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Basic Ionization Dosimetry for Radiological Protection Management

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Abstract

*Management of exposure to ionizing radiation based on the three principles of radiological protection (justification, optimization, and dose limits) requires precision and accuracy of the entire dosimetry chain starting with the absorbed dose and ending with the effective dose. Approved absorbed dose limits below about 100 mSvper five years are important to prevent the induction of stochastic effects.In this sense, the paper reviews the basics of exposure to external and internal radiation using ionization dosimetry. After presenting the characteristics of ionizing radiation, radiometric quantities (Φ, $\dot{\Phi}$, Ψ,$\dot{\Psi}$) and dose constants (Γ_X, Γ_K, Γ_D), the paper presents, first, the basic dosimetry quantities (X, K, D), the factors calibration coefficients (N_X, N_K, N_D,), photon interaction coefficients (μ/ρ, μ_{tr}/ρ, μ_{ab}/ρ), electron interaction coefficients (S_{col}, S_{rad}, L_Δ).Continue with the quantities of radiological protection valid for external and internal radiation exposure (D_{TR}, H_T, E), based on the quality factor and the radiation and tissues weighting factors(Q, w_R, w_T), which in turn are based on RBE and the risk coefficients(Ra,Rr,s),to prevention of stochastic effects. Next, the quantities of protection for external exposure, operational quantities of area (H * (10), H '(d, Ω)) and individual monitoring (H_p (10), H_p (d)), quantities of radiological protection for internal exposure, in the MIRD methodology, ($\bar{D} = \tilde{A}_S S(r_S \leftarrow r_T)$) and in the ICRP methodology, (H_T (50), E (50)) based on the nuclear decay law. Derived quantities for internal exposure control (ALI, DAC), Radon exposure to air (WL, WLM) and dose limits are shown. The paper concludes with the formula for the annual effective dose,E(mSv) =H_p (10) + E (50) +E_{sub}(DAC) +E(mSv←WLM), for workers and members of the public.*

Keywords: Ionizing Radiation, Dosimetry Quantities, Protection Quantity, Operational Quantities, External Dosimetry, Internal Dosimetry, Effective Dose

Introduction

This paper analyzes the basic quantities along the dosimetric chain, the absorbed dose - the effective dose, in external and internal exposure, to ionizing radiation, based on the nominal risk coefficients to avoid induced cancers and hereditary diseases. In this paper, we mention some aspects as follows. The paper begins with the presentation of the radiation characteristics, regarding the radiation field, particle fluence, Φ energy fluence Ψ and their rates (Φ, Ψ) necessary for defining and measuring the fundamental dosimetric quantities, exposure, X, kerma, K, and absorbed dose, D, under charged particle equilibrium and Bragg-Gray conditions. These dosimetric quantities support the management of radiation exposure after justification and optimization, in the application stage of the dose limits, ensuring that the dose (applied - limit) by calibration factors (N_X, N_K, N_D) is traceable to the national dosimetry standards PSDL. The determination of the absorbed dose in case of photon irradiation is done with the quantities resulting from the application of the Law of Attenuation and Absorption of Photons: the total linear attenuation coefficient, μ / ρ, the mass energy transfer coefficient, μ_{tr} / ρ, and the mass energy absorption coefficient, μ_{ab} / ρ. For the absorbed dose from charged particles, the Law of Mass Stopping Power allows the definition of electron interaction coefficients by inelastic collisions with matter, the mass electronic (or collision) stopping power, S_{col} / ρ, and linear energy transfer, L_Δ, which are used to determine and measure the dose. Radiation Protection Quantities for External and Internal Exposure are also presented: Absorbed dose, D_{TR}, in target (organ or tissue) T, multiplied by the radiation weighting factor, w_R, determines the equivalent dose, H_T, in target organ, T. Based on epidemiological data for each target organ, a nominal risk coefficient for cancer and hereditary effects at single or annual low dose,(≤ 100mSv), is estimated. This coefficient amplified by the equivalent dose, H_T,

determines the risk of cancer induction in the target. The nominal coefficient per organ divided by the sum of the nominal coefficients of all the organs that compose the ensemble, represents the tissue weighting factor w_T, normalized so $\sum w_T = 1$. This, in turn, multiplied by the organ equivalent dose determines the effective dose of the considered organ, E. The paper further presents operational quantities that replace the quantities of radiation protection for external exposure i.e. when the radiation source is outside the body.Because the doses (H_T, E) are not measurable in practice, the evaluation of the effective dose is made with the operational quantities, the equivalent of the area dose H* (10), and the equivalent of the personal dose, H_p (10).They represent, the monitoring of strongly penetrating radiation (d = 10 mm) and neutrons.The quantity H"(d, Ω),) is used for the monitoring of low penetration radiation, e.g. β particles, in the case of dose control in the lens of the eye (d = 3 mm) and dose control in the skin, hands and feet (d = 0.07 mm). For internal exposure, when the radiation source is in the body, the quantities of radiological protection are based on the law of nuclear decay. In the MIRD methodology, the internal dose, $\bar{D} = \tilde{A} \cdot S (r_T \leftarrow r_S)$, is the product of the physical parameter S $(r_T \leftarrow r_S)$, a tabulated "dose coefficient" and the cumulated activity, $\tilde{A} = \int A(t)dt$), for an infinite interval of time. In the ICRP methodology, the organ equivalent dose H_T and the internal effective dose E are replaced by the committed equivalent dose H_T (50) in organ or tissue, T, and the committed effective dose E(50), resulting from an incorporation of radionuclides followed by a period of 50 years old. In the formula of the equivalent dose, $H_T(50)$, enter the number of transformations in the source organ, $S_U(50)$, and the specific effective energy, SEE (T ← S). Derived amounts for the control of internal exposure to ingestion and inhalation (ALI), the activity concentration of radionuclide in breathed air by man(-DAC), the external exposure to tritium or noble gases (DAC) and the radon exposure in air (WL, WLM). Limits of the absorbed dose and the effective dose are presented. For occupational exposure, the effective dose, the value of which depends on the prevention of induction of stochastic and non-stochastic effects, is calculated by the formula, $E=H_p(10)+E(50)$. For the public, the equivalent of the annual effective dose is 2.4 mSv of which 1.6 mSv belongs to internal exposure and 0.8 mSv of external exposure due to the background of natural radiation.

Radiation Characteristics

RaditionCategories

Charged ionizing particles and uncharged ionizing particles. In the context of radiation safety radiation is divided into three categories: directly ionizing radiation produced by charged particles, indirectly ionizing radiation produced by electromagnetic waves with wavelengths of a few nanometers, and non-ionizing radiation, which is radiation of electromagnetic waves whose wavelength exceeds that of ionizing radiation [1].

Directly Ionizing Radiation

These aregenerated by electrically charged particles such as electrons e- (4-25 MeV), protons p (50-250 MeV) and carbon ions (100-450 MeV/u), for radiotherapy, alpha particles α++, positron e+, antiproton p+, etc. They lose energy through interaction at low energies (ionizations) and have defined ranges. We mention the radon in the atmosphere where the public exposure is about 50%. This disintegrating radionuclide emits directly ionizing radiation, alpha particles, α, beta particles, β, and γ photons (without electric charge).

Indirectly Ionizing Radiation

These are generated by electrically uncharged particles, such as X-rays (generated outside a nucleus), γ-gamma rays (emitted from a nucleus), neutron, neutrino and braking radiation with an energy of 4-25 MeV for radiotherapy.The attenuation of uncharged particles in the substance takes place according to an exponential law.

As main sources of braking radiation for radiodiagnosis and radiotherapy, we mention mammography sources 22-40 keV; dental radiography 50-100 keV; computer tomograph (CT) 100-150 keV; ortovoltage 150-500 keV; supervoltage 500-1000 keV; megavotage> 1000 keV.

As radioactive materials we mention: Cesium-137 (γ, 30 y, $1.6 \cdot 10^6$ Ci) - Food irradiator, Cobalt-60 (γ, 5 y, 15,000 Ci) – Cancer therapy, Indium -192 (74 d, 100 Ci); Strontium - 90 (29 y, 0.1 Ci) – Eye Therapy Device, Iodine - 131 (8 d, 0.015 Ci) – Therapy, Technetium 99m (6 h, 0.025)- Imaging, Americiu-241 (432 y, 5 μCi) – Smoke Detectors, Radon -222 (4d, 1 pCi/ℓ) –

Nonionizing Radiation

Examples

Electric wave, Radiation heat, RF wave, Microwave, and Optical wave.Non-ionizing radiation (NIR) generally means electromagnetic radiation whose quantum energy is less than 3.26 eV, the limit of the visible spectrum (λ = 380 nm, ε = 3.26 eV, T= $3.8 \cdot 10^4$ K, where K (≡ °K, 1976, 0° K = -273.15° C).

Photons of infrared radiation have energies ranging from $4 \cdot 10^{-3}$ to 1.6 eV. However, although called "non-ionizing" radiation, photons with an energy of up to 3 eV can ionize certain molecules; an electron in a sodium atom, for example, requires only 2.3 eV of energy to remove it from the atom.

Photons of solar radiation reaching the earth, having wavelengths in the range (0.28 - 2.5) μm and photon energies from $4 \cdot 10^{-3}$ eV to 4.4 eV do not produce ionization or excitation. The photon energies are determined by Planck's formula.

Also, the residual radiation from the Big Bang, the cosmic background radiation, which exists throughout the space, has a temperature of 2.7 K and a frequency of 56 GHz. Using the Boltzmann constant, the energy of the radiation photons is $2.3 \cdot 10^{-4}$ eV, small for ionization and excitation of atoms.

Radiation Field

Agent between the Radiation Source and the Irradiated Substance. According to ICRU85, radiation fields consisting of various types of particles, such as photons, electrons, neutrons, or protons, are characterized by radiometric quantities that apply in free space and in matter.

In all categories of radiation, the radiation field represents the agent, the cause of the phenomena that arise in the irradiated substance. It is determined by the quantities that depend on the number of particles, N, i.e.,the number of particles that are emitted,

transferred, or received, and other quantities that depend on the energy transported, E,by them,[2].

From radiometric quantities associated with a radiation field, we mention: particle fluence, Φ = dN / dA, as the number of particles incident on a sphere of cross-sectional area, dA, of incident particles per area element dA, in units of m^{-2} and the particle energy fluence, Ψ = dE / dA, defined as the radiant energy incident on a sphere of cross-sectional area dA, in units of J / m^2. The radiant energy, R, is the energy (excluding rest energy) of the particles that are emitted, transferred or received.

Variations in these quantities over time are also defined: fluence rate ($\dot{\Phi}$ = dΦ/dt, unit: $m^{-2}s^{-1}$) and energy fluence rate, ($\dot{\Psi}$ =dΨ/dt in J/m^2s).All radiometric quantities can be scalar quantities and vector quantities. Among the vector quantities, necessary for determining the absorbed energy, we mention:the vector energy fluence, Ψ (J/cm^2), i.e., the energy of all particles that cross the diametrical area dA, of an elementary sphere, in the direction specified by theversor, (Ω), and the vector energy fluence rate, $\dot{\Psi}$(W/m^2) as a derivative of it over time, as

$$\vec{\Psi} \equiv \left(\frac{dR}{dA}\right)\vec{\Omega}, \qquad \dot{\vec{\Psi}} = \left(\frac{d^2R}{dA \cdot dt}\right)\vec{\Omega}. \tag{1}$$

The applications of particle fluence Φ in determining the dosimetry quantities (X, K, D), in conditions of electronic equilibrium, are included in equations (2),

$$\Phi\left[\frac{1}{m^2}\right] = \frac{0.00873}{h\nu\left(\frac{\mu_{ab}}{\rho}\right)_{air}} X(R) = \frac{D_{air}}{\left(\frac{S_{el}}{\rho}\right)_{air}} = \frac{K_{air}(1-g)}{\left(\frac{S_{el}}{\rho}\right)_{air}}, \tag{2}$$

Where hν is the energy of the photon and μ_{ab}/ρ is the mass energy absorption coefficient, the interaction coefficient of the photon with the matter.

The relation between the energy fluence of the beam Ψ and the three dosimetric quantities is given by the system of equations (3),

$$\Psi\left[\frac{J}{m^2}\right] = \frac{0.00873}{\left(\frac{\mu_{ab}}{\rho}\right)_{air}} X[R] = \frac{D_{air}}{\left(\frac{\mu_{ab}}{\rho}\right)_{air}} = \frac{K_{col,air}}{\left(\frac{\mu_{ab}}{\rho}\right)_{air}}, \left[\frac{Gy}{\frac{m^2}{kg}}\right], \tag{3}$$

Characteristic Quantities of Radioactive Sources

Radiation fields, which represent the agent in the case of irradiation of the substance, are in turn the effect of radiation sources. For a radioactive source with N_0 the number of existing nuclei, at the arbitrarily chosen moment $t_0 = 0$, in a given mass of radioactive nuclide, with N nuclei remaining undintegrated at time t, the variation dN of the nuclei that disintegrate in the time interval dt, is dN = λNdt, where λ is the proportionality constant, called the decay constant, i.e.,the fraction of the atoms that undergo decay per unit time

$$\lambda = \frac{1}{dt}\frac{dN}{N} \quad \text{or} \quad N = N_0\, e^{-\lambda t}\,. \tag{4}$$

From equation (4), results the first characteristic, the activity of the radioactive sourceA(t), a measure of the rate of decay

$$A(t) \equiv -\frac{dN}{dt} = \lambda N = A_0 e^{-\lambda t}. \tag{5}$$

The unit of measurement of activity in SI is s-1, which received the special name of Becquerel Bq (1974), 1Bq = 1 s^{-1}.

Other main features of radioactive decay are [3]:

- the effective half-life, $T_{1/2} = \ln(2)/\lambda$;
- the effective average life, $T_{ave} = 1/\lambda = 1.443\ T_{1/2}$;
- the number of radioactive atoms in sample, it is equal with the total number of desintegrations, $N = A/\lambda = A\ T_{ave} = 1.44AT_{1/2}$;
- the number of atoms decayed, $N_{dis} = N(1 - f)$; $f = e^{-\lambda t}$;
- the secular equilibrium between the generating nuclide and the derived nuclide, $A_1(t) = A_2$ at $T_{1,1/2} >> T_{2,1/2}$,
- the transient equilibrium,when the activity of product A2(t) is higher than that of the generator A_1 (t),$T_{1,1/2} > T_{2,}$ ½, and,
- the cumulated activityin the S source organ, $\tilde{A}S = \int A(t)dt$.

Relation between a Radioactive Source and theRadiation Field For a point source of activity, A(t), the dosimetric quantities, measured at a point at a distance ℓ from the source, at which the self-absorption and self-spreading corrections, are neglected, constant Γγ have the following expressions [4],[5].

