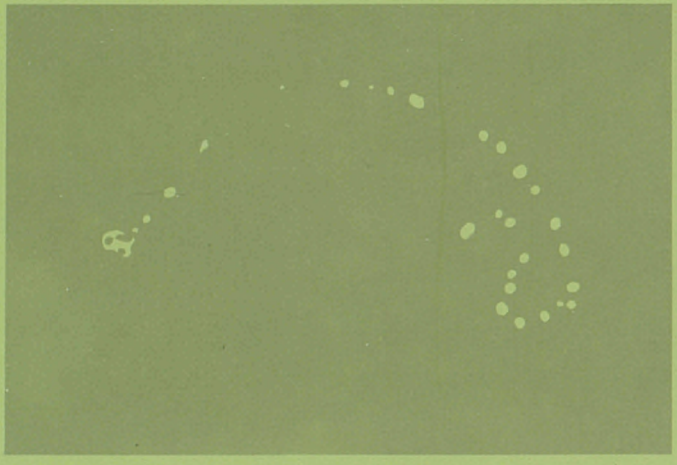
COMMISSION OF THE EUROPEAN COMMUNITIES

Third information seminar on the radiation protection dosimeter intercomparison programme

Beta intercomparison - Grenoble
6 to 8 October 1980



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Radiological protection - 24

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Directorate-General
Employment, Social Affairs and Education

Health and Safety Directorate

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PREFACE

Since 1964 the Health and Safety Directorate of the Commission of the European Communities, in collaboration with competent institutes and laboratories in the Member States, has been conducting intercomparisons of personal dosemeters. The objective of these intercomparisons is to improve monitoring of exposure to ionizing radiation and to establish a common basis for dose assessment. They therefore have a direct bearing on one of the aspects of physical control, as regulated by the Directive of the Council of the European Communities of 15th July 1980 which lays down the basic safety standards for the health protection of the general public and workers against the dangers of ionizing radiation.

This is the first time the Commission has carried out an intercomparison exercise for beta-ray dosemeters.

There is still a great deal of uncertainty in this type of dosimetry and it is desirable to improve measuring techniques, particularly as the number of persons exposed to beta radiation is expected to increase in future as a result of the increasing production and use of radioisotopes in the medical and industrial sectors and because of the increase in the reprocessing of nuclear fuels.

The result of this intercomparison have been discussed at the 'Third Information Seminar of the European Radiation Protection Intercomparison Programme', organised by the Commission of the European Communities in conjunction with the Commissariat à l'Energie Atomique, Centre d'Etudes Nucléaires de Grenoble, from 6 to 8 October 1980.

This document contains the original papers given on the occasion of this seminar together with the comments and conclusions of the editing committee.

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H. ERISKAT E. BENNETT

PART A

The Intercomparison Programme

Session I + II

Chairman : H. DE CHOUDENS

1. AIMS AND HISTORICAL SURVEY

G. PORTAL
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In 1964 the Health and Safety Directorate initiated the dosemeter intercomparison programme together with a Working Party on Personal Dosimetry.

PRINCIPLE

In this programme reference radiations available at several centres are used to irradiate the dosemeters, submitted by the various participants, to known radiation doses. The values of these doses are made known to the participants by the secretary of the Working Party, after all the results of the evaluations have been received. On completion the participants meet to analyse the results and to use the experience gained to assist in the planning of future intercomparison exercises.

NATURE OF THE INTERCOMPARISONS

From the outset two essential principles were agreed to define the line of approach in these intercomparisons. They would be carried out mainly for dosimetric systems normally used in routine operations and dosemeters should be evaluated using the same methods during the intercomparison. Moreover they would be carried out in conditions which would guarantee anonymity. The results would be published with a code number known only to the laboratory concerned. These measures were taken to avoid discouraging any laboratory from taking part.

AIMS

These intercomparisons have a variety of aims. The ultimate objective is to ensure regular surveillance of the quality of personal dosimetry in the various laboratories in the Community.

This presupposes that, during an initial period, technical assistance will be given to the less sophisticated laboratories to encourage them, under the protection of anonymity, to improve their methods. After this has

been achieved a new objective is envisaged namely standardization of performance at Community level. Of course the large laboratories should also benefit from these intercomparisons. The advantages for them would be the stimulating effect of the comparisons and the direct contact with other laboratories, facilitated by the programme. Thus, techniques in all the laboratories may be expected to improve, along with a steady improvement in dosimetry.

Finally, during these exercises, the laboratories which are particularly well equipped as regards reference radiation would be expected to help where possible with the irradiation of the dosemeters submitted by the participants. Obviously this would also give the other participants a chance to periodically recalibrate their own reference systems. In addition the reference laboratories would inevitably make contact with one another during this intercomparison, thus having the opportunity to standardize their own reference installations.

INTERCOMPARISON PROGRAMMES

(a) For photon radiations

The first campaign started in 1964 on a modest scale. At the time there were only a small number of participants and the first irradiations were carried out using only ^{60}Co radioactive sources. The irradiation centres at the time were :

- the PTB in Brunswick,
- the Rijksinstituut in Bilthoven,
- the GSF in Munich,
- the CEA in Fontenay-aux-Roses.

The number of participants has increased over the years. Thus during one of the most recent campaigns 30 participants were involved.

At the same time the radiations have been diversified with the introduction of X-rays and X-ray and gamma-ray mixtures. Initially only film dosemeters were tested. Subsequently, RPL and RTL dosemeters were included and this added a new dimension to the intercomparison.

The analysis of the first campaigns brought to light a number of divergent results, showing that certain laboratories needed Community aid. This aid was effective because the disparities rapidly disappeared. Subsequent advances have been slower but substantial for all laboratories.

An optimum has thus been obtained and the conclusion was that there is no point in maintaining the photon programme at its present level; a campaign every 3 to 4 years should from now on be enough to ensure that standardization of performance is maintained. A new type of radiation might then be introduced.

(b) For neutron radiations

The same type of intercomparison campaign was carried out for neutrons. Two initial campaigns which related exclusively to nuclear emulsions showed that this dosimetric system cannot be used for neutrons with an energy lower than 0.7 MeV. Moreover between 0.7 and 2 MeV the results are not very precise.

During the second campaign the only progress made was by the laboratories that applied correction factors i.e. a correction which is based on an estimate of the neutron energy. As this is not practicable for routine application it was decided to abandon this type of comparison and to replace it by that of albedo dosemeters.

The intercomparisons of this dosemeter type were carried out satisfactorily. For the present the results have shown - for all the participants - the practical limits of this system, in that it can only be used in areas throughout which the neutron spectrum remains constant. Special calibration is required for each particular area. In my opinion these campaigns should be carried out with a different aim than the one referred to at the beginning of this note. They should be devoted to the development of new systems, even if these are not yet in routine use.

(c) For beta radiations

The intercomparison which is the subject of the present seminar was carried out in 1979 as a pilot intercomparison. The aim was to broaden the scope of the intercomparison programme following the preliminary intercomparison between the irradiating laboratories (NPL, PTB, LMRI).

The theme of our meeting today is the continuation of this programme, which will be presented to you by the next speaker.

CONCLUSION

One conclusion is perfectly obvious - i.e. the complete success of the photon intercomparison programme. The performances of the various laboratories have been standardized and there has been a general advance in techniques. An unexpected by-product of this Working Party's activities was the decision to set up a Working Party within the ISO to prepare standards to be used for the production of reference radiation.

Finally, we hope that the neutron and beta intercomparisons will run just as smoothly and provide equally successful results.

2. DESCRIPTION OF THE FIRST INTERCOMPARISON PROGRAMME FOR PERSONAL BETA-RAY DOSEMETERS

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The extensive use of radioactive sources in industry, medicine and research often requires a beta-ray dose assessment for radiation protection purposes. Some improvement in the performance of personal dosemeters for the measurement is required and the Commission of the European Communities therefore decided to organize an intercomparison programm for personal dosemeters irradiated with beta-radiation. As it was the first intercomparison if this kind the number of participants was limited to twelve. Only standard reference beta-radiations given in ISO/DP 6980, 1980 (draft proposal), would be used and the irradiations would be carried out by three primary standard laboratories, the PTB (Germany), NPL (United Kingdom) and LMRI (France).

The radionuclides $^{90}Sr+^{90}Y$, ^{204}Tl and ^{147}Pm were employed. Their maximum and mean energies are given in Table 1. ^{147}Pm was used to irradiate thermoluminescence dosemeters with very thin covers and - in one case only - a film. Participants were instructed to mark dosemeters not to be irradiated with ^{147}Pm.

	Radionuclide		
	$^{90}Sr+^{90}Y$	^{204}Tl	^{147}Pm
Maximum beta-energy at the dosemeter, MeV	2.1	0.68	0.19
Mean beta-energy at the dosemeter, MeV	0.8	0.26	0.07

Table 1 : Maximum and mean energies of the beta radiations used in the intercomparison.

The physical quantity which it was agreed to measure was the absorbed dose D in soft tissue (composition see ICRU report 33, 1980) 70 μm below the surface of a soft tissue equivalent phantom with a density of 1 g cm^{-3} assumed to be an infinitely thick slab (1). The factor for converting D into the absorbed dose in soft tissue averaged over tissue depths between 50 μm and 100 μm was also given for all the irradiations by the primary standard laboratories.

The intercomparison was organized as follows (see Table 2) : Each participant sent 18 personal dosemeters (including 3 control dosemeters) to the PTB and NPL, and 7 personal dosemeters (including 1 control dosemeter) to the LMRI. Participants were allowed to submit more than one type of dosemeter provided for any irradiation they could all be mounted within an area of 10 cm x 10 cm. The PTB and NPL irradiated 5 personal dosemeters for each radionuclide given in Table 1 for every participant and the LMRI irradiated 6 personal dosemeters for each participant with (^{90}Sr + ^{90}Y) beta-radiation. The PTB and NPL informed every participant of the absorbed dose D for two dosemeters per radionuclide. The LMRI gave this information for three irradiations to each participant. By means of the D values obtained from the primary standard laboratories, every participant could determine the calibration factor for the three types of beta-radiation, and thus had a basis for determining the unknown absorbed doses D_{norm}. The results have been evaluated on the basis of the ratio D_{norm}/D_{st}.

Irradiating institute	Number of irradiated dosemeters per participant		
	Radionuclide		
	^{90}Sr + ^{90}Y	^{204}Tl	^{147}Pm
PTB Germany, F.R.	5 (2; 5)	5 (2; 5)	5 (2; 10)
NPL U.K.	5 (2; 5)	5 (2; 5)	5 (2; 10)
LMRI France	6 (1; 5), (1; 10), (1; 20)	-	-

Table 2 : Organization of the intercomparison. The first number in brackets indicates the number of personal dosemeters for which the absorbed dose was communicated by the primary standard laboratories, the second number after the semicolon is the absorbed dose in mGy for these personal dosemeters.

(1) The introduction of the 30 cm diameter soft tissue sphere as the phantom in which the quantity to be measured is defined would have no practical consequences.

3. DESCRIPTION OF THE BETA PERSONAL DOSEMETER IRRADIATIONS AT LMRI, NPL AND PTB

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The schematic arrangement for the irradiations at each laboratory is shown in Fig 1., details of the radionuclide sources and the absorbed dose rates at the calibration planes are given in Tables 1, 2 and 3.

Each laboratory measured for each irradiating source the absorbed dose rate to tissue at the calibration plane using its primary standard; the LMRI and PTB standards are air-filled extrapolation ionization chambers in semi-inifite tissue-equivalent phantoms (Böhm, 1976) and measure directly the absorbed dose to tissue. The NPL standard is a parallel-plate ionization chamber in which a relatively large volume of air is defined by a thin plastic film electrode system (Owen, 1972). The standard measured absorbed dose to air and conversion factors are necessary to derive absorbed dose to tissue (BCRU, 1977). The absorbed dose to tissue was specified at two depths, at 7 mg cm^{-2} ($\dot{D}$) and averaged between 5 and 10 mg cm^{-2} ($\bar{D}$) below the tissue surface. Independent intercomparisons between LMRI and PTB, and PTB and NPL have shown that agreement between the irradiating laboratories for absorbed dose to tissue measurements is about 1% for Sr + Y-90, 2% for Tl-204 and 7% for Pm-147.

As shown in Fig 1, most sources were used with beam-flattening filters to produce an area of uniform dose rate about 7 cm in radius at the calibration plane. The same filter constructions were used by all three laboratories; the filter material was polyethylene terephthalate (Melinex, Mylar, Hostaphan).

Sr + Y-90	3 concentric discs each 25 mg cm^{-2} thick and radii of 2, 3 and 5 cm
TL-204	2 concentric discs one 25 mg cm^{-2} thick, 2.75 cm radius and one 7 mg cm^{-2} thick and 4 cm radius
Pm-147	1 disc 14 mg cm^{-2} thick, 5 cm radius with a 0.975 cm radius hole at centre.

The dosemeters were placed on a 1 cm thick methylmethacrylate sheet (Perspex, Lucite, Plexiglas). The dosemeters were irradiated singly at LMRI, confined within a 5 cm radius of the source axis at PTB and within a 7 cm radius at NPL. The dose uniformity within the restricted areas at NPL and PTB was assessed as better than $\pm$ 4%. The dosemeters were separated by at least 1 cm at NPL and 0.5 cm at PTB. The irradiation times varied from 1 min to several days.

The primary standard measurements required several corrections to derive the dose given to the dosemeters; radioactive decay for the time between the measurement and the irradiation, corrections for changes from the measurement conditions in the air path density between the source and the dosemeter, and corrections for the distance between the calibration plane and the dosemeter reference plane. The air path density changes were most significant in altering the attenuation of low energy beta radiation, the corrections for Pm-147 being typically 10%. At the PTB the dosemeter reference plane was adjusted to be at the calibration plane and no corrections were necessary; at LMRI and NPL corrections were applied being a maximum of about 20% for Pm-147 irradiations of the thickest dosemeters.

The total uncertainty, at the 95% confidence level, of the stated dose delivered to a dosemeter was the root of the quadratic sum of the statistical and systematic uncertainties due to the primary standard measurement of the source dose rate, the irradiation of the dosemeter and the correction and conversion factors. The total uncertainties are given in Tables 1, 2 and 3.

The majority of dosemeters were sent through the post; the NPL and PTB irradiations took place during February-March 1979 and the LMRI irradiations in early June. The dosemeters were returned to the participants as soon as possible after irradiation together with the unirradiated control dosemeters whose purpose was to record any unintentional irradiation for example in the post.

The irradiation of the personal dosemeters in the PTB was performed with a commercially available beta-ray secondary standard, the prototype of which was developed in the PTB as illustrated in Fig 2 (Böhm, 1979).

The personal dosemeters, mounted on a 1 cm thick perspex plate were irradiated by a beta source screwed to the jig of the secondary standard. The irradiation time was digitally set at the control unit within the range of 1 s to 99999 S. During the time t the beta source was exposed by the

shutter, 11 ms were needed for closing the shutter. The relative uncertainty of the irradiation time was $(10^{-4} + 0.005/t)$, with t in seconds.

A diagram of the calibration and irradiation equipment used at LMRI is shown in Fig 3. The 1 cm perspex sheet which formed the base and calibration plane for the dosemeters being irradiated was supported on the front window F of the extrapolation chamber.

A diagramatic representation of the source and beam-flattening filter in use for irradiation of dosemeters in NPL is shown in Fig 4.

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	^{90}Sr + ^{90}Y source from		
	PTB	NPL	LMRI
Nominal activity in MBq (in mCi)	1850 (50)	40 (1)	5550 (150)
Thickness of the inactive silver foil "window" in mg cm^{-2}	50 ± 5	50	-
Protection against corrosion	0.1 mm stainless steel	gold flashing	0.15 mm stainless steel
Mean beta particle energy in MeV	0.8	0.8	0.8
Beam flattening filter	No	Yes	Yes
Source to calibration plane distance in cm	50	30	30
Correction for distance between calibration plane and dosemeter reference plane in % / mm	-	0.7	0.74
Absorbed dose rate $\dot{D}$ at the calibration plane in mGy h^{-1}	104	2.4	359
Factor for converting $\dot{D}$ into $\dot{D}$	1.003	1.00	1.00
Total relative uncertainty of the stated dose D_{st} in %	2.2	4.2	2.8

Table 1 : Characteristics of the ^{90}Sr + ^{90}Y sources employed

	^{204}Tl source from	
	PTB	NPL
Nominal activity in MBq (in mCi)	16.6 (0.45)	20 (0.5)
Thickness of the inactive silver foil "window" in mg cm^{-2}	20 ± 3	20
Protection against corrosion	1 µm thin gold flashing	gold flashing
Mean beta particle energy in MeV	0.24	0.24
Beam flattening filter	Yes	Yes
Source to calibration plane distance in cm	30	30
Correction for distance between calibration plane and dosemeter reference plane in % / mm	-	0.7
Absorbed dose rate $\dot{D}$ at the calibration plane in mGy h^{-1}	0.90	2.1
Factor for converting $\dot{D}$ into $\bar{\dot{D}}$	0.993	0.994
Total relative uncertainty of the stated dose D_{st} in %	3.5	4.3

Table 2 : Characteristics of the ^{204}Tl sources employed

	^{147}Pm source from	
	PTB	NPL
Nominal activity in MBq (in mCi)	440 (12)	800 (25)
Thickness of the inactive silver foil "window" in mg cm^{-2}	5 ± 1	3
Protection against corrosion	0.5 mg cm^{-2} electroplated nickel	0.5 mg cm^{-2} electroplated nickel
Mean beta particle energy in MeV	0.06	0.06
Beam flattening filter	Yes	Yes
Source to calibration plane distance in cm	20	20
Correction for distance between calibration plane and dosemeter reference plane in % / mm	-	2.3
Absorbed dose rate $\dot{D}$ at the calibration plane in mGy h^{-1}	0.3	0.7
Factor for converting $\dot{D}$ into $\dot{D}$	0.941	0.944
Total relative uncertainty of the stated dose D_{st} in %	7.4	9.1

Table 3 : Characteristics of the ^{147}Pm sources employed

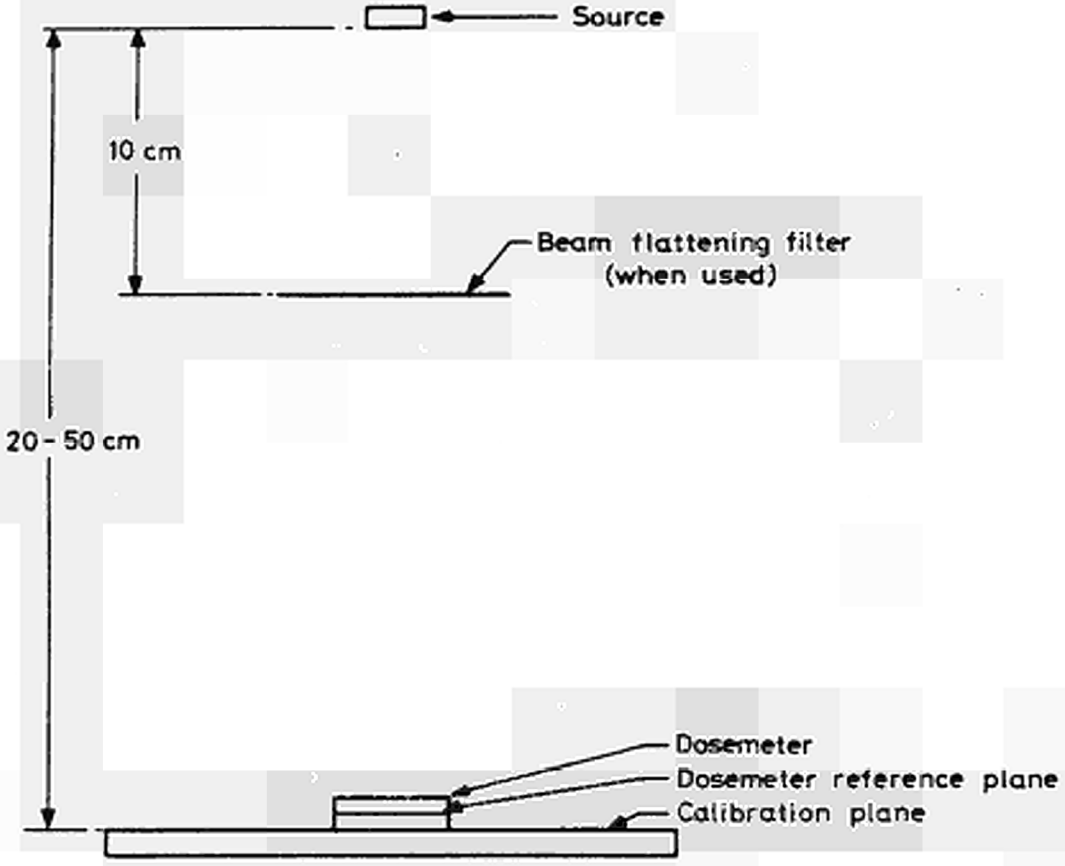


FIG 1. SCHEMATIC ARRANGEMENT OF IRRADIATION CONDITIONS

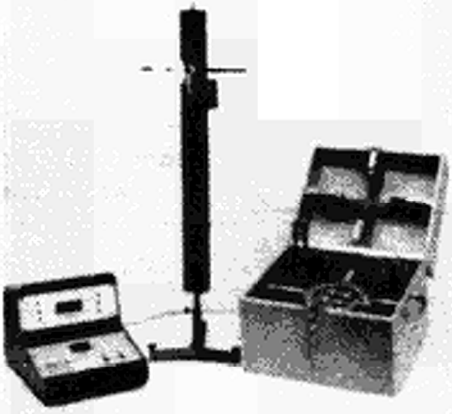


Figure 2: Components of the beta-ray secondary standard.
Left: Control unit for remote operation of the shutter and for digital preselection of the irradiation time.
Middle: Jig for β-source support with shutter and beam flattening filter.
Right: Wooden box for transport and storing of a special container for 4 β-sources, a handling tool for manipulating the sources, and 4 spacing bars.

Figure 3

LMRI variable cavity ionization chamber

A micrometic device to measure the distance between the source and the reference plane

B beam flattening filter

C collection electrode

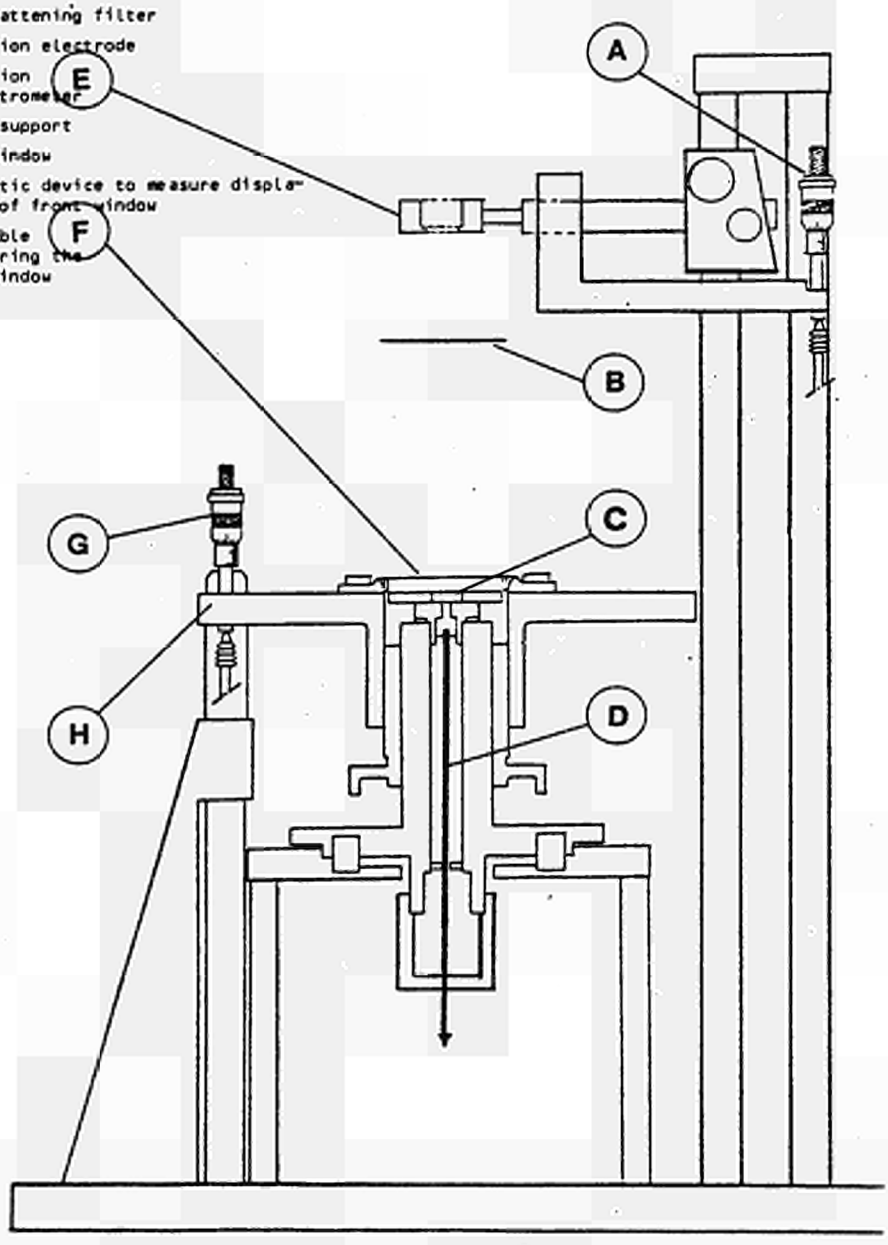
D connection to electrometer

E source support

F front window

G micrometic device to measure displacement of front window

H adjustable jig bearing the front window



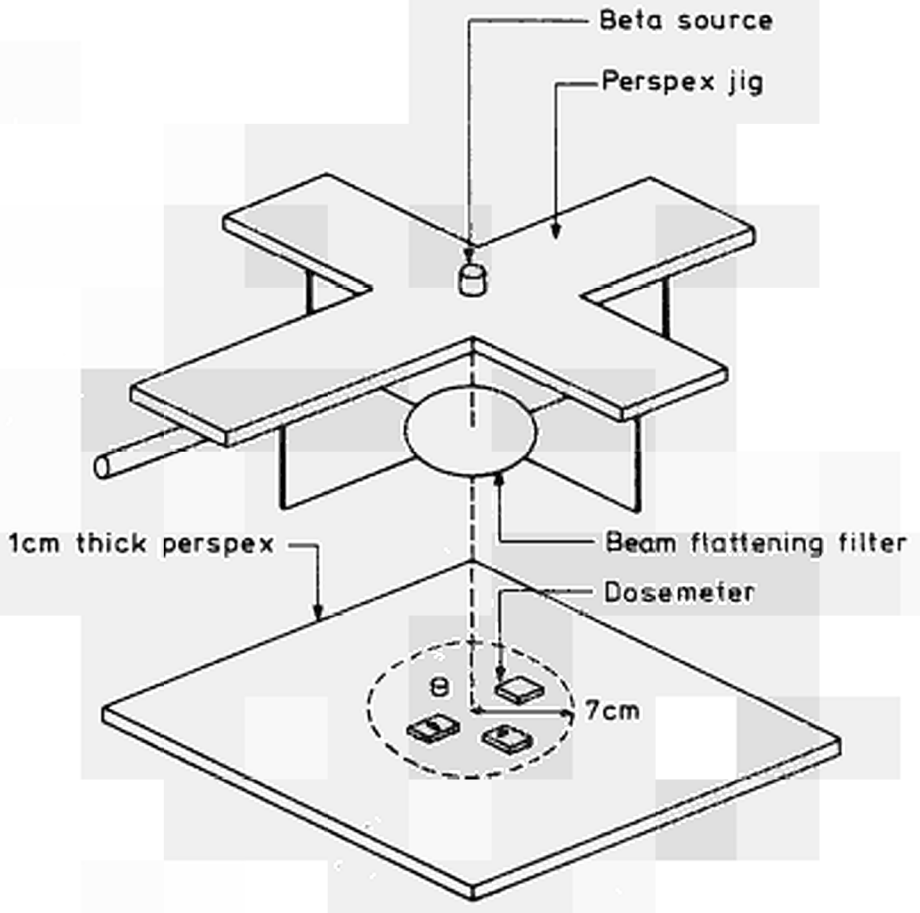


FIG. 4. NPL BETA IRRADIATIONS - CEC 1979

4. COMMENTS BY PARTICIPANTS

(a) Commissariat à l'Energie Atomique, Fontenay-aux-Roses - F. BERMANN

Our measurements were carried out with three types of dosemeters :

- films in multifilter badges for personal dosimetry;
- lithium fluoride pellets as used for personal dosimetry;
- thermoluminescent lithium borate, which is still in the experimental stage.

Table 1 shows the results obtained, as compared with the stated doses, both for the film dosemeters and the thermoluminescent dosemeters. On the whole there was close agreement which concealed a number of disparities detailed below.

1) Response of the thermoluminescent dosemeters

Frequently the two types of thermoluminescent dosemeters give systematically different results (Table 2).

N.B. The results supplied to the CEC correspond to the sum of the lithium fluoride and lithium borate results.

For the same beta source the same type of dosemeter sometimes gives systematically different results depending on the irradiating laboratory :

e.g. 204 Tl $\frac{NPL}{PTB} \simeq 1.2$ (cf. Table 2)

2) Beta response of the film dosemeter under PSI

Firstly it should be noted that this type of dosemeter is mainly used to measure penetrating radiation (X-rays, gamma rays, thermal neutrons) with a view to monitoring whole body irradiation.

In fact the dosemeter is very badly suited to measuring beta radiation, mainly because of the casing covering the film and the dosemeter itself ($\simeq$ 20 mg/cm^2). Measurement of ^{147}Pm irradiation was possible only via the associated X-rays.

The film used is made up of three emulsions of different sensitivities which are placed back to back on the same support. Accordingly, the

emulsion facing the source has a higher sensitivity than the emulsion placed behind it (Table 3).

Very probably, a further factor is involved. The response of a given emulsion varies significantly with the irradiation laboratory for the low-energy beta sources ^{204}Tl and ^{147}Pm (Table 4). This variation indicates that the sensitivity is dependent to some extent on the spectrum of these sources and on the contribution of the associated bremsstrahlung.

For all these reasons the results we have given have had to be normalized using the results obtained with dosemeters irradiated with known doses. In fact, as regards the film dosemeter, the main value of this intercomparison is no doubt that it improves our knowledge of its response to beta radiation and that it has shown us how unreliable it is for energy sources lower than 1 MeV.

CONCLUSION

Present practice is to evaluate beta doses on the basis of calibration via gamma rays from a ^{60}Co source, possibly corrected by a factor evaluated once and for all on the basis of irradiation by beta rays from a $^{90}Sr + ^{90}Y$ source.

This intercomparison has shown us the limits of precision which can be obtained even when using specific calibrations carried out with beta sources identical to the irradiation sources. Table 5 shows the coefficients used to compute the beta doses from the responses of the thermoluminescent dosemeters, expressed as a function of calibration by gamma rays from a ^{60}Co source (D_{ap}). These coefficients were obtained from the known intercomparison doses. They were found to be constant and close to 1 for beta radiation from $^{90}Sr + ^{90}Y$ and to vary greatly from one source to another for the lowest energies.

	Films	TLD
$^{90}Sr + ^{90}Y$	0.93 ± 0.16	0.90 ± 0.06
^{204}Tl	1.10 ± 0.16	0.97 ± 0.09
^{147}Pm	1.05 ± 0.13	0.97 ± 0.19

Table 1 : Combined results (D_{norm}/D_{stated})

Source	Dosemeter	NPL	PTB	LMRI
$^{90}Sr + ^{90}Y$	FLi	1.07 ± 0.02	0.97 ± 0.11	0.97 ± 0.01
	$Li_2B_4O_7$	0.86 ± 0.03	0.89 ± 0.08	0.97 ± 0.03
^{204}Tl	FLi	1.17 ± 0.05	1.01 ± 0.12	-
	$Li_2B_4O_7$	1.05 ± 0.02	0.86 ± 0.05	-
^{147}Pm	FLi	1.04 ± 0.12	0.96 ± 0.14	-
	$Li_2B_4O_7$	0.84 ± 0.16	1.09 ± 0.14	-

Table 2 : Thermoluminescent dosemeters (D_{norm}/D_{stated})

	NPL	PTB	LMRI	CEN G
1st emulsion	1.45 ± 0.12	1.71 ± 0.25	1.60	1.27 ± 0.09
2nd emulsion	0.82 ± 0.07	0.67 ± 0.09	0.88 ± 0.09	0.75 ± 0.08

Table 3 : Response of the film dosemeter to beta radiation from $^{90}Sr + ^{90}Y$; $\left\{\frac{D_{window} - D_{Al}}{D_{stated}}\right\}$

N.B. The responses are expressed in terms of a calibration by a ^{60}Co gamma source (D_{ap}).

	NPL	PTB
^{204}Tl	0.41 ± 0.04	0.53 ± 0.08
^{147}Pm	0.13 ± 0.02	0.30 ± 0.04

Table 4 : Response of the first emulsion of the film dosemeter (D_{ap}/D_{stated})

Source	Dosemeter	NPL	PTB	LMRI
$^{90}Sr + ^{90}Y$	FLi	1.04	1.09	1.05
	$Li_2B_4O_7$	1.07	1.07	1.00
^{204}Tl	FLi	3.63	3.31	
	$Li_2B_4O_7$	3.22	2.52	
^{147}Pm	FLi	30.4	14.7	
	$Li_2B_4O_7$	10.0	14.3	

Table 5 : Calibration coefficients determined on the basis of the known doses (D_{stated}/D_{ap})

4. COMMENTS BY PARTICIPANTS

(b) CEA - Centre d'Etudes Nucléaires, Grenoble - Y. HERBAUT

Our measurements were carried out with four types of dosemeter :

- multi-screen film dosemeter;
- three types of lithium fluoride thermoluminescent dosemeters of different thicknesses (from 30 $mg.cm^{-2}$ to 230 $mg.cm^{-2}$).

This intercomparison enabled us to evaluate the reproducibility and the response of these dosemeters in respect of the different beta radiation energies.

The results are presented in Tables 1 and 2. D_{ap} represents the dosemeter reading for dosemeters calibrated with a ^{60}Co photon source.

The films and the thermoluminescent dosemeter types 1 and 3 cannot measure beta radiation from a ^{147}Pm source as they are too thick.

In the case of the type 2 thermoluminescent dosemeter, the measurement uncertainties for ^{147}Pm are very high because of the effects of the variation of dosemeter thickness on dosemeter response to low-energy beta radiation.

Source	Film	TLD type 1	TLD type 2	TLD type 3
$^{90}Sr + ^{90}Y$	0.97 ± 0.05	0.93 ± 0.09	1.03 ± 0.06	0.94 ± 0.15
^{204}Tl	0.95 ± 0.09	0.99 ± 0.06	1.03 ± 0.13	1.02 ± 0.14
^{147}Pm			0.96 ± 0.28	

Table 1 : $\frac{D_{norm}}{D_{st}}$ for the different dosemeters

Source	TLD type 1	TLD type 2
$^{90}Sr + ^{90}Y$	0.97 ± 0.1	0.94 ± 0.03
^{204}Tl	0.30 ± 0.01	0.80 ± 0.10
^{147}Pm		0.08 ± 0.02

Table 2 : Response of the thermoluminescent dosemeters (D_{ap}/D_{st})

4. COMMENTS BY PARTICIPANTS

(c) CEGB Berkeley Nuclear Laboratories - I.M.G. THOMPSON

Assessment of the beta doses from the films are made using the following equation :

$D_B = F(O-P)$ where O is the apparent dose in the "open window" and P is the apparent dose in the "plastic filter" (300 $mg.cm^{-2}$)

F is the energy correction factor whose value depends on the ratio R where

$R = \frac{O-P}{PW-P}$ where PW is the apparent dose in the "plastic window" (50 $mg.cm^{-2}$)

The relationship between F and R is determined from the data and graph in BCS 0821, 1977 (*), details of which are attached. During routine estimations of beta doses this graph is described by a quadratic relationship, except when $R \leq 2.2$ when F is taken to equal 1.05. The original D_{eva} results for the $^{90}Sr/^{90}Y$ irradiations have been calculated using this value of F, i.e. as per the British Calibration Service recommendations. For the $^{90}Sr/^{90}Y$ irradiations R had a mean value of 1.03 for the NPL irradiations, 1.04 for the PTB irradiations and 1.11 for the LMRI irradiations. The BCS curve, for the combined emulsions, shown in the attached notes is only plotted for R values exceeding 1.3. If this curve is extrapolated back to the above three values of R then new values of F are obtained which have a higher value than 1.05.

For the $^{90}Sr/^{90}Y$ irradiations these higher F factors have been used to calculate new values of the dose, $D_{eva,new}$, and these are listed below together with the original estimates D_{eva}.

Further, it should be noted that the BCS data assumes that the badge is irradiated at 35° whereas the intercomparison was performed at normal, i.e. 0°, incidence.

(*) Supplementary Criteria for Laboratory Approval. Provision of Personal Dosimetry Services using Film Dosemeter for Beta, Gamma, X- and Thermal Neutron Radiations - British Calibration Service Publication 0821, October 1977.

Irradiation Lab	Absorbed doses at 7 mg/cm^2, mGy		
	D_{st}	D_{eva}	$D_{eva,new}$
NPL	4.95	3.4	4.56
	4.95	3.3	4.48
	2.47	1.7	2.37
	49.5	-	-
	21.0	16.2	21.6
PTB	5	5.5	7.27
	5	5.6	7.52
	2.4	3.5	4.6
	16	12.3	16.3
	100	130	-
LMRI	6.2	3.3	4.48
	10.5	8.6	8.86
	21.0	Films damaged on receiot (when arrived at BNL they were already damaged)	

It was thought that the inconsistent results for the PTB irradiation compared with LMRI and NPL might be due to the different irradiation conditions used at the PTB, this could result in the beta spectrum having a different ^{90}Sr to ^{90}Y ratio.

BETA RADIATION 0.5 TO 3.5 MeV | Burt, AK and Smith, JW Film dosimetry with the AERE/RPS film holder UKAEA report 6156, 1972

$$\text{Dose equivalent in rems} = F\left[D_{open} - D_{thick}\right]$$

The factor F is related to the ratio R, where

$$R = \left[\frac{D_{open} - D_{thick}}{D_{thin} - D_{thick}}\right]$$

The relation between F and R is given in Figure 1 and may be described by the quadratic

$$F = A + BR + CR^2$$

over the range $2.2 < R < 7.0$, where $A = 0.070$, $B = 0.510$, $C = -0.029$

It should be noted that $F = 1.05$ when $R \leq 2.2$

and $F = 2.20$ when $R \geq 7.0$

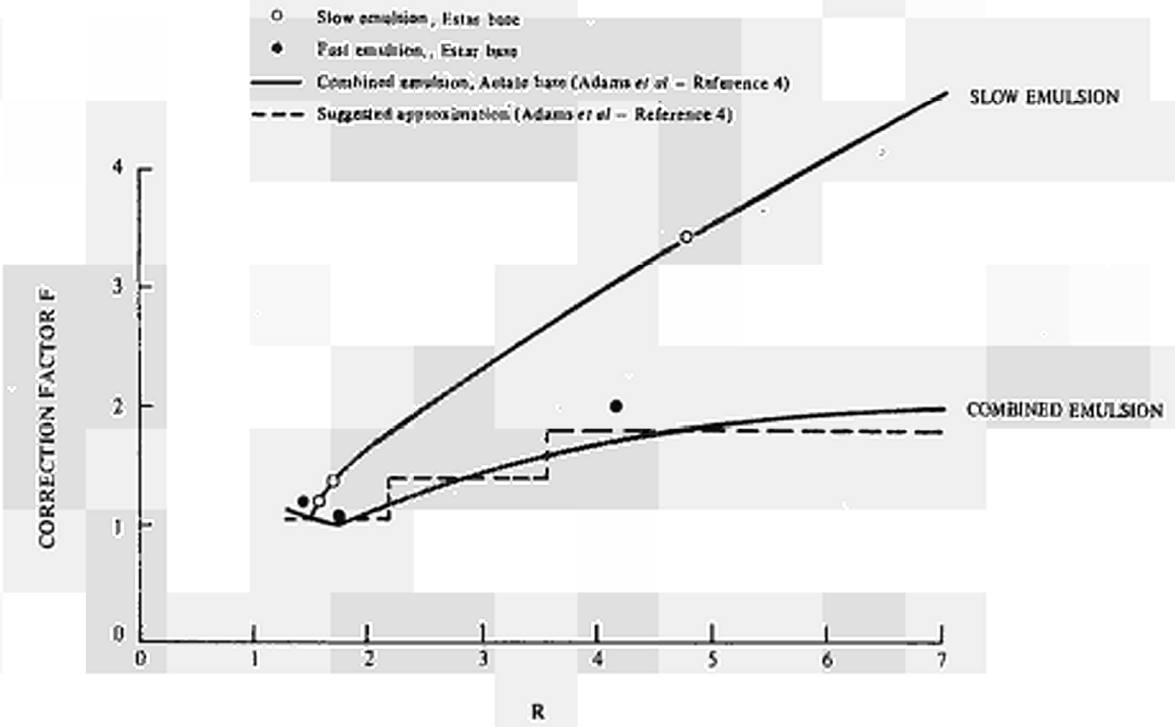


Figure 1. The relationship between F and R for beta radiation

4. COMMENTS BY PARTICIPANTS

(d) Kernforschungsanlage, Jülich - M. HEINZELMANN

DOSEMETER EVALUATION AND RESULT

The beta dosemeter intercomparison programme of the Commission of the European Communities was implemented at a time when the Health Physics Division of the Jülich Nuclear Research Centre started tests for determining the dose of beta radiation using TLD. We were therefore pleased, on the one hand, to be able to participate in an international comparison programme at such an early stage while, on the other hand, we had gathered little experience yet with beta dosimetry using TLD.

For our experiments, we used TLD-100 ribbons of the size 0.125" x 0.125" x 0.015". The response of these dosemeters to beta radiation is known to depend on the energy of radiation. Since, however, our aim is to arrive at a possibly energy-independent dose determination, we tried to obtain additional information about the radiation field by using several TLD's in one dosemeter. For our initial experiments, we used three TLD's arranged behind one another and covered with a foil of 1 mg/cm^2 (Fig. 1). From the readings of these three TLD's, we wanted to gain information about the energy of beta radiation. Our measurements revealed that the third TLD virtually did not provide any additional information on the radiation field, so that we only used the readings D_1 and D_2 of the first two TLD's for our dosemeter evaluation.

