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Conversion of International MNSR – Reference Case of Ghana MNSR

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ABSTRACT

This study was undertaken to design an LEU core with similar operational capabilities as the original HEU core and with acceptable safety margins under both normal and accident conditions. In order to provide comparisons between the proposed LEU core and the initial GHARR-1 HEU core, thorough analyses were performed for both cores. The proposed LEU core consists of UO_2 fuel elements clad in Zircaloy-4 alloy. The control element of the control rod material will remain unchanged but the diameter of the cadmium absorber would increase, leaving the diameter of the control rod unchanged. It is revealed that throughout the lifetime of the proposed LEU core the shutdown margin meets specification limits and there will be no tradeoff in the thermal neutron fluxes in the experimental channels. The latter will be achieved by increasing the power of the LEU core by 13 %. Other major neutronic parameters that were computed and compared for the HEU and LEU cores are shown in this report.

1. Introduction

The International Atomic Energy Agency's Coordinated Research Project on "Core Conversion of MNSR facilities" was initiated in 2005. And in 2006, this was approved and studies started in 2006 to formulate work plan. Four phases of work was scheduled as follows: The first phase is to characterize current HEU core, Perform LEU feasibility study – U3Si2, U-9Mo, UO2 and select an LEU fuel. The second phase is review existing SAR to identify necessary changes and review OLCs and PIEs. The third phase is to complete Conversion SAR and prepare startup (Commissioning Plan) for the LEU Core. And finally, phase four is to prepare documentations for shipping and receiving spent HEU core and fresh LEU core. The fuel selected under that study is UO_2 with an enrichment of 12.45 %. The enrichment has now been fine tuned to 12.5% at a Technical Meeting on the Core Conversion Project.
The objective of this report is to show further studies that have been undertaken, using present parameters proposed at the meeting, to design an LEU core with similar operational capabilities as the original HEU core and with acceptable safety margins under both normal and accident conditions. In order to provide comparisons between the proposed LEU core and the initial GHARR-1 HEU core, thorough analyses were performed for both cores. The proposed LEU core consists of UO_2 fuel elements clad in Zircaloy-4 alloy. The control element of the control rod material will remain unchanged but the diameter of the absorber material would increase whiles leaving the diameter of the control rod unchanged. The current parameters for the proposed core are compared with the current HEU core parameters in table 1.

Table 1. Comparison of Key Parameters for HEU and LEU Core.

Parameter	HEU	LEU
Fuel Meat	UAl_4 in Al matrix	UO_2 pellets
Fuel Clad Type	303-1 Al alloy	Zircaloy-4
Enrichment (U-235 wt%)	90.2	12.5
Density of fuel meat(g/cm^3)	3.456	10.6
U-235 loading per pin / Core (g)	2.9 / 998.12	3.9 / 1357.86
Uranium wt. (%)	27.5	88
No. of Fuel Pins/ Power	344 / 30 kW	348 / 34 kW

In the following sections of this document, it is revealed that throughout the lifetime of the proposed LEU core: The shutdown margin meets Technical Specification limits; Reactivity coefficients meet required limits and are comparable to the existing HEU core; Fuel integrity is maintained under all operating conditions and that there will be no tradeoff in the thermal neutron fluxes in the experimental channels (achieved by increasing the power of the LEU core to 34 kW).

2. Method of Analyses

2.1. Neutronics Analysis

The 3-D GHARR-1 Monte Carlo code was simulated to estimate some reactor physics parameters such as nuclear criticality and core reactivities using the current parameters. Models were fine tuned to reflect current parameters. A new version of the code, MCNP5/X was used for these calculations. Neutron flux distribution in some selected locations of the reactor was also analyzed. In particular, neutron transport simulations were done for a clean fresh core (zero burn-up). The GHARR-1 Monte Carlo model was further simulated for total withdrawal and fully inserted of the control rod to determine control rod worth and shutdown margins. Simulations were also performed for different positions of the control rod for the control rod calibration curve show in fig. 2.1. The radius of control rod for the LEU is slightly increased as proposed for the Core Conversion of MNSR's.