Exposure rate constant in air:

$$\Gamma_\gamma = \ell^2 \cdot \frac{\dot{X}}{A}\left[\frac{10^{-11} R \cdot m^2}{MBq} - h\right], \tag{6}$$

Kerma rate constant in air, for photons emitted with an energy greater than δ:

$$\Gamma_\delta = \ell^2 \frac{K_\delta}{A}\left[\frac{m^2 Gy}{Bq} - s\right], \tag{7}$$

Specific dose-rate constant in air,

$$\Gamma_\gamma^D = \frac{W_a}{e}\Gamma_\gamma = \ell^2 \frac{\dot{D}_a}{A}\left[10^{-13}\frac{Gy \cdot m^2}{Bq} - h\right], \tag{8}$$

Specific beta radiation constant, Table 1:

$$\Gamma_\beta = \ell^2 \cdot \frac{\dot{D}_a^\beta}{A} \cdot f(\rho_a, \ell)^{-1}\,[Rm^2\ per\ Bq - h], \tag{9}$$

where ρ_a is a density of air, $f(\rho_a,\ell)$is a correction function that takes into account the energy lost by beta particles due to their interaction over the distance ℓ (Table 1, [4]).

Table 1: Values for Calculating Beta Radiation Exposure

Radionuclides	$\bar{E}_\beta$ (MeV)	Γ_β(10-11 Gy.m_2/Bq·h)	k1/k2	μ2 cm2/g	μ2 cm2/g
^{35}S	0.049	3.92	1/0	175	-
^{60}Co	0.093	2.62	1/0	60	-
^{90}Sr	0.20	2.00	1/0	22	-
^{90}Y	0.93	0.77	1/0.7	3.5	5.5
^{32}P	0.69	0.905	1/0.7	5.6	8.8

Table 2 [4] shows the dose and exposure rate constants for different radionuclides in air.

Table 2: Dose Rate Constants of Different Radionuclides in Air

Radio-nuclide	Γγ (10-11 R·m²/Bq·h)	Γ_{γ}^{D} (10-13 Gy·m² Bq·h)
^{41}Ar	1.62	1.73
^{56}Mn	2.21	2.36
^{59}Fe	1.52	1.63
^{60}Co	3.19	3.41
^{65}Ni	0.74	0.787
^{85}Kr	0.00295	0.00314
^{95}Zr	1.01	1.08
^{131}I	0.53	0.551
^{133}Xe	0.034	0.0368
^{137}Cs	0.79	0.846

Units of Radiation and Radioactivity

The equivalencies between classical and SI dose units are presented in Table 3.

Table 3: Dose units

Unit/ Functions in Management	Classical unit	SI unit	Unit conversion
Activity/ Intake limitation	Curie (Ci)	Becquerel (Bq)	1 Ci =3.7x1010 Bq
Absorbed dose/ Dose limitation	Radiation absorbed dose (Rad)	Gray (Gy)	100 Rad = 1 Gy
Equivalent dose/ Dose limitation	Roentgen equivalent man (Rem)	Sievert (Sv)	100 Rem = 1 Sv
Effective dose/ Dose limitation	Roentgen equivalent man (Rem)	Sievert (Sv)	100 Rem = 1 Sv
Collective effective dose/ Optimisation	Persons (P) x 100 Rem	Persons x Sievert	100 Rem x P = 1 Sv x P

Basic Quantities for Protection and Physical Dosimetry.

Dosimetric Concepts in Physical Dosimetry, X, K, D

Exposure, X It is a measure of the number of ionized atomsin air for X- ray and γ -rays of energy up to 3 MeV.The exposure is the quotient of dQ by dm, where dQ is the absolute value of the mean total charge of the ions of one sign produced when all the electrons and positrons liberated or created by photons incident on a mass dm of dry air are

$$X \equiv \frac{dQ}{dm} = \frac{dQ}{\rho dV}. \quad (10)$$

The unit of measurement in SI is C/kg (1974). The old unit from 1928, roentgen (R), defined as, 1R = 2.58 x 10^{-4} C·kg^{-1}, is also tolerated.1 R = 0.0087 Gy = 1,61x10^{12} ionized atoms/g-air.

The exposure rate $\dot{X}$, is the quotient of dX by dt, where dX is the increment of exposure in the time interval dt, thus,

$$\dot{X} = \frac{dX}{dt}, \quad (11)$$

The unit of measurement in SI is C/kg·s and R/s.

The exposure measured in a uniformly irradiated mass of air, with Ni atoms ionized in the given mass, is

$$X = \frac{Q}{m_a} = \frac{Q}{\rho_a V} = \frac{N_i e}{m_a}, \quad (12)$$

where e is electron charge = 1.6 x 10^{-19} C =4.8 x 10^{-10}u.e.s. Applications of exposure X are found when determining the dose absorbed in the medium m, D_m, under CPE (3.3) or Bragg-Gray (3.7.2) conditions.

$$D_m = \frac{W_a}{e}\frac{(\mu_{ab}/\rho)\,m}{(\mu_{ab}/\rho)a}X,\ D_m = \frac{W_a}{e}\frac{S_{col}/\rho)\,m}{(S_{col}/\rho)a}X. \quad (13)$$

Kerma, K

Ita measure of energy transferred to a medium from the indirectly ionizing radiation. The kerma is a ratio between dE_{tr} and dm,

$$K \equiv \frac{dE_{tr}}{dm} = \frac{1}{\rho}\frac{dE_{tr}}{dV}, \quad (14)$$

where d E_{tr} is the mean sum of the initial kinetic energies of all the charged particles liberated in a mass dm of a material by the uncharged particles (photons) or by reaction (f,e^-) and the mass dm.In the International System of Units (1960), unit of kerma is gray, Gy, (1974).1Gy = 100 rad (1953).

The kerma rate, $\dot{K}$, is the ratio between the kerma variation dK, in the time interval dt, thus

$$\dot{K} = \frac{dK}{dt} \quad (15)$$

The unit of kerma rate is gray per second (Gy/s).

According to the law of conservation of energy, the mean energy transferred to the volume V, is equal to the mean energy absorbed by collisions and excitations of atoms plus the mean energy of photons (braking radiation or bremsstrahlung) which leave this volume, i.e.

$$E_{tr} = E_{ab} + E_B. \quad (16)$$

As a result, the quantity Kerma K includes the electronic component that remains in the element of mass dm, $K_{col} = (dE_{tr}/dm)_{col}$ and the component of the braken radiation, $K_{rad} \equiv B = (dE_B/dm)_{rad}$ which leave the considered volume.Noting with $\bar{g}$ thethe fraction of energy transferred to the braking radiation, $\bar{g} = E_B/E_{tr}$, results

$$K_{col} = K - B = K(1 - \bar{g}). \quad (17)$$

The collision kerma in air at the calibration point P in free space is related to exposure X by

$$(K_{col})_a = \left(\frac{W}{e}\right)_a \cdot X. \quad (18)$$

Absorbed dose,D

Itis ameasure of energy imparted to a medium from all types of ionizing radiation and for any material.

The absorbed dose is the meanenergy imparted $d\bar{\varepsilon}$ by ionizing radiation to a mass dm of volume dV, thus

$$D = \frac{d\bar{\varepsilon}}{dm} = \frac{d\bar{\varepsilon}}{\rho dV}. \quad (19)$$

The special name for the unit of absorbed dose is gray (Gy).1Gy = 1 J·kg^{-1} = 100 rad =114 R = 6.24 x 10^{9} MeV/g.

The absorbed-does rate, Ḋ, is the increment of absorbed does, dD, in the time interval dt, thus

$$\dot{D} = \frac{dD}{dt}. \quad (20)$$

Unit: J·kg^{-1}·s^{-1}. If the special name gray is used, the unit of absorbed dose rate is gray per second (Gy/s).

The dose being defined for an elementary volume dV, in this elementary volume of a substance irradiated with a beam of radiation, the yielded energy $\bar{\varepsilon}$, is the difference between the sum of the energies of all particles directly (electrons) or indirectly ionizing (photons) given by E_{in} entering in the volume dV of mass dm and also the sum of the energies of all charged particles (delta electrons) or not (photons) coming out of that volume, Eout plus plus the mean sum, ∑Q, net energy derived from mass in V (m → E positive,E→ mnegative) [2], thus

$$\bar{\varepsilon} = E_{in} - E_{out} + \sum Q. \quad (21)$$

In volume dV, for example, there are interactions of photons with the substance that give rise to electrons (Compton, photoelectrons and the production of pairs (e^-, e^+) and interactions of electrons with the free electrons of the atoms of the substance.

The absorbed dose measured at a point in the air as a result of an X exposure, called specific ionization J_a is

$$D_a = \frac{W_a}{e} X = \frac{W_a}{e} J_a, \quad (22)$$

where W_{air} is the average energy expended by beta per ion pair in air and e is the electron charge.

The Gray unit, for radioactive materials content in the body, in 70 kg man (Bq /annual dose (mGy)), means ^{40}K (4,433/ 0.18) - whole body, 14C (3,217/ 0.011) - whole body, ^{226} Ra (1.48/0.007 - bone lining, ^{210}Po (18.5/0.006) – gonads, /0.03 – bone, $^{90}S_r$ (48.1/0.026) – endosteal bone, /0.018 - bone marrow [3].

Dosimetry Fundamental Equation

The fundamental equation of dosimetry is obtained from equaling the average energy imparted to volume V from the irradiated substance of density ρ, $\bar{\varepsilon} = \int_V D\rho dV$, from (19), with the field energy transferred to volume V by the vector energy fluence $\vec{\Psi}$ (see the Poynting's theorem) through the closed surface, A, which borders this volume, $E_{tr} = \oint A \vec{\Psi} d\vec{A}$.
Applying the Gauss's theorem, one can obtain [4]

$$\bar{\varepsilon} = \int_V D\rho dV = -\oint_A \vec{\Psi} d\bar{A} \equiv -\int_V div\, \vec{\Psi} dV. \quad (23)$$

Considering that the quantities below the integral are constant at sufficiently small volumes, the balance equation (23) becomes the fundamental equation of dosimetry, for any ionizing radiation field, directly or indirectly,

$$D = -\frac{1}{\rho} div\, \vec{\Psi} \quad \text{or} \quad \dot{D} = -\frac{1}{\rho} div\, \vec{\Psi}. \quad (24)$$

Here $\vec{\Psi}$ is the sum between the vector energy fluence of the directly ionizing particle beam, charged particles, e.g. electrons, $\vec{\Psi}_c$, or charged secondary particles, e.g. secondary electrons, $\vec{\Psi}$c,s and the vector energy fluence of the indirectly ionizing particle beams, e.g., photons γ, $\vec{G}_u$, i.e., $\vec{\Psi} = \vec{\Psi}_c + \vec{\Psi}_{c,s} + \vec{\Psi}_u$.

It expresses the fact that the energy absorbed in an elementary volume of irradiated material, of density ρ, is imparted by the radiation field by the vector energy fluence or its rate, $\vec{\Psi}$, in the case of the absorbed dose rate, (Ḋ), as follows:

$$D = -\frac{1}{\rho} div\, \overrightarrow{\Psi c} - \frac{1}{\rho} div\, \overrightarrow{\Psi c,s} - \frac{1}{\rho} div\, \vec{\Psi} u. \quad (25)$$

When irradiating matter is done with photons, using the relation (17), equation (25) becomes

$$D = -\frac{1}{\rho} div\, \vec{\Psi} c,s + K - B. \quad (26)$$

In the case of the charge particle equilibrium, (CPE), the first term in the second member is equal to zero (div $\vec{\Psi}_{c,s}$ = 0), and equation (26) become

$$D = K - B. \quad (27)$$

The absorbed dose is equal to kerma, D = K, when the energy transferred by secondary electrons to tertiary photons, especially through Bremsstrahlung, is negligible, B = 0.

In the case of irradiation with charged particle beams (electrons, protons, particle, α, heavy ions), K = 0, bremsstrahlung losses can be neglected, B = 0, and ecuation (23) becomes

$$D = -\frac{1}{\rho} div\, \vec{\Psi} c. \quad (28)$$

Principle of Charged Particle Equilibrium

The equilibrium of charged particles exists when the sum of the kinetic energies of the charged particles entering the elementary volume, dV, is equal to that of the outgoing particles, i.e., Rin = R_{out}, or div $\vec{\Psi}_c$ = 0, to which is added the condition, B=0.

The absorbed dose distribution and in-depth kerma are shown in Fig.1. The increment of kerma faithfully follows the variation of the photon fluence, decreasing exponentially with the depth x.
The absorbed dose increases, at the intersection point x_A, exist a rigorous electronic equilibrium,

$$D_a = (K_{col})_a = \left(\frac{W}{e}\right)_a \cdot X, \quad (29)$$

Just like at the radioactive decay - secular equilibrium.

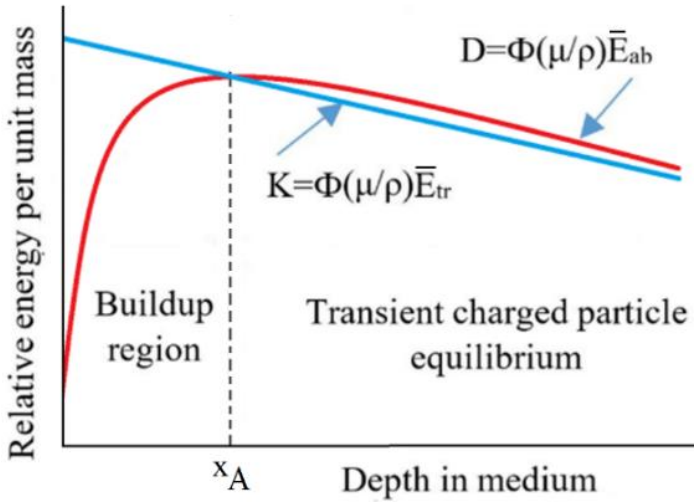

Figure 1: Rigorous Equilibrium for Kerma and Absorbed Dose

Beyond x_A, there is only transient electronic equilibrium, both kerma and absorbed dose decrease with depth after equilibrium occurs [3]. The equivalence between the dosimetric quantities is made at the intersection point, where CPE is performed.

Relationship between the Dozimetry Calibration Factors

Ionization chambers used in radiation therapy (photons/electrons, soft X–rays), radiology diagnostic (conventional, mammography, dental) and radiation protection (H*(10)), are calibrated at primary and / or secondary standards for measuring the absorbed dose in water, Dw [Gy], kerma aer, [Gy], specific ionization, J_S, or air exposure, X[C/kg]/[R] [6], [7].

