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EFFICIENCY OF THE DOSE RATE CALCULATION BY MONTE-CARLO METHOD AND POINT KERNEL METHOD WHEN HANDLING RADIOACTIVE WASTE

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For the gamma-radiation sources with the volume up to 1 m^3, filled with typical radioactive waste generated at NPPs with WWER-1000 reactors, the results of the dose rate (DR) calculations made by Monte-Carlo (MCNP) and by point kernel method (MicroShield and VOLUME) are compared. It is shown that the difference of the DR calculations made by the above methods does not exceed 10%. The values of DR calculated in MicroShield and VOLUME packages for the shields made of concrete and steel for such sources overestimate the MCNP data by 20…50%. The optimal correction of the buildup factor in the VOLUME package gives an accuracy of 10% in the shield calculations.

INTRODUCTION

At nuclear power plant (NPP) operation, a significant amount of radioactive waste (RW) of high, medium and low specific activity is produced. Interim storage of RW produced at Ukrainian NPPs is carried out at the site areas of these plants. As long as the storage facilities are, as a rule, full up, the technologies of minimizing the volume of RW waste are applied. Solid waste is pressed; combustible one is burned and pressed. The volume of liquid RW is significantly reduced by evaporation to turn it into a solid waste – "salt melt", what ensures its more safe storage.

Compacted RW in the form of pressed solid waste and ash, as well as of salt melt is packed in standard cylindrical containers (barrels) of 170 to 280 liters. It should be noted that after reducing the volume of liquid RW which passed the evaporation stage, as well as the compression of ash, its specific activity increases significantly. Handling of such waste: reloading, transportation and storage – requires some additional measures to ensure the regulations of radiation safety. While developing the technique for handling the compacted RW the radiation field calculation is required at all stages of technological processes not to exceed the permissible limits of radiation safety standards for the NPP personnel. That is, calculation of the dose rates (DRs) produced directly by a cylindrical source (or by several cylindrical sources) is required with different versions of biological shielding taken into account.

The DR (in this paper, DR means equivalent dose rate in relative units) of the external γ-radiation is determined by isotopic and elemental composition of RW loaded into the containers, as well as by variation in the spectral composition of γ-quanta at radiation transport both in the casks with RW and in biological shielding. At present, the most reliable data on the characteristics of external radiation from radiating sources of a complex geometric shape are obtained using Monte-Carlo (MC) method. First of all, these are packages, which are widely used in atomic energy: MCNP [1] – for calculating the transport of γ-quanta, electrons and neutrons, and PENELOPE [2] – for calculating the transport of γ-quanta and electrons. However, the programs based on MC method have a significant disadvantage: they consume a lot of calculation time and resources, what limits the application of this method when carrying out a large number of calculations. For prompt evaluation of the DR produced by surface and volume radiation sources of various configuration (cubic, cylindrical, etc.), the method of volumetric integration of point sources (the point kernel method) is applied including the calculation of biological shielding. This method is based on calculation of γ-quanta in the volume sources and shielding (exponential law of absorption) with allowance for the multiple scattering (in the form of buildup factors). This method is realized in a number of software products: MicroShield package [3], which is often applied in nuclear power engineering, and VOLUME package [4], etc. But when the biological shields are rather thick, the results of calculations in these packages give overestimated DRs as compared to the data obtained in the MCNP package [5]. It is known [6] that in order to calculate correctly the DR, when the detector is situated far away from the source, the buildup factor should be reduced because of the scattered radiation absorption not only in the detector direction, but also around the source.

The objective of this work is to analyze the effect of buildup factors in the source and in the shielding on the reliability of the DR calculation by the point kernel method in comparison to MC method.

1. CALCULATION TECHNIQUE

The radiation sources are assumed to be uniformly distributed in the volume V so that each element of volume dV isotropically emits γ-quanta with energy E and density n_γ. Then the number of γ-quanta $N_\gamma(R,E)$ in the detector at the distance R beyond the shielding with thickness t is determined by the expression:

$$N_\gamma(R,E) = \frac{n_\gamma(E)}{4\pi} \times$$

$$\times\int_V \frac{B(\mu(E)x)\cdot B_{PR}(\mu_{PR}(E)y)e^{-\mu(E)x}\cdot e^{-\mu_{PR}(E)y}}{R^2}dV. \quad (1)$$

In (1), $B(\mu(E)x)$ is the buildup factor in the source material; $\mu(E)$ is the linear attenuation coefficient of γ-quanta with energy E in the source material; x is the free path of γ-quanta from the volume dV to the intersection with the source boundary; $B_{PR}(\mu_{PR}(E)y)$ is the buildup factor in the shielding; $\mu_{PR}(E)$ is the linear coefficient of γ-quanta attenuation in the shielding; y is free path of γ-quanta from the volume dV in the shielding. The buildup factors depend on the value $\mu(E)\cdot x$ which is proportional to the mean free path of γ-quanta, so that $\mu(E)\cdot x = 1$, and consequently, $\exp[-(\mu(E)\cdot x)] = 1/e$.

