Original Research Article

PERFORMANCE OF 19.75% UO_2 FUEL MATERIAL IN THE CORE OF NIGERIA MINIATURE NEUTRON SOURCE REACTOR (MNSR)

ABSTRACT

Investigation has been done concerning the performance of 19.75% Uranium Dioxide (UO_2) fuel material in the core of the Nigeria Research Reactor-1 (NIRR-1), a Miniature Neutron Source Reactors (MNSRs) manufactured by China Institute of Atomic Energy (CIAE) using SCALE 6.1 code system and VENTURE-PC code. This is in line with the current efforts to convert the core of NIRR-1 from Highly Enriched Uranium (HEU) core (90.2% enriched UAl_4-Al fuel) to Low Enriched Uranium (LEU) core (19.75% enriched UO_2-Zircaloy-4 fuel). The geometry and dimensions of HEU and LEU cores were exactly the same except the increase in the fuel cell diameters from 1.2384cm to 1.632cm. Results obtained shows that the total control rod worth of 7.23mk (723pcm), clean cold core excess reactivity of 4.04mk (404pcm),k_{eff} of 1.0119634, shutdown margin of 3.19mk (319pcm) and neutron flux profile of $1.24 \times 10^{12}\ \mathrm{ncm^{-2}s^{-1}}$ for the potential LEU core were slightly greater than that of the current HEU core.

Keywords: NIRR-1, core, LEU, HEU, SCALE 6.1 code, VENTURE-PC code, Neutronics, geometry, dimension, control rod worth, excess reactivity, k-effective, shutdown margin, neutron fluxes, model.

