

DESIGN OF A NEUTRON THERAPY FACILITY FOR A 30-INCH CYCLOTRON

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ABSTRACT

A Monte Carlo computer code was used to optimize the collimator and shielding design for a proposed fast-neutron therapy facility. Using time-of-flight techniques, neutron energy spectra were obtained at various angles to provide an accurate input for the computations. Measurements were made on thick beryllium targets at bombarding energies of 8.3 and 16 MeV for the deuteron beam, and at 14.8 MeV for the proton beam. On the basis of energy, angular distribution, and relative yields, the ${}^9\text{Be}(d,n)$ reaction appears to be by far more favorable than the ${}^9\text{Be}(p,n)$ reaction for neutron therapy.

The beam contamination due to scattering in the collimator was computed for a polyethylene collimator of simple design. Several shielding configurations were investigated, leading to a greatly simplified room shielding design.

INTRODUCTION

Results of radiobiological research¹ have indicated that the use of a beam of fast neutrons for radiotherapy may have a significant advantage over the use of conventional x rays and γ rays. A fast-neutron therapy facility (Fig. 1) is being planned at the Argonne Cancer Research Hospital, using a 30-inch cyclotron² which delivers external beams of 8.3-MeV deuterons, 14.8-MeV protons, and 20.3-MeV ${}^3\text{He}$ ions with currents of about 100 μA .

It is well known³ that at energies in the MeV range, a thick beryllium target gives the highest neutron yields, but little has been reported about the energy spectra. Neutron spectra from bombardment of a thick beryllium target by deuterons up to 3 MeV have been measured;⁴ some information is also available for 15-, 20-, and 24-MeV deuterons.^{5,6} By means of activation techniques, neutron spectra and angular distributions from the bombardment of a thick beryllium target by beams similar to those obtainable from our cyclotron have been obtained,⁷ but the results are not precise enough for

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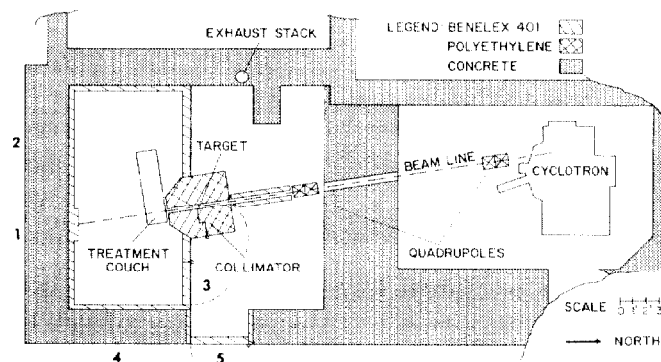


Fig. 1. The proposed fast-neutron therapy facility is located two flows below street level. The 24" concrete floor will be lined with 4" of Benelex; there will also be 9" of Benelex under the existing 12" concrete ceiling.

use as a starting point for calculations on collimator and shielding design. Furthermore, an accurate knowledge of the energy distribution of neutrons used in radiobiology and radiotherapy is necessary for improved dosimetry techniques, and to assist in the interpretation of biological responses of irradiated systems.

This paper describes the results of accurate determinations of the physical properties of neutron beams obtainable from a "compact" medical cyclotron, and shows the value of using Monte Carlo techniques for collimator and shielding design.

TIME-OF-FLIGHT MEASUREMENTS

Neutron energy spectra were measured by time-of-flight techniques, using a pulsed beam of deuterons or protons from the model FN Tandem Van de Graaff accelerator at the Argonne National Laboratory, which can accelerate singly charged particles from 4 to 17 MeV. Beam bursts of about 4 nsec duration were generated every 1068 nsec. The measurements were conducted in a large room, outside the accelerator vault, in order to reduce background effects to a minimum. Flight paths ranging from 1 to 3 m were used to emphasize different parts of the spectrum. A schematic diagram of the apparatus is shown in Fig. 2. The charged particle beam, after being magnetically analyzed, was collimated to $\frac{1}{4}$ " diameter by a tantalum collimator in front of the Faraday cup which was insulated and connected to a current integrator. The beryllium targets were always thicker than the range of the charged particles, and the entire target assembly was very light to avoid scattering of the neutron beam.

