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DETERMINING NEUTRON FLUX PROFILE IN HOMOGENOUS AND THERMAL REACTOR

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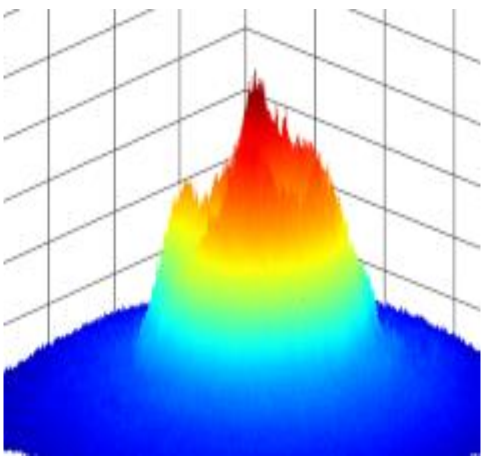
Graphical abstract



Abstract

The relationship of neutron flux and power density is directly proportional. It is essential to analyze the behavior of neutron flux to determine the thermal output power of nuclear reactor. This project is simulating the neutron flux profile using Octave and the user can manipulate some of the input such as fuel type, reactor shape, moderator and etc. In this project, neutron flux profile is determined by solution of diffusion equation for each shape such as cylinder, rectangular parallel-piped and sphere. Each shape has different equation for determining the neutron flux profile and the user can choose which shape they want. User will obtain the result of neutron flux in rectangular parallel piped reactor, sphere reactor and cylinder reactor. Each reactor shape will give their own profile based on the user input. In a nutshell, this project will benefit Nuclear Engineering Students who are taking Nuclear Reactor Theory for visualizing and understanding the neutron flux profile in reactor core.

Abstrak

Hubungan antara fluks neutron dan tenaga yang dihasilkan di dalam reaktor nuklear ialah berkadar secara langsung. Ianya sangat penting untuk mengenal pasti perkembangan fluks neutron untuk menentukan tenaga thermal yang akan dihasilkan oleh reaktor nuklear. Projek ini akan menghasilkan simulasi fluks neutron menggunakan perisian komputer seperti "Octave" dan pengguna boleh mengubah suai input program seperti jenis bahan bakar, bentuk reaktor, moderator dan lain-lain. Dalam projek ini profil fluks neutron ditentukan oleh solusi persamaan penyerapan fluks neutron. Setiap bentuk reaktor mempunyai persamaan yang tersendiri untuk menentukan profil fluks neutron dan pengguna boleh memilih bentuk yang dikehandaki. Pengguna akan menerima keputusan fluks neutron daripada reaktor berbentuk parallelepiped segi empat tepat, sphera dan silinder. Setiap reka bentuk reaktor akan memberi profil masing-masing berdasarkan input pengguna. Kesimpulannya, projek ini memberi manfaat kepada pelajar Kejuruteraan Nuklear yang mengambil subjek Teori Nuklear Reaktor untuk memahami dan menganalisis profil fluks neutron di dalam teras reaktor

1.0 INTRODUCTION

1.1 Research Background

The energy produced in nuclear power plant is from fission reaction where the bigger atom is split into two smaller atoms producing some neutrons and releasing energy. Neutron is bombarded to the fissile atoms usually ^{235}U to initiate the fission process. Fissile material is the material that its atom can undergoes fission reaction. Uranium-235, Plutonium-239 and Uranium-233 are fissile material.

This reaction occurs continuously if the critical mass of the fissile material achieved and this process is called chain reaction, where the neutron produced from the nuclear fission continue bombarded others atom and undergo nuclear fission [15]. In nuclear reactor, the huge amount of neutron produced and its travelling distance is defined as neutron flux. The chances of the fissile atom to undergo nuclear fission is increase when the neutron flux is increase and the power generated also higher.

In conventional nuclear power plant, neutron flux consists of different energy spectrums. Neutron flux in nuclear power plant is detected by fast and slow neutron detector placed in the reactor and recorded in the computer. The reading of the neutron flux in the real nuclear reactor recorded by using instrument.

However, in this project only mathematical equation is used to observe the neutron flux in the reactor. The type of reactor is homogeneous compared to the real nuclear reactor which is heterogeneous. This equation also valid for single group of energy spectrum of the neutron. Neutron cross section for moderators and fuels will be different for every energy spectrum. Mathematical equation used in this project is the solution of neutron diffusion equation. There are different shapes of homogeneous reactor can be observed using solution neutron diffusion equation such as slab, sphere and cylinder. The mathematical equation is converted into computer code to visualize the shape of neutron flux.

1.2 Research Aim

Creating simulation of monoenergetic neutron flux in homogenous reactor and by using software such as Octave or SCILAB to visualize the pattern. To observe the difference of neutron flux result if the parameters such as moderators, reactor shapes, fuel type and etc.

1.3 Problem Statement

Visualization of neutron flux by using neutron diffusion equation require a lot of steps. Each shape has different solution of neutron diffusion equation. Neutron diffusion equation is rate of change of neutron flux per length. By integrating the equation (1), the solution is obtained. Each shape, has different differential equation depending on their own Laplacian. The solution is used as governing equation in the computer codes to generate data and plotting the graph.

$$D\nabla^2\emptyset(r) - \Sigma_a\emptyset(r) + S(r) = 0 \quad (1)$$

Where:

$$S(r) = k_\infty \Sigma_a \emptyset(r) \quad (2)$$

$$k_\infty \Sigma_a = v\Sigma_f \quad (3)$$

There are a lot of parameters have to be considered for determining the neutron flux profile like critical mass of the fuel, critical size, macroscopic fission cross-section and etc. Each of the parameter has its own step and solution. The step and calculation in this project only valid for homogeneous reactor without reflector and thermal neutron.

In this project the neutron flux is observed with the power level of the reactor and size of the reactor. The reading of the neutron flux and graph generated is based on the power level and size of the reactor set by user. The user also can choose the shape of the reactor they want to observe. To achieve this, the computer program has to be written by converting the mathematical calculation to the computer code.

1.4 Objectives of Study

The objectives of the project are:

i. To generate the neutron flux data and graph for single energy spectrum of thermal group and the users can manipulate some of the parameters such as type of moderator, reactor shapes and sizes, operating power, energy spectrum and etc.
ii. To analyze the change of neutron flux profile with different value of parameters.

1.5 Scopes of Study

The scopes of the study of the project are:

i. Application of neutron diffusion equation.
ii. The study of parameters that affected the behavior of neutron flux profile.
iii. Applicable for the homogeneous nuclear reactor and without reflector and thermal neutron group.

1.6 Significance of Study

The significance of the study of this project are to observe the neutron flux profile in the homogeneous nuclear reactor.

