Data for Radiation Protection and Nuclear Data

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Various conversion coefficients have been used in external and internal dosimetry in radiation protection practices. Radiation doses in the human body cannot be directly measured in general situation and the conversion coefficient has been used to correlate the human body dose with physical quantities such as radioactivity, particle fluence and other dosimetric quantities to be used to describe the radiation field. Fluence-to-organ dose conversion coefficients have been calculated using Monte Carlo radiation transport codes in conjunction with an anthropomorphic mathematical phantom. Neutron and photon interaction cross-section libraries are indispensable for these calculations. ICRP Publication 74 gives tables of conversion coefficients for estimation of organ doses and effective dose for photons, neutrons and electrons. Based on these results, shielding calculation parameters have been prepared for simple and easy dose estimation in radiation facilities. Dose factors, organ doses and effective dose per unit intake of radionuclide, have been also calculated for internal dosimetry purpose. ICRP Publications 68 and 72 give tables of dose factors for a variety of radionuclides. Revision of radiation data library has been made to reflect updated information on radionuclides to internal dosimetry.

1. Introduction

The Medical Use Group on Atomic, Molecular and Nuclear Data, one of standing groups of the Japanese Nuclear Data Committee, has been studying various data necessary for medical diagnosis and treatment, and radiation protection. The Group's mission is to investigate needs coming from doctors, medical physicists and health physicists, and to convey them to developers and editors of nuclear data. The present paper describes one of the Group's activities focusing on radiation protection aspect of nuclear data.

2. Radiation Exposure and Dose Assessment

The dose assessment is one of the most important practices of radiation protection in reactor, accelerator and radionuclide-handling facilities. Radiation exposures are caused by radiations coming from the outside of the body and radiations originated from radionuclides taken into the body. Since it is practically impossible to measure directly the doses to the human body, the doses should be assessed using measurable quantities such as radioactivity taken into the body and radiation fluence incident on the body. Dose conversion coefficients play a very important role to transform these measurable quantities to a human dose, not measurable quantity. The conversion coefficients are usually estimated by numerical simulation using a radiation transport program and a mathematical human model. In the transport simulation, nuclear data such as interaction cross-section should be provided.

A set of dosimetric quantities used in radiation protection is defined by the International Commission on Radiological Protection (ICRP) in its Publication 60[1]. One of two basic quantities representing the human tissue or organ dose due to radiation R is the equivalent dose H_T, the absorbed dose $D_{T,R}$ multiplied by the radiation weighting factor w_R given by ICRP. The other is the effective dose E, a summation of the equivalent doses in tissues or organs, each multiplied by the relevant tissue weighting factor w_T given by ICRP and it is given by the expression

$$E = \sum_T w_T \cdot H_T = \sum_T w_T \sum_R w_R \cdot D_{T,R}$$

In the case of exposure from in-body radionuclide, the entire dose is not given at the moment of the intake, but the exposure extends over a certain period undergoing decay and removal of the radionuclide. For this case, the committed effective dose $E(\tau)$ is used, which is the effective dose accumulated over a period τ, where 50 and 70 years are recommended for adult and children, respectively. The conversion coefficients are therefore prepared for estimating these quantities.

3. Calculation of Dose Conversion Coefficients

Conversion coefficients correlating the fluence of incident radiation with the equivalent dose or effective dose are generally used in external dose assessment. These coefficients are calculated using a Monte Carlo radiation transport program in conjunction with a mathematical human model. The anthropomorphic mathematical phantom shown in Fig. 1 is widely used in such calculations, which is described with a variety of geometrical formulas. Various codes have been applied to calculate radiation transport from a source to organs or tissues in the body. Well-used codes are MCNP[2] for neutron and photon, and EGS4[3] for electron and photon. Some other codes used in radiation shielding are also applicable to such dose calculation. Nuclear data are indispensable in these calculations: neutron and photon interaction cross-section data and kerma factor or stopping power for charged particles. Since the issue of ICRP 1990 Recommendations[1], dose conversion coefficients for a variety of radiations were calculated by many groups in the world and compiled in ICRP Publication 74[4] and ICRU Report 57[5] for photons, neutrons and electrons of conventional energies. Dose conversion coefficients for these radiations of higher energies and other types of radiation have been calculated and discussed[6] after the issue of above-mentioned reports.

For internal dose assessment, data of dose factors are prepared, which represent the equivalent dose in organ and effective dose per unit intake of radionuclide. When calculation the dose factor, distribution of radionuclides in the body after the intake should be calculated using a biokinetic model for metabolism at the first step of calculation. Then the radiation energy transfer from source organ to target organ is calculated by Monte Carlo radiation transport and mathematical phantom in the same way of external dose calculation. Less penetrating radiations such as α particle and low energy β particle are treated as such they deposit their whole energy at the source organ as an approximation. In this calculation, radiation data file has a quite important role. It describes the type, energy, intensity per disintegration, and other important data. ICRP Publication 38[7] including data for 830 radionuclides has been widely used in the calculation, but it should be updated because it is based on an old database, 1970s ENSDF. For this reason, revising work of this radiation data library has being made by JAERI[8] to adopt new version of ENSDF[9]. ICRP Publications 68[10] and 72[11] give tables of dose factors for a variety of radionuclides. For the practical use, maximum permissible limits are more convenient, which represent activity concentration giving 50mSv per year in the exhaust or drainage.

