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CHARACTERIZATION OF FILTERS CARTRIDGES FROM THE WATER POLISHING SYSTEM OF IEA-R1 REACTOR: RADIOMETRIC METHODS

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ABSTRACT

The acceptance of radioactive waste in a repository depends primarily on knowledge of the radioisotopic inventory of the material, according to regulations established by regulatory agencies. The primary characterization is also a fundamental action to determine further steps in the management of the radioactive wastes. The aim of this work is to report the development of non-destructive methods for primary characterization of filters cartridges discarded as radioactive waste. The filters cartridges are used in the water polishing system of the IEA-R1 reactor retaining the particles in suspension in the reactor cooling water. The IEA-R1 is a pool type reactor with a thermal power of 5 MW, moderated and cooled with light water. It is located in the Energy and Nuclear Research Institute (IPEN-CNEN), in São Paulo, Brazil. The cartridge filters become radioactive waste when they are saturated and do not meet the required flow for the proper operation of the water polishing system. The activities of gamma emitters present in the filters are determined using gamma spectrometry, dose rate measurements and the Point Kernel Method to correlate results from both measurements. For the primary characterization, one alternative method is the radiochemical analysis of slices taken from each filter, what presents the disadvantage of higher exposures personnel and contamination risks. Another alternative method is the calibration of the measurement geometry of a gamma spectrometer, which requires the production of a standard filter. Both methods are necessary but can not be used in operational routine of radioactive waste management owing to cost and complexity. The method described can be used to determine routinely the radioactive inventory of these filters and other radioactive wastes, avoiding the necessity of destructive radiochemical analysis, or the necessity of calibrating the geometry of measurement.

Keywords: Filter, Waste, Characterization, Radiometric, Point Kernel

1. INTRODUCTION

Any nuclear activity generates radioactive wastes. It is necessary to carry out the management of these wastes, meeting regulations established by the responsible regulatory agencies. Methods for managing radioactive waste can be adapted, not necessarily following a standard model, considering their dependence on technological factors, political, social, type of waste being generated and quantity.

To determine which treatment should be given to the radioactive waste is necessary to perform the characterization. The characterization can be primary or final.

In primary characterization it is possible obtain the physical, chemical and radiological information of the waste and establish the radiation protection levels is needed. [1] The final characterization is intended to meet the safety objectives set by the regulator agency for the transport and deposition of waste in a final repository. The safety analysis of repositories require that the inventory of materials disposed radionuclide is known, ensuring the radiation safety in the long term. [2,3]

The methods used to characterize can be destructive or non-destructive. In destructive methods, the sample undergoes changes in its physical state, volume and chemical composition. In non-destructive methods, the material can be analyzed without any change in physical or chemical structure. The choice of method depends on some factors such as the type of radiation to be measured, type of waste and the possibility to minimize the generation of new wastes.

The IEA-R1 reactor is a pool-type research reactor, operating between 2 and 5 MW that uses water as coolant, moderator and biological shield. Besides research, it is used for production of radioisotopes and irradiation of samples with neutron and gamma beams. It is located in the Nuclear and Energy Research Institute at the University of Sao Paulo campus.
Expanded polypropylene filter cartridges are used in retention of suspended particles in water of the primary circuit of the reactor IEA-R1. These filters are part of the water polishing system as well as the activated charcoal bed and ion exchange resin. [4,5]

When these cartridges become saturated and stop meet the required flow of water, they are replaced and become radioactive waste. After weeks awaiting decay of activity, are transported to the Radioactive Waste Management Facility (GRR).
This department is located in the IPEN and is responsible for treatment and temporary deposition of waste generated within the IPEN and generated in other institutions.

The development of a primary characterization method, for determining the radioactive inventory of these filters is required for routine operation in the GRR. A protocol for the primary characterization of the material, so that the radionuclides present and their respective activities can be known in a simple and accurate manner, is a contribution to the GRR operational routine.

The aim of this work is present the developed method for primary characterization of filter cartridges by estimating of gamma emitters activity.

