Journal of Nuclear and Particle Physics 2017, 7(2): 27-35
DOI: 10.5923/j.jnpp.20170702.02

Consummation of 19.75% UO_2 Fuel Material in the Core of Nigeria Miniature Neutron Source Reactor (MNSR)

Samson D. O.*, **Buba A. D. A.**

Department of Physics, University of Abuja, Abuja, Nigeria

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 and VENTURE-PC codes. 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 1.24×10^{12} $\text{ncm}^{-2}\text{s}^{-1}$ with stability of $\pm 1\%$, vertical and horizontal variation less than three percent $(< 3\%)$ for the proposed LEU core are a bit more than that of the present HEU core. The model predictions of this parameters for NIRR-1 system in this study are perfectly in agreement with experimental results as well as the results from similar computations using different nuclear tools.

Keywords MNSR, NIRR-1, LEU, HEU, SCALE 6.1 code, VENTURE-PC code, Neutronics, Control rod worth, Excess reactivity, K-effective, Shutdown margin, Uranium dioxide

1. 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, the most common ceramic fuel (Sunghwan, 2013). Some of the benefits of using UO_2 as 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 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 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 (Jonah *et al*., 2007; Balogun, 2003; Salawu, 2012), 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 to about ten percent (10%) 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 work 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. This will give room for more moderators in the core and could increase the number of hydrogen available to thermalize the neutron in the potential LEU core for Nigeria Research Reactor-1 (NIRR-1). Hence the hydrogen to uranium ratio will increase with a corresponding increase in the thermal neutron flux. The current HEU NIRR-1 core has a tank-in-pool structural configuration, a nominal thermal power rating of 31.1kW, 230 x 230mm square cylinder, fueled by U-Al_4 enriched to 90.2% and was originally designed by China with a diffusion theory codes (HAMMER and EXTERMINATOR-2) (CERT, 2004; Jonah *et al*., 2005). 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

* Corresponding author:
saamdoftt82@gmail.com (Samson D. O.)
Published online at http://journal.sapub.org/jnpp

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 (Figures 1 and 2). With a built-in clean cold core excess reactivity of 3.77mk (377pcm) 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; Jonah *et al*., 2006). These design computations (such as neutron multiplication factors, neutron fluxes, fuel burn-up, fuel cycle length, isotopic inventory, cross section libraries etc.) was 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 (Butler and Albright, 2007; SCALE, 2011; Salawu, 2012). A licensed user of the codes performed the actual calculations and generated the output data used to perform this analysis. In this work the effective multiplication factor for the system, excess reactivity, 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 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. Achieving the core conversion of NIRR-1 to operate on the LEU fuel would be a helpful step forward in this international effort to reduce and eventually eliminate the civil use of HEU material in research reactors.

2. Materials and Methods

VENTURE-PC code has be used primarily in this work to compute group fluxes profiles and criticality information within the Nigeria Research Reactor-1 core region, while SCALE 6.1 serves as a mean to generate the N-group 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. The N-group cross section library used were copied from the SCALE run output file and stored in three different WordPad files. Uranium dioxide (19.75% UO_2) fuel of volume density 10.6g/cm^3 is the potential LEU fuel material chosen 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^3 with a natural zirconium of 98.23 weight percent (w/o) (Salawu, 2012; Samson, 2015). The geometry and dimensions of the proposed LEU fuel material will be invariant in some aspect but the fuel cell radius will vary from that of the present NIRR-1 fuel material, which will leads to a decrease in the number of active fuel pins to about 200. The three aluminium dummy pins and four 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. The uranium in the active fuel region of the LEU fuel material is enriched to 19.75% U-235 with each fuel rod containing 6.162g of U-235.

