FINAL REPORT

Reactions and Moderators for an Accelerator -Based Epithermal Neutron Capture Therapy Source for Cancer Treatment

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INTRODUCTION and BACKGROUND

The use of boron neutron capture therapy (BNCT) has been considered for nearly 30 years, and been practiced in Japan since the late 1970's. Early experiments in the USA were generally non-promising. However, new boron-containing ligand compounds were developed, which would seek out brain tumors. Concentration levels of the order of 30 micrograms of boron per gram of tissue become possible, and interest in the BNCT technique was revived in the USA beginning about 1985, with research reactors as the obvious source of the neutrons for the treatment. However, the limited number of research reactors in the USA (and the world) would mean that this treatment modality would be quite limited. With approximately 11,000 new cases of nominally inoperable brain cancer developing each year in the USA, it appeared that there would be a more general need for facilities to do BNCT. Consequently, the idea of having an accelerator source to produce the neutrons was developed. Accelerators do not have to be licensed by the NRC, nor would any activation products so produced be subject to NRC license. Furthermore, accelerators are much less expensive to build and install than are reactors (even the simple, ultra-safe systems such as the TRIGA).

This work originally had two goals:

- 1) Examine as many as possible reactions of charged particles on various targets of an accelerator, and determine those that would give high neutron yields of a convenient energy.
- 2) Determine, through calculations (using Monte Carlo stochastic computer codes), the best design for a moderator/reflector assembly which would give high thermal flux at a nominal 5 cm depth in the head of a patient, with minimal radiation dose from gamma rays and fast neutrons.

After completing these two studies, well within budget and time schedule, it was decided that the calculations were of questionable reliability without an experimental comparison, i.e. a benchmark. Hence it was proposed to add a third task, within the same initial budget, but with a longer time schedule, as follows:

- 3) Perform a benchmark experiment using a positive ion accelerator. The $\text{Li-7}(p,n)$ reaction was chosen for the benchmark, since it was readily available for most accelerators, and was one of the two highest yielding reactions from Task #1. Since the University of Missouri has no accelerator, possible accelerators at other universities were investigated, as to availability and cost. A unit having capability in the 2.5 MeV range was desired.

It was decided to perform the experiments at the Physics Department of the Idaho State University in Pocatello, Idaho, where a VandeGraff generator was available, at a nominal 2 MeV rating, but with operational capability up to about 2.3 MeV. Furthermore, a new radiofrequency quadrupole (RFQ) positive ion generator was being installed by the Idaho National Engineering Laboratory, at the Idaho State University Physics Department. It was indicated that the fixed energy (2.0 MeV) generator possibly could be available for use on the project.

2.0 BRIEF CHRONOLOGICAL SUMMARY

Initiation: The project commenced in November 1990, with simultaneous studies on tasks #1 and #2.

Task #1 - Study of Possible Reactions: The reactions studied (Task #1) were all obtained from literature, much of it dating back to the 1940's. . The Li-7(p,n) and H-3(p,n) reactions were the highest yielding reactions, with yields of 1.5 and 2.2 neutrons per 10,000 protons, respectively, for proton energies that would not give neutrons energies above 800 keV. Tables 1 through 4 give the results of the various reactions studied from the literature. For the most promising of these, the differential cross section data was used, along with energy loss of the charged particles in the target, to compute overall neutron production from a thick target. The last column in these tables gives the results of these calculations. In the case of the Li-7(p,n) reaction, a detailed study of the calculated differential cross section yields for the neutrons was done and compared with available experimental data in the literature. Figure 1 shows the results of these calculations compared with experiments, for the proton energy of 1.95 MeV, just slightly above the 1.88 MeV threshold energy. A summary of these results was presented in Reference #1. (Appendix A).

This task was essentially completed in March 1992

Task #2 - Moderator-Reflector Design: For the moderator-reflector design, only the Li-7(n,p) and H-3(p,n) reactions were studied. This effort had a criteria to limit the energy of the protons to levels that would not produce excessive inelastic scattering, and hence significant gamma ray production. For the Li-7(p,n) reaction, the proton energy was limited to 2.5 MeV, giving a corresponding maximum neutron energy of 0.787 MeV, as summarized below:

Li-7(p,n) Be-7* Reaction

Q-value of the reaction	-1.645 MeV
Threshold Energy, E-threshold	1.881 MeV
Neutron Energy at E-threshold	0.029 MeV
Incident Proton Energy for Study	2.50 MeV
Maximum Neutron Energy	0.787 MeV
Total Neutron Yield	1.48 neutrons per 10,000 protons

The H-3(p,n)He-3 reaction represents a difficult one to design the target. Nevertheless the computer study was made, assuming a "thick" H-3 target. The maximum neutron energy was limited to 1 MeV, thus resulting in the need for only 1.77 MeV proton energy. The results were as follows:

H-3(p,n)He-3 Reaction

Q-value of the reaction	-0.764 MeV
Threshold Energy, E-threshold	1.019 MeV
Neutron Energy at E-threshold	0.064 MeV
Incident Proton Energy for Study	1.77 MeV
Maximum Neutron Energy	1.0 MeV
Total Neutron Yield	2.54 neutrons per 10,000 protons

The moderator design concluded that, in the case of the Li-7 reaction, beryllium made the best moderator (slightly better than heavy water), and MgO was a slightly better reflector than the convention use of aluminum oxide.

The results of this study are given in more detail in Reference 2 (Appendix B). This task was essentially complete by August 1992, and a publication was submitted to the 4th International Conference on Neutron Capture Therapy held in Columbus, Ohio in September 1992 (Ref. 2).

Task #3 - Benchmark Experiment

The first attempt at a benchmark experiment was made in the Spring of 1992, using the VandeGraff generator at the Physics Dept. of Idaho State University. The machine was operated at 2.2 to 2.25 MeV, because it was desired to obtain results as close to the design proton energy (2.5 MeV) of Task 2 as possible, and also to be on the flat part of the curve. The experiments, which were run for a week, but were largely a failure because of voltage breakdown in the machine, which made it impossible to hold a steady beam for more than a few seconds. However, lessons were learned on the sensitivity of the integration flux measurement foils (gold and indium) to the neutron levels in the experiment. Beam current averaged no more than 4 microamps.

In October 1993 the measurements were again attempted, this time on the 2.0 MeV fixed energy RFQ machine. However, after setting up the experiment, the machine failed to work because of a burned out filament and no replacements available from the manufacturer.

In May 1994, the experiment was again attempted with the RFQ machine, this time successfully with a number of variations in the configurations. Even some multiplying configurations were measured. Machine output average about 70 microamps. Neutron production was about 4 times that from the VandeGraff experiments in April 1992, at the higher energy. Comparisons were made with each configuration and the results of the MCNP Monte Carlo computer code calculations of the same configuration. The conclusion was that the computer model

substantially over-predicted the flux at the 3.8 cm depth in the phantom, typical of a tumor depth. The predictions at the target and the entrance to the phantom agreed well with experiment. Gamma measurements were inconclusive, however, because of inadequate reliability of the TLD measurements.

