AAPM REPORT NO. 19

NEUTRON MEASUREMENTS AROUND HIGH ENERGY X-RAY RADIOTHERAPY MACHINES

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American Association of Physics in Medicine
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A REPORT OF TASK GROUP 27
RADIATION THERAPY COMMITTEE
AMERICAN ASSOCIATION OF
PHYSICISTS IN MEDICINE

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July 1986

Published for the
American Association of Physicists in Medicine
by the American Institute of Physics

Further copies of this report may be obtained from

Executive Officer
American Association of Physicists in Medicine
335 E. 45 Street
New York, NY 10017

Library of Congress Catalog Card Number: 86-73193
International Standard Book Number: 0-88318-518-0
International Standard Serial Number: 0271-7344

Published by the American Institute of Physics, Inc.,
335 East 45 Street, New York, New York 10017

Printed in the United States of America

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I. INTRODUCTION

Measurements of neutron fluence rates and corresponding absorbed dose and dose equivalent rates in high energy (greater than 10 MV) x-ray radiotherapy beams are especially difficult due to the large ratio of photons to neutrons and a lack of knowledge of neutron energy spectra. Photons interfere through photonuclear reactions in the detector and through pulse pile-up problems in detectors employing electronic pulse measurements. Responses of neutron detectors depend upon incident neutron energy and fluence-to-dose conversion factors vary strongly with neutron energy. These factors require a knowledge of neutron energy spectra which is very difficult to obtain. An exhaustive review of the physics of neutron production in high energy x-ray machines and measurement techniques, has been recently published by the National Council on Radiation Protection (NCRP)[1]. Despite the availability of this NCRP report and numerous scientific papers dealing with neutron measurements near high energy x-ray beams practical guidelines for such measurements are not easily available to clinical physicists. In response to this need, the American Association of Physicists in Medicine (AAPM) Science Council formed the Radiation Therapy Task Group No. 27 on "Neutron Measurements Around High Energy X-Ray Radiotherapy Machines" with the following specific objectives:

1. Recommend methods of measurement of neutron leakage around a high energy radiotherapy machine and provide a detailed step-by-step description of these methods.

2. Recommend instrumentation needed to perform these measurements.

3. Publish a summary of the recommendations.

The following report presents the recommendations of this task group in regard to these objectives. The first part of this report deals with neutron measurements in and near the primary photon beam, and the second part deals with measurements outside the treatment room.

II. MEASUREMENTS INSIDE THE TREATMENT ROOM

A. General Considerations

Inside the treatment room of a high energy x-ray radiotherapy machine neutron measurements prove to be very difficult because of unwanted interferences. Most electron accelerators are pulsed at a repetition rate of 100 to 400 pulses per second with a pulse duration of 1 to 10 microsecond. The photon leakage, although shielded to a large extent by the massive shielding in the treatment head, is still much more abundant than the neutron fluence, the fluence of high energy photons out of the primary photon beam being 10-100 times higher than that of the neutrons.[1] This intense photon pulse usually overwhelms

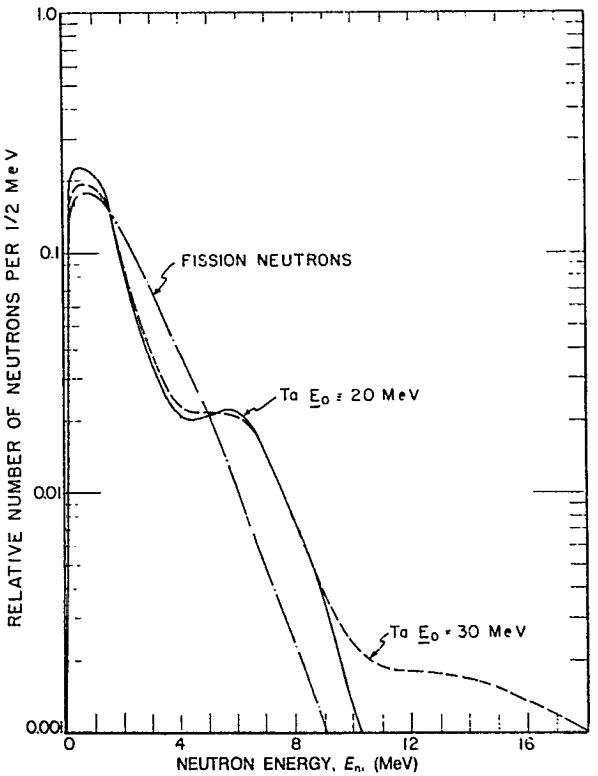

Fig. 1. Photoneutron spectra for tantalum with peak bremsstrahlung energies of 20 and 30 MeV. A fission neutron spectrum is shown for comparison.[1,2]

any active detector (a detector which detects particle events electronically) so that it only counts accelerator repetition rate. In the primary x-ray beam, of course, the relative photon fluence is much higher (1000-4000 times).[1] Therefore, it is recommended that only passive detectors, such as activation detectors, be employed for measurements inside the treatment room. It should be noted that it may be possible to adjust the accelerator to such a low output per pulse that active detectors can be used in the room. However, this condition is so far from the normal operating condition that there is no assurance that the ratio of neutron to useful photon fluence would be the same.

Activation detectors include thermal neutron detectors inside a moderator and bare threshold detectors. One type of interference that may be encountered during measurement in the primary photon beam is photoneutron production in the materials of the detector itself. This will be minimized by the use of a bare threshold detector. If a moderator-thermal neutron detector combination is used in the primary beam, the results must be corrected for the production of photoneutrons in the moderator (as described later). Outside the photon beam, photoneutron production in the detector may be neglected in most cases.

As will be apparent later, one of these methods (Moderated-Foil method) requires a determination of the average neutron energy at the point of measurement. For primary neutrons i.e. directly from a target without any further interaction, the neutron energy spectrum closely resembles a fission spectrum[2] as shown in Fig. 1. The average energy, E_{pri} for primary neutrons does not vary greatly with peak photon energy and values of 1.8, 2.1, 2.2, and 2.4 MeV are recommended for 15, 20, 25, and 30 MV x rays, respectively. As the neutrons penetrate through the shielding in the treatment head, a considerable degrading of the neutron energy spectrum follows. McCall, Jenkins, and Shore[3] have shown that an estimate of the average energy, E_{dir}, of primary neutrons which have traversed a layer of high atomic number shielding (lead, tungsten, or iron) can be obtained by using the half-energy-layer value, HEL, shown in Fig. 2. If the thickness travelled in the collimator shielding is x cm, and the half-energy-layer thickness is HEL cm, then E_{dir} is related to E_{pri} as follows