The volume fraction (f_i) for the zones in the fuel cell is determined by first calculating the volume of each zones and then divide each values obtained by the total volume of the equivalent fuel cell.

$$Total\ volume\ of\ the\ core\ per\ fuel\ pins = \frac{Total\ volume\ of\ the\ fuel\ regions\ in\ the\ core}{Number\ of\ fuel\ elements\ in\ the\ NIRR-1} \tag{1}$$

We then obtain the region atom density of the nuclide within the moderator (H_2O) region of the fuel cell by using the mass density of 0.9982g/cm^3 of water as available in SCALE manual (SCALE, 2011).

$$N_{H_2O} = \frac{\rho_{H_2O} \times N_A}{M_{H_2O}} \tag{2}$$

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 (Equations 3 and 4).

$$f_i = \frac{Volume\ of\ each\ zones}{Total\ Volume} \tag{3}$$

$$N_{iz} = \frac{\sum_{j \in Z} N_{ij} V_j}{\sum_{j \in Z} V_j} = \sum_{j \in Z} N_{ij} f_i \tag{4}$$

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, 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 (N_i^{eff}) of the nuclides in the moderator region and that of the mixture of the four aluminium tie rods and three 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 for NIRR-1 core is shown in Table 1, the results of the atom densities of various isotopes in zircaloy-4 were tabulated in Table 2, while the average homogenized atom density in the water mix region for the zircaloy-4 were illustrated in Table 3.

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 of the control rod as shown in Table 4 for the LEU NIRR-1 core model, while the corresponding HEU core were tabulated in Table 5.