Calibration factors N_X (=X_c/M_c) for exposure in [R] can be converted in calibration factors N_K (= $K_{air,c}/M_c$) for air kerma in [Gy] by

$$N_K[Gy] = N'_X\left(\frac{C}{kg}\right)\frac{W}{e}\frac{1}{1-g} = N_X[R]\cdot 2.58\,10^{-4}\frac{C}{kg.R}\cdot 33.97\,\frac{J}{C}\cdot\frac{1}{0.997} = N_X[R]\cdot 0.00877\,\frac{Gy}{R}. \quad (30)$$

To these is added the calibration factor of the dose absorbed in the water and air; $N_{(D,w)}^{60Co} = D_w^{60Co}/MC$, and $N_{D,air} = N_{D,W}/(S_{W,air}pQ)$ C0-60.

Typical units of N_X and N_K are R/nC and Gy/nC, respectively. A typical unit for both $N_{D,air}$ and $N_{D,w}$ is Gy / nC.

The dosimetric measurements developed with these quantities are traceable to the reference (primary) standard [8],[9].

Photon Interaction Coefficients based on the Attenuation and Absorption

Total linear attenuation coefficient, μ/ρ

The external irradiation of a medium absorbing with the collimated beam of monoenergetic photons of energy hν has as effect, the attenuation of the photon fluence, Φ, through the three major effects that occur: the Compton effect, the photoelectric effect and the pair generation effect (e^- and e^+) [10].

Experimentally it is found that for a layer of infinitely small thickness dx, from a substance, the variation of the photon fluence is - dΦ = μ·Φ·dx, where Φ is the photon fluence and μ is a proportionality constant, called the total linear attenuation coefficient for the three major processes.

The mass attenuation coefficient is defined by (μ/ρ), where ρ is the density of the absorber material. The unit of μ/ρ is m^2/kg, or in conventional units, cm^2/g, [11].

In absolute value, the proportionality constant μ/ρ, for uncharged particles, can be put in the form

$$\frac{\mu}{\rho} = \frac{1}{\rho d\ell}\frac{d\Phi}{\Phi} \quad \text{or} \quad \Phi = \Phi_0 e^{-\frac{\mu}{\rho}\rho\ell}. \quad (31)$$

Here, μ/ρ, Este the average fraction of particles that undergo interactions in crossing the distance dℓ in the density material ρ. After the integration of the relation (31) the law of attenuation of the fluence is obtained; Φ_0 is the incident fluence, Φ is the fluence remaining after a thickness, dℓ.

The total linear attenuation coefficient, μ/ρ = (τ + σ + κ) (1/ρ), τ/ρ is the linear attenuation coefficient for the photoelectric process, σ/ρ = (σ_S + σ_a)/ρ - the linear attenuation coefficient for Compton made of the linear attenuation coefficient for Compton scattering, σ_S/ρ, and the linear absorption coefficient Compton (σ_a/ρ), and κ/ρ - the linear attenuation coefficient for the pair- production process [12].

Mass Energy Transfer Coefficient, μ_{tr}/ρ

When irradiating an elementary volume dV with a beam of N photons of energy hv, E = N.hv considering that the average energy transferred in that volume is E_{tr} results that the variation of the energy transferred in elemental dℓ, is $dE_{tr} = E_{tr}\cdot\mu\cdot N d\ell = \mu_{tr}\cdot E d\ell$ or $d\Psi tr = \mu_{tr}\cdot\Psi_{tr}\cdot d\ell$. From this equation, the relation for the mass energy transfer coefficient, defined by (μ_{tr}/ρ), is obtained

$$\frac{\mu_{tr}}{\rho} = \frac{1}{\rho d\ell}\frac{dE_{tr}}{E}\frac{1}{\rho d\ell}\frac{d\Psi}{\Psi} \quad \text{or} \quad \Psi = \Psi_0 e^{-\frac{\mu_{tr}}{\rho}\rho\ell}, \quad (32)$$

or

$$\frac{\mu_{tr}}{\rho} = \frac{\mu}{\rho}\frac{E_{tr}}{h\nu} \qquad \frac{\mu_{ab}}{\rho} = \frac{\mu}{\rho}\frac{E_{ab}}{h\nu}, \quad (33)$$

where μ_{ab}/ρ is the mass energy absorbed coefficient. Relationship (16) makes the connection between the two coefficients, transfer and absorption

$$\frac{\mu_{ab}}{\rho} = \frac{\mu_{tr}}{\rho}(1-\bar{g}) = (1-\bar{g})\frac{1}{\rho dx}\frac{dE_{tr}}{E}. \quad (34)$$

This gives the connection between the two coefficients of interaction, mass transfer and mass absorption.

Calculation formula for μ_{tr}

After removing the radiative energies of the three major interaction processes (Figure 2 [13]), δ – the average energy of the characteristic X-ray for the photoelectric effect, fS - the energy fraction in Compton collision taken by the scattered photons, and $2m_0c^2/h\nu$ – fraction of the incident energy taken by (e^+ e^-) annihilation radiation photons, is obtained the mass energy transfer coefficient, for photons

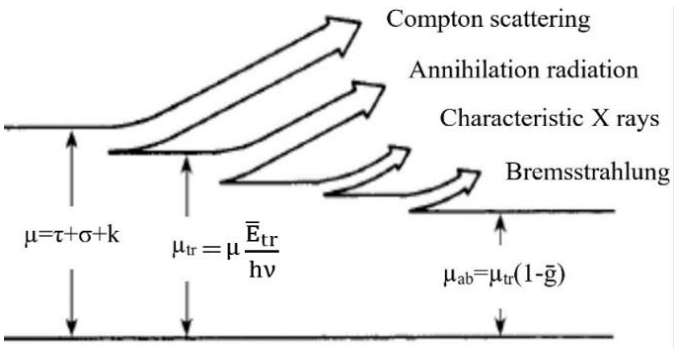

Figure 2: The relation of the attenuation coefficient of the photon μ, of energy transfer $\mu_{tr} = \mu - \sigma_s$, and energy absorption μ_{ab}.

$$\frac{\mu_{tr}}{\rho} = \frac{1}{\rho}\left[\tau\left(1-\frac{\delta}{h\nu}\right) + \sigma\frac{T_{avg}}{h\nu} + \kappa\left(1-\frac{2m_0c^2}{h\nu}\right)\right]. \quad (35)$$

Using (34) relation one can obtain the calculation expression for the total mass energy-absorption coefficient

$$\frac{\mu_{ab}}{\rho} = \frac{1}{\rho}\left[\tau\left(1-\frac{\delta}{h\nu}\right) + \sigma\frac{\bar{E}_c}{h\nu} + \kappa\left(1-\frac{2m_0c^2}{h\nu}\right)\right](1-g). \quad (36)$$

Equation (32), can be written like this

$$-d\dot{\Psi} = \mu_{tr}\dot{\Psi}\,d\ell \quad \text{or} \quad \frac{1}{\rho}\frac{d\dot{\Psi}}{d\ell} = \frac{\mu_{tr}}{\rho}\dot{\Psi}, \quad (37)$$

where kerma rate $\dot{K} \equiv \dot{\Psi}/\rho\ d\ell$ and kerma $K \equiv d\Psi/\rho \cdot dl$ or

$$\dot{K} = \frac{\mu_{tr}}{\rho}\dot{\Psi} \quad \text{or} \quad K = \frac{\mu_{tr}}{\rho}\Psi, \quad (38)$$

where Ψ este photon energy fluence (in units of $J.m^{-2}$) and **μtr /ρ** (in units of $m^2.kg^{-1}$) is the total mass energy- transfer coefficient in the considered medium. Analogous to equation (38) is the expression for absorbed dose rate and absorbed dose in intervalul de iradiere t

$$\dot{D} = \frac{\mu_{ab}}{\rho}\dot{\Psi} \quad \text{or} \quad D = \frac{\mu_{ab}}{\rho}\Psi = \dot{D}\cdot t. \quad (39)$$

Evaluation of Photon Dose from the Energy Fluence and Transfer Coefficients

The relation (38) written for air and water, assuming that the energy fluence of the photons Ψ is identical for the two mediums, allows to obtain kerma col in water, $(K_{col})_w$, as a function of the kerma col in air, $(K_{col})_a$, or kerma as a function of the dose in medium, D_m ,

$$(K_{col})_w = (K_{col})_a\left(\frac{\mu_{ab}}{\rho}\right)_a^w, \quad K_m = D_m\frac{(\mu_{tr}/\rho)m}{(\mu_{ab}/\rho)m}. \quad (40)$$

Using the relation (39), the dose absorbed in air D_a and the dose absorbed in medium Dm, at a point in the X-ray field with energy fluence Ψ, can be written as follows

$$D_a = \Psi_a\cdot\left(\frac{\mu_{ab}}{\rho}\right)_a \qquad D_m = \Psi_m\cdot\left(\frac{\mu_{ab}}{\rho}\right)_m. \quad (41)$$

In both situations, assuming the identical energy fluence of the photons $\Psi_a = \Psi_m$, results the expression for calculating the dose absorbed in the medium, as a function of the dose in air

$$D_m = D_a\frac{(\mu_{ab}/\rho)_m}{(\mu_{ab}/\rho)_a}. \quad (42)$$

The exposure rate (or exposure) in air, at photonic radiation with energies up to 3 MeV, in conditions of electronic equilibrium, it is calculated with the relation,

$$\dot{X} = \dot{\Psi}.\frac{\mu_{ab}}{\rho}\frac{e}{W_a}, \quad (43)$$

Obtained with the equations (18) and (39). Exposure rate X is given in units: C/kg·s, for Ψ ($J\cdot m^{-2}\cdot s^{-1}$), μ_{ab}/ρ (m^2/kg) and $(e/W_a) = 1/33.97$ (J/C). Ẋ is expressed in R/s, when the result obtained with equation (43), is divided with 2.58×10^{-4} C/kg.R.

It should be mentioned that to exist CPE, the thickness of the chamber walls must be at least equal to the maximum range of the secondary electrons in their material. In addition, the dimensions of the chamber cavities are significantly larger than the range of the secondary electrons, to satisfy the CPE principle of using air ionization chambers [1].

Electron Interaction Coefficient Based on Stopping Power

Stopping power for charged particles, S

A quantity that characterizes the incident particle, the environment and the interactions that take place.The main interaction of the secondary electrons in the irradiated environment with a photon beam, is the Coulomb interaction with the orbital electrons of the substance atoms, following which two processes take place: the transfer of an orbital electron to a higher energy level, a process called atom excitation, and the separation of an orbital electron, process called atom ionization.

Both processes, ionization and excitation are considered inelastic collisions, because by interaction a part of the kinetic energy of the incident electron is consumed due to the binding energy of the target electron in the atom.

Excitation and ionization are considered local processes of interaction for the range relatively low (see Table 4).

Table 4: Ranges of ionizing particles

Particle	Energy in MeV	Approximate range in soft tissue
Electron or β-particle	0.01	3 μ
	0.02	7 μ
	0.1	130 μ
	1.0	0.4 cm
	2.0	1 cm
Proton	1.0	28 μ
	2.0	80 μ
α - Particle	5.0	35 μ

The concept of the stopping power of a substance is defined as the average energy loss per unit length of the trajectory of a charged particle,

$$S = \frac{dE}{d\ell}, \quad (44)$$

where E is the energy of the electron (excluding the rest energy), and ℓ is the length of the trajectory. To eliminate the dependence on the mass density of the substance, the definition relation is divided by its density ρ. In this way, the mass stopping power is obtained

$$\frac{S}{\rho} = \frac{dE}{\rho d\ell}. \quad (45)$$

Considering that the average energy loss of the electron along its trajectory, dE/dℓ, occurs mainly by inelastic collisions and by the generation of braking radiation, the mass stopping power of the electron, S/ρ, is the sum of two independent components: the mass electronic (or collision) stopping power due to interactions with atomic electrons resulting in ionization or excitation, S_{col}/ρ, and the mass radiative stopping power due to emission of bremsstrahlung in the electric fields of atomic nuclei or atomic electrons, S_{rad}/ρ, thus

$$\frac{S}{\rho} = \frac{1}{\rho} S_{col} + \frac{1}{\rho} S_{rad}. \quad (46)$$

The SI units are in J·m²/kg or MeV.cm²/ g. The boundary between the two stopping powers is defined by the relation, S_{rad}/S_{col} = T·Z/800, where T is the kinetic energy of the particle charged in MeV, and Z is the atomic number of the target material.

Linear Energy Transfer, L_Δ

The linear collision stopping power, L_Δ, of a material, for a charged particle, is the ratio dE_Δ, on $d\ell$, where dE_Δ is the mean energy lost by the charged particles due to electronic interactions in traversing a distance dℓ, minus the mean sum of the kinetic energies in excess of Δ of all the electrons released by the charged particles, thus

$$L_\Delta = \frac{dE_\Delta}{d\ell}. \quad (47)$$

It is measured in keV/μm. This quantity is introduced for retain only the local effects in which the energy released by the charged particles is effectively absorbed by the substance in the elementary region considered. In this way, the fraction of energy transferred to both the braking radiation and the delta electrons, capable of subsequent ionizations, on a longer trajectory is eliminated.

The index △ indicates that the definition retains only the collisions in which the transferred energy is less than an imposed limit △, for example △=100 eV. If the collision energy limit moves to the maximum possible value, then linear energy transfer is identified with the collision stop power, L → L∞ = Scol and the limit becomes half of the maximum electron energy, ie. $\Delta = E_0 / 2$.

Electron Mass Stopping Power by Collision

The total mass stopping power, S/ρ, or the coefficient of interaction of directly ionizing radiation with the environment, defined in ICRU 1971, includes all losses of kinetic energy, dE, of an electron passing through a certain length, dℓ, in density substance ρ, which lead to the production of secondary electrons and atomic excitations.

This component, which characterizes both the incident particle and the environment passed through, as well as the interactions that take place, can be calculated using the relation Berger and Seltzer, 1964 [14]:

$$\left(\frac{S}{\rho}\right)\text{col} = \frac{2\pi r_0^2}{\beta^2} . m_0 c^2 . \text{NA} . \frac{Z}{M_A} . \{ \log \left[\frac{\tau^2(\tau + 2)}{2(I\, m_0\, c^2)2}\right] + \quad (48)$$

$$+ F(\tau) - \delta \},$$

where δ is the density correction (Sternheimer, 1952, 1953, 1956) and $F(\tau) = 1-\beta^2 + [\tau^2/8 - (2\tau+1)\log 2]/(\tau+1)^2$, m_0c^2 = rest energy of the electron (=0.511 MeV), $\tau = E/m_0c^2$ = ratio of kinetic energy of the electrons to the rest energy, β = v/c, v = velocity of the electrons, c = velocity of light, N_A = Avogadro constant (= $6.025 \cdot 10^{23}$ mol^{-1}), r_0 = electron radius, Z = atomic number, M_A = molar mass of substance A (unit: g/mol) and I = mean excitation energy of the medium

Knowing that the stopping power is inversely proportional to the depth of the particle, the relative stopping power, S_{rel}, is obtained using the comparison method. It is used to compare the energy lost in different substances, taking air as a reference medium. When the relative power is known, the path R_m in a range can be determined by the relationship

$$S_{rel} = \frac{(dE/dx)_m}{(dE/dx)_a} = \frac{R_a}{R_m}, \quad (49)$$

Where R_a and R_m are ranges in air and medium.

