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A pilot study of an epithermal neutron source based on a low-energy proton accelerator for boron neutron capture therapy

Wang, Chang-Kwang Chris, Ph.D.
The Ohio State University, 1989

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A PILOT STUDY OF AN EPITHERMAL NEUTRON SOURCE 
BASED ON A LOW-ENERGY PROTON ACCELERATOR 
FOR BORON NEUTRON CAPTURE THERAPY 

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School of the Ohio State University 

By 

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To My Parents
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# TABLE OF CONTENTS

ACKNOWLEDGEMENTS ................................................................. iii
VITA ................................................................................................. v
TABLE OF CONTENTS ........................................................................ viii
LIST OF FIGURES ............................................................................. xi
LIST OF TABLES ................................................................................. xvi

## CHAPTER

<table>
<thead>
<tr>
<th>CHAPTER</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>I.</td>
<td>INTRODUCTION ................................................................. 1</td>
</tr>
<tr>
<td>1.1</td>
<td>Background ........................................................................... 1</td>
</tr>
<tr>
<td>1.2</td>
<td>Biological Justification .................................................. 4</td>
</tr>
<tr>
<td>1.3</td>
<td>New Developments ............................................................. 7</td>
</tr>
<tr>
<td>1.3.1</td>
<td>Boron Compound Development ....................................... 7</td>
</tr>
<tr>
<td>1.3.2</td>
<td>Neutron Source Development .......................................... 8</td>
</tr>
<tr>
<td>1.3.2.1</td>
<td>Epithermal Neutron Source for BNCT ......................... 8</td>
</tr>
<tr>
<td>1.3.2.2</td>
<td>Hospital-Based Neutron Source for BNCT ................. 11</td>
</tr>
<tr>
<td>1.5</td>
<td>A RFQ-Based Neutron Irradiation Facility for BNCT ......... 13</td>
</tr>
<tr>
<td>1.6</td>
<td>Objectives and Scope of Study ....................................... 16</td>
</tr>
<tr>
<td>II.</td>
<td>CALCULATIONAL METHODS .................................................. 19</td>
</tr>
<tr>
<td>2.1</td>
<td>Introduction ......................................................................... 19</td>
</tr>
<tr>
<td>2.2</td>
<td>Nature of the Problem ....................................................... 19</td>
</tr>
<tr>
<td>2.3</td>
<td>Neutron Source Calculation ............................................. 20</td>
</tr>
</tbody>
</table>
Appendix A  FORTRAN listing of the NSRCE ..................................................133
Appendix B  FORTRAN listing of the SOURCE subroutine for
MORSE calculation .............................................................................138
Appendix C  FORTRAN listing of the next-event point and track-
length estimators for MORSE calculation ........................................143
Appendix D. FORTRAN listing of the estimator subroutine which
calculates the neutron response functions for the
boron-shell spectrometer ................................................................147
Appendix E. FORTRAN listing of the spectrum unfolding code
SPUNIT used for the boron-shell neutron
spectrometer .....................................................................................152
# LIST OF FIGURES

<table>
<thead>
<tr>
<th>FIGURES</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>1-1. The total neutron fluence-to-kerma factors for the standard man and the most important individual contribution to the total</td>
<td>10</td>
</tr>
<tr>
<td>1-2. The general configuration of a RF Quadrupole</td>
<td>15</td>
</tr>
<tr>
<td>1-3. The schematic detail of a RFQ-based neutron irradiation facility to be used for BNCT in a hospital</td>
<td>17</td>
</tr>
<tr>
<td>2-1. The geometric description of the neutron transport problem</td>
<td>21</td>
</tr>
<tr>
<td>2-2. The neutron production cross section for $^7$Li(p,n)$^7$Be reaction versus the incident proton energy</td>
<td>23</td>
</tr>
<tr>
<td>2-3. Histogram of normalized angular differential neutron yield for neutrons emitted from a thick lithium-7 target bombarded by a 2.5 MeV Proton beam</td>
<td>26</td>
</tr>
<tr>
<td>2-4. Histogram of the normalized energy dependent yield for neutrons emitted from a thick lithium-7 target bombarded by a 2.5 MeV Proton beam</td>
<td>27</td>
</tr>
<tr>
<td>2-5. The flow diagram of a typical MORSE run</td>
<td>30</td>
</tr>
<tr>
<td>3-1. The illustration of the lithium target assembly which was constructed and tested in the experiment</td>
<td>41</td>
</tr>
<tr>
<td>3-2. The illustration of the neutron moderator assembly which was constructed and tested in the experiment</td>
<td>42</td>
</tr>
<tr>
<td>3-3. The illustration of the phantom which was used in the experiment for absorbed dose measurement</td>
<td>43</td>
</tr>
<tr>
<td>Section</td>
<td>Description</td>
</tr>
<tr>
<td>---------</td>
<td>-----------------------------------------------------------------------------</td>
</tr>
<tr>
<td>3-4.</td>
<td>The geometric configuration for the first experimental setup</td>
</tr>
<tr>
<td>3-5.</td>
<td>The geometric configuration for the second experimental setup</td>
</tr>
<tr>
<td>3-6.</td>
<td>The geometric configuration for the third experimental setup</td>
</tr>
<tr>
<td>3-7.</td>
<td>The approximate energy ranges covered by different neutron spectrometer types</td>
</tr>
<tr>
<td>3-8.</td>
<td>The internal structure of a spherical proton proportional counter</td>
</tr>
<tr>
<td>3-9.</td>
<td>The electronics for the proton recoil proportional counter measurement</td>
</tr>
<tr>
<td>3-10.</td>
<td>The pulse height distribution recorded by the proton recoil proportional counter in the OSU subcritical assembly</td>
</tr>
<tr>
<td>3-11.</td>
<td>The gas multiplication gain versus bias voltage for the proton recoil proportional counter</td>
</tr>
<tr>
<td>3-12.</td>
<td>The energy resolution versus bias voltage for the proton recoil proportional counter</td>
</tr>
<tr>
<td>3-13.</td>
<td>The pulse height distribution for Na-22 gamma-rays recorded by the proton recoil proportional counter</td>
</tr>
<tr>
<td>3-14.</td>
<td>The geometric configuration of the boron-shell neutron spectrometer</td>
</tr>
<tr>
<td>3-15.</td>
<td>The pulse height distribution of thermal neutrons recorded by the $^3$He proportional counter</td>
</tr>
</tbody>
</table>
3-16. The set of response functions of the $^3$He proportional counter for the seven combinations of boron shells and paraffin jacket shown in Table 4 ...................................................... 72

3-17 The unfolded results performed by the modified SPUNIT code for a single-bin neutron spectrum ..................................................... 75

3-18. The calibration curve of GM counts versus absorbed dose using a Cs-137 source .......................................................................... 77

4-1. The calculated and the measured spectra of neutrons emitted at $0^\circ$ angle, from the lithium target which was bombarded by 2.5 MeV protons ........................................................................ 80

4-2. The calculated and the measured spectra of neutrons emitted at $45^\circ$ angle, from the lithium target which was bombarded by 2.5 MeV protons ........................................................................ 81

4-3. The calculated and the measured spectra of neutrons emitted at $90^\circ$ angle, from the lithium target which was bombarded by 2.5 MeV protons ........................................................................ 82

4-4. The calculated and the measured spectra of neutrons emerging from the 20 cm thick D$_2$O moderator ........................................... 84

4-5. The calculated and the measured absorbed gamma-ray doses in the phantom which was irradiated by the D$_2$O moderated neutron field ................................................................. 85

4-6. The calculated and the measured absorbed boron doses in the phantom which was irradiated by the D$_2$O moderated neutron field ........................................................................ 86

5-1. The preliminary design configuration of the neutron moderator assembly .................................................................................. 93
5-2. The energy dependencies of the macroscopic neutron scattering cross section for various neutron moderator materials.................................................................95

5-3. The energy dependencies of the average increase of lethargy per neutron collision for various neutron moderator materials.................................................................96

5-4. The relationship of RUFTED versus neutron energy for a monoenergetic neutron beam with no gamma-ray contaminations.........................................................................................104

5-5. The relationship of RUFTED and \( \phi_u \) versus the moderator (BeO) thickness at the irradiation point, which is 5 cm from the moderator assembly on the beam centerline. 105

5-6. The relationship of RUFTED and \( \phi_u \) versus the moderator (D\(_2\)O) thickness at the irradiation point, which is 5 cm from the moderator assembly on the beam centerline. 106

5-7. The relationship of RUFTED and \( \phi_u \) versus the moderator volume ratio (D\(_2\)O/Al\(_2\)O\(_3\)) at the irradiation point, which is 5 cm from the moderator assembly on the beam centerline.................................................................108

5-8. The relationship of RUFTED and \( \phi_u \) versus the reflector (Al\(_2\)O\(_3\)) thickness at the irradiation point, which is 5 cm from the moderator assembly on the beam centerline. 110

5-9. The relationship of RUFTED and \( \phi_u \) versus the \(^6\)Li loading at the interface between the moderator and the reflector. reflector ..............................................................................................................111

5-10. The relationship of RUFTED and \( \phi_u \) versus the \(^6\)Li loading at the front end of the moderator assembly. 113

5-11. The configuration of the optimum neutron moderator assembly for the proposed low-energy proton accelerator neutron irradiation facility.................................................................115

xiv
6-1. The neutron spectrum at the irradiation point of the optimum moderator assembly. ......................................................... 117

6-2. The relationship for four energy groups of neutron flux versus depth within the phantom on the beam centerline. ... 119

6-3. The depth distributions in a phantom for various components of the absorbed dose.......................................................... 120

6-4. The depth distributions in a phantom for various components of the dose equivalent................................................ 121

6-5. The distribution of therapeutic gains in a spherical phantom on the beam centerline..................................................... 123
LIST OF TABLES

<table>
<thead>
<tr>
<th>TABLES</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>2-1. The angular and energy dependence of the doubly differential neutron yield per mCoul of proton, calculated by NSRCE</td>
<td>25</td>
</tr>
<tr>
<td>2-2. The neutron kerm factors for tissue and 10 μg of 10B/g-tissue extracted from BUGLE-80 cross section library</td>
<td>37</td>
</tr>
<tr>
<td>2-3. The gamma-ray kerm factors for tissue extracted from BUGLE-80 cross section library</td>
<td>38</td>
</tr>
<tr>
<td>3-1. The material compositions of the boron-shell neutron spectrometer</td>
<td>71</td>
</tr>
<tr>
<td>5-1. The macroscopic cross section $\Sigma(n,\gamma)$ for thermal neutrons for various moderating materials</td>
<td>97</td>
</tr>
<tr>
<td>5-2. The calculated neutron fluence spectra, neutrons/cm2 per source neutron, and absorbed dose, cGy per source neutron generated by a 2.5 MeV proton beam impinging upon a lithium-7 target imbedded in the optimum moderator assembly</td>
<td>101</td>
</tr>
</tbody>
</table>
CHAPTER I

INTRODUCTION

1.1 Background

Many advances have been made in developing neutron capture therapy (NCT) since it was first suggested by Locher\(^1\) as a means of treating malignant tumors; but because of its complexity, the obstacles to its ultimate clinical success are still many. However, its limited clinical success has already been demonstrated\(^2\). In principle in NCT, a capture agent is carried by a pharmaceutical to tumor cells, where it attaches to or is incorporated within the tumor cells. Then, the tumor is irradiated with thermal neutrons. The thermal neutrons are absorbed by the capture agent, and then release high-LET charged particles which kill the tumor cells. Boron neutron capture therapy (BNCT) is NCT with boron-10 as the capture agent. The capture of a neutron by a boron-10 nucleus leads to the following reactions

\[
\begin{align*}
\text{n} + ^{10}\text{B} & \rightarrow ^{7}\text{Li} + \alpha + 2.79 \text{ MeV} \\
& \quad \text{6\%} \\
& \quad \text{94\%} \\
& \text{\downarrow} \\
& ^{7}\text{Li} + \gamma (0.48 \text{ MeV}) \\
& \quad \text{\downarrow} \\
& ^{7}\text{Li} + \alpha + 2.31 \text{ MeV}
\end{align*}
\]

The \(^{10}\text{B}(n,\alpha)^{7}\text{Li}\) reaction is attractive for NCT, because both the \(\alpha\) particle and the \(^{7}\text{Li}\) ion have ranges which are comparable to the size
of a cell; and should therefore only damage the cells in which they deposit their energy, while leaving the surrounding tissue uninjured. Therefore, ideally, if there exists a $^{10}$B-containing compound which can deliver a large amount of $^{10}$B to the tumor cells, while keeping $^{10}$B content low in the surrounding normal tissue, then the tumor cells can be killed selectively.

In its early years (1950-65), BNCT research was led by Dr. William Sweet, then a neurosurgeon at Massachusetts General Hospital. Clinical trials were carried out between 1959 and 1961, for patients with cerebral gliomas. However, they ended with failure. The failure was attributed to: (1) the excessive $^{10}$B concentration in the circulating blood, which caused damage to cerebral vascular endothelial cells by neutron capture reactions; (2) the $^{10}$B-containing compound, paracarboxyphylboronic acid or perhydrodecaborate, used by Dr. Sweet has non-uniform distribution in the tumor and high uptake by the surrounding normal brain tissue; (3) the narrow thermal neutron beam used for irradiation has poor penetration in tissue, and therefore patients with deep-seated tumors were not effectively treated.

The renewed interest in BNCT in recent years is attributable to the persistent work performed by the Japanese neurosurgeon, Dr. H. Hatanaka, and his collaborators. In the past two decades, Dr. Hatanaka has treated with BNCT 98 patients suffering with brain gliomas. He has used the second generation boron compound, Sulfhydryl Boranes ($\text{Na}_2\text{B}_{12}\text{H}_{11}\text{SH}$, or abbreviated as BSH) and a thermal neutron beam. Because of the poor penetration of thermal neutrons, all the patients
had to go through a surgery to reflect the scalp, trepan the skull, and reveal the main tumor bed, before irradiation. With such a procedure, neither repeated surgery nor long-term maintenance of a patient is desirable or practical, and the patients were therefore all treated with a single session irradiation. In spite of all the difficulties in its implementation, the most recent survival calculation, for Dr. Hatanaka's BNCT procedure, indicates 19 patients with grade III-IV gliomas (some of them called glioblastoma) in the cerebral mantle (meaning within 6 cm of the cortical surface) showed 48% 5-year survival and 19% 16-year survival rate. This survival rate is strikingly good comparing to the five-year survival rate of less than 5% for even the best currently available systematic multimodality (meaning the combination of radiation and chemotherapy) treatment schedules. Hatanaka's work has stimulated researchers all over the world to reevaluate the BNCT. Since 1983 there have been three international Symposia and many workshops devoted to the subject.

So far, the clinical applications of BNCT have been focused on brain tumors. This is because the breakdown of the blood brain barrier (BBB) in brain tumor cells allows boron-containing compounds to enter them; whereas, because of the existence of the BBB, the compound cannot enter the normal cells. This biological selectivity makes brain tumors amenable to BNCT. Theoretically, however, BNCT should not be limited to treating brain tumors only, because in general, all malignant cells are actively synthesizing, dividing, and differentiating, and thus are constantly nutrient-hungry and uptake more chemical substances than the surrounding normal cells. This is
the basis for the biological selectivity for the drugs used in a chemotherapy, and this may also serve as the basis for the selectivity of future boron-containing drugs, which may be used to treat other types of malignant tumors with BNCT. As a matter of fact, Dr. Mishima, a dermatologist in Japan, has recently started using BNCT with a compound called p-boronophenylalanine (BPA) to treat patients with superficial melanoma. It may be still too early to draw any conclusions from Mishima's clinical results, but so far the results have been encouraging.

1.2 Biological Justification

Although there has been limited clinical success with BNCT, many fundamental mechanisms are still not understood, and many obstacles are still left to be overcome. For example, animal studies using ethylnitrosourea-induced glioma, suggest that for the B-10 compound, BSH, which is now in clinical use for BNCT of brain tumors, boron atoms are not uniformly distributed among tumor cells. The vasculature and the degree of capillary permeability vary widely, depending mainly on the size of the glioma, and are often heterogeneous even in the same tumor. The distribution of B-10 also varies, correlating to capillary permeability. The B-10 concentration and the tumor/blood concentration ratio in large and medium sized gliomas are often adequate to BNCT; but the small gliomas, or the peripherals of large gliomas, often have much lower B-10 concentrations, and therefore have a much better chance to survive after irradiation. In addition, for a B-10 concentration of 30 µg/g in
tumor, the recommended thermal neutron fluence\textsuperscript{5} to a brain tumor in a single-session irradiation is \(5 \times 10^{12}\) n/cm\(^2\). The corresponding absorbed high-LET dose due to \(^{10}\)B(n,\(\alpha\))\(^7\)Li reactions is about 1300 cGy. For a typical number of \(10^9\) cells/g tissue, this dose translates to an average number of 35 \(^{10}\)B(n,\(\alpha\))\(^7\)Li reactions per cell. This many \(^{10}\)B(n,\(\alpha\))\(^7\)Li reactions in one cell certainly overkill most of the tumor cells, but are required to overcome Poisson statistics. The Poisson distribution predicts that, even if there is a new compound which can uniformly deliver enough boron-10 atoms to all tumor cells, there are always tumor cells, in which not enough \(^{10}\)B(n,\(\alpha\))\(^7\)Li reactions occur; and therefore these cells survive after irradiation. If this is the case, then one might wonder how BNCT can work at all? These are some of the questions for which the author has been seeking answers. Of course, the bottom line of these questions are: Can we really explain Hatanaka’s clinical results based on the most fundamental mechanisms? Does BNCT really work? If so, then how can we improve BNCT modality therapeutically, and how can we make it more available to medical communities? The followings are a few author’s thoughts toward these questions.

Indeed, BNCT may not be able to kill every single cancer cell in a treatment, while leave the surrounding normal cells intact. However, for many malignant tumors, there exist a critical mass. The tumor growth rate can be controlled by the body immune surveillance system if the tumor is smaller than the critical mass. For glioblastoma the critical mass is suggested to be \(10^5\) cells\textsuperscript{6}. The tumor mass after surgery usually contains about \(10^9\) cells (which is about 1 gram), and
therefore it is necessary to achieve 4 log kills (that means a survival fraction of $10^{-4}$) in a treatment following the surgery, in order to bring the tumor under control. The conventional radiation therapy or the chemotherapy alone can only achieve 1 to 2 log kills. The recent trials using systematic multi-modality, have achieved 2 to 3 log kills$^6$, which is still not good enough for the glioblastoma multiforme. The better results achieved by the multi-modality is attributable to the independence between the radiation dose (which is physically selective) and the chemical drug dose (which is biologically selective). Radiation dose is referred as physically selective because it is concentrated on the tumor volume. Chemical drug dose is referred as biologically selective because more drug is incorporated into tumor cells than that is into normal cells. The author perceives that there are similarities between BNCT and the multi-modality just described, in that boron drug (and thus the high-LET dose imparted by $^{10}$B(n,α)$^7$Li reaction) is biologically selective, and the gamma-ray and proton doses induced by $^1$H(n,γ)$^2$H and $^{14}$N(n,p)$^{14}$C reactions are physically selective. However, BNCT should be superior to the multi-modality for two reasons. First, the toxicity of the boron drugs is much less than that of the chemical drugs used in chemotherapy. Secondly, the proton dose from $^{14}$N(n,p)$^{14}$C reaction is high-LET, and therefore, may be beneficial in treating hypoxic tumor cells which are refractory to the low-LET gamma rays. By perceiving these possible advantages of BNCT, the author proposes that, indeed, BNCT does not kill every single tumor cell in a treatment, but it may have better chance to achieve 4 log kills than any other therapy modalities existing today.
be more specific, that is, indeed there may be many tumor cells (especially of those in peripheral regions of a tumor) in which not enough number of $^{10}$B(n,α)$^7$Li reaction occur during irradiation, but large portion of them may be killed by the induced gamma rays and protons, so that a total of 4 log kills may be achieved in a BNCT treatment. As a matter of fact, this may be the actual mechanisms which resulted in Hatanaka's successful clinical cases; because in most of his cases the non-boron dose (which is independent on the boron distribution) to the tumor bed was estimated to be about 9 Grays, which is more than 40%$^7$ of the total absorbed dose of 20 Grays received by well-boronated tumor cells, and therefore may well be responsible for 1 to 2 log kills.

1.3 New Developments

The future success of BNCT as a method of treating malignant brain tumors mainly depends on two developments: (1) new $^{10}$B-containing drugs, and (2) a new generation of neutron source.

