Fundamentals of Radiation Dosimetry.

LABORATORY WORK № 5

EXPERIMENTAL PART
FULL INSTRUCTION

THEORETICAL SECTION.

Ionizing radiation (ionising radiation) is radiation that carries enough energy to liberate electrons from atoms or molecules, thereby ionizing them. Ionizing radiation is made up of energetic subatomic particles, ions or atoms moving at high speeds (usually greater than 1% of the speed of light), and electromagnetic waves on the high-energy end of the electromagnetic spectrum.

Dosimetry is called the section of nuclear physics and measuring technique, which studies the values that characterize the action of ionizing radiation on substances, and methods and devices for their measurement. Initially, the development of the dosimetry was determined by the necessity of considering the action of x-rays on human.

Ionizing radiation has an effect on the substance only, when this radiation interacts with the particles, that consist the substance.

Regardless of the nature of ionizing radiation interaction can be quantitatively evaluated by the ratio of the energy ΔE, transferred by the element of irradiated matter, to the mass of this element Δm. This feature is called the **radiation dose (absorbed radiation dose)** D:

$$D = \frac{\Delta E}{\Delta m} \tag{1.1}$$

The various effects of ionizing radiation are primarily determined by the absorbed dose. It depends in a complex way on the type of ionizing radiation, the energy of its particles, the composition of the irradiated substance, and is proportional to the time of irradiation. The dose related to time is called the **dose rate N_D**. With a uniform radiation action, the dose rate is equal to the dose **D** ratio at time t, during which ionizing radiation was active:

$$N_D = \frac{D}{t} \tag{1.2}$$

The unit of absorbed radiation dose is Gray (Gy), which corresponds to the dose of radiation, at which an irradiated substance weighing 1 kg transmits ionizing radiation energy of 1 J; the radiation dose rate N_D is expressed in Gray per second or Gray per hour (Gy/s, Gy/h).

In practical dosimetry, a non-systemic unit of absorbed dose is usually used - rad (1 rad = 10^{-2} Gy=100 erg/g).

It would seem, that to find the absorbed dose of radiation, it is necessary to measure the energy of ionizing radiation falling on the body E_1, the energy passed through the body E_2, and their difference $\Delta E = E_1 - E_2$ divided by the mass Δm of the body. However, in practice this is difficult to do, since the body is not homogeneous, energy is dissipated by the body along all possible directions, etc. Thus, a very concrete and clear concept of the "radiation dose" turns out to be of little use in the experiment.

But it is possible to estimate the body absorbed dose by the ionizing effect of radiation in the air, surrounding the body.

In connection with this, another notion of the dose for X-ray and gamma radiation is introduced: **the exposure dose of radiation X**, which is a measure of the ionization of air by X-rays and γ-rays.

For the exposure dose unit, a Coulomb per kilogram (C/kg) was taken. In practice, use a unit called the roentgen (R). It is the exposure dose of x-ray or γ-radiation at which as a result of complete ionization in 1 cm^3 of dry air (0,001293 g) at 0 °C and 760 mm Hg – $2{,}08\cdot10^9$ ion pairs are formed. 1 P = $2.58\cdot10^{-4}$ C/kg.

The unit of exposure dose rate:

$$N_X = \frac{X}{t} \tag{1.3}$$

is 1 C/(sec·kg)=1 A/kg, and the non-system unit is 1 R/s or 1 R/hour.

Since the dose of radiation D is proportional to the incident ionizing radiation, there should be a proportional relationship between it and the exposure dose X:

$$D = f \cdot X \tag{1.4}$$

where ***f*** is a certain transition coefficient, depending on a number of reasons, and first of all on the irradiated substance and photon energy.

It is most simple to set the value of the coefficient ***f***, if the irradiated substance is air. At ***X***=1 R in 0,001293 g of air, $2.08\cdot10^9$ ion pairs are formed.

Therefore, in 1 g of air, there are $2.08\cdot10^9$/0,001293 ion pairs. On average, an energy of 34 eV is expended on the formation of one pair of ions. This means, that in 1 g of air, the radiation energy absorbed is equal to:

$$\frac{2{,}08\cdot10^9}{0{,}001293}\cdot34\cdot1{,}6\cdot10^{-19}\left[\frac{J}{g}\right]=88\cdot10^{-4}\left[\frac{J}{kg}\right]$$

So, the absorbed dose of $88\cdot10^{-4}$ [J/kg]=0,88 [rad] in air is energy equivalent to 1 P. Then by the formula (1.4) we have:

$$D = 0.88\cdot X,\ f = 0.88$$

where X in [Roentgen], D in [rad].

For water and soft tissue of the human body ***f***=1; therefore, the dose of radiation in [rad] is numerically equal to the corresponding exposure dose in [R]. This makes the convenience of using non-SI units - [rad] and [R].

For bone tissue, the coefficient ***f*** decreases with an increase in the photon energy from about 4.5 to 1 (see table 1.1).

Table 1.1

D=*f*·X	
Substance	f, [rad/R]
Air under normal conditions	0,88
Water and soft tissues	≈1
Bone tissue (the value of ***f*** increases with increasing wavelength)	1 ÷ 4,5

Quantitative rating of the biological effect ionizing radiation. The equivalent dose.

For a given type of radiation, the biological effect is usually the greater with increasing the dose of radiation. However, different radiations, even with the same absorbed dose, have different effects.

In dosimetry, it is common to compare the biological effects of different radiations with the corresponding effects caused by x-ray and γ-radiation.

The coefficient ***k***, showing how many times the effectiveness of the biological action of a given type of radiation is greater than the x-ray or γ-radiation, with the same radiation dose in tissues, is called the **quality coefficient**. In radio biology, it is also called r**elative biological efficiency (RBE)**.

The coefficient of quality is established on the basis of experimental data. It depends not only on the kind of particle, but also on its energy. We give approximate values of ***k*** (Table 1.2) for some radiations.

Table 1.2

Kind of radiation	***k***
X-ray, γ and β-radiation	1
Thermal Neutrons (~0,01 eV)	3
Protons	5
α-particles	20

The absorbed dose ***D*** together with the quality factor ***k*** gives an idea of the biological effect of ionizing radiation, therefore the product ***D·k*** is used as a single measure of this action and is called the equivalent dose of radiation H:

$$H = D \cdot k \qquad (1.5)$$

If the body is exposed to several types of radiation, then their equivalent doses ($\boldsymbol{H}_i$) are summarized:

$$H = \sum_i H_i = \sum_i k_i \cdot D_i \qquad (1.6)$$

where $\boldsymbol{D}_i$ is the absorbed dose of radiation type i, $\boldsymbol{k_i}$ is quality factor for for a given type of radiation.

Since $\boldsymbol{k}$ is a dimensionless coefficient, the equivalent dose of radiation $\boldsymbol{H}$ has the same dimension as the absorbed radiation dose $\boldsymbol{D}$, but is called a Sievert (Sv). The non-system unit of the equivalent dose is rem (roentgen equivalent man),

$$1\ [\text{rem}] = 10^{-2}\ [\text{Sv}].$$

The equivalent dose in [rem] is equal to the radiation dose in [rad], multiplied by the quality factor $\boldsymbol{k}$.

Biological effects of radiation doses. Effective dose.

Natural radioactive sources (cosmic rays, radioactivity of subsoil, water, radioactivity of nuclei that make up the human body, etc.) create a background corresponding to an approximately equivalent dose of 125 mrem during the year. An officially acceptable equivalent dose for professional work is 5 rem per year. The minimum lethal dose from y-radiation is about 600 rem. These data correspond to the irradiation of the whole organism.

