MONTE CARLO NEUTRON TRANSPORT SIMULATION OF THE GHANA RESEARCH REACTOR-1

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Abstract

Stochastic Monte Carlo neutron particle transport methods have been applied to successfully model in 3-D, the HEU-fueled Ghana Research Reactor-1 (GHARR-1), a commercial version of the Miniature Neutron Source Reactor (MNSR) using the MCNP version 4c3 particle transport code. The preliminary multigroup neutronic criticality calculations yielded a k_{eff}*=1.00449 with a corresponding cold clean excess reactivity of 4.47mk (447pcm) compared with experimental values of* k_{eff}*=1.00402 and excess reactivity of 4.00mk (400pcm). The Monte Carlo simulations also show comparable results in the neutron fluxes in the HEU core and some regions of interest. The observed trends in the radial and axial flux distributions in the core, beryllium annular reflector and the water region in the top shim reflector tray were reproduced, indicating consistency of the results, accuracy of the model, precision of the MCNP transport code and the comparability of the Monte Carlo simulations. The results further illustrate the close agreement between stochastic transport theory and the experimental measurements conducted during off-site zero power cold tests.*

1. Introduction

The Ghana Research Reacdtor-1 (GHARR-1) is a commercial version of the Miniature Neutron Source Reactor (MNSR) and belongs to the class of pool-in-tank-type reactors [1-2]. It is under-moderated with an H/U atom ratio of 197. Thermal power is rated at 30kW with a corresponding peak thermal neutron flux is 1.0×10^{12} n/cm^2.s. For fresh core, its cold clean excess reactivity is about 4mk. Cooling is achieved by natural convection using light water. Presently, the GHARR-1 core consists of a fuel assembly HEU (U-Al alloyed) fuel elements arranged in ten concentric rings about a central control rod guide tube which houses the reactor's only control rod. The control rod's reactivity worth is about -7mk, providing a core shutdown margin of -3mk of reactivity. The small core has a low critical mass. However, its relatively large negative temperature coefficient of reactivity is capable of boosting its inherent safety properties [3]. The small size of the core facilitates neutron leakage and escape in both axial and radial directions. To minimize such loses and thereby conserve neutron economy, the core is heavily reflected respectively on the side and underneath the fuel cage by a thick annulus and slab of beryllium alloy material. Adding regulated shims of beryllium to the top tray can compensate loss of reactivity due to axial neutron leakage.

A schematic drawing of the cross sectional view through the reactor is shown in Fig.1.

