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EUROPEAN WORKING GROUP « HOT LABORATORIES AND REMOTE HANDLING »
5 / 6 JUNE 1997 - STUDSVIK, SWEDEN

NEUTRON DOSIMETRY IN NUCLEAR FACILITIES

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Abstract

The evaluation of ambient and personal doses due to neutrons in nuclear facilities is one of the most difficult problems in radiation protection field.

Neutrons are present in many installations concerned by the nuclear fuel cycle (fuel rod manufacturing, nuclear power plants, transportation packaging, research laboratories involved in radiographic applications, neutron radiotherapy purposes, ...). The neutrons produced in those facilities cover a large spectrum of energy, from 10^{-3} to 10^{8} eV. In a practical point of view the dosimetric systems shall be able to measure doses on a neutron energy range of 10 decades.

Neutrons are neutral particles and are for this reason not directly detectable. Dose due to neutrons is obtained by the production of secondary particles and detection of neutrons is also achieved by measuring the secondary particles emitted during interaction between neutrons and absorption material. So the evaluation of « neutron doses » is a very complex phenomena which is directly linked to the nature of the reaction produced in the body or in the detection materials which are used in the dosemeters.

On the other hand, the biological effect of the secondary particles on organs or tissues, created during the interactions of neutrons with matter, is strongly dependant on the energy of the neutrons. So it is required to know adequately the energy distribution of neutrons for the study and the evaluation of the dosemeters.

Most of the radiation protection instruments available on the market at the present time do not satisfying completely these different requirements, especially those which measures personal doses. These different arguments require a new approach of neutron dosimetry.

The present communication intends to give some explanation on the previous radiation protection concepts, on the different techniques in use for neutron measurement and on the main research programs which are achieved in different countries in order to develop new neutron dosimetric systems.

1 - Introduction

Neutron dosimetry needs to know the energy distribution of the incident neutrons because of their interaction with the matter constituting the environment and the human bodies. Due to the mulptiple absorption and diffusion reactions inside the tissues, a lot of secondary charged particles are created into the body, which participate to the energy deposit.

While X or γ rays produce constant biological effects for a same amount of dose, in the whole energy range, neutrons present a higher biological effect and are strongly dependent on their energy.

Neutrons can not be directly measured because they are neutral particles; their detection is achieved by measurement of the secondary radiations or particles created during the interactions with a material which is specially chosen for its behaviour of producing these kind of interactions.

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(1) ICRP : International Commission on Radiological Protection
(2) ICRU : International Commission on Radiation Units and Measurement

Moreover, in many applications γ radiations are produced simultaneously with neutrons particles, so that the detection of "neutron dose" in mixed radiation fields needs special measurement techniques. Neutron dosimetry could be finally very complex and difficult to achieve. So its objectives should be clearly defined.

The first objective, with respect to the radiation protection regulations, is to comply with the regulatory dose limits defined for the different categories of workers involved in the nuclear facilities and the general public around these installations.

Monitoring of external exposure is achieved using **personal dosemeters** or **ambient dosemeters**. According to their principle of use, these dosemeters are divided into two families:

- **passive dosemeters**, which integrates effects of particles and gives total doses; they do not take into account the factor "time of exposure",

- **active dosemeters** (or ratemeters) which gives the dose rate or the integrated dose; they can deliver more sophisticate informations.

The second objective is to evaluate the doses corresponding to emergency or accidental situations. Such exposure corresponding to high doses or high dose rates can occur in case of critically accidents; they need specific dosimetric techniques.

In special cases, where the ratio neutron doses / γ doses is well known, neutron doses can be proceeded from the datas issued by the γ monitoring. Finally, complex dosimetric systems can be installed for example when a particular work place has to be designed or when annual exposure limits are near the regulatory dose limits.