1.3.1 Boron Compound Development

The new of $^{10}$B-containing drugs should not only have good affinity to tumor cells, but is also distributed uniformly among tumor cells. As of this writing, several new $^{10}$B-containing drugs have been developed and some of them are being tested with animals. Promising compounds which are being tested in tumor-bearing animals include $\text{Na}_4\text{B}_{24}\text{H}_{22}\text{S}_2$ (BSSB), $\text{Na}_4\text{B}_{24}\text{H}_{22}\text{S}_2$ (BSOSB), and $\text{C}_9\text{H}_{12}\text{NO}_4\text{B}$ (BPA). Other new compounds are: antibodies, porphyrins, nucleosides, low-density lipoprotein, amino acids, and boronated liposomes, and the
melanin-affinic agents chlorpromazine (CPZ), and thiouracil (TU); Among them, porphyrins already show good specificity and high concentration in tumor. A detailed review of boron compound development can be found in references 8 and 9.

1.3.2 Neutron Source Development

1.3.2.1 Epithermal Neutron Source for BNCT

Based on Hatanaka's experience, the whole brain should be treated in the neutron irradiation. Therefore, a broad epithermal neutron beam, or a large epithermal neutron field have advantages over a narrow thermal neutron beam.

Regarding neutron energy, the use of epithermal neutrons for BNCT not only has the advantage over thermal neutrons of greater penetration in tissue (and therefore greater usefulness in treating deep-seated tumors), but also has the possibility of dose fractionation. Dose fractionation is possible because the open-skull surgery before irradiation is not needed, if an epithermal neutron beam is used. Although it still needs more clinical study, dose fractionation is thought to be beneficial for the following reason. The majority of the normal tissue dose is due to gamma rays, which are low-LET radiation, whereas the boronated tumor tissue is exposed to the high-LET $^{10}$B capture products. Since cell damage by low-LET radiation is partially repaired at relative low doses, dose fractionation allows normal tissue to undergo repair during the period between two irradiation sessions.

Studies have been conducted to determine the neutron energy range which is most suitable for BNCT. The most suitable energy range
for treating deep-seated tumors is generally believed to be between 1 eV and 1 keV for two reasons. First, neutrons in this energy range have the lowest kerma factor (kerma per unit fluence), and therefore impose a low skin dose per unit fluence to a patient. The neutron kerma factors for a standard man, and the most important contributions to the kerma factors are shown graphically in Figure 1-10. Secondly, it is believed that, protons recoiling in collisions with neutrons having energy less than 1 keV, have energies too low to cause ionization in tissue. Thermal neutrons are damaging to biological tissues not by virtue of the recoil protons, but by the 615 keV protons released from $^{14}$N(n,p)$^{14}$C reactions. Therefore, neutrons with energies less than 1 keV but above thermal (~1.0 eV), may have near-zero RBE values.

The biological effectiveness of keV neutrons is currently being studied by a group at Harwell laboratory11. The Harwell group has measured depth-survival profiles for HeLa cells, which were irradiated by 24 keV neutrons. Their results show that, except at very high concentrations of $^{10}$B (which may not be realizable in vivo), cell killing at near-surface depths for unboronated tissue is greater than the cell killing for boronated cells at the depths which would be important clinically. Their preliminary results suggest that 10 keV neutrons provide a significant surface-sparing effect in comparison to 24 keV neutrons, and should be more suitable for BNCT. Since epithermal neutrons with energies between 1 keV and 100 keV have similar penetration abilities in tissue12, the Harwell group concluded that 2 keV than 10 keV neutrons. Harwell's study certainly has demonstrated
Figure 1-1. The total neutron kerma factor for a standard man and the most important contributions to the total.
that the RBE value of neutrons, with energies in the range of few keV to a few tens of keV, varies drastically. Also, Harwell's study has more or less confirmed that the upper limit of the neutron energy range which is suitable for BNCT is less than a few keV.

Nuclear reactors with thermal powers from 100 kilo-watts to several mega-watts have been used to produce thermal neutron beams for BNCT studies and clinical trials\textsuperscript{13}. As of this writing, neutron filter and moderator assemblies have been designed, constructed, and tested at various reactor facilities\textsuperscript{14,15,16}. Filter and moderator assemblies are used to extract epithermal neutrons from a reactor. Reactors may serve well for BNCT studies. However, the fact that most nuclear reactors are located distant from major hospitals, makes BNCT very difficult to implement widely. If BNCT is to be widely used to treat malignant tumors, then a hospital-based neutron source must be developed.

1.3.2.2 A Hospital-based Neutron Source for BNCT

At the First and Second International Symposia on Neutron Capture Therapy, a number of hospital-based neutron sources for BNCT were considered. They are the californium-252 isotope neutron source, the spallation neutron source, and the low-energy proton accelerator neutron source. Each of them is discussed separately below.

Californium-252 is a spontaneous fission neutron source, and has been used as an implant in fast neutron therapy. However, because of its relative low neutron yield, hard spectrum, and short half-life (2.6
years), a large quantity (> 1 g) of the isotope, and a massive neutron moderator are required for BNCT. The high cost of the large quantity of the isotope, which must be supplied constantly, makes this option impractical.

A spallation neutron source is currently being investigated at the Swiss Institute of Nuclear Research (SIN), using a 72 MeV proton beam impinging upon a copper target. This spallation neutron source has a high neutron yield of $10^{14} - 10^{15}$ n/s; but it also has two disadvantages. First, the proton energy required for a spallation source is between 60 and several hundred MeV. A proton beam of this energy can only be produced by large cyclotrons, synchrocyclotrons, or large linear proton accelerator. All of these accelerators are rather large and expensive. Secondly, the spectrum of spallation neutrons emitted from the target is too hard for BNCT, and therefore a massive neutron moderator and shielding are necessary to make this source suitable for BNCT. These two disadvantages make a spallation source not amenable to a hospital.

The low-energy proton accelerator neutron source is based on a 2.5 MeV proton beam impinging upon a lithium-7 target and interacting with the target via the $^7\text{Li}(p,n)^7\text{Be}$ reaction. The $^7\text{Li}(p,n)^7\text{Be}$ reaction has been used for a long time in laboratories, as a way of generating monoenergetic neutrons for cross section measurement. The advantages of such a neutron source are: (1) Neutrons which are emitted from the lithium target have energies between 3 keV and 800 keV and an average energy of about 400 keV. This energy spectrum is relatively soft compared to those of the
alternatives, and therefore less neutron moderator is needed. Less moderator reduces the distance between the target and the patient, and thus the neutrons can be used more economically. (2) The \( ^7 \text{Li}(p,n)^7 \text{Be} \) reaction is basically gamma-ray free, and therefore does not require gamma-ray shielding. This makes the neutron moderator and shielding much less massive than for the alternatives. (3) Because of the recent development of the high-current radio frequency quadrupole (RFQ) accelerator, it is possible to build a neutron irradiation facility which is compact and inexpensive, and therefore is amenable to a hospital.

Based on the above discussions, the author concluded that among the three proposed hospital-based neutron sources for BNCT, a neutron source based on a low-energy proton accelerator, may hold the greatest promise and therefore should be further developed.

1.4 A RFQ-Based Neutron Irradiation Facility for BNCT

In the past, low-velocity ion acceleration presented one of the main technological difficulties for high-current accelerators, because space charge forces are strong for low-velocity ions, and conventional magnetic quadrupole lenses are ineffective for focusing low-energy ion beams. In contrast to focusing with magnetic quadrupole lenses, RFQ relies on strong electrostatic focusing in a narrow channel; this allows proton beam currents in the range of 10 to 100 mA. In addition, the RFQ combines the functions of acceleration and bunching. This is accomplished by varying the geometry of electrodes, so that the relative magnitudes of transverse and longitudinal electric fields vary
through the machine. Figure 1-2 is a general configuration of a RFQ. A high-current proton RFQ is not commercially available, but the technology required to build one exists. Recently, a group at Los Alamos National Laboratory (LANL) has build a RFQ which successfully accelerated a 2.5 MeV proton beam with currents as high as 25 mA\textsuperscript{19}. This high-current RFQ was originally built for the fusion material irradiation testing (FMIT) project at LANL, with the aim of delivering 100 mA of D\textsuperscript{+} current.

A few years ago, Lone et al.\textsuperscript{20} proposed to bombard a lithium target with a 10 mA 2.5 MeV proton beam delivered by a RFQ, as a way to produce clean thermal neutrons for neutron radiography. Lone's calculation shows that the total neutron yield of a thick lithium target is $8 \times 10^{12}$ n/s for a 10 mA, 2.5 MeV proton beam; and with the lithium target surrounded by water, a peak thermal neutron flux of $2.15 \times 10^{11}$ n/cm\textsuperscript{2} -s can be achieved at the target. Both the neutron yield and the thermal neutron flux appears to be much lower than those of a reactor or a spallation neutron source. However, because the $^7$Li(p,n)$^7$Be reaction is gamma-ray free and because its neutron spectrum is relatively soft, it may be possible to develop an epithermal neutron field which is useful for BNCT. The key issue here is how to design a compact neutron moderator assembly which will produce a satisfactory neutron field for BNCT.

Very recently, at the author's request, the LANL group has performed a preliminary design of a 30 mA, 2.5 MeV proton RFQ. This RFQ is 3 meters long and 15 cm in diameter\textsuperscript{21}. Such a facility can be easily deployed in a hospital. The total cost was estimated to be
Figure 1-2. The general configuration of a RFQ
$3.7 \times 10^6$, which is rather inexpensive in comparison to its alternatives. At this point, the author does not rule out other possible high-current, low-energy proton accelerators, such as the dynamitron and the tandem cascade accelerator (TCA) which was recently proposed. However, the maximum proton current which can be delivered by these alternatives, at present, is only a few mA, and this current is a little too low for BNCT.

Figure 1-3 is the schematic detail of a RFQ-based neutron irradiation facility proposed by Blue et al. The facility includes a RFQ powered by a Klystron through a wave guide, a proton source, bending and focusing magnets, a vacuum pump, a lithium target and its associated cooling system, and a neutron moderator assembly. The RFQ technology is basically available, but the following challenges still remain: (1) to remove 75 kW of heat (deposited by the 30 mA 2.5 MeV proton beam) from the lithium target, and (2) to design a compact moderator assembly which produces a good neutron field for BNCT.

1.5 Objectives and Scope of this Study
The objectives of this thesis are to perform a thorough neutronic study for the proposed low-energy proton accelerator neutron irradiation facility (LPANIF), and to design a neutron moderator assembly which optimizes the neutron field for BNCT, and to evaluate its performance. This thesis does not address the heat removal problem of the lithium target.

The neutronic study was performed in three stages: (1) The neutron fields generated by 2.5 MeV protons bombarding a thick lithium
Figure 1-3. The schematic detail of RFQ-based neutron irradiation facility to be used for BNCT in a hospital
target, were first studied, both by calculations and by experiment. An effort was then made to validate the calculational methods using the experimental results. (2) Once the calculational methods were validated, a design process based solely on calculations was then undertaken to obtain the optimum configuration of the moderator assembly. The moderator assembly was optimized to obtain a neutron field with a good energy spectrum and a high flux. (3) Finally, the optimized neutron field obtained in step (2), was evaluated for its performance in a BNCT treatment.

Regarding the organization of the thesis, Chapter II describes the calculational methods. Chapter III describes the experimental setups, and the measurement techniques. Chapter IV presents both the calculational and the measured results for the experimental neutron fields, and the calculational methods are validated based on the consistencies between the two results. Chapter V describes the criteria and methodology which were used to design the optimum neutron moderator assembly. The geometric configuration of the optimum moderator assembly is also described in Chapter V. Chapter VI evaluates the performance of the optimized neutron field for BNCT. Conclusions and remarks drawn from this study are presented in Chapter VII.
CHAPTER II

CALCULATIONAL METHODS

2.1 Introduction

This chapter discusses the calculational methods which were used to study the neutron field generated by the proposed LPANIF, and briefly presents some calculational results. The calculational results are presented in more detail in Chapter IV, where they are compared with the measured results from experiment; and in Chapters V and VI, where they are used to design the optimum neutron moderator assembly, and to evaluate its performance in a BNCT treatment.

2.2 Nature of the Problem

As indicated in section 1.3.2.1, the most suitable energy range for neutrons in deep-tumor BNCT is between 1 eV and 1 keV. The average energy of neutrons from the $^7\text{Li}(p,n)^7\text{Be}$ reaction is about 400 keV, which is too energetic for BNCT; and therefore these source neutrons must be moderated. A neutron moderator assembly serves this purpose. Thus, the problem being investigated is a "fixed-source", "deep-penetration" neutron transport problem. "Fixed source" means that the source particles are in a sense fixed, and independent of the
geometry or material compositions of the problem. "Deep penetration" means that the source particles have to go through many mean free paths of a scatterer media in order to reach the point of interest.

The geometric description of the problem is illustrated in Figure 2-1. Neutrons are initially born at the lithium target, collide, and slow down in the moderator assembly; and finally emerge from the moderator assembly. The quantities which are most important, and therefore must be calculated are: (1) the neutron flux spectrum at the irradiation point, and (2) the neutron and gamma-ray absorbed doses in a phantom, which is placed at the irradiation point. The neutron flux spectrum is the most fundamental quantity that defines a neutron field. The absorbed doses in a phantom indirectly verify the incident neutron flux.

The calculational methods for this fixed-source neutron transport problem include two parts: (1) calculation of the yield, energy, and angular distributions of the neutrons generated in the lithium-7 target, which is bombarded by 2.5 MeV protons, and (2) neutron and gamma-ray transport calculations. The details of each part of the calculations are described separately in the following subsections.

2.3 Neutron Source Calculation

The yield, energy, and angular distributions of neutrons generated in the lithium target were calculated by simulating a 2.5 MeV proton slowing down in the lithium-7 target, using the doubly differential cross section for the $^7\text{Li}(p,n)^7\text{Be}$ reaction, and the stopping
Figure 2-1 The geometric description of the neutron transport problem. This configuration is the basic model used in the calculational study.
power for protons in lithium. Both the neutron production cross section and the stopping power are well known\textsuperscript{24,25}. Figure 2-2 illustrates the neutron production cross section versus the incident proton energy. As it is shown, the neutron production cross section has a threshold energy of 1.88 MeV, and peaks at a proton energy of 2.3 MeV. The average energy of the neutrons produced by the \textsuperscript{7}Li(p,n)\textsuperscript{7}Be reaction increases as the incident proton energy increases. Choosing 2.5 MeV as the energy for the incident protons is obvious, since protons with energies higher than 2.5 MeV do not increase by much the neutron yield, but do generate a harder neutron spectrum. In addition, a proton energy of 2.5 MeV is about the upper limit which a RFQ can achieve.

A computer program called NSRCE was written to perform the calculation of the yield, energy, and angular distribution of the source neutrons. The FORTRAN listing of NSRCE is provided in Appendix A. In NSRCE, the neutron source spectrum was calculated by reducing the energy of a 2.5 MeV proton by small increments of energy, $\Delta E$, and calculating on the basis of the proton stopping power, $dE/dx$, the distance, $\Delta S$, which is required to reduce the proton energy by $\Delta E$. For the calculated $\Delta S$, the probability $p(E,E_n,\theta)dE_n d\theta$ has been calculated based on the given neutron production cross section, where $p(E,E_n,\theta)dE_n d\theta$ is the probability per unit path length that a proton with energy $E$ will produce a neutron with energy in $dE_n$ about $E_n$ and direction in $d\theta$ about $\theta$. Note, the factor $2\pi \sin \theta$, arising from the relationship between the differential solid angle and $d\theta$, is incorporated
Figure 2-2. The neutron production cross section for $^7\text{Li}(p,n)^7\text{Be}$ reaction versus the incident proton energy
into the expression for $p(E,E_n,\theta)$. The probability $p(E,E_n,\theta)dE_nd\theta d\Delta S$ was summed for all $\Delta S$ with corresponding proton energies above the 1.88 MeV $^7\text{Li}(p,n)^7\text{Be}$ reaction threshold to produce $p(E_n,\theta)$, where $p(E_n,\theta)dE_n d\theta$ is the probability that a 2.5 MeV proton will produce, as it stops in a thick lithium-7 target, a neutron with energy in $dE_n$ about $E_n$ and with direction in $d\theta$ about $\theta$. That is,

$$p(E_n,\theta)dE_n d\theta = \int_{1.88\text{MeV}}^{2.5\text{MeV}} p(E,E_n,\theta)dE_n d\theta dE / (dE / dx) \quad \text{Eq. 2.1}$$

The angular and energy dependence of the doubly differential neutron yield per mCoul of proton charge on target, $p(E_n,\theta)$, as calculated by NSRCE, is summarized in Table 2-1, and is illustrated by Figures 2-3 and 2-4, respectively. Figure 2-3 is a histogram of the normalized angular differential neutron yield, $p(\theta) = \int \int p(E_n,\theta)dE_n d\theta$, integrated over $20^\circ$ increments of $\theta$. That is, the first element of the histogram in Figure 2-3, $F(0,20)$, represents the fraction of the neutrons which are emitted with $\theta$ between $0^\circ$ and $20^\circ$; i.e.

$$F(0, 20) = \int_0^{20^\circ} p(\theta)d\theta, \quad \text{Eq. 2.2}$$

and the elements of the histogram sum to unity. Figure 2-3 shows that 35% of the neutrons are emitted with $\theta$ less than $45^\circ$, that 45% of the neutrons are emitted with $\theta$ between $45^\circ$ and $90^\circ$, and that 20% of the neutrons are emitted into the backward hemisphere. Figure 2-4 is a histogram of the normalized energy dependent differential neutron yield, $p(E_n) = \int \int p(E_n,\theta)dE_n d\theta$, integrated over 100 keV increments of $E_n$, for each of the three groups of neutrons described.
Table 1  Neutron source strengths for a 2.5 MeV proton beam impinging upon a lithium-7 target
(Neutrons/mCoul)

<table>
<thead>
<tr>
<th>Energy (keV)</th>
<th>Neutron Emitting Angles,</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>0-15°</td>
</tr>
<tr>
<td>820</td>
<td>2.44E+10</td>
</tr>
<tr>
<td>742</td>
<td>1.16E+11</td>
</tr>
<tr>
<td>608</td>
<td>1.22E+11</td>
</tr>
<tr>
<td>497</td>
<td>7.13E+10</td>
</tr>
<tr>
<td>368</td>
<td>1.47E+10</td>
</tr>
<tr>
<td>297</td>
<td>2.65E+10</td>
</tr>
<tr>
<td>183</td>
<td>3.72E+10</td>
</tr>
<tr>
<td>111</td>
<td>5.85E+10</td>
</tr>
<tr>
<td>67</td>
<td>6.99E+09</td>
</tr>
<tr>
<td>41</td>
<td>3.26E+10</td>
</tr>
<tr>
<td>32</td>
<td>9.88E+09</td>
</tr>
<tr>
<td>26</td>
<td>1.08E+10</td>
</tr>
<tr>
<td>24</td>
<td>1.42E+09</td>
</tr>
<tr>
<td>22</td>
<td>1.66E+10</td>
</tr>
<tr>
<td>15</td>
<td>2.44E+09</td>
</tr>
<tr>
<td>7</td>
<td>8.33E+08</td>
</tr>
</tbody>
</table>

Note: The blank area are zeros
Figure 2-3. Histogram of the normalized angular differential neutron yield, \( p(\theta) \), integrated over 20° increments of \( \theta \) for neutrons emitted from a thick lithium-7 target bombarded by 2.5 MeV proton beam.
Figure 2-4. Histogram of the normalized energy dependent neutron yield, $p(E_n)$, integrated over 100 keV increments of $E_n$ emitted from a thick lithium-7 target bombarded by 2.5 MeV proton beam.
in Figure 2-3, i.e. for $0^\circ < \theta < 45^\circ$, $45^\circ < \theta < 90^\circ$, and $90^\circ < \theta < 180^\circ$.

That is, the first element of the histogram for group 1 in Fig. 2-4, $F_1(0,100)$,

$$F_1(0, 100) = \frac{\int_{0}^{100\text{keV}} \int_{0}^{90^\circ} p(E_n, \theta) d\theta dE_n}{\int_{100\text{keV}}^{600\text{keV}} \int_{0}^{90^\circ} p(E_n, \theta) d\theta dE_n}$$

Eq. 2.3

and the elements of the histogram sum to unity for each group. Figure 2-4 shows that the most forwardly directed group of neutrons have energies between 100 and 800 keV, the $45^\circ$ to $90^\circ$ group have energies between 20 and 650 keV, and the backwardly directed neutrons have energies between 3 and 450 keV. The total neutron yield per proton, $Y$, $Y=\int\int p(E_n, \theta) dE_n d\theta$, is $1.474 \times 10^{-4}$ neutrons per proton. The total neutron yield calculated by NSRCE has been compared with Lone's result $^{20}$, and with the results obtained from hand calculations, and all three results agree within 15%. For a 30 mA proton beam, the total neutron yield is $2.76 \times 10^{13}$ neutrons per second.