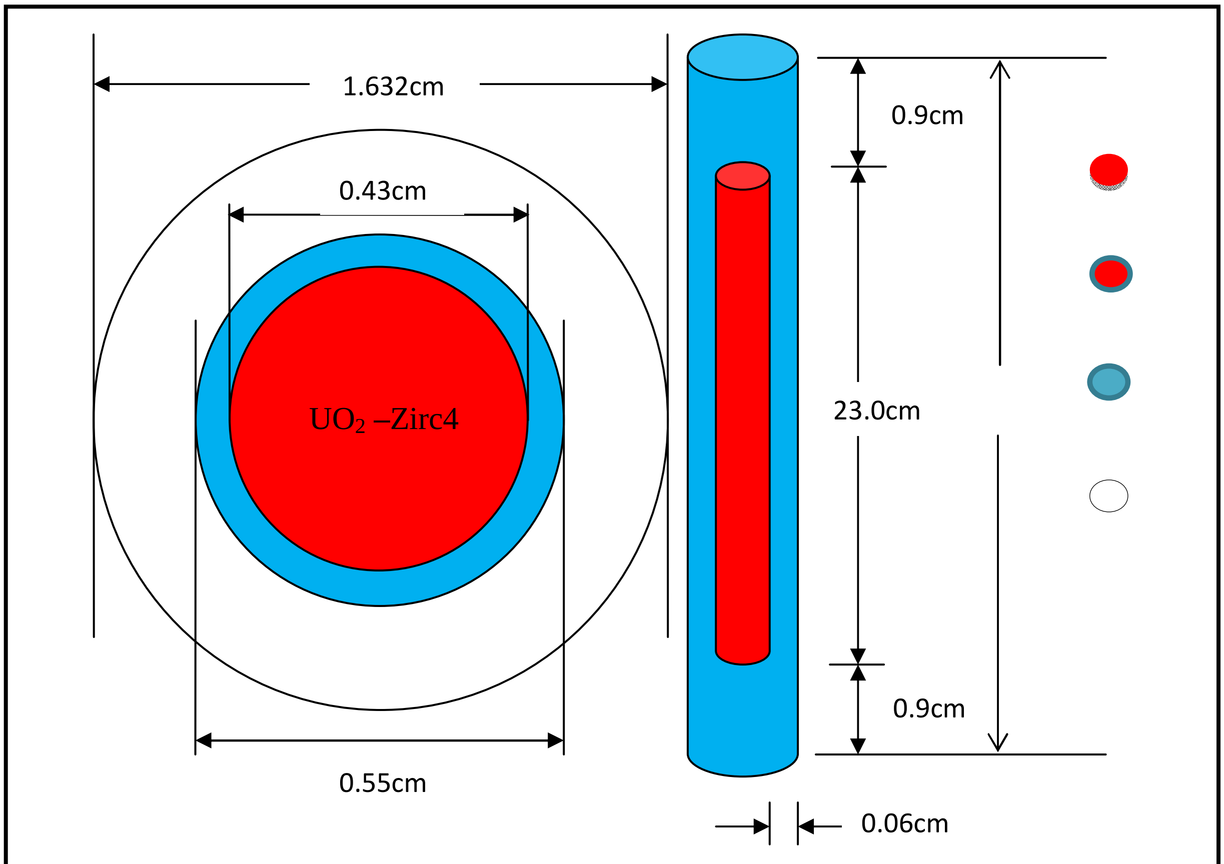
INTRODUCTION

Despite the availability of numerous type of nuclear fuel materials that can be used in reactor systems, majority of commercial reactors in the world are using Uranium Dioxide (UO_2) as fuel (Sunghwan, 2013), the most common ceramic fuel. Some of the benefits of using UO_2as reactors fuel include:strong non-proliferation characteristics, chemical inertness,compatibility with potential cladding materials such as stainless steel and Zircaloy, dimensional stability under irradiation, very high melting pointand excellent resistance to corrosion when exposed to high temperature and pressure (Lyons et al., 1972; Sunghwan, 2013). The Nigeria Research Reactor-1 (NIRR-1) is one of the few reactors in the world with a core that requires conversion from HEU to LEU fuel. A number of feasibility studies have been carried out for this reactor to investigate the possibility of using 12.5% UO_2 material to convert the core from HEU to LEU fuel (Jonah et al., 2009; Salawu, 2012; Jonah et al., 2012; Ibrahim et al., 2013). The results from these types of studies using various nuclear analysis tools (such as MCNP, CITATION and VENTURE-PC), has shown that there will be a slight reduction in the thermal neutron flux in the core of NIRR-1 when fuel with 12.5% UO_2 material. In addition, these studies have also revealed that the hydrogen to uranium ratio will decrease from about 180 in the current HEU core of NIRR-1 to about 18 in the proposed LEU core (Salawu, 2012). This could be the possible cause of the observed reduction in the thermal neutron flux of NIRR-1 as the core is left with less number of hydrogen to thermalize the neutron. Our major interest in this particular study is to find a means of increasing the hydrogen content in the core by replacing 12.5% UO_2 material in the proposed LEU core with 19.75% UO_2 material plus a corresponding decrease in the number of fuel pins in the core. Decreasing the number of fuel pins in the core from 347 to 200 will give room for more moderators in the core and this could increase the number of hydrogen available to thermalize the neutron in the proposed LEU core for NIRR-1. Hence the hydrogen to uranium ratio will increase with a corresponding increase in the thermal neutron flux. The current Nigeria research reactor core was originally designed by China with a diffusion theory codes, HAMMER and EXTERMINATOR-2 (FSAR, 2005). These design calculations were repeated in Nigeria with WIMS/CITATION and MCNP (Balogun, 2003). A recent version of the diffusion theory code called VENTURE-PC were used in this work to perform the neutronics analysis with a recent version of SCALE (SCALE 6.1) code system to generate a cross section library for the proposed LEU core for NIRR-1. A licensed user of the codes performed the actual calculations and generated the output data used to perform this analysis. We carryout comprehensive neutronics analysis of Nigeria Research Reactor-1 (NIRR-1) core using 19.75% UO_2 material as the fuel. In this work the effective multiplication factor for the system, excess reactivity, the reactivity worth of the control material, the shim worth and power distribution at different locations within the Nigeria Research Reactor-1 (NIRR-1) core were determined. In addition the relative flux levels at different location within the system were calculated. These locations include the inner and outer irradiation sites in the core of NIRR-1 system using 19.75% UO_2 material as the fuel.