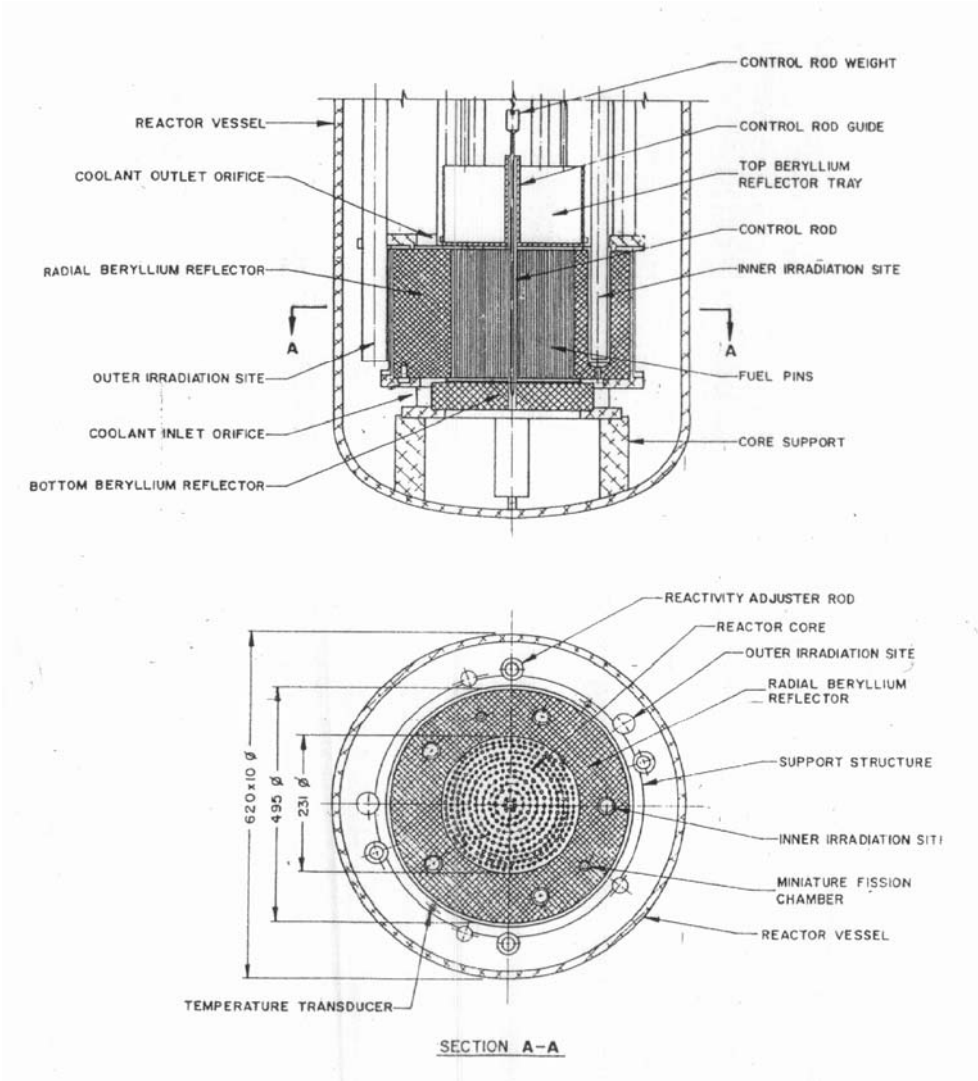

Fig.1: Cross section through the GHARR-1 reactor

The HEU-fueled GHARR-1 reactor has a theoretically designed core lifetime of ten years if it is operated at its maximum flux for 2.5 hours day, five days a week. At this rate, the HEU fuel with a burnup of 1% [4] would have been fairly depleted resulting in a fall of core excess reactivity to limits below 2.8mk When all the regulated beryllium shims have been added and core excess reactivity can no longer be sustained, the fuel would have been depleted, and the core would have reached its designed lifetime. Refueling can be done with either LEU or HEU fuels, a decision which is determined by several factors including reactor physics.

For this reason, core conversion studies on the GHARR-1 facility have been initiated. Previously, deterministic methods have been used to perform neutronic

core calculations and analysis on GHARR-1. However, due to its small core, complicated geometry and other associated structures, it has become increasingly necessary to employ more versatile methods such as Monte Carlo transport methods to accurately model the reactor in three-dimensions (3-D), simulate particle transport behavior and estimate reactor physics design and safety parameters. The Monte Carlo modeling and simulation will be applied to the core conversion program of the initiated on the GHARR-1 facility.

In this paper, the results of the Monte Carlo model, neutron transport and simulation of the GHARR-1 reactor using the MCNP4C transport code are presented.

2.0. 3-D Monte Carlo Model and Method of analysis

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 behavior of different particles. Additionally, MCNP has the ability to treat complex nuclear interaction processes [5].

2.1. Physical Model

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 modeling [6]. 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 modeled 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 mid-point for the Monte Carlo model.

2.2. Nuclear Model: fission sources, tallies and nuclear data

In the MCNP model of GHARR-1, all the 344 HEU fuel elements of cylindrical geometry were modeled as fission sources. The gram mass loading or weight content of U-235 per fuel element ($g_{U\text{-}235}$) for MNSR reactors can be calculated from the expression [1-2]

$$g_{U-235} = \rho_f V(1-e\%)Xf_m \qquad (1)$$

where V is the volume of the active zone of the fuel element of porosity $e\%$, X is the total mass fraction of uranium in the fuel and ρ_f is the fuel density in g/cm^3. The quantity f_m is defined as

$$f_m = \frac{m_{U-235}\varepsilon}{m_{U-235}\varepsilon + m_{U-238}(1-\varepsilon)} \qquad (2)$$

where m_{U-235} *and* m_{U-238} are the atomic masses of U-235 and U-238 isotopes respectively and ε is the U-235 enrichment. Substituting Eq. (2) into Eq. (1), the gram loading of U-235 becomes

$$g_{U-235} = \rho_f V(1-e\%)X\left(\frac{m_{U-235}\varepsilon}{m_{U-235}\varepsilon + m_{U-238}(1-\varepsilon)}\right) \qquad (3)$$

from which the density of the fuel can be calculated as

$$\rho_f = \frac{g_{U-235}}{V(1-e\%)X\left(\frac{m_{U-235}\varepsilon}{m_{U-235}\varepsilon + m_{U-238}(1-\varepsilon)}\right)} \qquad (4)$$

