INTERNAL DOSIMETRY FOR NUCLEAR POWER PROGRAM

(Dosimeter dalaman untuk Program Kuasa Nuklear)

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Abstract

Internal dosimetry which refers to dosage estimation from internal part of an individual body is an important and compulsory component in order to ensure the safety of the personnel involved in operational of a Nuclear Power Program. Radionuclides particle may deposit in the human being through several pathways and release wave and/or particle radiation to irradiate that person and give dose to body until it been excreted or completely decayed from the body. Type of radionuclides of concerning, monitoring program, equipments and technique used to measure the concentration level of such radionuclides and dose calculation will be discussed in this article along with the role and capability of Malaysian Nuclear Agency.

Abstrak

Dosimetri dalaman yang merujuk kepada penganggran dos dari bahagian dalaman badan seseorang individu adalah satu komponen mustahak dan wajib untuk menjamin keselamatan personel yang terlibat dalam pengendalian Program Kuasa Nuklear. Patikel radionuklid boleh memasuki ke dalam manusia melalui beberapa laluan dan membebaskan sinaran gelombang dan/atau zarah untuk mengiridiasi manusia dan memberi dos kepada badan sehingga ia disingkirkan atau menyepai habis dari badan manusia. Jenis radionuklid yang berkenaan, program pemantauan, peralatan dan teknik pengukuran tahap kepekatan radionuklid tersebut serta pengiraan dos akan dibincang dalam artikel ini bersama peranan dan kemampuan Agensi Nuklear Malaysia.

Keyword: dose; internal dosimetry; Nuclear Power Program; monitoring;

INTRODUCTION

When a personnel is dealing with ionizing radiation, he may be exposed to the radiation either external or internally. Therefore, the dose from the radiation shall be quantified in order to ensure the safety of the personnel. Dose received by personnel working in the radiation field shall not be more than 20mSv (total internal and external) as stipulated in the Basic Safety Radiation Protection Regulations [AELB, 2010]. The dose for external radiation to personnel can be estimated (measured) easily using a personal device such as film badge, thermoluminesence or pocket type dosimeters. However, such dosimeter is not suitable for the internal exposure. Therefore, the intake of radionuclides results in internal dose and to assess the internal dose it is necessary to evaluate the internal contamination of the worker.. This is crucial especially when Malaysia is planning to operate her maiden Nuclear Power Plant by year 2021, the demand to monitor the personnel due to internal exposure will increase tremendously.

The overall aim of internal dosimetry is to provide methods for the estimation of ionizing radiation doses and the associated risks from radionuclides incorporated inside human body [Paquet *et al*., 2007] and this is particularly important to personnel who is working within the nuclear facilities. The radionuclide deposited in various body organs/tissues will releases wave and/or particle radiation to irradiate that person and will give dose to the body until it been excreted or completely decayed from the body. In recent years, there have seen considerable effort devoted to this field and important progress in the development of modeling approaches used in the calculation of doses, in assessments of doses from bioassay data and in applications including epidemiological studies.

ROUTES OF INTAKE

Basically, there are four major routes of intake (of radionuclide) into human body namely, 1) Inhalation of contaminated airborne/aerosol/gases; 2) Ingestion of contaminated foods, drinks, etc; 3) Injection through accidental sharp object; and 4) Absorption through wound and cut skin. Among these, inhalation had been the most prominent route of intake by an individual either during routine occupational exposure or accidental occasion. When one is inside a radioactive area, radionuclide particulate may be suspended in the air and can easily enter the body by inhalation. These particulates may be deposited in different parts of the respiratory tract depending upon their aerodynamic equivalent diameter (AED) [US EPA, 2011].

