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REVIEW ARTICLE

RADIATION PROTECTION IN THE NUCLEAR MEDICINE CENTER FROM PET RADIONUCLIDE

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Abstract. *The purpose of this article is to present a methodology for determining sufficiency of the wall thickness to protect patients from the non-therapeutic effects of a medical source of ionizing radiation and calculating its increase in a way alternative to the use of the Monte Carlo method. The article is based on a case with a specific source of ^{18}F activity of 4 Ci in a tungsten container with a wall thickness of 29.5 mm, which are used in the Center of Nuclear Medicine of the Republic of Kazakhstan. The objectives of the study are the following: calculation of the exposure dose rate from a radionuclide source of certain activity; calculation of the multiplicity of the dose attenuation rate to ensure human safety and establishing the necessary thickness of protection from the opted building material. The results of these calculations may differ depending on the method of direct and reverse recalculation of various radiation doses units, and therefore the conclusion suggests the most optimal of them in terms of physical efficiency, economic feasibility and regulatory performance. Operational dosimetry solves the problem of implementing the ALARA principle, problems of optimizing radiation safety are solved. Reducing the dose load on a person is achieved by reducing the operating time, increasing the distance to the radiation source and using a protective screen. The article describes the choice of protective material and its thickness for protection against radionuclide sources.*

Keywords: *exposure dose, absorbed dose, equivalent dose, effective dose, dose rate, attenuation rate, protective shield.*

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ШОЛУ МАҚАЛАСЫ

ЯДРОЛЫҚ МЕДИЦИНА ОРТАЛЫҒЫНДАҒЫ ПЭТ РАДИОНУКЛИД КӨЗІНЕН РАДИАЦИЯЛЫҚ ҚОРҒАНЫС

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Аңдатпа. *Бұл мақаланың мақсаты науқастарды иондаушы сәулеленудің медициналық көзінің терапиялық емес сәулеленуінен қорғау және Монте-Карлоға балама әдіс бойынша оның ұлғаюын есептеу үшін қабырға қалыңдығының жеткіліктілігін анықтау мәселесін шешуін ұсыну. Есептеу үшін бастапқы деректер ретінде Қазақстан Республикасының Ядролық медицина орталығында пайдаланылатын қорғаныш қабырғасының қалыңдығы 29,5 мм вольфрам контейнеріндегі белсенділігі 4 Ки болатын* ^{18}F *радионуклидтік көзі қаралды. Зерттеудің міндеттері: белгілі белсенділіктің радионуклидті көзінен экспозициялық дозаның қуатын есептеу, адам қауіпсіздігін қамтамасыз ету үшін оның әлсіреу жиілігін есептеу және бұл үшін қажетті құрылыс материалынан қорғаныс қалыңдығын анықтау. Бұл есептеулердің нәтижелері сәулеленудің әртүрлі түрлерінің бірліктерін тікелей және кері санау әдісіне байланысты өзгеруі мүмкін, осыған байланысты қорытынды физикалық тиімділік, экономикалық орындылық және нормативтік-құқықтық орындау тұрғысынан олардың ең оңтайлысын ұсынады. Оперативті дозиметрияда негізінен радиациялық қауіпсіздікті оңтайландыру қағидатын іске асыру жөніндегі мәселелер шешіледі, ол нормалаумен және негіздеумен тығыз байланысты, сондай-ақ онда жұмыс уақытын азайту, радиоактивті материалға дейінгі қашықтықты ұлғайту және иондау ағынын әлсірететін қорғаныш экранды орнату есебінен иондаушы сәулелену көздерін пайдалану кезінде адамға түсетін дозалық жүктемені іс жүзінде қол жетерлік мәнге дейін азайту ұйғарылады. Операциялық дозиметрия ALARA принципін енгізу мәселесін шешеді және радиациялық қауіпсіздікті оңтайландыру мәселелерін шешеді. Адамға дозалық жүктемені азайту жұмыс уақытын қысқарту, сәулелену көзіне дейінгі қашықтықты ұлғайту және қорғаныс экранын пайдалану арқылы қол жеткізіледі. Мақалада радионуклидті көздерден қорғау үшін қорғаныс материалын таңдау және оның қалыңдығы сипатталған.*

Түйін сөздер: *экспозициялық доза, сіңірілген доза, эквивалентті доза, тиімді доза, доза қуаты, әлсіреу еселігі, қорғаныс экраны*.