1.7 Neutron Flux in Nuclear Reactor

Neutron flux can be defined as the number of neutrons circulating per length for every second based on its units. Neutron flux is introduced as total density of neutron striking the target in all directions [5]. The formula of neutron flux can be observed in eq. (4).

$$\emptyset = nv \ (4)$$

The pattern of or reading of neutron flux is important in the nuclear reactor for determining the power production. In reactor, neutron flux is govern using control rod made from good neutron absorber such as Boron, Cadmium, Silver and etc [2]. This is shown by the eq. (5). Based on the equation we can conclude that that neutron flux is directly proportional to the power.

$$P = E_R \Sigma_f \int ø(r)\, dV \ (5)$$

1.8 Neutron Diffusion Equation

In the reactor where the fission source is exists, $\Sigma_f \neq 0$. The eq. (1) can be defined as.

$$D\nabla^2\emptyset(r) - \Sigma_a\emptyset(r) + k_\infty \Sigma_a \emptyset(r) = 0$$

After dividing the eq. (1) with Σ_a, the differential equation will be,

$$\frac{1}{L^2}\nabla^2\emptyset(r) - \emptyset(r) + k_\infty\emptyset(r) = 0 \ \ (6)$$

$$\mathrm{B}^2 = \frac{k_\infty - 1}{L^2} \ \ (7)$$

$$\nabla^2\emptyset(r) + \mathrm{B}^2\emptyset(r) = 0 \ (8)$$

1.8.1 Differential Equation for Rectangular Parallel Piped

$$\left(\frac{\partial^2}{\partial x^2} + \frac{\partial^2}{\partial y^2} + \frac{\partial^2}{\partial z^2}\right)\emptyset(r) = -\mathrm{B}^2\emptyset(r) \ \ (8.1)$$

1.8.2 Differential Equation for Sphere

Only r-direction is considered:

$$\left(\frac{\partial^2}{\partial r^2} + \frac{2}{r}\frac{\partial}{\partial r}\right)\emptyset(r) = -\mathrm{B}^2\emptyset(r) \quad (8.2)$$

1.8.3 Differential Equation for Cylinder

Only r-direction is considered:

$$(\frac{\partial^2}{\partial r^2} + \frac{2}{r}\frac{\partial}{\partial r}))\emptyset(r) = -\mathrm{B}^2\emptyset(r) \quad (8.3)$$

1.8.4 Solution of Neutron Diffusion Equation

The Laplacian for each shape is inserted into the differential equation and solved by using differential equation method and conditional boundary is applied to obtain the solution of neutron flux. Each Laplacian is different for every shape as stated in **Appendix A**.

The conditional boundary for each shape is similar. When the position is minimum or at the center, the neutron flux will be maximum. However, if the position is further from the center the neutron flux will be decrease and approached to the zero [10]. **A** is the constant for the solution of the equation. The value of A is can be found by solving the eq. (5).

Table 1.1 Neutron Diffusion Equation Solution

Shape	Solution	A
Rectangular Parallel piped	$A\cos\left(\frac{\pi x}{a}\right).\cos\left(\frac{\pi y}{b}\right).\cos\left(\frac{\pi z}{c}\right)$	$\frac{3.87P}{VE_R\Sigma_f}$
Sphere	$A.\frac{1}{r}.\sin\frac{\pi r}{R}$	$\frac{P}{4R^2E_R\Sigma_f}$
Infinite Cylinder	$A\,.J_0(\frac{2.405r}{R})$	$\frac{0.738P}{R^2E_R\Sigma_f}$
Cylinder	$A\,.J_0\left(\frac{2.405r}{R}\right).\cos\left(\frac{\pi z}{H}\right)$	$\frac{3.63P}{VE_R\Sigma_f}$

At the center of the reactor, when the coordinate is (0,0), the neutron flux is maximum and equal to A.

1.9 Thermal and Homogeneous Reactor

In thermal and homogeneous reactor, the fission factor and resonance escape probability are equal to unity, $\epsilon\ \&\ p = 1$ due to there is no fertile material exist and the microscopic cross section absorption of neutron is in the 1/v region. Therefore, only reproduction factor **η** and thermal utilization **f** is considered in this project [13].

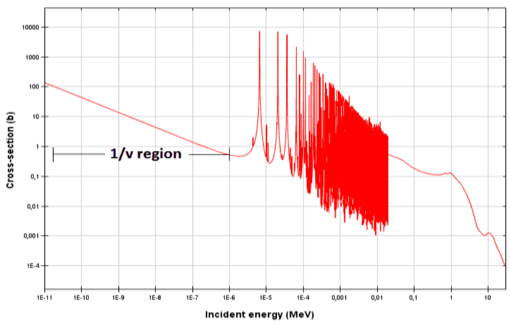


Figure 1.1 Neutron Cross-Section

Based on Figure 1, the only energy spectrum applicable for this type of reactor is ranged between 1E-5 MeV and below. In conclusion, all the cross section evaluated in this research are in thermal group.

1.10 Critical Mass and Size of the Reactor

To sustain chain reaction of fission reaction, the critical mass of fissile material is needed. In nuclear reactor, the critical mass is depending on its size. This factor is due to the relation of geometrical and its fuel and moderator composition. The value of geometrical buckling and material buckling is equal ,$B^2{}_m = B^2{}_g$ [12]. Table 2 list all the geometrical buckling for each reactor shape as mentioned in [8].

Table 1.2 Buckling Equation

Shape	Buckling ($B^2{}_g$)
Slab	$(\frac{\pi}{a})^2$
Rectangular Parallel Piped	$(\frac{\pi}{a})^2 + (\frac{\pi}{b})^2 + (\frac{\pi}{c})^2$
Sphere	$(\frac{\pi}{R})^2$
Infinite Cylinder	$(\frac{2.405}{R})^2$
Cylinder	$(\frac{2.405}{R})^2 + (\frac{\pi z}{H})^2$

When the dimension of the reactor is given, the critical mass of the reactor can be computed by relating the equation and steps below [9]. **Z** is introduced to make the step simpler. **Z** can be defined as:

$$Z = \frac{\Sigma_{af}}{\Sigma_{aM}} = \frac{N_f \sigma_{af}}{N_m \sigma_{aM}} \quad (9)$$

Thermal utilization **f** can be defined as:

$$f = \frac{Z}{1+Z} \quad (10)$$

In thermal and homogenous reactor, k_∞ can be defined as:

$$k_\infty = M_T^2 + B^2 \; (11)$$

Where:

$$M_T^2 = L_T^2 + \tau_T^2 \quad (12)$$

M_T^2 is defined as migration area where the neutron is being moderated [5]. L_T^2 is diffusion length while τ_T^2 is introduced as neutron age. The constant value of both parameters can be obtained at Evaluated Nuclear Data File (ENDF) [4].

By deriving the equations above **Z** can be defined as:

$$Z = \frac{1 + B^2 M_T^2}{\eta - 1 - B^2 \tau_T^2} \quad (13)$$

Mass of fuel, m_f can be obtained by,

$$m_f = Z \frac{\sigma_{am} V M_f}{N_a \sigma_{af}} \quad (14)$$

V is the volume of the reactor.

