CONTROL ROD CALIBRATION OF NIGERIA RESEARCH REACTOR -1 (NIRR-1) USING POSITIVE PERIOD METHOD

BY

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DECLARATION

I, Abdulsamad ASUU, declare that the work in this Thesis titled **"control rod calibration of Nigeria research reactor -1 (NIRR-1) using Positive Period method"** 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 at this or any other institution.

…… ……..

Asuku Abdulsamad Date

CERTIFICATION

This thesis titled "CONTROL ROD CALIBRATION OF NIGERIA RESEARCH REACTOR -1 (NIRR-1) USING POSITIVE PERIOD METHOD" by Abdulsamad ASUKU meets the regulations governing the award of the degree of Masters of Science (Nuclear Physics) of Ahmadu Bello University, and is approved for its contribution to knowledge and literary presentation.

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DEDICATION

To

my father, Cpl. Asuku Yisa

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ABSTRACT

Calibration of control rod has become an essential tool for measuring the reactivity changes in a reactor. The Positive Period Method of control rod calibration was shown to be a relatively independent, more accurate and well suited to the core configuration and control rod mechanism of low power reactors. The Nigeria Research Reactor-1 (NIRR-1) is a low power Miniature Neutron Source Reactor (MNSR). In this work, NIRR-1 control rod was calibrated from 107mm to 177mm withdrawal length at a low flux of 5×10^9 ncm^2s^{-1}. This flux and length was chosen to contain effects on reactivity due to temperature variation. Doubling times measurements were carried out in conjunction with the "Inhour" data to obtain reactor periods and their corresponding reactivities. The integral reactivities obtained were plotted against control rod withdrawal positions. The plot was compared with the s-shape integral curve earlier obtained for NIRR-1 during the zero power experiments in China and the two figures were found to be in agreement. The integral rod worth of (3.34 ± 0.11) mk obtained here was also compared with the in-built full rod length total reactivity worth of 7mk and was found to have accounted for 47.8% of the length. This result is an indication that 10mm withdrawal length is approximately equivalent to 0.56mk reactivity insertion and the length used here is the most reactive portion of the control rod. It also shows that only 30% of the control rod length is required to achieve the characteristic s-shape curve using the positive period method. The uncertainty of 3.29% obtained in the integral rod worth is found to be within the margin of experimental results. This has satisfied the safety requirements of MNSR for shutdown margin. Our work has demonstrated that routine calibration of the MNSR control rod via positive period method is possible and satisfied all safety requirements, and thus recommended.

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NOMENCLATURE

β_{eff} = Effective delayed neutron fraction

α_i = Fraction of the delayed neutrons that are of the i^{th} kind

C = Effective beryllium photoneutron contribution factor

β_γ = Effective delayed beryllium photoneutron fraction

α_j = fraction of the delayed beryllium photoneutrons that are of the j^{th} kind.

ϕ = Neutron flux (neutrons/cm^2-sec)

n = neutron density (neutrons/cm^3)

v = Neutron velocity (cm/sec)

$\sum_f$ = Macroscopic fission cross-section of the core fuel = $1.013x10^{-2}$

T = Reactor period in (s)

ρ = Reactivity (mK)

β_i = Delayed neutron fraction of the i^{th} delayed group

l = Prompt neutron generation time

λ_i = Decay constant of the i^{th} delayed group

C_i = Concentration of the i^{th} precursor

CHAPTER ONE

INTRODUCTION

1.1 INTRODUCTION

The central characteristics of a nuclear reactor are its controllable, self-sustaining fission chain reaction which releases useful neutrons and energy (CNSC, 2003). This controllability was shown to be greatly achieved via control rod motion and as such the knowledge of a reactor's response to specific control rod motions is essential for the safety, efficiency, operation and durability of a nuclear research reactor (Souza and Mesquita, 2010).

Research reactors are nuclear reactors that are majorly neutron source. Also called non-power reactors, the major contrasts between them and power reactors are that power reactors are utilized for electricity production, heat generation, or maritime propulsion while the neutrons produced by a research reactor are used for several purpose which include scattering, non-destructive testing, analysis and testing of materials, production of radioisotopes, research and public outreach and education. Research reactors comprise a wide range of civil and commercial nuclear reactors which are generally not used for power generation (Ahmed *et al*, 2006). Over 270 research reactors are currently operating in more than 50 countries (Parrish and Nicholas, 2007). Research reactors are separated into various classes of which one of such is the Miniature Neutron Source Reactor (MNSR) which the Nigeria Research Reactor-1 (NIRR-1) fall under.

The Nigeria Research Reactor-1 (NIRR-1), is the 8th commercial Miniature Neutron Source Reactor (MNSR) designed by China Institute of Atomic Energy (CIAE). First criticality was achieved on 03 February 2004. It is particularly designed for use in neutron activation analysis (NAA) and limited radioisotope production. 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 230mm long and 3.9mm in diameter with stainless steel of 0.5mm thickness as the cladding material. The length of the control rod (CR) is 230 mm 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 that functions such as reactor start-up, steady state operation, and shutdown are achieved (Ahmed *et al.,* 2006). Control rods are an important technology for maintaining the desired state of fission reactions within a nuclear research 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 (James, 2011).

Generally, control rod movements are used to compensate for other reactivity changes resulting from fuel burn-up, temperature changes, etc. Control rod calibration enables the determination of the reactivity associated with such changes in the conditions within the reactor (Souza and Mesquita, 2010).The change in reactivity of nuclear research reactors during operation may be due to various causes. Calibration of control rod (determination of reactivity worth per unit movement of control rod) is thus essential when the control rods are used as reactivity standards to measure the reactivity changes caused by any other perturbation in a reactor. Hence, it provides the reactor operator with a direct indication of net reactivity, which is a much more perfect indication of the nuclear status of the core (Nusrat et. al,, 2011). It has been revealed that once a control rod is calibrated, it is possible to evaluate the extent of other reactivity changes by comparing the critical rod positions before and after the change (Amir and Rose Mary, 2008). Calibration of control rod offers greater safety in reactor operation and aids the task of reactivity management; this is what the present work dwelled on.

There are several methods for determining the characteristic curve of control rods (Eugene, 2003). The various factors of reactor type, individual rod values, available instrument and time are what the methods of determining the characteristic curves of a control rod (calibration) are dependent upon. These methods include the Rod Drop method, the Mutual Calibration method, the Inverse Rate method and the Positive Period method amongst others. Most prominent of the methods are the mutual calibration method which presumes that to obtain reactor criticality, it is necessary to compensate reactivity change with moving rod and thus introducing reactivity of the same size but with opposite sign (Eugene, 2003). The method seems to be safer, but it requires a second rod that plays the role of compensation and thus allows for a certain degree of uncertainty (Fredrick, 1963). The rod-drop method is more cumbersome and has several disadvantages. The rods must be dropped from criticality, therefore limiting the rod combinations that can be measured. Also, the rod drop time is not instantaneous as is theoretically assumed, therefore limiting the accuracy of the method (Leslie, 1967). The inverse rate method is a modification to the rod drop method. It requires measurement of the weight of the rod by the Rod Drop method, however, errors in rod drop method will be propagated to the inverse rod results. A number of control rod calibrations have been done with the positive period method (Hidekini et. at., 1998, Agustin et. al, 2009, Nusrat et. al, 2011,). In all these calibrations, the method was shown to be a relatively independent and more accurate method, well suited to accommodate the core configuration and control rod mechanism of low power reactors like NIRR-1 (Agustin et.al., 2009; Leslie, 1967; Mueller and Brayan, 1963). Their result showed remarkably small uncertainty in calibration particularly for low power reactors. It is these favorable features of the positive period over other methods that the present work exploited in calibration of the NIRR-1 MNSR control rod.