2.4 Neutron and Gamma-Ray Transport Calculations

The standard techniques used to calculate neutron fluxes in a fixed-source, deep-penetration problem are: (1) the discrete ordinate method (also called the $S_N$ method), which solves the neutron transport equation (discretized in space, energy, and direction of motion) deterministically; and (2) the Monte Carlo method which solves the neutron transport equation by tracking each particle in a computer-simulated random walk process.
In general, the discrete ordinate method is appropriate for problems with simple geometries such as slabs, spheres, or cylinders, when the flux map of the entire transport medium is of interest. On the other hand, the Monte Carlo method is appropriate for problems with complex three-dimensional geometry, when the flux at only a few specific locations is of interest. In this calculational study, the main goals are: (1) to obtain the neutron and gamma-ray fluxes at the irradiation point (see Fig. 2-1 for an illustration), and (2) to obtain the absorbed doses at various depths in phantom (see Fig. 2-1 for an illustration). Because the problem geometry is rather complex, and because only the fluxes at a few dose points are of interest, Monte Carlo methods are the appropriate choice for this calculation. The Monte Carlo code MORSE-CG\textsuperscript{26} was used in conjunction with the cross section library BUGLE-80\textsuperscript{27} to perform the calculation. Both the code and the cross section library were obtained from the Radiation Shielding Information Center (RSIC) of the Oak Ridge National Laboratory (ORNL); and they are briefly reviewed below.

MORSE is a three-dimensional multi-group neutron and gamma-ray coupled Monte Carlo transport code, which was first released in 1970, and is now the most widely-used Monte Carlo code for shielding calculations in the nuclear industry. The flow diagram of a typical MORSE run is illustrated in Figure 2-5. As it is shown, a given problem is performed by following a number of batches of particles, which then constitute a run. The batch process is used so that statistical variations between groups of particles can be determined. At the beginning of
Figure 2-5. The flow diagram of a typical MORSE run
each batch, a batch of source particles (which consists of the initial position, direction, energy, and statistical weight associated with each particle) is generated and stored in the bank. The random walk for each batch consists of picking one source particle out of the bank, transporting it from collision to collision, generating secondary particles, and storing them in the bank for future processing. The parameters in a random walk process such as the path length between collisions, direction, and energy after each collision, are all determined from a set of probability density functions, which are generated from the multi-group cross section library. A particle is killed when (1) its weight is too low, and therefore it is not worth the calculational effort to track it; or (2) its energy is out of the range of interest; or (3) it escapes from the system. When the source particle is killed, the secondary particles produced by that source particle are tracked. When all secondaries (and possibly tertiaries, etc.) are killed, the second source particle is selected and tracked. This process is repeated, until all source particles for a given batch are killed. Then the source particles are generated for the next batch, and the process is repeated. When all batches are completed, the results and the statistics are summarized, and the run is finished.

BUGLE-80 is a coupled, 47 neutron group, 20 gamma-ray group, \( P_3 \), cross section library. \( P_3 \), implies that the angular distribution of the neutron scattering cross section is represented by Legendre polynomials of order 1,2 and 3. BUGLE-80 was chosen for its fine group structure in the energy range of interest. There are 25 groups
for neutron energies between 1 eV and 800 keV. The fine structure of the energy groups is essential to the accuracy of the transport calculation.

As it is indicated in Fig. 2-5, for a fixed-source MORSE calculation, there are two subroutines which are problem specific, and need to be provided by the user. They are the "source generator" subroutine and the "detector" subroutine.

The source-generator subroutine assigns position, energy, and direction to each particle at the beginning of its life. In this study, each source particle (neutron) is initially assigned with a specific energy and direction based on the probability density functions created from Table 2-1. The FORTRAN listing of the source-generator subroutine is provided in Appendix B.

The detector subroutine monitors a particle's position during its random walk process, and makes contributions to the fluence or dose at the detector, accordingly. The detector subroutine is either a simulation of a physical detector, and records the contribution to the detectors response when a particle arrives at the detector; or the detector subroutine is a so-called next-event estimator, and makes a contribution to the detectors response for each collision.

In the simulation of a physical detector, the particle fluence may be obtained by boundary-crossing, track-length, or collision-density estimators. In a boundary-crossing estimator, the particle fluence, $\Phi$, is defined as
where \( \text{AREA} \) denotes the area of the surface over which the fluence is to be estimated;

\( W_i \) is the statistical weight of the \( i \)th particle that crosses the surface which represents the detector;

\( \bar{\Omega}_i \) is the particles' direction at crossing; and

\( \bar{n}_i \) is a unit vector normal to the surface at the point that the particle crosses.

Therefore, the particle fluence obtained with the boundary-crossing estimator is an area-averaged value. A difficulty occurs with this type of detector, if a particle crosses the surface at a grazing angle, i.e. \( \bar{\Omega}_i, \bar{n}_i \to 0.0 \). Then very large contributions to the fluence are encountered. Thus, the boundary-crossing estimator cannot be used inside, or too close to, a scattering medium.

For a track-length estimator, the particle fluence is defined as

\[
\Phi = \frac{\sum W_i}{\text{AREA}} \quad \text{Eq. 2.4}
\]

\[
\Phi = \frac{\sum T_i}{\text{VOL}} \quad \text{(track length/volume)} \quad \text{Eq. 2.5}
\]

where \( \text{VOL} \) is the volume over which the fluence is to be estimated; and \( T_i \) is the sum of the track lengths in \( \text{VOL} \) contributed by the \( i \)th particle. The track-length estimator can be used both inside and outside of a scattering medium.

The collision-density estimator is based on the relation

\[
N = \Sigma_i \times \Phi \times \text{VOL} \quad \text{Eq. 2.6}
\]
where \( N \) equals the number of collisions in the volume (VOL). Therefore, the particle fluence is defined as

\[
\Phi = \frac{N}{\Sigma_i \cdot \text{VOL}} = \frac{\sum W_i}{\Sigma_i \cdot \text{VOL}}
\]  
Eq. 2.7

It is obvious that the collision density estimator is only good in a medium where many interactions occur. According to their definitions, the particle fluence estimated by both the track-length and the collision-density estimators, is volume-averaged.

These three physically-simulated detectors usually do not give good statistics in deep-penetration problems, because there are very few source particles that actually arrive at the detector. The majority of the source particles do not arrive at the detector, and are wasted. If one makes the detector very large in order to collect more counts, then the particle fluence which is obtained is not locally representative. This is the same dilemma faced by experimentalists when performing a very-low-count rate measurement of the fluence. The next-event point estimator may overcome this problem in certain circumstances, because each source particle is forced to make contribution to the detector at each collision, and therefore no source particle is wasted. In the next-event estimator, the fluence contribution to the detector at each collision is expressed by

\[
\Phi = \frac{W}{\Delta A} P(\tilde{\Omega}) e^{-\tilde{\Omega}} \Delta \Omega
\]  
Eq. 2.8

where \( W \) is the random walk particle weight before the collision;
\( P(\Omega) \) is the probability per steradian of scattering toward the detector;

\( \Delta \Omega \) is the solid angle, in steradians, subtended at the collision site by the detector;

\( \Delta A \) is the area of the detector projected onto a plane perpendicular to \( \Omega \);

\( \eta \) is the number of mean free paths from the collision site (\( r' \)) to the detector at 
\[
\int_0^r \frac{1}{\Sigma_i(r - R') \Omega} dR'.
\]

and \( R \) is the distance between the collision site and the detector, 
\[
R = |r - r'|.
\]

Another advantage of next-event estimator is that the detector can be as small as a point. This is not possible for physically-simulated detectors. In the limit as the detector becomes vanishingly small, the ratio \( \Delta \Omega / \Delta A \), becomes \( 1/R^2 \) and Eq. 2.8 becomes

\[
\Phi = \lim_{\Delta A \to 0} \frac{\Delta \Omega}{\Delta A} WP(\Omega) e^{-\eta} = \frac{W P(\mu) e^{-\eta}}{2\pi R^2} \tag{Eq. 2.9}
\]

where \( \mu \) is the cosine of the scattering angle of the collision. It should be noted, however, that next-event point estimator does not work for dose points which are inside a scattering medium, because in such a case, a few heavy-weight contributions from nearby collisions (for which \( 1/R^2 \to \infty \)) can ruin the statistics.

Following the rules discussed above, a next-event point estimator, RELCOL, was written to estimate the neutron and gamma-
ray fluxes at the irradiation point (see Fig.2-1 for the description); and a track-length estimator, TRKLH, was written to estimate the absorbed neutron and gamma-ray doses in the head phantom located at the irradiation point. RELCOL is called whenever a collision event occurs. TRKLH is called whenever a particle crosses the detector boundary. The FORTRAN listings of both RELCOL and TRKLH are provided in APPENDIX C. In order to improve the statistics when tracklength estimators were used, the path-length stretching technique, which is an option in MORSE, was used to bias neutrons to travel toward the detector. However, the improvement of the fractional standard deviation of the flux was marginal.

The absorbed dose rates in a phantom, are obtained by multiplying the group fluxes by their corresponding KERMA factors. The KERMA was assumed to equal the absorbed dose for both the neutron and gamma-ray dose for all dose points in the phantom. The KERMA factors of each element for the 47 neutron and 20 gamma-ray groups were available through BUGLE-80. The neutron KERMA factors and energy/group structures based on BUGLE-80, for normal tissue and for boron-10 at a concentration of 10 μg of $^{10}$B/g-tissue are provided in Table 2-2. The tissue composition is assumed to be $(C_5H_{40}O_{18}N)_n$. Since all neutrons produced by the lithium-7 target have energies less than 800 keV, KERMA factors are provided only for neutron energies less than 800 keV. The gamma-ray KERMA factors are presented in Table 2-3.
Table 2-2. The neutron KERMA factors for tissue and 10μg of $^{10}\text{B}/\text{g}$-tissue Extracted From BUGLE-80 Cross Section Library

<table>
<thead>
<tr>
<th>BUGLE-80 Group No.</th>
<th>Upper Energy Boundary (eV)</th>
<th>KERMA Factors, cGy-cm$^2$ Tissue($C_5H_{40}O_{18}N_4$)$_n$</th>
<th>$10\mu g$ of $^{10}\text{B}/\text{g}$-tissue</th>
</tr>
</thead>
<tbody>
<tr>
<td>20</td>
<td>8.20E+05</td>
<td>1.977E-09</td>
<td>1.288E-14</td>
</tr>
<tr>
<td>21</td>
<td>7.43E+05</td>
<td>1.825E-09</td>
<td>1.616E-14</td>
</tr>
<tr>
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<td>6.08E+05</td>
<td>1.653E-09</td>
<td>2.175E-14</td>
</tr>
<tr>
<td>23</td>
<td>4.98E+05</td>
<td>1.567E-09</td>
<td>2.546E-14</td>
</tr>
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<td>24</td>
<td>3.69E+05</td>
<td>1.319E-09</td>
<td>2.723E-14</td>
</tr>
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<td>2.97E+05</td>
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<td>3.250E-14</td>
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<td>1.83E+05</td>
<td>8.108E-10</td>
<td>4.058E-14</td>
</tr>
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<td>27</td>
<td>1.11E+05</td>
<td>5.780E-10</td>
<td>4.980E-14</td>
</tr>
<tr>
<td>28</td>
<td>6.74E+04</td>
<td>4.089E-10</td>
<td>6.096E-14</td>
</tr>
<tr>
<td>29</td>
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Table 2-3 The gamma-ray Kerma Factors for Tissue Extracted from BUGLE-80 Cross Section Library

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CHAPTER III

EXPERIMENTAL METHODS

3.1 Introduction

The experiment for this study was performed at the OSU Van de Graaff facility. The 15° North beam line was used to produce 2.5 MeV protons. Several components were constructed for this experiment. These include a lithium target assembly, a neutron moderator assembly, and a phantom. Several techniques were developed to measure the neutron spectra, and the absorbed doses in the phantom for the experimental neutron field. The components, experimental setups, measured quantities, and measurement techniques are all described in this chapter. The proton current was adjusted to match the counting rates of the measurement system, so that the detector system dead time was no greater than about 5%. However, the maximum achievable proton current for the OSU Van de Graaff accelerator is only about 1 μA. This low proton current caused difficulties in the experiment for insensitive detectors. These difficulties are explained later in this chapter.

3.2 Descriptions of Components

The target assembly consists of a flange which is compatible with the Van de Graaff beam line, a gate valve, and a lithium metal
target embedded at the end of a short drift tube. The gate valve is kept closed, when the lithium target is not in use, so that a good vacuum is maintained on the lithium surface to prevent it from oxidizing. The target is backed with copper, and cooled with water. Figure 3-1 illustrates the geometric configuration of the target assembly.

The moderator assembly consists of an aluminum right circular cylindrical tank which is filled with D$_2$O. D$_2$O has been loaned to us by the Department of Energy. The tank rests horizontally on a cradle, so that the tank's centerline can be aligned with the beam centerline at a height of 42" from the ground. The tank wall is 1/8" thick, and the tank is 36.5 cm in diameter, and 30 cm in height. The inside of the tank is divided into two compartments by a 1/8" thick aluminum bulkhead. The bigger compartment is 20 cm in length, and the smaller one is 10 cm in length. The purpose in this experiment was to generate epithermal neutrons, so only the bigger compartment was filled up with D$_2$O, and the smaller compartment was left empty. The same tank can be used to generate thermal neutrons, if the smaller compartment is also filled with D$_2$O. The inlet and outlet of the tank were connected with a short Tygon tube to provide an expansion chamber. Figure 3-2 illustrates the geometric configuration of the moderator assembly.

The phantom is a water-filled lucite box. The box has dimensions of 20x20x30 cm, and water is filled to a height of 20 cm to create a 20x20x20 cm water cube. The lucite wall thickness is 0.635 cm. Figure 3-3 illustrates the configuration of the phantom.
Figure 3-1. The illustration of the lithium target assembly which was constructed and tested in the experiment.
Figure 3-2. The illustration of the neutron moderator assembly which was constructed and tested in the experiment.
Figure 3-3. The illustration of the phantom which was used in the experiment for absorbed dose measurements.
3.3 Experimental Setups

In order to be parallel to the calculational study, the experiment was performed in three steps.

The first step was to measure the spectra of neutrons directly emitted from the lithium target, so that they can be compared with the spectra calculated by the NSRCE program. Neutron spectra were measured at a distance of 30 cm from the lithium target for three angles: 0°, 45°, and 90°. The geometric configuration of the experimental setup is illustrated in Figure 3-4.

The second step was to measure the spectrum of neutrons emerging from the moderator assembly, so that it can be compared with the spectrum calculated by the MORSE code. The moderator assembly was placed 2 cm from the lithium target, with its centerline collinear with the beam line. The neutron spectrum was measured at a point which is 8 cm from the moderator assembly on the beam centerline. The geometric configuration of the experimental setup is illustrated in Figure 3-5.

The third step was to measure absorbed doses in the head phantom for the neutron field emerging from the moderator assembly. The absorbed dose-to-depth distributions in the phantom not only have direct meaning for BNCT, but also indirectly verify the spectrum of the neutron field. Figure 3-6 illustrates the geometric configuration of the experimental setup. As it is shown, the absorbed doses were measured at various depths in the phantom on the centerline of the neutron field. The original plan for this part of the experiment was to measure the absorbed neutron doses, gamma-ray doses, and the doses
Figure 3-4. The geometric configuration for the first experimental setup. The neutron spectra were measured at positions 1, 2, and 3.
Figure 3-5. The geometric configuration for the second experimental setup. The neutron spectrum was measured at the irradiation point as indicated.
Figure 3-6. The geometric configuration for the third experimental setup. The absorbed doses of gamma-rays and $^{10}$B(n,α)$^7$Li reactions were measured at several depths in the phantom on the beam centerline.
created by $^{10}\text{B}(n,\alpha)^7\text{Li}$ reactions (or called boron dose for short). However, because of the low proton current (< 1 $\mu$A), neutron dose rates were too low to be measured. Therefore, only the gamma-ray doses and boron doses were measured. Also, for boron dose measurements, a cadmium sheet was placed on the front surface of the moderator assembly to prevent thermal neutrons from entering the phantom.

3.4 Measurement Techniques

Several techniques were developed to measure the neutron spectra, and the absorbed doses in phantom, for this experiment. These techniques are fully described in the subsections that follow.

3.4.1 Neutron Spectrum Measurement Techniques

A screening process was done to decide the most appropriate technique for neutron spectrum measurement. Figure 3-7 shows the approximate energy ranges covered by different types of spectrometers$^{29}$. All of these techniques require some sort of unfolding process to obtain the neutron spectrum from the measured data. The energy range of interest in this study is between 1 eV and 800 keV. As it is shown in Fig. 3-7, four types of spectrometer, either cover, or greatly overlap with, the energy range of interest. They are time-of-flight (TOF), $^6\text{Li}$ sandwich, proton recoil proportional counters (PRPC), and Bonner spheres. TOF requires either a beam chopper, or a pulsed proton beam; and neither of them were easily available to us. In addition, TOF is more amenable to a well-collimated neutron beam
Figure 3-7. The approximate energy ranges covered by different neutron spectrometer types.
than a semi-isotropic neutron field, which is expected for this experiment. Therefore, the idea of using TOF was dropped. According to ref. 15, the $^6$Li sandwich spectrometer has the lowest neutron detection efficiency among all the types of spectrometers. The neutron detection efficiency is important to the spectrum measurement, because the low proton beam current ($<1$ μA) of the OSU Van de Graaff translates to a low-flux neutron field, which was to be measured. Besides, for neutrons with energies below 100 keV, a $^6$Li spectrometer requires complicated electronics and spectrum unfolding techniques$^{30}$. Therefore, the idea of using a $^6$Li spectrometer was also dropped. Bonner sphere spectrometer covers a wide range of energy, and has high detection efficiency, but its energy resolution is the poorest among all the spectrometer types. A PRPC has good neutron detection efficiency and good energy resolution, and therefore was the final choice for the neutron spectrum measurement for this experiment. However, as it is shown in Fig. 3-7, a PRPC can only detect neutrons with energies above a few keV. For neutrons with energies less than a few keV, a new spectrometer called a boron-shell spectrometer was developed to measure the spectrum. Descriptions of the spectrometers, and their calibration, measurement, and spectrum unfolding techniques are provided in the section below.

3.4.1.1 Neutron Spectrum Measurement by Proton Recoil Proportional Counter (PRPC)

Neutron spectrum measurements using a PRPC are based on the fact that monoenergetic neutrons of energy $E_n$ give rise to a uniform ("rectangular") distribution of proton energies from 0 to $E_n$, in a n-p
scattering. For a rectangular response function, the neutron spectrum unfolding is quite simple. However, in practice, PRPCs do not yield an ideal rectangular response function, mostly because of the "wall and end" effects. In addition, the degree of distortion of the response function is often unknown, and hence creates an inherent uncertainty in the neutron energy resolution. Some inherent uncertainty may be unavoidable, but it can, and should, be minimized. As a matter of fact, much of the effort made in this measurement involves minimizing the wall and end effects of the counter.

The PRPC used in this experiment is a type 27050 counter, made by LND Inc. The counter is spherical, with a 2 inch diameter, and is filled to 10 atmospheres with H₂. The counter has an internal capacitance of 50 pF. Figure 3-8 illustrates the internal structure of such a PRPC. A spherical counter was chosen over a cylindrical one, because its response function is isotropic with respect to the incident neutrons. However, a spherical counter has more pronounced end effect, as a consequence of nonuniformity in the electric field in the counter, and this makes its energy resolution worse than that of a cylindrical counter. In addition, the maximum gas multiplication for a spherical counter is usually smaller than that of a cylindrical counter. High gas multiplication is always desirable, because it allows one to recover very small signals which would otherwise be buried in the preamp noise. A gas pressure of 10 atm of hydrogen was chosen to minimize the counter wall effect. Calculations show that, at 10 atm, more than 80% of the most energetic (800 keV) protons deposit their full energies in the counter. Gas pressures higher than 10 atm usually
Figure 3-8. The internal structure of a spherical proton recoil proportional counter
full energies in the counter. Gas pressures higher than 10 atm usually degrade the counters' resolution, because continuous discharge begins to occur in the counter.

Figure 3-9 illustrates the measurement system, which includes the PRPC, a CANBERRA 2006E preamplifier, a 5 kV detector bias voltage supply, a CANBERRA 4225 spectroscopy linear amplifier, and a PC-based multichannel analyzer (MCA). During the experiment, the electronic part of the measurement system suffers a great deal of the noise problem. It was observed that operation of the CANBERRA 4225 amplifier in the bipolar rather than the unipolar mode, greatly reduced electronic noise in the measurement system.

