COMPUTATIONAL STUDY OF 19.75% UO_2 FUEL FOR THE CORE CONVERSION OF NIGERIA RESEARCH REACTOR-1 (NIRR-1) FROM HEU TO LEU

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ABSTRACT

In this study use has been made of SCALE 6.1 code system and VENTURE-PC code system for the core conversion of Miniature Neutron Source Reactor (MNSR) from Highly Enriched Uranium (HEU) system (90.2% enriched UAl_4 fuel) to Low Enriched Uranium (LEU) system (19.75% enriched UO_2-zircaloy-4 fuel). All other structure materials and dimensions of HEU and LEU cores are the same except the increase in the fuel cell diameters for the proposed LEU core. Results obtained show that the peak power density of 4.310033Watts/cc, maximum neutron density of 6.94535e-6 n/cc, total control rod worth of (723 ± 0.049)pcm, clean cold core excess reactivity of (404 ± 0.009)pcm, k_{eff} of $(1.0119634 \pm 0.0072434)$, shutdown margin of (319 ± 0.1003)pcm and neutron flux profile of $(1.24 \times 10^{12} \pm 0.11)\ ncm^{-2}s^{-1}$ for the potential LEU core are slightly greater than those of the current HEU core. These results also indicate that the LEU core can operate perfectly in natural convection mode which shows the accuracy of the model and precision of the transport code system used.

Keywords: NIRR-1, MNSR, LEU, HEU, SCALE 6.1 code, VENTURE-PC code, peak power density, Neutronics, control rod worth, excess reactivity, k-effective, shutdown margin, and neutron fluxes.

1. INTRODUCTION

The present HEU NIRR-1 has a tank-in-pool structural configuration and a nominal thermal power rating of 31.1kW (Jonah *et al*., 2005). The current core of the reactor is a 230 x 230mm square cylinder and fueled by U-Al_4 enriched to 90.2%. It is in Al-alloy cladding whose thickness is 0.6mm. Light water is used as moderator and coolant while metallic beryllium is used as reflector. It has a total number of 347 fuel pins, three Al dummy pins and four tie rods. The length of the fuel element is 248mm; the active length being 230mm with 9mm Al-alloy plug at each end. The diameter of the fuel meat is 4.3mm, fuel meat volume density is 3.456g/cm^3 and the U-235 loading in each fuel element is about 2.88g. The control rod is made up of a cadmium (Cd) absorber of 266mm long and 3.9mm in diameter with stainless steel of 0.5mm thickness as the cladding material and overall length of 0.450m. With a built-in clean cold core excess reactivity of 3.77mk measured during the on-site zero-power and criticality experiments, the reactor can operate for a maximum of 4 hours 30 minute at full power, mainly due to the large negative temperature feedback effects (FSAR, 2005). Under these conditions, with the same fuel loading, the reactor can run for over ten years with a burn-up of less than 1%. In this work we focused upon the computational study of Nigeria Research Reactor-1 (NIRR-1) core conversion using uranium dioxide (UO_2) as fuel, the most common ceramic fuel (Sunghwan, 2013). Some of the benefits of using UO_2 as reactor fuel include chemical inertness, compatibility with potential cladding materials such as stainless steel and zircaloy, dimensional stability under irradiation, very high melting point and 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 allows 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 NIRR-1 core from HEU to LEU fuel (Jonah *et al*., 2009; Salawu, 2012; Jonah *et al*., 2012; Ibrahim *et al*., 2013). The results of these studies based on 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. 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 in addition to 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. A recent version of the diffusion theory code called VENTURE-PC (White, 2012) were used in this work to perform the neutronics analysis with a recent version of SCALE code system (SCALE 6.1) (Salawu, 2012) 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. The effective multiplication factor for the system, excess reactivity, and reactivity worth of the control material, shim worth and power distribution at different locations within the Nigeria Research Reactor-1 (NIRR-1) core were determined in this work. 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. The information available to us from literature has shown that a research has not been conducted on NIRR-1 using 19.75% enriched UO_2 material as the fuel with VENTURE-PC as the computational tools.

2. MATERIALS AND METHODS

The NIRR-1 fuel cell is enriched to 90.2% U-235 with each fuel pins containing 2.88g of U-235 while UAl_4-Al is the fuel material in the active fuel region and has a density of 3.456g/cm^3. The geometry of the active fuel material for the current HEU NIRR-1 is shown in table 1. Uranium dioxide (19.75% UO_2) fuel of volume density 10.6g/cm^3 is the proposed material selected to perform the core conversion study for NIRR-1 with zircaloy-4 as the cladding material. Zircaloy-4 has a volume density of 6.56g/cm^3 with a natural zirconium of 98.23 weight percent (w/o) (Salawu, 2012). All other structure materials and dimensions of HEU and the proposed LEU cores are the same except a decrease in the fuel cell radius caused by a reduction in the number of fuel pins in the core of NIRR-1. It is proposed that approximately 200 active fuel rods of LEU fuel materials (19.75% UO_2) be installed in the proposed core for NIRR-1. In addition, it is also suggested that three (3) aluminum dummy pins and four (4) aluminum tie rods in the HEU core be replaced by zircaloy-4 material. The proposed dimensions are: 23.0cm for fuel rod length, 0.43cm for fuel rod diameter and 1.632cm for fuel cell diameter, as illustrated in figure 1. In this figure, the active LEU fuel region is indicated in red color, where each fuel rod contain 6.162g of U-235.