Cell tallies for neutron flux and fission energy deposition were adopted in the GHARR-1 Monte Carlo model in accordance with the MCNP input data preparation, providing for the simulation of neutron fluxes (radial and axial) distributions and other physics design and safety parameters. The fission sources were also modeled to simulate the fission energy deposition in the reactor core. A 16-subdivision structure of the Hansen-Roach continuous neutron energy group was used in the Monte Carlo model. Nuclear data for fissile and non-fissile isotopes associated with materials (fuel, fuel clad, coolant, moderator, control rod and clad, reflectors, structural components) of the physical model was chosen from ENDF/B-VI nuclear data libraries. For cadmium, cross section data from ENDF/B-V data libraries was used. The special $S(\alpha\beta)$ scattering feature was applied in the nuclear model to treat thermal scattering in beryllium and hydrogen in light water for the reflector material and water regions respectively of the GHARR-1 Monte Carlo model.

2.3. Method of Analysis

The 3-D GHARR-1 Monte Carlo model was simulated to estimate some reactor physics parameters such as nuclear criticality and core reactivities, neutron flux distribution in some selected locations of the reactor. In particular, neutron transport simulations were done for a clean fresh core (zero burnup). Criticality calculations were performed by invoking the MCNP KCODE criticality source card to determine k_{eff} and corresponding core excess reactivities using all fuel elements as fission source points. In this analysis, 3000 neutron particle source histories per cycle were made for 2000 cycles. The MCNP criticality calculations were normalized to the steady-state power level of 30kW. Using the *FM* card, a value of

2.434 was computed for the average number of neutrons per fission which is used in computing the scaling factors or source strength by which all the tallies are scaled.

The GHARR-1 Monte Carlo model was further simulated for partial and total withdrawal (or inserted) of the control rod to different positions to determine control rod worth and shutdown margins. The 3-D model also provided for the simulation of the ten homogenized fuel lattice zones in order to calculate the radial and axial thermal neutron fluxes as well as the fission energies averaged over each lattice zone. Other regions such as the reflector materials and irradiation n channels were also simulated for radial and axial neutron flux distributions. The results are presented and discussed in the next section.

3.0. Results and discussion

The 3-D Monte Carlo MCNP plot of the GHARR-1 core configuration is shown in Fig. 2. The vertical cross section of the reactor with the control totally in the core is illustrated in Fig.3, showing a vertical slant tube installed for out-of-vessel irradiation and other reactor physics experiments.

Fig. 2: MCNP plot of GHARR-1 core configuration

Compared with the equivalent drawings shown in Fig.1, the good agreement establishes the fact that the model is good and that Monte Carlo techniques can be effectively deployed to model geometrically complicated systems

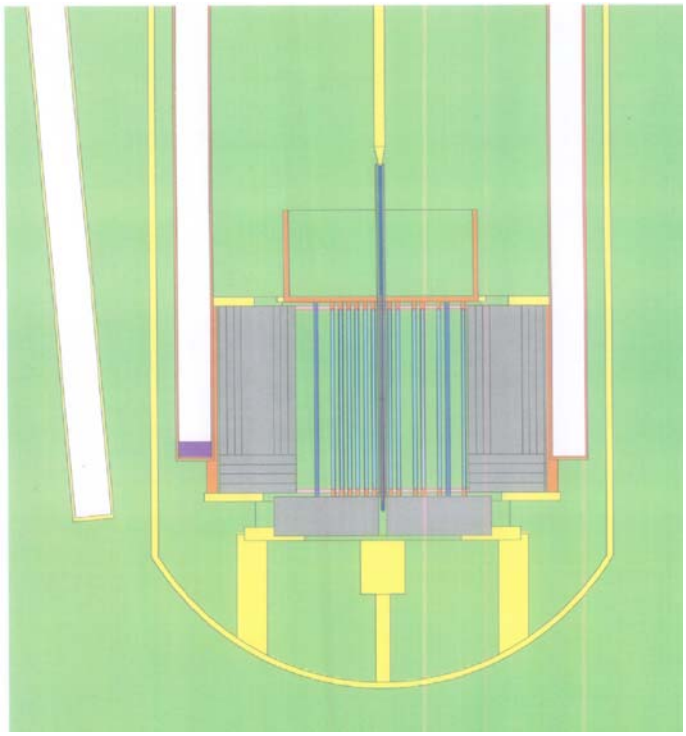

Fig.3: MCNP plot of GHARR-1 vertical cross section

The results of the MCNP Monte Carlo criticality calculations for k_{eff}, excess core reactivity, control rod worth and shutdown margins as compared with experimental measurements obtained during zero power criticality and cold test are shown in Table 1.