The calibration of the PRPC was done in the OSU subcritical assembly. The PRPC is constructed to be self-calibrating, by incorporating a small quantity ($10^{-3}$ atm) of $^3$He into the detector fill gas. The pulses generated by the $^3$He(n,p)$^3$H reaction, in a thermal neutron field give a peak (called the $^3$He peak, for convenience) in the pulse height distribution (PHD). The Q value of the $^3$He(n,p)$^3$H reaction is 764 keV. But, because the triton generates more electron-ion pairs per unit energy than the proton does, the $^3$He peak is equivalent to a proton with an energy of 780 keV$^{31}$. Figure 3-10 shows the calibration PHD which was recorded with the counter bias voltage at 2500 volts. The calibration curve of gas multiplication versus counter bias voltage is shown in Figure 3-11. The calibration was done at the OSU subcritical assembly, and was based on the following procedures:
Figure 3-9. The electronics for the proton recoil proportional counter measurement.
Figure 3-10. The pulse height distribution recorded by the proton recoil proportional counter in the OSU subcritical assembly. The proportional counter was operated at 2500 volts.
Figure 3-11. The gas multiplication versus bias voltage for the proton recoil proportional counter
(1) Set the counter bias voltage at 1500 volts, and adjust the amplifier gain, so that the $^3$He peak is located on the right half of the MCA screen. Record both the amplifier gain settings, and the channel number of the $^3$He peak on the MCA.

(2) Set the counter bias voltage at 1750 volts, and vary the amplifier gain, so that the $^3$He peak is located at the same channel number as that which was recorded in step 1. Then record the amplifier gain settings.

(3) Set the amplifier gain the same as in step 1. Disconnect the PRPC from the preamp, and feed the preamp with test pulses from a mercury switch pulser. Then adjust the pulser dial to locate the its peak on the right half of the MCA screen, and record the channel number of the pulser peak.

(4) Keep the pulser dial setting the same as in step 3, and change the amplifier gain setting to the value which was recorded in step 2. Then record the channel number of the pulser peak.

(5) Calculate the ratio of the channel number recorded in step 3 to that recorded in step 4. This ratio is the counter gas multiplication ratio for the two bias voltages set in steps 1 and 2.

(6) Repeat steps 1 through 5 many times. Each time increase the counter bias voltage by 250 volts, until the counter bias voltage reaches 5000 volts.

As it is shown in Fig. 3-11, the gas multiplication of the counter starts to increase at about 2000 volts, and reaches its full value of about 20, at 4750 volts. Above 4750 volts, continuous discharge begins to occur and the shape of the $^3$He peak is totally distorted. Based on Fig. 3-11,
the useful proportional region of the counter bias was determined to be between 2000 and 4750 volts.

It should be noted that, even when the counter is operated within its proportional region, the counter does not have a constant energy resolution. The resolution here is defined as the full width at half-maximum (FWHM) of the $^3$He peak divided by the channel number of the peak. An asymmetry in the $^3$He peak was observed for bias voltages above 3000 volts. This asymmetry has also been observed by others\textsuperscript{32}, and was attributed to the nonuniform increase in the electric field in the end region of the counter, as the gas multiplication increases. Figure 3-12 shows the relationship between the resolution and the gas multiplication for the PRPC.

Because the energy range of the neutron spectrum (1 eV to 800 keV), which was measured, is too wide to be covered by a single set of measurements; three sets of measurements were made, with the counter operated at 2500, 4000, and 4750 volts, respectively. The measurement with the detector biased at 2500 volts covers the spectrum between 200 and 800 keV. The measurement with the detector biased at 4000 volts covers the spectrum between 50 and 200 keV. The measurement with the detector biased at 4750 volts covers the spectrum between 10 and 50 keV. Because the gas multiplication is not large enough, the signals, for neutrons with energies less than 10 keV, are buried in the preamp noise, and thus cannot be recovered.

Since gamma rays also interact with the counter walls, some of the electrons generated in the wall may get into the counter fill gas
Figure 3-12. The energy resolution versus bias voltage for the proton recoil proportional counter.
and generate undesirable pulses. These gamma-ray pulses are mostly small signals, but they do overlap with the signals of neutrons with energies less than 100 keV. In this experiment, gamma rays are expected from the $^{27}\text{Al}(n,\gamma)^{28}\text{Al}$ reactions which occur in the moderator assembly wall. Therefore, a gamma-ray discrimination technique must be developed to remove the gamma-ray signals from the recorded PHD. The first attempt at gamma-ray discrimination was made using the standard technique called pulse-shape discrimination (PSD), which is based on the fact that the average rise time of neutron pulses in a PRPC is shorter than that of gamma-ray pulses. Therefore, by measuring the rise time, as well as the amplitude, of all the pulses, one should be able to separate neutron signals from gamma-ray signals. The attempt was totally unsuccessful, however, because the rise time difference was not observed in the PRPC. Another technique for gamma-ray discrimination is to use PRPCs with various gas pressures so that gamma-ray signals can be removed simply by pulse-height discrimination. This technique was not amenable to us, because we do not have a gas-fill system associated with the PRPC. The gamma-ray discrimination was finally accomplished by making an additional measurement with a small GM counter (model GM-2, made by Far West Technology Inc.). The GM counter, which responds only to gamma rays, was placed beside the PRPC during the measurement. The gamma-ray counts recorded by the GM counter were converted to a PHD, which would have been recorded by the PRPC in the absence of neutrons. The conversion, of GM counts to a gamma-ray PHD for the PRPC, was based on a calibration experiment, in which both the GM
counter and the PRPC were exposed to a Na-22 source. Figure 3-13 shows two PHDs for the Na-22 gamma rays recorded by the PRPC with the bias voltages at 4000 and 4750 volts. These gamma-ray PHDs was then subtracted from the PHDs, which were measured at the corresponding voltages by the recoil detector, to obtain the net neutron PHDs. For the PRPC biased at 2500 volts, the gamma signals are quite small compared to the neutron signals of interest, and therefore the gamma-ray contribution to the PHD can be discriminated from the neutron contribution on the basis of channel number.

In addition to the undesirable gamma-ray signals, significant (~5% of the total countrate) background signals were also observed during the PRPC measurement. These background signals were also removed from the total PHD. Thus,

\[
\text{net neutron PHD} = \text{total PHD} - \text{gamma-ray PHD} - \text{background PHD}
\]

The net neutron PHD was then used to unfold the neutron spectrum. The neutron spectrum was unfolded using a spectrum unfolding code called SPEC-4\textsuperscript{33}, which was specifically written for spherical proton recoil proportional counter. The input data required are the net neutron PHD, the counter parameters, and an estimate of the neutron spectrum above \( E_{\text{max}} \), the highest energy to be analyzed. The unfolded spectrum is presented as neutron fluence per unit lethargy interval, in groups of width equal to the counter resolution. The unfolding method used in SPEC-4 is based on a knowledge of the PHD produced in the counter by mono-energetic neutrons (including the calculated "wall
Figure 3-13. The pulse height distribution for Na-22 gamma-rays recorded by the proton recoil proportional counter with bias voltage at: (A) 4000 volts, and (B) 4750 volts.
effect"). An analytical expression developed by Snidow\textsuperscript{34} is used to generate the "response function" for a given neutron energy. The integrated response to neutrons, with energies above $E_{\text{max}}$, can thus be calculated and subtracted from the measured PHD. The neutron spectrum, in defined energy groups is then found by unfolding a matrix of response functions at the specified energies, using "back substitution". The three sets of measured PHD data, at counter bias voltages of 2500, 4000, and 4750 volts, were analyzed by SPEC-4 in one continuous computer run. Each set of PHD data covers about one third of the full spectrum. The data were analyzed in the order of increasing counter bias voltages. That is, SPEC-4 first used the PHD data measured with the counter bias voltage at 2500 volts, and assumed zeros for the spectrum above 800 keV, to unfold the neutron spectrum between 200 and 800 keV; this neutron spectrum was then used with the PHD data measured with the counter bias voltage at 4000 volts, to unfold the neutron spectrum between 50 and 200 keV; the unfolded spectrum between 50 and 800 keV was then used with the PHD data measured with the counter biased at 4750 volts, to obtain the neutron spectrum between 10 and 50 keV. All three steps described above were accomplished by SPEC-4 in one continuous computer run. The final SPEC-4 output displays the full neutron spectrum from 10 to 800 keV.
3.4.1.2 Neutron Spectrum Measurement by Boron-Shell Neutron Spectrometer

For neutrons with energies less than 10 keV, a new instrument called "boron-shell neutron spectrometer" was developed to measure the spectrum. The spectrometer is similar in principle to Bonner Spheres. A Bonner Sphere spectrometer does not work for neutron energies less than 10 keV, because its response functions do not show any peak-like characteristics in this low energy region. The boron-shell spectrometer was developed to overcome this problem. The response function is defined here as the spectrometer counts per unit of neutron fluence.

The boron-shell spectrometer is based on a set of interchangeable hemispherical shells, which contain various amounts of $^{10}$B, and a small (one inch in diameter) spherical $^3$He proportional counter, which is located at the focus of the hemispherical shells (which will hereafter be called boron shells for convenience). Also, for the boron shells which contain larger amounts of $^{10}$B, the $^3$He proportional counter is surrounded by a spherical paraffin jacket. The thickness of the paraffin jacket is larger for shells with larger amounts of $^{10}$B. Figure 3-14 illustrates the geometric configuration of the boron-shell spectrometer. With different combinations of boron shells and paraffin jackets, it is possible to develop a set of response functions, which have peak-like characteristics in the neutron energy region between 1 eV and 10 keV. Arguments which support such a design are provided below. Incoming neutrons are filtered by the boron shell, before they reach the $^3$He counter. Because the boron
Figure 3-14. The geometric configuration of the boron-shell neutron spectrometer
shell is spherical, the neutrons that strike the $^3$He counter are restricted to those which are nearly perpendicularly incident upon the boron shell, and these neutrons pass through nearly the same thickness of shell material in reaching the counter. Therefore, the filtering of spectra, and hence the response function of the spectrometer, depends only on the neutron energy ($E_n$) and not on the neutron angular distribution. Since the neutron capture cross section of $^{10}$B is proportional to $E_n^{-1/2}$, the $^{10}$B shell filters out low-energy neutrons in preference to high-energy neutrons. However, since the cross section for the $^3$He(n,p)$^3$H reaction is also proportional to $E_n^{-1/2}$, of the neutrons reaching the proportional counter, the proportional counter responds most strongly to the neutrons with the lowest energies. Therefore, by using a set of $^{10}$B shells with various $^{10}$B loadings and a $^3$He proportional counter, one can design a neutron spectrometer which has a set of response functions which peak at various energies between 1 eV and 10 keV. Similar techniques have been developed by Eisen et al. for a personnel neutron dosimeter. The simple spectrometer based on a set of $^{10}$B shells and a $^3$He proportional counter, unfortunately only responds very weakly to neutrons with energies above 100 eV. In order to enhance the response function for neutron energies above 100 eV, a paraffin jacket is placed around the $^3$He counter. Then neutrons with energies above 100 eV are moderated somewhat in the paraffin, thus increasing the probability that they will be registered by the $^3$He counter.

The fabrication of the hemispherical boron shells was quite difficult. Because of the large $^{10}$B loading (~1.0 g/cm$^2$) required for the
measurement, it was necessary to use highly $^{10}\text{B}$-enriched boron powder to make the shells. The boron powder was supplied by Eagle-Picher Inc., and is 90% $^{10}\text{B}$-enriched. Hemispherical models and rams were made, to press the boron powder into hemispherical shells of varying radii and thicknesses. Because boron powder does not hold its shape, a casing for the boron powder was formed by wrapping the power in a 5 mil thick aluminum foil before pressing. Also, before pressing a shell, the quantity of the boron powder was carefully weighed, so that the average $^{10}\text{B}$ loading of each shell in gm/cm$^2$ could be well known. Unfortunately, the $^{10}\text{B}$ loading in g/cm$^2$ varies from its average over the shell's surfaces, because of wrinkling of the aluminum casing, deviations from sphericity of the mold and press and misalignment of the centers of the hemispheres formed by the mold and press.

Calibration of the $^3\text{He}$ proportional counter was performed at the OSURR thermal column, to determine the operating bias voltage and to verify the $^3\text{He}$ fill gas pressure. Measurements were made with the $^3\text{He}$ counter biased at several voltages between 1000 and 3000 volts (which is the specified proportional region), and the PHDs were recorded on an MCA. 1500 volts was selected as the operating voltage, because the PHD at this voltage not only has good resolution for the $^3\text{He}$ peak, but also has a clear separation between the signals and the noise. Figure 3-15 shows the PHDs for a bias voltage of 1500 volts. The $^3\text{He}$ gas pressure (in atm) of the counter was determined based on the following formula:
Figure 3-15. The pulse height distribution of thermal neutrons recorded by the $^3$He proportional counter operated at 1500 volts.
\[ p = \frac{C}{\sigma_{th} \Phi_{th} N} = \frac{C k T}{\sigma_{th} \Phi_{th} V} \]  

**Eq. 3.1**

where  
- \( C \) = total counts recorded by \(^3\)He proportional counter  
- \( N \) = number of \(^3\)He atoms in the counter per atmosphere  
- \( \sigma_{th} \) = microscopic thermal neutron absorption cross section of a \(^3\)He atom, in barns/atom  
- \( \Phi_{th} \) = thermal neutron fluence, in neutrons/cm\(^2\)  
- \( k \) = the Boltzmann constant  
- \( T \) = temperature of the counter  
- \( V \) = gas volume of counter

\( \Phi_{th} \) was obtained by activation of gold foils. The total counts, \( C \), was obtained from the PHD, by summing up all the signals which were above the noise level. The calibration shows that 5.3 counts recorded by the \(^3\)He counter are equivalent to 1.0 n/cm\(^2\) of thermal neutron fluence. Based on a detector response of 5.3 counts-cm\(^2\), and the assumed values of 8.58 cm\(^3\) and 5400 barns for \( V \) and \( \sigma_{th} \) respectively, the \(^3\)He pressure was found to be 4.2 atms, which is quite different from the vendor-specified value of 10 atm. This calibration was done twice in the two-year period of this project, and the results of the calibrations are consistent. Since this calibration is quite straightforward, we decided to use 4.2 atms as the \(^3\)He pressure, to calculate the response functions for the boron-shell spectrometer.
The response functions of the boron-shell spectrometer were calculated using the MORSE\textsuperscript{26} code in conjunction with the BUGLE-80 cross-section library\textsuperscript{27}. The FORTRAN listing of the estimator which calculates the response functions is provided in Appendix D. The seven combinations of boron shell and paraffin jacket used for this experiment are provided in Table 3-1. The calculated response functions for the seven combinations are presented in Figure 3-16. As it is shown, each of the seven response functions has a gentle peak. In addition, on a logarithmic energy scale these peaks are evenly distributed in the energy range of 1 eV to 10 keV. Therefore, this set of response functions should serve well in the unfolding process for obtaining neutron spectra in this energy range. The neutron spectrum was unfolded based on a set of counts recorded for all seven boron-shell/paraffin combinations.

The code used in the unfolding process is a modified version of SPUNIT, which was originally developed at Battelle Northwest Laboratory, for Bonner Sphere systems. The basic equation describing the unfolding problem is:

$$C_k = \sum_{i=1}^{n} \phi_i R_{ik}$$

\text{Eq. 3.2}

where $C_k$ is the measured count rate for the $k^{th}$ detector, $\phi_i$ is the neutron flux in the $i^{th}$ energy bin, $R_{ik}$ is the response function relating the response of the $k^{th}$ detector to the flux in the $i^{th}$ energy bin, and $n$ is the total number of energy bins needed to define the neutron spectrum. The spectrum unfolding technique used in SPUNIT is based on an iterative algorithm developed by Doroshenko et al.\textsuperscript{36} The
Table 3-1. The material compositions of the boron-shell neutron spectrometer

<table>
<thead>
<tr>
<th>Detector Number</th>
<th>Boron-10 Loadings (g/cm²)</th>
<th>Paraffin Thickness (cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.0362</td>
<td>0.0</td>
</tr>
<tr>
<td>2</td>
<td>0.0674</td>
<td>0.0</td>
</tr>
<tr>
<td>3</td>
<td>0.15</td>
<td>0.0</td>
</tr>
<tr>
<td>4</td>
<td>0.39</td>
<td>0.64</td>
</tr>
<tr>
<td>5</td>
<td>0.55</td>
<td>0.64</td>
</tr>
<tr>
<td>6</td>
<td>0.79</td>
<td>0.95</td>
</tr>
<tr>
<td>7</td>
<td>0.95</td>
<td>0.95</td>
</tr>
</tbody>
</table>
Figure 3-16. The set of response functions of the $^3$He proportional counter for the seven combinations of boron shells and paraffin jacket shown in Table 3-1.
iterative technique uses the following equation:

\[
\phi_{i,p+1} = \frac{\phi_{i,p}}{\sum_{j=1}^{m} R_{ij}} \sum_{k=1}^{m} \frac{C_k}{N_{k,p}}
\]

Eq. 3.3

where \(\phi_{i,p}\) = the flux for energy bin \(i\) calculated during the \(p^{th}\) iteration

\(m\) = the number of detectors

\(N_{k,p}\) = the calculated count rate for \(k^{th}\) detector at \(p^{th}\) iteration

\(= \sum_{i=1}^{n} \phi_{i,p} R_{ik}\)

The iteration process is started with an assumed initial spectrum. As it is shown in the above equation, the fluxes for each iteration are calculated by using the results of a previous iteration and the ratio of the measured and calculated detector count rate. The iteration is continued until a convergence criterion is met. In equation 3.1, there are \(m\) equations and \(n\) unknowns. Usually \(n\) is greater than \(m\), hence no unique solution exist; and even when \(n<m\), the response function matrix \(R_{ik}\) is usually ill-conditioned, resulting in widely oscillating, sometimes negative, solutions having little physical significance. That is, the solution obtained using equation 3.2 may fit the measured data nicely, but be physically unreasonable. To overcome this problem, a spectrum smoothing algorithm, suggested by Sanna\(^{37}\) was incorporated into SPUNIT. The smoothing used is as follows:
\( (\phi_i)_s = (s\phi_{i-1} + \phi_i + s\phi_{i+1})/(1 + 2s), \quad i = 2,3,\ldots,n \)

\( (\phi_1)_s = \phi_1 \)

\( (\phi_n)_s = \phi_n \)

Eq. 3.4

where \( \phi \) is the spectrum before smoothing, and \( (\phi)_s \) is the spectrum after smoothing. The FORTRAN listing of the unfolding code, which includes the response function matrices, and the iterative and smoothing algorithms, is provided in Appendix E.

The performance of the unfolding code was tested with an artificial input. The artificial input was the detector counts derived from the response functions for a mono-energetic beam of neutrons, producing a fluence of 1000 n/cm\(^2\) in energy bin 7. Figure 3-17 presents the unfolded spectrum from this input. The unfolded spectrum reproduced the input spectrum: bin 7 had a fluence of 677 n/cm\(^2\), bin 6 had 148 n/cm\(^2\), and bin 8 had 175 n/cm\(^2\). Thus 32% of the fluence ended up in the "wrong" bins. This is not a poor performance for an unfolding code. For the relatively smooth spectrum, which is unfolded from the spectrometer data, the error incurred in the unfolded spectrum is probably less than 32%, because error incurred by the placement of fluence from the \( n^{th} \) bin into the \( n-1^{th} \) bin is offset by placement of fluence from the \( n+1^{th} \) bin into the \( n^{th} \) bin, and vice versa.
Figure 3-17. The unfolded results performed by the modified SPUNIT code for a single-bin neutron spectrum
3. 5 Absorbed Dose Measurement

3.5.1 Gamma-Ray Dose Measurement

The absorbed gamma-ray doses at various depths in the phantom were measured with the small GM counter, which has been described in Section 3.2.1. A calibration of counts versus absorbed dose was accomplished by the use of a 1 Curie Cs-137 source provided by the OSU Office of radiation Safety. A calibration curve of GM count rate versus absorbed dose rate is provided in Figure 3-18. The curve is quite linear up to 5 cGy/hr.

3.5.2 Boron-10 Dose Measurement

The boron doses at various depths in the phantom was measured with a small cylindrical BF₃ counter. The active volume of the BF₃ counter is 0.635 cm in diameter and 3 cm long. It is made by Reuter Stokes Inc. The operating voltage for the BF₃ counter was determined to be 1800 volts, based on the same technique used for the ³He counter described previously. The BF₃ counter efficiency was calibrated in the OSU subcritical assembly against the ³He counter, which was described in the previous section. The ratio of the thermal neutron detection efficiency of the ³He counter to that of the BF₃ counter was calibrated to be 26.5 to 1. Since the counts versus thermal neutron fluence for the ³He counter has been calibrated to be 5.2 to 1, the counts versus thermal neutron fluence for the BF₃ counter is therefore 0.2 to 1. Based on this relationship, the counts recorded by the BF₃ counter inside the phantom were then converted to obtain an "equivalent thermal neutron fluence". Finally, the boron
Figure 3-18. The calibration curve of GM counts versus absorbed dose using a Cs-137 source.
dose was obtained by multiplying the equivalent thermal neutron fluence by a kerma factor corresponding to a specified $^{10}$B concentration.
CHAPTER IV

RESULTS AND DISCUSSIONS

4.1 Introduction

Both the calculational results and the experimental results are presented in this chapter. Comparisons between the two results are made, and the possible causes of the discrepancies are discussed. Efforts are made to validate the calculational methods based on the consistencies between the two results, so that the calculational methods can be used to design the optimum moderator assembly and to evaluate its performance for BNCT.