1.2 STATEMENT OF RESEARCH PROBLEM

Critical experiments are carried out on various rectors to ascertain both the safety and stability of the reactor. Calibration of control rod has direct implication on such critical experiments. Since most operational activities of the NIRR-1 are steady state operation, there is need for accurate knowledge of the reactor response to transient characteristics due to control rod motion. This is to ascertain whether the reactor satisfies the safety requirement for MNSRs which dictates that the reactor must be stable under both the transient and steady state conditions. Thus it is important to determine experimentally, with emphasis on accuracy, the response of the reactor via its excess reactivity to control rod motion.

1.3 RESEARCH AIM AND OBJECTIVES

The aim of this research is to adopt the Positive Period method in the calibration of Nigeria Research Reactor-1 (NIRR-1) control rod. The objectives of this research work are to:

1. Calibrate NIRR-1 control rod using the Positive Period method
2. Investigate the reactivity worth of the control rod.
    a) at different time intervals
    b) at different positions.
3. Obtain the characteristic calibration curves of the control rod to determine the integral rod worth.
4. Compare theoretical predictions and experimental results of the Positive Period method in the calibration of NIRR-1 control rod
5. To recommend a simple procedure to be adopted for routine calibration of the NIRR-1 control rod by the operators.

1.4 JUSTIFICATION

To control NIRR-1, it is necessary also to affect its reactivity. This is realizeable via the motion of the absorbers (control rods) that are placed in the action parts of the control mechanisms, the function of which provides a reliable, remote and control of reactivity in the core (Eugene, 2003). It is therefore very vital to know the relationship between the reactivity change and the absorber position change with efficient accuracy for operation purposes. This relationship can be determined through control rod calibration (Eugene, 2003). More so in order to determine the excess reactivity available in the reactor, due to various operational activities and reactivity perturbations caused by prolonged utility of the NIRR-1, it is necessary to calibrate the control rod via the most relatively accurate method. The calibrated rod can then be used to measure the magnitude of the reactivity effect of other parameters on the reactor. The calibration can be achieved by observing the effects of a movement of the rod on the power level of the reactor (Ann, 1958).

The positive period method of control rod calibration was reported to be accurate and independent on the outside influences (Eugene, 2003). The measurements are carried out for relatively long time and the possible deviations are excluded with averaging from more values (Eugene, 2003)..

1.5 PREVIOUS WORK

An on-line 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

reactor control rods, the rod drop experimental technique 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, Bangladesh and Validation was performed by comparing the experimental data from developed system with the experimental data using positive period method and the calculated data from theoretical method based on the computer code like Monte Carlo code MVP. For instance, the average reactivity worth of the Safety rod measured by RECD and by ROMU was found to be $2.72, $2.7 respectively which is very close to that value calculated by MVP ($2.74). The result of validation agreed quite well within the margins of the experiment. This present research varies substantially from that of Nusrat *et al*, 2011, in that, it aims at not just determining the reactivity worth of NIRR-1, but doing so at different control rod position (calibration) while adopting the positive period method.

Neutronic test in the IPR-R1 TRIGA reactor were performed SOUZA, and MESQUITA, in the year 2010. Their experiments followed the determination of the IPR-R1 TRIGA safety analysis report which stated the necessity to perform them to ensure the safety the reactor. In the course of their experiment the control rods were calibrated using the positive period method and the worth of the regulating, shim and safety rods obtained. They concluded that there was a need to increase the excess reactivity of the core because of the negative reactivity values observed in the core. This according to them would be achieved by adding new fuel elements in other to operate the reactor at its new power. In the research carried out by SOUZA, and MESQUITA, the reactivity changes associated with the reactor were observed via the calibration of the control rods using the positive period method. This enabled their controlled study of the reactivity changes associated with

the reactor. However, their research was majorly concerned with general neutronics of the reactor rather than the method of calibration of the control rod and so no statement was made on the accuracy of the method. The present research is specifically aimed at calibrating the control rod of NIRR-1 using the positive period method which is a more precise method (Eugene, 2003}.

Agustin *et al.* 2009, earlier calibrated the control rod of the Portuguese research reactor (RPI) which was made up of 4 Cd and 1 stainless steel rod for power calibration using the positive period method. They conclusively reported that the calibration method had good agreement of experimental measures with theoretical curve. It is obvious therefore from their research that the positive period method of calibration holds strong balance between theoretical consideration and experimental result and thus good for control rod calibration. Large uncertainty also affected their result due to measurement uncertainty. For instance integrated reactivities of 66.50pcm, 107.94pcm, 156.09pcm, 217.17pcm, 240.98pcm, 246.94pcm have uncertainties as low as 4.11pcm, 8.87pcm, 1.98pcm, 10.42pcm, 9.66pcm, and 0pcm respectively. The difference between this present research and that of Agustin *et al.* 2009 lies in the fact that the NIRR-1 has a single control rod which makes its core configuration and control mechanism quite variant although the positive period method will be adopted in its calibration.

Frederick in 1963 calibrated the control rods at the ORR and LITR using the rod-drop method. The rod-drop method was examined as a means of determining the shut-down worth of the LITR and ORR shim rods. A total estimated error of 29% was obtained from the rod worth. They concluded that since the rod-drop method quickly and with no additional equipment can determine a rod worth reproducible to within 12%, .it followed that the method may be used as a tool to compare reactivity values of a rod over a period of

time or under slightly varying core conditions and that if inaccuracy of greater than 30% can be tolerated, the method may be used to determine the rod worth. They rightly noted that it may be of value if other means are not feasible or worth the time and effort necessary to obtain greater accuracy. The question therefore to ask is what if other means and worth time are feasible? Then, accuracy becomes a thing of great concern and thus a need to adopt a more accurate method with little uncertainty and error, in this light, the positive period method, applied to the NIRR-1, which is the aim of this present research.

Calibration measurement of the control rod and safety rods in the Enrico Fermi reactor were carried out by Mueller and Branyan in 1963. The control and safety rod calibration measurements were made during the low-power testing of the Enrico Fermi reactor. These measurements were made to confirm the rod design and also to investigate different calibration techniques. The regulating rod was calibrated entirely by means of positive and negative period measurements. The rod positions were obtained from the Gilnnore indicators. The period data were obtained from the two BF_3 detector channels connected to scalers, and from the ion chamber channel connected to the Keithley micro-microannmeter (Section III-C-2). Integrated counts taken from the scalers at 15-sec intervals were plotted on semi- logarithnnic graph paper to obtain the periods. The Keithley signal operated a timer set to start and stop automatically on e-fold power increases, thus giving a direct readout of the reactor period. The three periods obtained were averaged for use in the analysis. The average period was converted to reactivity using the inhour relationship for the reactor. This was available in tabular form for increased accuracy. They concluded that accurate calibrations of the control and safety rods were obtained for various degrees of shadowing. In general, the rod worths found were somewhat less than had been predicted from critical experiment data. However, the differences can largely be accounted

for by subsequent changes in the core and rod design. They recommended that although the differences in the positive period and subcritical count rate results were very small; it would have been adequate to calibrate the two control rods entirely by the subcritical count rate method. If the latter had been done, only one fuel loading would have been necessary, and the calibration measurements for both rods could have been completed in one day. Their recommendations in my opinion strongly stems from the fact that the positive period method takes longer time but this should not be much of a problem if the longer time will manifest into a more accurate result which is ultimately the uniqueness of any scientific research. For instance in their research, their results gave an estimated accuracy for the regulating rod calibration of approximately ± 4 per cent for the total rod worth and ± 2 per cent for the slope of the rod calibration curve. This is why the positive period method should be adopted in the calibration of the Nigeria Research Reactor-1 (NIRR-1).

