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Paper

Radioprotection issues in uraniferous minerals collections with reference to an actual case

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Abstract

Our work investigated the radioprotection implications associated with the possession of a collection of uraniferous minerals. Considering different scenarios, we developed (and applied to an actual collection) specific formulas for radiation doses evaluation. We discussed the shielding necessary to reduce the gamma irradiation down to the required values. A mathematical model was developed to estimate the minimum air flow rate to reduce the radon air concentration below the reference values. The radiation risks associated to the handling of single specimens was also addressed, including hand skin irradiation and shielding capabilities of surgical lead gloves. Finally, we discussed the radiation risks associated to the exhibition of a single specimen. The results, compared to the safety standards of the EU Directive 13/59, show that the exhibition of uraniferous samples with activity of a few MBq do not need specific radioprotection requirements nor for the involved personnel nor for visitors.

Introduction

Many museums all around the world do possess collections of uraniferous minerals. However, the correlated unavoidable emission of radiation poses problems for their storage and, even more, for their exhibition to the public[(1–8)]. Possibly many of those problems arise from an incomplete knowledge of the actual activity of the collection and the related amount and quality of the emitted radiation.

Worker exposition can be evaluated using specific calculators available online, such as *wise-uranium.org*. However, in a previous work[(9)], the authors developed an accurate method for the measurement of the activity content of uraniferous mineral specimens that was successfully applied to the 361 specimens of the mineralogical collection in possession of the Natural History Museum (SMA) of the University of Florence, Italy.

In the present work, the authors proceed further and investigate from the radioprotection point of view the various activities possibly related to the possession of a collection of (radioactive) uraniferous minerals, from the simple storage up to the specimens handling and exhibition to the public. For each considered scenario, the appropriate formulas for dose calculation have been derived and, taking advantage from the results of the above mentioned work, such formulas have also been applied to an actual collection. In the specific, our results, when compared with the statements reported in EU Directive 13/59 "laying down basic safety standards for protection against the dangers arising from the exposure to ionising radiation", show that while the simple storage of the whole collection may rise radioprotection concerns, on the other hand the exhibition of single specimens turns out to be safe.

In order to properly address the dose coming from the uraniferous ores, in relation to their isotopes content, the quantity U_{nat} is introduced, referred to a point-like sample of uranium ore that contains the U_{235} isotope with its 10 daughters and the U_{238} with

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its 13 daughters. The relative abundance in mass of U_{235} family, in secular equilibrium, is 0.7%, and of U_{238} family, in secular equilibrium, is 99.3%. This quantity represents the actual relative abundance found in natural ores of uranium isotopes.

Among the different contribution to the effective dose, radon emission and accumulation is a non trivial issue which has several implications in terms of radiation risk assessment and consequent radioprotection measures[10]. Unlike other intermediate elements in the uranium decay chains, radon is, under standard conditions, gaseous and easily inhaled, and therefore may represent a significant hazard for lung health, due to its α radiation emission. Even if radon represents one of the larger contribution to the background radiation dose, often its outdoor concentration is low enough so that the associated radiation risk is negligible. However, its spatial confinement is favoured by inadequate air change and ventilation rate, by small spaces and by the lack of external channels for air flow, conditions very common in basements and underground areas, which can let its concentration rise well above the reference levels stated by national regulations. When such conditions occur, radioprotection measures must be carefully determined and adopted.

From a general point of view, we presume that our work can be considered as a guide for the holders of similar collections for the determination, in a quantitative way, of the actual related radioprotection risks so as to allow the aware undertaking of the proper actions.

Materials and methods

In this section, radiometric measurements focused to radioprotection when dealing with collections of uraniferous minerals are presented, with a practical reference to the above mentioned collection. Ionising radiation risk is caused by the exposure to the mineral specimens' gamma rays irradiation, by the inhalation of radon gas produced by radioactive decay of uranium, and by the beta and gamma skin irradiation when handling the samples directly by hand.