The major problem, that arises while formulating the relations of type (1) in specific cases, is to determine x and y values. The scheme of calculating the radiation from the self-absorbing cylindrical volume source in the radial direction beyond the flat protective barrier at the point P is shown on Fig. 1. The expression for the gamma-ray flux in the detector P placed in the base plan, so that the radius vector from the source center and the normal to the protective barrier make angle α [7], takes the form:

$$N_\gamma(b,H) = \frac{n_\gamma}{2\pi}\int_0^H dz\int_0^R \rho d\rho \int_0^\pi \frac{B(\mu(E)x)B_{PR}(\mu_{PR}(E)y)e^{-\mu(E)x}e^{-\mu_{PR}(E)y}d\varphi}{\rho^2+b^2+z^2-2b\rho\cos\varphi}. \quad (2)$$

In (2), b is the distance from the cylinder axis to the observation point P; R is the cylinder radius; H is the cylinder height,

$$x = \frac{\rho^2 - b\rho\cos\varphi + \sqrt{(\rho^2+b^2-2b\rho\cos\varphi)R^2 - \rho^2 b^2\sin^2\varphi}}{\rho^2+b^2-2b\rho\cos\varphi}\sqrt{\rho^2+b^2+z^2-2b\rho\cos\varphi}, \quad (3)$$

$$y = \frac{t\sqrt{\rho^2+b^2+z^2-2b\rho\cos\varphi}}{(b-\rho\cos\varphi)\cos\alpha - \rho\sin\varphi\sin\alpha}. \quad (4)$$

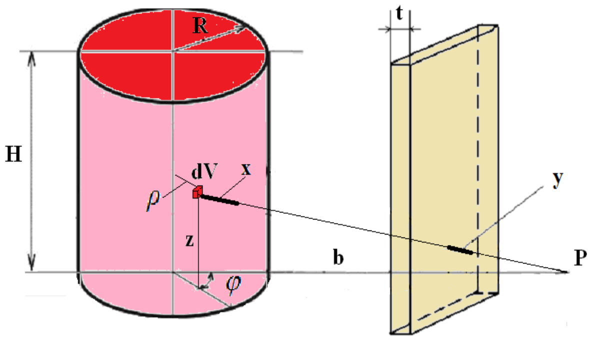

Fig. 1. Schematic of the self-absorbing cylindrical volume source with a shield

Expression (4) differs from the known relations given, for example, in [6] by introduction of angle α relative to the protective barrier. For the case when the observation point P is located at the height h_z from the source bottom, the expression for the radiation flux has the following form:

$$Nh_\gamma(b,h_z) = N_\gamma(b,h_z) + N_\gamma(b,H-h_z). \quad (5)$$

Correction of the buildup factors in the source and the shielding is defined as follows:

$$BC(\mu(E)x,K) = 1 + (B(\mu(E)x)-1)/K, \quad (6a)$$

$$BC_{PR}(\mu(E)x,K_{PR}) = 1 + (B_{PR}(\mu_{PR}(E)y)-1)/K_{PR}. \quad (6b)$$

In expressions (6), $K > 1$ and $K_{PR} > 1$; for $K = K_{PR} = 1$, the corrected values of the buildup factors are equal to the initial ones. The relations (1)–(6) are used in VOLUME package [4], which calculates γ-quanta flux in the given points, as well as DR with the dose coefficients [8] taken into account.

MCNP (Monte-Carlo N-Particle) code [1] is a multipurpose code based on MC method and designed for calculating the particles transport in a matter. This code allows processing 34 kinds of particles in a wide range of energies. For photons it makes from 1 keV to 100 GeV, for electrons – from 1 keV to 1 GeV, for neutrons – up to 150 MeV. The MCNP code is widely used to analyze shielding in the US nuclear industry. More than 400 person-years were spent to develop and test the code.