2.2. Thermal Hydraulic analysis.

The PLTEMP/ANL V4.1 was validated for the Ghana's MNSR. Four input data files were used in the PLTEMP/ANL V4.1 code to calculate the safety margins in the steady-state operation of GHARR-1 with HEU core. In addition, an input file giving the axial power shape of the fuel pin modeled (the average power pin or the peak power pin in the HEU core) was also used with the four input data files. Another set of four similar input data files were used to calculate steady-state safety margins of GHARR-1 with LEU core at both 30 kW and 34 kW. In addition, an input file giving the axial power shape of the fuel pin modeled (the average power pin or the peak power pin in the LEU core) was also used with each set of the four input data files; this is required by the PLTEMP/ANL V4.1.

One set of input models one (average fuel pin) of the 344 or 348 fuel pins in the HEU or LEU core respectively with a reactor power of 15 kW and a coolant inlet temperature of 24.5 °C. The pin is modeled as a solid rod of radius 2.15 mm in a 0.6 mm thick cladding, without any gap resistance in the case of HEU core. This input data file was used to calibrate the hydraulic resistance in the PLTEMP/ANL model to reproduce an experimentally measured coolant temperature rise of 13 °C (from 24.5 °C to 37.5 °C).

PARET: "Program for the Analysis of REactor Transients" code was developed for testing methods and models and for subsequent applications in the analysis of transient behaviour in research reactors. For PARET applications, the reactor core can be represented by one to four regions. Each region may have different power generation, coolant mass flow rate, and hydraulic parameters as represented in a single fuel pin with its associated coolant channel. The heat transfer in each fuel element is computed on the basis of one-dimensional conduction solution, providing for a maximum of 21 axial segments. The code has been used for transient analysis of GHARR-1.

3. Results and Discussion.

3.1. Criticality Results

The core excess reactivity calculated for the LEU UO_2 fuel with 344 fuel pins was below the 3 mK which is insufficient for the design of MNSR core. Hence the number of pins was increased to 348 to achieve the design reactivity of MNSR which is between 3. 5 mK and 4.0 mK . This is evident in table 2.

Table 2. Comparison of Reactivities for various cores.

Fuel / No. of Pins	Keff	Reactivity, mK
HEU; 344 pins	1.00375 ± 0.00005	3.74±0.05
LEU; 344 pins	1.00289 ± 0.00006	2.88±0.05
LEU; 348 pins	1.00389 ± 0.00004	3.87±0.04

The Criticality results for the HEU and 348-pin LEU cores are shown in table 3. The Multiplication factors, K_{eff}, and of course the reactivities are quite comparable and also compare well with values stated in the HEU SAR. The delayed neutron fractions for the two cores as estimated by Monte Carlo N Particle Code are 3.3 % and 3.9 % higher than MNSR manufacturer's quoted value of 0.00808 [Guo Chengzhan et al., 1991] respectively. Nevertheless, the two compares well with the delay neutron fraction of 0.00857 reported for NIRR-1 [Jonah, S. A. et al., May 2008].

Table 3. Comparison of Criticality Results for HEU and LEU

Criticality Result	HEU	Sigma	LEU	Sigma
Keff –Control rod completely withdrawn	1.00375	0.00005	1.00385	0.00004
Keff - Control rod fully inserted	0.99680	0.00004	0.99714	0.00004
Ractivity, mK	3.74	0.05	3.87	0.04
Delayed neutron fraction (β_{eff}), mK	8.347	0.0641	8.395	0.0566
Prompt Neutron lifetime (Λ), s	8.46×10^{-5}	0.06×10^{-5}	7.39×10^{-5}	0.06×10^{-5}
Control rod worth, mK	6.95	0.018	6.74	0.017
Shutdown margin, mK	3.21	0.012	2.87	0.011

The design control rod worth of the reactor is 6.8 mK and the shutdown margin is 3.0 mK for maintaining the reactor in safe shutdown conditions. The total cold excess reactivity to be compensated is about 4.0 mK by the control rod. The Monte Carlo MCNP calculation of the control rod worth is about 10.5 % more for the HEU core even though the control rod diameter has been increased (for the LEU core with the same CR dimensions as HEU, the worth will be lesser).

