DOE/EV/01105--T2

DE83 017507

WISCONSIN MEDICAL PHYSICS

WMP-168

NEUTRON/GAMMA DOSE SEPARATION BY
THE MULTIPLE ION CHAMBER TECHNIQUE

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

by Steven J. Goetsch, Ph.D.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

January, 1983

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

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

DISCLAIMER

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

DOE/EY/01105--T2

NEUTRON/GAMMA DOSE SEPARATION BY
THE MULTIPLE-ION-CHAMBER TECHNIQUE

by STEVEN J. GOETSCH

A thesis submitted in partial fulfillment of the
requirements for the degree of

DOCTOR OF PHILOSOPHY
(Medical Physics)

at the

UNIVERSITY OF WISCONSIN MADISON

1983

DISCLAIMER

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

NOTICE

PORTIONS OF THIS REPORT ARE ILLEGIBLE.

**It has been reproduced from the best
available copy to permit the broadest
possible availability.**

MASTER

EAB
DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

NEUTRON/GAMMA DOSE SEPARATION BY
THE MULTIPLE-ION-CHAMBER TECHNIQUE

Steven J. Goetsch

Under the supervision of Professor F. H. Attix

Mixed neutron/gamma radiation dosimetry is more complex and less well understood than gamma dosimetry. Fundamental disagreements still exist over neutron/gamma dose separation in phantom and in patients.

Many mixed n/ γ dosimetry systems rely on two dosimeters, one composed of a tissue-equivalent material and the other made from a non-hydrogenous material. The paired chamber technique works well in fields of neutron radiation nearly identical in spectral composition to that in which the dosimeters were calibrated. However, this technique is drastically compromised in phantom due to the degradation of the neutron spectrum. The three-dosimeter technique allows for the fall-off in neutron sensitivity of the two non-hydrogenous dosimeters. Precise and physically meaningful results were obtained with this technique with a D-T source in air and in phantom and with simultaneous D-T neutron and ^{60}Co gamma ray irradiation in air.

The MORSE-CG coupled n/ γ three-dimensional Monte Carlo code was employed to calculate neutron and gamma doses in a water phantom. Gamma doses calculated in phantom with this code were generally lower than corresponding ion chamber measurements. This can be explained by

the departure of irradiation conditions from ideal narrow-beam geometry.

ACKNOWLEDGEMENTS

I would like to acknowledge the help and support of a great many people without whom this work would not have been possible. I thank Herb Attix, my advisor, and originator of many of the ideas and methods upon which this work was based, for all his insight and advice over the years; Paul DeLuca for his patient explanations and constant challenges to my abilities; to Dave Pearson for his invaluable assistance and generous sharing of the folklore of experimental physics; to Gary Herling for his inestimable contributions to my ability to use a powerful computer code; to the staff of the Radiation Shielding Information Center (RSIC) for their help and advice; to Murray Thompson for his leadership in providing a first-rate user-friendly computing facility; and to my fellow graduate students for their help and encouragement.

I also wish to thank the members of my thesis committee: Professor John R. Cameron, Chairman of the Medical Physics Department, Professor H.H. Barschall of the Physics and Nuclear Engineering Departments, Professor Bhudatt R. Paliwal of the Medical Physics and Human Oncology Departments, as well as Professors Herb Attix and Paul DeLuca, both of the Medical Physics Department.

Above all I wish to thank my wife Mona, an accomplished professional educator in her own right, for her loving support and sacrifices throughout this long endeavor.

Finally, I wish to acknowledge the support of the U.S. Department of Energy through contract number DE-AC02-76EV01105 and the National Cancer Institute through PHS Grant Number 5 T32 CA09206 CT.

TABLE OF CONTENTS

	<u>Page</u>
Abstract	ii
Acknowledgements	iv
List of tables	viii
List of figures	xi
 Chapter I. Introduction	
a. Statement of Problem	1
b. Early Neutron Dosimetry	3
c. Development of Twin Ion Chamber Method	9
 Chapter II. Instrumentation and Methods	
a. Lead Filtration Method	23
b. Water Phantom Measurements	31
 Chapter III. Ion Chamber Measurements	
a. Lead Filtration Measurement Results	36
b. Early Water-Phantom Measurements at U.W.	43
c. Recent Water-Phantom Results	51
d. Combined D-T and Co-60 Irradiation in Free Space	70
 Chapter IV. Monte Carlo Simulation	
a. MORSE-CG Code	76
b. Simulation of GTNS Facility with Water Phantom	83
 Chapter V. Comparison of Experimental and Theoretical Results	
a. Comparison of MORSE-CG with Pure D-T Source	89
b. Comparison of Measured Results with Mixed n/ γ Beam	99

c. Comparison of Results with Other Workers	104
Chapter VI. Conclusions	115
Appendix A Calculation of Zirkle's n-unit in Terms of Rads	119
Appendix B Calculation of Ionization Ratio in Tissue	121
Relative to Air	
Appendix C Evaluation of f_n (from Bretscher and French)	123
Appendix D Revised Method for Estimating Void Corrections	125
Bibliography	128

List of Tables

	<u>Page</u>
1. Early Tissue-Equivalent Materials	13
2. Simultaneous Equations for Lead Filtration Method	24
3. Composition of Muscle Tissue & Tissue-Equivalent Materials	28
4. Two- and Three-Dosimeter Methods	32
5. Ion Chamber Calibration Data	41
6. Summary of B/A Measurements at 14-15 MeV by Other Groups	42
7. Gamma Doses at 5cm Depth April, 1979	44
8. Gamma Doses at 10cm Depth April, 1979	45
9. Partition of Neutron Dose at 5 and 10cm Depth Apr., 1979	48
10. $\overline{(B/A)}$ at 5 and 10cm Depth Apr., 1979	49
11. Gamma Fractions at 3cm Depth June, 1982	56
12. Gamma Fractions at 5cm Depth June, 1982	56
13. Gamma Fractions at 7.5cm Depth June, 1982	57
14. Gamma Fractions at 10cm Depth June, 1982	57
15. Gamma Fractions at 15cm Depth June, 1982	58

16. Gamma Fractions at 20cm Depth June, 1982	58
17. Gamma Fractions on Axis June & July, 1982	59
18. Uncertainties in n/γ Dose Estimates	62
19. $\overline{(B/A)}$ at 5 and 10cm Depth June, 1982	63
20. $\overline{(B/A)}$ on Axis June & July, 1982	64
21. Partition of Neutron Dose at 5 and 10cm Depth, June, 1982	65
22. Partition of Neutron Dose on Axis June & July, 1982	66
23. n/γ Doses from D-T Neutrons & ^{60}Co Gammas	74
24. Gamma Fractions for INDI Experiment	89
25. n, γ and Total Dose On-Axis from MORSE-CG	90
26. n, γ and Total Dose at 3cm Depth from MORSE-CG	91
27. n, γ and Total Dose at 5cm Depth from MORSE-CG	92
28. n, γ and Total Dose at 15cm Depth from MORSE-CG	92
29. Gamma Fractions On Axis from Revised MORSE-CG	101
30. Gamma Fractions at 3 and 15cm Depth from Revised MORSE-CG	102
31. Comparison of Neutron Dosimetric Parameters	105
32. Environmental Gamma Dose for the INDI Experiment (Burger)	110

33. Gamma Dose Fractions for the INDI Experiment	112
34. Revised Environmental Gamma Dose for the INDI Experiment	113
35. Environmental Gamma Dose in Phantom at GTNS	114

List of Figures

	<u>Page</u>
1. Schematic plan view of tritium gas target	29
2. Geometry for water phantom measurements	34
3. Fluence-to-kerma factors for H and C	47
4. Fluence-to-kerma factors for H and Mg	47
5. Neutron dose surface in air at 20cm	52
6. Configuration of dosimeters in water phantom	53
7. Neutron dose surface in phantom	67
8. Gamma dose surface in phantom	68
9. Gamma dose fraction in phantom	69
10. Dual beam irradiation facility	70
11. Configuration of dosimeters for n/ γ irradiation	72
12. Random walk diagram for MORSE-CG	78
13. Computer simulation of GTNS facility	84
14. Neutron dose fractions at 3, 5 and 15cm depths	93
15. Neutron energy fluence as a function of depth in phantom	94
16. n and γ dose in phantom relative to total incident dose	96
17. n & γ dose fractions at 3cm depth in phantom	97
18. n & γ dose fractions at 5cm depth in phantom	97
19. n & γ dose fractions at 15cm depth in phantom	98
20. Gamma fractions on axis in phantom	101
21. Gamma fractions off-axis at 3cm depth in phantom	103
22. Gamma fractions off-axis at 15cm depth in phantom	103

CHAPTER I

INTRODUCTION

a. Statement of Problem

Neutron dosimetry has long been less well understood than the dosimetry of the three other major categories of ionizing radiation (γ , α & β). The dosimetry of gamma radiation has been on a firm theoretical basis since the promulgation of the Bragg-Gray theory in 1936.¹ Gamma calibrations of dosimeters are routinely available at all major national laboratories and many regional calibration laboratories as well. The dosimetry of the non-penetrating particulate alpha and beta radiations is also well understood.

Experimental neutron dose measurement usually suffers from the problem of concomitant gamma radiation. Many neutron interactions lead to the direct or indirect production of gamma radiation. Almost all neutron dosimeters have some sensitivity to γ -rays as well, and almost all gamma dosimeters also respond to neutrons. The unavailability of a "pure" neutron beam for calibration purposes has been a major obstacle to neutron dosimetry.

The difficulty posed by cross-sensitivities of neutron and gamma detectors is that, although both forms of radiation are penetrating, the biological and physical effects of neutrons are markedly different

from those of gammas. Since the response of such a detector is the product of its neutron sensitivity times the neutron dose plus the gamma sensitivity times the gamma dose (assuming no synergistic action), no clear biological or physical effect can be predicted with confidence. The International Commission on Radiation Units and Measurements² has stated:

"The objective of neutron absorbed dose determinations is to describe the energy deposition in such detail that workers in biology and medicine may make unambiguous correlations with observed responses, or predictions of responses, of irradiated biological systems."

Medical physicists have not yet completely met the challenge of providing such unequivocal physical measurements or calculations. The biological effect caused by the deposition of a given amount of energy from neutron radiation is far more severe than that caused by the deposition of the same amount of energy from photon interactions. The U.S. Nuclear Regulatory Commission, in its radiation protection regulations³ has administratively adopted a factor of 10 which must be used in converting neutron absorbed doses to neutron dose equivalent, while the corresponding factor for photons is unity. Recent work by Rossi⁴ based on analysis of medical data from survivors of the Hiroshima and Nagasaki atomic bombs has suggested that this factor should be much higher in some circumstances. Thus, the need for precise and accurate neutron dosimetry has never been more critical.

b. Early Neutron Dosimetry

The existence of the neutron was first reported by Chadwick in 1932.⁵ The effects of neutron radiation on biological objects were studied almost immediately.^{6,7} It was at once apparent that fast neutrons deposit energy primarily through elastic recoil interactions. Therefore, although the available ionization chambers responded to neutron radiation, the necessity of developing an entirely new experimental methodology and theoretical dosimetry framework for neutrons was immediately recognized.

A variety of neutron dosimetric methods and units were proposed in the 1930's and 1940's. Ernest O. Lawrence in Berkeley, California reported early biological experiments in terms of the response of a Victoreen condenser "r" meter.⁸ No attempt was made at first to assess the relation of the indicated meter readings to absorbed dose in biological samples. The Berkeley group continued to use the Victoreen meter in early neutron radiation biology work and cancer therapy trials.^{9,10} They adopted the terminology of Zirkle and Lampe.¹¹ Zirkle et al referred to the quantity of neutron radiation which discharged the Victoreen "r" meter to the same extent as did one r (now 1 roentgen, abbreviated 1 R) of X-rays as a "neutron unit" or "n-unit". One r of X-rays was defined at that time as the quantity of photon radiation sufficient to produce one electrostatic unit of charge of either sign in 0.001293 gram of dry air. Since the 60-inch "medical" cyclotron at

Berkeley produced a spectrum of neutron energies up to about 19 MeV, the mean energy was probably about 6.4 MeV. Using presently accepted values for physical factors, an estimate of 0.90 rad for 1 n-unit can be computed (see Appendix A).

L.H. Gray and co-workers in England defined the "neutron energy unit" in 1940.¹² This unit, which was distinctly different from the Berkeley group's "n-unit", was such that "tissue which has received one energy unit of neutron radiation will have suffered the same increment of energy as an equal volume of water exposed to one roentgen of gamma radiation." Using ^{60}Co γ rays, the dose absorbed by a volume of water exposed to one R may be computed as follows:

$$\begin{aligned} (1.) \quad D &= X \cdot f_{\text{med}} \cdot A_{\text{eq}} \\ &= 1\text{R} \cdot 0.966 \text{ rad/R} \cdot 0.985 \\ &= 0.95 \text{ rad} \end{aligned}$$

where D is the dose in rads, X the exposure in Roentgens, f_{med} the Roentgen/rad conversion factor for ^{60}Co γ rays in water and A_{eq} is the attenuation correction factor for γ rays penetrating into a spherical mass of tissue large enough to produce charged-particle equilibrium at its center. Thus, the energy unit defined by Gray should have been about 0.95 rad. Gray did not state how it was known that an equal amount of energy was absorbed in a tissue sample. Gray further claimed that a given fluence of neutron radiation "produces seven times as much ionization in tissue as in an equal mass of air".¹³ Using presently

accepted values for physical factors, a value of 6.4 for the ionization ratio of tissue to air can be calculated (see Appendix B), assuming a mean neutron energy of 2.7 MeV.

Aebersold reviewed the early Berkeley neutron radiation therapy trials in a 1946 paper and concluded that a "free-air exposure of 1 n" from neutrons of average energy 6.4 MeV (produced by the bombardment of 19 MeV deuterons on beryllium) caused the same energy deposition in tissue as an exposure of 2.5 R of 220 kVp X-rays.¹⁴ The absorbed dose in tissue from this exposure is:

$$\begin{aligned}
 (1.) \quad D &= X \cdot f_{\text{med}} \cdot A_{\text{eq}} \\
 &= 2.5 \text{ R} \cdot 0.964 \text{ rad/R} \cdot 1.00 \\
 &= 2.4 \text{ rad}
 \end{aligned}$$

Dr. Robert S. Stone, in a retrospective lecture on neutron radiotherapy in 1948, noted that Parker had suggested the use of a new dose unit called the "roentgen equivalent physical", or rep, for radiations that could not be measured in roentgens.¹⁵ Stone pointed out that since 1 rep was defined [‡] by Parker to be the energy absorbed per gram of tissue from neutron radiation equivalent to the absorbed dose in tissue from an exposure of 1 roentgen from photons, then 1 n = 2.5 rep. The rep was considered to be essentially the same as Gray's energy unit.

[‡] Several definitions for the rep were current at that time.

Very little comment was made in early papers about separation of absorbed dose into neutron and photon components. Lawrence's first report in 1936 on biological research with neutrons made no mention of this matter.¹⁶ Spear, Gray and Read claimed in 1938 that cells bombarded by their 2.4 MeV neutrons "were completely protected from stray X-radiation", but gave no details.¹⁷ Gray and co-workers used both ionization chambers and gamma-insensitive neutron activation pellets in later experiments, but no discrepancies in doses calculated from these two dosimeters were noted.

The Berkeley workers had become concerned enough by 1939 to assign Paul C. Aebersold to investigate the problem of photon contamination.^{18,19} Aebersold designed and constructed a collimator consisting of a channel through a water tank lined with thick lead walls followed by a 3cm-thick lead filter.²⁰ The filter was designed "to suppress gamma rays from the source." Further, relative ionization from various filling gases in a 100cm³ cylindrical brass ionization chamber exposed to the mixed neutron/gamma beam was described as "very different" from that due to gamma radiation alone. Two unspecified methods were used to calculate ionization due to photons from these data and corrections were then made to obtain ionization due to neutrons alone. These specific neutron ionizations were compared to elastic scattering cross sections and found to be in very good agreement in most cases. During the Stone-Larkin neutron radiation therapy series, the beam was found to consist of "95 per cent neutrons and 5 per cent gamma rays".²¹ The intensity of gamma radiation was

adequate to cause 1-2 R exposure to the total body during a 100 "neutron unit" irradiation. Thus, absorbed dose to the patient from the mixed beam was presumably 1-2 rad gamma dose per 240 neutron rads.

Later review indicated serious problems with all neutron dosimetry performed with dosimeters designed for photon exposure measurements. Rossi noted in a review of neutron dosimetry in 1956 that the 25-r and 100-r Victoreen dosimeters, both of which had been extensively used in early fast neutron dosimetry, may have had appreciably varying neutron sensitivities.²² Measurements indicated that a dose of 1.0 n actually represented 2.0 to 2.5 times the energy imparted to tissue as that imparted by 1.0 R of gamma rays. Other proposed units such as Gray's "v-unit" (the amount of neutron radiation necessary to produce 1 esu per 0.001293 gram of dry air from secondary charged particles in water)²³ and Failla's "tissue roentgen" (defined for any type of radiation and any medium but under restrictive conditions)²⁴ did not come into general use.

Ultimately, two units were adopted by the International Commission on Radiological Units which gained widespread acceptance. The rad[†] was

[†] The ICRU has since adopted the Gray (defined as the amount of radiation sufficient to cause the absorption of 1 joule/kilogram) as the unit of absorbed dose. However, the rad is still in widespread use and will be used in this thesis.

adopted by the ICRU in July, 1954.²⁵ It was defined as the amount of any form of radiation sufficient to cause the material being irradiated to absorb 100 ergs/gram. The rem (an acronym for roentgen equivalent for man) was adopted in 1962 by the International Commission on Radiological Units and Measurements as the unit of dose-equivalent.²⁶ It is formally defined as follows:

$$(2.) \quad H = D \cdot Q.F. \cdot D.F. \cdot \dots$$

where H is the dose equivalent in rems, D is the absorbed dose in rads, Q.F. is the Quality Factor and D.F. is the Distribution Factor (for internally deposited radionuclides only). Other multiplicative factors may also be used as deemed appropriate. The Quality Factor is an administrative concept, used in personnel protection, which is based on relevant biological data such as the Relative Biological Effectiveness (RBE). The RBE of a given type and energy of radiation is defined as the ratio of the absorbed dose of 250 kVp X-radiation needed to produce a given biological effect, to the absorbed dose of the radiation in question needed to produce the same effect. The Q.F. for neutron radiation is taken by most national radiation protection agencies to be 10.

c. Development of the Twin Ion Chamber Method

After a decade or more of neutron research and early beginnings of neutron dosimetry, it was generally recognized that any practical neutron generator would also produce a significant amount of gamma radiation. Further, no known single dosimeter could at once measure both neutron and gamma dose and partition the dose into two separate components. Neutron radiation biology had conclusively demonstrated that equivalent absorbed doses of neutron and gamma radiation caused considerably different results. Therefore, researchers began making simultaneous measurements with two detectors of widely differing neutron sensitivities.

Cavity ionization chambers which fulfill (at least approximately) the requirements of Bragg-Gray theory were adopted for neutron dosimetry for many reasons. They are relatively small, robust, well understood, capable of being used as an absolute dosimeter (for photons), accurate for photon doses, easy to operate, and they give highly reproducible results. The "twin" ionization chamber method was apparently originated by Bretscher and French in 1944.²⁷ In a document de-classified about 1957, they described the construction of two ionization chambers designed to have quite different neutron responses, yet constrained to yield identical photon sensitivities.²⁸ This was done by constructing an ethylene gas-polythene wall chamber and a deuterio-ethylene gas-deuteropolythene wall chamber. The deuterium-substituted

plastic and deuterated gas was chemically almost indistinguishable from the other chamber, and the density and energy required to create an ion pair (W/e) was expected to be almost identical. However, the neutron sensitivity of the deuterated chamber was much lower because the energy transferred to the wall and the gas by elastic interactions is much less for the deuterated compounds than for normal hydrogen compounds. The photon sensitivity was adjusted to be equal for the two chambers by varying the gas pressure.

Bretscher and French later reported operational results from their hydrogenous/deuterogenous paired chambers.²⁹ They calculated from theoretical considerations the difference ratio $J_{\text{diff}} = (J_H - J_D) / J_H$ where J_H is the response (ionization) of the hydrogenous chamber to neutrons alone, and J_D is the neutron-only response for the deuterated chamber. These quantities were then simultaneously measured in a relatively "pure" beam of D-D (900kV) neutrons and found to be in agreement within experimental error. Further measurements were then carried out in a beam of neutrons known to be contaminated with gamma radiation. The fractional ionization current f_n in the neutron-sensitive (hydrogenous) chamber in the mixed beam due to neutrons was shown to be:

$$(3.) \quad f_n = \frac{(J_H' - J_D')/J_H'}{(J_H - J_D)/J_D}$$

where J_H' is the response of the hydrogenous chamber in the mixed beam

and J_D is the response of the deuterated chamber in the same beam (see Appendix C). Results of this method were as accurate as any other then available and represented the state of the art.

Reinhardt at the Clinton Laboratories pursued a similar line of research in 1947 with the modification that two paired chambers (of unspecified wall composition) were electrically connected in such a way that the imbalance in the ionization current could be read on a Lindemann electrometer.³⁰ One chamber was paraffin-coated inside to enhance neutron response and filled with butane (C_4H_{10}) at 30psi. The other chamber was not lined and was filled with argon at about 35psi. Again, the pressures were adjusted so that the gamma-ray ionization currents of the paired chambers balanced. Recoil protons ejected from the paraffin and butane by fast neutrons produced a differential ionization current which was read on the electrometer. A fast neutron flux density of as little as 5 neutrons per cm^2 per second could be detected.

Further developments along these lines were carried out by Marinelli,³¹ who noted that the hydrogenous/deutero-genous approach could be expected to work only when a "pure" reference beam of neutrons at the energy of interest was available so that the sensitivity ratio $(J_H - J_D) / J_H$ could be measured. Furthermore, although the CH_2 - CD_2 pair offered widely differing neutron sensitivities, the sensitivity ratio was strongly dependent upon energy, declining from 0.615 at 250 KeV to 0.10 at 10.0 MeV. This technique also became less reliable in

the presence of neutron-gamma fields dominated by the neutron component since it required the measurement of a small difference between two large currents. Marinelli also suggested the use of photon absorbers but pointed out the consequent perturbing effect on the neutron spectrum. Dainty found good agreement between experimentally measured and theoretically calculated results for dose to "standard tissue" at 1 cm from a 1 Ci PoBe neutron source.³²

Berlman at Argonne National Laboratory extended the twin chamber technique to scintillation detectors.³³ Scintillators with equal electron but dissimilar proton (i.e., hydrogen) content were chosen. Equal response for a given photon fluence was obtained by compensating differences in light emission by adjustments to the two detectors.