The allowed intensity of the proton beam on the target, without damaging the target, is the main limitation to obtaining the necessary neutron yield. This limitation has been readily recognized by a number of other investigators as well. However, upon completing these experiments at the lower energy (2.0 MeV protons), it became obvious that the lower maximum neutron energy (0.228 MeV) would make it feasible to use much less moderator around the target, and hence the patient could be placed much closer to the target. Studies on this option have been initiated using the computer and the MCNP code. These are being carried out at the University of Missouri, and some additional measurements using internal funding are planned for the future, in cooperation with Idaho State University Department of Physics.

3.0 Personnel Involved, Degrees Awarded from Work, and Publications

3.1 Personnel

Dr. Robert Brugger was the original principal investigator, assisted by Dr. Jay Kunze. Dr. Brugger retired from the University of Missouri in August 1991, and continued his work related to BNCT, but at Brookhaven National Laboratory. Dr. Kunze continued supervising the work, and defined Task #3, carrying it to its conclusion, as stated above. However, Dr. Brugger continued to keep informed of the progress of the work and provided advice.

Students who received degrees from this work were:

- 1) Lo, Yi-Wang (Frank), PhD (scheduled for August 1995), "Comparison of Monte Carlo Techniques with Integral Neutron Spectral Measurements on a Accelerator Neutron Source for a BNCT"
- 2) Luo, Jianbin, M.S., Nuclear Engineering, 1995, "Benchmark Experiments for BNCT System"
- 3) Kumuduni Kulasekere, MS, Nuclear Engineering, 1994. "A Calculation Using 'QAD' to Evaluate Gamma Dose Rates Produced by Accelerator-based Neutron Sources for BNCT"
- 4) Terry H. Wu, PhD, Nuclear Engineering, 1992, "A Low Energy Accelerator-Based Neutron Source for Neutron-Capture Therapy"
- 5) D. Jay Moreno Jalandoni, MS, Medical Physics emphasis in Nuclear Engineering, 1992, "An Experiment to Evaluate Calculations for the Design of an Accelerator for Boron Neutron Capture (Cancer) Therapy"

In addition, one student (Rajat Kudchadker) is well along the way toward completing his dissertation, having participated actively in the benchmark experiments of Task #3. However, his work is not being funded by the DOE grant (which has expired).

Dr. Frank Harmon, Chairman of Physics at Idaho State University, was most instrumental in making Task #3 possible to accomplish, and participated directly in the benchmark experiments.

It was through his efforts that the RFQ machine, owned by the Idaho National Engineering Laboratory, was made available for this work.

3.2 Publications

- 1) Brugger, R.M., X. Luo, J.F. Kunze, "Targets for the Production of Neutrons from Low Energy Positive ion Accelerators", TRANS. Am. Nuclear Society, 65, p.146 June 1992
- 2) Wu, H-S, R.M. Brugger, and J.F.Kunze, "Low Energy Accelerator-based Neutron Source for Neutron Capture Therapy", Advances in Neutron Capture Therapy , edited by A. Soloway et al, p.105, (Pergamon, 1993)
- 3) Kunze, J.F., R.Kudchadker, J.Luo,Y.Lo, F. Harmon, "Results of Benchmark Experiment of Accelerator/Moderator Configuration for BNCT - Using RFQ Accelerator", Proceedings of 1st International Conference on Accelerator Sources for Neutron Capture Therapy, Sept. 1994, p.243 to 253.

4.0 EXPENDITURES

Total expenditures on the Project, over the 4-year period, were approximately \$200,000, of which \$143,700 was provided by the U.S. Dept. of Energy grant, and the remainder was "cost sharing" from the University of Missouri.

Table 1 (p,n) Reactions with $E_{nmax} = 800$ keV.

Reactions	Z	Melt.Pt (K) [21]	Isotope Compo. (%) [22]	Linear Exp. 10 ⁻⁶ /K [23]	Target Thick- ness (μ m)	Reaction Thresh. Epth (MeV)	Enth (keV)	Epmax (MeV)	n Yield (10 ⁻⁶ n/p)
Zr96-Nb96*	40	2125	*2.8	23.3	11.7	0.6062	0.069	1.401	
Ta181-W181	73	3287	99.99	6.3	8.1	0.9751	0.032	1.771	5.91
Mn55-Fe55	25	1517	100	21.9	10.2	1.0279	0.356	1.814	
Ga71-Ge71*	31	303*	39.9	-	14.8	1.0320	0.216	1.821	
Mo100-Tc100*	42	2890	*9.63	4.8	10.0	1.1314	0.306	1.923	0.210
Ge73-As73*	32	1214	*7.8	5.8	17.3	1.1376	0.226	1.926	
K41-Ca41*	19	336*	*6.73	-	71.6	1.2333	0.763	2.011	
Tel30-I130	52	723	33.8	18.8	20.1	1.2495	0.080	2.043	
Ru101-Rh101	44	2523	17.0	6.4	9.3	1.3495	0.142	2.140	
Cr53-Mn53*	24	2130	*9.50	4.5	12.3	1.4061	0.529	2.187	
Ti49-V49 *	22	1943	*5.5	8.7	19.0	1.4121	0.630	2.192	
V51-Cr51*	23	2175	*9.75	8.5	15.3	1.5642	0.637	2.342	0.436
Fe57-Co57*	26	1809	*2.2	11.9	10.9	1.6480	0.541	2.428	
As75-Se75	33	1081	100	15.5	19.9	1.6693	0.032	2.454	1.32
Zr94-Nb94	40	2125	17.38	5.7	18.7	1.6993	0.208	2.487	0.927
Zn67-Ga67*	30	693	*4.1	30.3	15.7	1.8095	0.434	2.591	
Li7-Be7	3	454	92.5	47	98.8	1.8804	32.045	2.513	145.0
Co59-Ni59	27	1768	100	13.1	12.5	1.8874	0.582	2.665	3.19
Tel28-I128	52	723	31.69	18.8	25.7	2.0522	0.137	2.842	
Be9-B9	4	1560	100	11.5	37.3	2.0567	22.609	2.714	
Zr91-Nb91	40	2125	11.22	5.7	20.8	2.0669	0.569	2.852	2.44
Cu65-Zn65	29	1358	30.83	16.6	14.4	2.1663	0.556	2.944	2.52
Se77-Br77*	34	494	*7.6	45.8	28.3	2.1751	0.399	2.987	3.59
Cr54-Mn54*	24	2130	*2.37	4.5	13.9	2.1974	0.812	2.97	
Mo98-Tc98	42	2890	24.13	4.8	14.7	2.3904	0.274	3.175	2.01
Mo95-Tc95	42	2890	15.92	4.8	15.2	2.5098	0.306	3.293	3.52
Se80-Br80	34	494	49.7	45.8	32.1	2.6876	0.462	3.466	3.48
Zr92-Nb92	40	2125	17.15	5.7	24.9	2.8170	0.369	3.598	2.37
Sc45-Ti45	21	1812	100	10	37.4	2.9087	1.556	3.669	0.942
Bi11-C11	5	2300	80.1	4.7	32.2	3.0177	23.495	3.66	37.3
Ga69-Ge69*	31	303*	60.1	-	26.6	3.0513	0.707	3.824	
Sn120-Sb120	50	505	32.59	22.2	28.9	3.4921	0.273	4.275	
Zn68-Ga68	30	693	18.8	30.3	21.4	3.7566	0.907	4.525	
Mo96-Tc96	42	2890	16.68	4.8	19.6	3.7606	0.46	4.538	3.89
Ti47-V47 *	22	1943	*7.3	8.7	34.6	3.7865	1.888	4.540	
Cu63-Zn63	29	1358	69.17	16.6	19.9	4.2146	1.191	4.977	8.05
Se78-Br78	34	494	23.6	45.8	44.1	4.4127	0.821	5.182	4.80
Sr88-Y88	38	1041	82.58	22.5	82.4	4.4515	0.653	5.224	5.45
Ru100-Rh100	44	2523	12.6	6.4	18.5	4.4567	0.508	5.232	
Ca44-Sc44*	20	1112	*2.09	22.4	116.2	4.5332	2.599	5.277	
Cd110-In110	48	594	12.49	30.9	29.1	4.7565	0.45	5.533	
Ni62-Cu62*	28	1726	*3.59	13.5	19.0	4.8066	1.413	5.565	
Bi10-C10 *	4	2300	*9.9	11.5	45.9	4.8790	44.851	5.441	
Ti48-V48	22	1943	73.8	8.7	41.2	4.8987	2.383	5.644	
Mg26-Al26	12	922	11.01	25.0	79.9	4.9728	7.942	5.673	
Na23-Mg23*	11	371*	100	71*	127.9	5.0511	10.209	5.738	
Mo94-Tc94*	42	2890	*9.25	4.8	20.9	5.0962	0.662	5.868	0.0218
Mg25-Al25	12	922	10.00	25.0	83.4	5.2658	9.1	5.959	
Fe56-Co56	26	1809	91.72	11.9	23.5	5.4469	1.972	6.197	
Cr52-Mn52	24	2130	83.79	4.5	25.8	5.5985	2.349	6.344	6.29
Al27-Si27	13	933	100	23.2	61.1	5.8010	8.707	6.495	2.57
Si29-P29 *	14	1685	*4.67	2.6	68.4	5.9252	7.764	6.625	