The control rod system of the X-10 reactor was also calibrated by Jules and John, 1949. According to their research team, during the week of September 13 through September 17 1948, experiments were performed on the Clinton X-I0 reactor to determine the calibration of the number one and number two regulating rods, the number six shim rod, and the number seven safety rod. Absolute calibration of the regulating rods was obtained by allowing the cold pile to rise exponentially as a result of the partial withdrawal of one of the rods, while the relative effectiveness of various parts of the rods was obtained by introducing small increments of reactivity into the pile at full equilibrium power and observing the resultant power transient. The number six rod was calibrated by balancing with the number one and number two rods during a constant power run which started with the reactor in a cold condition. The value for the number seven rod was obtained by balancing against the number one and number two regulating rods during a run at full

power with the reactor in temperature equilibrium. No specific method was applied but the resultant calibration curve showed in hours as a function of position. In the research, the positive period method was not considered. This research therefore seeks to utilize the positive period method in the calibration of the NIRR-1.

CHAPTER TWO

THEORETICAL CONSIDERATIONS

2.1 CONTROL ROD CALIBRATION

Installation of rods of neutron-absorbing materials is carried out in most research reactors to provide precise, modifiable control of reactivity. These rods are able to be moved into or out of the reactor core and typically contain elements such as silver, indium, cadmium, boron, or hafnium. There are different materials used for control rods depending on the reactor design. Generally, good absorption cross section for neutrons and a long lifetime as an absorber should be the criteria for selecting the materials (Henry, 2005). The rods are grouped majorly according to their function and located at different places in the core where their function is maximized (Jarmo *et. al*., 2008). Control rods have several classifications. Three classifications will emerge based on functions. These classifications and their corresponding functions include;

a. Shim rods: for removal of reactivity in relatively large amount
b. Safety rods: for very fast shutdown when an unsafe condition presents itself
c. Regulation rods: for moderate adjustment and in the maintenance of a desired power or temperature.

Depending upon the type of reactor and the controls necessary, it is probable to use dual-purpose or even triple-purpose rods (Henry, 2005). The Nigeria Research Reactor -1 core (figure 2.1) has single control rod that serves triple purposes. The NIRR-1 control rod (figure 2.2) serves as a shim rod, regulation rod as well as safety rod. It 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 450 mm in length. The functions of reactor startup, steady-state operation, and shutdown are

accomplished by moving the control rod which is made up of a Cd absorber 266 mm long and 3.9 mm in diameter with stainless steel of 0.5 mm thickness as the cladding material. (Jonah et al 2006).

\

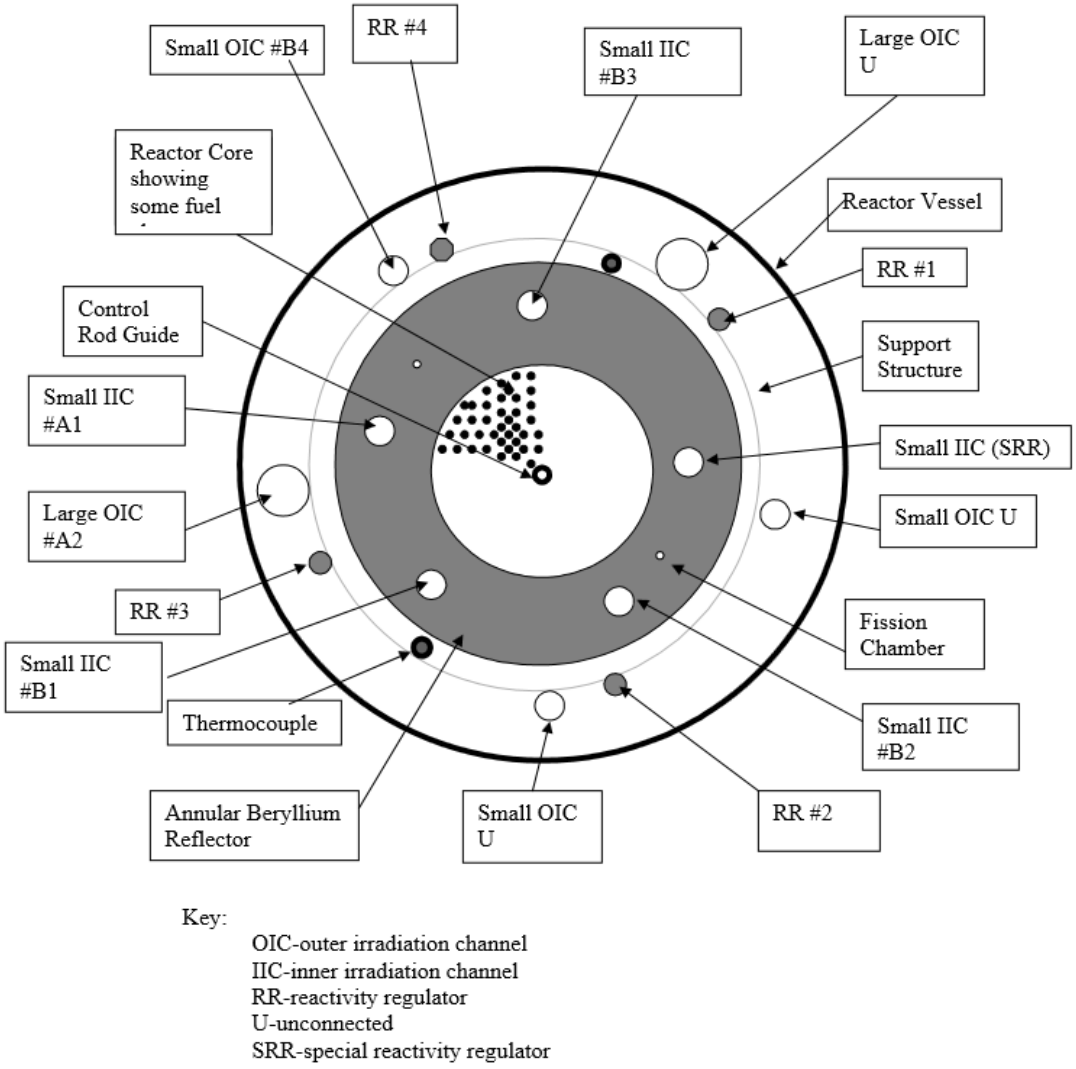


Fig. 2.1: Lay-out of NIRR-1 core showing irradiation channels and control devices (Jonah, 2008)

