



## Re-design of irradiation channels in americium-beryllium (Am-Be) neutron irradiation facility in NNRI using MCNP

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### ABSTRACT

The Americium-Beryllium neutron irradiation facility at the National Nuclear Research Institute (NNRI), Ghana, was re-designed with a 20 Curie source using MCNP5 to investigate the maximum amount of flux that is produced by the source. The main objective was to enable us harness the maximum amount of flux for optimization of neutron activation analysis. The design was compared with the values recorded in the existing Am-Be source design. The thermal neutron flux ranges from  $(1.80 \pm 0.0008) \times 10^6$  n/cm<sup>2</sup>.s to  $(2.28 \pm 0.0008) \times 10^5$  n/cm<sup>2</sup>.s. The epithermal neutron flux ranges from  $(8.20 \pm 0.0008) \times 10^5$  n/cm<sup>2</sup>.s to  $(3.73 \pm 0.0008) \times 10^4$  n/cm<sup>2</sup>.s and that of the fast neutron flux ranges from  $(4.20 \pm 0.0008) \times 10^5$  n/cm<sup>2</sup>.s to  $(1.72 \pm 0.0008) \times 10^4$  n/cm<sup>2</sup>.s. The  $k_{eff}$  of the Am-Be source design was obtained as  $0.00115 \pm 0.0008$ .

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### Introduction

The development of appropriate instrumentation to perform neutron activation analysis can be very useful in the simultaneous determination of elements in complex samples of several matrices. Neutron activation analysis (NAA) method provides a highly sensitive non-destructive technique for the elemental composition determination in samples. The energy and intensity of the neutrons produced by the neutron source facilities are very essential in NAA. The IAEA supplied and installed 20 Ci <sup>241</sup>Am-Be neutron irradiation facility at National Nuclear Research Institute (NNRI) of Ghana Atomic Energy Commission in 1977 for NAA (Osae and Amoh, 1996; Osae and Akoto-Bamford, 1986; Tetteh, 1980). Figure 1 and 2 show schematic diagrams of the Am-Be source at NNRI. The isotropic Am-Be source is however under utilised due to its lower neutron fluence rate relative to that of the Ghana Research Reactor -1 facility (Asamoah et al, 2011). The main advantage of the Am-Be neutron source irradiator is its very stable neutron flux, thus eliminating the need for standard material in the measurement of induced activity in samples (Zevallos-Chavez and Zamboni, 2005).

Am-Be neutron source is widely employed as calibration source for neutron instrumentation, and portable source for a variety of applications. It is well known that the neutron source also gives off penetrating  $\gamma$ -rays of 4.438MeV which are mainly associated with the neutron group leaving <sup>12</sup>C in the first excited state in the <sup>9</sup>Be( $\alpha$ , n)<sup>12</sup>C reaction. There is a compacted mixture of AmO<sub>2</sub> and <sup>9</sup>Be fine powder in the source construction. It is obvious that the cluster size, mixture ratio and compacted density of the active zone and the physical size of the source have strong influence on the neutron yield and the finer structure in the emerging neutron spectrum (Zhenzhou et al, 2007). The irradiator can be used as an alternative to the research reactor

most especially considering the cost of design, construction and operation of a research reactor.

Am-Be neutron source is an important neutron source for NAA; relatively inexpensive, easy to shield and portable besides producing stable neutron flux. The neutron flux distribution in the irradiation sites of the Am-Be neutron source has been determined by experiment by Mensimah et al (2011) and by theoretical means using MCNP5 by Asamoah (2011). The main objective of this research work is to redesign the irradiation channels in order to get a higher neutron fluence rate that will enable smaller samples to be irradiated thereby reducing the pressure on the GHARR-1.

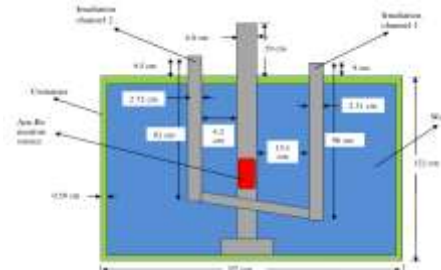


Fig 1 Schematic vertical cross-sectional view of the Am-Be neutron source facility at NNRI.