A significant advance in neutron dosimetry was reported by Rossi and Failla in 1956.³⁴ An ionization chamber was manufactured from "tissue-equivalent" plastic which had elemental composition resembling that of human tissue except for gross replacement of oxygen by carbon (Table 1). The counting gas was also compounded to be "tissue-equivalent", being somewhat closer to muscle in composition. Since these chambers and gases simulated tissue composition, they could be used for the determination of absorbed dose of any ionizing radiation. However, since neutron and gamma radiations differ in biological effectiveness, the biological effect depends on the proportion of dose due to each type of radiation. Rossi and Failla therefore used a neutron-insensitive chamber (graphite filled with carbon dioxide) as

Table 1. Early Tissue-Equivalent Materials

<u>Material</u>	<u>Compound</u>	<u>Fraction</u>	<u>Element</u>	<u>Z</u>	<u>Wt. %</u>
Solid (gel)	Water	66.2 %	H	1	9.8
	Gelatin	20.2	C	6	15.7
	Glycerol	5.2	N	7	3.6
	Sucrose	8.4	O	8	70.9
Plastic	Not given		H	1	10.1
			C	6	86.4
			N	7	3.5
			O	8	Trace
Gas 1	Hydrogen	38.1 vol. %	H	1	9.8
	Methane	22.2	C	6	15.7
	Nitrogen	2.1	N	7	3.5
	Oxygen	37.6	O	8	71.0
Gas 2	Methane	64.4	H	1	10.2
	Carbon dioxide	32.4	C	6	45.6
	Nitrogen	3.2	N	7	3.5
			O	8	40.7

Source: Rossi and Failla, (1956), Ref. 33

well as the tissue-equivalent chamber. The following pair of equations was suggested to describe the measurement:

$$(4.a) \quad T = 0.98 \cdot N + 1.07 \cdot \Gamma$$

$$(4.b) \quad C = K \cdot N + 1.07 \cdot \Gamma$$

where T and C are the responses of the tissue-equivalent and graphite-carbon dioxide chambers, respectively, and N and Γ are the

doses in tissue rads delivered by neutrons and gamma rays. The coefficient of the gamma dose is the reciprocal of the rad/roentgen conversion factor accepted at that time, namely 0.93 tissue rad/roentgen. The 0.98 coefficient was to correct for a presumed difference in W/e for electrons and protons. Rossi and Failla found that K was a neutron energy-dependent factor, ranging from 0.05 at 0.5 MeV to 0.25 at 14 MeV. This is the first description of a neutron/gamma field in terms of two simultaneous equations in two unknowns and four parameters in the literature.

The same set of equations was adopted by the National Committee on Radiation Protection and Measurements in 1961 with modified coefficients: ³⁵

$$(5.a) \quad T = 0.97 \cdot N + 1.03 \cdot \Gamma$$

$$(5.b) \quad C = k \cdot N + 1.03 \cdot \Gamma$$

The quantities T , C , N and Γ are defined to be the same as those defined by Rossi and Failla, although C represents the response of a Teflon- CO_2 chamber-gas combination, rather than graphite- CO_2 . The coefficient of Γ in both equations represents the reciprocal of the revised rad/roentgen conversion factor, namely 0.97 tissue rad/roentgen. The coefficient of N in equation (5.a) is a factor to allow for the difference in W/e between protons and electrons. The NCRP further stated that this procedure made possible the evaluation of radiation doses in a mixed field on the basis of the X-ray calibration

of the two chambers. Knowledge of the neutron energy was required so that the corresponding neutron sensitivity (k) could be determined from theoretical calculations. Rossi and Failla had observed values of k for the Teflon-CO₂ chamber ranging from 0.08 R/rad at 0.5 MeV to 0.24 R/rad at 8.0 MeV. The NCRP recommended a value of 0.16 R/rad for k if the neutron energy was unknown. The error in neutron dose was estimated at 10% from 0.5 to 8.0 MeV independent of the neutron/gamma dose ratio. The error in the gamma dose, however, ranges from negligible values with high gamma dose to large errors (up to 100% or more) for gamma dose fractions of less than 10%.

Further modification to the evaluation of the twin-ionization chamber technique was suggested by the International Commission on Radiological Units and Measurements in 1962.³⁶ The ICRU noted that an exposure of 1 R of approximately 1 MeV photons would lead to an absorbed dose of 0.96 rad in a tissue-equivalent ionization chamber. The same ionization would be produced in this chamber by an absorbed dose of neutrons equal to $0.96 \cdot W_{\alpha}/W_{\beta}$ rad, where W_{α} and W_{β} are the average energies required from an alpha particle or beta ray, respectively, to produce an ion pair in a gas. This ratio, W_{α}/W_{β} , was taken as 1.04, leading to the following response equation:

$$(6.a) \quad T = \frac{N}{1.00} + \frac{\Gamma}{0.96} = N + 1.04 \cdot \Gamma$$

where T is the ratio of the ionization produced in the chamber by Γ

tissue rads of photons and N tissue rads of neutrons to the ionization produced by 1 R of photons. The coefficient 1.00 represents the product $0.96 \cdot 1.04$, and the coefficient 0.96 is the tissue-rad/roentgen conversion factor. Similarly, the response equation for the neutron-insensitive Teflon- CO_2 chamber is given by:

$$(6.b) \quad C = 1.04 \cdot \Gamma + k \cdot N$$

where k was referred to as the neutron sensitivity of the non-hydrogenous chamber.

A modification of the formulation of the response equations for hydrogenous and non-hydrogenous ion chambers was proposed by Attix at the Naval Research Laboratory in 1973.³⁷ A tissue-equivalent ion chamber was used in conjunction with a graphite ion chamber to allow for neutron/gamma dose separation. The familiar equations relating chamber response to neutron and gamma dose were written in the following manner:

$$(7.) \quad Q_{n,\gamma} = A \cdot D_{\gamma} + B \cdot D_n$$

where $Q_{n,\gamma}$ is the response (in coulombs) per unit charge from the monitor ionization chamber of one of the two paired ion chambers, A is the response of that chamber per tissue-rad of gamma rays, B is the response of that chamber per tissue-rad of neutrons, and D_{γ} and D_n are the gamma and neutron doses in tissue rads under charged-particle

equilibrium conditions. The gamma sensitivity A, which is relatively independent of γ -ray energy for low-Z, tissue-like materials, can be determined for each chamber by means of a calibrated ^{60}Co gamma-ray exposure. This calibration is valid in mixed neutron/gamma fields if it is assumed or demonstrated that the 1.25 MeV mean energy of the ^{60}Co photons is representative of the gamma-ray spectrum encountered in the mixed field. However, there is no standard photon-free fast neutron beam of any energy available to determine the neutron sensitivity B.

The gamma and neutron sensitivities for ionization chambers may be calculated from the Bragg-Gray theory. W.S. Bragg³⁸ first proposed and L.H. Gray³⁹ later developed a theory relating the ionization caused by radiation in a small cavity to the absorbed dose in the walls of the surrounding medium. This relation may be stated as follows:

$$(8.) \quad E_m = S_g^m \cdot W_g \cdot J_g$$

where E_m is the energy deposited per unit mass in the surrounding medium (in ergs per gram), S_g^m is the dimensionless ratio of mass stopping powers for the secondary charged particles in the medium relative to that in the gas filling the cavity, W_g is the energy required for secondary charged particles to produce a unit charge in the gas (in ergs per coulomb) and J_g is the ionization produced in the gas (in coulombs per gram). This relation may be rewritten:

$$(8.a) \quad D_m = \frac{S_g^m \cdot W_g \cdot J_g}{100}$$

where D_m is the dose to the medium in rads, since the rad is defined as 100 ergs/gram. The ionization in the cavity J_g may be expressed as the ratio $Q_g/(V \cdot \rho)$ where Q_g is the ionization produced in the cavity (in coulombs), V is the volume of the cavity (in cm^3), and ρ is the density of the gas (in g/cm^3). The Bragg-Gray relation may now be written as:

$$(8.b) \quad D_m = \frac{S_g^m \cdot W_g \cdot Q_g}{100 \cdot V \cdot \rho}$$

For a truly tissue-equivalent ionization chamber in a pure photon field equation (8.b) may be rewritten as:

$$(8.c) \quad D_\gamma = \frac{(S_g^t)_\gamma \cdot W_\gamma \cdot Q_\gamma}{100 \cdot V \cdot \rho}$$

where D_γ is the absorbed gamma dose, $(S_g^t)_\gamma$ is the ratio of the effective mass stopping powers for secondary electrons in tissue relative to gas, W_γ is the energy per unit charge in the gas for γ -rays, and Q_γ is the response in coulombs due to gamma rays. Equation (7.) may be simplified in the absence of neutron radiation to give:

$$(7.a) \quad Q_\gamma = \Lambda \cdot D_\gamma$$

All the ionization in the ion chamber is now caused by photons so

equation (7.a) may be inserted into equation (8.c) and a theoretical value for the photon sensitivity may be calculated:

$$(9.a) \quad A \equiv \frac{Q_{\gamma}}{D_{\gamma}} = \frac{100 \cdot V \cdot \rho}{(S_g^t)_{\gamma} \cdot W_{\gamma}}$$

Similarly, the same ionization chamber would have a neutron sensitivity given by:

$$(9.b) \quad B \equiv \frac{Q_n}{D_n} = \frac{100 \cdot V \cdot \rho}{(S_g^t)_n \cdot W_n} \approx \frac{100 \cdot V \cdot \rho}{(S_g^t)_p \cdot W_{\alpha}}$$

where $(S_g^t)_n$ is the mean mass stopping power ratio for the secondary charged particles from neutrons, W_n is the energy per unit charge produced in the gas by fast neutrons, $(S_g^t)_p$ is the mean mass stopping power ratio for protons and W_{α} is the energy per unit charge produced in the gas by alpha particles. The latter relation is only approximate and is used due to availability of better data for S_g^t and W (for protons and alphas, respectively) than for neutrons. Moreover, W is only very weakly dependent upon the type or energy of the radiation, and protons are the main component of the charged particles crossing the ion chamber cavity.

The ratio of the neutron sensitivity to the gamma sensitivity is found by dividing equation (9.b) by equation (9.a):

$$(9.c) \quad \frac{B}{A} = \frac{(S_g^t)_\gamma \cdot W_\gamma}{(S_g^t)_n \cdot W_n} \approx \frac{(S_g^t)_\gamma \cdot W_\gamma}{(S_g^t)_p \cdot W_\alpha}$$

The preceding argument can be extended to the case of non-tissue-equivalent ion chambers by application of the Bragg-Gray relation. Equation (9.a) may be re-written for the case of a non-tissue equivalent ion chamber as follows:

$$(10.) \quad A \equiv \frac{Q_\gamma}{(D_\gamma)_t} = \frac{Q_\gamma}{(D_\gamma)_x} \cdot \frac{(D_\gamma)_x}{(D_\gamma)_t} = \frac{Q_\gamma}{(D_\gamma)_x} \cdot \frac{(\mu_{en}/\rho)_{\gamma,x}}{(\mu_{en}/\rho)_{\gamma,t}} \text{ CPE}$$

where the subscript t signifies real muscle tissue, x refers to the chamber wall material and CPE indicates the condition of charged particle equilibrium. This condition permits the substitution of the mass energy absorption coefficients for the corresponding dose ratio. Equation (9.a) may be re-written as:

$$(11.) \quad \frac{Q_\gamma}{(D_\gamma)_x} = \frac{100 \cdot V \cdot \rho}{(S_g^x)_\gamma \cdot W_\gamma}$$

This relation may be substituted into equation (10.) to obtain:

$$(12.) \quad A \equiv \frac{Q_\gamma}{(D_\gamma)_x} = \frac{(\mu_{en}/\rho)_{\gamma,x}}{(\mu_{en}/\rho)_{\gamma,t}} \cdot \frac{100 \cdot V \cdot \rho}{(S_g^x)_\gamma \cdot W_\gamma}$$

Equations for the corresponding neutron case are as follows:

$$(13.) \quad B \equiv \frac{Q_n}{(D_n)_t} = \frac{Q_n}{(D_n)_n} \cdot \frac{(D_n)_x}{(D_n)_t} = \frac{Q_n}{(D_n)_x} \cdot \frac{k_{n,x}}{k_{n,t}} \text{ CPE}$$

$$(14.) \quad \frac{Q_n}{(D_n)_x} = \frac{100 \cdot V \cdot \rho}{(S_g^x)_n \cdot W_n} \approx \frac{100 \cdot V \cdot \rho}{(S_g^x)_p \cdot W_\alpha}$$

$$(15.) \quad B = \frac{k_{n,x}}{k_{n,t}} \cdot \frac{100 \cdot V \cdot \rho}{(S_g^x)_n \cdot W_n} \approx \frac{k_{n,x}}{k_{n,t}} \cdot \frac{100 \cdot V \cdot \rho}{(S_g^x)_p \cdot W_\alpha}$$

where $k_{n,x}$ and $k_{n,t}$ are the fluence-to-kerma conversion factors for the neutrons (in units of rads per n/cm^2) in wall material x or in tissue, respectively. As before, equation (15.) can be divided by equation (12.) to obtain:

$$(16.) \quad \frac{B}{A} = \frac{k_{n,x}}{k_{n,t}} \cdot \frac{(\mu_{en}/\rho)_{\gamma,t}}{(\mu_{en}/\rho)_{\gamma,x}} \cdot \frac{(S_g^x)_\gamma \cdot W_\gamma}{(S_g^x)_n \cdot W_n}$$

$$\approx \frac{k_{n,x}}{k_{n,t}} \cdot \frac{(\mu_{en}/\rho)_{\gamma,t}}{(\mu_{en}/\rho)_{\gamma,x}} \cdot \frac{(S_g^x)_\gamma \cdot W_\gamma}{(S_g^x)_p \cdot W_\alpha}$$

The chamber gas g can be chosen to match the chamber wall (x) in stopping power, thus effectively making the S ratio unity (this can also be done for tissue-equivalent chambers with tissue equivalent gas). The ratio B/A then depends on the kerma ratio, energy absorption coefficient ratio and the W ratio. The most important of these cases is the case of the TE plastic-TE gas chamber-gas combination. The

commonly used A-150 muscle-equivalent plastic contains much more carbon and far less oxygen than real muscle tissue. However, since oxygen and carbon have practically identical (μ_{en}/ρ) values over a wide range of gamma energies, the ratio $(\mu_{\text{en}}/\rho)_{\gamma,t} / (\mu_{\text{en}}/\rho)_{\gamma,x}$ may be taken as unity. The hydrogen content of A-150 plastic and the methane-based TE gas suggested by Rossi and Failla are carefully chosen to match that of ICRU muscle tissue. The discrepancy in carbon and oxygen content does cause the fluence-to-kerma ratio to be significantly different from unity, though. The weighted average of $(k_{n,\text{TE plastic}}/k_{n,\text{muscle tissue}})$ over the neutron spectrum at the Naval Research Laboratory cyclotron was found to be 1.047. Assuming a value of W_{γ}/W_{α} of 0.948 for TE gas, the B/A for the TE plastic chamber was approximately 0.99. The same kerma ratio for 14.8 MeV neutrons may be computed using kerma values given by Caswell and Coyne.⁴⁰ This ratio is $(7.01 \cdot 10^{-9})/(6.75 \cdot 10^{-9})$ or 1.051, giving a B/A for the TE chamber in a 14.8 MeV neutron beam of 0.996.

CHAPTER II

INSTRUMENTATION AND METHODS

a. Lead Filtration Method

A lead filtration method was devised by Attix at the Naval Research Laboratory to provide the first experimental way of determining the neutron sensitivity (B) of a non-hydrogenous ion chamber in relation to that for a "tissue equivalent" ion chamber.⁴¹ The technique used a series of lead filters in narrow-beam geometry to progressively remove most of the gamma rays while transmitting the neutrons. A total of six measurements was required at one specified point in the beam to calibrate one ion chamber. Measurements were made with the hydrogenous and non-hydrogenous chambers in a collimated unfiltered beam, in the same beam with a lead filter, and in the same beam with a massive steel plug blocking the beam. It is possible to write six simultaneous equations in six unknowns which describe this situation (Table 2). Each equation is of the form of Eq. 7, with two modifications. The equations have been divided through by the (known) gamma sensitivity A to yield:

$$(17.) \quad \frac{Q}{A} = D_{\gamma} + \frac{B}{A} \cdot D_n$$

Table 2. Simultaneous Equations for Lead Filtration Method

$$\begin{aligned}
 (1.) \quad (Q/A)_{TE} &= D_Y^a + D_Y^s + (B/A)_{TE} \cdot D_n \\
 (2.) \quad (Q/A)_x &= D_Y^a + D_Y^s + (B/A)_x \cdot D_n \\
 (3.) \quad (Q/A)_{TE} &= \exp(-\mu \cdot t_1) \cdot D_Y^a + D_Y^s + (B/A)_{TE} \cdot \exp(-\sigma \cdot t_1) \cdot D_n \\
 (4.) \quad (Q/A)_x &= \exp(-\mu \cdot t_1) \cdot D_Y^a + D_Y^s + (B/A)_x \cdot \exp(-\sigma \cdot t_1) \cdot D_n \\
 (5.) \quad (Q/A)_{TE} &= \exp(-\mu \cdot t_2) \cdot D_Y^a + D_Y^s + (B/A)_{TE} \cdot \exp(-\sigma \cdot t_2) \cdot D_n \\
 (6.) \quad (Q/A)_x &= \exp(-\mu \cdot t_2) \cdot D_Y^a + D_Y^s + (B/A)_x \cdot \exp(-\sigma \cdot t_2) \cdot D_n
 \end{aligned}$$

Unknowns: $D_Y^a, D_Y^s, D_n, \mu, \sigma, (B/A)_x$ Simplifying:

$$\begin{aligned}
 (1.) \quad (Q/A)_{TE} &= D_Y^a + D_Y^s + (B/A)_{TE} \cdot D_n \\
 (2.) \quad (Q/A)_x &= D_Y^a + D_Y^s + (B/A)_x \cdot D_n \\
 (3.) \quad (Q/A)_{TE} &= C_1 \cdot D_Y^a + D_Y^s + (B/A)_{TE} \cdot D_n' \\
 (4.) \quad (Q/A)_x &= C_1 \cdot D_Y^a + D_Y^s + (B/A)_x \cdot D_n' \\
 (5.) \quad (Q/A)_{TE} &= C_2 \cdot D_Y^a + D_Y^s + (B/A)_{TE} \cdot D_n'' \\
 (6.) \quad (Q/A)_x &= C_2 \cdot D_Y^a + D_Y^s + (B/A)_x \cdot D_n''
 \end{aligned}$$

Unknowns: $D_Y^a, D_Y^s, D_n, D_n', D_n'', (B/A)_x$

Additionally, the gamma dose for this measurement has been partitioned into two components D_Y^a and D_Y^s representing gamma rays transmitted through the collimated aperture and those scattered into the detector (primarily from the neutron shield), respectively.

The equation set in Table 2 is transcendental due to the presence of the exponential attenuation terms and has no unique solution. Therefore, the coefficient of D_Y^a in Equations (3) and (4) of that table was replaced by a constant C_1 (equal to $\exp(-\mu \cdot t_1)$) and the coefficient of D_Y^s in Equations (5) and (6) was replaced by the constant C_2 (equal to $\exp(-\mu \cdot t_2)$), where μ is the linear attenuation coefficient for gamma radiation in lead and t_1 and t_2 are the lead thicknesses in cm. Additionally, the neutron dose D_n' replaced the term $\exp(-\sigma \cdot t_1) \cdot D_n$ in Eqs. (3) and (4) and D_n'' replaced the term $\exp(-\sigma \cdot t_2) \cdot D_n$ in Eqs. (5) and (6), where σ is the linear absorption coefficient for primary neutrons in lead. It is now possible to choose an estimated value of μ and solve these six equations for the six unknowns D_Y^a , D_Y^s , D_n , D_n' , D_n'' and the desired quantity, $(B/A)_x$ for the non-hydrogenous ion chamber. The ratio of neutron to photon sensitivity $(B/A)_{TE}$ for the tissue equivalent ion chamber is taken as a known quantity from the Bragg-Gray calculations summarized in Equation (16).

Attix obtained a value of 0.318 for a graphite ion chamber filled with air and 0.1585 for a magnesium ion chamber filled with flowing argon using this method for the cyclotron neutron spectrum at NRL, generated by 35 MeV deuterons on thick Be.⁴²

Improvements to this method were reported by Attix, Pearson and Capestrain at the University of Wisconsin.⁴³ Four thicknesses of lead were used as filters in narrow-beam geometry in this work, in contrast to the single lead filter plus steel plug used in the earlier work. An average value of (B/A) over four sets of measured points was obtained, thus providing an internal check on variation of this ratio with beam degradation due to the presence of the lead filters. Measurements were made at NRL, and with a beam of neutrons from a small, low-intensity $^3\text{H}(d,n)^4\text{He}$ (hereinafter referred to as D-T) neutron source at the University of Wisconsin. A value of $0.319 \pm 0.6\%$ S.D. for the (B/A) of a graphite ion chamber filled with air at NRL was reported using a value of μ for 0.2 MeV photons. Values of (B/A) for D-T neutrons were also reported, but varied widely from one set of filters to another, and included some negative values when μ for 4 MeV photons was used. These poor results were attributed to a low signal-to-noise ratio for this generator, and a confined working space near the target which resulted in less than ideal narrow-beam geometry. The most consistent estimates of these (B/A) ratios were for an assumed value of μ of 11.4 cm^{-1} , corresponding to 0.2 MeV photons. Values were as follows: $0.263 \pm 7.5\%$ S.D. for the graphite ion chamber filled with air, $0.168 \pm 9.2\%$ S.D. for the magnesium ion chamber filled with flowing argon, and $0.046 \pm 75\%$ S.D. for the GM counter.

Initial research for this thesis commenced with measurement of the B/A ratios for two ion chambers plus a miniature Geiger-Müller counter.⁴⁴ These measurements were performed at the University of

Wisconsin Gas Target Neutron Source, which provided an intense source of D-T neutrons,^{45,46} which could be collimated for narrow-beam geometry.

The ion chambers employed in this work were all Model IC-17 series guarded 3-terminal chambers manufactured by EG&G (Goleta, CA). All have walls thick enough to reach equilibrium for ^{60}Co γ -rays or for secondary charged particles produced by 14.8 MeV neutrons. The A-150 plastic "tissue equivalent" ion chamber (serial number 193-TG) has a volume of 0.946 cm^3 and a wall thickness of 0.508 cm (570 mg/cm^2). The atomic composition of A-150 plastic is given in Table 3,⁴⁷ along with the composition of muscle specified by the ICRU, and the present formulation of TE gas. A graphite-walled ion chamber (serial number 229-GG) of 2.43 cm^3 volume with walls of 0.303 cm thickness (530 mg/cm^2) was also used. A magnesium-walled ion chamber (serial number 225-MG) of volume 2.40 cm^3 with walls of 0.303 cm thickness (530 mg/cm^2) completed the set. A miniature GM counter (EG&G Model GM-2, serial number 1), patterned after the design described by Wagner and Hurst,⁴⁸ was also employed. It had a wall composed of 90 mg/cm^2 of iron, 200 mg/cm^2 of aluminum, 480 mg/cm^2 of ^6LiF and 807 mg/cm^2 of lead, designed to flatten out the inherent over-response of GM counters to low energy photons.