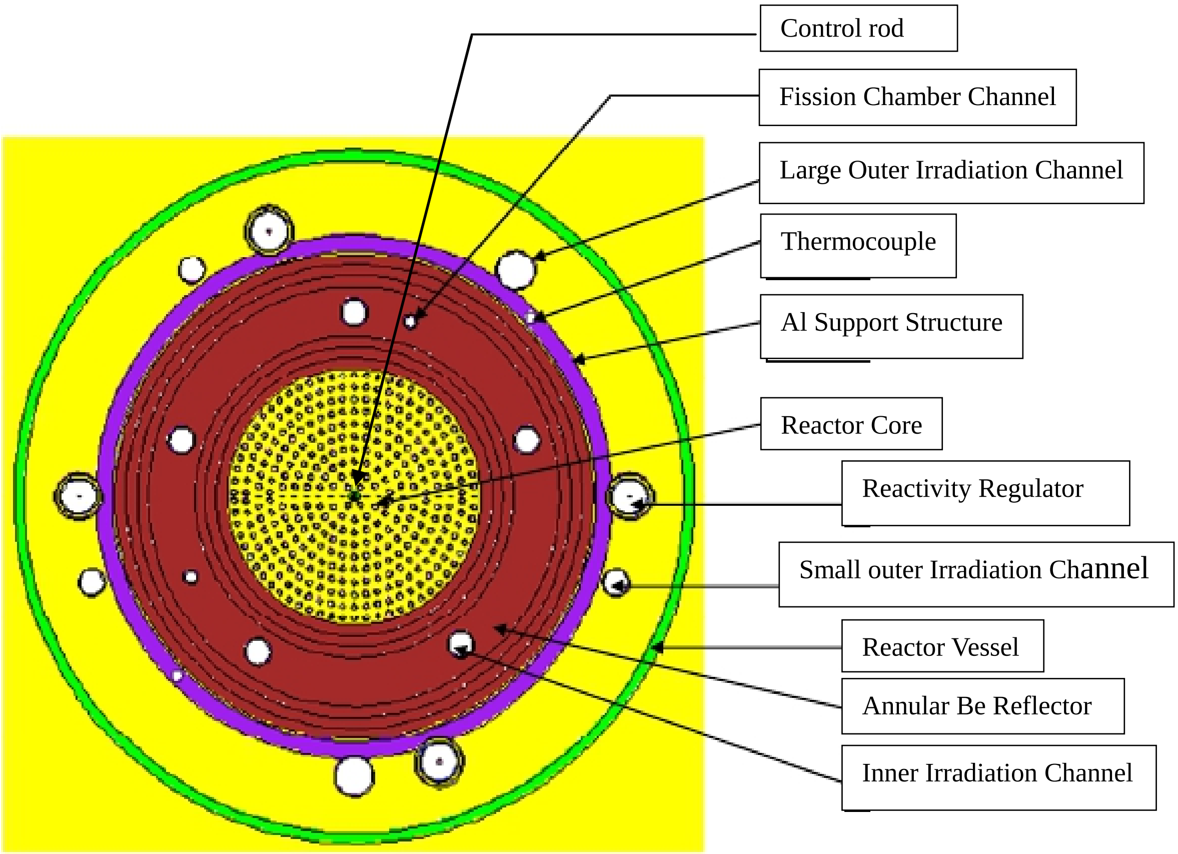

Figure 1. Front View of NIRR-1 core configuration

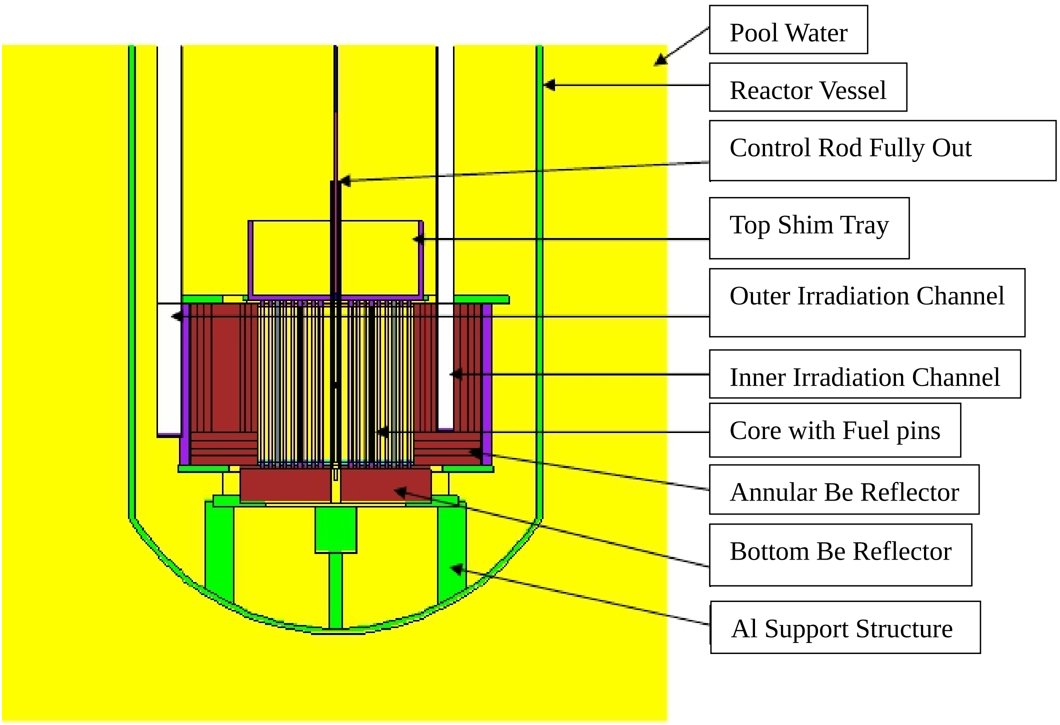

Figure 2. A close Side View of NIRR-1 core configuration

Table 1. Nuclides average homogenized atom density (atoms cm/b) for the LEU fuel cell model for NIRR-1 core

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
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	Combined with fuel
		zircaloy-4	See Table 3		

Table 2. The atom density (atoms/b-cm) of various nuclides present in the Zircaloy-4

Matl name	Nuclide name	Nuclide ID	M_i	w_i (w/o)	N_i(at./b-cm)	γ_i (a/o)	N_{mix}
Zircaloy-4	Zirconium	40000	91.224	98.23	4.2538e-2	1.0110	4.2075e-2
	Tin	50000	118.710	1.45	4.8253e-4	0.8973	5.3776e-4
	Iron	26000	55.845	0.21	1.4855e-4	0.8350	1.7790e-4
	Chromium	24000	51.996	0.10	7.5976e-5	9.1114	8.3386e-6
	Hafnium	72000	178.490	0.01	2.2133e-6	0.4928	4.4913e-6