All the necessary measurement have been described with reference to the actual mineralogical collection in possession of the Natural History Museum of the University of Florence, which is located in a basement, specifically selected to store the uraniferous specimens. The basement consists in a spacious warehouse connected to the storage room hosting the collection. The access to the storage room, allowed only to authorised personnel, is possible via a locked up door, equipped with signs bearing the radiation symbol, as required by national radioprotection regulations. The specimens are placed in polystyrene boxes and packed within

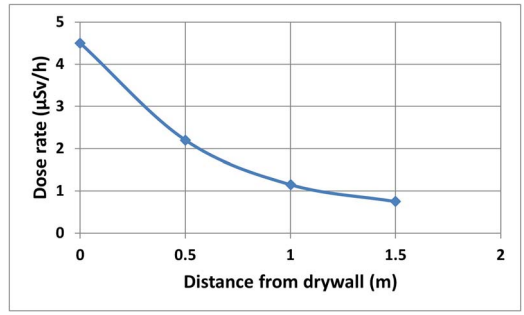

Figure 1. Measured dose rate as a function of the distance from the bricks wall surrounding the boxes containing minerals.

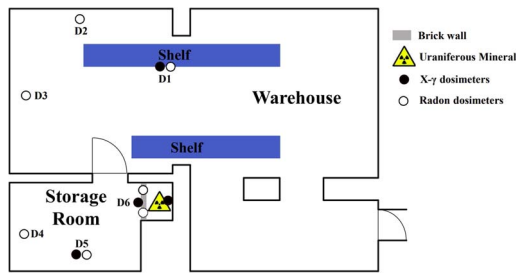

Figure 2. Map of the basement: positions of thermoluminescence dosemeters are indicated by red dots and positions of radon dosemeters by green dots.

shock-proof plastic sheets. The boxes are stacked up together and stored in a recess, protected by a 5-mm lead shield covering front and upper side of the recess. A bricks wall 20-cm thick encloses the recess frontally, while posterior and lateral bricks walls are 30-cm thick (see also map in Figure 2).

Gamma radiation measurements

Electron Ion Chamber is the instrument of choice to measure ambient dose rate due to gamma radiation. In our case, we used a Fluke Victoreen 451. The dose rate in contact with the upper side of the boxes (no bricks shielding) was 120 μSv h^{-1}. Dose rate was also measured along the horizontal plane as a function of the bricks wall distance: the results are reported in Figure 1, showing the dose decreasing approximately according to an inverse square relationship, as expected.

An additional ambient dose measurement was carried out using thermoluminescence dosemeters (TLD 100) provided by a service accredited ISO 17025 (UO Fisica Sanitaria, Careggi University Hospital, Florence, Italy). These dosemeters were placed in several locations within the basement, identified by red dots in Figure 2. Dosemeters were exposed for 33 d. Results are reported in Table 1: it is noticeable that the dose

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Table 1. Ambient dosimetry. The position of each dosemeter is reported in the map of Figure 2.

Dosemeter	Ambient dose rate (μSv h^{-1})	Position
1	0.41 ± 0.10	D1
2	2.3 ± 0.4	D5
3	4.1 ± 0.5	D6
4	122.0 ± 24.3	D7

Table 2. Evaluation of radon gas concentration in the air.

Dosemeter	Radon concentration (MBq m^{-3})	Position
1	14.9 ± 3.8	D6
2	15.3 ± 4	D6
3	14.2 ± 3.3	D5
4	14.0 ± 3.1	D4
5	1.33 ± 0.09	D3
6	0.31 ± 0.04	D2
7	0.22 ± 0.03	D1
8	0.07 ± 0.01	Office
9	0.07 ± 0.01	Office

rate in the warehouse is comparable with the local natural background radiation. The dosimetry results are in agreement with the ionisation chamber measurements as reported in Figure 1.

Radon radiation measurements

Quantification of radon concentration was performed by means of a diffusion type Ion chamber provided BY the Radon Service from the Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA), made of radon-permeable graphitic nylon (CR39-Intercast).

The results of radon measurements in different locations, indicated by green dots in the map in Figure 2 are summarised in Table 2. It is noticeable a spatial relationship of the dosemeters readings: the dosemeters placed in the warehouse (D1, D2, D3) provided results much lower than the ones placed in the storage room (D4, D5, D6, D7). The maximum values of radon concentration are found in correspondence with the locations closest to the collections (D6).

Skin dose measurements

Contact dose measurements are required in case of direct handling of the uraniferous specimens by hand, in order to evaluate the skin dose to the operator as well as possible adoption of radioprotection measures such as hands shielding materials.

We performed contact dose rate D_{cr} measurements on three selected samples of different weight, bulk composition and activity, by employing passive dosemeters placed on the surface of the specimens. Subsequent additional measurements were performed by inter-placing a stripe of a surgical lead glove between the samples and the dosemeters, in order to evaluate the dose reduction when wearing the gloves (Figure 3).