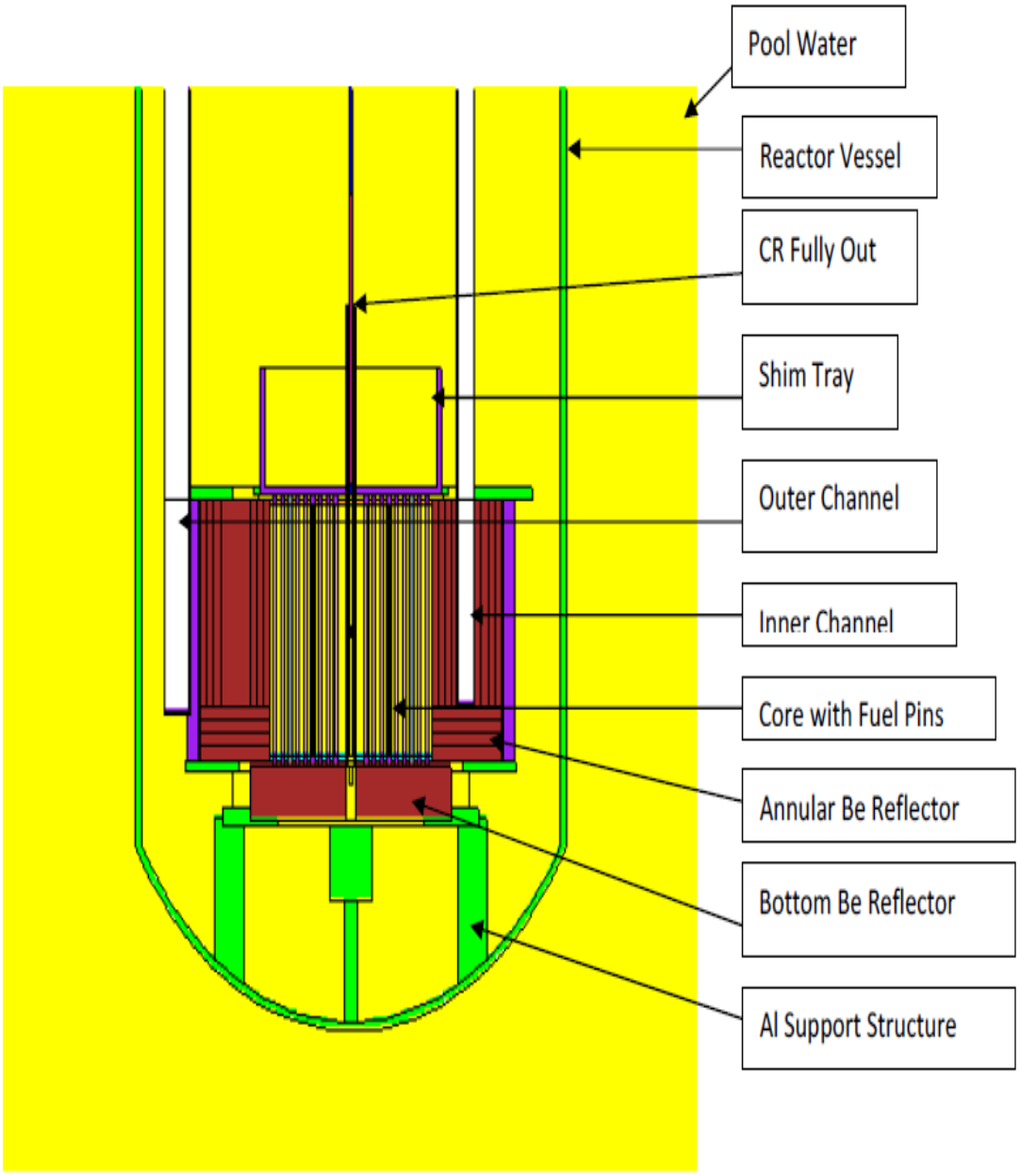


Fig. 2.2: An MCNP geometric representation of NIRR-1 (Jonah *et al*, 2008)

The effect of a given control rod penetrating into the active core of a reactor on its reactivity is dependent primarily upon two factors;

a). the neutron flux density value at the given place

b). penetration length into the active core (Eugene, 2003).

To control a reactor it is necessary to affect its reactivity (Eugene, 2003). This is realized by moving the absorbers that are placed in the action parts of the control mechanisms, the purpose of which provides the reliable remote and controlled movement of the absorbers in the core. Then, for operating purposes, it is essential to know the relationship between the reactivity change and the absorber position change with efficient accuracy. The relationship can be determined via calculations and is crucial after making any modification of the core, (Eugene, 2003).

The calculations come from the assumptions of the cosine thermal flux density distribution within the height of the reactor and of the proportionality of the absorber contribution to its square (perturbation theory). It therefore holds under this assumption that

$$\rho = K\int_{-\frac{H}{2}}^{x}\cos^2\frac{\pi t}{H}\ dt \ = \frac{K}{2}\left(x+\frac{H}{2}+\frac{H}{2\pi}+\frac{H}{2\pi}\sin\frac{2\pi x}{H}\right) \quad \ldots \quad (2.1)$$

Where K = constant that can be determined from the boundary condition

$\rho(-\frac{H}{2}) = 0$, $\rho(\frac{H}{2}) = \rho_0$

ρ_0 = rod weight

H = the total height of the core

x = the distance between the absorber end and the lower edge of the cores

After determining K and substituting into equation (2.1) above, the following is obtained

$$\rho(x) = \left(\frac{x}{H}+\frac{1}{2}+\frac{1}{2\pi}\sin\frac{2\pi x}{H}\right)\ \rho_0 \quad \ldots \quad (2.2)$$

From equation (2.2) above, it is evident that all rods have the same course of the characteristics and they differ only with the total weight ρ_0. Infact, the rod characteristics have different courses to a certain extent because simplified calculation does not respect the neutron flux density changes at the core reflector boundary (Eugene, 2003). As a rule, the mutual position of the absorber and the fission material cannot be also determined exactly i.e. the value of *H* or *x* cannot be determined exactly`. In spite of this, when the total weight of the rod is known, the characteristics can be determined with accuracy of about 10-20 by using this method (Eugene, 2003).

From the argument above, one can conclusively say that the determination of rod characteristics (calibration) is dependent on the determination of the reactivity change with respect to the change in the position of the absorber material. Different methods are applied in the calibration of control rods. These different methods include; the rod drop method, mutual calibration method, Inverse Rate method and the positive period method among others.

2.1.1 Rod drop Method of Control Rod Calibration

The rod-drop method: it is one of the most often used methods of reactivity determination. This method is based on the transient response of the reactor to a rapid time variation in the flux due to the dropping of the reactor control rods when the reactor is in a critical state. The rod drop time is not instantaneous as is theoretically assumed, therefore limiting the accuracy of the method (Leslie, 1967). Rod drop method is adopted to measure the negative reactivity of a control rod by suddenly inserting it into the core after adjusting criticality (Misawa et al., 2010).

If, when a reactor is operating at constant power, the reactor is scrammed, the power will decrease rapidly about two decades within one second and will then decrease with the characteristic delayed-neutron periods. We may derive a relation between the shutdown reactivity worth of the rod scrammed and the time behavior of the power during the first minute after scramming. For reactors with fuel-follower shim rods, this relation is (Ferdrick, 1963)

$$1 - k = \frac{R(0)}{R(t)} f\left[\beta_{eff} \sum_{i=1}^{6} \alpha_i e^{-\lambda_i t} + C\beta_\gamma \sum_{j=1}^{9} \alpha_j e^{-\lambda_j t}\right] \quad \ldots \ (2.3)$$

where

R(0) = detector response at steady power.

R(0) = detector response at a time, t, after dropping rod

f = fraction of total power generated in the core minus power generated in the dropped rod. This represents the fraction of delayed-neutron precursors which remains in the core.

β_{eff} = effective delayed neutron fraction

α_i = fraction of the delayed neutrons that are of the i^{th} kind

C = effective beryllium photoneutron contribution factor

β_γ = effective delayed beryllium photoneutron fraction

α_j = fraction of the delayed beryllium photoneutrons that are of the j^{th} kind.

The transient readings should be made after the higher modes of flux distribution have decayed and the fundamental mode persists, and before the intermediate deiayed neutron groups are decayed. The acceptable times are from 10 to 60 seconds (Ferdrick, 1963).The assumptions of the rod drop method are:

a. Prior to the drop, the delayed neutron precursors are at equilibrium.

b. The space independent, one-energy group model kinetics equation for the delayed neutron precursors is applicable.

c. Source multiplication is given by

$$M = \frac{1}{1-k} \quad \ldots(2.4)$$

d. After the rod drop, the generation of delayed neutron precursors may be neglected.

e. The detector samples the same fraction of the neutron population before the drop and at a time, t, after the drop.