$$E_{dir} = E_{pri}(1/2)^{x/HEL} \qquad (1)$$

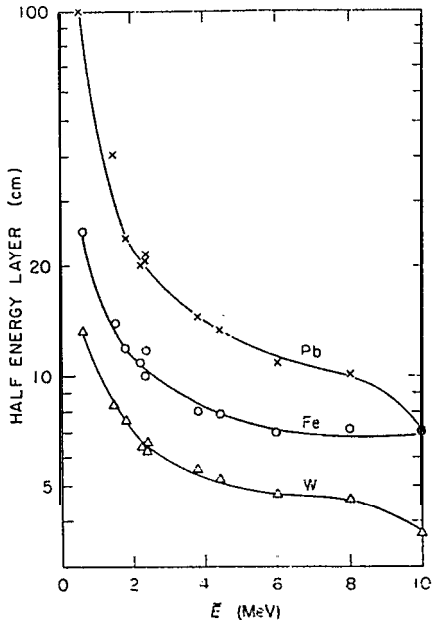

Fig. 2. The thickness of a spherical shell shield required to reduce the average energy of a neutron spectrum by one-half as a function of the unshielded average energy of the spectrum Data for iron, lead and tungsten are from McCall et al[3].

The value of x could be determined for a given direction by taking measurements from the manufacturer's drawings of the collimator shielding. A practical way to approximate x is to use the thickness and composition of collimator jaws and apply this value of x and associated HEL to neutrons emitted in all directions through the shielding. For these calculations, it is reasonable to assume that most of the neutrons originate from the x-ray target.

In addition to these direct neutrons, the neutrons scattered by the walls contribute varying amounts to the total neutron fluence. The average energy of these scattered neutrons excluding thermal neutrons is independent of room size and is given by the equation,[3]

$$E_{sc} = 0.24\ E_{dir} \qquad (2)$$

Finally, the combined total average energy for the direct and scattered neutrons i.e. the entire neutron spectrums at a point in the treatment room is given by the expression,[3]

$$E_{tot} = \frac{\frac{E_{dir}}{4\pi R^2} + \frac{5 \cdot 4\ E_{sc}}{S}}{\frac{1}{4\pi R^2} + \frac{5 \cdot 4}{S}} \qquad (3)$$

where R = distance in cm from the x-ray target, and
S = inside surface area of the treatment room in cm^2.

The maze area can be ignored for the purpose of calculating S. Using equation (2) and (3), the total average energy of neutrons can be expressed as

$$E_{tot} = E_{dir} \left[1 - \frac{16.4\ \pi R^2}{S + 21.6\ \pi R^2} \right] \qquad (4)$$

Knowing the total average energy of neutrons, the total dose equivalent rate at any point can be approximated from the fast neutron fluence rate by using the average conversion factors for neutron spectra, using the results of Monte Carlo simulations by McCall, Jenkins and Shore[3] (see Fig. 3a). These average conversion factors for different average neutron energies were obtained by using the shape of the neutron energy spectra generated by Monte Carlo code, MORSE[4] and the ICRP-21 fluence-to-dose equivalent conversion factors[5] for monoenergetic neutrons. This empirical relationship can be employed for conversion of fast neutron fluence rate to dose equivalent rate in a situation where the neutron energy spectrum or results of a Monte Carlo simulation are not available. For comparison, the fluence-to-dose equivalent conversion factors for monoenergetic neutrons from ICRP-21 are also shown in Fig. 3. Similar calculations for fluence-to-dose conversion factors for spectrum-averaged neutrons obtained from ICRP-21 values for monoenergetic neutrons are shown in figure 3b. The moderated-foil method, described later in this report, measures the fast neutron fluence rate which is then converted to a dose equivalent or dose rate using the procedure described above.

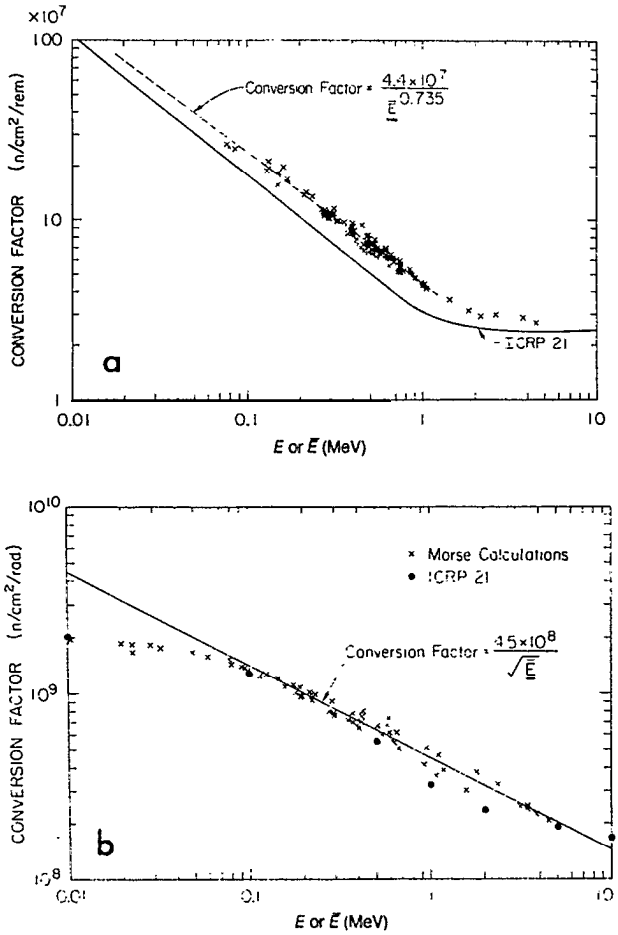

Fig. 3a. Fluence-to-dose equivalent conversion factors as a function of average neutron energy. The points are results of many different neutron spectra and shielding situations from McCall et al.[3]. The dashed line is a least squares fit to the points. Also shown is the fluence to dose equivalent conversion factor for monoenergetic neutrons as given in ICRP Publication 21[5].

Fig. 3b. Fluence-to-absorbed dose conversion factors as a function of average neutron energy.

The activation detectors can be conveniently supported on the accelerator couch for measurements in the patient-plane. A fixture may be used that positions the detectors accurately without the necessity of repeated measurement each time it is desired to position the detectors. If the maximum leakage dose equivalent in the patient plane outside of the treatment area is to be measured, the diameter of the circle of exclusion is set by the diagonal of the largest field possible. Thus, for a 35 cm x 35 cm maximum field size (diagonal of field equals 50 cm), edge of the closest detector would be positioned so that it is 25 cm from the isocenter. The irradiation is then carried out with the movable collimators closed. This measures the neutron leakage through the shielding, analagous to x-ray leakage measurements, which must be made with the collimators closed in order to avoid the inclusion of scattered radiation with the leakage radiation. This is *not* the neutron fluence that would be observed in that location during routine patient treatment when the collimators are open.

