



Ionizing Radiation Metrology



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Ionizing Radiation Metrology

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EDITORIAL NOTE

We are constantly changing. These changes are needed towards new achievements. The accurate information and hard work are key components in the construction of scientific knowledge.

The Ionizing Radiation Metrology Book emerged as an initiative to make accessible to a broader public issues related to metrology of ionizing radiation presented during the First Brazilian Congress of Metrology of Ionizing Radiation (CBMRI), held from November, 23-25, 2014. With this work, we intend to bring a contribution to new generations of researchers and professionals.

The metrology of ionizing radiation experiences a growing demand for high-level technical services in different segments. We must act with quality, conduct research and training professionals with excellence. Based on these roles, the country is strengthening its metrological structure and investing in the workers' radiation protection, general public and environment.

It is important to highlight that metrology has fundamental importance in health, industry, environment, among other sectors, since the entire work will provide a guarantee reliable measurements for the safe performance of activities. So the society can enjoy the benefits of the applications of nuclear technology with maximum reliability.

The focus on the knowledge dissemination, processes quality, realization, maintenance and dissemination of magnitudes related to ionizing radiation in Brazil are more than a mission, they are a challenge. We hope that future generations use that knowledge and construct new research levels and technical and scientific achievements based on sustainability, with great wisdom.

The best result for an institution, their professionals and the enthusiasts of an area is to see that his work benefits millions of people and provides welfare to society. The most direct and immediate return of a research activity.

José Guilherme Pereira Peixoto

FOREWORD

This book, which marks an important milestone in the field of metrology of ionizing radiation is sponsored, among other organizations, by the Technology Brazilian System (SIBRATEC), Brazilian Society of Metrology (SBM), National Institute of Metrology, Quality and Technology (INMETRO) and the Institute of Radiation Protection and Dosimetry (IRD), scheduled for releasing during the I Brazilian Congress of Metrology of Ionizing Radiation (CBMRI). The main purpose of this publication is to review various concepts, fundamental topics and methods related to the primary or secondary measurements of ionizing radiation.

The book is an outgrowth of the authors many years of experience and assembled into chapters, recounts in detail the historical development of measurements with ionizing radiation, as well as guides the reader to the abundant and updated available scientific literature.

Students and professionals who are dealing with experiments, calibration systems and metrological techniques normally applied to health, industry, environment, nuclear safety and radiation protection will certainly benefit from the book. Over all there is an expectation that this volume will become a reference not only for those professionals who are already intrinsically involved in

this field of metrology, as well as for those who are taken the first contact with such measurements.

Following the approach proposed by the BIPM Comité Consultatif pour les Etalons de Mesures des Rayonnements Ionisants, the book is fully devoted to three different aspects of metrology, namely: radionuclides and radioactivity; X-rays, Gamma, Electron and Charged Particles; Neutron metrology. Approaches inherent traceability, primary standard (absolute) and secondary (relative), assessment of uncertainties, nuclear instrumentation, and laboratory infrastructure were also contemplated herein.

Finally, the IRD as manager of the National Laboratory of Metrology of Ionizing Radiation (LMNRI) owe a sincere debt of gratitude to all of the generous colleagues, who kindly wrote the chapters and made valuable contributions to the book. The institute could not fail to express its gratefulness to Dr. José Guilherme Pereira Peixoto, a researcher of this institution, for having the idea, coordinated and editing of the whole process, which led to the publication of this book.

Dejanira Lauria da Costa
IRD Director

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Chapter I

Standard application in photon dosimetry

José Guilherme Pereira Peixoto
Maria da Penha Potiens

The characteristic ionizing radiation response is defined by the relationship between the stimulus and the correspondent response, which converge for many specifications that can be found using specific measurement instruments and that can be used for applications in Ionizing Radiation - IR. However, we looked for a response divided by the corresponding current or charge from the ionizing radiation stimulus; the sensitivity of these measurement instruments is related to their metrological property.

We found many response characteristic examples for measurement devices in different Ionization chambers, shapes and volumes and for solid-state dosimeters from thermoluminescence or semiconductor materials. These results depended on the application: for use as therapy, protection or diagnostic measurement instrument.

The typical secondary standard device that is used for IR for X and gamma rays is the ionization chamber; its discrimination threshold is related not only to the sensitivity of the measuring instrument but also to its stable response overtime and ease of use, i.e., the largest change in a slowly and monotonically changing stimulus that produces no detectable change in the response of a measuring instrument.

Under these conditions, the dc ion chamber shows these characteristics because the charges or current is efficiently

collected, measuring its ion current and considering recombination to be negligible.

1.1 Properties

We could not obtain the measurement directly or invasively but rather non-invasively by looking at the gas cavity. The ionization chamber is the most widely used type of cavity for ionization measurements, is commercially available and has a wide variety of projects and designs for dosimetry applications.

The first view of one ionization chamber cavity is given by a gas volume in the presence of an electric field. The drift of the positive and negative, as represented by the ions and electrons, constitutes an electric current [1], and the rate of the ion pair that is formed is constant and directly proportional to the volume.

For any kind of gas volume, the rate of ion pairs is associated with the gas volume through recombination, diffusion or migration. However, the experimental difficulty of obtaining this ionization measurement generally requires the study of the charge-particle equilibrium.

For additional information and details, see *Radiation detection and measurement* [1], but figure I.1 illustrates the basic elements of a rudimentary ion chamber. A volume of gas is enclosed within a region in which an electric field can be created by the application of an external voltage.

At equilibrium, the current flowing into the external circuit is equal to the ionization current collected at the electrodes, and a sensitive ammeter in the external circuit can thus measure the ionization current. Usually, the electrometer that is connected to the chamber is used as a sensitive ammeter to measure the collected charge in real time.

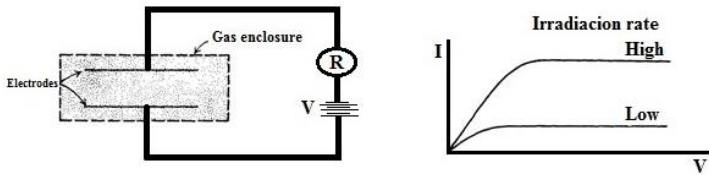


Figure I.1: The basic components of an ion chamber and the corresponding current-voltage characteristics [1].

An ionizing radiation chamber device should be connected to the measurement system, where the ionizing chamber serves as the sensor, connected to a power supply and an electrometer or a lower device counter, and at the end, displaying a device that is also connected to the air density measurement device.

1.2 Absolute dosimeter

The absolute dosimeter, similar to a free air chamber, as used to determine the photon X ray tube potential for diagnostic, protection and therapy applications has various special designs. However, it depends on this requirement to replace lost electrons [2] and has evolved into so many primary standard dosimetry laboratories in the countries.

We found that more traditional design is the plane-parallel geometry, as in figure I.2, where the plate system inside the box consists of three coplanar plates on one side of the beam and a parallel high-voltage plate opposite. The plates are all parallel to and equidistant from the X ray beam axis [2].

All types of free air ionization chambers are enclosed in a Pb-shielding box to exclude X rays scattering from elsewhere, and a tungsten-alloy diaphragm is at the front of the box and aligned with the X ray beam. Thus, the beam passes across of the section area in the plane of the axial point.

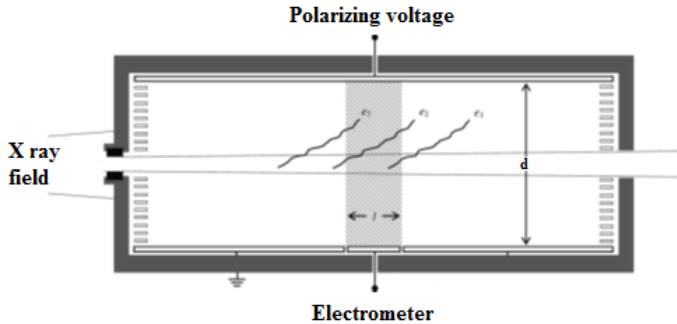


Figure I.2: Free-air chamber schematic diagram. Photons enter through an aperture of radius r_{ap} and interact with the air of the chamber to produce secondary electrons (e_1 , e_2 , and e_3). If the electrode separation d is sufficiently large, the secondary electrons come to rest within the chamber. In the course of slowing down, charge is liberated and swept in the electric field between the electrodes. An isolated section of electrode creates an air region of length l (shaded) from which charge is collected and measured as ionization current [3].

A cylindrical chamber, such as a variable volume of free air ionization chamber, was proposed by Attix [2, 4].

These new free-air ionization chambers are made by two telescopic cylinders, where the air can move independently, and the variable air volume, which is maintained at the same longitudinal axis, opens and closes the chamber volume, as shown in figure I.3. The polarizing potential is applied to a pair of telescoping cylinders [5, 6].

The collecting electrode is an off-axis rod that extends the entire length of the cylinders; this arrangement has a non-uniform electric field but does not interfere with the operating principle of the chamber [7].

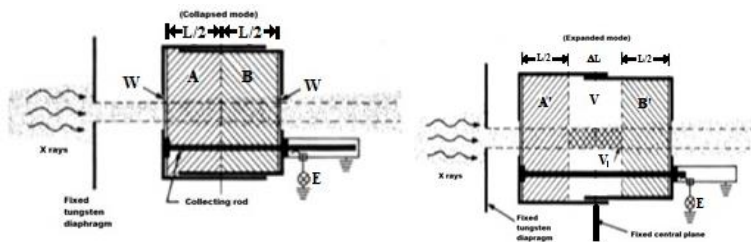


Figure I.3: Design of the cylinder free-air ionization chamber with variable volume [2,4].

These chambers show differences between the plane parallel; as in geometry and operations, knowing the charge-particle equilibrium and electronic loss, electric field uniformity and better air mass definition are not needed because the electrode collector length uncertainty is eliminated.

The point (P) at the border of the diaphragm is the reference point at the cavity chamber to be compared with the free air chamber when centered. The diaphragm is positioned such that the beam axis coincides with the plate or concentric cylinders axis center. The diameter of the cylinders is subject to the same dimensional constraints as the electrode separation d in parallel-plate chambers, namely electron-loss considerations.

The quantity is obtained by the direct measurement for the plane-parallel and that for the concentric variable cylinder is determined differentially by measuring the ionization current charge in open and closed cylinders.

The cylinders are equipped with precision movement such that the distance from the fixed diaphragm to the center of the collecting region remains constant. The charge that is collected for the collapsed position arises from photon interactions in regions A and B in figure I.3. The charge that is collected for the

expanded position by the increased volume V creates regions A' , B' and V . The photon interactions in the A' region will be greater than those for the collapsed measurement in region A because it is closer to the diaphragm. However, this difference will be compensated for by a good approximation and by reduced photon interactions in the B' region. The secondary electrons that are generated in the V region will stop within the chamber and do not depend on the homogeneity of the electric field.

The main advantage of the concentric cylindrical chamber is that the difference in the measurement by the first cylinder expanded and by the second one collapsed. When the first cylinder collapsed and the second expanded, we could determine the air mass effect directly related to the cylinder movement, figure I.4 shows into the Victoreen model 480 [4] implemented as a primary standard dosimetry for medium X Ray energy.

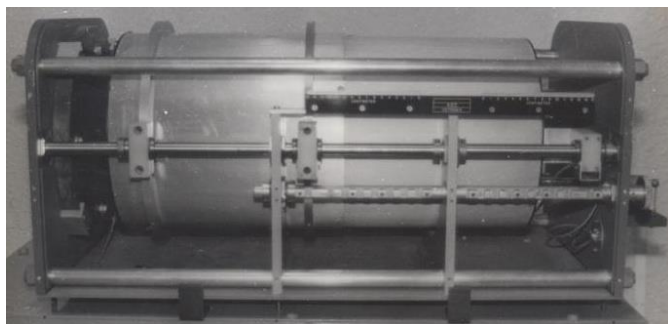


Figure I.4: Victoreen Free air ionization chamber, model 480, implemented as a primary standard dosimetry for medium X ray energy [4].

An expanded perspective of the air ionization chamber variable volume pieces is given by Victoreen model 481, figure I.5. In detail, these pieces are the "diaphragm" (1), the "large cylinder" (2) and the "lower cylinder" (3) and are fundamental in obtaining the interest volume and the air

attenuation correction factor, depending on the cylinder positioning, figure I.6 shows into the Victoreen model 481, implemented as a primary standard dosimetry for low X ray energy [8].

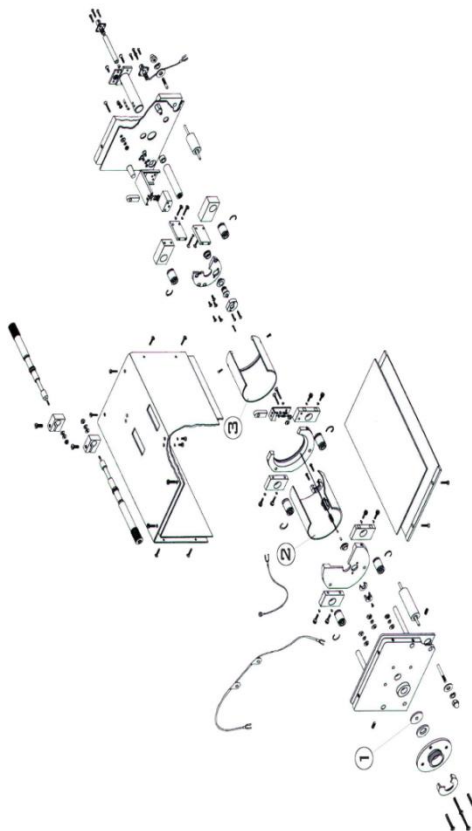


Figure I.5: Expanded perspective of the air ionization chamber variable volume pieces set in Victoreen model 481. In detail, these pieces are the "diaphragm" (1), the "large cylinder" (2) and the "lower cylinder" (3) and are fundamental parts in obtaining the interest volume and the air attenuation correction factor, depending on the cylinder positioning [8].



Figure I.6: Victoreen Free air ionization chamber, model 481[8].

The concentric cylinder free air ionization chamber has been implemented as the primary standard for medium-energy X rays in Italy [9] and in Taiwan [10] and for mammography X rays in the USA [8, 9] and Brazil [5, 10, 11].

1.3 traceable dosimeter

We found many geometry varieties of the cavity ionization chamber, but all basically consist of a solid envelope, such as spherical, plane parallel and cylindrical, surrounding a gas-filled cavity, as seen at table I.1, and an electric field is established to collect the ions that are produced by radiation. If the sensitive gas enters the atmosphere, it is considered the vented type [12–14].

We could determine this type by the chamber shape in each area; in particular, the spherical designs were used more in the isotropic irradiations. Conventionally such ‘thimble’ chambers, as they are sometimes called, are irradiated at mono-directional beams and at therapy beams.

The chambers can be designed as very compact or larger as needed for applications for therapy, diagnosis or protection,

including whether mono- or multi-directional radiation fields are used.

Observing the specific application of the ionization chambers, low to high energy beams and different dose measurements for photons or electrons ensure that the solid wall material is necessary for the range of the secondary electrons.

Ionization chambers are technical devices that are defined specifically with metrological characteristics applied to make ionization radiation measurements. Applications in X ray diagnostics include fluoroscopy, interventional radiology, mammography, CT and dental. Applications in X and Gamma rays include protection and therapy.

The ionizing standard chambers are calibrated in order to ensure traceability of their measurement, known as secondary standards, which have advantages in working with dosimeters with calibrations that are traceable in the laboratory. Thus, before ordering the calibration of the secondary standard, it should be established that the Primary Standard Dosimetry Laboratory – PSDL – that should be employed for the task is capable of providing traceability, i.e., in the energy region considered, the free-air chamber or the calorimeter (water or graphite) could be a primary standard for realizing the air kerma or absorbed dose (water or graphite) quantities. When the ion-collecting gas volume is precisely known, the chamber is an absolute dosimeter [2].

Now, the Secondary Standard Dosimetry Laboratory – SSDL – is traceable to stated references. This traceability chain should make it possible to trace the calibration results back to PSDL, which is acceptable for the customer. This secondary standard must reflect the traceability of the standard to the workshop level or user.

1.4 Secondary standard dosimeters and their characterization

An SSDL must have at least one secondary standard dosimeter that has been calibrated at the BIPM or at a PSDL. This dosimeter should conform to the specifications that are given in IEC 60731 [18] for reference class instruments. For each category of measurements, the SSDL should have two reference class ionization chambers.

Each reference chamber (or set of reference chambers) should provide a useful operating range of radiation qualities applicable to all qualities of sets that have been approved for that category. Some specific characteristics for diagnostic radiology instruments performance are recommended by IEC 61674 [19]. It is recommended that the secondary standard be recalibrated at intervals of approximately three years, although this period can depend on its demonstrated long-term stability and might therefore differ between instruments [20–22].

The secondary standard can be used either directly for the routine calibration of user instruments or periodically to calibrate one or more working standard instruments or to determine the air kerma or absorbed dose rate for subsequent use in calibration. The overall calibration uncertainty that is attributed to the user instrument might be slightly less when calibrated against the secondary standard rather than a working standard, but the difference should be small and must be balanced against the increased damage risk of a calibration coefficient change of the secondary standard if used regularly.

The SSDL dosimetry depends on the secondary standard stability is preserved with the maximum care and is stored in a safe place under stable environmental conditions that minimize the possibility of calibration coefficient change.

The secondary standard dosimeter ionization chamber must have a high degree of long-term stability and low energy dependence and must be vented and have sealed chambers that are generally less stable in the long term. For the measurement of air kerma, suitable buildup caps might be necessary for Co⁶⁰ and Cs¹³⁷ sources. If the water phantom chamber is used, a waterproof sleeve must be available unless the ionization chamber is designed to be inserted directly into water.

The measuring assembly, usually an electrometer, measures the charge or current from the ionization chamber and often also provides the polarizing potential. The measuring assembly can either be calibrated together with the ionization chamber, or the recommended methods are calibrated separately. In the latter case, the measuring assembly calibration in terms of electric current or charge must be traceable to electrical standards. Special high insulation coaxial cables are necessary to connect the ionization chamber to the measuring assembly [23–25].

Electrometers measure currents equal to or less than 10^{-9} A with a high gain, negative feedback, and operational amplifier with resistor or capacitor standard in the feedback path to measure the current or charge that is collected over a fixed time interval, as shown schematically in figure I.7 [26].

1.4.a Ionization chamber properties

Ionization chambers come in various shapes and sizes depending on the specific requirements, but they generally all have the following properties [26]:

These chambers are basically gas-filled cavities surrounded by a conductive outer wall and with a central collecting electrode, as shown in figure I.8. The wall and the collecting electrode are

separated with a high-quality insulator to reduce the leakage current when a polarizing voltage is applied to the chamber.

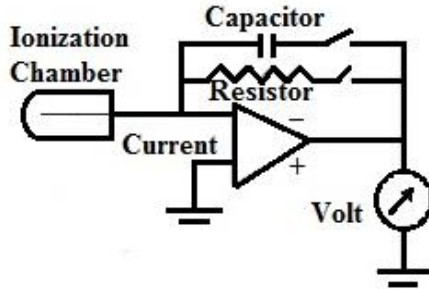


Figure I.7. Electrometer in feedback mode of operation. Using *rate mode* is $V = R.I$, and using *integrate mode* is $V = (I.t)C$, where V is in voltage, R is in ohm, I is in ampère, t is in second and C is in farad units.

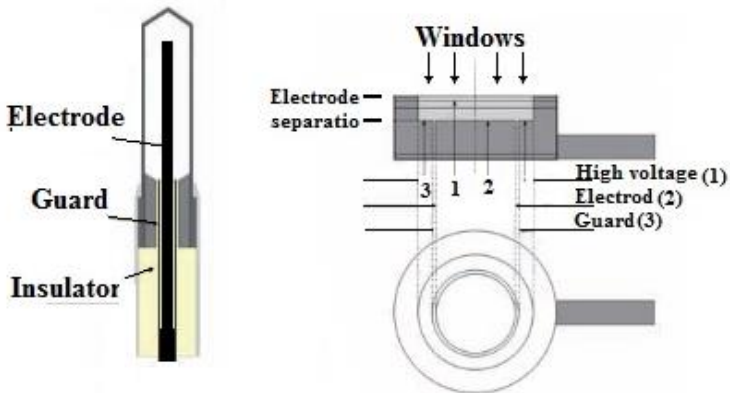


Figure I.8. Schematic diagram of cylindrical (left) and plane parallel (right) ionization chambers. That could see the electrode, guard, insulator and electrode separation of the air cavity.

A guard electrode is usually provided in the chamber to further reduce chamber leakage. The guard electrode intercepts

the leakage current and allows it to flow to the ground, by passing the collecting electrode. This guard also ensures improved field uniformity in the active or sensitive volume of the chamber, with resulting advantages in charge collection.

Measurements with open-air ionization chambers require air density correction to account for the change in the mass of air in the chamber volume, which changes with the ambient temperature and pressure. The ionization current was standardized using 20 °C and 101.325 kPa as reference conditions. Using the ideal gas law for the air density correction inside the chamber, volume sensitive measurement and the relative humidity were maintained between 40 and 70 %, as shown in equation I.1.

$$k_{TP} = \frac{(273.15 + T)}{(273.15 + 20)} \cdot \frac{101.325}{P} \quad \text{I.1}$$

I.4.b Ionization chambers (shape and volume)

Ionization chambers are technical devices that are defined specifically with metrological characteristics applied for ionization radiation measurements. Protection, therapy and X ray diagnostic applications include fluoroscopy, interventional radiology, mammography, CT and dental uses.

An ionizing radiation chamber device should be connected to the measurement system, where the ionizing chamber serves as the sensor, with a power supply and an electrometer or a lower device counter, and at the end, should display a device that also connects to the air density measurement device.

The ionizing radiation measuring system has different measuring instrument data processors and auxiliary devices, which are assembled in the laboratory with environmental

controlled conditions and linked together to carry out specified measurements.