In all the previous fields, and in order to take into account the new requirements issued by the different radiological boards (ICRU, ICRP, AIEA, EURATOM,....) considerable research programs have been achieved these last years in order to develop new neutron dosimetric instruments. The main progress have been made in the three following directions [1] [2]:

a - **Revolution** in the design of individual dosemeter of passive type, due to the work of APFEL [3] since 1997, derived from the principle of the bubble chamber (neutron bubble detector). In the same field, new electronic systems which facilitate the reading of the passive dosemeters are developed.

b - **Adaptation** of the technology of spectrometry developed in the field of high energy for example to radiation protection purposes. We can mention proportional counters (for microdosimetry) or semi conductors of silicium type.

c - **Use** of microelectronic systems, allowing a great number of functions in a small volume.

2 - Radiation Protection Dosimetric Quantities

2.1 - General

Neutron dosimetry in the radiation protection field deals with two kind of dosimetric quantities :

Radiation protection quantities, which are also called primary limiting quantities. These quantities, used for the definition of the dose limits according to the International Commission on Radiological Protection (ICRP) recommendations, in its Report n° 60 (1991) [4], take into account the interaction of radiation with tissues and organs; they are linked to the stochastic effects of exposed people.

These new dose limits for different categories of exposed people to ionizing radiation are indicated in the table 1 here after. We can note a reduction of a factor 2,5 for workers and of a factor 10 for the general population.

Table 1 : Actual dose limits and those recommended by ICRP

	Actually		Recommended by ICRP	
	Workers (1)	Public	Workers (1) (4)	Public
Annual effective dose (for the whole body)	50 mSv per year	5 mSv per year	20 mSv per year (averaged over 5 consecutive years) (2)	1 mSv per year (3)

(1) Applies to the total dose resulting from external exposure as well as from internal irradiation
(2) The annual dose limit should not exceed 50 mSv per year
(3) The dose limit can in certain circumstances exceed 1 mSv, as soon as the average value over 5 years don't exceed 1 mSv.
(4) For other organs (lens of the eyes, skin and extremities) the dose limits are respectively 150 mSv, 500 mSv and 500 mSv per year.

Operational quantities defined by the International Commission on Radiological Units and Measurements. These quantities are metrologic quantities introduced for personal and working area monitoring in case of external exposure (ICRU 1985, ICRU 1988) [5]. They are used for the calibration of radiation protection instruments.

Both types of families of dosimetric quantities consist each one of a coherent system and can be linked to the « basic physical quantities » such as the fluence (ϕ), the kerma in air (Ka), the absorbed dose (D), ... (ICRU 1980, ICRU 1993).

In each family of dosimetric quantities, the relevant conversion coefficient, linking respectively the radiation protection quantities and the operational quantities to the basic physical quantities, can be calculated using transport computer codes and appropriate mathematic models. The present paper will not present details about the calculation of these conversion factors.

2. 2 - Radiation protection quantities (or limiting quantities)

The radiation protection quantities defined by the publication 60 of ICRP (ICRP, 1991) are the following:

$$H_T = \sum W_R \times D_{T,R}$$

H_T (Sv) represents the "dose equivalent" in a tissue or in an organ where $D_{T,R}$ (Gy) is the absorbed dose averaged over a tissue or organ and weighted for the radiation quality that is for interest. The weighting factor, also called the radiation weighting factor, W_R, is selected for the type and energy of radiation incident upon the body.

$$E = \sum W_T \times H_T$$

E (Sv) represents the "effective dose" where W_T is the appropriate weighting factor of a specific tissue T. It could be noted that for the whole body $\sum W_T = 1$.

The effective dose is the key concept in radiation protection monitoring. It represents the dosimetric quantity which has to be calculated when a risk assessment is achieved (it is admitted than the risk is proportional to the effective dose). For the radiation protection regulation point of view, it is also considered as the reference quantity which is directly linked to the regulatory dose limit.

One must notice that the effective dose which represents the equivalent dose received by the exposed people, is essentially theoretical. The effective dose can only be determined using antropomorphic phantoms approaching the human body.