For the purpose of measuring neutron dose to the patient, the collimators should be opened to a specified field size in order to assess more accurately the neutron fluence in the patient plane accompanying an actual treatment. This task group recommends the use of a 20 x 20 cm^2 field for this measurement. The detector must be placed on the central-axis for in beam measurements and completely outside the useful radiation field for leakage measurements. The maximum fluence outside of the useful field can then be determined for the purpose of patient dose measurements.

Some regulations require measurements in the plane of the target, above the target and at other locations in the treatment room. In the latter case, it is more-or-less conventional to sample at isocenter height above the floor.

It is important that during activation measurements the exposure time be measured accurately and radiation dose be recorded, since both are necessary to reduce measurements to fluence per unit dose of x-rays at isocenter.

B. Phosphorus Activation Method

A bare activation detector of phosphorus may be used to measure both fast and thermal neutron fluences utilizing the reactions $^{31}P(n,p)^{31}Si$ and $^{31}P(n,\gamma)^{32}P$, respectively. The (n,p) reaction has a threshold of 0.7 MeV. The abundance of ^{31}P in natural phosphorus is 100%. ^{31}Si has a half life of 2.62 hours and emits 1.48 MeV beta rays (99%) and 1.26 MeV gammas (.07%). ^{32}P has

a half life of 14.28 days and emits 1.71 MeV beta rays (see table I). It has been shown[6] that all unwanted

Table 1. Properties of Activation Products

Reaction	Cross Section (barn)	Percent Abundance (%)	Product Half Life	Decay Radiation (MeV)	Branching Intensity
$^{115}_{(n,\gamma)}$ ^{116m}In	194	95.7	54 m	β^- 1.00	1.00
				γ 0.138 to 2.111	
^{197}Au (n,γ) ^{198}Au	99	100	2.698 d	β^- 0.962	0.99
				γ 0.412	0.99
^{31}P (n,p) ^{31}Si	*	100	2.62 h	β^- 1.48	0.99
				γ 1.26	0.07
^{31}P (n,p) ^{32}P	0.190	100	14.28 d	β^- 1.71	1.00

*varies with neutron energy, threshold energy is 0.7 MeV

activation products produced in a photon and neutron field are short-lived and are of no importance in the analysis of ^{31}Si and ^{32}P. The only other interference is due to neutrons produced in the phosphorus pentoxide itself which are then captured by ^{31}P nuclei. This interference has been shown to be small, being less than 5%[6]. The phosphorus technique has been used successfully in intense high energy photon beams by Price, Holeman, and Nath[6] and by Bading, Zeitz and Laughlin[7]. This method is now described in greater detail.

Step 1. Sample Preparation

In the original work described by Price et al[6] phosphorus pentoxide was obtained in its standard laboratory packaging as phosphorus pentoxide powder. Approximately 2.5 grams of material was placed in a 0.5

dram shell vial and capped. The shell vial was then positioned in the radiation beam and irradiated.

Bading et al[7] have shown that commercially prepared 85% orthophosphoric acid can be employed instead of the phosphorus pentoxide powder. They used 2 ml of the acid in 2 ml polypropylene cryotubes, as the irradiation activation detector.

Step 2. Irradiation

Many samples of phosphorus can be irradiated simultaneously because the sample size is small (2 to 3 ml in volume). This technique has the highest spatial resolution of all the methods presented in this report. For activation analysis, the samples must be irradiated at a constant dose rate. The samples should be irradiated to an approximate dose to water of 40 Gy (4,000 rad) for 15 MV x rays and 10 Gy (1,000 rad) for 25 MV x rays in order to obtain sufficient induced activity. The samples should be placed such that their long axis is normal to the central-axis of the primary x-ray beam in order to minimize neutron attenuation through the detector.

Step 3. Measurement of Induced Activity

Since the activation products are beta emitters, use of a liquid scintillation counter provides a practical means of determining induced activity with high counting efficiency (greater than 95%).

If phosphorous pentoxide powder is used[6], it must be carefully dissolved in distilled water so that the final ratio of powder to water is 0.32 gm of powder to 1 ml of water. This causes a violent reaction and must be done in a well-ventilated area, preferably within a hood. After the powder is dissolved, 3 ml of the solution is pipetted into a 20 ml liquid scintillation vial containing 15 ml of liquid scintillate fluid (Insta Gel., Packard Instruments Laboratory or an equivalent compound). The prepared sample is then placed in an automated and refrigerated liquid scintillation counter.

If the premixed orthophosphoric acid is used[7], 2 ml of irradiated orthophosphoric acid is pipetted into a 20 ml liquid scintillation vial after irradiation and mixed with 10 ml of liquid scintillation cocktail for aqueous compounds (Atomlight, NEN Corporation or an equivlaent compound). Cold scintillate should be used in sample preparation and refrigeration is necessary to insure a stable scintillate-sample mix.

The liquid scintillation vials containing the irradiated material are placed in a refrigerated and automated liquid scintillation counter and counting is initiated after a waiting period of at least one-half hour. A series of five to ten counts of two to minutes each are taken for the determination of ^{31}Si activity, which is followed by a 12-24 hour wafting period and 3-5 counts of IO-30 minutes each are taken to measure the ^{32}P activity. The usual precautions of background subtraction must be observed and are described in many references[8,9].

The measured count rate of ^{32}P and ^{31}Si must be corrected for their counting efficiencies, which can be determined by using standard sources of ^{32}P (1.71 MeV betas), ^{36}Cl (0.714 MeV beta) and ^{14}C (0.156 MeV betas), as described in Appendix I. The measured count rate is then converted to saturation count rate and activity per target nucleus using the procedure outlined in Appendix II.

Step 4. Calculation of Fluence Rates

It has been shown by Price, Holeman and Nath[6] that the fast neutron fluence rate in $n . cm^{-2} . s^{-1}$ is related to the saturation activity of ^{31}Si in Bq per target atom, $A_s(^{31}Si)$, as follows,

$$\phi_F = A_s(^{31}Si)\, I_\phi \qquad (5)$$

where I_ϕ is a ratio of two integrals which depends upon the relative shape of the neutron energy spectrum and the phosphorus activation cross section. This integral ratio has been calculated for a number of neutron spectra shown in references 6, 7 and 10. Since the average neutron energy at a point in the direct photon beam varies quite slowly with peak photon energy, the integral ratio is also relatively insensitive to peak photon energy for points in the primary photon beam. It has been found that a value of $1.5 \times 10^{25} cm^{-2}$ for the integral ratio can be used for photon peak energies in the range of 15 to 30 MeV[6]. However, the neutron spectrum degrades rapidly as neutrons pass through the collimator shielding resulting in a lowering of the average neutron energy. Therefore, for points out of the primary photon beam, the uncertainty in the determination of the integral ratio is larger using the phosphorus method.