The Primary Standard Dosimetry Laboratory – PSDL – is capable of providing traceability, i.e., in the energy region considered, the free-air chamber or the calorimeter (water or graphite) could be a primary standard for determining the air kerma or absorbed dose (water or graphite) quantities. The ionizing standard chambers are calibrated in order to ensure the traceability of their measurements, known as secondary standards.

Now, the Secondary Standard Dosimetry Laboratory – SSDL – is traceable to stated references. This traceability chain should make possible tracing the calibration results back to PSDL.

The more common ionization chambers that are used for these purposes should be of the vented type, i.e., their sensitive gas volume should communicate with the atmosphere, independent of their shape or volume design.

The codes of practice TRS 398 [27] and TRS 457 [28] and standards IEC61267 [29] and ISO4037 [18–21] generally follow ICRU 74 [34] on patient dosimetry for X and Gamma rays used for medical or protection purposes. The dosimetry quantities are divided into basic and application-specific quantities. Basic quantities are fundamental quantities as defined in ICRP 60 [23, 24, 35 and 36].

I.4.b.i Cylindrical or spherical ionization chamber

The response of cylindrical or spherical chambers is very symmetrical with respect to the axis. These chambers are usually oriented with the cylindrical or spherical axis of the chamber perpendicular to the X or gamma ray beam and are measured from all directions; back scattering is included [37].

I.4.b.ii Plane parallel ionization chamber.

Plane parallel ionization chambers use two parallel, flat electrodes that are separated by a few millimeters. These chambers are calibrated with their plate oriented perpendicular to the beam axis, which is also the orientation in which they should be used. Some of these chambers have different windows for entrance and exit, in which case it is important that the entrance window faces the beam focal point and that they measure only from one direction.

Different ionization chamber shapes for X and Gamma ray applications for measuring air kerma or absorbed dose (water or graphite) quantities are found in the market.

1.5 Calibration method

The measurement conditions cannot be kept perfectly constant, as by air density imperfections or same cavity effect and electric field, causing random changes in the indications that are obtained by an ionizing measuring system. Thus, it is necessary to repeat the measurement an appropriate number of times under repeatable conditions in order to minimize the random influences, and a calibration method will be more accurate for the capability of a measuring instrument. Independent of the calibration method that is used, the repeatability is shown by the arithmetic mean and standard deviation of the stability result indications.

1.5.a Substitute method

For the reference point at this calibration, each chamber is placed successively at the measurement point. Note that the reference point of a cylindrical or thimble ionization chamber is located on the chamber axis at a distance from the tip either as stated by the manufacturer or as indicated on the instrument.

For a plane-parallel chamber, the reference point is normally taken to be at the center of the inner surface of the front window (for the thin-window chambers that are used for low-energy X rays, the outer surface is taken) [38]. X ray calibration by substitution normally requires extra control equipment, such as an X ray tube current or monitor chamber.

1.5.b Tip-to-tip method

The two ionization chambers are placed coaxially with the ends of the chambers close to each other and irradiated simultaneously. If either sensitive volume has a length much greater than its diameter or if measurements are being carried out in a phantom, it might be better to place the chambers side by side with the chamber axes parallel (still referred to as tip-to-tip calibration).

In both cases, the reference points of the two chambers should be positioned symmetrically with respect to the beam axis and at the same distance from the radiation source [38].

Conventional X ray tubes usually have reflection targets (in contrast to the transmission targets that are used with accelerators, for example). As a result, there can be significant variation in the output rate and photon energy along the cross section of the beam parallel to the anode–cathode direction of the X ray tube (the heel effect). For tip-to-tip calibration in X rays, therefore, the reference points of the two chambers should be positioned on a line that is perpendicular to the anode–cathode direction.

To compensate for any residual radial non uniformity of the beam, the measurement should be repeated with the chambers interchanged in position, and if time allows, the positions should be interchanged several times. The mean of the calibration coefficients that are obtained with the chamber in the two positions should be used as the best estimate [38].

In tip-to-tip calibration, each chamber receives scattered radiation from the other. The error that is introduced by this effect is minimized when the two chambers are similar in design. Tip-to-tip calibration might be considered the method of choice in X ray beams if there is no monitor chamber or if it has become unreliable [38].

1.5.c Known radiation field or Dosimetry method

For the dosimetry reference point at this calibration, the reference chamber is placed at the measurement point for all of the dosimetry measurements that are needed for the radiation conditions set.

The chambers or dosimeter to be calibrated should be stated at the same point and measured. The time measurement should be used. Note that the reference point of a cylindrical or thimble ionization chamber is located on the chamber axis at a distance from the tip either as stated by the manufacturer or as indicated on the instrument.

For a plane-parallel chamber, the reference point is normally taken to be at the center of the inner surface of the front window (for the thin-window chambers that are used for low-energy X rays, the outer surface is taken).

1.6 Calibration Results

From the definition of accuracy of a measuring instrument, it follows that the quantitative expression for the measuring the instrumental capability to provide reliable indications would be the difference between its indication and the true value of the corresponding input quantity.

This difference is called the error (of indication) of a measuring instrument. However, upon recognizing the fact that a

true value cannot be determined, a conventional true value is used instead. In most cases, the reference value provides a suitable measurement standard. The error of a measuring instrument is estimated through instrument calibration.

Therefore, the measurement conditions cannot be kept perfectly constant, as by air density imperfections or same cavity effect, causing random changes in the indications that are obtained by an ionizing measuring system. Thus, it is necessary to repeat the measurement an appropriate number of times under repeatable conditions in order to minimize the random influences.

At the end of this repeat, we use the arithmetic mean and standard deviation of the indications as the result and type A uncertainty, respectively. The difference between the value thus obtained and the measured yields a reference value of a systematic error estimative of the indication of a measuring instrument. In addition to random variation in the charge or current, this result depends to some extent on the measurement conditions, and the uncertainty should be determined [39].

The accredited National Metrology Institute – NMI – or SSDL provides result traceability to measure the standard following the ISO/IEC 17025 [40], ensuring appropriateness and relevance. Once the air kerma or absorbed dose of a beam is established, the reference class ionization chamber is calibrated using the substitution method. If the customers do not find a suitable accredited SSDL, the employed laboratory may be able to provide traceability to the ionizing chamber, but the customer should ensure that the calibration is carried out using an appropriately documented calibration method.

Each calibration carried out by the SSDL must be reported accurately, clearly and objectively on a calibration certificate. The most important information on a calibration certificate is a

list of calibration coefficients with their uncertainties, which must be clearly indicated and determined using the ISO and IAEA recommendations; however, additional information is necessary for the correct interpretation and subsequent use of the calibration results. The information contained in a calibration certificate is specified in the international standard ISO/IEC 17025 [40]. The following list of items is an interpretation of these general requirements for the calibration of dosimeters:

- (a) A title (e.g., Calibration Certificate).
 - (b) Name and address of the calibrating laboratory.
 - (c) A unique certificate number, printed on every page.
 - (d) Date of issue of the certificate.
 - (e) Page number on every page, in the form “Page x of y”.
 - (f) Name and address of the user.
 - (g) Unique identification of the instrument(s).
 - (h) Date of calibration measurements and staff performing the calibration.
 - (i) Results of the calibration (preferably in tabular form):
 - Beam quality specified (HVL, gamma ray source);
 - Calibration coefficients, stating quantity and unit;
 - Uncertainty of measurement and coverage factor.
 - (j) Reference conditions.
 - (k) Calibration conditions.
 - (l) Instrument operation.
 - (n) Results of additional measurements.
 - (o) Information about the beams.
- Miscellaneous information:
- (i) Calibration traceability.
 - (ii) Name, position and signature of the responsible person.

Chapter II

X and gamma ray secondary standard metrology

José Guilherme Pereira Peixoto
Maria da Penha Potiens

II.1 Introduction

The Secondary Standards Dosimetry Laboratory (SSDL) role [15, 16, 25, 28–30] is crucial in providing traceability, disseminating calibrations at specific radiation qualities, and appropriately using radiation measuring instruments. An SSDL may be either national or regional. A national SSDL is a laboratory that has been designated by competent national authorities to undertake the duties of necessary radiation dosimetry traceability to national/international standards for country users. A regional SSDL is designated by an intergovernmental agreement or by an international organization not only to carry out national functions but also to provide calibration services and advice to other countries within the concerned geographical area [44].

An SSDL is equipped with secondary standards that are traceable to the primary standard dosimetry laboratories participating in the international measurement system, Primary Standard Dosimetry Laboratories (PSDL) and the Bureau International des Poids et Mesures (BIPM). Figure II.1 illustrates the global metrological links of the international measurement system (SI – Système International) for radiation dosimetry [26].

In 1976, the International Atomic Energy Agency (IAEA) and the World Health Organization (WHO) established an SSDL

network called the "IAEA/WHO Network of Secondary Standard Dosimetry Laboratories". The objective of this SSDL network was to improve accuracy in applied radiation dosimetry throughout the world; it is an association of SSDLs that agrees to cooperate to promote the objectives of that network under international auspices [42].

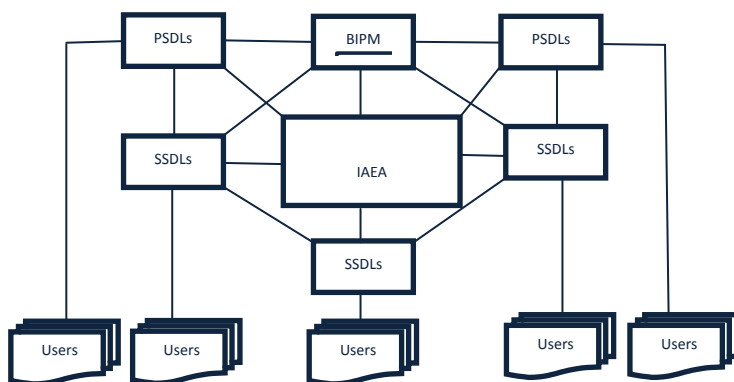


Figure II.1. Global metrological links of the international measurement system [26].

Historically, although the first SSDLs provided mainly radiation therapy-level calibrations, the scope of their work has expanded over the years [45]. Today, many SSDLs provide traceability for amplified range measurements, applied in radiation protection and diagnostic radiology in addition to radiotherapy following the IAEA recommendations and code of practices [15, 16, 27, 30].

The requirements for traceable and reliable calibrations performed at SSDLs are becoming increasingly important, and the demonstration of their competence can be achieved through comparisons and the establishment of a quality system following the International Organization for Standardization (ISO) standard [40]. One important requirement of the quality system is the

assessment of the measurement uncertainty and a general guidance on the uncertainty estimation published by ISO [33–36], based on which the IAEA prepared a practical guide for SSDLs on how to assess and report measurement uncertainties [49].

II.2 SSDL Responsibilities

The SSDL responsibilities include but are not limited to the following activities [50]:

- Maintaining secondary standard instruments in agreement with the international measurement system and performing recalibrations at least every 3 years.
- Performing calibrations of radiation measurement equipment and issuing calibration certificates with all of the necessary information, including the estimated uncertainties.
- Organizing dose comparisons for institutions within the country or region and participating in measurement comparisons within the IAEA/WHO SSDL network and with other standardizing laboratories.
- Cooperating with the IAEA/WHO network and with other metrological laboratories in the exchange of information and improvement of measurement instruments and techniques.
- Documenting and preserving records of all of the calibration procedures and results.
- Keeping up to date on progress in radiation measurement to improve calibration techniques as required, thereby providing a better service to the users of radiation.
- Providing training in radiation measurement, calibration techniques and relevant instrumentation use and maintenance as appropriate to the users of radiation as served by the SSDL.

- Secretariat reporting, at least annually, on the secondary standards status, radiation sources, calibrations performed and related activities.

II.3 Determination of a calibration coefficient: The model equation

Three methods can be used to calibrate instruments in a radiation field: tip-to-tip, substitution or calibration method in a known radiation field [51].

Using the tip-to-tip method, the reference dosimeter and the dosimeter to be calibrated are placed in the radiation beam and irradiated simultaneously.

In calibration by substitution, first, the reference dosimeter is placed at the calibration point to determine the reference output rate of the beam through a set of readings. It is then replaced by the dosimeter to be calibrated, and a similar set of readings is taken.

To perform a calibration in a known radiation field, it is assumed that the basic radiation quantity characterizing the field is already known and that no reference instrument is needed at the time of the irradiation of the dosimeter to be calibrated.

Each method has advantages and disadvantages, and an SSDL may select one or another procedure. Most SSDLs use the substitution method [41]. The calibration coefficient can be easily derived from the substitution method and can be determined in two steps [49]:

Step 1: The radiation beam output rate¹, $\dot{K}_{a,Q}$, of quality Q is determined with the SSDL reference standard, traceable at a PSDL:

$$\dot{K}_{a,Q} = N_{K,Q_0}^{ref} M_{corr}^{ref} k_{Q,Q_0} \quad \text{II.1}$$

where N_{K,Q_0}^{ref} is the SSDL reference standard calibration coefficient for the beam quality Q_0 ; M_{corr}^{ref} is the reference dosimeter reading corrected for the quantities influence; and k_{Q,Q_0} is the factor to account for the difference in beam qualities of the PSDL and the SSDL [27].

Step 2: The instrument to be calibrated is placed at the same position as the SSDL reference standard in the beam of quality Q. The calibration coefficient N_Q^{user} for the beam quality Q of the instrument to be calibrated is determined as the ratio of the output rate $\dot{K}_{a,Q}$, determined in step 1, to the mean reading that is obtained from the instrument to be calibrated and corrected for the influence quantities.

$$N_Q^{user} = \frac{\dot{K}_{a,Q}}{M_{corr}^{user}} k_{source} \quad \text{II.2}$$

where k_{source} is the correction for the effect of a change in the source position, and M_{corr}^{user} is the reading that is obtained with the user instrument, already corrected for influence quantities.

II.4 Dosimetry Protocols: Codes of Practice

One of the principal goals of the SSDL network in the radiation dosimetry field is to guarantee that the dose that is delivered to patients and/or received by individuals undergoing radiation fields within internationally accepted levels of accuracy. This regulation is accomplished by ensuring that the calibrations of

¹The model equation is also valid for integral kerma

instruments that are provided by the SSDLs are correct, emphasizing the participation of the SSDLs in quality assurance programs, promoting the contribution to support dosimetry quality audits and assisting if needed in performing the calibration of equipment in hospitals.

II.4.a Radiotherapy

The Code of Practice TRS 398 [27] determines the absorbed dose for the water methodology in low-, medium- and high-energy photon beams, electron beams, proton beams and heavy-ion beams used for external radiotherapy.

The determination of absorbed dose to water formalism in high-energy photon and electron beams uses an ionization chamber or a dosimeter calibrated in terms of absorbed dose to water in a Co⁶⁰ source.

It is assumed that the absorbed dose to water, D_w , is known at a depth of 5 g/cm² in a water phantom for Co⁶⁰ gamma rays. This determination is realized at the SSDL by means of a calibrated cavity ionization chamber performing measurements in a water phantom. The user chamber is placed with its reference point at the same depth, and its calibration factor $N_{D,w}$ is obtained from

$$N_{D,w} = \frac{D_w}{M} \quad \text{II.3}$$

where M is the dosimeter reading corrected for influence quantities; recommended reference conditions for the ionization chamber calibration in Co⁶⁰ are given in table II.1.

Medium- or low-energy X rays measuring the chamber must be calibrated in similar-quality beams, but only a few PSDLs have primary standards of absorbed dose to water for kilovoltage X ray qualities [38, 39]. However, it is possible to derive calibration factors in terms of absorbed dose to water from air

kerma calibration factors using one of the accepted protocols or Codes of Practice for the dosimetry of X ray beams.

Table II.1: Reference conditions for ionization chamber calibration in ^{60}Co gamma radiation for absorbed dose in low- and medium-energy X ray beams in standard laboratories.

	Reference value or reference characteristics		
	Gamma	X Rays	
Influence quantity	^{60}Co	Low	Medium
Phantom material	Water	PMMA ^c	Water
Phantom size (cm ³)	30×30×30	12×12×6	30×30×30
Source-chamber distance (SCD)	100 cm	Specified by user	Specified by user
Air temperature		20°C	
Air pressure		101.3 kPa	
Relative humidity		50%	
Reference point of the ionization chamber	cylindrical ^A	plane-parallel ^B	cylindrical ^A
Depth in phantom ^D	5 g/cm ²	Surface	2 g/cm ²
Field size at the position ^D		3×3 cm ²	10×10 cm ²

^A on the central axis at the center of the cavity volume

^B on the central axis at the outside of the entrance window

^C water equivalent plastic

^D of the reference point of the chamber

Typical reference conditions for the ionization chambers calibration in kilovoltage X ray beams are given in table II.1. The reference radiation qualities are those that are recommended by BIPM and their main characteristics are presented in tables II.2 and II.3 [54].

The radiation conditions or dosimetry quantities that were used for X and Gamma ray applied for Protection, Therapy and

Diagnosis are shown in tables II.2 to II.16. In the past, various radiation conditions have been used for the specification of the dose in IR, and there has been ambiguity because the same name has been used for different radiation conditions, expressed by the tube voltage, added filtration, half value layer (HVL) and homogeneity coefficient.

Table II.2: Low-energy X ray qualities recommended by BIPM [54].

Tube Voltage(kV)	Added Filtration(mm Al)	Half Value Layer(mm Al)	Air Kerma rate(mGy/s)
10	--	0.037	1.00
25	0.208	0.169	1.00
30	0.372	0.242	1.00
50(a)	3.989	2.262	1.00
50(b)	1.008	1.017	1.00

Table II.3: Medium-energy X ray qualities recommended by BIPM [54].

Tube Voltage kV	Added Filtration (mm)		HVL ¹ (mm)		Air Kerma rate mGy/s
	Al	Cu	Al	Cu	
100	3.431	--	4.030	0.149	0.50
135	2.228	0.232	--	0.489	0.50
180	2.228	0.485	--	0.977	0.50
250	2.228	1.570	--	2.484	0.50

¹Half Value Layer

II.4.b Diagnostic Radiology

Various examination techniques are used in X ray diagnostic radiology and include fluoroscopy, interventional radiological procedures, mammography, Computed Tomography (CT), dental and general radiography. X ray beams with tube voltages from 20 to 150 kV are used.

Table II.4: Radiation qualities for calibrations of diagnostic radiology dosimeters

Radiation Quality	Radiation beam	Material of an additional filter	Application
RQR	X ray assembly	No phantom	General radiography and dental applications ^A
RQA	added filter	Aluminum	Measurements behind the patient ^B
RQT	added filter	Cooper	CT applications
RQR-M	X ray assembly	No phantom	Mammography applications
RQA-M	added filter	Aluminum	Mammography studies
W or Rh Anode	added filter	Mo, Rh, Pd and Al	Mammography studies

^Afree in air

^Bon the image intensifier

The tube voltages in fluoroscopy, CT, dental and general radiography range from 50 to 150 kV; the anode material is usually tungsten. Mammography examinations are conducted with tube voltages between 22 and 40 kV, and various combinations of anode and filtration materials are used; the most common materials are molybdenum anode and molybdenum

filtration, but for calibration, we could use also tungsten, molybdenum and Rhodium anode, combined with Aluminum, Molybdenum, Palladium, Rhodium and Silver filtration [41,42,56–58]; see table II.4.

In diagnostic radiology, the specification of radiation qualities is important because the response of all dosimeters depends, at least to a certain extent, on the spectral distribution of the X rays employed. Radiation qualities are usually specified in terms of the X ray tube voltage first HVL and homogeneity coefficient [60].

The dosimetry formalism based on air kerma determination is given in detail by the TRS 457 [28], and the recommended radiation qualities are described by the IEC 61267 [29], as presented in Tables II.5 to 8. This Code of Practice generally follows ICRU 74 [34] on patient dosimetry for X rays that are used in medical imaging.

Table II.5: Characterization of non-attenuate radiation quality series RQR; the number 5 is the reference radiation quality [16 and 17].

Radiation Quality	X ray tube voltage(kV)	First HVL (mm Al)	Homogeneity coefficient
RQR2	40	1.42	0.81
RQR3	50	1.78	0.76
RQR4	60	2.19	0.74
RQR5*	70	2.58	0.71
RQR6	80	3.01	0.69
RQR7	90	3.48	0.68
RQR8	100	3.97	0.68
RQR9	120	5.00	0.68
RQR10	150	6.57	0.72

Table II.6: Characterization of attenuate radiation quality series RQA; number 5 is the reference radiation quality [16 and 17].

Radiation Quality	X ray tube voltage(kV)	Added Filtration(mm Al)	Nominal first HVL(mm Al)
RQA2	40	4	2.2
RQA3	50	10	3.8
RQA4	60	16	5.4
RQA5*	70	21	6.8
RQA6	80	26	8.2
RQA7	90	30	9.2
RQA8	100	34	10.1
RQA9	120	40	11.6
RQA10	150	45	13.3

Table II.7: Characterization of mammography radiation quality series RQR-M and RQA-M; number M2 is the reference radiation quality [16,17 and61].

Radiation Quality	X ray tube voltage kV	Added Filtration mm	Nominal first HVL mm Al
RQR-M1	25		0.28
RQR-M2*	28	0,03 Mo	0.31
RQR-M3	30		0.33
RQR-M4	35		0.36
RQA-M1	25		0.56
RQA-M2	28	0,03 Mo+2 Al	0.60
RQA-M3	30		0.62
RQA-M4	35		0.68

Table II.8: Characterization of Computed Tomography radiation quality series RQT; number 9 is the reference radiation quality [16 and 17].

Radiation Quality	X ray tube voltage(kV)	Added Filtration(mm Cu)	Nominal first HVL(mm Al)
RQT8	100	0.2	6.9
RQT9*	120	0.25	8.4
RQT10	150	0.3	10.1

II.4.c Radiation Protection

Occupational radiation protection is a major component of the support for radiation safety provided by the IAEA Member States. The International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (BSS) presents the requirements for occupational radioprotection [43 and 44].

Occupational exposure to ionizing radiation can occur in industry, medical institutions, research establishments, universities and nuclear fuel cycle facilities. The IAEA Technical Safety Report 16 [43] provides guidance on the establishment and operation of calibration facilities for radiation monitoring instruments based on the operational quantities. The recommended radiation qualities were established by the International Organization for Standardization [18–21] and are provided in Tables II.9 to 13.

