THE UNIVERSITY OF ASTON IN BIRMINGHAM

"NEUTRON DOSIMETRY"

A THESIS SUBMITTED FOR THE DEGREE

DOCTOR OF PHILOSOPHY

by

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SUMMARY

Neutron Dosimeter Project 1967/70

The basic problem was to investigate the feasibility of designing a personal neutron dosimeter with neutron and gamma ray sensitivities proportional to the dose equivalent rather than the absorbed dose. The condenser ionisation chamber was made the basis of the investigation.

Various materials for lining the ionisation chamber were tested and their response to neutrons of different energies measured. In order to overcome the high sensitivity of the chamber to gamma rays, two chambers with different linings were combined in one instrument such that there was automatic gamma ray compensation.

It was shown that by a suitable choice of dimensions and chamber capacitances a suitable gamma compensated instrument could be produced and the sensitivities of prototype instruments to neutrons and gamma rays was measured.

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I - INTRODUCTION

I.I General Note on types of Radiation

The neutron causes a higher degree of damage to tissue than any other identified radiation. Therefore when attempts are made to measure neutron flux it is first necessary to examine the different energies at which neutrons are commonly produced and the safety levels which are commonly accepted for these different energies.

As neutron irradiation will always produce subsequent gamma rays it is also necessary to examine gamma radiation in a similar fashion.

I.I.I Interactions of Neutrons with Matter

As the neutron is an uncharged particle it may be thought of as interacting with the nuclei only. There are interactions between neutrons and electrons but of negligible cross section and these can be neglected for practical purposes. The main processes of neutron interaction are:-

(I) Elastic Scattering

Kinetic Energy is transferred from the scattered neutron to the recoil nucleus. The sum of all the kinetic energies of all the particles within the system remains constant.

(2) Capture.

The neutron is captured by the target nucleus, forming a compound nucleus which is usually excited and undergoes a (n,α), (n,p) or (n,γ) reaction. At high enough energies the $(n,2n)$ reaction may occur.

(3) Inelastic Scattering

Can be taken as a special case of capture. The neutron is absorbed and a neutron re-emitted with a lower energy leaving the nucleus in an excited state, from which it decays to the ground state by the emission of one or more gamma rays.

I.I.2 Neutron Energy Groups

In considering neutron energies it is convenient to define four groups:-

(I) Thermal Neutrons

Have a Maxwellian distribution of velocities and are in thermal equilibrium with matter. In this distribution the most probable velocity per unit velocity at $295^{o}K$ is 2,200 m / sec corresponding to an energy of 0.025 ev. The predominant interaction is capture, usually the (n,γ) being produced. Occasionally (n,p), (n,α) or fission may occur. In many nuclides the neutron cross section is inversely proportional to the velocity of the neutron, thus measurements of neutron density can be achieved by the activation of $^{I}/v$ foils. (ie Activation $n v\sigma = n v (^{I}/v) = n$).

(2) Intermediate Neutrons (0.5 ev to I0 Kev)

These neutrons are contained in a region where there are many large resonant peaks in the neutron cross sections and hence are often called resonance neutrons. Neutron slowing down is the predominant interaction and leads to a neutron flux which is inversely proportional to energy giving the $^{I}/E$ spectrum. (When scattering cross sections are constant.)

(3) Fast Neutrons (10 Kev to 10 Mev)

Predominant part of the cross section here is elastic scattering, but with increase in energy inelastic scattering and reactions producing other particles become increasingly important.
For example in tissue, fast neutrons are mainly elastically scattered off the hydrogen nucleus. Inelastic scattering (occuring above I Mev for most nuclei) becomes more prevalent with increase in neutron energy. Above 10 Mev it may be as probable as elastic scattering. It is important as a source of γ rays in the moderation process, and causes large neutron energy losses in high Z materials where energy losses by elastic scattering can only be small.

(4) Relativistic Neutrons.

Involves energies mainly above 20 Mev and therefore as neutron sources of this energy range were not readily available this energy was not investigated. But briefly inelastic scattering takes over from elastic scattering at this energy, with many protons and or neutrons emitted from the target nucleus.

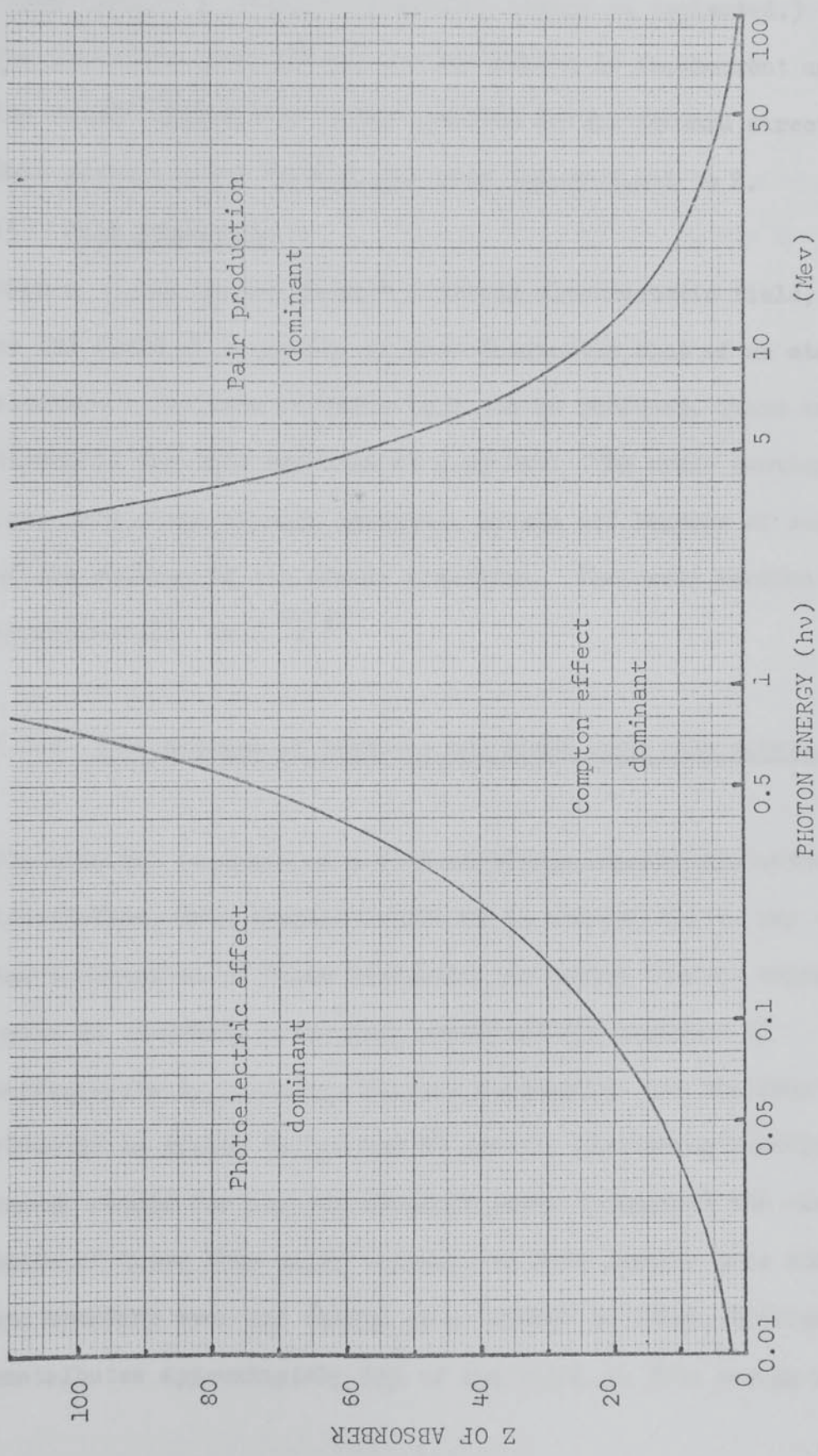
I.I.3 Interactions of Gamma Rays with Matter

In γ ray interactions, for dosimetry considerations, three main events occur. These are summarised in the graph [I] given in fig (I.I).

(I) Photoelectric effect

Here a γ ray photon ejects an atomic electron from an electron shell. At low energies the cross section decreases very rapidly as the energy of the γ ray increases rapidly with increasing Z (about as Z^{4-5}). [2]

Fig 1.1



Relative importance of the three major types of gamma ray interaction. The lines showing the values of Z and hν for which the two neighbouring effects are just equal.

(2) Compton scattering

Compton scattering predominates over the range I to 5 Mev. Here the photon may be thought of as colliding directly with a "free" atomic electron (the binding energy is neglected.) The photon is degraded and the scattering is incoherrent and the recoil electron is always emitted in the forward direction. This gives a cross section per atom proportional to Z.

(3) Pair production

When a γ ray passes through a strong electrostatic field, such as the field of a nucleus or less frequently that of an atomic electron a positron electron pair can be produced. There is a threshold for this reaction at I.02 Mev. The cross section for pair production at high energies, levels off because of screening of the nucleus by the atomic electrons. The cross section varies approximately as Z^2.(2)

I.I.4 Interactions of neutrons and Gamma rays with tissue.

For neutrons incident on a slab of tissue penetration without interaction, or a reaction outlined in section I.I.I, may occur.. Any combination of these reactions can happen (except capture which is obviously a terminal reaction) for neutrons of sufficiently high energy. Elastic scattering with the four main elements in tissue (H,C,O and N) are all decreasing functions of energy except for the odd resonant peak. In general the contribution of these four major elements to fast neutron dose and kerma are constant over the energy range 250Kev to I4Mev. Hydrogen contributes approximately 85% of the total in this energy range

while the hydrogen recoils at about 10 Kev contribute 97%.(See fig I.3).

Capture of low energy neutrons in tissue is a significant factor in dose contribution for most neutron spectra incident on anthropomorphic phantoms. The two main reactions are the $^{14}_{N}(n,p)^{14}_{C}$ and $^{1}_{H}(n,\gamma)^{2}_{H}$. (59) In the former reaction the proton and recoil nucleus contribute 0.62 Mev of energy at the capture site and the proton has a range of roughly 10 microns in tissue. However the gamma ray produced in the second reaction with 2.2 Mev would, in certain cases, escape from a mass of tissue even as large as a human phantom. So for small amounts of tissue the proton reaction will contribute the significant dose while larger masses the gamma producing reaction will predominate.

With inelastic scattering the main reactions of neutrons in soft tissue is with Oxygen, Carbon and Nitrogen. In this process the neutron may lose energy and leave the scattering nucleus above the ground energy state. The excited nucleus usually decays with the emission of a gamma ray.

With nonelastic scattering contribution becomes significant above 5 Mev and increases with neutron energy to about 15 Mev.(59) Most of these reactions are accompanied by gamma rays, but the alpha particle and proton producing reactions are of importance because of the high L.E.T. of these particles and their low range in tissue. The main capture and non elastic reactions which produce charged particles are listed in fig I.4 together with

Fig 1.2

Reaction	Photon energy (MeV)	HPRR[a]	5 MeV[a]	7 MeV[a]	10 MeV[a]	14 MeV[a]
1. H(n, n)H	—	6.9232	22.44	28.26	33.58	34.10
2. C(n, n)C	—	0.1780	0.57	0.59	0.63	0.60
3. N(n, n)N	—	0.0146	0.06	0.06	0.06	0.06
4. O(n, n)O	—	0.4228	1.36	1.97	1.50	1.44
5. C(n, n')*C; *C → C + γ_1	1.75	—	—	—	0.05	0.03
6. C(n, n')*C; *C → C + γ_2	4.43	0.0099	0.03	0.26	0.59	0.69
7. C(n, n')*C; *C → C + γ_3	6.6	0.0001	—	—	—	0.14
8. N(n, n')*N; *N → N + γ_1	1.63	0.0004	—	—	—	0.01
9. N(n, n')*N; *N → N + γ_2	2.31	0.0002	—	—	0.02	0.03
10. N(n, n')*N; *N → N + γ_3	5.1	0.0001	—	—	0.03	0.02
11. N(n, n')*N; *N → N + γ_4	10.0	—	—	—	—	0.03
12. N(n, n')*N; *N → N + γ_5	11.0	—	—	—	—	0.03
13. O(n, n')*O; *O → O + γ_1	6.1	0.0083	—	0.14	1.36	1.54
14. O(n, n')*O; *O → O + γ_2	7.12	0.0036	—	—	0.30	0.38
15. O(n, n')*O; *O → O + γ_3	3.8	—	—	—	—	1.58
16. O(n, n')*O; *O → O + γ_4	4.8	—	—	—	—	0.77
17. C(n, α_2)*Be; *Be → Be + n; Be → 2α	—	—	—	—	—	0.58
18. C(n, n')*C; *C → Be + α	—	—	—	—	—	0.51
19. N(n, 2n)N	—	—	—	—	—	0.01
20. H^1(n, γ)H^2	2.2	1.7250	1.64	1.41	1.21	0.89
21. C(n, α_0)Be	—	0.0006	—	—	0.11	0.36
22. N(n, α_0)B	—	0.0127	0.10	0.06	0.10	0.11
23. O(n, α_0)C	—	0.0284	0.55	0.69	0.73	1.11
24. C(n, α_1)*B; *Be → Be + γ	1.75	0.0001	—	—	—	0.05
25. N(n, α_1)*B; *B → B + γ_1	2.1	0.0006	0.01	0.01	0.02	0.04
26. N(n, α_2)*B; *B → B + γ_2	4.5	0.0001	—	0.01	0.02	0.05
27. N(n, α_3)*B; *B → B + γ_3	5.0	0.0004	—	—	0.04	0.06
28. O(n, α_1)*C; *C → C + γ_1	3.1	0.0014	—	—	0.11	1.03
29. O(n, α_2)*C; *C → C + γ_2	3.8	0.0009	—	—	0.22	2.46
30. O(n, α_3)*C; *C → C + γ_3	7.0	—	—	—	—	1.00
31. N(n, p)*C	—	0.1495	0.19	0.22	0.22	0.18
32. O(n, p)*N; *N → N + γ	6.1	—	—	—	—	0.51
33. N(n, t)C						
Total doses		9.4809	26.95	33.69	40.91	50.40

[a] In units of 10^{-10} rad neutron^{-1} cm^2.

Contributions from individual reactions to average absorbed dose in a whole anthropomorphic phantom.

HPRR = Health Physics Research Reactor at Oak Ridge

Fig 1.3

Reaction	Photon energy (MeV)	HPRR	5 MeV	7 MeV	10 MeV	14 MeV
1. H(n, n)H	-	73.12	83.11	83.88	82.08	67.66
2. C(n, n)C	-	1.89	2.11	1.75	1.54	1.19
3. N(n, n)N	-	0.15	0.22	0.17	0.15	0.12
4. O(n, n)O	-	4.46	5.05	5.85	3.67	2.86
5. C(n, n')*C; *C → C + γ_1	1.75	-	-	-	0.12	0.06
6. C(n, n')*C; *C → C + γ_2	4.43	0.10	0.11	0.77	1.44	1.37
7. C(n, n')*C; *C → C + γ_3	6.80	-	-	-	-	0 28
8. N(n, n')*N; *N → N + γ_1	1.63	-	-	-	-	0.02
9. N(n, n')*N; *N → N + γ_2	2.31	-	-	-	0.04	0.06
10. N(n, n')*N; *N → N + γ_3	5.10	-	-	-	0.07	0.04
11. N(n, n')*N; *N → N + γ_4	10.0	-	-	-	-	0.06
12. N(n, n')*N; *N → N + γ_5	11.0	-	-	-	-	0.06
13. O(n, n')*O; *O → O + γ_1	6.10	0.09	-	0.42	3.32	3.06
14. O(n, n')*O; *O → O + γ_2	7.12	0.04	-	-	0.73	0.75
15. O(n, n')*O; *O → O + γ_3	3.80	-	-	-	-	3.13
16. O(n, n')*O; *O → O + γ_4	4.80	-	-	-	-	1.53
17. C(n, α_2)*Be; *Be → Be + n Be → 2α	-	-	-	-	-	1.15
18. C(n, n')*C; *C → Be + α	-	-	-	-	-	1.01
19. N(n, 2n)N	-	-	-	-	-	0.02
20. H^1(n, γ)H^2	2.20	18.19	6.09	4.18	2.96	1.76
21. C(n, α_0)Be	-	-	-	-	0.27	0.73
22. N(n, α_0)B	-	0.13	0.37	0.18	0.24	0.27
23. O(n, α_0)C	-	0.30	2.04	2.05	1.78	2.20
24. C(n, α_1)*B; *Be → Be + γ	1.75	-	-	-	-	0.10
25. N(n, α_1)*B; *B → B + γ_1	2.10	-	0.04	0.03	0.04	0.08
26. N(n, α_2)*B; *B → B + γ_2	4.50	-	-	0.03	0.04	0.10
27. N(n, α_3)*B; *B → B + γ_3	5.00	-	-	-	0.10	0.12
28. O(n, α_1)*C; *C → C + γ_1	3.10	0.01	-	-	0.27	2.04
29. O(n, α_2)*C; *C → C + γ_2	3.80	-	-	-	0.54	4.88
30. O(n, α_3)*C; *C → C + γ_3	7.00	-	-	-	-	1.98
31. N(n, p)*C	-	1.58	0.70	0.65	0.54	0.36
32. O(n, p)*N; *N → N + γ	6.10	-	-	-	-	1.01
33. N(n, t)C						

HPRR = Health Physics Research Reactor at Oak Ridge

Contributions from individual reactions expressed as percentages of the total average absorbed dose in a whole anthropomorphic phantom

Reaction	Q Value (Mev)	E_{max} (Mev)	R_{max} (cm)	E_{max} (Mev)	R_{max} (cm)
C^{12} (n,α) Be^{9}	-5.70	7.89	6.6×10^{-3}	1.28	3.8×10^{-4}
N^{14} (n,p) C^{14}	0.63	14.60	2.2×10^{-1}	7.62	6.9×10^{-2}
N^{14} (n,t) C^{12}	-4.01	9.67	4.5×10^{-2}	2.98	6.3×10^{-3}
N^{14} (n,α) B^{11}	-0.16	12.80	1.5×10^{-2}	6.30	4.6×10^{-3}
O^{16} (n,p) N^{16}	-9.63	4.19	2.4×10^{-2}	-	-
O^{16} (n,d) N^{15}	-9.90	4.03	1.3×10^{-2}	-	-
O^{16} (n,α) C^{13}	-2.21	11.04	1.2×10^{-2}	4.57	2.6×10^{-3}

Capture and nonelastic reactions which produce charged particles in tissue, with maximum ranges and energies for secondaries produced by 14 and 7 Mev neutrons

Fig 1.4

their Q values.[75] The total list is in fig 1.2.

Gamma ray dose contributions generally obey energy variations as laid down in fig 1.1. This is a generalisation because, for example, in the irradiation of bone, soft tissue cells lie adjacent to, or are embedded within a mineralised matrix having different properties of gamma ray absorption. Consequently a relatively new concept of transition zone dosimetry has arisen [76,77](a transition zone concerns the distribution of absorbed dose near an interface between two dissimilar materials.) However, it is certain that the absorbed dose per exposure does vary with energy. An example is given in fig. 1.5 which shows how soft tissue in bone behaves with different photon energies. This demonstrates the extra photoelectron dose to tissue is most marked around the 50 Kev. region falling off for higher energies as the Compton effect becomes more dominant. This is further evidence to show that the assumption of unity for the Quality Factor of gamma rays, regardless of energy, is rather approximate.

1.2 Permissive Levels of Radiation.

From cosmic radiation there exists a constant neutron flux of roughly 50n cm^{-2} $hr.^{-1}$ at sea level. This increases with altitude, reaching a value approximately at 10,000 ft. of 500n cm^{-2} $hr.^{-1}$. This corresponds to a dose in the order of 10^{-1} mrem/week at sea level and 1 mrem/week at 10,000 ft. Thus individuals are continually exposed to neutrons but this background level is very low compared with the generally accepted permissible dose for critical organs of 100 mrem/week. The effects of ionising radiation can

Photon energy (Kev)	Osteocyte (5 μ)		"Average soft tissue" in cortical bone	10 μ lining of 50 μ Haversian canal	Trabecular marrow (skeletal average)
	Plane slab model	Cylindrical model			
25	2.80	3.49	1.94	1.50	0.96
35	3.12	3.75	2.42	1.76	0.99
50	3.25	3.65	2.71	1.89	1.03
75	2.40	2.55	2.10	1.60	1.05
100	1.52	1.57	1.47	1.26	1.04
200	1.05	1.06	1.05	1.02	0.99

Ratios of Absorbed Dose to Exposure (Rad / R) for the soft tissues contained in bone.

Fig 1.5

be classified into:-

(a) Somatic.

Affecting the individual, whether these effects develops early or be delayed.

(b) Genetic.

Affecting the offspring and ultimately the population as a whole.

The knowledge of early somatic effects are well established. The delayed somatic effects cause much concern as these include leukaemia and shortening of the life-span. Studies of the survivors of the atomic bombs at Hiroshima and Nagasaki have yielded much useful information.

The genetic effect of exposure to ionising radiations gives gene mutations of the germ cells, or cells, ancestral to them, in the reproductive organs and damage to the chromosomes. The genetic dose to a population is the dose which, if it is received by each person from conception to the mean age of childbearing (30 years), would result in the same genetic burden to the whole population, as do the actual doses received by the individuals. The I.C.R.P. recommends that the genetic dose to the population should not exceed 5 rems from all sources additional to the dose from natural background radiation and medical exposure.(87)

For workers in radiation areas (designated personnel) the permissible exposure to radiation is that dose which may be received without undue risk to the health of the individual. At the present time 5(N-18) rem is the maximum permissible cumulative dose up to the end of a calendar year, where N is the age of the individual.(3,81) Generally speaking all persons working in a radiation area (where

the radiation dose rate averaged over any one minute exceeds 0.75 mrem/hr.) are classified as "designated".[4] A designated person is required to wear film badges and have an annual examination. This basically means that a higher level of radiation, up to 3 rems a quarter year may be received. A permanent record of the film badge results is also kept. However, the limitations of the Commercial Film Badge are numerous.[5,6,7] For example fading and latent image fading cannot be avoided and it has been estimated that the film dosimeter is now fully developed, so no significient improvements can be expected.[8] This gives much encouragement for a neutron dosimeter of worthwhile performance.

Fig. 1.6 gives the maximum permissible organ and tissue dose equivalent. As can be seen there are special clauses for young workers. The dose equivalents are to be evaluated from quality factors given in Fig. 1.7.

Fig. 1.8 demonstrates the variation of quality factor with neutron energy. This means that any accurate dosimeter will have to be more sensitive to higher than lower energy neutrons. At the present time a radiation level of rate 2.5 mrem/hour is taken as equivalent to one maximum permissible level (M.P.L.) [4] For clarity in understanding it would be far better to calibrate all dosimeters in rem instead of rad. This would bring home to the operator the significance of the level of radiation being observed. However, if changes in quality factor are made the calibration of the instrument is then useless. With this point in mind a prospective specification for a military instrument requires calibration in rads.[9] This has the dubious advantage of concealing from the wearer the significance of the reading. Interpretation being left to the Military scientist from a knowledge

Fig 1.6

Exposed part of the body	Designated Persons		Non designated Persons		
	Over 18 years	16 - 18 years	Over 18 years	16 - 18 years	Under 16 years
Whole body, blood forming organs, gonads	5(N - 18)	1.5	1.5	1.5	0.5
Skin and bone (except the skin and bone of the hands, forearms, feet and ankles), thyroid	30.0	3.0	3.0	3.0	3.0
Other single organs	15.0	1.5	1.5	1.5	1.5
Hands, fore-arms, feet and ankles	75.0	7.5	7.5	7.5	7.5

Units are in rem / year.

Maximum Permissible Organ and Tissue Doses for Dose Equivalent.

Fig 1.7

Radiation	Quality Factor
X rays, gamma rays electrons and beta rays of maximum energy greater than 30 Kev	1.0
Beta rays, maximum energy not greater than 30 Kev	1.7
Fast neutrons and protons up to 10 Mev	10 (30 for eyes)
Naturally occuring alpha particles	10
Heavy recoil nuclei	20

General values of Quality Factor used in calculating Dose Equivalent.

Fig 1.8

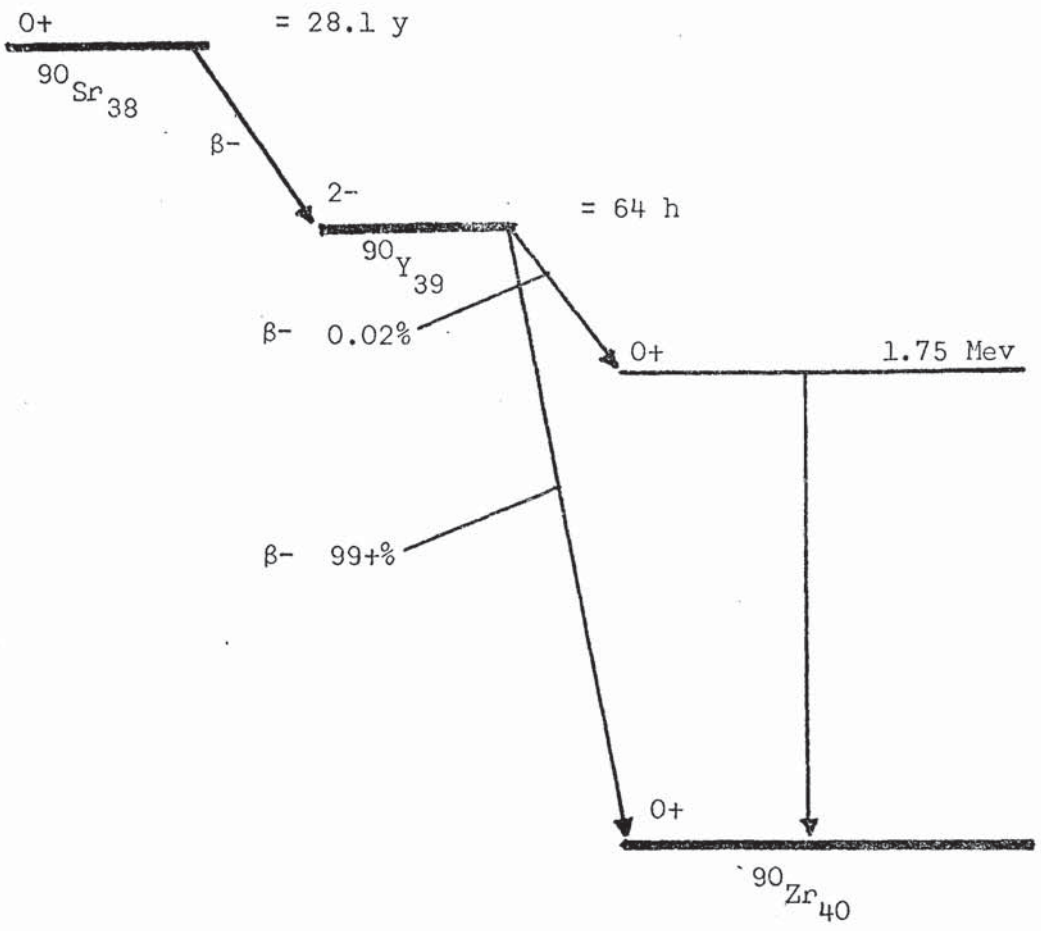
Neutron Energy	Quality Factor	Flux equivalent to 2.5 mrem / hour (n / cm^2 sec)	Integrated flux equivalent to 1 rem (n / cm^2 x 10^6)
Thermal	3.0	670	960.0
5 Kev	2.5	570	820.0
20 Kev	5.0	280	400.0
100 Kev	8.0	80	120.0
500 Kev	10.0	30	43.0
1 Mev	10.5	18	26.0
5 Mev	7.0	18	26.0
10 Mev	6.5	17	24.0
14 Mev	6.2	10	14.0
20 Mev	5.9	10	14.0
40 Mev	5.6	7.0	10.0
50 Mev	5.2	7.0	10.0
70 Mev	5.0	6.7	9.6
110 Mev	4.6	6.1	8.8
190 Mev	4.1	4.7	6.8
300 Mev	3.8	3.4	4.9
475 Mev	3.5	2.3	3.3
650 Mev	3.4	1.7	2.4
1,000 Mev	3.4	1.1	1.6

Relationship between Quality Factor, Dose equivalent for 1 M.P.L., and neutron flux equivalent to 1 rem.

of the nuclear bomb spectrum. However "fall out" conditions would still be prevailent and the interpretation of a rad dose is prone to many approximations. Obviously the military instrument would operate in very high dose conditions and would need therefore to be a relatively insensitive, general purpose dosimeter. It is envisaged that the military instrument would require 600 rad for F.S.D. It is still very strongly felt that such an instrument would, apart from being used in a disastrous atomic bomb war, give a rather futile reading. There is also an increasing body of opinion that the somatic effects of radiation have been greatly underestimated. (10)

Also in a recent report serious implications have been drawn between the genetic effects of the "fall out" from the Atomic Bomb tests completed in the 1950's and the deaths of very young children. (11) This challenges the assumption about the length of time strontium 90 remains in the bone. Strontium decays into radioactive yttrium which after about four years is redistributed to organs, including the reproductive ones. (See fig 1.9). So in scientific circles it is now gradually becoming apparent that the less radiation the body receives as a whole, the better it is for the general health in the long run.

Fig 1.9



Decay scheme of $^{90}Sr_{38}$ (Strontium 90)

SECTION 2. UNITS AND TERMINOLOGY

2.I Historical Background.

The position of radiological units has in past years been very confused. Only recently in I962 the International Commission on Radiation Units (I.C.R.U.) clarified the basic concepts and gave standard terms and definitions which are now becoming generally accepted.

In tracing the development of radiation dosimetry it is easy to see how this confusion arose. Probably the earliest unit used in dosimetry was the skin erythema dose. This was practical (too practical in fact!) to the extent that it indicated the radiation level at which certain undesirable skin changes became apparent. It was unsatisfactory as skin erythema varied according to:-

(a) The part of the body on which the measurement was made.

(b) The energy of the incident radiation.

For exposure to the back of the hand, individual, an erythema dose corresponds approximately to I00 R, 350 R, I,000 R, and I,500 R, for radiations of 0.0I, 0.I, I and 2 Mev respectively. It was not until I928 that the I.C.R.U. adopted the Roentgen as an international unit of X-Ray radiation.

2.2 Roentgen and Exposure Dose.

The Roentgen is a good example of the confusion which existed in radiation units. It was first defined in I928 by the I.C.R.U. as (I2) :- The quantity of X radiation which, when the secondary

electrons are fully utilised and the wall effect of the chamber is avoided, produces in one cubic centimetre of air at S.T.P. such a degree of conductivity that one electrostatic unit of charge is measured at the saturation current. Confusion arises from two points

(a) This is a definition of a unit, but the quantity for which the roentgen is a unit was not defined.

(b) The word "quantity" is synonymous with "amount" (ie quantity is that for which a unit is devised, mass is the quantity for which gram is a unit.)

In I937 an important change was made to the definition[I3]:- The international unit of quantity or dose of X rays shall be called the roentgen. This will be the quantity of X or gamma radiation such that the associated corpuscular emission per 0,00I293 gm of air produces, in air, ions carrying I esu of electricity of either sign. However this definition leads to three confusing points

(a) The inclusion of the phrase associated corpuscular emission changes the concept of the roentgen. It now described the field at a certain point rather than the energy locally absorbed there.

(b) The interpretation of the word "quantity" is even less obvious than before. It is now synomymous with dose, which in turn remains undefined.

(c) It seems that the I.C.R.U. evidently intended"dose" as a measure of the radiation field itself rather than the ionisation imparted to the air.

Up to I937 "dose" had taken three interpretations. As well as

those described above it could also be "the energy deposited in tissue". The latter definition was ruled out in the I937 report. It was not until I956 (I4) that the I.C.R.U. eventually made roentgen a unit of exposure dose and six years after in I962 (I5) the enigma surrounding the definition of "dose" was finally resolved. It is necessary not only to list the units used in present day dosimetry but to give their current interpretation. It does seem at long last that most of the units and measurements required in dosimetry are now defined. The remaining difficulty is to devise improved, and in some cases new methods, of measuring them. With the recommendation that roentgen should be the unit of exposure (I4) dose came the following statement. "The I.C.R.U. feels that it is desirable to orient the thinking of radiologists to a more general use of the rad and the term 'absorbed dose'. Since the introduction in I953 of energy units (rad) and the term 'absorbed dose', it has become necessary to avoid possible confusion between the term 'absorbed dose' and the more generic term 'dose' as measured in roentgens. In I953, the Commission considered the term 'dose' to be a rather general and sometimes vaguely used medical and pharmaceutical concept that could not very well be defined and the decision as to its definition was left open. The Commission is still of this opinion and feels that the term for the quantity expressed in roentgens should be more restrictive 'exposure dose' is therefore recommended." In I962 (I5) 'exposure dose' was changed further to 'exposure', thus at last divorcing it from dose completely.

Exposure (X) was defined as :-

$$X = \Delta Q / \Delta m$$

and exposure rate as :-

$$= \Delta X / \Delta t$$

Where Δq = Sum of the electrical charges on all the ions of one sign produced in air when all the electrons are completely stopped.

Δm = Mass of the matter in the specified volume element.

Δt = Time period.

The roentgen, R = 2.58×10^{-4} coulombs / kg (ie 1 esu of charge per 0.001293 gm of air). An X-Ray beam for which the exposure is 1 roentgen imparts to 1 gm of air approximately 87 ergs and for soft tissue 94 to 97 ergs.[16] The concept of exposure is limited to electromagnetic radiation, but it is meaningful for any point within an irradiated material, vacuum or "free air", as it can characterise the radiation field prior to the introduction of a biological object. Attempts have been made to convert exposure to absorbed dose [17], but it has not been found possible for a given amount of gamma radiation to use a factor which does not contain a number of variable parameters and is

Fig 2.1

Quantities and units currently used in dosimetry.

exact under all conditions to convert from exposure to absorbed dose.

2.3 Rad and Absorbed Dose

Again after much confusion over the use of rep units (The definition of rep was changed three times!) in 1953[18] the I.C.R.U. made a real step forward in improving dosimetry terminology. Absorbed dose was defined as the amount of energy, of any ionising radiation, which is imparted to matter by ionising particles per unit mass of irradiated material at the place of interest. It is a considered as a macroscopic point function The absorbed dose is measured in rads, and the rad is equivalent to 100 ergs of energy imparted by ionising radiation to 1 gm of any material (ergs /cm^2 / sec^{-1}). For biological measurements this unit has no practical use as the material is not specified.[19]

Absorbed dose (D) = $\Delta E_d / \Delta m$

and absorbed dose rate = $\Delta D / \Delta t$

Where ΔE_d = Energy imparted to the matter in a volume element.

Δm = Mass of matter in a volume element.

The rad unit replaces the rep and has the advantage that its magnitude is not dependent on the average energy required to form an ion pair in air. However absorbed dose depends on geometric and material configuration, thus precise experimental measurements must be carried out in a biological specimen or suitable phantom.

2.4 Rem and Dose Equivalent

From a dosimetry viewpoint when considerations of neutron and gamma radiation effects are being made the dose equivalent is the only measurement which can meaningfully be applied. In 1962 (I5) the commission decided to use the term "R.B.E. dose" for radiobiology only. Dose equivalent is defined as the product of absorbed dose (D), quality factor, distribution factor and other necessary modifying factors. The linear-energy-transfer dependent factor by which absorbed doses are to be multiplied to obtain, for purposes radiation protection, a quantity that expresses the irradiation incurred by exposed persons be called Quality Factor. This puts on a common scale all ionising radiations. A distribution factor is also used to express the modifications of biological effect due to non-uniform distribution of internally deposited isotopes. The report also states "Provisions for other factors are also made". What these are and how they affect the dose equivalent is not explained, thus making the measurements rather general.

The unit of dose equivalent is the Rem. This can be said to be numerically equal to the dose in rads multiplied by the appropriate modifying factors. Although these quantities do not cover a number of theoretical aspects such as the physical dimensions, they do give an unequivocal specification of a scale that may be used for numerical expression in radiation. However in practice the dose equivalent is usually taken as absorbed dose multiplied by quality factor. It appears to be impossible with present day measuring devices, and attitudes, to define this quantity more precisely.

Perhaps it would be best in the long run to cut the losses and define a new unit for biological damage based on an entirely different concept.

2.5 Quality Factor and R.B.E.

Quality Factor (Q.F.) is a constant used for the conversion of absorbed dose to dose equivalent. In 1962 (I5) the I.C.R.U. required a name for this linear-energy-transfer dependent factor by which absorbed doses are to be multiplied to obtain for all ionising radiations the irradiation incurred by exposed persons. Provisions for other factors had also to be made. Thus a Distribution Factor (D.F.) was to be used to express this modification of biological effect (As previously stated).

Originally Q.F. developed from the concept of Relative Biological Effect (R.B.E.).(3) It was found that equal absorbed doses delivered by different ionising radiations produced varying degrees of injury to body tissue. The R.B.E. of one radiation with respect to another is defined as the inverse ratios of the absorbed doses required for equal effect. The R.B.E. of different kinds of ionising radiations is usually indicated as relative to that of conventional therapeutic X-radiation (200 Kv) as unity. The I.C.R.U. however recommends that the term R.B.E. be used only in radiobiology. (I5) Tables of Quality Factors for the different radiations is given in section I. Quality Factor is progressively more difficult to measure with increase in neutron energy. However the slow decrease of Q.F. with increase in neutron energy has been observed using a Rossi type tissue

equivalent chamber.[20]

2.6 Radiation Field

The obvious way to describe a radiation field at a point is to count the number of quanta, particles or rays per unit area per unit time at this point. There are now two acceptable approaches to the definition of radiation field. The first is represented by fig 2.2 with a parallel beam of radiation passing near a point P. It is clear that the number per unit time and unit area of the circular plane is then equal to the particle flux density. For multidirectional radiation fields the fixed plane cannot be traversed by all the rays perpendicular ly so the number of rays striking the plane will be dependent on the orientation of the plane (unless the field is 100% isotropic.) However if the plane is allowed to move about its centre so that it can theoretically intercept each incident ray or particle perpendicularly the maximum number of rays will be counted. This can be seen in fig 2.3 where the periphery of the circular plane sweeps out a spherical surface as it rotates. So a definition can be formed of particle flux density as:-

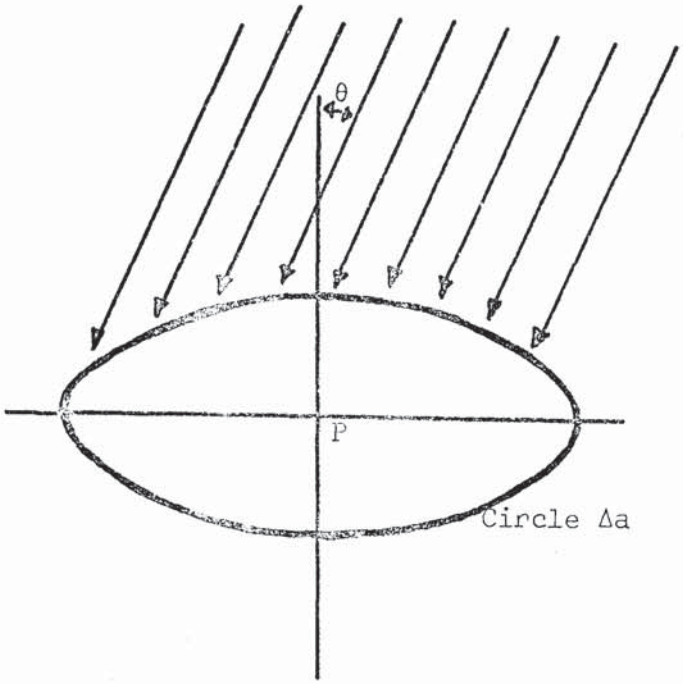
$$\phi = {}^{\Delta N}/_{\Delta a \Delta t}$$

Where Δa = area of the circle that sweeps out the sphere

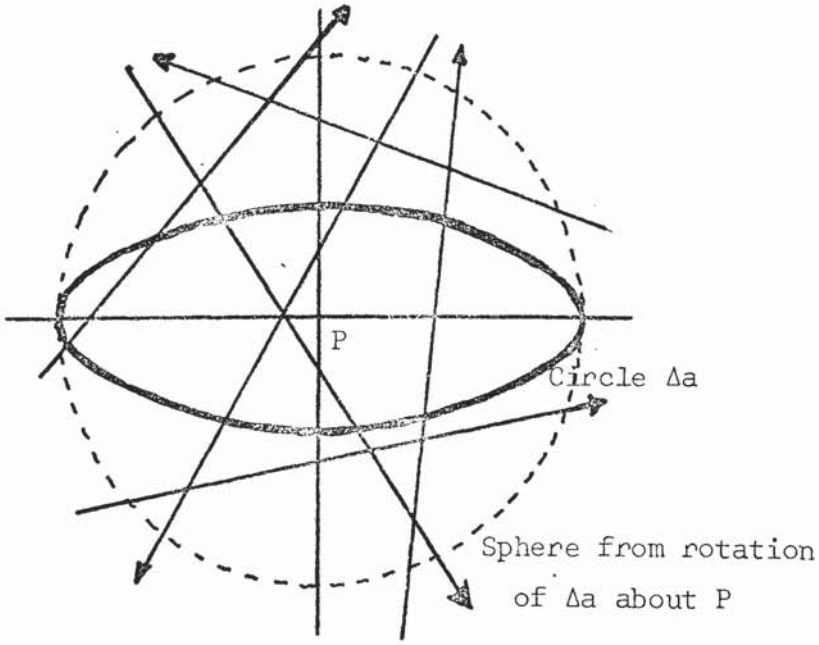
ΔN = number of particles that enter this sphere in time Δt

For brevity this may be called flux density, but not flux. It is

Fig 2.2 and 2.3



Flux density and fluence representation in a monodirectional and multidirectional field.,



more appropriate to define flux density for dosimetry purposes in this way as interest is centred on the effects produced at a point by radiation irrespective of the direction of incidence. It may be necessary sometimes to express the total number of particles or rays per unit area entering the sphere during any interval of time. This is the particle fluence or fluence:-

$$\Phi = {}^{\Delta N}/_{\Delta a}$$

This quantity is the time integral of flux density. Lastly similar arguments can be applied to energy considerations. If the amount of energy is measured entering the sphere per unit area and time this will give an expression for intensity or energy flux density:-

$$I = {}^{\Delta E_f}/_{\Delta a \Delta t}$$

ΔE_f = Sum of all the energies, exclusive of rest energy, entering the sphere.

The time integral of energy flux density is the energy fluence:-

$$F = {}^{\Delta E_f}/_{\Delta a}$$

2.7 Kerma

Just as absorbed dose describes the deposition of energy in a material through Coulomb interactions so kerma (kinetic energy

released per unit mass.) describes the transfer of energy by the uncharged particles to the charged particles per unit mass of the irradiated medium. This is basically the same as the concept of "first collision dose", which has proved to be so useful in fast neutron dosimetry.(I5) It is also closely related to the energy equivalent of exposure in an X-ray beam.

$$K = {}^{\Delta E_k}/_{\Delta m}$$

$$\text{Kerma rate} = {}^{\Delta K}/_{\Delta t}$$

In actual measurements Δm should be so small that its introduction does not appreciably disturb theradiation field. In practical terms it is considered a macroscopic point function. When charged particle equilibrium exists and bremsstrahlung losses are negligible the kerma is then equal to the absorbed dose at that point. The energy of charged particles is taken here to include those particles produced in secondary processes occurring within the volume element. Thus Auger electrons are included in ΔE_k (2I)
The fundamental physical description of a radiation field as given in section 2.6 is based on the knowledge of energy flux density at all relevant points. In dosimetry and neutron monitoring for radiological protection it may be more convenient to describe the primary radiation in terms of a kerma rate for that specified material. In fact this is now common practice and many figures have been evaluated for kerma / fluence. Figs 2.5 and 2.4 gives the ratio of kerma in some commonly used materials in dosimetry and kerma in standard man.(73) Standard man consists

Fig 2.4

Ratios of kerma in materials and composition gases commonly used in dosimetry, to that of standard man (10% H, 18% C, 3% N, 65% O and 4% other elements such as sulphur, phosphorus, etc).

Fig 2.5

Units are in ergs / gm / 10^7 n / cm^2 .

Kerma (Energy transfer) for elements, compounds and gases commonly used in dosimetry

of 65% oxygen, 18% carbon, 10% hydrogen and 3% nitrogen by weight.(74)

2.8 Linear Energy Transfer L.E.T.

Is a direct descendent from linear stopping power (ie mass stopping power times density) The I.C.R.U. however wished to emphasize the energy deposited in the medium rather than that lost by the charged particle and also be applied to energy losses that could be considered as local.(15) The linear energy transfer (L) of charged particles in a medium is the quotient of dE by dl:-

$$L = {}^{dE}/_{dl}$$

Where dE = average energy locally imparted to the medium by the charged particles of specified energy.

dl = distance traversed by the charged particle.

The I.C.R.U. then state that the term "locally imparted" may refer either to a maximum distance from the track or to a maximum value of discrete energy loss by the particle beyond which losses are no longer considered as local. In either case the limits chosen should be specified. Also the concept of L.E.T. is different from that of stopping power. The former refers to energy imparted within a limited volume, the latter to loss of energy regardless of where this energy is absorbed.

In practice irradiated material is traversed by charged particles

having a range of kinetic energies and hence a range of L.E.T's. The resultant distribution can then be expressed in a number of ways. For example the distribution of track length in L.E.T., can be expressed as the fraction of particle track length at a L.E.T. of L, per unit interval of L. (ie; L and L + dL). Alternatively the fraction of the dose delivered at L is specified by the distribution function D(L). As previously stated it is virtually impossible to establish generally acceptable criteria that distinguish between local and distant energy losses. An assessment toward a more realistic value of local energy loss has been attempted by a number of researchers in considering delta rays in excess of some cut off energy as separate particles. This has the advantage of having delta rays contributing short sections of high L.E.T. track to the overall distribution, without their kinetic energy being included in the energy transfer loss of the primary particle. Depending on the value of the chosen cut off energy value, rather different L.E.T. spectra result particularily at very high particle energies.

To further complicate events the I.C.R.U. and I.C.R.P. (1963) suggested that whenever the L.E.T. value is of importance a subscript be added to the acronym L.E.T. Thus $L.E.T._{500}$ and $L.E.T._{\infty}$ would respectively represent a particle L.E.T. when 500 ev delta rays are being considered to be separate particles and the latter when all delta rays are considered part of the primary track. Calculations involving L.E.T., other than $L.E.T._{\infty}$, are consequently complex and uncertain because of incomplete knowledge of the rate of energy loss of electrons below a few Kev.

Therefore in radiation protection where the Q.F. has been related to L.E.T., the latter is understood to be L.E.T.$_\infty$. Further investigation and understanding of L.E.T. are beyond the scope of this thesis as they envolve considerations of event size and local energy density.

2.9 Charged Particle Equilibrium

This is one of the important concepts in dosimetry as it is usually described as the balance of incoming and outgoing electrons at the boundary of a specified volume. An easier way to understand the phenomenon is to consider the equality of kerma and absorbed dose at the point of interest.

$$\Delta E_{d/\Delta m} = \Delta E_{k/\Delta m} = F\mu_{k/\rho}$$

Basically for a specified volume V Charged Particle Equilibrium exists if each charged particle leaving V, carrying a certain energy, is replaced by another identical particle entering with the same energy. Most practical γ ray and neutron dosimetry is based on the equality of kerma and absorbed dose under C.P.E. conditions. However notable cases where C. P.E. fails are

(a) Close to a point source of radiation (where intensity changes rapidly with distance)

(b) Near boundaries between materials of differing composition.

(c) At high primary radiation energies where the range of the secondary particles is no longer negligible compared to the mean free path of the primary rays. Unfortunately these three cases

represent formidable dosimetery problems and are the main contributors to the present day tolerances of up to ± 30% on neutron dosimeter accuracy.

2.10 Instrument scale calibrations

It is worthwhile at this point to show how the foregoing confusion seen amongst units and unit interpretations has been handed down to the manufacturers of present day dosimetric instruments. For this illustration it is more relevant in the circumstances, to investigate the position of the commercial dosimeters (for definition see sub-section 7.1). Firstly the units adopted for X-Ray and gamma ray instruments manufactured by three leading companies are given. The following information is taken directly from their respective advertising literature :-

(a) Manufacturer A The indicator moves up the scale and its position gives the radiation exposure in roentgens or m.roentgens. Essentially rate-independent for gammas up to 600,000 r / hr.

(b) Manufacturer B Radiation measured is X-Ray and gamma - Accepted standard for precise roentgen measurement - Valid readings in pulsed gamma fields of as high as 5×10^4 R / sec.

(c) Manufacturer C Instruments are manufactured in standard ranges between 0 - 200 mr and 1000 r. - Extremely efficient ion collection, extending the use of the dosimeter to the measurement of high dose rate - Measures integrated dose.

Before an appraisal is made of the above, the following points should be borne in mind, all taken from the I.C.R.U. report 10a :-

(a) The rad to be used solely for absorbed dose.

(b) The roentgen to be used solely for exposure.

(c) When the word rad may be confused with the symbol for radian, it is permissible to use "rd" as the symbol for rad (ie not "r" or "R")

(d) The symbol for roentgen is "R". (see fig 2.I)

Further to the above points the National Commission on Radiological Protection (N.C.R.P.) recommend exposure levels all of which are given in rem. (ie not rad or roentgen)

Therefore on analysis instruments of manufacturer A and B give exposure (although B calls it roentgen dose). Manufacturer C however is far more subtle the unit which they give their instrument is not defined "r" could be roentgen or rad. Also no indication is given as to the reading being exposure or absorbed dose, it is just termed integrated dose. The dosimeter of Manufacturer A is rate-independent for gammas up to 600,000 r / hr (rads / hr or roentgens / hr ?), the dosimeter of Manufacturer B is valid in pulsed gamma fields of 5×10^4 R sec but dosimeter of Manufacturer C is valid for the general term "high dose rate"

One further point, if the nationally accepted safety levels for personnel are given in units of rem then it does seem logical that personal dosimeters should be calibrated in rem. This is too evident as a glaring example of the position which develops when the accepted governing body continually changes ideas and definitions of the units. It is to some extent inevitable in the natural process of development, but the manufacturers concerned stand little chance of keeping pace with the ever changing scene

of radiation units and permissive levels.

