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Feasibility study of photo-neutron flux in various irradiation channels of Ghana MNSR using a Monte Carlo code

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ABSTRACT

Computer simulation was carried out for photo-neutron source variation in outer irradiation channel, inner irradiation channels and the fission channel of a tank-in-pool reactor, a Miniature Neutron Source Reactor (MNSR) in sub-critical condition. Evaluation of the photo-neutron was done after the reactor has been in sub-critical condition for three month period using Monte Carlo Neutron Particle (MCNP) code. Neutron flux monitoring from the Micro Computer Control Loop System (MCCLS) was also investigated at sub-critical condition. The recorded neutron fluxes from the MCCLS after investigations were used to calculate the power of the reactor at sub-critical state. The computed power at sub-critical state was used to normalize the un-normalized results from the MCNP.

1. Introduction

1.1. Photo-neutron physics

In a nuclear reactor that has been operated, another source of neutrons becomes significant. This source of neutrons may be produced by the interaction of a gamma ray with a deuterium nucleus. This reaction is commonly referred to as a photo-neutron reaction because it is initiated by electromagnetic radiation and results in the production of a neutron. The (γ, n) reaction which results in photo-neutron production is shown below:

$$^{2}_{1}H + ^{0}_{0}\gamma \rightarrow (^{2}_{1}H) \rightarrow ^{1}_{1}H + ^{1}_{0}n$$

This tells us that any source based on (γ, n) reaction is called photo-neutron source. Such reaction occurs due to interactions of mainly absorption of both prompt and delayed gamma rays with reactor materials at specific energy thresholds (Hetrick, 1971). Some of this photo-neutron production is due to very long-lived fission product decays. In general, reactors have no sources of neutrons other than the ones generated by spontaneous induced fission in fuel. The photo-neutron emission is very significance in MNSRs because it gives appreciable contribution to the start-up of neutron source after reactor shutdown. Photo-neutrons also make small contributions to the overall reactivity load in reactors involving the use of beryllium. The photo-neutron source usually persists after the reactor is shutdown and the prompt and delayed neutrons die down. The number of photo-neutrons produced by fission product; gamma photons is of importance during design and operation of reactors since their behaviour is similar to delayed neutrons in their effect on the reactor kinetics (Khermis, 2001).

1.2. Photo-neutron source in Ghana MNSR

The side (annular), bottom and top reflectors of GHARR-1 is beryllium metal alloy. They function not only as reflectors but also as moderators. In addition to its good reflection properties, it also produces photo-neutrons from the interactions of gamma rays through the (γ, n) reaction and provides a useful startup source once the core is activated. Long-term reactivity control is exercised by periodically increasing the thickness of this reflector to compensate for reactivity loss caused by fuel burn-up, samarium and xenon poison. Under normal operating conditions of the reactor, the top shims need to be added less frequently than once every one and half years. In addition to the initial excess reactivity of the core, the presence of the shims ensures that the core life of the reactor fuel elements shall be longer than 10 years. Fig. 1 shows the schematic diagram Ghana MNSR depicting how the reactor core is surrounded by beryllium annular and slab reflectors (Akaho et al., 2000).

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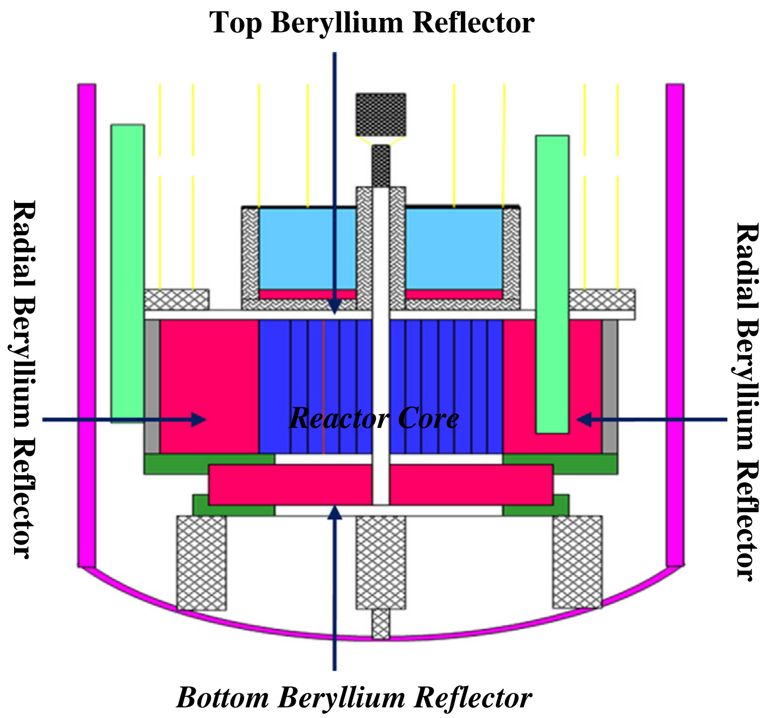

Fig. 1. Schematic diagram depicting MNSR core surrounded by beryllium annular and slab reflectors.

