

Simulated Thermal Neutron Flux Distribution Pattern in a Neutron Tank at the Schuster Labouratory University of Manchester

NWOSU, O.B

Department of Physics and Industrial Physics,
 Nnamdi Azikiwe University Awka, Anambra state, Nigeria.
 bo.nwosu@unizik.edu.ng, Zigma02@yahoo.com

Abstract—MCNP4c code was used to model and tally the thermal neutron flux in the neutron tank in Schuster labouratory University of Manchester, so as to observe the way thermal neutron distribute themselves across the diameter of the tank analogous to critical thermal reactor core. Four he-3 detectors was modelled to have been inserted at different distances from the Pu – Be neutron source at the centre of the tank made of polythene wall 113cm high and 165 cm in diameter. The wall was later assumed to be made of stainless steel and again tallying for flux at each point. This was done to observe the effect this will have on the neutron flux pattern anologous to reflector and bare core. A unique thermal neutron flux diffusion pattern was observed with both, a bell shaped curve with neutron flux maximum at the center of the tank and decreasing towards the edges analogous to the distribution pattern in a critical thermal reactor core. However, the graph from the simulation with polythene walled neutron tank showed a little rise at the edges of the tank, while the stainless steel graph had no such rise in flux towards the edges of the tank. This may be due to the fact that the polythene wall acted as a moderator and reflector thermalizing and preventing the escape of fast neutron leading to such little rise at the edges, while stainless which has no moderator property could not. These effects are analogous to reactor with a reflector or bare core. The Maxwellian-Boltzmann distribution of neutron flux against energies was also done and it showed approximate resemblance and confirmation of this theory. The knowledge from the spatial flux distribution pattern from this study reveals power level and distribution in the reactor core, and is significant in that it has helped to avoid economic wastage because the non-uniform thermal neutron flux distribution is unfavourable, and it results in a non-uniform burn-up of the fuel rods. The central rods are known to burn out soon while rods arranged at the edge of the core are hardly burnt except a reflector material is used around the core to help in fuel utilization or enriching fuels at the edges higher than those at the centre to balance thermal flux deficiency and sufficiency

Keywords—Nuetron flux, MCNP, He-3 detector, thermal neutron

1.0 Introduction:

He-3 detectors are known to be very efficient in detecting thermal neutrons (6). The water in the neutron tank will thermalized the fast born neutron from the Pu-Be neutron at the center of the tank. By assuming and modelling a situation where four He-3 neutron detectors are placed inside the neutron tank at different distances from the neutron source, thermal neutron flux could be tracked and tallied at these distances from the source to obtian their distribution pattern within the tank.

2.0 Theoretical framework.

The theories behind this work was discussed briefly here

2.1 Neutron Transport Theory

The neutron Boltzmann equation, also called the neutron transport equation describes neutron behaviour like scattering and absorption in a medium filled with nuclei. Formulation of this useful equation is based on the concepts of phase space (1). This Boltzmann equation for radiation transport have been expressed as:

$$\frac{1}{v} \frac{\partial \Phi}{\partial t}(r, E, \Omega, t) + \underline{\Omega} \cdot \underline{\nabla} \Phi(r, E, \Omega, t) + \sum_i \cdot \Phi(r, E, \Omega, t) = \int_{\Omega'} \int_E \sum_s (E' \rightarrow E, \Omega' \rightarrow \Omega) \Phi(r, E, \Omega, t) dE' d\Omega' + S^+(r, E, \Omega, t) \quad 2.1$$

Where

$\frac{1}{v} \frac{\partial \Phi}{\partial t}(r, E, \Omega, t)$ -refers to the in-balance term in the equation

$\underline{\Omega} \cdot \underline{\nabla} \Phi(r, E, \Omega, t)$ - refers to the streaming term describing particles or

neutrons that went through our reactor or the neutron tank without interaction.