As a last illustration of this point a fast neutron and gamma dosimetric system is investigated. Again quoting from the advertising literature:- "The model consists of two direct reading pocket dosimeters. The first collision tissue dose of fast neutrons is determined by subtracting the reading of the neutron insensitive dosimeter from that of the tissue - equivalent dosimeter, the range is 200 millirads full scale." When gamma radiation dose is measured the absorbed dose (rad) is directly equal to the dose - equivalent (rem) as the Q.F. is unity for the whole gamma ray energy spectrum. However, as mentioned previously this is not the case with neutron dose measurements. So in the field of personal monitoring it is important that unit of measurement is the rem not rad. Also the manufacturer has now been "caught" with the term first collision tissue dose. It is now obsolete and is replaced by "kerma in tissue" [72] as the correct term.

Finally one point which covers both gamma ray and neutron measurements, if the units are given in rads on the instrument then it should be made clear whether these units are air rads or tissue rads.

2.II Specific gamma - ray constant

This constant has no specific use in dosimetry other than evaluating the strength of a source. It is, as the name suggests only used for gamma - ray emitting nuclides such as cobalt 60.

The symbol for the constant is Γ :-

$$\Gamma = {}^{\Delta X}/_{\Delta t}\ l^2/_A$$

Where l = Distance of specimen from source.

${}^{\Delta X}/_{\Delta t}$ = Exposure rate at a distance from the source

A = Activity in curies of the source.

For radium the value of the constant is determined for a filter thickness of 0.05 cms of platinum and in this case the activity (A) is replaced by the weight of radium used in grams (g). Where 1 mg of radium equals 0.988 mCi.

2.12 Summary

It is evident that after the I.C.R.U. Report 10a in 1962 a real step forward was taken in the development of concepts, and mathematical techniques related to macroscopic distribution of energy absorbed from ionizing radiation in irradiated matter. However it seems that the controversy surrounding ron, rad and rem has been passed onto L.E.T. While a variety of averages (particularly dose averages of L.E.T) have been used in many radiobiological considerations, little or no use has been made of detailed distributions previously discussed. In the case of L.E.T. this is probably due to the indirect connection between distribution and energy deposition in irradiated tissues. For example an approach has been outlined for the utilization

of Z distributions in interpretations of survival curves, but because of paucity of both physical and biological data no correlations have as yet been attempted.[23] So it seems that as more sophisticated parameters of radiation quantity are developed giving greater precision, much simplicity and generality is lost as a result. This is plainly evident in considerations of L.E.T.

In the foregoing section only those units and terms which have been a basis for those units and terms are included. There are many others (see fig 2.1) with which this project is not directly concerned. The unit used in this project for sensitivity is expressed as a function of a scale division per rem of radiation. Basically this means that an absorbed dose rate is calculated and from a reading taken after a subsequent irradiation, a corresponding divisions per millirem figure given. For the liner chambers a neutron to gamma ratio was taken as only the relative sensitivity of the liners was required. This ratio was based on neutron dose equivalent to gamma dose equivalent and neutron absorbed dose to gamma absorbed dose (see section 6).

SECTION 3 CONDENSER IONISATION CHAMBERS.

3.1 Mechanisms involved in Dosimetry

The position up to 1967 of neutron dosimetry was investigated with a view to choosing a possible system for development. It was decided to concentrate on a system which would if possible give :-

(a) An immediate read out in terms of dose - equivalent

(b) Be completely self contained

(c) Measure neutron radiation and neutron plus gamma radiation in the desired dose - equivalent, giving automatic allowance for the differences in Q.F. between neutron and gamma rays.

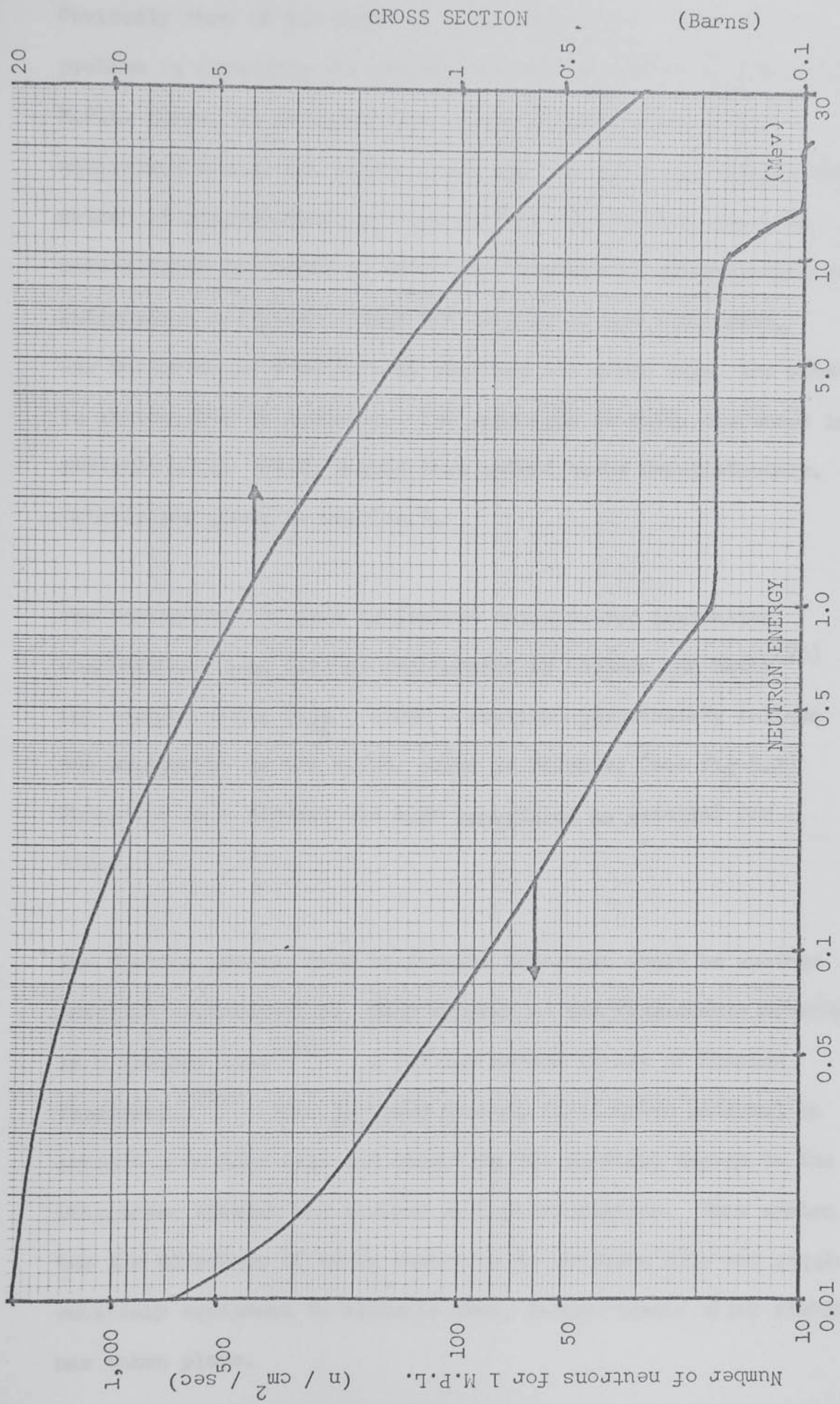
One method was to attempt to modify in some way the present Stephen gamma ionisation chamber to neutron and / or neutron plus gamma radiation. From observed literature the amount of work completed on neutron pocket quartz fibre dosimeter with a read out in rems, was very little. However some relevance was found in the work done on proportional counters and ionisation counters. Some ideas for wall liners and filler gases were taken from these, with a view to adaption in a condenser ionisation chamber. The neutron being uncharged requires "conversion" into a charged particle before any ionising takes place. One method commonly used in proportional counters is to use the (n, p) reaction. The knock - on proton is most easily obtained from the hydrogen nucleus, therefore the liner used is a hydrogenous plastic, this being a most convenient form of supplying hydrogen

in a dense form. The plastic containing the most hydrogen per molecular weight is Polythene. However the hydrogen cross - section over the energy - range does not entirely follow the reciprocal of the M.P.L. curve as can be seen from fig 3.1. The gas fillings also affects the response of the chamber and are often used to manoevre the response, for an increase or decrease in sensitivity[55] One method of combating this problem is to employ a device which behaves according to Bragg's Law, which is :-

"The amount of energy imparted by ionising radiations does not depend on the molecular arrangement of the absorbing atoms. Therefore if tissue equivalent material is used the energy lost per gram of tissue equivalent material will be equal to that of actual tissue."

Thus the stopping power of a compound is equal to the sum of the stopping powers of its atomic constituents. For a high degree of accuracy in following the response, Fano's theorem is linked with the above :-[24] In a medium of given composition exposed to a uniform flux of primary radiation, the flux of secondary radiation is uniform, independent of the density of the medium and independent of density variations from point to point, provided that the interactions of the primary radiation and the secondary radiation with the atoms of the medium are both independent of density. Thus the mass stopping power ratio for a "matched" cavity ionisation chamber will be unity, in the absence of any density effect whatever the size of the cavity and the pressure in it .

Fig 3.1



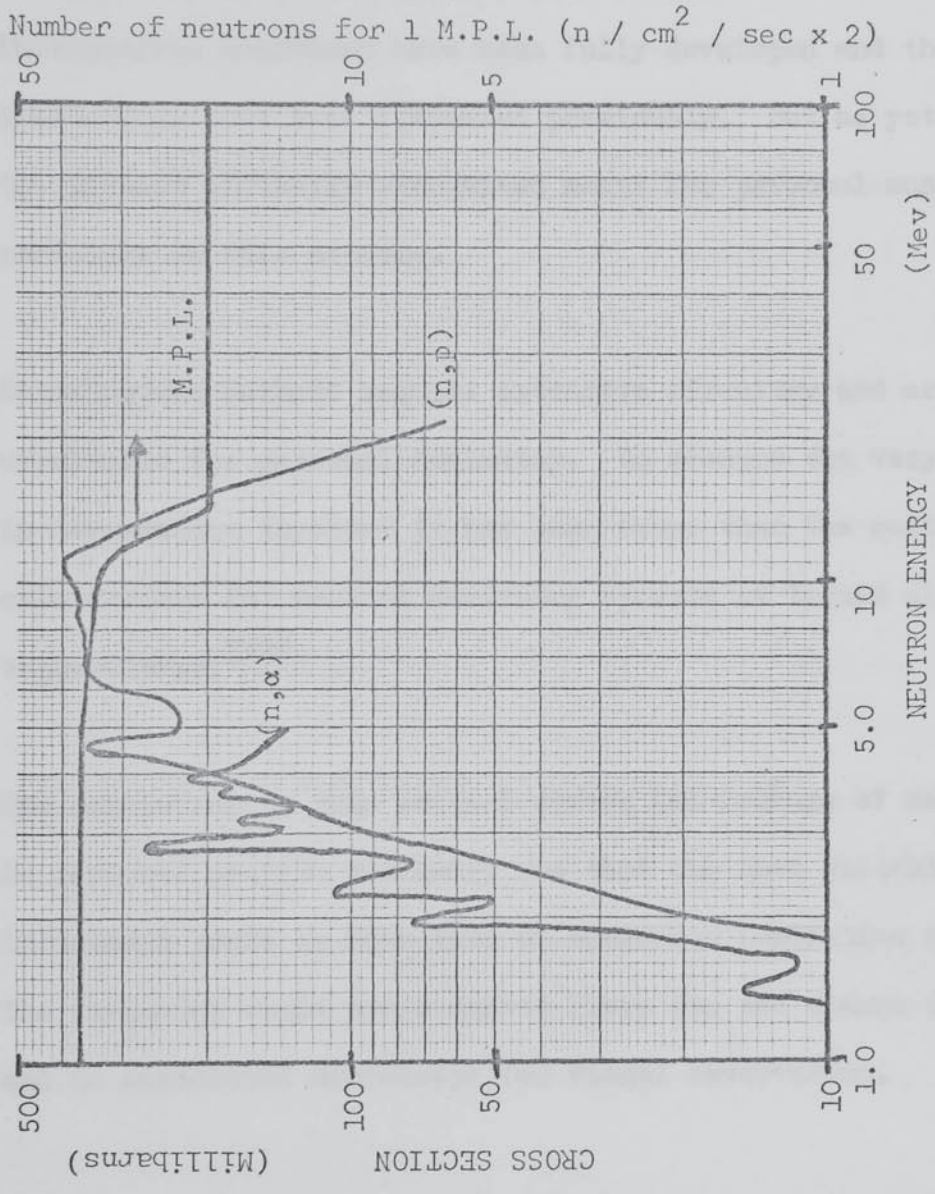
Neutron cross section for hydrogen and the M.P.L. curve.

Obviously then if the liner is tissue equivalent there is no problem in arranging the response of the instrument to follow the M.P.L. curve, it is automatic. Unfortunately these tissue equivalent liners are highly gamma ray sensitive and require some method of compensation for this effect. One leading American manufacturer has tried to solve this problem by incorporating two independent dosimeters under the heading of one instrument. One dosimeter is sensitive to neutrons and gamma rays, the other is insensitive to neutrons. The principle is fully discussed in section 7.2.2. Tests showed this method to be unsatisfactory. Details are given in section 7.

The measurement of dose equivalent using paired ionisation chambers has been further recommended by Goodman and Rossi[25] One element whose (n,p) cross - section approximately follows the reciprocal of the M.P.L. curve is Sulphur. (see fig 3.2) Therefore this element was also considered as material for a liner.

The fission process used in fission detectors would be another possible consideration. Here uranium as the fissionable material is a chamber liner[26]. A further method is use of Fission Fragments.[27] This involves placing fissionable material in contact with mica foil and observing the physical damage to the mica after fission has occured during irradiation. This system has the advantage of being sensitive to neutrons only but requires subsidary equipment to evaluate these damage tracks after etching has taken place.

Fig 3.2



Neutron cross sections for the principle charged particle reactions in sulphur and the M.P.L. curve.

Thermoluminescent devices are being extensively developed, but in most cases the integrated dose reading is destroyed by the measurement process.[12,28] However all solid state integrating devices, with their information erased by a read-out or not, require equipment to convert the radiation effects to a dose reading. There is no single device giving immediate read-out. Photographic emulsions have been fully developed and their limitations have been discussed previously. But as yet they are the present offically recognized means for personal monitoring, certainly in this country.

Calorimetric Methods require extensive circuitry and are unsuitable for personal dosimetry. To measure the very small rise in temperature involved it has been found that the most successful calorimeters for neutron dosimetry operate at liquid nitrogen temperatures.[29]

The conclusion to this initial survey and summary of methods used in personal neutron dosimetry was that the most suitable form of instrument would be some form of condenser ionisation chamber. The dosimeter would use a quartz fibre for the charge indication and an associated microscope for visual observation.

3.2 Mechanisms involved in the Condenser Ionisation Chamber

Since the most satisfactory method of measuring mixed neutron and gamma radiation appears to be some form of modified condenser

ionisation chamber it was thought necessary to examine this device in detail and appreciate the problems in manufacture, with a view to ultimate adaption for neutron measurement.

From fig 3.3. it can be seen that the dosimeter is a sophistication of the simplest form of integrating ionisation chamber[30]. The centre electrode is charged to about 150 volts and then isolated by the mechanical retraction of the charging bellows. The electrical capacitance (a few picofarads) between the electrodes provides the necessary charge storage. The amount of charge in the instrument is indicated by means of the image of a quartz fibre on a graduated scale. Radiation passing through the chamber will provide ionisation and reduce the charge in proportion to the radiation. This causes the quartz fibre to collapse across the scale. The fibre movement can be observed through a microscope system when the instrument is held up to the light. Thus an immediate indication of the integrated gamma dose is given when required.

3.3. Terminology and basic design problems of Condenser Ionisation Chambers.

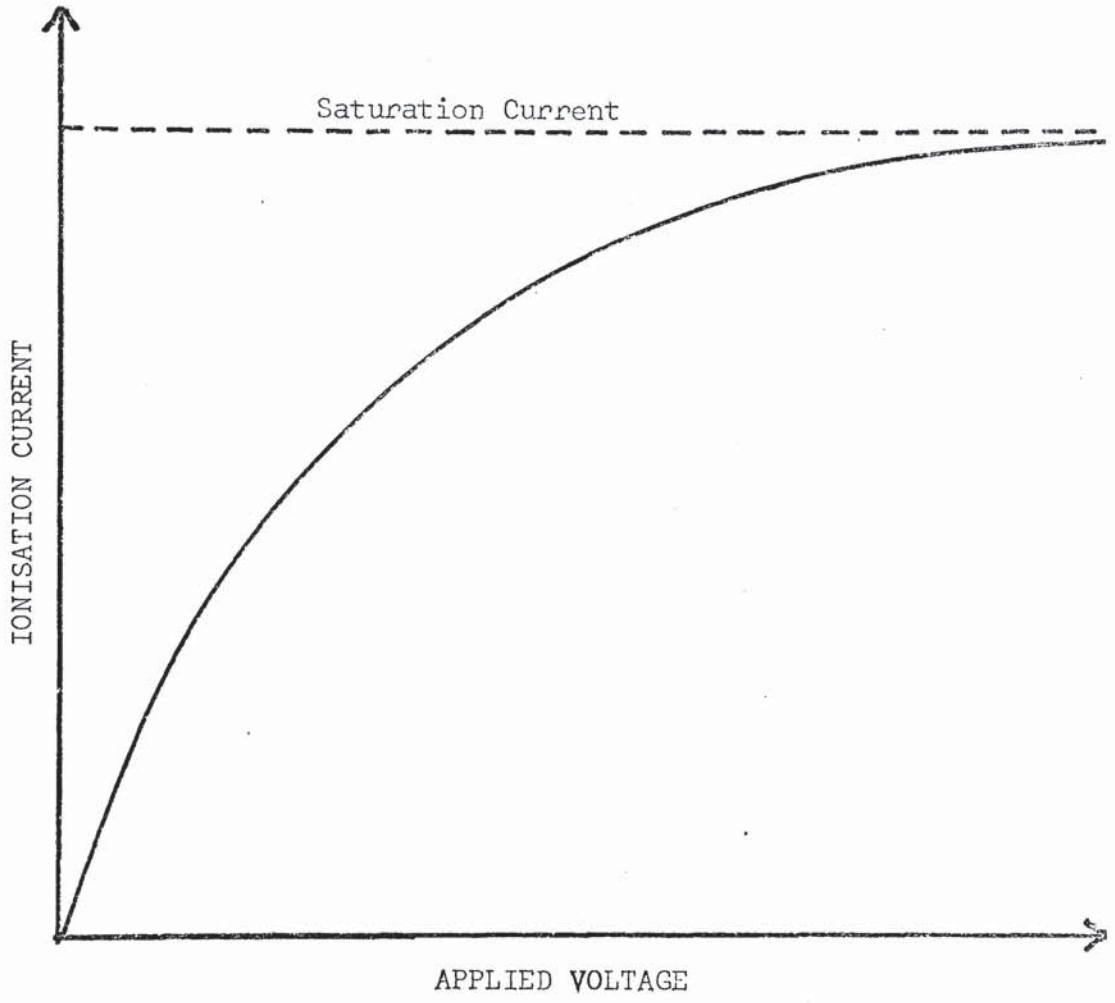
3.3.1. Saturation Current

As the voltage difference between the ionisation chamber electrodes is increased the current collected increases, as in fig 3.4. It moves towards an asymptotic value which is known as the saturation current for the given radiation intensity. The plot

Fig 3.3

Simplified sectional drawing of a gamma ray quartz fibre dosimeter, manufactured by R.A.Stephen and Co Ltd.

Fig 3.4



Typical saturation current.

of voltage versus current is called saturation curve. At low voltages recombination occurs which can be reduced by increasing the field strength by raising the voltage and / or reducing the distance between the electrodes. The maximum field strength however is governed by the onset of ionisation by collision giving charge multiplication etc. Usually with an ionisation chamber containing a centre electrode the saturation current can be obtained simply from a graph of voltage against current. However it has been found that with the more complicated centre electrode configuration of a quartz fibre device resulting in correspondingly complicated field shapes, this is not the case. There is no apparent saturation level! (90)

3.3.2. Initial Recombination

This occurs when the positive and negative ions formed in the track of a single ionising particle meet and recombine. This effect however only becomes important when there is high ion density. For example with alpha particle tracks at atmospheric pressure or electron tracks in a high pressure gas. The chambers used in this project are at atmospheric pressure, and there are no alpha particle considerations, therefore recombination losses will be a minimum. (at less than 1%.)(31) However in condenser chambers the voltage falls during radiation exposure. So if recombination losses are to be a minimum an initial voltage must be chosen which leaves the final voltage well along the plateau of the saturation curve.

3.3.3. General recombination

This covers recombination when the positive and negative ions meet as they drift towards the opposite electrodes. From an observation of the above effect the collection efficiency of the chamber can be defined as the ratio of the actual current to the ideal current. There have been many studies on current conductors by ionized gas, but these are not within the scope of the project.[32, 33] In ionisation chambers recombination effects only become appreciable at high dose rates when there are many ionisation tracks close together. [34]

3.3.4. Cluster Theory

When ionisation chambers are taking radiation dose measurements, the principal cause of non saturation of the current when the potential is sufficiently increased to avoid volume recombination, is the initial recombination. Jaffe has set forward a theoretical treatment of columnar recombination.[35] But as the majority of ionisation in the tracks occurs in small clusters, the theory of Lea, and the modification of the Jaffe theory are more applicable.[36] In recent years there have been very interesting proposals to measure the effect of L.E.T. on the Q.F. of radiation using recombination effects.[37, 38, 39]

3.3.5. Polarity Effects

Sometimes when the polarity of the collecting potential is reversed the collection effeciency alters. This may be due to a variety of reasons:-

(a) Contact, thermal or electrolytic emf's in the measuring circuit.[40,41]

(b) The emission of secondary electrons in the direction of the incident photons.

This Compton current for high energy photons can become appreciable.[42] However from tests taken with the liner chambers and the commercial dosimeters this effect is negligible.

3.4. Condenser Ionisation Chamber Wall Liner.

For an ionisation chamber to have neutron or gamma ray sensitivity some form of liner is required which undergoes a neutron or gamma ray initiated reaction. This produces charged particle radiation which in turn causes ionisation in the gas filling of the chamber. For reasonable accuracy the chamber should be designed to comply with the Bragg - Gray theory (discussed in section 3.5). This imposes no explicit restrictions on the geometrical shape of the cavity and thus holds for cylindrical chambers.[43] For measuring absorbed dose in a particular medium both liner and gas filling should be matched to that medium. Two materials are said to be matched for a particular type of radiation if there is equal flux density and energy distribution of secondary ionising particles in the liner and the gas.[44] With regard to cavity theory, a liner thickness equal to the maximum range in the liner of the most energetic secondary electrons is equivalent to an infinite medium. Since no secondary

particles originating at greater distances can reach the cavity. But the liner absorbs and scatters the secondary electrons and it is obviously undesirable to have the liner thicker than necessary. Gray (1937)[45] found that in graphite most energetic secondary electrons generated from the gamma rays from radium have a range of 0.6 cms. But over half of these electrons were produced within the last 0.2 mm of the surface of the liner, and only 10% is due to electrons coming from a depth greater than 0.7 mm. Therefore it is not necessary to have a wall thickness equal to the maximum range, but a compromise should be found. The absorption of primary radiation in the wall may reasonably be neglected.[46] As it has been estimated that experimental values of correction coefficients of primary photons in the walls of ionisation chambers for a ^{60}Co source are in the order of 3.2% for a wall thickness of 1 gm / cm^2. On average the wall thickness of condenser ionisation chambers is 250 mgm / cm^2. (taking into account the maximum range of 14 mev protons) showing very little primary radiation absorption.

3.5. Cavity Theory.

The application of cavity theory presupposes ideal experimental conditions eg:-

(a) Collection Efficiency

It is assumed all the ion pairs formed in the cavity are collected at the electrodes without recombination losses or the occurrence of further ionisation by collision. At high dose rates this is not the case.

(b) Leakage Current

The leakage current across the insulators in the chamber is

negligible compared with the ionisation current. (Stephen claim less than 2% per 24 hrs on the gamma pocket dosimeter.)

(c) Liner Material

As stated Gray has shown that over half of the ionisation in the cavity is due to electrons produced within the 0.2. mm of the cavity boundary. If a layer of graphite is deposited on the inner surface of a nonconducting liner, then the liner is not homogeneous and contributions from the graphite will have to be considered. The presence of an ion - collecting electrode differing from the liner causes difficulties.

Bragg - Gray relation

The main assumption here is that:- That energy lost by the electrons in crossing a volume is equal to the energy absorbed within that volume for both the solid and the gaseous volume elements.(47)

The basic equation is :-

$$E = S\ W\ J$$

Where E = Energy deposited per unit mass.

S = The mass stopping power ratio averaged over the spectrum

W = Average energy expended in producing an ion pair.

In practice delta ray production upsets this equation as the Bragg - Gray relation employs a continuous ("On the spot") energy loss model. Laurence in 1937, (48) modified the above to account for electron stopping power ratio not being independent of the energy of the electrons. In 1955 Spencer and Attix (49) derived theory for the delta ray production and also for a liner and gas of different atomic composition. The Spencer Attix

Theory holds well for small cavities.

3.6 Summary

In general there is a low degree of accuracy designated for a commercial neutron dosimeter (± 20% claimed by American manufacturers for the fast neutron dosimeter sensitivity).
The main reasons for this are :-

(a) For the cavity size it is difficult to operate under charged particle equilibrium.

(b) The chamber cannot in practise have matched liner and gas filling (to comply with Cavity Theory).

(c) This results in difficulty in adhering to the M.P.L. curve for fast neutrons (The response to fast neutrons over the energy range does vary, but not in the correct way).

Also it is generally understood that this size of dosimeter cavity could operate under saturation conditions. (In practice it does not for reasons given in section 3.3.1.).

However if the instrument is to be pocket sized there is straight away an upper limit to its length and circumference. (Assuming a cylindrical instrument is the cheapest to produce). Therefore with this in mind a compromise was chosen for the chambers in which the liners were tested, between the lower limit governed by the measurements of the commercial dosimeter and the sheer practicality of producing small gold leaves. It has been found that in most cases the response of the smaller commercial dosimeter, containing the unusual centre electrode configuration,

was very difficult to estimate. However it was possible in all cases to back theoretical predictions with experimental results, as can be seen in sections 4 and 6.

SECTION 4. THEORETICAL CHOICE OF LINER

4.I Introduction

In neutron and gamma ray dose measurements there has to be a means of converting the uncharged incident particle or ray, to a charged secondary particle. This enables ionisation to take place within the sensitive volume of the chamber. The amount of ionisation indicating the amount of radiation received.

The choice of material for the liner is governed by the type of radiation and the unit in which the radiation is required to be measured. For example if the chamber is to measure exposure then the liner has air equivalence. To measure absorbed dose in a particular medium the liner and the filling gas should be matched to this medium. (eg; ethylene gas and polythene). The reasons for this requirement are given in section 3.5. The liner material should have a reasonably high cross section for a charged particle for the incident radiation which is to be measured. The charged particles produced are typically protons or electrons. From a literature survey taken of the various approaches to this problem in the field of proportional counters and electronic ionisation chambers, four types of liner were chosen for investigation:-

(1) Fissile Material.(neutron sensitive).

(2) Plastic (neutron sensitive and insensitive).

(3) Sulphur (neutron sensitive).

(4) Pure element of low Z (neutron insensitive).

The need and general requirements for a neutron insensitive

chamber is explained in section 8 . For an indication of the behaviour of the four liners listed previously typical sensitivities and discharge rates were calculated in the following section. These calculations were all based on the sensitive volume size for a commercial dosimeter. (Definition of commercial dosimeter given in section 7).

4.2 Gas filling

It is worthwhile when considering the liner requirements for an ionisation chamber to consider also the effect of the gas filling the sensitive volume. To achieve Bragg Cavity requirements gases should match the atomic composition of the liners. Therefore with a polythene $(CH_2)_n$ liner, ethylene (C_2H_4) or cyclopropane (C_3H_8) should be used. These two gases were investigated and the results are given in section 7.3. For the tissue equivalent plastics there is a tissue equivalent gas. Two such gases were obtained by mixing different gases:-[51]

(a) Hydrogen 38.1% Methane 22.2% Oxygen 37.6% Nitrogen 2.1%

(b) Methane 64.4% Carbon Dioxide 32.4% Nitrogen 3.2%

There are other mixtures which are in common use,[50] most of these however are explosive (as is mixture (a).), and contain water vapour which is detrimental to the correct functioning of the dosimeter.[22] For the standard commercial dosimeter used to measure exposure nitrogen is used to achieve air equivalence. This project made use of dried air as this was very easy to reproduce and from a production point of view would be easiest to apply and manufacture. (No ingassing, explosive effects,

anaesthetic properties,etc;). However if it is possible within the experimental arrangement to change the gas filling then investigations will give some indication of how W (the ionisation efficiency) affects sensitivity. If a charged particle is slowed down in a gas, the total number of ion pairs produced (I) is approximately proportional to the energy expended (E). The efficiency is specified in terms of its reciprocal ie:-

$$W = {}^{E}/_{I}$$

Slow heavy particles, such as alpha particles and fission fragments, ionise less efficiently than do the faster particles. For air the value of W is taken to be about 32.5 ev / ion pair while for ethylene it is about 26.5 ev / ion pair.[68] So there could be a possible 20% increase in sensitivity for ethylene filled dosimeters. The results however gave a much higher increase than 20% (see section 7). Presumably there was some neutron interaction with the hydrogen in the ethylene gas molecule which also contributed to the greater increase in neutron sensitivity.

A further variable parameter is the pressure within the sensitive volume. With some proportional counters the pressure is altered to give the required sensitivity for the particular flux density of the incident radiation. In general with an increase in pressure there is a corresponding increase in sensitivity until ionic recombination effects occur. With decrease in pressure there is an initial decrease in sensitivity then a rapid increase due to ionisation by collision. All the experimental investigations into gases are contained in section 7.

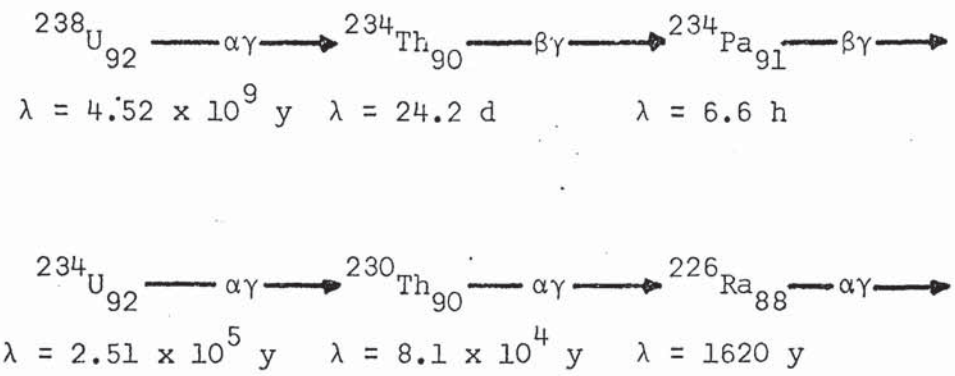
4.3 Fissile liner

The first liner considered is a coating of fissile material. Under neutron bombardment fission will occur and the energetic fission fragments produced are the ionising agents. Fission fragments have the advantage of a small range which is totally accomodated within the sensitive volume of a typical commercial dosimeter. These however, have the advantage of being able to discriminate against the alpha particle background which is always present due to the natural radioactivity of all fissile materials. In condenser ionisation chambers it is not possible to electronically discriminate against this alpha particle background, all events which cause ionisation will be recorded. Therefore in an initial consideration of a fissile liner it will be necessary to compare this ionisation caused by the background alpha particles with that which is produced by fission fragments from a typical neutron flux. Among the various fissionable sources which can be used for the production of fission fragments uranium is the most easily obtainable. It has the advantage also of low radiotoxicity allowing easier handling. Other fissionable materials, such as plutonium and neptunium, have been investigated but have not shown any marked advantage over uranium.[27] Therefore the calculations have been made for ^{238}U.

4.3.1 Alpha background discharge rate

To illustrate how the alpha particle background is produced, the decay scheme of ^{238}U is given in fig 4.1. Only ^{238}U is given as it constitutes 99.3% of natural uranium. From the decay scheme

Fig 4.1



Initial elements in the uranium 238 series.

it can be seen that the next alpha emitter after ^{238}U is ^{234}U, and this isotope has a half life of 2.48 x 10^5 years. Therefore assuming a reasonable standard of purification is obtained with the uranium, it will contain ^{238}U in equilibrium with ^{234}U. This would give on average two alpha particles per disintegration of ^{238}U. For the purposes of these calculations a coating or liner will assumed to be in a standard size commercial dosimeter. (The definition of the term "commercial dosimeter" is given in section 7).

Calculation

The approximate size of the surface area of the sensitive volume of a standard commercial dosimeter is given in section 3 as:-

Radius (r) = 0.60 cms

Length (l) = 1.88 cms

Therefore the coating will have to cover an area of:-

$2\pi rl$ = 7.1 sq cms ....(4.1)

If the density of deposition is approximately 100 $\mu g / cm^2$, then the mass of ^{238}U deposited is:-

= 7.1×10^{-5} gms

Now the number of ^{238}U atoms corresponding to this mass is:-

= $^1/_A(N_a \times M)$

Therefore N = 1.81×10^{17} atoms

The number of alpha particles given off from the above number atoms can be found directly from the decay constant of ^{238}U:-

Number of atoms decaying is given by

$^{dN}/_{dT} = -\lambda N$

$$= \frac{0.693}{t} \times N$$

= 0.89 atoms / sec

= 0.89 alpha's / sec

The average energy of an alpha from uranium is 4.5 Mev. This energy alpha will travel a range of 3 cms in air at N.T.P. From the Bragg Curve for alpha particles in air (fig 4.2) it can be estimated that with a chamber of 1.5 cms in diameter, 30% of the maximum possible ionisation is completed by the alpha particle before it crosses the chamber and strikes the opposite wall. Another assumption is that half of all the alpha particles produced will actually enter the chamber. Therefore the number of alpha's per sec entering the sensitive volume:-

= 0.13 alpha's / sec

Each of these particles has 4.5 Mev, therefore the amount of energy entering the sensitive volume.

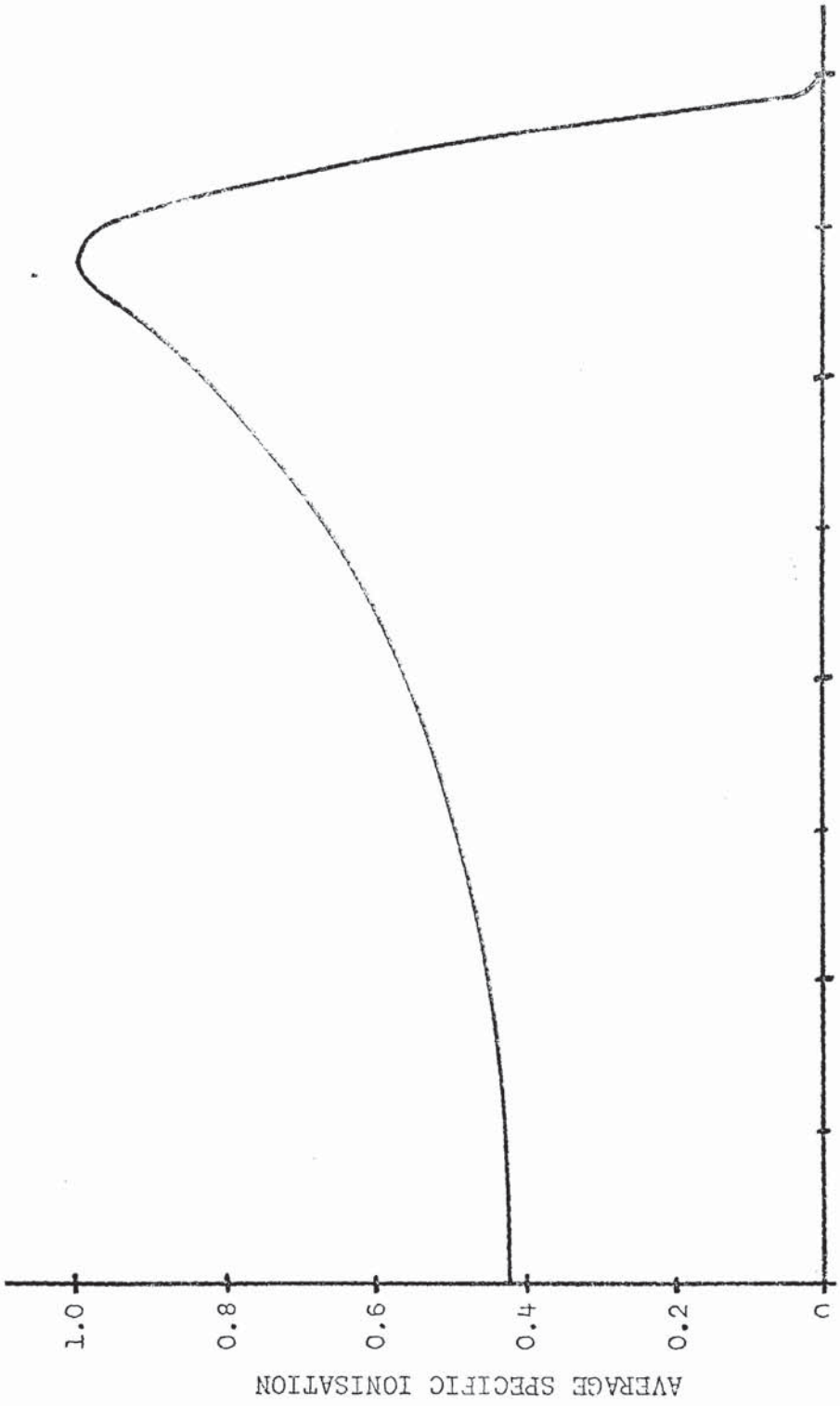
= 1.17×10^{6} ev / sec

(Taking into account two alpha's for every disintegration).

This will produce ion pairs, which for air requires an average about 3 5 ev, and this can be related directly to charge, on multiplying by e:-

Amount of charge E = 5.3×10^{-15} coulomb / sec

Fig 4.2



Bragg ionisation curve for polonium alpha particles at 15°C and 760 mm of Hg.

Now the charge on the chamber will be :-

$$Q = CV$$

$$= 7.0 \times 10^{-11} \text{ coulomb} \quad \ldots(4.3)$$

Where C = 1.0 μF

V = 70 volts for F.S.D.

Therefore the time taken to discharge the instrument will be Q / E

Therefore $T = 2.6 \times 10^4$ secs.

$T = 72$ hours.

4.3.2. Fission Fragments discharge rate

This is now compared with the amount of charge that Fission Fragment instrument would give in typical neutron flux. With a 100 μgm / cm^2 coating it is reasonable to assume that a negligible amount of fragments are absorbed in the coating. It has been estimated that 80 Mev / fission is produced with each neutron interaction.(52) Assuming as before that 3 5 ev is needed to form an ion pair and half of the fission fragments enter the chamber:-

Charge per fission $= 3.65 \times 10^{-13}$ coulomb

The number of fissions in time t:-

$$= N \sigma\phi t$$

Therefore the total charge produced in the sensitive volume (E_f):-

$$E_f = 3.65 \times 10^{-13} \times N \sigma\phi t$$

Where σ = 0.66 barns for 14 Mev neutrons.(53)

ϕ = 10 n cm^{-2}sec^{-1} for 1 M.P.L. of 14 Mev neutrons.

The time to discharge the instrument is given by :- $T = {}^{Q}/_{E_f}$

Therefore $T = 1.59 \times 10^6$ secs

$= 4,400$ hours

It can be clearly seen from these calculations that a discharge time of 72 hours for the alpha particle background is prohibitive for a condenser ionisation device, as there is no means of discrimination. This is further highlighted by the discharge time for a 1 M.P.L. flux taking 4,400 hours. The conclusion being that a Fissile Liner is unsuitable for a condenser ionisation device and not worthy of investigation.

4.4 Plastic Liner

It can be seen from considerations of neutron cross section that the reaction with the highest cross section is the knock-on proton reaction with hydrogen. Plastic liners offer a convenient method of supplying hydrogen in a concentrated, solid form. The neutron sensitivity of a plastic being calculated as directly proportional to the hydrogen content of the plastic molecule. Therefore the investigations seven plastics were chosen occupying the range between P.T.F.E. with no hydrogen, to polythene with maximum hydrogen content. Polythene is already extensively used in proportional counters as its neutron energy dependence curve for the production of charged particles is very similar to that of wet tissue.[54] To achieve even closer resemblence to the M.P.L. curve a varying thickness liner has been used.[55,56]

It was therefore interesting to compare the different behaviour of the 0.1 cm (0.040 inch) and the 0.15 cm (0.060 inch) polythene liners used in condenser ionisation conditions (see section 6). Tissue equivalent plastic is in common use for proportional counters,[51] but as one American manufacturer has produced a condenser ionisation chamber with this liner this plastic was not included for test (see section 3). There are many formula's for tissue equivalent plastic and many different compounds are used to achieve the correct proportions. The formula generally accepted at the present is $C_5H_{40}O_{18}N$.

In the theoretical calculations it has been necessary to consider the effect that the various elements in the plastic have on the range of the recoil proton. These ranges are given in fig 4.3 for two typical neutron energies. It can be seen that the plastics containing the heavier elements have a lower range for protons than those containing the lighter elements. This would contribute to the lower neutron sensitivity of the heavier element plastics, such as P.T.F.E. and P.V.C., as these liners were not made correspondingly thinner. Each liner being 0.10 cm (0.040 inches) in thickness.

For a calculated estimate of the behaviour of a plastic liner polythene was chosen. From this calculation the formula by which the maximum ranges, given in fig 4.3, are calculated can be seen. The range of a particle in any material is calculated from the formula:-[57,58,80]

Fig 4.3

Plastic	Formula	5 Mev	14 Mev
P.T.F.E.	$C_2 F_4$	21.6	129
P.V.C.	$C_2 H_3 Cl$	37.4	218
Polythene	$C_8 H_8$	40.0	270
Nylon	$C_{12} O_2 N_2 H_{22}$	40.6	271
Polythene	$C_2 H_4$	48.0	288

Units are in 10^{-3} cms.

Table of the maximum range of the maximum energy recoil proton from 5 and 14 Mev incident neutrons.

$$R_z/R_a = 0.90 + 0.275\ Z + (0.06 - 0.0086\ Z)\log E/M \quad \ldots\ (4.4)$$

4.4.1. Discharge rate for 5 Mev neutrons

Using the above formula the range of a 2.5 Mev recoil proton was calculated (It is a reasonable approximation to take the recoil proton energy as half the incident neutron energy).

$$\text{Range} = 13.4 \text{ mgm / cm}^2$$

The mass of polythene contained in a liner of this thickness in a typical commercial dosimeter is from section 4.3 :-

$$= 9.5 \times 10^{-2} \text{ gms}$$

As approximately 85% to 95% of the energy absorbed in tissue from a fast neutron beam is by the recoil protons from the elastic scattering with hydrogen, it is reasonable to ignore the contribution of the carbon nuclei. Therefore the number of hydrogen atoms in 9.5×10^{-2} gms of polythene is obtained by multiplying equation 4.2 by $1/7$:-

$$N = 8.21 \times 10^{21} \text{ atoms}$$

A 2.5 Mev proton will travel 10.4 cms in air. So with a chamber of diameter 1.5 cms, using the Bragg curve for protons, approximately $1/15$ of the maximum possible ionisation is completed by the proton before it crosses the chamber and strikes the opposite wall. Also assuming half of all protons produced actually enter the chamber gives:-

$$N\sigma\phi = 1.02 \times 10^{-2} \text{ interactions / cm}^2$$

Where σ = 2.0 barns.[53]

ϕ = 18 neutrons / cm^2 / sec for 1 M.P.L. of 5 Mev neutrons

Each of the protons entering the chamber has on average 2.5 Mev. Therefore as in section 4.3 relating this to charge:-

Charge per collision E = 1.14×10^{-14} coulomb

Time taken to discharge the instrument is Q/E where

Q = 4×10^{-11} coulomb (see section 4.3.1.). Therefore :-

T = 6.3×10^5 secs

= 1.68 hours

Sensitivity = 420 mrem F.S.D.

4.4.2. Discharge rate for 14 Mev neutrons

Using the range formula (equation 4.4) the range of a 7.0 Mev recoil proton was calculated:-

= 76.1 mgm / cm^2

The mass of polythene contained in a liner of this thickness in a commercial dosimeter (from section 4.3) :-

= 0.54 gms

Using the assumptions in section 4.4.1; a 7.0 Mev proton travelling 61 cms in air gives approximately $1/40$ of the maximum ionisation. From equation 4.2:-

$$N\sigma\phi = 4.17 \times 10^{-3} \text{ interactions / cm}^2$$

Where σ = 0.7 barns.[53]

ϕ = 10 neutrons / cm^2 / sec for 1 M.P.L. of 14 Mev neutrons.

Each of the protons entering the chamber has on average 7.0 Mev Therefore as in section 4.3 relating this to charge:-

Charge per collision E = 3.2×10^{-14} coulomb

Time taken to discharge the instrument is Q/E where $Q = 4.0 \times 10^{-11}$ coulomb (see section 4.3) :-

Therefore T = 5.1×10^5 secs

= 138 hours

Sensitivity = 356 mrem F.S.D.

From the above calculations it can be seen that polythene (ie a plastic) would be a worthwhile investigation as a possible liner for a commercial type dosimeter. It is unusual to note that the estimated time of discharge for 5 Mev neutrons was approximately 25% below that for 14 Mev neutrons. However this compares favourably with the actual experimental results. The sensitivity of 5 Mev average energy neutrons being 20% below that for 14 Mev neutrons. Thermal Neutron sensitivity was very difficult to predict using similar calculating methods. The ionisation in this case is caused by electrons from gamma rays emitted from neutron capture. In this capture process there is a 2.2 Mev gamma emitted from hydrogen and a 7.6 gamma emitted from aluminium. The aluminium forms part of the external case to

the commercial dosimeters and an even greater degree in the Liner Chambers. Also the other elements in the plastic would contribute high energy gamma rays, at roughly the same cross section, thus greatly increasing the sensitivity. When actual thermal neutron sensitivity measurements were made these predications were realised (see section 6 and 7). That is the hydrogen in the plastic was not, as for fast neutrons, the significant factor affecting sensitivity. For thermal neutrons the sensitivity seemed to be directly dependent on the amount of aluminium contained in the instrument.

If a comparison is made between the calculated sensitivity for polythene and the experimentally observed sensitivity of a liner consisting of 40% polythene and 60% carbon the latter is lower by a factor of $\simeq$ 6.0 (see section 7). However the neutron to gamma ray ratio's obtained with the liner chambers taken for 5 Mev average energy neutrons were used to estimate the effect of the carbon in the above liner. It was then found that with due allowance for the 60% carbon content this reduced the difference of the observed result to the calculated result to a factor of 3.5 lower. This is not an unreasonable figure considering the large number of approximations which were made in the calculated result.

4.5 Sulphur Lining

From considerations of neutron cross sections for the various elements it was thought that sulphur might be a suitable material

for a sensitive liner, see fig 3.2. Here the yield of the two competing reactions, the (n,α) and the (n,p), varied with neutron energy in approximately the right manner required to follow the reciprocal of the M.P.L. curve. However as sulphur had not been used before in this field no indications could be obtained as to its suitability. The two main charged particle reactions are :-

$^{32}S\ (n,p)\ ^{32}P$ useful energy range 2 - 20 Mev.

$^{32}S\ (n,\alpha)\ ^{29}Si$ useful energy range 2 - 6 Mev.

ie :-

$$^{32}S + n = {}^{32}P + p - 0.93 \text{ Mev}$$

$$^{32}P = {}^{32}S + \gamma + 1.71 \text{ Mev}$$

$$^{32}S + n = {}^{29}Si + \alpha + 1.16 \text{ Mev}$$

There is also a $^{32}S\ (n,2n)\ ^{31}Si$ reaction which takes place at much higher energies, but with low cross section,[59] so this will not contribute to the detection efficiency. However the useful energy range for the reaction is above 2 Mev, indicating that a sulphur lined dosimeter could only be used above this energy. Calculations as before, are for 5 and 14 Mev neutrons.

4.5.1. Discharge rate for 5 Mev neutrons

From the (n,p) reaction it can be seen that a 5 Mev neutron would approximately give a 4 Mev proton. Using equation 4.4 the range of a 4 Mev proton in sulphur is :-

$$= 37.0 \text{ mgm / cm}^2$$

The mass of sulphur contained in a liner of this thickness in a commercial dosimeter is from section 4.3 :-

$$= \quad 0.263 \text{ gms}$$

Number of atoms in this mass is from equation 4.2 :-

$$= \quad 5.0 \times 10^{21} \text{ atoms}$$

Half of the charged particles produced go into the chamber. Also a 4 Mev proton will travel in air approximately 28 cms, so using the assumption in section 4.3 this gives $^{1}/_{20}$ of the maximum possible ionisation within the sensitive volume, therefore :-

$$N\sigma\phi \quad = \quad 5.6 \times 10^{-4} \text{ interactions / cm}^2$$

Where σ = 0.25 barns (59)

ϕ = 18 neutrons / cm^2 / sec for 1 M.P.L. of 5 Mev neutrons.

