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ISSUES AND CLARIFICATION ON USE OF GAMMA RAY FLUENCE-TO-DOSE FACTORS FOR PACKAGE SHIELDING EVALUATIONS

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ABSTRACT

NRC Standard Review Plan for Transportation Packages for Radioactive Material, NUREG-1609, suggests that the flux-to-dose conversion factors in ANSI/ANS 6.1.1-1977, "Neutron and Gamma-Ray Fluence-to-Dose Factors", should be used when estimating the dose rates for compliance with regulations. This standard presents data recommended for computing the biologically relevant dosimetric quantity in neutron and gamma-ray radiation fields. Specifically, this standard is intended for use by shield designers to calculate effective dose equivalent. ANSI/ANS-6.1.1-1991, which is the latest version of the standard, is designated as a "historical" standard because it was officially withdrawn by ANSI as an American National Standard in 2001. ANS was unable to develop a revision to the standard within the ten-year period specified by ANSI for all standards, but intends to issue a revision in the future. The dose equivalents in the two versions of the standard (i.e., 1977 and 1991) are based on different data. The 1977 standard ceased to be recognized as authoritative in 1991 with the issuance of the later version. The NRC guidance recognizes that there are significant differences in the dosimetric models and the dosimetric quantities that form the basis of the factors in the two versions of the standard. The current NRC position remains that the 1977 version of ANSI/ANS 6.1.1 should be used for 10 CFR Part 71 applications. Codes commonly used by shield designers for estimating dose rates, for example MicroShield®, use recent ICRP data that is consistent with ANSI/ANS-6.1.1-1991. A review of the recent data has been done to consider what conversion factors may be applicable to and acceptable for use in radioactive material package shielding evaluations.

BACKGROUND

ANSI/ANS-6.1.1-1991 [1], the latest version of the standard, is designated as a "historical" standard because it was officially withdrawn by ANSI as an American National Standard in 2001. ANS was unable to develop a revision to the standard within the ten-year period specified by ANSI for all standards, but intends to issue a revision in the future. The information contained in this historical standard is believed to be correct, but a formal review to determine its accuracy has not been performed.

The dose equivalents in the two versions of the standard (i.e., 1977 and 1991) are based on different data. The 1977 standard ceased to be recognized as authoritative in 1991 with the issuance of the later version. The 1977 version of the standard was based on the maximum dose equivalent and the 1991 source bases the effective dose equivalent on the sum of weighted organ dose equivalents for an anthropomorphic representation of the human body. Caution must be exercised in taking specific gamma-ray constants from the literature because they can be in exposure units rather than in tissue-dose units. Also, doses computed for low-energy photons can be quite different when the conversion coefficients from the two versions of the standard are used.

Radiation Protection Principles [2]

Two very important documents in relation to radiation protection are the current recommendations of the International Commission on Radiological Protection (ICRP) and the International Basic Safety Standards for Protection against Ionizing Radiation and the Safety of Radiation Sources. Published in 1990, the latest ICRP recommendations are contained in ICRP 60 [3]. The Basic Safety Standards (BSS) were prepared jointly by the Food and Agriculture Organization of the United Nations, the IAEA, the International Labour Organization, the Nuclear Energy Agency of the Organization for Economic Co-operation and Development, the Pan American Health Organization, and the World Health Organization. They were published in 1996 by the IAEA as Safety Series No. 115 (SS-115) [4]. Paragraph 101 of the Transport Regulations SSR-6 [5] declares that the principles set out in both SS-115 and in IAEA Safety Series No. 120, “Radiation Protection and the Safety of Radiation Sources” [6], are fully utilized.

Most radiation protection quantities are either not measurable or their values are not directly traceable to primary standards. For this reason, knowledge of the numerical relations between the basic physical quantities and the dose quantities is very important prerequisite for the practical implementation of the whole system of radiation protection quantities. It is therefore absolutely vital that an agreed set of data accepted by national and international authorities is available in order to avoid confusion. In 1987 the ICRP Publication 51 [7] published a complete set of fluence-to-dose conversion coefficients to be used for the implementation of their recommendations published in 1977. Because of ICRP 60 this set of data must now be revised. In 1992 the ICRP and ICRU had therefore set up a joint task group. Its main objectives were to study the impact of ICRP 60 on data of conversion coefficients between basic physical quantities and primary limiting or operational quantities, to supply a new consistent set of data necessary for the practical implementation of the whole concept and to discuss the relations between the various types of quantities.

