17 Radiation Protection at an Accelerator Laboratory

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17.1 Introduction

The phenomenon called "radiation" is the transport of energy in the form of a stream of atomic particles or electromagnetic quanta (photons). No supporting medium is required. Radiation can be divided into ionizing and nonionizing. Ionizing radiation has a higher energy than nonionizing. (As a rule of thumb, ionizing radiation has an energy of the order of atomic or molecular binding energies, that is, 10 eV or higher, while nonionizing radiation has an energy below 10 eV.) Ionizing radiation can – as the name suggests – ionize material when interacting. Ionizing means that electrons are removed from the atoms/molecules in the material by the radiation. In this way, charged particles, i.e. ions, are produced. If this happens in a human body, radiation injuries can result. The interaction of radiation with matter is discussed in Sect. 17.2 and its consequences in a human body are discussed in Sect. 17.4.

Ionizing radiation is not a new phenomenon connected with human activity. It has always been available and is present throughout the environment. However, it is only during the last century that man has learned to detect ionizing radiation and to produce artificial ionizing radiation. In Sect. 17.3, the quantities and units used within the field of radiation protection are discussed, and in Sect. 17.5, detectors for ionizing radiation are briefly presented. In Sect. 17.6, dose measurements are outlined.

Many accelerators are used at low intensities and have therefore normally a low radiation level. However, it should be remembered that in general any accelerator can produce hazardous levels. Even if the ion (or electron) source is switched off, stray electrons can be accelerated over the high-voltage gap, producing bremsstrahlung when hitting material. The radiation hazards for different types of accelerators and for different kinds of radiation are discussed in Sect. 17.7.

For standard use with low intensities, only minimal shielding is normally required. This could be a risk, as personnel may become careless if they consider the radiation hazard as negligible. It is therefore important to measure the radiation level whenever an uncertainty exists. Safety considerations in an accelerator laboratory is discussed in Sect. 17.8. Finally, in the Appendices

of this chapter, the regulatory standards for radiation protection, attenuation of photons, and conversion between fluence and dose are given.

In Box 8, the nonradiation hazards (i.e. typical industrial hazards such as high-pressure gas hazards, electrical hazards, and fire and explosive hazards) in an accelerator laboratory are discussed.

17.2 Radiation and Its Interaction with Matter

Ionizing radiation may stem from three different origins: radioactive decay, accelerated particles or extraterrestrial (cosmic) sources. The properties of the more common type of ionizing radiation are summarized in Table 17.1. In the decay of a radioactive nucleus, its surplus energy is transferred to photons or to ionizing particles in a complex manner. The radiation emitted directly from the nucleus can be predominantly electrons (β^--particles), positrons (β^+-particles) or photons (γ-quanta), and for heavier nuclei also ^{4}He ions (α-particles). Part of the surplus energy may support processes causing vacancies in the electron shells outside the nucleus. When these vacancies are refilled, X-ray photons or so-called Auger electrons with discrete energies are emitted from the atom. The yield per decay of individual X-ray photons and Auger electrons, as well as for the primary emitted particle/photon, is fixed for a specific decay, but is only seldom 100%. Thus, the number of photons of a specified energy generated in a radioactive source is only occasionally identical to the number of decays.

The *radioactive decay* of the nucleus is statistical in nature. Therefore, it is impossible to predict when any given nucleus will disintegrate. Extensive experiments on radioactive materials have shown that the decay for a given initial mass of material is accurately exponential:

Table 17.1. Examples of different types of ionizing radiation. Charge is given in units of the elementary charge (1.602×10^{-19} C)

Type	Origin	Process	Charge	Rest Mass (kg)	Energy Spectrum
α-particle	Nucleus	Nuclear decay or reaction	+2	6.664×10^{-27}	Discrete
β^--particle	Nucleus	Nuclear decay	−1	9.110×10^{-31}	Continuous
β^+-particle	Nucleus	Nuclear decay	+1	9.110×10^{-31}	Continuous
γ-ray	Nucleus	Nuclear de-excitation	0	0	Discrete
X-ray	Electron cloud	Atomic de-excitation	0	0	Discrete
Neutron	Nucleus	Nuclear reaction or spontaneous fission	0	1.678×10^{-27}	Continuous or discrete
Fission fragment	Nucleus	Fission	10–30	$1.4 - 2.8 \times 10^{-25}$	Continuous

$$N_t = N_0 e^{-\lambda t} \tag{17.1}$$

which is in accordance with its stochastic nature. Here N_t is the number of independent radioactive nuclei at time t in a sample, N_0 is the number of radioactive atoms present at the beginning of the observation ($t = 0$) and λ is a constant called the disintegration or decay constant. The time interval $t_{1/2}$ during which half of the atoms disappear by decay, denoted the half-life, is given by $t_{1/2} = (\ln 2)/\lambda$. The *activity* of a radioactive material is the number of decays per unit time, and the number of decays per second is a convenient unit of measurement. In the SI system, this unit is called the becquerel (Bq). However, it should be observed that the activity says nothing about the kind of radiation emitted, nor about its energy.

The kinetic energy available from radioactive transformation is at most a few MeV, while electrostatic accelerators may generate electrons and singly charged ions of higher energies, but rarely above 5 and 10 MeV, respectively. The interaction with matter for ionizing particles in the MeV energy range will be very briefly outlined in the following paragraphs. The range of these particles is schematically illustrated in Fig. 17.1 (see also Sect. 17.7.1 and Table 17.6).

Photons lose their kinetic energy to atomic electrons, either partly (a Compton collision) or totally (the photoelectric effect). Photons above 1.02 MeV passing near a nucleus may, additionally, create an electron–positron pair. The large and few energy losses experienced by photons before they are annihilated mean firstly that the attenuation can be described by an exponential expression, $\Phi(x) = \Phi(0)e^{-\mu x}$, and secondly that the number of ion pairs created by a photon itself is vanishly small compared with the number of ionizations caused by the generated photoelectrons, Compton-scattered electrons and pair production electrons. Photons are accordingly denoted as *indirectly ionizing*. (In the expression, Φ is the photon fluence, x is the material thickness and μ is the attenuation coefficient).

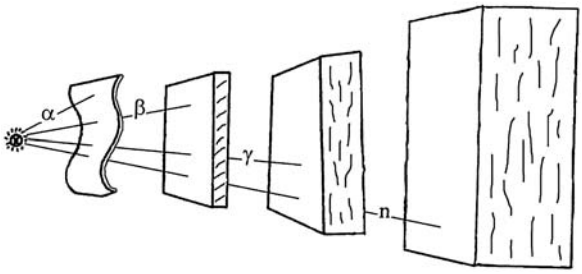

Fig. 17.1. A simple illustration of what is needed to stop energetic α, β, γ and neutron rays: a 0.2 mm sheet of paper, a 100 mm thick piece of wood, half a meter of concrete and a few meters of concrete, respectively

The two major energy loss processes for *electrons* (β-particles) are collision with atomic electrons and bremsstrahlung emission (see Sects. 17.3.2 and 17.7.1). In collisions with atomic electrons, the incident electron may lose up to half of its kinetic energy in a single encounter. Collision losses clearly dominate over bremsstrahlung losses in the electron energy range discussed here, particularly in materials of low atomic number. The fairly long range and irregular path of energetic electrons slowing down in matter means, on a microscale, that the ionization events are separated by distances of the order of a micrometer. Electrons and photons belong to the category of *sparsely ionizing radiation*.

Protons, *deuterons*, ^{4}*He ions* (α-particles) and other heavy charged particles are more than three orders heavier than an electron and can therefore lose only a minute fraction of their kinetic energy even in a head-on collision event with an atomic electron. On the other hand, the probability (cross section) for the event is very large, which means that heavy charged particles are *densily ionizing*. The slowing-down path is short and straight, and losses due to bremsstrahlung are insignificant.