Average excitation energy, I

The average excitation energy I [eV] appears in the expression of the collision stop power, the rest of the quantities being known. Table 5 shows the average excitation energy values in water and other properties of human tissues ICRP 1975.

Table 5: Mean excitation energies, water content and other properties of selected human tissues (ICRP 1975)

Substance	I [eV]	ρ (g/cm3)	<Z/A>	Water content [%]
H_2O	75.0	1.00	0.55509	
Adipose tissue	63.2	0.92	0.55847	15.3
Skin	72.7	1.10	0.54933	61.5
Brain	72.3	1.03	0.55423	78.6
Testes	75.0	1.04	0.55108	80.0
Blood	75.2	1.06	0.54995	80.0
Lung	75.3	1.05	0.54965	78.0
Skeletal muscle	75.3	1.04	0.54938	78.6
Cortical bone	106.4	1.85	0.52130	15.0
Striated muscle	74.7	1.04	0.55005	78.6
Compact bone	91.9	1.85	0.53010	15.0

Average Energy Expended Per Ion Pair in Air, W

From a global point of view, the average energy dissipated in the gas for a pair of ions is defined, the ratio

$$W = \frac{E}{N}, \qquad (50)$$

where *N* is the average number of ion pairs formed by primary and secondary processes when the initial kinetic energy E of a charged particle is completely dissipated in the gas. Average values of the energy for a pair of ions in the case of several gases, for electrons and α particles shows in Table 6.

Table 6: Mean energy W(eV) for generating an ion pair (i.p.)

	Gas						
	H_2	H_e	N_2	O_2	A_r	CO_2	Air
Electrons,	36.3	41	34.7	31.0	26.4	32.8	33.97
Alpha particles, α	36.5	44	36.5	32.4	26.4	34.4	35.3

To form a pair of ions in dry air under STP conditions (760 mm Hg, 0°C and ρ = 0.001293g / cc), the mean energy is 33.97 eV, and 33.77 eV under NTP conditions (760 mm Hg, 20^0 C and ρ = 0.001205g / cc).

Calculations of ionized atoms show that, 1 R = $2.08 \cdot 10^9$ i.p. /cm^3 of air, and 1 Gy = $2.37 \cdot 10^{11}$ p.i./cm^3 of air. Lethal dose (at which 50% of the population exposed to gamma radiation dies in 30 days), $L_{50/30}$ = 9.18 x 10^{17} ionized atoms for 5 Gy.

Considering a number of neutral atoms, 9.5 x 10^{25} (atoms per kg of tissue), results about 10^8 unaffected atoms. This means that at about 10^{18} ionizations in the tissue, the lethal dose occurs. Ionization distributions versus depth in air with maximum ionization at the range end are called Bragg curves and are characterized by their shape, as shown in Figure 3 [13].

The variation of the specific ionization, i.p. per mm, is presented in Fig. 5 for strongly ionizing alpha particles of ^{210}Po (Ea = 5.304 MeV) and radium Ra C 'or ^{214}Po (E_a = 7.687 MeV) at 760 mm and 15°C. These particles produce an average about 50,000 pairs of ions/cm and range up to about 80,000 pairs of ions/cm near the end of their path. Beta particles produce a similar curve, but with a much lower millimeter ionization rate, usually about 50 of ion pairs / cm on average.

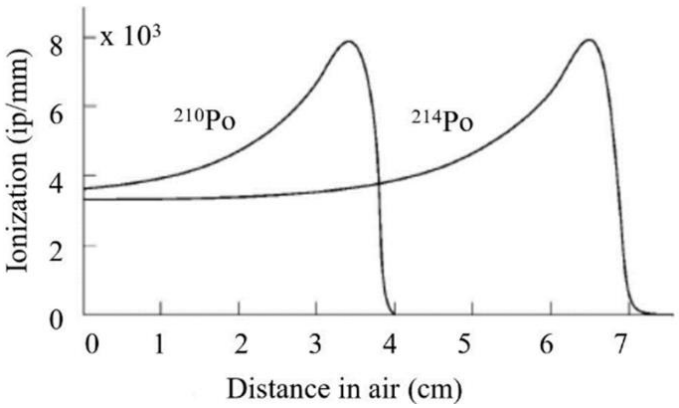

Figure 3: Bragg-Gray curves of apecific ionization (ion pairs per cm) of alfa particles from ^{210}Po and radium C' (^{214}Po) in air.

Principle of Bragg-Gray for Evalution Dose from Particle Fluence and Stopping Power

The determination of the absorbed dose in an environment irradiated with photon beams using the ionization method consists in the introduction of an electron detector (ionization chamber) of Bragg-Gray (BG) type, in the respective medium. The BG chamber is an ionization chamber that has a cavity small enough to not disturb the fluence of secondary electrons. The BG principle is illustrated in Figure 6.

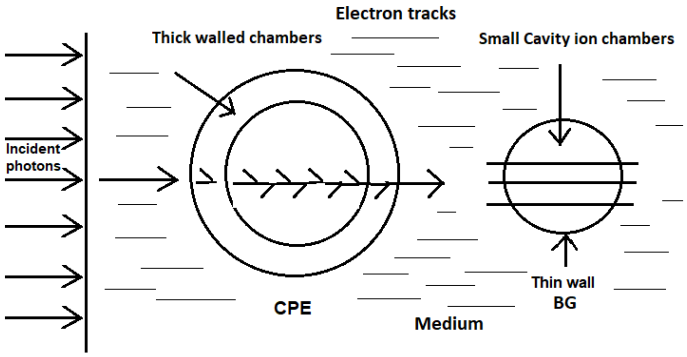

Figure 4: Cavities for illustrate CPE and Bragg-Gray principles

Knowing the collision stopping power, (S / ρ) col, and the absorbed dose can be calculated directly by amplifying it with the electron fluence, Φ, produced by irradiation with a photon beam, of a gas that is in the chamber cavity, so

$$D_g\,(Gy) = \Phi \left(\frac{S}{\rho}\right)_g. \qquad (51)$$

The dose absorbed in the environment surrounding the cavity, i.e., the wall, w, considering that the fluence of the charged particles Φ

that crosses the interface between the environments w and g does not change represents the relationship BG,

$$D_w = D_g\left(\frac{S_{col}}{\rho}\right)_g^w, \quad (52)$$

where we marked with $S_{w,g} \equiv (S_{col}/\rho)_g^w = [(S_{col}/\rho)w] / [(S_{col}/\rho)g]$ the mass stopping power ratio for wall and cavity gas.

In clinical dosimetry, when measuring the dose in a water phantom, the ionization chamber with tissue equivalent walls and the size of the small air cavity in relation to the secondary electron range are used.

In this way, the two BG conditions are met. First: the cavity must not disturb the charged particle fluence (including its distribution in energy) existing in the medium in the absence of the cavity, and second: the absorbed dose in the cavity to be imparted by charged particles crossing it. These two conditions represent the Bragg-Gray principle [3].

Evaluation of Photon Dose from the Particle Fluence and Stopping Power

In the situation where the gas in the cavity is air and the environment is water, the absorbed dose measured with chamber BG is given by the relation (52) in which we used the notations m and a,

$$D_m = D_a \frac{(S_{col}/\rho)m}{(S_{col}/\rho)a} = s_{m,a}\frac{W_a}{e}\frac{Q}{m}, \quad (53)$$

Where the mass stopping ratio for medium m, and air a is noted thus $S_{m,a} \equiv (S_{col}/\rho)_a^m = (S_{col}/\rho)_m/(S_{col}/\rho)_a$.

This is the Bragg-Gray relationship, in the case of the ionization chamber, having the sensitive volume filled with air. It should be mentioned that the dose calculation factor, $s_{m,a}$, depends only on the properties of interaction with the electrons of the environment and on the material in the ionization chamber cavity.

Using the ratios (28) for the absorbed dose in air, and (45) for the electronic stop power, we obtain the calculation relations for exposure rate, $\dot{X}$, in units (C/kg/s), or exposure X in units (R),

$$\dot{X} = \dot{\Phi}\cdot\frac{e}{W_{air}}\left(\frac{S_{el}}{\rho}\right)_{air} \text{ or } X = \Phi\cdot\frac{e}{W_{air}}\left(\frac{S_{el}}{\rho}\right)_{air}. \quad (54)$$

The exposure rate in X-ray units per second (R/s) is obtained by dividing the result obtained in C / kg·s by the factor
2.58×10^{-4} C / kg·R.

Protection Quantities for External and Internal Exposure

Mean Absorbed Dose in an Organ or Tissue, $D_{T,R}$

The definition relationship (ICRP, 1993b) for the Mean Absorbed Dose in an Organ or Tissue ($D_{T,R}$) from radiation R is

$$D_{T,R} = \frac{\bar{\varepsilon}_T}{m_T}, \quad (55)$$

Unde $\bar{\varepsilon}_T$ is the average total absorbed energy in the tissue or organ and m_T is the mass of the tissue or organ T. The unit of measurement in SI is gray (Gy), 1 Gy = 100 rads.

Equivalent Dose in Organ, H_T

The equivalent dose in any organ or tissue T of the human body, H_T, defined in ICRP 1991b Pub 60, is the product of the absorbed dose with the weighting factor for radiation of type R, w_R, i.e., $H_{T,R} = w_R D_{T,R}$. The unit dose equivalent to an organ or tissue is, J / kg, and has the special name of Sievert (Sv), after the name of Prof. Sievert. It was introduced in 1977. When several types of radiation are involved, the sum of their influence is made, and the total equivalent dose is obtained in each organ or tissue,

$$H_T = \sum_R w_R D_{T,R}. \quad (56)$$

Limit values for H_T: $H_T \leq 500$ mSv for all organs and tissues, except the lens of the eye; and $H_T \leq 150$ mSv, for the lens of the eye. H_T, $50 + H_{T,ext} \leq 500$ mSv.

Radiation Weighting Factors, w_R

The new "radiation weighting factor" (w_R) introduced in ICRP, Pub. 60, replaces the average qualiy factor $\bar{Q}$ from Publication 26. Its values mentioned in the three important ICRP publications published in the period 1977-2007, (1977, Pub 26 [15], 1979 Pub 30 [16], 1990 Pub 60 [17], 1991 Pub 61 [18], 1994 Pub 66 [19], and 2007 Pub 103 [20], are presented in Table 7.

Table 7: Recommended Weighting Radiation Factors

Radiation type	Publication 26-1977, $\bar{Q}$	Publication 60-1991, w_R	Publication 103-2007, w_R
Photons	1	1	1
Electrons and muons	1*	1[a]	1
Protons, energy > 2 MeV, others than those of recoil	10*	5	-
Protons and charged pions	-	-	2
Alpha particles, fission fragments, heavy ions	20	20	20
Neutrons of fissions	10	-	-
Neutrons, energy < 10 keV	-	5	A continuous function of neutron energy (see Fig. 6)
10 keV - 100 keV	-	10	
100 keV - 2 MeV	-	20	
2 MeV - 20 MeV	-	10	
20 MeV	-	5	

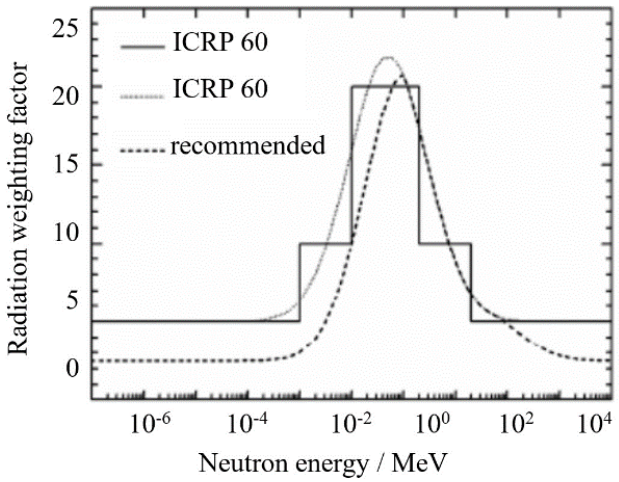

Figure 5: Radiation Weighting Factor, W_r, For Neutrons versus Neutron Energy

Weighting factors for neutron beams, w_R, depending on the neutron energy are shown in Figure 5 [20]. The values of w_R factors have been defined, based on the relative biological effectivness (RBE) for different radiations, considering their differences in producing stochastic effects.

Effective Dose, E

The effective dose, E, in Pub 103, is the sum of the weighted equivalent doses in all tissues and organs of the body. It is given by the expression

$$E = \Sigma_T \, w_T \, H_T, \qquad (58)$$

Tissue Weighting Factors, w_T

The tissue factor values for the three reference publications are presented in Table 8.

Table 8: Recommended Tissue-Weighting Factors

Weighting Factors			
Organ/Tissue	**w_T'' (ICRP 26)**	**w_T' ICRP (60)**	**w_T ICRP (103)**
Bone surface	0.03	0.01	0.01
Bladder	-	0.05	0.04
Brain	-	-	0.01
Breast	0.15	0.05	0.12
Colon	-	0.12	0.12
Gonads	0.25	0.20	0.08
Liver		0.05	0.04
Lungs	0.12	0.12	0.12
Esophagus	-	0.05	0.04
Red bone marrow	0.12	0.12	0.12
Salivary glands	-	-	0.01
Skin	-	0.01	0.01
Stomach	-	0.12	0.12
Thyroid	0.03	0.05	0.04
Remainder	0.30[a]	0.05[b]	0.12[c]

[a]5 additional organs and tissues; [b]10 additional organs and tissues; c13 additional organs and tissues for each sex.

The tissue weighting factor w_T'' from Publication 26 (1977), has the values given for 6 organs and tissues presented in Table 1 column 1, of which five of most organs and tissues are included in the rest, each having w_T'' = 0.06. This value is applicable to each of the five remaining organs or tissues (such as liver, kidneys, spleen, brain, small intestine, upper large intestine, lower large intestine, etc., but excluding skin, lens of the eye and the extremities) receiving the highest doses.

The factor w_T, normalized so that $\sum_T w_T = 1$, corresponds to the fractional contribution of the organ or tissue T depending on its sensitivity to the total risk of stochastic effects when the whole body is irradiated uniformly.

