PROGRESS IN THE NEUTRONIC CORE CONVERSION (HEU-LEU) ANALYSIS OF GHANA RESEARCH REACTOR-1

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Abstract

As required for all nuclear reactors, neutronic and thermal hydraulic analysis are being performed for the HEU-LEU core conversion studies of the Ghana Research Reactor-1 (GHARR-1) facility, which is a commercial version of the Miniature Neutron Source Reactor (MNSR). Stochastic Monte Carlo particle transport methods and tools (MCNP4c/MCNP5) were used to fine–tune a previously developed 3-D MCNP model of the GHARR-1 facility and perform neutronic analysis of the 90.2% HEU reference and candidate LEU (UO_2, U_3Si_2, U-9Mo) fresh cores with varying enrichments from 12.6%-19.75%. In this paper, the results of the progress made in the Monte Carlo neutronic analysis of the HEU reference and candidate LEU fuels are presented. In particular, a comparative performance assessment of the LEU with respect to neutron flux variations in the fission chamber and experimental irradiation channels are highlighted

1. Introduction

The Ghana Research Reactor-1 (GHARR-1) is a commercial version of the Miniature Neutron Source Reactor (MNSR) and has operated at different power levels since its commissioning in March 1995 [1]. The GHARR-1 facility is initially designed as a low power, pool-in-tank-type research reactor with a compact core consisting of a single fuel assembly comprising 344 U-Al (admixed in aluminum matrix) pins enriched to 90.2% [2-5]. The fuel assembly consists of ten concentric zones or rings of 354 fuel and structural lattices distributed about a central control rod guide tube. The core is under-moderated with an H/U atom ratio of 197. Thermal power is rated at 30kW with a corresponding peak thermal neutron flux of 1.0E+12n/cm^2.s. Cold clean excess reactivity for fresh core is limited to about 4mk (1/2 β_{eff}). Cooling is achieved by natural convection using light water. The integral reactivity worth of the control rod is about -7mk, providing a core shutdown margin

of -3mk of reactivity. The small HEU core has a low critical mass (<1kg). However, it has a relatively large negative temperature coefficient of reactivity to boost 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.

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

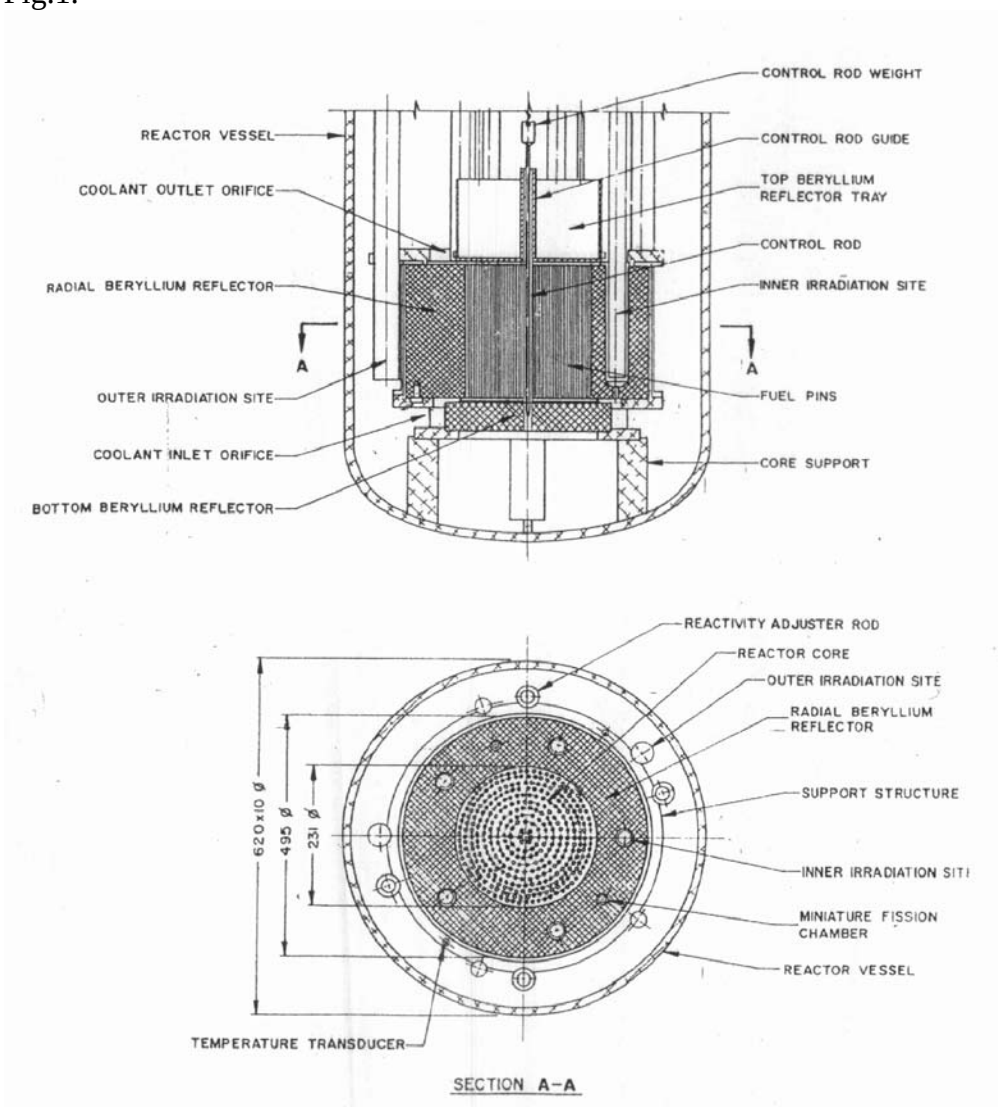


Fig.1: Cross section through the GHARR-1 reactor

Theoretically, the HEU-fueled GHARR-1 reactor has a designed core lifetime of ten years if it is operated at full power or its maximum flux for 2.5 hours day, five days a week. Under this condition, the HEU fuel, with a burnup of 1% [2-4], would depleted resulting in a drop in core excess reactivity to limits in the range of 2.5mk-

2.8mk. Loss of core excess reactivity due to axial neutron leakage is compensated for by the addition of regulated beryllium shims to the top tray. In particular, when the total reactivity worth contributed by the top shims (10.95cm total thickness) is consumed, and the excess reactivity can no longer be sustained, the core would have reached its end of life (EOL). Future cycle operations will require a replacement of the spent or depleted fuel with a fresh fuel assembly. Refueling can be done with either LEU or HEU fuels.

There are external indicators which influence considerations for use of LEU fuels as likely candidate replacement core for the GHARR-1 facility and other MNSR facilities. It has been observed in recent times, an international trend in replacing HEU-fueled nuclear reactor cores and other associated facilities with LEU assemblies. International politics, nuclear safeguard protocols and non-proliferation treaties are other important issues which influence LEU considerations in nuclear refueling. The absence of fuel fabrication facilities in some countries also heavily impacts on the choice or consideration of LEU fuel as candidates for fuel replacements.

For these and other important reasons including future cycle activities, it became necessary to establish core conversion studies on the GHARR-1 facility in 1996. The HEU-LEU core conversion analysis is presently been pursued jointly by the Department of Nuclear Engineering and Materials Science in the National Nuclear Research Institute of Ghana Atomic Energy Commission, and the RERTR Program of the Division of Nuclear Engineering, Argonne National Laboratory. 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 MCNP4c and MCNP5 transport codes were used to improve on the previously developed model [1] and simulate criticality, kinetic and other reactor physics design parameters of the HEU reference core as well as for UO_2, U_3Si_2 and U-9Mo candidate cores. The codes were also used to neutronically evaluate possible core configurations, arrangements and enrichments of the LEU fuels in order to produce results as close to the reference HEU as possible.

In this paper, the brief description of the GHARR-1 Monte Carlo model and the results of the simulation and neutronic transport analysis for the HEU-LEU conversion studies using the MCNP transport codes are presented.

2.0. 3-D Monte Carlo Model

Generally, multipurpose Monte Carlo particle transport codes have the capability to model and treat different complicated geometries in 3-D and also simulate the transport behavior of different particles and nuclear interaction processes. For these

reasons, the versatile and widely utilized MCNP code particle transport code was employed for the modeling simulation and neutronic analysis.

2.1. Physical Model

The physical Monte Carlo model of the GHARR-1 facility (Ghana MNSR) was developed following the approaches used in the 3-D combinatorial, generalized geometry and continuous energy methods applied in MCNP [1]. 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, experimental channels, shim tray, reactor vessel, reactor pool and other structural components were modeled accordingly using available design data. 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. Typically, MNSR reactors are designed to have different core loadings. This means that the fuel loading of a typical MNSR facility could vary or differ from other operating MNSR facilities. Taking cognizance of this variation, the GHARR-1 Monte Carlo model provided for the modeling and simulation of MNSR reactors with varying core configurations ranging from 342 – 350 fuel lattices. The model has been adapted to the Nigerian MNSR as well for the same activity.

The preliminary MNSR MCNP model, developed in 2001 [1], required a re-examination of some dimensional and material composition data. In this present work, the model has been “fine-tuned” to account for geometrical and material compositions of some structural components including the honey-combed top grid plate, insertion of reactivity regulators, and update of fuel element and beryllium material compositions.