Table 1: The geometry representation of HEU NIRR-1 fuel element

Fuel pin dimensions	
Active fuel diameter	0.43cm
Active fuel length	23.0cm
Total pin length	24.8cm
Cladding thickness	0.06cm
Fuel cell diameter	1.2384cm
Homogenized fuel radius	11.55cm
Guide tube radius	0.60cm

The average homogenized atom density (N_{iz}) is calculated by multiplying the region atom density (N_{ij}) by the region volume fraction (f_i) for the zones in the NIRR-1 fuel cell (equations 1 and 2).

$$N_{iz} = \frac{\sum_{j \in Z} N_{ij} V_j}{\sum_{j \in Z} V_j} = \sum_{j \in Z} N_{ij} f_i \tag{1}$$

$$f_i = \frac{Volume\ of\ each\ zones}{Total\ Volume} = \frac{V_j}{\sum_{j \in z} V_j} = \frac{V_j}{V_z} \tag{2}$$

Where, N_{ij} is the atom density of isotope i in region j, f_i is the volume fraction (VF) of region j in 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.

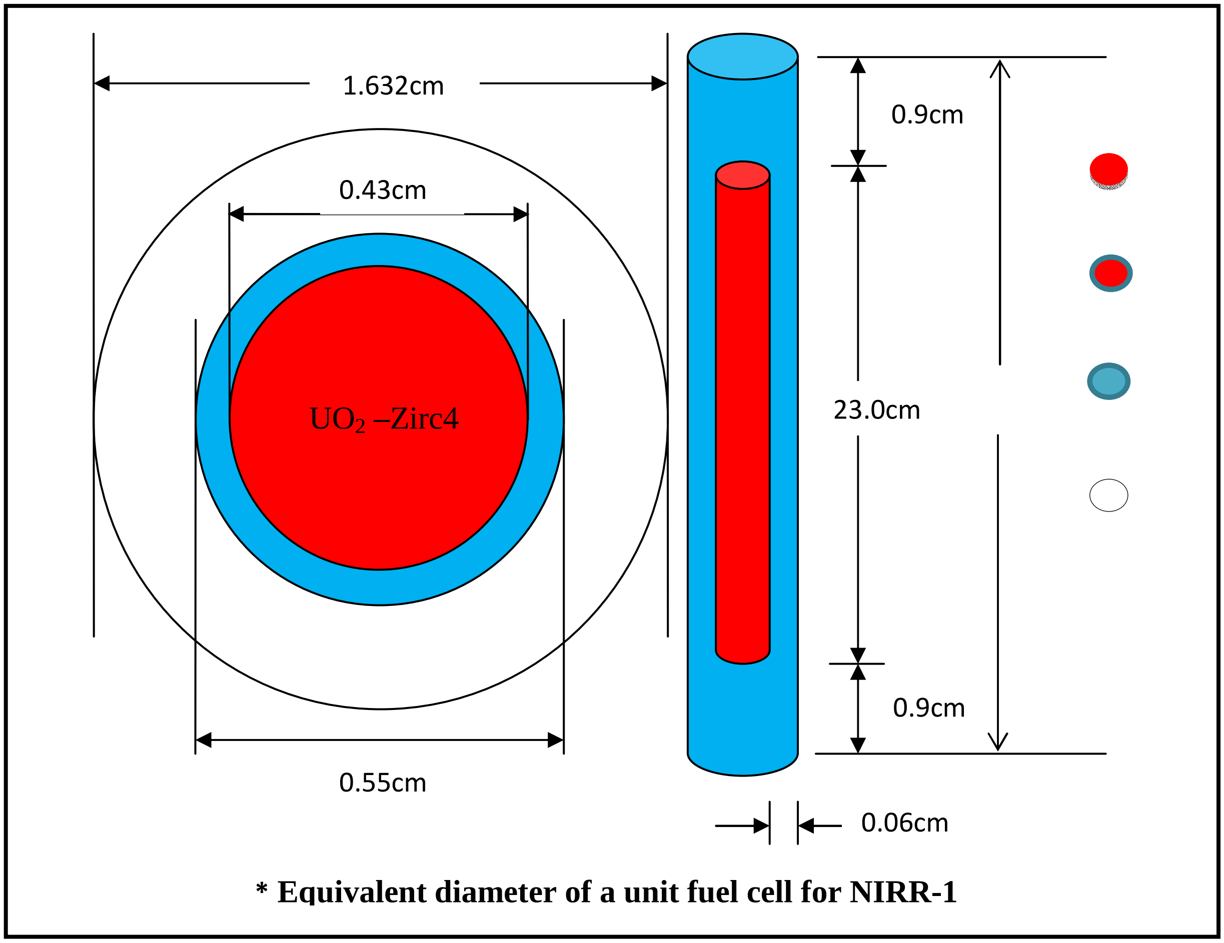
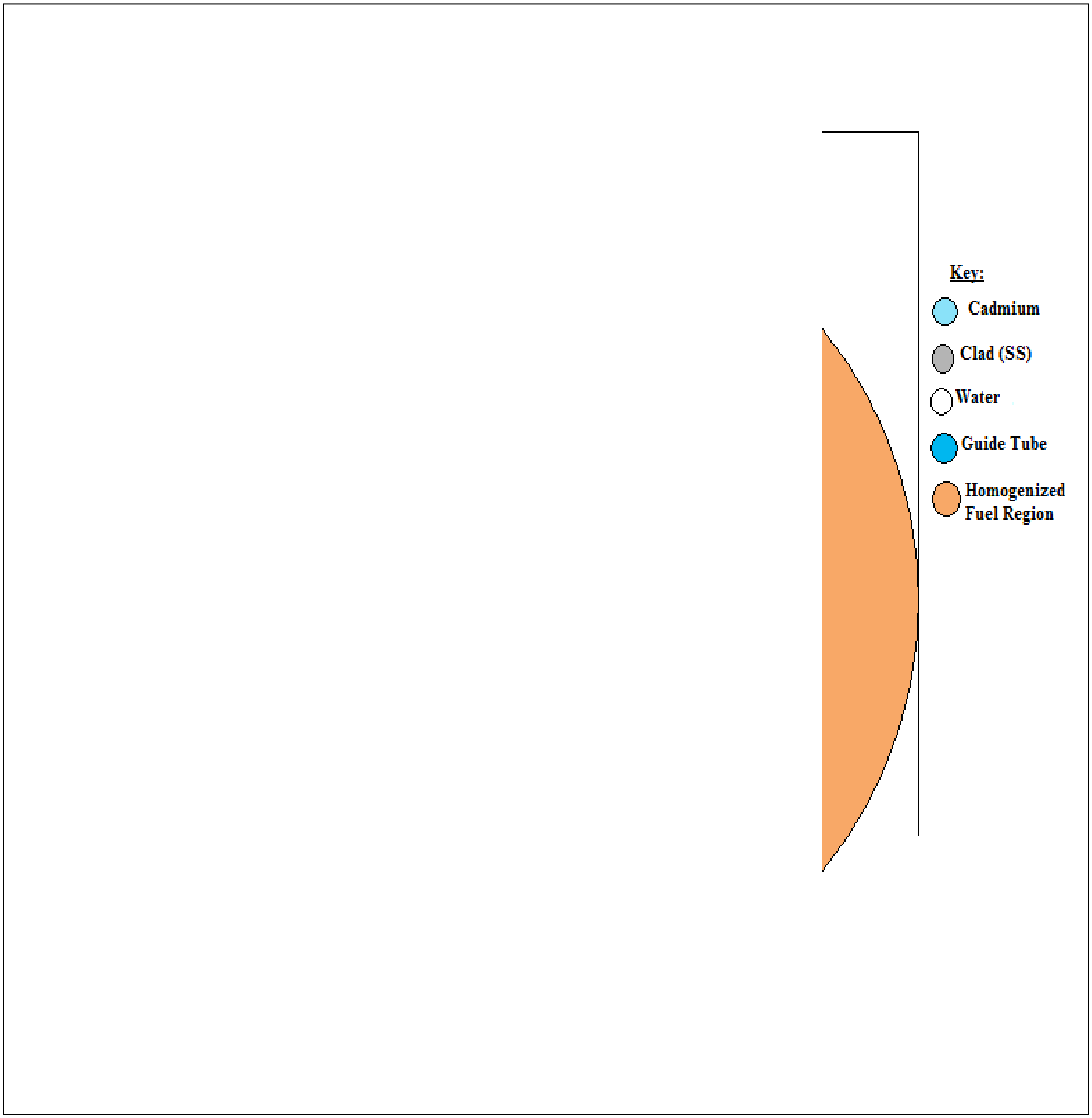