In spite of limitation in accuracy, ,the rod-drop method has been used to report rod worths for the Ford Nuclear Reactor, yankee Power Reactor, and intermediate spectrum reactors and critical assemblies of the Naval Reactors program.(Gallaher, 1961, Shapiro, 1958, Stehn. 1958). Furthermore, papers by Henry. 1958 and McMurry and Metcalf, 1960, indicated that sound theoretical basis for the method is available.

2.1.2 The Mutual Method of Control Rod Calibration

The Mutual Calibration method presumes that to observe the reactor criticality after the rod position change that involves the reactivity change $\Delta\rho_p$, it is necessary to compensate this change with moving another rod and thus with introducing reactivity of the same size but with opposite sign $\Delta\rho_k = -\Delta\rho_p$. If the reactivity change is compensated for by using another rod for which reactivity-position relationship (calibration curve) is known, then from the movement of this rod $\Delta\rho_p$ the value of the reactivity can be read. With repeating the measurements for the next sections of the unknown rod, its characteristics can be determined (Eugene, 2003). Although, the mutual calibration method is safe, it requires a compensating rod and thus allows for a certain degree of uncertainty.

2.1.3 Inverse rate method of Control Rod Calibration

In inverse rate method, the following is valid for the rod characteristic

$$\Delta\rho(x) = \rho_0 \frac{\frac{1}{N(x)} - \frac{1}{N\downarrow}}{\frac{1}{N\uparrow} - \frac{1}{N\downarrow}} \quad \ldots (2.5)$$

Where

$\Delta\rho(x) =$ reactivity released by tye rod movement from 0 to x

ρ_0 = reactivity at 0 rod position

N(x) = neutron rate at rod position x

$N \uparrow and\ N \downarrow$ are neutron rates at the upper and lower rod position mark respectively (Eugene, 2003). During the course of the experiment, the following procedures are observed

a. The measurement is realized by using the experimental measuring channel.
b. The detector is inserted into the dry channel of the reactor in the reflector immediately near the core into the depth according to the requirement of the initial rate.
c. The initial state of the reactor (positions of the control rods) is determined.
d. The exercise begins with determining the total weight of the rod. The initial state is the critical reactor at power 20 to 30000 pulses per second in the automatic control mode. The rate measured with the experimental measuring channel is about 20000. With the measured rod in the upper end position
e. The weight of the rod is determined with rod drop method,
f. After rod drop, the weight of the rod can be checked with the source jerk method.
g. The number of measured points along the height of the rod and their coordinate is selected in order that the rod characteristic could be drawn.

h. From the upper end position, the rod is adjusted step-by-step into the selected rod positions and the rates are measured for such times that are at least 10000 pulses is measured in each position. To exclude possible error measurement (affected with eclectic in dereferences etc.), the necessary measuring time is divided into several partial intervals

i. The corresponding evaluations are then carried out according to equation 2.5.

The inverse rate method is also safer method of control rod calibration except that its experimental procedure requires measurement of the weight of the rod by the Rod Drop method, which in addition to it not being very accurate; it also increases the sub-criticality of the reactor.

The standard method of control rod worth determination was shown to be the positive period method (Leslie, 1967). One advantage of the positive period method (which will be explained in the methodology of this research in details) is that it requires no extra equipment (Leslie, 1967), in addition to it being relatively very accurate.

2.2 INTEGRAL AND DIFFERENTIAL CONTROL ROD WORTH

2.2.1 Integral Rod worth

Reduction or increment of thermal utilization factor (f) is an important property of Control rods and thus it changes K_{eff}, depending on whether the rod is inserted or withdrawn from the core. The change in K_{eff} results in change in the reactivity of the core. The worth of a control rod, is therefore, directly related to its effect on reactivity and its usually measured in the same units. The effectiveness, or worth, of a control rod depends mostly upon the value of the neutron flux at the location of the rod. The change in reactivity caused by control rod motion is referred to as control rod worth (DOE, 1993). For a reactor

with a single control rod like NIRR-1, the control rod will have maximum effect if it is placed in the center of rector core where the flux is a maximum, as shown in figure 2.3

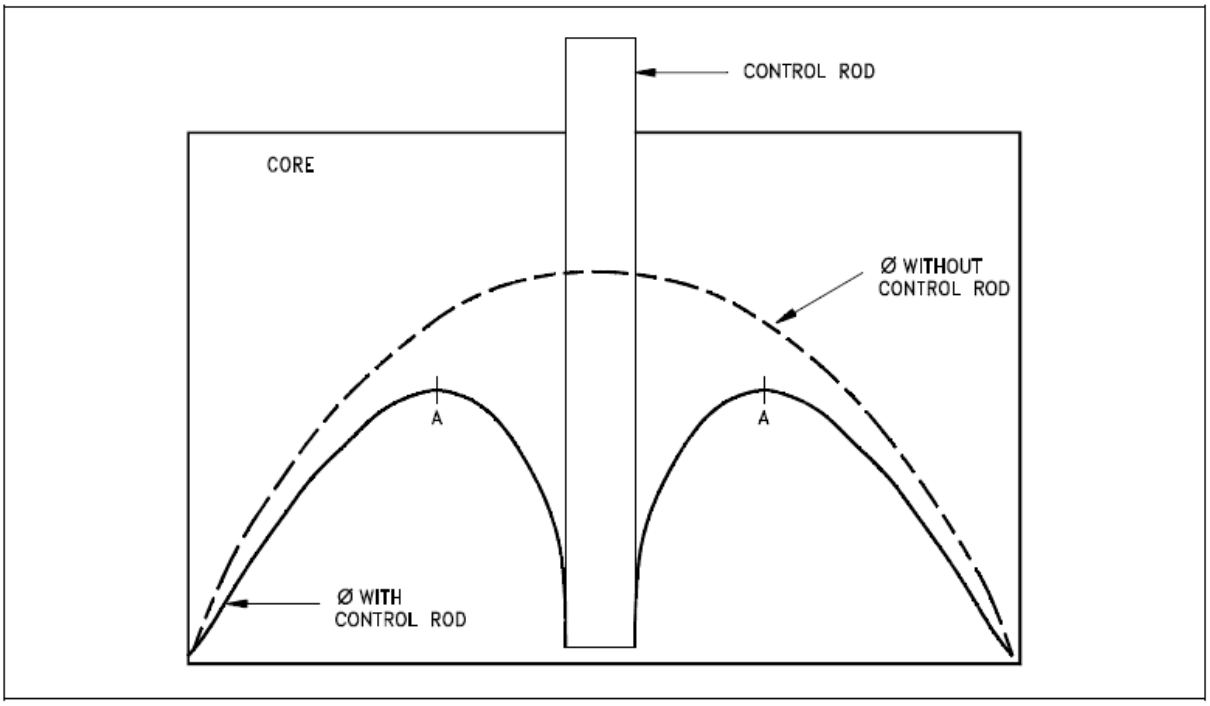


Fig. 2.3 Effect of single control rod on neutron flux (DOE, 1993)

The difference in the worth of the rod as it is inserted or withdrawn from the reactor is dependent on the axial flux shape. Generally, the flux is typically less at the top and bottom of the reactor than in the middle (DOE, 1993).. Therefore, the worth per unit length of the control rod is less at the top and bottom of the core than it is in the middle during insertion or withdrawal. This behavior is typically illustrated in the differential and integral rod worth curves as shown in fig. 2.3 & 2.4 respectively. The integral control rod worth curve is particularly significant in research reactor operation. For a reactor which has a large amount of excess reactivity, several control rods will be required (DOE, 1993),. To gain the full effectiveness of the rods and a relatively even flux distraction, the rods would need to be distributed appropriately. The exact worth of each control rod is dependent upon the design of the reactor. 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 1 cm, 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 that in Fig.2.4 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 (DOE, 1993).