The exact effect of control rods on reactivity can be determined experimentally. For example, a control rod can be withdrawn in small increments, such as 1 cm, and the change in reactivity can be determined following each increment of withdrawal. By plotting the resulting reactivity versus the rod position, a graph obtained for both Cores are shown in figure 1. The graph depicts integral control rod worth over the full range of withdrawal.

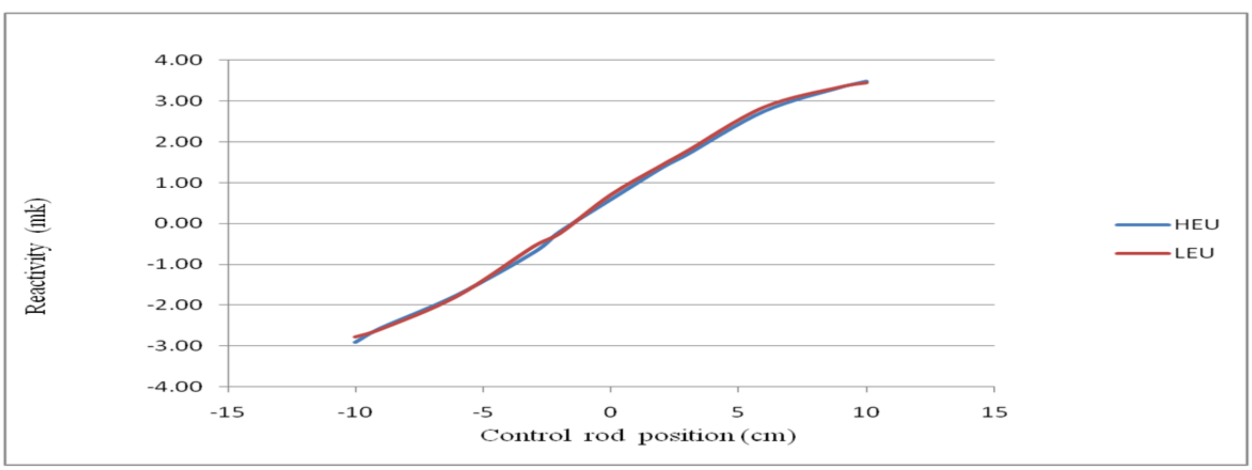

Fig. 1: The Integral Control Rod Curve.

The *integral control rod worth* is the total reactivity worth of the rod at that particular degree of withdrawal and is usually defined to be the greatest when the rod is fully withdrawn.
Measurement of neutron flux and neutron energy spectrum parameters in the inner irradiation sites can be utilised to determine linearity, repeatability and stability of the neutron measurement system, which includes detectors and secondary instrument. The LB1120 miniature fission chamber is employed as a neutron detector for the reactor. It has a small size and can be put into the side annulus. In the linear range of this detector the absolute neutron flux over 4-5 decades could be measured with both gold and manganese foils.
The average flux distributions in the inner irradiation channels, outer irradiation channels and that of the fission chambers are shown in the figures 2 to 5 respectively.