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ОБЗОРНАЯ СТАТЬЯ

РАДИАЦИОННАЯ ЗАЩИТА ОТ РАДИОНУКЛИДНОГО ИСТОЧНИКА ПЭТ В ЦЕНТРЕ ЯДЕРНОЙ МЕДИЦИНЫ

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Аннотация. *Цель данной статьи заключается в представлении методики определения достаточности толщины стены для защиты пациентов от нетерапевтического облучения медицинским источником ионизирующего излучения и расчету ее увеличения по методу, альтернативному Монте-Карло. В качестве исходных данных для расчета рассмотрен радионуклидный источник ^{18}F с активностью 4 Ки в вольфрамовом контейнере с толщиной защитной стенки 29,5 мм, какие используются в Центре ядерной медицины Республики Казахстан. Задачами исследования являются: расчет мощности экспозиционной дозы от радионуклидного источника известной активности, вычисление кратности ее ослабления для обеспечения безопасности человека и установление необходимой для этого толщины защиты из выбранного строительного материала. Результаты данных вычислений могут варьироваться в зависимости от способа прямого и обратного пересчета единиц различных видов доз облучения, в связи с чем вывод предлагает наиболее оптимальный из них с точки зрения физической эффективности, экономической целесообразности и нормативно-правовой исполнительности. Оперативная дозиметрия решает задачу реализации принципа ALARA, решаются задачи оптимизации радиационной безопасности. Снижение дозовой нагрузки на человека достигается сокращением времени работы, увеличением расстояния до источника излучения и использованием защитного экрана. Статья описывает выбор защитного материала и его толщины для защиты от радионуклидных источников.*

Ключевые слова: *экспозиционная доза, поглощенная доза, эквивалентная доза, эффективная доза, мощность дозы, кратность ослабления, защитный экран.*

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1 INTRODUCTION

According to the Hygienic standards **(Ğiniat, 2022)** and Sanitary and epidemiological requirements for ensuring radiation safety **(Tsoy, 2020)**, the main criterion for the safety of systems used for storing and transporting ionizing radiation sources is not exceeding the radiation doses of personnel (20 mSv) per 1700 hours for "A" group and 5 mSv per 2000 hours for "B" group) and the population (1 mSv per 8800 hours).

The multiplicity of attenuation of the dose rate of ionizing radiation generated by the radionuclide source contained in the package is ensured by implementing the ALARA principle by using the walls of the transport packaging set as a protective shield in 4π geometry. In accordance with the requirements of hygienic standards **(Ğiniat, 2022)** %and sanitary rules **(Tsoy, 2020)** on radiation safety, the acceptable equivalent dose rate when working with an ionizing radiation source, including its transportation in a container, to ensure uniformity of occupational exposure of personnel should not exceed 11.76 mSv/h under normal operating conditions or 117.65 mSv/h – in case of emergency (based on the dose of the planned increased exposure equal to 200 mSv **(Ğiniat, 2022)**.

In the simplest case, the protective material and its thickness for shielding from ionizing radiation source is chosen taking into account the multiplicity of the half attenuation of the flow of photons and ionizing particles. There is a directly proportional relationship between the thickness of the protection and the density of the shielding material.

2 LITERATURE REVIEW

In the simplest case, the number of half-attenuation layers, depending on the required attenuation multiplicity, is determined by the formula:

$$k = 2^n, \tag{1}$$

where *k* is the required attenuation multiplicity, and *n* is the number of layers of half attenuation.

In order to avoid irrational calculation of n through the logarithm of k by the formula:

$$n = \frac{\ln k}{0,693}, \tag{2}$$

we can use approximate data from **Table 1**.

Table 1
Approximate relationship between the multiplicity (k) and the number of half-attenuation layers (n) [3]

k	2	4	8	16	32	64	125	250	500	1000
n	1	2	3	4	5	6	7	8	9	10

If it is required to attenuate the radiation intensity by 8000 times, i.e. 8000 = 2^n, then k can be calculated by adding n layers necessary to provide 8- and 1000-fold attenuation:

$$n = 3 + 10 = 13. \tag{3}$$

To determine the effective thickness of protective shield when working with radionuclide source of specific activity, we need to use the formula:

$$P_X = \frac{\lambda \times A}{R^2}, \tag{4}$$

где P_X is the exposure dose rate (R/hr); λ – is the decay constant (for ^{18}F, $\lambda = 6.95\ \frac{R \times mCi}{hr \times cm^2}$ **(Melikhova, 2023)**); A is the activity of the source (mCi); R is the distance from the source (cm).

