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Methods of calculation of radiation protection for operational safety optimization at working with radionuclide photon radiation sources

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Operational dosimetry solves the problem of implementing the principle of optimizing radiation safety, which is closely related to regulation and justification and implies a realistic achievable reduction in the dose load on a person when using ionizing radiation sources by reducing operating time, increasing the distance to radioactive material and shielding for attenuation of ionization flux. The article describes two ways of determining the thickness of shielding under necessity of making operational decision to protect a person from radionuclide source of known activity or the dose rate created by it. Based on the required multiplicity of its attenuation, which allows determining the number of half attenuation layers in the shield or its thickness, is also dependent on the energy of radiation, emitted by the source.

Keywords: radiation safety, dose rate, attenuation rate, optimization principle, exposure dose, absorbed dose, equivalent dose, effective dose, protective shield.

Introduction

Radiation safety of personnel, population and the natural environment is ensured by implementation of a set of organizational, technological, technical, sanitary-epidemiological and medical preventive measures aimed at realizing the basic principles of radiation safety: justification, optimization and regulation.

The basis of the justification principle is prohibition of all types of work on the use of ionizing radiation sources, in which the benefits obtained for people and society do not exceed the risk of possible harm caused by irradiation that is additional to the natural radiation background.

Ways to implement the ALARA principle involve the provision of radiation protection by reducing the time of interaction with an ionizing radiation source, increasing the distance to it, shielding with a material that attenuates penetrating radiation, and minimizing the amount of radio active substance or material at a workplace in order to comply with the norms of positivity, which determine the levels of monofactor exposure to radionuclide sources of ionizing radiation (for radionuclide depending on the route of intake or the type of external exposure) established in the hygienic standards for radiation safety [1].

The regulation principle is closely related to the principles of justification and optimization, since comparing the results of measuring the physical factors causing the radiation hazard with their permissible or normalized values underlies the assessment of the radiation risk, as well as calculations of the duration of work with sources of ionizing radiation in specified amount at a certain distance from them behind a protective shield of the certain material and thickness. The regulation principle is implemented mainly by measuring or calculating a person's radiation dose to determine compliance of labor conditions with health regulations and hygiene standards of radiation safety, which is the general task of an individual dosimetric control (IDC).

Formulation of the problem

Not exceeding the permissible doses of radiation in occupational conditions is provided by:

– limiting the time of interaction with ionizing radiation sources as a result of dividing the maximum permissible dose by the measured value of the dose rate at the place of work according to the formula:

$$t = \frac{E_L}{P_H \times K_T}, \tag{1}$$

where t is the allowable time of working with IRS (hr), E_L is the effective dose limit (Sv), P_H is the equivalent dose rate (Sv/hr); K_T is the transition rate from equivalent to effective dose, equal to 0.642; 1 [1] or 1.903 [3, 4], depending on the approach to determining the effective dose;

– limiting the distance to IRS;

– shielding from IRS.

Calculation of relatively safe distance from radiation source and the thickness of shielding from it can be considered by the case of using gamma NDT with a radionuclide source (usually with the iridium radioisotope ^{192}Ir) of high activity. If the activity of ionizing radiation sources with iridium ^{192}Ir isotopes is 200 curies, then the safe distance from it can be determined by the formula:

$$P_X = \frac{K_\gamma \times A}{r^2}, \tag{2}$$

where, P_X is the exposure dose rate (R/hr); $K_\gamma = 4.6\frac{R \times mCi}{hr \times cm^2}$ is the gamma-radiation constant for ^{192}Ir [5]; A is the source activity (mCi); r is the distance from the source (cm).

At operations with ionizing radiation sources of high activity, radiation monitoring should be carried out using an instrument with an upper limit of measurement of at least 100 mSv/hr and not enter radioactively contaminated zones with ambient dose rates of more than 200 mSv/hr, and the sanitary post should be deployed at a distance where the dose rate will not exceed 0.3 μSv/hr [6, 7].

Goals

Radiation shielding in order to reduce the level of radiation risk for humans and the environment is important and even more difficult task of radiation safety compared with the limitation of the operating time and distance from the radionuclide ionizing radiation source.

Methods for determining the shielding thickness

Selection of protective material and its thickness for shielding from IRS in the simplest case is made taking into account the half-attenuation of photons flow and ionizing charged particles. There is direct proportionality between the shielding thickness and the density of shielding material. For example, for half attenuation of gamma-radiation emitted by the cobalt isotope of ^{60}Co of a certain activity, the shielding thickness of lead is 1.3 cm, of iron – 3.6 cm, of concrete – 13 cm, of modified arbolite – 20 cm [8], and of water – 27 cm.

The number of half-attenuation layers depends on the required attenuation ratio and is determined by the formula:

$$k = 2^n, \tag{3}$$

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

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

$$n = \frac{\ln k}{0.693}, \tag{4}$$

than the approximate data of Table 1 can be used.

In the case of need to weaken the radiation flow 8.000 times, i.e. 8.000 = 2^n according to (1), then k can be calculated either by (3) or by adding the layers n necessary to ensure 8 - and 1000 - fold attenuation:

$$n = 3 + 10 = 13, \tag{5}$$

Table 1.
Approximate relationship between multiplicity (k) and number of half - attenuation layers (n).

k	2	4	8	16	32	64	125	250	500	1.000
n	1	2	3	4	5	6	7	8	9	10

To determine in absolute terms the effective shielding thickness when working with radionuclide source of certain activity, it is necessary to use formula (2) again. By calculating the equivalent dose rate (P_H) or measuring its value using a dosimeter, one can calculate the attenuation ratio, which will allow choosing the thickness of shielding from a particular material using the reference book [5].

