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Survey of dose reconstruction in radiological accidents

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Abstract: Dose reconstruction is a comprehensive analysis of the exposure received by individuals in the vicinity of facilities that release contaminants to the environment (i.e., real doses to real people).Performing dose reconstruction analyses which are extremely important elements for improving efficiency in radiological emergencies and accidents have been studied in this paper. For dose reconstruction purposes, the organ doses from internal and external sources are used to evaluate the risk of stochastic detriment and to determine limit values which have been exceeded. For contaminant releases, radiation doses estimates are needed for individual exposures in release procedures. Internal organ doses are not directly measured and there are different biokinetic models to estimate organ doses. These models are not always directly applicable to individuals who undergo exposures, mainly due to significant differences between healthy and pathological organ/metabolism (clinical conditions). There are two case studies in this paper for internal exposures and external exposures; In the first, we have estimated mean absorbed doses in internal exposures to organs by three different methods; using the conversion coefficients of ICRP publications 53, 62 or 80, for radio-nuclides, Through Monte Carlo simulation, using the Visual Monte Carlo (VMC), Biokinetic models simulation using Dose and risk CALculation (DCAL). In the second one we have estimated mean absorbed dose in external exposure with ^{60}Co as source with three methods; using IAEA absorbed dose, VMC calculations, and DCAL calculations. The comparison between the results was discussed.

Key words: *Dose reconstruction; Radiological accidents*

1. Introduction

For occupational and public exposures, dose reconstructions are needed to define dose levels. It should be possible to reconstruct doses from internal and external sources for tissue and internal organs. Internal organ doses are not directly measured and there are different biokinetic models to evaluate them. For dose reconstruction purposes, organ doses are used to evaluate the risk of stochastic detriment, and to define threshold values to avoid deterministic effects.

In nuclear medicine, studies with radio-pharmaceuticals are used to confirm diagnostic hypotheses where, in the majority of the cases, the patient presents abnormal radio-pharmaceutical capitation, at least in one organ or tissue, which our approach is to some extent the same. For this reason, the models used for internal dosimetry are not, in general, directly applicable to reconstruct dose of accumulation of radio-nuclides in the source organ, since there are significant differences between healthy and pathological individuals. For persons with physical characteristics similar to the mathematical phantoms and standard conditions of Radio-nuclides absorption, it is possible to reconstruct absorbed doses to organs with acceptable level of confidence, in the case of dose reconstruction. On the other side, for therapeutic purposes, it is necessary to apply individual dosimetry of the person.

In the last decades, different softwares have been developed to be used specifically in nuclear medicine such as MIRDOSE (Stabin, 1996). The program Visual Monte Carlo (VMC) (Hunt et al., 2000) was written at the Instituto de RadioproteÇão e Dosimetria specifically for photon transport through voxel phantoms for adults; originally to evaluate exposed workers contamination by long lived radio-nuclides. In this paper we have used the MIRD cook book to compute organ doses due to some radio-nuclides exams and compared them with values obtained through VMC simulations and DCAL simulations. Since VMC code does not take into account biokinetic models the DCAL would be a good approach to take account biokinetic models.

There are two case studies in this paper for internal exposures and external exposures; in the first, organ doses were calculated for internal exposures for three different types of exams: bone scintigraphy (^{99m}MDPTc), lung scintigraphy with MAA-^{99m}Tc, and whole-body scintigraphy with ^{18}F-FDG. The mean activities were selected according a national survey performed among public and private nuclear medicine facilities in Brazil (Velasques de Oliveira, 2005). The radiopharmaceutical residence time for each source organ was chosen according to (ICRP, 1975). The main target organs were chosen among the highest absorbed doses to organs.

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In the second one we have estimated mean absorbed dose in external exposure with ^{60}Co as source using MAX phantom and different regions of interest in this phantom with VMC, DCAL, and IAEA method has been calculated.

2. External dose calculation

The assessment of external doses can be divided into two main steps: Monte Carlo photon transport simulations and dose calculations. The results of Monte Carlo simulations are the energies transferred to specific points (various locations) from each contaminated surface in the environment during the photon transport (given as air kermas per photon per unit area). The contaminated surfaces can be called intervention elements (e.g. roof, paved areas, walls, etc.) and the specific points are the evaluation locations (indoor and outdoor locations).