For assessing the inhalation risk, the probability of inhaling dust particles (including aerosol) is related to the settling rate. The slower the settling rate, the longer the particles will be suspended in the air and more likely to be inhaled. For dusts of similar shape and density, the rate of settling is proportional to the AED of the dust particle. The deposition in the respiratory tract is also influenced by the AED. Inhalable particulate is large diameter particulate matter (around 20 µm) that is inhaled either through the nostrils or the mouth. Meanwhile, small diameter particulate (≤ 10 µm) can reach the respiratory bronchioles and the gas-exchange region of the lungs. These are respirable particulate. In other words, the smaller the particulate, the higher changes it will be inhaled (as in Table 1) [DOSH, 2002; DOSH, 2005]. The routes of intake, transfer and excretion of radionuclide in human is shown as in Fig. 1.

INITIATION OF PERSONAL MONITORING FOR INTERNAL EXPOSURE

Personnel will be suspected to be exposed to the radionuclides and cause internal exposure when he is in the situation as below:

- when there is an indication of air contamination which may lead to significant inhalation (Dusty radiation environment)
- when personnel involved in accident of unsealed sources/nuclear facilities
- simple test of personal surface such as, hair, skin, nose blow and etc. indicating contamination
- personal air sampling wore by personnel *via* filter indicates contamination

RADIONUCLIDES OF CONCERNS

Depending on the type of nuclear facilities, the radionuclides of concern for the monitoring will be different. As this article mainly reflects to the Nuclear Power Program, Table 2 summarized the type of radionuclides of concern following two accident scenarios in a nuclear reactor [IAEA, 1989].

MONITORING TECHNIQUES

To investigate the level of concentration for the concerning radionuclides as mentioned in the Table 2, the monitoring of personnel for internal exposure shall be carry out. This can be done using two techniques, namely: *in-vivo* and *in-vitro*. Usually *in-vivo monitoring* is much widely used because it is simple, fast and most radionuclides while decaying will emit gamma radiation associated with the particle expelling. However, under certain circumstances, *in-vitro* will provide more informative.

In-vivo monitoring

This is the technique for internal dose monitoring of the radionuclides which emit radiation that is capable to penetrate out of the body, such as the x-rays or gamma rays of sufficient energy. The system can be designed for the measurement of the whole body (Whole Body counter, WBC) or specific organs (Lung/Thyroid counter) depending on the behavior of the radionuclide of concern.

There are many ways a person can be positioned during this measurement, i.e. sitting, lying, and standing. The detectors can be single or multiple and can either be stationary or moving. The advantages of whole body counting are that it measures body contents directly and not does rely on indirect methods (such as urinalysis) and that it can measure insoluble radionuclides in the lungs. On the other hand, disadvantages of whole body counting are that it can only be used for gamma emitters, except in special circumstances, and it can misinterpret external contamination as an internal contamination.

A Whole Body Counter (including other counters) is a device with low background arrangement connected to counting systems using either

- NaI(Tl) detectors for high energy photon detection
- Phoswich detectors with Be window and thin NaI(Tl) crystal and thick CsI(Tl) or CsI(Na), for low energy (<100 keV) photon detection
- HPGe detectors are replacing detectors for measuring the low energy and high energy photons with appropriate electronic systems.

Calibration of these systems is carried out with different type of physical and mathematical phantoms. Well known physical phantoms include BOMAB, LLNL, JAERI, thyroid and the knee phantoms. On the other hand, some of the renowned mathematical phantoms are MIRD, CRISTY and nowadays VOXEL phantoms. [Wikipedia, 2011]

The system is placed in a chamber with low background radiation in order to achieve a low level measurement. To avoid misinterpreting external contamination as an internal contamination, the personnel is always requested to take a shower and change the attire to prevent radiation from the contaminated skin or clothing prior entering the chamber for a certain period of time. Any radiation picking up by the detectors will be translated into qualitative and quantitative information using the counting system software.

In-vitro monitoring
When the radionuclide presents inside the body is not capable to penetrate the human body due to the low penetrating power (such as alpha, beta and weak gamma emitters), whole body counter is no longer helpful to provide information about the radionuclides. Measurement of such radionuclides shall be performed using the biological (bioassay) sample taken out of the body. In practice, alpha emitters which generally much toxic as compared to others are more preferably measured using alpha spectrometry compare to WBC due to better detection limit. The main sources of bioassay data are normally urine, faeces, breath and blood, although other sources such as hair and teeth have also been employed in special cases. Additional to this, the analysis of activities from nose blow or nasal swab can be used to provide an early estimation of the identities and relative levels of radionuclides in an inhalation mixture, even though these data can provide only a crude indication of potential intakes [IAEA, 2000].