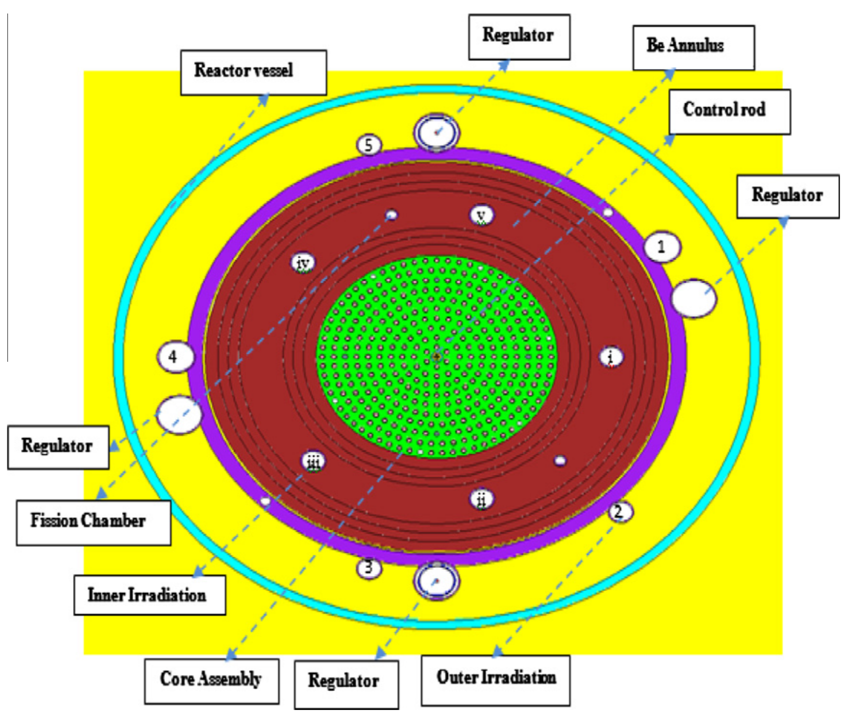

Fig. 2. MCNP5 plot of GHARR-1 core configuration (Anim-Sampong, 2001).

1.3. Epigrammatic description of Ghana MNSR reactor

The reactor GHARR-1 is a 30 kW (th) tank-in-pool type, low power research reactor, operating presently with a highly enriched uranium (HEU) core. The MNSR is a Chinese version of the Canadian SLOWPOKE. The reactor is inherently safe with a very strong capability of power self-limitation. These characteristics have been confirmed through various transient experiments. The total cold excess reactivity is 4.0 mK and it is controlled by 1 central rod of worth 6.8 mK, which ensures a shutdown margin of about 3 mK. This is sufficient to maintain the reactor in safe shutdown condition. It utilizes light water as coolant and moderator and cooled by natural convection. The reactor is used mainly for Research and Development (R&D), Neutron Activation Analysis