4. Current Topics

Regulation laws regarding radiation protection are going to be amended in April 2001 to adopt ICRP 1990 Recommendations, which include revisions of dose limit, dose quantities, etc. Necessary data were prepared prior to the enforcement of the new regulation laws. In radiation shielding calculation for reactor, accelerator and radionuclide-handling facilities, the effective dose in Anterior-posterior (AP) irradiation geometry, instead of the ambient dose equivalent $H^*(10)$, shall be estimated at the boundary of controlled area. Database of effective dose attenuation factors were developed for various RI γ-ray, neutron and β-ray sources and for six different shielding materials[12]. In this work, radiation transmission calculations were performed using JENDL-3.2[13].

A criticality accident happened on September 30,1999 at the uranium fuel processing plant in Tokai, Japan. Three workers were exposed to intense neutrons and γ-rays on the spot and two of the three died. A computer simulation has been performed to estimate detailed dose distribution in the body due to neutrons γ-rays from fission reaction[14]. In this simulation, Monte Carlo transport calculation code MCNP was used together with a special mathematical phantom that enables to simulate any working postures as shown in Fig. 2. Neutron cross-section library JENDL-3.2 was used in this calculation. The result is going to be published elsewhere in the near future.

5. Conclusion

Radiation dose to human body cannot be directly measured. Dose conversion coefficients play a very important role to transform measurable quantities such as particle fluence and radioactivity to the human dose. Various nuclear data have been used in the calculation of these conversion coefficients. In this context, we can say that nuclear data are supporting the radiation safety in the recent society.

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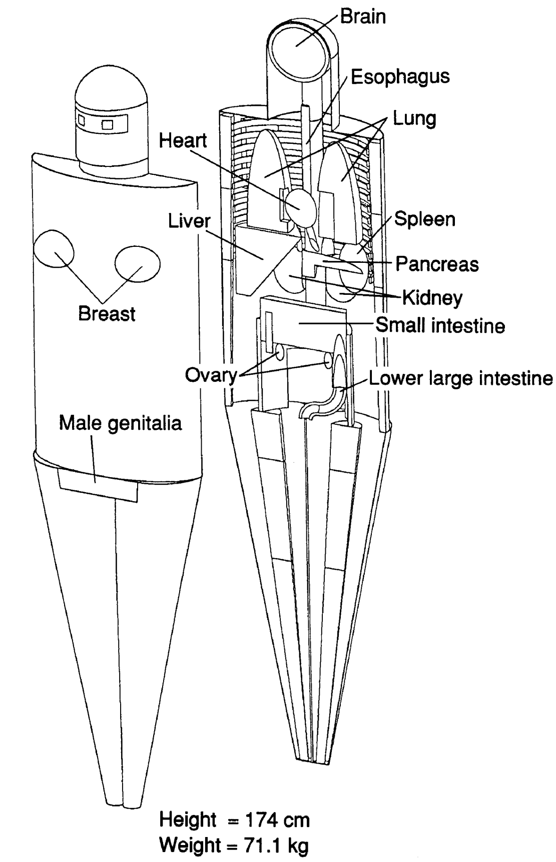

Fig. 1 Anthropomorphic mathematical phantom used in dose calculation

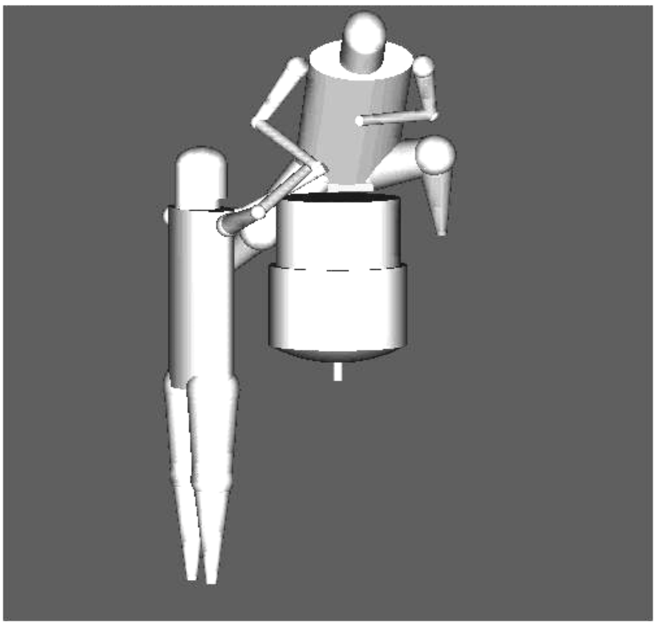

Fig.2 Simulation model to calculate detailed dose to two workers in the criticality accident