Table 1: Comparison of experimental and Monte Carlo (MCNP4C) criticality results

		Experimental	**MCNP4C**
Criticality (k_{eff})		1.00402	1.00449
Reactivity	Core excess in cold, clean state	4.00	4.47
	Control rod worth	-7.00	-7.70
	Shutdown margin	-3.00	-3.27

From the table, Monte Carlo MCNP calculations for the k_{eff} and cold clean cores excess reactivity differ from the experimental results within 0.047% and 11.75% respectively. Similarly, the differences in the control worth and shutdown margin

are 10% and 9% respectively. The differences in values between the experimental and MCNP4C calculations are however small. The relatively higher values obtained for the MCNP4C Monte Carlo simulations are due to the fact that the masses, dimensions and compositions of some material and structural components were approximated at the time of the modeling and simulation. Generally, as since factors such as size (mass, dimensions), type and material composition (material balance), geometry, etc, influence nuclear criticality, the observed differences between the experimental and MCNP4C measurements are anticipated. It is expected that correct masses and material balance would show more agreeable results.

An experimental plot of the radial distribution of the thermal neutron flux intensity is shown in Fig. 4.

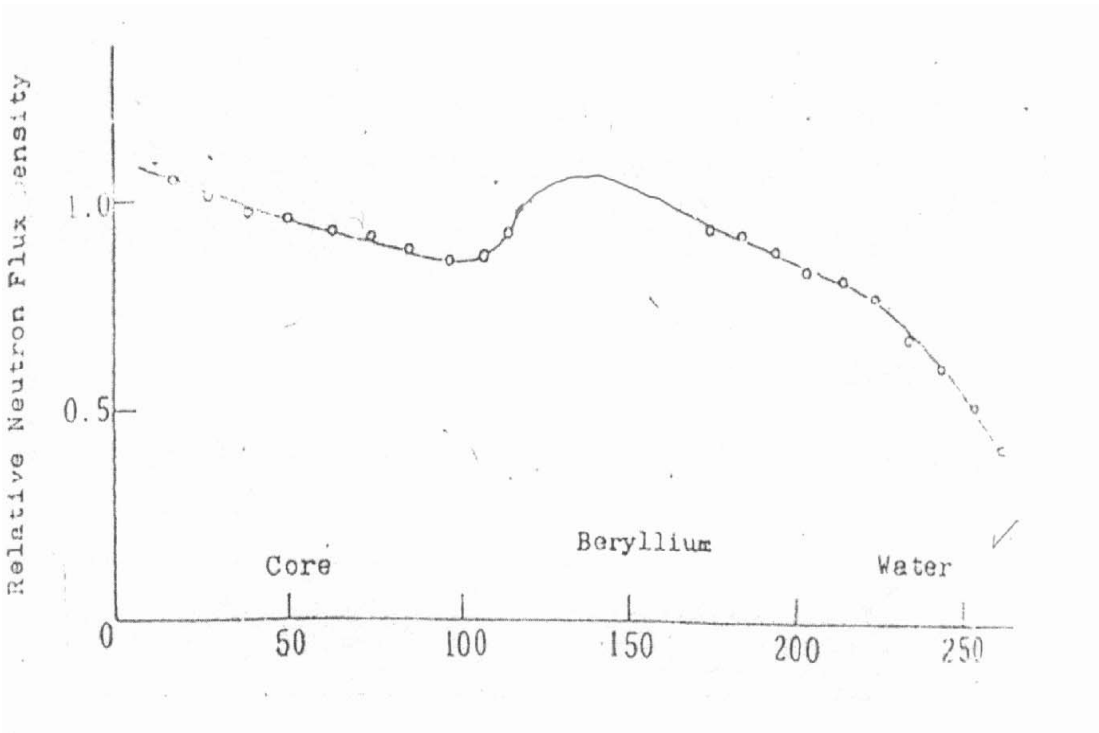

Fig. 4: Radial flux intensity distributions (experimental)

Comparatively, the MCNP4C results for the radial flux distribution in the core, annular beryllium reflector and outer irradiation channel is shown in Fig. 5. As expected, the neutron flux decays as the core is traversed outwards but peaks up in reflector zones such as the beryllium material where (n,2n) reactions occur. Comparatively, the MCNP4C results agree very well with experimental results measured during the GHARR-1 zero power criticality test as both show the same trend in flux distribution.