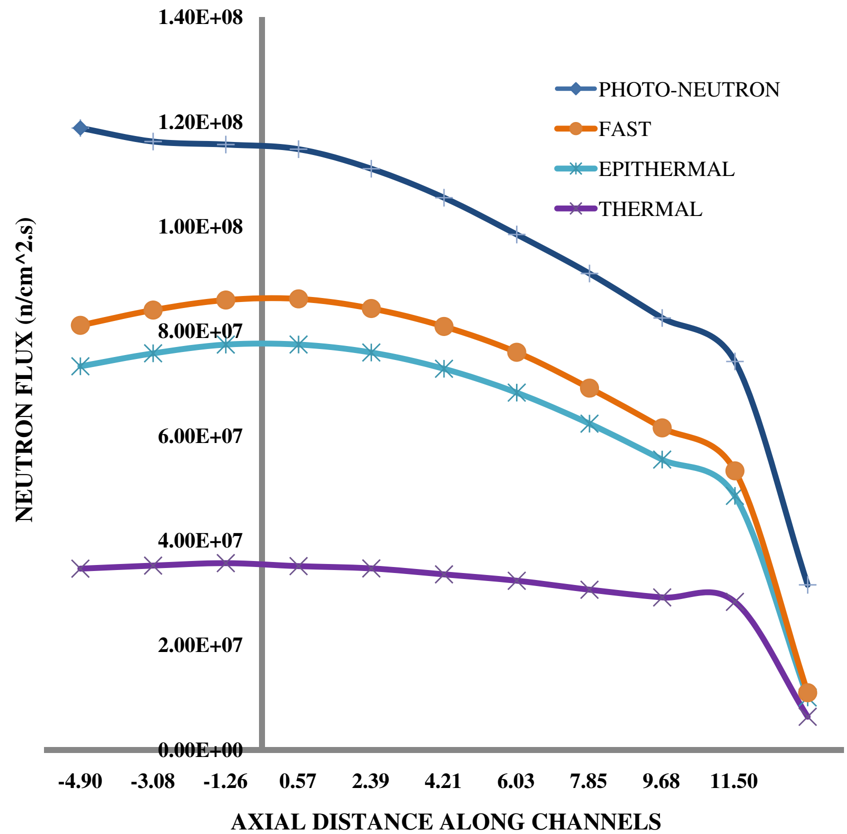

Fig. 3. Variation of neutron fluxes within the inner irradiation channel for GHARR-1 at sub-critical state using MCNP.

(NAA), production of short-lived radioisotopes and Education and Training (E&T). The peak or maximum thermal neutron flux in the core and its inner channels at full power operation is 1×10^{12} n cm^{-2} s^{-1}. The reactor assembly consists of the reactor core, beryllium (Be) reflector, small fission chambers for detecting neutron fluxes and thermocouples for measuring inlet and outlet temperatures of the coolant. There are five inner irradiation tubes installed within the beryllium annulus. Five outer irradiation tubes are also installed outside the beryllium annulus (Akaho et al., 2000).

1.4. Physical model of the GHARR-1 reactor

The physical Monte Carlo model of the GHARR-1 reactor was done following the approaches used in the 3-D combinatorial and generalized geometry methods applied in MCNP geometry modelling (Anim-Sampong, 2001). Thus different geometries in planar, conical, spherical and cylindrical configurations of the various zones, sections and materials such as the fuel assembly, control systems, reflectors, irradiation channels shim tray, reactor vessel, reactor pool and other structural components were modelled accordingly. Further use was made of available design data of structural components and materials of the reactor. The centre of the GHARR-1 core assembly which has a cylindrical configuration with ten fuel lattices concentrically arranged about the central control rod guide was chosen as the geometrical midpoint for the Monte Carlo model. Fig. 2 shows the MCNP plot of GHARR-1 core configuration.

1.5. 3-D Monte Carlo model analyses

MCNP is a popular, versatile multipurpose Monte Carlo particle transport code used worldwide. It has the capability to model and treat different geometries in 3-D and also simulate the transport behaviour of different particles. Additionally, MCNP has the ability to treat complex nuclear interaction processes (Briesmeister, 2000). The Monte Carlo method is a numerical procedure for solving mathematical problems based on statistical (or probability) theory. The procedure of Monte Carlo requires extensive computer time since it is necessary to follow the histories of many neutrons through a large number of collisions in order that the results may have statistical significance (Glasstone and Sesonske, 1994).

2. Theory

2.1. Monte Carlo calculations for GHARR-1

2.1.1. Computation for the sub-critical power

The maximum rated thermal power of Ghana MNSR is 30 kW which is equivalent to maximum thermal neutron flux of 1.0×10^{12} n cm^{-2} s^{-1}. The Monte Carlo normalization expression for normalizing the un-normalized MCNP results of GHARR-1 was evaluated using the critical rated power of the MNSR. To obtain a new rated power at sub-critical condition; Eq. (1) was used. The new computed rated power was used to replace the thermal rated power used in the normalization expression. The rated thermal power, P in Watts (W) which is specific to the reactor type understudy can be expressed as

$$P = 3.0 \times 10^{-8}\phi \tag{1}$$

Eq. (1) can be simplified to obtain the thermal neutron flux as shown in the following equation