$\sum_i \cdot \Phi(r, E, \Omega, t)$ - referred to as the total loss term which takes care of all the reactions leading to absorption and loss of neutron from the medium.

$$\int_{\Omega} \int_E \sum_s (E' \rightarrow E, \Omega' \rightarrow \Omega) \Phi(r, E, \Omega, t) dE' d\Omega'$$

- referred to as the scattering term describing all the scattering reaction like the elastic and inelastic reaction taking place in the neutron tank by the hydrogen atom or nuclear reactor leading to direction and energy change resulting to the moderation of the neutron etc.

$S^+(r, E, \Omega, t)$ Referred to as the extraneous source term, which takes care of particle (neutron) source in the medium which includes Pu/Be(α, n) source in the neutron tank and fission, (γn ,) etc. in the reactor.

Two major methods, the deterministic and the Monte Carlo methods are used for solving the Boltzmann neutron transport equation stated above. The Monte-Carlo method (MCNP) was employed in solving this equation for the neutron transport in the neutron tank, where the source term is the Pu/Be neutron source, the scattering term the water of the tank, the absorption term is the Helium gas in the detector.

2.2 Monte Carlo

Monte Carlo methods can be loosely described as statistical simulation methods utilizing sequences of random numbers to perform the simulations. In this method, a model of the neutron tank under study was set up in the computer and individual particles (neutrons) are thrown at it from the source. The various events like collisions absorptions etc in which the particle participate are followed and recorded by the program. These events in which the particle will undergo are called its history. Many simulations or histories are performed and the desired result is taken as an average over the number of observations. Monte Carlo calculations because of its statistical nature usually involve a considerable computer time to attain reliable result. Examples of Monte-Carlo code includes. MCBEND, MONK, GEANT4, MCNP. The one used in this work is the MCNP

2.2 MCNP:

This is the general- purpose Monte Carlo N-Particle transport code that can be used to track for neutron (as done in the neutron tank) as well as, photons and electrons. It uses a continuous –energy nuclear and atomic data libraries, to treat three – dimensional configuration of materials in geometric cells bounded by first, second and third degrees (2). The maximum line length in MCNP input file is 80 columns and it has a fixed structure that must be adhered to, if the programme is to give a reliable output. These are; the title card, Cell cards, Surface card, and Data cards. Each of these cards is defined using a series of commands containing parameters, starting each card on a new line. The important concepts that were followed in MCNP in-put files in this work is:

1. **Geometry** (cells and surface): Defined by 3-dimensional configuration of user define materials in geometrical cells bounded by first and second degree surface. MCNP uses Cartesian axes of x,y,z and it's chosen arbitrarily.

- Cells card: used to define intersections, unions and complements of the regions bounded by the surfaces of our objects.

- Surfaces card: Defined by supplying coefficients to the analytic surface equations; with a combination of planes, spheres and cylinders we can construct simple geometric shapes as done here. To plot this geometry we use the command mcnp ip n=filename (2).

- **Source definitions card:** defined by using the source definition command (sdef card), which carries some parameter like

- Pos; for x, y, z position of our point source.
- Erg; which define the energy of the emitted particle, with energy distribution d1.

- **Tallies card (F's cards):** This is used to specify the type of information we want to obtain from the MCNP simulation, like particle current, flux, energy etc thermal neutron flux was tracked in this work with F4 cards.

- **Material specification Card:** Defines the isotropic composition of the materials in the cells and the cross section library to be used.

We usually set a specific time we want the programme to run with the command ctme= 120. To run MCNP file we command as follows: mcnp ixrz n=filename. When this computer time has elapsed, it will display 3 additional output files, but the one with the tag "o" has the output of our tallies from which I considered the flux with the energy close to the thermal neutron energy range.