Se76-Br76*	34	494	*9.0	45.8	48.1	5.9636	1.191	6.724	7.57
Zn66-Ba66	30	693	27.9	30.3	29.6	6.0485	1.598	6.803	
Sr86-Y86 *	38	1041	*9.86	22.5	90.4	6.1264	0.961	6.891	6.97
P31-S31*	15	317*	100	-	98.3	6.6246	7.446	7.125	
Ni60-Cu60	28	1726	26.10	13.5	24.9	7.0256	2.264	7.77	
K39-Ca39*	19	336*	93.26	-	234.0	7.4938	5.632	8.206	
Zn64-Ga64	30	693	48.6	30.3	36.1	7.9763	2.287	8.72	
Ti46-V46 *	22	1943	*8.0	8.7	58.7	9.0087	4.387	8.73	
Fe54-Co54*	26	1809	*5.8	11.9	34.3	9.2006	3.728	9.927	
Ni58-Cu58	28	1726	68.27	13.5	31.2	9.5123	3.36	10.244	
Si32-Cl32	16	1685	95.02	2.6	131.6	13.9606	16.3	14.604	
Mg24-Al24	12	922	78.99	25.0	164.3	15.2815	31.292	15.866	
Ca40-Sc40	20	1112	96.94	22.4	258.0	15.4880	9.537	16.154	
Si28-P28	14	1685	92.23	2.6	125.9	15.6642	29.47	16.275	
Cl2-N12	6	4100	98.9	1.1	95.5	19.6484	123.092	20.000	

* -- Low melting point, & low percentage in composition of the natural element.

E_{pth} --- Energy of proton at threshold.

E_{nth} . --- Kinetic energy of the produced neutrons at threshold in the forward direction
in the laboratory system.

E_{pmax} --- Kinetic energy of proton necessary to produce 800 keV neutrons in the
forward direction in the laboratory system.

Table 2a. Target design with $E_{nmax} = 800$ keV.

Elements	Melt. Pt. (K)	Composition (%)	Thresh. (MeV)	Neutron Energy at Epth(keV)	Max.Proton Energy (MeV)	Neutron Yield (10^{-6} n/p)
Ta-181 or Mn-55	3287 1517	99.988 100	0.98 1.03	0.0319 0.356	1.77 1.81	5.91 ---
Zn-67 or Li-7	693 453.7	4.11 92.5	1.81 1.88	0.434 32.05	2.59 2.51	--- 148.00
Se-80	494	49.7	2.69	0.462	3.47	3.48
Sn-120	505	32.59	3.49	0.273	4.28	---
Cu-63	1358	69.17	4.21	1.191	4.98	8.05
Mg-26	922	11.01	4.97	7.942	5.67	---
Cr-52	2130	83.79	5.60	2.349	6.34	6.29

Total Yield: $\sim 1.84 \times 10^{-4}$ n/p.

Table 2b. Target design with $E_{nmax} = 800$ keV.

Element	Melt.pt (K)	Compositio n (%)	Thresh. Epth (MeV)	Neutron Energy at Epth(keV)	Max.Protn Energy (MeV)	Neutron Yield (10^{-6} n/p)
Mn-55	1517	100	1.0279	0.356	1.814	
Te-128	723	31.69	2.0522	0.137	2.842	
Sn-120	505	32.59	3.4921	0.273	4.275	
Cu-63 or Sr-88	1358 1041	69.17 82.58	4.2146 4.4515	1.191 0.653	4.977 5.224	8.05 5.45
Mg-25	922	10.00	5.2658	9.1	5.959	

Total yield: $\sim 4.025 \times 10^{-5}$ n/p.

Table 2c. Target design with $E_{n\max} = 800$ keV.

Elements	Melt. Pt. (K)	Composition (%)	Thresh. (MeV)	Neutron Energy at Epth (keV)	Max.Proton Energy (MeV)	Neutron Yield ($10^{-6}n/p$)
Zr-96	693	2.8	.61	0.0689	1.4	--- x
Cr-53	2130	9.501	1.41	0.529	2.19	---
Cr-54	2130	2.365	2.20	0.812	2.97	--- x
B-11	2300	80.1	3.02	23.495	3.66	---
Zn-68	693	18.8	3.76	0.907	4.525	---
Ca-44	1112	2.086	4.53	2.599	5.277	--- x
Mg-25	922	11.01	5.27	9.10	5.959	---

Table 2d. Target design with $E_{n\max} = 800$ keV.

Elements	Melt. Pt. (K)	Composition (%)	Thresh. (MeV)	Neutron Energy at Epth(keV)	Max.Proton Energy (MeV)	Neutron Yield ($10^{-6}n/p$)
Mo-100	2890	9.63	1.13	0.306	1.92	0.210 x
Co-59 or	1768	100	1.89	0.582	2.67	3.19
Te-128 or	723	31.69	2.05	0.137	2.84	---
Cu-65	1358	30.83	2.17	22.61	2.94	2.52
Se-80 or	494	49.7	2.69	0.462	3.47	3.48
Zr-92 or	2125	17.15	2.82	0.369	3.60	2.37
Sc-45	1812	100	2.91	1.556	3.67	0.942
Sn-120 or	505	32.59	3.49	0.273	4.28	---
Zn-68	693	18.8	3.75	0.907	4.53	---
Cu-63 or	1358	69.17	4.21	1.191	4.98	8.05
Sr-88 or	1041	82.58	4.45	0.653	5.22	5.45
Ca-44	1112	2.086	4.53	2.599	5.28	--- x
Mo-94 or	2890	9.25	5.10	0.662	5.87	0.0218 x
Mg-25 or	922	10	5.27	9.10	5.96	---
Fe-56 or	1809	91.72	5.45	1.972	6.20	---
Al-27	933	100	5.80	8.707	6.50	2.57

Total yield: $\sim 1.57 \times 10^{-5} n/p$.

