. In doing so, AMSSNuR has established a strategic plan 2017-2021, covering all actions associated with its assigned functions such as authorization, inspection, regulation and enforcement measures.

2- The AMSSNuR has adopted a proactive approach to achieve all strategic objectives as stated in its strategic plan 2017-2021 to satisfy the needs of internal and external stakeholders as well as interested parties through simple and flexible processes and systems which lead to continuous improvement of safety and security.

3- In this framework, Based on IAEA standards such as GSR part 2 and the requirements of ISO 9001: 2015 standard « Quality Management System - Requirements, AMSSNuR has established a strategy to develop and implement its specific integrated management system "IMS" aiming at anchoring the safety and security culture and consequently at maintaining a high level of safety and security at the national level.

4- Moreover, an IMS steering committee headed by Director General was designated and WGs for specific areas concerning AMSSNuR macro-processes, processes and associated sub-processes were created as well. The main results, achieved so far, are listed as follow:

   a. the manual of the IMS (French version)
   b. AMSSNuR regulatory oversight Policy
   c. IMS strategy
   d. 9 core processes and their associated procedures "authorization, inspection, regulation and enforcement"were drafted internally by the working groups and reviewed twice by IAEA -EU experts.
   e. more than 80% of the support processes were drafted and reviewed by IAEA -EU experts.
   f. more than of the management process and associated procedures were drafted and reviewed by IAEA -EU experts

5- The purpose of this paper is to describe the approach followed by AMSSNuR in establishing its IMS and to comment on the result achieved in this particular area and challenges that AMSSNuR is facing in its implementation.

Key Words: AMSSNuR, IMS, macro-process, core process, support process, management process, sub-process, and stakeholders.

247-INTRODUCTION OF DOSE CONSTRAINT CONCEPT INTO RADIATION PROTECTION FRAMEWORK OF THE REPUBLIC OF BELARUS

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The Publication 103 of International Commission on Radiological Protection (ICRP) and revised IAEA basic safety standard No GSR Part 3 highlighted the use of dose constraint (DC) and reference levels (RLs) as the key aspects of the optimisation of radiation protection process, thus setting the task for regulators to enhance implementation of DC and RLs into national regulatory system. In the Republic of Belarus, like in many other countries, this task is being completed by establishing appropriate requirements in the radiation hygienic standards and regulations (further - SANPINs), developed and approved by the Ministry of Health of the Republic of Belarus (MoH).

The existing SANPINs (2012 -2013), which are based on provisions of GSR Part 3 safety standard, set DC and RL definitions and indicate the need of their application to the optimization of protection of public and workers against exposure of radiation sources. The DC and RL concepts are reflected in a new Law of the Republic of Belarus “On radiation safety” that was put in force on June 18, 2020. The Law authorizes MoH to set up the procedure for establishing and application of DC and RLs in the radiation protection system.

To be effective in fulfilling this government mandate the specialists of Scientific Practical Hygiene Centre of MoH studied experience of interpretation and implementation of DC and RLs in other countries while also investigating the DC concept as defined in ICRP Publication 103 and GSR Part 3 safety standard. The results allow speaking that despite of the using DC or similar instruments for radiation protection purposes in many countries, in most of them the used approaches do not consider DC concept in a way, defined by the ICRP. The various guidance documents providing support in this domain do not help much. Finally, draft Regulation on the procedure for establishing and application of DC and RLs (further - DC Regulation) has been developed based on national regulatory experience and national regulations (SANPINs) with due account to ICRP and IAEA recommendations.

DC Regulation states that DC and RLs must be used within the context of optimisation of radiation protection, related to a given source of ionising radiation. In planned exposure situations DC shall be established:

- for public exposure from those types of facilities, which could give rise to exposure of members of the public (a representative person) with individual dose higher than 10 micro Sv/ year in consideration of all likely exposure pathways;
- for exposure of all radiation workers at nuclear facilities and those at radiation facilities, whose work with open sources is classified as 1 or 2 class, or who are working with sealed radiation sources of I and II safety category;
- for carers and persons exposed in biomedical research (medical exposure).

DCs are set as one or more of the following values:

- annual individual effective dose of a representative person (for public exposure) or standard worker (for occupational exposure);
- annual equivalent dose to an organ or tissue;
- risk constraint for exposure to a single (specified) source.

The DC Regulation clearly defines the responsibilities of the source operator (employer for use) and the regulator (Gossannadzor) for establishing DC, as summarised in Table 1 below.

In all cases the justification materials, required for endorsement of DC value by Gossannaadzor, must demonstrate that design and shielding specification of the facility ensure compliance with the proposed DC value in consideration of all likely pathways that could result in exposures to workers or members of the public.

TABLE 1. RESPONSIBILITIES FOR ESTABLISHING DC

<table>
<tr><th>DC</th><th colspan="2">Responsibilities</th></tr>
<tr><td>DC for public exposure in the design phase of building a new facility</td><td>Designer (in cooperation with source operator or employer for use):
— determines an appropriated value of DC based on project radioactive discharges into the environment and available data on impact on population from similar facilities;
— submits to Gossannadzor the standard form with the proposed DC value together with the documents required for its justification</td><td rowspan="2">Gossannadzor:
— approves (or make corrections and reconsiders) the proposed DC value, based on the results of sanitary-hygiene expertise, conducted in the process of the sanitary authorization of activities with the source of ionizing radiation</td></tr>
<tr><td>DC for public exposure from operating facility</td><td>Source operator (employer for use):
— determines an appropriate value of DC based on the data from safety assessment report;
— submits the DC standard form and justification materials to Gossannadzor</td></tr>
<tr><td>DC for occupational exposure in the design phase of building a new facility</td><td>Employer for use:
— defines DC values for workers, involved in specific tasks and practices based on the experience of the similar facilities and practices with comparable working conditions;
— submits the DC standard form to Gossannadzor for DC endorsement;
— demonstrates to Gossannadzor that the facility will be designed in such a way that no occupationally exposed worker is expected to receive a dose exceeding the DC;
— approves the DC value (after endorsement by Gossannadzor)</td><td rowspan="2">Gossannadzor:
— advises and gives formal agreement on the DC value based on the results of the sanitary-hygiene expertise;
— lays down DC values in the sanitary passport, that is a permit issued by Gossannadzor within the process of the sanitary authorisation of the activities with the sources of ionizing radiation</td></tr>
<tr><td>DC for occupational exposure at operating facility</td><td>Source operator (employer for use):
— specifies DC values based upon data on actual doses received by the workers of the facility and data from other existing radiation protection practices with comparable working conditions;
— submits the DC standard form to Gossannadzor for DC endorsement;
— approves DC values (in consensual agreement with Gossannadzor)</td></tr>
</table>

In the process of optimizing the source operator (employer for use) first of all takes measures to prevent exceeding the established DC values, and then directs the efforts to reduce doses of exposure to achievably low levels, considering economic and social factors. DC values can be revised, for example in situations, where existing facilities are being upgraded or modified or when new equipment is being installed in existing facilities.

In the event of DC values being exceeded the source operator shall inform Gossannadzor about taken corrective measures. Violation is not the exceeding of the DC, but if measurers are not taken.

DC Regulation also specifies that it is the responsibility of Gossannadzor to establish in the Hygiene Standard RLs for exposure doses to the public and workers in specific existing exposure situations, as well as RLs for exposure to the public and emergency workers and other persons participating in liquidation of consequences of a radiation accident in emergency exposure situation. The revised Hygiene Standard “Criteria for assessing radiation exposure" establishes upper levels of RLs for all categories of exposure in emergency exposure situation and existing exposure situation as well as specifies permissible levels of radionuclide activity concentrations in commodities (food, drinking water, construction materials) corresponding to the established RLs. The Hygiene Standard is subject to approval of the Council of Ministers of the Republic of Belarus, that makes it mandatory for all organizations in the territory of the Republic of Belarus.

Further work on adapting DC and RL concepts to the practice of the state sanitary supervision will include the development of practical guidelines, as well as close collaboration and training of all concerned on how to use DCs and RLs in optimisation process.

The abstract information is intended to contribute to the general discussion on implementing the DC and RLs concepts into the national radiation protection framework.

248-AAPM TG-111 DOSE DESCRIPTORS COMPUTED FROM ACCUMULATIVE DOSE PROFILES PRODUCED BY HELICAL CT ACQUISITIONS ON PMMA CYLINDRICAL PHANTOMS: USE OF MONTE CARLO BASED SIMULATIONS AND COMPARISON WITH EXPERIMENTAL DATA

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Introduction:

A comprehensive theory predicting relative dose distributions in cylindrical dosimetry phantoms associated to helical Computed Tomography (CT) scan series has been developed by Dixon et al. [1-2]. This formalism allows to derive CT dose descriptors applied to cylindrical phantoms that improve the efficiency and expand the applicability of the traditional and current quantities based on the $CTDI_{100}$ methodology. In the last decades, the limitations of the $CTDI_{100}$ to represent the accumulated dose at the midpoint (z=0) in the interval (-L/2, L/2) of a CT scanning length L had been discussed and demonstrated [1]. Therefore, the Task Group 111 of the American Association of Physics in Medicine (AAPM TG-111) [3] proposed novel CT dose descriptors supported by the unified theory of dose distribution in cylindrical phantoms [2]. AAPM TG-111 dose descriptors come from the accumulated dose profile which in analytic framework can be computed from the convolution product between the single axial dose profile and a rectangular function of length equal to the CT scanning length L [1].

The present study computes the AAPM TG-111 dose descriptors derived from accumulative dose profiles, $D_L(z)$, computed from Monte Carlo modelling of a helical CT acquisition. The advantage is that for MC calculation the accumulated dose is totally computed, not just at a specific point but for its long length. Dose profiles computed by Monte Carlo calculation include the effects of the anode self-attenuation (heel effect), x-ray spectrum and anode angle. Additionally, experimental measurements using a 0.6cc ion chamber of equilibrium dose, D_{eq}, and rise to equilibrium function were performed to validate the MC predictions.

Materials and Methods

A. CT DOSE METRICS IN CYLINDRICAL PHANTOMS

Dose line integral DLI is the line integral over $(-\infty,\infty)$ of the accumulated dose profile, $D_L(z)$, resulting from a helical CT scan acquisition of scan length ($L = vt_0$) (t_0=total beam-on time, v=couch velocity for helical scans). DLI is directly proportional to the scan length L with term of proportionality the equilibrium dose, D_{eq}, as shown in equation (1). D_{eq} is the asymptotic value of the maximum dose $D_L(0)$, which occur at $z = 0$, achieved for a sufficient long scan length [1].

$$DLI = \int_{-\infty}^{\infty} D_L(z)dz = D_{eq}L \tag{1}$$

$D_L(z)$ could be computed along the central or peripheral axes of a cylindrical phantom. The ratio between $D_L(0)$ with D_{eq} defines the approach to equilibrium function $H(L)$.

B. MONTE CARLO MODELLING

The 2015 version of the PENELOPE/penEasy Monte Carlo software was used to model in two stages a helical CT acquisition. First, the x-ray emission characteristics of the GE Discovery CT750 HD scanner (General Electric Company, Boston, USA) were modelled taking into account the beam quality, and dose profiles in a cylindrical phantom. Second, translation and rotation transformations were applied on position and direction of the simulated x-ray photons to model the helical trajectory of the x-ray source around the phantom in a helical CT acquisition. The geometry was coded according to the PENGEOM format and syntax. To model a helical CT acquisition, head and body clinical protocols were chosen with characteristics of pitch values of 0.516, 0.984, and 1.375, and nT=40 mm for body protocols, and pitch values of 0.531, 0.969, and 1.375, and nT=20 mm for head protocols. X-ray spectra corresponding to 80, 100, 120, and 140 kV were evaluated.

Results and Discussions

FIG. 1.a shows the dependence of the equilibrium dose in units of mGy/100mAs against the pitch, along the central axis of a PMMA cylindrical dosimetry phantom of 32cm diameter for 80 and 120 x-ray spectrum. The MC predictions corresponding to 80kV x-ray spectrum show good agreement with the experimental data. For 120kV x-ray spectrum the comparison produces a discrepancy of about 13% with respect to the experimental data. FIG. 1.b presents the rise to equilibrium function $H(L)$ defined by the quotient between $D_L(0)$ and D_{eq} for 120kV x-ray spectrum and pitch values of 0.516 and 0.984 corresponding to pitch values of body protocols. FIG. 1.b illustrates qualitatively that the MC predictions follow the same tendency of the experimental data at least qualitatively.

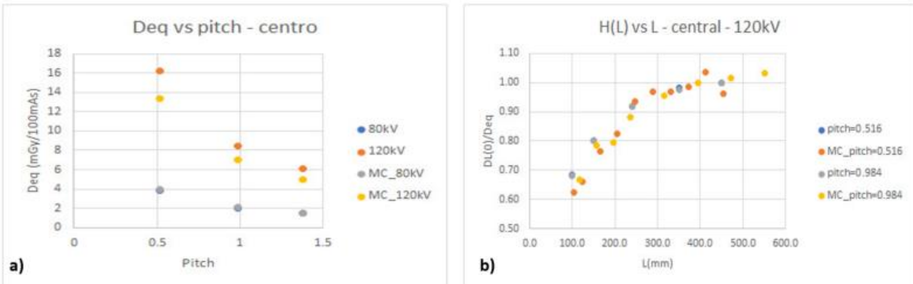

FIG. 33. a).Equilibrium Dose versus pitch and b) approach to equilibrium function H(L). Computed by Monte Carlo simulation and measured with a small 0.6cc ion chamber. For 80 and 120 X-ray spectrum.

Conclusions

A comprehensive data related to the AAPM TG-111 dose descriptors were obtained from the accumulative dose profile computed by MC simulation and calibrated in terms of mGy/100mAs using the air-kerma length product (PKL) reading of a 100mm pencil ion chamber. The results show good agreement with the experimental data, with discrepancies up to 13% in the case of the equilibrium dose for 120 kV x-ray spectrum. These data will be used to investigate the correlation between the AAPM TG-111 dose descriptors with organ doses.

ACKNOWLEDGEMENTS

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250-THE APPLICATION OF THE RISK MATRIX METHODOLOGY, TO REDUCE RADIOLOGICAL ACCIDENTS IN INDUSTRIAL RADIOGRAPHY

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INTRODUCTION

A long list of radiological accidents has been reported worldwide related to industrial radiography due to equipment failures, human errors, external events or combinations of them, which have occasionally caused high doses, with serious consequences for health, such as deterministic effects. [1] For this reason the Iberoamerican Forum of Radiological and Nuclear Regulatory Bodies (FORO) supported the creation of a methodology using the risk matrix as a tool to detect and correct anomalies, deviations and errors during the use of radioactive material and x-ray generators devices in industrial radiography. [2] This methodology describes a large list of initiators events, barriers, consequence reducers and frequency reducers. The application of the risk matrix approach is a useful tool to reduce the risk of accidental exposure to radiation.

The document developed by FORO using the risk matrix in industrial radiography, has been accompanied by the SEVRRA software, to simplify the use of these concepts and offer to industrial users of radioactive material and x-ray generators, a simple mechanism that allows them to apply the concepts of risk matrix methodology, to prevent radiological incidents.

MATERIALS AND METHODS

The risk matrix is a systematic, structured, simplified and conservative method that is based on the application of the risk equation, R = f * P * C, and consists in assessing the logical sequence in which accidents occur, considering that a certain human error or equipment failure (*initiator event*) occurs with a *frequency* determined (f); the facility or activity has one or several *defenses or barriers* (interlocks, alarms or procedures) capable of detecting and controlling the error or failure and act to prevent the initiating event from becoming an accident, however, there is always a *certain probability* (P) that these barriers can fail, and in in such case the accident will occur, and this is manifested with *certain consequences* (C).

To perform a risk assessment [3], this method provides each one of the variables independent of the risk equation at various levels (Example: Very High, High, Medium, Low and Very Low). In this sense, the risk matrix is a representation of all combinations of the levels of: f, P, C and the resulting level of risk.

The resulting level of risk (R) is obtained using the logical combination of the different levels of the variables independent so defined, that is, the frequency of the initiator event (f), the probability of failure of the planned defenses (P) and the severity of the consequences (C) that characterizes to a certain accident sequence.

The methodology of risk matrix has the advantage that it is relatively easy to apply in facilities and activities with a minimum of material and human resources.

CRITERIA FOR ASSIGNING CONSEQUENCES LEVELS

It was divided in 4 levels:

— Very high, catastrophic or very serious (CVH): These are those that cause severe deterministic effects, being fatal or causing permanent damage that reduces the quality of life of the affected people.

— High or Serious (CH): Are those that cause deterministic effects.

— Medium or moderate (CM): Are those that cause abnormal exposures that are below the thresholds of the effects deterministic. (only represent an increase in the probability of occurrence of stochastic effects).

— Low (CL): There are no effects on workers and the public, but security measures are degraded.

SEVRRA SOFTWARE (Radiological risk assessment system)

SEVRRA is software that includes the theoretical concepts developed by the risk matrix methodology relating the frequency of the initiating event (f), the probability of failure of the planned defenses (P) and the severity of the consequences (C) that characterizes to a certain accident sequence. SEVRRA software can be applied by any industrial radiography company that wants to assess the risk of the practice taking into account different stages: acquisition of the radioactive source / equipment, storage area, transport, operation, etc. After the data is loaded into the SEVRRA software by the authorized user of industrial radiography, the program performs the analysis, and the result obtained shows the risk for a given practice (using radioactive source or x-ray devices) and its comparison with a reference installation. (Fig. 1) This comparison allows understanding the risks detected and which of them should be analyzed quickly so as to reduce them as soon as possible.

Fig. 1. Comparison of results between the evaluated installation and the reference installation using SEVRRA Software, (Risk Low, Medium or High)

CONCLUSIONS

The presented methodology and software can detect deviations, failures and errors during practice in industrial radiography using radioactive materials or x-ray generators, and can apply for mobile or fixed practices.

The authorized user can easily know the risk level of their practice and thus apply methods and processes to reduce the risk.

The software is friendly and simple to apply. No special or specific knowledge about the methodology is required.

Risk assessment is an excellent tool to reduce the rate of incidents and accidents and can avoid unjustified exposures of workers and members of the public.

It is useful for authorized users in industrial radiography practices and also for regulatory bodies.

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251-IMPROVING RADIATION PROTECTION IN PRACTICE: FROM A TECHNICAL SERVICE PROVIDER (TSP) STANDPOINT

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INTRODUCTION

Radiation protection in developing and non-nuclear power countries could face different sets of challenges. For one the inventory of radioisotopes and radiation generators are mostly in medical and industrial applications with a relatively small amount from the nuclear fuel cycle. The legislative and regulatory framework may not be as comprehensive and sufficient compared to international standards. Also, due to limited resources, the needed physical infrastructure and human expertise may not be readily available. As safety requirements increase due to new standards such as GSR Part 3, building new capabilities to address the gaps could take time to develop. And owing to the inventory and type of utilization, the number of technical service providers (TSP) and the range of its capabilities could be limited.

TSPs contributes to the implementation of radiation protection in practice by providing services such as occupational exposure monitoring, workplace monitoring, and calibration. In the Philippines, the Radiation Protection Services of the Philippine Nuclear Research Institute (RPSS-PNRI) is one the TSPs but is the only government-owned service provider. The services provided are individual monitoring, calibration, workplace monitoring, leak testing of sealed sources and radioactive waste management.

In this paper, the work done by the RPSS-PNRI in the last 5 years to help further improve radiation protection in practice through the provision of technical services is presented. The actions taken to build the needed capabilities and the ways for which TSPs could help in developing regulations are also discussed.

NEEDS, TRENDS IN OCCUPATIONAL EXPOSURE CONTROL IN THE PHILIPPINES

Although the PNRI-RPSS has been providing technical services for many decades, there are still many areas that remained to be developed to address both the requirements of national regulations and the added recommendations of the GSR Part3. However, due to limited resources, it is not possible to address all of it at the same time. It is therefore important to determine which of the gaps will be prioritized.

As a TSP, the steps taken by the RPSS-PNRI are a) to first identify the requirements/needs for occupational radiation protection that is applicable to the current and future utilization in the country, b) determine the gaps in the services provided based on this needs, c) search and secure the funds to build the needed capabilities, and d) identify ways for which the data from services provided can be used as tool to improve regulations.

In the last 5 years the number of licensees of ionizing radiation have been increasing especially in the medical sector. Particularly for RPSS-PNRI, the volume of clients served has increased by almost 70% and the number of workers monitored by 50% since 2015.

Among the gaps in capabilities of the RPSS-PNRI is the radiation qualities used for the calibration of radiation monitoring instruments, which was limited to Cs-137 only. The individual monitoring service provided are also limited to external exposure monitoring while the extremity dosimeters available are not sufficient to meet the demand of the clients. Although there is presently no operating nuclear reactors in the country, the number of nuclear medicine and cyclotron facilities are increasing. There are also radioisotope dispensing facilities. There is therefore the need for an internal exposure monitoring of workers. The number of users of neutron radioisotopes and the planned operation of a subcritical research reactor also brings into consideration the need for neutron monitoring and calibration services.

Another requirement in the GSR Part 3 is the establishment of a dose registry of occupational exposure records. The country has currently no state dose registry of workers. But with the increasing number of workers, many of which are working in multiple facilities and the cessation of operations of some facilities, there is a growing need to establish a dose registry in the country.

CONTRIBUTIONS OF A RPSS-PNRI AS A TSP IN RADIATIO PROTECTION

Through a combination of IAEA Technical Cooperation Project, Grants-in-Aid Projects and national funding, the radiation protection service facility was upgraded and additional capabilities were developed starting 2016. The SSDL was also upgraded to enable calibrations for neutron, beta and higher intensity photons. This will give the radiation facilities access to more calibration services and thereby better accuracy and traceability of their monitoring instruments.

In occupational external exposure monitoring, processing equipment was upgraded to automatic system leading to a much improved throughput. Additional dosimeters were also acquired thereby increasing number of personnel monitored by 50% in 5 years. Also, the capability for neutron exposure monitoring was developed allowing better occupational exposure monitoring for workers that deal with neutron radiation.

With the absence of a state dose registry, RPSS-PNRI also developed a web-based tool, PhilDose, which is a dose registry of occupationally exposed subscribed to its individual monitoring service. It provides occupational exposure records of the worker, annual dose summary of a facility and notification of doses exceeding the limits. PhilDose is now in full implementation and maybe adopted as a state registry.

Over the years, RPSS-PNRI had collected information on the profiles of workers and facilities in the country. For instance, through the PhilDose, assessment of worker exposures show that majority of the conventional diagnostic radiology workers have received doses below the MDL of the dosimeters [2]. Results of extremity monitoring on the other hand show the need for dose monitoring of extremities of all workers in nuclear medicine facilities for better assessment of their occupational exposures [3]. Trends in gross radioactivity analysis, from results of leak testing of sealed sources services provided, show that the likelihood of degradation of the integrity of sealed sources are extremely low to nil. These findings of a TSP could be used as basis in reviewing existing regulatory requirements of radiation safety programs. TSPs can therefore also contribute in developing of better regulations in the country.

Technical service providers play an important role in radiation protection by giving operators and workers access to occupational exposure control and calibration services. In the Philippines, the RPSS-PNRI had been working over the years to build further its capabilities to provide Filipino workers with services that would aid them in their radiation protection programs. RPSS-PNRI not only considered the current needs, but also looked into the trends in utilization and new international safety standards. Despite the limitations common in developing countries, the RPSS-PNRI as a TSP played a key role in helping improve the radiation protection in the country.

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253-RADIATION SAFETY ASPECTS FOR DECOMMISSIONING OF DECAYED COBALT-60 RADIOACTIVE SOURCE AND DEPLETED URANIUM USED IN TELETHERAPY MACHINE

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Cobalt-60 Teletherapy (Telecobalt) machines are essential part of radiation therapy departments in developing world due to their cost effectiveness, great throughput and robustness. Many developing countries including India heavily depend on these machines to combat large burden of cancer cases. Due to this radiation safety aspect of decommissioning and transport of decayed teletherapy source is matter of great concern. The Telecobalt unit 'Picker Advance Tele Cobalt (ATC) C/9' (Advanced Medical Systems Inc., USA) was installed in 1995. The head capacity of the machine was 333 TBq (9000 Ci). This machine has provided the treatment of about 800 patients per year and worked for a period of 24 years (1995-2019). In March 2019, the output of Telecobalt source became below 50 cGy/min at normal treatment distance (NTD) of 80 cm which is clinically not acceptable for patient treatment because of the increase in treatment time. The decommissioning of the machine was considered due to ageing of machine and the vendor was not providing services anymore. The present study describes the radiation safety aspect of planning of decommissioning of the teletherapy machine and transport of decayed Cobalt-60 radioactive source for disposal and personnel dose measurements during the procedure. The Depleted Uranium (DU) was used by manufacturer for shielding of gamma rays in the Telecobalt machine and the present study becomes more important providing guidance for dismantling of Telecobalt machine, retrieval of DU and safe disposal of DU. This study recognizes potential hazards possessed by radioactive materials to environment and health impact during decommissioning, dismantling of machine and transport of radioactive material. To formally address the situation, a contingency plan was prepared and discussed considering various emergency scenarios.

The decommissioning agency was contacted for unloading of source as supplier of machine stopped providing the services. Decommissioning of the cobalt machine began with preparation of a plan and its submission for approval by national regulatory authority. The plan has foreseen all necessary steps, which were considered as appropriate to perform this process in a safe manner and with minimum expenses. The technical documents of the installation and the functioning of the machine as well a short description of the main characteristics of the cobalt source were prepared with the same goal and decommissioning was planned. Necessary advices were given by the Radiological Safety Officer (RSO) to the team considering source characteristics and dummy trials were performed in order to maintain the dose levels minimum (ALARA). A written contingency plan was also prepared in case of emergency. The decommissioning of the machine began with removal of the couch and collimator from the gantry and then alignment of the source container with the gantry head. Radiation survey of the source head was performed before the source transfer operation by the certified service engineers. Source unloading done by a certified service engineer under the supervision of authorized RSO. Then decayed source was unloaded in an approved shielded transport flask and was kept in another outer transport container. A radiation protection survey of container was done in a matrix format of the all sides of this package. Radiation levels were measured at 1 m distance from source and the external surface of package. The container was planned to be transported as Type B (U) package. After unloading the source from machine, a swipe test was also performed to check the contamination in the area in which the source moves and on the external surface of source head. To measure the personnel doses, pocket dosimeter, Thermo Luminescence

Dosimeter (TLD) badge and Optically Stimulated Luminescence Dosimeter (OSLD) were used. The machine was old and DU was present in form of shielding. Additional special arrangements were made in order to dismantle depleted uranium (DU) parts which were used for shielding in the source head of the Telecobalt machine. The machine head containing DU was transported in a separate transport container and it was transported as an Excepted package. The decommissioned source and DU were packed as per national regulatory requirements and transported for safe disposal to an approved waste disposal agency on receipt of road transport permission from national regulatory authority. The machine was dismantled locally in such a way that the unit cannot be put again for use. The area was monitored with contamination monitor for any residual contamination and declare it to be free from radiation contamination. The DU removed from collimator head and rotor by cutting head and rotor as per technical diagram of machine head. The DU retrieval was done in the facility of waste disposal agency.
Licensee is responsible for fulfilling the regulatory requirements and necessary arrangements/ tools for ensuring radiation safety while decommissioning. The dose received by the personnel was monitored by the pocket dosimeter, TLD badge and OSLD. The pocket dosimeter being an active dosimeter gives instantaneous reading received by the personnel. The maximum dose received by the source decommissioning engineers was observed 0.03 mSv during the decommissioning procedure. The decommissioning report were informed to the national regulatory authority through an online web-based Information and Communication Technology (I&CT) portal Electronic Licensing of Radiation Applications (e-LORA). Radiation level was measured 2.3 mR/hr around wheel after removing the collimator. Maximum radiation level on the surface of the package was found 6.0 mR/hr and maximum radiation level at 1 m from the external surface of the package was found 0.83 mR/hr. On the basis of transport index value, source was kept in category Yellow-II. The depleted uranium parts weight 21 Kg (15 Kgs in Head and 6 Kgs in Rotor) retrieved from head and rotor of Telecobalt machine. The high density and high atomic number of DU makes it suitable for the shielding of gamma rays. Earlier, DU was used as shielding material in Teletherapy machines. Today Tungsten is used as shielding material in Teletherapy machines instead of DU. However, there are several old Teletherapy machines exist in the world that uses DU as shielding material. Individuals can be exposed to DU by inhalation, ingestion and dermal contact (including injury by embedded fragments) during dismantling of Teletherapy machine, retrieval of DU and transport. Special emphasis was given in the contingency plan to DU related chemical and radiological toxicities with the two important organs being kidneys and lungs. The development of the new concepts and practices in the field of the decommissioning includes the preparation of a decommissioning plan. This study provides guidance for the actual decommissioning operations as well as the safe management of the waste arising from the decommissioning or dismantling of Teletherapy machines. Such plans need to be developed for each facility with radiation sources prior to decommissioning.

255-RADIATION MONITORING OF PATIENTS AND PERSONNEL FOR DOSE OPTIMIZATION AND RISK ESTIMATION DURING ENDOSCOPIC RETROGRADE CHOLANGIOPANCREATOGRAPHY

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Endoscopic Retrograde Cholangio-Pancreatography (ERCP) uses ionizing radiation and hence associated with radiation hazards. The presentation analyses the prospective estimates of the radiation doses received by patients and personnel during ERCP, optimizes the radiation dose during therapeutic ERCP, estimates the risk for personnel and patients and compares the doses with available data and reference levels.

EVALUATION OF PROCEDURAL PARAMETERS

Dose Area Product (DAP) measurements can be used to estimate the effective dose (ED) to patients undergoing ERCP. A DAP meter was fitted to the x-ray tube before each of the ERCP procedure. 40 ERCP procedures of therapeutic intent are evaluated in the presentation. All the procedures were performed on Allengers HF fluoroscope.

The procedure completion time observed varied between 10 minutes to 35 minutes with a median of 20 minutes. Table 1 show a comparison of median fluoroscopic Time and DAP values observed with few published data.

TABLE 1. COMPARISON OF MEDIAN FLUOROSCOPIC TIME AND DAP VALUES

Author	No. of cases	Median T min	Median DAP $Gy.cm^2$
SMS Jaipur [2019]	40	2.2	9.77
Tsapaki et al. [2017]	1632	3.5	15.6
Hadjiconstanti et al. [2017]	15	2.1	1.7
Tsapaki et al. [2016]	200	4.4	7.8
Saukko et al. [2015]	227	2.7	5.83
Seo et al. [2015]	126	4.52	27.35
Liao et al. [2015]	331	3.4	9.56
Tsapaki et al. [2011]	157	2.6	3.1
Rodriguez et al. [2011]	340	2.75	9.4
IAEA part E [2010]	10	3.3	11.1
IAEA part F [2010]	39	1.7	23.6
Tsalafoutas et al. [2003]	21	7.1	48.9
Buls et al. [2002]	54	8.3	60.3

ESTIMATION OF RADIATION DOSES

Optically Stimulated Luminescence Dosimeters (OSLDs) were used to measure radiation exposure to the abdomen, thyroid surface, forehead and hands of the personnel performing the procedure and center of anterior and posterior body surface of the patient undergoing the procedure. Median dose to abdomen, thyroid surface, forehead and hands measured for personnel are tabulated in Table 2.

TABLE 2. RADIATION DOSES MEASURED FOR PERSONNEL

Personnel	Abdomen cGy	Forehead cGy	Left Wrist cGy	Right Wrist cGy	Thyroid Surface cGy
Endoscopist	0.0047	0.0069	0.0072	0.0086	0.0081
Resident	0.0085	0.0093	0.0187	0.0306	0.0132
Technologist	0.0105	0.0204	0.0159	0.0437	0.0100

RISK ESTIMATION AND DOSE OPTIMIZATION FOR PATIENTS AND PERSONNEL

PCXMC dose calculations (Version 2.0.1.4), was used for the Monte-Carlo simulations. The patient age, sex, body weight and height was noted prior to the procedure to create a phantom analogous to the patient. The machine parameters filtration, anode angle and projection angle were obtained. kVp, mAs, DAP reading for each procedure are noted to simulate ESD and verified with ESD measured using calibrated OSLD. The imaging field size, Focus to Surface Distance (FSD) and Focus to Image receptor Distance (FID) also aided in the simulations. The risk of exposure induced cancer death estimated for patients in the presentation is 8.86×10^{-4} (1 in 1200) which is similar to the findings of Larkin et al. and Naidu et al. who have estimated the risk of radiation-induced cancer between 1 in 1700 (6×10^{-4}) to 1 in 3500 (3×10^{-4}) per procedure. The risk of exposure induced cancer death and loss of life expectancy (LLE) estimated in the presentation for the endoscopist, resident and assistant were 3.86×10^{-5} (1 in 26000), 6 minutes; 9.05×10^{-5} (1 in 12000), 6 minutes and 1.09×10^{-4} (1 in 9200), 12 minutes respectively.

The diagnostic reference level for ERCP in the presentation is 13.82 Gy.cm^2 (75th percentile DAP) while the UK DRL is 16.7 Gy.cm^2. Routine radiation dose monitoring is a valuable tool for reducing both patient and staff radiation doses. The ERCP type (diagnostic or therapeutic), fluoroscopic parameters and the complexity of each clinical case will definitely affect radiation dose received. The number of procedures performed routinely plays a vital contribution to the cumulative dose to the personnel.

CONCLUSION

The patient doses in ERCP can be optimized by the presence of an experienced interventional endoscopist in the team by reducing the fluoroscopic time and reduction of radiographic images. The radiation absorbed doses to the different organs are relatively low reasons be the high level of experience and the proper use of radiation protection measures. Proper use of lead curtains and lead aprons is important in reducing the personnel dose. The available data is not enough to establish local reference levels, but this could be a baseline for the optimization of dose with regard to avoiding unnecessary radiation risks.

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256-MEASUREMENT OF RADIATION DOSES FROM VARIOUS RADIOLOGICAL PROCEDURES IN A TIRTIARY CARE PUBLIC MEDICAL COLLEGE AND HOSPITALS

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Dose Reference Levels (DRLs) may be defined for each radiographic procedure for multiple dosimetric parameters as discussed in the available literature. In the presentation the entrance surface dose (ESD) measured using Optically Stimulated Dosimeters (OSLD) and the Dose Area Product (DAP) for various radiological procedures were evaluated to optimize the radiation protection and safety standards of the radiological procedures carried out at the institute and compared it with the available international data.

RADIOGRAPHIC EXAMINATIONS

The entrance surface dose for various radiographic imaging procedures were carried out on 33 X-ray generating equipment. A total of 4835 adult patient examinations were evaluated in the presentation with a minimum of 40 examinations of each part. Most of the machines are dedicated for a specific procedure. The 3rd quartile (the 75th percentile) value of the distribution of patient doses obtained for various sites of examination is tabulated and compared with the international reference guideline and other national DRLs as shown in Table 1.

TABLE 1. COMPARISON OF DRL FOR VARIOUS RADIOGRAPHIC PROCEDURES MEASURED AS ENTRANCE SURFACE DOSE (mGy)

Examination Part	IAEA	Germany	EU Rad Prot 2014	Spain 2009	UK 2014	Japan 2015	Austria 2010	SMS Jaipur 2019
Abdomen AP	-	-	-	-	4.4	3	-	1.83
Ankle	-	-	-	-	-	0.2	-	0.29
C Spine	-	-	-	-	-	0.9	-	1.02
Chest PA	0.4	5	0.3	0.3	0.15	0.3	0.3	0.38
Femur	-	-	-	-	-	2	-	2.82
Forearm	-	-	-	-	-	0.2	-	0.22
L Spine AP	10	10	10	10	5.7	4	10	10.15
L Spine LAT	30	30	30	30	10	11	16	12.64
Pelvis	10	10	10	10	3.9	3	6	9.25
Skull AP/PA	-	-	5	-	1.8	3	-	4.26
Skull LAT	-	-	3	-	1.1	2	-	2.62
T Spine AP	-	-	-	-	3.5	3	-	3.90
T Spine LAT	-	-	-	-	7	6	-	6.70

CT EXAMINATIONS

CT examinations performed on Philips Ingenuity 128 CT scanners (2 no.) for various sites. A total of 244 patient's CT scans were analyzed to optimize the radiation dose levels for the respective procedures and compared to the recent data from Japan as shown in Table 2. The dose levels observed in the presentation are lower in comparison. The differences in the procedure protocol, no. of slices acquired and the length covered in the presentation results in the lower dosimeteric values observed.

TABLE 2. COMPARISON OF DRL ($CTDI_{vol}$ AND DLP) FOR VARIOUS CT PROCEDURES

Procedure	$CTDI_{vol}$ mGy Japan 2015	$CTDI_{vol}$ mGy Jaipur 2019	DLP mGy.cm Japan 2015	DLP mGy.cm Jaipur 2019
Brain	85	32	1350	899
Chest	30	29.67	2350	1216
Abdomen	35	25.4	2300	1423

INTERVENTIONAL CARDIOLOGICAL PROCEDURES

Interventional coronary angiography (CA) and percutaneous transluminal coronary angioplasty (PTCA) were performed on Siemens Axiom Artis and Siemens Axiom Artis Zee units with monoplane digital flat panel systems and under the couch X-ray tube geometry. These interventional units have dose area product meter mounted in the X-ray tube housing, which was used for the measurement of patient entrance surface dose in the presentation. Relevant patient and examination related parameters as patient name, sex age, type of procedure, total fluoroscopy time, DAP meter reading and number of cine series were recorded for 40 patients each who underwent CA and PTCA procedures.

The patients included in the presentation were aged between 35 to 78 years indicating the higher incidences of cardio vascular diseases in the old age. The 75 percentile value of the DAP readings found was 33.63 $Gy.cm^2$ for angiography procedures and 80 $Gy.cm^2$ for angioplasty procedures. The average procedure completion time observed for angioplasty procedures was about 40 minutes, which was approximately double the time for angiography procedures.

European diagnostic reference levels for coronary angiography and percutaneous coronary intervention are 35 $Gy.cm^2$ and 85 $Gy.cm^2$ respectively based on 2018 data.

CONCLUSION

The radiation administered to patients for radiological examinations is expected to decrease as emphasis is placed on optimization of procedure and as equipment improves. The radiation dose to patients should be brought down as low as reasonably achievable (ALARA) as no dose is safe. As optimization occurs and practice improves, institutional reference dose levels require periodic updating.

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257-THE MOROCCAN APPROACH REGARDING CAPACITY BUILDING IN RADIATION SAFETY

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Abstract

The Moroccan Agency for Nuclear and Radiological Safety and Security (AMSSNuR) is the new National Regulatory Body of Morocco, established by the Law No. 142-12 on "nuclear and radiological safety and security", as an independent authority under the responsibility of the Head of Government.

One of the main strategic objectives of AMSSNuR is to upgrade the national regulatory framework covering nuclear and radiological safety and security and nuclear safeguards, in accordance with IAEA's standards and guidance to fully discharge its national responsibilities and comply with international obligations of the Kingdom of Morocco. For this purpose, the IAEA approach for Capacity Building (CB) was adopted and implemented through: (a) Human resource development; (b) Education and training; (c) Knowledge management; and (d) Knowledge networks.. In particular, the following activities have been implemented:

(a) Establishment and implementation of a national strategy on education and training on radiation safety, including the adoption of the IAEA training levels leading to qualify three categories of radiation protection professionals, namely: radiation protection expert (RPE), radiation protection officer (RPO) and qualified operator (QO);

(b) Developing and maintaining the necessary infrastructure concerning the radiation protection training and qualification, through the approval by AMSSNuR of training centers and related service providers, as recommended by the requirement 13 of GSR part 1 of the IAEA. Modalities and conditions of recognition of technical service providers are regulated by decree; and

(c) Participation in and contribution to regional and international networks such the Global Nuclear Safety and Security (GNSSN), the Forum of Nuclear Regulatory Bodies in Africa (FNRBA), the Arab Network on Nuclear Regulators (ANNuR)…etc.

The purpose of this paper is to present the strategy, methodology and approach followed by AMSSNuR in establishing and implementing Capacity Building for radiation protection and safety to sustain and continuously improve the national radiation safety infrastructure.

Key Words: AMSSNuR, Radiation Safety, RPE, RPO, QO, National Strategy for Education and Training, Regulatory Control.

258-THE ROLE OF REGIONAL AND INTERNATIONAL COOPERATION IN RADIATION SAFETY: THE MOROCCAN EXPERIENCE

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The benefits of ionizing radiation to mankind are enormous when used safely and securely. These benefits are reflected in a broad spectrum of radiation technologies used in, inter alia, medicine, industry, energy production, and research, including, during abnormal conditions such as the pandemic outbreak of coronavirus disease (COVID-19). The obvious and explicit acquaintance of humankind with ionizing radiation has lasted for more than a century. However, the potential radiation risk still needs thorough assessment, meticulous control, and continuous improvement.

To ensure the protection of human life and the environment from harmful effects of ionizing radiation, the IAEA has (a) established fundamental safety principles, requirements and measures to control the radiation exposure of people and the release of radioactive material into the environment; and (b) has been providing assistance to its Member States through national, regional and interregional cooperation projects. These standards are not only applicable during normal operations, but also showed their efficiency during emergency situations.

The Kingdom of Morocco, as a Member State, is well-aware of its international obligations of diligence and duty of care and ensures its compliance with the IAEA safety requirements and relevant international instruments. The Moroccan Agency for Nuclear and Radiological Safety & Security (AMSSNuR) has committed itself to respect this international consensus and made it not only a national responsibility but also a regional and international commitment.

Aligned with the royal directives, AMSSNuR assigns a high priority to cooperation at regional and international levels and acknowledges the importance of its role to enhance and promote safety globally by exchanging experience, sharing information and improving capabilities through, inter alia, regional and international knowledge networks such as the Forum of Nuclear Regulatory Bodies in Africa (FNRBA), the Global Nuclear Safety and Security Network (GNSSN), International Network of Education and Training in Emergency Preparedness (iNET-EPR), Regulatory Cooperation Forum (RCF) …etc.

Besides its active participation in knowledge networks, AMSSNuR pays attention to regional and international cooperation through: (a) hosting workshops and meetings, (b) expert missions, (c) scientific visits, (d) bilateral cooperation and (e) multilateral cooperation.

Since its creation, AMSSNuR has been effective in developing approaches and mechanisms of cooperation through several Memoranda of Understanding with regional and international players and similar regulatory authorities in the field of radiation safety; all in line with the overarching goal of promoting safety culture, ensuring competence development and sharing lessons learned and good practices.

259-ESTABLISHMENT OF A FUNCTIONAL, EFFECTIVE AND SUSTAINABLE REGULATORY CONTROL SYSTEM: THE MOROCCAN EXPERIENCE

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Created by law n ° 142-12 of August 22, 2014 relating to nuclear and radiological safety and security, AMSSNuR regulates and monitors nuclear and radiological safety and security, to protect people and the environment from the harmful effects of Ionizing Radiation, to honor Morocco's international obligations related to the peaceful use of nuclear energy, and to inform the public objectively about regulatory processes and the safety aspects of the authorized activities.

In accordance with its 2017/2021 strategic objectives, approved by its Board of Directors at its first meeting in 2016, AMSSNuR has set up a road map, an organization and multi-year action plans structured around four major programmes mainly aimed at:

(a) Upgrading the national regulatory framework, through a concerted and collaborative approach, and continuous efforts for the establishment of a viable common regulatory basis applicable to all applications of ionizing radiation sources. This is done by bringing together the main principles enforceable as part of an integrated and proportionate approach inherent risks of activities for people and the environment.

(b) Establish and reinforce the regulatory control of practices and activities involving ionizing radiation sources by emphasizing and strengthening the graded approach to enforcement with a view to encouraging compliance and deter future non-compliances.

(c) Enhancing its human capital and skills as well as setting up the tools of governance in accordance with the legislation in force, government guidelines and IAEA standards with regard to the integrated management system for regulatory authorities.

(d) Promoting and sustaining a strong safety culture through the establishment of a communication strategy with stakeholders and the public about radiation risks posed by planned, existing and emergency exposure situations.

Contributing to the promotion of harmonized international doctrine of regulatory control of nuclear and radiological safety and security through the establishment and implementation of a concrete cooperation strategy at the regional and global levels. This is achieved through bilateral agreements reached with the sister authorities, and regional and global collaborative networks for sharing knowledge, experience and lessons learned (e.g. the Global Nuclear Safety and Security Network (GNSSN); the Forum for Nuclear Regulatory Bodies in Africa (FNRBA); the International Network on Education and Training for Emergency Preparedness and Response (iNET6EPR) …etc. This article presents the policy, strategy and diversity of the actions carried out and the efforts deployed by AMSSNuR to establish and continuously improve the national regulatory control in order to comply with relevant national legislation, the relevant international instruments, and the IAEA International Safety Standards.

Key Words

AMSSNuR, Radiation Safety, Regulatory Control, Safety Culture, Communication strategy, International Cooperation, harmonized international doctrine.

260-SAFETY SECURITY INTERFACE MOROCCAN REGULATORY BODY EXPERIENCE

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It is widely accepted that nuclear safety and security have a common overall objective of protecting human life and health and the environment from the harmful effects of ionizing radiation. While they share a common overall objective, they have different specific objectives and there are some points of potential conflict between them. To promote a common understanding of these issues and identify areas of synergies between safety and security, numerous recommendations have been made and meetings organised, in particular by the International Atomic Energy Agency (IAEA) with a view to sensitizing relevant stakeholders that security measures and safety measures should be implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

In the newly promulgated Law 142-12 on safety and security, and the creation of the unified and independent Regulatory Body (the Moroccan Agency for Nuclear and Radiological Safety and Security (AMSSNuR) [1]), regulating safety and security, it is clearly stated that, at the legislative and regulatory levels, safety and security measures must be implemented in an integrated manner based on international safety standards and security guidance and available good practices.

The paper describes AMSSNuR's regulatory approach in dealing with safety-security interface and, where justified, in integrating them. Initiatives and actions to implement safety and security integration include regulatory provision, continuous communication with various stakeholders in terms of activities related to the development, assessment and upgrading of the national regulatory framework, self-assessment of safety and security culture within AMSSNuR, having personnel from safety and security units, working together in fulfilling AMSSNuR's core functions and involving them in trainings in the areas of safety and security. These actions among others are intended to promote and harmonize efforts made to ensure that safety and security are taken in account in an integrated manner.

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261-SAFETY AND RADIATION PROTECTION FOR OPERATORS AND LABS PERFORMING NEUTRON ACTIVATION ANALYSIS AT THE RP10 RESEARCH REACTOR

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Neutron activation analysis is the second application of research reactors worldwide [1]. Instrumental neutron activation analysis (INAA) is the most commonly used method. It implies inducing radioactivity in a material (sample) using neutrons and consecutive measurement of radiation emitted in the process of radionuclides decay, until stable nuclei is formed. The variety of applications of this technique leads to use of irradiation facilities installed in the research reactor. Despite individual practices, the common denominator must be safety and radiation protection of personnel and the laboratory where samples are manipulated. National and international safety and radiation protection policies and standards are strictly enforced.

The RP-10 research reactor at IPEN is a 11 m pool type reactor that operates at a maximum power of 6 MW. To perform INAA two irradiation facilities are mostly used: one corresponds to the pneumatic channel where the thermal neutron flux is 3.0×10^{13} cm $^{-2}$s $^{-1}$ and the other one is a box in the periphery of the reactor core, with a thermal neutron flux of 4.47×10^{13} cm $^{-2}$s $^{-1}$. Operation of the pneumatic transfer system is made from the radiochemical laboratory. Samples inside a polyethylene capsules are sent in a time of 8 s to irradiation. At this position, biological and siliceous material are irradiated between 200 – 500 s to determined short lived radionuclides. The core position is used to irradiate the same materials inside aluminium cans (25 pellets within samples, comparators and monitors): biological samples for 18000 s and siliceous samples for 1800 – 2400 s to determine intermediate and long-lived radionuclides. All measurements are done in the gamma spectrometry laboratory located to 10 meters from the radiochemical laboratory. Moreover, the aluminium must be transported from the reactor hall to the radiochemical laboratory, adding 10 more meters to the transportation distance, therefore time of exposure. Hence, there are three environments involved in the handling of samples and exposure to ionizing radiation of personnel.

One way of ensuring health and minimizing the effects of ionizing radiation is to know current regulation and interaction of activities of the instrumental neutron activation analysis process with activities to be carried out in the safety and radiation protection process.

REGULATORY FRAMEWORK

Local laws, regulations, and international safety standards govern safety and radiation protection of installations and personnel using ionizing radiation. IPEN uses the Radiological Safety regulation and the regulation of the use of ionizing radiation sources (Law 28028). The first one regulates protection against exposure to ionizing radiation and safety of radiation sources that cause such exposure [2] and the latter regulates the practices that lead to exposure to ionizing radiation in order to prevent and protect the health of people, the environment and property from its harmful effects [3]. It also uses international standards issued by the IAEA on radiation protection and safety of radiation sources. This establish requirements for protection of people and the environment from harmful effects of ionizing radiation and for the safety of radiation sources. [4], [5]. Other guidelines establish the generally applicable requirements to be fulfilled in safety assessment for facilities and activities, with special attention paid to defense in depth, quantitative analyses and the application of a graded approach to the ranges of facilities and of activities that are addressed [6]. Knowledge of these guidelines is sufficient to guarantee protection against ionizing radiation of all actors involved.

GOOD PRACTICES CONSIDERATIONS

The regulatory body has established a constraint dose of 20 mSv per year over a period of 5 consecutive years. In the case of the AAN laboratory it has been declared a controlled area since the radiation is less than 3/10 of the annual dose limit (6 mSv). Based on these dose limits and knowledge of these practices, it is possible to guarantee the protection of the occupationally exposed personnel and the environment.

Figure 1 shows the AANI process and its interaction with the safety and radiation protection (S and RP) process. They are both linked. As the application of the technique is a planned exposure situation, the aim is to reduce the dose as low as possible. The beginning of the S and RP process is to know the type of sample to be analyzed and identify the chemical elements that compose it and then predict its activity at the end of irradiation (activity A). The input for this is to have characterized the irradiation positions (know the neutron flux) and define the analysis protocol (elements to be analyzed). On the other hand, the INAA starts with an analysis requirement and has the samples as input and the neutrons produced in the research reactor. Activity A is used in activity 1, 2, 3 and 5 to define the sample mass and irradiation, decay and measurement time. For activity 2 it is necessary to request it from the reactor operators by filling in a form for their approval.

Activity B should be performed after activity 2. At this point, the approval of the Radiological Protection officer (RPO) is required, prior to handling the samples when they are unpacked from the irradiation capsule. After the decay time (3.2, 3.3) the samples must be taken to the gamma spectrometry laboratory for measurement (activity 4). With the inspection of the radio-protection officer (activity C), shielding and appropriate distance for transfer must be defined. The dose of samples in the shielding must also be measured (activity D). It should be noted that only samples to be measured should be transported to prevent the conversion of the measurement laboratory into a storage room and cause unnecessary irradiation of other personnel. As the measurement activity is carried out in three times (activity 5), the dose of the samples must be measured before each measurement (activity E).

A next activity corresponds to the storage of the samples once they have been measured and become radioactive waste (activity 5.1 and 5.2). This storage (which is usually in the radiochemistry laboratory) should be inspected by the RPO (F). This is practically the last interaction of the safety and radiation protection process with the INAA process. The RPO then approves the transport of the samples to confinement.

Since several activities require the inspection and approval of a specialist in radioprotection, it is good practice to have a Radiological Protection Officer in the group of operators of the technique. The National Regulatory Body must license him. In addition, he/she should supervise the use of personal dosimeters and keep a record of the personnel dose.

Common good practices to both processes, which are also part of the ISO/IEC 17025 requirements, are [7]:

- Trained and authorized personnel to carry out the planned activities. In the case of S and RP, licensed to operate the technique by the regulatory body.
- To establish an annual training program
- To maintain the facilities in optimal conditions to perform work: temperature, humidity, ventilation, cleanliness, avoid contamination, use of fume hoods, monitor the indoor work area.
- Use of equipment with preventive maintenance and calibration: balances, spectrometers, radiation monitors, pneumatic transfer system, etc.

- To have procedures for the activities to be carried out
- To establish and maintain records of implemented controls during the process
- To consider risks associated with activities in order to prevent or reduce undesired impacts and potential failures in laboratory activities
- To conduct audits/inspections at planned intervals to provide information if requirements of standards and guidelines are met.
- To plan, implement and maintain an audit/inspection program.
- To identify and select opportunities for improvement by considering the review of procedures, audits results, risks assessment, evaluation of recorded information, etc.

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262-AMSSNUR'S GRADED AND RISK-BASED APPROACH TO REGULATORY ACTIVITIES

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Abstract

The Moroccan Agency for Nuclear and Radiological Safety and Security (AMSSNuR) is the National Regulatory Authority, established by the Law 142-12 [1] covering nuclear and radiological safety and security, and nuclear safeguards. The main mission of AMSSNuR, is to ensure compliance, of the radiological and nuclear safety and security of all the activities and facilities involving ionizing radiation sources, with the provisions of the Law 142-12, associated regulations, and Morocco's international obligations.
Since its establishment in October 2016, AMSSNuR has been working to upgrade and strengthen its regulatory control system to meet current and future challenges. Thus, the national regulatory control system focuses on building a graded approach commensurate with the risks associated with ionizing radiation sources.