2.3 The Maxwellian–Boltzmann distributions

This distribution commonly referred to as Maxwellian distribution tries to assume at first approximation the behaviour of the kinetic energies of neutrons in an infinite non-absorbing scattering medium. That is neutrons in a medium where they are neither lost by absorption nor lost by leakage. The kinetic energy distribution of neutron in thermal equilibrium at absolute Kelvin temperature T as expressed in

(3), shows that

$$\frac{n(E)}{n} = \frac{2\pi}{(\pi KT)^{3/2}} e^{-E/KT} E^{1/2} \quad .2.8$$

Where the left side of the equation represent the fraction of the neutrons having energies within a unit energy interval at energy E, while the right side can be evaluated for various E's at a given temperature; and the Maxwellian curve obtained in this manner

indicating the variation of $\frac{n(E)}{n}$ with the kinetic energy of the neutron is shown in fig1.

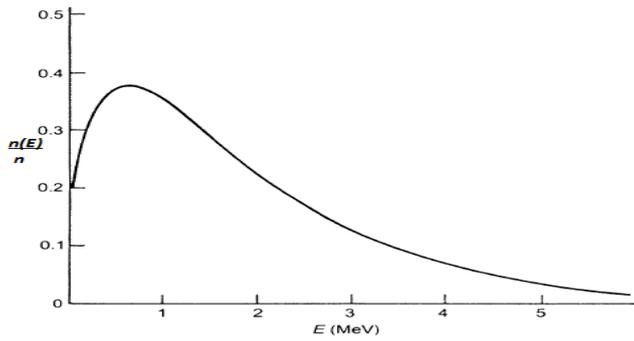


Fig1; Maxwell- Boltzmann energy distribution (4,1)

$n(E)$ is the number of neutrons of energy E per unit energy interval, with $\frac{n(E)}{n}$ indicating the variation with the kinetic energy of the neutrons

In a thermal reactor with or without a reflector as schematically shown of fig 2.

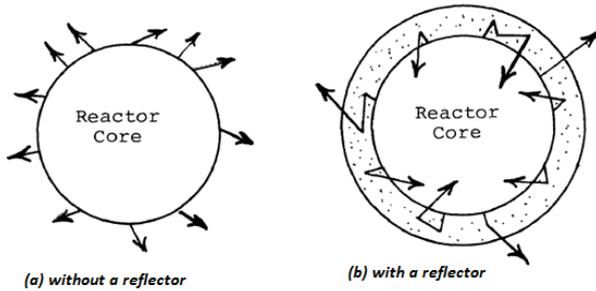


Fig 2: the schematic representation of reactor core without and with a reflector (5)

The general thermal neutron flux distribution as presented in (4,1) is as shown on fig 3.

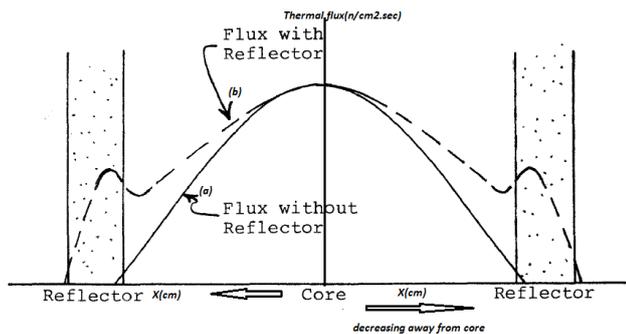


Fig 3: Thermal neutron flux distribution in a spherical ^{235}U water-moderated reactor with and without a reflector (4,1)

3.0 COMPUTER MODELLING AND SIMULATION WITH MCNP CODE

The thermal neutron flux distribution in the neutron tank was modelled using MCNP4C; Monte Carlo N-Particle Transport Code System Oak ridge national laboratory managed by UT-BATTELLE, LLC for the U.S department of energy. The neutron tank is 113cm high and 165 cm in diameter shown on fig4



Fig 4 snapshot of the neutron tank in Manchester laboratory used for the modelling.