Table 3 Target design with $E_{nmax} = 500$ keV.

Elements	Melt. Pt. (K)	Isotope Compositin (%)	Thresh. (MeV)	Neutron Energy at Epth (keV)	Max.Proton Energy (MeV)	Neutron Yield (10^{-6} n/p)
Ta-181	3287	99.988	0.975	0.0319	1.472	0.345
V-51	2175	99.75	1.564	0.637	2.044	0.264
Te-128	723	31.69	2.052	0.137	2.543	---
Mo-95 or	2890	15.92	2.51	0.306	2.995	1.29
Se-80	494	49.7	2.69	0.462	3.170	1.42
Sc-45 or	1812	100	2.91	1.556	3.374	0.412
Ga-69	303	60.1	3.051	0.707	3.528	---
Sn-120	505	32.59	3.49	0.273	3.977	---
Cu-63	1358	69.17	4.21	1.191	4.683	2.40
Cd-110	594	12.49	4.757	0.450	5.237	---
Mg-25	922	10.00	5.266	9.10	5.675	---

Total yield: $\sim 9.2 \times 10^{-6}$ n/p.

Table 4. Target design with $E_{n\max} = 50$ keV.

Elements	Melt. Pt. (K)	Compositin (%)	Thresh. (MeV)	Neutron Energy at Epth(keV)	Max.Proton Energy (MeV)	Neutron Yield ($10^{-6}n/p$)
Ta-181	3287	99.988	0.975	0.0319	1.0235	
Mn-55	1517	100	1.028	0.356	1.072	
Mo-100	2890	9.63	1.131	0.306	1.178	x
K-41	336	6.73	1.233	0.763	1.274	x
Ru-101	2523	17.0	1.349	0.142	1.396	
Cr-53	2130	9.501	1.41	0.529	1.4485	x
V-51	2175	99.75	1.564	0.637	1.6056	
Fe-57	1809	2.2	1.648	0.541	1.69	x
Zr-94	2125	17.38	1.699	0.208	1.7445	
Zn-67	693	4.1	1.81	0.434	1.853	x
Co-59	1768	100	1.887	0.582	1.929	
Te-128	723	31.69	2.052	0.137	2.098	
Cu-65	1358	30.83	2.166	0.556	2.208	
Cr-54	2130	2.365	2.197	0.812	2.238	x
Mo-98	2890	24.13	2.390	0.274	2.435	
Mo-95	2890	15.92	2.510	0.306	2.554	
Se-80	494	49.7	2.687	0.462	2.73	
Zr-92	2125	17.15	2.817	0.369	2.861	
Sc-45	1812	100	2.909	1.556	2.945	
B-11	2300	80.1	3.018	23.5	3.025	
Ga-69	303	60.1	3.051	0.707	3.092	
Sn-120	505	32.59	3.492	0.273	3.537	

Total yield: $\sim 2 \times 10^{-7}$ n/p.

x ---- The composition of the isotope is too small to make target, but enriched material might be available for the application.

Figure 1. Differential Cross Section of Li-7 for Proton Energy at 1.95 MeV.

Appendix 1

Brugger, RM., X. Luo, J.F. Kunze, "Targets for the Production of Neutrons from Low Energy Positive ion Accelerators", TRANS. Am. Nuclear Society, 65, p.146 June 1992.

gists, radiobiologists, and dermatologists on the many technical and scientific tasks that must be resolved before human trials are possible. This paper reviews some aspects of this project.

Neutron capture therapy is a bimodal therapy requiring the delivery of the neutron capture agent to the cancer cells, where it must concentrate to significantly higher levels than in healthy tissue, and the delivery of neutron irradiation to the tumor with suitable neutron beams. Currently, NCT with ^{10}B as the capture agent (BNCT) offers the most promise. The results reported here are based on the ^{10}B delivery agent boronophenylalanine (BPA). Methods to quantify the BPA (^{10}B) in blood and tissue are necessary. These techniques should be sensitive down to at least the parts-per-million level, should be suited for rapid and accurate quantification in bulk samples, and, if at all possible, should allow the microscopic determination of ^{10}B with subcellular spatial resolution. Toxicity and efficacy studies with animals are a necessary prelude to testing a new therapeutic modality in humans and have been completed to a large extent.

Neutron beams for NCT were a major consideration in the original design of the MIT research reactor (MITR) and received further emphasis during the upgrading of this reactor to MITR-II. At the time of the upgrade in the mid-1970s, high-quality thermal neutron beams were judged to be the most desirable for NCT. Such a beam was constructed with small contamination from fast neutrons and gamma rays. Fortunately, however, a spectrum shifter was also incorporated into the beam line, which could be used to provide an intense source for an epithermal beam. Epithermal beams have deeper useful penetration in tissue, 7 to 8 cm versus 3 to 5 cm for thermal beams. For successful brain-cancer therapy, it is believed that it must be possible to achieve a therapeutic effect at all depths in the brain. This requires epithermal neutrons. An epithermal beam has been constructed and implemented on the MITR-II medical beam line, which permits BNCT with therapeutic effectiveness at any location in the human brain. Figure 1 shows the MITR-II, the medical room below the reactor core, and the beam line with the epithermal neutron filter installed. The filter uses sulfur and aluminum to preferentially remove fast neutrons and to pass epithermal neutrons. Cadmium is used to remove thermal neutrons, and a final shield of bismuth removes unwanted gamma rays.

Boron quantization and imaging at the microscopic level are based on prompt gamma analysis and on track etch autoradiography. A prompt gamma facility has been constructed based on a diffracted beam. This unusual approach provides a slow neutron current in the sample that is considerably lower than can be achieved by a direct reactor beam. However, the background at the solid-state detector is greatly decreased by using Bragg diffraction. This permits the optimum placement of the germanium detector, and the increased solid angle makes up for the intensity lost in the Bragg diffraction. Furthermore, the lower intensity beam with greatly reduced fast neutron and background components greatly facilitates access to the sample position with minimal shielding. Very small samples, e.g., 10^{-2} cm^3 , with low concentrations of ^{10}B are analyzed by quantitative neutron track etch autoradiography (QNR). The QNR is cross-calibrated with the prompt gamma.

To determine where the ^{10}B is absorbed within the cell structure, we have developed a unique technique, high-resolution autoradiography (HRA). This HRA has adequate resolution to determine the spatial distribution of ^{10}B on the subcellular level, a critical element in the assessment of neutron capture effectiveness. The resolution is 2 to 3 μm , and the sensitivity is better than 1 ppm.