$$\phi = \frac{P}{3.0 \times 10^{-8}} \tag{2}$$

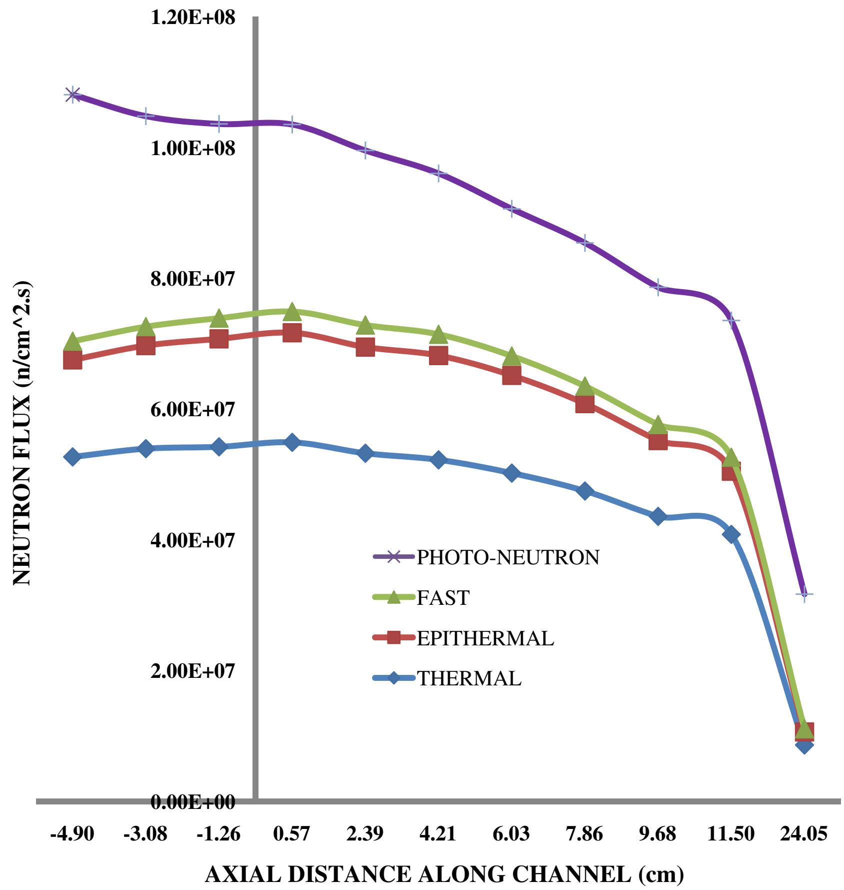

Fig. 4. Variation of neutron fluxes within the outer irradiation channel for GHARR-1 at sub-critical state using MCNP.

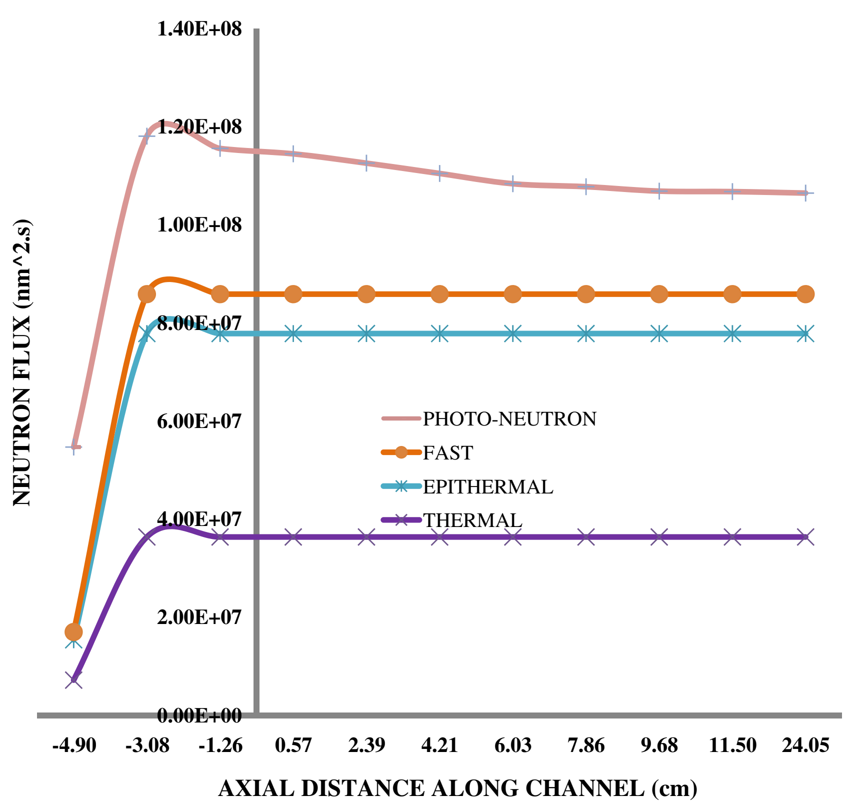

Fig. 5. Variation of neutron fluxes within the fission channel for GHARR-1 at sub-critical state using MCNP.

