

# Neutronic Calculations of Ghana Research Reactor-1 LEU Core

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## Commentary

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### ABSTRACT

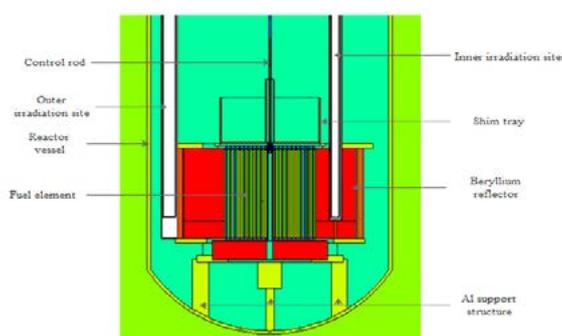
Presenting neutronic calculations relating to the Ghana Research Reactor-1 (GHARR-1) LEU core is the goal of this article. These are key factors for maintaining safe and reliable core operations of the reactor. The neutronic parameters are: multiplication factor ( $k_{eff}$ ), clean cold core excess reactivity ( $\rho_{ex}$ ), control rod worth (CRW), neutron flux distribution in the inner and outer experimental channel and the delayed neutron fraction. A Monte Carlo-N-Particle (MCNP5) neutronic simulation code was employed in this study. These calculations were done before the final core loading and the subsequent experimental Zero Power Test (ZPT) of the GHARR-1. The excess reactivity of the LEU core was calculated to be  $4.03 \pm 0.05$  mk. This study indicates that, the reactor is operated under safe and reliable condition.

## INTRODUCTION

Ghana Research Reactor-1 is a miniature neutron source reactor (MNSR), developed by the Chinese Institute of Atomic Energy (CIAE). The reactor is a compact research reactor designed to mimic the Canadian SLOWPOKE reactor design **Figure 1**. The MNSR is favored all over the world because of its inherent safety <sup>[1]</sup>. The reactor has simple auxiliary facilities, reliable cooling systems and shielding. The MNSR which is a 34 kW power tank-in-pool type of research reactor that uses low enriched uranium (13.0% U-235) as fuel, beryllium as reflector and light water as coolant, moderator and as a biological shielding. Primarily, the MNSR is used for teaching and learning and the neutron source for research activities such as neutron activation analysis (NAA) <sup>[2,3]</sup>. The reactor core consist of the fuel cage, the control rod guide tube which houses the only control rod, fuel rods, dummy rods and tie rods. Heat generated during reactor operation is removed by natural convection. The MNSR has five inner experimental channels, four reactivity regulators and two fission chambers which are located inside the annulus beryllium reflector while five outer experimental channels are located outside, as shown in **Figure 2**.

Technical summary of the reactor is presented in **Table 1**.

Maximum thermal neutron flux is obtained in the inner experimental sites that correspond to clean cold excess reactivity of approximately 4 mk ( $mk=10^{-3} (\Delta k/k)$ ). To compensate a long-term reactivity of the reactor, one or more beryllium shim is added to the top shim tray. A Monte Carlo-N-Particle (MCNP5) code was employed to simulate the excess reactivity ( $\rho_{ex}$ ), shutdown margin (SDM), average flux distribution in the inner and outer experimental sites, control rod worth (CRW) and delayed neutron fraction.



**Figure 1.** Vertical cross-section view of GHARR-1 MNSR

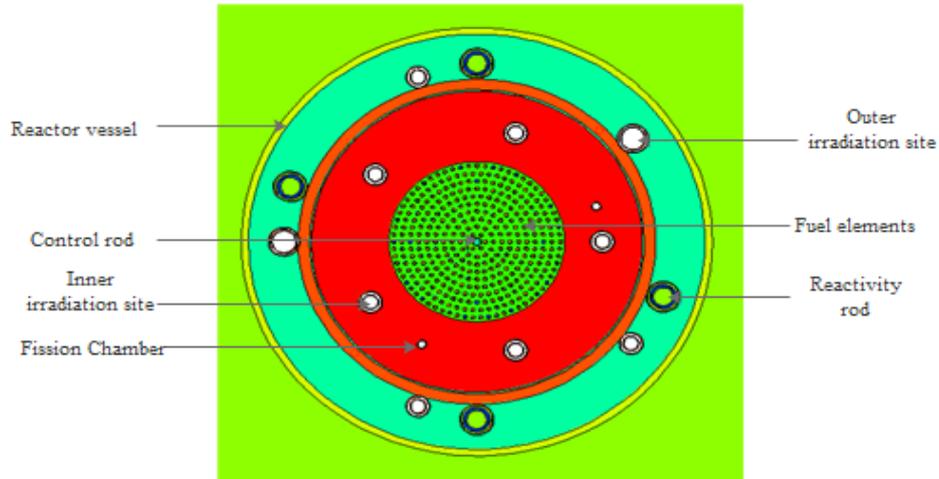


Figure 2. Horizontal cross-section view of GHARR-1 MNSR

Table 1. Technical summary of GHARR-1 MNSR [CIAE].