The water in the tank besides shielding effect moderate and thermalized the fast born neutron. The simulation was done with four He-3 neutron detectors (cell 10, 21, 22 and 23) (30cm high and 2.5cm diameter detector; normal length of 100cm was reduced so as to get more direct impingement and avoid dividing the detectors into cells). Four detectors was used so as to cover much distance within just few number of simulations and each detector was placed at the four cardinal points inside the water tank as shown on fig 5, but at different distances from the source, to simultaneously measure the thermal neutron flux from the assumed isotropic neutron source.

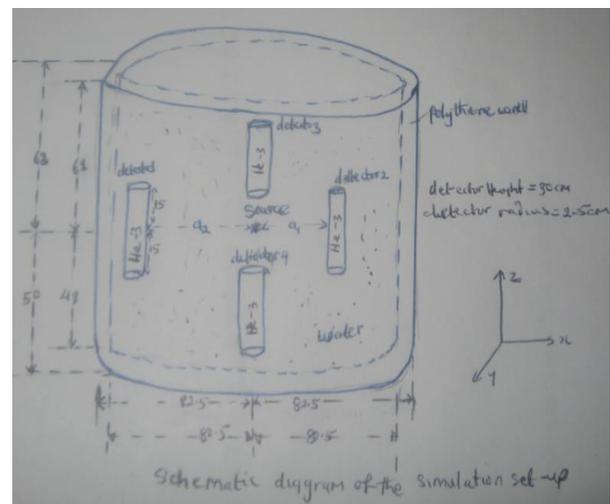


Fig 5. The schematic diagram for MCNP simulation set-up (sketched and snapped) The 1st -set-up has the wall of the tank thick and made of polythene, while the 2nd -set-up has the wall of the tank bare and made of stainless steel. I was trying to see if the two set-up could behave like a bare and reflected reactor while also observing the flux distribution pattern.

3.1. MCNP GEOMETRIC MODEL AND ASUMPTIONS

Two set of simulation was done with four He-3 detectors placed at each cardinal point and moved to a new position for another simulation.

The Pu/Be was assumed to be a point isotropic source placed at the centre of the tank. Material 1 was taken as the detector, material 2 as the water moderator and material 3 as polyethylene or stainless steel as the case may be.

4.1 First Simulation set-up: A second layer was added to the outer layer of the cylindrical neutron water tank to make it thick and assumed its material to be made with polythene. Thermal treatment card for lwrt and for polythene was included for thermal treatment and densities as well. The geometry of this set-up gotten from the MCNP plot window before running the programme is as shown on fig6.

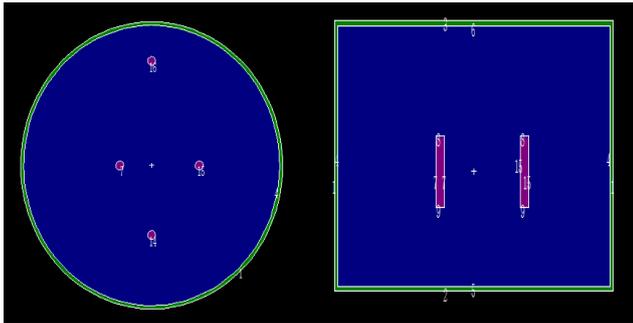


Fig 6. The obtained Geometry from the MCNP plot window (a) in the Z direction i.e. from top and (b) in the x and y direction. (The pink object is the detector, the blue colour is the moderating water, and the green is the polythene wall.)

Table1 Simulated thermal neutron flux distribution in the neutron polythene walled tank as a function of distance from centre.

distance(cm)	$\phi_{th} (n / cm^2 . sec)$
5	1.83E+03
10	8.69E+02
15	3.61E+02
20	1.57E+02
25	6.56E+01
30	2.77E+01
35	1.26E+01
40	6.23E+00
45	2.72E+00
50	1.27E+00
55	6.20E-01
60	2.82E-01
65	2.22E-01
70	1.30E-01
75	1.55E-02

After running the programme, the output 'fileo' contain neutrons with different energy, but only that corresponding to the thermal energy range was picked noting the corresponding distance at which that detector was placed. The table of the simulated result

of the thermal neutron flux distribution in the polythene walled neutron water tank described above is displayed in table 1.