Table II.9: Radionuclide sources that are used for the production of gamma radiation.

Radionuclide	Energy (MeV)	Half-life (days)	Air kerma rate ^A ($\mu\text{Gy}\cdot\text{h}^{-1}\cdot\text{m}^2\cdot\text{MBq}^{-1}$)
⁶⁰ Co	1.1733 1.3325	1924	0.31
¹³⁷ Cs	0.6616	10976	0.079
²⁴¹ Am	0.05954	84753	0.003

^A The air-kerma rate constant is valid only in the case of an unshielded point source. It is therefore given only as a guide and not as a means of determining the air-kerma rates.

Table II.10: Characteristics of low air kerma rate series

Tube Potential (kV)	Mean energy (keV)	Additional Filtration (mm)				First HVL (mm)
		Pb	Sn	Cu	Al	
10	8.5				0.3	0.058 Al
20	17				2.0	0.42 Al
30	26			0.18	4.0	1.46 Al
35	30			0.25		2.20 Al
55	48			1.2		0.25 Cu
70	60			2.5		0.49 Cu
100	87		2.0	0.5		1.24 Cu
125	109		4.0	1.0		2.04 Cu
170	149	1.5	3.0	1.0		3.47 Cu
210	185	3.5	2.0	0.5		4.54 Cu
240	211	5.5	2.0	0.5		5.26 Cu

Table II.11: Characteristics of the narrow spectrum series.

Tube Potential (kV)	Mean energy (keV)	Additional Filtration (mm)				First HVL (mm)	Second HVL (mm)
		Pb	Sn	Cu	Al		
10	8				0.1	0.047Al	0.052 Al
15	12				0.5	0.14 Al	0.16 Al
20	16				1.0	0.32 Al	0.37 Al
25	20				2.0	0.66 Al	0.73 Al
30	24				4.0	1.15 Al	1.30 Al
40	33			0.21		0.84 Cu	0.091Cu
60	48			0.6		0.24 Cu	0.26 Cu
80	65			2.0		0.58 Cu	0.62 Cu
100	83			5.0		1.11 Cu	1.17 Cu
120	100		1.0	5.0		1.71 Cu	1.77 Cu
150	118		2.5			2.36 Cu	2.47 Cu
200	164	1.0	3.0	2.0		3.99 Cu	4.05 Cu
250	208	3.0	2.0			5.19 Cu	5.23 Cu
300	250	5.0	3.0			6.12 Cu	6.15 Cu

Table II.12: Characteristics of the wide spectrum series.

Tube Potential (kV)	Mean energy (keV)	Additional Filtration (mm)		First HVL mm Cu	Second HVL mm Cu
		Tin	Copper		
60	45		0.3	0.18	0.21
80	57		0.5	0.35	0.44
110	79		2.0	0.96	1.11
150	104	1.0		1.86	2.10
200	137	2.0		3.08	3.31
250	173	4.0		4.22	4.40
300	208	6.5		5.20	5.34

Table II.13: Characteristics of the high air kerma rate series

Tube Potential (kV)	Mean Energy (keV)	Additional filtration			HVL (mm)			
		(mm)			First		Second	
		Al	Cu	Air	Al	Cu	Al	Cu
10	7.5			750	0.036	0.010	0.041	0.011
20	12.9	0.15		750	0.12	0.007	0.16	0.009
30	19.7	0.52		750	0.38	0.013	0.60	0.018
60	37.3	3.2		750	2.42	0.079	3.25	0.11
100	57.4	3.9	0.15	750	6.56	0.30	8.05	0.47
200	102		1.15	2250	14.7	1.70	15.5	2.40
250	122		1.6	2250	16.6	2.47	17.3	3.29
280	146		3.0	2250	18.6	3.37	19.0	3.99
300	147		2.5	2250	18.7	3.40	19.2	4.15

Chapter III

Traceability and Uncertainty

Malcolm McEwen

III.1 Introduction

It is often the case that “traceability” and “uncertainty” are presented in the literature as completely separate subjects. However, it is better to see them as two essential parts of the measurement itself. Without a traceability chain and an uncertainty analysis the determination of some specific quantity has no meaning beyond the local environment. In this case there can be no *meaningful comparison* with any other measurement of the same quantity at a different institution and without an uncertainty estimate there can be no *meaningful interpretation* of that comparison.

A complementary way of looking at these concepts is that they are necessary for the implementation of a quality assurance program, by which, and through a measurement, it is demonstrated that a particular device, piece of equipment, or procedure is “fit for purpose”. Traceability and uncertainty are not only essential for demonstrating the quality of the measurement obtained with an instrument, but the development of the traceability chain and uncertainty budget involves a process review that feeds directly into the quality documentation and procedures.

Given the page limitations, the aim of this chapter is not to provide a complete answer to the question of traceability and

uncertainty but to provide a complementary approach to that already to be found in the open literature.

III.2 Traceability

“Traceability” is the concept that describes the chain that links a measurement of a quantity in the field to a standard. Traceability is obtained by the process of “calibration” and the entire path between the end user and a standard is called the calibration chain. The term “standard” has two distinct meanings in the fields of metrology and quality assurance but both denote a form of standardization. To be completely clear, one should talk about a “measurement standard” or a “documentary standard” but often the qualifying term is dropped in many discussions on this subject. Ultimately, both indicate something (an artifact or procedure) against which others of a similar type are judged or compared.

III.2.a Measurement standard

The starting point for a calibration chain is an artifact or measurement that is usually referred to as a primary national standard. These are generally maintained and disseminated by National Measurement Institutes - NMIs, which in the dosimetry field are also referred to as Primary Standard Dosimetry Laboratories - PSDLs. A PSDL is designated by a national government for the purpose of developing, maintaining, and improving primary standards in radiation dosimetry. These standards are then disseminated through the calibration of secondary standards. The standards typically maintained at the PSDLs cover a range of end-use applications: radiation protection, diagnostic radiology, radiation therapy and industrial processing, for a range of beam modalities: kV X-rays, ¹³⁷Cs and

^{60}Co irradiators, radioactive sources for brachytherapy, linear accelerator beams (both photons and electrons). The measurement quantities include air kerma, air-kerma strength, activity, absorbed-dose and absorbed dose equivalent, depending on the modality and application.

III.2.b Role of the Bureau International des Poids et Mesures - BIPM

Although for a single country, the calibration chain starts (or stops) at the PSDL, the need for international equivalence means that there is an ongoing system of comparison of primary standards from the different NMIs around the world. This is accomplished through the Bureau International des Poids et Mesures - BIPM. The BIPM is an international laboratory created by the Meter Convention of 1875. The Ionizing Radiation Section of the BIPM, set up in 1960, develops and maintains primary standards for air kerma and absorbed dose used for radiotherapy and dose equivalent for radiation protection, in addition to maintaining radioactivity standards. BIPM also serves a second, important role – in the case where a NMI does not maintain a PSDL, or where a PSDL does not maintain primary standards for all the necessary radiation quantities, the BIPM's standards are disseminated by calibration of secondary standards maintained at the NMI.

Through the coordination by the BIPM of international comparisons and the dissemination of standards, the NMIs are able to declare their calibration and measurement capabilities - CMCs relative to each other and publish these in the BIPM key comparison database (KCDB). For an updated status of key comparison results of PSDLs see the web site: <http://kcdb.bipm.org/>.

This system of open access to key-comparisons leads to the concept of mutual recognition (MR) of measurement capabilities, currently between 150 metrology institutes, in accordance with the mutual recognition arrangement - MRA drawn up by the Comité International des Poids et Mesures - CIPM in 1999 and known as the CIPM - MRA. This MRA has enabled the equivalence of national standards dosimetry laboratories to be established, which essentially allows an institution to have its instrument calibrated at any laboratory that is part of the MRA.

III.2.c Secondary Standards Laboratories

It may be the case that a PSDL disseminates its standard directly to the end-user through a single calibration step. This is the case in Canada, where the population and a variety of industry sectors makes direct calibration at the PSDL. In other situations there may be a layer (or layers) of secondary standards laboratories that provide a much larger calibration capability to meet end-user needs. This is the case in the USA and also in Germany.

A special situation that requires further discussion is where a country does not have a PSDL. In this case there may be a Secondary Standard Dosimetry Laboratory - SSDL, which is a laboratory designated by competent national authorities to undertake the duties of providing the necessary link in the traceability of radiation dosimetry to national/international standards for users within that country.

An SSDL is equipped with secondary standards calibrated at one or more PSDLs but always with ultimate traceability to the BIPM. Additionally, the IAEA and the World Health Organization - WHO administer the SSDL Network (www-naweb.iaea.org/nahu/dmrrp/SSDL/) that provides a forum in which national SSDLs perform dosimetric comparisons and thus strengthen radiation dosimetry consistency worldwide.

III.2.d Documentary standards

Documentary standards can cover a wide range of documents, reports, regulations, etc., but in the discussion here the most relevant are the standards exemplified by the International Standards Organization (ISO). The ISO-9000 series of generic standards have been adopted by manufacturers and service providers as a way of demonstrating conformity of process. For calibration laboratories (primary and secondary) ISO-17025 [40] combines the Quality Assurance - QA aspects of ISO-9000 with some specific technical requirements to ensure an adequate level of technical competence in the dissemination of measurement standards. There may also be national guidelines and, for the end-users, protocols and best practice procedures developed by national and/or international bodies such as the IAEA [27] or AAPM [64].

III.2.e Traceability as process review

From wherever you are in the chain (end-user, PSDL, secondary laboratory) it's important to note that traceability is more than simply a label attached to a measurement. A review of how that traceability has been achieved is a very useful review of the quality of the calibration process, essentially telling you the "How" of what is done to produce the statement like "NIST-traceable". For any QA program this "How" is important information as it demonstrates not only that there is an unbroken measurement chain linking the end-user to the primary realization of the quantity but that this chain has been executed in the appropriate manner (using the correct measurements standards and following the correct documentary standards). It is not something to do in detail very often, but a worthwhile exercise for anyone wanting to ensure that the measurements they carry out are as good as they would like.

III.3 Uncertainty

As noted at the beginning of this chapter, it is essential to estimate the uncertainty in the result obtained for any measurement. There is much in the literature that describes the evaluation of measurement uncertainties and one finds as many different recommendations and approaches as there are authors in this field. The starting point, however, is the ISO Guide on the Uncertainty in Measurement, ISO GUM [39], also known as the JCGM Report 100 [46] which provides a comprehensive approach to uncertainties in general. A very detailed review of uncertainties for radiation dosimetry is given by Mitch et al. [65] and Castro et al. [66] provide an example of an uncertainty analysis and budget for a typical measurement in a clinical radiation therapy facility.

Here we will examine a very common situation for radiation dosimetry, a reference absorbed dose to water measurement in a photon beam from a linear accelerator. The approach for any other situation can be adapted from this example. The starting point is to write down the governing equations:

$$D_w^Q = Mk_Q N_{D,w}^{60Co} \quad \text{III.1}$$

Where $N_{D,w}^{60Co}$ is the calibration coefficient for the detector in a reference field (in this case ^{60}Co), k_Q is a factor that converts the calibration from the reference field to the user field, Q , and M is the detector measurement. In many radiation dosimetry measurements the equation to obtain the absorbed dose can be written in this way.

It is useful to break down the measurement component further:

$$M = M_{raw} \cdot P_{TP} \cdot P_{ion} \cdot P_{pol} \quad \text{III.2}$$

Where M_{raw} is the detector reading and the P-factors are a series of corrections applicable to the chosen detector, not all are shown in equation III.2. The reader is referred to the AAPM's TG-51 protocol and its Addendum [67] for a full description of the equations and correction factors for this example.

In thinking about the measurement process further it is useful to list the parameters on which each of the components depends:

$$M = M_{raw}(x,y,z,SSD,FS) P_{TP} P_{ion}(V) P_{pol}(V, D_{pp}) \quad \text{III.3}$$

The effect of experimental setup is revealed in equation III.3 in that the chamber reading is seen to be a function of position within the water phantom (x, y, z) distance from the radiation source (SSD) and the size of the radiation field (FS). Correction factors for polarity and ion recombination are a function of the polarizing voltage of the chamber and the dose per pulse of the incident beam (D_{pp}).

One can further decompose equations III.1 and III.3 by listing the equipment that is used for each step or correction factor and thus identify more dependencies and/or uncertainty components. It can be therefore seen that with this comprehensive break-down of the measurement equation one has already begun to examine the measurement procedure in some detail (rather than only focusing on the final result).

This makes the derivation of the overall uncertainty budget more robust and provides the process review by which one can then identify the measurement steps or components that need addressing to reduce the overall uncertainty. In this respect it is similar to the general QA approach of Failure Mode Effects Analysis - FMEA laid out by the AAPM's Task Group 100 [68].

To continue with this example, we will examine a number of components in equations III.1 and III.3 in more detail. The aim is not to derive an overall uncertainty budget but to show steps that can be used for any radiation dosimetry situation. In each case, it is important to analyze both the procedure itself and any underlying assumptions.

III.3.a Example #1 - Measurement, $M_{raw}(SSD)$

For a detector positioned at some distance from the radiation source (in this case a linac) there will be some dependence on that distance, specifically the difference between where it is assumed to be and where it actually is.

III.3.a.i Procedure followed

With the mechanical positioning devices available with modern linacs (e.g., a mechanical front pointer or distance stick) and careful technique, the Source-Surface-Distance can be determined with a measurement precision of 0.5 mm. We then use the inverse square law to give the relative uncertainty at the specific SSD. The use of other distance indicators (light-field distance indicator or room lasers) will likely lead to a larger uncertainty. Such systems are convenient but, as the AAPM TG-106 [69] report notes, their accuracy must be verified first.

III.3.a.ii Assumption

The mechanical positioning device (front pointer) correctly indicates the SSD. This would normally be verified during other QA activities related to the linear accelerator. This highlights the fact that further dependencies (and therefore, potential uncertainties) are exposed as we ‘un pack’ the procedure.

Similar analysis would then look at the positioning procedure to place the detector at the reference point in the water phantom,

$M_{\text{raw}}(x,y,z)$, and the effect of the size of the radiation field, or difference from what is defined, $M_{\text{raw}}(\text{FS})$. This will lead us to look at, among other things, the distance calibration of the water phantom and the congruence of the light and radiation fields of the linac.

Since this is an activity where the skill of the user has an impact, it may also lead us to look at person-to-person variability, an uncertainty component not often taken into account. If the uncertainty estimate for this component is judged to be too large then this would feedback into a review/improvement of the set-up procedure to minimize or eliminate such dependence on the individual, or result in revised training plans to ensure all users can complete the task at the required level. We see that from what appears to be a narrow activity – determination of the uncertainty related to detector position – a wide range of outputs are possible.

III.3.b Example #2 - Calibration data, $N_{D,w}^{60\text{Co}}$, k_Q

With the possible exception of the choice of calibration laboratory (see note on the CIPM-MRA earlier), the user has no influence on the uncertainty of the calibration coefficient. However, it is worth noting that the certificates issued by calibration laboratories often provide a lot of useful information, including the procedure used in the calibration, the uncertainty in the calibration, and guidelines for using the calibration data.

II.3.b.i Assumption

The calibration coefficient is valid for the detector on the date of measurement. This raises the question of the stability of the detector between the time of calibration and time of use and leads to a potential review/development of any monitoring procedure to ensure the correct operation of the detector.

The conversion factor, k_Q , may also be provided by the calibration laboratory but often the data has to be taken from the research literature or national/international protocols. Again, the end-user does not affect the uncertainty in any tabulated data but the procedure used to select and implement this data also has an uncertainty component and therefore requires some analysis.

III.3.b.ii Procedure followed:

In the case of an ionization chamber where the k_Q data are given as a function of some beam quality specifier then the user-dependent components include:

1. measurement of user's beam quality specifier;
2. selection of k_Q values for the user's detector from tabulated data;
3. interpolation of tabulated data to user's beam quality.

III.3.b.iii Assumption

The tabulated data applies to the specific detector being used for the measurement. The uncertainty in this assumption will depend on how the tabulated data was determined (e.g., from experiment or calculation, consensus or a single data set).

It may turn out that some of these uncertainty components are insignificant, depending on other uncertainty components and the overall target uncertainty, but the activity of going through the measurement procedure in detail is still very useful.

III.3.c Example #3 - Correction factor, P_{TP}

An ionization chamber reading usually needs to be corrected for variations in air density in the sensitive volume of the detector. This is done through what is referred to as the temperature-pressure correction, P_{TP} :

$$P_{TP} = \frac{(273.15 + T_{det})}{(273.15 + T_{ref})} \cdot \frac{P_{ref}}{P_{det}} \quad \text{III.4}$$

III.3.c.i Procedure followed

Equation III.4 shows that the evaluation of this correction requires a thermometer and barometer, and these should be calibrated and have sufficient resolution, 0.1 °C and 0.1 kPa, respectively. The actual measurements are the temperature of the phantom in which the chamber is placed and the room pressure.

This thermometer should span the temperature range from -0.5 to 30 °C, with a resolution of 0.2 °C or better. A barometer capable of measuring the atmospheric pressure with a calibration uncertainty of 0.1 % or better is required. These references should be kept as the laboratory standard [38].

III.3.c.ii Assumption

The measured temperature should equate to the actual temperature of the ion chamber. If the water phantom is not in equilibrium with the room temperature then there will be temperature gradients within the phantom which could affect both the determination of P_{TP} and the measurement of M_{raw} . This could impact the experimental procedure but also result in a review of the environmental control in the room where the measurements are being carried out (especially when one takes the potential effect of humidity into account as well). If the ion chamber is not in thermal equilibrium with the water then the temperature measurement will be incorrect [70]. This leads to guidelines for the measurement procedure for ensuring that thermal equilibrium is achieved (e.g., minimizing temperature differences, allowing sufficient time for stability to be achieved, etc.). To complete the uncertainty analysis, one must also

consider the thermal expansion of the chamber thimble, although this effect is usually very small [71].

III.3.c.iii Assumption

To apply equation III.4 using a room pressure measurement the assumption is that the ion chamber sensitive volume is not sealed from the environment. Some calibration laboratories include an air-communication test to verify this but the user may need to implement some chamber care procedure as a further check.

III.3.c.iv Assumption

Equation III.4 is generally valid but it has been shown that measurements deviate from the predicted correction for low energy x-ray beams [72]. This depends on the detector size, the X-ray energy and the local atmospheric correction.

Other corrections such as those for ion recombination, leakage currents can be analyzed in a similar way. The aim is always to obtain an uncertainty estimate, but, as shown above, very often there is an analysis/review of the measurement procedure. A lack of space prevents further discussion, but it should be noted that equation III.1 is not completely correct as the dose measurement is a function of some delivery parameter (e.g., Monitor Units for a clinical linac, or irradiation time for a radioactive source). This then introduces the performance of the irradiator as an uncertainty component that needs to be estimated.

III.4. Conclusion

This brief chapter has merely scraped the surface of the twin topics of traceability and uncertainty but the aim has been to attempt to show that these are more than academic concepts relevant to calibration laboratories and those involved in drafting documentary standards. Both traceability and uncertainty are

fundamental to any measurement that must stand up to external scrutiny and the process of incorporating these two components in the measurement procedure has significant, positive, implications. As, hopefully, shown in the sections above, by looking at these two aspects of the measurement one considers the entire measurement procedure and gains a greater understanding of the linkages between calibration steps and the linkages between measurement steps and should result in improved confidence in both the procedures used in any particular dosimetry situation and, ultimately, the results produced.

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Chapter IV

SI Dissemination

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The unit dissemination for such radiation quantity is realized by calibration chain and must be reliable and traceable to the SI measurement, but the calibration services follow the countries' and organizations' development strategies, which become a more competitive trade market. The quality gains contributions, among others, for services and products eliminate waste and rework, reducing costs.

The preliminary discussion about dissemination concept requires traceability understanding, which is an essential property of laboratory calibration services that is offered to clients to obtain and demonstrate ability quantity dissemination.

The metrological traceability concept is presented in the International Vocabulary of Basic and General Terms in Metrology – VIM [31 and 54] –as a “measurement result property that can be related to a reference through a documented unbroken chain of calibrations, each contributing to the measurement uncertainty” [73]. Therefore, the corresponding terms for quantity measurement value were obtained from the reference standard and not from the institute that the results come from, e.g., “traceable to a National Metrology Institute ...”

The traceability not only relates the current level to technological development, maintenance and improvement but should be more reliable than the calibration service accreditation

by the institute and should be objective and widely recognized for the demonstration of reliability and competence.

The measurement could be correctly interpreted anywhere essential; the Bureau International of Weights and Measures (BIPM) since 1875 has led efforts from standard establishment, developing concepts and performing agreements to metrological traceability, to growing international metrological infrastructure and being able to certify the Inspection and Compliance agencies' (OACs) competence. The OACs are known as calibration, performance test and clinical analysis laboratory, reference materials and proficiency test producers that are performed by the International Laboratory Accreditation Cooperation (ILAC) [74].

The measurement science known as Metrology has been developed for thousands of years because man first began to live in society when measurements were needed to build things and for production control, services and goods exchange. The first social group developed its own measurement system; however, there was divergence between regional measurement systems in part because quantities are expressed in unreliable measurement units, such as human anatomical parts (i.e., feet or inches), which vary between people.

Probably the first known standard was the Egypt pharaoh Khufu, 2900 AC, during pyramid building, using black granite standards that were traceable at the pharaoh's forearm to hand and using rods and ropes as secondary standards or work standards. The result was a secured base to an almost square pyramid; each average value side length was 228.6 m, and the standard deviation was 0.064 m (0.05 %).

Among so many interesting standard concepts and constructions, one was in 1305 DC, when the English King

Edward I decreed to be considered tree dry barley grains measured side by side as one inch. Immediately, the British shoemakers implemented the concept and began to manufacture standard size shoes, based on these units; thus, the forty-barley-grain shoe came to be known as size 40.

