Design Studies of an Epithermal Neutron Beam for Neutron Capture Therapy at the Musashi Reactor

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Studies were carried out to design an epithermal neutron beam for neutron capture therapy at the Musashi reactor, a TRIGA-II of 100 kW. The idea in this design is to use spent fuel elements as a converter assembly to convert thermal neutrons from the core to fission neutrons which would then be moderated to epithermal neutrons. In the design, 43 elements were placed outside of the graphite reflector but inside the tank. These elements could be cooled by water in the reactor tank and also be easily removed for inspection. Monte Carlo computations indicated that by using a 63 cm-thick Al₂O₃ moderator, an epithermal neutron beam with an intensity of 0.34 × 10⁹ n·cm⁻²·s⁻¹ and fast neutron and γ-doses per epithermal neutron of 4.3 × 10⁻¹¹ cGy·cm⁻²·n⁻¹ and 0.3 × 10⁻¹¹ cGy·cm⁻²·n⁻¹, respectively, could be produced. The multiplication factor of the 43 spent fuel elements was <0.95 and negligible reactivity coupling to the core by the spent fuel was calculated. Such an epithermal neutron beam, if built into the Musashi reactor, would significantly increase the chance of success of Boron Neutron Capture Therapy (BNCT) in Japan.

KEYWORDS: epithermal neutron beams, boron neutron capture therapy, TRIGA-II-Musashi reactor, spent fuels, fuel elements, converters, aluminum oxides, Monte Carlo method, fast neutrons, gamma radiation, radiation doses, current-to-flux ratio, criticality

I. INTRODUCTION

The reactor at the Musashi Institute of Technology Research Reactor (Musashi reactor) has been used since 1976 for treatment of brain tumors and malignant melanoma by boron neutron capture therapy (BNCT)(1)-(3). The BNCT results in Japan have encouraged researchers around the world to reevaluate BNCT(4)-(6).

The Musashi reactor is a TRIGA-II type reactor. With a thermal power of only 100 kW, a thermal neutron beam with satisfactory characteristics was produced for use in BNCT(7)(8). The BNCT results in Japan have encouraged researchers around the world to reevaluate BNCT(4)-(6).

The Musashi reactor was shut down in 1990 because of a water leak trouble in the reactor tank. An expanded purpose to utilize the reactor to treat deep-seated tumors with an epithermal neutron beam would help justify installing a new tank and restarting the reactor. In addition to extensive experience by thermal neutron capture therapy at the Musashi Institute of Technology, we now have a chance to design an epithermal neutron beam with wider applicability.

TRIGA type reactors are generally accepted as being safe for operation and about 60 TRIGAs are operating throughout the world. The advantage of using a TRIGA reactor for
BNCT have already been pointed out and several ideas to produce beams of thermal and epithermal neutrons have been reported. Recently, a method of generating an epithermal neutron beam using a $^{235}$U fission plate in the thermal column of a TRIGA reactor was proposed and this looks promising. This idea has been adapted to upgrade the present epithermal neutron beam at the Brookhaven Medical Research Reactor (BMRR), and to produce a very good beam in terms of intensity and quality. In this paper, we present the evaluation of using spent fuel elements of TRIGA fuel ($UZrH\,8.4/90/1.6$ wt%, $20\%$ enrichment $^{235}$U) as a converter assembly coupled to the 100 kW Musashi reactor to produce an epithermal neutron beam.

An epithermal neutron beam with neutron energies from 0.4 eV to 10 keV is promising for treating deep-seated tumors by BNCT. We have the possibility to design an epithermal neutron beam at the thermal column. To predict neutron fluxes and doses in this study, the Monte Carlo Neutron Photon (MCNP) code has been used. The epithermal neutron beam needed for BNCT is necessary to have a high flux of neutrons and a low background dose to normal tissues from fast neutrons and $\gamma$-rays. The design goals in this study are: (a) an epithermal neutron flux $>0.5 \times 10^9 \text{ cm}^{-2}\cdot\text{s}^{-1}$, (b) the dose from fast neutrons $<5 \times 10^{-11} \text{ cGy} \cdot \text{cm}^2 \cdot \text{n}^{-1}$, (c) the dose from $\gamma$-rays $<3 \times 10^{-11} \text{ cGy} \cdot \text{cm}^2 \cdot \text{n}^{-1}$, (d) the multiplication factor $(K_{\text{en}})$ in the spent fuel elements $<0.95$. The first goal was selected to have a full-dose treatment of time in about 2 h. A peak (2~3 cm deep) thermal neutron fluence of $5 \times 10^{13} \text{ n} \cdot \text{cm}^{-2}$ could be delivered. The second and third are deduced from the experiences by thermal neutron capture therapy at the Musashi reactor. The fourth is generally recognized as a safety value in Japan.

