Original Research Articles

NEUTRONICS STUDY OF NIRR-1 FUELLED WITH 19.75% UO_2MATERIAL USING VENTURE-PC AND SCALE 6.1 CODES

ABSTRACT

A comprehensive neutronics analysis using VENTURE-PC and SCALE 6.1 computer codes system has been performed for the core conversion study of Nigeria Research Reactor-1 (NIRR-1).The computed reactor core physics parameters include: group neutron fluxes profiles, power density distributions and neutron flux distributions. The number of active fuel pins used for this analysis is approximate 200 pins, which show that the fuel pins have been reduced to about 58% when compared with the present Highly Enriched Uranium (HEU) fuel (UAl_4-Al) of 347 pins. These reductions in the number of fuel pins has given room for more moderators in the core and hence increase the number of hydrogen available to thermalize the neutron in the potential 19.75% UO_2 fuelled core for Nigeria Research Reactor-1 (NIRR-1). The diffusion theory based calculated values of thermal flux profiles for the vertical as well as for the horizontal radial directions has been found to agree well with similar calculations using different nuclear analysis tools. The results obtained will qualify uranium dioxide (UO_2) fuel as the potential material for future Low Enriched Uranium (LEU) core conversion of Nigeria Miniature Neutron Source Reactors (MNSRs).

Keywords: Neutronics, VENTURE-PC code, SCALE 6.1 code, NIRR-1, MNSR,parameters, neutron, fluxes, core, power density, LEU fuel, HEU fuel, model, Zircaloy-4, cross section.

INTRODUCTION

Uranium dioxide(UO_2), a ceramic fuel is presently the most commonly used nuclear fuel for both research and power reactors. Uranium dioxide material is the fuel of choice for most reactorsdue to its high melting point (2800^{o}C), high neutron utilization, excellent irradiation stability, exceptional corrosion resistance in conventional coolants, good fission product retention,no phase change up to the melting point, compatibility with cladding (Zircaloy and stainless steel), ease of fabrication, and high specific power and power per unit length of fuel pin (Sundaran and Mannan, 1989; Sunghwan, 2013). UO_2 has a fabrication density of 10.3g/cm^3 and offers a relatively high uranium loading of about 9.1g/cm^3. The use of highly enriched uranium (HEU) material in research reactors has leads to a set of inevitable proliferation risks. HEU fuel is the last remaining civilian application of a direct-use of material, which is easily utilize in a nuclear explosive device (Franz et al., 2005). NIRR-1 is one of the world MNSR reactors that still uses HEU as fuel and therefore presents a potential nuclear proliferation threats to global security. NIRR-1 core was initially designed by China Institute of Atomic Energy (CIAE) with computer codes HAMMER and EXTERMINATOR-2 and 90.2% enriched UAl_4-Al as fuel (FSAR, 2005). HAMMER and EXTERMINATOR are the first set of codes used to solve reactor physics problem using diffusion theory method followed by WIMS and CITATION codes (Balogun, 2003). To model and analyze the NIRR-1 core a 2-D (r,z) neutronics models was developed using the recent version of diffusion theory codes (VENTURE-PC and SCALE 6.1).VENTURE-PC consists of VENTURE module (which solves reactor physics problems based on the multigroupneutronics finite different diffusion theory) and EXPOSURE computational module plus several other processing modules. It is a recent version of diffusion theory based deterministic code (White and Tooker, 1999) that can be used like CITATION (Balogun, 2003) to perform neutronics analysis for the NIRR-1 system.The VENTURE-PC code is used to generate the group fluxes profiles, power density distributions and criticality information within the Nigeria Research Reactor-1 core.It is also used to compute the effective multiplication factor,the neutron flux distributions at different location within the coreof the proposed 19.75% UO_2 material for core conversion studies of NIRR-1 from HEU to LEU.SCALE code provides a “plug-and-play” framework with 89 computational modules, including three deterministic and three Monte Carlo radiation transport solvers that are selected based on the desired solution (Salawu, 2012). The SCALE 6.1 code is used to generate the cross section libraries used in this work.Achieving the core conversion of NIRR-1 to operate on the LEU fuel would be a useful step toward, an effort to reduce and eventually eliminate the civil use of HEU material in research reactors. This work attempt to design a system with a higher performance in the area such as high neutron fluxes, higher fuel burn-ups, longer fuel cycle length and other reactor design parameters as compared with a similar study that uses 12.5% UO_2 material as the fuel.