It can be observable that there is a decrease of flux with distance obeying the Fick's law of diffusion. Plotting the thermal neutron flux against distance on a log scale, we obtain a graphical relationship shown on fig7 .

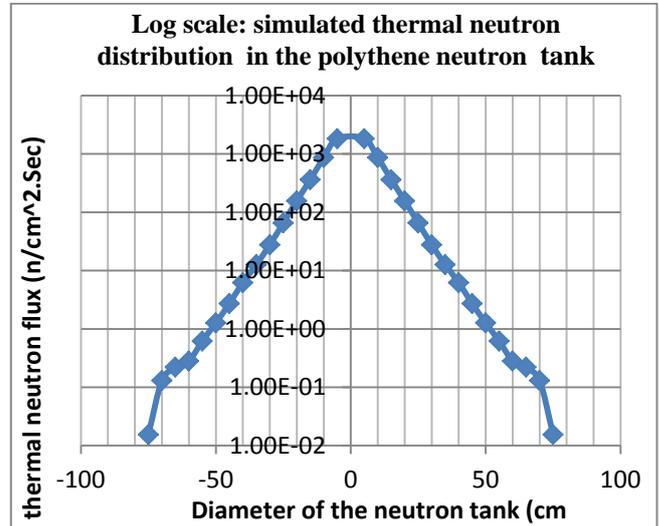


Fig 7 Simulated graph of neutron flux distribution pattern across the neutron tank made of polythene material.

The above figure shows the thermal neutron distribution pattern in the the nuetron tank in Schuster laboratory, University of Manchester. It was observed that thermal neutron flux decreases and tends to zero as we move from the centre towards the edge of the tank, having a definite maximum at the centre. Small neutron flux rise was observed at the end of graph (edges of the tank). This could be due to the effect of the polythene wall which has a moderating property that helped in thermalizing and preventing the escape of some fast neutrons from the tank. In comparison with fig2 (5.4.1) this is analogous to what is obtainable from a thermal reactor with a reflector round it.

4.3.2 The second set –up

This is just like the first but just a single wall made of stainless steel (chromium, nickel and iron). The thermal treatment card present in the previous set for polythene was removed, only that for water is left and densities as well. The MCNP plot of that file to check for geometry before simulation is shown on fig8

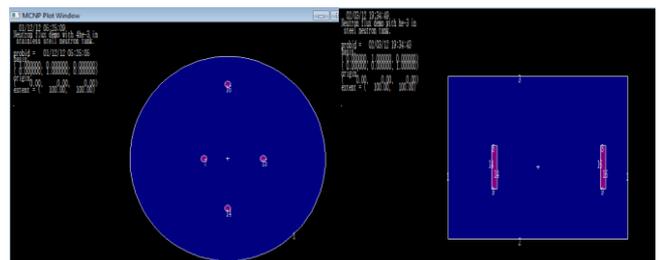


Fig 8. The obtained Geometry from the MCNP plot window for stainless steel wall (a) in the Z direction ie from top and (b) in the x and y direction.(the pink object is the detector, the blue colour is the moderating water, and the single white line is the stainless steel wall of the tank.)

Running the programme as previously described the following reading on table 2 was obtained of thermal neutron flux against distance

Table2. Simulated thermal neutron flux distribution in the neutron stainless steel walled tank as a function of distance from neutron source.

Distance (xcm)	$\phi_{th}(n/cm^2.sec)$
5	1.8E+03
10	8.7E+02
15	4.1E+02
20	1.6E+02
25	6.6E+01
30	2.8E+01
40	6.2E+00
45	2.7E+00
50	1.3E+00
55	6.2E-01
60	3.0E-01
65	2.2E-01
75	5.1E-02

Plotting the above reading on the we obtain the thermal flux distribution pattern in the neutron tank made of stainless steel as shown on fig9.