MATERIAL AND METHOD

Uranium dioxide (19.75% UO_2) fuel of volume density 10.6g/cm^3 is the proposed LEU fuel material selected to perform this core conversion study for NIRR-1 with Zircaloy-4 as the cladding material. Zircaloy-4 has a density of 6.56g/cm^2 with a natural Zirconium of 98.23 weight percent (w/o) (Salawu, 2012). The geometry, dimensions and material composition of other reactor components in the proposed LEU core will be the same as inthe HEU core except a decrease in the fuel cell radius caused by a reduction in the number of fuel pins in the core of

NIRR-1. There are approximate 200 active fuel rods of LEU fuel materials (19.75% UO_2) in the proposed LEU core for NIRR-1. The three (3) aluminum dummy pins and four (4)aluminium tie rods in the HEU core were replaced by Zircaloy-4 material of the same dimensions. The active fuel length, active fuel diameter and fuel cell diameter are 23.0cm, 0.43cm and 1.632cm respectively (figure 1.0). The uranium in the active fuel region (indicated with red colour) of the LEU fuel material in figure 1.0 is enriched to 19.75% U-235 with each fuel rod containing 6.162g of U-235.



*** Equivalent diameter of a unit fuel cell for NIRR-1**

Figure 1.0: The height and diameter of the active fuel cell and fuel rod models for the potential LEU core for NIRR-1

The computed average homogenized atom density (N_{iz}) is done by multiplying the region atom density (N_{ij}) by the region volume fraction (f_i) for the zones in the NIRR-1 fuel cell (equations1.0 and 2.0).

$$N_{iz} = \frac{\sum_{j \in Z} N_{ij} V_j}{\sum_{j \in Z} V_j} = \sum_{j \in Z} N_{ij} f_i \qquad (1.0)$$

$$f_i = \frac{V_j}{\sum_{j \in z} V_j} = \frac{V_j}{V_z} \qquad (2.0)$$

Where, N_{ij} is the atom density of isotope i in region j, f_i is the volume fraction of region j in zone z, isotope i, region j, zone z, V_j is the volume of region j and V_z is the composite volume of all the regions within the zone of interest.The effective density of the nuclides in the moderator region and that of the mixture of the 4 aluminium tie rods and 3 dummy pins of the LEU fuel cell model were obtained by multiplying the region atom density (N_i) by the volume fraction (f_i) obtained from the 200 active fuel rods of LEU fuel materials in the core of NIRR-1. The results of the calculated average homogenized atom density (N_{iz}) for the LEU fuel material is shown in table 1.0 and the average homogenized atom density in the water mix region for the Zircaloy-4 were tabulated in table 2.0.

Table 1.0: The average homogenized atom density (atoms/b-cm) for the LEU fuel cell model

Matl Name	Volume fraction (f_i)	Nuclide ID	N_{ij} (atom/b-cm)	$N_{ij} f_i$(atom cm/b)	N_{iz} (atom cm/b)
Fuel	0.0694	92235	4.7267e-3	3.280e-4	3.280e-4
		92238	1.8963e-2	1.316e-3	1.316e-3
		8016	4.7380e-2	3.288e-3	3.273e-2
		40090	2.165e-2	9.569e-4	1.043e-3
		40091	4.721e-3	2.087e-4	2.275e-4
		40092	7.217e-3	3.189e-4	3.476e-4
		40094	7.314e-3	3.233e-4	3.524e-4
		40096	1.178e-3	5.207e-5	5.676e-5
		50112	5.054e-6	2.234e-7	2.435e-7
		50114	3.549e-6	1.569e-7	1.710e-7
		50116	7.818e-5	3.456e-6	3.767e-6
		50117	4.130e-5	1.825e-6	1.989e-6
		50118	1.302e-4	5.755e-6	6.273e-6
		50119	4.619e-5	2.042e-6	2.226e-6
		50120	1.752e-4	7.744e-6	8.441e-6
		50122	2.490e-5	1.101e-6	1.200e-6
Clad	0.0442	50124	3.113e-5	1.376e-6	1.499e-6