Our dosemeters were calibrated on the secondary standard developed by PTB (Physikalisch-Technische Bundesanstalt), and this calibration was used first of all to evaluate the dosemeters irradiated under the intercomparison programme. Since in each case the dose was known for two of the dosemeters irradiated at one location with the radiation of one nuclide, the values obtained initially could still be corrected. For the dosemeters irradiated with a known dose, the ratio of dose in tissue at 7 mg/cm^2 to evaluated dose was established, and the results of the dosemeters irradiated with an unknown dose were corrected by multiplication by this ratio. Correction was only required for the Sr-90 and Pm-147 irradiations at NPL and for the Sr-90 irradiation at LMRI. As the LMRI data were received later, the correction for the Sr-90 irradiation at LMRI was carried out after our

results had been sent to Euratom. The results according to our own calibration were about 13% higher than the values of NPL for the Sr-90 irradiation, about 6% higher than the values of LMRI for the Sr-90 irradiation and about 15% lower than the values of NPL for the Pm-147 irradiation.

Among our results, two values deviated substantially from the actual value. However, these were due to computational errors in evaluation. The dosemeters had indicated the dose correctly. After correction of these values, our results differ by not more than $\pm$ 5% from the actual value in 57% of the cases. For 29% of the results, the deviation from the actual value was between $\pm$ 5% and $\pm$ 10%, and only for 14% was the deviation higher. The maximum deviation from the actual value amounted to 20%. In view of the fact that this is a first intercomparison, we are satisfied with our results.

DISCUSSION

The results of dosemeters irradiated by PTB under the intercomparison programme did not have to be corrected after evaluation with the aid of our own calibration. This was to be expected, since we use the secondary standard developed by PTB for our calibrations. The results of dosemeters irradiated with Pm-147 at the NPL had to be corrected by 15% on account of the results obtained for dosemeters irradiated with known doses. According to Owen, the preliminary results of the NPL-PTB intercomparison differ by 8% for the Pm-147 source. Our correction was twice as high as the differences in the NPL-PTB intercomparison. The reason for the requirement of a correction of our results for the Sr-90 irradiations at NPL and LMRI is not known. Perhaps, the depth doses distribution for irradiations at the Sr-90 source of NPL and LMRI differs from that at the Sr-90 source of PTB.

From the ratio $D_1 : D_2$ of the reading of the first TLD to the reading of the second TLD, we may gain information about the energy of beta radiation. However, the ratio varied more than expected for irradiations on one and the same nuclide. This was caused by the fact that we had not selected our dosemeters with sufficient care. The weight of dosemeters of 0.015" thickness is not always the same. This applies to both the individual TLD's of one shipment and the mean weight of TLD's from various shipments. In the case of 100 dosemeters of one shipment, the weight of the individual dosemeters varied between 6.7 and 9.1 mg. The mean dosemeter weight for one shipment was 8.5 mg, whereas it amounted to 10.1 mg for a second shipment.

These differences in weight influence the dosemeter sensitivity and, in particular, the ratio $D_1 : D_2$ of the readings of two dosemeters arranged behind each other during irradiation. If, for instance, two TLD's are arranged behind each other and irradiated in front of the Tl-204 source (Fig 2), the ratio of the readings $D_1 : D_2$ is 5.9 when the first TLD weighs 9.1 mg and the second TLD 8.1 mg. However, the ratio $D_1 : D_2$ is only 4.2 when the first TLD weighs 8.2 mg and the second TLD 9.1 mg.

FURTHER RESULTS WITH DOSEMETERS CONTAINING SEVERAL TLD's

As mentioned above, we only started with our tests on beta dosimetry with TLD's at the time of the intercomparison programme, and in the following, we will briefly report on the experience with and advancement of our dosemeters. From the ratio of the readings $D_1 : D_2$ of the two TLD's in the dosemeter and from the value of D_1, the dose in tissue at 7 mg/cm^2 depth can be clearly determined for radiation of the nuclides Pm-147, Tl-204 and Sr-90 (Fig 3). However, a dosemeter should not only be suitable for these nuclides, but also for more general application, for instance also in mixed beta-gamma radiation fields. As the response of TLD's to gamma-radiation is known, the ratios $D_0 : D_1$ (D_0 = dose in tissue at 7 mg/cm^2 depth) and $D_1 : D_2$ can be calculated for mixed radiation fields composed of gamma- and Tl-204-beta-radiation, gamma- and Pm-147-beta-radiation, as well as Tl-204 and Pm-147-beta-radiation. The results of these calculations are shown in Fig 3, and it may be seen that a value of a ratio $D_1 : D_2$ may be associated with two very different values of $D_0 : D_1$. This means that, in unknown radiation fields, the ratio $D_1 : D_2$ can no longer be used to clearly determine the dose. The dosemeters used for the intercomparison programme are not suited in mixed radiation fields.

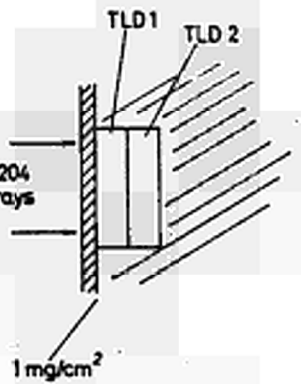
For further experiments, we arranged the TLD's side by side (Fig 4) and added a thin absorber in front of one of the TLD's. The results of measurements with these dosemeters on beta emitters of the secondary standard and calculations for mixed radiation fields are shown in Fig 5. For these dosemeters, a value of $D_1 : D_2$ is only associated in a first approximation to one value of $D_0 : D_1$. Such a dosemeter would be suited for measurements in the mixed radiation fields considered to date.

We then extended our measurements to cover measurements on a C-14 radiation source, i.e. a beta emitter with even lower maximum energy. For this radiation, the ratio $D_0 : D_1$ is smaller than for Pm-147 radiation. We

did not carry out calculations for mixed radiation fields with C-14-beta-radiation. However, it is obvious that a value of $D_1 : D_2$ will no longer be associated with only one value of $D_0 : D_1$ when considering C-14-beta-radiation.

The reason why the ratio $D_0 : D_1$ for C-14-beta-radiation is smaller than the corresponding ratio for Pm-147-beta radiation may be easily explained. The absorber foil in front of the first TLD is only 1 mg/cm^2 thick. Between 1 and 7 mg/cm^2, C-14-beta radiation is attenuated considerably more than Pm-147-beta radiation, so that there is a higher resoonse of the dosemeter to C-14-beta radiation for doses in tissue at 7 mg/cm^2 depth.

By using a foil thicker than 1 mg/cm^2 in front of the first TLD (Fig 4), one may easily succeed in obtaining a higher ratio of $D_0 : D_1$ for C-14-beta radiation than for Pm-147-beta radiation. Relevant experiments have been performed. According to initial preliminary measurements and calculations, it will be possible to clearly determine the dose in mixed radiation fields including C-14-beta radiation.



$\frac{D_1}{D_2} = 5.9$ if TLD 1 has a mass of 9.1 mg and TLD 2 has a mass of 8.1 mg

$\frac{D_1}{D_2} = 4.2$ if TLD 1 has a mass of 8.2 mg and TLD 2 has a mass of 9.2 mg

Fig. 2: Ratio of dosimeter readings for different masses of the dosimeters

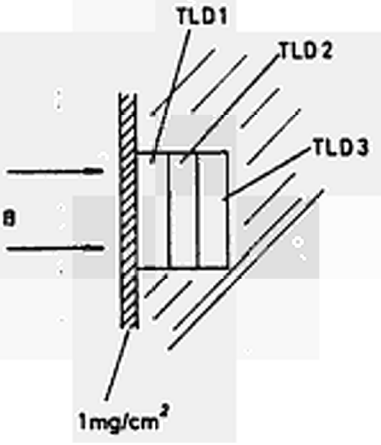


Fig. 1: Construction of the dosimeter

TLD 1 reading D_1
TLD 2 reading D_2
TLD 3 reading D_3

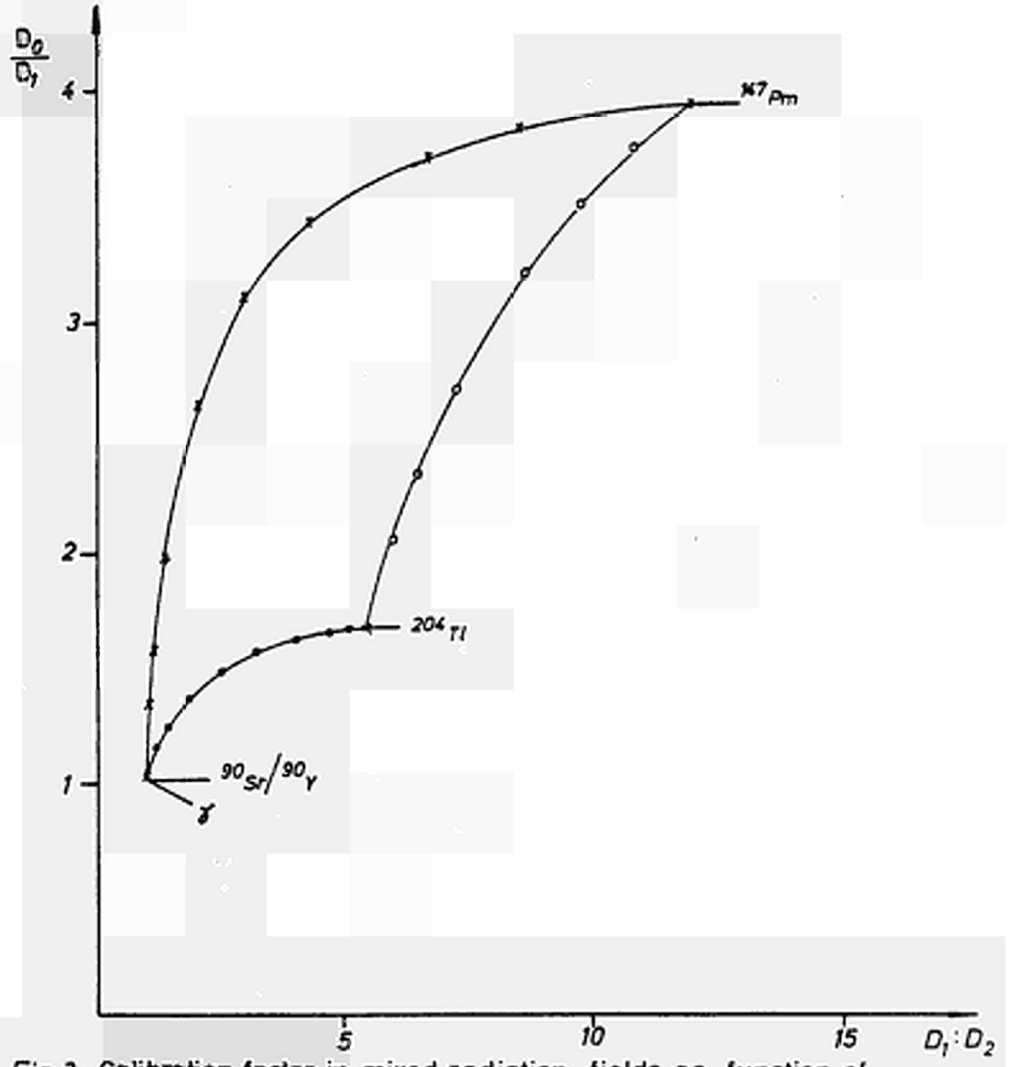


Fig. 3: Calibration factor in mixed radiation fields as function of the ratio $D_1 : D_2$
2 TLD behind one another
D_0 = Absorbed dose in tissue at 7mg/cm^2
D_1 = Reading of the first TLD
D_2 = Reading of the second TLD

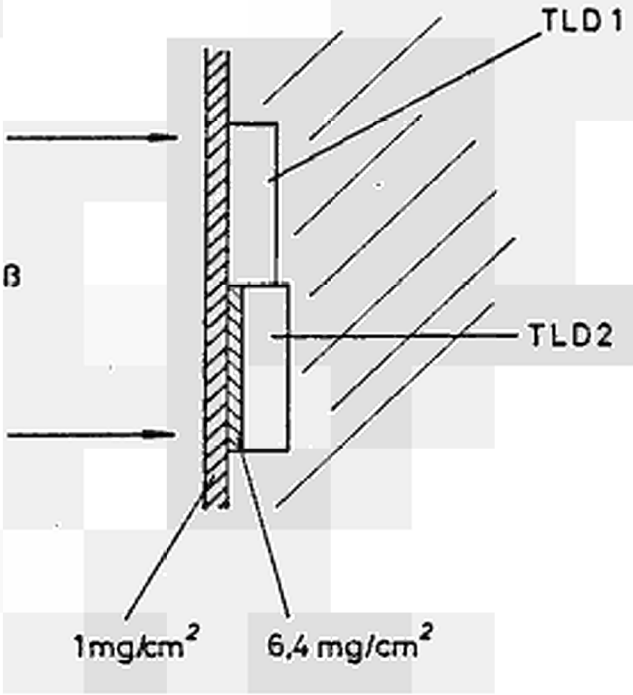


Fig. 4: New construction of the dosimeter

TLD 1 reading D_1

TLD 2 reading D_2

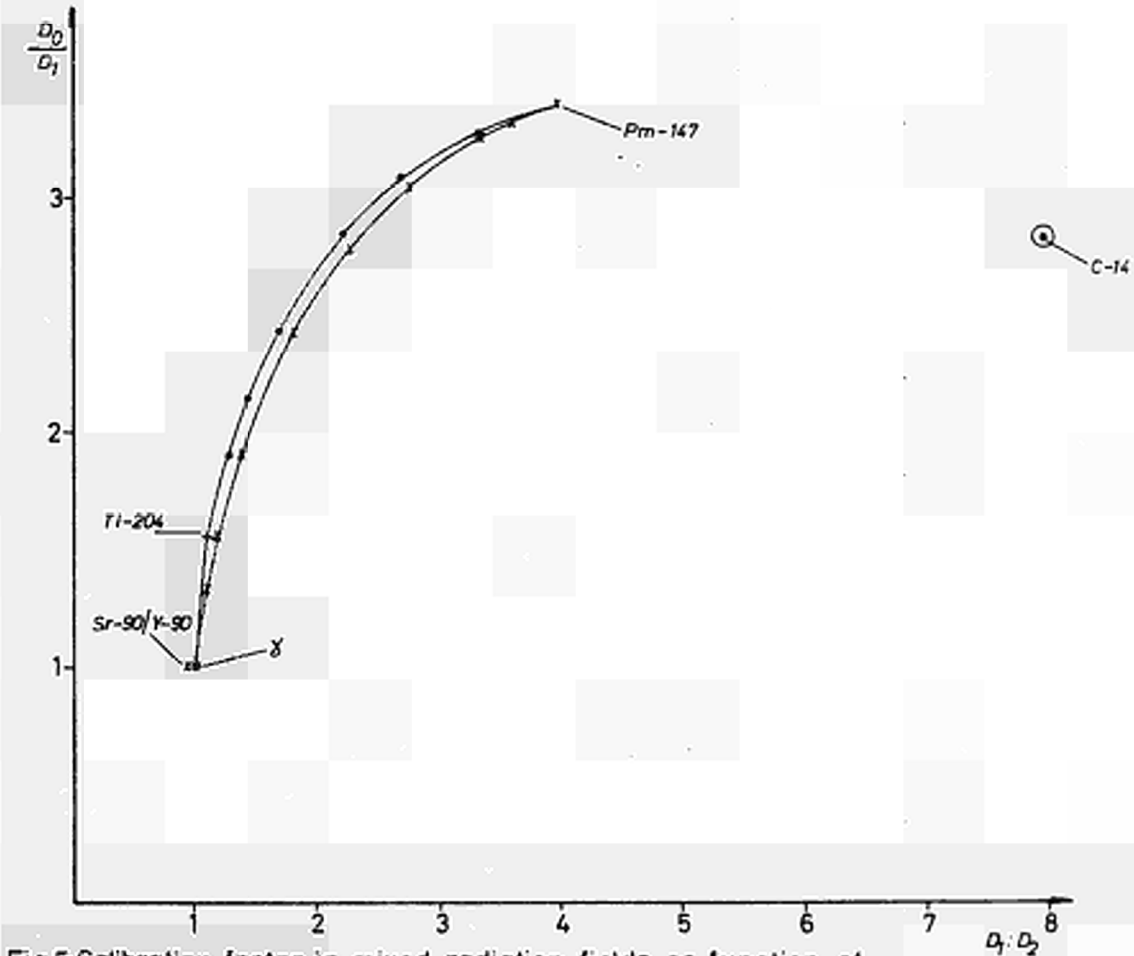


Fig.5: Calibration factor in mixed radiation fields as function of the ratio $D_1 : D_2$

2 TLD side by side

D_0 = Absorbed dose in tissue at 7 mg/cm^2

D_1 = Reading of TLD 1 behind an absorber of 1mg/cm^2

D_2 = Reading of TLD 2 behind an absorber of 7mg/cm^2

4. COMMENTS BY PARTICIPANTS

(e) Nuclear Research Centre, Karlsruhe - B. BURGKHARDT

INTRODUCTION

The Karlsruhe Nuclear Research Centre participated in the European intercomparison experiment 1979/1980 (Beta-Dosimetry). The TLD 700 dosemeters have been calibrated at Karlsruhe by using a beta-secondary standard manufactured by Buchler. The beta-ray fields of two (^{90}Sr + ^{90}Y) sources of different activities had been calibrated by the PTB by means of an extrapolation chamber as the primary standard. On the basis of the certificated beta-dose at several distances from the sources, the calibration of the same TLD detectors resulted in response factors which differed by more than 10%. Taking into account the known transmission factors of the beta-sources in a suitable way, this systematical uncertainty can be reduced significantly.

EXPERIMENT

^{7}LiF dosemeter chips of the size of 3 x 3 x 0.2 mm^3 and 3 x 3 x 0.9 mm^3 have been irradiated on the surface of a tissue equivalent phantom to an absorbed dose of 25 mGy according to the standard calibration procedure of the beta-secondary standard. The irradiation of the bare dosemeter has been performed in a vertical position of the secondary standard without any dosemeter support or additional covers.

For the beta-calibration TLD batches of 5 chips have been used. In addition, 5 TLD chips have been irradiated with ^{137}Cs gamma-ray to an exposure of 2.5 R (absorbed dose in soft tissue of 23.75 mGy). The dosemeters have been evaluated together by placing the detector in the reader with the irradiated detector surface in direction to the photomultiplier.

TRANSMISSION FACTORS

For the estimation of the relative beta-response of TLD detectors of different thicknesses transmission factors have been taken into account to correct for the individual energy distribution of the beta-sources caused by self-absorbtion in the source, window thickness of the source and distance

between source and detector. From the PTB calibration values of transmission factors are given as a function of tissue depth for different distances of the sources. The beta-dose is presented in the quantity absorbed dose in soft tissue at the surface of a semi-infinite tissue equivalent phantom.

In Fig 1 the transmission factor t_β is presented as a function of the tissue depths for the different beta-sources according to the certificate of the beta-secondary standard (1) and the data given by the PTB for the tissue depths of 390 mg/cm^2 (2). For a larger distance between source and detector the t_β -value decreases significantly. For tissue depths higher than 100 mg/cm^2 the transmission factor was found by linear interpolation of the t_β values between 100 and 390 mg/cm^2. The t_β factor given in Fig 1 can be applied to calculate the absorbed dose in the tissue depths or in the detector depth of interest.

The optical transmission factor t_L takes into account the absorption of TL light in the detector during evaluation. The t_L values as a function of detector thickness presented in Fig 1 have been found by gamma-irradiated TLD detectors of 0.2 mm thickness which was covered during read-out consecutively by additional unirradiated chips of 0.2 mm thickness.

The response of the TL detector for beta- and gamma-rays is given by the equations :

$$R_\beta = \frac{\alpha_\beta}{D_\beta}$$

$$R_\gamma = \frac{\alpha_\gamma}{X.k} = \frac{\alpha_\gamma}{D_\gamma}$$

R_β , R_γ	β-response / γ-response for absorbed dose in soft tissue in counts per rad
α_β , α_γ	detector reading in counts
D_β , D_γ	absorbed dose in soft tissue in rads
X	exposure in R
k	conversion factor in rad/R

On the basis of Fig 1 the mean transmission factor T_β can be calculated taking into account the absorption of β = particles and of the TL light in the detector of the thickness s.

$$T_\beta = \frac{1}{s} \cdot \int_0^s t_\beta \cdot t_l \, ds$$

The T_l light absorption in the gamma irradiated detector is given by

$$T_l = \frac{1}{s} \int_0^s t_l \cdot ds$$

The response values R_β and R_γ are corrected by the transmission factor T_β and T_l in order to be independent of the detector thickness. This results in

$$R^o_\beta = \frac{\alpha_\beta \cdot s}{D_\beta T_\beta}$$

$$R^o_\gamma = \frac{\alpha_\gamma \cdot s}{X \cdot k \cdot T_l}$$

The ratio between β-response and γ-response is then

$$\frac{R^o_\beta}{R^o_\gamma} = \frac{\alpha_\beta}{\alpha_\gamma} \cdot \frac{X \cdot k}{D_\beta} \cdot \frac{T_l}{T_\beta}$$

RESULTS

In Table 1 the experimental results of the response ratio R_β / R_γ, the factor T_l / T_β and the corrected response ratio is presented for the TLD detectors of 0.2 mm and 0.9 mm thickness and for the calibration positions of the $^{90}Sr + ^{90}Y$ sources.

The uncorrected response ratio scatters for the different irradiation positions between 0.97 and 1.075 for the 0.2 mm thick TLD 700 and between 0.846 and 0.995 for the 0.9 mm thick TLD 700. Taking into account absorption of beta-rays and light in the detector the corrected response ratios have been found to be nearly constant within $\pm$ 1.5%.

By considering a backscatter factor of 1.1 for the ($^{90}Sr + ^{90}Y$) source (2), the mean values of 0.951 and 0.927 for the phantom irradiation resulted in corresponding response ratios for free air conditions of 1.05 and 1.02.

The systematical uncertainty for the β / γ reference dose is in the order of 2% / 3% and for the TLD irradiations 2% / 2%. The results of the response ratios are found to be within the total measuring uncertainty of about $\pm$ 5%.

With respect to intercomparison experiments, a comparison of the results found in the laboratories with different calibration sources is thus only possible, if a correction of the detector thickness on the basis of known transmission factors has been made. This reduces the systematic uncertainty from 15% to about 1%.

REFERENCES

(1) Certificate of the beta secondary standard No 4
PTB-Berichte No 6.41/04/78 SB

(2) J. Böhm, PTB, private communication

Table 1: Beta/gamma ray response ratio for TLD700 and detector specific transmission factors

SOURCE	ACTIVITY $^{90}Sr+^{90}Y$	DISTANCE	DETECTOR THICKNESS [1]					
			0.2 mm			0.9 mm		
No	mCi	cm	$\frac{T_1}{T_\beta}$	$\frac{R_\beta}{R_\gamma}$	$\frac{R_\beta}{R_\gamma}\frac{T_1}{T_\beta}$	$\frac{T_\beta}{T_1}$	$\frac{R_\beta}{R_\gamma}$	$\frac{R_\beta}{R_\gamma}\cdot\frac{T_1}{T_\beta}$
1	2	30	0.977	0.971	0.948	1.117	0.948	0.945
2	50	11	0.874	1.075	0.94	0.927	0.995	0.922
		30	0.896	1.068	0.956	0.948	0.969	0.918
		50	0.915	1.05	$\underline{0.961}$	0.979	0.945	$\underline{0.924}$
					$0.951^{2})$			$0.927^{2})$

[1]) 0.2 mm LiF ≙ 52 mg/cm², 0.9 mm ≙ 2.35 mg/cm²

[2]) mean value

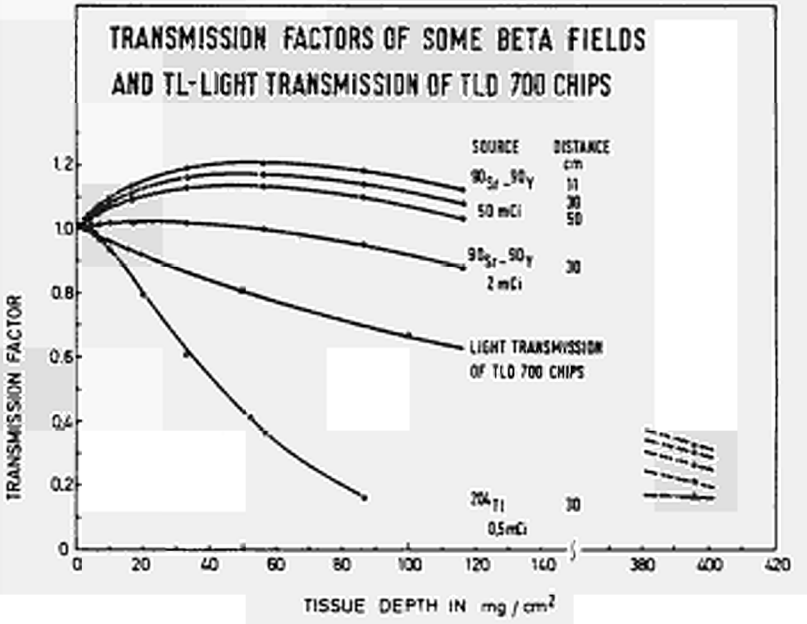


Fig. 1: Transmission factors as a function of tissue depth

4. COMMENTS BY PARTICIPANTS

(f) National Radiation Monitoring Service, Dublin - D.J. MURNAGHAN

The Film Badge Dosemeter Service operated by the National Radiation Monitoring Service (N.R.M.S.) in Ireland uses Kodak R.M. Film and the U.K.A.E.A./N.R.P.B. type holder. The evaluation methods in use are primarily based on the system devised by Jones and Marshall (1964) in the United Kingdom.

The overall N.R.M.S. results for the beta intercomparison are shown in the first diagram (figure 1) where the patched areas refer to Thallium-204 irradiations. The next three diagrams (figures 2, 3 & 4) show the beta results for each of the irradiating laboratories; the P.T.B., L.M.R.I. and N.P.L. The order in which the laboratories are listed is of some relevance as shall be seen later. It should be pointed out that this is a factor associated with the time scale of the intercomparison exercise and not directly with the irradiating laboratories. In the next diagram (figure 5) the time interval between the exposure of the film dosemeters at the irradiating laboratories and their subsequent development in Dublin is illustrated. The line indicated by "CAL" refers to our set of calibration and reference films which was prepared at a time which then seemed likely to coincide with the actual exposure dates at the irradiating laboratories. A considerable period of time elapsed between the exposure and development of the test dosemeters due to delays in their return to Dublin. These delays were outside the control of the irradiating and evaluation laboratories being due to a prolonged disruption of the postal services.

The first exposed dosemeters to reach our laboratory were from the P.T.B. and their development was postponed in the expectation that the exposed dosemeters from the L.M.R.I. and N.P.L. would soon arrive. After waiting for a month the decision was made to develop the P.T.B. and calibration films and so to salvage as much as possible from the intercomparison programme. Two additional sets of calibration films were then prepared. Set A being developed shortly after the exposure and set B being held for development with the films from the remaining two irradiating laboratories. The exposed dosemeters from both the L.M.R.I. and the N.P.L. did not reach our laboratory until July when they were developed together with calibration set B.

Thus, we experienced more technical problems than we had expected at the outset of the programme. For instance, there was a lapse of nearly six months between the exposure of the dosemeters at N.P.L. and their development in Dublin.

There is an apparant correlation between our results for each irradiating laboratory's batch of film dosemeters and the delay time in their return to our laboratory. From the time delays shown in the diagram (figure 5) the results would be least effected for the P.T.B. dosemeters and most effected for the N.P.L. exposed dosemeters. This is supported by the laboratory results.

In the evaluation of those film dosemeters exposed to beta radiation from Sr-90 the quality ratios, derived from the filter areas, were outside the range of our main evaluation system. We had thus to use the sets of films which we had exposed to a Sr-90 source in Dublin as part of our calibration and reference film sequences. In passing, we would note that the film dosemeters returned to the laboratory from our routine users usually have quality ratios, within the range of our main evaluation system.

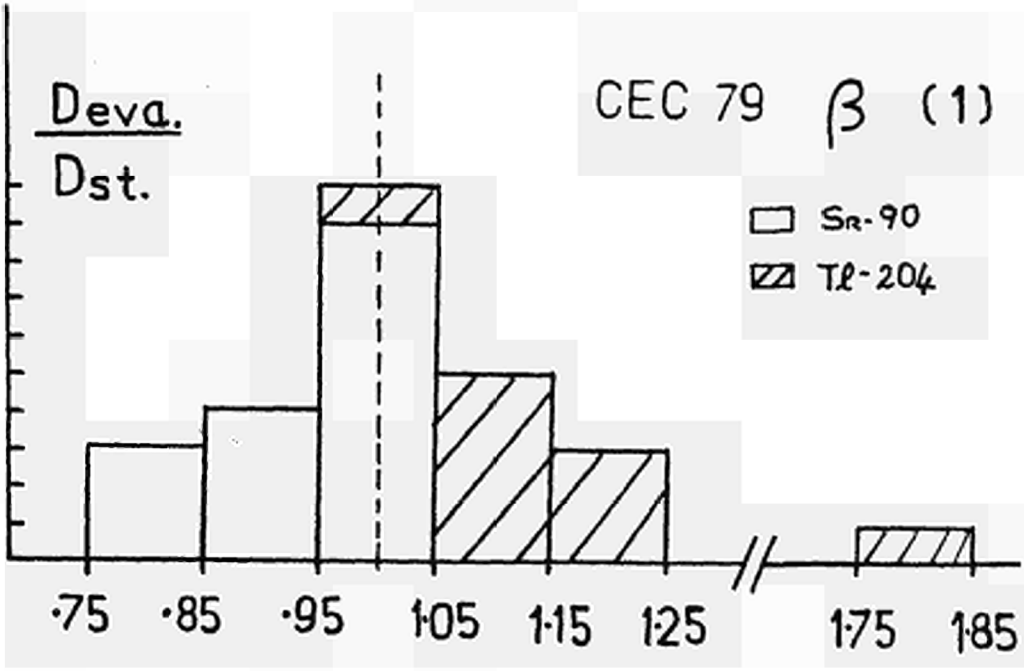
In several of the batches of film dosemeters returned to us we experienced the need to apply corrections for fading. The overall pattern of fading was not clear due to the unavoidable delays between exposure and development. This was further complicated by the time delays being different for the dosemeters exposed at each of the irradiating laboratories. The evaluation of fading effects was thus not as conclusive as we would have wished.

The conditions prevailing outside the dosimetry laboratory are always a factor in the accuracy attainable by an operational service. In this intercomparison programme the conditions we experienced were inferior to those usually present during routine operation. It was an encouraging picture nevertheless which emerged, as the results in those circumstances, with the unscheduled time delays were not unreasonable.

In conclusion, we would say that when the time delay problem first appeared our initial pessimism, about the possible results of our participation was not, in the end, fully justified. The intercomparison programme was thus of assistances to us and in fact gave us reassurance on several aspects. However, our results may be different to those from other participating laboratories due to the particular external difficulties which applied only to

our participation. This point should be borne in mind when forming an overall view of the results of this intercomparison exercise.

It is important to say that these intercomparison exercises initially for photons and now for beta radiations, are most useful to operational dosimetry services and we have gained useful knowledge from our participation in them.


Deva.
Dst.
CEC 79 β (1)
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Tℓ-204
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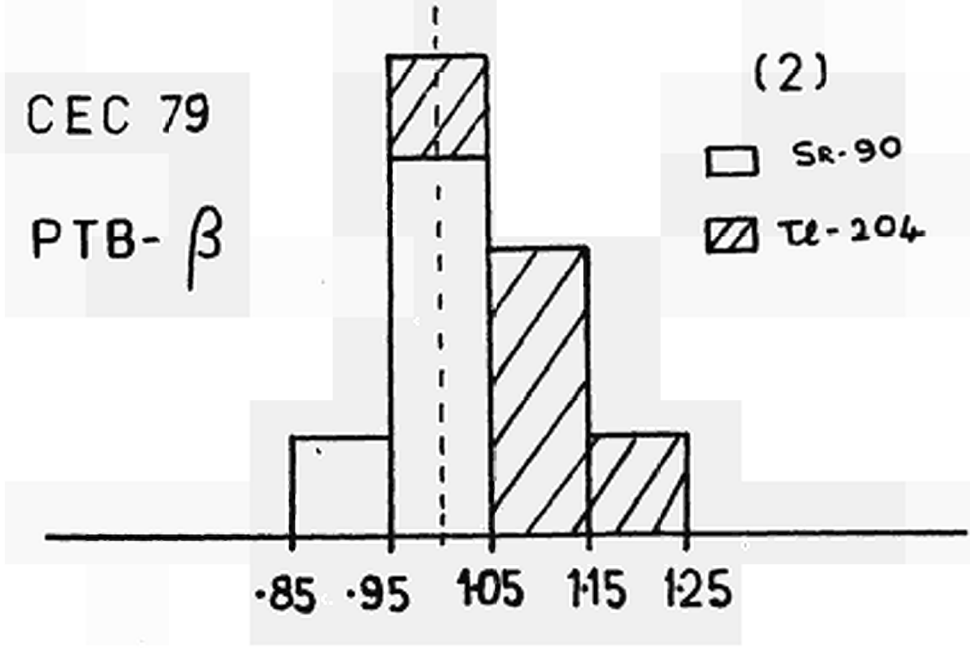
1·85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PTB- β
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Sr-90
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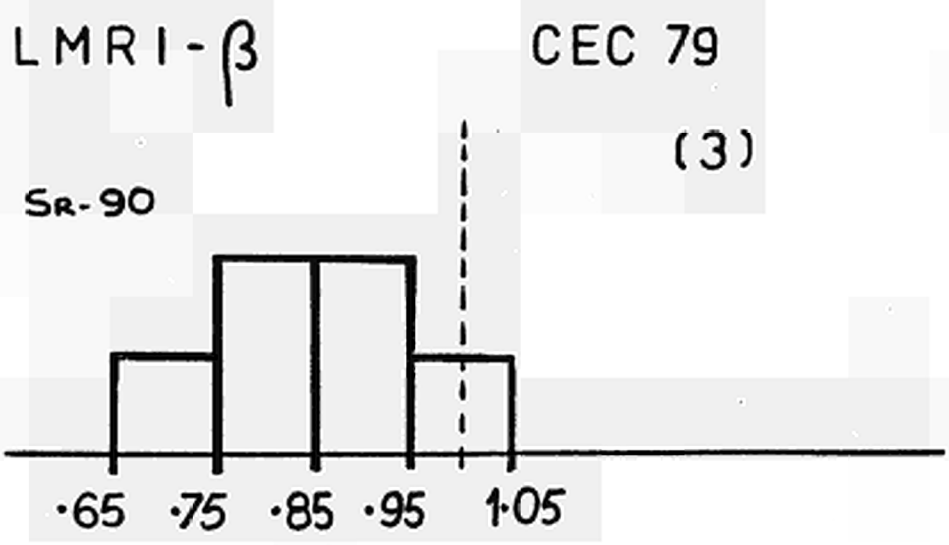
LMRI-β
CEC 79
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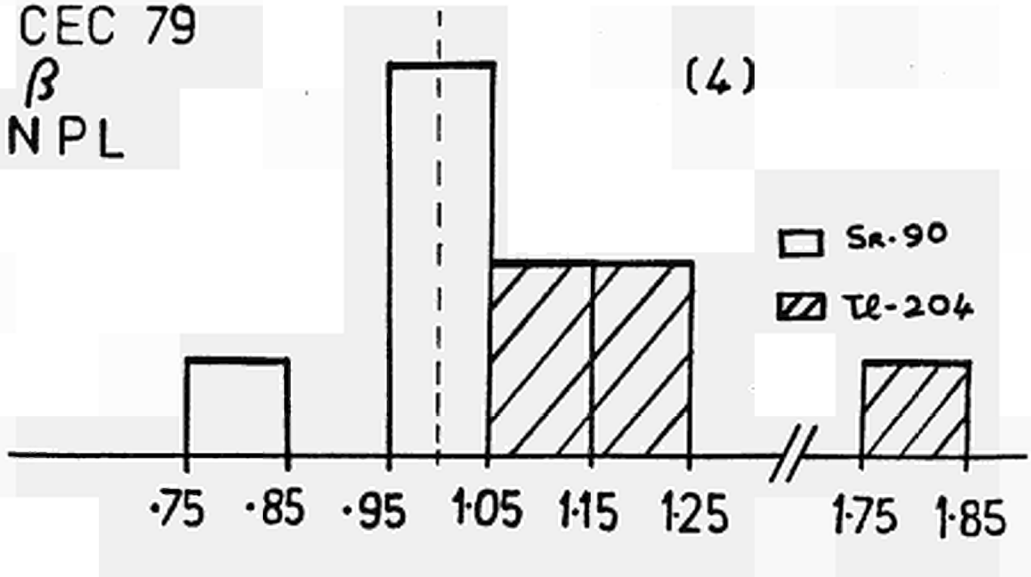
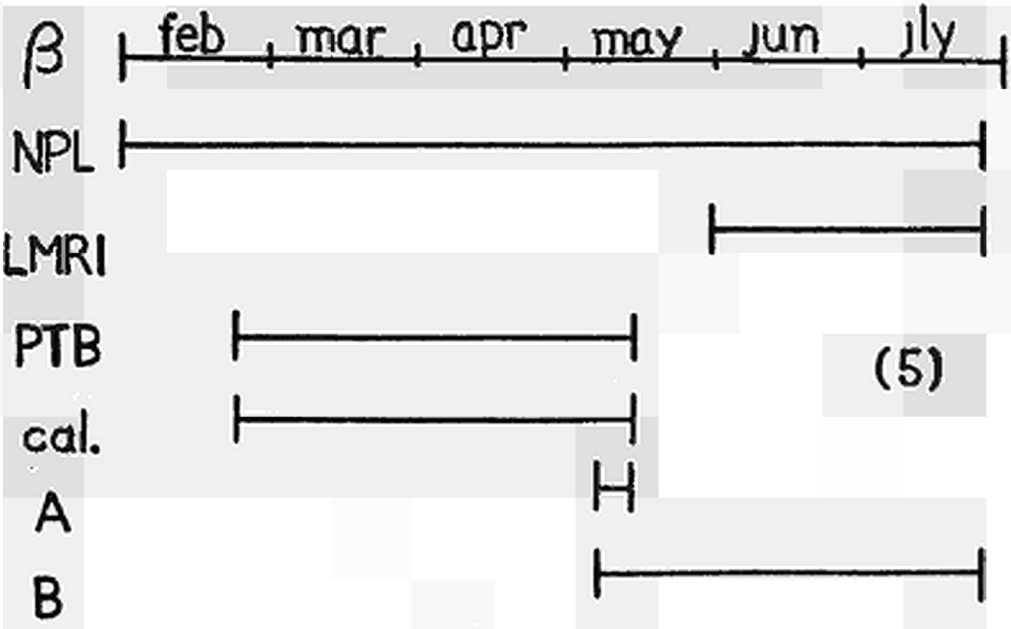

CEC 79
β
NPL
(4)
Sr-90
Tl-204
·75
·85
·95
1·05
1·15
1·25
1·75
1·85



(5)

4. COMMENTS BY PARTICIPANTS

(g) National Radiological Protection Board, Chilton - T.G. MARSHALL

1. INTRODUCTION

I would like to begin by emphasizing that both film and thermoluminescent dosemeters submitted by the NRPB are dosemeters used routinely in a commercially operated service and that the dosemeters were evaluated in the normal routine manner.

We welcome the opportunity to take part in intercomparison exercises of this type because they have a useful role in maintaining dosimetric standards. We were disappointed therefore that some uncertainty in the results was introduced by the extended period of about 4 months taken to irradiate the dosemeters. We waited until all the dosemeters had been returned before evaluating the results which resulted in a lapse of about 5 months between irradiation and read-out for some of the dosemeters. The resultant fading will have no doubt effected the accuracy of the results and may have effected the precision. It is hoped that such delays can be avoided in future intercomparisons.

2. FILM RESULTS

The NRPB/AERE film dosemeter is based, for beta-ray dose assessment, on calibrations with thin beta-ray sources, at an angle of incidence of 35° and at a distance of 20 cm. It is known (reference 1) that this dosemeter is sensitive to the source construction and irradiation geometry for beta-ray measurements so that we expected our results to differ markedly from the true results in this intercomparison exercise. For example, our results for the $^{90}Sr + ^{90}Y$ irradiations were consistently about 40% low. We have therefore submitted only normalized values for our film dosemeters. The results were as follows :

$$^{90}Sr + ^{90}Y \quad \left\{\frac{\overline{D}_{norm}}{\overline{D}_{st}}\right\} = 1.005 \quad \sigma = 0.09$$

$$^{204}Tl \quad \left\{\frac{\overline{D}_{norm}}{\overline{D}_{st}}\right\} = 1.2 \quad \sigma = 0.22$$

3. THERMOLUMINESCENT DOSEMETER RESULTS

Both normalized and evaluated results are given for the thermoluminescent dosemeters. Those for the normalized results are as follows :

$^{90}Sr + ^{90}Y$ $\left\{\frac{\bar{D}_{norm}}{D_{st}}\right\}$ = 1.033 σ = 0.07

^{204}Tl $\left\{\frac{\bar{D}_{norm}}{D_{st}}\right\}$ = 1.04 σ = 0.11

The results for the evaluated values were as follows :

		NPL	PTB	LMRI
$^{90}Sr + ^{90}Y$	$\left\{\frac{\bar{D}_{eva}}{D_{st}}\right\}$	0.84	0.869	0.911
		σ = 0.059	σ = 0.062	σ = 0.063
^{204}Tl	$\left\{\frac{\bar{D}_{eva}}{D_{st}}\right\}$	0.653	0.665	
		σ = 0.074	σ = 0.040	

For the thermoluminescent dosemeters a single calibration factor (based on the sensitivity to 104 keV X-rays) is used for the evaluation of beta-, X-, and gamma-ray doses and as such we expect the $^{90}Sr + ^{90}Y$ results to be about 5% low and the ^{204}Tl results to be about 25% low. However, the results for the LMRI irradiations were about 4% lower than expected and the NPL and PTB results were about 12% lower than expected. The LMRI results point to a systematic error of about 4% in the calibration of our system and the NPL and PTB results indicate about 8% fading over 5 months. Both these possibilities are being investigated.

