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PROGRESS REPORT

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THE UNITED STATES ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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BIOLOGICAL AND CLINICAL DOSIMETRY

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BIOLOGICAL AND CLINICAL DOSIMETRY

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BIOLOGICAL AND CLINICAL DOSIMETRY

Abstract

The Sloan-Kettering portable tissue equivalent calorimeter system was employed for a series of measurements in two neutron fields generated at our cyclotron which compared specific charge, measured with a tissue equivalent ionization chamber, to absorbed dose measured with the calorimeter in the same water phantom. The ratio measured for ionization chamber response in a ^{60}Co gamma-ray field to the response of the same chamber in the neutron fields was found to be 1.08 and 1.06 for fields whose average neutron energies were about 3.5 and 6 MeV respectively. This is a reflection of the dependence of $\bar{W}_n$ upon neutron energy, and the values are compared with recent calculations. Initial measurements of the thermal defect in tissue equivalent plastic have established the precision and accuracy of the experimental system. Preliminary indications confirmed our estimates of the thermal properties of the calorimeter. Microdosimetric measurements were carried out in fast neutron, and other fields including those produced by ^{125}I and soft x-ray sources. A G-M dosimeter was developed and employed to measure the gamma-ray portion of dose in a fast neutron field, and preliminary work was started on development of a compact spherical calorimeter.

BIOLOGICAL AND CLINICAL DOSIMETRY

INVESTIGATORS TIME & EFFORT ON PROJECT

The principal investigator has devoted approximately 15% of his time and the co-principal investigator approximately 80% of his time to the projects described herein. These percentages are not expected to change during the remainder of the current term.

I. SUMMARY

Our research performed during this last contract year was focused on solving some of the remaining problems which limited the absolute accuracy of our calorimetric dosimetry systems. In addition, we employed the tissue equivalent (TE) calorimeter for a series of measurements at our cyclotron and microdosimetric measurements were extended to low LET fields.

The primary accomplishments of the past year have been:

- (1) the calorimetric measurement of $\rho \cdot \bar{W}_n$ (the product of the stopping power ratio for secondary charged particles between the wall and gas of a tissue equivalent ionization chamber, and the average energy expended in the gas per ion pair collected) for two spectra of fast neutrons,
- (2) the final tests of the thermal defect calorimetry system at Brookhaven National Laboratory, (3) microdosimetric measurements in low LET fields and (4) the design of a G-M dosimeter and a simple spherical TE calorimeter.

II. CALORIMETRIC MEASUREMENT OF $\rho \cdot \bar{W}_n$

A. Ionization Chamber

The ionization chamber described in our previous annual report (1) with its surrounding phantom was placed in a 30 cm cubic water-filled lucite tank so that the geometry was nearly identical to the calorimeter. This is indicated schematically in Figure 1. In order to insure that temperature and pressure corrections applied to the measured charge were stable and relatively insensitive to temperature gradients, gas adsorption on internal surfaces, or insulator polarization, the chamber was allowed to equilibrate within the water tank with high voltage on and gas flowing for periods of from one to two hours.

Charge was collected using a Keithley Model 640 vibrating capacitor electrometer with a polystyrene standard capacitor in the feedback loop. The potential across the capacitor was monitored with a digital voltmeter. Constancy was checked with a circuit consisting of a polystyrene standard capacitor charged using a mercury battery, which provided a reproducible source of charge. Both capacitance values were measured with a General Radio Model 1608-A impedance bridge, and a General Radio 1000 pF standard air capacitor.

Collection efficiency at the chamber potentials normally employed (± 400 V) was determined to be 99.3% in the fast neutron fields and 99.85% in the ^{60}Co field. Dose rates in the two fields were approximately 30 rad/min and 200 rad/min respectively, and the charges which were measured using positive and negative collecting potentials, differed by approximately 0.6%. The average of both potentials was employed for these measurements.

FIGURE 1

B. Radiation Fields

Fast neutrons were generated at the Sloan-Kettering cyclotron by bombarding a thick beryllium target with either 8 MeV deuterons or 23 MeV ^3He ions. The neutron spectra resulting from these interactions are broad and continuous with mean energies of approximately 3.5 and 6 MeV respectively (2). The target assembly has been described previously (3) and it included an electrical connection to the beryllium target along with a provision for electrostatic suppression of secondary electrons which are produced by the incident charged particles. Target current was measured using a Brookhaven Instrument Corporation current digitizer and total charge, accumulated on a scaler, was employed for beam monitoring. An electro-mechanical timer was employed for monitoring the photon exposures which were carried out using an AECL Theratron-80 irradiator at Memorial Hospital.