The interaction of *neutrons* with matter is different from that of charged particles, as neutrons readily collide and interact with any nuclei encountered. A neutron is not itself ionizing, but if it hits a nucleus, it may activate it or cause emission of a γ-ray or a charged particle, indirectly giving rise to ionizing radiation. In hydrogenous material, fast neutrons slow down rapidly owing to collisions with protons. In a collision with a heavy nucleus, such as uranium for instance, the neutron loses very little of its kinetic energy. The probability of a neutron capture process is large for some light nuclei, including hydrogen, and in practice neutron radiation is always accompanied by a more or less significant amount of photons. The optimum strategy to eliminate fast neutrons is to use a proton-rich material such as water or concrete to slow them down to thermal energies and then to capture them with a material with a high capture cross section (see Sect. 17.7.2). The penetration of fast neutrons through concrete is higher or comparable to that of 1 MeV photons, and the order of half a meter or more of ordinary concrete may be needed in order to reduce the fast fluence rate by a factor of one hundred. Fundamental details about ionizing radiation and its interaction with matter can be found in [1] and [2], respectively.

17.3 Quantities and Units

The quantities and units used within the field of radiation protection are somewhat impenetrable to the layman owing to the vast range of options of measurable and nonmeasurable, mean-value and stochastic, plain physical and risk-weighted, source-related and target-related, and rate and time-integrated quantities. In this section, only the most essential quantities are mentioned and commented on. Quantities and units used within the radiation

protection field are defined by the International Commission on Radiation Units and Measurements (ICRU), and the interested reader should consult its reports on this matter for further guidance [3, 4].

17.3.1 Radiation Field Quantities

Despite the fact that ionizing radiation is quantized and its interaction is stochastic in nature, the radiation field and the energy transfer from the field to matter are mostly described in mean-value, nonstochastic terms. The particle fluence rate $\dot{\Phi}$, for instance, is thus the *mean number* of particles incident on a sphere of unit cross-sectional area per unit time. In textbooks on physics $\dot{\Phi}$ is sometimes denoted as the "particle flux", but the recommendation by the ICRU [3] is to reserve this term for the number of particles per unit time. The particle fluence Φ (m^{-2}) is numerically identical to the total particle path length traveled per unit volume, a relation useful in dosimetric calculations. Staying with dosimetry issues, it is the kinetic energy available in a radiation field that is of prime interest, not the number of particles carrying the energy. Consequently, such quantities as the energy fluence Ψ ($\mathrm{J\ m^{-2}}$) and the energy fluence rate $\dot{\Psi}$ ($\mathrm{J\ m^{-2}\ s^{-1}}$) are defined.

When one is performing detailed calculations of how radiation energy is transferred to matter, the basic field quantity is the particle fluence distribution with respect to direction and kinetic energy, $\dot{\Phi}_{\Omega,T}$ ($\mathrm{m^{-2}\,s^{-1}\,J^{-1}}$ steradian^{-1}). The physical meaning of $\dot{\Phi}_{\Omega,T}\, d\Omega\, dT(\mathbf{r})$ is the number of particles per second and unit area at a point $\mathbf{r}$ in the room that fulfill the criteria of having kinetic energies between T and $T + dT$ and of being confined to the solid angle $d\Omega$ in the direction defined by the unit vector $\mathbf{\Omega}$. The quantity $\dot{\Phi}_{\Omega,T}$ is also known as the angular flux in the field of radiation transport theory.

17.3.2 Interaction Quantities

The probability that an ionizing particle or photon will interact with atoms (or electrons) in its path is described by the (microscopic)*cross section*, σ (m^2). If a particle can undergo different and independent kinds of interactions, the total cross section σ equals the sum $\sum \sigma_i$, where σ_i is the cross section for the interaction of type i. The photon cross section, for instance, can be split into five components, $\sigma = \tau + \sigma_c + \sigma_{coh} + \kappa + \nu$, where τ denotes the photoelectric effect, σ_c Compton scattering, σ_{coh} coherent scattering, κ pair production and ν interactions with the nucleus.

The *attenuation coefficient* μ (m^{-1}) is the *macroscopic* cross section, i.e. the number of target entities (atoms or electrons) per unit volume, n_t, times the (microscopic) cross section σ. The *mean free path* of an uncharged particle equals μ^{-1} (m), i.e. the reciprocal of the attenuation coefficient μ.

In radiation dosimetry, the focus is not on the interaction as such, but instead on how much of the particle energy is transferred to the material by the

interaction. For photon interactions, for instance, the attenuation coefficient μ is weighted by the mean fraction $\overline{f}$ that is transferred to charged particles, taking into consideration the partial probabilities of the photoelectric effect, Compton scattering etc. The resulting coefficient, $\mu_{tr} = \overline{f}\mu$, is called the *energy transfer coefficient*. A certain fraction g of the kinetic energy transferred to atomic electrons in photon interactions will be converted to photon energy (bremsstrahlung etc.) during the retardation of these electrons. The *energy absorption coefficient* $\mu_{en} = \mu_{tr}(1 - g)$ takes this loss into consideration.

The *stopping power* S (J m^{-1}) of a material, for charged particles, equals dT/dl, where dT is the energy lost by the particle in traversing a distance dl in the material. The stopping power S is the sum of at least three independent components, $dT/dl = (dT/dl)_{el} + (dT/dl)_{rad} + (dT/dl)_{nuc}$, where the index *el* denotes energy losses due to collisions with electrons, the index *rad* denotes radiative energy losses in bremsstrahlung processes, and the index *nuc* denotes energy losses in which the transferred recoil energy is imparted to atoms. The first two components are usually referred to as the *collision stopping power* and the *radiative stopping power*, respectively.

17.3.3 Dose Quantities

The basic quantity for estimating radiation risk is the *absorbed dose* D (J kg^{-1}), the specific energy imparted. The SI unit J kg^{-1} in this case has been given the special name "gray" (Gy). D is a *mean-value* quantity and does not take into account the stochastic character of the absorption process. In the high-dose range, say for D larger than 100 mGy, there are many energy deposition events per human cell, and the absorbed dose D becomes a good descriptor of the energy imparted and can be expected to correlate well with the severity of acute radiation effects.

The higher biological effect of densely ionizing radiation per unit dose has led to the attachment of a weight factor, the *radiation quality factor* ω_R, to D. The weighted quantity $H = \omega_R D$ is called the *equivalent dose* and the SI unit in this case, J kg^{-1}, has been given the special name "sievert" (Sv). The value of ω_R picked for different types of radiation is a compromise, an adaptation of the variable RBE (relative biological to effectiveness) values obtained in irradiation experiments. For sparsely ionizing radiation, such as γ-rays, X-rays and electrons (β), the radiation quality factor ω_R equals unity.

The equivalent dose H_T in tissue T is the quantity used for dose limits for individual organs. As H_T is a weighted quantity, it is normally not experimentally measurable. This remark is also valid for the *effective dose* $E = \sum \omega_T H_T$ (Sv), where the *tissue weighting factor* ω_T weights the body organs with respect to carcinogenic and hereditary effects. The sum of all ω_T factors is normalized to unity. Dose limits in working life are expressed in units of E, as this is considered to be the best quantity available for estimating the probabilities of cancer and hereditary effects. As the numerical

value of E depends on the carcinogenicity and hereditary damage of the radiation, it should not be used outside this area, for instance in forecasting acute radiation effects.

