Nuclear Reactors

A Discussion of Neutron Flux, Power Density, and Gamma Ray Buildup Flux

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Abstract

This project examines basic concepts and calculations of nuclear engineering and attempts to give a basic background of nuclear reactors. The scope extends only to the inner workings of the nuclear reactor itself. The power producing components such as the steam generator and condenser will not be discussed. Concepts, definitions, and calculations were obtained from *Introduction to Nuclear Engineering* by Lamarsh and Barratta. Guidelines for the basic setup of nuclear reactor are based on a Westinghouse AP1000 nuclear reactor. This project discusses several key points: neutron flux, heat flux, power, gamma ray emission, and shielding of both neutron emission and gamma ray emission. Base information, such as decay modes, cross sections, and types of interaction between particles and a nucleus, regarding each topic, the problems encountered, and limitations regarding calculations will be examined.

Introduction

This project aims to answer several questions: 1) what is the neutron flux produced in active reactor with a certain setup and fuel, 2) what is the thermal power of the reactor, 3) what power density is required to produce to produce the thermal power, 4) given a neutron flux, what is the gamma ray flux, and 5) given a gamma ray flux, how much shielding would make the system tolerable?

To answer these questions I began with a known reactor: the Westinghouse AP1000 nuclear reactor. Regarding neutron flux in particular and several other parameters can be calculated with a computer program called SCALE. However, the cost and training demands deemed this approach unsuitable for this project. Therefore all calculations were done either manually or with a computational program like Mathematica. However, before these calculations will be answered, basic information regarding atomic structure, cross sections, decay modes, gamma ray and neutron interactions will be discussed.

Background

The neutron and photon is extremely important to nuclear engineering. Photons and individual, unbound neutrons can be thought of products of some nuclear reactions within a reactor given some conditions (Lamarsh and Baratta, 5-7). Both products are involved in several kinds of interactions with the atom.

There are several decay modes an atom can go through to reach a combination of protons and neutrons that is stable. The first decay mode is alpha decay. In alpha decay an atom ejects a single particle composed of two protons and two neutrons. This particle is called the alpha particle (it can also be called a helium atom). The second decay mode is beta decay. Beta decay contains two types of decay: beta positive and beta negative. In beta positive decay, a proton is converted into a neutron, positron, and an neutrino. In beta negative decay, a neutron is converted into a proton, electron, and an antineutrino. Some very heavy elements such as uranium go through spontaneous fission: unstable atom will split, unevenly into two other atoms plus photons of varying energies (Lamarsh and Baratta, 18-22).

Fission releases on average for U235 2.42 neutron per fission. As a free neutron, created from spontaneous fission for example, approaches and strikes a nucleus it forms a compound nucleus. From this compound nucleus several different things can happen: the neutron can scatter elastically, the neutron can scatter inelastically, the neutron can simply be absorbed into the nucleus which will release a gamma ray from the nucleus, or the nucleus will undergo fission from the added energy given to it by the neutron (Lamarsh and Baratta, 52-53). Each process is weighted depending on the incident neutrons energy and depending on what the target nucleus is e.g. uranium and carbon will cause the incident neutron to behave different once it has formed a compound nucleus. These weights are called cross sections which are in units of area. Cross sections can be described as the probability that a neutron interacts with the target nucleus in some way e.g. scattering, absorption, and fission an effective area that the incident neutron sees. The higher the cross section of a nucleus, the more area an incident neutron sees. Each kind of interaction e.g. inelastic, elastic, absorption, and fission has its own particular cross section (Lamarsh and Baratta, 54-57). The reader may note that thermal neutrons are generally defined as having an energy of about 0.025eV.

Fission for nuclear engineering is the most important of all of the processes. Fission occurs generally for elements that are at least above iron on the periodic table. Fission starts when an incident neutron, that may or may not have energy, strikes a nucleus. The two particles form a compound nucleus which then immediately splits or undergoes fission. The particles will undergo some sort of decay to become stable (Lamarsh and Baratta, 74-76). There are two types of elements that can undergo fission: fissile elements and fissionable elements. Fissile elements (U235)only require an incident neutron to strike because the binding energy the neutron lets off is enough for the atom to undergo fission. Fissionable elements (U238) require the incident neutron to carry some energy for the target nucleus to undergo fission (Lamarsh and Baratta, 77-78). The result of fission is two products, gamma rays, antineutrinos, and neutrons which total about 200MeV (recoverable energy). All of the energy released in fission can be recovered except the energy contained in the antineutrinos (Lamarsh and Baratta, 88-89).

The neutrons released as a result of fission will interact with more nuclei that have the possibility to under fission which will release more neutrons to interact with more nuclei. One can measure how many neutrons are in the system. This measurement is called neutron flux.

Neutron flux is measured from a point in the system (Lamarsh and Baratta, 60-62). Assuming one is working in a cylindrical system, neutron flux is depending on several different things. The most obvious being where the point is in the system, height with respect to the fuel rod and the radial distance away from the rod. The power of the system is also a variable in the flux in the sense that power is attributed to fission as seen above and the fission releases neutrons (Lamarsh and Baratta, 279-280).

