



Universität Hamburg



German-Armenian Joint Practical Course on Accelerator Physics

Beam-Matter Interaction and Radiation Dose Measurements

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1 Introduction

A secondary radiation is being produced when a particle accelerator beam interacts with matter (walls, equipment, air, etc.). The motivation for radiation measurements is determined by the needs of radiation safety, accelerator and particle beam physics, as well as by investigations related to materials and life science.. Radiation dose measurements allow monitoring and control of radiation levels. Measurement results provide the basis for construction of radiation protection walls and shields, and for the development of safety procedures. Radiation measurements are useful for the determination of particle accelerator beam parameters. They are based on measurements by tools and detectors of the beam diagnostic system. Combining beam diagnostics results with numerical simulations of particle transport, radiation dose measurements enable one to calculate parameters for irradiation of samples for materials and life science experiments.

Equipment and tools for radiation measurements available at CANDLE include: dosimeter based on Geiger-Muller counter for the measurement of radiation dose due to natural background and for residual dose measurements; a ionization chamber that can cope with high rate radiation and which is suitable for the measurement of prompt radiation produced during accelerator operation; Faraday cups, YAG screen stations, spectrometer and dosimeters.

Experimental tasks include: 1) search for high levels of Radon concentration in the basement rooms applying a high precision Geiger-Muller counter dosimeter; 2) measurements of the dose rates during AREAL accelerator operation by an ionization chamber and study of the radiation attenuation achieved by the protective wall.

The figure depicts a 2D dose distribution in the AREAL accelerator hall simulated by the FLUKA particle transport code.

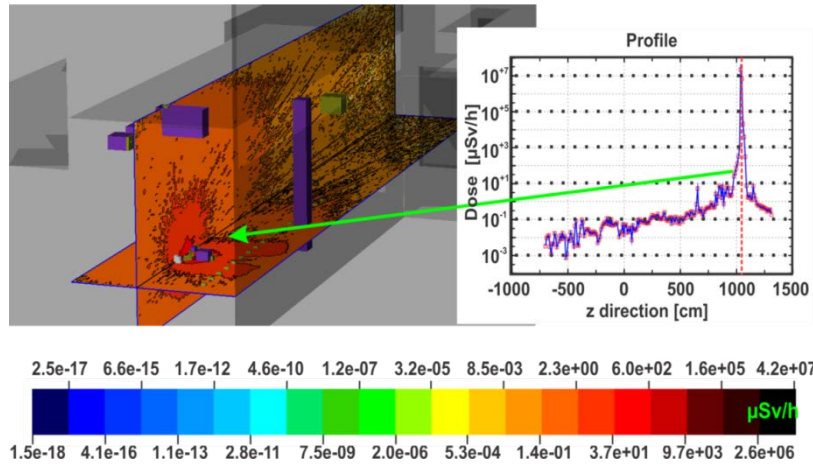


Figure 1.1: Equivalent dose distribution in the tunnel in Microsievert per electron hitting the Faraday Cup at the 95cm distance from the electron gun for 5 MeV beam energy (FLUKA simulation). Radiation penetrated tunnel walls carries $7.6 \cdot 10^{-5}$ MeV energy per electron.

2. Passage of electrons through the matter

2.1 Energy Loss by Electrons

Electrons interact with material through several mechanisms:

Ionization $e^- + A \rightarrow e^- + A^+ + e^-$;

Moller scattering $e^- + e^- \rightarrow e^- + e^-$;

Bremsstrahlung $e^- + e^-(A) \rightarrow e^-(A') + \gamma$;

Pair production $e^- + e^-(A) \rightarrow e^-(A') + (e^+ + e^-)$;

etc.

Electron (positron) scattering is considered as ionization when the energy loss per collision is below 0.255 MeV , and as Möller's (Bhabha) scattering when it is above.

At high energies (starting from a few tens of MeV) bremsstrahlung mechanism prevails.

2.2 Energy loss by ionisation (electron and positron)

In collisions of charged particles with matter, both excitation and ionization occurs. Energy loss during bremsstrahlung must also be considered for relativistic particles. Interacting neutral particles may produce charged particles, which are then detected during their specific interaction processes. For photons these processes are known as Compton scattering, photoelectric effect and pair production of electrons. The

electron can be observed through a sensitive ionization detection, which is generated during photon interactions.

Exact calculation taking into account the specific differences between the incident heavy particles and electrons gives a more accurate formula for the loss of electron energy as a result of ionization and excitation:

$$-\frac{dE}{dx} = 4 \pi N_A r_e^2 m_e c^2 \frac{Z}{A} \cdot \frac{1}{\beta^2} \left(\ln \frac{\gamma m_e c^2 \beta \sqrt{\gamma-1}}{\sqrt{2} I} + \frac{1}{2} (1 - \beta^2) - \frac{2\gamma-1}{2\gamma^2} \ln 2 + \frac{1}{16} \left(\frac{\gamma-1}{\gamma} \right)^2 \right) \quad (2.1)$$

Here, Z is the atomic number, A is the atomic mass, m_e is the electron mass, r_e is the classical electron radius, and N_A is Avogadro constant. This expression agrees with the general Bethe–Bloch relation within 10%–20%. The kinematics of electron-electron collisions and screening effects are also taken into account. The treatment of the ionisation loss of positrons is similar to that of electrons if one considers that these particles are of equal mass, but not identical charge.

For completeness, we also give the ionisation loss of positrons:

$$-\frac{dE}{dx} = 4 \pi N_A r_e^2 m_e c^2 \frac{Z}{A} \cdot \frac{1}{\beta^2} \left(\ln \frac{\gamma m_e c^2 \beta \sqrt{\gamma-1}}{\sqrt{2} I} - \frac{\beta^2}{24} \left[23 + \frac{14}{\gamma+1} + \frac{10}{(\gamma+1)^2} + \frac{4}{(\gamma+1)^3} \right] \right) \quad (2.1)$$

Since positrons are antiparticles of electrons, there is, however, an additional consideration: if positrons come to rest, they will annihilate with an electron normally into two photons which are emitted anti-collinearly. Both photons have energies of 511 keV in the centre-of-mass system, corresponding to the rest mass of the electrons. The cross section for annihilation in flight is given

$$\sigma(Z, E) = \frac{Z \pi r_e^2}{\gamma+1} \left[\frac{\gamma^2+4\gamma+1}{\gamma^2-1} \ln(\gamma + \sqrt{\gamma^2-1}) - \frac{\gamma+3}{\sqrt{\gamma^2-1}} \right] . \quad (2.3)$$

2.3 Energy losses by electron and positron via collision

For electrons and positrons, stopping power is different from stopping the power of heavy particles. The difference is kinematics, charge, spin and the characteristics of the electron causing ionization. A large part of electron's energy is transferred to atomic electrons (taken as free), which is described by the Møller cross section. In a single collision the maximum energy transition equals the total kinetic energy $W_{max} = m_e c^2 (\gamma - 1)$ but in case of identical particles, the maximum is half of this at $W_{max}/2$. (The results will be the same if transferred energy equals ϵ or $W_{max} - \epsilon$. The stopping power calculation is done for the faster of the two emerging electrons

by convention). In the formula provided below stopping power is the first moment of the Møller cross section (divided by dx)

$$\langle -\frac{dE}{dx} \rangle = \frac{1}{2} K \frac{Z}{A} \frac{1}{\beta^2} \left[\ln \frac{m_e c^2 \beta^2 \gamma^2 \{m_e c^2 (\gamma-1)/2\}}{I^2} + (1 - \beta^2) - \frac{2\gamma-1}{\gamma^2} \ln 2 + \frac{1}{8} \left(\frac{\gamma-1}{\gamma} \right)^2 - \delta \right] \quad (2.4)$$

Here the notion $= 4 \pi N_A r_e^2 m_e c^2$. By substituting a logarithmic term in the Bethe equation by $W_{max} = m_e c^2 (\gamma - 1)/2$ it can be compared with the logarithmic term in the above formula. The two forms differ by $\ln 2$. For describing electron-positron scattering a more complicated cross sectional formula called Bhabha cross section is used. The identical particle problem doesn't exist in this case so we can consider that $W_{max} = m_e c^2 (\gamma - 1)$. The first moment of the Bhabha equation results in the following.

$$\langle -\frac{dE}{dx} \rangle = \frac{1}{2} K \frac{Z}{A} \frac{1}{\beta^2} \left[\ln \frac{m_e c^2 \beta^2 \gamma^2 \{m_e c^2 (\gamma-1)\}}{2I^2} + 2 \ln 2 - \frac{\beta^2}{12} \left(23 + \frac{14}{\gamma+1} + \frac{10}{(\gamma+1)^2} + \frac{4}{(\gamma+1)^3} \right) - \delta \right] \quad (2.5)$$

The density effect correction δ was added to Uehling's equations following ICRU 37 in both cases (see Literature [4]).

According to **Figure 2.1** stopping powers for e^- , e^+ and heavy particles do not differ dramatically. In silicon material, the minimum value for e^- is $1.50 MeV cm^2/g$ (at $\gamma = 3.3$); for positrons, $1.46 MeV cm^2/g$ (at $\gamma = 3.7$), and for μ^- $1.66 MeV cm^2/g$ (at $\gamma = 3.58$).

2.4 Radiation Length

High-energy electrons interact with matter mainly by bremsstrahlung, while high-energy photons interact by electron-positron pair production. The characteristic amount of matter, traversed by those particles is called the radiation length X_0 (measured in $gram cm^{-2}$). The radiation length is the mean distance over which a high-energy electron loses all but $1/e$ of its energy by bremsstrahlung. Equivalently, it is the $7/9$ part of the mean free path for pair production by a high-energy photon. For high-energy electromagnetic cascades, the radiation length is also suitable. **Eq. (2.6)** presents a calculation of X_0 which is tabulated by Y.S. Tsai

(See Lit. [1] and references herein):

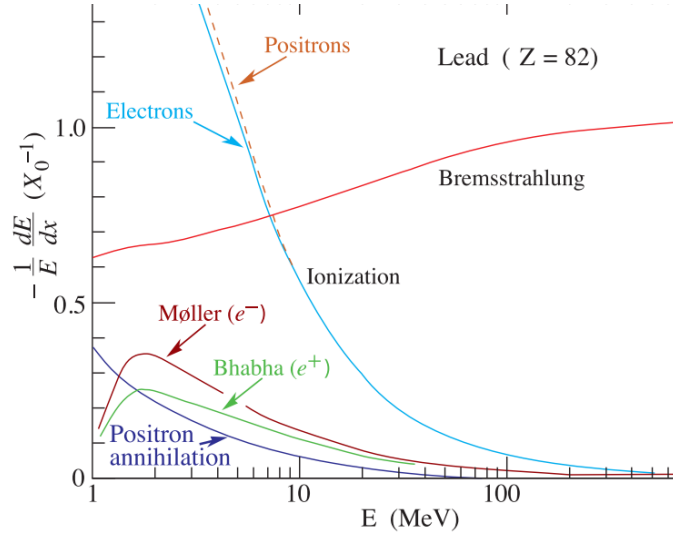


Figure 2.1: Fractional energy loss per radiation length in lead as a function of electron or positron energy (See Lit. [1] and references herein).

$$\frac{1}{X_0} = 4\alpha r_e^2 \frac{N_A}{N} \{Z^2 [L_{rad} - f(Z)] + ZL'_{rad}\} \quad (2.6)$$

where α is the fine-structure constant.