References

1. Adams N., Heard M.J., and Holt P.D. "Film Dosimetry Practice with the AERE/RPS Film Dosemeter: A Collection of Experimental Data". AERE-R4669 (1965).

4. COMMENTS BY PARTICIPANTS

(h) Centre d'Etudes de l'Energie Nucléaire, Mol - L. GHOOS

The S.C.K./C.E.N. Mol took part in the beta-ray intercomparison programme with its film dosemeter. To determine the doses, densities are measured in three different zones of the film corresponding to three different filter thicknesses.

To evaluate doses due to beta-rays some correction factors have to be applied which are related to the energy of the incoming particles; the energy is assessed by calculating the ratio of the apparent doses obtained from the densities behind the thinner filters.

The results we have obtained during this intercomparison are normal for the irradiations with ^{204}Tl; some anomalies are present for the ^{90}Sr + ^{90}Y irradiations.

When in the sixties the design of our dosemeters was finished and all correction factors had been determined, for a final test some dosemeters were irradiated in a standard laboratory. For X-rays this was P.T.B. at Braunschweig, for beta-rays this was Radiological Protection Services at Belmont (U.K.). This was done in 1966. We proceeded in the same way as we did in the present Euratom intercomparison : we determine the doses in the routine-way and afterwards the results were compared with the real doses.

In Table I we give the results we obtained, at that time, for the dosemeters irradiated with ^{90}Sr + ^{90}Y (doses in mGy).

Table II gives the results we communicated in the present intercomparison for dosemeters irradiated with ^{90}Sr + ^{90}Y.

In Table I the ratio $\frac{B}{A}$ is practically constant up to 50 mGy.

In Table II this ratio is constant up to 30 mGy while for 50 mGy and higher doses our doses are systematically lower by a factor of 2.

The results however were obtained in the same routine-way as was done in 1966. This can now be explained but it was not obvious initially.

Indeed we are still using the same film in our film dosemeters - Agfa Gevaert Structurix D10 - as we did in 1966. The less sensitive film D2 is not used for dose determination in the dose range involved here.

Dose R.P.S. A	Our dose B	Ratio $\frac{B}{A}$
3	2.97	0.99
9.9	10.7	1.08
29.7	27.5	0.93
49.5	51.6	1.04

TABLE I

Euratom dose mGy A	Our dose (communicated) mGy B	Ratio $\frac{B}{A}$
2.4	3.02	1.26
2.44	2.9	1.20
4.89	4.8	0.98
4.89	4.96	1.01
5	4.96	0.99
5	5.02	1
5.23	5.44	1.04
10.45	7.03	0.67
16	15.0	0.94
18.3	11.6	0.63
20.9	23.2	1.11
31.4	34.9	1.11
51.3	31.6	0.62
52.2	36.6	0.70
100	51.7	0.52
104.6	55.7	0.53

TABLE II

Looking more closely however we compared the sensitivity-curves for the 1966-emulsion and the 1979-emulsion and we observed a much faster saturation for the 1979-emulsion.

For the 1966-emulsion, for all doses considered in this comparison, the points on the sensitivity-curve on semi-log paper are all situated in the linear part of the S-curve. The ratio between the apparent doses behind the thinner filters is constant over the whole range of doses and so is the correction factor.

Due to a faster saturation in the 1979-emulsion the points on the sensitivity-curve, corresponding to the higher doses were not any longer in the linear part of the S-curve but already in the curved upper part. The ratio of the apparent doses was not constant any longer over the whole range. The correction factors are different and there is a difference between the results for the lower and higher doses.

You can wonder how it is possible to wait for about 15 years (from 1966 up to now) before checking the method of calculating doses. In fact we didn't. When we got the beta secondary standard from Buchler irradiated our film dosemeters with beta-emitting isotopes. As however in practice we never encounter beta-doses higher than 15 mGy our calibration for film dosemeters was not extended above this value: and so the discrepancy for higher doses was still present and it still exists.

We want to stress these results have been obtained in our normal routine way.

4. COMMENTS BY PARTICIPANTS

(i) CNEN - Laboratorio Fisica Sanitaria, Bologna - G. Busuoli

The dosemeter used was a prototype containing four BeO (Thermalox 995) TL detectors, 6 mm in diameter covered with 4 filters from 2 mg/cm^2 to 70 mg/cm^2 thick.

The dosemeter was calibrated using the secondary standard beta kit provided by NPL.

The results obtained in the intercomparison for the irradiations performed at PTB, NPL and LMRI are summarized in table 1.

The data reported in the table show that the prototype dosemeter used during this intercomparison needs to be improved even though some of the results have been affected by an unknown fading due to the long time between irradiation and evaluation in some cases.

The experiment was very useful and has demonstrated that the four element dosemeter is some what redundant and that the filter thickness is not the optimum one.

TABLE 1

	Source	Stated Dose mGy	Normalised Dose mGy	$D_{norm.}/D_{Stat.}$
IRRADIATIONS AT LMRI	$^{90}Sr/^{90}Y$	31.2	28.9	0.93
		51.9	54.4	1.05
		103.9	92.0	0.89
IRRADIATIONS AT NPL	$^{90}Sr/^{90}Y$	51.1	35.6	0.70
		18.2	13.9	0.76
		2.43	1.5	0.62
	^{204}Tl	16.8	31.0	1.85
		33.6	28.0	0.83
		2.1	2.1	1.00
	^{147}Pm	15.8	15.3	0.97
		2.1	43.9	20.9
		4.93	4.33	0.88
IRRADIATIONS AT PTB	$^{90}Sr/^{90}Y$	2.0	1.0	0.50
		24.0	21.8	0.91
		80.0	63.8	0.80
	^{204}Tl	1.6	16.8	1.68
		10.0	10.6	1.06
		24.0	22.1	0.92
	^{147}Pm	2.4	5.8	2.42
		5.0	10.5	2.10
		10.0	13.1	1.31

4. COMMENTS BY PARTICIPANTS

(j) GSF, Neuherberg - D.F. Regulla

General remarks

The GSF participated in the CEC Beta Intercomparison Programme with thermoluminescence and film dosemeters. The intercomparison programme fits well the present scientific activities at GSF in the field of beta dosimetry which are documented in a Ph.D. thesis (L.V.E. Caldas : Methods of Calibration and Dosimetry for Beta Radiation, GSF/University of Sao Paulo, 1980) and in the normal Research Programme aiming to set up a personal beta dosemeter. The scientific activities are complemented by the installation of a secondary standard dosimetry laboratory (SSDL) for beta radiation which is based on standard beta sources and extrapolation chamber techniques. The beta standards are operated within the GSF Calibration Centre which is a member of the IAEA/WHO international SSDL network and WHO collaborating Centre for Secondary Standard Dosimetry and Radiation Protection (see H. Eckerl and U. Nahrstedt : Das Sekundärstandard-Labor der GSF für Photonen- und Betastrahlung, GSF-Rep. S 587, Neuherberg, 1979).

Thermoluminescence dosimetry

All measurements were performed with thin-film $CaSO_4$ detectors with a TL active layer of about 5 mg/cm^2 bonded to a similarly thin aluminum foil. For protection during transport and exposure the detectors were covered with an aluminized Hostaphan foil of 1.7 mg/cm^2 mass per area excluding also light effects (O.D. = 3)(*). For convenience, the detectors were positioned in the open window of the film badge (see A.4). Evaluation was performed on a hot gas reader. For interpretation of the measurements in terms of dose, a Co-60 gamma-ray calibration was used under charged particle equilibrium conditions.

The results are given in Table 1. All ratios between evaluated dose and stated dose are relatively close to unity. The results vary with the irradiating laboratory. The best results for the unknown doses were achieved for Sr-90/Y-90 (LMRI), the least good results were found for Pm-147 (NPL).

(*) (Optical Density = 3)

The problem of discrimination between betas and photons is under consideration. In principle, thin-film detectors designed for beta dosimetry will underestimate the high-energy gamma component ($E_{Ph} \gtrsim 0.3$ MeV) due to lack of charged particle equilibrium. The present $CaSO_4$ detectors will, moreover, overestimate doses resulting from photons with $E_{Ph} < 0.3$ MeV.

Film dosimetry

The film dosemeter used for the Beta Dosimetry Intercomparison is based on the Agfa-Gevaert personal monitoring film which in the intercomparison period was still paper-enveloped, and is used routinely in the governmentally authorized GSF Film Dosimetry Service. The films were exposed inside the official badge containing 4 filters (0,05 mm Cu; 0,3 mm Cu; 1,2 mm Cu; 0,8 mm Pb) besides the open window. The calibration for evaluating the beta doses refers to Co-60 gamma irradiation which procedure is applicable for high-energy betas and electrons. For low-energy betas appropriate corrections must be applied provided the radiation quality is known.

The film results of the Beta Intercomparison are also given in table 1 for two different calibration procedures : (1) Routine GSF calibration based on Co-60 for all beta energies without correction; (2) calibration based on known doses provided by the irradiating laboratory.

As a result, the routine Co-60 calibration holds only for Sr-90/Y-90 except where the beta radiation quality can be estimated. In this case the necessary correction factors are about 2 to 2.5 for Tl-204 and about 5 to 10 for Pm-147. However, the determination of the beta radiation quality is difficult particularly in cases of mixed photon/beta fields. Under these conditions, the discrimination of the beta radiation component is promising only in the presence of high-energy photons.

It should be mentioned that the beta particles of Pm-147 did not penetrate the film envelope. The blackening of the film resulted from bremsstrahlung (see B.7, Fig. 3).

Conclusion

The CEC Beta Dosimetry Intercomparison 1979/80 offered the unique occasion to document the state of beta dosimetry in personal monitoring

within the European Community. From analysis of our own results as well as from comparing these results with those from other member states the situation appears rather encouraging. It is particularly worthwhile mentioning the high level of beta calibration facilities found in the community which is far above the one recently reported for the United States.

Despite the already promising results there is still research work to be done. This holds especially for the preparation of appropriate detectors of preferably low atomic number, new and eventually more appropriate dosimetry techniques, arrangements to discriminate between betas and photons and thus measure in mixed fields. CEC should assist the efforts in the member states by installing a steering committee for research coordination, research contracts and guidance.

The research work should be complemented by proper recommendations concerning the criteria for personal beta dosimetry, e.g., the beta energy range of interest. If according to the ISO draft (TC85/SC2/WG7) the range should be 500 keV $< E_{\beta, max} <$ 3 MeV, this would seriously influence the future work and dosemeter design. Again, CEC is invited to set up a necessary working committee.

Last but not least, CEC should take care of the organization of repeated beta intercomparisons in reasonable intervals of time. This will help to test and intercompare progress in beta dosimetry and give further confidence in the personal monitoring programme of the member states.

Table 1 CEC Beta Dosimetry Intercomparison 1979/80 :
Results of the GSF thermoluminescence and film dosemeters

Irrad. inst.	Beta source	Stat. dose cGy	Evaluated / Stat. dose TLD		Film	
			GSF cal.	Ext. cal.	GSF cal.	Ext.cal.
LMRI	Sr/Y-90	3.090	1.010	1.019	1.06	0.94
		5.150	1.019	1.029	1.01	0.88
		10.310	0.984	0.993	0.90	0.79
NPL	Sr/Y-90	4.980	1.056	1.006	1.00	0.88
		0.243	1.152	1.069	1.01	0.88
		1.940	1.062	1.010	0.97	0.85
	Tl-204	3.380	1.036	0.988	0.36	0.85
		0.212	0.943	0.920	0.42	0.99
		1.690	1.000	0.953	0.43	1.00
	Pm-147	1.840	0.783	0.962	0.10	0.74
		0.602	0.781	0.955	0.11	1.00
		0.226	0.929	1.106	0.09	0.78
PTB	Sr/Y-90	0.160	0.938	0.875	1.06	0.94
		2.000	1.060	1.025	1.01	0.88
		12.000	1.054	1.022	0.90	0.79
	Tl-204	0.240	0.875	0.917	0.48	0.88
		1.200	1.033	1.058	0.49	0.91
		1.600	1.013	1.037	0.44	0.81
	Pm-147	0.160	1.000	1.000	0.12	0.60
		0.500	0.920	0.940	0.18	0.86
		0.813	0.923	0.947	0.17	0.83

4. COMMENTS BY PARTICIPANTS

(k) Risø National Laboratory, Roskilde - P. Christensen

Introduction

Risø National Laboratory participated in the beta-ray intercomparison programme with two types of thermoluminescent dosemeters, which are both used in the personnel dosimetry programme at Risø :

- Type A, a 25 mm$^{\phi}$ x 9 mm polyethylene button containing three 0.9 mm x 3.2 mm x 3.2 mm LiF TLD-700 chips (Harshaw). The chips are kept in depressions at the top-side of the button and covered by a 1 mg.cm^{-2} thick aluminized Mylar foil. The dosemeter is used for beta dosimetry, in particular for experiments where low-energy beta emitters are involved. The dosemeter can contain up to twelve different TLD detectors.

- Type B, a polyethylene sachet containing two 0.9 mm x 3.2 mm x 3.2 mm LiF TLD-700 chips. The front-side and back-side thickness of the sachet is 17 and 39 mg.cm^{-2}, respectively. For the ^{147}Pm-irradiations a special sachet with a front-side thickness of 4 mg.cm^{-2} was used. The TLD sachet is routinely used at Risø for finger-dose monitoring.

Beta calibration data for the dosemeters (see Table 1) were obtained from a PTB/Buchler secondary standard beta calibration unit.

Results

The mean values of normalized and evaluated results are given in Table 2. Two results, a ^{204}Tl-dose stated as 2.4 rad and measured to be zero and a ^{147}Pm-dose stated as 10 rad and measured to be 0.23 rad were not included in the data shown in Table 2. As all detectors from the dosemeter in each case gave identical results, it was assumed that the two dosemeters have not received the stated doses.

It appears from Table 2 that our estimated values for the ^{147}Pm-doses are 0-30 % below the stated doses. Influence from 22 keV Ag K X-radiation may be one of the sources for this disagreement.

Table 1

Dosemeter responses to beta irradiations

Source	Resp. per rad beta dose in tissue at 7 $mg \cdot cm^{-2}$ / Resp. per 1 R ^{60}Co-exposure	
	Type A	Type B
$^{90}Sr+^{90}Y$ 50 mCi, 50 cm	1.00	1.02
$^{90}Sr+^{90}Y$ *) 2 mCi, 30 cm	0.944	0.95
^{204}Tl *) 0.5 mCi, 30 cm	0.296	0.221
^{147}Pm *) 14 mCi, 20 cm	0.117	0.072

*) With beam flattening filter

Table 2

Mean values of evaluated and normalized results

Institute	Source	$\left\{\frac{D_{eva}}{D_{st}}\right\} \pm \sigma$		$\left\{\frac{D_{norm}}{D_{st}}\right\} \pm \sigma$	
		Type A	Type B	Type A	Type B
PTB	Sr/Y	0.995±0.013	1.025±0.009	0.987±0.013	0.991±0.009
	Tl	1.008±0.011	0.955±0.027	0.998±0.013	1.013±0.028
	Pm	0.990±0.020	0.866±0.075	0.994±0.020	1.022±0.089
NPL	Sr/Y	1.021±0.008	1.036±0.008	1.002±0.008	1.005±0.008
	Tl	1.037±0.026	1.093±0.039	0.983±0.025	0.976±0.035
	Pm	0.847±0.019	0.708±0.064	0.983±0.023	1.075±0.091
LMRI	Sr/Y	0.993±0.013	1.013±0.025	1.009±0.013	1.013±0.023

5. EVALUATION OF THE RESULTS OF THE FIRST INTERCOMPARISON PROGRAMME FOR BETA-RAY DOSEMETERS

J. BÖHM
Physikalisch-Technische Bundesanstalt, Braunschweig

Twelve participants sent personal dosemeters to the primary laboratories of the PTB, NPL and LMRI for irradiation. The dosemeters of the participants are shown in Figure 1. Eleven participants reported the absorbed doses D_{norm}, on which this evaluation is based. Some of the participants utilized the maximum allowed area of 10 cm x 10 cm per irradiation to enter more than one dosemeter. A numbering system is therefore not given to the participants but to the types of dosemeters. The 19 types of dosemeter are distinguished by the letters F and T along the centre scale of Figures 2, 3 and 4 to indicate the radiation-sensitive material : F - film, T - thermoluminescence dosemeter. The numbers of dosemeter types between broken lines belong to one participant.

The absorbed doses reported by the participants, D_{norm}, are compared with the absorbed doses D_{st} stated by the irradiating primary laboratories. Figures 2, 3 and 4 contain all the D_{norm}/D_{st} ratios obtained during this intercomparison for the radionuclides $^{90}Sr + ^{90}Y$, ^{204}Tl and ^{147}Pm. The values of $1 - D_{norm}/D_{st}$ for the different laboratories varied between + 3% and - 3% and + 150% to - 50%. It is evident that due to these large differences, one scale for all D_{norm}/D_{st} values is not convenient. The scatter of the small $(1 - D_{norm}/D_{st})$ values can no longer be discernible. Thus, two scales for D_{norm}/D_{st} have been introduced in Figures 2, 3 and 4 : one from 0.8 to 1.2, and one from 0.5 to 2.5. If at least one of three D_{norm}/D_{st} values belonging to the irradiation with one radionuclide of one primary laboratory does not fit into the smaller scale, all three values are presented in the diagram with the larger scale.

Figures 5 to 18 contain histograms of D_{norm}/D_{st}. A Gaussian distribution is fitted to each histogram, and the mean value $\bar{x}$ and the standard deviations of each Gaussian distribution is given. Figures 5 to 7 show histograms for the irradiations by the three primary laboratories with $^{90}Sr + ^{90}Y$. No significant differences of the three Gaussian distributions can be recognized. It therefore seems justified to combine the D_{norm}/D_{st}

values of the three primary laboratories in one histogram (Figure 8). If the results obtained with dosemeters irradiated to absorbed doses above 25 mGy are combined, significant differences between dosemeters containing films and thermoluminescence materials can be seen (Figures 9 and 10). While the mean value for the thermoluminescence dosemeters is very close to unity and the relative standard deviation is small (9%), the mean value deviates markedly from 1 (0.9) for films and the relative standard deviation is about twice as great (19%). Such differences were not observed at absorbed dose rates below 25 mGy (Figures 11 and 12).

The results of irradiations by the PT3 and NPL for ^{204}Tl are shown separately in Figures 13 and 14. As there are no significant differences, the results are combined in Figure 15. The histograms for the irradiations with ^{147}Pm by the PTB (Figure 16) and NPL (Figure 17) exhibit larger values of s than for ^{90}Sr + ^{90}Y and ^{204}Tl. The results for ^{147}Pm contained in Figures 16 and 17 are combined in Figure 18.

Although beyond the scope of this intercomparison, some institutes used their own calibration factors to determine the absorbed dose. It is of interest to compare the absorbed dose D_{eva} evaluated by their own calibrations with the absorbed dose D_{st} stated by the primary laboratories. Figures 19 to 21 show the results for the three radionuclides. The $\bar{x}$ and s values are close to those obtained for the D_{norm}/D_{st} values.

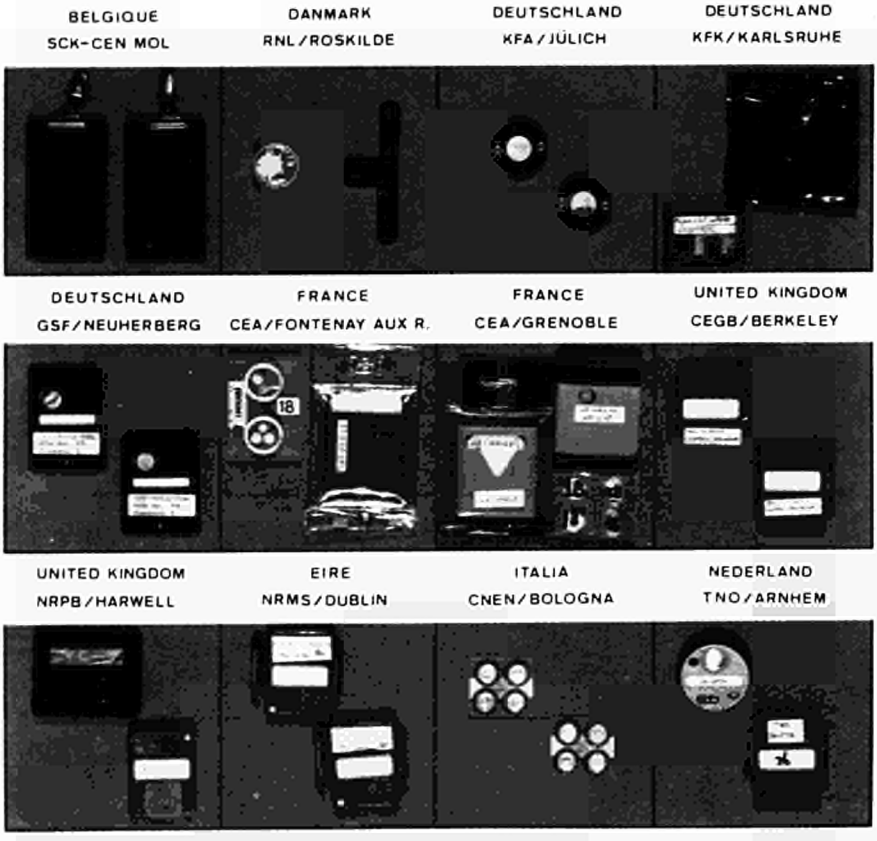


Figure 1: Personal dosemeters which were sent by the 12 participants to the three irradiating primary laboratories.

Figures 2, 3, and 4: D_{norm}/D_{st} values obtained during the intercomparison program for the three radionuclides (^{90}Sr + ^{90}Y), ^{204}Tl, and ^{147}Pm. For further explanation see text.

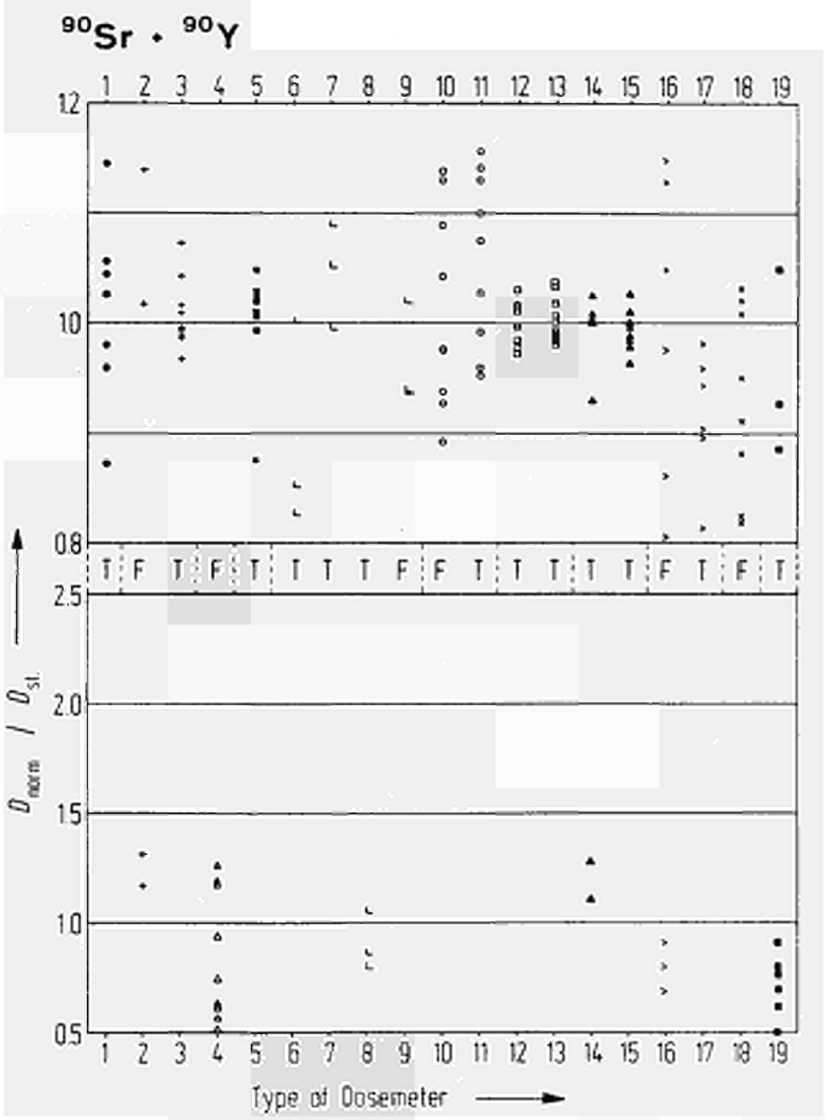

^{90}Sr + ^{90}Y
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19
1.2
1.0
0.8
T F T F T T T T F F T T T T T F T F T
2.5
2.0
1.5
1.0
0.5
D_{norm} / D_{st}
Type of Dosemeter

Figure 2

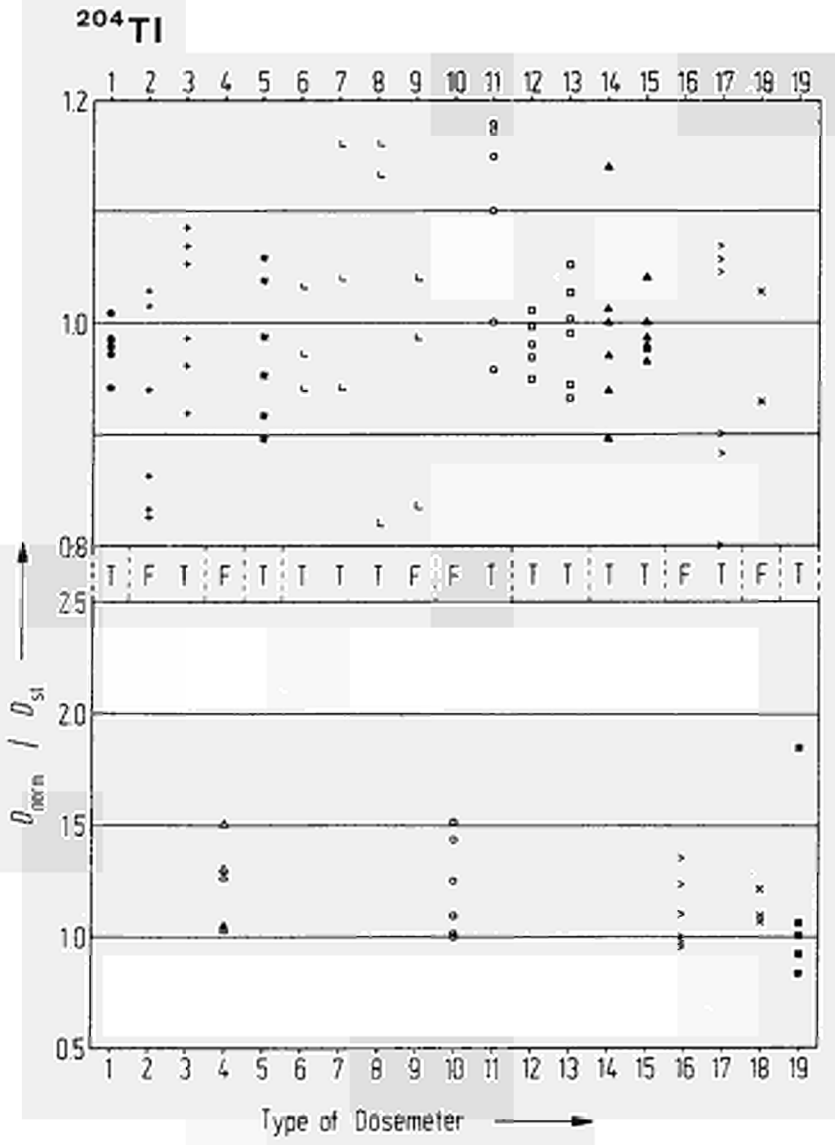

204Tl
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19
1.2
1.0
0.8
T F T F T T T T F F T T T T T F T F T
2.5
2.0
1.5
1.0
0.5
D norm / D st
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19
Type of Dosemeter

Figure 3

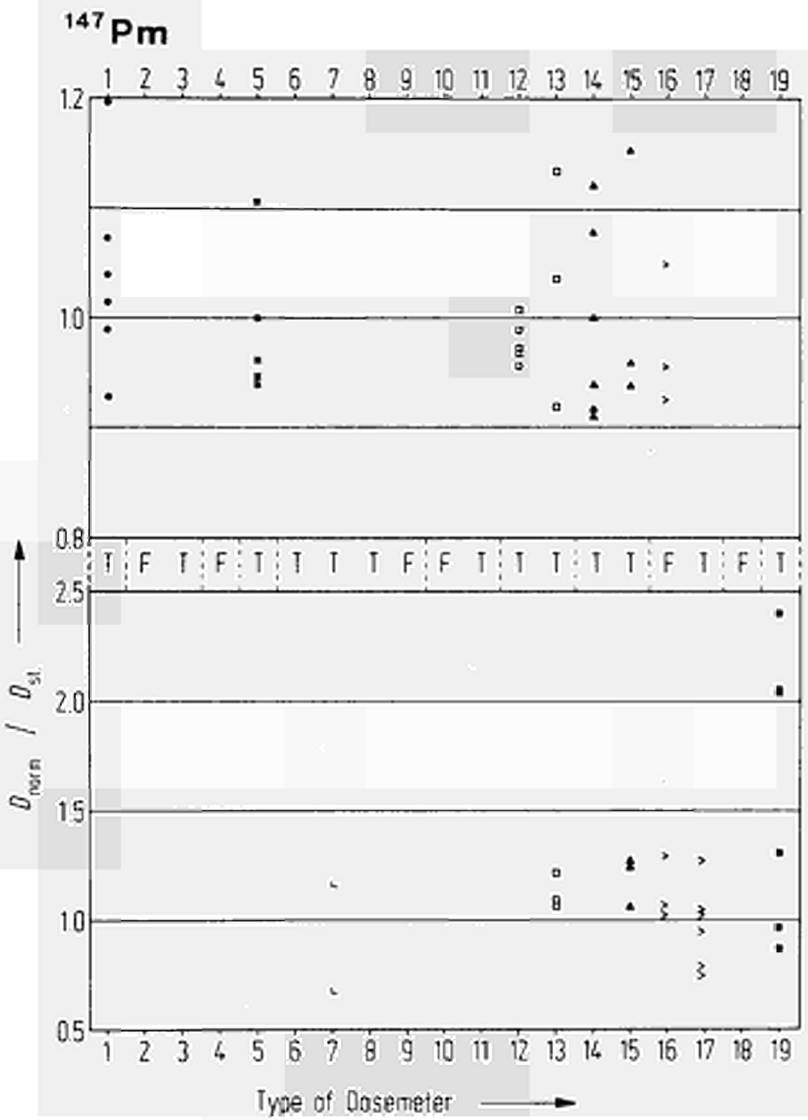

147Pm
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19
1.2
1.0
0.8
T F T F T T T T F F T T T T T F T F T
2.5
2.0
1.5
1.0
0.5
D norm / D st
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19
Type of Dosemeter

Figure 4

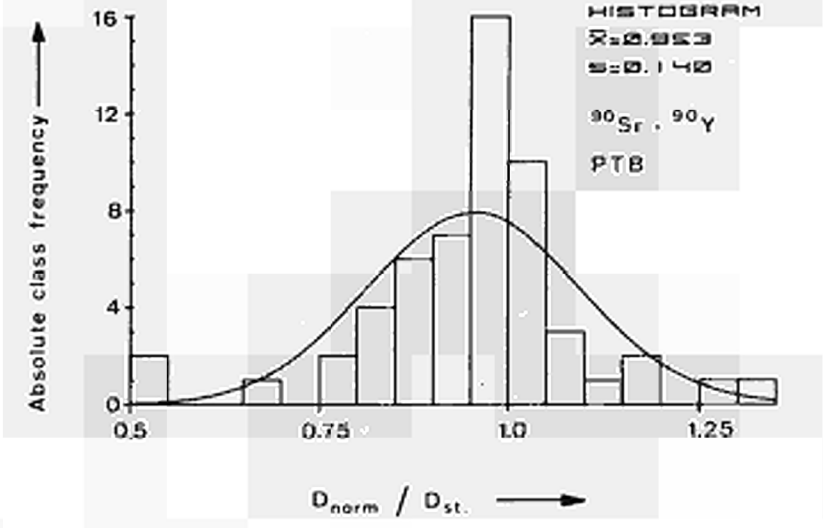


Figure 5

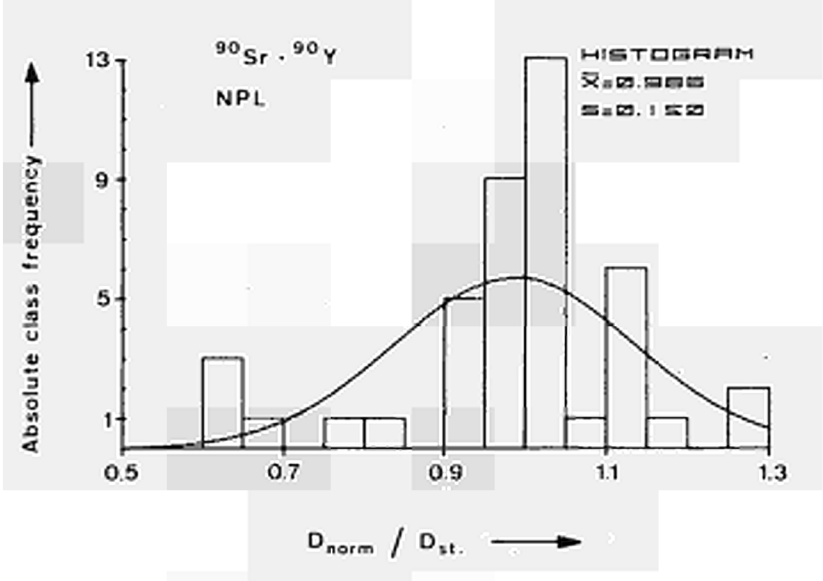


Figure 6

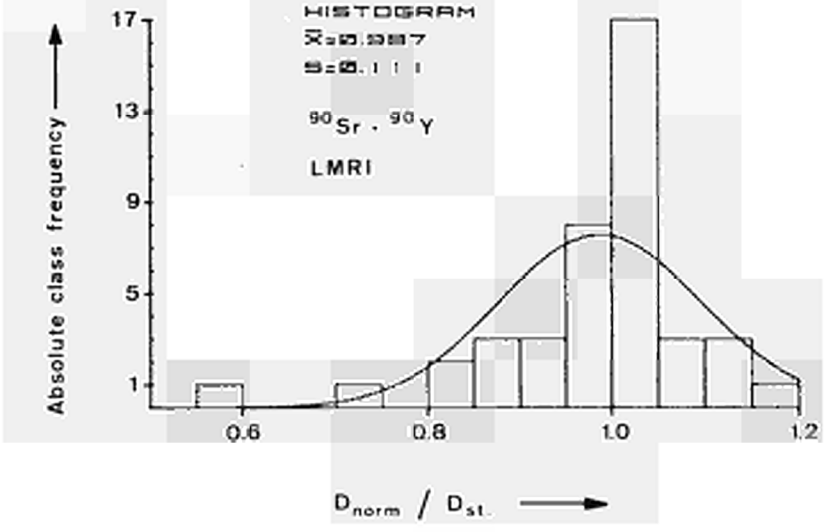
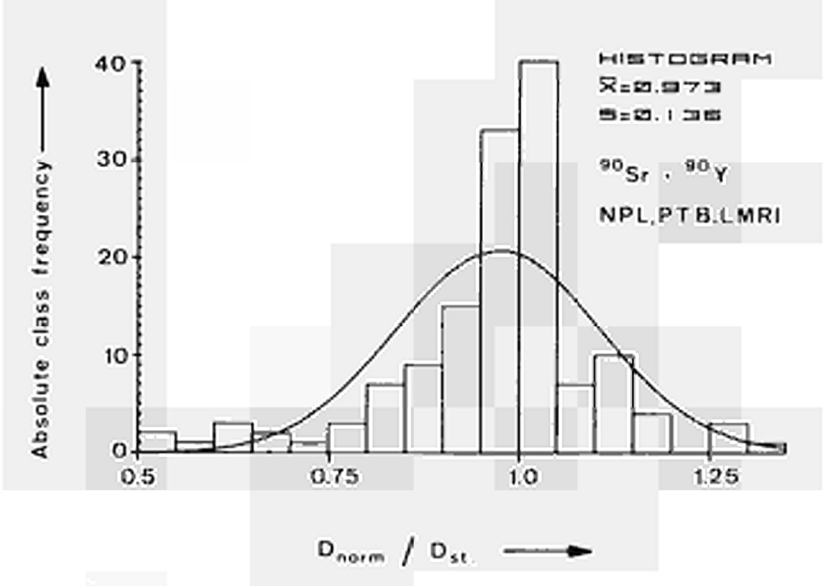


Figure 7



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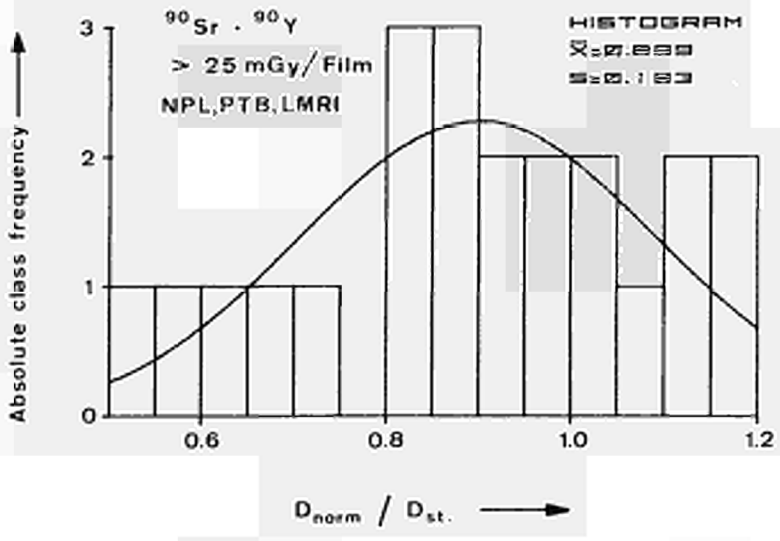

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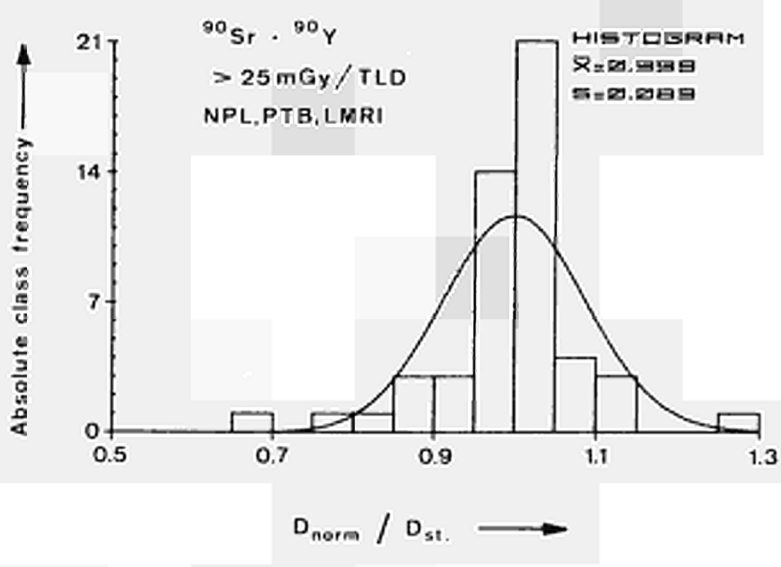

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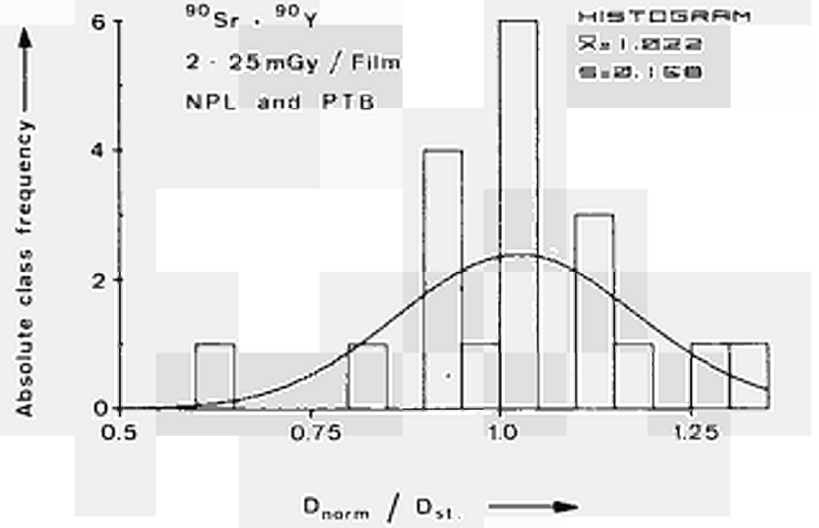

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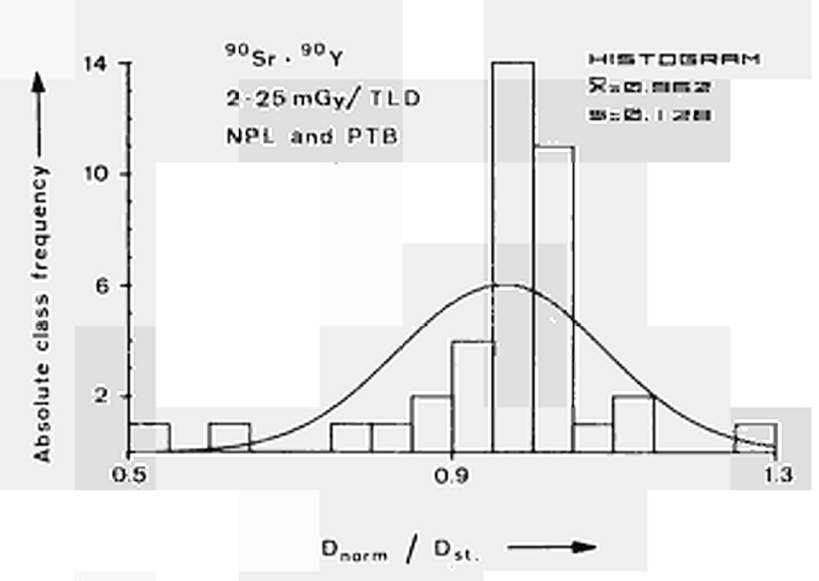