Apart from the HEU reference core consisting of 344 U-Al alloy fuel elements enriched to 90.2%, five other LEU “candidate” fuels were considered in the Monte Carlo neutronic analysis for the GHARR-1 HEU-LEU conversion feasibility studies. The LEU fuels included pelletized UO_2, as well as U-9Mo and U_3Si_2 dispersed in aluminium matrix. Different LEU core configurations and enrichments were modeled and simulated for the conversion analysis. In particular, three UO_2 configurations consisting of 344 fuel elements (12.6% enriched), 201 fuel elements (19.75% enriched, 149 water dummies), and 238 fuel elements (19.75% enriched, 112 Al dummies) were analyzed. For the U-9Mo, U_3Si_2 dispersant fuels, a core configuration of 344 fuel elements as the reference HEU was adopted. Fissile material enrichment was however reduced to a theoretical value of 19.75%.

2.2. Neutronics Model of the GHARR-1

The MCNP4c and MCNP5 [6] transport codes were used to perform the Monte Carlo calculations. 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. Some nuclear data from LLNL, LANL were utilized. 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. A 7-subdivision structure of the Hansen-Roach continuous neutron energy group was used in the Monte Carlo model. Neutronic analyses were performed using a condensed 3-group neutron energy structure: up 0.625eV for thermal neutrons, <8.21 eV for epithermal neutrons and up to 20 MeV for fast neutrons.

3. Method of Analysis

The 3-D GHARR-1 Monte Carlo model was used to simulate 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 clean fresh cores (zero burnup). Criticality calculations were performed to determine k_{eff} and corresponding core excess reactivities using all fuel elements as fission source points. For better statistics in this present analysis, the criticality specifications in the MCNP model provided for 400 kcode cycles and 500,000 source particles giving a total of 200 million neutron particle source histories. An initial criticality guess of k_{eff} =1.004 was used. The MCNP criticality calculations were normalized to the steady-state power level of 30kW.

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 and calibrate rod worths 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 or power averaged over each lattice zone. Other regions such as the reflector materials and irradiation channels were also simulated for radial and axial neutron flux distributions.

The MCNP plots of the present GHARR-1 Monte Carlo model showing the core configuration (344 fuel elements), and vertical cross sectional views of the reactor in both operational (control rod withdrawn) are presented in Figs. 2-3 respectively.

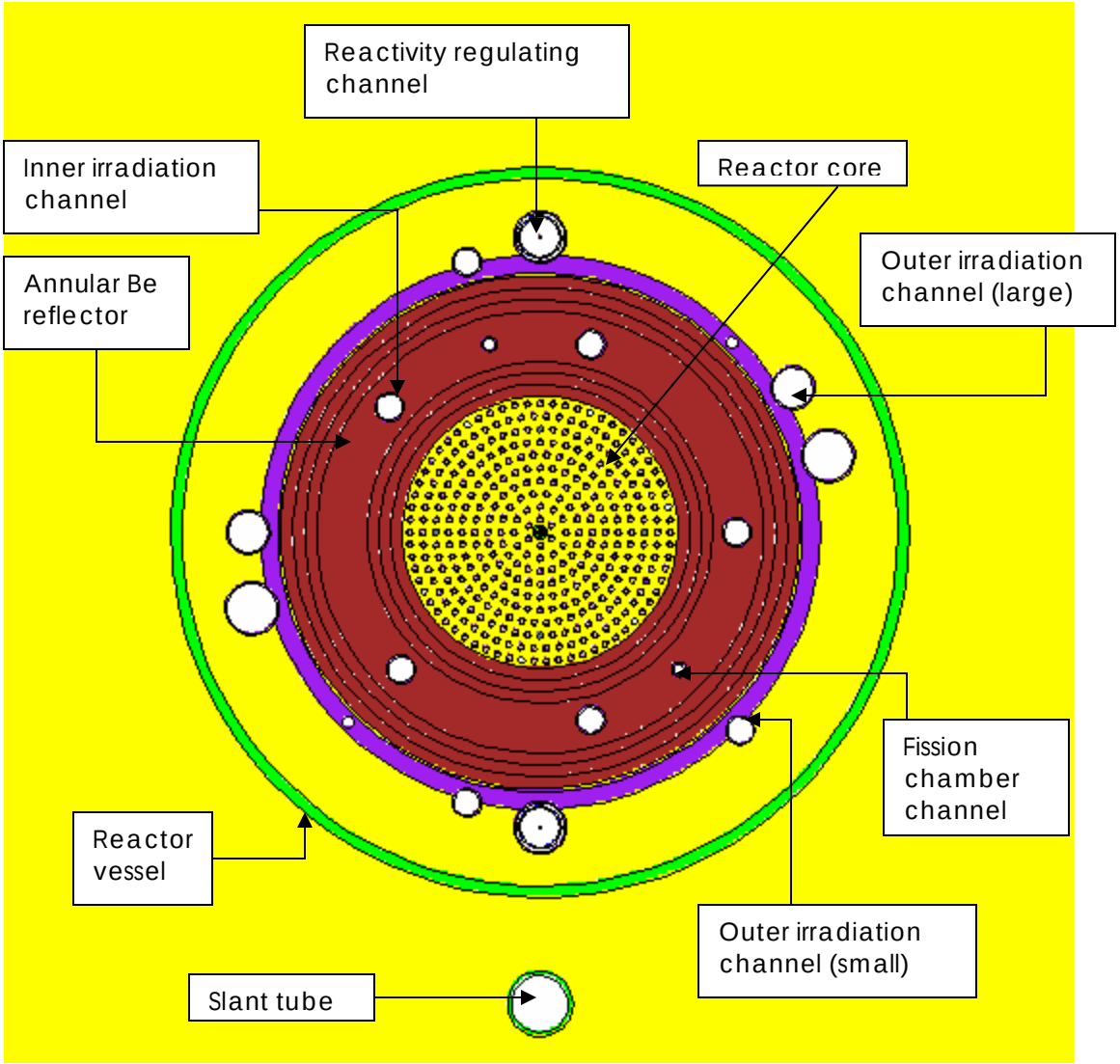


Fig. 2: MCNP5 plot of GHARR-1 core configuration showing fuel region (reactor core), channels for irradiation, fission chamber, regulating, slant and annular beryllium reflector

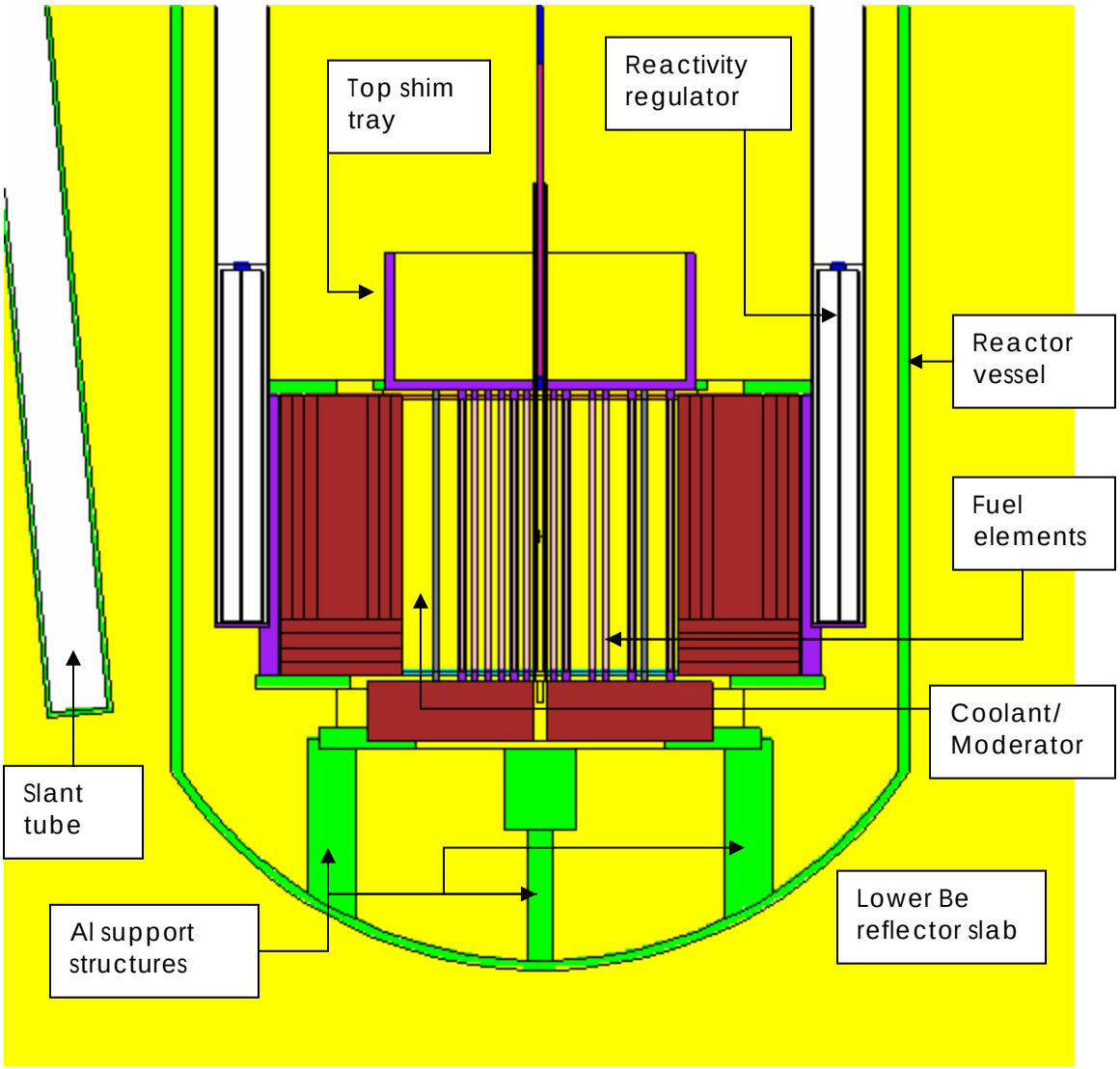


Fig.3: MCNP plot of vertical cross section of GHARR-1 reactor (control rod in full withdrawn position) showing structural supports, reactor vessel, etc.