For $A = 1 \text{ g mol}^{-1}$, $4\alpha r_e^2 N_A/A = (716.408 \text{ g cm}^{-2})^{-1}$. L_{rad} and L'_{rad} are given in **Table 2.1**. The function $f(Z)$ is an infinite sum. The elements up to uranium can be represented to 4 digit accuracy by the following formula

$$f(Z) = a^2[(1 + a^2)^{-1} + 0.20206 - 0.0369a^2 + 0.0083a^4 - 0.002a^6]$$

where $a = \alpha Z$.

Although there are accurate formulae to calculate, O. I. Dahl provides a compact fit to the data:

$$X_0 = \frac{716[g \times \text{cm}^{-2}]A}{Z(Z+1) \ln(287/\sqrt{Z})} \quad (2.7)$$

where A is the atomic mass and Z is the atomic number of the absorber(See [1] and references herein).

Table 2.1. Tsai's L_{rad} and L'_{rad} for use in calculating the radiation length according to Eq.(2.6)

Element	Z	L_{rad}	L'_{rad}
H	1	5.31	6.144
He	2	4.79	5.621
Li	3	4.74	5.805
Bi	4	4.71	5.924
Others	> 4	$\ln(184.5 Z^{-1/3})$	$\ln(1194 Z^{-2/3})$

Results obtained using this formula agree with Tsai's accurate formulae to better than 2.5% for all elements. The radiation associated with a mixture or compound may be approximated by the formula $1/X_0 = \sum w_i/X_{0i}$, where w_i and X_{0i} are the fractions by weight and the radiation length for the i – th element.

2.5 Bremsstrahlung

An electron loses energy by bremsstrahlung at a rate nearly proportional to its energy, while the ionization loss rate varies only logarithmically with the electron energy. The critical energy E_c is sometimes defined as the energy at which the two loss rates are equal. The value of the critical energy when bremsstrahlung starts to prevail over ionization mechanism can be obtained by the expressions: $E_c = \frac{610\text{MeV}}{Z+1.24}$ for solids and liquids and $E_c = \frac{710\text{MeV}}{Z+0.92}$ for gases. Alternatively, Rossi defines the critical energy as the energy at which the ionization loss per radiation length is equal to the electron energy. Experimental results prove that Rossi's form of the critical energy definition describes transverse electromagnetic shower development more accurately.

Bremsstrahlung radiation is emitted when the fast-moving charged particle is decelerated in the Coulomb field of the atoms. Though radiation takes place mainly due to the field of the nuclei, atomic electrons also contribute to the process. Since the probability of the bremsstrahlung process is proportional to $1/M^2$ (M being the mass of the particle), bremsstrahlung becomes the dominant process in the interaction of the lightest charged particles (i.e. electrons and positrons) with most materials already above a few tens of MeV. The bremsstrahlung process probability increases with Z^2 .

The Bethe-Heitler formula gives the energy loss rate according to

$$-\frac{dT_e}{dx} = n T_e \alpha Z^2 r_e^2 \left(4 \ln \frac{183}{Z^{1/3}} + \frac{2}{9} \right) \quad (2.8)$$

where n is the concentration of the atoms, α is the fine structure constant, $r_e = e^2/(m_e c^2)$ and it is assumed, that the kinetic energy of the electrons $T_e \gg m_e c^2 / (\alpha Z^{1/3})$. This condition ensures consideration of the full screening effect. The electron cloud of the atoms contributes to the bremsstrahlung proportional to Z . Calculation of the radiation spectrum calculation cannot be accurate without taking into account the so-called LPM effect, see below.

An approximation for the spectrum of the bremsstrahlung cross section in the “complete screening case” at high energies is given by the formula

$$d\sigma/dk = (1/k) 4\alpha r_e^2 \left\{ \left(\frac{4}{3} - \frac{4}{3}y + y^2 \right) [Z^2 (L_{\text{rad}} - f(Z) + ZL'_{\text{rad}})] + \frac{1}{9} (1 - y)(Z^2 + Z) \right\} \quad (2.9)$$

where $y = k/E$ is the part of the electron’s energy transferred to the radiated photon. At small y which is the case in the “infrared limit” the second term varies from 1.7% (low Z) to 2.5% (high Z) of the total expression. If it is neglected and the first line shortened with the definition of X_0 given in **Eq. (2.6)**, we have

$$\frac{d\sigma}{dk} = \frac{A}{R N_A k} \left(\frac{4}{3} - \frac{4}{3}y + y^2 \right) \quad (2.10)$$

The top curve in **Fig. 2.2** shows this cross section (times k). This formula takes place except for the points $y = 1$ and $y = 0$. In the $y = 1$ case, screening may become incomplete, and in the $y = 0$ case, the infrared divergence is removed by the interference of bremsstrahlung amplitudes with nearby scattering centers (the LPM effect) and dielectric suppression.

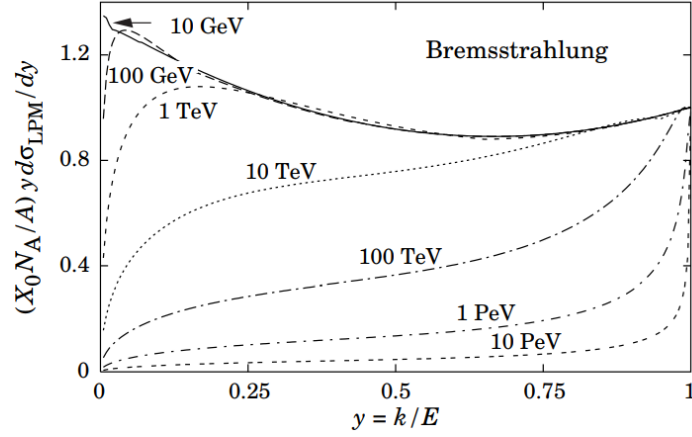


Fig 2.2: Dependence of the normalized bremsstrahlung cross section $k d\sigma_{LPM}/dk$ versus the fractional photon energy $y = k/E$ in lead. The vertical axis is given in units of photons per radiation length

An approximate expression for the energy loss rate is given by $-dE/dx = E/X_0$. Thus, the energy loss per unit path length is proportional to the energy of the charged particle and charged particle energy attenuation takes place exponentially:

$$E(x) = E(0) \exp(-x/X_0) \quad (2.11)$$

The number of photons with energies between k_1 and k_2 radiated by an electron traveling a distance $d \ll X_0$ is

$$N_\gamma = \frac{d}{X_0} \left[\frac{4}{3} \ln \left(\frac{k_2}{k_1} \right) - \frac{4(k_2 - k_1)}{3E_e} + \frac{k_2^2 - k_1^2}{2E_e^2} \right] \quad (2.12)$$

The mean value of the photon emission angle in the bremsstrahlung process does not depend on the photon energy. It can be found from the formula

$$\bar{\vartheta} = \frac{m_e c^2}{E_e} \quad (2.13)$$

2.6 Energy loss by photons:

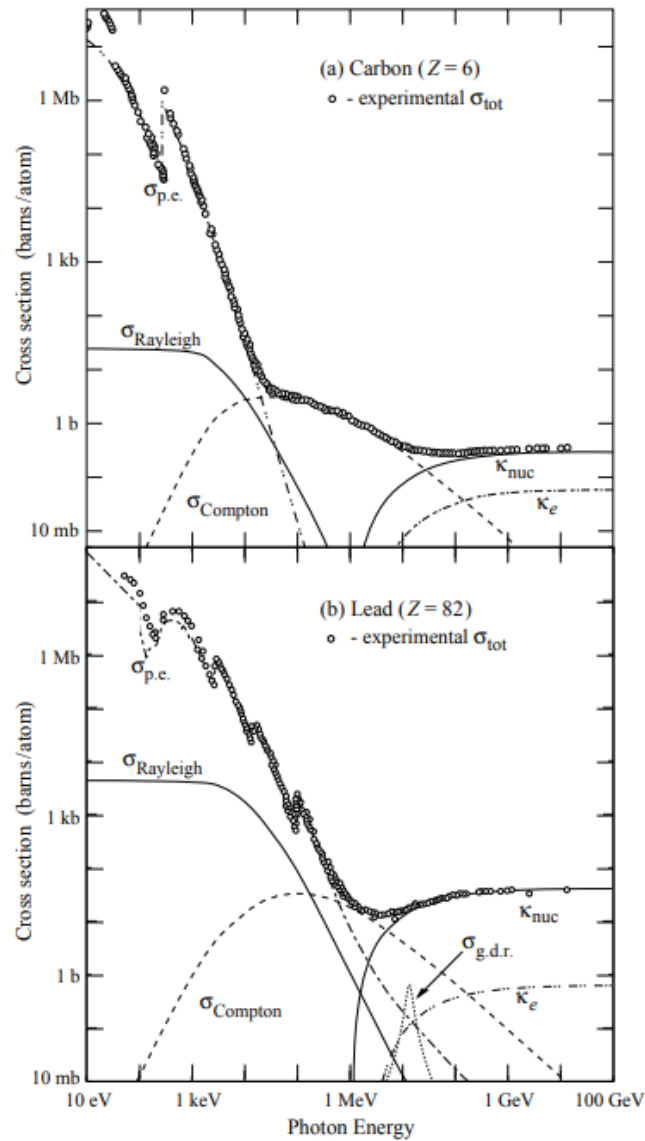


Figure 2.3: Photon total cross sections as a function of energy in carbon and lead with contributions of different processes:

- $\sigma_{p.e.}$ = Atomic photoelectric effect
- $\sigma_{Rayleigh}$ = Rayleigh (coherent) scattering
- $\sigma_{Compton}$ = Incoherent scattering (Compton scattering on electron)
- κ_{nuc} = Pair production in nuclear field
- κ_e = Pair production in electron field
- $\sigma_{g.d.r.}$ = Photonuclear interactions via Giant Dipole Resonance mechanism

Contributions to the photon cross section in a light element (carbon) and a heavy element (lead) are shown in **Fig. 2.3**. At low energies it is seen that the photoelectric effect dominates, although Compton scattering, Rayleigh scattering, and

photonuclear absorption also contribute. The photoelectric cross section is characterized by discontinuities (absorption edges) as thresholds for photoionization of various atomic levels are reached. Photon attenuation lengths for a variety of elements are shown in **Fig. 2.4**, and data for $30 \text{ eV} < k < 100 \text{ GeV}$ for all elements are available from the web pages given in the caption. Here k is the photon energy.