Table 3. 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	

Table 4. The 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

Table 5. The control rod withdrawal distance and reactivity for the HEU fuel cell model for NIRR-1

S/N	Control rod withdrawal length (cm)	Reactivity (mk)
1.	0.0	0.000
2.	2.0	0.455e-3
3.	4.0	1.045e-2
4.	6.0	1.636e-2
5.	8.0	2.364e-2
6.	10.0	3.182e-2
7.	12.0	4.000e-2
8.	14.0	4.773e-2
9.	16.0	5.545e-1
10.	18.0	6.136e-1
11.	20.0	6.636e-1
12.	22.0	7.000e-1
13.	23.0	7.209e-1

3. Results and Discussion

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 front and side view of NIRR-1 core configuration is illustrated in Figures (1 and 2). A plot of the variation in k-infinity as a function of hydrogen to uranium ratio is shown in Figure 3, 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 4. 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 6, while the data generated for k-infinity as a function of hydrogen to uranium (H/U) is shown in Table 7. A Matlab programming language was used to plot this data as shown in Figure 3 (Gerald, 2001; Matlab R2013b).

Table 6. 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	6.6403e-2	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		6.4801e23
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

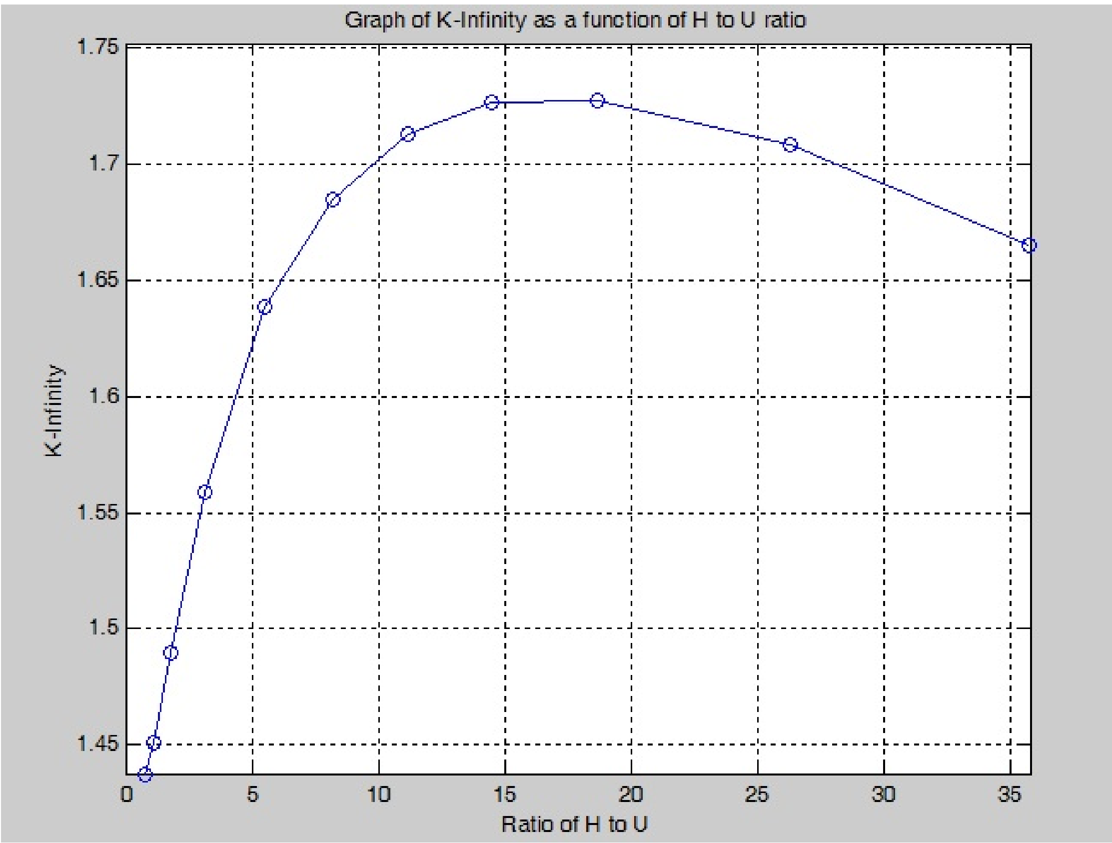

Figure 3. Plot of k-infinity as a function of H to U ratio for the LEU (19.75% UO_2) core

Table 7. 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	7.913e22	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		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

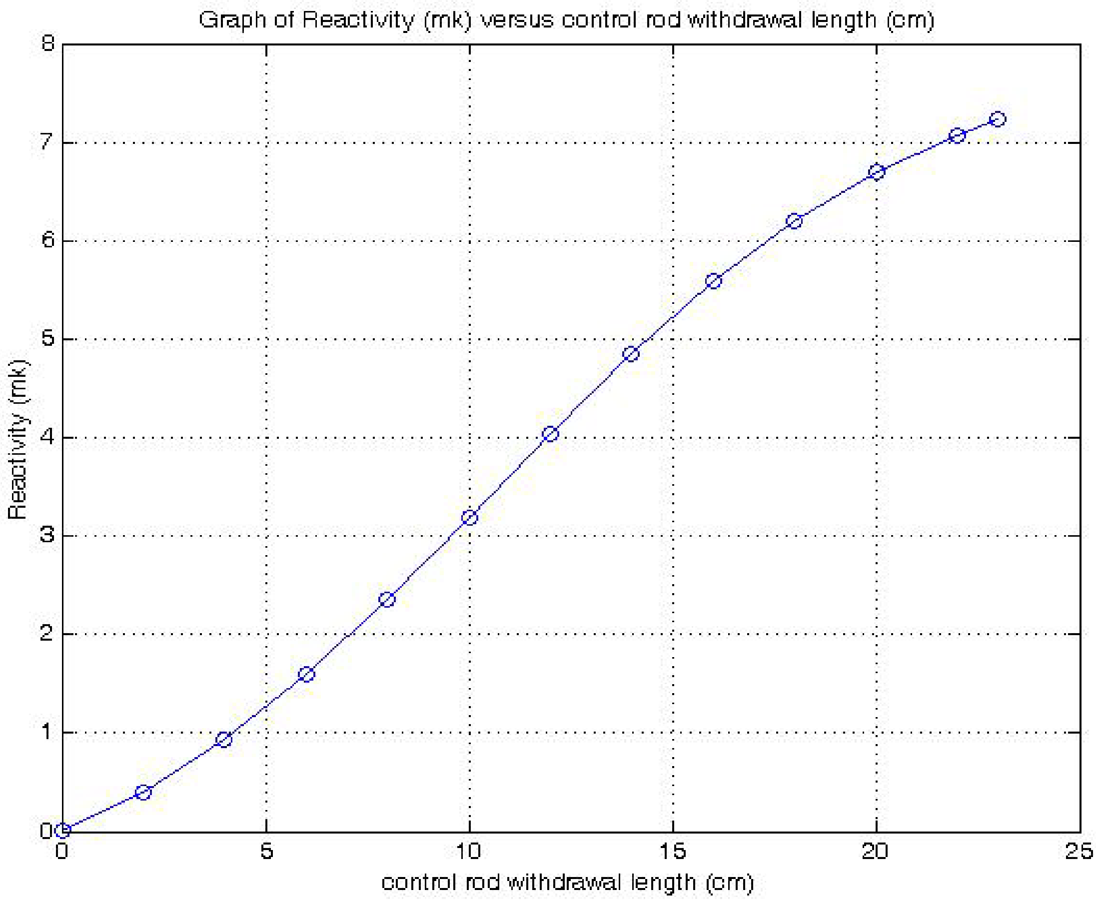

Figure 4. Reactivity (mk) versus control rod withdrawal length (cm) of active LEU (19.75% UO_2) fuel region

The Figure 3 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 factors (k_{eff}) at different level of control rod withdrawal length were used to compute the reactivity worth of the control rod (see Table 3). 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).