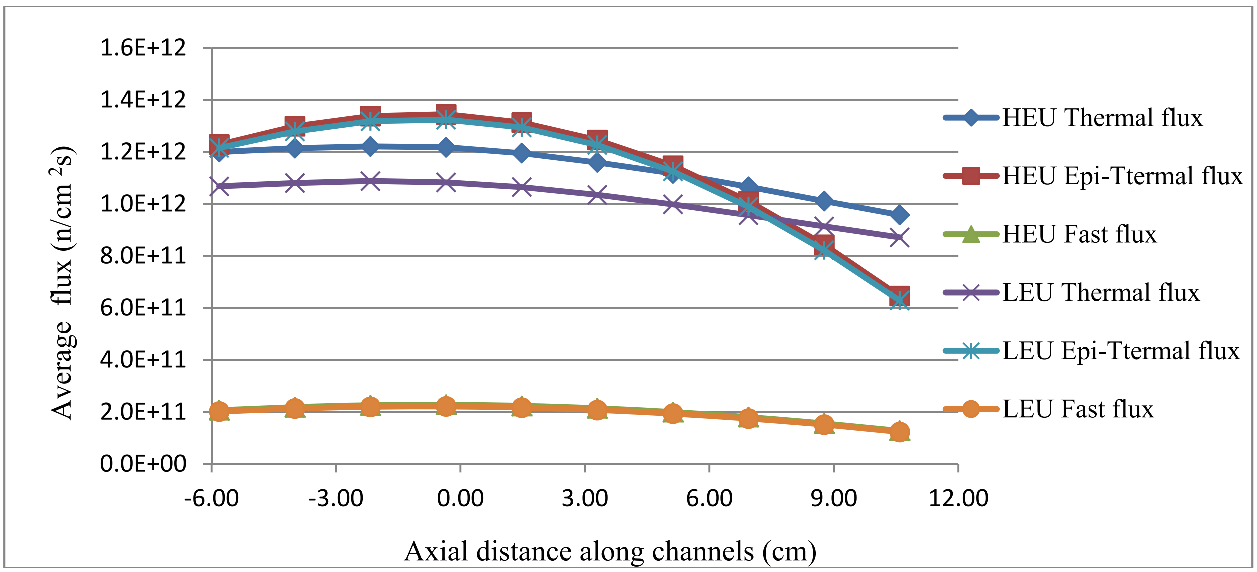

Fig. 2: Comparison of average flux distribution in inner irradiation channel at 30 kW.

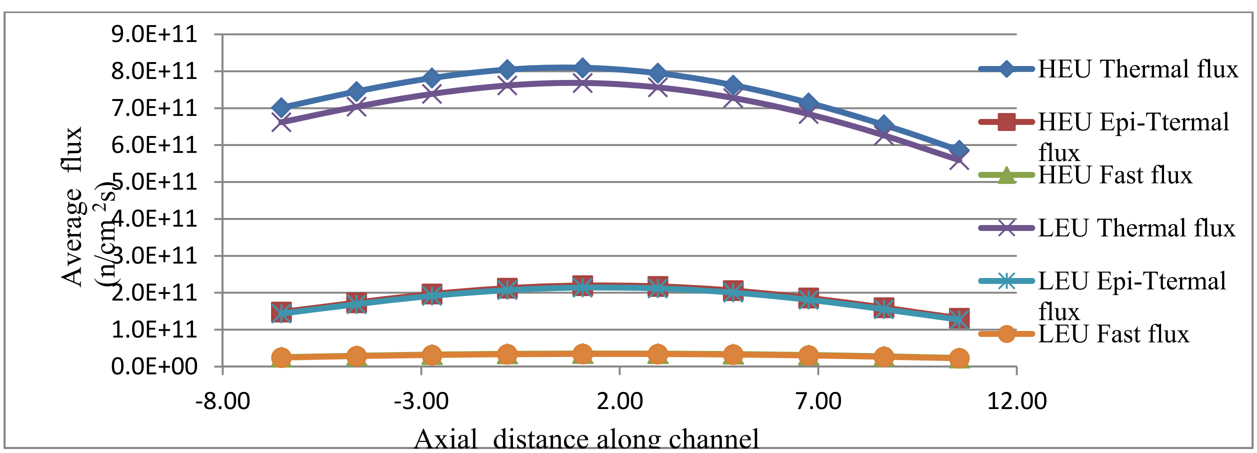

Fig. 3: Comparison of average flux distribution in outer irradiation channel at 30 kW.