The neutron flux does not create electrical energy we use to power our houses. The neutrons, fission products, and gamma rays interact with the entire reactor itself, maybe elastically, to raise the temperature of the reactor as a whole. A pipe usually filled with some mixture of water and other chemicals is built into the reactor to take away some of the heat that is generated. This water, or the coolant, is pumped into a steam generator, which turns a separate system of water into system which then creates electricity. The heated water cools slightly and is then sent back into the reactor as a constant loop. The heat generated by the reactor is related directly to the neutron flux. The amount of heat generated by the reactor is very important in determining the power output and it is therefore called the heat flux. The power density is a good measurement to determine how the heat is spread throughout the reactor (Lamarsh and Baratta, 410-412).

Gamma rays behave in a similar way neutrons do. Gamma rays are not only a result of fission directly but the fission products do emit gamma rays to some degree. Gamma rays do have a calculated flux to go along with them. The first kind of interaction gamma rays undergo with matter is the photoelectric effect. In this interaction a gamma ray interacts with an electron in an electron cloud of an atom. The electron is then ejected and carries the energy of the gamma ray minus the binding energy. The second kind of interaction is pair production. The gamma ray of a certain energy creates a positron and a negatron. The gamma ray must have an energy equal to or greater than the energy of the rest energy of the negatron and positron. The two will lose energy through collisions and eventually collide back together resulting in two x-rays. The last interaction is Compton scattering. This is inelastic scattering. The gamma ray continues to travel (Lamarsh and Baratta, 90-100).

As a result of gamma rays going through Compton scattering, the gamma flux is not a simple exponential decay. The proper calculation to use is called the buildup flux which contains the regular uncollided flux but contains a correction term called the buildup factor. The buildup factor directly corrects for the Compton scattering effect and varies with the material that is used to shield with and the energy of the gamma ray (Lamarsh and Baratta, 548-557). Since the object producing gamma rays is a cylindrical shape, the position in which the gamma ray flux is measured at should be very important (Lamarsh and Baratta, 566-568)

Methodology

This project based calculations off a Westinghouse AP1000 reactor (specifications can be found on the National Regulatory Commission's website). There are 157 fuel assemblies in this setup arranged in an approximate circle. See figure 1-1. All of the fuel assemblies have a length of 388.1 cm and a diameter of 0.820 cm. The reactor radius is 152.019 cm. Each fuel assembly contains a 17x17 setup of fuel rods. There are 264 fuel rod slots the remaining 25 slots are used for guide rods and electronics. See figure 1-2. The fuel rods are made of a uranium dioxide alloy. There are 41,448 fuel rods in the total reactor setup. The power generated by this reactor is 3400MW (NRC, 44-47). This power is heat generated.

A good reactor fuel will have a very high fission cross section. The enrichment of this reactor is 5% Uranium 235 and 95% Uranium 238. The fission cross section for uranium 238 is almost negligible. The fission cross section for U235 is extremely high: 587 barns. U235 has a very low abundance when found naturally, an enrichment process is very difficult to do and, therefore, very expensive. So, a U235 enrichment of 5% gives a good fission cross section for the fuel, and is very cost efficient. Most reactors undergo refueling every 2 years.

Neutron Flux

Neutron flux is a solution to the diffusion equation (Lamarsh and Baratta, 237). The diffusion equation is the differential equation

$$\nabla^2\phi - \frac{1}{L^2}\phi = \frac{s}{D} \tag{1}$$

where del squared is the Laplacian, in this case in cylindrical coordinates as our reactor is arranged in a cylindrical manner, L is the diffusion length, s is the neutron emissions from sources per cubic centimeter, and D is the diffusion coefficient. The solution to the diffusion equation yields, the neutron flux is given by

$$\phi(r,z) = AJ_0\left(\frac{2.405r}{R+d}\right)\cos\left(\frac{\pi z}{H+d}\right) \tag{2}$$

where d is the extrapolation distance from the surface of the reactor (Lamarsh and Baratta, 280). J_0 is the zeroth order Bessel function. R and H are the radius and height of the reactor, respectively. The origin is set to the center of the system. See figure 2-1. A is a constant

$$A = \frac{3.63 * P}{VE_r\Sigma_f} \tag{3}$$

P is the thermal power. V is the total volume of the reactor. E_r is the total energy recoverable per fission. Σ_f is the total fission macroscopic cross section. The reader may note that the diffusion equation is technically invalid if calculated outside the reactor. The extrapolation distance corrects for this. Units are given in neutrons per second per square centimeter.

Power

Power is proportional the neutron flux of the system and therefore cannot be calculated without already knowing the neutron flux.

Power Density

The power density, q''', is the rate of heat energy production per unit volume. As all neutrons can be classified as thermal neutrons, power density is

$$q'''(r) = E_d \overline{\Sigma}_{fr} \phi_T(r) \tag{4}$$

E_d is energy deposited locally (Lamarsh and Baratta, 410). Where the energy deposited locally accounts for the fission products, gamma ray energy, and beta ray energy but does not include the energy deposited in the coolant (Lamarsh and Barratta, 409). Σ_{fr} is the total macroscopic fission cross section per fuel rod. ϕ_T is the thermal neutron flux which is equal to the neutron flux above. The Σ_f term in thermal neutron flux can be rewritten in terms of Σ_{fr}

$$\underline{\Sigma}_f = \frac{\underline{\Sigma}_{fr} n\pi a^2 H}{\pi R^2 H} = \frac{\underline{\Sigma}_{fr} n a^2}{R^2} \tag{5}$$

n is the number of rods, A is the individual rod size, H is height of the rod, R is the radius of the entire reactor. When Σ_f in (2) is replaced (5), the result is

$$\phi_T(r, z) = \frac{3.63PR^2}{E_R \underline{\Sigma}_{fr} V a^2 n} J_0\left(\frac{2.405r}{R + d}\right) \cos\left(\frac{\pi z}{H + d}\right) \tag{6}$$

where V is the volume of the reactor

$$V = \pi R^2 H \tag{7}$$

therefore,

$$\phi_T(r, z) = \frac{3.63P}{E_R \underline{\Sigma}_{fr} \pi H a^2 n} J_0\left(\frac{2.405r}{R + d}\right) \cos\left(\frac{\pi z}{H + d}\right) \tag{8}$$