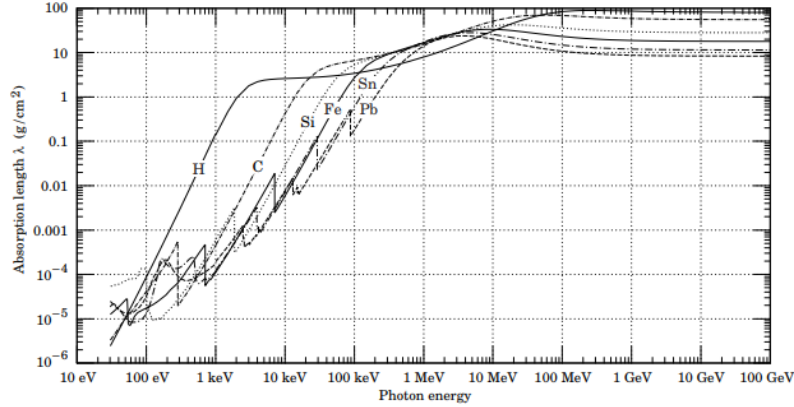


Figure 2.4: The photon mass attenuation length (or mean free path) $\lambda = 1/(\mu/\rho)$ for various elemental absorbers as a function of photon energy. The mass attenuation coefficient is μ/ρ , where ρ is the density. The intensity I remaining after traversal of thickness t (in mass/unit area) is given by $I = I_0 \exp(-t/\lambda)$. The accuracy is a few percent. For a chemical compound or mixture, $1/\lambda_{eff} \approx \sum_{elements} w_Z/\lambda_Z$, where w_Z is the proportion by weight of the element with atomic number Z .

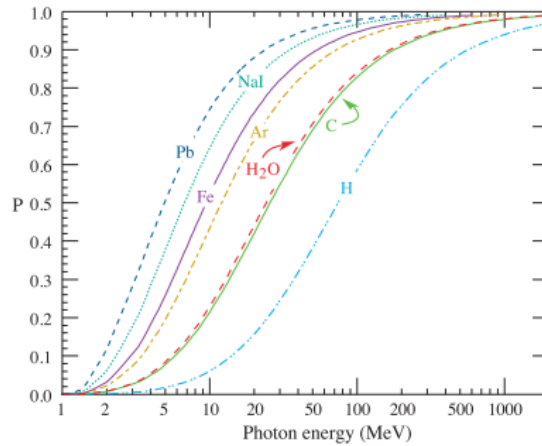


Figure 2.5: Probability P that a photon interaction will result in conversion to an e^+e^- pair. Except for a few-percent contribution from photonuclear absorption around 10 or 20 MeV , essentially all other interactions in this energy range result in Compton scattering off an atomic electron. For a photon attenuation length λ (**Fig. 2.4**), the probability that a given photon will produce an electron-positron pair (without first Compton scattering) within the thickness t of the absorber is $P[1 - \exp(-t/\lambda)]$.

The increasing dominance of pair production as the energy increases is shown in **Fig. 2.5**. Using approximations, Tsai's formula for the differential cross section reduces to

$$\frac{d\sigma}{dx} = \frac{A}{X_0 N_A} \left[1 - \frac{4}{3}x(1-x) \right] \quad (2.14)$$

in the complete-screening limit valid at high energies. Here $x = E/k$ is the fractional energy transfer to the pair-produced electron (or positron), and k is the incident photon energy.

2.7 Electromagnetic Cascade

When a thick absorber is hit by a high-energy electron or photon it initiates an electromagnetic cascade as pair production and more electrons and photons with lower energy are generated by bremsstrahlung. The cascade evolution in the longitudinal direction is governed by the high-energy particles and therefore scales as the radiation length in the material. The electrons' energy decreases and falls below the critical energy, and later dissipates by ionization and excitation rather than by the generation of more shower particles. When describing the behavior of electromagnetic showers it is convenient to bring up the scale variables $t = x / X_0$; $y = E / E_c$. With this convention, distance is measured in units of radiation length and energy in units of critical energy.

In **Fig. 2.6**, longitudinal profiles are shown from the simulation of a 30 GeV electron-induced cascade in iron by the simulation code EGS4. The number of secondary particles which are crossing a plane (very close to Rossi's Π function) depends very sensitively on the cutoff energy, here chosen as the a total energy of 1.5 MeV for both e^- and e^+ . The number of electrons decreases more drastically than energy deposition. This happens because when increasing the depth at a later portion of the cascade the energy is carried by photons. Calorimeter measurement depends on the device characteristics, however, it is not likely to be exactly any of the profiles shown in **Figure 2.1**. In case of gas-based counters, it may be very close to the electron number, but in case of glass-based "Cherenkov" detectors, which have "thick" sensitive regions, it is closer to the energy deposition (total track length). In "Cherenkov" detectors, the signal of the detectors is proportional to the "detectable" track length T_d which is usually less than the total track length T . Real devices are sensitive to particles with energy higher than some threshold E_d , and $T_d =$

$TF(E_d/E_c)$. An analytic formula for $F(E_d/E_c)$ as obtained by Rossi is given by Fabian, Amaldi (See [1] and references herein).

The mean longitudinal profile of energy deposition in an electromagnetic cascade is described in detail by a gamma distribution (See Lit. [1] and references herein)

$$\frac{dE}{dt} = E_0 b \frac{(bt)^{a-1} e^{-bt}}{\Gamma(a)} \quad (2.15)$$

The maximum t_{max} occurs at $(a - 1)/b$. Fits to shower profiles in elements range from carbon to uranium, at energies from 1 GeV to 100 GeV. The energy deposition profiles are well described by **Eq. (2.15)** with

$$t_{max} = (a - 1)/b = 1.0 \times (\ln y + C_j), \quad j = e, \gamma \quad (2.16)$$

where $C_e = -0.5$ for electron – induced cascades and $C_\gamma = +0.5$ for photon-induced cascades. To use **Eq. (2.15)**, one finds $(a - 1)/b$ from **Eq. (2.16)**, then finds a either by assuming $b \approx 0.5$ or by finding a more accurate value from **Figure 2.7**.

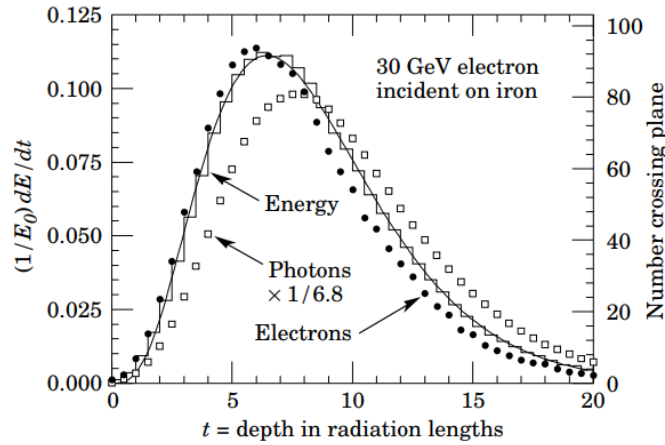


Figure 2.6: 30 GeV electron-induced cascade simulation in iron. Fractional energy deposition per radiation length is shown by the histogram, and the curve is a gamma-function fit to the distribution.

The results are identical for the electron number profiles, but it depends on the atomic number of the medium. The same form for the electron number maximum was obtained by Rossi in the assumption, but with $C_e = -1.0$ and $C_\gamma = -0.5$; we regard this as superseded by the EGS4 result.

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Parameterization of “shower length” $X_s = X_0/b$ is done less conveniently, since b depends upon both Z and incident energy, as shown in **Figure 2.7**. As a consequence of this Z dependence, the number of electrons that are crossing a plane near the shower maximum is underestimated using the approximation of Rossi for carbon and significantly overestimated for uranium. Necessarily the same b values are obtained for incident e^- and e^+ . For most cases, it is sufficient to take $b \approx 0.5$.

Ultra-high energy photons and electrons have a greater length of showers than at lower energies since the first or first few interaction lengths are increased via the mechanisms described above.

Near the origin, the EGS4 cascade (or a real cascade) increases more sharply, whereas the gamma function distribution is very flat. As a result (**Eq. 2.15**) fails abominably. For the first two radiation lengths cases; it was thus necessary to remove this region in making fits.

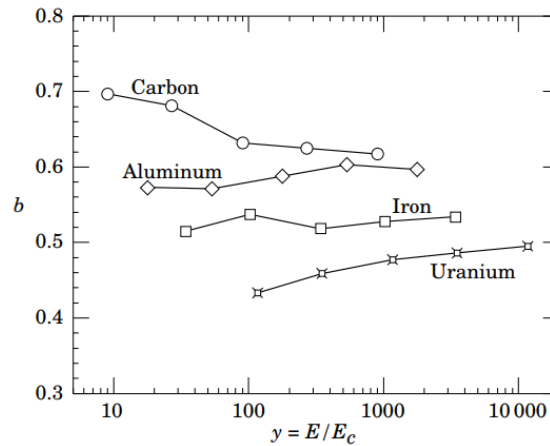


Figure 2.7: Fit values for the scale factor

b fit values for energy deposition profiles for a variety of elements for incident electrons with $1\text{ GeV} \leq E_0 \leq 100\text{ GeV}$

Since fluctuations play an important role, (**Eq.2.15**) should be used only in applications where mean behavior is adequate. Fast simulation algorithms were developed by Grindhammer et al. in which the fluctuation and correlation of a and b are obtained by fitting (**Eq. 2.15**) to individually simulated cascades. Then from profiles generated for cascades, the correlated distributions using a and b were chosen.

The development of electromagnetic showers in the transverse direction in different materials scales quite accurately with the Moliere radius R_M , given by $R_M = X_0 E_s / E_c$, where $E_s \approx 21 \text{ MeV}$, and the Rossi definition of E_c is used.

In a material with weight fraction w_j of the element, with radiation length X_j and critical energy E_{cj} , the Moliere radius is expressed by

$$\frac{1}{R_M} = \frac{1}{E_s} \sum \frac{w_j E_{cj}}{X_j} \quad (2.17)$$

On average, only 10% of the energy falls outside the cylinder with radius R_M . About 99% falls inside $3.5R_M$, but at this radius and beyond, composition effects play an important role and the scaling with R_M fails. The distributions are represented by a narrow core, and broaden as the shower develops. They are often represented as the sum of two Gaussians, and Grindhammer (See Lit. [1] and references herein). defines them with the function

$$f(r) = \frac{2rR^2}{(r^2 + R^2)^2} \quad (2.18)$$

where R is a phenomenological function of x/X_0 and $\ln E$. The LPM effect reduces the cross sections for bremsstrahlung and pair production at high energies and hence can cause significant elongation of electromagnetic cascades.