Where mass of the moderator, m_M is

$$m_M = \frac{V M_f}{N_a} \; (15)$$

$\mathbf{m_f}$ can be defined as:

$$m_f = Z \frac{\sigma_{am} M_f}{\sigma_{af} M_m} \mathrm{m_m} \; (16)$$

With $\mathbf{m_f}$, we can obtain atomic density N_f of the fuel by the formula:

$$N_f = \frac{N_a m f}{V} \; (17)$$

Where density, p of the fuel in reactor is,

$$p = \frac{mf}{V} \; (18)$$

Then macroscopic fission cross section, Σ_f can be computed by

$$\Sigma_f = N_f \sigma_f \quad (19)$$

The constant value of τ_T^2, L_T^2 and all the microscopic cross section of neutrons for moderators and fuels can be obtained at Evaluated Nuclear Data File **(ENDF)** [4].

1.11 Moderator and Fissile Material (Fuel Type)

Moderator is needed in the nuclear reactor for slowing down the fast neutron produced from fission reaction. The purpose of slowing down the neutron is for having larger microscopic cross section of scattering neutron for the moderator [6]. For this project, the moderator chosen is Light Water, Heavy Water, Graphite and Beryllium. These materials are selected due to their microscopic cross section of scattering is higher compared to other materials.

For creating fission reaction, fissile material is needed. Fissile material is the material that has high microscopic cross section of fission. There are only 3 types of nuclides that can undergo fission process which are ^{235}U, ^{233}U and ^{239}Pu [11]. However, these materials are difficult to obtain. ^{235}U are the isotope of natural uranium, ^{238}U. While ^{233}U and ^{239}Pu are the product of absorption of neutron by natural Thorium, ^{232}Th and natural Uranium respectively. In this project those materials are used as fuel in the reactor.

2.0 METHODOLOGY

2.1 Introduction

This chapter will clarify how to visualize neutron flux profile graph and generating its reading value. The solution in Table 2.1 is used as a governing equation for producing neutron flux graph and reading. All the steps on how to compute all the parameters in the equation will be shown in this chapter.

2.2 Governing Equation and Software

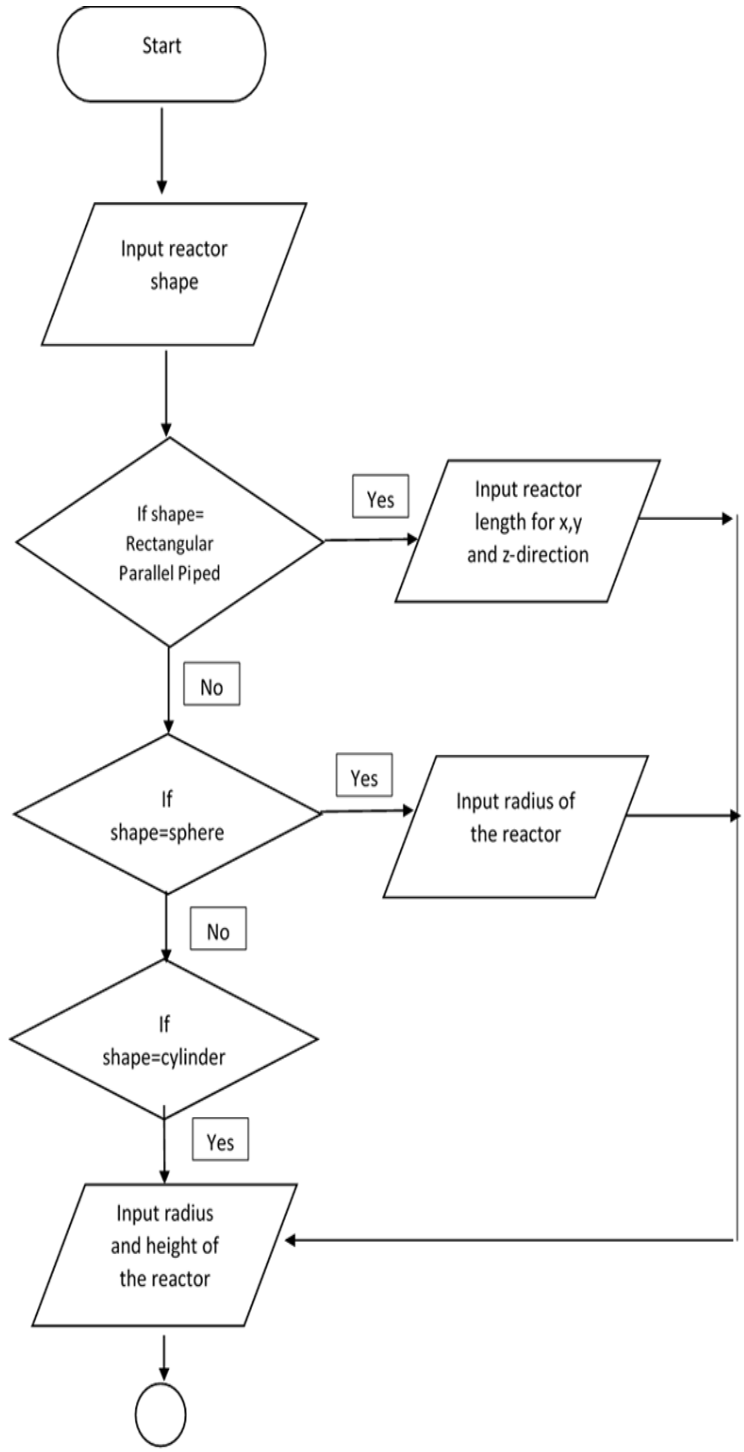
After observing Table 2.1, only Σ_f and V has to be computed to get the value of neutron flux, while the other parameters are constant. Value of V can be calculated based on the formula of volume for each shape respectively. Meanwhile, value of Σ_f can be retrieve by following the step of eq. (10) to eq. (20).