In operational radiation protection work, the three measurable quantities *ambient dose equivalent* H^*, *directional dose equivalent* H' and *personal dose equivalent* $H_p(d)$ have been introduced as substitutes for the effective dose and equivalent dose. The depth d is mainly limited to values of 10 mm, in popular terms the "deep dose", and 0.07 mm, the skin dose or "shallow dose". If the numerical value of H^*, H' or H_p, obtained with a properly calibrated instrument, is below the relevant dose limit, it is considered that conformity with the legal demands of the radiation protection system has been demonstrated. The definitions of H^*, H' and H_p are intricate and are aimed at radiation standardization laboratories. In operational protection work, knowledge of the precise definition of these quantities is not necessary.

The *kerma* K (Gy) is defined by the relation $K = dT_{tr}/dm$ $(= \Psi\mu_{tr}/\rho)$, where dT_{tr} is the sum of the kinetic energies of all the charged particles liberated by uncharged particles in a mass dm of the material. If the energy inflow and outflow of secondary charged particles cancel out (charged-particle equilibrium, CPE) in dm, the kerma K numerically equals the absorbed dose D.

The exposure X is an old quantity emanating from the use of open-air ionization chambers as primary-standard instruments for calibration purposes. The old special unit röntgen (R) for exposure corresponds to $2.58 \times 10^{-4}\,\mathrm{C\,kg^{-1}}$ (or approximately 0.0087 Gy expressed as kerma in air). Dosimetry standard laboratories are phasing out both the kerma and the exposure quantities in favor of the operational quantities H^*, H' and H_p.

17.4 Radiation and Living Material

Ionizing radiation interacts on the atomic level as outlined in Sect. 17.2. Any charged secondary particle created by the primary beam ionizes the material along its track while slowing down. The mean energy absorbed to create one ionization (one ion pair) is 34 eV for electrons, and about the same value also for heavier charged particles, when stopped in living tissue. If the ionization takes place within a biomolecule such as DNA (deoxyribonucleic acid), the lesion is denoted as "direct". If an ion pair created in any other type of molecule causes damage to a biomolecule, this is an indirect effect caused by attack from chemical radicals. Water plays an important role here, as the water content of the human body is so high.

17.4.1 Early Radiation Effects in Humans

Whole-Body Response

High irradiation doses, of order of 1 Gy and higher within less than a few hours, cause cell killing to such an extent that the function of organs will be significantly impaired or destroyed all together. Below a threshold dose of about 0.25 Gy, the cell-killing effect, even in sensitive organs is small enough to be compensated for and is not clinically detectable unless very sophisticated types of chromosomal or physiological analysis are applied.

The response to a whole-body dose in excess of 1 Gy is almost immediate, within hours, owing to damage to sensitive cells in the gastrointestinal tract. The symptom of nausea that follows, and at higher doses vomiting and diarrhea, increases in severity with dose.

As the variation in individual sensitivity to early radiation effects is small, all individuals irradiated above a certain threshold will show symptoms, and early effects are synonymously denoted as "deterministic". Another, perhaps more common, designation is "acute", emphasizing the short-termness (days, weeks or months) and distinguishing it from late effects (years), such as cancer and hereditary disorders.

The approximate dose thresholds for different acute radiation syndromes following brief and protracted exposures are listed in Table 17.2. Modern medicine offers treatments that relieve the symptoms following accidental overexposure, but these are rarely curative when the dose exceeds 6 Gy. On the other hand, single whole-body irradiations below 2 Gy are considered nonlethal.

The Skin Response

Soon after Röntgen's discovery of X-rays, skin redness (erythema) and, after massive and extended exposures, loss of skin and ulceration were observed from this new type of radiation. The soft (i.e. low-energy) X-rays of the early 1900s made the skin, or to be more precise, the basal cells just below the skin surface, a critical organ for acute radiation damage. A dose of about 6 Gy is the threshold for a so-called main erythema reaction about 1–2 weeks after irradiation. Before that, within hours, the skin reacts with a mild and transient redness if the absorbed dose exceeds about 2 to 3 Gy.

Another indication of a high dose to the skin is loss of hair (epilation). The threshold dose for temporary epilation is about 3 Gy, while a dose in excess of about 7 Gy is necessary to make the hair loss permanent.

Effects During Pregnancy

Animal studies and the epidemiological results from the bombing of Hiroshima and Nagasaki (H–N) indicate that a growing embryo is prone

Table 17.2. Estimates of the thresholds for deterministic effects in the adult human testes, ovaries, lens and bone marrow ([5], p. 103)

Tissue and Effect	Threshold		
	Equivalent Dose, Single Brief Exposure (Sv)	Equivalent Dose, Highly Fractionated or Protracted Exposure (Sv)	Annual Equivalent Dose Rate for Many Years (Sv y^{-1})
Testes			
Ṫemporary sterility	0.15	NA*	0.4
Permanent sterility	3.5–6.0	NA	2.0
Ovaries			
Sterility	2.5–6.0	6.0	>0.2
Lens			
Detectable opacities	0.5–2.0	5	>0.1
Visual impairment (cataract)	5.0	>8	>0.15
Bone marrow			
Depression of hematopoeisis	0.5	NA	>0.4

* NA denotes "not applicable", since the threshold is dependent on dose rate rather than on total dose

to radiation damage. When the embryo is irradiated during 8–15 weeks after conception in humans, the probability of severe mental retardation is believed to be 40% per Sv and 10% per Sv during weeks 16–25 ([6], p. 231). A downward shift in IQ score is interrelated with this risk, about 30 IQ units per sievert during the most sensitive period of 8–15 weeks [5]. These risk figures are mainly based on the H–N statistics, which also indicate that there is a threshold of about 0.2 Sv for these interferences with the developing human brain.

Taking into account the risks of both induction of malformations and childhood cancer (0–19 y) after in utero exposure, the ICRP concludes that only for fetal doses in excess of 100 mGy may there be medical reasons for terminating a pregnancy [7].

17.4.2 Late Effects in Humans

Radiation-Induced Cancer

That ionizing radiation is a carcinogen has been proven beyond doubt for brief or extended exposures to high doses. In contrast to the acute effects discussed above, radiation-induced cancers are characterized by the following:

1. They are stochastic.
2. There is a long delay (years or decades), the latency period, between the exposure event and the outbreak of the disease.
3. In an irradiated population, the radiation-attributable cancer risk to still-unaffected individuals is in force for a long period after the minimum latency period.
4. The damage to an affected person will not increase with dose.

The first point means randomness in the sense that it is not possible to predict the individual persons in an irradiated population that will be affected, just an expected value of how many. The minimum latency periods for solid tumors and leukemia are assumed to be about 10 and 2 years, respectively [8].

The proof of a connection between ionizing radiation and cancer stems from many different areas, for example medical irradiation procedures of the past and uranium miners. The dominating source of statistics, however, is from the follow-up of the Japanese bomb victims [9].

Cancer Risks After High-Dose Exposures

Natural and radiation-induced cancers cannot be distinguished from each other. In order to isolate the influence of radiation, the number of radiation-induced cancers must outnumber the expected variation in natural cancer incidence rates. Only extensive irradiations to many people can achieve this as, fortunately, the gathered experience from accidents and old medical procedures reveals that ionizing radiation is a fairly weak carcinogen. The excess lifetime morbidity risk is of the order of a few percent following a single dose of 100 mSv.

Compared with adults, small children and the human embryo are more prone to develop radiation-attributed cancers later in life. The rarity of natural childhood cancers also favors the prospect of identifying any such radiation effects. Cancer induction after in utero exposure to diagnostic X-rays has been studied extensively in the past and indicates an increase in childhood cancer risk by about 40% for doses of about 10–20 mGy ([10], Appendix G, §245). Todays investigation of the fetus with ultrasound has essentially removed this childhood cancer risk from the scene.