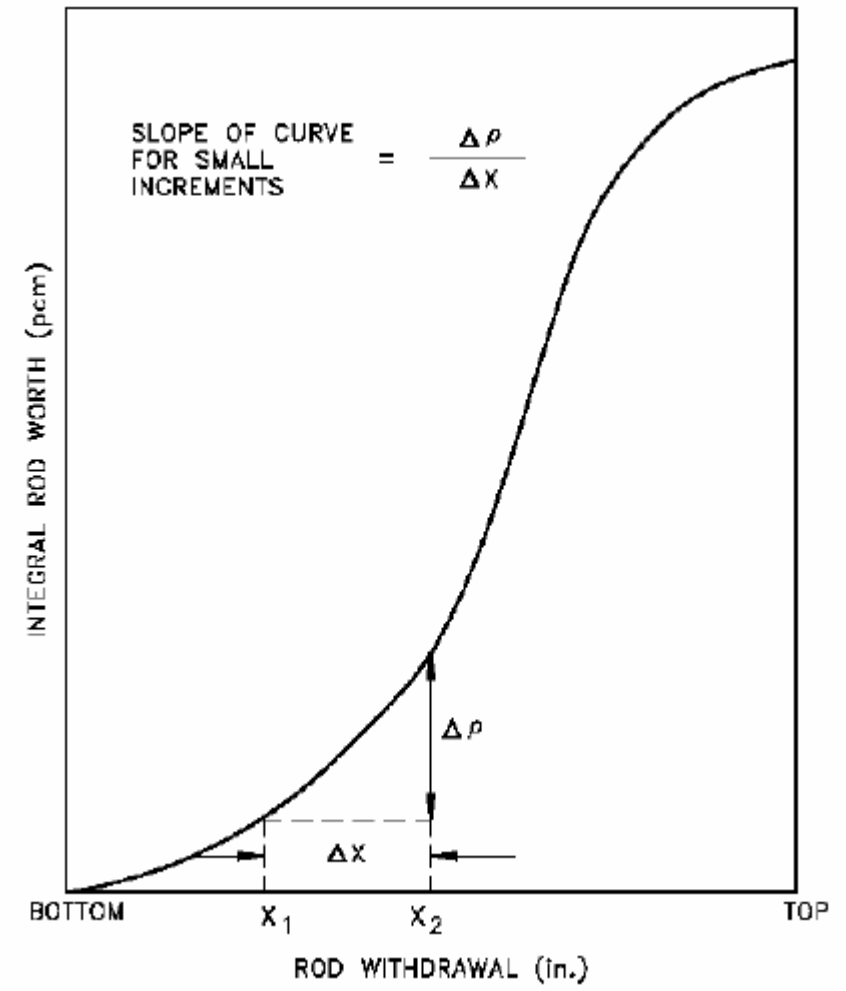


Fig. 2.4 Integral worth at different withdrawal positions (Henry, 2005)

2.2.2 Differential rod worth

The slope of the curve ($\Delta\rho/\Delta x$), 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 Fig.2.4 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 Fig.2.5. 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 unit length 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 unit length is basically the inverse of the rod worth per unit length from the center to the bottom.

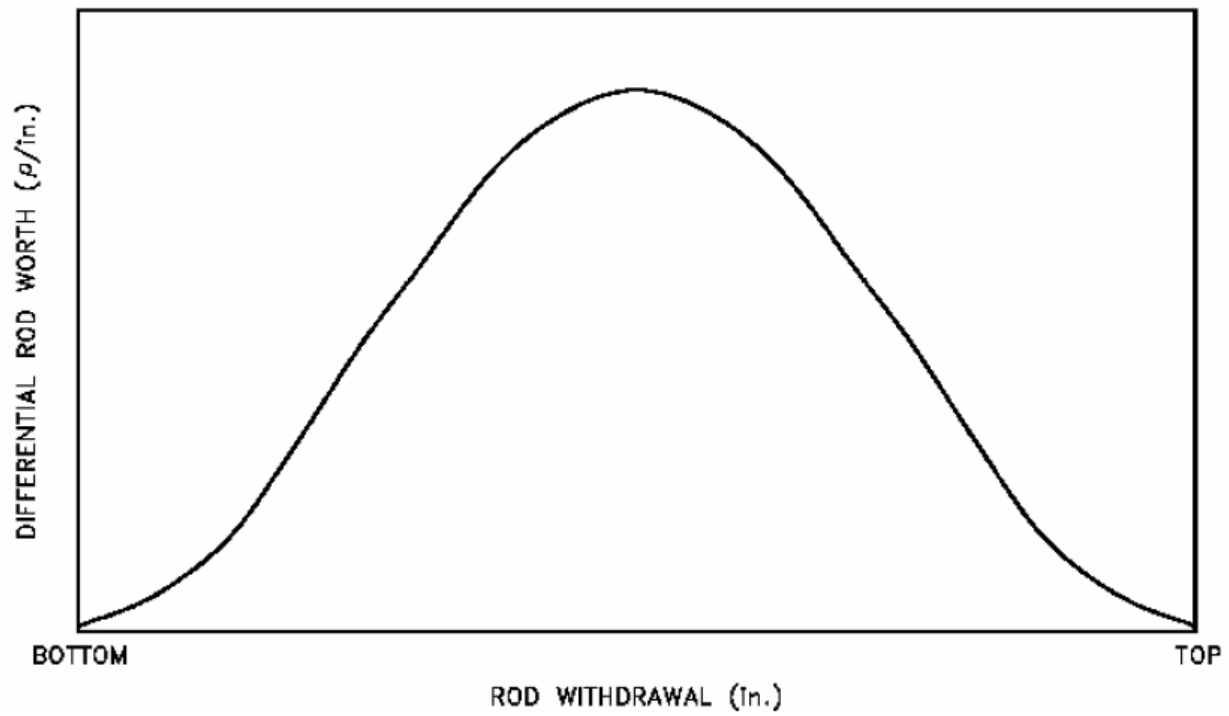


Fig. 2.5 Differential rod worth at different withdrawal position (Henry, 2005)

The determination of the reactivity worth of individual control elements and the effects of such elements on the power distribution in the core is important to the safe and efficient operation of a nuclear reactor. The NIRR-1 control rod was calibrated by the positive period method.

2.3 NEUTRON SOURCES

After sufficient power operation there will be neutrons available from various sources other than thermal fission in the reactor. They make it possible to see the approach to criticality on the NIRR-1 monitoring equipments. A power (neutron flux) increase can be seen on the NIRR-1 monitoring equipments as the control rods are withdrawn. If there were no sources present, the NIRR-1 monitoring equipments would not be sensitive enough to detect a positive, increasing period until the power was significantly higher. By the time the power was high enough to indicate on the NIRR-1 monitoring equipments, the period could be very short and a startup accident could occur.(CNSC, 2003). These neutrons produced in the reactor rather that of the neutron-induced fission are called source neutrons and those neutrons producing reactions that always occur in reactor materials are called intrinsic neutron sour.ces. Examples are

i. Phtoneutron sources:

$$\gamma + H_1^2 \rightarrow H_1^1 + n_0^1$$

Some radioisotope gamma-ray emitters can also be used to produce neutrons when combined with an appropriate target material. The resulting photoneutron sources are based on supplying sufficient excitation energy to a target nucleus by absorption of a gamma-ray photon to allow the emission of a free neutron. It was opined that only two target nuclei,

$Be^9$and, H^2 are of any practical significance for radioisotope photoneutron sources (Glenn, 1991).