MATERIALS AND METHOD

Reactor physics codes are typically used to support the performance of the core as well as to provide results used in the system thermal hydraulic codes for accident analysis. The VENTURE-PC code is used in this work to compute the group fluxes profiles, power density distributions and criticality information within the Nigeria Research Reactor-1.The SCALE 6.1 code is usedto generate the cross section libraries, perform the multigroup neutron flux calculations, as well as provide k-infinity from the one dimensional criticality calculations for the proposed 19.75% UO_2 material for core conversion studies of NIRR-1. The core ofNIRR-1 contains fuel cage, fuel pins (which serves as the main energy source), control rod with a guide tube (which is provided to facilitate the movement of the control rod), three dummy rods and four tie rods. The specific reactor physics calculation for the generation of the cross section libraries in this reactor required a collection of three cell models for the primary core elements, one each for the fuel rod, the control rod and materials outside the core regions.The top and bottom plate of the fuel cage is replaced with zirconium material in the LEU core. About nine different modules of SCALE code system is used to perform the cross section libraries development and these modules include AJAX, WORKER, CRAWDAD, BONAMI, CENTRM, PMC, PALEALE, XSDRNPM and WAX modules (Salawu, 2012).The volume fraction (f_i) for the zones in the fuel cell were determined by first calculating the volume of each zones and then divide each values by the total volume of the equivalent fuel cell. Table 1.0 shows the calculated region volume fraction for the zones in the NIRR-1 fuel cell model.

Table 1.0: The region volume fraction (f_i) for the zones in the NIRR-1 fuel cell model (with 200 active fuel pins)

S/N	Region	Volume (cm^2)	Tot. vol. (cm^2)	Vol. fraction (f_i)	Tot. vol. fraction
1.	Fuel	0.1452	2.0918	0.0694	1.0000
2.	Clad	0.0924		0.0442	
3.	Moderator	1.8542		0.8864	

The molecular weight of uranium and atom densities of various isotopes in Zircaloy-4 in the proposed LEU fuel material was calculated from the relations:

$$\frac{1}{M_u} = \sum_i \frac{w_i}{M_i} \quad (1.0)$$

$$N_i = \frac{w_i \rho N_A}{M_i} = \gamma_i \times N_{mix} \quad (2.0)$$

$$\gamma_i = \frac{Atoms\ of\ i}{Total\ atoms} \quad (3.0)$$

Where, M_u is the molecular weight of natural uranium in the fuel material,N_i is the region atom densities, N_A is the Avogadro's number, ρ is the density of Zircaloy-4, w_i is the weight percent of i and M_i is the molecular weight of i.

The results of the atom densities of various natural isotopes present in the individual nuclide of Zircaloy-4 are used to obtain the material composition data (atoms/b-cm) for the LEU fuel cell modelas shown in table 2.0. This result is then used to calculate average homogenized atom density (N_{iz}) for the LEU fuel material in the water mix region for the Zircaloy-4.

