Neutron sources for BNCT: a general review

V. Giusti[1]

[1]Department of Mechanical, Nuclear and Production Engineering, University of Pisa, Italy.

ABSTRACT

A review of different neutron sources used or potentially useful for Boron Neutron Capture Therapy (BNCT) applications is here presented. As the neutron source energy spectrum is not in general directly suitable for BNCT applications, the actions that should be undertaken in order to optimize it without affecting too much the neutron flux intensity will be also described.

Introduction

BNCT is a binary modality which is expected to selectively deliver lethal, high linear energy transfer (LET) radiation to tumor cells dispersed within normal tissues.[1] The two heavy fragments emitted by the nuclear reaction which occurs when the ^{10}B captures a neutron (namely $^{10}B(n,\alpha)^{7}Li$) have a range, in the tissue, which is of the same order of the cell dimensions. As a consequence they release their energy only to the cells which surround the position where the reaction occurs. It is evident that to get a successful BNCT a sufficiently high amount of boron must be selectively delivered to each tumor cell[2] while a neutron beam with an energy spectrum such that most of the neutrons can reach the target after being slowed down to thermal energy must be provided. For the latter purpose, a neutron spectrum in the epithermal energy range (0.4 eV – 10 or 20 keV) is generally considered optimal for postoperative treatment, through an intact scalp and skull, of deeply located intracranial tumors.[3] Finally, it is important to remind that the intensity of such an epithermal neutron flux should be enough to perform a BNCT treatment in a conveniently short time, i.e. not longer than 30 minutes.

Neutron sources

Nuclear reactor based neutron sources

From the beginning of the BNCT history, the nuclear reactors, in particular those used for experimental purposes, have been the only neutron sources able to provide the correct energy spectrum with, at the same time, an adequate thermal neutron intensity. This became even more evident when the BNCT moved from the use of a thermal neutron beam to the use of a more energetic, epithermal, neutron beam. In fact, while the moderation until the thermal energies is pretty easy to perform, the tailoring process aimed at enhancing the energy spectrum in an intermediate energy range like the epithermal one is unavoidably quite expensive in terms of the final useful neutron flux intensity.

In a nuclear reactor, neutrons are generated by the fission reactions of the ^{235}U and, to a minor extent, ^{238}U (although some reactors can use the uranium in combination with different fissile nuclides like Pu, Th etc.) occurring in the reactor core. The energy spectrum of the neutrons emitted by a ^{235}U fission reaction is spread over a large energy range whose mean energy is equal to 1.98 MeV. Assuming an energy yield of 180 MeV per fission and that each fission reaction is emitting a number of neutrons equal to 2.45, a nuclear reactor provides a neutron source intensity of some $8.4 \cdot 10^{10}$ $n \cdot s^{-1} \cdot W^{-1}$. If a typical experimental reactor power of 250 kW is considered, this corresponds to a source intensity of $2.1 \cdot 10^{16}$ $n \cdot cm^{-2} \cdot s^{-1}$. As pointed out before, the fission neutron energy spectrum can not be directly employed for BNCT, but it requires suitable modifications in order to be enhanced in the epithermal part and depressed in the thermal and fast components. In the following the actions that should be undertaken in order to tailor the neutron energy spectrum will be described.

Finally, it should be pointed out as all the BNCT clinical trials performed until now relied on nuclear reactor based neutron sources. A good review of them has also been recently published.[4]

Accelerator-based neutron sources

In a second step, when BNCT has been proved to be successful and it has to move to a wide clinic use, a different kind of neutron sources should be employed. In general, experimental nuclear reactors are not very close to the hospitals and the idea to build new small reactors just for BNCT purposes is not very attractive owing to the high investment cost and the low acceptability of such a structure inside a hospital environment. Neutron sources able to satisfy the requirements of reasonable cost, small size, easy maintenance and high acceptability are, instead, the accelerator based neutron sources. These devices accelerate light charged particles until a defined energy and let them impinge on a suitable target where the nuclear reactions which produce the neutrons take place. In the case of an accelerator based neutron source it is possible to refer to different nuclear reactions corresponding to a different combination of charged particle and target material. Some examples of useful nuclear reactions are reported in Table 1.[5] Comparing the neutron production rate and the average and maximum neutron energy it turns out that the combination of an accelerated proton beam with a lithium target would provide interesting results. However, while the use of a target material like beryllium or carbon does not introduce big constraints as it concerns the heat removal system, in the case of lithium, owing to its quite low melting point (181°C) and low thermal conductivity (85 $W \cdot m^{-1} \cdot K^{-1}$), it probably represents the main issue that has to be overcome.

