MODELING OF THE ECCENTRIC CONTROL ROD IN NIGERIA RESEARCH REACTOR (NIRR-1) USING WIMS AND CITATION CODES

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JULY, 2018

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DEPARTMENT OF PHYSICS,
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AHMADU BELLO UNIVERSITY,
ZARIA, NIGERIA

JULY, 2018

DECLARATION

I, Musa Adam, declare that the work in this Thesis titled "Modeling of the Eccentric Control rod in Nigeria Research reactor (NIRR-1) using WIMS and CITATION codes"has been carried out by me in the Department of Physics. The information derived from the literature has been duly acknowledged in the text and a list of references provided. No part of this thesis was previously presented for another degree or diploma in any university or any other institution.

…… ……..

Musa Adam Date

CERTIFICATION

This thesis titled "Modeling of the Eccentric Control rod in Nigeria Research reactor (NIRR-1) using WIMS and CITATION codes" by Musa Adam meets the regulations governing the award of the degree of Master of Science (Nuclear Physics) of Ahmadu Bello University, Zaria and is approved for its contribution to knowledge and literary presentation.

…… ……
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…… ……
Head of Physics Department Date
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Dean, Post Graduate School Date
(Prof. A.Z Abubakar)

DEDICATION

Dedicated to my parentsLate*Malam Adam Musa*and*MalamaHindatuYahaya*

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I sincerely express my gratitude to ALLAH the Almighty for the gift of life He granted me as well as good health, wisdom, knowledge, understanding, finance and ability to carry out this project research work.

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ABSTRACT

In many reactors, there are both control rods and shutdown or safety rods with the latter specifically designed for rapid shutdown of the chain reaction. But Nigeria Research Reactor 1 (NIRR-1), which is a Miniature Neutron Source Reactor (MNSR) has only one control rod which serves the purpose of both reactivity control and shutdown of the reactor. This work present a modeling of the eccentric control rod (ECR), which will be an important safety measure in the MNSR and will provide another alternative way to shutdown the reactor in an emergency. The codes WIMS and CITATION were used to model the ECR within the annular beryllium reflector with cadmium as an absorber material. The following reactor core physics parameters were determined: thickness of the ECR 1.25cm, optimum location of the ECR 14.5cm, rod worth of the ECR 6.11mk, shutdown margin of the ECR 1.61mk and safety reactivity factor of the ECR 1.36. The results indicate that an additional safety rod can be used to enhance safety of the MNSR.

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CHAPTER ONE

1.0 Introduction

The initial model for the Nigeria Research Reactor (NIRR-1) core physics calculations was designed by the China Institute of Atomic Energy with computer codes HAMMER and EXTERMINATOR-2 (FSAR, 2005). There are several methods for modeling the control rods. The applied method depends on various factors such as the reactor type, the individual rod values, the available instrument and time.

The Nigeria Research Reactor-1 (NIRR-1) is a Miniature Neutron Source Reactor (MNSR) designed by China Institute of Atomic Energy (CIAE). The reactor assembly comprise of a reactor core, Beryllium reflector and a centrally located Cadmium control rod.

The Nigerian Research Reactor-1 (NIRR-1) is located in Zaria, Kaduna Stateand operated by the Center for Energy Research and Training (CERT), Ahmadu Bello University (ABU). The reactor, which was critical for the first time on 3rdFebruary, 2004 is presently in its first fuel cycle (Balogun and Jonah, 2004). It isspecifically designed for use in neutron activation analysis (NAA). It is also suitable for teaching (Jonah et. al 2006).

The control rod of the Nigeria Research Reactor-1 (NIRR-1), comprises of a cadmium absorber 266mm long and 3.9mm in diameter with stainless steel of 0.5mm thickness as the cladding material. The overall length of the control rod (CR) is 450mm in length. The function of the control rod in NIRR-1 is to serve as shim rod, regulation rod as well as safety rod. The movement of the control rod is a process that ensures the functions such as reactor start-up, steady state operation, and shutdowns are achieved. Control rods are important for maintaining the desired state of fission reactions within a nuclear reactor.

The fact that the MNSR has only one central control rod is a serious shortcoming. The present work aims to redress this shortcoming.

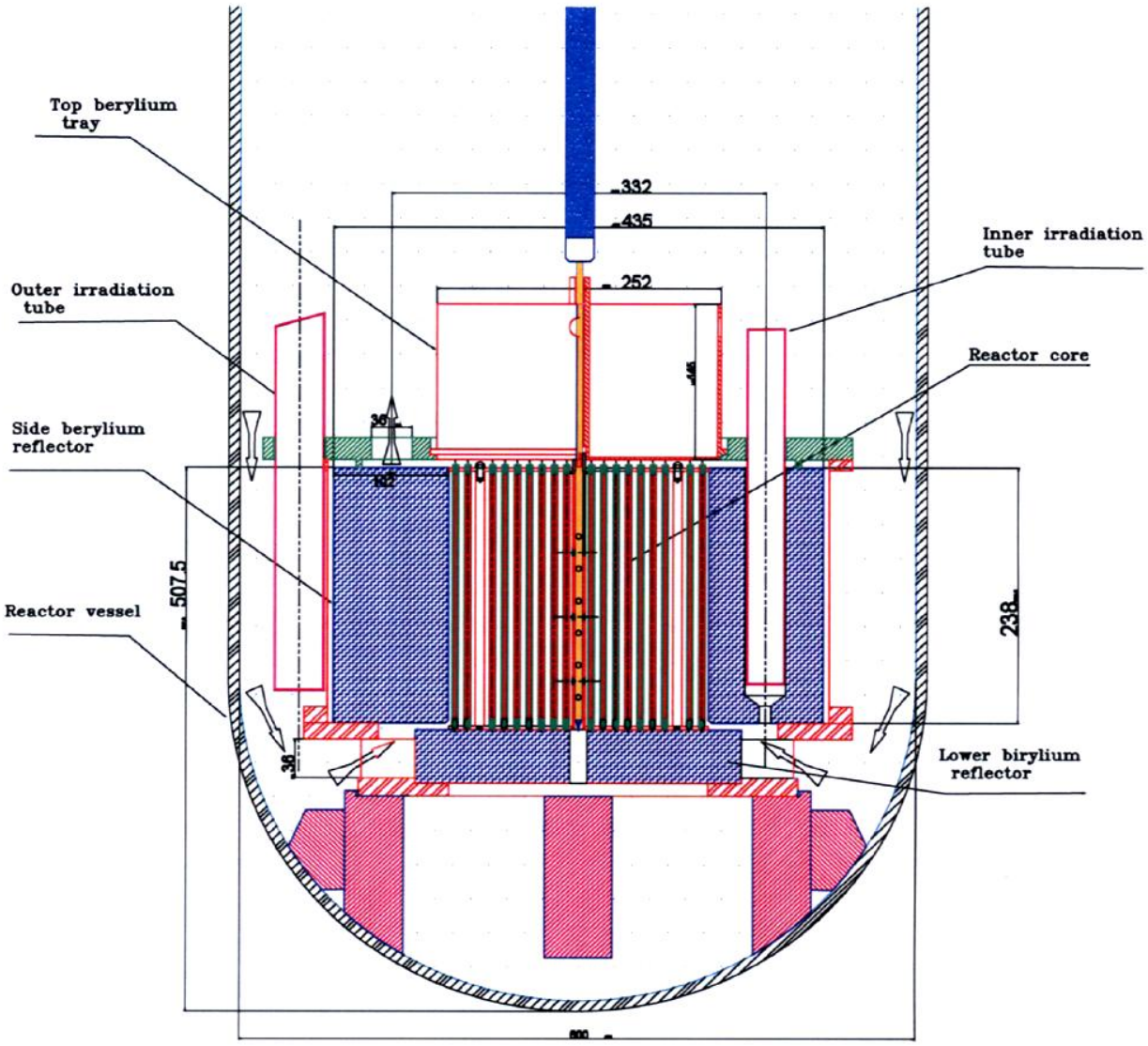


Figure 1.1: Schematic longitudinal section of MNSR vessel with fuel elements, control rod, and beryllium reflector (all dimensions are in mm).(Azande*et al*., 2010).

As the name implies, the NIRR-1 is designed mainly to serve as a neutron source. The core is a cylindrical fuel assembly containing 347 fuel elements. NIRR-1 has only one central

control rod (figure 1.2) which performs the functions of safe startup, shutdown, power regulation and reactivity control of the reactor. The reactor core assembly, surrounded by beryllium reflectors, is located at the bottom of the reactor vessel, which is suspended from the "I" beam structures that are embeddedonto the reactor pool wall and can operate for a maximum of 4.5 hours at full power(i.e. equivalent to a thermal neutron flux of $1x10^{12}$ $ncm^{-2}s^{-1}$ in the inner irradiationchannels); (Balogun and Jonah, 2004). Under these conditions, the reactor fuelloading can run for over ten years with a burn-up of less than one per cent.

The travel length of the rod in the core is 230mm. A weight of about 850g isattached to the control rod to facilitate its movement. The control rod moves througha hole in the top reflectorshim tray having a travel length of about 230mm. the total rod drop time from fully withdrawn position to fully inserted position is 26.5s.

