CHAPTER 4: NEUTRONICS OF GASEOUS CORE REACTORS

The preliminary design of any reactor system depends on the behavior of the neutrons and their energy levels within the specified geometries. Since gaseous core reactors will be operated at supercritical conditions to support rocket applications, the power densities will be comparatively high. The idea of investigation of neutronics in gaseous core reactors which can be used for rocket applications will be based on fission process, where self-sustained fission reaction generates the require energy. In the fission process heavy nucleus absorbs a neutron and reaches an exited state from which it can be escaped by radiative decay or by fission. In the advancement of nuclear reactor technology has addressed reactor neutronics with moderated power densities. When it comes to reactor systems that will be designed for interstellar rocket applications will have high-power density with compact geometries, this approach is more challenging in flyable reactor systems design with effective safety considerations. This chapter deals with two-group analysis in which two kinds of neutronics, like fast and thermal groups are considered over various reflector systems with varying thickness.

The effective parameters like Keff and energy flux densities will be investigated using Monte Carlo based two group diffusion models code. The results will be used to obtain input parameters for the heat transfer analysis between the propellant and the fission region. Validation to the code will be done for the case of solid core reactors experimental data available from Los Almas Space laboratory model investigated for fuel channels with hydrogen propellant based system. The geometry that was considered in the current work is give in the figure 4, the reflector thickness considered is 10 cm with a core radius of 1.5. For
simplification pup uses the core geometry model was scaled when grid was read into the neutronics code. The gas considered as fuel inside the reactor is U-C-F and enrichment levels are varied continuously with the different cases. The overall thickness of the core structure is considered as 100 cm.

![Diagram of reactor core system](attachment:image.png)

**Figure 4.1: Geometry of GCR Model Consider for Analysis**

The reactor model operates with the vortex generation through the pressure difference created by radial flow. Since these reactors are not having the control rods the reaction is controlled by the pressure difference created in the flow. If the outlet temperature of the propellant reaches its maximum design value the flow rate increase thought he wall to create extra cooling as a same time the reactor core is going to have more concentration of hydrogen. The propellant itself starts slowing down the neutrons and the reactivity is controlled (Glasston, 1955). The schematic diagram of actual reactor core system used in nuclear rockets is given in the fig. 4.2. Buffer region is created in the core to improve the reactivity time
between the neutrons and the isotopic material, since the nuclear fuel is so costly the flow rate need to be regulated at 100:1 ratio so that travel distance can increase. The designs parameters are considered for generating reactor core geometry are taken from Van douman, 1996 reactor analysis work. In the present thesis three reactor models are investigate with a variation in percentage of enrichment of the fuel. The mixture of U-C-F gas is also varied with the enrichment levels.

Figure 4.2: The hydrogen fuel (propellant) is injected in gaseous form, as well as the Uranium fuel in gaseous form

In the reactor core the fission process takes place with neutron interactions, besides the energy produced in the process there are various fragments released along with the neutrons. In the order of priority neutrons are the interesting elements in the core to maintain self sustained fission reaction. Along with the neutrons γ rays
are released. The fast neutrons whose energy levels are high compared to thermal neutrons they are also going to obstruct the interactions. The propellant or the moderator or the reflector starts slowing down the fast neutrons so that neutron populations increase with the added quantity. Some of the neutrons get scattered due to collisions (Ragsdale, 1993). The complete fission process in the reactor core is given below with the level of extraction of different particles. Some of the neutrons starts getting absorbed with reflector material, so the total number of neutron population need to be randomized.

Figure 4.3: The process of fission along with the by-products
4.1 NEUTRON BEHAVIOR IN REACTOR CORE

The process of fission reaction can occur due to interactions of neutrons with nuclei and it absorbs the neutron and splits into two different fragments. In this process six new neutrons are released. These neutrons again start participating in the process and the reaction becomes chain process. In the reactor core this reaction can be self-sustained with the neutron production and their interactions with the radioactive materials (Runback, 1964). This process can continue with supplying any external energy and the stable environment can be maintained with the heat produced during the process. The behavior of the neutrons depends on the materials used in the core and neutron reflector thickness and its selection differentiates the process.

Figure 4.4: Simplified Neutron Cycle
The neutrons generated in the fission process will be normally in the energy range between 0.1 to 10 Mev, with a mean energy between 1 and 2 Mev. The generated neutrons will have different energy levels and different behavior based on its form of emitting. The fast neutrons move unimpeded through space until they interact with other nuclei (poorer, 1954). The complete cycle of the neutrons in the reactor core is given in the below figure 4.4.

4.2 APPROXIMATE METHOD OF ANALYSIS

Neutronics analysis is going to change with the reactor configuration to the operational conditions. The basic difference starts with the atomic ratio of the fuel which is going to use in the fission process, the reflector materials used, reflector thickness, Moderator usage, reactor type like fast reactor, breeder reactor, power reactor, reactors used for propulsion applications. The design criteria for any reactors depend on the parameters given in the table 4.1. The investigation conducted on neutronics of any reactor system is differentiated by its keff factor. In case of gaseous core reactors the design is based on mission time and the rocket operational conditions like the expected specific impulse, thrust of the system intended to operate. The temperature limitations over the reflector materials and the reactor chamber are going to set the upper limit over the system. The critical value of keff can be selected for the temperature of operation. In case of controlling the reactor for safety purposes the design is made based on the selected keff factor based on the maximum assurance limit (Robert, 1989). The keff accident condition is taken a reference for the safety maximum limit. The neutronics analysis is mainly conducted to estimate the neutron flux and the neutron density in a core. So that reactivity can be estimated.
If the probability of neutrons are getting leaked though the core then the reflector thickness and the material need to be reconsidered. In reaching the desired keff value the reactor system need to be supplied with effective fuel composition. In case of solid core reactor systems the fuel composition varies with the material used in preparing fuel rods. Since the GCRs fuel composition depends upon the atomics ratios in which fuel gas is processed. In case of this thesis the fuel mixture is prepared from Uranium and floride compounds, the percentage of enrichment is directly proposal to the keff value. The core where is fission process is occurring the peak temperature value is measured and the distributed temperature profile depends on the effectiveness of the heat transfer with the
reactor buffer region. In most of the cases while designing the gas core nuclear reactors the neutron reflector and the external radiation shielding is selected on a single configuration to reduce the total weight of the system.