Neutrons and γ rays were detected by a scintillation counter consisting of a stilbene crystal, 2" in diameter and 1" thick, coupled to an RCA 8575 photomultiplier. The counting rate of the detector was kept to about 1000/sec by controlling the target current. Time of flight in the range of 0-400 nsec was measured with a conventional time-to-amplitude converter (TAC). The "start" signal was provided by the scintillation detector and the "stop", properly

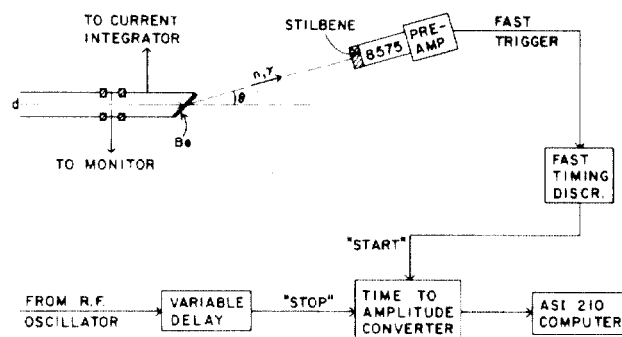


Fig. 2. Schematic diagram of the apparatus for time-of-flight measurements.

γ -ray peak, since the flight time of γ rays is known. Typically a spectrum was obtained in about 20 minutes.

delayed, was derived from the rf oscillator of the pulsed source. The time calibration and performance of the TAC were checked using a precision time calibrator;⁸ the output was stored on-line in a computer for preliminary data analysis.

A representative time-of-flight distribution is shown in Fig. 3. The zero time is determined from the position of the prompt

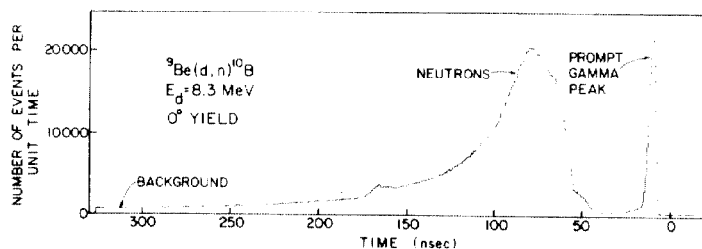


Fig. 3. Time-of-flight spectrum.

The data in Fig. 3 were not corrected for the energy dependence of the detector efficiency. Since the only constituents of stilbene are H and C, for which the scattering and interaction cross-sections are known, this efficiency can be calculated. The solid line in Fig. 4 is taken from the work of Jones and Toms.⁹ At energies below 1 MeV, the efficiency was measured relative to a long counter with a flat response,¹⁰ using monoenergetic neutrons.

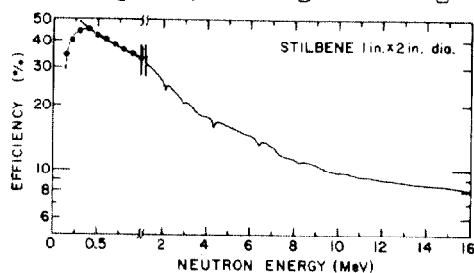


Fig. 4. Efficiency of the neutron detector. (Note break in the energy scale.)

than zero in Fig. 5 is consistent with previously reported data on

The energy spectra shown in Figs. 5, 6, and 7 were derived from time-of-flight distributions after background subtraction and correction for the counter efficiency. In Figs. 5 and 6, the area under each curve is proportional to the total yield at the indicated angle, and the sets of spectra in each figure are for an equal integrated charge at the target. The observed increase in the intensity of the higher-energy neutrons at angles larger

thin targets.¹¹ In Fig. 7 the spectra and yields are more nearly the same at all angles. Angular distributions of the total neutron yield for the three cases investigated are shown in Fig. 8.