Each of the protons entering the chamber has on average 4.0 Mev. Therefore as in section 4.3.1. relating this to charge :-

Charge per collision E = 1.82×10^{-14} coulomb.

Therefore the time taken to discharge the instrument is (as in section 4.3.1.) :-

$$T_p \quad = \quad 7.0 \times 10^{6} \text{ secs}$$

$$= \quad 1{,}920 \text{ hours}$$

Similarly for the alpha particle, a 5 Mev neutron would give approximately a 6 Mev alpha particle. As the method of evaluation

for all the following calculations is similar to that of the 4 Mev proton, only the differences in the basic facts and assumptions will be given. Therefore a 6 Mev alpha travelling 4.5 cms in air would give $^1/_4$ of the maximum possible ionisation. The cross section for the reaction σ is 0.25 barns and ϕ for 1 M.P.L. is again 18 neutrons / cm^2 / sec. This results in a total discharge time of :-

$$T_\alpha = 5.7 \times 10^6 \text{ secs}$$
$$= 1{,}575 \text{ hours}$$

Therefore the two reactions considered together would give a combined discharge time of:-

$$T = 3.15 \times 10^6 \text{ secs}$$
$$= 875 \text{ hours}$$
$$\text{Sensitivity} = 2.2 \text{ rem F.S.D.}$$

4.5.2. Discharge rate for 14 Mev neutrons

Here the (n,p) reaction would give a 13 Mev proton which travels 183 cms in air, giving $^1/_{90}$ of the maximum possible ionisation. The reaction cross section σ is 0.21 barns and ϕ for 1 M.P.L. is 10 neutrons / cm^2 / sec. This gives a discharge time of :-

$$T_p = 2.9 \times 10^6 \text{ secs}$$
$$= 835 \text{ hours}$$

Similarly for the alpha particle, a 14 Mev neutron would give approximately a 15 Mev alpha particle. This travels 21 cms in air giving $^1/_{16}$ of the maximum possible ionisation. The cross

section σ is 0.10 barns[59] and ϕ for 1 M.P.L. is again 10 neutrons / cm^2 / sec. This gives a discharge time of :-

$$T_\alpha = 7.0 \times 10^6 \text{ secs}$$
$$= 1,930 \text{ hours}$$

Therefore the two reactions considered together would give a combined discharge time of :-

$$T = 2.56 \times 10^6 \text{ secs}$$
$$= 712 \text{ hours}$$
$$\text{Sensitivity} = 1,780 \text{ mrem F.S.D.}$$

4.5.3. Discharge rate for thermal neutrons

It is possible in the case of sulphur to indicate thermal neutron sensitivity as there is a (n,α) reaction at this energy. From the basic equation (section 4.5) a 1.16 Mev alpha particle is produced. This travels 0.5 cms in air so on average the alpha particle will reach the end of its track within the sensitive volume of the chamber. The reaction cross section σ is 0.007 barns[59] and ϕ for 1 M.P.L. is 670 neutrons / cm^2 / sec. This gives a discharge time of :-

$$T = 5.42 \times 10^7 \text{ secs}$$
$$= 1,500 \text{ hours}$$

Thermal neutron sensitivity does appear to be much lower than that of the fast neutron sensitivity. However if account is made for the capture gamma rays emitted from the hydrogen and the aluminium case the thermal neutron sensitivity would be much higher.

It is possible to evaluate figures for ^{34}S, but as natural sulphur contains 95% of ^{32}S this was not considered worthwhile. On comparing the calculated values of sensitivity with the values obtained experimentally sulphur compared reasonably well. The values for 5 and 14 Mev being within 20%.

4.6. Pure element liner of low Z

After the idea of a compensation device was drafted it became apparent that this would require a neutron insensitive liner. From further observations of neutron cross sections it was evident that such a liner could be provided by a pure element of low atomic weight. This would have the combined benefit of low neutron cross section without possessing a high photo-electric effect dominance for the gamma ray interactions. (See fig 1.1). The two obvious choices for this liner were carbon and aluminium, both have minimal neutron cross section for the production of charged particles . The results from both these liners were as predicted, each having a low neutron to gamma ratio in the liner chamber tests and the aluminium having less than half the sensitivity of the polythene-graphite liner in the commercial dosimeter experiments.

4.7 Glossary

r = Radius of sensitive volume in a commercial dosimeter.

l = Length of sensitive volume in a commercial dosimeter.

N_a = Avogadro's Number.

M = Mass of coating or liner around the sensitive volume of a commercial dosimeter.

A = Atomic Weight of coating or liner.

N = Number of atoms in coating or liner.

t = Half life of coating.

λ = Decay Constant of coating.

e = Charge on the electron.

T = Time taken to discharge the dosimeter.

σ = Neutron cross section.

ϕ = Neutron flux.

R_z = Range of a particle in element of atomic number Z given in mgms / cm^2.

R_a = Range of a particle in air given in mgms / cm^2.

m = Mass number of the particle.

E = Initial energy of particle in Mev.

Z = Atomic number of element.

Q = Charge on the chamber.

C = Capacitance of the chamber.

V = Voltage on the chamber.

F.S.D. = Full scale deflection.

Section 5 Radiation Sources

5.1 Introduction

Four main types of radiation sources were used in the project. For convenience of identification the sources are:-

(a) Defined as neutron or gamma ray.

(b) Numbered according to the geometrical configuration in which the sources and specimen were confined.

The following section will give also a comprehensive survey of the type of source and the conditions under which it was used.

5.2 Neutron Sources

5.2.1 Americium Beryllium

This source basically converts alpha radiation to neutrons by the (α,n) reaction on the beryllium target. The main desirable properties of radioactive (α,n) sources are that:-

(a) The alpha emitter should have a high specific activity measured in curies / gm.

(b) The source should have a low gamma ray emission rate.

(c) The neutron energy spectrum should be reproducible from sources of different strength, but of the same type.

(d) The source should have a predictable variation versus time in its emission.

(e) Long half life.

(f) The alpha emitter and target material should be closely bonded, preferably as an alloy or compound, to achieve good mechanical stability.

The source of alpha particles, americium 241, has a half life of 458 years. This is now a standard source produced by the Radiochemical Centre at Amersham. The two stage reaction is:-

$$^{241}Am = {}^{237}Np + \alpha + 5.6 \text{ Mev}$$

$$^{9}Be + \alpha = {}^{13}C^{*} = {}^{12}C + n + 5.7 \text{ Mev}$$

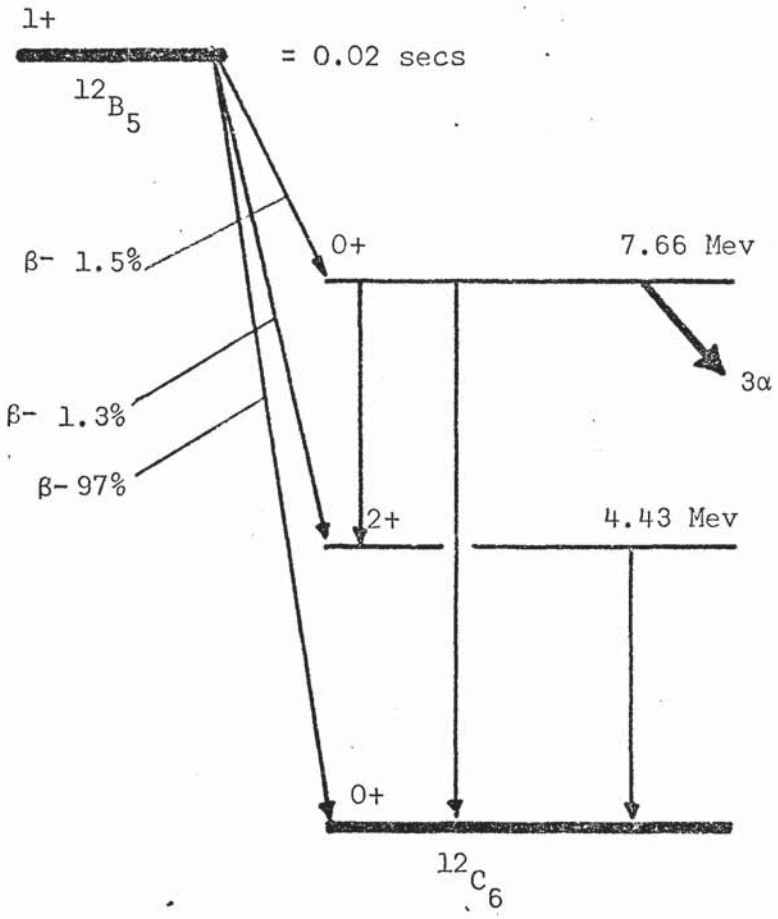
The final nucleus ^{12}C can be formed in either its ground state or the 4.43 or 7.65 Mev energy excited states. This can be seen from fig 5.1 which gives the decay scheme of $^{13}C^{*}$. (60) From the estimated limit of 9.77 Mev for the excess energy contained within the $^{13}C^{*}$ nucleus, three energy levels of ^{12}C can be reached. These are the ground state, the 4.43 and 7.65 Mev excited states. The direct transition of the 7.65 Mev level to the ground state is forbidden, as the two states have angular momentum equal to zero.(ie, no multipole radiation exists with l, the angular momentum carried by each quantum, equal to zero). Therefore it is possible to observe a 3.2 Mev + 4.43 Mev transistion. However it is found that the 7.65 Mev state decays directly by multibody breakup:-

$$^{13}C^{*} = {}^{4}He + {}^{8}Be + n - 1.67 \text{ Mev}$$

$$= 3\,{}^{4}He + n - 1.57 \text{ Mev}$$

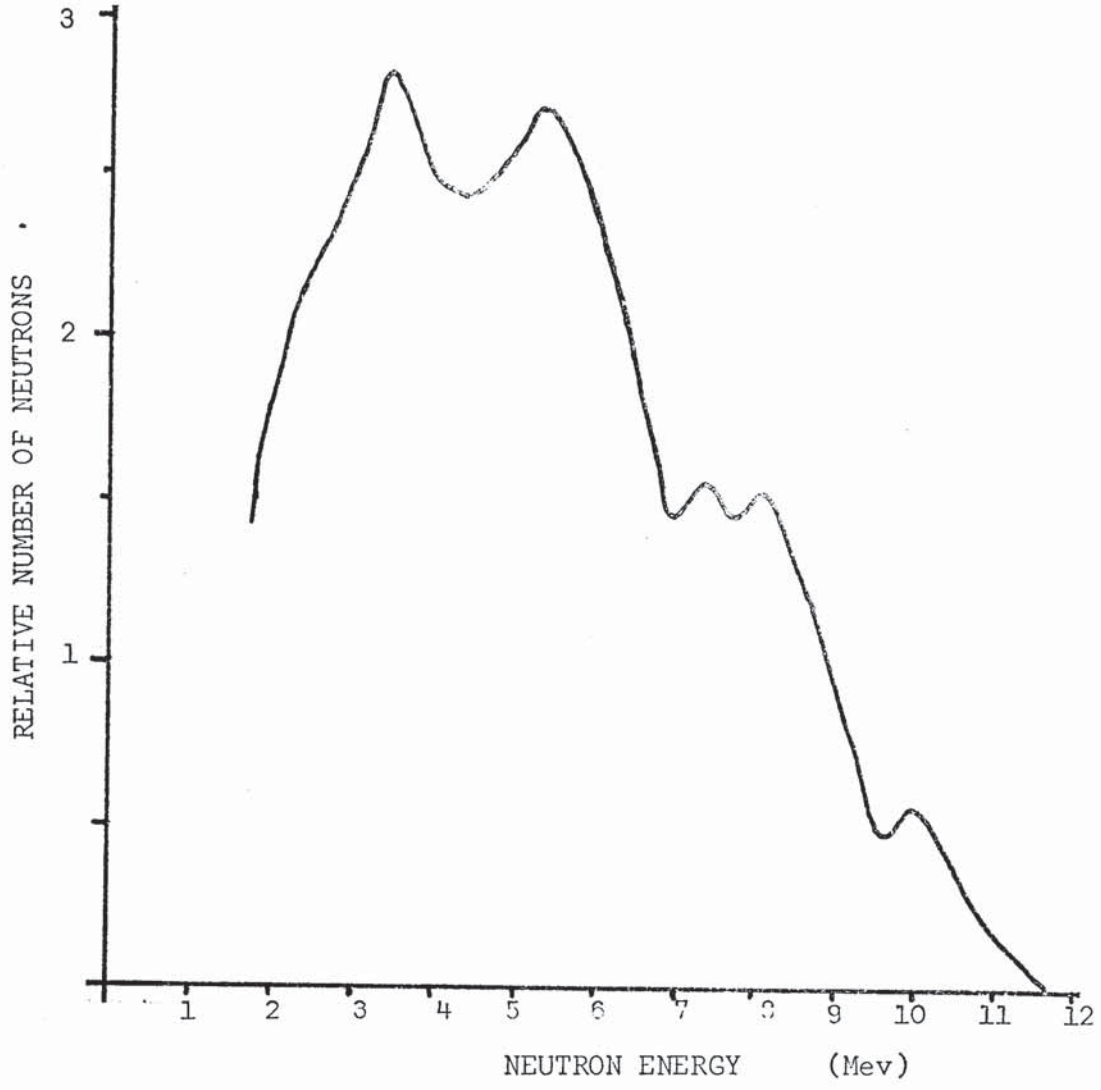
The ^{8}Be immediately breaks down into two alpha particles with the release of 94 Kev. It is noticeable with the neutron spectra from a americium beryllium source (61) (fig 5.2) that there are two peaks at 3.5 and 5.0 Mev. These two peaks probably correspond to the transition levels at which multibody breakup and 4.4 Mev level decay respectively occur. The further smaller peaks around 8.0 Mev would be directly due to the ground state transition.

Fig 5.1



Decay scheme of the $^{13}C^*$ nucleus ($^{12}B_5$ nucleus).

Fig 5.2



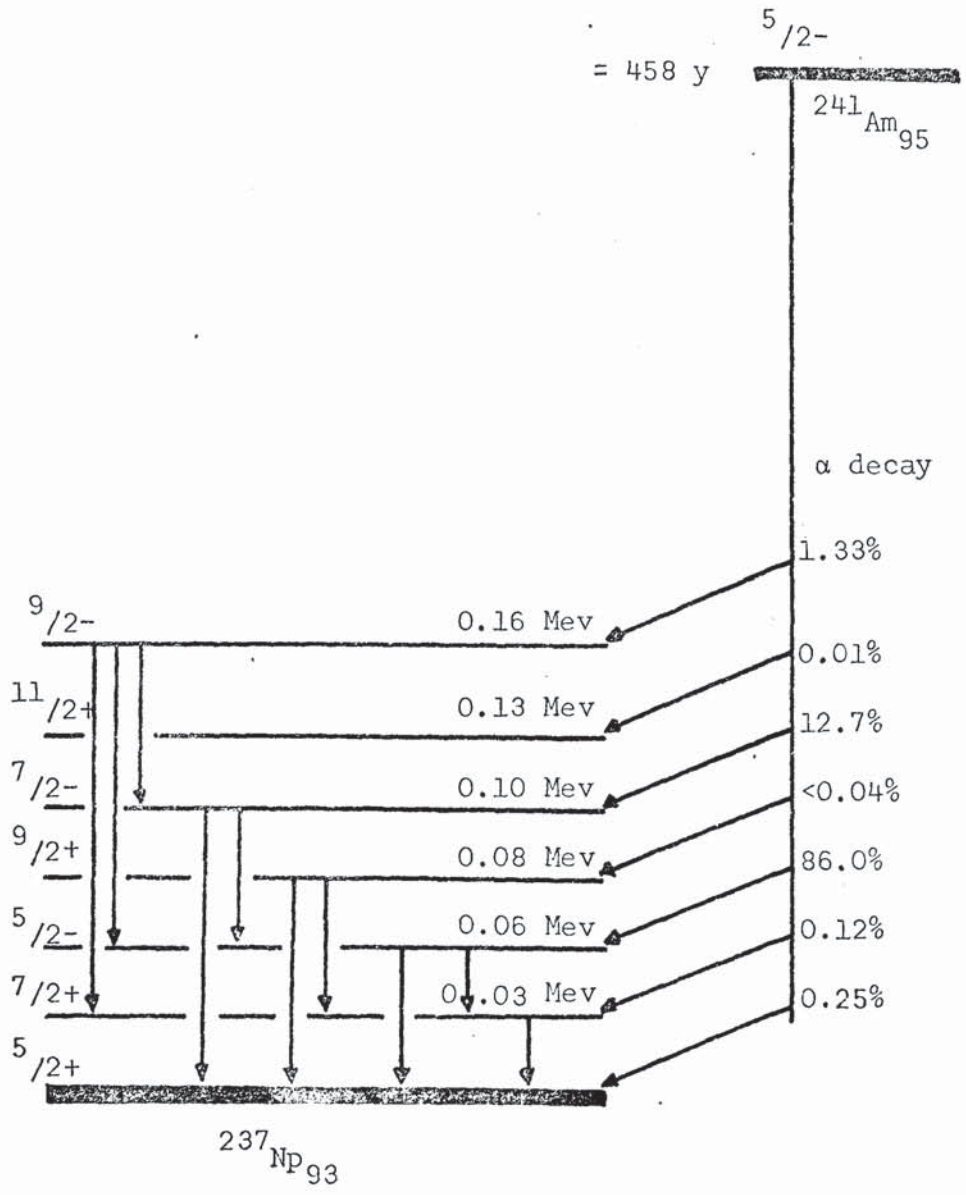
Spectrum of neutrons from Am-α-Be source.

Considering the 4.4 Mev level from the graph in fig 5.2 it can be seen that this corresponds approximately to 30% of the total neutron yield. Thus with a 300 mCi source it can be estimated that there are $7.5 \times 10^5 \times 30\%$ gamma's / sec of 4.4 Mev energy. This is equated directly to a gamma ray source of 4.4 Mev energy and the graph in fig A.4 used to evaluate the dose rate. This gives an absorbed dose rate of 3.6 mrads / hr at 5 cms. Comparing this directly with the neutron absorbed dose of 49 mrads / hr from the 300 mCi source at 5cms gives approximately 7% of high energy gamma rays to be taken into account. For dose equivalent, with a value of 345 mrem / hr for the neutron source the gamma rays represent about 1%.(Assuming a quality factor of one for gamma rays). It would therefore seem that there is a significant amount of high energy gamma radiation from an americium beryllium source if absorbed doses are being considered. However this is a topic which is rarely covered in text books on the subject.

The other gamma rays which significantly contribute to the dose rate are from the decay scheme of the americium (see fig 5.3). These gamma rays are predominately in the 40 to 60 Kev region and at a distance of 5 cms would give an absorbed dose rate of approximately 300 millirads per hour. However a lead shield 0.3 cms thick was placed round all the americium beryllium sources used in the project. This would attenuate gamma rays of this energy to an insignificant level.(62)

For the project two americium beryllium sources were used. These

Fig 5.3



Decay scheme — Americium 241

sources manufactured by the Radiochemical Centre are a compacted mixture of the oxide, with beryllium metal and are doubly encapsulated in stainless steel by argon arc welding.

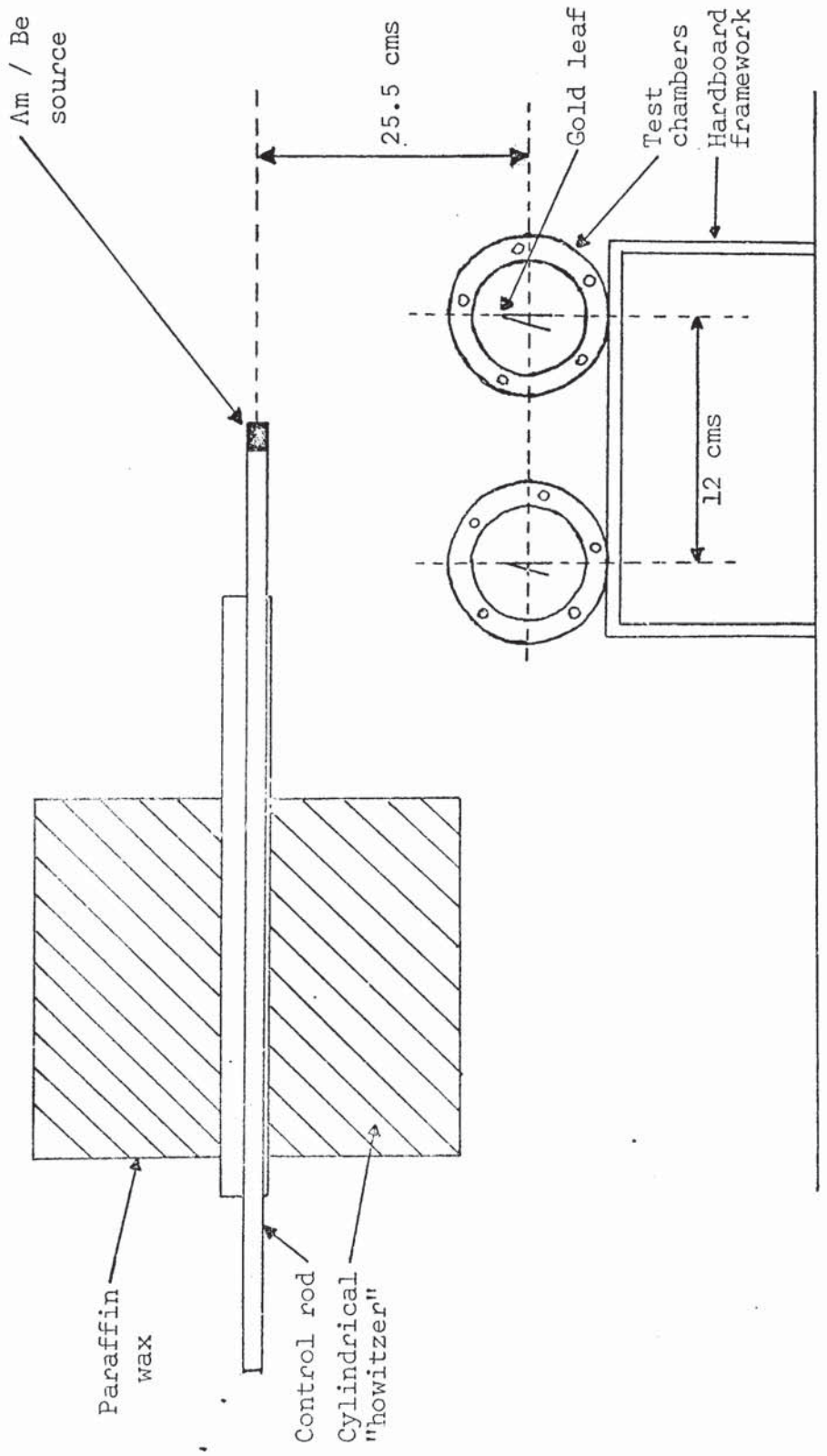
Source N 1

This is a 3 curie source, shielded with 0.3 cms of lead, at a distance of 29 cms from the chambers. The dose rate was therefore in the region of 100 mrem / hour (see appendix A). The chambers were supported on a hardboard framework 20 cms above a wooden bench, as in fig 5.4. This improved the ease of observation of the gold leaves and minimised scattering effects. The source itself was supported from a "howitzer" of paraffin wax into which the source was withdrawn while readings were taken.

Source N 2

300 millicurie source, shielded with 0.3 cms of lead. The source was held in a perspex sleeve and positioned centrally between two liner chambers (or commercial dosimeters) at a jigged source to chamber centre distance of 5 cms. Care was taken to ensure that this distance was from the central point of the sensitive volume to the centre of the source, as any error at this small distance of 5 cms would radically alter the dose rate. The jig was constructed of aluminium and was supported on a dexion framework, see fig 5.5 This gave an estimated dose level of 345 mrem / hr or 49 mrad / hr. (See appendix A)

Fig 5.4



Schematic representation of the apparatus arrangement for source N.1.

Fig 5.5

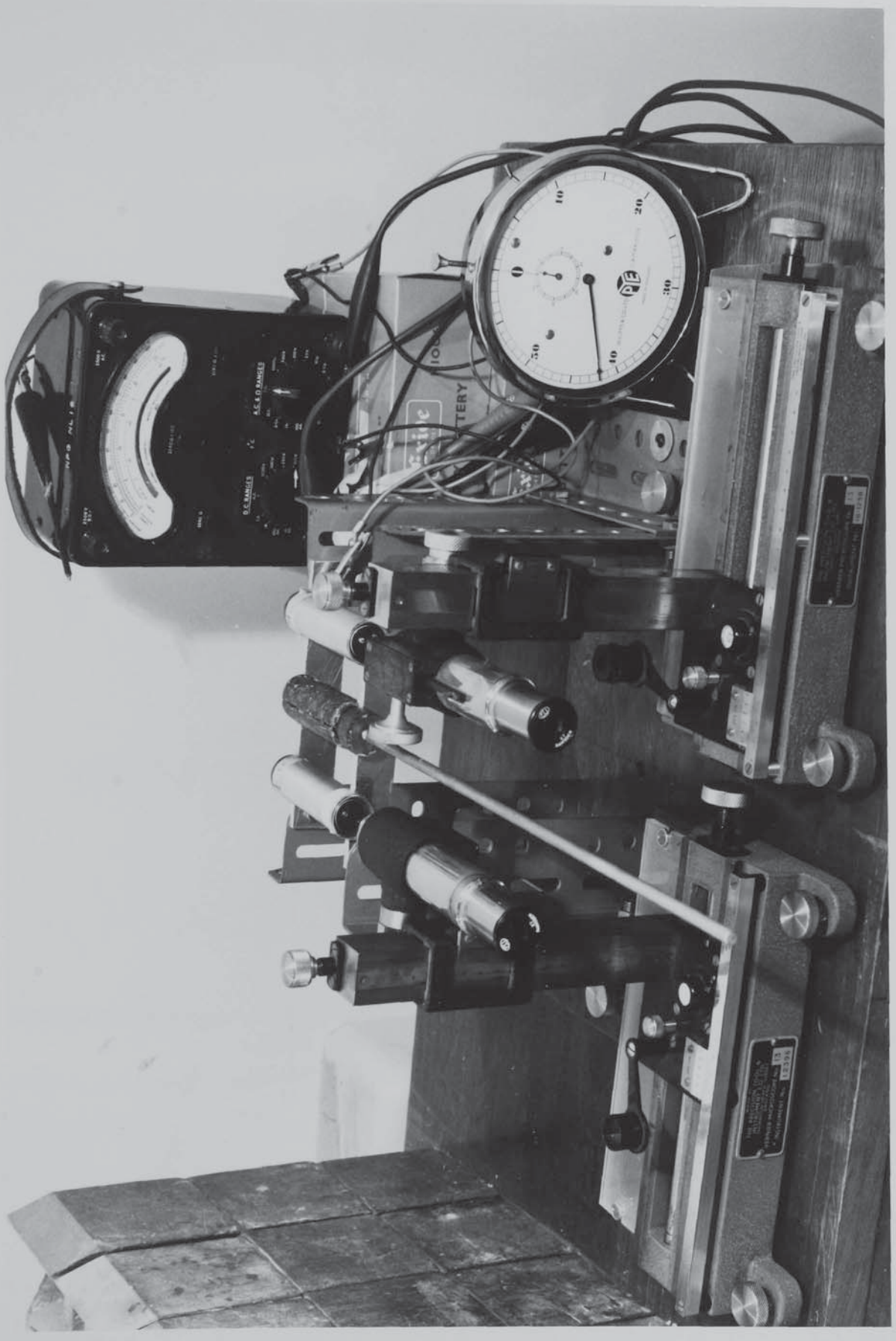
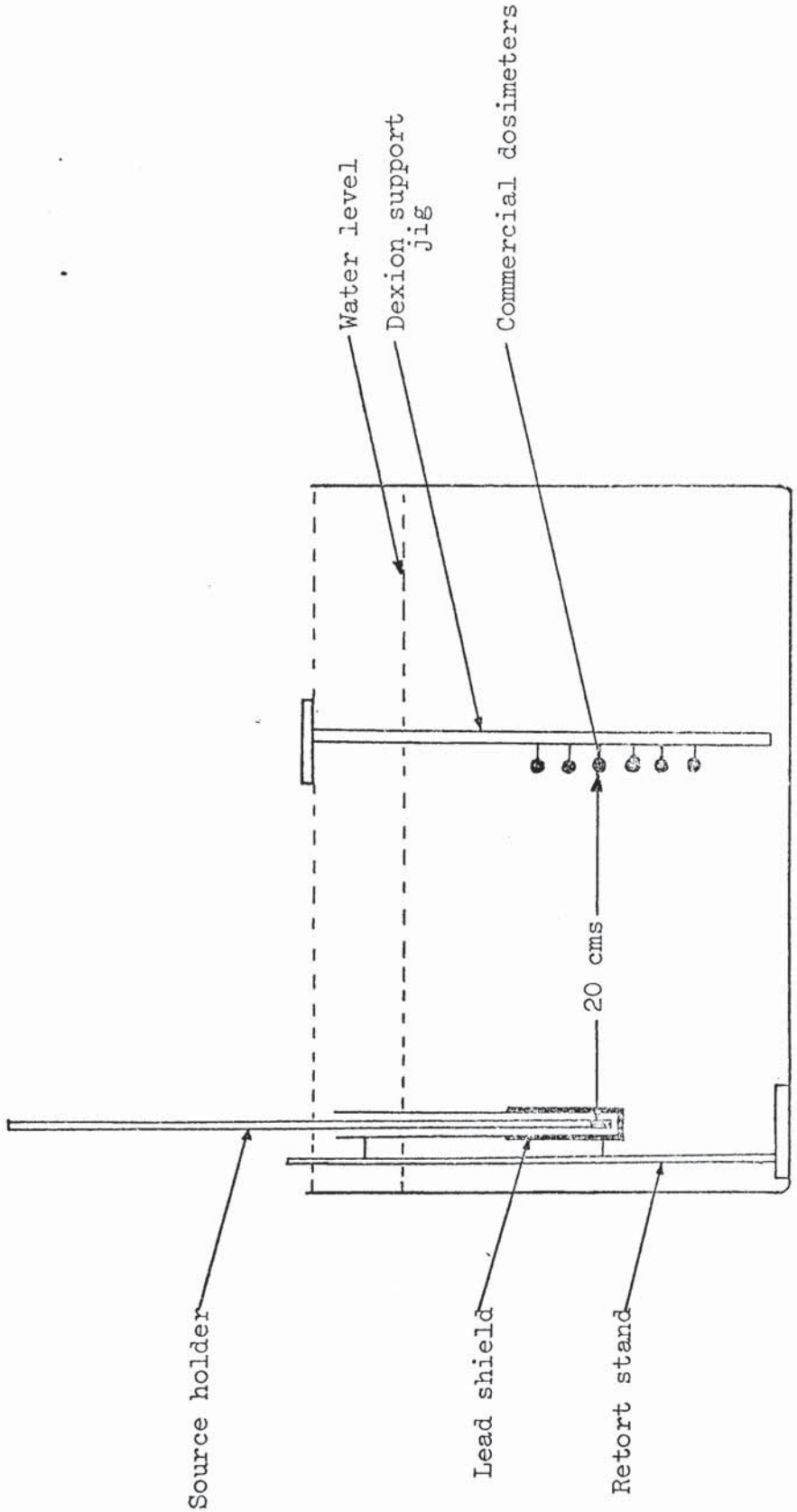


Fig 5.6



Schematic representation of the apparatus arrangement for source N.3.

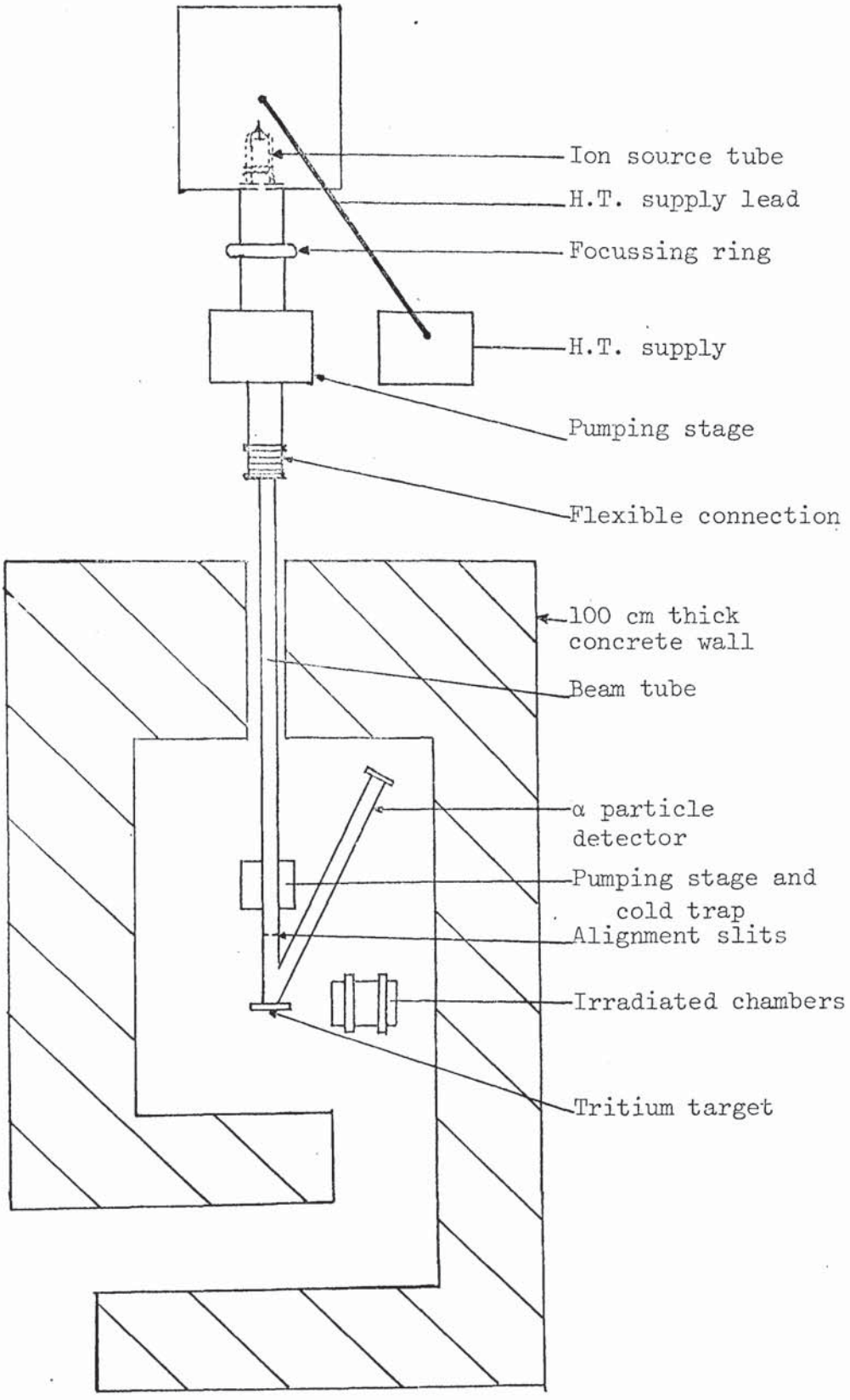
Source N 3

3 Curie source shielded with 0.3 cms of lead. The source was placed at a distance of 10 to 30 cms from a rack containing six commercial type dosimeters. (See fig 5.6.) The rack was a dexion framework which could easily be moved with respect to the source. Dose equivalent levels between 160 and 250 mrem / hr were used. On calculation of the dose rates appropriate geometrical modifying factors were given to those dosimeters at the bottom and the top of the rack as were a greater distance from the source. Further investigations were made with arrangement given above for the production of thermal neutrons. The rack and source were placed inside a large container filled with water where it was hoped to produce thermal neutrons by moderation. However it was found that the level of fast neutron flux contamination and the 2.25 Mev capture gamma rays from the $^{1}H\ (n, \gamma)\ ^{2}H$ reaction made it difficult to obtain a thermal neutron continuum of reasonable purity.

5.2.2. 14 Mev neutrons

This source of neutrons was provided by a "J type" S.A.M.E.S. linear accelerator (see fig 5.7.) This was used to accelerate deuterons onto a tritium target and produce neutrons by the D.T. reaction. All the irradiations were made at 90° to the incident deuteron beam direction. At this angle the 14.1 Mev neutrons produced are virtually unaffected by the deuteron energy.

Fig 5.7



Plan view of S.A.M.E.S. accelerator connected to the "Igloo" facility

The reaction is:-

$$^{3}H + ^{2}H = ^{4}He + n + 17.6 \text{ Mev}$$

ie :- $$T + D = \alpha + n + 17.6 \text{ Mev}$$

The extremely high Q value of this reaction makes it possible to produce 14 Mev neutrons from accelerating potentials of less than 100 k V. The energy dependence is given in fig 5.8, where it can be seen that the highest cross section occurs at 110 Kev. The deuterons are provided from deuterium gas which is leaked into the radio frequency ion source and then accelerated. The target is tritium absorbed in a thin layer of titanium of approximately 1.5 mg / sq in thickness. The titanium is backed by 0.01 inches of copper. The tritium content of the targets varied but was always in the region of < 1 curie / sq in. A target current of approximately 100 μ A was used for the higher flux levels of 1.5 rem / hour. The target was air cooled to a temperature of about 150°c. The neutron flux was measured by the associated alpha particle technique. This basically makes use of the fact that one alpha particle is produced for every neutron. The consequent neutron flux can be calculated at any angle from the target by a knowledge of the alpha - particle detector geometry, its observed counting rate and the calculated angular distribution of neutrons and alpha particles in the reaction. The detector was placed at 150°. (see fig 5.9.) and the detector distance from the target was one metre. The actual flux levels used for 14 Mev irradiations was subject to

Fig 5.8

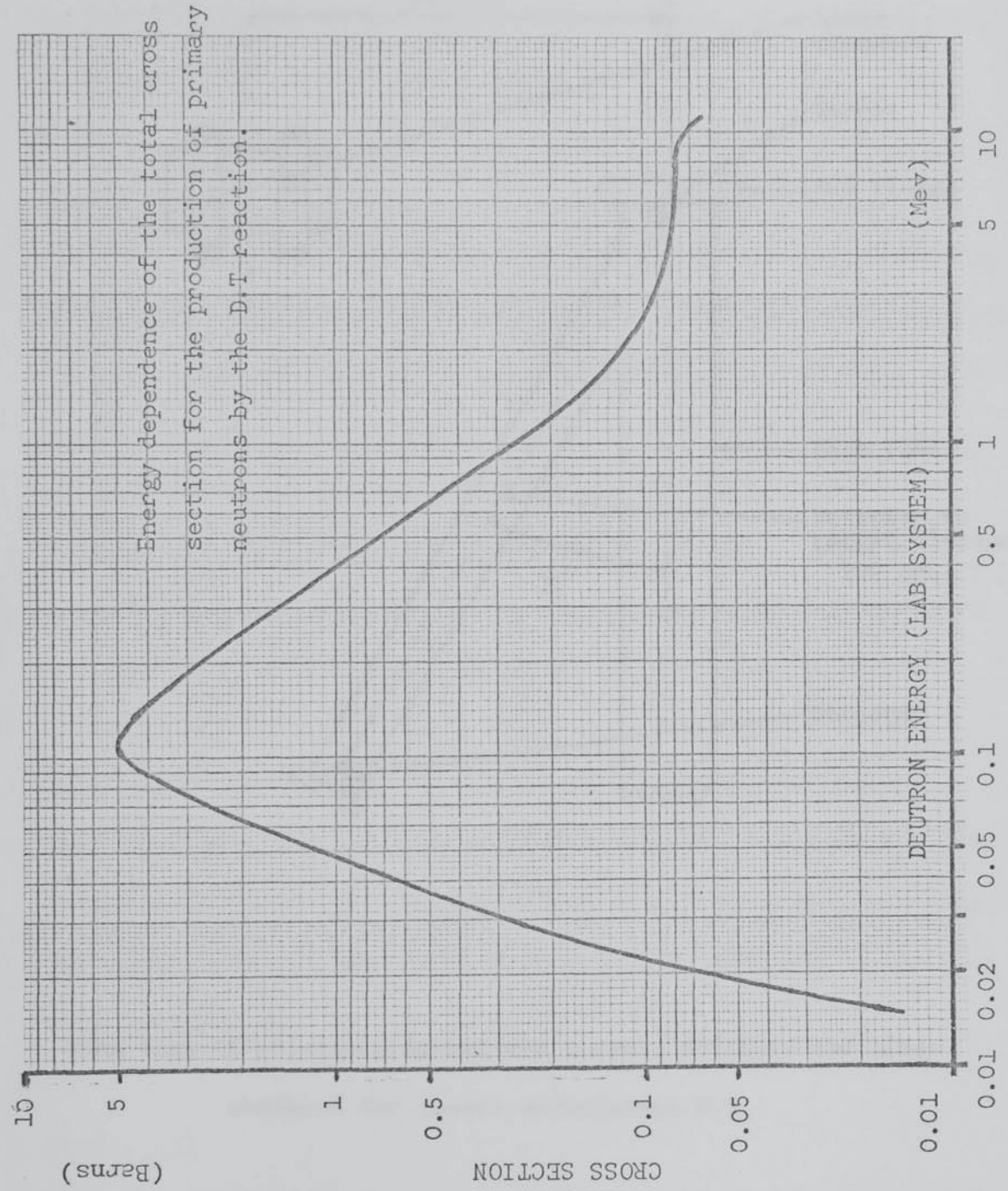
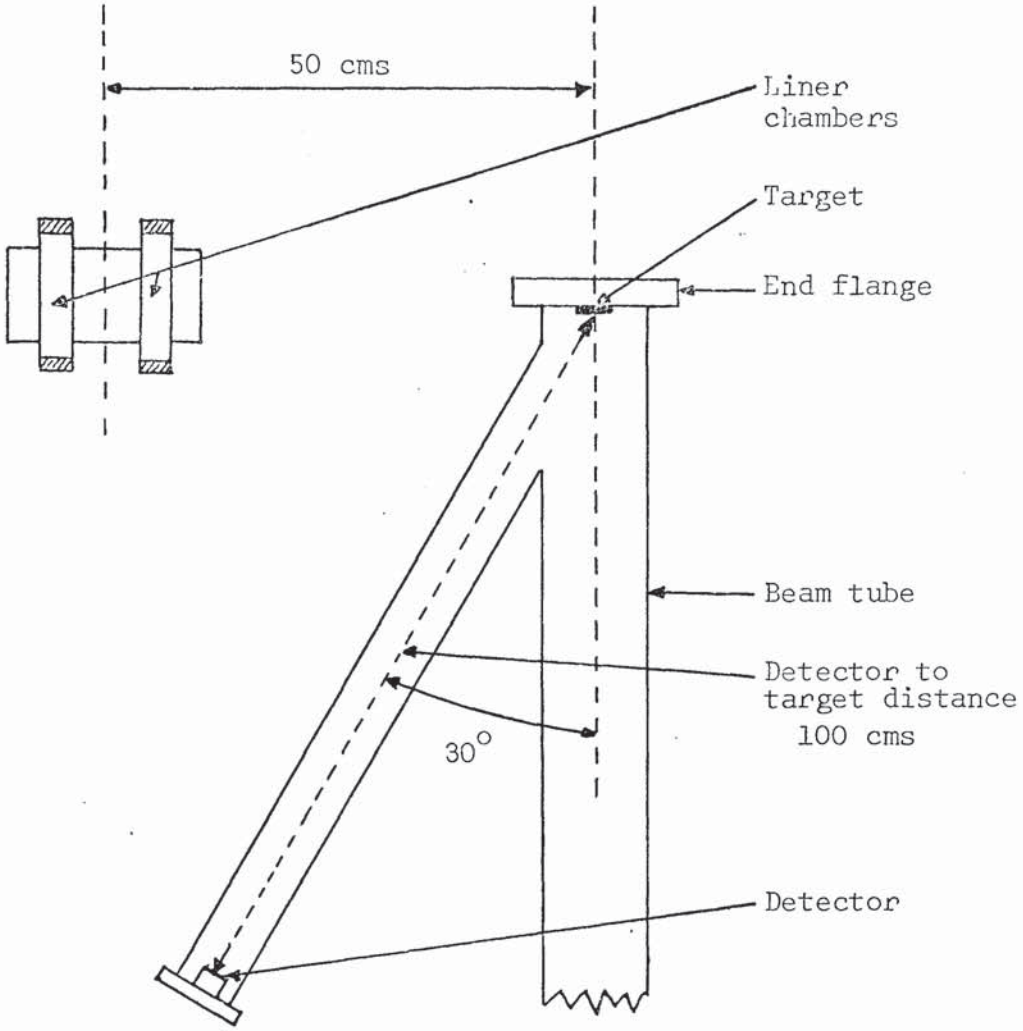

Energy dependence of the total cross section for the production of primary neutrons by the D.T reaction.
CROSS SECTION
(Barns)
DEUTRON ENERGY (LAB SYSTEM)
(Mev)
10
5
1
0.5
0.1
0.05
0.01
0.01
0.02
0.05
0.1
0.5
1
5
10

Fig 5.9



Plan view of relationship between target, detector and liner chambers for source arrangement N.5.

much variation. This was mainly due to it being impractical to run the accelerator at the preferred lower dose levels, because this would require 4 - 5 hours for one reading to be taken. With the liner chambers however, this was possible to achieve, as the time for each irradiation was two minutes for a reasonable gold leaf deflection to be observed.

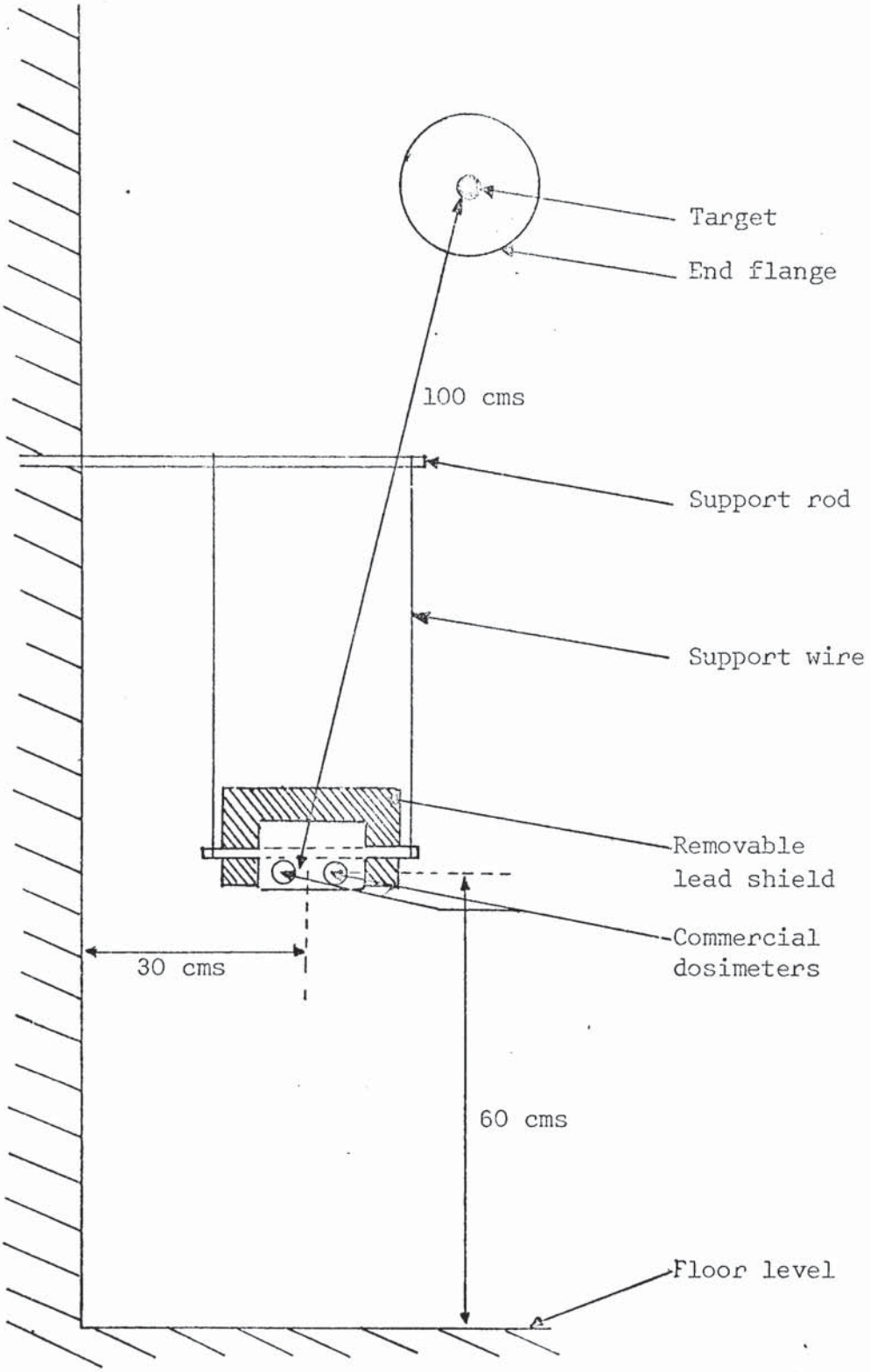
Source N 4

Was used solely with the commercial dosimeters. A wire framework constructed with a removable shield of lead 2.5 cms thick was used. With the flux levels produced at this radiation level there is an appreciable amount of gamma radiation from the brass end flange of the beam tube. (In the region of 0.5 Mev energy) The 2.5 cm thick lead shield attenuated these gamma rays by a factor of 2×10^2 (62). However the lead also attenuated the neutron flux by approximately 12%. The framework was suspended to reduce backscattering from the walls and floor. This positioned the dosimeters 1 metre from the target and 60 cms off the floor. (see fig 5.10) Two sets of results were taken, one with the lead shielding covering the dosimeters and the other without the lead shielding. Flux levels of up to 1 rem / hr. were used (see appendix A).