Figure 1: The height and diameter of the active fuel cell and fuel rod models for the potential LEU core for NIRR-1.

Figure 2 show the description of the control model for NIRR-1 core while the 1-D full core geometry with zone dimensions and descriptions for NIRR-1 is illustrated in figure 3. This figure give the detail Y and X dimensions of the physical core model for NIRR-1 core model with the control rod fully inserted into the core. The case with control rod fully withdrawn from the core is similar to this figure except that the poison material in the control region is replaced with water.



ntrol model for NIRR-1 core.

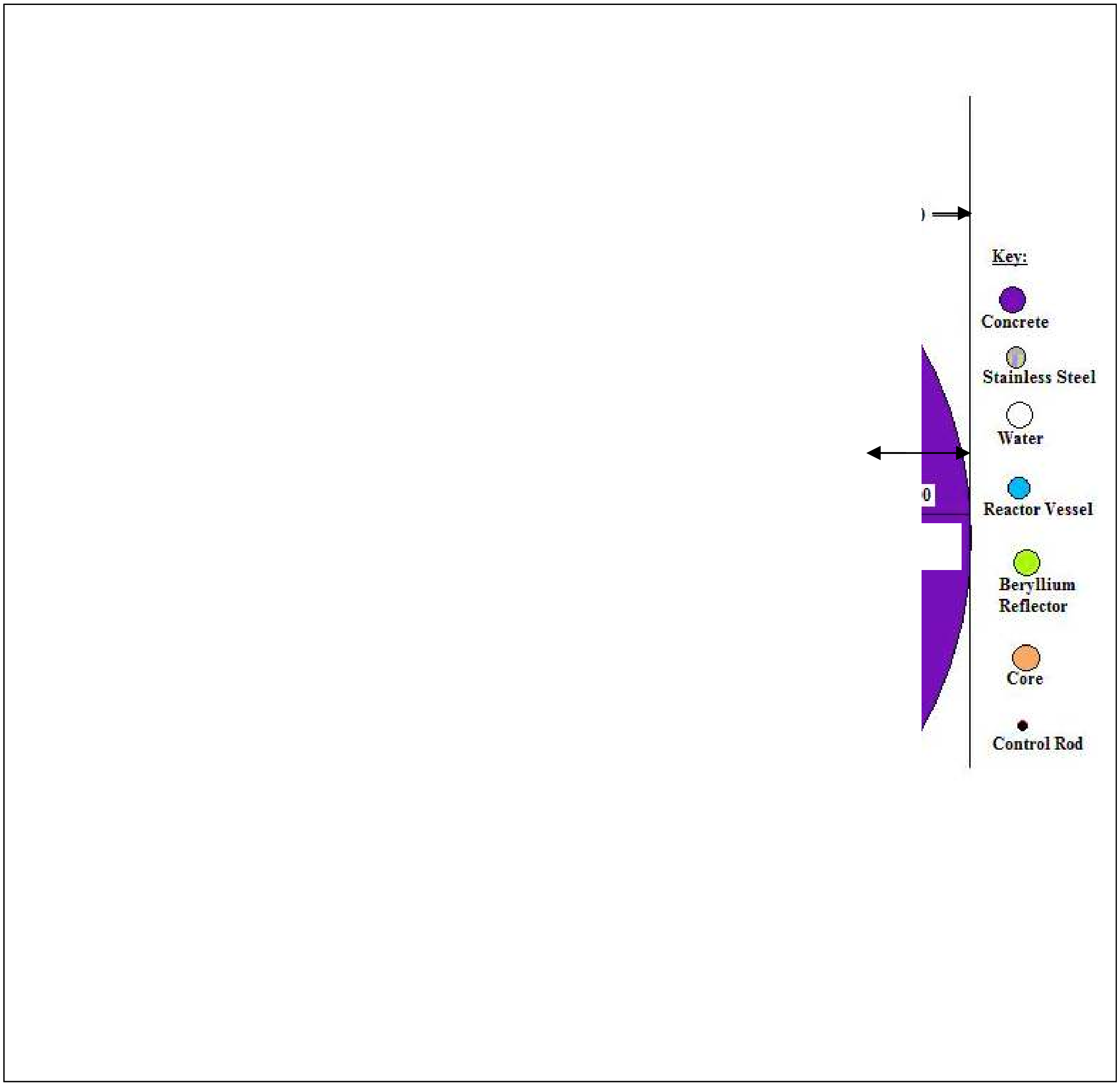


Figure 3: The geometry and dimensions of various components of NIRR-1 core.

The effective density of the nuclides in the moderator region and that of the mixture of the 4 aluminum 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). This procedure was carried out for the proposed assembly of two hundred 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 are presented in table 2 and the average homogenized atom density in the water mix region for the zircaloy-4 in table 3. The corresponding values for the HEU fuel cell model are presented in table 4 and table 5.

Table 2: The average homogenized atom density (atoms cm/b) for the LEU fuel cell model

Material 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
Clad	0.0442	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
		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	3.273e-2
		zircaloy-4	See table 3		

Table 3: The zircaloy-4 average homogenized atom density in the water mix region for the LEU fuel cell model

Material 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	1.043e-3
		40091	2.1194e-5	1.8786e-5	2.275e-4
		40092	3.2399e-5	2.8718e-5	3.476e-4
		40094	3.2835e-5	2.9105e-5	3.524e-4
		40096	5.2885e-6	4.6877e-6	5.676e-5
		50112	2.2689e-8	2.0112e-8	2.435e-7
		50114	1.5933e-8	1.4123e-8	1.710e-7
		50116	3.5098e-7	3.1111e-7	3.767e-6
		50117	1.8541e-7	1.6435e-7	1.989e-6
		50118	5.8452e-7	5.1812e-7	6.273e-6
		50119	2.0737e-7	1.8381e-7	2.226e-6
		50120	7.8654e-7	6.9719e-7	8.441e-6
		50122	1.1179e-7	9.9091e-8	1.200e-6
		50124	1.3976e-7	1.2388e-7	1.499e-6
		26054	4.6689e-8	4.1385e-8	5.011e-7
		26056	7.3312e-7	6.4984e-7	7.868e-6
		26057	1.6934e-8	1.5010e-8	1.817e-7
		26058	2.2537e-9	1.9977e-9	2.419e-8
		24050	1.6265e-8	1.4417e-8	1.745e-7
		24052	3.1365e-7	2.7802e-7	3.366e-6
		24053	3.5569e-8	3.1528e-8	3.817e-7
		24054	8.8531e-9	7.8474e-9	9.501e-8
		72174	3.2261e-11	2.8596e-11	3.462-10
		72176	1.0604e-9	9.3994e-10	1.138e-8
		72177	3.7499e-9	3.3239e-9	4.024e-8
		72178	5.4995e-9	4.8748e-9	5.902e-8
		72179	2.7462e-9	2.4342e-9	2.947e-8
		72180	7.0708e-9	6.2676e-9	7.589e-8

Table 4: The average homogenized atom density for the HEU core model (with control rod in)