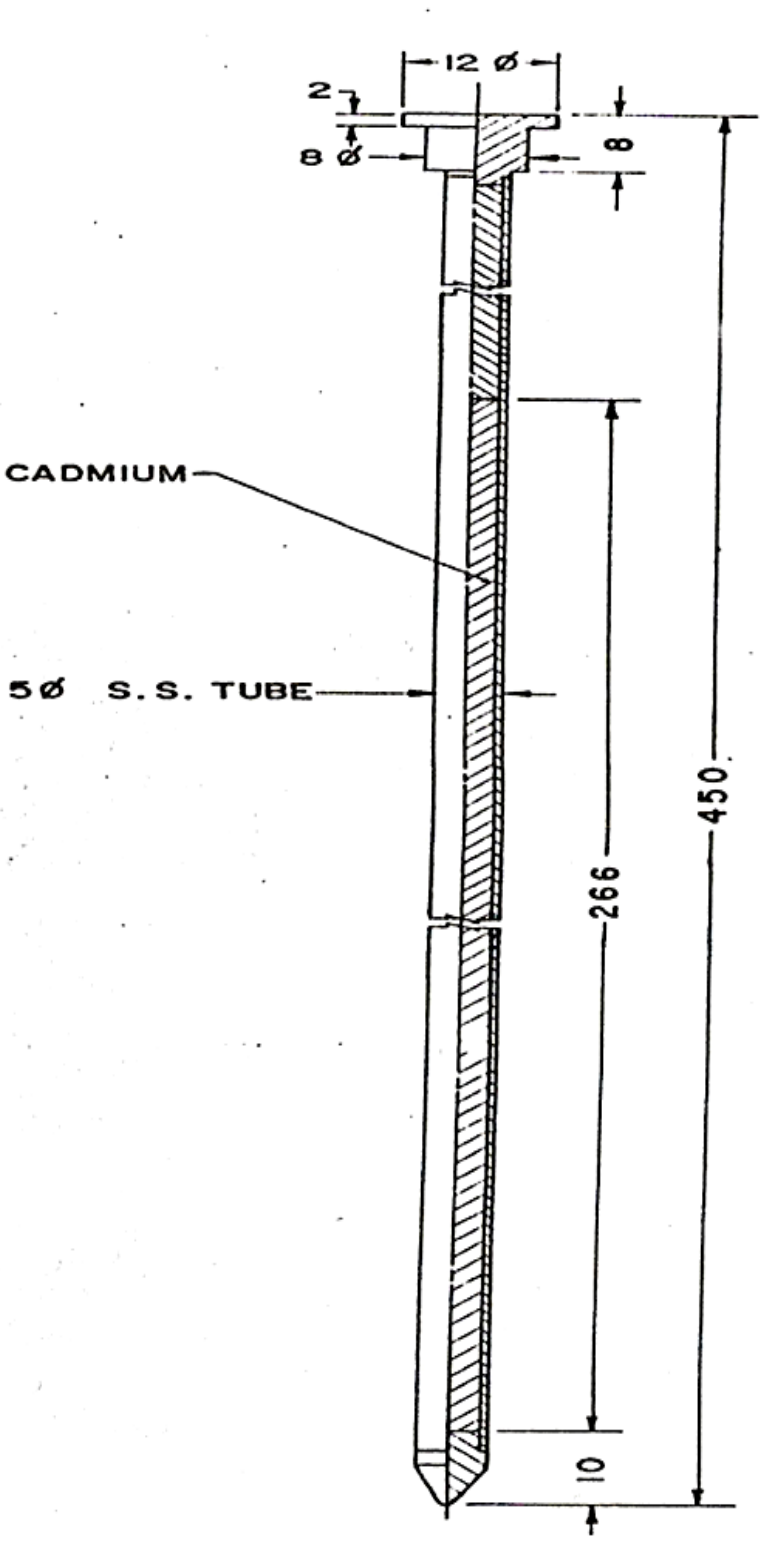


Figure 1.2: The Control Rod of Typical MNSR Reactor (dimensions in mm) (Azande*et al.*, 2010).

1.1 Research Question

- Why is it necessary to include an eccentric control rod?
- What will be the optimum location of the eccentric control rod, within the annular beryllium reflector?
- What is the optimum thickness of the eccentric control rod?

1.2 Statement of Research Problem

MNSR has only one control rod located at the centre of the reactor core, which serves for regulating the power level, starting up and shutting down the reactor. (SAR, 1992). The NIRR-1 does not have any control rod thatspecifically serve as safety rod. The International Atomic Energy Agency (IAEA) frowned at this. Therefore, the addition of the eccentric rod is expected to address this problem.

1.3Aim and Objectives

1.3.1 Aim

The aim of this research is to model the eccentric control rod in NIRR-1 using WIMS and CITATION codes.

1.3.2 Objectives

- To create a model of NIRR-1 with an eccentric control rod included.
- To determine the optimum location of the eccentric control rod
- To determine the rod worth of the eccentric control rod.

- To determine the safety reactivity factor of the eccentric control rod.

1.4 Justification of the Research

In this work, we have decided to design an additional safety rod which will serve as an important safety measure in the NIRR-1 and will provide an alternative way to shut down the reactor in an emergency situation, if the mechanism of the central control rod failed to do so.

1.5 Scope of the Study

This study focuses on safety of the NIRR-1 and covers the modeling of the eccentric control rod within annular beryllium reflector using WIMS and CITATION codes.

1.6 Previous Work

Presently, there are few works published on eccentric control rod.An online Full-scale modeling of the MNSR reactor to simulate normal operation, transients and reactivity insertion accidents under natural circulation conditions using the thermal hydraulic code Analysis of Thermal- Hydraulics Leaks and Transiants(ATHLET) was conducted by Hainoun and Alissa, (2005). And they found that the analysis that verified ATHLET input model is applicable for testing the modification of MNSR cooling system in order to improve core cooling and increase the maximum continuous reactor operation time.

Measurement of reactivity worth of TRIGA MARK-II research reactor control rods was developed by Nusrat*et.al*, 2011. Here for the TRIGA MARK-II research reactor, the reactivity measurement system was implemented with a dedicated circuit assembly and a conventional personal computer. A high-level visual basic real-time programming was

developed for data acquisition, reactivity calculation, online display (numerically as well as graphically), saving data. To measure reactivity worth of TRIGA 6 reactor control rods, the rod drop experimentaltechnique was adopted. The results of tests experiments, carried out with the rod-drop method for measuring various reactivity worth of control rods was presented in their paper. In their conclusion, they noted that the reactivity worth measurement system was tested successfully in the TRIGA Mark-II research reactor.

Boffie*et.al*,.(2012)studied the Design of an additional safety rod for Ghana Research Reactor-1 using Monte carlo N-Particle (MCNP5) code. The work presents an investigative study into thedesign of an additional safety rod, which will be an important safety component in the MNSR and willprovide another alternative way for emergency shutdown ofthe reactor.

Ibrahim *et al*., (2012), studied the Monte Carlo simulation of additional safety control rod for commercial MNSR to enhance safety using MCNP5 code. The Results obtained indicate that itwould be feasible to include two additional safety control rods to improve safety level of the MNSR.

Salawu*et.al*,. (2013) studied Designing a 2D RZ Venture Model for Neutronic Analysis of theNigeria Research Reactor-1 (NIRR-1) and they found that the control rod worth, as well as the top beryllium shim worth, were slightly higher than the experimental result.Mefarton and Collins(1982) worked on control rods in LMFBRS: using a physics assessment and they found that the Rod worth measurements inplutonium-fueled critical assemblies have been eliminated as a major sourceof design uncertainty. This situation is the result of accomplishments overthe last ten years.

Control rod worth calculation forVodo-vodyanoicEnergetichesky Reactor Russian for Water-water Power Reactor(VVER-1000) nuclear reactorusing WIMS and CITATION codes was carried out by Fadaei and Setayeshi (2009). In their study, the VVER-1000 reactor core was modeled usingWIMS-D4 and CITATION codes. Dynamic analysis for control rodworth calculation was performed by preparing software withFORTRAN program for coupling neutronic codes. Comparing between the results of this study and thereference, it was discerned that the manner of fuel assembly modelingand transport equation solving have important effects on calculationresults.

Dimitrov*et. al.*, (2013) worked on Numerical Modeling and Calculation of theFuel Cycle for the IRT-Sofia Research Reactor and concluded that the results for the effective neutron multiplicationfactor k_{eff}, obtained with MCNPX are higherthan the values calculated with REBUS-PC. Thus providing the possibility ofimproving the assessment of the IRT-Sofia reactivity balance and operating reactivitymargins.

CHAPTERTWO
THEORETICAL BACKGROUND TO THE STUDY

2.1 Control Rod in Nuclear Reactors

Control rods are important for maintaining the desired state of fission reactions within a nuclear reactor. They constitute a real-time control of the fission process, which is crucial for both keeping the fission chain reaction active and preventing it from accelerating beyond control.

The nuclear fission chain reaction is the fundamental process by which nuclear reactors produce usable energy. Most commonly, Uranium (U-235) is the fissile isotope used in this chain reaction, although the basic outline is applicable generally. In this process, a U-235 atom is struck by an incident neutron, causing the atom to fission into two smaller atoms, (example Krypton K-92 and Barium, B-141) and also release an average of 2.5 new neutrons. These new neutrons can then collide into more U-235 atoms, which undergo the same fission process, creating a chain reaction that releases substantial energy with each fission event. The key to sustaining the fission chain reaction there is the amount of neutrons that propagate to the next generation of fissions. However, not all of the fission-produced neutrons trigger another fission event (some may simply escape the reactor or be absorbed by non-fissile isotopes, for example),so it is necessary to carefully engineer every parameter of the reactor to ensure that at least one neutron from each fission event is able to triggerand otherfission.(Bodansky, 2004).

There are several ways to classify the types of control rods. One classification method is by the purpose of the control rods. Three purposes of control rods are listed below:

- Shim rods - used to remove reactivity in relatively largeamounts.
- Regulating rods - used for fine adjustments and to maintain desired power or temperature.
- Safety rods - provide a means for very fast shutdown in the event of an unsafe condition.

Addition of a large amount of negative reactivity by rapidly inserting thesafety rods is referred to as a "scram" or "trip."(DOE 1993).