Parameter	Description
<b>Plant</b>	
Reactor type	Tank-in-pool
Thermal power	34 kW
Cooling system	Natural convection
Inner experimental sites	5
Outer experimental sites	5
<b>Fuel Element</b>	
Fuel	UO <sub>2</sub>
<b>Enrichment</b>	
Density	13.00%
Fuel diameter	10.6 gcm <sup>-3</sup>
Length	4.3 mm
Cladding Material	230 mm
<b>Core</b>	
Core diameter	230 mm
Core height	230 mm
Number of concentric circles	10
Number of rod positions	354
Number of fuel rods loaded	335
Number dummy rods	15
Tie rods	4
Excess reactivity	~4 mk
<b>Control rods</b>	
Number of control rods	1
Number of reactivity rods	4
Absorber material	Cadmium
<b>Reflector</b>	
Side reflector	204 mm thick
Bottom reflector	50 mm thick
Top reflector	Variable thickness
Material	Metallic Beryllium

## METHODOLOGY

A 3-D Monte Carlo transport code of GHARR-1 LEU-core reactor was modified to reflect the changes in the equipment. Nuclear data for fissile and non-fissile isotopes associated with components of the physical model were selected from a number of evaluated data libraries; ENDF/B-VI, ACTL, ACTI, etc. A special scattering,  $S(\alpha, \beta)$  was applied specifically to treat thermal scattering in the metallic beryllium reflector and in light water for water regions of GHARR-1 [4]. Neutronic analysis were performed using

condensed three-group energy structure; 0 – 0.55 eV, 0.55 eV – 0.1 MeV and 0.1 MeV – 20 MeV for thermal, epithermal and fast neutrons respectively. The computations were performed using 600000 source neutrons per cycle for a total of 530  $k_{eff}$  cycles and an initial  $k_{eff}$  guess of 1.004 was chosen for which 30  $k_{eff}$  cycles were skipped to allow source neutrons to adjust to equilibrium before final  $k_{eff}$  was estimated. The criticality source card with initial criticality was used to estimate the  $k_{eff}$  and its corresponding excess reactivity  $\rho_{ex}$  with all the fuel elements as fission source points. TOTNU NO card was also used to estimate the criticality for average prompt neutrons for all the fissionable nuclides. The model also provided for the simulation of average neutron flux distribution in the inner and outer irradiation sites by evoking the F4 tally which calculates the flux distribution using the track length within the specified energy ranges in a unit volume [5]. The out-put data were then normalized using the expression in equation 1.

## THEORY

### Normalization Expression

The un-normalized out-put obtained after the MCNP simulation was normalized to get the actual neutron fluxes. This is accomplished by utilizing some parameters like loss to fission ratio, neutron fission q-value provided in the MCNP code.

$$C = \left( \frac{1J/s}{W} \right) \left( \frac{1MeV}{1.6020 \times 10^{-13} J} \right) \left( \frac{fission}{180.88 MeV} \right) \quad (1)$$

$$C = 3.467 \times 10^{+10} \frac{fission}{Ws}$$

Then the normalization factor for the un-normalized tallies obtained from the MCNP simulation was calculated by the expression below [3];

$$(3.467 \times 10^{+10} \times P \times tallies^* \times \bar{v}) \text{ neutrons / cm}^2 s \quad (2)$$

$$\text{where } \bar{v} = \frac{1}{\text{loss to fission}} \left( \frac{\text{neutron}}{\text{fission}} \right), \text{ power} = P(W) \text{ and tallies}^* (\text{per unit volume})$$

### Delayed Neutron Fraction

In Monte Carlo calculations, consider the introduction of a neutron with properties  $\vec{r} \in E'$  and in a critical reactor system with zero power. This neutron will induce fission thereby producing other neutrons and these neutrons will in turn cause another fission which lead to further neutrons production. The number of fission produced in this way will approach a limit, which is given by the iterated fission probability which is proportional to the adjoint function.

Using the prompt method,

$$\beta_{eff} = \frac{\chi_d^{v_d}}{\chi^v} = 1 - \frac{\chi_p^{v_p}}{\chi^v} \quad (3)$$

$$\beta_{eff} = 1 - \frac{k_p}{k} \quad (4)$$

Where  $k$  and  $k_p$  are the eigen-values for all neutrons and only prompt neutrons respectively.

