

DOSIMETRY PRINCIPLES, DOSE MEASUREMENTS AND RADIATION PROTECTION

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1. INTRODUCTION

In radiation processing dosimetry is used to quantify the energy deposited in a material or absorbed by a human from radiation sources.

Different dosimetry systems are used for different purposes in industry and research irradiation facilities, which have different requirements for dose determinations. Radiation safety standards and issues involving the radiation protection of humans against radiation exposure have their own dosimetry metrology.

Radiation dosimetry is a branch of physical science exploring different methods for the quantitative determination of energy, which is deposited in a given material by ionizing radiation, either through direct or indirect exposure. Dosimetry deals with determinations and calculations of quantities (dose) that describe the energy absorbed in a material and to some extent its rate of deposition (dose rate). Dosimetry determinations that are performed by exposing a dosimeter to a radiation source help in evaluating the radiation-induced effects, physical, chemical, and/or biological, on an irradiated material [1].

To assure that the desired radiation effects (biological, chemical, and/or physical) are achieved and that the irradiation process is performed safely, validation and process control procedures are implemented. Process controls rely on the establishment of a relationship between the source parameters and the absorbed dose in an irradiated object (for isotropic radioactive source irradiators: dwell time, position in the source rack, and conveyance speed; for accelerator sources: beam voltage, beam current, scanning width, scanned uniformity and conveyance speed). Absorbed dose and dose distribution is

inferred from determinations made with a suitable dosimetry system having some level of accuracy and precision [2].

Many dosimetry systems used in the radiation processing of materials are implemented in accordance with corresponding ISO/ASTM international standards (www.astm.org): ISO/ASTM 52628:2013 (general issues) [3], ISO/ASTM 51261:2013 (calibration) [4], ISO/ASTM 52701:2013 (characterization) [5], ISO/ASTM 52303:2015 (dose mapping) [6], ISO/ASTM 51707:2015 (uncertainties) [7], ASTM E2232-10 (mathematical modelling) [8], taking into account that the irradiation facilities also fulfil the standard requirements: ISO/ASTM 51649:2015 (high energy electron beam) [9], ISO/ASTM 51818:2013 (low energy electron beam) [10], ISO/ASTM 51608:2015 (X-ray beam) [11], ISO/ASTM 51702:2013 (gamma facility) [12]. These standards for dosimetry systems will be referenced in the appropriate sections below.

2. DOSIMETRIC QUANTITIES

A quantitative radiation dose is used:

- to predict associated radiation effects and possible material transformation and modification caused by irradiation;
- to ensure that overall radiation protection and safety are implemented when working with ionizing radiation.

A number of quantities and units have been defined by the International Commission of Radiation Units and Measurements (ICRU Report 60) [13] for describing the radiation sources. The most commonly used quantities in dosimetry and their units are listed in Table 1.

It was already noted in Chapter 2 “Radiation interaction with condensed matter” that absorbed dose is the main characteristic that allows the evaluation of radiation processing’s impact on an irradiated material.

The absorbed dose is related to the stochastic quantity of the energy imparted by a source which is the sum of all energies entering the volume of interest, minus all of the energy leaving this volume, taking into account any mass energy conversion within this volume [14] and is defined as follows:

$$D = \frac{d\bar{\epsilon}}{dm} \quad (1)$$

where: $\bar{\epsilon}$ is the mean energy imparted, and dm is the mass of a finite volume, V . The unit of absorbed dose is joule per kilogram (J/kg), or gray (Gy).

In the case of uncharged radiation (photons and indirect ionizing radiation), energy is imparted to matter in a two-step process.

Table 1. Radiation beam quantities (dN – a number of particles incident on a sphere of a cross-sectional area dA ; E – the energy of particle; $\Phi_E(E)$ and $\Psi_E(E)$ – short notations for the particle fluence spectrum and the energy fluence spectrum differential in energy E , respectively).

Monoenergetic beam	Polyenergetic beam
Particle fluence, Φ [m^{-2}]	
$\Phi = \frac{dN}{dA}$	$\Phi_E(E) = \frac{d\Phi}{dE}(E)$
Energy fluence, Ψ [J/m^{-2}]	
$\Psi = \frac{dE}{dA} = \frac{dN}{dA} E = \Phi E$	$\Psi_E(E) = \frac{d\Psi}{dE}(E) = \frac{d\Phi}{dE}(E)E$
Particle fluence rate, $\dot{\Phi}$ [$\text{m}^{-2} \cdot \text{s}^{-1}$]	
$\dot{\Phi} = \frac{d\Phi}{dt}$	
Energy fluence rate (intensity), $\dot{\Psi}$ [$\text{J}/\text{m}^2 \cdot \text{s}$]	
$\dot{\Psi} = \frac{d\Psi}{dt}$	

In the first step, the indirect ionizing radiation transfers energy to secondary charged particles. The mean energy transferred from indirectly ionizing radiation to charged particles (electrons) per unit mass of the material is defined as kerma (an acronym from: kinetic energy released per unit mass) [15]:

$$K = \frac{d\bar{E}_\alpha}{dm} \quad (2)$$

In the second step, these charged particles transfer some of their kinetic energy to the material (resulting in absorbed dose) and lose some of their energy in the form of radiative losses (bremsstrahlung, annihilation).

Different possibilities of energy absorption in the volume due to photon interactions with material are shown in Fig. 1.

Energy absorbed in the volume V of matter is expressed as:

$$E_{ab} = (\sum \varepsilon_i)_1 + (\sum \varepsilon_i)_2 + (\sum \varepsilon_i)_3 + (\sum \varepsilon_i)_4 \quad (3)$$

where $(\sum \varepsilon_i)_1$ is the sum of energy lost by collisions along the track of the secondary particles within the volume, V .

Secondary electrons are losing their energy along their tracks (blackened parts of the tracks shown in Fig. 1). Energy is not absorbed at the same location where it was first transferred to an electron.

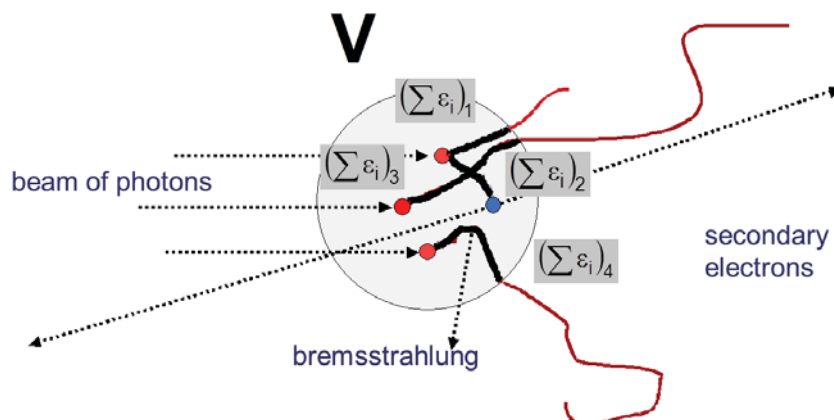


Fig.1. Illustration for the energy absorbed in volume. (Adapted from Ref. [16]).