3 Environmental radioactivity

3.1 Environmental radioactivity:

The long-lived natural radio-nuclides K^{40} , Th^{232} , and U^{238} with average abundances of 1.6, 11.1 and 2.7 *ppm* (corresponding to 412, 45 and 33 *Bq/kg*, respectively) in the earth's crust have large local variations. In most cases, γ radiation emitted due to the decay of natural radioactivity and its unstable daughters constitutes the dominant share in the local radiation field. Typical low-background applications contribute levels of natural radioactivity on the order of *ppb* or *ppt* in the detector components. Passive or active shielding is used to suppress external γ radiation down to an equivalent level. **Fig. 3.1** shows the energy-dependent attenuation length $\lambda(E_\gamma)$ as a function of γ -ray energy E_γ for three common shielding materials (water, copper, lead). The thickness ℓ required to reduce the external flux by a factor $f > 1$ is estimated, assuming exponential damping:

$$\ell = \lambda(E_\gamma) \cdot \ln f$$

At 100 *keV*, a typical energy scale for dark matter searches (or 2.615 *MeV*, for a typical double-beta decay experiment), attenuation by a factor $f = 105$ requires 67(269) *cm* of H_2O , 2.8(34) *cm* of *Cu*, or 0.18(23) *cm* of *Pb*. Such estimates allow for an order-of-magnitude determination of the experiment dimensions.

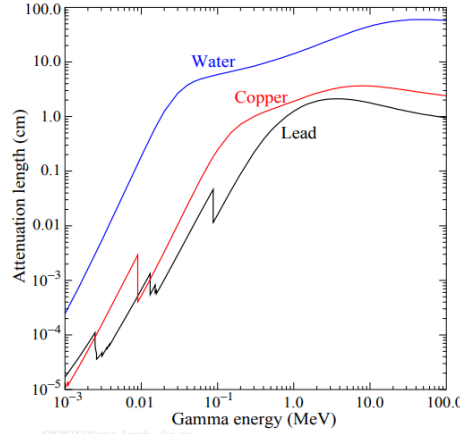


Figure 3.1: γ -ray attenuation lengths in some common shielding materials. The mass attenuation data has been taken from the NIST data base XCOM (pdg.lbl.gov).

A precise estimation of the magnitude of the external gamma-ray background, including scattering and the effect of analysis-energy cuts, requires Monte Carlo simulations based on the knowledge of the radioactivity field present in the laboratory. Detailed modeling of the γ -ray flux in a large laboratory, or inside the hermetic shielding, needs to cope with a very small probability of generating any signal in the detector. It is often advantageous to calculate the solid angle of the detector with respect to the background sources and mass attenuation of the radiation shield separately, or to employ importance sampling. These approaches reduce the computation time required for a statistically meaningful number of detector hits to manageable levels.

Water is commonly used as shielding medium for large detectors, as it can be obtained cheaply and purified effectively in large quantity. Water purification technology is commercially available. Ultra-pure water, instrumented with photomultiplier tubes, can serve as an active cosmic-ray veto counter. Water is also an effective neutron moderator and shield. In more recent underground experiments

that involve detectors operating at cryogenic temperature, liquefied gases (e.g. argon) are being used for shielding as well.

3.2 Radioactive impurities in detector and shielding components:

After suppressing the effect of external radioactivity, radioactive impurities, contained in the detector components or attached to their surfaces, become important. Every material contains radioactivity at some level. The activity can be natural, cosmogenic, man-made, or a combination of them. The determination of the activity content of a specific material or component requires case-by-case analyses, and is rarely obtainable from the manufacturer. However, there are some general rules that can be used to guide the pre-selection. For detectors designed to look for electrons (for example in double-beta decay searches or neutrino detection via inverse beta decay or elastic scattering), intrinsic radioactivity is often the principal source of background. For devices detecting nuclear recoils (for example in dark matter searches), this is often of secondary importance as ionization signals can be actively discriminated on an event-by-event basis. Decay induced nuclear reactions become a concern.

For natural radioactivity, a rule of thumb is that synthetic substances are cleaner than natural materials. Typically, more highly processed materials have lower activity content than raw substances. Substances with high electro-negativity tend to be cleaner as the refining process preferentially removes *K*, *Th*, and *U*. For example, Al is often found to contain considerable amounts of Th and U, while electrolytic Cu is very low in primordial activities. Plastics or liquid hydrocarbons, having been refined by distillation, are often quite radiopure. Tabulated radioassay results for a wide range of materials. Radioassay resulting from previous underground physics experiments are being archived at an online database.

The long-lived U^{238} daughter Pb^{210} ($T_{1/2} = 22.3$ y) is found in all shielding lead, and is a background concern at low energies. This is due to the relatively high endpoint energy ($Q_{\beta} = 1.162$ MeV) of its beta-unstable daughter Bi^{210} . Lead refined from selected low-*U* ores have specific activities of about 5–30 Bq/kg. For applications that require lower specific activity, ancient lead (for example from Roman ships) is sometimes used. Because the ore processing and lead refining removed most of the U^{238} , the Pb^{210} decayed during the long waiting time to the level supported by the *U*-content of the refined lead. Lining the lead with copper to range out the low-energy

radiation is another remedy. However, intermediate-Z materials carry additional cosmogenic-activation risks when handled above ground, as will be discussed below. Pb^{210} is also found in solders.

Man-made radioactivity, released during above-ground nuclear testing and nuclear power production, is a source of background. The fission product Cs^{137} can often be found attached to the surface of materials. The radioactive noble gas Kr^{85} , released into the atmosphere by nuclear reactors and nuclear fuel re-processing, is sometimes a background concern, especially due to its high solubility in organic materials. Post-World War II steel typically contains a few tens of mBq/kg of Co^{60} .

Surface activity is not a material property per se but is added during manufacturing and handling. Surface contamination can often be effectively removed by clean machining, etching, or a combination of both. The assembly of low-background detectors is often performed in controlled enclosures (e.g. clean rooms or glove boxes) to avoid contaminating surfaces with environmental substances, such as dust, containing radioactivity at much higher concentrations than the detector components. Surfaces are cleaned with high purity chemicals and de-ionized water. When not being processed, components are best stored in sealed bags to limit dust deposition on the surface, even inside clean rooms. Surface contamination can be quantified by means of wipe-testing with acid or alcohol wetted Whatman 41 filters. Pre-soaking of the filters in clean acid reduces the amount of Th and U contained in the paper and boosts analysis sensitivity. The paper filters are ashed after wiping and the residue is digested in acid. Subsequent analysis by means of mass spectroscopy or neutron activation analysis is capable of detecting less than 1 pg/cm^3 of Th and U .

The most demanding low-rate experiments require screening of all components, which can be a time consuming task. The requirements for activity characterization depend on the experiment and the location and amount of a particular component. Monte Carlo simulations are used to quantify these requirements. Sensitivities of the order $\mu Bq/kg$ or less are sometimes required for the most critical detector components. At such a level of sensitivity, the characterization becomes a challenging problem in itself. Low-background α , β , and γ -ray counting, mass spectroscopy, and neutron activation analysis are the commonly used diagnostic techniques.

3.3 Radon and environmental radioactivity:

The noble gas Rn^{222} , a pure α -emitter, is a U^{238} decay product. Due to its relatively long half-life of 3.8 d it is released by surface soil and is found in the atmosphere everywhere. Rn^{220} (Th^{232} decay product) is mostly unimportant for most low-background experiments because of its short half-life. The Rn^{222} activity in air ranges from 10 to 100 mBq/L outdoors and 100 to thousands of mBq/L indoors. The natural radon concentration depends on the weather and shows daily and seasonal variations. Radon levels are lowest above the oceans. For electron detectors, it is not the Rn itself that creates background, but its progeny Pb^{214} , Bi^{214} , Bi^{210} , which emit energetic beta and γ radiation. Thus, not only the detector itself has to be separated from contact with air, but also internal voids in the shield which contain air can be a background concern. Radon is quite soluble in water and even more so in organic solvents. For large liquid scintillation detectors, radon mobility due to convection and diffusion is a concern. To define a scale: typical double-beta-decay searches are restricted to $< \mu\text{Bq/kg}_{\text{detector}}$ (or 1 decay per $\text{kg}_{\text{detector}}$ and per 11.6 days) activities of Rn^{222} in the active medium. This corresponds to a steady-state population of $0.5\text{ atoms/kg}_{\text{detector}}$ or $50\text{ }\mu\text{L/kg}_{\text{detector}}$ of air (assuming 20 mBq/L of radon in the air). The demand on leak tightness can thus be quite demanding. The decay of Rn itself is a concern for some recoil type detectors, as nuclear recoil energies in α decays are substantial (76 keV in the case of Rn^{222}).

Low-background detectors are often kept sealed from the air and continuously flushed with boil-off nitrogen, which contains only small amounts of Rn. For the most demanding applications, the nitrogen is purified by multiple distillations, or by using pressure swing adsorption chromatography. Then only the Rn outgassing of the piping (due to its intrinsic U content) determines the radon concentration. Radon diffuses readily through thin plastic barriers. If the detector is to be isolated from its environment by means of a membrane, the choice of material is important.

Prolonged exposure of detector components or raw materials to air leads to the accumulation of the long-lived radon daughter Pb^{210} on surfaces. Due to its low Q -value of 63.5 keV , Pb^{210} itself is only a problem when extreme low energy response is important. However, because of its higher Q -value, the lead daughter Bi^{210} is a concern up to the MeV scale. The alpha unstable Bi -daughter Po^{210} ($E_{\alpha} =$

5304 keV) contributes not only to the alpha background but can also induce the emission of energetic neutrons via (α, n) reactions on low-Z materials (such as *F, C, Si...etc*). The neutrons, in turn, may capture on other detector components, creating energetic background. The (α, n) reaction yield induced by the α decay of Po^{210} is typically small ($6 \cdot 10^{-6} n/\alpha$ in Teflon, for example). Some data is available on the deposition of radon daughters from air onto materials. This data indicates effective radon daughter collection distances of a few cm in air. These considerations limit the allowable air exposure time. In case raw materials (e.g. in the form of granules) were exposed to air at the production site, the bulk of the finished detector components may be loaded with Pb^{210} and its daughters. These are difficult to detect as no energetic gamma radiation is emitted in their decays. Careful air-exposure management is the only way to reduce this source of background. This can be achieved by storing the parts under a protective low-radon cover gas or keeping them sealed from radon.

State-of-the-art detectors can detect radon even at the level of few atoms. Solid state, scintillation, or gas detectors utilize alpha spectroscopy or are exploiting the fast $\beta - \alpha$ decay sequences of Bi^{214} and Po^{214} . The efficiency of these devices is sometimes boosted by electrostatic collection of charged radon from a large gas volume into a small detector.