Material Name	Homogenized Material Name	Volume fraction (f_i)	Nuclide ID	N_{ij} (atom/b-cm)	$N_{ij}f_i$ (atom cm/b)	N_{iz} (atom cm/b)
Cadmium	Control	0.07671	48106	6.116e-5	4.692e-6	4.908e-5
			48108	4.355e-5	3.341e-6	3.495e-5
			48110	6.112e-4	4.689e-5	4.904e-4
			48111	6.263e-4	4.804e-5	5.025e-4
			48112	1.181e-3	9.059e-5	9.474e-4
			48113	5.979e-4	4.586e-5	4.798e-4
			48114	1.406e-3	1.079e-4	1.128e-3
			48116	3.665e-4	2.811e-5	2.941e-4
Clad (Stainless Steel)			14028	9.593e-5	7.359e-6	1.278e-4
			14029	4.872e-6	3.737e-7	6.491e-6
			14030	3.216e-6	2.467e-7	4.284e-6
			24050	4.638e-5	3.558e-6	6.179e-5
			24052	8.945e-4	6.862e-5	1.192e-3
			24053	1.014e-4	7.778e-6	1.351e-4
			24054	2.525e-5	1.937e-6	3.363e-5
			25055	1.064e-4	8.162e-6	1.417e-4
			26054	2.098e-4	1.609e-5	2.791e-4
			26056	3.288e-3	2.522e-4	4.379e-3
			26057	7.595e-5	5.826e-6	1.012e-4
			26058	1.011e-5	7.755e-7	1.346e-5
			28058	3.219e-4	2.469e-5	4.288e-4
			28060	1.240e-4	9.512e-6	1.652e-4
			28061	5.390e-6	4.135e-7	7.181e-6
			28062	1.719e-5	1.319e-6	2.289e-5
			28064	4.377e-6	3.358e-7	5.831e-6
Water		0.80274	1001	2.639e-2	2.118e-2	1.711e-1
			8016	1.314e-2	1.055e-2	8.552e-2
Guide tube			13027	2.636e-2	2.116e-2	7.093e-2
Fuel	Fuel	0.12055	92235	2.663e-4	3.210e-5	3.210e-5
			92238	2.857e-5	3.444e-6	3.444e-6
			13027	1.159e-2	1.397e-3	Combined with control
Water		0.80274	1001	5.330e-2	4.279e-2	
			8016	2.665e-2	2.139e-2	
Beryllium	Reflector		4009	1.236e-1	9.922e-2	9.922e-2
Water	Water		1001	6.674e-2	5.357e-2	Combined with control
			8016	3.337e-2	2.679e-2	
Al Vessels	Vessel		13027	6.026e-2	4.837e-2	
Water	Water		1001	6.674e-2	5.357e-2	
			8016	3.337e-2	2.679e-2	

Table 4 represent the x-direction of NIRR-1 core configuration with a single control rod in the central core region. It include all the major components of NIRR-1 core such as the control rod, fuel rod, beryllium reflector, water and reactor vessel. The control region represent a homogenized mixture of cadmium poison material with a stainless steel clad and a zone which surrounds the control material containing water plus the aluminum made control rod guide tube.

Table 5: The average homogenized atom density for the HEU core model (with control rod out)

Material Name	Homogenized Material Name	Volume fraction (f_i)	Nuclide ID	N_{ij} (atom/b-cm)	$N_{ij}f_i$ (atom cm/b)	N_{iz} (atom cm/b)
Water	Water	0.80274	1001	3.754e-2	3.013e-2	1.801e-1
			8016	1.877e-2	1.507e-2	9.004e-2
Guide tube			13027	2.636e-2	2.116e-2	7.093e-2
Fuel	Fuel	0.12055	92235	2.663e-4	3.210e-5	3.210e-5
			92238	2.857e-5	3.444e-6	3.444e-6
			13027	1.159e-2	1.397e-3	Combined with control
Water		0.80274	1001	5.330e-2	4.279e-2	
			8016	2.665e-2	2.139e-2	
Beryllium	Reflector		4009	1.236e-1	9.922e-2	9.922e-2
Water	Water		1001	6.674e-2	5.357e-2	Combined with control
			8016	3.337e-2	2.679e-2	
Al Vessels	Vessel		13027	6.026e-2	4.837e-2	
Water	Water		1001	6.674e-2	5.357e-2	
			8016	3.337e-2	2.679e-2	

The data generated from the tables were combine in a single library for use in the 1-D full core computational models for NIRR-1. The ANISN formatted output library from this 1-D calculation passes through a number of processing before it was used in the VENTURE-PC code system. Three input card modules were identified in this work as the basic modules (control module, input processor module and special processor module), both necessary and available to simulate the core physics of the NIRR-1 using the VENTURE-PC code system. In this modules we select the basic particle transport methodology, indicates the tallies to be printed, defines the geometry of the reactor, and assigns nuclides to their specific geometric zones. The VENTURE-PC code system was then used to compute group fluxes profiles, power density distributions, effective multiplication factor (k_{eff}) at different depth of insertion of control rod and criticality information within the Nigeria Research Reactor-1 core region. 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 6) for the LEU NIRR-1 core model and table 7 show similar results for the

HEU NIRR-1 core. The SCALE 6.1 code system serves as a mean to generate the cross section libraries, perform the neutron flux calculations, as well as provide k-infinity from the criticality calculation for the proposed 19.75% enriched UO_2 material for core conversion studies of NIRR-1 core. Three major different cross section libraries were generated using the SCALE 6.1 code system. The version of SCALE package generated consists of about 89 different computational modules as well as the current nuclear data libraries and problem dependent processing tools for neutronics calculations and other reactor physics calculations (SCALE, 2011; Salawu, 2012). About nine different modules of SCALE code system was used to perform the cross section libraries development.

Table 6: Control rod withdrawal distance, k-effective and reactivity for the LEU fuel cell model for NIRR-1

S/N	Depth of control rod insertion (cm)	k-effective	Reactivity (mk)
1	0.0	1.0080631±7.6627e-2	0.0049999±9.9713e-5
2	2.0	1.0084626±7.6657e-2	0.3950460±7.8784e-3
3	4.0	1.0090033±7.6698e-2	0.9297100±1.8541e-2
4	6.0	1.0096743±7.6749e-2	1.5932200±3.1774e-2
5	8.0	1.0104493±7.6808e-2	2.3595690±4.7057e-2
6	10.0	1.0112886±7.6872e-2	3.1895000±6.3608e-2
7	12.0	1.0121440±7.6937e-2	4.0353515±8.0477e-2
8	14.0	1.0129660±7.6999e-2	4.8481758±9.6687e-2
9	16.0	1.0137091±7.7056e-2	5.5829809±1.1134e-1
10	18.0	1.0143388±7.7104e-2	6.2056518±1.2376e-1
11	20.0	1.0148377±7.7142e-2	6.6989828±1.3359e-1
12	22.0	1.0152160±7.7170e-2	7.0730600±1.4106e-1
13	23.0	1.0153714±7.7182e-2	7.2267253±1.4412e-1