Each ion chamber was connected to a separate Keithley Model 616 Digital Electrometer. The responses of the GM counter and the monitor ionization chamber surrounding the neutron source were displayed on

Table 3. Composition of Muscle Tissue and Tissue Equivalent Materials

<u>Material</u>	<u>Compound</u>	<u>Percent</u>	<u>Element</u>	<u>Z</u>	<u>Wt. %</u>
A-150 Plastic ¹ (muscle equiv.)	Polyethylene	45.14 (by wt.)	H	1	10.2
	Nylon (Zytel 69)	35.22	C	6	76.8
	Carbon	16.06	N	7	3.6
	Calcium Fluoride	3.58	O	8	5.9
			Ca	20	1.8
			F	9	1.7
TE Gas ² (muscle equiv. with methane)	CO ₂	32.4 (by vol)	H	1	10.19
	N ₂	3.2	C	6	45.62
	CH ₄	64.4	N	7	3.52
			O	8	40.67
ICRU Muscle ³	C ₅ H ₄₀ O ₁₈ N	100%	H	1	10.2
			C	6	12.3
			N	7	3.5
			O	8	72.9
	Na, Mg, P, S, K, Ca		-	1.1	

¹ Smathers, et. al., (1977), Ref. 47.

² Rossi and Failla, (1956), Ref. 34.

³ ICRU 26, (1977), Ref. 2.

digital scalers. A specially designed electronic controller allowed the electrometer and scaler readings to be automatically started and stopped for meter and scaler readings to be automatically started and stopped for a chosen period of time. All measurements were made for a period of 100 seconds, and multiple readings were averaged at each point.

A flow of approximately $5\text{cm}^3/\text{min}$ of gas was passed through each chamber as follows: TE gas with the TE chamber, carbon dioxide with the graphite chamber, and argon with the magnesium chamber. The gamma-ray sensitivity coefficients A were measured in a calibrated beam of photons from a ^{60}Co teletherapy source.

Measurements were made in air at 25cm from the collimator of the Gas Target Neutron Source. These measurements were later repeated at a point 70cm from the collimator (Figure 1) when it was discovered from unfolding the six-equation set that the closer position had an

Fig. 1 Schematic plan view of the tritium gas target.

undesirably high level of scattered gamma radiation. The primary neutron beam was confined to a diameter of 6cm by a wax insert placed inside the Benelex (pressed wood) collimator. Measurements for all four dosimeters were made simultaneously. The three ion chambers were positioned in a plane 70cm from the collimator with their sensitive volumes within the useful beam. The GM counter was placed immediately behind this plane. No correction was made for the displacement of the GM counter with respect to the ion chambers, or for any shielding effect which the ion chambers may have had. Later measurements took these minor corrections into account.

All ionization measurements were made with both polarities and 400V bias on the ion chambers, while the GM counter was operated with +500V bias at all times. Dosimeter readings were normalized to the neutron output by dividing by the charge collected by the monitor ionization chamber. Raw results were corrected for temperature and pressure, electrometer correction factor, extrapolated to infinite voltage (for the ion chambers) and normalized to 1mCoulomb of charge in the monitor ion chamber. Typical dose rates at the ion chamber position were about 1 rad/min, and precision of the measurements was about 2% Standard Deviation (or $\pm 2\%$ FSD).

b. Water Phantom Measurements

Attix, August and Shapiro⁴⁹ demonstrated that measurements with TE plastic, graphite and magnesium ion chambers in the mixed n- γ beam issuing from a Benelex collimator can provide enough information to resolve the total dose into three components. The components were chosen to represent the primary fast (D_{fn}) and scattered slow (D_{sn}) neutron doses, and the total gamma dose, D_γ . The general form of the three equations to be solved for these three unknowns is shown in Table 4. This procedure requires a knowledge of the n/ γ sensitivity ratio for each dosimeter in both the primary and scattered radiation fields (namely, (B/A) and $(B/A)'$ in Table 4). The gamma-ray sensitivity A can be determined from a calibrated photon source. The response of each dosimeter must be measured at each point of interest.

Evaluation of $(B/A)'$ is dependent on the assumptions made about the scattered neutron spectrum. Kuchnir demonstrated in 1975 that $(B/A)_{TE}$ is relatively independent of energy over a wide range of neutron energies.⁵⁰ However, $(B/A)'$ for non-hydrogenous ionization chambers decreases strongly with decreasing neutron energy in the 1-15 MeV energy range. Attix et al⁵¹ assumed that the scattered neutrons could be represented by a monoenergetic component below 1 MeV, leading to the assumption that $(B/A)' \approx 0$ for a graphite chamber filled with air or CO₂ and for a magnesium chamber filled with argon. The revised

Table 4. Two- and Three-Dosimeter Methods

a. Two-Dosimeter Method

$$(1.) \quad (Q/A)_{TE} = D_Y + (B/A)_{TE} \cdot D_n$$

$$(2.) \quad (Q/A)_x = D_Y + (B/A)_x \cdot D_n \quad \phi$$

Unknowns: D_Y, D_n

b. Three-Dosimeter Method

$$(1.) \quad (Q/A)_{TE} = D_Y + (B/A)_{TE} \cdot D_{fn} + (B/A)_{TE} \cdot D_{sn}$$

$$(2.) \quad (Q/A)_x = D_Y + (B/A)_x \cdot D_{fn} + (B/A)_x \cdot D_{sn}$$

$$(3.) \quad (Q/A)_y = D_Y + (B/A)_y \cdot D_{fn} + (B/A)_y \cdot D_{sn}$$

Unknowns: D_{fn}, D_{sn}, D_Y

Simplifying:

$$(1.) \quad (Q/A) = D_Y + (B/A)_{TE} \cdot (D_{fn} + D_{sn})$$

$$(2.) \quad (Q/A) = D_Y + (B/A)_x \cdot D_{fn}$$

$$(3.) \quad (Q/A) = D_Y + (B/A)_y \cdot D_{fn}$$

equations representing this situation are then greatly simplified (Table 4).

It was suggested by Attix et al that this three-dosimeter method might be useful for n- γ dose separation in a water phantom where low-energy scattered neutrons cause an undetermined decrease in the effective (B/A) value for the non-hydrogenous dosimeters. The three-dosimeter method supplies the missing information on how much the (B/A) value is degraded by the scattered neutrons. An effective value of (B/A) for non-hydrogenous dosimeter x is suggested by the following:

$$(18.) \overline{(B/A)}_x = (B/A)_x \cdot \frac{D_{fn}}{D_{fn} + D_{sn}}$$

which equals (B/A)_x for D_{sn} = 0, and equals 0 for D_{fn} = 0 (a completely degraded neutron spectrum). This equation implicitly assumes that B/A is a linear function of neutron energy, obviously an over-simplification of a much more complex relation.

A series of measurements was therefore made in a water phantom utilizing the ion chambers and GM counter described previously.⁵² The water phantom used in this work consists of a Lucite box having dimensions of 40cm by 36cm with 41cm depth, and walls 7mm thick. The phantom was positioned outside the collimator with the front face 60cm from the face of the collimator (Figure 2). A depth of 10cm in phantom (measured from the front face of the phantom) down the beam axis

Fig. 2 Geometry for the water phantom measurements.

corresponded to the point at which the (B/A) measurements were made. The wax insert was removed for the in-phantom measurements and a square beam approximately 13cm by 13cm was produced at the phantom entrance, diverging to about 15cm by 15cm at the exit face. The beam centerline was about 13cm below the water surface. Each chamber was isolated from the water with a double layer rubber sheath and was approximately 19cm from the bottom of the tank.

Dosimeters were mounted on a remotely controlled X-Y scanner. The ionization chambers were mounted side-by-side in a plane parallel to the front face of the phantom and spaced at 2.2cm intervals (center to

center). The GM-counter was mounted directly behind the TE plastic chamber. The effect, if any, of the 1.5cm displacement of the GM-counter to a greater depth (which was partially compensated by the void caused by the presence of the ion chambers between the neutron source and the GM-counter) was not corrected for. Each chamber was irradiated at positions from 4.4cm on the west side of the beam centerline (which ran North-South) to 17.6cm east of the centerline. All dosimeters were simultaneously irradiated, then translated 2.2cm east and again irradiated, until the entire field at that depth was mapped out along the line of motion. This process was performed at depths of 5 and 10cm from the front face of the phantom. As in the lead filtration measurements, two or more 100 second irradiations were performed at each point, for both positive and negative polarities. Corrections were made for temperature, atmospheric pressure, electrometer calibration and ionic recombination losses. Precision was about 1-2% FSD. Dosimeter responses were normalized to the monitor ionization chamber response during the irradiation.

CHAPTER III

ION CHAMBER MEASUREMENTS

a. Lead Filtration Measurement Results

The lead filtration technique described in Chapter II was used to obtain the neutron-to-gamma sensitivity ratios (B/A) for the graphite ion chamber filled with CO_2 , the magnesium ion chamber filled with argon, and for the miniature GM counter.⁵³ Six different solution sets to the equations shown in Table 2 were found for the TE-GM pair, with the linear photon attenuation coefficient μ varied as an input parameter from 11.5cm^{-1} (corresponding to 0.2 MeV) to 0.471cm^{-1} (corresponding to 4.0 MeV). Solutions were obtained for four sets of filters at each value of μ . The smallest variance in the results for the five unknowns (D_n , D_γ^a , D_γ^s , $(B/A)_{\text{GM}}$ and σ) were obtained in each case when a value of 0.471cm^{-1} was used for μ . This is a reasonable result since it would be expected that most of the secondary gamma rays created by interactions of 14.8 MeV neutrons would be produced by inelastic reactions with nitrogen, oxygen, iron and copper and from hydrogen capture of thermal neutrons. All of these gamma radiations have energies of 2 to 8 MeV. Since the linear photon attenuation coefficient is constant within $\pm 15\%$ from 2 to 10 MeV, a value of 0.471cm^{-1} was used throughout this work. The B/A for the GM counter was thus found to be $0.186 \pm 6.3\%$ Standard Deviation (6.3% FSD).

The six-equation set was used to unfold the results of measurements made with the graphite and magnesium ion chambers utilizing a value of 0.471cm^{-1} for μ . Fractional standard deviations of the results were significantly larger for the TE-Mg pair (4 to 46% FSD) than for the TE-GM pair (1 to 7% FSD). The results for the TE-C pair were even worse, ranging from 3 to 380% FSD, and including a physically inadmissible negative mean value for D_Y^a . These results were not surprising, since the two-dosimeter method works best when one dosimeter is very neutron insensitive. Therefore, the results for D_Y^a , D_Y^s and μ from the TE-GM pair were used as input data to obtain the B/A for the non-hydrogenous ion chambers. An evaluation of B/A for each ion chamber could then be obtained for each lead filter used and for the open beam. The value of $(B/A)_C$ obtained in this manner was $0.297 \pm 2.6\%$ FSD, while $(B/A)_{Mg}$ was found to be $0.172 \pm 1.7\%$ FSD. These results were in reasonable agreement with those previously measured by Attix.

The values thus obtained in 1979 for these dosimeters were immediately used in the unfolding of ion chamber measurements made in a water phantom (described in the following section). However, it became apparent sometime later that there were several defects in these measurements. First, the response of the GM counter was not corrected for deadtime. It had been assumed at the time of the measurements that the relatively low counting rates in these experiments (e.g., 100 to 1000Hz) resulted in a negligible counting rate correction. However, these rates actually result in a 0.3 to 3.4% correction, if the

algorithm suggested by Evans ⁵⁴ is utilized. The form of the correction is as follows:

$$(19.) \quad N = \frac{n}{(1 - n \cdot \tau)}$$

where N is the true counting rate, n is the observed counting rate, and τ is the deadtime (which is given by the manufacturer as 33 μ seconds).

The second error was revealed when later measurements were performed in which the TE plastic ion chamber was used to traverse the useful beam area immediately outside the collimator. These scans showed a distinct lack of beam uniformity, especially in the exact center of the field, where the neutron beam passed through a 2cm diameter cooling-water pipe at the end of the 55cm-long gas target vessel. This problem was subsequently alleviated by construction of a paraffin compensating filter which was fitted around the water pipe to cause equal attenuation of the entire useful beam before passing through the collimator. Additionally, the measurement technique was modified to place each ion chamber in the exact geometric center of the beam (now indicated by a 5mm diameter HeNe laser spot), at the same distance from the source. Previously, all four dosimeters were exposed simultaneously in the non-uniform field of the filtered beam, with the GM counter in a plane 1.5cm behind the ion chambers.

Finally, it was noted that the responses of each ion chamber and the GM counter (when normalized by the response of the 55cm-long TE plastic monitor ion chamber which surrounds the target vessel) were not always constant. The normalized responses were found to increase linearly with the gas pressure in the target vessel. Since the present target vessel suffers a steadily increasing pressure change with the beam on, it was necessary to correct for this phenomenon. A 20% pressure change (from 5.0 to 6.0 Torr) in the target will normally cause an increase in the normalized response of the dosimeters of approximately 5%.

It was decided that the use of carbon dioxide as a matched gas with the graphite ionization chamber was unnecessarily troublesome, due at least partially to continuing problems with the porosity of the graphite wall of that device⁵⁵. The chamber was found to be quite sensitive to flow rate changes when surrounded by a tight fitting rubber sheath and used with flowing CO₂. It was prone to backfilling with air, resulting in severe distortion of the response. The graphite chamber is now used exclusively with static ambient air.

The lead-filtration method was therefore repeated with the above modifications to derive better values of B/A for the ion chambers and GM counter. The ion chambers were first re-calibrated at the University of Wisconsin Medical Physics Department ⁶⁰Co teletherapy source in an inter-comparison with the Midwest Center for Radiological Physics (MWCRP). The MWCRP PTW-119 NBS-calibrated ionization chamber

was used to obtain the gamma sensitivity for each of the ionization chambers. The precision of the measured charge ratios was less than 1%. Since the dose rate from the ^{60}Co source was too high to be used with the miniature GM counter, this device was calibrated with an open beam ^{137}Cs irradiator which had in turn been calibrated with the PTW-119 chamber. All of the Keithley 616 Electrometers were also calibrated utilizing the MWCRP Keithley Picoampere Source.

Each dosimeter was then sequentially exposed on the centerline of the unfiltered neutron beam, and then to the neutron beam filtered by 2.0, 4.2, 6.2 and 8.0cm of lead, and finally to the beam plugged by a 61cm steel rod. The B/A ratio for each chamber was derived by comparison with the response of the TE plastic chamber under the same conditions. The B/A of the TE plastic chamber was again determined from theoretical considerations to be 0.996. The graphite-air B/A was found to be 0.306, the magnesium-argon B/A was found to be 0.150 and $(\text{B/A})_{\text{GM}}$ was found to be 0.016. Results of the calibration of all of these devices are summarized in Table 5.

The values found for these dosimeters can be compared to those found by other investigators. Table 6 shows results at 14-15 MeV for the graphite ion chamber with both air and CO_2 , the magnesium chamber with argon, and for a variety of miniature GM counters. An alternative method of determining the neutron/gamma sensitivity ratio was developed by Kuchnir in 1975.⁵⁶ This method relies on the fact that most neutron generators (other than D-T) produce a neutron field which varies with

Table 5. Ion Chamber Calibration Data

<u>Chamber</u>	<u>Gas</u>	<u>B/A</u>	<u>A (nCoulombs/rad)⁽¹⁾</u>
TE (193-TG)	TE*	0.996	0.3440
C (229-GG)	Air	0.306	0.7733
Mg (225-MG)	Argon	0.150	1.194
GM (S.N. 1)	---	0.0160	1.742 $\cdot 10^6$ (2)

(1) Ion chamber ^{60}Co γ -ray response per rad of tissue absorbed dose under charged-particle equilibrium.

(2) GM counter response to ^{137}Cs gamma radiation in counts/rad

* Analyzed gas composition:

CH ₄	64.552 vol.%
CO ₂	32.258
N ₂	3.190

angle relative to the direction of the charged particle beam, while the gamma component is isotropic. The response of a dosimeter due to neutrons alone can therefore be unfolded from measurements made at more than one angle. Results at the bottom of Table 6 are taken from an International Neutron Dosimetry Intercomparison utilizing the D-T neutron source at the Radiological Research Accelerator Facility (RARAF) at Brookhaven in 1977.⁵⁷ All values for $(B/A)_{\text{C-CO}_2}$ in that report are high relative to Attix in 1979 and Waterman in 1973. This may be attributed to lack of understanding of the effect that the porosity of the graphite walls has on the composition of the gas contained in its sensitive volume. The closeness of the results for $(B/A)_{\text{C-Air}}$ obtained by Attix in 1974 and this work in 1982 lends confidence to the reliability of these results. Again, lead-filtration

Table 6. Summary of B/A Measurements at 14-15 MeV by Other Groups

<u>Workers</u>	<u>Date</u>	<u>C-CO₂</u>	<u>C-Air</u>	<u>Mg-Ar</u>	<u>GM</u>	<u>Method</u>
Attix, <u>et al</u>	1974		0.318			Lead filtration
Kuchnir, <u>et al</u>	1974			0.215		Spectral diff.
Cleland	1974		0.28			
Lewis & Young	1977				0.0130	Spectral diff.
Attix, <u>et al</u>	1977			0.1585		Lead filtration
Lewis & Hunt	1978				0.0163	Spectral diff.
Waterman, <u>et al</u>	1979	0.238*		0.180*		Spectral diff.
Attix, <u>et al</u>	1979	0.297		0.172	0.0186	Lead filtration
This work	1982		0.306	0.150	0.0160	Lead filtration
INDI REPORT:‡ (at 15.1 MeV)	1978					
ANL				0.238		
AFRRI				0.144		
CHRISTIE		0.356				
GSFF		0.33				
GSFM		0.341				
NIRS		0.365				
ORNL					0.	
RARAF					0.015	
TAMVEC				0.189		
TNO				0.14	0.	

* Data were interpolated between 14 and 16 MeV

‡ Reference 57.

results for the magnesium-argon pair obtained by Attix in 1977 and this work in 1982 are extremely close. The B/A for a miniature GM counter seems to be well-defined at 0.013 to 0.019.

b. Early Water-Phantom Measurements at U.W.

Measurements were made in the 14.8 MeV neutron beam at the Gas Target Neutron Source in April of 1979. The TE plastic ion chamber was filled with flowing methane-based TE gas, the graphite ion chamber was filled with flowing CO₂ and the magnesium ion chamber was filled with flowing argon. The dosimeters were positioned and translated as previously described.

The data obtained in these measurements were analyzed using both the paired-dosimeter and three-dosimeter technique represented by the equations in Table 4. Each dosimeter was paired with the TE chamber for analysis and three combinations of dosimeters were used in the three-dosimeter method. These techniques may be summarized as follows:

Method 1.: TE-C
Method 2.: TE-Mg
Method 3.: TE-GM
Method 4.: TE-C-Mg
Method 5.: TE-GM-Mg
Method 6.: TE-C-GM

Results of the two- and three-dosimeter techniques are summarized in Table 7 for the 5cm depth and Table 8 for the 10cm depth.

Table 7. Gamma Doses at 5cm Depth* April, 1979

Dist. Off Axis	Method (1) TE-C	Method (2) TE-Mg	Method (3) TE-GM	Method (4) TE-C-Mg	Method (5) TE-GM-Mg	Method (6) TE-C-GM
0cm	1.46 (2.7%)	2.68 (4.9%)	3.39 (6.2%)	3.86 (7.1%)	3.46 (6.4%)	3.48 (6.4%)
2.2	1.63 (3.0)	2.65 (4.9)	3.22 (5.9)	3.65 (6.7)	3.27 (6.0)	3.24 (6.1)
4.4	1.53 (2.8)	2.65 (4.9)	2.79 (5.1)	3.73 (6.9)	2.80 (5.2)	2.85 (5.2)
6.6	0.25 (0.5)	2.86 (5.3)	2.24 (4.1)	5.38 (9.9)	2.18 (4.0)	2.34 (4.3)
8.8	0.84 (1.5)	1.61 (3.0)	1.84 (3.4)	2.36 (4.4)	1.86 (3.4)	1.89 (3.5)
11.0	0.90 (1.7)	1.34 (2.5)	1.57 (2.9)	1.77 (3.3)	1.59 (2.9)	1.60 (2.9)
13.2	0.91 (1.7)	1.26 (2.3)	1.39 (2.6)	1.59 (2.9)	1.41 (2.6)	1.42 (2.6)
15.4	0.85 (1.6)	1.15 (2.1)	1.25 (2.3)	1.43 (2.6)	1.26 (2.3)	1.27 (2.3)
17.6	0.83 (1.5)	=== ---	1.15 (2.1)	--- ---	--- ---	1.16 (2.1)

*Doses given in tissue rads per 1 mCoulomb of charge collected by the monitor chamber. Figures in parentheses are percentages of total dose on beam axis (54.3 rads). Data were not obtained at this depth for the magnesium chamber at 17.6cm off-axis.

Table 8. Gamma Doses at 10cm Depth* April, 1979

Dist. Off Axis	Method (1) TE-C	Method (2) TE-Mg	Method (3) TE-GM	Method (4) TE-C-Mg	Method (5) TE-GM-Mg	Method (6) TE-C-GM
0cm	1.09 (2.6%)	2.35 (5.6%)	2.92 (7.0%)	3.56 (8.5%)	2.97 (7.1%)	3.00 (7.2%)
2.2	1.08 (2.6)	2.25 (5.4)	2.82 (6.7)	3.37 (8.1)	2.87 (6.9)	2.90 (6.9)
4.4	1.19 (2.8)	2.31 (5.5)	2.54 (6.1)	3.40 (8.1)	2.56 (6.1)	2.60 (6.2)
6.6	0.22 (0.5)	--- ---	2.14 (5.1)	--- ---	--- ---	2.23 (5.3)
13.8	0.70 (1.7)	1.24 (3.0)	1.33 (3.2)	1.77 (4.2)	1.33 (3.2)	1.35 (3.2)
15.0	0.89 (2.1)	1.12 (2.7)	1.19 (2.8)	1.35 (3.2)	1.20 (2.9)	1.21 (2.9)
17.2	0.83 (2.0)	--- ---	1.09 (2.6)	--- ---	--- ---	1.10 (2.6)

*Doses given in tissue rads per 1 mCoulomb of charge collected by the monitor chamber. Figures in parentheses are percentages of total dose on beam axis (41.8 rads). Data were not obtained for the magnesium chamber at this depth at 6.6 or 17.2cm off-axis.