This approach is adopted in different regulatory process, administrative measures and support actions carried out by AMSSNuR, in particular:

- **(a) Graded approach applied to the review, assessment and authorization:** In order that practices or activities involving ionizing radiation sources are framed appropriately to the challenges they present, Law No. 142-12 provides for two administrative regimes. These are the declaration regime or the authorization regime. This approach allows AMSSNuR to manage its regulatory control system according to the magnitude of the potential radiation risks arising from the facility or activity by focusing on higher risk and simplifying requirements placed on those of lower risk, without compromising the safety of the workers, public and the environment;
- **(b) Graded approach applied to operational radiation protection:** Administrative, organizational and technical measures established by the operator or under his/her responsibility, in order to ensure the appropriate radiation protection of workers, the public and the environment, are major elements of the operational radiation protection programme required by AMSSNuR before granting authorizations. This programme is defined on the basis of the risk associated with the activities, practices and associated installation;
- **(c) Graded approach applied to inspection and enforcement:** AMSSNuR is adopting a planned and systematic inspection programme. The scope of this programme and the frequency of inspections are commensurate with the potential hazard posed by practices, activities and associated facilities;

(d) **Graded approach applied to the integrated management system:** AMSSNuR is in the final phase of developing its Integrated Management System (IMS) combining all systems and processes and the ways with which they can be applied in a graded approach. This includes, deploying appropriate resources and requirements on the basis of consideration of importance to radiation safety and complexity of each activity, hazards and the magnitude of the potential impact, and possible consequences if an activity is carried out incorrectly.

The aim of this paper is to present the policy, the methods and tools for applying the graded approach in different regulatory activities performed by AMSSNuR.

Keywords: AMSSNuR, Regulatory Control, Radiation Safety, Graded Approach, Regulatory process.

263-REGULATORY INSPECTION PROGRAMME : CHALLENGES AND GOOD PRACTICES FROM THE MOROCCAN EXPERIENCE

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ABSTRACT

During the last three years, the Moroccan Agency for Nuclear and Radiological Safety and Security 'AMSSNuR', the unified and independent Regulatory Body in the Kingdom of Morocco, has been upgrading its regulatory framework covering nuclear and radiation safety and security. Through the cooperation with and support from the International Atomic Energy Agency (IAEA), AMSSNuR has developed and implemented a systematic and gradual inspection programme based on a graded approach for planning, conducting and monitoring inspection programmes in accordance with the provisions of the law n° 142-12 related to nuclear and radiological safety and security and the creation of AMSSNuR. The later is well equipped to effectively and efficiently implement the national regulatory control regime over practices and activities involving the use of radiation sources as established in its strategic objectives for 2017-2021.

AMSSNuR's integrated safety and security inspection system is based on its multi-annual progarmme of inspection commensurate with the potential magnitude and nature of the hazard associated with the facilities and activities involving radiation sources as required by the IAEA safety standards and security guidance, in particular the General Safety Requirements N°. GSR Part 1. The multiannual programme is translated annually on an inspection programme taking into account, as far as practical, human and financial resources of AMSSNuR and the risk associated with the radiation source and the complexity of the practice, as well as the possible consequences of an accident and the type and frequency of any violations found through inspections with a view to ensuring an adequate protection of workers, patients, the public and the environment.

As a results of this approach, AMSSNuR carried out more than 580 inspections of facilities and activities during 2017-2019 at twelve regions of the Kingdom of Morocco, covering medical facilities 83%; industrial facilities 13%

and others 4%, to verify that the authorized facilities and activities are in compliance with the regulatory provisions and with the conditions specified in the authorization. For the non-compliance identified during an inspection, the licensee is required to establish a remedial action plan to correct these deficiencies.

For some cases of non-compliance with the national legislative and regulatory provisions, AMSSNuR applies the oversight and enforcement provisions to maintain arrangements for safety and security as defined in the law n° 142-12 based on a graded approach. According to the gravity and the scope of the violation, AMSSNuR applies administrative measures or undertake prosecutions.

This paper aims to provide an overview of AMSSNuR's integrated inspection system and to outline the rationale for developing a systematic and gradual inspection programme based on a graded approach for planning, conducting and monitoring inspection programmes. This approach is considered a significant step forward in enhancing nuclear and radiological safety and security, and in ensuring compliance in all relevant sectors. It will be supported by the IT system for further improvement in the AMSSNuR's regulatory approach.

Key Words: AMSSNuR, Radiation Safety, regulatory inspection, graded approach, coercive measures.

264-DOSE CONSTRAINTS' LEGALLY "NON-BINDING" ROLE IN NUCLEAR APPLICATIONS

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ABSTRACT

This paper is challenging legally "non-binding" nature of Dose Constraints and its role in design, construction, commissioning, operation, and decommissioning of a facility conducting regulated activity.

Dose constraint levels for public and occupationally exposed workers are generally used as legally non-binding, but also present un-avoidable criteria in planning and implementation of radiation protection arrangements in any facility conducting regulated activity from the facility's design phase to its closure. Dose constraints also serve as important tool in assuring consistent regulatory approach to controlling groups of facilities in different industries. Derived levels pertaining to dose constraints provide important tangible handles to review and modify procedures, protocols, behaviours, and maintain overall safety culture. In this regard, the UAE regulatory framework mandates compliance as described in international guidance, mainly IAEA GSR Part 3.

The purpose of this paper is to discuss meaning and application of the legally non-binding nature of dose constraints in facilities conducting regulated activity according to the FANR experience gained over past 10 years.

265-EVALUATION OF AN INCIDENT LEARNING SYSTEM FOR RADIATION THERAPY: FIVE YEARS OF EXPERIENCE IN CLINICA LAS CONDES, CHILE

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Radiation therapy has shown a broad effect in cancer control and other diseases, on the other hand there have been some accidents related to both technical and human errors, nevertheless the use of new technologies has evolved then, in order to minimize risks to patients it has become necessary to develop more advanced quality assurance processes, such as the Incident Learning System (ILS), is defined as the ability of an organization to identify, report, and investigate incidents, then to take corrective action that improves the patient care system and this way reduces the risk of recurrence [1]. Although ILS for radiotherapy is gaining importance, such the recommendations published by: WHO's Radiotherapy Risk Profile, Toward Safer Radiotherapy, and ICRP Report No. 112. In addition to those protocols implemented by IAEA (SAFRON), AAPM and ASTRO (RO-ILS), ESTRO (ROSEIS), the terminology is diverse and fuzzy so must be clearly chosen to be able to implementing in the ILS to be developed.

In our ILS we use the terminology presented below, that is indistinctly all terms are catalogued as events as follows: a) Incident: are patient safety events that reached the patient, whether or not there was harm involved [2]; b) Adverse event: is an incident which results in harm to the patient [3]; c) Near miss: is an event that could have resulted in an accident or injury but did not either by chance or through timely intervention [2]; d) Error: is a failure to carry out a planned action as intended or application of an incorrect plan, and may manifest by doing the wrong thing or by failing to do the right thing, at either the planning or execution phase [3][4]. This retrospective study shows the results of five years of experience applying the ILS implemented in the Department of Radiation Therapy of Clinica Las Condes, Chile. Data were collected through the use of an incident report template developed and filled by Microsoft Excel® from January 2015 to December 2019, which refers to patients treated with an Oncor Impression Plus linear accelerator (Siemens), including the whole of the treatment techniques performed, using both Eclipse (Varian) and Prowess Phanter (Prowess Inc.) treatment planning system (TPS), and both Aria (Varian) and Lantis (Siemens) Record and Verify software (R&V), also using paper-based patient record until early 2017 and electronic patient record in the following years using Aria software.

The events implemented in this ILS has been categorized as follows: 1) Registration: which is related to the lack of information on patient data or treatment parameters included in the patient record, 2) Revision: refers to a verification of all the information related to the treatment plan by a member of the radiotherapy team, 3) Inappropriate dose: corresponds to a dose other than the prescribed dose in the treatment, 4) Dose in a different area: when the prescribed dose is delivered in an anatomical area other than the planned one, 5) Bolus: corresponds to the misuse of the bolus in the treatments that use it, 6) Accessories: it is related to the mishandling of all other accessories, immobilizers and supplies used in treatments, 7) Patient positioning: it is related to a poor positioning of the patient by the technician, 8) TPS: it concerns all related to TPS bugs and errors, 9) R&V: related to Aria bugs, 10) Hit with linac: corresponds to the blows made to the patient with linac, 11) Linac failure: concerns incidents related to technical accelerator failures, 12) Contrast media extravasation (CMEV): it is related to phlebitis caused by extravasation of contrast medium injected into the patient at the time of CT imaging. In addition, the following categories are used adverse events: 1) Inappropriate dose and 2) Dose in a different area: already defined above, 3) Patient fall: is any fall of the patient within the service, 4) Asphyxia: corresponds to airway obstruction, 5) Medical equipment and accessories (ME&A): they are related to accessories and medical equipment used in treatment, 6) Traumatic injury: those caused by the linac and accessories, 7) CMEV: already defined above. The ILS implemented contains the stages shown in Figure 1. The whole events are reported non-anonymously by any service staff and then classified according to their relevance in one of the categories mentioned above. For the adverse events, it is immediately reported on an electronic general notification form within the hospital's patient safety protocol, additionally a root cause analysis is subsequently perfomed within a committee formed of all team members. In some occasions, there is a feedback with external professionals in charge of the adverse event system of the hospital. The remaining event cases are presented and analysed weekly in a team meeting. In both cases, an action plan is implemented to increase the barriers that lead to the occurrence of events.

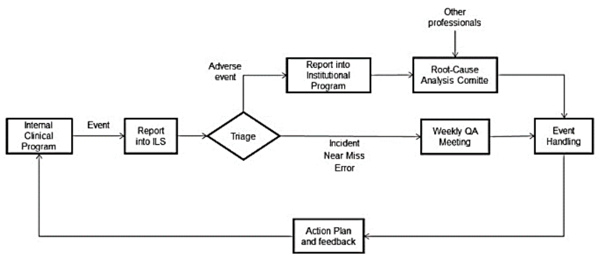

FIG 1. Schematic representation of the ILS implemented.

Along this 5 years evaluated, 177 events were registered, all of them were reported in the ILS (FIG. 2a), while nine of them were registered like adverse events according to the classification and characteristics of the ILS mentioned above (FIG. 2b) then were reviewed in the root cause analysis committee. The highest number of events was registered in 2016 (FIG. 2c) and the categories with the highest number of events were revision and registration during the first two years, corresponding to the errors with the most reported events (FIG. 2d). However, for the year 2017, the number of events for the registration category decreased by around 40% and for revision category by 90% due to the implementation of electronic patient record and correlation in the R&V system. Likewise, the number of errors has decreased by around 80% (FIG. 2d). The number of dose-related adverse events in a different area has decreased to no reported case. Although the total amount of events in general have decreased considerably over these years, events are being reported between 5 and 10% of patients treated each year in the service.

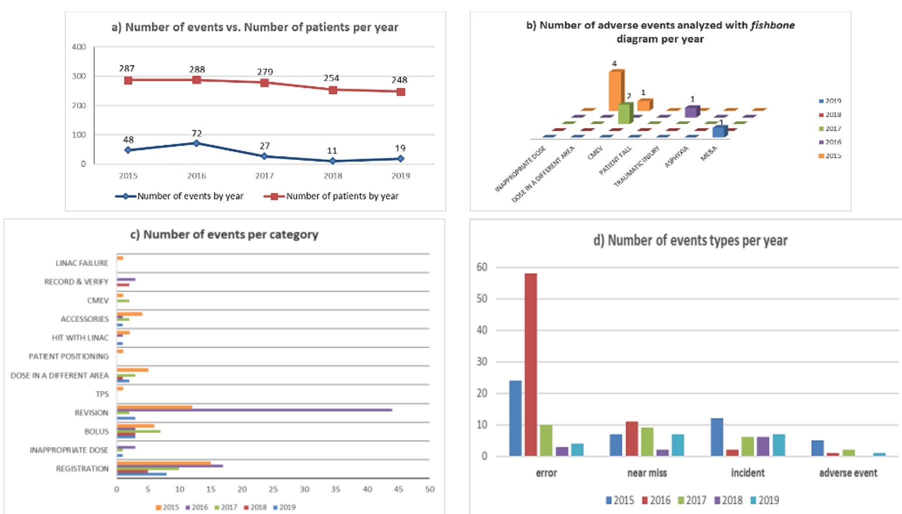

FIG. 2. Results of the five years ILS evaluation.

It is important to perform the design, implementation and evaluation of an ILS in a radiation therapy service therefore include it within the quality assurance program, since it enable us to know specifically the entire process to increase and optimize the barriers to the occurrence of any type of event in the treatment of patients, in addition to establishing a collaboration commitment with the entire work team as a key piece in patient safety.

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267-REGULATORY EVALUATION OF RADIOACTIVE DECORATIVE PENDANTS

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ABSTRACT

Radioactive consumer products, including jewellery and pendants, are shipped all over the world. The presentation summarizes some of the regulatory issues associated with commercially available decorative pendants commonly imported into the United Arab Emirates. An example pendant is shown in figure 1. The regulatory issues identified in the presentation are based on a technical evaluation of the decorative pendants conducted by the Federal Authority for Nuclear Regulation's (FANR's) radiological environmental laboratory. The presentation summarizes the gamma isotopic analysis, the technical evaluation of that analysis, the dose assessment of the decorative pendants, and how seemingly trivial technical nuances can impact decisions associated with regulatory actions. The presentation will review some of the challenging technical issues encountered with respect to implementing the guidance in IAEA Safety Guide RS-G-1.7, Application of the Concepts of Exclusion, Exemption and Clearance, which is adopted by FANR through FANR-REG-19 "Existing Exposure".

FIG. 34. Example pendant with partial metal cladding and ceramic interior.

The presentation will address the entire chain of events comprising the evaluation process including the analytical method for quantification of radioactive materials, the method for dose assessment, and application of regulatory requirements. Special challenges encountered during the evaluation process will be discussed. For example, the radioactivity content of the jewellery is impacted by the particular analytical method chosen for quantification. Initial screening analysis with common button sources indicated activity of the pendants may approach regulatory limits, but the analytical uncertainties proved problematic. A special calibration source, in the exact shape of the pendants, was used to develop an efficiency curve for gamma isotopic analysis, shown in figure 2. This technique reduced analytical uncertainties to level acceptable for potential future regulatory actions.

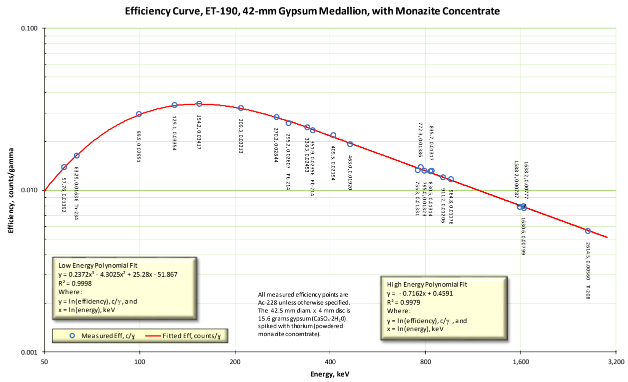

FIG. 2. Efficiency calibration curve for gamma isotopic analysis.

Skin dose represented the limiting exposure. Dose evaluation with VARSKIN computer software showed how dose is impacted by self-shielding of pendant materials. Techniques of bounding analysis helped to estimate exposure for the cases of both typical product use and improper use of the product. All parts of the analysis and evaluation process had implications regarding application of the regulatory limits (i.e., for concentration and dose).

The purpose of the presentation is to highlight the regulatory issues identified in the associated technical report, identify potential policy issues that may need to be resolved, highlight technical issues that affect compliance with IAEA guidance, and propose recommendations that will allow the national regulator to make an informed decision regarding the appropriateness of any future regulatory actions with respect to distribution of these products in the UAE.

269-NUCLEAR SAFETY CULTURE IN VIETNAM

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Culture is deeply rooted, from within the roots of society, not just superficially, so it is quite stable over time. Culture is shared and related with not only an individual, but a group, a community, or an organization, a society. The culture is vast and encompasses all aspects of the external and internal relationship of a group, a community, or an organization, a society. Building blocks of culture are values, attitudes and social norms.

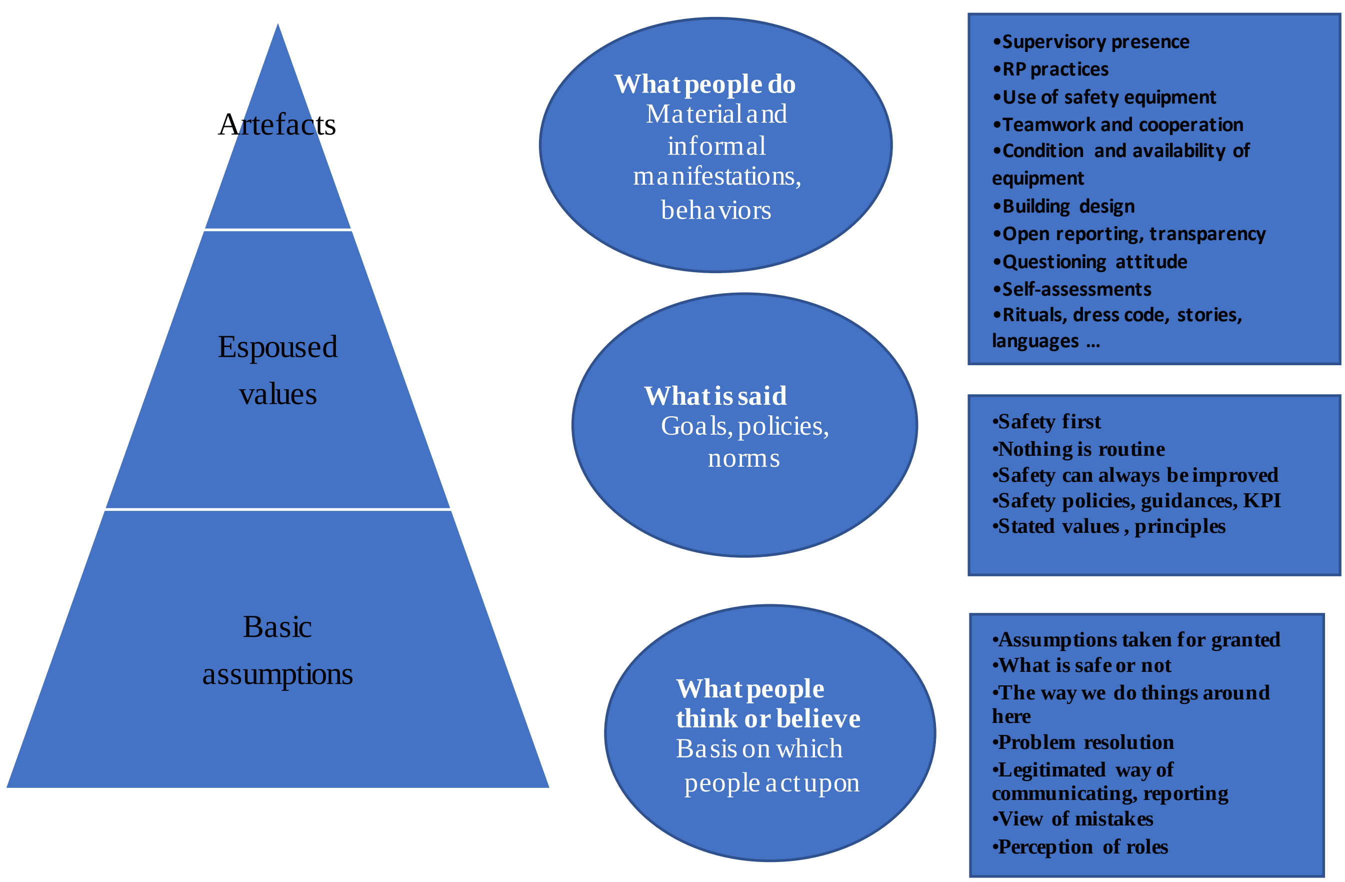
I. Safety Culture

As INSAG 4 Definition:

"Safety culture is that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance"

So Nuclear Safety Culture, whether spoken or not, has existed in countries with nuclear reactors and nuclear facilities at different levels.

In Vietnam, Nuclear Safety Culture exists in nuclear facilities (such as Da Lat nuclear reactors) and related organizations and individuals at different levels.



II. Safety Culture at the government level

Government policies for the use of nuclear power set broad safety objectives, establish the necessary institutions and ensure adequate support for its safe development. To ensure safety, the Vietnamese government has established a nuclear regulatory body, which is responsible for helping the government ensure the safe operation of research nuclear reactors as well as other radiation facilities and ensure safety for nuclear power plants in the future. The regulatory body is provided by the government with both human and financial resources to carry out the task of ensuring national safety in the field of radiation and nuclear.

III. Responsibility of Regulatory body for Nuclear Safety Culture

- Responsible for ensuring radiation and nuclear safety throughout the country.
- Developing legal documents related to radiation and nuclear safety.
- Inspecting and examining the assurance of radiation and nuclear safety at the radiation and nuclear facility.

IV. Nuclear Safety Culture at the Organization level

- Commit to implement necessary measures to ensure safety at facility.
- Develop safety implementation plan.
- Establishing related units to ensure safety.
- Develop safe working processes.
- Examining the radiation safety implementation on the attached functional units and agencies.
- Periodically assess the safety at the unit and the safety performance at work for each individual.

V. Nuclear Safety Culture for individual

- Develop an individual safety plan.
- Comply with the safety regulations of the unit.
- Regularly review the implementation process to ensure a sentence.
- Self-training and participation in safety and professional training courses.

Conclusion

Safety culture in general, nuclear safety culture in particular is very important. To ensure the safety of operation of nuclear and radiation facilities, compliance with safety regulations from the governmental level to operator must be strictly enforced. In Vietnam, safety culture has been gradually and increasingly widely deployed at different levels. We believe that the nuclear safety culture will contribute to the safety of nuclear facilities in Vietnam.

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270-NUCLEAR AND RADIOLOGICAL SAFETY AND SECURITY EDUCATION AND TRAINING STRATEGY: LESSONS LEARNED FROM THE MOROCCAN APPROACH AND METHODOLOGY

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In order to strengthen national capacities in safety and security and ensure safe and secure use of ionizing radiation sources in the sectors of health, industry, mines, agriculture, transport, research and higher education, AMSSNuR carried out, in 2019, the national strategy study for education and training in nuclear and radiological safety and security (NRSS). This study is based on IAEA's training methodology and on royal guidelines.

Through the implementation of the training strategy, AMSSNuR aims to strengthen national skills in a sustainable manner in the fields of NRSS; and to contribute to the development of NRSS skills in Africa at the service of our country's regional cooperation.

1. Results of the Study

The strategy identified more than 14,000 persons to be trained or qualified in accordance with the provisions of the Law No. 142-12 [1], most of whom operate in medical and industrial sectors.

Besides, in accordance with royal guidelines in terms of development of cooperation in Africa, the study integrated, also, training offers that the Kingdom of Morocco can provide to other African countries to respond to their needs for developing skills in NRSS. Those needs are estimated at 500 people per year.

The study revealed that over the next five years, the training needs are distributed as follows:

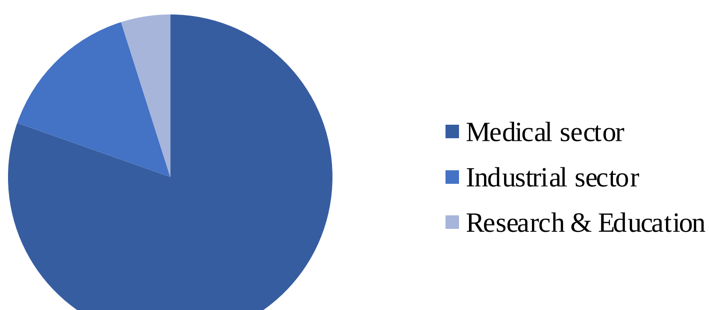

FIG. 1. Chart showing the qualification needs per sector.

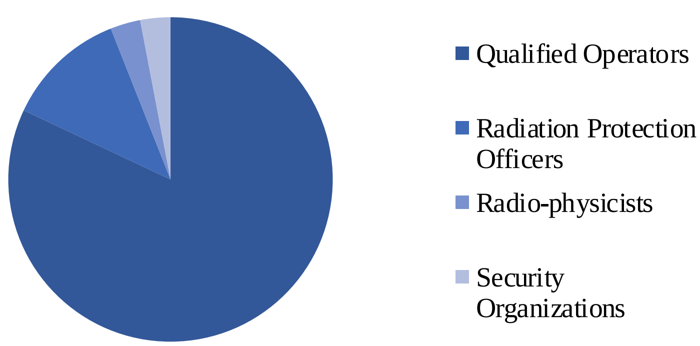

FIG. 2. Chart showing the new training needs per profile.

It should be noted that national training needs are centered at more than 52% within the cities of Casablanca and Rabat.

2. Training Offers Analysis

In terms of basic training offers, the study revealed that national universities have predispositions to teach subjects related to nuclear and radiological safety and security, as is the case for the universities of Rabat, Settat, Casablanca, Tetouan, Fez, Kenitra and Oujda.

For professional training, the training centers of the Ministry of Health, the Professional Training and Job Promotion Office and the National Centre for Energy, Nuclear Sciences and Techniques (CNESTEN) could play a very important role in the execution of training programmes.

Regarding security authorities, the study revealed specific training needs within the National Defense Administration, the Royal Gendarmerie, the General Direction of the National Security and the Directorate General of Civil Protection.

3. Main achievements

In order to implement this strategy, AMSSNuR drafted an agreement, with the Ministry of Education, the Ministry of Health, the Ministry of Energy and Mines and the CNESTEN, aiming to define the partnership framework and the mechanisms to implement professional training programs identified within the national strategy for theoretical and practical training in the field of nuclear and radiological safety and security.

Furthermore, a steering committee and a monitoring committee were set up to proceed to the development of descriptive sheets of two priority programs relating to:

(a). Basic training level of two years (Specialized Technician in Radiation Protection);
(b). Continuous education in the fields of nuclear and radiological safety and security.

The implementation of these two programs will make it possible to meet, in a progressive manner, the professional training needs necessary for the medical and industrial sectors estimated at more than 8,000 persons.

4. Supporting measures

Regarding the regulatory framework, AMSSNuR drafted an ordinance fixing the required training and qualification levels for qualified operators, Radiation Protection Officers and Radiation Protection Experts.

Besides, AMSSNuR reiterates its will and its disposition to:

(a). Work together with concerned parties to meet the needs identified in professional training in NRSS;
(b). Involve all concerned parties in the definition, the implementation and the evaluation of training programmes;
(c). Contribute to the governance structures of the national NRSS training strategy;
(d). Contribute to the communication with training providers;
(e). Develop mechanisms for evaluating the effectiveness of NRSS training;
(f). Establish agreements between AMSSNuR and the various concerned parties.

Conclusion

As short-term future actions, AMSSNuR is aiming to:

(a). Organize seminars for master's students in the field of NRSS;
(b). Review training programmes delivered by medicine faculties in order to propose additions relating to radiation protection;

(c). Schedule meetings with representatives of security authorities to discuss possible training offers.

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271-IMPLEMENTATION OF A CONTAMINATED MEDICAL WASTE MANAGEMENT SYSTEM *IN HOUSE* ACCORDING TO THE BSS DIRECTIVE IN THE UNIVERSITY HOSPITAL OF SASSARI

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A new system for contaminated medical waste has been implemented in 2019 in order to reduce the costs due to of the authorized company and the amount of contaminated medical waste.

Until 2017 in AOU Sassari (Azienda Ospedaliera Universitaria of Sassari) all wastes from Nuclear Medicine Unit and Radio Immunological Assessment (RIA) Laboratories are sent to a company with an annual cost equal to 12,2 k€. With a program of sensibilization, information and formation of workers, the amount of total contaminated waste has been reduced from 74 to 40 casks, through the re-organization of the management system following IAEA Safety Standards and requirements [1-3].

In AOU Sassari these radionuclides are administered:

- diagnostic use ^{18}F, ^{99}Mo, ^{123}I, ^{99m}Tc, ^{111}In, ^{201}Tl, ^{67}Ga, ^{131}I;
- therapeutic use ^{223}Ra, ^{131}I;
- laboratory use ^{125}I;

Radioactive wastes are materials that have been contaminated by radionuclides in Nuclear Medicine patients' procedures or in vitro procedures. Radionuclides used in therapeutic and laboratory uses have a longerr half life than diagnostic radionuclides [4].

The total activity held from 2012 to 2017 was:

	Total Activity (Tbq)	Total Activity with $T_{1/2}$> 4d (GBq)	% Activity with $T_{1/2}$> 4d
2012	1,006	24	2,37
2013	0,923	30	3,23
2014	1,403	19	1,35
2015	11,334	20	0,18

2016	8,848	17	0,19
2017	9,409	18	0,19

Time to dismiss the quantity of ^{131}I according with the Italian Normative Clearance condition was estimated in 7-8 months.

The Contaminated waste produced by AOU Sassari is both liquid and solid, although the liquid portion (obtained only from iodine used in the RIA laboratory) has significantly reduced since RIA practices have been fallen with the introduction of new methods that use non-radioactive substances.

Workers have been trained to differentiate contaminated sanitary material (needles, gauze, sheets, etc) in order to separate those with a shorter half-life [1].

In Fig.1 the trend of waste production in the period considered:

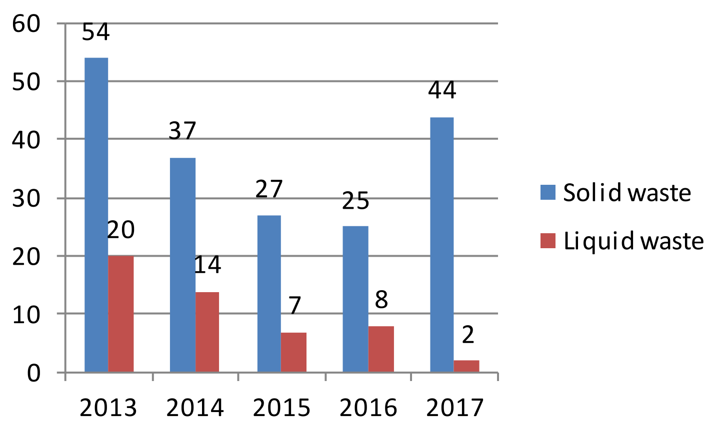

Fig.1 Number of waste cask (60 liter) in the period 2012-2017

The cost of single cask is estimated, referred by authorised company contract, in:

Type of waste	Cost (€/kg)
Solid contaminated waste	15,33
Liquid contaminated waste	7,67
Health-care waste	1,25
Urban waste	0,35

In 2018 and 2019, respectively, 54 and 83 casks have been managed through the new procedure, for saving of 12k€ and 19k€ respectly, justifing the investment in a waste deposit (moveable container) and instruments (mobile spectrometer with a NaI detector). A “Delay and Decay” approach was applied in order to transform radioactive waste in Health-care waste [5].

At the time of delivery as special medical waste, the contact dose rate has always been measured equal to the environmental background (<150 nSv/h)). Likewise, upon arrival at the temporary storage, measurements at the external wall of the container have always been equal to the environmental value, thus preserving the exposure of any workers.

The construction of the temporary warehouse resulted in a reduction in costs for the hospital, net of the material costs of the container and the equipment, and at the same time it brought greater awareness for the workers that identifying contaminated materials in advance reduces the amount of waste contaminated product.

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272-RADIOLOGICAL IMAGING OF ASYMPTOMATIC INDIVIDUALS: GUIDANCE FOR JUSTIFICATION, GOVERNANCE AND DECISION MAKING IN CT-IHA

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BACKGROUND

The use of medical imaging technologies has greatly increased over the past few decades. It has served medicine well in addressing the needs of patients and opening new horizons to improved care. In addition, computed tomography (CT) is increasingly applied to screen asymptomatic people for the early detection of disease. A limited study indicates that this practice occurs in most countries of the world. These screening practices are in both organized population-screening programmes and less structured settings. Even with organized programmes, balancing benefits and harms is critical in ensuring positive outcomes. Such procedures are denoted as individual health assessment (IHA) and may often challenge justification [1]. There is little evidence that IHA or CT-IHA is of benefit in asymptomatic individuals. The issues that arise are identified and the possibility of developing guidance is addressed in a new WHO publication [1,2]. To illustrate the scope of the problem, a plausible scenario is developed involving CT scanning of an asymptomatic person without a medical referral.

HEALTH GOVERNANCE, BENEFITS, HARMS AND IMPROVING PRACTICE

At an early stage of the CT-IHA project, it was clear that health governance issues needed to be considered and had much to offer. For example, there is extensive experience in non-radiological screening programmes that could help formulation of guidance (e.g. diabetic retinal screening). Guidance must reflect radiation quality and safety regulation and standards; the requirements for equitable and quality health services; societal values; ethics; health economics; public health considerations; and high levels of uncertainty involved throughout. Even where authorities are aware of the CT-IHA challenges to justification, there is reticence about taking action, for many reasons, including important concerns about personal freedom and empowerment.

Widely used medical terminology of benefits and harms is used in preference to detriment and risks. The obvious harms include probable radiation harms, mortality, morbidity, false-positives and negatives, incidental findings, overdiagnosis/overtreatment, related stress, and direct/indirect costs. In practice it is difficult to directly compare these with the expected benefits of early diagnosis.

GUIDANCE/ REGULATION AND EXISTING STANDARDS

A taxonomy of CT radiological screening activities was essential to developing governance advice for CT-IHA activity, which is classified into five types, based on eight features. Types 1 and 2 characterize screening programmes that are well structured and conducted within reliable governance, regulatory, licensing, inspection and audit frameworks. Types 3, 4 and 5 address IHA and CT-IHA where elements of an appropriate framework may be absent. For further information, see reference [2].

The *International* and *EURATOM Basic Safety Standards (BSS)* provide the basis for well-regarded radiation regulatory systems [3, 4, 5]. Key to ensuring quality services are notification, authorisation, and licensing which place clear responsibilities on the licensees and professionals. In addition, it is important to provide appropriate regulations, guidelines, standards, compliance verification processes, arrangements for education and training, and ensure respect for ethics.

The *International BSS*, co-sponsored by the IAEA with WHO and 6 other international organizations, differentiate between screening programmes and IHA in requirements 3.159 and 3.160. In formal screening programmes responsibility for collective justification of those screened rests with the health authority/ radiation protection competent authorities in conjunction with the appropriate professional bodies. In IHA the approach reverts to the requirement for justification for the individual presenting for the procedure.

Accepting and successfully implementing a comprehensive system dealing with both approaches is one of the major challenges for screening programmes and CT-IHA, and requires the following be addressed:

- consistency with the international standards mentioned above;
- definition of eligibility criteria based on risk profiles;
- provision of adequate information about benefits and harms to the individual;
- provisions concerning self-referral and self-presentation;
- provisions to ensure that promotion and advertising are honest;
- provisions to ensure adherence to accepted ethical values and standards of professionalism;
- adequate staff training and education with reference to screening;
- adequate equipment with reference to screening;
- adequate protocols for imaging techniques including appropriateness;
- adequate protocols for follow-up based on the screening results;
- adequate QA programmes, including documentation, at all levels;
- adequate certification and recertification processes;
- adequate mechanisms for process and outcome evaluation at all levels; and
- adequate arrangements for audit and inspection, including of justification.

In well-established screening programmes most, if not all, of these are already addressed. With CT-IHA, the situation is different, and much of the framework may be absent. In some cases, this might be acceptable, but the complete absence of the framework requirements is unlikely to be acceptable. The legal framework for any screening practice, either performed as a formal programme or as CT-IHA, should be consistent with health policy.

ACKNOWLEDGEMENTS

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273-ETHICS FOR RADIATION PROTECTION IN MEDICINE: A WHO/ICRP STAKEHOLDER INITIATIVE AND THE ETHICS RESPONSE TO COVID-19.

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The prevailing imperatives for professional ethical behaviour in medicine have radically altered since the introduction of the present system of radiation protection. Until recently the values underpinning the system were not explicitly identified, and this has now been achieved for its general application in radiation protection [1]. An additional report on medical applications is awaited [2]. In support of the latter, the World Health Organization (WHO) organised a joint workshop of relevant stakeholders with ICRP in September 2019. A parallel literature on a pragmatic approach for ethics in radiation protection in medicine has developed and matured. It involves a set of five values, that are widely accepted throughout the world and are closely allied with those identified by ICRP [3][4]. The pragmatic value set is rooted in the medical tradition, in tune with modern social expectations, satisfies a non-negotiable medical ethics requirement, and can be expected to improve radiation protection in practice.

WHO also has a unique experience of large public health events up to and including pandemics like Covid-19. This gives it an exceptional understanding of the practical impact of the relevance of ethics in managing such events. These can be helpful to parallel situations in radiation protection. Radiation protection professionals will recognise striking similarities between the issues involved in the Covid-19 crisis and their own experience. Both deal with uncertainties although those associated with Covid-19 are greater. Both also contend with unknowns in individual clinical outcomes, and in the allocation of expensive therapeutic interventions with limited availability (e.g. radiosurgery and intensive care of seriously ill Covid-19 patients). In both, compliance with a recognised ethics system allows a profession to assert its behaviour as being of a high moral standard. This is reassuring to health professionals, citizens and decision makers alike.

WHEN IS ETHICS USEFUL?

Professional codes of ethics and value systems can be useful where formal advice is lacking, inadequate for the situation, temporarily suspended or unavailable. Examples of situations where a personal sensitivity to ethics is both necessary and useful occur where:

- Law/protocols lack maturity.
- What ought to be done can't be done.
- There is inherent uncertainty.
- Accidental exposure or incidental findings occur.
- New services or ways of managing existing services are being contemplated.
- The working environment changes rapidly and significantly.
- Services are being reconfigured to render them more patient centred.

Well accepted value systems agree that dignity/autonomy, beneficence (do good), non-maleficence (do no harm), justice, and prudence ought to drive ethical reflections. Balancing these values can be problematic in practice. In situations where experience, regulation or guidance is inadequate additional values may be required and many suggestions in this regard emerged from the WHO Stakeholder workshop, including solidarity, empathy and inclusiveness. These additional values have already been widely implemented in the field of Public Health where population (as opposed to individual) health is the focus. We already have limited experience of some of these values in screening programmes such as mammography and in large accidents.

DISCUSSION

Public health professionals have a tremendous reservoir of both practical and ethics experience from which we can learn much. The WHO longstanding work on global health ethics is a useful source of experience for radiation. It has a deep experience of ethical issues and of the considerations important to delivery of quality services across whole populations. Its contribution may be particularly helpful where a wider group of values may need to emerge to improve radiation protection in practice in general, but also in the context of catastrophic accidental situations

It is of interest that WHO has consistently advised that the most powerful weapon available in dealing with Covid-19 is solidarity. Solidarity is an ethics value, which can be derived from those above, and places greater emphasis on the health of all than on financial security for a few. Solidarity is to society, in part, what beneficence and non-maleficence are to the individual and as such may have a powerful role to play in radiation protection. To paraphrase Meskens who asserts that: - solidarity requires dialogue, exchange of words, ideas, and knowledge [5]. This value beckons us to recognize the knowledge we collectively have; to acknowledge the imperative character of the complexity of the issues we are addressing; and to seek intellectual solidarity and engage in deliberation with other concerned participants. Even more important, solidarity, wedded to prudence and precaution, obliges us as scientists to recognise the reality of the uncertainties in our current situation and collectively adopt a responsible attitude to it.

As radiation scientists, healthcare professionals and citizens, we must be aware of the bonds that hold us together and act for the good of society. Thus, we respond favourably when asked to come to work in complex demanding situations, when many others will be safe at home. This is also why we share knowledge of many matters of fact, great and small. We are educated on aspects of managing uncertainty, and in voluntarily accepting higher occupational radiation risks than those borne by the public or patients. We are given a unique trusted position to advise on complex, incompletely understood risk management situations. Solidarity, and the idea that 'we are all in it together', help make sense of the complexity of health risk governance in general and of diagnostic and therapeutic radiological practices in particular [5].

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274-RADIATION SAFETY IN NON-MEDICAL APPLICATIONS IN THE KINGDOM OF MOROCCO

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The use of ionizing radiation sources is increasing in the non-medical sector in the Kingdom of Morocco. Indeed, around 20% of the activities involving ionizing radiations relate to non-medical applications in several socio-economic sectors, such as industry (with 2 cyclotrons for the production of Fluorine-18, more than 15 industrial radiography companies, more than 400 nuclear gauges, etc.), agriculture (with one agro-food irradiator), security control (with more than 12 cargo inspector and thousands of baggage scanners), etc. This comes down to the many benefits derived from ionizing radiations in their various applications; However, these benefits do not obscure the dangers associated with ionizing radiations nor the need to protect radiation workers, the public and the environment against them. This is where turns out the importance of the role of the Moroccan Agency for Nuclear and Radiological Safety and Security, "AMSSNuR", which is to regulate the use of ionizing radiations. Indeed, AMSSNuR is an independent regulatory authority that has been, since October 2016, responsible for ensuring the compliance of practices involving sources of ionizing radiations with legislative and regulatory requirements related to nuclear and radiological safety and security.

In this regard, and according to its strategic plan 2017/2021, AMSSNuR has been upgrading the national regulatory framework for safety, security & safeguards, in accordance with the provisions of the Law No.142-12 and Morocco's international commitments, by adopting IAEA Safety Standards and Security Guidance. Regarding radiation safety of non-medical activities, this upgrading process covers important improvements such as:

(a) new provisions on the qualification of qualified operators, as a measure of enhancing national capacities and competencies, allowing the operators to fulfil their responsibilities; The list of devices used in non-medical activities and that are required to be handled by qualified operators, as well as the modalities of training, qualification and recognition by the regulatory authority of qualified operators are set by the regulation that is being developed.

(b) specific provisions for enhancing the safety and security of activities involving vulnerable radioactive sources such as sources used in industrial radiography, in particular the rules for the design and layout of storage facilities for radioactive sources, as well as of industrial radiography sites.

(c) provisions for the recognition of technical service providers; Conditions are being defined for obtaining approval for the provision of a technical service in radiation safety.

AMSSNuR has developed a significant experience in the regulatory control of non-medical activities and facilities involving ionizing radiations, and aims to constantly enhance it and share it within and outside Africa region, through:

— Review and assessment and granting authorizations using procedures and operational tools (notification form, application forms for authorization, authorization procedures) inspired by IAEA TECDOC-1525, considering the graded approach. These tools are developed for different classes of activities and facilities in accordance with GSR-Part 3 requirements;
— Conducting regulatory inspections (over 16% of total inspected facilities in the Kingdom of Morocco are in the non-medical field) according to a graded approach, which is taken into account in defining the scope and the frequency of inspections, in accordance with AMSSNuR's inspection procedures and specific checklists used during inspections. These procedures and checklists are part of an Integrated Management System based on the IAEA standards and guidance, specifically IAEA TECDOC-1526;
— Digitalization of its regulatory processes, allowing, at the organizational level, a better mastering of all the decision-making information gathered from different processes, and a better interaction with license applicants and licensees.

This article presents the actions carried out by AMSSNuR in establishing and improving necessary regulatory requirements and administrative measures to ensure compliance with Law No.142-12, and compatibility with the requirements of GSR part 3 and the recommendations of code of conduct for the safety and security of radioactive sources to further improve the radiation safety and security in non-medical applications.

Key words: AMSSNuR, Radiation Safety, non-medical activities, regulatory requirements, ionizing radiations.

275-NON-MEDICAL HUMAN IMAGING IN UKRAINE

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The use of radiation sources for human imaging for non-medical purposes is an additional exposure to human body that requires special justification in each individual case. The imaging in airports and customs is carried out without a doctor's prescription and does not bring benefits to human health.

On the other hand, such an imaging is a need, because it contributes to the increase of the safety of society: prevents the carriage of prohibited items and substances, minimizes the probability of terrorist attacks.

How to avoid unjustified exposure? Are there protective measures envisaged in European countries? What is the position of IAEA and national nuclear regulatory authority?

The studies conducted by the World Health Organization, as well as the studies carried out in the IAEA member states, confirm the probability for risks of negative effect on human health from non-medical imaging.

How is it explained? First at all, exposure of people, including children, pregnant women in airports and customs checkpoints can be carried out without warnings and explanations about possible harm to the health for a particular person.

According to international data, effective exposure dose in case of using low-energy x-rays (backscatters) varies from 0.02 µSv to 0.1 µSv per one scanning. In case of using of x-rays transmission imaging the value varies between 0,1 µSv to 5 µSv per screening [1]. Both methods are used to detect explosives and drugs under human clothes. Applying of computer tomography leads to exposure level over 10 mSv per scanning. The computer tomography allows scanning of human body to identify drugs and other prohibited materials, the removal of which is impossible without surgical intervention.

Recently, radiation sources have been used more often in countries for human imaging in the places of mass gathering of people: shopping centers, banks, stadiums, buildings with strategic importance, check-points at the airports. One should not forget about risks of scanning cargo at customs. Use of mobile and stationary radionuclide facilities, as well as linear accelerators for cargo scanning can cause and sometimes lead to additional exposure to human in many countries. First of all, this means truck drivers, illegal migrants who cross the border in trucks or cars, and customs officers at checkpoints.

Today, we already have some experience in the radiation safety of such systems. Recently, seven units of stationary systems using linear accelerators were installed in Ukraine to scan trucks. During their commissioning, special attention was paid to ensuring the safety of not only the customs staff, but also the drivers and passengers of the vehicles being inspected. In addition, some organizational measures aimed at preventing accidental exposure of unauthorized persons have been introduced in the Radiation Safety Regulations for customs staff. These include: a quick visual inspection of the car; interview with the driver; moving the driver and passengers to their temporary location for the duration of the scan, analysis of supporting documents for the purpose of establishing risk routes and cargoes prior to the start of the scan.

Particularly difficult cause the question of avoiding the accidental exposure of people moving across the border illegally, hiding between the cargo, or in places of a vehicle that are difficult for survey. Unfortunately, this scenario cannot be completely ruled out yet. Although, it should be noted that it is not very typical for Ukraine, especially since linear accelerator scanning systems are now located on the western border towards the entrance to Ukraine, where illegal people traffic is not typical or, moreover, mass.

Considering this, Ukraine support the initiative of the IAEA and EU and started to implement the unified international recommendations on radiation protection, ethical aspects of using non-medical imaging. There is a need to implement dose limits for the public, stuff and other persons, who may be subject to such exposure.

According to a new IAEA standard [3], human imaging using radiation sources for the detection of concealed objects (for smuggling or use for malicious purposes) shall normally be deemed to be not justified. When the government or regulatory authorities decide that the justification of such human imaging is to be considered (benefit to the society is higher than harm to a certain person under exposure), such activities shall be obligatory subject to regulatory control through the licensing and oversight.

This issue is also addressed in Article 22 of Council Directive 2013/59/Euratom dated 5 December 2013. The document establishes the main safety regulations for protection against hazards caused by radiation. Annex V of the article indicates the list of such practices, where it is possible to use radiation for non-medical human imaging. It also specifies requirements for the regulation of this sphere.

Currently, it is planned to use several such installations in Ukrainian airports as auxiliary surveillance equipment (in case of a suspicion of drugs or explosives). This equipment will not be used for mass inspection.

We need to be very careful about introducing such equipment to use in the country. There should be a very clear line between when such equipment can be used in terms of national security or the safety of large numbers of people and when such equipment simply facilitates the work of security guards. It is unacceptable if tomorrow such equipment will appear uncontrollably on the entrances to office, shopping or entertainment centers in large quantities. Using it at airports also requires a prudent approach, as different categories of people may be exposed to different frequencies, such as aircraft crew, couriers or frequent flyers, comparing to passengers who board one plane once a year, on vacation.

Taking into account international experience, the SNRIU considers that each such practice shall be justified. The benefit of using imaging shall be higher that harm to health of a certain person subject to exposure. We developed a draft regulatory document that sets requirements for safety justification of using such facilities. Soon, the document will be issued for the public and professional discussion according to the procedure established by the law.

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277-LIMITED PROJECTION X-RAY CT RECONSTRUCTION FOR MEDICAL APPLICATIONS BY USING ITERATIVE RECONSTRUCTION TECHNIQUE

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In medical applications, radiation imaging has played a critical role in assisting physicians for diagnosis of diseases, as well as radiation therapy (RT) planning and treatment. Generally, the duration of RT for cancer treatment can last several sessions throughout days or weeks. The patient may undergo some anatomical changes between sessions in response to the treatment. The X-ray CT imaging is taken at the beginning of each RT session to ensure that the anatomical changes do not exceed the pre-defined threshold. Otherwise, the treatment needs to be revised again to avoid any damages that could cause normal tissues from the changes of patient's anatomy. Since the X-ray CT imaging needs to be taken before every session of the Image-guided Radiation Therapy (IGRT), the radiation exposed by the patient can be rather significant. This has become an issue of concern, as well as the growing usage of radiation imaging during clinical examinations in other medical applications.

It is of the utmost importance to control an amount of radiation exposed to the patients to be as low as reasonably possible, but still be able to produce a good quality of image. One way that can be implemented for this purpose is by reducing a number of projection images used for image reconstruction process. Instead of acquiring a set of projections from an entire 360-degree circle around the patient, only some projections from intervals of angles are obtained. Generally, the projection images will go through a mathematical process of reconstruction algorithm to compute the final reconstructed image that represents the map of tissue densities inside the patient. In general, the reconstruction algorithm that is widely implemented is filtered backprojection method of Feldkamp, Davis and Kress (FDK) [1]. The FDK method has gained its popularity among the commercial CT scanners because it is a straightforward method and easy to implement. However, the FDK algorithm requires a complete set of projection images from all angles to produce a good quality reconstructed image. When the projections are acquired in a limited number to reduce the amount of radiation exposure to the patient, the FDK algorithm perform less efficient and often produces reconstructed image with severe artefacts[2].

When the projection data is sparse, the X-ray CT image reconstruction problem is ill-posed and sensitive to measurement noise. As an alternative to FDK algorithm, another category of reconstruction algorithm namely iterative algebraic reconstruction algorithm is used. The iterative algebraic reconstruction algorithms are capable of handling undersampled data and producing reconstructed images with high quality. It has been reported in [3] that the iterative reconstruction algorithms provide similar image quality to that achievable with the FDK algorithm at 35% less dose. As the name implies, this category of reconstruction algorithm iteratively solves the CT reconstruction task through a closed-loop system as shown in Fig. 1.

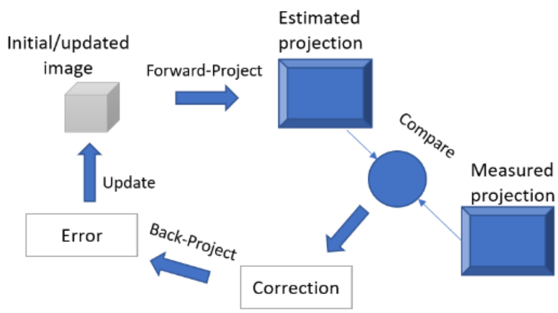

FIG. 35. Iterative reconstruction process

In the first iteration, the iterative algorithm starts with an initial image. Then, the initial image is forward projected to produce estimated projection, which is compared with the measured projection from data acquisition process. The discrepancy between the estimated projection and measured projection is back projected to compute an error that is used to update the estimated image for the next iteration. Then, the second iteration starts using an updated image from the first iteration as a starting point. The process is repeated iteratively until the image with the desired properties is achieved.

There are many algorithms which implement the iterative reconstruction scheme. One of these algorithms is called Adaptive-weighted Projection-Controlled Steepest Descent (AwPCSD) proposed by Lohvithee et al [4]. The AwPCSD is a total variation (TV) regularization algorithm, which minimises the TV norm of the image as expressed in equation $x^* = argmin\|x\|_{TV}$, subject to the following two constraints: 1) Data fidelity constraint, $|Ax - b| \leq \varepsilon$ and 2) Non-negativity constraint $x \geq 0$, where x is a vector containing the X-ray linear attenuation coefficient of the image, A is a system matrix describing the intersections between each particular ray and the image voxels, b represents the projection data measured on the image detectors at various projections, ε is an error bound that defines the amount of acceptable error between the predicted and observed projection data. In each iteration of the AwPCSD algorithm, there are two phases. The first phase is the implementation of simultaneous algebraic reconstruction technique (SART) to enforce data-consistency according to the two constraints mentioned above. The second phase is TV optimization, which is performed by the adaptive steepest descent method. These two steps are implemented alternatively and iteratively until the stopping criterion is satisfied. The AwPCSD algorithm is able to reconstruct a good quality of image from limited numbers of X-ray CT projection data.

Despite advantages of iterative algorithms, one main drawback of iterative algorithms, especially TV regularisation algorithm is the presence of a set of parameters in the implementation of two-stage approach. The roles of these parameters are for weighting and balancing effects between two approaches within one iteration, as well as effects between iterations, depending on the definitions of parameters. The values of parameters need to be selected carefully for different input X-ray CT measurements such that the iterative algorithm perform at its best and produces good reconstructed images. Normally, selecting optimal values for parameters are difficult and can only be determined by trials and errors. It is a time consuming and tedious process, which makes the iterative algorithms difficult to use in real practices, hence, limit the availability and popularity of iterative algorithms.

Recently, an efficient parameter selection in total variation-penalised X-ray CT image reconstruction is proposed in [5]. The work provided a systematic way of choosing parameters by combining Hedge method of Freud and Shapire with the AwPCSD algorithm reconstruction algorithm. The numerical studies showed that the method was able to select the best performance parameters from the user-defined values. Also, the results showed a potential of readily applying the best parameters from the training dataset to other different datasets, which can save lots of time to avoid re-calibrating the parameters.