All water-cooled reactors like the NIRR-1 have some deuterium present in the coolant in the reactor core because a small fraction of natural hydrogen is the isotope deuterium. The atom percentage of deuterium in the water ranges from close to the naturally occurring value (0.015%) for light water reactors to above 90% deuterium for heavy water reactors. Therefore, the required conditions for production of photoneutrons exist. The supply of gamma rays decreases with time after shutdown as the gamma emitters decay; therefore, the photoneutron production rate also decreases (DOE,1993).

ii. Alpha reactions with boron

$$\alpha_2^4 + B_5^{11} \rightarrow N_7^{14} + n_0^1$$

Intrinsic neutron sources mostly do not register significant neutrons in a new reactor's initial startup. The reason behind the installation of neutron sources which are operational is the relative weakness of intrinsic neutron sources. The strength of intrinsic neutron sources also depends on the power history of a reactor.

2.3.1 Prompt and delayed neutrons

Initially the fission fragments produced in a nuclear fission are highly excited and thus get rid of their excitation energy by emitting neutrons and gamma rays. Most of the fission neutrons are emitted almost immediately (~10^{-17} s) after the fission takes place; these are known as prompt fission neutrons, while the gamma rays are known as prompt gammas. Following de-excitation, a nucleus formed by fission does not have sufficient excitation energy to throw off any more neutrons. It usually decays by beta emission; in a few cases, the daughter nucleus created by beta decay has sufficient energy to emit a

neutron. This process results in approximately ½ % or so of the fission neutrons showing up long after the fission event. These are the delayed fission neutrons (CNSC, 2003).

One of the most important aspects of the fission process from the point of view of reactor control is the presence of delayed neutrons. Fission Products also release neutrons with some delay based on half life -Precursors (Andrew, 2008). A delayed neutron is a neutron emitted by an excited fission product nucleus during beta disintegration some appreciable time after the fission. How long afterward, is dependent on the half-life of the delayed neutron precursor (CNSC, 2003). The kinetic behavior of a reactor is critically dependent on the existence of these delayed neutrons as they have the effect of increasing the average lifetime of neutrons arising from fission. The average lifetime in heavy water moderated reactors is about 1 millisecond while in light water moderated reactors it is less (Chaplin, 2006) The symbol β_i is used to denote the total fraction of delayed neutrons. Delayed neutrons are commonly classified in six groups for an accurate description of neutron kinetics. Each of this group (i) is characterized by a fractional yield β_i and decay constant λ_i.

Table 2.1: Six Precursor-Group Decay constants And Delayed Neutron Fraction For Thermal Fission of U-235 (Daniel and Siamak, 1997)

Group	$\lambda_i\ (s^{-1})$	$\beta_i(\%)$
1	0.0129	0.0251
2	0.0311	0.1545
3	0.1340	0.1476
4	0.3310	0.2663
5	01.260	0.0756
6	3.2100	0.0293

2.4 NEUTRON LIFETIME

Neutron lifetime is an important quantity in reactor kinetics. It is the average time elapsing between the release of a neutron in a fission reaction and its loss from the system by absorption or escape. The overall neutron lifetime during thermalization and diffusion from production due to fission until absorption to produce fission is extremely short (Chaplin, 2006).

2.5 REACTOR PERIOD

Exponential power rise characterizes a supercritical nuclear reactor; the parameter that characterizes the rate of power rise is the reactor period, τ. The reactor period is defined as the time required for reactor power to change by a factor of "e," where "e" is the base of the natural logarithm and is equal to about 2.718.The reactor period is usually expressed in units of seconds. From the definition of reactor period, it is possible to develop the relationship between reactor power and reactor period that is expressed by the following Equation:

$$P = P_0 e^{\frac{t}{\tau}} \qquad \ldots. (2.6)$$

where P = transient power

P_0 = initial reactor power,

t = time during the reactor transient.

Here τ is the reactor period. A long reactor period corresponds to a slow power rise, and the smaller the value of τ, the more rapid the change in reactor power is. If the reactor period is positive, reactor power is increasing. If the reactor period is negative, reactor power is decreasing. The power in equation (2.6) is neutron power; it does not include decay heat which distinguishes it from reactor thermal power that includes decay heat (CNSC, 2003). The reactor period, which governs the

rate of stable rise, is proportional to the delayed neutron lifetime (1/λ), and also depends on the ratio of the reactivity to the delayed neutron fraction (Δk/β).

2.6 INHOUR EQUATION

Period-reactivity relationship for any reactor is an important and useful relationship. Aside forming the basis of several kinds of reactivity measurements like control rod worth measurements, it is also very crucial in their calibration and fuel and reflector worth measurements (Om Pal Singh *et. al*., 2014). The inhour equation is one such equation that relates reactor period and reactivity.

The relationship between a step change in reactivity (ρ) and the resulting reactor period (T) allowing for the delayed neutrons, can be derived in the form of an equation such as:

$$\rho = \frac{l}{KT} + \sum_{i=1}^{6} \frac{\beta_i}{1+T.\lambda i} \qquad \text{…….. (2.7)}$$

where β_i = delayed neutron fraction for group i

λ_i = decay constant of delayed neutron group i (s^{-1}) = $In2/t_{1/2}i$

l = Prompt neutron lifetime (s)

T = reactor period.

This equation is often called the reactivity equation or the "Inhour" equation. For a given reactor, the quantities β_i, λ_i, l and k are known (often k ≈ 1 and is left out of the equation) so that the period can be determined when the reactivity is known and vice versa. This is best done numerically and then the results presented in tables or as a graph.

The "inhour equation" is derived through the reactor kinetics equations, which are derived from the diffusion theory (Leslie, 1967). Attempts have been made by Hurwitz, Ussachoff, Henry, Lewins and Gozani (Hurwitz, 1949, Ussachoff, 1955, Henry, 1958, Gozani, 1963) among others to explain the operational and conceptual significance of

reactor kinetics parameters. Lewins has shown that the formulation of the reactor kinetics equations in a form parametric in the generation time is more accurate and simpler to solve than the corresponding form in the lifetime. The reactor kinetics equations are as follows:

$$\frac{dn}{dt} = \frac{\rho-\beta}{l}\mathrm{n} + \sum_{i=1}^{6} \lambda_i c_i \quad \ldots(2.8)$$

$$\frac{dc_i}{dt} = \frac{\beta_i}{l}\mathrm{n} - \lambda_i c_i \quad \ldots(2.9)$$

Where n = neutron density

ρ = reactivity

β= total delayed neutron fraction

β_i = delayed neutron fraction of the i^{th} delayed group

l = Prompt neutron generation time

λ_i = decay constant of the i^{th} delayed group

C_i = concentration of the i^{th} precursor

Since equations (2.8) and (2.9) are linear first order and the variables are separable, solutions of the following form can be assumed:

$$n(t) = n_0 e^{tw} \quad \ldots(2.10)$$

$$c(t) = c_{i0} e^{tw} \quad \ldots(2.11)$$

w is an undefined parameter with reciprocal time units. Substituting these equations (2.10) and (2.11) back into the kinetics equations (2.8) and (2.9) respectively and solving for reactivity, we obtain

$$\rho = \quad wl + \sum_{i=1}^{6} \frac{w\beta_i}{w+.\lambda i} \quad \ldots(2.12)$$

Since this equation is seventh degree in w, the neutron density has the form

$$n(t) = A_0 e^{tw_0} + A_1 e^{tw_1} + \cdots + A_6 e^{tw_6} \quad \ldots(2.13)$$

where the w's are the roots of the reactivity equation.

