APPLICATION OF THE MONTE CARLO METHOD TO ESTIMATE DOSES IN A RADIOACTIVE WASTE DRUM ENVIRONMENT

J. Ródenas[1], T. García[1], M. C. Burgos[1], A. Felipe[2], M. L. Sánchez-Mayoral[2]

[1] Departamento de Ingeniería Química y Nuclear
Universidad Politécnica de Valencia
Apartado 22012 E-46071 Valencia, Spain

[2] IBERINCO, Departamento de Generación Nuclear
Av. de Burgos, 8B (Edif. Génesis) 28036 Madrid, Spain

1. INTRODUCTION.

During refuelling operation in a Nuclear Power Plant, filtration is used to remove non-soluble radionuclides contained in the water from reactor pool [1]. Filter cartridges accumulate a high radioactivity, so that they are usually placed into a drum. When the operation ends up, the drum is filled with concrete and stored along with other drums containing radioactive wastes [2].

Operators working in the refuelling plant near these radwaste drums can receive high dose rates. Therefore, it is convenient to estimate those doses to prevent risks in order to apply ALARA criterion for dose reduction to workers.

The Monte Carlo method has been applied, using MCNP 4B [3] code, to simulate the drum containing contaminated filters and estimate doses produced in the drum environment.

In the paper, an analysis of the results obtained with the MCNP code has been performed. Thus, the influence on the evaluated doses of distance from drum and interposed shielding barriers has been studied. The source term has also been analysed to check the importance of the isotope composition. Two different geometric models have been considered in order to simplify calculations. Results have been compared with dose measurements in plant in order to validate the calculation procedure.

This work has been developed at the Nuclear Engineering Department of the Polytechnic University of Valencia in collaboration with IBERINCO in the frame of an R&D project sponsored by IBERINCO [4].

2. CALCULATION METHOD.

The MCNP code allows us to estimate photon flux averaged over a surface. As well, using conversion factors provided by the code dose rates at points of interest can be obtained

An appropriate geometric model has been developed for the drum. The drum is 873 mm high with a diameter of 573 mm, made of steel with a thickness of 2 mm to get the required tightness. The drum is filled with concrete up to 77,5 mm high. Then, the filter cartridge is introduced on the centre of the concrete basis and the drum is filled up with concrete leaving two holes for pipes transporting water from reactor pool through filter cartridges. This concrete is the first shielding barrier against radiations emitted by radionuclides deposited in the filters. When the filtration is accomplished pipes are cut off and the drum is sealed after filling with concrete the remaining space. A layout of the drum model can be seen in figure 1.