Table 3. Characteristics of the three selected samples used to estimate skin dose.

Inv. no.	Species	Activity (MBq)	Approx. size [l, h, d](cm)]
G47197	*Cuprosklodowskite*	17.0 ± 1.5	7, 6, 6
G47214	*Vandenbrandeite*	4.67 ± 0.51	7, 6, 5
G47221	*Metatorbernite*	1.70 ± 0.19	7, 5, 4

Also in this case, thermoluminescence dosemeters (*ExtRad* calibrated for skin dose measurements) were provided by the same service accredited ISO 17025 previously mentioned. The dosemeters were exposed for 40 d. Measurements were carried out in the warehouse room, where the three samples were placed far enough one another to avoid interference. Additional six dosemeters were used to evaluate the background value to be subtracted.

The chosen samples have different chemical composition and different activity, though shape and size adequate to appropriately perform the measurements. These properties are summarised in Table 3.

In Table 4, the measured contact doses are reported, obtained as the average value of the doses resulting from the dosemeters placed on the samples surface, together with the standard deviation. The large uncertainties are possibly due to the inhomogeneous distribution of the surface crystals which contain high uranium concentrations.

In the presence of radioprotection gloves, an approximate dose reduction of 30% has been measured (see Table 4).

Conservation issues

Gamma irradiation issues

The radiometric measurements described in the previous sections showed that radioprotection measures are necessary for the safety of the personnel and the public.

The radiation dose due to gamma radiation is dependent on the irradiation time, the distance and attenuation between the source and the person. With D_d being the dose rate in absence of shielding material at a distance d, the yearly cumulated dose calculated for a person at the same distance d from the source for a

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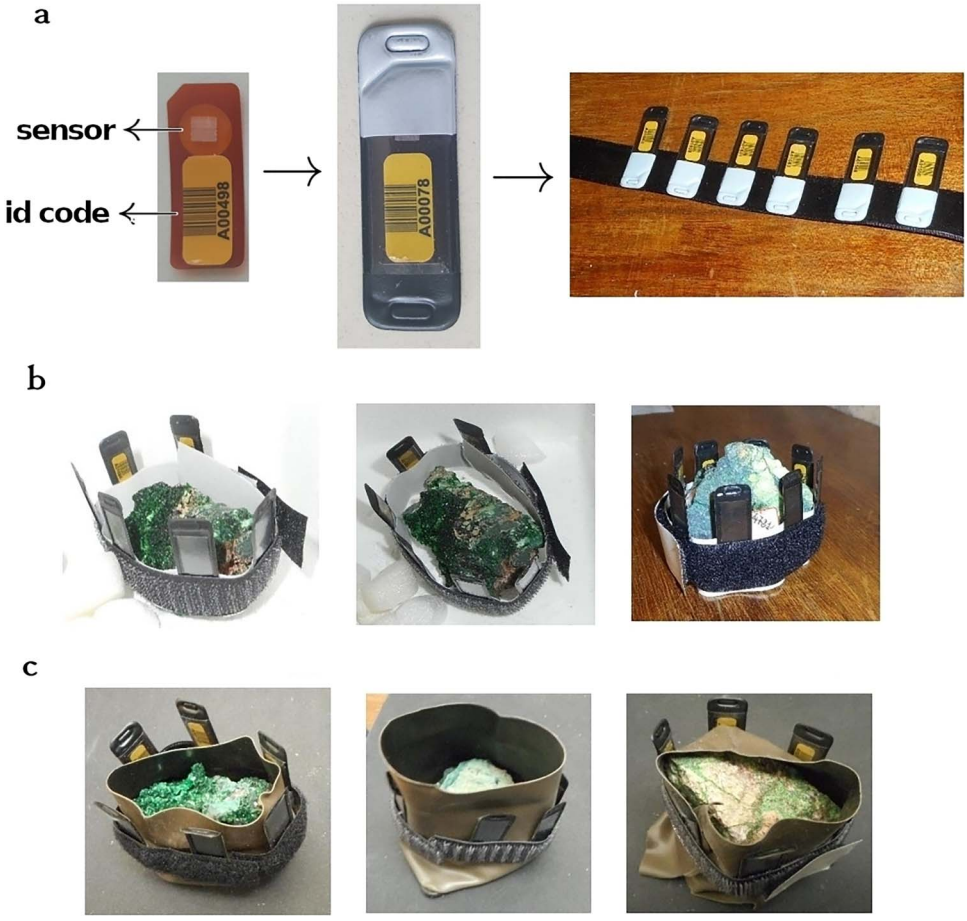

Figure 3. Pictures of the stand-alone detector (TLD chip), the TLD chip inside its holder and the belt (Velcro strip) holding the dosemeters to be located around the samples; (b) measurement configuration with the belt arranged around the samples; (c) measurement configuration with a gloves strip interposed between dosemeters and samples.