In the case of the charged particles beam (electrons), most of the energy is directly absorbed at the place within the volume where it was deposited (energy absorption). The absorbed dose to a material or medium is related to electron fluence in the material or medium:

$$D_{\text{med}} = \Phi_{\text{med}} \left(\frac{S_{\text{col}}}{\rho} \right)_{\text{med}} \quad (4)$$

where $(S_{\text{col}}/\rho)_{\text{med}}$ is mass collision stopping power of the material or medium for the electrons penetrating material or medium with a certain energy.

Due to the slowing down of electrons in the material or medium, a primary fluence spectrum, $\Phi_{\text{med},E}$, that ranges from the kinetic energy, E_k , down to zero, will be always present, even if the electron beam is defined as monoenergetic. So absorbed dose to the material or medium is an integral of D_{med} (Eq. (4)):

$$D_{\text{med}} = \int_0^{E_{\text{max}}} \Phi_{\text{med},E} \left(\frac{S_{\text{col}}}{\rho} \right)_{\text{med}} (E) dE = \Phi_{\text{med},E} \left(\frac{\bar{S}_{\text{col}}}{\rho} \right)_{\text{med}} \quad (5)$$

The dose rate, which is another important parameter in materials processing, may be obtained by differentiation of absorbed dose:

$$\dot{D} = \frac{dD}{dt} \quad (6)$$

In materials processing, one should take into account dose rate effects. High dose rates lead to the creation of high free radical concentrations \rightarrow back reactions \rightarrow reduced effect; low dose rates lead to the consumption of dissolved oxygen \rightarrow less peroxy radicals \rightarrow reduced effect [1]. The same reduction of effects may be obtained by irradiating materials in an oxygen-free atmosphere.

It is possible to calculate the dose rate from the natural radioactive sources (^{60}Co , ^{137}Cs , *etc.*) that are used in different applications such as medical product sterilization. The calculation is based on the evaluation of the exposure at a certain distance from the source and followed by the conversion of the exposure to absorbed dose.

The exposure, X , is usually defined as the sum of the electric charges on all ions of one sign that are produced when all electrons liberated by the radiation in a volume of air are completely stopped, divided by the mass of air in that volume: $X = dQ/dm_{\text{air}}$. An exposure X of 1 C/kg provides an absorbed dose in air, D_{air} , of 33.97 Gy [13]:

$$D_{\text{air}} [\text{Gy}] = \frac{W_{\text{air}}}{e} X \quad (7)$$

with $W_{\text{air}}/e = 33.97 \text{ J/C}$, where W_{air} is the average energy, expended in air to produce ion pair.

The dose to a material or medium, D_{med} , is related to the dose in air, D_{air} , at the same location:

$$D_{\text{med}} = 33.97 \frac{\left(\frac{\mu_{\text{en}}}{\rho}\right)_{\text{med}}}{\left(\frac{\mu_{\text{en}}}{\rho}\right)_{\text{air}}} X \quad (8)$$

The exposure rate in air, \dot{X} , is inversely proportional to the squared distance from the point source of activity, $A(t)$ [17]:

$$\dot{X} = \frac{\Gamma A(t)}{r^2} \quad (9)$$

where Γ is the specific exposure gamma-ray constant at 1 m distance from the source.

Specific exposure gamma-ray constants for the nuclides, most commonly used in radiation processing, are: $\Gamma(^{60}\text{Co}) = 2.50 \times 10^{12} \text{ Cm}^2/\text{kg}\cdot\text{MBq}\cdot\text{s}$ and $\Gamma(^{137}\text{Cs}) = 6.64 \times 10^{13} \text{ Cm}^2/\text{kg}\cdot\text{MBq}\cdot\text{s}$ [18]. Γ values, provided in different literature sources, differ slightly due to new recalculations of the radionuclide properties. Instead of a specific exposure gamma-ray constant, a specific dose gamma-ray constant might also be used when exposure is converted to absorbed dose. The formulae used are valid only for point sources. Field arrangements and geometry factors must be taken into account if calculating the dose from the complex sources, such as pencil sources in gamma cell, or sources in the racks of the ^{60}Co unit.

3. DOSE DETERMINATIONS

3.1. STANDARDS

Radiation measurements cover a broad area of instruments and methods focusing on assessing different parameters of radiation processing. A radiation dosimeter is a device, instrument or system that measures or evaluates, either directly or indirectly, dosimetric quantities. A dosimeter along with its reader is referred to as a dosimetry system. The inference of dose is extremely important in industrial and research uses as well as in clinical applications, which are not covered in this book.

Validation, verification and radiation process control depend on the assessment of absorbed dose. The main requirements for performing proper assessments of absorbed dose and related quantities) should be fulfilled [3, 5, 19-21]:

- Assessments of absorbed dose shall be performed using a dosimetry system or systems having some level of accuracy and precision.
- The calibration of each dosimetry system shall be traceable. Traceability is defined as a property of the result of a test that can be related to stated references, usually national or international standards, through an unbroken chain of comparisons, all having stated uncertainties. For example, certificates issued by a national metrology laboratory in compliance with the International Committee on Weights and Measures (CIPM; Bureau International des Poids et Mesures – BIPM) Mutual Recognition Arrangement (MRA) or any other laboratory accredited to ISO/IEC 17025:2005 can be taken as proof of traceability and no further action is required by the user [22].

Accurate, traceable dose determinations provide independent, inexpensive means for quality control in radiation processing.

The classification of dosimetry systems (ISO/ASTM 52701:2013) [5] is based on the metrological properties of a dosimetry system (Type I and Type II) and field of its application. Reference or standard systems are Type I. Systems for routine use are Type II.

- Type I dosimeter – a dosimeter of high metrological quality; its response is affected by individual influence quantities in a well-defined way so, that it can be expressed in terms of independent correction factors.
- Type II dosimeter – a dosimeter, the response of which is affected by influence quantities in a complex way that cannot practically be expressed in terms of independent correction factors.
- Reference standard systems (Type I) are used to calibrate dosimeters for routine use and therefore require high metrological quality; low uncertainty and traceability to appropriate national or international standards are needed.

- Routine systems (Type II) are used for routine absorbed dose assessments, such as dose mapping and process monitoring. Traceability to national or international standards is needed.

The hierarchy of standard dosimetry systems as well as particular ISO standards for radiation processing are shown in Fig.2.