C. Results

In the ^{60}Co field and in each neutron field, a set of calorimeter measurements was taken followed by electrical calibration and then a set of ionization chamber measurements. This procedure was carried out in the same order, and required approximately one working day at each source. An example of the output of a calorimeter run obtained in the ^3He induced neutron field is shown in Figure 2. The relative displacement due to irradiation (or calibration) heating $\Delta R/R_0$ is computed using the displacement of a known resistance change. In this case a one ohm change in the bridge balancing resistance is employed. When the radiation is turned on, there is a slight overshoot which is caused by irradiation of bridge components. Since this rise is compensated by an equivalent drop, which can be discerned at the turn-off point, the effect is ignored in data analysis. More recent experiments have shown that the bridge can be operated, with equivalent noise levels, in the control room. This configuration eliminates the problem of signal injection and simplifies bridge operation. During this run the bridge was rebalanced to null in order to maintain the greatest accuracy. There is virtually no heat loss, therefore the drift before and after irradiation is nearly identical. The displacement due to radiation heating is determined by graphical analysis and the known amount of resistance change necessary to null the bridge circuit. Table I shows a series of such measurements carried out in the fast neutron field.

D. Thermal Defect

The thermal defect results from radiochemical processes in the A-150 plastic, therefore it is to be expected that there is a dependence upon total accumulated dose, dissolved oxygen content and, possibly, ionization density. Initially, the plastic may contain dissolved oxygen so that radiochemical reactions are exothermic, resulting in the liberation of excess heat. When this oxygen is removed either by diffusion or by radiochemical depletion, the thermal defect becomes stable

FIGURE 2

TABLE I. SERIES OF CALORIMETER MEASUREMENTS IN NEUTRON FIELD

Run	Resistance Change	Initial Resistance	Relative Percentage Change
	ΔR (ohm)	R_0 (ohm)	$\Delta R/R_0$ ($\times 10^{-3}$)
1	3.107	31,103	9.988
2	3.099	31,101	9.965
3	3.052	31,099	9.813
4	3.077	31,097	9.896
5	3.106	31,096	9.989
6	3.066	31,094	9.861
7	3.089	31,093	9.933
8	3.095	31,091	9.955
9	3.103	31,090	9.979
10	3.066	31,088	9.862

Mean = 9.924×10^{-3}

Standard Error of the Mean = 0.19%.

and is approximately 4%, and we have demonstrated this effect (4). However, after the accumulation of several megarads Fleming and Glass found that radiation damage reduced the thermal defect to about 3.7%. This suggests that a TE plastic calorimeter has a finite lifetime, and core replacement should be considered after accumulating several megarads. Our calorimeter core has received approximately 200 kilorads to date.

Since there is no data presently available on the dependence of the thermal defect upon charged particle type or energy, the value of 4% is subject to a degree of uncertainty. The types of endothermic radiochemical reactions which occur in A-150 plastic include: hydrogen gas production, double bond breakage, vinylene production, main chain scission and some cross-linking. The G- values for these reactions are dependent upon ionization density, but the currently available data do not indicate an obvious relationship which could be employed to predict yields for different charged particle types. The uncertainty presently assigned to the value of the thermal defect has been taken as $\pm 2\%$ until better data is available. This is discussed in the next section.

The object of this set of experiments was to determine the response of the tissue equivalent ionization chamber in two different fast-neutron fields and in a ^{60}Co field. Since the energy of the neutrons is relatively low, about 90% of the dose is deposited by recoil protons while the remainder is deposited by alpha particles and heavier recoil ions. If the formalism of ICRU Report 26 (5) is adopted, then one can write the Bragg-Gray equation which relates absorbed dose in the chamber wall for two radiation fields as follows:

$$\frac{D_n}{D_g} = \frac{Q_n}{Q_g} \frac{m S_g}{m S_n} \frac{\bar{W}_g}{\bar{W}_n}$$

Where D_n and D_g refer to the absorbed doses measured in the neutron and the gamma ray field, Q_n and Q_g are the normalized charges per unit mass collected in the chamber gas in each field, $\bar{W}_n$ and $\bar{W}_g$ are the average energies expended in gas per ion pair collected in each field and $m S_g$ and $m S_n$ are the mass-stopping power ratios between wall and gas for secondary charged particles generated. The response sensitivities can be expressed as the chamber output per unit absorbed dose of fast neutrons relative to the output per unit absorbed dose of gamma-rays, which becomes:

$$\frac{Q_n/D_n}{Q/D} = \frac{m S_n}{m S_g} \frac{\bar{W}_n}{\bar{W}_g}$$

The ICRU Report 26 recommends that if the ratio of mass-stopping powers is nearly unity then this ratio will be approximately 1.05 ± 0.05 for most fast neutron fields.