		26054	1.040e-5	4.597e-7	5.011e-7
		26056	1.633e-4	7.218e-6	7.868e-6
		26057	3.772e-6	1.667e-7	1.817e-7
		26058	5.020e-7	2.219e-8	2.419e-8
		24050	3.623e-6	1.601e-7	1.745e-7
		24052	6.987e-5	3.088e-6	3.366e-6
		24053	7.923e-6	3.502e-7	3.817e-7
		24054	1.972e-6	8.716e-8	9.501e-8
		72174	7.186e-9	3.176e-10	3.462-10
		72176	2.362e-7	1.044e-8	1.138e-8
		72177	8.353e-7	3.692e-8	4.024e-8
		72178	1.225e-6	5.415e-8	5.902e-8
		72179	6.117e-7	2.704e-8	2.947e-8
		72180	1.575e-6	6.962e-8	7.589e-8
Moderator	0.8864	1001	6.6434e-2	5.889e-2	5.889e-2
		8016	3.3217e-2	2.944e-2	Combined with fuel
		Zircaloy-4	See table 2.0		

Table 2.0: The Zircaloy-4 average homogenized atom density in the water mix region

Matl Name	Volume fraction (f_i)	Nuclide ID	N_{ij} (N_i^{eff})	$N_{ij} f_i$(atom cm/b)	N_{iz} (atom cm/b)
Mixture of dummy pins and tie rods in the moderator	0.8864	40090	9.7196e-5	8.6155e-5	Combined with homogenized atom density of similar isotopes in the clad
		40091	2.1194e-5	1.8786e-5	
		40092	3.2399e-5	2.8718e-5	
		40094	3.2835e-5	2.9105e-5	
		40096	5.2885e-6	4.6877e-6	
		50112	2.2689e-8	2.0112e-8	
		50114	1.5933e-8	1.4123e-8	
		50116	3.5098e-7	3.1111e-7	
		50117	1.8541e-7	1.6435e-7	
		50118	5.8452e-7	5.1812e-7	
		50119	2.0737e-7	1.8381e-7	
		50120	7.8654e-7	6.9719e-7	
		50122	1.1179e-7	9.9091e-8	
		50124	1.3976e-7	1.2388e-7	
		26054	4.6689e-8	4.1385e-8	
		26056	7.3312e-7	6.4984e-7	
		26057	1.6934e-8	1.5010e-8	
		26058	2.2537e-9	1.9977e-9	
		24050	1.6265e-8	1.4417e-8	

		24052	3.1365e-7	2.7802e-7	
		24053	3.5569e-8	3.1528e-8	
		24054	8.8531e-9	7.8474e-9	
		72174	3.2261e-11	2.8596e-11	
		72176	1.0604e-9	9.3994e-10	
		72177	3.7499e-9	3.3239e-9	
		72178	5.4995e-9	4.8748e-9	
		72179	2.7462e-9	2.4342e-9	
		72180	7.0708e-9	6.2676e-9	

The diffusion theory analysis code (VENTURE-PC) was used to generate values of the effective multiplication factor (k_{eff}) at different depth of insertion of control rod. These data were then used to calculate the reactivity worth (i.e. measure of the deviation of a reactor from criticality) of the control rod (see table 3.0) for the NIRR-1 core model.

Table 3.0: Control rod withdrawal distance, k-effective and reactivity for the LEU fuel cell model for NIRR-1

S/N	Control rod withdrawal length (cm)	Reactivity (k)
1.	0.0	-3.1894951e-3
2.	2.0	-2.7944545e-3
3.	4.0	-2.2597901e-3
4.	6.0	-1.5962802e-3
5.	8.0	-8.2993124e-4
6.	10.0	0.0000
7.	12.0	8.4585152e-4
8.	14.0	1.6586759e-3
9.	16.0	2.3934809e-3
10.	18.0	3.0161519e-3
11.	20.0	3.5094829e-3
12.	22.0	3.8835600e-3
13.	23.0	4.0372254e-3