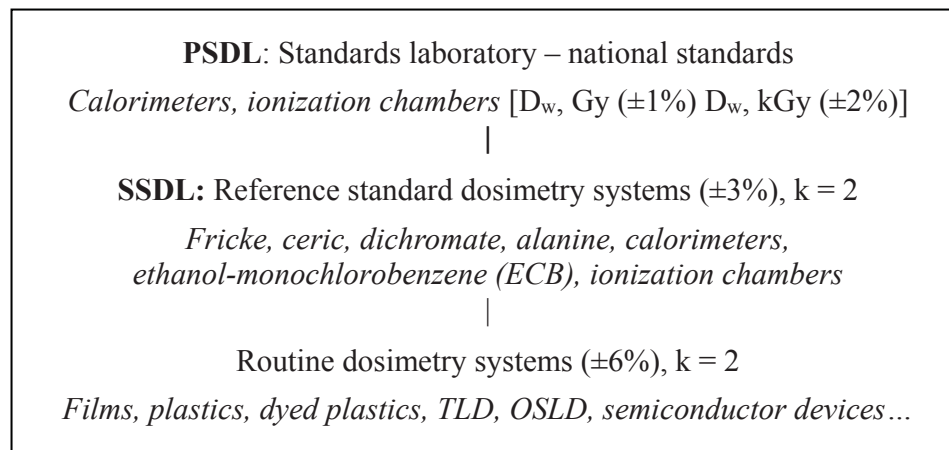


Fig.2. Dosimetry standards hierarchy.

Primary standards are instruments of the highest metrological quality that permit determination of the unit of a quantity from its definition. The accuracy of primary standards has been verified by comparison with standards of other institutions operating at the same metrological level. Primary standards are recognized by the primary standards dosimetry laboratories (PSDLs) in about 20 countries worldwide. Regular international comparisons between the PSDLs and with the CIPM (BIPM) ensure international consistency of the primary dosimetry standards. Radiation detectors used for the calibration of radiation sources for industry, research or medicine, must have a calibration coefficient traceable (directly or indirectly) to a primary standard. Primary standards are not used for routine calibrations, since they represent the unit for the quantity at all times. Instead, the PSDLs calibrate dosimeters for secondary standards dosimetry laboratories (SSDLs) that in turn are used for calibrating the reference instruments of users, such as therapy level ionization chambers (hospitals) or calorimeters (radiation processing). This allows for the transfer of information from an accredited standards laboratory to an irradiation facility with established traceability (comparing absorbed dose assessments). Reference dosimeters are further used for calibration of dosimetry systems for the routine determination of absorbed dose.

The result of a dose determination is only an approximation or estimate of the dose value and is complete only when accompanied by a quantitative statement of its uncertainty. In dosimetry, the uncertainty associated with the determinations is often expressed in terms of accuracy and precision (Fig.3).

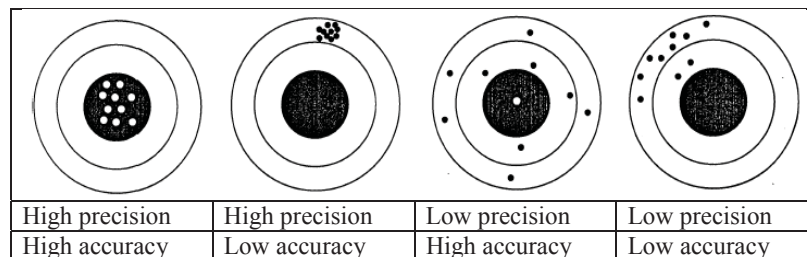


Fig.3. Illustration of precision and accuracy. (Adapted from Ref. [16]).

Accuracy specifies the proximity of the mean value of a measurement to the specified value.

Precision specifies the degree of reproducibility of a determination. High precision is associated with a small standard deviation of the distribution of determination results.

3.2. CALIBRATION AND VERIFICATION OF DOSIMETRY SYSTEMS

In radiation processing, validation and process control (as for sterilization, food irradiation, *etc.*) and the whole irradiation performance depend on the assessment of absorbed dose. This is performed using dosimetry systems having an acknowledged level of accuracy and precision. In dosimetry, accuracy is associated primarily with a quantity, which can be directly traceable to a primary standard system. Traceability is achieved by the calibration of a dosimetry system. Calibration is performed by determining the relationship between dosimeter response and the absorbed dose of reference or a primary standard. The effects of the influence factors, such as dose rate, temperature, storage time, storage conditions, humidity and light, should be minimized to attain optimal calibration conditions. Calibration is needed for reference standard dosimeters as performed in national or accredited laboratory; for transfer standard dosimeters, as performed according to class distinction requirements; and for routine dosimeters, as performed in a calibration laboratory, or in-plant facility. Primary standard dosimeters do not require any calibration.

Both components of a dosimetry system – dosimeters and the measuring or read-out equipment – shall be calibrated and traceable to internationally recognized standards:

- Calibration of measurement equipment is performed measuring certain parameters of the equipment and comparing them with a set of reference values. If measurement equipment cannot be calibrated (*e.g.* signal amplitude from an EPR (electron paramagnetic resonance) spectrometer), the stability of the equipment has to be demonstrated by the use of measurement standards (*e.g.* stable EPR spin standards).

- Calibration of dosimeter is performed by following a sequence of steps: a) irradiate the dosimeters, b) read the information provided by the dosimeters using a calibrated instrument, c) generate a calibration curve or response curve, d) compare the response curve with an initial calibration verification having periodic confirmation of validity, and e) compliance with a traceability chain that provides consistency of the performed measurements with the appropriate national or international standard.

More details on calibration procedures are found in ISO/ASTM 51261:2013 [4], and NPL Report CIRM 29 [23].

In order to verify the calibration, calibration curves prepared for routine dosimeters in a calibration laboratory or in an in-house calibration facility should be verified for the actual conditions of use in the irradiation production facility. Routine dosimeters should be irradiated together with reference or transfer standard dosimeters to at least three different absorbed doses. Absorbed dose results given by two types of dosimeters should be analysed with respect to any systematic trends for potential corrections if needed.

3.3. UNCERTAINTIES

The current methodology for estimating measurement uncertainty is given in the document “Evaluation of measurement data – Guide to the expression of uncertainty in measurement” (JCGM 100:2008) [24]. This is referred to as the “GUM”, which defines the uncertainty as a parameter associated with the result of a measurement that characterizes the dispersion of measured values (= range of the values) that could reasonably be attributed to the measurand (= absorbed dose). This document is prepared in line with a standard ISO/ASTM 51707:2015 [7]. Uncertainties are based on probabilities and are often expressed as standard deviations (standard uncertainties), σ . The distribution of possible values often approximates a Gaussian or normal distribution (Fig.4).

There two types of uncertainties are investigated in performing measurements: type A (random) and type B (non-random, systematic):

- Type A standard uncertainties, u_A , are evaluated by statistical analysis of series of measurements (*e.g.* standard deviation of the mean) and are related mainly to some precision of the dosimeter response.