The selection of bioassay sample will depends not only on the major route of excretion, as determined from the biokinetic model for the particular physico-chemical form and route of intake, but also depends on factors such as ease of collection, analysis and interpretation [IAEA, 2000]. In general, urine samples are the easiest to collect and will be the basis for the determination of intakes of materials that are readily absorbed, and of the levels of systemic activity in body tissues. Ideally, 24 hours urine sample should be used for routine monitoring.

On the other hand, intakes of materials that are poorly absorbed by the body, after either inhalation or ingestion, are usually assessed from faecal samples, but these samples are not readily collected and their measurement results are also difficult to interpret. Usually, the nominal transit time for material passing directly through the Gastrointestinal tract is about two days, but this varies considerably with diet, health of the individual and other factors. In addition, due consideration must also been given to the type of monitoring being undertaken. Special monitoring carried out to assess a specific intake may involve more comprehensive sampling than would be necessary for routine or task related monitoring [IAEA, 2000].

In either case, contamination of the sample during collection must be avoided. In general, bioassay samples should be collected only outside contaminated work areas after the person had following through decontamination process, especially, of the hands, so that activity found in the sample is representative of activity within the body.

For a rough estimation of total alpha activities, sample normally require a simple preparation and can be measured using equipments such as Zinc Sulphite (ZnS) scintillator detector or a Gross Alpha/Gross Beta gas flow Proportional counter. However, if an individual species of radionuclide is required (for example, the concentration level of each ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{242}Pu), then the sample shall first undergo a complicated radiochemical separation such as ashing, calcinations, solvent extraction, ion exchange and etc. Finally, the purified radionuclides are deposit onto a disc using either electroplating or co-precipitation technique. The disc is then measured for quantitative and qualitative information using an Alpha Spectrometry System.

Meanwhile, for the gross beta activities, it can be measured using a Gross Alpha/Gross Beta Proportional counter or undergone a complicated radiochemistry separation such as of alpha emitters, or simple distillation (only for ^{3}H) before measured using Liquid Scintillation coincidence counter to quantify the nuclide especially for low energy beta emitters such as ^{3}H, ^{14}C, ^{35}S and ^{241}Pu.

DOSE ASSESSMENT AND CALCULATION

The concentration of the radionuclide measured from either ways shall be transformed into dose value to estimate the risk associated to a personnel. For intake and dose assessment calculations, several computers codes such as sophisticated LUDEP and the less complicated MONDAL/MONDES are available.

LUDEP which was designed by National Radiation Protection Board, UK is a suite of programs, developed in Turbo and Power Basic and compiled for the IBM-compatible personal computer. It enables the user to calculate doses and dose rates to the respiratory tract regions and other body organs for a wide range of user-defined conditions. The software has been designed to be as flexible as possible, permitting the user to change most of the parameters used in the dose calculation and display the results obtained at each stage of the calculation. [IAEA, 2011]

Meanwhile, MONDAL/MONDES from National Institute of Radiological Sciences, Japan enables dosimetrists to estimate the intake of radionuclides and the resulting committed effective dose based on the measurement results of individual monitoring such as *in-vivo* counting or bioassay measurement. This software consists of three electronic look-up tables for retention and/or excretion of inhaled or ingested radionuclides:

- MONDAL1.xls: for inhalation of particles of 1 μm AED by personnel
- MONDAL5.xls: for inhalation of particles of 5 μm AED by personnel
- MONDES.xls: for ingestion by personnel,

and also a program: CAL.xls, by which the intake of radionuclides and the committed effective dose are readily calculated. [IAEA, 2011]