Source N 5

This was also a D.T. source using neutrons emitted at 90° to the beam direction but here the target to chamber distance was 56 cms This arrangement was used for both the commercial dosimeters and

Fig 5.10



Schematic representation of the apparatus arrangement for source N.4.

the liner chambers. The same position jig as in source N 2 was used. Thus two liner chambers or dosimeters could be irradiated at any one time, with due account being made for the difference in distance from the target. (see fig 5.11). The flux level was kept at 345 mrem / hr. (to be the same as the americium beryllium source) and the irradiations were for two minutes. At this low flux level it was found that there was no appreciable activation of the end flange, therefore lead shielding was not deemed necessary .

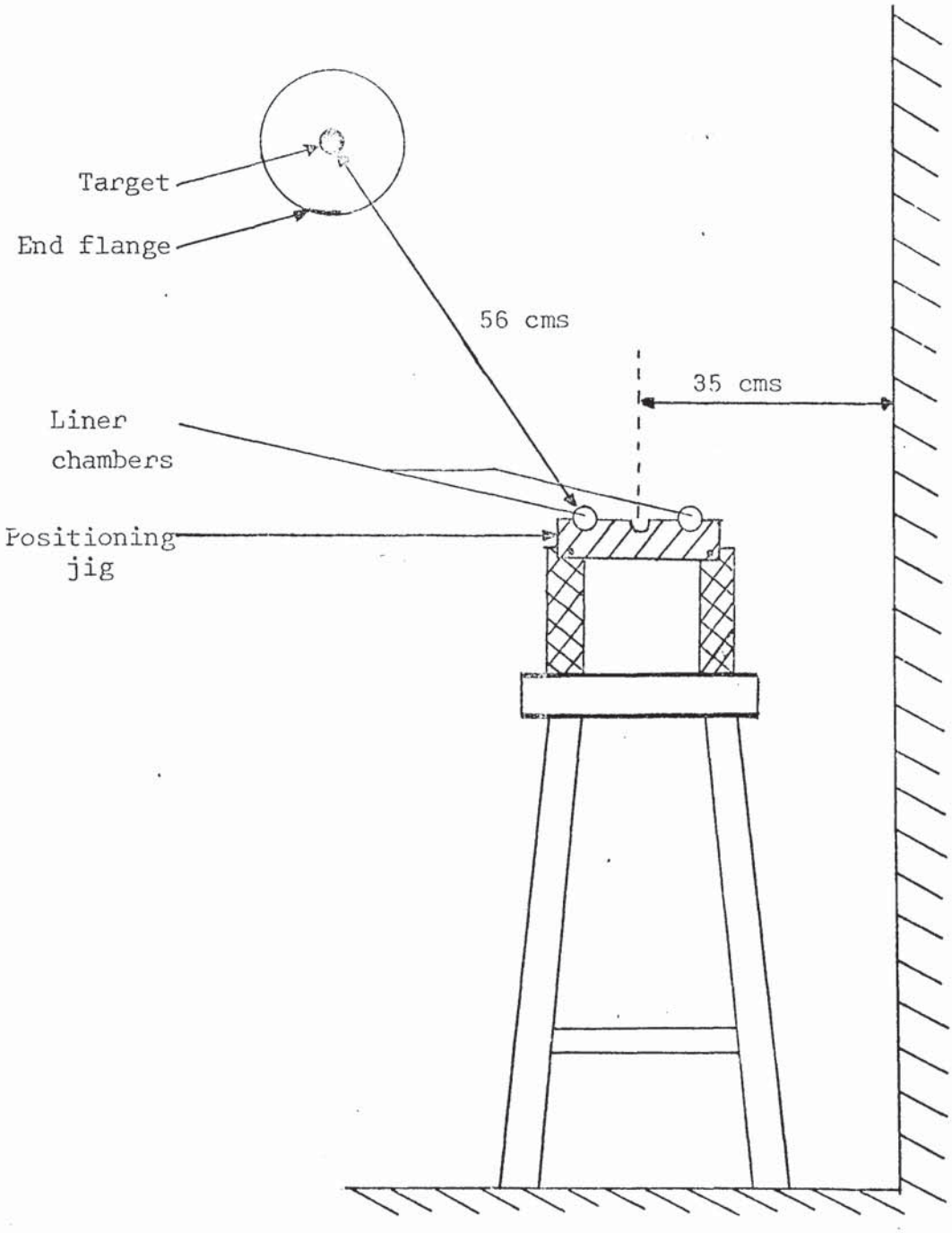
5.2.3. Thermal Neutrons

These were obtained from the Experimental D Hole in G.L.E.E.P. reactor at A.E.R.E. Harwell. This hole was chosen as it was in the reflector parallel to the fuel rods and thus gave a low gamma ray flux. The hole was 7 inches in diameter and the dosimeters were held in a clamp at the end of an aluminium pole. The pole was mounted on a railway so that it could be moved into the hole to achieve the required flux level. See fig 5.12. The fast neutron flux is less than 1% of the thermal neutrons in D hole [63].

Source T. N. 1.

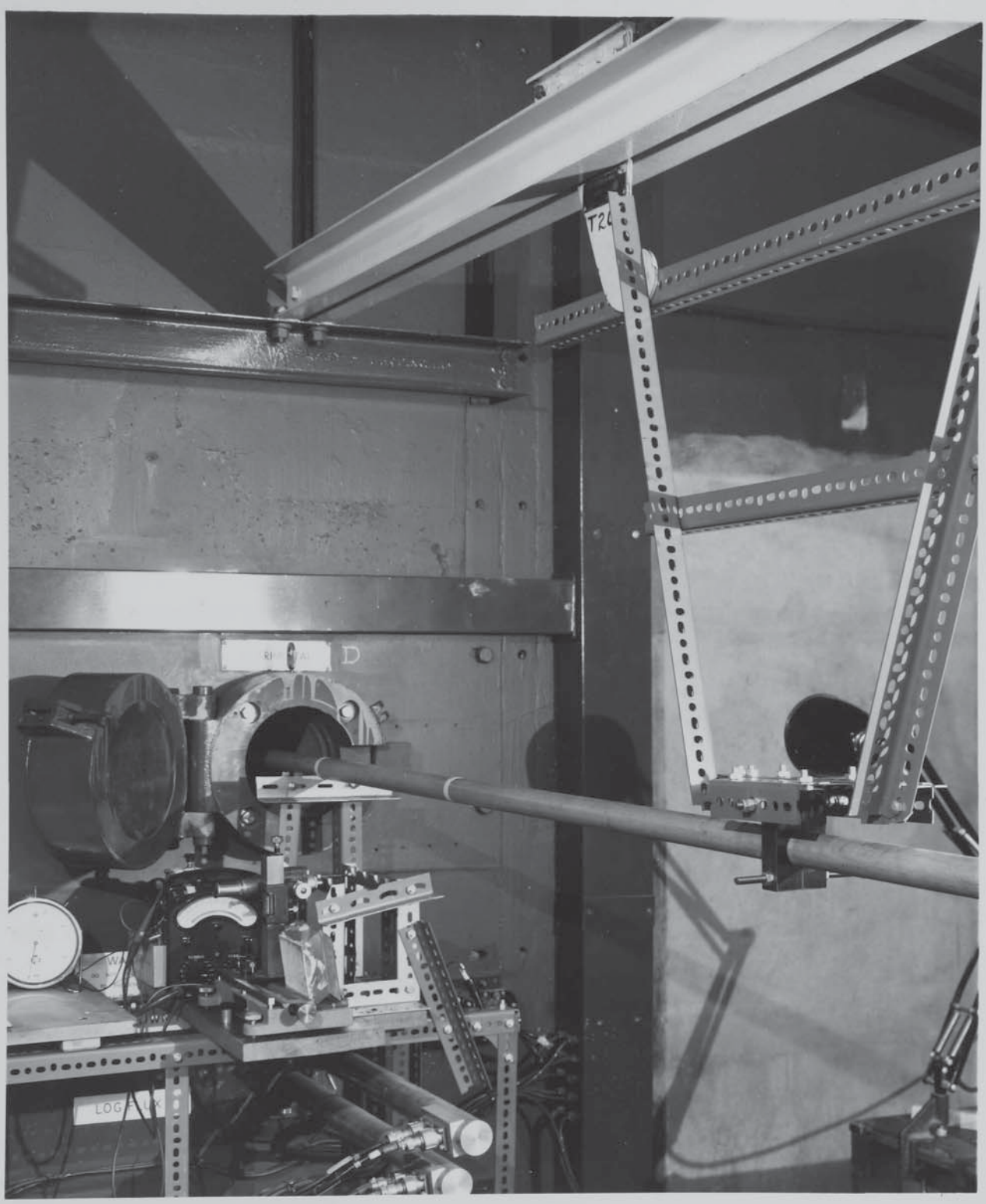
Was a distance of 1 ft 8 inches down the hole corresponding to a flux level of 7×10^4 neutrons cm^{-2} sec^{-1}, This gave a dose rate of 294 mrem / hour. There is an estimated 10 mrem / hour of gamma rays at this point. Irradiations were for the commercial dosimeter in batches of three for 15 or 20 minutes.(see fig 5.12)

Fig 5.11



Schematic representation of the apparatus arrangement for source N.5.

Fig 5.12



Source T. N. 2.

Was at the surface of the hole. This gave a flux level of 10^4 neutrons / cm^{-2} / sec^{-1} and gave a dose rate of 42 mrem / hr. This was checked on a Slow Neutron Survey Meter Type 1399A[20]. The gamma ray contribution was estimated to be 3 mrem / hr[63] The liner chambers were irradiated in these conditions. Each chamber was clamped at the end of the aluminium pole, (see fig 6.9) irradiated for two minutes and then lowered out of the flux into a fixed position for a reading to be taken. The jig for the fixed position was a slightly modified version of that used previously in N. 2 and N. 5. Thus it was possible to use this jig for normalising with a gamma ray source as previously done in N. 2 and N. 5 using the source fixed position for the thermal neutrons and gamma irradiations.

5.3. Gamma Ray Sources.

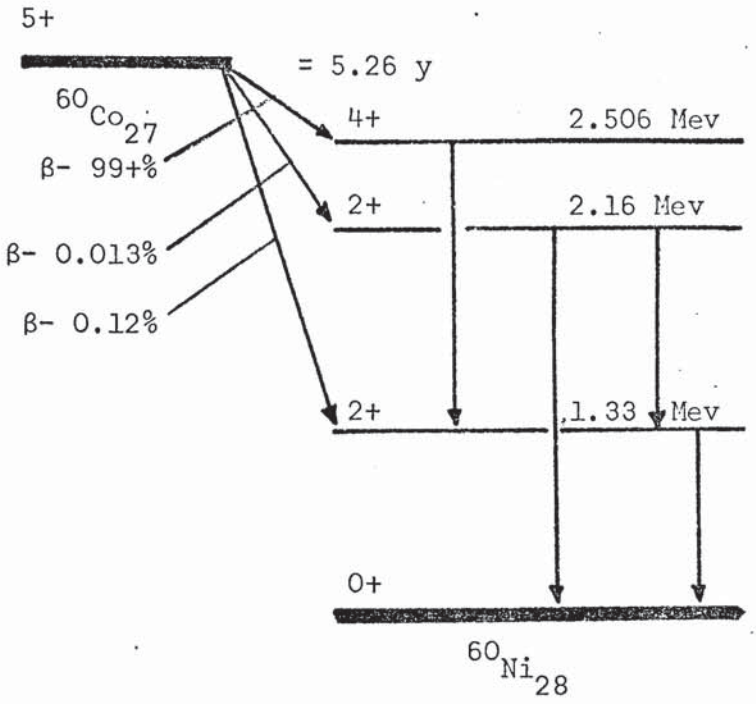
Two sources were used for the production of gamma rays cobalt 60 and caesium 137. Cobalt 60 is produced by the neutron bombardment of natural ^{59}Co. and ^{137}Cs is a fission by - product which is obtained by processing spent uranium fuel rods. Both of these sources are produced by the Radiochemical Centre at Amersham.

5.3.1. Cobalt 60 source.

This isotope has a half life of 5.26 years and emits beta and gamma rays in its decay to the stable nickel 60. (see fig 5.13)

$$^{60}Co = {}^{60}Ni^{*} + \beta = {}^{60}Ni + 2\gamma + 2.7 \text{ Mev}$$

Fig 5.13



Decay scheme of Cobalt 60

The rays have energies of 1.17 and 1.33 Mev. This source was a metal cylinder of cobalt encapsulated in welded stainless steel.

Source G.1.

Half millicurie cobalt 60 source used in the same configuration as source N 1. Dose rate approximately 160 mrem / hr.

Source G 2.

50 microcurie cobalt 60 source used in the same configuration and jigged position as source N 2. The source was contained in a perspex sleeve such that its position would be central to the sensitive volume of the irradiated chambers. Dose rate was approximately 20 mrem / hour.

Source G 3.

Half millicurie cobalt 60 source used in the same configuration and jigged position as source N 3. The source to chamber distance was again varied from 10 to 30 cms giving a dose rate varying between approximately 120 to 210 mrem / hour.

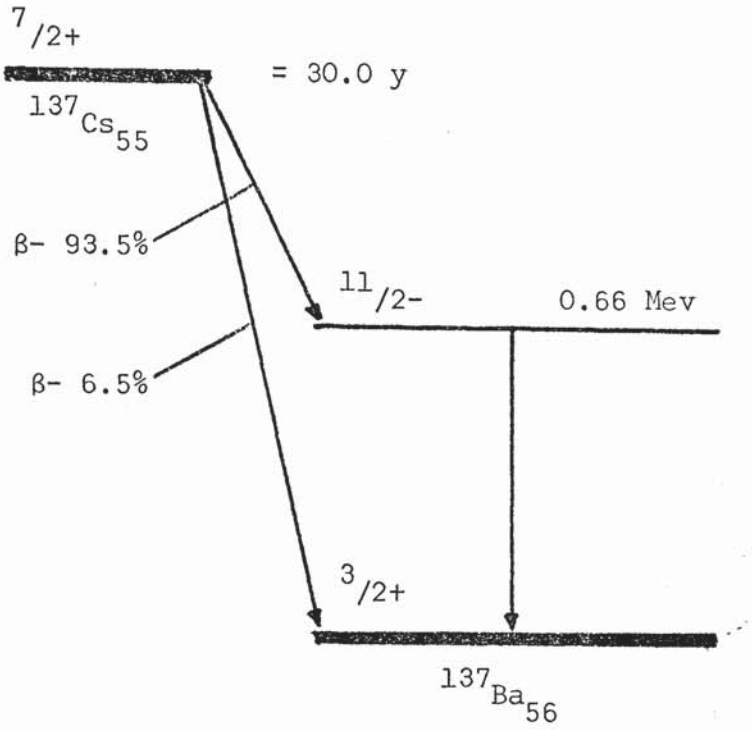
5.3.2. Caesium 137

This isotope has a half life of 30 years and emits beta and one gamma ray in its decay to the stable Barium 137. (see fig 5.14)

$$^{137}Cs = {}^{137}Ba + \beta + \gamma + 0.66 \text{ Mev}$$

This source was a bead of caesium glass encapsulated in welded stainless steel.

Fig 5.14



Decay scheme of Caesium 137

Source G 4

95 microcurie Caesium 137 source used in the same configuration as source G 3. (This was intended to investigate any difference in response to various energy gamma rays.) Dose rate approximately 9 mrem / hour.

Section 6 Liner Investigations and Testing

6.1 Introduction

From initial tests taken with the commercial dosimeters as described in section 7 it was decided that sulphur was unsuitable as a liner for the reasons stated. Fulfilling most requirements for a neutron sensitive and insensitive liner (for the compensation principle given in section 8) would be a plastic or some pure element of low Z, (other than sulphur). From consideration of molecular composition and neutron cross-sections it was seen that a plastic with a high neutron sensitivity was polythene, and that with a low neutron sensitivity was P.T.F.E. However as neutron cross-sections are not always reliable and geometrical effects and liner thickness calculations uncertain, it was decided to investigate both these plastics with five others having a hydrogen content intermediate to polythene and P.T.F.E. The pure elements carbon and aluminium were also included. For convenience of identification in the text the ionisation chambers used for liner investigation are referred to as "liner chamber". The tests were divided into four stages :-

(a) 5.1 Mev average energy neutrons from source N 2.

(b) 14 Mev neutrons from source N 5.

(c) Thermal Neutrons from source T.N.2.

(g) Gamma rays from source G 3. This source was used for normalisation of (a), (b) and (c). The term normalisation will be explained later.

6.2 Construction of Liner Chambers.

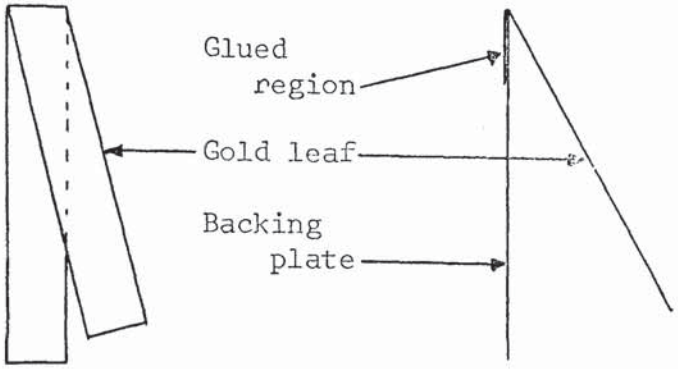
The chambers were of aluminium with a central brass electrode supporting a gold leaf. The brass electrode was held in position by a perspex insulator (see fig 6.7.).

6.2.1. Gold Leaf.

This was used to give an indication of the charge in the chamber. Their movement was observed through the perspex end plug and tracked by a travelling microscope. These leaves were specially selected for uniformity of movement and linear voltage response over a selected region of their swing. To find out if a leaf was suitable the following procedure was adopted. A varying voltage was applied to each gold leaf in turn, and its movement from fully charged to discharged, observed through the travelling microscope. If the leaf had kinks along its length or it was incorrectly glued to the support irregular, jerky movements would be observed during the travel of the leaf as the voltage was varied. This was undesirable because instead of several readings being taken with one charging of the liner chamber, only two or three could be taken if the leaf had only a small region of linear response. A diagram of the assembly and glueing of a typical gold leaf is given in fig 6.1. Only after much practice was it possible to construct a gold leaf of satisfactory size. The smallest leaf constructed was for the compensation device described in section 8.

Fig 6.1

(a)



(b)

Gold leaf assembly. Showing in (a) a leaf correctly glued and (b) a leaf which has been overglued so that it is partially stuck onto the front face of the backing plate.

6.2.2 Perspex Insulator

The design factors involved here proved to be much more difficult than first anticipated. The requirement was for a high resistance insulator of good mechanical strength, sufficient to rigidly support a gold leaf and allow an "O" ring fitting. It would have to be of sufficient transparency to permit the passage of light for illumination of the gold leaf and allowing easy observation of the gold leaf. Therefore an insulator was constructed as in fig 6.2. This had a groove for a size OS 14 "O" ring .
Initial tests showed the results to be inconsistent. The liner chamber showing a decrease in sensitivity with irradiation time. The cause of this effect was eventually traced to a phenomenom commonly termed as "Soakage".(64) If an ionisation chamber has been charged for some time and is then momentarily short circuited, it will recover some charge which is "supplied" by the perspex insulator. This recovered or soaked charge is mainly dependent on the volume of the dielectric. Therefore the insulator was redesigned as in fig 6.3, thus lowering the volume without undue reduction of the surface area or the mechanical strength. This also reduced the length of the electrode held in the perspex, thus reducing the overall capacitance of the system and improving sensitivity. The two other factors considered were ionisation of "free" air and electrical leakage. From the diagram it can be seen that the end of the electrode protrudes beyond the perspex support. When this occurs some charge is lost to the "free" air ions. This can be reduced by minimising the surface area of the end of the electrode by recessing into the perspex as in fig 6.4.

Figs 6.2, 6.3 and 6.4

Fig 6.2

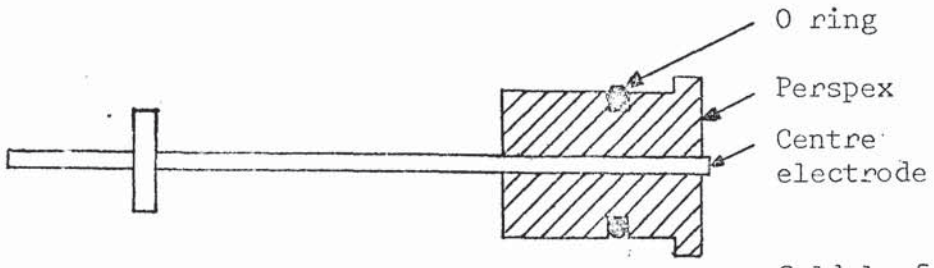


Fig 6.3

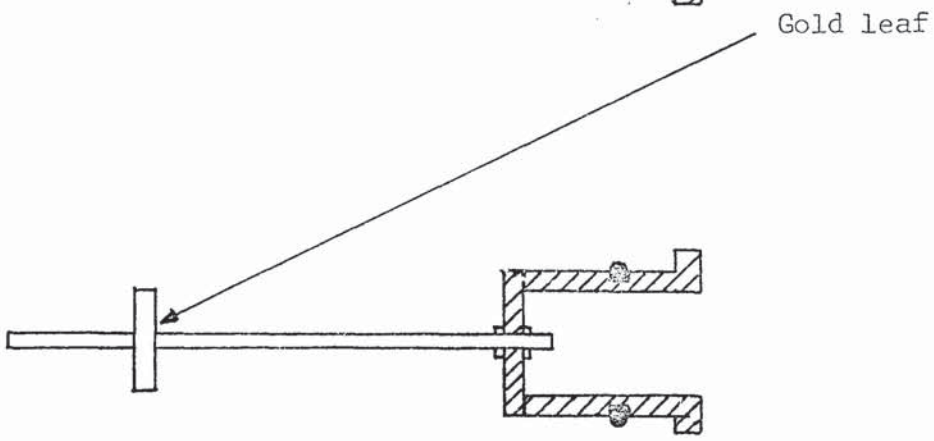
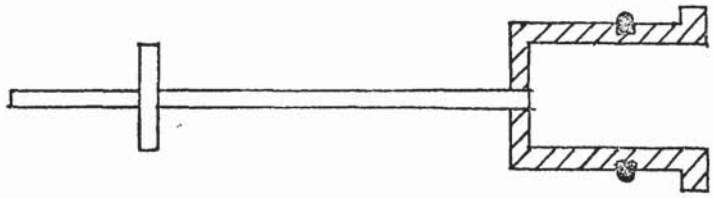


Fig 6.4



Liner Chamber insulator. Showing the original design in Fig 6.2, modified version in Fig 6.3 and the ultimate version in Fig 6.4.

Electrical leakage was minimised by polishing the insulating surface and keeping it throughly clean. Improvement was also noted when the insulating surface was recessed into the aluminium tube. This had the two-fold advantage of keeping the surface dust free and preventing circulating air currents contributing to the "free air" ionisation.

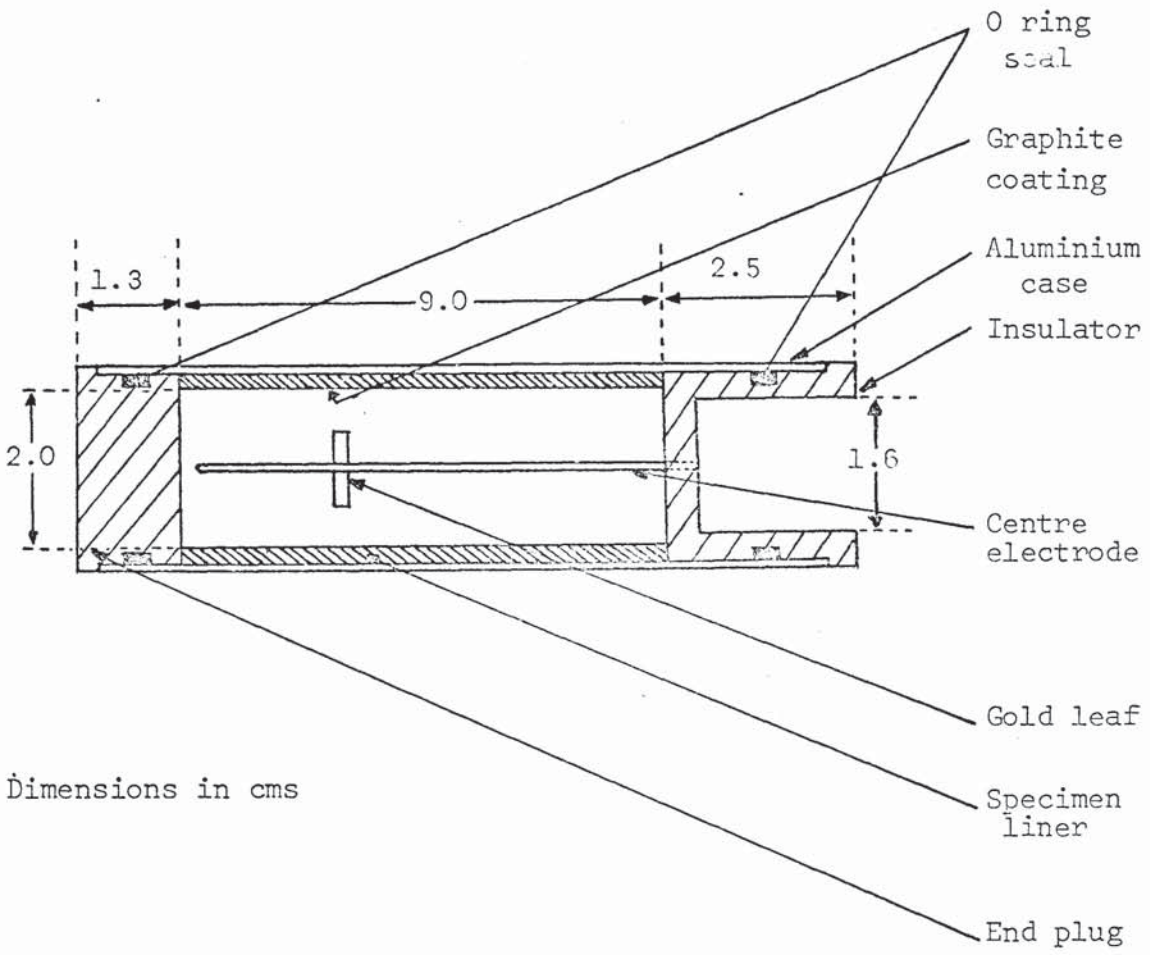
6.2.3 Perspex plug

This also had a OS 14 size "O" ring fitting and was required to be optically flat to allow observation of the gold leaf over its region of travel without undue distortion. Dimensions as in fig 6.5 The aluminium case had a small hole drilled through it just below the end of the perspex plug. This removed any pressure build up from pressing in the perspex plug in the sensitive volume. After the plug had been positioned it was sealed.

6.3 The Liner

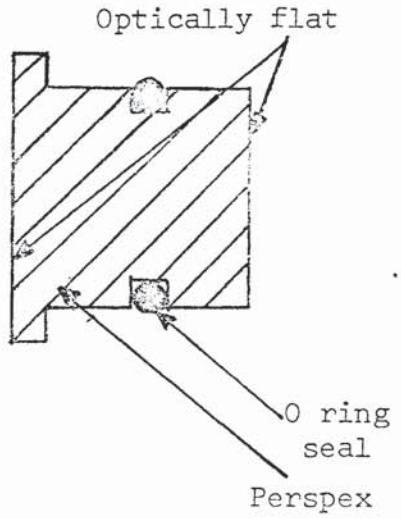
From basic considerations of thickness requirements as given in section 3 a liner thickness of 0.10 cms (0.040 inches) was chosen. However because of the low density of polythene a second liner of this material of thickness 0.15 cms (0.060 inches) was also included. The plastic liners except polystyrene were cut from flat sheet and bent, with the aid of heating, into the required cylindrical form. The polystyrene liner was machined from the solid as its brittle nature prevented bending to this degree. The aluminium liner was cut from flat sheet and bent to cylindrical form while the carbon liner was machined from reactor grade graphite.

Fig 6.7



Assembled liner chamber

Fig 6.5



End plug

Figs 6.5 and 6.7

6.4 The Liner coating

Each of the plastic liners was given a graphite coating. This was achieved by painting on a aquadag solution, allowing it to dry, and then rubbing it firmly into the surface. The surface was then repainted, dried, and finally polished. The graphite thickness was gauged by taking resistance measurements from the centre of the liner to the outside edge. (see fig 6.7) A coating was considered satisfactory if this resistance was in the order of 10 ohms which corresponded to a thickness of approximately 0.003 cms (0.001 inches). The aquadag was a suspension of colloidal graphite in water made by the Acheson colloids Company of Plymouth. It is necessary to achieve a reasonably high graphite purity as any impurities will affect the gamma ray sensitivity. The photoelectric effect is the cause being approximately proportional to $Z^{4.5}$ (2) so the chambers will be particularly sensitive to high Z materials. This was clearly demonstrated in section 8 where the polythene chamber was first coated with Molykote. This is a molybdenum disulphide solution. The gamma ray sensitivity decreased by a factor of six when the graphite coated was substituted for the molybdenum. The thickness of the coating will also influence the sensitivity. It has been estimated that with 250 kv X-rays more than 90% of the total ionisation arises from photon absorption in a wall thickness of 0.003 cms (0.001 inches) thick.(79) This may to some extent account for the generally uniform sensitivity of all the liner chambers to gamma radiation. Aluminium is the exception being noticeably higher (from the results in section 6). Aluminium being electrically

conducting was not given a graphite coating and having a higher Z number than graphite may be the reason for the higher gamma ray sensitivity (reasons given in section 6.8). It was considered using aluminium as a conductive coating for the plastic. This would be achieved by vacuum coating[80]. However because of the inconvenience and uncertainty in actually improving results by this method, it was not attempted.

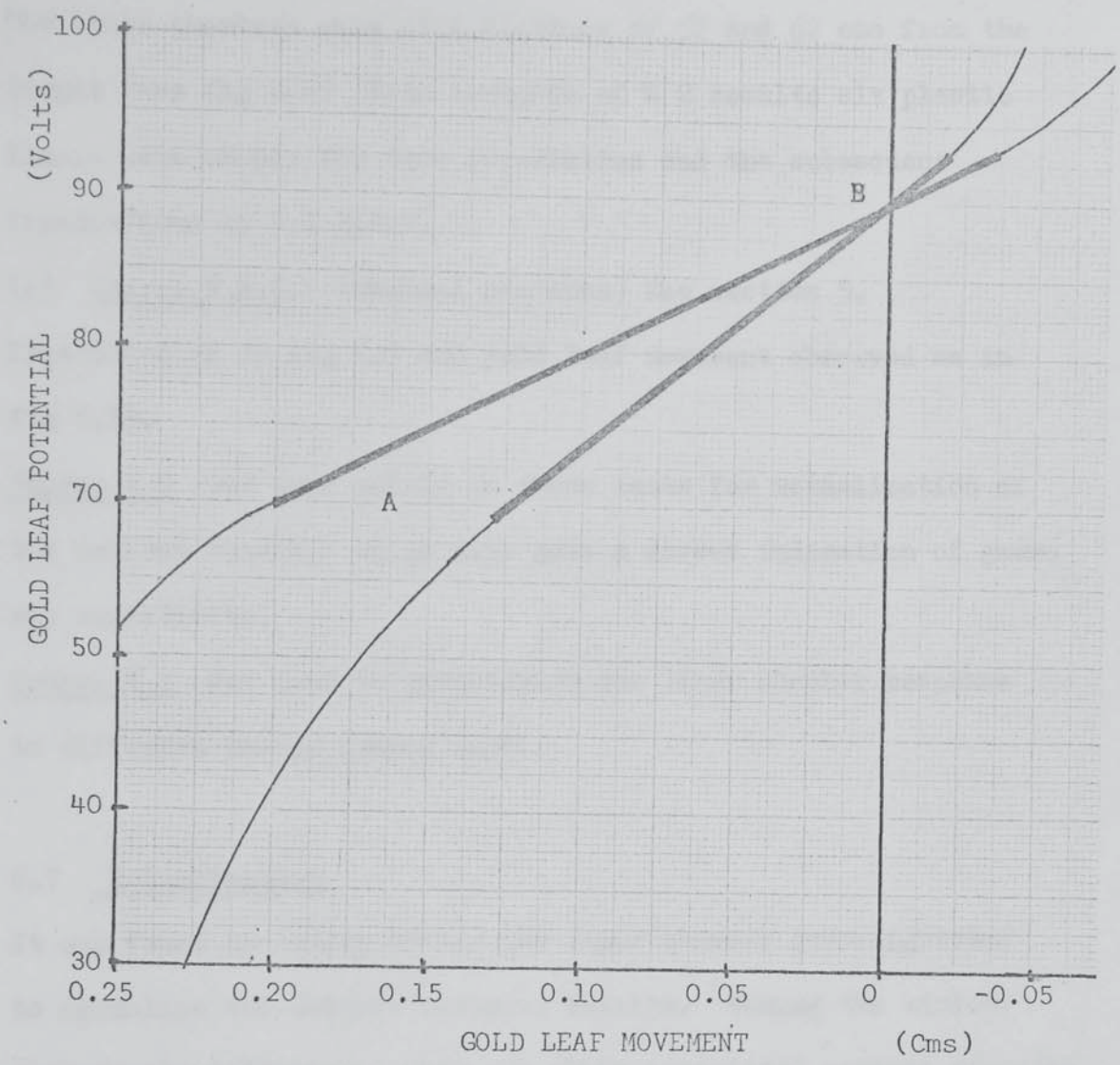
6.5 Liner Chamber calibration.

After construction the liner chamber was then placed in the jig (fig 5.5 but without the source present) and the voltage leads connected. If the gold leaf had satisfactory travel over the scale (section 6.2.1) a voltage calibration was then taken prior to commencement of the irradiations. The calibration consisted of taking leaf position as a function of voltage. A typical graph of such a calibration is given in fig 6.6. The leaf was then only used over this calibrated region. Also if the liner chamber was accidentally knocked or physically moved it was then found necessary to make another calibration. Using this voltage it was possible to compare one liner chamber with another.

6.6 Irradiation procedures.

(a) Source N 2 (5 Mev average energy neutrons) see section 3. The chambers were charged to approximately 80 volts and then given a succession of two minute irradiations until the leaf collapsed beyond the linear response region. The position of the leaf recorded after each irradiation by the travelling microscope

Fig 6.6



Typical voltage calibration curves for two liner chambers

Region A to B is the linear response region over which the instrument readings were taken. B being the region where the gold leaf is approaching the outer electrode and A the region where the leaf is approaching the backing plate. Calibration always being taken from the selected point B down to A.

(b) Source N 5 (14 Mev neutrons) See section 5 .
The liner chambers were at a distance of 57 and 62 cms from the target (see fig 6.8) From analysis of N 2 results six plastic liners were chosen for this irradiation and the subsequent irradiations in G.L.E.E.P.

(c) Source T.N.2. (Thermal neutrons) See section 5.
Irradiated as in fig 6.9 and gold leaf movement observed as in fig 6.10.

Source G 3 was used solely in these tests for normalisation of the neutron readings which also gave a direct indication of gamma ray sensitivity.

Source G 2 was used to investigate the liner chamber response to different energy gammas rays.

6.7 Normalisation

It was found necessary during the liner chamber investigations to normalise the neutron response results. During the various experiments different leaves and their associated perspex supports tried consecutively in the same chamber gave differing absolute neutron sensitivity results. This was probably due to the basically differing sensitivities of each gold leaf, their relative position in the chamber and the different electrical leakage rates of the perspex supports. One further effect would be the contribution to secondary ionisation by the perspex insulator and end plug. This contribution was impossible to estimate with any reasonable accuracy, but as it was common to all readings taken with a given geometrical configuration; On taking a comparison of the neutron sensitivity of one liner against another this effect would be minimised. However the most effective method for minimising the above errors was to normalise each

Fig 6.8

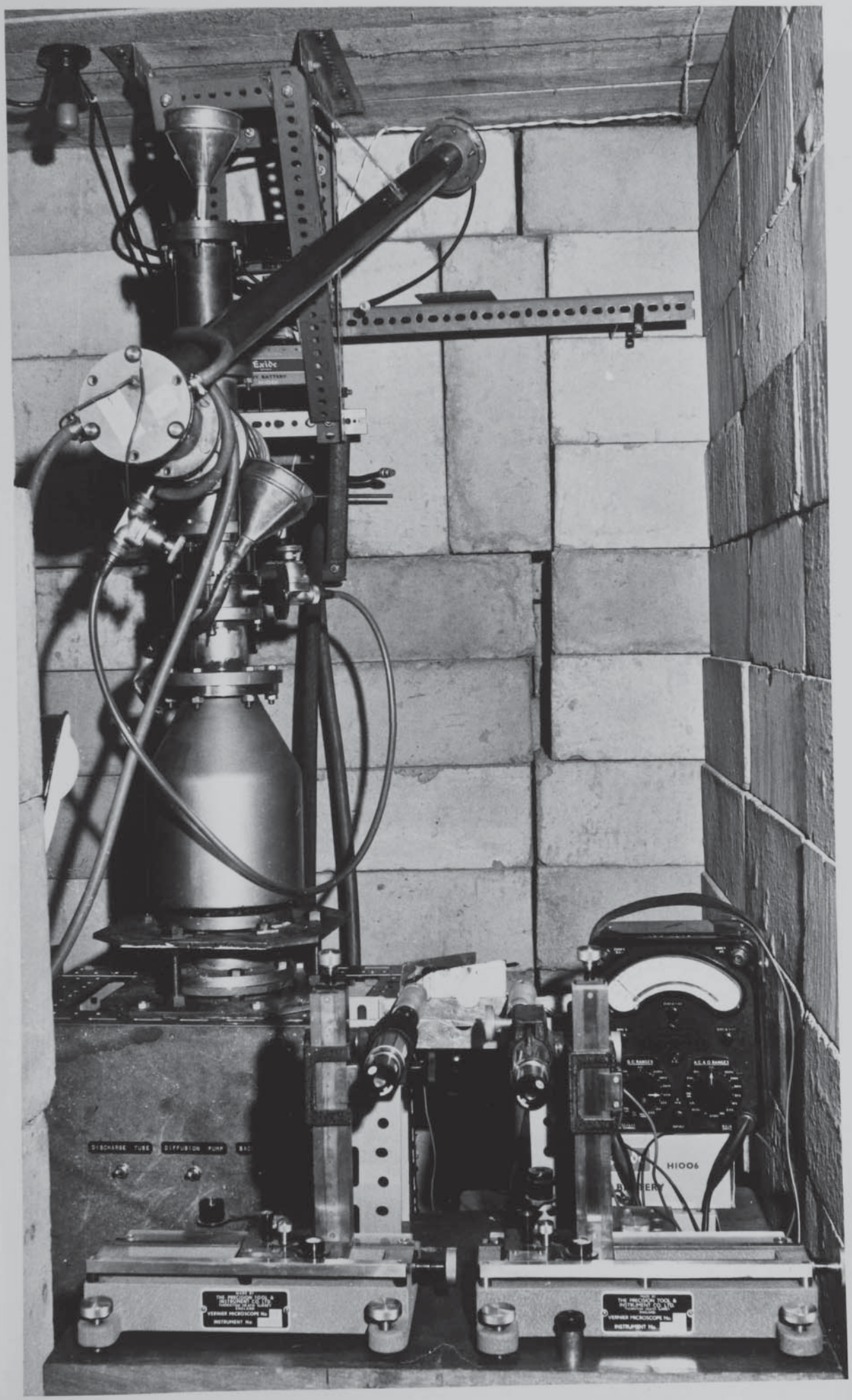


Fig 6.9

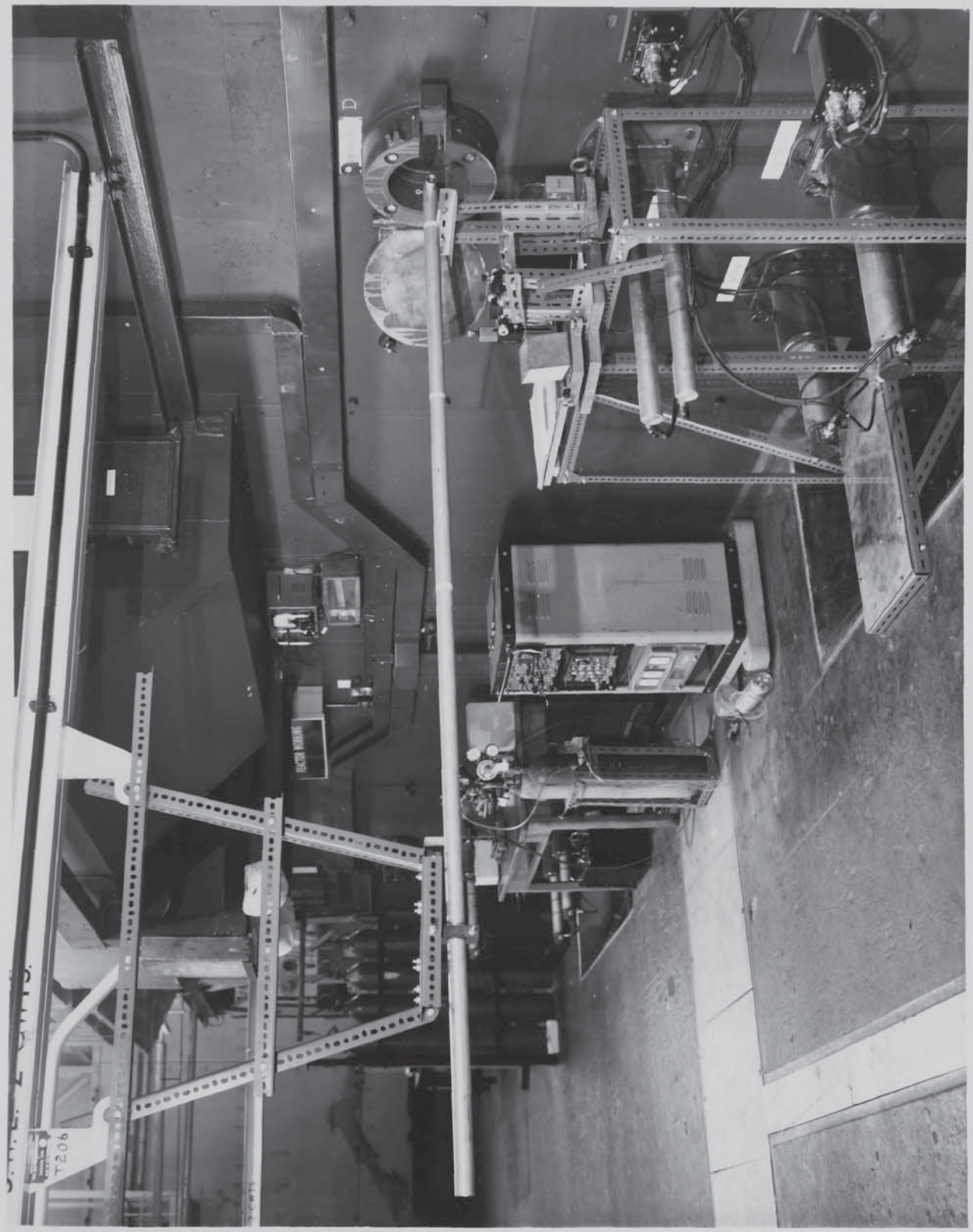
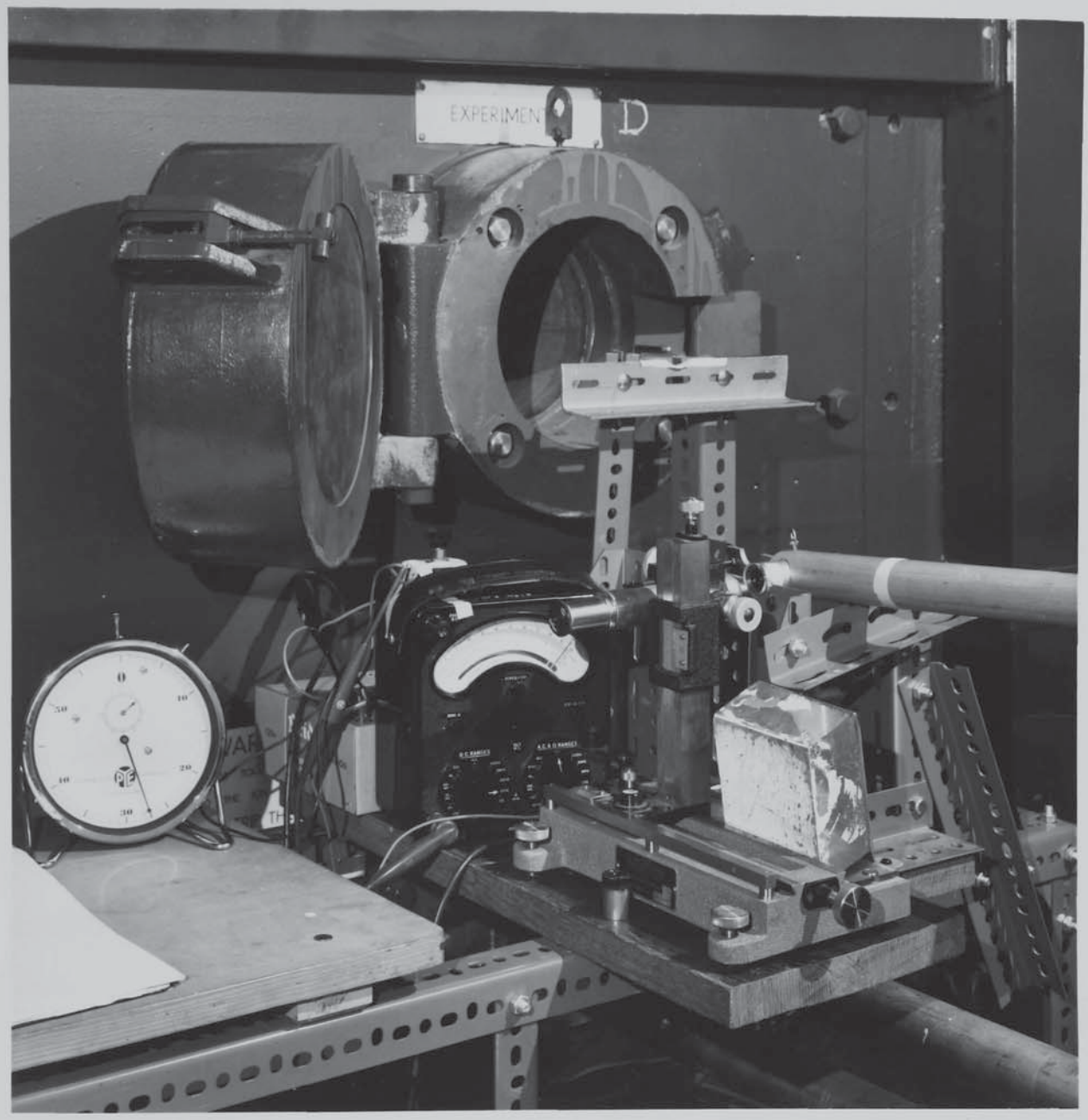


Fig 6.10



neutron reading to a reading taken with the gamma ray source. All readings were taken for a period of two minutes thus electrical leakage and other time dependent errors would be common to all chambers. Thus the majority of errors would be minimised by normalisation and the important parameter became the neutron to gamma ray sensitivity ratio. Thus a comparison of the ability of each liner to convert the incident neutron radiation to charged particle radiation (ionisation or conversion) efficiency was achieved.

6.8 Analysis of Results

A table of results for this experiment is given in fig 6.11. As the predominant reaction of neutrons in hydrogenous plastic is with the hydrogen nucleus a graph was drawn (fig 6.12) showing neutron to gamma ray sensitivity ratio against the hydrogen weight content of the plastic molecule. Hydrogen weight content being the ratio of the weight of hydrogen in the plastic molecule to the total molecular weight of the plastic. As can be seen from the graph for 5 Mev and 14 Mev neutrons, neutron to gamma ray sensitivity is directly proportional to hydrogen weight content. For thermal neutrons there does not appear to be much linearity, but this is probably due to the sensitivity being dependent on the gamma rays from neutron capture not recoil protons. In the 14 Mev neutron graph the only points off the line are nylon and the thin polythene. The increase in sensitivity for nylon is probably due to the oxygen in the nylon molecule giving an (n, α) reaction of 0.3 barns and a (n, p) reaction of 0.05 barns.

Fig 6.11

5 Mev irradiations

Liner	$H/Total$	n	γ	n/γ
Aluminium	0.000	1.87	2.53	0.74
Carbon	0.000	0.89	1.70	0.53
P.T.F.E.	0.000	1.08	1.87	0.58
P.V.C.	0.049	1.56	1.62	0.96
Thiokol S.T.	0.058	3.03	2.87	1.06
Cellulose Acetate	0.059	1.43	1.34	1.07
Polystyrene	0.077	2.85	2.36	1.21
Nylon 66	0.097	3.20	2.20	1.45
Polythene (Thick)	0.143	2.52	1.41	1.79
Polythene (Thin)	0.143	3.10	1.73	1.79
14 Mev irradiations				
P.T.F.E.	0.000	1.82	2.31	0.79
P.V.C.	0.049	2.44	2.17	1.12
Polystyrene	0.077	3.77	2.93	1.29
Nylon 66	0.097	4.16	2.24	1.86
Polythene (Thick)	0.143	5.40	3.10	1.74
Polythene (Thin)	0.143	3.76	2.51	1.49

Liner chamber n/γ ratios

(For thermal neutron irradiation results and table legend see following page).