P and ϕ are the thermal rated power and thermal neutron flux respectively (Akaho et al., 2000). The research based it reference on the thermal power and thermal neutron flux because the reference reactor which is Ghana MNSR is a thermally based reactor.

2.1.2. Normalization expressions

The un-normalized results obtained after the MCNP simulation was normalized to get the actual neutron fluxes. To accomplish this, some parameters such as the neutron fission q-value, loss to fission ratio provided in the MCNP code were utilized to calculate the normalization factor as described in the MCNP manual and presented herein. The conversion factor, C below was used in the conversion

$$C = \left(\frac{1\ \mathrm{J/s}}{\mathrm{W}}\right)\left(\frac{1\ \mathrm{Mev}}{1.60205\mathrm{E}^{-13}\ \mathrm{J}}\right)\left(\frac{\mathrm{fission}}{180.88\ \mathrm{MeV}}\right) \tag{3}$$

$$C = 3.467\mathrm{E} + 10^{\mathrm{fission}}/\mathrm{Ws}$$

The source strength of the reactor is calculated by the factor

$$\left(3.467 \times 10^{10} \times P \times \bar{\nu}\right)\ \mathrm{neutrons/s} \tag{4}$$

where P is the newly calculated power. P was calculated as a new power after shutdown using Eq. (2). For a steady state operation, the number of neutrons/fission $\bar{\nu}$ is approximately 2.597.

The normalization factor for the un-normalized tallies was calculated using the expression below.

$$\frac{3.467\mathrm{E} + 10(\mathrm{W})\mathrm{tally}^{*}\bar{\nu}}{\mathrm{volume}} \tag{5}$$

where $\bar{\nu}$ is the number of neutrons/fission and it is given by

$$\bar{\nu} = \frac{1}{\mathrm{loss\ to\ fission}} \tag{6}$$

(CCC-700/MCNP4C, 2000).

3. Monte Carlo code simulation

3.1. MCNP simulation method

A preliminary Monte Carlo code simulation was done using the various reactor parameters (i.e., power and neutron flux) for the MCNP input deck. Neutron transport simulations were run for 200,000,000 particle histories (500,000 particles and 400 criticality cycles) with an initial criticality (keff) guess of 1.004. Twenty cycles were initially skipped before the actual runs.

3.2. Computed power for normalization

Data of neutron fluxes from the reactor Micro Computer Control Loop System (MCCLS) were recorded after the reactor has been under sub-critical condition for a period of one week. The recorded neutron flux results within that period were used to compute for a new thermal rated power using Eq. (2). The results were then normalized using Eq. (5). The results obtained for the neutron fluxes from the various irradiation channels were plotted against the axial distance along the channels.

4. Results and discussion

The results from the MCNP code after simulation was used to prove the decay variation of neutron fluxes at the various irradiation channels. This proves were done by using graphical representation. A graph of neutron flux versus axial distance along the irradiation channels were plotted as shown in Figs. 3–5 for the three main irradiation channels for the Ghana MNSR reactor. The graph depicts the decay trend of neutron flux at subcritical condition and how the neutron flux decreases as it moves away from the centre of the reactor core. There is a little similarity for the inner and the outer irradiation channels as shown from Figs. 3 and 4. The minor similarities for the outer irradiation channel compare to the inner irradiation channel is due to the slight closeness of the outer channel to the core as compare to the inner channel which is more and more closer to the core. So the outer channel which might have little more fission heat below that of the inner channel. The variation of neutron flux in the fission channel is far different from the inner and outer channel due it positional distance away from the centre of the core. The flux results were obtained by segmenting the channel into cells. The results from MCNP were taken from the cell closer to the core.

5. Conclusion

In conclusion, as the decay trend depicts from graphs, it is agreed without any bias that the reactor can also be used to perform the task being performed by the Americium Beryllium Source (ABS) facility in the National Nuclear Research Institute (NNRI). Since the reactor's residual flux is slightly similar to flux generated by the ABS. The function of the ABS facility is for photo-Neutron Activation Analysis. The research has also proved without doubt that the existing residual heat flux can still facilitate the initialization of the reactor after shut-down of almost 14 years of operation.

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