4.0. Results and discussion

The results of the Monte Carlo criticality (***k_{eff}***) calculations and simulation runs for some reactor physics design parameters and core excess reactivity calculations for the reference HEU and candidate LEU cores are indicated in Table 1.

From this table, it is observed that the Monte Carlo calculated k_{eff} values for the dispersant U-9Mo and U_3Si_2 LEU fuels are 0.023% lower with respect to the reference HEU core. Higher k_{eff} values were however simulated for the zirc-4 clad UO_2 LEU cores. Using the same core configuration as the reference HEU, the

12.6% enriched UO_2 core yielded the same k_{eff} results as for the reference 90.2% HEU core.

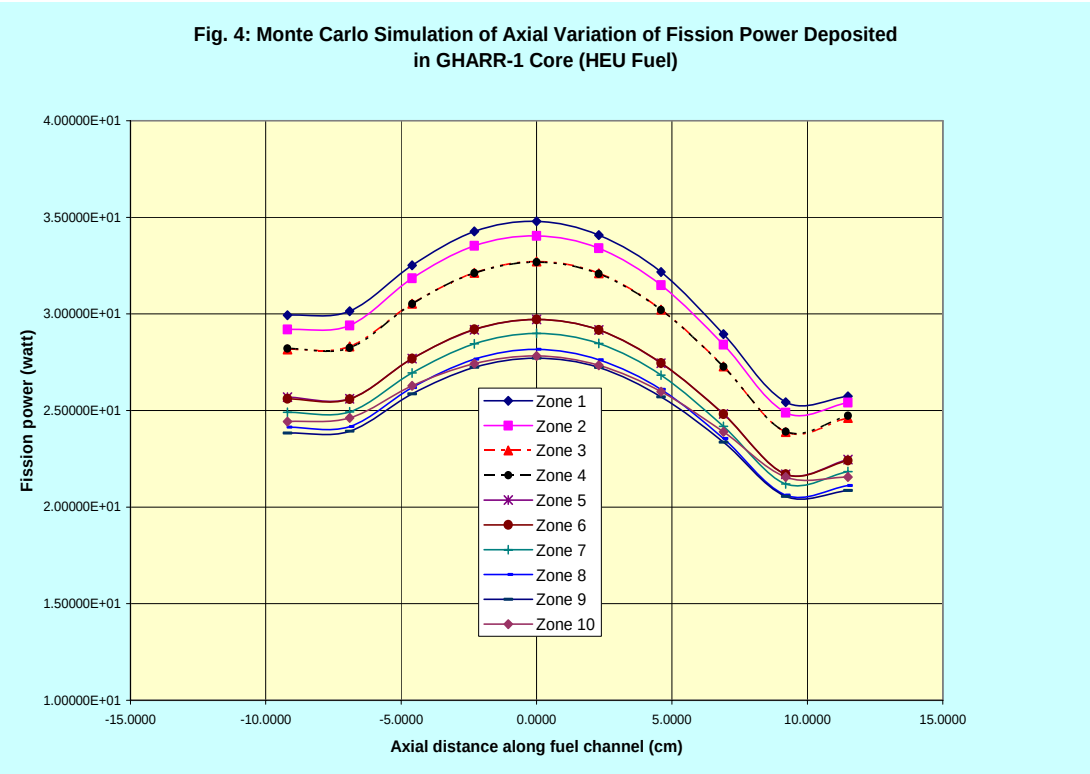
Table 1: Monte Carlo (MCNP4c) simulated criticality parameters for GHARR-1 Facility: HEU vrs. LEU Fuel

	Criticality parameter				
	Reactor status				
Fuel type	Control rod withdrawn		Control rod inserted		
	k_{eff}	**Excess reactivity (mk)**	**k_{eff}**	**Rod worth (mk)**	**Shutdown margin (mk)**
HEU	1.00454 ± 0.00007	4.5195	0.99701	-7.5526	-3.0331
U-9Mo	1.00432 ± 0.00007	4.3014	0.99779	-6.5445	-2.2345
U_3Si_2	1.00431 ± 0.00007	4.2717	0.99767	-6.6555	-2.3655
U_3Si	1.00406 ± 0.00007	4.0436	0.99749	-6.5865	-2.5429
UO_2 (12.6%)	1.00454 ± 0.00007	4.5195	0.99797	-6.5834	-2.0624
UO_2 (19.75%, (Al dummies)	1.00481 ± 0.00006	4.7870	0.99862	-6.1985	-1.4115
UO_2 (19.75%, (H_2O dummies)	1.00434 ± 0.00006	4.3213	0.99785	-6.5040	-2.1827

A reactivity worth of 6.8mk-7.0mk for the centrally located control rod for the Ghana MNSR facility is reported [2, 7]. From Table 1, the simulated Monte Carlo control rod worths for the various HEU and LEU fuels are reported. The lowest rod worth value of -6.2mk corresponding to a maximum deviation of 9% -11% was calculated for the UO_2 core (238 fuel elements with aluminium dummies). According to the specifications required for reactivity control as listed in the Operating Limits and Conditions (OLC) for the Ghana MNSR facility, the minimum and maximum worths of the central control rod shall be 5.5mk and 7mk respectively [8]. Thus, for a core with excess reactivity of 4mk, the corresponding minimum and maximum shutdown margins shall therefore be 1.5mk and 3mk respectively. From Table 1, all the 5 LEU fuels satisfy these conditions and thus

qualify for use as candidate LEU fuels for converting the HEU fuelled GHARR-1 facility and hence any other MNSR core with LEU fuels.

In particular, the Monte Carlo model of the GHARR-1 reactor also provided for the simulation of the fission energy deposited on each fuel element cell and also per 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 MNSR HEU-LEU core conversion studies. A graphical representation of the results of the Monte Carlo simulations of the fission energy deposited in for *all the 10 fuel lattice zones* in the GHARR-1 HEU reference core is illustrated in Fig. 4.

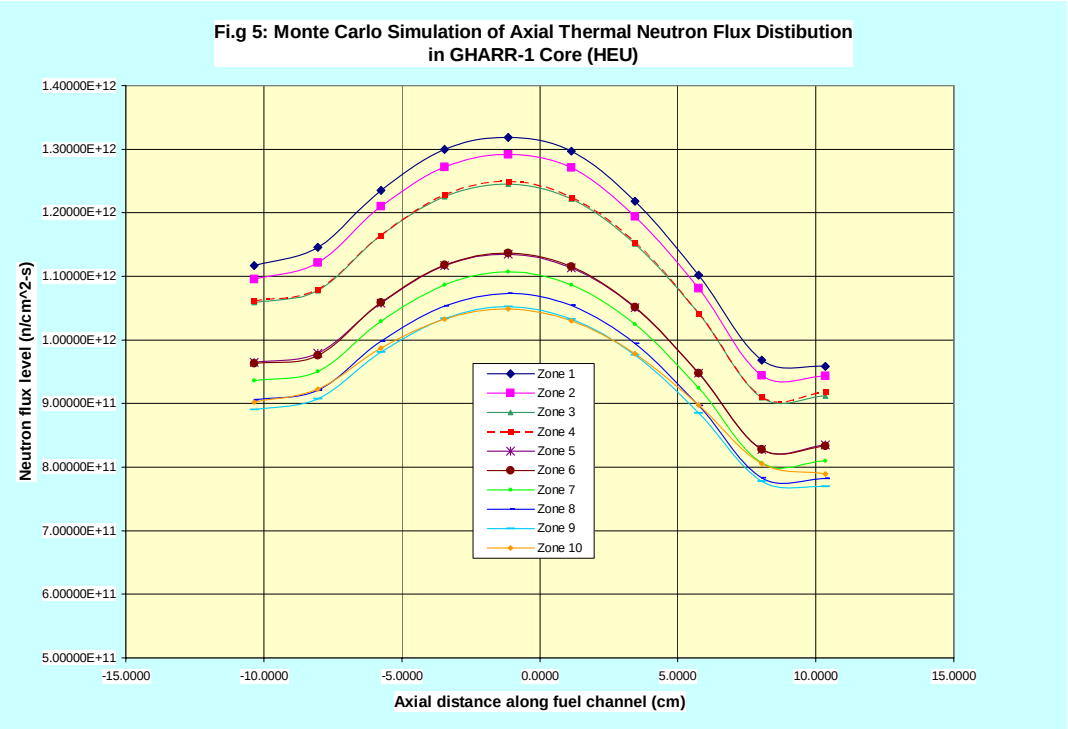


In particular, the fission power and fluxes peak at the centre of the fuel channels which was selected as the geometrical centre of the core in this model. The same trend was observed for all the six fuel considered in this study. In general, for a given axial location, the fission power and fluxes were observed to be higher as the number of fuel elements in the fuel lattice zones increased.