Fig 6.11 continued

Thermal neutron irradiations

Liner	$^{H}/_{Total}$	n	γ	$^{n}/_{\gamma}$
P.T.F.E.	0.000	8.78	3.17	2.76
P.V.C.	0.049	7.94	2.32	3.42
Polystyrene	0.077	9.10	2.37	3.84
Nylon 66	0.097	4.22	1.21	3.49
Polythene (Thick)	0.143	7.14	1.49	4.79
Polythene (Thin)	0.143	4.03	1.06	3.80

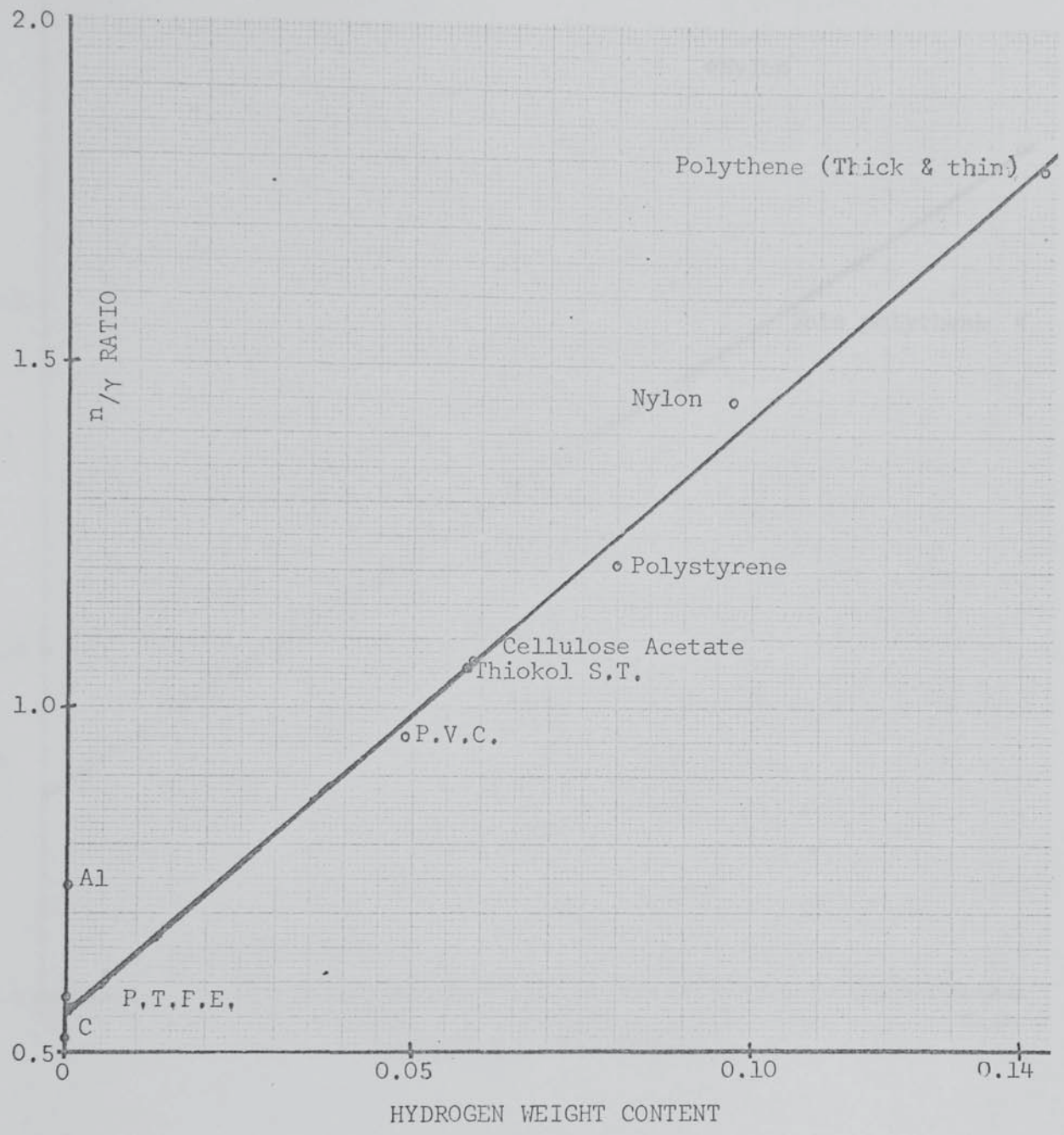
Liner chamber $^{n}/_{\gamma}$ ratios

$^{H}/_{Total}$ represents the fraction of the total weight of hydrogen present in the molecule to the total molecular weight of the complete molecule.

The "n" column represents the number of volts that the liner chamber discharges in two minutes with the neutron source. This is the average of approximately six readings taken along the linear response region of each chamber (see fig 6.6).

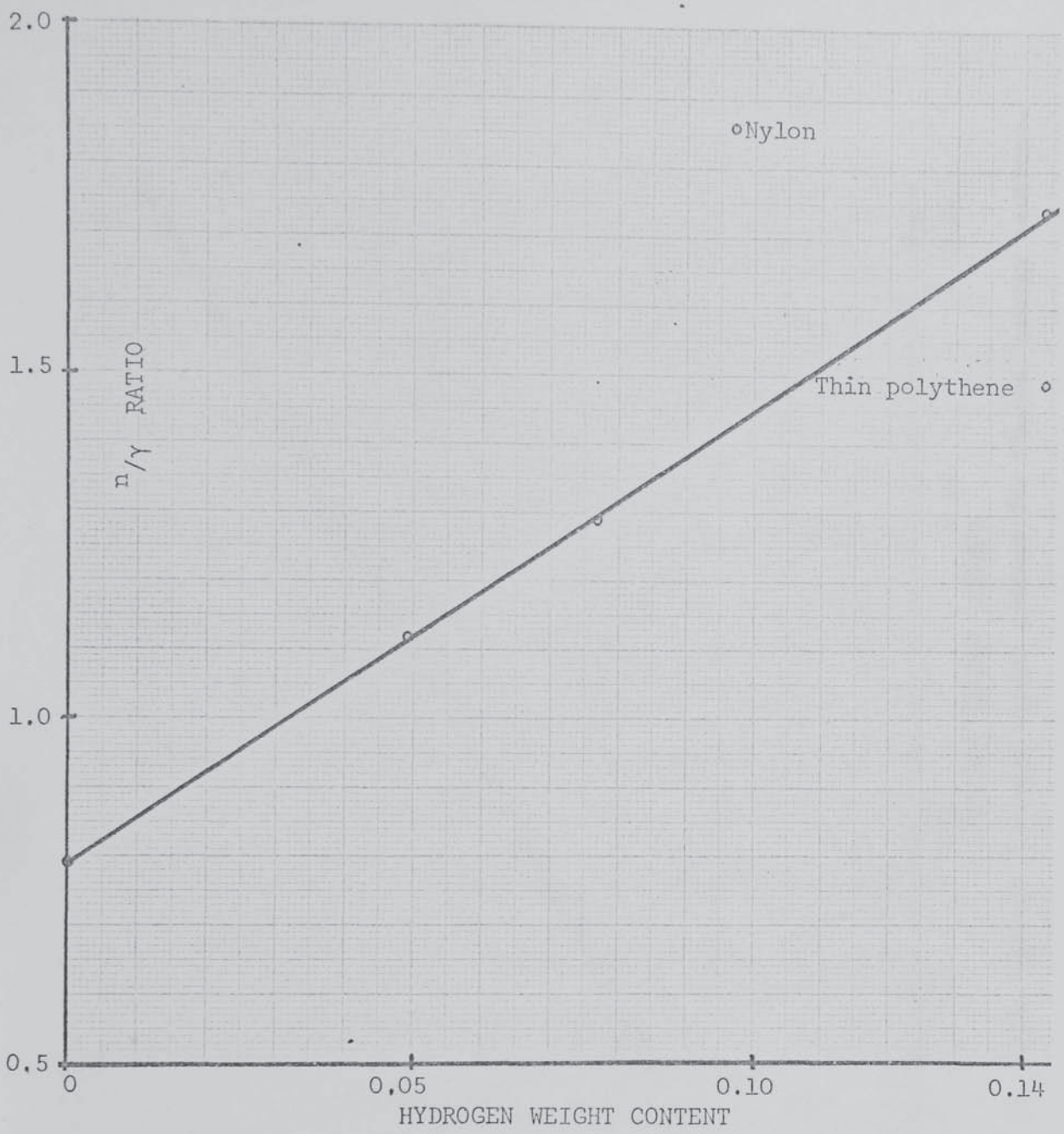
The " γ " column is the same for the gamma ray source.

Fig 6.12a



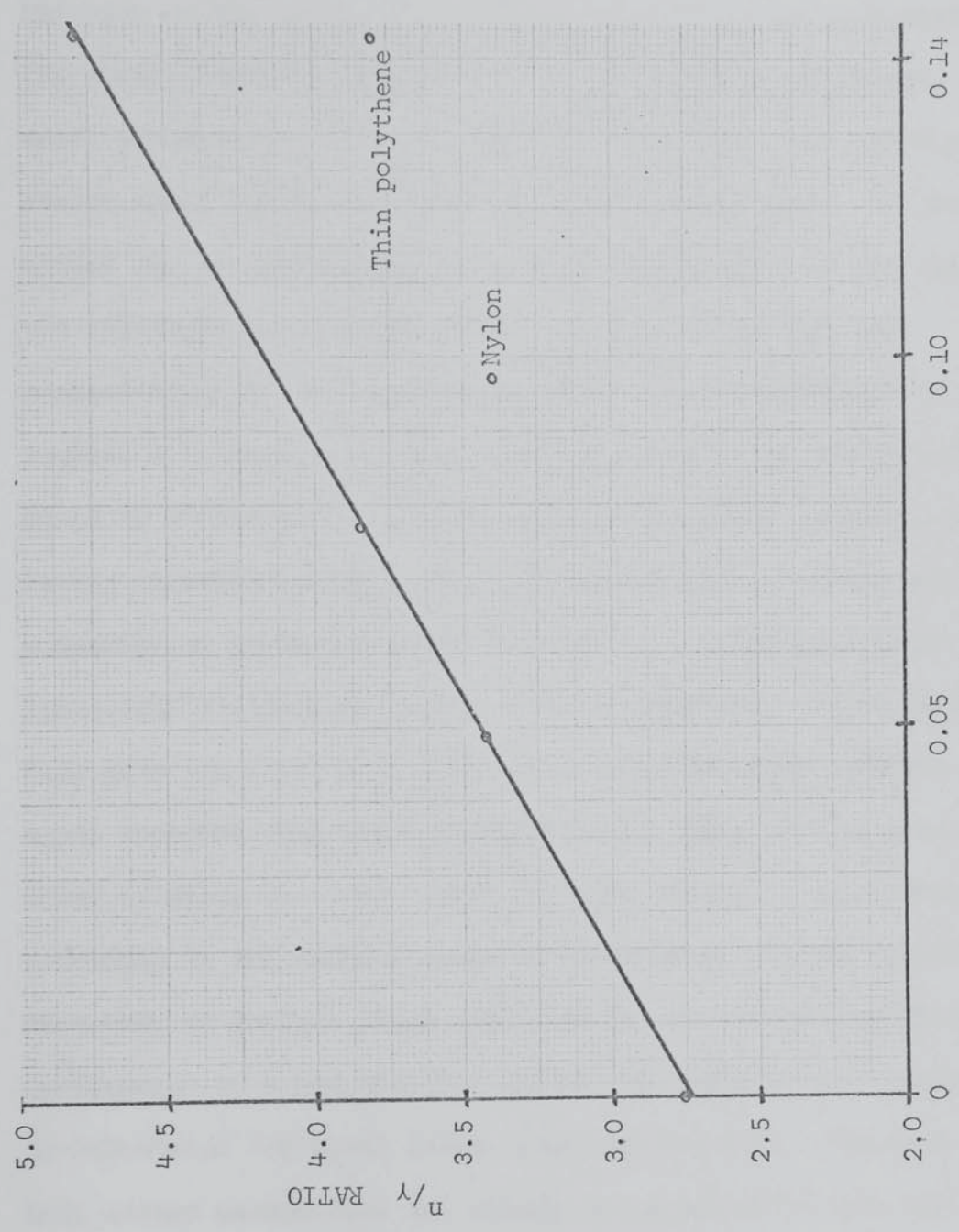
Liner chamber $^{n}/_{\gamma}$ ratio against hydrogen weight content for 5 Mev average energy neutrons.

Fig 6,12b



Liner chamber $^{n}/_{\gamma}$ ratio against hydrogen weight content for 14 Mev energy neutrons.

Fig 6.12c



Liner chamber $^{n}/_{\gamma}$ ratio against hydrogen weight content for thermal energy neutrons.

Thinner polythene gives a low sensitivity because its thickness is too small a fraction of the maximum recoil proton energy range at this higher energy. For the pure elements Carbon and Aluminium, together with the plastic P.T.F.E., there is no hydrogen in the liner. The higher than calculated sensitivity for these liners is probably due to the perspex insulators, which themselves would act as proton radiators. The perspex covers about 15% of the total interior surface area. The actual amount the perspex contributed to the sensitivity of the liners was difficult to estimate as the bulk of the perspex was perpendicular to the irradiation flux. This represented in some regions a "liner" which was over 2 cms thick. As the contributions would be virtually be the same for all the liner chambers (each having identical perspex insulators)and would be eliminated after a neutron to gamma ray ratio was taken , a straight inter-comparison between the liners would be possible. Fig 6.13 and 6.14 give the results of 6.11 after they have been adjusted for equal absorbed dose and dose equivalent. These tables have been drawn up using the dose rate for each source as calculated according to the methods given in Appendix A. It was considered necessary to produce these tables as in the calculation of the performance of a compensation device the sensitivities could only be calculated for equal doses. (see section 8.4). Fig 6.15 puts into better perspective the amount of compensation that has to be achieved by such an instrument. From the graphs it would seem possible to predict the neutron to gamma sensitivity of any plastic for energy below about 10 Mev. Reactions with heavier

Fig 6.13

Liner	Gamma rays	Thermal neutrons	5 Mev	14 Mev
Polythene (Thick)	1.0	4.39	0.80	0.69
Polythene (Thin)	1.0	3.48	0.80	0.59
Nylon 66	1.0	3.19	0.66	0.74
Polystyrene	1.0	3.52	0.56	0.51
P.V.C.	1.0	3.13	0.46	0.44
P.T.F.E.	1.0	2.53	0.26	0.31

Liner chamber results from fig 6.11 adjusted for equal absorbed doses.

Fig 6.14

Liner	Gamma rays	Thermal neutrons	5 Mev	14 Mev
Polythene (Thick)	1.0	2.34	0.114	0.111
Polythene (Thin)	1.0	1.85	0.114	0.095
Nylon 66	1.0	1.71	0.092	0.119
Polystyrene	1.0	1.87	0.077	0.083
P.V.C.	1.0	1.67	0.062	0.072
P.T.F.E.	1.0	1.35	0.037	0.051

Liner chamber $^{n}/_{\gamma}$ ratios from fig 6.11 adjusted for equal dose equivalent.

nuclei producing charged particles become more common for higher energy neutrons. From a dosimetry view point the plastic most suited for a neutron sensitive chamber is nylon. This can be seen from the results in fig 6.15. When a plastic is used in a compensation device (as given in section 8) based on subtraction of two chambers, ie a neutron and gamma sensitive and neutron insensitive chamber, nylon is still the most suitable. This can be seen in fig 6.16 where the various neutron sensitive plastics have been matched with the results for P.T.F.E. The only real discrepency being the results for thermal neutrons, reasons for this are explained later. However nylon is by no means identical to tissue equivalent materials. The only resemblance in fact is with the hydrogen weight content (see fig 6.17). Although the remaining elements are the same as in tissue they they are not in the same proportion as those in tissue equivalent plastic [50]. (Formula given in section 4) The plastic which is most insensitive to neutrons from the results in fig 6.11 is P.T.F.E., as predicted. It was also found that the ratios of neutron absorbed dose sensitivity to gamma absorbed dose sensitivity for the smaller commercial dosimeters (see section 7) and the larger liner chambers were both approximately 0.08. Which to some extent showed that scaling down the results from the larger to the smaller chambers was not unreasonable. However as hydrogen weight content appears to govern the fast neutron response up to 10 Mev presumeably any plastic with a 10% hydrogen weight content would be suitable. Above 10 Mev reactions with the heavier nuclei became predominant

Fig 6.15

Liner	Gamma rays	Thermal neutrons	5 Mev	14 Mev
Polythene (Thick)	-	144	7.0	6.8
Polythene (Thin)	-	114	7.0	5.8
Nylon 66	-	130	7.0	8.9
Polystyrene	-	170	7.0	7.5
P.V.C.	-	188	7.0	8.1
P.T.F.E.	-	256	7.0	9.6
Ideal ratio	-	3.0	7.0	6.2

Liner chamber $^{n}/_{\gamma}$ ratios from fig 6.14 expressed as a "Quality Factor" relationship.

This was achieved by multiplying the 5 Mev column up to 7.0, and then using the same multiplication factor on the thermal neutron and 14 Mev columns.

Fig 6.16

Liner $^{n}/_{\gamma}$ ratios from fig 6.13 with the P.T.F.E. ratio subtracted

Liner	Gamma rays	Thermal neutrons	5 Mev	14 Mev
Polythene (Thick)	-	1.83	0.54	0.38
Polythene (Thin)	-	0.92	0.54	0.28
Nylon 66	-	0.65	0.40	0.43
Polystyrene	-	0.99	0.30	0.20
P.V.C.	-	0.60	0.20	0.13

Thus for the flat response that would be required for absorbed dose measurements nylon 66 appears to be the most suitable liner.

Liner chamber $^{n}/_{\gamma}$ ratios from fig 6.14 with the P.T.F.E. ratio subtracted and also expressed as a "Quality Factor" relationship

Liner	Gamma rays	Thermal neutrons	5 Mev	14 Mev
Polythene (Thick)	0.011	0.99	0.077	0.060
Q.F.	1.0	90	7.0	6.2
Polythene (Thin)	0.011	0.50	0.077	0.044
Q.F.	1.0	46	7.0	4.0
Nylon 66	0.008	0.36	0.055	0.68
Q.F.	1.0	46	7.0	8.7
Polystyrene	0.006	0.52	0.040	0.032
Q.F.	1.0	91	7.0	5.6
P.V.C.	0.003	0.32	0.025	0.021
Q.F.	1.0	90	7.0	5.9

Element	Nylon $C_{12}\ H_{22}\ O_2\ N_2$	T.E.P. $C_5\ H_{40}\ O_{18}\ N$	T.E.P. $C_7\ H_{70}\ O_{32}\ N$	Acetal $C\ O\ H_2$	Cellulose Acetate $C_8\ O_7\ H_{13}$
Hydrogen	9.7 %	9.9 %	10.3 %	6.6 %	5.9 %
Oxygen	14.2 %	71.6 %	75.3 %	53.3 %	50.7 %
Carbon	63.7 %	14.9 %	12.4 %	40.0 %	43.4 %
Nitrogen	12.4 %	3.5 %	2.1 %	-	-

Table showing the weight content (expressed as a percentage) to total of the molecule, contained in 5 plastics (see section 6.8).

Fig 6.17

such as the (n, α) reaction of 0.3 barns and a (n, p) reaction of 0.05 barns with the oxygen in the molecule.(59) For this reason the oxygen containing plastics, acetal and cellulose acetate have been included in the table. For further interest both these could be investigated at 14 Mev to verify the significance of oxygen at this higher energy. Also included in the table are two of the latest multi-composition wall liners used in a wide range personnel dosimeter, based on the proportional counter.(65) Here presumably the sensitivity to higher energy neutrons being obtained by recoil carbon atoms. This was later modified.(66) The increase in nitrogen content was to achieve a higher thermal neutron sensitivity. This, it was claimed was achieved by the ^{14}N (n, p) ^{14}C reaction which provided 580 Kev protons from neutrons below 10 Kev. However a condenser chamber device having no discrimination against gamma rays would have protons produced from this reaction swamped by the capture gammas produced in thermal neutron reactions. This can be seen by the high sensitivities obtained at G.L.E.E.P. For example in hydrogen a 2.2 Mev gamma is produced. If this gamma ray interacts further, usually by Compton scattering a recoil electron of average energy 1.0 Mev will be produced.(67) For aluminium the figures are 7.5 Mev gamma producing a 4.4 Mev recoil electron. For the liner chambers it is suspected that the majority of thermal neutron sensitivity comes from the comparatively large amount of aluminium surrounding the chamber compared to that of the commercial dosimeters. A 4.4 Mev gamma produces 2.6 Mev average energy electron which would penetrate a 0.10 cm

liner easily. This can be seen in the results where the liner chambers have the higher thermal neutron sensitivity (see fig 6.11 and 6.13). So from an absorbed dose viewpoint a condenser ionisation chamber has a high thermal neutron sensitivity because of the gamma rays produced by neutron capture outside the limits of the liner. From these results and investigations it seems clear that there is no ideal liner, according to the requirements given in section 3, but that nylon approaches nearest to this ideal. Thus to cover the whole neutron energy range it is suggested that a nylon lined device would be incorporated in an instrument, constructed with the minimum of aluminium (to reduce thermal neutron sensitivity). This indicates a device perhaps, where the case and liner were made of nylon. This would certainly be possible if the rigidity of nylon was acceptable for this purpose. It is also interesting to note that aluminium has a higher neutron to gamma ray ratio than carbon. This can to some extent be explained by the consideration of their respective neutron cross sections. From these it can be seen that aluminium has more (n,γ) resonance peaks, and a (n,α) reaction with a threshold of about 6 Mev which carbon does not possess.

However if the thermal neutron results achieved with the liner chambers can be taken as unecessarily high due to the large amount of aluminium around the sensitive volume then from the tables in fig 6.15 and 6.16 polythene is the most suitable liner. For this reason and those given in section 7.4 it was chosen for further investigation.

Section 7 Tests on Quartz Fibre Dosimeters

7.1 Introduction

For convenience of identification within this text the quartz fibre (condenser ionisation) dosimeter is referred to as a 'commercial dosimeter'. (see fig 3.3).

Basically if a commercial dosimeter is to made that measures dose equivalent for neutrons it must :-

(a) Have the desired variation of neutron sensitivity as a function of neutron energy.

(b) Have the correct ratio of neutron to gamma ray sensitivity, in terms of dose equivalent, without the use of extensive lead shielding.

After the decision had been made to adopt the condenser ionisation chamber and following investigations of possible liners (see section 6) the actual investigation into the behaviour of 'life size' commercial dosimeters was begun. This work fell into three stages :-

Stage 1 Initial experimental work used to investigate the general response of existing instruments (ie, in current commercial production) and a sulphur lined dosimeter. Sulphur (used for reasons given in section 4) was not included in section 6 as it was found possible to construct a commercial dosimeter with a sulphur liner.

Stage 2 Investigations into the variation of response of different gas fillings and gas pressures.

Stage 3 Experiments to investigate the final choice of liner which would be used in a possible compensation device. A compensated commercial dosimeter is being produced by R.A.Stephen and Co based on the results of these investigations . However test results for this instrument are not yet available.

7.2 First stage

This involved the testing of three sulphur lined dosimeters, six standard gamma ray commercial dosimeters (produced by R.A.Stephen and Co) and three commercial dosimeters produced by an american manufacturer.

7.2.1 Sulphur lined dosimeter

These were basically standard commercial dosimeters produced by R.A.Stephen and Co with the bakelite (air equivalent) lining substituted by a sulphur lining. The liner had to be thick enough to rigidly support the quartz fibre movement and be at least 70% of the maximum range of 14 Mev neutrons (to produce a reasonable number of charged particles). Therefore the lining was made 0.127 cms (0.050 inches) thick and had a 0.0008 cms (0.0003 inches) graphite coating. On exposure to a Harwell calibrated radium source the relative sensitivity of the dosimeters was established. These results were in agreement with the results obtained from a cobalt 60 source (source G 3) at Aston (see fig 7.4). Each dosimeter was fitted with a copper capillary tube (pumping stem) to facilitate a change of gas filling or change of gas pressure within the sensitive volume.

This meant that investigations could be made into the change of response with various gas fillings. This was particularily interesting as it would give information regarding how the change of W (ionisation efficiency) affected the sensitivity. (see section 4). The organic alkene gases were chosen for investigation as they could be ultimately used with a polythene liner. (Thus complying with the Bragg cavity principle of a matched gas).

7.2.2 Tissue equivalent dosimeter

A neutron dosimeter constructed by an american manufacturer actually consisted of two quartz fibre condenser instruments. The mode of operation is given in section 2.9. On assessment of performance the device was found to be unsatisfactory on four counts:-

(a) Scale divisions in rads (discussed in section 2.9)

(b) The subtraction of the gamma reading from the neutron plus gamma reading was tedious and gave widely differing results. These readings were found to be below the calculated value of neutron absorbed dose on all occasions by 5% to 15% for 5.1 Mev average energy neutrons, and 20% to 40% for 14.0 Mev energy neutrons. Results are in fig 7.4.

(c) The neutron plus gamma dosimeter was more sensitive to gamma radiation than the neutron insensitive dosimeter. Consequently for an exposure to a pure gamma source an apparent "neutron dose" was always given an subtraction.

(d) There were two neutron plus gamma dosimeters the sensitivities of which differed by up to 50%. Both these dosimeters were

included in the investigations as their actual absolute sensitivity gave an indication of what one manufacturer considered acceptable for this type of device.

7.2.3. Stephen dosimeters

The readings taken under this heading were the average of six standard quartz fibre gamma ray dosimeters produced by R.A. Stephen

7.2.4. Irradiation Sources.

Four neutron sources and two gamma ray sources were used. For neutrons, Source N 4 had the two sets of results taken with 14 Mev neutrons. (with and without the lead shielding.). Source N 3, 5 Mev average energy neutrons (fig 5.4) and the attempted production of thermal neutrons (Thermal neutron irradiations were included in the second stage as G.L.E.E.P. facilty was not available at this time) The gamma - ray sources were 1.17 and 1.33 Mev from source G 3 and 0.66 Mev energy from source G 4. and as there was no perceivable difference in results observed between G 3 and G 4, they are both included under the general heading of gamma ray sensitivity results.

7.2.5. Result analysis

From fig 7.4 it can be seen that in terms of absorbed dose the overall sensitivity of the dosimeters to neutron and gamma radiation is fairly constant. However when the results are considered in terms of dose equivalent, the neutron sensitivity is lower than the gamma ray sensitivity on average by a factor

of ten. The intention of a compensation instrument is to give neutron and gamma radiation measured in terms of dose equivalent (rems). This requires neutron and gamma radiation to be measured in the ratio of their respective quality factors. This can be a value of 3 to 1 for thermal neutrons and up to 10 to 1 for fast neutrons. Therefore the compensation will have to raise a neutron sensitivity from a factor of ten lower, to a factor of ten greater than that for gamma rays. This requires a compensation of the order of a 100 times. (See fig 6.16.)

Alternatively the method described in section 7.2.2. could be adopted. This uses the subtraction of the reading of one device from that of another. However if this is to operate satisfactorily it necessitates having a matched pair of dosimeters. Unfortunately in a batch of dosimeters identically constructed, but there is shown on investigation a differing sensitivity between individual dosimeters of up to 5%. This manufacturing problem together with the points raised in section 7.2.2. gave rise to the obvious need for a single device with automatic compensation. (A compensation device was developed, details are given in section 8).

7.3 Second stage

This was completed in late 1968 and early 1969. Two alkene gases were chosen for investigation as it seemed from observations made in section 3, there would be a possibility of using this gas filling with a polythene liner in the ultimate instrument. (This

would then preserve the Bragg Gray cavity requirements). Three sulphur lined commercial dosimeters, each fitted with a pumping stem, were used for the investigations. The gases were ethylene $(C_2H_4)_n$ and cyclopropane $(C_3H_6)_n$ at pressures of from 0 to 5 lbs per square inch above atmospheric pressure. The dosimeters were connected to a pumping/filling rig with standard demountable unions and $\frac{3}{4}$ inch Genevac vacuum taps. This enabled gas mixtures to be easily changed and also gas to be kept in the dosimeter at pressures above atmospheric without being continually coupled to the gas cylinders. A standard rotary and diffusion pump were used to evacuate the dosimeters. Pressures were achieved at the manifold down to 10^{-6} torr. However it is rather unlikely that this was the pressure achieved within the dosimeter due to the high impedance of the pumping stem. Using the $\frac{3}{4}$ inch Genevac vacuum taps the dosimeters were connected to the pumping system as in fig 7.1 and fig 7.2.

The radiation sources, irradiation conditions and experimental procedures were identical to those of the first stage. The gas fillings for the irradiations were alternated as follows :-

(a) With nitrogen.

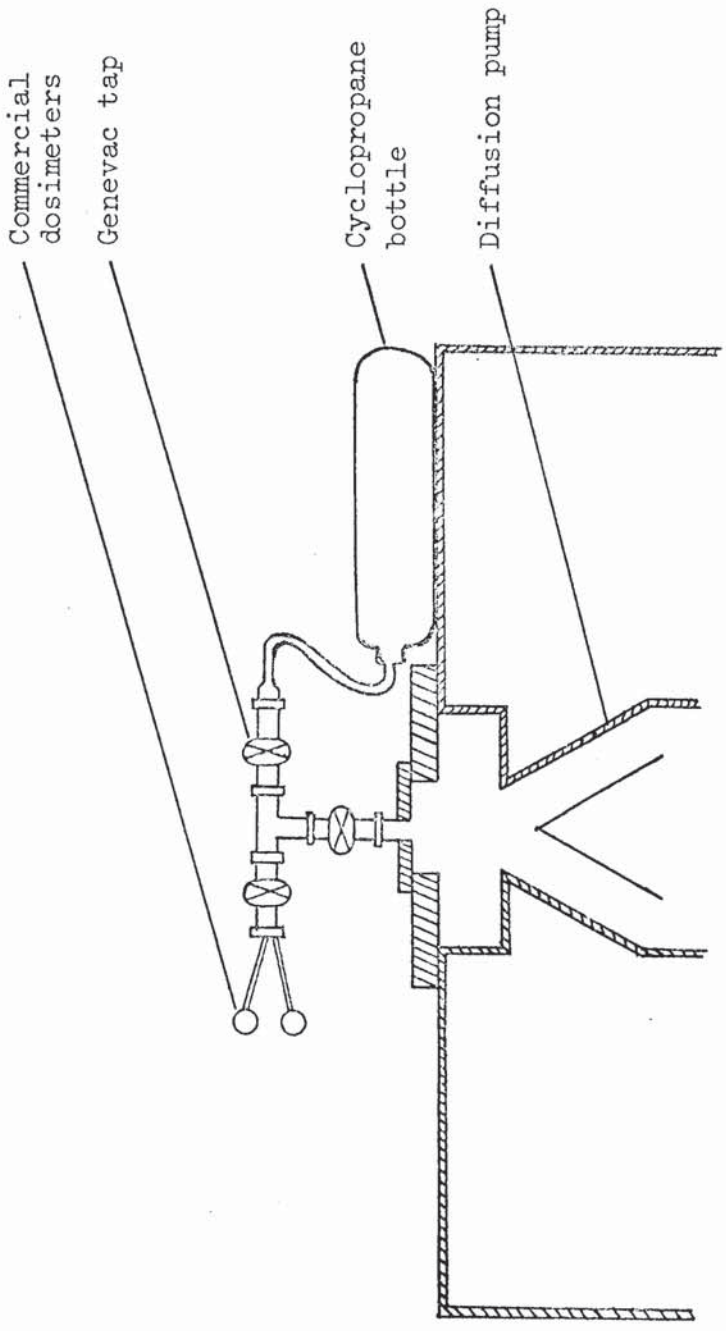
(b) With ethylene at atmospheric pressure.

(c) With ethylene at 5 lbs per square inch above atmospheric pressure.

(d) With cyclopropane at atmospheric pressure.

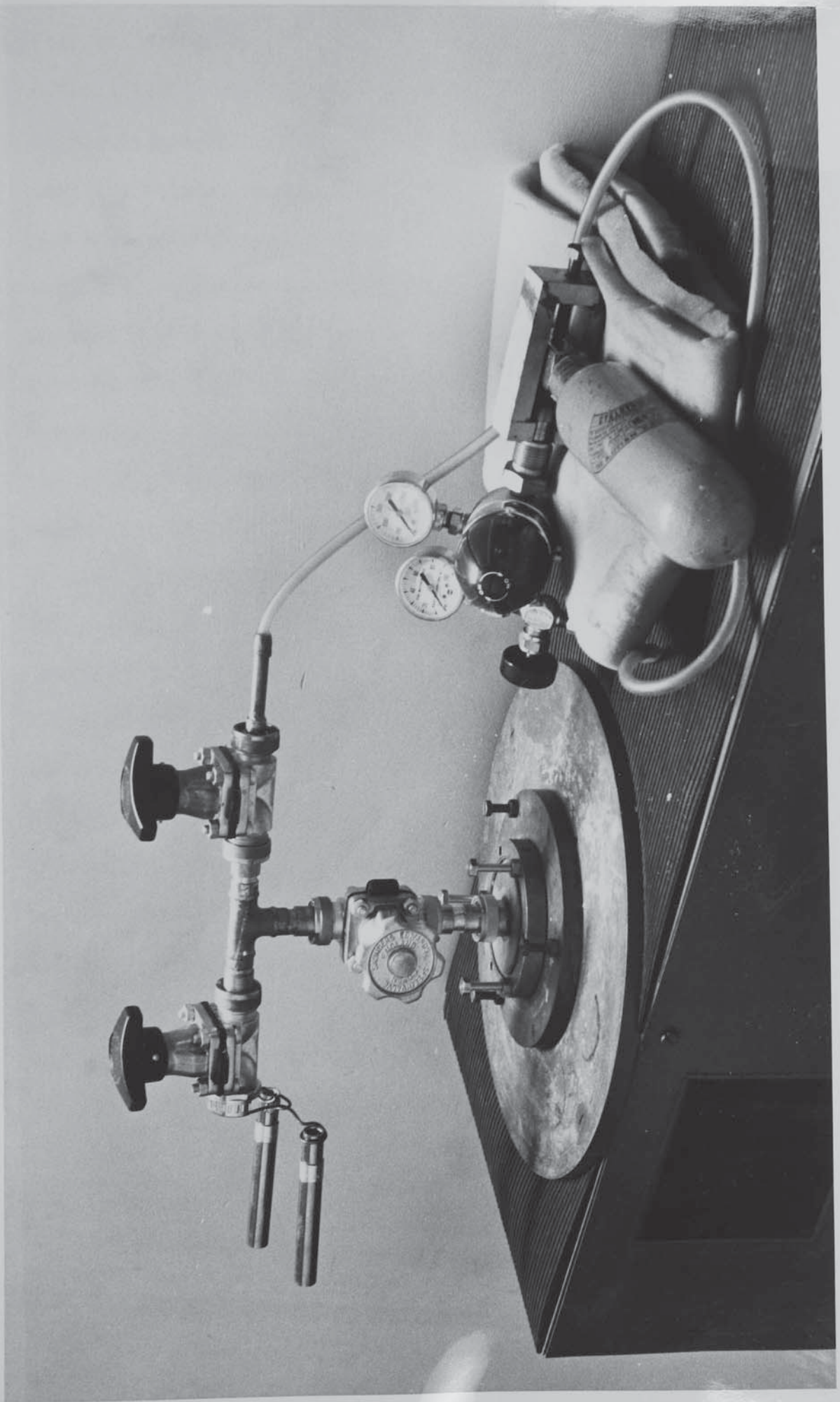
(e) With cyclopropane at 5 lbs per square inch above atmospheric pressure.

Fig 7.1



Pumping arrangement for commercial dosimeters (Second stage).

Fig 7.2



After the dosimeter was given the appropriate gas filling it was sealed off with the tap and the irradiations were completed with the tap kept well away from the direct line between source and sensitive volume. Care was taken not to exceed 5 lbs per square inch above atmospheric pressure in case of damage to the dosimeter. A complete day was allowed for pumping and the chambers were heated to about 60°c to improve "outgassing" of the internal components of the dosimeters. (78) However complete outgassing was impossible and it was found that after sixty days the relative sensitivity of a sealed ethylene gas-filled dosimeter had dropped by 25% to its initial sensitivity. However this problem is overcome in industry by preheating the components or sub-assemblies in a vacuum dessicator for a few days and then "soaking" them at a similar temperature in the gas-filling chosen for the instrument. Then cooling to room temperature while still in the gas. Thus this type of problem would be greatly reduced in a commercially produced instrument, since this phenomenon and its solution are already understood it was not thought worthwhile pursuing any further investigation. The table of results in fig 7.3 show that there is a marked increase in sensitivity with an alkene gas-filling. Rough figures show that compared to the normal nitrogen gas-filling there was:-

(a) 50% increase in sensitivity with $C_2 H_4$.

(b) 55% " " " " $C_2 H_4$ at 5 lbs pressure

(c) 65% " " " " $C_3 H_6$.

(d) 70% " " " " $C_3 H_6$ at 5 lbs pressure

Fig 7.3

Commercial dosimeter sensitivities and $^{n}/_{\gamma}$ ratios for various gas fillings

Dosimeter	Gas	Pressure	Radiation Sources				A $^{n}/_{\gamma}$	B $^{n}/_{\gamma}$
			N 4a	N 4b	N 3	G 3		
59	N_2	A	0.77	0.80	0.83	8.47	9.26	9.80
61	N_2	A	0.56	0.61	0.65	9.09	6.44	7.09
63	N_2	A	0.83	0.80	0.90	10.86	7.52	8.26
59	$C_2 H_4$	A	1.25	1.31	1.41	16 67	7.69	8.47
61	$C_2 H_4$	A	1.20	1.33	1.27	14.08	9.05	9.00
63	$C_2 H_4$	A	1.49	1.64	1.61	16.95	9.24	9.52
59	$C_2 H_4$	A + 5	-	-	1.61	16.67	-	9.71
61	$C_2 H_4$	A + 5	1.45	1.61	-	16.67	9.20	-
63	$C_2 H_4$	A + 5	1.85	2.04	-	23.81	8.59	-
61	$C_3 H_6$	A	1.89	1.67	1.92	22.22	8.00	8.62
63	$C_3 H_6$	A	2.38	2.27	2.70	28.57	8.13	9.43
61	$C_3 H_6$	A + 5	2.13	-	2.17	27.03	7.87	8.06
63	$C_3 H_6$	A + 5	2.70	-	2.94	30.30	8.93	9.71

59 = Sulphur lined dosimeter, serial number DP-25-2 No 59.
61 = " " " " " " No 61.
63 = " " " " " " No 63.
N 4a = Source N 4 with lead shielding.
N 4b = Source N 4 without lead shielding.
Pressure = Atmospheric (A) or 5 lbs / sq in above atmospheric (A + 5).

Sensitivity in scale divisions (20 divs for F.S.D.) / mrem x 10^{-2}
$^{n}/_{\gamma}$ ratios are also x 10^{-2}. Where A = $^{N\ 4}/_{G\ 3}$ and B = $^{N\ 3}/_{G\ 3}$.

These increases are constant for irradiations with 5 Mev average energy neutrons, 14 Mev neutron and gamma rays. Therefore it would seem feasible to improve sensitivity by 50% to use an ethylene filled dosimeter. The reason for this increase is probably due to:-

(a) The average energy needed to form an ion pair (W) is much lower for ethylene than for nitrogen. W for nitrogen is 36.4 eV/ ion pair, while for ethylene it is about 26.5 eV / ion pair (68)

(b) The hydrogen in ethylene would contribute knock on protons which would increase the ionisation within the sensitive volume.

The main disadvantage with a gas filling is the difficulty of "pre-soaking" (defined earlier) and the effect that it would have on the overall response of the dosimeter to different energy neutrons as well as gamma rays. As it has been found that polythene liners do follow the correct energy response curve for neutrons, but on average in a ratio of 1 to 1.45 greater than tissue.(54) Therefore it is reasonable to assume that an ethylene gas-filling would improve the overall response of a dosimeter. This would also apply to a compensation device as described in section 8, as the alkene gas would increase gamma ray sensitivity but not to the same extent as neutron sensitivity. Therefore the combined effect would still improve the sensitivity of a compensation device. Facilities were on hand to provide alkene gas fillings but the complications involved in maintaining a constant level of alkene gas within the sensitive volume, (ie without following the long and expensive procedure of outgassing) would necessitate a constant flushing process . Therefore it was decided that the most

satisfactory method was to use a dry air filling for the dosimeters and liner chambers. Therefore before commencement of experimental procedure the instruments were kept open in a dessicator for approximately seven days. However it was born in mind that with an ultimate device the sensitivity, using an alkene gas filling, could be improved.

7.4 Third Stage

This was completed in 1970 and consisted of investigating the response of commercial dosimeters constructed to specification by R. A. Stephen, resulting from work completed earlier in the project. These were constructed with two differing liners, one sensitive to neutrons and gamma rays the other insensitive to neutrons. The liners used were selected from the various liners investigated in section 6 and ultimately to be used in the compensation device as given in section 8. The main point of interest at this stage was to find out if

(a) The degree of sensitivity.

(b) The differing ratio's found between the various liners .

(c) The differing response to varying neutron energies and gamma rays were of the same order for the much smaller sensitive volume and associated equipment used in the commercial dosimeter. Therefore with that in mind graphite loaded polythene and aluminininium were chosen.

Polythene was chosen in preference to nylon because it is easier

to mould and is easiest to mix with graphite. The graphite was added so that the liner would be electrically conducting, Aluminium was chosen in preference to carbon or P.T.F.E. because it is easier to machine. Also it is electrically conducting and would rigidly support the quartz fibre mechanism in the focal plane of the optical system (unlike P.T.F.E.) . The liners were 0.09 cms (0.035 inches) thick. The dosimeters were assembled in the usual way as for gamma ray dosimeters, and filled with nitrogen. Four specimens of each were made.

7.4.1 Irradiation sources

Three neutron sources and one gamma ray source were used. Source N.2 for 5 Mev neutrons, source N.5 for 14 Mev neutrons, source T.N.1 for thermal neutrons and source G.2 for gamma rays.

7.4.2 Results analysis

The results are divided into two tables given in fig 7.4 and 7.5. The former gives the average readings of four polythene/graphite lined dosimeters and four aluminium lined dosimeters, the latter gives the results of two selected chambers one of which was polythene/graphite and the other aluminium lined. These chambers were selected from the results in fig 7.4 as the two most likely dosimeters to operate successfully in a compensation device.

From fig 7.4 it can be seen that if the average polythene/graphite chamber is matched to the average aluminium chamber for the compensation principle as laid down in section 8 (ie simple automatic subtraction), then the results obtained do not give

Fig 7.4

Various commercial dosimeter sensitivities and $^{n}/_{\gamma}$ ratios

Dosimeter	Gas	Radiation Sources				A $^{n}/_{\gamma}$	B $^{n}/_{\gamma}$
		N 4a	N 4b	N 3	G 3		
59	N_2	0.77	0.80	0.83	8.47	9.26	9.80
61	N_2	0.56	0.61	0.65	9.09	6.44	7.09
63	N_2	0.83	0.80	0.90	10.86	7.52	8.26
S	N_2	0.81	0.85	0.90	10.00	8.33	9.00
B_1	?	1.56	1.58	1.81	12.19	12.89	14.92
B_2	?	1.23	1.30	1.51	10.53	12.05	14.49
B_3	?	0.67	0.70	0.74	8.54	8.03	8.70
		T.N.	N 4				
P	N_2	5.02	1.35	0.77	8.94	15.10	8.60
Al	N_2	5.31	0.58	0.33	8.94	6.50	3.70

59 = Sulphur lined dosimeter, serial number DP-25-2 No 59
61 = " " " " " " No 61
63 = " " " " " " No 63
S = Average results of 6 standard Stephen gamma ray dosimeters.
B_1 = Fast neutron and gamma ray dosimeter Ser. No. B100238.
B_2 = " " " " " " " B020031.
B_3 = Gamma ray dosimeter. " 0288.
P = Average results of 4 Polythene/graphite lined dosimeters.
Al = " " " Aluminium lined dosimeters.

N 4a = Source N 4 with lead shielding.
N 4b = Source N 4 without lead shielding.
T.N. = Thermal neutron source T. N. 1.

Sensitivities in scale divisions (20 divs = F.S.D.) per mrem x 10^{-2}.
$^{n}/_{\gamma}$ ratios are also x 10^{-2}. A = $^{N\,4}/_{G\,3}$ and B = $^{N\,3}/_{G\,3}$.

Fig 7.5

Sensitivity results for two selected commercial dosimeters

Dosimeter	Radiation Sources			
	T.N.1	N 3	N 4	G 3
4	53.5	8.60	10.75	93.7
11	52.1	2.94	5.99	96.7
11 adjusted	49.9	2.82	5.74	92.7
Subtraction	3.6	5.78	5.01	1.0
Ideal Q.F. ratio	3.0	7.0	6.2	1.0

From the results for the polythene/graphite and aluminium lined dosimeters (given in fig 7.4) the sensitivities of two dosimeters has been isolated, that would give the most favourable response when used in a compensation device based on the subtraction principle.

4 is a polythene/graphite lined dosimeter.

11 is an aluminium lined dosimeter.

"11 adjusted" has been reduced so that the overall response on subtraction of 11 from 4 is unity (ie the Q.F. for gamma rays). The remaining results have been adjusted by a similar proportion (ie multiplied by 0.9586).

the correct Q.F. relationship over the neutron energy range and for gamma rays. The basic reason for this is that in terms of dose equivalent the neutron sensitivity is about ten times lower than that of the gamma ray sensitivity. Thus when the dosimeters are selected for compensation subtraction the gamma ray sensitivities of both dosimeters have to be very close to one another to obtain a net reading that is direct function of the neutron dose equivalent to the neutron reading, ie a slight variation of gamma ray sensitivity drastically affects the neutron sensitivity. If the chambers are individually selected as in fig 7.5 the resultant neutron to gamma ray sensitivity ratio can be seen to be closer to the desired Q.F. ratio for dose equivalent measurements. In production of a compensation device on a commercial scale this selection process would provide a difficulty. Therefore if a neutron dosimeter based on this compensation principle is to be made commercially, a means must be found of evaluating two chambers for a matched system without individual assembly of each chamber and individual testing, before a final choice of chambers is made. One possible solution is to use the volume to capacitance ratio of the chamber. (See section 8.4). This parameter is directly linked to the neutron and gamma ray sensitivities by some constant. Therefore on measuring the capacitance of each chamber (with its centre electrode) before final assembly an adjustment could be made to the volume of one or both to enable a matched pair to be produced. Alternatively a selection could be made from a batch of neutron sensitive chambers pairing each off with a neutron insensitive chamber,

all these selections made on measurement of capacitance alone.

It is noticeable that although the thermal neutron sensitivities for all the dosimeters measured are high compared to the gamma ray sensitivities, they are not as high as the thermal neutron to gamma ray sensitivities for the liner chambers, see fig 6.11. This indicates that thermal neutron induced reactions in the aluminium case are contributing significantly to the thermal neutron sensitivity, see section 6.6. It is found that there is more than three times the thickness of aluminium around the liner chambers than there is for the average commercial dosimeter. The approximate 7.5 Mev energy gamma ray produced in the capture process would have little difficulty in reaching the sensitive volume of the chamber, together with a high proportion of the associated beta particles (see section 6.6). It was found that on average the thermal neutron to gamma ray sensitivity ratio was for the liner chambers, seven to eight times larger than that for the commercial dosimeters.

Section 8 Compensation Device

8.1 General Requirements

The first requirement of any neutron measuring device is that it should have some means of discriminating against gamma radiation. This has been fully explained in section 2. Briefly the reason is that the secondary ionisation indirectly produced by both forms of radiation are approximately equal for equal absorbed dose rates, but the relative neutron response is up to ten times lower compared to gamma rays when equal dose equivalent rates are considered.

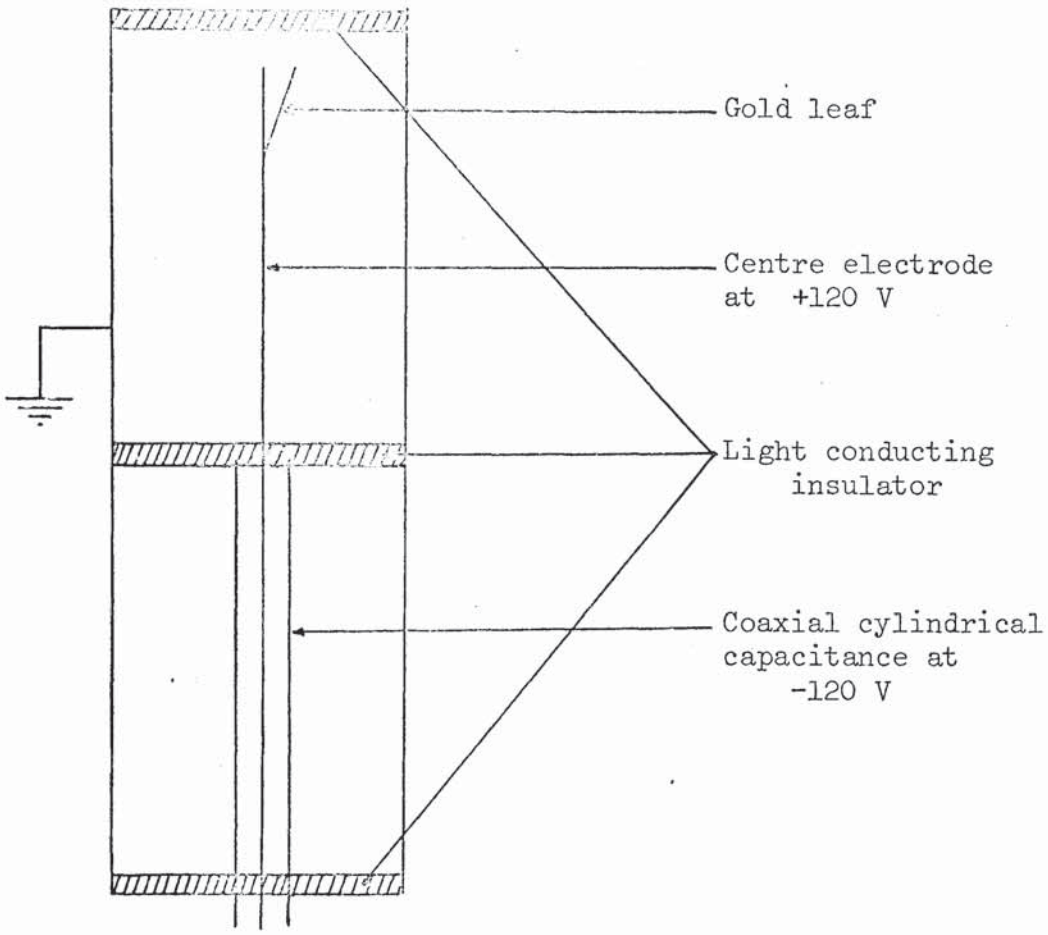
Discrimination is achieved in proportional counters by having a bias voltage which removes the pulses of smaller pulse height due to the gamma ray.[54,69] An alternative method is to have a twin ion chamber system.[70,71] Here one chamber is connected in opposition to the other. This is achieved by having one chamber with a high sensitivity to neutrons and the other a low neutron sensitivity, each having the same gamma ray sensitivity. Thus when connected there is an unbalance for neutrons while the gamma ray contributions cancel.