Radon (a radioactive noble gas, originating from the soil's uranium and thorium content) is considered responsible for more than half the average natural radiation dose for humans and one of the major causes of lung cancer. The health concerns of radon made regulatory control necessary and many countries implemented some measures for handling exposure to radon. The usual regulatory approach is specifying dose limits, an amount of radiation dose that is acceptable, these would be translated to reference levels, meaning permissible activity concentrations in various media (set in a way not to reach the dose limits). These reference levels then would be compared to the measured activity concentrations and if those concentrations exceed the reference levels the appropriate measures set in the national regulations would have to be implemented. Developments in the dose conversion calculations further raised the importance of radon (the conversion factors were approximately tripled from the previously used ICRP 65). The European Union has included exposure to radon in the 2014 Basic Safety Standards, which requires the Member

States to introduce reference levels for indoor radon concentrations not exceeding (as an annual average) 200 Bq m^{-3} for new dwellings and new buildings with public access, 300 Bq m^{-3} for existing dwellings, and 300 Bq m^{-3} for existing buildings with public access, allowing for low occupancy time a maximum of 1000 Bq m^{-3} . Since regulating radon concentrations requires a large number of measurements some method is necessary to optimize the allocation of the limited resources available for each country. The indoor radon concentration and the exposure from radon are dependent on many factors, but an assumption can be made that geology is a major control on the variation of indoor radon. This radon potential can be described by many different ways.

3.3.1 Geogenic radon potential concepts

The EU BSS describes radon-prone areas as a geographic area or administrative region where surveys indicate that the percentage of dwellings expected to exceed national reference levels is significantly higher than in other parts of the country. This is a good concept for national regulations, however it can't be used across borders, it is highly dependent on national regulations and gives only a sense of risk related to the average concentration of the particular country, as high or as low it may be. One of the often (Gruber et al. 2013, Szabó et al. 2014, Bossew 2015, Pásztor et al. 2016 see [1]) used methods assessing the geogenic radon potential is the continuous variable originally proposed by Neznal et al. 2004. , Based on research conducted in the Czech Republic, three categories of GRP were set: low ($GRP < 10$), medium ($10 < GRP < 35$) and high ($35 < GRP$) (Szabó et al. 2014 see [1]). In practice there are some variations on providing the values for c and k (Gruber et al. 2013 see [1]). If c and k values are not available, then the radon potential is usually estimated from proxies. Such proxies are the standardized indoor radon concentration (measured in defined standard conditions such as ground floor rooms, presence of a basement, etc. to 'factorize out' anthropogenic factors) The standardized indoor radon concentration is correlated to the GRP, with inaccuracies caused by remaining factors unaccounted for or poorly assessed. Other quantities such as equivalent uranium (eU) or dose rate have similarly describable relations to the GRP, however these relations can be locally different, according to the regional predominance of some factors. The controlling factors have to be taken into account

when using substitutes for the soil radon in the formula (Gruber et al. 2013, see [1]). A different way of defining radon potential is based on multivariate cross-tabulation. This method results in an index with a categorical-ordinal quantity, the results are given in classes such as (I, II, III, IV) or (low, medium, high). Classes are assigned based on scores either assigned to a combination of input quantities or calculated as the sum of points delegated to the input quantities. The second type allows for the consideration of multiple factors. Available quantities are soil radon, permeability, standardized indoor concentration, equivalent uranium concentration or other geochemical quantities, external terrestrial gamma dose rate, geological categories, quantities related to tectonics, and the presence of 'special features' like mines, caves, water bodies and other extraordinary conditions, which are coded binary (yes, no) . For compiling maps, similarly to the definition, several options exist. First the definition of the target variable has to be decided upon. Then the mentioned variable has to be matched to spatial units (area), which will serve as the basis of the map. These spatial units can take various shapes and forms such as administrative or geological units or a grid cells. Geographical units might be a practical choice for the radon potential, and if desired those units can be decompiled into a grid system. The spatial units are then assigned a value derived from the measured target variables inside (arithmetic mean, geometric mean, median, etc.) (Gruber et al. 2013 in [1]). If insufficient data is available for the mean calculation to be representative of the area that technique shouldn't be used. Various estimation or interpolation techniques (local regression methods, disjunctive kriging, Bayesian inference or extensive Monte Carlo simulations) can be implemented during the construction of such maps, but it should be kept in mind that the interpolated concentration is only an estimate, not the actual radon concentration, even though it can be useful for the visualization of the data and in defining areas with higher risk probability (Cafaro et al. 2014, see [1]). The different spatial units offer different advantages and disadvantages. Administrative boundaries make administrative action easier, but disregard the relation between the radon potential and the geology and soil properties. Grids makes mapping independent from other variables, but ignores variation within the grid cells. Geological boundaries are much more closely related to the radon potential but still there can be variations in the radon potential inside the geological units (Ielsch et al. 2010 see [1]).

3.4 Cosmic rays:

Cosmic radiation is a source of background for just about any non-accelerator experiment. Primary cosmic rays are about 90% protons, 9% alpha particles, and the rest heavier nuclei. They are totally attenuated within the first few hundreds g/cm^2 of atmospheric thickness. At sea level secondary particles ($\pi^\pm : p : e^\pm : n : \mu^\pm$) are observed with relative intensities 1 : 13 : 340 : 480 : 1420.

All but the muon and the neutron components are readily absorbed by overburden such as building ceilings and passive shielding. Only if there is very little overburden ($\lesssim 10 g/cm^2$) do pions and protons need to be considered when estimating the production rate of cosmogenic radioactivity.

Sensitive experiments are thus operated deep underground where essentially only muons can penetrate. The muon intensity falls off rapidly with depth. Active detection systems, capable of tagging events correlated in time with cosmic-ray activity, are needed, depending on the overburden. The muonic background is related to low-radioactivity techniques insofar as photonuclear interactions with atomic nuclei can produce long-lived radioactivity directly or indirectly via the creation of neutrons. This happens at any overburden, however, at strongly depth dependent rates. Muon bremsstrahlung, created in high-Z shielding materials, contributes to the low energy background too. Active muon detection systems are effective in reducing this background, but only for activities with sufficiently short half-lives, allowing vetoing with reasonable detector dead time.

Cosmogenic activation of detector components at the surface can be an issue for low-background experiments. Proper management of parts and materials above ground during manufacturing and detector assembly minimizes the accumulation of long-lived activity. Cosmogenic activation is most important for intermediate-Z materials such as Cu and Fe. For the most demanding applications, metals are stored and transported under sufficient shielding to stop the hadronic component of the cosmic rays. Parts can be stored underground for long periods before being used. Underground machine shops are sometimes used to limit the duration of exposure at the surface. Some experiments are even electro-forming copper underground.

3.5 Neutrons:

Neutrons contribute to the background of low-energy experiments in different ways: directly through nuclear recoil in the detector medium, and indirectly, through the production of radio-nuclides, capture γ s and inelastic scattering inside the detector and its components. The indirect mechanisms allow even remote materials to contribute to the background by means of penetrating γ radiation. Neutrons are thus an important source of low-energy background. They are produced in different ways:

1. At the earth's surface the flux of cosmic-ray secondary neutrons is exceeded only by that of muons;
2. Energetic tertiary neutrons are produced by cosmic-ray muons by nuclear spallation in the detector and laboratory walls;
3. In high- Z materials, often used in radiation shields, nuclear capture of negative muons results in the emission of neutrons;
4. Natural radioactivity has a neutron component through spontaneous fission and (α, n) -reactions.

A calculation with the hadronic simulation code FLUKA, using the known energy distribution of secondary neutrons at the earth's surface, yields a mass attenuation of 1500 g/cm^2 in concrete for secondary neutrons. In case energy-dependent neutron-capture cross sections are known, such calculations can be used to obtain the production rate of particular radio-nuclides.

At an overburden of only few meters water equivalent, neutron production by muons becomes the dominant mechanism. Neutron production rates are high in high- Z shielding materials. A high – Z radiation shield, discussed earlier as being effective in reducing background due to external radioactivity, thus acts as a source for cosmogenic tertiary high-energy neutrons. Depending on the overburden and the radioactivity content of the laboratory, there is an optimal shielding thickness. Water shields, although bulky, are an attractive alternative due to their low neutron production yield and self-shielding.

Shields made from plastic or water are commonly used to reduce the neutron flux. The shield is sometimes doped with a substance having a high thermal neutron capture cross section (such as boron) to absorb thermal neutrons more quickly. The

hydrogen, contained in these shields, serves as a target for elastic scattering, and is effective in reducing the neutron energy. Neutrons from natural radioactivity have relatively low energies and can be effectively suppressed by a neutron shield. Ideally, such a neutron shield should be inside the lead to be effective for tertiary neutrons. However, this is rarely done as it increases the neutron production target (in form of the passive shield), and the costs increase as the cube of the linear dimensions. An active cosmic-ray veto is an effective solution, correlating a neutron with its parent muon. This solution works best if the veto system is as far away from the detector as feasible (outside the radiation shield) in order to correlate as many background-producing muons with neutrons as possible. The vetoed time after a muon hit needs to be sufficiently long to assure muon bremsstrahlung and neutron-induced backgrounds are sufficiently suppressed. An upper limit to the allowable veto period is given by the veto-induced deadtime, which is related to the muon hit rate on the veto detector. This consideration also constitutes the limiting factor for the physical size of the veto system (besides the cost). The background caused by neutron-induced radioactivity with live-times exceeding the veto time cannot be addressed in this way. Moving the detector deep underground, and thus reducing the muon flux, is the only technique that addresses all sources of cosmogenic neutron background.

4 Gaseous ionization detectors

4.1 Introduction

Gaseous ionization detectors are designed to measure the ionization produced when an incident particle traverses some medium. For ionizing radiation to occur, the kinetic energy of particles (photons, electrons, etc.) of ionizing radiation is sufficient and the particle can ionize (to form ion by losing electrons) target atoms to form ions. Basic principle of gaseous ionization detectors is illustrated by Fig. 4.1.

The W-value is defined as the average energy lost by the incident particle per ion pair formed. Due to the competing mechanism of the energy loss, i.e. excitation, W-value is always greater than the ionization energy.

Table 4.1: Ionization potential and W-values for some gases for electrons.

Gas	1 st Ionization Potential (eV)	W-Value (eV/ion pair)
Ar	15.7	26.4
He	24.5	41.3
N ₂	15.5	34.8
Air		33.8

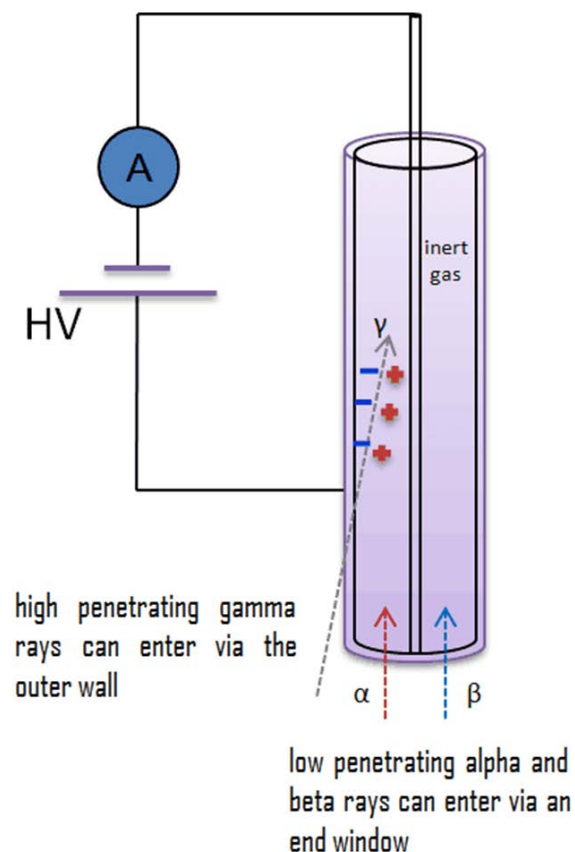


Figure 4.1: Principal scheme of gaseous ionization detectors

An electric field should be applied to collect electron-ion pairs produced in a gas-filled detector.