Fig. 2 presents the geometry for a self-absorbing cylindrical volume source with shielding, developed in MCNP code.

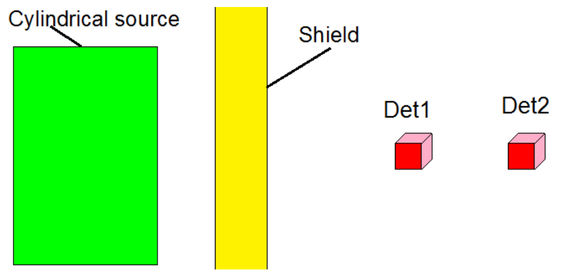

Fig. 2. Schematic of a self-absorbing cylindrical volume source with shielding in MCNP code

The DR is calculated in MCNP code as the mean DR produced by γ-quanta passing through the cube with the edge of 10 cm, located at different distances from the source surface (see Fig. 2).

MicroShield [3] is the code designed to calculate the DR for different scenarios. The MicroShield code is a complex software designed for evaluation of the photon/gamma-radiation shielding and determination of the DR from radiation sources of different shape. The MicroShield code is widely used for designing biological shields, for estimating the intensity of radiation sources, for minimizing the body burden. After developing the shape of the radiation source, its intensity and shielding at various points of the dose registration are calculated by the method of point sources. Fig. 3 presents the schematic for a self-absorbing cylindrical volume source with shielding developed in MicroShield package.

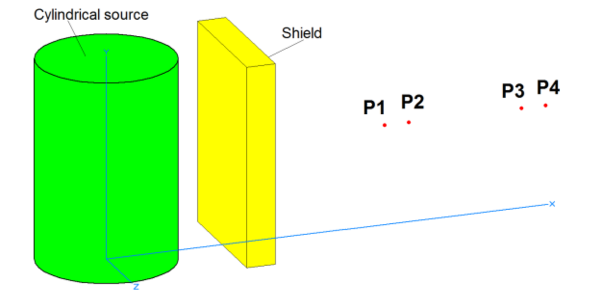

Fig. 3. Schematic of a self-absorbing cylindrical volume source with shielding in MicroShield code

In some cases, for example, when calculating the external radiation from the storage containers with spent nuclear fuel, the point kernel method is not applicable because of the large thickness of the biological shielding of the container itself [4]. When calculating the biological shielding from the set of sources in the form of storage containers, application of the combined technique is preferable. The combined technique includes both MC method and the analytical method of integration of point sources over the container surface.

This technique is realized in Serf_MC package [5]. This is our product, which is a modification of the package VOLUME. Monte-Carlo method is used to calculate spectral and angular distributions, as well as the density of gamma-ray flux on the surface of the cylindrical container loaded with spent nuclear fuel. Using the data obtained by integrating the point sources over the container surface, the DR produced by γ-quanta at an arbitrary distance is calculated with biological shielding taken into account.

The radiation flux from the cylindrical surface source in the radial direction at the point *P* beyond the biological shielding (Fig. 4) can be written as:

$$N_\gamma(b,Z,E)/dE = 2\cdot I_E(E)\cdot\int_0^H\left[\int_0^{\pi/2-\Psi}\frac{I_\phi(\phi)\cdot n_\gamma(z,\varphi)\cdot BC_{PR}(\mu_{PR}(E)y)\cdot e^{-\mu_{PR}(E)y}\,d\varphi}{R^2+b^2+(z-Z)^2-2bR\cos\varphi}\right]dz\,. \qquad (7)$$