Similarly, the axial neutron flux (thermal) distributions along the fuel rod channels in the core region is shown in Fig. 5. The corresponding experimental plot for the axial neutron flux intensity measured at *three* locations *between fuel lattices* is depicted in Fig.6.

As observed from these plots, the experimental and Monte Carlo simulated results both show the same trend in the parametric (power and flux) distributions in the

core. The observation shows that the GHARR-1 MCNP model is very good and thus establishes the fact the Monte Carlo model can be accurately and successfully utilized as an excellent tool in performing neutronic analysis of the MNSR reactor.



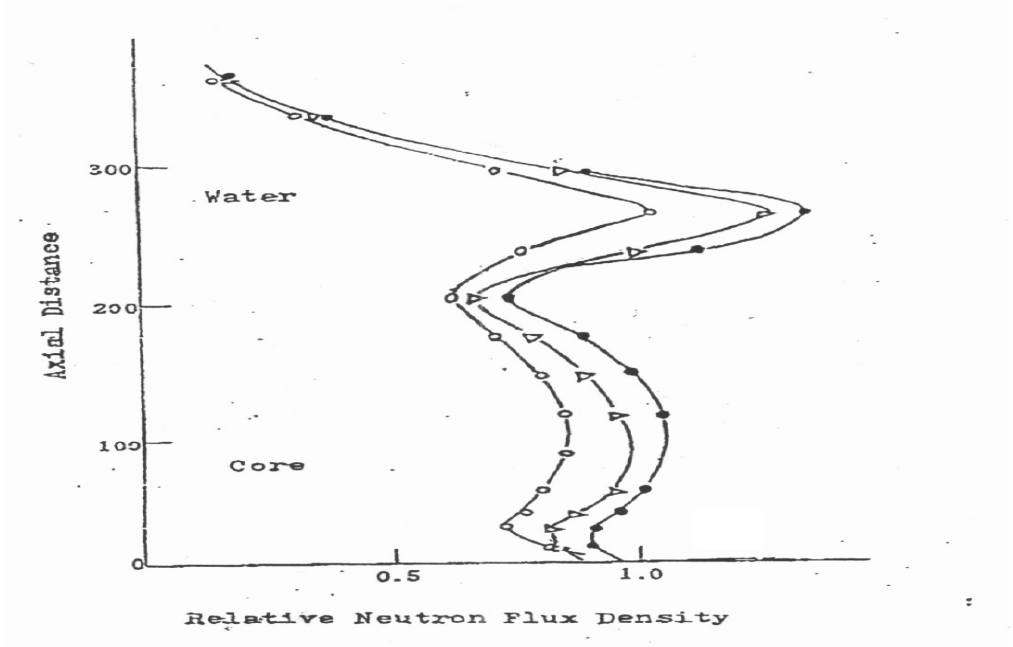
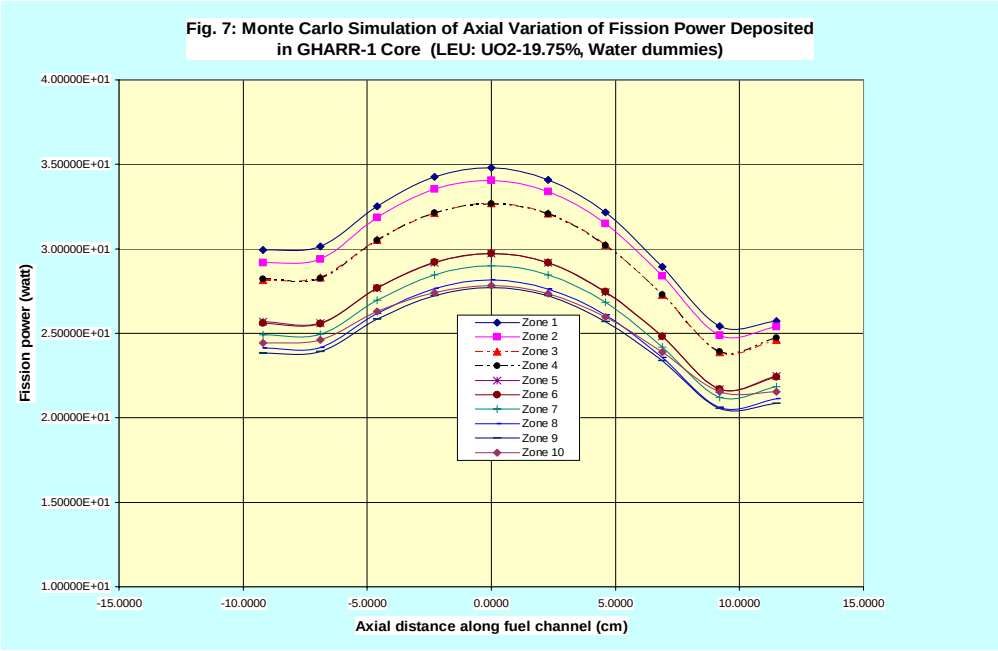
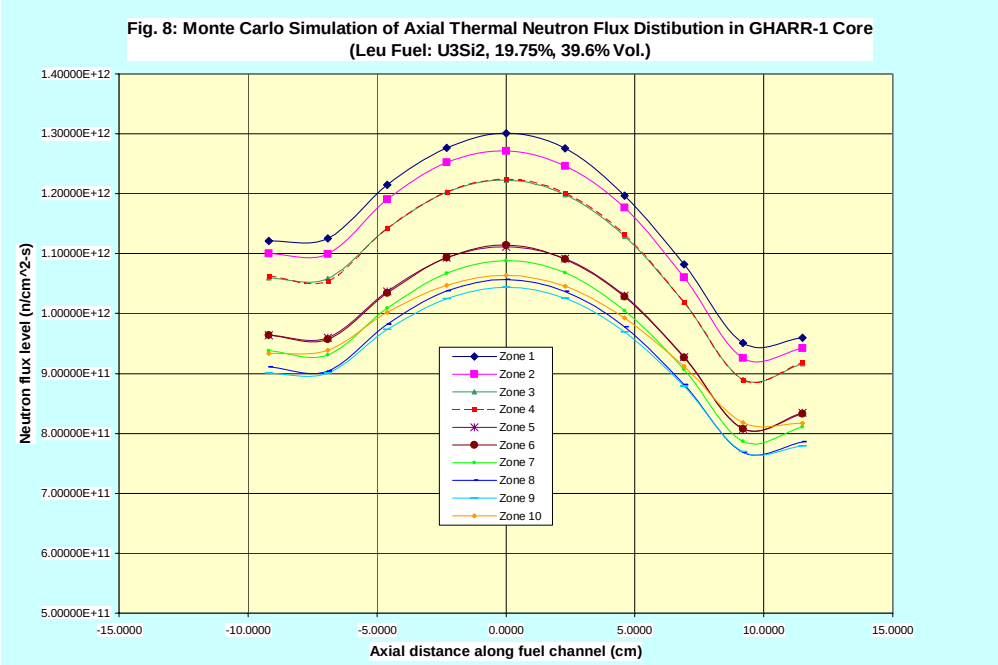


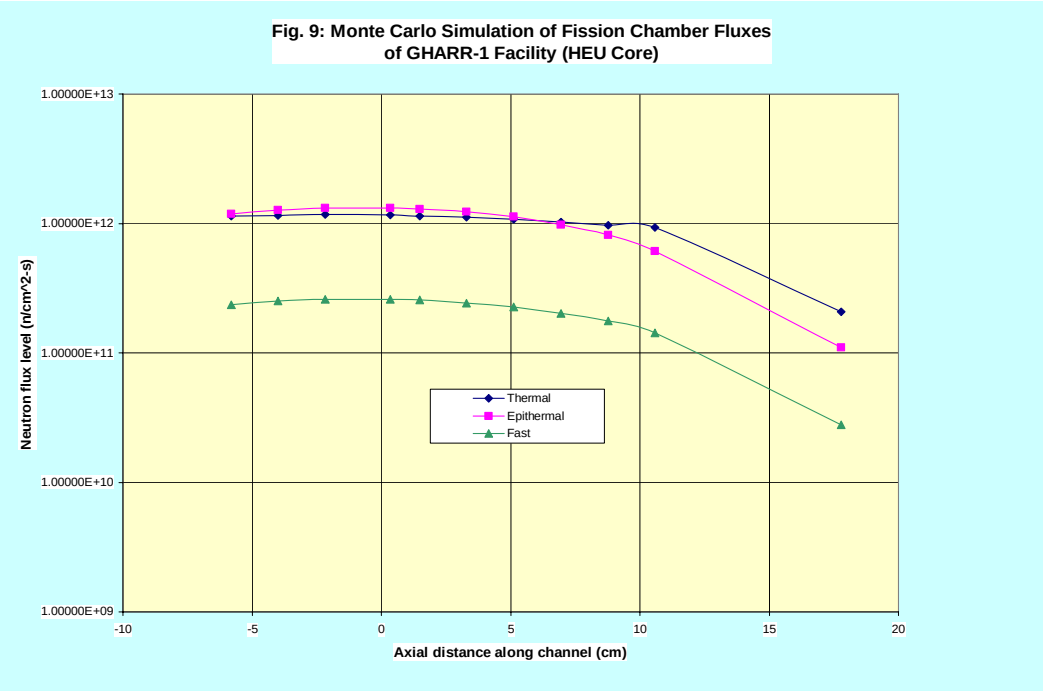
Fig.6: Relative axial flux density (experimental)