Table 7: Control rod withdrawal distance and reactivity for the HEU fuel cell model for NIRR-1

S/N	Control rod withdrawal distance (cm)	Reactivity (mk)
1	0.0	0.000
2	2.0	$0.455 \pm 9.104e\text{-}3$
3	4.0	$1.045 \pm 2.091e\text{-}2$
4	6.0	$1.636 \pm 3.273e\text{-}2$
5	8.0	$2.364 \pm 4.729e\text{-}2$
6	10.0	$3.182 \pm 6.366e\text{-}2$
7	12.0	$4.000 \pm 8.003e\text{-}2$
8	14.0	$4.773 \pm 9.549e\text{-}2$
9	16.0	$5.545 \pm 1.109e\text{-}1$
10	18.0	$6.136 \pm 1.228e\text{-}1$
11	20.0	$6.636 \pm 1.328e\text{-}1$
12	22.0	$7.000 \pm 1.401e\text{-}1$
13	23.0	$7.209 \pm 1.442e\text{-}1$

3. RESULTS AND DISCUSSION

The structure materials and dimensions of various components in the proposed LEU core for NIRR-1 have been kept identical with those 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 illustrated in figures 1, 2 and 3. A plot of the variation in k-infinity as a function of hydrogen to uranium ratio is presented in figures 4 and 5 for the LEU and HEU cores, while reactivity as a function of control rod withdrawal distance for the proposed 19.75% LEU core and 90.2% HEU core for the NIRR-1 system is illustrated in figure 6. The method used involve no apparent spatial dependence of cross sections in the active fuel region, because it is treated as constant in the homogeneous regions. However, 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 8, the data generated for k-infinity as a function of hydrogen to uranium (H/U) is illustrated in table 9 while Table 10 show similar results of k-infinity versus H/U for the HEU core for the active fuel.

Table 8: Total number of hydrogen atoms in each of the LEU 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	6.6403e-2	6.3236e22±6.2940e-3
2	0.306	1.3014		8.6417e22±8.6013e-3
3	0.324	2.1208		1.4083e23±1.4017e-2
4	0.357	3.7446		2.4865e23±2.4749e-2
5	0.408	6.5637		4.3585e23±4.3381e-2
6	0.459	9.7587		6.4801e23±6.4498e-2
7	0.510	13.3295		8.8512e23±8.8098e-2
8	0.561	17.2763		1.1472e24±1.1418e-1
9	0.6192	22.2394		1.4768e24±1.4699e-1
10	0.714	31.3717		2.0832e24±2.0735e-1
11	0.816	42.6481		2.8319e24±2.8187e-1

Table 9: 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±6.294e-3	7.913e22	0.799±6.293e-3	1.437±8.073e-2
2	0.306	8.642e22±8.601e-3		1.092±8.601e-3	1.451±8.152e-2
3	0.324	1.408e23±1.402e-2		1.779±1.401e-2	1.490±8.371e-2
4	0.357	2.487e23±2.475e-2		3.143±2.475e-2	1.559±8.759e-2
5	0.408	4.359e23±4.338e-2		5.509±4.339e-2	1.638±9.203e-2
6	0.459	6.480e23±6.449e-2		8.189±6.449e-2	1.685±9.467e-2
7	0.510	8.851e23±8.809e-2		11.185±8.809e-2	1.713±9.624e-2
8	0.561	1.147e24±1.142e-1		14.495±1.142e-1	1.726±9.697e-2
9	0.6192	1.477e24±1.469e-1		18.665±1.470e-1	1.727±9.703e-2
10	0.714	2.083e24±2.074e-1		26.324±2.073e-1	1.708±9.650e-2
11	0.816	2.832e24±2.819e-1		35.789±2.819e-1	1.665±9.354e-2

Table 10: The ratio of hydrogen to uranium (H/U) and k-infinity for the HEU (90.2% UAl_4) fuel cell model

S/N	H to U ratio	k-infinity
1	4.545±4.191e-3	1.705±8.746e-2
2	6.818±6.287e-3	1.747±8.962e-2
3	18.182±1.677e-2	1.797±9.218e-2
4	32.955±3.039e-2	1.830±9.388e-2
5	54.545±5.029e-2	1.840±9.439e-2
6	79.545±7.335e-2	1.834±9.408e-2
7	109.091±1.006e-1	1.812±9.295e-2
8	140.091±1.292e-1	1.783±9.146e-2
9	177.273±1.635e-1	1.750±8.977e-2
10	211.364±1.949e-1	1.715±8.798e-2
11	250.0±2.305e-1	1.681±8.623e-2