By calculating the equivalent dose rate (P_H) or measuring its value using a dosimeter, it is possible to calculate the attenuation multiplicity, which would allow to select the thickness of shield according to the reference book **(Melikhova, 2023)**.

The frequency of attenuation of ionizing radiation is calculated as the ratio of the effective dose rate from a certain source to the permissible annual dose rate limit set for “A” group personnel:

$$k = \frac{P_E}{P_{per}}, \tag{5}$$

where k is the multiplicity of attenuation of ionizing radiation, P_E is the effective dose rate from the source of ionizing radiation, P_{per} is the permissible dose rate for the personnel of group A, B or the population, depending on the category of protected persons. $P_{add.}$ for the personnel of “A” group is 11.76 mSv/h, for the personnel of “B” group it is 2.5 mSv/h, and for the population – 0.11 mSv/h.

It is known that the activity of the produced source with the isotope ^{18}F is 4 curies (A = 4,000 mCi). This means that the exposure dose rate from an unprotected source at a distance of 1 meter (R = 100 cm) would be:

$$P_X = \frac{6.95 \times 4{,}000}{100^2} = 2.78\, {}^{R}\!/_{hr}. \tag{6}$$

To further determine the multiplicity of attenuation of the effective dose rate, it is necessary to carry out a number of transformations that cannot be limited by the assumption that 1 Sievert is equal to 100 roentgens (1 Sv ≠ 100 R). If the exposure dose rate (P_X) from ^{18}F source out of protective container, measured at a distance of 1 meter (100 cm) or calculated by formula (4), is 2.78 R/hr, then it should be taken into account that in the conditions of electronic equilibrium, when the total energies of electrons leaving and entering a certain volume are equal, the following correlation is established between the units of measurement of the exposure and absorbed dose in the air:

$$1R = 0.0087Gy, \tag{7}$$

i.e., the exposure dose of 1 R corresponds to the absorbed dose in the air equal to 0.87 rad, however, this value differs from the dose that a person would receive if exposed to radiation in the same field. The dose in the air is caused mainly by the presence of nitrogen, carbon and oxygen, whereas biological tissue contains hydrogen (10.1% by weight), characterized by twice the absorption capacity compared to other elements, therefore, the biological tissue exposure dose of 1 R corresponds to the absorbed dose equal to 0.0095 Gy.

Since for X-ray, beta and gamma-radiation, the equivalent dose is equal to the absorbed one, then 1 R/hr can be taken equal to 0.0095 Sv/hr, and the equivalent dose rate (P_H) from the considered source ^{18}F at a distance of 1 m would be:

$$P_H = 2.78 \times 0.0095 = 26.41 mSv/hr \tag{8}$$

The effective dose rate (P_E) of external irradiation is calculated by the formula:

$$P_E = P_H \times K_T, \tag{9}$$

where K_T is the coefficient of transition from an equivalent dose to an effective one, which is equal to:

1. 0.642 – due to the division of 0.7 by 1.09. The UN Scientific Committee on the Effects of Atomic Radiation recommended to calculate the effective dose from all radionuclides in the environment on the base of the absorbed dose in the air by using a conversion factor of 0.7 **(Atamanyuk et al., 2023)** to account for radiation shielding by various organs and tissues and its backscattering, and indicated more accurate coefficients for converting the absorbed dose in the air into an effective dose for ionizing particles and photons of various energies. When measuring the absorbed dose with air equivalent dosimeters, it is required to translate the measured results into units of equivalent dose in biological tissue, which requires knowledge of the ionizing radiation spectrum. In the absence of data on the spectrum and small differences in the dose rate from the background, the transition coefficient from the absorbed dose in the air to the equivalent dose in the tissue can be assumed to be equal to 1.09 as a result of the ratio of the equivalent of an off-system unit of the exposure dose (1 R) in biological tissue (0.0095 Gy) to a similar indicator in the air (0.0087 Gy);

2. 1.903 is the maximum value of the conversion coefficient of the air KERMA into an equivalent dose ($H_p(6)/K$) in a plate phantom according to Table III.1a **(IAEA Safety Standards, 2015)**;

3. 1 – according to the definition of the effective dose in the sanitary rules **(Tsoy, 2020)**, since the sum of the weighting coefficients taking into account the radiosensitivity of various organs and tissues is equal to 1.