Radiation weighting factor W_R

The purpose of this weighting factor W_R is to take into account the relative biological efficiency either of the charged particles during their interaction with tissues or of all the particles constituting the incident beam. It replaces the previous quality factor Q(L), which is only used for the determination of the operational quantities as explained in section 2.4 here after.

Numerical values of W_R are given by the ICRP Report 60 for different types of incident radiation, including neutrons. Their comparison with previous quality factor Q (L) are indicated in the following table.

Table 2: From the quality factor Q(L) to the weighting factor W_R (ICRP, 1991)

Type of radiation	Weighting factor W_R	Quality factor Q(L)
Protons, all type	1	1
Electrons and muons, all energies	1	1
Neutrons		
less than 10 keV	5	2
10 keV to 100 keV	10	5
100 keV to 2 MeV	20	10
2 MeV to 20 MeV	10	10
higher than 20 MeV	5	5
Protons other than backscattering protons (energy higher than 2 MeV)	5	5
Alpha particles, fission products, heavy nucleons	20	20

Concerning neutrons, ICRP recommends to use instead of the previous discreet quantities, the following relation giving the value of the weighting factor W_R according to the energy E_n of the *neutrons (see also figure 1):*

$$W_R(E_n) = 5 + 17 \exp -(\ln(2 E_n))^2 / 6$$

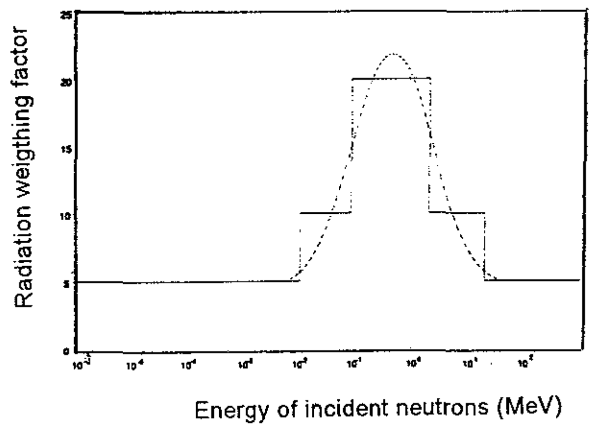


Figure 1: Variation of the weighting factor W_R for neutrons

Comment: *The introduction of these two changes* (*Quality factor replaced by the Weighting factor and* modification of dose limits) reduces the acceptable neutron doses by a factor of 10. It means that the future neutron dosimeters should be able to measure doses 10 times lower than presently. Not all the existing dosemeters have the needed sensitivity.

2. 3 - Operational quantities

The operational quantities (O.Q.) have been introduced in 1985 by ICRU in order to be able to make a reasonable estimation of the effective dose equivalent H_E, which in principle is not directly measurable and to be sure not to exceed the regulatory doses, as defined by ICRP for external exposure. Changes have appeared latter in publication 47 of ICRU (ICRU, 1992).

The operational quantities have been defined for penetrating radiations (for example photons of energy higher than 12 keV or neutrons of all energy) as well as for weakly penetrating radiation (alpha or beta particles).

Concerning working area monitoring, the radiation protection instruments shall be able to measure the ambient dose equivalent H*(10) and the directional dose equivalent $H'(d,\alpha)$. For individual monitoring, the personal dosimeters which are designed to be worn on the surface of the body (trunk) shall be able to measure the personal dose equivalent Hp(d). For neutrons, the operational quantities are respectively H*(10) and H' (0,07) for working area monitoring and Hp(10) and Hp(0,07) for personal monitoring.

Dose equivalent H

The dose equivalent H (Sv) at a specific point is defined by the following relation:

$$H = \int Q\,(L) \times D\,(L) \times dL$$

where Q (L) is the quality factor corresponding to particles with a linear energy transfer coefficient of L and D (L) x dL is the absorbed dose for particles having a linear energy transfer coefficient TLE which is comprised between L and L + dL at this point.