The biological effect of radiation with a different equivalent dose is shown in Table 1.3.

Table 1.3. Biological effect of single doses.

Equivalent dose $\boldsymbol{H}$, [rem]	Biological effect
5 – 10	Registration of individual mutations
10 – 25	For an adult there are no visible violations. Damage to the brain is possible for the embryo.
25 – 50	Temporary male sterilization, blood changes are possible
50 – 100	Changes in blood, immunity disorders
100 – 200	Immunodeficiency
200 – 400	Loss of capacity for work, disability
400 – 500	Severe bone marrow injury, 50% mortality
600 – 1000	Severe damage to the intestinal mucosa, death within 3 — 12 days
1000 – 10000	Coma, death within 1 to 2 hours
H>10000	Immediate death, death under a ray

With the general single irradiation of the body, different organs and tissues have different sensitivity to the action of radiation. Thus, at the same equivalent dose ***H***, the risk of genetic damage is most likely when irradiating reproductive organs. The risk of lung cancer when exposed to radon α-radiation under equal irradiation conditions is higher than the risk of skin cancer, etc. Therefore, it is clear that the radiation doses of individual elements of living systems should be calculated taking into account their radiosensitivity. For this, we use the weighting coefficients b_T (T is the index of the organ or tissue) given in Table. 1.4.

Table 1.4

Organs and tissues	b_T
Gonads	0,20
Red marrow	0,12
Small intestine	0,12
Lungs	0,12
Liver	0,05
Esophagus	0,05
Thyroid	0,05
Skin	0,01
Stomach	0,12
Bladder	0,05
Breast	0,05
Bone marrow cells	0,01
Other	0,05

The effective dose (***E***) is a value used as a measure of the risk of the long-term effects of irradiation of the entire human body, taking into account the radiosensitivity of its individual organs and tissues. Effective doses, like equivalent doses, are measured in [rem] and Sievert [Sv].

The effective dose ***E*** is equal to the sum of products of equivalent doses in organs and tissues to the corresponding weight coefficients:

$$E = \sum_t b_t \cdot H_t \qquad (1.7)$$

To obtain an effective dose, the calculated absorbed organ dose $\boldsymbol{D_t}$ is first corrected for the radiation type using factor ***k*** (see 1.5 formula) to give a weighted average of the equivalent dose quantity $\boldsymbol{H_t}$, received in irradiated body tissues, and the result is further corrected for the tissues or organs being irradiated using factor b_T, to produce the effective dose quantity ***E***.

The sum of effective doses to all organs and tissues of the body represents the effective dose for the whole body. If only part of the body is irradiated, then only those regions are used to calculate the effective dose. The tissue weighting factors summate to 1,0 ($\sum_t b_t = 1{,}0$), so that if an entire body is radiated with uniformly penetrating external radiation, the effective dose for the entire body is equal to the equivalent dose for the entire body.

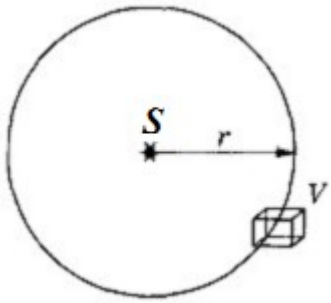

Figure 1.1

We establish the relationship between the activity of a radioactive preparation — a source of γ-photons — and the power of the exposure dose ***X***. From source ***S*** (Figure 1.1), γ-photons fly out in all directions. The number of these photons permeating 1 m^2 of the surface of a sphere in 1 sec is proportional to the activity of ***A*** and inversely proportional to the surface area of the sphere ($4\pi r^2$). The exposure dose rate ($\frac{X}{t}$) in the volume ***V*** depends on this number of photons, since they cause ionization. So, we have:

$$\frac{X}{t} = k_\gamma \cdot \frac{A}{r^2} \qquad (1.8)$$

where k_γ is the gamma constant, that is characteristic of a given radionuclide. Units of k_γ in (1.8) is $\left[\frac{\mathrm{R}\cdot\mathrm{cm}^2}{\mathrm{mCi}\cdot\mathrm{hour}}\right]$, 1 mCi=10^{-3} Ci; activity ***A*** in [mCi]; ***r*** in [cm]. In this case, unit of exposure dose rate $N_X = \frac{X}{t}$ is: [R/hour].

The activity ***A*** of the substance is equal to 1 Ci (Curie), if dN/dt=3.7x10^{10} radioactive decays occur in it every second. In this way:

$$1\ \mathrm{Ci} = 3.7\cdot 10^{10}\ \mathrm{Bq}\ \text{(exactly)}$$
$$1\ \mathrm{Bq} \approx 2.7027\cdot 10^{-11}\ \mathrm{Ci}.$$

The *curie* (symbol Ci) is a non-SI unit of radioactivity originally defined in 1910. According to a notice in Nature at the time, it was named in honour of Pierre *Curie*, but was considered at least by some to be in honour of Marie *Curie* as well.

Since the 1 Curie unit is large, smaller units are often used: millicuries (10^{-3} curies) and microcuries (10^{-6} curies).

Becquerel (Bq) is a unit of the activity of a radioactive source in the International System of Units (SI). One Becquerel is defined as the activity of a source in which, in one second, one radioactive decay takes place on average:

$$1\ \mathrm{Bq} = 1\ \text{decay/sec}$$

The law of inverse squares is generally applicable, when the energy diverges (propagates) in the radial direction from the source (fig. 1.2). As the area of sphere (which is determined by the formula $4\pi r^2$ grows in proportion to the square of the distance from the source (the radius of the sphere ***r***), and as the emitted radiation moves farther from the source, this radiation must pass through a surface, whose area grows in proportion to the square of the distance from the source Consequently, the intensity of radiation, passing through the same area, is inversely proportional to the square of the distance from the source.

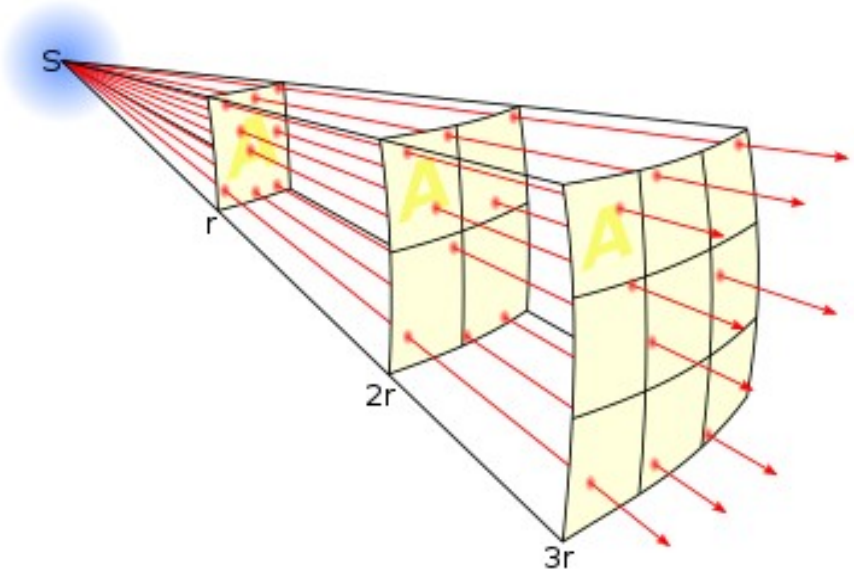

Figure 1.2. Lines mean the flux, emanating from the source. The total number of flux lines depends on the power of the source and remains unchanged with increasing distance from it. A higher density of lines (the number of lines per unit area) means a stronger field. The density of the flux lines is inversely proportional to the square of the distance from the source, since the surface area of the sphere grows in proportion to the square of the radius. Thus, the field strength is inversely proportional to the square of the distance from the source.