The centre of the Core is equidistant from the inner irradiation channels and the fission chamber which houses the device used in measuring the neutron flux experimentally. The various graphs follow the same pattern and also depict the reduction in the thermal neutron flux of the LEU

Core at 30 kW.
In order not to compromise the thermal neutron flux especially in the inner irradiation channel, the power of the LEU core is increased by 13 % to recompense the fall in flux at 30 kW for the LEU core. Base on the average ratio of the thermal neutron flux in the inner irradiation channel at 30 kW of the LEU core to that of the HEU core, the power for LEU core is increase to 34 kW. This is to normalize the thermal neutron flux ratio in the inner irradiation channels to unity. So the two profiles of the thermal flux are almost completely superimposed on the other as observed in figure 2.6 below. The effects of the increase in power of the LEU core on the neutron fluxes in the other locations are shown in fig. 2.7 and 2.8.

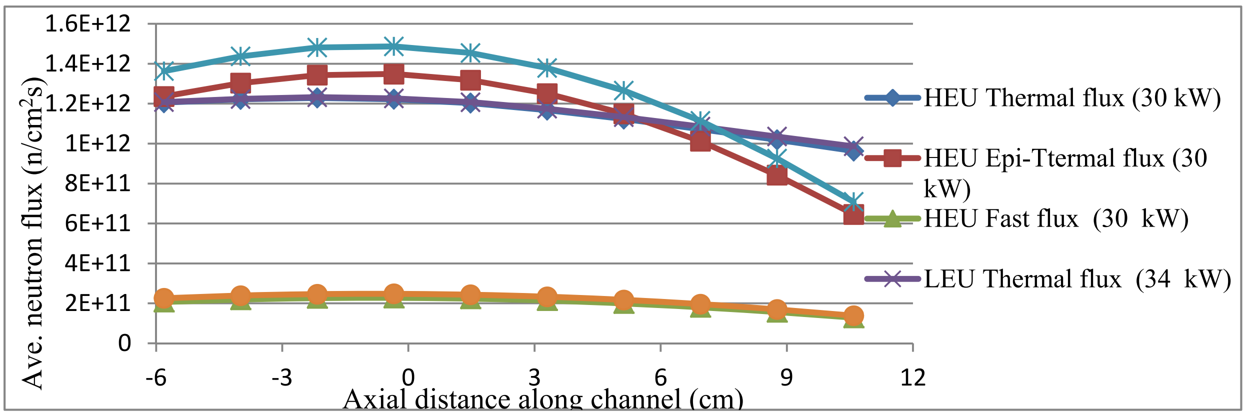

Fig. 4: Comparison of average flux distribution in inner irradiation channel at nominal powers.