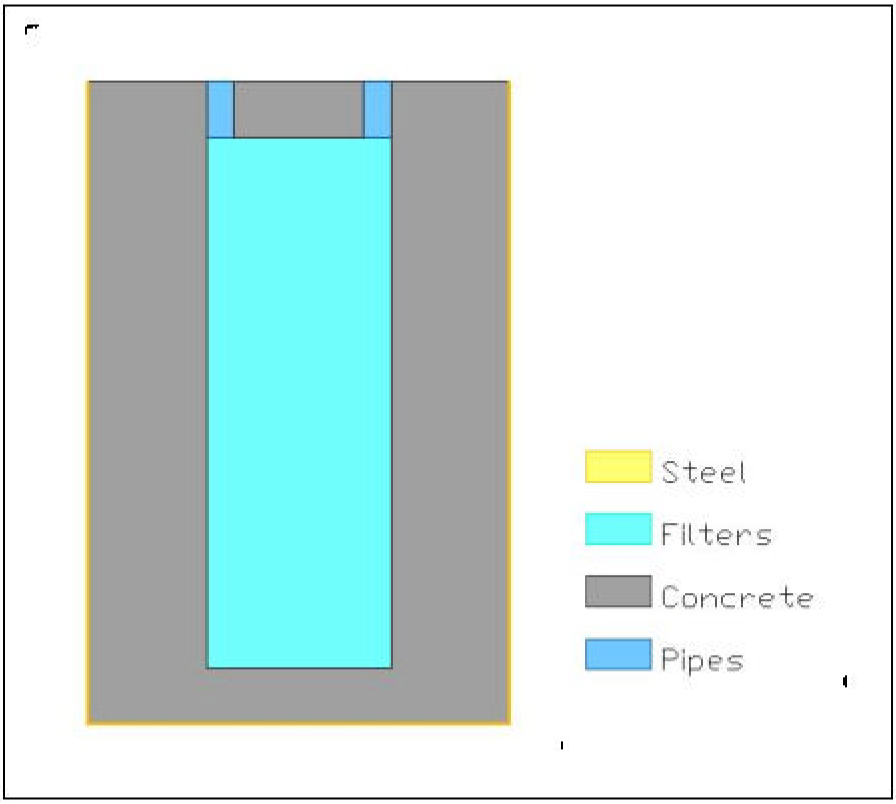


Fig. 1 Layout of the drum model.

The source term has been modelled supposing that non-soluble radionuclides such as Co^{60}, Cs^{137}, Zn^{65}, Mn^{54} and Cs^{134} are retained in the filters. A typical isotopic composition for a filter of those characteristics has been considered [5, 6] as listed in table 1.

Table 1. Isotopic composition.

Radionuclide	Energy (MeV)	Percentage (%)
Co-60	1.1732	70
	1.3325	70
Cs-137	0.6616	10
Zn-65	1.1115	9
Mn-54	0.8348	8
Cs-134	0.57	0.69
	0.605	2.94
	0.796	2.82
	1.038	0.03
	1.168	0.57
	1.365	0.102

Surrounding the drum at a distance of 30 cm some lead slabs of 4 mm thick are placed for radiological protection.

The MCNP code has been run for this model obtaining dose rates at different points from the drum where the photon flux is scored using planes.

Results (dose rates) given by MCNP are evaluated per emitted particle, so they have to be multiplied by the activity contained in the drum in order to obtain true dose rates at points of interest.

3. ANALYSIS OF THE SOURCE.

First, the importance of considering global activity of the source versus individual activity of each radionuclide has been analysed. Thus, calculations have been repeated using two different energy data. In the initial model a unique energy spectrum has been considered, and the obtained dose rate has been multiplied by the global activity of the source. In the modified model, with a better agreement with the actual source, a different case is separately run for each radionuclide. The obtained dose rate in each run has been multiplied by individual activity summing over all radionuclides included in the source.

Results obtained at 1 cm from the drum are listed in table 2, while the comparison of both models is presented in table 3, where it can be seen that the initial model is a good approximation, as deviations are less than 10 % for all considered distances. That is,

$$\dot{D}_T \cdot A_T \approx \sum_i \dot{D}_i \cdot A_i$$

where

$\dot{D}_i$ is the dose rate obtained for radionuclide i.

A_i is the activity of radionuclide i.

$\dot{D}_T$ is the dose rate obtained for the entire energy spectrum.

A_T is the total activity in the drum.

Table 2 Dose rate at 1 cm from the drum.

Radionuclide	Dose Rate/per Particle (mSv/h per Bq)	Activity (Bq)	Dose Rate (mSv/h)
Co-60	1,52141E-10	6,61E+09	1,00565
Cs-137	4,91304E-11	9,44E+08	0,04638
Zn-65	1,23606E-10	8,50E+08	0,10507
Mn –54	7,60289E-11	7,55E+08	0,05740
Cs-134	6,20763E-11	2,83E+08	0,01757
		Total	**1,23207**

From Table 2 it can be seen that some radionuclides such as Co-60 give the main contribution to the global dose rate, due not only to its greater activity but also to their higher energy.

$$\beta = \dot{D}_T \cdot A_T \qquad \alpha = \sum \dot{D}_i \cdot A_i$$

Table 3 Comparison between individual and global models.