Recall, that the **dose rate (N)** is a quantity, that determines the dose, received by the object per unit time.

With a uniform radiation action, the dose rate is equal to the dose ratio at time t during which ionizing radiation was active, see 1.2, 1.3, 1.8 formulas.

Dosimetric instruments.

Dosimetric devices, or dosimeters, are devices for measuring the doses of ionizing radiation or dose-related quantities (see fig. 2.1). Constructively, the dosimeters consist of a nuclear radiation detector and a measuring device. They are usually graduated in units of dose or dose rate. In some cases, an alarm is provided for exceeding the set dose rate.

Figure 2.1. Some types of dosimeters.

Depending on the detector used, there are dosimeters ionization, luminescent, semiconductor, photodosimeter, etc. The dosimeters can be designed to measure the doses of any particular type of radiation or to record mixed radiation.

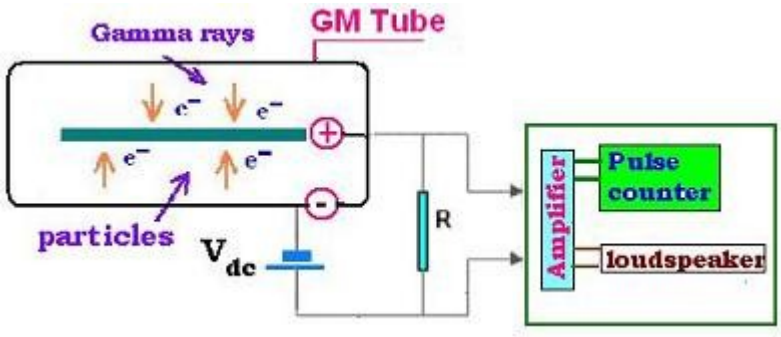

Figure 2.2. Typical structural diagram of the dosimeters

There are dosimeters, whose detectors are gas-discharge counters.

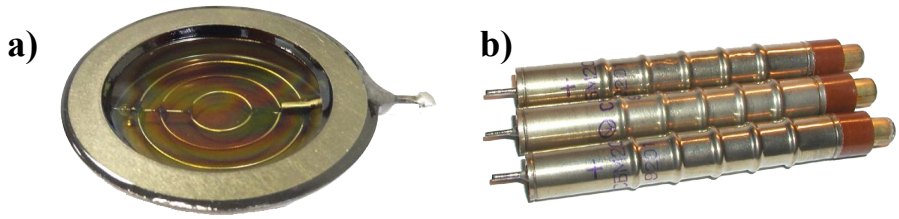

Figure 2.3. Some Types of Detectors. a) – Pancake G-M tube used for alpha and beta detection; the delicate mica window is usually protected by a mesh when fitted in an instrument; b) – The "classical" Geiger tube filled with a mix of inert gases

In conclusion, we note, that the general structural diagram of all dosimeters is similar to that shown in Fig. 2.2. The role of the sensor (measuring transducer) is performed by the nuclear radiation detector (fig. 2.3). As the output devices can be used point devices, recorders, electromechanical counters, sound and light signaling devices, etc.

Protection against ionizing radiation.

Work with any sources of ionizing radiation requires the protection of personnel from their harmful effects. This is a big and special problem. Let us briefly consider some aspects of this problem. There are three types of protection: protection of time, distance and material.

Let us illustrate the first two types of protection in the model of a point source of gamma radiation. We transform the formula (1.8):

$$X = k_{\gamma} \cdot \frac{A}{r^2} \cdot t \qquad (2.1)$$

As can be seen from the formula, the longer the time and the shorter the distance, the greater the exposure dose. Consequently, it is necessary to stay under the influence of ionizing radiation for a minimum time and at the maximum possible distance from the source of this radiation.

The protection of the material is based on the different ability of substances to absorb different types of ionizing radiation (fig. 2.4).

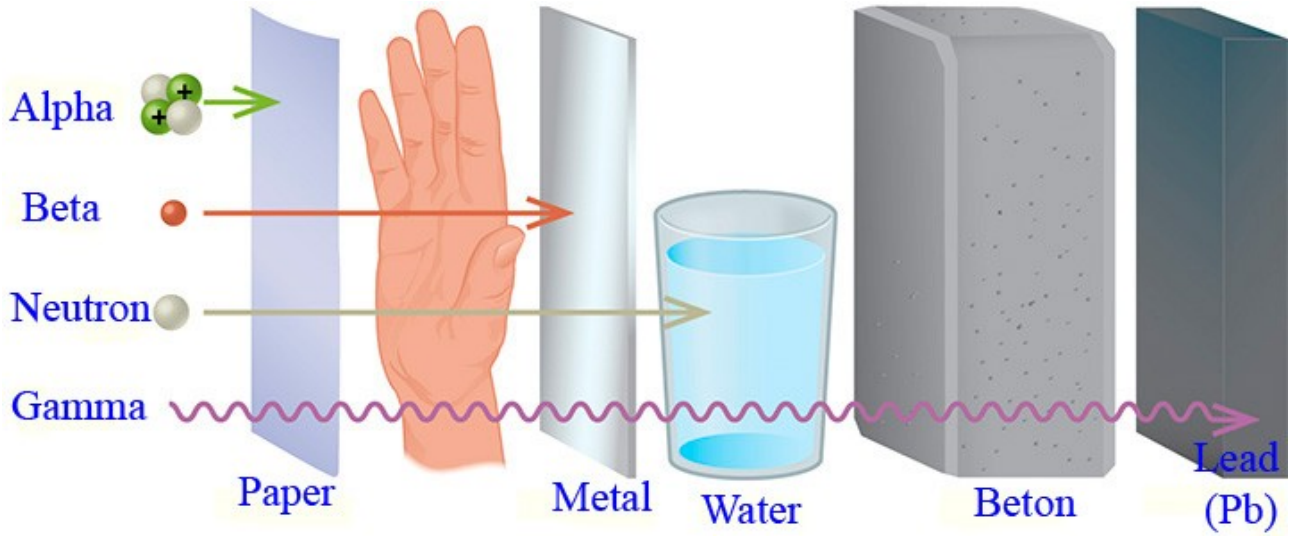

Figure 2.4. Penetrating power of various types of radiation.

Protection from α-radiation is simple: just a sheet of paper or a layer of air a few centimeters thick to fully absorb the α-particles. However, when working with radioactive sources, avoid getting α-particles into the body during breathing or eating food.

To protect β-radiation, plates made of aluminum, plexiglas or glass a few centimeters thick are sufficient. In the interaction of P-particles with matter, bremsstrahlung X-rays may appear, and from β^+ particles may be β^+ radiation, which occur when these particles are annihilated with electrons.

The most difficult protection from "neutral" radiation: X-ray and γ-radiation,

neutrons. These radiations are less likely to interact with particles of matter and therefore penetrate deeper into the substance. The attenuation of the beam of X-ray and γ-radiation approximately corresponds to the law:

$$N(x) = N_0 e^{-\mu x} \tag{2.2}$$

where N_0 is the number of particle, which flux falling on the absorber; $N(x)$ – is the number of particle, which flux passed the absorber thickness x; μ is called the linear attenuation coefficient of radiation. The linear attenuation coefficient μ has a dimension of [cm^{-1}].