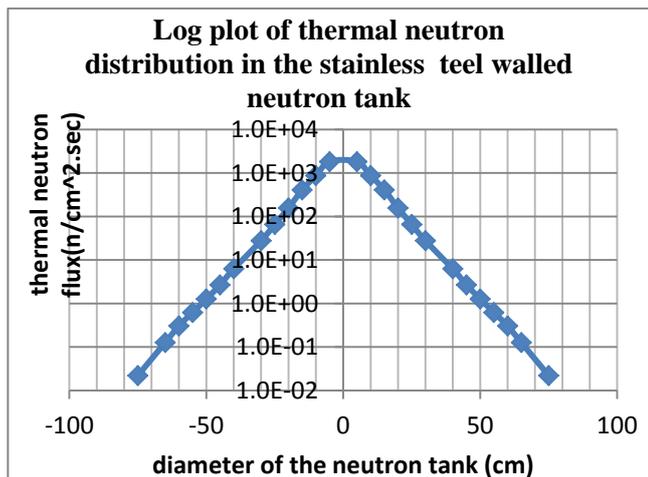


Fig 9 Simulated graph of neutron flux distribution against distance from the neutron source in the neutron tank made of stainless steel material.

The thermal neutron pattern still maintain its distribution of decreasing as we move toward the edge of the tank, however the kink(little flux rise) seen in the previous graph has disappeared probably because the stainless steel cannot thermalized and reflect neutron back into the tank. It is obvious that the

property of the polythene wall is different from that of the stainless steel wall. In comparison with fig2 (5,4,1) this is analogous to what is obtainable from a thermal reactor without a reflector round it, ie bare core.

5.0 The simulated Maxwell-Boltzmann distribution

Neutrons of different energy range against the flux (table3) was plotted to give the graph shown on here on fig 1

Table3Maxwell-Boltzmannneutron distribution table

cell 22 (4th detector) 60cm from source		
energy	flux	rel.error
5.00E-07	1.57E-08	0.126
1.00E-06	2.61E-09	0.6081
5.00E-06	8.21E-09	0.355
1.00E-05	5.15E-10	1
5.00E-05	3.95E-09	0.5852
1.00E-04	0.00E+00	0
5.00E-04	1.23E-08	0.3655
1.00E-03	6.35E-09	0.4618
5.00E-03	7.00E-09	0.4494
1.00E-02	4.66E-09	0.5473
5.00E-02	3.30E-09	0.622
1.00E-01	1.22E-08	0.3645
5.00E-01	2.10E-08	0.2711
1.00E+00	1.50E-08	0.3256
5.00E+00	5.85E-08	0.1581
1.00E+01	5.67E-08	0.1517

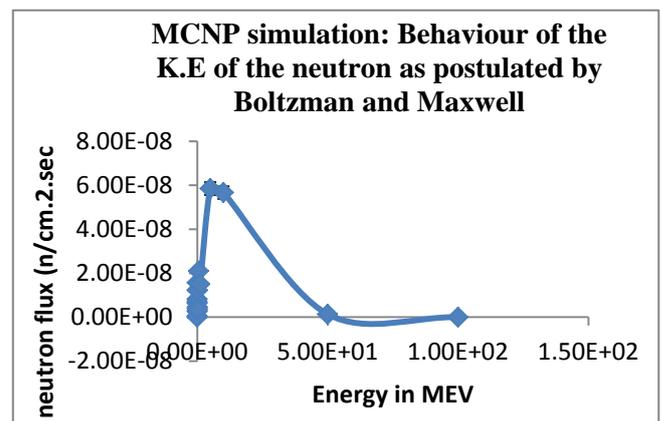


Fig10. the simulated Behaviour of the K.E of the neutrons and flux as postulated by Maxwell and Boltzmann

The Maxwellian curve obtained as shown above indicating the variation of $\frac{n(E)}{n}$ with the kinetic energy of the neutron, is approximately similar with the theoretical distribution shown previously on fig1 by (3,1) portraying how neutrons behave in a scattering and non-absorbing medium

5.2 Comparison between the MCNP simulations: (polythene wall (red) and stainless steel wall (blue))

From the graphs of two set up of polythene wall(fig7) and stainless wall(fig9), we combined to obtain that on fig11. we could see both behaving like the theoretical graph on fig2 showing the graph for reflected and bare reactor core.