8.2 Construction and Development

It was decided that the last method mentioned above had possibilities if it would be capable of adaption to a device based on the condenser ionisation chamber.

One feature of the condenser ionisation chamber which made this adaption possible was that its response is not governed by the polarity of the charge placed on the central electrode. Thus this electrode (holding the quartz fibre) can be positive or negative with respect to the outside wall electrode.
Therefore a twin chamber system could be devised where one chamber had positive charge on the centre electrode with respect to the outer electrode, and the other chamber the reverse. By use of the appropiate wall liners and gas fillings for the sensitive volume one chamber could be made neutron sensitive and the other neutron insensitive, with both chambers having the same gamma ray sensitivity. Therefore on charging the chambers and exposing them to the same amount of neutron irradiation there would be an unbalance. (The neutron sensitive chamber discharging faster than the other chamber). However with the same gamma irradiation both chambers would lose charge at the same rate and the net loss of charge would be zero. Thus if connected together the chambers could be made to operate a gold leaf or quartz fibre. Alternatively the response of each chamber could be arranged such that the gamma irradiation did cause an unbalance of charge loss, but in the correct Q.F. relation to that charge loss experienced under the neutron irradiation. However to achieve the correct relation between the neutron and gamma ray discharge rates requires that the chambers should be capacitively linked as in fig 8.1, which shows the first simple concept.

Fig 8.1

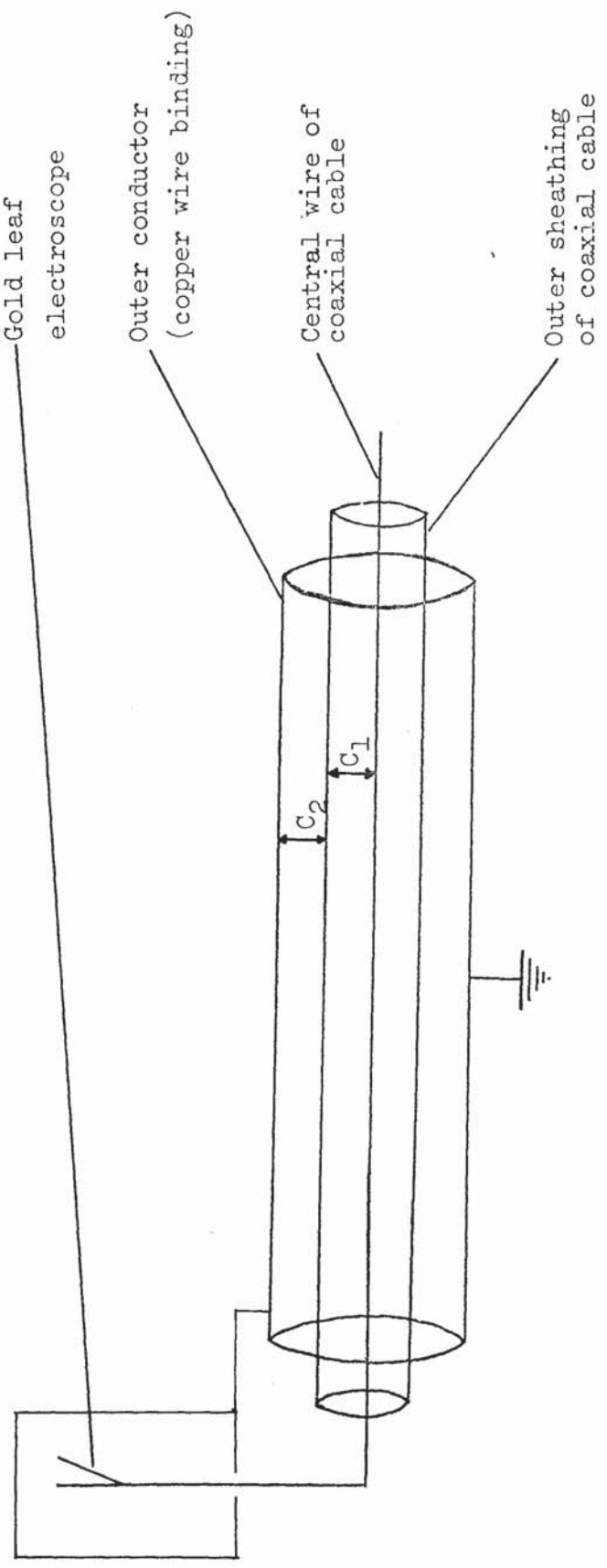


First simple concept of capacitance linked ionisation chambers

The idea of compensation was first investigated in the laboratory using a quick and very simple method with a coaxial cable and a gold leaf electroscope. The coaxial cable was bound with copper wire and voltages applied as in fig 8.2, the centre wire being connected directly to the electroscope. The capacitance C_1 representing one chamber of a compensation device and C_2 representing the other chamber. It was clear from direct observation of the gold leaf that reducing the charge on C_2 increased the charge on C_1. Thus it was seen that the charge on one chamber could significantly influence the charge on an adjoining chamber if connected in the correct way.

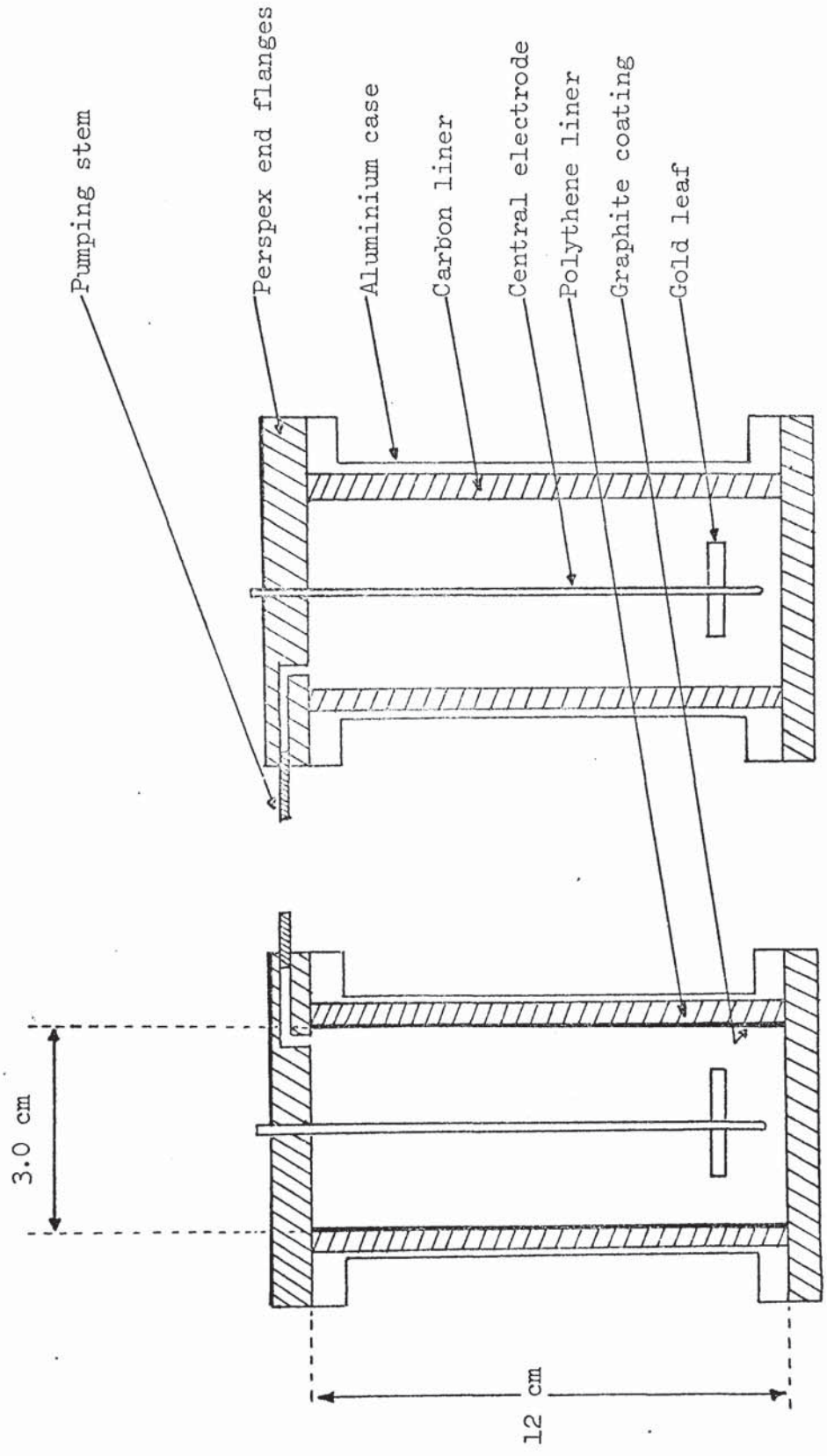
Therefore following this result two chambers were constructed for use under irradiation conditions using this principle. These were of aluminium length 12 cms and diameter 3 cms. The aluminium was approximately 0.35 cms in thickness ($^1/8$ inches). Each of the chambers had perspex end flanges. 1.3 cms thick ($\frac{1}{2}$ inch) to support a gold leaf, allow light transmission for easy observation of the gold leaf and to provide electrical insulation. One flange was drilled and fitted with a pumping stem so that the sensitive volume of the chamber could be pressurised if necessary. (See fig 8.3 and 8.4) Chamber A had a carbon lining machined from reactor grade graphite and Chamber B had a polythene lining cut from flat sheet. Each liner was 0.2 cms thick. The experiments were performed with source N 1 and source G 1. It was found that the experiments fell into three

Fig 8.2



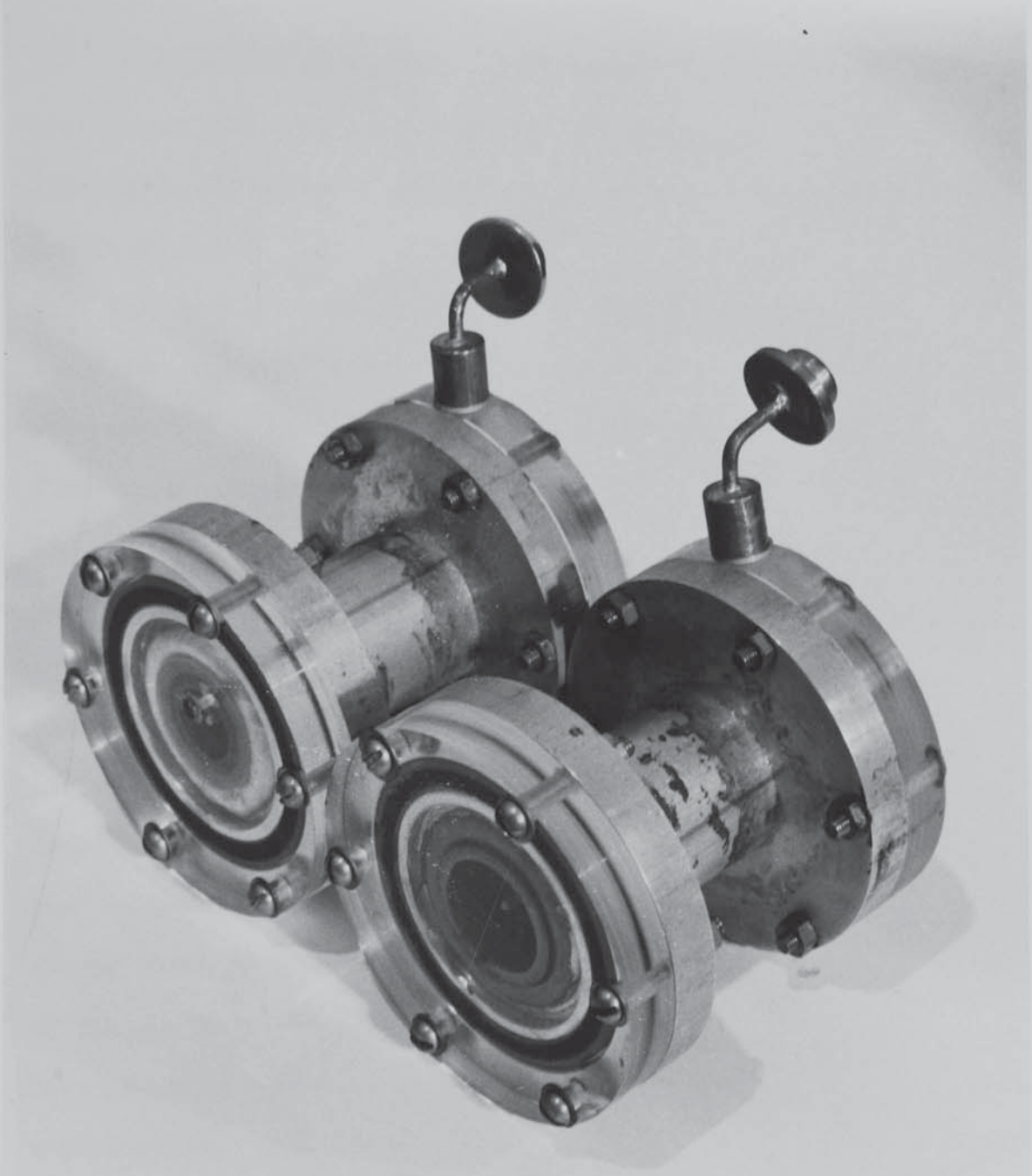
Coaxial cable investigation.

Fig 8.3



The first compensation chambers tested

Fig 8.4



distinct stages. Each stage was a logical development of the previous one :-

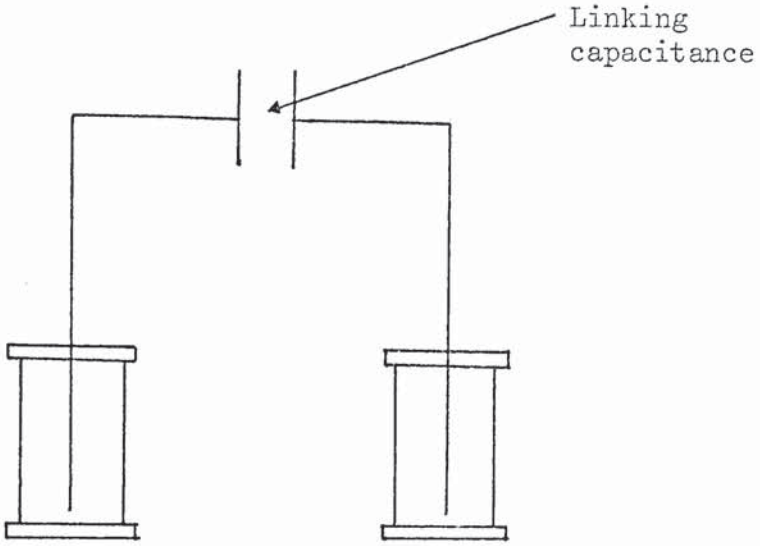
(a) Connected as in fig 8.5. This proved to be unsatisfactory because of the supposed irradiation effects on the external linking capacitance. (Later found to be radiation effects on perspex insulators)

(b) The chambers were rearranged and connected in fig 8.6. Improvements in the consistency of the results were noted but there were still unexplained irradiations effects (see section 6). These were now thought to be on the external leads connecting the chambers to the linking capacitance.

(c) Connected as in fig 8.7 and fig 8.8. This was the final development and had the linking capacitance moved inside the chamber by now placing it between the outer electrode of the neutron insensitive chamber and earth. This eliminated any external connecting leads as the chambers could now be placed back to back. It was found that complete compensation for gammas could be achieved by this means by altering the pressure of the gas in the sensitive volume of the neutron sensitive chamber. However over a period of hours these results were not reproducible. This was due to an effect, (which was not fully appreciated at the time of the experiments,) in the perspex end insulators. Known as "soakage", the effect is explained in section 6.

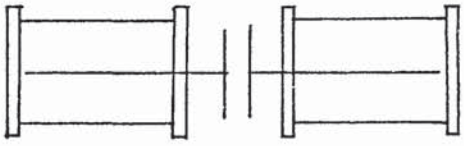
As a result of these experiments a Provisional Patent was drafted and submitted, the basic claims being that:-

Fig 8.5



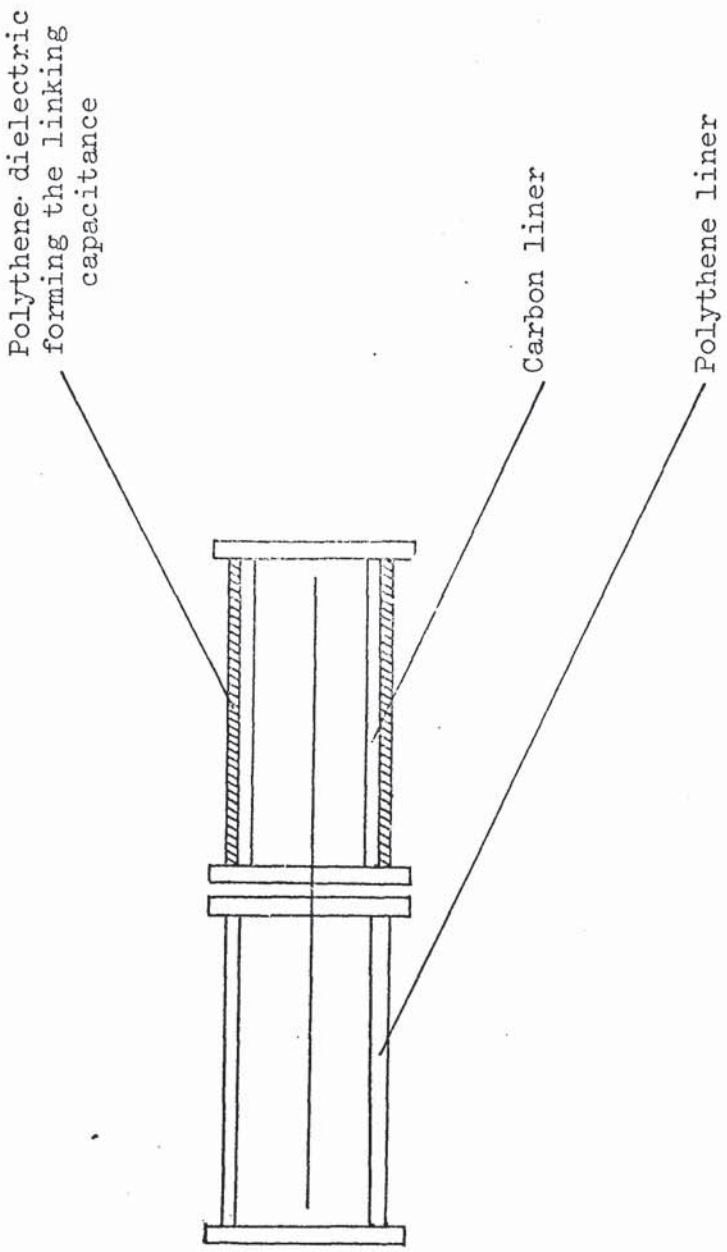
First method of chamber connection

Fig 8.6



Second method of chamber connection

Fig 8.7



Third method of chamber connection

(a) A neutron and gamma radiation sensitive chamber of lower relative neutron sensitivity which when combined in an instrument give an indication of the effective biological dose of neutron and gamma radiation or neutron dose alone.

(b) The charging of the two chambers mentioned in (a) to have electric fields in opposite senses such that there is cancellation or partial cancellation of the charges collected.

(c) The addition of a linking capacitor between the two chambers as in (a) so that the response to neutron and gamma radiations is proportional to the effective biological dose in the right quality factor ratio.

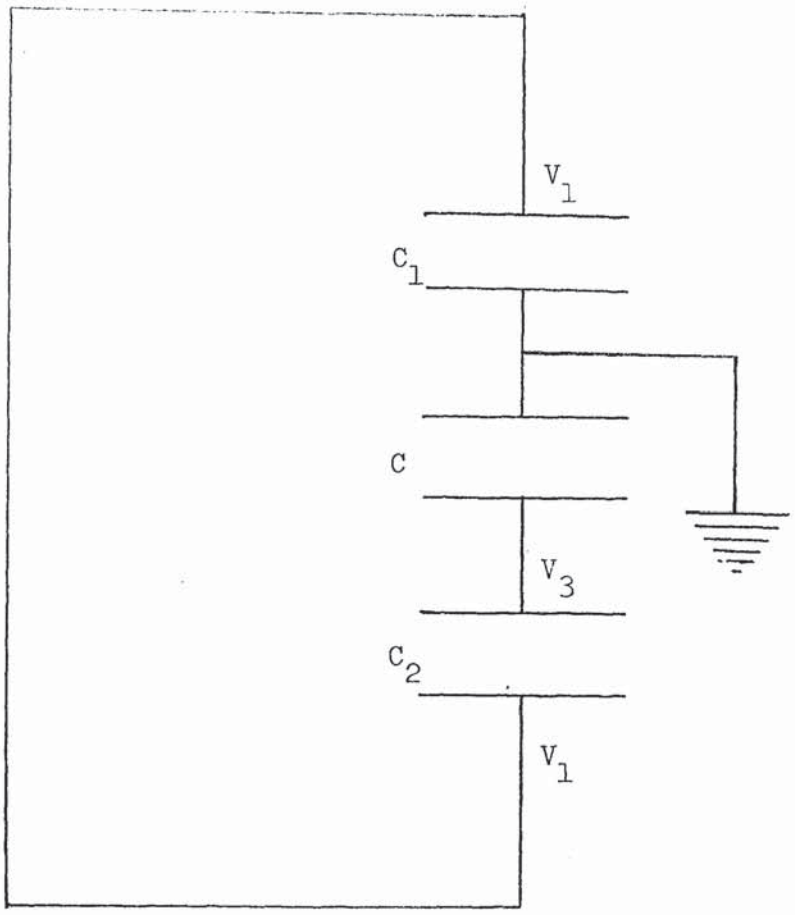
(d) The addition of a linking capacitor between the two chambers as in (a) such that there is complete compensation of the gamma radiation dose.

The complete patent is given in appendix B.

8.3 Compensation Theory

The circuit diagram for the compensation device is given in fig 8.9. It is simply three capacitances connected in series. For mathematical convenience it is first assumed that C is not in the circuit. C_1 is charged to voltage V_1 and C_2 to voltage $V_3 - V_1 = V_2$. ($V_3 > V_1$). Equal doses of radiation are now applied simultaneously to C_1 and C_2. C_1 undergoes a voltage drop of δV_1 and C_2 a voltage drop of δV_2. C at voltage V_3 is now connected. Immediately a quantity of charge δq flows from C into

Fig 8.9



Compensation device circuit diagram

C_1 and C_2. Thus the voltage of C will drop by an amount $\delta q/C$ and that of C_1 and C_2 will rise by an amount $\delta q/C_1$ and $\delta q/C_2$ respectively until an equilibrium is reached. Therefore :-

$$\delta q/C + \delta q/C_1 + \delta q/C_2 = \delta V_1 + \delta V_2 \quad ....(8.1)$$

For C_1, the neutron sensitive chamber containing the quartz fibre, the actual net change in potential is :-

$$\delta V_1^* = \delta V_1 - \delta q/C_1 \quad ....(8.2)$$

Substituting for δq using equation (8.1) and (8.2) :-

$$\delta V_1^* = \delta V_1 - \frac{\delta V_1 + \delta V_2}{\left[1/C + 1/C_1 + 1/C_2\right] C_1}$$

Therefore :-

$$\delta V_1^*/\delta V_1 = 1 - \frac{1 + \delta V_2/\delta V_1}{\left[1/C + 1/C_1 + 1/C_2\right] C_2}$$

Therefore :-

$$\delta V_1^*/\delta V_1 = \frac{1/C + 1/C_2 - 1/C_1\left[\delta V_2/\delta V_1\right]}{1/C + 1/C_1 + 1/C_2} \quad ....(8.3)$$

If $C_1 = C_2$ and $C >> C_1$ and the two chambers are arranged to have the same sensitivity (ie $\delta V_2 = \delta V_1$), then equation 8.3 becomes :-

$$\delta V_1^*/\delta V_1 = 0 \qquad \text{ie} \quad \delta V_1^* << \delta V_1$$

so there will be complete compensation.

Alternatively if one chamber is insensitive to one particular form of irradiation (ie neutrons) then :-

$$\delta V_2 << \delta V_1 \qquad \text{therefore} \quad \delta V_2/\delta V_1 = 0$$

and from equation(8.3):-

$$\delta V_1^*/\delta V_1 = \tfrac{1}{2}$$

For general simplification of equation(8.3)the following can be substituted :-

$S = \delta V_2/\delta V_1$ is the relative sensitivity of both chambers when they are not connected in a compensation circuit.

$R_1 = \delta V_1^*/\delta V_1$ is the relative response of the first chamber to that of the first chamber in a compensation circuit as defined in equation (8.2).

$K_1 = C_1/C$

$K_2 = C_2/C$

Therefore on multiplying equation (8.3) by C_1C_2/C and substituting the above :-

$$R_1 = \frac{K_1(1 + K_2) - SK_2}{K_1K_2 + K_1 + K_2}$$

$$= \frac{1 + K_2 - S\,C_{2/C_1}}{1 + K_2 + C_{2/C_1}} \qquad \ldots.(8.4)$$

If $C_1 = C_2$ and $C \to \infty$ $K_2 \to 0$ then:-

$$R_1 = \frac{1 - S}{2} \qquad \ldots.(8.5)$$

Equation (8.4) can be used to solve for any desired degree of compensation that may be required, ie for the comparisen of the neutron to gamma ray response :-

$$R_{1n/R_{1\gamma}} = \frac{K_1(1 + K_2) - S_nK_2}{K_1(1 + K_2) - S_\gamma K_2}$$

$$= \frac{1 + K_2 - S_n\,C_{2/C_1}}{1 + K_2 - S_\gamma\,C_{2/C_1}} \qquad \ldots.(8.6)$$

Equation (8.6) can now be used to determine the size of the linking capacitor C for any desired ratio of $R_{1n/R_{1\gamma}}$

Rearranging for K_2 :-

$$K_2 = \frac{S_n\, C_2/C_1 - 1 - R_{n/R_\gamma}(1 - S_\gamma\, C_2/C_1)}{1 - R_{n/R_\gamma}}$$

Therefore :-

$$C/C_2 = \frac{R_{1n/R_{1\gamma}} - 1}{1 + R_{1n/R_{1\gamma}}(S_\gamma\, C_2/C_1 - 1) - S_n\, C_2/C_1} \quad \ldots.(8.7)$$

If the quality factor (Q.F) is introduced into equation (8.7) the expression can be written in a more convenient practical form :-

$$R_{1n} = {}^{\delta V^*_{1n}}/_{\delta V_{1n}}$$

$$R_1 = {}^{\delta V^*_{1\gamma}}/_{\delta V_{1\gamma}}$$

Therefore :-

$$R_{1n/R_{1\gamma}} = {}^{\delta V^*_{1n}}/_{\delta V^*_{1\gamma}}\ {}^{\delta V_{1\gamma}}/_{\delta V_{1n}} \quad \ldots.(8.8)$$

If the instrument is to measure dose equivalent the quality factor of the radiation will determine the discharge rate of the instrument. If this is the case, then for exposure to the same absorbed dose of neutrons and gamma rays will give the value of quality factor for that particular neutron energy, ie :-

$$QF = {}^{\delta V^*_{1n}}/_{\delta V^*_{1\gamma}} \quad \ldots.(8.9)$$

Therefore substituting in (8.8) for the value of Q.F obtained in (8.9), and multiplying above and below by C_1 :-

$$R_{1n}/R_{1\gamma} = QF\ q_{1\gamma}/q_{1n} \qquad \ldots.(8.10)$$

Substituting (8.9) in (8.7) :-

$$C/C_2 = \frac{QF\ (\ q_{1\gamma}/q_{1n} - 1)}{1 + QF\ q_{1\gamma}/q_{1n}\left[q_{2\gamma}/q_{1\gamma} - 1\right] - q_{2n}/q_{1n}} \qquad ..(8.11)$$

As $S_\gamma\ C_2/C_1 = \delta V_{2\gamma}/\delta V_{1\gamma}\ C_2/C_1 = q_{2\gamma}/q_{1\gamma}$ ....(8.12)

and $S_n\ C_2/C_1 = \delta V_{2n}/\delta V_{1n}\ C_2/C_1 = q_{2n}/q_{1n}$ ....(8.13)

For complete cancellation of the gamma ray dose :- $\delta V_1^* = 0$

Therefore from equation (8.9) :- $QF \to \infty$

Rearranging(8.11) :-

$$C/C_2 = \frac{q_{1\gamma}/q_{1n} - 1/QF}{1/QF + q_{1\gamma}/q_{1n}\left[q_{2\gamma}/q_{1\gamma} - 1\right] - q_{2n}/q_{1n}\ QF}$$

$$= \frac{q_{1\gamma}/q_{1n}}{q_{1\gamma}/q_{1n}\ q_{2\gamma}/q_{1\gamma} - 1}$$

Therefore :-

$$C/C_2 = \frac{1}{q_{2\gamma}/q_{1\gamma} - 1} \qquad \ldots.(8.14)$$

8.4 Calculation of the theoretical response of a Commercial Dosimeter

If a condenser quartz fibre dosimeter (commercial dosimeter) were to be made incorporating a compensation chamber, some idea of the expected behavior of the response of the instrument to variation in chamber and electrode size would be required. Unfortunately there are several variable parameters which govern the absolute response of such an instrument. Therefore before an attempt is made to calculate the response of such an instrument the effect of some of these variables must be simplified. The main variables are :-

(1) Chamber diameter.

(2) Chamber length.

(3) Chamber capacitance.

(4) Chamber wall thickness and type of wall material.

(5) Central electrode size and configuration.

(6) Liner thickness and type of liner material.

(7) Gas filling and gas pressure.

In the calculations which follow the capacitance of the chambers is related directly to their length and is expressed as pF / cm.

Also the volume is linked directly to the capacitance of the chamber, expressed as D/C. This enables charge to be expressed as a function of chamber volume :-

For chamber one :-

$$\begin{aligned} q_{1\gamma} &= Q_{1\gamma} D_1 \\ q_{1n} &= Q_{1n} D_1 \end{aligned} \qquad \text{.....(8.15)}$$

For chamber two :-

$$\begin{aligned} q_{2\gamma} &= Q_{2\gamma} D_2 \\ q_{2n} &= Q_{2n} D_2 \end{aligned} \qquad \text{.....(8.16)}$$

Substituting in equation (8.14), multiplying the right hand side above and below the line by C_1 and rearranging :-

$$1/C = 1/C_1 \left[Q_{2\gamma}/Q_{1\gamma} \left[\frac{D_2/C_2}{D_1/C_1} \right] - C_1/C_2 \right] \qquad \text{.....(8.17)}$$

The response of the neutron chamber under compensation conditions will be :-

Substituting (8.17) in (8.4) :-

$$R_{1n} = \frac{1 + C_2/C_1 \left[\frac{Q_{2\gamma} D_2/C_2}{Q_{1\gamma} D_1/C_1} - C_1/C_2 \right] - \delta V_{2n}/\delta V_{1n} \; C_2/C_1}{1 + C_2/C_1 \left[\frac{Q_{2\gamma} D_2/C_2}{Q_{1\gamma} D_1/C_1} - C_1/C_2 \right] + C_2/C_1}$$

Substituting equations (8.13) and (8.16) and rearranging :-

$$R_{1n} = \frac{1 - \left[\frac{Q_{2n}/Q_{2\gamma}}{Q_{1n}/Q_{1\gamma}}\right]}{1 + \left[\frac{D_1/C_1}{D_2/C_2}\right] Q_{1\gamma}/Q_{2\gamma}} \quad \ldots.(8.18)$$

There are a number of simplifications which can be made to the above equation. Firstly if the ratio of the response of an uncompensated neutron sensitive chamber to neutrons, compared to that for gamma rays, for the same absorbed dose is :-

$$\delta V_{1n}/\delta V_{1\gamma} = q_{1n}/q_{1\gamma} = Q_{1n}/Q_{1\gamma} \quad \ldots.(8.19)$$

and similarily for the neutron insensitive chamber is :-

$$\delta V_{2n}/\delta V_{2\gamma} = q_{2n}/q_{2\gamma} = Q_{2n}/Q_{1\gamma} \quad \ldots.(8.20)$$

From the results in fig 6.13 this directly represents the neutron to gamma ray sensitivity ratios. Therefore if nylon and P.T.F.E. are chosen as the liners for 5 Mev average energy neutrons then :-

$$\frac{Q_{2n}/Q_{2\gamma}}{Q_{1n}/Q_{1\gamma}} = 0.4 \quad \ldots.(8.21)$$

Also if $Q_{2\gamma}$ is approximately equal to $Q_{1\gamma}$, then equation (8.18) becomes :-

$$R_{1n} = \frac{0.60}{1 + \left[\frac{D_1/C_1}{D_2/C_2}\right]} \qquad \ldots\ldots(8.22)$$

One interesting fact which arises from this equation is that the neutron response of the instrument does not depend on the actual size of the chambers, but in the way that the volume to capacitance of the chambers is related. For a maximum sensitivity instrument it would seem that :-

$$D_1/C_1 \ll D_2/C_2$$

and for a minimum sensitivity instrument :-

$$D_1/C_1 \gg D_2/C_2$$

There are no mathematical limits to the volume/capacitance ratio from this equation, but obviously these would be set in practise by the physical dimensions required in the field. However so that a direct comparison could be made with the liner chamber results, the ratios given in (8.19) and (8.20) were completed on the basis that the two chambers have similar volumes and capacitances. On this assumption the response becomes 0.3 (or 30%). This figure could be improved by say doubling the volume of the neutron insensitive chamber keeping the capacitance

constant, giving about 10% increase in neutron response. Further, if the capacitance C_2 was also halved the neutron response would increase by another 8%. In practise this would be feasible as increasing D_2 would decrease C_2, unless the actual length of the chamber was also reduced. However there would a further limit to this manoevre,

For complete gamma ray compensation in the device $Q_{2\gamma} = Q_{1\gamma}$ and if this is substituted in (8.17), D_2/D_1 has the lower limit of unity :-

$$1/C = 1/C_2 \, (D_2/D_1 - 1) \qquad \ldots.(8.23)$$

Therefore :-

$$D_2/D_1 > 1$$

The neutron response from (8.20), could then conceivably vary from 0 to 60% (ie 0 to 0.6) of that of the uncompensated chamber.

However it is evident from consideration of all the derived equations that (8.4) contains the most significant information. It can be seen from (8.4) that the response to neutrons and gamma rays is governed by the capacitance ratios C_1/C_2 and the relative sensitivity S. For a high neutron sensitivity C_1 should be larger than C_2. If an arbitary choice is made such that $C_1 = 10C_2$ then $R_{1n} = 88\%$, but instead of this producing a value of $R_{1\gamma}$ in the region of 11%, $R_{1\gamma} = 83\%$.

Upon consideration of the physical principles involved it is easy to see how this has occurred. The gamma response of the neutron sensitive chamber is now too large to be significantly compensated by that fed back from the neutron insensitive chamber. However this assumption is based on the value of S_γ being unity and $D_1 = D_2$, which need not necessarily be the case if C_1 was increased. The capacitance C_1 varies inversely, but not linearly with D_1 for a fixed value of chamber length. Thus if C_1 was increased by a factor of 10, D_1 would decrease by a factor of approximately one half. Therefore the effective value of S_n and S_γ would double giving a reduction in R_{1n} to 85% and $R_{1\gamma}$ to 75%, which still results in an incorrect Q.F. ratio.

Upon substitution of figures in this equation (8.4), it becomes apparent that although there are various combinations of the parameters which will give the correct value of Q.F. , there is no maximum or minimum sensitivity instrument which can be achieved by this means. For example if a low neutron sensitivity instrument is required, from inspection of the equation it is apparent that $1 + C_2/C_1$ should approach $S_n C_2/C_1$. Therefore if $S_n = 0.4$ and $C_2 = 3C_1$ a value of R_{1n} is given at as 0.65% . However this results in a value of $R_{1\gamma}$ equal to -29%, this means that under gamma irradiation the leaf will actually gain charge. (This effect was observed in section 9).

The effect on response which results with the variation of

these parameters can be demonstrated further by consideration of equation (8.6) and (8.10) :-

$$QF\ q_{1\gamma}/q_{1n} = \frac{1 + C_{2/C} - S_n C_{2/C_1}}{1 + C_{2/C} - S_\gamma C_{2/C_1}} \quad \ldots.(8.24)$$

If an absorbed dose is required (where Q.F. = 1 for neutrons and gamma rays), for the instrument reading then $S_n = S_\gamma$. The difficulty in achieving this result is inherent in the choice of liner. From the results given in section 6 (fig 6.13) the investigations have not produced a liner which would give these required response ratios.

Equation (8.24) can be further modified by using the relationships derived in (8.15), (8.16), (8.19) and (8.20) :-

$$QF\ q_{1\gamma}/q_{1n} = \frac{1 + C_{2/C} - Q_{2n}/Q_{1n} D_{2/D_1}}{1 + C_{2/C} - Q_{2\gamma}/Q_{1\gamma} D_{2/D_1}}$$

Substituting known values for Polythene/P.T.F.E. and Nylon/P.T.F.E. compensation devices and rearranging for $C_{2/C}$:-

$C_{2/C} = 1.061\ D_{2/D_1} - 1$ (for 5 Mev neutrons with nylon)

$C_{2/C} = 1.081\ D_{2/D_1} - 1$ (for 14 Mev neutrons with nylon)

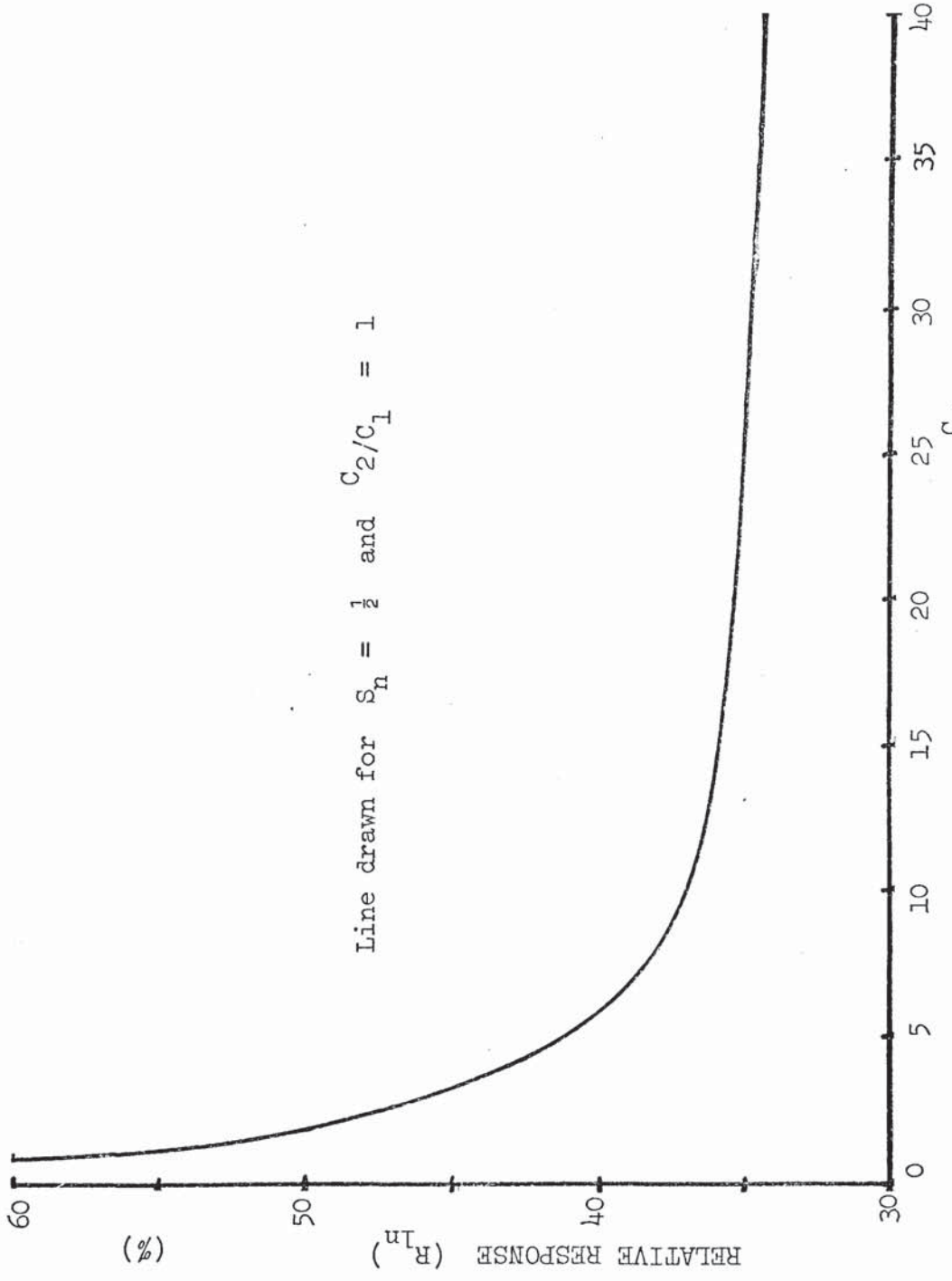
$C_{2/C} = 1.071\ D_{2/D_1} - 1$ (for 5 Mev neutrons with polythene)

$C_{2/C} = 1.075\ D_{2/D_1} - 1$ (for 14 Mev neutrons with polythene)

The last four equations constitute ....(8.25)

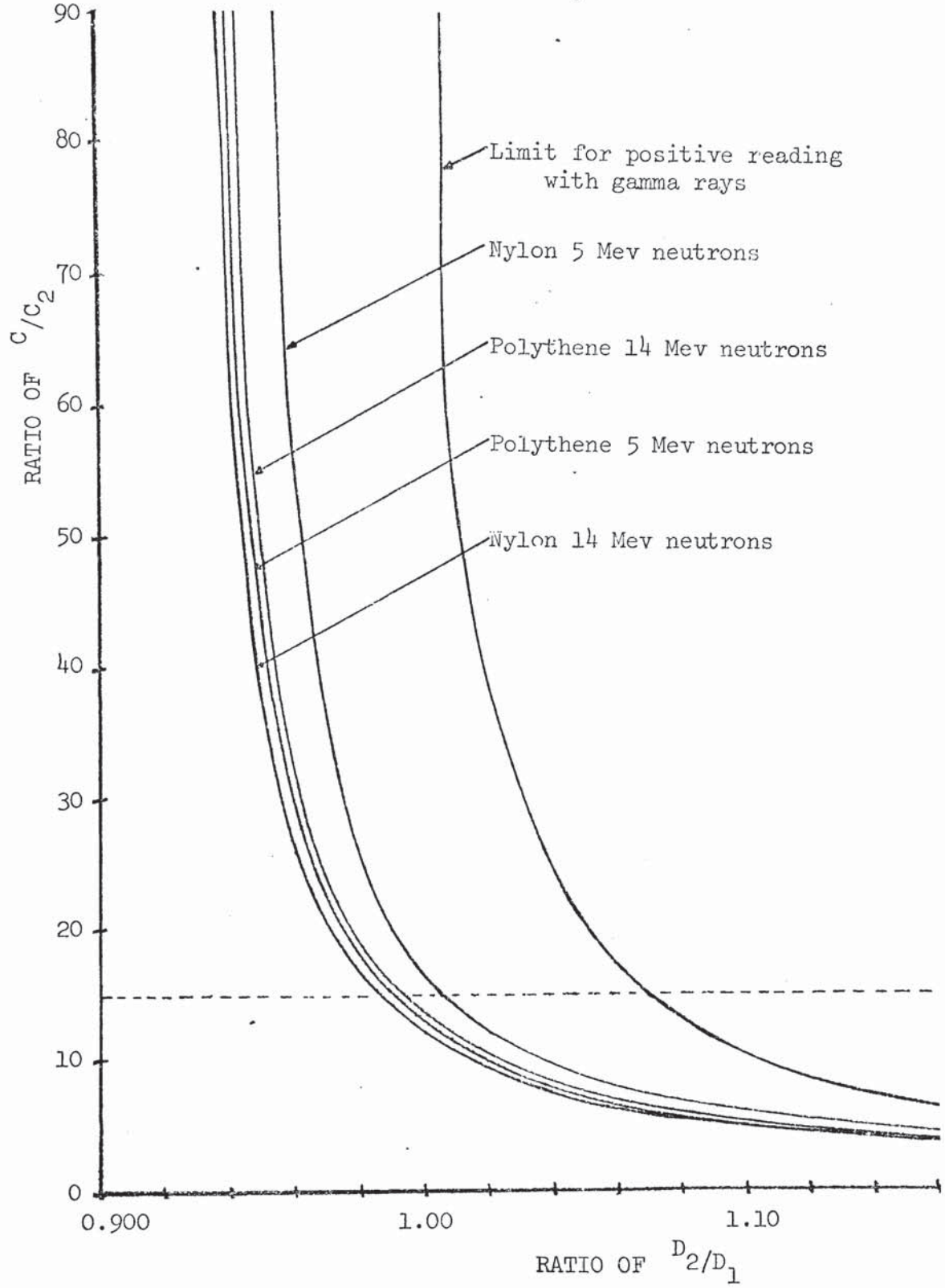
Values of $q_{1\gamma}/q_{1n}$ being found from fig 6.13. This gives a lower limit to D_2/D_1 of between 0.92 to 1.0 depending on which liner is used and what energy of neutron irradiation is considered. This equation (8.25) has much practical significance as it combines parameters involving both chambers and the linking capacitance of the compensation device. Fig 8.11 shows a graph of (8.25), with C_2/C against D_2/D_1 for various values of $q_{1\gamma}/q_{1n}$. (The vertical axis is presented as C/C_2 not C_2/C as it has more practical significance in this form). An upper limit to any choice of parameters is the line representing a value of Q.F. equal to infinity (ie $q_{1\gamma}/q_{1n} = 0$). If a combination of parameters is chosen above this line the net result will be to produce an instrument which gives an overall increase in charge (ie a negative reading) for gamma irradiation. This effect was obseved in section 9 for this reason. The minimum value of C/C_2 which can be reasonably achieved with a coaxial cylindrical arrangement, having a radius within $\pm$ 10% of that of the currently manufactured gamma ray dosimeter, is in the region of 15. Thus the horizontal dotted line drawn on the graph represents this "cut off" value. Therefore the region above this dotted line and to the left of the line for $q_{1\gamma}/q_{1n} = 0$, represents that combination of parameters that give dose equivalent measurements with any degree of accuracy.

Fig 8.10



Showing how response is less affected by large values of C/C_1.

Fig 8.11



Variation of C/C_2 against D_2/D_1 for polythene and nylon 66 at 5 and 14 Mev.

Also seen from (8.4) is that the value of C does not significantly affect the response of the instrument, unless C approaches the value of C_1 and C_2 . This can be more clearly demonstrated by reference to fig 8.10. When the capacitance ratio C/C_2 alters from 1 to 15 the response changes by approximately 25% , but from 20 to 40 the response changes by only 1.5% . As previously explained there is no likelihood of C/C_2 being less than 15, therefore this problem should not arise.

Lastly if any one of the equations in (8.25), say the equation for nylon with 5 Mev neutrons, is rearranged into the form :-

$$D_{2/D_1} = 0.94 \left(C_{2/C} + 1\right)$$

and this equation is then substituted into (8.4) :-

$$R_{1n} = \frac{0.623 \left(C_{2/C} + 1\right)}{1 + C_{2/C} + C_{2/C_1}}$$

From this equation it can be seen that as the value of $C_{2/C}$ increases so the value of R_{1n} decreases, for a constant value of C_{2/C_1} . Thus when any combination of parameters is chosen from fig 8.11 to operate a compensation device, it is advisable to select a value of $C_{2/C}$ which is as low as possible. (ie for higher relative sensitivity try and achieve a high value for C/C_2).

