EVALUATION OF THE RADIOLOGICAL SAFETY OF THE NUCLEAR TECHNIQUES LABORATORY OF THE ESCUELA SUPERIOR POLITÉCNICA DE CHIMBORAZO (ESPOCH)

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Abstract

The main objective of this investigation was to evaluate the radiological safety of the Nuclear Techniques laboratory of the Faculty of Sciences of ESPOCH because it contains a "radioactive rock", which emits radionuclides that were unknown in principle. Initially, the existing areas in the Laboratory were delimited so that it is possible to determine the environmental equivalent dose rate, for which data were taken at different points both near and far from the source, obtaining as a result (0.085 ± 0.038) μSv/h. The radionuclides present in the rock were known through a Radioactive Content Analysis by gamma spectroscopy, which allowed the identification of several radioactive isotopes, the most relevant for their specific activity being ^{238}U, ^{232}Th, ^{226}Ra, ^{214}Bi. The shielding calculation was carried out using the Maximum Project Dose Rate (TDMP) method and, as a result, it was obtained that the shielding must have a thickness of 4.52cm in lead (Pb), to comply with the dose limits recommended by the International Atomic Energy Agency (IAEA).

Keywords: radiological safety, radiactive content, radionuclides, dose rate, gamma spectroscopy

1. Introduction

Radioactivity is a physical phenomenon that studies the processes of spontaneous disintegration of atomic nuclei through the emission of different subatomic particles (Vásquez et al., 2019; Vassileva & Holmberg, 2021). Radiation has brought benefits and mistakes to the scientific world and humanity (Kang et al., 2015). In general, various applications have been determined, from the generation of electricity to medicine (Draganić et al., 2020; Vassileva & Holmberg, 2021) and, despite its innumerable uses, its use carries many risks if it is not used properly. appropriate, applying the national and international safety recommendations and standards already established (Biegon et al., 2022; Donya et al., 2014; Jenkins et al., 2021).

In this context, the International Atomic Energy Agency (IAEA) has published various safety standards in which the requirements for the regulatory control of radioactive materials are established, in which guidance is provided on the physical security of the sources, to control risks related to ionizing radiation (International Atomic Energy Agency, 2014).

In its scope of application, generating and emitting radiation sources are considered, which are used for different purposes and care and control processes are recommended. In the case of the transmitter type, solid containers are used to prevent the leakage of radioactive material. The main objective of these containers is to reduce the exposure to external radiation that this source produces (Tumanov, 2019).

In several cases, radioactive materials have been used, without first establishing a surveillance system based on requirements and recommendations from international organizations. Consequently, there may be radioactive sources outside of regulatory control, which could be introduced into the environment in general and cause accidents or contamination processes (International Atomic Energy Agency, 2014). Throughout history, according to the IAEA, orphan sources have been found, which were recovered. This problem is due to many factors, one of which is the lack of professionalism of the personnel who work with these radiation-emitting elements, which may be forgotten due to staff retirement or facility closure. There are also cases where traffickers may be illegally storing this type of sources for nefarious purposes (Tomek et al., 2022; Tsilikis et al., 2019)

At Escuela Superior Politécnica de Chimborazo, the Nuclear Techniques Laboratory bunker has a radiation emitting source, which is not confined within adequate shielding. In the same way, the storage place does not comply with the safety and radiological protection standards established by the national regulatory entity, which, in the case of Ecuador, is the Undersecretary of Nuclear Control and Applications (SCAN).

Framed in this reality, it is considered extremely important to analyze the existing radioactive rock in the ESPOCH, with the main objective of complying with the regulatory processes related to the management and control of radioactive sources (IAEA, 2020; Miller, 2016). These studies have not been carried out before, but they are necessary to control radiation exposure. This investigation will allow the use of the laboratory and ensure that the radioactive source is in its respective container to avoid danger for students and workers of the institution

2. Materials and Methods

2.1 Methodology

An experimental design was carried out to determine the appropriate shielding thickness, to guarantee compliance with the established dose limits to safeguard the radioactive source and put the bunker of the ESPOCH Nuclear Techniques Laboratory to good use.