RESULTS AND DISCURSION

The geometry and dimensions of various components in the proposed LEU core for NIRR-1 were kept identical with that of the present HEU core of the system. This is to ensure that the thermal-hydraulics characteristic of NIRR-1 system remains unaltered. The geometry of the LEU fuel cell model used in this calculation is given in figure 1.0. A plot of the variation in k-infinity as a function of hydrogen to uranium ratio is shown in figure 2.0 while that of reactivity as a function of control rod withdrawal distance for the proposed 19.75% LEU core for the system is shown in figure 3.0.The method used involve no apparent spatial dependence of cross sections in the active fuel region because they were treated as constant in the homogeneous regions, but in the actual system of NIRR-1, there is a spatial dependence of cross sections in the active fuel region because each fuel pin is surrounded with clad and water and there are several configurations of fuel/clad/water within the NIRR-1 core. The results generated for the total number of hydrogen atoms in each of the fuel cell radii is shown in table 4.0 while the data generated for k-infinity as a function of Hydrogen to Uranium (H/U) is shown in table 5.0. A Matlab programming language was used to plot this data as shown in figure 2.0.

Table 4.0: Total number of hydrogen atoms in each of the fuel cell radii

S/N	Fuel cell radii (cm)	Moderator volume (cm^3)	Hydrogen region atom density (atoms/b-cm)	H-atoms (atoms)
1	0.298	0.9523		6.3236e22
2	0.306	1.3014		8.6417e22
3	0.324	2.1208		1.4083e23
4	0.357	3.7446		2.4865e23
5	0.408	6.5637		4.3585e23
6	0.459	9.7587	6.6403e-2	6.4801e23
7	0.510	13.3295		8.8512e23
8	0.561	17.2763		1.1472e24
9	0.6192	22.2394		1.4768e24
10	0.714	31.3717		2.0832e24
11	0.816	42.6481		2.8319e24

Table 5.0: The ratio of Hydrogen to Uranium (H/U) and k-infinity for the LEU (19.75% UO_2) fuel cell model

S/N	Fuel cell radii (cm)	H-atom (atoms)	U-atom (atoms)	H to U ratio	k-infinity
1	0.298	6.324e22		0.799	1.437
2	0.306	8.642e22		1.092	1.451
3	0.324	1.408e23		1.779	1.490
4	0.357	2.487e23		3.143	1.559
5	0.408	4.359e23		5.509	1.638
6	0.459	6.480e23	7.913e22	8.189	1.685
7	0.510	8.851e23		11.185	1.713
8	0.561	1.147e24		14.495	1.726

9	0.6192	1.477e24		18.665	1.727
10	0.714	2.083e24		26.324	1.708
11	0.816	2.832e24		35.789	1.665