4.2 Results and Comparisons

Figures 4-1, 4-2, and 4-3 show both the calculational and the measured results, for the three cases of the first experimental setup (see Fig. 3-4 of chapter III for a description). Figure 4-1 presents the neutron spectra for the case of $\theta = 0^\circ$. Figure 4-2 presents the neutron spectra for the case of $\theta = 45^\circ$. Figure 4-3 presents the neutron spectra for the case of $\theta = 100^\circ$. The solid lines are the calculated spectra obtained from NSRCE. The dotted lines are the neutron spectra unfolded from the pulse height distribution (PHD) recorded by the proton recoil proportional counter (PRPC) using the SPEC-4 code. The boron-shell spectrometer was not used in this set of measurements, because the neutrons directly emitted from the
Figure 4-1. The calculated and the measured spectra of neutrons emitted at 0° angle, from the $^7$Li target which was bombarded by 2.5 MeV protons.
Figure 4-2. The calculated and the measured spectra of neutrons emitted at 45° angle, from the $^7$Li target which was bombarded by 2.5 MeV protons.
Figure 4-3. The calculated and the measured spectra of neutrons emitted at 90° angle, from the \textsuperscript{7}Li target which was bombarded by 2.5 MeV protons.
lithium target all have energies above a few keV, and were covered quite nicely by the PRPC alone. The spectra are presented in units of neutron flux per unit lethargy. As it is shown, the shape of the calculated spectra matches quite well with the shape of the measured spectra. However, in general, the absolute values of the measured spectra are lower, by a factor of 2 to 3, than those of the measured spectra.

Figure 4-4 presents the neutron spectra for the second experimental setup (see Fig. 3-5 of Section III.2 for a description). The spectra are presented with the same format as Figures 4-1, 4-2, and 4-3. The solid line is the calculated spectrum obtained from the MORSE code. The dotted line is the measured spectrum, and combines results obtained from the PRPC measurement and the boron-shell spectrometer measurement. As it is shown, the neutron spectrum above 10 keV, which was obtained from the PRPC, is again lower by a factor of 2 to 3, than that of the calculated spectrum. But, the neutron spectrum below 10 keV, which was obtained from the boron-shell spectrometer, matches quite well with that obtained from the MORSE calculation.

Figure 4-5 and 4-6 present the results for the third experimental setup (see Fig. 3-6 of Section III.2 for a description). Figure 4-5 shows the absorbed gamma-ray dose-to-depth curves. These include the doses contributed from both the incident gamma rays and the gamma rays induced in the phantom. The solid line is the calculated dose-to-depth curve obtained from the MORSE code. The
Figure 4-4. The calculated and the measured spectra of neutrons emerging from the 20 cm thick D$_2$O moderator.
Figure 4-5. The calculated and the measured absorbed gamma-ray doses in the phantom which was irradiated by the D$_2$O moderated neutron field.
Figure 4-6. The calculated and the measured absorbed boron doses in the phantom which was irradiated by the D$_2$O moderated neutron field. The B-10 concentration in the phantom was assumed to be 30 $\mu$g/g of tissue.
dotted line is the dose-to-depth curve obtained from the measurements made with the small GM counter. As it is shown, the code results agree with the measured results within 30%, and for depths greater than 2 cm, the agreement is nearly perfect. Figure 4-6 presents the boron dose-to-depth curves. The solid line is the calculated boron dose-to-depth curve obtained from the MORSE code. The dotted line is the boron dose-to-depth curve obtained from the measurements made with the small BF\textsubscript{3} counter. Again, the code results agree well with the measured results.

4.3 Discussions

In making comparisons between the calculational results and the measured results, one should bear in mind that there are inherent errors associated with both the calculational methods and the measurement techniques. The errors associated with the calculational methods result from:

(1) the use of the multigroup Monte Carlo technique, because the grouped BUGLE-80 cross section library was collapsed based on a typical light water reactor neutron spectrum. This error is more serious when there are many materials with sharp resonance neutron absorption peaks. In this study, aluminum is the only material which has sharp resonance peaks. But, because the quantity of aluminum is relatively small compared to the neutron moderating material (D\textsubscript{2}O) in the experimental moderator assembly, this error is not serious in this study; and
(2) the use of the track-length detector in estimating the flux in the phantom. In order to achieve reasonable statistics, the track-length detector has a diameter of 2 cm. The estimated flux is therefore, the average value in a detector of this volume. Because the mean free path for thermal neutrons in tissue is 0.35 cm, which is much less than the detector diameter, the flux based on the response of such a relatively large volume detector may deviate by a significant amount from the value of the flux at the point at which the average flux is calculated.

The errors associated with the measurement techniques result from:

(1) the uncertainty in the $^{10}$B loading of each boron shell of the boron-shell spectrometer. The $^{10}$B loadings, which appear in Table 4, were obtained assuming that the quantity of boron in each shell is correctly measured, and that the boron areal density is uniform over the shell region.

(2) the uncertainty in the H$_2$ gas pressure of the proton recoil proportional counter (PRPC). The H$_2$ gas pressure could be easily measured, if there were a gas-filling system associated with the counter. This, unfortunately was not the case for the PRPC used in the experiment.

(3) the possibility of oxidization of the lithium target, and a consequent reduction in the neutron yield per unit charge striking the target.
(4) the possibility of a portion of the proton beam missing the lithium target, but still contributing to the measured charge striking the target. This also reduces the neutron yield per unit charge.

Despite so many possible causes of errors in calculations and measurements, the calculational results agree reasonably well with the measured results. A discrepancy exists only for the measurements made with the proton recoil proportional counter (PRPC). In addition, as it is shown in Figures 4-1 thru 4-4, the discrepancy follows a very consistent pattern. That is, all four neutron spectra obtained with the PRPC have the same shape as those obtained from calculations, but the absolute value of the flux measured with the PRPC is lower, by a factor of 2 to 3. This suggests that the actual values of the response functions of the PRPC may be lower, by a factor of 2 to 3, than the values which were used in the spectrum unfolding process. Since the absolute value of the response function of the PRPC is proportional to the hydrogen gas pressure, this then suggests that the hydrogen pressure of the PRPC may be lower, by a factor of 2 to 3, than the vendor-specified value of 10 atm s. It is interesting that both the $^3$He counter and the PRPC used in this experiment were made by LND Inc., and they both seem to have gas pressure which are a factor of 2 to 3 times lower than their specified values.

In summary, the Van de Graaff experiment has served very well as a benchmark problem for the neutronic study for the proposed LPANIF. The measured results have certainly improved the confidence in the calculational methods, which were used in the rest of this study.
to design the optimum moderator assembly, and to evaluate its performance for BNCT.
NEUTRON MODERATOR ASSEMBLY DESIGN

5.1 Introduction

This chapter describes the criteria and methodology which were used to design the optimum neutron moderator assembly for the proposed low-energy proton accelerator neutron irradiation facility (LPANIF).

5.2 Design Criteria for the Moderator Assembly

The design criteria for the moderator assembly are summarized below:

1. The absorbed neutron dose to fluence ratio (ANDFR) should be less than $1.0 \times 10^{-10}$ cGy-cm$^2$, which corresponds to the kerma factor of a 10 keV neutron. ANDFR is defined as

$$\text{ANDFR} = \frac{D_n}{\Phi} = \frac{\sum_i K_i \Phi_i}{\sum_i \Phi_i} = \bar{K}$$

Eq. 5.1

In other words, ANDFR is nothing but the fluence-weighted kerma factor of a neutron spectrum. This criterion represents the hardest neutron spectrum that can be used for BNCT without excessive damage to the skin.

2. The flux of useful neutrons (i.e. those with energy greater than ~1 eV) emerging from the moderator assembly should be large (> 1.0
x 10^8 neutrons/cm²-sec), so that a single-session treatment can be achieved in less than a few hours. This criteria is based on the suggested neutron fluence of 5 x 10^{12} n/cm² at the tumor location for a single-session treatment.

3. The absorbed dose due to the incident gamma rays should be significantly lower than that due to the gamma rays which are induced in phantom.

5.3 Design Methods

The design of the moderator assembly is based solely upon Monte Carlo code results. The design process includes three steps: (1) Preliminary design (2) Parametric study of the preliminary design, and (3) Final design synthesis. The details of each step are discussed in the following subsections.

5.3.1 Preliminary Design

The preliminary design of the moderator assembly consists of a cylindrical moderator surrounded by a reflector. Materials and thicknesses of the moderator and the reflector have been studied. The configuration of the preliminary design is illustrated in Figure 5-1. In addition to the moderator and reflector, a thin layer of $^6$Li is placed in the front window of the irradiation port to filter out thermal neutrons. The selection of materials for the moderator and reflector are discussed below.

5.3.2 Moderator Material Selection

The selection of the moderator materials is based on neutron slowing-down theory and has been confirmed by Monte Carlo,
Figure 5-1. The preliminary design configuration of the neutron moderator assembly
calculations. For our moderator assembly, the properties which a good moderator material should possess are (1) large $\Sigma_s$, the neutron scattering cross section, (2) moderate $\xi$, the average increase in the lethargy of a neutron per collision, and (3) small $\Sigma_a(n,\gamma)$, the neutron cross section for radiative capture. For a moderator possessing properties 1 and 2, source neutrons are slowed down in a short distance, and yet the spectrum of neutrons emerging from the moderator is not spread too much. Closer examination of property one reveals that both the magnitude of $\Sigma_s$ and its energy dependence are important. The ideal energy dependence of $\Sigma_s$ is for $\Sigma_s$ to be large for energies above 1 keV and zero for energies below 1 keV. Then the moderator rapidly slows neutrons into the desired energy range (i.e. below 1 keV), and then keeps them there. The importance of property 3 is obvious, since neutrons which are absorbed in the moderator cannot contribute to therapy. Also, neutrons which are absorbed in radiative capture events, create a photon flux within the moderator which may contaminate the neutron field entering the patient.

In reality, there is no material possessing all three desired properties. Figure 5-2 shows the energy dependencies of $\Sigma_s$ for H$_2$O, D$_2$O, BeO, Be, and C, the moderators commonly used in nuclear reactors; and for alumina(Al$_2$O$_3$), since it was suggested as a good neutron spectrum shifter for a reactor epithermal neutron beam$^{15}$. Figure 5-3 shows the energy dependencies of $\xi$ for these same materials. As it is shown, $\xi$ does not vary as much with neutron energy as does $\Sigma_s$. Table 5-1 provides values of $\Sigma_a(n,\gamma)$ for thermal neutrons for the moderating materials.
Figure 5-2. The energy dependencies of the macroscopic neutron scattering cross section for various neutron moderator materials.
Figure 5-3. The energy dependencies of the average increase of lethargy per neutron collision for various neutron moderator materials.
Table 5-1. $\Sigma_a(n,\gamma)$ for thermal neutrons for various moderating materials.

<table>
<thead>
<tr>
<th>Materials</th>
<th>$H_2O$</th>
<th>$D_2O$</th>
<th>Be</th>
<th>BeO</th>
<th>C</th>
<th>$Al_2O_3$</th>
<th>Al/D2O*</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\Sigma_a(n,\gamma)$</td>
<td>2.2(-2)</td>
<td>3.4(-5)</td>
<td>1.2(-3)</td>
<td>7.2(-4)</td>
<td>3.4(-4)</td>
<td>1.8(-2)</td>
<td>2.2(-2)</td>
</tr>
</tbody>
</table>

Synthesizing the data presented in Figs 5-2 and 5-3, and Table 5, we conclude that, $H_2O$ has large $\Sigma_s$, which is good, but it also has very large $\xi$ which results in poorly shaped neutron spectra. In addition, $H_2O$ has relatively large $\Sigma_a(n,\gamma)$, and generates 2.2 MeV gamma rays, by the absorption of neutrons. $D_2O$ has smaller $\xi$, and therefore produces neutron spectra with better shapes than those produced by $H_2O$. Also, $D_2O$ has an advantage relative to materials with higher atomic number, which is not apparent from Figs 5-2 and 5-3, and Table 5-1. This advantage is that the average cosine of the neutron scattering angle is larger for $D_2O$ than for higher atomic number materials, thus resulting in better transmission through the $D_2O$ than would result if the scattering were isotropic. Finally, the fact that $D_2O$ is a liquid at room temperature may be an advantage in clinical application, since one can simply adjust the thickness of $D_2O$ to make the neutron spectrum harder or softer to fit the patient's needs.

Beryllium and graphite have moderate $\xi$ and large $\Sigma_s$. However, for beryllium and graphite, the values of $\Sigma_s$ are smaller at high
neutron energies, and larger at low neutron energies. As it has been discussed, this kind of energy dependence of $\Sigma_s$ is not desirable. Beryllia not only has large $\Sigma_s$, comparable to that for beryllium, but also the energy dependence of $\Sigma_s$ for beryllia is more ideal than for beryllium or graphite.

The energy dependence of $\Sigma_s$ is also good for alumina. However, its $\xi$ is too small, and the distance between the lithium target and patient is consequently large, thereby reducing the neutron flux to the patient. Besides, aluminum generates hard gamma-rays as a consequence of radiative capture of thermal neutrons. Although Al$_2$O$_3$ alone is not a good moderating material, a mixture of D$_2$O and Aluminum may have moderating properties comparable to those of BeO. Oka et al $^{38}$ have shown that a mixture of aluminum and heavy water, with a volume ratio of 85/15, is the best neutron spectrum shifter for a reactor beam.

The qualitative discussions given above were confirmed by a set of Monte Carlo calculations. On this basis, D$_2$O, BeO, and mixtures of Aluminum and D$_2$O were selected as moderating materials for further evaluation.

5.3.3 Reflector Material Selection

Various materials have been evaluated for the reflector region of the moderator assembly. A good reflector material for the moderator assembly should have reasonably small $\xi$, but large $\Sigma_s$, so that it reflects epithermal neutrons in a short distance, without greatly degrading their energy. Several materials possess the above properties and they are lead, bismuth, alumina, and graphite.
Monte Carlo calculations show that lead and bismuth have values of $\xi$ which are too small, and consequently many of the reflected neutrons have energies which are too high for BNCT as they leave the moderator assembly. Graphite has a value of $\xi$ which is too large, and consequently many of the reflected neutrons are thermalized in the moderator assembly, and therefore are not appropriate for BNCT. Alumina reflects neutrons so that they leave the moderator assembly with energies which are most suitable for BNCT. However, an undesirable property of alumina is its large cross section for radiative capture of thermal neutrons and the subsequent emission of hard gamma-rays. The effects of this property, however, can be minimized by placing small amount of $^6$Li at the interface between the moderator and the reflector. Then thermal neutrons, which are generated in the moderator, are captured by $^6$Li at the interface, before they can be absorbed by aluminum in the reflector. Alumina was therefore chosen to be the reflector material for the moderator assembly.

5.4 Parametric Study of the Preliminary Design

The design parameters subjected to study are (1) The optimum thickness for a moderator which is made of D$_2$O or BeO. (2) The optimum volume ratio of Aluminum to D$_2$O in the moderator, if it is made of such a mixture. (3) The optimum thickness for the alumina reflector. (4) The optimum thickness for the $^6$Li layer at the interface between the moderator and the reflector.(5) The optimum thickness for the $^6$Li layer at the front window of the irradiation port.
The design parameters were evaluated based on the suitability for BNCT of the neutron flux at the irradiation point shown in Figure 5-1. The diameter of the moderator was fixed at 25 cm, the thickness of the alumina reflector was fixed at 30 cm, and the $^6$Li layer at the front window was fixed at 0.05 g/cm$^2$. In addition to the components of the moderator assembly which are shown in Fig. 5-1, the preliminary design which was the basis of the parametric studies included a 0.1 g/cm$^2$ thick $^6$Li layer which separates the moderator and the reflector.

The calculated neutron spectra is presented in Table 5-2, for seven energy groups for various thicknesses of BeO and D$_2$O as moderator, and for various mixtures of Aluminum and D$_2$O as moderator, with the mixture thickness fixed at 25 cm. Groups 5 and 6 cover an energy region from 0.88 eV to 1.58 keV, and therefore are the most desired groups for BNCT. Other groups of neutrons with energies higher than that of group 5 are also useful for BNCT, but are less desirable because their kerma factors are larger. In addition, Table 5-2 introduces a new quantity, $\phi_u$, the fluence of neutrons with energies greater than 0.88 eV. Since all these neutrons are useful for BNCT, $\phi_u$ is called the useful neutron fluence, and it will be used to optimize the moderator assembly design in the later part of this section.

A number of parameters have been previously used to judge the suitability of a neutron beam for BNCT. Two frequently used parameters are the therapeutic gain (TG) (the ratio of dose equivalent to the tumor relative to the maximum dose equivalent to normal tissue
Table 5-2. The calculated neutron fluence spectra, neutrons/cm$^2$ per source neutron, and absorbed dose, cGy per source neutron generated by a 2.5 MeV proton beam impinging upon a lithium-7 target imbedded in the moderator assembly described by Figure 5-1.

<table>
<thead>
<tr>
<th>Neutron Fluences</th>
<th>$\phi_1$ (a)</th>
<th>$\phi_2$</th>
<th>$\phi_3$</th>
<th>$\phi_4$</th>
<th>$\phi_5$</th>
<th>$\phi_6$</th>
<th>$\phi_7$</th>
<th>$\phi_u$ (b)</th>
<th>$D_n$ (c)</th>
<th>$D_8$ (c)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>D$_2$O thickness</strong> (cm)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>15.0</td>
<td>1.46E-5</td>
<td>2.28E-5</td>
<td>5.04E-5</td>
<td>8.14E-5</td>
<td>5.72E-5</td>
<td>6.42E-5</td>
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<td>2.91E-4</td>
<td>5.46E-14</td>
<td>1.54E-15</td>
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<td>17.5</td>
<td>5.88E-5</td>
<td>1.07E-5</td>
<td>2.73E-5</td>
<td>5.21E-5</td>
<td>4.30E-5</td>
<td>5.76E-5</td>
<td>2.57E-6</td>
<td>1.97E-4</td>
<td>2.62E-14</td>
<td>1.55E-15</td>
</tr>
<tr>
<td>20.0</td>
<td>2.92E-6</td>
<td>5.67E-6</td>
<td>1.73E-5</td>
<td>3.66E-5</td>
<td>2.91E-5</td>
<td>4.29E-5</td>
<td>2.02E-6</td>
<td>1.35E-4</td>
<td>1.48E-14</td>
<td>1.51E-15</td>
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<td>22.5</td>
<td>1.17E-6</td>
<td>2.76E-6</td>
<td>8.70E-6</td>
<td>2.25E-5</td>
<td>2.26E-5</td>
<td>3.30E-5</td>
<td>1.96E-6</td>
<td>9.07E-5</td>
<td>7.28E-14</td>
<td>1.46E-15</td>
</tr>
<tr>
<td>25.0</td>
<td>8.31E-7</td>
<td>1.58E-6</td>
<td>5.12E-6</td>
<td>1.49E-5</td>
<td>1.49E-5</td>
<td>2.36E-5</td>
<td>1.81E-6</td>
<td>6.09E-5</td>
<td>4.55E-15</td>
<td>1.35E-15</td>
</tr>
<tr>
<td><strong>BeO thickness</strong> (cm)</td>
<td></td>
<td></td>
<td></td>
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<td></td>
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<td></td>
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<td></td>
</tr>
<tr>
<td>15.0</td>
<td>4.79E-6</td>
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(a) $\phi_i$ is the energy-dependent neutron fluence for group $i$, in neutrons/cm$^2$ per source neutron, where $i = 1$ for 183.2 keV < $E$ < 800 keV
2 for 67.4 keV < $E$ < 183.2 keV
3 for 15.0 keV < $E$ < 67.4 keV
4 for 1.58 keV < $E$ < 15.0 keV
5 for 214.5 eV < $E$ < 1.58 keV
6 for 0.88 eV < $E$ < 214.5 eV
7 for $10^{-5}$ eV < $E$ < 0.88 eV

(b) $\phi_u$ = useful neutron fluence = $\sum_{i=1}^{6} \phi_i$

(c) $D_n$ and $D_g$ represent the absorbed neutron and gamma dose, respectively.
and the maximum usable depth (MUD, the depth in tissue where the absorbed dose to the tumor equals the absorbed dose which can be tolerated by normal tissue for specified $^{10}$B concentrations in tumor and blood). A strength of these parameters is their usefulness in radiation treatment planning. A disadvantage of these parameters is that it is difficult to use them for beam comparisons, since they depend strongly upon the phantom, especially the boron concentrations of the tumor and normal tissue. In addition, it is difficult to obtain good statistics with a Monte Carlo code for dose calculations inside a phantom, because the neutron flux estimator has to be physically located inside the phantom where not many events occur. Therefore, TG and MUD are not suitable for a design study of the moderator assembly.