Not all reactors have different control rods to serve the purposes mentioned above. Dependingupon the type of reactor and the controls necessary, it is possible to use dual-purpose or even triple-purpose rods.(DOE 1993).

2.2 Control Rod Effectiveness

The effectiveness of a control rod depends largely upon the value of the ratio of the neutron flux at the location of the rod to the average neutron flux in the reactor. The control rod has maximum effect (inserts the most negative reactivity) if it is placed in the reactor where the flux is maximum. If a reactor has only one control rod, the rod should be placed in the center of thereactor core. The effect of such a rod on the flux is illustrated in Figure 2.0.

If additional rods are added to this simple reactor, the most effective location is where the flux is maximum, that is, at point A. Numerous control rods are required for a reactor that has a large amount of excess reactivity (that amount of reactivity in excess of that needed to be critical). The exact amount of reactivity that each control rod inserts depends upon the reactor design. The change in reactivity caused by control rod motion is referred to as control rod worth.

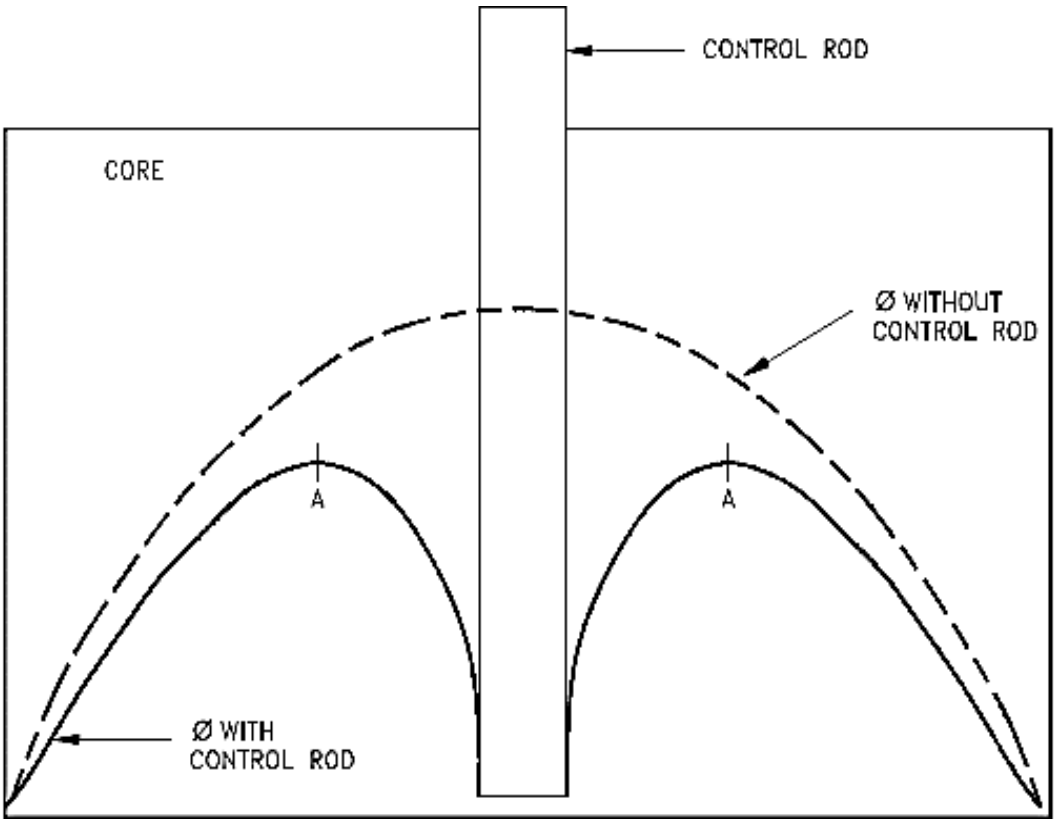


Figure 2.0 Effect of Control on Radial Flux Distribution (DOE 1993).

2.2.1 The Eccentric Control Rod

The calculation of control rod worth is somewhat more involved when the rod is not located along the axis of the reactor.(Lamarsh, 1983).

2.3Control Rod Worth

The reactivity worth of control rods which is their efficiency to absorb excess reactivity is an important parameter in the design and analysis of a nuclear reactor core (Duderstadt and Hamilton, 1997). The control rods worth is affected by their position in the core, their operational time, surrounding materials, fuel burnup as well as the concentrations of fission

products such as Xenon and Samarium. A reactors' control system has the following basic functions:

• Provide a means of starting the reactor by bringing the reactor power up to the desired level

• Maintaining the power at that level

• Shutting the reactor down for routine operations as well as in accidental conditions (Fadaei and Setayeshi, 2009).

The reactivity worth of control rods (control rod worth), ρ_W can be defined as

$$\rho_W = \rho_o - \rho_i \tag{2.1}$$

Where ρ_0 and ρ_i are the system reactivity at control rod withdrawal and insertion, respectively. This definition is mainly used, when measuring the insertion depth of control rod required to change the reactor power level. Another definition of control rod worth can be used to compensate for the reactivity change due to a long-time fuel burn up. In the former definition, there is no variation in physical properties of the core materials even though the control rods are inserted.Meanwhile, the physical properties vary, with the fuel burn up in the latter definition. But, it was eventually found that the two definitions of control rod worth were equivalent. (Oka*et al*, 2010).

2.4 Integral and Differential Control Rod Worth

2.4.1 Integral Rod Worth

The exact effect of control rods on reactivity can be determined experimentally. For example, a control rod can be withdrawn in small increments, such as 1cm, and the change in reactivity can be determined following each increment of withdrawal. By plotting the resulting reactivity versus the rod position, a graph similar to Figure 2.1 is obtained. The graph depicts integral control rod worth over the full range of withdrawal. The integral control rod worth is the total reactivity worth of the rod at that particular degree of withdrawal and is usually defined to be the greatest when the rod is fully withdrawn.

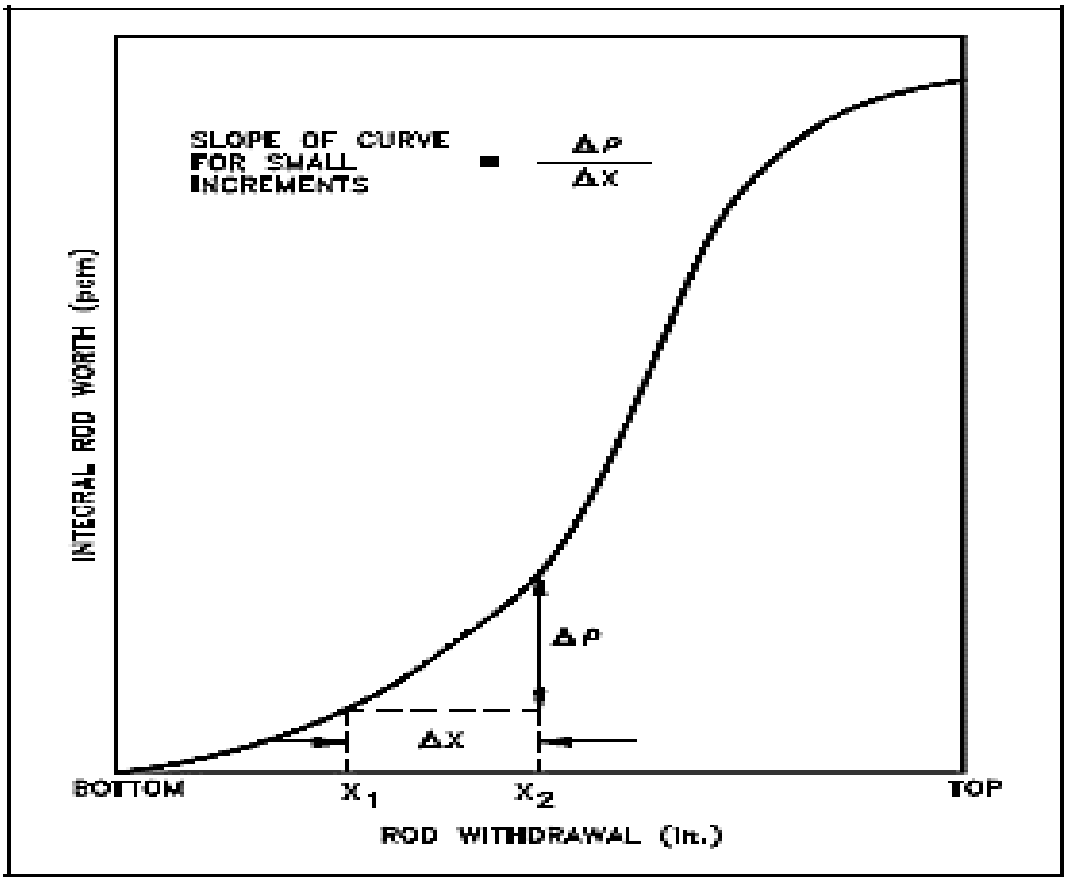


Figure 2.1 Integral Control RodWorth (Henry, 2005).

The slope of the curve ($\Delta r/\Delta x$), and therefore the amount of reactivity inserted per unit of withdrawal, is greatest when the control rod is midway out of the core. This occurs because the area of greatest neutron flux is near the center of the core; therefore, the amount of change in neutron absorption is greatest in this area. If the slope of the curve for integral rod worth in

Figure 2.1 is taken, the result is a value for rate of change of control rod worth as a function of control rod position. A plot of the slope of the integral rod worth curve, also called the differential control rod worth, is shown in Figure 2.2. At the bottom of the core, where there are few neutrons, rod movement has little effect so the change in rod worth per inch varies little.

As the rod approaches the center of the core its effect becomes greater, and the change in rod worth per inch is greater. At the center of the core the differential rod worth is greatest and varies little with rod motion. From the center of the core to the top, the rod worth per inch is basically the inverse of the rod worth per inch from the center to the bottom.