Table 2.0: The material composition data (atoms/b-cm) for the LEUfuel (19.75% UO_2) cell model

Material #	Material Name	Nuclide ZAID #	Region atom density (N_i)
1.	LEU Fuel	92235	4.7267e-3
		92238	1.8963e-2
		8016	4.7380e-2
2.	Zircaloy-4 (Clad)	40090	2.165e-2
		40091	4.721e-3
		40092	7.217e-3
		40094	7.314e-3
		40096	1.178e-3
		50112	5.054e-6
		50114	3.549e-6
		50116	7.818e-5
		50117	4.130e-5
		50118	1.302e-4
		50119	4.619e-5
		50120	1.752e-4
		50122	2.490e-5
		50124	3.113e-5
		26054	1.040e-5
		26056	1.633e-4
		26057	3.772e-6
		26058	5.020e-7
		24050	3.623e-6
		24052	6.987e-5
		24053	7.923e-6
		24054	1.972e-6
		72174	7.186e-9
		72176	2.362e-7
		72177	8.353e-7
		72178	1.225e-6

		72179	6.117e-7
		72180	1.575e-6
3.	Moderator (H_2O)	1001	6.6434e-2
		8016	3.3217e-2

The effective multiplication factorvalues generated by the VENTURE-PC code for different shim thickness were used to compute the reactivity worth of the top beryllium shims (see table 3.0), the value for k_{eff} at critical height is 1.0112886. The reactivity worth and the control rod worth may be expressed mathematically as:

$$Reactivity, \rho = \frac{k_{eff} - k_{eff}\ at\ critical\ height}{k_{eff}\ at\ critical\ height} \quad (4.0)$$

$$\text{Control rod worth} = \text{Reactivity worth} + \text{Shutdown margin} \quad (5.0)$$

Where the shutdown margin is the negative reactivity the reactors core present when the control rod is fully inserted (Henry et al., 2014).

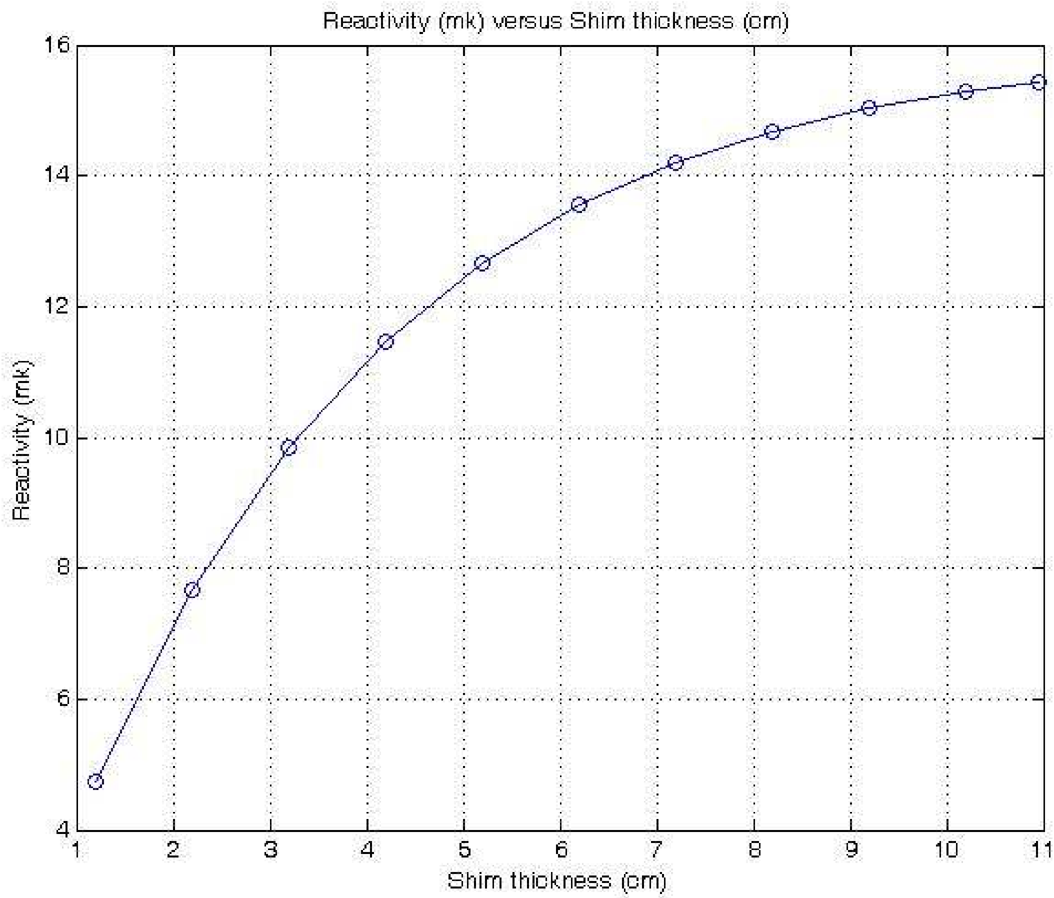
Table 3.0: Shim thickness, k-effective and reactivity for the NIRR-1 2-D model