Putting (8) into (4) we finally get,

$$q'''(r,z) = \frac{1.16PE_d}{Ha^2nE_R} J_0(\frac{2.405r}{R+d})\cos(\frac{\pi z}{H+d}) \quad (9)$$

Note that, d the extrapolation distance also comes into effect here. Units are given as kilowatts per liter.

Shielded Gamma Ray Flux

To safely shield from the gamma ray flux the buildup flux equation is used. This solution comes in two parts depending on if the place of measure is from the top and or the sides. Measurements are taken directly on the surface of the shield. The gamma ray source can be assumed to be a line. The units used are gamma rays per second per square centimeter (Lamarsh and Baratta, 569).

$$\phi_b = \frac{S}{4\pi R}\sum A_n\{F[\theta_1, (1+\alpha_n)\mu R] + F[\theta_2, (1+\alpha_n)\mu R]\} \quad (10)$$

See figure 3-1. Or if θ_n is in terms of length and the equation is expanded to account for A_1 and A_2, then

$$\begin{aligned}\phi_b = \frac{S}{4\pi R}\{&A_1F[\tan^{-1}\frac{H+z}{R}, (1+\alpha_1)\mu R] + A_2F[\tan^{-1}\frac{H+z}{R}, (1+\alpha_2)\mu R] \\ &+ A_1F[\tan^{-1}\frac{H-z}{R}, (1+\alpha_1)\mu R] + A_2F[\tan^{-1}\frac{H-z}{R}, (1+\alpha_2)\mu R]\}\end{aligned} \quad (11)$$

Where S is equal to the number of gamma rays emitted per second, R is the shield thickness of the reactor, $A_n = A_1 + A_2 = 1$, F is a Sievert function with arguments θ_1 and θ_2 which measure the angle between the point to measure and the bottom and top of the source respectively, α_n is a function of the shielding material (the n refers to the same n on the A term. So when multiplied together A and α will have the same subscript), and μ is a function of the shielding material atom density.

Results

Neutron Flux

The values listed below are calculated in Mathematica. See appendix for more values. We must first initial several conditions that will be used throughout the paper.

1*	Number of Assemblies	157
2*	Number of Rods Per Assembly	17*17-25=264
3*	Total Number of Rods (2)*(1)	41,448
4*	Reactor Height (cm)	388.1
5*	Reactor Radius (cm)	152.019
6*	Reactor Volume (cm^3)	2.8176*10^7
7*	Reactor Power (MWh)	3400
8	Power Per Rod (Wh) (7)/(3)	82030.5
9	Energy Per Fission (W)	3.204*10^-11
10	Fissions Per Second (7)/(9)	1.06117*10^20
11*	Rod Radius (cm)	0.374
12*	Rod Volume (cm^3)	170.544
13	Macroscopic Fission Cross Section (barns)	1.40689

*NRC see references

To calculate the neutron flux we break up the equation (2) to make it easier to calculate. The first step is to calculate (3)

$$A = \frac{3.63 * \text{ Power Per Rod}}{(\text{Reactor Volume}) * (\text{Energy per fission}) * (\text{Macrscopic Fission Cross Section})} = 3.87 * 10^{13} \tag{12}$$

Therefore the flux (2)

$$\phi(r, z) = AJ_0\left(\frac{2.405r}{\text{Rod Radius} + d}\right)\cos\left(\frac{\pi z}{\text{Reactor Height} + d}\right) \tag{13}$$

Note since the diffusion equation is invalid beyond the source, so is its solutions. Therefore the extrapolation distance, d, is equal to a constant multiplied by D which is the diffusion coefficient for water

$$d = D * 2.13 = 0.16 * 2.13 = 0.3408\text{cm} \tag{14}$$

Since there is no way to know the flux of the reactor as a whole, an approximation is given. Each rod in the reactor has a flux listed above. Since there are 41,448 rods this number is multiplied to the individual rod's flux. Total flux is denoted by

$$\phi_{Tot}(r, z) = \phi(r, z) * 41{,}448 \tag{15}$$

A robust answer to this equation is given in the Mathematica program. An example would be combining (12), (13), and (14) into (15)

$$\phi_{Tot}(\text{RodRadius}, 0) = 1.03011 * 10^{18}\text{neutrons per cm}^2\text{per second} \tag{16}$$

This total flux is evaluated at each fuel rod's surface at half of its height then multiplied by the number of rods. Several more graphs of this value can be found in the appendix.

Power Density

The power density (9) is based off the neutron flux but the constants in the front of the equation are different.

$$q'''(r,z) = \frac{1.16 * (\text{Reactor Power}) * (\text{Energy Deposited})}{(\text{Reactor Height}) * (\text{Rod Radius})^2 * (\text{Number of Rods}) * (\text{Energy Recoverable})} J_0\left(\frac{2.405r}{\text{Reactor Radius} + d}\right)\cos\left(\frac{\pi z}{\text{Reactor Height} + d}\right) \quad (17)$$

The extrapolation distance d is the same as in the neutron flux. Note that this function does not have to be calculated per fuel rod. It does take into account the number of rods. When (17) is evaluated at the origin of the system

$$q'''(0,0) = 1577.75\text{MWh/liter} \quad (18)$$

Several more graphs of this value can be found in the appendix.

