000-3522-14

PROGRESS REPORT
TO
THE UNITED STATES ATOMIC ENERGY COMMISSION
CONTRACT NO. AT(11-1)-3522

BIOLOGICAL AND CLINICAL DOSIMETRY

Contract Period: November 1, 1973 to December 31, 1974
Report Period: July 1, 1973 to June 30, 1974
Investigator: John S. Laughlin, Ph.D., Member
Co-Investigators: Lowell L. Anderson, Ph.D., Associate
Thomas R. Canada, Ph.D., Associate
John F. Detko, M.A., Research Associate
Joseph C. McDonald, M.S., Research Associate

Sloan-Kettering Institute for Cancer Research
410 East 68th Street
New York, New York 10021

October 31, 1974

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

MASTER

DISTRIBUTION OF THIS DOCUMENT UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

TABLE OF CONTENTS

	<u>Page</u>
I. SUMMARY	4
II. DIRECT CALORIMETRIC MEASUREMENT OF ABSORBED DOSE	6
1. Physical Properties of Shonka A-150 Tissue Equivalent Plastic	6
2. Preparation of Calorimeter Central Absorbers	9
3. Thermal Defect Measurements	10
III. NEUTRON DOSIMETRY	14
1. Total Dose Measurements	14
a. Angular Distributions	14
b. Transmission Measurements	18
2. Neutron and Gamma-Ray Components of the ($^3\text{He},n$) and (d,n) Fields	20
a. Paired Chamber Technique	20
b. TLD Measurements	22
3. Radiobiological Measurements	23
a. HeLa Cell OER and RBE	23
b. Experimental Animal Radiotherapy	27
IV. MICRODOSIMETRIC MEASUREMENTS	29
1. Experimental Apparatus	29
2. Results and Discussion	33
V. DOSIMETRY OF ^{252}CF MEDICAL NEUTRON SOURCES	39
1. Dosimetry	39
2. Radiation Protection	41

	<u>Page</u>
VI: GERMANIUM GAMMA CAMERA	47
1. Introduction	47
2. Background	47
3. Tests on a Enlarged Camera	52
4. Stability and Quantum Efficiency of the Enlarged Camera	52
5. Tests on Small Matrix Units	54
6. Tests on Passivated (2x2) Matrix Units	61
7. Matrix Electronics	65
VII. PUBLICATIONS AND PRESENTATIONS	69

I. SUMMARY

The thermal defect, which occurs in plastics used for absorbed dose calorimeters, has been studied in detail for Shonka A-150 tissue equivalent (T.E.) plastic. A thermal excess of 10% is attributed to a failure to anneal dissolved oxygen. It has been found that thermal annealing results in an absorber whose output stabilizes quickly and remains constant with accumulated dose. Details of sample preparation and other physical measurements are discussed.

A number of physical properties of the fast neutron fields produced when 8 MeV deuterons or 23 MeV ^3He ions are incident upon a thick beryllium target have been measured. These include: the angular distributions of total dose, the linear dose attenuation coefficients of steel, lead, and a polyethylene mixture, the dependence on depth of the neutron dose and the gamma-ray dose.

The radiobiological properties of the ($^3\text{He}, n$) and (d, n) neutron fields have been studied as a function of depth within a water phantom. Aerobic HeLa S3 cell survival data were obtained at three depths between 2 and 12 cm. The resulting survival curves, plotted as a function of neutron dose, show no dependence on depth. Aerobic and hypoxic HeLa S3 cells were irradiated in-vitro at depths of 2 and 12 cm. Although the resulting survival curves show a small decrease in sensitivity with increasing depth, the oxygen enhancement ratio remains constant and has a value of 1.8 ± 0.1 in both fields.

Microdosimetric measurements were made for cyclotron produced fast neutrons and for ^{252}Cf sealed sources at various positions in phantoms. Differences in quality and their significance to radiobiological experiments and clinical applications have been considered.

The neutron and gamma-ray components of dose have been determined for ^{252}Cf sources by a paired ionization chamber method. Extensive thermoluminescent dosimeter measurements were performed to determine the beta-gamma components of dose. The evaluation of NTA neutron film for personnel monitoring and clinical applications are also discussed.

The current area of the germanium gamma camera detector is just under 11 cm^2 . Evaluation tests during the current year revealed serious loss of quantum efficiency with time in the device. This appeared after exposure of the matrix detector to room ambient, an occurrence which is expected in the construction and maintenance of a full scale camera. The cause of the degradation appears to be the result of surface

contamination, particularly at the interelectrode regions. A group of test matrix units was constructed from the original (2x2 cm) prototype detector and used to evaluate a commercially available passivation technique. Initial results are encouraging with two of four passivated units undetermined after two and four weeks respectively of room ambient storage. Preparations for passivation and testing the current camera detector are in progress.

A specially designed electronic monitor system is under commercial construction for use with the germanium camera after further enlargement of the detector.

II. DIRECT CALORIMETRIC MEASUREMENT OF ABSORBED DOSE

(J. C. McDonald, T.R. Canada, R. Freeman and J.S. Laughlin)

I. Physical Properties of Shonka A-150 Tissue Equivalent Plastic

Our calorimeter has been designed to measure absorbed dose in neutron fields, specifically those produced by our cyclotron using the $^9\text{Be}(^3\text{He},n)$ and $^9\text{Be}(d,n)$ reactions. In order to maintain a kerma ratio relative to tissue close to unity, the central absorber and all major components have been constructed of Shonka type A-150 (1) tissue equivalent (T.E.) plastic.

When certain plastics are used in absorbed dose calorimeters, endothermic radiochemical effects occur. It was initially observed in our lab (2), and subsequently by others (3,4,5), that approximately 3% of the energy absorbed in tissue equivalent plastic mixtures does not appear as heat.

Some earlier calorimeters designed by Laughlin and co-workers (2,6) have made use of plastic mixtures which contained carbon so that they were electrically conducting. A calibration current was then used to deposit a known amount of energy by Joule heating of the absorbing wafer. This method eliminated the need to insert a heating element which resulted in a more nearly tissue equivalent absorber. The present design relies on this same method of electrical calibration.

Results of resistivity measurements made with Shonka A-150 plastic are shown in Figure I-1. Samples of varying lengths with the same cross section (2x2 cm) were suspended in a thermal enclosure which was held at a constant temperature. Resistance is plotted as a function of the ratio of length to area so that the slope in each case is the resistivity. The lines were generated by a linear least squares fit of the data, and the y-intercept indicates a value for the contact resistance of about two ohms.

The variation of resistivity with temperature is shown in Figure I-2. We have found that accumulated doses from 100 to 150 kilorads of orthovoltage x-rays produce no measureable change in the value of resistivity measured for the calorimeter absorber.

FIGURE I-2

2. Preparation of Calorimeter Central Absorbers

Cylindrical wafers of Shonka A-150 are machined from stock to dimensions of 2 cm diameter by 0.2 cm thick in the SKI machine shop. A hole 0.3 cm deep is drilled in the edge of the wafer using a #60 drill to accomodate the thermistor. Surface preparation consists of sanding each side first with 220 grit and then with 280 grit sandpaper. The surfaces are then cleaned using 100% ethyl alcohol followed by distilled water. To achieve good adhesion and low contact resistance in later steps it is now necessary to thermally anneal the wafer by placing it on a non-metallic surface in a vacuum oven at 100°C and 20 inches of Hg for a period of 48 hours.

After cooling to room temperature, the wafer is weighed using a Mettler balance to give the first of a series of weights of materials making up the thermally active element. The weight of the Veco 43A1 thermistor and 30 cm of constantan lead wires is also recorded at this time. A small amount of Emerson & Cumming Stycast 2850 GT thermally conducting epoxy (9 BTU in/ft²-hr-°F) is mixed with catalyst and allowed to stand under vacuum to remove air bubbles. We then fill the hole in the wafer with epoxy and insert the thermistor. To insure good thermal contact this assembly is again placed in a vacuum to force air out of the unhardened epoxy. Upon hardening (~ 6 hours) the assembly of wafer, epoxy, thermistor and wires is weighed to give the weight of the epoxy in the thermally active element.

Two 20 cm lengths of #40 AWG copper enameled wire are weighed and cemented to the flat faces of the Shonka wafer with small dots of silver electrically conducting epoxy supplied by Tra-Com Co. These leads and lands are trimmed under a microscope to minimum size and weight consistent with good electrical contact and inevitable minor stresses. After trimming the current assembly is again weighed yielding the weight of silver epoxy on the wafer.

In the final assembly step the faces of the wafer are spray coated with a thin layer of Acheson Electrodag 415 conducting paint. This coating covers the lands and forms an equipotential surface for uniform electrical heating of the wafer. Upon drying, the total weight of the assembly is recorded and the weight of the Electrodag calculated from that. The completed wafer is then baked again in the vacuum oven for 12 hours. This step, we have found, reduces the contact resistance to approximately 1 ohm. The resistance of the wafer is then measured using a General Radio impedance bridge and it is then assembled into the calorimeter.

3. Thermal Defect Measurements

Bewley has observed both exothermic and endothermic radiochemical reactions in a calorimeter absorber constructed of Shonka A-150 T.E. plastic (4). His measurements indicated an approximately 10% larger response per rad relative to an ionization chamber. After exposure to 100 kilorads of 7 MV x-rays, the relative response was approximately 4% less than expected from the chamber measurement. This is consistent with the expected value for endothermic radiochemical effects.

We have also investigated this problem by comparing the response of our calorimeter to a cylindrical T.E. ionization chamber whose collecting volume is identical in size to the calorimeter absorber. It is constructed of Shonka A-150 plastic and filled with a gas mixture of methane, carbon dioxide, and nitrogen (6) so that the atomic composition of the chamber wall and the gas is nearly identical. The irradiation geometry is indicated in Figure I-3. The ionization chamber is placed in a phantom whose outer dimensions approximate those of the calorimeter. The same amount of T.E. plastic is placed in front of both detectors. During the course of x-ray irradiations, exposure measurements were performed by members of the Memorial Hospital Regional Calibration Laboratory using a Baldwin-Farmer ionization chamber. Continuous monitoring during these experiments was achieved with a solid state transmission monitor located near the tube window.