Upon comparing the response sensitivities of our TE ionization chamber normalized to the absorbed dose measured at the same position using the TE calorimeter in both neutron fields compared to the response sensitivity in the ^{60}Co field, we find the result shown in Table II. Both ratios appear to be somewhat larger than 1.05 but are within the assigned range of uncertainty. The estimated experimental uncertainties are shown in Table III.

The set of measurements compared the response sensitivity ratios for a TE ionization chamber with flowing TE gas measured in two neutron fields and a ^{60}Co gamma-ray field. The sensitivities were determined by comparison of charge collected by the TE ion chamber to the absorbed dose measured with a TE calorimeter. The ratios for two neutron fields relative to the ^{60}Co field are somewhat higher than expected using the recommendations of ICRU Report 26. The difference could reflect a slight increase in the value of $\bar{W}_n$ for these neutron fields, or an increase in the mass-stopping power ratio between chamber wall and gas, or a combination of both. Recent measurements and calculations (6) of $\bar{W}_n$ for methane based TE gas suggest that previously employed values were somewhat low. Our results are consistent with these findings and future work will focus on extending comparisons of ionometric and calorimetric measurements to neutron fields with differing energy spectra.

III. THERMAL DEFECT MEASUREMENTS

The thermal defect value measured by Fleming and Glass for TE plastic (7) could be employed for neutrons of relatively low energy since the dose is predominately deposited by recoil protons. However, for higher energy cyclotron-produced fast neutron beams an increasingly larger fraction of dose is deposited by alpha particles and heavy recoil nuclei. Radiochemical yields for these densely ionizing charged particles are usually quite different than those obtained with sparsely ionizing radiation. In order to properly evaluate the thermal defect for fast neutron irradiated A-150 TE plastic, the technique of Fleming and Glass will be extended by using different energy protons and other charged particles.

This project is a collaborative effort of the Sloan-Kettering, Biological Dosimetry Laboratory and The Radiological Research Accelerator Facility at Brookhaven National Laboratory. Thermal defect measurements will be performed on a number of TE plastic samples from three laboratories in the United States and one in Europe. An attempt will be made to evaluate samples molded at the extremes of temperature and pressure which will demonstrate whether the thermal defect is dependent on sample processing technique. Hydrogen, helium, carbon and possibly oxygen ions of various energies will be accelerated by a 4 MV Van de Graaff. The calorimeter constructed by Fleming and Glass is being employed, however, certain critical internal components have been rebuilt or replaced. A block diagram of the system is shown in Figure 3.

TABLE II. RATIO OF TE IONIZATION CHAMBER RESPONSE IN TWO
NEUTRON FIELDS RELATIVE TO ^{60}Co AND NORMALIZED
TO DOSE MEASURED BY THE TE CALORIMETER

Neutron Field	$^9\text{Be} (d,n)$	$^9\text{Be} (^3\text{He},n)$
Mean Energy	$\bar{E}_n \approx 3.5 \text{ MeV}$	$\bar{E}_n \approx 6 \text{ MeV}$
Response Sensitivity Ratio	$1.08 \pm .03$	$1.07 \pm .03$

TABLE III. ESTIMATED EXPERIMENTAL UNCERTAINTIES

Calorimetric Measurements		
Item	Random %	Systematic %
Thermal Defect	--	2.0
Geometry	0.2	0.2
Irradiation Precision	0.8	--
Irradiation Uniformity	--	0.5
Electrical Calibration	0.3	--
Graphical Analysis	0.3	0.1
Mass and Composition of Core	0.02	0.1
Total Uncertainty		$\pm 2.5\%$
Ionometric Measurements		
Measurement of Integrating Capacitor	0.01	0.1
Measurement of Potential with Electrometer	0.01	0.1
Geometry	0.2	0.2
Irradiation Precision and Uniformity	0.8	0.5
Correction to Standard Temperature and Pressure	0.1	0.1
Correction for Recombination	0.1	0.1
Total Uncertainty		$\pm 0.7\%$

Note: The uncertainties in $\bar{W}_n$ and mS_n are not indicated since they are not employed in measuring the ratio of response sensitivities.