2.2 Methodology to determine the environmental dose rate of the internal areas of the bunker

The environmental dose rate was determined using a Geiger-Müller counter, with a valid calibration certificate. Initially, the existing areas in the Nuclear Techniques laboratory bunker were identified, with four rooms identified as rooms A, B, C, D and a bathroom, as shown in figure 1.

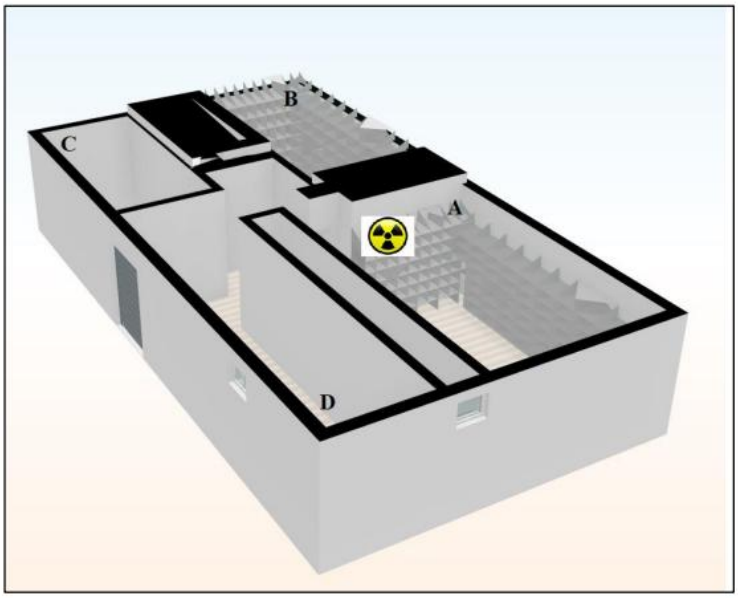

Figure 1: Diagram of the interior of the bunker of the Laboratory of Nuclear Techniques

Once the zones were identified, before taking measurements, the background radiation inside the bunker was defined, which was (0.085 ± 0.038) μSv/h. Subsequently, data were collected inside the room where the radioactive source is located (controlled area) and later in the areas surrounding the source (supervised areas).

To obtain data, 5 values of the environmental dose rate were determined at each point, the average dose was calculated and multiplied by the detector calibration factor, which is 1.01 ± 0.03 uSv/h. To obtain the error, the standard deviation of the values of each point was calculated by adding the value of 0.0001, which is the minimum value marked by the radiation detector with which the work was carried out.

The measurements in room A were obtained at a source-detector distance of 20, 50, and 100 cm. In-room B, radiation was measured at a source-wall distance of 5 and 100 cm with an offset of 50 cm. Afterward, values of area D were acquired, at the points closest to the source and, in addition, it was measured at the door of room C. Finally, data was taken at points E and F corresponding to the bathroom and shower, respectively.

Figure 2: Diagram of the procedure for obtaining the environmental dose rate of the bunker

2.3 Methodology for shielding calculation

2.3.1 Determination of radioactive isotopes by gamma spectrometry

The radioactive isotopes present in the emitting source existing in the bunker of the Nuclear Techniques Laboratory were determined using an "Analysis of Radioactive Content in the sample(s) of rock material", which was carried out in the "Laboratorio de Análisis de Radiactividad" of the "Subsecretaría de Control y Aplicaciones Nucleares" (SCAN), belonging to the Ministerio de Energía y Recursos Naturales No Renovables (Ecuador).

The analysis was carried out using gamma spectrometry, one of the most used techniques for studying nuclide dispersion in the environment. Its main objective is the quantitative determination of the gamma-emitting radioelements, the system receives the radiation spectrum of the sample and compares it with the discrete energies of the reference radioelements, thus achieving the identification of the isotopes of the sample (Egozi et al., 2020; Ho et al., 2018; Obaid et al., 2018).