S/N	Shim thickness (cm) (Top Beryllium thickness)	Reactivity (k)
1.	1.2	4.7336636e-3
2.	2.2	7.6673464e-3
3.	3.2	9.8483262e-3
4.	4.2	1.1468734e-2
5.	5.2	1.2669479e-2
6.	6.2	1.3556070e-2
7.	7.2	1.4208110e-2
8.	8.2	1.4686114e-2
9.	9.2	1.5035569e-2
10.	10.2	1.5290788e-2
11.	10.95	1.5437037e-2

The SCALE 6.1 code was used to generate the three major different cross section libraries developed in this work.

RESULTS AND DISCURSION

Results have been obtained for the neutronics characteristic of the projected LEU fuel (UO_2-Zircaloy-4-clad) for NIRR-1 MNSR core using the VENTURE-PC code and SCALE code system. The proper averaged cross section used for the homogeneous model in this work was generated using the heterogeneous fuel cell and control cell models. This model was used by SCALE 6.1 code to determine neutron flux as a function of both space and energy for each location in the cell model.



The above figure show the graph of measured reactivity worth versus shim thickness during the off-site zero-power test for the LEU core using the VENTURE-PC code. The value of the total control rod worth obtained for this work from the four group calculation for the LEU core is 7.23mk which is in good agreement when compared with similar results calculated by G.I Balogun using CITATION code (Balogun, 2003), S.A Jonah using MCNP code (Jonah et al., 2007) and S. Abdulhameed using VENTURE-PC code (Salawu, 2012). The value of reactivity worth of the top beryllium shim was obtained for each shim thickness ranging from 1.2cm to

10.95cm.The calculated magnitudes of axial distributions of thermal, epithermal, resonance and fast neutron flux in the inner and outer irradiation channels for the proposedLEU (19.75% UO_2) core for NIRR-1 were plotted as shown in figures 2.0 and 3.0 respectively. The inner and outer irradiation channels locations were selected at radial distances of 16.77cm and 26.79cm from the core center. The figures (2.0 and 3.0) also shows the computed results of the 2-D vertical Y-directed neutrons flux profiles expected at various region within the NIRR-1 (MNSRs) core model for the potential LEU core (19.75% UO_2). The thermal neutrons flux level, peak power density and maximum neutron density calculated in the potential LEU core are $1.24 \times 10^{12}\ ncm^{-2}s^{-1}$,$4.31033e+00\ W/cc$and $6.94535e-6\ neutron/cc$ respectively. The calculated thermal neutron flux level is approximately equal to that of the present HEU core of $1.1 \times 10^{12}\ ncm^{-2}s^{-1}$ for the present HEU core. We observed that the thermal neutrons flux level in the potential LEU core is slightly lower when compared with a similar neutrons flux in the HEU core. This implies that the total number of neutrons that were able to get to the thermal energy in the moderation or thermalization processes is lower in the LEU core than in the HEU core. Since the outer irradiation location is filled with water, weobserved a symmetric neutron flux profile in this region. We did not observed any reflector peak in the high energy neutron flux as shown in figures (2.0 and 3.0), this is due to the fact that fast neutrons are not affected on the average by the reflector materials (i.e. water or beryllium) in the NIRR-1 core.