Gamma Ray Buildup Flux

The gamma ray buildup flux is rather complicated. The shield is assumed to be lead approximately 1.76 cm. One gamma ray is assumed to be release for every fission at an energy of 1 MeV. The mass attenuation coefficient, which relates an energy of a gamma ray to how it is absorbed, is 0.0684. The density of lead is 11.34 grams per cubic centimeter. Therefore the attenuation coefficient is equal to

$$(\text{Mass Attenuation Coefficient}) * (\text{Density of Lead}) = 0.775656\text{cm}^{-1} \quad (19)$$

The buildup factor is 1.37 at this energy level for lead. If we divide the buildup factor by the attenuation coefficient we can verify the shield radius. A_1 and A_2 are also determined by the energy of the gamma ray which are 2.84 and $A_2 = 1-A_1=-1.84$. α_1 and α_2 are determined once again by the gamma ray energy. α_1 is -0.03503 and α_2 is 0.13486.

The buildup flux requires the gamma ray to be in terms of per unit area. Therefore we must divide the production of gamma rays by the reactor area

$$\frac{\text{Fissions Per Second}}{(2 * \text{Pi} * (\text{Reactor Radius})^2 + 2 * \text{Pi} * (\text{Reactor Radius}) * (\text{Reactor Height}))} = 2.05693 * 10^{14}\text{Gamma Rays per cm}^2\text{per second} \quad (20)$$

θ_1 and θ_2 are defined as follows

$$\theta_1 = \tan^{-1}\frac{H+z}{R}$$
$$\theta_2 = \tan^{-1}\frac{H-z}{R} \quad (21)$$

Where R is the radius of the reactor and z is the height measured from the middle. Four Sievert functions are found within the buildup flux. It is in terms of

$$\int_0^{\theta} \gamma\, e^{\sec x}\, dx = F(\theta, \gamma) \quad (22)$$

So when combined with (20) and (22) is

$$\phi_b = \frac{\text{Emitted Gamma Per Area Per Second}}{4\pi \text{Reactor Radius}}\{A_1F[\theta_1, (1+\alpha_1)\text{BuildupFactor}] + A_2F[\theta_1, (1+\alpha_2)\text{BuildupFactor}] + A_1F[\theta_2, (1+\alpha_1)\text{BuildupFactor}] + A_2F[\theta_2, (1+\alpha_2)\text{BuildupFactor}]\} \quad (23)$$

Note that the buildup flux is not a function of the height necessarily. It is only meant to give a rough estimate of the thickness of a shield. In this case when shielding from 1MeV gamma rays with a lead shield of thickness 1.76 cm, the buildup factor is approximately $5.90734*10^{12}$ gamma rays per square centimeter per second which is reduced from the original value. This value is the same for all values of height.

Discussion of Results

Most of the values that are calculated in Lamarsh's book are very close approximations because fuel rod interaction is very complex. The values calculated have several limitations associated along with them. Fick's law which is used in the diffusion equation is not valid outside a neutron source. Therefore, the diffusion equation and its solutions are also invalid outside of the source. As stated prior, the extrapolation distance, d, corrects for this because otherwise the Bessel Functions causes a zero or negative flux which is according to Lamarsh, nonphysical and unreal. The second limitation for neutron flux is the fact that the neutron interaction does not occur between two rods, but in a working reactor it does. The total neutron flux calculated above would be higher had the neutrons been able to interact with one another to produce more fission reactors thus creating more neutrons. The total neutron flux calculated seems to be reasonable.

Though the power density is calculated using some of the same terms as neutron flux, it did not have to be calculated per rod but its only limitation is once again the extrapolation distance which is corrected for as the neutron flux. The values calculated at the center of the reactor, radial distance and height are zero, seem reasonable since the total power is 3400MWh. At the edge of the reactor the power density is small compared to the center.