The thermal neutron fluence rate in $n.cm^{-2}.s^{-1}$ is given by

$$\phi_{th} = A_s(^{32}P) / \sigma_{n\gamma} \quad (6)$$

where $A_s(^{32}P)$ is the saturation activity of ^{32}P in Bq per target atom and $\sigma_{n\gamma}$ is the thermal cross section[11] of ^{31}P which has a value of 190×10^{-27} cm^2.

Step 5. Determination of Dose Equivalent Rate and Dose Rate

For points within the primary photon beam, dose rate and dose equivalent rate are obtained from the fast neutron fluence rate by using a fluence-to-dose-equivalent conversion factor of 2.25×10^{-6} rem/min per $n.cm^{-2}s^{-1}$ and a fluence-to-dose conversion factor of 2.47×10^{-7} rad/min per $n.cm^{-2}s^{-1}$ (numerical values obtained from an update of reference 6).

It should be pointed out that the integral ratio reported above is for neutrons above the $^{31}P(n,p)^{31}Si$ reaction threshold of 0.7 MeV. For a 25 MV x-ray beam, the ratio of integrated intermediate energy neutrons (i.e. less than 0.7 MeV) to the fast neutrons (i.e. greater than 0.7 MeV) has been evaluated using Monte Carlo generated neutron spectra and found to be 0.30 for in-beam points. This leads to a ratio of corresponding dose equivalent rates which is of the order of 0.01. Thus, for in-beam measurements, the phosphorus technique described above yields a value of neutron dose equivalent which is 98 to 99% of the total neutron dose equivalent.

C. Moderated Foil Method

Based upon the work of Stephens and Smith[12], a practical method for measuring neutron dose equivalent rates using a thermal neutron activation foil in a moderator has been developed by McCall[3,13]. The moderator is a cylinder of polyethylene 15.2 cm (6 in) in diameter by 15.2 cm (6 in) in height covered with a 0.5 mm (0.020 in) thick cadmium foil or an equivalent layer of 25% borated silicone rubber, 3 mm (1/8 in) thick. The purpose of a moderator is to provide an energy independent thermal neutron fluence at the foil proportional to the incident fast neutron fluence, for energies up to a few MeV. The moderator can be used with any thermal neutron detector but activation foils of indium or gold are the materials of choice. The properties of these two foils are given in Table I. Each foil has some advantages and disadvantages as described below.

Step 1. Selection of Moderator-Foil System

Indium provides greater sensitivity than gold because of its large cross section and the shorter half life of the activation product. For a given neutron fluence delivered in a short time (minutes) the indium foil will yield about 100 times the activity of a comparable gold foil. Indium foils are considerably cheaper. On the other hand, the short half life (54 min) of ^{116m}In means that the counting equipment must be reasonably close to the measurement point. Indium is very soft but in the preferred 0.125 mm (5 mil) thickness it can be obtained mounted on aluminum support discs. The thermal capture product of aluminum is ^{28}Al, which does not interfere with the counting because the cross section is small and the half-life is only 2.24 min. The intense photon fluence will produce ^{26}Al and ^{26m}Al via (gamma, n) reactions, but because of the small cross section (about 16 mb) and long half-life (7 x 10^5 yr) for ^{26}Al production, and the short-life (6.34 s) of ^{26m}Al, the interference produced is negligible after a few minutes of waiting. Plastic support discs made of lucite, polystyrene etc. may lead to interference from induced activity of ^{11}C which emite 511 keV photons and has a half life of 20m.

Gold foils are quite expensive but 0.025 mm (1 mil) thickness is sufficient to provide reasonable mechanical strength. ^{198}Au produced by thermal neutron capture has a half life of 2.698 days which allows for substantial delays between irradiation and counting. It is therefore possible to send the moderator and foil to a calibration laboratory to calibrate one's counting system. A disadvantage of the longer half life is that the gold foils cannot be reused until the activity has decayed to an acceptably low level, which may take several weeks.

The resonance peaks for thermal neutron capture in In and Au are very similar in energy and magnitude, both peaking a few electron volts above the cadmium cut-off energy of approximately 0.4 eV.

Moderators can be made quite easily or purchased commercially.* Prepared gold and indium foils can also be purchased from a commercial vendor*. Foils of indium and gold can also be purchased and cut or punched to a desired size. The foils can be identified easily by fastening them to a sheet of paper with post-it note tape or other lightly

*Reactor Experiments, San Carlos, CA 94070 or any other supplier of high purity foils and moderators, for example, Indium Corp of America.

sticking tape and numbering them on a typewriter. The easiest way to identify foils is to simply write on them with any 'permanent' marker pen. Several colors easily differentiate runs and the writing can be removed by alcohol swabs. Gold is soft enough that the numbers can be embossed and the carbon can be cleaned off with alcohol. Stamp collector's glassine envelopes are convenient holders for the foils.

Step 2. Counting and Calibration

A counting system must be calibrated to measure the induced activity in the foil. Either gold or indium foil can be counted for beta rays with a thin window GM or a proportional counter. Either foil can also be counted with a scintillation or GeLi detector detecting the gamma rays. If a NaI scintillation counter is used, care must be taken to set a discrimination window around 412 keV gammas from ^{198}Au, in order to avoid counting the gamma rays from ^{196}Au. This interference is produced by the (gamma, n) reaction in gold. If these precautions are taken, this interference from ^{196}Au can be reduced to less than 2% for a 25 MV x-ray beam. A preferred method is to use an intrinsic germanium detector with a multi-channel pulse height analyzer which can easily discriminate against the unwanted gamma rays from ^{196}Au.

The counting equipment is all commercially available from many sources. One should build a light-weight holder with shelves so that foils can be reproducibly positioned at several distances near the detector. Normally, the detector should be shielded with lead. This not only reduces the background but prevents disturbance by radioactive sources being moved in nearby areas. Care must be taken to ensure that there is no significant resolving time loss.

One of the simplest detectors is a thin-window Geiger tube as used for measuring low-activity beta ray sources. An example of such a counting setup is a 1 inch diameter tube with an end-window of 1.4 mg/cm^2 thickness, coupled to a Ludlum Model 220 Portable Scaler Rate Meter (Ludlum Measurements, Inc., Sweetwater, Texas). In a lead shield 5 cm thick, the unit has a background of about 8 cpm and a negligible coincidence loss rate (dead time correction) at 20,000 cpm. The foils are normally counted on a shelf close to the window, but if necessary the rate can be decreased by about a factor of 10 by using the lowest shelf position. The counting efficiency of a thin window Geiger counter for ^{196}Au is only about 1/6th of that for ^{198}Au, so the interference from ^{196}Au is minimal.

