THE PRESENT STATE OF NUCLEAR ACCIDENT DOSIMETRY

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1. INTRODUCTION

During the last two decades various types of nuclear accident dosemeters have been developed and tested. Important contributions to this development have been made through a series of intercomparisons organized by the Oak Ridge National Laboratory in the USA (1), and a co-ordinated research programme, including four international intercomparisons, established by the IAEA (2). The IAEA has issued a compendium of neutron spectra in criticality accident dosimetry (3) and is now preparing a comprehensive technical manual on nuclear accident dosimetry (2).

By "nuclear accident dosimetry" is usually meant criticality accident dosimetry, since it is for this purpose that special accident dosemeters have been developed. For other types of radiation accidents, involving, for example, contamination or sealed radioactive sources, the dosimetry methods used routinely for low-level personnel monitoring are usually adequate, provided that dosemeters are worn.

The primary purpose of nuclear accident dosimetry is to provide, in case of an accident, a rapid assessment of radiation doses for the guidance of the medical services in the appropriate treatment of more heavily exposed persons and for reassuring personnel who have been only lightly exposed. A rapid dose assessment is also desirable for public relations reasons. Furthermore, reliable estimates of dose are important for record purposes and of scientific value in studies of the effects of acute radiation exposure in man.

In this paper the present state of nuclear accident dosimetry is discussed on the basis of a critical review of nuclear accident dosemeters prepared by the author (4).

2. NUCLEAR ACCIDENT DOSIMETRY METHODS

Nuclear accident dosimetry systems are generally based on a combination of personnel dosemeters and area dosemeters and, in addition, the induced activity in the body of the exposed person will usually be measured after an accident involving a neutron exposure. Installed neutron dosemeters can contain more components than it is practicable to use in a personnel dosemeter, so area dosemeters can provide more detailed information on the neutron spectrum and consequently a more accurate neutron dose estimation. However, area dosemeters may be subject to recovery difficulties following an accident, and information from personnel dosemeters is generally needed for the transfer of the results obtained with

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area dosemeters to the positions of the exposed persons. Simple personnel dosemeters can provide a fairly rapid dose assessment after an accident, but the use of a simple dosemeter means that greater emphasis is to be placed on the methods of interpretation.

As in low-level personnel dosimetry, a basic problem in criticality accident dosimetry is that generally no single neutron detector is available which adequately covers the whole energy range of interest. Therefore, the determination of the neutron dose received by a person involved in a criticality accident generally requires a knowledge of the neutron spectrum incident on the body. Several nuclear accident dosimetry systems have been designed to provide some spectral information by including at least two neutron detectors with different energy responses.

2.1. Neutron dosemeters

In existing nuclear accident dosimetry systems the measurement of neutron doses is usually based on activation detectors or to a smaller extent on fission fragment track detectors. The main advantages of activation detectors are that they are normally insensitive to beta- and gamma-radiation, and most activation detector foils are inexpensive. Generally, a combination of fast-neutron (threshold), intermediate-energy neutron (resonance) and thermal-neutron detectors is used, and the neutron dose is derived from the induced activities using assumptions on the neutron spectrum.

Like activation detectors, fission fragment track detectors are normally insensitive to beta- and gamma-radiation. A further advantage is that their information content following an accident is not subject to radioactive decay. On the other hand, fission fragment track detectors are more expensive, and the radiotoxicity of the fissile material is a disadvantage which in some countries excludes their use in personnel dosemeters.

Silicon diodes may be utilized for fast-neutron detection in the energy range from about 0.2 to 15 MeV. Their main advantages are a relatively flat energy response, easy and fast evaluation and negligible response to other types of radiation. Their main disadvantage is a limited overall accuracy.

Albedo dosemeters are being increasingly used for routine personnel monitoring, but so far their use in nuclear accident dosimetry systems have been very limited owing to their wide variation in response as a function of neutron energy.