Figure I-4 shows the relative response of the calorimeter as a function of accumulated dose. The output in each case has been determined by measuring the percentage change in resistance of a miniature thermistor embedded in the calorimeter's central absorber. Each point represents at least three measurements, and the error bars indicate a standard deviation of about 0.5%. The response has been normalized to the initial value measured, and the lines are smooth curves drawn through the data. The upper curve shows results obtained with a sample that had been thermally annealed before the application of electrical contacts. A small decrease of a few percent in relative response during the first 20 to 30 kilorads is followed by a constant output thereafter. The second set of data is the result of an absorber which had the electrical coating applied and was then heated. The lower curve shows the result of using an absorber which was not thermally annealed. The decrease in response with accumulated dose is approximately 15%, which is consistent with the total change of 14% observed by Bewley.

IRRADIATION GEOMETRY

Figure I-4

It is likely that dissolved oxygen is involved in the initial exothermic effect. During irradiation, radical formation occurs in polyethylene, the major component of Shonka plastic, and the production of oxides in excited states which decay with the release of heat is entirely possible. This effect would continue until the dissolved oxygen was depleted. We cannot propose exact mechanisms at the present time since little is known about the radiation chemistry of this particular tissue equivalent plastic mixture. The thermally annealed samples, particularly the absorber heated before the coatings were applied, show a reduced effect due to the removal of oxygen from the plastic during heating. The small initial decrease is apparently due to the radiochemical consumption of any remaining oxygen. The equilibrium portion of the upper curves corresponds to a thermal defect of approximately 3-1/2%.

Our measurements have shown, and we therefore recommend, that thermal annealing should be done before electrical contact application in T.E. calorimeter absorbers since it virtually eliminates large initial exothermic radiochemical effects.

At the present time the calorimeter is in the target room of the MSKCC cyclotron. Neutron doses of a few hundred rad have been measured with no difficulty, and further work is under way.

REFERENCES:

1. Shonka, F.R., Rose, J.E., and Failla, G. Proc. 2nd Internat. Conf. Peaceful Uses Atomic Energy, 21:184, 1958.
2. Milvy, P., Genna, S., Barr, N., and Laughlin, J.S. Proc. 2nd Internat. Conf. Peaceful Uses Atomic Energy, 21:142, 1958.
3. Fleming, D.M. and Glass, W.A. Rad. Res. 37:316, 1969.
4. Bewley, D.K., McCullough, E.C. Page, B.C., and Sakata, S. Phys. Med. Biol. 17:95, 1972.
5. Sabel, M., Schmidt, Th., and Pauly, H. Health Phys. 25: 519, 1973.
6. Rossi, H.H. and Failla, G. Nucleonics 14:32, 1956.

III. NEUTRON DOSIMETRY

(J.C. McDonald, T.R. Canada, R. Freeman, and J.S. Laughlin)

1. Total Dose Measurements

a. Angular Distributions

Dose measurements were made with ionization chambers described in our last Annual Report (1). These chambers are shown in Figure III-1. Chambers B and C are constructed of Shonka A-150 tissue equivalent plastic and are filled with a gas mixture of 64.4% CH₄, 32.4% CO₂ and 3.2% N₂ by partial pressures. The chambers are operated in a semi-sealed mode wherein filling is repeated before each run and the gas is maintained at approximately 15 mm Hg above atmospheric pressure by a gas regulation system. The larger cylindrical T.E. chamber (B) had a nominal active volume of 0.8 cc with a wall thickness of 2.5 mm. The smaller thimble-shaped T.E. chamber had a nominal active volume of 0.06 cc and a wall thickness of 0.50 mm.

A neutron collimator was designed and installed in the high density concrete wall separating the cyclotron and experimental rooms. This system is considered a prototype, which would allow dosimetric and biological experiments to continue while studies of the proper material composition and physical dimensions necessary for a final system design were conducted. The collimator had an overall length of 40 cm: 10 cm of steel, 5 cm of lead and 25 cm of a polyethylene-lead-boron mixture (80% lead, 1% boron by weight). It defined a circular field with a diameter at the exit surface of either 5 or 8 cm.

The measurement of angular distributions for the two neutron producing reactions $^9\text{Be}(d,n)^{10}\text{B}$ and $^9\text{Be}(^3\text{He},n)^{11}\text{C}$ have been compared to time of flight measurements performed at the Argonne National Laboratory (2) using a Tandem Van de Graaff accelerator which produced neutron beams very similar to those generated at the MSKCC cyclotron. The comparison of these measurements in Table I shows close agreement between these two methods which differ, by at most, $\pm 5\%$. A comparison is also shown in Figure III-2.

TABLE I

Relative angular distributions in total dose measured
by T.E. ion chamber and number of neutrons/unit
energy by time-of-flight at Argonne National
Laboratory (Kuchnir, et al. 1972)

Angle	0°	15°	30°	60°
T-O-F	100%	76	47	24
Chamber	100%	76	52	26

Angle	0°	15°	30°	60°
T-O-F	100%	81	62	28
Chamber	100%	86	65	~30

+ T.O.F

Figure III-2

b. Transmission Measurements

Dose attenuation curves for the ${}^9\text{Be}({}^3\text{He},n)$ and ${}^9\text{Be}(d,n)$ fields were measured in the experimental arrangement shown schematically in Figure III-3. The 0.8 cc T.E. ionization chamber was positioned 45 cm in front of an 8 cm diameter, collimated field. Steel, lead and polyethylene-lead-boron (80% Pb, 1% B by weight) bricks (cross sectional dimensions of 10 x 20 cm) were placed between the collimator face and the ionization chamber. The initial brick was placed 10 cm in front of the ionization chamber. Additional bricks were added as shown in the diagram. The results for the two fields and the three materials are summarized in Fig. III-3.

The data were fit to a simple exponential function of the absorber thickness plus a constant. This constant term is added to account for the tailing observed in the data at larger thicknesses which arises from scatter within the absorbing media, scatter from the walls of the room, and some variation in the absorption efficiency of a given material as the neutron energy spectrum is degraded with thickness. The resulting fits are shown as solid lines in Fig. III-3. The corresponding values of the linear dose attenuation coefficients, μ , for the deuteron induced field are shown in Table II.

TABLE II

Narrow Beam Linear Dose Attenuation Coefficients

(In cm^{-1})

<u>Reaction</u>	<u>Particle Energy</u>	<u>Approx. Neutron Energy</u>	<u>Steel</u>	<u>Lead</u>	<u>Polyethylene Mixture</u>
${}^9\text{Be}(d,n){}^{10}\text{B}$	8 MeV	3 MeV	0.18	0.14	0.15
${}^9\text{Be}({}^3\text{He},n){}^{11}\text{C}$	23 MeV	8 MeV	0.17	0.13	0.12

Figure III-3

The constant term was small in all cases, ranging from 0.5 to 1.2 percent. The values of μ are consistent with those measured for neutron fields produced by deuterons, incident upon a beryllium target, with energies between 16 and 50 MeV (3). The μ for steel ($\mu \approx 0.18$) appears to be approximately independent of the incident deuteron energy at $8 < E_d < 50$ MeV, while that for hydrogenous materials increases as the average energy of the neutron spectrum decreases. This suggests that bulk shielding material might be composed of the minimum thickness of steel necessary to degrade the neutron energy spectrum to a region where lightweight hydrogenous materials are equally efficient.

2. Neutron and Gamma-Ray Components of the ($^3\text{He}, n$) and (d, n) Fields

a. Paired Chamber Technique

The response of the previously described ionization chambers, Q (in coulombs), in a mixed n - γ field is given by (3).

$$Q = X_\gamma D_\gamma + X_n D_n \quad (1)$$

where D_γ and D_n are the γ and n tissue doses (rads) and X_γ and X_n are the absorbed dose conversion coefficients (coulomb/rad) for γ -rays and neutrons. The values of X_i ($i = \gamma$ or n) may be calculated using the Bragg-Gray relation, and assuming the atomic composition of the chamber wall is identical to tissue:

$$X_i = \frac{Q_i}{D_i} = \frac{100 \text{ pV}}{W_i (S_g^t)_i} \quad (2)$$

where V is the volume (cm^3) of the filling gas with density ρ (gm/cm^3), W_i is the average energy per unit charge formed in the gas (ergs/coulomb) and $(S_g^t)_i$ is the effective mass stopping power ratio of tissue to gas.⁸ Values of X_γ equal to 3.14×10^{-10} and 2.15×10^{-11} coulombs/rad ($\pm 1.5\%$) were measured for the 0.8 cm^3 and 0.06 cm^3 T.E. chambers by placing them at a position of charged particle equilibrium in a polystyrene phantom and exposing them in a calibrated ^{60}Co field.

The value of X_n was calculated from the ratio X_n/X_γ which, from equation (2), is equal to $(W_\gamma (S_g^t)_\gamma) / (W_n (S_g^t)_n) - 1$. Values of W_n were obtained by integrating the mean W_n values calculated at discrete neutron energies by Dennis (4) over the energy spectra of reference (2). This yielded W_n values

for the ^3He and deuteron induced fields which differed by approximately 1.5%. An average value, $w_n = 31.4$ eV/ion pair was used here. The effective mass stopping power ratio is equal to unity in a homogeneous chamber for both the neutrons and the gamma rays. Using these values and setting $w_\gamma = 29.9$ eV/ion pair (5), the ratio X_n/X_γ was calculated to be 0.927. An uncertainty of 2% is estimated for this value.

The third chamber (designated Al-Ar) was designed to have a volume approximately equal to that of the 0.06 cc T.E. chamber. It was constructed of aluminum with a wall thickness of 0.50 mm, and was filled with argon. X_γ for this chamber was measured, as above in a ^{60}Co field, to be 3.36×10^{-11} coulomb/rad ($\pm 1.5\%$).