ROLE AND PRESENT CAPABILITY IN NUCLEAR MALAYSIA

As a centre for nuclear technologies in Malaysia, Nuclear Malaysia shall develop her owns capacity to stay involve in the internal dosimetry especially as the focal point for reference to other organizations. At present, there is still no proper monitoring program had being carried out either for *in-vivo* or *in-vitro* monitoring. In terms of facilities, there is only one unit of Whole Body Counter and one unit of Thyroid Counter to serve for the *in-vivo* purpose. Meanwhile, to serve the purpose of *in-vitro* monitoring on bioassay sample, equipments such as two units of Gross Alpha/Gross Beta gas flow Proportional Counter, 48 silicon surface barrier alpha detectors and three units of Liquid Scintillation Counter are already available here. However, equipments along were not sufficient to carry out the monitoring program without proper human capital development especially knowledge and skills related to the radiochemistry processing of bioassay samples.

CONCLUSION

Internal dosimetry is important and compulsory component for a Nuclear Power Program (NPP) and other nuclear facilities in order to know the dose from radionuclides that deposited inside the human body in order to ensure the safety of the personnel. Items and particulars as discussed above will be helpful for one to estimate the dose internally to personnel. Each NPP shall develop it owns capacity and a proper plan to carry out such program so that it will prevent personnel to suffer in the later stage. Meanwhile, Nuclear Malaysia serve as the centre for nuclear technologies in Malaysia shall develop her owns capacity to get involved in the internal dosimetry especially as the reference centre to the other organizations.

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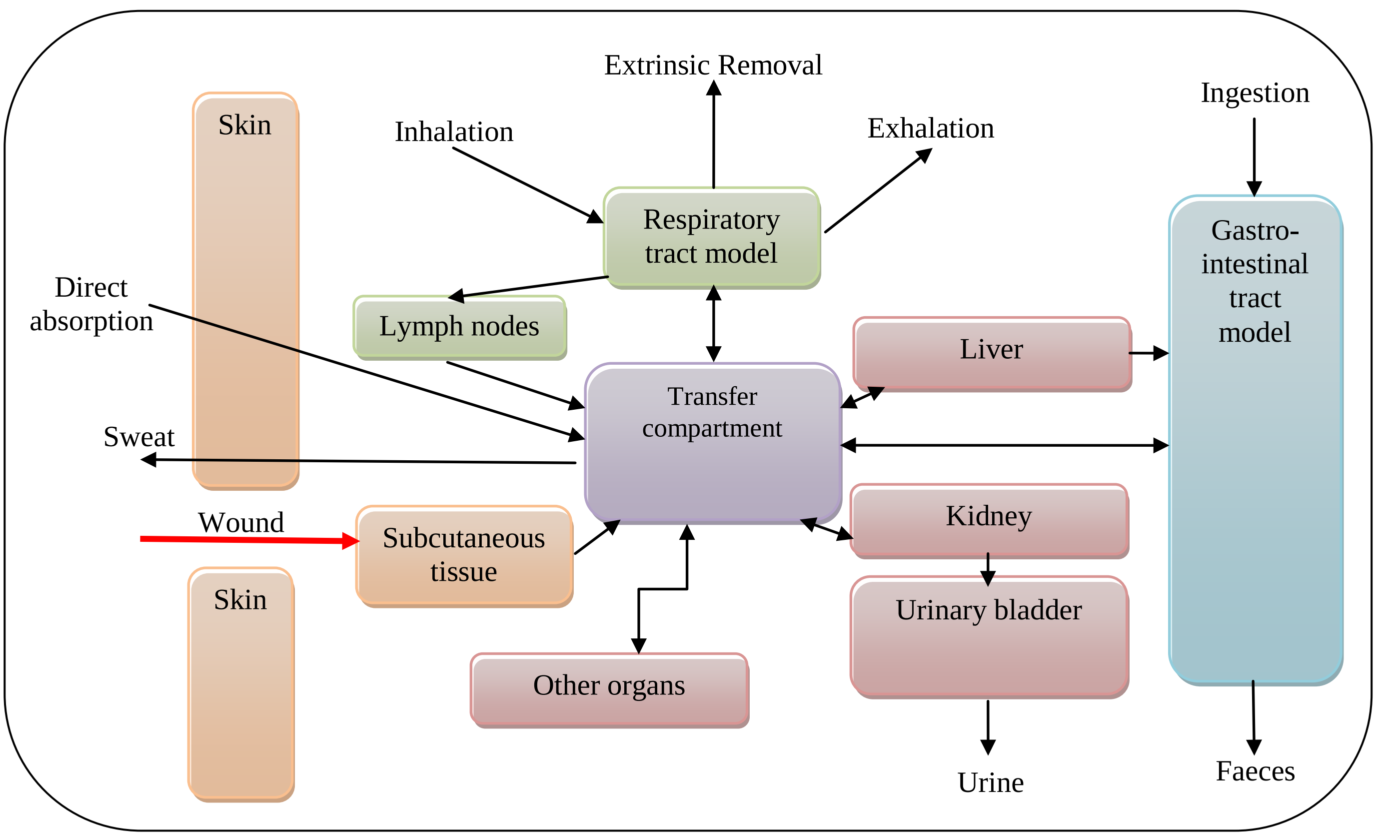