![Neutron Tracks Leaving the Reflector with Angle Compared to the Reactor Surface](image)

**Figure 4.5: Neutron Tracks Leaving the Reflector with Angle Compared to the Reactor Surface**

In case of shielding design and analysis the dosage limit is mentioned in reactor systems design. In case of space reactors the considerations will change for a manned mission or space vehicle which travels the longer distance. The reactor selection for the rocket vehicles is done based on the specific power density of the designed system. Since the total power requirement to attain required gravitational acceleration is a major interest.
4.3 CODE DEVELOPMENT FOR NEUTRONICS INVESTIGATION

The code is developed for solving time-independent two groups discrete originates in the form of Boltzmann Transport Equations for different geometries defined by the user. The grid used in solving this problem is based on standard diamond differencing method for space angle discretization, and for spatial and angle discretization adaptive weighed diamond differencing method was used. The Monte Carlo method and the functions generated for solving the two group neutron equations detailed in 4.3.1 and the input model is attached in appendix A, the code is also attached in Appendix B. This code developed with the help Dr. Ozoner from Los Almost Space Laboratory and the neutron groups are used from the common format which can be used for MCNP, the code has grid libraries the name space step 28 class is created. This class is capable of reading the fission neutron groups and the probability functions are created with the unsigned materials. The distance for the cell zones are given as an input and it can solve for the reflector surfaces. MaterialData Xs object is created to store the material properties of each fission group and the reflector configurations. The isotopic material data is written in Material ID get_fission spectrum and the variation of fuel composition is varied through the input. To distinguish the neutron spectra energy group class is created along with the identified cell zones. The flow chart describes the user input details required to run the code and the final data to analyses the reactor criticality format can be identified. The data generated with the code is processed through Statistical number crunching system and the required values can be extracted, the data is plotted using gnu plot for the parameters required to explain the behavior in gas core reactors.
Figure 4.6: Flow Chart for the Code Developed to Solve Two Group Model
4.3.1 MONTE CARLO NEUTRON TRANSPORT

In the analysis of reactor physics neutrons play an important role to obtain self-sustain fission reaction. Monte Carlo codes are used to analyst the radiation shielding problems, neutron interactions with the nuclei, reflector modulator problems, and safety criticality analysis. Besides neutron transport equations the fundamental particles behavior also studies using Monte Carlo method. These codes accept physical geometries to analyses major interactions by the particles with input data conditions. The neutron transport equations are solved using stochastic method by Monte Carlo code. The second method that can be used for solving the neutron interaction by using Boltzmann equations, this method takes a statistical averaging of net leakage of neutrons from the core and assuming that all the neutrons in the reactor are with uniform energy levels and are taking part in the fission reaction.

\[
\nabla \cdot \phi(r, \Omega) + \sum_{a} \phi(r, \Omega) + \sum_{a} \phi(r, \Omega) = \int_{\Omega'} P(\Omega' \Omega) \phi(r, \Omega') d\Omega' + S(r, \Omega) \quad (4.1)
\]

The above equation explains the loss rate of neutrons from a unit volume with a unit angle around a point. The neutron balance can be obtained by quantifying the neutron loss rate by absorption, reflection; scattering and the loss of energy by neutrons can be balanced. The integral equation on the right side will express the scattered neutrons quantity in all directions. S indicates the source of fission neutrons that are liberated in each chain for a specific location in the given volume. The difficulty with the above method lies with solving the flux distribution and which needs high computational power as well as the specific geometrical models to solve. In case of Monte Carlo algorithms the development of statistical model and the average rate of interactions can be defined base on the geometrical considerations. Monte Carlo algorithm performs the calculation on a
specific neutron and its path throughout grid at a time. The collection results are based on discreet events with the individual neutron behavior. The code developed in this thesis is not concentrated on adjoin calculations, since the running time to reverse the process is a parallel algorithm development approach. The energy variation is taken as a continuous function to describe neutron interactions in a 1 D cylindrical geometry. The code takes random selection individual neutrons to solve for required parameters. The rest of the analysis based on infinitesimal cylinder with a one dimensional cross-section behaves as a complete physical system and generates the data for the reactor criticality.

The probability distribution functions considered in selecting the neutrons is taken from probability distribution f(x) where x as a variable. The event of interactions occurrence predicated between xs and x+dx

\[ dP = f(x) \, dx \quad (4.2) \]

The probability density function was generated using a monte carlo integration method for the following equation in the interval of \([a,b]\) for a variable in the interactions.

\[ P(a < x < b) = \int_a^b dP = \int_a^b f(x) \, dx \quad (4.3) \]

The behavior of the equation with a probability function will vary with the intervals and the values become infinity at a certain point of time. The convergence criteria given under boundary conditions to take the intervals for which the solution is obtained. This can be calculated from cumulative distribution function by a direct integral.
\[
F(x) = P(x' < x) = \int_{-\infty}^{x} f(x')dx'
\]  
\hspace{1cm} (4.4)

The limit defined in the code to solve the cumulative probability function space must yield \( P=1 \) and hence

\[
\lim_{x \to \infty} F(x) = 1
\]  
\hspace{1cm} (4.5)

In case of solving this problem the random variables are distributed between the \( a \) and \( b \) the function is written in the form of

\[
f(x) = \begin{cases} 
  \frac{1}{b-a} \text{ when } a \leq x \leq b \\
  0 \text{ when } x < a \text{ or } x > b 
\end{cases}
\]  
\hspace{1cm} (4.6)

Now the cumulative distribution function is going to change in the form of

\[
F(x) = \begin{cases} 
  0 \text{ when } x < a \\
  \frac{x-a}{b-a} \text{ when } a \leq x \leq b \\
  1 \text{ when } x > b 
\end{cases}
\]  
\hspace{1cm} (4.7)

When the uniform distribution of the neutrons are considered in the geometry the limits are set for \( a=0 \) and \( b=1 \). Then the function is going to generate random variable which can be uniformly distributed in the given interval. In case of solving complete geometry Pseudo-random number generation method is used to create the sample for various intervals inside the geometry, the other random number is derived from an inversion method of sampling. The cumulative distribution function is solved at a uniformly distributed variable at unit interval. With the obtained data sample a uniformly distributed variable is generated called
\( \zeta \). This is used to apply the sampling method the CDF by using the value at 0.8979.

\[
F(x) = \zeta \iff x = F^{-1}(\zeta)
\]  

(4.8)

At a given CDF the corresponding value of the x is set at 7.1552, the data samples given in the neutron groups can be randomized in the similar process and the functions are generated at each specific location of the geometry. In case of the values outside the interval the solver identifies the group as a different function and obtains the samples by using rejection technique. In simpler terms the density function \( g(x) \) and the constant of rejection limit is set for \( f(x) \) at \( c \geq 1 \) and the \( f(x) \) is generated through the following relation

\[
f(x) \leq cg(x)
\]  

(4.9)

In this case the uniformly distributed variable is obtained from the relation given below

\[
\zeta < \frac{f(x)}{cg(x)}
\]  

(4.10)

If the inequality does not hold then the value will be discarded and the procedure is going to repeat from the beginning with the help a loop this function was considered and the if- else statement is used to differentiate the variable limit. If the values are acceptable with in the domain the \( f(x) \) follows exactly the same relation and data can be generated. The method of selection of sampling is done based on a specific relation given by dividing both the integrals with an infinity limits.
\[ E = \frac{\int_{-\infty}^{\infty} f(x) \, dx}{\int_{-\infty}^{\infty} c g(x) \, dx} \]  

(4.11)

The difference between the functions are relatively small and the ratio of integral have a possibility of reaching unity, since the functions are used to generate random data this factor will not affect the solution accuracy. The above method can only generate the distribution inside the core; in case of fission reaction the third method is used to account for energy and the neutrons emission in the fission reaction. The consideration taken for the two-body scattering with equip-probability interval of 12 and 14. The random variable \( x \) is generated from the following equation to integrate the function.

\[ x = x_n + (N\zeta - n)(x_{n+1} - x_n) \]  

(4.12)

The sampling algorithm starts with integrating the function between \( x_n < x < x_{n+1} \)

\[ \int_{x_n}^{x_{n+1}} f(x) \, dx = \frac{1}{N} \]  

(4.13)

The first interval sample is taken from the relation given below to solve the integral part

\[ n = N\zeta + 1 \]  

(4.14)

In most scattering reactions the cosine interval is chosen as 32 and the solution, number of intervals considered differentiates the accuracy of the solution with the functions used for creating the random data points in the reactor core. The reactor core is divided into different geometrical regions to track the single neutron in
each of the region. The path traced by each neutron is said to be a track and the path is traced once the set points on the geometry given by a specified path of neutrons are attributed to the generation point to the escape point. The total tracks traced by each neutron create a neutron history for that particular core. The free path length between the collision points need to be defined with the help of user input, the sample input values are defined in the Appendix I, which described the set points and the paths traced are chosen from eighteen value matrix. The matrix generated with different set points using a trace angle is directly read by the code. The subroutines used in the code are having the functions to generate the eighteen values for the path tracing. This values can be tallied with the number of neutrons emitted at the source and the number of histories generated.