2.2. Gamma dosemeters

The measurement of doses from gamma-rays following an accident will generally be made with the same types of dosemeters as are used for routine personnel monitoring, i.e. film, thermoluminescence and radiophotoluminescence dosemeters. In some systems gamma-ray detectors are incorporated into the nuclear accident dosemeter badge, whereas in others the gamma dose is determined using a routine personnel dosemeter.

2.3. Special nuclear accident dosimetry methods

A chemical nuclear accident dosemeter has been developed by Dvornik and co-workers (5). The dosemeter measures the total (neutron + gamma) absorbed dose.

Analysis of chromosome damage produced in human blood lymphocytes following over-exposure is a useful biological dosimetry technique. In a mixed gamma and neutron radiation field, the gamma and neutron dose components can be estimated from the chromosome aberration yields if the ratio of the gamma to neutron doses is known(6).

If a person who is not wearing a dosemeter is accidentally exposed, the dose may be estimated from chromosome aberration analysis, from the induced activity in the body (e.g. Na-24 in blood or P-32 in hair) or from reactions in items carried by the exposed individual. Chromosome-aberration analysis and the measurement of body sodium activity are well-established methods, but in mixed radiation fields additional information is needed in order to derive gamma and neutron doses. More unusual methods (measurement of induced activity in coins etc.) may provide some information, but difficult interpretation problems are to be expected if methods that are not in routine use or have not been designed and tested for nuclear accident dosimetry must be relied on.

2.4. New possibilities

A recent promising development in fast-neutron dosimetry, which may find application in nuclear accident dosimetry, is the registration of fast-neutron-induced recoil-particle tracks in polycarbonate foils or in other plastics using electrochemical etching.

Further developments in albedo neutron dosimetry could result in an increased application of albedo dosemeters in nuclear accident dosimetry.

Self-irradiation of thermoluminescence phosphors containing elements which form radioactive isotopes after neutron capture could also be utilized in nuclear accident dosimetry, and the use of lyoluminescence dosemeters (7) is a further possibility. However, lyoluminescent materials are still too insensitive to be used for routine personnel monitoring.

3. DISCUSSION AND CONCLUSIONS

The experiences gained at national and international intercomparisons (1,2), and from experimental studies (8) show that even simple personnel criticality dosemeters perform satisfactorily in neutron spectra that are not too degraded from an uncollided fission spectrum. For spectra rich in intermediate-energy neutrons, on the other hand, the performance of simple dosemeters is generally not satisfactory when routine methods of interpretation are used.

Installed neutron dosemeters containing several activation and/or track detectors can provide an estimate of the general shape of the neutron spectrum, so usually they can give a reasonable determination of the neutron dose at the position of the dosemeter. However, in well-moderated spectra, the commonly used activation detectors (In,S,Mg etc.) may be insufficient. The performance may be improved by adding a neptunium track detector to the activation detector set. However, even for this combination the neutron dose may be underestimated by a factor of about 2 in particularly difficult cases (8). Threshold detector systems that include a set of fission-fragment track detectors contained in a B-10 shield can usually give a satisfactory determination of the neutron dose. However, such threshold detector systems are expensive so they are used to only a very limited extent.

Accidental gamma doses can usually be measured with an uncertainty of less than 25%, using the same personnel dosemeters (film, TLD,RPL) as are used in routine personnel monitoring. However, the uncertainty may be increased in a mixed radiation field with a heavily moderated neutron spectrum if appreciable corrections have to be applied to the reading of the gamma dosemeter for the response to thermal neutrons and to the radiations from activated foils in the vicinity of the dosemeter.

The present state of nuclear accident dosimetry is satisfactory in the sense that several nuclear accident dosimetry systems are now available that perform within the criteria proposed by a panel of experts convened by the IAEA in 1969 (9). However, the proper interpretation of the responses of existing nuclear accident dosemeters may require a high degree of expertise, so, in the authors opinion, there is a need for a further development that should aim at simplification. It would be desirable if accidental neutron doses, as is usually the case for the gamma doses, could be measured with the same dosemeters as are used in routine personnel monitoring. Whether or not this simplification will eventually be achieved is dependent on further research and development in personnel neutron dosimetry.

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