Table 4. Samples contact dose rate D_{cr} in presence and absence of shielding gloves, and dose reduction in the presence of shielding gloves. The reported doses are the average values, with standard deviation, resulting from the dosemeters placed on the surface.

Inv. no.	D_{cr} (μSv h^{-1})	D_{cr} w/gloves (μSv h^{-1})	Dose reduction (%)
G47197	631 ± 163	389 ± 76	38%
G47214	279 ± 61	187 ± 49	33%
G47221	250 ± 88	193 ± 59	23%
Average	387 ± 65	256 ± 36	34%

time period T is

$$D_\gamma = D_d \cdot T \cdot B \quad (1)$$

where B is the attenuation factor provided by the shielding. The cumulative activity of the whole collection may result high enough as to require appropriate shielding of the gamma rays emission in order to reduce the yearly dose below its limit ($D_\gamma < D_L$)[11, 12].

Most commonly used materials for gamma rays shielding are lead and bricks (or concrete). Measurements of the shielding performances of these materials were carried out in order to allow a generalisation of the shielding problem for different situations such as collections with different cumulative activity or storage conditions. We measured attenuation efficacy of lead and bricks (actually the materials that were chosen for the radiation shield of our reference collection).

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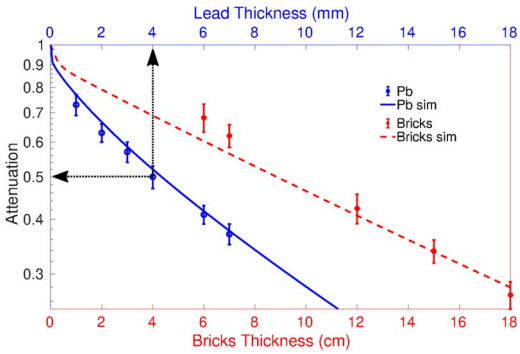

Figure 4. Attenuation measurements of the radiation emission from a high activity sample from lead and bricks as a function of their thickness. Is it noticeable that the attenuation capacity of lead is much superior than that of bricks for the same thickness (note: lead thickness is in mm and concrete is in cm).

The measurement instrumentation consisted in an ionising chamber located at a fixed distance from of a particular sample characterised by an elevated activity (17 MBq), as found by previous results (sample G47197[9]). Dose measurements were performed in different conditions: in absence of any shielding material, and in the presence of lead sheets and bricks of different thickness interposed between the ionisation chamber and the sample. The attenuation factor B was then computed as the ratio between the dose rate in presence and in absence of the attenuation materials. The results are reported in Figure 4. It is noticeable that the attenuation capacity of lead is superior than that of bricks, due to its higher density and larger atomic number. However, the choice of the proper material will depend on practical conditions such as cost, actual applicability of the shielding, available space (to get the same attenuation, bricks or concrete need a much larger space).

Figure 4 reports the measured transmission factors of the U_{nat} emission for lead and bricks. The first half-value layer of lead is 4 mm. This value is almost the same as for a monochromatic gamma radiation of about 500 keV and is in agreement with the mean energy of 520 keV of the uranium gamma ray spectrum (weighted with the branching ratio of each emission). In the same figure are reported the calculated transmission factors considering the full U_{nat} spectrum and the attenuation coefficient of each gamma emission. The calculated first TVL and equilibrium TVL are 21 and 30 mm, respectively.

Radon airborne issues

Most possibly the conservation of a collection of uraniferous minerals in a confined room induces the presence of a significant airborne due to the radon emission. Direct measurement of radon air concentration with active instrumentation is not trivial: the most viable (and precise) measurement can be performed by means of specific passive dosemeters, as in our case. EU Directive 13/59 states that the annual average activity concentration in air shall not be higher than 300 Bq m^{-3}. When radon air concentrations exceed such critical value, actions must be taken, with the support of a radiation protection expert, in order to reduce the radon air concentration below that level. In many cases such actions consist in the introduction of adequate ventilation systems. With the aim of evaluating the radon emission rate, a mathematical model was developed, assuming conservation conditions similar to those of our reference collection (storage room connected to a second—warehouse—room). The model requires as input the radon concentration levels in both rooms.