According to ICRP recommendations (ICRP, 1991), Q (L) is respectively represented by the following formula for different neutron energies:

$Q\,(L) = 1$	for L less than	10 keV / μm
$Q\,(L) = 0{,}32\,L - 2{,}2$	for $10 \leq L \leq$	100 keV / μm
$Q\,(L) = 300 / \sqrt{L}$	for L higher than	100 keV / μm

Ambient dosimetry

The ambient dose equivalent, H*(10) (Sv), at a specific point in a radiation field, is the dose equivalent that would be produced by the corresponding expanded and aligned field in the ICRU sphere, at a depth of 10 mm on the radius opposing the direction of the aligned field.

This quantity is generally not directly measurable, but can be determined from the measurement of the neutron fluence and the use of the appropriate conversion coefficient of the fluence to ambient dose equivalent.

$$H^*(10) = h^*\phi\,(E) \times \phi$$

An ambient dosemeter is « ideal » if the response in terms of fluence is independent of the direction of the incident neutrons (isotropical response), with only the same dependence of the energy of neutrons as of the conversion coefficient fluence → ambient dose equivalent.

Personal dosimetry

The individual (or personal) dose equivalent, Hp(d) (Sv) is the dose equivalent obtained in soft tissues as defined in ICRU 51 below a specified point on the body at an appropriate depth d.

As previously, this quantity is not directly measurable. It shall be determined using an antropomorphic phantom, from the measurement of the neutron fluence and the use of the appropriate conversion coefficient of the fluence to personal dose equivalent.

$$Hp(10) = hp\,\phi\,(E) \times \phi$$

The conversion coefficient $hp\,\phi\,(E)$ depends, in this case, on the energy of the incident neutrons and an the angle of incidence calculated according to the normal of the phantom front surface.

A personal dosemeter is « ideal » if the response in term of fluence presents the same energy and angular dependence as the conversion coefficient fluence → ambient dose equivalent (isodirectional response).

3 - Neutron dosimetric techniques

3.1 - Passive dosemeters

• **Thermolumescence albedo dosemeters (TLD dosemeters)** detect thermical and epithermical neutrons, using reaction ^{6}Li (n,α). Neutrons of higher energy are detected indirectly after having been thermalized through the tissues of the body, *i.e.* albedo neutrons. The response of those kinds of detectors decreases rapidly over 10 keV, so that their sensitivity, expressed in dose equivalent, is directly linked to the neutron operational spectral. They need consequently a calibration at each work place. Recent developments made in Russia and in China, using copper doped lithium fluoride or carbon doped aluminium, show that certain sensitivity can be multiplied by 30 to 50.

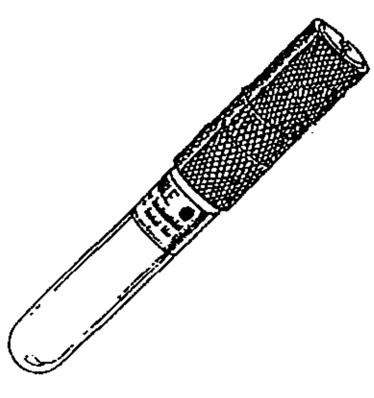
• **Nuclear photographic emulsions (NPE dosemeters)** detect only fast neutrons by measurement of the back-scattering protons issued by the interaction with the hydrogenous material present in the detector. Due to the fact that the lowest measurable energy of neutrons is around 1 MeV, this kind of dosemeters is only suitable for high energy accelerators. They are not used in nuclear power plants or in reprocessing facilities.