The coefficient of attenuation μ depend on the ordinal number Z of the absorber element of matter and for energy of photons E_{ph} in the range of 60 - 120 keV can be approximately determined by the formula:

$$\mu_m = k \cdot \lambda^3 \cdot Z^3, \quad \mu_m = \frac{\mu}{\rho} \tag{2.3}$$

where λ - is radiation wavelength; k – coefficient of proportionality; μ_m - mass attenuation coefficient [cm^2/g]; ρ – is density of absorber [g/cm^3].

The absorption of X-rays is almost independent of the compound in which the atom is present in the substance, and therefore it is easy to compare the mass coefficients of weakening μ_{m_bone} of bone $Ca_3(PO_4)_2$ and $\mu_{m\_soft\_tissie}$ soft tissue or water H_20 by the formula (2.3). Atomic numbers Ca, P, O, and H are respectively 20, 15, 8, and 1. Substituting these numbers in (2.3), we obtain

$$\frac{\mu_{\text{m bone}}}{\mu_{\text{m soft tissue}}} = \frac{3 \cdot 20^3 + 2 \cdot 15^3 + 8 \cdot 8^3}{2 \cdot 1^3 + 8^3} = 68.$$

A significant difference in the absorption of X-rays by different tissues allows us to see in the shadow projection the images of the internal organs of the human body.

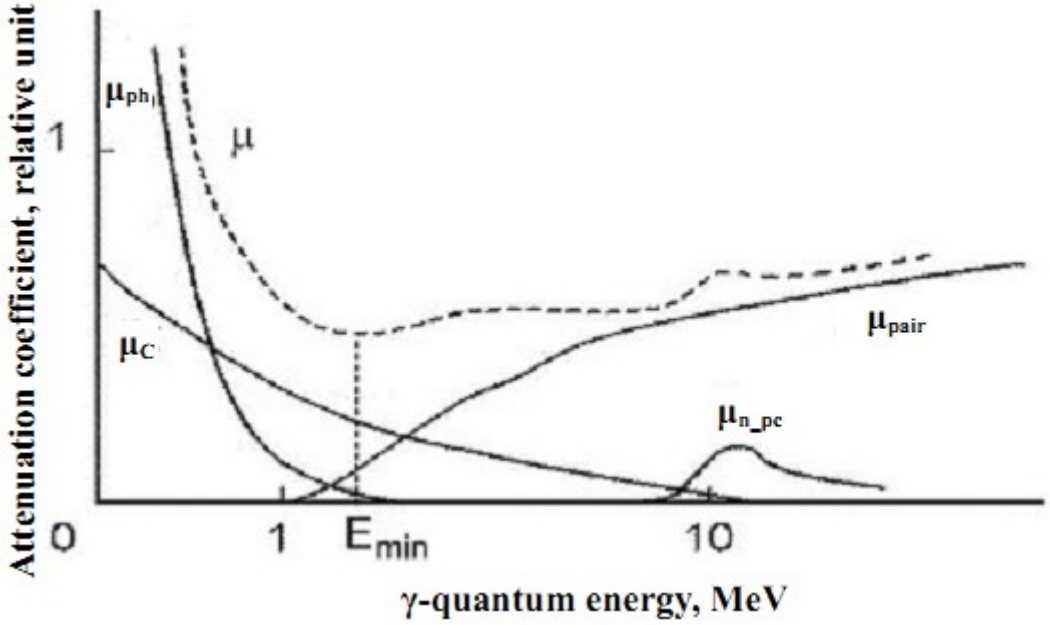

Fig. 2.5. The dependence on the energy of the full coefficient attenuation μ and the contribution of its partial components when the interaction of gamma radiation with matter. where μ_{ph}, μ_C, μ_{pair}, μ_{n_pc} is the mass attenuation coefficients, accordingly, due the photoelectric effect, Compton scattering, electron-positron pairs formation and nuclear photo effect.

Also **μ** depend on the energy of the γ-photons (see Figure 2.5).

When calculating protection, these dependencies, photon scattering, and secondary processes are taken into account. Protection from neutrons is most difficult. Fast neutrons first slow down, reducing their velocity in hydrogen-containing substances. Then other substances, for example cadmium, absorb slow neutrons.

Respirators are used for individual protection of respiratory organs from radioactive dust.

In emergency situations involving nuclear accidents, you can take advantage of the protective properties of residential buildings. So, in the cellars of wooden houses, the dose of external irradiation decreases 2 – 7 times, and in the cellars of stone houses – 40 – 100 times (Figure 2.6).

When radioactive pollution of the area is monitored the activity of one square kilometer, and when food products are polluted, their specific activity is monitored. As an example, if the area is infected more than 40 Ci/km^2, the population is fully resettled. Milk with specific activity 100 Bq/Liter and higher is not subject to consumption.

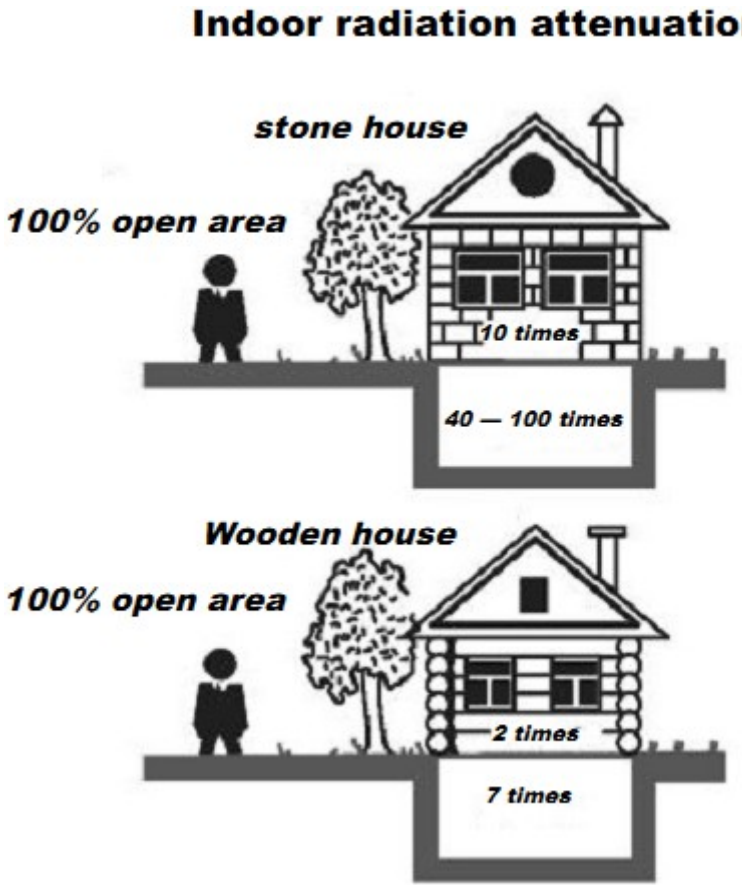

Figure 2.6. the protective properties of residential buildings.

EXPERIMENTAL PART.
Hardware. The instruments and equipment.

Laboratory work is performed on the training complex, having a mate with a PC. All experiment parameters set and measured parameter values are displayed in the program window. A block diagram of the device is shown in Fig. 3.1.

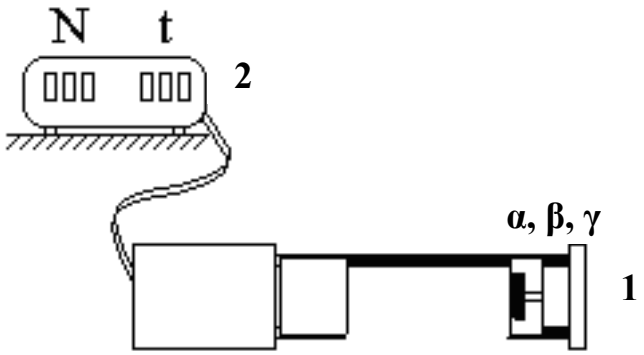

Fig. 3.1.