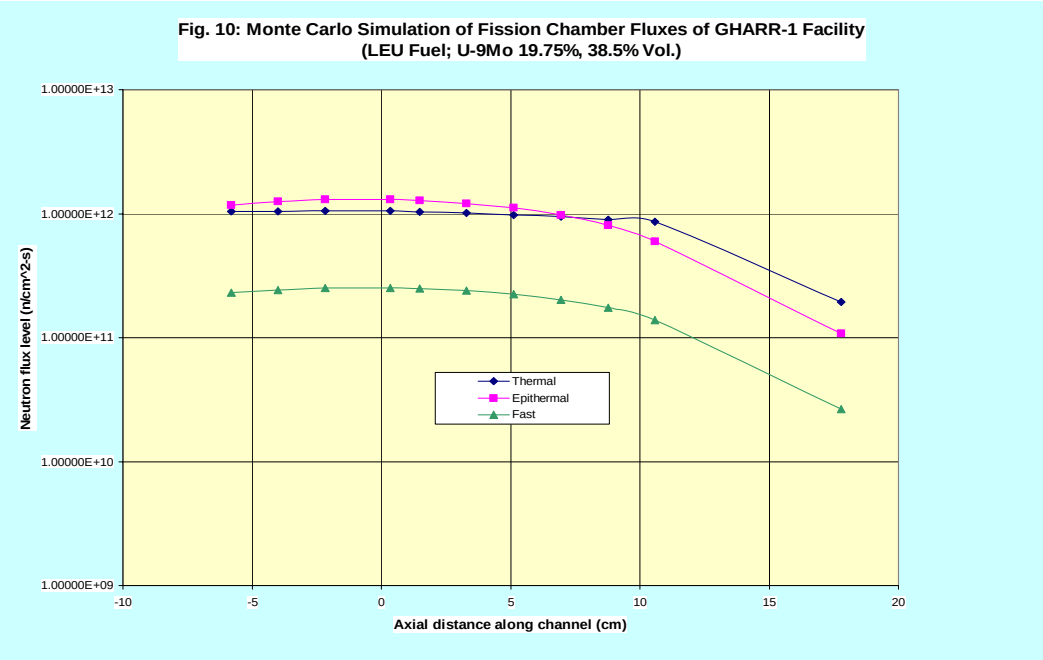
The results obtained for the LEU candidate fuels are similar. In particular, the same trend of the parametric variations is repeated for all the fuel types. Thus for purposes of brevity, only few representations of each fuel type are illustrated. The axial power and flux distributions for the 19.75% enriched UO_2 core (201 fuel elements and water dummies) and U_3Si_2 LEU fuels are shown in Figs. 7 and 8 respectively.



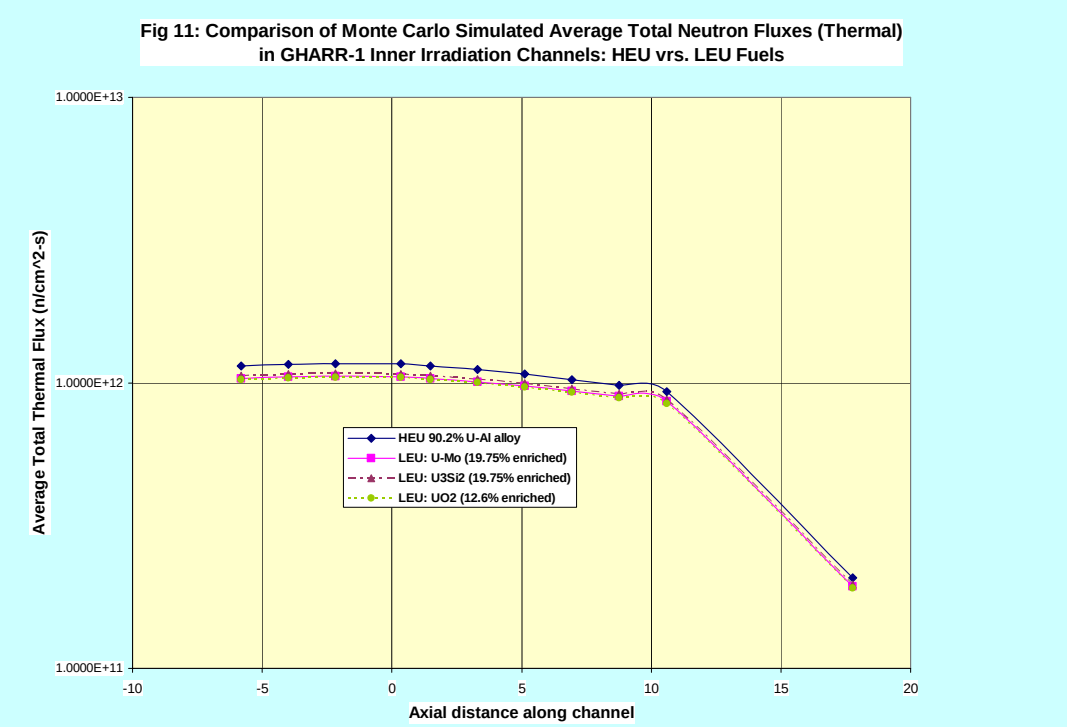


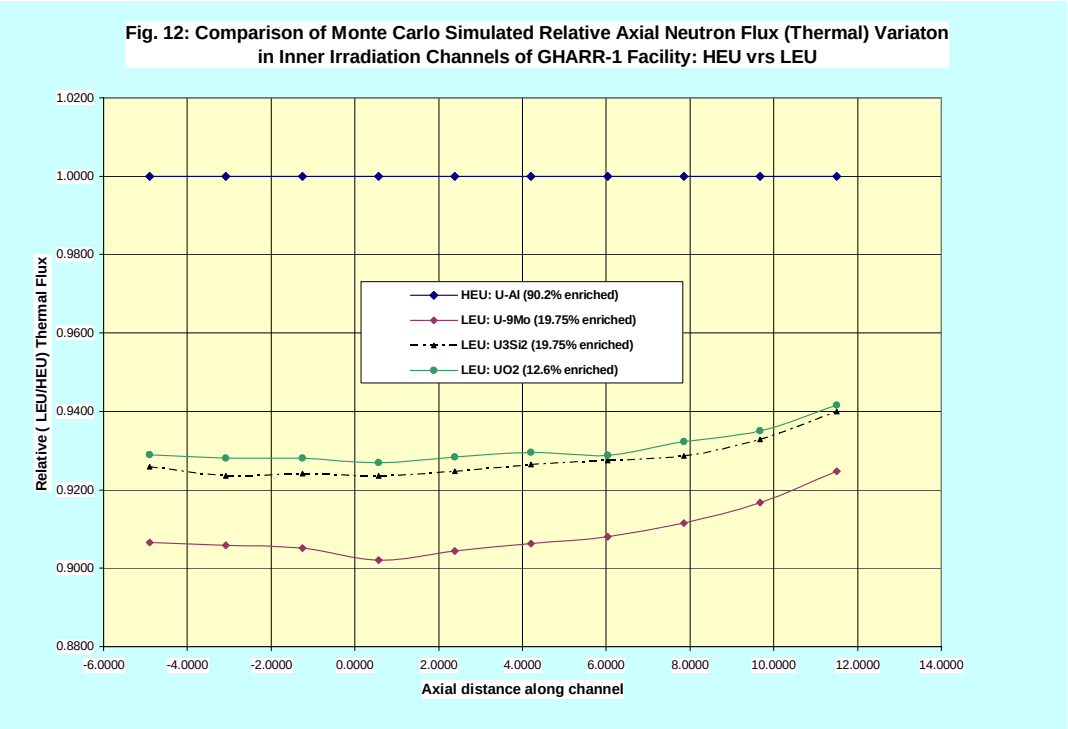
Further progress has been achieved in the development of the GHARR-1 MCNP Monte Carlo model. As performed earlier for the preliminary model [1], the present improved model further provided for neutronic analysis of some other regions of the reactor. The results of the 3-group (thermal epithermal and fast) neutron flux variations in the experimental (fission chamber, inner and outer irradiation) channels for the reference HEU and LEU cores are presented in Figs. 9-14.