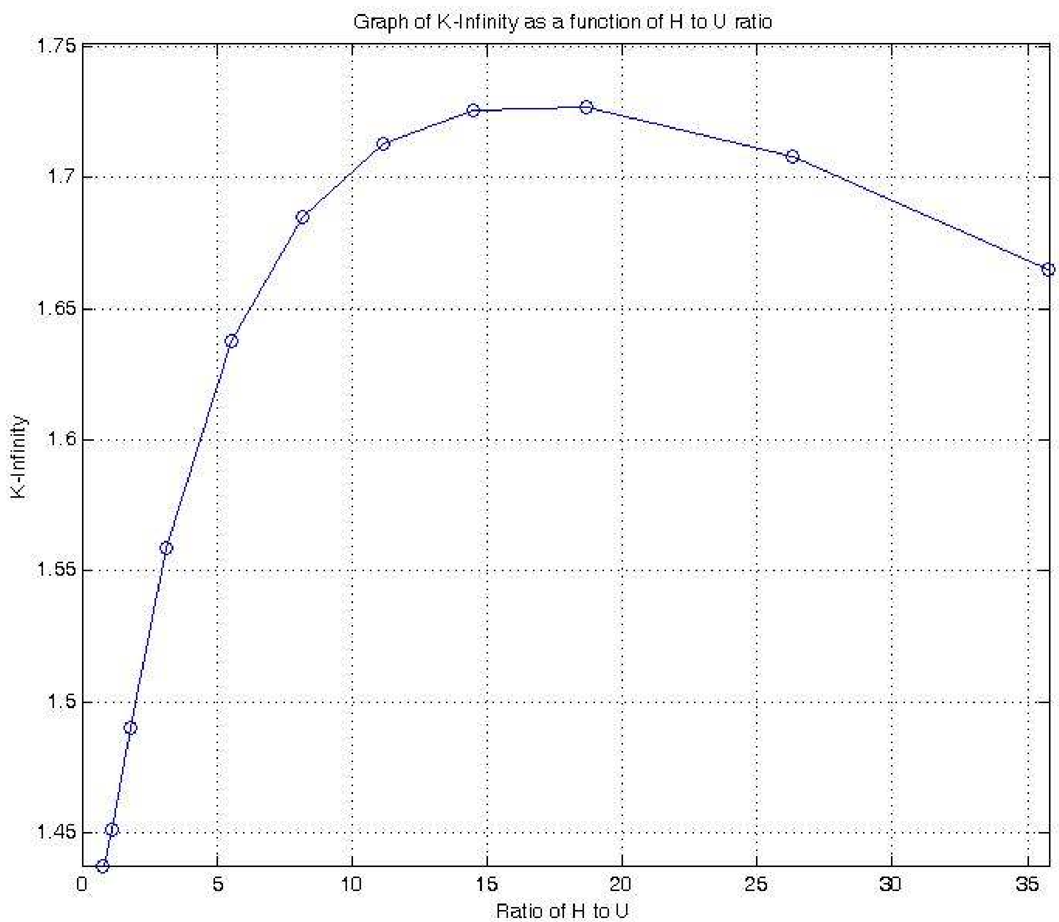


Figure 4: Plot of k-infinity as a function of H to U ratio for the LEU (19.75% UO_2) core

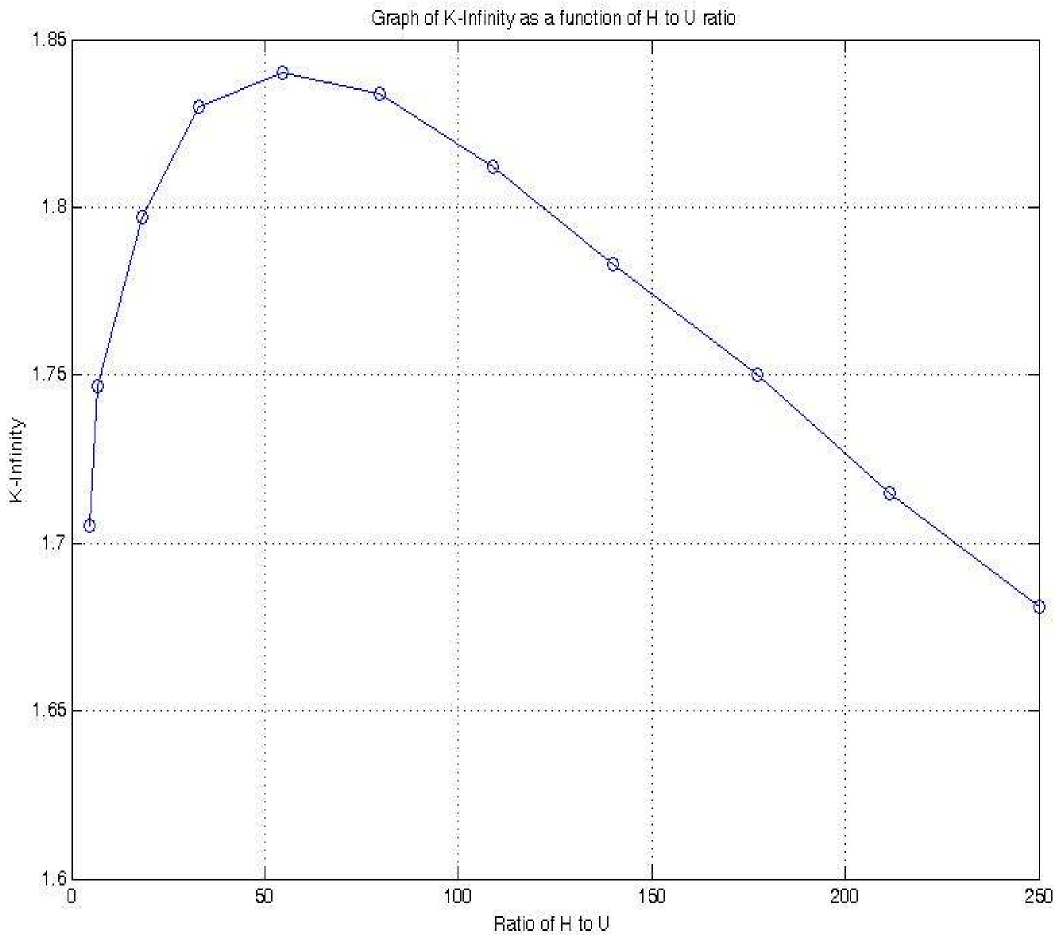


Figure 5: Plot of k-infinity as a function of H to U ratio for the HEU (90.2% UAl_4) core

For the LEU core 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 while for the HEU core the reference position is at 177.24. The ratio of hydrogen to uranium for the proposed LEU core is ten times less than that of the HEU core, this as a result of the decreases in the multiplication factor as hydrogen to uranium ratio increases. 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 factors (k_{eff}) at different level of control rod withdrawal length for the LEU core were used to compute the reactivity worth of the control rod as shown in table 6. This was used to produce the graph of reactivity versus control rod withdrawal length for the proposed 19.75% LEU core for NIRR-1 as illustrated in figure 6. This figure also show the corresponding plot for the 90.2% HEU reactivity against control rod withdrawal distance from bottom of the active fuel.