To estimate the keff value to probability of interactions by neutrons with the nuclei can be found by sampling the free path length. The geometrical cross section is set to the nuclei and the track history can be generated for the microscopic interaction of the neutrons. This can be obtained from the constant interaction probability with the homogenous medium traveled by the neutron; the homogenous medium can be a reflector or a propellant as well as gaseous uranium (Feisbee, 2003). The probability that the neutron can undergo interaction with the nuclei surface with a distance change to dx can be obtained from $\Sigma_t$.

$$dP = \Sigma_t \, dx \quad (4.15)$$

This can trace the path from x to dx and the probability of interaction in the nuclei region. An arbitrary zero position is considered and the probability needs to be found for the initial track of the neutron path. The idea of creating initial probability function is to find out the chance of reaching x distance without any interactions and also to find out the probability with which the interactions takes place within the given interval.
\[ dP_0 = -P_0(x) dP = -P_0(x) \Sigma_r dx \] (4.16)

The non-interaction probability also can be found from the differential equation given by

\[ P_0(x) = e^{-x \Sigma_r} \] (4.17)

The first move distance can be identified from the function given below with a specific cross section

\[ P_0(x) = P_0(x) \Sigma_r \, dx = \Sigma_r \, e^{-x \Sigma_r} \, dx \] (4.18)

The free path length can be written as

\[ f(x) = \Sigma_r \, e^{-x \Sigma_r} \] (4.19)

This can be included in cumulative probability function written to the microscopic cross-section

\[ F(x) = \int_0^x \Sigma_r \, e^{-x \Sigma_r} \, dx = 1 - e^{-x \Sigma_r} \] (4.20)

The initial length characteristics can be found using \( x \) and the function that can be expressed based on cross sections selected in solving the defined problem. The cross sections that can solve are up to the limit of 0.3188 cm^{-1}.

\[ x = -\frac{1}{\Sigma} \ln(1 - \zeta) = -\frac{1}{\Sigma} \ln \zeta \] (4.21)

The mean free path, \( l \) can be used identified to calculate the non-interaction probability and the code written is with the values that can solve up to the limit of
3.1364 cm. The interactions found are at a distance of 0.004 to 0.006, it can be a
good method of solving the mentioned possibilities with the functions generated.
Deta Tracking probability method was used to identify the neutron path length
probabilities. The generated data is interpreted with the homogeneous mediums,
since the solution of non-homogeneous region needs multiple integration
solutions which will become complex in developing the code. It takes ample
amount of time to solve integral equation; more over the probability functions
becomes closed form equations. In real cases the geometry is considered as
regions by counting the cell zones. The diamond differencing method used to
create the specific cell zones to number them in terms of groups.

\[ l = \int_0^\infty x e^{-x^{-\Sigma}} dx = \frac{1}{\Sigma_i} \]  

(4.22)

The problem in generalizing solution in case of neutron mean free path changes
with the materials and the region of interest. The cell zones are identified based
on spatial neutron coordinates and the boundary surfaces are excluded from the
cell zones, in such cases the re-sampling the remaining distance to the next
collisions. This approach is included in a subroutine by using ray coding method.
The non-interaction probability can be expressed as

\[ e^{-x_2 \Sigma_2} = e^{-(x_1 - d) \Sigma_1} \Rightarrow -\Sigma_{i_1} \Sigma_{i_2} = -(x_1 - d) \Sigma_{i_1} \]  

(4.23a)

\[ \Rightarrow x_2 = (x_1 - d) \frac{\Sigma_{i_1}}{\Sigma_{i_2}} \]  

(4.23 b)

In case of non-interactive probability the path length changes to

\[ x = d + x_2 = d + (x_1 - d) \frac{\Sigma_{i_1}}{\Sigma_{i_2}} \]  

(4.24)
In this case infinite circular cylinder of radius R, is considered in a Cartesian coordinate system with three direction vectors, and the distance d can be resolved by using the equation below

\[
d = \sqrt{\left(\Omega_x^2 + \Omega_y^2\right) r^2 - \left[\Omega_x (y - y_0) - \Omega_y (x - x_0)\right]^2}
\]

Based on the line of site the distances between the cell zones are identified so that the nearest surface can be integrated and shortest distance can be found. The microscopic total cross section of the isotopic material needs to be identified to generate a probability distribution for the isotope channels

\[
P_m = \frac{\sum_{i,m}}{\sum_{t}}
\]

The conditional probability is written to indicate a selecting reaction in the ith group if m isotopes are written as

\[
P_{i,m} = P_i P_m \frac{\sum_{i,m}}{\sum_{t,m}} = \frac{\sum_{i,m}}{\sum_t}
\]

In the fission process the truncated integer values are considered for N of \(\nu\) and the probability was determined with the decimal fractions. In reality if the neutron absorbed by isotopic material the neutron history comes to termination and the new source of generation led to neutron release. The number of emitted neutrons in this process can be identified by a fission nubar \(\nu\).

\[
\text{No. of emitted neutrons} = \begin{cases} N + 1 & \text{if } \zeta - \bar{\nu} \leq N \\ \end{cases}
\]
The time of emission is exponentially distributed and it can be similar to the neutron free path length

\[ t = -\frac{1}{\lambda_j} \ln \zeta \]  \hspace{1cm} (4.29)

The distributions need to be made for neutron energy levels independent of its decay, the variation in the energy levees differentiate the total group emission time.

4.3.2 GRID GENERATION USING DIAMOND DIFFERENCING METHOD

The method used in solving the neutronics in an infinite cylinder was discretized using a finite volume based diamond difference method which creates non uniform grid. Since the specified geometry is divided into regions to identify fission groups, so structured grids are not so accurate in such kind of applications to perform analysis, the number of grid elements generated are 100,000 and the number of geometrical zones divided are 250, four groups are taken on each zone to analyses the changes with respect x+dx. The data used to specify neutron secondary interaction are given in the Appendix c. To estimate the quality of the mess generate and the accuracy of the solution a relation is generated to figure of marite.

\[ FOM = \frac{1}{R^2T} \]  \hspace{1cm} (4.30)

R indicates relative error and T indicates the computer run time, in the current problem three different cases are solved on a single grid. This identification of the error need to be compared with the each set of solution, in order to get more accurate solution when FOM value increase the R values starts decreasing. The
most accurate solution can be obtained at unity. The estimated error is obtained to generate a profile in calculating keff value with the help of code and generated grid. The error range started coming down with the increased iterations.