**II. MATERIALS AND METHODS**

1. **Musashi Reactor**

Figure 1 shows a vertical cross section of the Musashi reactor BNCT facility. The reactor and experimental facilities are surrounded by a concrete shield structure. The reactor core and graphite reflector assembly are located at the bottom of a 2 m diameter Al-tank which is 7 m in height. Approximately 6 m of water sits above the core which provides sufficient shielding in the vertical direction. The core (44 cm in diameter and 40 cm in height) is shielded by a minimum of 1 m thick concrete, 45 cm thick water and 30 cm thick graphite.

![Fig. 1 Vertical cross section showing Musashi reactor “TRIGA-II” of 100 kW](image-url)
There are two irradiation columns on opposite faces of the reactor core. One is called the thermalizing column and the other the thermal column. These extend from the outer face of the reflector assembly into the concrete shield structure. The thermal column was originally a large boral-lined, graphite-filled Al-container with an area of 1.2 m × 1.2 m and 1.7 m in length. This was remodeled in 1976 for BNCT\(^{(7)}\). Figure 2 shows the configuration of the remodeled thermal column including the setup for treating a brain tumor patient. The thermal column consists of graphite block moderators, a void region (cavity), a bismuth (Bi) shield to reduce γ-rays, and outer graphite blocks for extracting thermal neutrons to the patient port. A \(^6\)LiF sheet was used as a thermal neutron shield (or collimator) to protect normal tissues outside the irradiation aperture. A patient was set up at the irradiation port which is 175 cm away from the center of the core and within a shielded room with 1 m-thick concrete walls. Table 1 shows beam characteristics of the Musashi thermal neutron beam at the patient port. In 1985, fuel elements were changed from Al clad to stainless steel clad to enhance operation safety (see Fig. 3). The water pool is used for fuel storage and 63 Al-clad fuel elements are stored in the pool since 1985. The fuel elements consists of 3 parts; meat, graphite reflector, and stainless steel end fixture as shown in Fig. 3. The meat is a solid, homogenized mixture of U and ZrH\(_{1.6}\) containing about 8.4 wt% of U enriched to 20% \(^{235}\)U. The H-to-Zr atom ratio is approximately 1.6. Each fuel element is clad with a 0.05 cm thick stainless steel. A fuel element is 3.73 cm in diameter with a total length of 75 cm. Figure 4 shows the core arrangement used for operation until 1990. Water flows in the space between each element to provide natural convection cooling for the core. For the

![Diagram of thermal column](image)

**Thermal Column**

Figs. 2 Remodeled thermal column structure including setup of patient treated for brain tumor by BNCT

<table>
<thead>
<tr>
<th>Table 1</th>
<th>Characteristics of Musashi thermal neutron beam at patient port</th>
</tr>
</thead>
<tbody>
<tr>
<td>Item</td>
<td>Value</td>
</tr>
<tr>
<td>---------</td>
<td>----------------------</td>
</tr>
<tr>
<td>Thermal neutron flux ( (n \cdot cm^{-3} \cdot s^{-1}) )</td>
<td>( 0.8 \times 10^9 )</td>
</tr>
<tr>
<td>Cadmium ratio</td>
<td>100</td>
</tr>
<tr>
<td>Fast neutron flux ( (n \cdot cm^{-3} \cdot s^{-1}) )</td>
<td>( 1.0 \times 10^6 )</td>
</tr>
<tr>
<td>Fast neutron dose rate ( (cGy \cdot h^{-1}) )</td>
<td>25</td>
</tr>
<tr>
<td>Gamma-ray dose rate ( (cGy \cdot h^{-1}) )</td>
<td>60</td>
</tr>
</tbody>
</table>
MCNP computations, core components were considered as a homogenized mixture within the core.