A series of small-animal experiments has been carried out to assess the toxicity of BPA and to determine the efficacy of neutrons plus BPA as compared to conventional therapy modalities. Toxicity and biodistribution studies of the BPA in a limited number of human volunteers have also been completed.

2. Targets for the Production of Neutrons from Low-Energy Positive Ion Accelerators, Robert M. Brugger,* Xunshan Luo, Jay F. Kunze (Univ of Missouri-Columbia)

INTRODUCTION

To produce neutrons for neutron capture therapy at a hospital, a compact positive ion accelerator, such as a radio-frequency quadrupole machine, with a moderator blanket would be more readily accepted than a reactor with a moderator. Unfortunately, designs to date of accelerator-based epithermal neutron beams fall short of similar beams from reactors of a few megawatts. To try to identify improvements, we have evaluated many neutron-producing reactions. The $^7\text{Li}(p,n)$ and $^3\text{H}(p,n)$ continue to be the most promising for neutron yield, but both have troubles with target design.

For the $^7\text{Li}(p,n)$ reaction, the threshold is 1.88 MeV, but protons up to 2.5 MeV can be used with a thick target before gamma rays are produced by excitation of the first excited state in ^7Be . We calculate the yield of neutrons to be 1.48×10^{-6} neutrons per proton. However, for this target, three-fourths of the heat is produced by protons below the threshold. At 2.5 MeV, the maximum neutron energy is 800 keV, and a moderator must surround the target to reduce the energies of the neutrons to less than ~ 10 keV. This moderator significantly reduces the beam current of epithermal neutrons so that proton currents of 20 mA or more must be used to produce an epithermal beam of $>1 \times 10^9 \text{ n}\cdot\text{cm}^{-2}\cdot\text{s}$. Lithium or hydrogen targets that can withstand 50 kW of heat are difficult to design. It is suggested that a target of lithium on a high-conductivity backing would have to be 15 cm in diameter to spread this heat load.¹ Such sizes compromise the moderator design. If a target could be designed to more effectively use the protons in the energy range where neutrons are produced, these target design problems might be overcome.

MULTIPLE-LAYER TARGETS

Solid targets are usually designed to have one thin layer of the metal on top of a backing. If the target could be a stack of thin layers of elements in a sequence in which the highest threshold energy element is in the first layer closest to the incoming protons while the lowest threshold energy element is in the last layer, one might use more of the proton's energy to increase the neutron yield and less of the proton energy unconstructively by merely producing heat. Each layer would be thin and take advantage of the proton energy in a range where that material had a reasonable cross section for the neutron-producing reaction.

We have examined the literature for various reactions that produce neutrons from relatively low energy protons and have identified 67 such reactions below 20 MeV, and 16 of these are below the threshold of the $^7\text{Li}(n,p)^6\text{Li}$ reaction of 1.88 MeV. Unfortunately, the neutron yield for many of these latter reactions is unknown from the literature, and for those which are known, the yield is usually quite small compared to the ^7Li reaction yield. The lowest threshold was 0.6 MeV (^{90}Zr).

Table I provides an example of a layered target design in which seven materials, including ^7Li , have been arranged in a multiple target layer arrangement. The maximum proton beam energy is 5 MeV, and each stage allows production of neutrons up to 800 keV before the next target material is introduced. The overall yield from this arrangement is 20% more than the yield from a 2.5-MeV beam on ^7Li . This would not seem to be a useful gain, obtaining only 25% more production from a doubling of the proton beam power. Enhancing just a ^7Li target (2.5-MeV maximum proton energy) to lower energies

*Currently at Brookhaven National Laboratory.

TABLE I
Multiple-Layer Target Design

Material	Melt T (K)	Threshold Proton, E (MeV)	Maximum Proton Energy (MeV)	Neutron Yield
Enriched ^{96}Zr	2125	0.606	1.0	6 ^a
^{181}Ta	3287	0.98	1.77	5.9
^7Li	454	1.88	2.51	148
Enriched ^{80}Se	494	2.69	3.5	3.5
Enriched ^{120}Sn	505	3.49	4.28	6 ^a
Enriched ^{63}Cu	1358	4.2	4.98	8
Total yield (10^{-6} n/p)				178

^aEstimated yield; no data found.

by adding a ^{181}Ta layer and a ^{96}Zr layer under the ^7Li layer enhances the output by only 8%.

REACCELERATED PROTON TARGETS

Another potential solution for increasing the yield without increasing the nonproductive proton beam energy is to construct the target in layers of material separated by vacuum and reaccelerate the protons between layers, such as with 100-keV potential differences. Each target layer would have just the appropriate thickness for the protons to lose on the order of 100 keV of energy. These layers could continue to be added until the reacceleration did not bring the protons back to the threshold. It is best to have as much target material as possible see the higher energy protons because there the total cross section and the yield of neutrons are the greatest. Target thickness for a given energy loss is another advantage, being ~15% more at 2.5 MeV than at 1.88 MeV for lithium. However, the

thickness of such a lithium layer is only 35 μm for a 200-keV energy loss at the 2.4-MeV level. Clever target design, such as by use of a small honeycomb, is required to remove the heat generated without sacrificing any of the proton energy above 1.88 MeV in materials other than lithium. The benefits are worth seeking, however, because every 100-keV reacceleration, especially in the 2.5-MeV region, will enhance the target output by ~20%.

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3. Potential Impact of Low-Enriched Uranium Research Reactor Core Conversions on Irradiation Facilities for BNCT, Lisa A. Heimberger, Tunc Aldemir (Ohio State)

INTRODUCTION

A critical requirement for boron neutron capture therapy (BNCT) is a strong source of thermal (<1-eV) or epithermal (1-eV to 10-keV) neutrons. The currently available neutron sources with sufficient intensity are research and test reactors. Since the neutrons generated in nuclear reactor cores cover a wide spectrum of energies and the effectiveness of BNCT strongly depends on how well the fast (>10-keV) neutron dose to healthy tissue is minimized, the core neutron spectrum is an important consideration in the design of an irradiation facility for BNCT.

A large number of research and test reactors in the world use highly enriched uranium (HEU) fuels. In view of the restrictions on the export of HEU from the United States and also the 1986 U.S. Nuclear Regulatory Commission ruling, these reactors are in the process of being converted to use low-enriched uranium (LEU) fuels. The conversion process usually leads to core spectrum hardening (i.e., shift to higher neutron energies). The objective of this paper is to present the results

Fig. 1. Facility model for calculations.

Appendix 2

LOW ENERGY ACCELERATOR-BASED NEUTRON SOURCES FOR NEUTRON CAPTURE THERAPY

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INTRODUCTION

Neutron beams for neutron capture therapy (NCT) have been studied by a number of investigators,^{1,2,3} using high flux research reactors as the source of neutrons. There are very few research reactors in the world, probably less than ten, that can meet the conditions of both flux intensity and neutron energy spectrum. Hence the availability of NCT for patients with malignant brain tumors would be quite limited, even if all of these reactors would be able to accommodate the necessary beam design and hospital room atmosphere into the operational program.

For a number of years it has been suggested that serious design work should be undertaken to develop accelerator-based sources for the neutrons, using compact accelerator designs. The relatively new RFQ (radiofrequency-quadrupole) linear accelerators have the promise of achieving this objective.