Interesting are also the fusion reactions D-T and D-D.[6,7] Both reactions are producing monoenergetic neutrons (2.45 MeV for the D-D and 14.1 MeV for the D-T case) and require a pretty low beam energy ranging from 100 and 120 keV. At these beam energies the neutron yield of the D-T reaction is of two orders of magnitude larger than the D-D yield, however, the presence of the tritium in the D-T reaction can arise some concern if employed in a hospital environment. As reported by one of the main laboratories developing this kind of sources, the neutron source intensity for a D-D based device is expected to be equal to $1.2 \cdot 10^{12}$ $n \cdot s^{-1}$ (an upgraded version is expected to produce $1.6 \cdot 10^{13}$ $n \cdot s^{-1}$).[8]

Table 1. Characteristic of some nuclear reaction of interested for accelerator based neuron source. Data from Ref. 5.

	Beam energy (MeV)	Neutron yield ($n \cdot s^{-1} \cdot mA^{-1}$)	Average neutron energy (MeV)	Maximum neutron energy (MeV)
$^{7}Li(p,n)^{7}Be$	2.5	$8.90 \cdot 10^{11}$	0.55	0.79
$^{9}Be(p,n)^{9}Be$	4.0	$1.00 \cdot 10^{12}$	1.06	2.12
$^{9}Be(d,n)^{10}B$	1.5	$2.16 \cdot 10^{11}$	2.01	5.81
$^{13}C(d,n)^{14}C$	1.5	$1.81 \cdot 10^{11}$	1.08	6.77

Neutron spectrum tailoring

In any case both the nuclear reactor based neutron sources and the accelerator based neutron sources can not be directly used for BNCT, but they require a suitable moderator/reflector assembly in order to tailor the neutron energy spectrum.

Filtering materials

A good moderator, in the case of BNCT, is represented by a material that can allow for a rather efficient fast to epithermal slowing down, while the neutron removal from the epithermal group (slowing down plus absorption) should occur, on the contrary, at a slower rate: this gives rise to an accumulation of the neutrons in the epithermal energy range. If a simple scheme of three energy groups, fast (E>10 keV), epithermal (1 eV<E<10 keV) and thermal (E<1 eV), is adopted, an important parameter in order to identify a good moderator is the ratio $\Sigma_{s,fn\rightarrow epi}/\Sigma_{r,epi}$, where $\Sigma_{s,fn\rightarrow epi}$ is the fast to epithermal slowing down macroscopic cross section and $\Sigma_{r,epi}$ the removal macroscopic cross section from the epithermal group.[9] Table 2 summarizes the values of that ratio together with some other parameters for different moderators. Materials containing fluorine or magnesium are those presenting the better performance in terms of neutrons accumulation inside of the epithermal energy range during the slowing down from the source energies. In particular the mixture of 30% of Al and 70% of AlF_3 shows the best performance in terms of the ratio $\Sigma_{s,fn\rightarrow epi}/\Sigma_{r,epi}$. The above mentioned ratio corresponds to the ratio that could be obtained between the epithermal and fast neutron flux in an infinite medium with a stationary source generating neutrons at a suitable high energy (enough to establish the asymptotic neutron flux in the fast energy range). With a filtering column of finite size, however, e.g. for a bare 60 cm long cylindrical column with a radius of 30 cm and a fast neutron source entering from one side, the use of the mixture Al/AlF_3 above described leads to a ratio between epithermal and fast neutron flux on the other side equal to 6.8, while using the AlF_3 it is possible to obtain a value of 19.0. This is due to the fact that the larger is the epithermal macroscopic scattering cross section, the

Table 2. Macroscopic cross section of some interesting moderators. More details about the calculation procedure of these parameters can be found in Ref. 9.