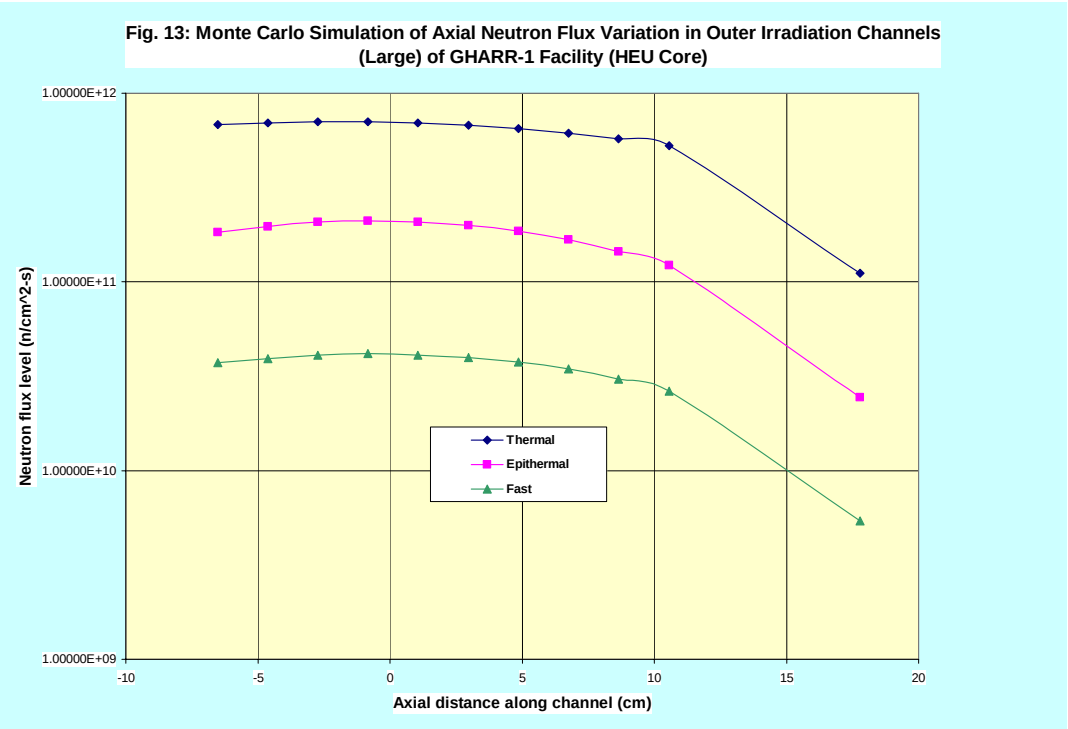
As indicated earlier, the same trend is observed for both the HEU and LEU fuels. A comparison of the thermal neutron flux variation in the inner irradiation channels for HEU and 3 LEU fuels are illustrated in Fig. 11. The differences however, exist in the relative values simulated for each fuel type as shown in Fig. 12

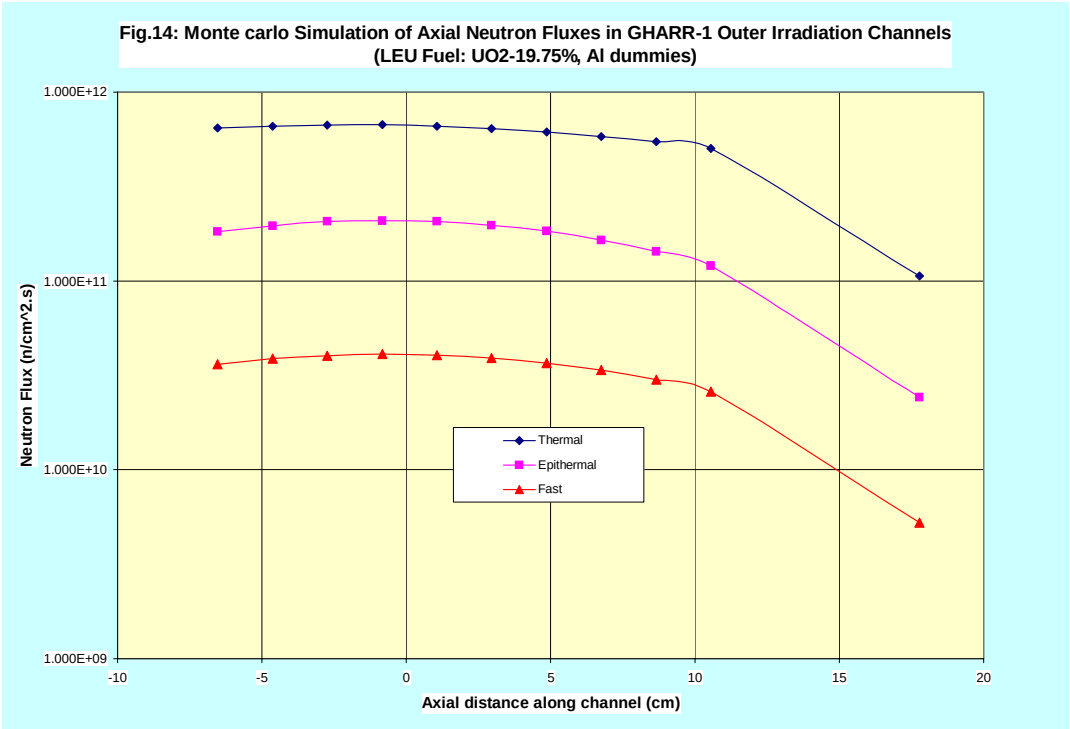




It has been reported worldwide that HEU-LEU core conversions oftentimes result in a neutron flux "trade-off". In this study as illustrated in Fig. 12, the neutron flux "trade-offs" or losses reported for three of the LEUs considered in this study are within 6%-10% with respect to the reference HEU core from the low flux UO_2 (12.6% U-235). The "losses" observed for the two dispersant fuels (U_3Si_2, U-9Mo) are relatively lower compared with the pelletized UO_2 fuel. The reduced thermal neutron fluxes in the inner irradiation channels as shown by the Monte Carlo simulation are, however, higher than 5.0E+11 n/cm^2.s In general, the GHARR-1 facility is operated at half full power (15kW) corresponding to a thermal neutron flux of 5.0E+11 n/cm^2.s. Hence these "losses" will not affect utilization of the Ghana MNSR facility since the simulated results show that all fuels are capable of producing thermal fluxes within 10% of the rated maximum of 1.0E+12 n/cm^2.s recorded experimentally for the reference HEU core. The Monte Carlo simulations for the other LEU fuels also show similar results. Thus, neutronically, it can be concluded that all the five LEU fuels qualify as candidate LEU options for core conversion of the GHARR-1 facility.

The results obtained for the Monte Carlo simulation of the outer irradiation channels are similar to that observed for and exhibit the same trend for all the fuel types. In Figs. 13 and 14, the axial neutron flux variation in the outer irradiation channels for the reference HEU and UO_2 (19.75% + Al dummies) are illustrated.



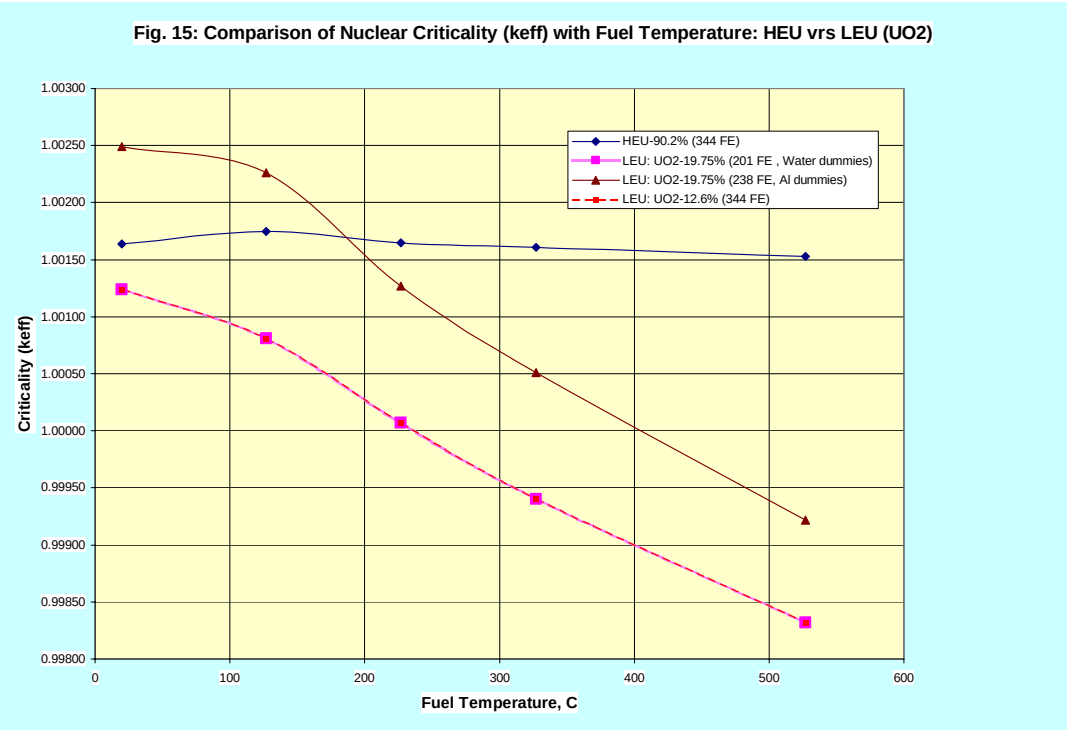


Progress or advances in calculating the kinetic parameters of the Ghana MNSR using the Monte Carlo approach have also been made in the neutronic analysis for the GHARR-1 HEU-LEU conversion studies. Table 2 shows the Monte Carlo (MCNP5) simulated results for the delayed neutron fraction for the reference HEU and candidate LEU cores.