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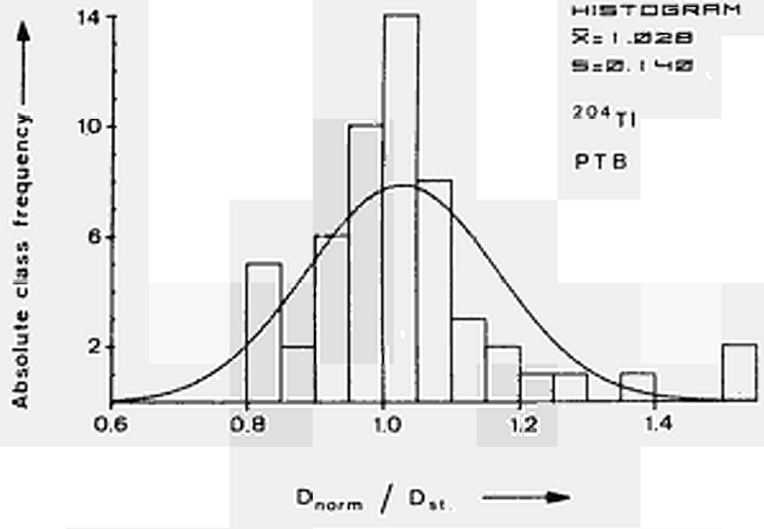

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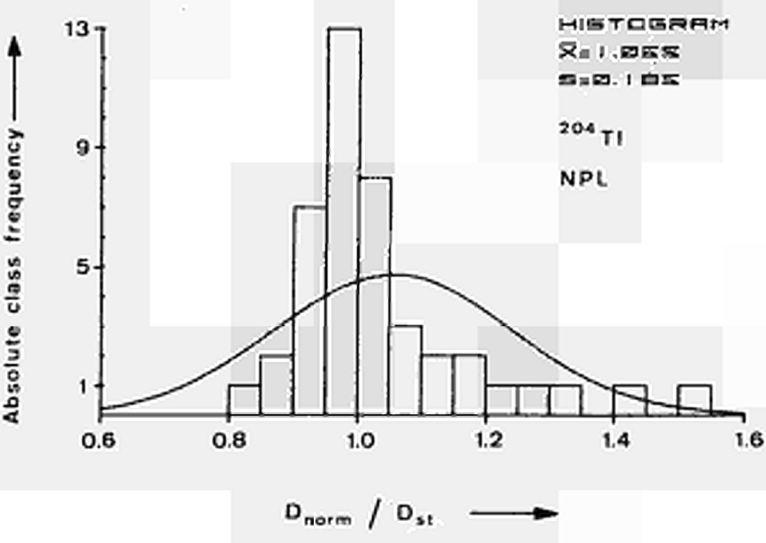
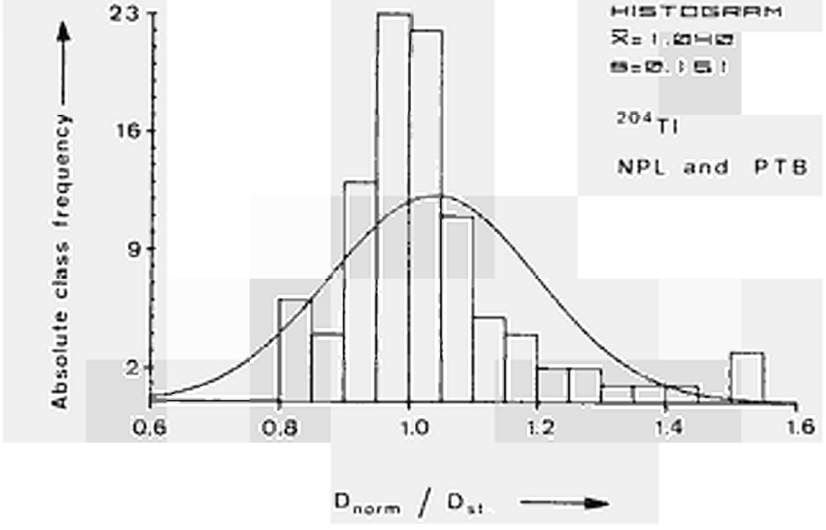


Figure 14



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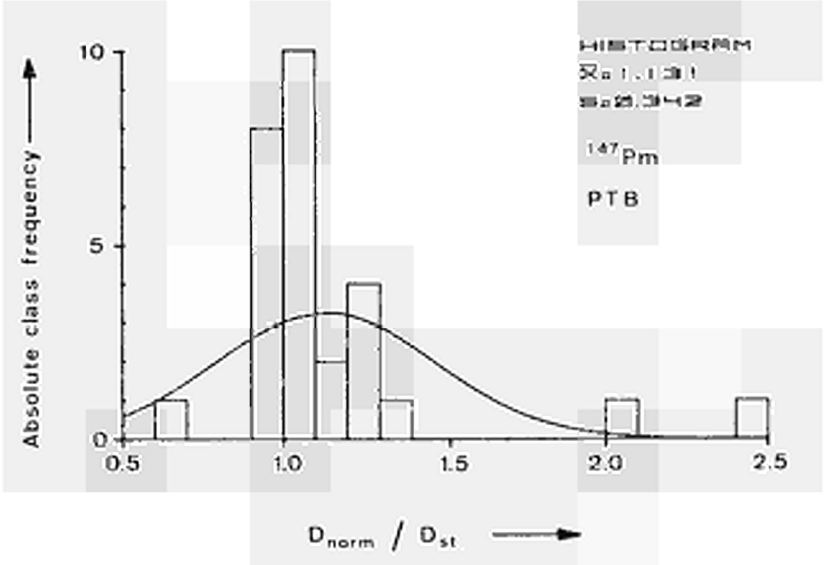

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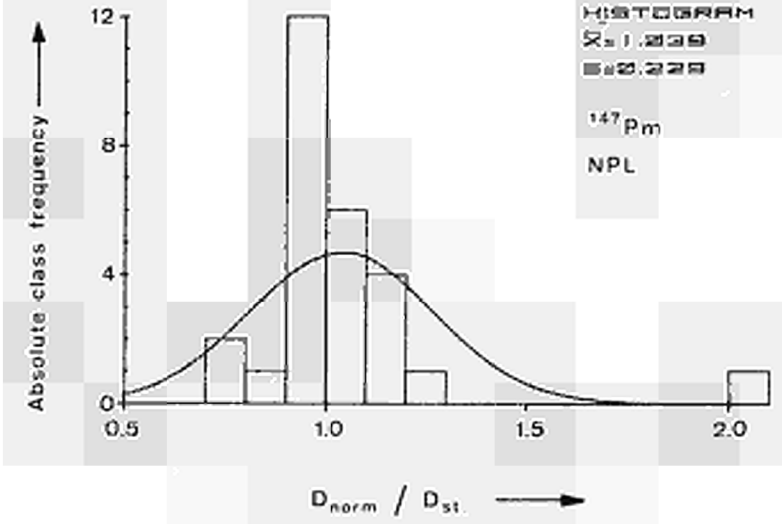

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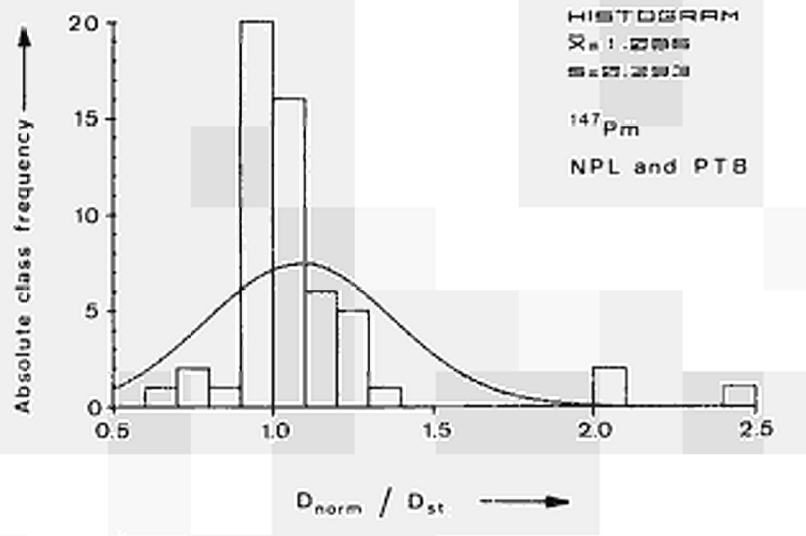

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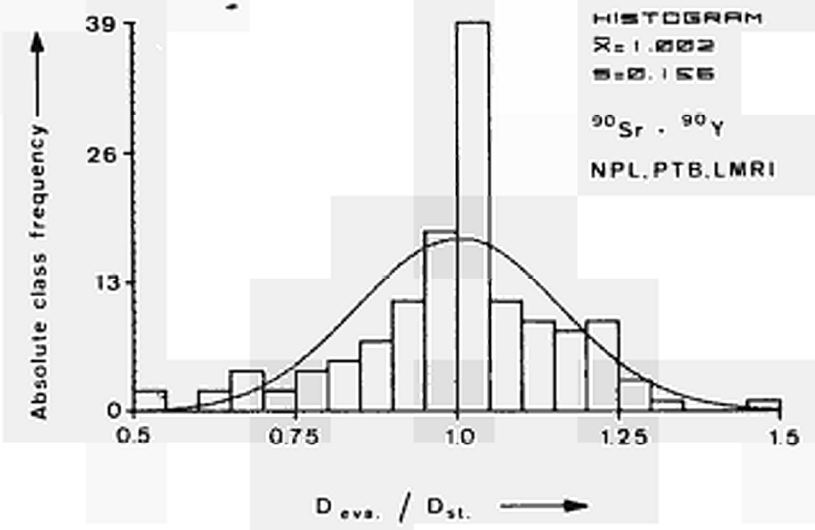

Figure 19

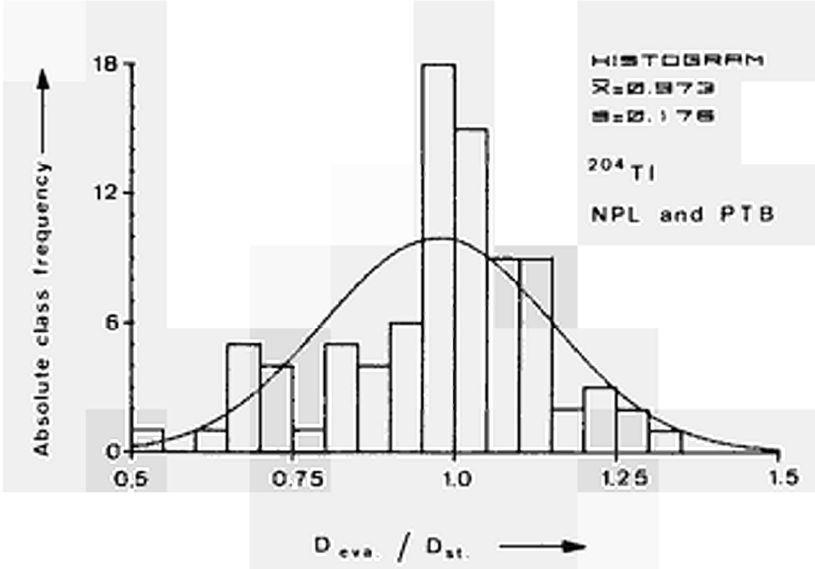

Figure 20

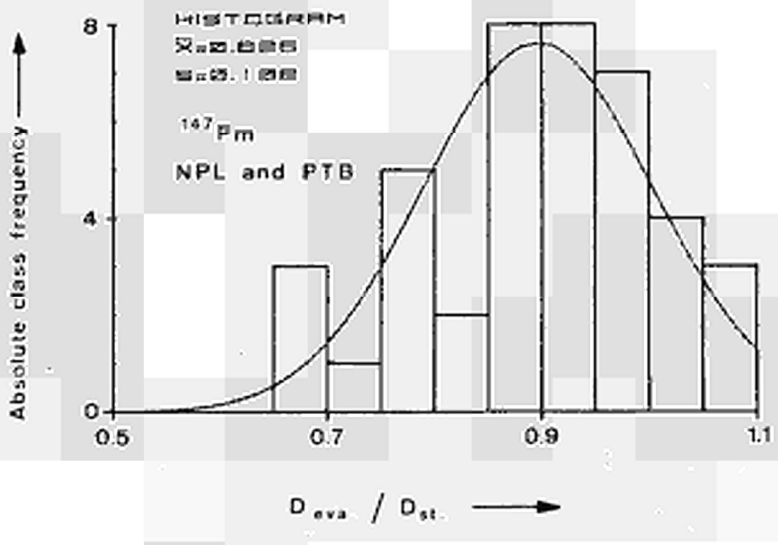
HISTOGRAM
147Pm
NPL and PTB
Absolute class frequency
8
4
0
0.5
0.7
0.9
1.1
D eva / D st

Figure 21

6. REVIEW OF THE INTERCOMPARISON

- Comments of the Editing Committee -

1. AIMS AND ORGANISATION

The long-term objective is to encourage the beta dosimetry services within the Community to operate at the level of performance recommended by the Community so that personal dose records in member countries may be compiled on a common basis of measurement. As an initial step this was the first Community personal dosemeter intercomparison for beta radiation and one aim of the exercise was to investigate the reproducibility of the dosemeter systems submitted by the participants over a range of doses and energies. An analysis has been made of the ratios of the normalised doses reported by the participants, to the doses stated by the irradiating laboratories; this analysis gives information about the precision of the dosemeter systems. The normalised doses for each type of dosemeter were derived by comparison with the 2 or 3 dosemeters of the same type with stated doses from each of the radionuclides Sr+Y-90, Tl-204 and Pm-147 which were sent to each participant with the 'unknown' irradiated dosemeters. The majority of participants also reported evaluated doses which were their assessment of the dose based on their own calibration technique. An analysis of the ratios of the evaluated to stated doses gives information about the accuracy of the dosemeter systems.

The main purpose of the comparison was to gain information about operational dosemeters but some experimental systems were also submitted and their results have been included in the analysis.

The irradiations were not carried out as planned due to unforeseen circumstances, personal and postal, which resulted in delays in irradiating some dosemeters and between the irradiation and read-out of some dosemeters. The NPL and PTB irradiations were carried out during February-March 1979 and the LMRI irradiations in June. The possible effects of these delays are discussed later and in future intercomparisons it would be important to attempt to avoid such delays.

2. ANALYSIS OF THE RESULTS

The CEC report EUR 5287 (1975), 'Technical Recommendations for Monitoring the Exposure of Individuals to External Radiation', section 4.10 suggests the following specification for the performance of a personal dosemeter :

(a) Random uncertainties on a single dosemeter should not exceed 20 %, at the 95 % confidence level, for doses approaching the maximum permissible; or for smaller doses, 10 % of the average permissible dose for the monitoring period, whichever is the greater.

(b) Systematic uncertainties should not exceed 40 % for doses approaching the maximum permissible; or for smaller doses, 10 % of the average permissible dose for the monitoring period, whichever is the greater.

With the publication of ICRP 26 (1977), the concept of 'permissible dose' has been replaced by 'dose-equivalent limit' which for radiation workers and irradiation of the skin is 500 mSv per year and for the lens of the eye is 150 mSv per year, ICRP 30 (1980).* Assuming monthly monitoring periods then the average monthly dose limits are 42 mGy for skin and 13 mGy for the lens. The dose range used in the comparison was from 2 to 120 mGy with most doses in the range 5 to 30 mGy, which is of the same order as the monthly dose limits for the skin and the eye lens. Thus the random and systematic uncertainties should be those recommended for doses approaching the dose limit.

In analysing the data it has been assumed that all the data for a particular dosemeter type and radionuclide are members of the same statistical population. In some cases this assumption is questionable, for example fading of some dosemeter information and the variation in the irradiation dose levels may affect the precision of the dosemeter system. So it should be borne in mind that the following analyses and conclusions are based on the limited set of data available from this comparison.

(a) Precision of the Dosemeter System

The recommended precision limits i.e. random uncertainty limits, are $\pm$ 20 % about the mean dose at the 95 % confidence level. The number of normalised/stated dose ratios reported for each nuclide and type of dosemeter ranged between 3 and 16. The 95 % confidence limits have been derived by multiplying the standard deviation for a single dosemeter by a factor of 2. The factor of 2 strictly applies to a large number of measurements so its use tends to underestimate the 95 % confidence limits and the use of the same factor for all dosemeter types irrespective of the number of measurements avoids penalising those types with relatively few measurements.

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* However the CEC Basis Safety Standards retains 300 mSv (30 rem) per year for the lens of the eye.

Table 1 lists the 95% confidence limits for each type of dosemeter and nuclide.

TABLE 1

TLD Systems		Sr + Y-90	Tl-204	Pm-147
Participant No.	Dosemeter type No.	95% confidence limits, %	95% confidence limits, %	95% confidence limits, %
1	1	± 16	± 4	± 18
2	3	± 6	± 10	
5	5	± 8	± 10	± 11
6	6	± 20	± 11	
6	7	± 9	± 18	± 50
6	8	± 23	± 27	
7	11	± 14	± 20	
8	12	± 3	± 4	± 8
8	13	± 3	± 7	± 17
9	14	± 24	± 17	± 18
9	15	± 4	± 5	± 26
10	17	± 14	± 24	± 39
12	19	± 43	± 72	± 80
Average 95% limits		± 14	± 18	± 30

Film Systems		Sr + Y-90	Tl-204	Pm-147
Participant No.	Dosemeter type No.	95% confidence limits, %	95% confidence limits, %	95% confidence limits, %
2	2	± 33	± 18	
4	4	± 54	± 47	
6	9	± 10	± 22	
7	10	± 19	± 36	
10	16	± 34	± 29	± 25
11	18	± 17	± 33	
Average 95% limits		± 28	± 31	

In view of the limited number of measurements and the method used for deriving the 95% confidence limits, there is some uncertainty on the limits given in Table 1 so the precision of the types of dosemeter has been divided into three categories : those types well-within the 20% limits, ie those with 95% confidence limits up to 15%, those types close to the limits, ie with 95% confidence limits from 16 to 24% and those well-outside the limits, ie with 95% confidence limits of 25% and greater. Table 2 summarises the precision of the dosemeter types falling within these three categories.

TABLE 2

TLD systems		Sr + Y-90	Tl-204	Pm-147
Total No. of dosemeter types		13	13	9
95% confidence limits	No. of dosemeter types	8	7	2
0-15%	Type Nos.	3,5,7,11,12, 13,15,17	1,3,5,6,12, 13,15	5,12
95% confidence limits	No. of dosemeter types	4	4	3
16-24%	Type Nos.	1,6,8,14	7,11,14,17	1,13,14
95% confidence limits	No. of dosemeter types	1	2	4
25% and greater	Type Nos.	19	8,19	7,15,17,19
Film systems		**Sr + Y-90**	**Tl-204**	**Pm-147**
Total No. of dosemeter types		6	6	1
95% confidence limits	No. of dosemeter types	1	0	0
0-15%	Type Nos.	9	-	-
95% confidence limits	No. of dosemeter types	2	2	0
16-24%	Type Nos.	10,18	2,9	-
95% confidence limits	No. of dosemeter types	3	4	1
25% and greater	Type Nos.	2,4,16	4,10,16,18	16

The averages of the 95% confidence limits for all TLD and film systems given in Table 1 show that for Sr + Y-90 and Tl-204 doses, TLD systems have about twice the precision, ie half the random uncertainty of the film systems. This is confirmed for doses greater than 25 mGy in Figs 9 and 10 in the paper by Böhm analysing the results of the intercomparison in these proceedings although Figs 11 and 12 show that for lower doses TLD and film systems have comparable precisions.

(b) Accuracy of the Dosemeter Systems

Seven laboratories reported evaluated data which first should be analysed to ensure satisfactory agreement of the stated doses from the three irradiating laboratories. The ratios of evaluated/stated doses have been averaged for each irradiating laboratory, radionuclide and type of dosemeter, the results are shown in Figs 1 and 2. The mean ratios for PTB and LMRI have been normalised to the NPL values and the average ratios calculated for each irradiating laboratory for all TL and film systems. The TL and film ratios were combined, weighting the mean according to the number of measurements in the two groups and their variances. The overall ratios compare the stated doses of PTB and LMRI with those of NPL and are given in Table 3.

TABLE 3

SUMMARY OF THE EVALUATED/STATED DOSE RATIOS AND THEIR STANDARD DEVIATIONS FOR PTB AND LMRI, RELATIVE TO NPL

	Sr + Y-90		Tl-204		Pm-147
	TLD	Film	TLD	Film	TLD
PTB/NPL	1.01±0.04	0.81±0.35	1.03±0.06	1.06±0.04	0.88±0.06
LMRI/NPL	1.00±0.06	1.08±0.07			
Combined TLD and film weighted ratios					
PTB/NPL	1.01 (1.01)		1.05 (0.98)		0.88 (0.93)
LMRI/NPL	1.02				

The agreement between the stated doses from the different irradiating laboratories is within 2% for Sr+Y-90, 5% for Tl-204 and 12% for Pm-147. Direct measurement of sources exchanged between NPL and PTB gave agreement of the primary standards within 1%, 2% and 7% respectively, the ratios are shown in brackets in Table 3.

Having shown that the stated doses from the different irradiating laboratories are in satisfactory agreement, the evaluated/stated dose ratios for all irradiations can be combined to give an average ratio and standard deviation of a single dosemeter for each nuclide and type of dosemeter, the results are shown graphically in Fig 3. The average evaluated/stated dose ratio has some uncertainty associated with it and this uncertainty has been taken as the standard deviation of the mean ratio at the 95% confidence level (assuming Student's t = 2); this uncertainty has been added to or subtracted from the average ratio to give the average ratio with the maximum difference from unity. Table 4 gives the average evaluated/stated dose ratio, the 95% confidence limits and the average ratio after the addition or substraction of its uncertainty (max. ratio).

TABLE 4

TLD Systems Type No.	Sr + Y-90 Ratio	95% confidence limits	Max. ratio	Tl-204 Ratio	95% confidence limits	Max. ratio	Pm-147 Ratio	95% confidence limits	Max. ratio
1	1.08	± 4%	1.12	1.00	± 1%	0.98	0.94	± 6%	0.89
3	1.21	± 2%	1.23	0.82	± 3%	0.80			
5	1.03	± 2%	1.05	0.99	± 5%	0.94	0.89	± 4%	0.86
12	1.00	± 1%	1.01	1.03	± 1%	1.04	0.93	± 3%	0.91
13	1.02	± 1%	1.03	1.02	± 2%	1.05	0.78	± 5%	0.74
11	0.88	± 3%	0.85	0.66	± 7%	0.62			
Film Systems									
2	0.95	± 11%	0.86	1.09	± 6%	1.15			
4	0.90	± 14%	0.79	1.08	± 15%	1.23			
18	0.93	± 4%	0.89	1.13	± 10%	1.25			

The required accuracy limits i.e. the recommended systematic uncertainty limits are ± 40 % about the evaluated/stated dose ratio of unity. From Table 4 it can be seen that all 9 types of dosemeter with the exception of type No. 11, have accuracies well-within the ± 40 % limits; type No. 11 (TLD, Tl-204) has an accuracy close to the limit.

(c) Overall Performance of the Dosemeter Systems

Table 5 summarises the precision and accuracy of the dosemeter systems and gives the number of types of dosemeter in the three categories : well-within the recommended limits, close to the limits, and well-outside the limits.

TABLE 5

	Precision limits ± 20 %			Accuracy limits ± 40 %		
	Within limits	Close to limits	Outside limits	Within limits	Close to limits	Outside limits
Sr + Y-90	9	6	4	9	0	0
Tl-204	7	6	6	8	1	0
Pm-147	2	3	5	4	0	0

3. CONCLUSIONS

It is important to bear in mind that the following conclusions are based on the limited data from this intercomparison. The number of participants, the number of dosemeters and the dose ranges were all limited in this comparison and the results were a mixture of those from systems under development as well as from operational dosimetry systems. The conclusions have been drawn from the available results but care should be taken before interpreting the conclusions are applying to personal beta dosimetry in general.

The general conclusions drawn from this comparison are as follows :

(a) The precision of the dosemeter systems with respect to the Community's

recommended limits of ± 20 % at the 95 % confidence level, is shown in Table 5. The precision of the dosemeter systems generally cannot be considered entirely satisfactory and consideration should be given to improving the precision of those systems close to the limits, and those well-outside the limits should either be improved in their precision or abandoned in favour of a more precise system.

(b) The accuracy of all 9 dosemeter systems for evaluating Sr + Y-90 and Tl-204 doses satisfied the Community's recommended limits of ± 40 %, as did all 4 systems evaluating Pm-147 doses. Only 1 system was close to the limits for Tl-204 doses and consideration should be given to improving the accuracy of this system. In general, it is considered that the accuracy of the dosemeter systems was satisfactory.

(c) Comparing the performance of TLD and film systems :

(i) TLD and film systems had similar accuracies, but in general the TLD systems had somewhat better precision, about 1/2 the random uncertainty of film systems.

(ii) None of the film systems was able to evaluate low energy beta doses from Pm-147 due to absorption of the beta-radiation in the film packaging. However some film and TLD systems had significant responses to bremsstrahlung and the 22 keV Ag K X-radiation from the silver encapsulation of the Pm-147 source.

(iii) Film systems had, in general, built-in energy discrimination and required no further knowledge of the irradiation energy whereas most TLD systems lacked energy discrimination and required this knowledge for accurate dose assessment.

(d) There are some reservations about the state of personal beta dosimetry within the Community, revealed by this comparison, as to whether the results are typical of the performance of routine dosimetry services throughout the Community for the following reasons :

(i) It is not clear how much of the data came from routine dosimetry services as opposed to systems under development and to what extent the results from these systems may have biased the conclusions.

(ii) Most TLD systems require knowledge of the radiation energy for accurate dose assessment and this was available in this inter-

comparison. It is uncertain what the effect of unknown or mixed photon/beta irradiations would have on the results.

(iii) Most participants in the comparison had access to PTB or NPL secondary standard sources for calibration purposes whereas many other dosimetry services rely on other means of calibration.

As this has been the first intercomparion of this kind the number of participants was limited. Therefore the participants in this exercise do not represent all the communities major dosemeter services in terms of the number of people monitored for beta radiation.

(e) In view of these reservations on the results of this first comparison, it would be worthwhile to repeat the exercise under the following conditions :

(i) All the Community's major beta dosimetry services should be invited to participate.

(ii) The comparison should concentrate on routine dosimetry systems rather than on experimental systems.

(iii) The irradiation energies should be unknown to the participants and no calibrating doses should be supplied.

(iv) Mixed photon/beta irradiations should be included in the exercise.

(v) The participants should be encouraged to report all their evaluated results.

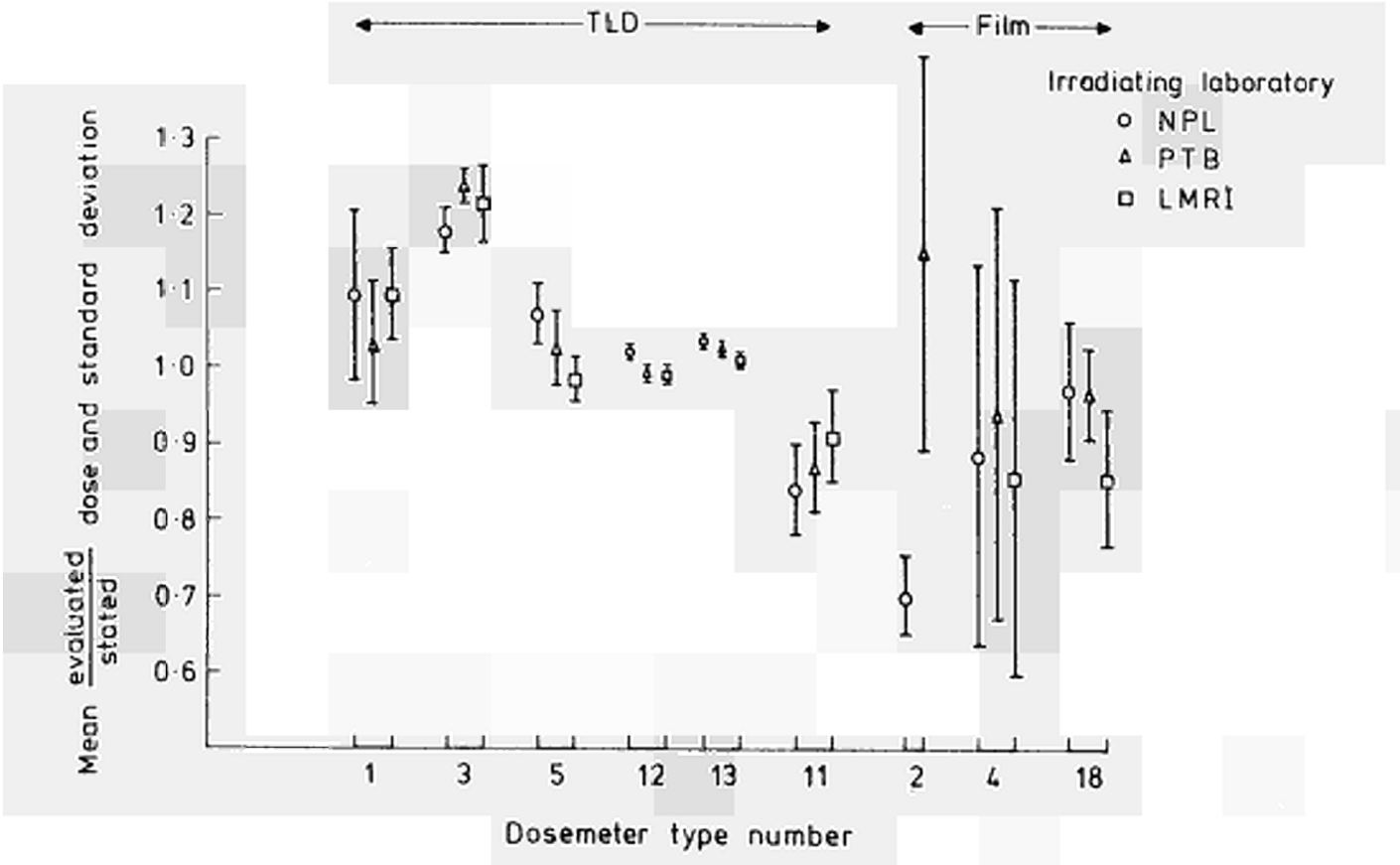

FIG. 1. Sr + Y - 90 IRRADIATIONS

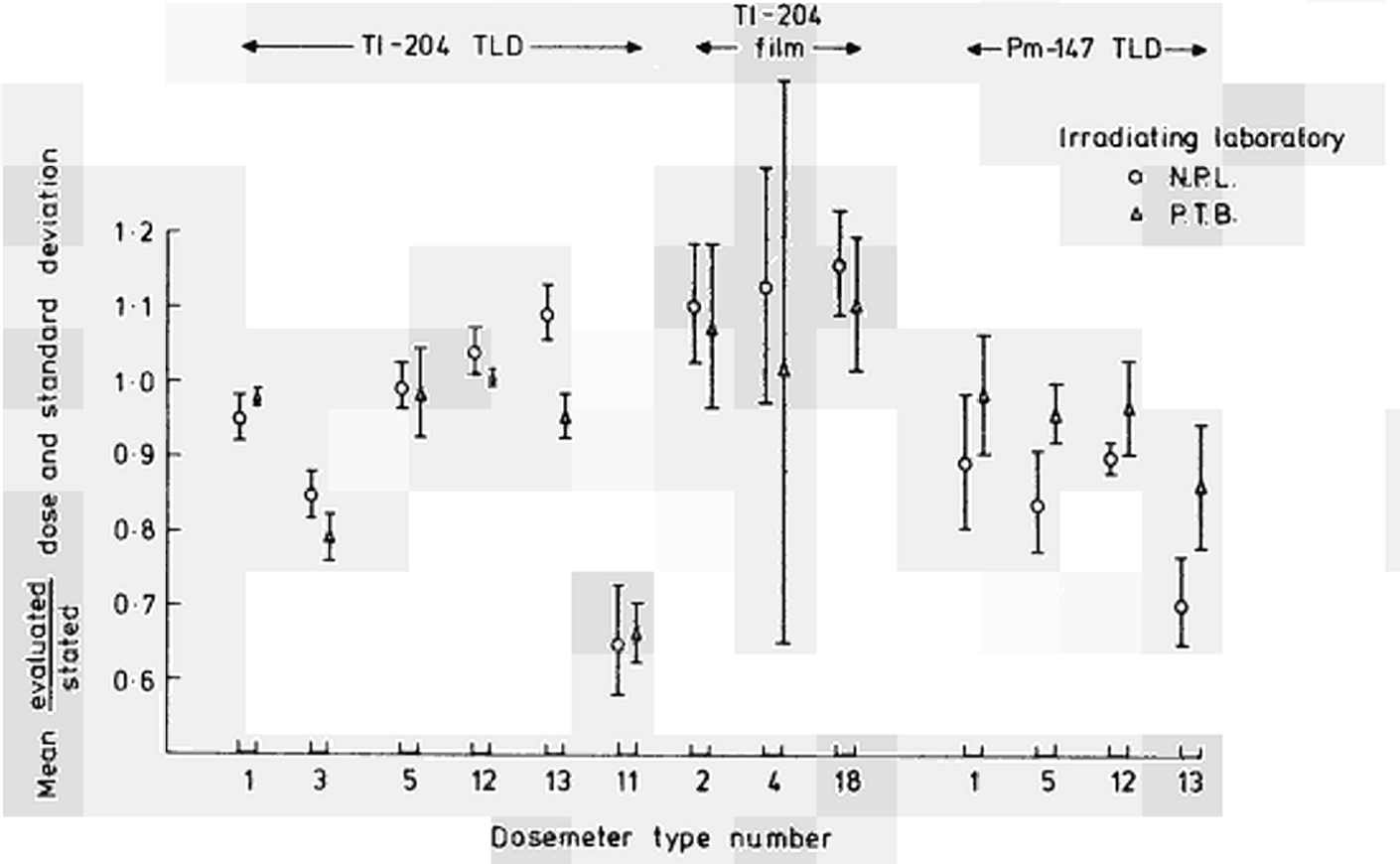

FIG. 2. Tℓ - 204 & Pm - 147 IRRADIATIONS

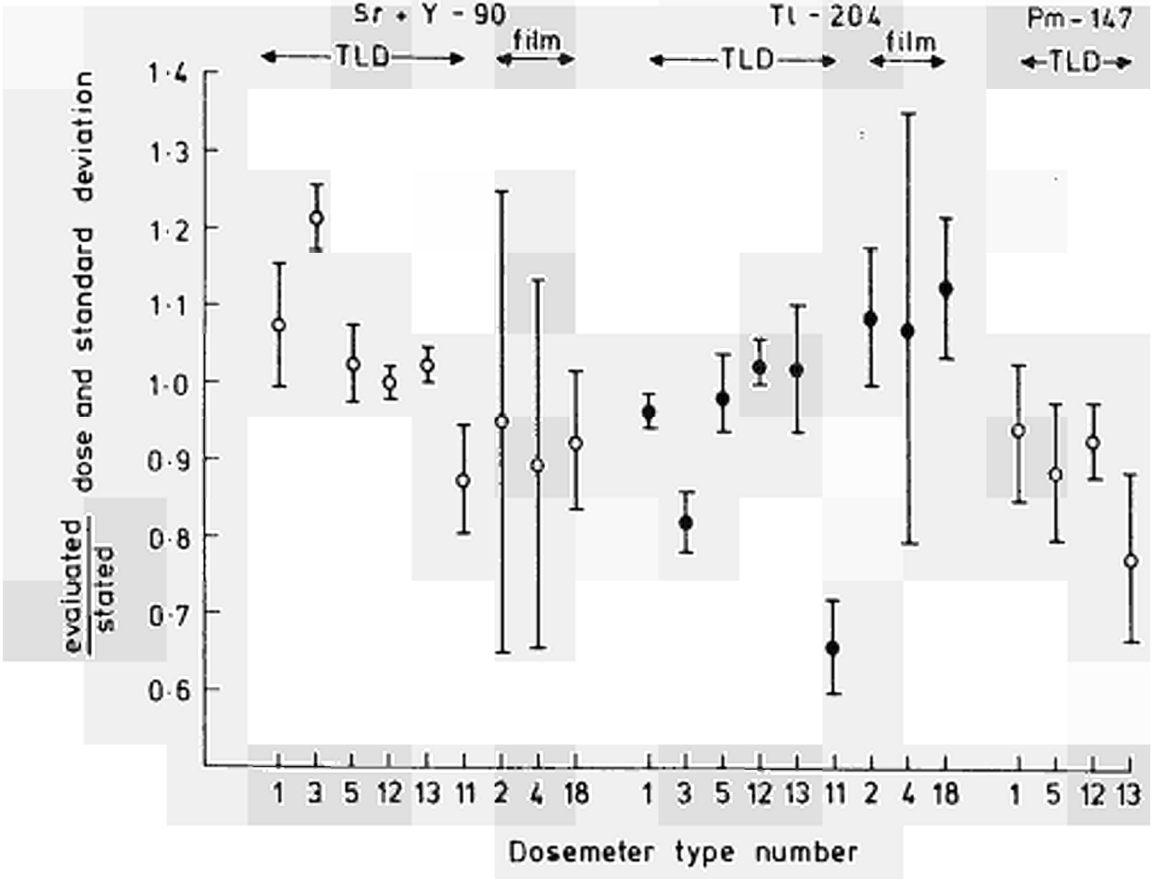

FIG. 3. OVERALL MEANS

Session III

Chairman : G.H. HOFMEESTER

7. FUTURE TRENDS IN BETA DOSIMETRY

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1. INTRODUCTION

Any future trends in beta dosimetry must be influenced by the existing legislation on dose limits, by the latest available data on the structure of the relevant biological organs and by the impending change to SI units. Any dosimetry system developed should also ideally be flexible enough to allow for possible future changes in the dose limits and for new or improved data on the organs that can be irradiated by external beta radiation.

This report deals with various dose limits applicable to beta radiation, the relevant dimensions of the human skin and eyes and the significance of the beta dose in radiation fields associated with the operation of the CEGB nuclear power stations.

Recent developments in personal dosimetry systems and radiation measuring instruments are discussed and the International Electrotechnical Commission approach to the practical measurement of the Shallow dose equivalent index ($H_{I,S}$) is also briefly described.

Any system used for the measurement of beta dose requires proper evaluation and calibration and recent recommendations by the ISO on beta reference radiations for these purposes are given.

2. DOSE LIMITS APPLICABLE TO BETA RADIATION

The ICRP make recommendations on appropriate limits for any occupational or other exposure to radiation. In ICRP Publication 26 the Commission's recommendations are intended to prevent non-stochastic effects and to limit the occurrence of stochastic effects to an acceptable level. The Commission believes that non-stochastic effects will be prevented by applying a dose-equivalent limit of 0.5 Sv (50 rem) in a year to all tissues except the lens, for which a limit of 0.3 Sv (30 rem) in a year is recommended. The Commission is only able to offer guidance and recognises itself that because of the differing conditions that apply in various countries, detailed guidance on the application of its recommendations,

either in regulations or in codes of practice, have to be made by the various international and national bodies that are familiar with what is best for their needs.

This can be illustrated by the differing requirements of the European Communities and the USA for the limitation of non-stochastic effects. In Article 9 of the European Council Directive of 15 July 1980, the requirement is that the dose limit for the lens of the eye shall be 300 mSv (30 rem) in a year with a limit of 500 mSv (50 rem) in a year applicable to the skin.

This contrasts with the requirements for the USA given in NCRP Report 39 where the occupational exposure limit for unlimited areas of the skin (other than hands and forearms) is 15 rem in any one year. The dose equivalent limit to the hands is 75 rem and the forearms 30 rem in any one year. The lens of the eye is singled out for a limit of 50 mSv (5 rem) in a year.

The USA limit for the eye is thus a factor of 6 lower than that specified within the European Community. Recent biomedical evidence has led the ICRP to reconsider their recommendations for the eye limit and this has been lowered to 150 mSv (15 rem). The fact that there are different limits for the same organ in various countries is of practical importance to the designer of new personal monitoring systems or monitoring instruments. Not only may these different limits imply differing system sensitivities but they may also effect the basic design philosophy of the detecting element.

3. RELEVANT BIOLOGICAL ORGANS

In the previous section it was shown that within the CEC the lens of the eye and the skin have higher values of dose limits than the whole body. In radiation fields having a significant fluence of beta or low energy photon radiation the spatial dose distribution may be considerably non-uniform and the superficial tissues may receive the limiting dose rather than the whole body. This has led in recent years to many investigations and measurements of the depth of the basal layer in the skin and of the depth of the epithelial cells in the equatorial region of the lens of the eye.

Values of the depth of the basal layer for different locations of the body have been measured by whitton and are shown in Fig. 1. Apart from the finger tips, palms and soles where the thickness is within the range 30 to 50 $mg.cm^{-2}$ the majority of sites have values of 4 to 8 $mg.cm^{-2}$. The histograms show that a few individuals have a skin thickness of only 2 $mg.cm^{-2}$. Based upon such measurements the ICRP 1977 proposed that the skin dose be measured by the mean value between depths of 5 and 10 $mg.cm^{-2}$ for most parts of the skin that are unprotected by clothing. For practical dose assessments they recommended that this could be determined by measurements at a depth of 7 $mg.cm^{-2}$. For the measurement of skin dose an ideal detector should therefore have a 5 $mg.cm^{-2}$ deep tissue equivalent detector beneath a 5 $mg.cm^{-2}$ tissue equivalent "window".

A detailed investigation of the cells at risk in the lens of the eye has been made by Charles and Brown. They observed that the epithelial cells in the equatorial region of the lens are those involved in cataract induction and that for people aged between 20-65 years, these were at a depth of 2.3 $\pm$ 0.4 mm. Their calculations for beta radiation showed that the mean equatorial dose could be reasonably measured using a planar dosemeter integrating between 2.5 and 3.5 mm. Thus the ideal detector for measurement of the eye dose would be a 100 $mg.cm^{-2}$ thick tissue equivalent detector beneath a 250 $mg.cm^{-2}$ tissue equivalent window.