The neutron sensitivity of this chamber was estimated in two ways. In the first, X_n , was calculated from the ratio of neutron and gamma ray sensitivities as described above. However, in this case, equation (2) must be modified to account for the ratio, F , of the mass energy-absorption coefficient of aluminum to that of tissue for ^{60}Co gamma-rays, and the ratio of kerma, k , for fast neutrons in argon and tissue so that: $X_n/X_\gamma = (w_\gamma(SW)_\gamma k) (w_n(SW)_n F)^{-1}$. Values of $w_\gamma = 26.2$ eV/ion pair (5), $w_n = w_p = 26.4$ eV/ion pair (5), $(SW)_\gamma = 1.09$ (6), $(SW)_n = (SW)_p = 1.16$ (7) and $F = 0.87$ (7) were used in the calculation. The kerma for argon, as a function of neutron energy, was estimated including only the elastic scattering process. The cross-section was assumed to be equal to the total cross section values given in ref. (9). All angular distributions were assumed to be isotropic in the center of mass and the average recoil energies were calculated. The ratios of these kerma values to those for tissue given by Bach and Caswell (10) were averaged over the energy spectra yielding average k values of 1.9×10^{-2} and 2.4×10^{-2} for the deuteron and ^3He induced spectra, respectively. The corresponding values of X_n/X_γ were calculated to be 2.0×10^{-2} and 2.6×10^{-2} , respectively. A large uncertainty of $\pm 25\%$ is assigned to these values because of the numerous approximations which were made.

The second estimate was based upon the X_n/X_γ values measured by Goodman (11) for an argon filled aluminum chamber at neutron energies of 0.6, 2, 5 and 15 MeV. These data were plotted as a function of neutron energy and an arbitrary smooth curve passed through them. Average values of X_n/X_γ were then obtained by integrating over the neutron energy spectra of ref. (2). The value obtained for the deuteron induced spectrum, $1.6 \pm 0.4 \times 10^{-2}$, is in reasonable agreement with that calculated above, while that obtained for the ^3He induced field, $4.3 \pm 1 \times 10^{-2}$, is somewhat larger. In each case, an

average of the two estimated values, 1.8 ± 0.5 and $3.3 \pm 1.0 \times 10^{-2}$, was adopted. These relatively small values of neutron sensitivity allowed this chamber to be used in conjunction with the T.E. chamber to estimate the gamma ray contaminant present in the total dose.

b. TLD Measurements

Lithium fluoride and calcium fluoride loaded teflon thermoluminescent dosimeters were also used to study the level of gamma ray contaminant in the neutron fields. The ^7LiF and $\text{CaF}_2\text{:Mn}$ teflon discs³ were 0.13 mm thick and 12.7 and 6 mm in diameter respectively. They were read out 24 hrs after exposure with a Con-Rad Model 7100 TLD reader. Annealing after readout followed the procedures outlined in reference (12). The discs were exposed, both in the neutron fields and the ^{60}Co calibration field, encapsulated in 110 mg/cm² of aluminum.

The response of these dosimeters in a mixed n- γ field is also given by equation (1). Estimates of the ratios X_n/X_γ are available in the literature. Bewley and Parnell (13) measured the limits of the ratio for both dosimeters to be 0.06-0.09 at an $E_n \approx 7$ MeV and Wingate et al. (14) have measured the LiF ratios as a function of neutron energy. On the basis of these results X_n/X_γ values of 0.075 in the ^3He and 0.035 in the deuteron induced fields have been assigned, with an error of approximately $\pm 20\%$.

Figure III-3 shows the percentage of the total dose at a given depth which arises from the gamma ray contamination. These results were obtained by solving pairs of simultaneous equations of the form given by equation (1). T.E. ionization chamber data were paired with those obtained with the Al-Ar chamber (the solid lines), the ^7LiF TLD's (the solid data points) and the CaF_2 TLD's (the open data points), respectively. In the deuteron induced field, reasonable agreement is found between the two types of dosimeters at depths of ~ 1 cm to 10-12 cm. Beyond this depth, the TLD results diverge from those obtained with the Al-Ar chamber. In the ^3He induced field, the TLD results lie consistently ($\sim 4\%$) below those measured with the ionization chamber. These differences may be due to the directional sensitivity of the TLD's, which would result in a low estimate of the gamma ray dose, and/or, in the case of the ^3He induced field, an under estimate of the Al-Ar chamber neutron sensitivity, X_n/X_γ , of this chamber, from the $3.3 \pm 1.0\%$ used here, to 8% would result in the data obtained with the two types of dosimeters being in reasonable agreement.

3. Radiobiological Measurements

a. HeLa Cell OER and RBE

HeLa S-3 cells, plated in plastic flasks, were irradiated simultaneously in the fast neutron fields at depths of 2.2, 7.4 and 11.8 cm in a 50x50x70 cm water filled polystyrene tank. The flasks were suspended by their caps and oriented such that the primary dose originated from the secondary charged particles generated in the cell medium rather than in the plastic.

In the case of the cells in suspension, the ampules were mounted to polystyrene discs which were placed at depths of 2 and 12 cm in another water filled polystyrene tank (30x30x35 cm). These discs were rotated at 10 revolutions per minute in order to maintain the cells in suspension and thereby assure that the primary dose was generated in the cell medium rather than in the glass.

The total dose rates were measured at each depth before every irradiation period with tissue equivalent ionization chambers. In the ^3He induced field the dose rate varied from approximately 16 rads/min at the 2 cm depth to approximately 4.5 rads/min at 12 cm. In the deuteron induced field this variation was from approximately 40 rads/min to 10 rads/min.

^{60}Co survival curves were obtained by irradiating the suspended cell system at a depth of 2 cm in a 16x16 cm field at an SSD of 80 cm. The dose rate was approximately 80 rads/min. These exposures were performed on the same day as the fast neutron irradiations and with cells prepared at the same time.

Following the procedures described previously (I) the plated and suspended cell systems were exposed to both fast neutron fields a minimum of two times, over a period of months. The plated cell results are shown in Figure III-4, where the survival data, S , obtained at the three depths are plotted as a function of the neutron dose, D_n . The sample error bars result from a $\pm 15\%$ uncertainty assigned to the colony scoring and multiplicity correction and a $\pm 3\%$ uncertainty in the dose reproducibility.

These data show the functional dependence of percentage survival on neutron dose to be, within the scatter of the data, insensitive to depth. The solid lines are fits to the combined data of a simple exponential function, $S = ne^{-D_n/D_0}$, using the least square fitting technique described in ref. (7), and yield values of $D_0 = 55 \pm 1$ rads $n = 0.9 \pm 0.1$ and $D_0 = 54 \pm$ rads $n = 0.7 \pm 0.1$ in the ^3He and deuteron induced fields respectively. The data obtained at the individual depths were also fit as a function of the total dose (neutron plus gamma ray). The resulting parameters are summarized in Table III.

An example of the data obtained with the cell suspension system in the deuteron induced fast neutron and ^{60}Co fields is shown in Figure III-5. The errors in this case arise from $\pm 10\%$ and $\pm 2\%$ uncertainties in the colony scoring and dose reproducibility, respectively. These survival curves were fit, as a function of the total dose, D , to $S = 1 - (1 - e^{-D/D_0})^n$ where n and D_0 are the extrapolation number and the 37 percent dose-slope which characterize a given curve, respectively. The results are summarized in Table III-1.

The D_0 values obtained with the two cell systems are in good agreement. The low extrapolation numbers which characterize the plated cells may be a result of too large a multiplicity correction arising from incorrect scoring and/or cell clumping. The RBE values as determined from the suspended cell data at a 50% survival level, are, in terms of the ratios of total dose, 2.7 and 2.5 at the 2 and 12 cm depths in the ^3He induced field and 3.2 and 2.9 at the corresponding depths in the deuteron induced field.

The principle measured variation in the composition of the total dose as a function of depth in these two fast neutron fields is the increasing fraction due to gamma radiation. The distribution in linear energy transfer of the dose arising from the heavy secondary charged particles, protons and alpha particles, over the range of depths involved shows little variation within either the deuteron or ^3He induced fields. An estimate of the biological effect of this gamma ray dose increase can be made by assuming that the "gamma ray" and "neutron" doses act independently and that the resulting survival curve is a product of the two corresponding multi-target single hit type functions. Such an approximation allows the inhibition of clone formation due to the gamma-ray dose to be estimated but assumes the "sublethal damage" arising from the neutron and gamma ray doses to be non-additive. Thus, this gives the minimum expected gamma-radiation response and is reasonable only at small gamma ray doses like those involved here (~ 60 rads). In order to make such an estimate, parameters for the neutron dose, $D_0(N)$ and n , were obtained by fitting the 2 cm depth suspended cell data to

$$S = (1 - (1 - e^{-xD/D_0(\gamma)})^{n_\gamma}) (1 - (1 - e^{-(1-x)D/D_0(N)})^{n_N})$$

where $D_0(\gamma)$ and n_γ were taken from the measured ^{60}Co survival curve and the value of x is the measured percentage of the total dose, D , due to gamma rays. Assuming that these parameters do not vary with depth, (i.e. the fast neutron energy spectrum does not vary) survival data were calculated from the above equation for a range of x values and fit to a multi-hit single target type equation.

TABLE III-1

HeLa Cell Survival Curve Parameters

Depth (cm)	Cell System	$^9\text{Be}(^3\text{He},n)^{11}\text{C}$ Fast Neutrons		$^9\text{Be}(d,n)^{10}\text{B}$ Fast Neutrons		^{60}Co Gamma Rays	
		D_0 (rads)	n	D_0 (rads)	n	D_0 (rads)	n
2	Suspension	56 ± 2	2.5 ± 0.3	56 ± 1	1.9 ± 0.2	146 ± 4	2.6 ± 0.3
2.2	Plated	57 ± 2	0.9 ± 0.1	53 ± 1	0.8 ± 0.1		
7.4	Plated	61 ± 2	0.9 ± 0.1	59 ± 2	0.7 ± 0.1		
11.8	Plated	59 ± 2	1.3 ± 0.1	63 ± 3	0.6 ± 0.1		
12.0	Suspension	59 ± 2	2.4 ± 0.3	68 ± 2	1.6 ± 0.2		

The results of this estimate are shown in Figure III-3, where the measured D_0 values (in units of total dose) are plotted as a function of depth and compared with those calculated (dashed lines) for the level of gamma ray contaminant measured at the corresponding depth. The calculated extrapolation numbers showed no significant variation over this range of depths. It appears that the depth dependence of D_0 observed in these fields is reasonably consistent with that calculated on the basis of the stated assumptions, and that, within the accuracy of the data presented, no variation in the biological effectiveness of the neutron dose with depth in a water phantom is observed.

b. Experimental Animal Radiotherapy

The relative response of Rigeway Osteogenic Sarcoma, borne in the hind limbs of mice, was studied using fractionated treatments of neutrons and x-rays followed by local tumor hyperthermia (42.5°C). The fractions were delivered three times a week (MWF) for one week. Neutrons generated by the $^9\text{Be}(d,n)^{10}\text{B}$ reaction ($E_n = 3.5$ MeV) were found to have an RBE of approximately 2.2 relative to 300 kVp x-rays in the range of doses from approximately 10 to 75 neutron rads per fraction. This result is consistent with reports from other cyclotron produced fast neutron studies.