Differential control rod worth is the reactivity change per unit movement of a rod and is normally expressed as r/inch or $\Delta k/k$ per inch. The integral rod worth at a given withdrawal is merely the summation ofall the differential rod worth up to that point of withdrawal. It is also the area under the differential rod worth curve at any given withdrawal position.

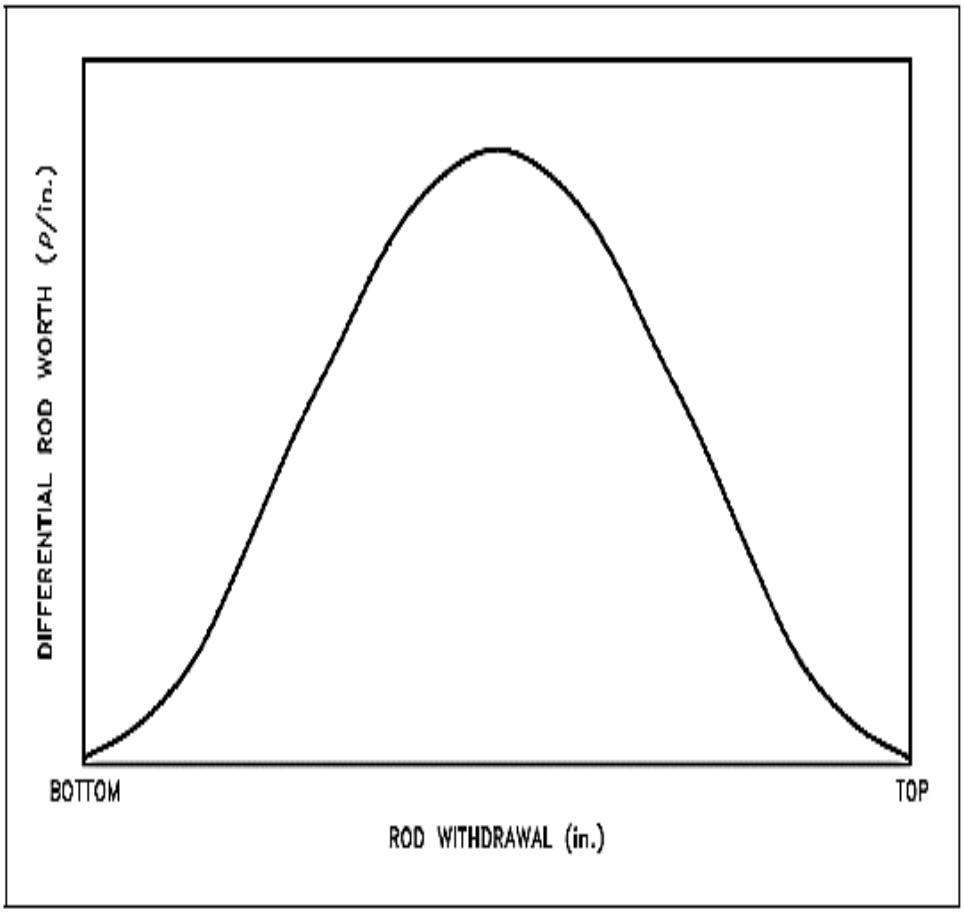


Figure 2.2 Differential control rod worth (henry, 2005).

2.5Shutdown Margin and Safety Reactivity Factor.

Shutdown marginis the instantaneous amount of reactivity by which a reactor is subcritical orwould be subcritical from its present condition assuming all control rods are fully inserted except for the single rod with the highest integral worth, which is assumed to be fully withdrawn.

Shutdown margin is required to exist at all times, even when the reactor is critical. It is important that there be enough negative reactivity capable of being inserted by the control

rods to ensure complete shutdown at all times during the core lifetime. A shutdown margin in the range of one to five percent reactivity is typically required.

The stuck rod criterion refers to the fact that the shutdown margin does not take credit for the insertion of the highest worth control rod. The application of the stuck rod criterion ensures that the failure of a single control rod will not prevent the control rod system from shutting down the reactor. (DOE, 1993).

The shutdown margin (SDM) and safety reactivity factor (SRF) are obtained from the simple formulae:

$$SDM = \rho_{cr} - \rho_{ex} \quad \text{And} \tag{2.2}$$

$$SRF = \frac{control\ \ rod\ \ wort\ h}{core\ \ excess\ \ reactivy} = \frac{\rho_{cr}}{\rho_{ex}} \tag{2.3}$$

Where

$$\rho_{cr} = \frac{k_{out} - k_{in}}{k_{out}\ k_{in}} \text{and} \rho_{ex} = \frac{k_{out} - 1}{k_{out}} \tag{2.4}$$

With k_{out} being the value of k when the control rod is fully withdrawn and k_{in} the value when the rod is fully inserted. (Balogun, 2003).

2.6 Diffusion Equation

The simplest and most widely used mathematical description of the neutron distribution in nuclear reaction, is provided by the neutron diffusion theory (Glasstone and Sesonske, 1981). For simplicity of explication, the neutrons are treated as if they are all of one effective speed, and effects associated with change in neutron energy are suppressed. Such a simplification would be justified, in practice, if the cross-section were averaged over the appropriate neutron energy distribution (Okumura *et. al.*, 2014).

2.6.1 The Group Diffusion Method

One of the most powerful ways of treating a multi-region reactor is by the group diffusion method. In this method the energies of all the neutrons are divided into a number of energy groups. The neutrons within each group are lumped together and their diffusion, scattering, absorption, and other interactions are described in terms of suitably averaged diffusion coefficients and cross sections, which are collectively known as group constant.(Lamarsh, 1983).

2.6.2 One Group of Neutrons

The simplest group diffusion problem involves only one group of neutrons, and for the purpose of illustration it will be assumed that these neutrons are all thermal neutrons. Thus, the fact that fission neutrons actually appear in a reactor at high energies, will be ignored and it will be assumed that they are emitted at thermal energies.(Lamarsh, 1983).

2.6.3 Two Group Method

The one–group method is obviously limited by the fact that all neutrons are lumped together into only one group. A more accurate procedure, particularly for thermal reactors, is to split the neutrons into two groups. In this case the thermal neutrons are included in one group called the thermal or slow group, and all other neutrons are included in the fast group. The energy separating the two groups is thus of the order of 5KT. (Lamarsh, 1983).

2.6.4 Applications of two-group method

It must be emphasized that the two group method is not an exact technique for making reactor criticality calculations. Nevertheless it does provide a relatively simple way of

obtaining reasonable estimates of critical mass or size, flux distributions, and so on. Two group calculations are generally far more realistic than one-group calculations, particularly for thermal reactors. By contrast the same degree of improvement is not obtained by going to two groups in the case of fast reactors, and first-order calculations of these systems as might well be made with one group as with two groups. Although two-group computations tend to be time consuming when done by manually, they can be carried out very rapidly on a computer. For this reason the two-group method has been widely used for parametric studies of the gross characteristics of reactor systems (Lamarsh, 1983).

2.6.5 The Multi group Method

Calculations of reactor criticality can be greatly improved by the obvious device of increasing the number of neutron groups. In this way, the slowing down of fast neutrons in thermal reactors can be treated in a far more realistic fashion than in two-group theory. With the fast and intermediate reactors, on the other hand, since it is the fast neutrons that cause the bulk of the fissions, a more detailed understanding of the behavior of the neutrons is of central importance, in predicting criticality and other reactor properties. Multi-group calculations are therefore essential for the proper design of these reactors.(Lamarsh, 1983). Therefore, the multi-group diffusion equation

$$-D_g \Delta^2 \phi_g^{(n)} + \sum_{r,g} \phi_g^{(n)} = \sum_{g'=1}^{g=1} \sum_{s,g'} \rightarrow g\phi_{g'}^{(n)} + \frac{1}{\lambda(n-1)} \chi_g F^{(n-1)} \qquad (2.5)$$

Where g=1,. Here g and n denote the energy group and the iteration index, respectively, G is the total number of energy groups, D is the diffusion constant, φ is the neutron flux, λ is the multiplication factor, χ is the fission spectrum function, υ is the number of neutrons per

fission, Σr and Σs are macroscopic removal and scattering cross sections respectively. F is the fission source and it is defined as

$$F = \sum_{g'=1}^{G} \nu_{g'} \sum_{r,g'} \phi_{g'} \tag{2.6}$$

The system of equation (2.5) is solved by fission source iteration, which starts by guessing the fission source and neutron flux of the groups. Next, the multiplication factor is calculated. This iterative strategy goes on until a required value of k_{eff} is satisfied.

2.6.6 Diffusion Coefficient

In the one-speed diffusion approximation to transport theory, it is found that the diffusion coefficient can be represented in terms of macroscopic cross sections by

$$D = \frac{1}{3(\Sigma_t - \Sigma_s \bar{\mu}_0)} \tag{2.7}$$

Where Σ_t, the total cross section is the sum of the absorption cross section, Σ_a, and the scattering cross-section, Σ_s. $\bar{\mu}_0$ is the average cosine of the neutron scattering angle for collisions in the laboratory system. The quantity in parenthesis is called the transport cross-section, designated as Σ_{tr} and its reciprocal is the transport mean free path λ_{tr}; thus:

$$\Sigma_t - \Sigma_s \bar{\mu}_0 \equiv \Sigma_{tr} = 1/\lambda_{tr} \tag{2.8}$$

So that

$$D = \frac{1}{3\Sigma_{tr}} = \tfrac{1}{3}\lambda_{tr} \tag{2.9}$$

In a weakly absorbing medium, Σ_a is small and Σ_t may be replaced by Σ_s hence we have:

$$D \approx \frac{1}{3\Sigma_s(1-\bar{\mu}_0)} = \frac{\lambda_s}{3(1-\bar{\mu}_0)} \tag{2.10}$$

Where$\lambda_s(=1/\Sigma_s)$ is the scattering mean free path of the one-speed neutrons in the diffusion medium.