4. RADIATION FIELDS ASSOCIATED WITH THE OPERATION OF NUCLEAR POWER STATIONS

Many of the maintenance and repair jobs arising in the operation of power stations and its associated research involves the close handling of contaminated and activated materials. The control of such jobs depends more upon the dose delivered to the skin than upon the whole body doses. Beta/gamma ratios are observed ranging from 48:1 to 800:1 for fission product contamination from natural uranium irradiated under Magnox reactor conditions.

Investigations of the depth dose within the CEGB were determined using a stack of LiF/teflon dosemeters. Fig. 2 shows typical depth doses from different areas around cooling ponds, contaminated by aged fission products at levels of about 10^{-4} $\mu Ci.cm^{-2}$.

Clearly there is a need in these areas to accurately measure the tissue absorbed dose rate for control purposes and the tissue dose received by the individuals working there.

5. MEASUREMENT OF BETA DOSE

5.1. Personal dosimetry

The size of the photographic dosemeter precludes its use for the measurement of extremity doses, where a small dosemeter that does not interfere with the operators movements is required. TLDs are therefore almost exclusively used for measurement of the extremity dose. These dosemeters are usually the same as those used for assessment of whole body doses and are therefore too thick (typically 30 to 80 $mg.cm^{-2}$) for measurements of the skin dose between tissue depths of 5 and 10 $mg.cm^{-2}$. Recently a number of ultra thin dosemeters have been developed. The ultra thin bonded (UTB) dosemeter of Charles and Khan consists of a 6 $mg.cm^{-2}$ LiF teflon disc which is bonded to a thick ($\sim$0.2 mm) teflon disc to make the dosemeter more robust. A 40 μm thick Melinex window simulates the overlying tissue. Calculations based on the beta attenuation data of Cross show that the detector and its window are both within a few percent of the required equivalent tissue mass thickness of 5 $mg.cm^{-2}$. Its minimum detectable dose (2 σ above background) is approximately 1.7 mGy (170 mrad) but this can be reduced by a factor of 2-3 by using special sensitising and annealing procedures. The response of this detector as a function of beta energy is shown in Fig. 3 together with that of three other commonly used dosemeter types.

The UTB dosemeter has a good response at all energies whereas the other types are within $\pm$ 30% of the required dose for energies above 0.5 MeV, whilst at 0.15 MeV they can be low by factors of 5 to 25. Ultra thin dosemeters were not available in 1975 when the Commission of the European Communities published their technical recommendations on the use of TLDs. Whilst recognising the need to measure skin dose at a depth of 5 to 10 $mg.cm^{-2}$ they found it expedient to specify a cut-off energy of 0.5 MeV due to either the thickness of the window or of the existing detectors themselves. Such TLDs could of course be used at energies below 0.5 MeV provided they were specially calibrated for known and invarient beta fields. TLDs having detectors that are 20 $mg.cm^{-2}$ thick with a 30 $mg.cm^{-2}$ window are more suited for measuring the skin dose to the hands whereas UTB-TLDs are more appropriate for the measurement of skin dose to exposed, uncovered areas of the arms, legs and face.

5.2. Beta survey instrumentation

Most existing survey instruments used for assessing beta dose are exposure rate instruments with an end cap which when removed leaves a thin end window exposed to the radiation. The difference between the exposure rate readings with the cap removed and with the cap in position is normally equated to the low penetrating or beta dose rate. Different manufacturers produce instruments whose thin end windows have a thickness in the range 1 to 30 $mg.cm^{-2}$. Now that the impending introduction of SI units sounds the death knell of the quantity exposure the purist can rejoice that the same quantity will be used for operational measurements of doses from beta and photon radiation. It has proved extremely difficult, however, to find a universally accepted replacement quantity for exposure. Even when agreement is reached on this quantity the biggest errors in the assessment of beta dose rates will still result from geometrical factors.

The Eberline RO2 is probably one of the most widely used beta/gamma survey instruments in the UK. It has a 208 cm^3 volume ionisation chamber of diameter 7.62 cm. Its chamber walls are constructed from 200 $mg.cm^{-2}$ thick Phenolic and its end cap of the same material is 400 $mg.cm^{-2}$ thick. The thin end window has a total thickness of 7 $mg.cm^{-2}$. Fig. 4 shows the beta response of the RO2 to a range of $^{90}Sr/^{90}Y$, ^{204}Tl and ^{147}Pm sources for various source to instrument end window distances.

A CEC Working Party on Calibration Problems is likely to require that a detector's reference point for calibrations performed with beta radiation shall be specified by the manufacturer. To make his instrument appear to have a good beta response a "wise" manufacturer will most likely soecify the geometrical centre of the chamber. If this is done for the RO2 the beta response is modified to that shown in Fig. 5. Clearly this response would appear to be better, but surely it should be argued that the purpose of calibration and radiation protection measurements is to prevent individual workers being subjected to excessive radiation. By calibrating to the front entrance window the user will have a much better indication of the errors likely to be obtained for dose measurements approaching surfaces, and manufacturers will be encouraged to design purpose-built instruments. Perhaps a future trend in beta dose measuring instruments will be the appearance of a commercial instrument having a response similar to that shown in Fig. 6. This instrument's ionisation chamber has a depth of 4 mm and a window thickness of 7 $mg.cm^{-2}$. The small volume of the

chamber limits its minimum reading to 100 mrad.h^{-1} but advances in electronics would more readily permit the measurement of currents of 10^{-15} A produced at 1 mrad.h^{-1}.

One further trend is likely to be the development of commercial instruments to measure dose rates associated with the release of the long lived noble gas Krypton 85 during fuel reprocessing. For example submersion in a ^{85}Kr concentration of 3 pCi .cm^{-3} gives a whole body dose rate of 100 mrem per 168 hour week.

A survey instrument has been developed by Hajnal for the measurement of beta/gamma radiation fields around fuel reprocessing plants. The concentric, cylindrical ionisation chamber design can measure and display the beta and gamma components of the field. The outer beta-ray chamber has walls 7.7 mg.cm^{-2} thick and the inner "gamma", aluminium-wall ionisation chamber is completely separated and guarded from the outer chamber. For both the beta and gamma components the dose rate range is from about 0.1 mrad.h^{-1} to 10 rad.h^{-1}.

To attempt to make a practical measurement of the Shallow dose equivalent index the IEC have produced a draft standard on dose equivalent instruments which defines the requirements for the measurement of skin dose equivalent. The variation of response with beta radiations energy requirements are as follows : E_{max} 100 keV to 500 keV - 50% to + 100%, E_{max} 500 keV to 1 MeV $\pm$ 35% and E_{max} 1 MeV to 4 MeV $\pm$ 30%.

6. BETA REFERENCE RADIATIONS FOR THE ASSESSMENT AND CALIBRATION OF BETA DOSEMETERS AND DOSERATEMETERS

The International Standards Organisation have completed a draft proposal on beta reference radiations which is at present subject to voting. Clearly it is important to be able to compare the results of tests done in different laboratories. A necessary step in achieving this is the use of well proven and internationally accepted methods of producing the radiations and of standardising the radiation fields.

Two series of radiations have been listed. Series 1 consists of radionuclides of $^{90}Sr/^{90}Y$, ^{204}Tl and ^{147}Pm used with beam flattening filters designed to provide uniform dose rates over a large area at a specified distance. This series has been used in this present beta intercomparison. Series 2, which does not use beam flattening filters, provides

higher dose rate by using smaller source to detector distances than series 1 and also extends the energy range by the addition of ^{14}C and $^{106}Rh/^{106}Ru$.

One future trend in calibration will most likely be the extended use of these ISO beta reference radiations.

7. CONCLUSIONS

Future trends in beta dosimetry may include the use of several personal dosemeters located on the individual at the specific exposed organs and each uniquely designed to measure the dose at the depth of that organ. They will be used to demonstrate compliance with dose limits that are specific to individual organs and depths.

The rapid development of integrated circuits will permit the use of more sensitive radiation detectors which will consequently be more readily designed to have a response to match that of the exposed organs.

Other more likely trends will be the use of the same quantity for the measurement of dose from both beta and photon radiation, all calibrations and assessments on beta instruments being performed with the ISO beta reference radiations and the appearance on the market of more automated TLD systems for the measurement of extremity doses.

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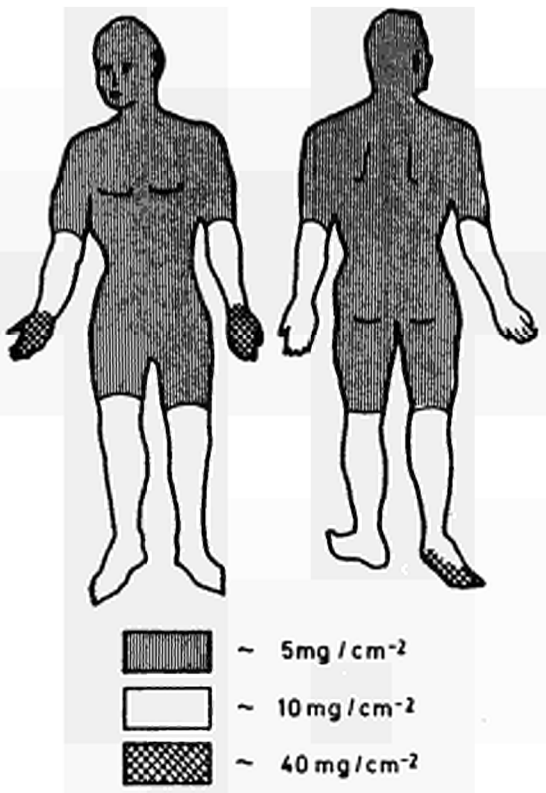

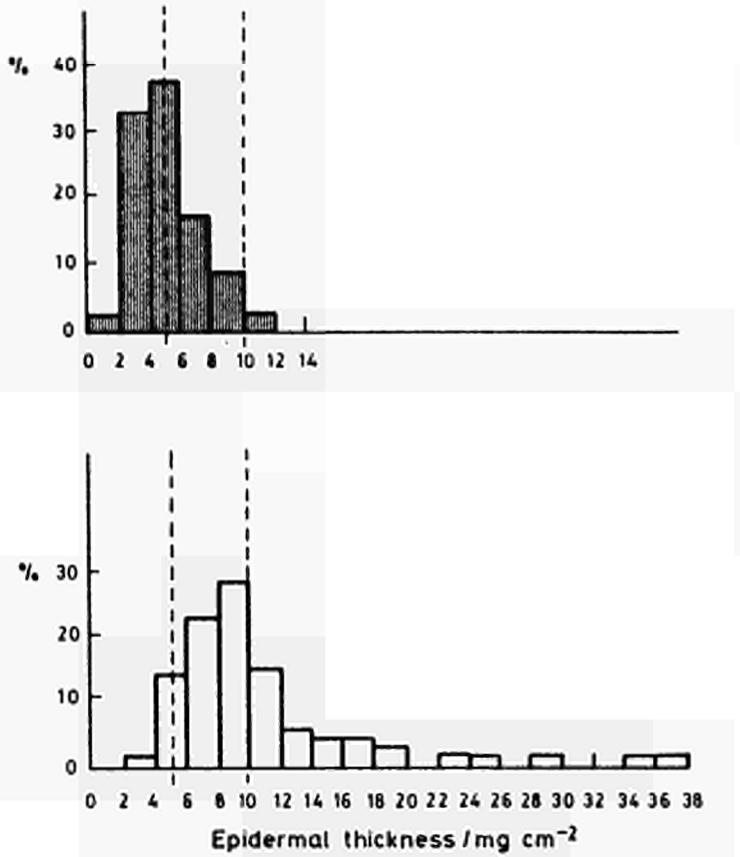

FIG. 1. Variation of Epidermal Thickness with Body Site.

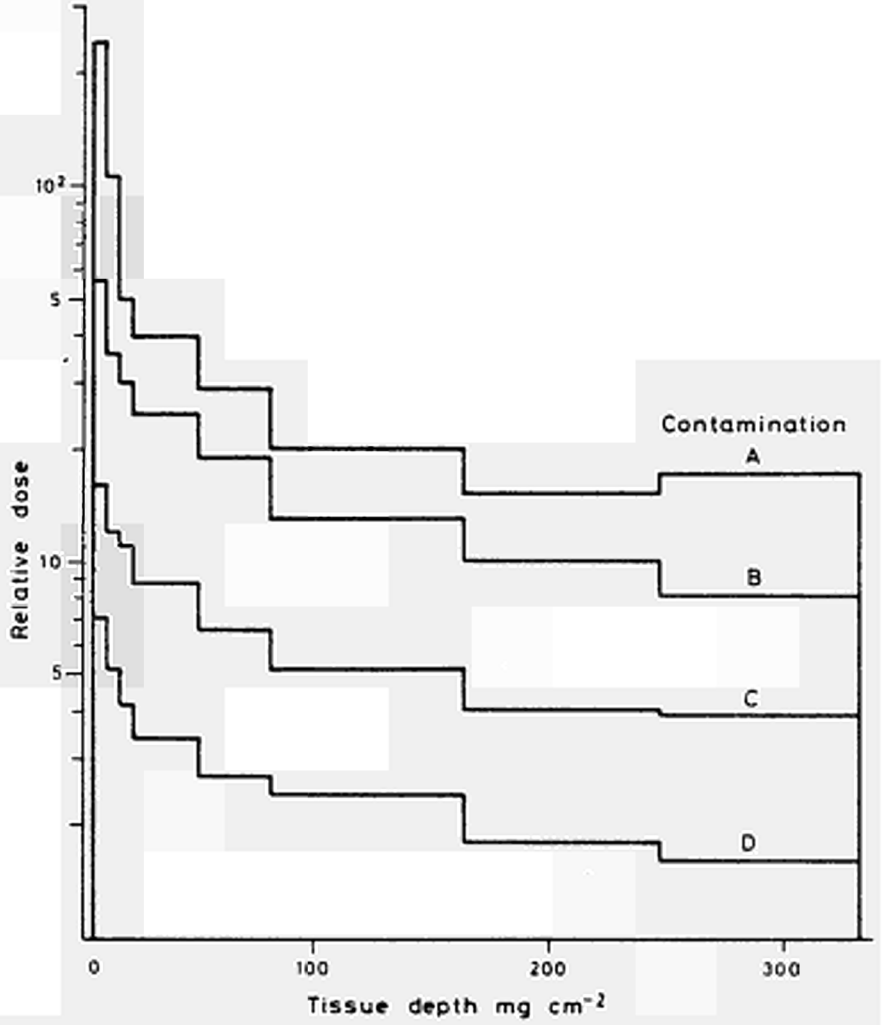

FIG.2. Typical Depth Doses in Magnox Pond and Boiler Sites.

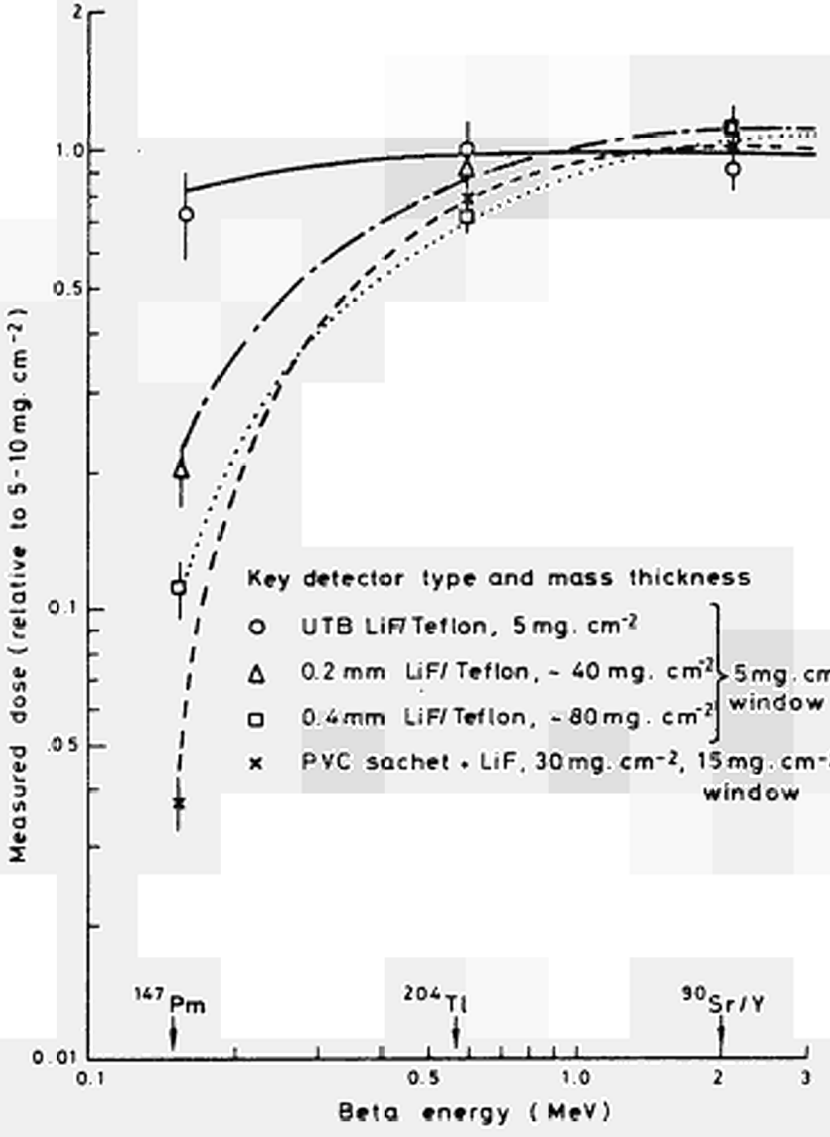

FIG.3. The Beta Response of Various Skin Dosemeters.

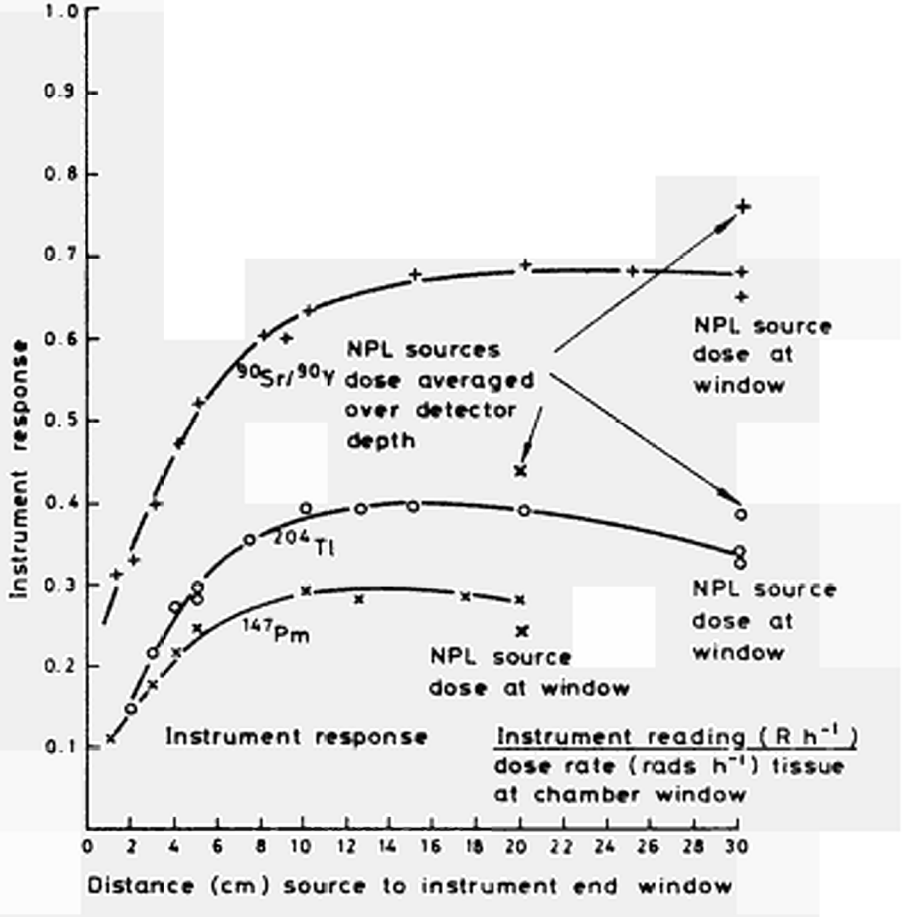

FIG. 4. Beta Response of Eberline Portable Ion Chamber, Model RO-2

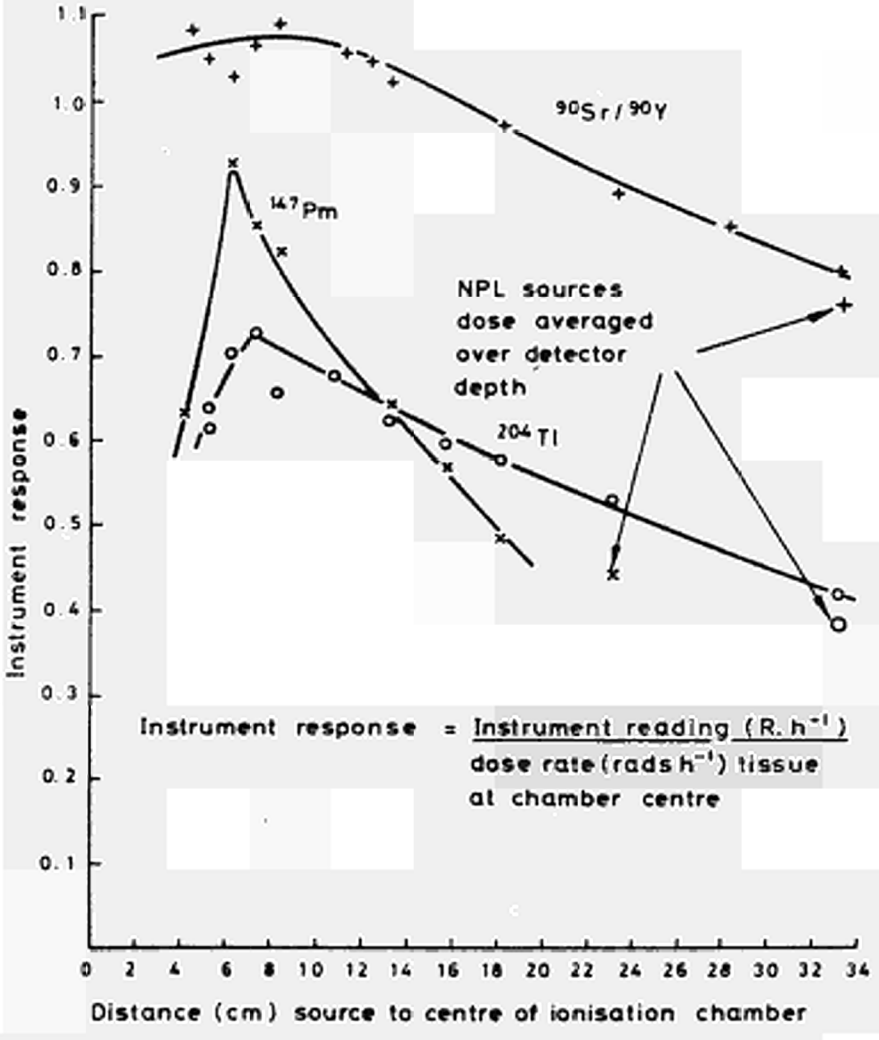

FIG. 5. Beta Response of Eberline Portable Ion Chamber Model RO-2

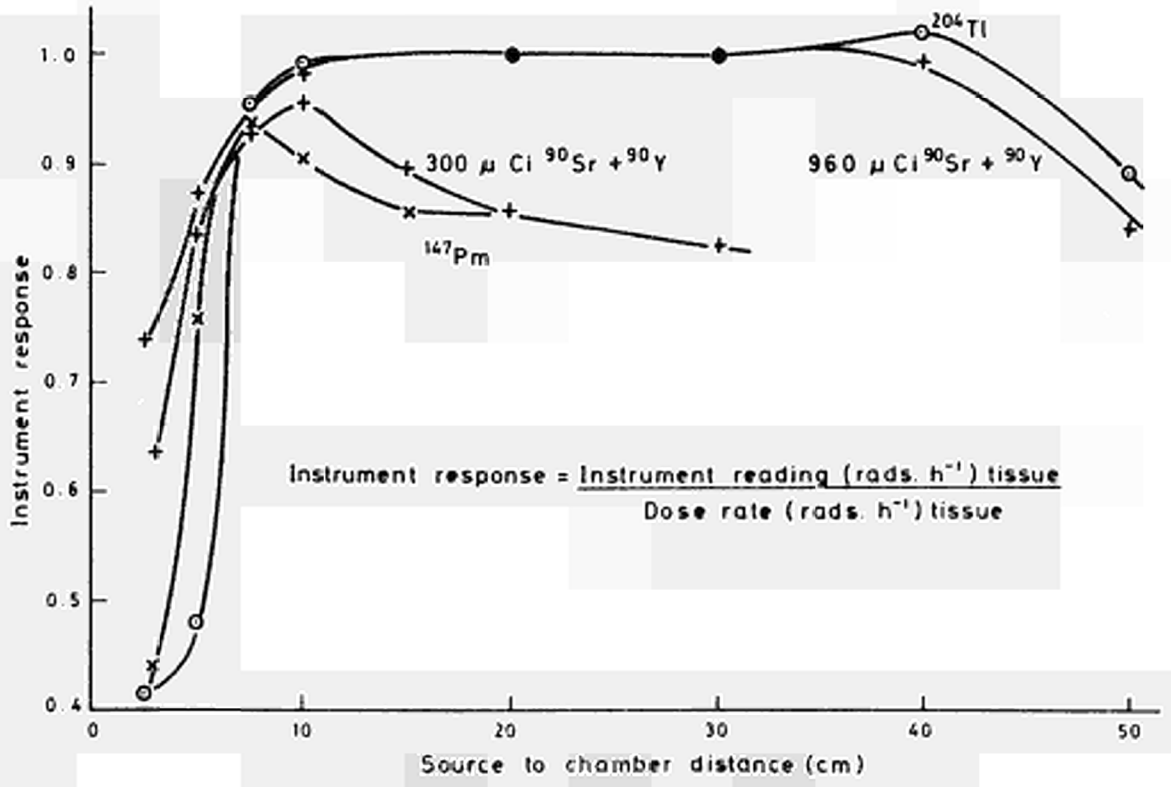

FIG. 6. Beta Response of B.N.L. 3 Survey Meter.

8. FUTURE INTERCOMPARISON PROGRAMMES

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The following considerations are intended to help further discussions rather than provide a readily and easily applicable approach for further intercomparison measurements in beta dosimetry for radiation protection purposes.

Trying to characterize the last intercomparison on the basis of the reports presented during this seminar, it could be said that this was more a workshop than an intercomparison proper, as the methods, intentions, and expectations varied among the participants according to the available technical facilities, and the degree of experience and sophistication in the laboratories involved. Nevertheless, this first beta dosimetry intercomparison can be regarded as a worthwhile and valuable undertaking, and in particular, as the beginning of future joint efforts.

There is no need to elaborate on existing discrepancies among so-called primary standards of the national laboratories, as these will be the subject of further investigations in and among these laboratories. Notwithstanding these endeavours it will certainly be important to agree internationally upon the operational quantity to be measured in beta dosimetry. There is much in favour of the absorbed dose in a defined depth e.g. 70 μm in a phantom, the shape of which is not at all critical for beta dosimetry, and which consists of standard soft tissue as defined by ICRU (1). The logical though not, of course, inevitable consequence of such a choice would then be to establish a primary standard realizing the unit of this quantity in order to keep any corrections small when calibrating operational instruments, i.e. in disseminating the unit of the operational quantity. But establishing the fundamentals of the necessary procedures will again be the duty of the national laboratories.

To come now to the aims of intercomparison measurements, these are as follows :

- to verify and control the standard of performance, and
- to ensure that this standard of performance does not deteriorate in the course of time as a consequence of careless routine.

This also applies to well-established methods of measurement. As an example, reference is made to German legislation, according to which individual film dosemeters for photon radiation are exempted from the requirement of official verification if they are issued and evaluated by officially recognized laboratories participating in an annual intercomparison organized and supervised by an authorized standard laboratory. In individual beta dosimetry, however, this situation has not yet been attained.

The principal goal of future intercomparison must then be

- to encourage and promote the further development and improvement of existing methods,
- to stimulate the development of new methods and to provide means of testing them reliably; and finally
- to provide assistance in developing a simple and sufficiently accurate method for routine use which is applicable under normal working conditions without requiring exceptional skill.

For this, in future intercomparisons, it would appear to be necessary

- to leave the calibration of individual dosemeters to the laboratory or establishment which issues and evaluates them,
- to irradiate the dosemeters with beta radiation, the quality and quantity of which is unknown to the participant,
- to use mixed beta and photon radiations, and
- to use lower doses besides those used in the last intercomparison.

In order to facilitate and improve the calibration of dosemeters by the participants, information on the depth dose distribution should be included in the certificate specifying the beta sources used thereby.

Concerning the choice of beta energies used in intercomparisons, it does not appear necessary to put much emphasis on very low energies from the beginning, as these raise special problems as to the appropriate location of the dosemeters on the exposed person. For example, the finger tips are frequently the most exposed part of the body when handling low energy beta sources, but it does not seem to be expedient to place a dosemeter there.

Special care should be taken over the decision on the quantity to be determined in beta dosimetry. It should, of course, be satisfactorily

measurable, but it must also fit into a consistent scheme of quantities for radiation protection. This means that it must be possible to combine it directly, or after appropriate but simple conversion, with the quantities used with other kinds of ionizing radiation, in such a way that the sum can be compared with the primary or secondary radiation protection limits without undue uncertainty.

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PART B

General topics of interest

Session IV

Chairman : G. PORTAL

1. LEGAL REQUIREMENTS ON BETA DOSIMETRY

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INTRODUCTION

The topic I have been asked to deal with at this Seminar refers to an aspect of external irradiation and to the provisions governing in this connection items such as maximum permissible doses, dose limits, dose evaluation and related surveillance; this item however may be brought within the general radiation protection surveillance measures.

The aim of this contribution being essentially that of stimulating discussion, I shall limit myself to pointing out briefly what the main problems seem to be and what the regulatory position appears to be like in some countries of the Community in connection with the above points. Regarding the latter, France, Germany and Great Britain, besides my own country, are considered.

1. INTERNATIONAL ORIGIN OF LEGISLATION

The domain of beta irradiation (involving mainly, as it is known, skin and lens of the eye) is definitely among the sensitive ones within radiation protection legislation which, as it is also known, has a common basis in the Euratom Directives whose latest revised version has just entered into force.

Apart from other international sets of standards (ICRP, IAEA) dealing with the subject, we would like to recall briefly the above Directives as they firstly appeared in 1959 (1) and have evolved so far, with particular reference to the provisions expressly touching upon skin and lens of the eye.

The Directives originally stated maximum permissible doses for skin external irradiation of occupationally exposed workers (8 rems in 13 weeks and 30 rems in a year) while the lens yearly dose (5 rems) was assimilated to that provided for whole body irradiation. Dose limits were also stated for the lens regarding particular groups of the public (1.5 rems and 0.5 rems).

The 1966 Directives (2), with reference to the "special external irradiation" context, extended the yearly limits of normal irradiation to the case of planned special irradiation; in the case of unplanned such irradiation, the limits of 60 rems for skin and 30 rems for the lens of the eye were introduced.

According to the next version (1976) (3) of the Directives, with reference to "partial body doses" the quarterly limit for skin irradiation of workers is raised to 15 rems, while in the case of planned special exposure the dose received "must be kept lower than one half of the annual limits" adopted as partial maximum permissible body doses. Moreover, yearly dose limits for members of the public were established, resulting in 3 rems for skin exposure and 1.5 rems for the lens of the eye.

Finally, the latest revision of the Directives (1980) (4) on considering the "partial body exposure" raises the yearly limits for skin and lens irradiation of workers to 50 and 30 rems respectively; corresponding limits for persons of the public are provided being ten times lower than the above. Yearly limits for hands skin vary throughout the time, ranging from 15-60-120 rems (1959 and 1966) to 75 (1976) to 50 (1980) for workers and from 60 rems (1959) to 7.5 (1976) to 5 (1980) for the public. Some principles useful for individual dose assessment are also introduced under "collective and individual monitoring of exposure".

The Directives also contain general provisions regarding dose assessment and recordings (surveillance) as well as controls to be performed by the competent authorities (supervision and inspection).

All the above illustrated changes have taken place over a time-span of more than twenty years, taking into account the evolution in attitude at an international level with regard to radiation protection in general expressed especially in the ICRP recommendations. The International Commission on Radiological Protection, after publication of No. 26 volume (1977) (5) has also produced a statement in 1978 (6) touching upon - among several points - whole skin exposure to beta radiations; another statement of March 1980 (7) considers doses accumulated in the lens of the eye and their effects on the basis of the ICRP 26 corresponding recommendations.

On the other hand, in Publication 12 (1969) (8) the Commission had already dealt with items such as individual monitoring and dose surveillance of external irradiation and monitoring in the case of skin contamination.

These principles lie under the 1980 Directives, although some parts could not be made wholly consistent with the latest ICRP positions, as for example concerning the limits for skin and lens of the eye. But it is equally important to remember that so far the adoption of the Directives in the Member countries is not a uniform one, due to several reasons such as national institutional framework, enlargement of the Community, internal difficulties, etc.

2. TECHNICAL LEGAL REQUIREMENTS FOR CONSIDERATION

Having recalled the main interested features of the Directives, we should now try and outline what we think could be the main elements for consideration as a technical basis for a legal regime of beta irradiation and dosimetry.

One aspect concerns dose evaluation, regarding which it should be stated if such evaluation ought to be carried out by instruments or otherwise by appropriate calculation; alternatively, indication should be given on whether the real dose amount should be worked out or rather an indication such as "lower than ..." would be acceptable ?

Another aspect regards the persons concerned by the dose evaluation, that is workers and groups of the population. Several questions arise here, such as to whether the evaluation should be carried out on an individual basis or rather involve selected workers, in which case could a "sampling" method be applied satisfactorily ? As to the public, should some legal provisions state that the actual dose must be the object of an "on the person" calculation or rather be worked out as a mean "per capita" value, or else should a determination within a given range be required ?

This rapid review is certainly not intended to be an exhaustive one, nor have I the competence and time to discuss the several points on technical grounds; however, it might prove useful to agree on their impact from the legal angle and on whether, how and to what extent they could be proposed as a common basis for many regulatory texts at least in the Community, as long as this does not tamper with practice and operational needs.

In this connection, let us now have a look at some national regulatory positions concerning in particular beta irradiation.

3. NATIONAL PROVISIONS

French legislation (9) is almost entirely consistent with the 1966 Directives but the case of unplanned special partial irradiation, where particular provisions are introduced for the case of limits not exceeding twice those established for planned irradiation. Some provisions implementing those on workers' radiation protection deal specifically with conditions for using individual dosemeters (10).

In Germany, the 1976 Directives were incorporated in a legal text (11) of the same year, providing maximum body doses (including skin and lens of the eye) for occupationally exposed persons; however, the hands skin limits (cat. B workers) were lowered to 20 rems, those of the rest of the skin to 10 rems and the lens limits to 5 rems. Special exposure doses are provided for accordingly. Rather detailed provisions may also be found concerning body (including partial and individual) "dose determination" and modalities for fulfilling such a task (instruments, safety measures, recordings and notifications, etc.).

British laws and regulations are being worked on very actively in order to make them consistent with the Directives, although to our knowledge no legal or regulatory provisions have so far been issued specifically intended to adopt them. The existing provisions (12), based upon international recommendations (ICRP), stipulate that the quarterly dose to hands and the lens shall not exceed 40 rems and 8 rems respectively; they also establish certain measures for individual protection by dosemeters and for relevant recordings. More detailed and specific standards regarding external irradiation are dealt with in the well known "codes of practice" (13) equally (although indirectly) binding due to the legal structure and largely adopted in Britain.

In Italy, provisions concerning the matter under examination are contained in the radiation protection law and its implementing decrees (14), by which the 1966 Directives were adopted, including the skin, hands and lens limits of both normal and special irradiation. Similar limits are established also for non occupationally exposed workers (in line with those of the said Directives, and for particular groups of the population). No specific provisions are found regarding ways for evaluating body doses, apart from those including the evaluation of individual dose among the tasks to be performed by the "qualified expert" and, as to external irradiations, those stating that such evaluation must be carried out "by one or more individual dosemeters".

FINAL REMARKS

This short survey can only provide a very general idea of some national situations,which on the other hand appear somehow lacking uniformity, detail and clarity. Truly, part of the requirements may be found unsuitable for inclusion in legal provisions, whether "per se" or depending on national systems; besides, compliance with the Directives may be satisfactorily assured at a national level as long as the aims thereby indicated are accomplished by means of equivalent provisions in terms of radiation protection. In any case, it must be said that the Euratom Commission has made a considerable effort, especially by the 1980 Directive, to bring principles and standards as much as possible in line with the actual uses of nuclear energy, of ionising radiations in general, and with the radiation protection needs.

In presenting our contribution, it was difficult to avoid dealing with aspects relating to dosimetry in general, mainly since on the whole national legislations do not seem to be specifically and extensively devoted to beta irradiation and dosimetry: which does not necessarily imoly that such particular item should not or could not be in fact developed, as the case may be, in those legislations. For example, the criteria provided by the Directives since 1976 for collective and individual monitoring of exposure and for the dose evaluation aspects resulting from the new classification of exposed workers, could be developed at a national level; moreover, a proper, clear and completely articulated set of maximum permissible doses for skin and lens irradiation of workers and the public could be set forth.

A closer analysis of the regulatory position of the countries considered as well as of the remaining ones, might reveal that some or most existing provisions, whether or not specifically adopting the Directives, are already substantially consistent with them. In some cases, we have noticed that national provisions seemed to go even beyond their strict scope, either by being more particular (e.g. maximum permissible doses for "particular groups of population" in Italy) or more stringent (e.g. dose limits for cat. B workers in Germany). But then, the essential objective is not only that of merely observing the Euratom Directives (which remains in any way a fundamental one) but also that of harmonising at the best the radiation protection laws and regulations within the Community. We trust that the discussion developing in the course of our meeting will help in achieving such harmonisation.

SOURCES

(1) Basic safety standards for health protection of the general public and workers against ionizing radiation, Brussels, 2 February 1959.

(2) Basic safety standards of 27 October 1966 for health protection of the general public and workers against ionizing radiation, amending the basic standards of 2.2.1959 and 5.3.1962.

(3) Council Directive of 1 June 1976 laying down the revised safety standards for the health protection of the general public and workers against the dangers of ionizing radiation.

(4) Council Directive of 15 July 1980 amending the Directives laying down the basic safety standards for the health protection of the general public and workers against the dangers of ionizing radiation.

(5) Recommendations of the International Commission on Radiological Protection (adopted 17.1.1977), ICRP Publication 26.

(6) Statement of the International Commission on Radiological Protection (Stockholm, May 1978).

(7) Statement and recommendations of the International Commission on Radiological Protection (Brighton, March 1980).

(8) General principles of monitoring for radiation protection of workers (adopted 24.5.1968), ICRP Publication 12.

(9) Decree of 15 March 1967 embodying public administrative regulations concerning protection of workers against the dangers of ionizing radiation.

(10) Decree of 19 April 1968 concerning conditions for the use of individual dosemeters.

(11) Radiological Protection Order of 13 October 1976.

(12) The Factories Act of 1961.
The Ionizing Radiations (unsealed and sealed sources) of 1968 and 1969.

(13) Code of Practice for the protection of persons exposed to ionizing radiations in research and teaching, 1968, HMSO, London.
Code of Practice for the protection of persons against ionizing radiations arising from medical and dental use, 1972, HMSO, London.

(14) Presidential Decree nO; 185 of 13 February 1964 on Safety of plants and health protection of workers and the public against the risk of ionizing radiation arising from the peaceful use of nuclear energy.
Decree of June 1968 of the Ministry of Work establishing maximum permissible doses and concentrations for health protection of workers against ionizing radiation.

2. QUANTITIES AND UNITS

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INTRODUCTION

Since its inception in 1925 the ICRU has been preparing recommendations on the definition of the physical quantities used to describe the effects of ionizing radiation.

In line with our improved knowledge in the various fields of radiology and the development of the relevant techniques, it draws up new concepts and tries to give them a universal and lasting character. This is not to say that, once formulated, these concepts must remain fixed; on the contrary, the ICRU constantly reviews these concepts to take account of the opinions it receives and technical developments.

To give a better understanding of the ICRU's role and to grasp the broad lines of its work, it is interesting to glance back over its publications and to give a broad historical outline of its work in connection with the various radiological quantities, before analysing the present situation.

EXPOSURE

In 1928 - three years after its foundation at the first international radiological congress - the ICRU established the first unit used in radiology; the roentgen. This was not yet a quantity but a unit corresponding to the quantity as measured by the free-air-chamber, an instrument which at the time was universally used in X-ray installations. An examination of the definition which was proposed at the time and which contained implicit reference to the means of measurement i.e. the free-air-chamber, shows that the Commission was responding to an immediate need in radiology without great concern for the future of this concept.

The ICRU had not yet developed in detail its philosophy as regards the physical quantities.

This initial definition had to be modified nine years later. In 1937 the ICRU proposed a new definition of the unit which no longer referred to the means of measurement and which did not correspond exactly to the

same quantity. Initially the number of roentgens corresponded to the number of esu charges produced in a given volume of air irrespective of the origin of the ionizing particles produced, whereas in the new definition it corresponded to the number of esu charges produced anywhere by the particles created in this volume of air.

This important development was inevitable because it was necessary to extend the field of utilization of this unit to photons with an energy greater than 300 keV (the case of radium). The free-air-chamber was unsuited for this purpose and was suoerceded by the cavity ionization chamber, the theory of which was developed by L.H. Gray.

Not until 1956 was a name finally given to the quantity corresponding to the roentgen; it was called the 'exoosure dose'. Later, in 1962, in Report 10a the ICRU deleted the term 'dose' and retained only the 'exposure' in order to avoid any confusion with the quantity 'absorbed dose'.

This first example is a good illustration of the role and the working methods of the ICRU in this field.

ABSORBED ENERGY

After the Second World War the enormous progress made in the various applications of nuclear energy revolutionized the field of application of radiation dosimetry which had been limited initially to photons with an energy lower than 3 MeV. It was necessary to draw on a new concept suitable for all types of radiation, whether directly or indirectly ionizing (alpha, beta, gamma, neutrons, etc.), irrespective of their energy.