REFERENCES:

1. Annual Rpt. to the United States Atomic Energy Commission, Biological and Clinical Dosimetry, AEC Contract AT(11-1)-3521, 1972-73.
2. Kuchnir, F.T., Skaggs, L.S., Elwyn, A.J., Mooring, F.P., and Frigerio, N.A. Design of a neutron therapy facility for a 30-inch cyclotron. Cyclotrons 1972, Proc. Sixth Internat. Cyclotron Conf., Vancouver, Canada, July 1972, pp.638-645, American Institute of Physics, New York, 1972, and private communication.
3. Attix, T.H., Theus, R.B., Shapiro, P., Surratt, R.E., Nash, A.E., and Gorbics, S.G. Neutron beam dosimetry at the NRL cyclotron. Phys. Med. Biol. 18:497-507, 1973.
4. Dennis, J.A. Computed ionization and kerma values in neutron irradiated gases. Phys. Med. Biol. 18:379-395, 1973.

5. Myers, I.T. Radiation Dosimetry (ed. F.H. Attix, W.C. Roesch and E. Tochilin), Vol. I, 317-330, Academic Press, New York, 1968.
6. Berger, M.J. and Seltzer, S.M. Tables of energy losses and ranges of electrons and positrons. NASA SP-3012, 1964.
7. Barkas, W.H. and Berger, M.J. Tables of energy losses and ranges of heavy charged particles. NASA SP-3013, 1964.
8. Physical Aspects of Irradiation. N.B.S. Handbook 85, 1962.
9. Stehn, J.R., Goldberg, M.D., Magurno, B.A., and Weiner-Chasman, R. Neutron Cross Sections, Vol. 1, BNL 325 Second Ed., Suppl. No. 2, 1964.
10. Bach, R.L. and Caswell, R.S. Energy transfer to matter by neutrons. Rad. Res. 35:1-5, 1968.
11. Goodman, L.J. Private communication, 1974.
12. Attix, F.H. and Cameron, J.R. Manual on Radiation Dosimetry. (Ed. N.W. Holm and R.J. Berry), pp. 397-409, Dekker Inc., New York, 1970.
13. Bewley, D.K. and Parnell, C.J. The fast neutron beam from the M.R.C. cyclotron. Brit. J. Radiol. 42:281-288, 1969.
14. Wingate, C.L., Tochilin, E., and Goldstein, N. Response of lithium fluoride to neutrons and charged particles. AEC Conf. 650637, 421-434, 1967.

IV. MICRODOSIMETRIC MEASUREMENTS

(J.C. McDonald, T.R. Canada, B.R. Freed, and J.S. Laughlin)

1. Experimental Apparatus

A commercially produced, 12.7 mm diameter, spherical proportional counter of the Rossi type (1) was used to measure the spectral distribution of event size at various positions within the neutron fields. The wall of the chamber is constructed of T.E. plastic, 1.27 mm thick, and is enclosed in a 0.18 mm thick aluminum shield. The counter was filled with the T.E. gas described above to a pressure of 5.25 mm Hg, giving an effective diameter (d) of 1.0 μ m.

The chamber was operated at a potential difference of 550 volts. Standard low noise electronics were used to amplify the counter signals which were analyzed and stored in a 256 channel multichannel analyzer. The analyzer is interfaced to an IBM 1800 computer making the data available for immediate analysis. The counter was calibrated with an internal ^{241}Am alpha source which gave a peak at 84.8 KeV/ μ .

Simple total dose characteristics of high linear energy transfer (LET) radiation fields are inadequate to describe the observed response of biological systems exposed to them. This response is found to have a strong dependence upon the LET of the radiation. In the case of a fast neutron field, the dose is deposited by charged secondary particles which have a large range of LET values. In hydrogenous materials, neutrons, in the energy range of interest here, lose their energy primarily by elastic scattering with protons. The recoil protons have a minimum LET depending upon the maximum energy available and a maximum LET (~ 90 keV/ μ) at the Bragg peak. Similar behavior is exhibited by alpha particles, produced in (n, α) reactions, which have a maximum LET at the end of their tracks of approximately 200 keV/ μ . Heavier reaction products and recoiling nuclei are more densely ionizing, resulting in LET values of greater than 200 keV/ μ . Any variation of the neutron energy spectrum will alter this complex distribution of dose in LET and may result in a change in the biological response.

To study the distribution of dose in LET for small tissue equivalent volumes, event-size distributions were obtained using the spherical proportional counter described above, with an effective diameter of 1 micron. Because the range of event-sizes exceeds the dynamic range of the multichannel analyzer, it was necessary to obtain the data at three different gain settings. The data were reduced by a computer program which smoothed and combined the spectra, using a cubic spline technique.

A typical event-size distribution, $N(Y)$ versus Y , measured with a Rossi-type spherical proportional counter, spans a range of counts per channel of five to six orders of magnitude and a Y range of three to four orders of magnitude. Because the latter exceeds the dynamic range of available multichannel analyzers, it is necessary to obtain the data in several overlapping spectra. Various distributions derived from these data involve the products of Y or Y^2 with $N(Y)$ and the derivative $dN(Y)/dY$. These distributions are very sensitive to any local statistical fluctuations in the data. This Appendix describes a computer program which smooths the input data, combines overlapping spectra, and calculates a wide variety of parameters from these data.

Event spectra are either transferred directly from the multichannel analyzer to the disk storage of an IBM 1800 MPX system, or punched on paper tape. From paper tape or disk storage, a punched card deck and raw data listing are provided by an auxiliary program. All input to the main program is via punch cards.

The main program, written in Fortran, accepts a maximum of 3 spectra, each containing up to 256 channels. Each spectrum must overlap the adjacent one for purposes of normalization. The user specifies the following parameters: the number of spectra; the lower and upper channel limits of data to be used for each spectrum; the calibration constant (keV/ μ /channel) for each spectrum; the form of data weighting to be used in the smoothing routine (the exponent of the counts per channel); overall range of LET within which all distributions are to be normalized; distributions to be plotted (discussed in detail below); and the number of decades desired for the log ordinate of the selected plots (if not specified, the minimum number necessary is determined by the program).

Each spectrum is smoothed individually, using a cubic spline curve fitting technique developed by Powell (2). Initially, several multi-point averaging formulas were tried (linear, parabolic, gaussian) but were found to be unsatisfactory. When used a sufficient number of times to produce an adequately smooth first derivative, statistically unacceptable trends away from the raw data were introduced. Powell's subroutines VCO3A and VBO6A (2) fit the data in sections, delineated by "knots" which are generated in the program, to a cubic function in such a fashion that the overall fit has a smooth first derivative and a continuous second derivative.

Two adjacent spectra, i and $i + 1$ obtained at gain settings of C_i and C_{i+1} keV/ μ /channel respectively ($C_{i+1} > C_i$), are normalized with respect to one another by

fitting a straight line to the number of counts/channel in spectrum $i+1$ versus the number of counts in the appropriately summed "narrower" channels of spectrum i . If the intercept of this fit is unacceptable large, spectrum $i+1$ is shifted relative to spectrum i and the data is refit. This procedure is continued in an iterative process, until the absolute value of the intercept is suitably small. The resulting relative zero offset, B_{i+1} , is generally less than 1.5 channels, if the values C_i are accurate. The inverse of the slope of the line is defined as the relative intensity-time normalization T_{i+1} . As an example of this procedure, Fig. IV-1 shows the overlap region of two spectra, differing in gain by a factor of approximately four. The value of B_{i+1} in this case is 1.4 channels. The smooth curve is the cubic spline fit to the C_{i+1} data.

Following the smoothing and normalization, normalized distributions of frequency in event size, $P(Y)$, dose in event size, $D(Y)$, dose in LET, $D(L)$, and the cumulative $D(L)$ distribution, $E(L)$, are calculated from

$$Y_{ij} = C_i (j - B_i)$$

$$P(Y)_{ij} = \frac{T_i Q_{ij}/C_i}{\sum_{i,j} T_i Q_{ij}}$$

$$D(Y)_{ij} = \frac{T_i T_{ij} Q_{ij}/C_i}{\sum_{i,j} T_i Y_{ij} Q_{ij}}$$

$$D(L)_{ij} = \frac{T_i Y_{ij} Y_{i,j+1} (Q_{ij} - Q_{i,j+1} Y_{ij}/Y_{i,j+1})/C_i^2}{\sum_{i,j} T_i Y_{ij} Y_{i,j+1} (Q_{ij} - Q_{i,j+1} Y_{ij}/Y_{i,j+1})/C_i}$$

$$E(L)_{ij} = 1 - \sum_{L_{min}}^L C_i D(L)_{ij}$$

WHERE

- i = spectrum number (1 to 3)
- j = channel number (1-256) of spectrum i
- Q_{ij} = smoothed counts in spectrum i , channel j
- C_i = keV/ μ /channel in spectrum i
- B_i = zero offset of spectrum i ($B_1 = 0$)
- T_i = intensity-time normalization factor for spectrum i ($T_1 = 1$)
- Y_{ij} = event size (keV/ μ) represented by channel j , spectrum i ; corresponds to end-point of triangular distribution of track-lengths for LET = Y_{ij} .