2.7 Neutron Flux

Macroscopic cross sections for neutron reactions with materials determine the probability of one neutron undergoing a specific reaction per centimeter of travel through that material (DOE, 1993). If one wants to determine how many reactions will actually occur, it is necessary to know how many neutrons are traveling through the material and how many centimeters they travel each second. It is convenient to consider the number of neutrons existing in one cubic centimeter at any one instant and the total distance they travel each second while in that cubic centimeter. The number of neutrons existing in a cm^3 of material at any instant is called neutron density and is represented by the symbol n with units of neutrons/cm^3. The total distance these neutrons can travel each second will be determined by their velocity. A good way of defining neutron flux (ϕ) is to consider it to be the total path length covered by all neutrons in one cubic centimeter during one second. Mathematically, this expressed as (DOE, 1993):

$$\phi = nv \tag{2.10}$$

ϕ = neutron flux (neutrons/cm^2-sec)

n= neutron density (neutrons/cm^3)

v = neutron velocity (cm/sec)

The term neutron flux in some applications (for example, cross section measurement) is used as parallel beams of neutrons traveling in a single direction. The intensity of a neutron beam is the product of the neutron density times the average neutron velocity. The directional beam intensity is equal to the number of neutrons per unit area and time (neutrons/cm^2-sec) falling on a surface perpendicular to the direction of the beam. Since the atoms in a reactor do not interact preferentially with neutrons from any particular direction, all of these directional beams contribute to the total rate of reaction. In reality, at a given point within a reactor, neutrons will be traveling in all directions. (DOE, 1993).

2.7.1 Microscopic Cross Section

The probability of a neutron interacting with a nucleus for a particular reaction is dependent upon not only the kind of nucleus involved, but also the energy of the neutron. Consequently, the absorption of a thermal neutron in most materials is much more probable than the absorption of a fast neutron. Also, the probability of interaction will vary depending upon the type of reaction involved. (Foster and wright, 1997).

The probability of a particular reaction occurring between a neutron and a nucleus is called the microscopic cross section (σ) of the nucleus for the particular reaction. This cross section will vary with the energy of the neutron. The microscopic cross section may also be regarded as the effective area the nucleus presents to the neutron for the particular reaction and the larger the effective area, the greater the probability for the reaction (DOE, 1993).

Because the microscopic cross section is an area, it is expressed in units of area, or square centimeters. A square centimeter is tremendously large in comparison to the effective area of a nucleus, and it has been suggested that a physicist once referred to the measure of a

square centimeter as being "as big as a barn" when applied to nuclear processes. The name has persisted and microscopic cross sections are expressed in terms of barns.

Let I, beam of parallel monoenergetic neutrons per m^2 impinge perpendicularly for a given time on a thin layer, δx thick of a target material containing *N* atoms per m^3. Suppose C is the number of individual processes (e.g. neutron captures per m^2). Then the nuclear cross section σ for a specified reaction (at a given energy) is defined as the average number of individual processes occurring per target nucleus per incident neutron in the beam:

$$\sigma = \frac{C}{(N\delta x)I} \text{ m}^2/\text{nucleus} \tag{2.11}$$

Since nuclear cross sections are frequently in the range of 10^{-26} to 10^{-30} m^2 per nucleus, it is the practice to express them in terms of a unit of 10^{-28} m^2 per nucleus, called the *barn* and abbreviated by the letter b. Thus a cross section of 3.5×10^{-29} (i.e. 0.35×10^{-28}) would be called 0.35 b. The relationship between barns and cm^2 is shown below: (DOE, 1993).

$$1\text{barn} = 10^{-24}\,\text{cm}^2 \tag{2.12}$$

2.7.2 Macroscopic Cross Section

For target material with N nuclei per m^3, the quantity Nσ is equivalent to the total cross section of the nuclei per m^3; this is called the macroscopic cross section of the material for the process. It is represented by the letter Σ and given by

$$\Sigma = \text{N}\sigma \tag{2.13}$$

where

Σ = macroscopic cross section (cm^{-1})

N = atom density of material (atoms/cm^3)

σ = microscopic cross-section (cm^2)

2.8 Concept of Criticality

The state of a fission chain reaction can be concisely summarized by the effective multiplication factor, k, which indicates the change in total number of fission events during successive generations of the chain reaction. It is defined as:

$$k = \frac{total\ \#\ of\ fission\ \ events\ \ in\ a\ given\ \ geration}{total\ \#\ of fission\ \ events\ \ in\ the\ previous\ \ generation} \tag{2.14}$$

A reactor that is in a steady state (i.e. each individual fission event triggers exactly one subsequent fission event) has k = 1 and the reactor is said to be critical. Ifk< 1, the reactor issubcritical and the chain reaction cannot be sustained.Ifk> 1, the reactor is supercritical and the reaction will grow exponentially.(Bodansky, 2004).

The most important number for nuclear power reactors is therefore **1**, as any other value of the multiplication factor k implies a very unsafe reactor. Maintaining precisely k = 1 is difficult, as this precise balance is influenced by a huge number of factors. Some of these factors are inherent to the fissile fuel or reactor materials themselves, such as the number of neutronsproduced in a fission event or the amount of neutron absorption due to fuel rod casings or moderators. However, even if engineered to perfect balance initially, the multiplication factor of a reactor will necessarily vary over time, as many byproducts of the fission reaction are neutron absorbers (referred to as poison) and will lower the overall neutrons population as they accumulate.(Bodansky, 2004; larmarsh, 1983).

2.9 Reactivity and Its Relationship with Criticality

Reactivity is a basic parameter that characterizes the dynamic behavior of a reactor. Its changes in time and absolute quantity have immediate influence on reactor safety and therefore they are strictly limited (Eugene, 2003). Reactor subcriticality (i.e. size of negative reactivity before start of a reactor), maximal reactivity excess of reactor core, changes of reactivity caused by insertion or release of fuel element or experimental device and determination of weight and course of characteristics of absorption elements (scams, regulation and compensation rods) or influence of concentration of boric acid and burn-up absorbers, Practically, all these are data about reactivity that needs to be calculated theoretically in advance and verify (in most cases) practically by means of measurement (Eugene, 2003). A reactor is in the critical state if thermal neutron flux density that is invariant in time is kept in a reactor free of contribution of outside neutron source. It is important to note that criticality depends on neutron balance, not on power level; saying that a reactor is critical is not the same as saying it is at high power {CNSC, 2003). Under normal operating conditions a reactor operates close to criticality (that is, k_{eff} is nearly equal to one). It is more convenient under these circumstances to talk in terms of the amount by which k_{eff} differs from one than it is to keep quoting the value of k_{eff} itself. The effective multiplication constant is defined in terms of a detailed neutron balance everywhere in the core (CNSC, 2003).

If there are N_o neutrons in the preceding generation, then there are $N_o(k_{eff})$ neutrons in the present generation. The numerical change in neutron population is $N_o\{(k_{eff})$ -1} (DOE, 1993). Expressing the gain or loss of neutron population as a fraction of the present

generation $N_o(k_{eff})$,the reactivity (ρ) will be obtained, which is a fractional change in neutron population per generation i.e

$$\rho = \frac{\{N_o(k_{eff}) - N_o\}}{N_o(k_{eff})} \tag{2.15}$$

Where the N_o in the numerator and denominator of equation (2.15) cancels out, the reactivity is then obtained as

$$\rho = \frac{\{(k_{eff} - 1)\}}{(k_{eff})} \tag{2.16}$$

Reactivity can be defined as the deviation of a reactor from critical state (DOE, 1993). Thus the reactivity

$$\rho = \delta k_{eff} = \frac{\{(k_{eff}) - 1\}}{k_{eff}} \tag{2.17}$$

Where δk_{eff} the deviation of a reactor from its critical state and k_{eff} is the effective multiplication factor.

If the reactivity value in equation (2.16) above has a value of $\boldsymbol{\rho} < 0$ then the reactor is said to be subcritical. On the other hand, if the reactivity value, $\rho > 0$, the reactor is said to be supercritical whereas, if the reactivity value, $\rho = 0$, the reactor is said to be critical. The larger the absolute value of reactivity in a reactor core, the further the reactor is from criticality (DOE, 1993).

2.9.1 Coefficients of Reactivity

Reactivity is the most essential parameter in nuclear reactor operation. When it is positive the reactor is supercritical, zero at criticality, and negative the reactor is subcritical (Amir

and Rose Mary, 2008). What the neutron population, and consequently the reactor power are doing at any given time are determined by the amount of reactivity (ρ) in a reactor core. The reactivity can be affected by many factors (for example, fuel depletion, temperature, pressure, or poisons). (DOE, 1993).

2.9.2 Temperature Coefficient of Reactivity

Reactivity can be controlled in various ways: by adding or removing fuel; by changing the fraction of neutrons that leaks from the system; or by changing the amount of an absorber that competes with the fuel for neutrons. A change in power, for a reactor whose power increases after a certain doubling time due to control rod withdrawal, generally alters the temperatures of the fuel, moderator, and coolant. A change in temperature of any of these components causes a change in reactivity that, in turn, affects reactor operation (a feedback effect). The temperature coefficients of reactivity (the change in reactivity per unit change in temperature) determine the size and direction of the reactivity change (CNSC, 2003). In a nuclear research reactor, temperature changes can introduce reactivity changes. This property is called the "temperature coefficient of reactivity." In water-cooled nuclear reactors, the NIRR-1 for instance, the predominant reactivity changes are brought about by changes in the temperature of the coolant water. In this case the temperature coefficient is negative, which means that an increase in coolant temperature causes a decrease in reactivity, and vice-versa (Amir and Rose Mary, 2008).