For these reasons, we have defined a new parameter with which to assess the suitability of a beam for BNCT. This parameter is the ratio of useful neutron fluence to entrance dose equivalent (RUFTED). A strength of this parameter is that it depends upon the neutron beam alone and not upon the phantom. The neutron fluences are estimated at the beam port, in the absence of a phantom. Therefore, one can take advantage of using the next event estimator to obtain neutron fluxes with very good statistics. The RUFTED is defined as:

$$\text{RUFTED} = \frac{\Phi_u}{K_n \ast \text{RBE} + K_g}$$

Eq. 5.2

where $\Phi_u$ is the useful neutron fluence at the irradiation point, and $K_n$ and $K_g$ are the incident neutron and gamma-ray kermas in tissue at the irradiation point. In other words, the RUFTED is the fluence of
neutrons which are useful for BNCT per unit of dose equivalent. In this study, a widely accepted RBE value of 2.0 was used for neutrons. The cut-off, between neutrons which are useful for BNCT and neutrons which are not, is somewhat arbitrary. It has been chosen to be 1 eV, because neutrons with energies less than 1 eV have a much shorter penetration depth in tissue than neutrons with energies greater than 1 eV.

Larger values of the RUFTED correspond to more ideal beams. For example, for a beam with no gamma-rays, the RUFTED is proportional to the reciprocal of the neutron flux to kerma rate conversion factor for neutron energies greater than 1 eV, and zero otherwise. In this case, the RUFTED is a maximum for the neutron energy for which the kerma factor is a minimum, and is thus consistent with conventional wisdom regarding the optimum neutron energy for BNCT. Figure 5-4 is a graph of RUFTED versus neutron energy for monoenergetic neutron beams with no gamma contamination. This figure is presented to provide the reader with a basis on which to judge the goodness of the RUFTED for our accelerator-based neutron source. Physically speaking, larger RUFTED corresponds to larger TG and larger MUD.

The optimization of the design parameters of the moderator assembly was based upon maximizing the RUFTED, and the useful neutron flux, $\phi_u$, per unit proton current. Larger $\phi_u$ per unit proton current means shorter treatment times. Figure 5-5 and 5-6 show the relationships of the RUFTED and $\phi_u$ versus moderator thickness for D$_2$O and BeO respectively. As they are shown, the RUFTED increases
Figure 5-4. The relationship of RUFTED versus neutron energy for a monoenergetic neutron beam with no gamma-ray contaminations
Figure 5-5. The relationship of RUFTED and $\phi_u$ versus the moderator (BeO) thickness at the irradiation point, which is 5 cm from the moderator assembly on the beam centerline.
Figure 5-6. The relationship of RUFTED and $\phi_u$ versus the moderator ($D_2O$) thickness at the irradiation point, which is 5 cm from the moderator assembly on the beam centerline.
with increasing $D_2O$ and BeO thickness, for moderator thicknesses less than approximately 24 cm. Further increases in moderator thickness reduce the value of the RUFTED, because the useful neutron flux decreases while the accompanying gamma-ray dose remains about the same. Figure 5-7 shows the relationships of the RUFTED and $\phi_u$ versus the volume ratio of $D_2O$ to Aluminum, for a 25 cm thick moderator made of such a mixture. As it is shown, the RUFTED is approximately constant, for volume ratios of $D_2O$ to Aluminum less than 30/70. Further increases in volume ratio slightly reduce the value of RUFTED. This is because a softening of the neutron spectrum, due to a higher percentage of $D_2O$ in the mixture, is accompanied by an increase in generation of thermal neutrons, and hence more hard gamma rays from $^{27}Al(n,\gamma)^{28}Al$ reactions.

Hence, the design problem, which is faced in determining the type and optimum thickness of a moderator, is that the RUFTED is improved at the price of decreases in the neutron flux. Generally speaking, among the three materials $D_2O$ provides the highest useful neutron flux, but low RUFTEDs. The mixtures of Aluminum and $D_2O$ are not superior to pure $D_2O$. BeO gives reasonably good values for both RUFTEDs and useful neutron fluxes. Among the various combination of moderators and moderator thicknesses, 20 cm of BeO has the best combination of RUFTED and useful neutron flux. However, it should be noted from Figs. 5-5, 5-6 and 5-7 that 22.5 cm of $D_2O$ and 25 cm of a mixture of $D_2O$ and Aluminum with 30/70 volume ratio, provide comparable values of RUFTED and useful neutron flux.

To obtain the optimum thickness for the alumina reflector,
Figure 5-7. The relationship of RUFTED and $\phi_u$ versus the moderator volume ratio ($D_2O/Al_2O_3$) at the irradiation point, which is 5 cm from the moderator assembly on the beam centerline. The thickness of the moderator was 25 cm.
calculations were performed using various thicknesses of alumina reflector, for the same preliminary moderator assembly design which was the basis of the moderator parametric studies described above. The moderator was assumed to be the optimum moderator type and thickness described above, BeO, 20 cm thick. Figure 5-8 shows the calculational results of the relationships of RUFTED and $\phi_u$ versus the thickness of alumina reflector. As shown in Figure 5-8, both RUFTED and $\phi_u$ increase with increasing alumina reflector thickness for thicknesses up to 20 cm. Increases in alumina reflector thickness from 20 to 30 cm cause $\phi_u$ to increase slightly to its asymptotic value, but decrease RUFTED. This is because the neutron reflection efficiency approaches that of an infinite reflector at 30 cm, while increases of reflector thickness beyond 20 cm generate more thermal neutrons in the reflector. This increases the gamma-ray dose from $^{27}$Al(n,$\gamma$$^{28}$Al reactions without increasing $\phi_u$ significantly, and thus decreases RUFTED. On the basis of maximizing $\phi_u$ without greatly degrading RUFTED, we judge the optimum alumina reflector thickness to be about 30 cm.

To find the optimum loading of $^6$Li at the interface between the moderator and reflector regions, calculations were performed using various loadings of $^6$Li, for the same preliminary moderator assembly design which was the basis of the moderator and reflector parametric studies described above. That is, BeO, 20 cm thick, was again used as the moderator, and the reflector was assumed to be alumina, 30 cm thick. Figure 5-9 shows the calculational results of the relationships of RUFTED and $\phi_u$ versus $^6$Li loadings. As it is shown, 0.05 g/cm$^2$ of $^6$Li
Figure 5-8. The relationship of RUFTED and $\phi_u$ versus the reflector ($\text{Al}_2\text{O}_3$) thickness at the irradiation point, which is 5 cm from the moderator assembly on the beam centerline.
Figure 5-9. The relationship of RUFTED and $\phi_u$ versus the $^6$Li loading at the interface between the moderator and the reflector. The RUFTED and $\phi_u$ were estimated at the irradiation point, which is 5 cm from the moderator assembly on the beam centerline.
gives the largest RUFTED, and at the same time gives a good value for \( \phi_u \). Increasing the \( ^6\text{Li} \) loading above 0.05 g/cm\(^2\) decreases both RUFTED and \( \phi_u \), because too many useful epithermal neutrons are absorbed. On the basis of maximizing RUFTED, without greatly degrading \( \phi_u \), we judge the optimum loading of \( ^6\text{Li} \) at the interface between the moderator and reflector regions to be 0.05 g/cm\(^2\).

To find the optimum loading of \( ^6\text{Li} \) at the front window of the moderator assembly irradiation port, calculations were performed using various \( ^6\text{Li} \) loadings at the front window, for the same preliminary moderator assembly design which was the basis of the moderator, reflector, and \( ^6\text{Li} \) interface thickness parametric studies described above. That is, BeO, 20 cm thick, was used as the moderator, alumina, 30 cm thick, was used as the reflector, and \( ^6\text{Li} \), 0.1 g/cm\(^2\) thick, separated the moderator from the reflector regions. Figure 5-10 shows the calculational results of the relationships of RUFTED and \( \phi_u \) versus \( ^6\text{Li} \) loadings. As it is shown, 0.01 g/cm\(^2\) of \( ^6\text{Li} \) gives the largest RUFTED and at the same time gives a good value of \( \phi_u \). Increasing the \( ^6\text{Li} \) loading above 0.01 g/cm\(^2\) filters too many of the useful epithermal neutrons from the beam, and therefore reduces both \( \phi_u \) and RUFTED. It should be noted that, even with no \( ^6\text{Li} \) at the front window, the RUFTED is reasonably good. This simply means that the contamination by thermal neutrons at the irradiation point is not serious. However, for the sake of completeness, the optimum \( ^6\text{Li} \) loading at the front window is judged to be 0.01 g/cm\(^2\) on the basis of maximizing RUFTED.
Figure 5-10. The relationship of RUFTED and $\phi_u$ versus the $^6$Li loading at the front end of the moderator assembly. The RUFTED and $\phi_u$ were estimated at the irradiation point, which is 5 cm from the moderator assembly on the beam centerline.
5.5 Final Design Synthesis

The final design of the moderator assembly is the synthesis of the optimum values of the design parameters presented in the previous section, and it is shown in Figure 5-11. As it is shown, the final design consists of a cylindrical moderator made of BeO, which is 25 cm in diameter and 20 cm in height, surrounded by a 30 cm thick alumina reflector. A loading of 0.05 g/cm$^2$ of $^6$Li is placed at the interface between the moderator and reflector regions. A loading of 0.01 g/cm$^2$ of $^6$Li is placed at the front window of the irradiation port to reduce the thermal neutron contamination. In addition, a layer of D$_2$O, 10 cm thick, surrounds the alumina reflector, except at the irradiation port, and the D$_2$O is in turn surrounded by an outer skin of $^6$LiF. These two layers function as a neutron shield to reduce the patient's whole-body dose. The escaping neutrons are thermalized by the D$_2$O and are then captured by $^6$Li. Conventional neutron shielding, such as borated polyethylene, is not appropriate for our moderator assembly because both hydrogen and boron produce secondary gamma rays by capturing thermal neutrons.
Figure 5-11. The configuration of the optimum neutron moderator assembly for the proposed low-energy proton accelerator neutron irradiation facility.
CHAPTER VI

THE PERFORMANCE EVALUATION

The neutron spectrum at the irradiation point created by the optimum moderator assembly described in the previous chapter is presented in Figure 6-1. For a 30 mA proton beam producing $2.76 \times 10^{12}$ neutrons/sec, the useful neutron flux evaluated at the irradiation point is $1.0 \times 10^9$ neutrons/cm$^2$-sec. The corresponding total absorbed dose rates for neutron and gamma-rays are 2.85 cGy/min and 2.5 cGy/min, respectively. The gamma-ray dose can be much reduced by adding 1 or 2 inches of bismuth to the front end of the moderator assembly. This arrangement, of course, will reduce the neutron flux at the irradiation point, but only slightly. It can be derived from Fig. 6-1, that 89% of the useful neutrons have energies less than 10 keV. The absorbed neutron dose to fluence ratio is $4.7 \times 10^{-11}$ cGy-cm$^2$, which is equivalent to the kerma factor of a 5 keV neutron. Since the biological damage caused by neutrons with a few-keV is still being investigated, it is too early to say if this neutron field is good enough for BNCT. However, according to the latest Harwell's paper$^{11}$, a 5 keV beam may be marginal.

The performance of the neutron field was further evaluated based upon the absorbed dose and dose equivalent distributions in a phantom. The phantom represents a human head. Its composition is assumed to be (C$_5$H$_{40}$O$_{18}$N)$_n$ and its shape is a sphere, 9 cm in radius.
Figure 6-1. The neutron spectrum at the irradiation point of the optimum moderator assembly.
The center of the spherical phantom is located 12 cm from the $^6$LiF front window on the centerline of the moderator assembly. Figure 6-2 shows the relationship for four energy groups of neutron flux versus depth within the phantom on the beam centerline. In order to achieve reasonable statistics in the MORSE calculations, the original 12 groups in the epithermal region, as appear in the BUGLE-80 cross section library, have been collapsed into group 2 and 3 in Figure 6-2. Figure 6-2 clearly shows that the incident neutron fluence is dominated by group 2, which is between 0.88 eV and 1.58 keV; and at depth beyond 1 cm, the group 4 ($E_n < 0.88$eV) starts to dominate.

Figure 6-3 shows the depth distributions in a phantom for various components of the absorbed dose. The curve of the absorbed dose due to $^{14}$N(n,p)$^{14}$C reactions, shown in Figure 6-4, has been corrected from the absorbed dose calculated based on the formula of $(C_5H_{40}O_{18}N)_n$ by reducing the absorbed dose by one-half, since the brain nitrogen concentration is one half of that described by the formula. The corresponding depth distributions of dose equivalent are shown in Figure 6-4. In these figures, the $^{10}$B concentration was assumed to be 30 $\mu$g/g in the tumor and 10 $\mu$g/g in normal tissue, consistent with the concentrations assumed by others for studies done for reactor beams. Also, in order to compare the results of this study with other studies done for reactor beams, the RBE values used in Figure 6-4 are, 2.0 for neutrons and 2.5 for $^{10}$B(n,α)$^7$Li reaction products, consistent with those reported by others. As it is shown in Figure 6-4, more than 50% of the entrance absorbed dose to normal tissue is contributed by incident neutrons, mainly those with energies...
\( \phi_i \) is the \( i \)-th grouped neutron fluence, where
- \( i = 1 \) for \( 67.4 \text{ keV} < E_n < 800 \text{ keV} \)
- \( i = 2 \) for \( 1.58 \text{ keV} < E_n < 67.4 \text{ keV} \)
- \( i = 3 \) for \( 0.88 \text{ eV} < E_n < 1.58 \text{ keV} \)
- \( i = 4 \) for \( 10^{-4} \text{ eV} < E_n < 0.88 \text{ eV} \)

Figure 6-2. The flux-depth distributions in a spherical phantom on the beam centerline calculated by the MORSE code for the optimum moderator assembly.
A: Total absorbed dose to tumor with 30 µg of $^{10}$B/g tissue
B: Total absorbed dose to normal tissue with 10 µg of $^{10}$B/g tissue
C: Absorbed dose due to the incident gamma-rays, and gamma-rays induced from $^1$H(n,$\gamma$)$^2$H reaction
D: Absorbed dose due to $^{14}$N(n,p)$^{14}$C reactions
E: Absorbed dose due to $^1$H(n,n')$^1$H reactions

Figure 6-3. The absorbed dose-depth distributions in a spherical phantom on the beam centerline calculated by the MORSE code for the optimum moderator assembly.
A: Total dose equivalent to tumor with 30 μg of $^{10}$B/g tissue
B: Total dose equivalent to normal tissue with 10 μg of $^{10}$B/g tissue
C: Dose equivalent due to the incident gamma-rays, and gamma-rays induced from $^1$H(n,$\gamma$)$^2$H reaction
D: Dose equivalent due to $^{14}$N(n,p)$^{14}$C reactions
E: Dose equivalent due to $^1$H(n,n)$^1$H reactions

Figure 6-4. The dose equivalent-depth distributions in a spherical phantom on the beam centerline calculated by the MORSE code for the optimum moderator assembly
higher than a few keV. However, the entrance absorbed dose to
normal tissue is less than two-thirds of the peak absorbed dose to
normal tissue, which occurs at a depth of 3.5 cm; and therefore the
entrance absorbed dose is tolerable. According to Oka et al\textsuperscript{38}, "the
MUD occurs where the $30 \mu g \text{^{10}B/g}$ dose curve falls 0.4 times the
maximum-dose level of the $10 \mu g \text{^{10}B/g}$ curve." It is worth noting that
Oka’s definition of MUD was advanced from that originated by
Zamenhof\textsuperscript{17}, to take clinical results into consideration. These results
are that it requires only 20 Gy to kill tumor cells at a $\text{^{10}B}$
concentration of 30 $\mu g/g$, whereas, normal tissue can tolerate up to 50
Gy at a $\text{^{10}B}$ concentration of 10 $\mu g/g$. We used Oka’s definition of MUD
in our study, because it is more specific than Zamenhof’s. From Figure
6-4, the MUD is found to be about 7.5 cm. Figure 6-5 shows the
relationship between therapeutic gain and tumor depth. In this Figure,
the $\text{^{10}B}$ concentrations in the tumor and normal tissue were again
assumed to be 30 $\mu g/g$ and the 10 $\mu g/g$, respectively

According to Figure 6-3, the maximum absorbed dose to tumor
is $3.71 \times 10^{-14}$ cGy/source neutron for a $\text{^{10}B}$ concentration of 30 $\mu g/g$
tumor. Therefore, for a 30 mA 2.5 MeV proton beam, the maximum
absorbed dose rate is 1.02 cGy/sec, or 61.5 cGy/min. For a single-
session irradiation of 20 Gy to the tumor, the treatment time can be as
short as 33 minutes. This treatment time is quite reasonable.
Figure 6-5. The distribution of therapeutic gains in a spherical phantom on the beam centerline calculated by the MORSE code for the optimum moderator assembly.
CHAPTER VII

CONCLUSIONS AND FUTURE STUDIES

7.1 Conclusions

The major conclusions for this study are:

(1) A low-energy proton accelerator neutron irradiation facility (LPANIF) for BNCT has been described. The neutrons are generated by 2.5 MeV protons impinging upon a lithium-7 target. The proposed proton accelerator is a radio frequency quadrupole (RFQ), which delivers 30 mA of proton current.

(2) The neutron fields generated by 2.5 MeV protons impinging upon a lithium target, with or without a moderator assembly, have been studied both by calculations and by experiments. The experimentally measured quantities include the neutron spectra of the irradiation field, and the absorbed gamma-ray and boron doses in a phantom which is exposed in the field. The fast neutron dose could not be measured, because the neutron flux of the experimental field was too low. The calculational results, in general, agree reasonably well with the experimentally measured results, except for the neutron spectra above 10 keV, where the measured results are lower by a factor of 2 to 3 than the calculated results. This discrepancy is thought to be caused by the uncertainty in the $H_2$ gas pressure of the proton...
recoil proportional counter, which was used to measure the neutron spectra.

(3) Efforts were made to design the neutron moderator assembly, which produces the most suitable neutron field for BNCT. The design methods were based solely on Monte Carlo code results. The moderator assembly was optimized for a neutron field with a good energy spectrum and a high flux. It was found that a good neutron spectrum can be obtained, at the price of a lower flux. Beryllia (BeO) was found to be the best moderating material for obtaining an epithermal neutron field for BNCT; because, among other attributes, it has large neutron scattering cross sections in the several-hundred-keV range, to knock down fast neutrons, and an appropriate mass number to slow the fast neutrons down to the desired energy range (which is between 1 eV and 1 keV). The optimized moderator assembly has been illustrated in Figure 43. It basically consists of a cylindrical BeO moderator, which is 25 cm in diameter and 22.5 cm in height, and is surrounded by 30 cm of alumina (Al₂O₃) reflector.

(4) The neutron field produced by the optimum moderator assembly has a neutron absorbed dose to fluence ratio of $4.7 \times 10^{-11}$ cGy-cm², which is equal to the kerma factor of a 5 keV neutron. The total useful neutron flux (meaning $\phi$ for $E_n>1$eV) evaluated at the irradiation point is $1.0 \times 10^9$ n/cm²-sec for a 30 mA 2.5 MeV proton beam. Also, 89% of the total neutron flux is from neutrons with energies less than 10 keV, and 84% of the total total neutron flux is from neutrons with energies less than 1 keV. Both the neutron
spectrum and the flux are comparable to those of a mega-watt range nuclear reactor. The boron dose (or the thermal neutron flux) peaks about 3.5 cm from the surface of a 9 cm diameter spherical phantom. Because of the large neutron field, the boron dose (or the thermal neutron flux) is slowly attenuated in the phantom. This, therefore provides a good maximum usable depth of about 7.5 cm. The irradiation time for a single-session treatment of BNCT is estimated to be as short as 33 minutes.

7.2 Suggestions for Future Studies

This dissertation is basically a proof-of-principle study for the proposed low-energy proton accelerator neutron source for BNCT. Both the calculational methods and the measurement techniques developed for this study can still be improved to obtain better results. More studies are still needed, before the proposed hospital-based irradiation facility can be built. The suggestions for future studies are summarized below.