Stemming from the success of the work in [13], the presentation proposes an extending work by applying the automated parameter selection for X-ray CT imaging with a number of real datasets from different patients in the Faculty of Medicine, Ramathibodi Hospital. The objective is to obtain statistical results on the optimal set of parameters from different patients in the similar imaging contexts (e.g. similar parts of the body, similar CT scanner), in order to readily apply the set of parameters for an efficient implementation of iterative algorithm (specifically for the AwPCSD algorithm). The result of the presentation will make the iterative algorithm more generalised to be implemented by a general user. Hence, the limited projection X-ray CT reconstruction can be efficiently implemented to reduce the amount of radiation exposure delivered to the patient during clinical examinations/treatments.

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279-RISK COMMUNICATION: CHALLENGES AND GOOD PRACTICES FROM THE MOROCCAN EXPERIENCE

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Information and communication are strategic instruments that support organizations to achieve their regulatory objectives. The law n°. 142-12 related to safety and security and the creation of the Moroccan Agency for Nuclear and Radiological Safety and Security "AMSSNuR" (the Moroccan Regulatory Body) assigns clear responsibility to AMSSNuR for information of and communication with interested parties, including the public, about the potential risks associated with facilities and activities involving ionizing radiation sources as well as the regulatory decision making processes.

In its strategic vision, AMSSNuR aims to establish itself as an independent, efficient, credible and transparent regulatory body. For this purpose, AMSSNuR developed the strategic plan 2017-2021, which covers all its regulatory functions including information and communication as an important element in helping to build public confidence. Thus, the development and implementation of a reliable and transparent communication policy which aims, inter alia, to inform interested parties about the risk aspects of installations and associated activities involving ionizing radiation sources is one of the strategic objectives of AMSSNuR.

AMSSNuR, in its communication activities, relays on the IAEA Safety Standards, especially: (a) the Fundamental Safety Principle 2 highlighting in para 3.10 the key role of the Regulatory Body in consulting and informing interested parties on safety-related issues; (b) Requirement 36 of GSR Part 1 (General Safety Requirements on Governmental, legal and regulatory framework for safety); (c) Requirement 5 of GSR Part 2 (General Safety Requirements on Leadership and management for safety); where the Regulatory Body's communication and consultation with interested parties are addressed; and (d) GSG-6 on Consultation with Interested Parties by the Regulatory Body

Communication about risks, needs to be a structured and implemented. The establishment and implementation of communication strategies and plans help to ensure effective radiation risk communication and to promote the establishment of appropriate means of informing and consulting interested parties and the public about the possible radiation risks associated with facilities and activities. For this reason and to meet all the national legal and regulatory provisions in accordance with the IAEA standards, AMSSNuR established a communication strategy that revolves around:

(a) Social Media; AMSSNuR has developed strong social media presence on most social media channels to disseminate information and to provide interested parties with timely, reliable, comprehensive, understandable and easily accessible information on safety, radiation risks and regulatory issues through Facebook, Instagram, LinkedIn, Twitter, YouTube.
(b) Public Relations; has been addressed by AMSSNuR using a communication plan based on a continuous interaction with relevant parties, including awareness promotion of all kind of mass media concerning, inter alia, risk communication in nuclear and radiological fields (TV, Radio, Newspaper, Electronic press).
(c) Public communication during a nuclear or radiological emergency; AMSSNuR has established procedures to disseminate information on safety to interested parties, such as information on incidents in facilities and activities, including accidents and abnormal occurrences, as well as radiation risks associated with facilities and activities. In this framework, AMSSNuR works for building trust and confidence in normal communication

to ensure more effective communication in case of a possible nuclear or radiological emergency and better credibility to respond to misinformation and misunderstanding on all media channels.

(d) Internal communication; the staff of AMSSNuR is kept informed about the decisions and activities of their organization, and other relevant information. They should be made aware that their communication might affect the public's risk perception.

(e) Stakeholder involvement; AMSSNuR has also set up an inclusive approach for engaging stakeholders by organizing several meetings and public hearing sessions. The approach adopted by AMSSNuR is based on dialogue/exchange of ideas and information concerning the regulatory process and the radiation risks culture

The purpose of this paper is to share the Moroccan experiences and good practices related to risk communication, and to discuss current challenges faced by the Moroccan regulatory body in communicating radiation risks and safety issues with the public and other interested parties, especially those related to transparency, reputation and trustworthiness, constraints and knowledge building, public trust, risk perception, dissemination of radiation safety and security information, message development, information clarity, and stakeholder engagement.

Key words

Radiation Risk Communication, Safety Communication, Public Information, Stakeholders Involvement

281-EVALUATION OF ESTABLISHMENT AND PRACTICAL IMPLEMENTATION OF GSR PART 3 REQUIREMENTS IN A GENERAL HOSPITAL

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Since the publication in Spanish of the GSR Part 3, this has been a conceptual reference in terms of radiation protection for the regulatory body and for user of ionizing radiation. The comprehensive evaluation of the practical implementation of the provisions of this Standard, in specific facilities, would make possible to check the feasibility of it compliance. The most representative facilities for this process are the general hospitals where the three exposure categories are present, for situations of planned exposure and eventually for emergency exposure situations.

The presentation sets out the result of the level of establishment and implementation GSR part 3 requirements analysis, in the General Hospital "V. I. Lenin", where teletherapy, high dose rate brachytherapy, superficial X-ray therapy, diagnostic nuclear medicine ("in vivo" and "in vitro"), metabolic therapy with ^{131}I, with and without hospitalization, interventional radiology in cardiologist surgery, tomography, mammography and conventional radiodiagnosis are carried out.

The analysis is performed from the perspective of facility's radiation protection service, regulatory authority and an entity that provides radiation protection support services, based on the provisions of safety reports of each practice, institutional licenses and regulatory inspections reports, realized by the national regulatory authorities for radiation protection and safety, in the triennium 2017-2019, following the order of the requirements that appear in table No.1 of GSR Part 3, referring to general requirements and the requirements of planned exposures, both generic and those related to occupational, public and patient exposures. It also presents the results of the analysis of the preparation for emergency exposures, related to occupational and public exposures, as established in the hospital radiological emergency plan.

For the analysis, each requirement has been classified, with respect to its formal establishment in the country, as regulated in 22 regulatory dispositions on radiation safety applicable to medical practices, as: established (E) or not established (NE), some of them are not applicable to any or all medical practice (NA).

The established requirements have been categorized according their practical implementation level in the hospital as: fully implemented (I), in the process of implementation (P) and by starting their implementation (SI). Then, for each practice, the number of requirements has been quantified, according to this classification, and finally the general appreciation of requirements implementation level in the hospital is expressed.

The analysis result reported that, for daily situations of planned exposure, the requirements for occupational exposure, public exposure and medical exposure are in their great majority established and implemented, as shown in Table 1.

TABLE 1. ESTABLISHMENT AND IMPLEMENTATION OF GSR PART 3 REQUIREMENTS FOR PLANNED EXPOSURE SITUATIONS

	Occupational Exposure (28) Section 2 (1-5); Section 3: paras. 3.5 to 3.67 (6-18) and 3.68 to 3,116 (19-28)					Public Exposure (23) Section 2 (1-5); Section 3: paras. 3.5 to 3.67 (6-18) and 3,117 to 3,144 (29-33)					Medical exposure (27) Section 2 (1-5); Section 3: paras. 3.5 to 3.67 (6-18) and 3,145 to 3,185 (34-42)				
Practice	**E**	**NE**	**I**	**P**	**SI**	**E**	**NE**	**I**	**P**	**SI**	**E**	**NE**	**I**	**P**	**SI**
Radiotherapy	26	2(NA)	24	2	-	21	2(NA)	20	1	-	25	2(NA)	23	2	-

Nuclear Medicine	26	2(NA)	24	2	-	21	2(NA)	20	1	-	26	1(NA)	24	2	-
Interventional Cardiology and CT	26	2(NA)	15	9	2	20	3(NA)	10	8	2	25	2(NA)	12	11	2

For exposure situations of emergencies, specifically for potentials, since in the period no emergency has arisen, the requirements establishment and implementation is also very favourable for preventing and mitigate possible exposure of the public and the workers who would participate in the consequences of the emergency evaluation and liquidation, as Table 2 shows.

TABLE 2. ESTABLISHMENT AND IMPLEMENTATION OF GSR PART 3 REQUIREMENTS FOR EMERGENCY EXPOSURE SITUATIONS

	Emergency Workers Exposure (7) Section 2 (1-5); Section 4 (43-45)					Public Exposure (7) Section 2 (1-5); Section 4 (43-44)				
Practice	**E**	**NE**	**I**	**P**	**SI**	**E**	**NE**	**I**	**P**	**SI**
Radiotherapy	7	-	6	1	-	7	-	6	1	-
Nuclear Medicine	7	-	6	1	-	7	-	6	1	-
Interventional Cardiology and CT	7	-	1	6		5	2(NA)	1	4	-

The evaluation shows that, although the entire Cuban radiation safety regulatory framework applicable to medical practices, entered into force before GSR 3 publication, it can be stated that most of them allow compliance with the provisions of said standards. For the case study, the level of implementation for radiotherapy and nuclear medicine is very satisfactory and for imaging still need to continue implementing some requirements, mainly those related to safety assessment.

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282-OPTIMIZATION OF THE IODINE 131ADMINISTRATION METHOD IN ABLATIVE THERAPIES

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INTRODUCTION

Ablative therapy with iodine (I-131) is used to remove the remaining thyroid tissue after its removal for presenting oncological pathology. Ablative therapies have increased in the last decade; Peru currently has 57 facilities dedicated to this practice, generating the need to investigate the doses received by the occupationally exposed worker (TOE) and the doses received in radiosensitive organs of the patient, such as the lens.

The administration of I-131 is carried out orally and is generally administered by the Medical Technologist through two methods whose choice depends on the patient's clinical condition: method No. 01 employs direct administration of I-131 and method No. 02 employs the administration by system of communicating vessels. Method N ° 01 has demonstrated advantages in costs, times, less difficulty in administration, reduction of radioactive residues, low levels of TOE dosimetry and dose received by the patient's lens.

The main objective of the study was to prospectively and randomly evaluate the efficacy of method No. 01 in the radiological protection of TOE and to estimate the dose received by the patient's lens. As a secondary objective of the study, we seek to initiate an investigation to establish reference dose levels (NDR) in this practice for both TOE and patients.

METHOD AND MATERIALS

Two groups of 49 patients were established at random, for group A, method No. 01 was used, and for group B, method No. 02. The activity administered to each participant was 100 mCi of I-131. For method No. 01, the parameters were had: 53 cm distance between operator-patient during administration, average time of 9 seconds for administration, and time to leave the room of 5 seconds, see Fig. 1.

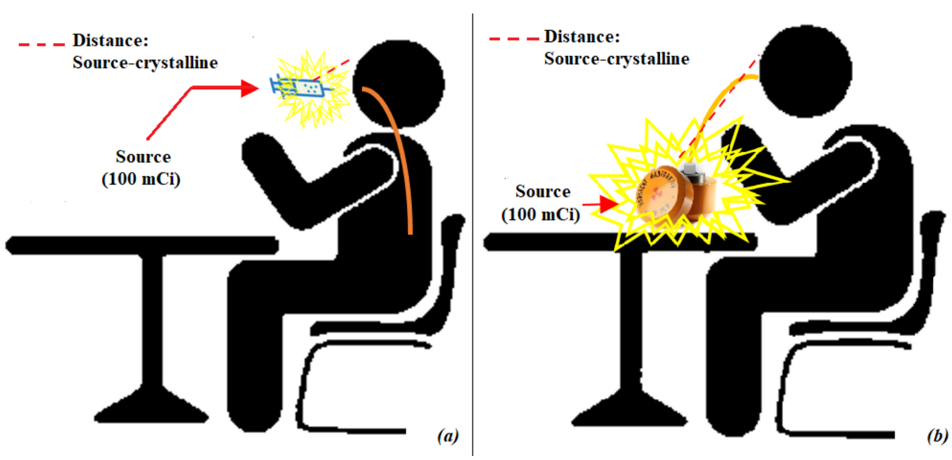

Fig 01. (a) Method N° 01 of direct administration, (b) Method N°02 using communicating vessel system

For method No. 02 (see Fig. 1) were considered: 243 cm distance between operator-patient during administration, time of 25 seconds used by TOE to install the communicating vessel system with bare source, average time of 157 seconds of administration, distance of 44 cm with a time of 37 seconds to remove the installed system, and time 5 seconds to leave the room, see Fig. 2.

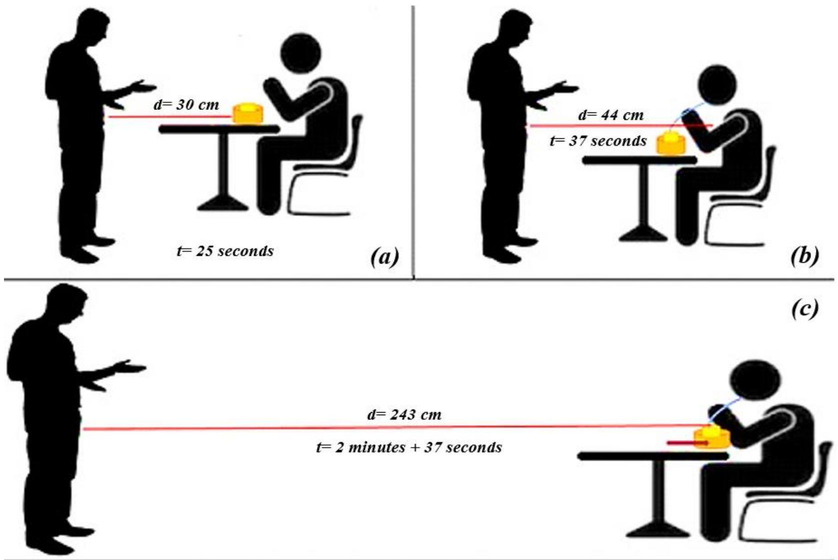

Fig. 2. (a) Positioning of the communicating vessel system before administration. (b) Collection of the administration system by means of communicating vessels installed. (c) Supervision of the I-131 administration procedure. Times were measured with a stopwatch for both methods.

In both methods, the parameters of time, distance and shielding were used. The calculation of the patient's lens dose was made using the gamma factor of 2.2 $R.cm^2.mCi^{-1}.h^{-1}$, considering a point source and compared with the analyzed data of the dose rate obtained by a radiation monitor. Ludlum Measurements, for which the conversion factor of 114.94 Sv/R was used to make the change units from Roentgen to Sievert.

RESULTS

For the TOE, personal dosimetry reports were considered, those who administered with method No. 02 obtained an average value of 0.60 mSv, while those who administered with method No. 01 obtained a value of "M", that is, minimum measurable. The dose estimate obtained for the patient's lens is presented in Table 1.

Table 1. Dose obtained for the patient's lens in mSv.

	Method N° 01	Method N° 02
Theoretical	658.5	1512.8
Experimental	775.9	1782.3

The two values of crystalline dose have a variation of 15.0%. Recommended The study of the two administration methods of I-131 using a TLD measurement system for the determination of doses in crystalline for patients, and, in wholebody and crystalline for TOE.

CONCLUSIONS

- In method N° 02 the TOE and patient are more exposed to the radioactive source.
- Method No. 01 optimizes dose, resources, times, and produces less radioactive waste than method No. 02.
- Occupational and medical reference levels are required for the optimization of procedures for I-131 therapies.

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283-AN APPROACH TO PUBLIC COMMUNICATION PROCESS FOR A DEVELOPING COUNTRY (SUDAN) FIRST NPP

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INTRODUCTION

Public communication is a method to improve democratic governance by helping governments consult their community on the key public policy issues the government faces [2]. A public consultation is a necessary tool used to improve the transparency, efficiency, and effectiveness of regulation in addition to other tools like Regulatory Impact Analysis (RIA), regulatory alternatives, and improved accountability arrangements [3].

Sudan is embarking upon electricity generation through nuclear energy and has decided to build a 1000 MWe PWR on a super turnkey basis, though this decision is yet to be declared publicly by the government [1].

The significance of having an effective Public Communication and Stakeholder Management (PCSHM) strategy cannot be overstated. Adverse effects of public protests against nuclear power, which often delay the progress of work, can be contained by a sound, clear, responsive, and authentic PCSHM strategy. Public Communication, as the title implies, is communicating with the public and keeping them informed. To ensure that the public understands and acknowledges what has been told to them; it is necessary to address the common public concerns and worries [1].

Status of Public Communication in Other Countries

The success of a nuclear power program in any country is influenced by a well-structured PCSHM strategy, moreover the political maturity and continuity in governmental policies. E.g., in Austria, a fully constructed NPP was not permitted to be commissioned owing to the public opinion which was 50.2% against and 49.8% pro-nuclear. Similarly, the recent retreat in the nuclear power program in Germany is due to the political decision-makers [1].

Vietnam Atomic Energy Agency (VAEA) has established an extensive public communication and stakeholder management plan for its first NPP, in which it has emphasized on the developing communication objectives for different groups of audience. Execution of this plan will be carried out by using tools such as exhibitions, facility tours, public hearings, mass media, online promotion, support & education programs, and public relation centers [1].

The US is communicating with the public using an issue-based approach, instead of focusing only on the technological issues, the social aspects related to nuclear power generation, such as health, air pollution, the environment, and the economy, are being discussed as well [6].

In Indonesia, the years from 2010 to 2016 witnessed the first public opinion survey on nuclear energy [4]. The survey results illustrated that the level of public support nationally is above 70% in the last 3 years, this high support for the NPP program, probably due to the urgent need for stable and reliable electricity. TV media is the most effective tool for socialization, while economically, radio can be selected as an advertising tool with widespread nationally. The ability of the Indonesian government to maintain public trust is unique, therefore the public expects an explanation from the Government on the benefits and risks of the nuclear power program [4].

METHODOLOGY

Coherent communication with the public will be carried out, because of the sensitivity and complexity in nuclear power production. One crucial guideline for communicating with the public before any engagement with them is recognizing the communication's objectives differences during various phases of the NPP project [5]. These phases are:

— Pre-Project Phase;
— Project Decision Making Phase;
— Plant Construction and Operation Phase;
— Plant Decommissioning.

There are main issues and concerns for the public that should be addressed in each phase of the project. These issues can be summarized as follows:

— The necessity of nuclear power for the nation; availability of sufficient national infrastructure for nuclear power to be included in the national energy plan;
— Location of the site and environmental effects;
— The decision to build an NPP is to be defended
— Independence and integrity of the information provider;
— Benefits due to generating electricity by nuclear power;
— Public opinion during the commissioning of the power plant;
— Concerns of the public regarding the operation of the NPP; Radiation hazards due to effluents and spent fuel;
— Safety of the transportation and storage of the radioactive waste/spent fuel through public areas;
— Emergency plan and emergency communication;
— Public Understanding of the decommissioning phase of the project;
— Following environmental safety and security management standards;
— Continued financial support for the decommissioning of the power plant.

The number and sort of tools to be used depending on the message, audience, timing, resources, and legal and regulatory requirements. Such tools include:

— The announcement in press, TV;
— Facility tours;
— Seminars / Workshops;
— Website;
— Exhibitions;
— En masse distribution of fact sheets, mails to students, public, etc.

An indicative timeline for the implementation of the public communication plan is given below:

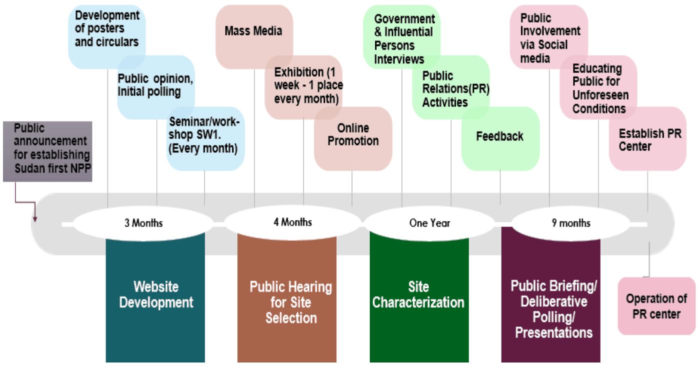

FIG.36. Timeline for Implementation Plan

FINDINGS

— Communication at all times should be transparent and in a language that is effortlessly understandable by the target audience;
— Public memory is usually short-lived and hence the public communication program has to be reiterated frequently;
— The messages should be continuous and consistent, and part of a regular program of information release and distribution through all existing delivery systems;
— There is a considerable number of the public that doesn't have access to the internet therefore, it must be sure that all online messages reach them via offline media.

CONCLUSION

Evidently, the public acceptance of nuclear power program is a prerequisite for the long-term success of nuclear power program in any country. The pessimism about nuclear power can be overcome by proactively contacting the public, addressing their likely concerns, and removing the misgivings.

All suitable tools (print, electronics, audio-visual documentaries, models, science fairs, exhibitions, seminars and workshops, career fairs, etc.) should be used for various target audiences like students, homemakers, and teachers, experts, professionals from the industry and decision-makers.

The proactive public communication plan recommended in this document can help the public to cooperate with the government for a sustainable nuclear power program in Sudan.

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284- IMPROVING NATIONAL RADIATION PROTECTION REQUIREMENTS IN PLANNED EXPOSURE SITUATIONS BY IMPLEMENTING IAEA GENERAL SAFETY REQUIREMENTS (GSR PART 3)

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Since the promulgation of the law 142-12 related to nuclear and radiological safety and security and the creation of the Moroccan Agency for Nuclear and Radiological Safety and Security (AMSSNuR) in 2016, the Kingdom of Morocco through AMSSNuR, its national Regulatory Body, has been upgrading its regulatory framework to ensure compliance with the Law 142-12 and consistency with the international safety standards, in particular the IAEA general safety requirements (GSR part 3).

For planned exposure situations, the draft regulations clarify the practical procedures for applying radiation protection within activities involving ionizing radiation sources in order to ensure the implementation of the fundamental principles of radiation protection (justification, optimization, and limitation). The draft regulations define several provisions regarding the licensee's responsibilities, procedural and technical arrangements for the designation of controlled and supervised areas, local rules, requirements for individual and area monitoring and personnel qualification and training.

On another side, regarding the workers exposure dose limits, the Kingdom of Morocco introduces the International Commission on Radiological Protection (ICRP) recommendation for lens equivalent dose limit reduction. This development will considerably strengthen the implementation of the optimization principle, particularly in the medical environment in facilities where interventional procedures are carried out.

As support measures for the implementation of these provisions, the licensee can hire a recognized radiation protection service provider that must work in close collaboration with the occupational physician, the radiation protection officer and the medical physicist in the case of a medical facility.

AMSSNuR ensures its regulatory functions to control that licensee implement a radiation protection programme that meets the regulatory requirements related to the protection of workers, the public and the environment and ensures that doses to persons are optimized, taking into account social and economic factors. The management of review and assessment systems, granting of authorizations, inspection and monitoring of regulated activities involving ionizing radiation sources, allow to have a high level of protection against the risks associated with activities involving the use of Ionizing Radiation Sources.

The aim of this paper is to present AMSSNuR's approach for the transposition of the GSR part 3 requirements related to planned exposure, in the national regulation for the application of the provisions of the law 142-12, and the practical aspects for their implementation.

Key Words: AMSSNuR, Radiation Safety, Regulatory Control, Planned Exposure Situation, Lens equivalent Dose Limit

285-RADIATION SAFETY IN MEDICAL PRACTICES: THE MOROCCAN EXPERIENCE AND CHALLENGES

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In the Kingdom of Morocco, the medical sector is close to 80% of all the practices and activities involving ionizing radiation sources. This sector comprises various and large number of installations, including more than 5700 diagnostic radiology machines with a number of CT-scans exceeding 324, more than 5100 dental X-ray machines including Cone Beam technologies, 26 nuclear medicine facilities with 43 radioactive iodine (I-131) therapy isolation rooms, 14 PET-CT and 12 SPECT- CT machines, as well as 38 radiotherapy facilities with 58 LINACs, 2 Gamma knifes and 23 high dose rate (HDR) brachytherapy Systems.

The Moroccan law N°142-12, promulgated in 2014, has made significant changes in nuclear and radiological safety and security compatible with relevant international instruments, the international safety standards and security guidance. It has also reserved a large part of its legal provisions for the medical uses of ionizing radiation sources. Given the specific aspects of this sector, the law sets, in addition to the general provisions, several requirements regarding the qualification of medical and health professionals, maintenance and quality control measures and further radiation protection requirements.

With a view to further strengthening its regulatory infrastructure in the medical field, in accordance with GSR part 3 requirements, the draft decree, implementing the law n° 142-12, on the use of ionizing radiation sources for medical purposes, clarifies the practical implementation of the general requirements related to justification and optimization, provides for the development of evaluation and control procedures for the implementation of those principles, requires the use of Diagnostic Reference Levels (DRLs) as a tool to optimize the diagnostic protocols, defines the list of the authorized health professionals for the use of radiation sources, determines compliance standards for medical radiation equipment and strengthens the requirements for quality assurance and quality control procedures as well as incidental and accidental exposures.

The draft decree takes into account the need for consistency with all laws and regulations relating to the environment, radiation protection and in particular the national health system, including those specific to the practice of medicine and medical devices.

The aim of this paper is to present AMSSNuR's approach regarding the implementation of the new provisions of "GSR part 3" and "SSG -46", to improve the practice of radiation protection and safety in medical uses of ionizing radiation sources.

Keywords: AMSSNuR, Radiation Safety, Radiation Protection, Medical Exposure, Patient Safety.

287-CONDITIONING OF DISUSED SEALED RADIOACTIVE SOURCES AT THE PHILIPPINE NUCLEAR RESEARCH INSTITUE RADIOACTIVE WASTE MANAGEMENT FACILITY

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The majority of the radioactive waste generated in the Philippines comes from the medical and industrial fields because there is no operational nuclear power plant in the country. Most of these are in the form of sealed radioactive sources (SRS). The SRS consist of radioactive materials (RAM) that are encapsulated in small, metallic vessels with radioactivity ranging from kilo-Becquerels to Terra-Becquerels. If the source is no longer usable for its intended purpose or the equipment in which the RAM is embedded is obsolete, damaged, or leaking, it is considered spent and/or disused. However, the SRS may still be radioactive and may still pose a health hazard and environmental contamination if it is not managed well [1].

The Philippine Nuclear Research Institute (PNRI) operates the Radioactive Waste Management Facility (RWMF) to safely manage and secure these disused SRS (DSRS). The RWMF is authorized to treat, condition, and store radioactive waste until a final disposal facility is operational. The Philippines has not yet developed a formal national policy and strategy for radioactive waste management; however, existing regulations are being implemented to control the use of RAM in the Philippines. This regulation is known as the Code of PNRI Regulations (CPR) and is being enforced by the Nuclear Regulatory Division (NRD) of PNRI as the regulatory authority in this field. The CPR Part 3 entitled "Standards for Protection Against Radiation" which states that the optimization of the protection and safety measures associated with radioactive sources shall be subjected to dose constraints for the radiation workers and the public to avoid exceeding the dose limits set by the regulatory authority [2]. At present the RWMF continuous to provide better services and accommodate the increasing number of DSRS in the Philippines following this regulation. The paper aims to provide valuable radiation safety practices in conditioning DSRS performed inside the RWMF and how it is being implemented following existing regulations.

Managing of DSRS consists of simple steps from receiving to the actual conditioning of DSRS. Figure 1 shows the steps in managing DSRS in the PNRI-RWMF. The first step is receiving the DSRS from the waste generators. In this stage, the waste generator must comply with the CPR Part 27 which describes the safety and security requirements when transporting DSRS and the waste acceptance guidelines (WAG) as set by the PNRI-RWMF. The dose rate of the waste package must be 2 mSv/h and this will be verified upon arrival to the PNRI-RWMF. Aside from this, the surface contamination of the DSRS is monitored as well. The PNRI-RWMF will not accept the package if the WAG has not been fulfilled. On the other hand, if the WAG is met, the DSRS is accepted and transferred inside the PNRI-RWMF for further processing.

The next process is planning the conditioning activity for DSRS. The information provided by the waste generator is verified based on the actual source information on the DSRS. Distance, time, and shielding are being considered during planning. Once done, the conditioning process is performed. This process includes multiple steps namely dismantling, retrieval, encapsulation, and storage. Dismantling is the process of reducing the volume of the waste by removing parts of the DSRS which are considered to be scrap materials and may be disposed of as regular solid wastes. Next is retrieving the RAM inside the dismantled DSRS, this is performed by removing the RAM inside the DSRS. Encapsulation is the next procedure in which the RAM is transferred in a specialized stainless-steel capsule. These capsules are fabricated according to the standard designs provided by the IAEA. Storage of the capsule to a standard 200 L drum with a conrete and lead lining is the final step in conditioning DSRS.

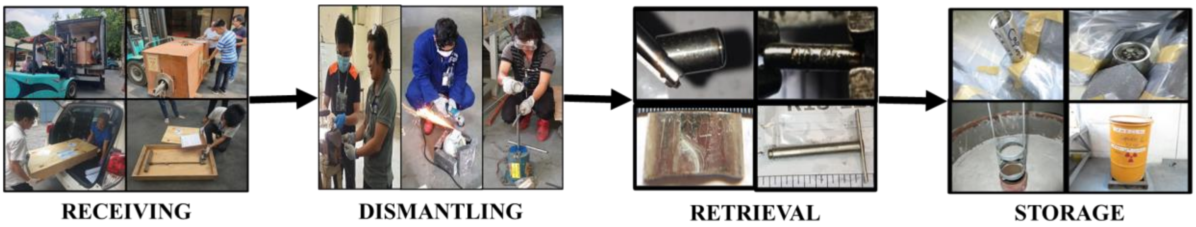

FIG. 37. DSRS management strategy of PNRI-RWMF.

Before and after performing these processes, the PNRI-RWMF has different provisions not only to comply with the requirements of CPR Part 3 but also to protect the safety of its personnel. Before entering the facility, the operational area monitoring is performed to monitor the background radiation levels that are currently present inside the facility. This is also perfomed after the operation to monitor any significant increase in the radiation levels in the facility. According to CPR Part 3, the occupational whole-body dose limits for radiation workers must be an average of 20 mSv for 5 consecutive years or 50 mSv in a single year. On the other hand, the extremity (hands, feet, or skin) dose limit must be 500 mSv in a single year. With these limits in place, personnel of the PNRI-RWMF is equipped with personal dosimeters to monitor the radiation absorbed by the personnel during the operation inside the facility. The annual dose registry for the personnel of PNRI-RWMF is summarized in Table 1 for the year 2019.

TABLE 1. ANNUAL DOSE REGISTRY OF PNRI-RWMF PERSONNEL, YEAR 2019

Personnel ID	Whole-body Dose (Hp 10), mSv	Extremity Dose (Hp 0.07), mSv
RPSS-A	<0.10	2.21
RPSS-B	0.13	3.36
RPSS-C	0.15	1.08
RPSS-D	<0.10	0.87
RPSS-E	<0.10	<0.10
RPSS-F	0.12	4.02
RPSS-G	<0.10	3.05

The annual dose received by the PNRI-RWMF personnel is below the occupational dose limit as prescribed by the regulatory authority. This means that the existing guidelines, operational procedures, and radiation protection program is effective in protecting not only the public but also its workers. At the end of each operation, the radiation control and monitoring officer checks for contamination of the personnel exiting the facility to avoid cross-contamination between the buildings of the facility.

In summary, management strategies of PNRI-RWMF abide by safety standards and procedures in managing DSRS to maintain the highest safety and security of the people and the environment.

ACKNOWLEDGEMENTS

The PNRI-RWMF would like to send its gratitude to the IAEA, through international cooperation projects, in helping the operators develop operational and managerial skills concerning DSRS.

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288-COMPETENCE IN RADIATION SAFETY ISSUES AFFECTED BY THE NEW LEGISLATION

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One of the suggestions of IRRS-mission (2016) for Belarus was to consider establishing appropriate requirements for the qualification, and make sufficient arrangements for the training and the recognition of radiation safety specialists (i.e., radiation protection officers, qualified experts) in order to ensure a reliable availability of such specialists.

The new Law of Belarus "On Radiation Safety" of 18 June 2019 (entering into force on the 27 June 2020), as well as the appropriate by-laws, have been developed in order to enhance the legislation and regulations in the field of radiation safety, including the issues of competence in radiation safety.

Among other things the concept of radiation safety consultation was first introduced. The term of "radiation safety consultant" is equal to "qualified expert" in accordance with the IAEA terminology, or "radiation protection expert" in terminology of EU Directive.

The following provisions were established in the Law "On Radiation Safety" in this regard:

— The possibility of provision the service of consultation in the sphere of radiation safety by individual entrepreneurs or employees of facilities.
— The consumers of consultation in the sphere of radiation safety: manufacturers and users of ionizing radiation sources, organizations carrying out design, construction, installation, commissioning, diagnosis, repair, maintenance, decommissioning of ionizing radiation sources and associated equipment, scrap metal companies.
— Open list of questions, which radiation safety consultant can cover (justification of practice; optimisation and establishment of appropriate measures; commissioning and decommissioning of ionizing radiation sources; employment conditions for exposed workers; radiation monitoring organization; training and knowledge testing programmes for exposed workers; arrangements for radioactive waste management; arrangements for prevention of accidents and incidents; preparedness and response in emergency exposure situations; preparation of appropriate documentation for carrying out practices in the sphere of ionizing radiation use).
— Requirements to the radiation safety consultants: possession of special knowledge and skills in the sphere of radiation safety (in relation to a specific issue as mentioned before); higher education, which allows to provide consultation; at least three years of experience as a technical manager or specialist in radiation safety issues, for which consultation is expected; at least once every five years training and knowledge testing on radiation safety issues.
— The possibility to confirm the quality of consultation using the procedure of recognition (certification) by the Regulatory Body.
— Maintenance (and placement on the Regulatory body's web-site) of the unified register of certified consultants in the sphere of radiation safety.

The existing system of training and knowledge testing on radiation safety issues is also under revision now in order to more clearly define the categories of specialists, who are required to undergo training and knowledge testing on radiation safety issues, and the procedure itself.

Also the Strategy for enhancing competency in radiation safety is provided to be defined by the Government of Belarus on the proposal of the Ministry of Emergencies, agreed with the Ministry of Education, the Ministry of Health, the Ministry of Natural Resources and Environmental Protection, the Ministry of Defense, the State Border Committee, in order to form and ensure the functioning of a state system of training on radiation safety, as well as in the field of radiation technology.

In the context of working on the Strategy for enhancing competency in radiation safety in Belarus methodological materials, kindly provided by the IAEA, will be used. Taking them into account the appropriate stages will be included into the Strategy for enhancing competency in radiation safety. Undoubtedly this work will go better if a high level steering committee or commission composing the representatives of all the interested parties (ministries, departments, and organisations) is organized.

289-A DESIGN OF DECISION SUPPORT SYSTEM FOR JUSTIFYING THE USE OF IMAGING TECHNOLOGY IN DETECTING COVID-19 BASED ON INDONESIAN REGULATIONS

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INTRODUCTION

On Wednesday morning April 15, 2020, the Indonesian government spokesman for the corona case explained that the update of the number of Covid-19 corona virus patients positive cases increased by 297 in a total of 5,136 [1]. The data shows that virus transmission within the community is still occurring. At present, the fear of the community is mounting in the development of the corona virus in the community. This incident was later captured by the hospitals and clinics in Indonesia to offer some corona virus detection options to the community. They offered Covid-19 health check-up screening packages completed by chest radiographic or CT scan examination. There are many differences of opinion on this case among radiologists, hospitals, clinics, and associations. Most people argued that this type of diagnosis is permitted, but others rejected and argued that it is malpractice. This problem does not only occur in the case of Covid-19, but it is repeated every time a new disease arises. Advanced imaging technology has opened new horizons especially for medical diagnoses of patients. In Indonesia, justification has become a public policy following the presence of many new technologies and the implementation of technology to solve new problems. Decision making for justification requires a comprehensive consideration. The paper aims to discuss the construction of the decision support system as a tool for justification based-on regulations in the case of the implementation of imaging technology for the detection of Covid-19 in Indonesia.

METHODOLOGY

Stivaros et al [2] reviewed the decision support systems for clinical radiological practice. Stojkovsca et al [3] proposed a model for medical knowledge presentation and reasoning, which is used in a clinical decision support system for managing asthma in school-aged children. The paper proposed the design of a decision support system with forward chaining approach for justification in the case of imaging technology utilization in detecting Covid-19 based-on nuclear regulations in Indonesia. The justification in the paper is constructed on three levels. Generally, the proposed decision support system software can be utilized for all kinds of justification problems in the medical area.

IMAGING TECHNOLOGY FOR COVID-19 DETECTION

Many kinds of researches in China showed a correlation between people with Covid-19 and the results of examinations by imaging technology. However, on March 22, 20202, ACR (American College of Radiography) explained that the Centers for Disease Control (CDC) in the USA does not currently recommend CXR or CT to diagnose Covid-19 [4]. An area of justification discusses the expediency of radiation when a clinical evaluation or other imaging modalities reportedly can provide an accurate diagnosis. When the hospital or clinic will provide a diagnosis, standard referral guidelines should be utilized for medical imaging which based on the best available evidence to assist the decision-making process when choosing the best imaging procedure for a given patient. Consultation and communication between the referring physician, the radiology team, and the patient or patient's family are also needed.

JUSTIFICATION BASED-ON NUCLEAR REGULATION IN INDONESIA

Justification has become public policy in Indonesia. The regulation on justification is under preparation. It contains three levels of justification i.e. justification for diagnoses in general medical, the justification for specific purposes, and the justification for each patient before the examination. Indonesia regulates justification for radiation facilities and radioactive substances; or nuclear installations and nuclear materials. Justification evaluates documents contain at least a description and purpose of the types of nuclear technology utilization; complete characterization of nuclear technology to be used and actions to be taken to ensure radiation safety, the security of radioactive sources, and/or safeguard; assessment of the benefits and disadvantages of the types of nuclear technology utilization; and development plan for nuclear technology utilization. The evaluation considering the benefit and disadvantages, as described previously, covers aspects of safety, health, security, technology, social, and economy. The initial evaluation is based on the level of risk to safety and security that can be caused; complexities of the operation of nuclear technology facilities and/or equipment; benefits gained compared to the risks resulting from the use of nuclear technology; and history of accidents involving nuclear technology utilization [5].

DESIGN OF DECISION SUPPORT SYSTEM (DSS)

The medical DSS attempted to simulate the judgment of a human domain expert. A rule-based DSS has at least a knowledge database and inference engine. The knowledge database contains knowledge in referencebooks and doctor's experience, where the inference engine contains rules that connect between cause and effect. Then, the medical DSS must assimilate the knowledge and decision-making framework of the domain expert. The proposed DSS will provide recommendations concerning diagnosis and further imaging based on the regulatory provision in which the recommendation reflects a conclusion of forward chaining searching based on doctor's experiences and regulation in Indonesia. The software flow chart is shown in Fig. 1. The figure reflects a general justification of level 1 up to level 3 for new technology that will be implemented in Indonesia. Processes in the justification level 1 up to level 3 follow the provision in Indonesia as described in the previous chapter. Forward chaining is utilized to perform conclusions as output from correct evidence as input.

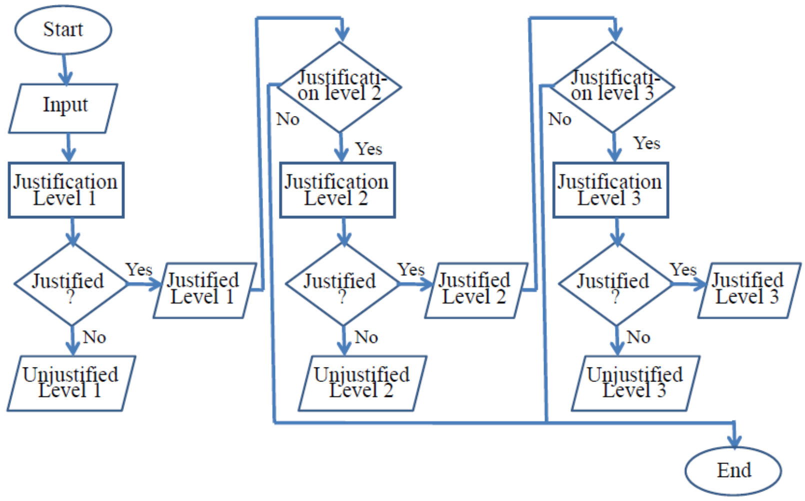

Fig.1. DSS Flowchart

RESULT AND ANALYSIS

In the paper, the design of DSS software is implemented to justify the utilization of imaging technology for detecting Covid-19 as a case study. The software experimental result showed that imaging technology can be justified to detect Covid-19 with the condition that it is treated as a complement to other diagnoses. It is also required that the diagnosis using imaging technology should be based on hospital/clinic referral guidelines, and all treatment should be communicated to the patient and/or patient's family.

CONCLUSION

The design of DSS to justify new technology or implementation has been constructed. The justifications include level 1 up to level 3. The DSS is implemented to the issue of utilization based on nuclear regulations in Indonesia. Generally, the DSS software can be utilized for all kinds of justification in the medical area. The result analysis showed that imaging technology can be justified to detect Covid-19 with the condition that it is treated as a complement to other diagnosis. The diagnosis using imaging technology should be based on hospital/clinic referral guidelines, and all of all treatment must be communicated to the patient and/or patient's family.

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291-REFLECTIONS ON LOW-DOSE RADIATION, THE MISCONCEPTIONS, REALITY AND MOVING FORWARD

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BACKGROUND

Radiation is ever present, and has significant variations in our daily lives with most people not considering it in their everyday activities. Recent documentaries on the April 1986 accident at Chernobyl and the one at Fukushima in March 2011 have focused the public's attention on the topic of radiation once again. However, the health risks associated with low doses of radiation are perceived by the public to be substantially greater than they actually are.

The nuclear industry has a remarkable safety record. The industry operates under strict radiation protection standards that require, for example, the use of advanced technological systems to protect individuals and the environment from the harmful effects of ionizing radiation. Where there are operating nuclear power plants, local communities do not experience any detrimental health effects. In 2012, the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) confirmed that an increase of risk to public health *"cannot be attributed reliably to chronic exposure to radiation at levels that are typical of the global average background levels of radiation"* [1].

As shown in *Figure 1*, the nuclear industry has successfully improved practices and reduced dose exposures for workers to approximately 1 mSv per year. The doses to workers associated with the nuclear fuel cycle are well below levels that are received from natural background radiation, which is 2.4mSv/y as a global average.

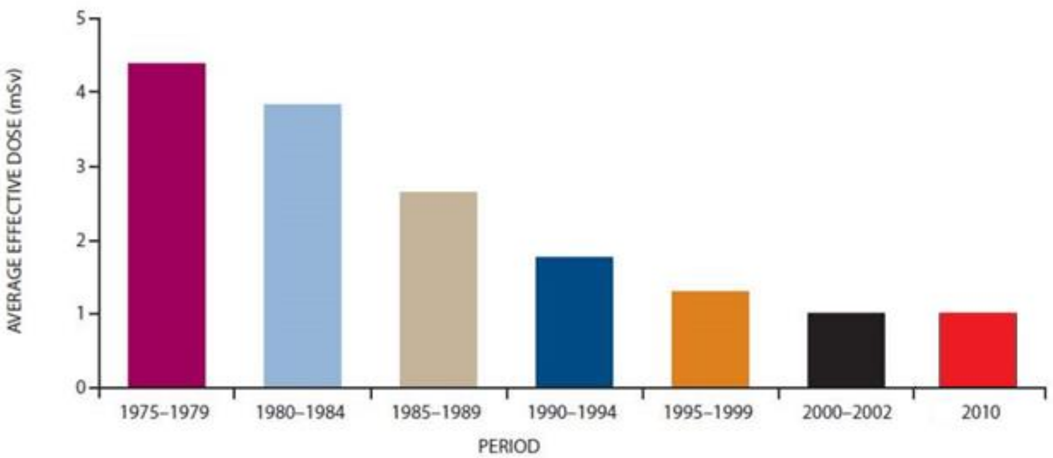

FIG. 1. Chart showing annual average dose over time in the nuclear industry [2]

THE UNDERLYING ISSUES

The World Nuclear Association advocates that radiation protection standards should be based on the best current scientific understanding of the effects of radiation, as well as proportionate public policy generated through political debate, driven by informed public opinion.

However, the ability of the nuclear industry to provide clean, affordable and low-carbon electricity is being compromised as a result of increasingly onerous regulation, coupled with misconceptions about radiation.

Several nuclear radiation applications (e.g. medicine, industry, agriculture) have resulted in a rise in standards of living and the use of nuclear energy provides clean, affordable and low-carbon electricity for many millions of people around the world. The very low risks associated with nuclear technology are massively outweighed by its benefits and additional constraints on radiation exposure or emissions limit the extent to which society can take advantage of these benefits.

The World Nuclear Association's Radiological Protection Working Group notes that the principle of "optimisation" is often misused with social and economic factors not being properly taken into account in decision making. This leads to an expectation of lower and lower doses for no net benefit. However, when used properly and as intended, optimisation is a powerful and practical tool for risk management and continuous improvement. While the original intent of optimisation was to consider both socio-economic and public health aspects, its current implementation has shifted from its original aims.

This shift in use is also seen with the concept of collective dose. In health-risk assessment, collective dose is a useful aid for optimisation of decision making, however, applying very low possible doses to large populations has many uncertainties and should not be used as a basis for public policy.

Additionally, the linear no-threshold (LNT) model, which is extrapolated from high dose and high dose rate exposures assumes that radiation risk is proportional to dose, so that even doses typical of a person's average exposure (i.e. a few mSv per year) is assumed to carry a small but defined risk of cancer. Such a model, if not correctly communicated, can lead to adverse public health effects due to fear of being exposed to low doses.

Due to the renewed interest among practitioners, industry, interest groups, experts and policy-makers on low-dose level exposure to radiation, the presentation will discuss current knowledge and realities of low-dose radiation as informed by current scientific understanding. Additionally, the presentation will explore possible forward trends and future application of radiation protection principles involving low-dose radiation.

CONCLUSIONS

Current science shows any risk associated with very low doses of radiation (as experienced in the nuclear industry) is extremely low, if it exists at all. However, misconceptions about radiation directly affect the ability of the nuclear industry to provide clean, affordable and low-carbon electricity.

The burden from excessive regulation imposes unnecessary costs to the global nuclear industry – while at most achieving minor decreases in exposure levels without any benefits.

Effective communication and stakeholder engagement about the low health risk from exposure to low levels of radiation may prevent adverse health effects associated with the fear of radiation

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293-THE DOSE DISTRIBUTION OF RADIATION FIELD IN THREE COMMON INTERVENTIONAL PROCEDURES

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OBJECTIVE

To test the dose distribution of radiation field in three common interventional procedures ,aiming to provide basic data for radiation protection and safe operation of the interventional radiology staff.

METHODS

Place thermoluminescent dosemeters(TLDs) in different points according to the horizontal aspect around the interventional table and the vertical plane of interventional radiology staff often stay. Based on the selected experimental conditions, the TLDs were grouped to irradiate.After the experiment, the TLDs were measured in the laboratory and calculated the dose of radiation field. Results It is when the liver and the cerebrovascular interventional procedures that the radiation doses of it are relatively low,besides the radiation doses can be ignored over three meters.To the cardiovascular interventional procedure, the greater the distance from the X-ray tube, the less the dose. According to the date of cardiovascular interventional high-dose group , we can see the radiation field dose distribution not only with distance, but also with angle, the doses near the angle of 30 and 180 and 330 are higher and the angle of 60 and 90 and 270 are low.

CONCLUSION

This article is the first investigation that studies the dose distribution of radiation field through Anthropomorphic phantom and TLDs during digital subtraction angiography (DSA) for interventional procedures of cerebrovascular, cardiovascular and liver at home and abroad. Referring to the dose distribution of radiation field, interventional radiology staff should be far from the field of high dose irradiation and set lead shields, reducing exposure to interventional radiology staff.

【Key Words】 Interventional procedure ; Radiation field ; Dose distribution

295-DOSE CONSTRAINTS CONFIRMATION IN INTRODUCED MODIFICATIONS AT ALREADY AUTHORIZED MEDICAL PRACTICES.

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In Cuba, since 2001, quantitative dose constraints values, ratified in 2011, were established in the Radiotherapy [1] and Nuclear Medicine [2] Safety Guidelines and their concept appears in the Basic Standards in force in the country, issued in 2011 [3], as recommended in the International Basic Radiation Safety Standards 115.

One of main applications of dose constraints, which the regulatory authority controls from then on, is their compliance both in the calculations of radiological shields in medical practice [4,5,6], as in the checks, after completing the modifications, in the premises where ionizing radiation sources are used [7]. So far in our country, the decision to define the proper value of the dose constraints for occupational exposure by the user entity of these sources, as referred to in the GSR Part 3, has not been included in the legislation [8].

The presentation sets out how regulatory authority controls dose constraints compliance in evaluating the optimization process, during construction of new facilities or modification of existing ones for occupational and public exposure, in planned exposure situations, in Radiotherapy (for Occupationally Exposed Workers (OEW)-10 mSv/year and Public- 0.5 mSv/year) and in Nuclear Medicine, (OEW- 6 mSv/year and Public- 0.2 mSv/year).

Representative cases of this act by the regulatory authority, show the evaluations of optimization process in the modifications made in last five years, in two cancer hospitals, based on the increase theirs workload due to new technologies installation at these services.

The information for this analysis, has been obtained from technical report [9] issued by regulatory authority, which supports the evaluation of the information submitted by Construction Licenses applicants required in the country [10], a stage that includes authorization of buildings remodelling, equipment installation and commissioning, at the proposal of licensees. Table 1 shows the result of the evaluation by regulatory authority of the proposal presented for each case.

TABLE 1. COMPLIANCE WITH THE DOSE CONSTRAINTS IN THE APPLICATIONS SUBMITTED.

No	Hospital/ Practice	Dose constraints Compliance		Measures	Correction	Final result
		OEW	Public			
1	**I**/ Brachytherapy	yes	yes	no	Not necessary	Licensed
2	**I**/ Surface Therapy	no	no	-Define location of the team command post. -Specify shielding calculation.	Corrected	Licensed
3	**I**/Teletherapy-Accelerator	no	no	-Check treatment room maze design and specify neutron calculation.		Not authorized
4	**II**/ Teletherapy-1	no	yes	- Make the reinforcement of the primary wall once the radioactive source has been installed and its need has been verified.	Corrected	Licensed
5	**II**/ Surface Therapy and Brachytherapy	yes	yes	- Surface Therapy workload- 30 patients in 6 hours a day a week. - Workload in Brachytherapy -10 patients in 8 hours a day a week.	Not necessary	Licensed
6	**I**/ Nuclear Medicine	no	N/A	- Specify the calculation of shielding for the SPECT operator's workplace.	Corrected	Licensed
7	**II**/ Nuclear Medicine	no	no	-Calculate radiochemical hood shields at the Radiopharmacy Technician workplace.	Corrected	Licensed

8	**II**/ Nuclear Medicine	no	no	-Specify shield calculations for SPECT-CT premises	Corrected	Licensed

In a complementary way, the dose constraints compliance verification, by regulatory authority, of in the inspections carried out once the modifications are completed [11] is exposed, as expressed in Table 2 and by the reports of Personal Dose Equivalent $H_p(10)$ in each year [12].

TABLE 2. COMPARISON OF MEASURED DOSE RATE WITH DOSE CONSTRAINTS.

Rooms	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**8**
Value of the measured dose rate, with respect to the dose constraints (in fraction) DR/DC	0,12 ±10%	0,11 ±10%	No built	0,25 ±10%	0,20 and 0,40 ±10%	0,1 ±10%	0,01 ±10%	No equipment installed

As results of individual dosimetric surveillance of the OEWs linked to aforementioned modifications show, it has been possible to observe established in Radiotherapy and Nuclear Medicine Safety Guides dose constraints, since the dose values in one year have not exceeded these values in all evaluated period.

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299-INSPECTION OF INDUSTRIAL RADIOGRAPHY AND FIVE MAJOR OPERATOR'S MISTAKES PRACTICAL ADVICES AND LESSONS LEARNED

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Industrial radiography using radiation sources is associated with substantial risk to workers and members of the public. According to UNSCEAR, 30-40% of all reported accidents associated with clinical consequences are a result of an event in this radiation practice. Inspecting such radiation practice might be a challenge in particular for a new regulatory body or its new staff members. Any additional limitations, e.g. limitations related to COVID-19, put additional stress on inspectors who should perform efficient and effective inspections. Practical advices on preparation and carrying out inspections are described. The analysis of industrial radiography accidents shows that five major operator's mistakes initiated or contributed to an accident, e.g. operator's staff do not use a survey meter to identify the location of a radioactive source after an exposure. Some of these accidents led to fatal consequences. Advices on how to systematically identify these mistakes during an inspection well before an accident happens, are described. By identifying the so-called *accident precursors* or *near misses* the inspector's activity effectively contributes to prevention of accidents. The discussion given might encourage regulatory bodies to develop practical inspection protocols or manuals focusing on the identification of such *near misses* and to efficiently assess compliance with safety requirements including safety culture of an industrial radiography operator during an inspection. Such approach might be also used for inspections of other radiation practices.

INTRODUCTION

Inspection as one of the core functions of any regulatory body (RB) is very often perceived by general public as the most visible activity of the RB. Developing inspector's competences might be a challenge as a "school" for inspectors does not exist. Practical inspector's protocols shall be developed. Sharing inspection's experiences shall be encouraged, in particular those related to practices involving radioactive sources of Category 1 and 2 [1]. Among such practices is industrial radiography. According to UNSCEAR, 30-40% of all reported accidents associated with clinical consequences are related to this practice. Typical sources used in this practice include radioisotopes, e.g. Se-75 and Ir-192, and X-ray generators. The practice, its safety guides and accidents are described elsewhere [2, 3]. The practice is carried out all over the world. So nearly all RBs are confronted with a question how to efficiently inspect industrial radiography and its long list of safety issues [4].