This study investigates the optimization of the moderator and reflector system that would be used between the accelerator and the patient. The study is similar to that of Blue et al at Ohio State University⁴, and Dolan⁵ at INEL, but goes further in attempting to optimize the useful neutron flux for boron NCT with respect to the unwanted dose from neutrons and gamma rays. This study also investigates both the $\text{Li-7}(p,n)$ and $\text{H-3}(p,n)$ proton accelerator target reactions. Though these two targets are studied separately, and each would involve an entirely different target design, a combination target of tritiated lithium hydride deserves to receive attention in future designs.

The yields from the targets were calculated from the differential cross sections and stopping powers in the literature, assuming thick targets for the incident protons. The yields were divided into five angular groups, which were used as input to the MCNP Monte Carlo computer code. Neutron fluxes, neutron doses, and gamma doses were calculated at the patient position (skull) and at a tumor at various depths.

BASIC SYSTEM DESIGN AND COMPUTATIONAL DESIGN TOOLS

The basic system design that was eventually adopted, after a number of design iterations, is shown in Figure 1. The moderator adopted was 20 cm of BeO, for the Li reaction, and 34 cm of Al₂O₃ for the ³H reaction. The moderator and the 3 cm diameter accelerator vacuum (drift) tube were surrounded by the reflector of MgO for the Li-7 reaction, Al₂O₃ for the tritium reaction. These selections were made on the basis both of maximizing desired neutron flux and of minimizing other neutron and gamma doses (see following section). Total diameter of the outside of the reflector was 85 cm, and it extended 33 cm back along the accelerator drift tube.

The goal of this design was to produce neutrons, at the patient, predominantly in the energy range of 1 eV to 10 keV. Our studies, and those of other investigators, had shown that this range would optimize the treatment for tumors at a 3 to 7 cm depth from the surface of the skull. The reference tumor used for the calculations was 6 cm. It was assumed that the tumor contained 30 μg of boron-10 per gram of tissue, and that there was a 10:1 ratio between tumor and healthy tissue, i.e. the healthy tissue had 3 μg/g concentration of B-10.

The neutron source reactions studied were from the two reactions shown in Table 1. The proton energy on target was limited to a maximum value that would not produce neutrons above nominally 1 MeV, since higher energy neutrons are much more difficult to moderate down to the desired intermediate energy (epithermal) neutron energy range desired at the patient position. Since the threshold proton energy for the two reactions is not the same, the needed maximum proton energy differs between the two reactions, being only 1.77 MeV for the tritium target, but 2.5 MeV for the Li-7 target. Furthermore, the neutron yield for the hypothetical pure tritium target is nearly twice that of a pure Li-7 target. However, the target design to obtain the full utilization of the proton beam is not as obvious for the tritium target as it is for the lithium target.

Table 1. - The neutron-producing proton target reactions employed in the design, and the limiting conditions.

	$\text{H}^3(\text{p},\text{n})\text{He}^3$	$\text{Li}^7(\text{p},\text{n})\text{Be}^7$
Q-Value (MeV)	-0.764	-1.645
Threshold Energy (MeV) in lab	1.019	1.881
Neutron Energy (MeV) @ threshold	0.064	0.029
Incident Proton Energy (MeV)	1.77	2.50
Maximum Neutron Energy (MeV)	1.00	0.787
Total neutron yield, n/p	2.54×10^{-4}	1.48×10^{-4}

DOSE RESULTS

All dose calculations were done with reference to a thin tissue-equivalent cell at the patient irradiation position, used to determine the neutron and gamma fluxes that would be present at the surface of the skull. The intent was to optimize the desired intermediate neutron flux, but also to minimize the doses to the patient from the other neutrons and gamma rays.

The selection was done on a somewhat arbitrary basis, using plots similar to that of Figure 2, which shows the selection of reflector material for the lithium target reaction. The dotted lines indicate the minimum acceptable intermediate neutron flux, and the minimum ratio of that flux to total neutron dose to healthy tissue.

The gamma dose was an item of major concern, and this was minimized by adding a small amount of Li-6 neutron absorber to both moderator and reflector. The Li-6 absorbed the thermal neutrons, reducing their direct dose, and also reduced the radiative capture in the moderator and reflector materials, thus reducing the gamma dose. Figure 3 is a plot which resulted in the selection of 0.05 atom percent as the "optimum" amount of Li-6 poison to recommend.

Figure 1. Moderator-Reflector Assembly Design. Moderator is BeO. Reflector is MgO for Li-7 target and Al_2O_3 for the tritium target.

Figure 2. Selection of reflector material for Li-6 target.

Figure 3. Selecting the amount of Li-6 poison to use.

OVERALL RESULTS and DISCUSSION

Table 2 lists the overall results from the design, for the desired (useful) intermediate energy neutron flux, and the undesirable doses to healthy tissue. Results are quoted with respect to one milliamperere of proton beam on the target. Present RFQ machine technology has achieved proton beam currents of a few milliamps, with up to 20 milliamps believed to be achievable with current state-of-the art operation.⁶

These beam currents would give the desired treatment, 2000 cGy (≈ 5000 cSv or Rem) to a tumor having 30 μg of B-10/g of tissue, in approximately 45 minutes for the H-3 target (55 minutes for Li-6 target). This is a factor of 4 to 5 times longer than for beams designed for some of the higher powered research reactors, such as that at the University of Missouri. However, the 45 minute treatment period appears to be within the tolerable range for a patient treatment time.

TABLE 2 - Calculated Results, per mA of Proton Beam

	H-3 target	Li-7 target
AT PATIENT IRRADIATION POSITION IN AIR		
Total Neutron flux ϕ (n/cm ² sec)	1.29E+8	1.31E+8
Epithermal Neutron flux ϕ (1eV-10keV)	1.10E+8	0.94E+8
Neutron dose in cGy/epithermal n/cm ²	4.1E-11	4.3E-11
Gamma dose in cGy/epithermal n/cm ²	2.8E-11	2.7E-11
Thermal neutron ϕ (n/cm ² sec)	1.14E+8	1.00E+8
Total dose to tumor in cGy/minute	2.23	1.78
AT 6 cm DEPTH TUMOR WITH 30 μg B-10/g of TISSUE		
Fraction of dose from B-10(n, α)	65%	67%
Fraction of dose from Fast neutrons	4%	5%
Fraction of dose from N-14(n,p)	1%	1%
Fraction of dose from gammas	30%	27%
"Advantage Depth"*	9 cm	8 cm
*Depth at which B-10-caused dose is equal to the neutron and gamma doses to the tumor.		

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Appendix 3

RESULTS OF BENCHMARK EXPERIMENT OF ACCELERATOR/MODERATOR CONFIGURATION FOR BNCT - USING RFQ ACCELERATOR

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KEYWORDS: BNCT, ACCELERATOR, BENCHMARK

ABSTRACT

A number of design studies involving Boron Neutron Capture Therapy (BNCT) have been done using the MCNP code. There is always the question of how accurate are these studies, and do they conform to reality. A relatively simple benchmark experiment has been conducted using the 2.0 MeV RFQ accelerator at Idaho State University. Result of the measurements that can be considered reliable enough to be used as a benchmark are given, and a discussion of the need for additional reliable measurements is included.