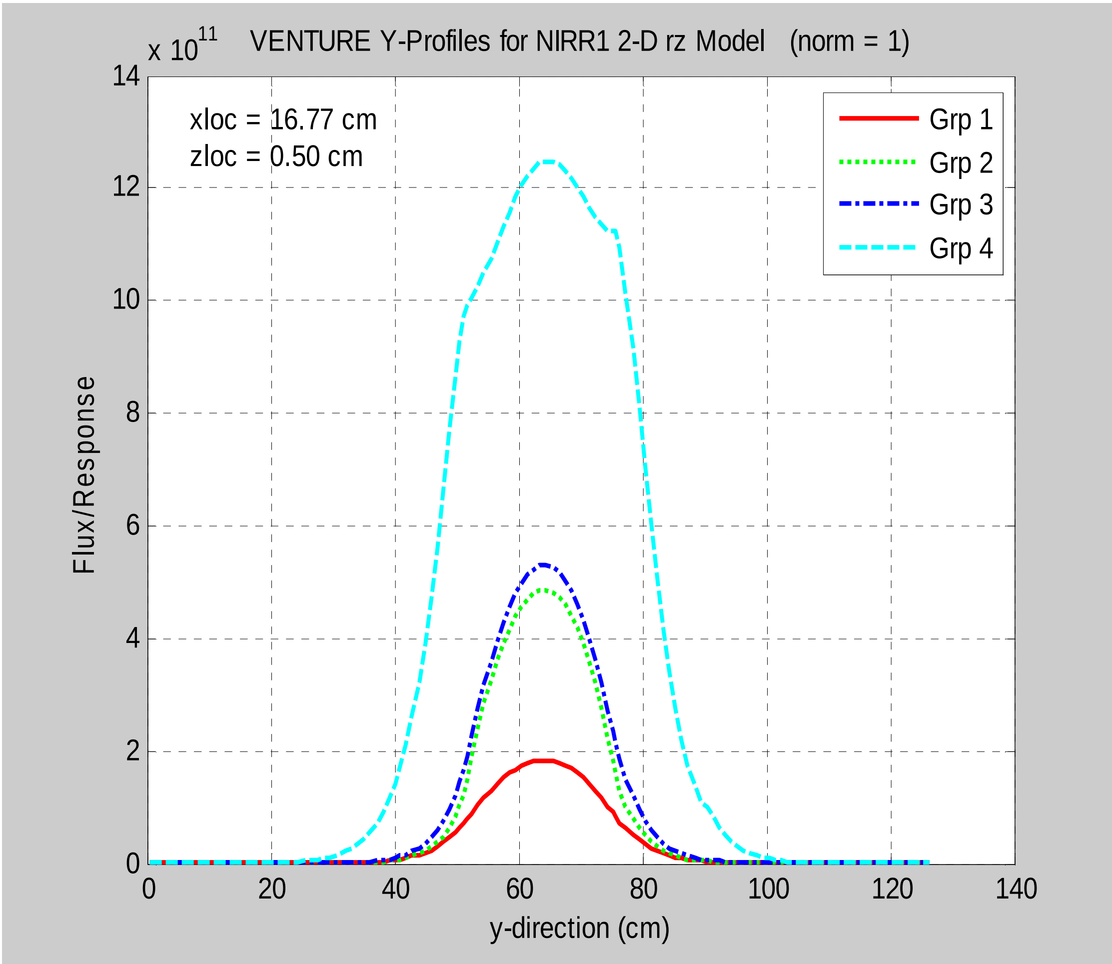


Figure 2.0: Vertical (Y)-directed neutrons flux profiles through the radial beryllium region for the LEU core (19.75% UO_2) for NIRR-1