The clean cold core excess reactivity calculated for the 19.75% LEU core for NIRR-1 was (4.04 ± 0.0001)mk, the shutdown margin was (3.19 ± 0.001)mk and the corresponding value of k_{eff} was (1.0119634 ± 0.0001) for the proposed LEU (UO_2) fuel. These values are in good agreement with the design specification of 3.5mk – 4.0mk for MNSR as reported by (Chengzen, 1993) and HEU (UAl_4) of 4.0mk value reported by (Balogun, 2003) using

CITATION code and 4.7mk by (Jonah et al., 2009) using MCNP5 code version 1.40, shutdown margin of 2.87mk by (Jonah et al., 2009) using MCNP code and 2.18mk by (Azande et al., 2011) using a simplified scheme of WIMS and CITATION codes, k_{eff} value of (1.00472 ± 0.0002) by (Jonah et al., 2009; Ibrahim et al., 2013) using MCNP code. The slight difference between the measured and computed clean cold core excess reactivity shows that during measurements the shim plates were stacked over each other and submerged in water. There is always a thin film of water remaining between the adjacent shim plates. During calculation no water was taken into account while modelling the beryllium shim plates in the shim tray. As a result of this the measured reactivity for the HEU core is higher than the computed reactivity for the proposed LEU core. The shutdown margin for the LEU (UO_2) is large in comparison with the recommended safety limit (i.e. ≥ 2.5mk). The k_{eff} value obtained show that the potential LEU (UO_2) core is said to be critical since for fission chain reaction to be just continuing at a steady state; k_{eff} has to be approximately unity. Expectations are that when the control rod is fully withdrawn, the super-criticality should be attained while sub-criticality should be attained when the control rod is fully inserted. It is therefore, desirable that criticality should be attained about midway between full withdrawal and full insertion of the control rod so as to allow for easy control and regulation of reactivity (see Table 4). The thermal neutrons flux level calculated in the 19.75% LEU core for NIRR-1 was $1.24 \times 10^{12}\ ncm^{-2}s^{-1}$ with stability of $\pm 1\%$, vertical and horizontal variation less than three percent $(< 3\%)$. This value is in good agreement with the nominal value of $1.1 \times 10^{12}\ ncm^{-2}s^{-1}$, stability of $\pm 1\%$, vertical and horizontal variation $< 3\%$ 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 due to hardening of the neutron spectrum. This implies that the total number of neutrons that were able to get to the thermal energy in the thermalization processes 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.

4. Conclusions

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 and the model developed in this work will be of great important in the development of computational models for future analytical studies of the proposed LEU-fuelled NIRR-1 core.

ACKNOWLEDGEMENTS

The authors wish to appreciate the staff of Physics department, Nigeria Defence Academy, Kaduna, Nigeria for their assistance during the course of this research work.

REFERENCES

[1] Azande, T.S., Balogun, G.I., Ajuji, A.S., Ahmed, Y.A., Jonah, S.A., (2011): Determination of LEU Core Safety related parameters of NIRR-1 using WIMS and CITATION Codes. Continental J. Applied Science 6 (1), 71-76, ISSN: 1597-9928, http://www.wiloludjournal.com.

[2] Balogun, G.I., (2003): Automating some Analysis and design calculation of Miniature Neutron Source Reactors at CERT. Annals of Nuclear Energy, 30, pp. 81-92.

[3] Butler, C., Albright, D., (2007): Core Physics and Kinetics Calculation for the Fissioning Plasma Core Reactor, Paper Presented at the Institute for Scientific Research, Inc., Fairmont, West Virginia, available at; http://www2.sti.nasa.gov.