Figure IV-1

In addition to these distributions, values of the mean LET and the absorbed dose are calculated. The expression for $D(L)$ is adapted from that of Rossi (3)

$$D(L) = \int_0^L Y N(Y) dY - Y^2 \frac{dN}{dY} \bigg|_{Y=L}$$

to multichannel analyzer data taken with a gain C keV/ μ /channel. In this case, assuming for two adjacent channels (Y_1, N_1) and (Y_2, N_2) , that $dN/dY = (N_2 - N_1)/(Y_2 - Y_1)$ then the dose fraction per unit LET is given by

$$D(L_1) = \text{const.} \frac{L_1 L_2 (N_1 - N_2 L_1 / L_2)}{C^2}$$

The results of these calculations are listed and plots of the superimposed raw and smoothed count data, $D(Y)$, $D(L)$, $Y \cdot D(Y)$, $L \cdot D(L)$, and $E(L)$ are available to the user.

2. Results and Discussion

An example of the event size distribution, $N(Y)$ vs Y , obtained in this fashion is shown in Fig. IV-1. These data were obtained at a depth of 2 cm in a water phantom placed in the ^3He induced field. Figure IV-2d shows the normalized distribution of dose in event size, $D(Y)$, times Y versus $\log Y$. Figure IV-2b shows the distribution of dose in linear energy transfer, $D(L)$, times the linear energy transfer, L , as a function of $\log L$. The latter was obtained following the procedure of Rossi (3); i.e. assuming a triangular distribution of event size. The function $E(L) = 1 - \frac{L}{L_{\min}} \int_0^L D(L) dL$ is shown in Fig. IV-3. These data clearly show that, with this neutron energy spectrum, the neutron energy is absorbed primarily in the (n,p) scattering reaction ($\sim 90\%$).

In order to obtain some qualitative information concerning the variation of the neutron dose quality as a function of position in the field, event spectra were obtained at a number of points along the depth axis and in the penumbra of the deuteron and ^3He -induced collimated fields. The measurements were made in water and included an event size region of from 6 to 250 keV/ μ .

The results of these measurements are summarized in Figures IV-4 and IV-5. In both fields the distribution of dose in LET shows a small increase at lower values of LET and a corresponding decrease at values between 15 and 90 keV/ μ as the depth in phantom is increased. This indicates a hardening of the neutron energy spectrum with depth which is consistent

Figure IV-2

Figure IV-3

Figure IV-4

Figure IV-5

with the activation foil data presented above. The large displacement of dose to higher values of LET 2 cm into the penumbra is the expected consequence of the average neutron energy being significantly reduced due to scattering in the collimator and phantom.

This reduction in energy will result in an increase in the relative biological effectiveness of the neutron dose in the penumbra relative to that on axis, while the small variation observed with depth is expected to be of little biological consequence.

REFERENCES:

1. Rossi, H.H. and Rosenweig, W. A device for the measurement of dose as a function of specific ionization. Radiology 64:404-411, 1955.
2. Powell, M.J.D. Curve fitting by splines in one variable. IN: Numerical Approximation to Functions and Data (ed. J.G. Hayes) pp. 65-83, Athlone Press, London, 1970.
3. Rossi, H.H. Radiation Dosimetry. Ed. F.H. Attix W.C. Roesch and E. Tochilin. Vol. I, 43-92, Academic Press, New York, 1968.

V. DOSIMETRY OF ^{252}Cf MEDICAL NEUTRON SOURCES

(L. Anderson and J. S. Laughlin)

Between January and April, 1974, three patients were implanted with ^{252}Cf . A fourth patient had both an ^{192}Ir and a ^{252}Cf implant. All of these patients had metastatic lesions, previously untreated. Table I summarizes the pertinent data relative to implantation, tumor dose, tumor size, and tumor or normal tissue response for the short period of follow-up. Tumor regression vs. time elapsed since implantation is shown in Figure V-1. The median regression (50%), as expected, varies from case to case. The technique of implantation is similar to the one previously described for removable iridium implants (1).

1. Dosimetry

For the patients treated to date with ^{252}Cf , an interim implant technique has been used, with thin-wall stainless steel sources (ALC-S4X) contained in gold tubing which is after-loaded into surgically-implanted plastic tubing. The sources are spaced at 1 cm between centers with the gold tubing extending 1 cm beyond the terminal source on each end. Nylon fishline is used for spacing and to provide a long leader on one end and a short leader on the other. In the afterloading procedure, all of the long nylon leaders are inserted through the implanted tubes before any of the gold tubes is withdrawn from the shield. Each source tube is then drawn into its implanted tube where it is stopped in the correct position by a metal button which is affixed to the short leader. The number of sources per tube and the position of the stop is determined from tube length measurements as shown in Table II.

Orthogonal radiographs are taken of the implanted region either with dummy sources in place or, if a final verification of position is deemed necessary, with the gold tubes in position. In the latter case, since individual sources are not visualized radiographically, film coordinates of randomly-chosen points along each gold tube are digitized and a special computer program is used to determine the coordinates of both the centers and the ends of the actual (4 mm active length) sources.

The Memorial Hospital computer program BRACHY is used to compute ^{252}Cf dose distributions. For determining the neutron dose, the program considers the sources to be point sources and makes use of an interpolation table based on the formula

$$D_n = \frac{2.58}{r^2} e^{-0.823 r}$$

where D_n is the neutron dose in rad/ μ gh at a point r cm from the source. This formula represents measurements by Colvett and co-workers (2). In the case of the beta and gamma ray dose computations, oblique filtration in the gold tube makes necessary a line source calculation, using the film coordinates of source ends and a two-dimensional interpolation table based on thermoluminescent dosimetry measurements at Memorial Hospital.

The objective in ^{252}Cf implants at Memorial Hospital is to deliver the same effective dose as that delivered in ^{192}Ir implants, which provide the basis for comparison in the clinical study. The ^{192}Ir minimum effective dose to the tumor volume is 6000 rad in 6 days, for a dose rate of 41.7 rad/h. Therefore, the ^{252}Cf dose rates r_n from neutrons and $r_{\beta\gamma}$ from beta and gamma rays must satisfy the relation

$$R_n r_n + R_{\beta\gamma} r_{\beta\gamma} = 41.7 \text{ rad/h}$$

where R_n and $R_{\beta\gamma}$ are the corresponding RBE's, both relative to ^{192}Ir radiation at 41.7 rad/h.

RBE values are derived from a cell-survival study by Hall (3) which suggested an RBE of 6.7 for ^{252}Cf neutrons relative to ^{226}Ra gamma rays at 35.7 rad/h (6000 rad in 7 days). The same study elucidated the effect of dose rate on cell survival for ^{226}Ra irradiations and thus permitted us to determine dose-rate corrected RBE's for both neutron and beta-gamma components of ^{252}Cf radiation. The resulting values are $R_n = 6.4$ and $R_{\beta\gamma} = 0.86$.

The ratio $r_n/r_{\beta\gamma}$ was found to have an average value of 1.12 over a plane 0.5 cm from the source plane of an idealized "typical" planar implant consisting of four lines of five sources each with 1 cm spacing in both directions. This ratio was used to solve the above equation, with the results $r_n = 5.8$ rad/h and $r_{\beta\gamma} = 5.2$ rad/h. The corresponding doses for a 6 day treatment are $D_n = 840$ rad and $D_{\beta\gamma} = 750$ rad.

Clinical dosimetry for ^{252}Cf patients has been made more comprehensive by the use of special localization and computation techniques. Both transverse tomography and orthogonal radiography have been explored as localization methods for skin dosimetry. In each case thin lead wires on the skin surface are visualized relative to source locations. Orthogonal radiography has the advantage of requiring no additional films beyond those necessary for tumor dosimetry. It makes use of the same computer program used for localizing sources in gold.

tubes, i.e., for translating randomly digitized points along curvilinear images in each film into the coordinates of equally spaced points along actual wires or tubes. A further auxiliary computer program has been developed for the precise rotation of the implant plane (defined by three specified source points) into a plane parallel to the computational plane. This feature permits a more definitive statement of dose in the 0.5 cm plane and hence a more reliable intercomparison of implants.

2. Radiation Protection:

Dose rates measured near the four patients implanted are summarized in Table 1. The "surface" values were measured at the distance of closest approach for the instrument used. For the neutron measurement the detector probe was of the moderator type, nine inches in diameter. Dose rates were also measured in rooms adjoining those of the implanted patients and on the floors above and below. These measurements yielded dose rates, even for these relatively small loadings, of approximately 2.0 mrem/h (assuming a Q.F. of 1 for the gamma radiation), which is the current recommended limit for uncontrolled areas. For future implants involving greater total activities additional shielding will be required.

Measurements were also made using film packs placed near the eyes of patients in an attempt to measure dose to the lens of the eye. The patient (No. 1) with an implant in a groin node had reported doses of only 240 mrem neutron and 270 mrem γ -ray. In one patient (No. 2), the presence of the ^{192}Ir sources made the γ -ray background of the nuclear trace film so grainy that no neutron dose could be reported. In another case (No. 4), the patient's spouse was in nearly constant attendance, acting as a private duty aide, but spending most of the time at the door of the patient's room. The doses received by the spouse, determined from a standard film badge worn throughout the six day period, were 220 mrem neutron and 60 mrem γ -ray. The reported doses to the eye monitors for the patients with implanted neck nodes were approximately 5 rem γ -ray and 25 rem neutron for a 6 day treatment time.

REFERENCES:

1. Henschke, U.K., Hilaris, B.S. and Mahan, G.D. Afterloading in interstitial and intracavitary radiation therapy. Amer. J. Roentgenol. Rad. Ther. Nucl. Med. 90:386, 1963.

2. Colvett, R.D., Rossi, H.H. and Krishnaswamy, V.
Dose distributions around a californium-252 needle.
Phys. Med. Biol. 17:356, 1972.
3. Hall, E.J. A comparison of radium and californium-252
using cultured mammalian cells. Radiology 102:173,
1972.