2. Monte Carlo Code

The Monte Carlo for Neutron and Photon (MCNP) transport code, which was developed at the Los Alamos National Laboratory, is a general-purpose Monte Carlo code\(^{15}\). The MCNP can be used to model 3-D transport of neutrons, photons, coupled neutron/photons, or coupled photon/electrons. The MCNP uses either continuous or discrete nuclear cross section data such as ENDF\(^{16}\) or ENDL\(^{17}\) to describe the probability of reactions. The code is well designed and experimentally validated so it is a good computational tool for applications to the neutron beam design. The MCNP code, version 4.2a, which is running on an IBM RISC-6000 workstation in the Medical Department of Brookhaven National Laboratory (BNL) was used throughout the design work.

The MCNP computations were carried out for the design of an epithermal neutron beam at the Musashi reactor, starting each time with fission neutrons from the homogenized core and traveling out to the patient port. The number of fission neutrons in the core were normalized to \(7.6\times10^{14}n\cdot\text{s}^{-1}\) for an operating power of 100 kW. We used the nuclear data files based on ENDF-V. Among these nuclides, \(\gamma\)-ray production data of zirconium (Zr) was not included and only prompt \(\gamma\)-rays were considered. This would underestimate the dose from \(\gamma\)-rays from the core at the patient port. To verify the calculation method, neutron flux distributions in the old thermal column construction were calculated first to compare with the experimental results.

The \(K_{\text{eff}}\) was calculated by K code in MCNP for the core used since 1985. MCNP has the capability to calculate \(K_{\text{eff}}\) eigenvalues for fissile systems. The K code is a standard feature. A \(K_{\text{eff}}\) value was also calculated for the core with the spent fuel elements which placed outside the reflector. The nominal source size (3,000 particles) per cycle and 200 cycles were made to obtain a statistical error estimate <0.1\%. The K code computation was also used to investigate the criticality of the spent fuel elements to assure safety. The fission heat of the spent fuel elements was also calculated for thermodynamic estimates.
3. Proposed Design

We have designed an epithermal neutron beam for the irradiation port in the thermal column. Figure 5 depicts the design for the irradiation port. The core was homogenized (Fig. 4) and the reflector material was graphite. Spent fuel elements as a converter assembly were placed outside the graphite reflector in a spent fuel basket at 69 to 81 cm from the center of the core. After considering criticality, the spent fuel elements were arranged in 3 rows having 14, 15 and 14 elements in each row, total 43 elements corresponding to 1.68 kg $^{235}$U. The distance from center to center of each element is 4 cm. The idea to install these elements within the tank allows them to be cooled by the water in the tank and also they can be easily inspected at any time. Several materials were examined to find an effective moderator to slow fission neutrons to epithermal neutrons as will be shown in Sec. III-3. To allow for easy setting up a patient at the irradiation port, the patient location is 175 cm from the center of the core. This patient port is the same distance as that of the old thermal column used for BNCT. At the irradiation port near the patient position we included a void space. Thus, we can adjust the intensity of the beam at the irradiation port by changing the length of this beam path. To collimate the proposed epithermal neutron beam, 20 cm thick Pb was used, which was also useful for whole body shielding from $\gamma$-rays. Bismuth blocks of 10 cm thickness were used in the beam line to shield $\gamma$-rays.

III. RESULTS AND DISCUSSION

1. Musashi Thermal Neutron Beam at the Old Thermal Column

Figure 6 shows the MCNP calculated fast, epithermal, and thermal neutron fluxes along the beam central line from the reactor core to the patient port. Thermal, epithermal, and fast neutron energy ranges were selected as $<0.4$ eV, $0.4$ eV~$10$ keV and $>10$ keV, respectively. The old thermal column consists of graphite, void space, bismuth blocks, and another void space around the patient port (see Fig. 2). Experimental data of thermal neutron fluxes in the core center, at the Bi surface, and also at the patient port were $3.5 \times 10^{12}$, $3 \times 10^9$ and $8 \times 10^8$ n·cm$^{-2}$·s$^{-1}$, respectively. For an estimate of the epithermal neutron flux, the beam was filtered with a Cd sheet and flux measurements were made in a head phantom. The thermal neutron flux at 2 cm deep in the phantom was measured to be $3 \times 10^7$ n·cm$^{-2}$·s$^{-1}$ (8). The same value of the thermal neutron flux could be obtained from an incident flux of epithermal neutrons of $1.5 \times 10^7$ n·cm$^{-2}$·s$^{-1}$ (shown in the Fig. 6) into the phantom. The fast neutron flux was measured by activating indium foils using the $^{115}$In($n, n'$)$^{115m}$In reaction and is
shown in Table 1. The value is lower than the calculated result because the threshold energy of the $^{115}$In($n, n')^{115m}$In reaction is about 500 keV. The fast-neutron dose rate was calculated to be 23 cGy·h⁻¹ which was in agreement with the experimental data obtained by the paired chambers (see Table 1). The consistency of the calculated and measured results is satisfactory.