Estimates of D_Y and the fraction of the total on-axis dose for each depth may be directly compared for each of the six unfolding methods in Tables 7 and 8. It should be noted that each of the six equation sets is constrained to give the same total dose estimate at any one point (e.g., 4.4cm off-axis at 5cm depth always yields 52.5 rads total dose per monitor unit). It is at once apparent that the TE-GM data set almost always gives the highest estimate of D_Y (and therefore the lowest estimate of D_n) of any of the three two-dosimeter

sets (Methods 1 through 3). The TE-C data set always yields the lowest estimate of D_γ of any of the three two-dosimeter sets (and therefore the highest estimate of D_n). The TE-Mg data set (Method 2) almost always gives results intermediate between the TE-GM and TE-C methods.

The implication of the two-dosimeter unfolding results is that the (B/A) value for the graphite chamber is probably too high in a water phantom. The neutron sensitivity of the graphite chamber should fall off gradually with decreasing neutron energy, since the fluence-to-kerma factor declines steadily from 14.8 MeV downward (Figure 3). Thus the graphite chamber, in phantom, significantly overestimates the neutron dose. The $(B/A)_{Mg}$ value is also too high in this situation, for the same reason, but to a lesser degree. The magnesium fluence-to-kerma factor also declines fairly rapidly from 14.8 MeV downward (Figure 4). However, since the (B/A) for the magnesium chamber with argon is less than half of that for the graphite chamber with air to begin with, the resulting error in the estimates of D_γ and D_n is less. By this same argument, the TE-GM pair should give the best estimate of neutron and gamma dose.

Unfolding methods 4, 5 and 6 partition the neutron dose into a fast component (D_{fn}) and a slow component (D_{sn}). Table 9 gives the results of this neutron dose division at both 5 and 10cm depth for all three methods. The fast neutron dose decreases with depth due to attenuation and energy-losing interactions while the proportion of dose

Fig. 3 kerma: H and C

Fig. 4 kerma: H and Mg

(Ordinate in $\text{rad-cm}^2/\text{neutron}$ in both figures)

at depth due to photons and slow neutrons increases. Several negative values are present in this table for D_{fn} and D_{sn} . These values are physically inadmissible and probably indicate a fault in input data. It should be noticed that these negative values most often occur when the ion chambers were straddling the beam edge, which was located at about 6-7cm off-axis. At this location both the accuracy of positioning and the differences in size of the TE ion chamber and GM counter compared to the larger magnesium and graphite ion chambers are probably important.

The three-dosimeter technique provides an estimate of the effective neutron-to-gamma sensitivity ratio $\overline{(B/A)}_x$ in-phantom for each of the two dosimeters used with the TE plastic ion chamber. These

Table 9. Partition of Neutron Dose at 5 and 10cm Depth* April, 1979

5cm Depth

Dist. Off- Axis	Method (4) (TE-C-Mg)		Method (5) (TE-GM-Mg)		Method (6) (TE-C-GM)	
	<u>D_{sn}</u>	<u>D_{fn}</u>	<u>D_{sn}</u>	<u>D_{fn}</u>	<u>D_{sn}</u>	<u>D_{fn}</u>
0cm	5.67	44.80	3.74	47.14	4.77	46.09
2.2	4.77	45.52	2.97	47.69	3.93	46.71
4.4	5.18	43.60	0.73	48.98	3.10	46.56
6.6	12.11	-3.39	-3.25	15.19	4.92	6.85
8.8	3.59	-0.30	1.21	2.57	2.48	1.28
11.0	2.06	0.03	1.21	1.06	1.66	0.60
13.2	1.62	-0.29	0.73	0.78	1.20	0.30
15.4	1.36	-0.37	0.54	0.63	0.98	0.18
17.6	---	---	---	---	0.81	0.11

10cm Depth

0cm	5.83	32.44	3.00	35.85	4.51	34.32
2.2	5.42	32.81	3.02	35.71	4.30	34.41
4.4	5.23	31.34	1.19	36.21	3.34	34.03
6.6	---	---	---	---	4.77	8.29
13.8	2.53	-1.20	0.45	1.31	1.56	0.18
15.0	1.07	0.02	0.36	0.87	0.74	0.49
17.2	---	---	---	---	0.65	0.17

* Doses given in tissue rads/lmC of charge in monitor chamber.

Table 10. $\overline{(B/A)}$ at 5 and 10cm Depth* April, 19795cm Depth

Dist Off Axis	$\overline{(B/A)}_C$		$\overline{(B/A)}_{Mg}$		$\overline{(B/A)}_{GM}$	
	Method (4) TE-C-Mg	Method (6) TE-C-GM	Method (4) TE-C-Mg	Method (5) TE-GM-Mg	Method (5) TE-GM-Mg	Method (6) TE-C-GM
0cm	.264	.269	.153	.160	.0172	.0169
2.2	.269	.274	.156	.162	.0175	.0172
4.4	.266	.279	.154	.170	.0183	.0174
6.6	-.115	.173	-.067	.219	.0237	.0108
8.8	-.028	.101	-.016	.117	.0127	.0063
11.0	.0045	.0787	.0026	.0804	.0087	.0049
13.2	-.064	.0596	.037	.0889	.0096	.0037
15.4	-.110	.0462	.064	.0925	.0100	.0029
17.6	---	.0371	---	---	---	.0023

10cm Depth

0cm	.252	.263	.146	.159	.0172	.0164
2.2	.255	.263	.148	.159	.0171	.0164
4.4	.255	.271	.148	.167	.0180	.0169
6.6	---	.212	---	---	---	.0133
13.8	.269	.0313	.156	.128	.0133	.0020
15.0	.0044	.118	.0025	.122	.0132	.0074
17.2	---	.0613	---	---	---	.0038

* Effective value of $(B/A)_x$ as defined in Eq. (18.), Chapter II.

estimates from the February, 1979 data are given in Table 10. The effective (B/A) for each dosimeter decreases slightly between the 5 and 10cm depths for points measured in the useful beam. Several negative values were noted on the beam edge and at points well off the beam axis. These negative values were attributed to poor signal-to-noise ratios at points out of the useful beam, leading to large uncertainties in the results. The estimated value of $\overline{(B/A)}_x$ drops rapidly for each dosimeter for points in the beam penumbra. The drop-off in $\overline{(B/A)}_x$ for the graphite and magnesium ion chambers occurs in precisely those areas where the average neutron energy is significantly lower than the incident 14.8 MeV beam. Since the fluence-to-kerma factors drop off rapidly with energy, it is to be expected that the effective (B/A) should also decline.

These early dose measurements in a water phantom demonstrated the feasibility and physical meaningfulness of the three-dosimeter method. The very low neutron sensitivity of the GM counter appeared to be extremely helpful in separating the neutron and gamma doses. Gamma dose estimates from each of the three unfolding methods (Methods 3, 5 and 6) in which the GM counter was utilized were dominated by the presence of the GM counter. These estimates were internally consistent and appeared to be physically meaningful. Results from the other three methods had predictable flaws which were exacerbated as exposure conditions became more dissimilar to those under which the graphite and magnesium counters were calibrated. Further measurements at other depths and along the beam axis were indicated.

c. Recent Water Phantom Results

Subsequent to the in-phantom dose measurements taken in April, 1979 it was noted that there were several deficiencies in the measurement techniques employed, method of analysis and the uniformity of the neutron field. The lead filtration measurements had previously been repeated for similar reasons. With these new calibrations in hand it was important to repeat the water phantom measurements utilizing better technique in a more uniform neutron field. Therefore new measurements were made in the same water phantom in June and July, 1982.

Prior to these measurements in the water-filled phantom, the TE plastic ion chamber and the CM counter were employed to obtain a neutron/gamma "map" of the field in air at a distance of 20cm from the outside face of the collimator. Three horizontal (east-west) traverses were performed at 2.54cm above, 2.54cm below, and coplanar with the beam centerline. Each scan extended from 7cm to the west of the centerline to 7cm east of the centerline. A three-dimensional plot of the neutron surface at 20cm from the collimator face was thus obtained (Figure 5). This plot indicates the success of the beam flattening filter which had recently been installed in creating a relatively uniform neutron field from about 4cm west of the centerline to 3cm east. The beam edge is relatively sharp, especially on the eastern side.

Fig. 5 Neutron dose surface in air at 20cm from the collimator.

The gamma contamination in the useful beam in air was found to be relatively constant at 3.0 to 3.6% of the total dose. The amount of contamination increases as a function of distance off-axis due to the fact that the gamma dose declines less rapidly than the neutron dose, indicating that the collimator provides a sharp edge for the neutron field while probably contributing to the scattered γ -ray dose.

A number of improvements in technique were employed in these new measurements. The experimental geometry was improved by attaching the dosimeters to the X-Y scanner in such a manner that all four dosimeters were held in the same Y plane (equidistant from the collimator) at all times (Figure 6). The spacing was altered from the previous experiment to balance the dosimeters on either side of the movable block from which they were suspended. The magnesium ion chamber was at $X=+12\text{cm}$ when the TE plastic ion chamber was at $X=0$, while the graphite ion chamber was at $X=+9\text{cm}$ and the GM counter was at $X=+3\text{cm}$. The front face

Fig. 6 Configuration of dosimeters in water phantom.

of the Lucite water phantom was positioned at 20cm from the exit face of the collimator instead of the 50cm position used earlier. The geometric beam width at that point is 10.4cm square, expanding to 11.7cm square at 20cm depth. The gas pressure inside the target vessel was carefully monitored for later corrections and kept between 5.0 and 6.2 Torr. The target pressure and GM counter deadtime corrections were used in data analysis.

A correction was also made for the attenuation of the neutron beam in the water displaced by the volume of the dosimeter; we made no correction previously. Shapiro et al⁵⁸ calculated a multiplicative correction factor of 0.970 for the IC-17 model TE ion chamber used in this work in the NRL neutron beam having an average energy of 14 MeV. A different method was used in calculating the displacement corrections used here, as suggested by Attix⁵⁹ and outlined in Appendix D. This new relationship yields a displacement factor of 0.981 for the IC-17 TE plastic, magnesium and graphite ion chamber, each of which has an outer radius of 1.143cm. The 1% difference between this calculation and the result of Shapiro et al may be due to their failure to consider the dosimeter as a probe used to deduce the absorbed dose in the center of a sphere of tissue-like material sufficient in radius to achieve charged particle equilibrium (CPE). The mass-radius of this sphere is taken to be 0.5 g/cm^2 for the present radiations. (Shapiro's correction factor in effect corrects twice for the attenuation of neutrons in the TE chamber wall). A displacement factor of 0.988 was calculated for the GM counter we used, which has an outer radius of

0.89cm. These factors were used in unfolding the in-phantom measurements.

Two sets of measurements were made along the beam axis with each dosimeter consecutively placed on the beam centerline at depths of 3.0, 5.0, 7.5, 10.0, 15.0 and 20.0cm. Cross-beam scans were made with each dosimeter consecutively placed on-axis, then 3.0, 6.0, 9.0 and 12.0cm off-axis, at depths of 3.0, 5.0, 7.5, 10.0, 15.0 and 20.0cm. The horizontal traverses in the phantom were performed not only to map out the beam profile, but also to choose a series of points which are characterized by different neutron and gamma spectral distributions. The point on the centerline and the point 3cm to the east are in the primary field defined by the beam collimator. The point 6cm to the east is located in the penumbral region, and the final two points are totally out of the primary field. Each of the three paired dosimeter methods and the three three-dosimeter methods used previously was employed to unfold the data acquired in-phantom.

Estimates of the gamma fraction at each measured point in phantom are given in Tables 11-17. All of the qualitative conclusions about the dosimeter responses and the results of the two- and three-dosimeter methods are confirmed by these data. The TE-GM pair almost always yields the highest estimate of gamma dose, the TE-C pair the lowest, and the TE-Mg pair is usually intermediate. This result is true for each point given for the two sets of on-axis measurements shown in Table 17. The two three-dosimeter methods which employ the GM counter

Table 11. Gamma Fractions at 3cm Depth (in Percent)* June, 1982

Dist. Off Axis	Method (1) TE-C	Method (2) TE-Mg	Method (3) TE-GM	Method (4) TE-C-Mg	Method (5) TE-GM-Mg	Method (6) TE-C-GM
0cm	0.82	5.94	4.95	8.15	5.15	5.32
3.0	0.55 (0.51)	5.82 (5.37)	4.70 (4.34)	8.13 (7.51)	4.88 (4.50)	5.07 (4.68)
6.0	2.56 (0.52)	4.17 (0.84)	12.69 (2.56)	4.13 (0.83)	14.12 (2.85)	13.54 (6.2)
9.0	40.29 (2.04)	40.37 (2.05)	39.83 (2.02)	37.46 (1.90)	41.40 (2.10)	41.17 (2.09)
12.0	54.31 (1.88)	40.77 (1.41)	50.71 (1.75)	28.78 (1.00)	53.72 (1.86)	52.28 (1.81)

*Numbers in parentheses are percentages of total dose on axis at 3 cm depth (114.6 rads per 1mCoulomb of charge in the monitor chamber).

Table 12. Gamma Fractions at 5cm Depth (in Percent)* June, 1982

Dist. Off Axis	Method (1) TE-C	Method (2) TE-Mg	Method (3) TE-GM	Method (4) TE-C-Mg	Method (5) TE-GM-Mg	Method (6) TE-C-GM
0cm	2.38	5.00	5.82	5.75	6.18	6.16
3.0	2.58 (2.31)	3.56 (3.18)	5.42 (4.85)	3.28 (2.93)	5.87 (5.26)	5.73 (5.12)
6.0	9.80 (1.73)	12.94 (2.29)	17.05 (3.01)	13.68 (2.42)	18.10 (3.20)	17.85 (3.15)
9.0	47.65 (2.69)	48.23 (2.72)	40.97 (2.31)	45.72 (2.58)	41.68 (2.35)	41.91 (2.36)
12.0	76.33 (2.80)	65.75 (2.41)	54.61 (2.00)	55.04 (2.02)	55.39 (2.03)	55.37 (2.03)

*Numbers in parentheses are percentages of total dose on axis at 3 cm depth (92.5 rads per 1mCoulomb of charge in the monitor chamber).

Table 13. Gamma Fractions at 7.5cm Depth (in Percent)* June, 1982

Dist. Off Axis	Method (1) TE-C	Method (2) TE-Mg	Method (3) TE-GM	Method (4) TE-C-Mg	Method (5) TE-GM-Mg	Method (6) TE-C-GM
0cm	3.84	5.68	6.44	5.81	6.84	6.78
3.0	3.75 (3.42)	3.59 (3.27)	6.06 (5.53)	2.49 (2.27)	6.62 (6.03)	6.38 (5.82)
6.0	0.93 (0.31)	-0.75 (-0.25)	10.98 (3.64)	-2.67 (-0.88)	12.62 (4.18)	11.74 (3.89)
9.0	27.39 (2.40)	22.38 (1.96)	32.00 (2.80)	17.16 (1.50)	34.16 (2.99)	33.19 (2.91)
12.0	51.06 (2.49)	42.07 (2.05)	48.22 (2.35)	33.28 (1.62)	50.60 (2.46)	49.61 (2.42)

* Numbers in parentheses are percentages of total dose on axis at 3 cm depth (79.5 rads per lmCoulomb of charge in the monitor chamber).

Table 14. Gamma Fractions at 10cm Depth (in Percent)* June, 1982

Dist. Off Axis	Method (1) TE-C	Method (2) TE-Mg	Method (3) TE-GM	Method (4) TE-C-Mg	Method (5) TE-GM-Mg	Method (6) TE-C-GM
0cm	4.25	6.27	7.20	6.53	7.63	7.57
3.0	4.57 (3.42)	5.16 (3.27)	6.92 (5.53)	4.52 (2.27)	7.42 (6.03)	7.25 (5.82)
6.0	8.33 (2.13)	9.45 (2.42)	16.06 (4.12)	8.96 (2.30)	17.35 (4.45)	16.87 (4.32)
9.0	32.99 (3.27)	29.97 (2.97)	32.76 (3.24)	25.80 (2.56)	34.24 (3.39)	33.76 (3.34)
12.0	56.77 (3.33)	43.66 (2.56)	46.73 (2.74)	32.15 (1.89)	48.70 (2.86)	47.75 (2.80)

* Numbers in parentheses are percentages of total dose on axis at 3 cm depth (62.1 rads per lmCoulomb of charge in the monitor chamber).

Table 15. Gamma Fractions at 15cm Depth (in Percent)* June, 1982

Dist. Off Axis	Method (1) TE-C	Method (2) TE-Mg	Method (3) TE-GM	Method (4) TE-C-Mg	Method (5) TE-GM-Mg	Method (6) TE-C-GM
0cm	3.69	10.03	8.43	13.01	8.65	8.90
3.0	4.64 (4.24)	9.86 (9.02)	8.36 (7.64)	12.11 (11.07)	8.58 (7.85)	8.79 (8.03)
6.0	8.69 (3.64)	9.79 (4.10)	13.94 (5.84)	9.29 (3.89)	14.92 (6.25)	14.59 (6.11)
9.0	29.93 (4.49)	23.53 (3.53)	29.98 (4.49)	17.42 (2.61)	31.71 (4.75)	30.89 (4.63)
12.0	33.41 (2.89)	35.79 (3.10)	43.36 (3.75)	34.97 (3.02)	45.66 (3.95)	45.04 (3.89)

* Numbers in parentheses are percentages of total dose on axis at 3 cm depth (43.7 rads per lmCoulomb of charge in the monitor chamber).

Table 16. Gamma Fractions at 20cm Depth (in Percent)* June, 1982

Dist. Off Axis	Method (1) TE-C	Method (2) TE-Mg	Method (3) TE-GM	Method (4) TE-C-Mg	Method (5) TE-GM-Mg	Method (6) TE-C-GM
0cm	8.23	10.25	11.30	10.31	11.89	11.80
3.0	8.14 (7.35)	8.03 (10.17)	11.27 (10.17)	6.78 (6.12)	12.07 (10.90)	11.77 (10.62)
6.0	8.95 (4.14)	7.37 (3.41)	16.34 (7.57)	5.16 (2.39)	17.88 (8.28)	17.15 (7.94)
9.0	41.44 (7.03)	32.27 (5.47)	36.88 (6.25)	23.86 (4.05)	38.69 (6.56)	37.84 (6.42)
12.0	43.35 (5.25)	38.76 (4.69)	45.48 (5.50)	33.10 (4.01)	47.79 (5.78)	46.95 (5.68)

* Numbers in parentheses are percentages of total dose on axis at 3 cm depth (25.7 rads per lmCoulomb of charge in the monitor chamber).

Table 17. Gamma Fractions on Axis (in Percent)*

June 9, 1982

Depth In Water	Method (1) TE-C	Method (2) TE-Mg	Method (3) TE-GM	Method (4) TE-C-Mg	Method (5) TE-GM-Mg	Method (6) TE-C-GM
3 cm	1.56 (1.27)	5.17 (4.20)	5.26 (4.27)	6.64 (5.40)	5.51 (4.48)	5.58 (4.53)
5 cm	2.39 (1.67)	5.50 (3.84)	5.75 (4.01)	6.62 (4.62)	6.04 (4.22)	6.07 (4.24)
10 cm	2.78 (1.32)	6.61 (3.13)	7.18 (3.40)	8.16 (3.86)	7.54 (3.57)	7.57 (3.58)
15 cm	3.92 (1.21)	6.79 (2.10)	8.77 (2.72)	7.70 (2.38)	9.32 (2.89)	9.23 (2.86)
20 cm	7.86 (1.52)	8.46 (1.64)	10.88 (2.11)	7.79 (1.51)	11.54 (2.23)	11.33 (2.19)

July 30, 1982

3 cm	2.65 (2.05)	4.66 (3.61)	5.19 (4.02)	4.96 (3.84)	5.53 (4.28)	5.49 (4.25)
5 cm	3.10 (2.14)	4.73 (3.27)	5.80 (4.01)	4.77 (3.30)	6.21 (4.29)	6.12 (4.23)
7.5cm	3.84 (2.21)	5.68 (3.28)	6.44 (3.72)	5.81 (3.35)	6.84 (3.95)	6.78 (3.91)
10 cm	5.55 (2.59)	6.36 (2.97)	7.32 (3.42)	5.77 (2.69)	7.77 (3.63)	7.66 (3.58)
15 cm	6.91 (2.07)	7.32 (2.19)	9.07 (2.72)	6.42 (1.92)	9.66 (2.90)	9.47 (2.84)
20 cm	8.23 (1.53)	10.25 (1.91)	11.29 (2.11)	10.28 (1.92)	11.89 (2.22)	11.80 (2.20)

*Numbers in parentheses are percentages of total incident dose (137.7 rads) on front face of phantom (in absence of phantom).

(Methods 5 and 6) give results that are almost always within 3-4% of those given by the TE-GM pair. The TE-C-Mg method (Method 4) gives the most inconsistent results of any unfolding technique.

An analysis of uncertainties in neutron and photon dose estimates from ion chamber measurements in a mixed field has been reported by Goodman.^{60,61} Using Goodman's terminology in which k_U replaces $(B/A)_x$ and Δk_U is the uncertainty in k_U , he deduced the expression for the uncertainty in the neutron and photon absorbed doses:

$$(20.) \quad \frac{\Delta D_n}{D_n} = \frac{\delta(k_U)}{[(1/k_U) - (1 + \delta(k_U))]}$$

$$(21.) \quad \frac{\Delta D_\gamma}{D_\gamma} = \frac{\delta(k_U)}{[1 + \delta(k_U)) - (1/k_U)]} \cdot \frac{D_n}{D_\gamma}$$

where $\delta(k_U)$ is the fractional uncertainty in the neutron-to-photon sensitivity ratio for the non-hydrogenous dosimeter. The ratio of these two equations may be expressed as:

$$(22.) \quad \frac{\Delta D_\gamma}{D_\gamma} = - \frac{D_n}{D_\gamma} \cdot \frac{\Delta D_n}{D_n}$$

These relationships demonstrate several pertinent features. Even a large uncertainty in $(B/A)_x$ for a dosimeter having a small (much less than unity) (B/A) contributes only a small uncertainty to the absorbed

dose estimates D_n and D_γ . It is important to minimize Δk_U for dosimeters having a moderate k_U , such as the graphite and magnesium ion chambers. The magnitude of the uncertainty in the gamma dose ΔD_γ is equal to the magnitude of the uncertainty in the neutron dose ΔD_n . However, the magnitude of the fractional uncertainty in D_γ is equal to (D_n/D_γ) times the fractional uncertainty in D_n . Thus for a mixed beam having a 5% gamma contamination, a 1% uncertainty in the neutron dose corresponds to a 20% uncertainty in the gamma dose.