The count rate C during counting time of duration t_c, is determined after a wafting time tw. The saturation count rate, C_s, is obtained by

$$\dot{C}_s = \frac{\dot{C}\ \lambda t_c\ \exp(\lambda t_w)}{[1-\exp(-\lambda t_1)][1-\exp(-\lambda t_c)]} \quad (7)$$

Details of this procedure are described in Appendix II. Note that if the irradiation time t_i and the counting time t_c are short compared to the half life of the isotope (as will usually be the case for gold), then the exponentials in the denominator of this expression can be approximated to give

$$\dot{C}_s = \frac{\dot{C}\ \exp(\lambda tw)}{\lambda t_1} \quad (8)$$

The saturation activity is then given by

$$A_s = \frac{\dot{C}_s}{e\ N\ f_d} \quad (9)$$

where the constants N and f_d are described in Appendix II. The purpose of a calibration procedure is to obtain the counting efficiency e so that the system can be used to measure a fast neutron fluence rate. Knowing the absolute saturation activity, the fast neutron fluence rate can be determined by the expression,

$$\phi_F = f_\phi \cdot A_s \quad (10)$$

where

$$f_\phi = 7.9 \times 10^{21}\ \text{n.cm}^{-2}.\ \text{s}^{-1}.\ \text{B q}^{-1}\ \text{for gold.} \quad (11)$$

This conversion factor has been obtained by McCall measuring the absolute saturation activity produced by a known fast neutron fluence rate. In practice this procedure is very difficult because of the need to determine absolute counting efficiency and activity. For this reason, it is not advised unless the user is well-versed in such measurements. Two alternatives to this procedure are presented below.

First, if one has a calibrated neutron source, one can use it to calibrate the detector. A ^{252}Cf neutron source of suitable strength (2 x 10^7 neutron/sec, minimum) may be used for this purpose. Other sources such as PuBe, have

neutron spectra which do not match accelerator neutron spectra. This calibration, however, is an involved measurement which requires many corrections including the following:

1. Scattering (Jenkins[14], Eisenhauer & Schwartz[15].
2. Anisotropy of neutron emission.
3. Decay correction for the source.

One cannot hope to obtain high accuracy in such measurements without a great deal of care, since these corrections can be large. The first two, for example, may each amount to 30% or more and are in the same direction. The systematic errors, including that of the neutron source output itself, may well be on the order of ± 25%.

The recommended alternative for calibration is to send a moderator and a gold foil (indium has too short a half life) to a facility that can expose the combination to a known neutron fluence or dose equivalent. Two such calibration laboratories are National Bureau of Standards (NBS)*** and the SEFOR Calibration Center**. The exposure necessary for the gold foil described above, is approximately 10^7 neutrons/cm^2. Note that a gold foil giving even 10^3 cpm contains only about 16 Bq (2 nCi) of ^{198}Au, so there is no problem in sending it through the mail.

This procedure provides an overall calibration of the detector system directly in terms of a known fast neutron fluence rate and is less likely to result in errors. If indium is to be used, then it should be calibrated by the user in terms of the gold foil which has been already calibrated.

The calibration laboratory will state the fast neutron fluence I_F, the irradiation time t_i and the time of the end of the irradiation, The user must measure the saturation count rate C_s. One may then calculate a simple system calibration factor, C_F, using

$$I_F = \dot{C}_S \cdot C_F \qquad (12)$$

**Southwest Experimental Fast Oxide Reactor, SEFOR, Mechanical Eng. Dept., University of Arkansas, Fayetteville, AR 72701 Attn: Prof. Leon West or Cecil Cogburne.

***Radiation Dosimetry Section, National Bureau of Standards, Washington, D.C., Attn. Robert Schwartz.

or carry out the somewhat more elaborate analysis that results in the counting efficiency of the detector system, e.

Once a calibration has been obtained, a long lived radioactive source should be used as a consistency check to monitor the sensitivity of the counting system whenever it is used. If one is counting beta rays, a ^{36}Cl source (half life of 3 x 10^5 y) is a good choice. The energy of the prominent gamma ray from ^{116m}In is 1.293 MeV, hence either ^{22}Na with a gamma ray at 1.27 MeV or ^{60}Co with gamma rays at 1.17 and 1.33 MeV could be used for checking gamma ray counting systems. For ^{198}Au, the energy of the prominent gamma ray is 0.412 MeV, so one could use ^{113}Sn with a gamma ray at 0.392 MeV. All of these calibration nuclides have half lives long enough to make them useful, although their activities must be corrected for decay.

Step 3. Irradiation

A moderator-foil system is placed with its center at the point to be measured. If several moderators are available, these can be exposed at the same time provided they are not too close to each other (seperation between adjacent moderators should be at least 30 cm which is about twice the diameter of the moderator). The required photon dose at the isocenter to give a reasonable count rate from the foil is a function of all the parameters of the detection and counting system plus the energy of the accelerator and the room size. For typical detection and counting systems described below, one needs about 40 or 10 Gy (4000 or 1000 rad) of photons at the isocenter for 15 MeV and 25 MeV accelerators, respectively. These exposures give a convenient count rate of about 300-600 cpm with gold at points within 2 meters of the isocenter.

Note that when one makes measurements in a room, one will need to know several factors in order to analyze data. These factors are the distance from the x-ray target to the foil, R, the surface area of the inside of the room, S, times at which the exposures began and ended for each foil (in order to calculate t_i and t_w), and the composition and thickness, x, of the variable collimators.

If measurements in the primary photon beam are to be performed, care must be taken to ensure that the moderator is well inside the radiation field. Because of the large size of the moderator, field size at the moderator must not be smaller than 20 x 20 cm for this requirement.

Step 4. Correction for In-Beam Measurements

If this detector is used in the primary photon beam, a correction for the production of photoneutrons in the moderator-foil system must be made. McCall, Jenkins and Tochilin[16] have calculated the apparent neutron fluence produced by a given photon dose delivered to the moderator-foil system. Results of these simulations are shown in Fig. 4. Measurements have been also made at three photon energies to test these Monte Carlo calculations. It can be seen from Fig. 4 that for a 25 MV x-ray beam, photons produce an apparent neutron fluence in the detector itself of 1.1×10^{5} n/cm^{2}/rad. This could be a correction of 18 to 30% depending upon the particular model of accelerator producing 25 MV x rays. For 18 MV x rays, this correction is about half as much as that for 25 MV x rays. This correction can be reduced greatly by using boron (borated silicone rubber) in place of cadmium foil. A quantitative estimate of this correction for boron-covered detectors is not yet available, but the correction is likely to be small for x-ray energies below 18 MV. The correction can be estimated by using the carbon curve in Fig. 4. Since this correction is large (about 30%) for cadmium-covered detectors when used in higher energy photon beams, this method is not the preferred method for use in beam with collimators open, for photon energies above 20 MV.