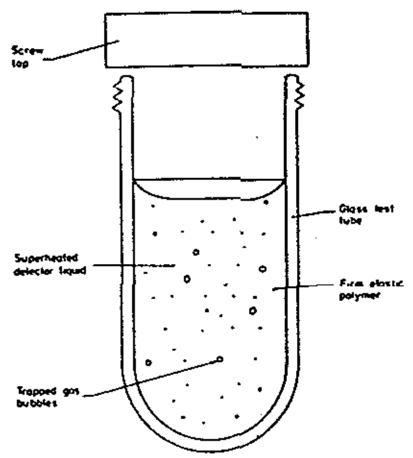
• **Etched track detectors (ETD dosemeters),** whose the most common one is « CR-39 », are based on the revelation of tracks produced in the detector during interaction of ionizing particles. This kind of detector allows the measurement of neutrons of 50 keV using the (n,p) reaction and the measurement of thermical neutrons using the (n,α) reaction. Their responses are acceptable, but important variation of background contribution limits their operational use at different work places.

• **Bubble or superheated drop dosemeters,** which technology is based on a principle similar to the one used for the bubble chambers employed in high energy physics [3] [6]. They consist of a gel or transparent polymer in which micro droplets of freon have been dispersed. In the absence of radiation these droplets are thermodynamically stable and capable of remaining in a metastable state over a period of several months. When exposed to a neutron field, charged secondary particles induce these droplets to undergo a liquid-gaseous phase change, while they can be counted with the naked eye.

The dosemeter composition is such as to assure proportionality between the number of bubbles and the dose equivalent. Dosemeters commercialized by Bubble Technology Industries cover a range of sensitivity (0,03 to 3 bubbles per µSv) together with different energy ranges (thermal energy, energies in the 200 keV to 15 MeV range, etc.).

These dosemeters present indisputable qualities; but their use for workers monitoring is for the moment limited because several dosemeters are necessary for cover the whole neutron spectrum and doses, their use is not convenient and they may be influenced by temperature. Their cost is also high.





This type of dosemeters have been qualified by IPSN/CTHEN for the french nuclear companies (EDF, COGEMA, CEA,...) [7].

Following table, extract from a Griffith work [7], gives the main characteristics of the different passive techniques in use for neutron dosimetry.

	Albedo (TLD)	Bubble detector	CR-39 (DST)	Emulsion NTA
Energy response	mediocre	excellent	good	acceptable
Lowest limit of dose equivalent (mSv)	0,005 - 0,2	0,005 - 0,02	0,02 - 0,3	0,3 - 0,8
Highest limit of dose equivalent (mSv)	> 100	1 - 10	> 50	> 100
Sensitive to photons	yes	no	no	yes, a little
Cost of the detector	moderate	high	low	low
Influence of the environment	very low	temperature and chocks	very low	humidity and temperature
Ease of lecture	excellent	good	good	tedious

3.2 - Active dosemeters

• **Proportional counters**, which are applied for microelectronic dosimetry. These kind of detectors are suitable for ambient monitoring. The measurement is achieved using a tissue equivalent proportional counter (TEPC) designed in order to simulate the energy deposit produced during the absorption of the charged particles in the biological structures. The secondary charged particles created in the holder react with the internal gas and produce ionisations. According to Bragg-Gray principle, the TEPC reacts as an ionisation chamber, sensitive to the neutrons and to γ rays.

The heigh of the pulse is related to the linear energy transfer (TLE) coefficient of the secondary charged particle, so that this counter should allow the measurement of the absorbed dose as well as the quality factor. Due to this behaviour, this detector is suitable for radiation protection measurements on complex fields. Sensitive to all kind of radiations, these devices allow the determination of the respective contributions of doses due to photons (low TLE) and to neutrons (high TLE) in mixed fields. This technique has been applied in several applications: space orbital stations, in aircrafts, near medical accelerators,...

Extensions of the previous technology are actually under study in several laboratories, based on different techniques (semi-conductor technology, multifil technology, CIRCE apparatus developed by IPSN, single or multi diode systems,...).

Their use can however not be extended to personal monitoring because the reduction of the volume of the chamber would lead to a reduction of the sensitivity by a factor of 10 to 20, which is unacceptable for neutron dosimetry.