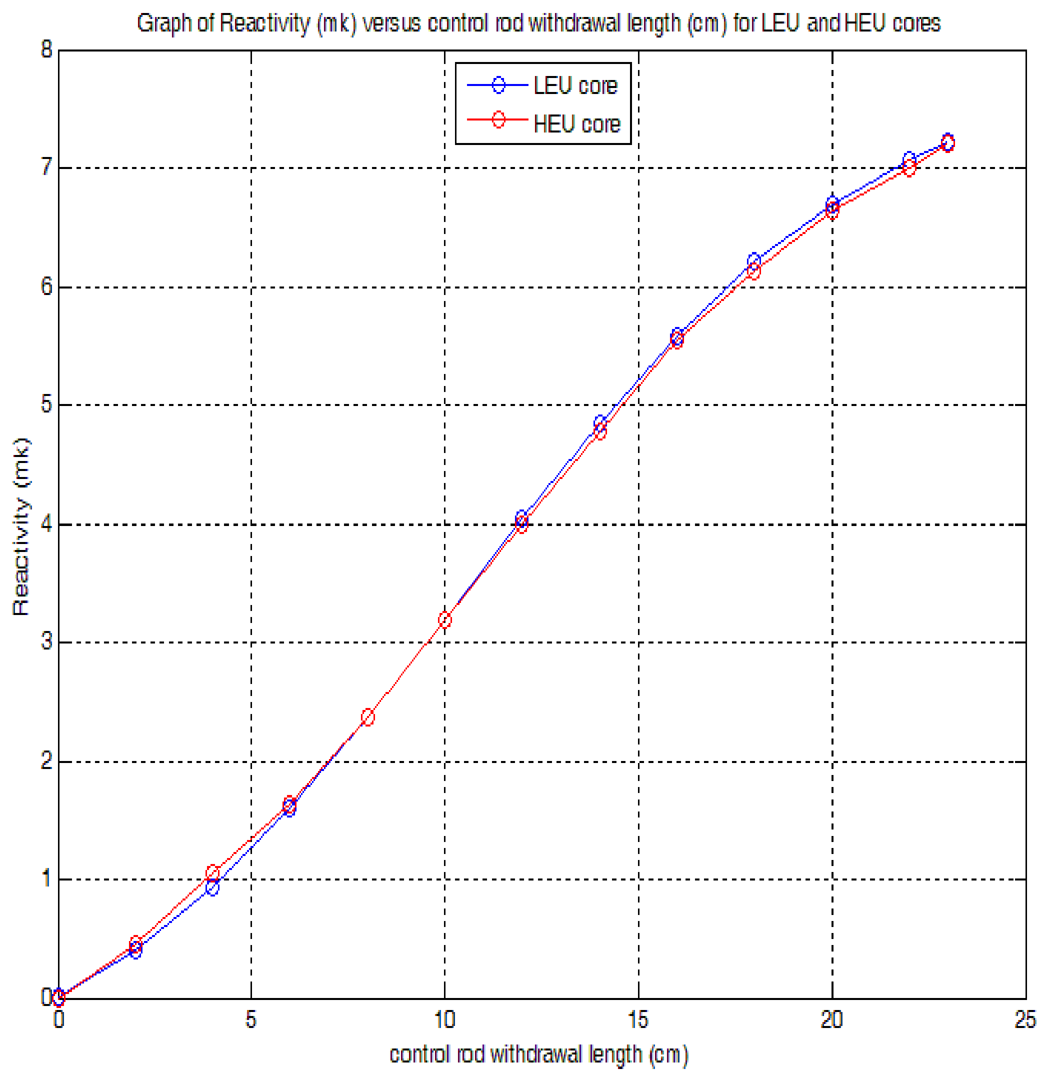


Figure 6: Reactivity (mk) versus control rod withdrawal length (cm) of the active fuel region

The clean cold core excess reactivity calculated for the 19.75% LEU core for NIRR-1 was (404 ± 0.009)pcm, the shutdown margin was (319 ± 0.1003)pcm, peak power density of 4.310033Watts/cc, and maximum neutron density of 6.94535e-6 n/cc and the corresponding value of k_{eff} was $(1.0119634 \pm 0.0072434)$ for the proposed LEU (UO_2) fuel. The thermal neutrons flux level calculated in the 19.75% LEU core for NIRR-1 was $(1.24 \times 10^{12} \pm 0.11)\, ncm^{-2}s^{-1}$. 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 and 90.2% UAl_4 with 347 pins.

4. CONCLUSION

The group constants generated by VENTURE-PC code system were used in the SCALE 6.1 code system which supports more complex geometry descriptions for this model of NIRR-1 and the results obtained has been compared with the corresponding experimental data. This comparison clearly shows that the model is accurate for conducting neutronics analysis for NIRR-1. The proposed 19.75% LEU core is very reactive relative to the core of the present 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 have clearly shown that 19.75% enriched UO_2-zircaloy-4 fuel will be very useful in the core conversion MNSR to LEU core.

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