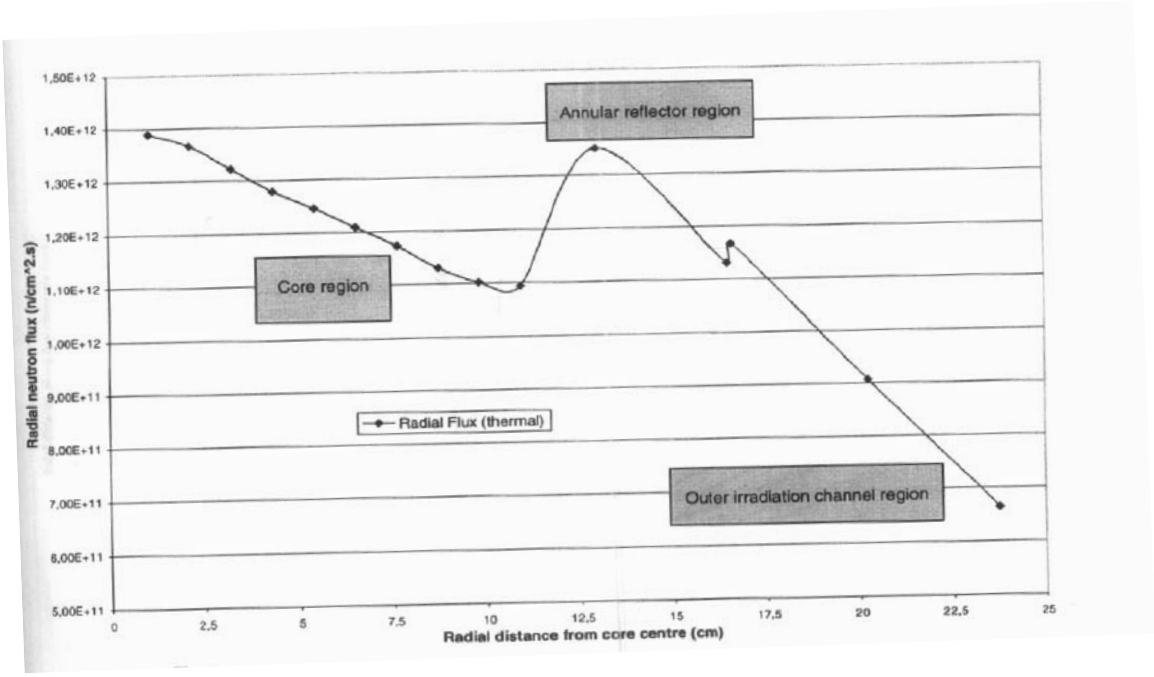

Fig. 5: Radial flux distribution in the core and some other regions (MCNP),

The axial neutron flux distributions along the fuel rod channels for *all the 10 fuel lattice zones* in the GHARR-1 HEU core obtained via the MCNP4C Monte Carlo simulation are shown in Fig.6. The corresponding experimental plot for the axial neutron flux intensity measured at *three* locations *between fuel lattices* is depicted in Fig.7.