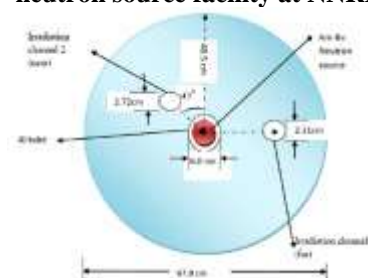
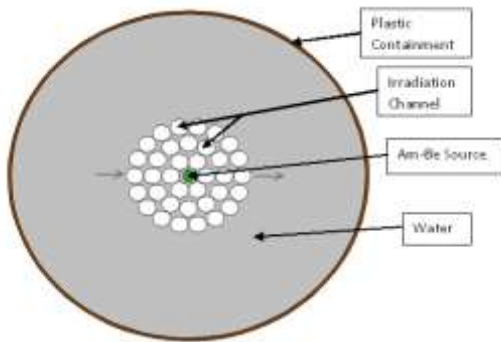


Fig 2 Horizontal cross-section of the Am-Be neutron source facility at NNRI.

**Methodology**

The simulation of the neutron particle histories was performed in the computer simulation laboratory of NNRI in March, 2001. The MCNP5 code was used to model thirty-six channels around the source in order to monitor the neutron fluxes in all directions. The Am-Be source design contains an AmO<sub>2</sub> compound in a mixture of beryllium powder in a double layered stainless steel shell (Zhenzhou et al, 2007). The source is surrounded by water for moderation and shielding effect. The whole system is housed in a plastic containment. Fig.3 shows a well labeled cross-sectional view of the single source Am-Be design that was used in the study. The arrows in the diagram show the pattern by which the radial neutron fluxes distribution were monitored.



**Fig.3. Cross-sectional view of the single Am-Be neutron source irradiation facility**

125000 particles were simulated for 400 cycles to enable the simulation of 50000000 particle histories. The more the particles simulated, the less the error margin. The Monte Carlo method can exhibit a small bias in the calculated  $k_{eff}$ . If at least 5000 (preferably 10,000) neutrons per  $k_{eff}$  are used, this bias should not be a factor in MCNP results. The larger the number of neutrons per  $k_{eff}$  cycle requested in the first entry on the KCODE card, the smaller the bias. Therefore, it is wise to run as many neutron histories as you can afford, combined with using at least 200 active  $k_{eff}$  cycles (X-5 Monte Carlo Team, 2003). For this reason, we decided to run 125000 particles. An initial  $k_{eff}$  guess of 0.001 was used based on the knowledge that the facility is a subcritical system and the previous computations performed on the facility using MCNP5 (Asamoah et al, 2011). The initial guess of  $k_{eff}$  only affects the creation of fission source points for the second  $k_{eff}$  cycle. A severe underestimation of the initial guess will result in the creation of too many source points and vice versa. Source points in future cycles are unaffected by the initial guess of  $k_{eff}$ . Thirty-six channels were created around the source in concentric ring pattern to monitor the flux in every direction around the source (fig 3). Radial volume flux tallies (F4) were created in the channels to get the desired flux. The results from the tallies retrieved from the output file are normalized using the MS-Excel worksheet and the results obtained are interpreted into graphs as shown in fig.4. The unnormalised results were then normalized to get the actual fluxes. To accomplish this, some parameters such as the fission q-value and the loss to fission which are included in the output are called from the MCNP output to calculate the normalization factor. The search menu was used to find these parameters. The conversion factor  $C$  was used in the interpretation of the MCNP tally results in the output file (X-5 Monte Carlo Team, 2003).

$$C = \left( \frac{1J/sec}{watts} \right) \left( \frac{1MeV}{1.60205e^{-13}J} \right) \left( \frac{fission}{190.83MeV} \right) = 3.2709 * 10^{10} \text{ fission/watts} \cdot \text{sec} \tag{1}$$