The device consists of two units: a detection unit (1) and a control unit (2). The electronic unit (1) (Fig. 3.1) consists of a Geiger exchangeable counters (see fig. 2.3).

To investigate the dependence of radiation dose on distance, a dosimetric ruler and a set of replaceable detectors can be used.

The training dosimeter is a combined device that allows carrying out various experiments both in the field of dosimetry and in nuclear physics. The device can be used both for counting the number of particles, and directly for dosimetry. The device includes replaceable detectors, bracket and dosimetric ruler.

The device can operate in standalone mode from the built-in battery or from the 220 V Power Line through the power supply unit. The power supply of the device is realized from a power source with a voltage of 9 V and a current of 2 A.

Programs for use.

For the working device with a personal computer need to use a special software package (see fig. 4.1), named «LabVisual».

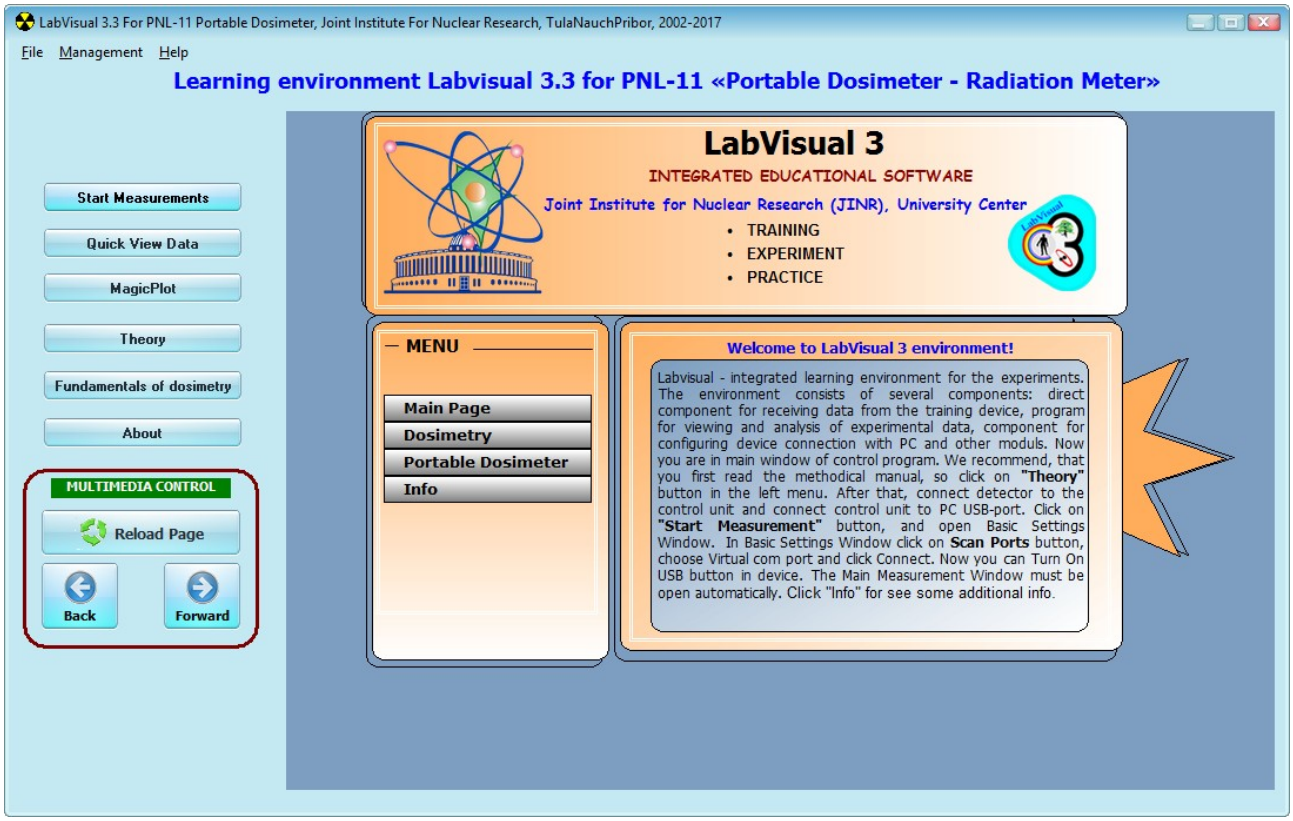

Figure 4.1. Software Package «LabVisual 3» for Dosimetry.

For work in PC-MODE, connect device to USB port of PC. The device work in virtual com port mode. You need to install the necessary drivers to operate with device (lowcdc driver), according to the instructions for installing the drivers for you used operating system. Supports 32 and 64 bit versions Windows XP - Windows 10.

If the installation was successful, the system will display the virtual com port, which later must be selected in the control program LabVisual.

Connect one of detector to the Device in "OUT to Detector" socket. Turn ON Power Switch button. If initialization mode is active you must press and hold USB/Del.Files Button, until initialization is finished. See LCD Screen on device (PNL-11M initialization message). This option can be turn off from LabVisual Program in Advanced Settings Window (before it, connect device to PC). **When initialization is finished, device USB transmitter is turned off, device automatically in manual mode!**

Start LabVisual Program. Now you in Main Window (fig. 4.1). Press "Start Measurement" button. Now you in basic settings window (fig. 4.2). Press "Scan Available Ports", choose need port from menu Input Com Port. You can open Device Manager Window for viewing Virtual Com Port Number. Press CONNECT button in LabVisual Program. You must see label in program: "Connection Status: CONNECTED". **Now you can press USB button in Device for Turn ON Usb transmitter in device.** Window measurements must be open automatically (see fig. 4.3). Also, You can make initialization by click "INITIALIZE DEVICE" button in Main Program Window (connect to device first, choose Virtual Com Port Number and after press CONNECT button).