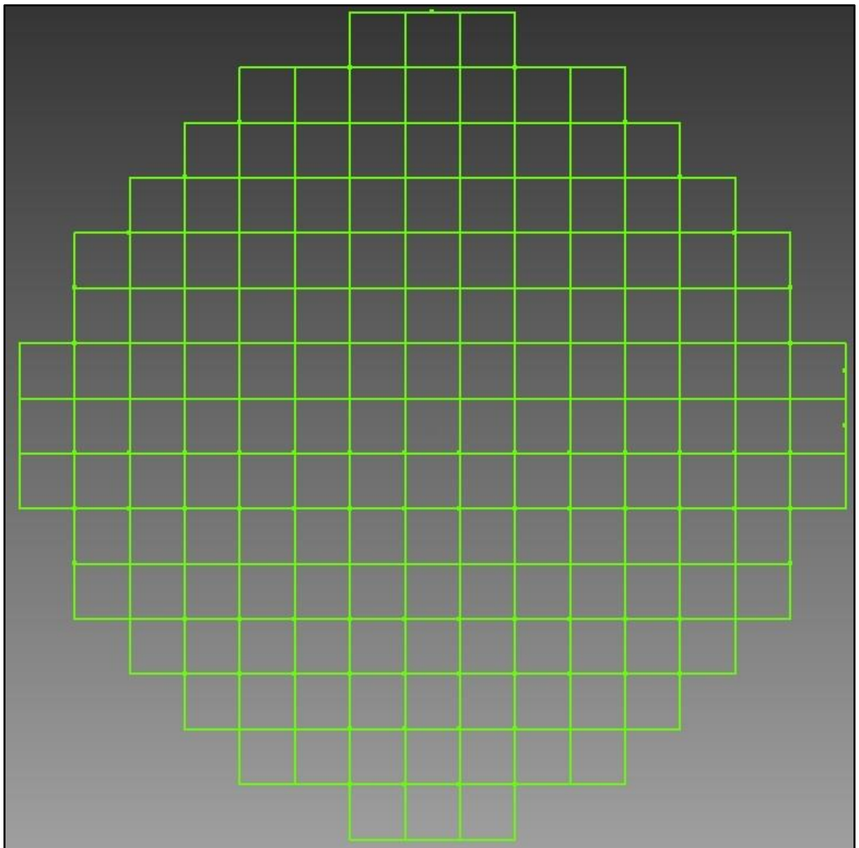
The gamma ray flux was a much more difficult procedure. Several of the terms depended on the energy of the gamma ray. Lamarsh only gave a few terms to work from. For example, to shield a gamma ray with an energy of 1MeV with lead the mass attenuation coefficient, A_n, and α_n, are all dependent on the gamma ray flux. However, only a few of these values are easily found in Lamarsh's text. Therefore only a few different shielding radii can be found. This is not a huge problem but the texts quoted seem to be no longer in print. The other main problem with the buildup flux is that it does not seem to be a function of the height. There are four Sievert integrals in the buildup flux equation that is examined. If the height changes, the variables correct themselves to equal to final value no matter the height. In that respect, it is hard to say that the buildup flux is a function of height. It does however give a good approximation of what the resultant gamma flux is after shielding. The attenuation coefficient and other constants that are dependent on the gamma ray energy assume that the gamma rays are monoenergetic. Fission does not release monoenergetic gamma ray fluxes meaning that this value is not necessarily correct.

Conclusion

I believe this project produced logical and feasible results. The neutron flux of the reactor reached a maximum of 10^{18} neutrons per square cm. The power density of the reactor reached a maximum of 1500kW per cubic centimeter. The gamma ray buildup flux is reduced by a factor of ten with two cm of lead shielding. A large piece of the study was devoted to understanding the core concepts behind reactors. If more time and money were provided a program called SCALE would be worth looking into as it does many of these complex calculations and provides more accurate answers. However, the key concepts: what is happening inside a reactor, why certain fuels are used, how they are used, neutron interactions, fission, and shielding are still understood which regarding the time I had to work on this project is the most important.