A first assumption is that radon gas generated by the mineral specimens spreads from the storage room to the warehouse, where it is confined. The time evolution of radon concentration in the two rooms can be expressed as follows:

$$\frac{dC_0(t)}{dt} = -\lambda \cdot C_0(t) + \frac{F}{V_0} \cdot A - \frac{R}{V_0} \cdot [C_0(t) - C_1(t)] \quad (2)$$

$$\frac{dC_1(t)}{dt} = -\lambda \cdot C_1(t) + \frac{R}{V_1} \cdot [C_0(t) - C_1(t)] \quad (3)$$

with initial conditions

$$C_0(0) = 0$$
$$C_1(0) = 0 \quad (4)$$

C_0 (t) and $C_1(t)$ are the time-dependent radon concentrations $(Bq\,m^{-3})$ in the storage room and in the warehouse respectively, A(Bq) is uranium activity of the whole collection, F is the fraction of A emitted in the environment as radon gas in the time unit (s^{-1}), R is the air flow rate (m^3 s^{-1}) between the two rooms, V_0 and V_1 are the volumes of the storage room and the warehouse, respectively, and λ is the radon decay probability per unit time (equal to 2.1×10^{-6} s^{-1}).

In the condition of equilibrium concentration in the two rooms (for $t = \infty$, in practice for $t >> \frac{1}{\lambda}$), solutions of equations (3) and (4) are

$$C_0 = \frac{A \cdot F \cdot (R + V_1 \cdot \lambda)}{\lambda \cdot (R \cdot (V_0 + V_1) + V_0 \cdot V_1 \cdot \lambda)}$$
$$C_1 = \frac{A \cdot F \cdot R}{\lambda \cdot (R \cdot (V_0 + V_1) + V_0 \cdot V_1 \cdot \lambda)} \quad (5)$$

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These formulas allow the calculation of radon concentrations in both rooms if the values of F and R are known, or, conversely, their calculation after the measurements of C_0 and C_1 (in equilibrium conditions), which is

$$F = \frac{C_0 V_0 + C_1 V_1}{A}\lambda \tag{6}$$

$$R = \frac{C_1}{C_0 - C_1} V_1 \lambda \tag{7}$$

In Appendix A, different scenarios are considered, which, starting from the above equations, allows to evaluate the minimum air flow rate to maintain the radon concentration below the stated threshold value.

Manipulation issues

Manipulation of radioactive minerals may be necessary in situations such as transport of one or more specimens to different locations, cataloguing operations or radiometric measurements, such as the ones performed in[9]. In order to address the radioprotection issues in this context, we analyse the following scenario, taking place in the storage room and in the adjacent warehouse room. In the proposed scenario, the operations under considerations are as follows:

1. A sample of the collection is selected and withdrawn from the storage room
2. The sample is carried in the adjacent room, where it is positioned on a table.
3. The sample is visually examined, the inventory number is read and a picture is taken.
4. A radiometric measurement is performed by placing a detector at a specific distance (previously marked on the table) from the sample.
5. The specimen is then put back in the storage room, into its polystyrene box.

In order to quantify the dose received by the personnel who conducts the handling, we hypothesise the timing of the different phases and the source-to-worker distances during the procedure as summarised in Table 5. It is noticeable that during phases (1) and (5), the personnel is inside the storage room containing the full collection.

Although the numerical dose results reported below do refer to our specific collection, considered as reference, the dose calculation procedure can be considered of general applicability to any collection, by simply introducing into the equations the appropriate numerical values.

Effective dose due to gamma radiation

This effective dose consists in two contributions: the first, D_1, is due to the gamma radiation emitted from the whole collection,the other, D_2, is due to the handled sample.

The first contribution, occurring during phases (1) and (5), can be quantified from the ambient radiation measurement. In our reference case, from the radiometric results reported above, we obtain

$$\begin{aligned} D_{1(1+5)} &= 4.1\left[\frac{\mu\text{Sv}}{\text{h}}\right] \cdot \frac{40}{3600}[\text{h}] \\ &\simeq 45\,\text{nSv} \end{aligned} \tag{8}$$

where we have considered 40 s of residence time in the storage room and a dose rate of 4.1 μSv h^{-1} (see Table 2), corresponding to the maximum rate measured close to the bricks wall (*D*6 in map n Figure 2).