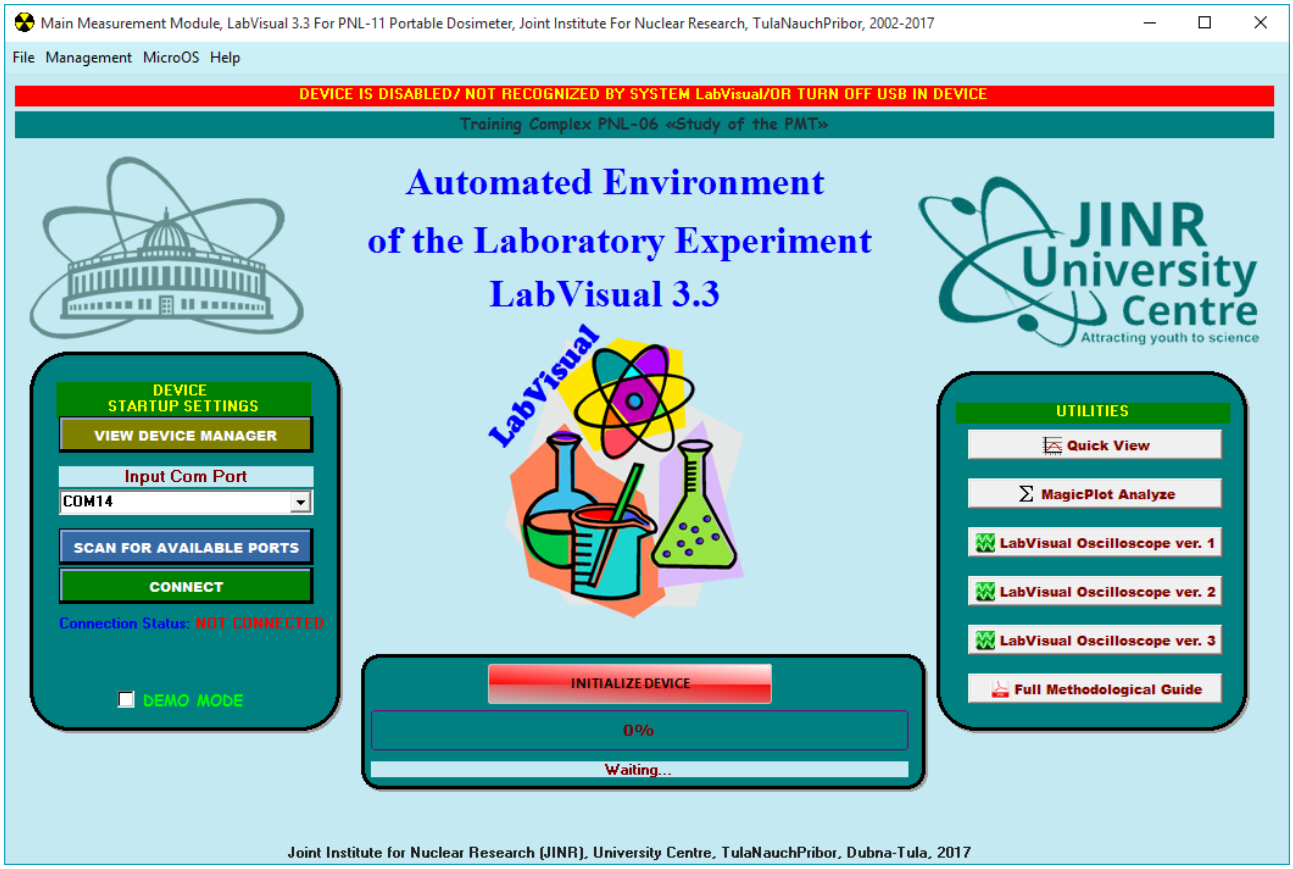

Figure 4.2. Basic settings window Software Package «LabVisual 3» for Dosimetry.

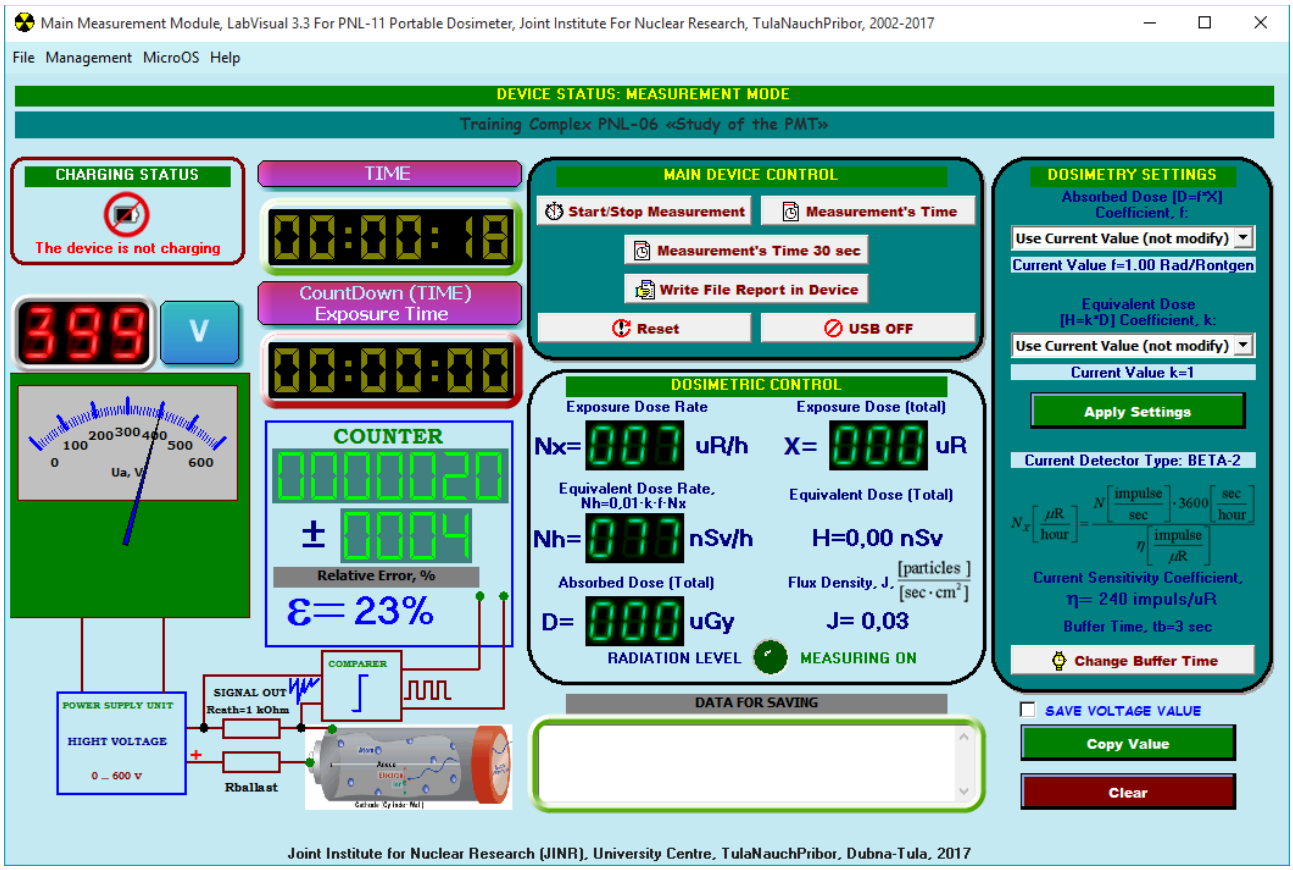

Figure 4.3. Window measurements Software Package «LabVisual 3» for Dosimetry.

More detailed information about the device can be found in the passport for it device.

Execution Order.

Task 1. Verification of the inverse square law (1.8) and estimation of the activity of the source, using PNL-11 dosimeter.

1. **Before the experiment is extremely recommended to become familiar with the software and processing data described in supplement for PNL-11 device and in LabWork 1 — 4.**
2. Before work, make sure, that the control unit «Turn OFF» from power line (~220 V).
3. Connect detector (1) to the control unit (2) see fig. 3.1 (recommended use Beta-2 detector).
4. Connect Control Unit to PC, using USB-cable.
5. Plug Control Block to power line ~220 V and Turn ON it.
6. Turn On PC and start OS.
7. Start the measurement program.
8. Set an acceptable measurement time Δt (5, 10, 20, 30 sec) by “CountDown” (“SET TIME”) button (recommended Δt=30 sec).
9. Set the Geiger counter's voltage about 400 V (about the middle of the plateau). If necessary, adjust the level of the comparator.
10. Measure background (number of background particles) $\boldsymbol{N}_{bckgr}$ at least 5 times for measurement time Δt and calculate the average value of the background counts $<\boldsymbol{N}_{bckgr}>$.
11. For this, start measuring by pressing the button “START COUNTER/TIMER” without isotopes. Wait, until the measurement is complete, and write N value. Repeat it at least 5 times.
12. Set the γ-source (^{137}Cs) on a single line with the detector, so that their centers coincide. In practical applications, it is important to remember, that expressions (1.8) and (2.1) are valid for a point source (i.e., the size of the source must be much less than the distance to the source), and also in the absence absorption of radiation in a matter. In addition, there is always a natural radiation background on the Earth, it will also make amendments.
13. So, set the minimum distance $\boldsymbol{r}$ between γ source and detector, at which the source can be regarded as a point ($\boldsymbol{r} \approx 5 – 6$ cm). At this distance, we can use the formula (1.8, 2.1). Because sources, used in this task, emit not only gamma rays, but alpha and beta radiation, and the detector has high sensitivity, the measurement should start with 1 plate, thickness=1 mm of Pb (lead) filtr-absorber. So, set one plate of lead (Pb) between source and detector.
14. Repeat measurements with γ-isotope, according to item 11, at least 5 times, does not changing distance $\boldsymbol{r}$ between γ-source and detector and does not changing time of measurement Δt.
15. Calculate the average value of $<\boldsymbol{N}_{\gamma}'>$ for this distance $\boldsymbol{r}$ between the source and detector. After that, calculate the average number of gamma quanta $<\boldsymbol{N}_{\gamma}>$, registered by the detector, taking into account the correction to the background:

$$<\boldsymbol{N}_{\gamma}>=<\boldsymbol{N}_{\gamma}'> – <\boldsymbol{N}_{bckgr}> \qquad (3.1)$$

16. Change the distance r between γ – source and detector on 1 cm and repeat the measurement, according to paragraphs 11, 14, 15.
17. Measure for other distances between the source and the detector, until $\langle N_\gamma' \rangle \approx \langle N_{bckgr} \rangle$, i. e. until the γ flux $\langle N_\gamma \rangle$ does not become to 0 (usually in this case $r \approx 15$ — 20 cm).
18. Fill in the table 2.1.

Table 2.1

Isotope Type=^{137}Cs

k_γ= … $\left[\frac{\text{R}\cdot\text{cm}^2}{\text{mCi}\cdot\text{hour}}\right]$

$\langle N_{bckgr} \rangle$= …

r, cm	$\frac{1}{r^2}$, cm^{-2}	$\langle N_\gamma' \rangle$	$\langle N_\gamma \rangle = \langle N_\gamma' \rangle - \langle N_{bckgr} \rangle$

Activity, A= … mCi

19. Build the dependence $N_\gamma = N_\gamma(r)$ of the number gamma quanta, registered by the detector, from the distance r between the gamma source and detector. Draw it on the chart.
20. Build the dependence $\mathbf{N}_\gamma = \mathbf{N}_\gamma(1/r^2)$ of the number gamma quanta, registered by the detector, from the inverse square of the distance $1/r^2$ between the gamma source and detector. Draw it on the new chart. For an example of the experimentally obtained dependences, see Fig. 5.1.

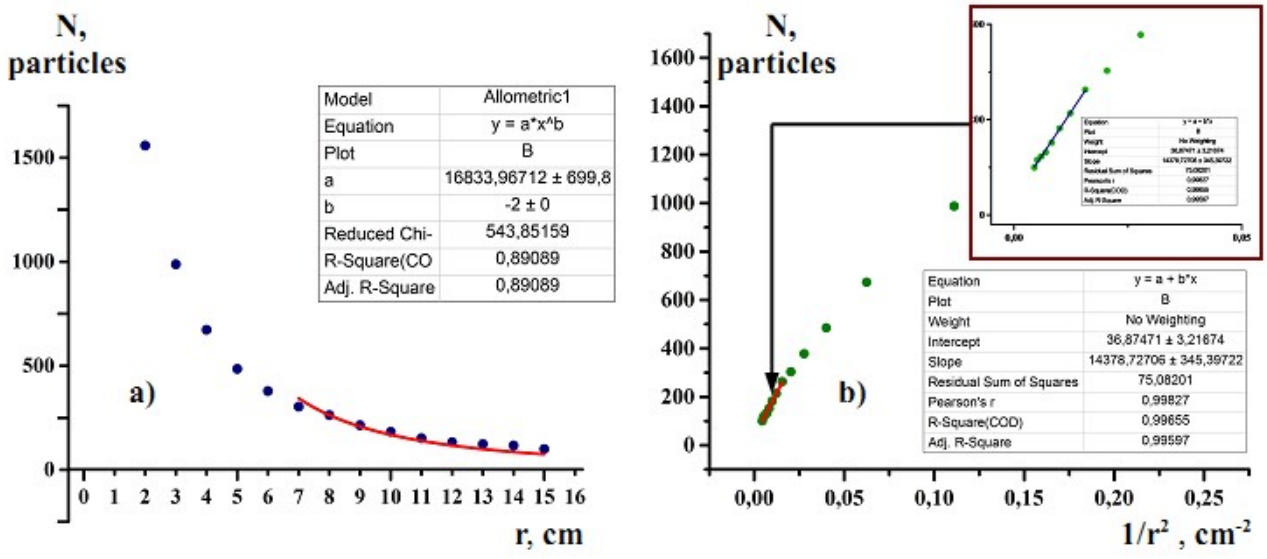

Fig. 5.1. An example of experimental dependences: a) — $N_\gamma = N_\gamma(r)$; b) — $\mathbf{N}_\gamma = \mathbf{N}_\gamma(1/r^2)$. The approximation of the experimental data is shown in the region, where the radiation source can be regarded as a point (r>6 cm, $\mathbf{1}/r^2 < 0.0278$ cm^{-2}).

21. There are two ways to analyze the resulting dependencies: a) You can analyze the dependency $\mathbf{N_\gamma = N_\gamma(r)}$, using the non-linear least squares method by formula $y(x)=a_1 \cdot x^{b1}=a_1/x^{-b1}$ (see fig. 5.1, a), where $y=N_\gamma$ — the number of registered gamma quanta minus the background; b1= -2 – fixed index; $x = r$ — distance between source and detector; a_1 – is some coefficient, in which the activity $\boldsymbol{A}$ and gamma constant $\boldsymbol{k_\gamma}$ of the source (see 1.8, 2.1 formula). b) You can analyze the linear range of the dependency $\mathbf{N_\gamma = N_\gamma}(1/\boldsymbol{r}^2)$ (see fig. 5.1, b), using the linear least squares method. In this case, the dependency will be determined by the line formula: $y(x)=a_2+b_2 x=a_2+b_2 \cdot \left(\frac{1}{r^2}\right)$, in which we can accept $a_2 \approx 0$, if the background correction is taken into account . So, $y=N_\gamma$ — the number of registered gamma quanta minus the background; $x=1/\boldsymbol{r}^2$; b_2 – is some coefficient, in which the activity $\boldsymbol{A}$ and gamma constant $\boldsymbol{k_\gamma}$ of the source. If we taken $a_2=0$, then b_2 is equal to the coefficient a_1 from paragraph (a): $b_2=a_1$.
22. A linear formula $y(x)=a_2+b_2 x=a_2+b_2 \cdot \left(\frac{1}{r^2}\right)$ can be easily fitted in an integrated environment Labvisual component “MagicPlot” or “LabVisual Quick View”.
23. So, find the linear range of the dependency $\mathbf{N_\gamma = N_\gamma}(1/\boldsymbol{r}^2)$ (see fig. 5.1, b) and fit it, using LabVisual.
24. Find $\boldsymbol{b_2}$ slope coefficient.
25. We show, how the coefficient $\boldsymbol{b_2}$ is related to the activity $\boldsymbol{A}$ and gamma constant $\boldsymbol{k_\gamma}$ of the source. Write (1.8) in

$$\frac{X}{t}=N_X=k_\gamma \cdot \frac{A}{r^2} \cdot e^{-\mu \cdot d} \qquad (3.2)$$

where $\boldsymbol{N}_X$ is exposure dose rate in [R/hour], $\boldsymbol{e^{-\mu d}}$ – factor, that takes into account the attenuation of gamma radiation in a lead (Pb) plate with d=1 mm=0,1 cm and $\mu \approx 1{,}2$ cm^{-1} for $E_\gamma=0{,}662$ MeV (gamma-ray energy of ^{137}Cs isotope). So, for ^{137}Cs isotope attenuation factor $\boldsymbol{e^{-\mu d}}=0{,}89$.