FIGURE 3.

This calorimeter system has been described previously (7) and a brief summary of its operation will now be given. The charged particle beam from the accelerator is swept across the absorber face by electrostatic deflection plates, in the manner of a video raster, in order to produce a uniform irradiation pattern. The calorimeter is adiabatically controlled by a thermopile which detects the temperature difference between the absorber and the surrounding adiabatic jacket. The thermopile signal is amplified and used to control a thermal regulator which provides current for a heater surrounding the adiabatic jacket.

Since the absorber and adiabatic jacket are constrained to rise in temperature simultaneously, the measuring thermistor is placed in the jacket in order to minimize power dissipation in the absorber. Electrical calibration is achieved through the use of a resistance element between the TE plastic and aluminum faces.

The cross-sectional diagram in Figure 4 shows the various elements of the calorimeter. The aluminum isothermal jacket is maintained at a set temperature by a thermal regulator, and its interior is at vacuum. Within this element is the adiabatic jacket, which is thermally and electrically insulated from the outer aluminum wall. The adiabatic jacket is maintained at a negative potential in order to suppress the emission of secondary electrons from the absorber and assure a reliable measurement of the beam current. The measuring thermistor and thermopile are mounted with a high thermal conductivity epoxy and are electrically guarded from the negative high potential to prevent leakage currents.

The absorber can be rotated so that either face will completely stop the charged particle beam. One face is aluminum 0.018 cm thick and 1.4 cm in diameter and the other is 0.025 cm thick TE plastic of the same diameter. The ratio of power measured using both sides of the absorber yields the thermal defect. Details of the absorber assembly are shown in Figure 5.

IV. MICRODOSIMETRY

A. Neutron Measurements

An E.G. and G. tissue equivalent proportional counter was employed to measure event size spectra in air for the neutron field produced using our deuterium gas target (1). The result of one such measurement is shown in Figure 6. For comparison, the event size spectrum produced by the $^3\text{He}, n$ interaction is also shown. The higher neutron energies present in the D-D field are evident in the increased number of events between about 3 and 30 keV/ μm due to fast recoil protons. In addition, there are fewer events due to slow and stopping protons near 100 keV/ μm . The number of low lineal energy events due to gamma-rays

FIGURE 4

FIGURE 5

CROSS SECTION OF DUAL ABSORBER

FIGURE 6

is small, and the dose due to gamma-rays in air has been estimated to be approximately 2.5%. This compares favorably with the value of 3.5% as measured using the G-M detector described in section V. The total dose per unit target current as measured by proportional counter and the 1 cc E.G.G. spherical ionization chamber were found to agree to within 2%.

Schraube et al. (8) have calculated the neutron spectra for various deuteron energies, and a comparison of two spectra for 6.9 and 10.6 MeV deuterons is shown in Figure 7. These energies correspond to the values for the deuterons generated at the Sloan-Kettering and the Heidelberg cyclotrons. The mean neutron energies are not too different due to the increased number of events due to the low energy break-up interaction which becomes more predominant at higher deuteron energies.

Figure 8 shows a comparison of data taken by Menzel et al. (9) at the Heidelberg cyclotron and our data. Several features are evident: the higher energy content of the neutron spectrum at the Heidelberg cyclotron is reflected in the increased number of low lineal energy events due to fast protons and high lineal energy events resulting from alpha particles and heavy recoil nuclei. In addition, there are also relatively more events due to slow and stopping protons resulting from low energy neutrons from the break-up interaction. The most probable event size in both spectra is nearly the same.

Figure 9 shows the calculated values of $\bar{y}_F$ and $\bar{y}_D$ for monoenergetic neutrons generated by Caswell and Coyne (10). If we assume that the calculated energy spectrum for 6.9 MeV neutrons represents the spectrum produced at our cyclotron then one can attempt to employ the monoenergetic values for $\bar{y}_F$ and $\bar{y}_D$ weighted by the energy spectrum in the manner indicated in Figure 10. The expected values are computed by integrating over the neutron spectrum. The value calculated for $\bar{y}_F$ is in reasonably good agreement with the value computed using the measured event size spectrum. The value for $\bar{y}_D$ is higher due to the fact that it represents the second moment of the distribution so that larger event sizes are weighted more heavily. Since the proportional counter is constructed of TE plastic which contains a higher percentage of carbon relative to ICRU tissue, an increased number of large events and a higher $\bar{y}_D$ would be expected.