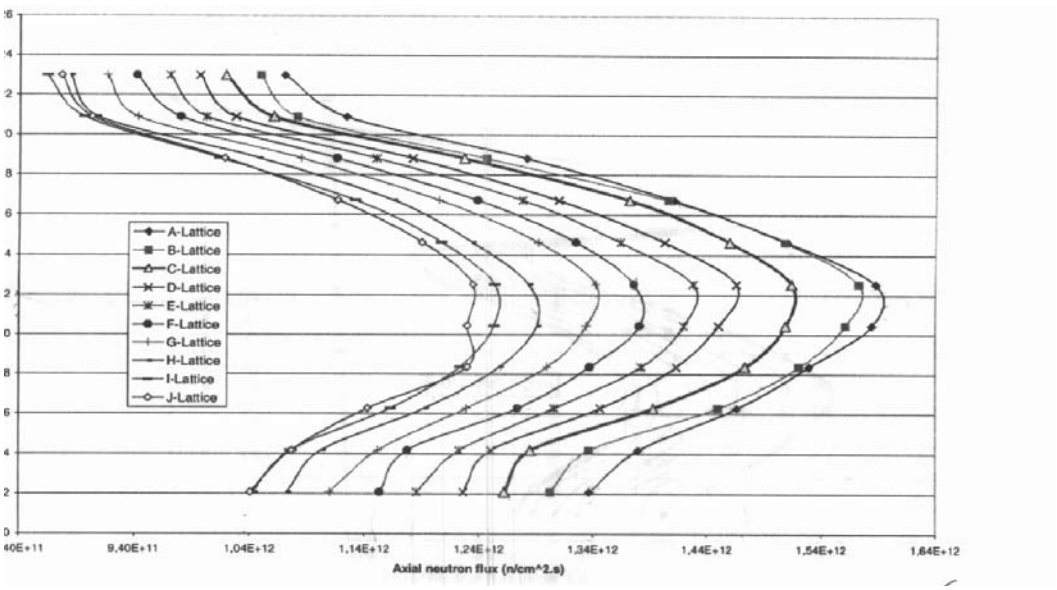

Fig. 6: Axial flux distribution along fuel rod channels in GHARR-1 core (MCNP)

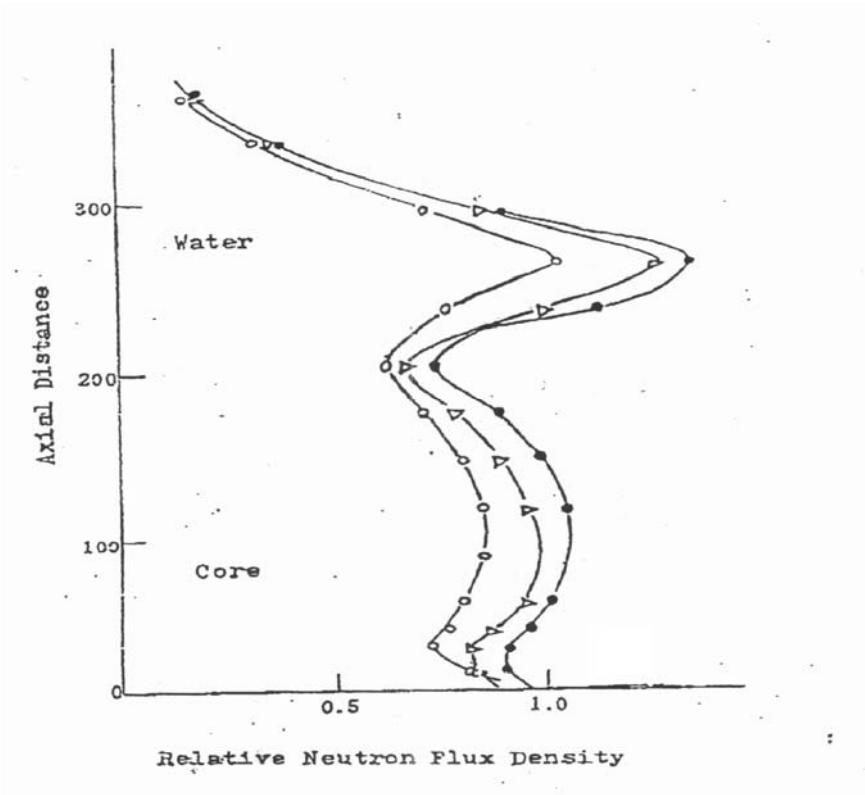

Fig. 7: Relative axial flux density (experimental)

Both approaches (Monte Carlo and experimental) show the same observed trend with a conspicuous peaking at the geometrical centre and a fall off as the fuel channel rod is traversed away from the centre. The close agreement between both approaches in the trends is further evidence of a good model and proves the suitability Monte Carlo stochastic methods to modeling reactor systems.