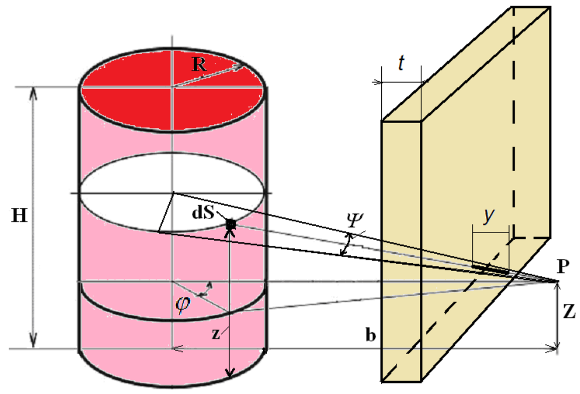

Fig. 4. Schematic of the cylindrical surface source with shielding

In (7), *b* is the distance from the cylinder axis to the observation point *P* in the radial direction; Z is the coordinate of the point *P*; *R* is the cylinder radius; *z* and φ are the coordinates of the surface element *dS*; $\Psi(b,z,Z)=\arccos\left[R/\sqrt{(z-Z)^2+b^2}\right]$ is the flat angle, under which one sees the lateral surface of the cylinder from the point *P*, the angle being formed by a straight line directed to the center of the cylinder bottom and a tangent to the lateral surface of the cylinder; $\varphi(\phi,b,z,Z)=\pi-\arccos\times$ $\times\left((R-b\cos\phi)/\sqrt{R^2+b^2+(z-Z)^2-2bR\cos\phi}\right)$ is the angle between the normal to the cylinder surface and the straight line from the element *dS* to the observation point *P*; *H* is the cylinder height; $I_E(E)$ is spectral, and $I_{AN}(\phi)$ is angular distributions relative to the normal to the cylinder surface; $n_\gamma(z,\varphi)$ is the density of γ-quanta on the container surface. The mean free path of γ-quanta from the surface element *dS* in the shielding is determined by the expression (4), and the corrected buildup factor in the shielding – by the expression (6b).

Let us consider the variation in the dose characteristics of the external radiation produced by RW sources containing the most active isotopes formed during the WWER-1000 reactors operation. After compacting RW into salt melt, compressed ash and solid radioactive waste, more than 95% of the external radiation is determined by the following isotopes: ^{54}Mn, ^{60}Co, ^{110m}Ag, ^{134}Cs, ^{137}Cs. Therefore, the effect of the three major isotopes: ^{137}Cs, ^{54}Mn, and ^{60}Co with the most intensive energy lines of γ-quanta is studied. We put these isotopes in the order of the energy lines increase: 0.661, 0.834 and 1.17, 1.33 MeV. The major energy lines for ^{110m}Ag and ^{134}Cs lie in the same range.

2. NUMERICAL RESULTS

The following characteristics of the external radiation of a cylindrical source are determined using MCNP package: spectral composition and angular distribution. As an elemental composition for the most commonly used source with a volume of 200 liters we choose a "concrete": this means worldwide Portland concrete compound, with material composition according to database of the National Institute of Standards and Technology, USA [9] and a density in the range from 0.5 to 3 g/cm^3.

The spectral composition of photons on the surface of the cylindrical source induced by uniformly distributed isotopes ^{54}Mn, ^{60}Co, and ^{137}Cs for the filling density of 1 g/cm^3 is shown in Fig. 5. Some of γ-quanta lose their energy not only because of absorption, but also due to Compton scattering. The portion of such photons grows with increase of RW density inside the radiation source. Table shows the ratio of the number of photons with initial energy to all the γ-quanta emitting from the object as a function of the source density for the isotopes: ^{54}Mn, ^{60}Co, and ^{137}Cs.

The ratio of the number of photons with the initial energy to all the γ-quanta emitting from the object (Spectrum) and DR as a function of the source density

Density, g/cm^3	^{137}Cs		^{54}Mn		^{60}Co	
	Spectrum, %	DR, %	Spectrum, %	DR, %	Spectrum, %	DR, %
0.5	72.9	64.6	73.8	67.9	75.2	73.7
1	57.4	54.7	59.6	57.9	63.1	63.9
2	35.7	49.9	37.9	52.6	41.8	58.0
3	25.2	48.6	27.0	51.2	30.2	56.4

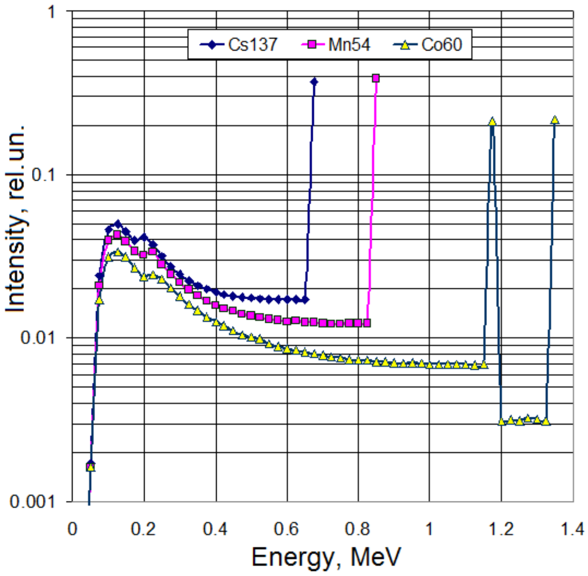

Fig. 5. Spectral composition of photons on the surface of the cylindrical source induced by isotopes ^{54}Mn, ^{60}Co, and ^{137}Cs uniformly distributed in the volume