Fig. 1: Route of intake, transfer and excretion [NCRP, 1987]

Table 1: Percentage of particulate mass inhaled

particle aerodynamic diameter (micrometer)	**respirable particulate mass (%)**
0	100
1	97
2	91
3	74
4	50
5	30
6	17
7	9
8	5
10	1

Table 2: Radionuclides of concern with the time dependent change for two possible nuclear reactor accidents

Reactor accident scenario	**Monitoring Duration**	**Radionuclides of concern***
Reactor meltdown with particle containment	Short Term	**^{3}H**, ***^{89}Sr***, **^{90}Sr**, ^{103}Ru, ^{105}Rh, **^{106}Ru**, *^{131}I*, ^{133}I, *^{140}Ba*, *^{140}La*
	Long Term	**^{3}H**, **^{89}Sr**, ***^{90}Sr***, **^{99}Tc**, ^{103}Ru, **^{106}Ru**, ***^{129}I***, *^{134}Cs*, *^{137}Cs*
Reactor meltdown with or without failed containment	Short Term	**^{86}Rb**, ***^{89}Sr***, **^{90}Y**, **^{91}Y**, *^{95}Nb*, *^{95}Zr*, ^{96}Nb, ^{99}Mo, ^{160}Tb, ^{103}Ru, ^{105}Rh, ^{111}Ag, **^{112}Pd**, ^{115}Cd, ^{115m}Cd, **^{121}Sn**, ^{124}Sb, ^{125}Sn, ^{127}Sb, ^{131}I, ^{131m}Te, ^{132}Te, ^{133}I, ^{136}Cs, *^{140}Ba*, *^{140}La*, ^{141}Ce, ^{143}Ce, **^{143}Pr**, ^{147}Nd, ^{149}Pm, ^{151}Pm, ^{153}Sm, ^{239}Np
	Long Term	**^{3}H**, **^{89}Sr**, ***^{90}Sr***, **^{91}Y**, ^{93m}Nb, ^{95}Nb, ^{103}Ru, **^{106}Ru**, ^{110m}Ag, ^{113m}Cd, ^{115m}Cd, ^{121m}Sn, ^{123}Sn, ^{124}Sb, **^{129}I**, *^{134}Cs*, *^{137}Cs*, ^{141}Ce, *^{144}Ce*, ^{147}Pm, ^{160}Tb, ***^{238}Pu***, ***^{239}Pu***, ***^{240}Pu***, ***^{241}Pu***, **^{242}Pu**, ***^{241}Am***, **^{243}Am**, ***^{242}Cm***, ***^{244}Cm***

Note:

- ^{3}H, ^{90}Sr and ^{106}Ru: Pure beta emitters.
- Others bolded radionuclides: emitting alpha or weak gamma, preferably determine using bioassay.
- Italic type denotes radionuclides are of major concern.