Step 5. Determination of Dose Equivalent Rate

Conversion of neutron fluence rates to dose equivalent rates depends on the average total energy of the neutrons, which may be obtained by calculation as described in section I.A. Fluence-to-dose equivalent conversion factors are obtained from the curve shown in Fig. 3 as described in section I.A. These conversion factors were obtained by using ICRP-21 factors and Monte Carlo generated neutron spectra. These factors are valid for neutron spectra only. For monoenergetic neutrons, ICRP-21 values of the fluence-to-dose equivalent conversion factors are appropriate. Fluence-to-absorbed dose conversion factors for neutron spectra as well as for monoenergetic neutrons are shown in figure 3.

D. Activation Rem Meter Method

The Anderson-Braun rem meter with a BF_3 tube in the original version[17] yielded a response vs neutron energy curve that was proportional to the dose equivalent vs neutron energy curve within about ± 30% between thermal energies and 10 MeV, except for the region around 1 to 10 keV where the response was high by up to 60%[18]. The

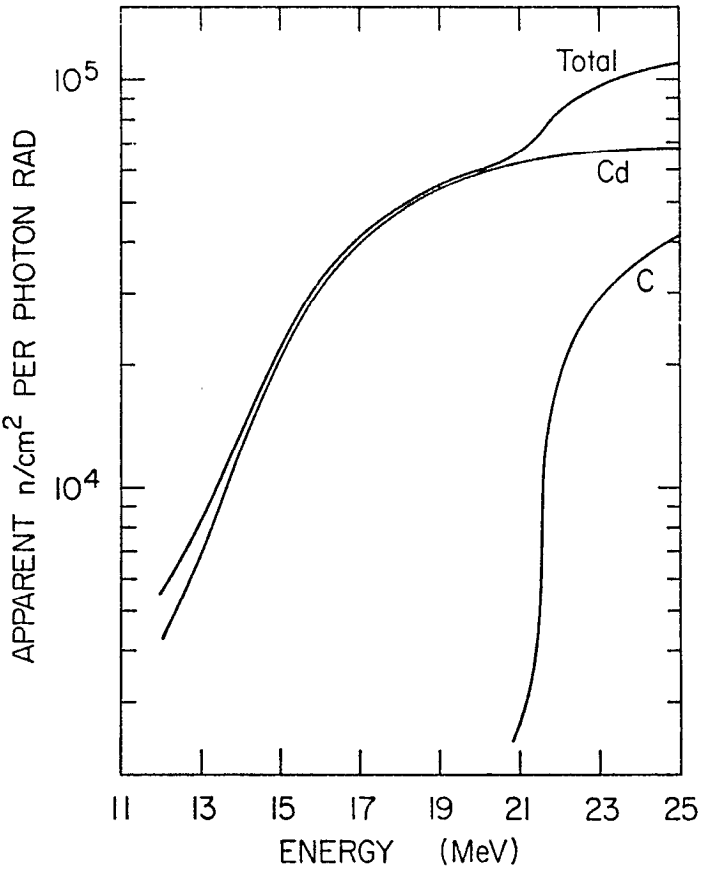

Fig 4. Apparent neutron fluence per photon rad as a function of peak photon energy for the Cd-shielded moderated-foil. The curves labeled C, Cd and total represent the values for neutrons from the carbon in the moderator, cadimium and the total, respectively.

range of neutron energies around a medical accelerator is easily covered by this instrument, but in-room use will lead to erroneous results (except perhaps in the entrance maze) because the BF_3 tube is simply overwhelmed by the pulse rates due to x-rays. The detection system tends to lock-on to the accelerator pulse repetition rate, yielding entirely spurious results. Replacement of the BF_3 tube with one not enriched in ^{10}B and substitution of fast scaling circuitry for the stock rate meter does not ameliorate the situation significantly. One answer is to use a passive thermal neutron detector such as an activation foil in the Andersson-Braun moderator as developed by Rogers and Van Dyk[19].

The most important advantage of the rem meter is that no knowledge of the neutron spectrum is required. Data reduction is straightforward and simple. The disadvantages are larger size and weight, larger angular dependence and the relatively high cost. In spite of the multitude of errors and other shortcomings attributed to the Andersson-Braun rem meter, it has been in use for almost 20 years and, with a minimum of caveats, meaningful results have been obtained under a wide range of conditions. With a foil detector it can be used in the treatment room. However, for open collimator measurements in a primary beam above 16 MV, a correction would have to be applied for the overresponse due to photoneutron production in the moderator and this has not been evaluated quantitatively. Under closed collimators the rem meter can be used "in-beam" at any medical accelerator energy because the intensity and average energy of high energy photons through the collimator shielding is significantly degraded. At 25 MV for example, it has been calculated that some 5-6% of the on-axis photon **fluence** is above 15 MeV, whereas under closed collimators virtually no photons with energy greater than 15 MeV are present at all.

Step 1. Selection of Foil

A moderated gold foil method has been used successfully by Rogers and Van Dyk[19]. Their commercial Anderson-Braun rem meter was slightly modified to accept a gold foil (2.2 cm in diameter, 0.13 mm thick with an approximate weight of 1 g) placed in the center of the cavity provided for the BF_3 tube.

The function of the moderator is to produce an energy response that Is proportional to the dose equivalent index curve. As might be surmised, the design is complex and therefore the Anderson-Braun moderator, or one of the spherical designs available, must be purchased. Some of

the moderated neutrons are captured by the gold foil and produce the radioisotope ^{198}Au which has been described in section II.C. An alternative is to use indium foil. The advantages and disadvantages of the two foils have been already described in section II.C. The response per unit neutron fluence in this type of moderator is smaller by a factor of 10 to 30 compared to the detector of the previous section which makes use of indium foil preferable.

The BF_3 tube is easily removed from the moderator, and the foil positioned reproducibly in about the center of the cavity by means of a simple plastic fixture.

Step 2 Irradiation

The moderator is placed in the desired location, preferably side-on to the accelerator and irradiated. The procedures for irradiation are essentially the same as in the moderated-foil method described in previous section, except that the lower response of the rem meter requires a longer exposure. A typical irradiation schedule for indium ranges from 4 to 20 m, depending on distance of moderator from isocenter, at a dose rate of 4 to 5 Gy/m for a 25 MV machine. For a 15 MV x-ray beam, the exposure must be approximately four times longer.

Step 3. Counting and Calibration

After exposure, the induced activity in the foil is assessed using methods identical to those described previously in section II.C. As before perhaps the simplest method is to use a thin window GM tube as used for measuring low activity beta ray sources. For details, see the previous section II.C.

Measured count may be converted to saturation count rate and activity as described in Appendix II. Dose equivalent rate is then simply related to the saturation activity by a constant factor, as shown below,

$$DE = f_{DE} A_s \qquad (13)$$

where A_s is the saturation activity in Bq per target nucleus

f_{DE} is 4.07×10^{23} mrem/(Bq per target nucleus) using gold as the activation detector, as determined by Rogers and Van Dyk[19]. They quote a value of 140 mrem / (photon / s per gm). This factor is for the 412 keV photons (95% emission per decay) from ^{198}Au and applies to a gold foil thickness-of 60 mg/cm^2 (0.13 mm or 0.005" thick).