IV.1 Metrology development and its infrastructure

Through technological and social development during the XX and XXI centuries, metrological concept and process improvements were demanded. Measurements are present in daily activities, including at the supermarket, at the gas station, in the medical field, in medicine production and application, and in the environmental and people protection systems, among others. However, the commercial activities were the main introducers of the actual international metrology infrastructure.

During the late eighteenth century in France, one successful initiative was proposed to use the 10^{-7} equator line fraction to the North Pole distance and the meridian that passes through the Paris observatory as the length unit. Then, in 1799, the platinum and iridium alloy rod was deposited in the France National Archives as result of these units, and the metric system was instituted. Following, several countries have adopted this system.

Conversion measurements are important for exchanging money, and one scientist group during the Paris Universal Exhibition in 1867 formed a “Committee of Weights and Measures and Money” aimed at standard measurements.

Following this direction, the "International Committee of the Metro" meeting in 1872 with 30 countries' delegates distributed across various countries the reference standards of meter and kilogram, starting the international metrology structure. Three years later, on May 20, 1875, the Metre Convention was signed,

establishing the metrology international authority; this was revised in 1921 and 1960 when the Metric System was replaced by the International System of Units or SI. Currently, 56 countries are signatories of the Metre Convention.

Three international organizations promote and maintain the metric standards:

- Bureau International of Weights and Measures (Bureau International des Poids et Mesures) – BIPM. This bureau maintains an international metrology center in Sèvres, France. The BIPM's mission is to establish the units and the main international standards for physical quantities and prototypes; performing standard comparisons; and coordinating and making determinations related to the physical constants that are involved in those units. Initially, the activities of the BIPM were limited to length and mass quantities, but in the twentieth century, the activities extended to other quantities, such as electrical in 1927, photometric and radiometric in 1937, ionizing radiation in 1960, time scales in 1988 and chemistry in 2000.
- General Conference on Weights and Measures (CGPM - Conférence Générale des Poids et Mesures) establishes policies and adopts management measures, including the BIPM budget. It must also ensure the SI dissemination and improvement through the CGPM meeting every four years.
- International Committee for Weights and Measures (CIPM - Comité international des poids et mesures) has 18 committee members elected by the CGPM, is responsible for managing strategic advisory committees with an annual expert meeting, and prepares proposed work to be submitted to the General Conference. The Consultative Committee of Ionizing Radiation

(CCRI) has expert Electron, Gamma and X ray, Radionuclide and Neutron committees.

IV.2 National Metrology Institute - NMI

The National Metrology, Quality and Technology Institute (INMETRO) in 1973 succeeded the National Institute of Weights and Measures (INPM) to manage the Brazilian metrological policy for technological services infrastructure for calibrating instruments and to evaluate and certify the processes, services and quality products. The Brazilian Metrology System (SINMETRO) is related to metrology, standardization and compliance certification, consisting of the public and private institutions that perform activities. The National Councils of Metrology, Standardization and Industrial Quality (CONMETRO) were created at the Ministry of Industry and Trade.

IV.3 Mutual Recognition Arrangement – MRA

The world trade commerce established intense and complex rules for business in 1995. In general, trade relations are accompanied by product checks (goods or services) for type tests, and standard compliance became the trade market process, avoiding unnecessary obstacles in the creation of international trade, additional costs and "tested once, accepted everywhere". The established mechanisms would be able to assess the structure metrology and conformity for each country to achieve recognition by the appropriate international metrological system.

The BIPM was naturally included in these discussions because of its international metrological scenario leadership and its mission. There is the National Metrology Institutes (NMIs) cooperation with other institutions designated by these NMIs, as well as a number of international and intergovernmental

organizations, such as the Regional Metrology Organizations (RMO):

- Asia Pacific Metrology Program – APMP
- Euro-Asian Cooperation of National Metrological Institutions - COOMET
- European Association of National Metrology Institutes – EURAMET
- Intra-Africa Metrology System - AFRIMETS
- Inter-American Metrology System - SIM

The international comparison process of NMIs was held more than one hundred years and could provide the technical basis for some kind of recognition, but this recognition was insufficient for the intended purposes; a more formal recognition of national measurement standards was necessary. Discussions with the ILAC reinforced this view.

The BIPM, motivated not only by the international trade demands but also by science and technology, formulated a series of recommendations to improve the global traceability of NMI measurement standards.

During the 21st CGPM meeting in 1999, the Mutual Recognition Arrangement (MRA) [75] was signed. This agreement improves the technical basis for international agreement on trade markets, services and regulation exchange. INMETRO signed the MRA and designated the National Metrology Laboratory of Ionizing Radiation (LNMRI) from the Institute of Radiation Protection and Dosimetry (IRD / CNEN) for ionizing radiation NMI. Thus, the LNMRI / IRD / CNEN participate under the supervision of INMETRO in all of the planned Agreement activities.

Briefly, the MRA objectives are to:

- Establish the equivalence degree of national measurement standards maintained by NMIs.
- Provide the mutual recognition of calibration and measurement certificates issued by NMIs.
- Provide a consistent technical basis for relating to international trade and regulatory activities for more comprehensive agreements.
- It is expected that these objectives are achieved through:
 - Key comparisons: international comparison results of measurement standards, allowing a quantitative measure of the equivalence degree of national primary standards, available in the Key Comparison Data base - KCDB on the BIPM webpage (www.bipm.org).
 - Peer Review audits: NMIs competence demonstration and quality systems.
 - Complementary comparisons promoted by RMO: successful involvement of NMIs.

Together, these three procedures demonstrate the confidence degree results reported to the participating institutions and thus promote mutual trust.

Formally, the equivalence degree NMIs recognize participants provided by MRA for each national measurement standards obtained from specific quantities and value key comparisons as promoted by the BIPM Consultative Committees or MRA, recognizing the calibration and measurements validity to magnitudes and specific intervals. The results of Peer Review audits carried out by experts from other NMIs and MRA approve the metrology processes, employing confidence by NMIs. The BIPM webpage published the MRA, the official document for each NMI informing of the calibration service offered,

metrological quantity used, measurement range and uncertainty, standard and, if needed, traceability.

The phrase "tested once, accepted everywhere" is widely achieved, not only in the calibration certificates agreement but also in the conformity assessment. The conformity assessment covers calibration and test laboratories, clinical analysis, reference material producers and test proficiency.

The document's recognition is necessary to establish multilateral agreement techniques. The most direct and robust technique would be established multilateral agreements, defining conditions regarding calibration and certificate acceptance. Supported by several international institutions that have contributed to standardizing the conformity assessment processes [51, 54 and 55], by metrology, each country develops its own conformity assessment process [55, 57 and 58].

International Laboratory Accreditation Cooperation (ILAC) [74] is an international organization that covers national laboratory accreditation and inspection, developing efforts to achieve a mutual international recognition agreement. The ILAC Arrangement was signed in 2000, and in 2001, the accreditation systems equivalence recognized certificates and reports that were issued by conformity assessment and accredited inspection organism.

In 2005, the statement was signed by CIPM and ILAC, establishing the roles and responsibilities of NMIs and National Accreditation Institutes to improve the traceability and the worldwide acceptance of the measurements under the MRA and the ILAC Arrangement. The uniform focus application established accreditation rule requirements, while ILAC guidelines and rules were published to establish policies for specific topics.

IV.4 Metrological traceability and ILAC policy

The metrological result traceability is considered a key issue, being necessary for the establishment of a policy to harmonize the concept and implementation understanding of the requirement. The MRA [75] and ILAC [74] arrangement does not provide guidelines or rules regarding metrological traceability. The National Accreditation Institutes use the general requirements for tests and calibration laboratories to serve this purpose [40]; the requirement for traceability in section 5.6.1. is “All equipment used for tests and/or calibrations, including equipment for subsidiary measurements (e.g., for environmental conditions) having a significant effect on the accuracy or validity of the result of the test, calibration or sampling shall be calibrated before being put into service. The laboratory shall have an established program and procedure for the calibration of its equipment.”

Calibration justifies the laboratories that are needed, as shown in other traceability requirements for the Calibration Laboratories as defined in section 5.6.2.1.1 [40] as “A calibration laboratory establishes traceability of its own measurement standards and measuring instruments to the SI by means of an unbroken chain of calibrations or comparisons linking them to relevant primary standards of the SI units of measurement. The link to SI units may be achieved by reference to national measurement standards. National measurement standards may be primary standards, which are primary realizations of the SI units or agreed representations of SI units based on fundamental physical constants, or they may be secondary standards, which are standards calibrated by another national metrology institute. When using external calibration services, traceability of measurement shall be assured by the use of calibration services

from laboratories that can demonstrate competence, measurement capability and traceability. The calibration certificates issued by these laboratories shall contain the measurement results, including the measurement uncertainty and/or a statement of compliance with an identified metrological specification.”

In traceability requirements, reference standards cases are defined in section 5.6.3.1 [40] as “The laboratory shall have a program and procedure for the calibration of its reference standards. Reference standards shall be calibrated by a body that can provide traceability as described in 5.6.2.1. Such reference standards of measurement held by the laboratory shall be used for calibration only and for no other purpose, unless it can be shown that their performance as reference standards would not be invalidated. Reference standards shall be calibrated before and after any adjustment”.

For the ILAC Policy on the Traceability of Measurement Results [74], the instruments and reference standard should be calibrated.

IV.5 NMI policies for traceability

The NMI policy [76] states that "Metrological Traceability in accrediting conformity assessment bodies and recognition of conformity to the principles of Good Laboratory Practice", in section 8.2 states "Reference standards calibration and measuring instruments by external laboratories" and establishes that "In order to assure the metrological traceability performs on conformity assessment and test facilities should ensure that their reference standards calibration and their measuring instruments need to be calibrated by laboratories that can demonstrate competence, measurement capability and traceability for specific

calibration is performed. The following organizations meet these requirements".

IV.6 Brazil SI dissemination: X and gamma ray calibration laboratories

INMETRO is a signatory of the MRA; therefore, all of the NMI members of the Scientific and Industrial Metrology Division (DIMCI/INMETRO), the Time Service Division of the National Observatory (DSHO/ON) and the National Metrology Laboratory of Ionizing Radiation (LNMRI/IRD/CNEN) as designated laboratories are also included in the MRA.

The LNMRI/IRD MRA scope includes the three considered areas in the ionizing radiation metrology: X and Gamma rays, Electrons and Charge Particles, Radionuclides and Neutrons. In this chapter, we only show the aspects that are related to the X and gamma radiation fields, which have calibration services for used instruments in radiation protection (measurements in environments and worker exposure), radiotherapy and diagnostic radiology.

Seven other Brazilian calibration laboratories have instrument calibration services at these quantities; only one of them has calibration in radiotherapy service, and six have calibration services in radiation protection and in diagnostic radiology.

The Radiation Protection and Nuclear Safety Section of the National Nuclear Energy Commission –DRS/CNEN –requires for radiation protection used instruments to calibrate by certified laboratories approved at Test and calibration services committee (CASEC/IRD). The technical requirements for agreement are shown in the "Technical Requirements for Certification of Laboratories Calibration used in Radiation Protection," where, "The instrument used as reference standard should be calibrated

by secondary or primary standard laboratory with a 3 years interval," and it is not required that the calibration services be accredited by NMI.

The Technology Brazilian System – SIBRATEC was established through Decree 6259/07 by the Science, Technology and Innovation Ministry – MCTI. SIBRATEC is composed of several networks with the goal of providing an infrastructure laboratory capable of offering companies services for conformity assessment (calibration, testing, analysis, and certification). It is expected that these technological services assist companies in overcoming techniques for domestic and foreign markets access, meeting the National Regulatory Agencies requirement demands. The Network Technology Services consist of 21 thematic networks; one of these networks is for Radiation Protection and Dosimetry (Metroradi).

The metrological traceability of reference standards for calibration services in Brazil is given by the LNMRI reference standards, which is the INM by INMETRO's designation. The LNMRI is traceable at Physikalisch-Technische Bundesanstalt (PTB) reference standards for radiation protection and diagnostic radiology and at BIPM for radiotherapy. Figure II.1 shows the hierarchy system of the gamma and X ray measurement. The Peer Review is the audited form indicating the INMETRO accreditation policy; the LNMRI calibrations services can be found in appendix C of the KCDB on the BIPM webpage.

Chapter V

Radiation Protection Quantities and Units: Desirable Improvements

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V.1 Introduction

The purpose of this paper is to review critically the global system of quantities² and units³ that was created for the purpose of protecting people against the detrimental effects attributed to exposure to ionizing radiation (or radiation, in short), under the aegis of the International Commission on Radiological Protection

²The term *quantity* is used to describe the measurable property of a phenomenon, such as radioactivity or radiation; it is NOT used a synonym of amount, i.e., the total of something in number, size, value, or extent (It is noted that in Latin derived languages the equivalent to the term *magnitude* is used to mean *quantity*, and the equivalent of the term *quantity* is used to mean *amount*; this has exacerbated translation problems). Quantities can be *extensive* or *intensive* as follows: an extensive quantity is equal to the sum of that quantity for all of its constituent subsystems (examples include volume, mass, and electric charge); conversely, an intensive quantity is independent of the extent of the system (examples include temperature, pressure, and density). This is an important distinction that is relevant for radiological protection; for instance, the quantities *dose* and *collective dose* (namely the summation of all individual doses in an exposed group) are considered akin, however dose is an intensive quantity and collective dose is an extensive quantity –they are separate entities.

³The term *unit* is used to mean an amount of a quantity used as a standard of measurement; e.g., units of time are second, minute, hour, day, week, month, year and decade.

(ICRP) [58, 59] and the International Commission on Radiation Units and Measurements (ICRU) [60, 61]. This system was adopted by the Bureau International des Poids et Mesures [82] and has been formally established in international intergovernmental standards [39,63].

The system has proved successful in helping radiological protection to become a globally uniform, consistent and coherent professional discipline. However, as it happen with any other successful development, the experience gained over time is showing that the system my benefit from some improvements. The time seems to be ripe for undertaking a deep review of the current system of radiological protection quantities and units and suggest the necessary revisions to update it, by taking into account a number of lessons learned, particularly in the aftermath of nuclear accidents and in the protection of patients in the practices of radio-diagnosis, interventional radiology and radiotherapy. The paper analyses difficulties with the system and implicitly suggest some feasible solutions.

V.2 Quantities

The history of radiological protection reflects the attempts to identify quantities for measuring human radiation exposures while also providing a metric for inferring the risk associated with the exposure. After many decades, ICRP and ICRU converged upon a system of protection quantities that are related to the risk associated with radiation exposure. The system evolved around the following basic physical quantities:

- the “exposure”, which is the sum of the electric charges on all ions of one sign that are produced when all electrons liberated by the ionizing radiation in a volume of air are completely stopped, divided by the mass of air in that volume;

- the “kinetic energy released per unit mass”, known by its acronym kerma, which is defined as the sum of the initial kinetic energies of all the charged particles liberated by uncharged ionizing radiation;
- the “absorbed dose”, which represents the mean energy imparted to matter per unit mass by ionizing radiation. In the medical area especially in radiation therapy, the evolution from the quantities exposure to kerma and to absorbed dose was relatively smooth. The realization and consistency of the later is now done in a very robust system of primary standards, based on a water and a graphite calorimeter, chemical dosimetry and ion chamber with very low uncertainties compared to what is the normal practice in radiation protection.

A system of so-called protection quantities was developed by ICRP and ICRU for purposes of radiological protection. These protection quantities allow quantification of the extent of exposure of the human body to radiation from both whole and partial body external irradiation and from intakes of radionuclides.

The system is founded on the physical quantity absorbed dose and comprehends: the protection quantities termed equivalent dose, effective dose; the operational quantity termed dose equivalent and its derivatives ambient dose equivalent and personal dose equivalent, as well as definitions of recording quantities.

The protection quantities are strange quantities because they do not meet the more elementary requirements for a quantity: they are neither measurable nor traceable; accuracy or precision in their amount cannot be formally defined. In spite of these formal shortcomings, they are universally used and have greatly helped radiation protection and nuclear safety regulators, who are fortunate to have a single quantity-concept to measure the levels

of protection and safety bringing together many other variables characterizing the relationship exposure-risk-protection.

Interestingly, and somehow surprisingly, regulators have been able to standardize all the complex balancing in that relationship to achieve a unique and universal set of quantities that governs radiation and nuclear protection and safety. Moreover, they have been able to agree on international normative establishing standardized conversion factors between basic physical quantities that can be measured in nature and the protection quantities.

Few human endeavors have achieved this level of sophisticated simplification to characterize and regulate exposure to a detrimental agent. However, the protection quantities are unique and universal, but "by definition" and not in the mathematical sense of "existence and uniqueness". Because if they were in that sense, they should be able to solve all problems covering the entire field of interest without further clarification (and this chapter would not be needed!).

As in every niche of human endeavors, the description and the model that explains and rationalize what is needed to protect humans and their habitat against the harmful effects of exposure to ionizing radiation is a finite and simplified representation of the complex reality of radiation exposure and its health effects. While such a reality is not infinite in its descriptors, at least it exceeds its usual modeling in size and complexity.

The models used in practice are aimed at being elegant and simple and they are usually tailored to the principle of intelligibility, namely of what can be comprehended by the human mind in contrast to sense perception. This is done using that body of techniques for investigating phenomena, acquiring new knowledge, or correcting and integrating previous knowledge usually termed 'scientific method'.

When the scientific method is used to describe and model the reality associated with radiation exposure and its health effects, inevitably some qualities are lost, including subtle, specific anomalies and phenomena associated with such complicated problem. Intelligibility, usually need to simplify and average, forgetting singularities, losing some details of the phenomenon and its consequences. Of course it is remarked that this happens in all areas of science and technology but it seems to be particularly sensitive in the sciences of quantifying radiation exposure, perhaps due to the common (and wrong) connotation of associating huge harm with radiation exposure situation.

Radiation protection is essentially based on qualifying and quantifying inferred biological damage that would result from radiation exposure, with the starting point being obtaining measurement of basic physical quantities, such as the activity of a radioactive substance and flux of radiation from a source emitting radiation exposing the target, and the ultimate physical quantity that is the absorbed dose by the target.

While this initial valuation is governed by the objective laws of physics and chemistry, it is followed by a semi-objective weighting of this reality, which includes biological knowledge but also perceptions and 'insights'. The epistemology of this second part, namely from physical certainty to perception of harm, still is basically an unresolved problem.

Accordingly, there is a systemic interaction of both sets of approaches that are epistemologically very different and this affects intelligibility. In fact, it should be recognized that physics; chemistry and biology have not yet settled a unified theory able to generate a description of phenomena such as the interaction of radiation with living matter and its ultimate consequences, at

least without significant gaps that are dominant at the operative field.

The theoretical plexus underpinning radiation protection and safety is based on conservative assumptions and the radiation protection quantities are tailored to this reality. Unsurprisingly, there are some issues that need to be improved in both scientific description and the definition of concrete operations to solve a problem. For instance, a recurrent conundrum has been working with physical quantities vis-à-vis protection quantities at intermediate ranges of dose, for instance near that border region where the so-called ‘deterministic health effects’ start to dominate in detrimental importance the so-called stochastic health effects, namely the border region between the stochastic and deterministic.

Under the dominance of these fundamental epistemological problems the system of radiation protection quantities and units was build. It is briefly described hereinafter.

V.2.a Absorbed Dose

The fundamental quantity of the protection quantities is the mean of physical quantity termed absorbed dose, which is defined in specified organs and tissues in the human body, i.e. it is the mean energy deposited in a tissue or organ (T), divided by its mass. The mean absorbed dose averaged over the tissue or organ T, D_T is given by the expression V.1

$$D_T = \varepsilon_T / m_T \quad \text{V.1}$$

Where ε_T is the mean total energy imparted in the target region of a tissue or organ T, disregarding the beam geometry, and m_T is the mass of that tissue or organ.

The target region is the anatomical region within the body in which radiation is absorbed, which may be an organ or a specified tissue as in the gastrointestinal tract, urinary bladder, skeleton, and respiratory tract, all treated as a homogenous media.

V.2.b Equivalent Dose

The quantity absorbed dose is not directly related to radiation risk because different radiation types have different efficiencies to produce harm. Therefore, in order to define a quantity related to radiation risk, the organ and tissue absorbed doses are weighted by dimensionless radiation weighting factors to account for the differences in biological effectiveness of different types of radiations from external and internal sources.

The radiation-weighted organ and tissue absorbed doses are termed equivalent dose. Thus, the equivalent dose, H_T , in a tissue or organ, T, is defined as the mean absorbed dose from radiation in the tissue or organ T, weighted by appropriate radiation weighting factors, and it is given by the expression V.2:

$$H_T = \sum_R w_R D_{T,R} \quad \text{V.2}$$

where $D_{T,R}$ is the mean absorbed dose from radiation R in a tissue or organ T, and w_R is the radiation weighting factor for radiation R.

The radiation weighting factor, w_R , is an over simplified dimensionless factor by which the organ or tissue absorbed dose is multiplied to reflect the higher biological effectiveness of high-linear energy transference (LET) radiations compared with low-LET radiations. Linear energy transfer (LET) is defined as the average linear rate of energy loss of charged particle radiation in a medium, i.e., the radiation energy lost per unit length of path through a supposedly homogeneous material.

Consequently, LET can be expressed as the quotient of dE by dl where dE is the mean energy lost by a charged particle owing to collisions with electrons in traversing a distance dl in matter, $LET = dE / dl$. The radiation weighting factors are chosen on the basis of experimental values of the relative biological effectiveness (RBE) of various radiation types for various endpoints.

The RBE is defined as the ratio of a dose of a low-LET reference radiation to a dose of the radiation considered that gives an identical biological effect. RBE values may vary with the dose, dose rate, and biological endpoint considered. The currently recommended radiation weighting factors are:

- 1 for X- and γ photons, β particles, electrons and muons,
- 2 for protons and charged pions,
- 20 for α particles, fission fragments and heavy ions and a value derived from a continuous function for neutrons with a highest values of 20 for energies of around 1 MeV.