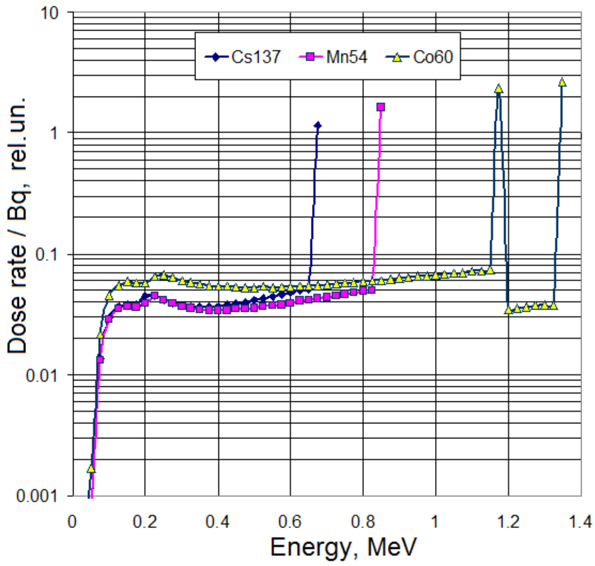

Fig. 6. DR of photons on the surface of the cylindrical source induced by isotopes ^{54}Mn, ^{60}Co, and ^{137}Cs uniformly distributed in the volume

When the density increases from 0.5 to 3.0 g/cm^3, the number of initial γ-quanta decreases almost threefold, and the DR produced by the initial photons decreases by ~ 1.3 times, since γ-quanta with lower energy contribute less to the DR.

The calculations show that unlike the energy spectra, the angular distributions of the external radiation for different isotopes differ insignificantly. When the cylindrical container is loaded with "concrete" with the density of 0.5…3 g/cm^3, the angular distributions of γ-quanta for the considered isotopes are described with good accuracy by the following model function**:**

$$I_{AN}(\varphi) \approx \cos^{4.8}(\varphi). \quad (8)$$

The model function describes the calculated data in the angular range of 0…60°, which includes more than 99% of the photons emitted by RW, with the accuracy of several percent. Fig. 6 presents the DR distribution as a function of the energy of gamma-rays released to the surface. The DR relative intensity is calculated per decay (1 Bq) of each isotope. Expressions (7), (8) as well as the spectral and angular distributions on the source surface calculated by MC method in Serf_MC package are used for the DR calculation with biological shielding taken into account.

MicroShield and VOLUME packages are used to calculate the DR in specified points by integrating over the source volume. To compare the results of the MC calculations of the DR (in MCNP package) with those obtained in the MicroShield and VOLUME packages, the procedure of averaging over similar detectors with the shape of a "box" was used. The DR as a function of the distance from the source surface is calculated as the average DR value at the front and back surfaces of the "boxes" at the points *P*1 and *P*2 for detector 1, at the points *P*3 and *P*4 for detector 2 (see Figs. 2 and 3). In vertical axis, the difference in DR within ± 5 cm is 0.5%, and when the detectors are moved away from the radiation source surface, the difference in DR values at the points *P*1, *P*2 and *P*3, *P*4 decreases.

2.1. DOSE RATE PRODUCED BY A CYLINDRICAL SOURCE

Fig. 7 presents the dependence of the DR on the source density at the distance of 100 cm in absence of the shielding between the source and the detector. These results are obtained using three techniques (MCNP, MicroShield and VOLUME) for calculation of the DRs produced by ^{137}Cs and ^{60}Co isotopes. The accuracy of the calculation data obtained in the MCNP package is better than 5%, and these data are the benchmark for comparison with the results obtained in MicroShield and VOLUME packages.