The uncertainty relationship given by Goodman has several implications for this work. The dosimeter having the lowest neutron sensitivity, namely the GM counter, likely will yield the least uncertain dose estimates when used with the TE chamber. Conversely, the graphite ion chamber, having the highest neutron sensitivity of the three "neutron insensitive" dosimeters, probably must yield the most unreliable dose estimates. For example, a 40% uncertainty in $(B/A)_{GM}$ (taken to be 0.016) leads to only about $\pm 0.5\%$ contribution to the uncertainty in the neutron dose estimate while even a 10% uncertainty in $(B/A)_C$ (taken to be 0.306) leads to a $\pm 5.0\%$ contribution to the uncertainty of D_n . Again, for a neutron/photon dose ratio of 95% to 5%, the above uncertainties in $(B/A)_x$ would lead to uncertainties in the gamma dose estimate of 10% and 100%, respectively. The $(B/A)_x$ values experimentally determined in this work and their uncertainties may now be analyzed to assess their contribution to the uncertainty in neutron and gamma dose estimates (Table 18). The GM counter contributes no more than 3.1% to the gamma dose uncertainty even for a

Table 18. Uncertainties in n/γ Dose Estimates

Dosi- meter	5% D _γ		10% D _γ		50% D _γ	
	$\frac{\Delta D}{D}$	$\frac{\Delta D}{D}$	$\frac{\Delta D}{D}$	$\frac{\Delta D}{D}$	$\frac{\Delta D}{D}$	$\frac{\Delta D}{D}$
C-air	1.2%	22.2%	1.2%	10.5%	1.2%	1.2%
Mg-Ar	0.59%	11.2%	0.59%	5.3%	0.59%	0.59%
GM	0.16%	3.1%	0.16%	1.5%	0.16%	0.16%

<u>Dosimeter</u>	<u>(B/A)</u>	<u>Δ(B/A)</u>
C-air	0.306	0.008
Mg-Ar	0.150	0.005
GM	0.016	0.0016

5% gamma dose field, while the graphite chamber contributes over 22% for the same case.

The results of calculations of the effective $(B/A)_x$ in-phantom are given in Table 19 for the 5 and 10cm scans and in Table 20 for the two on-axis scans. The value of $(B/A)_C$ at first declines with greater depth and with distance off-axis, but rises again in each case. This is due to several factors, including extremely poor signal-to-noise ratios far off-axis, but also indicating a fundamental weakness in the assumptions of linearity of $(B/A)_x$ in phantom and that $(B/A)'_x = 0$. The same pattern holds for the estimate of $\overline{(B/A)}_{Mg}$ when using the TE-C-Mg method of analysis. The effective value of $(B/A)_{GM}$ is nearly constant at 0.015, again giving confidence to the TE-GM method.

Table 19. $\overline{(B/A)}$ at 5 and 10cm Depth* June 24, 19825cm Depth

Dist Off Axis	$\overline{(B/A)}_C$		$\overline{(B/A)}_{Mg}$		$\overline{(B/A)}_{GM}$	
	Method (4) TE-C-Mg	Method (6) TE-C-GM	Method (4) TE-C-Mg	Method (5) TE-GM-Mg	Method (5) TE-GM-Mg	Method (6) TE-C-GM
0.0	.281	.278	.138	.134	.014	.015
3.0	.301	.283	.148	.124	.013	.015
6.0	.275	.238	.135	.088	.009	.012
9.0	.331	.374	.162	.220	.023	.020
12.0	.633	.630	.310	.305	.033	.033

10cm Depth

0.0	.289	.281	.142	.132	.014	.015
3.0	.306	.286	.150	.124	.013	.015
6.0	.301	.235	.148	.061	.007	.012
9.0	.373	.298	.183	.078	.008	.016
12.0	.557	.425	.273	.039	.004	.022

* Effective value of $(B/A)_x$ as defined in Eq. (18.), Chapter II.

Table 20. $\overline{(B/A)}$ on Axis* June 9 and July 30, 1982June 9, 1982

Depth In	$\overline{(B/A)}_C$		$\overline{(B/A)}_{Mg}$		$\overline{(B/A)}_{GM}$	
	Method (4) TE-C-Mg	Method (6) TE-C-GM	Method (4) TE-C-Mg	Method (5) TE-GM-Mg	Method (5) TE-GM-Mg	Method (6) TE-C-GM
H ₂ O						
3.0	.268	.276	.132	.142	.015	.014
5.0	.275	.279	.135	.140	.015	.015
10.0	.265	.270	.130	.136	.015	.014
15.0	.278	.265	.136	.121	.013	.014
20.0	.307	.279	.150	.114	.012	.015

July 30, 1982

3.0	.289	.285	.142	.137	.015	.015
5.0	.294	.284	.144	.131	.014	.015
7.5	.292	.284	.143	.133	.014	.015
10.0	.304	.290	.149	.131	.014	.015
15.0	.310	.286	.152	.121	.013	.015
20.0	.290	.278	.142	.127	.013	.015

*Effective value of $(B/A)_x$ as defined in Eq. (18.), Chapter II.

Tables 21 and 22 give the partition of neutron dose into fast and slow components for the 5 and 10cm scans and for the on-axis scans, respectively. Several negative values for D_{sn} are noted at points far off-axis, indicating extremely poor signal-to-noise ratio at these points.

Table 21. Partition of Neutron Dose at 5 and 10cm Depth* June, 1982

5cm Depth

Dist. Off- Axis	Method (4) (TE-C-Mg)		Method (5) (TE-GM-Mg)		Method (6) (TE-C-GM)	
	<u>D_{sn}</u>	<u>D_{fn}</u>	<u>D_{sn}</u>	<u>D_{fn}</u>	<u>D_{sn}</u>	<u>D_{fn}</u>
0cm	7.05	80.15	9.30	77.50	7.90	78.92
3.0	1.30	78.76	13.47	64.43	5.89	72.14
6.0	1.44	12.68	5.53	7.86	2.98	10.45
9.0	-0.23	3.06	-1.42	4.46	-0.68	3.71
12.0	-1.63	3.16	-1.56	3.08	-1.61	3.12

10cm Depth

0cm	3.19	54.84	7.07	50.29	4.65	52.74
3.0	-0.06	52.09	8.88	41.56	3.31	47.22
6.0	0.23	14.26	7.79	5.37	3.08	10.15
9.0	-1.00	5.56	1.93	2.11	0.11	3.97
12.0	-2.03	4.50	1.38	0.49	-0.74	2.64

* Doses given in tissue rads per mCoulomb of charge collected by the monitor chamber.

Table 22. Partition of Neutron Dose on Axis* June-July, 1982

June 9, 1982

Depth In H_2O	Method (4) (TE-C-Mg)		Method (5) (TE-GM-Mg)		Method (6) (TE-C-GM)	
	D_{sn}	D_{fn}	D_{sn}	D_{fn}	D_{sn}	D_{fn}
3cm	12.88	91.58	5.74	99.98	10.19	95.46
5.0	9.19	80.54	6.05	84.23	8.01	82.24
10.0	7.94	51.90	5.63	54.61	7.07	53.15
15.0	3.65	35.73	7.57	31.11	5.13	33.59
20.0	-0.04	24.63	5.62	17.96	2.09	21.55

July 30, 1982

3cm	5.58	95.69	9.00	91.67	6.87	93.83
5.0	3.60	87.10	11.35	77.97	6.52	82.88
7.5	3.54	71.30	8.19	65.83	5.29	68.77
10.0	0.31	60.29	7.61	51.69	3.06	56.31
15.0	-0.46	39.09	7.12	30.17	2.40	34.97
20.0	1.19	21.83	3.53	19.08	2.08	20.56

* Doses given in tissue rads per 1 mCoulomb of charge collected by the monitor chamber.

Three-dimensional "maps" of the neutron and photon dose fields and the local gamma fractions are shown in Figures 7, 8 and 9. Although each scan was performed only in the east direction, symmetry around the

beam centerline was assumed for purposes of illustration. The qualitative results are just those that would be expected. The neutron dose profile is relatively flat from -3 to +3cm and rapidly falls off at 6cm off the centerline. The neutron dose also declines steadily from 3cm depth to 20cm depth. The gamma dose field generally follows the neutron dose field, although the magnitude is far less. The decline with depth and distance off-axis is much slower for the gamma dose than for the neutron dose field. The local gamma fraction shows that the gamma dose is a small component in the useful neutron dose field, even at considerable depth in phantom, but it rises very

Fig. 7 Neutron dose surface in phantom.

prominently out of the useful beam, becoming equal to the neutron dose at 12cm off-axis.

Fig. 8 Gamma dose surface in phantom.

Fig. 9 Gamma dose fraction in phantom.

d. Combined D-T and Co-60 Irradiation in Free Space

The University of Wisconsin Gas Target Neutron Source is uniquely equipped to provide simultaneous irradiation with 14.8 MeV D-T neutrons and Co-60 gamma rays.⁶² A Picker C9000 ^{60}Co Teletherapy Source has been modified and mounted on a movable cart which runs along the same rails as the neutron collimator (Figure 10). The ^{60}Co collimator may be positioned to direct a beam of gamma rays down the centerline of the neutron beam. Dosimeters or other devices may then be positioned on

Fig. 10 Dual beam irradiation facility.

the 90 by 110cm low-density aluminum honeycomb tray affixed perpendicularly to the neutron collimator and 21cm below the beamline. The gamma source may be positioned with its collimator immediately adjacent to the tray or at any distance up to about 2.5 meters away.

A series of irradiations was performed in air with the gamma source alone, the neutron source alone, and both sources simultaneously. The unfiltered output of the 1000 Curie ^{60}Co source was too high to allow the GM counter to be used without unacceptably high deadtime corrections. Therefore, a series of lead filters of 4.1, 5.4 and 9.5cm thickness were sequentially inserted into the gamma source collimator. The three ion chambers and the GM counter were mounted in a plane 30cm from the face of the collimator. The TE chamber and Mg chamber were vertically suspended (as before) in a fixed position from the X-Y scanner and were used with flowing gases (Figure 11). The GM counter was mounted horizontally from the west side of the tray with its sensitive volume immediately below the Mg chamber while the C chamber was mounted horizontally from the east with its sensitive volume just below the TE chamber. The neutron beam centerline was midway between the center of the TE chamber and the center of the Mg chamber.

The placement of the dosimeters illustrated in Figure 11 allowed all four dosimeters to be exposed to both a uniform neutron field and a uniform gamma field. The necessity of sequentially positioning the dosimeters in the center of the neutron field was thereby eliminated.

Fig. 11 Configuration of dosimeters for n/γ irradiation.

This was important for simultaneous n/γ irradiations since the neutron source output declines significantly with time, while the gamma field is virtually constant. Normalization to the neutron monitor chamber of measurements made in a combined D-T neutron and ^{60}Co -gamma field would therefore have been invalid.

The dosimeters were irradiated for a series of 100-second exposures with the ^{60}Co source filtered by 4.1, 5.4 and 9.5cm of lead. The dosimeters were also exposed to the neutron source alone at both a moderate output and at an output rate about 3.5 times higher than the first. The lower neutron source output was combined with the 4.1 and

5.4cm lead-filtered ^{60}Co beams, and the higher neutron source output was combined with each of the three filtered gamma beams.

Results of these measurements are given in Table 23. The gamma doses from the ^{60}Co source per 100 second reading could be determined directly from each of the four chambers by dividing its response by its A value. The gamma source filtered by 4.1cm lead (Experimental Condition I) yielded 166 ± 3 mrad per 100 seconds, 5.4cm lead filtration (Condition II) yielded 76.9 ± 1.9 mrad, and 9.5cm lead filtration (Condition III) gave 11.8 ± 0.5 mrad. The "unfolding" of these gamma-only irradiations yielded gamma fractions of $99.5 \pm 4.9\%$ from the six methods for the first filter, $101.9 \pm 4.8\%$ for the 5.4cm filter and $104.8 \pm 6.8\%$ for the 9.5cm filter. As with the water phantom data, the methods in which the GM counter appeared gave the most accurate results. The lower neutron output (Condition IV) gave 62.0 ± 0.1 neutron rads and 1.8 ± 0.1 gamma rads per 100 seconds, according to Methods 3, 5 and 6. The same three methods gave 218.6 ± 0.3 neutron rads and 6.5 ± 0.3 gamma rads for the higher neutron output (Condition V).

It was possible to predict in advance the neutron and photon dose rates when any of the first three experimental conditions was combined with either Condition 4 or 5. Below the results in Table 23 for each of the last five experimental conditions given in Table 23 is the predicted dose based on measurements of the separate neutron and gamma sources. The predicted response is not simply the algebraic sum of the

Table 23. n/γ Doses (mrads/100 sec.), D-T n and Co-60 γ Oct., 1982

Irrad [#] Cond.	Meth. (1) TE-C	Meth. (2) TE-Mg	Meth. (3) TE-GM	Meth. (4) TE-C-Mg	Meth. (5) TE-GM-Mg	Meth. (6) TE-C-GM
I.	(170)	(176)	(163)	(181)	(161)	(162)
II.	(79.7)	(82.5)	(76.1)	(84.3)	(75.5)	(76.0)
III.	(12.0)	(13.0)	(11.8)	(13.7)	(11.6)	(11.8)
IV.	63.3 (0.5)	61.1 (2.7)	62.0 (1.8)	59.6 (4.2)	62.1 (1.7)	62.0 (1.9)
V.	222.6 (2.6)	214.1 (11.0)	218.5 (6.6)	208.5 (16.6)	219.0 (6.1)	218.4 (6.7)
VI.	65.3 (170)	54.1 (181)	70.2 (165)	46.5 (188)	71.8 (163)	70.3 (165)
Predicted:		63 mrad n, 166 mrad γ				
VII.	65.6 (78.8)	59.5 (84.9)	66.6 (77.8)	55.4 (88.9)	67.3 (77.1)	66.6 (77.8)
Predicted:		62 mrad n, 78 mrad γ				
VIII.	142.7 (170)	131.6 (181)	147.0 (165)	124.1 (188)	148.6 (164)	147.2 (165)
Predicted:		140 mrad n, 168 mrad γ				
IX.	201.6 (81.6)	192.4 (87.5)	201.4 (81.8)	186.2 (96.9)	202.3 (80.9)	209.4 (81.8)
Predicted:		202 mrad n, 82 mrad γ				
X.	135.6 (12.4)	128.2 (20.0)	133.0 (15.0)	125.8 (24.6)	133.5 (14.5)	132.9 (15.1)
Predicted:		130 mrad n, 15.5 mrad γ				

- [#] I. Co-60 with 4.1cm lead filtration.(γ doses in paren.)
 II. Co-60 with 5.4cm lead filtration.
 III. Co-60 with 9.5cm lead filtration.
 IV. D-T neutrons alone, lower output.
 V. D-T neutrons alone, higher output.
 VI. D-T neutrons (low), + Co-60 (4.1cm).
 VII. D-T neutrons (low), + Co-60 (5.4cm).
 VIII. D-T neutrons (high), + Co-60 (4.1cm).
 IX. D-T neutrons (high), + Co-60 (5.4cm).
 X. D-T neutrons (high), + Co-60 (9.5cm).

separate neutron and gamma exposures, since the neutron source output declined throughout these measurements. The predicted doses were normalized to the recorded monitor chamber readings during the combined irradiations. The three GM methods (Methods 3, 5 and 6) give results extremely close to predictions for the gamma dose but yield 2-10% higher neutron doses than predicted.

CHAPTER IV

MONTE CARLO SIMULATION

a. MORSE-CG Code

Many common radiation shielding and dosimetry problems are inherently too complex to permit calculation of absorbed doses in closed mathematical form. The general time-dependent integro-differential Boltzmann transport equation may be used to describe any radiation field, but explicit solutions are not always possible. Herman Khan proposed in 1950 the use of stochastic (Monte Carlo) techniques to solve such problems.^{63,64} He identified virtually all of the principal concepts at that time, but widespread application of these techniques had to await the development of modern high-speed cost-efficient computers.

The MORSE (Monte Carlo Oak Ridge Stochastic Experiment) computer code was completed by Straker, Stevens, Irving and Cain in 1970 and is still in widespread use today, in revised editions.⁶⁵ The code allows for the transport of neutrons only, photons only, or neutrons with secondary photons in either the forward or adjoint (backward) mode. Unlike one-dimensional discrete ordinate codes such as ANISN⁶⁶ and two-dimensional discrete ordinate codes such as DOT,⁶⁷ MORSE does not provide a complete solution to the (simplified) Boltzmann equation everywhere in space. Rather, it provides a time-varying particle

fluence differentiated into energy and angle bins (if desired) at points and/or surfaces of interest. The MORSE-CG ⁶⁸ version of the code, however, permits full three-dimensional representation of the source-medium-detector geometry. The Combinatorial Geometry (CG) module allows the user to combine nine different kinds of three-dimensional solid bodies (e.g., spheres, cubes, arbitrary polyhedrons) into zones and regions descriptive of the problem. The code has been extensively validated by years of experience.

The MORSE-CG code, like all Monte Carlo codes, utilizes a pseudo-random number generator, which is actually a suite of computer codes, to provide "random" numbers for use in constructing hypothetical particle case histories. A fluence or absorbed dose estimate from MORSE-CG is obtained from one or more runs of the code, each containing perhaps 10^3 to 10^5 case histories. Reasonably reliable statistical estimates can usually be achieved with about this degree of effort. Cost of computing time is almost always the limiting factor in precision of computed results. The fundamental limitation of accuracy of the results is, of course, the uncertainty in the interaction cross-sections employed. It is interesting to note the vast disparity between the number of case histories used in these calculations vs. the actual number of particles created during ion chamber measurements. While 10^5 case histories would be almost prohibitively expensive to compute, the Gas Target Neutron Source creates about 10^{14} neutrons during a single 100-second irradiation. The $1000\text{Ci } ^{60}\text{Co}$ source would emit about $7.4 \cdot 10^{12}$ gamma rays during this same interval.

The process of creating each individual case history is referred to as the Random Walk. The logic of the Random Walk is shown below:

Figure 12.

The first operation consists of choosing source parameters. Starting particle type, location, energy and direction are randomly selected from distributions which incorporate probability density functions representative of the source to be simulated.

The second operation of the Random Walk consist of selecting a path length for the chosen particle to travel to its first collision. The path length L is typically selected from a distribution given by:

$$(23.) \quad P(L) = 1 - e^{-\Sigma_t(E)L}$$

where $P(L)$ is the probability of interaction at some path length $\leq L$, and $\Sigma_t(E)$ is the total macroscopic cross section for the particle of energy E (prior to collision) in the region to be traversed. The random number n (uniformly distributed on the interval $0.0 \leq n \leq 1.0$) is set equal to $P(L)$. Once a random number n has been chosen, Equation (23) may be inverted to give the path length L :

$$(24.) \quad L = \frac{\ln(1-n)}{\Sigma_t(E)} = \frac{\ln n}{\Sigma_t(E)}$$

since $1-n$ is distributed in the same manner as n .

The third operation of the Random Walk consists of selection of collision parameters at the site just chosen. Both the total interaction cross section Σ_t of the medium and the individual interaction cross sections σ_{ti} for each element i in the medium must be known. The total macroscopic cross section (in cm^{-1}) may be expressed in terms of the elemental cross sections (in $\text{cm}^2/\text{nucleus}$) as follows:⁶⁹

$$(25.) \quad \Sigma_t = \rho_1 \sigma_{t1} + \rho_2 \sigma_{t2} + \dots + \rho_n \sigma_{tn}$$

where ρ_i is the atomic density of element i (in $\text{nuclei}/\text{cm}^3$). The probability of a particular species being involved in the collision is given by the fraction of the total cross section represented by that species: $\rho_i \sigma_{ti} / \Sigma_t$. The nucleus involved can be obtained by selecting a new random number r and comparing it to a cumulative distribution function:

$$(26.) \quad P(k) = \frac{\sum_{i=1}^k \rho_i \sigma_{ti}}{\Sigma_t} = r$$

This relation may then be solved for k , where k can be any integer between 1 and n , inclusive. In like manner, the type of interaction can be determined by comparing another random number s to:

$$(27.) \quad P(i) = \frac{\sum_{j=1}^1 \sigma_{ij}}{\sigma_{ti}} = s$$

where the σ_{ij} are the various components of the different interactions composing the total cross section σ_{ti} of element i .

The fourth operation of the Random Walk consists of computing post-collision parameters for the particles which survive (or are created during) a collision. These parameters include the type, energy and direction of the surviving incident particles as well as those for any secondary particles created in the interaction. Calculation of these parameters depends on the type of interaction, which may include neutron elastic scattering, neutron inelastic scattering, fission, Compton photon scattering, photoelectric effect, pair and triplet production. Modification of the statistical weight W associated with each surviving particle is also performed. It is possible, in the interest of cost efficiency, to force a scattering event at each collision. The weight W of particles actually absorbed in collisions is reduced by multiplying by the non-absorption probability.

The fourth module of the Random Walk also includes particle scoring, which produces the final dose and/or fluence estimate. MORSE-CG performs calculations in batches of particles, and runs most efficiently when the number of batches and number of particles per batch are approximately equal. For example, 10,000 case histories might

be run as 100 batches of 100 particles per batch. Batch-to-batch statistics (Fractional Standard Deviations) are calculated by this module for each detector employed.

Four different types of estimation routines are provided with MORSE-CG. Subroutine RELCOL is termed a point detector (or "next event") estimator. This routine computes at every collision site S the contribution Φ_i that would be made to the flux at a given point D if a particle leaving that collision were forced to scatter through an angle θ_s in the direction of point D . The contribution Φ_i is given by:

$$(28.) \quad \Phi_i = W_0 \cdot \frac{\sigma_s(E)}{\sigma_t(E)} \cdot f(\theta_s, E) \cdot \frac{e^{-\Sigma_t(E')R}}{R^2}$$

where W_0 is the weight of the incident particle before collision, $\sigma_s(E)$ is the total scattering cross section of the element involved in the collision, $\sigma_t(E)$ is the total cross section of that element, $f(\theta_s, E)$ is the probability density per steradian of scattering through angle θ_s , $\Sigma_t(E')$ is the total macroscopic cross section of the scattered particle, and R is the distance between S and D . This estimation routine was used exclusively in this work. Other estimation routines include a boundary crossing surface estimator (Subroutine BDRYX), a regional track-length estimator ⁷⁰ and a regional real collision density estimator.