We hope to develop this technique of utilizing monoenergetic microdosimetric parameters along with planned measurements of the neutron energy spectrum in order to attempt a calculation of the secondary charged particle spectrum generated by this neutron field. In this way it may be possible to compare calculations of the relative biological effectiveness using the methods of Kellerer and Rossi and those of Katz, and thereafter to measured biological data.

CALCULATED NEUTRON SPECTRA FOR THE $D(d,n)^3\text{He} + D(d,np)D$ INTERACTIONS
(SCHRAUBE et al. 1974)

FIGURE 7 .

FIGURE 8

CALCULATED MICRODOSIMETRIC PARAMETERS
FOR MONOENERGETIC NEUTRONS
(CASWELL AND COYNE, 1975)

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FIGURE 9

$$\langle \bar{y}_F \rangle = \frac{\int_0^{E_n^{\max}} \bar{y}_F(E_n) \cdot \phi(E_n) dE_n}{\int_0^{E_n^{\max}} \phi(E_n) dE_n}$$

FIGURE 10

	$\bar{y}_F$	$\bar{y}_D$
Calculated	16.7	63.8
Measured	16.9	72.7

B. Photon Measurements

Encapsulated ^{125}I sources were employed along with two x-ray tubes for these studies since these same tubes were used in radiobiological experiments in order to simulate the radiation emitted from ^{125}I at dose rates sufficient to obtain cell survival curves. The two x-ray tubes were a Machlett OEG-50 molybdenum target tube with a 1.0 mm beryllium window and a Machlett OEG-50-G tungsten target tube with a 0.25 mm beryllium window. The tube potential, obtained with a capacity filtered supply (11), was monitored by means of a digital microammeter which measured the current flowing through a resistor connected from the high potential to ground. The uncertainty in the value of this resistor was estimated to be $\pm 3\%$. A modification in the voltage supply made it possible to operate the tube with x-ray outputs of sufficiently low intensity so as to remain within the counting capacity of the instrumentation. The x-ray tube currents employed ranged from about 5 to 100 μA .

As previously discussed (11), the potential applied to the tubes with these low currents would be essentially constant potential. At the voltage and currents employed for the radiobiological studies, 40 kV and 25 μA , it was computed that the instantaneous potential would be 0.96 that of the peak potential. Therefore, one might expect slight differences in the fields employed for the radiobiological experiments and these microdosimetric measurements, although these differences are not expected to be greater than a few percent.

The proportional counter assembly consists of a spherical A-150 TE plastic (1) shell with an inner diameter of 1.27 cm and a wall thickness of 0.127 cm. A thin (0.018 cm) aluminum enclosure surrounds the counter shell and it is vacuum tight permitting the enclosure to be filled with TE gas at reduced pressure. The density ratio between the wall and the gas creates an effective sphere diameter of a few micrometers of tissue equivalent material. The TE gas employed is the propane based mixture devised by Srdoc (1), and a pressure of 34 Torr yields an effective sphere diameter of 1 μm . This pressure was computed using the relationship: $P = 43.2 \text{ Torr} \cdot \text{cm} / \mu\text{m}$ (12) which assumes a density for the gas of 1.929 g/l. Since the presence of water vapor, oxygen and other contaminants in the counter gas can severely degrade its performance, the gas filling system contains in-line filters and all metal fittings with the exception of some nylon tubing.

The proportional counting region is defined by an anode, which is a thin stainless steel wire fixed across a diameter of the TE plastic shell, and a concentric helical electrode whose potential is a fixed fraction of the anode potential. This design is based on the original work of Rossi and Rosensweig (13). During these experiments, the anode potential was maintained at either +600 or +650 volts. Calibration was carried out using the built-in ^{241}Am source which is mounted with a small swing assembly on the outer surface of the TE plastic shell. This arrangement allows alpha particles from the source to enter the counter through a small hole in the wall, which serves as a collimator, and then cross the spherical volume along a diameter. The calibration source is in this position in only one particular counter

orientation, and when the instrument is accumulating data the source is held away from the hole in the counter wall by gravity.