Another manipulation of the equations derived in this section is to obtain an estimation of the linking capacitance required for complete compensation in terms of that required for partial compensation. If C^* and C are the linking capacitance required for partial and complete compensation respectively, then from (8.4) and (8.11) :-

$$C^*/C_2 = \frac{(QF\ q_{1\gamma}/q_{1n} - 1)}{1 + QF\ q_{1\gamma}/q_{1n}\ (q_{2\gamma}/q_{1\gamma} - 1) - q_{2n}/q_{1n}}$$

Substituting the above in equation (8.14) :-

$$C^*/C = \frac{(QF\ q_{1\gamma}/q_{1n} - 1)(q_{2\gamma}/q_{1\gamma} - 1)}{1 + QF\ q_{1\gamma}/q_{1n}\ (q_{2\gamma}/q_{1\gamma} - 1) - q_{2n}/q_{1n}}$$

$$= \frac{1}{1 + \dfrac{1 - \left[\dfrac{q_{2n}\ q_{1\gamma}}{q_{2\gamma}\ q_{1n}}\right]}{(QF\ q_{1\gamma}/q_{1n} - 1)(1 - q_{1\gamma}/q_{2\gamma})}}$$

The boundary conditions for (8.26) are :-

$$\frac{q_{2n}\ q_{1\gamma}}{q_{2\gamma}\ q_{1n}} = \frac{Q_{2n}\ Q_{1\gamma}}{Q_{2\gamma}\ Q_{1n}} = 0.4 \quad \text{(from equation 8.21)}$$

$$q_{1\gamma}/q_{2\gamma} = Q_{1\gamma}/Q_{2\gamma}\ D_1/D_2 = D_1/D_2 \qquad (Q_{2\gamma} = Q_{1\gamma})$$

$$q_{1\gamma}/q_{1n} = Q_{1\gamma}/Q_{1n} = 1.67 \quad \text{(for nylon from fig 6.13)}$$

From fig 1.8 a Q.F. of 7 is used for 5 Mev average energy neutrons. Therefore substituting these values in (8.26) :-

$$C/C^* = 1 + \frac{0.056}{1 - D_1/D_2} \qquad \ldots\ldots(8.27)$$

From inspection of (8.27) it can be seen that the value of D_1/D_2 only significantly affects the equation when D_1 approaches the value of D_2 .

8.5 Substitution of derived results

In the development of any device it is often desirable to know, when the instrument is aimed to provide for one special aspect of a market, what particular performance range the instrument can be expected to cover. From consideration of the equations derived in this section an estimation can be made of the overall sensitivity. From fig 6.13 the ratio of neutron sensitivity of a nylon liner to that of a polythene liner is 0.82 for 5 Mev average energy neutrons. From fig 7.4 the sensitivity of a polythene/graphite lined chamber is approximately 0.0077 divisions per millirem. (130 millirem per division). Therefore with a commercial dosimeter containing a nylon/graphite liner the sensitivity would be in the region of 0.0063 divisions per millirem. (158 millirem per division).

To discharge the instrument over the full 20 divisions (F.S.D.) would therefore require 2.6 rem for the polythene/graphite liner and 3.2 rem for the nylon/graphite liner.

This can now be related to an estimate of the actual sensitivity of a compensation instrument. To achieve this estimate the relative response of the instrument is calculated from (8.4) and (8.20), using values of D_1/C_1 and D_2/C_2 that could be reasonably be expected in a compensated practical device. The average size of the sensitive volume in a quartz fibre condenser ionisation chamber is 2 cc and the typical capacitance is in the region of 1 pF. From charge to discharge (ie one F.S.D.) takes about 70 volts. Therefore from fig 8.11 a value of D_2/D_1 must be chosen which results in a reasonably high sensitivity and a value of C/C_2 chosen which is above the dotted line. Therefore if $D_2/D_1 = 0.97$, C/C_2 is approximately 23. From inspection of (8.4) it can be seen that for a high sensitivity C_2/C_1 should be as small as practically possible. Therefore $C_1 = 1$ pF and C_2 is taken as 0.7 pF. Thus from equation (8.4) the relative neutron response would be :-

$$R_{1n} = 40\% \quad (0.396)$$

and from (8.22) for a device with total gamma ray compensation :-

$$R_{1n} = 35\% \quad (0.348)$$

Therefore for a polythene/graphite liner :-

R_{1n} = 6.4 rem for F.S.D. (with partial gamma compensation)

R_{1n} = 7.9 rem for F.S.D. (with total gamma compensation)

and for a nylon/graphite liner :-

R_{1n} = 8.0 rem for F.S.D. (with partial gamma compensation)

R_{1n} = 9.7 rem for F.S.D. (with total gamma compensation)

This is on average about 40 times less sensitive than the highest sensitivity gamma ray commercial dosimeter.

8.6 Estimated dimensions

Using the ideas developed in section 8.4 of volume and capacitance ratios, it would be useful to estimate the actual dimensions that could be expected in a commercial dosimeter for neutrons and gamma rays employing the compensation principle. The following serves to illustrate one method of achieving this result.

As previously stated the size and associated capacitance of a typical commercial dosimeter sensitive volume is approximately 2 cc and 1 pF respectively. The relationships given in section 8.4 of $D_2/D_1 = 0.97$, $C/C_2 = 23$ and $C_2/C_1 = 0.70$ are used for this exercise. Therefore :-

If $D_1 = 1.90$ cc, $D_2 = 1.84$ cc.

If $C_1 = 1.0$ pF, $C_2 = 0.70$ pF and $C = 33$ pF.

Therefore using the graph in fig 8.12, for the neutron sensitive chamber :-

$D_1/C_1 = 1.90$ giving $a/b = 0.24$ and $P = 0.49$ pF / cm

For the neutron insensitive chamber :-

$D_2/C_2 = 2.60$ giving $a/b = 0.14$ and $P = 0.30$ pF / cm

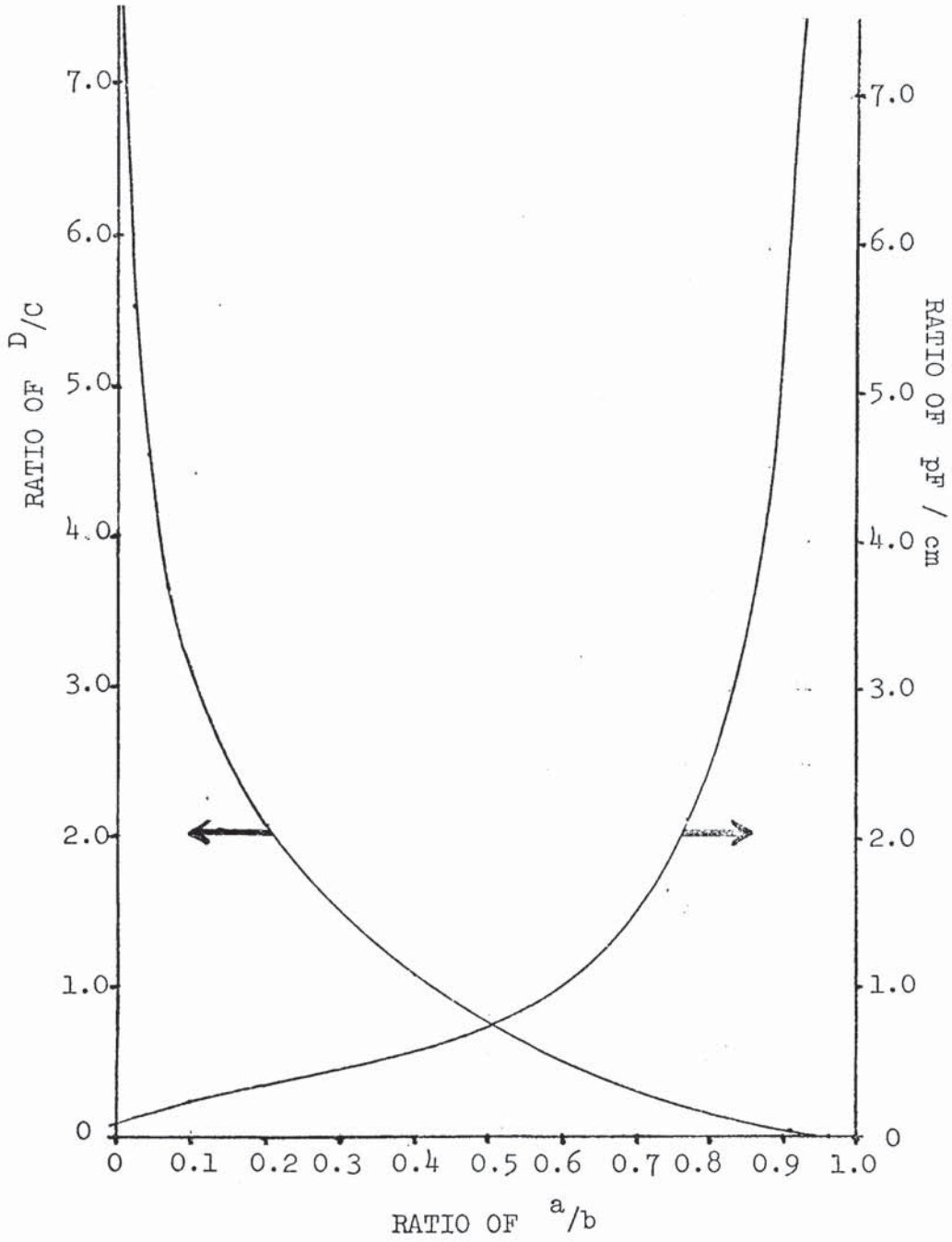
Therefore using these values of P produces a neutron sensitive chamber of 2.04 cms and a neutron insensitive chamber of 2.33 cms in length.

The linking capacitance, estimated from the graph in fig 8.11, is about 33 pF. This capacitance is produced across the P.T.F.E. liner. (The liner acting as a dielectric). Therefore the thickness of P.T.F.E. required to achieve this capacitance can be expressed as a ratio of b/a , found from a knowledge of P^* and then employing fig 8.12. P^* is calculated from the length of the neutron insensitive chamber and the assumption that the dielectric constant of P.T.F.E. is 2. Therefore :-

$P^* = 7.08$ pF / cm giving $a/b = 0.92$ and $b/a = 1.087$

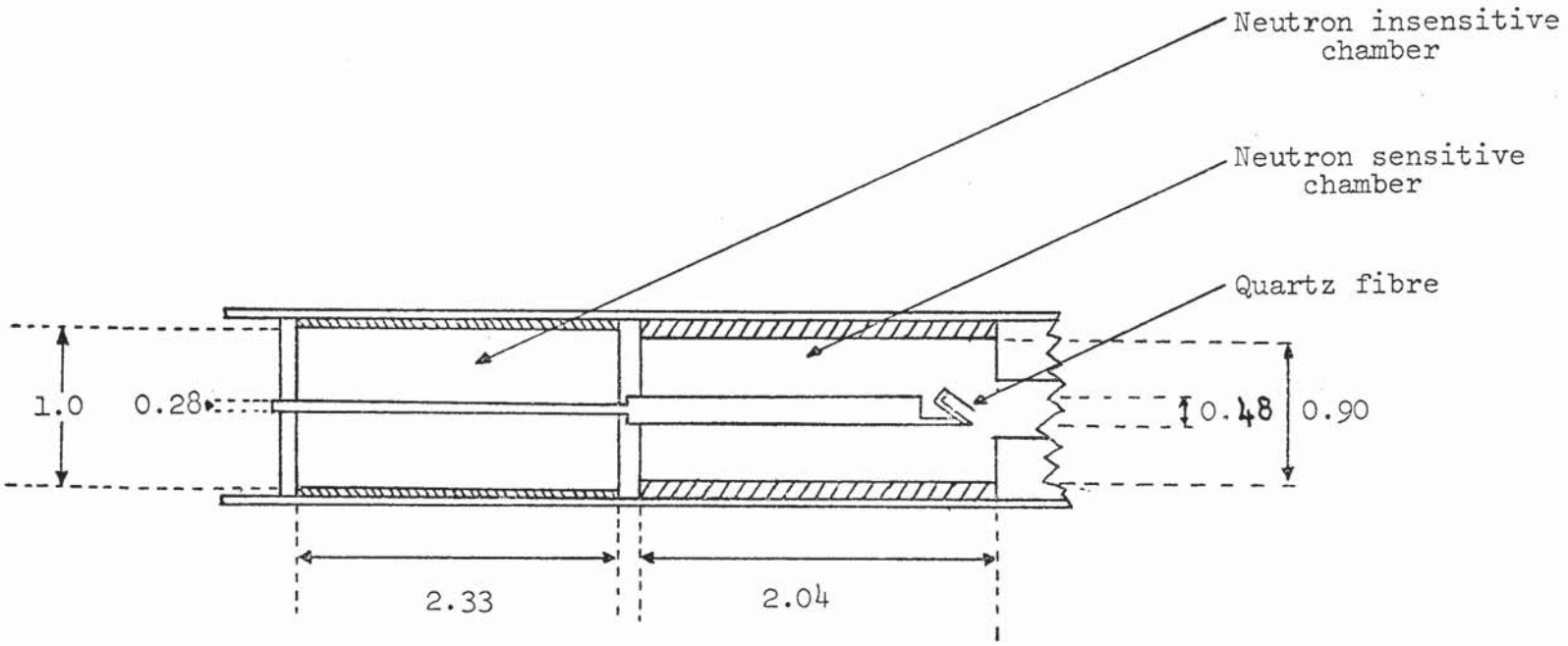
Therefore if $a = 0.500$ cm, $b = 0.543$ cm . The liner must therefore be 0.043 cm thick (0.017 inches). The thickness for the liners used in the compensation device in section 9 was 0.10 cm.

Fig 8.12



Graph of $^{D}/_{C}$ and pF / cm against $^{a}/_{b}$.

In fig 8.13 the relevant dimensions for the compensation device given in this section are drawn. These dimensions are all based on the assumption of 1 cm for the internal diameter of the neutron insensitive chamber.



All dimensions in cms

Estimated dimensions of a quartz fibre compensation device.

Fig 8.13

8.7 Glossary

a = External diameter of central electrode.

b = Internal diameter of outer electrode.

C_1 = Capacitance of the neutron sensitive chamber in the compensation device.

C_2 = Capacitance of the neutron insensitive chamber in the compensation device.

C = Linking capacitance in the compensation device.

C^* = Linking capacitance required for partial compensation of the gamma radiation in the compensation device.

δV_1 = Voltage change on C_1 due to a given radiation dose.

δV_2 = Voltage change on C_2 due to a given radiation dose.

δV_{1n} = Voltage change on C_1 due to a given neutron dose.

$\delta V_{1\gamma}$ = Voltage change on C_1 due to a given gamma ray dose.

δV_{2n} = Voltage change on C_2 due to a given neutron dose.

$\delta V_{2\gamma}$ = Voltage change on C_2 due to a given gamma ray dose.

δV_1^* = Net change in potential on C_1 due to the compensation effect, after a given radiation dose.

δV_{1n}^* = Net change in potential on C_1 due to the compensation effect, after a given neutron dose.

$\delta V_{1\gamma}^*$ = Net change in potential on C_1 due to the compensation effect from a given gamma ray dose.

D_1 = Volume of the neutron sensitive chamber (C_1).

D_2 = Volume of the neutron insensitive chamber (C_2).

K_1 = C_1/C.

K_2 = C_2/C.

P = Capacitance expressed in terms of chamber length in pF / cm.

P^* = Linking capacitance expressed in terms of chamber length in pF / cm.

q_{1n} = Alteration in charge on C_1 due to a given neutron dose.

q_{2n} = Alteration in charge on C_2 due to a given neutron dose.

$q_{1\gamma}$ = Alteration in charge on C_1 due to a given gamma ray dose.

$q_{2\gamma}$ = Alteration in charge on C_2 due to a given gamma ray dose.

δq = Compensating charge flow due to a given radiation dose.

Q_{1n} = Constant for neutron sensitive chamber under neutron irradiation.

Q_{2n} = Constant for neutron insensitive chamber under neutron irradiation.

$Q_{1\gamma}$ = Constant for neutron sensitive chamber under gamma irradiation.

$Q_{2\gamma}$ = Constant for neutron insensitive chamber under gamma irradiation.

QF = Quality factor.

R_1 = Relative response of neutron sensitive chamber (as defined in equation 8.2).

R_{1n} = Relative response of the neutron sensitive chamber to neutrons.

$R_{1\gamma}$ = Relative response of the neutron sensitive chamber to gamma rays.

S = Relative sensitivity of both chambers when not connected in a compensation circuit.

S_n = Relative sensitivity of both chambers to neutrons.

S_γ = Relative sensitivity of both chambers to gamma rays.

Section 9 Compensation Device Testing

9.1 Introduction

Initial investigations completed into the theory of compensation, as given in section 8, suggested that from a practical viewpoint the idea of compensation was feasible. However the results obtained with the first chambers were unsatisfactory due to their overall inconsistency. It was only after investigations had been completed with the liner chambers (see section 6) that the reason for this inconsistency became apparent. The perspex plates of the first compensation device were made very thick to withstand the high pressures experienced within the sensitive volume during gas filling investigations. Consequently there was a considerable volume of perspex around the centre electrode, causing a considerable "soakage" effect. (Soakage is defined in section 6.2.2). Therefore as a completion to the project a further compensation device was designed to reduce the amount of this "soakage" effect on the dosimeter behaviour. Further modifications were also added to reduce electrical leakage and remove end electrode "free" air (explained in section 6.2.3) ionisation effects. The irradiations were divided into two stages :-

(a) 5.1 Mev average energy neutrons from source N 2.

(b) Gamma rays from source G 3.

9.2 Construction of the Compensation Device

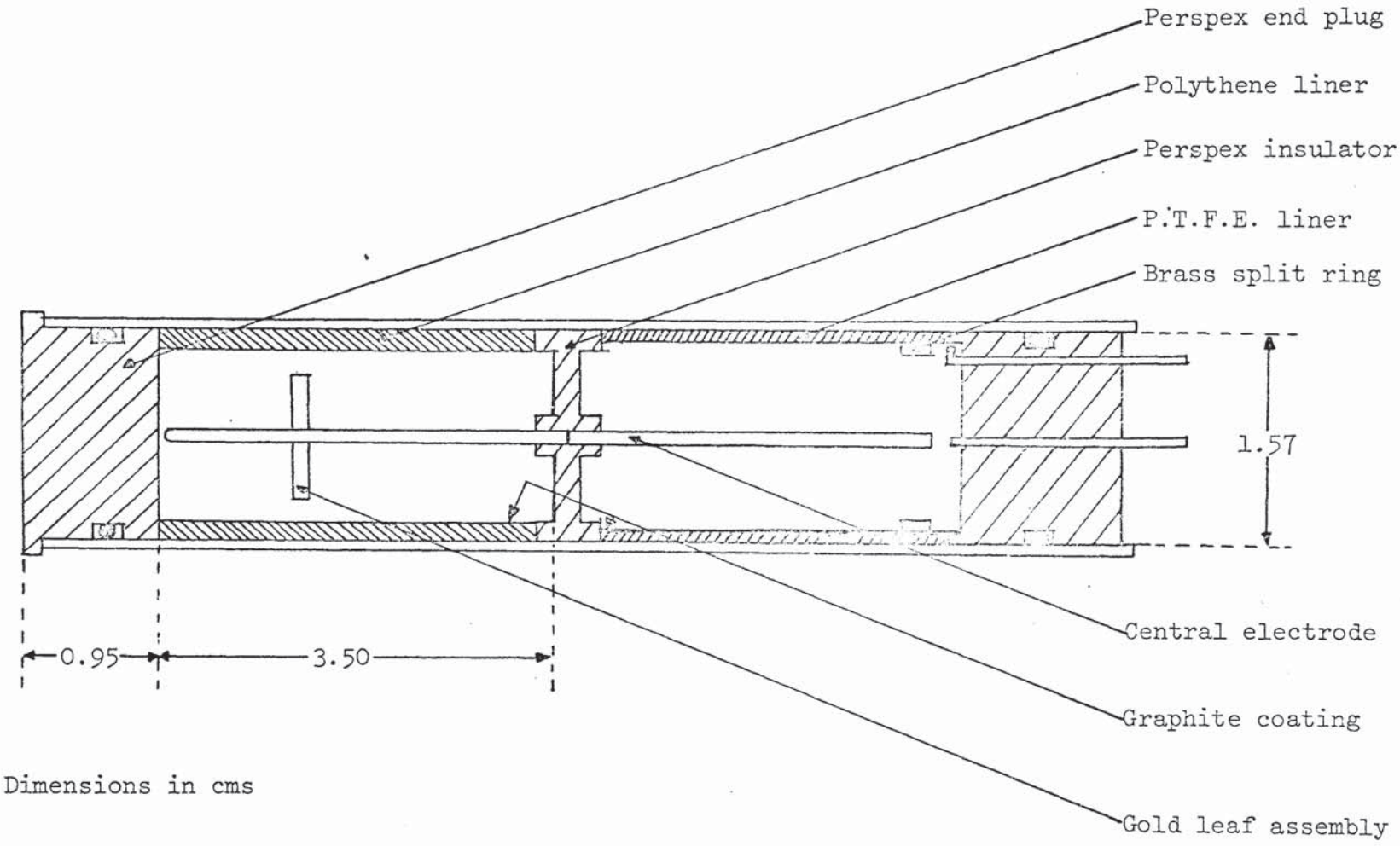
Two instruments of the same design were constructed. Both instruments possessed the basic requirements of two chambers, one sensitive to neutrons and gamma rays the other insensitive to neutrons, for the compensation principle. The volume and capacitance ratios of the chambers, together with the linking capacitance to neutron insensitive chamber capacitance ratio, was chosen from the requirements and procedure given in section 8. The devices were constructed of aluminium with a central brass electrode. The division between the two chambers was achieved by a central perspex insulator which also acted as a brass electrode support (see fig 9.2). The complete device is given in fig 9.1.

9.2.1 Gold leaf

The procedure adopted for producing and selecting a gold leaf was exactly the same as that given in section 6.2.1 . It was possible however with improved technique to obtain a much smaller, and hence more sensitive, gold leaf than used in previous experiments. The size used in this device was in the region of 0.5 cms length and 0.15 cms width.

9.2.2 Perspex Insulator

It was fortunate that many of the problems associated with the construction of an insulator for this compensation device (based on the ionisation chamber) had been experienced during development



Dimensions in cms

Assembled compensation device

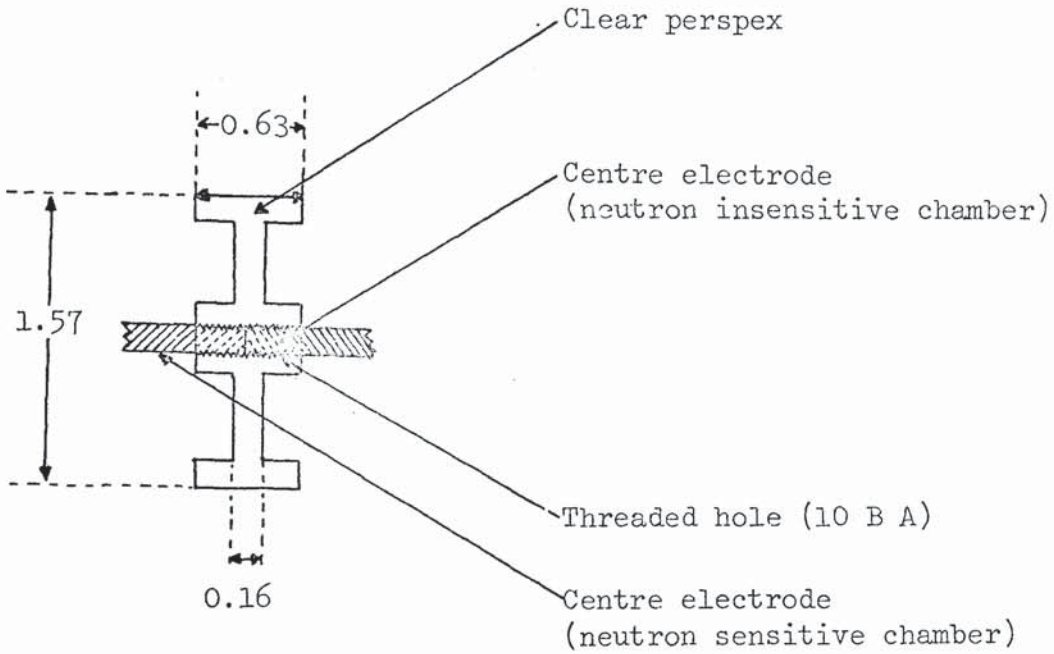
Fig 9.1

of the liner chambers. The requirements were for a high resistance insulator of good mechanical strength sufficient to support two electrodes. The electrodes were placed on either side of the insulator and held in position by a 10 BA threaded central hole at the centre of the insulator. Therefore a further requirement was the ability of the insulator to support such a small thread without small pieces breaking off when the electrodes were screwed into position. A diagram of the insulator is given in fig 9.2 . The insulator must also allow passage of light for observation of the gold leaf and provide an impermeable boundary between the two chambers. An initial attempt was made to construct the insulator of polystyrene. This was found to be unsatisfactory as besides being difficult to machine, stress cracks occured around the centre. Therefore perspex was again used and this proved to be satisfactory. Before installation the insulator was highly polished to reduce electrical leakage and light absorption.

9.2.3 Charging mechanism

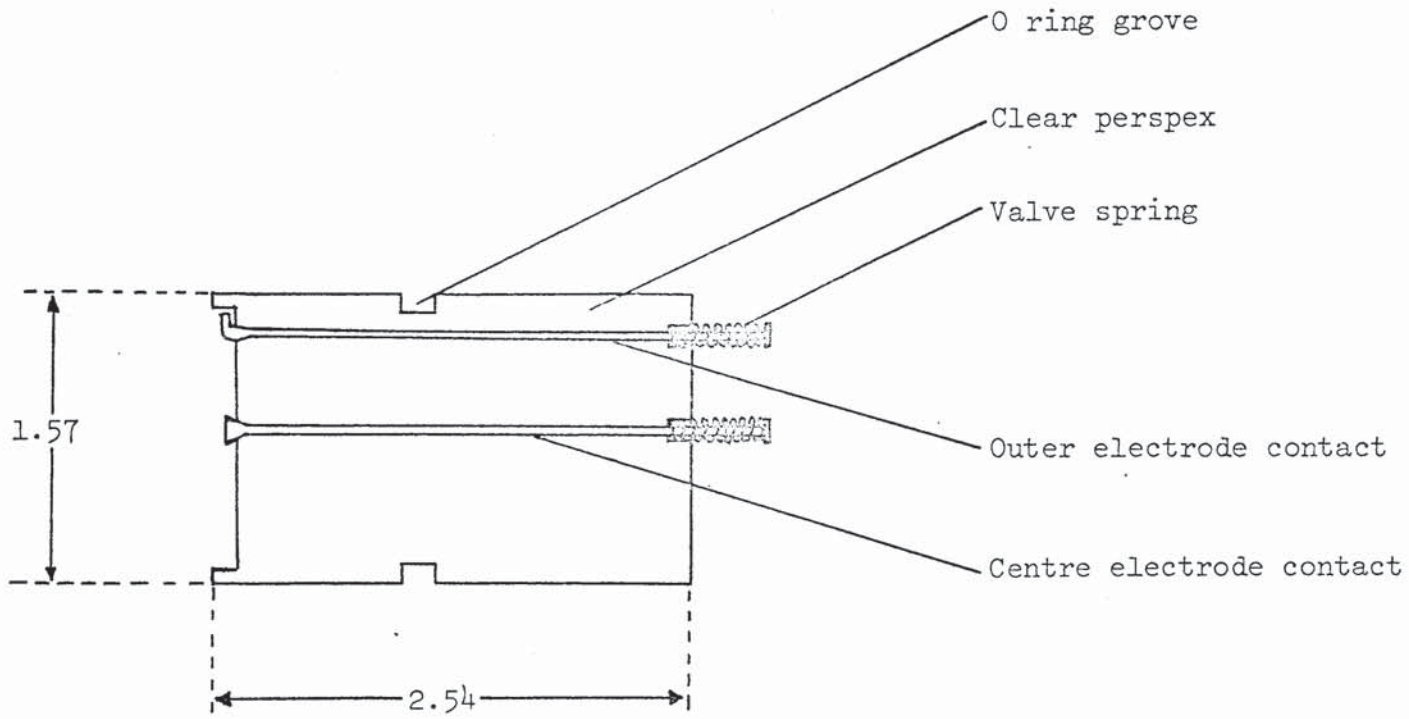
It was not certain how much the "free air" ionisation effect contributed to the overall sensitivity of the liner chambers, but it was decided with this compensation device to eliminate the effect. Therefore a perspex end piece was constructed, as in fig 9.3, which contained two spring loaded charging pins. When depressed the centre pin made contact with the centre electrode and the other pin made contact with the outside wall electrode of the neutron insensitive chamber.

Fig 9.2



Dimensions in cms

Compensation device insulator



Dimensions in cms

Compensation device charging mechanism.

Fig 9.3

As this end piece was no longer a centre electrode support (as with the liner chamber) it was now possible to construct the mechanism from a large volume of perspex. This had the advantage of not splitting when positioned in the tube from pressure at the region of the "O" ring seal. The "O" ring had the ability of rigidly positioning the charging mechanism as well as sealing the sensitive volume. Obviously with the electrodes having no direct external access to the air, there was no charge lost by the electrodes on this account. There would be charges induced on the pins of the charging device, but it was found in practice that these were negligible.

9.2.4 Perspex plug

This also had an "O" ring fitting (size O S 9), and was required to have a good optical finish to allow observation of the gold leaf without undue distortion of the gold leaf. As in the case of the liner chambers the aluminium tube forming the body of the instrument had a small hole drilled through it, just below the end of the perspex plug. This removed any pressure build up which would occur when the plug was positioned. The hole was then sealed after positioning of the plug, so that the device contained dry air at atmospheric pressure.

9.2.5 The liners

From the results obtained with the liner chambers, polythene and P.T.F.E. were chosen as the liners. Polythene was chosen in preference to nylon as the former had greater sensitivity to

5 Mev neutrons. (The investigations were conducted with this lower energy and not at the 14 Mev neutron energy.)
The polythene liner was 0.15 cms (0.060 inches) thick and the P.T.F.E. liner was 0.10 cms (0.040 inches) thick, both liners being cut from flat sheet and bent with the aid of heating to the required cylindrical shape. The conductive coating was applied by a procedure similar to that described in section 6.4.
The P.T.F.E. chamber had a brass split ring positioned against the conducting surface . This gave the charging pin a region of good electrical contact when the pin was depressed during charging of the outer electrode.

9.3 Method of Assembly

Due to the more complicated construction of this compensation device compared to that of the liner chambers, difficulty was experienced in the assembly of the device. However various methods of assembly were tried and the one finally adopted is given below :-

(a) The P.T.F.E. liner (neutron insensitive) was positioned in the aluminium case and the brass split inserted.

(b) The central insulator electrode support was positioned and the neutron insensitive chamber centre electrode srewed into the insulator.

(c) This electrode was then adjusted for length, until it was of such a length to give sufficient clearance to the central pin of the charging mechanism.

(d) The charging mechanism was positioned and a continuity check taken for both pins. So that when one of the spring loaded pins was depressed it made electrical contact with its corresponding electrode.

(e) The gold leaf electrode (for the neutron and gamma ray sensitive chamber) was then positioned and an electrical continuity check made between it and the neutron insensitive chamber electrode.

(f) Finally the perspex plug was pushed into position and the pressure escape hole sealed.

9.4 Calibration of the Compensation Dosimeter

As the device had a much smaller gold leaf than the previous instruments it also had the advantage of requiring a lower initial charging voltage. It was found that the gold leaf would obtain the desired starting position (roughly $\frac{3}{4}$ of the maximum possible swing) with only 45 volts on the centre electrode. This compared favourably with the 80 to 90 volts of the liner chambers and 150 volts for the initial compensation device (see section 8). A lower charging voltage on the centre electrode meant lower electrical leakage, but more important it kept the voltage on the outer electrode of the neutron insensitive chamber at around 70 to 80 volts thus reducing electrical leakage here as well.
It is important that the outer electrode should have a reasonably low voltage because there is a much smaller leak path (from the high voltage to earth) in this case than for the centre electrode.

From measurements taken on both instruments (designated A and B) it was found that the electrical leakage from the outer electrode was a factor of 5 greater than that from the centre electrode. However the electrical leakage was found to be much improved on that of previous devices and only in the region of 2 volts per hour. As long as the outer electrode did not receive or lose any charge the deflection of the gold leaf was unaffected. Therefore the voltage calibration of the gold leaf was completed with the outer electrode shorted to the aluminium case. The voltage required to cover the "working" range of the device was from 30 up to 45 volts. Therefore the leaf was calibrated over this voltage region, the calibrations were taken before and after the series of irradiations.

9.5 Charging procedure

With this compensation device it was found that a complication was added to the charging procedure by the capacitive effect across the charging pins and their respective electrodes. To prevent damage to the instrument and obtain a meaningful reading a strict sequence of events had to be observed during charging of the instruments. The sequence used in as follows :-

(a) Earth centre electrode.

(b) Charge outer electrode to 80 volts.

(c) Earth outer electrode charging pin by a brief touch with an earthed probe.

(d) Charge the inner electrode to 40 volts.

(e) Earth inner electrode charging pin by a brief touch with an earthed probe.

(f) Repeat (b), (c) , (d) and (e).

The main problem was caused by the fact that with this device only one electrode could be charged at a time. Therefore it was found necessary to observe this sequence as if say (b) and (d) were reversed the leaf would be likely to suffer damage because the excess charge would cause the leaf to reach the earthed outer electrode of the neutron sensitive chamber. If this were repeated too often damage would be caused to the end of the gold leaf, rendering it useless for microscopic observation. This problem could be overcome in a commercial instrument by ensuring that both electrodes were charged simultaneously, thus preventing excess charge build up on the centre electrode. Or alternatively it would be possible to arrange the values of the capacitances in such a way that, by charging the outer electrode the correct charge is induced on the central electrode. Thus only one external charging operation is required to achieve the correct voltages on the electrodes. Sequence (e) was necessary because the charging pins being exposed to air, would lose their charge far more rapidly than the internal electrodes. This would adversely affect the charge induced on the capacitance between the end of the inner electrode and the charging pin. Thus if the pins are earthed before irradiations commence this effect does not interfer with the readings on the instrument. A similar procedure is adopted on the charging bellows of present day commercial dosimeters.

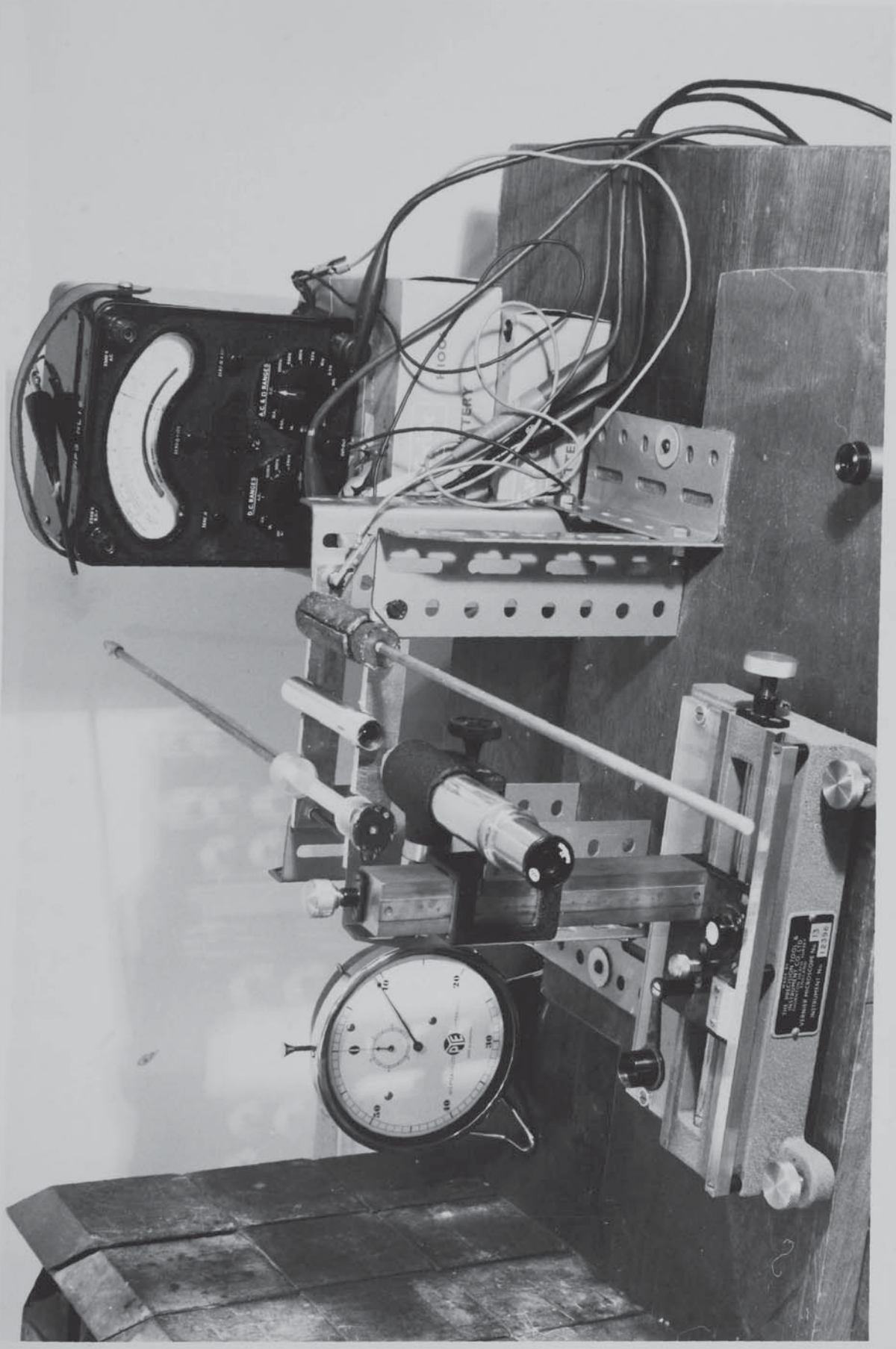
9.6 Irradiation procedure

(a) Source N 2 Both compensation devices were charged according to the procedure given in section 9.5 . The devices were then irradiated for ten minute periods until the leaf collapsed beyond the linear response region. The position of the leaf was recorded after each irradiation by a travelling microscope.

(b) Source G 2 Again both devices were charged according to the procedure given in section 9.5 . They were then irradiated for twenty minute periods and the leaf observed as in (a) for source N 2. The increased irradiation time was to offset the difference in absorbed dose rate from N 2 and G 2. By doubling the exposure time for G 2 the absobed dose rates were made approximately equal. This was permissible as the electrical leakage rates were comparatively small. As an alternative to the above alternate neutron and gamma ray irradiations were taken. A sequence was taken of neutron N 2 for ten minutes followed by G 2 for twenty minutes.

(c) Source N G 1 This was basically a combination of N 2 and G 2. One compensation device was positioned centrally between the 300 millicurie neutron source (americium/beryllium) and the 50 microcurie cobalt 60 source, (See fig 9.4). Readings were then taken for a ten minute period. The charging and observation procedure were as given in (a) for source N 2. The object of this irradiation was to verify the claim that the compensation device would be able to measure neutron plus gamma radiation in their respective dose equivalent proportions . It was found

Fig 9.4



the results taken from the combined irradiation were approximately the same as the total of the readings obtained with each source taken separately for a ten minute period. However due to the somewhat imprecise nature of this measurement (ie a proper jig was not used) these results are not included. A table of results for radiation procedure (a) and (b) is given in fig 9.7 .

9.7 Experimental considerations

It is worthwhile before proceeding to analyse the results to consider three parameters which govern the performance of a condenser ionisation device. These parameters also have special significance for the compensation device tested in this section.

9.7.1 Electrical leakage

Electrical leakage from the outer electrode is a very significant factor affecting the performance of the compensation device. As explained previously, the construction of the outer electrode is basically a P.T.F.E. dielectric (which also serves as a neutron insensitive liner) contained within cylindrical plates, which can be approximated as plane parallel plates. One plate is the aquadag coating (at approximately 80 volts) and the other plate is the aluminium case of the device (at earth potential). To achieve similar capacitance for the neutron sensitive and the neutron insensitive chamber, the aquadag coating on the P.T.F.E. (neutron insensitive) liner covered the whole of the inner surface of the liner. Similar capacitances were desirable to

enable the volumes of the two chambers to be nearly equal. (As requested by the theory given in section 8). This was necessary as the diameter of the two centre electrodes was the same for both chambers. However a direct result of coating the whole of the inner surface of the chamber with aquadag was to create a relatively short leakage path for the charge on the inner surface. Obviously this charge would leak at a rate that was in proportion to the voltage on the outer electrode. So as the voltage of the outer electrode fell during irradiation the electrical leakage rate would also drop and so the compensation rate could be affected. Therefore directly from fig 8.11, values of D_1 and D_2 were chosen such that with C/C_2 at approximately 30 the gamma ray readings would be negative, (ie an increase in deflection of the gold leaf) assuming that there was negligible electrical leakage. This choice of parameters thus helped to offset the "positive" electrical leakage and produce approximately the correct value for neutron to gamma ray sensitivity ratio. The leakage rate, as explained later, was estimated from a knowledge of the resistivity of P.T.F.E. From physical considerations of the behaviour of the device it can be seen that any leakage from the outer electrode is fed back to the neutron sensitive chamber, but this is not of a sufficient value to lower the potential on both chambers by an equal amount. This, together with the probability of there also being a non saturation current effect, would create an undesirable large gamma ray sensitivity. Thus the neutron to gamma ray sensitivity ratio would be too small (ie well below 7) if the correct theoretical value of D_2/D_1 and C/C_2. , taken from

fig 8.11 , had been chosen.

It was found that the electrical leakage for the centre electrode charged to 45 volts was negiligible compared to the leakage observed when the centre electrode was at 45 volts and the outer electrode, instead of being earthed, was at 80 volts. It is therefore reasonable to suppose the electrical leakage effect was due to the outer electrode. The electrical leakage measurements, given as an average of six readings, are shown in fig 9.7 .

9.7.2 Nor saturation current effect

As the compensation device was operated at low voltages (40 to 45 volts, for voltage across each of the chambers) there was a probability that the ionisation current was being collected under non saturation conditions (see fig 3.4) . The net result of this effect would be to create a variation in the compensation rate of the device with irradiation time. Sensitivity to neutron irradiation would fall with irradiation time and sensitivity to gamma rays may produce a negative reading. The reason for this is that the neutron sensitive chamber discharges at a faster rate than the neutron insensitive chamber. For any given radiation dose, the voltage across the neutron sensitive chamber would be falling relative to that across the neutron insensitive chamber. Therefore the chambers would be operating at continually changing positions on the saturation curve, thus continually altering the current to each capacitance. This produces an alteration in the compensation rate of the device. Therefore if this was the

case in practice, then the sequence in which the irradiations were alternated would be of significance. Thus for comparable positions of the gold leaf a series of gamma only irradiations would show differing results than those from a series of alternate neutron and gamma irradiations. It was evident during the tests that this effect was occuring. Therefore an attempt was made measure this alteration in saturation current and determine at which point along the saturation curve the irradiations were being taken. A vibrating reed electrometer was used, with a claimed current measure down to 10^{-17} amps. However it was found that this instrument was very prone to radiation effects and insufficient shielding could be given to the connecting leads to make the measurements valid.

9.7.3 Capacitance

As stated previously the prime purpose of this particular series of investigations was to prove the feasibility of the compensation principle and not to achieve exact neutron and gamma ray dose equivalent response. However to achieve aneutron to gamma ray sensitivity ratio that was reasonably close to the Quality Factor, allowance had to be made for the electrical leakage from the outer electrode. Therefore with a knowledge of the thickness of the P.T.F.E. liner and the resistivity of P.T.F.E., an estimation of the leakage was made (see later). With reference to fig 8.11 , a volume ratio of D_2/D_1 equal to 1.10 was chosen. This would, as explained previously, give an instrument with a "negative" response for gamma radiation. Thus with a capacitance for each

chamber of around 1 pF , the linking capacitance, governed by the coaxial design of the device, would be approximately 35 pF. It was found from the results that even though this estimate was of an approximate nature it did produce a neutron to gamma ray sensitivity ratio of a realistic value (ie around 7). These results are given in fig 9.7 .

The two chambers of the compensation device differed in three respects :-

(a) The neutron sensitive chamber contained the thicker liner. (0.15 cms compared to 0.10 cms).

(b) The neutron sensitive chamber contained the gold leaf assembly.

(c) The neutron insensitive chamber contained the brass split ring (for charging of the outer electrode).

Thus calculations of capacitance would be difficult for C_1 and C_2 , therefore it was decided to take capacitance measurements with a Wayne-Kerr bridge. The calculated and observed values of capacitance are given in fig 9.5 . This shows that the important capacitance ratio C/C_2 is as expected in the region of 35 . The capacitance of C_1 and C_2 is also as predicted in the region of 1 pF.

The following procedure was adopted for the measurements taken with the Wayne-Kerr bridge :-

The springs were removed from the charging pins and a lead soldered to each pin, so that direct contact could be made to the

bridge electrodes. After balancing the bridge, so that the capacitive effect of the charging pins within the perspex end piece was taken into consideration, the following measurements were taken :-

(a) Capacitance C. For the linking capacitance the centre electrode was shorted to the case and measurements taken as in fig 9.6a. Thus the value of capacitance taken was C in parallel with C_2 .

(b) Capacitance C_1. Here the linking capacitance was shorted to the case and measurements taken as in fig 9.6b. Thus the value of the capacitance measured was C_1 in parallel with C_2 .

(c) Capacitance C_2. In this case the central electrode of the neutron sensitive chamber was removed (effectively removing C_1) and measurements taken as in fig 9.6c. C as in procedure (b) remaining shorted to the case.

Calculated and observed values of capacitance are given in fig 9.5.

9.8 Analysis of results

A table of results for the two compensation devices tested in this section is given in fig 9.7 . The first problem that arose with these measurements, (compared to those of the liner chambers) was that the irradiations were for a much longer period of time. The reason for the increased irradiation time was to enable a significant gold leaf deflection to be attained. The principle of compensation lowers the overall sensitivity, (see section 8)

Fig 9.5

Calculated and observed values of compensation device capacitance

Capacitance	Calculated	Observed	
		A	B
C	35.1	34.8	36.7
C_1	1.00	1.16	1.19
C_2	0.98	1.18	1.07

Units are in pF.

Devices A and B were as near as possible of identical construction.

Fig 9.6a

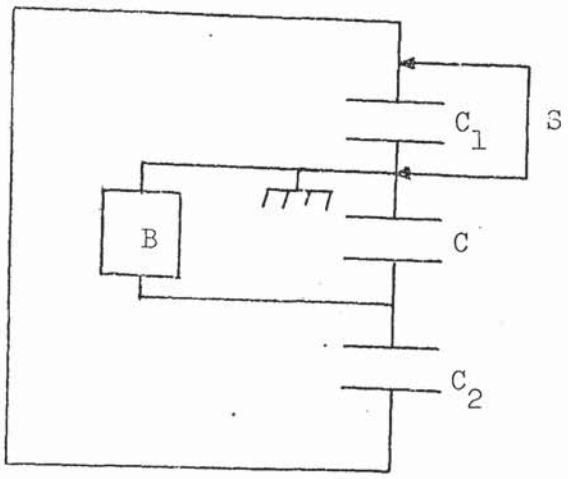


Fig 9.6b

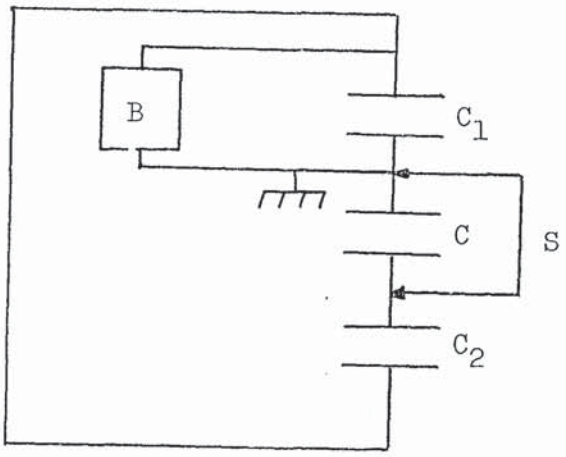
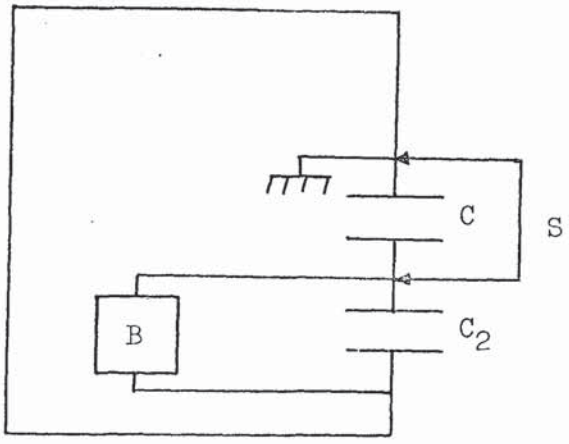


Fig 9.6c



B = Wayne-Kerr capacitance bridge.
S = Shorting connection.

Circuits for compensation device capacitance measurements.

so this was the only method readily available for increasing gold leaf deflection. As previously stated, to enable the absorbed dose to be approximately equal for both irradiations, the neutron irradiations (10 mins) were half the time of the gamma irradiations. This compares with the two minute irradiation period of the liner chambers. One consequence of this longer irradiation time was to increase to a significant amount the effect of electrical leakage. Therefore it was also found necessary to include electrical leakage results in fig 9.7 . The average value of leakage of both instruments was assessed and a straightforward subtraction made from the observed results. It was found that on average the uncorrected results for the neutron and gamma ray measurements were in the correct ratio (as calculated) for dose equivalent comparisons to be made. As can be seen from the graph in fig 9.8 , the neutron to gamma ray ratio is fairly well maintained until the voltages have fallen by about 30% . At this point the electrical leakage and/or the non saturation current effect significantly alter the compensation rate and thus result in a lowering of neutron sensitivity and an actual negative reading for gamma ray sensitivity. However this situation was predicted from initial calculations. Thus the results in fig 9.7 show that if the correct value of chamber volume ratio and linking capacitance ratio were chosen and electrical leakage was significantly reduced, the instrument could give a reading directly related to rems.