For this, is used the measure of dose rate with handheld gamma detectors and the Point Kernel method. It is possible to use this method without the need for calibration in geometry, allowing its application in the operational routine of radioactive waste management to characterize filter cartridges as well other materials.

2. METHODS AND MATERIAL

From a group of several dozen filters already collected as waste, a sample of 15 filters was selected with dose rates in contact about 0.3 mGy/h or less, using a survey meter (Automess GmbH, model 6150 AD). This value was chosen to keep low the doses in the individuals during the subsequent measurements and, at the same time, allowing the determinations with reasonable accuracy, for a sample of this size. Each selected filtration unit was individually wrapped in polyethylene bags and labeled.

The homogeneity of each filter was checked to guarantee that it is possible to apply the calculation method, which assumes that the distribution of activity is uniform. To this purpose, a shielding was constructed with lead bricks, in the form of a tunnel, so that each filter could be completely shielded in the horizontal position (FIG. 1). The shielding was long enough to allow the filter to move along its axis without loss of shielding. A beam-hole 1 cm in diameter in the central brick allowed the radiation detector to measure the radiation emitted by a thin slice of the filter each turn. The readings were taken at every 3 cm, totaling seventeen slices for each filter. According to IAEA [6], a waste can be considered homogeneous if the concentration of one radionuclide used as indicator, in different parts of the waste, is within the range of ± 30% around the average concentration.

Figure 1: experimental arrangement used in the homogeneity checking

After this, the dose rates were measured at the distances of 20 cm, 40 cm and 60 cm from the surface of the filter in the median plane of the cylinder, using the Radiagem 2000 (Canberra, Radiagem 2000 Personal Portable Dose Rate and Survey Meter), and the 6150 AD Automess detectors. The two detectors were used as a means of comparing results and detecting any discrepant values.

The filters were then positioned near a hyperpure germanium detector (Ortec, HPGe, model EPCG-15-190-R), calibrated in energy. Each filter was measured during 600 seconds (FIG. 2).

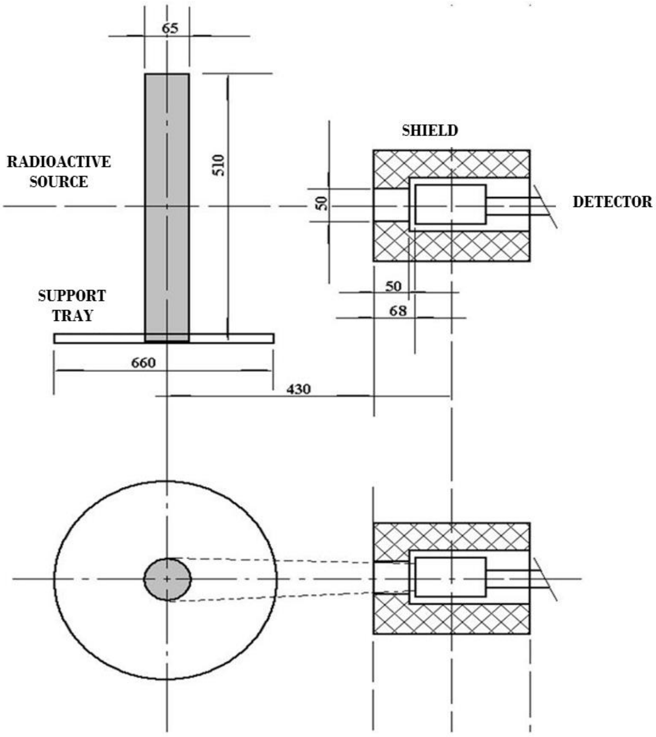


Figure 2: experimental arrangement used in the HPGe

The photopeak heights were corrected by the detector energy efficiency curve and the ratios between the peak heights were used as an estimate of the ratios of emission rates between any pair of photon energies of radionuclides in the filter. The ratios were input data for calculating the dose rates expected in the measurement positions, assuming that the proportions between the emission rates of photons of each energy are the same proportions that were detected. This simplification is acceptable since the differences in the self-absorption of radiation in the filters and the absorption in the air between the filter and the detector can be neglected, for the energies considered.