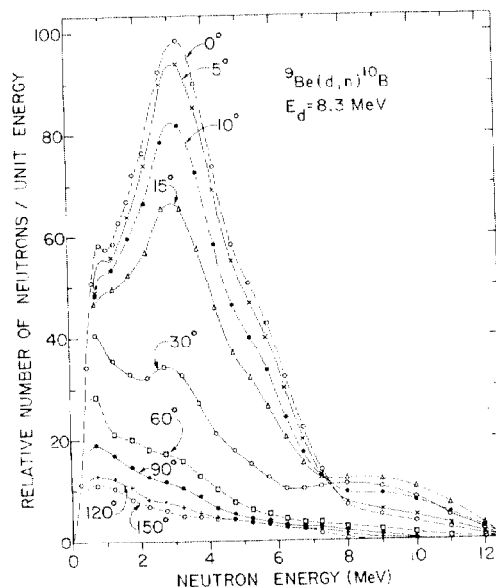


Fig. 5. Energy distributions

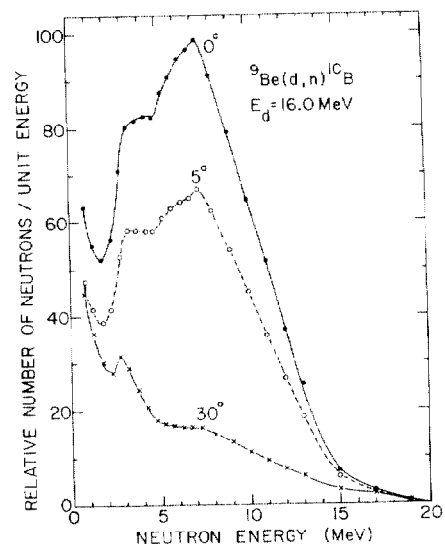


Fig. 6. Energy distributions

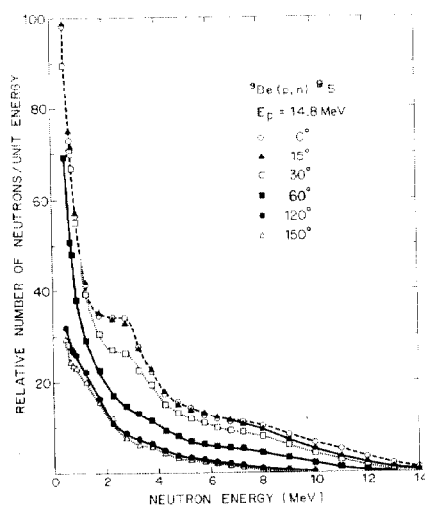


Fig. 7. Energy distributions

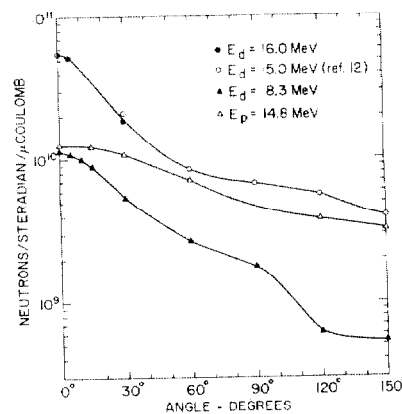


Fig. 8. Angular distributions of the total neutron yield from a thick beryllium target.

The total neutron yields integrated over 4π are (10.7, 2.7, and 7.1) $\times 10^{10}$ neutrons/ μC for the (d,n) reaction at 16.0 and 8.3 MeV and the (p,n) reaction at 14.8-MeV, respectively. Data on absolute

neutron yields from thick beryllium targets at these energies are scarce in the literature. Smith and Kruger¹³ using a 4π counter, obtained 3.2×10^{10} neutrons/ μC for 10-MeV deuterons on a thick beryllium target. This is in good agreement with our results according to the empirical $E_d^{3/2}$ law for the increase of neutron yield in the $^9\text{Be}(d,n)$ reaction.⁴

The production of neutrons at the target also gives rise to γ rays. Since we have no information about the spectral distribution of γ rays, we cannot obtain absolute γ flux values.