The DRs calculated in MicroShield package exceed the MCNP data on average by 20% for ^{137}Cs, and by more than 10% for ^{60}Co. The DRs calculated in

VOLUME package at $K = 1$ exceed the MCNP data on average by 10% for ^{137}Cs and by ≈ 5% for ^{60}Co. When the detectors are moved away to the distance of 300 cm, these differences for MicroShield and VOLUME calculations at $K = 1$ decrease by a factor of ≈ 2. An increase in the correction factor K (decrease of the buildup factor) in VOLUME package leads to the decrease of DR; when $K = 1.2$, the difference from the results obtained in MCNP package is less than 5%. A further increase in K leads to a decrease in the DR, and at $K = 2$ the DR values are less than those obtained in the MCNP package for all the isotopes (see Fig. 7).

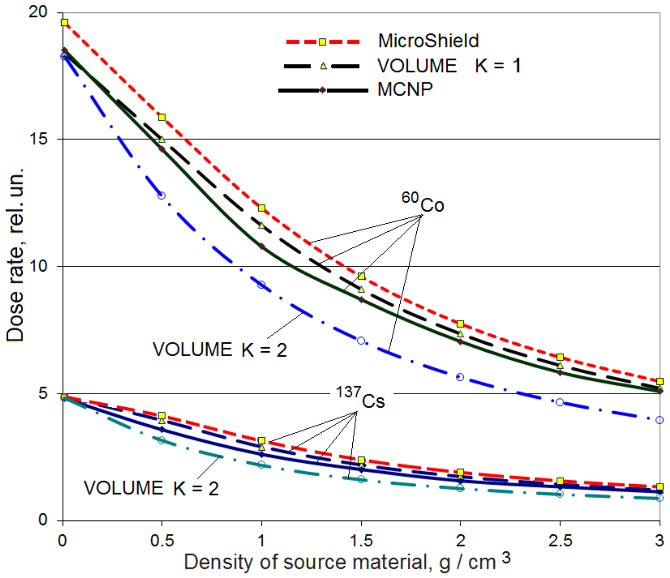

Fig. 7. Dependencies of DR caused by ^{137}Cs and ^{60}Co isotopes on the source density when shielding is absent

Similar dependencies are obtained for ^{54}Mn; the differences between the calculated DR values in VOLUME and MCNP packages are less than for ^{137}Cs and more than for ^{60}Co.

The buildup factors increase with the growth of the mean free path of γ-quanta which is determined as $\mu(E)\cdot x$. For the cylindrical sources, the same volume can be obtained with very different height and radius. We assume that the average source size, as well as the photon free path x, can be determined as the cubic root of the source volume, $R_{AV} = (V)^{1/3}$. Then the average free paths of γ-quanta $\mu(E)\cdot R_{AV}$ in the 200-liter source with the density of 0.5…3 g/cm^3 belong for ^{60}Co to the range of 1.67…10.04 cm, for ^{54}Mn – to the range of 2.04…12.26 cm, and for ^{137}Cs – to the range of 2.28…13.65 cm, while $R_{AV} = 58.5$ cm.

The characteristics of the radiation from the sources of the cylindrical shape with the volume up to 1000 liters are compared taking the same ratio $H/2R = 1.513$ as for the source with the volume of 200 liters.

The calculations show that the average free path of γ-quanta (in the source with the volume of 1000 liters and the densities of RW from 0.5 to 3 g/cm^3 and $R_{AV} = 100$ cm) belongs to the range of 2.86…17.17 cm for ^{60}Co, to the range of 3.49…20.97 cm for ^{54}Mn, and to the range of 3.89…23.34 cm for ^{137}Cs. It is shown that for the sources with a volume up to 1000 liters the characters of the DR dependencies calculated in different packages are similar to those shown in Fig. 7 for isotopes ^{137}Cs, ^{54}Mn, and ^{60}Co with the densities in the range from 0.5 to 3 g/cm^3.

Thus, the DR produced by a cylindrical container of 200 liters, filled with RW of typical isotope composition: ^{54}Mn, ^{60}Co, ^{110m}Ag, ^{134}Cs, and ^{137}Cs, which is most commonly used at NPPs, can be calculated both by MC method and by the point kernel method in the RW density range of 0.5…3 g/cm^3. The calculations show that similar conclusions should be made when calculating the DR produced by the "cubic" containers of 200…1000 liters.