The most extensive radiation detriment data on an adult population stem from the follow-up of the Hiroshima and Nagasaki populations. The fatal-cancer incidence in a cohort of 50 000 persons who were significantly exposed, i.e. received a whole-body dose >5 mGy, has been compared with a control group of 36 500 persons exposed to <5 mGy. As seen in Tables 17.3 and 17.4, the lifetime risk attributable to the bombs exhibits a strong dependence on dose and age at exposure. As an average over gender and age, the lifetime cancer mortality probability in the H–N statistics is 10–12% and about 1% per Sv for solid cancers and leukemia, respectively.

Table 17.3. The lifetime excess probability due to radiation in the significantly exposed group in Hiroshima and Nagasaki with an average effective dose of 0.2 Sv. The natural lifetime risk for an unexposed person is shown for comparison [9]

	Probability			
Age at Exposure	Excess Women	Natural Women	Excess Men	Natural Men
10 y	0.05	0.19	0.03	0.26
30 y	0.03	0.20	0.02	0.28
50 y	0.01	0.15	0.01	0.18

Table 17.4. The percentage of fatal cancers attributable to the bombing of Hiroshima and Nagasaki as a function of dose [9]

Effective Dose Range	Attributable Fraction	
	Leukemia	Solid Cancer
5–200 mSv	14%	2%
200–500 mSv	45%	12%
500–1000 mSv	74%	23%
>1000 mSv	84%	39%

Cancer Risks After Low-Dose Exposures

Considering that a single or at least very few ionizations may alter cell genetics, radiation protection expert organizations such as the ICRP consider it pertinent to base radiation hygiene recommendations on a linear–no-threshold (LNT) hypothesis of cancer induction. This hypothesis simply states that the probability of dying from a radiation-caused tumor (including leukemia) is directly proportional to the effective dose E and that no threshold exists below which the probability is zero. Applying the epidemiological data given above to working-life conditions, the slope of the LNT curve is reduced by a factor of 2. This reduction is motivated by the fact that working-life exposures are typically protracted and of low dose rate, leaving room for cellular repair. This reasoning has led the ICRP to recommend so-called nominal risk coefficients for fatal cancer of 4.0 and 5.0% Sv^{-1} applicable to a working population and a population including also children, respectively, and to low-dose and low-dose-rate exposures.

The LNT hypothesis, which has given birth to concepts such as ALARA (as low as readily achievable) and collective dose (man-sievert), is a cornerstone of modern radiation protection recommendations. In recent years, however, the uncritical use of LNT has been questioned. The criticism is based on mainly two things:

1. Predicting low-dose (<50 mSv) consequences, for example the number of cancer mortalities, using the LNT hypothesis gives a false impression of knowledge that is not there.
2. The LNT hypothesis leads to a regulation of risks trivial to the individual, which causes mistrust in the regulatory authority.

Radiation epidemiology is not sensitive enough to resolve the low-dose issue, and one has to await future research in fields such as cellular genetics and signaling in order to quantify cancer probabilities after mSv (or lower) irradiations. Meanwhile, it is wise to extract, at most, a precautionary principle from the LNT hypothesis. The cancer risk at a dose below say 50 mSv presumedly resides in the interval from zero up to at most 10 times the predictions of LNT. With such an uncertainty, it is sound to be very restrictive with risk quantifications. The complexity of cellular response to external agents such as ionizing radiation, revealed by modern cell physiology, strongly support such a cautionary posture.

Hereditary Effects

The detailed mapping of the human genome in recent years and the improved understanding of human hereditary processes indicate that the probability of hereditary effects following exposure to ionizing radiation is negligible compared with the natural risks in the dose range below 100 mGy (see Table 17.5).

Hereditary diseases are broadly classified as either single-gene (Mendelian), chromosomal and multifactorial types. Multifactorial diseases result from a

Table 17.5. Estimates of hereditary risks to offspring from exposure to a single parent (1st) generation and two (1st and 2nd) parent generations. The assumed doubling dose is 1 Gy. From [11]

Disease Class	f_s	r_D (Gy^{-1} per 10^6 progeny in generation)		
	Per 10^6 live births	one 1st generation exposed	two	two both gen. exp.
Mendelian				
Dominant + X	16 500	750–1500	500–1000	1300–2500
Recessive	7 500	0	0	0
Chromosomal	4 000	[a]	[a]	[a]
Multifactorial				
Chronic	650 000[b]	250–1 200	250–1 200	250–1 200
Congenital[a]	60 000	2 000	400–1 000	2 400–3 000

[a] The chromosomal damage is included partly in the class of dominant and X-linked diseases, and partly in the class of congenital abnormalities.
[b] Frequency in population.

large number of interacting genetic and environmental factors, and the natural incidence is high. The multifactorial class includes both abnormalities seen at birth (congenital abnormalities), and adulthood diseases such as high blood pressure and diabetes.

The so-called doubling dose is a basic concept for hereditary-risk estimations. The doubling dose D_{dbl} is the absorbed dose that provokes as many mutations as those which occur spontaneously in one generation.

For dominant single-gene disorders, a mutation will ultimately, if not lethal, give a hereditary ailment in the living offspring. For the recessive type of disorders, and for polygenic and multifactorial diseases, the probability of manifested hereditary damage in the first-generation offspring may be close to zero. In order to take this into consideration, the probability of mutation is multiplied by a disease incidence factor Q. The hereditary risk per unit dose, r_D, can then be written in the form $r_D = f_s D_{dbl}^{-1} Q$ (Gy^{-1}), where f_s is the baseline spontaneous-incidence rate. For the incidence factor, we have $0 \leq Q < 1$, and on the basis of mouse data, it is believed that the doubling dose for most human gene locations is at least 1 Gy.

In Table 17.5, an estimation of the hereditary risks is summarized. It must be remembered that hereditary damage may affect quality of life in a way that ranges from the almost unnoticeable to the most severely disabling. This implies that it is not very meaningful to add the incidence rates of Table 17.5 together without weighting for the severity of the hereditary disorder in question. It is also obvious from the figures in Table 17.5 that the assumed doubling dose of 1 Gy far from doubles the incidence rates in the first few generations of offspring, i.e the Q factor is of the order of one percent or lower. This statement is also valid in the case where the irradiation is duplicated in the first few generations.

A permanent and significant increase in gonad dose to a population will, however, in the long run, change the baseline frequency f_s, and in this new hereditary equilibrium, the increased mutation rate will result in an identical increase in disease frequency.

17.5 Detecting Ionizing Radiation

All detectors for ionizing radiation are based on the same fundamental principle – the transfer of energy to the detector. In the detector, the energy is converted into some other form that can be registered. Modern detectors are essentially electrical – at some point, the information from the detector is transformed into electrical pulses that can be treated by electronic means.

It should be recognized that all instruments are limited to measuring certain types of radiation within a fixed range of energy. Outside these limits, instrument readings are not to be trusted.

17.5.1 X-Ray, β-Ray and γ-Ray Detection

Ionization detectors were the first electrical devices developed for radiation detection. These instruments are based on the direct collection of the ionization electrons and ions produced in a gas by passing radiation. Three basic types of detectors have been developed – the ionization chamber, the proportional counter and the Geiger–Müller counter. These types of detectors are today mostly used as radiation monitors as they are cheap, simple to operate and easy to maintain.

The scintillation detector makes use of the fact that certain materials when irradiated emit small flashes of light, i.e. they scintillate. When coupled to an amplifying device such as a photomultiplier, this scintillation light is transmitted through a shaped light pipe to the photocathode of a photomultiplier. There, photons release electrons, which are accelerated and focused onto the first dynode. For each primary electron hitting a dynode, between two and five secondary electrons are released. Up to 15 multiplying stages can be used, reaching overall multiplying factors of up to 10^9. A few incident photons therefore produce a measurable electrical pulse, which can then be analyzed and counted electronically.