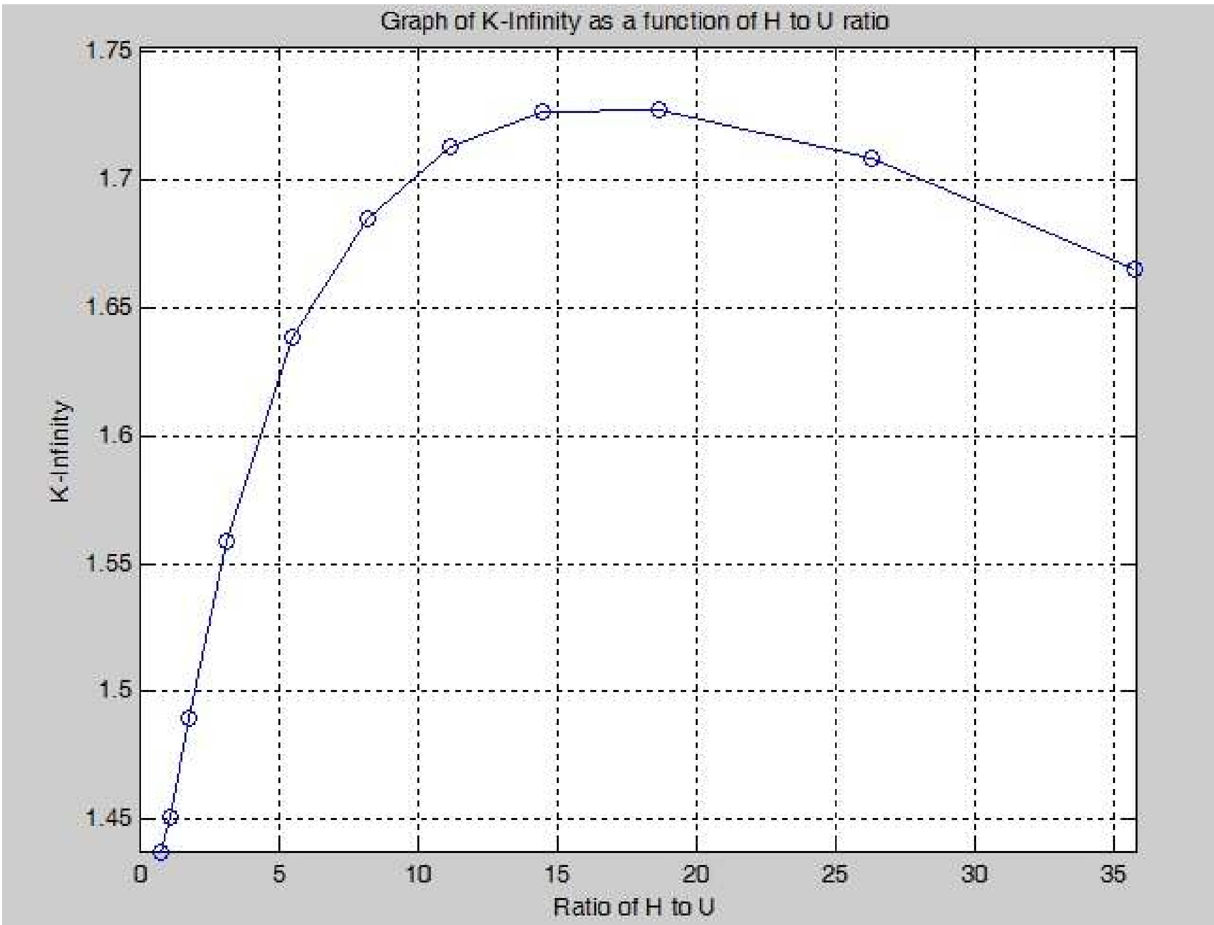


Figure 2.0: Plot of k-infinity as a function of H to U ratio for the LEU (19.75% UO_2) core

The figure above show the result of an increase in the multiplication factor as hydrogen (H) to uranium (U-235) ratio increases up to a value of 1.735 at 18.7 the position of the reference NIRR-1, this value decrease for any further increases in the hydrogen (H) to uranium (U-235) ratio. Due to the vital role of hydrogen in the scattering process in a typical thermal reactor system, the high hydrogen to uranium ratio in the LEU core will result to an increase in the thermal neutron flux and decrease in flux level in the high energy region of the composite flux spectrum of the LEU fuel system. The data generated for the effective multiplication factor (k_{eff})at different level of control rod withdrawal length were used to compute the reactivity worth of the control rod (see table 3.0). This was used to produce the graph of reactivity versus control rod withdrawal length for the proposed 19.75% LEU core for NIRR-1 (figure 3.0).