2.2. DOSE RATE PRODUCED BY A CYLINDRICAL SOURCE BEYOND THE SHIELDING

The reliability of the calculations carried out by the point kernel method for the DR produced by a cylindrical container of 200 liters filled with RW is analyzed with the shielding made of concrete and iron taken into account. The DR values at distances of 100…300 cm from the source surface are calculated. The angles under which γ-quanta penetrate the shielding belong to the range ~ 0…23°, when the detectors are at the distance of 100 cm, and ~ 0…8° – at the distance of 300 cm. Fig. 8 shows the DR produced by the emission of isotopes ^{54}Mn, ^{60}Co, and ^{137}Cs (at the decay of 1 Bq) at the distance of 100 cm as a function of the thickness of the shielding made of concrete (with the density 2.3 g/cm^3). The calculation of the DR in VOLUME package minimizes the deviation $\delta_{DR} = (DR_{VOLUME} - DR_{MCNP})/DR_{MCNP}$ due to the variation of the correcting factor K_{PR} in the buildup factor of the shielding according to the expression (6). When $K_{PR} = 1$, the calculated DR values in the VOLUME and MicroShield packages are practically the same and significantly exceed DR_{MCNP} (see the data in Fig. 8).

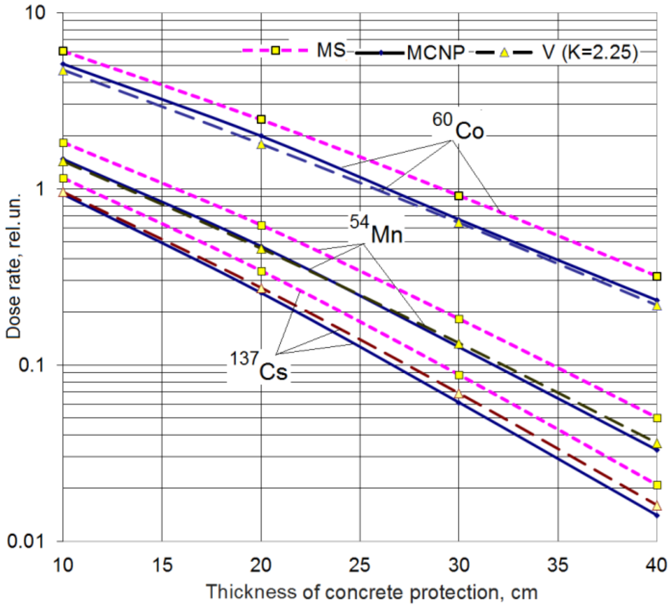

Fig. 8. Dependences of DR produced by radiation of ^{54}Mn, ^{60}Co, and ^{137}Cs isotopes on the thickness of the concrete shield

When K_{PR} = 2.25, what is close to the optimal value, the deviation δ_{DR} does not exceed 10% for ^{54}Mn and ^{60}Co at the shielding thickness of 10…40 cm; and it is a little greater: 12.8 and 14.2% for ^{137}Cs at the shielding thickness of 30 and 40 cm, respectively. Note, that DR_{MS} calculated in MicroShield package exceeds DR_{MCNP} by ~ 20% at the shielding thicknesses of 10 cm, and by ~ 50% at the shielding thickness of 40 cm. For the steel (iron) shielding arranged at the distance of 100 cm these deviations are by ~ 5…7% less. Fig. 9 presents the DR produced by radiation of isotopes ^{54}Mn, ^{60}Co, and ^{137}Cs at the distance of 300 cm at the decay of 1 Bq as a function of the thickness of the iron shielding (*x*-axis shows equivalent thicknesses of the concrete). When K_{PR} = 2.25, what is close to the optimal value, the deviation δ_{DR} does not exceed 10% for all three considered isotopes in the range of the shielding thicknesses equivalent to those of 10…40 cm thick concrete.