### Shutdown Margin

The shutdown margin characterizes the core multiplication capabilities in the shutdown state and also relates to the rate at which the power level may be reduced in an emergency shutdown. Shutdown margin  $\rho_{sm}$  can be obtained from the relation [6].

$$\rho_w = \rho_{ex} + \rho_{sm} \quad (5)$$

where  $\rho_w$ , control rod worth,  $\rho_{ex}$  the excess reactivity and  $\rho_{sm}$  the shutdown margin,  $k_{in}$  and  $k_{out}$  being the eigenvalues for total rod insertion and withdrawal respectively. Hence,

$$\rho_w = \frac{k_{out} - k_{in}}{k_{out}} * k_{in} \quad (6)$$

which gives,

$$\rho_{sm} = \frac{k_{out} - k_{in}}{k_{out}} * k_{in} - \rho_{ex} \quad (7)$$

**Table 2.** Calculated neutronic parameters for GHARR-1 LEU core.

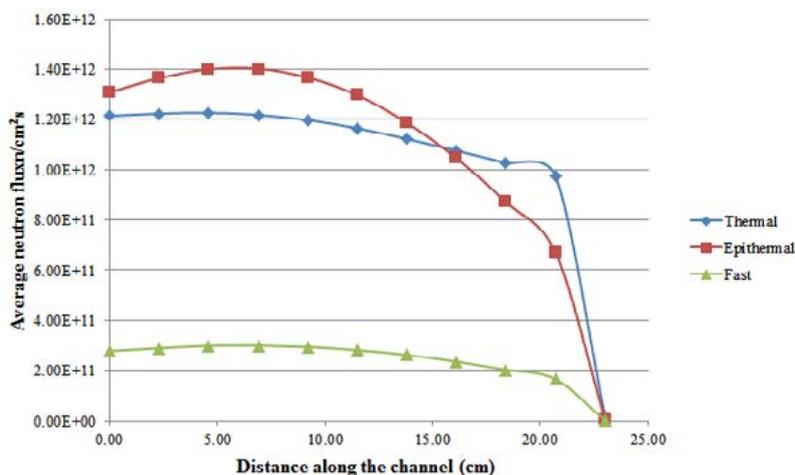
Parameter	LEU
$k_{eff}$ (rod out)	$1.00405 \pm 0.00005$
$k_{eff}$ (rod in)	$0.99687 \pm 0.00005$
$k_p$	$1.00365 \pm 0.00005$
Core excess reactivity, $\rho_{ex}$ (mk)	$4.03 \pm 0.05$
Control rod worth (mk)	$7.13 \pm 0.02$
Shutdown margin (mk)	$3.09 \pm 0.11$
Effective delayed neutron ( $\beta_{eff}$ )	$0.00084 \pm 0.00005$
Safety reactivity factor (SRF) $\geq 1.5$	$1.77 \pm 0.02$

## RESULTS AND DISCUSSION

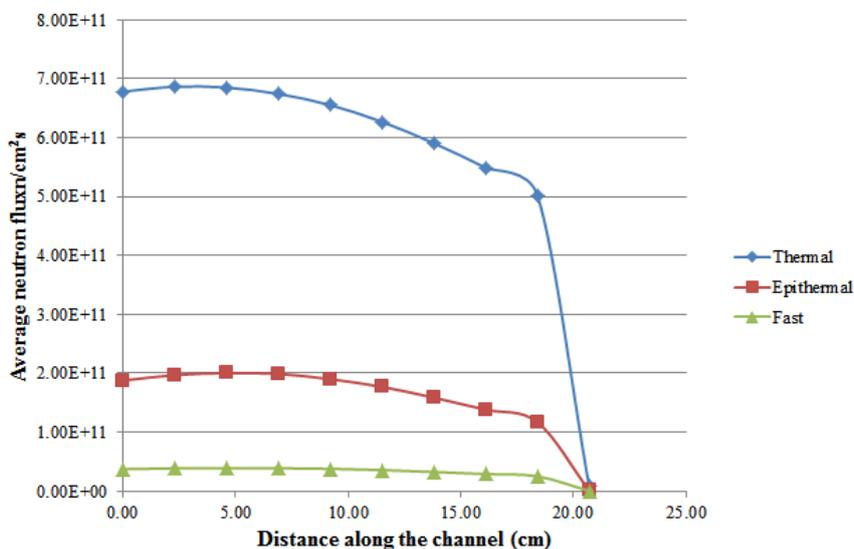
The result obtained for the nuclear criticality, core excess reactivity and the control rod worth of the LEU are presented in **Table 1**. The table shows some eigenvalues with its associated errors for GHARR-1 LEU core when the control rod is fully withdrawn from the core and when it is fully inserted into the core. The worth of the control rod was found to be 7.13 mk which is close to the value reported by other authors. The core excess reactivity was calculated to be 4.03 mk.

The TOTNU NO card was used with the control rod withdrawn to estimate the effective delayed neutron fraction as presented in **Table 2**.

Shutdown margin of 3.09 mk was obtained for the control rod of the LEU core. Again, the average neutron flux distribution in the inner and the outer irradiation sites of GHARR-1 LEU core is similar to those reported by other authors (**Figures 3 and 4**).



**Figure 3.** Average Neutron flux distribution in inner irradiation channels.



**Figure 4.** Average neutron flux distribution in outer irradiation channels.

## **CONCLUSION**

In this work, neutronic calculations have been performed for the Ghana Research Reactor-1 LEU core using MCNP5 simulation code. The results showed that the criticality calculations of the LEU are in good agreement with the HEU.

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