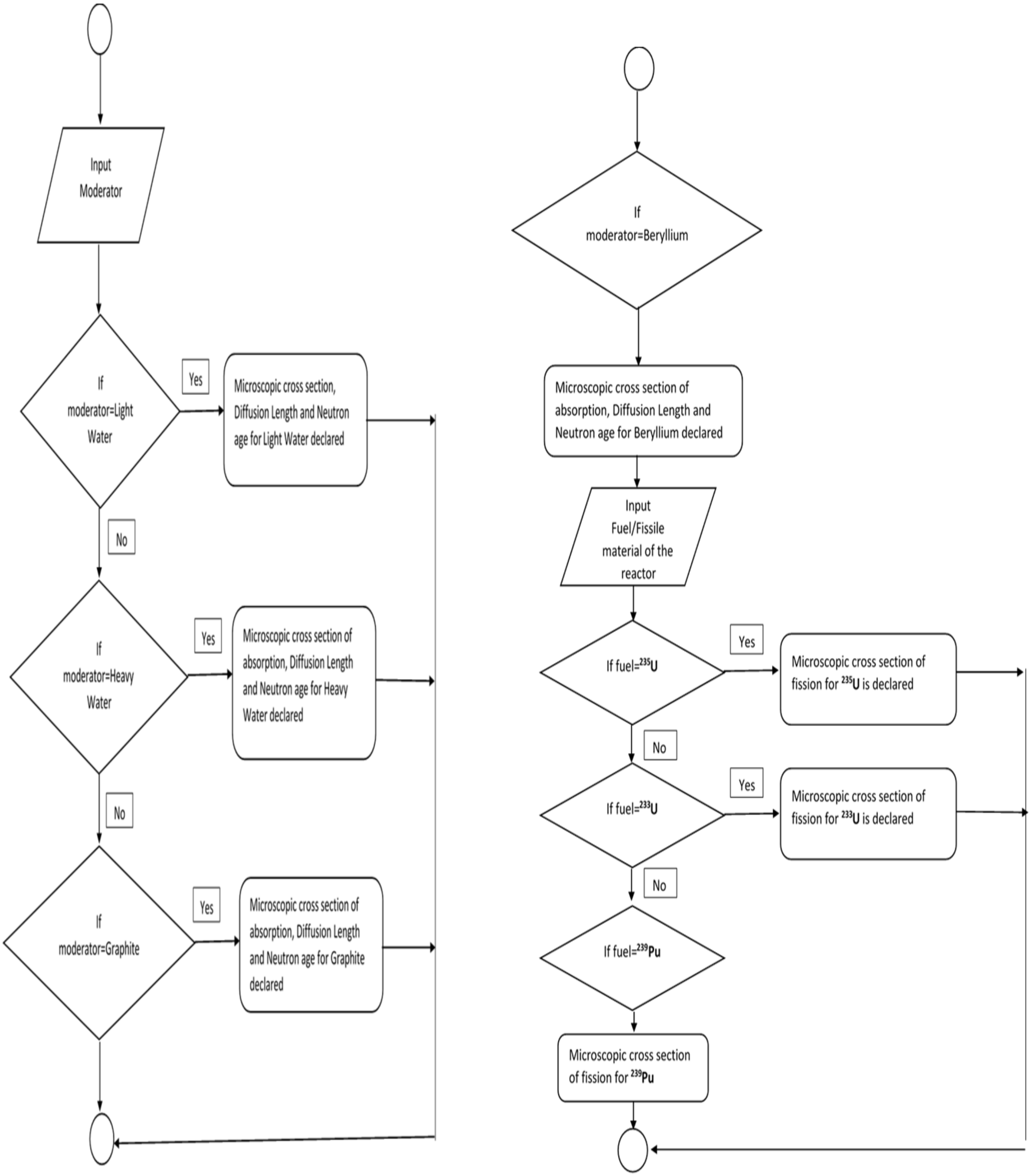
All the solution of neutron flux provided in Table 1 are the governing equation for this project. The equations will be converted to the computer codes for creating the simulation of neutron flux in thermal and homogeneous reactor. For this project, the open-source software which is Octave will be used.

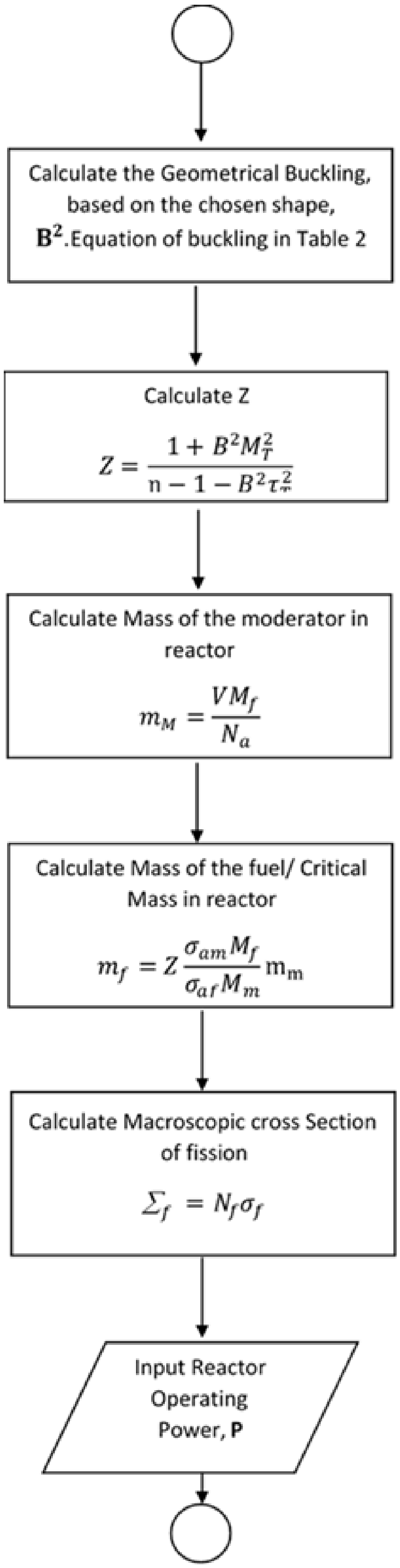
Moreover, SuperMC developed by FDS Team is used for determining neutron flux in 17 x 17 fuel assembly. This software used Monte-Carlo method to solve the problem. Monte Carlo simulation analyses flux by creating models of probable outcomes by substituting a range of values—a probability distribution—for any element with intrinsic uncertainty. It then repeats the calculation, each time using a different set of random values from the probability functions.

2.3 Flowchart for Computer Program

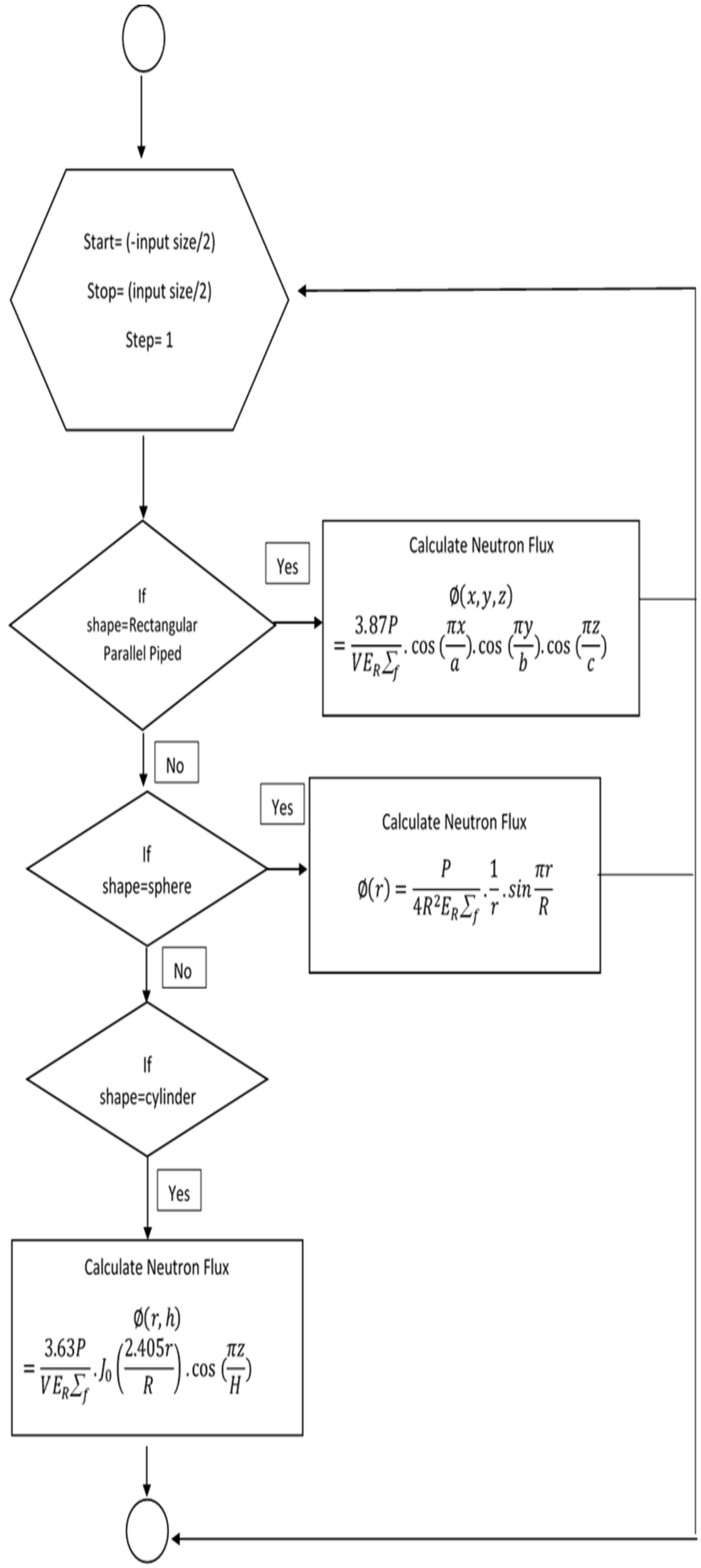
The user can select the reactor shape, moderator, fissile material and insert the operating power of the reactor. Each moderator and fuel have their own microscopic cross section and will be declared as user input request. Below is the flowchart of the computer program.