In the first case, the exposure dose rate of 2.78 R/hr would be equal to the effective dose rate of 16.96 mSv/hr, in the second case – 50.26 mSv/hr, in the third case – 26.41 mSv/hr.

The necessary multiplicities of attenuation of these dose rates, calculated according to formula (5) for various categories of protected persons, would be respectively:

- for “A” group personnel – 1,442; 4,274 and 2,246;
- for “B” group personnel – 6,784; 20,104 and 10,564;
- for the population – 154,182; 456,909 and 240,091.

If we use the classical definition of the effective dose from the sanitary rules **(Tsoy, 2020)** as the result of multiplying the equivalent dose by the sum of the weighting coefficients of the radiosensitivity of organs and tissues of the whole organism, equal to 1, and also take into account the margin factor in the design of biological protection equal to 2, then according to the principle of hyperprophylaxis of potential radiation hazard, the calculation of the effectiveness of protecting personnel and population from the source ^{18}F of 4 Ci activity in a tungsten container with a wall thickness of 29.5 mm should be based on the following data:

– the effective dose rate (P_E) at a distance of 1 m from an unprotected source ^{18}F of 4 Ci activity is 50.26 mSv/hr;

– the multiplicity of attenuation (k) of the effective dose rate from an unprotected source ^{18}F of 4 Ci activity is equal to 4,274 – for “A” group personnel; 20,104 – for “B” group personnel and 456,909 – for the population.

At first glance, the validity of the chosen calculation option is not confirmed by the dose equivalents that had being measured for over 2 hours by means of the Harshaw 6600 Lite individual dosimeters at various distances from ^{18}F source with an activity of 4 Ci **(Table 2)**.

Multiplication of the measured dose rate of 19.7 mSv/h by the maximum value of the conversion coefficient of the air kerma into an equivalent dose of 1.903 equals to 37.49 mSv/h, which is 25% less than the conservative calculated value of 50.26 mSv/h. This is due to the fact that the measurements were carried out by the Harshaw 6600 Lite instrument manufactured in the USA, where a different approach to determine the individual equivalent of the whole body radiation dose $H_p(10)$ is used. According to the domestic standard, the entire ionization energy is converted into

$H_p(10)$, whereas the American colleagues attribute the Compton scattering energy of the charged particles entering the limit of annual irradiation to the dose on the skin $H_p(0.07)$ **(Kim et al., 2019)**.

Table 2
2-hour individual dose equivalents by exposure to ^{18}F radionuclide source with an activity of 4 Ci [author's material]

N/a	Distance from the source R, m	Individual dose equivalent, mSv	Dose rate, mSv/hr.
1.	0.1	3.8×10^3	1.9×10^3
2.	0.5	1.5×10^2	75
3.	1	39.4	19.7
4.	5	1.6	0.8
5.	10	0.38	0.19

Considering that in order to determine the thickness of the protection in absolute terms by applying the data in **Table 1**, it is necessary to know the thickness of the layer of the deep attenuation of the absorbed dose rate of brake-, X-ray, gamma- and beta-radiation, which is 0.3 cm (30 mm) **(Dreyzin et al., 2022)** for tungsten with a density of 19.25 g/cm^3. In this case, the tungsten protection thickness of 5, 6 and 8 cm, respectively, is required to protect the personnel of groups "A", "B" and the population at 1 m from the source ^{18}F of 4 Ci activity during the full working day. However, this is a very rough approximation, since these results were obtained using the express method of choosing a protective material and its thickness, applicable only in an emergency situation.

Knowing the decay energy of ^{18}F radionuclide (E = 0.25 MeV) and the necessary multiplicity of attenuation of the created dose rate, it is easy to find the thickness of the tungsten wall, which would be:

– 3.6 cm – to protect "A" group personnel at 1 m from the source ^{18}F of 4 Ci activity during a full working day;

– 4.2 cm – to protect "B" group personnel at 1 m from the source ^{18}F of 4 Ci activity during a full working day;

– 5.5 cm – to protect the population at 1 m from the source ^{18}F of 4 Ci activity during a full working day.