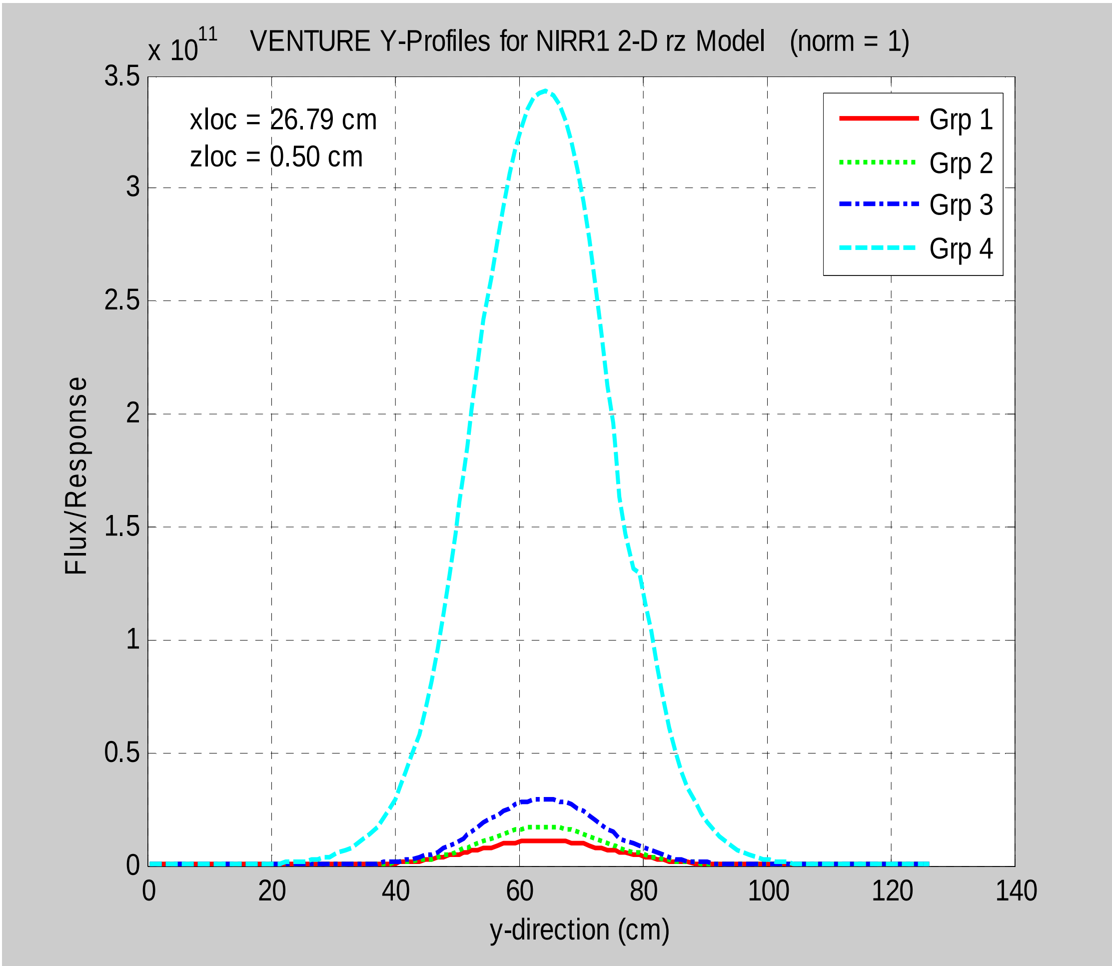


Figure 3.0: Vertical (Y)-directed neutrons flux profiles through the outer irradiation location for the LEU core (19.75% UO_2) for NIRR-1