TABLE 1. CLINICAL EXPERIENCE WITH ^{252}Cf AT MEMORIAL HOSPITAL

Patient	Diagnosis	Site & Date of Implant	Planar Implant			Dose (rad) at 0.5cm*			Tumor Size (cm)	Skin Dose* (rad)	Skin Reactions	Tumor Regres- sion
			# tubes	# sources	Activity	n	8y	^{192}Ir equiv				
1	metastatic Ca vulva	rt.groin 1/10/74	4	20	7.3 μg ^{252}Cf	840	620	5900	3.8x2.3 x1.6	840 n	erythema, pigmentation	complete
2	metastatic Ca submaxi- lary gland	lt. neck 2/8/74	4	12	4.3 μg ^{252}Cf	1075	1000	7800	1.4x1.4 x1.2	1030 n	erythema, pigmentation	incomplete
2	metastatic Ca submaxi- lary gland	ant.neck 2/8/74	4	12	6.5 mg Ra Eq ^{192}Ir	-	-	5400	1.4x1.4 1.2	3800 Y	erythema pigmentation	incomplete
3	metastatic Ca naso- pharynx	lt.neck 2/13/74	4	18	6.4 μg ^{252}Cf	1240	1120	8500	2.8x2.8 x2.1	850 n	pigmenta- tion	complete
4	metastatic Ca nose	rt.neck 4/19/74	5	29	9.8 μg ^{252}Cf	1250	950	8800	5x5 x2.3	390 n	pigmenta- tion	incomplete

* Opposite center of implant plane. Dose at 0.5 cm from implant plane is average of doses above and below.

POSITIONING GUIDE FOR REMOVABLE INTERSTITIAL IMPLANTS

^{252}Cf

^{192}Ir Direction sources to be inserted

DIMENSIONS IN CENTIMETERS

Suggested BC EF

[illegible]

- 1) $BF = AG - (AB+FG)$
- 2) ^{192}Ir : $CE = n-0.7$
 ^{252}Cf (gold tube): $CE = n + 1.6$
 ^{252}Cf (plastic tube): $CE = n-0.4$
- 3) Sources centered: $BC = EF = \frac{BF-CE}{2}$

Sources not centered: $BC = AD - \frac{CE}{2}$

$$EF = BF - (BC + CE)$$

- 4) $AC = AB+BC$, $EG = EF+FG$

*AD to be measured in OR if intraoral loop implant; otherwise, AD to be determined from radiograph of dummy sources at known spacing, starting at A.

Table III. Radiation Levels Near Patients with ^{252}Cf Implants

<u>Patient</u>	<u>"Surface" Dose Rates</u> (mrem/h)		<u>Dose Rates at 1 meter</u> (mrem/h)	
	<u>Neutron</u>	<u>γ-ray</u>	<u>Neutron</u>	<u>γ-ray</u>
1	500	180	5	1
2	300	1500	10	10
3	600	380	20	2
4	1000	400	20	15

FIGURE V-1

Tumor regression for ^{252}Cf patients at Memorial Hospital.

VI. GERMANIUM GAMMA CAMERA

(J. Detko and J. S. Laughlin)

1. Introduction

Effort in the solid state detector program has been focused on the development and construction of a germanium gamma ray camera utilizing ultra pure germanium and an orthogonal strip geometry developed earlier in this program for such use. During the current period a major focus of experimental work has been the determination of the degree of operational stability that can be achieved for the orthogonal strip detector. Adequate stability of detector response characteristics is obviously indispensable to the construction of a practical camera for clinical use. Accumulated experimental response data has caused increased concern that the long term stability of the orthogonal strip matrix diode might not be adequate relative to the requirements of a practical gamma ray camera. This concern was also heightened by the implications of work done elsewhere involving both lithium drifted and ultra pure germanium diodes (1). This report describes some of the problems connected with matrix instability and provides preliminary results that indicate an available method for substantially improving stability of the orthogonal strip gamma camera.

2. Background

The early phases of this work dealt with the development and testing of a detector geometry suitable for use with the large array of diodes which would be required in a clinically useful gamma camera. An array of individual diodes was considered impractical. Among the diode configurations examined was the orthogonal strip geometry which was successfully used first by Hofker et al. (2) in a 200 micron thick undrifted diode employed as a de/dx charge particle counter. The orthogonal strip geometry is shown in Figure VI-1. It use makes tractable the problems of diode mounting and electrical contacts. It is also extendable to large arrays. In the course of work at SKI toward a gamma camera it was shown that this geometry could be effectively incorporated into a lithium drifted diode whose thick depletion layer provided necessary gamma ray absorption efficiency not obtainable from the conventional p-n junction. Although originally tested in silicon (3), this diode configuration was intended for use with germanium to utilize the greater photoelectric absorption efficiency associated with the higher z of this material. Parker subsequently reported on an orthogonal strip matrix constructed from lithium drifted germanium (4).

B

POSITION INDICATION FOR GAMMA RAY ABSORBED IN ORTHOGONAL STRIP MATRIX

Figure VI-1A Orthogonal-strip geometry. The dark lines represent grooves of high electrical resistance formed on the p and n faces of the diode. The grooves on each face are at right angles to those on the opposite face, thus producing isolated electrode strips on each face of the diode.

1B Acquisition of position information from a signal generated by an absorbed gamma ray. Spatial resolution depends upon the width of the individual electrode strips which make up the matrix.

The availability of ultra pure germanium from General Electric has eliminated the need for the lithium drift process and with it serious instability problems in the diode compensated region which threatened the feasibility of ever producing a germanium gamma camera of clinically significant area. Upon the availability of the ultra pure material it was decided to suspend work with lithium drifted germanium and to produce an orthogonal strip matrix with ultra pure germanium. The successful operation of the initial orthogonal strip matrix from ultra pure germanium was reported previously (5). Detailed data on the electrical characteristics of the (2x2x.5) cm matrix were obtained along with preliminary image studies using isotope phantoms and in-vivo small mammal organs. Illustrative images obtained from the original orthogonal strip camera are shown in Figures VI-2 and VI-3.

Initially it appeared that the elimination of the lithium drift process in the fabrication of the ultra pure germanium matrix would result in an inherently stable device. Repeated measurements of the energy resolution and electrical characteristics of the first matrix were made throughout the initial test program. Not until the matrix was deliberately exposed to room ambient did a clear decline in operational characteristics finally take place, including a loss in energy resolution. In particular, skewing or lobing on the low energy side of the photopeaks was observed on a multi-channel analyzer display. This behavior in effect reduced the operating quantum efficiency of the device. This occurs as follows. The matrix output signal, before it is accepted, must pass through an energy window to eliminate signals corresponding to scattered radiation. Therefore, depending on the magnitude of the charge deficit some partially collected charge pulses from photoelectric absorptions will be rejected. Thus, fewer photoelectric absorption events will be counted than actually took place in the detector's depletion region. The origin of the charge loss appears to be related to gradual changes taking place on the interstrip surface associated with the matrix structure. This effect probably corresponds to the progressive loss of quantum efficiency reported by Parker who described an orthogonal strip matrix from lithium drifted germanium (6). Due note has also been taken of the work by Baertch (7) who used narrow beam radioisotope sources to examine the uniformity of depletion in planar diodes produced from ultra pure germanium. A consistent finding was that the depletion of the diode near the edges occurs at higher bias voltages than for more central regions even for diodes known to have a high degree of uniformity in impurity concentration. The existence of similar conditions at the edges of the strips of the

Figure VI-2A Gamma camera image of ^{99m}Tc (140 Kev) phantom containing 3 and 4 m.m. "hot" spots separated by 9 m.m.

2B 5 m.m. hot spot introduced between other two. Although the edge separation between the center spot and other is less than 4 m.m., the three are resolved.

Dot image of thyroid of rat injected with ^{123}I

Figure VI-3 Lower part is a gamma camera storage scope image of a rat thyroid following injection of the animal with ^{123}I (159 Kev). Upper section is a photograph of the thyroid taken after its excision. The two thyroid lobes are resolved although the maximum dimension of the organ is less than 1 cm.

orthogonal strip matrix was considered a potentially serious problem because of the possibility of a progressive worsening with time. This could eventually lead to high current breakdown at bias voltages too low to assure full charge collection over the entire region of each electrode strip. The occurrence of such conditions in a germanium camera would result in partial or total malfunction of the device.

3. Tests on a Enlarged Camera

As mentioned above the initial ultra pure germanium matrix produced at Sloan-Kettering was a .36 cell (2x2x5) cm device from which spatial resolution of less than 4 mm was obtained. The area of the camera's current detector is nearly 11 cm² with dimensions of (3.25x3.25x.9) cm. An assembly of matrix units of this size, or slightly larger, operated in parallel, will comprise the full scale detector array required in a gamma camera of clinical size.

The enlarged detector consists of (10x10) orthogonal strips with center to center separation of 3.3 mm. The interelectrode regions, currently linear grooves, are .5 mm wide. The matrix p⁺ contact was constructed from ion implanted boron using a gaseous discharge technique developed at General Electric. The n⁺ contact was formed using the conventional lithium diffusion process. The spatial resolution of the 100 cell device near the face of a collimator has been measured to be coincident with the geometric resolution of 3.3 mm. Energy resolution of (3-4.5) KeV (FWHM) was obtained for ⁵⁷Co (122 KeV).

4. Stability and Quantum Efficiency of the Enlarged Camera

Following the preliminary operational tests of the larger matrix array, short term quantum efficiency stability was examined using a fixed ⁵⁷Co source which was placed above the matrix outside of the cryostat. The response of a single representative electrode strip was monitored. Each measurement consisted of an integration of the number of counts under both the 122 KeV and 136 KeV photopeak of ⁵⁷Co. The results are shown in Figure VI-4. They extend over a period of 77 days. The values are normalized to that measured on day zero. The error bars represent the fractional standard deviation of each measurement. The extreme variation in response is within one percent and much of this is probably due to thermal effects. A figure of 58% was obtained for the photoelectric absorption efficiency of the matrix for ⁵⁷Co relative to a totally absorptive sodium iodide crystal. The measured efficiency for the diode prior to formation of the matrix was 63%. The difference is probably the result of charge loss near the edges of the electrode strips.