As mentioned before in section II.C, it is quite difficult in practice to measure absolute activity and the preferred method is to calibrate the whole system by exposing the detector with gold foil to a known dose equivalent at NBS*** or SEFOR**. If it is necessary to use an indium foil, it should be calibrated relative to a gold foil in the user's beam.

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***Radiation Dosimetry Section, National Bureau of Standards, Washington, D.C., Attn. Robert Schwartz.

III. MEASUREMENTS OUTSIDE THE TREATMENT ROOM

Outside the treatment room both the neutron and photon fluence rates are considerably lower and the neutron pulse is spread out over several hundred microseconds. Therefore, electronic counting detectors such as BF_3 counters, LiI scintillation detectors, etc., can be successfully employed for neutron measurements. The neutron energy spectrum outside the treatment room resembles that of a heavily shielded fission source, and the average neutron energy is markedly lower than that inside the room. For these reasons, neutron detectors outside the treatment room must have higher sensitivity, especially for lower energy neutrons (in the range of 100's of keV and lower). A number of rem meters are available commercially.[+] These employ an electronic thermal neutron detector inside a moderator, and are designed to give a direct reading of dose equivalent in rem (Sv). All of these instruments respond well to fast neutrons in the range of MeV's but have poorer sensitivity to lower energy neutrons. Outside a properly shielded treatment room, most of the neutrons are less than 0.5 MeV in energy, and the response of these rem meters is not as accurate but is adequate[20]. An alternative to the rem meter is to employ a Bonner spectrometer which is sensitive to the entire energy range. However, this multi-sphere spectrometer system is a complex instrument to use, requires extensive data analysis, and can give misleading results if not properly used.

Recently, neutron badges have become available from a commercial vendor (R.J. Landauer Company) who provides three types of neutron badges. Neutrak is a polycarbonate film track-etch dosimeter for fast neutrons. The manufacturer claims a minimum reportable dose equivalent of 30 mrem (0.3 mSv), a neutron energy threshold of 1 MeV, and a flat energy response from 3 to 14 MeV. The high neutron energy threshold makes this detector virtually useless for

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[+]For example, Eberline, 505-471-3232, BF-3 moderated survey instrument, models PRS-2P/NRD and PNR-4/NRD; Ludlum, 617-826-9118, Neutron Spectrometer, models 42-5 and 16; Nuclear Enterprises, Neutron Dose Equivalent Meter, model NM2; Technical Associates, 213-883-7043, REM-PUG Rem Monitor; Victoreen, 216-795-8200, SNOOPY, Portable Neutron Remmeter, model 478, and other equivalent instruments.

neutron measurements outside of medical accelerator treatment rooms. Neutrak 144 badge is a more sensitive carbonate, CR-39, which is claimed to be sensitive to neutron energies as low as 144 keV. An extended range neutron monitor, Neutrak ER, which is a combination of the Neutrak 144 and an albedo dosimeter, is also available. Minimum detectable dose is claimed to be 10 mrem (0.1 mSv) and the minimum energy threshold is less than 144 keV. The Neutrak and Neutrak ER may be of use in the near future, but the task group does not recommend its use now as a sole method of neutron measurements because this detector has as yet not been evaluated by the scientific community.

We point out however that the ordinary x-ray film badge serves well as an indicator of the presence neutrons, since gamma doses are always associated with neutrons. Thus a zero or low-reading x-ray film badge is a reliable indicator of the absence of any significant neutron dose.

IV. SUMMARY OF RECOMMENDATIONS

The recommendations of the task group are summarized below:

a. In the primary photon beam with peak photon energy

(i) below 20 MV, with collimators open
moderated-foil method (section IIC)
or phosphorus activation method (section IIB)

(ii) below 20 MV, with collimators closed
moderated-foil method (section IIC)

(iii) above 20 MV, with collimators open
phosphorus activation method (section IIB)

(iv) above 20 MV, with collimators closed
moderated-foil method (section IIC)

b. Out of the primary photon beam, but inside treatment room

moderated-foil method (section IIC)
or activation rem meter method (section IID)

C. Outside the treatment room

commercial rem meters (section III)

V. DISCUSSION

The phosphorous activation method is most suitable for neutron measurements in the primary photon beam because of its low sensitivity to high energy photons and small size. The small size of this detector allows a determination of neutron profiles across the treatment field as well as neutron dose equivalent rate per photon rad as a function of field size[21,7]. Phosphorus method is not recommended for the closed collimator or for very small field sizes which may attenuate the primary neutrons and hence alter their average energy and detector response. As one increases the field size from 5 x 5 to 40 x 40 cm^2, the neutron dose equivalent per photon rad at the isocenter has been shown to increase by a factor of two for a 25 MeV accelerator[21]. It may be noted that this does not necessarily mean that neutron production in the treatment head (the neutron source strength) increases as the field size increases. In fact, the neutron source strength is likely to remain constant as collimators are opened[7]. The increase in measured neutron dose equivalent per photon rad at isocenter with field size may arise from neutron transport from the source to the detector. The moderated-foil detector has a diameter of 15 cm and cannot be used accurately for fields smaller than 20 x 20 cm^2, and the activation-rem meter employs a moderator which has an even larger diameter of 22 cm. In addition to their larger size, the moderated-foil detector and activation rem meter method both suffer from significant overresponse in the primary photon beam above 20 MV because of photoneutrons generated in the detector-moderator system itself. For the moderated-foil method, this correction has been evaluated quantitatively by McCall et al[16] and is approximately 18-30% for a 25 MV x-ray beam and about half as much for an 18 MV x-ray beam. This correction can be reduced by using boron in place of cadmium, but a quantitative estimate of this correction for a boron-covered detector is not presently available. This correction is even larger for the activation rem meter and its magnitude has not been determined systematically. For these reasons, it is recommended that of these methods for neutron measurements in the primary x-ray beam with peak energies greater than 20 MV, phosphorus activation is the most reliable. When the collimators are closed, the moderated-foil method does not suffer from this problem.

For measurements inside the treatment room, but outside the primary photon beam, the phosphorus method is not as reliable for the following reasons. As described in previous sections, the neutron energy spectrum changes rapidly as one moves away from the primary photon beam, which necessitates the use of different conversion factors

in the phosphorous method. It has been shown that the integral ratio changes by almost a factor of 1.5 as one moves a distance of 1 m from the isocenter. Since these ratios depend on the neutron energy spectrum, and these spectra can only be obtained by extensive experiments[22] or by Monte Carlo simulations[23], the phosphorus method is not easy to use outside the primary photon beam. It also suffers from a lack of sensitivity.