As for the second contribution D_2, we consider the average activity A_{av} in a sample U_{nat} of the collection, computed as the total activity of the collection (1.46 GBq, as measured in our previous work[9]) divided by the number of samples (361), is equal to 4 MBq. Using the specific gamma-ray dose constant Γ[13, 14] of U_{nat} at a distance d_{ref}=30 cm (0.208 μSv/(h MBq^{-1})[9]), it is possible to estimate the dose received by an operator at a certain distance d_k from the sample for any of the k operations reported above as follows:

$$D_{2(k)} = A_{av} \cdot \Gamma \cdot \left(\frac{d_{ref}}{d_k}\right)^2 \cdot \tau_k \tag{9}$$

where d_k is the sample-operator distance and τ_k the time duration of the k phase (k = from 2 to 4).

The dose values can be easily obtained by introducing in equation 9 the actual times τ_k and the distances d_k, of the different phases, for example those reported in Table 5.

The total dose received by the operator is obtained by summing the contributions of equations 8 and 9 (in the case of a single operator in charge of all the handling procedures):

$$D_{gamma} = D_{1(1+5)} + \sum_{(2)}^{(4)} D_{2(k)} \simeq 70\,\text{nSv} \tag{10}$$

Effective dose due to radon

The dose due to radon inhalation can be estimated by considering the conversion factor J between radon concentration C (in equilibrium conditions) and the ambient dose rate DR, which is equal to $3 \cdot 10^{-9}$Sv (Bq^{-1}/m^{-3} h^{-1})[15]. In our specific reference case, taking into account the radon concentration

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Table 5. Estimate of the main exposure parameters during the handling operations.

Measurement phase	Time (s)	Operator-to-source distance (cm)	Location	Exposed body parts
(1)	20	30	Storage room	Body and hands
(2)	40	30	Storage room	Body and hand
(3)	20	40	Warehouse	Body
(4)	30	50	Warehouse	Body
(5)	20	30	Storage room	Body and hand

reported in Table 3, the maximum effective dose rate results $DR_w = 4\ \mu\text{Sv h}^{-1}$ in the warehouse and $DR_s = 46\mu\text{Sv h}^{-}$ in the storage room.

Considering the measurement phases reported in Table 5, it is possible to compute the dose due to radon (that is proportional to the residence time in each room) for a single handling specimen:

$$\begin{aligned} D_{radon} &= DR_s \cdot (\tau_1 + \tau_2 + \tau_5) + DR_w \cdot (\tau_3 + \tau_4) \\ &\simeq 1\mu Sv \end{aligned} \quad (11)$$

Equivalent skin dose (hand) evaluation

Since the specimens are handled directly by hand, equivalent skin dose evaluation is necessary. Skin dose arises from gamma and beta emission contributions, being the latter characterised by a high linear energy transfer within the tissues. On the contrary, alpha particles do not contribute to the dose because they are completely blocked by the first stratum corneum of the skin (of thickness around 70 μm), therefore their contribution to the skin dose (and to the dosemeters reading) as well is negligible. The relevant organ at risk is the hand skin.

Considering the superficial dose rate of a specimen, DR_{sup}, the skin dose D_{skin} due to the handling of a specimen can be calculated by multiplying DR by the handling time τ. By example we can refer to the results of the contact dose measurements reported in Table 4 ($DR = 256\mu\text{Sv h}^{-1}$). Is it possible to estimate the average equivalent skin dose due to the handling of a single specimen, during the phases 1, 2 and 5 reported in Table 5:

$$D_{skin} = D_{skin(1)} + D_{skin(2)} + D_{skin(5)} \simeq 2\mu Sv \quad (12)$$

Final consideration about personnel dose during handling

The total effective dose received during the full procedure, by both source irradiation and radon inhalation, is given by the sum of the results of equations 10 and 11.

From the results discussed above, it appears clear that the considered handling procedure can be conducted with negligible radiation risk[15].If the procedure is carried out by more than one worker, then the individual dose is even lower.

Exhibition issues

Radioprotection of workers

In the case of an exhibition of a few samples of the collection, evaluation of the ionising radiation risk for the visitors and the workers, such as custodian or cleaning staff, is necessary. Parameters to be taken in consideration are the geometry of the exhibition room, in particular the minimum distance from the specimens, the time spent by workers or visitors in proximity to the exhibited samples and the dose reduction achievable by means of the shielding materials.

