EVALUATION OF THE ENVIRONMENTAL DOSE AND THE ^{131}I CONCENTRATION IN A LABORATORY OF RADIOACTIVE MATERIALS PRODUCTION

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INTRODUCTION

Among the radioactive materials produced at the Department of Radioactive Materials Production of Instituto Nacional de Investigaciones Nucleares to be used in nuclear diagnosis and therapy in Mexico, ^{131}I which emits approximately equal numbers of medium-energy beta particles and gamma rays, has special importance notwithstanding its short half-life due its affinity with vital glands (1). Therefore, the air must be monitored because significant levels of airborne contaminators can occur in working areas (2).

In this laboratory there are working 10 workers making the dilution of $Na^{131}I$ to be used in Nuclear Medicine for medical examination and treatment of thyroid condition.

The aim of this work is to determine the ^{131}I annual incorporation levels as well as the personal and environmental dosimetry in the working area. Concentration of ^{131}I in the laboratory of radioactive materials production atmosphere and the absorbed dose produced for this concentration are determined. External dosimetry both personal and environmental were obtained with locally made thermoluminescent dosimeters.

PERSONAL MONITORING

a) Air monitoring for intake.

The inhalation route of the personal subject to potential intake of ^{131}I was monitored at the breathing zone (3). This was done with personal air samplers Victoreen 08-430 consisting of a charcoal filter and a small battery-driven pump that draws air though it. The filter was fastened to the lapel of the worker's lab coat to be located near the breathing zone. The pump were operated at a pumping rate of 6.99 liters /min.

Air samplers were used by the workers along six months. Filters were evaluated every day at the end of the work's day by means of gamma spectroscopy considering sampling times in the range of 120 to 150 minutes. Iodine-131 concentration was determined at room temperature (20°C) and at atmospheric pressure of 532 mmHg.

Radioactive concentration (Bq/liter) of ^{131}I was evaluated by means of gamma spectroscopy of the filters by dividing the net counts (counts - background) by the product of the detector efficiency (counts/gamma-ray) by the counting time (s) and by the sample volume (liters) (4):

$$C = \frac{counts - background}{E\, t_c\, Y\, V_c}$$

Dose equivalent values were calculated by multiplying the ICRP dose conversion factor, 2.9×10^{-4} mSv/Bq, by the air derived concentration in Bq/m³, and by the volume of air breathed in over the working period under conditions of "light activity", 1.2 m³/h x 5 h (5). In equation form this is expressed as follows:

$$D(mSv) = ADC(Bq/m^3) x 1.2(m^3/h) x 5(h) x 2.9 x 10^{-4}(mSv/Bq)$$

b) External dosimetry

Personal monitoring of the workers was performed using locally made thermoluminescence dosimenters. Personal dosimeters were constituted by four LiF:Mg,Ti+PTFE pellets placed in plastic badges of the thickness required to fulfil electronic equilibrium conditions. Monitoring devices were worn on the trunk, between waist and shoulder level at the site of the highest exposure level to which the body is subjected. Personal thermoluminescence dosimeters were worn by the workers for a month between changes because the risk of excessive exposure is low.

Environmental monitoring of the working area was performed using $CaSO_4$:Dy+PTFE locally made thermoluminescence dosimeters. Environmental dosimeters were placed at 1.60 m floor level to be exposed along one month.

Thermoluminescence readings were made in a Harshaw 4000 TL analyzer at a heating rate of 10°C/s integrating the signal from 30°C to 300°C during 30 seconds. All readings were made in N_2 atmosphere.

Previous to be used and reused, the dosimeters were submitted to an specific thermal annealing. Thermal annealing consisted in heating at 400°C during 1 hour followed by 100°C during two hours for LiF dosimeters and 300°C during 30 minutes for $CaSO_4$ dosimeters.

Doses were evaluated interpolating in a calibration curve previously elaborated by irradiating a batch of dosimeters at given doses using a Victoreen 64-764 (12.79 GBq) ^{137}Cs irradiator.

RESULTS AND DISCUSSION

Results showed that ^{131}I derived air concentrations received by occupationally exposed personal and the absorbed doses resulting from these concentrations were lower than the limit established by the ICRP, except that obtained for a worker, labouring during two months at the radiopharmaceuticals packing area, which received a dose three fold the established limit.

Personal dosimetry revealed that two workers received an absorbed dose two fold higher than the limit due that these workers were labouring at the radiopharmaceutical packing area.

Environmental monitoring results showed that two points, located at the packing area and at the labelled radiopharmaceuticals area, registered radiation levels 5 times higher than the typical environmental values.

CONCLUSIONS

These results suggested that the hot cells in the radiopharmaceuticals packing area could have any radiation leakage which was corrected immediately reducing these radiation levels until admissible values. However, these areas must be classified as controlled areas.

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