V.2.c Effective Dose

Different organs and tissues present different sensitiveness to radiation. Therefore, in order to account for such sensitiveness, equivalent doses have to be weighted with tissue weighting factors. The weighted summation of equivalent doses is termed effective dose and is defined as the sum of the equivalent doses in all specified tissues and organs of the body, each weighted by tissue weighting factors representing the relative contribution of that tissue or organ to the total health detriment. In sum, the effective dose, E , is the tissue-weighted sum of the equivalent doses in all specified tissues and organs of the body, given by the expression

$$E = \sum_T w_T H_T \quad \text{V.3}$$

where: $H_T = w_R D_{T,R}$ is the equivalent dose in a tissue or organ, T, and w_T is the tissue weighting factor for tissue T. Thus, the quantity *effective dose* became the risk-related (or risk-informed) dose for the whole body.

The *tissue weighting factor*, w_T , is the factor by which the equivalent dose in a tissue or organ T is weighted to represent the relative contribution of that tissue or organ to the total health detriment resulting from uniform irradiation of the body [35]. The weighting is such that: $\sum_T w_T = 1$. In this definition, no attempt is made to separate acute from protracted doses or the organ or tissue heterogeneity.

It is also not clear if the definition for *effective dose* uses age- and sex-averaged *tissue weighting factors*. Nevertheless, for a population of both sexes and all ages these tissue weighting factors are applied to the sex-averaged organ equivalent doses of the reference person and not to a specific individual [79].

The values of each *tissue weighting factors* are less than 1 and the sum of all *tissue weighting factors* is 1. The values are chosen by the ICRP considering epidemiological studies of organ-specific detriment factors, in particular of Japanese A-bomb survivors.

The currently recommended *tissue weighting factors* are 0.12 for bone-marrow, colon, lung, stomach and breast, 0.08 for gonads, 0.04 for bladder, esophagus, liver and thyroid, 0.01 for bone surface, brain, salivary glands and skin, and a value of 0.12 is assigned to remainder tissues adrenals, extra thoracic region, gall bladder, heart, kidneys, lymphatic nodes, muscle, oral

mucosa, pancreas, prostate (♂), small intestine, spleen, thymus, and uterus/cervix (♀).

V.2.d Dose Equivalent

The quantity equivalent dose cannot be measured directly in body tissues and the quantity effective dose is by definition immeasurable as it is a summation of weighted equivalent doses. This immensurability of the basic radiological protection quantities create problems for ensuring compliance with standards, e.g., throughout environmental or personal monitoring of the incurred radiation exposure. In order to solve this serious problem of applicability, ICRP and ICRU proposed measurable operational quantities [58, 60, 61] to be used in practical applications for monitoring and investigating situations involving external exposure. They are defined for measurements and assessment of doses in the body.

The fundamental operational quantity is termed *dose equivalent*, H, which is conceptually defined as the product of D and Q at a point in tissue, where D is the absorbed dose and Q is a *quality factor*, which is defined for the specific radiation at this point, thus $H = DQ$. Q characterizes the biological effectiveness of a radiation, based on the ionization density along the tracks of charged particles in tissue. Q is defined as a function of the unrestricted LET of charged particles in water. The value of Q as function of LET is as follows: Q (LET)

- = 1 for $LET < 10 \text{ keV}/\mu\text{m}$;

- = $0.32 \text{ LET} - 2.2$ for $10 \leq LET \leq 100 \text{ keV}/\mu\text{m}$;

- = $300/\sqrt{LET}$ for $LET > 100 \text{ keV}/\mu\text{m}$.

Radiation monitors for external radiations are calibrated in terms of operational quantities derived from the *dose equivalent*. These operational quantities are termed *ambient dose equivalent*

and *personal dose equivalent*. The *ambient dose equivalent*, $H^*(10)$, is the dose equivalent at a point in a radiation field that would be produced by the corresponding expanded and aligned field in the do-called *ICRU sphere*⁴, at a depth of 10 mm on the radius vector opposing the direction of the aligned field. The *personal dose equivalent*, $H_p(d)$, is the dose equivalent in soft tissue (which is commonly interpreted as the '*ICRU sphere*') at an appropriate depth, d , below a specified point on the human body, where the specified point is usually given by the position where the individual's dosimeter is worn. Measurements in terms of dose equivalent are used to estimate effective dose (see 'recording quantities' below).

V.3 Recording Quantities (Missing Protection and Operational Quantities)

The protection quantities and the operational quantities become mixed-up when doses are recorded for regulatory purposes. ICRP has defines a *dose of record* as the effective dose of a worker assessed by the sum of the measured personal dose equivalent $H_p(10)$ and the committed effective dose retrospectively determined for the Reference Person⁵ using results of individual

⁴ The *ICRU sphere* is a sphere of 30 cm diameter made of tissue equivalent material with a density of 1 g/cm³ and a mass composition of 76.2% oxygen, 11.1% carbon, 10.1% hydrogen and 2.6% nitrogen, which is used as a reference phantom in defining dose equivalent quantities [81]

⁵ The *Reference Person* is an idealized person for whom the organ or tissue equivalent doses are calculated by averaging the corresponding doses of the Reference Male and Reference Female, namely idealized male or female with characteristics defined by the ICRP for the purpose of radiological protection, and with the anatomical and physiological characteristics defined in the report of the ICRP Task Group on Reference Man, ICRP Publication 89 [148].

monitoring of the worker and ICRP reference biokinetic and dosimetric computational models.

The equivalent doses of the Reference Person are used for the calculation of the effective dose by multiplying these doses by the corresponding tissue weighting factors. ICRP clarifies that the *dose of record* may be assessed with site-specific parameters of exposure, such as the type of materials and the Activity Median Aerodynamic Diameter (AMAD), which is the value of aerodynamic diameter such that 50% of the airborne activity in a specified aerosol is associated with particles greater than the AMAD, but the parameters of the Reference Person shall be fixed as defined by the ICRP. *Dose of record* is assigned to workers for purposes of recording, reporting and retrospective demonstration of compliance with regulatory dose limits. It is obvious that the dose of record is a quantity needed for practical reasons; however, it should be recognized that they add further 'impurity' to the questionable purity of the protection quantities.

V.4 Units of the Quantities

The unit of the fundamental quantity *dose* is the unit corresponding to energy per unit mass, namely *joule per kilogram* (J kg^{-1}) in SI units. The unit of absorbed dose is therefore absorbed J kg^{-1} . The protection quantities *equivalent dose* and *effective dose* and the operational quantity *dose equivalent*, as well as those derived from it, such as the *personal dose equivalent* or the *ambient dose equivalent*, also have the same unit, J kg^{-1} , because both are obtained by multiplying absorbed dose with dimensionless weighting factors.

However, in order to avoid confusion, within the system of SI units it was internationally agreed to use the special name *gray* (Gy) for the J kg^{-1} of absorbed dose and the special name *sievert*

for the J kg^{-1} of all the other quantities [82]. This policy was endorsed by the Consultative Committee for Units (CCU) [83].

V.5 Analysis

The successful system of radiation protection quantities and units has shown some shortcoming, as follows.

V.5.a General difficulties

In general terms, it seems that the system includes a myriad of quantities and there has been substantial confusion among professionals and the general public on their distinction, use and even need. There has also been misunderstanding on the perception of the units used to express the values of such quantities. It should be recognized that some of these problems are simply linguistic and grammatical, including difficulties in translation. These are issues of concern to the metrological community responsible for the physical realization of the quantities and its worldwide dissemination and meaningful traceability.

A more serious problem is that the system is intended by definition to deal with low levels of exposure, namely exposures involving radiation doses sufficiently low as to ensuring the appropriate protection to people. It is not intended for dealing with high-dose exposure situations such as those than may occur after an accident and also in some medical practices, for instance in radiotherapy and interventional radiology. Since people involved in high exposure situations have also to be protected, the absence of an ad hoc system of quantities for those situations have forced the use of either the existing system, or ad hoc extensions of it, also for such high exposure situations. This has caused and continues to cause many problems. The interpretation of the dose levels measured or estimated may in several

situations also lack a direct correlation with the way the quantities were defined and realized. The reference conditions such as distance, field size, energy fluence, beam geometry and homogeneous media may differ substantially from the real exposure situation.

V.5.b Difficulties with the absorbed dose

There is a problem with the definition of absorbed dose as a 'protection' quantity, which has further implications. The absorbed dose is a physical intensive quantity that in theory is definable at a point, e.g., as $D_T = \delta\varepsilon_T/\delta m_T$, where $\delta\varepsilon_T$ is the differential total energy imparted in the target point of a tissue or organ T, and δm_T is the differential mass of that point in the tissue or organ. However, the absorbed dose is defined as a mean. This may create problems for some organs where the mean is not necessary representative of the potential harm radiation can impose. In reality this is a simplified approach that neglects the fact that the harm to an organ depends on the level of damage to its function. The structural organ tolerance depends on the cell radio sensitivity and not on the irradiated volume and the functional organ tolerance depends on the type of organization and its functional reserve. For instance, tissues that have a parallel organization i.e. the lung, the inflicted harm will depend on the irradiated volume, while tissues with serial organization i.e spinal cord and coronary depend more on the punctual dose. Hence, in some medical practices the concept of mean is not necessarily representative and must be used with care when interpreting dosimetric results.

V.5.c Difficulties with the dose equivalent

The equivalent dose is not a stable quantity because its definition varies according to the values given to the radiation weighting

factors and the uncertainties associated with the simplified definition of mean absorbed dose and the tissue specificities, subject treated earlier. This presents problems for record keeping as the same quantity may have different values depending on the values of the radiation weighting factors at the time the equivalent dose was incurred.

V.5.d Difficulties with the effective dose

While the long search for the effective dose as a quantity suitable for setting exposure limits was completed in 1977, the concept and application of this quantity remains elusive and it is not easily understood. This is particularly severe in the case of medical uses (see hereinafter). The quantities absorbed dose and equivalent dose retain (in spite of their averaged nature) some of the conditions required for being defined as intensive quantities, but it is questionable to define the effective dose as an intensive quantity, and it remains somehow in a limbo between intensive quantities and extensive quantities. Moreover, its definition is not stable because it varies according to the values given to the tissue weighting factors. This presents problems for record keeping as the same quantity may have different values depending on the values of the tissue weighting factors at the time the effective dose was incurred.

Confusedly, the effective dose is 'risk-informed' and is a quantity used in protection to limit risks, but it is not a quantity to be used for risk assessment since it incorporates sex-, age- and tissue-specific averaging for a referent individual and not for specific individuals or populations.

In spite of these difficulties, it should be underlined that effective dose has nonetheless proven to be successful for risk limitation and for risk management, in particular for occupational exposure situations, where the radiation field may be considered

as a broad beam and the energy fluence incorporates the primary and scatter radiation. An assessment of the energy fluence for a particular geometry requires a quite elaborate measuring design hence, the use Monte Carlo codes are used to simulate and help the understanding the results.

It should be underlined that effective dose enables the summation of doses due to exposures from external and internal exposures and takes account of scientific information on radiation risks. Unsurprisingly, effective dose is the dose quantity used in the majority of countries for purposes of radiation protection.

The ICRP has created an ad hoc group, ICRP Task Group 79, to deal with the generic issue of the use of effective dose as a risk related radiological protection quantity [84]. The group will produce a report to provide guidance on when the quantity ‘effective dose’ can be used and when it should not. The group noted that experience has shown that ‘effective dose’, which has been defined and introduced by ICRP for risk management purposes, i.e. for risk limitation and optimization, is widely used in radiological protection and related fields beyond its original purpose, incorrectly in some cases. The group recalled that useful guidance on restrictions on the use of the quantity is provided by ICRP Committee 2 in annex B to the main recommendations [79]. This guidance needs to be further expanded, and proposals made for the control of exposures and risk management in situations where ‘effective dose’ should not be used. Specific advice on the use of dose coefficients may also be given.

V.5.e Difficulties with the dose equivalent

The use of the term dose equivalent as the operational parallel of equivalent dose is prone to confuse people. Moreover the term is grammatically questionable because ‘dose’, which is a noun, is

used as an adjective; in fact, the term is untranslatable to many languages that shall use the expression ‘equivalent of dose’ rather than ‘dose equivalent’. Furthermore, Q has been superseded by the radiation weighting factor in the definition of the protection quantity equivalent dose, but it is still used in calculating the operational quantity dose equivalent. This cumbersome usage is a substantial cause of confusion. An additional difficulty derive from the fact that no operational quantities have been defined for internal dosimetry, namely there is not a defined quantity for providing a direct assessment of equivalent or effective dose incurred due to internal emitters.

Although it did not play a significant role after this Fukushima reactor accident, the use of the operational quantity dose equivalent was another cause for uncertainty and difficulty because it is easily confused with the quantity equivalent dose, i.e. the same words are used but just in reverse order. The names of these quantities provide semantic problems in many languages including Japanese. The usage is grammatically questionable in English because while equivalent can be used as an adjective or noun, dose is a noun (or verb) and its forced use as an adjective should be done with care (e.g. the expression ‘dose equivalent’ might be more appropriately written as ‘equivalent dose’).

Not surprisingly, the translation of equivalent dose vis-à-vis dose equivalent has been problematic in languages using ideograms such as Japanese. The term dose equivalent is translated to Japanese as, while the term equivalent dose is translated as. Namely, the character for dose, a combination of beam, (here is the short form of, meaning radiation) and amount, is preserved as an adjective in the first case and as a noun in the second. But the term equivalent is translated as (a combination of matching and amount), in the expression dose equivalent; and, as

(a combination of same and value), in the expression equivalent dose. If you are not versed in Japanese, these explanations may be difficult to understand which in itself may provide an example of the difficulties that language translation and inexact word usage might or does have on understanding and communicating.

Fortunately, the operational quantity dose equivalent is used primarily by dosimetrists whereas the protection quantities, equivalent dose and effective dose, are used in communication with the public and non-experts. Thus, this issue is of less importance than others, although use of the same words to define different quantities remains problematic, and it is not entirely uncommon for dose equivalent to be used incorrectly when equivalent dose is the proper term.

V.5.f Difficulties derived from changing names of the protection quantities.

The names used for the protection quantities have evolved over time. ICRP Publication 26 [85] and its amendment issued by the ICRP's 1978 Stockholm statement introduced and defined the quantities 'organ or tissue dose equivalent' and 'effective dose equivalent'. ICRP Publication 60 [35] changed the terms to 'equivalent dose in a tissue or organ' and 'effective dose'. The reason for the change was explained as follows: 'the weighted dose equivalent (a doubly weighted absorbed dose) has previously been called the effective dose equivalent but this name is unnecessarily cumbersome, especially in more complex combinations such as collective committed effective dose equivalent'. ICRP Publication 60 also states that 'the Commission has decided to revert to the earlier name of equivalent dose in a tissue or organ'. However, searching for the name 'equivalent dose' in previous ICRP reports failed to find clear evidence for this statement. For example, in ICRP

Publication 2 [86] the name ‘RBE dose’ was used and in ICRP Publications 6 [80] and 9 [87] the name ‘dose equivalent’ was used.

Therefore, there were over the years a de facto coexistence of two names for the organ or tissue related radiological protection quantities: equivalent dose and dose equivalent. Such perplexing coexistence appears to be due to changes introduced by the ICRP in Publication 60. The coexistence of the two different names for the same quantity has added confusion and misunderstanding within an already complex dosimetric system for radiological protection. In its latest recommendations in ICRP Publication 103 [79], ICRP uses equivalent dose but without the specification ‘in a tissue or organ’. This additional over simplification can add to misunderstanding with effective dose if the quantity is not clearly specified since the units of both quantities are the same.

V.5.g Difficulties derived from the absence of radiation-weighted quantities for high doses

The equivalent dose and the effective dose are defined only for low doses. A radiation-weighted dose quantity applicable to high doses for radiation protection purposes is not available.

The problem created by the lack of a formal quantity for a radiation-weighted dose for high doses was specifically identified at the time of the Tokai-Mura accident in Japan [88]. At the time, a de facto neutron weighted dose had to be created to deal with the situation. The problem was never resolved however and remains unsolved today. Should the doses from the Fukushima Daiichi accident have been very high, this deficiency could have caused problems of dose specification.

Surprisingly and confusedly, the dose limits for tissue reaction effects (formerly termed deterministic effects) for

exposures at higher doses are given in sievert, the units of equivalent dose, effective dose and dose equivalent, usually without explicit specification of the quantity to be used.

The fundamental quantities to be used for quantifying exposure in such situations are organ and tissue absorbed doses (given in gray). If high-LET radiation is also involved, absorbed dose weighted with an appropriate ‘relative biological effectiveness (RBE)’ might be used. Such RBE-weighted absorbed doses are not defined quantities, although they are being used in clinical practice [79]. For the special situation of astronauts, the gray-equivalent (Gy-Eq) is also used [70–73].

The ICRU is studying this issue of iso-effective or equi-effective dose in the context of radiation therapy and the outcome of this study could be of interest in addressing accidental exposures.

V.5.h Difficulties with the units

The same unit, the sievert, is used for all the protection quantities and the operational quantities. Thus, the protection quantities equivalent dose and effective dose, the operational quantities dose equivalent and those derived from it, namely personal dose equivalent and ambient dose equivalent, as well as the recording quantity dose of record, all of them, uses the common unit sievert. Consequently, if the name of the quantity is not specified together with the unit, there could be serious confusion and misunderstanding. A further complication is that the older, pre-SI system of units expressed energy per unit mass in erg per gram rather than joules per kilogram. In that old system the special names given were rad for the unit for absorbed dose and rem for unit for the protection and operational quantities. This system of units is still used in some countries, e.g. in the United States of America.

A major example of the confusion triggered by the use of the unit sievert without stating the quantity was evident in the aftermath of the accident at the Fukushima Daiichi nuclear power plant. The unit was used in reporting of thyroid doses without reference on whether it was equivalent dose or effective dose. The fact is that incorporation of radioactive iodine into the body results in radiation exposure almost exclusively to the thyroid. Usually the equivalent dose is the relevant quantity for reporting organ doses but, if the dose is reported indicating only the unit, it can easily be confused with the effective dose.

There can be a two orders of magnitude difference in the risk to be inferred from the same number of sieverts of equivalent dose versus effective dose. For example, a high effective dose might mask a high equivalent dose to the thyroid. Moreover, since the adult thyroid gland is less sensitive to the carcinogenic effects of radiation than other organs, this 'dose' may or may not be of major health importance unless the dose were incurred by children. As seen, this lack of specificity in using the sievert can be a major source of confusion for decision-makers trying to interpret the potential impact of exposures on workers and the public [93].

There may be purist reasons of theoretical nature to keep the same unit for equivalent dose and effective dose, since the latter is just a weighted average of the first. However, some have proposed a quick fix by creating yet another name for the unit of effective dose. The confusion created by not specifying the dose quantity when giving numerical values in terms of sieverts merits a careful analysis of the possibilities of improving reporting and communication. The practice of not specifying the dose quantity has produced confusion when reporting doses from radioiodine intakes, because whether the number of sieverts reported are of

thyroid equivalent dose or whole body effective dose makes a difference of a factor of about 25 in terms of radiological protection. This is because the tissue weighting factor for thyroid used in the computation of effective dose is 0.04 (i.e. the dose to the thyroid is reduced by a factor of 0.04) [93].

V.5.i Difficulties during accidents

Many problems have been encountered in the use of the current system of radiation protection quantities and units during accidents. Some of them are described hereinafter.

Accidents involving very high doses (The RA2 Accident): A typical radiation accident may involve so high radiation doses that the subtlety of the quantities used for characterizing the exposure become irrelevant in practice. An example is given by an accident in a critical facility [94]. In September 1983 the experimental reactor RA-2, a critical ensemble of variable configuration with fuel elements of 90% enriched uranium, using light water as moderator and reflector and with a 0.1 W nominal power, underwent an accidental criticality excursion involving very high neutron exposures.

These caused the prompt death by over exposure of the operator, and lower doses to the people who were in the commanding room and in the surrounding laboratories. The dosimetric evaluation required using alternative methods because none of the persons involved had their personal dosimeters in place. Measures were done of the induced activity of ^{24}Na in blood and of ^{32}P in hair samples and activation of personal elements like rings, chains, keys, sweaters, etc., and absorbed dose rate after the event. In sum, for estimating the absorbed dose, the evaluation methods included: using the values of ^{24}Na activation in blood, ^{32}P and the characteristics of the critical facility; assessing the thermic, epithermic and rapid component

of the neutrons fluence and applying ad hoc conversion factors to dose equivalent in tissue; measuring the rate of gamma absorbed dose, estimating the components of the gamma dose due to fission products and to prompt radiation, including in this last component, the gamma radiation coming from neutronic captures; modelling an homogeneous estimated cylindric source based on the core's accidental configuration; defining a transference factor between the gamma absorbed dose measured in a certain place of the facility and in a place of the core; etc.

Besides evaluations using experimental measurements calculations used a moderate neutron spectrum with peak energy centered around 1 MeV and extended between energies of 0.01 MeV to 10 MeV. After all this quasi-jugglery with numbers, the total absorbed dose in the whole body was estimated adding gamma and neutron components.

The weighted neutron fraction in total body was estimated to be 22 Gy, while the gamma fraction in total body was assigned to be equal to the maximum dose in trunk, namely 21 Gy, thus total absorbed dose in whole body was estimated to be 43 Gy. This accident is representative case of a situation in which the values of the absorbed dose are so high that it becomes irrelevant the need to convert the physical quantity in a representative magnitude of the biological effects of the accident such as the dose equivalent.

Accidents involving uneven irradiation (The La Plata accident):

Other typical cases for which the current system is not tailored include accidents involving extremely uneven irradiation. A traditional example quoted in the literature [95] refers to an accident (usually referred to as the La Plata accident) involving doses ranging from 0.5 Gy to 34000 Gy! In this accident a

worker carried a ^{137}Cs source belonging to an industrial gamma radiography unit in his trouser pockets for around a day.

While the doses incurred by the hematopoietic and gastrointestinal organs were sufficiently low as to inhibit the occurrence of acute radiation syndrome, some local doses were extremely high. The front of the thighs, the inguinal-scrotal region, and to a less extent the hands, were the areas where the doses were highest and unsurprisingly they were the most affected by lesions. These started with the appearance a few days of wet radiation dermatitis, which gradually grew until they reached approximately the limit represented by the 10 Gy isodose line.

