SPECIAL MONITORING FOR INTERNAL CONTAMINATION BY STRONTIUM-90

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ABSTRACT

This work is concerned with the procedures adopted during a special monitoring carried out for internal contamination evaluation. In this event an individual of the public was exposed to ^{90}Sr. During a preliminary monitoring, it was detected external contamination on the clothes, skin, hair and internal contamination was suspected. Samples of 24 hours urine were collected during the three days following the detection of the event. The ^{90}Sr activity in the urine was obtained by beta counting after radiochemical separation. The applied method was shown to be appropriate for this purpose presenting a lower detection limit of 0.05 $Bq.L^{-1}$. The assessment of the intake activity and the equivalent dose were carried out according to the models proposed by the ICRP. The results obtained for the most exposed organ and the whole body did not reach the limits for the individual member of the public.

INTRODUCTION

One of the most important radioisotope of the strontium for the internal contamination point of view is the ^{90}Sr, because of its longest radioactive half-life (T½ = 29 years). The ^{90}Sr decays by beta emission to the ^{90}Y, an emitter of short half- life (T½ = 64 hs) that also decays by beta emission. The ^{90}Sr is an alkaline earth whose metabolism is similar to that of calcium. The strontium incorporation patterns are predominantly the ingestion and the inhalation. Its absorption along the GI tract is between 20 to 50% for soluble compounds of Sr. According to the ICRP-30 model Class D compounds for 1μm AMAD particles 63% of the amount inhaled will be deposited in the respiratory tract of which 83% (52% of the inhalation intake) will be eventually transferred to blood. Most Sr is deposited in skeleton from which it is slowly released.[1,2]

In this accident an individual presented widespread external contamination with ^{90}Sr on his feet, hands, buttocks, face, arms, chest, hair and clothes. The problem was detected during a routine inspection carried out by a radioprotection technician. In this inspection, the technician found two lead recipients used as shielding of the material, wich was deposited in an inappropriate place. Immediately he started the monitoring around this place and a radioactive contamination was detected in several points near the lead recipients with highest dose rate of 0.24 mSv/h. These areas were isolated for further descontamination. Then, a detailed monitoring was made in the facility and also on the people present there. It was verified radioactive contamination in the aprons of some workers, in a tennis shoes and in an individual. The contaminated material was picked up and the individual descontamination was immediately proceeded. In this oportunity a 24 hours urine sample was requested for the internal contamination evaluation. It was also requested two additional urine samples in the subsequent days, in order to estimate the intake.

This work discusses the radioprotection measurements used in the individual internal monitoring after suspicion of a contamination with ^{90}Sr. These measurements include: the determination of ^{90}Sr in urine, estimating intakes and calculation of the committed effective dose.

METHOD - Analysis of ^{90}Sr, in urine with chemical separation.[3]

Face the importance of a quantitative evaluation, the analysis was accomplished using the methodology with chemical separation for the ^{90}Sr and its measure for beta couting.

The method for ^{90}Sr analysis consisted of strontium preconcentration from urine sample, elimination of interferents by ferric hydroxide scavenger, ingrowth ^{90}Y, separation of ^{90}Sr from ^{90}Y and beta counting of the ^{90}Y in a low background Geiger-Müller detector. The strontium yield was determined with ^{85}Sr tracer and the yttrium yield gravimetrically. The ^{90}Y purity was verified by decay curves.

Urine samples of 24 hours were collected after days of the accident, in a total of 3 samples.

The equipment was a low background anticoincidence Geiger-Müller multicounter system GM-25-5, gas flow, and high counting efficiency (28% for Y-90), from RisØe, Denmark.

The results obtained for the Sr-90 actitivities in the samples analysed are presented in

Table 1. The lower limit detection (LLD) was of 0.05 $Bq.L^{-1}$. The ^{90}Y purity was verified by decay curves, the half-life of this radionuclide from different analysis performed were 64.8 ± 2.1 hours.

Table 1: ^{90}Sr activity concentration in urine samples.