The MORSE-CG code also includes a package of statistical biasing routines and techniques (also called variance reduction or importance sampling routines) which have as their objective the reduction of variance of calculated fluences or doses for a given investment of computer time. Each of the stochastic selection processes illustrated in the Random Walk module may be altered to enhance unlikely but important events. Particles may be transported in a preferred direction, for example, or particles of a particular type may be either suppressed or enhanced. The MORSE-CG code allows for path-length stretching, "Russian roulette" and splitting. Russian roulette eliminates a specified fraction of particles below a specified statistical weight, and splitting creates two interesting particles in place of one. All such biases must be removed by adjustment of the statistical weight of the particles involved during the evaluation process. Since these variance reduction schemes often result in an increase in computer time per case history, their use is not always beneficial.

A given particle history is terminated when the particle's weight, energy or spatial position exceeds predetermined bounds. The code checks to determine whether more histories remain to be performed, and if so, returns to the first operation depicted in Figure 12. Otherwise, the code prints out the results.

The MORSE-CG code uses discrete energy bins in all calculations. Each energy bin utilizes a set of cross sections which represent average probabilities for each type of interaction. A particular collision may result in the surviving particle being down-scattered into a lower energy group. Secondary photon production is treated as a special case of down-scattering, while photo-neutron production could be treated as a type of up-scattering. The time required to run a given problem rises as the number of particle energy groups is increased. Preliminary calculations were performed using the 22-neutron, 18-gamma ray CASK data set.⁷¹ All final calculations were done utilizing the 37-neutron, 21-gamma ray FEWC1 data set, based on cross sections from the ENDF/B-IV compilation.⁷²

b. Simulation of GTNS Facility with Water Phantom

The MORSE-CG computer code (CCC-203C) was acquired from the Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory in April, 1980.⁷³ The IBM version of the code was chosen for translation into useable form since the 16 bit word length of the IBM computer is the same as that for the Digital Equipment Corporation VAX 11/780 computer used in this work. The length and complexity of the MORSE-CG package, which contains over 9,000 lines of FORTRAN IV code, caused substantial difficulty. The sample problem provided with the code for verification of operation was not successfully run for three months. The sample problem consisted of the degenerate one-dimensional

case of a point source in an isotropic medium, utilizing spherical boundary detectors. Generalization to extended sources, restricted source particle directions, multiple media, full three-dimensional geometry and use of the point detector estimator occupied extensive effort over the next 12 months. The final suite of codes and subroutines produced by these efforts has been contributed to the national radiation shielding code library at RSIC.

The source-collimator-phantom geometry characteristic of the GTNS facility was simulated for these calculations (Figure 13). Although

Fig. 13 Computer simulation of GTNS facility.

the code would allow for the creation of a physically realistic 76-cm thick collimator composed of concrete, iron and wax, it is more efficient to consider the collimator to be a perfect absorber. A 55cm line source was used to simulate the 55cm-long 2.6cm-diameter gas target. The beam line was chosen as the +z-axis of the code, with the origin at the entrance of the deuteron beam into the target vessel. The +x direction was chosen to be up (with respect to the floor) and the +y direction was chosen to be toward the west. A starting z coordinate for each source particle was randomly chosen between 0 and 55.0cm. The starting x and y coordinates were restricted to be 0.0cm, thus defining a line source. The starting direction cosines chosen for each particle confined the beam to a cone which just enclosed the 9.1cm square exit from the Benelex collimator insert, since this was the limiting aperture. Particles whose initial trajectories would carry them outside the 9.1cm square aperture were rejected and a new starting location was chosen. In this manner it was possible to produce a beam of neutrons which randomly filled the volume of space limited by the collimator. The penumbral region expected of a line source viewed through a square aperture was reproduced as well.

A mathematical problem was presented by this square beam. The Subroutine SDATA computes the contribution made to the dose or flux at a point by uncollided source particles, since they are not analyzed by the RELCOL routine. The contribution Φ_1 for an isotropic point source

is given by the expression:

$$(29.) \quad \Phi_1 = \frac{(WATE)e^{ARG}}{4\pi R^2}$$

where WATE is the statistical weight of the source particle, ARG is the negative of the number of mean free paths from the source to the detector and R is the distance from the source to the detector. It is at once apparent that for a source confined to one hemisphere, the numerator in this equation should be changed to $2\pi R^2$, since the solid angle of such a source is simply 2π . For a cone-shaped beam, the numerator is $2\pi(1-\cos(\theta_{\max}))$, where $\theta_{\max}$ is the maximum angle between the z-axis and particles entering such a cone. However, the correct numerator for a beam limited to a square is not easy to compute. To a very good approximation, the expression for the solid angle subtended by a cone-shaped beam can be multiplied by $2/\pi$, the ratio of the area of a square to the circle in which it is inscribed. Gary Herling of the Naval Research Laboratory, however, generously contributed a nearly exact solution to this problem.⁷⁴ Herling's expression is as follows:

$$(30.) \quad \Phi_1 = \frac{(WATE) \cdot e^{ARG}}{4 \sin^{-1}[(1+b^2)^{-1/2}]} , \quad b = \frac{2d}{l}$$

where d is the distance from the origin to that limiting aperture, and l is the width of the aperture. This expression was used in Subroutine SDATA.

One additional problem was posed by the use of the point detector estimator within a water phantom. Kalos has shown that an infinite variance may result from this technique, caused by collisions very near the point detector (due to the presence of the $1/R^2$ term in the flux estimate, Eq. (28)).⁷⁵ Although this seldom presents a problem in gaseous media, due to the rarity of collisions very near the detector, it often causes problems in condensed media. A solution to this problem was employed by Herling for calculations in a water phantom irradiated by a D-T neutron beam.⁷⁶ He surrounded the point detector by a cylindrical void, thus eliminating the possibility of collisions near the detector. All MORSE-CG calculations were performed in phantom with point detectors surrounded by a spherical void of 0.5cm radius. This coincidentally happens to be nearly the same radius as the void introduced by the gaseous (sensitive) volume of a Model IC-17 ion chamber in-phantom. Calculations were performed with one detector at a time to avoid disturbing the interaction media.

A number of assumptions were made throughout these calculations:

- (1.) The source consisted entirely of a beam of monoenergetic neutrons from the 14.2 to 14.9 MeV neutron energy bin.
- (2.) No interactions with the target material (e.g., copper, water) were permitted.
- (3.) The collimator was assumed to be a perfect absorber.
- (4.) Air between the target and phantom was replaced by an interaction-free void.
- (5.) The Lucite walls of the phantom were replaced by an equal thickness of water.

All of these assumptions were investigated, and only assumptions (1)

and (3) were found to be questionable. Interactions of D-T neutrons with the target material were found to be insignificant along the beam axis, thus verifying the second assumption. Fewer than 1 neutron per 1000 emitted interacted in air between the source and the front face of the phantom, proving assumption (4) to be valid. The gross attenuation of D-T neutrons in Lucite was found to be only about 7% greater than that in water. Since the walls of the phantom are 0.7cm thick and dose calculations were performed at depths of 3.0cm or greater, this difference was shown to be insignificant. Scattered gamma rays from the collimator were found to contribute less than 0.2% of the total dose in air at the center of the front face of the phantom. However, a later revision of the code allowed for the emission of 2.2 MeV hydrogen capture gamma rays (produced by thermal neutrons) from the face of the collimator, as described later. Determination of the origin of the ~3% gamma dose fraction in air at the phantom proved to be the most difficult problem encountered in the use of this Monte Carlo code.

CHAPTER V

COMPARISON OF EXPERIMENTAL AND THEORETICAL RESULTS

a. Comparison of MORSE-CG with Pure D-T Source

Herling ⁷⁷ has compared the results of calculations utilizing MORSE-CG with experimental measurements made by Attix et al ⁷⁸ for the INDI ⁷⁹ project, and found good agreement. These calculations treated the Radiological Research Accelerator Facility as a point source of D-T neutrons. Neutron and photon fluxes were calculated at 5, 10 and 20cm depth along the beam centerline inside a cylindrical water phantom. The gamma fraction $R = 100 \cdot D_{\gamma}/D_t$ was also calculated (Table 24). Results were compared to those measured by Attix et al. These calculations were repeated at the University of Wisconsin and were well reproduced (Table 24).

Table 24. Gamma fractions* for INDI experiment

<u>Depth</u>	<u>Herling</u>	<u>Attix</u>	<u>This work</u>
5 cm	3.5 ± 0.2	2.4	3.6 ± 0.2
10	5.4 ± 0.2	6.0	5.3 ± 0.2
20	10.2 ± 0.7	10	10.3 ± 0.6

*Gamma fraction defined as $100 \cdot D_{\gamma} / D_{\text{total}}$

Calculations were performed to simulate the CTNS facility using the code and techniques described in Chapter IV. The FEWG1 data set was used as input for a series of calculations at depths of 3 to 20cm along the beam centerline and from 0 to 12cm off-axis at depths of 3, 5 and 15cm. Results of these calculations are given in Tables 25 to 28. The calculated neutron, gamma and total doses per source neutron (incident on the phantom) were normalized by multiplying by 10^{12} source neutrons. The precise number of neutrons emitted by the CTNS source during the time it takes the monitor ionization chamber to collect 1mCoulomb of charge is not well known. The normalization factor selected gives doses in the range of 0.2 to 60 rads. Neutron doses relative to total dose on axis are plotted in Figure 14 for the 3, 5

Table 25. Neutron, Gamma and Total Dose On-Axis from MORSE-CG

<u>Depth in Phantom</u>	<u>Neutron Dose*</u>	<u>Stan. Dev.</u>	<u>Gamma Dose*</u>	<u>Stan. Dev.</u>	<u>Total Dose*</u>	<u>Stan. Dev.</u>	<u>Gamma Fraction</u>
1.0	61.06	1.22%	1.11	6.99%	62.17	1.22%	1.79%
3.0	56.16	1.41	1.18	7.12	57.34	1.39	2.06
5.0	51.37	1.71	1.28	6.22	52.65	1.68	2.43
7.5	42.76	1.84	1.26	5.73	44.02	1.79	2.86
10.0	36.01	2.45	1.24	14.13	37.26	2.41	3.34
15.0	23.65	4.00	0.93	9.96	24.59	3.86	3.79
20.0	14.21	3.06	0.62	6.55	14.83	2.94	4.17

*Dose in tissue rads per 10^{12} neutrons incident on phantom.

Table 26. Neutron, Gamma and Total Dose at 3cm Depth from MORSE-CG

<u>Distance Off-axis</u>	<u>Neutron Dose*</u>	<u>Stan. Dev.</u>	<u>Gamma Dose*</u>	<u>Stan. Dev.</u>	<u>Total Dose*</u>	<u>Stan. Dev.</u>	<u>Gamma Fraction</u>
0.0	56.16	1.41%	1.18	7.12%	57.34	1.39%	2.06%
1.0	55.82	1.24	1.07	4.76	56.90	1.22	1.89
2.0	56.48	1.74	1.26	5.38	57.74	1.71	2.18
3.0	55.92	1.45	1.13	5.10	57.06	1.43	1.99
4.0	56.25	1.99	1.13	7.92	57.38	1.96	1.97
5.0	52.11	1.49	0.83	6.74	52.94	1.47	1.57
5.5	36.92	2.52	0.69	5.19	37.61	2.47	1.83
6.0	6.78	6.51	0.64	6.09	7.43	5.97	8.64
7.0	3.72	5.95	0.48	4.02	4.20	5.29	11.48
8.0	2.37	5.48	0.40	6.35	2.78	4.78	14.48
9.0	1.91	7.51	0.35	6.61	2.26	4.85	15.39
12.0	0.62	6.47	0.20	7.67	0.82	5.25	23.89

*Dose in tissue rads per 10^{12} neutrons incident on phantom.

and 15cm scans. The gradual divergence of the neutron beam with depth is well visualized.

Each position in phantom was calculated separately using the point detector estimator, as discussed in Chapter IV. The water phantom was approximated by a rectangular parallelopiped with a front face 40cm square. The depth of the phantom (measured along the beamline) was 13 to 30cm, depending on the depth of the detector. The water depth was always minimized to reduce the amount of computer time spent calculating unimportant parts of a particle history. It was found that 10cm

Table 27. Neutron, Gamma and Total Dose at 5cm Depth from MORSE-CG

<u>Distance Off-axis</u>	<u>Neutron Dose*</u>	<u>Stan. Dev.</u>	<u>Gamma Dose*</u>	<u>Stan. Dev.</u>	<u>Total Dose*</u>	<u>Stan. Dev.</u>	<u>Gamma Fraction</u>
0.0	51.37	1.71%	1.28	6.22%	52.65	1.68%	2.43%
3.0	50.49	1.66	1.13	6.73	51.62	1.63	2.20
5.0	46.38	1.74	0.87	5.74	47.25	1.71	1.83
5.5	37.83	2.02	0.89	7.10	38.73	1.98	2.32
6.0	8.00	4.73	0.79	14.45	8.80	4.50	9.01
7.0	4.68	5.36	0.56	4.52	5.24	4.82	10.63
8.0	3.55	8.09	0.50	6.50	4.05	7.14	12.38
9.0	2.20	5.54	0.40	6.97	2.61	4.80	15.54
12.0	0.70	5.11	0.22	4.04	0.92	4.02	23.50

*Dose in tissue rads per 10^{12} neutrons incident on phantom.

Table 28. Neutron, Gamma and Total Dose at 15cm Depth from MORSE-CG

<u>Distance Off-axis</u>	<u>Neutron Dose*</u>	<u>Stan. Dev.</u>	<u>Gamma Dose*</u>	<u>Stan. Dev.</u>	<u>Total Dose*</u>	<u>Stan. Dev.</u>	<u>Gamma Fraction</u>
0.0	23.65	4.00%	0.93	9.96%	24.59	3.86%	3.79%
3.0	23.10	3.84	0.81	7.93	23.90	3.72	3.38
5.0	20.76	4.35	0.69	8.26	21.45	4.22	3.24
5.5	20.11	4.04	0.64	6.49	20.75	3.92	3.09
6.0	14.97	4.76	0.69	9.66	15.65	4.58	4.38
7.0	5.98	6.33	0.60	8.45	6.58	5.80	9.14
8.0	5.07	15.04	0.56	8.75	5.63	13.57	9.93
9.0	4.16	16.29	0.46	9.79	4.60	14.74	9.68
12.0	1.55	8.98	0.27	6.96	1.83	7.71	14.83

*Dose in tissue rads per 10^{12} neutrons from the source.

Figure 14. Neutron dose fractions at 3, 5 and 15cm depths.

of water behind the detector was adequate to give full backscatter. Depths in excess of this amount contributed little to the fluence estimates at the detector location while increasing computing time substantially. A limit of 40 to 90 minutes of central processor unit (CPU) time was allocated for each problem, depending on the detector location. At a depth of 3cm, 45 minutes of CPU time was sufficient to compile 24,000 case histories, while 60 minutes of CPU time allowed only 6000 histories at 20cm depth. Occasionally, a given problem was rerun with a new starting random number to allow pooling of several runs in order to achieve better statistical precision. Ultimately, a precision of 7% FSD or better was achieved for almost all neutron, gamma and total dose estimates at each point.

The neutron fluence at each point was divided into nine energy bins from thermal energies to the primary (14.2 to 14.9 MeV) energy bin. A three-dimensional plot of fluence as a function of depth on axis (at 1, 3, 5, 7.5, 10, 15 and 20cm depth) is given in Figure 15. The axis labeled "Depth in Phantom" gives the depth from 0cm at the origin to 20cm in the foreground. The axis running from left to right gives the neutron energy, while the height represents the differential fluence in $\text{n/cm}^2\text{eV}$. The prominent D-T peak is continually attenuated with depth and is surpassed by the 13.8 to 14.2 MeV peak at 15cm depth. The lower energy peaks tend to reach a flat maximum at about 5 to 10cm depth.

Neutron Spectra in Phantom

Figure 15. Neutron energy fluence as a function of depth in phantom.

It is now possible to compare directly these Monte Carlo calculations with results from the ion chamber measurements reported in Chapter III. The neutron and gamma doses (relative to incident dose) vs. depth in phantom are illustrated in Figure 16 for the MORSE-CG and TE-GM results. The fraction of the total dose due to neutrons is quite similar for both the MORSE-CG calculations (dashed line) and for the TE-GM experimental method (solid line) at any depth. However, the fraction of dose due to photons is quite different for the two methods, and is well outside the error bars representing the uncertainty in the two methods. It is important to note that this calculation assumes that only primary D-T neutrons are incident upon the phantom.

The neutron and gamma dose fractions at 0, 3, 6, 9 and 12cm off the beam axis at depths in phantom of 3, 5 and 15cm are illustrated in Figures 17, 18 and 19, respectively. The dose fractions relative to total dose on axis at that depth are plotted in each figure. The neutron dose fractions calculated by MORSE-CG are in good agreement with dose fractions determined from the TE-GM method at nearly all locations. Again, the gamma dose fractions from these Monte Carlo calculations are not in close agreement with the measured results anywhere in phantom.

Figure 16. Neutron and gamma dose in a water phantom relative to total incident dose on axis for the GTNS D-T neutron source. MORSE-CC results are shown by the dashed line, ion chamber results by the solid line.

Since the measured and calculated neutron dose fractions are in good agreement with each other, and the FEWGL cross section data set is believed to have accurate values for the gamma ray production probabilities, some other explanation for the disagreement between experimental and theoretical results for gamma dose fractions must be found. One possible cause of this disagreement is the failure of these calculations to allow for the known gamma fraction in air at 20cm from the collimator in the absence of the water phantom. Measurements

Figure 17. n & γ dose fractions (%) at 3cm depth in phantom. The geometric beam edge is at 5.28cm. (Dashed line: MORSE-CG; solid line: ion chamber results.)

Figure 18. n & γ dose fractions (%) at 5cm depth in phantom. The geometric beam edge is at 5.35cm. (Dashed line: MORSE-CG; solid line: ion chamber results.)

reported in Chapter III showed that 3.0 to 3.5% of the total tissue absorbed dose rate in air at the front face of the phantom is attributable to gamma radiation. The energy of these gamma radiations may be estimated from the lead filtration measurements to be in the range of 1 to 10 MeV. Further calculations using MORSE-CG were therefore performed in order to attempt to achieve closer agreement with measured results.

Figure 19. n & γ dose fractions at 15cm depth in phantom. The geometric beam edge is at 5.67cm. (Dashed line: MORSE-CG; Solid line: ion chamber results.)

b. Comparison of Measured Results with Mixed n/ γ Beam

A new source routine for the MORSE-CG code was written. This routine created a plane γ -ray source 20cm from the front face of the water phantom with dimensions of 195cm (east to west) by 108 cm (above and below the beam centerline), approximating the size and position of the collimator. It was known from foil activation and TLD measurements outside the collimator that substantial quantities of thermal neutrons were present. Since it is reasonable to assume that some of these thermal neutrons are actually transmitted through the collimator (which contains large amounts of hydrogen), an energy of 2.2 MeV corresponding to the energy of hydrogen capture gamma radiations was used.

A series of calculations utilizing this new version of MORSE-CG which produced only a plane source of 2.2 MeV gamma rays and no D-T neutrons was performed. Gamma doses were computed at six points along the beam axis in a water phantom at depths of 3 to 20cm. Gamma doses were also calculated at four off-axis points at 3 and 15cm depths, corresponding to locations where ion chamber measurements were performed. One calculation was made at a point in air with no phantom present at a distance of 20cm from the face of the collimator. This calculation provided the weighting factor necessary to produce a 3.25% gamma dose fraction in air at the face of the phantom. Results of all of these new gamma dose calculations were then algebraically added to

the combined neutron and gamma dose calculations described in the previous section.

Results of these new calculations combined with the previous calculations are presented in Table 29 for points on the beam axis. These results are illustrated in Figure 20. The new gamma dose estimates are much closer to the results from the TE-GM measurements than were the calculations described earlier. The Monte Carlo results are still below the ion chamber results at each depth, but the disagreement is within the experimental uncertainties at two of the points. The discrepancy between the two data sets appears to increase with increasing depth.

Table 30 gives results of the combined calculations at 3 and 15cm depth. These results are also illustrated in Figures 21 and 22. The gamma fractions calculated by the revised Monte Carlo technique are in quite good agreement with ion chamber results at 3cm depth, lying within the experimental uncertainties at each point. The ion chamber measurements indicate a sharper drop between the points at 3 and 6cm off-axis at 3cm depth than do the MORSE-CG calculations. A slight error in the positioning of the GM counter could cause a substantial difference in measured gamma dose fractions in the penumbra region (near 6cm off-axis), however. Results at 15cm depth are consistently lower for the MORSE-CG calculations than for the TE-GM results. This is consistent with the earlier observation that the Monte Carlo dose estimates of gamma fractions at 15 and 20cm depth are substantially

Table 29. Gamma fractions on axis from revised MORSE-CG

Depth	D_n^*	D_γ^*	R	$D_\gamma/D_{axis} (\%)$
0 cm	58.68	1.97	3.25	3.25 ± 0.40
3	56.16	2.52	4.29	4.15 ± 0.40
5	51.37	2.43	4.52	4.01 ± 0.45
7.5	42.76	2.21	4.92	3.65 ± 0.56
10	36.01	2.06	5.40	3.39 ± 0.50
15	23.65	1.53	6.06	2.52 ± 0.42
20	14.21	1.08	7.05	1.78 ± 0.34

*Dose in tissue rads per 10^{12} neutrons incident on phantom.

Figure 20. Gamma fractions on axis in phantom.

lower than the measured results. The disagreement is more pronounced in the primary beam, namely on axis and 3cm off-axis.

Table 30. Gamma fractions at 3 and 15cm depth from revised MORSE-CG

<u>Depth</u>	<u>Distance Off-Axis</u>	<u>D_n[*]</u>	<u>D_γ[*]</u>	<u>R</u>	<u>D_γ/D_{axis} (%)</u>
3.0cm	0.0cm	56.16	2.52	4.29	4.29 ± 0.40
	3.0	55.92	2.18	3.75	3.71 ± 0.35
	6.0	6.78	1.67	19.74	2.85 ± 0.50
	9.0	1.91	1.37	41.78	2.33 ± 0.45
	12.0	0.62	1.10	64.06	1.87 ± 0.45
15.0	0.0	23.65	1.53	6.06	6.06 ± 0.65
	3.0	23.20	1.33	5.46	5.30 ± 0.53
	6.0	14.97	1.21	7.48	4.81 ± 0.48
	9.0	4.16	0.98	19.02	3.88 ± 0.39
	12.0	1.55	0.82	34.61	3.75 ± 0.38

*Dose in tissue rads per 10¹² neutrons incident on phantom.

Figure 21. γ fractions off-axis at 3cm depth in phantom.