Prompt and inherent reactivity feedbacks are considered important factors for research reactor safety. For the MNSR and SLOWPOKE reactors, inherent safety features are provided in their design since they are built with a negative reactor temperature coefficient.

The mechanisms that affect reactivity in these reactors are the fuel and moderator expansion, void production among others (Alhassan*et. al,* 2010).

2.9.3 Excess reactivity

Core excess is the reactivity available which is above that necessary to achieve criticality. In other words, it is what the reactivity of the system would be if all of the control rods in the reactor were completely withdrawn. The core excess of a reactor is constantly changing due to many of the variables already discussed, such as fuel depletion, temperature, and fission product poisons. It is also an important parameter which is routinely assessed to provide assurance that there is always sufficient negative reactivity in the control rods to shut the reactor down; even perhaps with one rod, or one group of rods stuck out. If the core excess is too low then it may not be possible to go critical during the xenon peak after shutdown. (DOE, 1993).

CHAPTER THREE

MATERIALS AND METHODS

3.1Materials

The following tools were used to perform the simulation:

- WIMS and CITATION
- Computer

3.1.1 Simplified Scheme of Wimsand Citation Codes

The computer codes WIMS-D4 (Winfrith Improved Multi-group Scheme), (Askew *et al*1966) and CITATION, (Fowler*et al*., 1987) are a pair of versatile reactor design tools. The input requirements for some of the applications of the thermal reactor neutronics code WIMS-D4 can be demanding to setup. The code is both powerful and flexible enabling a wide range of thermal reactor calculations to be performed. However, for standard calculations there would be some benefit to be gained from a simple user interface, where the major choices leave the user with the much simpler task of specifying basic geometrical, material and operational data. The provision of sensible defaults and ability to select from pre-checked libraries of such data would further aid WIMS user; (Balogun, 2003).

CITATION is a package designed to provide the user with framework for easily setting up WIMS models and performing sequences of calculations. The aim is for CITATION to provide a menu of basic application options for the user to select from. For each distinct type of fuel element, WIMS is asked to perform say 30 burn-up cycles through about a dozen code modules; (Balogun, 2003). In addition, at most of these cycles, cross section data are created for up to 20 variants of the standard operating conditions such as temperature changes, coolant density changes, the decay of Xe-135 due to a shut-down, or

the insertion of control rods, making in all about 600 cycles of input to WIMS. The simplified scheme is an interactive program, which prompts the user to give the names of data files which will be used in the calculation. Some of these files are provided by the CITATION package and can be used unchanged. Others can be created or modified locally for the particular application being modeled and several variants may exist from which the user will be invited to make a selection.

3.2 Methodology

A 2-D (r, z) model for NIRR-1 was developed to determine the optimum location of the eccentric control rod. In this model, the half core was divided in the r-direction into six regions which were: the central control rod, the fuel, the first part of beryllium annulus, the second part of beryllium annulus, the first part of the water and the second part of the water. The number of mesh points and the width in cm were: 5, 0.62500; 10, 10.925; 5, 3.00; 4, 7.00; 14, 30.00; 4, 10.000; respectively. The core was divided in the z-direction into eleven regions which were (from the top to the bottom): first part of the water, second part of the water, shim tray, water, the upper non-fuel, the fuel, the second part of the water, the first layer of lower non-fuel, the second layer of the lower non-fuel, the bottom beryllium and water. The number of mesh points and the width in cm were: 14, 20.000; 14, 20.500 2, 0.200; 2, 0.400; 2, 0.900; 14, 19.000; 4, 4.000; 2, 0.400; 2, 0.500; 5, 5.000; 24, 30.000; respectively. The model was run andthe values obtained from the output file were used to determine the optimum location of the eccentric control rod.

A 3-D (r, θ, z) model for the NIRR-1 wasdeveloped to designan eccentric control rod, to analyze the reactivity worth of the eccentric control rod and to investigate the shutdown

margin of the eccentric control rod. The half core was divided in the r-direction into six regions which were: the control rod, the fuel, the first part of the beryllium annulus, the second part of the beryllium annulus and the water. The number of mesh points and width in cm in each region were: 3, 0.62500; 8, 10.925; 3, 2.600; 3, 1.250; 7, 6.150; 17, 40.000; respectively. The half core was divided in the θ-direction into six regions. The number of mesh point and the angle in radians in each region were as follow: 4, 0.10606; 5, 1.04550; 4, 0.21212; 5, 0.06469; 4, 1.19290; 3, 0.52275; respectively.

Finally, the core was divided in the z-direction into ten levels and each level has six layers as shown in appendix c.

To calculate the reactivity worth of an eccentric control rod (ECR), 3-D citation input data with the CCR fully withdrawn such that the ECR was fully inserted and the value of $k(= k_{in})$ was noted; next the ECR was fully withdrawn and again the value of $k(= k_{out})$ was again noted. The ECR worth was returned as $.\frac{k_{out} - k_{in}}{k_{out}\, k_{in}}$

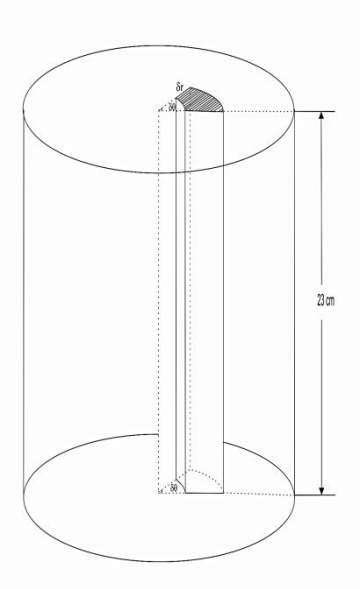


Fig.3.1Sketch of an eccentric control Rodin 3-dimention

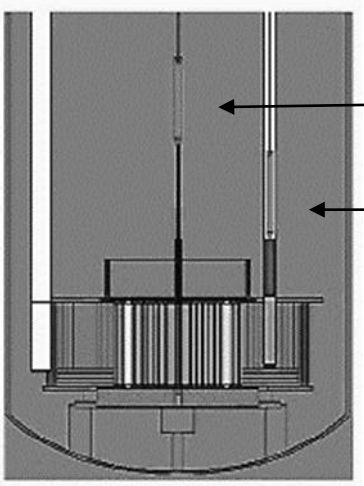


Fig 3.2 Longitudinal section of MNSR with CCR and ECR

3.3Calculation and Description of ECR

In order to calculate the cross-sectional area of the ECR we should note that the area of a sector is equivalent to the area of ECR. This implies that

Area of the sector = Area of ECR=$\frac{\theta (r_2^2 - r_1^2)}{2}$

Where $\delta r = r_2 - r_1 = 1.25$cm and $\delta\theta = 0.06469$rad

Therefore Area of ECR =$\frac{0.06469\,(15.8^2 - 14.55^2)}{2}$

Area of ECR = 1.227cm^2.

The reactor was modified to include one additional safety rod with cadmium as an absorber material and stainless steel as clad. The safety rod inserted in the inner irradiation site located at angle 0.065rad. The inner irradiation channels are place 40mm above the bottom of the core through which the ECR move to cover a distance of 190mm (i.e 230-40mm) in the core. The total vertical distance travelled by the ECR is 192mm. The cross-sectional area of ECR is 1.227cm^2 and the change in radius of 1.25cm. TheSketch ofthe ECR in 3-dimention andlongitudinal section of MNSR with CCR and ECR as modeled using WIMS and CITATION codes is shown in Fig. 3.1 and 3.2.

CHAPTER FOUR
RESULTS AND DISCUSSION

The result of the parameter calculated using WIMS and CITATION on modeling the eccentric control rod to improve safety of NIRR-1 is presented in Table 4.1.

Table 4.1: Calculated parameter values for ECR

Parameter	This work	GHARR-1	Others
Thickness of ECR	1.25cm	……	----------
Optimum location of ECR	14.55cm	--------	16.5cm (SAR, 2005)
Control rod worth of ECR	6.11mk	5.90mk	6.16mk NIRR-1 (Jonah, 2014)
Excess reactivity of ECR	4.5mk	3.93mk	--------
Shutdown margin of ECR	1.61mk	1.97mk	1.44mk NIRR-1 (Jonah, 2014)
Safety reactivity factor of ECR	1.36	1.5	----------

Data for the neutron flux against radial distance, (Appendix A) is presented in 4.1