INSPECTIONS AND FIVE MAJOR OPERATOR'S MISTAKES IN INDUSTRIAL RADIOGRAPHY

As a rule, inspections of industrial radiography are conducted regularly, i.e. inspections are listed in *Annual Inspection Programs*. In Slovenia, each industrial radiography operator is inspected at least once per year. The importance of detailed preparation of each inspection cannot be overestimated [4]. When at operator's site the inspector should focus mainly on those safety issues which cannot be checked otherwise. To address issues in line with a graded approach, a site-specific check list should be prepared by the inspector. The preparation is even more important when restrictions such as COVID-19 measures or construction site restrictions at night are in place. In preparation phase the inspector should not study only the authorization file. Valuable data might be found on the web, taking of course into account appropriate caution. Such data might include information on compatibility of equipment, obsolete exposure devices and information on transport of sources from a country of origin. Useful videos related to practical issues might be available, such as a use of "Go-No-Go test".

At a site, five major mistakes which have already led to accidents described in [5] shall be in a focus of inspections. How can the inspector systematically identify such so-called *accident precursors* or *near misses* at an inspection?

Lack of using a survey meter

A lack of using a survey meter after each exposure seems to be the most frequent initial cause leading to an accident. The inspector shall check if appropriate equipment is used, its calibration, e.g. a label on an instrument, and documented daily checks. The inspector shall visually observe the use of a survey meter, e.g. who in the team is using it and how it is used. Discussion with radiographers shall follow, e.g. discussion on normal readings related to exposures and background and who sets alarm set point and changes batteries. In that way the inspector identifies the implementation of written procedures and if radiographers are familiar with them.

Inappropriate management of safety and warning systems of the enclosure

The inspector shall prepare a map of inspector's tour around and inside the enclosure, not to forget areas above and below it. The neighbouring areas should be included. This map based on the authorization file shall contain a location of radiographers and other persons during exposure and all safety and warning systems, such as maze, radiation monitoring probe and its display. All signs and labels should be noted as well as security features as appropriate. At the site the verification of this map shall be conducted. When talking to persons at a site the inspector shall check their understanding of the language used for warnings. When appropriate, functionality of safety features, e.g. interlocks, should be demonstrated by radiographers using the procedures.

Lack of procedures for recovery operations

The inspector shall not only check availability of recovery equipment and procedures but shall also verify if radiographers are familiar with them. The inspector might put detailed questions or ask for demonstration.

Inappropriate management of equipment used for industrial radiography

This *near miss* might be checked with analysis of equipment procurement procedures and data, e.g. checking how equipment compatibility is assured, and with verification of *Maintenance Programme* records.

Poor management of communication with on-site clients

Poor communication might be of course observed at on-site client site. But the inspector shall also conduct interviews with operator's decision makers to assess how safety issues are incorporated in their everyday discussions with on-site clients, e.g. in a discussion taking place before a contract with an on-site client is signed. Decision makers should demonstrate the awareness of safety rules to be used in this practice, i.e. safety culture.

CONCLUSIONS

Focusing inspection of industrial radiography on identification of five major mistakes leading to accidents in the past might be a useful approach to efficient and effective inspections. As such *near misses* might be difficult to identify by an operator, a role of an inspector in that respect cannot be overestimated. In that way the inspector contributes to prevention of accidents and encourages safety culture of an operator. RBs shall develop inspection protocols focusing on the identification of such *near misses*. This approach might be also used for developing other inspector's protocols as they facilitate on-site inspections and help new inspectors to enter the field.

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300-HOW DO I DEMONSTRATE TO THE REGULATOR THE ADEQUACY OF MY OFF-SITE EMERGENCY PLAN?

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The UK government introduced new legislation in 2019 to strengthen the national emergency preparedness and response arrangements for radiological emergencies both on-site and off-site. The Radiation (Emergency Preparedness and Public Information) Regulations 2019 [1] are designed to improve public protection and reduce adverse consequences in the event of an emergency. The changes deliver a consistent approach to radiation emergency preparedness and response across the civil nuclear, defence nuclear and radiological sectors. The role of the regulator is to regulate this on-site and off-site capability and ensure compliance to law and associated relevant good practice.

The purpose and scope of the poster presentation is to outline and identify the key facets of what would constitute as an adequate test of off-site emergency planning arrangements. Indeed, the purpose of testing emergency plans is to demonstrate their ability to deliver an effective response to any radiation emergency. An adequate test should therefore give confidence in the accuracy, completeness, practicability and adequacy of the plans and should identify how plans can be improved. The presentation will also identify and explain the processes and requirements to satisfy and ensure alignment between on-site and off-site emergency planning.

IAEA guidance (GSR Part 7, Requirement 25) [2], refers to exercise programmes being developed and implemented to ensure that all specified functions required are performed for emergency responses. In addition, exercise regimes should include the 'participation in some exercises of, as appropriate and feasible, all the organisations concerned, people who are potentially affected, and representatives of news media'. Furthermore, it should articulate how the exercises are systematically evaluated and in the UK and how / when key exercises are evaluated by the regulatory body (ONR). The presentation will reference the planning arrangements through to the exercise / test. See figure 1.

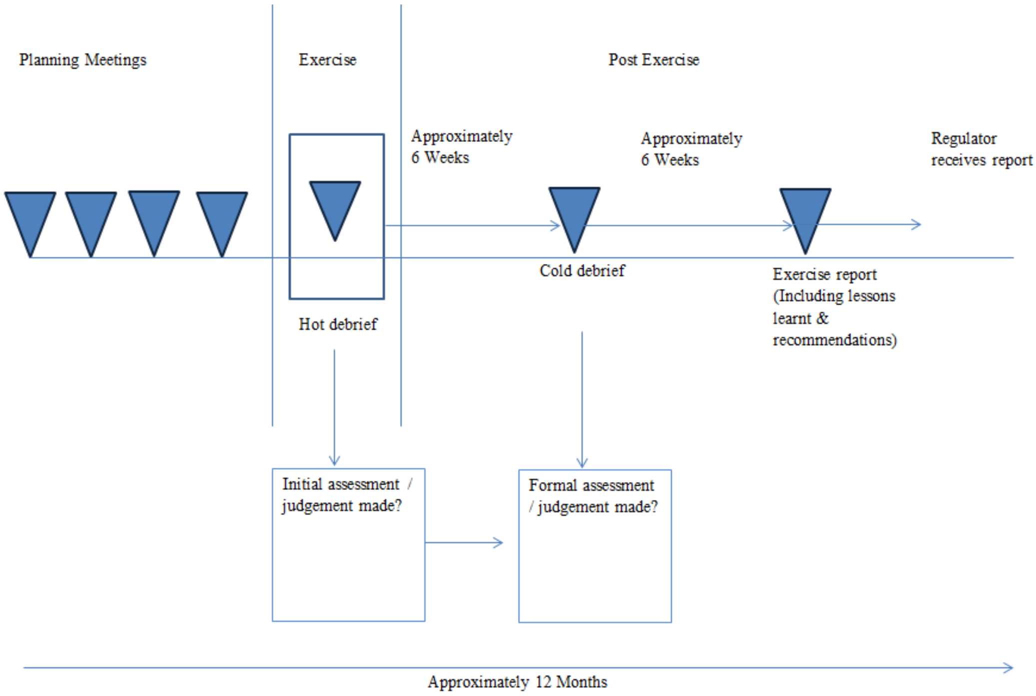
This will be explained using the UK approach to emergency exercise demonstrations, these are categorised as:

- Level 1, 2 & 3 tests - detailing the differences and expectations of each.
- A deeper description / understanding and requirements of Level 2 exercises.

Regulatory assessment / inspection of the process (see figure 1).

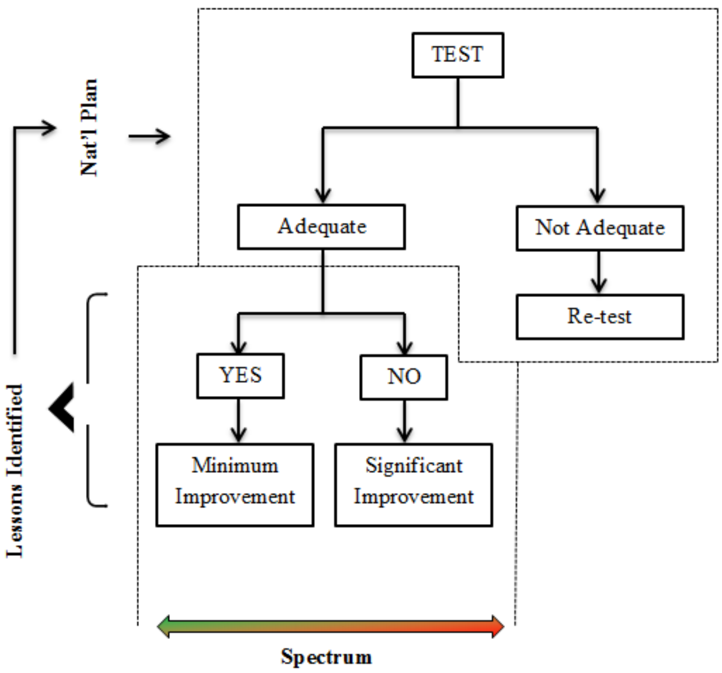
In addition to the above, the presentation will make reference to the current challenges associated against with the back drop of the Covid-19 pandemic and the strategies being developed to ensure continued legislative compliance.

Figure 1



The presentation will link the external attendance and influencing (by regulators) through local and national working groups such as the UK Lessons Learnt, Blue Lights, Local Authority and Nuclear Radiological Emergencies Working Groups and a national programme of testing which demonstrates continuous improvements. This will include demonstrating that the exercises are evaluated against pre-established objectives of emergency response to demonstrate that identification, notification, activation and response actions can be performed effectively to achieve the goals of emergency response and how lessons are learnt and recommendations taken forward both locally and nationally. See figure 2.

Figure 2



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301-DEVELOPMENT OF A TECHNICAL SKILLS COMPETENCE FRAMEWORK FOR REGULATORY INSPECTORS

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As part of an effective regulatory regime, regulators must ensure their inspectors are suitably qualified and experienced to undertake their regulatory roles. IAEA provides a number of guides on this topic, which include; Managing Regulatory Body Competence (No.79, 2013) [1], Methodology for the Systematic Assessment of the Regulatory Competence Needs (IAEA-TECDOC-1757, 2015) [2] and Building Competence in Radiation Protection and the Safe Use of Radiation Sources (RS-G-1.4, 2001) [3]. A significant proportion of this focus is on the regulatory skills; however, regulators also need individuals with specific technical competencies. Over time regulators also need to assess how their skills profile is developing and manage specific deficits introduced by changing work or movement of staff by aiding the targeting of either retraining, providing experience, recruiting or employing the use of a technical support contractor to close these deficits. This poster discusses the development of technical skills and behavioural competency framework and its deployment in supporting UK team of Radiological Protection & Criticality Safety inspectors in the assessment of the regulator's skills profile.

This poster will describe how the UK regulator:

- Defines sets of technical skills and behavioural competencies (under the topics of Generic, Emergency Preparedness & Response, Radiological Protection, Criticality Safety, Shielding and newly developed Radiological Consequences) for a large team of Radiological Protection & Criticality Safety inspectors;
- Utilises the process for assessing the competencies of inspectors by grading each from 'no knowledge' to 'expert' in each of 54 competency areas;
- Uses the output to assess the overall profile of the team and the individuals within it; and
- Uses the assessment results to retrain, provide experience, recruit and employ technical support contractors to close deficiencies.

ONR identifies a number of competency requirements in relation to the regulatory aspects of the regulatory roles which are detailed in a series of documents including:

- Competence Framework and Training and Development for ONR Staff, ONR-HR-GD-003 [4]
- Issue and Control of ONR Warrants and AVO Identification Cards, ONR-HR-GD-005 [5]
- Regulatory Competence Framework, ONR-HR-GD-009 [6]

There are additional competency requirements and processes for employees who are entering at a graduate level and or with little or no knowledge of nuclear regulation and is intending on developing the individuals into a nuclear regulatory role by providing experience, training and knowledge acquisition opportunities in a structured manner. This procedure and associated competencies are detailed in Application of ONRs equivalence process, ONR-HR-GD-001 [7].

ONR delegates responsibility for the maintenance of overall competency to the Professional Leads (PLs) for their Specialism. This activity is essential in order to secure the current and long term capacity, capability and resilience of the Specialisms. Reviews of competency are used to monitor and report on the status of competency across the Specialism and inform decisions relating to development, recruitment and deployment of staff. PLs also use them to provide assurance to Regulatory Leadership Team as necessary and inform ONR's Operational Effectiveness Indicators.

ONR allows a level of discretion in how each Specialism defines and develops their technical skills and behavioural competence suites, ensuring the flexibility so as not to constrain and set a one size fits all for the specialisms and reduces the risk of inappropriately influencing the definition of competencies leading to omissions or errors.

During 2016 the Radiological Protection and Criticality (RP&C) Specialism initiated a review of its technical skills and behavioural competencies. It recognised its current matrix of competencies and discrimination (scoring) required updating and specifically required greater detail in relation to requirements and specifically to attainment/success criteria to allow accurate self-assessment by its members when assessing themselves against the competency requirements.

The technical and behavioural competencies were developed with reference to the competency frameworks developed by a number of professional bodies, international guidance, and the internal ONR Equivalence process e.g., The Society of Radiological Protection Chartered Radiation Protection Professional [8].

These have been used to develop and underpin the following five categories of competencies that together define the competency of the RP&C Specialism as a whole:

— Generic Competences, which apply to all RP&C Specialism;
— Radiological Protection;
— Emergency Preparedness & Response;
— Criticality Safety; and
— Radiation Shielding.

Detailed descriptions of each competency were developed by teams of relevant ONR inspectors and employees and subject to internal peer review. An "Excel" spreadsheet tool was developed to allow individual inspectors and para-technical employees to assess themselves and score against each of the competencies. These self-scores are consolidated into the tool. The tool provides results of the scoring for each competence and suite of competencies in the consolidated tool and also in "Competency Snapshots". These snapshots use a weighting factor to modify scores for individuals against each competency to help differentiate between a large total score based on lots of inexperienced individuals scoring low and one or two experienced individuals scoring high and a broader range of scores.

This poster will conclude by providing a summary of the findings of the first analysis of the self-scoring activities carried out during 2019. It will also provide a set of recommendations and actions following that analysis.

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302-GSR PART 3 APPLICABLE REQUIREMENTS FINDINGS IN NUCLEAR MEDICINE MODIFICATION

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One of GSR Part 3 aspects that is addressed in several of its requirements is the treatment that, from the point of view of radiation safety, should be given to modifications that may have important consequences for protection and safety [1].

The presentation shows, for a nuclear medicine department that performs bone and thyroid scans, and metabolic therapy, both, ambulatory and hospitalized with ^{131}I, the justification analysis carried out of modification consisting in radioactive inventory increase (100% for technetium generator and 50% for ^{131}I), maintaining reception frequency and staff participating in the benefits (Requirement 10). This modification arises at the request of clinical oncology service, as a result of positive impact that bone and thyroid gammagraphy has had on the diagnosis precision, follow-up and treatment of cancer patients, after around a year of work.

It also sets the timely notification to regulatory body of intention to introduce to practice such modification (Requirement 9) and describes how the safety assessment that the Standards set forth in Requirement 13 has been carried out, as provided for these cases in the applicable Cuban regulation [2].

In accordance with the provisions of Requirement 30, regarding public exposure, it was submitted to examination and approval by the regulatory body before the modification materialized, since members of the public can be exposed during the reception at the airport and sources transfer from this to the Hospital.

As a demonstration of effectiveness of actions taken to guarantee the least possible impact on protection and safety, the result of the retrospective evaluation of the doses received by the Occupationally Exposed Workers (Personal Dose Equivalent for whole body Hp (10) and hands Hp (0.07)) has been made.

For this, the reports of these doses supplied by the dosimetry laboratory of the Radiation Protection and Hygiene Center [3], a provider of the state service, were evaluated. Comparing the maximum and average monthly doses in the period of March 2018, when the Department began to work with previous inventory to July 2019, with the period beginning from July 2019, as shown in table No. 1, the increase in inventory has only affected the increase in doses received (mSv/month) for preparing and ^{131}I administer technicians, at values always below the assumed value of annual limit fraction for one month, (40 mSv/month) for the equivalent of personal dose Hp (0.07) and of the order of the fraction of the Occupationally Exposed Workers dose constraints for one month [4], equivalent to 0.6 mSv/month.

TABLE 1. REPRESENTATIVE VALUES OF MONTHLY RECEIVED DOSE BY INVOLVED IN MODIFICATION STAFF.

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Responsability	Before modification				After modification			
	Maximum dose		**Average dose**		Maximum dose		**Average dose**	
	$H_p(10)$	$H_p(0,07)$	$H_p(10)$	$H_p(0,07)$	$H_p(10)$	$H_p(0,07)$	$H_p(10)$	$H_p(0,07)$
Technicians	0.39	24.62	0.28	7.53	0.9	17.9	0.63	16.8
Radiopharmaceutical	0.33	3.08	0.33	1.96	0	0.13	0	0.13
Nurse	0	1.43	0	0.67	0	1.6	0	0.8

The work shows that in this case, for made modification, compliance with regulatory authority provisions in the applicable Cuban legislation to nuclear medicine, is consistent with compliance with the provisions of the previously expressed GSR Part 3 requirements, regarding to carrying out modifications that may have important consequences for protection and safety.

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303-STRENGTHENING SUSTAINABILITY AND EFFECTIVENESS FOR NATIONAL NUCLEAR SAFETY REGIME BY IMPROVING RADIATION PROTECTION CAPABILITIES IN MYANMAR

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ABSTRACT

Myanmar, a State with very limited quantities of nuclear material, has gone through many improvements. Myanmar has acceded to the Convention on Nuclear Safety (CNS) and Convention on the Physical Protection of Nuclear Material (CPPNM) and its amendment on 6th December 2016 [1]. Furthermore, Myanmar has expressed a political commitment with regards to the Code of Conduct on the Safety and Security of Radioactive Sources.

As a legislation issue in Myanmar, Division of Atomic Energy (DAE) under the Ministry of Education (MOE), promulgated Atomic Energy law on 8th July 1998, and it was mainly based on radiation safety, does not cover nuclear safety, nuclear security and safeguards (3S strategies). In order to strengthen its national nuclear related legislation, DAE has just recently completed the drafting of the Myanmar Nuclear Law that prohibits the use, production, storage, distribution and import/export of nuclear and other radioactive materials without government license. Furthermore, the development of a number of regulations namely Nuclear Safety Regulation, Nuclear Security Regulation and Safeguards Regulation will follow soon. [2]

There is no radioisotope production facility in Myanmar and radioisotopes are still imported from other countries. The utilization of radiation sources and irradiating apparatus are limited to the use in medicine, industry, agriculture, education and training, livestock breeding and research.

The DAE has full regulatory competence; its mission is to oversee the safety and security of all the peaceful applications of atomic energy. All new radioactive sources and Radiation Apparatus imported to Myanmar are taken the Prior Permission first. The DAE issued the prior permission, registration, user license and re-export license step by step. During the licensing process and inspection activities, the DAE intends to assess both safety and security aspects at the same time. All of the inspection records and issued documents are recorded. [3]

When such radioactive sources loss their efficiency due to their half-life, they are designated as 'disused sources'. Safety and security of such sources are major concerns for regulatory authorities and licensees. Spent radioisotopes from medical and industrial sectors are sent back to the country of origin. Thus, safety problem related to spent radioisotopes and radioactive waste in Myanmar is very low/small. Accidents/incidents involving radiological and radioactive sources have not been occurred in Myanmar so far. The arrangements for mitigation actions have not been made yet. [3]

Being a developing country, Myanmar people are unfamiliar with the effective risk communication, radiation safety culture and system of radiological protection that is vital importance for the protection of

workers, patient, public and the environment. Furthermore, a few serious accidents in nuclear power plants in other country, primarily based on concerns about safety and cost, were the key influences on the loss of public confidence.

In order to enhance positive public opinion on radiation protection in State, Myanmar raises awareness to maintain and further strengthen national nuclear safety regime in strengthening nuclear safety globally using mass media (print media, television, radio, etc.), web space (including social networks), social and professional events, educational and cultural programme, translation of technical document into the national language, promoting core concept of radiation protection and supporting interested people and public in their daily use.

The main challenges of the DAE include generation gaps. Many of those working in the nuclear organizations are retiring and attrition of competent personnel is a real risk for Myanmar nuclear society. Knowledge transfer from one generation to the next is seriously suspend. To overcome these challenges, Government has adopted long-term education policies and put necessary efforts in the implementation of educational and training programmes on nuclear field.

Undergraduate programme on Nuclear Engineering was launched in 2008 at Technological University-TU (Kyauk-Se). These undergraduate students have been introduced to the Radiation Protection syllabus, Waste Management syllabus, Nuclear Instrumentation syllabus and Nuclear Security and safeguards syllabus as general materials. As of 2012, undergraduate program was implemented and over 80 well-trained person were graduated. The postgraduate programme has been recently introduced at Mandalay Technological University (MTU) and TU (Kyauk-Se).

In Myanmar, inadequate training of the staff, mishandling of sources are some of the challenges in our country. To overcome these challenges, the DAE, in collaboration with stakeholders, has been conducting a series of workshops and training to exchange information, knowledge and experiences on the system of radiological protection. Moreover, radiation protection training for radiographers from Government Hospitals are being conducted each year.

At the international level, in order to strengthen the State's nuclear safety regime and to fulfill the international obligation, the DAE, under the auspices of IAEA, is hosting international and regional meetings, to share the information and good practices in radiation protection.

The paper will present an approach to strengthen national safety regime, with special focus on regulatory effectiveness of inspection, control of radioactive sources, information regarding the best methods of effective training and educational programmes to concentrate on nuclear safety, still strives for continuous improvements to its performance of promoting the public awareness on radiation protection system and experiences and practice in safety culture in the use of radiation sources, as provided in the IAEA safety standards, to the protection of workers, patients, the public and the environment.

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304-REGULATOR'S APPROACH ON MEDICAL EXPOSURE IN LATVIA SUPERVISION OF OPERATORS AND COLLABORATION WITH STAKEHOLDERS, ESTABLISHING NATIONAL DRLS, ENHANCING QUALITY OF INSPECTIONS

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The presentation shows the measures taken to improve the work of the Radiation Safety Centre of State Environmental Service (RSC SES) in Latvia in the last few years, when increased attention has been paid to medical exposure and its supervising. Measures have been taken in a number of areas simultaneously, ranging from strengthening collaboration with stakeholders, improving the knowledge and skills of regulator’s inspectors, and strengthening patient protection at medical facilities through the development and implementation of national Dose Reference Levels (DRLs).

SUPERVISION OF OPERATORS AND COLLABORATION WITH STAKEHOLDERS

Radiation Safety Centre of State Environmental Service (RSC SES) performs state supervision and control in the field of radiation safety and nuclear safety. According to data as of 1 January 2020 the total number of operators in different areas is 1051, including 709 operators in dentistry and 116 operators in medical area [1]. The number of operators in medicine and dentistry determines the increased interest in medical exposure. Number of X-ray equipment used in medical applications and dentistry (by type) is shown in Table 1. Data is based on RSC SES compiled statistics [1].

TABLE 1. NUMBER OF X-RAY EQUIPMENT BY TYPE

Type of X-ray equipment	Number
Conventional radiography	186
Computed tomography	70
C-arc X-ray machines	74
Mammography	55
Mobile X-ray machines	35
Interventional radiology	16
Bone densitometers	25
Dental intra-oral X-ray	974
Panoramic dental X-ray	40
Conic beam computed tomography	32

Participation in the IAEA regional project RER/9/147 “Enhancing Member States' Capabilities for Ensuring Radiation Protection of Individuals Undergoing Medical Exposure” (2018-2021) made a significant contribution to identifying the problems in field of medical exposures. A Project on implementation measures in the field of medical exposure was set and plan with time schedule was created. The Project Implementation Team was established in 2018 (formally approved in March, 2019 by Order of director of RSC SES). The Team includes representatives from RSC SES (4 representatives), Health Inspectorate (1), professional societies (6), hospitals and medical centers (14), universities (2), technical service companies (6). Measures for improvement are discussed by Team and further forwarded for implementation in medical institutions.

In 2018 RSC SES elaborated Guidelines for operators “On patient doses’ analysis and optimization” and “Guidelines on DRL application”. In collaboration with Latvian Society of Radiology was updated “Referral guidelines and guidelines for individual justification on radiological procedures”, “Referral guidelines applicable to children”, “Clinical Audit guidelines” [2].

ESTABLISHING NATIONAL DRLS

One of main tasks of the Project in the field of medical exposure was establishing national Dose Reference Levels (DRLs). There were only international DRLs set in “Regulations Regarding Protection Against Ionizing Radiation in Medical Exposure” (2014). DRLs were expressed in terms of Entrance Surface Dose (ESD).

In 2018 a National Survey was proposed by RDS SES and discussed with the Project Implementation Team. The aim was to obtain values of DRL in terms of Dose Area Product (DAP), which could be more convenient for DRL application and dose analysis by operators. In 2019 data collection and analysis was carried out by RSC SES to establish national DRLs for CT and conventional radiography, using ICRP “Third Quartile” method [3]. Results [2] are going to be implemented into Regulations.

Next step commenced in 2020 with the task to establish national DRLs for mammography and Conic beam computed tomography.

ENHANCING QUALITY OF INSPECTIONS

Since 2018 RSC SES Inspection Division significantly improved quality of inspections carried out at medical facilities. The RSC SES approach includes:

- Frequency of inspections at facilities is set according to the level of hazards and risks (graded approach is implemented in the planning inspections);
- Checklists are elaborated for specific fields (radiotherapy, nuclear medicine, radiological diagnostic, X-ray dentistry);
- During inspections the inspectors interview the staff, check documentation, make indicative measurements (workplace monitoring, surface contamination, conformity of technical parameters of radiological diagnostic equipment), check image quality etc.

In 2018 RSC SES organized procurement on measurement equipment for Quality Control (QC) of technical parameters of radiological diagnostic equipment to increase capabilities of inspections. For X-ray equipment were purchased: Multifunction X-ray meter (kV, ms, etc.); Patient Dose Calibrator (for checking DAP meter); mHVL filters; Image quality test phantom (for testing homogeneity, resolution etc.). For Computed Tomography CT dose phantoms (head and body) and ionization chamber was purchased. In 2020 with support of IAEA was obtained also CT and mammography image quality test phantoms.

In May, 2019, an expert mission (supported by IAEA) was organised for inspectors. The aim was to increase the quality of inspections in diagnostic radiology, including image quality and dose, optimization and basic QA/QC. The training for inspectors was carried out for assessment of clinical images and dose data, interpretation of the observations. After the training, the inspectors were able to perform the measurements themselves. Since 2019 inspectors occasionally include QC measurements and image quality tests in the scope of inspections at radiological diagnostic facilities.

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305-CHALLENGES FOR RADIATION PROTECTION IN THE MANAGEMENT OF RADIATION AND NUCLEAR LEGACIES

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Many countries with nuclear energy and related programs, involving man-made and naturally occurring radioactive material, are facing challenges with radiation and nuclear legacy sites and installations [1]. They must be managed in an open and transparent way, addressing the views of relevant stakeholders, so as to build confidence in the solutions being developed. The development of an effective process that addresses these complex and interacting issues will help to avoid the creation of more legacies in the future. Common characteristics that bound the development of a practical approach to legacy site management and regulation include the following [2]:

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

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

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

This presentation will review the conclusions and recommendations of report of the EGLM [2] connected specifically with:

— Holistic Optimisation;
— Type of Exposure Situation, and
— Prescriptive and Performance Related Regulations.

These issues will be discussed further in the light of continuing activities of the NEA's Committee on Decommissioning and Legacy Management, and related international developments, including the results of presentations and discussion at the international workshop held in Tromsø in November 2019 on the subject of the Regulatory Framework of Decommissioning Legacy Sites and Wastes from Recognition to Resolution: Building Optimization into the Process [3]. The discussion will highlight the complex circumstances at major sites, the role of reference levels, the value of stakeholder engagement, the concept of sustainability, the use of an integrated all-hazards approach to risk management, and the role of adaptability in approaches to management and regulatory supervision.

Based on the above material, options will be presented to address challenges for radiation protection in the management of radiation and nuclear legacies.

ACKNOWLEDGEMENTS

The vital contribution to this paper of all the members of the NEA Expert Group on Legacy Management and the organisations that shared information for case studies is gratefully acknowledged, alongside the Secretariat at the NEA.

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306-THE RESULTS OF THE STUDY PERFORMED FOR THE CHARACTERIZATION OF REPRESENTATIVE PERSON FOR BELARUSIAN NPP

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A “representative person” concept according to the ICRP 101 Publication recommendations [1] is set up in the Belarusian national radiation safety standards [2] for the purpose of radiaion protection of the public. That is why a set of parameters describing the “representative person” had to be defined for the dose assessment purpose for public living around Belarusian NPP.

According to the Belarusian regulation one of the elements of the radiation safety provision is performing of radiation-hygienic monitoring which includes dose assessment to public [3].

The Republican Scientific Practical Center of Hygiene under the Ministry of Health (RSPCH) has developed and tested the method for the defining of parameters characterizing the Representative Person for the Belarusian NPP. The method includes a specially-tailored questionnaire for the public living in the vicinity of the Belarusian NPP and the data collection from regional and local authorities.

Since 2014 and up to now the RSPCH is caring out pre-operational studies for Belarusian NPP to establish ‘baseline’ activity concentrations of natural and artificial radionuclides in food and drinking water. At the same time the RSPCH is making surveys of socio-hygienic characteristics of the region (households, sources of drinking water supply, type of housing, consumption of local food, mushrooms, berries, game, fish, etc.)., habits of local population for subsequently determining the impacts of this facility to public health. The study is carried out for the observation zone of Belarusian NPP (12.9 km radius around the site with the population of about 8 thousand people) and in the Ostrovets city (the nearest to the site big settlement).

The developed questionnaire consisted of three parts: 1st part of the questionnaire was the passport part, which determines age, gender, place of residence, level of education, profession, position, as well as family and parent statuses; 2nd part contained questions that will help determine the parameters, significantly affecting the intake of radionuclides (living conditions, time spent outdoors, consumption local products, water supply, etc.); 3rd part contained questions determining the frequency of consumption of products, which allowed to study of the actual nutrition characteristics of different age and gender groups.

The analysis of the data for the period from 2014 to 2018 years demonstrated that in Ostrovets region prevails the rural population (61.5–53.7%). Urban population is represented exclusively by residents of the regional center - the Ostrovets city.

The number of women per 1,000 men in different years ranged from 1049 to 1123 in the Ostrovets district (in general in the Republic of Belarus at the beginning of 2019 y, per 1000 men were 1,146 women). This indicator depends on the average life expectancy of men, which is noticeably lower than that of women [4].

A comparative analysis of the age structure of population was also made. The percentage of groups employable persons is 54.6-55.6%, younger than employable (below 15 yrs) is 18.6-19.8%and older than employable ages (more than 64 yrs) is 24.6-26.6% [4].

The results of this survey demonstrated that the majority of the population (81%) is living in the private houses. In summer people spend about 7.5-8.0 hours outdoors and in the winter – up to 3 hours. About 90% of respondents have private gardens (78-85% of consumed vegetables are self-grown) and about 57% of them have livestock (41%-chicken, 33% - pigs, 12% - cows). About 72% of respondents consume milk purchased in a store (no more than 1 liter per day) and 67% of respondents consume meat from private farms. The consumption rates of local foods for particular age groups and particular food products (including wildly grown, game and fish from local rivers and lakes) were also defined.

The results of the survey were inserted into a specially created database.

This study will help to define the characteristics of the “representative person” for the Belarusian NPP. Using the regional data for public dose assessment helps to reduce the level of conservatism in the dose estimates by several times.

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307-THE HORMESIS-THRESHOLD BIODOSIMETRIC MODEL FOR ESTIMATION OF LOW-SLOW DOSE EFFECTS FROM INTERNAL SOURCES OF IONIZING RADIATION

Comparative study of hormesis-threshold biodosimetric model and the optomagnetic imaging spectroscopy method in characterization of biophysical properties of pulmonary tissue

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Introduction: Hormesis-threshold model for individualized lung cancer risk assessment in smoking and low-dose radiation exposure has presented firstly in 2015 [1], and later in the monograph related to the low-slow dose effects from internal sources of ionizing radiation [2]. Recently the differences in paramagnetic/diamagnetic properties of pulmonary tissue were evaluated before and after external influences of different types of light sources: W- white LED, UV - ultraviolet light, IR – infrared, R- red, G- green [3,4].

Methods: Apoptotic parameters in cytocentrifuge preparations of bronchoalveolar lavage cell suspensions were evaluated by light microscopy using the TUNEL in situ cytochemical method in healthy nonsmokers and smokers and hard smokers with pathologically and clinically confirmed diagnosis of non-small-cell lung cancer. Based on the apoptotic index (ratio yielding by apoptotic cells) and apoptotic capacity (reflects a generation of apoptotic bodies and their clearance by alveolar macrophages) the graph was carried out by a neural network method (Fig. 1). Optomagnetic imaging spectroscopy (OMIS) method (NanoWorld AG, Belgrade) was performed by ILT 350, International Light Technology, USA. Transmittance W/m^2 was evaluated at 380-780 nm.

Results: The graph obtained by the neural network method (Fig. 1) can be used for the assessment of the dose-response relationship regarding smoking and radiation effects at low-doses. Both approaches, biodosimetric method based on apoptotic parameters and artificial intelligence method, as well as OMIS provide a more clear distinction of nonsmokers, smokers, and smokers with lung cancer according to apoptotic parameters and smoking exposure (Fig. 2).

Interpretation: The findings fit with the hormesis-threshold model of tissue response to low-dose radiation. The method represents a step toward individualized screening and lung cancer risk assessment. The proposed model does not require an exact measuring of tissue doses in conditions of exposure to low doses of radiation, especially alpha-emitting radioisotopes. This method enables quick orientation about the extent of damage of complex tissue regulatory mechanisms *in situ*, and indirectly, it may indicate the existence of adaptive, premalignant, or malignant lesions [1,2]. The results of OMIS measurement showed a distinction of control groups (nonsmokers and smokers) by diffuse and polarized W and UV light, advantageously with the influence of W light. When OMIS was performed using red light, a clear distinction between control groups (nonsmokers and smokers) and a group of patients with lung cancer was found. Some of the smokers' values overlap with those of nonsmokers, as well as cancer patients. The result corresponds to the finding of the fields of radiation hormesis (Fig. 1). The proposed model does not require an exact measuring of tissue doses in conditions of exposure to low doses of radiation, especially alpha-emitting radioisotopes. This method enables quick orientation regarding the extent of damage of complex regulatory mechanisms in the tissue in situ, and indirectly indicates the existence of premalignant or malignant lesions. The hormesis model of tissue response to radiation deviates from the threshold model and assumes that radiation in higher doses increases the risk of cancer, but not linearly. The model implies the existence of a threshold that cannot be clearly defined because of the tune changes in individual immunocompetence of tissue [1]. In conclusion, low-dose radiation induces mixed hormesis and threshold tissue response. OMIS method may support lung cancer screening because the magnetic properties of tissue precede to immune tissue response.

FIGURES

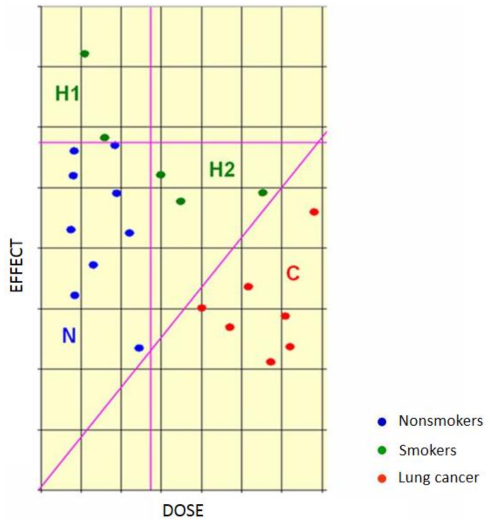

FIG. 1. Dose-response relationship in smoking and low-dose-radiation exposure based on neural network method. Dose (along x line) represents smoking and low-dose-radiation exposure. Response (along y line) represents the response of tissue and reflects a complex network of the cell to cell interactions and tissue remodelling, including the balance between apoptosis and apoptotic clearance. H1 and H2 fields represent areas of radiation hormesis. Adapted from primary source Zunic and Rakic, 2016 [2].

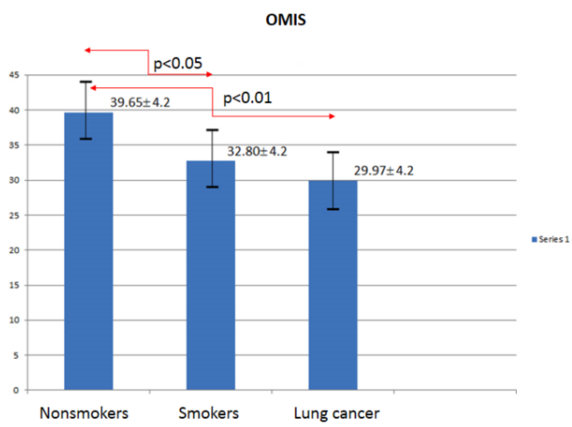

FIG. 2. OMIS results for groups of nonsmokers, smokers, and smokers with lung cancer.

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308-ASSESSMENT OF NEUTRON AIR CREW DOSES IN BRAZIL WITH THE MONTE CARLO METHOD

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As airline transportation becomes more important, the exposure of air crew to ionising radiation from cosmic rays also. This is due to two main reasons: an increasing number of hours in flight annually that come from the demand and the higher altitudes that regular flights can reach by technological advances, with a clear trend towards higher doses. Several studies on the subject [1][2][3] have advanced on the evaluation of those doses.

The United Nations Committee on the Effects of Atomic Radiation [4] estimates the world average annual effective dose of air crew to be in the range of 1.2 – 7.0 mSv. Other international organizations (ICAO, IATA, IFALPA) estimate this value to be between 2 and 5 mSv per year. The European Union recognizes air crew as occupationally exposed workers and so does the United States of America, since 1994.

In Brazil, federal regulations do not apply this same status of occupational exposure to air crew. As most of its territory under the influence of the South Atlantic Magnetic Anomaly (SAMA), that lowers the Van Allen radiation belt, allowing thus higher radiation doses, the evaluation of air crew annual doses is particularly important [5].

It has been shown that the main component of air crew doses in high altitude commercial flights in Brazil comes from neutrons [7]. Rochedo et al [8] evaluated individual doses in domestic flights in Brazil using the CARI-6 code. They found that doses are in the range of 0.03 – 8.8 μSv. A typical flight from Rio de Janeiro to São Paulo presents an average individual dose estimated by them to be around 1.8 μSv.

Common flight routes in Brazil were selected to evaluate the neutron doses. The neutron field present at those altitudes was generated by EXPACS (EXcel-based Program for calculating Atmospheric Cosmic-ray Spectrum) [9] that allows the calculation of cosmic ray fluxes of neutrons, protons, muons and photons in the Earth's atmosphere, given the altitude, longitude, local cutting stiffness and other parameters. Those parameters were selected from the airline route at regular intervals and for each point, given the calculated neutron spectrum, an effective dose was calculated. For this calculation, a Monte Carlo code was developed using the Geant4 toolkit. The ICRP 103 voxel phantoms for both genders were used in the calculation. Figure 1 shows an example of neutron and photon fluxes estimated by EXPACS for a commercial flight from São Paulo to Rio de Janeiro. The code also estimates the neutron component of effective dose to be 0.49 μSv/h. The analysis of annual air crew doses show that, for some high altitude flight routes, they may be subject to doses that exceed the limit for public justifying thus their monitoring.

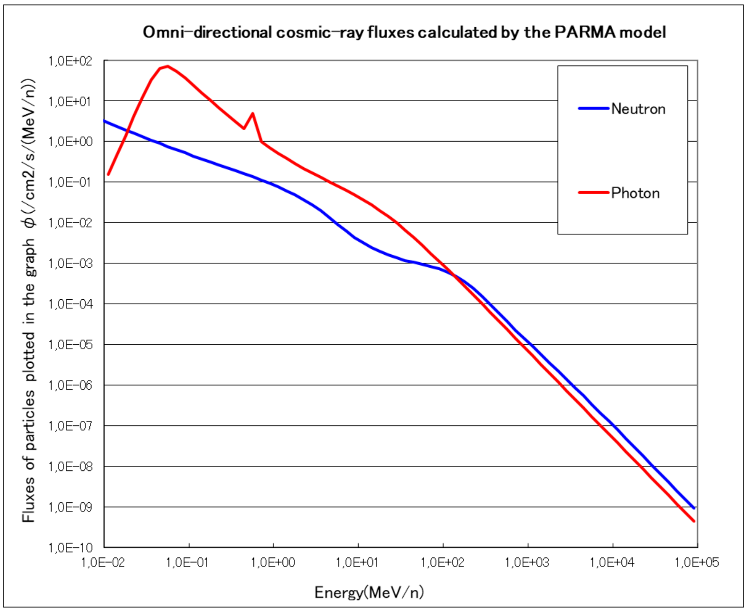

Figure 38 Neutron and photon fluxes calculated by EXPACS for a typical flight from São Paulo to Rio de Janeiro.

ACKNOWLEDGEMENTS

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309-WORKING ON DECISION MAKING AND STAKEHOLDER INVOLVEMENT IN THE ARGENTINE ADVISORY COUNCIL ON MEDICAL, INDUSTRIAL AND RESEARCH USES OF RADIOISOTOPES AND IONIZING RADIATION USES (CAAR)

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The presentation summarizes the role of the "Advisory Council on Medical, Industrial and Research Uses of Radioisotopes and Ionizing Radiation Uses" (CAAR, according to its Spanish acronym) of the Argentine Nuclear Regulatory Authority (ARN), describing the routine activities related to evaluating compliance with the requirements associated with radiation protection accordingly the ARN policy and the proposals from the stakeholders related to new technologies that imply a review of the requirements in force.

REGULATORY FRAMEWORK

The ARN has the mission to protect people and environment of the deleterious effects of ionizing radiations derived from nuclear activities in accordance with the provisions of the Nuclear Activity Act N° 24.804 [1] and its Regulatory Decree N°1390/98 [2].

The authorization process is one of the tasks among those performed by ARN for the different facilities and practices.

According to ARN, Class II facilities (facility or practice that only requires an operation license), the authorization processes are associated with the granting of individual license for the staff occupationally exposed.

ARN´s classification includes in the Class II: medical linear accelerators; tele-therapy; brachytherapy; nuclear medicine departments; self-shielded gamma irradiators; gamma radiography; mining and milling installations (not including mining tails disposal); nuclear installations without criticality potential; research and development in physical-chemical and biomedical areas and import, export, and storage of radioactive material. The total registered facilities for this class are around 2000.

OBJECTIVES

The objective of CAAR is advised to the Board of Directors of the ARN in the process of approval the applications of personal licenses for the use of ionizing radiation in medicine and industry (class II facilities with the exception of nuclear fuel cycle facilities, due to this is the task of the other ARN Council).

Their job is related to:

- Recommend the action to be taken on each application (approval or reject) for a new license or license renewal/modification.
- Advice on updating or proposing new requirements.
- Advice on education and the training and actualization programs that applicants must comply with.
- Analysis of other issues that may arise linked to individual license and formulation of relevant recommendations.

— In this assessment of background and competencies, the CAAR works as a complementary instance advising directly to the ARN Board of Directors in accordance with the art 4.39 of GSG-12: Organization, Management and Staffing of the Regulatory Body for Safety [4]:

—

— 4.39. Advisory committees should report to the highest level of authority within the regulatory body

CONFORMATION

The CAAR is a team conformed by stakeholders and personnel from ARN.

The stakeholders are recognized experts from the following profile: physicians specialized in radiotherapy (2); physicians specialized in nuclear medicine (2); medical physicist specialized in radiotherapy (1); medical physicist specialized in nuclear medicine (1); industrial uses expert (1); radiopharmaceutical industry expert (1).

The participation of stakeholders provides transparency to the ARN licensing process, meanwhile through their expertise and knowledge they help ARN regarding the regulation of new radiopharmaceuticals and technologies in medicine or new industrial practice to develop coherent regulation for assuring the radiation protection of workers, publics and patients.

The personnel of ARN accomplish the role of President, Technical Secretary and Administrative Secretary of CAAR.

Both the stakeholders and ARN personnel are designed by the Board of Directors of the ARN.

In the accomplishment of their duties, all members of the Council may express their opinions with complete independence of their relationship with departments and organizations to which they belong. This is in accordance with the provisions in GSR Part 1 - Governmental, legal and regulatory framework for safety [5]:

4.18. The regulatory body may decide to give formal status to the processes by which it is provided with expert opinion and advice. If the establishment of advisory bodies, whether on a temporary or a permanent basis, is considered necessary, it is essential that such bodies provide independent advice, whether technical or non-technical in nature.

CONCLUSIONS

The Advisory Council on Medical, Industrial and Research Uses of Radioisotopes and Ionizing Radiation (CAAR), according to its Spanish acronym) of the Argentine Nuclear Regulatory Authority (ARN) is a collegiate body, which main task consists of advising the Board of Directors directly regarding the granting of user licenses and all the issues associated with the assurance of radiation protection in practice. The commitment of the stakeholders contributing with their expertise and knowledge is a key in the process of licensing in Argentina.

The member´s expertise besides highest recognition in their specialty, determine a very qualify advice and it is a valuable complementary review in the ARN licensing process.

Finally, CAAR is an example of stakeholder commitment that contribute in regulatory decision making for affording greater confidence and quality to the process while providing greater transparency.

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[15] IAEA GSR Part 1 - Governmental, legal and regulatory framework for safety

311-ACCURATE FLUKA APPLICATION FOR RADIATION SAFETY ASSESSMENT AROUND CYCLOTRONS

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Monte Carlo codes are widely used in radiation safety assessment and proved their efficiencies in numerical calculations. But, the results depend mostly on the users' input including geometry design, assigned material, experimental setup, and physics models. However, our work is focused on the establishment of an accurate FLUKA input [1] dealing with safety assessment around medical cyclotrons and based on IAEA guides [2]. The advantages to building this input, are: first to save a lot of time to repeat this kind of work for every experiment within the same facility and second to allow predicting radiation protection results, which cannot be performed experimentally or being very costly. Always, within the scope where this optimised application is validated with the experiment. The architecture of the FLUKA input is presented in the left part of FIG.1 and the geometry of a standard cyclotron facility (bunker and casemate) with the setup of the experiment in the right part of FIG.1.

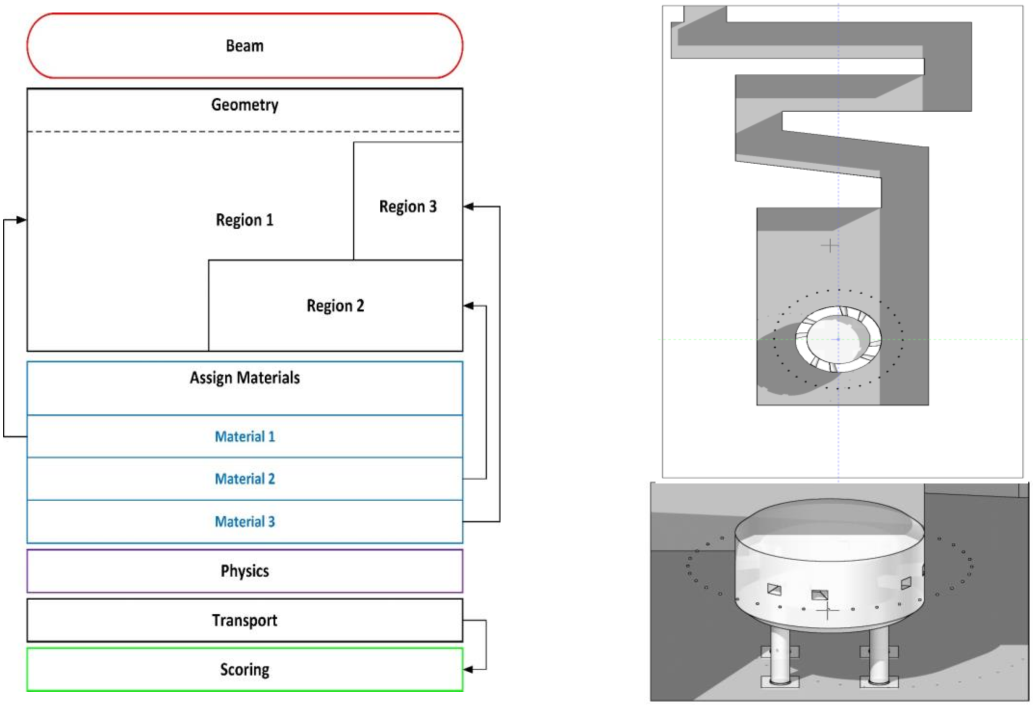

FIG. 39. FLUKA input structure (left). Horizontal overview of the layout (top-right) and a zoom on the cyclotron with dosimeters around (bottom-right)

When thinking in the protection of workers, the public and environment against ionizing radiation in cyclotron facilities, neutrons represent the particles with the highest risk. This is why our work is focused on neutron effects. Then, differential neutron fluence scoring in a liquid target for FDG production and the vault room air has been validated for two types of cyclotrons: IBA 18/9 MeV and PETtrace 16.5 MeV [3] (see FIG.2). Furthermore, an adjusted ICRU sphere was successfully calibrated for ambient neutron equivalent dose H*(10). Which gave the possibility to assess neutron H*(10) at any position in the cyclotron vault room where the experimental setup could be very hard to put in place as well as time management. Hence, the ALARA principle is justified when using FLUKA code instead of experimental measurements.

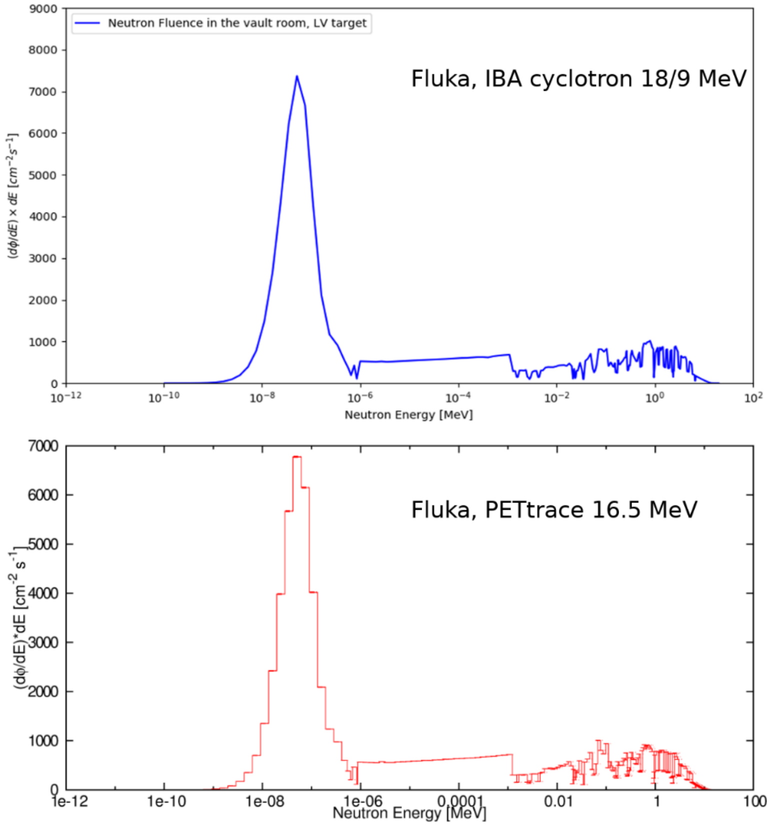

FIG. 40. Fluka Differential neutron fluence in the target. Top part is with IBA Cyclone®18 cyclotron (our results). Bottom is in PETtrace cyclotron (Infantino, 2015).

Besides, the uncertainty analysis has been carried out to identify the energy span, where FLUKA results are valid for neutron fluence assessment and ambient neutron equivalent dose "H*(10)" simulation regarding the locations of the detectors.

Given the obtained results and the agreement with the reference [3] findings, our input file is available for evaluation and use in the benefit of the radiation safety community.

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312-THE ROLE OF INTERNAL DOSIMETRY IN THE FOLLOW-UP OF METASTATIC CASTRATION-RESISTANT PROSTATE CANCER THERAPY WITH ^{177}LU-PSMA

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Quantitative imaging proved its efficacy in the assessment of the individualized internal dosimetry based on patient-specific data. Furthermore, hybrid dosimetry for Targeted Radionuclide Therapy (TRT) is a good approach to overcome the limitations of structure overlapping. Hence, the absorbed dose is easily estimated by avoiding organ overlap. For this purpose, a dosimetry study for the emerging ^{177}Lu-PSMA-617 radioligand treatment of seven patients with metastatic castration-resistant prostate cancer (mCRPC), has been reported. The dosimetry workflow is described. Absorbed doses and residence time metrics in organs-at-risks as well as the fraction of the whole-body absorbed dose and the initial injected activity, are given.