Distance from drum (cm)	(mSv/h)	(mSv/h)	b/a
1	1,23207	1,26271328	1,02487138
50	0,18490	0,19360024	1,04705376
100	0,07997	0,080106896	1,00171184
150	0,04220	0,045035691	1,06719647
200	0,02680	0,029190557	1,08919989

The source of radiation gamma has been modelled as a volumetric source. But this model can be simplified considering an isotropic point source located at the centre of the drum. In order to know the accuracy of this approximation, calculations have been repeated for the same distances listed in table 3 using the point source model. Representing in figure 2 the obtained dose rate in terms of the distance from the drum has compared results from both models.

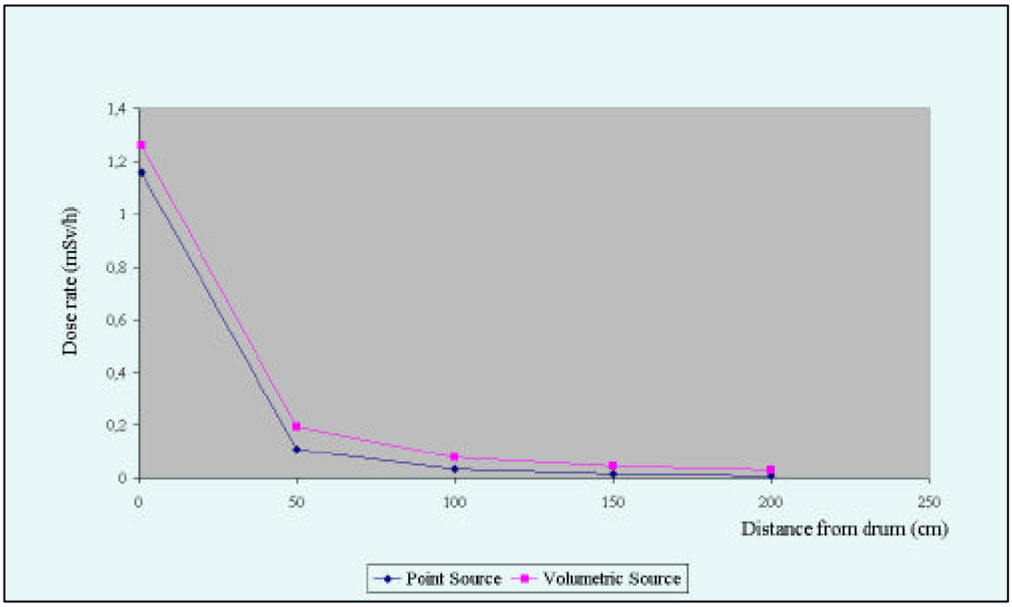


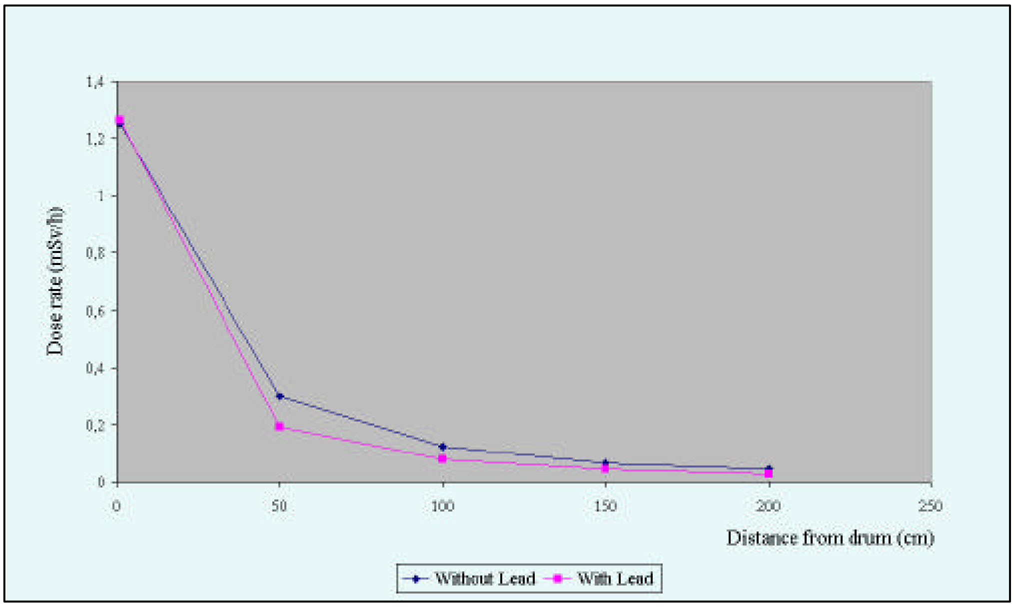
Fig. 2. Comparison between volumetric and point source models.