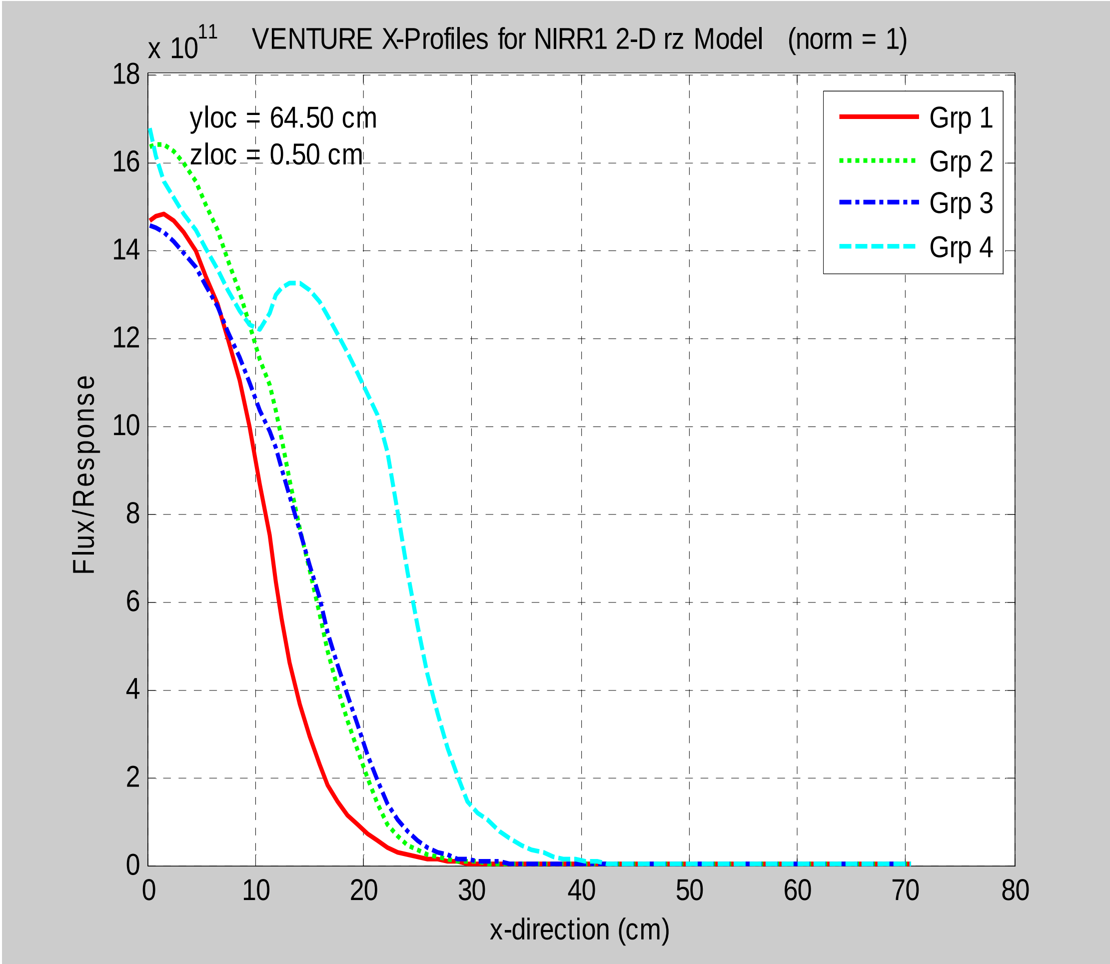


Figure 4.0: Horizontal (X)-directed neutrons flux profiles through the center of the LEU core (19.75% UO_2) for NIRR-1

Figure 4.0 shows the X-directed neutrons flux profiles starting from the center of the active fuel region in the axial direction. The reflector peak shown in the thermal neutron flux profile represents the effect of the radial beryllium on the thermal neutrons.The above plots reveal that the shapes of the neutron flux in the 19.75% LEU core were similar to that of the present HEU core of NIRR-1.The only different is that the neutron flux level is a little bit different in the 19.75% LEU core as compare to the present HEU core of NIRR-1. The detail information shown by these plots can be of great help in selecting a region with high neutron flux for positioning irradiation channels within the NIRR-1 reactor core.This result is consistent with the fact that Zircaloy-4 cladding gives higher neutron economy which has positive impact on the excess reactivity of the system.

CONCLUSION

The thermal neutron flux in the proposed LEU core (with 19.75% UO_2 material) is a little bit less than the present HEU core. This insignificant reduction in the flux for the same power level has shown that the NIRR-1 reactor can be fuelled with the potential 19.75% UO_2 (LEU) with about 200 active fuel pins without any significant reduction in the NIRR-1 performance.

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