Analysis	Sr yield (%)	Y yield (%)	Sr-90 Concentration ($Bq.L^{-1}$)
1	100.0 ± 7.7	83.8 ± 3.2	10.4 ± 2.1
2	46.6 ± 2.3	100.0 ± 3.9	10.0 ± 1.9
3	81.7 ± 5.4	72.4 ± 2.7	10.4 ± 2.0

ESTIMATES OF THE DOSE

As generally happens in the cases of an accident, the lack of information is a very serious problem. Collections of information in an exhaustive way were made to facilitate the evaluation of the involved individual's dose. In this accident, it was not availllable any data about the nature of the compound, the time of the occurrence, and the path of intake.

For the internal dose assessment it was considered that the intake happened through inhalation in virtue of the contamination of the hair, body and clothes. The available information indicated that the ^{90}Sr came as a compound of the solubility class D, although these data were not confirmed.

The activity was determined considering excretion of 24 hours, and taking into account the concentration in the urine given at Table 1, therefore:

- Concentration found in the urine = 10.4 ± 2.1 Bq/l
- Total volume of the individual's urine = 1.4 l
- Total activity in the urine = 14.6 Bq

The estimate of the intake is obtained starting from the data of the total activity and of the knowledge of the excretion fractions for the urine of 24-hours, IRF(t), presented in the Table 2.

Table 2: Fraction of initial intake in 24-hours urine for the ^{90}Sr, considering inhalation and class D[4].

Time after intake (days)	IRF(t)
1	8.57 E-02
2	4.63 E-02
3	3.69 E-02
4	2.98 E-02
5	2.45 E-02
6	2.03 E-02
7	1.71 E-02

The estimate of the Intake (I) is given by:

$$I = A(t) / IRF(t)$$

Where:

A(t) = the activity in the bioassay sample at time t after the intake.

IRF(t) = the intake retention fraction at time t after a unit intake of activity.

The estimate of the committed dose equivalent to the bone surfaces and to the total body are:

$H_{T,50}$ = I / Dose conversion factor for the critical organ

$H_{E,50}$ = I / Dose conversion factor for the total body

The dose conversion factors for the ^{90}Sr are presented in the Table 3.

Table 3 : Dose Conversion Factors for the ^{90}Sr[2].

Dose Conversion Factor (bone surfaces) $Sv.Bq^{-1}$	Dose Conversion Factor (whole body) $Sv.Bq^{-1}$
7.3x E-07	6.2x E-08

RESULT AND DISCUSSION

It was observed that the results of urine concentration obtained for the samples colected in

three serial days had not presented significant variation. Analyzing the Table 2, it is observed that the largest variation in the excretion fraction happens from the first to the second day. Considering the errors involved in the measurement technique and comparing them with the variations among the excretion factors of the Table 2, there is an indicative that the intake had likely happened four days before the first sample collection. In addition of this analysis, taking into account the information of the radioprotection technician it was estimated that the most probable date of the intake had occured at least at 4 days before the first collection.

The calculations of the dose was made based on these data and they are presented in the Table 4.

Table 4: Calculations of the committed equivalent dose considering the fraction of the initial intake in urine for the fourth day [4], and the dose conversion factor [5].

Days between the incorporation and the measurements	Activity incorporated Bq	Committed Equivalent Dose (bone surface) mSv	Committed Effective Dose mSv
4	489.9	3.58E-01	3.04E-02

CONCLUSION

The optimized method of measurement for ^{90}Sr showed to be adequate not only for special monitoring but also in routine monitoring, presenting a lower detection limit of 0.05 $Bq.L^{-1}$[3]. In view of the activities carried out by the individual involved in the accident, he was classified as an individual of the public for purpose of dose evaluation . It is known that the dose limit for the bone surface considering an individual of the public is of 100 mSv/yr and for the whole body it is of 1 mSv/yr[6]. Comparing these limits with the bioassay results it was concluded that this individual had not shown internal contamination for ^{90}Sr. The committed effective dose is only 3% of the public limit and for individual dose registration purpose it is considered as zero dose.[7]

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