4.3.3 CODE CAPABILITIES

This particular code can take input in the form of ASCII text with a free field format, a sample input is given in the appendix-I. It can be developed for solving in vacuum, reflective, periodic and surface source boundary conditions. It can generate data for inhomogeneous source fixed or Keff calculation as well as time absorption of alpha, nuclide concentration can be investigated. The code is capable of solving one dimensional and two dimensional geometries with different shapes and the range of the solution domain depends on cell zones identified.

4.4 METHODOLOGY

Nuceutronic analysis is conducted under uniform temperature and density distributions. In Gas core reactors the mean free path length of the neutrons are very high under such cases the investigation of neutron criticality in a one dimensional geometry needs certain assumptions. The temperatures variations are considered to be very small and the density distributions are taken as a reference. Criticality analysis is conducted to estimate the reflector thickness and the cylindrical core effects on neutron behavior by solving two-group, two- region calculations for specified 1-D geometry. The computational facility needed to solve multi group diffusion theory to investigate the complete three dimensional core model with a completely reflected material cross-section are out of the scope for this assignment. The energy range for neutrons, all reactions given in a particular cross-section data evaluation are accounted for, and cover the energy
range between $10^{-5}$ eV and 20 MeV. The nuclear data library was used to apply neutron-induced cross-sections at different temperatures. Thermal correction in the phonon band requires separate cross-section evaluation, the so-called $S(\alpha,\beta)$ cross-sections that are available for BeO, graphite and hydrogen at a temperatures varying from 294 K -15000K used in the present calculations. The $S(\alpha, \beta)$ thermal scattering treatment is a complete representation of thermal neutron scattering by molecules and crystalline solids. Two processes are allowed: (1) inelastic scattering with cross-section $\sigma_{\text{in}}$ and a coupled energy-angle representation derived from an $S(\alpha,\beta)$ scattering law and (2) elastic scattering with no change in the outgoing neutron energy for solids with cross-section $\sigma_{\text{el}}$ and an angular treatment derived from lattice parameters. The elastic scattering treatment is chosen with a probability of $\sigma_{\text{el}}/(\sigma_{\text{el}} + \sigma_{\text{in}})$. The representation of Thermal neutrons through Maxwellian Distribution is represented in the figure below.

Figure 4.7: Maxwellian Distribution of Neutron Energies
The reactor core was taken to be a homogenous mixture of propellant gas and the U-C-F, the propositions are obtained from the computational fluid dynamic analysis conducted on the core. The resonance escape probability is considered as one and the fast fission was neglected. The thickness of the beryllium reflector and the graphite reflectors are varied from the consideration made by Lafyatis near optimum value 0.6. The distance at which the neutrons are traveled is considered as a neutron track, in the current solution individual neutrons are traced for its path and probability function is used to estimate the number of neutrons interacted with the source and the neutron interaction regions are formed. The figure 4.8 shows the neutron trajectory between the source and the point of absorption.

Figure 4.8: Distance and Neutron Trajectory between the Points of Neutron Source to the Point of Neutron Absorption

In this work neutron investigation is conducted in a cylindrical core system with infinite length to consider the geometry as a one dimensional system. The analysis
is conducted through a code written on two group neutron transport theory using Monte Carlo method, the detailed analysis on neutron behavior in a GCR core can give the neutron balance from the production to mean by which neutron is getting utilized. This balance can give us an account for neutron production to the neutron loss. Also for analyzing thermal hydraulics this balance will help in estimating the total heat produced by the system. Neutron diffusion theory is used to describe the neutron flux density and the neutron production, leakage, fission and absorption. Solving multi group diffusion equations are very much complex and time consuming. Inters of computational power required to analyze the reactor geometries are even more complex. In the real analysis of reactor design simple apaches are taken to solve the diffusion equations like one-speed neutron transport theory or two group diffusion equations are solved to attribute the neutron behavior to the complete system. This is not a new process, most of the validation studies are conducted on the similar aspects and the results comparison is satisfactory, the similar approach is used to complete the analysis. The infinity multiplication factor is a notation for the reactor criticality analysis and which can be written in the equation below. This indicates the number of neutrons produced in the absorption of neutron per fission.

\[
k_\infty = \frac{Neutron\ production\ rate}{Neutron\ absorption\ rate}\]

(4.31)

In gaseous core reactors neutronics analysis the geometry considered is in infinity in length, in such cases the neutron leakage from the system is neglected. The only way of neutron getting lost depends on the absorption. In case of finite geometries with a specific dimensions when a core diameter is specified the effective neutron multiplication factor can be found from the following equation.
\[
k_{\text{eff}} = \frac{\text{Neutron production rate}}{\text{Neutron absorption rate} + \text{Neutron leakage rate}}
\]  

(4.32)

The ration of infinite multiplication factor to the effective multiplication factor can be found by using source as \( S \) and the leakage as \( L \), absorption as \( A \)

\[
\frac{k_{\text{eff}}}{k_{\infty}} = \frac{A}{A + L}
\]

(4.33)

The above relation represents the non-leakage neutron probability for a given system. In case of critical reactors the neutron multiplication factor should be above unity. If the size of the system is defined by \( a \), and the radial distance is taken as a measure for identifying the neutron leakage with respected to the surface a proposal relation can be used.

\[
\frac{L}{S} = \frac{A}{V} \frac{S A}{a^2} \frac{a}{a^2} \frac{1}{a}
\]

(4.34)

The infinite multiplication factor is a function coolant used and the meteor material selected along with the propellant used inside the system. In general cases the value varied from 0 to 1.2 in case of space reactor. Neutron source in case of homogenous system is represented by \( S(r) \) and it can be found from the balance written for the diffusion equation

\[
S(r) = \sum \kappa_a \Phi(r)
\]

(4.35a)

Lekage+absorption=Production  

(4.35b)

\[
-DV^2 \Phi(r) + \sum \kappa \Phi(r) = S = \sum \kappa_a \Phi(r)
\]

(4.35c)
The non-leakage probability needs to be found from the given relation to generate the functions through the cumulative functions in Monte Carlo Method.

\[
\nabla^2 \Phi(r) + \frac{\sum_a (k_a - 1)}{D} \Phi(r) = 0
\]

(4.36)

\[
\nabla^2 \Phi(r) + B^2 \Phi(r) = 0
\]

(4.37)

Here geometrical bulking is added to specify the region to generate the functions

\[
1 = k_\infty \left[ \frac{1}{B^2 m L^2 + 1} \right]
\]

(4.38)

\[
P_{\text{non-leakage}} = \left[ \frac{1}{B^2 m L^2 + 1} \right]
\]

(4.39)

\[
P_{\text{non-leakage}} = \left[ \frac{1}{B^2 m L^2 + 1} \right] = \frac{\sum_a \Phi(r)}{\sum_a \Phi(r) + B^2 m \phi} = \frac{\sum_a \Phi(r)}{\sum_a \Phi(r) + (-D \nabla^2 \Phi(r))}
\]

(4.40)

In two-group diffusion theory the considerations are made based on thermal energy group as well as the fast neutron group. The boundary set between these groups is given as 1 e. The fast neutrons can be thermalized with the help of moderator; fast neutrons are will be last in absorption. The neutron fluxes can be expressed for both the groups.

\[
\text{Fast} : \Phi(r) = \int_{10 \text{MeV}}^{100 \text{MeV}} \Phi(E, \hat{r}) dE
\]