Figure VI-4

The sensitivity of this device to surface contamination proved no less than the initial smaller matrix. It did not tolerate exposure to room ambient, an occurrence which is almost inevitable in the construction or maintenance of a full scale device. A steady decline in energy resolution of the entire matrix occurred following an exposure of under one half hour to room conditions. In the pulse height spectra, a marked lobing developed at the base of the photopeaks on the low energy side for the p-side electrode strips. This resembled the data obtained from the first device in its decline. Concurrent measurements for the n-side showed high count plateaus to the left of the photopeaks. Despite the deterioration, the detector was still operable as a position sensitive device capable of providing images. However, the response uniformity was grossly degraded. This can be seen in Fig. VI-5 which displays the normalized digitized output of the camera under exposure to a ^{57}Co flood source having a uniformity of $\pm 5\%$. The source was viewed by the matrix through a 3.5 cm thick straight bore collimator with .101 inch circular holes. Although the response was expected to include some non-uniformity due to imperfect alignment between the collimator and matrix, the gross response variation across the matrix is too large to be caused by misalignment, reaching more than a factor of three in the extreme even after discounting the response of narrowed edge strips. In contrast, the variation of response in the camera detector following restorative surface treatments was reduced to around 12%, excluding the response of the narrowed p-side strips.

The sensitivity of the matrix surface structure to low level contamination underscored the importance of finding some type of surface passivation which would preserve the surface characteristics required for uniform charge collection across the individual electrode strips.

5. Tests on Small Matrix Units

To provide a test group of matrix arrays for evaluation of passivation efforts, nine matrix units with two electrode strips on each side were fabricated from the original 36 cell ultra pure germanium matrix array. Response characteristics of one of these approximately 1/4 inch square arrays were measured in detail with a narrowly collimated beam of ^{57}Co . The data for this unit will be presented first. The data for two passivated members of the group will then be shown for comparison. Figure VI-6 is a photo of a multichannel analyzer display of a ^{57}Co (122 KeV, 136 KeV) spectrum obtained from the chosen matrix unit. An ^{241}Am (59.4 KeV) peak and a pulser peak appear on the left and right of the central pair of ^{57}Co peaks. A resolution of 1.2 KeV (FWHM) was obtained for the

Figure VI-5

Normalized digital output of U P Ge Mat-2
under ^{57}Co flood source

P

31	79	70	68	67	69	72	79	83	31
25	61	60	54	58	55	50	70	85	36
35	80	73	70	72	75	78	82	95	46
49	95	86	83	90	83	86	85	100	50
39	77	67	67	72	71	77	82	82	33
45	100	78	77	88	84	84	90	90	43
39	74	73	71	85	79	76	77	80	43
46	93	87	82	92	93	77	85	86	58
17	36	35	34	40	29	32	33	37	19
35	82	77	77	78	66	71	75	66	12

N→

Figure VI-6

122 KeV peak. The narrow beam isotope source used to examine the response characteristics of the (2x2) matrix unit was micrometer positioned. Although the beam diameter of the source was appreciable (1.25 mm) relative to the width of the individual electrode strips (around 3 mm), the measured data can be usefully compared to the calculated ideal response. Hatch made similar use of this fact in a doctoral thesis. The adequacy of this approach is seen in Figure VI-7, where the calculated and the measured beam intensity profiles are shown to be in good agreement. These curves trace the normalized detector output as the source is moved across the edge and on to a totally absorptive lead sheet placed between the source and a detector. Figure VI-8 indicates schematically the movement of the collimated source across the 2x2 cell matrix unit.

The response of the p-side of the selected matrix to the narrow beam source is shown in Fig. VI-9 for 500 volts bias. The matrix surface is drawn to scale at the bottom of the plot. The points on each curve are normalized and represent the integrated counts in both the 122 KeV and 136 KeV photopeaks for equal run times at each source position measured from the beam axis. The error bars indicate the fractional standard deviation of each measurement. The error bars are only shown in Fig. VI-8 but apply at corresponding points of the succeeding figures within a fraction of a percent. They are not explicitly shown on later figures to reduce the density of detail. In the calculated curves indicated in Fig. VI-9, each point represents the computed fraction of the total source emission which is incident on the surface of the electrode strip. This is proportional to the fraction of the circular beam cross section which intersects the electrode strip surface at each point. As the collimated beam is moved onto the matrix electrode strip, it can be seen in the plot that the measured response falls below the calculated response. Both the calculated and measured values eventually reach the same value as a consequence of the normalization process. The difference between the curves at the same points appears to be due to incomplete depletion at the matrix edge which results in both a thinner depletion region and an increased dead layer. It will be seen in Fig. VI-9 that the calculated expected response becomes less than the measured response to the source as it was moved to points near the edges of the groove. This is a result of charge collection at the electrode strip from gamma ray absorptions which occur within the groove. Ordinary charge diffusion partially accounts for this behavior.

A semi quantitative figure of merit for the matrix electrode strip relative to the ideal response of a uniformly depleted device can be obtained from the following considerations. In Fig. VI-9, the areas marked I and II refer to a part of the area under the two pairs of curves. The vertical dashed line indicates a lateral boundary of each area.

Figure VI-7

Figure VI-8

Schematic representation of movement of narrowly collimated ^{57}Co beam across (2 x 2) ultra pure Ge matrix

Figure VI-9

Each portion of a curve which bounds area I and II corresponds to that segment of the source path extending from the point where the beam edge first strikes the outer edge of the electrode strip (see Fig. VI-8) to the last point at which the entire beam cross section is on the electrode strip, i.e. just before the point at which radiation from the collimated beam "spills over" into the groove. The area under this section of the calculated curve is proportional to the total number of photons detected by the electrode strip during the transit of the source across the electrode strip. Since the curves are normalized, the ratio of the area under the measured curve to the corresponding area under the calculated curve yields an integral value of relative efficiency across the strip. This number provides an index of agreement between the measured and calculated curve. For area (I) this ratio is 92% while for area (II) it is 87%. The points on the measured response curves beyond the segments bounding areas I and II correspond to values which have been elevated by the collection of charge from gamma ray absorptions in the groove due to "spill over" of the beam. Since these points reflect the combined response of the electrode strip and the groove component, their values are quantitatively ambiguous.

In Fig. VI-10, the response of the same (2x2) matrix is presented for 200, 300 and 600 volts bias. The calculated ideal response is plotted with small dashed lines. The relative efficiency figures or index of agreement obtained by the method already described are 65%, 81% and 91% respectively. The depletion voltage for the matrix as obtained from capacitance measurements is around 220 volts. Therefore, the response data indicate that substantial overvoltage is necessary to assure adequate depletion of the matrix near the edges. Even at 300 volts bias a significant decline in relative strip efficiency occurs. Figure VI-11 shows the response curves for the n-side of the same matrix at 500 volts bias. The integral figures are comparable to the values obtained at the same bias voltage for the p-side scan.

The measured photoelectric absorption efficiency for the 5 mm thick germanium matrix was 34% for ^{57}Co .

6. Tests on Passivated (2x2) Matrix Units

A commercially developed diode passivation technique is currently under evaluation for applicability to the orthogonal strip matrix structure. Figure VI-12 presents the response data for the p-side of a passivated member of the test group of small matrix units, for bias voltages of 500, 400, 300 and 200 volts. This matrix was stored for two weeks under room conditions before it was tested. The calculated ideal response to the source at different positions is shown

Figure VI-10

(2x2) U.P. Ge Mat-1-1

Response of p Side Strip to Narrow Beam

^{57}Co Source for Different Bias Voltages

Figure VI-11

Figure VI-12

(2x2) U.P. Ge Mat-1-3-p

Response of p Side of Passivated Matrix to Narrow Beam ^{57}Co Source for Different Bias Voltages

as a dashed line. The integral area figures were obtained as above by comparing the partial areas under the measured and calculated curves. These figures ranged between 95% and 79% for strip #1 and between 92% and 74% for strip #2. As with the unpassivated device, the curves indicate that higher bias voltage than required to deplete the bulk material is necessary to achieve nearly uniform quantum efficiency across the entire matrix strip. The n-side response was found to be comparable to the p-side values. The energy resolution of ^{57}Co was measured to be 1.2 KeV FWHM.

Representative response curves for a second passivated device are shown in Figure VI-13 for the p-side strips at 500 and 400 volts bias. Resolution of ^{57}Co was measured to be 1.4 KeV FWHM. This last device was exposed continuously to room ambient conditions for one month prior to testing. As with the other passivated matrix units it was only protected from free exposure to dust. Two other passivated devices were also tested following one month of exposure to room conditions. One failed to hold bias voltage, probably because of damage in handling. The other was only marginally stable. Although further measurements are required before a definitive conclusion can be drawn regarding the usefulness of the particular passivation process, the successful operation of two orthogonal strip matrix units after two weeks and one month of storage under room conditions respectively is very promising. Similar treatment of unpassivated ultra pure germanium planar diodes would usually result in severe degradation or complete failure. Moreover, the response data obtained from the successfully passivated units is at least as good as was obtained from the unpassivated device which underwent testing shortly after fabrication without prolonged room ambient exposure.

7. Matrix Electronics

The current electronic readout system employed with the (10x10) strip matrix is based on one amplifier for each electrode strip. An alternative mode, which would utilize an external impedance network, was suggested in a previous report (8) and has since been implemented with some success elsewhere. Ultimately it may prove that a resistive divider network or delay line readout offers sufficient equipmental economy over the method of one data channel per electrode strip to compensate for the lower count rate capability of these monitoring methods. However a commitment to a particular electronic readout system for the full scale camera is not necessary at this time. It is desirable to operate the gamma camera with an electronic system which does not complicate the performance evaluation of

Figure VI-13

the detector assembly. Currently the use of one of the alternative electronic systems poses additional problems that can be circumvented by using one amplifier for each electrode strip.

A specially designed prototype electronic system is currently under commercial construction for use with an enlarged orthogonal strip camera. The package will be sufficient to monitor the planned (8x8) cm camera detector and can be expanded to a larger system if this proves advisable.