Increasingly, the quantity 'exposure' no longer corresponded to a measure of absorbed energy in any environment and thus could not be used. Accordingly, in 1950 the ICRU recommended that the 'dose' be expressed in terms of energy absorbed per unit of mass of irradiated material and in 1953 it introduced a new quantity: the 'absorbed dose'. This rapidly replaced the 'reo' which had been proposed by Parker in 1948 but which had the drawback of depending on W_{air}.

Finally in 1962 (Report 10a) a new quantity was proposed: the 'kerma'. The quantity was intended to complete the concept of absorbed dose to describe the entire process of interaction of non-directly ionizing radiation with matter: transfer of energy to charged particles, deposition of this energy in matter via the Coulomb interactions. It replaced the 'first collision

dose' which had never been defined in an unequivocal manner. The ICRU then concluded its work on the definition of physical quantities connected with energy deposited or absorbed in biological target.

DOSE EQUIVALENT

It was also necessary to quantify the effect of radiation on the living cell. In 1956 the ICRU adopted the proposals made by Parker (1948) and defined the quantity 'RBE dose' which took into account the biological effectiveness of radiation for radio-biological studies. In 1959 it decided, together with the ICRP to replace this concept, in the field of radiation protection, by a new quantity: - the 'dose equivalent'. This no longer referred to the RBE but to the quality factor Q whose values were chosen on the basis of published studies and laid down by the ICRP. ICRU Report No 19 and its supplement, which were published in 1971 and 1973 respectively, both included the definition of this quantity and introduced minor changes in the drafting and specified the values of the quality factor.

THE OPERATIONAL QUANTITIES IN RADIATION PROTECTION

The activities of the ICRU which we have just surveyed related to the basic quantities used in radiation protection for defining radiation fields. In the last decade the ICRU has been developing a new theme: the definition of an 'operational' quantity with which to specify levels of ambient radiation using a concept which can provide an indication of the maximum absorbed dose received by an individual placed in this radiation field.

It has long been recognized that, in the case of indirectly ionizing radiation, the measurement of exposure or of the kerma in air may be used as a basis for assessing (approximately) the maximum absorbed dose by the body. However, these concepts are not applicable in the case of directly ionizing radiation. In this case an attempt has been made to use the 'absorbed dose in free air' or 'in a small tissue mass' placed at the point under consideration. However, depending on whether this mass is 'isolated or not' the mean absorbed dose determined in this manner may assume different values. Accordingly, this concept cannot be used to characterize a radiation field.

The ICRU proposed a solution in Report No 19 published in 1971. An approximate value for the maximum absorbed dose or the maximum absorbed dose

equivalent in the human body can be obtained by determining these two values within a 30-cm diameter sphere of tissue-equivalent material. The ICRU defined two new quantities: the 'absorbed dose index' and the 'absorbed dose equivalent index'. To meet with certain criticisms, the ICRU included these two quantities in its Report No 25 (1976), calling them 'unrestricted indices' and considering separately the case of penetrating radiation and less-penetrating radiation. The ICRU created the 'shallow dose equivalent index' and the 'deep dose equivalent index', which it called the 'restricted dose equivalent index'. These concepts were included in ICRU Publication No 33 which was published in April 1980.

One might think that, with these new concepts, the question of operational quantities has finally been solved. But unfortunately this is by no means the case, because a number of gloomy critics have observed that the quantities proposed are not addable and that they are not quite so easy to apply in practice!

In the past few years we have witnessed a 'revolt' by dosemeter technicians who have not always found it easy to accept these latest concepts.

The Commission of the European Communities has been the melting pot of this movement. A working party has set itself the task of presenting a summary of the present situation relating to the Operational Quantity. Most of the quantities are analysed here and examined in the light of the latest proposals such as the 'ceiling dose' or the 'dose absorbed at two reference depths'.

The ISO (International Standard Organization) precipitated this movement. In effect, ISO Working Party TC 85/SC 2/GT 2, which draws up the standards for the production of reference radiation, has expressed its misgivings about the present situation in an official letter which it addressed to the ICRU last year.

It should be said that the ICRU reacted very rapidly by creating a new working party under the direction of Professors Burlin and Harder and Dr Cooper with a view to specifying the practical application of these concepts. The ICRU wisely invited the main 'contestants' to take part in its work. Let us wish them the best of luck.

3. THE CALIBRATION HIERARCHY FOR BETA-DOSIMETRY

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INTRODUCTION

Before outlining current ideas in the United Kingdom on a calibration hierarchy for beta dosimetry, which forms a part of a general hierarchical scheme for protection-level calibrations, I thought it might be interesting to outline the steps taken at the National Physical Laboratory in the development of primary and secondary standards for beta dosimetry. My own involvement with this area of work dates only from 3 or 4 years ago when I took on responsibility for British Calibration Service activities in the radiological field. I shall say more about BCS later, but I must refer you to Mr Owen for any detailed enquiries about the earlier primary measurements on beta sources and secondary standards.

DEVELOPMENT OF PRIMARY AND SECONDARY STANDARDS

The first steps were taken in 1967 when the British Committee on Radiation Units and Measurements (BCRU) discussed the matter in response to a request from the Ministry of Labour. This request related to the requirements of the Factory Inspectorate for the checking of beta survey instruments. Clearly NPL could not calibrate all these instruments itself, and so secondary standards needed to be developed as the first stage in the dissemination of standards from the primary laboratory.

A programme of work on the absolute measurement of beta-ray dose-rates commenced at this time and papers were presented to BCRU discussing the relative advantages and disadvantages of beta sources or calibrated instruments as secondary standards. In terms of the accuracy attainable it was concluded that radioactive sources were to be preferred, provided that adequate radiochemical ourity and robustness of construction could be achieved. In 1968 further papers were presented to BCRU arguing for measurements on the beta ray secondary standard to be made in terms of absorbed dose rate to air rather than to tissue, as better accuracy (of the order of 4%) would be obtainable. Proposals for the soecific isotope sources and dose rates were also made and the system tested by performing calibrations on two commercial

monitors - an ionization chamber and a geiger instrument. Additional uncertainties, random and systematic, approaching ± 10%, attach to the instrument calibration factors depending on the instrument response, dose-rate, and beta-ray energy (1).

Discussions with the Radiochemical Centre (now TRC Ltd.) were opened in 1969 in order to ensure that the beta sources could be made available on a commercial scale. However, it was NPL's intention from the commencement of the project, that it would itself market the calibrated sources together with source mounting jig and filters. NPL publicised the system to prospective customers in the UK and abroad. The possibility that TRC might itself undertake the calibration of sources was discussed but this did not materialise (unlike the situation for Ra, Co, Cs photon sources which are certified by TRC in terms of exposure rate at a distance). Some effort was expended between NPL and TRC in the development of sources with satisfactory construction and reproducibility of activity levels in terms of the required absorbed dose levels at a distance. The active material is bonded in silver foil with inactive silver as backing and window, and with gold or nickel anti-corrosive coating of the window. In fact the secondary standard sets, that is calibrated sources in the mounting assembly with appropriate filters, became available in 1972. Eight major radiation centres in the UK purchased them and 7 foreign centres. Limited sets of sources (in terms of the dose rate range covered) were sold to two further foreign customers.

NPL designed secondary standards for beta-ray dosimetry in the protection area were thus available before the NPL-originated photon radiation secondary standards at either protection-level or therapy-level. Here, of course, I am emphasising 'NPL-originated'. Clearly before NPL-designed secondary standard instruments were available, alternative ionisation chambers were calibrated against primary standards. In fact, as far as could be seen, the national requirement for beta-ray secondary standards was fulfilled in 1972, since when further demand was canvassed in 1975, there was insufficient response for TRC to undertake the manufacture of a second large batch of sources. Now only replacements for the shorter half-life sources Tl and Pm are available.

In 1973 factors were published for the conversion of beta-ray dose rate, measured in air to dose-rate in tissue at specified depths (2). Additional uncertainty (about ± 4%, but up to ± 10% for promethium at one depth)

arises by the use of these factors. BCRU later published these factors applicable to an average depth of 5-10 mg cm^{-2} (3), and subsequently slightly modified the conversion factor for ^{147}Pm (4).

BRITISH CALIBRATION SERVICE

At about this time the deliberations of the BCRU on the general topic of protection level calibration dissemination schemes, had led to the conclusion that the British Calibration Service (BCS) should undertake the formal approval of secondary level laboratories in the radiological field. The BCS scheme with its system of assessment and supervision assures traceability of calibrations at secondary laboratories to the primary standard, and also a proven calibration capability, and the scheme has been well established in the areas of electrical, mechanical and thermal measurements for more than a decade. The BCS scheme has some appeal for the primary standards laboratory as it provides some assurance on the way secondary standards are employed, and of the manner in which primary standard measurements are transferred to field instruments. A good network of secondary laboratories also relieves the primary laboratory of much routine work. Thus in respect of beta-ray monitor calibrations, the requirement that NPL secondary standard systems should be employed was written into the criteria documents to be used for the assessment of laboratories and these documents were published in 1977 (5). At about the same time BCS was incorporated into NPL.

In fact there are still no BCS approved laboratories in the radiological area. BCS approval is sought on a voluntary basis and the status of such approval under the expected new Ionising Radiation Regulations is as yet unclear. These regulations will become law in a year or so's time, and in the associated Code of Practice, the section dealing with the testing of monitoring instruments is expected to call for the approval of laboratories by the Health and Safety Executive. Laboratories performing type-testing and acceptance (or pre-use) testing of instruments are recognised, and we hope to ensure that BCS will preserve a role in their statutory approval. Radiation Protection Advisors are to be responsible to employers for the routine testing of instruments. A second consultative document in which these matters will be clarified is expected to be published soon by our Health and Safety Commission.

CONCLUSION

I hope that this short description of the development of the beta-ray dosimetry dissemination scheme in the UK has been of some interest. As this example shows, the problems of developing primary and secondary measurement standards may be equalled by the difficulties of establishing a formalised dissemination scheme to ensure that the best use is made of these standards for the benefit of the community.

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4. STANDARDIZATION OF BETA REFERENCE RADIATION

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In radiation protection the evaluation of doses received by the entire organism of an individual or by certain organs which are radiosensitive to a greater or lesser extent is deduced from data recorded by detectors or dosemeters placed in the neighbourhood of the areas under consideration. In general the sensitivity of a detector to different types of radiation will depend on the laws of response as a function of the energy of the radiation. These laws need not necessarily be complicated but it is essential to know them.

Thus it is necessary :

a) to have access to reference installations in which the essential characteristics of the dosemeters can be determined and studied;

b) to specify the characteristics of these reference installations in order to ensure that these means of irradiation are universally applicable. It is essential that the results, curves and coefficients obtained in the installations of laboratory X can if required be confirmed by data obtained in laboratory Y, or at least that they are identical to those which would have been obtained in the installations of laboratory Y.

The standardization of reference radiation involves the drafting of standards by the specialists in the different laboratories concerned. These specialists meet together at international level in Working Group No 2 of Sub-Committee No 2 of ISO 85.

The delegations from the different countries participating in these meetings include the experts from the national primary standard laboratories (PTB, NPL, NBS, LMRI, etc.) as well as experts from laboratories or centres for calibrating radiation protection apparatus attached to different national institutes or bodies (CEGB, CEA, GSF, etc.).

In the case of X-rays and gamma radiation the work of ISO 85/SC2/WG2 led to the publication of international standard ISO 4037. Subsequently the British delegation drew up a similar text on beta radiation. The draft was first presented in London in 1976, then amended, and was discussed again in

Bologna in 1977 and referred to a sub-group of WG2. The text was modified in Brunswick in 1979 and it is this version (draft standard ISO/DP 6980) that was submitted to the members of WG2 for approval.

In attempting to summarize the text I should like to distinguish between three main sections :

1) - definition of sources of beta radiation;
2) - realization of reference beams - two series or two types of beams are proposed,
3) - physical quantity used to calibrate the radiation protection instruments - annexes.

1. DEFINITION OF BETA RADIATION SOURCES

Five radionuclides were selected, bearing in mind their nuclear characteristics and the availability of sealed sources on the market :

- ^{14}C,	E_{max} = 0.156 MeV,	T = 5730 years
- ^{147}Pm,	E_{max} = 0.225 MeV,	T = 2.6 years
- ^{204}Tl,	E_{max} = 0.763 MeV,	T = 3.8 years
- $^{90}Sr+^{90}Y$,	E_{max} = 0.546 MeV and 2.274 Mev,	T = 28.5 years
- $^{106}Ru+^{106}Rh$,	E_{max} = 0.039 MeV and 3.54 MeV,	T = 368 days

The lower limit of beta radiation considered in this standard is 66 keV representing the energy of beta particles capable of reaching the sensitive layer of the skin, which is situated - in accordance with ICRP convention (Publication 26) - at 7 $mg.cm^{-2}$ under the surface of the skin.

The practical use of sealed radioactive sources makes it obligatory to use dense, metallic materials in constructing the container of the radionuclide and the outlet window for the beta radiation.

Under these conditions a sealed source is characterized by a residual energy $E_{res} < E_{max}$ as a result of the absorption of beta radiation by the different materials which it passes through.

To ensure that the sealed source is in line with the standard the two criteria mentioned in the standard must be observed :

a) the ratio E_{res}/E_{max} must be determined experimentally using an absorption method and must be compared with the values of Table II in the standard;
b) photon contamination as a proportion of total dose rate must be measured and must be lower than 5%.

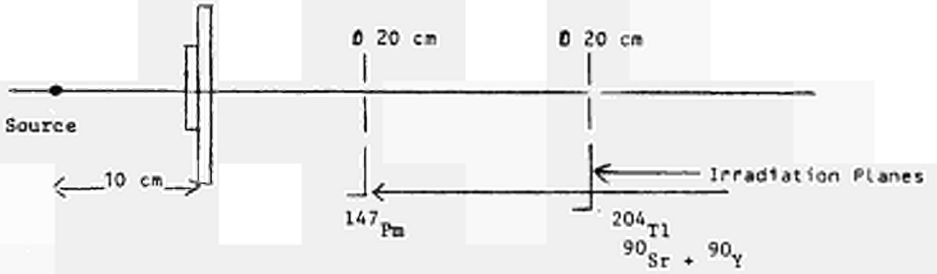
2. PREPARATION OF REFERENCE BEAMS

This standard proposes two series of reference beams. They relate to different objectives specified below.

2-1 Series No 1

This series of reference beams associates one filter or one set of beam flattening filters with each of the following three radionuclides respectively: ^{147}Pm, ^{204}Tl and (^{90}Sr + ^{90}Y); the filters are made of plastic (polyester or polyethylene terephtalate) which are placed at a fixed distance from the source and whose function is to ensure a uniform dose rate in the irradiation plane. This relates to one calibration distance only.

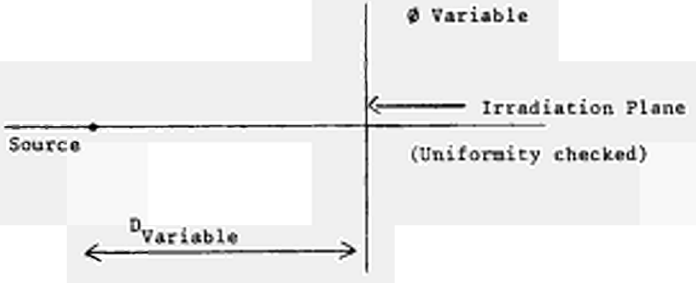
Beam - flattening filters



Under these conditions the variation of dose rate in the irradiation plane is obtained simply by substituting the source.

2-2 Series No 2

In order to round off the reference beams described above, both as regards the range of beta energies and the available dose rates, a second series is proposed which includes the same three radionuclides along with ^{14}C and (^{106}Ru + ^{106}Rh). The irradiation geometry is as follows :



In this case, depending on the distance between the source and the irradiation plane it is necessary to check whether the dose rate in the irradiation field used is uniform.

Depending on the nature and technological features of the source the dose rates estimated 'at the point of contact' range from 20 rads.h^{-1} (0.2 Gy.h^{-1}) for ^{14}C to several thousands rad.h^{-1} (a few tens Gy.h^{-1}) for the other radioelements.

3. CALIBRATION OF THE SOURCES

For beta radiation the quantity recommended for calibration of radiation protection devices is absorbed dose in tissue at a depth of 7 mg.cm^{-2}.

Accordingly, the beta radiation fields from the sealed sources used must be calibrated in this quantity. In general, dose rate in tissue at a given point in a radiation field from a beta source is measured with the aid of an extrapolation chamber (Bragg-Gray cavity). Construction of this device requires the use of materials similar to 'soft tissue' (ICRU 19); precise workmanship and a knowledge of the various correction factors which are difficult to determine theoretically or experimentally (effects of polarization, diffusion, geometric dimensions of the ion collection zone, relative attenuations of the materials used, etc.). Under these conditions, calibration of beta sources in terms of absorbed dose in soft tissue under 7 mg.cm^{-2} can only be carried out for distances between the source and the irradiation plane which minimize the influence of these correction factors. Accordingly, the conditions for reliable calibration of absorbed dose from a beta source must be strictly observed when such a source is being used or when its calibration is transferred to a source of different activity with the aid of appropriate equipment.

5. THE ROLE OF BETA DOSIMETRY IN RADIATION PROTECTION

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ABSTRACT

On the basis of ICRP 26 the practical aspects of beta dosimetry are discussed taking into account the beta radiation fields expected in radiation protection, the working conditions and the beta dose contributions in mixed beta/gamma fields. For the measurement of beta doses in extremity and personnel dosimetry a so-called 'shallow dose equivalent' in a tissue depth of 50 mg/cm^2 is proposed here which may serve as an operational quantity for the dose to the skin protected by clothing or gloves. This concept permits the use of relatively thick detector elements in mixed beta/gamma fields with a beta energy threshold of about 0.2 MeV.

1. INTRODUCTION

In radiation protection there is generally no need to wear a special beta dosemeter and to separate dose contributions from beta and gamma rays. Personnel dosemeters should, however, be optimized to detect also beta rays. An estimation of a beta-dose or (beta+gamma)-dose is then possible by taking into account the working conditions in beta fields in particular the distance to the beta source, the thickness of tissue which differs in the case of unprotected and protected skin as well as the corresponding lowest detectable beta energy.

Discussing the aspects of beta dosimetry in radiation protection the main questions of interest are :

- what kind of beta fields do we expect in radiation protection,
- what concept of personnel monitoring is required in beta/gamma fields,
- what type of beta dosemeter do we need,
- how important in practice is the estimation of beta dose compared to gamma monitoring ?

2. BETA FIELDS

In contrast to the common practice in gamma/neutron fields, beta fields are mainly described by the characteristic data of the beta source and not directly by field data at the point of interest. For the discussion in dosimetry the mean beta energy instead of the maximum energy is the more realistic term to describe beta sources.

What kind of beta fields do we expect in radiation protection? In Table 1 the characteristic data of typical beta fields are discussed in the sequence of importance with respect to the expected dose rate. We expect the highest dose rate for skin contaminations, here the beta dose may be calculated taking into account the activities per area, the irradiation time and the diameter of contamination. Small skin contaminations with a diameter below 1 cm may result in unexpected high dose rates (1).

Tab. 1: TYPE OF β-SOURCES IN RADIATION PROTECTION

Type of β-source	Source distance	β-dose estimation	Source characteristic	Expected dose rate in 70 µm tissue
SKIN CONTAMINATION	SKIN CONTACT, (hands, face)	CALCULATION from activity/area $\dot{H}$ >>for diam. <1 cm	$\dot{H} \sim \frac{A(E,d)}{\text{max (Diameter)}^k}$ k≥1	~8 $\frac{rad}{h}$ for $\frac{1\mu Ci}{cm^2}$
'POINT'SOURCE	0.1 - 10 cm from finger/hand	EXTREMITY DOSIM. in 27mg/cm² tissue depth	$\dot{H} \sim \frac{A(E,x/R)}{(\text{Distance})^2}$	~1 $\frac{rad}{h}$ for 1µCi in 1 cm
AREA SOURCE (room contamin.)	40-200 cm from floor	PERSONNEL DOSIM. Attenuation in Air: 1m ≙ 129mg/cm² Clothes: > 50mg/cm²	$\dot{H}$ variable function of $E_{max.}$, distance	~4to25 $\frac{mrad}{h}$ for $1\frac{\mu Ci}{m^2}$ in 75cm
VOLUME SOURCE (radioact. cloud, air contamination)	Distances ≤R_{max}(β) Attenuation in air/clothes	CALCULATION from activity/m³ for environmental monitoring	VOLUME SOURCE given by range R of β-particles	0.06-0.6 $\frac{mrad}{h}$ for $\frac{1\mu Ci}{m^3}$

In personnel monitoring point and area sources are of main practical interest. Here one of the characteristic field data is the source distance d which is relatively low (0.1 cm to 10 cm) for the unprotected skin of the fingers, hands or other extremities but larger (40 cm to 200 cm) for the chest of the body. Under normal working conditions the skin of the whole body is protected by clothing. Both, the clothing and the absorption in air for source distances of about 1 m reduce the effective dose rate in the beta field as a function of the relative beta particle range R.

Volume sources occur in environmental monitoring or in the case of air contaminations if the distance to the source is smaller than 15 m. The beta dose of a radioactive cloud or a room contamination may be calculated taking into account the specific activity of the air and the volume of the source.

The dose rate which we expect from a beta source of $1\,\mu$Ci is a maximum for a skin contamination of $1\,\mu Ci/cm^2$ and was found to be always about one or two orders of magnitude lower for a point source of $1\,\mu$Ci at 1 cm or for an area contamination of $1\,\mu Ci/cm^2$ at 75 cm or for a volume source of $1\,\mu Ci/cm^3$ at larger distances.

3. CONCEPT OF PERSONNEL MONITORING

The role of beta dosimetry in personnel monitoring is based on the radiation effects expected in different tissue depths and on the dose equivalent limits which are recommended by ICRP for the unprotected skin, the extremities, the lens of eye, and the uniform whole body irradiation (Fig. 1) (2,3).

The annual dose equivalent limits to be applied are :

- for the skin which is unprotected by clothing especially at the extremities 500 mSv (50 rem) in a recommended tissue depth of 7 mg/cm,
- for the lens of the eye 150 mSv (15 rem) in a recommended tissue depth of 300 mg/cm^2,
- for the uniform whole body irradiation and the effective dose equivalent 50 mSv (5rem). Here the tissue depth is not recommended and a depth of about 1000 mg/cm^2 may be adopted.

For the situations in routine practice Table 2 presents the tissue depth for the dose quantities of interest, the effective tissue depth of interest and the minimum beta particle energy which is detectable after transmission through clothing or a corresponding layer of air. The proposal discussed here makes use of the fact that the skin of the whole body - excluding unprotected extremities and the head - is protected by clothing. Taking into account the minimum layer of the clothing, 50 mg/cm^2 is proposed here to be the tissue depth for the skin protected by clothing and gloves, respectively.

In consequence, the current practice in personnel monitoring makes use of two types of dosemeter :

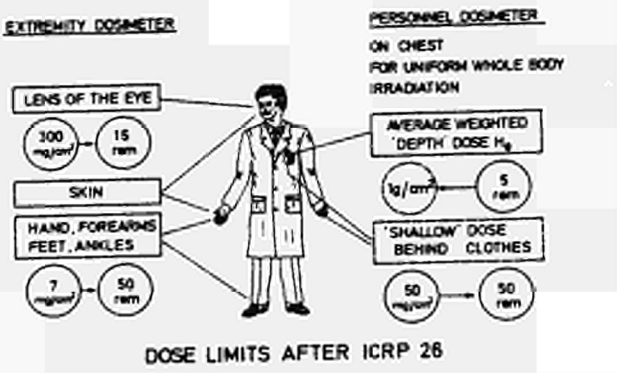

Fig. 1 : Dose equivalent limits after ICRP 26 (2,3) and the corresponding tissue depths required for personnel monitoring and extremity dosimetry. The concept proposed here is presented in dashed lines.

Tab. 2: β-RADIATION EFFECTS IN TISSUE AND TYPE OF PERSONAL MONITORING

RADIATION EFFECT FROM β-RAYS			β-DETECTION ABOVE	PERSONNEL MONITORING		
TISSUE OF INTEREST	LOCATION	TISSUE DEPTH (mg/cm²)	E_β(min) [1] (MeV)	DETECTOR SHIELDED *)BY (mg/cm²)	DOSIMETER TYPE	WORKING CONDITIONS
SKIN unshielded:	extremities, face	7	0.07	7	EXTREMITY DOS.	handling with β/γ sources/ contaminations
	hands inside	40	0.2			
shielded:	source distance ≥ 5 m; +skin shielded by clothes	130	0.4	50 [3]	PERSONNEL DOS. 'SHALLOW DOSE'	working in β/γ fields without handling with sources/contaminations
		> 50	0.6			
LENS OF EYE	eye unprotected	300	0.8			
WHOLE BODY	tissue depth representing the measure of H_g	1,000	2.0	1,000	'DEPTH DOSE'	uniform whole body irradiation

*) tissue depth in column 3 was used as the maximum range of β-particles with E_β(min)

*) for tissue equivalent materials representing the tissue depth of interest for a 'thin' detector

*) recommended tissue depth for the skin protected by clothing/glove of at least 50 mg/cm² thickness

- the extremity dosemeter applied for handling with beta/gamma sources of contaminations in non-uniform radiation fields where the source distance is in the order of 0.1 to 10 cm, to measure the skin dose at the fingers or at the face in the case of unprotected skin.
- the personnel dosemeter for whole body irradiations to be worn at the chest of the body. After the Euratom recommendations (4) the so-called discriminating basic dosemeter consists of at least two detector elements, one detector element to measure the dose equivalent in a shallow depth for the monitoring of the skin protected by clothing, the other detector element to measure the depth dose in terms of effective dose equivalent or a similar operational dose quantity. The readings of both detector elements are assumed to be representative for a uniform whole body irradiation and can be interpreted in practice only as the dose at the dosemeter's location.

The Euratom recommendation allows the interpretation that a personal beta dosemeter is a discriminating basic dosemeter with a thin detector element shielded by 50 mg/cm^2 which results in a beta energy threshold at the source of 0.6 MeV and a detector element shielded by a layer of about 300 to 1000 mg/cm^2 for the detection of high energy beta rays.

4. BETA DOSEMETERS

In the last years many attempts were made to develop thin detector elements and thus to improve the response of finger dosemeters. Fig. 2 discusses the properties of three types of dosemeters and the transmission factor for the beta dose equivalent which is expected behind a tissue layer of 7 and 50 mg/cm^2. From the TLD elements offered commercially and applied in large scale use at least a 0.2 mm thick LiF detector seems to fulfil the minimum requirement for a finger dosemeter up to now. Most of the monitoring services in the European Community, on the other hand, apply up to now for the purpose of extremity dosimetry TLD detector elements up to 0.9 mm thickness which in addition are shielded by a plastic window with a thickness of 7 mg/cm^2 up to about 150 mg/cm^2. In the case of a 0.9 mm thick detector and a window of 100 mg/cm^2 the beta response of such a finger dosemeter may be compared directly with that of a pen dosemeter. From this point of view the current practice in extremity dosimetry is insufficient and should be improved in many laboratories.

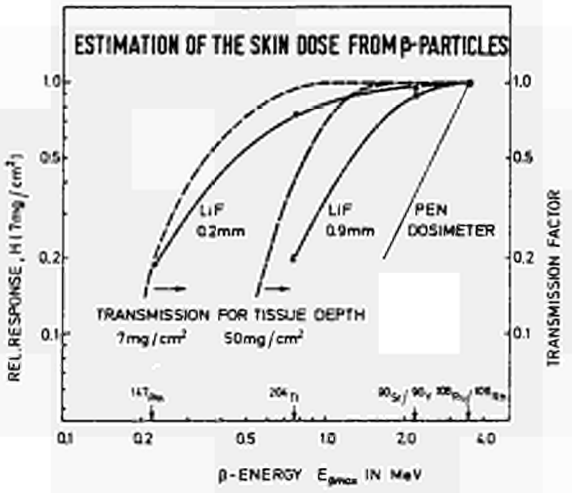

Fig. 2 : Beta-energy dependence of the response of LiF and pen dosemeters and of the transmission factor for a tissue equivalent layer of 7 mg/cm^2 and 50 mg/cm^2.

In the radiation protection practice, however, one of the main requirements for handling with beta sources or contaminations is the protection of the skin at the fingers and hands by means of gloves to avoid skin contaminations or unnecessarily high skin doses from low energy beta particles. When finger rings or finger tip dosemeters are worn outside the glove it is possible to use thicker detector elements and/or protection foils for the detector element which may exceed 50 mg/cm^2 corresponding to the thickness of the gloves in use. Consequently the concept of measuring the 'shallow dose equivalent' in 50 mg/cm^2 seems to be practicable also in extremity dosimetry.

For whole body irradiations in many situations a detector element of 0.9 mm thickness meets the minimum requirement for a personnel dosemeter which measures the 'shallow dose equivalent', i.e. the dose equivalent to the skin protected by clothing. With respect to the transmission factor for a tissue depth of 50 mg/cm^2, the 0.9 mm LiF detector underestimates the dose of interest by a factor of 2 approx. independent of beta energy especially above the required threshold of 0.6 MeV. For a uniform whole body irradiation the 0.9 mm thick detector overestimates, on the other hand, the dose

equivalent to the lens of the eye by a factor of 2. An 0.2 mm thick detector element in the discriminating basic dosemeter will, however, significantly overestimate the dose to the protected skin.

5. CONTRIBUTION OF BETA DOSE IN MIXED BETA/GAMMA FIELDS

In radiation protection, do we need at all the monitoring of skin by using a personnel dosemeter at the chest of the body?

After ICRP 26 the monitoring of skin is required only for working condition 1 if the annual dose to the unprotected skin exceeds 150 mSv (15 rem) which is three-tenth of the limit. In most of the real situations in beta/gamma radiation fields the beta dose reading of the personnel dosemeter was found to be one order of magnitude lower compared to this value. This is demonstrated in Fig. 3 which shows the annual dose results found for the occupational workers in the decontamination facilities of the Karlsruhe Nuclear Research Center. The highest beta dose results are only 1.4% of the annual dose equivalent limit. The corresponding ratios of the beta/gamma dose for a monitoring period of one month was found to be lower than a factor of 4 (Fig. 4).

For most of the persons monitored in beta/gamma fields there is obviously no need for a discriminating basic dosemeter with a special beta detector element. With respect to an application of extremity dosemeters quite different aspects of irradiation conditions have to be considered.

The situation found in routine monitoring may change significantly after incidents resulting in high level contaminations. Results of personnel monitoring in beta/gamma fields have shown in the past that the ratio beta dose/gamma dose may reach a factor of 20 in high level contaminated areas depending on the thickness of and the distance to the source. Here we obviously need a basic beta dosemeter as well as extremity dosemeters.

In high level contaminated areas with fission products the beta dose may be found to be significantly high at the fingers or at the head (Fig. 5). The results at extremities may be higher up to a factor of 10 compared to the dose reading at the chest. This situation calls for a broader application of extremity dosemeters for workers in high level areas.

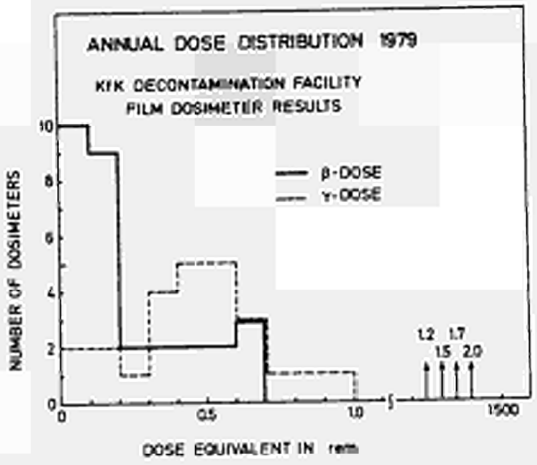

Fig. 3 : Frequency distribution of the annual beta-dose and gamma-dose found for the workers in the Decontamination Facility of the Karlsruhe Nuclear Research Center.

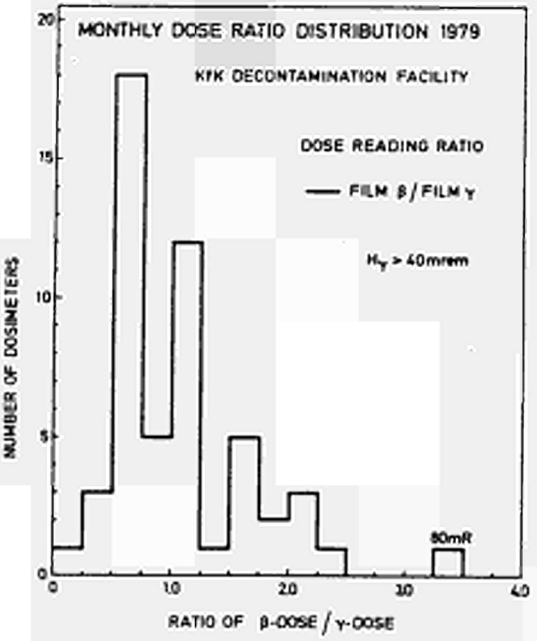

Fig. 4 : Frequency distribution of the monthly ratio beta-dose/gamma-dose.

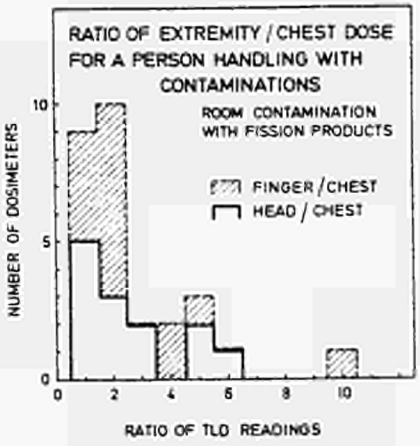

Fig. 5 : Frequency distribution of the dose ratio finger/chest head/chest during work in contaminated rooms found with TLD doseneters.

6. CONCLUSION

The importance of beta dosimetry is without any doubt the monitoring of extremities such as fingers in case of handling with beta/gamma nuclides or contaminations. There is, however, the general question how to correct for the systematic uncertainties of dose reading in the extreme non-uniform radiation field where the location of the dosemeter may differ significantly from the dose maximum expected.

The use of gloves and the suitable application of finger dosemeters to be worn outside the gloves makes clear that there is no necessity in practice to develop the ideal beta detector of 7 mg/cm^2 thickness and to measure skin doses from low energy beta sources below 0.2 MeV.

Because of the relative small range of beta particles and the large distance to the beta source the personnel dosemeter worn on the chest provides only information of secondary order compared to finger dosemeters or to the gamma dose contribution. According to the low beta dose contribution expected in mixed beta/gamma fields a discriminating basic dosemeter is useful only for a small group of persons working in high level contaminated fields.

The introduction of an operational quantity for the skin dose, for instance the 'shallow dose equivalent' in a tissue depth of 50 mg/cm^2 for

the protected skin seems to be a practical concept in beta dosimetry.

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6. PRACTICAL BETA DOSIMETRY IN RADIATION PROTECTION

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ABSTRACT

Various aspects of practical beta dosimetry in radiation protection are discussed. The capabilities of currently used personnel dosemeters to satisfy internationally recommended standards for beta-dose measurements are reviewed, and some proposals for improvements are reported. A description is given of a commercially available secondary standard beta-calibration unit which has proved very useful for practical radiation protection dosimetry. Furthermore, the present status of survey instruments for beta dose-rate measurements is briefly commented upon.

INTRODUCTION

A routine dosimetry system universally applicable for practical beta-radiation protection must be designed for measurements of exposures from a large variety of radioisotopes and therefore must be capable of covering a broad range of beta-ray energies.

The assessment of radiation doses to the unprotected skin from low-energy beta particles of unknown energy requires a thin effective detector thickness which normally involves an unattractively low sensitivity, and furthermore presents difficulties for the construction as well as handling of the detector. Consequently most currently used personnel badges for whole body beta/gamma monitoring cannot effectively measure exposures from low energy beta radiation.

In recent years various types of practical TL dosemeters with a skin dose equivalent response over a wide range of beta energies have been developed. Such dosemeters are now used mainly for extremity dosimetry. They can be introduced as a supplementary detector into the whole body personnel beta/gamma badge as well, which implies a more sophisticated badge design and dose evaluation procedure. However, since the protective clothing and working distances used in practical radiation work may prevent any significant dose from low energy beta particles to the skin, except for the extremities, it may be more important to concentrate on a proper extre-

mity dosimetry than to design a personnel badge with an ideal skin dose equivalent response covering the whole practical beta energy range down to approximately 50 keV.

At present there is not much data available to give information about the dose contribution from low energy beta emitters compared with the total dose. Further investigations in this area would be valuable for designing an optimal dosimetry system.

Dose Equivalent Control Limits

In radiation protection against external beta radiation, control must be considered for exposures to the skin, the lens of the eye, and the body tissue. The dose equivalent control limits recommended by ICRP (1,2) together with corresponding reference tissue depths for the dose measurements are listed in Table 1.

TABLE 1 : Dose equivalent limits (ICRP) and reference tissue depths for controlling exposures to skin, eye lens and body tissue.

Part of body	Symbol	Depth $mg.cm^{-2}$	Limits mSv
Skin	Hs	5-10	500
Eye	He	300	150
Body	Hd	1000	50

In routine personnel monitoring usually only the skin dose referring to a tissue depth of ideally 5-10 $mg.cm^{-2}$ and the body dose with a reference depth between 400 and 1000 $mg.cm^{-2}$ are estimated in accordance with the CEC recommendation for the use of a basic non-discriminating dosemeter (3,4). This presumes that the dose to the eye lens is automatically controlled if the dose limits for H_d and H_s are not exceeded which holds only when $H_s/H_e \geq 3{,}3$ or $H_e/H_d \leq 3$ (cf. Table 1).

From the depth dose curve for ^{90}Y beta radiation (E_{max} = 2.27 MeV) for absorption in Perspex shown in Fig. 1 (5), it can be seen that the energy of ^{90}Y beta particles is just at the upper limit of the range of energies for which the condition $H_s/H_e \geq 3.3$ is satisfied. So, for all

energies below approximately 2.3 MeV the eye dose will automatically be controlled when the skin dose is kept within the recommended limit. For higher energies the eye dose limit can be controlled by the body dose measurement if an appropriate reference depth is chosen for the measurement. As shown in Fig. 2 (5) the requirement $H_e/H_d \leq 3$ will be met for all beta energies above that of ^{90}Y if a reference depth of approximately 500 mg.cm^{-2} is used, whereas a depth of 650 mg.cm^{-2} is appropriate only for energies above that of ^{106}Rh (E_{max} = 3.6 MeV). The choice of a depth below 1000 mg.cm^{-2} results in an overestimate of the body dose and in practical radiation protection a compromise must be made between the allowed degree of overestimate of depth dose equivalent and the capability desired for control of the eye dose limit.

In a recent draft IEC document (6), considering this matter in relation to the design of portable radiation monitoring instruments, a depth of 800 mg.cm^{-2} has been proposed for measuring the depth dose equivalent. In accordance herewith the value in the CEC recommendations may be changed from 400-1000 mg.cm^{-2} to 700-1000 mg.cm^{-2} when revised in the future (7). This implies that for certain but undoubtedly very few radiation compositions, including high energy beta rays, a third measurement may be necessary for controlling the eye dose limit.

CALIBRATION PROCEDURE

The calibration of beta radiation protection monitoring instruments as well as personnel dosemeters requires some kind of standardized calibration procedure which may include either a standardized dose-rate instrument, e.g. an extrapolation chamber, a set of standardized beta sources, or both of them. The interpretation of data obtained from measurements with an extrapolation chamber is complicated and needs skilful and time-consuming evaluation work; so for radiation protection purposes the use of a set of calibrated sources seems preferable.

A complete secondary standard beta calibration unit including four beta sources, a jig for supporting the sources during irradiation, three beam-flattening filters and an electronic unit for controlling the exposure time has recently been developed by Physikalisch-Technische Bundesanstalt (PTB) in Braunschweig (8) in co-operation with the German firm Buchler. A photograph of the complete unit is shown in Fig. 3 and the technical data are given in Table 2. At Risó a calibration set-up of this type is now

applied for the calibration of dosemeters and ratemeters as well as providing a reference for other beta calibration facilities. In order to facilitate a vertical beam orientation which is useful for many calibration purposes, a holder with a distance indicator has been constructed for the unit as shown in Fig. 4. PTB provided each source with a calibration certificate which includes measured values for absorbed dose rates in air and in tissue referred to the surface of a semi-infinite phantom of tissue equivalent material, and furthermore depth dose data for absoprtion in tissue equivalent material (see Table 2 and Fig. 5). The facility is similar to a unit developed earlier at National Physical Laboratory (NPL) in UK (9) and it fulfills the specifications set up by ISO for the set of beta calibration sources categorised as series 1 sources (10).

Personnel Dosimetry

The assessment of skin dose, defined as the average dose in tissue at a depth between 5 and 10 $mg.cm^{-2}$ below the surface of the body (1,3,6,11) requires a dosemeter with ideally an effective dosemeter thickness of 5 $mg.cm^{-2}$ and a protective layer of the same thickness for covering a broad range of beta energies. The present state of whole body beta monitoring is far from satisfactory in meeting these requirements (12). For film badges the paper wrapper which has to be used to protect the film from exposure to light is too thick to allow beta particles of very low energy to penetrate, and for dosimetry based on thermoluminescence or radiophotoluminescence the thicknesses of the active dosemeters normally prevents a highly efficient detection of low energy beta particles.

A situation like this is not necessarily a serious problem for radiation protection if a proper extremity dosimetry is practiced, as the doses from beta rays are given mainly to hands and fingers. Fig. 6 shows beta energy response curves for the finger dosemeter used routinely at Risö given for the two reference depths below the surface of the body : 5-10 $mg.cm^{-2}$, and 40 $mg.cm^{-2}$. The latter is the average thickness of the insensitive protective skin layer of the palm surface of the hands including the front of the fingers and fingertips (13). As illustrated in Fig. 6, a dosemeter of this type with a considerable thickness can estimate doses to the front of the fingers with a reasonable accuracy whereas serious underestimations occur for skin doses from low energy beta exposures to other parts of the body.

Table 2

Technical data on the PTB/Buchler secondary standard beta calibration unit in use at Risø.