The plot of Table 4.2 is depicted in fig. 4.1

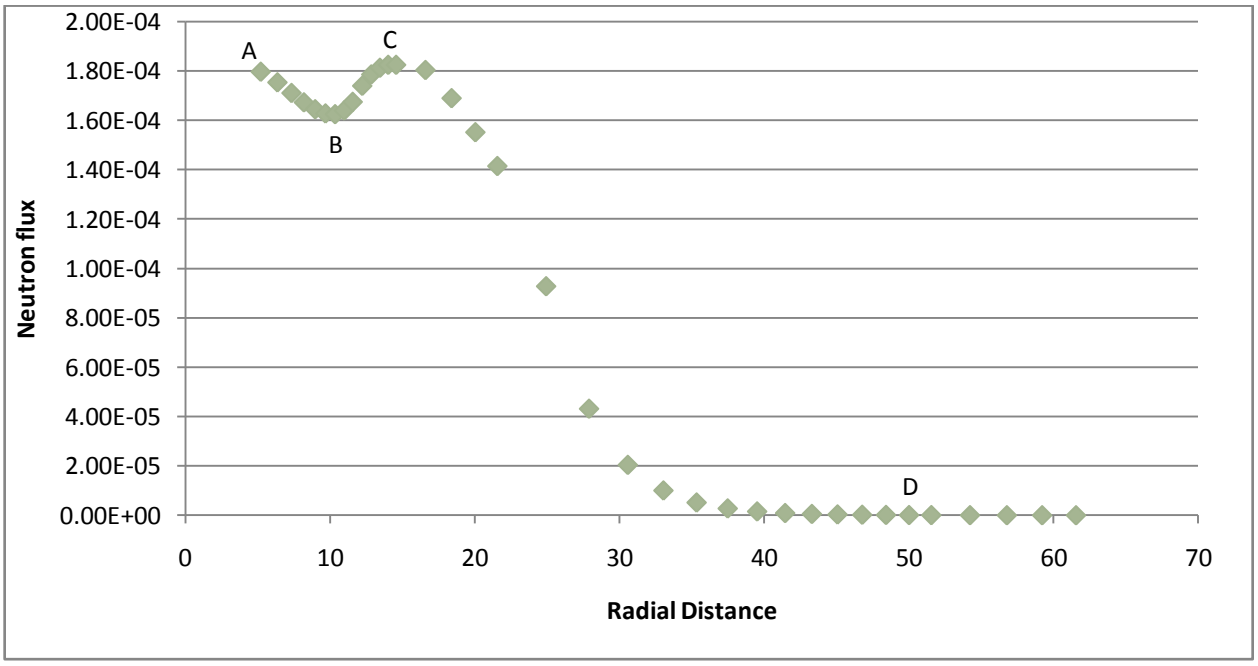


Fig. 4.1 Flux distribution in radial direction

. The optimum location of the ECR was found to be 14.55cm which is not in agreement with the value of 16.5cm reported in the NIRR-1 Safety Analysis Report (SAR, 2005).

From available literature, there was no publication on the methodology of the calculation of the value of 16.5cm as contained in the (SAR, 2005). Careful observation of Fig.4.1 clearly show the neutron flux from the centre of the core, which is point A decreases steadily to a minimum point B which is the outer edge of the core and then the neutron fluxrises steadily to maximum point C which corresponds to the optimum location of the ECR within the annular beryllium reflector. Finally, the flux declined to saturation at D. This result is in good agreement with the work of (Balogun *et., al.* 2005), who reported a decline in neutron flux within the beryllium annulus which is expected because of the removal processes, whereby high-energy neutron slow down towards thermal energies, iscontinuously taking place within the beryllium annulus.

4.1Significance of the results

Generally, the 2-D and 3-D models developed for the NIRR-1 have been used to model the eccentric control rod. In the 3-D, the rod worth of ECR decreases from the edge of the core to the outer edge of the annular beryllium reflector.This implies that, there was nopeak position where the ECR can be inserted.

However, 3-D is very effective in calculating the excess reactivity, SDM and SRF. While 2-D is also very effective in determining the optimum location of the ECR position.

4.1.1 Excess reactivity

The NIRR-1 cold excess reactivity measured at the time of commissioning was determined to be 4.97mk, which is more than the licensed range of (3.5- 4.0) mk. Hence a Cadmium

poison of worth $-1.2\ mk$ was inserted into one of the inner irradiation sites which act as reactivity regulating device in order to reduce the core excess reactivity to 3.77mk, which is within the recommended licensed range of (3.5- 4.0) mk. The core excess reactivity of 4.5mk is very close to the measured value.

4.1.2 Safety reactivity factor

On safety criteria,the Safety Reactivity Factor (SRF) of 1.36mk implies that the greater the ratioof the control rod worth to the core excess reactivity is, the better and safer the reactor would be.This allows for a wider shutdown margin enabling the reactor to be graduallyshutdown when in operation.

CHAPTER FIVE

CONCLUSION AND RECOMMENDATION

5.1 Conclusion

In this research, modeling of the eccentric control rod in NIRR-1 has been carried out using WIMS and CITATION codes. The ECR excess reactivity of 4.5mk and ECR safety reactivity factor of 1.36 were obtained and both were comparable to the valuesof 3.93mk and 1.5 respectively obtained by Boffies*et.al.* (2012),while the ECR control rod worth of 6.11mk and ECR shutdown margin of 1.61mk were in conformity with the values of 6.16mk and 1.44mk respectively reported by Jonah, (2014).

More so, the value of 14.55cm obtained as the optimum location of the ECR in this work needed to be revalidated using other deterministic codes as the value was not in agreement with the NIRR-1 Safety Analysis Report 2005 (SAR-2005).

5.2 Recommendations

1 A mechanical device should be design for inserting the ECR into the reactor during emergencies.

2 The effect of the control rod weight on the mechanical component should be carefully studied.

3 The construction and installation of the ECR is recommended.

4 Inthe feature design of an MNSR, the calculation of the optimum location ofECRneed to berevalidated using different codes such as MCNP5 code, MCNPX-2.7.0 code, VENTURE-PC code and SCALE 6.1 code.

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APPENDIX A

Data for the neutron flux against radial distance

Neutron Flux	Rod Distance
1.80E-04	5.195
1.75E-04	6.348
1.71E-04	7.321
1.67E-04	8.179
1.65E-04	8.955
1.63E-04	9.669
1.63E-04	10.334
1.64E-04	10.959
1.68E-04	11.55
1.74E-04	12.209
1.79E-04	12.834
1.81E-04	13.431
1.83E-04	14.002
1.83E-04	14.55
1.80E-04	16.579
1.69E-04	18.386
1.55E-04	20.031
1.42E-04	21.55
9.28E-05	24.921
4.32E-05	27.887
2.04E-05	30.567
1.01E-05	33.03
5.18E-06	35.322
2.79E-06	37.473
1.56E-06	39.508
9.08E-07	41.443
5.46E-07	43.292
3.38E-07	45.065
2.15E-07	46.77
1.40E-07	48.416
9.34E-08	50.008
6.36E-08	51.55
3.66E-08	54.223
1.91E-08	56.771
9.56E-09	59.209
3.40E-09	61.55

APPENDIX B

** CITATION ** INPUT DATA FOR NIRR-1 RESEARCH REACTOR

TWO DIMENSIONAL R-Z GEOMETRY

001

```
 0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  1  0  4  0  0

 1  0  1  1  1  1  0  0  1  1  0  1  1  1  1  0  0  0  0  0  0  0  0

550                                        60    60

 1.5

003

 0  0  0  0  7  0  0  0  0  0  1  0  0  0  0  0

     .0001      0.0001              0.0001

.4692       0.4692      0.3       1.0

004

5  .62500  10 10.925    5  3.00     4  7.00    14  30.00    4  10.000

14  20.000  14 20.500    2  0.200    2  0.400    2   0.900  14  19.000

4  4.000    2  0.400    2  0.500    5  5.000   24  30.000

005

2  2  2  2  2  2
```