If a measurement of a dosimetric quantity x is repeated N times, then the best estimate for x is the arithmetic mean of all measurements x_i :

$$\mu = \bar{x} = \frac{1}{N} \sum_{i=1}^N x_i \quad (10)$$

The standard deviation σ_x is used to express the uncertainty for an individual result x_i :

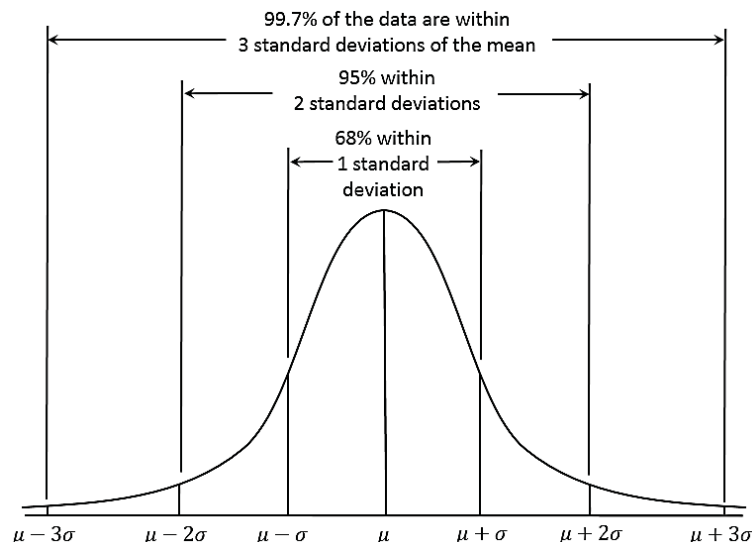


Fig.4. Gaussian distribution of measurement probability. Values μ corresponds to the arithmetic mean of all measurements; σ is standard deviation. The probability of a value being within $\pm 1\sigma$ is $\sim 68\%$ (coverage factor $k = 1$); the probability of a value being within $\pm 2\sigma$ is $\sim 95\%$ ($k = 2$); the probability of a value being within $\pm 3\sigma$ is $\sim 99.7\%$ ($k = 3$). (Adapted from Ref. [25]).

$$\sigma_x = \sqrt{\frac{1}{N-1} \sum_{i=1}^N (x_i - \mu)^2} \quad (11)$$

The standard deviation of the mean value is used to express the uncertainty for the best estimate:

$$u_A = \sigma_{\bar{x}} = \frac{1}{\sqrt{N}} \sigma_x = \sqrt{\frac{1}{N(N-1)} \sum_{i=1}^N (x_i - \mu)^2} \quad (12)$$

- Type B standard uncertainties, u_B , cannot be estimated by repeated measurements and are evaluated by means other than statistical analysis (based on judgement and previous experimental data). Type B uncertainties include influences on the measuring process, the application of correction factors or physical data taken from the literature and are related to a calibration (accuracy). Type B evaluations vary, but their outcomes should be converted into a standard uncertainty in order to allow for the mathematical combination of all components of uncertainty.

Having evaluated standard uncertainties associated with each component of a measurement, the combined uncertainty, u_C , associated with a particular measurement is obtained by summing in quadrature standard uncertainties of the individual component, *i.e.* by taking the square root of the sum of the squares of the individual components:

$$u_c = (u_1^2 + u_2^2 + u_3^2 + \dots)^{1/2} \quad (13)$$

The results of such analyses are often presented in a table, known as an uncertainty budget. An example of uncertainty budget of routine polystyrene calorimetric dosimetry system is provided in Table 2.

Table 2. Measurement uncertainties of routine polystyrene calorimetric dosimetry systems from Risø High Dose Reference Laboratory, at $k = 2$. (Data are adapted from Ref. [26]).

Component of uncertainty	Type A [%]	Type B [%]
Calibration of irradiation dose		3.2
Temperature measurement of calorimeter (at 3 kGy)	1.0	
Temperature extrapolation of calorimeter (at 3 kGy)	1.0	
Change of temperature sensitivity of specific heat of polystyrene		0.5
Heating effects	0.5	
Quadrature sum	1.5	3.2
Overall quadrature sum	3.5	

The combined uncertainty u_c is assumed to exhibit a Gaussian distribution (confidence level of 68%).

The expanded uncertainty, $U = k \times u_c$, with a coverage factor $k = 2$, corresponding to 95% confidence level, is often used to represent overall uncertainty, which relates to the accuracy of the measurement of a certain quantity.

4. DOSIMETRY SYSTEMS

Quality assurance programmes and implemented quality control measures are aimed to prove that the irradiation process was carried out within prescribed dose limits. This requires careful selection of the dosimetry system and its proper use after selection. The scope of application is one of the most important criteria for the selection of a suitable dosimetry system:

- field (industry, medicine, research and development),
- process to be controlled,
- standard (primary standard, reference, transfer, routine dosimetry),
- measured quantity (absorbed dose, dose rate and other dosimetric quantities, dose mapping and others),
- physical properties of dosimeter (radiation-induced physical and chemical processes in detector material that are related to the measured or evaluated dosimetric quantity),

- physical state (solid, liquid, gaseous, combined),
- measurement type (internal, external),
- measurement mode (*in-situ* or *ex-situ*),
- target substance related measurement (*in vivo*, *in vitro*).

Dosimetry systems are based on measurement of radiation-induced physical or chemical changes in the detector material which can then be attributed to the absorbed dose. Thus, dosimetry methods are classified as:

- ionization based (ionization chambers),
- temperature change related (calorimeters),
- thermoluminescence (LiF),
- colour change related (Perspex™, radiochromic systems),
- free radical concentration change related (alanine),
- conductivity change related (ECB, alanine solution),

Table 3. Dosimeter system standards. (Compiled on basis of Ref. [28]).

Dosimeter system	Method of analysis	Useful dose range [Gy]	Nominal precision limits [%]	References
Fricke solution	UV spectrophotometry	$3 \times 10^{-4} - 10^2$	1	ASTM E 1026-04 [29]
Ceric-cerous sulphate	UV spectrophotometry	$10^3 - 10^6$	3	ISO/ASTM 51205:2009 [30]
Potassium dichromate	UV-VIS spectrophotometry	$5 \times 10^3 - 4 \times 10^4$	1	ISO/ASTM 51401:2013 [31]
Ethanol-mono-chlorobenzene	Titration, or HF oscillometry	$4 \times 10^2 - 3 \times 10^5$	3	ISO/ASTM 51538:2009 [32]
L-alanine	EPR	$1 - 10^5$	0.5	ISO/ASTM 51607:2013 [33]
Perspex systems	VIS spectrophotometry	$10^3 - 5 \times 10^4$	4	ISO/ASTM 51276:2012 [34]
FWT-60 film	VIS spectrophotometry	$10^3 - 10^5$	3	ISO/ASTM 51275:2013 [35]
B3/GEX film	VIS spectrophotometry	$10^3 - 10^5$	3	ISO/ASTM 51275:2013 [35]
Cellulose triacetate	UV spectrophotometry	$10^4 - 10^6$	3	ISO/ASTM 51650:2013 [36]
Calorimetry	Resistance/temperature	$1.5 \times 10^3 - 5 \times 10^4$	2	ISO/ASTM 51631:2013 [37]
LiF (Sunna film)	Optically stimulated luminescence	$50 - 3 \times 10^3$	3	ASTM E2304-03 [38]
TLD	Thermoluminescence	$1 - 10^4$	2	ISO/ASTM 51956:2013 [39]

- radiation chemical oxidation based (Fricke),
- radiation chemical reduction based (dichromate, ceric-cerous),
- optically stimulated luminescence (OSL, Al_2O_3 , Sunna),
- production of radiation defects in semiconductors (diodes, MOSFETs),
- others.