INTRODUCTION

Following an extensive computer design effort using the MCNP code ¹, it was decided that the results that had been obtained needed to be verified by a benchmark experiment of similar dimensions and similar materials. Calculations alone may or may not represent reality, and since benchmarks of this type had never been performed, it was decided to use the VandeGraff generator at Idaho State University to perform a benchmark experiment of relatively simple geometry. The initial attempts in 1992 were relatively unsuccessful, because of a very unsteady beam, with significant arcing, and hence great uncertainty in correlating beam time and average energy with the measured neutron and gamma fluences. Part of the problem was attempting to run a 2.0 MeV VandeGraff machine at as high an energy as possible (some runs approached 2.3 MeV) so as to get closer to the optimum proton energy, nominally 2.5 MeV. At this energy the neutron yields are the highest, while the gamma contamination is relatively low. At 2.378 MeV proton energy, the product Be-7 nucleus has a threshold of producing its first excited state, from which a 430 MeV gamma ray is emitted. Also, the maximum neutron energy is 0.787 MeV, which allows for not too large of a spread of neutron energy from the proton target, and makes moderating the system to obtain maximum epithermal flux (1 eV to 30 keV) a bit more successful than from a reactor spectrum, for instance.

As plans to return and perform more experiments with the VandeGraff generator were being considered, it became apparent that even after a complete reconditioning of the unit, smooth, steady state

operation much over 2.0 MeV was going to be very uncertain. Consequently, a request was made to the INEL for permission to use the newly installed 2.0 MeV RFQ generator (Figure 1), which had been purchased by the INEL and placed at Idaho State University. It appeared that instead of an equivalent 4 microamps total proton current on target, the maximum possible from the VandeGraff, over ten times as much equivalent steady state yield could be obtained with the pulsed RFQ. The stability of the proton beam was indicated to be excellent, in both current and energy.

Figure 1: RFQ Positive Ion Accelerator (Los Alamos Design)

ACCELERATOR BEAM CHARACTERISTICS

The pulsed RFQ accelerator, (manufactured by AccSys Technologies Model PL-2) was operated in a direct linear beam mode (no magnetic deflection) through a long drift tube, onto a lithium target. The resulting reaction $\text{Li-7}(p,n)\text{Be-7}$ produced neutrons that spread forward into a heavy water moderator, and at other angles into an aluminum oxide reflector.

Neutron yield is a very strong function of proton energy (Figure 2, calculated from cross section data in Reference 2) near the 2.0 MeV range. Hence, conducting a benchmark based on proton current would leave a major concern about the actual proton energy. Furthermore, the structure of the machine with its long drift tube, made aiming of the beam and directing it onto the lithium target very difficult. Finally, the targets used were natural lithium, not lithium-7, and hence maximum neutron production could not be expected. At 2.0 MeV, the maximum neutron energy produced is 0.228 MeV in the laboratory system.

Figure 2: Neutron Yield as a Function of Proton Energy onto Li-7 Target

It was decided that the benchmark could not effectively include the proton beam production of neutrons, because of the various uncertainties. Fortunately, the Be-7 produced in the target reaction allows for neutron production to be measured after the fact. The Be-7 remaining has a half life of 53.29 days, and decays with a 478 keV gamma, with a branching ratio of 10.39%. (See Figure 3.) Each benchmark run was made with a new target, and the target material was collected after each run and counted in an absolute gamma spectrometer (intrinsic germanium crystal) to determine the activity at shutdown. Runs were typically one hour in length.

Figure 3: Be-7 Decay Scheme, used for determining absolute neutron production

Typical beam characteristics were as follows:

Pulsing Rate:	119 Hz
Pulse width	60 microseconds
Nominal pulse current*	10 mA

giving an effective "duty cycle" of 0.0070,
and an effective steady state current of $0.070 \text{ mA} = 70 \text{ microamps}$.

*However, the current was measured with a Faraday cup about 80 cm from the target. Beam alignment was a critical parameter (discussed in the section on RESULTS), and the Faraday cup measurement may be much higher than the current that reached the active area of the target.

This effective steady state current was about 18 times what could have been obtained with the VandeGraff generator.

DESCRIPTION OF EXPERIMENT

The geometry chosen was similar to that of the basic computer study conducted by one of the authors¹, essentially the same as that tried in the earlier benchmark experiment,⁴ and also similar to the design adopted in an earlier study by Blue and associates at Ohio State University.⁴ The benchmark configuration included several minor modifications, primarily in the thickness of the heavy water moderator used between the target and the phantom (patient). Thicknesses of both 10 and 12.5 cm of heavy water were used, consisting of cylinders of D₂O liquid, 14.5 cm in diameter, encased in a thin layer of plastic. The moderator, and parts of the beam drift tube, were surrounded by aluminum oxide reflector. The referenced work¹ had actually determined that beryllium oxide was a slightly superior moderator to heavy water for the accelerator-BNCT application using the Li-7(p,n)Be-7 reaction, and that magnesium oxide was a superior reflector. These decisions had been made on the basis of neutron yield, and of a high ratio of useful (1 eV to 30 keV) flux to both fast neutron flux and gamma flux at the patient location. Thermal neutrons were virtually eliminated by adding 0.5% ⁶Li to the reflector for those design studies. The Li-6 poison was not included in the reflector of the benchmark, however, because of cost considerations.

Figure 4 shows a cross sectional sketch of the moderator reflector arrangement. The aluminum oxide reflector consisted of trapezoidal blocks having a block density of 2.576 gm/cc. Because of gaps in the blocks, the effective overall density was somewhat less than the block density. A value of 2.5 gm/cc was used in the MCNP computer model of the experiment. Figure 5 shows two photographs of the arrangement. Though somewhat crude, as to the outer surface smoothness, the effective overall density in the regions near the beam tube, moderator, and phantom are believed to be effectively represented by $2.5 \pm 0.15 \text{ gm/cc}$ of aluminum oxide.

FRONT VIEW OF BNCT SETUP

SIDE VIEW OF BNCT SETUP

TOP VIEW OF BNCT SETUP

Figure 4: Moderator-Reflector Arrangement for the Benchmark Experiment

Figure 5: Photographs of the experimental benchmark set-up, with the RFO Accelerator

EXPERIMENTAL METHOD

Earlier experiments using the Van de Graaff generator had shown that standard thin gold and indium foils (gold thickness = 0.0025 cm, indium thickness = 0.0125 cm) receive adequate activation after an exposure of several hours. These indium foils were used, both bare and cadmium covered, for determining the neutron spectrum indicators as well as for measuring the neutron fluence from the response characteristics of the foils. For the RFQ runs, activation was quite adequate for both detectors, following only one hour of exposure at the above accelerator conditions.

For measuring the fast dose, neutron film badges were used, generally at outer edges of materials, so as to not perturb the other measurements. Nickel was inserted on all runs, in an attempt to measure the fast neutron flux, but no activation was found above background.

Gamma dose measurements were made using packages of three small TLD chips, for each location at which measurements were made. The response from these was also adequate (50 to 1000 mR range). Some of the film badges also reported gamma doses.