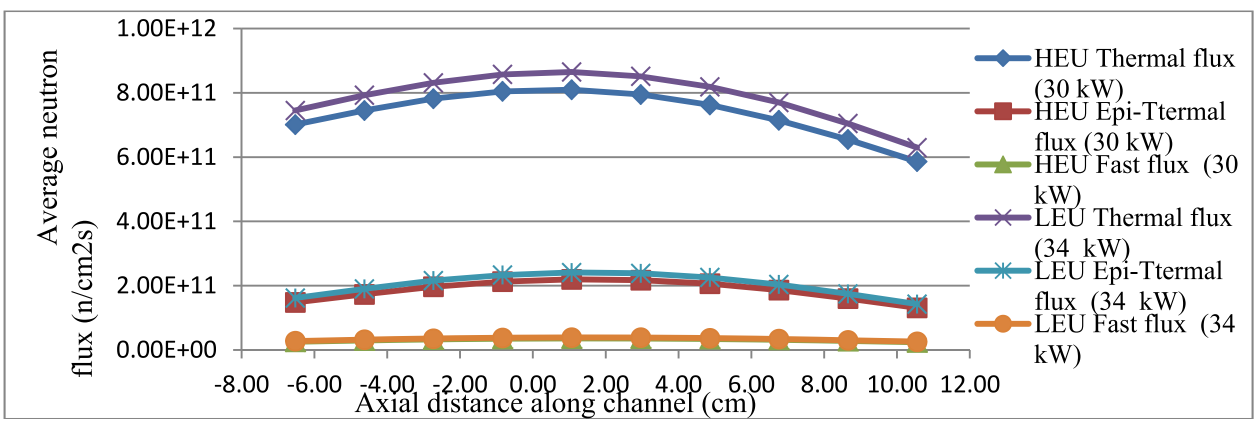

Fig. 5: Comparison of average flux distribution in outer irradiation channel at nominal powers.

Reactivity changes due to changes in the physical properties of the materials in the reactor. Reactivity coefficients are useful in quantifying the reactivity change that will occur due to the change in a physical property such as the temperature of the moderator. The temperature coefficient can conveniently be considered to consist of three partial contributions: nuclear temperature, density temperature and volume temperature coefficients. Some reactivity coefficients evaluated for the Core Conversion are shown below.
The fuel temperature coefficient for both cores at various temperatures is shown in table 4. That for the LEU fuel is consistent since the coefficients computed are all negative and hence makes

the core more inherently stable than the HEU core.

Table 4 Fuel Temperature Coefficient

Temperature (˚C)	HEU (mK / ˚C)	LEU (mK /˚C)
20	-	-
126.85	$(1.02\pm0.001)*10^{-3}$	$(-6.69\pm0.005)*10^{-3}$
226.85	$(4.80\pm0.020)*10^{-4}$	$(-9.57\pm0.004)*10^{-3}$
326.85	$(-9.71\pm0.002)*10^{-5}$	$(-10.2\pm0.003)*10^{-3}$
526.85	$(-9.79\pm0.002)*10^{-5}$	$(-9.92\pm0.002)*10^{-3}$
Average	0.000326682	$(-9.08\pm0.002)*10^{-3}$

The moderator temperate coefficient is shown in table 5 is done for varying temperature and corresponding density. The average values for the core cores calculated by the MCNPX code are comparable and acceptable since both values are negative indicating inherent safety of both cores.

Table 5. Moderator temperature coefficient.

Temperature (°C)	HEU (mK /°C)	LEU (mK /°C)
32		
50	-0.15883	-0.18768
60	-0.18148	-0.20405
70	-0.39561	-0.40581
100	-0.24293	-0.25832
Average	-0.24471	-0.26397

2.2. Thermal Hydraulic Results

Thermal hydraulic parameters obtained from studies undertaken on both the HEU and LEU cores at nominal reactor powers are show in the tables 6. The results of the calculations for the clad surface and coolant temperatures using an inlet temperature of 30 °C and a coolant pressure of 1 bar are also shown in this table.

Table 6. Comparison of HEU and LEU steady-state parameters using PLTEMP/ANL

Parameter	**HEU – 344 rods**	**LEU – 348 rods**	**LEU – 348 rods**
Power (kW)	30.0	30.0	34.0
Core Flow Rate (Kg/S)	1.1E-3	1.1E-3	1.2E-3
Peak Fuel Temp. (°C)	104	-	142
Max. Clad Surface Temp. (°C)	77.3	95.0	98.3
Max. Coolant Temp. (°C)	53.1	53.4	57.1

For the LEU core the nominal power is raised to 34 kW in order to meet the flux level of 1×10^{12} n/cm2.s. Hence the computations, using PLTEMP, were performed for the LEU core at this power and the steady-state parameters were also compared with those of HEU and LEU at 30 kW in table 6.

The effect of inlet temperature on temperature difference, as computed by PLTEMP, for both HEU and LEU are shown in table 7.