Evaluation of the time constant of the instruments, using the

Fig 9.7

Sensitivity results for the Compensation Devices

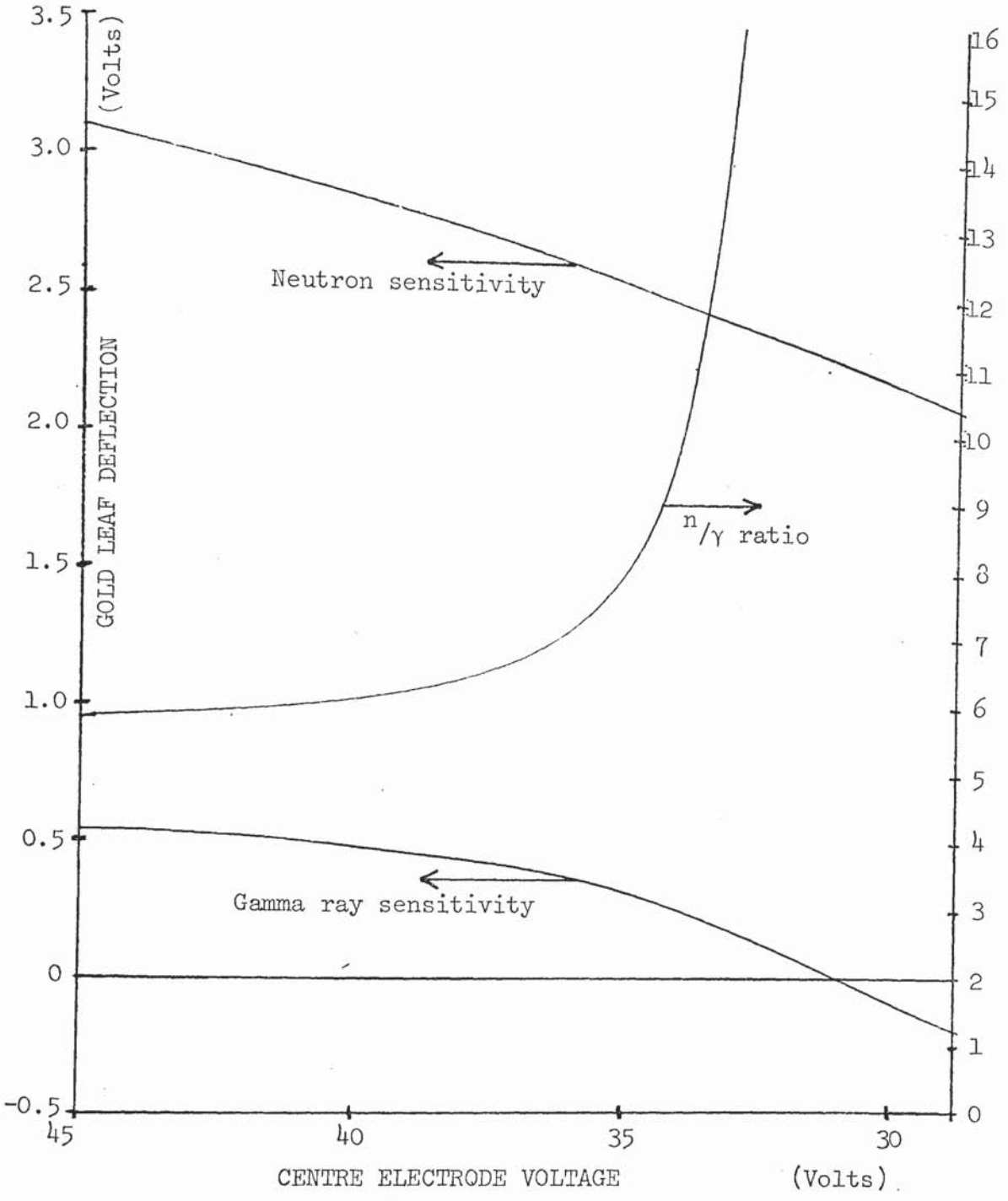
Measurement	Device		
	A	B	Average
Neutrons (n)	2.69	2.93	2.81
Gamma rays (g)	0.37	0.48	0.42
$^{n}/_{g}$ ratio	7.3	6.1	6.66
Leakage (L)	1.17	1.02	1.10
n - L	1.52	1.91	1.71
g - L	-0.81	-0.54	-0.68
Uncompensated gamma rays (G)	6.16	6.68	6.42
$^{g}/_{G}$ ratio	16.82	14.01	15.23

Units are (except for ratios) in volts discharged per 20 mins.

Devices A and B were as near as possible of identical construction.

Uncompensated gamma rays (G) were taken with the linking capacitance short circuited.

Fig 9.8



The above relationship is based only on results taken from alternate neutron and gamma ray exposures for the compensation device.

Variation of neutron and gamma ray sensitivity and $^{n}/_{\gamma}$ ratio with centre electrode voltage

observed electrical leakage rates, gives a value of approximately 12 hours. If the assumption is made that this electrical leakage is entirely due to the charge lost from the linking capacitance formed across the P.T.F.E. liner, then the apparent resistivity of the dielectric is found to be in the region of 3.0×10^{18} ohm cm. (90) This value compares favourably with that normally accepted for P.T.F.E. at around 10^{18} ohm cm. (91) This demonstrates that a plastic with high insulation properties such as P.T.F.E., giving a normally acceptable rate of electrical leakage, would be difficult to adapt in this coaxial configuration for use within a compensation device.

It was also noticeable from the results that there was a change in the compensation rate of the device when different irradiation procedures were adopted. (ie gamma ray only exposures compared with alternating neutron and gamma ray exposures.) This indicates that there might be some non saturation current effect, but it is virtually impossible to estimate its significance compared to that of electrical leakage without taking saturation current measurements. As stated previously, measuring such small currents is extremely difficult in the presence of radiation. Therefore it was decided that the effort and time involved in such a study was more than could be allotted at this late stage in the investigations.

9.9 Summary and suggestions for further research

As previously stated the main object of this last section was to prove that the idea of a neutron and gamma ray dosimeter, using the compensation principle given in Appendix B, was a feasible not only from a practical viewpoint but also a commercial viewpoint. From the results obtained it would seem that this object had been achieved.

The secondary consideration was to construct the device so that the neutron and gamma ray readings were given in the ratio of their dose equivalent. Thus giving the instruments equal absorbed doses of neutrons and gamma rays produces a neutron reading of approximately 6 to 7 times greater than that for gamma rays (for this particular neutron energy used). In the construction of the device allowance was made for electrical leakage therefore the results do roughly give the correct ratio. However subtracting electrical leakage from the results makes the gamma ray exposures have a negative reading (ie a charging up of the gold leaf). This shows that with electrical leakage at an acceptable level different chamber parameters would have to be evaluated and used. For example a better value of D_2/D_1 would be from fig 8.11, in the region of 0.950. This would then allow a wider choice of C/C_2, between approximately 35 and 90, for the correct response. However it is not envisaged that the volume to capacitance ratio of 5, used for the compensation device, could ever be used in a commercial instrument. If this value

of 5 was used, the ratio $^{a}/_{b}$ would have to be unduly large. (See section 8.5). The largest value of $^{D}/_{C}$ that could be used in a commercial dosimeter is in the region of 2. This figure is obtained from present day government specifications as to what size actually constitutes a personal dosimeter. Therefore from the testing of this prototype the compensation idea shows much promise. However from a commercial viewpoint there are difficulties still to be overcome. To adapt a device based on the operation of a gold leaf, to one of half the size using a quartz fibre, while maintaining accuracy limits for response and other specifications laid down by government bodies and safety committees presents a problem. Therefore from this commercial adaption viewpoint there are five immediate problems which present themselves :-

(1) Electrical leakage across the linking capacitance. Although P.T.F.E. a good electrical insulator for most purposes, has been used in this compensation device it still allows too high a leakage rate for the acceptable 2% (laid down for gamma ray dosimeters). There are various ways to tackle this problem, such as to use an even better insulator (ceramic, etc) in conjunction with P.T.F.E. in the form of a sandwich, or alternatively increase the value of C by external means. It can be seen from fig 8.11 and equation 8.4 that increasing the value of C to say, infinity is mathematically possible as it does actually increase the overall radiation response of the instrument. Therefore perhaps some means could be found of placing a large capacitance in parallel with that of the linking

capacitance.

(2) Dose rate dependence. The operational nature of the device (ie gas ionisation) would present a problem from the military viewpoint. Here it is envisaged that very high dose rates of small duration would have to be measured. Therefore if the instrument is planned for use as a radiation device for use in atomic warfare further tests would be required. For example tests could be taken with kilocurie cobalt 60 sources, Godiva reactor pulses and a pulsed X-Ray machine to determine dose rate dependence. (9)

(3) Neutron energy dependence. The response of this compensation device could also be investigated for a wider range of neutron energies. This project was conducted with one source of monoenergetic neutrons and a radioactive source with a broad spectrum of energies. With a higher energy accelerator of about 3 MV terminal voltage, readings could be taken at 2 Mev steps up to 14 Mev by employing various accelerating voltages, incident beams and target elements. Further tests could also be undertaken with thermal neutrons so that the significance of aluminium in and around the dosimeter could be better appreciated.

(4) Problems of a more technological nature involve the matching of chambers during construction to achieve the correct chamber volume ratio for the desired instrument response. One way of approaching this problem would be to consider some form of "tunable" insert which could be positioned in the neutron insensitive chamber during final assembly of the instrument. However if manufacturing tolerances are high enough this problem

might not arise.

(5) Finally the use of P.T.F.E. as a liner in the neutron insensitive chamber also poses problems. The resistivity of the plastic does not appear to be high enough. Also P.T.F.E. is impossible to mould in a factory environment without employing elaborate and costly safety precautions. However one non-hydrogenous plastic which may provide an alternative to P.T.F.E. is Kel F, (polytrifluorochloroethylene $C_2 F_3 Cl$). This has somewhat similar properties to P.T.F.E. but is much easier to fabricate. For example it is not possible to cast and impregnate P.T.F.E. easily while with Kel F there is no difficulty.

Appendix A Dose Rate Calculations

A.1 Introduction

That the intensity of radiation from a point source varies as the inverse square of the distance from a reciever to the source is a fundamental law universally accepted for radiation measurements. However measurements have been made which show that under certain conditions there are significant deviations from this law. Such apparent deviations have been attributed to various secondary effects such as :-

(1) Finite size of the chamber.

(2) Scattering.

The irradiations completed in this project were done under non ideal conditions where some form of modified inverse square law applies. Therefore for accurate dose rates to be applied to the readings some calculation must be made of the effect of these modifications.

A.1.1 Finite size of the chamber

In order to obtain a sufficiently high dose rate it is often necessary to place the specimen close to the point radiation source. With the intensity of the source obeying the inverse square law the closer the specimen is to the source, the more impossible it becomes to achieve an equal dose rate to all points within the volume of the specimen. The larger the specimen the greater is the effect. However this can to some extent be overcome by rotation of the specimen, but this would be an unsatisfactory

solution for the liner chambers used in this project. It has been found that when the source to chamber distance approaches the chamber dimensions, the finite size of the chamber (or specimen) produces an appreciable deviation from the inverse square law relationship. Fig A.1 gives an indication of the variation of absorbed dose within a specimen close to a point source of radiation.

Correction factors have also been calculated to adjust the readings of a finite size chamber to those of ideal point chambers.[83,84,85] These calculations have been evaluated by applying the inverse square law throughout the real chamber volume and then integrating over that volume, although the intergrations of Spiers [85] were confined to the interior surfaces of the chamber . This would yield a more precise correction factor as ionisation produced by exposure to neutron and gamma rays is generated by secondary charged particles originating in the internal surface layers of the chamber walls. The volume integrals, on the other hand, would be more applicable to a primary radiation consisting of energetic charged particles, such as beta rays.[86] The correction factors applied to the dose rates are based on those evaluated by Kondo and Randolph. [84]

A.1.2. Scattering

The scattering of radiation within the specimen and from nearby objects is one of the more uncertain factors in dose measurements.

Fig A.1

Influence of the inverse square law on the absorbed dose distribution in a specimen exposed to a point source.

The attenuation of the radiation in passing through the specimen is not accounted for, nor is the effect of inhomogeneity of the specimen material. It is also assumed that the secondary charged particles have negligible range.

Curve A Variation with specimen thickness. If a 10 cm thick specimen is placed with its proximal surface 1 metre from the source the ratio of thickness to SSD (source to sample distance) is 0.1, and the inverse square law causes the dose at the distal surface to be 0.83 of that at the proximal surface. If the specimen is moved to 40 cm, this ratio becomes 0.64.

Curve B Variation with specimen lateral "extension." An object of length 60 cm is placed with its long dimension at right angles to the line joining its centre and the source, the latter being 40 cm. The ratio of lateral extension to SSD is 30/40 = 0.75 and from B the dose at the ends of the specimen is 0.64 of that at its centre.

With the relatively small specimens (ie liner chambers) used in this project, the scattered radiation component occuring within the specimen was taken as negligible. However scattering from large objects irradiated by the field, such as a concrete wall, tends to contibute a significant amount to the dose. This is especially noticeable in the case of 14 Mev neutron irradiations. To obtain the desired amount of shielding for the required "safe" level to surrounding personnel, the experiments were conducted within a relatively small area surrounded by concrete blocks.

It is common practice in absorbed dose and dose equivalent measurements to conduct the investigations under conditions, as near as possible, to that of charged particle equilibrium.(see section 2.8). Trouble arises here when accurate absolute absolute measurements are required as the specimen can be subject to secondary charged particles originating from nearby material (such as jigging frameworks, walls, etc,). This kind of anomaly is removed in practice by covering the specimen with a buffer layer of material which approximates to the atomic composition of the liner used in the instrument being tested. This would not be possible within the normal use of any commercial dosimeter, therefore the investigations were conducted without the inclusion of any buffer layer.

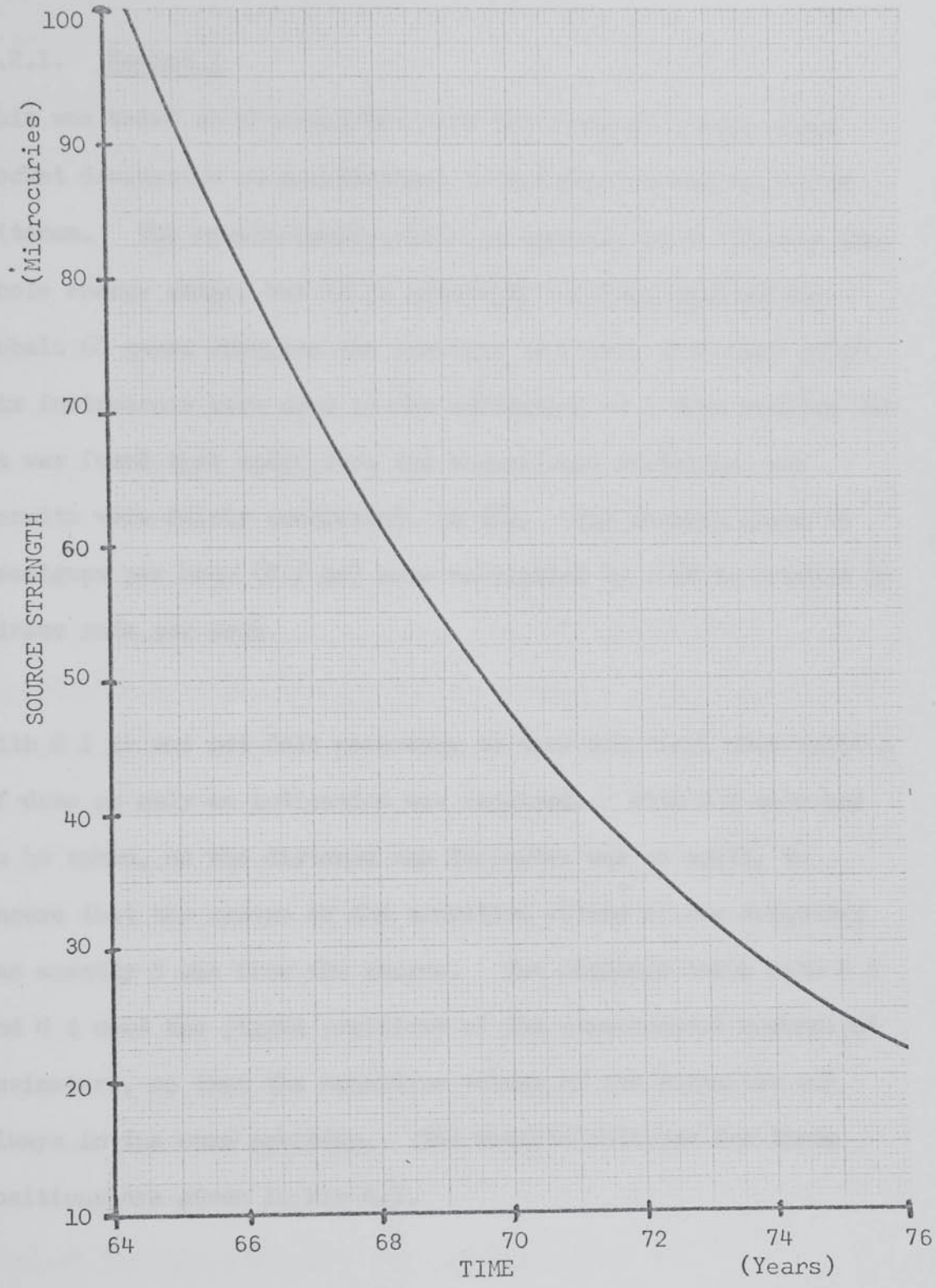
A.2 Gamma ray dose rate calculations

Gamma ray dose rates were primarily considered because it was possible to directly compare the theoretical calculations with the results obtained from quartz fibre condenser ionisation chamber dosimeters. Various methods were used to calculate the dose and each contributed to the final average value which was used for the irradiations. For the results taken with sources G 1 , G 3 and G 4 (all at 20 cms source to specimen distance) no appreciable difference was found between the calculated and observed dose rates. However for source G 2 (at 5 cms source to specimen distance) a difference was observed. This was thought to be primarily due to the "Finite size of the chamber" effect explained in section A.1.1.

The experiments conducted with G 2 were taken over a period of 18 months. Therefore as Cobalt 60 has a relatively short half life (5.26 years) the change in source strength was the first complication that had to be taken into account. (See fig A.2). The source, serial number 3013, was certified by the Radiochemical Centre at Amersham as being 100 microcuries on the 21st April 1964. As can be seen from the graph the source varied in strength over the 18 month period begining January 1969 from 55 to 45 microcuries.

In all four methods were used to estimate the gamma ray dose, one of these based on practical measurements, the other three on

Fig A.2



Cobalt 60 source strength decrease with time.

calculation :-

A.2.1. Method 1

This was based on observations with the standard quartz fibre pocket dosimeters as manufactured by R.A.Stephen and Co Ltd of Mitcham. The manufacturers claim an accuracy of ± 10% over the whole energy range, but it is generally accepted that at the Cobalt 60 gamma energies the accuracy is higher than this value. Six instruments were used in the evaluation of a dose reading and it was found that apart from the occassional deviation, the results were fairly consistent. (± 5%). The results given in roentgens per hour (R / hr) were multiplied by 1.04 to convert to tissue rads per hour.

With G 1 it was not felt necessary to take practical observations of dose as only an indication was required. With G 2 care had to be taken, as the distance and dosimeter was so small, to ensure that the centre of the sensitive volume of the dosimeter was exactly 5 cms from the source. The readings taken with G 3 and G 4 used the jigged positions of the experimental commercial dosimeters, so that the sensitive volume of the dosimeter was always in the same position. The results obtained for these positions are given in fig A.3.

A.2.2. Method 2

This made use of a commonly accepted general equation for exposure rate (roentgens per hour) at one metre from an unshielded point source. The equation, which gives a reasonable approximate value, is :-

$$X/t = 0.5 \times Ci \times Mev \times N$$

Where:-

Ci = Activity of the source in curies.

Mev = Average gamma ray energy in Mev.

N = Number of gamma rays emitted per disintegration.

Although the sources used were not strictly point sources, the approximation to a point source is reasonable and it was thought worthwhile to include these calculations in the results. The active source dimensions are in the region of 0.1 cms by 0.5 cms. The results are given in fig A.3 after the multiplying factor of 1.04 (to convert to tissue rads per hour) has been applied.

A.2.3. Method 3

This was based on a knowledge of the specific gamma ray constant (Γ) as given in section 2.11. For the two sources used in the project the specific gamma ray constant values are [88] :-

Cobalt 60 = 1.30 R m^{-2} h^{-1} Ci^{-1}

Caesium 137 = 0.326 R m^{-2} h^{-1} Ci^{-1}

In the cobalt 60 value account has been taken of the two gamma rays emitted per disintegration of the nucleus. The specific gamma ray constant variation with gamma ray energy for a source which emits one gamma ray per disintegration of the nucleus is given in fig A.4.[89] From the above values the exposure rate at the various distances at which the sources were used was evaluated. This was then converted to absorbed dose rate in tissue by using the factor explained in Method 2.

A.2.4. Method 4

This last method was based on direct calculation from first principles of the dose rate from a knowledge of the strength of the source:-

An isotope which emits one gamma ray of energy E Mev per disintegration of the nucleus is considered. At a distance of y cms from a point source of 1 millicurie of this isotope the energy flux will be :-

$$F = \frac{3.7 \times 10^{10} \times 3600 \times E}{10^3 \times 4 y^2} \text{ Mev/ cm}^2 \text{ mCi}^{-1} \text{ hr}^{-1}$$

In a small volume of air, the rate of energy absorption is equal to the probability of production of secondary electrons (which is given by the linear gamma ray absorption coefficient in air) multiplied by the energy flux. Therefore :-

$$Q/V = F$$

= Linear gamma ray absorption coefficient in air, (derived from the mass energy transfer coefficient).

If W, in units of ev, is the average amount of energy required to produce one ion pair in air, then the relationship between roentgens and the energy deposited per unit volume (given in the basic definition of the roentgen) is :-

$$\text{One Roentgen} = \frac{W \times 10^{-6}}{4.80 \times 10^{-10}} \quad \text{Mev / cm}^3$$

Where :-

W = 33.7 ev / ion pair.

e = Electronic charge = 4.80×10^{-10} esu.

Therefore 1 R = 7.02×10^{4} Mev / cm^3

and $Q/V = 1.51 \times 10^{5} \; E/y^2 \; R \; m^{-2} \; Ci^{-1} \; hr^{-1}$

As this calculated dose is actually the exposure dose, the tissue rad conversion factor of 1.04 is used as Method 2. This value for the dose rate was applied to all the sources and the results are given in fig A.3.

Fig A.3

Radiation source	Methods				
	(1)	(2)	(3)	(4)	Average
G 1	8.15	7.73	8.04	-	7.97
G 2	26.30	24.96	25.96	20.00	24.30
G 3a	68.5	65.0	67.6	64.0	66.27
G 3b	17.13	16.25	16.90	16.00	16.57
G 3c	7.61	7.20	7.51	7.10	7.35
G 4a	3.60	3.26	3.22	3.60	3.44
G 4b	0.93	0.81	0.81	0.90	0.86
G 4c	0.41	0.36	0.36	0.40	0.38

Explanation of gamma ray sources given in section 5.3 .

Explanation of methods given in section A.2 .

Units are tissue millirads per hour (millirem / hr).

Calculated and observed gamma ray dose rates for various sources

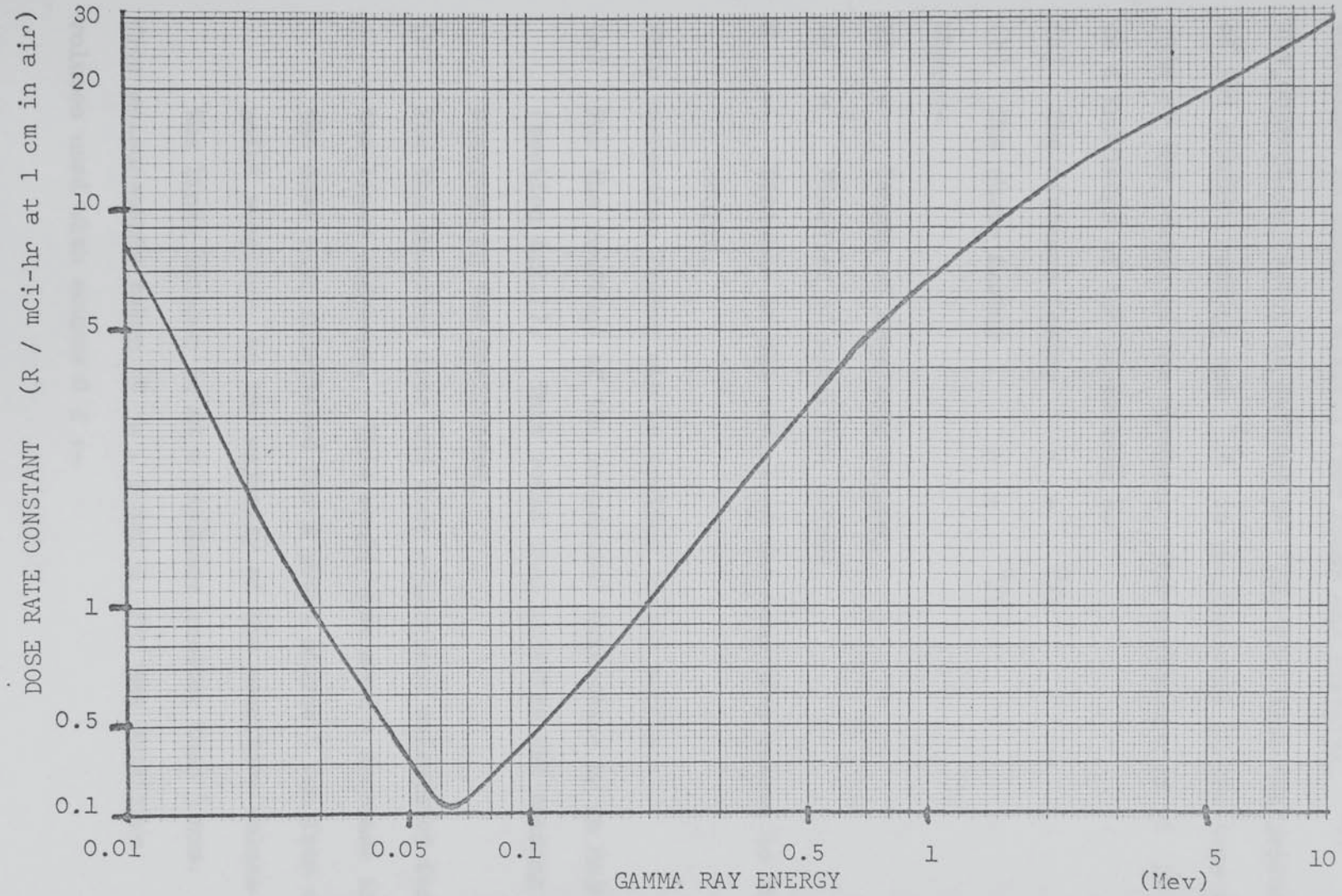


Fig A.4

Gamma ray dose rate constant(Γ) for a γ ray point source in air as a function of γ ray energy.

A.2.5. Correction factor

The correction factor used for the gamma ray dose rate calculations is for the "Finite size of the chamber" effect, explained in section A.1.1., from Kondo and Randolph. (84) The correction factor is defined as $^{1}/_{K}$, such that the product of the chamber reading and $^{1}/_{K}$ is proportional to the true dose rate at the centre of the chamber. The evaluation of K is based on a knowledge of two parameters :-

(a) The distance factor $\alpha = {}^{a}/_{D}$

(b) The shape factor $\sigma = {}^{a}/_{L}$

Where:-

2L = Length of sensitive volume.

2a = Diameter of sensitive volume.

D = Distance of the centre of the sensitive volume to the source.

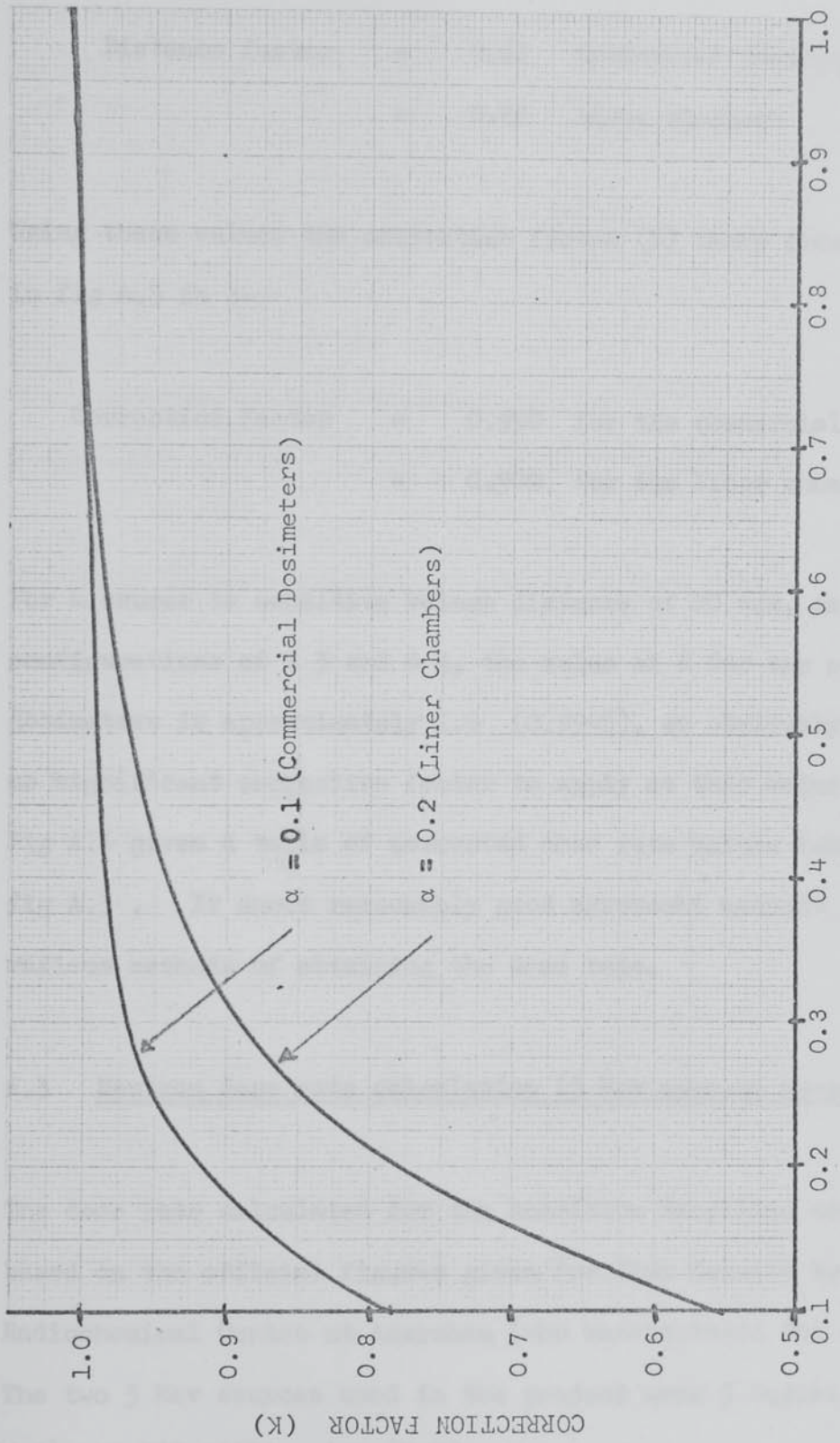
This correction factor can be applied in two senses :-

(a) For the readings of the commercial dosimeters as in Method 1 (section A.2.1). This would then increase the reading observed on the dosimeters.

(b) For the absorbed dose and dose equivalent calculated for the liner chambers. This would then show a decrease in the dose rate calculated for a point source 5 cms from a point receiver, as the majority of the sensitive volume of the liner chamber is at a distance greater than 5 cms.

From measurements taken of a , D and L for the sensitive volumes used with source G 2 :-

Fig A.5



Correction Factor (K) against Shape Factor (σ) for two values of Distance Factor (α).

Shape factor	=	0.053	Commercial dosimeters
	=	0.364	Liner chambers
Distance factor	=	0.10	Commercial dosimeters
	=	0.20	Liner chambers

Using these values the correction factor (K) taken from the graph in fig A.5 is :-

Correction factor	=	0.990	for the commercial dosimeters.
	=	0.908	for the liner chambers.

For a source to sensitive volume distance of 20 cms, as in the configurations of G 3 and G 4, the value of K for the commercial dosimeters is approximately 1.0 (0.9996), so obviously there is no significant correction factor to apply at this value of D . Fig A.6 gives a table of corrected dose rate values taken from fig A.3 . It shows reasonably good agreement amongst the various methods of obtaining the dose rate.

A.3 Neutron dose rate calculation (5 Mev average energy)

The dose rate calculated for the Americium Beryllium sources was based on the official figures given for flux density by the Radiochemical Centre at Amersham (who manufactured the sources). The two 5 Mev sources used in the project were 3 Curies (N 1 and N 3) and 300 millicuries (N 2).

Fig A.6

Calculated and observed gamma ray dose rates for source G 2 corrected for Finite size of Chamber effect.

Radiation source	Methods				
	(1)	(2)	(3)	(4)	Average
G 2	23.88	22.66	23.56	22.25	23.08

The correction is only applied to source G 2 as it is not significant beyond 5 cms source to chamber distance.

Units are tissue millirads per hour (millirem / hr).

The emission ratings were :-

$$3 \text{ Curies} = 7.5 \times 10^6 \text{ n / sec}$$

$$300 \text{ millicuries} = 7.5 \times 10^5 \text{ n / sec}$$

Using the inverse square law and the figure for 5 Mev neutrons (given in fig 1.8) for a neutron flux level equivalent to one M.P.L. a figure was evaluated for dose equivalent. This could then be converted to absorbed dose using the Quality Factor (also given in fig 1.8). It was appreciated that the Americium /Beryllium source was not a monoenergetic source, the energy of 5 Mev being the average. This meant that the Quality Factor of 7 was also an average figure.

It was clear that if a reasonable accuracy was to be achieved with these dose rate calculations, then a "Finite size of chamber" correction would have to be applied to the results taken with source N 2.(Similar to that of G 2). However the approach used for the gamma ray source is not applicable to a neutron source for the following reasons :-

(a) The distribution of the "Knock on" protons in the liner of the chamber is non isotropic.

(b) The secondary protons produced do not all effectively originate on the internal surface of the sensitive volume.

(c) The dimensions of the source N 2 are such that it can no longer be remotely considered as a point source.

The facts supporting these statements are briefly as follows. In (a) the distribution of neutrons tends to peak in the forward direction. For (b) many protons are found to originate well within the liner. In (c) it can be seen from the dimensions given by the Radiochemical Centre of neutron and gamma ray sources, that the neutron source used as N 2 is approximately 13 times the volume of the gamma ray source G 2. This would tend to indicate that perhaps the "Finite size of the chamber" effect for the neutron source would be less significant than for the smaller gamma ray source. This can be to some extent explained by considering the neutron source as supplying a "Finite size of source" effect in opposition to the "Finite size of chamber effect".

Placing a fast neutron monitor (type NE 046) 5 cms from the neutron source showed that the calculated result was 5 % greater than the observed dose level. However these results have not taken into account any "Finite size of chamber" effect which the neutron monitor may have at this distance. The chamber (containing the sensitive volume) for a NE 046 monitor is a right cylinder of approximately 5 cms diameter and 10 cms in length. Ideally the source to chamber distance should be chosen such that small displacements of the chamber, over a distance comparable to that of the chamber size, yield only small percentage changes (of 5 to 10 %) in the measurements observed by the chamber. This was not the case with measurements taken by the NE 046 monitor at 5 cms.

Therefore with these points in mind it was decided that as there was no relevant information on neutron source errors, and in any event these errors were anticipated to be less than those for gamma ray sources, the calculated result would be used. A table of calculated and observed dose rates is given in fig A.7 .

A.4 Neutron dose rate calculations (14 Mev energy)

The dose levels for this energy were based on a knowledge of the neutron flux assessed by means of the associated alpha particle count rate. A S.A.M.E.S. accelerator was used to provide the deuterons for the D.T. reaction, this having the advantage of providing an easily adjustable flux level. This was achieved by variation of the deuteron current to the tritium target. Therefore the number of neutrons leaving the target was known for any particular distance and direction, from a knowledge of the anisotropy factor and direct use of the inverse square law. (The chambers being at 90° and some 55 to 65 cms from the target).

With the commercial dosimeters a high dose rate was used to reduce exposure times to a reasonable value. This was necessary as it was not possible to run the accelerator for 3 or 4 hours just to complete one reading. The target lifetime and the actual economics of running an accelerator for this length of time being the main factors preventing this "ideal" dose rate. However with the liner chambers it was possible to adjust the ion current of the accelerator to give a flux level corresponding

Neutron dose rates for 5 Mev irradiations

Source	Calculated dose rate	Observed dose rate
N 1	99	91
N 2	345	325
N 3a	829	-
N 3b	207	198
N 3c	92	89

Units are in millirem / hr.

Calculated results are based on flux levels given by the Radiochemical Centre at Amersham.

Observed results are taken with a type NE 046 monitor.

to the required dose rate of 345 mrem / hr. Thus with the short two minute exposure time for the liner chambers this low flux level could be easily maintained.

It was necessary on some occasions to irradiate the commercial dosimeters in flux levels corresponding to 1 rem / hr at 1 metre. This appreciably activated the end flange of the target assembly, producing a significant gamma ray dose (see section 5.2.2. for source N 4). Therefore 2.5 cms of lead were used to attenuate the gamma rays to below 1% of the neutron dose equivalent . This produced a corresponding 12% reduction in the neutron flux level, however this was allowed for in the calculations.
It was found that with the much lower flux level used with source N 5, (corresponding to 345 mrem / hr) the gamma rays produced by the activation of the end flange were less than 1% of the total dose rate.

Appendix B Provisional Patent

The patent given in the following section was submitted to the Patents Office in London as a provisional patent specification under the patents act of 1949. In the original patent the drawings entitled Figures B.1, B.2 and B.3 are written as figures 1,2 and 3 respectively.

Improvements in or relating to Ionisation Dosimeters

We, R. A. Stephen & Company Limited, a British company of Miles Road, Mitcham, Surrey, do hereby declare this invention to be described in the following statement :-
This invention relates to ionisation dosimeters responsive to neutron radiation or to both neutron and gamma radiation and has for its object the provision of a neutron dosimeter in which at least partial compensation for gamma dosage may be obtained.

In practice, neutron radiation is always accompanied by some gamma radiation, so that it is difficult to discriminate between the two forms of radiation or to measure the effective biological dose in a mixed radiation field.

The need for discrimination is due to the differing biological effectiveness or quality factor of neutron and gamma radiation. Total ionisation is proportional to the dose in rads, whereas biological damage is proportional to the dose in rems (rads multiplied by quality factor). The ionisation produced by a

mixed radiation field cannot therefore be a measure of biological dose. For fast neutrons, with energies in the range of 0.1 Mev to 15 Mev, a dosimeter needs to be some 6 to 10 times more sensitive to neutrons than to gamma radiation if the ionisation is to represent biological dosage. Because of the widely different interaction cross-sections of nuclei for fast neutrons and for gamma radiation it has not been possible to achieve the required ratio of sensitivities in a single dosimeter. Previously two dosimeters have therefore been used, a first being sensitive both to neutron and to gamma radiation and a second being mainly sensitive to gamma radiation. The neutron dose is found by subtracting the gamma dose measured by the second dosimeter from the dose (neutron + gamma) measured by the first. This arrangement requires relatively bulky apparatus and is liable to error in the making and subtraction of two readings.

Features and advantages of ionisation dosimeters according to the invention will become apparent from the following description of embodiments thereof, taken in conjunction with the accompanying drawing, comprising figures 1 to 3, of which :-

Figure B.1 is a partial longitudinal cross-section through one embodiment of a dosimeter according to the invention;

Figure B.2 is a similar view of another embodiment of the invention; and

Figure B.3 shows a third embodiment of the invention.

A dosimeter according to the invention includes two ionisation

chambers which are arranged to have markedly different ratios of neutron to gamma radiation response. The two chambers are arranged in opposite polarities that the electric fields within them are opposed. A direct connection is made between one electrode of each chamber and the corresponding electrode of the other. The remaining pair of corresponding electrodes are capacitively coupled. Owing to the inverted polarity of the two chambers the capacitively coupled electrodes collect charge carriers of opposite sign, so that the net charge collected on isolated electrodes is the difference between the charges produced in the two chambers. The potential appearing on the isolated electrode is therefore proportional to the charge difference.

If equal charges are collected in the two chambers the change in the potential of the isolated electrode is zero. For unequal charges the electrode potential does change, the potential in the chamber giving the smaller charge being increased and that in the chamber giving the greater charge being decreased.

The charge difference is stored in the capacitor connected between the two chambers. A conventional quartz fibre electroscope is provided in the chamber which has the greater ratio of neutron to gamma ray sensitivity and the indication of this electroscope gives an overall measure of the potential change.

It is possible to select for the coupling capacitor connected between the electrodes a value such that the reading obtained is proportional to the effective biological dose in a mixed radiation field. Alternatively, the value of the coupling capacitor may be chosen to give complete cancellation of the potential charges,

in the chamber which is more sensitive to neutron radiation, which is due to the gamma radiation dose, thus giving a reading which is proportional to the neutron dose only.

To obtain a true measure of biological dose in a mixed radiation field the value C_t of the coupling capacitor is given by the equation :-

$$C_t = \frac{C_2 \, (QF \, q_{1*}/q_{1n} - 1)}{1 + QF \, q_{1*}/q_{in} \, (q_{2*}/q_{1*} - 1) - q_{2n}/q_{1n}}$$

Where:-

C_2 is the capacitance of the chamber having the lower sensitivity to neutrons;

q_{2*} is the charge sensitivity to gamma radiation of that chamber;

q_{2n} is the charge sensitivity to neutron radiation of that chamber;

q_{1*} is the charge sensitivity to gamma radiation of the chamber which is more sensitive to neutron radiation.

q_{1n} is the charge sensitivity of that chamber to neutron radiation; and

QF is the quality factor of the neutron radiation.

For complete compensation of the gamma radiation dose, the capacitance C_c of the coupling capacitor is given by :-

$$C_c = \frac{C_2}{(q_{2*}/q_{1*} - 1)}$$

The neutron response of the chamber with the greater rato of neutron to gamma sensitivity is given by:-

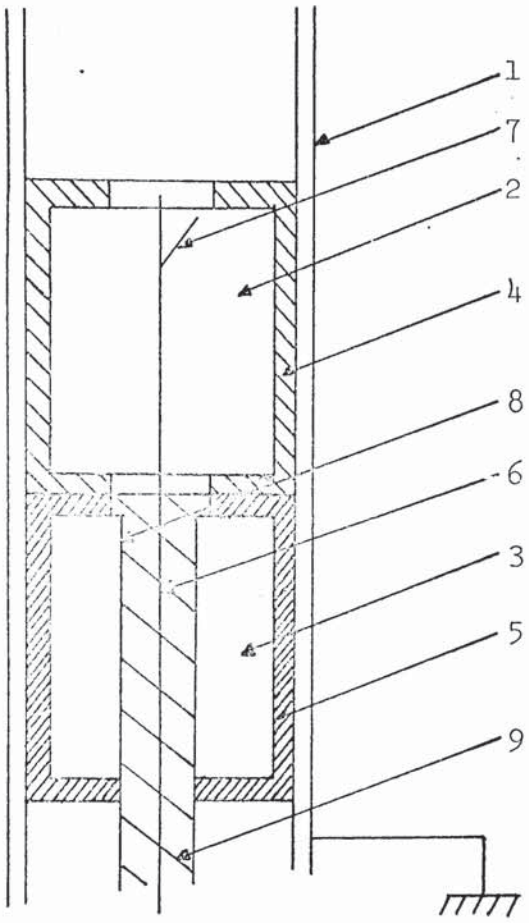
$$q_{1n}/C_1 \left[\frac{C_1/C_2 + C_1/C_t - q_{2n}/q_{1n} \; C_1/C_2}{C_1/C_2 + C_1/C_t + 1} \right]$$

Where C_1 is the capacitance of that chamber and the remaining symbols have the meanings already stated.

This compares with the neutron sensitivity of q_{1n}/C_t of the uncompensated chamber.

Figure B.1 illustrates an embodiment of the invention in which the isolated electrodes of the two chambers are capacitively coupled. A part only of the dosimeter is shown. To the components illustrated there may be added a conventional optical system for viewing the electroscope and some convenient arrangement for charging the isolated electrodes of the ionisation chamber. The device shown includes a casing 1 surrounding two coaxially disposed cylindrical ionisation chambers 2, 3. Chamber 2 has a wall coating 4 rendering the chamber preferentially sensitive to neutron radiation, while chamber 3 has a coating 5 which makes it preferentially sensitive to gamma radiation. The isolated axial electrode 6 of chamber 2 carries the measuring electroscope 7. This electrode also passes through chamber 3, within which it is surrounded by a second electrode and supported by 9. When the device is charged, electrode 6 is connected to a source of potent - ial which is positive with repect to the earthed casing 1 of

Fig B.1



the device. A source of negative potential with respect to the earthed casing 1 of the device, is connected to the conductive outer surface of the dielectric 8 in chamber 3. Thus charge transference due to ionisation produced within chamber 2 in response to neutron radiation will tend to reduce the positive potential of electrode 6, while charge transference in chamber 3 due to gamma radiation will tend to increase the positive potential of this electrode. The amount of compensation will depend upon the value of the coupling capacitance, which in this embodiment is constituted by the capacitance between electrode 6 and the conducting surface of dielectric sheath 8.

In the embodiment shown in figure B.2 the arrangements are generally similar to those of figure B.1 and corresponding elements therefore bear the same reference numerals. In this embodiment, however, the isolated electrode 6 of chamber 2 is coupled through a capacitor 10 to the isolated electrode 8 of chamber 3.

In the embodiment shown in figure B.3 a single isolated electrode 6 is common to both chamber 2 and chamber 3. Here, however, the lining 5 of chamber 3 is separated by a dielectric layer 11 from the casing 1 of the device and by a dielectric annulus 12 from the lining 4 of chamber 2.

Where complete compensation for gamma radiation dosage is required so that the reading of the device is related only to the neutron flux, it will be necessary so to construct the device that the charge sensitivities for gamma radiation q_{1*} and q_{2*} of the two chambers will be appropriately related to the ratio of the

Fig B.2

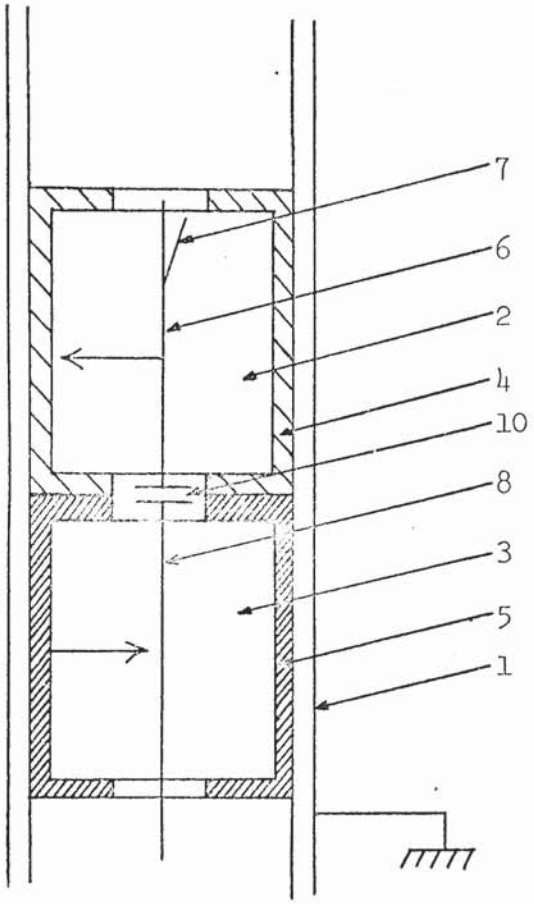
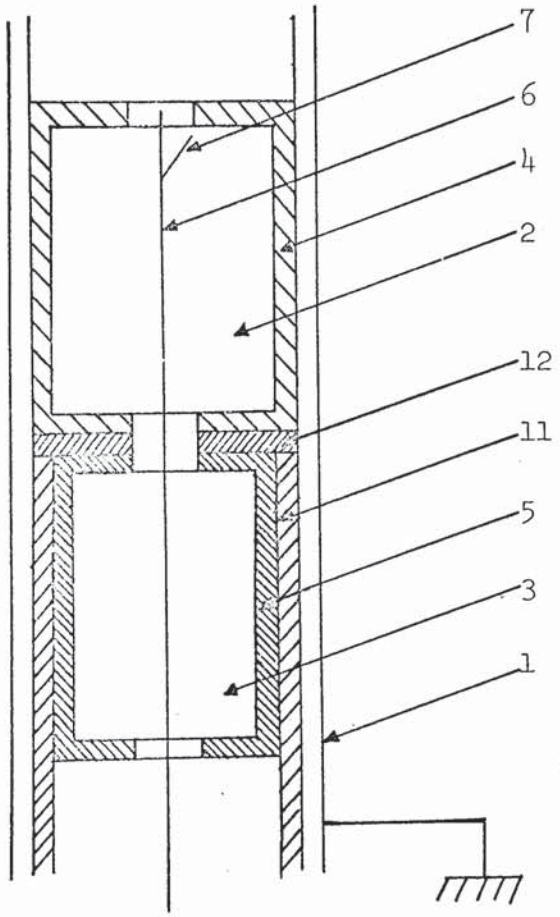


Fig B.3



coupling capacitance to that of the second (or gamma sensitive) chamber :-

$$q_{1*}/q_{2*} = \frac{C_c}{C_c + C_2}$$

We, PAUL COOPER and BRYAN TYMONS, hereby acknowledge that R. A. STEPHEN & COMPANY LIMITED are entitled to the sole benefits of this invention made by us and described in the foregoing specification and we hereby assign to the said R. A. STEPHEN & COMPANY LIMITED the right to apply for Letters Patent in respect of the said invention.

As witness our hands this 10th day of October, 1969.

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