Figure 22. γ fractions off-axis at 15cm depth in phantom.

c. Comparison of Results with Other Workers

A number of reports exist in the scientific literature of dosimetric measurements with high-intensity fast-neutron sources. Barschall has reviewed a variety of such sources used for medical and other purposes.^{80,81} These sources generally employ either a cyclotron to accelerate deuterons (or protons) to an energy of 8 to 66 MeV for bombardment of a beryllium target, or an accelerator to accelerate deuterons to a modest energy (100 to 500 KeV) for bombardment of a tritium target. The first category of neutron sources (referred to as (d,Be) sources) produce a spectrum of neutron energies from near-thermal to nearly the bombarding energy of the deuteron. The mean energy of the neutrons thus produced varies from about 0.4 to 0.6 times the bombarding energy for the (d,Be) reaction, depending on the target thickness. The D-T type of neutron generators produce a nearly mono-energetic (14 to 15 MeV) beam of neutrons.

It is possible to compare some of the dosimetric parameters of the 14.8 MeV neutron beam in water measured at the Gas Target Neutron Source with those reported by other investigators. Table 31 gives the physical parameters measured for the GTNS source together with those measured in phantom for other D-T sources and for two cyclotron sources. Only those (d,Be) sources with mean neutron energies of 15 MeV or less are shown. Cumulative results for all participants in the INDI report are also given. Not all investigators reported both the 50%

Table 31. Comparison of Neutron Dosimetric Parameters

Author	Neutron Source	50% Dose	Gamma Fraction at:				Technique
			5cm	10	15	20	
Greene (1968)	D-T, 50cm SSD 10x10cm	8.0cm	--	13%	12%	--	Ion chambers
Alsmiller (1974)	35 MeV d,Be pt. 10x10cm, 125cm	10 cm (TE liq)	--	--	--	--	HETC code
August (1977)	35 MeV d,Be pt. 10x10cm, 125cm	13 cm (TE liq)	5.4- 12.8	6.9- 14.7	8.4- 15.9	9.9- 17.2	Ion chambers $E_n=15\text{MeV}$ ave.
INDI (1978)	D-T, pt. source 20cm, uncoll.		6.3 --	8.3% --	-- --	12.0 --	Carried out in 1973
Attix (1975)	D-T, pt. source 20cm, uncoll.		2.4 --	6.0 --	-- --	10.0 --	INDI expt. Ion chambers
Herling (1975)	D-T, pt. source 20cm, uncoll.		3.5 --	5.4 --	-- --	10.2 --	INDI expt. MORSE/CASK
Burger (1977)	D-T source(INDI) 20cm, uncoll.		7.8 5.4	10.3 7.3	-- --	14.6 10.7	Ion chamber DOT code
Goetsch (1982)	INDI D-T source 20cm, uncoll.		3.5	5.0	--	8.4	MORSE/FEWG1 $E_n=15.9\text{MeV}$
Weaver (1977)	21 MeV d,Be pt. 10x10cm, 150cm		--	8.0	--	--	$E_n=8.5\text{MeV}$ ave.
Broerse, Greene	D-T, pt. source 8x10cm, 80cm	~8cm	-- --	11.5 8.3	-- --	-- --	Phillips tube Elliot tube
Mijnheer (1978)	D-T, pt. source 16x20cm, 80cm	9.7cm	15.0	17.0	19.0	20.0	Ion chambers
Schmidt (1980)	D-T, 14.1 MeV 11x11cm, 80cm	9.2cm (Al50)	5.6	5.7	7.7	9.5	Stilbene cry. ion chamb.
McDonald (1982)	RTNS D-T source Al50 phantom		5.9 4.4	-- --	-- --	-- --	TE-CM method Calorimeter
Hertel (1982)	D-T source, 166cm, uncoll.		-- --	10.0 (7.2)	15.0	22.0	NE213 scint. (TE liquid)
UW GTNS (1982)	D-T line source 10x10, 125cm	10.5cm 12.2cm	5.7 4.5	7.2 5.4	8.7 6.1	11.1 7.0	TE-CM method MORSE/FEWG1

depth dose and the fraction of the absorbed dose at depth due to gamma radiation. The source-to-surface distance (SSD) varied from 20 to 166cm, while the field size varied from 8 by 10cm to completely uncollimated beams.

The average source-to-surface distance for the 55cm-long Gas Target Neutron Source may be estimated from the MORSE-CG calculations. The line source was approximated to be of uniform lineal intensity, in good agreement with activation chip measurements. Source neutrons (which are emitted isotropically) originating in the downstream end of the source (adjacent to the collimator) are more likely to pass through the open part of the collimator without interacting than those originating near the target entrance. Therefore, for detectors outside the collimator, the effective center of the neutron source distribution is shifted downstream. The MORSE-CG code was used to select randomly the starting axial (z) coordinate of 10,000 source particles. The average z was found to be 32.6 ± 1.1 cm. The distance from the source entrance to the face of the water phantom was 155cm. Therefore, the effective SSD of the GTNS facility for this work was $155 - 32.6$, or about 122cm.

Weaver et al⁸² have reported on the variation of gamma dose fraction on axis with a 21 MeV (d,Be) neutron source as a function of field size, phantom size and depth in a tissue-equivalent liquid.⁸³ The increase of gamma dose fraction with depth and field size was approximately linear. The gamma dose fraction was nearly identical for a 30cm cube phantom and a 50 by 50 by 35cm phantom, suggesting a good approxi-

mation to the desired semi-infinite size. However, the gamma dose fraction decreased under otherwise identical irradiation conditions for 20cm and 10cm cube phantoms. This suggested to Weaver et al that a significant portion of the photon dose in the beam was produced by scattered photons outside the beam. They concluded that the effective field size for photon production was larger than the collimator-defined field. These investigators also noted that the gamma fraction increases substantially with distance off-axis. They observed that thermal neutrons at depth in phantom would result in the production of 2.2 MeV hydrogen capture gamma rays. Since the photon mean free path λ_γ in water is about 22cm, while the neutron mean free path λ_n was about 8cm, the photon fluence was more slowly attenuated with depth and distance off-axis than was the neutron dose. This observation is also valid for D-T sources, where λ_n is about 9.9cm.

Measurements utilizing D-T neutron generators have been reported by a number of centers.⁸⁴⁻⁸⁹ There is a wide variation in the geometric configuration of the neutron sources and phantoms, with SSD's of 50 to 166cm, and field sizes of 8x10cm to uncollimated beams. Greene, Broerse and Mijnheer used water phantoms, Hertel used a tissue-equivalent liquid and Schmidt and McDonald used a solid Al50 plastic phantom. Thus, comparisons even among neutron sources with nearly identical energy spectra are not totally meaningful. The 50% depth dose was 8.0cm at 50cm SSD, 8.5cm at 80cm SSD and 9.7cm at 80cm at another institution. Results for the GTNS neutron beam in phantom with a 10x10cm field size may be deduced from Figure 15, which yields

10.5cm for the TE-GM method and 12.2cm for the MORSE-CG calculations. The increase in the depth for 50% dose with increasing SSD is the expected physical result. The gamma dose fractions at depth are given for these investigators in Table 31, where reported. These results vary quite widely and are strongly influenced by the gamma dose fraction incident upon the phantom. The results obtained in this thesis for the GTNS facility are near the middle of the reported range at each depth.

Dosimetric properties at the NRL cyclotron were reported by August⁹⁰ in 1977 and modeled with the HETC transport code by Alsmiller⁹¹ in 1974. The NRL 35 MeV (d,Be) neutron source had an average neutron energy of about 15 MeV, making it roughly comparable to a D-T neutron source. The 125cm SSD at NRL is quite comparable to the 122cm effective SSD at GTNS. August reported a 50% depth dose at 13cm for a 10x10cm field size, while Alsmiller calculated about 10cm depth for that quantity, in reasonable agreement with our results. The gamma dose fractions were not well known in the liquid TE phantom at NRL and Alsmiller did not report such calculations. Nevertheless, all gamma dose fractions reported in this thesis are intermediate between the upper and lower limits reported by August (Table 31).

Another meaningful comparison to the GTNS facility is provided by the INDI experiment carried out in 1973 at the RARAF facility at Brookhaven. This facility created an uncollimated isotropic source of 15.1 MeV D-T neutrons with a gamma dose fraction of about 5% in air

(staff estimate) at the phantom position. A total of fourteen groups of scientists made measurements at RARAF during 1973. Measurements were made in air at a distance of 30cm from the target and at depths of 5, 10 and 20cm in a 30cm cubic water-filled Lucite phantom with front face 20cm from the target. Combined results of all INDI participants are given in Table 31, together with published reports by Attix,⁹² Herling⁹³ and Burger⁹⁴. Herling used MORSE-CG to model the experiment, as discussed earlier, while Burger used the two-dimensional discrete ordinates code DOT for simulation. Both workers used the CASK⁹⁵ data set previously described. The gamma dose fractions calculated by both MORSE and DOT were lower than either set of measured results or for the average of all reported results.

Three sets of Monte Carlo calculations discussed earlier have all given lower estimates of gamma dose fractions in phantom than measurements conducted under identical circumstances. Burger, Morhart and Schraube⁹⁶ suggested that the Monte Carlo calculations must necessarily produce lower gamma dose estimates than measured results when scattered gamma rays from the room are considered. They referred to the difference as $D_{\gamma,env}$ for the environmental gamma dose and gave a procedure for estimating this contribution. The calculated gamma dose is first normalized to the measured dose as follows:

$$(31.) \quad D_{\gamma,tot} = \frac{D_{n,calc}}{D_{n,meas}} \cdot D_{\gamma,meas}$$

The desired dose $D_{\gamma,env}$ is equal to $D_{\gamma,tot}$ minus $D_{\gamma,calc}$, and is always found to be a positive quantity. The environmental gamma dose component was found in this manner to be 30 to 33% of the measured dose. Alternatively, the calculated and measured absorbed doses can be normalized to each other by equating them at some point. Table 32 gives the same quantities (with the neutron doses set equal at 10cm depth) for the INDI experiment as calculated and measured by Burger et al. This normalization illustrates the close match between the measured and calculated neutron doses and still attributes about 32% of the measured gamma dose in phantom to scattered gamma rays from the room. Neglecting this environmental component in the Monte Carlo calculations leads to a substantial error in the estimation of gamma dose fractions in phantom.

Table 32. Environmental Gamma Dose for the INDI Experiment (Burger)

Depth	$D_{n,meas}^a$	$D_{n,calc}^b$	$D_{\gamma,meas}^a$	$D_{\gamma,calc}^b$	$D_{\gamma,env}^c$	$\frac{D_{\gamma,env}}{D_{\gamma,meas}}$
5 cm	5.34	5.46	0.452	0.309	0.143	31.6%
10	3.08	3.08	0.354	0.242	0.112	31.6
20	1.06	1.03	0.181	0.124	0.057	31.5

^aAbsorbed dose in tissue rads from ion chamber measurements.

^bAbsorbed dose in tissue rads from calculations (normalized).

^cAbsorbed dose in tissue rads (normalized to ion chamber results).

It is now possible to reconsider the INDI experiment in light of the suggested methodology of Burger et al. New Monte Carlo calculations have been performed for this thesis to estimate the gamma dose fractions in phantom for the INDI experiment. These new calculations differ from the calculations performed by Herling in several important ways. First of all the data set used was FEWG1 instead of CASK, which has several implications. The CASK data set had a maximum neutron energy bin of 12.2 to 14.9 MeV, with an average energy of 13.6 MeV. The energy of the D-T neutrons at RARAF was estimated by the staff to be about 15.5 MeV. The FEWG1 data set extends up to 20 MeV and has a neutron energy bin from 14.9 to 16.9 MeV, with an average neutron energy of 15.9 MeV. The FEWG1 data set is based on the more recent ENDF/B-IV data set rather than the ENDF/B-II and III data sets from which the CASK cross sections were derived. The FEWG1 data set also contains 37 neutron energy groups and 21 photon groups in contrast to the 22 neutron and 18 photon groups which CASK uses. The geometry of the INDI experiment was also more closely modeled this time, with a 30cm cube phantom replacing the cylinder of 16.926cm radius and 30cm length used by Herling. Lastly, the neutron and gamma dose estimates were prepared by allowing the MORSE-CG code to do a fluence-to-dose conversion for each of the 58 energy bins rather than by performing a manual calculation of these products for 5 groups of neutron energy bins and 5 groups of photon energy bins.

Results of this more rigorous calculation are given in Table 33. Also included in this table are results from Attix (measured) and from Herling and Burger (calculated). The revised MORSE-CG calculations were performed using first a neutron source bin with mean neutron energy of 13.3 MeV and then with a bin with mean neutron source energy of 15.9 MeV. Results obtained for the gamma dose fraction at 13.3 MeV were somewhat higher than those of Herling, while those obtained at 15.9 MeV were somewhat lower than Herling's results. Each of the new estimates is within the range of reported results for the INDI experiment. More confidence should be placed in the 15.9 MeV estimate since the energy more nearly matches the estimated neutron energy at the RARAF facility.

A revised estimate of the environmental gamma dose at INDI can now be prepared. Table 34 gives the new neutron and gamma dose estimates using the revised MORSE-CG calculations (at 15.9 MeV) and the neutron

Table 33. Gamma Dose Fractions* for the INDI Experiment

<u>Depth</u>	Attix (meas)	Herling 13.6MeV	Burger (meas)	Burger (calc)	FEWG1 13.6MeV	FEWG1 15.9MeV
5 cm	2.4	3.5	7.8	5.4	3.7	3.5
10	6.0	5.4	10.3	7.3	6.8	5.0
20	10.0	10.2	14.6	10.7	11.8	8.4

*Gamma dose fraction defined as $100 \cdot D_{\gamma} / D_{\text{total}}$

and gamma doses measured by Burger. These calculations indicate that the environmental gamma dose accounts for 40 to 66% of the total gamma dose present on axis in phantom. While this estimate may perhaps err on the high side, it certainly indicates that the environmental gamma dose at the RARAF site was too high to be reasonably neglected.

The environmental gamma dose component may also be computed for the gamma dose fractions in phantom at the GTNS facility. Table 35 gives the same quantities described by Burger et al for the previously reported Monte Carlo calculations and ion chamber measurements at GTNS. The calculated results have been normalized to the measured results by setting the neutron doses to be equal at 10cm depth. Calculated results given in this table are for the Monte Carlo calculations which utilized a neutron-only source (reported in Chapter 4). The contri-

Table 34. Revised Environmental Gamma Dose for the INDI Experiment

Depth	$D_{n,meas}^a$	$D_{n,calc}^b$	$D_{\gamma,meas}^a$	$D_{\gamma,calc}^b$	$D_{\gamma,env}^c$	$\frac{D_{\gamma,env}}{D_{\gamma,meas}}$
5 cm	5.34	5.10	0.452	0.151	0.301	66%
10	3.08	3.08	0.354	0.164	0.190	54
20	1.06	1.19	0.181	0.109	0.072	40

^aAbsorbed dose in tissue rads from ion chamber measurements.

^bAbsorbed dose in tissue rads from calculations (normalized).

^cAbsorbed dose in tissue rads (normalized to ion chamber results).

bution of scattered gamma rays to absorbed dose on axis in phantom varies from 56 to 66% of the total measured gamma dose. The environmental gamma dose declines with depth, indicating that the scattered gamma radiation comes predominantly from the direction of the neutron source. It is impossible to obtain a good agreement between measured and calculated results under these circumstances if the calculations ignore this contribution.

Table 35. Environmental Gamma Dose in Phantom at GTNS

Depth	$D_{n,meas}^a$	$D_{n,calc}^b$	$D_{\gamma,meas}^a$	$D_{\gamma,calc}^b$	$D_{\gamma,env}^c$	$\frac{D_{\gamma,env}}{D_{\gamma,meas}}$
3 cm	105.9	92.3	5.69	1.94	3.75	66%
5	89.1	84.5	5.48	2.10	3.38	62
7.5	74.3	70.3	5.12	2.07	3.05	60
10	59.2	59.2	4.62	2.04	2.58	56
15	38.8	38.9	3.72	1.53	2.19	59
20	23.3	23.4	2.90	1.02	1.88	65

^aAbsorbed dose in tissue rads from ion chamber measurements.

^bAbsorbed dose in tissue rads from calculations (normalized).

^cAbsorbed dose in tissue rads (normalized to ion chamber results).

CHAPTER VI. CONCLUSIONS

Mixed neutron/gamma radiation dosimetry is still poorly understood compared to gamma dosimetry. Even under the most carefully controlled circumstances (during the INDI experiment) the world's leading neutron research centers obtained a wide range of gamma dose estimates. Although lack of agreement on appropriate values for fundamental physical parameters (such as kerma factors and W/e) caused some of the dispersion of the results, basic differences of opinion still exist over the best way to make measurements in mixed n/γ fields.

The paired chamber technique, using a neutron "tissue-equivalent" chamber and a "neutron-insensitive" dosimeter, has long been regarded as one of the simplest, most accurate and most reliable methods of obtaining neutron and gamma absorbed doses in mixed n/γ fields. The neutron-to-photon sensitivity ratio (B/A) for the TE chamber may be obtained from theoretical considerations. Any reasonably tissue-equivalent ion chamber must have a B/A very near unity (0.996 for the IC-17 ion chamber used in this work). The B/A for the non-hydrogenous (neutron-insensitive) dosimeter relative to the TE chamber may be obtained from the lead-filtration method described in Chapters II and III.

The paired-dosimeter method suffers from a serious defect. The neutron sensitivity B for the non-hydrogenous chamber is correct only at the neutron energy used in the lead filtration calibration. Once these two dosimeters are placed in a water phantom, the calibration for the non-hydrogenous chamber is in error due to the presence of scattered neutrons whose mean energy is necessarily less than that of the bombarding neutrons. The neutron sensitivity B and gamma sensitivity A of the TE chamber are unaffected by the degraded neutron energy spectrum and secondary gamma radiation to the degree that the wall material is truly tissue-equivalent (since both B and A are given in coulombs per tissue rad). However, the response of the non-hydrogenous wall materials (e.g., carbon, magnesium, iron) typically used for the neutron-insensitive dosimeter declines rapidly with decreasing neutron energy. This error invariably leads to an underestimate of the gamma dose in phantom.

The three-dosimeter method (described in Chapters II and III) was devised as an experimental way of allowing for the known fall-off of neutron sensitivity B of the non-hydrogenous chambers in-phantom. The neutron dose is partitioned into fast and slow components. Two assumptions are inherent in this method: the non-hydrogenous dosimeter is completely insensitive to the "slow" neutron dose component, and the gamma sensitivity A is unaltered in-phantom relative to the gamma energy used for calibration. Since the latter assumption is implicit in virtually all reported gamma dose measurements in-phantom (e.g., at ^{60}Co radiation therapy facilities) only the former assumption is truly

questionable. This method, also provides an "effective" B/A for each non-hydrogenous dosimeter at each point in the phantom.

The two- and three-dosimeter methods were extensively tested in air and in-phantom with a D-T neutron source and in air with simultaneous neutron/photon irradiation. Uncertainty analysis of the paired equations for the two-dosimeter method leads to the conclusion that best results are obtained when the B/A ratio for the non-hydrogenous dosimeter is as small as possible. The TE-GM pair appeared to give the most consistent and reliable gamma dose estimates in each experimental circumstance, as expected. The GM counter also appeared to dominate gamma dose estimates when it was included as one member of the three-dosimeter method. Gamma dose estimates for the TE-GM, TE-C-GM and TE-Mg-GM methods were identical within the uncertainty of the measurement.

The MORSE-CG three-dimensional coupled neutron/gamma Monte Carlo computer code offers a totally independent method of estimating neutron and gamma doses in each of the measured experimental circumstances. The MORSE-CG code is one of the most extensively used and validated Monte Carlo shielding codes in the world and the FEWGL data set employed is a modern, tractable, well-tested reference. Unfortunately, computing costs do not normally permit an entire irradiation facility (including all walls, floors and ceilings) to be modeled. The environmental (room return) gamma radiation dose is therefore very difficult to predict. Monte Carlo shielding codes almost invariably predict

lower gamma doses in-phantom than measured results indicate. Nevertheless, physically reasonable models of scattered gamma radiation may be employed to obtain better Monte Carlo gamma dose estimates.

The two- and three-dosimeter methods have been shown to give reasonable and physically meaningful neutron and gamma dose estimates under a broad range of irradiation conditions. Ion chamber measurement techniques are well understood and offer a precise, reproducible means of making absorbed dose estimates. Intercomparison of ion chamber-derived absorbed dose estimates with those obtained from other means (e.g., calorimetry, Monte Carlo calculations) normally shows good agreement. The entire field of mixed neutron/gamma dosimetry remains one of the most challenging fields of radiological physics.

Appendix A Calculation of Zirkle's n-Unit in Terms of Rads

Zirkle defined the quantity of neutron radiation which discharged a Victoreen "r" meter to the same extent as one r of X-rays as one n-unit. Assuming the wall of the Victoreen meter to be air equivalent for neutrons as well as for photons, the absorbed dose of neutrons sufficient to cause this reading may be computed as follows:

$$(1.) \quad D(E) = K(E) \cdot \Phi(E)$$

where D is the dose in rads, K(E) and $\Phi(E)$ are the neutron KERMA factor and neutron fluence, respectively, both of which are functions of energy. The simplifying assumption may be made for the early neutron work by the Berkeley group that $\Phi(E)$ can be replaced by a monoenergetic spectrum of ~6.4 MeV neutrons, for which the KERMA factor is $0.890 \cdot 10^{-9}$ rad-cm²/n. This gives:

$$(2.) \quad D = (0.890 \cdot 10^{-9} \text{ rad-cm}^2/\text{n}) \cdot (\Phi_n \text{ n/cm}^2)$$

Assuming that the Victoreen chamber fulfills the Bragg-Gray requirements, then the following relation also applies:

$$(3.) \quad D = \frac{Q}{m} (\text{coulomb/kg}) \cdot \frac{W}{e} (\text{joule/coulomb})$$

where Q/m is the ionization per unit mass produced in the chamber and (W/e) is the energy per ion pair for neutrons in air. It is possible to compute the absorbed dose sufficient to yield ionization interpreted by the chamber as "1 r" as follows:

$$\begin{aligned}
 (4.) \quad D &= (1r) \cdot \frac{(2.58 \cdot 10^{-4} \text{ coul/kg})}{1 r} \cdot (35.8 \text{ joules/coul}) \\
 &= 9.03 \cdot 10^{-3} \text{ gray} \\
 &= 0.90 \text{ rad}
 \end{aligned}$$

where a value of 35.8 joules/coulomb for the University of Washington neutron beam (produced by 21 MeV deuterons on a beryllium target) has been used as an approximation of the Berkeley cyclotron-produced neutrons.