Table 7. Effect of Inlet Temperature on Temperature Difference at Nominal Operating Power for the HEU and LEU Cores

T_{IN} (ºC)	30 kW		36 kW	
	HEU – ΔT (ºC)	LEU – ΔT (ºC)	HEU – ΔT (ºC)	LEU – ΔT (ºC)
10	24.10	29.15	27.00	32.28
15	21.63	27.16	24.20	30.20
20	20.20	25.59	22.66	28.54
30	18.60	23.26	20.97	26.03
35	18.30	22.37	20.63	25.07
40	18.03	21.61	20.54	24.24

PARET code was utilized for the Transient Analysis in order to compare the reactor power, fuel temperature and clad temperature for the two cores. Results have been shown in figures 6, 7 and 8.

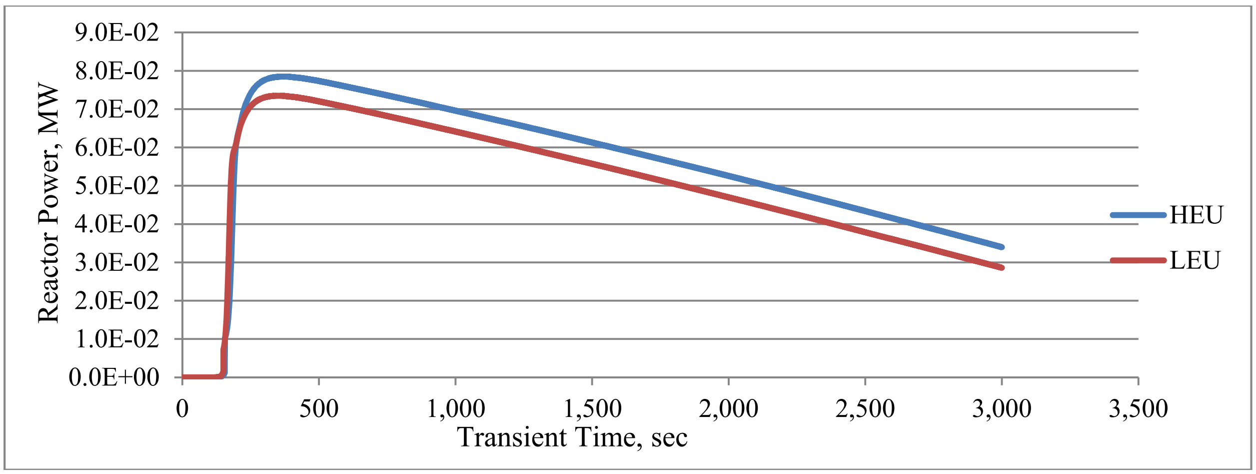

Fig. 6. Power vs. Time for a 3.8 mK Reactivity Insertion with HEU and LEU Fuel

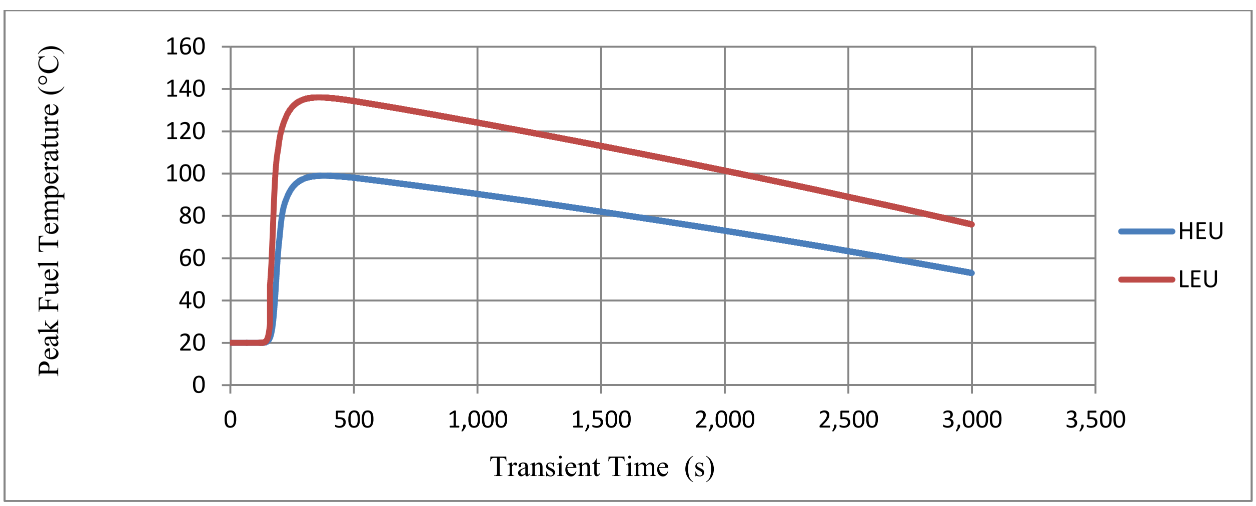

Fig. 7. Fuel Temperature comparison of HEU and LEU cores for 3.8 mK reactivity transient.

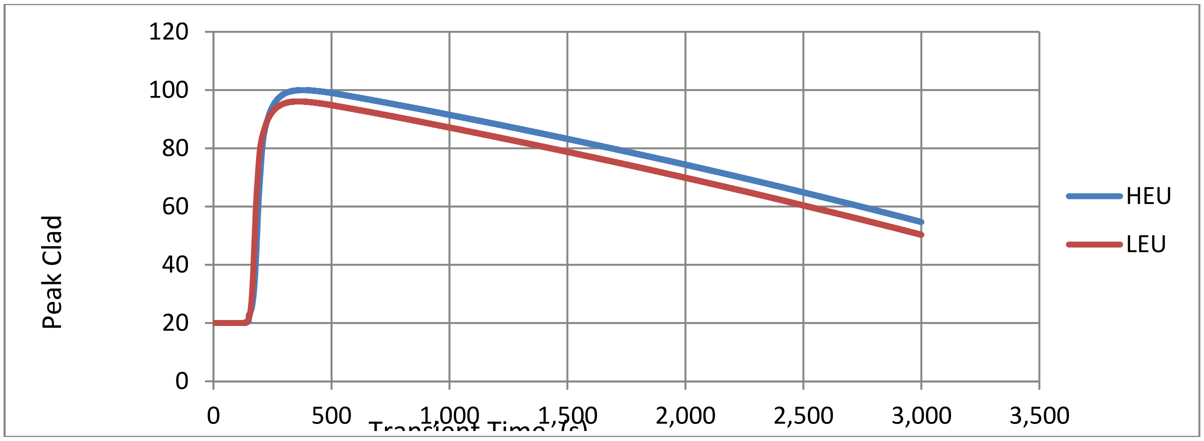

Fig. 8 Clap Surface Temperature comparison of HEU and LEU cores for 3.8 mK reactivity transient.

The peak temperature for the fuel as shown in the table 8, is far below it melting point of 2800 °C and that of the clad is also far below its melting point of 1850 °C, indicating good safety margins.

Table 8 Peak power, peak fuel temperature and peak clad temperature for various reactivity insertions.

Reactivity Insertion, mK	Peak power, kW	Peak Fuel Temp, °C	Peak Clad Temp, °C
3.8	73.5	136	96.1
6.0	140	200	122
8.0	350	254	126

4. Conclusion

Ghana is committed to ensuring the success of the IAEA-RERTR HEU-LEU conversion program and 12.5 % enriched UO2 has been chosen as fuel for LEU Core. For core excess reactivity of 4 mK, 348 fuel pins would be appropriate for the GHARR-1 LEU Core. Results indicate that flux distribution in the inner irradiation channels will not be compromised, if the power of LEU core is increased to 34 kW.

5. References

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