Depending on the electric field strength, operation of gas-filled detectors can be divided into several regions (Fig. 4.2).

Ionization region:

The electric field is strong enough to prevent recombination and collect all the charges efficiently. The steady state current measured at this condition is called ionization current, which is the basic principle of the DC ion chamber.

Proportional region:

In this region of applied voltage, the voltage is above the threshold for gas multiplication. The multiplication is linear; the collected charge is proportional to the number of ion pairs created by the incident radiation. At fixed applied voltage the measured pulse amplitude (or height) is proportional to incident particle energy.

The Geiger Müller region:

The avalanche develops until positive ions created reduce the electric field below the point at which the gas multiplication can take place. The avalanche terminates when the same total numbers of positive ions are created, and each pulse is of the same amplitude which does not reflect the energy absorbed.

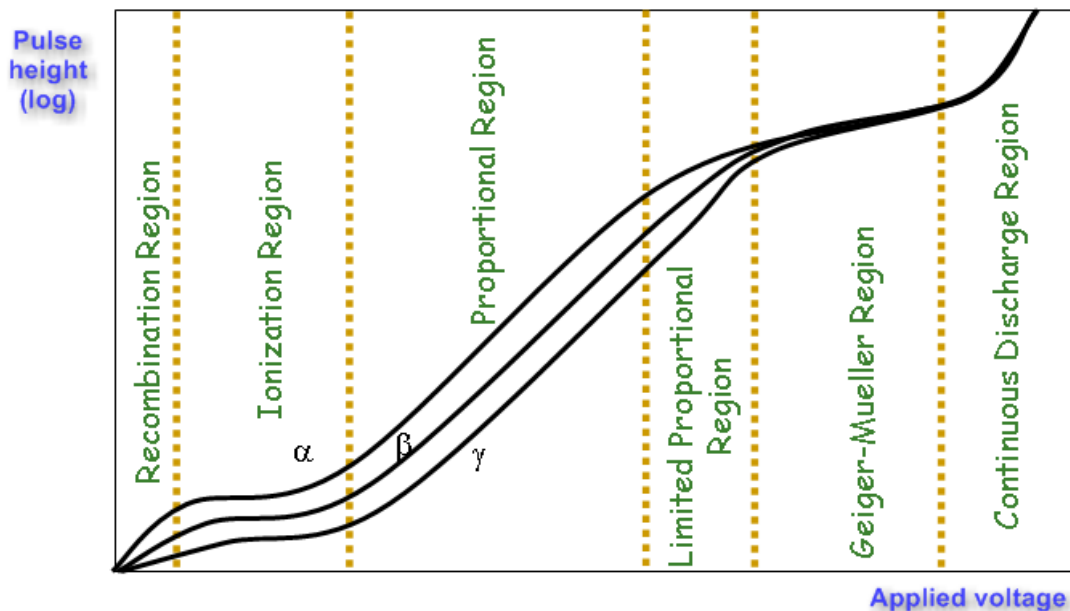


Figure 4.2: Operation regions of gas-filled detectors.

4.2 Operation principles of DC ion chamber

The basic principle of the DC ion chamber is the measurement of the steady state current which is called ionization current. In direct current mode, the negative charges are being collected either as free electrons or as negative ions.

Typical ionization current magnitude in most applications is small (less than 1 pA) and a highly sensitive electronic voltmeter measures the current indirectly by sensing the voltage drop across a series (Fig. 4.3). If the ionization current does not change

during the several values of the time constant RC , its steady state value is given by $I = V_R/R$. ([5] Med Phys 4R06/6R03 Radioisotopes and Radiation Methodology).

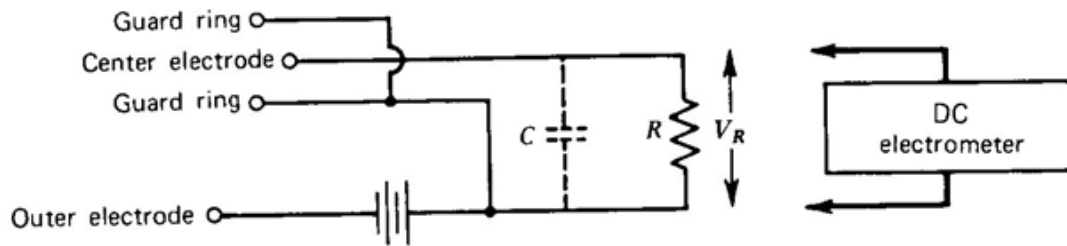


Figure 4.3: Low current measurement with a DC electrometer [5].

4.3 Principles of Geiger Müller counters

Main advantage of these instruments is that they usually do not require any signal amplifiers. Since the positive ions do not move far from the avalanche region, a positively charged ion cloud compensates the electric field and terminates the avalanche process. In contrast to proportional counters, the energy or even incident radiation particle cannot be distinguished by Geiger Müller counters, since the output signal is independent of the amount and type of original ionization.

The block diagram of a counting system for a Geiger Müller counter is shown in Fig. 4.4. Because no preamplifier is required, the counting system is simplest. The parallel combination of the load resistance R and the capacitance of the detector C_s (including associated wiring) give the time constant ($2 \sim 3$ microseconds) of the charge collection circuit. Thus, only the fast-rising component of the pulse is preserved. The coupling capacitor C_c blocks the high voltage, but transmits the signal at ground potential.

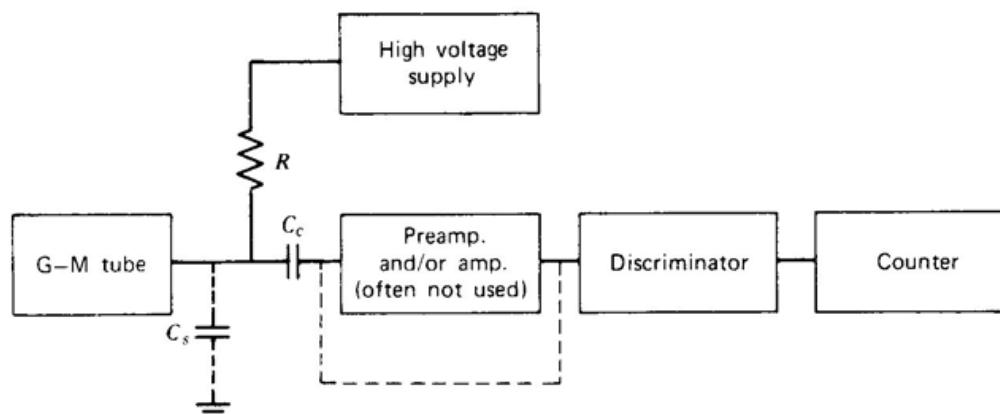


Figure 4.4: The block diagram of an electronics for counting pulses from a G-M tube[5].

5 Radiation Dose Measurements at AREAL

5.1 Introduction

Strong motivation for radiation measurement is determined by needs of radiation safety, accelerator and beam physics as well as by radiation related investigations on material science and life science. Radiation dose measurements allow monitoring and control of radiation levels.

Radiation dose measurement results provide the basis for construction of radiation protection walls and shields and for the development of safety procedures. Radiation dose measurements can be useful for the determination of beam parameters along with measurements by tools and detectors of the beam diagnostic system.

Combined with the beam diagnostic and the numerical particle transport simulation code FLUKA, results of radiation dose measurements enable one to calculate experimental irradiation parameters for several samples of material science and biological experiments that are being conducted on the AREAL facility.

5.2 AREAL radiation safety

Along with the RF electron gun, two acceleration structures will permit to reach electron beam energy of 20 MeV. The laser driven photo gun capability limits the beam current to 0.2 nA for both single and multi-bunch modes of the LINAC operation. An straight beam line is planned, without beam compressors and beam turns, making the beam dump at the end the sole source of secondary neutron and gamma radiation that safety system should cope with.

Table 5.1: LINAC parameters relevant to radiation safety.

Bunch charge	200 pC
Electron energy at the gun exit	5 MeV
Electron final energy	20 MeV
Beam current	0.2 nA
Beam final mean power	4 mW
Beam distance from the wall	1.5 m
Beam distance from the ceiling	2.2 m

5.3 Radiation by electrons:

The major radiation loss components of the 20 MeV electrons are:

- Giant-Resonance-Neutrons: Those neutrons with energy range between 0.1MeV and 20MeV are produced by the bremsstrahlung photons in the core of the electromagnetic shower.
- Direct Gamma: Photons with energies between 0.1MeV and 20MeV leaving the shower core.

High-energy electrons predominantly lose energy in matter by bremsstrahlung, and high-energy photons by $e^+ e^-$ pair production. The characteristic amount of matter traversed for these related interactions is called the radiation length R_l , usually measured in $[g/cm^2]$. It is both the mean distance over which a high-energy electron loses all but $1/e$ of its energy by bremsstrahlung, and $7/9$ of the mean free path for pair production by a high-energy photon. It is also the scale length for describing high-energy electromagnetic cascades.

The radiation length for a given material consisting of a single type of nuclei can be approximated (compact fit to the data) by the following expression:

$$R_l = \frac{716A}{Z(Z+1)\ln(287/\sqrt{Z})} [g/cm^2], \quad (5.1)$$

where A is the atomic mass and Z is the atomic number of the absorber nucleus.

For electrons at lower energies (below few tens of MeVs), the energy loss by ionization is predominant. An electron loses energy by bremsstrahlung at a rate nearly proportional to its energy, while the ionization loss rate varies only logarithmically with the electron energy. The critical energy E_c is sometimes defined as the energy at which the two loss rates are equal. The value of the critical energy when bremsstrahlung starts to prevail over ionization mechanism can be obtained by the expression

$$E_c = \frac{610MeV}{Z + 1.24} \quad (5.2)$$

for solids and liquids.

The Molière radius is a characteristic constant of a material giving the scale of the transverse dimension of the fully developed electromagnetic showers initiated by an incident high energy electron or photon. By definition, it is the radius of a cylinder

containing on average 90% of the shower's energy deposition. It is related to the radiation length R_l by the following approximate formula:

$$R_M = 0.0265 R_l (Z + 1.2), \quad (5.3)$$

where Z is the atomic number.

5.4 Beam Dump; Standard-Target:

Dose and shielding calculations presented are based on the algorithms and formulae borrowed from the SHIELD11 computer code.