In order to refer to a significant case, we consider to exhibit the sample of our collection characterised by the largest activity, already used to measure the shielding efficiency of the materials as reported in Figure 4 (sample G47197[9]).

As already described above, the dose rate at a fixed distance can be computed by means of the specific gamma ray dose constant of U_{nat} $\Gamma = 0.208 \frac{\mu Sv}{h^{-1} MBq^{-1}}$ at 30 cm[9]. For the considered specimen, whose activity previously measured is equal to 17 MBq, the dose rate at 30 cm is

$$DR_{30cm} = 0.208\left[\frac{\mu\text{Sv}}{\text{hMBq}^{-1}}\right] \cdot 17[\text{MBq}] \simeq 3.5\left[\frac{\mu\text{Sv}}{\text{h}}\right] \quad (13)$$

Let's now consider the personnel working in the exhibition room (custodian and cleaning staff), with the following working scenario:

- Custodian ($w1$): occupational factor 2000 h y^{-1}, average source distance 4 m;
- cleaning staff ($w2$): occupational factor 30 h y^{-1}, average source distance 70 cm.

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Under these conditions, we obtain

$$D_{w1} = 3.5\left[\frac{\mu\text{Sv}}{\text{h}}\right]\cdot\left(\frac{30[\text{cm}]}{400[\text{cm}]}\right)^2\cdot 2000\left[\frac{\text{h}}{\text{y}}\right] \simeq 40\frac{\mu\text{Sv}}{\text{y}} \quad (14)$$

$$D_{w2} = 3.5\frac{\mu\text{Sv}}{\text{h}}\left(\frac{30[\text{cm}]}{70[\text{cm}]}\right)^2\cdot 30\left[\frac{\text{h}}{\text{y}}\right] \simeq 20\frac{\mu\text{Sv}}{\text{y}} \quad (15)$$

In conclusion, no radioprotection measures should be adopted in case of personnel working within discussed scenario, consisting in the exhibition of few samples of the collection, being the average distance from the specimen large enough to keep the dose well below the limits of international regulations[15].

Let's now consider a situation characterised by a more relevant occupational factor: the presence of an office close (possibly behind) to the display cabinet, with a desk at a distance of 50 cm from the source and an occupational factor of 2000 h y^{-1}. The dose rate is

$$D_{w3} = 3.5\left[\frac{\mu\text{Sv}}{\text{h}}\right]\cdot\left(\frac{30[\text{cm}]}{50[\text{cm}]}\right)^2\cdot 2000\left[\frac{\text{h}}{\text{y}}\right] \simeq 2.6\frac{\text{mSv}}{\text{y}} \quad (16)$$

where we have neglected the shielding effect of the wall between the office and the exhibition room. In order to let the effective dose not to exceed the value of 1 mSv y^{-1}[15], the transmission factor B of the wall must be

$$D\cdot B < 1\text{mSv} \Rightarrow B < 0.38 \quad (17)$$

Therefore, an evaluation of the shielding performance of the wall is needed: if the protection ensured by the wall is not enough, additional shielding must be prearranged to guarantee the respect of the limit.

Radioprotection of visitors

Since monitoring of individual doses received by the visitors is not a viable procedure, precautionary measures must be adopted in order to guarantee the ionising radiation risk being negligible, in agreement with local and international regulations. If we consider the most cautelative scenario, this condition is satisfied when the individual dose is below 10 μSv/year (dose limit for artificial radionuclides[15]).

We cautionary consider an exhibition geometry such that the average visitor to source distance is 70 cm and the average time spent by the visitor in front of the specimen is 1 h. In such conservative conditions, the individual dose is

$$D = 3.5\left[\frac{\mu\text{Sv}}{\text{h}}\right]\left(\frac{30[\text{cm}]}{70[\text{cm}]}\right)^2\cdot 1\text{ h} \simeq 0.7\mu\text{Sv} \quad (18)$$

We can conclude that a visitor can observe the sample up to a maximum of 15 h in 1 y with no further precaution to be taken.

Finally, it is possible to assess the eye equivalent dose arising from the exhibited sample. Considering the eye lens of a visitor to be at a distance of 30 cm from the sample and an exposure time of 1 h, the equivalent dose due to gamma radiation, as computed from equation (18), is equal to 3.5 μSv, which is more than 1000 times lower than the dose limit for the public (15 mSv according to European Council Directive 2013/59[15]). The uriniferous samples also emit alpha radiation (fully absorbed by the interposed air) and beta radiation with about 0.5-MeV mean energy and with 3-MeV maximum energy. Considering the maximum energy and a glass display case with density 2.4 g cm^{-3}, the resulting range of the electrons (CSDA approximation)[16] is 0.7 cm (0.1 cm if considering the mean energy). Typical glass thickness of museum display cases is of the order of 1 cm; consequently, all the beta radiation is absorbed and does not contribute to the visitors exposure.