One of the limiting quantities in radiation protection for the exposure of persons is effective dose. As this is not a directly measurable quantity, operational quantities had to be created which are measurable. The operational quantities taken for radiation protection are ambient dose equivalent, relating to strongly penetrating radiation (gamma or neutron radiation), or directional dose equivalent, relating to weakly penetrating radiation (alpha and beta radiation). ‘Radiation level’ is defined as ‘dose rate’ in SSR-6 paragraph 233. This gives no hint about which “dose” is meant, but some explanation is provided in paragraphs 233.1 through 233.6 in the Advisory Material TS-G-1.1 [8]. The ‘radiation level’ should be taken as the value of the operational quantity ‘ambient dose equivalent’ or ‘directional dose equivalent’ as appropriate.

There is no technical instrument that has such a complicated response to radiation as living organs have in a human body. Therefore, it is necessary to find instruments that give an estimate of the effective dose. That means that the effective dose as such is no longer measured, but other quantities are measured instead. Suitable technical instruments can measure these so-called operational quantities.

Radiation Response Function [9]

Fluence-to-dose factors presented in ANSI/ANS-6.6.1 are intended for use in calculations of uniform, whole-body exposure resulting from routine operation of a radiation facility.

Does equivalent (H) is the product of the absorbed dose (D) and the average quality factor (Q)

$$H = D \times \bar{Q}$$

In nearly every radiation shielding design or analysis, it is necessary to relate a response, that is, a dose, and instrument response, or some other radiation effect experienced by a target, to the fluence of ionizing radiation. The response function is called by the ICRP a conversion coefficient when relating the fluence to a dosimetric quantity.

In its most general form for any one type of radiation, the response occurs over a sensitive volume and the response may be determined as

$$R = \int_0^\infty dE \int_{4\pi} dV \int_V dV \, \mathcal{R}\,(r, E, \Omega)\Phi(r, E, \Omega)$$

where the volume integration is over the sensitive volume V of the target. The function R(r, E, Ω) is from a physical point of view the expected response that may be attributed to a particle of energy E traveling in direction Ω at point r, per differential unit of path length traveled. The response function is very complicated for a large-volume such as the human body whose various organs and tissues respond quite differently to a given radiation dose. A radiation monitoring instrument is also a very complicated target. For analytical purposed practical problems are addressed by using an idealized response function or conversion coefficient.

Response functions for human as target [9]

Two types of dose quantities have been developed for radiation-protection purposes in occupational and public health. Both types are phantom related, in that they are determined on the basis of physical or mathematical models of the human body. The more precise quantities, called limiting dose quantities, are used by the ICRP and NCRP in their recommendation of dose limits in radiation protection. They are based on anthropomorphic phantoms and represent weighted averages of organ does in human subjects. For radiation protection purposes at doses well below limits for public or occupational exposure there is a need for operational dose quantities. These quantities are based on simple geometric phantoms. They are measurable, in principle, and instruments may be calibrated in their terms.

An analysis of a shielding design estimates a radiation field characterized in terms of flux densities or fluences of photons in the free field that is absent of the human body. The energy spectrum Φ(E) of the fluence is calculated at reference points. The dose quantity R for a phantom representation of the human body is calculated using an appropriate response function as

$$R = \int_0^\infty dE\ \mathcal{R}(E)\Phi(E)$$

where $\mathcal{R}$ is a phantom-related response function and Φ is the fluence energy spectrum, not perturbed by the presence of the phantom. Response function for phantoms are computed for a number of irradiation conditions. At selected points in the phantom, absorbed-dose values are determined. The resulting distributions of absorbed dose and does equivalent throughout the phantom are examined to obtain the maximum value. The response function is that value of either the absorbed does or dose equivalent divided by the fluence of the incident radiation.

Fluence-to-dose factor (he) is the quotient of the effective dose equivalent (He) by fluence Φ at a specific energy

$$h_E = \frac{H_E}{\Phi}$$

Flux-to-dose-rate factors presented in ANSI/ANS 6.1.1-1977 [10] are based on calculations made using multigroup transport calculation to determine the dose-rate distribution in a 30-cm-thick slab phantom for photon energies between 0.02 and 16 MeV.[1] The fluence-to-dose factors presented in ANSI/ANS 6.1.1-1991 are based on recommendation of ECRP, as given in ICRP 51. The ICRP recommendations are based on calculations of dose equivalent in the appropriate organ and tissue regions of male and female anthropomorphic phantoms. Fluence-to-effective dose equivalent factors are presented for male and female phantoms, the two of which are recommended by the ICRP for general radiation protection purposes. Calculations of gamma-ray fluence-to-dose factors have been performed for incident monoenergetic gamma rays having energies from 0.01 to 10.0 MeV [1]

Deep dose equivalent response function is the maximum dose equivalent within a spherical, slab, or cylinder phantom. Ambient dose equivalent is associated with the dose equivalent in soft tissue below a specified point on the body. [9] Effective dose equivalent is analogous to human exposure and dose accumulation in critical organs, which it seeks to accurately depict as a risk-weighted average. The use of effective dose equivalent as the basis for dose calculation provides accurate, unambiguous values within the context of occupational and nonoccupational exposure limits. [1]