The radionuclides determined in the samples with their associated activity and uncertainty are shown in Table 1, from the entire list those with the highest energy and specific activity are taken.

Table 1: Specific activity of the radioactive isotopes of the source

Radioactive isotope	^{214}**Bi**	^{214}**Pb**	^{226}**Ra**	^{222}**Rn**	^{228}**Ac**	^{212}**Pb**	^{232}**Th**	^{235}**U**	^{238}**U**	^{210}**Pb**
Activity (Bq/g)	10 400	9 700	15 000	7 500	260	560	13 00	670	1800 000	5700

Source: (Subsecretaría de Control y Aplicaciones Nucleares, 2019)

2.3.2 "Tasa de Dosis Máxima de Proyecto (TDMP)" Method

To carry out the calculation "Tasa de Dosis Máxima de Proyecto (TDMP)" Method was used, which is based on projecting a certain dose after having made the shielding. This method considers the definition of attenuation of an incident beam, which follows an approximation with a negative exponential function. By interposing a shield of thickness x between the radiation source and the point to be protected, an attenuation of the dose rate is produced, this is calculated by applying equation 1

$$\dot{H} = \dot{H}_0 . e^{-\mu x} \qquad \textit{(Equation 1)}$$

Where $\dot{H}$ is the expected shielded dose rate at the point of interest to be protected, that according to the IAEA Safety Requirements and the International Commission on Radiological Protection (ICRP), the project equivalent dose for areas of habitual permanence of OEP is 0.012 mSv/year and 0.012 mSv/h. $\dot{H}_0$ is the dose rate of the radioactive source when it does not have a shield. While $e^{-\mu x}$ is the attenuation factor, where μ is the linear attenuation coefficient and x is the intervening shield thickness.

To calculate the dose rate of each radioactive isotope that makes up the emitting source ($\dot{H}_0$) when it does not have an absorbent medium, equation 2 is used, considering that it is the dose rate per unit of time with a certain activity at a certain distance (Boal & Pinak, 2013; CSN, 2012; IAEA, 2013; Valentin, 2000)

$$\dot{H}_0 = \frac{A.\tau}{d^2} \qquad \textit{(Equation 2)}$$

Where A is the activity of the radiation source, τ is the gamma constant of each isotope, and d refers to the source-point distance to be protected.

To calculate the initial activity, the specific activity in Curies (Ci) is taken as the starting point. The value of the activity of the real source is calculated using values such as the half-life and the radioactive decay constant, which are specific to each nuclide since they will vary depending on the half-life of each isotope.

Finally, the thickness of the armor x is:

$$x = \frac{ln\left(\frac{\dot{H}}{\dot{H}_0}\right)}{\mu} \qquad \textit{(Equation 3)}$$

3. Results and Discussion

3.1 Environmental Dose Rate Results

Table 2 presents the values obtained for the dose rate in room A. Based on the distances previously considered, at 20 cm an average dose of (4.290 ± 0.163) µSv/h was determined; at 50 cm (3.476 ± 0.049) µSv/h and, finally, at 100 cm there is a dose of (1.394 ± 0.085) µSv/h.

As expected, it was observed that the closer to the radioactive source, the higher the values of the environmental dose rate. This would coincide directly with one of the aspects of the radiation protection principle, which establishes that by increasing the distance between occupationally exposed personnel (OEP) and a source of ionizing radiation, the exposure will decrease in the same proportion as the square of the distance increases (CSN, 2012).

Table 1: Ambient dose rate in room A

Distance (cm)	Dose rate ($\mu Sv/h$)
20	4.290 ± 0.163
50	3.476 ± 0.049
100	1.394 ± 0.085

In room B, next to the radioactive source, 60 data were obtained, since it was measured at 6 different points, 5 and 100 cm away from the wall. As shown in Table 3, 12 average environmental dose rate values were obtained, identifying that the lowest value occurs at point 1, 5 cm away from the source wall, and was (0.275 ± 0.038) µSv/h; while the highest was located at point 4, also at 5 cm with a value of (0.701 ± 0.048) µSv/h. This is mainly because the point was at a height in a straight line with the source emitting radiation.