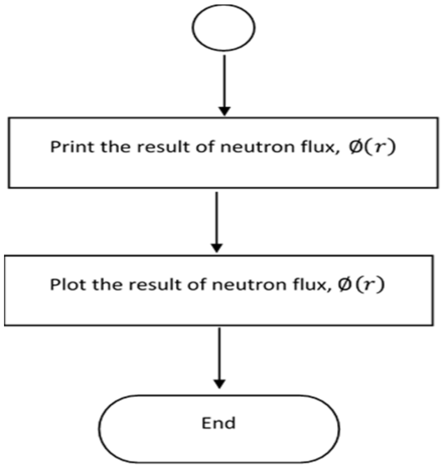

Input
Moderator
If
moderator=Light
Water
Yes
Microscopic cross section, Diffusion Length and Neutron age for Light Water declared
No
If
moderator=Heavy
Water
Yes
Microscopic cross section of absorption, Diffusion Length and Neutron age for Heavy Water declared
No
If
moderator=Graphite
Yes
Microscopic cross section of absorption, Diffusion Length and Neutron age for Graphite declared
If
moderator=Beryllium
Microscopic cross section of absorption, Diffusion Length and Neutron age for Beryllium declared
Input
Fuel/Fissile
material of the
reactor
If fuel=^{235}U
Yes
Microscopic cross section of fission for ^{235}U is declared
No
If fuel=^{233}U
Yes
Microscopic cross section of fission for ^{233}U is declared
No
If fuel=^{239}Pu
Microscopic cross section of fission for ^{239}Pu



Calculate the Geometrical Buckling, based on the chosen shape, $\mathbf{B^2}$.Equation of buckling in Table 2
Calculate Z
$Z = \frac{1 + B^2 M_T^2}{n - 1 - B^2 \tau_T^2}$
Calculate Mass of the moderator in reactor
$m_M = \frac{V M_f}{N_a}$
Calculate Mass of the fuel/ Critical Mass in reactor
$m_f = Z \frac{\sigma_{am} M_f}{\sigma_{af} M_m} m_m$
Calculate Macroscopic cross Section of fission
$\Sigma_f = N_f \sigma_f$
Input Reactor Operating Power, P



Start= (-input size/2)
Stop= (input size/2)
Step= 1
If shape=Rectangular Parallel Piped
Yes
Calculate Neutron Flux
$\emptyset(x,y,z) = \frac{3.87P}{VE_R\Sigma_f} . \cos\left(\frac{\pi x}{a}\right) . \cos\left(\frac{\pi y}{b}\right) . \cos\left(\frac{\pi z}{c}\right)$
No
If shape=sphere
Yes
Calculate Neutron Flux
$\emptyset(r) = \frac{P}{4R^2 E_R \Sigma_f} . \frac{1}{r} . \sin\frac{\pi r}{R}$
No
If shape=cylinder
Yes
Calculate Neutron Flux
$\emptyset(r,h) = \frac{3.63P}{VE_R\Sigma_f} . J_0\left(\frac{2.405r}{R}\right) . \cos\left(\frac{\pi z}{H}\right)$



2.4 Visualization of Neutron Flux Using SuperMC

To get better visualization of neutron flux in the reactor SuperMC software developed by FDS team is used. The software used is to determine the flux of the neutron in one fuel assembly. All the geometry and material assembly are based on SuperMC user manual [14].

Table 2.1 Design Parameters of Fuel Assembly

Design Parameters	
Fuel Assembly Design	17x17
Fuel Rod Length	20 cm
Fuel Rod Pitch	1.26 cm
Fuel Pallet Diameter	0.819 cm
Clad Tube Material	Steel 304
Clad Thickness	0.0654
Fuel Rod Fill Gas	He
Fuel Rod Fill Gas Gap	8.2x10-3cm
Water Tube Diameter	1.2348 cm
Fuel Pellet Material	UO2
Moderator	H2O

Geometry of Fuel Rod

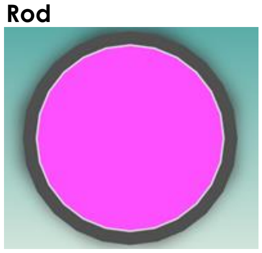


Figure 2.1 Top View of Fuel Rod

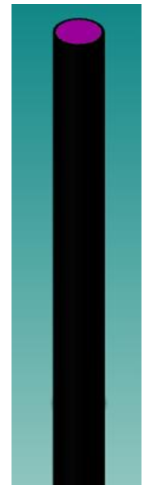


Figure 2.2 Isometric View of Fuel Rod

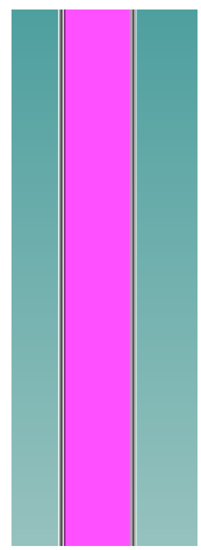


Figure 2.3 Cross-Section of Fuel Rod

Geometry of Water Tube

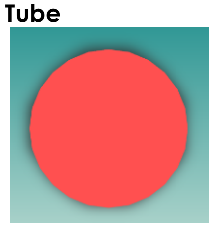


Figure 2.4 Water Tube Top View

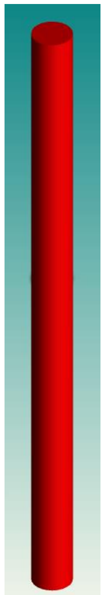


Figure 2.5 Water Tube Isometric View

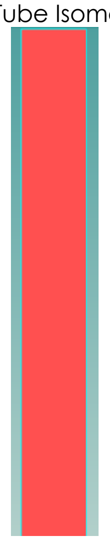


Figure 2.6 Cross-Section of Water Tube

Geometry of Fuel Assembly

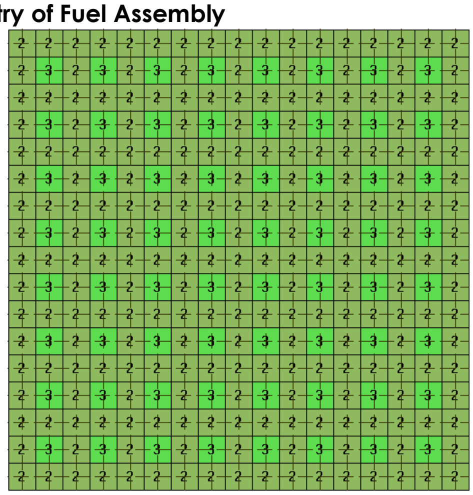


Figure 2.7 Fuel Lattice Design (17x17)