10 2 2 2 2 2

10 5 2 2 2 2

10 2 2 2 2 2

10 6 2 2 2 2

2 3 4 4 2 2

2 3 4 4 2 2

7 6 4 4 2 2

7 6 2 2 2 2

2 4 4 2 2 2

2 2 2 2 2 2

008

4 3 1

10 1 2.41207E+00 7.88399E-03 6.31941E-20

8.9558E-02 6.3211E-02 2.1822E-04 0.0000E+00

10 2 1.53515E+00 4.27834E-03 1.79990E-20

0.0000E+00 2.2360E-01 4.6835E-02 4.4852E-06

10 3 2.28844E+00 1.41589E-02 1.49671E-21

0.0000E+00 0.0000E+00 2.8710E-01 3.7316E-02

10 4 2.31641E+00 3.79598E-01 2.86356E-24

0.0000E+00 0.0000E+00 5.7308E-03 4.4936E-01

2 1 2.31474E+00 5.23423E-04 2.90232E-12

3.7445E-02 1.0538E-01 6.5618E-04 0.0000E+00

2 2 7.82476E-01 2.18910E-07 2.50896E-12

0.0000E+00 2.7873E-01 1.4726E-01 1.4398E-05

2 3 5.59551E-01 9.82805E-04 2.50711E-11

0.0000E+00 0.0000E+00 4.4785E-01 1.4688E-01

2 4 1.52079E-01 1.90763E-02 3.61806E-10

0.0000E+00 0.0000E+00 7.0056E-05 2.1727E+00

3 1 1.88301E+00 1.01817E-03 1.09486E-03

3.0825E-01 1.8712E-01 6.3132E-04 0.0000E+00

3 2 8.03203E-01 8.00045E-04 1.32638E-03

0.0000E+00 6.1492E-01 1.3281E-01 1.2224E-05

3 3 7.98725E-01 1.30986E-02 1.43315E-02

0.0000E+00 0.0000E+00 6.9401E-01 1.1784E-01

3 4 4.40473E-01 1.22843E-01 1.87736E-01

0.0000E+00 0.0000E+00 6.5989E-04 1.7909E+00

4 1 1.46108E+00-6.74210E-03 2.53996E-12

1.6395E-01 7.0933E-02 1.2198E-16 0.0000E+00

4 2 5.78656E-01 1.55444E-07 1.62893E-12

0.0000E+00 5.4857E-01 2.7478E-02 0.0000E+00

4 3 4.83681E-01 3.44555E-05 1.48119E-11

0.0000E+00 0.0000E+00 6.7205E-01 1.7071E-02

4 4 4.35424E-01 6.61386E-04 2.62700E-10

0.0000E+00 0.0000E+00 1.1841E-05 7.6486E-01

5 1 2.75003E+00 1.65147E-04 1.25887E-12

9.9553E-02 2.1492E-02 6.1437E-17 0.0000E+00

5 2 2.83584E+00 2.98368E-04 1.25616E-12

0.0000E+00 1.1603E-01 1.2154E-03 0.0000E+00

5 3 4.02073E+00 9.03633E-04 1.06141E-11

0.0000E+00 0.0000E+00 8.1715E-02 2.8526E-04

5 4 3.82302E+00 4.73577E-03 9.24719E-11

0.0000E+00 0.0000E+00 4.7915E-04 8.1976E-02

6 1 2.45647E+00 4.69608E-04 1.39361E-08

4.9364E-02 8.7307E-02 5.2105E-04 0.0000E+00

6 2 1.08610E+00 3.61085E-05 1.20967E-08

0.0000E+00 2.5165E-01 1.1644E-01 1.1366E-05

6 3 1.27043E+00 1.09886E-03 1.58643E-07

0.0000E+00 0.0000E+00 3.7653E-01 1.1698E-01

6 4 8.36959E-01 1.82978E-02 3.09132E-06

0.0000E+00 0.0000E+00 7.0252E-05 1.7409E+00

7 1 2.52241E+00 3.78575E-04 7.37550E-13

6.0937E-02 7.2772E-02 4.0621E-04 0.0000E+00

7 2 1.45448E+00 8.11154E-05 2.21958E-13

0.0000E+00 2.1782E-01 7.8261E-02 7.6137E-06

7 3 2.25705E+00 1.01120E-03 4.17746E-13

0.0000E+00 0.0000E+00 2.6807E-01 7.4881E-02

7 4 1.79423E+00 1.53062E-02 7.82986E-12

0.0000E+00 0.0000E+00 6.3911E-05 1.1280E+00

0

7.452795E-012.545717E-011.488919E-040.000000E+00

999

Our own fuel for NIRR-1 follws

3 1 1.88301E+00 1.01817E-03 1.09486E-03

3.0825E-01 1.8712E-01 6.3132E-04 0.0000E+00

3 2 8.03203E-01 8.00045E-04 1.32638E-03

0.0000E+00 6.1492E-01 1.3281E-01 1.2224E-05

3 3 7.98725E-01 1.30986E-02 1.43315E-02

0.0000E+00 0.0000E+00 6.9401E-01 1.1784E-01

3 4 4.40473E-01 1.22843E-01 1.87736E-01

0.0000E+00 0.0000E+00 6.5989E-04 1.7909E+00

APPENDIX C

**** TEST FILE 1/2 CORE, TEST FILE DATED 1ST APRIL 2008 ****

**** 3-D MODELLING OF NIRR-1 REACTOR CORE, ****

```
001

 0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  1  0  6  0  0  0

 1  0  0  0  0  0  0  0  1  0  0  0  0  0  0  0  0  0  0  0  0  0  0  1

600

        1.0E-03

003                        L  T  R  B  F  Bk

 0  0  0  0 12  0  0  1  1  0  1  0  1  0  0  0  0  0  0  0  0  0  0  0



                  1.000000E+15 1.0000000  0.5000000000

004

 4  0.10606  5  1.04550  4  0.21212  5  0.06469  4  1.19290  3  0.52275

3  .62500   8 10.925    3  2.600    3  1.250    7  6.150   17 40.000

 14 20.500    2  0.200    2  0.400    3   0.900  12  19.000   5   4.000

2  0.400    3  0.500    5  5.000   10  10.000

005
```

2 2 2 2 2 2 Level 1----------

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2 ----------

2 2 2 2 2 2 Level 2----------

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2 ----------

2 2 2 2 2 2 Level 3----------

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2 ----------

2 2 2 2 2 2 Level 4----------

6 6 6 6 6 6

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2 ----------

2 2 2 2 2 2 Level 5----------

3 3 3 3 3 3

4 4 4 4 4 4

4 4 4 10 4 4

4 4 4 4 4 4

2 2 2 2 2 2 ----------

2 2 2 2 2 2 Level 6----------

3 3 3 3 3 3

4 4 4 4 4 4

4 4 4 10 4 4

4 4 4 4 4 4

2 2 2 2 2 2 ----------

7 7 7 7 7 7 Level 7----------

6 6 6 6 6 6

4 4 4 4 4 4

4 4 4 4 4 4

2 2 2 2 2 2

2 2 2 2 2 2 ----------

7 7 7 7 7 7 Level 8----------

6 6 6 6 6 6

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2 ----------

2 2 2 2 2 2 Level 9----------

4 4 4 4 4 4

4 4 4 4 4 4

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2 ----------

2 2 2 2 2 2 Level 10----------

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2

2 2 2 2 2 2 ----------

008

4 2 1

2 1 2.31474E+00 5.23423E-04 2.90232E-12

3.7445E-02 1.0538E-01 6.5618E-04 0.0000E+00

2 2 7.82476E-01 2.18910E-07 2.50896E-12

0.0000E+00 2.7873E-01 1.4726E-01 1.4398E-05

2 3 5.59551E-01 9.82805E-04 2.50711E-11

0.0000E+00 0.0000E+00 4.4785E-01 1.4688E-01

2 4 1.52079E-01 1.90763E-02 3.61806E-10

0.0000E+00 0.0000E+00 7.0056E-05 2.1727E+00

3 1 1.88301E+00 1.01817E-03 1.09486E-03

3.0825E-01 1.8712E-01 6.3132E-04 0.0000E+00

3 2 8.03203E-01 8.00045E-04 1.32638E-03

0.0000E+00 6.1492E-01 1.3281E-01 1.2224E-05

3 3 7.98725E-01 1.30986E-02 1.43315E-02

0.0000E+00 0.0000E+00 6.9401E-01 1.1784E-01

3 4 4.40473E-01 1.22843E-01 1.87736E-01

0.0000E+00 0.0000E+00 6.5989E-04 1.7909E+00

4 1 1.46108E+00-6.74210E-03 2.53996E-12

1.6395E-01 7.0933E-02 1.2198E-16 0.0000E+00

4 2 5.78656E-01 1.55444E-07 1.62893E-12

0.0000E+00 5.4857E-01 2.7478E-02 0.0000E+00

4 3 4.83681E-01 3.44555E-05 1.48119E-11

0.0000E+00 0.0000E+00 6.7205E-01 1.7071E-02

4 4 4.35424E-01 6.61386E-04 2.62700E-10

0.0000E+00 0.0000E+00 1.1841E-05 7.6486E-01

6 1 2.45647E+00 4.69608E-04 1.39361E-08

4.9364E-02 8.7307E-02 5.2105E-04 0.0000E+00

6 2 1.08610E+00 3.61085E-05 1.20967E-08

0.0000E+00 2.5165E-01 1.1644E-01 1.1366E-05

6 3 1.27043E+00 1.09886E-03 1.58643E-07

0.0000E+00 0.0000E+00 3.7653E-01 1.1698E-01

6 4 8.36959E-01 1.82978E-02 3.09132E-06

0.0000E+00 0.0000E+00 7.0252E-05 1.7409E+00

7 1 2.52241E+00 3.78575E-04 7.37550E-13

6.0937E-02 7.2772E-02 4.0621E-04 0.0000E+00

7 2 1.45448E+00 8.11154E-05 2.21958E-13

0.0000E+00 2.1782E-01 7.8261E-02 7.6137E-06

7 3 2.25705E+00 1.01120E-03 4.17746E-13

0.0000E+00 0.0000E+00 2.6807E-01 7.4881E-02

7 4 1.79423E+00 1.53062E-02 7.82986E-12

0.0000E+00 0.0000E+00 6.3911E-05 1.1280E+00

10 1 2.41207E+00 7.88399E-03 6.31941E-20

8.9558E-02 6.3211E-02 2.1822E-04 0.0000E+00

10 2 1.53515E+00 4.27834E-03 1.79990E-20

0.0000E+00 2.2360E-01 4.6835E-02 4.4852E-06

10 3 2.28844E+00 1.41589E-02 1.49671E-21

0.0000E+00 0.0000E+00 2.8710E-01 3.7316E-02

10 4 2.31641E+00 3.79598E-01 2.86356E-24

0.0000E+00 0.0000E+00 5.7308E-03 4.4936E-01

0

7.452795E-012.545717E-011.488919E-040.000000E+00

999

cr constant

10 1 2.41207E+00 7.88399E-03 6.31941E-20

8.9558E-02 6.3211E-02 2.1822E-04 0.0000E+00

10 2 1.53515E+00 4.27834E-03 1.79990E-20

0.0000E+00 2.2360E-01 4.6835E-02 4.4852E-06

10 3 2.28844E+00 1.41589E-02 1.49671E-21

0.0000E+00 0.0000E+00 2.8710E-01 3.7316E-02

10 4 2.31641E+00 3.79598E-01 2.86356E-24

0.0000E+00 0.0000E+00 5.7308E-03 4.4936E-01

Following Group Constants (10 and 5) removed from Fresh3d2.cii :-

10 1 2.41207E+00 7.88399E-03 6.31941E-20

8.9558E-02 6.3211E-02 2.1822E-04 0.0000E+00

10 2 1.53515E+00 4.27834E-03 1.79990E-20

0.0000E+00 2.2360E-01 4.6835E-02 4.4852E-06

10 3 2.28844E+00 1.41589E-02 1.49671E-21

0.0000E+00 0.0000E+00 2.8710E-01 3.7316E-02

10 4 2.31641E+00 3.79598E-01 2.86356E-24

0.0000E+00 0.0000E+00 5.7308E-03 4.4936E-01

5 1 2.75003E+00 1.65147E-04 1.25887E-12

9.9553E-02 2.1492E-02 6.1437E-17 0.0000E+00

5 2 2.83584E+00 2.98368E-04 1.25616E-12

0.0000E+00 1.1603E-01 1.2154E-03 0.0000E+00

5 3 4.02073E+00 9.03633E-04 1.06141E-11

0.0000E+00 0.0000E+00 8.1715E-02 2.8526E-04

5 4 3.82302E+00 4.73577E-03 9.24719E-11

0.0000E+00 0.0000E+00 4.7915E-04 8.1976E-02