As we know from the passport of the device, exposure dose rate $\boldsymbol{N}_X$ [μR/h] is related to the counting rate N [impuls/sec] by formula:

$$N_X\left[\frac{\mu R}{hour}\right]=\frac{N\left[\frac{impuls}{sec}\right] \cdot 3600\left[\frac{sec}{hour}\right]}{\eta\left[\frac{impuls}{\mu R}\right]} \qquad (3.3)$$

where η is the sensitivity factor of detector (from the passport of detector).

Rewrite (3.3) for N_X in [R/h]:

$$N_X\left[\frac{\text{R}}{\text{hour}}\right]=\frac{N\left[\frac{\text{impuls}}{\text{sec}}\right]\cdot 3600\left[\frac{\text{sec}}{\text{hour}}\right]}{\eta\left[\frac{\text{impuls}}{\mu\text{R}}\right]}\cdot 10^{-6}\left[\frac{\text{R}}{\mu\text{R}}\right] \qquad (3.4)$$

Or:

$$N_X\left[\frac{\text{R}}{\text{hour}}\right]=\frac{N\left[\frac{\text{impuls}}{\text{sec}}\right]}{\eta\left[\frac{\text{impuls}}{\mu\text{R}}\right]}\cdot 3{,}6\cdot 10^{-3}\left[\frac{\text{sec}\cdot\text{R}}{\text{hour}\cdot\mu\text{R}}\right] \qquad (3.5)$$

By comparing (3.2) and (3.5), we get:

$$k_\gamma\cdot\frac{A}{r^2}\cdot e^{-\mu\cdot d}=\frac{N}{\eta}\cdot 3{,}6\cdot 10^{-3} \qquad (3.6)$$

Expressing from (3.6) the counting rate N [impuls/sec], we get:

$$N=\frac{277{,}8\cdot k_\gamma A\cdot\eta\cdot e^{-\mu\cdot d}}{r^2} \qquad (3.7)$$

As $N=\frac{N_\gamma}{\Delta t}$, where $\boldsymbol{N}_\gamma$ - is the number of particles, registered by the detector during measurement time Δt.
In the end, we get:

$$N_\gamma=\frac{277{,}8\cdot k_\gamma A\cdot\eta\cdot e^{-\mu\cdot d}\cdot\Delta t}{r^2}=\frac{b_2}{r^2} \qquad (3.8)$$

26. From (3.8) we obtain a calculation formula for activity $\boldsymbol{A}$:

$$A=\frac{b_2}{277{,}8\cdot k_\gamma\cdot\eta\cdot e^{-\mu\cdot d}\cdot\Delta t} \qquad (3.9)$$

where $\boldsymbol{b}_2$ – is the found experimentally slope coefficient of the linear range the dependency $\mathbf{N}_\gamma = \mathbf{N}_\gamma(1/\boldsymbol{r}^2)$ (see fig. 5.1, b); $\boldsymbol{A}$ **–** is activity in [mCi]; $\boldsymbol{k}_\gamma$ is the gamma constant of the source in $\left[\frac{\text{R}\cdot\text{cm}^2}{\text{mCi}\cdot\text{hour}}\right]$ from reference table; η – is the sensitivity factor of detector:

η=240 [impuls/μR] for BETA-2 detector
η=144 [impuls/μR] for BETA-1 detector
η=120 [impuls/μR] for SBM-20 detector (tube)

Δt is the measurement time, set in paragraph (8) (usually, Δt=30 sec);
d=1 mm=0,1 cm – lead's plate thickness; $\mu\approx 1{,}2\ \text{cm}^{-1}$ – is the linear attenuation

coefficient for E_γ=0,662 MeV for ^{137}Cs isotope; for other isotopes, μ can be determined from the graph in the annex.

27. Using formula 3.9, calculate the activity ***A*** of the ^{137}Cs sample. Formula 3.9 gives the value of activity in [mCi].
28. Compare the experimental value of the sample activity with the passport data (see annex). Note, that 1 Bq ≈ $2{,}7027 \cdot 10^{-11}$ Ci.
29. It is recommended to repeat the measurements and calculations with other isotopes and detectors.
30. At the end of work, disconnect all devices from the power line.

Execution Order.

Task 2. Calculation of doses from radiation sources and comparison with experimental values, using PNL-11 dosimeter.

1. **Before the experiment is extremely recommended to become familiar with the software and processing data described in supplement for PNL-11 device and in LabWork 1 — 4.**
2. Before work, make sure, that the control unit «Turn OFF» from power line (~220 V).
3. Connect detector (1) to the control unit (2) see fig. 3.1 (recommended use Beta-2 detector).
4. Connect Control Unit to PC, using USB-cable.
5. Plug Control Block to power line ~220 V and Turn ON it.
6. Turn On PC and start OS.
7. Start the measurement program.
8. Set the Geiger counter's voltage about 400 V (about the middle of the plateau). If necessary, adjust the level of the comparator.
9. Set an acceptable measurement time Δt by “CountDown” (“SET TIME”) button (recommended Δt=30 sec).
10. Measure background (number of background particles) N_{bckgr} at least 5 times for measurement time Δt and calculate the average value of the background counts $<N_{bckgr}>$.
11. For this, start measuring by pressing the “START/STOP MEASUREMENT” button without isotopes. Wait, until the measurement is complete, and write N_{bckgr} value (the number particles of background, registered by the detector during measurement time Δt). Repeat it at least 5 times.
12. Using formula (3.3), calculate manually the value of the background exposure dose rate $\boldsymbol{N}_{xman}$ in [μR/h]. Before it, calculate the counting rate $\boldsymbol{N}= <\boldsymbol{N}_{bckgr}>/\Delta t$ in [impuls/sec].
13. Click “START/STOP MEASUREMENT” button and measure the radiation background (exposure dose rate $\boldsymbol{N}_{xdev}$) by the device for a few (about ~2 – 3) minutes. See result on LCD screen or LabVisual Window.
14. Compare the results of calculations $\boldsymbol{N}_{xman}$ and measurements $\boldsymbol{N}_{xdev}$ values.
15. Express the exposure dose rate value $\boldsymbol{N}_x$, which you received in [μR/h] in SI units [C/(kg·sec)]=[A/kg]:

$$1\ \text{P/h} = 2.58\cdot10^{-4}\ \text{C/(kg·h)};\ 1/3600\ \text{[hour/sec]}$$

16. Calculate the absorbed radiation dose rate N_D (1.2), using (1.4) formula for air under normal conditions and water (soft tissues) in [rad/h]. Coefficient $\boldsymbol{f}$ can be taken from table 1.1.

17. Convert the absorbed radiation dose rate value N_D, which you received in [rad/h] in SI units in [Gy/sec]:

$$1 \text{ rad/h} = 10^{-2} \text{ Gy/h}$$

18. Calculate the equivalent dose rate of radiation N_H, using (1.5) formula in [rem/h].
19. Convert the equivalent dose rate of radiation N_H, which you received in [rem/h] in SI units in [Sv/sec]:

$$1 \text{ [rem/h]} = 10^{-2} \text{ [Sv/h]}.$$

20.