(4.41)

\[
\text{Thermal} : \Phi(r) = \int_{0 \text{ev}}^{1 \text{ev}} \Phi(E, \hat{r}) dE
\]

(4.42)
The complete set of equations used in two group neutron diffusion theory to explain the complete neutron process is given in the figure 4.9 based on the energy spectra.

\[
D_1 \nabla^2 \Phi_1(r) - \sum_{al} \Phi_1(r) + S_1(r) = 0
\]  

(4.43)

The balance is written as the fast neutrons produced from the source and the fast neutrons getting thermalize along with the neutrons leaked.
\[ k_{\text{eff}} = \frac{\nu_1 \sum_{f_1} \Phi_1(\vec{r}) + \nu_2 \sum_{f_2} \Phi_2(\vec{r})}{-D_1 \nabla^2 \Phi_1(\vec{r}) - D_2 \nabla^2 \Phi_2(\vec{r}) + \sum_{a_1} \Phi_1(\vec{r}) + \sum_{a_2} \Phi_2(\vec{r})} \] (4.44)

In the process of fission the thermal neutrons are absorbed and some of the fast neutrons are thermalized and the population gets balanced this relationship can be expressed as \( k_{\text{inf}}/p \) and the source term can be replaced with the help population balance and the equation four is converted in the form of equation 4.45 and this is used to ass in the code to develop fast neutron balance

\[ D_1 \nabla^2 \Phi_1(\vec{r}) - \sum_{a_1} \Phi_1(\vec{r}) + \frac{k_{\text{inf}}}{p} \sum_{a_2} \Phi_2(\vec{r}) = 0 \] (4.45)

Thermal energy group neutrons balance can be written in the similar pattern for the balance of source and the absorption and interaction with the nuclei.

\[ D_2 \nabla^2 \Phi_2(\vec{r}) - \sum_{a_2} \Phi_2(\vec{r}) + S_2(\vec{r}) = 0 \] (4.46)

In case of converting the fast neutrons into the thermalized neutrons, along with the source some extra neutrons are also added to the group. The probability of which the fast neutrons converted into thermal neutrons can be written as

\[ D_2 \nabla^2 \Phi_2(\vec{r}) - \sum_{a_2} \Phi_2(\vec{r}) + p \sum_{a_1} \Phi_1(\vec{r}) = 0 \] (4.47)

To calculate thermal and fast flux distribution of the neutrons inside the reactor system can be expressed as

\[ \nabla^2 \Phi_1(\vec{r}) + B^2 \Phi_1(\vec{r}) = 0 \] (4.48)

\[ \nabla^2 \Phi_2(\vec{r}) + B^2 \Phi_2(\vec{r}) = 0 \] (4.49)
The bulking of both the systems is quite similar since the effect of buckling calculated for geometries but not for the neutron groups, In the current woke single buckling effect is consider in the forms given below

\[- (D_1 B^2 + \sum_{a1} \Phi_1 (r)) + \frac{k_x}{p} \sum_{a2} \Phi_2 (r) = 0 \quad (4.50)\]

\[- (D_2 B^2 + \sum_{a2} \Phi_2 (r)) + \frac{k_x}{p} \sum_{a1} \Phi_1 (r) = 0 \quad (4.51)\]

\[
k_{\text{eff}} = \frac{k_x}{(1 + L_1^2 B^2)(1 + L_2^2 B^3)} = 1 \quad (4.52)\]

Where,

\[
L_1^2 = \frac{D_1}{\sum_{a1}} \quad (4.53)
\]

\[
L_2^2 = \frac{D_2}{\sum_{a2}} \quad (4.55)
\]

The groups considered for calculating in three cases in the form of fast neutrons and the thermal neutrons are given in the table 4.2. The fuel is used in the form of Uranium-Carbon-Fluorides mixture and the atomic configurations are given the table 4.3, and the enrichment levels considered in the case 1 is practically difficult to design a system. Since the availability of highly enriched gaseous form is difficult and the temperature control within the material contains is not possible. Since the enrichment percentage is high the core diameter is reduced to 110 cm and in the other cases the core diameter kept at 150 cm. The fuel temperature are
varied different cases, initial case is kept at 10000k and the other analysis is conducted at 4400 K, since the temperature is easily attainable and thermal-hydraulic management is comfortable. The model selected for GCR to investigate neutronics is based on mass distribution and uniform temperatures.

**Table 4.2: The Neutrons Groups Considered In the Analysis of Three Cases**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Group 1</th>
<th>Group 2</th>
<th>Group 1</th>
<th>Group 2</th>
<th>Group 1</th>
<th>Group 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>D (cm)</td>
<td>1.35</td>
<td>1.08</td>
<td>1.4</td>
<td>0.4</td>
<td>1.3</td>
<td>0.5</td>
</tr>
<tr>
<td>$\Sigma_a$ (cm$^{-1}$)</td>
<td>0.001382</td>
<td>0.0054869</td>
<td>0.01</td>
<td>0.15</td>
<td>0.008</td>
<td>0.05</td>
</tr>
<tr>
<td>$\nu$ (neutrons)</td>
<td>2.41</td>
<td>2.41</td>
<td>2.4</td>
<td>2.4</td>
<td>2.4</td>
<td>2.4</td>
</tr>
<tr>
<td>$\Sigma_f (cm^{-1})$</td>
<td>0.000242</td>
<td>0.00408</td>
<td>0.0035</td>
<td>0.1</td>
<td>0.0015</td>
<td>0.03</td>
</tr>
<tr>
<td>$\Sigma_{g, g+1} (cm^{-1})$</td>
<td>0.0023</td>
<td>0.0</td>
<td>0.01</td>
<td>0.0</td>
<td>0.01</td>
<td>0.0</td>
</tr>
<tr>
<td>$\nu$ (cm/s)</td>
<td>$3.0 \times 10^5$</td>
<td>$2.2 \times 10^5$</td>
<td>$10^5$</td>
<td>$2 \times 10^5$</td>
<td>$10^5$</td>
<td>$2 \times 10^5$</td>
</tr>
<tr>
<td>$\lambda$ (s$^{-1}$)</td>
<td>0.08</td>
<td>0.08</td>
<td>0.08</td>
<td>0.08</td>
<td>0.08</td>
<td>0.08</td>
</tr>
<tr>
<td>B</td>
<td>0.0064</td>
<td>0.0066</td>
<td>0.0075</td>
<td>0.0076</td>
<td>0.0074</td>
<td>0.0074</td>
</tr>
</tbody>
</table>

The reactivity coefficients of various properties are investigated, uniform reflector temperatures is assumed and the relation with the reflector temperature and the multiplication factor variations are studied. The model considered in the investigation is from Dam and Hoogenboom. In the analysis three different core setups are considered in analyzing the effects, the calculations are conducted using the code given in the appendix b. The enrichment levels are varied with the
notion that the gaseous form of uranium with high percentage of enrichment is a costly affair. The models are divided into three cases and here after the reference moves to case 1, case 2 and case 3. The fuel enrichment variation creates huge impact on the reactor thermal power, and the behavior of neutron and the reactor energy levels.