Table 2: Room B ambient dose rate

Distance (cm)	Point	Dose rate ($\mu Sv/h$)
5	1	0.275 ± 0.038
	2	0.533 ± 0.081
	3	0.465 ± 0.049
	4	0.701 ± 0.048

	5	0.513 ± 0.038
	6	0.422 ± 0.038
100	1	0.311 ± 0.058
	2	0.319 ± 0.031
	3	0.362 ± 0.051
	4	0.428 ± 0.019
	5	0.477 ± 0.065
	6	0.402 ± 0.030

In the rest of the rooms next to the radioactive source, the same procedure was carried out, taking 5 environmental dose data. Table 4 details the values obtained in zones C and D. At the door of room C, an average value of (0.295 ± 0.042) µSv/h was obtained, later it was measured at the point closest to the source of room D, having a dose value of (0.085 ± 0.038) µSv/h.

Table 3: Ambient dose rate of rooms C, D

Room	Ubication	Dose rate ($\mu Sv/h$)
C	Door	0.295 ± 0.042
D	Door	0.085 ± 0.038

Finally, measurements were taken at the bathroom door and in the shower, resulting in a dose of (0.503 ± 0.154) µSv/h and (0.166 ± 0.037) µSv/h respectively, as shown in Table 5.

Table 4: Environmental dose rate at the bathroom and shower door

Ubication	Dose rate ($\mu Sv/h$)
Bathroom door	0.503 ± 0.154
Shower	0.166 ± 0.037

To carry out adequate data analysis, it is considered that according to the IAEA (2013), the permissible limit for OEP is an effective dose of 20 mSv/year, which, transforming the units with which it is worked, results in a maximum dose of 2.28µSv/h. With this, it was determined that, in room A, the values obtained at 20 and 50 cm from the source exceed the permissible limits recommended by the IAEA, since the average doses of (4.290 ± 0.163) µSv/h and (3.476 ± 0.049) µSv/h, exceed with a value of (2.01 ± 0.163) µSv/h and (1.196 ± 0.049) µSv/h respectively, at the referential level.

No problem was found in the other rooms of the bunker since all the environmental dose values are below international recommendations.

3.2 Shielding calculation results

For the calculation of shielding, the radioactive isotopes that make up the natural source of radiation were identified, this was done through a gamma spectrometry study. The most relevant isotopes are shown in Table 6, together with their specific activity and the activity of each one.

Table 5: Isotopes with higher activity and specific activity

Isotope	Actividad específica	Actividad
^{238}U	$4.864\ x\ 10^{-5}\ Ci/g$	$9.701\ x\ 10^{-2}\ Ci$

^{232}Th	$3.512\,x\,10^{-7}\,Ci/g$	$7.026\,x\,10^{-4}\,Ci$
^{226}Ra	$4.054\,x\,10^{-7}\,Ci/g$	$8.108\,x\,10^{-4}\,Ci$
^{214}Bi	$2.810\,x\,10^{-7}\,Ci/g$	$5.620\,x\,10^{-4}\,Ci$

With the application of equation 3, the shielding result in Pb was determined. The shielding thickness results for the radioactive source were obtained with the TDMP method; at a source-point distance to be protected of 0.5 m. The attenuation coefficients in Pb and the values of the specific gamma constant were taken from the book of Laurie Unger "Specific gamma-ray dose constants for nuclides important to dosimetry and radiological assessment (Unger & Trubey, 1982)

3.2.1 Calculation of ^{238}U, ^{232}Th, ^{226}Ra, ^{214}Bi

The values obtained in the calculation for the radioactive isotopes with the highest activity are presented in Table 7. Initially, ^{238}U presented an initial activity at the time of the study of 0.0970 Ci, when the time elapsed until the shielding was carried out it was 0.0969 Ci. The value of the dose rate produced by radiation at the point of interest to be protected is 0.2527 mSv/h. This radioisotope exhibits alpha (α) decay, which means that a piece of paper could stop this type of radiation, but there is also the possibility that the emission of a photon is generated when going from an excited state to a stable state. For this reason, a shielding thickness of 4.52 cm in Pb is required to attenuate this radiation beam and thus comply with the environmental dose limits in the bunker.