Semiconductor detectors are based on crystalline semiconductor materials, most notably silicon and germanium. The basic operating principle of semiconductor detectors is analogous to that of gas ionization devices. However, the medium is now a solid material. The passage of radiation through the solid material creates electron–hole pairs (instead of electron–ion pairs). The advantage of a semiconductor is that the average energy required per electron–hole pair is some 10 times smaller than that required for gas ionization. Thus, the amount of ionization produced for a given particle energy is an order of magnitude greater, resulting in superior energy resolution. Moreover, semiconductor detectors have a greater density and therefore a greater stopping power than gas detectors. As a result, they are more compact in size and can have a very fast response time.

17.5.2 Tritium Detection

It is important that proper ventilation techniques are used to ensure that any release of tritium is adequately exhausted from the laboratory. It is also necessary to provide air monitoring for releases of tritium gas. A common device for tritium monitoring is an ionization detector, through which air is drawn at a fixed rate. The ionization chamber may be preceded by an ion collector and/or a filter in order to remove other ions or radioactive material. Tritium contamination can be detected and analyzed through the use of samples collected with filter paper or other type of traps and then transferred to a liquid scintillation solution. Liquid scintillation counters are also the instrument of choice for monitoring tritium in urine samples.

17.5.3 Neutron Detection

Neutron radiation may be accompanied by relatively high levels of γ-radiation. Consequently, in order to measure the neutron level adequately, it is necessary to have an instrument insensitive to γ-radiation. Ionization instruments, in general, are usually not satisfactory for measuring neutrons, since they are also sensitive to γ-radiation.

The boron trifluoride (BF_3) proportional counter provides a sensitive detector for a neutron survey instrument and can be relatively insensitive to γ-radiation. This instrument uses BF_3 gas enriched to more than 95% in the isotope ^{10}B, and typically the detector has about a 100 mm active length. The BF_3 counter is sensitive to thermal neutrons. It is also possible to detect intermediate and fast neutrons by enclosing the detector in polyethylene or paraffin wax, which brings down the neutron energy before the neutrons enter the BF_3 gas. With a suitable moderator configuration, it is possible to achieve a count rate which approximates the dose-equivalent rate $\dot{H}$ for neutrons in the energy range below 10 MeV.

The ^{3}He neutron detector is more sensitive than the BF_3 counter. The detector is based on the reaction ^{3}He(n, p)T. The proton and the tritium will ionize the gas and an electrical pulse will be obtained for every absorbed neutron. A disadvantage is that the ^{3}He gas is very expensive, and to have a sensitive detector, the detector needs to be large. The sensitivity is strongly dependent on the ^{3}He gas pressure, volume and enrichment and can be up to several hundred pulses per incoming neutron per cm^2.

Cerium-doped lithium silicate glasses are widely used as neutron detectors. Recent developments incorporate this scintillator into plastic fibers acting as light waveguides [12]. Both flat and bent large-area neutron detectors can be built by this technique. The light output is monitored at the end of the glass fibers.

17.6 Radiation Dose Measurements

Generally, suitable dosimeters tend to mimic human tissue in atomic composition in order to avoid the problems of converting the measured signal to another material that is too different. This is of special importance when dealing with the sometimes very strongly atomically dependent cross sections of neutrons, and in gamma radiation field measurements for less well-known spectral distributions. Now, this choice of a suitable atomic composition may not be a problem for the user at an accelerator laboratory, since the decision on composition has already been taken by the instrument manufacturer. The important thing is that the manufacturer can describe the application areas for the instrument and that the instrument is calibrated in quantities acceptable to the radiation protection authority surveying safety issues. The user's

responsibility is to use a dosimeter suitable for the radiation to be monitored, to check the functionality and to calibrate the dosimeter on a regular basis.

In accelerators emitting pulsed radiation, the capability of a dosimeter to cope with the high dose rates in the pulse peak needs to be considered. As mentioned in Sect. 17.6.1, some solid-state dosimeters can reproduce very high dose rates accurately. Many type of gas-filled detectors may be duty-cycle-dependent in the high-dose-rate domain. Further guidance on this subject is given in the ICRU Report 20 [13].

An issue that accelerator dosimetry measurement has in common with all high-energy photon applications involving small radiation field sizes is the dose buildup within surfaces facing air or gaseous media in the direction of the incoming photons. When photons irradiate a solid material in air, the maximum dose in the solid is reached at a depth corresponding to the range of the secondary electrons generated in the solid by the photons. Thin-walled ionization chambers, open dosimeters etc. must be covered with an appropriate amount of material to register this maximum properly. A set of buildup covers of different thicknesses may be necessary in order to handle the diverse photon geometries and energies encountered.

In order to perform dose measurements that can be used for legal purposes, the dosimeter used must be calibrated with respect to or traceable to established dose quantities, such as the environmental dose equivalent $H^*(10)$. With modern dosimeters, this is usually no problem.

17.6.1 Personal Monitoring

Individual dosimetry is normally mandatory in all controlled areas of an accelerator. The exposure of people can be measured by using pocket pen dosimeters, film badges, pocket ionization chambers, thermoluminescent dosimeters and diode detectors. There is, however, a trend today allowing only legally authorized dosimeter types from likewise authorized dosimetry laboratories, when the dosimeter is intended for personal record-keeping.

Film badges can be made sensitive to γ- and X-rays, electrons, and neutrons. The use of filters makes possible the separation of different kinds of radiation and different energies. The radiation produces blackening of the emulsion, which can be determined photometrically after development. Integrating dosimeters based on the thermoluminescence (TL) principle are gradually replacing the old film badge as the dominating legal dosimeter. A thermoluminescent material stores the ionization energy in the crystalline lattice in such a way that after exposure it can be released in the form of light when the material is heated. The amount of light is directly proportional to the absorbed dose accumulated by the dosimeter material. Following special routines for how the material is temperature-treated and stored when not in use, a TL dosimeter can be reused hundreds of times. Compared with the film badge, the TL dosimeter is superior in sensitivity, dose and dose-rate

linearity, and energy dependence for photons. The most common TL phosphor, LiF, is linear in dose rate up to 10^9 Gy s^{-1} [14], a useful capacity in pulsed radiation fields. A TL badge can hold several small TL tablets, for instance one with high sensitivity, e.g. ^{6}LiF, to neutrons and one with low sensitivity, e.g. ^{7}LiF. Film and TL badges should be read out on a regular basis. A common integration time is 1 month.

The pocket pen dosimeter is based on the old electroscope-discharging principle and is directly readable by the wearer. Modern direct-reading electronic personal dosimeters (EPDs), mostly based on multidiode designs, have made the pocket pen dosimeter obsolete. Today EPDs come in lightweight designs (<100 g) offering separate shallow- and deep-dose readings, dose and dose rate histories, and visual and audible alarms for dose and dose rate, and with computer software for easy changing of dosimeter parameters, visualization of dose history etc. EPDs have high dose sensitivity and are usually calibrated in both $H_p(10)$ and $H_p(0.07)$ (see Sect. 17.3.3).

17.7 Radiation Hazards

The primary beam creates secondary radiation by collision with atoms in its path, and induced activity by nuclear reactions. The secondary radiation is prompt and can be a problem only while the beam is on, while the induced activity constitutes a source that is also alive after the beam has been stopped. The small aperture and high intensity of the primary beam makes it a dangerous source of radiation that can cause local radiation injuries to the human body. Fortunately, in most accelerators the primary beam is enclosed in such way that it is not accessible. As a rule, bremsstrahlung and neutrons, owing to their high penetrating abilities, dominate the radiation hazard and determine the degree of protection necessary in an accelerator environment.