The industrial area can be considered as a special sub-urban environment; however, the principles of photon transport in the different areas (e.g. urban, industrial) are the same. There are two different ways of Monte Carlo simulations of photon transport in an industrial environment; so-called "global" and "local" approaches of photon transport. A detailed description of the simulations and the environment assumed can be found elsewhere (Kis et al., 2004).

In the first approach, the global one, the air kerma rate is calculated at a central evaluation location due to all different intervention elements being situated in the whole environment and having significant contribution to the air kerma rate at that evaluation location (Jacob et al., 1987). In this way, an assessment of the individual dose can be carried out. It should be noted that this sort of individual dose refers only to the dose contribution from an area where the person spends only part of his/her time.

In the second novel approach, the sources are distributed only in the central part of the environment (in the central component). In this way, the air kerma per photon per unit area due to each specific deposition area being in the central component can be determined in the whole environment at any arbitrary locations in and outside the central component, up to a distance at which the radiation from the source simulated actually will still give significant contribution. Therefore, the assessment of the whole contribution of a specific deposition area (e.g. a roof) to the collective dose can be calculated separately.

Estimating organ dose in a human phantom exposed to radiation from an external source consists of calculating an effect of interest in a geometrically complex object located in an otherwise geometrically simple (one- or two-dimensional) system. This process is mathematically described by the time-independent neutral-particle Boltzmann transport equation.

3. Internal dose calculations

Considering that the mean corporal weights of patients are similar to those of ICRP simulators, it was used ICRP dose conversion factors for estimating absorbed dose to organs. The absorbed dose DT in the target organ T due to the accumulated radionuclide in a single source organ S is

$$D_T = A_S.S(T \leftarrow S)$$

A_S is the time integrated or accumulated activity, that is, the total number of disintegrations in the source organ, and S(T←S) is the dose conversion factor (Table 1) which depends on the type of radiation, emitted energy per disintegration, the mass of the target organ and geometry of the simulators.

Table 1: Examples of ICRP dose conversion factors (mGy/MBq) for nuclear medicine diagnostic procedures in target organs

Procedure/ Radiopharmaceutical	Dose conversion factor $S(T \leftarrow S)$						
	Bone surfaces	Red marrow	Bladder	Lungs	Thyroid	Brain	Heart
Bone scintigraphy MDP-^{99m}Tc	6.3×10^{-2}	9.2×10^{-2}	4.8×10^{-2}	-	-	-	-
Lung scintigraphy MAA-^{99m}Tc	-	-	8.7×10^{-3}	6.6×10^{-2}	-	-	9.6×10^{-3}
Whole-body ^{18}F-FDG	-	-	1.6×10^{-1}	-	-	2.8×10^{-2}	6.2×10^{-2}

4. The Monte Carlo approach (VMC Code)

There is no doubt that Monte Carlo based simulation is the preferred option for the external dose assessment in complex environments. The so-called "location factor" method was applied in the Monte Carlo based dose calculation. The "location factors", defined as the ratio of the exposure at a given location to that at 1 m height above an infinite smooth and plane lawn source, have been used to characterize the external exposure in several environments. Each of these factors gives the exposure at a location taking into consideration the composite contributions of the different surfaces surrounding this location.

The first GSF codes, the EXPURT code by the NRPB, the EDEM2M code and the RISØs URGENT code applied the "location factor" method for dose calculation in urban area.

The program Visual Monte Carlo (VMC) was written at the Instituto de RadioproteÇão e Dosimetria specifically for photon transport through voxel phantoms. The program is written in Visual Basic, and has been applied to internal and external dose calculations due to photons the program was

later extended to include electron, proton and alpha particle transport through voxel structures. VMC has been extensively validated using comparisons with a number of physical phantoms other Monte Carlo programs, and also through international intercomparisons. The code does not take into account biokinetic models used for radionuclides used in nuclear medicine procedures, since it was originally written for occupational exposed workers. For this reason, it was assumed that the residence time in the main source organs is the same related in (Velasques de Oliveira, 2005)

5 The DCAL Code

DCAL consists of a series of computational modules, driven in either an interactive or a batch mode, for the computation of dose and risk coefficients. The system includes extensive libraries of biokinetic and dosimetric data and models representing the current state of the art. DCAL has unique capability for addressing intakes of radionuclides by non-adults. DCAL runs as 32-bit extended DOS and console applications under Windows 95/98/NT/2000/XP. It is intended for users familiar with the basic elements of computational radiation dosimetry. Components of DCAL have been used to prepare U.S. Environmental Protection Agency Federal

Guidance Reports 12 and 13 and a number of publications of the International Commission on adiological Protection. The dose and risk values calculated by this release are consistent with those published in Federal Guidance Reports 12 and 13.