3 MATERIALS AND METHODS

Today, the cornerstone in calculating the thickness of shielding against ionizing radiation is the determination of the effective dose, which characterizes the value of the risk of the long-term effects of radiation exposure overall human body and its individual organs and tissues, taking into account their radiosensitivity. The rule of calculation of the individual dose equivalent in the whole body to the effective dose is of the fundamental importance, since the concept of effective dose was introduced to assess the risk of stochastic effects of irregular irradiation of the whole body, and its values are now normalized values of radiation exposure on the human body, established by international and national standards. Calculation of effective dose according to the formula proposed by the sanitary rules **(Tsoy, 2020)** is rather complicated task due to the lack of information about the equivalent dose in organs, therefore, in practical work, there is used an approximate method based on the additivity of the effective dose determined by the formula (5).

Radiation safety hygienic standards **(Ğiniat, 2022)**, along with the main dose limits, also indicate dose coefficients in terms of effective or equivalent dose per unit of external radiation flux or 1 Bq of radionuclide in the body through the respiratory organs or food tract for the most critical age group and the most toxic chemical form of the radionuclide. Thus, if we include the multiplications of magnitudes of each monofactor effect on its dose ratio, then in sum, we obtain a value equal to or greater than the actual effective dose received. This method became the basis of the methodology for determining the maximum effective dose **(Kudryashev & Kim, 2019)**.

4 RESULTS AND DISCUSSION

Thus, the thickness of the tungsten protection of 29.5 mm is quite sufficient to ensure an acceptable level of radiation safety of “A” group personnel, especially if we take into account the short half-life of ^{18}F ($T_{1/2}$ = 109.77 min) and the exponential decrease in the dose load from one source per person over time.

If we assume that a tungsten container (wall thickness of 29.5 mm) with the source ^{18}F of 4 Ci activity is near an external wall or concrete partition adjacent to a permanent residence of population, in which the equivalent dose rate should not exceed 0.03 mSv/hr according to the sanitary rules **(Tsoy, 2020)**, then it is necessary to assess the sufficiency of the concrete thickness of 20 cm according to the project of the Center of Nuclear Medicine.

Calculated according to equations (6) and (8), the equivalent dose rate from the source ^{18}F of 4 Ci activity is 26.41 mSv/hr. If the wall of the tungsten container with a thickness of 29.5 mm, corresponding to the thickness of the polymer attenuation layer (30 mm), reduces the initial dose rate by half, i.e. up to 13.2 mSv/hr, then the additional attenuation multiplicity must be calculated by the formula (5):

$$k = \frac{13.2 \times 10^{3} \mu Sv/hr}{0.03 \mu Sv/hr} = 0.44 \times 10^{6} \quad (10)$$

Knowing the decay energy of ^{18}F radionuclide (E = 0.25 MeV) and the necessary multiplicity of attenuation of the created dose rate, we find that the required thickness of concrete protection is 70 cm (according to the reference book **(Melikhova,2023)**).

5 CONCLUSIONS

Thus, in order to comply with the requirements by maintaining the equivalent dose rate in a room for population at the level of 0.03 mSv/hr, it is necessary to increase the thickness of adjacent concrete walls and partitions up to 70 cm, if a tungsten container with ^{18}F source of 4 Ci activity stays near it. Depending on the method of determining the effective dose rate, its value varies by 3 times, which will certainly affect the magnitude of the attenuation ratio and, therefore, the shielding thickness, which would be either insufficient or unnecessarily expensive. In the first case, it is possible, guided by the principle of hyper-prevention of danger, to take advantage of the equality of effective and equivalent dose rate established by the sanitary rules. The choice of K_T equal to 0.642 would make it possible to heighten accuracy of calculation of the effective dose of internal irradiation, since the above coefficient takes into account the screening of radiation by various organs and tissues and its backscattering as well. In the second case, when the effective dose rate in corroboration with the requirements of the IAEA should be almost twice the equivalent, it is also reflected in Kazakhstan's sanitary rules, according to which the permissible equivalent dose rate in premises of permanent being of A group personnel is 6 μSv/hr, which is on average 1.9 times less than the quotient of the ratio of the annual limit of the effective dose (20 mSv) to the shortened working time (1,700 hr). The application of the third method, which implies the equality of the effective and equivalent dose modules, is convenient in assessing the external effects of multifactorial chronic exposure through instrumental IDC.

CONFLICT OF INTEREST

The authors state that there is no conflict of interest.

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