17.7.1 Electron Accelerators

Electrons

The primary beam of electrons may be accessible if the beam is brought out into the room air. The acceleration of electrons by an electrostatic accelerator means a limit on the kinetic energy T of the electrons of about 5 MeV. The maximum range of electrons is approximately proportional to the inverse density of the stopping material, with the consequence that electron ranges are usually expressed in surface weight units, e.g. g cm^{-2}. In the energy interval 2–5 MeV, the electron range in any material in cm can in practice be set to $0.6 \times \rho^{-1}T$, if T is in MeV and the density ρ is in g cm^{-3}. This means that 5 MeV electrons are stopped by a few cm of unit-density material, while the range in air is about one thousand times longer. It is a good practice to stop the electrons in a low-atomic-number material as this will minimize bremsstrahlung formation.

Gamma Radiation

Any charged ionizing particle may lose energy in a bremsstrahlung (X-ray) process. The energy distribution of the generated bremsstrahlung is continuous, with a maximum energy equal to the initial electron energy. The cross section for the process in a material is approximately proportional to Z^2/m^2, where Z and m are the atomic number of the material and the mass of the charged particle, respectively. Thus, bremsstrahlung is a potential safety problem only for electrons.

The angular dependence of the X-rays generated in thick targets is to a large extent governed by the angular distribution of the electrons that are slowing down. Increasing the electron energy from, say, 100 keV and upwards causes the X-ray beam to tilt more and more in the electron direction. The photon energy fluence at large angles, say larger than 60°, is dominated by low-energy photons created by energy-degraded electrons in the target. For increasing primary electron energies, this behavior will be more pronounced.

Low-energy X-rays are generated wherever low-energy electrons are deaccelerated, for example in cavities and klystrons. For electrons in the energy interval 0.5–5 MeV, the forward-scattered photons create a dose-equivalent rate of roughly $14\,T^3$ Sv h^{-1} at 1 m per mA if T is expressed in MeV [15].

Every point in an accelerator in which high-energy electrons strike matter is a bremsstrahlung source. Wherever the electron beam is deflected, or passes through slits, windows, collimators and so forth, enhanced emission of bremsstrahlung photons should be suspected. Poor vacuum is another source of bremsstrahlung, owing to collisions of electrons with residual-gas molecules. A partial loss of vacuum can be almost instantaneous, and the subsequent rapid and pronounced increase in dose rate along the beamline makes this kind of failure extra insidious.

Neutron Radiation

Photonuclear processes, such as (γ, n), show a broad resonance cross section (the giant resonance) centered around 15 to 25 MeV in most materials. The threshold is not lower than 6 MeV, with exceptions for deuterium (2.2 MeV) and ^{9}Be (1.67 MeV). Thus, neutrons are not produced by electrostatic electron accelerators unless deuterium or beryllium targets are deliberately exposed to bremsstrahlung.

17.7.2 Proton and Light-Ion Accelerators

Compared with electrostatic electron accelerators, the radiation environments around accelerators used for light ions up to and including helium ions are more complex. The much lower intensity of bremsstrahlung and the short range of the primary radiation facilitate the radiation protection work.

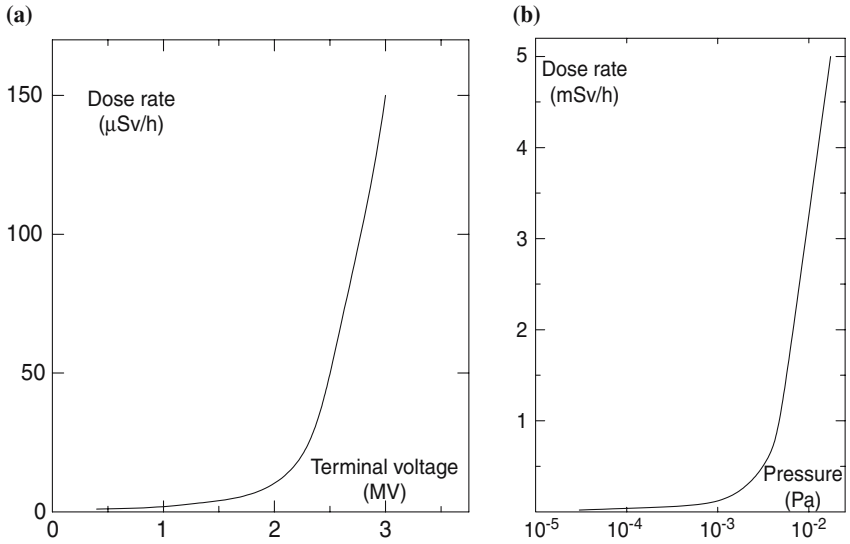

Fig. 17.2. Bremsstrahlung dose rate 1 m from the accelerator tank just opposite the terminal for ∼3 μA protons injected into a fairly new 3 MV tandem Pelletron. (**a**) As a function of the terminal voltage up to the maximum design voltage. The pressure at the low-energy side just outside the tank was 1×10^{-5} Pa and the pressure in the terminal was 5×10^{-4} Pa; (**b**) As a function of the mean pressure in the low-energy accelerator tube. The curve was obtained at a terminal voltage of 2.0 MV

Bremsstrahlung cannot be avoided, however, as electrons are unintentionally accelerated in all evacuated high-voltage parts of the accelerator.

Examples of the bremsstrahlung around an electrostatic accelerator are shown in Fig. 17.2. The curves were obtained with a 3 MV tandem Pelletron using a beam of protons. Similar curves have been obtained with other ion beams. The detector was an organic scintillator designed for monitoring low background levels. Figure 17.2a demonstrates the strong increase of the dose rate when the terminal voltage approaches the maximum design voltage. The strong dependence of the dose rate on the pressure in the accelerator tube is demonstrated in Fig. 17.2b. A similar strong increase of the dose rate can also be obtained by changing the optics of the beam entering the tube so that they become worse. Both figures were obtained for a constant beam current. If the beam current is changed (keeping the terminal voltage and tube pressure constant), a linear relation between dose rate and beam current is obtained (at least for moderate currents, 1–10 μA).

Many light-ion and proton reactions create neutrons, and it is usually these neutrons that determine the radiation shield necessary. The most common shielding material is concrete. The high content of hydrogen in the concrete efficiently slows down the neutrons to thermal energies and the neutrons

are then, with a high probability, lost by a capture process in hydrogen, i.e. $^{1}\mathrm{H}(\mathrm{n}_{th},\gamma)^{2}\mathrm{H}$. In the capture a 2.2 MeV photon is created, which, in turn, because of its high penetration, may cause a protection problem. The mean free path in ordinary concrete is about 0.1 m for 2.2 MeV photons. To avoid the production of 2.2 MeV gamma radiation, boron can be mixed into the concrete. The capture cross section of boron due to the reaction $^{10}\mathrm{B}(\mathrm{n}_{nt},\alpha)^{7}\mathrm{Li}$ is about 10 000 times bigger than the corresponding cross section of hydrogen capture ditto, and the emitted gamma ray has an energy of only 0.48 MeV.

Protons

The neutron yield from protons impinging on thick targets increases rapidly with energy, from the order of per mill per proton at 10 MeV to tens of percent at 100 MeV. Generally, the neutron yield for most materials is similar to that for copper and iron, or lower; it may be even a factor of ten lower for proton energies well above the threshold for neutron production in the material [15, 16]. (Materials with low thresholds are discussed in the next section.)