(1) To improve the inherent errors in the multigroup calculations performed by MORSE, one may have to use the MCNP\textsuperscript{41} code. MCNP is a Monte Carlo neutron and gamma-ray transport code developed at Los Alamos National Laboratory. It uses a huge cross section library, and therefore allows continuous energy degradation in a neutron slowing down process. Compared to MORSE, an MCNP run requires more computer memory and more computation time.
(2) To improve the calculational statistics in estimating the neutron fluence in a phantom, the importance functions should be generated and used to bias the neutron transport process. The importance functions may be obtained by running the code in the adjoint mode. The adjoint fluence usually provides good importance functions for the forward calculation. The adjoint calculation for this study, however, may be as difficult as the forward one, because the transport problem is basically a point-source-to-point-detector deep-penetration problem. Other biasing schemes such as the "exponential transform" and "geometry splitting" should also be tried.

(3) In the use of the proton recoil proportional counter (PRPC) to measure the neutron spectra, it is better to equip the PRPC with a gas-filling system so that measurements with several gas pressure can be made, and so that the gas pressure can be determined accurately as the detector is filled. Therefore, the use of a gas-filling system would not only improve the discrimination against gamma rays, but also would reduce the uncertainty in the response functions of the PRPC.

(4) To reduce the uncertainty in the response functions of the boron-shell neutron spectrometer, the $^{10}$B loading of each boron shell should be experimentally determined by shooting a Sc-filtered monoenergetic 2 keV beam through each shell. This can be done at the high flux reactor of Brookhaven National Laboratory.

(5) The LET distribution of the neutron field should be measured, so that the biological effectiveness of the field can be better evaluated.
Because of the low sensitivities of LET detectors, the LET measurement can only be best done at high-current accelerator.

(6) To ultimately determine if the neutron field produced by the proposed LPANIF is good enough for BNCT, cell irradiation experiment must be performed. This, again, can only be done with a high-current (~ 1 mA) accelerator, so that large neutron doses can be delivered in a relatively short period of time.

(7) Both the calculational and the experimental studies must be performed to determine how to remove the 75 kW of beam heat from the lithium target. The heat-pipe concept which was suggested by Blue et al. looks appealing, and certainly deserves more investigation.
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129


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40. N. M. Schaeffer, Reactor Shielding for Nuclear Engineers, USAEC Information Services (1973) p 614.

Appendix A  FORTRAN listing of the NSRCE
THIS IS PROGRAM "NSRCE" WHICH CALCULATES THE NEUTRON YIELD OF
A 2.5 MEV IMPINGING UPON A LITHIUM-7 TARGET. THE NEUTRON YIELD
IS EXPRESSED BY A MATRIX, AND EACH ELEMENT OF WHICH IS THE
DIFFERENTIAL NEUTRON YIELD FOR A SPECIFIC EMISSION ANGLE AND
ENERGY

```
DIMENSION A(5),XSC(24,37),EN(24,37),EP(24),ANG1(37),
&ANG2(12),ENG(25),XSC1(37),EN1(37),XSC2(181),EN2(181),
&XSC3(180),EN3(180),SN(180),SNG(12,25)
AREA=10.0
PI=3.1415926
CV=180.0/PI
CURENT=0.01
PRFLUX=CURENT*6.242197E*18/AREA
EPKEV=2500.0
DO 20 J=1,25
DO 20 1=1,12
SNG(I,J)=0.
20 CONTINUE
T5RCE=0.

READ DIFFERENTIAL XSEC & NEUTRON ENG. FROM NUCLEAR DATA TABLE
BY HORST LISKIEN & ARNO PAULSEN, 15, 57-84(1975)
READ(5,699)( (XSC(I,J),I=1,24),J=1,37)
READ(5,699)( (EN(I,J),I=1,24),J=1,37)

CALCULATE THE STOPPING POWER DE/DX IN KEV/CM

30 EPMEV=EPKEV/1000.
BETA=SQRT(2.0*EPMEV/(938.0*EPMEV))
A(I)=-0.5831
A(2)=0.562
A(3)=-0.1183
A(4)=0.009298
A(5)=0.0002498
AA=0.00153
B=2.147E4
BTSQ=BETA**2.0
SUM=0.
DO 40 I=1,5
SUM=SUM+A(I)*((ALOG(EPMEV)**(I-1))
40 CONTINUE
T1=ALOG(B*BTSQ/(1.0-BTSQ))
S1=AA*(T1-BTSQ-SUM)/BTSQ

CONVERT DE/DX FROM EV/10E-15(ATOMS/CM**2) TO KEV/CM

S2=S1*4.5924E-4
TH=1.0/S2
AN=4.5924E-22*TH*AREA

DEFINE THE CORRESPONDING PROTON ENERGIES GIVEN IN THE XSC TABLE
EP(1)=2500.0
DO 50 I=2,12
EP(I)=EP(I)-50.0*(I-1)
50 CONTINUE
EP(13)=1.921
```
EP(14)=1.909
EP(15)=1.902
EP(16)=1.896
EP(17)=1.892
EP(18)=1.890
EP(19)=1.887
EP(20)=1.886
EP(21)=1.885
EP(22)=1.884
EP(23)=1.882
EP(24)=1.881

C DEFINE THE CORRESPONDING ANGLES GIVEN IN THE XSC TABLE

ANG1(1)=0.
DO 100 J=2,37
ANG1(J)=ANG1(1)·5.0*(J-1)
100 CONTINUE
C
C DEFINE THE CORRESPONDING ANGULAR INTERVALS SPECIFIED BY S12 IN THE
C NEUTRON TRANSPORT CODE

ANG2(1)=21.0
ANG2(2)=43.0
ANG2(3)=59.0
ANG2(4)=75.0
ANG2(5)=85.0
ANG2(6)=90.0
ANG2(7)=95.0
ANG2(8)=105.0
ANG2(9)=121.0
ANG2(10)=137.0
ANG2(11)=159.0
ANG2(12)=180.0

C DEFINE THE CORRESPONDING ENERGY GROUPS IN THE NEUTRON TRANSPORT
C CODE

ENG(1)=820.85
ENG(2)=742.74
ENG(3)=608.1
ENG(4)=497.87
ENG(5)=368.83
ENG(6)=297.2
ENG(7)=183.16
ENG(8)=111.09
ENG(9)=67.397
ENG(10)=40.868
ENG(11)=31.828
ENG(12)=26.058
ENG(13)=24.176
ENG(14)=21.875
ENG(15)=15.034
ENG(16)=7.1017
ENG(17)=3.3546
ENG(18)=1.5846
ENG(19)=0.454
ENG(20)=0.21445
ENG(21)=0.1013
ENG(22)=0.037267
ENG(23)=0.010677
EN(24)=0.005044
EN(25)=0.001855

... OBTAIN THE ANGULAR DEPENDENT XSC & ENERGY FOR THE CORRESPONDING PROTON ENERGY BY INTERPOLATION FROM THE INPUT TABLE ...

DO 150 I=1,24
IF (EPKEV.LE.EP(I).AND.EPKEV.GT.EP(I+1)) GO TO 200
150 CONTINUE

DO 250 J=1,37
XSC1(J) = XSC(K,J) * (XSC(K,J) - XSC(K-1,J)) * (EPKEV - EP(K)) / (EP(K) - EP(K+1))
EN1(J) = EN(K,J) * (EN(K,J) - EN(K-1,J)) * (EPKEV - EP(K)) / (EP(K) - EP(K+1))
250 CONTINUE

EXTEND EN1(J) & XSC1(J) FROM EVERY 5 DEGREE TO EVERY 1 DEGREE

DO 300 I=1,181
XI = I
DO 280 J=1,37
IF (XI.GE.ANG1(J).AND.XI.LT.ANG1(J+1)) GO TO 290
280 CONTINUE

XSC2(I) = XSC1(J) * (XSC1(J-1) - XSC1(J)) * (XI - ANG1(J)) / (ANG1(J+1) - ANG1(J))
EN2(I) = EN1(J) * (EN1(J-1) - EN1(J)) * (XI - ANG1(J)) / (ANG1(J+1) - ANG1(J))
300 CONTINUE

START GENERATING ENERGY-ANGULAR DEPENDENT NEUTRON SOURCE

DEG1=0.
IA=1
DO 400 I=1,180
XI=I
DEG2=W1-0.5
RAD1=DEG1/CV
RAD2=DEG2/CV
STR=2*PI*(COS(RAD1)-COS(RAD2))
IF (I.GT.ANG2(IA)) GO TO 330
400 CONTINUE

XSC3(I) = (XSC2(I) - XSC2(I-1)) / 2.
EN3(I) = (EN2(I) - EN2(I-1)) / 2.
SN(I) = AN*PFLUX*XSC3(I)*STR*1.0E-27
SN(1) = AN*PFLUX*XSC3(I)*STR*1.0E-27
DO 360 II=1,25
IF (EN3(I).LE.ENG(II).AND.EN3(I).GT.ENG(II-1)) GO TO 370
360 CONTINUE

370 CONTINUE

SNG(IA,IG)=SNG(IA,IG)*SN(1)
SN(1) = AN*PFLUX*XSC3(I)*STR*1.0E-27
SN(1) = AN*PFLUX*XSC3(I)*STR*1.0E-27
DO 360 II=1,25
IF (EN3(I).LE.ENG(II).AND.EN3(I).GT.ENG(II-1)) GO TO 370
360 CONTINUE

EPKEV=EPKEV-1.0
IF (EPKEV.LE.1881) GO TO 450
GO TO 30

PRINT OUT RESULTS
450 WRITE(6,595) TSRC
     WRITE(6,599)((SNG(I,J),I=1,12),J=1,25)
500 CONTINUE
595 FORMAT(2X,'TOTAL NEUTRON SOURCE = ',E12.4)
599 FORMAT(6F10.4)
699 FORMAT(12(E10.4,1X))
 STOP
 END

//
Appendix B  FORTRAN listing of the SOURCE
SUBROUTINE SOURCE(IG,U,V,WfXfY,Z,WATE,AG)
R=1.25
PI=3.14159
RDS=SQRT(FLTRNF(0)*R**2)
THETA=FLTRNF(0)*2.*PI
Y=RDS*COS(THETA)
Z=RDS*SIN(THETA)
X=0.01
RN2=FLTRNF(0)
RN1=FLTRNF(0)
IF(RN1.LE.0.08426) GO TO 10
IF(RN1.GT.0.08426.AND.RN1.LE.0.30984) GO TO 20
IF(RN1.GT.0.30984.AND.RN1.LE.0.49457) GO TO 30
IF(RN1.GT.0.49457.AND.RN1.LE.0.65611) GO TO 40
IF(RN1.GT.0.65611.AND.RN1.LE.0.73801) GO TO 50
IF(RN1.GT.0.73801.AND.RN1.LE.0.77219) GO TO 60
IF(RN1.GT.0.77219.AND.RN1.LE.1.0) GO TO 5
10 DEGR=10.5
GO TO 130
20 DEGR=32.0
GO TO 150
30 DEGR=51.0
GO TO 170
40 DEGR=67.0
GO TO 190
50 DEGR=80.0
GO TO 210
60 DEGR=87.5
GO TO 230
CCC ANGLE1, 10.5 DEGREE
130 IF(RN2.LE.0.046443) GO TO 132
IF(RN2.GT.0.046443.AND.RN2.LE.0.3267) GO TO 134
IF(RN2.GT.0.3267.AND.RN2.LE.0.62717) GO TO 136
IF(RN2.GT.0.62717.AND.RN2.LE.0.81668) GO TO 138
IF(RN2.GT.0.81668.AND.RN2.LE.0.85626) GO TO 140
IF(RN2.GT.0.85626.AND.RN2.LE.0.92882) GO TO 142
IF(RN2.GT.0.92882.AND.RN2.LE.1.0) GO TO 144
132 IG=20
GO TO 400
134 IG=21
GO TO 400
136 IG=22
GO TO 400
138 IG=23
GO TO 400
140 IG=24
GO TO 400
142 IG=25
GO TO 400
144 IG=26
GO TO 400
CCC ANGLE2, 32.0 DEGREE
150 IF(RN2.LE.0.001306) GO TO 152
IF(RN2.GT.0.001306.AND.RN2.LE.0.18975) GO TO 154
IF(RN2.GT.0.18975.AND.RN2.LE.0.45365) GO TO 156
IF(RN2.GT.0.45365.AND.RN2.LE.0.72737) GO TO 158
IF(RN2.GT.0.72737.AND.RN2.LE.0.79308) GO TO 160
IF(RN2.GT.0.79308.AND.RN2.LE.0.87284) GO TO 162
IF(RN2.GT.0.87284.AND.RN2.LE.0.982) GO TO 164
IF(RN2.GT.0.982 .AND. RN2.LE.1.0) GO TO 166
152 IG=20
   GO TO 400
154 IG=21
   GO TO 400
156 IG=22
   GO TO 400
158 IG=23
   GO TO 400
160 IG=24
   GO TO 400
162 IG=25
   GO TO 400
164 IG=26
   GO TO 400
166 IG=27
   GO TO 400
CCC ANGLE3, 51.0 DEGREE
170 IF(RN2.LE.0.03774) GO TO 172
   IF(RN2.GT.0.03774 .AND. RN2.LE.0.23827) GO TO 174
   IF(RN2.GT.0.23827 .AND. RN2.LE.0.55742) GO TO 176
   IF(RN2.GT.0.55742 .AND. RN2.LE.0.68955) GO TO 178
   IF(RN2.GT.0.68955 .AND. RN2.LE.0.81373) GO TO 180
   IF(RN2.GT.0.81373 .AND. RN2.LE.0.88608) GO TO 182
   IF(RN2.GT.0.88608 .AND. RN2.LE.1.0) GO TO 184
172 IG=21
   GO TO 400
174 IG=22
   GO TO 400
176 IG=23
   GO TO 400
178 IG=24
   GO TO 400
180 IG=25
   GO TO 400
182 IG=26
   GO TO 400
184 IG=27
   GO TO 400
CCC ANGLE4, 67.0 DEGREE
190 IF(RN2.LE.0.07803) GO TO 192
   IF(RN2.GT.0.07803 .AND. RN2.LE.0.33879) GO TO 194
   IF(RN2.GT.0.33879 .AND. RN2.LE.0.5241) GO TO 196
   IF(RN2.GT.0.5241 .AND. RN2.LE.0.74117) GO TO 198
   IF(RN2.GT.0.74117 .AND. RN2.LE.0.84035) GO TO 200
   IF(RN2.GT.0.84035 .AND. RN2.LE.0.99537) GO TO 204
   IF(RN2.GT.0.99537 .AND. RN2.LE.1.0) GO TO 206
192 IG=22
   GO TO 400
194 IG=23
   GO TO 400
196 IG=24
   GO TO 400
198 IG=25
   GO TO 400
200 IG=26
   GO TO 400
202 IG=27
   GO TO 400
204 IG=28
    GO TO 400
206 IG=29
    GO TO 400
'CC ANGLE5, 80.0 DEGREE'
210 IF(RN2.LE.0.00042) GO TO 212
    IF(RN2.GT.0.00042.AND.RN2.LE.0.18364) GO TO 214
    IF(RN2.GT.0.18364.AND.RN2.LE.0.34646) GO TO 216
    IF(RN2.GT.0.34646.AND.RN2.LE.0.65718) GO TO 218
    IF(RN2.GT.0.65718.AND.RN2.LE.0.79934) GO TO 220
    IF(RN2.GT.0.79934.AND.RN2.LE.0.87286) GO TO 222
    IF(RN2.GT.0.87286.AND.RN2.LE.0.91147) GO TO 224
    IF(RN2.GT.0.91147.AND.RN2.LE.0.96304) GO TO 226
    IF(RN2.GT.0.96304.AND.RN2.LE.1.0) GO TO 228
212 IG=22
    GO TO 400
214 IG=23
    GO TO 400
216 IG=24
    GO TO 400
218 IG=25
    GO TO 400
220 IG=26
    GO TO 400
222 IG=27
    GO TO 400
224 IG=28
    GO TO 400
226 IG=29
    GO TO 400
228 IG=30
    GO TO 400
'CC ANGLE6, 87.5 DEGREE'
230 IF(RN2.LE.0.11693) GO TO 231
    IF(RN2.GT.0.11693.AND.RN2.LE.0.25539) GO TO 232
    IF(RN2.GT.0.25539.AND.RN2.LE.0.59223) GO TO 233
    IF(RN2.GT.0.59223.AND.RN2.LE.0.77321) GO TO 234
    IF(RN2.GT.0.77321.AND.RN2.LE.0.86021) GO TO 235
    IF(RN2.GT.0.86021.AND.RN2.LE.0.90569) GO TO 236
    IF(RN2.GT.0.90569.AND.RN2.LE.0.91942) GO TO 237
    IF(RN2.GT.0.91942.AND.RN2.LE.0.93255) GO TO 238
    IF(RN2.GT.0.93255.AND.RN2.LE.0.96268) GO TO 239
    IF(RN2.GT.0.96268.AND.RN2.LE.0.99566) GO TO 240
    IF(RN2.GT.0.99566.AND.RN2.LE.1.0) GO TO 241
231 IG=23
    GO TO 400
232 IG=24
    GO TO 400
233 IG=25
    GO TO 400
234 IG=26
    GO TO 400
235 IG=27
    GO TO 400
236 IG=28
    GO TO 400
237 IG=29
    GO TO 400
238 IG=30
    GO TO 400
239 IG=31
   GO TO 400
240 IG=32
   GO TO 400
241 IG=33
400 U=COS(DEGR*PI/180.)
   CALL AZIRN(SA,CA)
   WU=SQRT(1.-U**2)
   V=CA*WU
   W=SA*WU
   WATE=1.0
   RETURN
END
Appendix C  FORTRAN listing of the estimators used for MORSE calculations
SUBROUTINE RELCOL

... THIS VERSION IS FOR POINT DETECTORS LOCATED AT (XD,YD,ZD)

COMMON /USER/ AGSTRT, WSTRT, XSTRT, ZSTRT, ZSTRT, DFF, EBOTN, EBOTG,
1 TCU1, X, Y, IADJ M, NGPQT1, NGPQT2, NGPQT3, NGPQT4, NGPQT5, NITS, NLAST,
2 NLEFT, NGMP, NGMT, NSTRT,

COMMON /PDET/ ND, NNE, NE, NT, NA, NRESP, NEX, NEXN, NEND, NDNR, NTNR, NTNE,
1 NANE, NNDN, NENND, NAEN, LOCRSP, LOCXD, LOCYD, LOCZD, LOCUD,
2 LOCSD, LOCQE, LOCQT, LOCUE, LOCQA, LMAX, EFIRST, EGTOP,

COMMON /NUTRON/ NAME, NAMEX, IG, IG0, NAMED, MEDLD, NREG, U, V, W, W0D, VOLD,
1 WOLD, X, Y, Z, W0D, Y0L, Z0L, H0L, W0L, W0L0, W0L0, W0L0, W0L0,

COMMON BL(1)

DIMENSION NL(1)

EQUIVALENCE (BL(1), NL(1))

DATA NEST /1/, FNEST /1.0/

NEST • FNEST ARE THE NO. OF ESTIMATES TO BE MADE TO EACH DETECTOR

** ISTAT MUST BE EQUAL TO 1.

** NEX MUST BE AT LEAST 1

** NEXN MUST BE AT LEAST 1

DO 30 I=1, ND
1 A=LOCXD - I
2 XE = BL(A
3 YE = BL(A
4 ZE = BL(A
5 A = XE - X
6 B = YE - Y
7 C = ZE - Z
8 SD2=A*A - B*B - C*C
9 DS=SQRT(SD2)
10 IF(DS.LE.1.0) GO TO 2
11 GO TO 4
12 WRITE(6,129) IG0
13129 FORMAT(/,3X,14,'THE COLLISION IS L.T. 1 CM FROM DET')
14 GO TO 30

C ** COS DEPENDS ON THE ANGLE OF INTEREST

4 COS=C/DS
5 THETA = (A*U0L - B*V0L - C*W0L)/DS
6 IG0 = NGPQT3
7 IF (IG0.LE.NGPQT1) IG0 = NGPQT1
8 IA = LOCRSP * NRESP*NGMT + 1
9 CALL PTTHETA(NMED, IG0, IG0, THETA, BL(IA), NMTG)
10 NES = 0
11 PSUM = 0.
12 IA = IA - 1
13 DO 5 IL=IG0, IL = NGPQT3
145 PSUM = PSUM + ABS(BL(IA*IL))
15 R = FLTRNF(0) * PSUM
16 DO 15 IL=IG0, IL = NGPQT3
17 IF (R-ABS(BL(IA*IL))) 20, 20, 15
1815 R = R - ABS(BL(IA*IL))
19 IL = IG0
20 MARK=1
21 AGED = AGE • DS/BL(NMTG-IL)
22 MEDIUM=NMED
23 CALL EUCLID(MARK, X, Y, Z, XE, YE, ZE, DS, IL, ARG, 0, MEDIUM, BLZNT, NREG)
24 IF (ARG.LT.-64.) GO TO 25