Seven patients (57 ≤ Age ≤ 78), with mCRPC, underwent SPECT/CT imaging (Siemens Intevo Bold) up to four times at 2-4, 24, 48 and 72 hours after administration of prescribed ^{177}Lu-PSMA-617 (activities: 6.0 - 7.4 GBq) between 4 and 6 cycles. Before the therapy, patients were informed about the procedure and signed a consent according to national regulations and approval of the competent Local Health Authority was obtained. The level of aggressiveness, given by the Gleason score GS is between 6 and 8. Follow up examinations included kidney function tests, and measurement of serum PSA is taken into account.
The sensitivity of the SPECT was determined by quantifying the ^{177}Lu activity concentration in a Region Of Interest (ROI) of an image of a phantom containing six hollow spheres with different diameters (37, 28, 22, 17, 13, 10) mm and acquired at the same conditions as patients SPECT/CT scans. A dose calibrator (Isomed, MED nuclear 2010) was used to determine activities in the same background conditions and with the same vial [1-3]. With an initial activity background of 0.037 MBq/ml, the calibration factor is equal to 12 cps/MBq. Recovery coefficients are fixed to the unity for organs exceeding the 1 cm size.
Data reconstruction was determined using the HybridRecons tool (Hermes software) [4]. Coregistration and segmentation, using the quantitative HybridDosimetry 5.0.1, were established. Volumes Of Interest (VOI) of organs-at-risk (kidneys and liver) at each time point, were defined (Figure 1). Finally, the cumulated activity at each organ and time of administration to the patient were used to deduce absorbed doses.

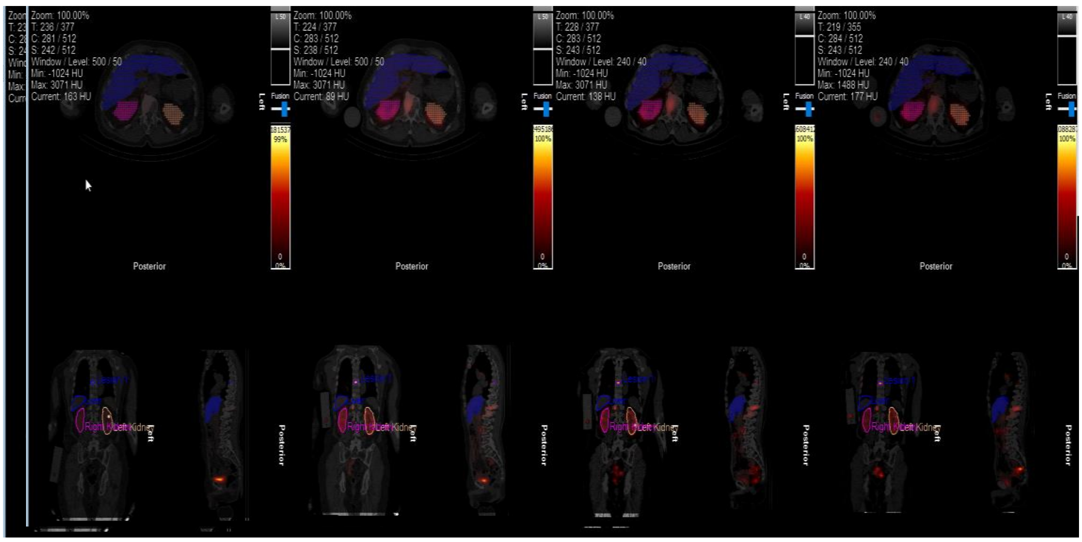

FIG. 1. Organs at risk contouring with Hermes software (Patient1 Cycle 1; 2-4h, 1d, 2d, 3d)

For the seven patients included in this study, the mean absorbed doses accumulated by organs-at-risk are calculated. Patients should not receive doses exceeding the dose limit that would lead to toxicity (27 Gy for kidneys) [5-6] and treatment safety has to be confirmed.
On average, the absorbed doses in kidneys for all patients does not exceed 4.8 Gy. The liver accumulates a dose between 163 and 734 mGy per cycle.
Since the total body doses are photon-components, the mean absorbed dose is still low (< 400 mGy). As a consequence, the whole-body absorbed dose per injected activity does not exceed 0.032 mGy/MBq.
The residence time in kidneys and liver is low (less than 2 hours) and high in the whole-body (from half a day to more than two days).
Knowing the cumulative absorbed dose in organs-at-risk during the treatment, principally kidneys enables optimization of the TRT efficacy by reducing the number of cycles or adjusting the administrative doses to each patient. This information is important in the treatment planning.

The workflow of the internal dosimetry in a daily clinical routine at Inselspital Bern is detailed and continued optimization of all parameters in the dosimetry process is ongoing. Keeping in mind that increasing the number of patients will lead to better accuracy in internal dosimetry, ongoing and future cycles will be considered in future dose calculation. Nevertheless, in vivo urinary and blood investigations are needed to determine the in-vivo-dose and for the effective half-life estimation.

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314-ESTABLISHING REGIONAL DIAGNOSTIC REFERENCE LEVEL (DRL) FOR CHEST RADIOGRAPHY IN MADAGASCAR

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INTRODUCTION

The radiation exposure resulting from the medical diagnostic examinations is very significant and represents the largest man-made source of the radiation exposure to population in Madagascar. Establishment of the Diagnostic Reference Level (DRL) helps to avoid radiation dose to the patients that does not contribute to medical diagnosis [1], but so far, there is a lack of data to allow such determination. The aim of this study is to evaluate the radiation doses to patients undergoing chest diagnostic X-ray examinations and to assist in the development of regional DRL for chest radiography in the capital Antananarivo City and the southern regions of Madagascar.

MATERIALS AND METHOD

Chest radiography is the most frequent X-ray examination for the six hospitals under investigation. Two are teaching hospitals (H1 and H2), two are private hospitals (H4 and H5), located in Antananarivo City, and two are regional hospitals located in Fianarantsoa City, southern Madagascar (H3 and H6). One X-ray machine is investigated for each hospital.

The information from the X-Ray machines, the technical factors, such as the applied tube potential (in kVp), the charge (in mAs), the focus skin distance (FSD in cm) and the patient characteristics such as the weight, height, age and thickness of the irradiated regions for chest examination were recorded. The output of the X-ray machines (in mGy/mAs) at 1 m distance were measured using thermoluminescent dosimeters [2]. Entrance Skin Dose (ESD) was calculated from the X-ray machine output, using backscatter factors and the technical factors [3, 4], following the equation:

$$ESD = OP * C * \left(\frac{100}{FSD}\right)^2 * BSF$$

OP: Output machine (mGy/mAs)
C: Charge (mAs)
BSF: Backscatter factor (a BSF value of 1.35 was used as suggested in the European guideline) [5]

The DRL was set at the third quartile value of the ESD distribution [6]. In all, data were recorded and determined for 176 patients who underwent chest examinations, during the period from January to February 2019

RESULTS AND DISCUSSION

The technical factors and the mean value of Entrance Skin Dose for chest examination from the six investigated hospitals are presented in Table-1.

As a result, the value of 0.60 mGy was obtained as DRL for chest examination for the two studied regions of Madagascar. The value obtained in this study is higher than data published by the IAEA-BSS-115 for chest examination [7]. Corrective actions include the need for adjustment of the technical factors and implementing quality assurance programme in these six hospitals in order to avoid unnecessary risks of increased radiation dose to patients.

TABLE 1: TECHNICAL FACTORS AND MEAN ENTRANCE SKIN DOSE (ESD)

Hospital	Patient number	kV	mAs	FSD (cm)	ESD (mGy)
H1	21	81 (70 – 95)	6.4 (5.6 – 7.1)	136 (105 – 150)	0,12 (0,07 - 0,16)
H2	62	75 (50 – 105)	25 (8 – 63)	137 (70 – 200)	0,28 (0.06 – 1.30)
H3	36	78 (60 – 86)	9.2 (4.6 – 19)	154 (70 – 170)	0.27 (0.13 – 1.07)
H4	27	76	29 (20 – 40)	117 (105 – 140)	0.64 (0.39 –1.12)
H5	16	49 (48 – 50)	18 (16 – 20)	105 (90 – 115)	0.54 (0.36 – 0.82)
H6	14	68 (58 – 86)	93 (40 – 200)	104 (100 – 110)	6.09 (1.79 – 8.51)

CONCLUSION

From 176 patients, a preliminary value of the DRL from two regions of Madagascar is set at 0.60 mGy, which is in excess of standard value. However limited, the result suggests that to mitigate the radiation harmfulness, the diagnostic practice should be improved and optimized, to avoid unnecessary radiation dose excess during X-ray radiography exercises in the two regions. To improve the current study, more data need to be gathered. To extend it, other examinations need to be investigated, but also for other regions, with the aim of producing national DRLs.

ACKNOWLEDGMENTS

The authors would like to thank the six hospitals and their staff that participated in this study, by making available the X-ray machines to carry out measurements. Gratefulness is also expressed to the INSTN-Madagascar Radiation Protection and Dosimetry Department team, for their invaluable help in processing the data.

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315-IMPROVEMENTS IN THE ESTIMATION OF INGESTED DOSE FROM NATURAL RADIOACTIVITY IN THE IRISH DIET

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In 2014, the Radiological Protection Institute of Ireland[9] produced a report which outlined the radiation doses received by the Irish population. This dose report estimated that the annual effective dose from the ingestion of food and water was 267 µSv, which accounts for 6.5% of the estimated average annual effective dose in Ireland [1]. It was estimated that 98% of the radioactivity ingested in the Irish diet is as a result of natural radioactivity, highlighting the importance of natural radionuclides in the determination of ingestion doses. The IAEA and UNSCEAR have provided reference values of 260 µSv for the annual effective dose for the ingestion of radioactivity in food [2], [3], which are in good agreement with the dose estimate for Ireland.

The goal of this study was to provide a more accurate estimation of the ingested dose through an improved duplicate diet sampling survey and the development of radioanalytical methods for specific natural radionuclides of interest. An outcome of this work will be the reassessment of the annual effective dose due to the ingestion of natural radioactivity in food for the Irish population.

The sampling methodology used was a duplicate diet survey accompanied with food diaries. In previous Irish assessments, duplicate diet surveys had been conducted by the collection of food samples in a large canteen facility in Ireland. The sampling approach in the current study involved the collection food from eight participants over a seven-day period. By sourcing food from individual participants throughout the whole day over a period of one week, it is expected that a more representative sample of varying consumption rates and dietary habits will be obtained. A summary of the improved sampling methodology will be discussed.

Radioanalytical methods in the laboratory were developed with the aim of improving detection limits to enable the measurement of low levels of natural radioactivity in food and to minimise the time required to conduct the analysis and assess the activity concentrations. The radionuclides that were analysed were ^{210}Po, ^{210}Pb and ^{14}C. An overview of the techniques used will also be provided.

Results of the radionuclides measured in the samples from the duplicate diet survey will be provided.

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316-MANAGING RADIUM LEGACIES FROM THE WATCH INDUSTRY: THE SWISS EXPERIENCE

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The Radium Action Plan 2015-2022 aims at mitigating the risks arising from radium legacies from the watch industry in Switzerland. This paper describes the measures put in place regarding the handling of contaminated properties and former landfills where wastes containing radium were deposited.

INTRODUCTION

In 2014, the problem of radiological legacies related to the use of radium-based luminescent paint in the watch industry resurfaced in Switzerland. Following the discovery of radium-bearing waste at a former landfill site, the press warned the public that many buildings are still potentially affected by radium contamination. The use of radium-based paint in watchmaking workshops or private dwellings was indeed quite common between 1920 and 1960. At that time, no systematic control of radium contamination was carried out in those former workshops or apartments that were used for work at home. Moreover, radium-bearing wastes from these activities were disposed of in municipal landfill sites together with ordinary household waste.

THE RADIUM ACTION PLAN 2015-2022

In order to permanently settle the problem of radium legacies from the watch industry and to address the concerns of the population appropriately, the Swiss government adopted the Radium Action Plan 2015-2022 and provided funding of CHF 9 million for the whole period. The most affected cantons as well as the watch industry support the Action Plan by means of a voluntary financial contribution. The Federal Office of Public Health (FOPH) is responsible for the implementation of the Action Plan, which is divided in two distinct parts with different objectives:

(h) The management of potentially contaminated properties (buildings and gardens) in order to avoid unacceptable exposure of the occupants to residual radium contamination;
(i) The management of former landfills potentially containing radium-bearing wastes in order to avoid the dispersion of radium in the environment and uncontrolled exposure of workers during excavation work.

Management of potentially contaminated properties

The approach consists of 3 steps: i) search for all properties where radium was handled, ii) measurement and assessment of each property (indoor and outdoor), iii) remediation of affected properties where the occupant exposure exceeds the reference level of 1 mSv/year [1]. The choice of this reference level was based on several criteria that will be discussed during the presentation: in particular legislative framework, public acceptability, resources, feasibility.

The historical research identified more than 1000 properties potentially contaminated, most of which are home workplaces. By the end of 2019, 668 of these properties had been examined, of which 113 exhibited levels of exposure above the reference level, indicating the need for remediation (nearly 90% of them are nowadays used for residential purposes), and in 97 cases the remediation had already been completed.

The screening consists of a measurement of the dose rate over the entire surface of a building or outdoor areas (gardens). If the net dose rate exceeds 100 nSv/h in indoor areas, the effective dose potentially received by the most exposed occupant is assessed based on the measurement results and exposure scenarios. If the calculated effective dose exceeds 1.0 mSv/year, a remediation is necessary. Among the cases that require remediation, the effective dose rates were found to be between 1 and 10 mSv/year, except for five properties where values between 10 and 17 mSv/year were obtained. For outdoor areas, remediation is required if the concentration of radium-226 in the soil exceeds the threshold of 1000 Bq/kg. The average values measured in soil sampled from gardens requiring remediation was 26’500 Bq/kg. In one case, a concentration of around 670’000 Bq/kg was measured.

The remediation includes planning, decontamination, rehabilitation, final control and waste disposal. The objective is to reduce the effective dose below 1 mSv per year indoors and the concentration of radium in the soil outdoors below 1000 Bq/kg. If possible and feasible, the residual contamination is reduced to achieve a net dose rate of 100 nSv/h throughout (ALARA). In accordance with the Swiss legislation and under specific conditions, remediation waste with low levels of radium are disposed of in conventional incineration plants or landfills. Other radium-bearing waste has to be sent to the Federal Collection Centre for Radioactive Waste. The average remediation costs are in the order of CHF 35’000 for decontamination and CHF 15’000 for rehabilitation.

Management of former landfills potentially containing radium-bearing waste

The serendipitous discovery of contaminated wastes on a former landfill site during construction work has shown that wastes with high levels of radium and significant dose rate (up to 0.25 mSv/h in contact) can still be found today in former landfills. Therefore, a strategy has been developed to identify former landfills that may contain radium contaminated waste and classify them into 3 risk categories with specific defined measures.

In Switzerland there are more than 15,000 former landfills registered in the cadaster of polluted sites. Since it is generally not possible to determine "a priori" whether or not a former landfill contains radium-bearing waste, assumptions have been made and criteria defined to assess the likelihood of the presence of such waste in a former landfill. Each landfill was then classified into one of the 3 risk categories on the basis of the defined criteria. The purpose of this classification is in particular to identify landfills that could pose a risk to a worker’s health or the environment in case of excavation work (category 2) or could endanger groundwater quality through radium contamination (category 3). Monitoring and radiation protection measures have therefore been defined for these two highest risk categories. On the other hand, no specific measures are foreseen for the lowest risk category (category 1). For landfills that require monitoring or radiation protection measures an entry describing the radium contamination will be added to the internal cadasters of polluted sites in order to ensure the permanent availability of this information. The radiological monitoring of former landfills will be a long-term task of the FOPH.

STAKEHOLDER INVOLVMENT

Informing the public and especially those individuals directly concerned (owners and tenants) was a crucial aspect of the Action Plan. This part called for the authorities to develop an active and transparent communication plan. In addition, ensuring that administrative bodies at the cantonal and local level were regularly informed and involved was of equal importance since the success of the Action Plan substantially depended on their cooperation.

CONCLUSIONS

The Radium Action Plan is supposed to last up to 2022 until all potentially contaminated properties are found and remediated if necessary. Several former industrial sites requiring remediation based on the criteria mentioned above are also contaminated with other (non-radioactive) pollutants. Dealing with such complex cases of chemical and radiological pollution may be challenging and a longer process. The management of the former landfills will be implemented in the long term in synergy with the management of conventional (non-radioactive) polluted sites.

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317-MASUREMENT OF RADON ACTIVITY CONCENTRATION IN SOIL GAS, GROUND WATER AND INDOOR AT AL-TUWAITHA SITE

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Measurement of Indoor Radon Concentration and Assessment of Effective Dose

The largest contribution to the average effective dose results from exposure to radon and its radioactive progeny [1]. The estimate of radon concentration in various parts of Al-Tuwaitha nuclear site (3km south of the southern edge of Baghdad, the capital of Iraq) has been realized. Indoor radon radioactivity concentration in 10 locations at Al-Tuwaitha nuclear site was measured using Durridge Rad 7system . It is easy to use, portable, and very sensitive device. The effect of climate parameters on the indoor radon concentration with different season was studied. Al-Tuwaitha is characterized by a hot, arid climate with intense sunshine for much of the year. In terms of maximum temperatures, nearby Baghdad is considered one of the hottest cities in the world [2].

To estimate the annual effected dose for indoor radon and its short-lived progeny used equation (1). Which is derived from epidemiological studies and physical dosimeter from UNSCEAR 2000 Report, [1].

$$\mathrm{AED} = \mathrm{C_0} \times \mathrm{F} \times \mathrm{T} \times \mathrm{CF} \qquad (1)$$

Where: C_0: the Rn concentration in the air in Bq/m^3,F: equilibrium factor between radon and its progeny and the recommends value is 0.4.[1], T: the average indoor occupancy time for worker 480 hr/y, CF: the conversion factor 9nSv. m^3/(Bq.hr)[3,4].

Table (1) was shown in summer season, the measurements of radon concentration are ranged between (5.85±11.7- 62.0±20.9)Bq/m^3 and the highest radon concentration was found at Alalwayat shrine with a dose of exposure 0.106 mSv/y while in winter season, the measurements of radon concentration are ranged between (7.37±8.85-162±43.3)Bq/m^3 and the highest radon concentration was found at the Medication/Equipment storage with effective dose of 0.278 mSv/y due to the long-time closure of the place and the presence of medical devices. Furthermore, the radon concentration in dwellings depends on the building material and the porosity and the density of the wall material. All values were found below the recommended value

TABLE 1.INDOOR RADON CONCENTRATIN AND DOSE RATE DURRING DEFFERENT SEASON

No.	Location	Indoor radon con. (Bq/m^3)	Dose rate (mSv/y)	Date	Temp. of room (°C)
1	Radiological and Nuclear Safety Directorate/Equipment storage	32.5 ±10.2	0.056	4/7/2019	34.1 summer
		115±39.9	0.198	10/2/2019	21.3 winter
2	Central Laboratories Directorate/sample preparation room	44.3±26.6	0.076	8/7/2019	34 summer
		64.9±4.82	0.112	11/2/2019	18.3 winter
3	Radiological and Nuclear Safety Directorate/sample preparation room	5.85 ±1.7	0.010	7/7/2019	34 summer
		20.7 ±7.62	0.034	12/2/2019	18.4 winter
4	Medication / Equipment storage	7.08±4.94	0.012	13/7/2019	33.5 summer
		162±43.3	0.278	13/2/2019	20.4 winter
5	Nuclear applications Directorate / storage	26.6±3.14	0.044	15/7/2019	34 summer
		62.0±24.4	0.106	18/2/2019	19.7 winter
6	Central Library / Basement	23.6±5.9	0.04	25/8/2019	34 summer
		14.8±7.62	0.024	14/2/2019	18.8 winter
7	Alalwayat shrine	62.0±20.9	0.106	26/8/2019	34 summer
		29.5±8.34	0.050	20/2/2019	20 winter
8	Decommissioning Directorate / equipment storage	5.9±0.0	0.010	27/8/2019	34 summer

		7.37±8.85	0.012	19/2/2019	20 winter
9	Managementand Treatment of Radioactive Waste	18.9±12.8	0.032	28/8/2019	34 summer
	Directorate / meeting room	42.8±26.6	0.072	21/2/2019	20 winter
10	Site Support Directorate / Technical Affairs	23.6±17.4	0.040	29/8/2019	32 summer
		20.7±3.41	0.034	17/2/2019	19.3 winter

Radon in groundwater

The second part of the present study deals with the monitoring and measurement of radon concentration in the ten wells at Al-Tuwaitha nuclear site through 2019, using a RAD7 system electronic radon meter with WAT250 mode. The results are presented in Table 2 for the building of concern , radon concentration with the calculated errors deviations, and borehole depth.All measurements are ranged between(1020-608 Bq/m^3) while the highest radon concentration was found at well 9 with depth of 20 m (regional zone) near the Radioactive Waste Management and Treatment Directorate and the lowest value was found at well 1,2,8, and 10 with depth 15 m (shallow ground water zone) near TAMMUZ-2 reactor, fuel processing facility, Russian Isotope Production Facility, and Russian Silos respectively. The radon concentration values were corrected to the other parameters as temperature, and relative humidity using capture software. Furthermore, all values were found below the recommended value by the USEPA (which is 11000 Bq/m^3)[5].

TABLE 2. RADON CONCENTRATIN IN GROUND WATER AT AL-TUWAITHA SITE

Monitoring Well	Building of concern	radon con. (Bq/m^3)	Borehole depth (m, bgs)
Well 1	Building 24 (TAMMUZ-2 reactor)	608±152	15
Well 2	Building 73 (fuel processing facility)	608±151	15
Well 3	Building 80 (nuclear physics laboratory)	734±152	26
Well 4	Assumed background area	734±214	27
Well 5	Building 13 (IRT-5000 reactor)	714±38	15
Well 6	Building 9 (radiochemistry laboratory)	830±304	45
Well 7	STP(Sewage Treatment Pond)	734±76	25
Well 8	Building 15B (Russian Isotope Production Facility)	608±37.8	15
Well 9	Building 35 (RWTS)	1020±484	20
Well 10	Building 40 (Russian Silos)	608±108	15

Radon gas in soil

The third part of the present study was concerned with monitoring radon gas concentration in the soil. The measurements of radon in soil were carried out at eight selected location as was shown in Table 3 .Which were taken from (25-30)cm depth across Al-Tuwaitha nuclear site, using a RAD7 system electronic radon meter with grab mode to generate a radon distribution map. The radon gas concentration in soil was ranged from 248 ± 58 Bq/m^3 at medication/ right side to 602 ± 50.4 Bq/m^3 near the Alalwayat shrine with mean value of 420.5 Bq/m^3. The results were indicated that the soil of Al-Tuwaitha site within the range reported by other investigators and without posing any health hazard.

TABLE 3. RADON CONCENTRATIN IN SOIL GAS AT AL-TUWAITHA SITE

No.	Monitoring region	Radon con. (Bq/m^3)	Temp. (°C)	Location (GPS)
1	Near the Radiation and Nuclear Safety Directorate	567±49	25.2	N 33.20652 & E 44.51349
2	Medication/ right side	248±58	34.6	N 33.20872 & E 44.51199
3	In front of the designs building	275±124	36.6	N 33.20912 & E 44.51144
4	Management and Treatment of Radioactive Waste Directorate/ beside stationary	274±25.5	34.2	N 33.20210 & E 44.51743
5	Behind the Central Laboratories Directorate	353±30.5	32.6	N 33.20750 & E 44.51207
6	Near the Alalwayat shrine	602±50.4	25.1	N 33.20345 & E 44.50884

7	In front of the Italian fuel building	397±53.4	27.5	N 33.20043 & E 44.51211
8	Near the Scientific Information Center (Central Library)	452±93	25.8	N 33.20998 & E 44.51575

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318-ASSESSMENT OF INDOOR RADON CONCENTRATIONS IN HOMES AND WORKPLACES IN YANGON REGION, MYANMAR USING RAD7 RADON DETECTOR

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INTRODUCTION

The radioactive radon gas is released in rock, soil, and water and can build up to create dangerous levels inside homes and work places. The indoor air concentration of radon was recognized as a radiation health hazard, potentially causing an increase in the incidence of lung cancer. The lower the radon concentration in a home, the lower the risk of lung cancer as there is no known threshold below which radon exposure carries no risk. Myanmar is making effort to improve radiation safety for the purpose of the protection of people and the environment from the harmful effects of ionizing radiation. Division of Atomic Energy (DAE) was established under the Ministry of Education to carry out research, development and training in the field of atomic energy and to ensure the safety of the radiation sources and the protection from nuclear radiation hazards.

METHOD

The experiment is the first study to assess the indoor radon concentrations in homes, monastery, dhamaryone and workplaces such as office rooms, classrooms and laboratories in Yangon Region for the radiation safety purpose. The indoor radon concentrations were measured by using DURRIDGE RAD7 electronic radon detector which is provided by the International Atomic Energy Agency (IAEA) to DAE. The measurement points were set one meter above the ground and in the center of the room. The measurement period for each room was half day in which a cycle time was set to be 30 min for 24 cycles. Fig. 1 shows the experimental setup for the indoor radon measurement in the class room using RAD7 radon detector and the measured alpha energy spectrum for maximum radon value.

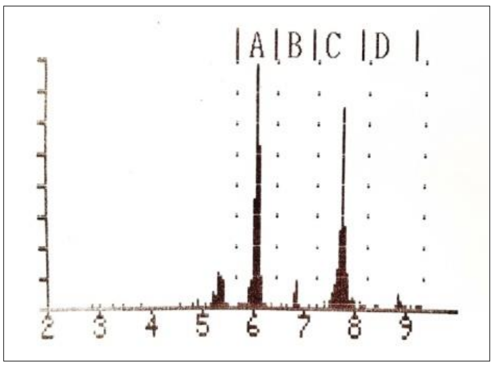

FIG. 41. The indoor radon measurement in the class room using RAD7 detector and the measured alpha energy spectrum for maximum radon value.

RESULTS AND DISCUSSION

The assessment of indoor radon concentration in the study buildings are shown in Table 1. The maximum and minimum values are 30.5 Bq/m^3 and 1.25 Bq/m^3 respectively. All concentration values are less than the lower limit of the recommended ranged 200-300 Bq/m^3 by The International Commission on Radiological Protection (ICRP, 2009). The annual effective doses received by the residents of the study area were calculated by using the equation introduced by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR, 2000). The calculated maximum and minimum annual effective doses are 0.77 mSv and 0.03 mSv respectively as shown in Table 1.

TABLE 1. INDOOR RADON CONCENTRATION AND ANNUAL EFFECTIVE DOSE ASSESSMENT

Sample code	Sample location	Indoor radon concentration (Bq/m^3)	Annual effective dose (mSv)
H1	G floor	5.89	0.15
H2	G floor	3.33	0.08
M1	1st floor	22.6	0.57
M2	1st floor	26.3	0.66
M3	1st floor	22.4	0.56
M4	2nd floor	15.8	0.40
M5	1st floor	4.77	0.12
M6	G floor	8.65	0.22
M7	G floor (Hall)	1.93	0.05
D1	G floor (Hall)	1.25	0.03
O1	G floor	10.0	0.25
S2	1st floor	6.32	0.16
S1	G floor	30.5	0.77
L1	G floor	17.5	0.44
L2	G floor	8.83	0.22

CONCLUSION

Radon concentrations in homes, monastery, dhamaryone and workplaces such as office rooms, classrooms and laboratories are well below the recommended safe limit values. The two lowest values 1.25 Bq/m^3 and 1.93 Bq/m^3T were recorded because these places were open and with high ventilation. The measurements revealed that the indoor radon concentrations depend on the building materials, dimension of the room, ventilation in the room and elevation from the ground. The annual effective dose values show that there is no significance risk to the human beings due to the presence of natural radon in the homes and workplaces.

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319-TRAINING NUCLEAR REGULATORS IN CRITICALITY SAFETY

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Criticality safety is often considered by inspectors' and dutyholders' alike as a specialist area where detailed expertise is required. Whilst it is true in that it requires underpinning by such specialists, criticality safety knowledge is required at many levels in order to achieve safe performance on nuclear sites. For nuclear regulators, this will range from a basic awareness amongst all its inspectors, to significant knowledge and experience by their experts.

The presentation provides an overview of the new criticality safety training delivered to inspectors at the United Kingdom (UK) Office for Nuclear Regulation (ONR). This training has recently been re-developed to provide a more targeted and proportionate level of training, commensurate to the level of understanding and knowledge required of the wide range of roles undertaken by regulatory staff. It has also been developed to include a range of different teaching techniques to improve the learning experience of the participants and overall training effectiveness.

An overview of the revised course contents and structures is given in the presentation, together with examples of some of the teaching techniques used. These include techniques that encourage participants to physically move around the area the training is delivered, to undertake various activities designed to engage different modes of learning, and techniques that make use of different mediums to convey the training material. How this developed form of training has enhanced ONR's regulatory activities is discussed along with the benefits this brings to the dutyholders that ONR regulates.

320-CONSERVATISM IN DOSE ASSESSMENT AND MODELLING WHEN CALCULATING CRITICALITY RISKS

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The poster presentation provides an overview and key findings of work undertaken by the United Kingdom (UK) nuclear regulator into the use of conservatisms that can be applied in criticality safety cases to estimate criticality risks, and the potential implications of these findings for the regulation and management of criticality safety.

When calculating the risk of criticality for fissile systems, conservative bounding assumptions must be made with regards those parameters that affect the reactivity of such fissile systems. Dutyholders balance these assumptions against operational requirements such that they are not overly restrictive and, for example, do not significantly hinder operations from taking place or disproportionately increase other risks. The Office for Nuclear Regulation (ONR), acting as the independent nuclear regulator across the UK, form regulatory judgements regarding dutyholders' safety cases and their use of these conservative assumptions.

An overview of two examples of conservatisms that may be used in UK criticality safety cases to support the estimate of criticality risk are presented. These examples include work undertaken on behalf of ONR in the form of research papers into the use of 'burn-up credit' [1] (used when assessing the criticality risk of spent nuclear fuel) and consideration of the effect temperature has on the criticality risk of fissile systems [2]. Both research papers are published on the ONR website.

Burn-up credit concerns the concept of taking credit for the reduction in reactivity (k_{eff}) associated with irradiation ('burn-up') of nuclear fuel during operation of a nuclear reactor. Credit for it has typically not been claimed in UK criticality safety cases, with a 'fresh fuel' approach generally being used. This is a conservative approach to take when considering criticality safety, but it is anticipated that in future taking credit for burn-up will be necessary (e.g. for transport, storage, or disposal of spent nuclear fuel). The research paper [1] considers burn-up credit methodologies and data currently used around the world, and presents a set of 53 Regulator Questions (RQ) that enable regulatory attention to be proportionately targeted when considering criticality safety cases that make use of burn-up credit. This can help form a regulatory judgement as to the appropriateness of a dutyholders application of burn-up credit and whether it remains suitably conservative.

Detail on a number of these RQ is presented, including the axial and horizontal variation of fuel burn-up, how the presence of neutron absorbing materials can affect burn-up, and how the cooling period of the spent nuclear fuel can affect the credit taken for burn-up. These topics are discussed and further specific RQ that can be asked are highlighted.

Temperature is one of a number of physical factors that can affect the neutronic behaviour of nuclear systems. In particular, the neutron interactions elements can experience can be a function of temperature, which in turn can affect the k_{eff} of fissile systems and hence whether the system is critical or not. The k_{eff} for these fissile systems may increase or decrease with varying temperature and until recently it has not been practicable for dutyholders to assess these temperature effects within their criticality calculations. However, the latest codes and nuclear data libraries now allow criticality calculations across a range of temperatures, which is discussed within the research paper [2]. This research helps form a regulatory judgement as to the appropriateness of a dutyholders consideration of temperature within their criticality safety case and whether it remains suitably conservative.

A summary of the nuclear processes that are affected by temperature and how these can influence the k_{eff} of a fissile system are presented. A range of generalised fissile systems and how the k_{eff} of these systems is affected with varying temperature is discussed, together with how these findings can be used to proportionately target those areas of most importance within a criticality safety case.

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321-STUDY OF CALIBRATION THE DOSEMETER FOR MEASURING EYE LENS DOSE IN TERMS OF $H_P(3)$ WITH DIFFERENT PHANTOM

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OBJECTIVE

To calibrate the dosemeter for measuring eye lens dose in terms of $H_P(3)$ and study the performance of the dosemeter, such as batch homogeneity, detection threshold, linearity, energy response, isotropy.

METHOODS

The standard for calibration is based on ISO 12794(2000), phantoms are slab phantom and cylindrical phantom, radiation quality is the narrow beam X-ray (N60~N120) taken from ISO 4037.

RESULTS

The performance of the dosemeter, as batch homogeneity, the coefficient of variation is 5.7%.The detection threshold is less than 0.1 mSv.As linearity, for two phantoms, response are both less than 10%.The energy responses to both phantoms, are in the range of 20%~40%.As isotropy, mean value of the response differ from the response of normal incidence is (1.05±0.03) for the slab phantom, and (0.99±0.08) for the cylindrical phantom.

CONCLUSION

For selected experimental conditions, both phantoms are suitable for eye lens dosemeter calibration, the calibration results conform to the requirements of ISO 12794, the dosemeter is suitable for using in interventional radiology.

Key words: Eye Lens Dose ; Thermoluminescent Dosemeter ;
Conversion Coefficients ; Calibration

322-ASSESSMENT OF THE METHODS FOR ESTIMATIONS OF THE RADIOACTIVE SURFACE CONTAMINATION IN THE LATIN AMERICAN AND CARIBBEAN REGION

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INTRODUCTION

Radioactive surface contamination is specified in terms of activity per unit area and the limits are based on the international recommendations. The limits are radionuclide-specific, since the number of the calibration sources are limited then in many case the specific calibration factors are not available for the monitors used in the direct measurements. It was identified the needs on improvement of the methods for evaluation of surface contamination in the Latin American and Caribbean region. The International Atomic Energy Agency (IAEA) have supported the efforts for improvements of the radioactivity measurements of the contamination through the regional projects RLA 9066, RLA 9075 and RLA 9085. The article presents the results of two proficiency tests run in the 2013-2019 period and coordinated by the Center for Radiation Protection and Hygiene (CPHR).

MATERIAL AND METHODS

The assessment investigation was organized in two steps. In the first step it was organised a proficiency test to assess the methods used in the region for estimation of the radioactive surface contamination. In the test participated 15 institutions from Argentina, Colombia, Costa Rica, Cuba, Ecuador, El Salvador, Guatemala, Honduras, Nicaragua, Paraguay, Perú and Uruguay. Three testing plane sources with unknown activity were measured by participants to determine the activity per unit area by the methods accepted by regulations in each country. It were measured the testing sources of ^{36}Cl and ^{241}Am, traceable to North-American primary laboratory NIST with beta and alpha emissions, respectively, and the ^{131}I with complex decay scheme of beta and photon emissions, prepared by Center of Isotopes (CENTIS) and traceable to German primary laboratory PTB. The results were compared with the limits of acceptable uncertainty stated in ICRP 75 [1] for those measurements in the workplace.

In the second step, it was planned a trainning course on evaluation of the surface contamination based on the ISO 7503 standard [2]. In the course participated 21 institutions from Argentina, Bolivia, Brazil, Chile, Costa Rica, Cuba, Dominican Republic, Ecuador, El Salvador, Guatemala, Honduras, Nicaragua, Paraguay, Perú, Uruguay and Venezuela. Two testing plane sources of ^{131}I and ^{125}I with unknown activity were prepared by CENTIS with traceability to Cuban national standards. The participant were separated in 4 groups and were asked to estimate the activity per unit area of the testing sources applying ISO 7503 method. The results were also compared with the limits of acceptable uncertainty stated by ICRP 75.

RESULTS AND DISCUSSION

It was plotted the trumpet curve with the upper and lower limits of the acceptable uncertainty for each measured source in the proficiency test [3]. Only eight participants were able to evaluate the surface contamination of ^{131}I because of the complex scheme of decay and the lack of the specific calibration factors for the monitors they operated. In the figure 1a is shown the results, six participants were able to evaluate satisfactorely the contamination. Three of those six participants uses the estimated calibration factor recommended by the CPHR laboratory based on ISO 7503. The good results in general are considered associated to the level of the surface contamination around 1,4 Bq·cm^{-2} that correspond to the level of records where expanded uncertainties are accepted of ± 100 %. The same considerations were applied to the results for ^{36}Cl shown in the figure 1b where only two participants were outliers. Following the same concepts for the test with ^{241}Am, shown in the figure 1c, the results were compared to the acceptable uncertainty but for activities per unit of area around 3,6 Bq·cm^{-2} that are far from the derived limits for removable contamination of 0,04 Bq·cm^{-2} in the case of ^{241}Am [4]. In this region the acceptable limits are of – 33 % and + 50% related to conventional true value. Only two results were satisfactories and correspond to the participants that uses ISO 7503 calibration factors recommended by CPHR laboratory.

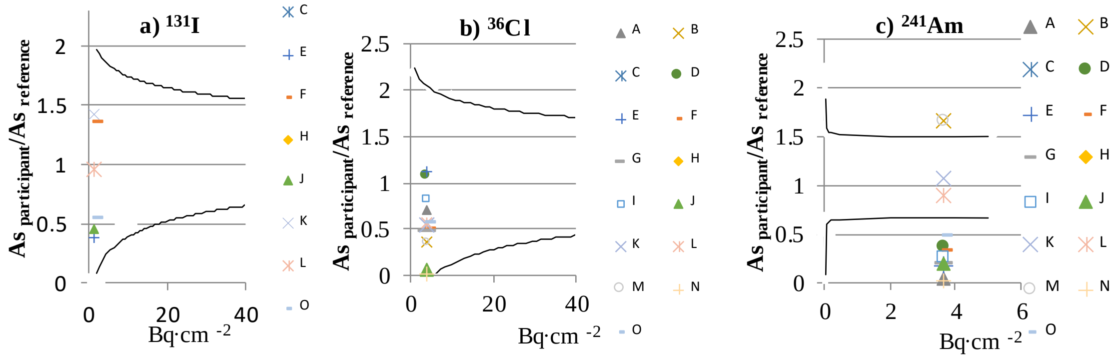

FIG. 42. Results of the proficiency test on estimation of radioactive surface contaminations using three radionuclides. The ratio of the activity per unit area reported by participant and reference is represented as $As_{participant}/As_{reference}$.

The ISO 7503 can be implemented by the end-users using the efficiencies or factors provided by the laboratories for some calibration sources that, in many case, are not the same as the radionuclides to be measured. In the second step test, 21 participants were trainned on the mentioned methods and they determined the activity per unit area of ^{131}I and ^{125}I plane sources. Only one from eight results was satisfactory for one group in direct measurements of ^{125}I source. The activity per unit of area of the sources were prepared around 18 kBq·cm^{-2} where acceptable limits are well defined (- 33% and + 50 %). The main problems encountered were the mistakes in applying the novel ISO 7503 method.

CONCLUSIONS

The methods for estimation of the radioactive surface contamination in the region have to be improved. The best results are mainly found when are employed the recommended factors by a laboratory using the ISO 7503 method. For direct implementation of the ISO 7503 methods by the end – users is needed more trainning.

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324-STATUS OF MANAGEMENT ON IMPORTING MATERIAL CONTAINING NATURALLY OCCURRING RADIOACTIVE MATERIAL(NORM) IN REPUBLIC OF KOREA

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Along with industrial development, the usage of material containing radionuclides of natural origin has increased. With the increase of naturally occurring radioactive material(hereinafter referred to as "NORM") and its product, related radiation risks have also increased. In order to prevent the problems arising from such risks, 「The Act on Protective Action Guidelines against Radiation in the Natural Environment」 (hereinafter referred to as "the Act") has been implemented in 2011. Materials containing NORM are classified as either raw materials, residues or products in the act. The registration level for raw materials, the same as the IAEA level, is K-40 10 Bq/g, 10,000kBq/yr, and that for uranium and thorium and their progenies is 1 Bq/g, 1000kBq/yr.

Regular inspection became a new requirement in 2019 with the revision of the act to reinforce safety management of natural environmental radiation. Handlers of raw materials and residues or products manufacturers (hereinafter referred to as "registrants") and recycled scrap metal handlers are subject to regular inspections. The regular inspection period is determined from 1 to 3 years, considering the quantity handled or the number of radiation monitors operated. Regular inspections were conducted on 51 companies with a one-year periodic inspection cycle, of which 47 were raw materials handlers, 1 residues handler, and 3 recycled scrap metal handlers. For the registrants, investigations are conducted on matters necessary for safety management, such as the distribution status of handled substances, the manufacturing or import/export status of processed products, and the degree of radioactive contamination around operating facilities. For recycled scrap metal handlers, compliance with operating standards, completion of training, record storage, and safety measures for suspicious materials, is checked.

TABLE 1. Status of operation of radiation monitoring at airport and sea port

as of December 2019

	Airport	Sea port	Sum
Number	14	114	128
Location	Incheon(13), Gimpo(1)	Busan(39), Incheon(28), Gunsan(9), the others(38)	

In order to monitor NORM in raw materials, products, and recycled scrap metal, radiation monitors are installed at airports and sea ports and entrusted to the airport and sea port operators for operation. In addition, not only airports and sea ports, but also recyclable scrap handlers who operate electric melting facilities with a unit capacity of 30 tons or more install and operate radiation monitors.

The process of importing NORM is as follows. Cargoes are monitored by port operators through the radiation monitoring system to identify whether their radiation dose rate exceeds the natural radiation dose rate or not. If an alarm occurs in the radiation detector, (1) the cargo is isolated, (2) detailed investigation and analysis are conducted, (3) an action order such as return to the importer is given. If radiation above a certain level is detected, measures such as confirmation of the cargo and the owner are taken. Based on the collected information of the alarm and the cargo, it is checked whether the subject is needed to be registered as raw materials / residues / products or not. The owner of the alarmed cargo is informed that the cargo needs to be returned to the original country or disposed of.

Products containing NORM are in some cases imported into the Republic of Korea. For example, the products include furnace dust, aluminium scrap, timber, zinc skimming, and zinc oxide containing Cs-137 and Th-232. In the case of recycled scrap metal, a radiation test result(radiation-free certificate) is requested in advance from the exporting country, and if not submitted, it is returned.

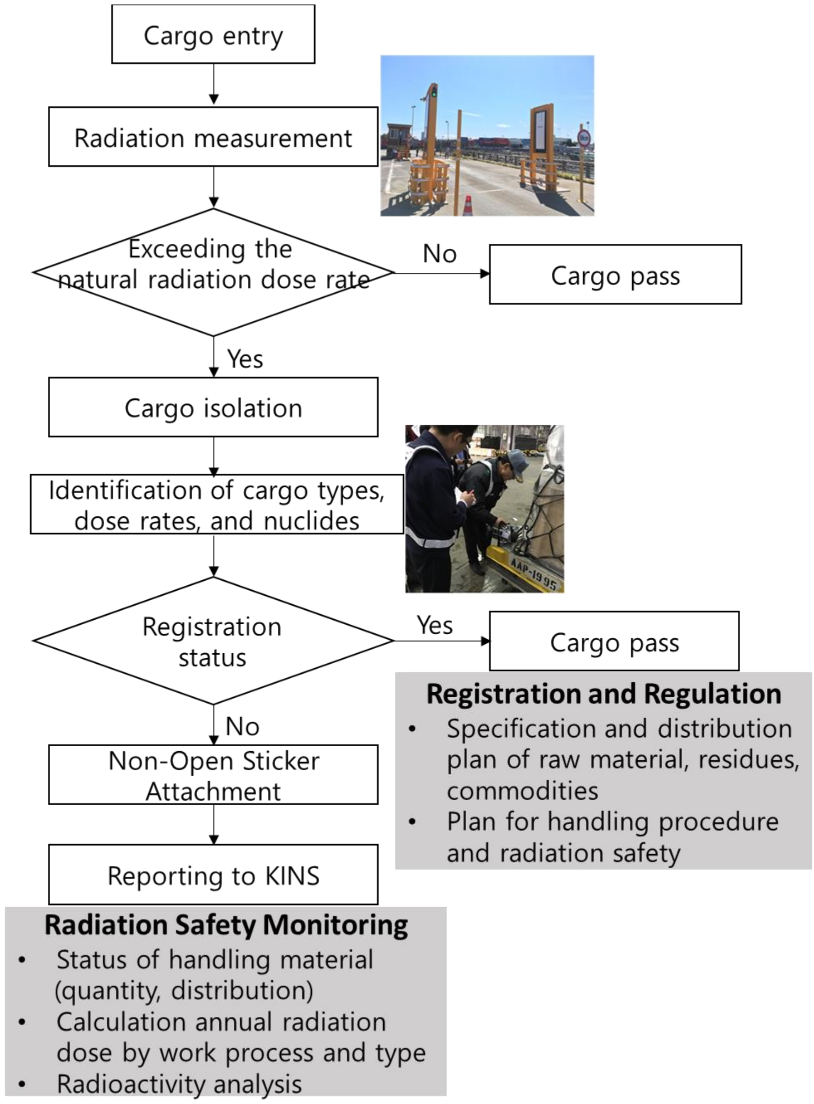

FIG. 43. Process for radiation monitoring at airports and sea ports

Recently, the Act was amended to prohibit the use of NORM at all for products worn or adhered to the body. Therefore, it is expected that radiation issues caused by products are expected to be decreased in the country. However, since there are no international trade guidelines for radioactive materials, many issues are expected to arise in trade due to different levels, standards, and positions between exporting and importing countries. Procedures, such as the issuance of no-radiation certificate, will not be problematic when it is required by the both sides.

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325-MYANMAR'S EFFORT TO SUSTAIN CAPACITY BUILDING AND PROMOTING SAFETY CULTURE IN RADIAITON PROTECTION

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Abstract

The widespread applications of radioactive materials and radiation generating equipment in the field of industry, medicine, agriculture and research in Myanmar necessitated establishment of an efficient regulatory framework and consequently the Division of Atomic Energy (DAE) constituted to exercise regulatory control over the safe usage of the radioactive materials and the radiation generating equipment.

Myanmar has gone through many improvements since 2016. Myanmar recently acceded to the Convention on Nuclear Safety (CNS) and Convention on the Physical Protection of Nuclear Material (CPPNM) and its amendment on 6th December 2016 [1]. Furthermore, Myanmar has expressed a political commitment with regards to the Code of Conduct on the Safety and Security of Radioactive Sources.

As a legislation issue in Myanmar, Division of Atomic Energy (DAE) under the Ministry of Education (MOE), promulgated Atomic Energy law on 8th July 1998 for safe use of radiation source in the country. It was mainly based on radiation safety and does not cover nuclear safety, nuclear security and safeguards (3S strategies). In order to strengthen its national nuclear related legislation, DAE has just recently completed the drafting of the Myanmar Nuclear Law that prohibits the use, production, storage, distribution and import/export of nuclear and other radioactive materials without government license. [2]

As a country with no nuclear power plants, the nuclear safety concerns in Myanmar are small compared many other developed countries. There is no radioisotope production facility in Myanmar and radioisotopes are still imported from other countries. The utilization of radiation sources and irradiating apparatus are limited to the use in medicine, industry, agriculture, education and training, livestock breeding and research.

The DAE acts as the Regulatory Body under the Atomic Energy Law and is responsible for all aspects of control, security and safe management of radioactive materials and radiation apparatus using in Myanmar. All new radioactive sources and Radiation Apparatus imported to Myanmar are taken the Prior Permission first. The DAE issued the prior permission, registration, user license and re-export license step by step. During the licensing process and inspection activities the DAE intends to assess both safety and security aspects at the same time. All of the inspection records and issued documents are recorded. [3]

Safety and security of such sources are major concerns for regulatory authorities and licensees. Spent radioisotopes from medical and industrial sectors are sent back to the country of origin. Thus, safety problem related to spent radioisotopes and radioactive waste in Myanmar is very low. DAE is using Regulatory Authority Information System (RAIS) since 1998 and now using RAIS 3.3. DAE Collect the list of the private clinics, hospitals, industries with their radioactive source and radiation apparatus by the help of relevant Ministries. [3]

Establishment of recording and reporting of incidents to regulatory authorities will come soon. Reporting systems for medical radiation incidents become Mandatory reporting as part of regulation. Document procedure

for investigation of medical radiation incidents including sharing information between DAE and Ministry of Health and Sports (MOHS) will be established soon.

Being a developing country, Myanmar people are unfamiliar with the effective risk communication, radiation safety culture and system of radiological protection. Users of radiation sources need to submit storage design and emergency procedures when they apply for licence. Users of radiation apparatus also need to submit room layout plans for their facilities. Inspectors discuss with users to help them for better radiation safety designs before the start of construction stage.

Inspectors from DAE are implementing safety culture in every practices and facilities which they inspect. Their safety points are for patients, radiation workers, public and environment. They measured the radiation levels inside and around facilities with instruments provided by IAEA to ensure the safety as described in the design's planned exposure. Then make open discussion with stakeholders about findings on radiation safety of their facility. If something is needed to modify or change to meet the required safety level, they discuss immediately with operators, owners about radiation safety remedies and precautions. OSLD badge wearing practice is also implementing to radiation workers by DAE inspectors for their safety and reading of exposure records from individual badge are issued regularly to users. These are also on the checklist of inspectors to take records.

DAE is giving radiation protection training for radiographers from Government Hospitals every year. They discuss about safety practices in the use of radiation, safe transport of radioactive materials and emergency response. DAE is also giving awareness raising discussion classes in firefighter officer trainings and police inspector training schools each year to implement radiation safety culture between frontline officers by working together with relevant ministries. In Myanmar, well developed and implemented safety culture is essential. To achieve that, DAE in collaboration with stakeholders, has been conducting workshops and trainings to exchange information, knowledge of radiation safety, experience and good practices on the system of radiation protection at the national level and regional levels with the help of IAEA and other donors such as EU, USDOE.

There is also increasing of new facilities in radiation technology and more inspectors need to fulfill the ever increasing workloads. Capacity building is need for new inspectors even the senior inspectors share their knowledge through on job trainings and to them. From these hands on trainings, new inspectors get chances to use of actual operating equipment and physically experience the complex scientific concepts and behavior of real-world usage whilst handling their actual inspection devices. Inspectors from DAE are also trying to do capacity building by self-learning from various resources and testing locally available materials to apply the IAEA guidelines for radiation safety in Myanmar.

Nowadays, technology is ever changing in every sector. For example, Linear Accelerators in cancer treatment, container scanners in transport sectors, etc. To follow with developing technology, inspectors need to upgrade their skills in inspections for the new things by the help of capacity building program. It can also help to ensure the radiation safety in their inspections. IAEA is helping DAE to accelerate in capacity building by giving appropriate trainings for inspection of specific practices and also support new instruments for inspections such as neutron detectors to inspect linear accelerators.

This paper will present an overview of the sustaining capacity building efforts to fulfill the demands of radiation safety in various sectors in Myanmar. It is not only the responsibility of government but also the keen participation from relevant organizations and stakeholders are needed to accomplish. Radiation safety culture needs to start from practice in everyday life activities of stakeholders around the world.

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326-ASSESSMENT OF RADIOLOGICAL IMPACT FOR ENSURING RADIATION SAFETY FOLLOWING LARGE SCALE NUCLEAR AND RADIOLOGICAL EMERGENCIES

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Inspite of all safety measures adopted and implemented during the design and operation of nuclear facilities like nuclear power plants(NPPs) and reprocessing facilities and for the radioisotopes during its manufacture, transport, storage and usage etc, possibility of major accidents having potential for release of large quantity of radioactivity to the environment cannot be completely ruled out. The experience gained during the response to nuclear accidents like Chernobyl and Fukushima and many radiological accidents has helped the international community to upgrade their emergency preparedness to enable reduction of radiological consequences significantly.

Following release of large quantity of radioactivity into the environment, if response actions and protection measures are not implemented, it can lead to exposure of public to significant level and radiological consequences. Generic criteria for doses received for which protective actions and other response actions are to be undertaken to avoid severe deterministic effects and to reduce the risk of stochastic effects and the guidance values for restricting exposure of emergency workers are given by IAEA in GSR part3[1]. Following large scale radioactive releases to environment, for effectively managing radiation exposure to the emergency workers/response teams, field monitoring teams, members of the public and environment, quick assessment of environmental radiological status is essential.

To meet the criteria and requirements in line with GSR part3 and GSR part7[2], development of capability for quick assessment of radiological impact following any major nuclear or radiological accidents is an important requirement. The quick assessment of environmental radioactivity levels, in application of OILs will enable the early implementation of protective actions and hence the reduction of radiological consequences.

The presentation is aimed at identifying and suggesting the combination of systems to be developed and maintained by member states for quick assessment of radiological status during any nuclear or radiological emergencies and for the upgradation of preparedness for response to large scale nuclear and radiological accidents or threats. These systems will help in the assessment of possible radiation exposures to the public and emergency responders and enable in the timely implementation of emergency response for reducing the radiological consequences ensuring radiological safety in public domain during emergency exposure situations. The capability developed by the member states can also be useful for offering assistance to other nations who may be undergoing a radiation emergency situation including a transboundary emergency.

NPPs, reprocessing facilities and irradiators have large quantity of radioactivity within them; but very well contained with security and safety ensured. Though probability is extremely low, severe accidents in these facilities or malicious acts can lead to release of large quantity of radioactivity to the environment. Depending on the accident scenario and integrity of the containment, quantity and quality of radioactivity released to the environment and period of releases etc can vary significantly. Topography of the site to which they are released, elevation of releases, meteorological conditions during the release and its transportation etc can significantly influence the environmental radioactive contamination and radiological impact on the public. In addition to accidents, malicious acts including dispersal of radioactive materials following explosion of Radiological Dispersal Devices(RDD) or Improvised Nuclear Device(IND) also can lead to significant level of radiation exposure to the public and environmental contamination depending on the conditions in which the malicious acts are carried out.

Identifying all potential accidents and malicious acts leading to nuclear and radiological emergency conditions including low probable ones, the following are to be considered for national level preparedness for emergency response.