The weighting factor, w_T' from Publication 60 (1991), presented inTable 7, column 2, has values given for 12 organs and for other organs and additional tissues in number of ten: adrenal glands, brain, upper large intestine, small intestine, kidneys, muscles, pancreas, spleen, thymus and uterus. It should be noted that the values were developed for a reference population with an equal number of both sexes and a wide range of ages. When determining the effective dose, they apply to workers, the entire population, and either sex.

Weighting factor w_T from Publication 103 (2007) has the values given for 14 organs and tissues, presented in Table 2 column 3. The remaining tissues (14 in total, 13 for each sex) are: adrenal, extrathoracic tissue (ET), gallbladder, heart, kidneys, lymph nodes, muscles, oral mucosa, pancreas, prostate, small intestine, thymus spleen, uterus / cervix) (male prostate and uterus/cervix for women,). The weighting factors represent the relative contribution of the different tissues to the total risk for the case of uniform irradiation of the whole body.

The definition ratio (57) for the effective dose, according to the recommendations of Pub 103, becomes

$$E = \sum_1^{27} w_T \left[\frac{H_T^M + H_T^F}{2}\right], \qquad (59)$$

where the equivalent dose is evaluated for the organ or tissue T of the reference man, H_T^M and the reference woman,H_T^F, including the remaining tissues [21]. Relationship (59) can be applied in MIRD methodology [22].

Collective Dose, S

The dosimetric quantities for radiological protection presented above refer to a reference person. For radiation exposure of groups of occupationally exposed persons or of the public, to optimize and reduce radiation exposure, according to the principles of justification and optimization of protection, collective dose quantities have been introduced [17].

The Commission defined the quantity of "collective equivalent dose in T tissue", S_T, to express the total radiation exposure of a

specific organ or tissue in a group of individuals, by the expression

$$S_T = \sum_i \bar{H}_{T,i} \cdot N_i, \quad (60)$$

Where Ni is the number of individuals in the population subgroup i who receive the average equivalent dose per organ $\bar{H}_{T,i}$.

The radiation exposure of a population is given by the quantity of the "collective effective dose", S. This is calculated with the relation

$$S = \sum_j \bar{E}_j N_j, \quad (61)$$

Where $\bar{E}_j$ is the average effective dose per population subgroup j, and Nj represents the number of individuals in that subgroup. The name of the unit of these quantities is the man Sievert (person xSv). For the summed doses, the time and population for the summed doses must be specified.

Limits for Absorbed Dose in Air

Limit values for E

Occupational dose limit for whole body is 50 mSv per year effective dose, whole body limit for members of public is 1mSv per year effective dose, and single tissue or organ limit if not coverd separately is 50 mSv/y effective dose (max. 500 mSv/y to a single organ).

Dose Conversion Coefficients and Dose Coefficients

Conversion coefficients in external exposure and dose coefficients in internal exposure acts as a conversion unit from radiometric, dosimetric or activity quantities to radiological protection or operational quantities.

At external exposure, the expressions (57) and (58) divided by radometric, dosimetric and activity quantities become definition expressions for dose rate conversion coefficients, as $\dot{h}_E = w_T'' \cdot \dot{h}_T$, in Pub 26, $\dot{e}_E = w_T' \dot{h}_T$, in Pub 60 and $\dot{e}_E = \Sigma_T w_T \dot{h}_T$ in Pub 103. Here, we have: $\dot{h}_T$ is the dose equivalent rate conversion coefficient for organ or tissue T, $\dot{h}_E$ is the effective dose equivalent rate (EDE) conversion coefficient and $\dot{e}_E$ is is the effective dose rate conversion coefficient.

In the case of external exposure to radionuclides in air, water, and soil, the external individual effective dose rate coefficients for contaminated soil, contaminated air and contaminated water are determined by dividing the effective dose definition equation with the radionuclide concentrations in the respective environments. The air submersion dose rate coefficients given by equation din Pub 103, sunt $h_{T,ext}$ - the equivalent dose rate per unit air concentration of radionuclide and $\dot{e}_{E,ext}$ - the effective dose rate coefficient (Sv/s per Bq/m^3). The dose rate is average concentration × dose rate coefficient. The duration of exposure before exceeding dose limits is the ratio between dose limits for workers or adult and dose rate. Dose coefficient values are calculated and tabulated in [23].

At internal exposure, in the MIRD methodology the internal dose coefficient is obtained by dividing the absorbed dose to the cumulated activity, $\tilde{A}_S$, or or intake I, i.e., the activity that enters the body through the respiratory tract or the gastrointestinal tract or the skin.

In ICRP Metodology, the expressions for dose coefficients are $h_{E,50} = w_T''.h_{T,50}$, in Pub 26, e (50) = $w'_T \cdot h_T(50)$ in Pub 60 and e(50) = $\Sigma_T w_T h_T(50)$, in Pub 103. Hear we have $h_{E,50}$ (or $\dot{h}_{E,50}$) the effective dose equivalent coefficient, i.e., the committed effective dose equivalent (CEDE) per unit intake of radionuclides, and $h_{T,50}$ is the organ dose equivalent coefficient i.e., the committed dose equivalent to organ or tissue T per unit of activity of the radionuclide taken in by the specified route (see 6.4.2).

Dose conversion factors are tabulated for inhalation and ingestion for workers and public in Pub 68 (ICRP1994), Pub 72 (ICRP 1996a), and Pub 119, ICRP 2012, [24-26].

Relationship between E and H_E

Absorbed dose in air per unit photon fluence and Quotient of effective dose equivalent or dose equivalent in the specified organ or tissue to absorbed dose in air are presented in Table 9.

The 1982 UNSCEAR report, Annex A, presents a value of 0.7 Sv/Gy for the ratio of the effective dose equivalent to the dose absorbed into the air for men and women, based on calculations for gamma emitting clouds in the atmosphere with an mean energy of approximately 1 MeV, [27].

Table 9: Absorbed Dose in Air per Unit Photon Fluence [28].

Photon Energy (MeV)	Absorbed dose in air (10-16 Gy·m^2)	Quotient of effective dose equivalent or dose equivalent in the specified organ or tissue to absorbed dose in air (Sv·Gy-1)			
		Effective	Gonads	Tyroid	Skin
0.010	7.52	0.0022	0.0040	0.00043	0.194
0.015	3.09	0.0139	0.0136	0.00017	0.350
0.020	1.69	0.0543	0.0696	0.0326	0.442
0.030	0.695	0.227	0.2220	0.268	0.581
0.050	0.309	0.570	0.4290	0.603	0.759
0.065*	0.278	0.630	0.460	0.710	0.800
0.100	0.381	0.768	0.530	0.970	0.904
0.200	0.869	0.796	0.732	0.763	0.955
0.500	2.32	0.717	0.568	0.631	0.908
1.0	4.56	0.712	0.568	0.551	0.980
1.5	6.18	0.798	0.700	0.845	0.912
2.0	7.52	0.778	0.646	0.765	1.000
4.0	12.1	0.968	0.710	1.58	0.948
10.0*	23.2	0.968	0.710	1.58	0.948

The result of the comparison in Figure 6 [28] between the effective dose E and the equivalent of the effective dose H_E, in the geometry AP, shows the difference is less than 12% for all photon energies over 100 keV and for all irradiation geometries.

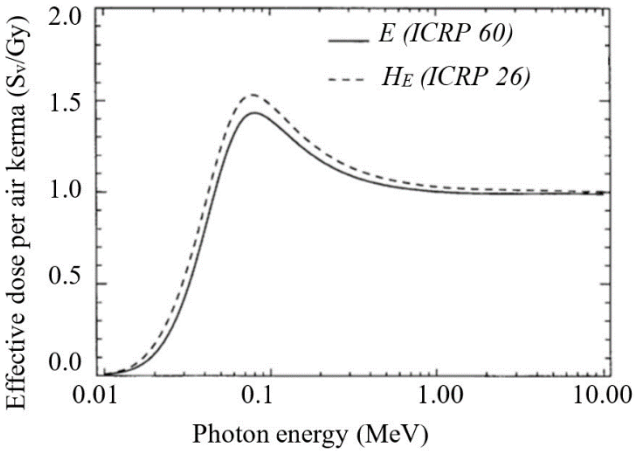

Figure 6: A comparison of conversion coefficients in anteroposterior irradiation (AP) irradiation geometry for effective dose, E, and the effective dose equivalent, H_E, per unit air kerma free in air as a function of photon energy

In rotational symmetry, the E/H_E ratio is between 0.95 and 1.00 for photonic energies greater than 100 keV.

Operational Quantities Replacing the Protection Quantities for External Exposure

Dose Equivalent and Quality Factor, H =QD

The quantity of the dose equivalent (H) was first defined in Publication 26 (ICRP, 1977) [15].

Publication 60 (ICRU 1991b) defined the dose equivalent as

$$H = D \cdot Q(L), \quad (62)$$

Where D is the absorbed dose at the point of interest in the tissue and Q (L) - the corresponding quality factor at this point, defined as a function of the linear, unrestricted energy transfer, L_∞ (often denoted L or LET), of the particles charged in water.

The values of the quality factor Q (L), as a function of linear transfer of energy in water, at the interaction point, are given in ICRU Pub 60 (ICRP, 1991a) through the following relations:

$$\begin{aligned} Q(L) &= 1 && (L<10), \\ Q(L) &= 0.32L - 2.2 && 10 \le L \le 100), \\ Q(L) &= 300/\sqrt{L} && (L>100), \end{aligned} \quad (63)$$

The relationship Q (L) - L was recommended in Pub. 60 for new calculations of the operational quantities H*, H ', and Hp. For the types of radiation and energy that are not included in ICRP Table 6 Pub 60, an approximation of w_R can be obtained by calculating the average quality factor $\bar{Q}$ at a depth of 10 mm in the ICRU sphere,

$$w_R = \bar{Q} = \frac{1}{D} \int_{L=0} Q(L) \cdot D_L \, dL, \quad (64)$$

Where $D_L = dD/d_L$ is the distribution of D by L for the charged particles that contribute to the dose absorbed at the point of interest.

The difference between $\bar{Q}$ and w_R is that while the former is a direct function of L, the latter is related to RBE.and is only indirectly related to mass stopping power.

Quantities for Environmental Monitoring and Individual Monitoring

Body protection quantities (equivalent dose and effective dose) are not measurable in practice and therefore cannot be used directly as radiation monitoring measures.

Operational quantities are used to evaluate the effective dose or equivalent doses to tissues and organs (Table 10) [20].

Table 10: Operational dose quantities for external exposure

Functions	Operational Dose Quantities	
	Area monitoring	Individual monitoring
Effective dose control	Ambient dose equivalent H*(10)	Personal dose equivalent $\mathbf{H_p(10)}$
Control of dose to skin, hands and feet	Directional dose equivalent H′(0.07,Ω)	Personal dose equivalent, $\mathbf{H_p(0.07)}$
Control of dose to the lens of the eye	Directional dose equivalent H′ (3, Ω)	Personal dose equivalent,$\mathbf{H_p(3)}$

Environmental Monitoring

Ambient dose equivalent, H(d)*

It is operational quantity for effective dose evaluation. The ambient dose equivalent, H*(10), at a point in a radiation field is the dose equivalent that would be produced by the field expanded and aligned accordingly in the ICRU sphere to a depth of 10 mm on the radius vector opposing the direction of the aligned field, defined by ICRU, 2001b.

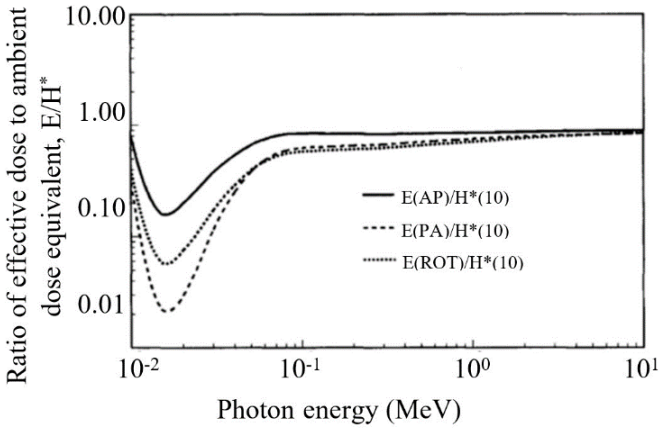

Figure 7: The ratio of E/ H*(10), for various irradiation geometries as a function of photon energy [28].

Figure 7 [27] shows that for energies between 60 keV and 10 MeV, the value of the E / H * (10) ratio varies from 0.75 to 0.92 for the AP geometry and for 0.48 to 0.85 for the ROT geometry. In these areas, the equivalent of the ambient dose indicates an overestimation of the effective dose exceeding 15%.

At low energies, there is a significant overestimation. At 25 keV the E / H * (10) ratio is a factor of approximately 3 for AP geometry and close to an order of magnitude for ROT and PA geometries.

The dose to the skin and the dose to the lens of the eye are of great importance in radiological protection, and in this situation, the quantities E and H*(10) have a practically limited application.

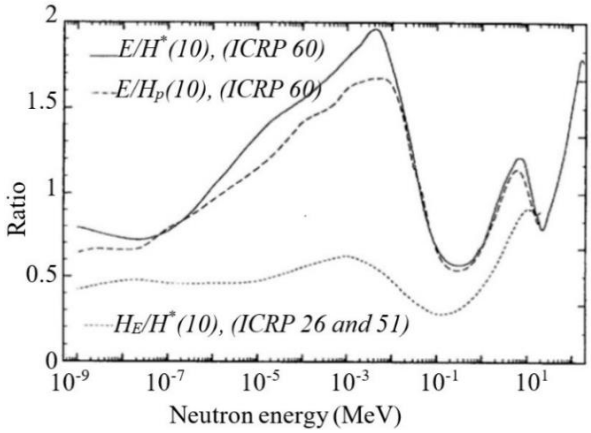

Figure 8: The ratio of E/H*(10), E/HP (10) and HE/H*(10), as a function of neutron energy

The ratio of the effective dose to the ambient dose equivalent, E/H*(10), in area monitoring, and the effective dose per the personal dose equivalent HP (10) from individual monitoring as a function of neutron energy, are shown in Fig.8.

Also, the ratio of the effective dose equivalent to the ambient dose equivalent, HE/H*(10), (ICRP26 / ICRP51), as a function of neutron energy, are shown in Figure 8 [28].

Directional dose equivalent, H'(d, Ω)
For area monitoring of weakly penetrating radiation, H' (d, Ω), the operational quantity is the directional dose equivalent, H' (0.07, Ω) and H'(3, Ω).