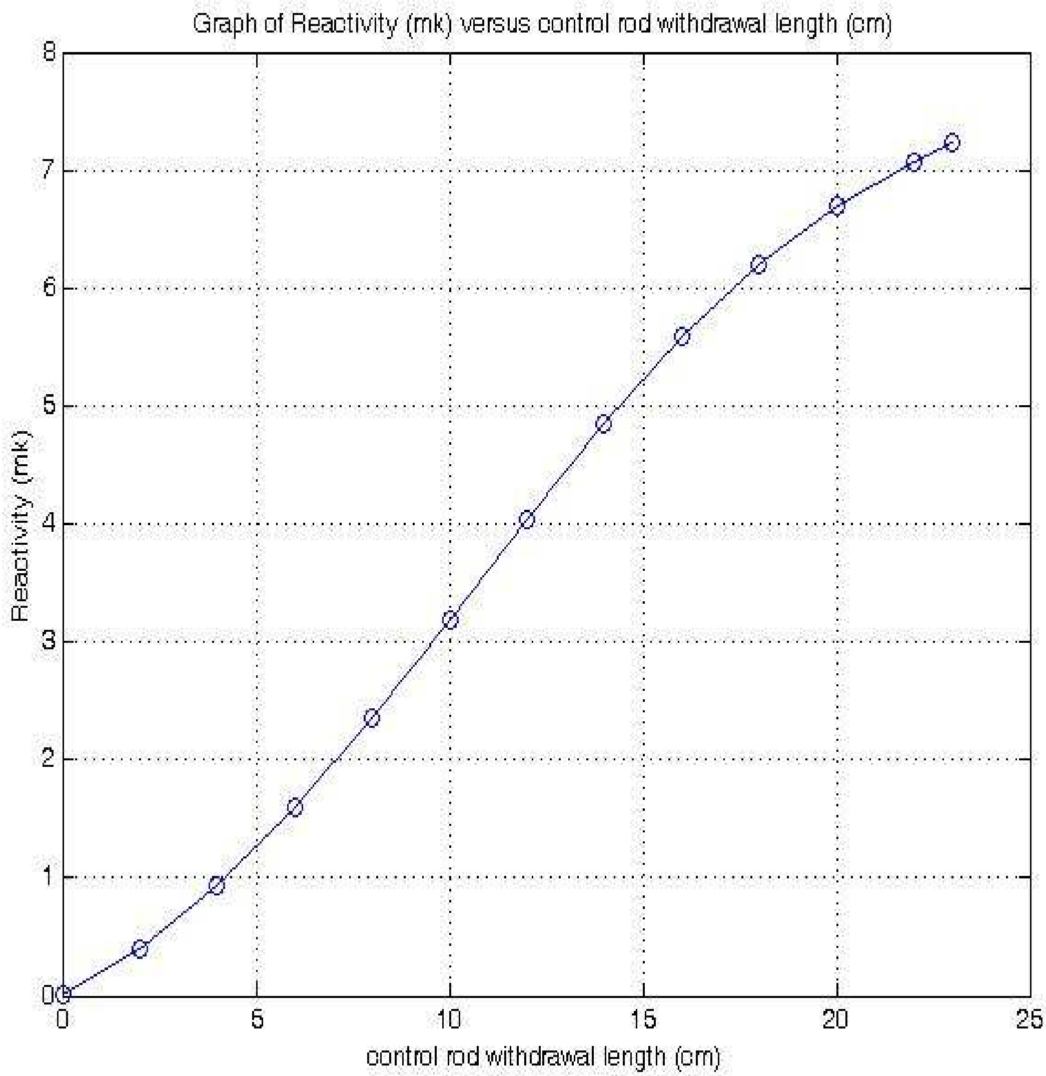


Figure 3.0: Reactivity (mk) versus control rod withdrawal length (cm) of active LEU (19.75% UO_2) fuel region

The clean cold core excess reactivity calculated for the 19.75% LEU core for NIRR-1 was 4.04mk, the shutdown margin was 3.19mk and the corresponding value of k_{eff} was 1.0119634 for the proposed LEU (UO_2) fuel. The thermal neutrons flux level calculated in the 19.75% LEU core for NIRR-1 was $1.24 \times 10^{12}\ ncm^{-2}s^{-1}$. This value is in good agreement with the nominal value of $1.1 \times 10^{12}\ ncm^{-2}s^{-1}$ for the present HEU core of NIRR-1. The thermal neutron flux in the 12.5% UO_2 core from similar calculation was observed to be slightly lower than the thermal neutron flux in the HEU core. This implies that the total number of neutrons that were able to get to the thermal energy is slightly higher in the 19.75% UO_2 core with 200 active fuel pins than in the 12.5% UO_2 core with 347 pins.

CONCLUSION

Comparison of the results obtained from the neutronics calculation using the VENTURE-PC computer code in this work clearly shows that the model is accurate for conducting neutronics analysis for NIRR-1 and useful in the core conversions of Miniature Neutron Source Reactors (MNSRs) to LEU. The proposed 19.75% LEU core is very reactive relative to the core of the HEU system. Therefore the number of regulatory rod in the current HEU core might not be sufficient to reduce the reactivity of the system to a critical level. The results from the calculation performed in this work has shown that 19.75% UO_2 material can also be considered for a more detail analysis for the core conversion studies of NIRR-1 from HEU to LEU. The results of the reactor parameters generated in this work were as expected. The model developed in this work will be very useful in the development of computational models for future analytical studies of the proposed LEU-fuelled NIRR-1 core.

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