	$\Sigma_{s,fn}$	$\Sigma_{s,epi}$	$\Sigma_{s,fn\rightarrow epi}$	$\Sigma_{s,epi\rightarrow th}$	$\Sigma_{r,fn}$	$\Sigma_{r,epi}$	$\dfrac{\Sigma_{s,fn\rightarrow epi}}{\Sigma_{r,epi}}$
Al/AlF_3*	2.47E-1	1.86E-1	1.18E-2	3.25E-3	1.20E-2	3.67E-3	3.23
AlF_3	3.40E-1	2.68E-1	1.23E-2	5.05E-3	1.30E-2	5.35E-3	2.30
MgS	1.55E-1	1.21E-1	5.10E-3	1.26E-3	5.37E-3	2.24E-3	2.28
MgF_2	3.51E-1	3.08E-1	9.83E-3	4.66E-3	1.14E-2	4.78E-3	2.06
^{7}LiF	3.25E-1	2.78E-1	1.21E-2	6.04E-3	1.31E-2	6.20E-3	1.94
Mg_2Si	1.35E-1	1.10E-1	2.32E-3	1.26E-3	2.86E-3	1.43E-3	1.62
$(CF_2)_n$*	2.84E-1	2.98E-1	9.41E-3	5.97E-3	1.07E-2	6.00E-3	1.57
Al	1.12E-1	8.04E-2	2.20E-3	7.45E-4	2.41E-3	1.41E-3	1.56
Al_2O_3	3.82E-1	3.29E-1	1.11E-2	6.92E-3	1.17E-2	7.48E-3	1.48
Al_2S_3	4.80E-2	4.08E-2	1.29E-3	1.83E-4	1.53E-3	9.14E-4	1.42
Bi	2.54E-1	2.61E-1	7.19E-4	4.72E-4	8.13E-4	5.61E-4	1.28
D_2O	2.59E-1	3.23E-1	3.78E-2	3.20E-2	3.92E-2	3.20E-2	1.18
BeO	5.45E-1	6.92E-1	2.58E-2	2.40E-2	3.08E-2	2.40E-2	1.07
Pb	2.72E-1	3.72E-1	7.48E-4	7.08E-4	7.81E-4	7.09E-4	1.05

**A 30% Al plus 70% AlF_3 mixture.*

shorter is the mean free path of the neutrons and consequently the number of particles that can reach the lateral surface of the column and then leave the system. However, the value of the epithermal macroscopic cross section is the 70% of the corresponding value reported for the AlF_3. This (partial) contradiction shows that the choice of the moderator material can not be separated from the geometry restrictions and even from the choice of the reflector material and vice versa.

Suitable actions have to be considered in order to reduce as much as possible the photon background at the irradiation position. This can be effectively obtained introducing one or more photon shields (along the moderator assembly). Thanks to its high mass attenuation coefficient, very good performance from this point of view can be achieved using high purity Bi. In the case of the nuclear reactor based neutron sources many photons are produced during the fission process inside the core or by radioactive decay of the fission products, while for accelerator based neutron sources the product of the neutron production reaction can result in an excited state (instead of the ground state) and produce one or more photons during the de-excitation process. However, in both cases, the more relevant part of the photon component is produced by thermal neutron absorption in the materials of the moderator/reflector assembly.

Like for the photon component, also the thermal neutron flux at the irradiation position should be reduced as much as possible. Thermal neutrons are in fact responsible for a too high radiation dose to the skin, the scalp and the first millimeters of the brain, thus limiting the BNCT treatment performance. To reduce this component, so avoiding the production of any photons, an original solution is that adopted in the Studsvik epithermal neutron beam where a ^6Li filter (the neutron absorption reaction is of the (n,t) type) positioned just at the end of the beam is decreasing the thermal neutron flux to negligible levels.[10]

Reflecting materials

The reflector material surrounds the moderator assembly and its role is that of increasing the number of the neutrons that, after leaving the moderator, following a series of scattering collisions, re-enter the moderator itself. By this way, a material can be considered an efficient reflector if the increase of the flux inside the moderator column is larger than the one which could be obtained by simply replacing the reflector by an equal amount of moderator material.