	Criticality		Effective delayed neuttron	
Fuel	k_{eff} (Total)	k_{eff} (delayed)	β_{eff} MCNP5	β_{eff} (EXT-2)
HEU: U-Al fuel (90.2%)	1.00454	0.99618	8.3541E-03	8.0800E-03
LEU: UO_2 fuel 344 FE, 12.6%)	1.00454	0.99621	8.3240E-03	
LEU: UO_2 fuel (19.75%, 238 FE + 116 Al dummies)	1.00481	0.99636	8.4402E-03	
LEU: UO_2 fuel (19.75%, 201 FE + 153 H_2O dummies)	1.00434	0.99609	8.2543E-03	
LEU: U_3Si_2 fuel (19.75%, 344 FE)	1.00431	0.99602	8.2874E-03	
LEU: U-9Mo fuel (19.75%, 344 FE)	1.00432	0.99591	8.4082E-03	

The results reported in the GHARR-1 SAR [8] were computed using the diffusion code EXTERMINATOR-2. From Table 2, the results of both the Monte Carlo (MCNP5) and the diffusion (EXTERMINATOR -2) calculations are agreeable.

Other kinetic parametric simulations performed in the HEU–LEU included calculations of the prompt neutron lifetimes and reactivity coefficients for the reference HEU and candidate LEU cores. For brevity, only a few of the Monte Carlo simulated results are presented in this paper. A comparison of the nuclear criticality and reactivity variations with fuel, moderator and void temperatures for HEU (U-Al) and LEU (UO_2) cores are illustrated in Figs. 15 -20 respectively.



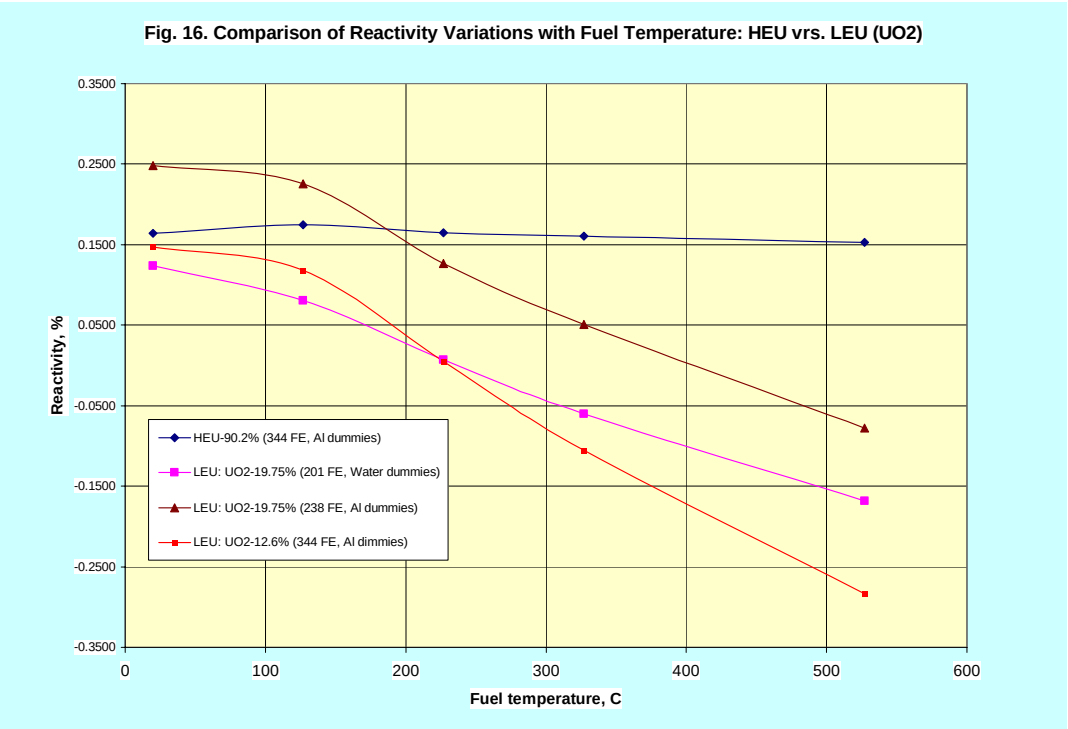


Fig. 16. Comparison of Reactivity Variations with Fuel Temperature: HEU vrs. LEU (UO2)
Reactivity, %
0.3500
0.2500
0.1500
0.0500
-0.0500
-0.1500
-0.2500
-0.3500
0
100
200
300
400
500
600
Fuel temperature, C
HEU-90.2% (344 FE, Al dummies)
LEU: UO2-19.75% (201 FE, Water dummies)
LEU: UO2-19.75% (238 FE, Al dummies)
LEU: UO2-12.6% (344 FE, Al dimmies)

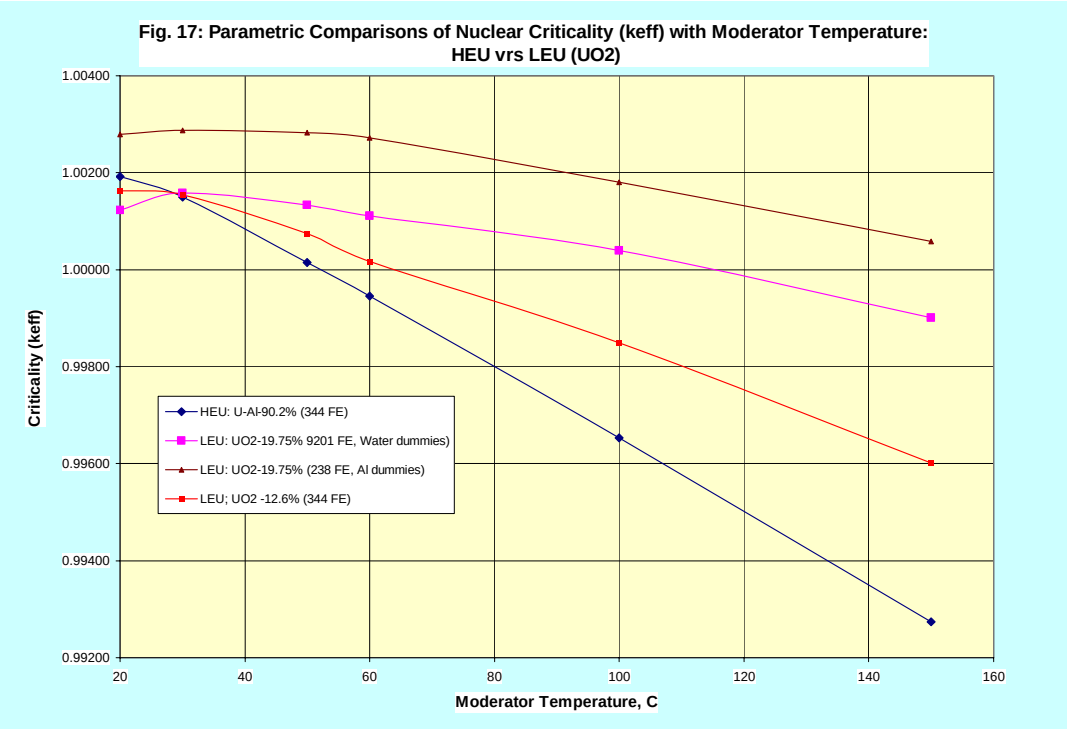


Fig. 17: Parametric Comparisons of Nuclear Criticality (keff) with Moderator Temperature: HEU vrs LEU (UO2)
Criticality (keff)
1.00400
1.00200
1.00000
0.99800
0.99600
0.99400
0.99200
20
40
60
80
100
120
140
160
Moderator Temperature, C
HEU: U-Al-90.2% (344 FE)
LEU: UO2-19.75% 9201 FE, Water dummies)
LEU: UO2-19.75% (238 FE, Al dummies)
LEU; UO2 -12.6% (344 FE)