To sum up, the elimination of the lithium drift process through the use of hyper pure germanium removes a serious bulk instability in the germanium diode and in the germanium orthogonal strip matrix. However, response uniformity and stability of the orthogonal strip matrix have been shown to be seriously degraded by progressive changes that take place on the interelectrode surfaces as a result of exposure to contaminants. Reduction of the size of the inter-strip region will improve but not eliminate the problem. Insufficient data exists at present to predict how long the response characteristics of an orthogonal strip matrix will remain stable in a "clean" vacuum. However it is clear that the conditions necessary to realize long term stability are extremely exacting. These conditions may already be too restrictive for a practical clinical device since they definitely exclude exposure of even a warmed matrix assembly to room ambient and possibly are incompatible with temperature cycling between liquid nitrogen and room temperature. This is a fragility which is undesirable in a costly system intended for practical clinical use. Preliminary results with a particular passivation process are extremely encouraging. If long term stability is demonstrated for passivated matrix units, the eventual construction of a practical full scale camera can be approached with greater confidence.

REFERENCES:

1. Hofker, W.K., Oosthoek, D.P., Hoeberechts, A.M., Van Dantzig, R., Mulder, K., Oberski, J.E., Koerts, L.A., Kieperink, J.H., Kok, E., and Rumphorst, R.E. The checker board counter. A semiconductor de/dx detector with position indication. I.E.E.E. Trans. 13:208, 1966.
2. Detko, J.F. Semiconductor diode matrix for isotope localization. Phys. Med. Biol. 14:245-253, 1969.
3. Annual Progress Report to the United States Atomic Energy Commission. Contract No. AT(30-1)-3510. Biological and Clinical Dosimetry, 1970.

4. Detko, J.F. A prototype ultra pure germanium orthogonal strip gamma camera. Proc. Med. Radioisotope Scint. IAEA Vol. I 241-254, Monte Carlo, 1972.
5. Parker, R.P. Gunnensen, E.M., Gunnensen, Ellis R. and Bell J. A semiconductor gamma-camera: assessment of results. Proc. Med. Radioisotope Scint. IAEA Vol. I 193-216, Monte Carlo, 1972.
6. Baertsch, R.D. Collimated beam scanning of pure germanium detectors. I.E.E.E. Trans. Nuc. Sci. NS 20 No. 1, pp. 488-493, February, 1973.
7. Baertsch, R.D. Surface effects on p-type high purity germanium detectors at 77°K. I.E.E.E. Trans. Nuc. Sci. NS 21 No. 1, February, 1974.
8. Annual Progress Report to the United States Atomic Energy Commission, Contract No. AT(11-1)-3522, Biological and Clinical Dosimetry, 1971.

VII. PUBLICATIONS AND PRESENTATIONS

Publications

1. Detko, J.F. A prototype ultra-pure germanium orthogonal strip gamma camera. Medical Radioisotope Scintigraphy, IAEA, Vol. I, 241-255, 1973.
2. Detko, J.F. Operational characteristics of a small ultra-pure germanium gamma camera. Semiconductor Detectors in Medicine. USAEC Conf. 730321, 1973.
3. Hahn, E.W., Canada, T.R., Alfieri, A.A. and McDonald, J.C. The radiation enhancement factor (REF) of hyperthermia in combination with fast neutrons on local tumor response. Rad. Res. 59:141, 1974. (Abstract).
4. McDonald, J.C., Canada, T.R., Freeman, R. and Laughlin, J.S. Physical properties of Shonka A-150 tissue equivalent plastic for an absorbed dose neutron calorimeter. Rad. Res. 59:228, 1974. (Abstract).

Presentations

1. Laughlin, J.S., Canada, T.R., Simpson, L., McDonald, J.C., Kuo, T.Y.T. and Mittleman, A. Compact cyclotron neutron dosimetry. Pres. at the German-Austrian Roentgen Conf., Vienna, Austria, April, 1973.
2. Hahn, E.W. and McDonald, J.C. Embryogenesis in the rat following fast neutron irradiation. Pres. at the Radiation Research Society Mtg., St. Louis, Mo., May, 1973.
3. Canada, T.R., McDonald, J.C., Freed, B.R., Kuo, T.Y.T. and Laughlin, J.S. Physical properties of compact cyclotron produced fast neutron fields. Pres. at Applications of Neutrons in Radiotherapy, Rijswijk, Netherlands, October, 1973.
4. McDonald, J.C., Canada, T.R., Freeman, R. and Laughlin, J.S. Physical properties of Shonka A-150 tissue equivalent plastic for an absorbed dose neutron calorimeter. Pres. at the Fifth International Congress of Radiation Research, July 14-20, 1973, Seattle, Washington.
5. Hahn, E.W., Canada, T.R., Alfieri, A.A. and McDonald, J.C. The radiation enhancement factor (REF) of hyperthermia in combination with fast neutrons on local tumor response. Pres. at the Fifth International Congress of Radiation Research, July 14-20, 1974, Seattle, Washington.
6. Detko, J.F. Current status of the orthogonal strip gamma camera. Invited speaker - Symposium for American Assoc. of Physicists in Medicine, July 28-31, 1974, Kansas City, Missouri.

Submitted for Publication

1. Laughlin, J.S., Canada, T.R., Simpson, L.D., McDonald, J.C., Kuo, T.Y.T. and Mittleman, A. Compact cyclotron neutron dosimetry. Invited presentation at the First German-Austrian Roentgen Congress, Vienna, 1973. To be published in Strahlentherapie.
2. Canada, T.R., McDonald, J.C., Freed, B.R., Kuo, T.Y.T. and Laughlin, J.S. Physical properties of compact cyclotron produced fast neutron fields. Accepted for publication in Radiation Research, November, 1974.
3. Canada, T.R., Djordjevic, B., McDonald, J.C., O'Neil, J., Zeitz, L. and Laughlin, J.S. A biological determination of the variation of fast neutron field quality with depth. I. RBE. Submitted to Radiation Research, 1974.
4. Zeitz, L., Canada, T.R., Djordjevic, B., Dymbort, G., Freeman, R., McDonald, J.C. and Laughlin, J.S. A biological determination of fast neutron field quality with depth. II. OER. Submitted to Radiation Research, 1974.

PUBLICATION
BY AEC
AUTHORIZED ☐NOTICE OF RESEARCH PROJECT
SCIENCE INFORMATION EXCHANGE

SMITHSONIAN INSTITUTION

U.S. ATOMIC ENERGY COMMISSION

SIE NO.

AEC CONTRACT NO.

SUPPORTING DIV. OR OFFICE: _____

NAME & ADDRESS OF CONTRACTOR OR INSTITUTION: (State the division, department, or professional school, medical, graduate or other, with which this project should be identified.)

The Sloan-Kettering Institute for Cancer Research
410 East 68th Street
New York, New York 10021

TITLE OF PROJECT:

Biological and Clinical Dosimetry

NAMES, DEPARTMENT, AND OFFICIAL TITLES OF PRINCIPAL INVESTIGATORS AND OTHER PROFESSIONAL SCIENTIFIC PERSONNEL: (not including graduate students) engaged on the project, and fraction of man-year devoted to the project by each person.

John S. Laughlin, Ph.D., Member
Lowell L. Anderson, Ph.D., Associate
Thomas R. Canada, Ph.D., Associate
John F. Detko, M.A., Research Associate
Joseph C. McDonald, M.S., Research Associate

NO. OF GRADUATE STUDENTS ON PROJECT: 0 NO. OF GRADUATE STUDENT MAN-YEARS: _____

SUMMARY OF PROPOSED WORK: (200-300 words, omit Confidential Data). Summaries are exchanged with government and private agencies supporting research, are supplied to investigators upon request, and may be published in AEC documents. Make summaries substantive, giving initially and for each annual revision the following: OBJECTIVE; SCIENTIFIC BACKGROUND FOR STUDY; PROPOSED PROCEDURE; TEST OBJECTS AND AGENTS.

Research will be directed toward the development and application of systems such as: a tissue equivalent calorimeter, neutron ionization chambers and tissue equivalent spherical proportional counters for the determination of absorbed dose in radiobiological research and clinical applications. Both primary and secondary dosimetric techniques are being developed and employed. Further investigations are also planned for the special problems associated with the dosimetry of ultra-high intensity electron beams and related studies of oxygen diffusion and depletion.

In addition, research will continue with the recently operational germanium semiconductor gamma camera which utilizes orthogonal strip geometry. In conjunction with the Metabolic Isotope Group kinetic studies of the in-vivo uptake of isotopes and labeled compounds are planned.

RESULTS TO DATE:

The thermal defect which occurs in plastics used for absorbed dose calorimeters, has been studied in detail for Shonka A-150 tissue equivalent (T.E.) plastic. It has been found that thermal annealing results in an absorber whose output stabilizes quickly and remains constant with accumulated dose.

A number of physical properties of the fast neutron fields produced

	PROGRAM CATEGORY NO. (contd)
BUDGET	
PRIMARY	
SECONDARY	

Signature of Principal Investigator

DATE: 11/8/74

INVESTIGATOR

DO NOT USE THIS SPACE

NOTICE OF RESEARCH PROJECT

TITLE OF PROJECT: Biological and Clinical Dosimetry

Results to Date: (Cont'd.)

when 8 MeV deuterons or 23 MeV ^3He ions are incident upon a thick beryllium target have been measured. These include: the angular distributions of total dose, the linear dose attenuation coefficients of steel, lead, and a polyethylene mixture, the dependence on depth of the neutron dose and the gamma-ray dose.

The current area of the germanium gamma camera detector is just under 11 cm² and evaluation tests during the current year revealed that surface contamination, particularly at the inter-electrode regions may result in loss of quantum efficiency. A group of test matrix units was constructed from the original (2x2) cm prototype detector and used to evaluate a commercially available passivation technique. Initial results are encouraging with two of four passivated units undeteriorated after two and four weeks respectively of room ambient storage. Preparations for passivation and testing the current camera detector are in progress.

A specially designed electronic monitor system is under commercial construction for use with the germanium camera after further enlargement of the detector.