Table 4.3: The Core Models Used for Analysis with Change in Enrichment

<table>
<thead>
<tr>
<th>Case</th>
<th>Fuel Mixture</th>
<th>Fuel Enrichment</th>
<th>R_c (cm)</th>
<th>T_f</th>
<th>T_r</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case 1</td>
<td>0.7:0.3:4.0</td>
<td>50 %</td>
<td>118</td>
<td>10000</td>
<td>2200</td>
</tr>
<tr>
<td>Case 2</td>
<td>0.7:0.18:4.0</td>
<td>30 %</td>
<td>150</td>
<td>6400</td>
<td>1900</td>
</tr>
<tr>
<td>Case 3</td>
<td>0.7:0.18:4.0</td>
<td>5%</td>
<td>150</td>
<td>4400</td>
<td>1200</td>
</tr>
</tbody>
</table>

The neutron reflector thickness optimization and the wall temperature need to be maintained are two important aspects of the analysis. In this work graphite reflector is selected with uniform temperature density. The fuel composition used in this work is with 50 % enrichment, derived fuel from U235, the U-C-F system contains a specific percentage of UF₄ and in molar fraction of CF₄ composition. The pressure obtained in the system from the heat transfer analysis is at 100 bars and variation of pressure with temperature does not affect the criticality and neutron behavior for a specific configuration. Since the variation is throughout all the parameters of the system. Using the Monte Carlo method probability functions is generated and the Eigen values are created in the form of matrix for thermal group and also with the fast neutron group. The code follows individual neutron tracing technique and identifies the interactions.
4.4.1 CRITICAL DENSITY CALCULATION

The total atoms density of the fuel gas is to be evaluated and the critical fuel density factor need to be evaluated for all the cases and the data is read by the code from the data libraries in the form of UF₄, UF₅ and UF₆. The critical density varies with the temperature of the gas composition at which the reactor system is operated. In this evaluation three different cases are analyzed at two different temperatures. The critical density is denoted by $n_c$, in order to estimate the neutron criticality the fuel density need to be evaluated. Using the data available at the given temperature for three cases the uranium fuel density is estimated by

$$ n_u = \frac{f_U}{f_U + f_C + f_F} n_c \times 10^{30} \quad (4.56) $$

The molecular structure of the uranium and carbon and floride are bonded together to form uranium floride compositions and the floride bonding becomes more in terms of atomic ration UC₄ compound will form. The molecular fuel density is represented by $n$ and which can be found from critical density.

$$ n = n_u + n_c = \frac{f_U}{f_U + f_C + f_F} n_c \times 10^{30} \quad (4.57) $$

In evaluation of core pressure the molecular density $n$ plays an important role since $p$ is calculated from $p = nkT_\ell$, $k$ is Boltzmann Constant

4.4.2. NEUTRON MEAN FREE PATH

The source term is considered in the calculation of neutron mean free path. In GCR the length of source is distributed along the center of the reactor. Since the gaseous form of fuel is used the neutron mean free path is comparatively high with solid core reactors. In the investigation of $k_{eff}$ variations in the reactor the
calculated data supports with identifying the variations in the reactor system. The neutron mean free path length is different for the source term, leakage term, as well as absorption term. The neutron reaction rate accounts in the investigation of the multiplication factor. In the calculation of neutron mean free path four neutron energy levels are considered at 2.8 ev, 4.32 Kev, 112 keV and 20 Mev. The neutron mean free path length for the source can be calculated from the microscopic cross section.

\[
\sum_i = \frac{RR_i}{\phi} \quad (4.58)
\]

\[
l_i = \frac{1}{\sum_i} = \frac{\phi}{RR_i} \quad (4.59)
\]

4.4.3 SPECIFIC POWER DENSITY

The specific power density indicates the power produced per specific quality of fuel inside the reactor system. With the input values of the fuel temperature and the reactor core temperature distribution is useful in calculating specific power density. The values can be compared with the radial position of the core since the generation rate difference from different radial positions. The reactor radial power if given by

\[
P_s[kW/\text{kg}] = \frac{\text{Reactor.Thermal.Power}}{\text{Total.Mass.of..fissionable.material}} \quad (4.60)
\]

The specific power density is also calculated based on the specific power calculated along the radial direction of the reactor.

\[
f(r)[W/\text{kg}] = \frac{\text{Fission.Power.Density.at.r}}{\text{Uranium.Density.at.r}} \quad (4.61)
\]
The reactor core where fuel occupies the cells is divided into specific power zones and in the current analysis 10 zones are considered to calculate the total power distribution along the cells.

### 4.4.4 FUEL DENSITY REACTIVITY COEFFICIENT

In the criticality analysis the fuel density of a reactor's place an important role, since the pressure and the temperature are continuously varying with the reactivity and the number of neutrons present at a particular instant. In order to account the neutron production and the criticality the fuel density reactivity coefficient need to be calculated. The $k_{\text{eff}}$ values are calculated for corresponding fuel densities. The coefficient of fuel density is express in terms of atomic fuel density and reactivity coefficient.

\[
\alpha_n = \frac{n_f}{k} \frac{\partial k}{\partial n_f}
\]  

(4.62)

The absolute reactivity coefficient is expressed in equation 4.63, but throughout the work the normalized values are interpreted, since the units are not so reasonable for understanding.

### 4.4.5 FUEL TEMPERATURE REACTIVITY COEFFICIENT

For a reactor to operate at stable conditions the reactivity temperature should be maintained at a stable value. In gas core reactors the idea of maintaining stable fuel temperatures reactivity coefficient is to operate the reactor at a specific criticality. The reactivity coefficient is expressed as

\[
\alpha_{r_T} = \frac{\partial k}{\partial n_f}
\]  

(4.63)
In case of highly enriched fuels the reactivity coefficient varies with small changes in the reactivity, in the reactor core besides the fuel. The propellant is also entered into the reactor, but the molecular weight balances the flow and the pressure deference maintained in the process of really supports the reactivity. The velocity ratios maintained in the fuel flow rate are at 100:1, so that the fuel flow rate is regulated.

4.4.6 REFLECTOR TEMPERATURE REACTIVITY COEFFICIENT

The reflector plays an important role in reactivity inside the core; the reflector temperatres is inversely proposal to the reactor criticality. The temperatres reactivity coefficients define the stability of the reactor system; the coefficient of the reflector temperatres is given below.

\[ \alpha T_r = \frac{\partial k}{\partial T_r} \]  

The thickness of the reflector material affects the neutron absorption and the criticality of the reactor operation.

4.5. RESULT AND DISCUSSIONS

The analysis conducted for three cases at different enrichment levels to investigate the behavior of the neutrons inside the reactor and the effect of various parameters like reactivity coefficients, multiplication factors are investigated for the same geometry. The grid generated in this particular solution is of high accuracy. The results are converged at higher degrees of freedom and within less amount of time. The residual plots in calculating the keff values with respect to degree of freedom are given in the figure 4.10. In the analysis case got converged a very fast rate since the percentage of enrichment is high and the neutron
distribution is enormous in count. So the criticality values are easily above 1 so that the percentage of error in his case is comparatively less and the error level almost all went to the level 0000.2 and in the other cases the error levels are also less but in case of case three the enrichment levels are very low so the error obtained in $k_{eff}$ calculation is reached to 0.0005 and all these results are satisfactory and can be comparable with the real time models.