Two different targets were used. One was vacuum deposited natural lithium on a copper backing. The deposit of lithium was nominally 7000 Angstroms (0.7 microns) thick. In this quite thin layer, the proton only loses 6 keV of energy. Thick targets (approximately 2 mm thick) were used on several of the runs, and this thickness was sufficient that the protons were slowed down to below the 1.88 MeV threshold energy before leaving the lithium target material.

After each run, the target material was carefully removed, and later counted for beryllium activity. The thick target holder was built to allow for a very high water flow for target cooling, and the cooling lines came out the rear of the target. Separation of the target from the moderator by approximately 1.5 inches has a negative effect on all of the fluxes in the forward direction. However, this reduction is considered to be within the accuracy of the measurements quoted in the RESULTS section.

MCNP CALCULATIONS

The MCNP calculations were done with a recent (Dec. 1993) release of the code package for DOS-based personal computer applications, from RSIC (Oak Ridge National Lab)⁵. The system was modelled according to standard combinatorial geometry input, approximating as closely as possible the actual configuration. Smoothing obviously occurred on the outer dimensions of the reflector.

MCNP was run in full neutron-photon mode, so that capture gamma ray production was obtained. Consequently, photon neutron production in the heavy water was also included, and resulted in a very small fluence of high energy (well above the 0.228 MeV maximum energy from the target) neutrons being present.

The neutron source term for the MCNP calculations was derived from the cross section data for the $\text{Li-7}(n,p)\text{Be-7}$ reaction, divided into five angular increments, with an energy spectrum developed from fractional increments of the proton beam energy between the 1.88 MeV threshold energy and 2.0 MeV maximum energy. The total neutron yield (for a "thick" target) was $1.97\text{E-}5$ neutrons per proton on target, nearly an order of magnitude less than the $1.48\text{E-}5$ n/p yield ratio at the higher value of 2.5 MeV used in the Ref. 1 work, as the optimum proton energy.

RESULTS

Four runs were made, two with 10 cm thickness of heavy water moderator, and two with 12.5 cm thickness. Target yields was very dependent on the centering and the focusing of the beam onto the lithium of the target. There were considerable variations in beam alignment from run to run, as could be observed by careful observation of the target appearance at the end of the run. Comparing the best yield from the thick target to the best yield from the thin target resulted in a ratio of 5 between the two. For the highest yield on the thick target, the cooling behind it was inadequate to prevent serious disintegration of the target, even to the extent of depositing lithium on the walls of the drift tube.

Because of the inconsistency in attempting to obtain uniform target performance, all the result given below are referenced to a baseline total target yield of 10^8 neutrons produced by the target. Because the thick target had the further perturbation of the water lines behind the target, and the nominal 4 cm of separation between target and moderator, results are not quoted for those runs. The precise mockup of

these water lines has not been done in MCNP.

The results are all quoted in rather specific units of "Total foil activations per 10^8 source neutrons", and for the particular foil mass and thickness designated. The reason for such a complex indicator instead of the more conventional indicators of thermal or epithermal flux per unit target yield is the fact that the latter require very specific definition of the integral flux shape. Normal convention is to quote a Maxwell-Boltzmann thermal flux and an epithermal flux, between certain energy limits, based on the criteria of a constant flux per unit lethargy. Neither of these conditions will be true for the moderator arrangements in these experiments. However, if both calculations and experiments employ the same detectors, i.e. specific foils, size, thickness, and weight, than a true comparison can be made between the calculations and experiment.

The results given in Table 1 below are for the idealized configuration shown in Figure 4, i.e. the target holder essentially in contact with the moderator.

Table 1: BENCHMARK RESULTS and COMPARISON WITH MCNP

All results are in TOTAL DETECTOR FOIL ATOMS ACTIVATED per 10^8 neutrons produced from the target.

	10 cm Moderator Thickness *		12.5 cm Moderator thickness **	
	Experiment	MCNP calc.	Experiment	MCNP calc.
Gold Foil Results, 0.001 inch thick, 0.062 grams each, 0.5 inch in diameter				
Phantom Surface	N. A.	118	194	102
Phantom at depth	215	112	135	52
Indium Foil Results, 0.005 inch thick, 0.120 grams each, 0.5 inch in diameter				
Phantom Surface	6944	7146	5313	5736
Phantom at depth	7524	8053	4731	3846

The comparison between the measurements and calculations shows a tendency for the gold foil experimental results to be higher than the MCNP calculations, by the order of a factor of two. In the case of the indium foil results, the version of MCNP available ⁵ did not contain indium activation cross section. The MCNP results were generated via a set of cross sections which the authors developed from the published cross sections in the epithermal range, and by developing these into a number of discrete energy groups. The MCNP results were then reduced into the same group structure, for generation of the foil response results quoted.

The statistical uncertainty of the MCNP results was typically 1 to 2 percent. The experimental data uncertainties were of the order of $\pm 5\%$, except for the results at the 5 cm depth in the phantom, which have an uncertainty of nearer $\pm 10\%$

Figure 6: Neutron Fluence Spectra Histogram, Face of Phantom

Of some interest is the variation of epithermal and thermal fluence across the depth of the phantom. The distribution of these two fluence types are shown in Figure 6.

Figure 7: Fluence behavior in the phantom

Two special situations were encountered in the experiments, which deserve mention because of the significance of these on the ultimate results.

a) The effect of the cooling water (and the aluminum casing) behind the target was considerable. MCNP runs were made with cooling water and without cooling water behind the target. The results are shown in Table 2.

b) Reflector material density would be expected to have a an effect on retention of neutrons within the system. Table 3 shows the effect between calculations with the theoretical aluminum oxide density of 3.95 gm/cc and the actual effective density in the experiment, 2.5 gm/cc.

Table 2: Relative Effect of Cooling Water, 1.1 cm thick (and the Al wall), behind the target.

	No Water - reference case	Water case result
Target epithermal flux	1.0	0.256
Phantom-face epithermal flux	1.0	0.656
Thermal flux 3.8 cm into phantom	1.0	0.785
Results of "water" case are relative to 1.0 for "no water" case		

Table 3: Relative Effect of Aluminum Oxide Reflector Density

	3.95 gm/cc	2.50 gm/cc
Target epithermal flux	1.0	1.000
Epithermal flux at Phantom Face	1.0	0.919
Thermal flux at 3.8 cm depth in Phantom	1.0	0.832
Gamma Dose at Phantom Face	1.0	0.720

CONCLUSIONS AND RECOMMENDATIONS

As can be seen from Table 1, the MCNP results differ significantly in a few respects from the measured results. In general, the MCNP results have a statistical uncertainty in the 1 to 2% range. However, the experimental errors are much higher. Obviously, more measurements, and improved experimental results would be desirable. However, the MCNP results appear to be predicting thermal fluxes about correct at the phantom face, but MCNP appears to over-predict neutron fluxes at a depth

of 3.8 cm into the phantom. The calculation appears to significantly underpredict the thermal flux at the target, but this is not a critical aspect of the design. Still, further investigation is needed to determine why this significant discrepancy between calculation and experiment.

Gamma dose rates have not been quoted, because of the wide variation in results from the duplicate detectors. Similar problems beset the attempts to determine the fast neutron dose, using neutron sensitive film. Results were inconclusive because of the high thermal neutron and gamma background.

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