Universe 2 contains fuel rod while Universe 3 contains water tube.

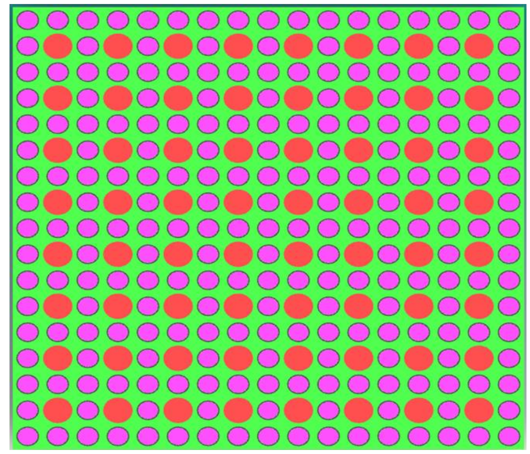


Figure 2.8 Top View of Fuel Assembly

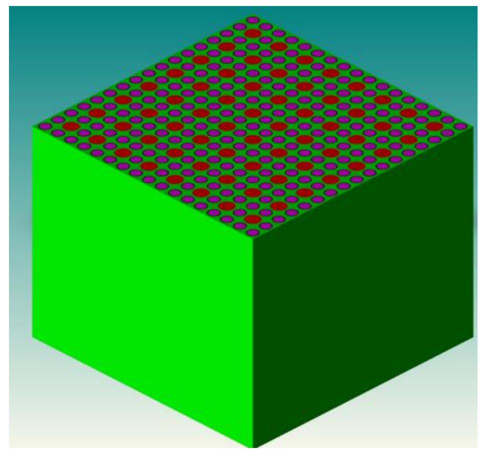


Figure 2.9 Isometric View of Fuel Assembly

Material For Geometry

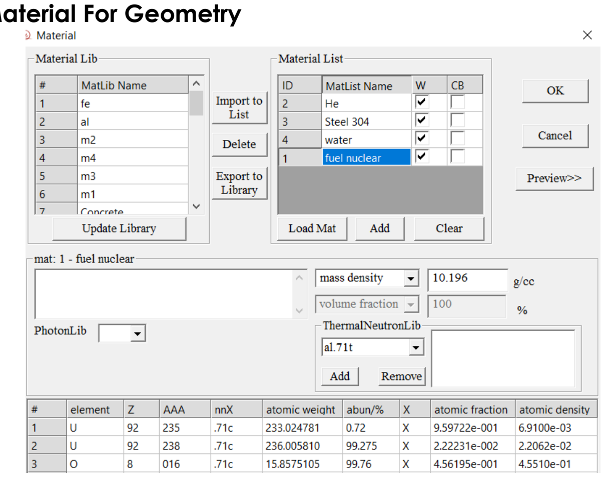


Figure 2.10 Material for Geometry

Based on Figure 2.10, the material declared is for each region in the geometry, He is used for the fill gas gap, Steel 304 is for cladding region while water is used as moderators. Lastly, the nuclear fuel as mentioned in the figure 2.10 is used at the fuel rod region.

3.0 RESULTS AND DISCUSSION

3.1 Introduction

In this chapter, the result will be discussed more detail by showing the graph that will generate after running the computer code and SuperMC software. Only cylinder shape reactor has two different types of graphs due to it has two directions with different function which is axial (cosine function) and radial direction (Bessel function). The other reactors will have similar type of graph although having more than 1 direction due to the function for neutron flux in each direction are same [3].

3.2 Result for Every Reactor Shape using Octave Computer Codes

Each reactor has different mathematical solution. For rectangular parallel-piped reactor, the function is cosine while sphere reactor consists of sinusoidal function. Finally, cylinder reactor consists of two function which are Bessel and Cosine function (Myers, 1960).

3.2.1 Graph for Rectangular Parallel-Piped Reactor Shape

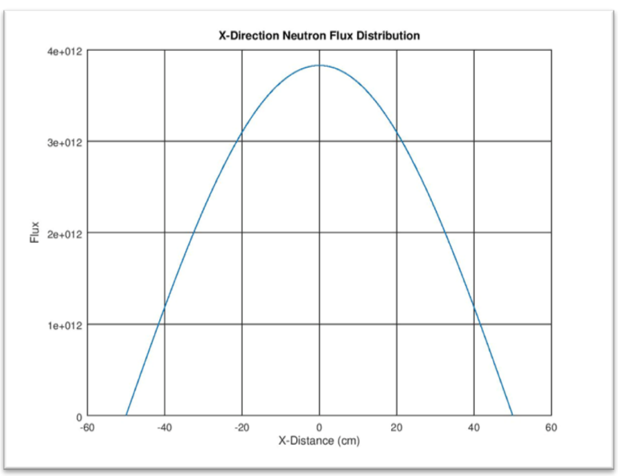


Figure 3.1 X-Direction Flux Distribution

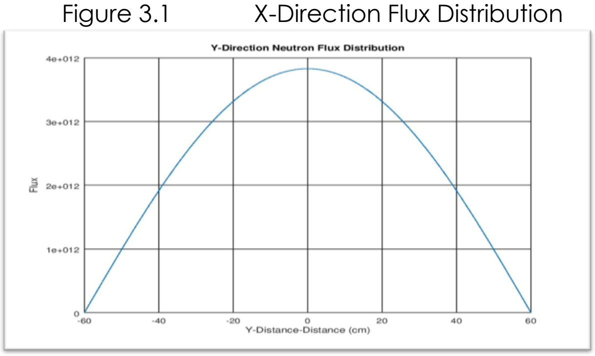


Figure 3.2 Y-Direction Flux Distribution

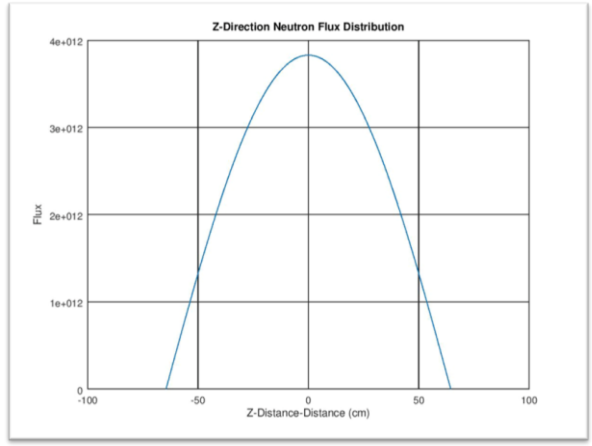


Figure 3.3 Z-Direction Flux Distribution

Figures 3.1,3.2 and 3.3 shows the cosine function of neutron flux in rectangular parallel piped reactor, the shape of the graph is same for every direction due to the similarity of the function. Length of X-direction is 100 cm, Y-direction is 120 cm and Z-direction is 129 cm.