2. Criticality

The $K_{\text{eff}}$ factor of the homogenized core used until 1990 was calculated to be 1.030 with no control rods included in the calculation. The criticality of the spent fuel elements in the converter region was also calculated. A $K_{\text{eff}}$ factor of 0.937 was calculated for 43 spent fuel elements in the spent fuel basket. A $K_{\text{eff}}$ factor of 1.030 was also calculated for the core with the 43 spent fuel elements. A negligible reactivity coupling to the core by the spent fuel was found.

3. Epithermal Neutron Beam

(1) Selection of an Effective Moderator

The epithermal neutron flux using aluminum as a moderator showed higher values than those for the Al₂O₃ and graphite moderators but the fast neutron flux was also high. The graphite was unsatisfactory because of a low flux of epithermal neutrons. A good ratio of epithermal to fast neutron fluxes was obtained for the Al₂O₃ with a thickness of 60 cm. The Al₂O₃ was selected as an effective moderator for the proposed design.

(2) Neutron Flux Distributions

Figure 7 shows, with the spent fuel elements in the thermal column, the fast, epithermal and thermal neutron flux distributions along the beam central line from the center of the core to the patient position. The fast neutron flux in the core was higher than the thermal neutron flux because of thermal neutron absorption in the fuel. The fast neutron flux was then rapidly decreased in the graphite moderator while the thermal neutron flux was increased. The distributions for fast and epithermal neutrons around the edge of the graphite reflector were different from those without the spent fuel elements (see Fig. 8). The fast and epithermal neutron fluxes were 50 and 15 times higher than those without the spent fuel elements, respectively. These increments indicate that spent fuel elements as a fission converter assembly could produce extra fission neutrons to enhance the epithermal neutron flux at the patient port.

The fast neutrons produced from the fission converter assembly and from the core were moderated by 63 cm thick Al₂O₃ inside the thermal column. The epithermal neutron flux decreases from $2\times10^{18}$ n·cm⁻²·s⁻¹ downstream the converter assembly through Al₂O₃ to $1.5\times10^9$ n·cm⁻²·s⁻¹ at the Cd-film while the
thermal neutron flux decreases from $1.2 \times 10^{10}$ to $2.5 \times 10^8 n\cdot cm^{-2}\cdot s^{-1}$ at the same location. After the 0.05 cm thick Cd-film, the thermal neutron flux was further reduced to $2.5 \times 10^6 n\cdot cm^{-2}\cdot s^{-1}$. The attenuation by the 10 cm thick Bi block was 50% for both fast and epithermal neutron fluxes while the attenuation of the $\gamma$-dose was a factor of 10. The 20 cm long void space surrounded by the lead collimator shows another 50% decrease for the neutron fluxes. The intensity of the epithermal neutron flux can be adjusted by changing the length of this void space without changing the flux ratio of epithermal to fast neutrons.