The source strength of the reactor is calculated by the factor

$$3.2709 * 10^{10} \text{ fission/watts} \cdot \text{sec} * P(\text{watts}) \tag{2}$$

where P is the power of the Am-Be source. The normalization factor for the unnormalised tallies is calculated by

$$\phi_i = \frac{3.2709 * 10^{10} (\text{fission/watts} \cdot \text{sec}) * P(\text{watts}) * \bar{\nu} (\text{particles/fission}) * \text{tally}(\text{cm})}{\text{volume}(\text{cm}^3)} \tag{3}$$

where  $i$  represents the energy group of the neutron being analysed, i.e.; thermal, epithermal or fast and  $\bar{\nu}$  is the number of neutrons/fission and it is given by (X-5 Monte Carlo Team, 2003)

$$\bar{\nu} = \frac{1}{\text{loss to fission}} \tag{4}$$

**Theory**

MCNP5 can be used to theoretically simulate the random process which characterises the interaction of nuclear particles with materials. It is useful in solving problems which are difficult to approximate with deterministic methods. The individual probabilistic events that comprise a process are simulated sequentially while the probability distributions governing the events are statistically sampled to describe the total phenomenon.

The MCNP5 codes tracks each particle from its birth to the time of its death (absorption or capture). As more particle histories are followed, the neutron and photon distributions become better known (X-5 Monte Carlo Team, 2003).

The probability of a first collision for a particle between  $l$  and  $l+dl$  along its line of flight is given by (X-5 Monte Carlo Team, 2003)

$$p(l)dl = e^{-\Sigma_t l} \Sigma_t dl \tag{5}$$

where  $\Sigma_t$  is the total macroscopic cross section of the medium and is interpreted as the probability per unit length of a collision. Setting  $\xi$  the random number on [0, 1], to be (X-5 Monte Carlo Team, 2003)

$$\xi = \int_0^l e^{-\Sigma_t s} \Sigma_t ds = 1 - e^{-\Sigma_t l} \tag{6}$$

It then follows that 
$$l = -\frac{1}{\Sigma_t} \ln(1 - \xi) \tag{7}$$

But because  $1 - \xi$  is distributed in the same manner as  $\xi$  and hence may be replaced by  $\xi$ , we obtain a well known expression for the distance to collision as outlined in the MCNP manual. (X-5 Monte Carlo Team, 2003)

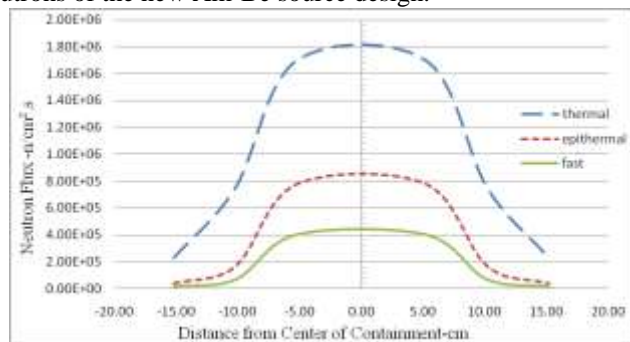
$$l = -\frac{1}{\Sigma_t} \ln(\xi) \tag{8}$$

## Results and discussion

After the 400 cycles and 50 million particle histories of the MCNP simulation the result obtained was  $0.00115 \pm 0.0008$  which compares well with the value of  $0.00118 \pm 0.0008$  (Asamoah et al, 2011).

### Neutron flux distribution in the single Am-Be source design

The average neutron flux yield in the first irradiation channel of the Am-Be source at NNRI are  $(4.476 \pm 0.007) \times 10^3$  n/cm<sup>2</sup>.s,  $(6.57 \pm 0.02) \times 10^2$  n/cm<sup>2</sup>.s and  $(3.70 \pm 0.02) \times 10^2$  n/cm<sup>2</sup>.s for thermal epithermal and fast neutron flux respectively (Asamoah et al, 2011). The average neutron flux yield for the other irradiation channel are also  $(1.9839 \pm 0.0014) \times 10^4$  n/cm<sup>2</sup>.s.,  $(5.945 \pm 0.007) \times 10^3$  n/cm<sup>2</sup>.s and  $(2.611 \pm 0.004) \times 10^3$  n/cm<sup>2</sup>.s for thermal epithermal and fast neutron flux respectively (Asamoah et al, 2011). Fig.3 shows the graph of the radial neutron flux distribution of the thermal, epithermal and fast neutrons of the new Am-Be source design.