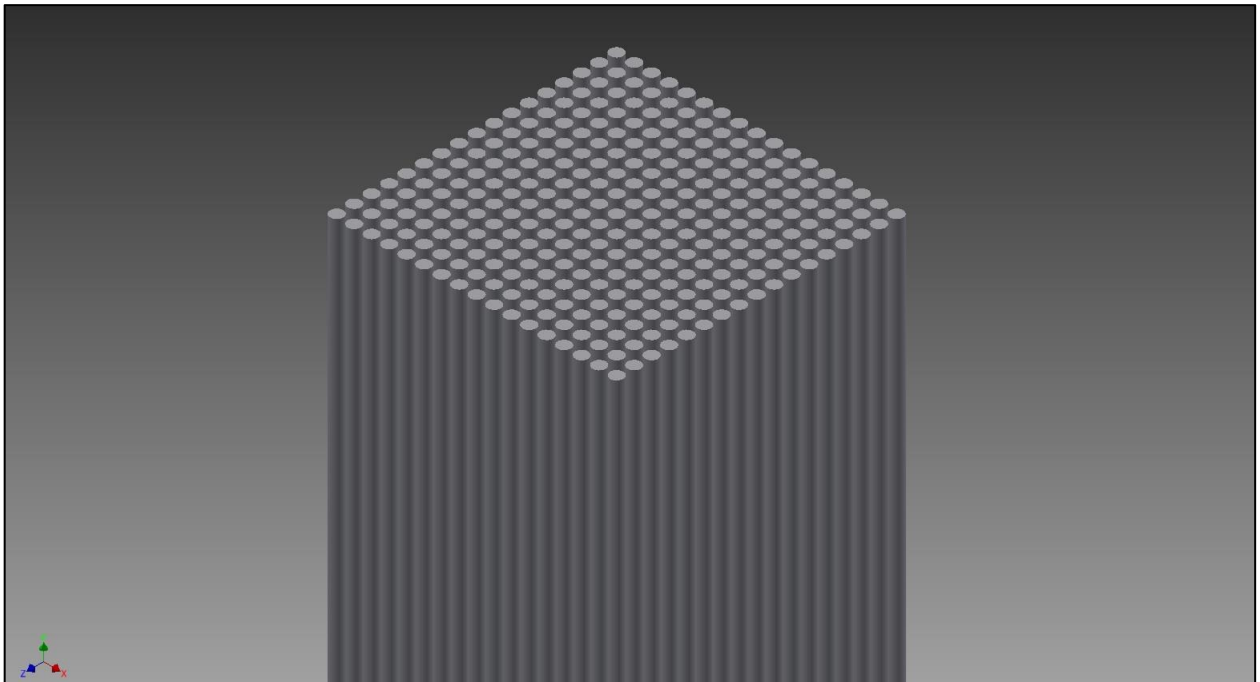
Appendix

Figure 1-1



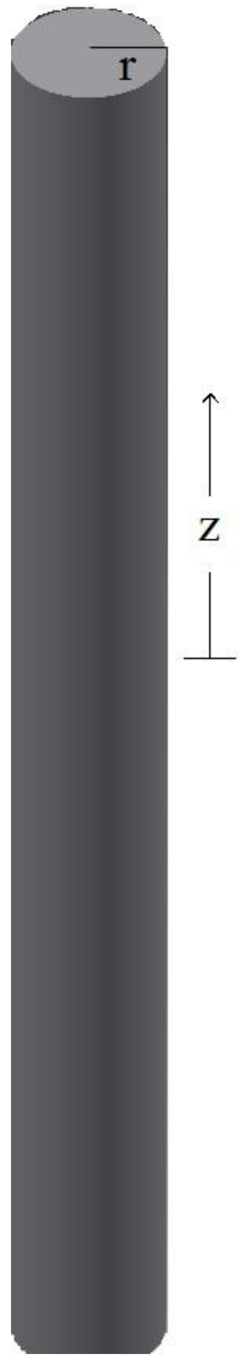
157 fuel assemblies arranged in a circle. All approximately square 9”x9”.

Figure 1-2



17x17 uranium dioxide fuel rods. 25 slots are guide rods or electronics.

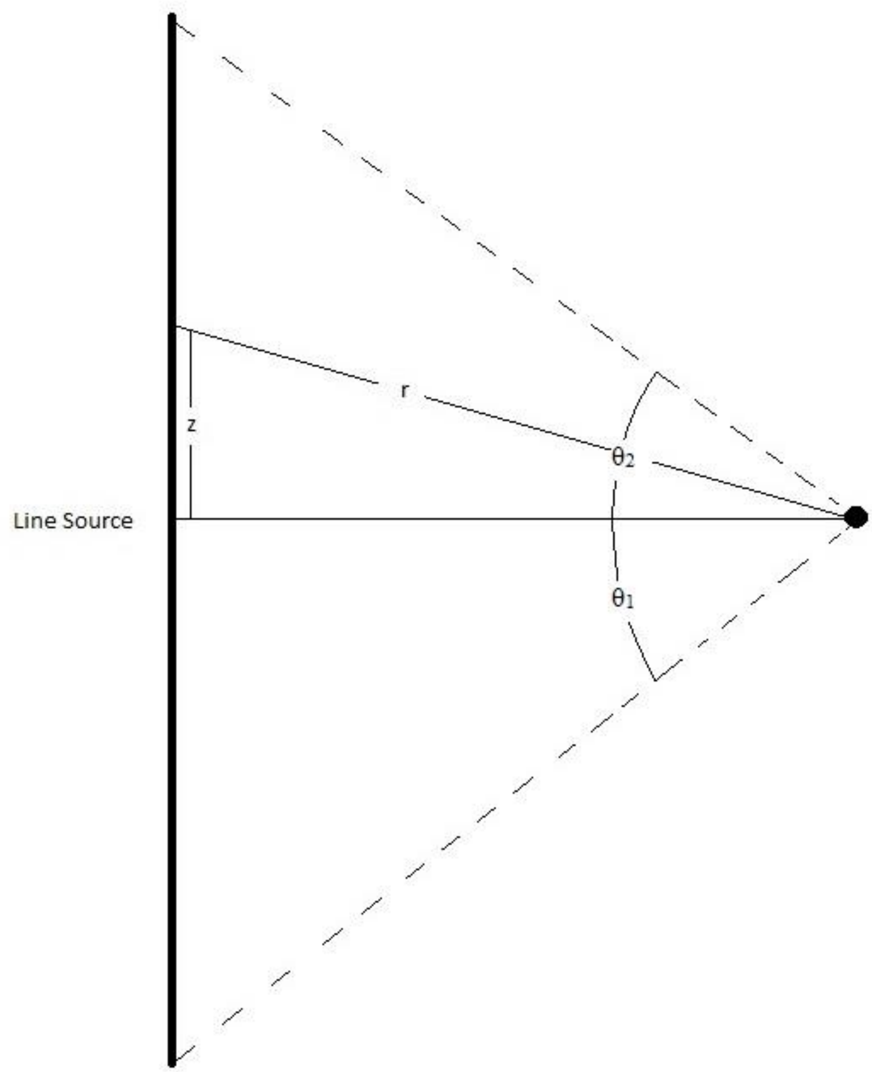
Figure 2-1



r is measured from the radial center of the fuel rod

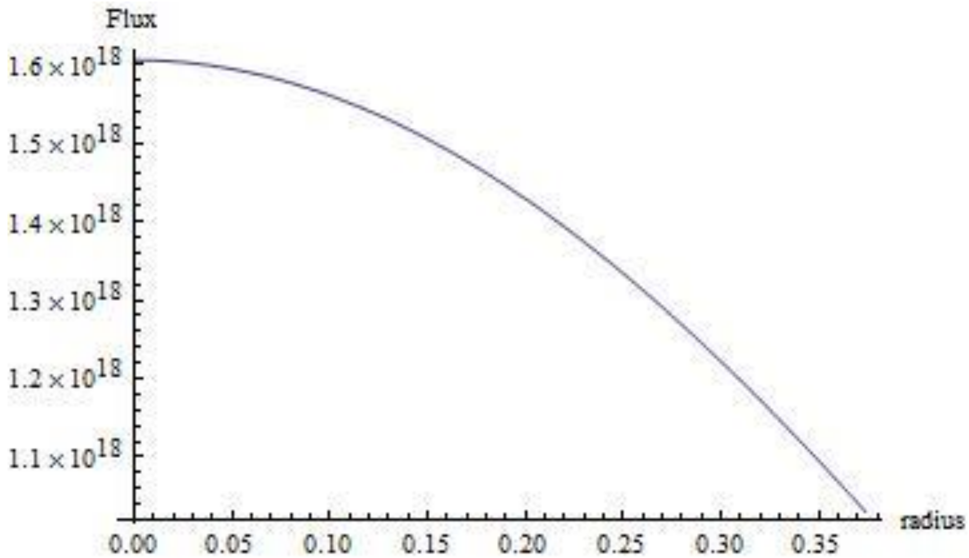
z is measured from the middle of the reactor