Alpha particles emitted by ^{241}Am have an energy of approximately 5.47 MeV and the energy loss in a 1 μm pathlength of propane based TE gas has been calculated to be 88.5 keV/ μm . This value was computed by averaging the data of Oldenberg and Booz (14) and the calculations of Janni (15) for protons which were scaled to the equivalent alpha particle energy. Since the data of Janni was tabulated for elements and some compounds, it was assumed that the stopping powers for the individual elements of the TE gas mixture could be linearly combined according to their mass fractions. Although it is difficult to estimate the uncertainty introduced by using this procedure, it is possible to compare certain stopping power values with experimentally obtained data (16) for alpha particles incident upon gasses such as propane. The agreement is within about 3% which may indicate the uncertainty of the procedure. An example of the event size distribution produced by alpha particles from the calibration source is shown in Figure 11. The full width at half-maximum is about 19%. This is due, in part, to the resolution of the counter, energy straggling (17) and the geometry of the collimator, which is not optimally designed since it is simply a hole in the wall of the counter. The centroid of the alpha particle distribution was computed in order to determine the energy scale.

Electronic signals were amplified and processed using commercially available low noise modular electronics and a multichannel analyzer (MCA). Since all of the events from the sources employed in this study were below about 20 keV/ μm , it was necessary to accumulate a distribution from the alpha calibration source, set the output of a precision pulser at the peak channel of the calibration distribution and then increase the amplification sufficiently to accumulate pulses. The gain was then re-checked with the pulser.

The number of events as a function of lineal energy was stored and then processed using a computer program described previously (3). In these measurements, event size distributions were accumulated in either two or three segments which were numerically smoothed and then fitted together. This procedure was necessary due to the limited dynamic range of the electronic system and the analog-to-digital converted in the multichannel analyzer.

Figure 12 shows the number of events, $n(y)$, as a function of lineal energy for ^{125}I . Individual data points, representing the normalized number of events per MCA channel, have been joined by straight line segments in the computer program. There are fewer events at higher lineal energy, therefore fluctuations in the data increase in this region and uncertainties in the data due to counting statistics ranged from approximately $\pm 3\%$ to less than $\pm 1\%$. A smooth continuous line was passed through the data by the method of cubic splines (3), and this smoothed data set was employed in subsequent data analysis.

FIGURE 11

FIGURE 12

Y (KEV/MICRON)

No attempt has yet been made to extrapolate the number spectrum to very small event size since noise limits accurate measurements in this region using our counter system. In addition, there is, to our knowledge, no theoretically justified method for such extrapolations. In view of this limitation, the event size distributions and the microdosimetric quantities reported are subject to additional uncertainty due to our inability to record extremely small events.

The distribution of dose as a function of lineal energy, $d(y)$, is obtained by multiplying the number of events at a particular value of lineal energy, $n(y)$, by the energy per event: $d(y) = y \cdot n(y)$. A comparison of these distributions for ^{125}I , ^{60}Co and ^{137}Cs is shown in Figure 13. A particularly useful form of the event size distribution is obtained by plotting $y \cdot d(y)$ versus the logarithm of y . In this type of display, equal areas between two values of y represent equal values of absorbed dose. Figure 13 shows a comparison of the distributions of dose in lineal energy for the above mentioned radiations where the data under each distribution has been normalized to unit dose. No individual data points are depicted in this presentation of the data since the smoothed data set was employed for these distributions. The event size distribution for ^{125}I is narrow and centered about a higher value of lineal energy than the other two isotope sources. Lower energy photons from this source interact with the counter wall to produce lower energy secondary electrons having higher stopping powers. The larger energy depositions of these secondary charged particles produce larger pulses in the counter. The peak roughly corresponds to the most probable deposition events and the relatively narrow shape is due to the low energy of the electrons and the fact that their slowing down spectrum is not very broad. Higher energy secondary electrons produced by ^{60}Co and ^{137}Cs deposit less energy in the counter resulting in a distribution of smaller events. This description assumes that ionization in the counter gas accurately reflects the deposition of energy and this assumption may be subject to some uncertainty due to the dependence of $\bar{W}$, the average energy expended by electrons per ion pair detected by the counter, on electron energy and non-tissue equivalence of the gas at low electron energies (18). The distributions each have a maximum event size of about $20 \text{ keV}/\mu\text{m}$ since electrons, like other charged particles, have a maximum in their stopping power versus energy relationship so that they have an energy deposition maximum. Figure 14 shows distributions of dose in lineal energy for ^{125}I along with those for three x-ray fields. The tungsten target x-ray tube was operated at two different potentials and the molybdenum target tube was employed for only one set of measurements at 40 kVcp. The tungsten target tube operated at 40 kVcp produces a distribution which is very similar to that of ^{125}I . This x-ray tube was employed for the radiobiological experiments previously mentioned (11) in order to simulate the radiation from ^{125}I , and it appears, from the event size distributions, that the two radiations are closely matched.