Figure 4.10: Residuals of the Code for Three Cases

The initialization of the solution started with the guess values from the boundary conditions inputted from the input desk file and the error obtained in the intial iterations are comparatively low. The convergence criteria is not set for the $k_{eff}$ maximum value rather it was obtained from the specific density value based on the Temprature value given as a input.
The accuracy of the solutions depends upon the time at which the relative error starts coming down and the where it stabilized. In case of high enriched uranium based case initially it went up and the value has come down very fast and at 0.25 seconds the value starts stabilizing. In case of 30 % enrichment levels the solution got stabilized with in the similar time lines. In the calculation of third case the time initially has come down and the error levels increased and then stabilized at 0.28 sec. The overall results obtained from the Monte Carlo based code are satisfactory and the results are discussed in his chapter with the merits in the reactor model.
In all the three cases the normalized specific power density with respect to the radial position is calculated as per the enrichment levels mentioned. In case of 5% enriched fuel the specific power density is comparatively low for the specified geometry configuration and it is touching 0.99 at the 100 cm radial position. In case of high enrichment levels the specific power density maintained at 0.99 at the intial stage and later stages the value started growing with the radial position. Since the enrichment levels and the atomic compositions vary in case of 30% enrichment and in case of 50% the relative effects are comparatively low in case of 50% enrichment. The radial distance place an important role in handling the heat transfer. The calculations are done for cell difference in specific power density for all the cases. The normalized values are plotted to compare all the three cases againest the change in radial position.
The fuel temperatures is calculated with respective to the radial position for three cases, in case one the fuel temperatures is maximum at 100 cm and reaching above 6800 k and the temperatures starts decline when the buffer interactions starts transferring the heat. Since the fission zone liberates the heat and transfers to the propellant in case three the fuel temperatures initially high and it starts decline when the radial position varies. In the core estimated. The figure 4.13 explains the radial position effects on fuel temperatures along the core. The idea is that when it cross the buffer zone the overall heat is effectively transferred to the propellant. The effect of fuel gas density on neutron density is calculated using the cumulative probability functions inside the core for three cases. Since the enrichment levels different in three cases the plots are generated individually, and the result are taken on a residual scale to indicate the neutron fluctuations.
Figure 4.14: Effect of fuel gas on neutron density for Case 1 with 50% enrichment

Figure 4.15: Effect of fuel gas on neutron density for Case 2 with 30% enrichment
In case 1 the enrichment levels are 50% the neutron density fluctuations are comparatively low and the peak value is obtained at 0.000315 since the criticality and the temperatures levels do effect the neutron population. In case 2 the enrichment levels considered is at 30% and the fluctuations are high since the core geometry is not changed and the maximum values reached at 0.00030 atoms/barn-cm. In case 3 the enrichment levels are maintained at 5 and which is more nearer to the real life reactor operations. The fuel gas densities variation is also an effect of radial Temprature distribution along the length of the core. Due to the neutron mean free path length the interaction can continue till the heat reservoir region.

Figure 4.16: Effect of fuel gas on neutron density for Case 3 with 5% enrichment
Figure 4.17: Variation in fuel gas density through radial position for Case 1

Figure 4.18: Variation in fuel gas density through radial position for Case 2
Figure 4.19: Variation in fuel gas density through radial position for Case 3

Variation in fuel gas density is calculated for the gas core reactor system and the results are plotted its change over radial position for three cases. In case one the critical density is considered at $3.18 \times 10^{-5}$ atoms /burn-cm and the calculations are done up to the temperatures range of 10000 K and the reactor power obtained in this case is at 962.2 Kw with a chamber pressure of 48.49 and the maximum temperatures that can be attainable under highest value of fuel gas density at60 kg/m$^3$ and tempratures resulted in 14624 K. In case of 30 % enrichment the atoms/ Burn-cm is chosen as $2.833 \times 10^{-5}$ and the fuel temperatures maintained inside the core is at 4400k at which the reactor is able to produce 104.4 kW of power at 8.477 bars of pressure and the maximum attainable temperatures at peak fuel density is 6048. In case of 5 % enrichment levels the reactor system operated at the same tempratures with critical density of $4.37 \times 10^{-4}$ atoms/ b-cm with
power levels of 40.5 and the pressure maintained is relatively high at 96.84, the peak value of the temperatures attainable at 162 kg/m³ fuel density.

Figure 4.20: Fuel Gas temperature along the radial direction of the core for case 1

Figure 4.21: Fuel Gas temperature along the radial direction of the core for case 2
Figure 4.22: Fuel Gas temperature along the radial direction of the core for case 3

The variation in fuel gas density with respect to the radial position is studied for the three cases with the variation in temperatures. The fuel gas temperatures also affects overall reactor power and the criticality of the reactor system. In case of 50% fuel enrichment the maximum fuel gas temperatures. In the initial stage the fuel enters the reactor core and the start gaining the heat due to the core conditions and starts participating in the fission reaction. In the process the fuel gas temperatures reaches the peak and then transfers the heat to the propellant and then the fragments temperatures comes down. In case one the maximum temperatures reaches to 10000 K at the center of the geometry in radial direction. In case of 30 % enrichment the level falls down to the 7000 K, in case of 5% enrichment the maximum fuel gas temperatures is in the range of 6000 K. The variation in fuel gas temperatures affects the reactivity in the core, the uniform temperature assumed in calculating neutronics by considering its peak value at a specified radial distance. In case of coupled solutions the results obtained from
the neutronics of the system can be taken in a computational fluid dynamics code to obtain complete radial distribution.

Figure 4.23: $K_{\text{eff}}$ variation with the change in Fuel Gas Density for Case 1

Figure 4.24: $K_{\text{eff}}$ variation with the change in Fuel Gas Density for Case 2
Figure 4.25: Keff variation with the change in Fuel Gas Density for Case 3

The multiplication factor keff is calculated for the relations mentioned in the methodology 4.2, and the effect of various parameters on the criticality of the reactor is studied. These effects are studied for three cases and the results are compared by curve fitting and critical points are identified in case of keff. In case 1 the critical point can be found at 0.28 specific reactivity coefficients at $3.18 \times 10^{-5}$ atoms/b-cm. These plots are used to find the critical density point by creating the linear fit for all the three cases. In case 2 the critical fuel density is at 0.31 with specific reactivity coefficients $2.833 \times 10^{-5}$ In case three the critical fuel density is at 0.18 with a specific reactivity coefficient of $4.370 \times 10^{-4}$. In all the three cases the least value is to the less enriched fuel due so that the multiplication factor also is less for reactor operated at 5% enrichment. The variation in fuel density occurs due to the change in reactor pressure and temperatures. Radius of the core varied in case of 50 % enrichment and other cases the radius considered for the core is at
150 cm, in case one it is 118 cm. In order to handle larger core the fuel need to be supplied and the reactor criticality are too high.

Figure 4.26: Effect of Reflector Temperature on $K_{eff}$ in case 1

Figure 4.27: Effect of Reflector Temperature on $K_{eff}$ in case 2
Reflector plays an important role in core design of the reactor system. In case of highly enriched fuels the thickness of the reflectors should be high, this is represented in the graph 4.30 and it illustrates for the three cases the effect of reflector thickness on the reactor criticality. The reflector temperatures play an important role in neutron absorption, in order to maintain the reactor core at the required criticality it should be operated at lower reflector temperatures. In case of gas core reactors creating a specific cooling cycle to the reflector walls improve the complexity of the system and which is not a feasible solution. In such cases the propellant flow channel is passes thought the reflector walls so that the desired reflector temperatures is maintained. In case 1 the reflector temperatures need to be maintained within the range of 1900 k to have higher keff factor since the neutron absorption leads to decrease in reactivity. In case 2 this can even manageable to reach 2200 k of the reflector temperatures. In case three the effect is not because of
the temperatures rather the most significant parameter can be reflector thickness and then the temperatures upper limit can be at 2200 K to 2400 K.