Figure 3-1

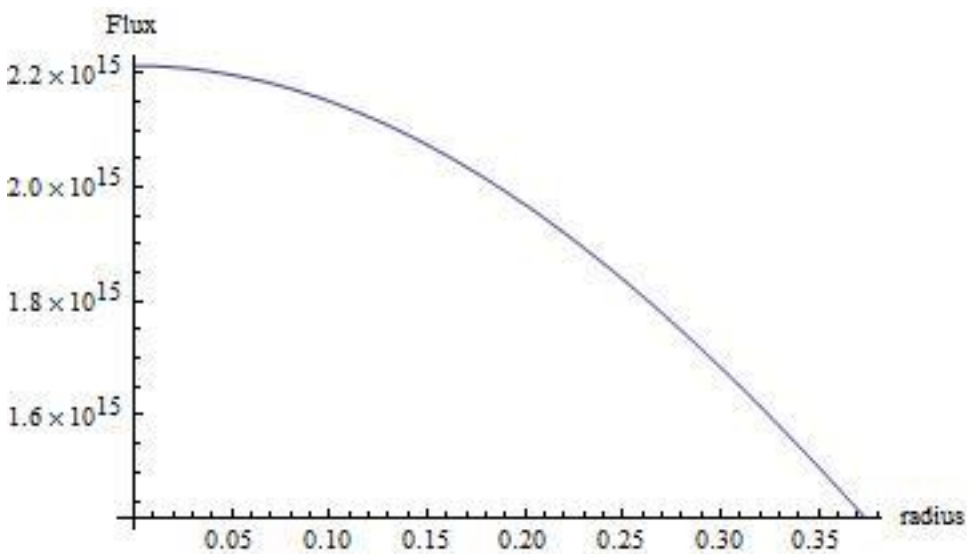


Appendix – Results

Neutron Flux

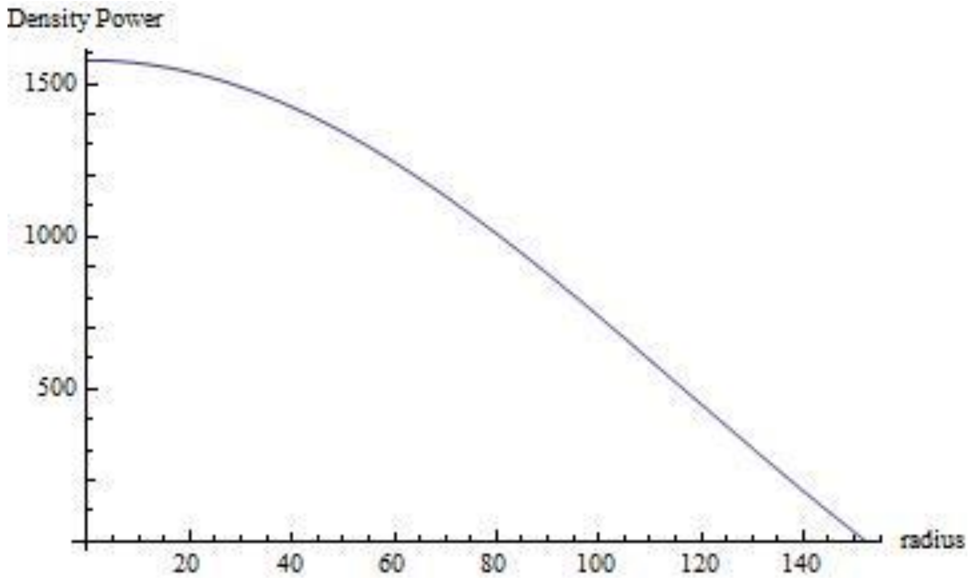


$z = 0.0cm$

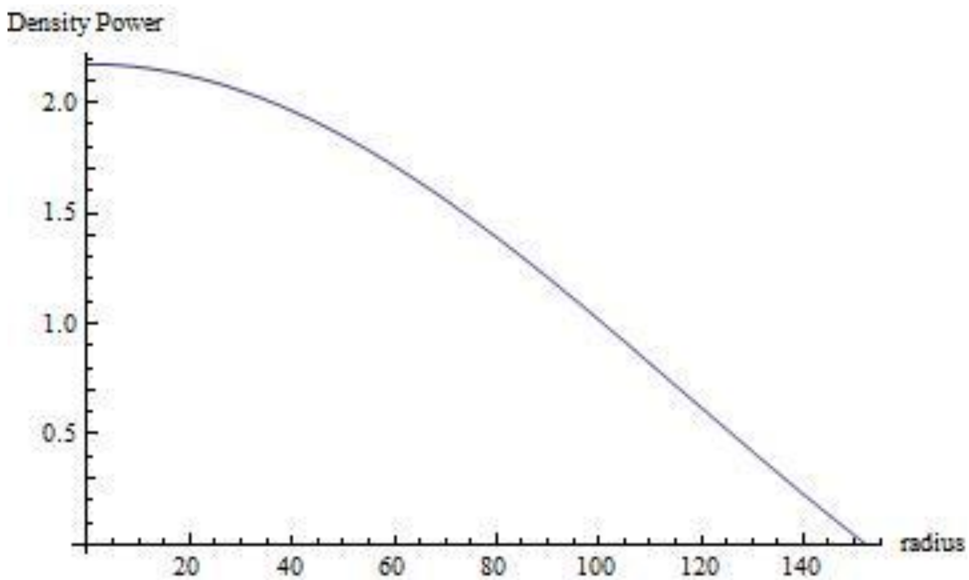


$z = \frac{ReactorHeight}{2} = -z$

Power Density



$$z = 0.0cm$$



$$z = \frac{ReactorHeight}{2} = -z$$

Appendix – Mathematica Code

```
(*Reactor Specifications*)
NumberofAssemblies = 157;
NumberofRodsPer=17*17-25;
TotalNumberofRods=NumberofAssemblies*NumberofRodsPer
41448
ReactorHeight=388.1;(*cm*)
ReactorRadius=152.019; (*cm*)
ReactorVolume = Pi*ReactorRadius^2*ReactorHeight (*cm^3*)
2.81766×10^7
(*Number of fission per second is equal to the reactor power divided by the energy released per
fission*)
ReactorPower = 3400*10^6 (*Watts*);
PowerPerRod=ReactorPower/TotalNumberofRods  //N
EnergyPerFission = 3.204*10^-11 (*Watts or 200MeV*)
FissionsPerSecond = ReactorPower/EnergyPerFission
82030.5
3.204×10^-11
1.06117×10^20

(*Neutron Flux Per Rod*)
RodRadius=0.374; (*cm*)
RodVolume = Pi*RodRadius^2*388.1 (*cm^3*);
MacFissCrossSec=.05*582.2*.04833+.95*0 ; (*The enrichment, thermal neutron fission cross
section, atom density of U235.  U238 has zero thermal neutron cross section c*)
A=3.63*PowerPerRod/(RodVolume*EnergyPerFission*MacFissCrossSec)(*Component in
finding the neutron flux.  See text.*)
3.87341×10^13
Flux[r_,z_]:=A*BesselJ[0,2.405*r/(RodRadius+2.13*.16)]*Cos[Pi*z/(ReactorHeight+2.13*.16)]
(*Actual neutron flux per rod as a function of radius away from rod and height measured from
the center of rod*)
TotalFlux[r_,z_]:=Flux[r,z]*TotalNumberofRods (*Neutron Flux for the entire system*)