In conclusion, the exhibition of even the most active sample of our reference collection does not require precautionary measures to be adopted because the public exposure is negligible. As already stated, every situation must be analysed in details to adjust the results to the specific case under consideration, since the final dose evaluation can strongly be affected by several parameters, such as the activity of the samples, the geometry conditions, and the crowding conditions of the museum.

Conclusion

In this work, we examined all the various radioprotection implications associated with the possession of a collection of uraniferous minerals, from the storage of the whole collection up to the public exhibition of a single specimen. For all the connected activities and for many of the possibly related scenarios we developed specific dose calculation formulas enabling the corresponding radiation doses evaluation, either effective or equivalent or both, for workers and public

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as well. Dose calculation formulas have been applied to an actual collection that had been previously fully characterised in terms of activity determination for every single specimen[9]. We found that while the amount of radioactivity within the whole collection may require radioprotection actions such as gamma rays shielding and radon air pollution recovery, the exhibition of a single specimen is, under normal conditions, safe and totally viable from the radioprotection point of view. Limited handling of a reduced number of specimens is also safe. However, if a prolonged handling of the specimens is planned, then wearing common medical lead gloves may help reducing significantly (up to 30%) the hands skin dose. We limited our investigation to the radioprotection implications relating to the possession of an uraniferous collection, avoiding addressing the formal aspects, possibly of cumbersome fulfilment, as strongly dependent on national regulations. As a final remark, we feel that we can consciously state that this paper can be used as a guide to other holders of uraniferous minerals to quantify the radiological risk associated to its detention, handling or exhibition.

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Ventilation required to maintain Radon concentration below reference levels. A review of significant scenarios

When considering a standard storage condition for the collection of uraniferous minerals such the one described in this paper, the radon air concentrations are ruled by the formulas of equation (4). Let now consider two particular scenarios. The first with $R \simeq 0$, i.e. there is no radon diffusion between two rooms, the concentration of radon gas in the first room is maximum and in the other is null. By applying this condition to equtaion (4), we obtain

$$C_0 = C_{Max} = \frac{A \cdot F}{\lambda \cdot V_0} \tag{A.1}$$

$$C_1 = 0 \tag{A.2}$$

In the second scenario, the value of R is maximum, i.e. there is a perfect mixing between radon gas in the two rooms; this is achieved if $R >> V_1 \cdot \lambda$. In this case, the concentration in both rooms is equal and is

$$C_{min} = \frac{A \cdot F}{\lambda \cdot (V_0 + V_1)} \tag{A.3}$$

This corresponds to the minimum achievable value of radon gas concentration in the storage room, that depends on the total of volumes of the two rooms.

According to the European regulations, the radon reference level is $C_{ref} = 300$ Bq m^{-3}[15]. Considering the proposed model, three possible cases occur:

$$\begin{aligned} &(a) \qquad C_{ref} > C_{Max} \\ &(b) \qquad C_{min} \geq C_{ref} \geq C_{Max} \\ &(c) \qquad C_{ref} < C_{min} \end{aligned} \tag{A.4}$$

(*a*) In this case, the radon gas air concentration is always lower than reference level. (*b*) The radon gas air concentration may be kept lower than reference level in both rooms if enough air flow rate R is implemented, that is (from equation 3)

$$R \geq V_1 \cdot \lambda \cdot \frac{AF - C_{ref} V_0 \lambda}{C_{ref} \lambda (V_0 + V_1) - AF} \tag{A.5}$$

(*c*) This scenario is characterised by an air concentration of radon gas larger than reference levels in the storage room for any R value.

This last event leads to a further consideration that is about V_1. If $V_1 >> V_0$ (i.e. the 1-room is outdoor) then the condition $C_0 < C_{ref}$ is always possible provided that enough air flow rate R is accomplished. By referring to equation (A.5), this assumption means that V_0 is negligible when compared to V_1 and, at the same time AF is negligible when compared to $C_{ref} \lambda V_1$; therefore, the condition is accomplished when

$$R \geq \frac{AF - C_{ref} V_0 \lambda}{C_{ref}} \tag{A.6}$$

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