^{232}Th and ^{226}Ra show initial activity and actual activity of 7.02 x 10-4 Ci and 8.11 x 10-4 Ci respectively. Because the half-life is very long, its activity is not significantly altered. The dose rate produced by radiation at the point of interest to be protected is 1.92 x 10-3 mSv/h and 3.92 x 10-4 mSv/h, obtaining that the thickness for 232Th is -2.73 cm and for 226Ra is -5.08 cm, values that are negative because the representative decay is alpha (α) and has very low penetrating power in matter. Since we worked with attenuation coefficients in Pb, it is evident that this attenuator is very large for these isotopes. For this reason, no shielding is needed to stop them and to comply with the standardized environmental dose limits in the bunker.

For the shielding calculation of ^{214}Bi, we worked with its initial activity, since the half-life of this element is 19.9 minutes, and its real activity reports a result of 0 Ci. However, we must consider that ^{238}U always produces daughter isotopes, and therefore ^{214}Bi will be present until the total decay of the parent isotope. The dose rate produced by radiation at the point of interest to be protected is 0.018mSv/h, in addition, ^{214}Bi presents 3 types of decay, alpha (α), negative beta ($\beta-$), and gamma (γ). For the first two decays there is no problem in their shielding, but for decay γ there is, due to its high penetration power when interacting with matter. With this background, a thickness of 0.602 cm of Pb is required to be able to stop this type of radiation and comply with the standardized limits of environmental dose for the study area.

Table 6: Values for calculation of shielding of radioactive isotopes

Magnitude	^{238}U	^{232}Th	^{226}Ra	^{214}Bi
Initial activity ($A_0\ en\ Ci$)	$9.701\,x\,10^{-2}$	$7.02\,x\,10^{-4}$	$8.11\,x\,10^{-4}$	$5.62\,x\,10^{-4}$
The activity of the source ($A\ en\ Ci$)	$9.609\,x\,10^{-2}$	$7.02\,x\,10^{-4}$	$8.10\,x\,10^{-4}$	$5.62\,x\,10^{-4}$
Ambient equivalent dose rate of the source ($H_0\ en\ \frac{mSv}{h}$)	$2.527\,x\,10^{-1}$	$1.92\,x\,10^{-3}$	$3.92\,x\,10^{-4}$	$1.8\,x\,10^{-2}$

Expected Environmental Equivalent Dose Rate ($H_L\ en\ \frac{mSv}{h}$)	$1.2\ x\ 10^{-2}$	$1.2\ x\ 10^{-2}$	$1.2\ x\ 10^{-2}$	$1.2\ x\ 10^{-2}$
Espesor ($x\ en\ cm$)	4.25	−2.73	−5.08	0.602

4 Conclusions.

Considering the technical requirements and international recommendations for the management of radioactive sources and, above all, to achieve adequate levels of radiological protection of the OEP, the environmental equivalent dose rate was analyzed within the bunker areas of the Nuclear Techniques Laboratory. It was obtained as a result that only in room A, at 20 and 50 cm, the dose limits recommended by the International Atomic Energy Agency (IAEA) are exceeded, yielding an average value of (4.290 ± 0.163) µSv /h and (3.476 ± 0.049) µSv/h respectively, since the recommended value is 2.28 µSv/h. To attenuate the radiation levels identified at the source and comply with the dose limits recommended by the IAEA (0.00228mSv/h), a thickness of 4.52 cm of Pb with a suggested density of 11.35 g cm^{-3}, which would contrast the emissions from the main source, which in this case is ^{238}U.

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