TotalFlux[1*RodRadius,0]
1.03011×10^18
Manipulate[Plot[TotalFlux[r,z],{r,0,RodRadius},AxesLabel→{radius,Flux}],{z,-
ReactorHeight/2,ReactorHeight/2}]

(*Power Density*)
EnergyDeposited=2.88391782*10^-11;(*EnergyDeposited Locally is taken as 180MeV or in this
case converted to Joules*)
PowerDensity[r_,z_]:=1.16*ReactorPower*EnergyDeposited/(ReactorHeight*RodRadius^2*Tot
alNumberofRods*EnergyPerFission)*BesselJ[0,2.405*r/(ReactorRadius+2.13*.16)]*Cos[Pi*z/(
ReactorHeight+2.13*.16)]
PowerDensity[0,0]
```

```
Manipulate[Plot[PowerDensity[r,z],{r,0,ReactorRadius},AxesLabel→{radius,Power Density}],{z,-ReactorHeight/2,ReactorHeight/2}]
1577.75

(*Gamma Ray Buildup Flux*)
(*Material is lead approximately *)
Sievert[θ_,γ_]:=NIntegrate[Exp[-γ*Sec[x]],{x,0,θ}]
(**)
Clear[A]
MassAttenuationLead=.0684;  (*1MeV*)
DensityLead=11.34; (*g/cm^3*)
AttenuationCoef=MassAttenuationLead*DensityLead
BuildupFactor=1.37; (*uR  1MeV*)
ShieldRadius=BuildupFactor/AttenuationCoef (*cm*)
A1=2.84;
A2=1-A1
ReactorHeight;
EmittedGamma=FissionsPerSecond/(2*Pi*ReactorRadius^2+2*Pi*ReactorRadius*ReactorHeight)
Alpha1=-0.03503;
Alpha2=0.13486;
Theta1[z_]:=ArcTan[(ReactorHeight+z)/ShieldRadius]
Theta2[z_]:=ArcTan[(ReactorHeight-z)/ShieldRadius]
BuildupFlux[z_]:=(EmittedGamma/(4*Pi*ShieldRadius))*(A1*Sievert[Theta1[z],(1+Alpha1)*BuildupFactor]+A2*Sievert[Theta1[z],(1+Alpha2)*BuildupFactor]+A1*Sievert[Theta2[z],(1+Alpha1)*BuildupFactor]+A2*Sievert[Theta2[z],(1+Alpha2)*BuildupFactor])
0.775656
1.76625
-1.84
2.05693×10^14
BuildupFlux[0]
5.90734×10^12
```

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