These operational quantities are defined as follows: The directional dose equivalent, H'(d, Ω), at a point in the radiation field is the dose equivalent that would have been produced by the corresponding expanded field in the ICRU sphere, at a depth, d, on a ray with a specified Ω direction.

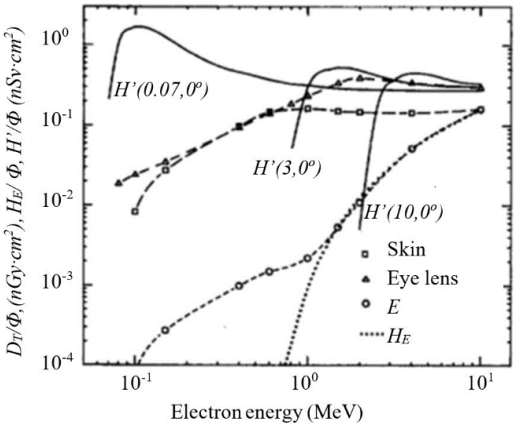

Figure 9: Protection and Operational Quantities for Electrons
The quantities shown in Fig. 9 [27] are: the effective dose, E, the effective dose equivalent, H_E, the organ absorbed dose, D_T, for the lens of the eye and skin, and the directional dose equivalent, H'(d,00), at depths, d, of 0.07, 3, 10 mm, as a function of electron energy (MeV) [29]

Table 11: Conversion coefficient for photons [29]

Photon energy (MeV)	Conversion cefficient, 10^{-12} Sv .Gy^{-1}		
	0.07 mm	3 mm	10 mm
1.0 x 10^{-2}	0.930	0.271	0.010
1.5 x 10^{-2}	0.974	0.686	0.271
2.0 x 10^{-2}	1.02	0.917	0.601
3.0 x 10^{-2}	1.19	1.19	1.09
4.0 x 10^{-2}	1.38	1.42	1.43
5,0 x 10^{-2}	1.52	1.59	1.63
6.0 x 10^{-2}	1.58	1.67	1.74
1.0 x 10^{-1}	1.55	1.60	1.65
5.0 x 10^{-1}	1.21	1.22	1.21
1.0 x 10^{0}	1.16	1.16	1.14
5.0 x 10^{0}	1.59	1.12	1.11
1.0 x 10^{1}	1.63	1.11	1.09

Dose equivalent at various depths on the principal axis of the ICRU sphere per unit-absorbed dose to air in free for a photon incident in a plane parallel beam is given in Table 11.

Table 12: Conversion cefficient for neutrons [29]

Neutron energy, MeV	Conversion cefficient, 10^{-12} Sv . cm^2		
	0.07 mm	3 mm	10 mm
2.5 x 10^{-8}	2.70	2.70	2.10
1.0 x 10^{-7}	2.80	3.10	2.90
1.0 x 10^{-6}	2.70	2.80	3.30
1.0 x 10^{-5}	2.30	2.10	2.80
1.0 x 10^{-4}	1.40	1.80	2.40
1.0 x10^{-3}	1.30	1.50	2.00
1.0x10^{-2}	4.30	4.20	3.10
5x10^{-1}	134	117	84.8
1	184	165	135
5	259	252	239
10	281	303	300
20	400	510	500

Dose equivalent per unit fluence at depth 0.07 mm, 3mm and 10 mm for neutron's incident an isotropic field on the ICRU sphere ICRP, is presented in Table 12.

Individual Monitoring

Personal Dose Equivalent, Hp (d)

The operational quantities for the evaluation of the effective dose in individual monitoring at external exposure is the personal dose equivalent Hp (d). The specified point is given by the position where the personal dosimeter is worn. For the evaluation of the ef-

fective dose a depth d = 10 mm, Hp (10) is recommended. For the personal dose equivalent, the SI unit is Joule per kilogram ($J \cdot kg^{-1}$) with the special name sievert (Sv).

Superficial Individual Dose Eqhivalent

The evaluation of the equivalent dose to the skin, hands and feet, $H_p(0.07)$, is made at a depth d = 0.07 mm. For monitoring the dose to the lens of the eye, the depth is d = 3 mm.

Figure 10 shows the dose absorbed in the organ, D_T, per unit of air kerma Ka, for the eye lens depending on the energy of the photon, in units (Gy / Gy) using irradiation geometries: antero-posterior - AP, postero-anterior - PA, right lateral - LAT, rotational - ROT and isotropic - ISO, [29].

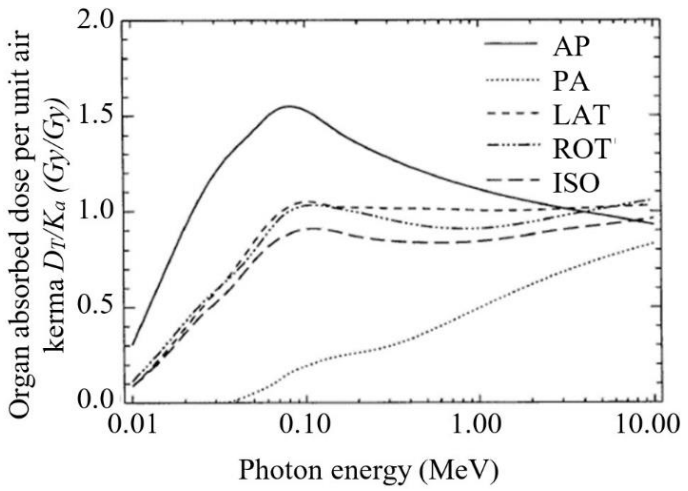

Figure 10: Eye lens absorbed dose per unit air kerma free in air, [29]

Individual dose limits Nominal risk coefficient

Radiation exposure of the human body produces side effects. They are grouped into two categories: deterministic effects at high doses, with threshold above 100 mSv, and stochastic effects at low doses below about 100 mSv, no-thereshold, for which exists a linear dose–effect relationship. At doses higher than 100 mSv there is an increased probability of deterministic effects and a high risk of cancer.

As an example, we mention the estimates for the threshold dose received in a single and short exposure for several tissues / effects of an adult person: testicle / temporary sterility: 0.15 Gy; testicle / sterility: 3–6 Gy; ovaries / sterility: 2.5-6 Gy; critalin / detectable opacities: 0.15-2 Gy; critalin / cataract: 5 Gy.

Also, exist the treatment of moderately radiosensitive cancer: 25-70 Gy, local, 2 Gy/day. The dose-survival relationship is described by its inflection point, $LD_{50/60}$, i.e., the dose at which half the number of individuals would be expected to die in 30-60 days. For a healthy adult human, it is estimated that $LD_{50/60}$ after acute exposure is between 3 and 5 Gy, dose at the midpoint.

The ICRP 2007, Pub 103 recommends a limit on effective dose of 20 mSv per year, averaged over 5 years (100 mSv in 5 years), with the further provision that the effective dose should not exceed 50 mSv in any single year.

Dose limits recommended in the Pub. 60 and Pub 103 are presented in Table 13 [20].

Table 13: Comparison of protection criteria between the 1990 and the 2007

Planned exposure situations	ICRP 1990 Pub 60	ICRP 2007 Pub 103
Individual dose limits		
Occupational exposure	20 mSv/y	20 mSv/y average on 5y
lens of the eye	average on 5y	150 mSv/y
skin	150 mSv/y	500 mSv/y
hand and feet	500 mSv/y	500 mSv/y
pregnant women	500 mSv/y	1 mSv to the
remainder of pregnancy	2 mSv abdomen 1mSv for intake	embryo/fetus
Public exposure	1 mSv in a year	1 mSv in a year
lens of the eye	15 mSv/y	15 mSv/y
skin	50 mSv/y	50 mSv/y

Individual dose limits are established to reduce stochastic effects (cancer sau efecte ereditare). The current radiological protection system is based on dose-response relationship of type “linear no-threshold” (LNT) model.

According to this model, at doses below 100 mSv, a given increment in dose will produce a directly proportionate increment in the probability of incurring cancer or hereditary effects attributable to radiation [20].

The number of the radiation induced fatal cancer cases, $dN_r(t)$, is directly proportional to the total number, $N_0(t)$, of members of the group exposed to ionizing radiation, consider living at time t, with the time interval dt and with a parameter $\lambda_r(t)$, i.e., $dN_r(t) = N_0(t)\ \lambda_r(t)\ dt$. The number $dN_S(t)$ of spontaneous fatal cancer cases is defined by analogy with the last equation as $dN_s(t) = \lambda_S(t)N_0(t)dt$. The parameters $\lambda_r(t)$ and $\lambda_s(t)$ represents the mortality rates, defined as follows

$$\lambda_r(t) = \frac{1}{dt}\frac{dN_r(t)}{N_0(t)} \qquad \lambda_s(t) = \frac{1}{dt}\frac{dN_s(t)}{N_0(t)}. \tag{65}$$

The linear dose-effect relationships, $\lambda_r(t) = a\ D(t)$, given by the LNT model, allows us to define absolute risk coefficient, $R_a = d\lambda_r(t) / dD = a$. The nominal absolute risk coefficient, Ra, is the probability of the occurrence of a stochastic effect per dose and per time or age interval. In the “absolute” or “additive” risk model, the excess probability rate is dose - dependent and ange- independent. The relative risk coefficient is defined as $Rr,s = d\lambda_r(t)/ \lambda_s(t)\ dD$. The calculation expressions for the two risk coefficients are

$$R_a = \frac{1}{Ddt}\frac{dN_r(t)}{N_0(t)}; \qquad R_{r,s} = \frac{dN_r(t)}{D\, dN_S(t)}. \qquad (66)$$

In the "relative" or "multiplicative" risk model, the excess rate increases with age at the same speed as the spontaneous rate of cancer due to background ($\lambda_S(t)$). These coefficients were called "detriment", [15], [17].

The term "detriment" represents the combination between the probability of occurrence of the stochastic effect and the appreciation of the severity of this effect. Detriment, known as "detriment - adjusted nominal risk coefficients", is the product of nominal risk coefficient Ra and detriment quality d [20], [30].

Detriment quality d, independent of radiation exposure, is a weighting factor for the radiation risk, related to the severity of the stochastic effect. It includes the lethality of the disease, the loss of quality of life, and the relative loss of life expectancy due to the disease.

The values for "detriment -adjusted nominal risk coefficients", based on the cancer incidence weighted data for lethality and deterioration of life, for both an active population and a general population, are given in Table 14 [20].

Table 14: Detriment-adjusted nominal risk coefficients at low doses and dose rates

Exposed population	Cancer		Hereditary effects		Total	
	ICRP	ICRP	ICRP	ICRP	ICRP	ICRP
	103	60	103	60	103	60
x 10^{-2} Sv^{-1}						
Whole	5.5	6.0	0.2	1.3	5.7	7.3
Adult	4.1	4.8	0.1	0.8	4.2	5.6

From Table 14, it results that the nominal risk coefficients for detriment - corrected for cancer are 5.5 x 10^{-5} per mSv for the entire population and 4.1 x 10^{-5} per mSv for adult workers. The probability coefficients corrected for detriment for hereditary diseases up to the second generation are 0.2 x 10^{-5} per mSv for the entire population and 0.1 x 10^{-5} per mSv for adult workers.

Protection Quantities Based on the Nuclear Decay Law for Internal Exposure

Internal dose

Internal dosimetry methodologies began with the single-compartment model ICRP 2/10, 1959 [31]. The methodologies of MIRD, [32-35], and Pub 26/30 (ICRP 1977) [15], [16] have developed a system for calculating the average absorbed dose expressed in rads at a target organ r_T from a radionuclide uniformly distributed in a source organ r_S. In the ICRP methodology, for the source organ and the target organ the notations (T, S) are used, and in the MIRD methodology - the notations (r_k, r_h) [33] or (r_T, r_S) [22]. There are several standard relationships between the source organ (S) and the target organ (T) in the human body. These can be the following: S identical to T, i.e., the whole body, T in the center and the source S radiates concentrically, S in the center and T concentrically, and S separate from T [36].

Internal Dose Assessment. ICRP 2/10

Internal dose is defined as dose from radiation released by radioactive materials deposited inside the body. The alpha, beta and gamma particles emitted by the organ source having different ranges due to their energy, can be absorbed inside the volume of the source or outside its volume.

The radioactive activity in the organ of interest is A (t) = A (0) exp ($-\lambda_e$ t), post uptake, where the initial activity in the organ A (0) is the initial intake activity of radionuclide in the body and $\lambda_e = \lambda_p + \lambda_b$ is the effective decay constant $\lambda_e = \lambda_p + \lambda_b$, with λ_p - the physical decay constant and λ_b - the biological decay constant.

The initial dose equivalent rate $\dot{D}_0$ to the organ of interest is

$$\dot{D}_0 = k\, A(0)\frac{E}{m}, \qquad (67)$$

where E = effective energy deposition, m = mass of the organ and k = constant of units. If A (0) is expressed in μCi, E in MeV, and m in grams, if D˙_0 is expressed in rad / h, the constant k = 2.13 (rad -g-dis / h-MeV-μCi).

The dose rate as a function of time t is written in terms of the initial absorbed dose rate:

$$\dot{D}(t) = \dot{D}_0 e^{-\lambda_e t}. \qquad (68)$$

The equivalent of the total dose accumulated in the organ of interest from a single intake at t = 0 to any post uptake time is obtained by integration from t = 0 at time T. After performing the integration in relation (67) the law of accumulation of dose absorbed for the T-assessment period is

$$D = k\,\frac{A(0)E}{m\lambda_e}\left(1 - e^{-\lambda_e T}\right)\phi, \qquad (69)$$

Where ϕ is the energy-absorbed fraction.
When t → ∞ for short-lived radionuclides used in nuclear medicine, exp (- λe t) → 0, and the total dose due to the intake is given by the fundamental ICRP-2/10 relationship:

$$D = 2.13\,\frac{A(0)(\mu Ci)}{m_T\,(g)\lambda_E\left(\frac{1}{h}\right)}\left[\Sigma_i E_i\left(\frac{MeV}{dis}\right)Y_i\right]\phi. \qquad (70)$$

Where DT is dose in rads, constant 2.13 represents (rad-g-dis / h-MeV-μCi), A (0) in (μCi) is initial activity in the organ T, m (g) - mass of organ, ϕ - absorbed fraction, i - number of photon emitted, E_i - mean photon energy (MeV) of the i-th photon, Yi - yield of the i-th photon of energy E_i, and $\lambda_{eff} = \ln 2 / T_{eff}$ is effective half- life (hours) [30].

Internal Dose Estimation using MIRD Methodology

The MIRD methodology refers to photons, electrons and positrons, which have the quality factor Q = 1 and the radiation weighting factor w_R= 1 and H (Sv) = D(Gy). The MIRD equation is based on two main parameters: the biological parameter $\tilde{A}$, the cumulated activity in the organ source, r_S, and the value of the physics parameter S, determined by the nature of the radiation and its absorption characteristics in the target organ, r_T.