Information on standards applicable for dosimetry systems used in radiation processing is presented in the Table 3. It can be also found on the web pages: www.iso.org, and www.astm.org/COMMITTEE/E61.htm. Essential and very useful information on the dosimetry systems used in materials processing is found in Refs. [2] and [27]. Table 3 summarizes the description of most of the important dosimetry systems.

Primary standard dosimetry systems

Calorimetry and ionization chambers are considered as primary standard methods to determine dose. Calorimeters are used when establishing standards for dose in the radiation processing of materials. Calorimeters and ionization chambers are used to establish dose for medical applications. Dosimeters for medical applications, not discussed here, are described in detail in several publications including Refs. [15, 40-43].

Calorimetry is the most fundamental method used as a primary standard for absorbed dose, since the measured rise in temperature is the most direct consequence of energy absorption (energy-to-heat conversion) in a thermally isolated mass. Measured energy per unit mass or the average dose to the medium assuming no heat loss is:

$$\bar{D} = \frac{h\Delta T}{1-\delta} \quad (14)$$

where h is specific heat capacity of the medium, and δ is the thermal defect (this small fraction of the energy that does not appear eventually as thermal energy because of a chemical reaction).

The principle of calorimetry is illustrated in Fig.5.

A *graphite calorimeter* is used by several PSDLs as a primary instrument to determine absorbed dose. Graphite is an ideal material for calorimetry. Graphite has a low atomic number, Z , so that all of the absorbed energy reappears as heat, without any loss of heat in other mechanisms (such as heat deflection) and its specific heat capacity is 7.1×10^2 J/K, so that temperature changes of a few $\mu^\circ\text{C}$ can be measured.

Water calorimeter is also used by PSDLs as a primary standard to determine the absorbed dose to water in a water phantom. An example of a water calorimeter is shown in Fig.6.

Water calorimeter at Physikalisch Technische Bundesanstalt [44], is able to determine D_w for ^{60}Co radiation under reference conditions with a standard measured uncertainty of approximately 0.2%. Highly purified water is used due to low operating temperature of 4°C .

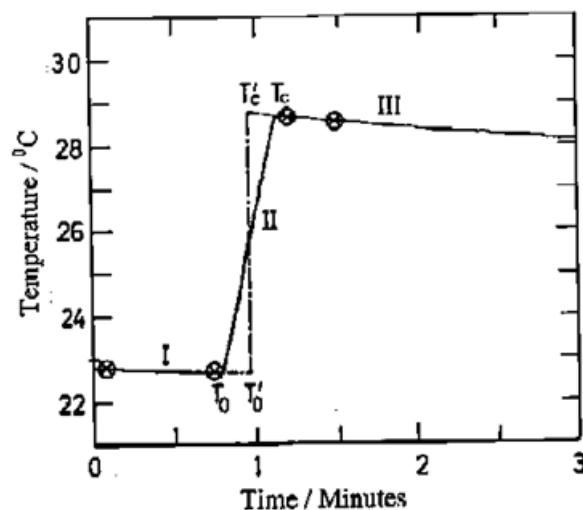


Fig.5. The temperature curves were extrapolated from T_0 and T_c to mid-point of the irradiation time at T_0' and T_c' , respectively. $\Delta T = T_c' - T_0'$ is used for dose calculation. Regions I, II and III are before, during and after the irradiation, respectively. (Adapted from Ref. [15]).

The advantage of using graphite instead of water is a lack of thermal defects.

Graphite, water and polystyrene calorimeters, classified as Type II detectors (ISO/ASTM 52628:2013) [3], are also defined as *process calorimeters*. These calorimeters may be used as internal standards at an electron beam irradiation facility, including their application as transfer standard dosimetry systems for calibration of other dosimetry systems, or they may be used as routine dosimeters provided they are used under static, non-moving conditions. Calorimeters are highly sensitive (water calorimeter – 3.4 kGy/°C, polystyrene calorimeter – 1.4 kGy/°C, graphite calorimeter – 0.75 kGy/°C) to absorbed dose within the range of 1.5-60.0 kGy measured using 4-10 MeV electron beams.

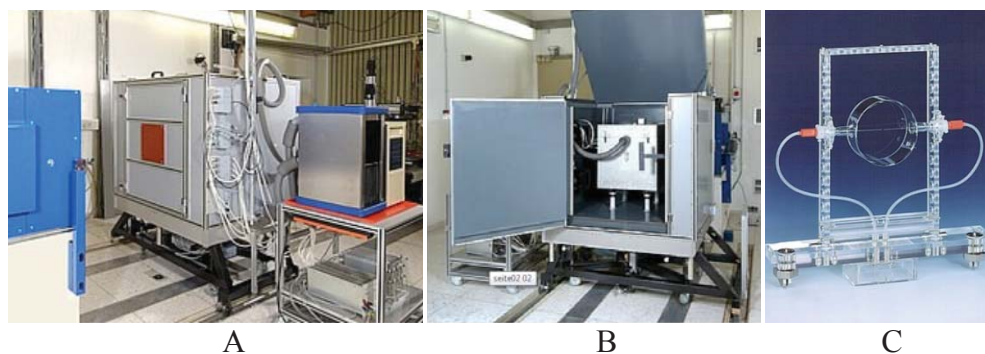


Fig.6. Water calorimeter at Physikalisch Technische Bundesanstalt (PTB, Germany): A – an overall view of the construction, B – cubic water phantom surrounded by polystyrene containing walls, C – calorimetric detector filled with highly purified water. (Adapted from Ref. [44]).

However, calorimetry has not been conducted under the high speed operating conditions common to electron beam processing.

Special polystyrene calorimeters are used for dose measurements at 1.5-4 MeV electron beam energies and that the development of a calorimeter for low energy (80-120 keV) electron beams under static, non-moving conditions is ongoing [28].

Reference standard chemical dosimetry systems

(Fricke, ceric-cerous, potassium dichromate, alanine, ECB dosimeters)

In chemical dosimetry systems, the dose is determined by evaluating the chemical change produced by radiation in the sensitive volume of the dosimeter.

Fricke dosimeters use a water solution of $\text{FeSO}_4(7\text{H}_2\text{O})$ or $\text{Fe}(\text{NH}_4)_2(\text{SO}_4)_2(6\text{H}_2\text{O})$ with additives of sulphuric acid (H_2SO_4) and sodium chloride (NaCl), and are the most widely used chemical dosimetry standard (Fig.7).