[4] CERT., (2004): Center for Energy Research and Training, Final Safety Analysis Report of Nigeria Research Reactor-1 (NIRR-1), CERT Technical Report-CERT/NIRR-1/FSAR-01, CERT, Abu, Zaria, Nigeria.

[5] Chengzen, G., (1993): Experiment of adding top beryllium shims for MNSR, CIAE Technical Report.

[6] FSAR., (2005), Final Safety Analysis Report of Nigeria Research Reactor-1 (NIRR-1), CERT Technical Report CERT/NIRR-1/FSAR-01.

[7] Gerald, W.R., (2001): MATLAB Programming. Department of Mechanical Engineering, Portland State University, Version 0.97, www.prenhall.com/recktenwald.

[8] Ibrahim, Y.V., Odoi, H.C., Thomas, J.W., and Jonah, S.A., (2013): Design Options of Control Rod for Low Enriched Uranium Fueled NIRR-1 Using Monte Carlo N-Particle Code. Journal of Nuclear Energy Science and Power Generation Technology, Vol. 2, Pp. 1-3, http://dx.doi.org/10.4172/2325-9809.1000104.

[9] Jonah, S.A., Balogun, G.I., Umar, I.M., Mayaki, M.C., (2005): Neutron Spectrum Parameters in Irradiation Channels of the Nigeria Research Reactor-1 (NIRR-1) for the K_0- NAA standardization. Journal of Radiational. Nucl. Chem., vol. 266, pp. 83-88.

[10] Jonah, S.A., Umar, I.M., Oladipo, M.O.A., Balogun, G.I., Adeyemo, D.J., (2006): Standardization of NIRR-1 Irradiation and Counting Facilities for instrumentation Neutron Activation Analysis. Journal of Applied Rad. and Isotopes, vol. 64, pp. 818-822.

[11] Jonah, S.A., Liaw, J.R., Matos, J.E., (2007): Monte Carlo simulation of core physics parameters of the Nigeria Research

Reactor-1 (NIRR-1). Annals of Nuclear Energy, vol. 54, Pp. 953–957.

[12] Jonah, S.A., Ibikunle, K., Li, Y., (2009): A Feasibility Study of LEU Enrichment Uranium fuels for MNSR Conversion using MNCP. Annals of Nuclear Energy 36, 1285-1286.

[13] Jonah, S.A., Ibrahim, Y.V., Ajuji, A.S., Onimisi, M.Y., (2012): The Impact of HEU to LEU Conversion of Commercial MNSR; Determination of Neutron Spectrum Parameters in Irradiation Channels of NIRR-1 using MNCP code. Annals of Nuclear Energy 39, 15-17.

[14] Lyons, M.F., Boyle, R.F., Davies, J.H., Hazel, V.E., Rowland, T.C., (1972): UO_2 Properties Affecting Performance, Nuclear Engineering and Design, Published by Elsevier B.V., San Jose, California 95125, U.S.A. doi:10.1016/0029-5493(72)90072-6.

[15] MATLAB R2013b: MATLAB Programming Language Setup.

[16] Salawu, A., (2012): Computational Modeling and Simulation to Generate Reactor Physics Parameters for Core Conversion of the Miniature Neutron Source Reactor from HEU to LEU. Ph.D thesis, Ahmadu Bello University, Zaria, Nigeria.

[17] Samson, D.O., (2015): Neutronics Study of NIRR-1 Fuelled UO_2 Enriched to 19.75% using VENTURE-PC and SCALE 6.1 Codes. M.Sc thesis, Nigerian Defence Academy University, Kaduna, Nigeria.

[18] SCALE, (2011): A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design. ORNL/TM-2005/39, Version 6.1, Oak Ridge National Laboratory, Oak Ridge, Tennessee, June 2011. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785.

[19] Sunghwan, Y., (2013): UO2-SIC Composite Reactor Fuel with enhanced thermal and mechanical properties prepared by Spark Plasma Sintering. Florida, USA.