(a) Systems and methodology for quick assessment of radioactive releases from NPPs.
   (i) Systems for the prediction of potential radioactive releases based on the accident scenario and plant parameters;
   (ii) Decision support system comprising of large number of radiation detectors around the plant, meteorological stations, dispersion prediction models to enable quick estimation of released radioactivity(at ground level, elevated or combination of both)[3].
(b) Systems for the prediction of airborne radioactivity and environmental ground deposition for long downwind distances using dispersion modelling and prevailing as well as anticipated meteorological conditions.

(c) Radiation monitoring Systems and methodology for quick qualitative and quantitative assessment of large scale environmental radioactive contamination (ground vehicle based monitoring, aerial gamma spectrometric survey as well as drone based system).
   (i) Aerial Gamma Spectrometry System (AGSS) to be installed in aircrafts for aerial radiation monitoring[4] for quick assessment of large area radioactive contamination including identification of radionuclides following radioactive releases and for searching of radioactive sources or hotspots –can help to quickly identify and confirm the area for implementation of protective actions including Iodine thyroid blocking and other response actions;
   (ii) Quad Rotor based Aerial Radiation Monitoring System (QUARMS)[5] for remote aerial surveys (especially for RDD and IND emergency scenario) for identification of contaminated area, locate and mark area of high radiation levels, search of 'hotspots' and orphan sources;
   (iii) Mobile Radiological Impact Assessment Laboratory (M-RIAL) installed with state of the art radiation monitoring systems with online computation and data transfer capability for detailed monitoring of large area around the nuclear facilities.

(d) Network of many Radiation Emergency Response Centres spread over the country with trained Emergency Response Teams (ERTs) equipped with various types of monitoring systems, protective equipment and other supporting labs and facilities as the back bone of the national emergency management system.
   (i) Dosimetry (external, internal) for quick assessment of exposure to the affected persons including large number of public;
   (ii) Biodosimetry laboratories having capability to cater into large number of blood samples from public in case of high level exposure cases in public (for radiological accidents, RDD and IND emergencies);
   (iii) Assessment of internal contamination of large number of personnel from public domain to be carried out using portable monitoring systems and Quick Scan type Whole Body Counting System in a short period of time. Quick whole body scanners for internal dosimetry and Bioassay for quick assessment of intake in public domain;
   (iv) Facilities for Medical management of radiation injuries and radioactive contamination (external and internal) of victims.

Having the above systems developed, calibrated, tested and maintained, periodic nuclear and radiological emergency exercises including field and aerial radiation monitoring surveys, environmental sampling, usage of dispersion models and impact assessment, dosimetric support etc are to be conducted to demonstrate that arrangements are in place for managing exposure to the Emergency Response Teams(ERTs), members of the public and for adopting optimised protection strategies.

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327-RADIATION SAFETY AND PROPORTIONATE RISK MANAGEMENT

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The system of radiological protection, as developed by ICRP and implemented through IAEA standards, has a long and distinguished history in providing a balanced approach to radiation safety, both in its design and its implementation. It has remained in place in much the same form since the publication of ICRP Publication 26 in 1977 [1]. According to this, both individual and utilitarian interests and perspectives are addressed through a process of constrained optimization, with optimization intended to lead to the most benefit to the most people, and constraints being operative to limit the degree of inequity among the individuals exposed. In 2007, ICRP Publication 103 [2] introduced the concept of Reference Levels, which provide for greater flexibility in the context of existing, as opposed to planned, exposure situations. Publication 103 also introduced an approach for developing a framework to demonstrate radiological protection of the environment, an issue which had not been explicit in previous ICRP recommendations. In 2016, ICRP Publication 138 [3] confirmed and re-emphasized the ethical foundations of the system of radiological protection.

Alongside these developments in recommendations, there have been continued technical developments and substantially expanded uses of ionising radiation. For example, nuclear power programs have expanded significantly, while a significant number of the older nuclear facilities and related sites and areas have fallen due for decommissioning and/or environmental improvement. Another major area involving a huge increase in use of ionising radiation is the medical field, both for humans and in veterinary services.

Nuclear power and medical uses of ionising radiation both bring very significant benefits to society as well as presenting risks to human health and the environment. The effective management of these risks, using a graded approach [4], is crucial to confidence in continued application of both of these areas of technology. However, while the potential for harm, in terms of the effects of ionising radiation on living organisms, is similar, the contexts in which exposures arise or could arise are very different. In addition, they are viewed very differently by stakeholders. Furthermore, regulation of radiation safety has evolved somewhat independently of other hazards. Questions therefore arise as to whether radiation risks are managed proportionately within and between each area, and in proportion to other risks, such as chemical, biological and physical, arising in connection with use of those technologies.

This paper will consider these questions with a focus on two areas: a) nuclear decommissioning and environmental improvement and b) medical applications. The scope will be limited to planned and existing exposure situations.

These examples have been chosen because:

— the application of the system of radiological protection has important implications for risk management but in two very different areas, technologically and in their risk-benefit profiles;

— both areas involve significant levels of radiation exposure at the individual and collective levels;
— both areas are also associated with a significant range of other non-radiological risks;
— both areas are complex technically and socially, and therefore generate a significant challenge to optimisation of risk management, economic and social factors being taken into account.

The basis for discussion will include reference to on-ongoing international activities, e.g. in ICRP, NEA and IAEA programmes, drawing on a broad range of technical and cultural perspectives, as well as examples of how international recommendations and guidance are applied in practice, e.g. [5], [6], [7].

The purpose in addressing these questions is to generate a discussion as to whether differences in approach makes reasonable sense and are proportionate, and whether there is scope for improvements that support a more beneficial, or at least a more informed, allocation of resources. It is hoped that the result may then contribute to a debate on how radiation safety may be placed proportionately into the overall optimisation of protection, incorporating an interdisciplinary and rigorous safety culture, encompassing all the aspects of the process and its socio-economic and geographic context.

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328-NEUTRON SHIELDING FOR A NEW PROTON THERAPY FACILITY IN THAILAND WITH A GEANT4 SIMULATION STUDY

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ABSTRACT

The King Chulalongkorn Memorial Hospital (KCMH) is building a proton therapy facility in collaboration with the Faculty of Medicine, Chulalongkorn University. The proposed facility has a single gantry room and will use a cyclotron as the source of protons. In this study, neutron considerations for the proposed proton facility were investigated using the Monte Carlo-based Geant4 simulation toolkit to calculate the effective dose due to the secondary neutron field produced a new proton therapy facility. Proton beams are originated with an energy of 100 MeV in the gantry room with different angles for the patient: a fixed 100 MeV proton beam was also modeled. The effective dose was calculated in several locations of interest, inside and outside the facility for different scenarios. The simulation results showing that the effective dose ranges obtained are reasonable. The effective doses calculated through the Geant4 simulation were compared to the Thailand regulatory limits. They showed that the facility would not pose a health risk for the public or staff, with a maximum effective dose rate of 20 mSv per year in control rooms and maze exit areas and 1 mSv in a year close to the walls, outside the facility, under very conservative assumptions. This study illustrates the first neutron shielding verification analysis of a new proton therapy facility. Moreover, it may perform a new baseline of comparison and validation for the international community, besides confirming the viability of the facility from a radioprotection approach.

Keyword: Proton therapy facility, Neutron shielding, Radiation Protection, Geant4

329-PRECAUTION TO MINIMIZE RADIATION EXPOSURE IN THREE DIFFERENT SPACE TOURISM DESTINATION

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Current advancement in launch vehicle technology will made affordable space tourism possible. Spaceflight today consisted of healthy, fit and well-trained astronauts doing research in low-earth orbit of 400km. With the advent of space tourism, people from the general populace with varying health condition will be exposed by similar radiation. Certain precaution has to be taken to keep the radiation dose of the soon-to-be space tourist below that of professional astronauts.

There are several possible destinations of space tourism with different exposure to radiation. These exposures depend on the altitude or distance from the surface of the earth, and the activity of the sun. Those destinations are sub-orbital flight, low earth orbit, and the moon. Due to differences in characteristics between several destinations, different precaution or procedure might be employed for each.

SUB-ORBITAL

A sub-orbital spaceflight is a form of spaceflight in which the spacecraft reaches space, but not completing a single orbit before re-entering earth atmosphere. Due to the nature of this kind of spaceflight, a heavy shielding incurs significant weight penalty and is uneconomical. The radiation dose received by passengers are not high enough to justify a heavy shielding. A dose of 0.34-2.64 μSv [1] is less than dose during one hour of flight at 35,000ft. Thus, a radiation protection level implemented on commercial flight might be enough for sub-orbital space tourism. A pilot in these flights might be exposed to higher level of radiation, but depending on the number of launches, it may not be higher than crews of a commercial flight [2]. Factor other than radiation may has more impact towards limitation of flight, such as high g-forces.

ORBITAL

Radiation doses experienced by crew and tourist on an orbital spaceflight is in several order of magnitude higher than those of sub-orbital flight. Thus, a shielding is necessary. Currently, International Space Station (ISS) astronauts which orbits at 410 km has experienced a dose of 1 mSv per day. For terrestrial radiation, ICRP has mandated occupational dose limit in 20mSv per year averaged, or 100mSv in 5 year. But, for spaceflight purposes, ICRP [3] has determined that typical dose is 1 mSv per day and dose is to be decided per mission. All uncertainties need to be assessed in determining the designed dose limits for specific mission, supported with data from computational radiation transport code and radiation forecasting. [3]

A space tourist might come from a different background from astronauts, and consequently has different medical conditions. Thus, lower dose limit might be preferred. Future regulation body may decide on new limit for spaceflight purposes if enough data sets from manned mission are available.

First line of defense in avoiding overexposure may done on the ground. There should be a year window between reservation and departure date. Within that window, applicant should have their dose monitored, If within one year the applicant exceed certain limit, departure date might be delayed or cancelled. This is done in addition to health checks that will be automatically in place for such flights.

The other defense against exposure is the spacecraft. Orbital tourism might be done long enough and need enough facilities to justify the need of a space station (or any form of permanent habitation in orbit). This enable more shielding to be used because the launching cost of the shielding will be a one-time cost. Optimal orbit should be calculated in a way that it is high enough so that the atmospheric drag is minimized yet low enough so that the need for shielding does not outweigh the benefit of having higher orbit. The station should have an area with specifically high shielding to protect occupant from increased radiation due to higher solar activity or any other cause. Several radiation detection and alarm system should be in place to warn occupants. Station design should also consider the possibility of secondary radiation production due to primary radiation interaction with structures, equipment and the astronauts [3]. For earth shuttle, certain trajectory should be calculated in a way to minimize transit time.

The other line of defense will be procedure and training that needs to be created to prevent overexposure. Dosage monitoring in orbital mission should be carried out [3]. To suppress the biological effects of radicals formed due to high radiation exposure, certain novel drugs might be employed in preventing cells death. Astronauts dose records should be included in a medical record and validated using biomarker dosimetry methods. The data is important in assessing the cancer risk of the astronaut and determining the allowable dose limit during the designed post-mission time window. During the post mission period, astronaut dose limit and health should also be monitored. [4]

THE MOON

Due to its location, a spaceflight to the moon will pass through the Van Allen Radiation Belt. Thus, significant radiation exposure might be expected in lunar excursion. To avoid mass penalty from carrying heavy shielding from the surface of the earth and moon, several vehicles might be used to transport tourist to the moon. A shuttle with low shielding might carry tourist and fuel from earth surface to a transit spacecraft with heavy shielding parked in low earth orbit. Another shuttle parked on the moon or lunar orbit then carry tourist to the surface of the moon, in which tourist are to be kept in heavily shielded environment. Any excursion should be done in a shielded spacesuit and/or vehicle. This three-vehicle configuration plus permanent lunar base reduce weight penalty that might be incurred if the shielding is to be carried by single vehicle from the surface of the earth to the surface of the moon. In addition, precaution and procedure that is in place for orbital tourism is also employed. Passenger requirement for lunar flight should be higher than that of orbital flight, since lunar flight involves higher amount of radiation and less flexibility for emergency evacuation.

CONCLUSION

With increasing possibility of space tourism, precautionary measure needs to be taken to minimize the effects of cosmic radiation exposure. Technical and procedural measure presented in this paper can minimize those effects but are rather complicated in their implementation without a proper regulatory framework. Regulation needs to be case-specific with several destinations and radiation condition in mind, but not necessarily different for every single person.

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331-RADIATION PROTECTION PRACTICE AT B.P.KOIRALA MEMORIAL CANCER HOSPITAL NEPAL

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Introduction: B. P. Koirala Memorial Cancer Hospital (BPKMCH) is the first cancer hospital of its kind in Nepal. It provides high-quality services for the prevention, diagnosis, treatment and research on Cancer since 1998. Diagnostic Radiology and Nuclear Medicine (RDNM) and Radiation Oncology (RO) have been using radiation for diagnostic and treatment proposes. RDNM department is equipped with CT scanner, Mammogram and X-ray radiation generators and RO two Linear Accelerators, a Cobalt-60 teletherapy unit, CTsimulator and HDR brachytherapy.

Methods and materials: The radiation protection assurance possible only with radiation equipment's standard bunker/room and door, quality control of equipments and personal monitoring of radiation workers. Medical physicists and RSO has been involving in the safety of radiation workers by involving in frequent monitoring of radiation area and providing the individual dosimeter. The area monitoring has been performed in equipment's optimum performance as well as clinical condition. The neutron survey was performed for 15MV photon beam in door, console and corridors. The survey was also performed near on couch at isocenter level as well as chest level of radiation worker at patient setup. Similarly, regular quality control along dose monitoring of machine and international dose audit which help to limit the dose to the patient.

Results and Discussions:

Personal Monitoring: The thermoluminescent dosimeter (TLD) has started since the beginning of the radiation departments. Radiation workers have been wearing chest and wrist badges as per radiation environment. Every three months used TLD badge used to send for evaluation. As shown in table 1, the dose report shows that annual or five year effective dose received by radiation workers is within limit, 20mSv and 100mSv [1]. The mean lifetime dose of 20 years is 1.07mSv. The mean cumulative dose of last five year is about 0.40mSv. The institute mean and collective effective dose of last single year is 0.09mSv and 5.20person-mSv respectively. The RDMP effective dose is slightly higher than the RT department. The result shows that the institute has assured radiation protection to the workers.

Area Monitoring: The radiation area monitoring of DRNM, RT equipments found that radiation leakage from walls and doors is in background level. In case high energy photon beam, 15MV, slightly higher than background level but within dose limit. Since the use of high energy photon beam is less than one percent. The neutron dose level activated from 15MV photon beam found maximum at door is 2mSv per hour and background level in rest place. During the exposure, the scatter neutron dose rate at Linac couch, near isocenter and radiation worker standing points are found around 42mSv per hour. Almost a half minute time takes to drop down to background level after beam off. In case of cobalt-60 teletherapy and brachytherapy, the radiation around the machine is present as usual. The radiation workers, patients and visitors are in safe environment.

Radiation Dose Audit: The proper delivery of prescribe RT dose to the patients also a part of radiation protection for patient. For this propose mechanical and dosimetric quality control need to perform frequently. Medical physicists have been involved since the establishment of services. The dose audit has been carried out with the support of IAEA using postal dose audit programme [2]. The audit result are very satisfactory and are within the limit as shown in table 2

TABLE 1. PERSONAL DOSE MONITORING REPORT

Description	Institute (mSv)	RT(mSv)	DRNM (mSv)
Mean lifetime cumulative effective dose upto 2018	1.07	0.83	1.30
Average cumulative effective dose of last five years	0.40	0.33	0.51
Average effective dose of a year	0.09	0.08	0.10
Collective effective dose (person-mSv) of a year	5.20	2.85	2.35

TABLE 2. POSTAL DOSE AUDIT RESULTS

Beams/machine	Year	User stated Dose (Gy)	IAEA mean (Gy)	% deviation relative to mean dose
6MV,Linac I	2007	2.0	2.05	-2.5
	2009	2.0	2.00	-0.1
	2011	2.0	2.00	1.0
	2013	2.0	2.07	3.3
	2015	2.0	1.97	1.5
6MV,Linac II	2007	2.0	2.01	-0.4
	2009	2.0	2.08	-3.7
	2011	2.0	2.10	-4.6
	2013	2.0	2.07	-3.3
	2015	2.0	2.02	-1.0
6MV,Linac III	2018	2.0	1.97	1.3
15MV,Linac III		2.0	1.99	0.5
Co-60	2007.0	2.0	2.04	-1.8
	2009.0	2.0	2.00	0.0
	2011.0	2.0	2.10	-4.8
	2013.0	2.0	2.11	-5.1
	2014.0	2.0	2.03	-1.4
	2015.0	2.0	2.06	-3.0

Conclusion: The personal dosimeter and radiation dose audit results shows that BPKMCH has been well-practicing the radiation protection principle for radiation worker, patients as well as for public. Still some implementation needs to apply in diagnostic radiology and nuclear medicine department. In RT department, patient safety has been doing using quality control. The radiation protection of radiation workers and patients is main responsibility of medical physicist/Radiation protection officer.

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332-OVERVIEW OF DIAGNOSTIC REFERENCE LEVELS IN NIGERIA

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Background

Radiation dose delivered to patients undergoing x-ray examinations are influenced by several factors, and these determinants are responsible for the wide dose variation [1]. A critical step towards harmonization of radiation dose delivered to patient is the establishment of dose reference levels (DRL). Diagnostic reference levels (DRL) helps to indicate dose levels that may be unacceptably high or low in order to adopt possible dose reduction strategies that will not compromise the required level of image quality [1-3]. Radiation regulatory bodies adopted and recommended the use of DRL globally. Consequently, several nations of the world have established and adopted the usage of national dose reference levels (DRL) in their countries [2-4]. Such countries develop their own DRL in order to achieve optimal image quality with minimum radiation exposure to the patient. In addition, diagnostic standards are enforced, governed and reviewed periodically in accordance with changes in clinical practice and equipment [2-4]. Contrarily, Nigeria is yet to develop its own national DRL. According to World Health Organization (WHO) using DRL as a reference and working within these levels will reduce variability, promote good practice and enhance radiation protection [5]. Establishment of national DRLs in a big country like Nigeria might be complex due to large area of coverage. However, regional DRLs within the country can be harmonized for adoption and subsequently upgrading the process to achieve the desired goal. Hence, the goal of this study is to propose a model for the establishment and adoption of national DRL for routine x-ray examinations in Nigeria.

Methods

A systematic literature search of Scopus, Google Scholar, Research Gate and Google, for studies that estimated diagnostic reference levels in Nigeria from 2015 to 2020 was conducted. We extracted all the reported dose reference levels (DRLs) having standard quality. Selected protocols were based on examination commonly reported in literature. The descriptive statistics was analysed using SPSS (Version 23.0).

Results

The estimated dose reference levels (DRL) for computed tomography dose indices for adult and children protocol respectively is as presented in Tables 1 and 2. Head, chest and abdomen protocol were recorded for adult while only head CT examination were reported for children. The DRL values for routine radiography examinations is depicted in Table 3. Chest PA has the least median dose value and lumbar spine (LS) LAT has the highest median dose value.

TABLE 1. SUMMARY OF REPORTED DRL FOR COMPUTED TOMOGRAPHY EXAMINATIONS IN NIGERIA FROM 2015-2020 [6-13]

Adult protocol	Head		Chest		Abdomen	
Dose Indices	$CTDI_{vol}$ (mGy)	DLP (mGy.cm)	$CTDI_{vol}$ (mGy)	DLP (mGy.cm)	$CTDI_{vol}$ (mGy)	DLP (mGy.cm)
Mean	64.1	1236.7	14.7	631.1	18.6	1186.9
Median	63.0	1310.0	15.2	691.3	19.6	1252.3

TABLE 2. REPORTED DRL FOR PAEDIATRIC HEAD COMPUTED TOMOGRAPHY EXAMINATIONS IN NIGERIA SHOWING DIFFERENT AGE GROUPINGS [14]

Dose Indices	New born	1 y	5 y	10 y
$CTDI_{vol}$ (mGy)	27.0	37.0	48.0	54.0
DLP (mGy.cm)	1040.0	988.0	493.0	1824.0

TABLE 3. SUMMARY OF REPORTED DRL (mGy) FOR ROUTINE RADIOGRAPHY EXAMINATIONS IN NIGERIA FROM 2015-2020 [15-24]

Protocol	Chest PA	LS AP	LS LAT	CS AP	CS LAT	Pelvis	Abdomen	Skull PA
Mean	1.52	5.42	9.58	1.36	1.27	3.11	3.90	2.88
Median	1.23	5.00	9.00	1.45	1.30	2.78	3.67	3.70

LS: Lumbar Spine; CS: Cervical Spine; PA: Posterior-Anterior; AP: Anterior-Posterior

Conclusion

The dose reference levels (DRL) obtained in this study are comparable with those from other countries and showed the possibility of dose harmonization in Nigeria. Adoption and implementation of DRL is recommended in order to promote radiation dose optimization and enhance patient safety.

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333-CAPACITY BUILDING TO ENHANCE RADIATION SAFETY CULTURE: PREGNANT PATIENT UNDERGOING F18-FDG PET-CT EXAMINATION.

Dedicated education & training of medical staff including ethical aspects, justification, optimization, fetus dose reporting & benefit-risk dialogue, response in case of unplanned exposure and unexpected pregnancy revealed.

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Capacity building is a base to set a radiation safety culture in the clinical environment. The aim of the comprehensive capacity building process is to use medical radiological examination every time when benefits overweight the possible risk and to avoid unjustified fear of radiation during planned exposure, as well as to support healthcare staff discussion about radiation benefit and risk. This presentation shows that, in order to enhance radiation safety culture when pregnant patient undergoes a PET-CT examination, the medical staff preparedness and response to unplanned exposure should be included among the topics of the Education & Training schemes. Unplanned exposure might occur despite of the number of the radiation safety procedures implemented in the clinical environment. One of such situation is an unexpected reveal of the pregnancy during radiological examination, especially during a PET-CT examination.

The F18-FDG PET-CT procedure is commonly used for patients, but for a pregnant patient the justification of the procedure is still a very sensitive topic. This relevant examination might not be performed due to staff concern about the foetus radiosensitivity and the legal aspect of the examination, despite that there are many international guidelines describing this topic and the procedures for planned exposures of pregnant patients [1], [2], [10]. In addition, usually all guidelines contain only quantitative description of the risk, which is not preferable to be used during healthcare staff discussion with a pregnant patient. The proposition of qualification for the increase of childhood cancer and proposition of description of the possible deterministic effects are given in TABLE 1. The details and level of the education & training scheme should depend on the targeted medical staff (MDs, RTTs, Radiographers, MPEs, RPEs). Proposed dedicated E&T, due to response in case of unexpected pregnancy revealed during examination, include:

- Ethical aspects, justification, optimization, communication: fetus dose reporting & benefit-risk dialogue, [2]
- Procedures in place to check weather patient might be pregnant or not,
- Examples of justification of PET-CT procedures [4], [5], [6],
- Image quality definition and assessment including e.g. Visual Grading Analysis,
- ALARA principle to enhance radiation safety culture,
- Classification of planned and unplanned exposure and radiation safety procedures,
- Overview of the methodology of the conceptus dose estimations (monte carlo simulations, physical phantoms measurements, available on-line tools), [7], [8],
- Analysis of the case and results of estimated conceptus doses, optimisation possibilities and impact of examinations parameters on fetus dose (eg. kV, F18-FDG activity, duration of the procedure), possibilities for the dose optimisation due to patient BMI (i.e. lower kV for patient with low or normal BMI comparing to kV used for obese patients),
- Case dose analysis (TABLE 1) and proposed benefit-risk communication including proposed qualification for increase risk of childhood cancer (TABLE 2), which might be used to support healthcare staff discussion with a pregnant patient, based on the international guidelines [1],
- Presentation of available data concerning radiation effects for fetus [9], [10],
- Overview of the legal requirements and international Basic Safety Standards,
- Comprehensive dose management, including recurrence exposures.

TABLE 1. Fetus dose estimation and proposed benefit-risk dialogue guidelines:

Dose to conceptus (mGy) above natural background	Probability of no malformation	Probability of no cancer (0-19 years)	Proposed qualification for increase risk of childhood cancer
0 [1]	97 % [1]	97,7 % [1] (means 0,3% risk of cancer)	N/A
5 [1]	97 % [1]	97,7 % [1] (means 0,3% risk of cancer)	Negligible
10 [1]	97 % [1]	97,6 % [1] (means 0,4% risk of cancer)	Minimal
50 [1]	97 % [1]	97,4 % [1] (means 0,6% risk of cancer)	Very low
Estimated for this case procedure: **77 mGy (7,7 cGy)** TABLE 2	97 % [1] (*case explanation*: i.e. unchanged relative to naturally occurring in the population or due to other factors; dose level well below the threshold for teratogenic deterministic effects or for a significant reduction in IQ	*Case estimation*: **97,2** % [1] (means **0,8%** risk of cancer)	*Case qualification*: **very low**
100 [1]	97 % [1]	97,1 % [1] (means 0,9% risk of cancer)	Low

TABLE 2. Total conceptus dose estimation for PET-CT examination - 77 mGy (7,7 cGy). Presented data concern case study F18-FGD PET/CT after combined treatment for axillary soft tissue Ewing's sarcoma, 23 years old female patient, 93 kg, 163 cm, around 16 week of pregnancy unexpected revealed during examination.

Exposure parameters and estimated dose of whole body CT	Exposure parameters and estimated dose of PET examination
140 kV, 172 mAs, pitch 0.8, colimation 3 mm, CTDIw (mGy) 18,92; CTDIw (mGy/100mAs) 11	F18-FDG, 344 MBq
Conceptus dose: 67 mGy (6,7 cGy)	Conceptus dose: 10 mGy (1 cGy)

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337-PARALLELISMS BETWEEN ORGANIZATION AUDITS AND HEALTH CHECKUPS

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Is there any parallelism between auditing an organization and doing a personal health checkup? The challenge here is to demonstrate they have much more in common than at first sight one would suspect! In fact, they share the same purpose: whether it is a complete (whole-body) approach or a more focused approach, dedicated to a specific process (partial body or organ), the outcome represents an image of the current situation; it is like a snapshot of the organization (or the organism).

The mains goals of organization audits (or health checkups) are to:

(j) Assure that everything is going as it should or planned.
(k) Identify opportunities for improvement.
(l) Identify the risks (as an organization needs to know which are the internal and external threats, the doctor needs to know the history and the risk factors of their patient).
(m) Prevent potential future problems (for example, in an organization might be identifying potential no-conformities, parameters or work practices that might be improved; establishing the parallelism with the human body, would be preventing a disease).
(n) Correct any imbalances.

(o) Take action (eliminate no compliances in an organization or manage the disease by prescribing the best therapeutic approach).

More than a mere tool of the quality management system (or patient management), audits (or health checkups) should be seen as a tool for the management board (or for the physicians).

The presentation goal is to demonstrate how regular auditing (or health checkup) contributes to raising the level of compliance (or getting a fitter body and mind) within an organization and improve the overall culture of quality and radiation protection (or acquire or maintain a healthy lifestyle).

The preliminary steps of auditing consist in its careful planning: decide when and what to audit, whether it is an external (physical exam) or internal audit (lab or imaging procedures) are critical steps to obtain a successful outcome. This is crucial to decide how many hours or days you need to allocate, which resources (human and material), which areas or processes are going to be audited. If there is any previous audit, it will be checked whether all the “findings” were properly addressed, exactly as the physician always asks for symptoms and previous lab or imaging results.

Full auditing is like a full health checkup: although necessary, depending on organization performance (body shape/health status), it might be expensive, disturb the normal workflow and time-consuming. A common approach to minimizing negative impact without compromising expected results is to perform regular/periodic audits/checkups focused on specific processes of the organization (organ or organ systems) planned in an integrated manner, to assure that nothing is “left behind” or overlooked or the other way around. Some of the processes to be checked in the Nuclear Medicine Department are: patients’ and referring physicians’ satisfaction, work practices regarding radiation protection, infection prevention and control, and document system (SOP, records and record-keeping, archiving, and data protection). Items of a checklist are scored according to the level of conformance. If no or partial conformance is identified, a recommendation is released (in this parallelism: a treatment is prescribed).

The experience acquired so far, has demonstrated that regular auditing had a significant impact on the overall quality of services (or health status) and on continuous improvement. It also led to changes in work practices such as the implementation of stricter rules regarding record keeping, equipment quality control, survey areas for radioactive contamination, cleaning procedures, including frequency adjustments, waste management and the correct use of shielding devices (in this parallelism, it could be translated as introducing healthier lifestyles, such as physical exercise, healthier eating habits and sleep hygiene).

Leading a successful and meaningful organization is like taking care of a living organism: it is a full-time job, requiring awareness of internal and external signs or threats, supported in an evaluation system designed to leverage the organization or body level to its highest performance.

To get the most of it, it should be used in a preventive way by identifying potential future issues and avoiding the disturbances or disruptions that might come up later on. Sometimes, it might work on a mitigation way, by identifying existing problems with no (or shorter) delays, allowing quicker and more assertive actions.

In summary, regular audits (or health checkups) improve the organization’s (or body) performance through better management of risks, cost containment, promotion of safety and overall staff motivation.

338-FETAL DOSE ESTIMATION DURING PREGNANT BREAST CANCER RADIOTHERAPY

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Typically, during radiotherapy, pregnancy should be avoided. Nevertheless, this can be done if the patient has strong clinical indications or even the occurrence of uncontrollable pregnancy during treatment. Therefore, it should appropriately optimize the tumor dose with reducing the fetal dose as low as possible. The radiation dose of the fetus that is not in the field is mainly due to radiation leaks from treatment head, scattered by collimators and beam modulation, and scattered radiation within the patient [1-4].

Purpose: This work aimed to determine the fetal dose in a pregnant breast cancer patient.

Materials and Methods: A 26-year-olds woman with right breast cancer had been treated by 10 MV photon beams for 13 fractions (2Gy/F, 5F/wk) before confirming her pregnancy with 6 weeks gestation period. The treatment plan was generated on Eclipse Treatment Planning system version 13.6 for two opposing tangential fields with field-in-field (FIF) technique and left supra clavicular area field with a total dose of 50Gy in 25 fractions, five days/week. On the CT images of an adult Alderson Rando anthropomorphic phantom, the position of the fetus for the first stage of gestation was simulateto be above the pubis and deep in the middle of the body(slide no 31) then approved by a gynecologist. The patient treatment plan was transferred to the anthropomorphic phantom images with re-calculation of the dose. The fetal dose was measured by using thermoluminescent dosemeters (TLDs-100) rods at the positions represented the fetus for every trimester. The measurements were repeated with 3 irradiations.

Results: As shown in Table 1, the total fetal dose with full treatment course for the first semester of the pregnancy is 0.105 Gy or0.056 Gy for 13 fractions of the treatment. The values for 2nd trimester varies from 0.119 to 0.255 Gy. The dose is increasing with the increasing trimester up to 0.562 Gy for the 3rd trimester.

Conclusion: The out of field dose of the fetus depends on the stage of gestation and the distance from the edge of field. The first trimester of gestation is important since the radiation dose as low as 0.10Gy may relate to the growth retardation (AAPM Report No.50)[5].

TABLE 1. The average fetal dose measured in Rando Phantom representing the pregnancy patients treated for breast cancer with the total dose of 50 Gy

Trimester	Slide No.	Distance from lower edge of the fields (cm)	Dose (Gy) ± SD
1st	31	27.5	0.105 ± 0.001
2nd	30	25	0.119 ± 0.001
	29	22.5	0.134 ± 0.001
	27	17.5	0.178 ± 0.004
	25	12.5	0.255 ± 0.009
3rd	23	7.5	0.389 ± 0.015
	21	2.5	0.562 ± 0.021

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339-PROBABILISTIC INVERSE MODEL FOR IDENTIFICATION OF ATMOSPHERIC CONTAMINATION SOURCE IN AN CONTINENTAL-SCALE BASED ON THE 106-RU 2017 EVENT

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In October 2017, a lot of European countries reported atmospheric detections of ruthenium 106-Ru. In many scientific papers, probable locations of the sources of 106-Ru agent in the atmosphere have been considered. Based on airborne concentration measurements and chemical assumptions, it is possible to assume that the release occurred in the Southern Urals region in the Russian Federation. Such reconstruction requires determining the set of source parameters and is defined by the Source Term Estimation (STE) field in the Continental scale. We present the Bayesian framework able to identify the source of the large scale airborne contaminant. The proposed model utilizes the adjusted two-stage inverse problem methodology. To increase the correctness of the estimated source location the prior distribution was assumed based on fast-backward model (HYSPLIT) calculations. Then to obtain posterior distribution the performed probabilistic calculations involved propagation of radionuclides from places with high probability values by the advance forward model (JRODOS-MATCH). The dedicated inverse model and Approximate Bayesian Computation (ABC) algorithm enhanced the estimation of the posterior distributions of contamination source parameters and reduced the number of computationally costly forward model runs.

STOCHASTIC RECONSTRUCTION METHODOLOGY FOR 106-RU 2017 EVENT

In emergency response management it is important to know the area that might become contaminated following the release of dangerous material. Given a source term and wind field, we can apply an appropriate atmospheric dispersion model to calculate the expected concentration. On the anther side, given concentration measurements and knowledge of the atmospheric air parameters identifying the release source is very difficult. This task can be understood as showcase of the dispersion model reproducing the encountered contamination and is defined by the Source Term Estimation (STE) problem [1]. Bayesian methods formulate this problem into searching for a posterior distribution based on efficient sampling of an ensemble of simulations using priori knowledge and observed data. In Bayesian inference, posterior distribution is related to prior distribution through a likelihood function: $\pi(\theta \mid d_{obs}) = \pi(d_{obs}|\theta)\pi(d_{obs})$. where θ and d_{abs} denote the source parameters of interest and the measurement. The goal of Bayesian inference is to compute the posterior distribution by sampling due to the multivariate distribution characteristics. The idea of Approximate Bayesian Computation (ABC) [2], which is one of the most popular sampling algorithms, is to accept θ as an approximate posterior draw if its associate data d is close enough to the observed data d_{obs}. The accepted parameters are a sample from $\pi(\theta|\rho(d, d_{obs}) < \varepsilon)$ where the $\rho(d, d_{obs})$ is the chosen measure of discrepancy, and ε is a threshold defining as closeness margin.
The primary purpose of this work is to locate possible source term of ruthenium released in an unidentified event in 2017. Different countries measured the 106Ru concentration by their measurements monitoring station during early October 2017. IRSN [3] and the Ro5 group have collected all measured data. To track location for source term, determination of the path line of travelled 106-Ru will be very informative and essential. HYSPLIT 4.0 computer [4] code has a backward trajectory model with different vertical motion which is useful to collect pathline and additional information about travelled air mass. Results obtained from different backward trajectory models will be used to form prior distribution $\pi(d_{obs})$ (see Fig.1).
The JRODOS [5] computer code with the MATCH model used as a forward concentration prediction method $JRODOS\_MATCH(\theta^i) \rightarrow d$, where d is simulated values in the places where the sensors are located. The possible sample θ^i were used as source term location (x, y) with different sensitivity cases such as the activity q of released Ru (form high=280TBq to Low=80TBq), the height of release h (from 10 to 100 meters). Another target is also

to get a possible date of release (t). After performing ABC simulations, it was found out that the most probable source term region is Mayak (see Fig. 2). Subsequently, the source term parameters were analyzed: emission level and date of unexpected event occurrence. The results and conclusions obtained coincide with those of the publications and reports dealing with the 106Ru event topic published by IRSN [3].

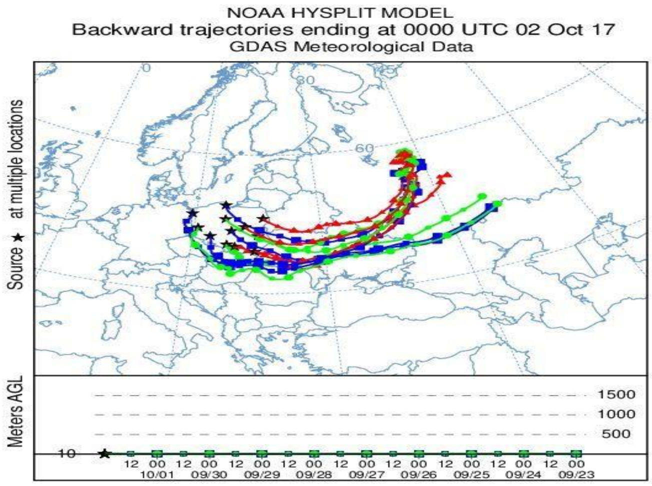

FIG. 44. Trajectories from Poland with the HYSPLIT 4.0 *Isosigma model*

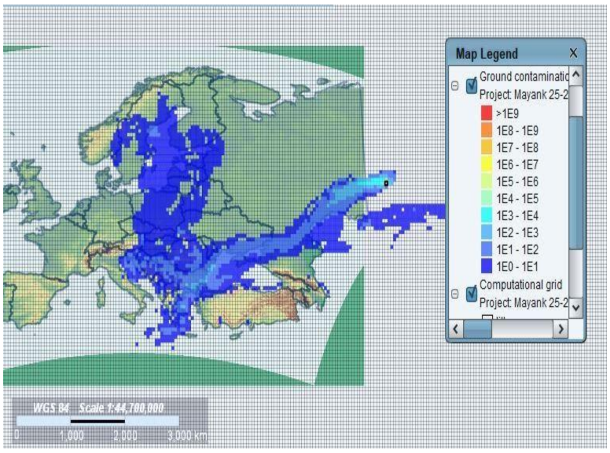

FIG. 2. Concentration and path line of 106-Ru from with the highest probability values θ^{MAP}

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340-AN APPRAISAL OF THE IMPACTS OF "RADIOPHOBIA" ON EFFECTIVE RADIATION PROTECTION, AND THE NEED FOR A NEW COMMUNICATIONS PARADIGM

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Radiation is one of the best-studied carcinogens, yet it remains the most feared, as a result of discursive and cognitive links with birth defects, death and destruction. Its invisibility to the senses, the relationship with cancer and the strong emotional response elicited in relation to nuclear energy has made radiation a classical case in risk literature. The perceived uniqueness of radiation health risks is reflected in the effects in the aftermath of nuclear accidents, where the socio-psychological impacts of "radiophobia" are causing considerable negative health detriments, in most cases more so than the actual radiological impacts. Despite considerable efforts by sections of the scientific community and the nuclear industry to normalise humankind's relationship with radiation there has been little shift away from the perceived uniqueness of the health risks of radiation.

This presentation will offer an appraisal of the impacts of "radiophobia" on ensuring effective radiation protection which in an appropriate fashion balances any potential health impacts of radiation with the many beneficial applications of nuclear technology. This presentation will also explore the need for a new radiation risk communications paradigm in order to achieve effective radiation protection, based upon psycho-social insights and novel applications in respect of cognitive and psychological research.

THE CONCEPT OF "RADIOPHOBIA"

"Radiophobia" as a term is often encountered in relation to radiation, be it risk perception, communications or radiation protection. Its definition, however, are as many as there are practitioner in the world of radiation, with no consensus as to what "radiophobia" actually is. Nevertheless, it is crucial to attempt to establish a stronger definition of the phenomenon, in order to be able to fully ascertain its impacts on radiation protection and risk communication in this particular field. The Mayo Clinic defines a phobia as *"an overwhelming and unreasonable fear of objects or situations that pose little real danger but provoke anxiety and avoidance. Unlike the brief anxiety you may feel when giving a speech or taking a test, specific phobias are long lasting, cause intense physical and psychological reactions…"* [1]. Based on this definition, "radiophobia" as a concept would most likely be limited to a very small number of individuals. However, it would be incorrect to apply such a narrow and clinical definition to "radiophobia", and it would be more fruitful to focus on a much broader socio-political and psychological phenomenon which is probably best defined as a latent anxiety – often out of mind, but resurfacing under specific conditions (e.g. nuclear accidents, involuntary irradiation events). The consequences of "radiophobia" as experienced after radiological accidents are well documented, including anxiety, fatalistic behaviour, substance abuse, stigmatisation, increased rates of suicide, to mention but a few [2, 3]. However, due to the fact that no unified, in-depth studies have been conducted into the combination of historical, psychological and cognitive factors which have created "radiophobia", there is only a fractured understanding of the phenomenon. In turn, this threatens to undermine efforts to avoid "radiophobic" responses to radiological and nuclear incidents.

This presentation will be offering a unified account of "radiophobia", building on historical narratives, psychological factors such as availability bias, affect heuristics, psychological anchoring and confirmation bias, as well as cognitive factors. The invisibility of radiation lends itself to the creation of mental constructs by the individual to build an "image" or a point of reference to understand radiation as a phenomenon, including the importance of popular culture in creating these images, in conjunction with the aforementioned heuristic factors. Building on this, the presentation will offer a novel application of these concepts to explain "radiophobia" and how it related to effective radiation protection.

THE LINK BETWEEN THE LOW-DOSE DEBATE, RADIATION PROTECTION AND "RADIOPHOBIA"

Given that historical reviews of humanity's relationship with radiation showcases a clearly dynamic nature, with its early signs emerging in the aftermath of the global nuclear testing controversy, it is clear that "radiophobia" co-evolved with the debate which began in the 1950s and 1960s over the potential health impacts of low-dose radiation (<100mSv). It is therefore of interest to explore the linkages between the low-dose debate, which still rages on, the current radiation protection paradigm and "radiophobia".

For decades, the application of linear exposure/risk models in the realm of public health and regulations has been commonplace. Low-dose radiation is a prime example of such a scientific and regulatory domain. The presentation will highlight how the discursive process which led to the adoption of the Linear No-Threshold (LNT) of radiological protection has helped shape and influence current risk perceptions, communications strategies and knowledge geographies of radiation. Despite increasing critique against the LNT and practises such as collective dose assessments to calculate hypothetical health detriments, the current radiation protection paradigm is strongly linked, both historically and discursively with "radiophobia" as the dominant radiation protection paradigm reflects the perceived uniqueness of radiation health risks. It will be shown that a "radiophobia"-driven radiation protection philosophy, which does not take into account socio-economical factors in an appropriate fashion, may inadvertently result in less effective radiation protection measures.

A MODEL FOR EFFECTIVE RADIATION RISK COMMUNICATION

In order to avoid future instances of "radiophobia" and ensure that radiation protection measures are as effective as possible, there will have to be a reappraisal in our relationship with radiation and bring it in line with scientific facts. However, the current radiation protection paradigm will make this difficult it is reinforcing to the public the notion of a unique threat. There is strong evidence of a continuous divergence in risk perception between the radiation protection community and the general public, a phenomenon which has been well-documented over several decades [4]. Therefore, in order to achieve such a reappraisal in terms of radiation risks, a different model for effective radiation risk communications will be offered, building not only on already-existing research in this area, but also drawing upon key insights from psychosocial and cognitive research. This will be done against the backdrop of the reliance amongst many stakeholders in radiation protection on facts-based approaches in radiological risk communication. By also drawing on a historical discourse analysis of radiation, this presentation offers a new interpretation of the risk communications paradigms currently in used by the nuclear industry, governments and regulators. This will be done alongside with analysis on how truth- & trust decay in modern societies impacts the prospect of effective risk communication in radiological contexts

ACKNOWLEDGEMENTS

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341-RADIATION SAFETY AT CLINICA LAS CONDES HOSPITAL. EIGHT-YEARS OF PERSONAL DOSIMETRY.

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BACKGROUND

Clinica Las Condes is a highly complex health institution with more than 3,500 workers, where various diagnostic and treatment modalities use ionizing radiation. It has, among others, Radiotherapy, Nuclear Medicine and Radiology. A great effort has been made by the entire community to achieve implementation of the Basic Safety Standards [1] in practice. In 2011, decision to create a Comprehensive Radiation Protection Program was made, this Program would include all the hospital radiological units and all occupationally exposed workers (OEW), more than 650 people.

To put this Program into practice, many protocols, procedure guides, indicators, databases, radiological surveillance, training and qualifications were created throughout these years, which today form the basis of the Radiological Safety Culture [2] at the Institution.

The objective of this work is show how personal dosimetry has behaved in the hospital from 2012 until today, analysing the percentage of workers in four ranges of defined annual doses, and get an idea of the level of compliance with the protocols adopted for radiological protection in the institution related to the Culture of Safety of personnel.

METHODS

Considering Chilean national regulations, quarterly follow-up was performed on the equivalent doses or Hp (10,) received by workers in the last eight years through a local dose management system, for this work annual doses corresponding to the sum of the four quarters were considered.

To carry out the analysis, different variables were considered: number of workers with dosimetry, average annual dose, quantity and percentage of values of dose within 4 ranges (0.1 mSv, 1 mSv, 5 mSv and 20 mSv) as well as the average global values.

RESULTS

Commercial software from Microsoft Corp., such as Access and Excel, was used to prepare and analyse the data within the integrated dose management system.

The data segregated annually can be seen in Table 1. None of the workers received an annual dose greater than 20 mSv.

TABLE 1. ANNUAL DOSE, CLASSIFICATION ACCORDING TO DOSE RANGE AND NUMBER OF WORKERS

	Average	2012	2013	2014	2015	2016	2017	2018	2019
Dose (mSv)	0,15	0,19	0,13	0,12	0,13	0,14	0,19	0,15	0,15
0 - 0,1 (mSv)	399	302	361	369	415	431	428	467	417
0,1 - 1 (mSv)	117	94	87	98	97	95	164	131	170
1 - 5 (mSv)	15	17	17	11	13	20	14	15	16
5 - 20 (mSv)	2	1	0	1	1	1	5	3	3
Workers	533	414	465	479	526	547	611	616	606

The global results are reflected in Fig. 1, where the average annual number of workers in each of the analysis intervals is shown.

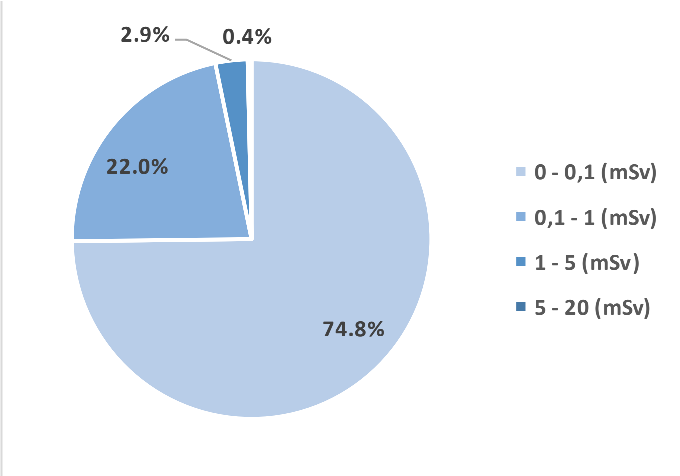

Fig. 45. Chart showing the percentage distribution of personal dosimetry values according to dose

CONCLUSIONS

This results shows that all occupationally exposed workers work in a safe environment considering majority of workers have annual doses below the permissible limit dose for the general public of 1 mSv according to ICRP 103 [3], even more than 99% of then are below 5 mSv that in some countries like Chile is recognized as a public limit dose [4,5].

There are not a lot of parameters to define level of the Safety Culture in a specific institution, but clearly the annual effective doses of occupationally exposed personnel is one of them, other variables such as the number of radiological events and accidents, dose received for patient, level of training and staff training. This work constitutes a starting point in that direction, to have a base to achieve excellence in the Safety Culture in our institution.

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342-ASSESSMENT OF LOCAL DIAGNOSTIC REFERENCE LEVELS FOR PEDIATRIC CT PROCEDURES USING A PYTHON-BASED TOOL FOR PATIENT CT DOSE TRACKING

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According to international and national guidelines [1], European Directive [2], countries should establish diagnostic reference levels (DRL). It is important to assess patient doses and periodically reassess local values of the DRL in healthcare institutions, in order to manage optimal radiation dose for patients. For benchmarking and optimizing local practice of radiology facilities, the challenge is to create a network of medical centres connected to a common patient dose tracking system. To meet this task numerous commercial dose monitoring applications have been released. However, in some (less-resourced) centres such tools are still not implemented and the hospital staff needs to collect data manually which is time consuming and heightens the risk of human errors. The aim of this study was to develop a Python-based tool for patient CT dose tracking to perform the assessment of paediatric exposure doses and establish the local DRL values for most common CT procedures.

MATERIALS AND METHODS

Custom data processing software was developed using the Python programming language. DICOM Structured Reports (SR) of CT examinations were used as an input. For each patient sequence CT dose metrics, including acquisition parameters, exposure and dose indicators were collected automatically. The developed tool was tested with data recorded from one Siemens CT scanner for paediatric examinations performed in 2018 and 2019. Patient-related and exposure-related parameters were collected. The median DLP values were estimated for head scans in different patient age groups. Local DRLs were calculated using the 75th percentile and comparison with and national DRLs [3] and European guidelines [4] are done.

RESULTS

A custom Python-based tool was developed to support medical physicists in their work with optimization of CT examinations. It provides an interim solution to speed up the process of collecting and analysing complete and accurate data in the absence of commercial dose tracking software, until such becomes available.

More than 1300 CT data were collected using Python-based software. Comparing to manual data reporting, more scanning parameters were included for CT dose analysis and substantially less time was needed for data collection. For most softwares, only a few parameters that affect CT dose are analyzed. Our tool enabled a deeper insight into CT dose dependences on acquisition parameters (exposure time, scanning length, collimation, pitch), X-ray source parameters (tube current, voltage, exposure time per rotation) as well as patient age, size and gender.

The patient exposure assessment studies at Vilnius University Hospital Santaros Klinikos were started at 2012 and after the analysis of the results the number of corrective actions, including optimised examination protocols, were performed. The study on head CT protocol optimisation was done with aim to determine the lowest tube current time product (mAs) values for successful non-syndromic craniosynostosis identification without iterative reconstruction algorithms using simulation techniques. Our results showed that images registered with 120 kV and 13 mAs can be used to diagnose nonsyndromic craniosynostosis with statistically same accuracy as with standard protocol and correspond to decrease of effective dose from 4.98 till 0.33 mSv. This method will be applied to other localizations and indication [5].

As a result of exposure optimisation actions done the dose for head CT decreased in all age groups more than 3 times in youngest patients from 2010 to 2018. In 2019, most common single phase CT examination for children was head CT, covering about 46 percent of performed scans. In 2019, the median DLP for head was 119, 215, 232, 275 and 300 mGy*cm for patients aged 0-1, 1-5, 5-10, 10-15 and 15-18 years, respectively. Patient were grouped by age differently, depending whether comparison was done with national or European DRLs. Patient doses showed a wide variation due to patient clinical indication and image acquisition parameters. The patient doses were 2-4 times lower than the national DRLs [3]. When comparing 2018 and 2019 data slight decrease in median DLP is observed in youngest age group (Fig. 1). In all age groups European DRLs are higher than calculated median DLPs.

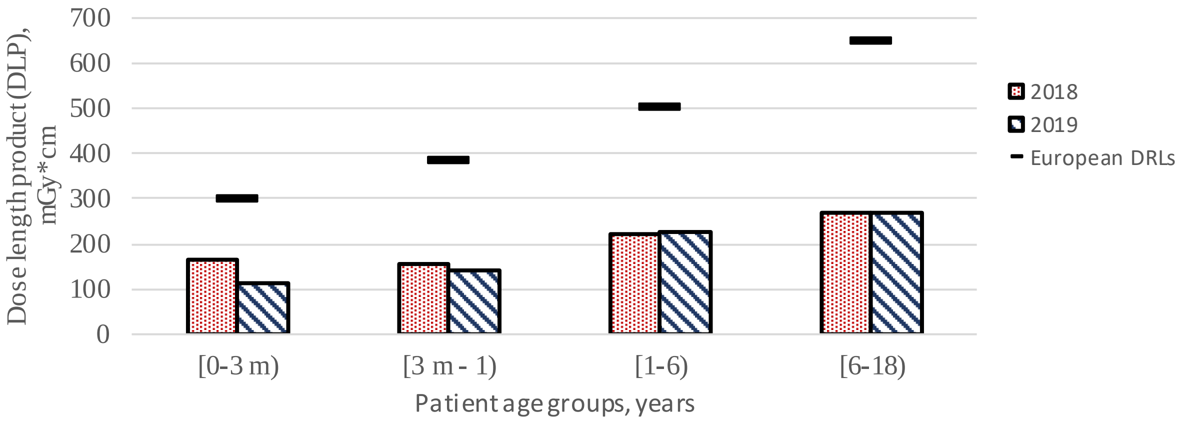

FIG. 1. The comparision of pediatric head CT DLP median values with European DRLs

Paediatric local DRLs were established only for single phase head procedures combining 2018 and 2019 data due to limited set of examinations of other anatomical regions illustrating the need for further dose monitoring (Table 1). The local DRLs are lower in comparison with European guidelines [4].

TABLE 1. LOCAL HEAD CT DRLS COMPARISON WITH EUROPEAN DRLS

Patient age groups	Number of patients	Local DRLs	European DRLs
0-3 months	18	180	300
3 months – 1 year	57	180	385
1-6 years	158	250	505
6 years and older	363	315	650

CONCLUSIONS: Implementation of accurate and well organized dose monitoring system is considered as a first step towards an effective patient dose management strategy. A custom Python-based tool enabled a substantially faster and improved tracking of CT data. It can help to identify primary source of variability in the dose calculation indicating key protocols that could benefit most from dose reduction activities. This in turn will facilitate the optimization of radiation protection of patients in CT procedures and will contribute to the national dose data collection.