17.7.3 Low-Voltage Neutron Generators

High neutron dose rates can also be created in low-voltage accelerators with proper projectile and target materials. Low-voltage neutron generators function by accelerating positive ions across a potential drop of a few hundred kV or less. Because of the high yield of energetic neutrons, the reaction of deuterium with tritium as the target is the method of choice. The reaction $^{3}\mathrm{H}(^{2}\mathrm{H},\mathrm{n})^{3}\mathrm{He}$ (also denoted as T(d, n) or D–T) is exoergic with a large energy release, 17.6 MeV. For thin targets, the maximum yield occurs at a deuteron energy of 107 keV. For thick tritium targets, the yield increases rapidly up to 600 keV and then more slowly [17]. The energy of the neutrons created is somewhere in the interval 13–16 MeV for accelerating potentials less than 400 kV, with the maximum and minimum energies in the forward and backward directions, respectively. By convention, these neutrons are referred to as 14 MeV neutrons, from the kinetic energy in the center-of-mass coordinate system.

The typical yield of 14 MeV neutrons per mA for 150 keV deuterons gives a fluence rate of about $8 \times 10^{5}\ \mathrm{cm}^{-2}\ \mathrm{s}^{-1}$ at one meter from the target. This fluence rate is equivalent to an effective dose rate of 1.4 in the AP and 1.1 Sv h^{-1} in the ROT geometry (see Sect. 17.B.5).

Another exoergic reaction used for low-voltage neutron accelerators is the D–D (deuteron-deuteron) reaction. The neutron yield at 150 kV is less than one percent of that for the D–T reaction and the neutron energy is much lower, in the range 2–3 MeV [17]. The effective neutron AP dose per mA at 1 m from the deuteron target is approximately 0.6 mSv h^{-1}. D–D accelerators

do not generally present any important radiation hazards compared with D–T generators. It should be noticed, however, that if deuterium is accumulated somewhere in the beamline before the actual target, extra neutron sources outside the expected one are created.

17.7.4 Heavy-Ion Accelerators

Typically, accelerating lithium and heavier ions generates only moderate radiation problems if the projectile ions are kept within the vacuum part of the accelerator. Hazards to pay attention to are neutron-producing reactions and the source of characteristic X-rays created in the material in which the ions are stopped, and, as in all "evacuated" electrostatic environments, bremsstrahlung from inadvertently accelerated electrons.

17.7.5 Induced Radiation

In contrast to the particle beam itself, the induced activity in accelerator materials and components constitutes a delayed source of radiation that exists also after the accelerating voltage has been switched off. The thresholds for nuclear reactions in ordinary metals and materials are such that induction of radioactivity is an almost nonexistent problem for electrostatic electron accelerators. Proton and deuteron beams and the accompanying neutrons may produce measurable quantities of activity, but the specific activity concentrations in accessible areas are usually too low to be hazardous. In addition, the majority of radionuclides are short-lived, and by delaying access to target areas after shutdown and delaying maintenance programs by one or a few days, the exposure of personnel can be minimized. Evidently, the highest activity concentrations should be expected in targets, beam dumps, slits and collimators.

The unavoidable induction of activity in components and materials is complex, and the reader is advised to consult the rich literature (e.g. [15, 18, 19]) on the subject to solve specific problems.

For light-ion electrostatic accelerators, thermal-neutron reactions induce activity in air, copper, steel, concrete etc. The resulting specific activity is usually less than 1 $\mathrm{Bq\,g^{-1}}$ and with negligible radiation risks to the personnel. Long-lived products, such as ^{60}Co, however, may cause administrative problems at decommissioning, owing to the very strict regulations for disposal of weakly radioactive materials.

17.7.6 Tritium Targets

Tritium, or ^{3}H, is the only radioactive isotope of hydrogen. It is a pure beta emitter with a half-life of about 12 years. The maximum energy of the beta particles is only 18.6 keV, corresponding to a range of less than about

$0.6\,\mathrm{mg\,cm^{-2}}$. Thus, tritium is only a potential hazard when taken into the human body, for example by inhalation. Even so, the dose factor is low, owing to a short biological half-life, roughly 2 weeks for most compounds, and the low ionizing energy emitted per decay. The low dose factor, about 30 mSv for an intake of 1 GBq tritium, must, however, be seen in the perspective of the extremely high tritium target activities that exist and the natural tendency of tritium to become airborne, particularly at elevated temperatures.

The typical target construction is tritium adsorbed in a thin metal layer. Tritium desorbs spontaneously from the metal at room temperature and is, furthermore, driven out by the bombardment with projectile particles. Under normal conditions, the tritium leaving the target is exhausted via the vacuum pumps and the ventilation. The tendency of tritium to become airborne implies that tritium targets not in use should be stored in airtight containers. Such containers must be opened in well-ventilated places.

The bulk of the tritium desorbing from a target will pass straight through the vacuum pumps used and be ventilated out. Small amounts of tritium can be found in pump oils and other components. One exception is pumps of the ion-getter type. These retain tritium and may contain TBq of tritium if used for many targets [17].

17.8 Safety Considerations

17.8.1 Administrative Safety Program

The work at an accelerator facility must be organized in such a way that risks from radiation and other hazards are minimized. The details of an accelerator safety program depend on local conditions, but the basic safety strategies to follow have been set out by various national and international bodies (see for instance [20–23]). Evidently, any accelerator safety program must conform with the current national legislation. For many accelerators, the normal situation is a low radiation intensity. This is a risk, as staff easily become careless, considering the radiation hazard as negligible. This may lead to insufficient notice being taken of changed conditions (increased pressure in the vacuum system, new components of unsuitable material along the beam path, use of a target material with a low (p, n) threshold etc.). It is necessary to avoid careless habits in an accelerator laboratory. An accelerator must be respected for its maximum radiation capability rather than its customary conditions. In this section, we give some general comments on the administrative part of a radiation safety program, leaving more technical aspects to Sect. 17.8.2.

To be effective, a radiation safety program must have full support, both mental and budgetary, from the management, but also be proportionate and well adapted to all interests in the organization. A radiation safety program should, in a radiation safety manual, clearly define the duties and responsibilities of all employee categories, including the management, and state the

qualification requirements at different levels in the organization. The duties and the possible delegation in the safety organization should preferably refer to named persons, and the delegation scheme should be known and easily accessible to all employees. The safety program should cover topics such as categorization of workers and workplaces, education and training of personnel, routine controls and inspections, and rules for visitors. The safety procedures may be quite extensive, and it is hence important to state in the manual how safety work should be documented and reported. The local radiation expert in charge is a key person concerning radiation safety issues and it is good practice to let him/her report directly to the management of the facility in order to avoid on-the-floor conflicts between research and safety interests. At large-scale accelerator facilities, the expert normally reports to some kind of radiation safety steering committee.

An accelerator workplace can be very dynamic, with shifting beam rooms and intensities, mobile protective shields, etc. The safety manual must define who is authorized to make changes to beamline setups and how changes with significance for safety shall be reported in the organization. Pre-prepared plans for how to handle emergencies and accidents are also an important part of an operational safety program. Conventional accidents should not be forgotten, as they are normally much more probable than irradiation mishaps involving radiation doses to employees.

17.8.2 Technical Safety

The radiation safety at an accelerator facility depends on numerous factors. In addition to administrative procedures and the shielding the building itself provides, technical systems aimed at minimizing radiation and other hazards to the staff are of great significance. The ideal technical safety system is, if necessary, redundant; works passively in the background; is not sensitive to human errors; and does not interfere unduly with the primary accelerator assignments.