The other two distributions reflect qualitatively the increased content of low energy photons in these fields. This results in more dose at higher lineal energy. However, the values of frequency

FIGURE 13

FIGURE 14

mean and dose mean lineal energy for these fields are not substantially higher than those for ^{125}I , suggesting that their radiobiological properties may not be very different.

V. DOSIMETRY INSTRUMENTATION DEVELOPMENT

A. Compact Spherical Calorimeter

Initial investigations into the feasibility of constructing a compact calorimeter took place and some results are reported here. Since calorimetric dosimetry systems have always been viewed as inconvenient to employ outside the realm of a standards laboratory, we have attempted to develop a simple instrument for more routine use. The general form of the instrument can be seen in Figure 15. It is similar in construction to E. G. and G. dosimeters which are currently being used by all of the neutron therapy trial centers in this country. Since the calorimeter core is spherical, constructed of A-150 plastic and has a diameter identical to that of the E. G. and G. IC-17 TE ionization chamber, direct comparisons can be made of kerma in A-150 as determined using both dosimeters. Perhaps this will help resolve some questions relative to the applicability of Bragg-Gray theory for the E. G. and G. chamber in neutron fields and, possibly, augment calibrations in radiotherapy beams on a more routine basis.

We are collaborating with W. Quam of the Santa Barbara Division of E. G. and G. on this project. Certain internal components such as the core and adiabatic jacket are being fabricated by Mr. Quam. The E. G. and G. facilities at Goleta are capable of plastic molding and lamination techniques not possible in our laboratory. In addition, if the design proves to be useful, then it is possible that E. G. and G. might make it available commercially.

B. Geiger-Mueller Dosimeter

We have constructed an energy compensated G-M dosimeter based on a design first described by Wagner and Hurst (19) and subsequently modified by Wagner (20). The gamma-ray energy response is relatively constant ($\pm 10\%$) over a wide range from approximately 55 keV to a few MeV. The energy compensating filter, constructed of tin and lead, has 30 equally spaced holes 2 mm in diameter drilled in the filter wall and 8 holes 1 mm in diameter in the filter end. In addition, 0.07 mm of lead was added to the wall but not the end. The neutron response of this dosimeter was estimated to be less than 0.5% of the gamma-ray response for neutron energies of a few MeV. Recent measurements have also determined the neutron sensitivity of a similar counter to be $\leq 0.4\%$ for several monoenergetic neutron beams ranging up to 15 MeV (21). This value is significantly lower than the neutron

FIGURE 15

response of non-hydrogenous ionization chambers which may be a few percent for neutrons up to 5 MeV and as much as 30 to 35% for higher neutron energies (22).

1. Description of the Instrument

The dosimeter was designed to be geometrically similar to the E. G. and G. IC-17 tissue equivalent (TE) plastic ionization chamber (1) so that both instruments can be interchangeably placed in an irradiation tank for the purpose of measuring the total absorbed dose due to neutron plus gamma-rays and the gamma-ray dose at a particular point using the TE ionization chamber and the G-M dosimeter respectively. Since the G-M dosimeter is relatively small, it can also be employed for the measurement of gamma-ray isodose distributions in air and in liquid, using a remotely controlled positioning mechanism.

Figure 16 shows a view of the assembled dosimeter. The cubical connector block is sturdy, for mechanical mounting, and it contains within it an emitter-follower circuit, powered by the high voltage line, which provides +6 volt pulses approximately 20 μ s long which can be counted using any standard scaler. A schematic diagram of this circuit is shown in Figure 17. The long stem allows positioning of the detector which is remote from the connector block, and all fittings below the block are waterproof to permit immersion in liquid. The aluminum cylinder surrounding the detector is double walled so that there is approximately 3 mm of ^6LiF powder surrounding the miniature Geiger tube. Figure 18 shows a detailed photograph of the detector with the cover removed. The purpose of this ^6LiF shield is to reduce the fluence of thermal neutrons reaching the counter when it is at some depth in a hydrogenous medium. A second version of the thermal neutron shield using $^6\text{Li CO}_3$ was also employed, but no detectable difference was found in measurements taken using either material. Two types of Geiger tube were evaluated. An Amperex 18529 and an ICC 1015/S were compared and their respective response characteristics are discussed in the next section.

2. Calibration

The G-M dosimeter was calibrated in a ^{60}Co gamma-ray field by placing the geometrical center of the counter tube at a point approximately 1 m from the source and accumulating counts for a fixed time period. The tissue dose rate at this point had been previously evaluated using a calibrated ionization chamber. The response sensitivities for both Geiger tubes are shown in Table IV.