Our measurements on neutron energy and angular distributions indicate that the $^9\text{Be}(d,n)$ reaction is more favorable than the $^9\text{Be}(p,n)$ reaction for the production of fast neutrons for therapy.

MONTE CARLO COMPUTATIONS

A Monte Carlo program, operational on a CDC-3600 computer, was developed by N. A. Frigerio, and has been tested against independent computations and experiments.^{14,15} It utilizes available cross sections for neutron capture and differential scattering as well as for γ -ray interactions. It is an ideal computational tool for providing guidance on the optimization of collimator and shielding design. Since the $^9\text{Be}(d,n)$ reaction at a deuteron energy of 8.3 MeV will be used as a neutron source at the ACRH facility, all computations were made for this case only.

As a starting point, we made calculations for a simple geometry consisting of a high-density polyethylene¹⁶ cube, 1.5 m on each edge, with an opening 90 cm long having a uniform cross-sectional area of 10 cm \times 10 cm. The neutron source, placed at the center of the cube, was taken as a square 2 cm \times 2 cm generating 31,196 neutrons in 4π . Each neutron was followed until it left the polyethylene cube or was captured. The neutron-produced γ rays were also followed completely. The results are summarized in Table I.

Table I. Number of particles with their average energies and doses integrated over the field size, leaving the collimator in the forward direction within square fields of increasing sizes at 75 cm from the source. D_n and D_γ were computed using appropriate flux-to-dose conversion factors.^{17,18}

Field size (cm ²)	Neutrons NI [*] =0 NI [*] >0		\bar{E}_n (MeV)	No. of γ rays	\bar{E}_γ (MeV)	$10^3 D_n$ (rad cm ²)	$10^3 D_\gamma$ (R cm ²)
10 x 10	368	76	3.20	78	1.14	1470	39
12 x 12	381	97	3.16	106	1.05	1568	49
14 x 14	384	104	3.17	133	1.07	1604	63
16 x 16	384	112	3.15	159	1.06	1624	75
18 x 18	384	119	3.15	188	1.08	1645	90
150 x 150	384	134	3.15	1994	0.91	1686	815

*NI = Number of interactions in the polyethylene cube.

From the computations for this simple geometry it follows that:

1. In a 10 cm x 10 cm field at 75 cm from the source, 17% of the neutron flux is due to "inscattered" neutrons with an average energy of 1.8 MeV.
 2. The neutron dose decreases rapidly outside the 10 cm x 10 cm field.
 3. From the 31,196 neutrons emitted at the source, 518 left the front face of the collimator, and only 2 left through the sides.
 4. Of the neutron-produced γ rays 30,676 were from capture in hydrogen and 724 from inelastic collisions with carbon. Of the γ interactions, 96% consisted of Compton scattering; a total of 7024 γ rays, having an average energy of 850 KeV, escaped the collimator.
 5. An addition of 2% by weight of boron to the polyethylene will cause 97% of the capture events to occur in boron, with the emission of a 478-KeV γ ray rather than the 2.223-MeV γ resulting from capture in hydrogen. Under these conditions the γ attenuation in the collimator increases by a factor of about 5.
 6. In the 10 cm x 10 cm field, the γ dose contributed by the presence of the collimator is about 4% of the neutron dose. With the addition of boron it can be reduced to 0.4%. This is much lower than the gamma component from the source, which we estimate from our measurements to be 5-10% of the neutron dose.
 7. When the values quoted in items 4 and 5 are scaled up to the true source emission, 2.7×10^{12} n/sec for 100- μ A deuterons, the boron-loaded polyethylene collimator will behave as an extended source of γ rays yielding a dose of about 2 mR/min at one meter.
- High-density polyethylene was chosen for these calculations because of its simple composition and high concentration of hydrogen atoms ($8.08 \times 10^{22}/\text{cm}^3$), which results in a shorter relaxation length.