Shielding the staff from external radiation, and the use of distance, stationary (wall) shields and locks are preferred as compared with warning lights, audio alarms and other techniques more dependent on human behavior. The amount of shielding material must be dimensioned in accordance with the maximum practicable workload of the accelerator and the most "unfavorable" particle accelerated. It must be ensured that radiation attenuation at doors, windows, ventilation ducts and all other penetrations through a shielding wall,, is sufficient or is remediated by additional shielding blocks. The effect of sky-shine through the roof must be considered, as this can provide a significant contribution to the total radiation levels observed outside thick shielding walls.

Concrete is a natural choice for a shielding material. Both as a part of the building construction and as additional mobile shields, concrete is inexpensive, reliable, structurally useful and easily installed. Owing to these

practical aspects and its content of hydrogen, concrete is especially suitable for shielding against neutrons (see Sect. 17.7.2).

Access control and radiation warning systems are essential parts of a radiation safety program, particularly at facilities provided with several target rooms. A common way to prevent entry to high-radiation areas is by interlocks. If the interlock is broken or loses power, the beam should be stopped at a safe position and a restart must commence at the position of the interlock. There is a great deal of variation in the interlocking arrangements to be found in different laboratories. This depends on differences in accelerator-specific space or component limitations and on differences in design philosophy, cost factors and users' requirements.

In high-level radiation areas accessible to staff, the control circuits of the accelerator should produce a warning by light or an easily identified sound prior to start-up, to allow anyone trapped in the laboratory to leave the area. The duration of the warning must be long enough for a person to safely leave the area or reach the nearest scram switch.

Detectors to monitor the X- and γ-ray and neutron radiation levels in various areas have been installed in some laboratories. Such area monitoring is not necessary if the radiation level is unlikely to change. Area monitors are sometimes used to switch off the beam or give a warning if the permissible level has been exceeded. One has to remember that even if the charging supply is switched off, a high voltage – and therefore an increased radiation level – can be obtained because of self-charging, whenever the belt (or chain) is moving.

It is important that survey and monitoring instruments are properly maintained and calibrated. The use of a check source for verifying the fitness of the detector prior to each measurement is highly recommended. At least every second year, the absolute sensitivity of the detector should be controlled against secondary-standard instruments or calibrated in a standards laboratory.

17.A Appendix: Recommendations by ICRP

Most national regulatory standards in the field of ionizing-radiation protection are based on recommendations issued by the International Commission on Radiological Protection (ICRP). Two principles are recommended for practices involving exposure to a workforce: firstly, the exposure must be justified, and secondly, all justified exposures must be optimized [5]. Justification means that no practice should be adopted unless the benefits to exposed individuals or to society balance out the harm caused by the radiation exposure. The aim of the optimization is to keep all exposures as low as readily achievable (ALARA), taking also economic and social factors into account.

In order to guarantee that justified and optimized exposures do not lead to unacceptable risks to any individual, the ICRP also recommends dose limits. The basic recommendation for workers is that the effective dose E should be kept below 100 mSv average over any consecutive period of 5 years and below 50 mSv in any single year. The annual equivalent dose to skin, hands and feet should be limited to 500 mSv, and for the lens of the eye, the corresponding value is 150 mSv.

Additional restrictions apply to occupational exposure of pregnant women. The ICRP recommends that the working conditions of a pregnant worker, after the declaration of pregnancy, should be such that it is unlikely that the additional equivalent dose to the conceptus will exceed about 1 mSv [21].

The recommendations by the ICRP are extensive, but generic in character, and for operational radiation protection work, applicable national legislation and local radiation safety rules need to be consulted for more practical guidance.

17.B Appendix: Half-Value Layer (HVL) and Fluence–Dose Conversions

17.B.1 HVL for Photons

The thicknesses needed to attenuate the fluence rate of primary photons to half of its value for the photon energy range between 0.1 keV and 3 MeV are given in Table 17.6.

Table 17.6. The thickness needed to attenuate the fluence rate of primary photons to half of its value (the HVL) for different materials. Note that in broad geometries, the contribution from secondary photons can be significant. In narrow-beam geometries, the HVL is related to the attenuation coefficient μ by HVL $= (\ln 2)/\mu$

Photon Energy keV	Air m	Water cm	Al mm	Fe mm	Pb mm	Glass mm	Concrete mm
0.1	1.1×10^{-4}	1.2×10^{-5}	–	–	–	–	–
1	1.5×10^{-3}	1.7×10^{-4}	2.4×10^{-3}	1.0×10^{-3}	1.2×10^{-4}	8.9×10^{-4}	8.7×10^{-4}
3	3.3×10^{-2}	3.6×10^{-3}	3.3×10^{-3}	1.7×10^{-3}	3.1×10^{-4}	5.5×10^{-3}	6×10^{-3}
10	1.1	0.13	0.10	5.2×10^{-3}	4.7×10^{-3}	0.17	0.11
30	15	1.9	2.3	2.3	2.0×10^{-2}	3.5	2.4
100	35	4.1	15	2.4	0.11	17	16
300	50	5.8	25	8.0	1.5	26	27
1000	84	9.8	42	15	8.6	44	46
3000	150	17	73	24	14	77	80

17.B.2 Photon Effective-Dose Factors

The effective dose (in pSv) per unit photon fluence (in cm^{-2}) for a standard adult person, as a function of photon energy, is illustrated in Fig. 17.3a. Curves for three different irradiation geometries have been included.

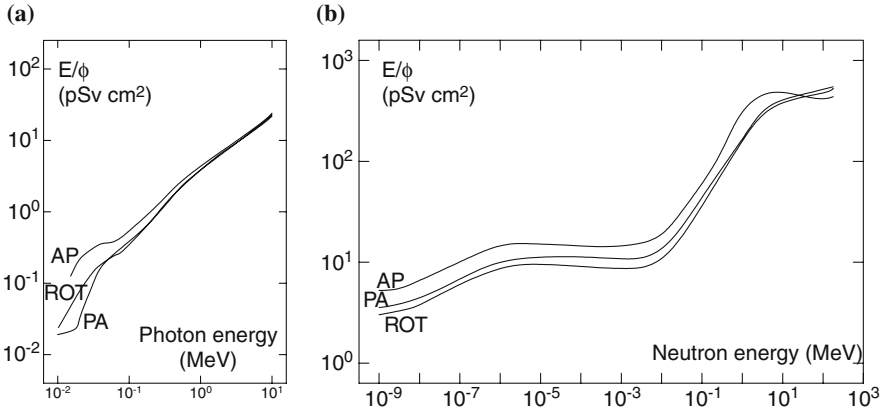

Fig. 17.3. The effective dose E per unit photon (**a**) and neutron (**b**) fluence in units of pSv cm^2 for a standardized adult person. AP = irradiation in the direction front to back, PA in the direction back to front, and ROT in all horizontal directions [24]

17.B.3 Neutron Cross Sections

Neutron interaction cross sections vary widely with the neutron energy and target material. This complexity of the interaction makes accurate neutron-shielding calculations demanding and extensive, and calculations utilizing intricate computer programs are the norm. Databases covering neutron cross sections, thick-target yields etc. can be found on the Internet, for example at http://www-nds.iaea.org/ and http://www.nndc.bnl.gov/nndc/exfor/.

17.B.4 Neutron Attenuation

Several textbooks and other publications have gathered data for neutron transmission through concrete and other materials (e.g. [2, 25]). The transmission factor depends on the initial neutron energy distribution, the angle of incidence, the isotopic content of the shielding material, the quantity of interest etc. The allotted space does not, unfortunately, permit us to cover the complex field of neutron attenuation in this book.

17.B.5 Neutron Effective-Dose Factors

The effective dose (in pSv) per unit neutron fluence (in cm^{-2}) for a standard adult person, as a function of the neutron energy, is illustrated in Fig. 17.3b. Curves for three different irradiation geometries have been included.

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