The values of dose rate obtained for a volumetric source are greater than those for a point source. Nevertheless, this difference tends to decrease as the distance increases, just as it can be seen in the figure 2. Beyond 1.5 m it is better the agreement between results from both models, that is, differences are negligible when the point model is used. Therefore, the simplified model should be used for a larger distance. Anyway, it is obvious the importance of shielding near to the drum.

4. INFLUENCE OF SHIELDING AND DISTANCE.

For both source models when a shielding barrier is interposed between the source and points of interest, dose rates decrease as expected. Nevertheless, for distances higher than 150 cm the influence of the lead shielding is less significant as it can seen in figure 3 where results obtained for volumetric source with lead shielding and without it are represented in function of the distance from the drum. Take into account that shielding is located at 30 cm from the drum, so it has no influence on results at 1 cm.

Fig. 3. Influence of shielding vs. distance.



Two shielding materials, lead and aluminium, have been compared in order to check if a lighter material could be used. However, results showed that the necessary thickness of aluminium is so high that the shielding would be heavier than using lead.

Results obtained from the point source model when the lead shielding is removed are represented in terms of distance in figure 4 for values between 50 and 200 cm. An adjustment has been made for the curve in order to check that the dose rate is inversely proportional to the square distance. The adjustment expression is:

$$\dot{D} = a \cdot x^{-b}$$

Obtaining values for b from 1.8 up to 1.91 depending on the number of points considered.

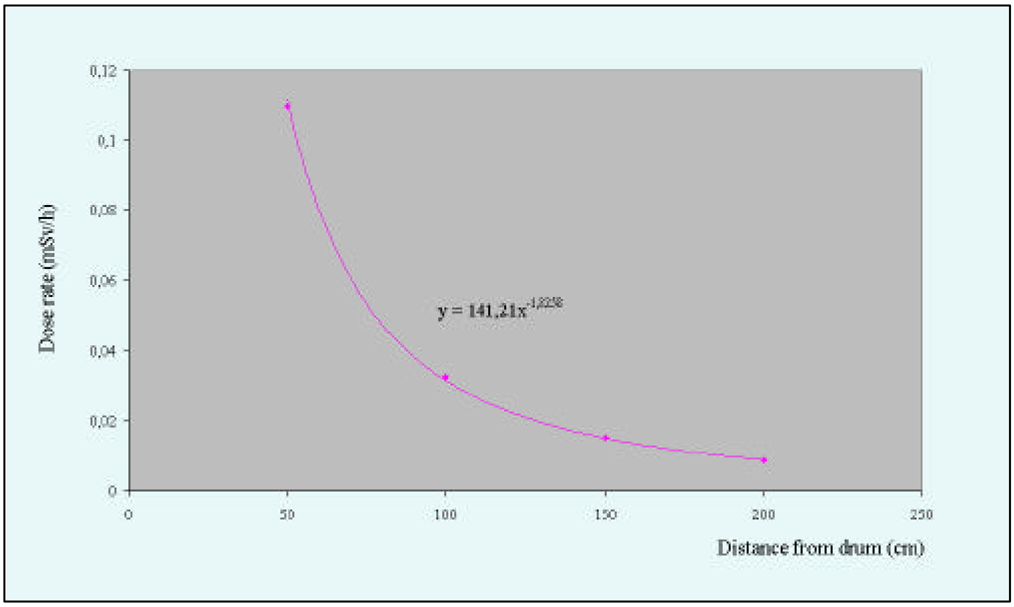


Fig. 4 Variation of dose rate with distance.