Langsdorf¹⁹ has carried out a computational analysis for neutron collimator design using an optical analogy. He suggests a collimator with two constrictions at critical points. Preliminary computations using such a collimator design indicate that the inscattered neutron flux can be reduced to < 5%.

B. Patient Treatment Room Design

The horizontal deuteron beam will be brought through the 6-ft wall of the cyclotron vault to the center of the collimator cube. The neutron irradiation facility will be built inside the adjacent room (Fig. 1) which requires additional shielding to ensure safe radiation levels. Benelex²⁰ was chosen over concrete because it is easy to fabricate, and has a neutron shielding efficiency similar to that of water,²¹ while the γ shielding efficiency is 0.62 that of concrete. Its simple composition allows more accurate calculations than for ordinary concrete. Furthermore, elements such as Na, Mg, Al, Si, Mn, Fe, Co, and Cu which are invariably present in concrete are activated under the bombardment of fast neutrons, causing a high residual γ -radiation level in the room. Another consideration is the albedo, which is much lower for Benelex.

Since the collimator will provide effective shielding at the source, Monte Carlo computations for the room shielding design were performed with a neutron beam originating at the target position and

restricted to a 10 cm x 10 cm field at 75 cm from the source. In order to take into account the effect of scattered radiation from the patient, we placed a tissue-equivalent²² man-like phantom with its head in the neutron beam at a distance of 83 cm from the source. Several shielding geometries were investigated, using 15,000 neutrons generated at the source to provide statistically significant figures. For each, the neutron and γ fluxes as well as the dose-equivalent values were computed at various test points.

The total dose rate in air for an uncollimated neutron beam was measured with a tissue-equivalent ionization chamber at 0° and at a distance of 1 m from the source. It was found to be 0.136 rads/min/ μ A for 8.3-MeV deuterons on a thick beryllium target, in agreement with measurements by Pinkerton.²³ Of this 5-10% is estimated to be due to γ rays. Using flux-to-dose conversion factors,¹⁷ we found the neutron dose per unit flux, averaged over the neutron spectrum, to be 3.47×10^{-9} rads/n/cm².

Table II summarizes the results of the computations for the final shielding configuration (Fig. 1), properly scaled up to a neutron source emission matching the dose measurement.

Table II. Radiation levels behind the final shielding

Location (see Fig. 1)	Thickness (inches)		(mrad/hr/100 μ A)	
	Benelex	Concrete	Neutrons	Gammas
1	8	30	.08	1.7
2	4	30	.005	1.2
3	8	--	.03	5.0
4	4	30	.0003	0.08
5	6	--	.0001	1.1
Ceiling	9	12	.0005	0.2
Floor	4	earth	--	--

The neutron facility is expected to be in operation for 15-20 hr/week. The proposed shielding is evidently more than adequate for radiation safety everywhere outside the room. A minimum of 4" of Benelex is proposed in order to keep the activation of the concrete and the albedo at a low value.

C. Depth Dose Distributions in the Phantom Head

Since the room shielding computations were done with a tissue-equivalent²² man-like phantom in the beam, they also provided a depth dose distribution within the phantom head. For a source-to-skin distance of 83 cm the neutron dose along the central axis falls to 50% of the entrance value at a depth corresponding to ~ 7 g/cm². When differences in geometry and techniques are taken into consideration, this result is in reasonable agreement with experimental measurements.²³

CONCLUSIONS

Our investigation indicates that, using 8.3-MeV deuterons on a thick beryllium target with careful collimation design, one can

obtain a fast neutron beam that is suitable for a clinical trial with patients having tumors in the head and neck regions.

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