The outstanding advantage of the activation rem meter is that it measures dose equivalent directly and a knowledge of the neutron energy spectrum is not necessary. However, some new regulations[24] require that neutron leakage be expressed in terms of dose in rad (Gy), and it is not easy to obtain an accurate neutron dose measurement using the rem meter. However, a reasonable estimate of neutron dose can be obtained by dividing the measured dose equivalent by a quality factor of ten. Thus, the preferred methods for leakage measurements outside the primary beam are the moderated-foil method and the activation rem meter.

Neutron measurements outside the treatment room can be performed adequately by using any of the commercially available neutron rem meters. Because neutron track films have not been fully evaluated by the scientific community, their use for monitoring neutron doses outside the room in not recommended.

These recomendations do not imply that the numerous other techniques for neutron measurements are not accurate or practical. This task group has chosen to describe the practical details of the methods which members of the task group have had successful experience with. We hope that a clinical physicist will be able to use this manual to perform reliable neutron measurements around his high energy x-ray machine.

APPENDIX I: CALIBRATION OF THE LIQUID SCINTILLATION SPECTROMETER

There are several important considerations in the use of a liquid scintillation counter for he determination of the activity of an irradiated sample[8,9]. These considerations are:

(a) Sample-Scintillate Mix

The irradiated sample must be readily dispersible in the scintillation fluor. The sample-scintillation mix should provide a homogeneous and stable mix which does not separate into layers with time. The miscibility of an aqueous sample-scintillate mixture is dependent upon the sample volume, scintillate volume, and temperature of the mix itself. If the sample is aqueous then an emulsifying cocktail such as Instagel (Packard-United Technologies), Biofluor, Aquasol-2, Aquasure (New England Nuclear Corp.), Ready-solv Hp and Gp (Beckman Instruments) should be used. Typically, at temperatures below 10°C a clear solution results if the water in the total sample plus scintillation cocktail mix is less than 10-12%. In the region of 12-17% sample-water mix a two phased mixture results and such mixtures should not be counted. For mixtures containing more than 18% water a gel results. It is in this gel configuration that samples containing radioactive materials may be counted without variable sample counting geometry becoming a problem.

In the analysis of phosphorous pentoxide solutions with. a concentration of phosphorus pentoxide in water equal to 0.32 gm/ml, 3 ml of sample may be added to 15 ml of Instagel, forming a gel at 4°C. Other types of solutions and concentrations should be experimented with to obtain optimum sample in a given volume of scintillate. Very important also is the maintenance of sample clarity with a given sample-scintillate mixture.

(b) Sample-Scintillate Clarity

The counting efficiency of the sample-scintillate is very dependent upon the opacity of the mix. The more turbid or cloudy the mix is, the lower will be the counting efficiency. One often has to trade off between efficiency and the amount of sample which may be added to a given scintillate. The underlying factor is the minimum activity which can be detected in a count time of 1 to 5 minutes. Because the analysis usually involves a decay curve, it is desirable to have a mix which is clear and high in counting efficiency. This allows one to maximize the counts observed in successive l-5 minute counting intervals.

(c) Liquid Scintillation Counter Calibration

A calibration of the liquid scintillation counting system involves the determination of the counting efficiency for the radiation of interest for a specific sample-scintillate mix. The most direct method by which this may be accomplished is the use of a calibration standard nearly identical to the expected activation product. A list of readily available beta calibration solutions is given below:

Table II. Beta Calibration Sources and Their Properties

Isotope	Max. Energy (MeV)	Avg. Energy (MeV)	Half Life
^{63}Ni	0.066	0.0172	100.1 y
^{14}C	0.156	0.0467	5730 y
^{35}S	0.167	0.0488	87.39 d
^{45}Ca	0.258	0.0772	165.1 d
^{36}Cl	0.714	0.312	308.000 y
^{32}P	1.710	0.694	14.28 d

A 0.050 ml (50 lambda) aliquots of each selected beta reference standard should be placed into an equal number of unirradiated sample volumes contained within 22 ml liquid scintillation counting vials. The appropriate amount of liquid scintillation fluid should then be added to each sample. The resulting mixture should be shaken vigorously and placed in the liquid scintillation counter for counting. The counting efficiency as a function of average beta ray energy may then be determined.

A sealed liquid scintillation standard (generally provided with the liquid scintillation counter) should also be counted along with the spiked samples as a future reference for system performance.

APPENDIX II. DETERMINATION OF SATURATION COUNT RATE AND ACTIVITY

When an activation detector is irradiated, the amount of induced activity depends upon the number of target atoms, neutron fluence rate, activation cross section, irradiation time, and decay of induced activity. In order to correct for the irradiation time, wafting time and decay of induced activity, it is customary to use the saturation count rate which includes all of the appropriate time parameter corrections. Simply stated, saturation count rate is the maximum count rate which can be induced in the sample by the given neutron fluence rate. The saturation count rate, $\dot{C}_s$, in cps is related to the measured count rate, $\dot{C}$, during a counting period by the equation

$$\dot{C}_s = \frac{\dot{C}(\lambda t_c)\exp(\lambda t_w)}{(1-\exp(-\lambda t_i))(1-\exp(-\lambda t_c))} \qquad (II-1)$$

where λ is the decay constant for the activation product under consideration, t_w is the wafting time from the end of the irradiation to the beginning of the counting, t_c is the counting time, and t_i is the irradiation time. The saturation count rate can be related to saturation activity per target nucleus using the following equation,

$$A_s = \frac{\dot{C}_s}{e\ N\ f_d} \qquad (II-2)$$

where e is the beta or gamma counting efficiency in cps per emitted particle/s

N is the number of target nuclei in the sample and

f_d is the branching intensity, i.e. the number of particles emitted per decay.

The number of target atoms N in a counting sample containing m gm of activation material is given by

$$N = 6.023 \times 10^{23} mna/A \qquad (II-3)$$

where 6.023×10^{23} is Avagadro's number,

A is the molecular weight of the activation material

n is the number of target atoms in one molecule of the activation material

a is the fractional natural abundance of the target element.

A specific example for phosphorous pentoxide powder follows: As described in section II.B, 3 ml of P_2O_5 solution is taken in a counting sample. Provided the ratio of 0.32 g per ml is maintained, this results in a mass of 0.96 gm of P_2O_5 powder in the sample. Using the values, m = 0.96 gm, n = 2, a = 1.0 and A = 141.94 in equation (II-3), one obtains N = 8.15 x 10^{21} atoms of phosphorus in this counting sample.

Another specific example for an indium foil follows: For an indium foil with a mass of 1 gm, the number of target atoms of ^{115}In would be determined by setting a = 0.957, A = 114.82, m = 1.0 and n = 1, leading to a value of 5.02 x 10^{21} atoms of ^{115}In.

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