Epithermal desquamation extending approximately as far as the 5 Gy isodose line was observed. With the passage of time there occurred muscular atrophy of both legs and extensive edema of the inguinal-scrotal area. The appearance of extensive femoral hemorrhages made it necessary to amputate first the lower left limb, and then the right. Meanwhile the rate of chromosomal aberrations in peripheral blood confirmed the low doses absorbed by the blood system at about 0.5 Gy.

The analysis of this practical case permits a number of reflections on the difficulties of the current system, for instance: How the overall radiation risk in this individual should be characterized?

Is the dose in peripheral blood a good surrogate of the effective dose? How to consider the contribution to other parts of the body exposed to very high doses?

Accidents involving a high neutron component (The Tokaimura Accident): The 1999 Tokaimura nuclear accident was a criticality accident occurred in an uranium reprocessing facility, resulting

in two deaths [70 and 78], as three workers, were preparing a small batch of fuel for an experimental fast breeder reactor, and using 18.8% enriched uranium.

It was the first batch of fuel for that reactor in three years, and no proper qualification and training requirements appear to have been established to prepare those workers for the job. A precipitation tank reached critical mass when its fill level, containing about 16 kg of uranium, reached about 40 l. Criticality was reached upon the technicians adding a seventh bucket of an aqueous uranyl nitrate solution to the tank. The nuclear fission chain reaction became self-sustaining and began to emit intense gamma and neutron radiation.

At the time of the criticality event, the workers closest to the tank promptly experienced pain, nausea, difficulty breathing, and other symptoms, one losing consciousness later in the decontamination room and began to vomit. The water that promoted the chain reaction served as a neutron moderator. The criticality continued intermittently because as the solution boiled, steam bubbles attenuated moderation from the liquid water due to the created void coefficient and the solution lost criticality; however, the reaction resumed as the solution cooled and the voids disappeared.

Finally volunteer workers permanently stopped the reaction by draining water from a cooling jacket surrounding the precipitation tank. A boric acid solution was then added to the tank to ensure that the contents remained subcritical. These operations exposed 27 workers to a mixed radiation field. These emergency workers incurred relatively low neutron doses but for which not clear weighting factors were defined.

Accidents involving a myriad of issues: The accidents at the Chernobyl NPP in the former USSR and at the Fukushima

Daiichi NPP in Japan presented several issues with quantities and units. For Chernobyl the issues were widely reported in the rich bibliography on this accident (e.g., see [97]). For the Fukushima Daiichi accident The ICRP convened a task group to compile lessons learned from the accident with respect to the ICRP system of radiological protection. The members of the task group have published their findings [93], which included many issues related to the quantities and units that are used for radiological protection purposes [98]. The ICRP group concluded that the radiological protection community has an ethical duty to learn from the lessons of Fukushima and resolve any identified challenges.

Before another large accident occurs, it should be ensured that *interalia* any confusion on protection quantities and units is resolved.

V.5.j Difficulties in radio-diagnosis and radiotherapy

The use of the radiation protection quantities for the protection of patients undertaking radio-diagnostic and radio-therapeutic procedures has been seriously questioned over a number of years. In radiotherapy the quantity of use is the absorbed dose and the derived quantities are not considered to express patient protection. Radiation protection in radiotherapy suffers from the same lack of appropriate quantities as during emergency situations involving high doses.

For radio-diagnosis, specifically, the use of effective dose in medicine has been particularly controversial [99, 100]. It has been recalled [101] that for medical exposures, the effective dose is supposed to be used just for comparing the doses from different diagnostic procedures – and in a few special cases from therapeutic procedure – and for comparing the use of similar technologies and procedures in different hospitals and countries

as well as using different technologies for the same medical examination. For risk estimation from medical exposures, several alternate approaches are being suggested; e.g., following simple adjustment to the nominal risk per unit effective dose to account for age (and sex?) difference [102]; or, replacing effective dose by effective risk in which the weighting factors would be evaluated for tissue-specific lifetime cancer risks per unit equivalent dose.

Thus the use of effective dose in medicine have been seriously questioned and the medical community was advised to use effective-dose base information wisely, realizing that effective dose represents a generic estimate of risk from a given procedure for a generic model of the human body, namely that the effective dose is not the risk for any one individual. Due to the inherent uncertainties involved in its estimations, effective dose should not be used for epidemiologic studies of patients undergoing radio-diagnosis or radiotherapy and for estimating the risks of such cohorts [103].

In a recent article, two authorized ICRP officers addressed the issue of using effective dose in medicine [104]. They recall that the protection quantity 'effective dose' was developed by the ICRP for use in the radiological protection of workers and the public, as a risk-adjusted dosimetric quantity to optimize protection, comparing received or planned doses with constraints, reference levels, and limits expressed in the same quantity. They think that considering exposures incurred during medical procedures, effective dose can be of practical value for comparing: doses from different diagnostic examinations and interventional procedures; the use of similar technologies and procedures in different hospitals and countries; and the use of different technologies for the same medical examination,

provided that the representative patients or patient populations for which the effective doses are derived are similar with regard to age and sex.

The ICRP officers (in their personal capacity) support the judicious use of effective dose as an indicator of possible risk, but caution against the use of effective risk as compared with the calculation of scientific best risk estimates with consideration of associated uncertainties.

In fact, ICRP has clearly stated that “risk assessment for medical diagnosis and treatment... is best evaluated using appropriate risk values for the individual tissues at risk and for the age and sex distribution of the individuals undergoing the medical procedures” [86, 87]. Radiation protection of patients is based on the principles of justification of the medical procedures and optimization of protection, for which the effective dose seems to be suited. Assessment of radiation risks for individuals or groups of patients is not a primary objective of radiological protection.

An important focus of the forthcoming report of ICRP Task Group 79 will be medical exposures [84]. The group has already recognized that the use of ‘effective dose’ for patient exposures is problematic particularly when it is used to assess risk in specific individuals, including children. However, effective dose may be a useful tool for comparisons of, for example, different diagnostic examinations and interventional procedures, the use of different technologies for the same medical examinations, and the use of similar technologies and procedures in different hospitals and countries.

In sum, effective dose will be explained as a dose quantity linked to risk, but intended for the control of exposures for protection purposes and not risk assessment. However, while

radiation doses can be measured and effective doses calculated down to low doses, the associated risk is uncertain and inferred on the basis of assumptions regarding risk projection across populations and dose-response relationships. The plan is to circulate a report for comment during 2015 to be revised at the forthcoming meeting of ICRP.

V.6 Outlook

The system of radiological protection quantities has been used successfully for more than 30 years in controlling occupational exposure and public exposure in normal situations, prospectively in the design of facilities and planning of operations and retrospectively for demonstrating compliance with regulations. However, the use has also demonstrated great difficulties in communicating radiological information to non-specialized experts and to the public. These difficulties in understanding the units and quantities appeared to be a consequence of the complexity of the system which uses more than one quantity and combines physical exposure data with scientific data on radiation risk for organs and tissues.

Although the system and the quantities have shown to be well suited for occupational radiological protection, they are less suited for use in the public domain where communication with non-experts is required, particularly in emergency situations. For instance:

- the differences between the quantities (e.g. effective dose and equivalent dose and absorbed dose) are not well explained and are not well understood even by educated audiences;
- the distinction between the quantities used in the radiological protection system (e.g. equivalent dose and effective dose) and the operational quantities used for radiation measurement (the

dose equivalent quantities, e.g. personal dose equivalent) is even more difficult to understand;

- the use of the same unit (i.e. sievert) for the quantities equivalent dose of an organ and the effective dose over the body, without specifying the quantity, and for the operational quantity dose equivalent, enhances confusion and misunderstanding; and, in sum,
- it is not understood why there are so many different quantities. It is particularly confusing that the different radiation protection quantities have a common unit, the sievert. The problem becomes particularly evident when reporting thyroid doses to workers and the public from intakes of radioactive iodine. The equivalent dose is the relevant quantity for reporting organ doses but, if the dose is reported indicating only the unit, it can easily be confused with effective doses. The effective dose is a risk-related quantity for the whole body and can differ appreciably from the equivalent dose to an organ for the same person.

There are a number of possibilities for improving the situation in the short term. For instance:

- Avoiding the use of equivalent dose without specification of the organ or tissue concerned, e.g. a thyroid equivalent dose;
- Using the shorter and simpler term 'organ dose' for organ equivalent dose in communications, e.g. thyroid dose, which is already usual in many radiological protection practices. Another solution to minimize confusion is to always add the quantity when the unit sievert is being used. Another solution would be to consider renaming the units, but this would require careful deliberation.

One important shortcoming is that the current system does not include simplifying quantities for the sole purpose of public information. Would it be desirable to fill this gap?

Purists working in quantities and units would probably reject the idea. Simplification will always imply a loss in the scientific rigor that is essential in quantification. But, is not rigor already violated in the current system of protection quantities?

In fact, as indicated before, the protection quantities do not comply with the essential requirements for quantities. A further simplification could be welcomed if this will make easier the serious problem of public communication.

A system of public information quantities would be tailored to convey, in a fully and easily understandable and credible manner, radiation effects and risks. This would at least avoid the serious psychological effects that are associated to the misunderstanding of radiation and its quantification. In fact, public distrust is generated when the authorities transmit information in a quantitative manner that is not understandable not only by the public at large but also to many experts.

Perhaps a system like this could include simplified quantities to convey, for instance, the presence of radioactive substances in the environment including its temporal variation. The ideal would be to have few, or even an unique, quantity, summarizing in a simplified manner all the elements currently covered by activity, absorbed doses, weighting factors, temporal variation, etc. Is this possibility really feasible?

It is difficult to answer this question. However, it is clear that it is feasible and desirable to study the possibility to develop a system of quantities for public communication.

The quantities for radiological protection purposes and for measurement purposes are somewhat sophisticated and their application requires professional knowledge. However, radiation protection practitioners are not alone in using these quantities, as

emergency decision-makers—who do not necessarily know the details—rely on them for their choices of intervention and in the receiving end the public claim for simplicity in understanding. Misunderstandings about the quantities in the aftermath of an accident may lead to untoward difficulties, incorrect interpretations of potential consequences and incorrect decisions and after all serious psychological and social detriment for member of the public.

Ways to improve and foster information exchange and education and to develop ‘easy-to-read’ material on the system of radiological protection quantities and units are sorely needed.

Chapter VI

Neutron Primary Standard Metrology

Walsan Wagner Pereira
Sandro Passos Leite

The method of Manganese Sulfate Bath (MSB), developed by researchers who were involved in the building of the nuclear bomb [107], was first developed for the study of secondary emission of neutrons in the fission process. This method was first adapted for use by O'Neal and metrological Sharff-Goldhaber [108], who carried out the measurement of the absolute rate of emission of a neutron source. Physically, an MSB system is a cavity, generally spherical, and contains approximately 0.5 m³ of concentrated solution of manganese sulfate. Currently, MSB is the main method that is used in metrology laboratories to measure the emission rate of radioisotope neutron sources [109].

VI.1 Early years

After the discovery of the neutron in 1932, neutron radiation soon was used as a tool to investigate the nuclear structure. The advantage of this radiation, compared to those that have charge, is that it can easily penetrate the atomic nucleus because it does not need to overcome the strong electric field that surrounds it.

A skilled explorer of this tool was Enrico Fermi (1901-1954). He demonstrated that it was possible to transform atomic nuclei by bombarding them with neutrons; by bombarding uranium, the heaviest of the natural elements, he expected to obtain transuranic elements, which would complete the periodic table of chemical elements.

Also working on experiments related to the bombardment of nuclei by neutrons, Hahn (1879-1968) and Strassmann (1902-1980) published a study in which they concluded that the bombing of uranium nuclei by neutrons gave rise to the formation of nuclei of mass intermediate. Immediately after this work, Meitner (1878-1968) and Frisch (1904-1979) reported the absorption of neutrons leaving the nuclei of unstable uranium, causing them to split into two fragments of roughly equal mass. In analogy to the process of cell division, they dubbed this phenomenon nuclear fission. After the discovery of fission, Fermi was the first to consider that fission fragments could induce more neutrons. One consequence of this induction would be the production of a chain reaction [110]. The first months of 1939 involved an intense search for evidence on the issue of secondary neutrons that could occur from highly excited fission fragments or at the time of fission.

Evidence of the secondary emission of neutrons [111] came from an experiment that consisted of placing a neutron source at the center of a large cylindrical tank [90 × 90 cm²] full of water, which was placed on packed uranium oxide rods. The finding of the possible increase of neutrons was generated by comparing the number of neutrons that were present in the tank dichotomous situations: one when only the source was present and the other when all source-uranium oxide was in the tank.

The number of neutrons that were present in the water was determined by measuring activation foils (rhodium foils) that were placed at different positions inside the tank. Activation foils were used to determine the density of neutrons as a function of distance, and this information was used to calculate the number of neutrons that were present in the water. This measurement method depends on the neutrons being moderated into a volume

with spherical symmetry [111]. A 5% increase was observed in the number of neutrons when uranium oxide was present in the tank.

However, the authors did not immediately accept that this increase was due to the presence of uranium oxide. A distrust of results emerged because the amount of neutrons that were emitted was enough to produce nuclear reactions of the type $(n, 2n)$ energy. To avoid this kind of reaction, they set up a source of RaBe (γ, n). This source consisted of a block of beryllium with a gram of radium.

Due to the larger size of the source, the assumption that the neutron moderated through water would have a spherical symmetry was not true, making the results that were obtained for the number of neutrons present in the tank in the two situations mentioned above inconclusive. Thus, it was not possible to verify the increase (or not) in the number of neutrons.

To overcome the dependence of moderate neutrons with respect to the spherical symmetry [112], the USA proposed an aqueous solution not only to moderate neutrons but also to activate them. The authors used an aqueous solution containing 10% Manganese Sulfate (MnSO_4). Manganese is the target element, and its activation, as induced by neutron capture activity, is proportional to the number of thermal neutrons that are present in the solution.

A calculation was performed to determine the number of thermal neutrons in the tank substituted by the measurement of the count of the solution after being homogenized. In this work, we not only developed the Method of the Manganese Sulfate Bath (originally called the "Method of Physical Integration") [108] but we also reported the first steps in a sustained chain

reaction. This allowed, in subsequent years, the construction of reactors and nuclear bombs.

VI.2. Middle years

The paper from Szilard, Fermi and Anderson reporting the manganese sulfate bath method was published on August 1, 1939, and on September 1, Germany invaded Poland, starting the Second World War. During this terrible period of human history, scientific communication was markedly impaired. Related to neutron research, articles were retained by the editors, but during this period, many advances were made but were only published after 1945. Some papers that brought us to the present development are cited here.

One of these studies was received in 1942 but not published until 1946 [108]. These authors were the first to develop a technique to absolutely measure the emission rate of neutron sources using the manganese sulfate bath. The technique for the measuring neutron emission rate (Q) was developed by these authors in the following steps: the neutron source to be measured was placed in the center of a tank with aqueous MnSO_4 until activity induced in the solution by the neutrons reached equilibrium.

Then, the solution was stirred, and counting was performed using a Geiger detector with thin walls immersed in it. Another irradiation was performed, but an absorber of neutrons was inserted into the solution. After again reaching a steady activity of the solution and withdrawing the absorber, the count of the solution was performed again. The emission rate of the neutron source was finally determined by the expression $N = N_a / (1-R)$, where N is the emission rate, N_a is the number of neutrons captured by the absorber, and R is the ratio of counting solution measured with and without the presence of absorber. The RaBe

(γ , n) source calibrated in this article had its emission rate determined with an uncertainty of 9.3%.

Others studies brought significant development to this technique or simply related the work conditions [94–103].

Since the first publication, this technique has changed due to the development of radiation detectors; currently, even the Monte Carlo simulation of some physical aspects of the manganese sulfate bath plays an important role in this accuracy and uncertainties of approximately 0.7% for evaluating neutron source emission. One of the most recently published international comparison shows results from eight national metrology institutes that have traceability and have been disseminated around the world.

VI.3. Determination of the neutron emission rate by manganese sulfate bath

The method of the manganese sulfate bath currently has two principal techniques: static and circulating ones. The difference between these two techniques is related to solution activity measurement.

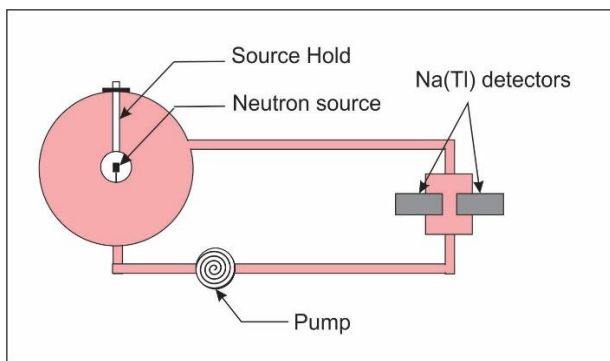


Figure VI.1 - Scheme of a circulating manganese sulfate bath.

In the circulating technique, figure VI.1, the solution flows continuously through tubes in contact with detectors, usually two, placed in a container external to the bath. In this system, both the growth and decay of activity in solution inducing the neutron source can be monitored.

In static one, figure VI.2, only the decay of the solution can be monitored because measurement is performed only after the removal of the source and detector immersion.

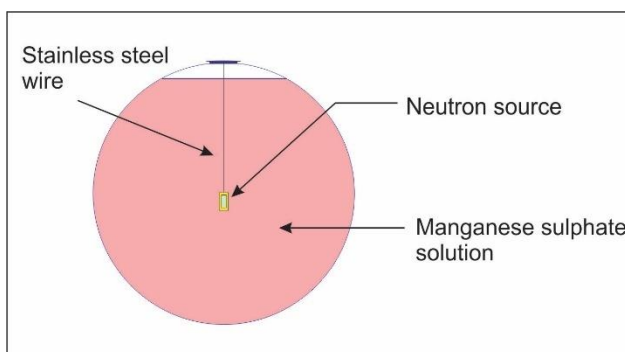


Figure VI.2 - Scheme of a static manganese sulfate bath

In this chapter, we present a mathematical model of the static technique that was used by three of the last participants for an international comparison [122]. This is the technique that was used by a neutron lab in Brazil.

Over the past 39 years, the Laboratory of Neutron Metrology of Brazil (LN) has measured the emission rate from neutron sources (Q) using the manganese sulfate bath method. However, in 1996, with the installation of the manganese sulfate bath system donated by the BIPM and through the cooperation program between the Bureau International de Poids et Mesure (BIPM), the National Institute of Metrology, Standardization and Industrial Quality (INMETRO) and the National Metrology

Laboratory of Ionizing Radiation (LNMRI), it became possible to establish the Laboratory of Neutron Metrology as a national reference.

Currently, the LN is responsible for the custody and maintenance of the Brazilian Standard Fluence Neutron Source ($^{241}\text{AmBe}$ (α, n) 3,7GBq) and determining the neutron fluence quantity. The LN/LNMRI also promotes the spread of this quantity for neutron measuring instruments that are used in industry, research centers, hospitals and universities, ensuring their traceability. Along with seven other laboratories, the LN takes part in Comité des Consultatif Rayonnements Ionisants (CCRI), Section III (Neutron), organized by the BIPM, where interlaboratory comparisons past and future are discussed related to neutron metrology. Figure VI.3 shows the Brazilian manganese bath.

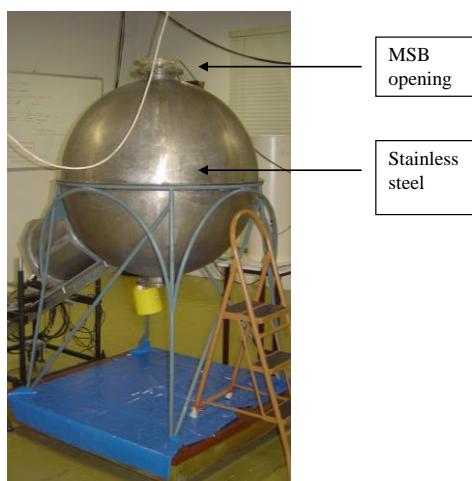


Figure VI.3 - LN/LNMRI absolute primary standard system for neutron sources

One simple mathematical model related to emission rate in manganese bath measurement can be stated as in VI.1.

$$Q(A, K, \varepsilon, F) = \frac{AK}{\varepsilon f} \quad \text{VI.1}$$

where A is the rate counting in reference time; K is the corrections due to neutron loss due to leakage from solution, neutrons captured by nuclides while moderating and those neutrons that come out from source material and subsequently are captured by source encapsulation; and f is the thermal neutron fraction captured by manganese to the other nuclides and ε is detection system efficiency.

Thus, the relative combined standard uncertainty for emission rate is given in VI.2.

$$\frac{u_Q^2}{Q^2} = \frac{u_A^2}{A^2} + \frac{u_K^2}{K^2} + \frac{u_\varepsilon^2}{\varepsilon^2} + \frac{u_F^2}{f^2} \quad \text{VI.2}$$

VI.3.a Calculating A

The experimental parameter that determines the emission rate of a neutron source is the counting of the ^{56}Mn -produced radionuclide reaction with neutron inside manganese bath (C_{ij}). This counting should be corrected for saturation (A) and then be used to calculate the emission rate of a neutron source. The saturation activity counting value is determined by VI.3. The saturation activity in solution is the moment at which the ^{56}Mn atom number being produced is equal to the ^{56}Mn atom number that is decaying. The saturation activity is reached asymptotically after 24 hours. After this time, the source is removed from the bath, and the ^{56}Mn decay process can be measured using an NaI(Tl) detector. The physical and mathematical decay measured model is given by VI.3.

$$A(C_{ij}, \lambda_s, \lambda_m, \tau_c, T_{is}, T_{ij}, T_i) = \frac{C_{ij} \tau_c e^{\lambda_s T_{is}} e^{\lambda_m T_{ij}} (\lambda_m - \lambda_s)}{(1 - e^{-\lambda_m \tau_c}) (e^{-\lambda_s T_{is}} - e^{-\lambda_m T_{ij}})} \quad \text{VI.3}$$

where C_{ij} is the count rate measured during the time interval τ_c after corrections for dead time and background radiation; λ_s is the constant decay of the neutron source; λ_m is the constant decay of ^{56}Mn ; τ_c is the duration of counting; T_{is} is the time interval between the reference date and the inclusion of the source in the bath; T_{ij} is the time interval between the removal of the source from the bath and the early count j-th; and T_i is the time interval in which the source remains in the MSB.