In some case the escape from the lateral surfaces of the moderator assembly, however, turns out to be somewhat useful, namely if it is concentrated in the fast energy range. A reflector that behaves in this way can be defined as an “active reflector” since it is actively participating to the spectrum tailoring process. On the other hand, a reflector that does not produce a selective action on the neutrons leaving the moderator can be defined as a “passive reflector”: it has no role in the spectrum tailoring process. A type of active reflector is well represented by the Ni: a 5 cm thick Ni reflector surrounding a cylindrical moderator assembly made by AlF_3 produces an epithermal albedo (fraction of the neutrons scattered back from the reflector to the moderator) equal to 75% (the epithermal albedo obtained increasing the AlF_3 column radius by 5 cm it will be equal to only 57%), while the fast albedo is limited to 40% (42%, by increasing the AlF_3 column radius by 5 cm). An example of passive reflector is instead represented by the Pb: for a thickness of the reflector larger than 12 cm, again considering a cylindrical moderator assembly made by AlF_3, this material shows the better performance in terms of the total albedo (that is the albedo for all the neutrons disregarding the energy). In particular a 20 cm thick Pb reflector scatters back the 80% of the neutrons which would otherwise leave the moderator.

The choice of an active reflector is preferable when the fast neutron component is constituted by neutrons of pretty high energy. The moderation until the epithermal energy of such a component would require a too large volume of moderator material. It could be advantageous to let them leave the moderator instead to try taking care of their slowing down: this in fact could result in a too low epithermal neutron flux (even if pretty pure) at the irradiation position. On the contrary, when the fast neutron component is expected to have a mean energy not so far from the upper limit of the epithermal energy range, it could be more useful to keep as large as possible a number of neutrons inside the moderator, disregarding their energy.

Conclusion

Nuclear reactors have been indeed very important in order to start the first clinical trials on BNCT, but if the success of this therapy will be proved it is evident that to move toa wide clinic use it will be necessary to refer to different neutron sources. Accelerator based neutron sources are able to meet all the requirements of a hospital environment. In particular, due to the fact that the mean energy of the neutrons produced is not so far from that of the fission reactions, it is possible to use all the experience accumulated during the design of the moderator/reflector assemblies for the reactor based neutron source to provide an optimized moderator/assembly also for the accelerator based sources.

References

[1] R. F. Barth, A. H. Soloway, J. H. Goodman, R. A. Gahbauer, N. Gupta, T. E. Blue, W. Yang, and W. Tjarks, "Boron neutron capture therapy of brain tumors: an emerging therapeutic modality", Neurosurgery, 3(44), 433-451, 1999

[2] R. G. Fairchild, and V. P. Bond, "Current status of 10-B neutron capture therapy: enhancement of tumor dose via beam filtration and dose rate, and the effects of these parameters on minimum boron content: a theoretical evaluation", Int. J. Radiat. Oncol. Biol. Phys. 11, 831-840, 1985

[3] E. Bisceglie, P. Colangelo, N. Colonna, P. Santorelli, and V. Variale, "On the optimal energy of epithermal neutron beams for BNCT", Phys. Med. Biol. 45, 49-58, 2000

[4] O. K. Harling and K. J. Riley, "Fission reactor sources for neutron capture therapy – a critical review", J. of Neuro-Oncology 62, 7-17, 2003

[5] T. E. Blue and J C. Yanch, "Accelerator-based epithermal neutron sources for boron neutron capture therapy of brain tumors", J. of Neuro-Oncology 62, 19-31, 2003

[6] J. M. Verbeke, K. N. Leung and J. Vujic, "Development of a sealed-accelerator-tube neutron generator", Appl. Radiat. Isot. 53, 801-809, 2000

[7] H. Koivunoro, T. P. Lou, J. Reijonen and K. N. Leung, "A compact neutron generator based on D-D or D-T fusion reaction for BNCT", in Research and Development in Neutron Capture Therapy, edited by W. Sauerwein et al. (Monduzzi editore, Bologna, 2002), 313-318, 2002

[8] http://www.lbl.gov/Tech-Transfer/collaboration/techs/lbnl1764.html

[9] B. Montagnini, N. Cerullo, V. Giusti and M. Sumini, "Feasibility of a small accelerator driven subcritical reactor for BNCT applications", J. Nucl. Sci. Technol. Supplement 1, 389-393, March 2000

[10] V. Giusti, P. M. Munck af Rosenschold, K. Skold, B. Montagnini and J. Capala, "Monte Carlo model of the Studsvik BNCT facility: description and validation", Med. Phys. 30 (12), 3107-3118, 2003