The fluence of 6.4 MeV neutrons needed to produce one n-unit may now be computed by inverting equation (1.), thus:

$$\begin{aligned}
 (5.) \quad \Phi_E &= \frac{D(E)}{K(E)} = \frac{0.90 \text{ rad}}{(0.890 \cdot 10^{-9} \text{ rad cm}^2/\text{n})} \\
 &= 1.01 \cdot 10^9 \text{ n/cm}^2
 \end{aligned}$$

Appendix B Calculation of Ionization Ratio in Tissue Relative to Air

Assume an equal fluence of 2.7 MeV neutrons (the energy spectrum of neutrons from the D-D reaction used in Gray's experiment was given as 2.4 to 2.9 MeV). Then, neutron dose is given by two expressions (assuming Bragg-Gray conditions are satisfied)

$$(1.) \quad D_x = \Phi \cdot K_x$$

$$(2.) \quad D_x = \frac{(Q)}{(m)} x \cdot \frac{(W)}{(e)} x$$

For tissue, utilizing a value of K_T from Ref. 2 and using (W/e) for 2 MeV neutrons in TE gas from Ref. 97, this yields:

$$(1.a) \quad D_T = \Phi \cdot K_T = \Phi \cdot (0.341 \cdot 10^{-8})$$

$$(2.a) \quad D_T = \frac{(Q)}{(m)} T \cdot \frac{(W)}{(e)} T = \frac{(Q)}{(m)} T \cdot (31.14)$$

For air, taking K_A from Ref. 2 and using (W/e) for 2 MeV alpha particles from Ref. 97, this yields

$$(1.b) \quad D_A = \Phi \cdot K_A = \Phi \cdot (0.583 \cdot 10^{-9})$$

$$(2.b) \quad D_A = \frac{(Q)}{(m)} A \cdot \frac{(W)}{(e)} A = \frac{(Q)}{(m)} A \cdot (35.08)$$

Equating (1.) and (2.) for air and tissue and solving for Φ gives:

$$(3.) \quad \Phi = \frac{(Q/m)_T \cdot 31.14}{(0.341 \cdot 10^{-8})}$$

$$(4.) \quad \Phi = \frac{(Q/m)_A \cdot 35.08}{(0.583 \cdot 10^{-9})}$$

Equating (3.) and (4.) gives:

$$(5.) \quad \frac{(Q/m)_T \cdot 31.14}{(0.341 \cdot 10^{-8})} = \frac{(Q/m)_A \cdot 35.08}{(0.583 \cdot 10^{-9})}$$

The ratio of ionization in tissue to air may now be computed:

$$(6.) \quad \frac{(Q/m)_T}{(Q/m)_A} = \frac{(W/e)_A}{(W/e)_T} \cdot \frac{(0.341 \cdot 10^{-8})}{(0.583 \cdot 10^{-9})} = 6.59$$

This is in good agreement with Gray's value of 7 for this ratio.

Appendix C Evaluation of f_n (from Bretscher and French)

Bretscher and French demonstrated in 1944 that the fraction f_n of the ionization current J_H' in their neutron sensitive ionization chamber was given by:

$$(1.) \quad f_n = \frac{(J_H' - J_D') / J_H'}{(J_H - J_D) / J_H}$$

where J_H' and J_D' are the chamber responses in the mixed neutron/gamma beam of the hydrogenous (neutron-sensitive) and deuterogenous (neutron-insensitive) chambers, respectively. J_H and J_D are the same quantities in a pure neutron beam.

A response equation may be written for each chamber as follows:

$$(2.a) \quad J_H' = A_H \cdot D_\gamma + B_H \cdot D_n$$

$$(2.b) \quad J_D' = A_D \cdot D_\gamma + B_D \cdot D_n$$

Similar equations may be written for the response of these chambers to a pure neutron beam (where $D_\gamma = 0$):

$$(3.a) \quad J_H = B_H \cdot D_n$$

$$(3.b) \quad J_D = B_D \cdot D_n$$

The expression for f_n given by Bretscher and French may now be evaluated:

$$(4.) \quad f_n = \frac{[(A_H \cdot D_\gamma + B_H \cdot D_n) - (A_D \cdot D_\gamma + B_D \cdot D_n)] / (A_H \cdot D_\gamma + B_H \cdot D_n)}{(B_H \cdot D_n - B_D \cdot D_n) / (B_H \cdot D_n)}$$

But $A_H = A_D$, since the gamma sensitivities of the two chambers were adjusted to be equal. Thus equation (4.) reduces to:

$$(5.) \quad f_n = \frac{[(B_H - B_D) \cdot D_n] / (A_H \cdot D_\gamma + B_H \cdot D_n)}{[(B_H - B_D) \cdot D_n] / (B_H \cdot D_n)}$$

which further reduces to:

$$(6.) \quad f_n = \frac{B_H \cdot D_n}{(A_H \cdot D_\gamma + B_H \cdot D_n)}$$

The numerator of this fraction is clearly the response of the hydrogenous chamber to neutrons alone, while the denominator is the response of that chamber to a mixed beam of neutrons and gammas. Thus, f_n is, in fact, the proportion of the response of the hydrogenous chamber in a mixed beam due to neutrons.

Appendix D Revised Method for Estimating Void Corrections

A revised method for estimating the void correction for an ion chamber in a water phantom was suggested by F.H. Attix. The derivation begins by considering the meaning of A and B in equation (7.) on page 16. The gamma sensitivity A is defined as the ratio of the dosimeter reading from γ irradiation to the tissue dose at the center of an equilibrium tissue sphere centered at the same point. For neutrons incident in air, the neutron sensitivity B is defined analogously. The radius of the dosimeter $\underline{l}$ must always be larger than the radius $\underline{l'}$ of the equilibrium tissue sphere. The value of $\underline{l'}$ is taken to be 0.5g/cm^2 of tissue for both ^{60}Co gamma rays and for D-T neutrons. (The latter only requires $\approx .2\text{--}.3\text{g/cm}^2$, but it is convenient to use the same value as for the γ rays.) Note that the outer edge of the detector is at a distance $\underline{l}$ from the center of the sphere (with $\underline{l} > \underline{l'}$) and hence the particles incident on the equilibrium sphere are actually attenuated by an additional path $(\underline{l} - \underline{l'})$ of air. This attenuation is negligibly small.

We may now define A' and B' in a water phantom:

$$(1.) \quad A' = \frac{\text{dosimeter reading (in phantom) for } \gamma\text{-rays}}{\gamma \text{ tissue dose at center of an eq. tissue sphere (radius } \underline{l'}) \text{ inside a spherical shell of water of thickness } (\underline{l} - \underline{l'}), \text{ centered at the same point}}$$

$$A' = A \cdot \frac{\gamma \text{ tissue dose at center of an eq. sphere of tissue centered at same point}}{\gamma \text{ tissue dose at center of an eq. tissue sphere (radius } \underline{l'}) \text{ inside a spherical shell of water of thickness } (\underline{l} - \underline{l'}), \text{ centered at the same point}}$$

We may now define a correction factor p :

$$A' \equiv A \cdot p$$

where

$$p = \frac{1}{\exp[-(\mu_{\text{en}}/\rho) \cdot (\underline{l} - \underline{l'}) \cdot \rho]} = \exp[\mu_{\text{en}}(\underline{l} - \underline{l'})]$$

where μ_{en}/ρ is the gamma ray mass energy absorption coefficient, which is roughly equal to the broad beam attenuation coefficient. Similarly,

$$B' = B \cdot q$$

where:

$$q = \exp[(\sigma_{\text{en}})_n \cdot (\underline{l} - \underline{l'})]$$

where the neutron mass energy absorption coefficient σ is used to evaluate q .

A' and B' are to be used in phantom in place of A and B , respectively, to account for the displacement or void correction. It should be noted that p and q depend only on the physical size of the

detector, not on the wall material. These correction factors are approximately energy-independent, and therefore are independent of position in phantom so long as the detector is completely immersed.

As an example, the IC-17 ion chamber has a physical radius of 1.143cm. We may compute the displacement factor p for γ rays as follows:

$$p = \exp[0.0297(1.143-0.500)] = 1.019$$

The inverse of p gives 0.981, compared to Shapiro's value of 0.970, which is erroneously low because the attenuation in the chamber wall (equivalent in thickness to the radius of the equilibrium tissue sphere) is counted twice. The GM counter has a radius of 0.38cm, leading to a $p = 1.012$, and $p^{-1} = 0.988$.

Bibliography

1. Gray, L.H. (1936), An Ionization Method for the Absolute Measurement of Gamma-Ray Energy; Proceedings, Royal Society of London, 156, 578.
2. International Commission on Radiations Units and Measurements, (1977), Neutron Dosimetry for Medicine and Biology, Report 26.
3. U.S. Nuclear Regulatory Commission (1982), Standards for Protection Against Radiation; Title 10, Energy, U.S. Code of Federal Regulations, U.S. Government Printing Office.
4. Rossi, H.H. and Mays, C.W (1978), Leukemia Risk from Neutrons; Health Physics, 34, 353.
5. Chadwick, J. (1932), Possible Existence of a Neutron; Nature, 129, 312.
6. Zirkle, R.E., Aebersold, P.C. and Dempster, E.R., (1937), Relative Biological Effectiveness of Fast Neutrons and X-Rays Upon Different Organisms; Am. Journ. Cancer, 29, 556.
7. Lawrence, J.H. and Lawrence, E.O., (1936), The Biological Action of Neutron Rays; Proc. Nat. Acad. Sci., 22, 124.
8. Lawrence, E.O., (1940); The Biological Action of Neutron Rays; Radiol., 29, 313.

9. Stone, R.S., Lawrence, J.H. and Aebersold, P.C., (1940), A Preliminary Report on the Use of Fast Neutrons in the Treatment of Malignant Disease; Radiol., 35, 322.
10. Stone, R.S. and Larkin, J.C., (1942), The Treatment of Cancer with Fast Neutrons; Radiol., 39, 608.
11. Zirkle, R.E. and Lampe, I., (1938), Differences in the Relative Action of Neutron and Roentgen Rays on Closely Related Tissues; Am. Journ. of Roent., 39, 613.
12. Gray, L.H., Mottram, J.C., Read, J. and Spear, F.G., (1940), Some Experiments Upon the Biological Effects of Fast Neutrons; Brit. Journ. Radiol., 13, 371.
13. Gray, L.H. and Read, J., (1939), Measurement of Neutron Dose in Biological Experiments; Nature, 144, 439.
14. Aebersold, P.C. and Anslow, G.A., (1946), Fast Neutron Energy Absorption in Gases, Walls and Tissue; Phys. Rev., 69, 1.
15. Stone, R.S., (1948), Neutron Therapy and Specific Ionization, Janeway Memorial Lecture; Am. Journ. Roent. and Rad. Ther., 59, 771.
16. Lawrence, (1936), *ibid*.
17. Spear, F.G., Gray, L.H. and Read, J., (1938), Biological Effect of Fast Neutrons; Nature, 142, 1074.

18. Aebersold, P.C. and Anslow, G.A., (1939), The Ionization Produced in Thimble Chambers by Fast Neutrons; Phys. Rev., 55, 1134.
19. Aebersold, P.C. and Anslow, G.A., (1939), The Ionization Produced in Gases by Fast Neutron Irradiation; Phys. Rev., 55, 680.
20. Aebersold, P.C., (1939), The Production of a Beam of Fast Neutrons; Phys. Rev., 56, 714.
21. Stone, (1942), *ibid.*
22. Rossi, H.H., (1956), Radiation Dosimetry, edited by Hine, G.J. and Brownell, G.L., Academic Press, New York.
23. Gray, L.H., (1943), The Ionization Method of Measuring Neutron Energy; Proc. Cambridge Phil. Soc., 40, 72.
24. Failla, G., (1937), The Measurement of Tissue Dose in Terms of the Same Unit for All Ionizing Radiations; Radiol., 29, 202.
25. International Commission on Radiological Units, (1954), Radiological Units-International Commission Recommendations; Nucleonics, 12, 11.
26. International Commission on Radiological Units and Measurements, (1962), Report 10.a, Radiation Quantities and Units, (NBS Handbook 84).

27. Marinelli, L.D., (1953), Annual Review of Nuclear Science, Vol. 3, 261.
28. Bretscher, E. and French, A.P., (1944), Neutron Flux Measurements (D+D and C+D neutrons), Cavendish Laboratories, U.K. Report BR-386.
29. Bretscher, E. and French, A.P., (1944), The Measurment of Neutron Flux in the Presence of Gamma Rays, Cavendish Laboratories, U.K. Report BR-517.
30. Reinhardt, P.W., (1947), A Portable Instrument for the Detection of Fast Neutrons in the Presence of Gamma Radiation; Phys. Rev., 71, 836.
31. Marinelli, (1953), *ibid.*
32. Dainty, J., (1950), Report on Fast Neutron Dosimetry; CRM-482, Nat. Res. Council of Canada, Chalk River, Ontario.
33. Berlman, I.B., (1953), U.S. Atomic Energy Commission, ANL 4948.
34. Rossi, H.H. and Failla, G., (1956), Tissue-Equivalent Ionization Chambers; Nucleonics, February, 1956.
35. National Committee on Radiation Protection and Measurements, (1961), Measurement of Absorbed Dose of Neutrons and Gamma Rays; Report Number 25.

36. International Commission on Radiological Units and Measurements, (1962), Physical Aspects of Irradiation; Report 10.b.
37. Attix, F.H., Theus, R.B., Shapiro, P., Surratt, R.E., Nash, A.E. and Gorbics, S.G., (1973), Neutron Beam Dosimetry at the NRL Cyclotron; Phys. Med. Biol., 16, 497.
38. Bragg, W.H., (1912), Studies in Radioactivity, Macmillan Press, London.
39. Gray, (1936), *ibid.*
40. ICRU, (1977), *ibid.*
41. Attix, F.H., Theus, R.B. and Rogers, C.C., (1974), Proc. of 2nd Sym. on Neutron Dosimetry in Biology and Medicine, Neuherberg/Munchen.
42. Attix, F.H., August, L.S., Shapiro, P. and Theus, R.B., (1977), Scattered Radiation from a Neutron Collimator; NBS SP456 and Med. Phys., 4, 118.
43. Attix, F.H., Pearson, D.W. and Capestrain, R.R., Jr., (1977), An Improved Method for Measuring the n/γ Response Ratio for γ -Ray Dosimeters; USERDA Progress Report C00-1105-252, Madison, WI.
44. Attix, F.H., DeLuca, P.M., Jr., Pearson, D.W. and Goetsch,

- S.J., (1979), Measurement of Neutron/Gamma Sensitivity Ratios, USDOE Report COO-1105-264, Madison, WI.
45. Chenevert, G.M., DeLuca, P.M., Jr., Kelsey, C.A. and Torti, R.P., (1977), A Tritium Gas Target as an Intense Source of 14 MeV Neutrons; Nucl. Instr. and Methods, 145, 149.
46. DeLuca, P.M., Jr., Torti, R.P., Chenevert, G.M., DeTorie, N.A., Tesmer, J.R. and Kelsey, C.A., (1978), Performance of a Gas Target Neutron Source for Radiotherapy; Phys. Med. Biol, 23, 876.
47. Smathers, J.B., Otte, V.A., Smith, A.R., Almond, P.R., Attix, F.H., Spokas, J.J., Quam, W.M. and Goodman, L.J.; (1977), Composition of Al50 Tissue-Equivalent Plastic, Med. Phys., 4, 74.
48. Wagner, E.B. and Hurst, G.S., (1961), A Geiger-Mueller γ -Ray Dosimeter with Low Neutron Sensitivity; Health Physics, 5, 20.
49. Attix, (1977), Med. Phys., *ibid*.
50. Kuchnir, F.T., Vyborny, C.J. and Skaggs, L.S., (1975), A Precise Method for Measuring the Neutron Response of a Neutron-Insensitive Dosimeter; IAEA-SM- 193/51, Univ. of Chicago, Dept. of Radiology.
51. Attix, (1977), Med. Phys., *ibid*.

52. Attix, F.H., Pearson, D.W., Goetsch, S.J. and DeLuca, P.M., Jr., (1979), Comparative n- γ Dose Measurements in a Water Phantom for 14.8MeV Neutrons; USDOE Progress Report COO-1105-265, Madison, WI.
53. Attix, F.H., (1979), Report COO-1105-264. *ibid.*
54. Evans, R.D., (1955), The Atomic Nucleus, p. 787; McGraw-Hill, New York.
55. Pearson, D.W., Attix, F.H., DeLuca, P.M., Jr., Goetsch, S.J. and Torti, R.P., (1979), Ionization Error Due to Porosity in Graphite Ionization Chambers; USDOE Progress Report COO-1105-262, Madison, WI. and *Phys. Med. Biol.*, 25, 333.
56. Kuchnir, F.T., Vyborny, C.J. and Skaggs, L.S., (1975), Experimental Determination of the Neutron Sensitivity Function of a Dosimeter; Proc. of 2nd Sym. on Neutron Dosimetry in Biology and Medicine, Neuherberg, Munchen.
57. International Commission on Radiological Units and Measurements, (1978), An International Neutron Dosimetry Intercomparison, Report 27.
58. Shapiro, P., Attix, F.H., August, L.S., and Theus, R.B., (1976), Displacement Correction Factor for Fast-Neutron Dosimetry in a Tissue-Equivalent Phantom; *Med. Phys.*, 3, 87.

59. Attix, F.H., (1982), Displacement Correction Factors, in press.
60. Goodman, L.J., (1974), Uncertainty Analysis for Dosimetry in a Mixed Field of Neutrons and Photons; Proc. of 2nd Sym. on Neutron Dosimetry in Biology and Medicine, Neuherberg/Munchen.
61. Goodman, L.J., (1980), Uncertainties and their Treatment in Mixed-Field Dosimetry; Ion Chambers for Neutron Dosimetry, EUR 6782 EN, Radiobiological Institute TNO, Rijswijk, the Netherlands, p. 295.
62. DeLuca, P.M., Jr., Higgins, P.D., Attix, F.H. and Pearson, D.W., (1980), Progress in Development of a Tandem Irradiation Fast-Neutron and ^{60}Co Source Facility; USDOE Report DOE/EV/01105-275, Madison, WI.
63. Kahn, H., (1950), ; Nucleonics, 6, (5), p.27.
64. Kahn, H., (1950), ; Nucleonics, 6, (6), p.60.
65. Straker, E.A., Stevens, P.N., Irving, D.C. and Cain, V.R., (1970), The MORSE Code, ORNL-4585, Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, TN.
66. Engle, W.W , (1967), A User's Manual for ANISN, K-1693, Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge. TN.

67. Rhodes, W.A., and Mynatt, F.R., (1973), The DOT-III Two-Dimensional Discrete Ordinates Transport Code, ORNL-TM-4280, Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, TN.
68. Emmett, M.B., (1975), The MORSE Monte Carlo Radiation Transport System, ORNL-4972, Oak Ridge National Laboratory, Oak Ridge, TN.
69. Schaeffer, N.M., Editor, (1973), Reactor Shielding for Nuclear Engineers, p. 234, U.S. Atomic Energy Commission, National Technical Information Service, Springfield, VA.
70. Chilton, A.B., (1978), A Note on the Fluence Concept; Health Physics, 34, 715.
71. CASK, (1975), 40 Group Coupled Neutron and Gamma-Ray Cross Section Data. DLC-23, Radiation Shielding Information Center, Oak Ridge, TN.
72. FEWG1, (1977), 37 Neutron, 21 Gamma Ray Coupled, P_3 , Multigroup Library in ANISN Format, DLC-31, Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, TN.
73. Straker, (1970), *ibid.*
74. Herling, G.H., (1981), Personal Communication.

75. Kalos, M., (1963), On the Estimation of Flux at a Point by Monte Carlo; Nucl. Sci. Eng., 16, III.
76. Herling, G.H., (1975), Monte Carlo Calculations of Neutron and Photon Spectra for the INDI Project, NRL Report 7886, Naval Research Laboratory.
77. Herling, (1975), *ibid*.
78. Attix, F.H., Theus, R.B., Gorbics, S.G. and Rogers, C.C., (1975), Report on NRL Measurements as a Participant in the INDI, Naval Research Laboratory Memorandum Report 3051, Washington, D.C.
79. ICRU, (1978), *ibid*.
80. Barschall, H.H., (1976), The Production and Use of Neutrons for Cancer Treatment; Amer. Scien., 64, 668.
81. Barschall, H.H., (1978), Intense Sources of Fast Neutrons; Ann. Rev. Nucl. Part. Sci., 28, 207.
82. Weaver, K., Bichsel, H., Eenmaa, J. and Wootton, (1977), Measurement of Photon Dose Fraction in a Neutron Radiotherapy Beam; Med. Phys., 4, 379.
83. Frigerio, N., Coley, R. and Sampson, M., (1972), Depth Dose Determinations, I. Tissue-Equivalent Liquids for Standard Man and Muscle; Phys. Med. Biol., 17, 792.

84. Greene, D. and Thomas, R.L., (1968), An Experimental Unit for Fast Neutron Radiotherapy; Br. J. Radiol., 41, 455.
85. Broerse, J.J., Greene, D., Lawson, R.C. and Mijnheer, B.J., (1977), Operational Characteristics of Two Types of Sealed-Tube Fast-Neutron Radiotherapy Installations; Int. J. Radiation Oncology Biol. Phys., 3, 361.
86. Mijnheer, B.J., Zoetelief, J. and Broerse, J.J., (1978). Build-up and Depth-dose Characteristics of Different Fast Neutron Beams Relevant for Radiotherapy; Br. J. Radiol., 51, 122.
87. Schmidt, R., Magiera, E. and Scobel, W., (1980), Neutron and Gamma Spectroscopy for Clinical Dosimetry; Med. Phys., 7, 507.
88. McDonald, J.C., Barschall, H.H. and Howell, R.H., (1982), Comparison of Dose Determinations for 15-MeV Neutrons; Med. Phys., 9, 37.
89. Hertel, N.E. and Murphie, W.E., (1982), The Use of Neutron and Gamma-Ray Spectral Measurements and Calculations to Obtain Dosimetric Information for DT Neutrons; Med. Phys., In Press.
90. August, L.S., Shapiro, P., Theus, R. B. and Schell, M.C., (1977), Activation Dosimetry for Neutron-Gamma Dose Separation in a Tissue-Equivalent Phantom; Proc. of Third Sym. on

Neutron Dosimetry in Biology and Medicine, Neuherberg/
München.

91. Alsmiller, R. G., Jr. and Barish, J., (1974), Calculations
Pertaining to the Use of Fast (≤ 50 MeV) Neutrons in Cancer
Radiotherapy; ORNL-TM-4442, Oak Ridge National Laboratory, Oak
Ridge, TN.
92. Attix, F.H., (1975), *ibid.*
93. Herling, G.H., (1975), *ibid.*
94. Burger, G., Morhart, A. and Schraube, H., (1977), Recent
Results in Neutron Depth Dose Calculations; Proc. of Third
Sym. on Neutron Dosimetry in Biology and Medicine,
Neuherberg/München.
95. CASK, (1975), *ibid.*
96. Burger, G., (1977), *ibid.*
97. International Commission on Radiation Units and Measurements,
(1979), Average Energy to Produce an Ion Pair, Report 31.