Fig.7. Fricke dosimeters: (A) as prepared Fricke solutions, (B) application of Fricke dosimeter in research related to food irradiation. (Adapted from Ref. [28]).

When irradiated, ferrous ions Fe^{2+} are oxidized into ferric ions Fe^{3+} , that exhibit a strong absorption peak at a wavelength of 304 nm, whereas ferrous ions Fe^{2+} do not show any absorption at this wavelength. The response of Fricke dosimeter is expressed in terms of the yield of ferric ions Fe^{3+} . It is nearly independent of the photon and electron energy in the range of 5-16 MeV. The average dose to Fricke solution is given by a change in optical density at 304 nm:

$$\bar{D}_F = \frac{\Delta\text{OD}}{\varepsilon G \rho L} \quad (15)$$

where: ε – the molar extinction coefficient (217.4 l/mol·cm at 25°C), G – the yield of ferric ions Fe^{3+} (1.617×10^{-6} mol/J), ρ – the density of Fricke solution (1.023 kg/dm³ at 25°C), L – the path length over which the optical signal was read (typically 2-4 cm). The absorbed dose within the range of 40-400 Gy can be measured using Fricke dosimeter.

Potassium dichromate is known as reference dosimetry system for gamma facilities. It uses a potassium dichromate ($K_2Cr_2O_7$) and silver dichromate ($Ag_2Cr_2O_7$) solution in perchloric acid ($HClO_4$). Dose measurements within the range of 10-50 kGy are based on colour changes in the solution at a wavelength of 440 nm due to the radiolytic reduction of dichromate ions to chromic ions. The $Ag_2Cr_2O_7$ solution in $HClO_4$ allows the determination of doses down to 2 Gy, but the dose response then (colour change) is observed at 350 nm (Fig.8).

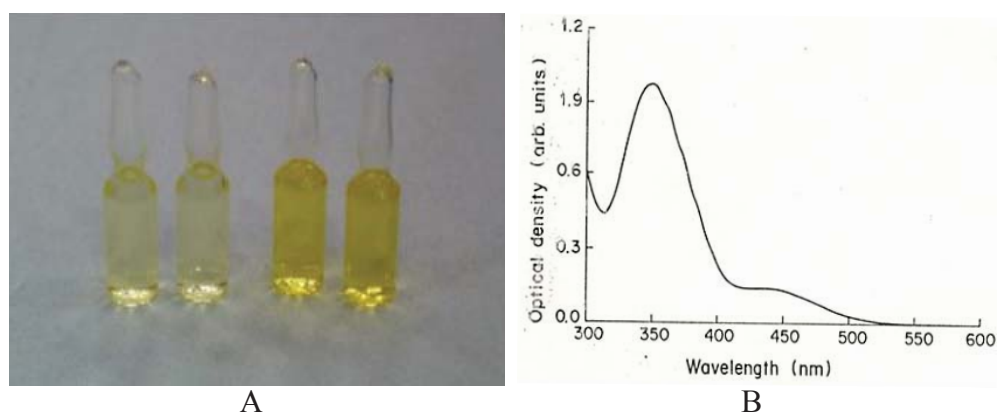
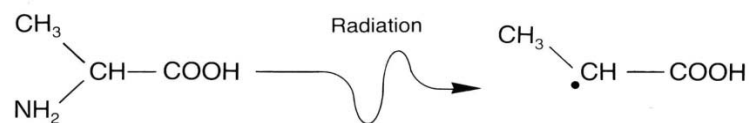


Fig.8. Dichromate dosimeters: A – as prepared (left) and irradiated (right) dosimeters, B – UV-VIS absorbance spectrum of irradiated potassium dichromate. (Adapted from Ref. [28]).

Ethanol-monochlorobenzene dosimeter contains monochlorobenzene (C_6H_5Cl) in an aerated ethanol-water solution. The concentration of monochlorobenzene may vary between 4 and 40 vol% based upon request, but in radiation processing a solution containing 24 vol% of monochlorobenzene is used. This dosimeter is based on the formation of hydrochloric acid (HCl) upon irradiation *via* dissociative electron attachment, since monochlorobenzene is a good electron scavenger and reacts with “dry” and solvated electrons. The determination of absorbed dose is carried out by measuring the concentration of HCl using alkalimetric or mercurimetric titration [45]. This dosimeter is used in reference (also transfer) dosimetry systems for dose measurements at electron and X-ray facilities. Doses from the range of 10 Gy-200 kGy can be measured when the electron beam energy is higher than 4 MeV, while the doses measured at X-ray beams (> 2 MeV) can be from the range of 10 Gy-2 MGy. This dosimeter is nearly independent of irradiation temperature.

Alanine dosimeters (pellets or films) are based on an amino acid which forms stable free radicals when irradiated:



The concentration of free radicals is measured using EPR spectroscopy (Fig.9) and is proportional to the absorbed dose, which can be determined within the range of 10 Gy-100 kGy. Alanine is used for gamma and electron reference dosimetry. However dose response dependency on environmental conditions (humidity, temperature) should be considered when performing measurements.

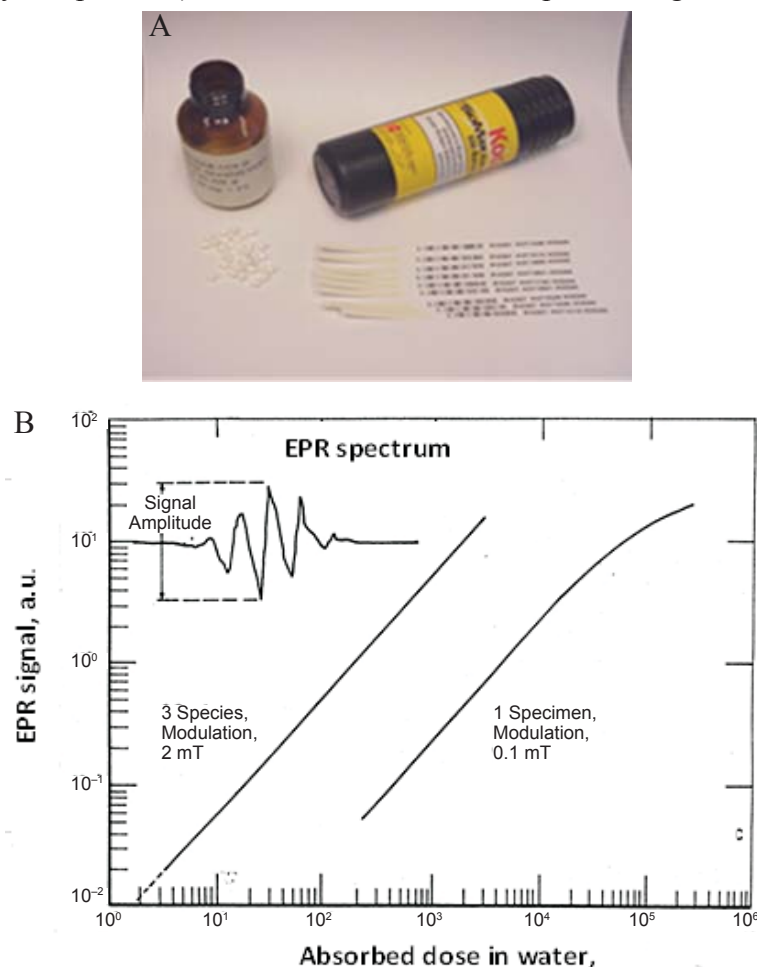


Fig.9. Alanine dosimeters: A – a variety of alanine dosimeters, B – EPR spectrum and measured value: peak to peak signal. (Adapted from Ref. [2]).