6. First case study

Organ doses were calculated by the three methods described above for selected radio nuclides procedures and selected target organs: bone scintigraphy with MDP-^{99m}Tc, lung scintigraphy with MAA-^{99m}Tc and whole-body scrennning with ^{18}F-FDG. The results are shown on Tables 2, 3 and 4 below.

Table 2 shows the mean absorbed doses (mGy) in bone surface, bone marrow and bladder for the bone scintigraphy (986.6 Mbq initial activity of MDP-^{99m}Tc and residence time of 3 hs).

Table 2: Mean absorbed dose to organs for bone scintigraphy exam with MDP-^{99m}Tc

Model	Absorbed doses (mGy)		
	Bone surface	Red marrow	Bladder
ICRP	62.20	9.10	47.40
VMC	30.20	4.50	20.00
DCAL	58.10	9.20	50.00

Table 3 shows the mean absorbed doses (mGy) in lung, bladder and heart due to lung scintigraphy (158 MBq initial activity of MAA-^{99m}Tcand residence time of 4.9 h)

Table 3: Mean absorbed dose to organs for lung scintigraphy exam with MAA-^{99m}Tc

Model	Absorbed doses (mGy)		
	Lung	Bladder	Heart
ICRP	9.48	1.37	1.52
VMC	6.06	-	-
DCAL	8.00	2.00	1.80

Table 4 shows the mean absorbed doses (mGy) in lung, bladder,spleen and brain due to whole-body scintigraphy with18F-FDG (430.68 MBq intial activity and residence time of 2 hs).

Table 4: Mean absorbed dose to organs for whole body scintigraphy exam with ^{18}F-FDG

Model	Absorbed doses (mGy)			
	Lung	**Kidneys**	Spleen	Brain
ICRP	49.60	9.04	6.02	19.08
VMC	30.02	3.87	5.69	0.27
DCAL	45.10	9.00	6.50	18.00

7. Second case study

The MAX/EGS4 exposure model has already successfully been used for absorbed dose estimations for the many radiological accidents such as (Kramer et al., 2005). In this study the phantom has been exposed by ^{60}Co source with activity of 28.1 PBq (760 kCi) and the arms are opened and the left side of the phantom was exposed to the radiation emitted by the source without shielding by the arms. The distance between the source and the phantom is 2.2 m and the source is placed in 1.2m above the ground.

In order to verify the dose assessment reported in the IAEA document (IAEA, 1996), absorbed doses have been calculated in regions of interest (ROIs) on the surface of the MAX phantom.

The ROIs are volumes at the surface of the MAX phantom with an area of about 26 cm^2 and a depth of 3.6 cm. Depending on their location in the MAX phantom, the ROIs represent mixtures of skin, muscle, adipose and skeletal tissues. Table 5 shows the results.

8. Discussion and conclusions

The values for absorbed doses in organs and tissues estimated with VMC code were smaller than using ICRP dose conversion factors because it was used only the organ self-dose, considering the percentage of the used activity during the residence time in the source organ, without other

contributions and without re-circulation of the radiopharmaceutical. The VMC may be implemented for radiopharmaceuticals internal dosimetry adding new source organs and considering all the target organ contributions. But DCAL code is doing better here.

Table 5: Absorbed dose per ROI for 2 minutes of exposure time

ROI	IAEA Absorbed Dose(Gy)	MAX Absorbed Dose(Gy)	VMC (Gy)	DCAL (Gy)
1	10.90	12.66	10.00	10.80
2	10.60	7.31	10.10	10.00
3	11.10	13.65	11.50	11.05
4	11.10	11.61	11.50	11.05
5	-	8.30	9.00	8.00
6	15.50	13.07	14.08	15.00
7	18.00	13.65	17.05	18.10
8	12.00	9.80	11.80	12.01
9	11.00	8.54	10.90	11.10
10	12.50	8.30	12.00	12.00

Although some information about the exposure conditions during the accidents will be missed, it is possible to successfully determine the absorbed dose distribution in external exposures using the MAX/EGS4 exposure model, VMC code, and DCAL code. Reasonable agreement was found between the data reported in the IAEA document and the results calculated for the MAX phantom and VMC and DCAL. The advantage of using the MAX phantom is that this exposure model can additionally calculated also the absorbed dose to body organs and tissues.

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