Comparison of data from ANSI/ANS 6.1.1-1977 (deep dose equivalent) and ICRP 51 (effective dose equivalent and ambient dose equivalent) show little difference among the fluence-to-dose conversion factors above approximately 100 keV gamma-ray energies. However, below 100 keV the values vary widely. Most gamma-ray fields encountered diminish rapidly in intensity below the Compton buildup region, 100 to 500 keV. Therefore, except in the case of low-energy bremsstrahlung or X-ray source, there is no reason to choose on of the fluence-to-dose conversion factor over another. [1]

ASSESSMENT OF EFFECTIVE DOSE USING FLUENCE-TO-DOSE FACTORS

The model-based protection quantities, equivalent dose and effective dose, are not directly measurable and their values must be assessed using the results of measurements of the related physical quantities in combination with the fluence-to-dose factors. The operational quantities for exposure to external radiation fields are directly measurable. For external radiation of the body, it is possible to define a set of operational quantities for specific external exposure geometries that are measurable in terms of the basic physical quantities 'fluence' and 'air kerma free-in-air'. [11] Monitoring of external exposure by active or passive means is carried out with devices calibrated in terms of the operational quantities. NRC regulatory guidance for review of radioactive material package designs recommends use of an earlier version of the standard, specifically, ANSI/ANS-6.1.1-1977. NUREG-1609 caution that use of the conversions in ANSI/ANS 6.1.1-1991 can result in a significant underestimation of external dose rates as defined by 49 CFR 173.403 and 10 CFR 20.1004. In addition NRC guidance states that dose rates determined with the 1991 standard do not correspond physically to dose rates measured by typical radiation monitoring instruments [12].

Calibration of survey meters is a three step process: (1) a physical quantity such as air kerma or fluence is determined at a reference point, (2) the device being calibrated is placed at this same point in order to determine the response to the reference quantity, and (3) the response of the device appropriate radiation protection quantity is then determined by application of fluence-to-dose factors that relate the physical quantity to the radiation protection quantity (personal, ambient, or directional dose equivalent) [13].

Maximum operational radiation levels

The requirement that the radiation level at the surface of an excepted package should not exceed 5 µSv/h was established in order to ensure that sensitive photographic material will not be damaged and that any radiation dose to members of the public will be insignificant. The radiation level limits inherent in the definition of the categories (and associated labels) have been derived on the basis of assumed package/cargo handling procedures, exposure times for transport workers and exposure times for photographic film [14]. Two basic dose rate limits apply to conveyances carrying radioactive material. The maximum allowed dose rate on the external surface is 2 mSv/h, and the maximum dose rate at 2 m from the external surface is 0.1 mSv/h

There are three main criteria for the segregation of packages containing radioactive material during transport. The first is for radiation control purposes to minimize the dose to personnel and to undeveloped photographic film. For workers, segregation distances are based on a dose limit of 5 mSv per year. For members of the public, they are based on 1 mSv per year. For undeveloped photographic film, the dose limit is 0.1 mSv per consignment of film (paragraph 563 of SSR-6).

NRC standards for radiation protection require the use of deep dose equivalent in place of effective dose equivalent for determining external exposure by measurement with an external personal monitoring device, unless the effective dose equivalent is determined by a dosimetry method approved by the NRC. [15]

CONCLUSIONS

Operational limits for radiation levels specified in the Transport Regulations are historically not based only on occupational and public health, but also take into consideration damage to photographic material. Operational quantities are measurable with survey instruments or estimated using calculation methods in terms of the basic physical quantities ‘fluence’ and ‘air kerma free-in-air’ that are calibrated using standard fluence-to-dose factors. It is important that the conversion factors used for relating fluence to an effective dose for measured and calculated are based on the same dosimetric model, and it is vital that an agreed set of data accepted by national and international authorities is available in order to avoid confusion. MicroShield [16] is a widely used point kernel analysis method and used fluence-to-dose factors from ICRP 51. The fluence-to-dose factors recommended by ANSI-6.1.1-1977 and ANSI-6.1.1-1991 have no relevance to operational limits set to preventing damage to photographic film. Furthermore, there is no reason to choose one of the ANSI-6.1.1-1977 fluence-to-dose conversion factors over the ANSI-6.1.1-1991 since the gamma energies important to shielding design for transportation, greater than 100 keV.

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END NOTES

[1] The 1977 version of the standard was based on the maximum dose equivalent in a 30-cm (diameter) x 60-cm tall, tissue-equivalent cylinder. The 1991 version was based on the quantity recommended in ICRP Publication 26 for the effective dose equivalent. [ANS Issues Clarification on ANSI/ANS-6.1.1-1991, "Neutron and Gamma-Ray Fluence-to-Dose Factors." **(**Nuclear News, **October 2004)]**