The multiplicity of attenuation of ionizing radiation is calculated as the ratio of the equivalent dose rate from a certain source to the allowable annual dose rate limit established for the personnel of A group:

$$K = \frac{P_H}{P_{adm}}, \tag{6}$$

where K is the multiplicity of attenuation of ionizing radiation, P_H is the equivalent dose rate from the source of ionizing radiation, $P_{adm.}$ is admitted dose rate for personnel of A group.

According to hygienic standards [1], the permissible dose of annual occupational exposure to personnel of A group, which includes all collaborators working directly with sources of ionizing radiation, is equal to 20 mSv. Also, the hygienic standards admits a shortened annual operating time of 1.700 hours for the personnel of A group [1]. Thus, it is possible to calculate the permissible dose rate for the personnel of A group:

$$P_{adm.} = \frac{20}{1.700} = 11.76\ \mu\text{Sv/hr}, \tag{7}$$

If the equivalent dose rate (P_H) at a distance of 0.5 m from IRS with ^{137}Cs isotope of 67 GBq (1.810 mCi) activity is 22.3 mSv/hr, then the required attenuation multiplicity (k) will be 1.896:

$$k = \frac{22.300}{11.76} = 1.896, \tag{8}$$

Knowing the required attenuation ratio, the energy of IRS with ^{137}Cs isotope (0.661 MeV) and the protective shield material, one can easily determine its thickness by the reference book [5]: when using lead, it will be 18.5 cm, and concrete – 66.1 cm.

The thickness of the additional protection of population is calculated in a similar way. Population should not be exposed to occupational irradiation at the rate above 0.11 μSv/hr, which is derived from the annual dose limit (1 mSv) and the admitted exposure duration (8.800 hr). To avoid exceeding the irradiation dose of population from the same IRS with ^{137}Cs isotope of 67 GBq activity, creating the equivalent dose rate of 22.3 mSv/hr at a distance of 0.5 m from it, the additional protection either from lead (8.8 cm) or from concrete (29.7 cm) is needed.

However, in the case of using a dosimetric device that measures not equivalent, but exposure dose rate, it is necessary to perform a number of transformations, which cannot be limited by the assumption of equality of 1 sievert (Sv) to 100 roentgen (R).

For example, if the exposure dose rate (P_X) of gamma radiation from ^{137}Cs source without a protective container, measured at a distance of 1 meter (100 cm) or calculated by the formula (2), is 200 mR/hr, then one needs to take into account that under the conditions of electron equilibrium, when the total energies of electrons leaving and entering a certain volume are equal, the following relationship is established between the units of measurement of the exposure and absorbed dose in the air:

$$1R = 0.0087\,\mathrm{Gy}, \tag{9}$$

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 mainly due to the presence of nitrogen, carbon and oxygen, whereas in biological tissue there is hydrogen (10.1% by weight), characterized by twice the absorption capacity compared with other elements, therefore, in biological tissue, an exposure dose of 1 R corresponds the absorbed dose of 0.0095 Gy.

Since for X-ray, beta- and gamma-radiation the equivalent dose is equal to the absorbed dose, 1 R/hr can be taken equal to 0.0095 Sv/hr, and the equivalent dose rate (P_H) from cesium source in question at a distance of 1 m will be equal to:

$$P_H = 200 \times 0.0095 = 1.9\ \mathrm{mSv/hr}, \tag{10}$$

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

$$E = H_T \times K_T, \tag{11}$$

where H_T is the equivalent dose (Sv), K_T is the conversion factor from the equivalent dose to the effective one, which is equal to:

– 0.642 due to division of 0.7 by 1.09. Approximately, to calculate the effective dose from all gamma-emitting radionuclides in the environment, based on data of the absorbed dose in the air, the United Nations Scientific Committee on the Effects of Atomic Radiation recommended using a conversion factor of 0.7 [2] to account for radiation shielding by various organs and tissues and its backscattering, and indicated more accurate coefficients for converting the absorbed dose in air to the effective dose for photons of various energies. When measuring the absorbed dose of gamma radiation by air-equivalent dosimeters, it is necessary to convert the measured results into units of equivalent dose in biological tissue, which requires knowledge of the gamma-radiation spectrum. In the absence of data on the spectrum and small differences in dose rate from the background, the transition coefficient from the absorbed dose in air to the equivalent dose in tissue can be taken equal to 1.09 as a result of the ratio of the non-systemic unit equivalent of the exposure dose (1 R) in biological tissue (0.0095 Gy) to a similar indicator in the air (0.0087 Gy);

– 1.903 is the maximum value of the conversion ratio of air kerma to the equivalent dose ($H_p(6)/K$) in a plate phantom according to Table III.1a in [3];

– 1 – according to the definition of the effective dose in [1], since the sum of weighing coefficients that take into account the radio sensitivity of various organs and tissues is equal to one.

In the first case, the exposure dose rate of 200 mR/hr will be equal to the effective dose rate of 1.22 mSv/hr, in the second case – 3.62 mSv/hr, in the third case – 1.9 mSv/hr. The necessary multiplicity of attenuation of these dose rates will be 104, 308 and 162, and the thickness of lead shields are 12.2 cm, 14.6 cm and 13 cm, respectively.

Results and discussion

Thus, 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 calculated by the formula (6) 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 [9]. 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 [3] should be almost twice the equivalent, it is also reflected in Kazakhstan's sanitary rules [9], 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 [1]) to the shortened working time (1.700 hr [9]). 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.

Conclusion

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 [9] 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 (6).

Radiation safety hygienic standards [1], 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 [10].

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