3.2.2 Graph for Sphere Reactor Shape

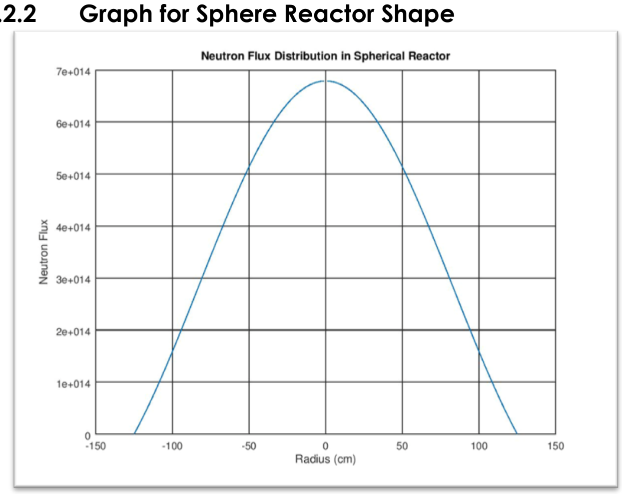


Figure 3.4 Neutron Flux Profile in Sphere Reactor

Figure 3.4 indicates the neutron flux profile in sphere reactor. The profile of the neutron flux is sinusoidal function. The length of the radius is 125 cm.

3.2.3 Graph for Cylinder Reactor shape

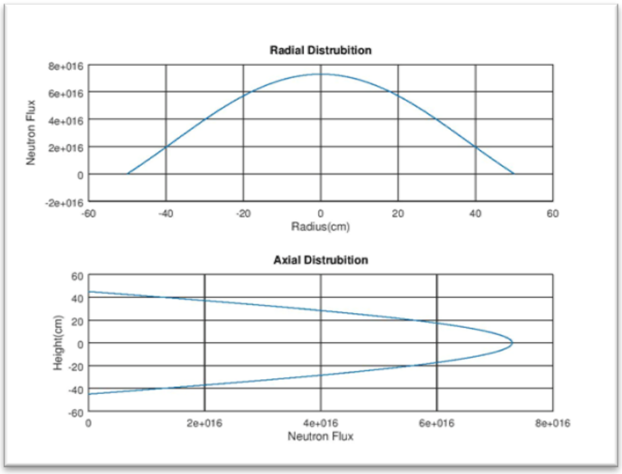


Figure 3.5 Radial and Axial Flux Distribution

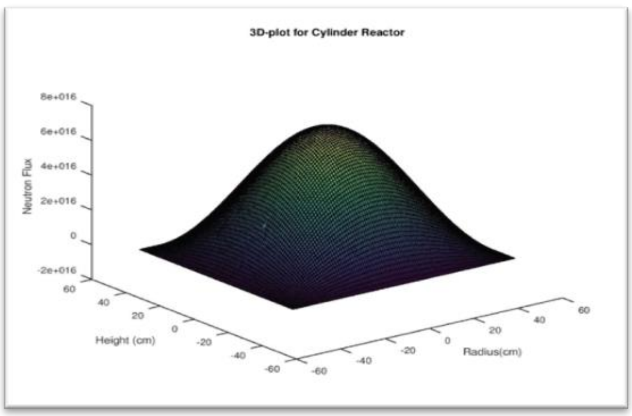


Figure 3.6 3-D Graph of Neutron Flux Distribution

Figure 3.5 and 3.6 visualize the neutron flux profile in cylinder reactor. There are 2 profiles existing in the reactor. In r-direction or radial direction the function profile is Bessel function. For z-direction or axial direction, the profile function is cosine function.

3.3 Result for 17x17 Fuel Assembly using SuperMC

3.3.1 Tallied Region

To visualize the neutron flux in the fuel assembly, tally must be set. The tally set as a mesh and the geometry is stated in Figure 3.7.

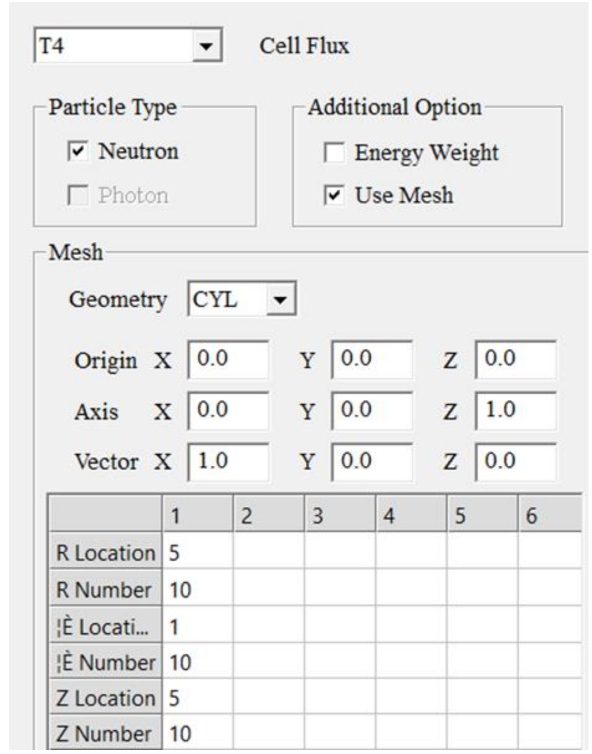


Figure 3.7 Geometry of Tally

To get more accurate result the length of tallied location and number must be set as low as possible. So, the length of tallied radius is 5 cm and we set the number radius at that location 10. While, the tallied angle of rotation is set to 1 means 360° and total number of the rotation is set to 1. Lastly, the tallied height is set to 5 cm and 10 numbers of height is set to be tallied. Figure 3.8 shows the tallied region (square border) in the fuel assembly.