The electron beam will be terminated by the beam dump target, which is an iron cylinder surrounded by shielding. The parameters of some common beam target materials are presented in **table 5.2**. For electrons radiation losses in iron the critical energy is 22.4 MeV (Formula 5.2). That means that 20 MeV electrons will lose about half of their initial energy through bremsstrahlung, while the ionization mechanism accounts for the rest.

The empiric formulae used by the SHIELD11 code are derived on the basis of the numerous measurements and particle tracking simulations applied to the standard target. The standard-target is a 12-inch long iron cylinder with 2-inch radius. Those dimensions correspond to 17.3 radiation length and 3.74 Moliere radii.

Secondary neutrons and gamma radiation will be produced also when beam electrons scrape the inner surface of the beam pipe or hit pieces of the beam diagnostics equipment. However, since no bunch compressors and beam deflections are foreseen, the main concern in terms of secondary radiation remains the beam dump.

Table 5.2: Material parameters for concrete, iron and lead built in and used in SHIELD11, effective atomic number: Z , atomic mass: A , density: ρ [g/cm³], radiation length: R_l [g/cm²], Moliere radius R_M : [g/cm²].

	Concrete	Iron	Lead
Z	13	26	82
A [g/mole]	26.98	55.85	207.19
ρ [g/cm ³]	2.35	7.87	11.35
R_l [g/cm ²], ([cm])	26.7, (11.36)	13.84, (1.76)	6.37, (0.561)
R_M [g/cm ²], ([cm])	11.1, (4.72)	10.7, (1.36)	14.2, (1.25)
E_c [MeV]	42.8	22.4	7.33

5.5 Giant-Resonance-Neutrons:

Giant-Resonance-Neutrons are emitted nearly isotropically in angle. The neutron yield per unit power loss is proportional to the beam energy and depends on the target material as $Y = 0.121Z^{0.662}$ where the yield is expressed in units [10¹² n/s/kW]. To obtain the dose at 1 cm distance from the source one should multiple it by the factor $3.2 \times 10^{-10} \text{ Sv} \times \text{cm}^2/\text{n}$:

$D_{ne} = 4.93 Z^{0.662} (E \times 10^{-16}) [Sv/e]$ where E is the beam energy in MeV. For a beam energy of 20 MeV and a current of 0.2 nA one gets neutron dose rate at the dump target

$$D_n = 1.775 Z^{0.662} \frac{EI}{q_e} \times 10^{-13} = 0.3835 \text{ Sv/h.} \quad (5.4)$$

5.6 Direct Gamma:

The angular and energy distribution of the gamma radiation emitted directly in the core of the shower at the radiation source can be expressed by the following formula [2]:

$$D_\gamma = \left(\frac{EI}{q_e} \right) (3.031 E e^{-0.959\sqrt{\vartheta}} \times 10^{-13} + 7.5 e^{-\vartheta/72.2} \times 10^{-8}), \quad (5.5)$$

where D_γ is the gamma dose rate at the radiation source in units of [Sv/h], E is the beam energy in [MeV] and ϑ is the radiation angle in degrees with respect to the beam direction. Putting in $E = 20 \text{ MeV}$ and $\vartheta = 90$ degrees one gets $D_\gamma = 0.5378 \text{ Sv/h}$. The first term in formula (5.5) describes radiation emitted in forward direction ($\vartheta = 0^\circ - 5^\circ$), while the second term describes lateral and backward radiation. Angular and energy dependence of the direct gamma radiation dose rate near the target is illustrated in **Fig. 5.1**.

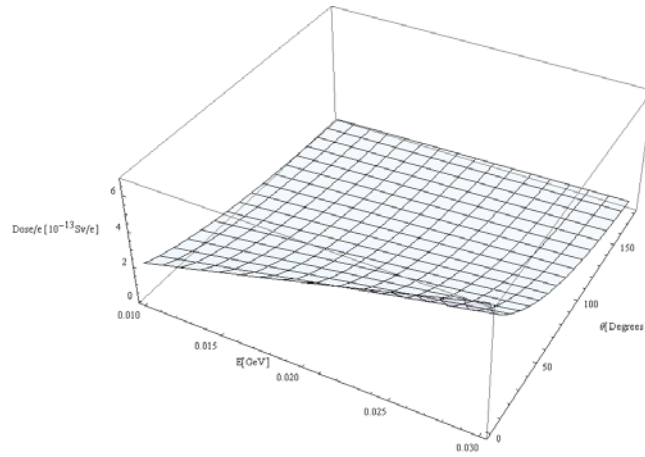


Figure 5.1: Direct gamma radiation dose per electron at standard target. Maximum of radiation lays in forward direction and increases with electron energy nearly linearly.

5.7 Shielding:

The shielding has to be dimensioned to keep the ambient dose equivalent rate below the limit of 0.125 $\mu\text{Sv/h}$ (the sum of both, the neutron-dose and γ -dose rates). The parameters of some common shielding materials are given in the Table 5.3. The mean free path λ [g/cm^2] is the parameter that defines the materials' ability to absorb radiation $F = F_0 e^{-\lambda/d}$, where F is the fluence and d [g/cm^2] is the material thickness. Neutrons are efficiently absorbed by concrete, since it contains hydrogen (in bounded water molecules). For the attenuation of gamma radiation, high Z materials (like lead) are being applied (formulae 5.1-5.3 and Table 5.3).

Table 5.3: Removal free path for some shielding materials

	Concrete	Fe	Pb
ρ [g/cm^3]	2.35	7.87	11.35
Neutrons [g/cm^2], ([cm])	30, (12.8)	47, (5.97)	97, (8.55)
Gamma [g/cm^2], ([cm])	42, (17.9)	33.6 (4.27)	24, (2.11)

5.8 Tunnel walls:

The distance between the straight beam pipe and the nearest tunnel wall will be 150 cm. Any air gap will reduce the dose rate outside the tunnel significantly, since both neutron and gamma lateral radiation are isotropic and thus decrease proportionally to $\sim 1/L^2$, where L is the distance from the source. Thick concrete walls screen the area outside from the radiation emitted not only at beam dump target but also from the radiation produced when the beam hits the vacuum chamber, slit walls, etc. To ensure 0.125 $\mu\text{Sv/h}$ ambient dose, 90cm total width of concrete shielding is

necessary (Fig.5.2). Introduction of only 10 cm lead as additional shielding decreases the necessary concrete thickness to 50 cm (Fig. 5.3).

5.9 Beam dump shielding

The beam dump will consist of iron core with lateral and downstream shielding. The iron target is a 30 cm long cylinder with 5 cm thickness. 10 cm lead shields surround the iron core (laterally and downstream). A 50 cm thick concrete block will constitute as the outer shielding of the beam dump.

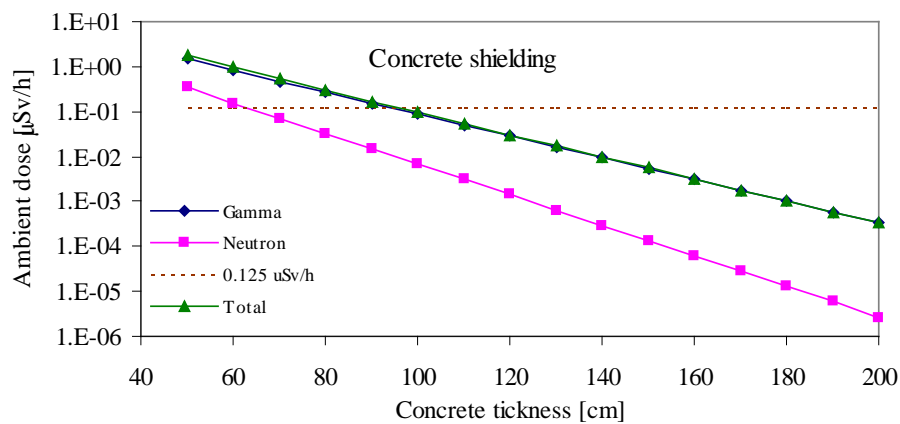


Figure 5.2: Ambient dose outside the tunnel vs. concrete wall thickness. The dotted line shows the target level of equivalent dose, i.e. $0.125 \mu\text{Sv}/h$.

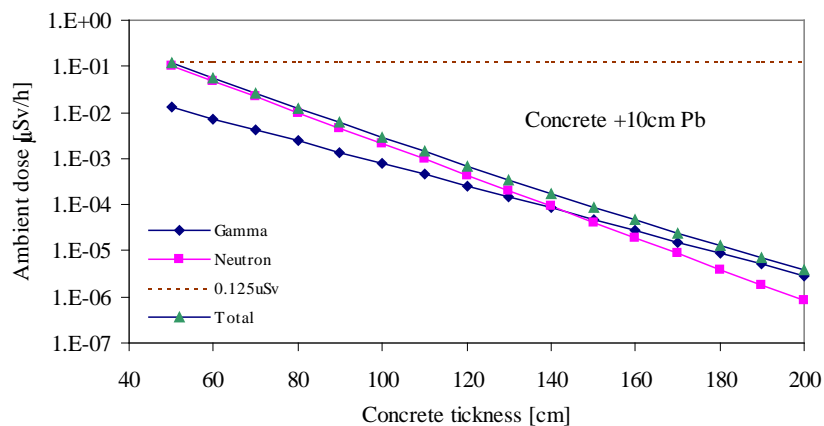


Figure 5.3: Ambient dose outside the tunnel vs. concrete wall thickness. The dotted line shows the target level of equivalent dose, i.e. $0.125 \mu\text{Sv}/h$. In this case, additional 10cm of lead shielding around the beam dump target effectively reduces the gamma dose.

5.10 Equipment

5.10.1 Gamma-scout dosimeter

Geiger-Muller counter tube based alpha, beta gamma radiation meter
GAMMA_SCOUT with data reading and display software

Display: Liquid-crystal display (LCD), 4-digit, numeric with description, quasi-analogue logarithmic bar chart,

Operating mode indicators

Radiation Detector: End-window counting tube according to the Geiger-Müller principle

Stainless steel housing,

Measuring length 38.1 mm, measuring diameter 9.1 mm,

Mica window 1.5 to 2 mg/cm²

Zero rate <10 pulses per minute with screening by 3mm Al and 50 mm Pb,
operating temperature -20 to +60°C,

operating voltage approx. 450 V calibrated scale from 0.01 μSv/h to 5,000.00 μSv/h

Radiation Types: α from 4 MeV

β from 0.2 MeV

γ from 30 keV

Selection shield: $\alpha + \beta + \gamma$ without shielding

$\alpha + \beta$ Al foil approx. 0.1 mm, shields α completely

γ Al shield of approx. 3 mm, shields α completely and β up to 2 MeV,

weakens γ by less than 7% based on Cs-137

Power Consumption: Less than 10 microamperes under environmental radiation

Memory: 256,000 byte (100,000 data sets)

Housing: Impact-resistant plastic

Dimensions: Length 163 mm x width 72 mm x height 30 mm.