Using the method of Point Kernel described by Rockwell [7] or the MicroShield® software and the proportions of the emitted intensities of each photon energy identified in the gamma spectrometry, the expected dose rates were calculated at each of the measurement distances above. In these calculations, the MicroShield is used in the mode where the individual photon energies and the emission rates are specified.

Each photon contributes to the fluence at the measuring position, so that the dose rate at a point P (FIG. 3) is given by [7]:

$$\dot{D} = \sum_i \Phi_i \cdot G_i \qquad (1)$$

Where:

$\dot{D}$: dose rate at the point P, given in $Gy.h^{-1}$,

Φ_i: flow of photons with energy i at the P position, given in $cm^{-2}.s^{-1}$
G_i: dose factor per unit of flow of photons with energy i, given in $Gy.\ h^{-1}.cm^2.s$

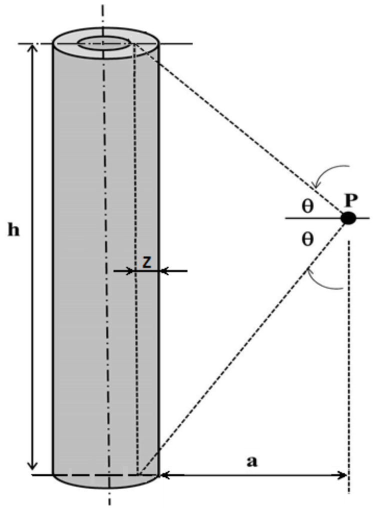


Figure 3: Geometry of the source and the detector for dose measurements at point P

The flow of photons of each energy, expected at the measuring position, is calculated by [7]:

$$\Phi_i = \frac{B_i \cdot S_{Vi} \cdot R_o^2}{2(a+Z_i)} \cdot F_i(\theta, b) \qquad (2)$$

Where:

B is the build up factor, calculated for each energy i,
Sv_i is the emission rate of photons of energy i, per unit of volume, given in $s^{-1}.cm^{-3}$
R_0 is the radius of the filter, given in cm;
a is the distance from the measuring position to the filter surface, given in cm;
Z_i is the self-attenuation distance of each photon of energy i inside the filter, given in cm;
$F_i(\theta, b_i)$ is the integral

$$F_i(\theta, b_i) = \int_0^{artg(\frac{h}{2a})} e^{b_i \sec\theta} d\theta \qquad (3)$$

in which $b_i = \mu_{Si} Z_i$
where μ_{Si} is the linear attenuation coefficient for the photons of energy i in the filter, in cm^{-1}.

The calculation of the dose rate at the measuring positions can be done with any values of Sv_i, as long as the proportions between the emission rates of any pair of photon energies agree with the emissions rates determined previously.

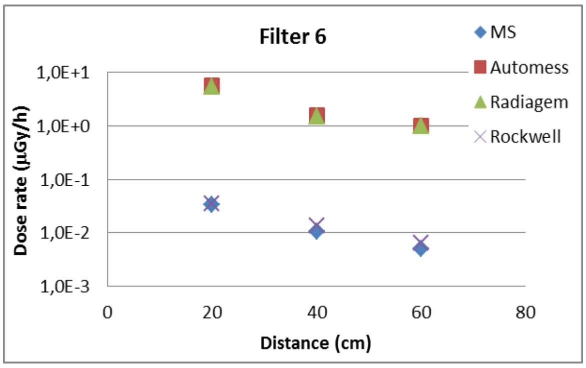
Finally, the Sv_i values are adjusted so that the calculated dose rate coincides with the measured value. The adjustment is done by averaging the proportions between the values measured at three distances from the filter and the values that were calculated with the initial estimates of the Sv_i values.