VI.3.b Calculating f

In addition to neutron interactions with manganese, other interaction processes occur for neutrons that are emitted by the source put inside the bath, which should be determined for accurate $Q(t)$ determination. One process is thermal neutron capture in solution by hydrogen, sulfur and oxygen atoms; thus, we need to correct for thermal neutrons that are not captured by manganese. The F value is determined by VI.4.

$$f = \frac{\sigma_{Mn}(1+a)}{\sigma_{Mn}(1+a) + \sigma_s + \sigma_o + \frac{N_H}{N_{Mn}} \sigma_H} \quad \text{VI.4}$$

where N_{MN} and N_H are the number of manganese and hydrogen atoms per cubic centimeter of solution, respectively, and σ_{Mn} , σ_H , σ_s and σ_o are the microscopic cross sections for thermal neutron capture by manganese, hydrogen, sulfur and oxygen, respectively. The factor $(1 + a)$ corresponds to the correction factor due to resonances in the manganese cross section, where the value of a is given by $(33N_H / N_{MN} + 0.8)$. Observing the expression of F , this quantity depends mainly on the ratio N_H/N_{MN} . This parameter is calculated considering the water concentration and

manganese sulfate in the solution. Concentrations are usually determined by gravimetric method.

VI.3.c Calculating K

Other interaction processes that compete with neutron capture by manganese atoms are collected in parameter K, which is determined by VI.5. This parameter is currently well determined by Monte Carlo simulation.

$$K = \frac{1}{1 - N - S - L} \quad \text{VI.5}$$

where the N, S, and L components are corrections due to:

N (nuclide): Fast neutrons captured in solution. This component is important when the neutron energy that is emitted by the source is greater than 2 MeV. Fast neutrons are captured by oxygen and sulfur nuclei through reaction types (n, α) and (n, p).

S (source): This component refers to the neutrons that are scattered by the solution and are possibly captured by the source material.

L (leakage): Neutron Escape from bath. L depends mainly on the size of the bath and the neutron energy that is emitted by the source.

VI.3.d Calculating ε

After correcting for F and K parameters, one still needs to determine the efficiency or sensibility of the detection system to finally determine the neutron source emission rate. The efficiency (ε) is obtained by counting a sample solution with standardized activity; its value is obtained by the ratio between the count rate of a known mass sample that is inserted in the bath ($N(C_{ij}, \lambda_m, \tau_c, T_0)$) and the determined absolute value of a mass

specific activity of same sample solution ($A(t)_a$) considering the same reference time (VI.6 to VI.8).

$$\varepsilon = \frac{N(C_{ij}, \lambda_m, \tau_c, T_0)}{A(t)_a} \quad \text{VI.6}$$

where

$$N(C_{ij}, \lambda_m, \tau_c, T_0) = \frac{C_{ij} \tau_c \lambda_m e^{\lambda_m T_0}}{(1 - e^{-\lambda_m \tau_c})} \quad \text{VI.7}$$

and

$$A(t)_a = m_a \cdot A(t)_{conc} \quad \text{VI.8}$$

where $(t)_{conc}$ is the concentration of the standard activity of the irradiated sample solution. This irradiation is generally performed in a nuclear reactor facility (^{56}Mn activity per mass of solution MnSO_4), and m_a is the irradiated mass that is inserted in the bath solution.

Generally, neutron metrology laboratories use a nuclear reactor to irradiate the solution sample and a coinciding system as $4\pi\beta-\gamma$ to standardize the activity from the irradiated sample.

It should be remarked that the calculation of any source rate emission must be followed from the uncertainty estimation as given [123].

The total process to obtain a neutron source emission rate is summarized in the diagram from figure VI.4.

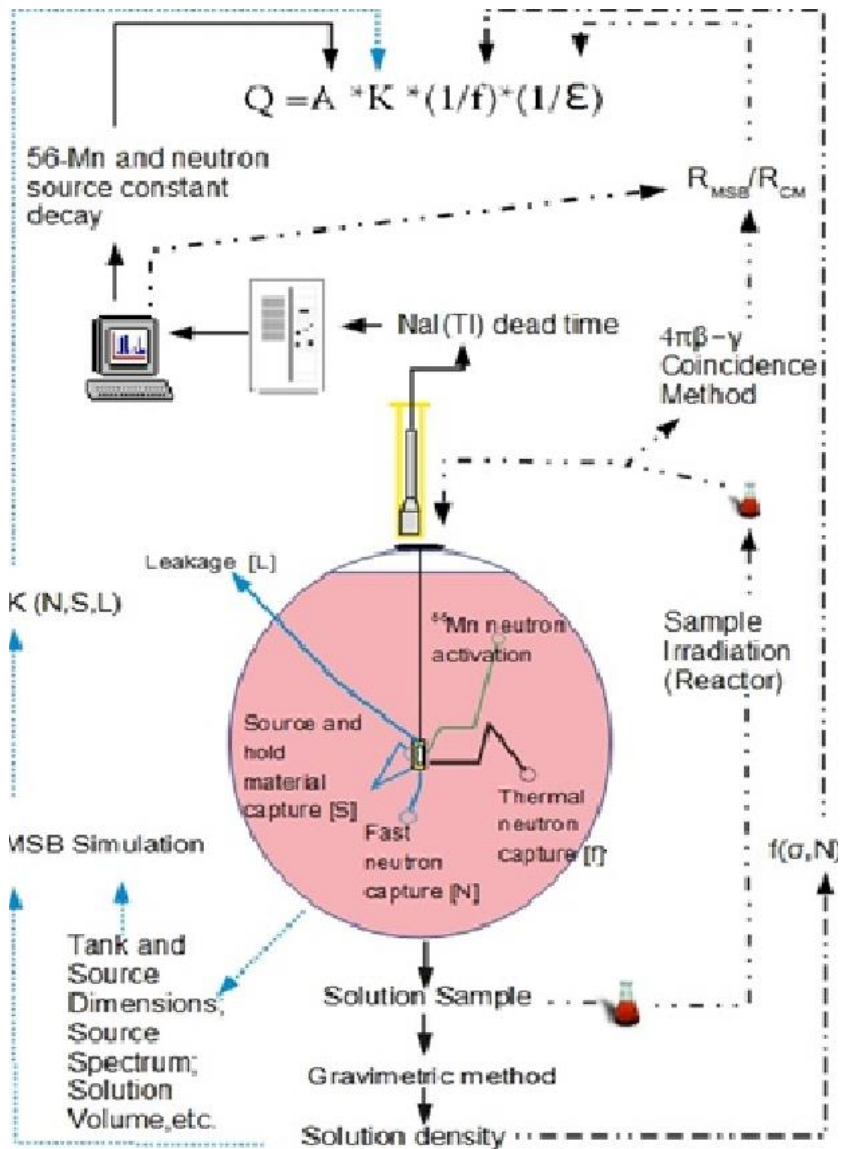


Figure VI.4 – Diagram for absolute neutron emission measurement based on manganese bath.

Chapter VII

Primary Standardization in Radionuclide Metrology

José Ubiratan Delgado

Understood as the area of scientific knowledge that studies the different nuclides with single decay modes, radionuclide metrology develops methods and techniques, including system calibrations with laboratory instrumentation, and can conduct measurements for the Activity quantity and is correlated with the highest possible level of metrological quality.

Under conditions of the physical quantity of a radioactive source, containing a radionuclide in a particular energy state and in a moment was defined by NCRP as "*the expected value at that time of the number of spontaneous nuclear transitions in unit time, that energy state*" [124].

The unit of Activity is *becquerel* (Bq), which corresponds to one disintegration per second, as proposed by the international system of units (SI) in 1975. The value of the Activity of a sample can also be registered by multiples or submultiples of Bq. The old unit that is still preserved in some recent manufacturing equipment and in several countries is the *Curie* (Ci), which, by definition, is equal to the number of transformations per second in one gram of ^{226}Ra ($1 \text{ g} = 3.7 \cdot 10^{10} \text{ dps}$). Thus, 1 Ci is equal to $3.7 \cdot 10^{10}$ Bq, and even though outside of SI this unit is still quite widespread, its use is tolerated by the General Conference on Weights and Measures only for nuclear medicine practices [125].

A calibration laboratory that produces radioactive standards in various shapes and geometries and whose activity is certified with reduced uncertainty must have several measurement methods to ensure robustness on values that are practiced and provided to users. Measurement systems, following the model of other physical quantities, can be classified as primary (absolute) or secondary (relative).

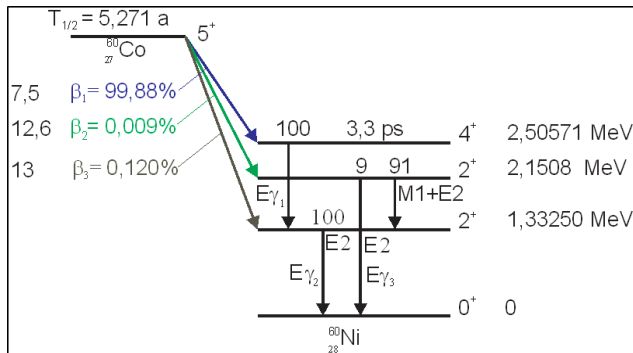
However, due to the diversity of radioactive particles and by the fact that the sources of radiation decay over time, there is no permanent standard for radioactivity. Therefore, the primary references are defined by combining specific instrumentation and measurement methods for each type of radionuclide, according to nuclear parameters that are associated with its decay scheme.

Thus, a given radionuclide should only be accepted as standard when the Activity is well established and the decay parameters are determined consistently. However, the same radionuclide must be submitted to inter-laboratory comparison programs with results within appropriate limits for accuracy and precision, and the instrumentation and methodologies that are adopted must be traceable to the international reference system (SIR) within the framework of the BIPM.

VII.1 Primary methods for radionuclide standardization

These methods are characterized by feasibility of determining the radioactive source Activity without the need for comparison with a standard or knowledge *a priori* of the detection system efficiency. The technical literature uniquely identifies the primary standardization for absolute or direct measurement methods so that the certified radioactive standards that originate in these methods are called primary standards. For the implementation of an absolute or direct method, in order to

obtain the Activity of a given radionuclide that decays by the emission of two or more radiations, only knowledge about the decay scheme is required, as seen in the example below for the decay of ^{60}Co [126, 127]. As a result, this Activity can be determined "directly" through the counting rates recorded by the absolute measurement system.

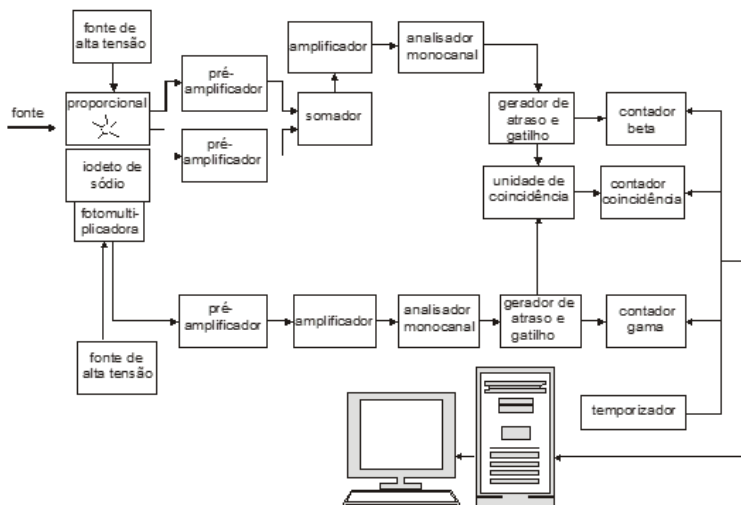


Radionuclide Metrology offers six absolute methods for primary standardization, namely Coincidence, Anticoincidence, Counting 4π , Counting on defined solid angle, Liquid scintillation counting (CIEMAT/NIST) and Peak-sum coincidence counting.

VII.2 $4\pi\beta\text{-}\gamma$ Coincidence counting method

The technique of counting with electronic signals in coincidence began to be used as a method of standardization in 1940. This technique had as goals determining the Activity of radioactive sources or obtaining the detector efficiency using two detectors suitable for each type of emission from a point source. It becomes advantageous when the decay scheme of the radionuclide to be standardized is an emitter of radiation α , β or x , immediately followed by the emission of gamma radiation.

However, it can be used for coincidences with electron capture- γ , γ - γ , x- γ , etc. A system of standardization in coincidence (see figure below) typically consists of a proportional detector for the counting of β events and a detector of NaI (TI) for the counting of γ events.



Counting in coincidence $4\pi\beta$ - γ , being the simplest system, supports the measurement due to one or more types of radiation with high efficiency in a 4π detector, whose geometric efficiency is equal to unity, called a beta detector, and registering counting N_β , with appropriate corrections for background, live time and decay. Simultaneously, the counts due to another type of radiation are measured in a second detector, called a gamma detector, which registers photons with N_γ count rates without the need for a geometric efficiency of 100 %. The gamma detector counting that is coincident in a given time interval with those of the beta detector is recorded as N_c .

When one considers the system of anticoincidence, $N_a = N_\gamma - N_c$ is recorded. The reason N_γ/N_c is called the β efficiency parameter is that, in practice, N_β , N_γ and N_c or N_a are registered for various beta efficiencies, and the activity is estimated by extrapolating N_β for $\varepsilon_\beta = 1$. This extrapolation is typically set to a polynomial function. The detector output signals are processed and recorded by electronic channels and are composed, in addition, of a high voltage unit, preamplifier, amplifier, pulse height analyzer, digital counter and multichannel analyzer [128].

In this simple case of β - γ decay, because the γ radiation is immediately issued in relation to β radiation, the count of coincidences between them indicates disintegration. The beta detector is sensitive only to β and the γ detector only to gamma radiation. For the point source of Activity per mass unit N_0 , detection efficiency ε_β and ε_γ , and count rates of β , γ and coincidence channels, we have:

$$N_\beta = N_0 \cdot \varepsilon_\beta; N_\gamma = N_0 \cdot \varepsilon_\gamma; N_c = N_0 \cdot \varepsilon_\beta \cdot \varepsilon_\gamma \text{ VII.1}$$

$$N_0 = \frac{N_\beta \cdot N_\gamma}{N_c} = \text{Activity} \text{ VII.2}$$

The ε_β and ε_γ efficiencies can be obtained by the counting rates in each channel, whereas the source is extensive, and the proportional detector is sensitive to all sources:

$$\varepsilon_\beta = \frac{N_c}{N_\gamma} \text{ and } \varepsilon_\gamma = \frac{N_c}{N_\beta} \text{ VII.3}$$

In a system for radioactivity measurement, however, some phenomena require corrections, such as resolution time; accidental coincidences; gamma sensitivity of the

proportional detector; and dead time. These corrections are introduced in the calculations, and detailed formulas are described in the literature, including for other cases of emissions, according to the decay scheme of nuclides.

VII.3 Anticoincidence $4\pi\beta\text{-}\gamma$

Although it is recognized as a well-established method, the coincidence counting method presents experimental difficulties, especially when dealing with nuclides that have metastable or isomeric nuclear states. Also in measurements with high counting rates, it is essential to make corrections for dead time and accidental coincidences. The anticoincidence method, on the other hand, was proposed by Bryant as complementary to that of coincidence to standardize simultaneous $\beta\text{-}\gamma$ emitters and, in 1967, was proposed to metastable emitters with delayed γ radiation [129].

In enabling the standardization of radionuclides that have simple or complex decay scheme, the anticoincidence method still extends its application to β -pure emitters that make use of the tracer technique. The accidental coincidences that occur in the method of anticoincidence, according to Bryant, are circumvented by registering the counts in each detector separately and the counts in anticoincidence in an electronic unit taking into account a detector with the other. Indeed, there is no basic difference between coincidence and anticoincidence methods.

The difference lies in the way to obtain the count of gamma pulses that are not correlated. In terms of a measurement system in a laboratory for primary

standardization, the anticoincidence method is complementary to the coincidence because it proposes the counting of γ pulses that are not correlated. The count rates in the three channels are represented by the following equations:

$$N_{\beta} = N_0 \sum \rho_r \left[\varepsilon_{\beta r} + (1 - \varepsilon_{\beta r}) \left(\frac{\alpha \cdot \varepsilon_{ce} + \varepsilon_{\beta \gamma}}{1 + \alpha} \right)_r \right] \quad \text{VII.4}$$

$$N_{\gamma} = N_0 \sum \rho_r \left(\frac{\varepsilon_{\gamma r}}{1 + \alpha_r} \right) \quad \text{VII.5}$$

$$N_{nC} = N_0 \sum \rho_r \left[(1 - \varepsilon_{\beta r}) \left(\frac{\varepsilon_{\gamma r}}{1 + \alpha_r} \right) \right] \quad \text{VII.6}$$

where: N_{β} , N_{γ} and N_{nC} are the β , γ and anticoincidence channel count rates, respectively, corrected for background, decay, accidental coincidences and dead time losses; N_0 is the activity of the source that is being measured; $\varepsilon_{\beta r}$ and $\varepsilon_{\gamma r}$ are the efficiencies of the β and γ detectors for branch r of the decay; ρ_r is the emission probability of branch r ; A is the total internal conversion coefficient; α_r is the total internal conversion coefficient of the γ -rays that are associated with branch β_r ; ε_{ce} is the efficiency of β detector to conversion electrons associated with the branch r ; and $\varepsilon_{\beta \gamma}$ is the efficiency of the detector β to γ -rays that are associated with branch β_r .

The anticoincidence system can be described as a coincidence system that is modified to count only no coincidences. Thus, for a radionuclide with β - γ decay, subtract the no coincidence counts N_{nC} of the gamma counts range N_{γ} to obtain the counts in coincidence. As seen in the equation below:

$$N_0 = \frac{N_{\beta} \cdot N_{\gamma}}{N_{\gamma} - N_{nC}} \quad \text{VII.7}$$

Then, the total count N_0 can be determined by following all of the theoretical principles that were used for the coincidence method, as previously seen.

To solve the problem of count losses, Baerg et al. [130] incorporated into the anticoincidence system a live countdown using a device with an extendible dead time [131]. Then, live time is characterized by the time that the system is free to perform a count, in contrast to the dead time, which is the time that the system is busy and cannot perform a pulse count. Therefore, the live time is the time that the system is able to register a count and is measured for β , γ and anticoincidence channels. The live time T_v is described by the equation below.

$$T_v = \left(\frac{F}{F_{ref}} \right) - C \cdot d \quad \text{VII.8}$$

where, F_{ref} is the reference frequency; F is the number of pulses that were counted from the oscillator while the channel is active, representing the time interval in which the system is not paralyzed by the dead time; C is the count over the sample period; d is the width of pulse oscillator; and $C \cdot d$ is the correction due to the pulse width generated by MTR2.

Therefore, the actual count rate N for the β , γ and anticoincidence channels will be:

$$N = \frac{C}{T_v} \quad \text{VII.9}$$

In this way, the count rate N_0 for a nuclide can be determined by the classical equation of the coincidence method, adapted to the anticoincidence method. The equation of N_0 is valid only if the real events in anticoincidence are counted. In radionuclide metrology, the dead-time correction is crucial. Therefore, instead of using idealized correction formulas for a standard

measurement system, electronic modules were developed by LNHB [132] in order to solve this problem by instrumentation.

VII.4 Counting 4π Method

Measurement for emission of charged particles: The activity can be determined by depositing the radioactive source on fine support, positioned inside a windowless detector in order to establish a solid angle of 4π steradians. In this configuration, the counting efficiency can reach 100 %, and there is no need to make corrections for losses due to absorption or scattering. In the use of gas proportional counters, the efficiency is also extremely high for performing measurements of radiation α or β .

Measurement for γ emitters: For radionuclides that have γ transitions of high intensity, the counting in a NaI(Tl) well-type can also achieve efficiencies close to 100 %, with reduced uncertainties. Under this condition, the $4\pi\gamma$ counting becomes more simple and feasible than the $4\pi\beta$ - γ coincidence technique, including low levels of uncertainty. However, the accuracy of this method is limited to the radionuclides that present direct transition from the nucleus to the ground state due to the uncertainties in the emission probabilities for the other transitions.

VII.5 Defined Solid Angle Counting Method

The counts to measure emitters α , β or photons (x and γ) are held at geometry of defined solid angle, where the value of the angle is less than or equal to 1π steradians. Along with the coincidence counting, this method provides an efficient alternative to the primary standardization of Activity because it aims to count all of the radiation that is emitted by a radioactive

source in any angle of emission. As in the case of measurement with gas counters or in liquid scintillation, the constituent material of the source is part of the sensitive volume of the detector. Usually, this material is deposited on a very thin support and is wrapped by the detector to form a 4π count geometry.

This methodology requires corrections for the fraction of the decay that was not detected, in general, due to particles or photons that are absorbed by source support in small emission angles. The observed count rates must be corrected by the geometric factor $\Omega / 4\pi$, which includes exact distances and mathematical modeling to the solid angle subtended. A strict control of the detection system geometry, as in, for example, the detailed design of mechanical device, is a single condition for successful measurements.

VII.6 Liquid scintillator counting method (CIEMAT/NIST)

Absolute standardization can also be determined by the CIEMAT/NIST method, which is used to determine the Activity of radionuclides that decay by the emission of particles β , β - γ , ec - γ or x . This method was developed in the '80s and is based on the model parameter free or figure of merit, requires exact calculations for the Spectra of the absorbed electron energy by scintillator and is widely used in several laboratories of radionuclide metrology worldwide.