• **Recombination chambers (RC)**. The principle of this apparatus is the same as the previous one except that the detector consists of an high-pressure ionisation chamber operating in a non saturated mode. In this mode a low collecting voltage allows the measured current to be influenced by the initial recombination of ions. This effect depends on the L of a charged particle such that the quality factor Q can be approximated by an empirical relation based on the ratio of the non-saturated to the saturated ion current. At this time only a few prototypes have been built.

• **Silicon diode detectors**. These kind of instruments are not yet available for operational purposes.

3.3 - Discussion

Globally the actual dosimetry techniques do not totally answer to the question of the users. Studies are on going in several laboratories for the development of active methods, especially those using silicium diodes. The major difficulties for this kind of detector are however their sensitivity to the incident photons and their response which is strongly dependant on the energy of the neutrons. If the response of those detectors can be predicted for monoenergetic neutrons, important improvement have to be made for complex neutron spectrum whose relative differences of more than a factor 10 appears for the moment. These results demonstrate the necessity to improve studies in realistic neutron spectrum. They lead also to the the necessity of specific calibration methods adapted to each working place.

3.4 - Spectometry techniques

Spectometry consist in the characterization of the field of radiation either in term of energy distribution (for this purpose, a measurement of the neutron fluence, according to their energy has to be improved and the relevant conversion coefficient taken into account) or in term of distribution of Linear Transfer Energy (TLE) coefficient as defined for microdosimetry application. Different techniques can be achieved. We present only here their general principles.

- **Multisphere systems:** The principle of the method consists to thermalize the incident neutrons with a multiple polyethylene sphere system of different diameters and to detect the thermalized neutrons in the middle of the spheres with proportional counters of ^{3}He or BF_3 type. Computer systems allow the calculation of the spectral distribution as well as the neutron fluence and the equivalent dose. The sensitivity of this system allows to cover all the neutron spectrum from 100 keV to 20 MeV.

- **Activation and nuclear fission systems:** This method uses several detectors realized in different material (Au, S, Cu,...). Due to the specific activation cross section of each of the material, these detectors allow the determination of the neutron spectrum distribution. This technique is particularly suitable for critically accident dosimetry. We will not describe in the present paper details of these measurements.

- **Systems based o the measurement of the time of flight:** This technique is suitable for pulsed neutron fields. The secondary particles produce during the interaction of the incident neutrons with the detector produce a signal which is detectable after a time related to the energy of the neutrons.

- **Backscaterring proton detection systems:** This method consists of the measurement of back-scattered protons produced during the interaction of the incident neutrons with a hydrogenous material (organic plastics or mineral materials). The detection is achieved by proportional counters. Computer systems permit to discriminate neutrons and protons. This technique allows a good energy resolution.

4 - Conclusions

The International Commission on Radiological Protection (ICRP) recommends a decrease in the mean annual exposure limits by a factor of 2,5 and an increase in the radiation weighting factors for neutrons. Both recommendations require the development of instruments of increasing sensitivity. In addition, during the past decade, the neutron energy range of practical relevance in radiation protection has been extended towards higher energies, *i.e.* 100 MeV and even more. In this energy region, secondary protons and possibly other particles contribute to the long-range component of the radiation field, in addition to the « *conventional* » neutron-gamma components at neutron energies < 20 MeV.

In addition to developments in the « *conventional* » range, the future instrumentation in dosimetry and spectrometry should also take the requirements of increased sensitivity and/or extended energies into account. Experimental research must be complemented by a further development of the methods and of the data base for numerical dosimetry. Improved area monitoring, possibly including spectrometric capabilities, must be combined with optimum individual monitoring which means the use of active, directly readable compact instruments.

Finally, it is of great practical importance that dosimetry research for radiation protection purposes is targeted towards application in realistic, operational situations and that the approaches with a view to enabling the transfer of research level methods to routine applications.

All these aspects demonstrate the need for further research and developments work and a closer investigation of workplaces.

5 - References

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