The count rate as a function of absorbed dose rate was measured to be linear over a range of 50 to 200 count/sec for a counter potential of +510 volts. Dead time was less than 0.5% for count rates up to 200 count/sec, therefore, no correction was applied to the measurements reported here since they were carried out at approximately 150 count/sec. There is some systematic uncertainty associated with this calibration procedure since the counter is neither tissue equivalent

FIGURE 16

FIGURE 17

FIGURE 18

Table IV. Response sensitivities of two Geiger tubes employed.

<u>Amperex 18529</u>	<u>ICC 1015/S</u>
9.5×10^{-7} rad/count	4.5×10^{-7} rad/count

Table V. Neutron sensitivity of graphite-CO₂ ionization chamber.

<u>Computed</u>	<u>Measured with respect</u>
<u>(9.5±1)%</u>	<u>to G-M dosimeter</u>
	(9.9±1)%

Measured gamma-ray percentage of total dose
in air at one meter = 3.3%

nor air equivalent. Therefore, the uncertainty in calibration is estimated to be approximately $\pm 5\%$. In addition, Thorngate (23) has found that the counter response rises at photon energies above about 4 MeV. This non-linearity might contribute to an overestimate of dose at depth in a hydrogenous medium, but this is also believed to be a small effect compared to the uncertainties associated with other methods of measuring gamma-ray dose in a neutron field.

3. Measurements in Neutron Field

The mixed fast neutron and gamma ray field produced by the $D(d,n)^3\text{He}$ and $D(d,np)D$ interactions in a pressurized deuterium gas target bombarded with 7.5 MeV deuterons from the Sloan-Kettering cyclotron has been studied previously using a number of dosimetric techniques (1). The neutron spectrum produced by these interactions is roughly triangular in shape from about 2 to 11 MeV with a mean energy of approximately 7.6 MeV. The gamma-ray fraction of total dose in air for a 13 cm diameter collimated field at a distance of one meter from the deuterium gas target entrance foil was measured using an ionization chamber constructed of graphite with CO_2 flowing through the collecting volume at a rate of approximately 5 cc/min. The G-M dosimeter was then placed at the same position so that intercomparison of these dosimeters could be performed. Accumulated target current was measured using a current integrator and a scaler to monitor these experiments. The effect of stray radiation upon the emitter follower circuit was evaluated by irradiating only the stem and connector block in the collimated neutron field. The number of counts produced per monitor unit was found to be less than 0.2% of the value measured when the entire instrument was irradiated. No correction was made for this small effect since it could have been due, in part, to incomplete shielding of the Geiger tube by the collimator.

The neutron sensitivity of the graphite- CO_2 chamber was estimated using a spectrum calculated by the method of Schraube (8) for a neutron field similar to the one employed for these experiments. Values of the ratio of neutron sensitivity to gamma-ray sensitivity as a function of neutron energy have been determined by Kuchnir (24). These values, along with the computed neutron spectrum, were employed to calculate the neutron sensitivity ratio for the graphite- CO_2 chamber in our neutron field. Results are shown in Table V along with the measured percentage of gamma-ray dose in air. The computed neutron sensitivity ratio compares favorably with the value obtained by determination of the graphite- CO_2 neutron sensitivity ratio necessary to achieve the same gamma-ray dose as measured by the G-M dosimeter (assuming its sensitivity to be 0.4%). The gamma-ray dose expressed as a percentage of total dose (neutron + gamma) versus depth in a 30 cm cubic Lucite walled water tank is shown in Figure 19. The measurements were carried out using the TE ionization chamber to determine total dose and both the graphite- CO_2 chamber and the G-M dosimeter to determine the gamma-ray percentage. Since the data obtained by both methods were nearly identical, the average value at each depth has been plotted.

FIGURE 19

The G-M dosimeter has been useful in determining gamma-ray dose as a percentage of total dose in a cyclotron produced fast neutron field of fairly low energy. Measurements with a similar detector have been carried out recently (25) in higher energy neutron fields generated by 26, 35 and 45 MeV protons incident upon Be and Li targets. It was estimated that the neutron response sensitivity for the G-M dosimeter in these fields was also below 1%. In conclusion, it appears that this dosimeter can be employed for use in neutron fields ranging in energy from a few MeV to several tens of MeV and, although its absolute accuracy is subject to some uncertainty, its small size, convenient read-out and low sensitivity to neutrons are very desirable features.

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