The detection system makes use of the liquid scintillator as a radiation detector, which transforms in light the wavelength ultraviolet [133], using two photomultiplier tubes in independent channels but operating in coincidence to transform light into electrical impulses. Once the pulses have been processed by the

system, data would be generated that could be used to obtain the sample Activity of the radionuclide. By means of the measurement of a set of samples of a standard of tritium, ^3H , when used in the form of a tracer forms an efficiency curve that characterizes the experimental condition [126].

VII.7 Sum-peak Method

The peak-sum coincidence count is a very simple method that uses a single detector of HPGe or NaI (Tl). This method was initially proposed by Brinkman and, as an absolute method, can be used for standardization of some types of radionuclides, unlike the general technique of spectrometry of photons, which demands the use of an efficiency curve by means of standardized sources. Its most successful application is made with point sources measured on geometry of almost 100 %, as the control of radionuclidic impurities is assured in the sample. It applies those radionuclides that decay with emission of at least two photons in coincidence in the transition into the ground state or β emitters, followed by an x-ray or gamma, such as ^{65}Zn and ^{51}Cr [134]. Thus, for the count in coincidence by peak-sum method, there is a need to use a photon radiation detector, in which intrinsic and geometric efficiencies should be high. An example application is given for ^{60}Co because this brings the advantage of owning two gamma lines in cascade, with almost 100% of intensity to its main energies. The "peak-sum" occurs in $E_1 + E_2$.

Then, the equation is:

$$A = N_T + (N_1 \cdot N_2) / N_p \quad \text{VII.10}$$

Where N_T is the count rate for the total spectrum, and N_1 , N_2 and N_p are the corresponding count rates to peak in energies E_1 , E_2 and E_p , respectively. These counts are directly related to the

activity of the source at photo peak and overall efficiencies, respectively, ε_1 , η_1 and ε_2 , η_2 , which represent energies E_1 and E_2 . This equation applies to ^{60}Co in the case of using the model of simplified decay, in which its main lines have almost 100 % of intensities. Then:

$$n_T = A(\eta_1 + \eta_2 - \eta_1 \cdot \eta_2) \quad \text{VII.11}$$

where: $n_1 = A \cdot \varepsilon_1(1 - \eta_2)$; $n_2 = A \cdot \varepsilon_2(1 - \eta_1)$ and $n_P = A \cdot \varepsilon_1 \cdot \varepsilon_2$

The previous equations should consider the correlation factors, ω_P and ω_T , which correspond to the partial and total energy deposition. Then:

$$n_1 = A \cdot \varepsilon_1(1 - \omega_T \cdot \eta_2); \quad n_2 = A \cdot \varepsilon_2(1 - \omega_T \cdot \eta_1) \quad \text{VII.12}$$

$$n_P = A \cdot \omega_P \cdot \varepsilon_1 \cdot \varepsilon_2; \quad \text{and} \quad n_T = A(\eta_1 + \eta_2 - \omega_T \cdot \eta_1 \cdot \eta_2) \quad \text{VII.13}$$

It is necessary to use radioactive sources of low activity in order to avoid dead time problems. As occurs for any absolute method, the identification and quantification of radionuclidic impurities should be taken into account. The uncertainty estimates depend on the determination of the peak area and of the correlated emission intensities.

VII.8 LNMRI Absolute Standardization

The primary methods for radionuclidic standardization can generate a basis for Activity quantity measurement, which is useful in radio analytical and nuclear chemistry, with applications in industry, medicine, research and environmental areas. Such measurements lead in practice to the development and dissemination of radioactive standards, which are initially produced under auspices of a national laboratory of metrology, as the LNMRI (IRD/CNEN) in Brazil. Absolute standardization tasks conducted by the LNMRI are characterized:

- by the validation of all of the available primary standards, which are supported and confirmed by one or more additional measurement methods. This is done to ensure that the result of standardization is confirmed by additional measurements and entirely self-contained;
- by binding the new standardization to previous standardization through comparisons with standard solutions available in stock or through the use of calibration factors for secondary standardization instruments, with uncertainty far below 1 %, except for nuclides of complex decay;
- for the verification of the consistency of the measurements at the international level, as demonstrated by comparison programs under the auspices of BIPM and metrological regions or comparisons with other national laboratories, such as PTB, NIST, LNHB, and NPL; and
- for research projects on the primary standardization of radionuclides, which were developed by the LNMRI and include the ability to perform absolute standardization for different radionuclides and dissemination by means of calibration and transfer standards.

Since 1980, more than 50 radionuclides have been standardized in the LNMRI and are available in the form of stable solutions or point sources for different nuclides emitting α , β , γ and x, as certificated by the total Activity or Activity per unit mass. The tasks in the lab are conducted for approximately 20 among the systems, processes or methods of measurement, supporting the provision to users of certified radioactive sources and the development of new standards.

In addition, inter-comparison programs are held to assess nuclear medicine services in the country using nuclides in the form of radiopharmaceuticals administered to patients. From 1998, the LNMRI has developed with the nuclear medicine services, a comparison program of Activity measurements for

radiopharmaceuticals applied to patients for diagnosis or therapy. The literature demonstrates that the radiopharmaceutical, before being applied to the patient, is not always measured properly. The causes are due to operating errors or malfunctions of activimeters and may result in excessive doses or even repeating exams due to low image quality. The goal of the program is to evaluate the quality performance of routine measurements made in such a way that the doses that are administered to the patient have correct activities, keeping their exposure to radiation as low as possible [135].

Moreover, LNMRI has developed a program to perform quality control and assess performance measurements of the radiopharmaceuticals generated in the main producing centers in the country.

Finally, since 1991, as an assignment inherent to a national institute of metrology, the IRD has been developing, promoting, coordinating and maintaining the Inter-comparison National Program in Radionuclide Analysis for environmental samples, known as PNI. The purpose is to evaluate the analytical capacity of Brazilian laboratories that monitor the occurrence of environmental contamination by radioactive materials and to determine the radiological conditions around a nuclear or radioactive facility.

Chapter VIII

X and gamma rays, electrons and charged particle examples and exercises

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VIII.1. Introduction

In recent years, many efforts have been made by the scientific community to present a formal and validated scheme [40] for metrological issues for ionizing radiation applications. These efforts have resulted in several publications that define the quantities, procedures and methods for data treatment and uncertainty estimation [117 and 118]. However, these formal approaches have a final aim of the adequate assessment of doses in medical, industrial and other applications of ionizing radiation.

A formal definition of the quantities, calibration methods and laboratorial frameworks is essential for assuring the quality of applications depending on the choice on a radiotherapy treatment or the warrantee that patient doses on a imaging facility are as low as possible, while producing images with the best diagnostic quality. For these and other reasons, modern society radiation dosimetry and radiation dosimetry require a qualified metrology scheme.

However, the application of the concepts and formalisms that have been proposed regarding technical documentation are not

simple and need attention for correct assumptions and estimations. This topic is usually not adequately addressed in the academic environment, and there are few opportunities for radiation users to specialize in this area. Therefore, this chapter tries to emphasize, using examples and exercises, the constraints, interpretations and limits of implementation of the formal approaches using practical situations. The chapter is far from exhaustive of all of the different possibilities of these applications, and the provided data are included only for illustrative purposes.

VIII.2. Gamma rays, electrons and charged particles

VIII.2.a. Practical examples

i) How to determine the calibration coefficient of a thimble ion chamber, for Ir-192 sources [136–140].

Background: Due to the complex spectrum of the Ir-192 sources, until now, there has not been a primary standard for the determination of the reference air kerma of this kind of source. The methodology that was adopted for the dosimetry of Ir-192 sources is indirect, in which a calibrated thimble is used to calculate the calibration coefficients for 250 kV and Co-60, which are interpolated for the Ir-192 energy [143].

Considering:

$$D_{air} = \frac{M}{m_{air}} \left(\frac{w}{e} \right)_{air} \quad \text{VIII.1}$$

Where D_{air} is the Air dose (Gy); M is the Electric current that is produced in the ion chamber (C); m_{air} is the dry air mass (kg); and $(w/e)_{air}$ is the mean energy per ion pair (J/C).

According to Di Prinzio [140], the thimble ion chamber calibration coefficient (Nk) can be determined by:

$$N_k = \frac{K_R}{M} \longrightarrow M = \frac{K_R}{N_k} \quad \text{VIII.2}$$

Where K_R is the reference air kerma rate.

Substituting (2) in (1):

$$N_k = \frac{K_{air}}{D_{air}} \times \frac{1}{m_{air}} \times \left(\frac{w}{e} \right)_{air} \quad \text{VIII.3}$$

Note: It is therefore important to note that the N_k determined in the equation VIII.3 is for one discrete energy. However, Ir-192 spectrum is extremely complex. Thus, the following should be considered:

$$D_{air}(Ir-192) = \sum D_{air}(E_i) \quad \text{VIII.4}$$

Where $D_{air}(E_i)$ is the Air dose due to the photon fluency of energy E_i .

Using equations VIII.2 and VIII.4:

$$\frac{K_{air}(Ir-192)}{N_k(Ir-192)} = \sum_i \frac{K_{air}(E_i)}{N_k(E_i)} \quad \text{VIII.5}$$

The N_k for Ir-192 can finally be determined using the following expression:

$$\frac{1}{N_k(Ir-192)} = \sum_i \frac{K_{air}(E_i)}{K_{air}(Ir-192)} \times \left(\frac{1}{N_k(E_i)} \right) \quad \text{VIII.6}$$

ii) How to use the Fricke Dosimetry as a primary standard of dose to water for Ir-192 sources.

Background: A dosimetry standard for the direct measurement of the absolute dose to water for the Ir-192 sources is currently not available. Fricke dosimetry is a technique that depends on the oxidation of ferrous ions (Fe^{2+}) to ferric ions (Fe^{3+}) by ionizing radiation. The Fricke dosimeter is 96% water

by weight; therefore, its dosimetric properties are close to those of water.

The AAPM TG-43 Report and its update uses a protocol for determining the dose to water based on an air kerma strength (S_k) measurement. The dose to water conversion is performed via the dose rate constant Λ , which converts the air-kerma strength to the dose to water:

$$\dot{D}_w = S_k \times \Lambda \quad \text{VIII.7}$$

Where \dot{D}_w is the Dose rate to water, in $\text{cGy}\cdot\text{h}^{-1}$; S_k is the Air kerma strength, in $\text{cGy}\cdot\text{cm}^2\cdot\text{h}^{-1}$; and Λ is the dose rate constant, in cm^2 ;

The Fricke dosimetry can be used as an alternative method to that proposed by the TG-43. The main advantage of the Fricke methodology is being independent of the dose rate constant and of the air kerma strength. As discussed by Klassen et al. [144], the absorbed dose to the Fricke solution, D_F , was obtained from the following equation:

$$D_F = \frac{\Delta OD}{G(\text{Fe}^{3+}) \cdot L \cdot \rho \cdot \varepsilon} \quad \text{VIII.8}$$

where ΔOD is the difference in the absorbance of the irradiated and non-irradiated solutions; L is the optical path length of the cuvette; ρ is the density of the Fricke solution at 25 °C; ε is the molar linear absorption coefficient of the ferric ions; and $G(\text{Fe}^{3+})$ is the radiation chemical yield of the ferric ions.

Important: As the temperature can affect the optical absorbance, it is therefore important to correct the ΔOD for differences in temperature, according to equation VIII.8:

$$\Delta OD = (OD_i - OD_c) \cdot [1 + 0.0012 \cdot (25 - T_i)] \cdot [1 + 0.0069 \cdot (25 - T_r)] \quad \text{VIII.9}$$

Where OD_i and OD_c are the optical densities of the irradiated and control solutions, respectively; T_i is the temperature in °C of the Fricke solution during the irradiation; and T_r is the temperature in °C of the Fricke solution during the spectrophotometer reading.

Because the final goal is to determine the dose to water, the equation VIII.10 is used for this purpose:

$$D_w = D_F \times F_{F/w} \times F_X \quad \text{VIII.10}$$

Where D_w is the dose to water; D_F is the dose to Fricke; $F_{F/w}$ is the conversion factor for Fricke to water; and F_X is other conversion factors.

Important: It is important to note that equation VIII.10 should always be adequate for the used measurement setup. The applied correction factors will mainly depend on the type of vessel that is used for the irradiation of the Fricke solution.

VIII.2.b. Exercises

- i. What are the procedures, as recommended by the TECDOC-1079, for the determination of the air kerma strength of brachytherapy gamma sources?
- ii. Three different methodologies are used to obtain the Ir-192 calibration factor $N_k(\text{Ir})$: that described by Ezzel; that described by Goestchet *al.* [145]; and that described by Marechal [143]. What are the main differences among these three methods?
- iii. Brachytherapy uses encapsulated radioactive sources to deliver a high dose to tissues near the source [146]. What is the recommended quantity for the specification of gamma sources? And for Beta ray sources? What are the reference distances for those quantities? What are the SI units of reference for those quantities?
- iv. How should the calibration of I-125 and Pd-103 low dose rate brachytherapy sources be performed? Which

- laboratories can provide reference air kerma rate calibrations for those kinds of sources?
- v. What is the accepted methodology for the primary standard determination of the absorbed dose rate of beta ray sources? Is this method acceptable for all kinds of beta ray sources?
 - vi. The Fricke dosimetry is being considered an option for the calibration of Ir-192 brachytherapy sources. How does this type of dosimetry work? How can the absorbed dose to water be obtained?
 - vii. According to the ISO 4037-1 [30], reference radiation in the energy range between 4 MeV and 9 MeV is provided because of the 6 MeV photon fields, which are widely used in radiotherapy treatment and the nuclear industry. An ionization chamber should be used to determine the air kerma at the point of testing. Does a removable cap have to be used on the detector with an air-equivalent material? Which corrections should be used? Justify.
 - viii. The ion chamber that is used for the dosimetry of photon energies between 4 MeV and 9 MeV should be calibrated in air in terms of air kerma and with a photon spectrum similar to that of the reference radiation. What should be done when it is not possible to obtain a similar spectrum for the calibration?
 - ix. What are the main requirements that a Co-60 source should have to be used for the calibration of ionization chambers?
 - x. What are the physical quantities that are used as a measure of the radiation in external beam therapy for Co-60? And for linear accelerators? Is it the same quantity that is used for charged particles?

VIII.3 Diagnostic X-rays

VIII.3.a Practical examples

- i) How is an estimate of the uncertainty of the HVL determined in a diagnostic x-ray equipment?

Background: An important parameter to be measured in quality assurance (QA) programs in diagnostic X ray facilities is the half-value layer (HVL), whose definition is well established [28]. The typical measuring arrangement consists of an ion chamber that is positioned at a fixed distance from the x-ray tube focal spot, and a group of filters are introduced between the source and the detector for the determination of the relation of the radiation intensity and filter thickness. Despite the simplicity of the measurement procedure and simple instrumentation and geometric assembly to HVL measurements, its uncertainty estimation must be conducted with care.

Some QA national guides adopt a simplified equation for the calculation of the HVL from an x-ray machine:

$$HVL = \frac{x_b \ln\left(2^{L_a/L_0}\right) - x_a \ln\left(2^{L_b/L_0}\right)}{\ln\left(L_a/L_b\right)} \quad \text{VIII.11}$$

where L_0 is the intensity of the x-ray beam without any filter between the source and the detector; L_a is the intensity of the beam after attenuation by thickness x_a ; and L_b is intensity of the beam after attenuation by thickness x_b .

Condition: $L_a \leq L_0/2 \leq L_b$

Consider a practical situation where $L_0 = 5\text{mGy}$, $L_a = 2,7\text{mGy}$ and $L_b = 2,2 \text{ mGy}$. The corresponding thickness in Aluminum for obtaining these attenuated beam intensities was $x_a = 3 \text{ mm Al}$ and $x_b = 4 \text{ mm Al}$.

The main instruments for measuring the HVL consist of a radiation detector, normally an ion chamber, and a caliper or micrometer for determining the Aluminum thickness of the attenuating filters. A system for determining the distances from the X ray tube focal spot and the set of filters and the radiation

detector could also be considered, but it is not critical for the final uncertainty and will not be considered in this example.

The HVL uncertainty is determined from the conventional method from the derivation of the equation VIII.11 on the main variables:

$$u_{HVL} = \sqrt{\left(\frac{\partial HVL}{\partial L_0}\right)^2 \cdot u_{L_0}^2 + \left(\frac{\partial HVL}{\partial L_a}\right)^2 \cdot u_{L_a}^2 + \left(\frac{\partial HVL}{\partial L_b}\right)^2 \cdot u_{L_b}^2 + \left(\frac{\partial HVL}{\partial x_a}\right)^2 \cdot u_{x_a}^2 + \left(\frac{\partial HVL}{\partial x_b}\right)^2 \cdot u_{x_b}^2} \quad \text{VIII.12}$$

where $u_{L_0}^L$ is the uncertainty of the L_0 measurement; $u_{L_a}^L$ is the uncertainty of the L_a measurement; $u_{L_b}^L$ is the uncertainty of the L_b measurement; $u_{x_a}^x$ is the uncertainty of the x_a measurement; and $u_{x_b}^x$ is the uncertainty of the x_b measurement.

These uncertainty values, however, are composed of combinations of Type-A and Type-B uncertainties that must be carefully estimated in order to be representative of the experimentally obtained values. Depending on the level of desired uncertainty estimation, the budget to be developed by the user can take into account different number of hypotheses that can be considered in the calculation of the equation VIII.12. A complete definition of these budgets is out of the context of the present chapter, and more information can be obtained from the references [28]. In particular, the appendix VIII of the reference presents examples of uncertainty budgets that can help the users of this methodology determine the correct approach for the criteria to be adopted in each case.

From the practical situation considered above, the uncertainty on the filter thicknesses can take into account the thickness-measuring procedure using a conventional instrument, such as a caliper rule or micrometer. In this case, the resolution, calibration and accuracy can be adopted for composing the type-B uncertainty, and an average of measurements can generate a

mean value and standard deviation to be incorporated on the type A uncertainty. However, the measurement process requires a superposition of sheets of filters, which introduces a combination of uncertainties that must be considered for a consistent budget.

On the other side, the measurement processes on a clinical environment usually have an additional constraint related to the time expended for conducting the measurement. The measuring procedures must be optimized in order to spend as little time as possible, as the X ray system must be used for its main purpose: producing diagnostic images. Therefore, it is a common restriction for the implementation of HVL measurement procedures to have few L_0 , L_a and L_b data, increasing the Type A uncertainty. Typical Type B uncertainties must also be considered when using ion chambers or other radiation detectors [39, 47].

VIII.3.b Exercises

- i. Considering the example of the estimation of the uncertainty on HVL presented above, what sources of uncertainties are important to be adopted for the budget? Consider Type A and Type B uncertainties for both ion chamber and micrometer measurements.
- ii. The metrology formalism introduced in the IAEA code of practice TRS 457 [28] proposes the following equation for air kerma measurements:

$$K = M_Q N_{k,Q_0} \prod_i k_i \quad \text{VIII.13}$$

Define the quantities M_Q and N_{k,Q_0} and identify the main correction factors k_i that are normally used in practical measuring procedures.

- iii. Suppose that you must determine an air kerma value of a diagnostic x-ray beam with quality Q, but the calibration

- certificate provided by the local SSDL only presents the calibration coefficient at quality Q_0 . How do you estimate the air kerma, and what is the uncertainty budget considering this restriction?
- iv. The IAEA code of practice TRS 457 [28] presents examples of typical uncertainty budgets for quantities that are directly measured using diagnostic dosimeters (Table 8.2) in three different scenarios. Describe in detail the differences between these scenarios and which criteria you hypothetically would use in each situation.
 - v. Suppose you perform a survey of entrance surface air kerma in a group of 25 real patients who will undergo chest imaging. All of the patients will undergo antero-posterior (AP) and lateral (LAT) imaging, and you will measure the fixing groups of 3 LiF TLD's directly on the patient body using Micropore© tape. Summarize the measurement procedure, the data analysis and the uncertainty budget.

VIII.4. General Questions

- i. What are the primary objectives of calibration?
- ii. What is the importance of the “*Bureau International des Poids et Mesures*”? What is its position on the International Measurement System?
- iii. In order to maintain a high accuracy of measurement, it is important to follow the reference conditions values or ranges for the parameters that affect the measuring instruments. What are those parameters? What are the reference ranges?
- iv. According to the TRS 469 [38], the SSDLs should follow internationally accepted standards. The IAEA has organized a comparison program using ionization chambers to help the SSDLs verify the integrity of their national standards. How does it work? What are the procedures that the SSDLs should follow?
- v. The quantity air kerma characterizes a beam of photons or neutrons in terms of the energy that is transferred to any

- material. Air kerma is defined as the total energy per unit mass transferred from an x-ray beam to air. This quantity is widely used for calibration services [147]. It is possible to establish a relationship between the air kerma and another quantity: the exposure. The exposure is defined as the total charge per unit mass liberated in air by a photon beam. Define and demonstrate the relationship between air kerma and exposure.
- vi. For the calculation of air kerma, using an ionization chamber, the mean energy that is required to produce an ion pair in air per unit charge (W_{air}/e) is necessary; it is usually assumed to be a constant value for the (W_{air}/e). However, the ICRU recently recommended two different methodologies to obtain this quantity. What are the methodologies?
 - vii. Several Primary Standard laboratories offer a calibration service for megavoltage X rays. Those labs calibrate secondary standard dosimetry, using two different models of dissemination. What are they?
 - viii. The quantity absorbed dose to water (D_w) is very important for the clinical procedures of therapy treatments. However, due to the very limited beam penetration and the relatively low dose rates involved in kilovoltage X rays, it is difficult to measure the D_w . Thus, almost all of the existing standards for these radiations are based on air kerma. How can the D_w be determined from the air kerma?
 - ix. The free air chamber is accepted as a primary standard for the determination of air kerma using kilovoltage X rays. How can the air kerma be obtained using this type of chamber? What is the final estimated uncertainty?
 - x. There are three accepted techniques that can be used to achieve a primary standard for the absorbed dose to water for high-energy X rays used in the linear accelerators. Describe briefly the three techniques, highlighting their main positive and negative points.

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CONGRESSO BRASILEIRO
DE METROLOGIA DAS
RADIÇÕES IONIZANTES

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