To check the accuracy of the method, the new calculated values of Sv_i of each energy are translated into the activity of the corresponding radionuclides, considering the yield on the photons in the decay, and the activities are used to calculate the dose rate at the measurement positions, using Microshield. In these calculations, the software is used in the mode where the radionuclides and the activities are specified.

3. RESULTS

After checking the homogeneity of the filter cartridges, the gamma spectrometry was accomplished. Analyzing the energy spectra obtained, 60Co, 108mAg e 110mAg were identified in all filter cartridges. These radionuclides are coming from the structures of the IEA-R1 reactor, such as racks for storage of fuel, the coating used in graphite reflectors and the control rods.

The measured dose rates and the calculated dose rates using the initial estimate of Sv_i, are shown in the graph (FIG. 4) for one filter of the sample, as an example of the results.



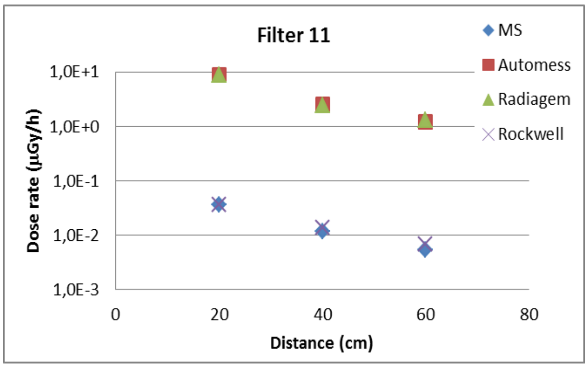


Figure 4: Measured dose rates and the calculated dose rates using the initial Svi estimate.

The average ratio between the measured and the calculated dose rates at the three distances from the surface for each filter was used to refine the initial estimates of S_{Vi}. The new values were used to calculate the activities of the corresponding radionuclides and this value is taken as the end point of the method. Using these activities in the MicroShield in the mode 'radionuclide & activity', the expected dose rates were calculated and compared with the measured values (FIG. 5).

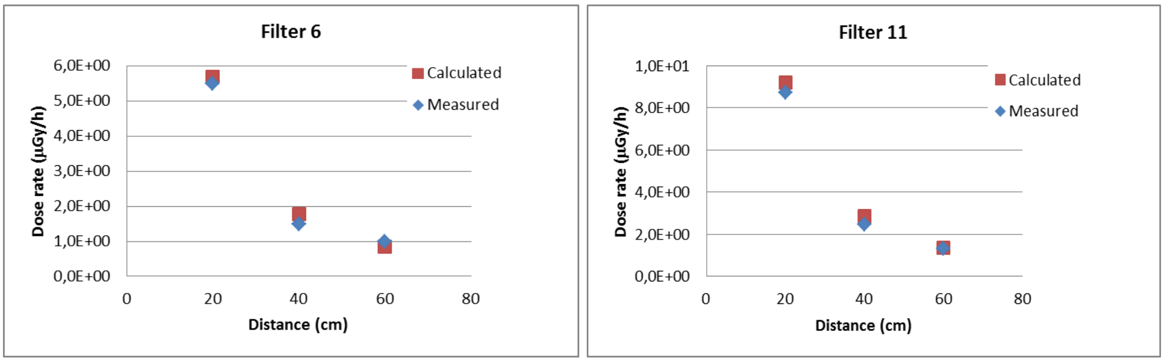


Figure 5: Measured and calculated dose rates for filters 6 and 11 of the sample, as an example of the method's internal consistency.

4.CONCLUSION

Through the results obtained it was concluded that by using a spectrometer gamma calibrated in energy , a handheld detector and the Point Kernel method is possible determine the activity present in the filter cartridges.

The method described allows for the primary characterization and obtaining the radioisotopic inventory of radioactive waste, becoming unnecessary the use of destructive techniques or the calibration in geometry of gamma spectrometers. The method can easily be used in the operating routine for the management of waste, not only for filter cartridges but also to other radioactive wastes.

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