Source type	1	2	3	4
Radionuclide	^{147}Pm	^{204}Tl	$^{90}Sr+^{90}Y$	$^{90}Sr+^{90}Y$
Nominal activity mCi	14	0.5	2	50
Thickness of the inactive silver foil "window" mg cm^{-2}	5±1	20±3	50±5	50±5
Protection against corrosion	0.5 mg cm^{-2} electroplating of nickel	1-µm thin gold flashing	1-µm thin gold flashing	0.1 mm stainless steel
Mean beta particle energy MeV	0.06	0.24	0.8	0.8
Filter material	Hostaphan	Hostaphan	Hostaphan	-
Filter dimensions	1 disc of 5 cm radius and 100 µm thickness with a hole of 0.975 cm radius in centre	1 disc of 4 cm radius and 50 µm thickness, plus 1 disc of 2.75 cm radius and 190 µm thickness	3 concentric discs each 190 µm thick and of radius 2,3 and 5 cm	
Absorbed dose rate in air at phantom surface at date of calibr. ($\mu Gy \cdot s^{-1}$)	0.4349 at 20 cm distance	0.2563 at 30 cm distance	2.312 at 30 cm distance	521.1-25.37 at 10-50 cm distance

Developments in TLD techniques in recent years have improved the possibilities for assessing skin doses to the extremities and to the total body.

A badge containing e.g. a 0.2 mm thick Teflon disc can measure the skin dose to the body from beta exposures to within 30% for maximum beta energies ranging from approximately 0.7 to 3.5 MeV (14). New TL-elements with an active dosemeter thickness of 10-20 $mg.cm^{-2}$ (15,10) provides further improvements for the assessment of doses from low energy beta rays.

The beta energy response of TLD-badges containing relatively thick dosemeters such as the commonly used 0.9 mm thick chips from the Harshaw Co. can be improved by introducing an additional skin dose equivalent detector into the badge e.g. a graphite-mixed, low transparent TL-dosemeter, see Fig. 7 (17).

Alternatively one may create a new glow peak in a thin surface layer of existing LiF TLD's using a boron-diffusion process. Thereby a skin dosemeter can be produced which at the same time can ensure the detection of infrequent skin doses from low energy beta radiation with a reasonable detection threshold and still maintain its high accuracy for measuring skin doses from more penetrating radiation (18). From the ratio of the additionally produced glow peak to the original one, information can be obtained about the dose contribution from low energy beta rays as can be seen from Fig. 8.

The use of two chips with two different filters for skin dose measurements, e.g. 10 and 50 $mg.cm^{-2}$, as proposed for the personnel badge at Idaho National Engineering Laboratory, USA (19), may be the approach that is most obvious for obtaining a practical skin dosemeter for those systems using dosemeters of relatively large thickness.

For extremity dosimetry several types of TL dosemeters are available. A LiF-Teflon disc, 8.9 $mg.cm^{-2}$ thick and covered with a 4 $mg.cm^{-2}$ protective layer shows a skin dose equivalent response for all exposures from beta rays with E_{max} above 100 keV (20). Ultra-thin LiF-Teflon discs (21) or $CaSO_4$:Dy-Teflon discs (22) thermally bonded to a thick Teflon base have improved the practical possibilities of handling thin Teflon discs. Other organic materials resistant to relatively high temperatures, e.g. polyimide and polyethersulfone, have been successfully applied as support for the TL-phosphor in developments of thin dosemeters (23,24). The change of transpa-

rency of TL dosemeters by the addition of graphite in the production process has proved to be a useful method of obtaining practical skin dose equivalent sintered or Teflon-based dosemeters (25,26). By fixing LiF powder to an Al disc using a cold-pressing method, practical dosemeters with low beta energy dependence can be produced (27). A great diversity of construction is seen for extremity dosemeters in practical use (5). A new design fitted for semi-automatic handling has recently been developed (28).

Dose Rate Measurements

Protection against beta radiation usually involves frequent control measurements of surface and air contamination levels as well as direct measurements of the dose rates arising from beta rays present in the working area. Hazards due to loose contamination on surfaces and its dispersal into the air are in general satisfactorily controlled by existing surface- and air-contamination monitoring systems. Also, reliable dose-rate monitoring of beta ray fields is feasible by using commercially available beta/gamma radiation survey instruments provided that the distance between the source and monitor is not too short and an energy-dependent conversion factor for the instrument reading is used. For examole, for the widely used Eberline R02 beta/gamma survey instrument that uses a 1.1 cm deep ionization chamber with an entrance end-window thickness of 7 $mg.cm^{-2}$, the following response data in terms of instrument reading (Rh^{-1})/dose rate ($rad.h^{-1}$) in tissue (at 7 $mg.cm^{-2}$) were obtained : 0.68 for $^{90}Sr/^{90}Y$, 0.39 for ^{204}Tl and 0.29 for ^{147}Pm (29). These data are only valid for distances from the source greater than approximately 12 cm. For shorter distances, a strong decrease of the response occurs. At present, there are only very few monitor possibilities available for estimating dose rates close to or in contact with the beta sources. Extrapolation chambers can be used, which however are not convenient for routine monitoring purposes. A 4-mm deep ionization chamber is usable for beta dose rate measurements at distances down to approximately 8 cm (29). Recent developments in scintillation detectors (30,31) and surface barrier detectors (32) look promising for obtaining energy-independent survey instruments for dose rate measurements at short distances from the source. An alternative method is to use solid state integrating dosemeters e.g. thin TL dosemeters (33).

CONCLUSION

The beta response of most currently used personnel dosemeters worn at the trunk of the body for beta/gamma monitoring, is very dependent on energy, and exposures from low energy beta radiation to the unprotected skin can normally not be efficiently monitored by these. Some improvements may be obtained quite simply, e.g. by reducing the thickness of the cover over the dosemeter element and by applying recent TLD developments. However, if the hazards from exposures to low energy beta rays are mainly due to doses to the hands and fingers it may be even more important to concentrate on the construction and the application of appropriate extremity dosemeters.

The choice of a depth for body dose estimations lying at the upper range of the interval, 400-1000 $mg.cm^{-2}$, e.g. 800 $mg.cm^{-2}$, means that for certain high energy beta exposures it may be necessary to take into account the control of the dose to the lens of the eye in addition to assessing skin and depth dose.

A set of calibrated beta sources like the secondary standard beta calibration unit recently developed at PTB has proved very useful for beta radiation protection dosimetry and a widespread distribution of units of this kind may contribute considerably to improvements of current beta radiation protection dosimetry.

There is a need for commercially available survey instruments for dose rate estimates close to or at the surface of beta radiation sources. New developments of scintillation and surface barrier detectors may be useful for this purpose.

ACKNOWLEDGEMENT

The author thanks Mr O. RASMUSSEN for constructing the support for the entire secondary standard beta calibration unit.

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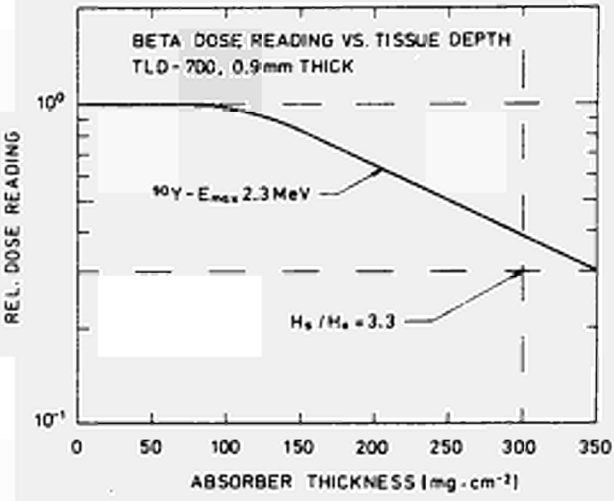

Fig. 1. Depth-dose curve for absorption of ^{90}Y beta rays in tissue measured with Harshaw, 0.9 mm thick, LiF TLD-700 chips (ref. 5).

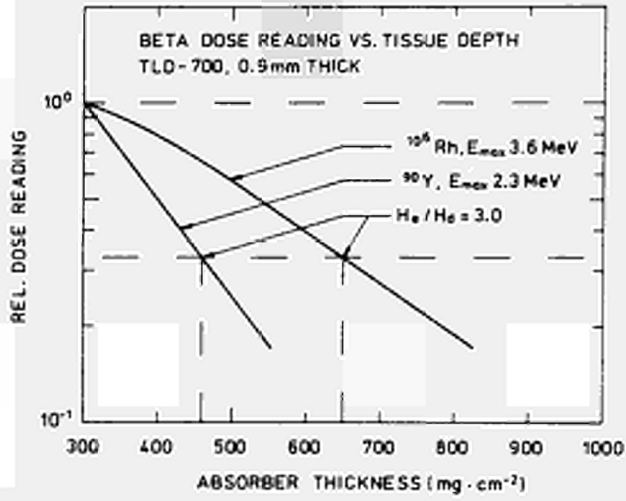

Fig. 2. Depth-dose curves for absorption of ^{90}Y and ^{106}Rh beta rays in tissue measured with Harshaw, 0.9 mm thick LiF, TLD-700 chips (ref. 5).

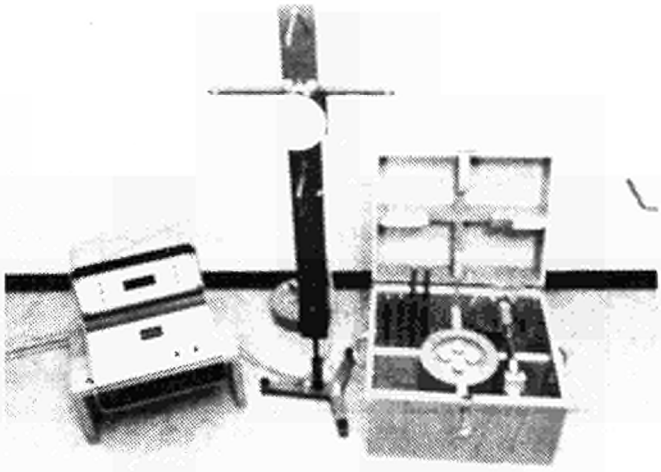

Fig. 3. Photograph of the PTB/Buchler secondary standard beta calibration unit.

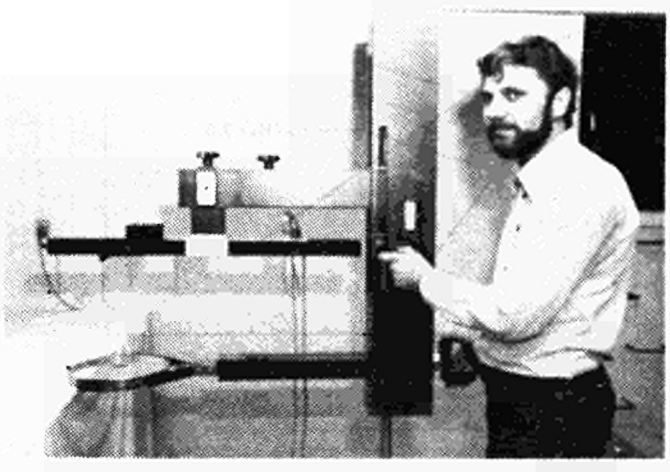

Fig.4. Photograph of the secondary standard beta calibration unit with a holder for vertical beam orientation.

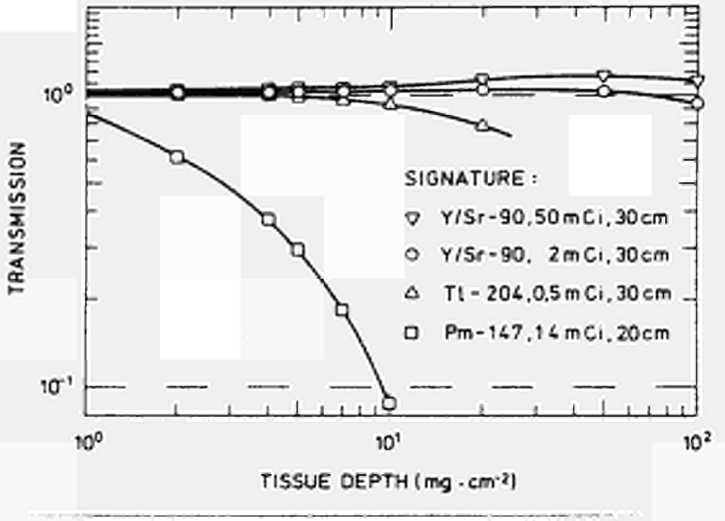

Fig. 5. Beta transmission curves of the four beta sources specified in table 2. The points indicate figures given in the calibration certificate.

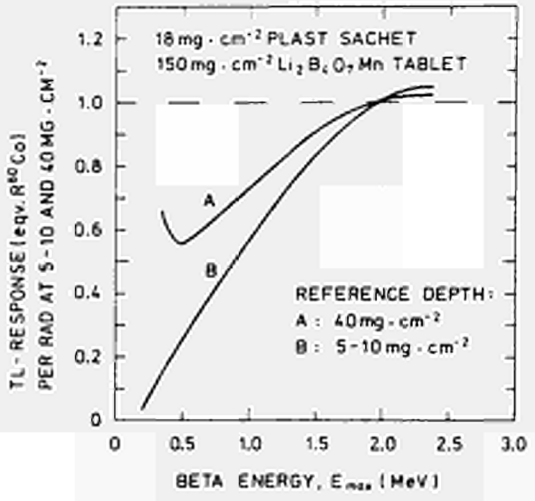

Fig. 6. Beta energy response curves of the finger dosemeter used at Risø containing three $Li_2B_4O_7$:Mn sintered tablets in a polyethylene sachet. The responses are normalized to the skin absorbed dose at the two depths, 5-10 mg·cm^{-2} and 40 mg·cm^{-2} respectively.

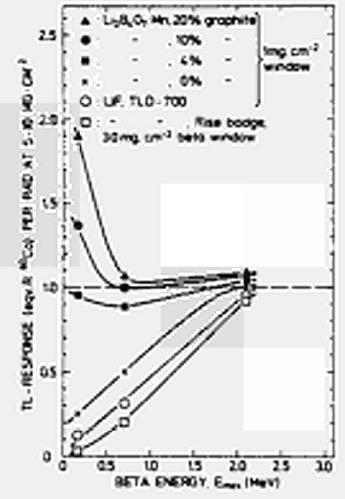

Fig. 7. Beta energy response curves of 2.3×2.3×0.9 mm^3 LiF TLD-700 chips and 5 $mm^{\varnothing}$×0.8 mm $Li_2B_4O_7$:Mn sintered tablets containing various amounts of graphite.

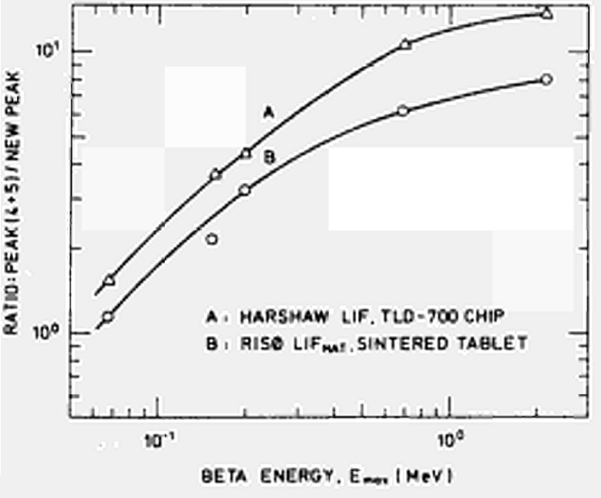

Fig. 8. The glow peak ratio: peak (4+5)/new glow peak as a function of beta energy shown for two types of boron-diffused LiF TLD's.

7. STATE OF THE ART IN BETA DOSIMETRY

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INTRODUCTION

To date, beta dosimetry plays a minor role in radiation protection. This is not because beta radiation represents a negligible hazard but results from the difficult measuring technique as compared with photon dosimetry. The reason is the strong electron absorption and scattering in the detector.

The number of occupational exposures to beta radiation is indeed considerable. From a statistical analysis of large scale personnel monitoring data (GSF Film Dosimetry Service with about 55.000 films a month) beta radiation was found in 0.5% of the utilities monitored. One third of these films showed also contamination. Accordingly beta radiation hazard is found mainly in the fuel cycle, at hospitals and in research.

Radiation fields in such environment may show up significant spatial gradients in dose rate depending on the location. For certain working conditions the beta-to-gamma dose rate ratio was found to vary widely, e.g. up to 30. The finding requests for proper beta measuring equipment in radiation protection in order to effectively control the occupational exposure for the benefit and safety of radiation workers. This is of concern particularly for organs near the body surface such as skin, eyes and gonads but also bone marrow.

The need of beta dosimetry was actualized by the Three Mile Island (TMI) emergency incident. Respective experiences were reported at the 25th Annual Meeting of the Health Physics Society at Seattle/USA 1980. It was stated that relatively little is known about beta dosimetry in radiation protection. Emphasis was given to national beta dosimetry standards (1).

BETA DOSE ASSESSMENT

From photon dosimetry we expect the dose rate in a certain source distance to be measured by means of a standard or field dosimetry equipment or to be calculated from the source activity.

For beta radiation, dose calculations are rather complex and depend strongly on the spectrum, source distance, field and absorber homogeneity etc.. Inspite of numerous publications in this field, the problem of beta dose rate calculation is by far not yet solved. Besides data achieved from the application of SPENCER's theory (2,3) to a homogeneous and infinite medium there are empirical dose functions for point sources which are less accurate but simple to use (4-6).

Accurate beta dose assessment is possible by measurement. However, beta dosimetry requires rather sophisticated equipment. There is a real need for small, rugged and energy independent beta dosemeters applicable in radiation protection.

CALIBRATION AND MEASURING QUANTITIES

The calibration quantity for protection level instruments in general is still under discussion, at present (dose equivalent in specified depths, dose equivalent indices).

The measuring quantity in personnel dosimetry for electron radiation is the dose equivalent, which can be derived from the absorbed dose to soft tissue in the semi-infinite soft tissue equivalent medium by application of the conversion factor 1 Sv/Gy.

For representation of the unit of absorbed dose rate for beta radiation, the British Committee on Radiation Units and Measurements (BCRU 1974) has proposed the absorbed dose rate in air measured free in air, $\dot{D}_a$, which may be considered an approach in analogy to the representation of the unit of exposure for photons using free-air-chambers.

The measuring quantity absorbed dose rate in soft tissue $\dot{D}_t$ at a specified depth of a semi-infinite phantom (ICRU 1964) was preferred by LMRI (France) and PTB (Germany) for reasons of the practical importance of $\dot{D}_t$ in therapy and radiation protection (7).

The concept of dose quantity $\dot{D}_t$ in beta dosimetry is also followed by a European Committee preparing a Technical Recommendation document on the "Calibration of Measuring Instruments for Use in Radiation Protection" (CEC, 1980). Accordingly, radiation protection instruments should be calibrated in terms of this quantity.

METROLOGY

There are two different types of primary standards for beta radiation due to the different approaches of dose representation.

For metrological realization of absorbed dose rate "in air free in air", $\dot{D}_a$, NPL has developed a primary standard instrument consisting of thin foils that form a parallel plate chamber perpendicular to the beam axis (8).

By contrast, LMRI, PTB and other national laboratories favour a beta primary standard metrology based on an extrapolation chamber to represent the unit of absorbed dose rate in soft tissue, $\dot{D}_t$ (9,10).

The extrapolation chamber allows the absorbed dose in a solid medium to be determined from the ionization in a small gas-filled cavity inside the medium by extrapolation to zero-volume. For the perturbation of the radiation field by the chamber due to differing scattering and stopping properties of the gas and the solid material, BÖHM (11) has recently determined perturbation correction factors.

Technically, extrapolation chambers consist of low-atomic number material, they are air-filled and use a variable spacing of a special parallel plate ionization chamber. An example of set-up is shown in Fig. 1. The procedure is to measure the ion current between the conductive entrance window and the collecting electrode for different electrode distances. Extrapolating the resulting curve back to zero distance, a good estimate of the superficial dose at the centre can be obtained.

The extrapolation chamber is to measure absorbed dose rate at the beta source surface or at some source distance. By varying the entrance foil thickness, the absorbed dose rate can be assessed in a specific depth, even at zero depth by extrapolation to zero thickness of the entrance foil.

$\dot{D}_t$ is calculated from the absorbed dose rate $\dot{D}_a$ in air and the collecting volume according to the Bragg-Gray theory by multiplying $\dot{D}_a$ with the mass stopping power ratio for tissue and air, $s_{t,a}$, and with the correction factors K_{back} and K_{foil} (12):

$$\dot{D}_t = \dot{D}_a \cdot s_{t,a} \cdot K_{back} \cdot K_{foil}.$$

$\dot{D}_a$ is evaluated from

$$\dot{D}_a = \frac{\bar{W}}{e} \cdot \frac{K'}{a \cdot \varrho} \cdot \frac{d}{dl} \quad (i.K(l)),$$

wherein

$$K' = K_{gap} \cdot K_{scat} \cdot K_{brems} \cdot K_{dec} \cdot K_{hum},$$

and

$$K(l) = K_{sat} \cdot K_{inh} \cdot K_{sw} \cdot K_{\varrho}$$

with K_{sat} = lack of saturation due to initial and volume recombination and diffusion loss. K_{inh} = inhomogeneity of primary radiation field inside collecting volume. K_{sw} = beta particles scattered from side walls. K_{ϱ} = air density correction. K_{back} = difference in backscatter between tissue and chamber material. K_{foil} = foil thickness and material referred to 2 mg/cm^2 tissue equivalent foil. K_{gap} = gap between collector electrode and guard ring (lack of back scatter, information of collecting volume). K_{scat} = beta particles scattered from material in the source-chamber neighbourhood. K_{brems} = bremsstrahlung emitted from source. K_{dec} = source decay. K_{hum} = humidity in air. $s_{t,a}$ = mass stopping power ratio tissue/air. i = ionization current. W/e = conversion factor 33.73 V.a = area of collecting electrode. l = chamber depth. ϱ = air density.

The reliability of the extrapolation chamber dosimetry is remarkable and may be understood from Fig. 2, which shows depth dose curves for plastic foils achieved at PTB with the primary standard measuring device, and at GSF with a commercial PTW extrapolation chamber for the low-energy beta emitter Pm-147 with the mean beta energy of 60 keV.

The extrapolation chamber is applicable as a reference or transfer instrument for calibration purposes but not for routine radiation protection purposes.

The same is true for extremely thin-walled ionization chambers used in radiation therapy. By contrast, flat ionization chambers primarily used in low-energy X-ray dosimetry are designed for standard purposes as well as for field dosimetry. The results of flat ionization chambers and of extrapolation chambers agree well, at least, for beta energies $\gtrsim$ 200 keV.

BETA PROTECTION LEVEL DOSEMETERS

Recently, portable dose equivalent rate meters were developed for photons and electrons on the basis of twin-ionization chambers. Calibration is in terms of dose equivalent rate at two specified depths (7 mg/cm^2, and 800 mg/cm^2 of a tissue equivalent 30 cm sphere using a solid backscatter rear body (13), respectively behind 1000 mg/cm^2 (14)).

Besides ionization chambers other radiation detectors are applicable to beta dosimetry, e.g., surface barrier detectors, GM counters, film and solid state dosemeters.

Surface barrier detectors operated in pulse counting mode show usually a high energy dependence of response. Some recent results are more promising where energy dependence is less than ± 25% for maximum beta energies between 0.23 MeV and 2.2 MeV (15).

Using a semiconductor as an extrapolation chamber where the sensitive volume is varied by variation of the detector voltage, the results agree well with a normal extrapolation chamber for P-32, Tl-204 and Sr-90/Y-90 (16).

GM counters can be applied for flux determinations of radioactive beta sources but not directly for dose rate assessment. The major drawback is the energy dependence. Even with a so-called beta-window, GM counter instruments may considerably underestimate or even ignore the beta component and thus pretend a too low dose rate. For reasons of misinterpretation, such instruments may even be dangerous for radiation protection practice.

A scintillation counter with a thin tissue equivalent scintillator reported by GÜHNE (17) allows accurate dose rate determination. However this method does not yet lend itself for routine application.

HAJNAL and McLAUGHLIN (14) present similar results on a survey instrument equipped with a scintillation detector to determine surface dose rates.

Film dosemeters are strongly energy dependent caused by the envelop material. Since the radiation quality of beta particles can approximately be estimated from the blackening pattern behind the filters, correction factors can be applied to achieve overall accuracies of better than ± 30% on the basis of a 95% confidence level. This is true particularly for particle energies above 0.5 MeV. Fig. 3 shows personnel monitoring films exposed

to beta radiation in the GSF film badge. For Pm-147 we found differences in the blackening pattern doubtlessly typical for the respective beta source under use : For an earlier GSF calibration source, the low-energy betas could be detected while in case of the CEC intercomparison only bremsstrahlung was found for Pm-147.

Thermoluminescent dosemeters (TLD) promise a low beta field perturbation and a high spatial resolution. However, the beta response depends strongly on the detector thickness. Usual TLDs with 0.9 mm thickness underestimate the beta dose rate by a factor 2 even for Sr-90/Y-90, by a factor 5 for Tl-204 and by a factor 30 for Pm-147, for reasons of particle absorption and, thus, inhomogeneous detector penetration (Fig. 4). However, with sufficiently thin-film TLDs ($\approx$ 5 mg/cm^2) beta dose rates can be assessed energy independently above 50 keV mean beta energy and this for specified depths of $>$ 2 mg/cm^2 (Fig. 5).

These results agree well with those achieved with ultra-thin bonded LiF/Teflon (5 mg/cm^2) behind a 5 mg/cm^2 window, whereas commercial 0.2 mm LiF/Teflon detectors (40 mg/cm^2) underestimate the Pm-147 dose by about a factor of 5 (18).

Also TSEE dosemeters lend themselves for even low-energy beta dosimetry, for reasons of the extremely thin EE active layer. Respective data obtained from BeO and $\alpha\beta$-Al_2O_3 on quartz substrates after evaluation on the GSF TSEE reader are given in Fig. 6. From this figure, dosimetry of electrons with mean energies even beyond 50 keV seems possible.

ROUTINE CALIBRATION

The use of calibrated beta sources is convenient for routine calibration purposes of protection instruments rather than the use of chambers.

Acceptable methods for the calibration of beta sources intended to be applied for instrument calibration are the

- direct comparison with a primary standard at a national laboratory
- measurement by a suitable reference instrument whose performance has been checked against a primary or secondary standard
- comparison with secondary standard sources using a suitable reference instrument.

Recently, PTB together with a commercial supplier have developed a Beta Secondary Standard containing 4 interchangeable beta sources. The specifications of the 4 beta sources are given in Table 1. Each has an active area of about 1 cm^2. Relative spectral electron flux densities as measured by means of uncooled 2 cm^2 Si(Li) semiconductor detectors have been shown by BÖHM (11) (Fig. 7). Calibration of each Beta Secondary Standard is tracing back to the PTB primary standard. The apparative set-up of the Beta Secondary Standard is shown in Fig. 8. There is the source stand, the control unit with digital time preset and LED real time display, and a transport box with the source shieldings, distance holders, field homogenizing plastic filters and a beta source manipulator.

For beta studies concerned, e.g., depth dose measurements, determination of transmission factors etc., the availability of a standard quality dosemeter is recommended. For this reason, the Beta Secondary Standard should be backed up, e.g. with an extrapolation chamber as is shown in Fig. 9 for the GSF beta calibration facility.

The extrapolation chamber used at GSF is from PTW, Freiburg, the type number is 23391 (see Fig. 1). The entrance foil consists of a graphited polyimid foil of 4.5 $mg.cm^{-2}$ mass per area. Optional foil thicknesses of 8.5 $mg.cm^{-2}$ and 12.5 $mg.cm^{-2}$ are available. The chamber walls are of aluminium except the frontal ring which is made of polyimid. The diameter of the entrance window is 60 mm, the spacing of the electrodes is variable from 0.5-25.5 mm. The guard ring and collecting electrode is available in brass or, alternatively, tissue equivalent (TE) material (A150*). The diameters of guard ring and collecting electrode are delivered in 5 corresponding sizes (see Fig. 1, insert). For the present studies, TE electrodes were used with a diameter of 30 mm of the collecting electrode and with a width of 19 mm of the guard ring.

INTERCOMPARISON OF TL BETA DOSIMETRY

The capabilities of the GSF thin-film TLD were proven in the CEC Dosimetry Intercomparison Programme 1979/80 with beta irradiations to unknown doses at 3 primary standard dosimetry laboratories (PSDL) (LMRI, NPL, PTB). Fig. 10 shows the intercomparison results referring to both a GSF or external calibration provided by the respective PSDL. The evaluation

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* for specifications see ICRU 26 (1977).

based on external calibration showed a relative mean absorbed dose of 0.99 with a coefficient of variation of 5.7%. For GSF calibration the relative mean absorbed dose was 0.98 resp. 1.000 with resp. without two outliers of around 28% from NPL for Pm-147. The corresponding coefficients of variation were 9.3% resp. 6.7%.

Considering the two NPL outliers in case of GSF calibration, it must be understood that GSF calibration is tracing back to the PTB primary standard. Thus the outliers may probably be explained by the different metrological approaches of both PSDLS to represent the absorbed dose to tissue (Table 2).

From the intercomparison results thin-film TLD may be considered a kind of secondary standard or reference beta dosemeter for many practical situations. Precondition is a precision TLD "metrology" with the uncertainties kept small by proper evaluation techniques.

CONCLUSION

Accurate beta dosimetry is possible on the basis of conventional ionization chamber technique. However, the application of this time-consuming method, e.g., with an extrapolation chamber is limited to standard laboratories or to special investigations. Routine beta dosimetry will profit from the new Beta Secondary Standard based on calibrated beta sources. This will most probably stimulate new activities concerning the detection of beta radiation with area surveillance instruments and personal dosemeters. For comparison of beta calibration factors, standardization of the calibration conditions should be provided.

The promising results of the CEC Beta Dosimetry Intercomparison 1979/80 have shown the principle usefulness of TLD in beta dosimetry. Further efforts will be necessary for improving its routine application and that particularly in mixed photon/beta fields.

The requirements to TLD personal dosemeters for photon and beta radiation are being drafted by different national and international bodies. They should be the guideline for future investigations.

Work has also to be done to relate the measurable quantities of beta radiation to quantities relevant to radiation protection by proper derivations or calculations.

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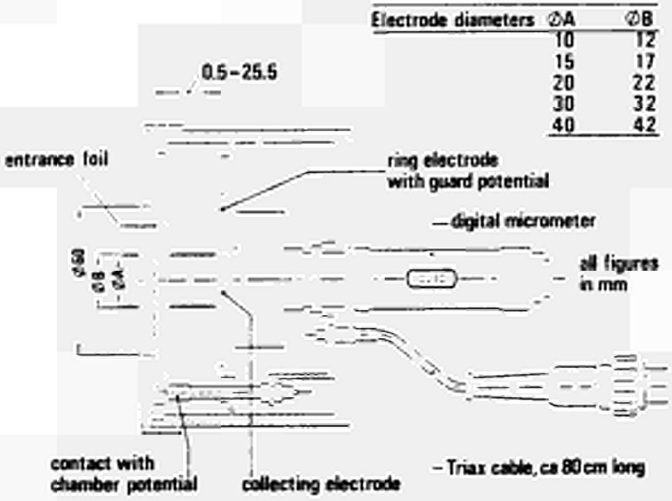

Fig. 1 Cross-sectional view of a commercial extrapolation chamber. With courtesy from PTW, Freiburg/FR Germany

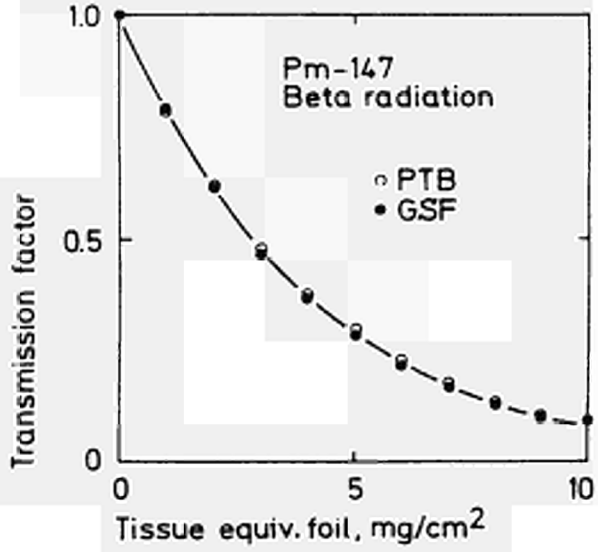

Fig. 2 Comparison of transmission factors determined at PTB and GSF for hostaphane foils and Pm-147 beta particles related to entrance foils of identical thickness (0.56 mg/cm^2).

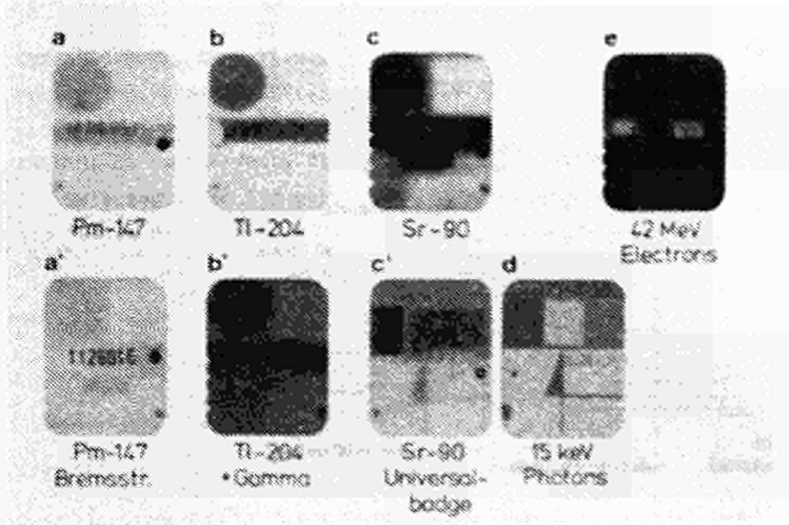

Fig. 3 Filter pattern of routine personnel monitoring films exposed to different types of beta radiation in the so-called roentgen-beta-gamma badge (a,b,c). Beta doses around 1 rad in soft tissue. Note the possible differences of filter pattern due to beta particles (a) and bremsstrahlung (a') in case of Pm-147. Effect of pure beta radiation (b) or mixed beta/gamma radiation (b') for Tl-204. Differences of filter pattern due to the roentgen-beta-gamma badge (c) or universal badge (c') for Sr-90/Y-90. Discrimination of beta particles (Sr-90/Y-90) (c') and low-energy photons (15 keV) (d) using a filter pair (Al/Cu) of equal mass per area but different atomic number in the universal badge. (e) Filter pattern due to 42 MeV electrons from a betatron.

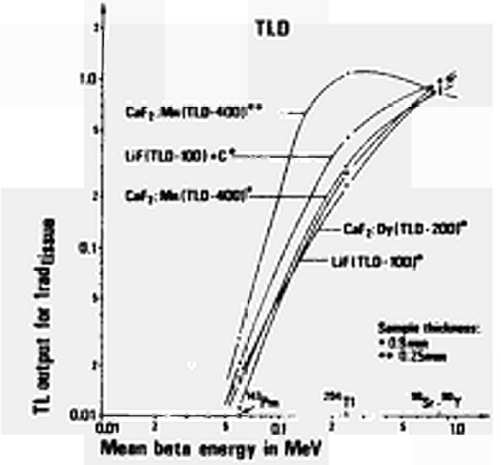

Fig. 4 Relative response of some commonly used thermoluminescent dosimeters (TLD) as a function of beta particle energy. Detector thickness 0,9 mm and 0,25 mm. Reference radiation Co-60 under CPE conditions. Beta dose 1 rad in soft tissue (19).

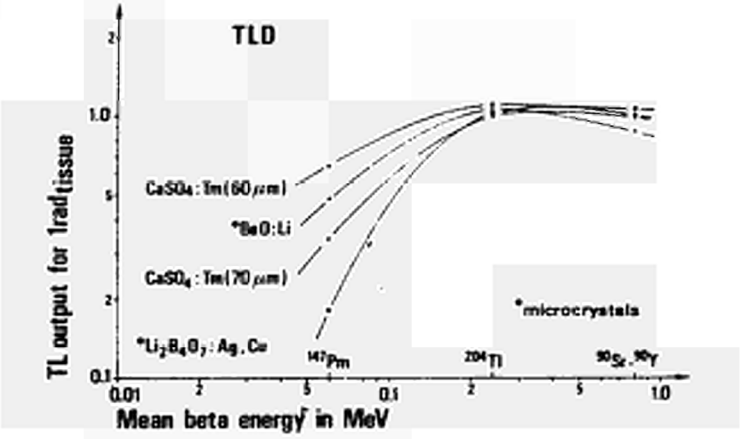

Fig. 5 Relative response of special thin and ultra-thin thermoluminescent dosimeters (TLD). Reference radiation Co-60 under CPE conditions. Beta dose 1 rad in soft tissue (19).

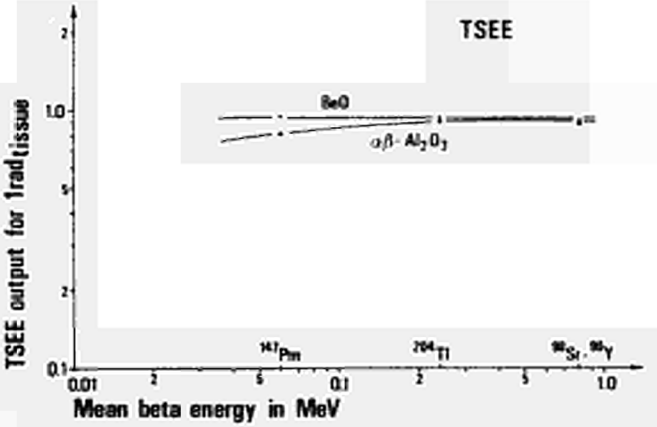

Fig. 6 Relative response of thermally stimulated exoelectron dosimeters (TSEE). Reference radiation Co-60 under CPE conditions. Beta dose 1 rad in soft tissue (19).

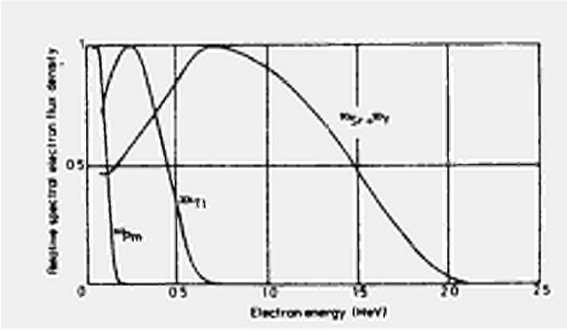

Fig. 7 Relative spectral electron flux density of the beta calibration sources given in table 1 (except the high-activity Sr-90/Y-90 source) with beam flattening filters. Spectra determined at the distance of the phantom surface by means of uncooled 2 cm^2 SI(Li)-semiconductor detectors. The flux densities are normalized at the same maximum value, but not corrected for instrumental resolution and backscattering loss of the detectors (from BÖHM [11]).

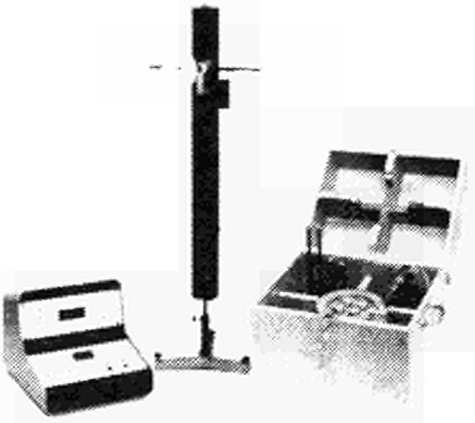

Fig. 8 View of the PTB developed and calibrated Beta Secondary Standard. Middle: Calibration stand with source, shutter and field homogeneizing filter. Left: Remote control unit. Right: Transport box with source container and accessories.

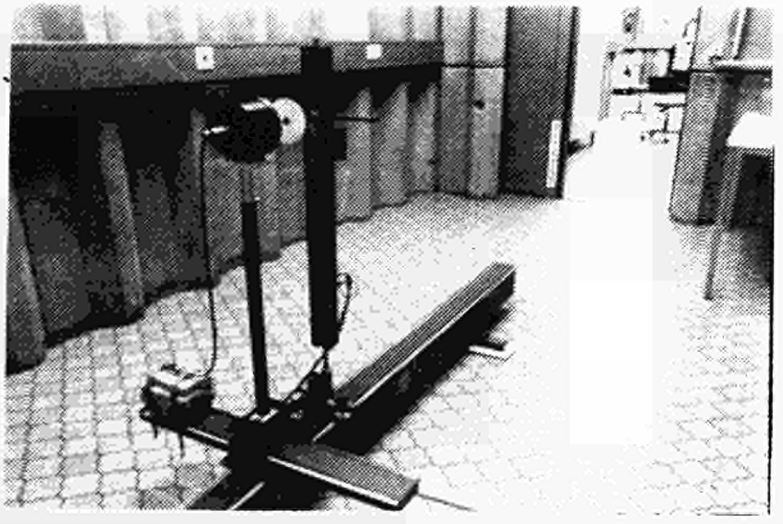

Fig. 9 Beta claibration installations at the GSF Calibration Centre, consisting of a Beta Secondary Standard, a high-precision optical bench and the PTW extrapolation chamber in a temperature and humidity controlled environment.

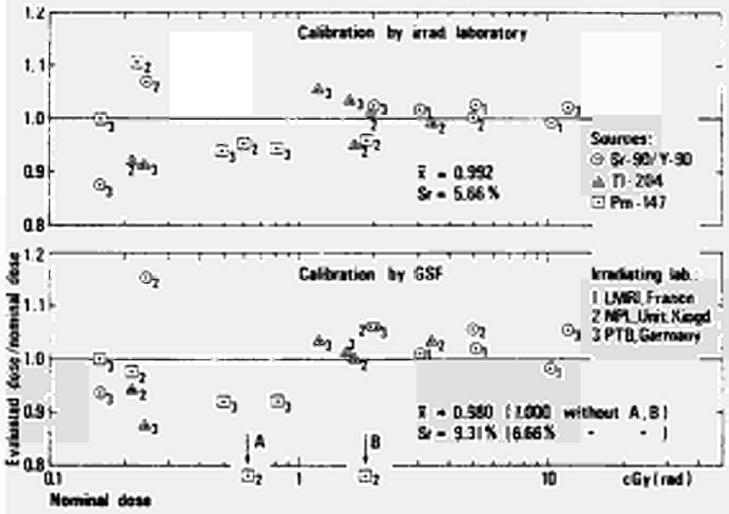

Fig. 10 Results of the GSF TL thin-film dosimeter within the CEC International Beta Dosimetry Intercomparison 1979/80.