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343-RSCS SURFACE CONTAMINATION MONITOR (SCM)

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Alpha and beta surface contamination surveys can be performed quickly and accurately with the one-of-a-kind RSCS Surface Contamination Monitor (SCM). The SCM system employs a Position Sensitive Proportional Counter based system that automatically acquires survey data and pinpoints the specific location of any radioactivity present. The large detectors can be up to 2 meters wide, allowing large surfaces to be surveyed up to 100 times faster than traditional survey scanning techniques for 100% coverage. The unique capability of identifying precisely on the anode where a detection occurred creates an efficient, yet pinpoint accuracy found in no other contamination survey system. The system data processing software rapidly analyzes the survey data and generates easy to interpret digital images, and detailed, objective survey reports. The system can be operated manually by a technician or autonomously using robotics and LiDAR-based Simultaneous Location and Mapping (SLAM).

When performing surface contamination surveys, data are collected for each 5 cm of linear detector travel as determined by precision wheel encoders. Meanwhile, the detector electronics records and localizes all the detected events along the length of the anode. The system software then combines all events located within each 5 cm x 5 cm (25 cm^2) area over the entire scanned surface. Each 25 cm^2 area is then combined as 1/4th of four separate 100 cm^2 areas, as illustrated in Fig. 1. This technique ensures that the highest 100 cm^2 area is identified because it is not sensitive to the mispositioning of the detector as may occur using more traditional systems that employ multiple non-overlapping detectors. Because the data is recorded for every 25 cm^2, the system records 400 measurements for every square meter it traverses.

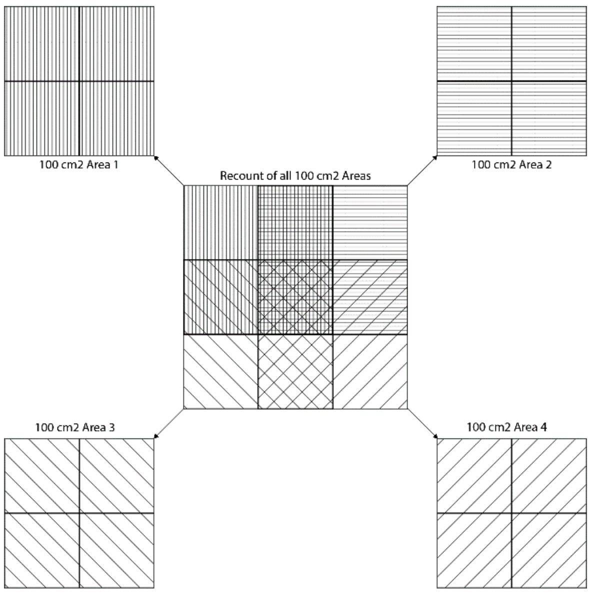

FIG. 46. 25 cm^2 Evaluation

The system automatically generates survey reports that typically include the following; however, reporting can be customized to meet site specific requirements:

- A summary of survey parameters;
- A cumulative frequency distribution plot (CFD) of the survey data;
- A 2-D color image of the survey area results to clearly indicate the location of any residual activity see Fig. 2;
- A statistical summary of the survey data (mean, maximum, minimum, & standard deviation);
- An exception report with a 2-D display of areas over action levels (both 100 cm^2 and 1 m^2);
- If the system used is equipped with SLAM capabilities, 3D pointclouds, and mesh files containing map coordinates and sample results for each 25 cm^2 virtual detector unit.

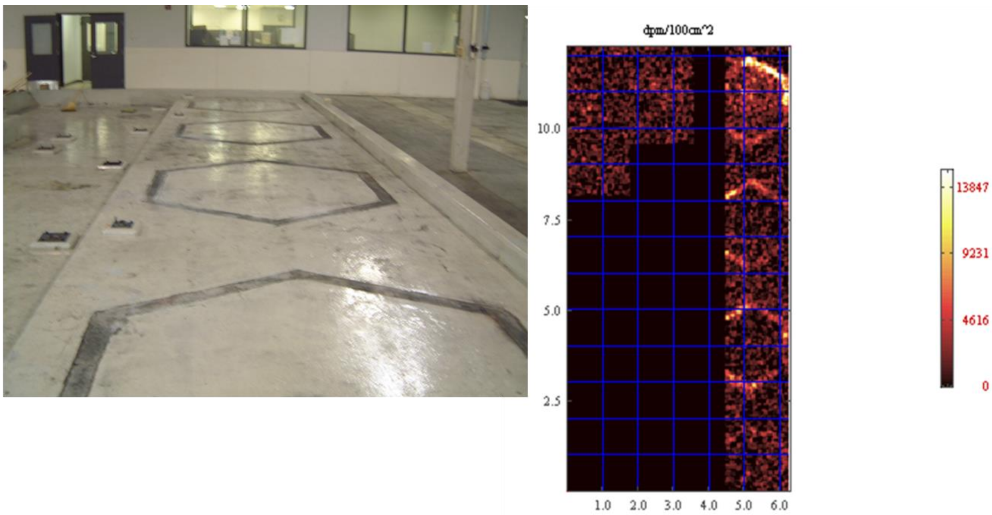

FIG. 2. 2-D Visualization of Survey Data

To meet the challenges of surveying for low-level alpha contamination, the SCM can be operated in the recount mode. The recount mode provides a second detector mounted at a fixed distance behind the primary detector. Data from both detectors is recorded by the SCM in a single pass, effectively performing two surveys concurrently. The processing software then compares the results of the two detectors for each 100 cm^2 area by applying coincidence" logic to identify actual events and eliminate false positives caused by background.

344-CLEARANCE CONCEPT IN TURKEY

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Turkey, like many countries adopted clearance levels for very low level radioactive materials, since if material is classified as radioactive its management can be complex and costly compared to its treatment as non-radioactive. After clearance, radioactive material can be treated as non-radioactive waste and subject to conventional requirements. Radioactive material's clearance is regulated in Turkey by Regulation on Clearance and Removal of Site from Regulatory Control in Nuclear Facilities [1]. Wastes are categorized as radioactive waste if the activity of the radionuclides within exceeds the clearance levels [2]. IAEA RS-G-1.7 for unconditional clearance and European Commission's scenario specific levels for conditional clearance are applied in national regulation [3,4,5]. As to latter case, clearance levels for reuse and recycling of metals, reuse of buildings without demolition for non-nuclear purposes, buildings for which demolition is planned, and building rubbers are all available. If the conditional clearance is to be applied, then the first destination of the material shall be informed to Regulatory Authority. For the clearance of the radioactive material of which the amount is less than 1 tone, the activity concentrations shall be less than exemption levels, as stated by BSS [6]. If the clearance levels for some radionuclides are not presented, then clearance can only be permitted if the public effective dose incurred from those radionuclides doesn't exceed 0.01 mSv/ year. For the measurements regarding the clearance, direct measurements or sampling and laboratory analysis can be carried out, scaling factors and other methods approved by Regulatory Authority can be used. The measurement procedure, including the averaging area and mass, should take into account the type of nuclear facility, the material to be cleared and the radionuclides involved. If the activity is heterogeneously distributed within the material, then;

- For the clearance of metals, averaging masses of 100 kg at most for the activity concentration measurement and an averaging area of 100 cm^2 at most for the surface contamination measurement shall be appropriate,
- For the building an averaging area of 1 m^2 at most for the surface contamination measurement shall be appropriate,
- For the building rubbles averaging masses of 1 tone at most for the activity concentration measurement shall be appropriate.

Representative amount of the sampling has to be done and each measurement's results shall be compared with clearance levels. If the activity is sufficiently homogeneously distributed larger averaging areas (up to 1 m^2) and masses (up to 1 tone) may be appropriate.

Having established recently Nuclear Regulatory Authority which took regulatory functions of former Turkish Atomic Energy Authority (now became Turkish Nuclear Energy Mining Research Authority), new regulatory infrastructure has been in force since 2018 [7]. Clearance will be subject to regulatory approval according to the new system. The existing regulation shall be revised according to new nuclear regulatory system since:

-It doesn't comply with IAEA GSR Part 3 where clearance levels of radionuclides of artificial origin in terms of activity concentration are presented only for bulk amount of materials in Table I.2. [8]
-There is a need in the country to regulate the clearance of radioactive material arising from facilities and activities other than nuclear installations, such as radioactive waste facilities.
-Low probability scenarios have to be taken into account hence a provision for the effective dose expected to be incurred by any individual for such low probability scenarios not to exceed 1 mSv in a year is needed.
-Clearance of natural radionuclides have to be regulated, since rare earth mining facilities in the country, which is regarded as planned exposure situation, will be authorized.

The contaminated site in Gaziemir/Izmir will be regulated under the new system. The source of the contamination has been Eu-152 radionuclide which contaminated lead factory recycling waste scrap batteries. The existence of mixed waste, i.e. both radioactive and hazardous waste (lead and arsenic) makes the situation complicated. The plan is to first segregate the radioactive wastes from the site, to deliver them to radioactive waste processing and storage facility operated by Turkish Nuclear Energy Mining Research Authority, then to transfer all remaining waste on the site, accepted as hazardous non-radioactive waste, to hazardous waste disposal facility licensed by Ministry of Environment and Urbanization and to remove the site from regulatory control. Since the public exposure is regarded "unlikely" at this site, the clearance criterion to be applied for public exposure when segregating the radioactive waste was decided as 0.1 mSv/year [9]. Site release criteria will be determined accordingly. The regulation for authorization of the contained sites is being drafted by Nuclear Regulatory

Authority. It is planned to first authorize the remediation activities and then to remove the site from the regulatory control.

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345-USE OF RSCS SIM-TEQ™ INSTRUMENT SIMULATORS FOR RADIATION SAFETY TRAINING

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The use of radiation instrument simulators provides an effective method for conducting realistic hands-on training in the use of portable instruments and electronic dosimeters. Practical training scenarios can be used without any restrictions, such as those required when using actual radioactive material.

SIM-Teq is a portable wireless training network of simulated electronic dosimeters and survey meters, designed to measure simulated radiation and contamination sources. The network-based system allows the instructor to combine simulated instrument auto-response with manual, remote control capability. The SIM-Teq system enables trainers to create a training environment that includes high fidelity simulated instruments and "live" detectable sources to instill the experience, self-assurance, and real understanding they want trainees to achieve - safely. The instructor is free to teach, observe, and assess the trainee, with the ability to remotely manage, view, or control any SIM-Teq device at any time.

The technology responsible for this realistic training includes ultra-wideband (UWB) radio technology for simulated radiation sources and a combination of RFID and miniature range finders for contamination sources. Multiple simulated radiation and contamination sources can be pre-staged in a training environment, allowing students to enter the area and measure radiation and contamination levels. In contrast, simulated electronic dosimeters accumulate dose associated with their distance to and time near the radiation sources. The simulated instruments use original equipment manufacturer (OEM) components for a high-fidelity response, including meter displays, control pushbuttons and dials, audible and visual alarms, and telemetry system operation for training in high dose rate situations. The system can support up to 10 simulated radiation sources (unlimited contamination sources) and 16 instruments simultaneously in the same training environment and can operate through most standard construction barriers (i.e., walls).

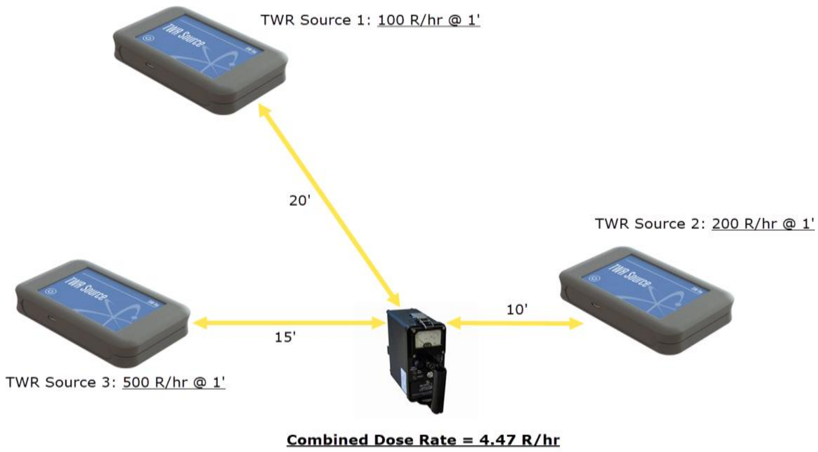

FIG. 47. Example of dynamic complex radiation field combining dose rates from three sources relative to distance and source strength.

Simulated sources are small; about the size of a deck of cards. Sources may be hidden inside backpacks, plastic containers, or PVC piping. UWB signals will not transmit through a solid metal casing, but when placed inside an automobile, enough signal escapes to provide a reading on a meter.

Each simulated instrument is designed to respond to simulated sources with near-identical response times and statistical fluctuations as actual instruments and sources. This fidelity provides the trainee with accurate feedback during a training event for visual meter readouts and audible alarm conditions. Shielded conditions may be simulated by placing materials between (or near) sources and/or meters. The UWB time-of-flight may be delayed slightly, resulting in a lower calculated dose rate. However, it should be noted that the UWB signal is not being affected in the same manner as gamma or X-rays are attenuated when passing through matter, but the resulting effects provide realistic feedback to the trainee.

An alternative solution for simulating dynamic radiation fields is to control the source strength during a training event manually. Each source may be controlled with the control software residing on a tablet computer with user-selected preset values that can be changed as needed by a trainer.

SIM-Teq systems are currently in use at over 20 nuclear facilities in the United States and Canada. The response is positive from students new to the field of radiation safety, allowing SIM-Teq to be part of the process of bringing in a new generation of Health Physics Technicians. It has also received positive responses from both seasoned technicians and First Responders, as it allows them the opportunity to participate in engaging, thought-provoking, and realistic training. One commercial nuclear power plant has found the SIM-Teq system so effective that it now uses it as part of their ALARA reviews for all high dose job planning.

346-ATMOSPHERIC DISPERSION STUDY OF TUNISIAN RADIOACTIVE ISOTOPE COMPANY SISORA

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ABSTRACT

In this work, the radiological evaluation of atmospheric emissions of the Isotope Radioactive Tunisian Company, SISORA and the evaluation of public exhibitions in normal operation were studied. To perform the tasks mentioned above, a "Comply" code simulator (1-2), simulating the Gaussian plume of the air dispersion model. The source term of the radioactive gaseous releases related to the nominal production activities of the SISORA site are beta emitting radioactive elements of short half-life period not exceeding 2 hours, of F-18 (T 1/2 = 109 min) 1000 Ci / year of Ar-41 (T 1/2 = 109.61 min) 23 mCi / year, of O-15 (T 1/2 = 2.3 min) 0,04 mCi / year, of N-13 (T 1 / 2 = 9.7 min) 0,04 mCi / year and C-11 (T 1/2 = 20.4 min) 0,04 mCi / year, during normal operation of a 18 Mev Cyclotron. The results of the simulation showed that the annual committed and external dose received by the individual near the installation 92 µSv / year is well below the regulatory limit 1 mSv/year.

RESULTS

The effective dose for an adult is estimated at 92 µSv / y which represent 9.2 % of the dose limit for this public person.

Based on the dose conversion coefficient adopted by Human Respiratory Tract Model for Radiological Protection (3), the effective dose for a child and a baby is estimated at the following values:

TABLE 3: EFFECTIVE DOSE ESTIMATION (5)

Exposure doses by irradiation	Units	Units Most Exposed housing		
		Adult	Child 10 Years old	Baby 1to 2 Years old
Plume Irradiation + Inhalation	µSv/h	1.049E-02	1.063E-02	1.078E-02
	µSv/j	2.51E-01	2.55E-01	2.58E-01
	µSv/an	92	93.26	94.53

CONCLUSION

The result of this calculation for the public located at the level of the building closest to SISORA which is present for 365 days 24 / 24h (30 meters in the direction of the prevailing wind) is estimated at 92 µSv per year.

SISORA is located in an industrial area, so the assumption of existence of a public person does not exceed 12 hours, which reduces the effective dose received.

For an adult who works 12 hours a day during a year, the effective dose received due to the emissions from the SISORA site at 46 µSv per year.

Effective dose received by a 10-year-old student for 8 hours a day for 5 days a week for 40 weeks, due to emissions from the SISORA site of 16,837 µSv per year

Effective dose received by a baby who remains in a crèche during the work of these parents is 8 hours a day for 5 days a week for 48 weeks, due to the emissions from the SISORA site 20.70 µSv per year.

Knowing that the site is an industrial area and the Cyclotron will be deployed between 2h00 and 5h00 am. A baseline radiological study was performed before the installation of the Cyclotron and periodic tracking of the radiological state of the SISORA site be done.

349-REFERENCE LEVELS AND DOSE CONSTRAINTS APPLIED IN OCCUPATIONAL AND PUBLIC EXPOSURE IN TURKEY

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The experience in regulating of radiation protection of thousands of radiation sources used in medicine and industry in Turkey has now extended to nuclear power plants one of which is under construction, the other of which has completed EIA stage. The regulation related to nuclear facilities mostly comply with 2013/59 EC Directive and IAEA GSR Part 3 [1,2,3]. However the regulation for radiation safety was in compliance with the old BSS, and needed to be revised according to the new international approach [1,2,4,5]. Having established recently Nuclear Regulatory Authority which took regulatory functions of former Turkish Atomic Energy Authority (now became Turkish Nuclear, Energy, Mining Research Authority) in 2018 by the Decree Law 702, the new radiation protection regulation which will cover both nuclear facilities and radiation sources is being developed considering the fact that radiation protection principles are the same for all types of radiation sources [6]. So, graded approach will be the most important principle when implementing this regulation. It aims to improve the existing system especially for radiation sources, and to adapt to new regulatory system established, and taking into account lessons learned from nuclear power plant's licensing. In the paper, existing and new system are presented in terms of dose constraints and reference levels.

Reference levels and dose constraints are used for optimization of protection and safety, the intended outcome of which is that all exposures are controlled to levels that are as low as reasonably achievable, economic, societal and environmental factors being taken into account [1]. These are part of national regulations which are under revision.

Reference levels in the regulation for radiation safety were not meant similarly with the new international approach. Reference levels in the existing regulation included recording levels, investigation levels and intervention levels covering both action levels and guidance levels, and were generally applied for planned and emergency exposure situations. Recording levels were 0.2 mSv/month for radiation workers and 0.01 mSv/year for the public. Investigation levels were 1/10 of the dose limits for radiation workers and the public. Intervention levels were equivalent doses incurred at one time higher than the annual limits or equivalent doses totally incurred in a year higher than the annual limits for radiation workers and the public. Activity concentrations of radon at workplaces were the same with the IAEA GSR Part 3 but not with EC Directive 2013/59; namely the permissible radon concentration at workplaces was regulated as 1000 Bq/m^3 and at dwellings 400 Bq/m^3 was permitted. However these activity concentrations were not called as "reference levels". The permissible effective dose incurred from radon and building materials were not addressed. The values of dose constraints for the public were not presented in the regulation, either [4]. The radiation protection regulation for nuclear facilities is rather new; even so the values of dose constraints for normal operation were absent[3]. Dose constraints are defined to be determined as facility specific by Nuclear Regulatory Authority. In the licensing of NPPs, the need emerged to have a standard public dose constraint for all facilities to be defined in the regulation. Dose constraints for radiation workers shall be determined by the licensee. In emergency exposure situations, the term "reference levels" in EC Directive 2013/59 were replaced in the regulation by "guidance levels" as recommended by IAEA GSR Part 7 [7]. Additionally, an effective dose of 50 mSv for the emergency response actions undertaken voluntarily by emergency workers was also adapted taking into account IAEA GSR Part 7, not EC Directive 3013/59. For transition from emergency exposure situations to existing exposure situations, namely for the areas contaminated as a result of nuclear accidents, 20 mSv/year effective dose for the public and 20 mSv/year equivalent dose for the fetus were already defined as again "guidance levels" in the regulation [3]. Reference levels applied in remediation are not mentioned in both regulations.

The new arrangements related to reference levels and dose constraints can be listed as follows;

-0.1 mSv/year will be fixed in the regulation as the public dose constraint for normal operating conditions of all nuclear, radioactive waste and radiation facilities and radiation applications.

-As to existing exposure situations, the reference level applying to indoor external exposure to gamma radiation emitted by building materials, in addition to outdoor external exposure will be regulated as 1 mSv per year.

-As to existing exposure situations, if the effective doses from radon exposure exceed 6 mSv per year the situation will be accepted as planned exposure situations, and
the reference level for the annual average activity concentration shall not be higher than 300 Bq/m^3 at dwellings.

- Determination of reference levels in remediation of contaminated areas by Nuclear Regulatory Authority will also be stated.

With these new arrangements, it is envisaged to establish a more effective system that will meet the radiation protection needs of the country and ensure the protection of radiation workers, public, environment and future generations from ionizing radiation.

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352-UTILIZATION OF INDONESIA'S LOCAL COMMUNITY IN THE FORMULATION OF RADIATION PROTECTION POLICY THE UTILIZATION OF MUSRENBANG AS AN ADDITIONAL STAKEHOLDER ON RADIOLOGICAL PROTECTION POLICYMAKING IN INDONESIA

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Since 1956, Indonesia has planned to use nuclear as an alternative energy source. Good preparation is attempted according to IAEA procedures, one of which is by establishing the National Nuclear Technology Agency (BATAN) and the Nuclear Energy Supervisory Agency (BAPETEN). Apart from the preparations made by establishing these institutions, in the 1990's, the construction of nuclear power plants still encountered obstacles one of which was rejection from civil society. [1] In 2016, a survey conducted by BATAN showed an increase in community acceptance of the construction of nuclear power plants to above 77.5%. [2]

The increasing of public acceptance on the construction of nuclear power plants also increases the possibility of the realization of the construction of nuclear power plants in Indonesia. Therefore, a long-term plan in the process of operating a nuclear power plant is needed, including a plan for the formulation of a radiation exposure protection policy. In this paper, the author will review the important role of *Musrenbang* in Indonesia as an additional stakeholder in the radiation protection decision-making.

The Development Planning Conference (*Musrenbang*) is a platform used by civil society in Indonesia to convey aspirations in the form of socio-economic development policy advice to governments at the village, city, provincial, and national levels. This platform runs bottom-up so as to produce strong citizen participation. The *Musrenbang* is proven to be effectively used by civil society because of the high likelihood of follow-up on policy advice given by the *Musrenbang* participants to the government. [3] Due to the high effectiveness, the *Musrenbang* has the potential to be used as a platform for formulating policy recommendations regarding radiation exposure.

The idea of involving *Musrenbang* as an additional stakeholder in the process of formulating a radiation protection policy is in accordance with one of ICRP's recommendations on 'Optimization and Protection'. In the recommendation, the process of formulating a radiation protection policy can involve other stakeholders outside the radiation protection expert. The aim of involving other stakeholders is to produce a comprehensive policy that takes into account social, ethical, and transparency considerations.[4]

The involvement of *Musrenbang* in the policy making process will make policy more inclusive because civil society from various backgrounds can be actively involved. Inclusiveness in policy formulation will have a positive impact on policy because it can make radiation protection policies take social aspects into account. This is because the community as the main party that will be directly affected by the radiation protection policy involved in the formulation process. In addition, the use of Musrembang in the process of formulating radiation protection policy recommendations reflects the transparency aspect. this is because the public can be directly involved in the process of preparing policy recommendations. so that, all information considered in the policy formulation process can be accessed by the public.

Conclusion

Increased public acceptance of the discourse of the construction of nuclear power plants in Indonesia has increased significantly. Thus, the possibility of developing nuclear power plants in Indonesia is increasing. Therefore, the government needs to immediately compile a roadmap for the construction and operation of nuclear power plants in Indonesia, including developing a scheme to fomulate a radiation protection policy. This is due to the risk of radiation exposure from the nuclear power plant to the local residents. Through the explanation above, it can be seen that Indonesia can utilize one of the local communities namely *Musrenbang*. The use of *Musrenbang* is in line with ICRP's recommendations on the importance of involving other stakeholders besides radiation protection experts. The *Musrenbang* succeeded in answering this challenge because the involvement of the *Musrenbang* can help formulate a radiation protection policy to be more comprehensive, especially in terms of ethical and social aspects, as well as transparency aspects.

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353-THE CALL FOR GUIDELINES CONTAINING PRACTICAL EXAMPLES OF JUSTIFICATION OF DIAGNOSTIC MEDICAL EXPOSURE IN CASE OF PREGNANT PATIENTS TO FURTHER FACILITATION GSR APPLICATION.

Call for a specific separate elaboration for emergency, dental, extremities, nuclear medicine, oncology etc. to facilitate GSR application in clinical practice and to ensure patient safety (both mother and child) as a strategic priority.

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Decision making and stakeholder involvement are crucial in setting the scene of radiation protection. Regular review of justification decision is needed due to upcoming information concerning e.g. epidemiology or dose estimations. This approach applies as well to justification of the medical exposure of pregnant patients. So far reference guidelines, like iRefere [4], provide detailed recommendation of justification of the medical exposure in general population, with only some general statements regarding pregnant patients. Some excellent material is available, like specific movies eg. “Pregnancy and Radiation” prepared by The Norwegian Radiation and Nuclear Safety Authority and Akershus University Hospital [5]. Nevertheless, it might be beneficial having written, detailed and clear recommendations.

Most diagnostic procedures are associated with doses well below 100 mGy considered as a safe value [1]. IAEA states that X-ray procedures in most cases are safe during pregnancy and IAEA emphasizes that medical radiological procedures with a proper justification should not be avoided during pregnancy [2].

Notwithstanding many international guidelines describing this topic and the procedures for planned exposures of pregnant patients [1], [2], the notification of pregnancy as a relative contraindication to perform medical radiological examinations is stated in National Standard Procedures. This in turn leads to a wrong clinical approach by the personnel involved in the implementation of medical examinations [10], including over-interpretation of existing law regarding the exposure of pregnant patients. For instance diagnostic intraoral examination could be refused due misbelieves that this kind of examination is prohibited by law [9]. On 10 November 2015, The Polish Ministry of Health has published an update of ‘national Diagnostic and Interventional Radiology (D&IR) standard procedures’ [3]. Regretfully, this update has not changed the approach to pregnant patients. In particular the standard procedure no. 65 ‘CT of multiple anatomical parts after polytrauma (abdomen) (3.065)’ indicates pregnancy as a relative contraindication for this examination.[8] CT in emergency situation is lifesaving procedure, so that balancing benefit versus risk should lead to the conclusion about the justification of such emergency CT examination. [8] On the other hand National Basic Safety Standard, so called “atomic Law”, doesn’t prohibit pregnant patients’ examinations, it just limits them to necessary cases.

This reported contraindication create confusion in professional staff, and in fact they have to balance not only benefits and risk for the patient but they also have to keep into account potential legal implication of their decision in terms of either existing law or its interpretations by lawyers. This pose significant pressure on the professional staff and creates real difficulties in the clinical practice. In fact this can led to the situation when the live-saving procedures are not performed or are not performed in due time. Health accidents, due to lack of in-time proper diagnosis of pregnant patients with stroke symptoms, has been reported [11].

In the presentation it is suggested that the new guidelines, which would provide practical examples of justification of diagnostic medical exposure in case of pregnant patients, could contribute to capacity building and enhancement of radiation safety culture in the clinical practice at the healthcare centres. This kind of document would help to further facilitation of the GSR (General Safety Requirements) application and thus securing access to diagnostic procedures for this group of patients. Probably the best approach would be to provide step-by step new specific elaboration for emergency, dental, extremities, nuclear medicine, oncology etc. to facilitate GSR application in clinical practice and to ensure patient safety (both mother and child) as a strategic priority. Such guideline could contain real clinical scenarios and or case-studies presenting justified up-to date examples [6], [7].

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354-ASSESSMENT OF SAFETY IN GAMMA AND ELECTRON IRRADIATION FACILITIES

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An assessment model is developed in this paper for assessment of safety in gamma and electron irradiation facilities. The model consists of a chick-list which require a simple Yes, Partly, or No answer to assess the safety related areas in the irradiation facilities. Methodology of assessment are described and illustrated with examples. It is shown that the assessment model is more realistic and flexible tool to identify areas of conformance with basic requirements and areas needing improvement. This model is applicable to gamma and electron irradiation facilities to improve radiation protection and safety of radiation sources.

INTRODUCTION

Uses of radiation sources in research and industrial irradiation facilities have many beneficial applications. However, protection of workers and public against radiation hazards and safety of radiation sources should be assessed during operation and conduction of activities of these applications [1]. Assessment can be performed through number of questions and discussions with management and personnel and review of documentation.

The assessment of safety in gamma and electron irradiators is developed using a check-list with pre-defined questions or considerations designed based on basic requirements of radiation protection and safety of radiation sources [SGR Part 3]. Conformance with theses requirements would significantly improve the radiation protection of irradiators and safety of radiation sources. The chick-list is developed in this paper using IAEA TECDOCs (1113 and 1367) list of questions to assess specific safety areas in gamma and electron irradiation facilities and also operating experiences of accident causes and lesson learned are included [2]. Overview of irradiation facilities and accidents occurred with operation are provided to understand the types of irradiators and the accident causes and lessons learned [1, 3, 4, 5].

METHODOLOGY

The assessment is intended to be performed periodically during the operation of the irradiators. The main inputs to the assessment are the organization personnel and examination of procedures, records, and documentation of an irradiator and also results of the previous assessment. The model for the assessment of safety in gamma and electron irradiators is based on a check-list approach covering the irradiator safety related areas such as (1) irradiator design, (2) worker and public protection, (3) emergency preparedness, and (4) records. This model provides the operating organization with flexible scope of the assessment, allowing covering an area or some areas of the irradiator. The check-list approach is also applicable to smaller irradiator operating organizations with low number of persons or resources.

In addition to the above mentioned areas, the operating organization can also perform assessment to cover specific safety areas of particular importance to operation of specific irradiator or conducting related activities. The assessment consists of list of questions or considerations which require a simple Yes, Partly, or No answer to assess the irradiator safety related areas. This assessment model aims at identifying the irradiator areas of conformance with the basic requirements in SGR Part 3 (and lessons learned) and areas needing improvements. Then the operating organizations prioritize and focus resources on the areas needing most improvement. For these reasons, this assessment model is a useful tool for implementation of the relevant requirements.

The process of assessment includes (a) preparation of the assessment and defining the scope and assessment areas, (b) execution to provide answers to the check-list questions and considerations in the assessment model, (c) identifying the areas needing improvement, and (d) implementing of improvements [6].

ASSESSMENT MODEL

An example of the developed assessment model is assessment of area1 (IRRADIATOR DESIGN) given in Table 1. Assessment of other areas is also applying similar model and procedures.

TABLE 1. EXAMPLE OF ASSESSMENT OF AREA 1 (IRRADIATOR or ACCELERATOR DESIGN)

List of questions or consideration	Answer		
1. The gamma irradiator model/type and radiation sources, number of sources (per pencil, per module, and per rack), and maximum design, total, and date of activity installed (or the accelerator model/type and energy of radiation) as described in the application approved by the regulatory authority.	Yes	Partly	No
2. A safety assessment was performed prior to any design modifications (if any)	Yes	Partly	No
3. Protection of the sources (or the accelerator) from adverse environmental conditions (heat, moisture, etc.) is provided and working.	Yes	Partly	No
4. Fire detection and protection in the irradiation and source storage areas (or the accelerator irradiation areas)is provided and working.	Yes	Partly	No
5. Adequate ventilation in the irradiation and source storage areas (or the accelerator irradiation areas) is provided and working.	Yes	Partly	No

CONCLUSIONS

An assessment model consists of predefined list of questions based on basic requirements of radiation protection and operating experiences is proposed for assessment of safety related areas in gamma and electron irradiators. The model is also flexible and allows one to add questions to an existing area or to introduce new areas. It is shown that the proposed model is useful tool to identify areas of conformance with basic requirements and areas needing improvement. In conclusion, the proposed assessment model can be used in practices of gamma and electron irradiators in practices of gamma and electron irradiators to improve radiation protection and safety of radiation sources with simple procedures.

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355-GRADED APPROACH IN AUTHORIZATION PROCESS TO REGULATE RADIATION SOURCES IN MEDICAL ACTIVITIES

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In Lebanon there are a large number of radioactive materials and ionizing radiation sources that are used in many fields such as medicine, industry, agriculture, and scientific research in universities and research centres, beside others that fall exclusively within the peaceful uses of atomic energy. The largest number of these sources is used in the medical field due to the necessity of diagnosis, radiography and radiotherapy, particularly for cancerous tumours treatment, where Linacs and PET centres are increasing considerably. In fact, in Lebanon there are hundreds of facilities using thousands of radiation sources where most of them (more than 80%) belong to the medical sector. As for the industrial field, it is used in measuring thicknesses, density, flow rate and packing level in different production plants, as well as XRF-XRD analysis for goods characterization. In addition, non-destructive industrial imaging is practiced by means of different X-rays and radioactive materials. Cement factories, civil aviation, road and oil companies are some examples of local industrial facilities that use radiation sources.

The Lebanese Atomic Energy Commission LAEC was assigned as the national regulatory authority to regulate and control of all practices and facilities that involve the use ionizing radiation sources, taking into account both safety and security when appropriate. This includes as well all related activities such as import, export, transport, storage, radioactive waste management, in line with the International Atomic Energy Agency standards related to nuclear safety and security. In this context, LAEC issues authorizations, regulations and guides to regulate the use of radiation sources, in a graded approach based on the particular activity and related radiation risk, where inspections are often conducted.

Beside a number of applicatory decrees from the council of ministers, LAEC is supported legally in its task by the applicatory decree 15512/2005 which is based on the decree law 105/83. However, the legal framework should be consolidated by a comprehensive nuclear law that is already prepared but it is not yet issued officially. This law is primarily covering Safety, Security and Safeguards concepts as well as liability. Meanwhile, the gaps in legislations are covered by a new applicatory decree and by the safety regulations that are developed based on IAEA GSR part 3 and should be issued soon.

In its role as regulatory authority in Lebanon, LAEC established an effective radiation safety infrastructure inside the country which covers the life cycle of the radiation and radioactive sources including the use, import, export, disposal and transport when applicable. LAEC places high value on all aspects of safety and nuclear security especially, considering the high priority to protect radiation workers, patients, the public and the environment from the hazardous effects of ionizing radiations.

LAEC has the database of radiation sources that is updated as evolving (import, export, transport, change of location, in or out of service, etc.). LAEC has a system of notification for such activities to be initiated by the applicant or licensee. This system is accompanied by inspection and authorization processes where the frequency and requirements could be different depending on the activity and the compliance with the radiation safety regulations, using a graded approach. Among the requirements for authorization in some activities, an emergency exposure plan should be presented. All available data, including doses of workers, is managed by the Regulatory Authority Information System (RAIS). The presentation will focus on the latest LAEC regulatory system based on the graded approach for authorizations.

356-ROLE OF THE TUNISIAN ASSOCIATION OF NUCLEAR SCIENCES AND AWARENESS IN COMMUNICATING BENEFITS AND RISKS OF RADIATION EXPOSURE TO PATIENTS AND PUBLIC.

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1 Tunisian Centre for Nuclear Sciences and Technology

Ionizing radiation arising from all nuclear techniques applications represents a risk to the public either directly or indirectly. Therefore, there is a need to effectively communicate useful information to the wide public especially because Tunisia is considered as a new embarked country in nuclear energy. For all these reasons, the establishment the Tunisian Association of Nuclear Sciences and Awareness was performed in July 2016. It is a non-profit association and aims to raise awareness of the nuclear techniques applications benefits in different domains: energy, health care, environment and agriculture.

The main challenges of the Tunisian Association of Nuclear Sciences and Awareness are the active communication in nuclear field of research with the public concerning the benefits, and the risks associated with any applications involving the use of radiation, we attempt first to vulgarize information's related to ionizing radiation associated to the safety and security. We try to show the benefit and the side effect of nuclear techniques applications. We strive to reassure people worried about ionizing radiation by providing effectively certain notions of nuclear in a comprehensive, timely and professional manner. Figure 1 shows different logistics of communications for different target levels.

FIG. 48. Communication in relation to the audience level

Since the majority of university hospitals are centralized in the capital and the east part of Tunisia, doctors are trying to handle the overload and they have no time to communicate to patients the benefits and the risks associated with any exam involving the use of radiation. As a result, patients decide not to travel from the western or southern part of Tunisia whether by fear of radiations or by fear of waiting. The benefit to the patient of having a speedier diagnosis and/or therapy may not be effectively communicated in a comprehensive, timely and professional manner. In this context, the Tunisian association of nuclear sciences and awareness, among its objectives, addresses the issue of communication of radiation risk and benefits to patients and the basis for such information. While there are different ways of communicating radiation risk, we recognize that certain basic parameters are essential for patients to enable them to make an informed choice about undergoing medical imaging and a therapy investigation under the direction of a well-trained and qualified individual.

In our strategy we propose a progressive communication approach. We first address target public and then the general public. We proceed by inculcating basic concepts associated with safety and security by addressing a target people who might be in the environment of nuclear activities without being directly involved in it such as cleaning agents, waste transporter, guardians or visitor manager for nuclear sites. We also address the general public by organizing events where it was a question of vulgarizing certain concepts related to ionizing radiation and various nuclear activities. Behind our actions we want first to dissipate the fear of nuclear among people. We aspire spreading the basic notions of security and safety in order to establish a culture of tolerance for nuclear activities. We seek to promote nuclear activities in Tunisia by organizing scientific events on nuclear science and technology for peaceful purposes, open days in collaboration with influential partners as well as media information (radio, newspapers, etc.)

The actions that we have carried out have enabled our association to make itself known, gaining public confidence (Table 1). By our actions, we certainly help the government to prepare Tunisia in terms of communication with the public in the event of a nuclear or radiological emergency.

TABLE1: Actions and activities carried out by our association.

Date and Place	Activities	Communication
Activities 2017		
February 2017, Tunis.	29ème Cancer Information Day, "Women's Cancers"	The fear of irradiation or cancer?
March2017, Monastir.	7th Tunisian Days of Nuclear Medicine	Presentation of the Association
April 2017, Tunis.	International Medical Technology Days	Presentation of the Association
August 2017, Beijing, China.	The 25th WiN Global Annual Conference “Women Nuclear Cooperation Harmony”	'Tunisian Women in Nuclear'
September2017 Kraków, Poland.	International conference on Developments and Applications of Nuclear Technologies NUTECH-2017	Presentation of the Association
September 2017, CNSTN, Sidi Thabet.	Open day “the nuclear every day” (radio-Jeune, RTCI, Radio-Diwen)	Awareness of the public and students
October 2017,Monastir.	Symposium "The role of scientific centers and museums in the dissemination of scientific culture in the Arab world”	The role of the Tunisian Association for Sciences and Nuclear Awarness in spreading science and knowledge in the field of peaceful applications of energy and nuclear technologies
Activities 2018		
March, 2018, Bariloche, Argentina.	The 26th WiN Global Annual Conference	'Tunisian Women in Nuclear'
April 2018 Marrakech (Marroco).	Journées Pratiques Francophones de Sciences Analytiques	Presentation of the Association
Activities 2019		
June 2020, Madrid, Spain.	The 27th WiN Global Annual Conference.	'Tunisian Women in Nuclear'
December 2019 Rabat, Morocco.	Conference on “Promoting and Strengthening Women in Nuclear in Africa	Presentation of the Association
Activities 2020		
February 2020, Vienna, Austria.	International Conference on Nuclear Security: Sustaining and Strengthening Efforts	Presentation of the Association

358-ESTABLISHMENT OF CALIBRATION SERVICE FOR RADIATION PROTECTION DOSIMETRY AT THE SSDL OF TAJIKISTAN

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The Nuclear and Radiation Safety Agency (NRSA) is a State Regulatory Authority of Tajikistan which is responsible for the state policy in the field of radiation safety and environmental radiation monitoring. The NRSA closely cooperates with the IAEA through its Technical Cooperation (TC) programme in improving radiation safety and nuclear security. The IAEA TC Project TAD9006 was aiming at strengthening the regulatory regime provided by NRSA for stakeholders in Tajikistan through expanding the range of national capabilities in the calibration of measuring equipment used for radiation protection dosimetry.

One of the project outcomes was to procure, install, and bring to an operational level a gamma beam Cs-137 protection level irradiation (GBI) system in the premises of a Secondary Standard Dosimetry Laboratory (SSDL). The main purpose of the SSDL is to assure traceability of ionizing radiation measurements to the International System of Units through reliable calibration services for end-users in Tajikistan.

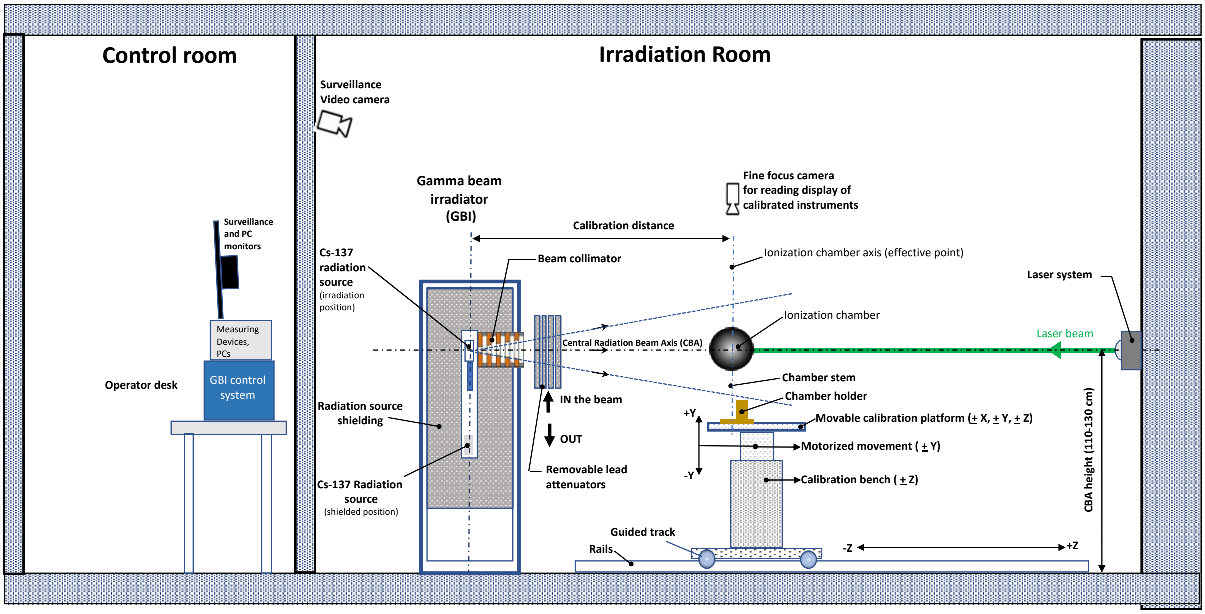


FIG. 1. Schematic drawing of the irradiation room with different components of calibration system.

GAMMA RAY PROTECTION LEVEL CALIBRATION. SYSTEM

The GBI system consists of one irradiation room joint through the controlled area and the main corridor to the GBI control room. The irradiation room is equipped with a gamma beam irradiator that includes a collimated Cs-137 radiation source, a calibration bench for positioning of calibrated items up to the distance of 5 m from the radiation source, and various ancillary and safety equipment. The control room contains a control system for the GBI, dosimetry equipment, and other ancillary equipment. The layout of the irradiation room with various GBI system components is shown in Fig.1.

The GBI include Cs-137 source with a nominal activity of 740 GBq. The size of radiation beam can be modified by two sets of removable beam collimators allowing to achieve the field size up to 50 cm on the central radiation beam axis at the distance of 300 cm. The air kerma rate can be modified by the calibration distance (1m to 5m) and a set of six beam attenuators allowing to decrease the radiation beam intensity by a factor up to 1000. The irradiator is operated through a control system that has connections to various interlocks and the GBI system components. The irradiation time could be set up between 1 sec and 999999 sec. The control system contains an electro-pneumatic switch that controls the actual position of the source.

REFERENCE STANDARDS

The SSDL has three air vented cylindrical ionization chamber used as reference dosimetry standards for protection level dosimetry: (i) PTW 32005, with sensitive volume of 30 cm3; (ii) PTW32002, 1000 cm^3, and (iii) PTW 32003, 10 liter including PTW UNIDOS electrometer. The dosimetry standards are traceable to the International Measurement System (IMS) through calibration at the IAEA Dosimetry Laboratory.

AIR KERMA RATES

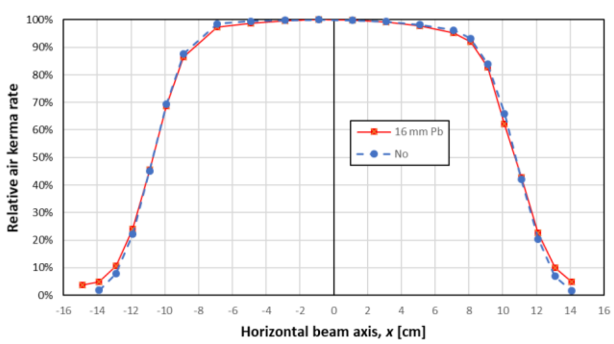
The air kerma rate values on the central beam axis at a distance > 200 cm from the radiation source were determined using the PTW 32002 (#341). The PTW32005 (#163) was used for the air kerma rate measurement at the distances below 200 cm. The selected chamber was connected to the PTW UNIDOS (#20863) electrometer. The polarizing voltage of each chamber was setup to +400 V. The results are given in Table 1.

TABLE 1 The measured air kerma rate values (10 October 2019)

Coll.	d	[cm]	50	70	100	150	**200**	250	300	350	400	450	500
SC	$\dot{K}_{a,SC}$	[µGy/min]	3428	1758	869	387	**217.7**	139	96.3	70.8	54	42.5	34.7
BC	$\dot{K}_{a,BC}$	[µGy/min]	3500	1793	887	393	**222.0**	141.9	98.3	72.2	55.1	43.6	35.9
$(BC/SC)_{ratio}$		[ ]	1.021	1.02	1.021	1.014	1.02	1.021	1.021	1.02	1.022	1.026	1.035

RADIATION BEAM PROFILES

The beam profile measurements were perfomed using a spherical ionization chamber the PTW 32005. Each point shown in the Fig. 2 represents the average of four chambers readings corrected for the effect of ambient temperature and pressure and normalized to the reading on the central radiation beam axis (CBA)



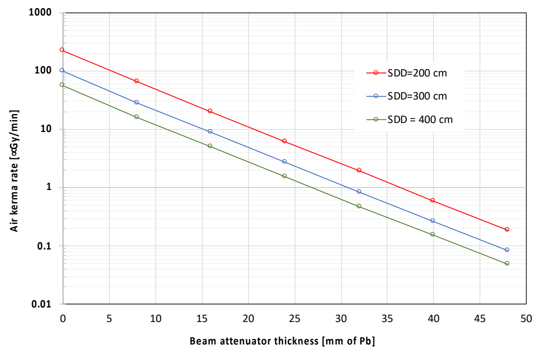


FIG.2. Left - comparison of the horizontal beam profiles measured with and without the beam attenuator at the distance of 100 cm from the source; Right - air kerma rate measured at different distances and thickness of the beam attenuator.

CALIBRATION AND MEASUREMENT CAPABILITIES

Calibration and Measurement Capabilities (CMCs) were designed according to the instructions described in the document “Calibration and Measurement Capabilities in the context of the CIPM” published at the BIPM website in 2017 [1]. The SSDL will calibrate radiation protection instruments used for personal, field and environmental monitoring of photon radiation (i.e. portable radiation survey meters, personal dosimeters).

The CMCs designed for the SSDL of Tajikistan includes seven entries in the field of radiation protection dosimetry (Tab.13). The calibration and measurement services include calibration of the instruments in terms of air kerma, ambient dose equivalent and personal dose equivalent in Cs-137 gamma rays in a wide range of dose rates.

TABLE 2 Calibration and Measurement Capabilities

Calibration or Measurement Service			Measurand Level or Range			Measurement Conditions/Independent Variable		Expanded Uncertainty				Comments (*to be published via the web page)*
Quantity/ Class	Instrument or Artifact	Instrument Type or Method	Minimum value	Maximum value	Units	Parameter	Specifications	Value	Units	Coverage Factor	Level of Confidence	
Air kerma rate	Ionization chamber	Calibration against a secondary standard free in air	6E-02	8E-01	mGy/min	Cs-137	ISO 4037: 2019; 740 GBq (2018); 1m to 4m distance, without Pb absorbers	**1.3**	%	2	95%	Radiation protection level calibrations for ionization chambers
Air kerma rate	Dosemeter	Irradiation in a calibrated field free in air	5E-06	2E-01	Gy h^{-1}	Cs-137	ISO 4037: 2019; 740 GBq (2018); 0.5m to 5m distance; with and without Pb absorbers	**1.6**	%	2	95%	Radiation protection level calibrations for survey meters, dosemeters
Air kerma	Dosemeter	Irradiation in a calibrated field free in air	5E-06	1E00	Gy	Cs-137	ISO 4037: 2019; 740 GBq (2018); 0.5m to 5m distance; with and without Pb absorbers, tirr> 10 sec	**1.6**	%	2	95%	Radiation protection level calibrations for survey meters, dosemeters
Ambient dose equivalent rate	Survey meter	Irradiation in a calibrated field free in air	6E-06	6E-02	Sv h^{-1}	Cs-137	ISO 4037: 2019; 740 GBq (2018); 1m to 3m distance; with and without Pb absorbers, tirr > 10 sec	**4.5**	%	2	95%	Radiation protection level calibrations for survey meters. Traceable to the air kerma standard at the IAEA with applied conversion factor from air kerma to ambient dose equivalent.
Ambient dose equivalent	Survey meter	Irradiation in a calibrated field free in air	5E-06	1E00	Sv	Cs-137	ISO 4037: 2019; 740 GBq (2018); 1m to 3m distance; with and without Pb absorbers	**4.5**	%	2	95%	Radiation protection level calibrations for survey meters. Traceable to the air kerma standard at the IAEA with applied conversion factor from air kerma to ambient dose equivalent.
Personal dose equivalent rate at 10 mm depth	Personal dosimeter	Irradiation in a calibrated field on the surface of the ISO rod (finger) phantom	1E-05	7E-03	Sv h^{-1}	Cs-137	ISO 4037: 2019; 740 GBq (2018), 3m to 4m distance, with and without Pb absorbers	**5.5**	%	2	95%	Radiation potection level calibration for passive personnel dosimetry monitoring devices.Traceable to the air kerma standard at the IAEA with applied conversion factor from air kerma to personal dose equivalent.
Personal dose equivalent, at 10 mm depth	Personal dosimeter	Irradiation in a calibrated field on the surface of the ISO slab phantom	5E-06	1E-01	Sv	Cs-137	ISO 4037: 2019; 740 GBq (2018), 3m to 4m distance, with and without Pb absorbers, tirr > 10 sec	**5.5**	%	2	95%	Radiation potection level calibration for active and passive personnel dosimetry monitoring devices.Traceable to the air kerma standard at the IAEA with applied conversion factor from air kerma to personal dose equivalent.

QUALITY MANAGEMENT SYSTEM

The Quality Management System of the SSDL was prepared in 2019 and accredited by the National Accreditation Agency (NAA) of Tajikistan according to IEC/ISO 17025 [2] in 2020.

CALIBRATION SERVICE

The calibration of survey meters is performed in terms of ambient dose equivalent rate $\dot{H}^*(10)$. The expected value of the ambient dose equivalent rate at the reference point of the measurements is calculated from the air kerma rate determined with the reference ionization chamber on the radiation beam axis at the reference distance and multiplied with the appropriate value of the conversion factor $h^*_K(10, Cs)$

The calibration of an electronic personal dosimeter (EPD) is performed in terms of personal dose equivalent $H_p(10)$. The EPD was placed on the surface of a water phantom (30cm x 30cm x 30cm) at the position of the central radiation beam axis. The personal dose equivalent at the reference point of the measurements is calculated from the air kerma rate determined with the reference ionization chamber on the radiation beam axis at the reference distance and multiplied with the appropriate value of the conversion factor $h_{pK}(10, Cs, 0°)$.

The operational dose quantities were calculated from the air kerma using appropriate conversion coefficient [Sv/Gy] published in the ISO 4037-3 [3].

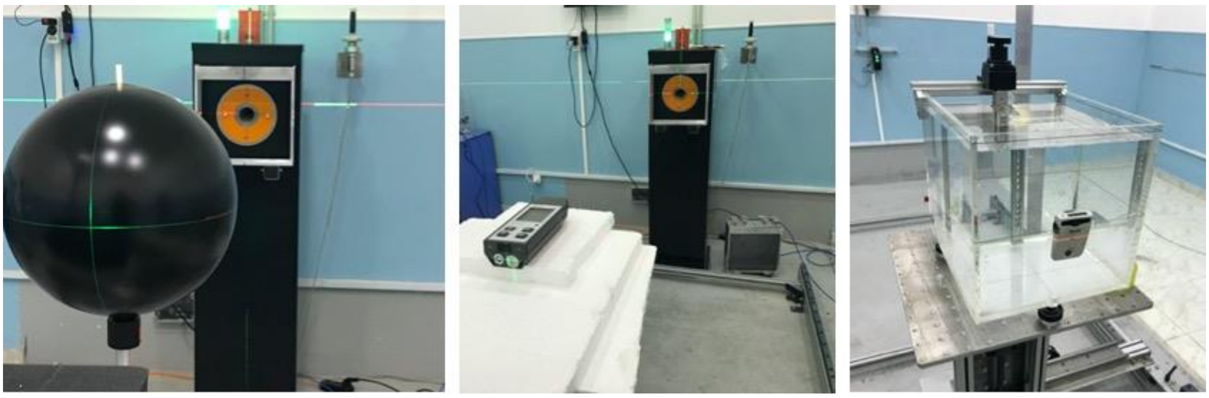


*FIG. 3. Calibration service provided by the SSDL of Tajikistan for different type of protection level dosimetry systems: LEFT – ionization chambers in terms of air kerma, K_a; MIDDLE – Radiation survey meters in terms of ambient dose equivalent, H*10; RIGHT – personal dosimeters in terms of personal dose equivalent, H_p10*

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362-STAFF EYE-LENS DOSE MONITORING DURING CARDIOVASCULAR AND NEURORADIOLOGY INTERVENTIONAL PROCEDURES

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INTRODUCTION

The ICRP 118 lowered the eye-lens equivalent dose limit for professional exposure from 150 to 20 mSv/year, averaged over defined periods of 5 years, with no single year exceeding 50 mSv. This change can be particularly relevant for medical staff involved in interventional radiology procedures. In this work we estimate eye-lens dose absorbed during cardiovascular and neuroradiology interventional procedures, using personal dosimeters placed outside non-shielded eyeglasses, close to the eye-lens. Results are reported for each staff member, for single procedure and normalised to KAP value. Effects of eye lens shielding and suspended ceiling are considered as well.