Really the analysis can be also made for the volumetric source as for higher distances both models have a similar behaviour.

5. COMPARISON WITH MEASUREMENTS.

Results obtained using the MCNP code have been compared with experimental measurements in order to check the reliability of the model developed.

Values of contact dose rate and contained global activity for 4 drums with similar characteristics to our model have been acquired from a Radiological Protection Service. They are listed in table 4.

Table 4. Measured values of contact dose rate

Drum	Activity (Bq)	Contact dose rate (mSv/h)
1	$9{,}44 \cdot 10^{9}$	6,75
2	$2{,}07 \cdot 10^{9}$	1,47
3	$2{,}73 \cdot 10^{10}$	19,2
4	$1{,}58 \cdot 10^{9}$	1,21

The Monte Carlo method has been applied to simulate dose rates at points very close to the drum, for example 1 cm. Results are listed in Table 5 as well as measured dose rates and deviation δ calculated as,

$$\delta = \frac{\left|\dot{D}_m - \dot{D}_c\right|}{\dot{D}_m}$$

being,

$\dot{D}_m$ the measured dose rate.

$\dot{D}_c$ the calculated dose rate.

Table 5 Comparison between calculated and measured dose rates.

Drum	**Calculated dose rate (mSv/h)**	**Measured dose rate (mSv/h)**	**Deviation (%)**
1	7,07	6,75	4,695
2	1,55	1,47	5,417
3	20,44	19,2	6,443
4	1,18	1,21	2,247

Deviations are always lower than 7%, then our model shows a good reliability. Furthermore, for the higher deviations the calculated dose rate is greater than the measured one. Therefore, it can be considered a conservative model.

6. CONCLUSIONS.

The Monte Carlo method is a powerful tool to determine dose rates in a waste drum environment. It has been applied by using the MCNP 4B code.

Results using this code with the model developed have been compared with experimental measurements obtaining a good agreement, as deviations are lower than 7%, which ensures a good reliability of the model.

Attenuation of gamma radiation with the distance from drum for distances greater than 100 cm is more important than the effect of a lead shielding. Also for those distances a point source model can be considered simplifying the actual volumetric source.

The source term can also be simplified in the model considering a global energy spectrum and global activity in the drum instead of individual activity and energy for each radionuclide present in the source. In this way, computer time can be substantially saved.

REFERENCES.

[1] Frank J. Rahn, Achilles G. Adamantiades, John E. Kenton, Chaim Braun, “A guide to nuclear power technology”, Wiley-Interscience, 1984.

[2] Eichholz, G. G. "Environmental Aspects of Nuclear Power", Ann Arbor Science, 1976.

[3] J. F. Briesmeister (Editor): “MCNP - A General Monte Carlo N-Particle Transport Code, Version 4B”, LA-12625-M, Los Alamos National Laboratory, Los Alamos, New Mexico, March 1997.

[4] A. Felipe, Á. Bayón, J. Ródenas, A. Pascual, I. Zarza, F. Sarti. “Cálculos de dosis en entornos de realidad virtual durante operaciones de recarga de combustible”, 27 Reunión Anual de la Sociedad Nuclear Española, Valencia, 24-26 Octubre 2001.

[5] R. Coello: “Caracterización de depósitos radiactivos en circuitos primarios”. Sociedad Nuclear Española, nº 143, Junio, 1995.

[6] J. Ródenas, A. Martinavarro, V. Rius: "Validation of the MCNP code for the simulation of Ge-detector calibration". Nuclear Instruments & Methods in Physics Research A 450 (2000) 88-97.