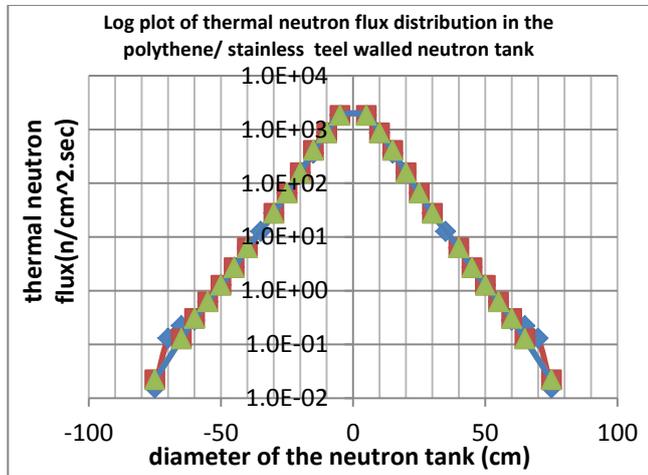


Fig 11 the simulated graph of the polythene/ stainless steel wall behaving a bit like the reflected core and bare core.

We could observe a little rise in thermal neutron flux which means little power rise if it's in a reactor. The flux near the edge rises a little about 10% of the maximum flux before dropping off finally to zero. May be the polythene which is a good neutron moderator is acting a little like a reflector; a reflector is a

material with moderating properties placed around a reactor core to reflect neutrons back into core for more fission. This isn't same with stainless steel walled graph, because the steel has no moderating properties. The flux still retains maximum value at the center in each case, but the average flux may have risen to about 40% in the reflected core when both the axial and radial distribution of flux is accounted for.

From what we have known, we could then say that having a reflector round the reactor is a good idea, the implication of a reactor without it is that thermal neutron flux distribution seen in the stainless simulation equivalent to that seen in bare reactors will lead to

- Poor use of fuel in the outer region of the core to produce power
- Differential burning of fuel
- Limitations of total power produced because of high ratio of the maximum flux to average flux

Most reactors have reflectors, AGR is said to have a graphite reflector around its core, and its presence leads to decrease of the linear dimensions of the critical reactor.

Acknowledgements:

I wish to acknowledge the staff and personnels of school of Physics and Astronomy and the Schuster Laboratory University of Manchester for providing me with these expensive necessary equipments and materials for this work. I wish to also acknowledge TETFUND Nigeria (tertiary education trust Fund) for providing all the necessary funding for this research.

References:

- (1)Weston M. Stacey,2007; Nuclear Reactor Physics, Second Edition, Completely WILEY-VCH Verlag GmbH & Co. KGaA, Weinheim
- (2)Andrew Boston; 2000. Introduction to MCNP-the Monte Carlo transport code Olive Lodge Laboratory
- (3)Samuel, G. and Alexander, S; 1981: Nuclear reactor engineering, Van Nostrand Reinhold Ltd New York N.Y 10020; PP13-
- (4) Kenneth, J.S and Richard E. F, 2008 Fundamentals' of nuclear Science and Engineering 2nd edition, CRC press Taylor & Francis Group Boca Raton p194
- (5) canteach.candu.org/library/20053707; Nuclear Theory –course 227; Neutron flux distribution
- (6)Maayouf R.M.A; Bashter, I.I; Maamly, W.M.; And Khalil, M.I; 2008.

Neutron Flux Measurements at the CFDF using Different Detectors; IX Radiation Physics & Protection Conference, 15-19 November 2008, Nasr City - Cairo, Egypt