Routine dosimetry systems

(Perspex™, films, ECB, ceric-cerous solutions, process calorimeters)

These systems are used in radiation processing facilities for absorbed dose mapping and process monitoring. They require calibration.

Perspex™ is the trade name of poly(methyl methacrylate) ($C_5O_2H_8$)_n. Certain dyes are responsible for the colour of Perspex (red, amber, gammachrome YR). *Perspex dosimeters* darken when irradiated (Fig.10). The increased absorption is a result of radiation-induced free radicals that initially react with

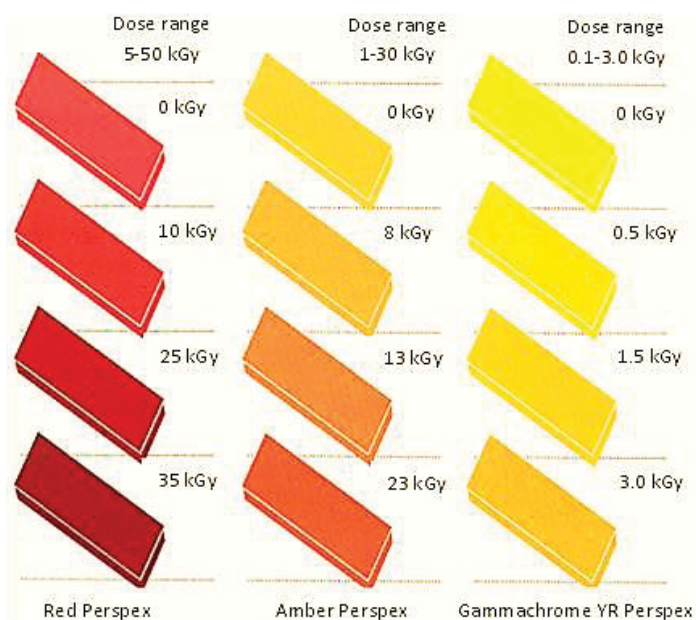


Fig. 10. Different types of Perspex dosimeters and their reaction to irradiation. (Adapted from Ref. [28]).

oxygen present in the polymer to yield peroxy radicals. Radiation-induced colour changes can be accurately measured by means of a spectrophotometer and are related to the absorbed dose. Absorbed doses within the interval of 0.1-3.0 kGy can be determined at 530 nm by the Gammachrome YR Perspex, within interval of 1-30 kGy – at 603 nm by the Amber Perspex and within interval of 5-50 kGy – 640 nm by the Red Perspex dosimeter. Perspex dosimeters are used in routine gamma dosimetry.

Film dosimeters are mainly used for absorbed dose mapping and process monitoring at an irradiation facility. Most important film dosimeters are:

- FTR-125 – cellulose triacetate dosimeters are used for routine dosimetry in electron beam, gamma-ray and ion beam irradiation facilities, mainly for dose mapping. Absorbed dose within the range of 5-300 kGy can be evaluated according to radiation-induced absorbance changes at 280 nm.



Fig. 11. A – irradiated Risø B3 film; B and C – GEX DoseStix and WinDose dosimeters made from Risø B3 film, respectively. (Adapted from Ref. [19]).

- FTW-60 is a colourless radiochromic film containing hexa (hydroxyethyl) pararosaniline cyanide in a nylon matrix. Radiation induces a film colour changes toward a deep blue. Absorbed dose can be estimated for the range of 3-30 kGy, if spectrophotometric optical density measurements are carried out at 605 nm, and for the range of 30-150 kGy, if spectrophotometric optical density measurements are carried out at 510 nm. Film response is independent of the energy and type of radiation (electron beam, gamma ray or X-ray) and of the dose rate up to about 10^{13} Gy/s. These dosimeters are used for process control for gamma as well as for electron beam irradiation.
- Risø B3 (GEX) radiochromic film (Fig.11) is colourless polyvinyl butyral film containing the leucocyanide of pararosaniline. Radiation induces film colour changes to deep pink. Heating at 60°C for 5 to 10 min after irradiation helps stabilize the colour. Absorbed dose can be estimated within the range of 2-100 kGy performing spectrophotometric measurements at 544 nm. Films are widely used in gamma and electron beam radiation processing.
- Gafchromic films are radiochromic films consisting of colourless transparent coatings of polycrystalline substituted diacetylene sensor layers on a clear polyester base. Radiation induces film colour changes to deep blue. Absorbed doses within the range of 1 Gy-40 kGy can be evaluated. Spectrophotometric readings are performed at different wavelengths (670, 633, 600, 500 and 400 nm) depending on the absorbed dose. Films are applicable as a routine dosimeters in medical applications, industrial radiation processing and food irradiation.
- Sunna film [46] is a novel OSL dosimetry system and is made by LiF uniformly dispersed in a polyethylene matrix (Fig.12). Irradiation stimulates formation of colour centres in LiF (F-, M-, N-, R-centre) that correspond to discrete optical absorption bands in the UV-VIS region. The information related to the absorbed dose can be inferred by excitation of the irradiated dosimeter with a light at the wavelength of the colour centre absorption and measurement of a characteristic luminescence at a significantly higher wavelength. The range of the evaluated absorbed dose depends on the absorption wavelength:
 - for the evaluation of UV absorbance at 240 nm, the dose range is between 5 and 100 kGy;
 - for the evaluation of green OSL at 530 nm, the dose range is between 200 Gy and 250 kGy;
 - for the evaluation of near infrared OSL at 670 and 1100 nm, the dose range is between 10 Gy and 10 kGy.The Sunna films are used in both gamma and electron beam processing for dose distribution measurements, as well as for routine process control.
- Tetrazolium films [47] are also based on the colour changes after irradiation. The main component of the film, a tetrazolium salt, is a heterocyclic organic compound which yields highly coloured water insoluble formazans