Figure 4.29: Enrichment vs. $K_{eff}$ for Representative Core

Figure 4.30: Change of $K_{eff}$ with the Increase in reflector Temperature
Figure 4.31: Change of Keft with the Increase in reactor core Temperature

The major factors that affect the reactor criticality are reactor core temperatures, fuel enrichment and the reflector temperatures. In all the three cases the results are compared with the parameters that can affect the system stability. A generalized relation is drawn to the fuel enrichment and the keff values for three cases since the fuel enrichment is different and it do effects the reactor criticality under similar radial positions. The major effect on criticality by reactor core temperatures is calculated for three cases and the results are plotted in 4.29, which represent the increase in criticality at higher reflector temperatures. In case one the peak keff value is obtained at 6000 K and the variation is very much limited till 10000 K. Since Keff is just not dependent on reactor core temperatures rather many factors like reflector temperatures and the critical fuel density also affect them. More interestingly the results indicate the enrichment levels differentiate the level of criticality reached by a reactor system with a specified operating temperatures. In case two and three the keff value of 1.02 is reached at 6000K at the same core
radius. In case 3 the enrichment level are low so the temperatures variations are very much effective.

Figure 4.32: Change of Keff with the Increase in Reflector Thickness

The material of the reflector and its thickness are two important parameters in designing a reactor core. The effective reflector thickness suggest for case 1 with graphite as a reference can be at 40 cm to 100 cm based on the reflector temperatures at which reactor core is exposed. Besides the reflector thickness the wall temperatures limitations also play an important role in neutron criticality analysis. In case 2 the reflector thickness is effected the keff and get stabilized with the maximum operational temperatures of the reactor core. In competition with the various cases with effecting parameters are giving an idea how criticality is maintained in a reactor so that higher power densities are investigated in the reactor system.
CHAPTER 5: CONCLUSIONS AND POSSIBILITIES FOR FUTURE WORK

In the previous chapters both the neutronics and heat transfer analysis results are presented with a detailed explanation. This chapter deals with the specific conclusion obtained from the work and the future developments that can be possible for GCR experimentation. The important parameters that affects the reactor criticality is studied in neutronics in a confined geometrical shape with a variation the fuel enrichment and the composition. The specific idea is to choose the best model for the interstellar travel or such kind of long distances in light years by using current technological possibilities. The kinetic heat transfer and dissociation of heat affects the stability of the reactor and the neutron generation rate. In such cases the fission reaction in a core is uncontrollable; in case of ground reactors different mechanisms are used. The core Temperature and pressure control impacts the reactor operation and its criticality.

5.1 CONCLUSIONS

The flow and criticality analysis conducted on gas core nuclear reactor to quantify the effects of fuel and propellant temperatures and densities on both the generation rate as well as on neutronics are given a particular attention. The effects are considered from an uncoupled fission zones and the maximum heat generation rates with in the design parameters. This work completely determined the fuel gas density effects and reflector Temperature variations on the criticality of a gas core nuclear reactor with an infinite length. The large reactivity coefficient of the fuel gas affected the criticality and density variations also affect the neutron flux in a given core. The reflector temperature affects the neutron spectra because of large mean free path of the GCR core. The Temperature distributions in the radial directions affect the isotropic interactions there by reactivity steps down in case of
less enriched fuels. In case of highly enriched uranium fuel the coefficient of fuel density varied from 0.36 pcm/K to -6.4 pcm/K, whereas in case of 5 % enriched fuel the variation is at 0.21 Pcm/K to the -0.3 pcm/K. This indicates in case of higher enrichment GCR core the density fluctuations create a variation in the reactivity interactions. In case of 5 % enrichment there is a sharp rise of the keff due to the density fluctuation in the reactor core and the thickness of the reflector also affected the reactivity coefficient. In case of higher enriched fuel usage with an effective thickness of graphite reflector has shown a neutron prompt removal time in 0.03 sec it may improve the reactivity and some time it can reach maximum criticality values. At the maximum thickness suggested in the results has a peak value of criticality at 0.98 and can be improved by a fraction of 0.02 with the change of material. In case of 30 % enrichment the U-C-F composition need to be taken as proposed whether the atomic ratio of U and F are high and can be a suggestible compound in the form of UF₆ so the results shown in the neutronics analysis supports the operation with higher criticality factor.

In heat transfer analysis the dissociation effects and kinetic heat transfer through convection and radiation are studied to develop conclusions on maximum heat transfer rate and the effectiveness of the core design at a given fuel enrichment. The flat temperature profile has been observed in the case of higher generation rates at 50 % fuel enrichments. The decrease in fuel enrichment effected the radial temperature distribution along the core and the maximum fuel gas temperature reached in the case of 5 % enrichment is around 4500 K with a reactor pressure of 2.4 Mpa. The difference in overall pressure and temperature distribution is not much in case of hydrogen and helium, whereas effects are more on the side of the reflector wall temperature. The temperature correction that can be made by using helium can be of 1% decrease in reflector wall temperature. The same fraction can be obtained through using different cooling channel mechanism for the input of
propellant into the core. In the benchmark calculations conducted on different generation rates resulted in creating a variation profile between the core pressure rise and the dissociation energy along the center of the core. The difference is visible in first two cases in a fraction of two and then the difference has reached of order 10. This indicates the radiative fluxes dominate the overall heat transfer in a reactor core.

5.2 DISCUSSION FOR FUTURE WORK

The code can be future developed to add one speed neutronics equation, multi group diffusion equations to solve on different geometries. Creating mode libraries to solve different grids for different geometries can be considered. A coupling can be written to analyze coupled neutronics and the thermal-Hydraulics for every fission zone; more fission groups can be analyzed with such kinds of libraries in the code. The current work focused on uranium gaseous fuel derived from U-235 in the composition of U-C-F, in the future studies there can be a great potential in comparing U-233 for nuclear rocket and conversion methods into gaseous state. In case of Am-239 very limited work is available, but there is a greater potential with the compound to develop most powerful rocket reactors. In case of reflector selection graphite was considered due to its characteristics of both reflecting ability as well as to use it as a moderator to slow down the neutrons. In case of manned missions nuclear rockets needs external radiation shields, in such cases Beo is used because it can act as a reflector and shielding material. The neutronics analysis can be conducted with a coupled approach to investigate the core behavior with the Beo reflector without having much moderating ability.

In order to develop a GCR system experimentally hydrogen, helium handling need to be studied completely for chamber parameters design in later stage
neutron investigations can be done. In the current developments only experimental facilities available for handling hydrogen till 3500 K and the where are the helium is tested experimentally at 1200 K Temprature since it has common practice for gas cooled reactors. In the development of GCR rocket systems the experimental testing on propellant behavior changes the design base done its flow properties and behavior in moderated environments. Before a rocket reactor development some work can be carried out to develop a ground based power generation system with research reactor capabilities to understand its neutronics.