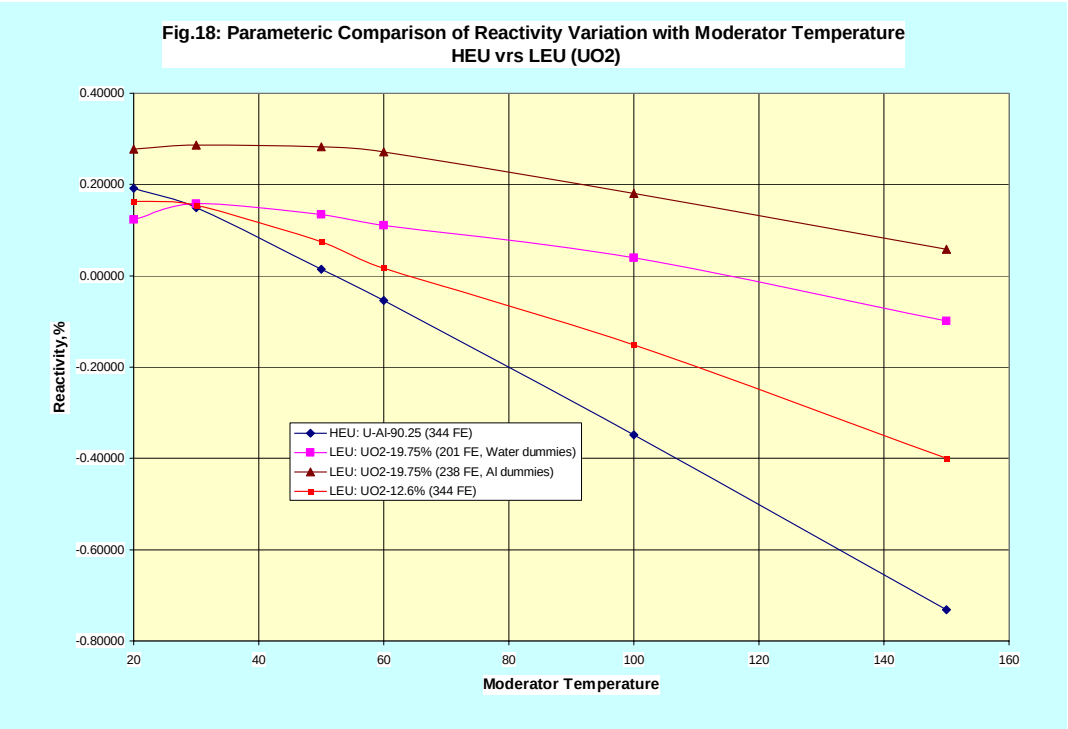


Fig.18: Parameteric Comparison of Reactivity Variation with Moderator Temperature
HEU vrs LEU (UO2)
Reactivity,%
Moderator Temperature
0.40000
0.20000
0.00000
-0.20000
-0.40000
-0.60000
-0.80000
20
40
60
80
100
120
140
160
HEU: U-Al-90.25 (344 FE)
LEU: UO2-19.75% (201 FE, Water dummies)
LEU: UO2-19.75% (238 FE, Al dummies)
LEU: UO2-12.6% (344 FE)

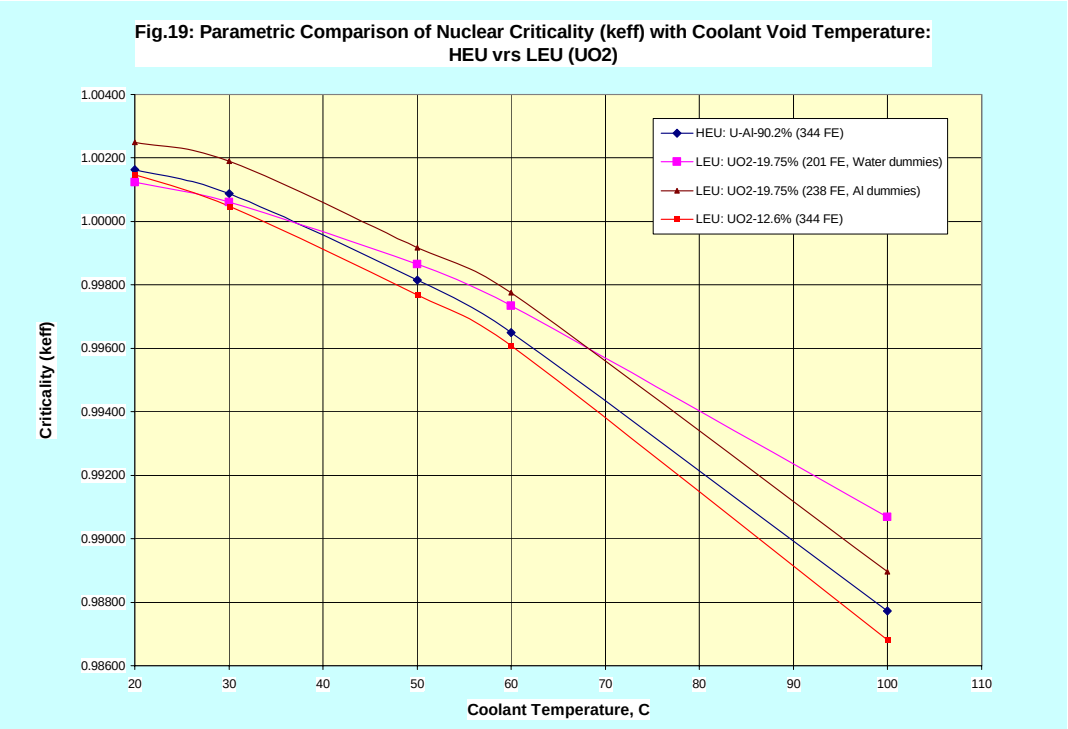
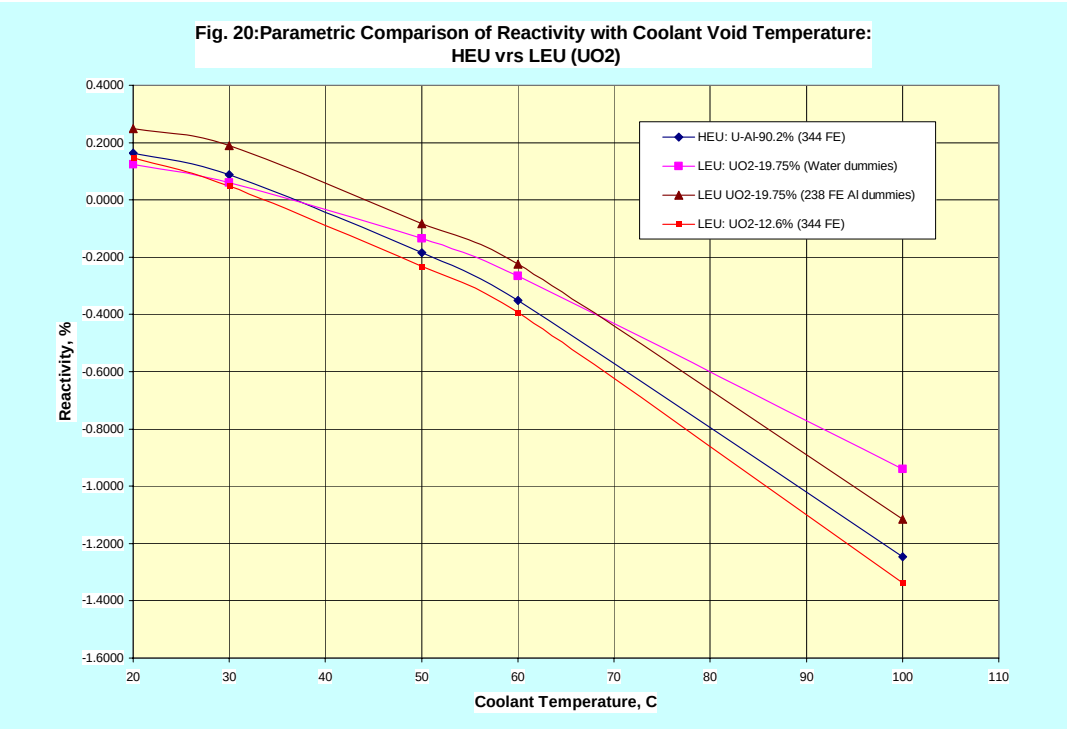


Fig.19: Parametric Comparison of Nuclear Criticality (keff) with Coolant Void Temperature:
HEU vrs LEU (UO2)
Criticality (keff)
Coolant Temperature, C
1.00400
1.00200
1.00000
0.99800
0.99600
0.99400
0.99200
0.99000
0.98800
0.98600
20
30
40
50
60
70
80
90
100
110
HEU: U-Al-90.2% (344 FE)
LEU: UO2-19.75% (201 FE, Water dummies)
LEU: UO2-19.75% (238 FE, Al dummies)
LEU: UO2-12.6% (344 FE)



5. Conclusion

Significant progress or advances have been made in the HEU-LEU core conversion activities for the GHARR-1 (Ghana MNSR) facility. In particular, the preliminary Monte Carlo model developed earlier using the MCNP4c3 particle transport code has been fine-tuned to account for some material compositions and improve computational capabilities. The improved model, as seen from the demonstrated results, correctly simulates the neutronic and other design parameters of the MNSR reactor. The model further demonstrates its capability as a competent tool for performing any neutronic analysis of MNSR reactors. Following this, the model has been adopted to correctly simulate the same activities for the Nigerian MNSR with success. Thus, the GHARR-1 Monte Carlo model developed using the versatile MCNP particle transport code has proved to be very effective for simulation of reactor physics design parameters and as well as performing global neutronics analysis of MNSRs, including HEU-LEU core conversions studies.

Further work on fuel depletion, burnup credit analysis and thermal hydraulic in favour of the HEU-LEU conversion studies are on-going presently. It is anticipated that the completed work will further provide sound neutronic and thermal hydraulic bases for selecting the most suitable candidate LEU core for converting the GHARR-1 and other MNSR facilities.

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