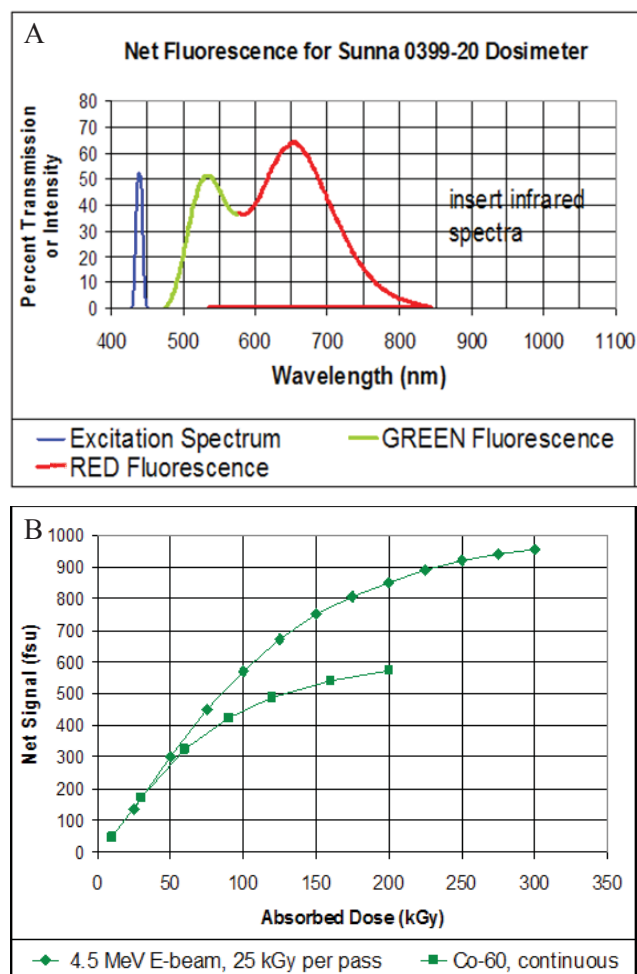


Fig.12. A – Sunna films UV, green or red (IR) OSL absorption bands; B – sensibility of Sunna film (green range) to different irradiations. (Adapted from Ref. [28]).

due to radiolytic reduction. The range of the inference of absorbed dose depends on the measured wavelength at which the absorption peak was identified:

- for tetrazolium violet (TV): 525 nm measured absorbance corresponds to the dose range of 10 Gy-30 kGy;
- for tetrazolium red (TTC): 490 nm measured absorbance corresponds to the dose range of 10 Gy-100 kGy;
- for tetrazolium blue (TB): 520 nm measured absorbance corresponds to the dose range of 10 Gy-10 kGy;
- For nitro blue tetrazolium (NBT): 522 and 612 nm measured absorbance correspond to the dose range of 10 Gy-25 kGy.

Some additional information related to Gafchromic films, Sunna dosimeters, tetrazolium films and other novel dosimetry systems could be found in Ref. [14].

5. RADIATION SAFETY AND RADIATION PROTECTION

Mankind greatly benefits from the use of electron beam, gamma rays and X-rays in industrial processing and in research and development. Radioisotopes and fissionable materials are used in medicine, research, and power generation. These uses involve potential exposure of personnel to radiation. Accidental exposure is also possible. Since radiation exposure presumably involves some risk to the individuals, the levels of exposure allowed should be worth the result that is achieved. The overall objective of radiation protection is to balance the risks and benefits from activities that involve radiation even if these risks and benefits are hardly measurable directly [48].

The IAEA recommendations for radiation protection are provided in the Basic Safety Standards (BSS) books [49, 50]. These basic safety standards represent internationally agreed standards that set out the requirements for a framework to regulate radiation safety in each country and that, in principle, have been accepted by all member states. They are based on knowledge of radiation effects and on established principles of radiation protection, recommended by the ICRP (International Commission on Radiological Protection):

- Benefit of practices must offset radiation detriment.
- Exposures and likelihood of exposure should be kept as low as reasonably achievable (ALARA principle).
- Dose limits should be set to ensure that no individual faces an unacceptable risk under normal circumstances.

The ICRP has proposed safety standards to protect the health of workers and the general public against the dangers arising from ionizing radiation (last upgraded version: ICRP publication 103 [51]). These recommendations were laid down in a European Directive: EU Council Directive 2013/59/EURATOM [52], which has been accepted by the Member States of the European Community.

The Directive has defined safety standards for the exposed workers in the following way:

- The limit on the effective dose is 100 mSv in a consecutive five year period, subject to a maximum effective dose of 50 mSv in any single year. In accordance with this, most Member States have defined an annual limit of 20 mSv. (Sievert, Sv, is equal to 1 joule of energy deposited in a kilogram of human tissue).
- The annual limit on the equivalent dose for the lens of the eye is 20 mSv.
- The annual limit on the equivalent dose for the skin is 500 mSv.
- The annual limit on the equivalent dose for the hands, forearms, feet, and ankles is 500 mSv.

The annual limit for the whole-body dose for the general population, is 1 mSv in most countries.

The ICRU has defined radiation protection quantities for dose limitations to the potentially exposed workers that are different from those described above in the introduction of this chapter: the equivalent dose (accounts for radiation type) and the effective dose (accounts radiosensitivity of different organs and tissues) [13].

Equivalent dose is the dose absorbed in an organ or tissue and multiplied by the relevant radiation weighting factor, w_R :

$$H_{T,R} = w_R D_{T,R} \quad (16)$$

where $D_{T,R}$ is the average absorbed dose in the organ or tissue T, and w_R is the radiation weighting factor for radiation R (alpha particles, electrons, photons, neutrons).

Effective dose is a summation of the tissue equivalent doses, each multiplied by the appropriate tissue/organ weighting factor, w_T :

$$E = \sum_T w_T H_T \quad (17)$$

where H_T is the equivalent dose in tissue T, and w_T is the tissue weighting factor for tissue T.

The unit of equivalent and also effective dose is J/kg, termed the sievert (Sv).

In order to minimize radiation risks to personnel working in potential exposure conditions, relevant radiation protection measures should be imple-

Table 4. Typical personal dosimeters [42].

Dosimeter	Principle of operation	Radiation type, measurement range	Advantages and disadvantages
Film badge	Photochemical blackening	γ, β 0.1 mSv-5.0 Sv	Can be documented, insensitive for low-energy rays
Pen-type pocket dosimeter	Ionization chamber	γ 0.03-2.00 mSv	Very sensitive, permanently readable, insensitive for α - and β -rays, cannot be documented
Permanently readable dosimeter (pocket dosimeter)	Ionization or proportional chambers and GM counters	γ 0.1 μ Sv-10.0 Sv	Permanently readable, cannot be documented
TLD dosimeter	Thermoluminescence measurement	$\gamma, (\beta)$ 0.1 mSv-10.0 Sv	Suitable for low dose measurements, cannot be documented
Phosphate glass dosimeter	Photoluminescence measurement	γ 0.1 mSv-10.0 Sv	Can be documented, can be read repeatedly
Albedo neutron dosimeter	Neutron moderation by the carrier	n, γ 0.1 mSv-10.0 Sv	Calibration depends on human carrier

mented in the work environment and the doses to workers must be permanently controlled and measured using the appropriate dosimeters.

Typical applications of different measurement techniques for personal dosimetry are listed in Table 4. More detailed information on personal dosimeters can be found in Refs. [41, 42, 53].

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