MATERIALS AND METHODS

Dosimeters were assigned to each staff member involved in the procedure during x-ray emission (Surgeon, Surgeon Assistant, Anesthetist, Nurse and Radiology technician (if present in surgical room): 3 dosimeters were attached on the left, front and right sides of non-shielded eyeglasses frames (LiF100 thermo luminescent dosimeters, model Ext-Rad, Harshaw Thermofisher); 1 reference dosimeter was attached externally to the lead apron on the left upper chest side and 1 close to the thyroid. The dosimeters were provided by a ISO-17025 accredited Dosimetry Service.

Measurements were carried out without interfering with the conduct of the monitored procedures. 30 Studies (mean KAP 40 $Gy \cdot cm^2$), 15 Arteriovenous Malformations (AVM, mean KAP 90 $Gy \cdot cm^2$) and 15 Aneurysms (ANE, mean KAP 130 $Gy \cdot cm^2$) for neuroradiology interventional procedure were supervised, performed using a biplanar angiographic system (model Philips Azurion), with skirt and ceiling-suspended shielding (0.5 mm Pb). 8 Endovascular Aortic Reconstruction (EVAR, mean KAP 450 $Gy \cdot cm$) for aortic surgery were supervised, performed with an angiography system (model Philips Allura Clarity), with skirt shielding (0.5 mm Pb).

RESULTS

The most exposed eye lens dose per procedure was found to be similar for Surgeon and Surgeon Assistance, in case of Studies, ANE and AVM, with values per procedure centered around 0.020 mGy, 0.04 mGy and 0.060 mGy respectively (procedures performed with the use of ceiling-suspended barrier, which attenuating factor was estimated to be 50). The eye-lens dose of Surgeon doubles that of Surgeon Assistance in case of EVAR (0.74 mGy versus 0.310 mGy) procedure, carried out without any shielding equipment. From dosimetric reading the eye-lens dose for Nurses and Radiographer was estimated less than 0.050 mGy/procedure in every kind of procedure. The eye-lens dose for Anesthetist is strongly dependent on the time spent in the surgical room with X-rays on, and hence on the typology of the procedure, ranging from around 0.01 mGy in case of Studies and ANE to around 0.25 mGy in case of EVAR.

CONCLUSIONS

The new eye-lens exposure equivalent dose limit of 20 mSv/year may be easily exceeded for Surgeon and Surgeon Assistance in interventional radiology procedures performed without an effective shielding equipment and/or Personal Protecting Equipment (PPE). Depending on the workload, a personal eye-lens monitoring should be considered for Surgeon and Surgeon Assistance, who can exceed the eye-lens dose annual limit after a few numbers of procedures (around 10 for ANE and AVM, around 20 for Study and EVAR, in absence of any shielding equipment).

For Nurse and Radiographer, the eye-lens dose may be estimated on the basis of dosimetric measurements performed on thyroid or external apron, while eye shielding equipment is required only for a large number of annual procedure, that we estimated to be around 1000 Studies and 500 ANE, AVM and EVAR procedures for Nurses and 800 EVAR procedures for Radiographer.

For Anesthetist, no eye shielding equipment is needed in case of Studies, ANE and AVM. However, in case of EVAR, more than 70 procedures a year could lead to an excess of eye-lens dose above the annual limit.

Our work highlights that adoption of PPE with an effective geometrical shielding respect to the scattered radiation is crucial.

363-THE IAEA PROGRAMME ON INTERNATIONAL DATA COLLECTION AND MODEL TESTING FOR RADIOLOGICAL ENVIRONMENTAL IMPACT ASSESSMENT

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Radiological Environmental Impact Assessment (REIA) is used to assess exposures to humans and impacts on the environment in planned, existing and emergency exposure situations. The results of such assessments are used, for example, in the evaluation of the radiological impact of routine and accidental releases of radionuclides, setting authorizations of radionuclide discharges for planned activities at nuclear facilities, to support decision making in the management of existing exposure situations, including remediation, for the performance assessment of nuclear waste disposal facilities, as well as for clearance and exemption of material with low levels of radioactivity.

The IAEA has published three General Safety Guides to provide generic guidance on meeting the requirements of IAEA Safety Standards Series No.GSR Part 3 for protection of the public and protection of the environment. These Safety Guides, together with the requirements of GSR Part 3, provide a basis for including environmental considerations in the assessment and management of radioactive releases. The Safety Guide GSG-10 [1] describes an approach for REIA to prospectively evaluate radiation exposures and radiation risks due to radioactive releases to the environment from new or existing facilities and activities from which the public and the environment might be exposed to radiation.

Radiological exposures to humans and impacts on the environment are the result of a complex interaction of the properties of the radionuclides involved, environmental conditions, agricultural practices and human habits. Environmental transport and dose assessment models are used alongside measurements of radionuclides in the environment to undertake REIA. Many model parameters are needed to characterize the specific exposure conditions and to quantify the transfer of radionuclides within an ecosystem. The results provided by such models are necessary to prove compliance with regulatory standards, to support decisions during and after nuclear emergencies and to optimize, for example, the remediation of contaminated sites. All estimated exposures are uncertain to some extent because the parameters used to calculate them are subject to a more or less pronounced variability due to the inherently incomplete knowledge about the exposure conditions. The uncertainties in parameter values used in the assessment of exposure resulting from the lack of site specific data are compensated for by the use of generic data with cautious assumptions; however, it is important that these should not be overly pessimistic.

The IAEA has been organizing programmes of international data collection and model testing for REIA since the 1980s. These programmes have contributed to a general improvement in models, in the collation of consensus data for use in REIA and in the capabilities of Member States to undertake such assessments. IAEA publications on this subject over the past three decades demonstrate the comprehensive nature of the programmes and record the associated advances which have been made. The IAEA programmes have had the following general objectives:

- To improve environmental assessment models and modelling methods by model testing, comparison and other approaches;
- To develop international consensus, where appropriate, on environmental modelling philosophies, approaches and parameter values;
- To develop methods for the assessment of radionuclides transfer in the biosphere in areas where such methods were not already available;
- To provide an international focal point for the exchange of information on environmental assessment modelling;
- To respond to environmental assessment modelling needs expressed by other international organizations and entities.

The most recent programme, MODARIA (Modelling and Data for Radiological Impact Assessment) programme was set up to continue the IAEA's activities in the assessment and management of environmental releases and, specifically, to address the following needs:

— support and facilitate the implementation of GSR Part 3 regarding exposures to the public in planned, existing and emergency exposure situations, as well as regarding radiological impacts to the environment;
— strengthen Member States' capabilities for the assessment of exposures to the public and radiological impacts to the environment as recommended by the IAEA Action Plan on Nuclear Safety;
— provide harmonized and easy-to-use assessment tools to support decisions on radiological issues;
— provide technical guidance to Member States launching, or planning to launch, nuclear power programmes on assessing radiological impacts arising from discharges of radionuclides into the environment;
— provide technical guidance to Member States across the world control residues containing enhanced levels of natural radioactivity that are produced during industrial activities or during mining of metals and uranium.

A large number of scientific papers have been developed and published by participants in the course of their work carried out within MODARIA and earlier programmes which underlines the general scientific importance of the programme. The IAEA publishes the output of the work undertaken in the Technical Document Series and compilations of date in the IAEA Technical Report Series, eg., on 'Parameter Values for the Prediction of Radionuclide Transfer in Terrestrial and Freshwater Environments' [2]. Summary reports of the key outputs of the Programmes are also published [3,4]. The main topics for which technical guidance and supporting data have been provided within the MODARIA programme are:

— approaches for the assessment and decision making of existing exposure situations for NORM and nuclear legacy sites using case studies;
— assessment of Exposures and Countermeasures in Urban Environments: testing of assessment models using data from case studies;
— assessment methodologies and tools for evaluating exposures to wildlife for planned releases to the environment;
— transfer processes and data for radiological impact assessment, specifically a database of k_d values for terrestrial and aquatic environments; a review of radioecology data for radiocaesium from the Fukushima Daiichi nuclear accident; and environmental transfer parameters for non-temperate areas;
— models and assessment approaches for assessing exposure and radiation effects to wildlife;
— an update to the BIOMASS 2005 methodology for biosphere modelling for long term safety assessments of high level waste disposal facilities;
— modelling approaches and model testing for the assessment of the fate and transport of radionuclides released in the marine environment.

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364-NATIONAL ACTION PLAN FOR CONTROL OF PUBLIC EXPOSURE TO RADON IN THE CZECH REPUBLIC

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The National Action Plan for Radon Exposure Control (hereinafter referred to as "RANAP") is a follow-up to the Radon Programmes of the Czech Republic, which were implemented on the basis of the Government Resolution between 2000 and 2009 and between 2010 and 2019.

The National Action Plan for Radon Exposure Control came into effect on 1 January 2020.

RANAP is aimed at controlling the public exposure to radon in buildings with residential rooms or rooms intended to be occupied by persons, school facilities, buildings providing social or health services and in workplaces with increased radon exposure.

We defined three long-term objectives of RANAP:

1. Informed and communicating state administration, involved public, educated professionals
2. Effective prevention in the construction and reconstruction of buildings
3. Effective control of existing exposure

Radon exposure often contributes significantly to the magnitude of public exposure. It is realistic and effective to reduce it, thereby reducing the potential health risks for the population.

Due to its specific geological subsoil, the Czech Republic (hereinafter referred to as the "CR") is one of the countries with higher levels of exposure to this source in the world. A long-term stay in buildings with an increased activity concentration of radon, which may be households, schools, workplaces, etc., and which are not sufficiently protected against the penetration of radon from the subsoil, is a risky situation. According to the latest estimates of the National Radiation Protection Institute, more than 4.5% of the housing stock in the Czech Republic is overburdened with radon.

In the Czech Republic, the legislation defines areas with increased risk from radon, namely individual municipalities, and explicitly stipulates obligations for the operators of workplaces on the underground or first floor of a building located in the areas of these municipalities. In these areas, the probability of exceeding the reference level set for radon is higher than 30%.

As a result of lifestyle changes in recent decades following the energy-saving measures for buildings, the radon load needs to be examined and appropriate measures proposed in case of its potential. [1]

LEGAL FRAMEWORK

The responsibility of The National Action Plan for Radon Exposure Control („RANAP") is established on the Act No. 263/2016 of Coll., Atomic act and Decree No. 422/2016 of Coll., on radiation protection and security of a radioactive source.

The law was in force on 1st January 2017. The RANAP is the first time included in the atomic act. The new atomic law based on our acquired experience and at the same time adopts the of Europe union legislation. The RANAP is in accordance with the requirements of COUNCIL DIRECTIVE 2013/59/EURATOM of 5 December 2013 laying down basic safety standards for protection against the dangers of exposure to ionising radiation, in accordance with the IAEA (International Atomic Energy Agency) document "Safety Standards General Safety Requirements Part 3.5 Existing exposure situation"

The issue of RANAP solves in cooperation with other authorities such as the Ministry of Industry and Trade, the Ministry of the Environment, Ministry of the Health, Ministry of the Finance, and Ministry of the Agriculture, Ministry for Regional Development and also in collaboration with regional authorities.

RANAP as a strategy document is part of the project call Radon program.

This Ministries shall participate in informing and educating the public and professional groups in the area of protection against exposure to radon and in developing methods and technologies for reducing this exposure.

RANAP defines part of the area which should the ministries support. The support could be as financial or by the project.

ACTIVITIES TO FULFILL RANAP IN 2020

We are concentrating on communication strategy for Radon workplace in 2020. We made an interesting micro webpage to help all people easily understand the topic, to find the measuring company, to identification workplace in the prone area map and use interactive forms to registration. The target included also help the employer step by step what is essential to do. Another interesting project is educative video for a building architect and builder expert. The professor from the university will teach and explain the main rule for protection on radon and remediation. Also, we improve the radon map, we provide a free long time measurement by the passive detector for people and school and more. The Czech Republic supports the Radon program more than 770 thousand EUR every year. We hope the support will also next years despite the COVID situation in the world.

The design of the website you can see below fig. 1 you should find on https://www.radonovyprogram.cz/pracoviste/ or by the QR code fig. 2.

FIG. 2 Part of www.radonovzprogram/pracoviste/

FIG. 49. QR code.

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365-HELPING HOSPITALS IMPROVE THE ENVIRONMENT FOR RAISING CONCERNS

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The presentation shares personal experience of challenges encountered by a Radiation Protection and Medical Physics Expert when applying the system of radiological protection to protect patients in a hospital, which had recently introduced a new senior management structure. These challenges relate to organizational culture and response to concerns he had raised whilst seeking to encourage radiation safety improvements. It describes how the national healthcare system is trying to improve 'speaking up' culture to make it safe for staff to raise concerns in the public interest.

Whilst events outlined took place in one country the author is aware that underlying issues are global (and not confined to radiation safety). He has been encouraged by international colleagues to raise awareness of these challenges so that they can be addressed by the international radiation protection community. He suggests possible solutions, in keeping with recent IAEA initiatives to improve radiation safety culture in medicine [1], [2].

The UK National Health Service (NHS) was set up in 1948 and is the world's fifth largest employer. It has a workforce of about 1.3 million people, undertaking some 350 different roles [3]. As with other healthcare systems it uses radiation widely, for diagnosis, treatment, and research. The NHS is a complex organization that is constantly changing. In England, its radiation services are mostly delivered by about 200 local healthcare providers, with central strategic direction. A recent report notes that the quality of NHS management is an issue of considerable national importance [4].

ISO 9000 quality management standards define *management* as coordinated activities to direct and control an organization [5]. The IAEA Safety Glossary describes *management system* as 'a set of interrelated or interacting elements for establishing policies and objectives and enabling the objectives to be achieved in an efficient and effective manner' [6]. Its description adds that components of an organization's *management system* include organizational structure, resources, processes, personnel, equipment and culture - and that its processes have to address the totality of the requirements on the organization as established in, for example, IAEA safety standards and other international codes and standards. It refers to *safety culture* as being the assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receive the attention warranted by their significance.

ICRP state that the *System of Radiological Protection* contributes to an appropriate level of protection from harmful effects of exposure to ionising radiation; is based on scientific knowledge, ethical values, and more than a century of practical experience; and forms the basis of standards, regulations, guidance, programmes, and practice, worldwide [7]. The overall *System of Radiological Protection* is described in ICRP Publication 103 [8]. This explains the three fundamental universal *principles of radiological protection*: justification, optimisation, and limitation.

IRPA have published guidance on the important topic of radiation safety culture [9]. In this guidance they note that culture can exist in several possible development stages, and that *RP professionals within an organization must take the central role in supporting management to drive and embed radiation protection culture throughout the organization*. Their guidance [9] refers to and describes traits of various types of radiation protection cultures: Pathological, Reactive, Calculative, Reactive and Generative, noting that *the objective of any culture development program is to move the organizational and individual behaviors towards the highest stage* [10].

IAEA have recently developed training material aimed at improving safety culture in medicine - their website notes that *the concept of safety culture is well-known in nuclear installations - less so in healthcare* [2]. Digital presentations accompanying this training material will include contributions from winners of its 2019 competition: *Towards a strong radiation safety culture in medicine* [1]. Competition entries were divided into 10 safety culture traits identified as essential for improving radiation safety culture in medical institutions:

— Personal accountability;

- Questioning attitude;
- Effective safety communication;
- Leadership safety values and actions;
- Decision-making;
- Respectful work environment;
- Continuous learning;
- Problem identification and resolution;
- Environment for raising concerns;
- Work processes.

The IAEA Radiation Protection and Safety of Radiation Sources *International Basic Safety Standards* require organizations using radiation to promote and maintain safety culture and outline (p.28) how this is to be achieved [11].

A 2015 independent review aimed at fostering an open and honest reporting culture in the NHS, the *Freedom To Speak Up* Review, revealed serious problems in NHS culture for raising concerns [12]. The review recommended implementation of a series of principles and actions aimed at achieving the culture change which it identified as being required to make it safe, and normal practice, for staff to 'speak up'. The poster outlines findings of this review and progress in implementation of its recommendations, illustrated by unexpected personal experience when seeking to maintain and develop local radiation safety culture and practice in line with IAEA, ISO, ICRP, IRPA and WHO precepts, and the Bonn Call for Action [13].

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366-IMPROVING OPERATIONAL EFFICIENCY, COMPLIANCE AND COMMUNITY PERCEPTION WITH REAL-TIME RADIATION AND RADON DATA

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The nuclear and medical radiation industry has always faced a significant burden in effectively balancing exposure and productivity. Maintenance operations and development of new techniques require significant staffing resources to monitor and review personnel exposures and changes in operational radiation levels. Currently utilised monitoring technology and review techniques remain unchanged since the 1950s, relying upon mostly analogue instrumentation and paper reporting. After utilising isolated area monitoring during a reactor decommissioning project, which continually failed without warning due to dirty power, I began investigating how current shortcomings could be addressed, security increased and exposure risks minimised, by utilising real-time monitoring and automated reporting.

The community and media expectations of reported data and nuclear monitoring systems continue to be coloured by sensational technology seen in movies and emerging technology available through smart homes and smart cities initiatives. In order for the general public to be more engaged with real radiation levels and effects, the nuclear industry needs to more effectively communicate safe levels by having greater control of reporting exposure levels through the controlled release of background and expected exposure levels. An increase in real world monitoring data communicated in a controlled and more effective way will aid in the general public perception around nuclear industry and other radiation facilities. This is particularly pertinent as Australia seeks to implement its Nuclear Waste Facility, community acceptance being the number one inhibitor of being able to move forward.

Increasing operational effectiveness while delivering operational cost savings sounds like an attractive proposition. Increased live data can be utilised to identify optimum practices for reducing operational and environmental exposures or releases. By reducing exposure and the detachment of required health physics resources from routine operations,there is greater availability for these resources to support maintenance operations and project-based work. The review of this increased data will create internal capability to identify low and high risk activities, resulting in the ability to reduce the number of highly skilled workers in an industry where the availability of this talent is reducing.

What stops industry from implementing technology that encourages agile work practices, while minimising security risks and personnel exposure?

FURTHER INFORMATION

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This body of work is an accumulation of radiation management and safety communication initiatives practiced in community, operations and researcher engagement while improving and articulating radiation assurance.

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367-COMMISSIONING DOSIMETRY AND DOSE MAPPING OF THE TUNISIAN GAMMA IRRADIATION FACILITY AND COMPARISON WITH MCNP MONTE CARLO SIMULATION

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Abstract

Cobalt-60 irradiation facility has been put into operation at the National Centre of Nuclear Sciences and Technology at Sidi-Thabet in Tunisia. Technical specifications were checked by dosimetry commissioning experiments and compared to Monte Carlo simulation data using MCNP. Installation qualification has been carried out to measure absorbed dose distribution in the irradiation cell and products. Two dosimeter systems were employed for measurements: Red and Amber Perspex and Cellulose Triacetate (CTA). The regions of minimum and maximum absorbed dose within a homogeneous dummy products and the dose uniformity ratio were determined. The products were loaded in carrier by sawdust with a bulk density of 0.114 g/cm3, potato with a bulk density 0.58 g/cm3 and syringe with a bulk density 0.123 g/cm3. The isodose curves and the three-dimensional views were built using the kriging method. An acceptable agreement in most sets between Monte Carlo simulation results and the experimental values.

Keywords: Dosimetry commissioning; Monte Carlo simulation; MCNP6.1; Isodose curves; Dose uniformity process, Gamma-irradiation facility

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368-RADIATION DOSE OF PO-210 INHALED BY TUNISIAN SMOKERS INTO LUNGS.

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The number of lung cancers is increasing in the world. Cigarette smoke is considered one of the causes of lung cancers. Among the several hazardous chemicals found in cigarettes, polonium 210, a natural radionuclide, is found in cigarette smoke and inhaled into smokers' lungs. The ^{210}Po is the most element investigated in cigarettes in the literature. This work aims to estimate the radiation dose of the intake of Po-210 by Tunisian smokers. Eleven brands, local and imported, were bought from the local market. Polonium 210 was determined by alpha spectrometry using PIPS detector after chemical treatment and spontaneous deposition in stainless disc. The activity concentrations of ^{210}Po ranged between 11.7±0.5 mBq.g^{-1} and 25.3±0.8 mBq.g^{-1}. The yearly effective dose were calculated in the basis that 22% of the concentration of ^{210}Po are retained into the lungs as reported Kubalek and the conversion factor is 3.3 μSv.Bq^{-1} recommended by ICRP (2012) [1,2]. Assuming that a smoker who consumes one pack of cigarettes bought in Tunisia per day, the average effective dose is about 90.6±3.3 μSv. This value is low compared to the values of the effective dose reported in Hungary, Egypt, Philippines and Vietnam corresponding respectively to 185.6 μSv, 190 μSv, 210 μSv and 223 μSv per year [3-6].

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369-DECAYED, EFFECTIVE AND RESIDUAL RADIOACTIVITIES OF SELECTED ^{99M}TC-LABELLED RADIOPHARMACEUTICALS AND THEIR IMPLICATIONS FOR NIGERIAN NUCLEAR MEDICINE PRACTICE

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The need for more formal documentation of nuclear medicine practice around the world has been highlighted by some authors [1]. It has also been suggested that data should be collated from different operational facilities to establish more specific Diagnostic Reference Levels (DRLs) so that practice in different jurisdictions can be properly regulated [2], [3]. The Nigerian practice is beleaguered by a myriad of challenges that force frequent operational disruptions [4], and consequent paucity of research. The country's regulatory bodies have not yet recommended any DRL for the Practice. The IAEA recommends that administration of radiopharmaceuticals should closely follow their preparation [5]. The presentation therefore aimed to report the radioactivity levels of frequently used radiopharmaceuticals in the Nigerian Practice, suggest modifications where necessary, and provide guidance levels.

The presentation emanated from a prospective clinical experimental study of 153 ^{99m}Tc-labelled radiopharmaceuticals prepared and administered to patients at the University College Hospital Ibadan. This was the ***only fully operational*** facility in Nigeria during the study period (July – September 2019). The radiopharmaceuticals included in the study were ^{99m}Tc-MDP, Sodium Pertechnetate, ^{99m}Tc-MAG3 and ^{99m}Tc-DMSA. Mandatory preliminary calibrations were carried out on the equipment used, notably the Capintec CRC-15R Radionuclide Dose Calibrator. Measurements of Decayed, Effective and Residual Radioactivites were systematically taken each morning for every preparation that met the inclusion criteria. Microsoft Excel and SPSS version 23 were used to treat data, and the findings were presented using tables and charts.

Out of the 153 radiopharmaceuticals administered, 82 (53%) met the inclusion criteria. The mean Decayed, Effective and Residual Radioactivities of the radiopharmaceuticals were as reported in Table 1 below.

TABLE 1. MEAN DECAYED, EFFECTIVE AND RESIDUAL RADIOACTIVITIES

	Decayed Radioactivity (MBq)	Effective Radioactivity (MBq)	Residual Radioactivity (MBq)
^{99m}Tc-MDP	80.72 +/- 53.45	653.72 +/- 77.94	17.07 +/- 6.96
	(14.06 to 186.11)	(461.76 to 849.85)	(8.21 to 37.37)
^{99m}Tc-Pertechnetate	25.75 +/- 11.91	161.32 +/- 67.19	12.28 +/- 5.81
	(6.29 to 67.34)	(61.57 to 221.19)	(2.44 to 34.78)
^{99m}Tc-MAG3	68.08 +/- 43.08	137.15 +/- 48.79	7.15 +/- 1.75
	(10.36 to 126.91)	(96.16 to 223.85)	(4.11 to 9.25)
^{99m}Tc-DMSA	11.99 +/- 9.62	147.25 +/- 68.20	9.33 +/- 5.61
	(0 to 22.94)	(52.91 to 207.87)	(4.40 to 18.50)

Radiopharmaceutical storage times, shown in Fig. 1 below, were within acceptable levels at 0.75, 1.13, 3.38, and 0.55 hours for ^{99m}Tc-MDP, ^{99m}Tc- Pertechnetate, ^{99m}Tc-MAG3 and ^{99m}Tc-DMSA, respectively. The ^{m99}Tc-MAG3 radiopharmaceutical group had the highest storage time, resulting in the highest amount of Decayed Radioactivities (up to 50%).

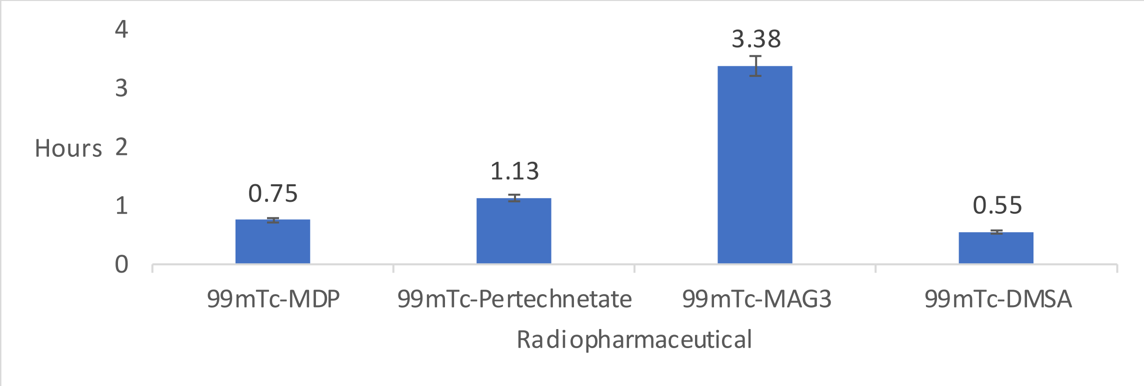

FIG. 1. Bar Chart Showing Radiopharmaceutical Storage Times

The presentation reveals that if radiopharmaceuticals that have decayed to up to 50% of their prepared doses because of long storage times were still useful for diagnostic imaging, lower doses can be prepared if the causes of prolonged storage times are removed. This can lead to overall reduction in both medical and occupational exposures. To achieve this, facilities in the country should increase equipment and personnel capacity. Furthermore, Effective radioactivities for all radiopharmaceutical groups were found to be satisfactorily high, exceeding 80% on average in three radiopharmaceutical groups, and their consequent Residual levels were low and of little practical concern. This shows that the administration techniques used were effective. However, they can be improved on to reach impressive Effective doses reported by researchers in Boston USA [6].

In summary, radiopharmaceutical storage times should be further reduced in the Nigerian practice, especially for ^{99m}Tc-MAG3 radiopharmaceutical. Radioactivity levels in the presentation are substantially similar to reference levels set in Europe [7] but differ from those in Australia [8] and the USA [9]. To establish national reference levels for the Nigerian practice, the country's regulatory bodies may consider both the 75th Percentile of the average values in the presentation, and the ***design*** and results reported by other local researchers [10].

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370-DEVELOPMENT OF CLINICAL DIAGNOSTIC REFERENCE LEVELS IN THE CZECH REPUBLIC
And Clinical Radiological Standards in Nuclear Medicine a Radiodiagnostics

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This poster will present the latest development in clinical radiological standards and clinical diagnostic reference levels (DRLs) in the Czech Republic.

The current DRLs in the Czech Republic are only partly clinical, some of them are getting out of date and for some standardised procedures proper DRLs still haven't been established, so continuous improvement is necessary. SÚJB's technical supporting organization the National Institute of Radiation Protection (SÚRO) usually runs the studies gathering data for establishing of DRLs. Than such proposed DRLs are presented to the relevant professional bodies and the Ministry of Health and after gained consensus the final DRLs are published by SÚJB in the legal Decree on Radiation Protection.

For properly established DRLs it is essential to create good clinical radiological standards that the hospitals accept and follow in standard practise. Of course, a graded approach should be used during these efforts – it is more important to create solid clinical radiological standards in nuclear medicine, interventional radiology and computed tomography, then in dental radiography. That's why the SÚJB focuses its efforts to the following projects:

- Last year a project to set the DRL in nuclear medicine was carried out. The Czech Society of Nuclear Medicine is validating the resulting data from a clinical perspective in these days. The project will continue with the setting of DRLs for CTs, which are part of hybrid imaging systems (PET – CT, SPECT – CT);
- SÚJB runs series of consultation meetings at interventional radiology departments nowadays. During them the SÚJB's inspectors speak with the interventional radiologists and radiographers about the standardised practise at their department (without any possibility of restrictions for the meetings are not inspections). For the purpose of these meetings the departments send to SÚJB beforehand a list of dosimetric data of at least 50 latest interventional procedures for three most common interventional procedures in the Czech Republic. The inspectors use these data for finding out if the local practise is standardised. After visiting most of the departments in the Czech Republic, the gathered information from the meetings will be used for creating a clinical National Radiological Standards for these three procedures and the dosimetric data will be used for establishing clinical DRLs for them;
- And SÚRO's project financed by Technology Agency of the Czech Republic of optimisation of high dose CT procedures is being finalised right now. The main result from this project is a very advanced proposal for National Radiological Standards for 6 most important high dose CT examinations in the Czech Republic. SÚRO has closely cooperated with Czech Radiological Society on this project. The result is based on a search of international and foreign recommendations and papers about these procedures and a national survey within almost all of the radiological departments asking about these procedures in details (indications, strategy, exposure settings, use of contrast, clinical evaluation, etc.). When these enhanced radiological standards will be applied in practise, national dose survey focused on these procedure can be made, resulting in establishment of clinical DRLs;
- This year the SÚRO will start a national dose survey focused on paediatric procedures in order to establish clinical paediatric DRLs in accordance with the latest international recommendations about paediatric DRLs.

371-OPERATIONAL ISSUES AT RUSSIAN AND UKRAINIAN POWER REACTORS:STATISTICS AND CAUSES

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Short Abstract. It was shown that severe nuclear disasters which occurred in several last decades, caused stagnation of nuclear power development over the world. It was discussed a response, taken by IAEA members in order to assure safety of power reactors. Authors analyzed tendency of annual operational issues occurrence by INES scale in Russian Federation and Ukraine within the period from 1991 to 2018 year. A decreasing trend of operational issues emergence was observed. It was studied root causes of these issues and customized mitigation strategies for issues prevention were suggested.

Up to the moment, there are 31 countries in the world operating commercial nuclear power plants (NPP) and using them for energy generation. Sustainable development of nuclear technology – the source of energy which is free from greenhouse gas emissions – is an important issue recognized internationally. Several severe nuclear disasters, occurred at nuclear power plant "Three mile island" (1979, the USA), Chernobyl NPP (1986, USSR), NPP Fukushima (2011, Japan) decreased the pace of civil nuclear technology [1]. This happened because the citizens of many countries have lost their confidence about safety of nuclear installations in general and its inherent risks [2]. In addition, after several social campaigns which took place in Switzerland and Germany the governments of both countries have substantially changed its national energy policies. Basically their strategical goals were redirected from nuclear energy towards renewables [2,3]. After a few years, however, this shift was reconsidered and founded redundantly strict because it lacked scientific justification [4]. So, the consequences of a nuclear disaster may have a) international and b) long-lasting effects.

Normal operation of a NPP unit is relied on stable functioning of its elements and assured by maintenance of the operational personnel. Taking into account the fact that the number of elements exceed thousands of items, and the circumstance that its lifetime may least up to 60 years, the probability of a failure increases over time. In case of a failure, the NPP operator fixes it and investigate causes. Then the operator shares some statistical updates with the regulator using standard forms. In order to improve international norms and share best practices internationally, some general failure data is consequently reported to the IAEA.

In 1990 the group of international experts of IAEA proposed following failure classification by its impact on the nuclear material safety barriers, people and environment: eight levels starting from 1 (minor impact) up to 7 (severe impact) including the level 0/below scale (not significant for safety) for statistical purpose [5].

Current research is aimed to derive general tendencies in safety insuring at nuclear power plants in Russian Federation and Ukraine. For this reason, the statistics of failure events occurred in Russian Federation and Ukraine between years 1991 and 2018 was studied (FIG 50) [6,7].

As it can be seen from the line graph, the total annual failures number of NPP elements of both countries declined continuously. Though in Russian Federation the failure frequency was not even so two periods can be noted: a steady decline phase between years 1992 and 2001 was changed by phase of stabilization, which started from 2002 to 2018 and characterized by low failure frequency fluctuation. The frequency trend in Ukraine is more smooth but has resembling pattern to that of Russian Federation. The correlation between two trends within the period from 1997 to 2018 can be estimated using following equation:

$$\rho = \frac{\sum_{i=1}^{n}(x_i-\bar{x})(y_i-\bar{y})}{\sqrt{\sum_{i=1}^{n}(x_i-\bar{x})^2}\sqrt{\sum_{i=1}^{n}(y_i-\bar{y})^2}} = 0.86 \quad (1)$$

where x_i – number of events in Russian Federation, recorded during year i; $\bar{x}$ – events population mean in Russian Federation; y_i – number of events in Ukraine, recorded during year i; $\bar{y}$ – events population mean in Ukraine.

High extent of failure frequency trends of Ukraine and Russian Federation indicates that both countries have undertaken counter failures efforts in the same time. An example of such efforts could be, for instance, implementation of the probabilistic safety assessment for NPP element safety [8].

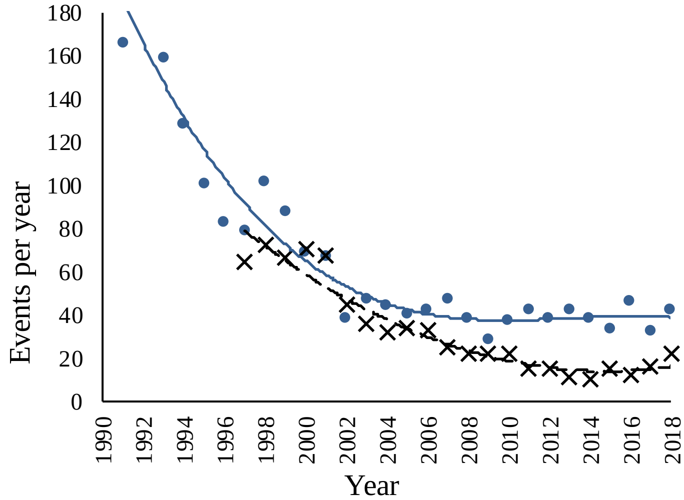

FIG 50. Operational issues at power reactors in Russian Federation and Ukraine between years 1991 and 2018 rated 0/below scale and higher by INES scale: – events in Russian Federation; – curvilinear approximation for Russian Federation; – events in Ukraine; ; – curvilinear approximation for Ukraine.

Despite that similarity, each county has unique root causes distribution of failures shown at the FIG. 51 and FIG. 52 [7,9]:

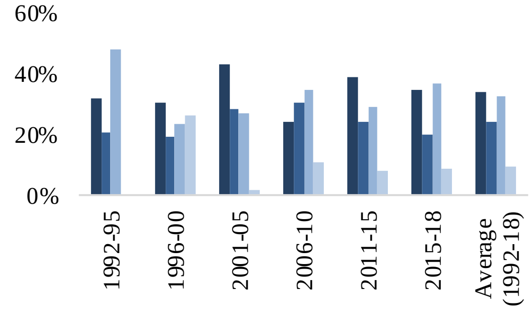

FIG. 51. Failure root causes by 5-year terms in Russian Federation:

– management issues; – design and construction issues; – manufacturing defects; – undetermined.

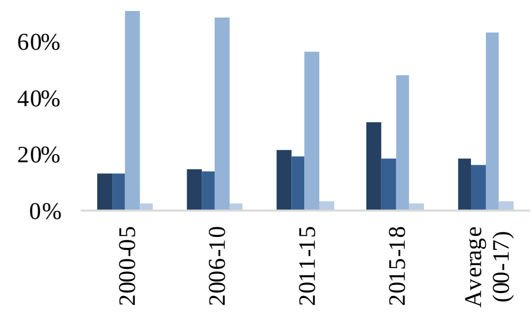

FIG. 52. Failure root causes by 5-year terms in Ukraine:

– management issues; – design and construction issues; – manufacturing defects; – undetermined.

Root causes of failures in Russian Federation have even spread and do no significantly alter over time. Therefore, a balanced failure mitigation strategy can be proposed. In Ukraine, however, management issues become more frequent but manufacturing defects are still dominating. Thus, it seems appropriate to improve management–oriented strategy. It can be also noted that 10% of all evens have unidentified origin and need to be decreased.

In conclusion, continuous improvement of NPP elements reliability is crucial for sustainable development of nuclear energy. Statistics of failures by INES scale can be use as indicator of safety. However, national mitigation strategies should base on type, event frequency and its impact on safety.

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372-OVERVIEW OF RISK MODELS AND RESULTS OBTAINED BY FORO PROJECT (SEVRRA 2) FOR IMRT AND DNM TECHNIQUES

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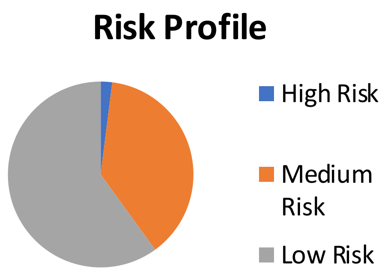
The Ibero-American Forum of Radiological and Nuclear Regulators (FORO) has developed risk models, under the risk matrix methodology, to analyse the possibility of having under and over radiation exposures to patients, workers and public in radiotherapy services that apply radiotherapies techniques such as 3D Conformal, Cobalt-60, Braquitherapy for high and low dose. Due to the fact that radiotherapy techniques are constantly evolving, the FORO decided to expand their risk assessment capabilities to cover new techniques such as Intensity Modulated Radiation Therapy (IMRT) and Diagnostic Nuclear Medicine (DNM) techniques. Therefore, risk models for IMRT and DNM techniques were developed under the SEVRRA 2 project. In the IMRT, 151 equipment failures/human errors capable to initiate accident scenarios condition were identified along with 216 safety elements (safety barriers and frequency and consequences reducers) with the capability to prevent, detect and/or stop accident conditions or mitigate their consequences. Accident sequences were grouped into 13 main processes, separated by consequences to patients, workers and the public. The risk model for DNM can be applied to services that perform diagnostic studies with conventional radiopharmaceuticals (I-131, Tl-201, Tc-99m, etc.) and different techniques for acquiring images in the corresponding hybrid SPECT/CT and PET/CT equipment. The DNM model identifies 96 potential accident starter events distributed in 12 stages of the process ranging from service design and construction to radioactive waste management. For each technique, the application of the risk matrix begins with the analysis of the main process followed in a high standard service, understanding their processes interactions, looking for sources of potential equipment and human failures as well as identifying the safety elements implemented in the service, for both technics an ideal facility was modelled having the highest safety standard identified in the Ibero-American region, which is used as reference facility. An initial risk model is obtained and tested in some facilities (hospitals/radiotherapy services) to get feedbacks from the final users over the risk models developed; therefore, risk assessments were performed in hospitals/services in the Latin American region and Spain, where the IMRT or DNM techniques are been applied. As result of the testing phase, the risk profile (distribution of accident sequences with risk very high, high, medium and low) contribution was obtained and their major risk contributors (safety elements in place in the facility) were identified. The risk profiles of the reference facility are showed in Figs 1 and 2.



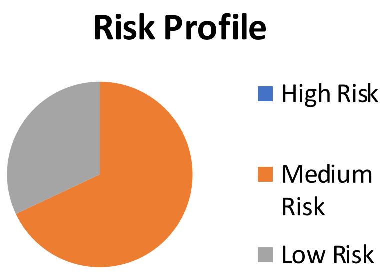
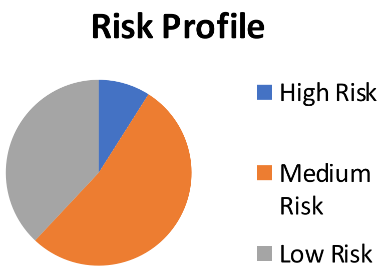


FIG. 1. IMRT Reference facility. *FIG. 2. DNM Reference facility.*

Application of the risk assessment in some facilities in the Ibero American region shows for the IMRT technique that the "Acceptance and Commissioning", "Treatment Planning" and “Treatment Delivery” process steps have the major number of accident sequences with high risk, mainly because of the lack of the safety barriers "In-vivo dosimetry in initial treatment session", "Redundant verifications" and "Peer review".

Whereas, in DNM there were not identified accidental sequences with Very High or High Risk, 68% of accidental sequences show Medium Risk (tolerable) and 32% have Low Risk (widely accepted). The "Radiopharmaceutical Preparation" and "Image Acquisition" stages in DNM include nearly 50% of the Medium Risks sequences identified, so special attention should be paid to compliance with the working procedures established in these two stages.

Examples of the risk profile obtained for IMRT and DNM facilities can be seen in Figs 3 and 4 respectively.



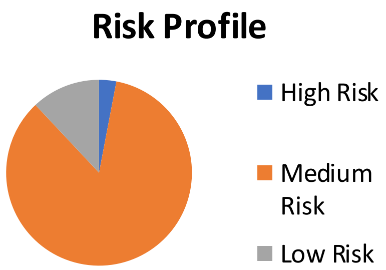


FIG. 3. IMRT facility . *FIG. 4. DNM facility*

The risk models developed allows regulators and safety officers of the facilities to identify the safety elements (barriers and reducers) with the greatest impact on risk reduction and risk increase for each facility analysed, which facilitates communication between them to look for risk management and radiological safety improvements. The paper presents an overview of the risk models and results obtained.

ACKNOWLEDGEMENTS

The SEVRRA 2 Project was developed within the activities framework of the Ibero-American Forum of Radiological and Nuclear Regulators (FORO). Medical Physicists and Regulators from FORO member countries: Argentina, Cuba, Chile, Spain, Mexico and Uruguay acknowledge the FORO support and funding for the development of the work done.

374-THE INDUSTRIAL GAMMA RADIOGRAPHY IN COLOMBIA IMPLEMENTATION OF THE SPECIFIC SAFETY GUIDE NO. SSG-11

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The objective of this work is to share the experience on the results obtained in the implementation of the Guide to Radiological Safety in Industrial Radiography (SGG-11), the challenges presented during the process of putting theory into practice, the feasibility of executing industrial radiography working within the framework of the International Basic Safety Standards (NBS) and other IAEA safety standards [1]; analysing its applicability in the particular case of TECNIENSAYOS S.A.S., a Colombian company that provides Non-Destructive Testing services with the use of radioactive sources, being the industrial gamma radiography operating organization with the highest operating capacity in the country and a trajectory of more than 20 years in the sector.

Industrial gamma radiography is a method of inspection within Non Destructive Testing that is used for the quality control of different materials and industrial components, allowing the detection, identification and dimensioning of discontinuities or anomalies present in the internal structure of the inspected element, such as cracks, pores, slags, etc. Mainly used in the inspection of welds during the construction of oil infrastructures, such as hydrocarbon transport lines, storage tanks and station assemblies; other applications are also found in the steel and shipbuilding industries.

This technique is carried out with sealed sources of gamma emitters with high activity, among the isotopes used we find Cobalt-60, Selenium-75, Cesium-137, Itherbium-169, Tulio-170 and Iridium-192, this being last the most widely used in Colombia. These sources are housed in fixed or portable containers, depending on the requirements of the installation; since most industrial gamma radiography activities take place on-site, the most commonly used containers are portable ones, also known as projector equipment.

Historically, industrial gamma radiography in situ is attributed the highest rates of radiological accidents worldwide, the most frequent causes are human errors or equipment failure [2], with high doses of radiation through which they generate serious health problems, not only of the workers, but also to members of the public, and in some occasions deterioration to the environment.

The handling of radioactive material in any practice involves a risk, which is controllable as long as all the safety and radiological protection measures are complied with, however, industrial gamma radiography works are carried out in the field, carrying out constant transport of the radioactive material from the storage site to the construction site, moving heavy equipment (projectors) and frequently exposing sources with high levels of radiation, usually under workload pressures as well as under difficult climatic and topographic conditions. All this generates an unfavourable scenario that can induce the inadequate execution of operating procedures, increasing the probability of incidents occurring. Hence the importance and need for the facilities to assume an organizational commitment to the safety culture [3], forged from the top management and rooted to the operational side.

In the national context, the industrial gamma radiography in Colombia had its first forays in the early 1970s, as a method used for the inspection of welds in the construction of oil pipelines and other oil infrastructure works. Although in the country there are not many reports of serious incidents related to sources used in this practice, the risk is always latent. Although it is true that the general responsibility for maintaining radiation safety falls on the facility in charge of the radioactive material, on the other hand, it is worth highlighting the management of the regulatory and control bodies, welcoming from the beginning the international recommendations and of the International Organization of Atomic Energy on the matter, adapting them to the needs and regulations of the country.

As a result of this work, the importance of the Specific Safety Guide No. SGG-11 is highlighted, being an effective tool for compliance with the requirements in terms of safety and radiation protection in industrial gamma radiography facilities, as long as it is applied in properly. Likewise, some recommendations are made aimed at improving certain aspects such as the types of incidents that are contemplated, the specific emergency procedures, simulations of emergency situations as part of operator training and a more applicable methodology in the safety assessment.

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375-NEA WORKSHOP ON "OPTIMISATION: RETHINKING THE ART OF REASONABLE"

Lisbon, Portugal
13 – 15 January, cosponsored by the Instituto Português de Oncologia de Lisboa-Francisco Gentil (IPOLFG), and the Technological and Nuclear Campus, Technical Superior Institute, University of Lisbon (IST-CTN)

Dr Edward Lazo
Deputy head of the NEA Division of Radiological Protection and Human Aspects of Nuclear Safety.

Dr Jacqueline Garnier-Laplace
Radiological Protection Specialist in the NEA Division of Radiological Protection and Human Aspects of Nuclear Safety.

The optimisation of protection, to keep radiological exposures As Low As Reasonably Achievable (ALARA), has been central to radiological protection for decades. However, because of scientific uncertainty in understanding the biological effects of low doses of ionising radiation on human beings, other living creatures, and ecosystems, and due to gaps in our knowledge of how ionising radiation might act on cell, tissue and whole organism biological functioning, precaution in regulation and application can result in minimising radiological exposures rather than optimising protection. The objective of this workshop was to discuss the nature and intention of optimisation of protection, and to see how reasonableness should be interpreted in practice.

While optimisation is defined in international recommendations and requirements, its implementation in regulation and application remains quite varied. Workshop presentations and discussions showed that the objective of optimisation of protection can be interpreted quite differently depending on the situation causing the need for radiological protection. For example, optimisation of protection in the context of a deep geologic disposal site will address choosing containers and a site geology to manage exposures in 10s of thousands of years, while optimisation of protection from domestic radon will address influencing personal behaviour. Such differences can promote an image of uncertainty and lack of knowledge. These, combined with incomplete scientific knowledge often result in protection choices taking significant levels of precaution, to the extent of minimising exposures. Optimisation of radiological exposure is not the optimisation of protection.

Situations causing the need for radiological protection will generally be complex, multi-disciplinary, and multi-dimensional, and radiological risks will be one of many different risks. Optimisation and reasonableness are informed by the scientific understanding of the risks involved, but are case-specific, stakeholder dependent, circumstance driven judgements. By broadening the risk aspects being considered, beyond those caused by exposure to ionising radiation, the nature of the objective of optimisation of protection shifts from radiological protection to well-being. This was the focus of the workshop, identifying protection addressing well-being in the broadest sense. Thus optimising radiological protection will not necessarily optimise well-being, and may, in fact focus efforts on radiological risk on health to the detriment of other risks. However, optimising well-being can focus protection solutions on the most serious risks, allocating resources in a more risk-prioritised fashion, in a more reasonable fashion.

376-RADIATION PROTECTION REGULATIONS IN THE USA - A NEEDED UPDATE

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The practice of radiation protection (RP) in the U.S.A. has evolved from recommendations for protection from X rays and radium in the early to mid-20th century, to a codified system of regulations since the early 1950s. These regulations have traditionally followed international and national recommendations. Initially the Federal Radiation Council (FRC) provided recommendations to the federal agencies in the 1960s, which would in turn be used to develop new RP standards, and quickly be codified. Similarly, the federal RP regulations would be adopted by state Radiation Control Program authorities. For decades the Atomic Energy Commission (AEC) was the lead agency for weapons, reactor, and developing medical and industrial applications of radioactive materials (RAM). The AEC developed the first national RP regulations for the protection of man in the early mid-1950s. However, these regulations were focused on the use and control of radioactive materials. During the 1950s and 1960s states utilized these RP regulations for materials, and also developed regulations related to medical and non-medical X-ray equipment. In the late 1960s, the Food and Drug Administration (FDA) gained statutory authority over the manufacture of X ray and other radiation generating machines. However, regulation of end-users of these machine sources was always at the state-level. In the early 1970s the U.S. Environmental Protection Agency was formed, and given the authority of the FRC for development of RP recommendations. They also developed and codified RP regulations related to the nuclear fuel cycle, clean-up of old uranium procession sites, and drinking water. About the same time, the Occupational Health & Safety Administration (OSHA) developed RP regulations to apply in general occupational settings not covered by the AEC or state agencies. Equally significant in the early 1970s, the AEC was split into the [now] U.S. Department of Energy (DOE) and Nuclear Regulatory Commission (NRC). Through the 1960s to present time, the U.S. Department of Transportation (DOT) utilized the International Atomic Energy Agency's (IAEA) standard to develop regulations for the safe transport of RAM on public highways. And as a direct result of the accident at Three Mile Island in March of 1979, the Federal Emergency Management Agency (FEMA) was formed to coordinate such major disasters. Thus, today there is a plethora of federal agencies (i.e., the EPA, FEMA, FDA, DOE, DOT, NRC and OSHA) that have promulgated and enforce various occupational, environmental, air / water, emergency response and clean-up RP regulations. In recent years more and more states have taken over the regulation of RAM from the NRC. There are currently 39 Agreement States. Clearly, the various states and territories oversee and control the vast majority of radiation sources in the USA. These programs direct statewide regulatory radiation control programs in functional areas that include: medical and non-medical X ray equipment and accelerators, radioactive materials, decommissioning, radon testing and mitigation, nuclear power plant emergency response, and environmental surveillance. The primary goal of these programs is to prevent unnecessary environmental contamination or exposure of the public, patients and workers to radiation from controllable sources - while allowing their beneficial use. Without a doubt, the states have a major role in inspection of radiation sources and users in the U.S.A. As inspections and monitoring are performed, occupational and public exposures records and sources are evaluated to ensure compliance with national and state standards and regulations. The licensing or registration of RAM and X-ray sources, combined with performing related compliance inspections, form the backbone for implementing RP regulations within a state. Inspections ensure the appropriate regulatory requirements and conditions of the use permits (e.g., a registration or licenses) are being followed. Given the level of radiation exposure is directly related to health risk, the potential for public, patient and/or worker radiation exposure is a key component to the risk-informed inspection frequency. Unfortunately, in that the responsibility for regulatory updates is spread across numerous federal agencies, there is a serious lack of uniformity in RP regulations and their basis in the U.S.A. And with limited exceptions, current national and international RP standards have not been utilized to update regulations in decades. Current federal NRC and state regulations require public and worker whole body exposure or dose to be limited to 1 mSv/a (0.1 rem/yr) and 50 mSv/a (5 rem/yr) respectively. In addition to these numerical dose limits, the international concepts of justification and optimization are codified in federal and state's regulations via ALARA requirements. These regulations are based on circa 1976 ICRP 26/30 recommendations and calculational methods. In that OSHA's regulations are linked to circa 1972 NRC regulations, occupational whole body dose limits in this regulatory framework can be up to 120 mSv/a (12 rem/yr). Similarly, OSHA limits for extremities and skin are 750 and 300 mSv/a (75 and 30 rem/yr) respectively. These outdated OSHA regulations have their basis in circa 1960 ICRP 2 recommendations. In practice, the more likely a RAM licensee's operations will approach these limits, the more frequent state and NRC inspection frequency will be (e.g., annual to several years). OSHA does not perform routine inspection, only reactive. Similarly, OSHA's radon occupational concentration limits are decades behind current standards of practice. Recently the National Council on Radiation Protection and Measurement (NCRP) reviewed the current literature regarding the linear non-threshold (LNT) model basis for our RP standards. Despite the ongoing debate of applying LNT to public and occupational RP standards, the NCRP re-affirmed that the majority of recent epidemiological studies support the LNT framework. Several years ago NCRP reviewed the International Commission on Radiation Protection's (ICRP) recommendation to reduce the lens of the eye dose limit, and concurred some reduction is warranted. DOT maintains conformance with IAEA standards. The DOE has made strides to update their RP regulations, NRC began a similar effort, but suspended that work a few years ago.

This work will describe the efforts of the Conference of Radiation Control Program Directors (CRCPD) to maintain uniform model state regulations, and a needed path forward for the U.S.A. to incorporate and update their RP regulations to current international and national radiation protection standards. If we do not utilize current science for the foundation for federal and state regulations – we are failing to fully protect the environment, public, patients and/or workers from controllable radiation exposure.