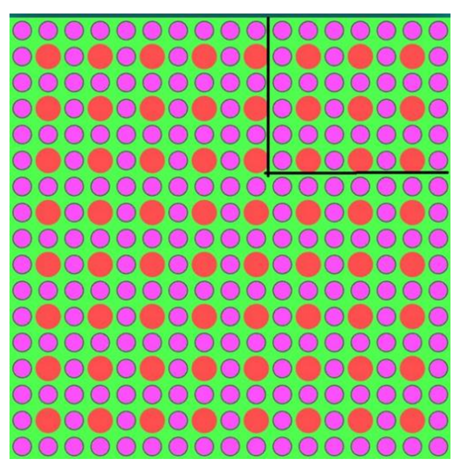


Figure 3.8 Top View of Tallied Region

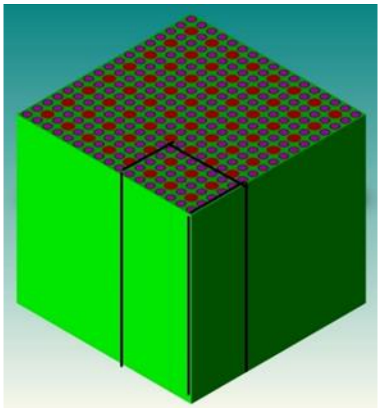


Figure 3.9 Isometric View of Tallied Region

Figure 3.8 and 3.9 indicate the location of tallied region in 17x17 fuel assembly. The location of the tallied region is based on Figure 4.7. All the calculation of flux is carried out on the mark region for getting the better visualization and more distinguish result. If the calculation is carried out for the whole geometry, the visualization for neutron flux is not as defined as tallied calculation.

3.3.2 Flux Visualization in Tallied Region

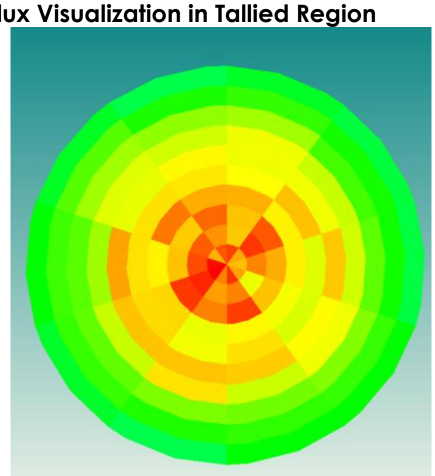


Figure 3.10 Top View of Flux

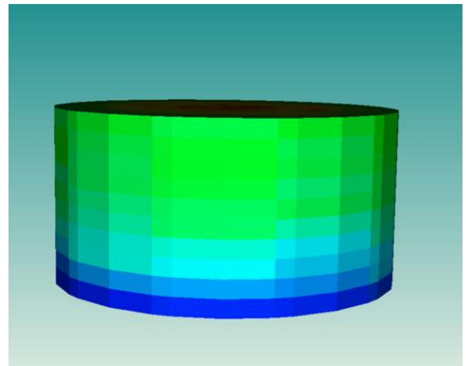


Figure 3.11 Isometric View of Flux

Based on Figure 3.10 and 3.11, the 3-D graph for neutron flux in tallied region is visualized. Based on the figures, The brightest colour (red) region indicates the highest population of neutron flux. The center has the highest population flux and the value of flux is decreasing. To get more detail explanation the 3-D graph is separated to 2-D graph for each axis.

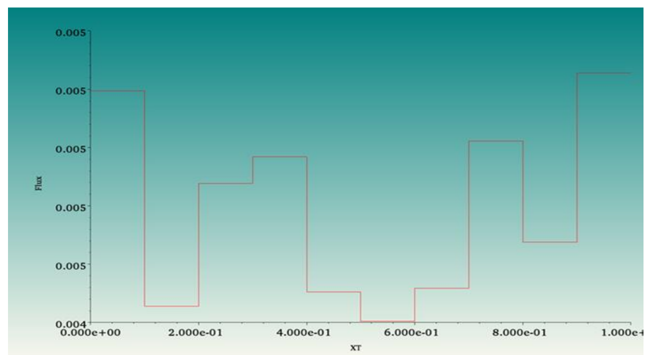


Figure 3.12 XT (angle of rotation) direction flux

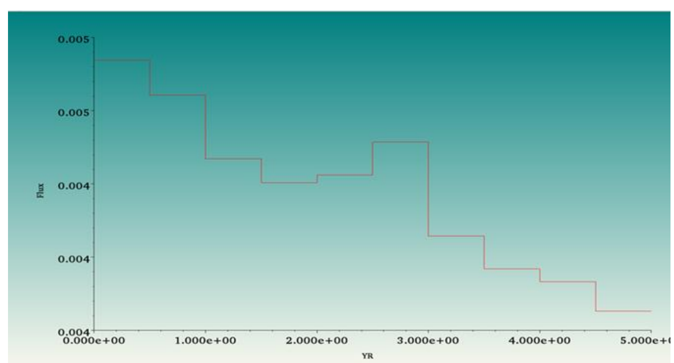


Figure 3.13 Y/R direction flux

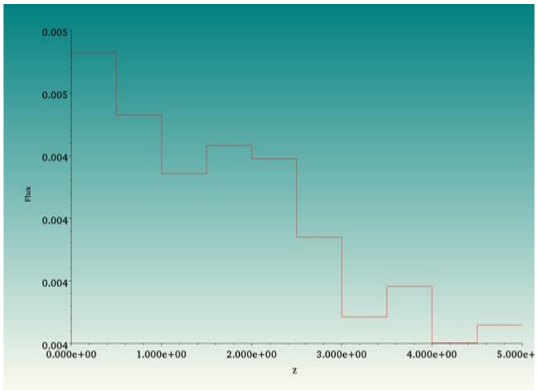


Figure 3.14 Z-Direction Flux

Based on the result of figures 3.12, 3.13 and 3.14 there are several peaks of flux along the distance for each axis. The center for each axis has the highest value of flux and the value is decreasing while the distance is increasing. However, the value of flux is not constantly decreasing like the result produced by Octave computer codes. There are some locations where the flux is higher to the nearby location although the flux is not as higher as the flux in center. The peak in the graph indicates the location of fuel rod while the lower value of flux indicates the location of cladding.

The difference of flux produced by Octave computer's code (Homogenous Reactor) and SuperMC software (Heterogenous Reactor) is the different number of peaks. In homogenous reactor there are only single peak while in heterogenous there are more than one peaks. In homogenous reactor, the flux profile is determined by a single mathematical equation and the value of single energy constant (neutron cross section) is used [7]. Furthermore, the number of fuel rod in homogenous reactor is considered as one without any cladding material. Meanwhile in heterogenous reactor, there are several fuel rods with cladding materials and each of them is calculated separately by neutron diffusion equation using their own respective geometry before merging the results with other fuel rods.

4.0 CONCLUSION

In a nutshell, the project is highlighting the visualization of neutron flux profile in thermal and homogeneous reactor. Each reactor has different function of the neutron flux profile as observed in Chapter 4, the profile is quite similar to each other. The homogenous reactor means we assumed the core as a source and mono energy value of neutron cross-section is used for every material [1]. The computer programming software (Octave) is utilized in this project for creating the user interface to observe the neutron flux pattern in the reactor.

This project will help the students who are taking Nuclear Reactor Theory subject to visualize the neutron flux in homogeneous reactor. The computer code will be given to the assigned lecturer and he/she can utilize this program to show the students what are happening in homogenous reactor.

To get the visualization of neutron flux profile in heterogenous reactor, SuperMC software is used developed by FDS Team in Heifei Institute. All the geometry and materials in the fuel assembly is based on SuperMC manual user. The results obtained indicate this software can be used to analyze the distribution of neutron flux in the reactor.

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