5.10.2 Ionization chamber based radiation area survey meter STEP OD-01 with software

Measuring value:	Ambient dose equivalent $H^*(10)$, Ambient dose rate equivalent $dH^*(10)/dt$, Directional dose equivalent $H'(0.07, \Omega)$, Directional dose equivalent $dH'(0.07, \Omega)/dt$,
Type of radiation:	Photon and beta radiation Mixed radiation fields Pulsed radiation fields
Display ranges:	Dose rate 0 μ Sv/h .. 2000 mSv/h Dose 0 .. 2000 μ Sv
Energy range:	Photon energy 6 keV to 7.5 MeV (15 MeV)(1) Beta energy 60 keV to 2 MeV Radiation direction referred to $\pm 45^\circ$ for $H'(0,07)$ preferential direction (axial) $\pm 90^\circ$ for $H^*(10)$
Detector type:	Air-opened ionisation chamber, Volume 600 cm ³ , PMMA-Shielding disposable, 550 mg/cm ² Face sided entry window 3.3 mg/cm ² (metal covered PETP foil) Preferential direction/point of reference axial, marked on detector Wall potential + 400 V (μ Sv, mSv/h) + 40 V (μ Sv/h)
Measurement uncertainties:	Linearity < 15 % (fine measurement range 20) \pm 10 % (fine measurement ranges 200 and 2000)
Warm-up time:	2 minutes
Power Supply:	Batteries, Accumulators 3Current consumption approx. 30 mA @ 6 V Battery lifetime approx. 100 h
Display unit:	3 1/2 – digit LC-display, back lighted Digit height: 12 mm Additional bar display Automatic switch for fine measurement range decades
Air pressure:	80 to 110 kPa
Relative humidity:	max. 80 %



Gamma-scout dosimeter, general view



STEP OD-01 survey meter

Calculation and simulation software: FLUKA, SHIELD11, MATHLAB MATEMATICA, MS EXCEL.

5.11 Practical hours

Task 1:

Measurements of the background dose rates in the AREAL machine hall and neighboring premises using Geiger-Muller counter and Ionization chamber survey meter;

In measuring mode of the GAMMA_SCOUT dosimeter (press the corresponding button), the display shows the current radiation measurement reading every 2 sec. GAMMA-SCOUT stores all registered pulses in its internal memory and keeps them ready for you to use when desired. Stored data in the memory can be read out and processed using GAMMA_SCOUT TOOLBOX software. Use mini USB cable to connect the dosimeter with the PC USB port and transfer the measurement data to the computer for visualization and processing.

Prior to measurement the zero of the STEP OD-01 device should be verified. Electrical zero can be considered as balanced, if the display indicates a value in the range of $0 \leq 0.05$. The meter switch must be set at the measuring range μSv . Note that after switching between the ranges it takes about 2 min before the internal circuit

come to balance and the device is ready for the measurements. Use the USB cable provided to connect the dosimeter to the PC, run dedicated software (that should be installed on PC) for the reading out, visualization and processing of the measurement data. Try to interpolate some possible discrepancy between the data obtained by two principally different type of dosimeters relying on the differences of the technical characteristics (energy ranges, sensitivities, etc.).

Task 2:

Search for high levels of Radon concentration in the basement rooms applying the GAMMA_SCOUT dosimeter;

Survey the basements rooms using the GAMMA_SCOUT dosimeter with the Geiger-Muller counter. Measure the background dose at several fixed positions with the counter window open (put the switch on the position $\alpha + \beta + \gamma$), then turn the switch to the position $\beta + \gamma$ thus preventing α particles to reach the counter window and repeat the measurement. The duration of the single measurement should be at least 2 minutes to gather some statistics. If the measured dose value in the first regime is significantly higher than that in the second regime, it can be interpreted as a clear indication of higher level of Radon concentration [8]. Note that the dosimeter is sensitive to α particles with energy more than 4 MeV and displays the results starting 10 counts per minute.

For the absolute measurement of the Radon concentration, elaborated techniques should be applied and that problem is beyond the scope of the current work.

Task 3:

Perform measurements of the dose rates during the AREAL accelerator operation by the ionization chamber at several fixed positions in the machine hall and compare with the FLUKA simulation results. A protection barrier/wall with single and then with two layer of concrete bricks should be assembled between beam target and dosimeter to demonstrate radiation protection by increased shielding.

Task 4:

For the radiation loss study of the AREAL beam at the target with various materials and different thickness measurement of the dose rate dependence on the beam current measured by the Faraday Cup should be conducted. It is anticipated to use

several 0.5 mm thick Al sheets and 1cm - 2cm long water filled polyethylene containers as targets.

Task 5:

Experimental study of the dose rate dependence on the beam energy as measured in the dispersion section using a bending magnet and a YAG screen station.

Remarks:

After detailed discussion of the course it was suggested to use a beam diaphragm to center the beam by minimizing the beam losses; compare the result of this attempt with the beam profile team in terms of their result on the beam size; compare results of measurement of background dose rates in the AREAL machine hall and environment; search for Radon concentration in the basement rooms; compare measurements with FLUKA simulation results; demonstrate radiation protection by increased shielding.

Add the radiation loss study of the AREAL beam at the target with various materials and different thickness provided.

Literature^{*}

1. M. Tanabashi et al. (Particle Data Group), Phys. Rev. D 98, 030001 (2018). (<http://pdg.lbl.gov>).
2. <http://www.fluka.org/>.
3. W.R. Nelson, The SHIELD11 Computer Code, T.M. Jenkins, SLAC-R-737, UC-414.
4. International Commission on Radiation Units and Measurements, Fundamental Quantities and Units for Ionizing Radiation, ICRU Report 60 (1998).
5. Gas Filled Detectors in Radioisotopes and Radiation Methodology, Med Phys 4R06/6R03.

^{*}ALL items in the list are available online.

APENDIX A: Radiological quantities and units

Physical quantities

The fluence (unit: $1/m^2$) is the number of particles incident upon a small spheric cross-sectional area.

It can be shown that the fluence is given by the sum of the particle trajectory lengths in the unit volume.

The Absorbed dose, (unit: gray, $1 \text{ Gy} = 1 \text{ J/kg} = 100 \text{ rad}$) is the energy deposited by ionizing radiation in a volume element of a material divided by the mass of this volume element.

The linear energy transfer, L or LET (unit: J/m, often given in $\text{keV}/\mu\text{m}$) is the mean energy, lost by a charged particle in traversing a unit distance in matter. Low-LET radiation: x rays and gamma rays or light charged particles such as electrons that produce sparse ionizing events at a molecular scale ($L < 10 \text{ keV}/\mu\text{m}$).

High-LET radiation: neutrons and heavy charged particles that produce ionizing events densely spaced at a molecular scale ($L > 10 \text{ keV}/\mu\text{m}$).

Protection quantities

Protection quantities are dose quantities that quantify the extent of exposure of the human body to ionizing radiation.

The equivalent dose (unit: Sievert, $1 \text{ Sv} = 100 \text{ rem}$) in an organ or tissue is equal to the sum of the absorbed doses in the organ or tissue caused by different radiation types weighted with radiation weighting factors. It expresses long-term risks (primarily cancer and leukemia) from low-level chronic exposure. The values for w_R recommended by ICRP [4] are given in Table 1.

Table 1: Radiation weighting factors, w_R .

Photons, electrons and muons	1
Neutrons, $E_n < 1 \text{ MeV}$	$2.5 + 18.2 \cdot \exp[-(\ln E_n)^2/6]$
Neutrons, $1 \text{ MeV} \leq E_n \leq 50 \text{ MeV}$	$5.0 + 17.0 \cdot \exp[-(\ln(2E_n))^2/6]$

The effective dose, (unit: Sievert) is the sum of the equivalent doses, weighted by the tissue weighting factors of several organs and tissues of the body that are considered to be most sensitive.

Operational quantities

The operational quantities are used for the assessment of effective dose or mean equivalent doses in tissues or organs, since protection quantities are not measurable in practice.

Ambient dose equivalent, (unit: Sievert): The dose equivalent at a point in a radiation field that would be produced by the corresponding expanded and aligned field in a 30 cm diameter sphere of unit density tissue (ICRU sphere) at a depth of 10 mm on the radius vector opposing the direction of the aligned field. Ambient dose equivalent is the operational quantity for area monitoring.

Personal dose equivalent, (unit: Sievert): The dose equivalent in ICRU tissue at an appropriate depth, below a specified point on the human body. The specified point is normally taken to be where the individual dosimeter is worn. For the assessment of effective dose, with a depth 10 mm is chosen, and for the assessment of the dose to the skin and to the hands and feet the personal dose equivalent, with a depth 0.07 mm, is used. Personal dose equivalent is the operational quantity for individual monitoring.

APENDIX B: Background radiation sources

The worldwide average of the annual whole-body dose equivalent due to all sources of natural background radiation ranges from 1.0 to 13 mSv with an annual average of 2.4 mSv (that corresponds to the rate of 0.27 $\mu\text{Sv/h}$) [1]. In certain areas values of 50 mSv have been measured. Typically more than 50% originate from inhaled natural radioactivity, mostly radon and radon daughters. Dose equivalent rates due to cosmic ray background radiation range from less than 0.1 $\mu\text{Sv/h}$ at sea level to a few $\mu\text{Sv/h}$ at aircraft altitudes. Artificial sources (medical, cigarettes, air travel, building materials, etc.) contribute 0.61 mSv.

Main sources for background radiation are shown in the table below. This classification can be retrieved from the two following documents:

- a) United Nations Scientific Committee on the Effects of Atomic Radiation (2008 (published 2010)). Sources and effects of ionizing radiation. New York: United Nations. p. 4.
- b) Ionizing radiation exposure of the population of the United States. Bethesda, Md.: National Council on Radiation Protection and Measurements. 2009.

Table 2: Main sources of background radiation. The numbers indicate annual averages in mSv units.

Radiation source	World	USA	Remark
Inhalation of air	1.26	2.28	mainly from radon, depends on indoor accumulation
Ingestion of food & water	0.29	0.28	(K-40, C-14, etc.)
Terrestrial radiation from ground	0.48	0.21	depends on soil and building material
Cosmic radiation from space	0.39	0.33	depends on altitude
sub total (natural)	2.40	3.10	sizeable population groups receive 10-20 mSv
Medical	0.60	3.00	world-wide figure excludes radiotherapy; US figure is mostly CT scans and nuclear medicine.
Consumer items	-	0.13	cigarettes, air travel, building materials, etc.
sub total (artificial)	0.61	3.14	
Total	3.01	6.24	millisievert per year

APENDIX C: Occupational exposure

The ICRP recommends limiting occupational radiation exposure to 50 *mSv* per year, and 100 *mSv* in 5 years. In 2002 IAEA recommended that occupational doses below 1–2 *mSv* per year do not warrant regulatory scrutiny. The limit in the EU-countries and Switzerland is 20 *mSv* per year, in the U.S. it is 50 *mSv* per year. The effective dose limit for general public is typically 1 *mSv/year*.