The Monte Carlo MCNP4C model of the GHARR-1 reactor also provided for the simulation of the fission energy deposited on each fuel element cell and also per homogenized fuel lattice zone. This was done to establish the reactor power distribution across the GHARR-1 HEU core that would be used as a design reference for the GHARR-1 HEU-LEU core conversion program. The MCNP4C results of the fission energy deposited in each fuel lattice zone is illustrated in Fig. 8. The observed decreasing trend is conclusive that there are more nuclear fissions in fuel elements nearer the core centre. The relatively higher peaks obtained for the tenth lattice compared with ninth could be explained by the contribution of radial neutron reflection provided by the annular beryllium reflector where the neutron fluxes are relatively higher (Figs.4 & 5).

It can be inferred from the results of the Monte Carlo simulated fission energies and peaking factors that lattice zones closer to the core centre would have relatively

higher fuel burnups. However, this can be confirmed when a detailed fuel depletion and burnup credit analysis is performed.

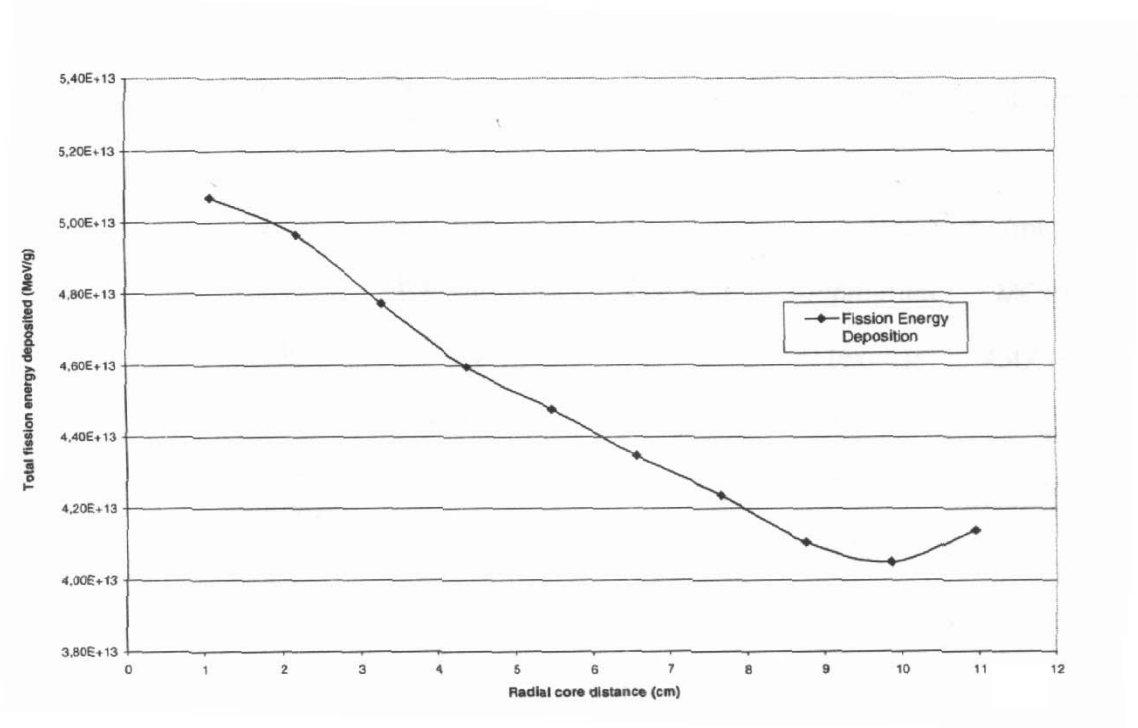

Fig.8: Fission energy deposition in fuel lattice zones of GHARR-1 core

Conclusion

Stochastic Monte Carlo methods have been applied to accurately develop a 3-D model of the GHARR-1 reactor using the versatile MCNP4C particle transport code.

Within limits of errors caused by approximations in material balance and dimensions of some components, the Monte Carlo model has successfully simulated some important reactor physics and criticality safety design parameters. The results are in very good agreement with experimental measurements obtained during the facility's zero power criticality and on-site cold tests. This work thus provides further evidence that Monte Carlo methods are accurate in modeling complicated geometries and also capable of simulating physics and engineering design parameters of nuclear reactors such as the GHARR-1 reactor.

The Monte Carlo model and the simulated results for the present GHARR-1 HEU fueled reactor would be used as design references for its HEU-LEU core conversion program.

References

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