(3) Neutron Beam Characteristics

Table 2 summarizes the characteristics of the proposal epithermal neutron beam at the thermal column of the Musashi reactor in comparison to the currently available epithermal neutron beams\(^{(19)}\). An epithermal neutron flux of $0.34 \times 10^9 n\cdot cm^{-2}\cdot s^{-1}$ (statistical error estimate 3\%) could be produced at the patient port by using a converter assembly of 43 spent fuel elements. This flux is smaller than the design goal ($0.5 \times 10^9 n\cdot cm^{-2}\cdot s^{-1}$) but by shortening the length of the void space or increasing the reactor power, the intensity would reach $0.5 \times 10^9 n\cdot cm^{-2}\cdot s^{-1}$. The beam intensity per unit of operating power which could be produced at the Musashi reactor is higher than those of the MITR and PETTEN epithermal neutron beams\(^{(19)}\). The contamination of fast-neutron dose per epithermal neutron was $4.3 \times 10^{-11}$ cGy\cdot cm$^2$\cdot n$^{-1}$ (statistical error estimate 10\%) which is less than the value of the design goal. The contamination of $\gamma$-dose per epithermal neutron was $0.3 \times 10^{-11}$ cGy\cdot cm$^2$\cdot n$^{-1}$ (statistical error estimate 10\%). The $\gamma$-dose might be increased when delayed $\gamma$-rays from the core and capture $\gamma$-rays from Zr are included. The value could be still less than the value of the design goal because the intensity of the delayed $\gamma$-rays is 3/4 times of prompt $\gamma$-rays while the capture cross section for Zr is small (0.19 barn)\(^{(21)}\). The epithermal neutron beam is also directed forward with a J/o ratio of 0.64; such anisotropic beam having better penetration in tissues than anisotropic beam.

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Reactor power (MW)</th>
<th>Epithermal$^*$ neutron flux $\times 10^9$ n\cdot cm$^{-2}$\cdot s$^{-1}$</th>
<th>$D_{fast}/n_{epi}$ $\times 10^{-11}$ cGy\cdot cm$^2$\cdot n$^{-1}$</th>
<th>$D_{\gamma}/n_{epi}$ $\times 10^{-11}$ cGy\cdot cm$^2$\cdot n$^{-1}$</th>
<th>J/O$^*$</th>
</tr>
</thead>
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<td>1.8</td>
<td>4.3</td>
<td>1.3</td>
<td>0.67</td>
</tr>
<tr>
<td>MITR</td>
<td>5</td>
<td>0.2</td>
<td>13</td>
<td>13</td>
<td>0.55</td>
</tr>
<tr>
<td>PETTEN</td>
<td>45</td>
<td>0.33</td>
<td>10.4</td>
<td>8.4</td>
<td>0.80</td>
</tr>
<tr>
<td>This work</td>
<td>Musashi</td>
<td>0.1</td>
<td>0.34</td>
<td>4.3</td>
<td>0.3</td>
</tr>
</tbody>
</table>

$^*$ Values in air at patient irradiation position

(4) Heat of Spent Fuel Elements

The fission heat was calculated for 43 spent fuel elements. The total fission heat was 2.0 kW which was 1/50 of the operating power of the reactor. The water temperature rise in the reactor tank is 4.75°C\cdot h$^{-1}$ for 100 kW operation\(^{(22)}\), therefore, 2.0 kW would increase the temperature by only 0.1°C\cdot h$^{-1}$. There is no need another cooling system to handle this heat.

IV. CONCLUSION

Design studies were carried out for producing epithermal neutron beams for neutron capture therapy in the thermal column, at the Musashi reactor. The approach to obtain epithermal neutrons is to use spent fuel elements as a converter assembly which would convert thermal neutrons to fission neutrons. Forty three spent fuel elements were placed outside the graphite reflector in
the region of 69~80 cm away from the center of the core. There would be a spent fuel basket to accommodate those spent fuel elements where they could be easily reached for inspection at any time and also would be cooled by water in the reactor tank. Aluminum oxide (Al₂O₃) was selected as an effective moderator to slow fission neutrons to epithermal neutrons. Monte Carlo computations indicated that by using 63 cm thick Al₂O₃ as a moderator, an epithermal neutron beam with an intensity of 0.34×10⁹ n·cm⁻²·s⁻¹ and fast-neutron and γ-doses per epithermal neutron of 4.3×10⁻¹¹ and 0.3×10⁻¹¹ cGy·cm²·n⁻¹, respectively, could be produced at the thermal column. The beam would be relatively forward-directed with a neutron current-to-flux ratio of 0.64. The criticality calculations showed that the Kₚ factor of the spent fuel assembly was <0.95 and negligible reactivity change was produced to the core by the spent fuel elements. The total fission heat of the spent fuel elements was 2.0 kW which would increase the water temperature by only 0.1°C·h⁻¹. Such an epithermal neutron beam, if built into the Musashi reactor, would significantly increase the chance of success of BNCT in Japan. Operating the reactor up to 250 kW would provide a shorter treatment time and would be more comfortable for patients. This design also suggests how TRIGA reactors in other countries might be modified to produce beams of epithermal neutrons for neutron capture therapy.

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