The energy emitted by the radionuclide incorporated in an organ is calculated for each discrete nuclear transition, as: $\triangle_i = k_{ni}\ E_i$, where E_i is the average particle energy emitted per nuclear transition, ni represents the number of particles emitted per nuclear transition, k - a conversion factor, i is index to designate type of radiation transition. The fraction of the emitted energy that contributes to the dose absorbed in the target is ϕ_i. The product of the quantities $\tilde{A}_S$, $\triangle_i$ and $\Phi_i = \phi / m_T$, represents the energy absorbed in the target mass m_T.

The average absorbed dose $\bar{D}$ ($r_T \leftarrow r_S$) expressed in rads (1974), at a target organ r_T from a radionuclide uniformly distributed in the source organ rS, was formulated by the MIRD Committee, as follows:

$$\bar{D}\ (r_S \leftarrow r_T) = \frac{\tilde{A}_S}{m_T}\ \Sigma_i \triangle_i\ \phi_i\ (r_T \leftarrow r_S) = \tilde{A}_S\ \Sigma_i \triangle_i\ \Phi_i\ (r_T \leftarrow r_S) = \tilde{A}_S\ S(r_T \leftarrow r_S)\ , \quad (71)$$

where $\tilde{A}_S$ (μCi—h) is activity in organ source r_S, m_T (g) is the mass of the target organ T, $\triangle_i$ (g-rad/μCi-h)) is the equilibrium dose constant for particles of a certain type and energy, indicated here by i, ϕ_i ($r_T \leftarrow r_S$) and $\Phi i(r_T \leftarrow r_S)$ represents, respectively, the absorbed fraction and the absorbed fraction specific to the energy for the target organ r_T, for the particles i emitted in the source organ r_S, and the factor

$$S(r_T \leftarrow r_S) = \Sigma_i \triangle_i\ \Phi_i\ (r_T \leftarrow r_S) = = \bar{D}\ (r_T \leftarrow r_S)\ /\ \tilde{A}_S), \quad (72)$$

is the mean absorbed dose to target organ r_T per unit cumulated activity in source organ r_S.

The average dose for a radionuclide is simple to calculate

$$\bar{D}\ (r_T \leftarrow r_S) = \tilde{A}_S\ S\ (r_T \leftarrow r_S). \quad (73)$$

In the case of several organ sources the total average dose is

$$\bar{D}\ (r_T) = \sum_h \bar{D}\ (r_T \leftarrow r_S) = \sum_h \tilde{A}_S\ S\ (r_T \leftarrow r_S), \quad (74)$$

Cumulative activity at source ($\tilde{A}_S$) is a biological factor and S factor is a physical factor because it depends on physical and anatomical data. The S factor is tabulated for radionuclides, source organs and target organs.

It should be noted that the values of the absorbed fractions, ϕ, were calculated for a large number of radionuclides and organs by Snyder et al., in 1969 [37]. The equilibrium dose constant, $\triangle_i$ (g-rad / μCi-h), which gives the absorbed dose rate, was calculated by Dillman (1969), on the assumption that all the emitted energy is locally absorbed, for most radionuclides of interest in nuclear medicine, [38].

Absorbed Dose Estimation using ICRP Methodology

The ICRP methodology takes into account the time variation of the activity in the source organ, by replacing the terms of effective dose and equivalent dose, with the committed equivalent dose in the organ and the committed effective dose in the whole body, during the working periods in radiation field, 50 years for adults and 70 years for children.

Committed Equivalent Dose, H_T (50)

ICRP Publication 26, using Qi quality factor values that are constant for any type of radiation i, defines the committed dose equivalent (H_{50}) for a period of 50 years followed by incorporation, thus $H_{50} = \sum_i Q_i\ \bar{D}_{50}$. Here $\bar{D}_{50}$, i is the total absorbed dose during the 50 years after the intake of the radionuclide into the body averaged throughout the specified organ or tissue for each radiation of type i.

ICRP Publication 30 introduces the target organ (T) and the source organ (S). The committed dose equivalent H_{50} in the target organ, expressed in Sv, for all types of radiation i emitted by radionuclide j is given by the expression

$$H_{50}\ (T \leftarrow S)_i\ = 1.6 \times 10^{-10}\ [U_S(50) \sum i\ SEE(T \leftarrow S)_i]j, \quad (75)$$

where Us is the number of radionuclide transformations in S over the 50 years following intake of the radionuclide (Bq-s) and SEE $(T \leftarrow S)_i$ (in MeV / g / transformation) is the specific effective energy for type i radiation, modified accordingly by the quality factor Qi, absorbed in T for each transformation in S, expressed as (Sv/(Bq-s) and $1.6 \cdot 10^{-10}$ represents Sv-g/MeV.

For any radionuclide j, SEE (T ← S) j for target T and source S, in conventional units MeV / (g-t), summed over all the radiations produced by the transformation of radionuclide j into the source organ S, is given by the expression

$$SEE\ (T \leftarrow S)j\ = \Sigma i\ \frac{E_i\ Y_i\ Q_i\ AF(T \leftarrow S)_i}{m_T}, \quad (76)$$

where Yi is the yield of radiations of type i per transformation of radionuclide j; Ei (in MeV) is the average or unique energy of radiation i as appropriate; AF $(T \leftarrow S)_i$, is the fraction of energy absorbed in target organ T per emission of radiation i in S, Q_i is the appropriate quality factor for radiation of type i and m_T (in g) is the mass of the target organ.

Finally, target T may be irradiated by radiations arising in several different sources S, the total value of H_{50}, in target T is then given

$$H_{50,T}\ = 1.6 \times 10^{-10}\ \Sigma_S\ \Sigma_j\ [U_S\ \Sigma_i SEE\ (T \leftarrow S)_i i]_j. \quad (77)$$

ICRP Publication 60, replaced the term "committed dose equivalent" $H_{50,T}$ with "committed echivalent dose" $H_T(50)$, "committed

effective dose equivalent" $H_{E,50}$ with committed effective dose E (50)) and quality factor Q with weighting factor for radiation w_R.

Committed Effective Dose (Whole Body), E(50)

The effective dose commited, E (50), from radionuclide incorporations, is evaluated with the equation

$$E(50) = \Sigma_n e_{n,inh}(50) I_{n,inh} + \Sigma e_{n,ing}(50) I_{n,ing}, \quad (78)$$

where $e_{n,inh}$ (50) is the committed effective dose coefficient for activity intakes by inhalation of a radionuclide n, In,inh is the activity intake of a radionuclide n by inhalation, en,ing (50) is the committed effective dose coefficient for activity intakes of a radionuclide n by ingestion, and In,ing is the activity intake of a radionuclide n by ingestion [20].

The e(50) values are the sex-mediated effective dose coefficients (female, male) for the incorporation of the specified radionuclides and they are calculated using the relation:

$$e(50) = \sum_{T=1}^{T=27} w_T \left[\frac{h_T^M(50) + h_T^F(50)}{2}\right], \quad (79)$$

Where w_T is the tissue-weighting factor for T tissue, and h^M_T (50) and h^F_T(50) are committed equivalent dose coefficients for T tissue, male or female, respectively. The amount (70) also includes the rest of the tissues in man and woman (ICRP Pub 103).

The e(50) values are the sex-mediated effective dose coefficients (female, male) for the incorporation of the specified radionuclides and are calculated using the relation:

$$E = e(50)*\text{Intake}, \quad (80)$$

Where the intake can be determined from the measured quantity, M, by:

$$\text{Intake} = \frac{M(t)}{m(t)}. \quad (81)$$

Here M (t) = the value measured at time t of activity in an organ, in the whole body, or in daily excretions [Bq or Bq-d^{-1}] and m (t) = the value estimated at time t of activity in an organ, throughout the body or in daily excretions [Bq or Bq-d^{-1}]. Details in [39-41].

Derived Quantities for Internal Dose Control

Annual Limits on Intake, ALI

It is the basic quantity for the control of internal exposure. For the control of internal exposure to radionuclides at work there are two derived quantities: the Annual Limit on Intake (ALI) and the Derived Air Concentration (DAC). These were defined in Published 60 (ICRP, 1991b). ALI as the incorporation of the activity (Bq) of a radionuclide that would lead to an effective dose corresponding to the annual limit of the effective dose, E_{limit}, assuming that the worker is exposed only to this radionuclide. ALI for radionuclide j is:

$$ALI_j\ (Bq) = \frac{E_{lim}(Sv)}{\sum w_T h_{T,50}\left(\frac{Sv}{Bq}\right)} = \frac{0.02 S_v}{e(50)} \quad (82)$$

where e(50) is the committed effective dose coefficient in Sv / Bq. The Commission recommended in Pub 60 that ALI be based on a dose limit of 0.020 Sv.

Derived Air Concentration, DAC

DAC represents the level of radioactivity inside the body. A Derived Air Concentration (DAC) is defined as that concentration of radionuclide in air, which, if breathed by Reference Man (ICRP 1975) for a work-year, would result in the intake of one ALI. The unit of measurement in SI is Bq/m^3.

The concentration of a radionuclide in air is limited by

$$\int C(t)B \leq ALI, \quad (83)$$

where C (t) is the concentration of the radionuclide in air at time t, B is the volume of air breathed by a worker per unit time, and the integration is carried out over the 2000 working hours per year (= 40 h / week x 50 weeks/y).

For the special case of constant air concentration, the DAC is related to the ALI (Bq/y) through

$$DAC\ (Bq/m^3)\ \frac{ALI\ (Bq)}{2{,}000h\ .1.2\ m^3/h} = \frac{ALI\ (Bq)}{2{,}400\ m^3}, \quad (84)$$

Based on a normal breathing rate B of 20 ℓ/min (= 0.020 m^3/min x 60min/h x 40h/w x 50w/y = 2400 m^3/y. DAC for inert gases that are not incorporated is limited to the effective dose, E_{limit} , given by the incident radiation to the body surface from the airborne activity. Thus, DAC is given by

$$DAC\ (Bq/m^3) = \frac{E_{limit}}{2{,}000\ \Sigma_T w_T \dot{h}_T} = \frac{E_{limit}}{2{,}000\ \dot{e}_{sub}}, \quad (85)$$

Where $\dot{e}_{sub}$ is the effective dose rate coefficient (mSv/h per Bq/m^3), that is, the effective dose per unit time-integrated exposure to a radionuclide, for submersion in a cloud containing the noble gas radionuclide and 2000 h is the annual working time, w_T is the tissue weighting factor, and $\dot{h}_T$ is the equivalent dose rate coefficient (Sv/s per Bq/m^3) for tissue T. Doza efectiva externa se calculeaza cu relatia, E_{sub} (Sv) = 2000 $\dot{e}_{sub}$ DAC. The reference person effective dose rate coefficients for air submersion sunt tabelati.

Secular Equilibrium and Transient Echilibrium

When a radioactive atomic species A gives birth, by its disintegration, to an atomic species B, we say that the two radioisotopes are "genetically linked." The first is called a "generator" and the second is a radioactive "derivative".

Radon 222 (T½ = 3.823 d, ρ = 0.00973 g/cm_3), derived isotope, is in secular equilibrium with the generating isotope, radium 226 (T½ = 1600 y).

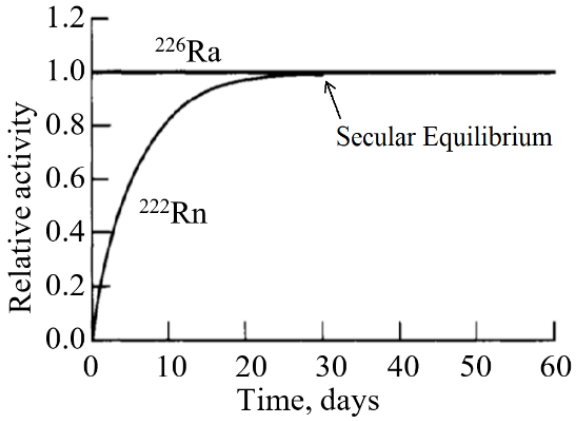

Figure 11: Echilibrium secular: the growth of Radon ($_{86}^{222}$)Rn, $T_{2,1/2}$ = 3.8 d) into a sample of Radium ($^{226}_{88}$)Ra Ra, $T_{1,1/2}$ =1600 y)

Radon after about 7 half-lives has the activity equal to that of radium, i.e., the number of radionuclides that disintegrate in a unit of time from radium is equal to the number of newly formed nuclides at the same time of radon, $\lambda_1 N_1 = \lambda_2 N_2$, Figure 11 [12].

Equilibrium factor F is the ratio of the equilibrium equivalent concentration C_{eq} to the radon activity concentration C_{Rn}. This factor describes the ratio of potential alpha energy concentration (PAEC) in a given compound of Rn-222 and its short-lived decay products to the PAEC, as would be the case in radioactive equilibrium [30].

The fifth characteristic of radioactive decay is the transient (regime) equilibrium between the activities of two genetically linked radionuclides, when $T_1 > T_2$ or $\lambda_1 < \lambda_2$. Fig.12 shows the transient equilibrium between the radionuclide Te-132 ($T_{2,1/2}$ = 76.8h) and the radionuclide I-132 ($T_{2,1/2}$ = 2.28h) genetically linked.

When there is equality between the activities of the two radionuclides, it is said that the two nuclides are in transient equilibrium, and the activity of the derivate acording to relation

$$A_1 T_1 = A_2 (T_1 - T_2) = const\ , \qquad (86)$$

is higher than that of the parents by the factor $T_1 / (T_1 - T_2)$. This is done because the meeting time of the two disintegrations exceeded the time when the derivat activity reached its maximum value [15].

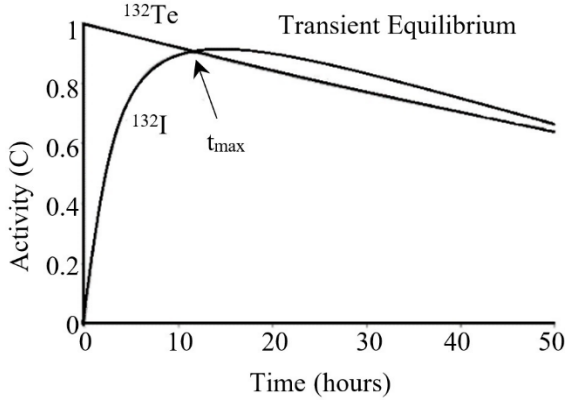

Figure 12: Transient equilibrium: Activity of 132I ($T_{2,1/2}$ = 2.28h) produced from initially pure 132Te ($T_{2,1/2}$ = 76.8h).