** BEWARE THIS VERSION WILL NOT WORK IF ENERGY BIASING IS USED
CON = WATE*EXP (ARG)*SIGN (PSUM,BL(IA+IL))/SD2/FNEST
IF (CON.LT.1.E-40) GO TO 25
CALL FLUXST (1,IL,CON,AGED,COS,0)

25 NES = NES + 1
INN = LOCXD + 6*ND + 1
NL(INN) = NL(INN) + 1
IF (NES=NEST) 10,30,30

30 CONTINUE
RETURN
END
SUBROUTINE TRKTH

"C" TRACK-LENGTH ESTIMATOR
"C"
COMMON /NUTRON/ NAME, NAMEX, IG, IG0, NMED, MEDOLD, NREG, U, V, W, UOLD, VOLD, EOLDBANKR
1, WOLD, X, Y, Z, XOLD, YOLD, ZOLD, WATE, OLDWT, WTBC, BLZNT, BLZON, AGE, OLDAGEBANKR
COMMON / PDET/ ND, NNE, NE, NT, NA, NRESP, NEX, NEXND, NEND, NENDR, NTNR, NTNE,
2, NEND, NTEND, NEND, LOCSP, LOCXD, LOCIB, LOCCO, LOC, LOCUD,
LOCSD, LOCCE, LOCCT, LOCQAE, LMAX, EFIRST, EGTOP
COMMON BC(1)
DIMENSION NC(1), RAD(3), XD(3), YD(3), ZD(3)
EQUIVALENCE (BC(1), NC(1))
DATA RAD(1), RAD(2), XD(1), XD(2), YD(1), YD(2), ZD(1), ZD(2)/1.0, 2.0,
6.26, 2.0, 0.0, 0.0, 0.0, 0.0,
2.6, 2.0, 0.0, 0.0, 0.0, 0.0/
CC CHECK FOR BOUNDARY CROSSING OF DETECTOR
DO 200 I=1, ND
DIST=(X-XD(I))**2+(Y-YD(I))**2+(Z-ZD(I))**2
RAD2=RAD(I)**2
CHECK=DIST-RAD2
IF(ABS(CHECK).LT.0.0001) CHECK=0.
IF(CHECK).NE.0.0 RETURN
200 CONTINUE
RETURN
CC CHECK FOR INWARD CROSSING
100 DIR=(X-XD(I))*U-(Y-YD(I))*V-(Z-ZD(I))*W
IF(DIR.LT.0.0) RETURN
GO TO 300
200 CONTINUE
RETURN
CC CALCULATE FLUENCE
300 TRK=WATE*SQRT((X-XOLD)**2+(Y-YOLD)**2+(Z-ZOLD)**2)
VOL=4.0*3.14159*RAD(I)**3.0/3.0
CON=TRK/VOL
CC STORE FLUENCE CONTRIBUTION
CALL FLUXST(I, IG, CON.0.0, 0.0, 0.0)
CC ACCUMULATE THE NUMBER OF ESTIMATE
NC(LOCXD*6*ND-1)=NC(LOCXD*6*ND-1)+1
RETURN
END
Appendix D. FORTRAN listing of the estimator subroutine which calculates the neutron response functions for the boron-shell spectrometer
SUBROUTINE RESP

C RESPONSE FUNCTION ESTIMATOR
C
COMMON /NUTRON/ NAME, NAMEX, IG, IGO, NMED, MEDOLD, NREG, U, V, W, UOLD, VOLD, WOLD
1, XOLD, YOLD, ZOLD, WATE, OLDWT, WTBC, BLZNT, BLZON, AGE, OLDAGEBANKR
COMMON/ PDET/ ND, NNE, NE, NT, NA, NRESP, NEX, NEND, NDNR, NTNR, NTNE,
1 NANE, NTNRD, NTNEND, NACEND, LOCSP, LOCX, LOCY, LOCZ, LOCQ, LOCQTE, LOCQAE,
2 LOCX, LOCY, LOCZ, LOCQ, LOCQTE, LOCQAE, LMAX, EFIRST, EGTOP
COMMON BC(1)
DIMENSION NC(1), SIG1(47), SIG2(47)
EQUIVALENCE (BC(1), NC(1))
DO 10 I=1,47
SIG1(I)=0.
SIG2(I)=0.
10 CONTINUE
DATA (SIG1(I), I=20,47) / 0.1529, 0.1714, 0.1967, 0.2183, 0.2299,
& 0.2346, 0.2310, 0.2441, 0.2627, 0.2767, 0.2869, 0.2943, 0.3164, 0.3841, 0.5119,
& 0.7126, 1.113, 1.762, 2.528, 3.89, 6.792, 11.05, 17.11, 26.37, 38.34,
& 68.05, 171.9/
CC INPUT THE MACROSCOPIC XSEC FOR B-10 AT 1 G/CM**2
SIG1(20)=0.1529
SIG1(21)=0.1714
SIG1(22)=0.1967
SIG1(23)=0.2183
SIG1(24)=0.2299
SIG1(25)=0.2346
SIG1(26)=0.2310
SIG1(27)=0.2314
SIG1(28)=0.2441
SIG1(29)=0.2627
SIG1(30)=0.2767
SIG1(31)=0.2869
SIG1(32)=0.2943
SIG1(33)=0.3164
SIG1(34)=0.3841
SIG1(35)=0.5119
SIG1(36)=0.7126
SIG1(37)=1.113
SIG1(38)=1.762
SIG1(39)=2.528
SIG1(40)=3.89
SIG1(41)=6.792
SIG1(42)=11.05
SIG1(43)=17.11
SIG1(44)=26.37
SIG1(45)=38.34
SIG1(46)=68.08
SIG1(47)=171.9

CC CALCULATE THE MACROSCOPIC XSEC FOR 10 ATM HE-3
DO 20 I=20,47
SIG2(I)=SIG1(I)*0.00793
20 CONTINUE
DO 25 I=20,47
SIG1(I)=SIG1(I)/8.
25 CONTINUE
RA=3.75
RB=1.25
RC=2.5
PI=3.1415926

CC CHECK FOR BOUNDARY CROSSING OF DETECTOR

DIST=SQRT(X*X+Y*Y+Z*Z)
CHECK1=DIST-RA
CHECK2=DIST-RB
IF(ABS(CHECK1).LT.0.0001) GO TO 30
IF(ABS(CHECK2).LT.0.0001) GO TO 40
RETURN

30 CHORD1=SQRT((X-XOLD)**2+(Y-YOLD)**2+(Z-ZOLD)**2)
ARG1=-SIG1(IG)*CHORD1
WATE=WATE*EXP(ARG1)
WRITE(6,35) IG,WATE,SIG1(IG),CHORD1
C 35 FORMAT(1X,'IG',I3,'WATE',E12.4,'SIG1(IG)',E12.4,'CHORD1',E12.4)
RETURN

CC CHECKOUT INWARD CROSSING

40 DIR=X*U-Y*V-Z*W
IF(DIR.LT.0.0) RETURN
CHORD2=SQRT((X-XOLD)**2+(Y-YOLD)**2+(Z-ZOLD)**2)
ARG2=-SIG2(IG)*CHORD2
CON=WATE*(1.-EXP(ARG2))*PI*RB**2
CCC MULTIPLIER CON BY 27 SINCE THERE ARE 27 GROUPS ALTOGETHER AND
CCC THEREFORE EACH GROUP ONLY SHARES 1/27 OF THE TOTAL NUMBER OF
CCC THE NEUTRON HISTORY
CON=CON*27.
WRITE(6,45) IG,CON,SIG2(IG),CHORD2
C 45 FORMAT(1X,'IG',I3,'CON',E12.4,'SIG2(IG)',E12.4,'CHORD2',E12.4)
WATE=WATE*EXP(ARG2)
CCC CALCULATE RESPONSE
CCC STORE RESPONSE CONTRIBUTION
CALL FLUXST(1,IG,CON,0.0,0.0,0)
CCC ACCUMULATE THE NUMBER OF ESTIMATE
NC(LOCXD.6*ND-1)=NC(LOCXD.6*ND-1)+1
RETURN
END
SUBROUTINE SOURCE(IG,U,V,W,X, Y, Z, WATE, AG)

R=1.25
RR=6.25
X=-6.25
Y=0.
Z=0.
RAM=FLTRNF(0)
IF(RAM.GT.0.666667) GO TO 5
GO TO 100
5 IF(RAM.GT.0.888889) GO TO 10
GO TO 35
10 IF(RAM.GT.0.962963) GO TO 20
GO TO 25
20 IG=21
GO TO 300
25 IF(RAM.GT.0.925926) GO TO 30
IG=22
GO TO 300
30 IG=23
GO TO 300
35 IF(RAM.GT.0.777778) GO TO 40
GO TO 60
40 IF(RAM.GT.0.851852) GO TO 45
GO TO 50
45 IG=24
GO TO 300
50 IF(RAM.GT.0.814815) GO TO 55
   IG=25
   GO TO 300
55 IG=26
   GO TO 300
60 IF(RAM.GT.0.740741) GO TO 65
   GO TO 70
65 IG=27
   GO TO 300
70 IF(RAM.GT.0.703704) GO TO 75
   IG=28
   GO TO 300
75 IG=29
   GO TO 300
100 IF(RAM.GT.0.333333) GO TO 105
    GO TO 200
105 IF(RAM.GT.0.555556) GO TO 110
    GO TO 135
110 IF(RAM.GT.0.629630) GO TO 120
    GO TO 125
120 IG=30
    GO TO 300
125 IF(RAM.GT.0.592593) GO TO 130
    IG=31
    GO TO 300
130 IG=32
    GO TO 300
135 IF(RAM.GT.0.444444) GO TO 140
    GO TO 160
140 IF(RAM.GT.0.518519) GO TO 145
    GO TO 150
145 IG=33
    GO TO 300
150 IF(RAM.GT.0.481481) GO TO 155
    IG=34
    GO TO 300
155 IG=35
    GO TO 300
160 IF(RAM.GT.0.407407) GO TO 165
    GO TO 170
165 IG=36
    GO TO 300
170 IF(RAM.GT.0.370370) GO TO 175
    IG=37
    GO TO 300
175 IG=38
    GO TO 300
200 IF(RAM.GT.0.222222) GO TO 210
    GO TO 235
210 IF(RAM.GT.0.296296) GO TO 220
    GO TO 225
220 IG=39
    GO TO 300
225 IF(RAM.GT.0.259259) GO TO 230
    IG=40
    GO TO 300
230 IG=41
    GO TO 300
235 IF(RAM.GT.0.111111) GO TO 240
GO TO 270
240 IF(RAM.GT.0.185185) GO TO 245
   GO TO 250
245 IG=42
   GO TO 300
250 IF(RAM.GT.0.148148) GO TO 255
   IG=43
   GO TO 300
255 IG=44
   GO TO 300
270 IF(RAM.GT.0.074074) GO TO 275
   IG=45
   GO TO 300
275 IF(RAM.GT.0.037037) GO TO 285
   IG=46
   GO TO 300
285 IG=47
300 RD=SQRT(FLTRNF(0)*R**2)
   U=RR/SQRT(RR**2+RD**2)
   CALL AZIRN(SA,CA)
   WU=SQRT(1.-U**2)
   V=CA*WU
   W=SA*WU
   WATE=1.0
RETURN
END
Appendix E. FORTRAN listing of the SPUNIT code used for the boron-shell neutron spectrometer
PROGRAM SPUNIT

SPECTRUM UNFOLDING WITH INFORMATION THEORY

CONVERTED TO IBM 3081 BY C-K CHRIS WANG, 11/88

THIS PROGRAM CALCULATES THE NEUTRON FLUENCE AS A FUNCTION OF NEUTRON ENERGY USING AN ITERATIVE PERTURBATION TECHNIQUE TO SOLVE THE MATRIX APPROXIMATION OF A FREDDHOLM INTEGRAL EQUATION OF THE FIRST KIND. THE WANG 26 GROUP RESPONSE MATRIX IS USED TO UNFOLD THE SPECTRA FROM "WANG" SPECTROMETER DATA.

DIMENSION AL(7,26),EN(26),SPC(27),BC(7),BCS(7),ALTR(7),
& LE(27),LRETH(26),SPL(26),SPLX(26,10),SPLI(27),SPLL(27),
& A1(26),A2(26),A3(26),A4(26),A5(26),A6(26),A7(26)

INTEGER TITL(40)

ENERGY INTERVAL END POINTS
DATA (E(I),I=1,27)/
11.000E-07,4.140E-07,8.764E-07,1.855E-06,5.044E-06,
21.068E-05,3.727E-05,1.021E-04,2.145E-04,4.540E-04,
31.585E-03,3.355E-03,7.102E-03,1.503E-02,2.188E-02,
42.418E-02,2.606E-02,3.183E-02,4.087E-02,6.738E-02,
51.111E-01,1.832E-01,2.972E-01,3.688E-01,4.979E-01,
66.081E-01,7.427E-01/

WANG 26 GROUP RESPONSE MATRIX FACTORS

DATA (A1(I),I=1,26)/
11.187E-01,3.174E-01,4.151E-01,4.397E-01,3.952E-01,
23.039E-01,2.031E-01,1.398E-01,1.021E-00,6.672E-02,
34.401E-02,3.198E-02,2.412E-02,1.995E-02,1.846E-02,
41.805E-02,1.749E-02,1.398E-02,1.027E-01,6.672E-02,
51.451E-02,1.459E-02,1.471E-02,1.380E-02,1.252E-02,
61.088E-02/

DATA (A2(I),I=1,26)/
17.845E-03,6.874E-02,1.450E-01,2.222E-01,2.543E-01,
22.317E-01,1.739E-01,1.264E-01,9.576E-02,6.382E-02,
34.278E-02,3.133E-02,2.376E-02,1.970E-02,1.824E-02,
41.784E-02,1.730E-02,1.331E-02,1.535E-02,1.446E-02,
51.438E-02,1.478E-02,1.457E-02,1.368E-02,1.242E-02,
61.081E-02/

DATA (A3(I),I=1,26)/
15.910E-06,1.198E-03,8.948E-03,3.647E-02,7.918E-02,
21.131E-01,1.153E-01,9.680E-02,7.951E-02,5.674E-02,
33.966E-02,2.968E-02,2.281E-02,1.906E-02,1.769E-02,
41.731E-02,1.680E-02,1.566E-02,1.500E-02,1.411E-02,
51.403E-02,1.442E-02,1.422E-02,1.336E-02,1.216E-02,
61.062E-02/

DATA (A4(I),I=1,26)/
11.129E-15,3.015E-09,8.863E-07,3.558E-04,5.883E-03,
23.638E-02,1.063E-01,1.606E-01,1.778E-01,1.729E-01,
31.577E-01,1.269E-01,1.004E-01,8.435E-02,7.591E-02,
47.717E-02,7.367E-02,6.602E-02,5.594E-02,4.378E-02,
53.675E-02,2.669E-02,2.547E-02,2.199E-02,1.867E-02,
61.607E-02/

DATA (A5(I),I=1,26)/
11.179E-21,1.208E-12,4.090E-09,1.306E-05,5.929E-04,
DATA (A6(1),I=1,26)/
17.094E-23, 5.297E-18, 6.908E-13, 5.745E-09, 2.007E-05,
21.508E-03, 2.166E-02, 6.917E-02, 1.207E-01, 1.737E-01,
32.024E-01, 2.010E-01, 1.770E-01, 1.499E-01, 1.518E-01,
41.531E-01, 1.446E-01, 1.327E-01, 1.210E-01, 9.247E-02,
57.597E-02, 5.952E-02, 4.578E-02, 3.743E-02, 3.668E-02,
61.661E-02/
DATA (A7(1),I=1,26)/
17.167E-23, 3.492E-21, 4.506E-15, 2.170E-10, 2.061E-07,
23.924E-04, 1.029E-02, 4.284E-02, 8.590E-02, 1.400E-01,
31.756E-01, 1.773E-01, 1.678E-01, 1.521E-01, 1.424E-01,
41.427E-01, 1.297E-01, 1.255E-01, 1.049E-01, 9.040E-02,
56.399E-02, 4.937E-02, 4.170E-02, 4.006E-02, 3.087E-02,
63.176E-02/
C
DO 1 I=1,26
AL(1,I)=A1(I)/2.36
AL(2,I)=A2(I)/2.36
AL(3,I)=A3(I)/2.36
AL(4,I)=A4(I)/2.36
AL(5,I)=A5(I)/2.36
AL(6,I)=A6(I)/2.36
AL(7,I)=A7(I)/2.36
1 CONTINUE
READ(5,15) TITL
15 FORMAT(40A2)
READ(5,49)(BC(I),I=1,7)
49 FORMAT(7E11.4)
SUMBC=0.
DO 40 I=1,7
40 SUMBC=SUMBC*BC(I)
IA=AVEBC
WRITE(6,59)
59 FORMAT(///,' CONVERGENCE CRITERIA:',/' THE iterations will',
1', cease when either maxima number of iterations are',
2', exceeded',' OR percent error (based on calculated vs',
3', input detector readings) is met,',
4', WHICHEVER CONDITION OCCURS FIRST.',
5', DEFAULT VALUES:',/' 500 ITERATIONS, MAX.',
6', 3.0% ERROR.')
NITER=5000
ERRTST=0.1
C
ECHOS OF THE INPUT VALUES
C
WRITE(6,55)TITL
55 FORMAT(/,'X,40A2,/)W R I T E (  6,  65) (B C ( I ) , I = 1 , 7 ) , N I T E R , E R R T S T
65 FORMAT(/,' input values for detector responses:',
6X, 'DET7', /, 5X, (F10.3), ///),
& ' CONVERGENCE CRITERIA: ',/,'5X, 'max. # OF iterations =',15,
& /, '5X, ' percent error =',F6.2,'%',//)
C
SET INITIAL SPECTRUM

READ(5,75)(SPL(I),I=1,26)
75 FORMAT(6E11.4)
DO 80 I=1,13
IP=I+13
80 WRITE(6,85)I,SPL(I),IP,SPL(IP)
85 FORMAT(3X,2(112,E11.2))
DO 100 I=1,26
100 SPL(I)=SPL(I)

CALCULATE THE DETECTOR RESPONSES AND SUM FROM INITIAL SPECTRUM

DO 110 M=1,7
BCS(M)=0.
DO 110 J=1,26
BCS(M)=BCS(M)*AL(M,J)*SPL(J)
110 CONTINUE
WRITE(6,115)(BCS(I),I=1,7)
115 FORMAT(/,3X,'BCS(I ) = ',7E12.4,/) 
SUMBCS=0.
DO 120 I=1,7
120 SUMBCS=SUMBCS+BCS(I)

CALCULATE ERROR ON FIT
ERROR=0.
DO 130 I=1,7
ERR=(BCS(I)-BC(I))/BC(I)
130 ERROR=ERROR+ERR*ERR

NUMBER ENERGY INTERVALS
DO 140 I=1,26
140 EN(I)=I

BEGIN ITERATION LOOP

DO 320 ITER=1,NITER
ITER=ITER
DO 260 N=1,26
FACT1=0.
DO 240 I=1,7
FACT1=FACT1+AL(I,N)/BCS(I)
240 CONTINUE
FACT2=0.
DO 250 J=1,7
FACT2=FACT2+AL(J,N)/BCS(J)
250 CONTINUE
SPL(N)=SPL1(N)*FACT1/FACT2
S=0.01
SPL(1)=(SPL(1)*S*SPL(2))/(1.+S)
DO 265 N=2,25
265 SPL(N)=(S*SPL(N-1)*SPL(N)*S*SPL(N+1))/(1.+2.*S)
SPL(26)=(SPL(26)*S*SPL(25))/(1.+S)

CALCULATE NEW DETECTOR RESPONSES

DO 270 I=1,7
BCS(I)=0.
DO 270 N=1,26
BCS(I)=BCS(I)+SPL(N)*AL(I,N)
270 CONTINUE
ERROR=0.
ERR=0.
DO 275 I=1,7
275 CONTINUE
ERR=(BCS(I)-BC(I))/BC(I)

ERROR=ERROR*ERR*ERR
FERROR=100.*SQRT(ERROR/7.)

SET SPL TO SPLI

DO 280 N=1,26
   SPLI(N)=SPL(N)
   END OF ITERATION LOOP

IF(PERROR.LT.ERRTST) GO TO 350
IF(MOD(ITER,100).NE.0) GO TO 320

WRITE(6,315) ITER, PERROR, (BCS(I),I=1,7)
315 FORMAT(15X,'ITERATION #',I7,5X,'ERROR =',F7.2,'%',/,
&3X,'BCS(I) =',7F11.4,/
320 CONTINUE
350 WRITE(6,360)
360 FORMAT(/,3X,'GRP #',7X,'SPECTRUM',/)
   DO 365 I=1,26
      WRITE(6,363)I, SPL(I)
363 FORMAT(4X,13,7X,E11.5)
365 CONTINUE
STOP
END