**Fig 4. Radial neutron flux distribution of the single 20Ci Am-Be neutron source design**

From the graph, the neutron fluxes were equally distributed on both sides of the single source obeying a Gaussian distribution. The thermal neutron flux ranges from  $(2.28 \pm 0.0008) \times 10^5$  n/cm<sup>2</sup>.s to  $(1.80 \pm 0.0008) \times 10^6$  n/cm<sup>2</sup>.s. The epithermal neutron flux ranges from  $(8.20 \pm 0.0008) \times 10^5$  n/cm<sup>2</sup>.s to  $(3.73 \pm 0.0008) \times 10^4$  n/cm<sup>2</sup>.s and that of the fast neutron flux ranges from  $(4.20 \pm 0.0008) \times 10^5$  n/cm<sup>2</sup>.s to  $(1.72 \pm 0.0008) \times 10^4$  n/cm<sup>2</sup>.s. The thermal flux at 5.23cm, 10.23cm and 15.19cm from the source are  $(1.78 \pm 0.0008) \times 10^6$  n/cm<sup>2</sup>.s,  $(7.49 \pm 0.0008) \times 10^5$  n/cm<sup>2</sup>.s and  $(2.31 \pm 0.0008) \times 10^5$  n/cm<sup>2</sup>.s respectively. These values recorded at these randomly selected distances were all higher than the average thermal flux values as reported by Asamoah et al, 2011. Epithermal neutron flux of  $(7.78 \pm 0.0008) \times 10^5$  n/cm<sup>2</sup>.s,  $(1.64 \pm 0.0008) \times 10^5$  n/cm<sup>2</sup>.s and  $(3.73 \pm 0.0008) \times 10^4$  n/cm<sup>2</sup>.s respectively was recorded at similar distances from the source. Fast neutron flux followed the same trend at similar distances from the source; (ie.  $(4.02 \pm 0.0008) \times 10^5$  n/cm<sup>2</sup>.s,  $(6.76 \pm 0.0008) \times 10^4$  n/cm<sup>2</sup>.s and  $(1.72 \pm 0.0008) \times 10^4$  n/cm<sup>2</sup>.s. It was observed that the flux values were higher in positions closer to the core due to their proximity (The neutrons do not need to be slowed down to attain thermal energy so the closer it is to the source, the higher the flux). The results recorded for thermal, epithermal and fast flux were all higher than that reported in the two irradiation channels of the Am-Be source presently in NNRI as reported by Asamoah et al. Consequently, irradiation channels could be installed closer to the source to be able to harness higher flux. This will enable smaller sample size materials to be irradiated. The Am-Be source is used in other facilities for prompt gamma neutron activation analyses (PGNAA). With the amount of fast neutron

flux recorded, a prompt gamma detection system could be installed to enable prompt gamma neutron activation analyses. Prompt gamma neutron activation analyses are a well known rapid non-destructive and experimental technique that compliments conventional NAA. The PGNAA can be applied to the Am-Be source to determine light elements (eg. H, B, N, Si, P, S and Cl).and elements with a large neutron capture cross section (eg. Cd, Sm and Gd) by irradiating them continuously with neutrons.

## Conclusion

It was realized that neutron flux obtained close to the source was higher than that reported in the two irradiation channels of the Am-Be neutron source in the NNRI. Therefore the installation of new irradiation channels in positions closer to the source will help harness the high neutron fluence.

A prompt gamma detection system could be installed to enable prompt gamma neutron activation analyses due to the amount of fast flux recorded thereby providing an added advantage to the Am-Be source (being able to determine light elements and elements with a large neutron capture cross section)

With the high neutron flux recorded, the new model can be adopted by universities in Africa and third world countries, so as to enable such countries to harness the benefits of nuclear science in terms of neutron activation analyses.

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