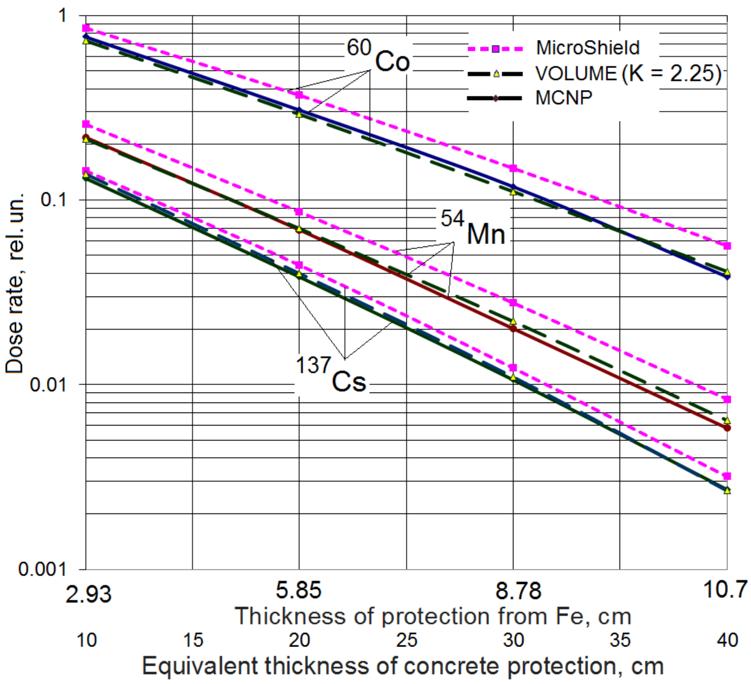

Fig. 9. Dependences of the DR produced by radiation of ^{54}Mn, ^{60}Co, and ^{137}Cs isotopes on the thickness of the iron shield

CONCLUSIONS

The results of the calculations of the dose rates produced by gamma-quanta from radioactive waste loaded into cylindrical and "cubic" sources, made by Monte-Carlo method and by the method of point sources integration, are compared. The results of the calculations obtained by Monte-Carlo method in MCNP package are used as the benchmark data.

It is shown that the DR produced by a cylindrical source of 200…1000 liters (containers of 200 liters are the most popular for storing RW at Ukrainian NPPs) filled with RW of densities from 0.5 to 3 g/cm^3 can be calculated with the accuracy of about 10% by the point kernel method realized in MicroShield and VOLUME packages.

When calculating the shielding made of concrete (steel) in the range of thicknesses up to 40 cm, the calculation results obtained in the packages MicroShield and VOLUME give overestimated values of DR (by 20…50%) as compared with the data obtained in MCNP package. The VOLUME package allows increasing the accuracy of the calculations to be better than 10% by appropriate choice of the buildup factor.

It is shown that, in order to reduce the time and necessary computing resources, costly calculations of the dose rates from sources of complex shape in MNCP can in some cases be replaced by calculations in the MicroShield and VOLUME packages without significant loss of accuracy. In this case, the MNCP package is used to calculate only single reference points.

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ЭФФЕКТИВНОСТЬ РАСЧЕТОВ ДОЗОВЫХ НАГРУЗОК МЕТОДАМИ МОНТЕ-КАРЛО И ТОЧЕЧНЫХ ИСТОЧНИКОВ ПРИ ОБРАЩЕНИИ С РАО

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Для источников гамма-излучений с объемом до 1 м^3, заполненных типичными радиоактивными отходами, образующимися на АЭС с реакторами ВВЭР-1000, проведено сравнение результатов расчетов мощности доз (МД) методами Монте-Карло (MCNP) и интегрирования точечных источников (MicroShield и VOLUME). Показано, что отличие в результатах не превышает 10% при расчетах МД методами Монте-Карло и интегрирования точечных источников. Результаты расчетов защит из бетона и стали для таких источников в пакетах MicroShield и VOLUME дают завышенные значения МД (20…50%) по сравнению с данными пакета MCNP. Оптимальная корректировка фактора накопления в пакете VOLUME дает точность расчетов защит около 10%.

ЕФЕКТИВНІСТЬ РОЗРАХУНКІВ ДОЗОВИХ НАВАНТАЖЕНЬ МЕТОДАМИ МОНТЕ-КАРЛО І ТОЧКОВИХ ДЖЕРЕЛ ПРИ ПОВОДЖЕННІ З РАО

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Для джерел гамма-випромінювань з об'ємом до 1 м^3, заповнених типовими радіоактивними відходами, що утворюються на АЕС з реакторами ВВЕР-1000, проведено порівняння результатів розрахунків потужності доз (ПД) методами Монте-Карло (MCNP) і інтегрування точкових джерел (MicroShield і VOLUME). Показано, що при розрахунках ПД методами Монте-Карло та інтегрування точкових джерел різниця в результатах не перевищує 10%. Результати розрахунків захисних споруд з бетону і сталі для таких джерел у пакетах MicroShield і VOLUME дають завищені значення ПД (20…50%) у порівнянні з даними пакета MCNP. Оптимальне коригування фактора накопичення в пакеті VOLUME дає точність розрахунків захистів приблизно 10%.