Source	Activity	E_{max}	E_{mean}	$\dot{D}^{d=0}_{tissue}$	Cal. distance
	mCi	MeV	MeV	$cGy \cdot h^{-1}$	cm
Pm-147	14.0	0.225	0.06	0.161	20
Tl-204	0.5	0.763	0.24	0.082	30
Sr-90/Y-90	2.0	2.274	0.80	18.731	30
	50.0	2.274	0.80	208.120	11

Table 1 Dosimetric specifications of the Beta Secondary Standard.

ABSORBED DOSE TO TISSUE

GSF BETA THIN FILM TL DOSEMETER — CEC Beta Intercomp. 1979/80

Source	NPL/PTB (direct)*	GSF/NPL (TLD)	GSF/PTB (TLD)	GSF/PTB / GSF/NPL = NPL/PTB (TLD)	NPL/PTB (TLD) / NPL/PTB (direct)
Sr-90/Y-90	0.97	1.05	1.04	0.99	1.021
Tl-204	1.02	1.03	1.00	0.97	0.951
Pm-147	1.08	0.80	0.98	1.23	1.139

*acc. NPL information / Data on GSF cal., incl. nom. dose evaluation

Table 2 Comparison of calibration of the NPL and PTB primary standard beta dosimetry via the GSF TL thin-film dosimetry within the CEC International Beta Dosimetry Intercomparison 1979/80.

8. BETA-RAY DOSIMETRY IN MIXED BETA-/GAMMA-RAY FIELDS

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1. INTRODUCTION

Beta-ray dosimetry is a complex subject and there are substantial additional complications if photons are also present. Nor is there a universally accepted opinion on the quantity which should be measured.

However, it is essential that neither the limit for the effective dose equivalent or the non-stochastic limits for individual organs should be exceeded. In the case of external radiation in most practical situations, the effective dose equivalent limit can be controlled by restricting the dose from penetrating radiations, measured at some depth in the body to the whole body limit. In the case of the skin and eyes the whole body limit is over-restrictive for less penetrating radiations which may be either beta or low energy photon radiations. Strictly from the viewpoint of control there is no need to distinguish between beta and photon radiations as long as the dose is to the skin and eyes and to the whole body can be adequately assessed. This point is recognised by the CEC document EUR 5287 in its recommendations for a basic dosemeter. Requirements for a discriminating dosemeter to distinguish between different types and energies of radiations may be based on the need to identify sources of exposure so that appropriate protective measures can be taken. It is difficult to define a good specification for such a dosemeter, since it would need to be designed for a specific work situation and with a knowledge of the radiation sources that might be employed. The argument that a discriminating dosemeter will enable the effective dose equivalent to be derived may be dismissed as impracticable unless a large number of such dosemeters are to be worn on the body to enable the orientational factors to be used in the assessment.

When a non-tissue equivalent dosemeter is used for the assessment of doses to the skin and eyes and to the whole body it may be necessary because of the dosemeter response to distinguish between different radiations in order to properly assess the doses. This is obviously the case for film dosemeters. But even with tissue equivalent dosemeters there may be practical difficulties in making adequate dose assessments.

2. TWO ELEMENT NON-DISCRIMINATING BASIC DOSEMETERS

Use is currently being made of approximately tissue equivalent detectors to produce two element non-discriminating dosemeters. In an ideal dosemeter detectors of appropriate thickness would be contained beneath tissue equivalent filters of appropriate thicknesses and the sensitivity would be independent of radiation type and energy and there would be no need to separate beta-ray doses from photon doses in a mixed beta-/photon field. It does leave one important question, however, i.e. the possibility of over exposure of the eye lens if only the skin and body doses are controlled.

The current ICRP dose limits are 500 mSv, 150 mSv and 50 mSv for the skin, eye lens and whole body respectively. Taking the worst case namely the largest value recommended in EUR 5287 for the depth at which the body dose is measured i.e. 1000 mg cm^{-2} then if the depth dose distribution of the radiation in question is such that the ratio

$$\frac{D_{5-10\ mg\ cm^{-2}}}{D_{300\ mg\ cm^{-2}}} \geqslant 3.3$$

or the ratio

$$\frac{D_{300\ mg\ cm^{-2}}}{D_{1000\ mg\ cm^{-2}}} \leqslant 3$$

then the eye lens dose limits will not be exceeded

We have used the beta-ray depth dose distribution published by Cross to determine for what beta-ray exposures and irradiation distances the above criteria are satisfied. This work has shown that for all energies less than 1.5 MeV the criteria are satisfied. Above 1.5 MeV if the irradiation distances are $<$ 5 cm the eye dose limits could be exceeded by about 25% or more especially for point sources and as the energy is increased. Examination of the depth dose distributions from point sources of ^{144}Ce + ^{144}Pr (2.98 MeV), ^{106}Rh (3.53 MeV) and ^{38}Cl (4.9 MeV) shows that the eye lens dose limits could be exceeded in the worst case by a factor of 3.

Thus for certain beta-ray energies and exposure distances which will be rarely experienced in practice, the eye lens dose may become limiting. One is tempted to reduce the depth at which the body dose is measured so that the eye lens dose limit is never exceeded. However, this has a disadvantage in that the ICRP limits change from time to time which would outdate such a dosemeter design. A better approach is to fix the dosemeter design to measure the dose equivalent at 5-10 mg cm^{-2} and 1000 mg cm^{-2}. If the radia-

tions given above are expected to be present at dose rates levels which could cause the eye lens dose limits to be exceeded then special procedures should be taken i.e.

(a) the skin dose should be controlled to the eye lens dose limits
or
(b) persons concerned should wear eye protectors
or
(c) they should be issued with eye dosemeters.

3. FILM DOSEMETERS

In the UK film dosemeters are designed to give the absorbed dose at or near the surface of the body. The result is separated into the dose due to penetrating and non-penetrating radiations. Photons of energy $>$ 20 keV are considered as penetrating and all beta-rays and photons of energy $<$ 20 keV as non-penetrating.

The response of film varies markedly with radiation type and energy which necessitates the separate evaluation of photon and beta-ray doses. In general the energy response problem is overcome by covering the film with a number of plastic and metal filters. For pure beta-ray dosimetry the ratio of the blackening between two appropriate filter areas (thin plastic for beta-rays) can be used to give an indication of the beta-ray energy. A correction factor can then be applied to the apparent dose to give the true dose.

For mixtures of photons and beta-rays, provided the photon energy $>$ 20 keV the beta-ray and photon doses can be evaluated separately, but if the photon energy is less than 20 keV the photons are also absorbed in the thin plastic filters and the appearance of the film is similar to that for beta-ray exposures. This leads to errors in the dosimetry of beta-rays in the presence of low energy photons.

Film dosemeters could be redesigned to reduce these errors but there is little point in doing this for situations which rarely occur in practice.

9. THE INFLUENCE OF BREMSSTRAHLUNG ON THE CALIBRATION OF DOSEMETERS FOR BETA RADIATION - DEMONSTRATED BY MEANS OF AN EXAMPLE -

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A suitable radiation source for calibration of dosemeters for beta-radiation is the secondary standard (1) developed by the Physikalisch-Technische Bundesanstalt for absorbed dose due to beta radiation in tissue. With this secondary standard the absorbed dose rate in air $\dot{D}_C$ is known for the radiation of the 3 nuclides Pm-147, Tl-204 and Sr-90/Y-90 for specific distances from the radiation sources and in addition the absorbed dose rate in tissue is known for various tissue depths.

As generally the dosemeters in common use have a low response to beta radiation of low maximum energy. During calibration of dosemeters of this type, a simultaneously present photon radiation, can perceptibly influence readings. Marked influence of photon radiation on the dosemeter reading can be expected particularly in the case of low energy beta radiation. Thus for the radiation of the Pm-147 source the exposure rate $\dot{J}$ by photons and the ratio of the exposure rate by photons to the exposure rate by beta radiation in air was determined and the influence of the photon radiation on TLD calibration was estimated.

ESTIMATION OF EXPOSURE RATE

The exposure rate $\dot{J}$ by photons at a distance of 20 cm in front of the secondary standard - the distance for which the absorbed dose rate $\dot{D}_C$ had been determined by the Physikalisch-Technische Bundesanstalt - was determined for the radiation of the Pm-147 source with a dosemeter with plastic scintillator for low energy gamma radiation (MAV-601) and with thermoluminescence and film dosemeters.

With the dosemeter with plastic scintillator the exposura rate $\dot{J}$ was determined without additional absorber and with perspex and aluminium absorbers of various thicknesses between Pm-147 source and dosemeter. Fig. 1 shows measurement results. The weight/unit area of the absorber is given along the abscissa. Without absorber the exposure rate is $\dot{J}$ = 0.54 mR/h.

According to the data of the Physikalisch-Technische Bundesanstalt (2) and taking into account the activity decay which has occurred since the time of measurement by the Physikalisch-Technische Bundesanstalt the absorbed dose rate in air is $\dot{D}_c$ = 122 mrd/h. From this it is possible to deduce the following ratio of exposure rate by photons to absorbed dose rate in air :

$$\frac{\dot{J}}{\dot{D}_c} = 4.4 \cdot 10^{-3} \ \frac{R}{rd}.$$

From Fig. 1 itfollows that the dose rate behind an aluminium absorber decreases at a considerably higher rate than behind a perspex absorber. From this it follows that photon radiation and not beta radiation is being measured.

From the curves in Fig. 1 it is possible to calculate the mass attenuation coefficient $\frac{\mu}{\rho}$. For low absorber thicknesses the dose rate decreases more sharply than is to be expected from the measurement results with larger thicknesses. The mass attenuation coefficients determined from the values in Fig. 1 are summarized in Table I. By interpolation of the values of the mass attenuation coefficient stated by Hubbel (3) it is possible to determine an effective energy of the photons from the values of $\frac{\mu}{\rho}$ in Table I. Here the effective energy is the energy of a monoenergetic radiation with the same mass attenuation coefficients as the radiation being investigated. The effective energy values determined from measurements with perspex absorbers and with aluminium absorbers agree most satisfactorily.

In determining the exposure rate $\dot{J}$ in front of the Pm-147 source with thermoluminescence dosemeters (TLD), 3 TLD's mounted behind one another (Type 100, dimensions 0.125" x 0.125" x 0.015") were exposed to radiation simultaneously. The beta radiation was absorbed in the first two TLD's. Only the reading of the third TLD is used to estimate the exposure rate produced by photon radiation. Averaged out from the results of three exposures to radiation - assuming high energy gamma radiation - the ratio of exposure rate by photons to absorbed dose rate in air was determined as follows :

$$4.3 \cdot 10^{-3} \ \frac{R}{rd}.$$

This value still has to be corrected. The response of the TLD is higher by a factor of 1.3 to 20 keV photons - this corresponds according to Table I approximately to the energy of the photons in the radiation field of the Pm-147 source - and the photon radiation is attenuated in the first two TLD's acting as absorbers. 20 keV photons are attenuated by 13% by the first two TLD's. If these two effects are taken into account, one obtains from the measurements with the TLD's the following ratio of exposure rate by photons to absorbed dose rate in air :

$$\frac{\dot{J}}{\dot{D}_c} = 3.8 \cdot 10^{-3} \ \frac{R}{rd}.$$

With the aid of the film dosemeter the ratio of exposure rate by photons to absorbed dose rate in air was determined as

$$2.4 \cdot 10^{-3} \ \frac{R}{rd}.$$

The values determined with the various dosemeters for the ratio of exposure rate by photons to absorbed dose rate in air are summarized in Table II. The individual values deviate perceptibly from one another. As however the beta radiation complicates determination of the exposure rate and as the photons are of very low energy, the values obtained with the various dosemeters agree with one another rather well. As a mean value we obtain the following from Table II :

$$\frac{\dot{J}}{\dot{D}_c} = 3.5 \cdot 10^{-3} \ \frac{R}{rd}.$$

DISCUSSION OF MEASUREMENTS

Determination of exposure rate is prone to considerable error on account of the low energy of the photons. The reading of the dosemeter with plastic scintillator is energy-independent within $\pm$ 15% in the energy range from 25 keV to 1 MeV. The effective energy of the photons, however, is less than 25 keV according to the results obtained from absorption measurement. For energy values lower than 25 keV the dosemeter should give dose rate underreadings. Because the energy of the photons is not known with sufficient accuracy it cannot be stated by what amount the dosemeter underesti-

mates the exposure rate. In the same way estimation of the effective energy of photon radiation from the measurements in Fig. 1 is only possible with reservation, as the values obtained for effective energy are below 25 keV and still lower values cannot be obtained, quite simply because the dosemeter cannot measure lower energy radiation.

The result of dose rate measurement with TLD's depends on how many photons are absorbed in the first 2 TLD's. The lower the energy of the photons, the more will be absorbed. An effective photon energy of approximately 20 keV was assumed. The result of film dosemeter evaluation indicates even lower photon energy.

Although it is not possible to dismiss the fact that the exposure rate was underestimated with the TLD's and the dosemeter with plastic scintillator, the results of film dosemeter evaluation are even lower. On account of the low energy of the gamma radiation and the simultaneously present beta radiation, the result obtained with the film dosemeter may even, however, be prone to rather larger errors.

Determination of the exposure rate by photons in front of the Pm-147 source can only be regarded as approximation. A statement of the error limits is hardly possible. Nor is it possible to exclude the fact that the ratio of exposure rate produced by photons to absorbed dose rate has in fact been underestimated.

ORIGIN OF PHOTON RADIATION

The photons can be gamma radiation of the Pm-147, gamma radiation of another isotope or bremsstrahlung.

The Pm-147 emits gamma radiation with an energy of 120 keV in 0.01% of all decays. It can be assumed that this low emitted gamma quantum number cannot be the cause of the photon radiation established.

The low effective energy value of approximately 20 keV is evidence against the assumption that gamma radiation of a different nuclide is the source of the photons.

A calculation of the number of photons produced by internal and external bremsstrahlung is complicated. According to simple estimates a bremsstrahlung quantum is to be expected in approximately 1% of all decays. The number of photons is sufficient to explain the measured values for exposure rate by photons.

INFLUENCE OF THE PHOTON RADIATION ON THERMOLUMINESCENCE DOSEMETER READINGS

The measured ratio of the exposure rate by photons to the absorbed dose rate of $3.5 \cdot 10^{-3} \frac{R}{rd}$ is low. Nevertheless a photon radiation must be taken into account with the readings of dosemeters for radiation protection monitoring. This will be shown with the examples of various TLD's.

In radiation protection monitoring in the case of beta radiation the value of the absorbed dose or absorbed dose rate has to be determined in tissue at 70 μ depth and not the absorbed dose rate in air. For the Pm-147 source of the secondary standard the value of the absorbed dose rate in tissue at 70 μ depth is smaller by a factor of 4.80 than the absorbed dose rate on the surface. As photon radiation is not attenuated in practice by 70 μ tissue, the ratio of the exposure rate of the photons in air to the absorbed dose rate in tissue at 70 μ depth is equal to $1.7 \cdot 10^{-2} \frac{R}{rd}$.

Although they are used for determining the beta dose rate, radiation protection dosemeters are frequently not sensitive enough to low energy beta radiation. Table III gives a number of values of the TLD response to Pm-147 beta radiation. In addition, on the assumption that the response of the TLD's is too high to 20 keV gamma radiation by a factor of 1.3, it has been calculated what fraction of the reading in the case of exposure to radiation in front of the Pm-147 source of the secondary standard is produced by photons and not by beta particles. From the values in the Table III it follows that with the only 0.39 mm thick TLD's 8% of the reading is produced by photons. With thicker TLD's the influence of photons is correspondingly larger and behind somewhat thicker absorbers (line 3 in Table III) practically the whole reading can be attributed to bremsstrahlung.

CONCLUSIONS FOR MEASUREMENTS AT OTHER RADIATION SOURCES

The results in Table III make it possible to generalise. The lower the response of a dosemeter to Pm-147 beta radiation, the larger will be the fraction of the reading caused by bremsstrahlung.

The response measured at the Pm-147 source of the secondary standard is not valid in every case for measurements in the vicinity of other Pm-147 sources because the ratio of exposure rate produced by photons to absorbed dose rate produced by beta radiation depends on the type and volume of material situated between radiation source and measuring point. The less

material located between source and measuring point, the lower will be the ratio. As calibration by the secondary standard is carried out at a relatively long distance from the source, the ratio in this case between exposure rate by photons to absorbed dose rate by beta radiation is relatively high. For this reason, excessively high response values will be found with dosemeters having low response to Pm-147 beta radiation at the secondary standard when compared with measurements carried out at other Pm-147 sources.

In the case of calibration with Tl-204- or Sr-90/Y-90 beta radiation it is not generally necessary to take into account the simultaneously present bremsstrahlung as the response of the dosemeters to Tl-204 and Sr-90/Y-90 beta radiation is considerably higher than to Pm-147 beta radiation.

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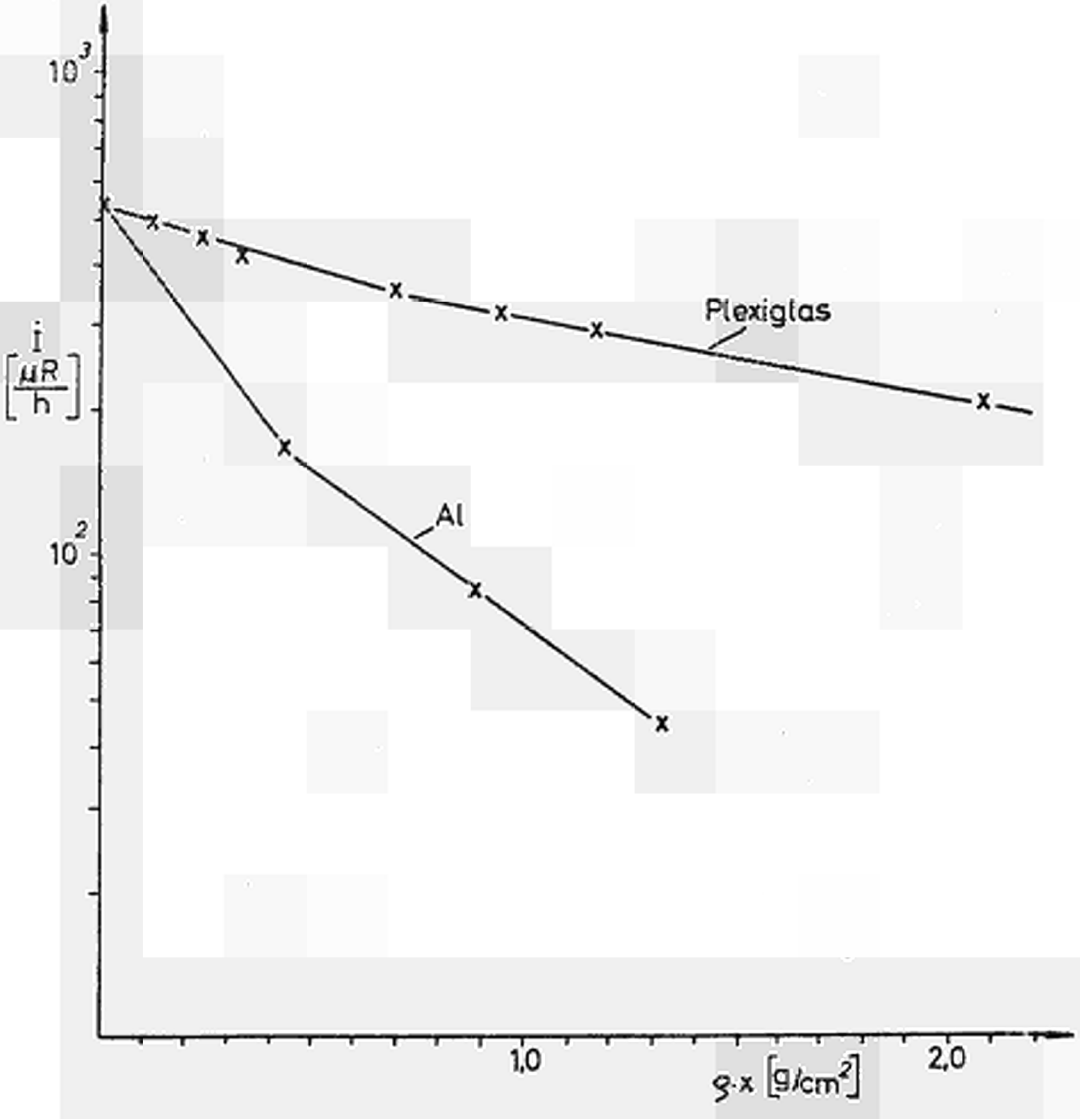

Fig.1: Exposure rate as function of the absorber thickness in a distance of 20 cm to the Pm-147 source of the secondary standard

Material	$\frac{\mu}{\rho}$ $\left[\frac{cm^2}{g}\right]$	E_γ [keV]	
Plexiglas	0,60	≈ 19	$\rho \cdot x < 0,4\ g/cm^2$
	0,39	≈ 24	$\rho \cdot x > 0,5\ g/cm^2$
Aluminium	2,67	≈ 22	$\rho \cdot x < 0,4\ g/cm^2$
	1,44	≈ 27	$\rho \cdot x > 0,5\ g/cm^2$

Table I: Mass attenuation coefficients and effective energies of the photons of the Pm-147 source

Dosimeter	$\frac{\dot{i}}{\dot{D}_C}$ $\left[\frac{R}{rd}\right]$
Dosimeter with scintillator (type MAB-601)	$4,4 \cdot 10^{-3}$
TLD-100	$3,8 \cdot 10^{-3}$
Filmdosimeter	$2,4 \cdot 10^{-3}$

Table II: Ratio of the exposure rate of photons to the absorbed dose rate of β-rays in a distance of 20 cm to the Pm-147 source of the secondary standard

Dosimeter*)	Response $\frac{R}{rd}$	Fraction of the reading induced by bremsstrahlung of the Pm-147 source [%]
TLD 100; 0,89 mm	0,13	17
TLD 100; 0,39 mm	0,29	8
TLD 100; 0,39 mm**)	0,024**)	92**)

*) Dosimeters covered by an absorber of 1 mg/cm^2

**) TLD irradiated behind another TLD of the same type

Table III: Response of TLD for Pm - 147 ß-radiation and fraction of the reading induced by bremsstrahlung

10. DETERMINATION OF ABSORBED DOSE RATES WITH AIR-FILLED EXTRAPOLATION CHAMBERS AT SMALL CHAMBER DIMENSIONS OR LOW GAS PRESSURES

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ABSTRACT

The interpretation of extrapolation chamber measurements performed at small chamber dimensions or low gas pressures becomes difficult if the current due to ionization in the chamber gas is falsified by a non-negligible current generated by secondary electrons emitted from the chamber walls. The disturbing current can be eliminated by using the double extrapolation chamber described here. Typical extrapolation curves obtained with this chamber at small chamber dimensions and low gas pressures are presented for (^{90}Sr + ^{90}Y) beta radiation. The disturbance of the ionization current by secondary electrons from the walls can be taken into account by a correction factor K_{ion}. Values of the product of K_{sat}, the correction factor due to incomplete charge collection, and K_{ion} are given for a two-electrode plane-parallel ionization chamber irradiated by (^{90}Sr + ^{90}Y) beta radiation.

1. INTRODUCTION

The determination of absorbed dose rates for beta radiation at the point of interest in a phantom is rather delicate, especially at the surface of a phantom or close to interfaces between materials of different atomic numbers. Extrapolation chambers are the instruments most often used for this purpose.

The absorbed dose rate in the phantom material, $\dot{D}_t$, is calculated from the absorbed dose rate $\dot{D}_a$ in the chamber gas according to the Bragg-Gray relation :

$$\dot{D}_t = \dot{D}_a s_{t,a} \qquad (1)$$

where $s_{t,a}$ is the mass-stopping power ratio for the phantom material and the chamber gas. $\dot{D}_a$ is obtained by [1]

$$\dot{D}_a = \frac{\bar{W}}{e.a.\varrho} \; K' \frac{d}{dx} (i_u \, K(x)), \qquad (2)$$

where K' and K are correction factors and

$\bar{W}$ is the average energy expended in air per ion pair formed,
e is the electron charge,
a is the effective area of the collecting electrode,
ϱ is the air density,
i_u is the ionization current due to the undisturbed ion dose rate for a chamber built with walls fully matched to the chamber gas with respect to the effective atomic numbers, and
x is the chamber depth.

$\frac{d}{dx} (i_u \, K(x))$ denotes the slope of the extrapolation curve, which is the plot of the ionization current, corrected by K(x), versus the cavity thickness x.

The average current i_e of an extrapolation chamber, measured for both polarities of the chamber voltage, is composed of three contributions :

$$i_e = i_u + i_f + i_s \qquad (3)$$

The current i_f is due to additional ionization caused by the interface-effects from the walls, and the current i_s is due to the charge transport of low energy non-ionizing secondary electrons liberated from the walls of the chamber.

At normal temperature and pressure, i_f and i_s can usually be neglected compared with i_u, and i_u is denoted as "ionization current". At very small chamber dimensions or low gas pressures however, the influence of i_f and i_s on the measured current becomes evident. By means of a conventional, two-electrode extrapolation chamber, the separation of i_u, i_f and i_s is impossible, whereas a three-electrode parallel-plate chamber with variable electrode distances and chamber gas pressure makes possible the determination of the different contributions to i_e. The arrangement of a three-electrode double extrapolation chamber is described and the contributions of $i_f + i_s$ to i_e for ($^{90}Sr + ^{90}Y$) beta radiation are discussed.

2. EXPERIMENTAL ARRANGEMENT

A schematic diagram of the double extrapolation chamber, operated in a gas-tight vessel is given in Fig. 1. Three electrodes (materials: tin - carbon - carbon) form two plane-parallel component chambers with a common central measuring electrode which is surrounded by a guard ring. A sensitive current measuring system capable of measuring currents down to 10^{-17} A was connected to the central electrode.

3. EXTRAPOLATION CURVES AT LOW GAS PRESSURES

Extrapolation curves obtained with the chamber gas air at pressures of 17.91 mbar and 0.001 mbar (vacuum) are shown in Fig. 2. The depth x_c of the carbon-carbon (C-C) chamber was varied between 773 μm and 2770 μm, while the depth x_t of the tin-carbon (Sn-C) chamber remained constant (708 μm). The polarities of the chamber voltages of both component chambers and the air pressure are indicated at the right of Fig. 2 for each extrapolation curve.

As one would expect, the signs of the slopes $\frac{di_e}{dx}$ of the extrapolation curves depend on the polarity of the chamber voltage of the C-C chamber at p = 17.91 mbar, while the absolute values of $\frac{di_e}{dx}$ differ for both polarities. For p = 0.001 mbar (vacuum), i_e is almost independent of x_c. The shaded areas indicate the range of i_e due to the variation of the chamber voltages of the C-C and Sn-C chamber between 13 Volt and 60 Volt. The voltage dependence is strong for equal polarities of both chamber voltages, but small for opposite polarities.

4. THE CONTRIBUTION OF i_f AND i_s TO i_e

In the "compensation mode" (2,3) (tin electrode positive, outer carbon electrode negative with respect to the central measuring electrode), the double extrapolation chamber allows easy determination of $\dot{D}_a$ as

$$\frac{d}{dx_c}\,(i_u\,K(x_c)) = \frac{d}{dx_c}\,(i_e\,K(x_c)) \qquad (4)$$

For two different depths x_{c1}, x_{c2}, the measured currents $i_e(x_{c1})$ and $i_e(x_{c2})$ yield the slope :

$$\frac{d}{dx_c}\,(i_u\,K(x_c)) = \frac{i_e(x_{c1})\,K(x_{c1}) - i_e(x_{c2})\,K(x_{c2})}{x_{c1} - x_{c2}} \qquad (5)$$

The slope was determined according to Eq. (5) for x_{c1} = 1583 μm and x_{c2} = 1178 μm in the air pressure range from 9 mbar to 194 mbar. The quotient of the slope and the air density turned out to be constant within $\pm$ 0.8% in the whole pressure range.

The slope was used to calculate the ideal current i_{id} of a two-electrode plane-parallel chamber. i_{id} was compared with the actually measurable average current i of a C-C chamber and a Sn-C chamber, obtained from measurements with both polarities of the chamber voltage. The ratio i_{id}/i can be considered as the product of two correction factors K_{ion} and K_{sat}:

$$\frac{i_{id}}{i} = K_{ion} \cdot K_{sat} \qquad (6)$$

where K_{ion} is the correction factor for the disturbance of the ionization current by secondary electrons, and K_{sat} that for the loss of positive and negative ions by thermal diffusion to the collector against the charge separating field. The variation of $K_{ion} \cdot K_{sat}$ with the air pressure is shown for two chamber depths in Fig. 3a and 3b. As the diffusion loss varies with U^{-1} (4) and not with E^{-1} (U denotes the potential difference between the electrodes, E the electric field strength), the heights of the plateaus for U = 13 V coincide for both chamber depths. The increase of i by secondary electrons is governed by back diffusion processes (2) and is assumed to vary with E/p, (E = electrical field strength, p = pressure of chamber gas). Thus, for pressures of $\sim$ 20 mbar, where the disturbance of the ionization current is caused predominantly by secondary electrons, the $K_{ion} \cdot K_{sat}$ values are closer to unity at lower chamber voltages. The curves for the Sn-C chamber lie below those of the C-C chamber because of the stronger secondary electron emission of the tin electrode compared with that of the carbon electrode.

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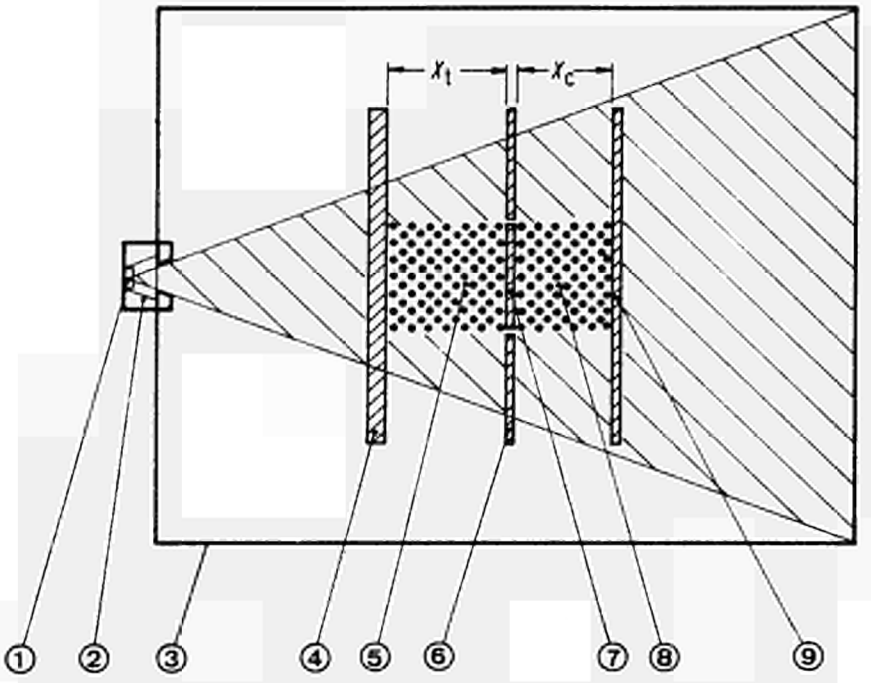

Fig. 1 Schematic diagram of the extrapolation chamber with ß-source mounted in a vacuum vessel. The lengths x_t and x_c are the respective chamber depths of the tin-carbon and the carbon-carbon chamber. x_t and x_c are measured by two laser interferometers. The shaded area indicates the cone of the ß-radiation within the vessel.

1. (^{90}Sr + ^{90}Y)ß-source of 4 GBq activity, covered by 50 µm stainless steel foil and 50 µm Al window.
2. Brass shielding, collimator and shutter of the source.
3. Stainless steel vessel to establish chamber gas pressures from 0 bar to 1 bar.
4. 50 µm tin foil entrance window, 100 mm in diameter.
5. Active volume of the tin-carbon component chamber.
6. Grounded guard ring, insulated against the charge collecting area by an annular groove about 0.5 mm in width.
7. Charge collecting area 30 mm in diameter on either side of central foil (carbon coated Hostaphan foil of 2 µm thickness.)
8. Active volume of the carbon-carbon chamber.
9. Exit window (carbon-coated Hostaphan foil of 2 µm thickness).

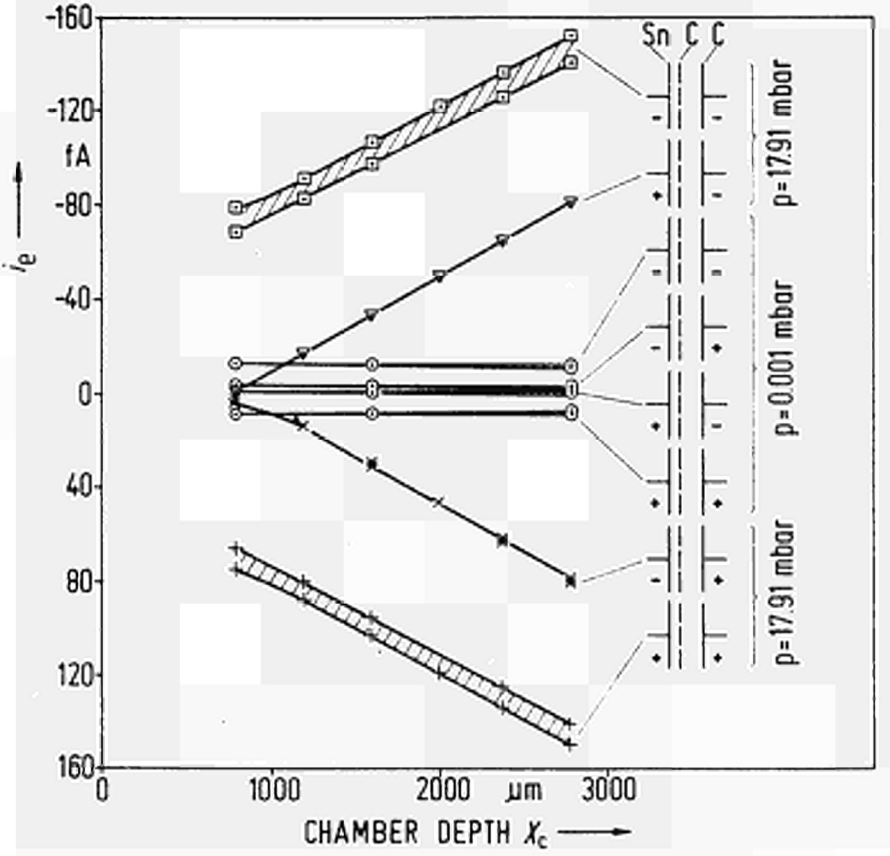

Fig. 2 Extrapolation curves (chamber current versus chamber depth x_c) for air pressure p = 0.001 mbar (vacuum) and p = 17.91 mbar. The different polarities of the chamber voltages of both component chambers are indicated. The shaded areas between the curves cover the current range measured for chamber voltages between 13 V and 60 V.

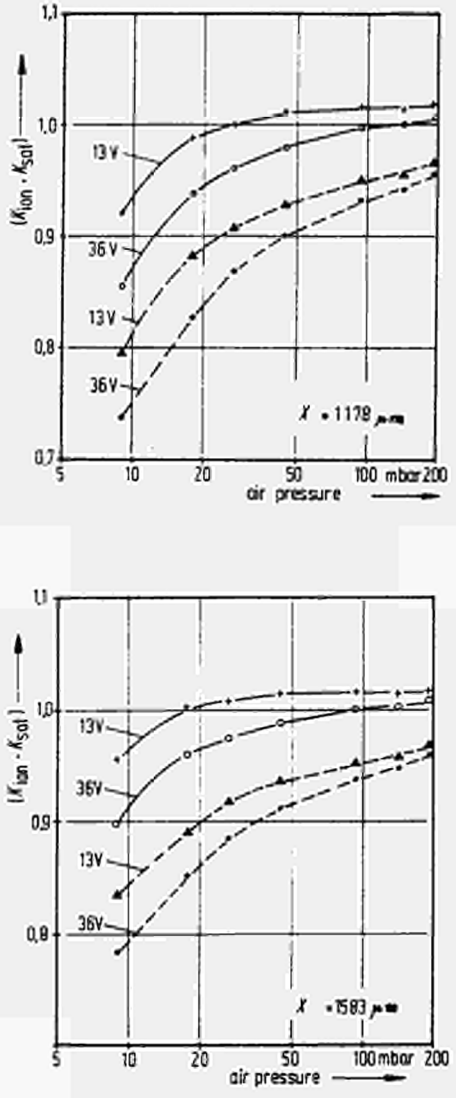

Fig. 3 Product of the correction factors $K_{ion} \cdot K_{sat}$ for an air-filled plane-parallel ionization chamber for chamber depth X of 1178 µm (Fig.3a) and 1583 µm (Fig.3b) in the pressure range of 9 mbar to 194 mbar. The full lines represent $K_{ion} \cdot K_{sat}$ for a chamber with "walls matched to the gas" (C-C chamber), the dashed lines indicate the correction factors of an unmatched chamber (Sn-C chamber), both irradiated by a ($^{90}Sr + ^{90}Y$)β-radiation. The parameter of the curves is the absolute value of the applied chamber voltage.

11. SUMMARY OF EDITING COMMITTEE

Session III of Part A and the whole of Part B of the Seminar were devoted to the presentation of papers on topics of major interest related to personal monitoring for beta raditations. These presentations and the ensuing discussions highlighted the technical difficulties which still remain in this area of personal monitoring and indicated that the objectives are not always clear.

Beta-ray monitoring is mainly concerned with limiting the dose to skin and the eye lens but irradiation of some of the deeper organs, such as the male gonads, by the more energetic beta-ray emitters must not be ignored. Over the past few years the ICRP has introduced changes in the dose limit for the eye lens which has adversely affected the performance of some dosemeters especially those for which the relationship between the dose limits for skin, eye lens and whole body is part of the design philosophy. Thompson pointed out the very large differences between the limits applied in the USA and in the European Communities for the skin and eye lens which does little to clarify the objectives in dosemeter design.

The present situation with regard to quantities and units was presented by Portal. The effective dose equivalent and dose equivalent indices are difficult if not impossible to measure with dosemeter systems having an acceptable degree of complexity so that some agreed simplification of these concepts is required for practical application. Because of the relatively low penetrating power of beta-rays the problem is less severe for beta-ray monitoring than it is for photon and neutron monitoring. Nevertheless it is important to ensure that the quantity chosen for beta-radiation fits into a consistent scheme for radiation monitoring as a whole. A number of ideas have been put forward for the quantity to be measured operationally and the practical application of dose equivalent index is under active consideration by an ICRU Working Party. Hopefully an early solution will emerge.

Interesting papers were given on calibration and standardization for beta-ray dosimetry. A draft ISO standard ISO/DP6980 specifies two series of beta-ray reference radiations. The first series consists of sources of Pm-147, Tl-204 and Sr/Y-90 together with beam flattening filters.

This series is meant for use under prescribed geometrical conditions in which case the dose rate is uniform over the calibration plane. The second series includes sources of C-14 and Ru/Rh-106 and is meant for use under variable geometrical conditions; the uniformity of the dose rate over the calibration plane must be checked in this case. The sources in the first series are now available and commonly used for beta-ray calibrations. The output from these sources is given in terms of absorbed dose to tissue at a depth of 7 mg/cm^2. When the quantity to be measured is clarified it may be necessary to supply the absorbed dose rate at other depths, for example, 300 mg/cm^2 and 1000 mg/cm^2. Heinzelmann drew attention to the fact that the X-ray dose rate from silver encapsulated Pm-147 sources becomes significant for some types of dosemeters and Böhm presented an interesting paper on air filled extrapolation chambers with small chamber dimensions or with low gas pressures.

Several authors discussed the required and actual performance of beta-ray personal dosemeters. One major uncertainty concerned the design and objectives of 2 element non-discriminating dosemeters. These dosemeters are usually designed to measure skin dose at a depth of 7 mg/cm^2 and body dose at some depth between 400 and 1000 mg/cm^2. Designs are such that if skin and body doses are controlled to their appropriate dose limits then the eye dose will automatically be kept below its dose limit. Unfortunately the recent change in the dose limit for the eye lens has meant that this no longer holds for all beta-ray energies and the possibility of further changes in the future has introduced some confusion. One approach suggested was to adjust the depth at which the body dose is measured so that the eye dose is still automatically controlled. Another suggestion was to fix the depth at 1000 mg/cm^2 and if beta-radiations are present at dose rates which could cause the eye lens dose limits to be exceeded then special procedures could be introduced. The skin dose could be controlled to the eye dose limits or the persons concerned could wear eye protectors or they could be issued with eye dosemeters. Further discussion is clearly necessary to reach an agreed approach to this problem.

The results of the intercomparison showed that most operational dosemeters were either unable to measure beta-ray doses from Pm-147 or gave rather poor results. This led to discussions on the importance of measuring doses from low energy emitters. The consensus was that improve-

ments in the dosimetry of Pm-147 were desirable but that dosimetry of lower energy emitters was unnecessary as the mean beta-ray energy of Pm-147, 0.07 MeV, only just exceeds the minimum beta-ray energy which would penetrate 7 mg/cm^2 of tissue.

To extend the energy range down to E_{max} = 0.2 MeV will require thinner detectors and the common view was that the development of such detectors should be encouraged. Several authors discussed on-going developments in this field with thermoluminescent detectors which clearly showed that it is possible to meet this objective.

To summarize, the view was that the intercomparison and the seminar had been worthwhile and successful. Most of the participants had learned something useful from the exercise. However only a limited number of participants had been invited to take part because this was intended as a pilot scheme and it was felt that the next phase of the intercomparison should be along the lines given in the paper by Professor Wagner and also given by the Editing Committee in the conclusion of section A 6.

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Track-autoradiography of a bacterium (Nitrobacter Winogradskyi) labelled with ^{14}C-acetate. The ^{14}C-acetate incorporated within the bacterium can be located at the track origin of the electron emitted by the tracer element.

Nuclear emulsion : Ilford K5 Exposure time : 4 days Amidol development.

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This document contains the original papers given at the occasion of this seminar with comments and conclusions worked out by the editing committee.

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