Exposure to Radon, WL and WLM

Radon is a noble, radioactive gas from the decay of radium, the latter coming from the uranium decay series. Radon from the atmosphere reaches the body through inhalation. Radon, present in the air about 40%, inhaled into the lung diffuses into the blood through which it reaches all other organs [42]. Radon decay products are presented in Table 15 [3].

Table 15: Radiation from Radon and its Decay Products

Radionuclide	Half-life	Major radiation energies in MeV and (percent of disintegrations)		
		α	β	γ
Radon-222	3.823day	5.49↑		
Polonium-218	3.03 min	6.00↑		
Lead-214	26.8 min		0.65-0.98	0.295(19)
				0.352(36)
Bismuth -214	19.7 min		1.0-3.26	0.609(17)
				1.120(17)
				1.764(17)
Polonium-214	164 μs	7.69↑		
Lead-210	21 y		0.016-0.061	
Bismutth -210	5 d		1.161(100)	
Polonium-210	138 d	5.31↑		
Lead-206	Stable			

The calculation of the concentration of alpha energy absorbed in the lung, with the data from Table 15, they show that the total energy of alpha particles associated with the decay of 3.7 Bq / ℓ of short-lived radionuclides accumulated in the lung is 1.3 x 10^5 MeV / ℓ = 2.08 x 10^{-5} J / m^3 = 0.02 mJ / m^3.

Based on these data, two quantities were introduced for radon exposure: Working Level (WL) - a measure of concentration in air, originally defined as 100 pCi /ℓ = 3,700 Bq· m^{-3}

$$1\ \text{WL} = 2.08 \times 10^{-5} \frac{\text{J}}{\text{m}^3} = 3{,}700 \frac{\text{Bq}}{\text{m}^3}, \qquad (87)$$

And Working Level Munth (WLM) - is a unit for the radon exposure a worker receives during a month (170 working hours) at 1 WL

$$1\ \text{WLM} = 3.54\ \text{mJ}\ \frac{\text{h}}{\text{m}^3} = \frac{0.64}{\text{F}}\ \text{MBq}\ \frac{\text{h}}{\text{m}^3}, \qquad (88)$$

Where F is the equilibrium factor. 1 WLM corresponds to a radon equilibrium equivalent concentration of 0.64 MBq h m^{-3}; 1 mJ h / m^3 = 0.282 WLM.

The conversion from radon exposure to effective dose, E, by convention, is given by egality between the detriment per unit exposure to radon progeny (mJhm^{-3}), and the total detriment per unit effective dose (mSv) [43].

The nominal probability coefficient (fatality) for males and females of 8.0 x $l0^{-5}$ per (mJ h m-3) was adopted in ICRP 65. It represents cases /10^6 person – years per unit of exposure.

The values for effective dose coefficients are obtained from the relation of equality of detriment. For the total detriment per unit effective dose of 5.6 x 10^{-5} per mSv for workers and 7.3 x 10^{-5} per mSv for public (Pub 60) are obtained the following dose coefficients: 1.43 mSv/ (mJ h m^{-3}) and 5.06 mSv/WLM for workers; 1.10 mSv/mJhm^{-3} and 3.88 mSv/WLM for members of the public.

The nominal risk coefficients, detriment coeficients for effective dose and the dose coeficients, and are presented in Table 16 [30].

Table 16: Dose coefficients for Radon based on the various ICRP Publications [30]

Risk coefficient	Category	Detriment 10^{-2}/Sv		Dose coefficient F (mSv/ MBq•h•m-3	Dose coefficient (mSv/ WLM)
2.83.10^{-4}	Public	7.3	ICRP	6.3	4
(ICRP 65)	Occupational	5.6	60	7.8	5
5.0.10^{-4}	Public	7.3	ICRP	10.9	7
(ICRP 115)	Occupational	5.6	60	14.1	9
5.0.10^{-4}	Public	5.7	ICRP	14.1	9
(ICRP 115)	Occupational	4.2	103	18.8	12

Annual effective dose of 10 mSv for Radon-222 has been established in Pub 65 [42]. The optimization of protection must ensure the annual levels of action: 3 –10 mSv (200 - 600 Bq/m3) in homes and 3-10 mSv (500 -1500 Bq/m3) at work [20], [43-45].

Occupational and Public Exposure

Currently, people are exposed to natural radiation (85%, consisting of 50% Radon gas from the ground, 14% gamma rays from the ground, biuldings, 10% cosmic rays and 11.5% from food, drink, air) and artificial radiation (15% composed of 14% medical, < 0.1 % nuclear discharges, < 0.1 % products, 0.2% fallout and 0.3% occupational, Fig. 13, [46].

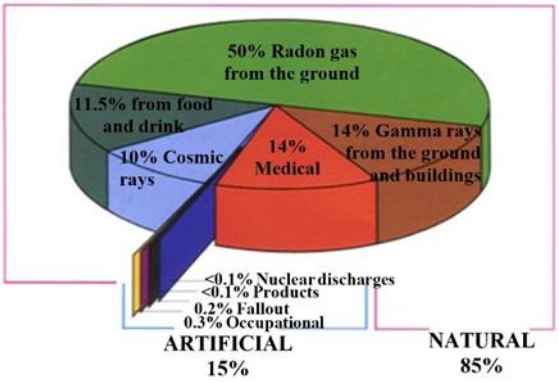

Figure 13: Radiation exposure of man from natural sources about (85%) and from artificial radiation sources, (15%) [46]

It can be said that radiation dosimetry has its origins in the medical applications of ionizing radiation that began after the discovery of X-rays

Occupational Exposure Assessment

When monitoring occupational exposures to external radiation, individual dosimeters measure the dose equivalent of the whole body, Hp (10). For internal exposure, the committed effective dose, E (50), is determined from the incorporation activities of the radionuclides with the equation (64). The total annual effective dose, E, from all sources (mSv), is given by the relationship (89),

$$E = H_p(10) + E(50) + Esub(DAC) + E(mSv \leftarrow WLM) \quad (89)$$

Where H_p is the personal dose equivalent for external exposure, or the external effective dose, E(50) is the committed effective dose from internal exposure i.e., the internal effective dose and E (WLM) is the exposure to radon progeny expressed in Working Level Months (WLM). In certain situations, for example, exposure of aircraft crews, in which individual monitoring with personal dosimeters cannot be performed, an assessment of the effective dose can be obtained from the values of the ambient dose equivalent, H*(10).

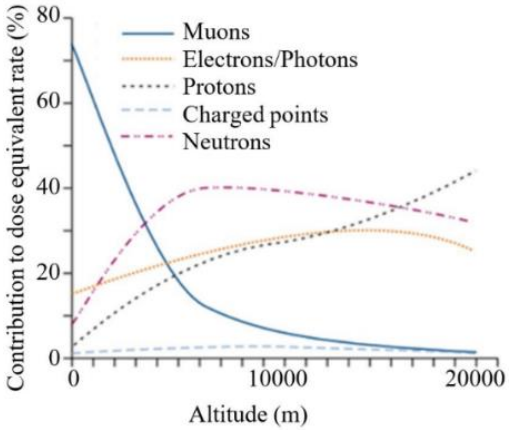

Figure 14: Dose equivalent rate components due to cosmic radiation in the atmosphere [47]

The contribution of radiation from the atmosphere (muons, electrons/photons, protons, charged pions and neutrons) to the dose of the dose equivalent depending on the flight altitude is presented in Figure 14 [47].

Public Exposure to the Natural Background

Doses associated with uncontrollable radiation sources (natural or other environmental sources) are given in Table 17.

Table 17: Average annual effective dose equivalent from natural radiation sources (after UNSCEAR Report 1988) [48]

Source of irradiation	Annual effective dose equivalent (mSv)		
	External	Internal	Total
Cosmic rays			
Directly ionizing radiation Neutron component	0.30 0.055	-	0.30 0.055
Cosmogenic radionuclides:	-	0.015	0.015
Primordial radionuclides			
Potassium K - 40	0.15	0.18	0.33
Rubidium Rb - 87	-	0.006	0.006
Uranium - 238 series:	0.1		1.34
Uranium - 238 to uranium - 234		0.005	
Thorium - 230		0.007	
Radium - 226		0.007 1.24	
Radon – 222 to polonium - 214		1.1	
Lead – 210 to polonium - 210		0.12	
Thorium – 232 series:	0.16		0.34
Thorium - 232		0.003	
Radium – 228 to radium - 224		0.13 0.18	
Radon – 220 to thallium - 208		0.16	
Total (rounded)	0.8	1.6	2.4

Examining the data in the tables shows that the average annual exposure from natural sources is about 2.4 mSv, of which 0.8 mSv external exposure and 1.8-mSv internal exposure. The background radiation exposure may vary with location due to differing radionuclide concentrations in rock and soil, water, and an increase of cosmic radiation with altitude. Activity of naturally occurring long-lived radionuclides in human body (pCi) is presented in Table 18 [3].

Table 18: Radionuclides in Human Body

Natural background	Radiation Activity in Body [pCi]
Uranium-238; α, 4,5 billion y	26
Radium-226; α, 1600 y	120
Radium-228; α, 5,8 y	21
Plumb-210; α, 21 y	600
Polonium-210; α, 138 d	200
Potasium-40; β 1,3 billion y	130 000
Carbon-14; β, 5,730 y	87 000
Hidrogen-3; β. 12,3 y	27 700
Rubidium-87; β, 4,8 x 1010 y	29 000
Strontiu-90, β, 28,1 y	2 900

The total radioactivity in the body is 277,567 pCi. This means that it exists 10, 267 radioactive decay per second (dps) and 887,068 800 disintegration per day in the human body. Each radioactive decay produces radiations.

The estimated total annual committed effective dose received by population because of ingestion of water was in the range 0.11–2.51 μSv/y for thermal water and in the range 0.11–38.8 μSv/y for mineral water [49]. In the staff working on the application of various treatment procedures in the spa resorts of Tusnad and Felix, H_E was evaluated between 0.02 and 2.2 mSv / y [50].

Finally, the annual effective dose to the public is the sum of the effective dose obtained in one year from external exposure with the committed effective dose due to the radionuclides incorporated in that year. Usually, the dose is not obtained from individual monitoring as in the case of occupational exposure, but is mainly determined by environmental measurements, data on habits and modeling, according to Pub 103 [20].

Conclusions

This paper, based on the principle of dose limits for radiation protection purposes, presented the basis of external exposure (when the radiation source is outside the body) and internal exposure (when the source of ionizing radiation is inside the body), using ionization dosimetry. These refer to occupational personnel and even to the population in the event of a nuclear incident or in medical radiation applications for diagnosis and treatment. Dose limits management is based on three fundamental principles of radiological protection: justification, optimization and dose limitation, this paper coveres the last principle.

The importance of dose limits is that irradiation of the total popu-

lation at low dose and low organ equivalent dose below 100 mSy, there is a probability of inducing stochastic effects (cancer or hereditary effects) in an internal organ, i.e., there is a linear relationship between equivalent dose and the stochastic effect, without threshold, LNT model.

The paper presented the operational quantities that refer to a point inside the phantom or body, radiometric quantities ($\Phi, \dot{\Phi}$ $\Psi, \dot{\Psi}$) used to define the dosimetric quantities (X, K, D), which in turn are applied to obtain conversion and dose coefficients for protection quantities in external and internal exposure.

After presenting the types of ionizing radiation, the fundamental equation of dosimetry valid for neutral particles, exposure X, and kerma, K, and for both types of ionizing radiation, absorbed dose, D, is presented. The determination of the absorbed dose from the charged particles is performed by applying the law of mass stopping of the charged particles, in this case electrons, in order to obtain their interaction coefficients with the environment. These coefficients are the following: the mass electronic (or collision) stopping power, S_{col}/ρ, in ($J \cdot m^2$ / kg) and the linear energy transfer, L_Δ in (J / m) or (keV / μm). Under Bragg-Gray conditions, the absorbed dose based on the electron fluence Φ, (m^{-2}), is D (Gy) = $(S_{el} / \rho)_{col} \cdot \Phi$, and D (Gy) = $(L_\Delta / \rho) \cdot \Phi$.

The quantities of radiological protection valid for both external and internal radiation exposure are the following. Average absorbed dose in organ T from R radiation. The equivalent dose in organ T, $H_T = w_R D_{TR}$, which is associated with a nominal risk factor, R_T, to determine the probability of inducing the stochastic effect in that organ T. The effective dose, $E = w_T H_T$, where the tissue weighting factor w_T is normalized, to take into account the fractional contribution of the risk of each organ or tissue T, to the total risk of stochastic effects, when the whole body is irradiated uniformly.

Because protection quantities (H_T equivalent dose and E effective dose) are not directly measurable, in external dosimetry, they are assessed using ICRU operational quantities for area monitoring and individual monitoring, such as ambient dose equivalent, H^* (10) and individual dose equivalent. Hp (10), used to estimate the external effective dose.

The internal dose is determined by applying the law of nuclear decay. Dose evaluation in the target organ, r_T, in the MIRD methodology, is done by multiplying the cumulative activity, $\tilde{A}_S$(Bq·s), in the source organ, rS, with the physical factor S (Gy / Bq·s), as follows: D ($r_T \leftarrow r_S$) =$\tilde{A}_S$ S ($r_T \leftarrow r_S$). The ICRP methodology uses the committed equivalent dose H_T (50) as the product of the specific effective energy, SEE (T ← S) in (Sv / Bq·s), and the number of U_S nuclear transformations in (Bq·s) in 50 y at organ source S, thus $H_T(50) = U_S$ (50) SEE (T ← S). The committed effective dose, denoted by E (50), is $E(50) = e_{ing}(50) \cdot I_{ing} + e_{inh}(50) \cdot I_{inh}$, where: the effective dose coefficient is e (50) (= $\sum_T w_T h_T$ (50)) and I (Bq) is the radionuclide intake by inhalation, and ingestion.

In most occupational exposure situations, the effective dose E can be deduced from the relationship $E = H_p$ (10) + E (50). To the second member of the equation can be added the incorporation of radionuclides through the skin, exposure to radon, and exposure to inert gas, which are not incorporated, during the 2000 working hours per year, DAC (Bq/m^3) = E_{limit} / 2000 $\dot{e}_{sub}$, where E_{lim} is the annual effective dose limit in Sv, and the effective dose rate coefficient, e_{sub} (Sv / h per Bq / m^3), for submersion in a cloud contaminated with a noble gas radionuclide per 2000 h - annual working time.

In special situations, the concentration of radon in the air is considered. The conversion from radon exposure to effective dose is obtained using a egality between the detriment per unit exposure to radon progeny (mJ h m-3 or WLM) and the detriment per unit effective dose (mSv), e.g 1 WLM = 5.06 mSv.

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