For positive reactivity, it is found that one root (w_0) is positive and the other six roots are negative.

Thus, after a short time, the neutron density reduces to

$$n(t) = A_0 e^{tw_0} \quad \ldots(2.14)$$

By definition, the stable reactor period (T) is

$$T = \frac{1}{w_0} \quad \ldots(2.15)$$

Therefore,

$$n(t) = n_0 e^{t/T} \quad \ldots(2.16)$$

Substituting $w = 1/T$ in the reactivity equation, we obtain

$$\rho = \frac{l}{KT} + \sum_{i=1}^{6} \frac{\beta_i}{1+T.\lambda i}$$

Where *K* is equal to 1, which is the same as equation (2.7) above.

This is the general relation between the reactivity and the stable reactor period (*T*) for a single group model of a bare core (Zheng, 1993). The two-point model is adopted for MNSRs because of the beryllium is used as reflector in the reactor which will produce some photoneutrons in the reflector acting as additional delayed neutron groups and thus the inhour equation for MNSRs becomes (Zheng, 1993)

$$\rho = \frac{l}{KT} + \gamma \left[\sum_{i=1}^{6} \frac{\beta_i}{1+T.\lambda_i} + Q \sum_{i=1}^{15} \frac{\beta_j}{1+T.\lambda_j} \right] \quad \ldots(2.17a)$$

$$\gamma(effective\ delayed\ neutron\ \ coefficient) = 1.23\ for\ MNSR$$

$$Q(effective\ delayed\ photoneutron\ \ coefficient) = 0.2\ for\ MNSR$$

Apart from the fact that this relation is very important in reactor physics, *it also forms the physical background of the positive period method of control rod calibration.*

This equation can be simplified if some assumptions are made. Assuming one average group of delayed neutrons,

$$\rho = \frac{\beta}{1+T.\lambda} \qquad ...(2.17b)$$

This assumption is accurate for reactivities of about 0.025 percent .or less.

2.7 CONCEPT OF CRITICALITY

The neutron population at any instant in a nuclear reactor depends on the rate of neutron production (due to fission processes) and the rate of neutron losses (via non-fission absorption mechanisms and leakage from the system). Critical is the condition where the neutron chain reaction is self-sustaining and the neutron population is neither increasing nor decreasing (DOE, 1993).Before the first start-up of any reactor a critical experiment has to be performed. At this time the accuracy of theoretical calculations can be compared with the experiment, while the reactor core is loaded stepwise with fuel elements. In one position a neutron source emits some 106 neutrons per second. These source neutrons introduce fission in ^{235}U atoms and thus fission neutrons are produced. The relative number of neutrons near the core is measured with a fission chamber (Eugene, 2003). When the reactor's neutron production exceeds losses, characterized by increasing power level, it is considered "supercritical", and; when losses dominate, it is considered "subcritical" and exhibits decreasing power.

The "Six-factor formula" is the neutron life-cycle balance equation, which includes six separate factors, the product of which is equal to the ratio of the number of neutrons in

any generation to that of the previous one; this parameter is called the effective multiplication factorK_{eff} given by,

$$K_{eff} = L_f \rho L_{th} f \eta \varepsilon \quad \ldots(2.18)$$

Equation (2.18) is called the six factor formula. This six factor formula, ensures the possibility of tracing the entire neutron life cycle from production by fission to the initiation of subsequent fissions.

where

$$L_{th}(\text{thermal non} - \text{leakage factor}) = \frac{\textit{number of thermal neutrons that do not leak from reactor}}{\textit{number of neutrons that reach thermal energies}} \quad \ldots(2.18a)$$

$$L_f(\text{fast non} - \text{leakage factor}) = \frac{\textit{number of neutrons that do not leak from reactor}}{\textit{number of fast neutrons produced by all fission}} \quad \ldots(2.18b)$$

$$\rho\ (\text{resonance escape probability}) = \frac{\textit{number of neutrons that reach thermal energy}}{\textit{number of neutrons that start to slow down}} \quad \ldots(2.18c)$$

$$\eta\ (\text{reproduction factor}) = \frac{\textit{number of fast neutrons produced by thermal fission}}{\textit{number of thermal neutrons absorbed in the fuel}} \quad \ldots(2.18d)$$

$$f\ (\text{thermal fuel utilization factor}) = \frac{\textit{number of thermal neutrons absorbed in the fuel}}{\textit{number of thermal neutron absoirbed in any reator material}} \quad \ldots(2.18e)$$

$$\varepsilon(\text{fast} - \text{fission factor}) = \frac{\textit{number of fast neutrons produced by all fissions}}{\textit{number of fast neutrons produced by thermal fissions}} \quad \ldots(2.18f)$$

When the reactor is critical, K_{eff} = 1.

2.7.1 Subcriticality

Subcritical is the condition in which the neutron population is decreasing each generation (DOE, 1993). From equation (2.18) above, when a reactor is subcritical, K_{eff}< 1. Even in a subcritical assembly such as a shut-down reactor core, any stray neutron that happens to be present in the core (for example from spontaneous fission of the fuel, from radioactive decay of fission products, or from a neutron source) will trigger an exponentially decaying chain reaction. Although the chain reaction is not self-sustaining, it acts as a multiplier that increases the equilibrium number of neutrons in the core. This *subcritical multiplication* effect can be used in two ways: as a probe of how close a core is to criticality, and as a way to generate fission power without the risks associated with a critical mass. The degree of subcriticality of a multiplying assembly having a spontaneous fission-neutron source may be inferred by inducing an additional change in multiplication and by following the flux profile through the course of this change to its final stationary state (Bennett, 1976).

2.7.2 Supercriticality

Supercritical is the condition in which the neutron population is increasing each generation (DOE, 1993).As stated above, when the reactor is supercritical, K_{eff} > 1. That an assembly is supercritical does not guarantee that it contains any free neutrons at all. At least one neutron is required to "strike" a chain reaction, and if the spontaneous fission rate is sufficiently low it may take a long time (in ^{235}U reactors, as long as many minutes) before a chance neutron encounter starts a chain reaction even if the reactor is supercritical.

The Nigeria Research Reactor -1 (NIRR-1) first criticality was accomplished on the 3rd of February 2004 (Balogun et al., 2004) and has since then been working securely for

neutron activation analysis (Jonah et al., 2006). When large positive reactivity is inserted into a core via the withdrawal of a control rod, and ΔK becomes far greater than β, the reactor becomes supercritical on prompt neutrons alone without having to wait for delayed neutrons. A reactor with effective multiplication factor large enough to make the prompt neutron fraction increase enough to make the reactor critical is said to be prompt critical (CNSC, 2003).

2.8 REACTIVITY AND ITS RELATIONSHIP WITH CRITICALITY

The reactivity (as also the criticality factor K and ΔK) provides a measure of the over-all neutron economy of a chain-reacting system and, in this sense, it is a global parameter characterizing the state of the system as a whole. It is an indicator of the degree of criticality and hence of the response of the system to external and internal perturbations (Malaviya *et. al.,* 1964). 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 a 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_0 neutrons in the preceding generation, then there are $N_0(K_{eff})$ neutrons in the present generation. The numerical change in neutron population is $\{N_0(K_{eff}) - N_0\}$(DOE, 1993). Expressing the gain or loss of neutron population as a fraction of the present generation $N_0(K_{eff})$

the reactivity (ρ) will be obtained, which is a fractional change in neutron population per generation i.e

$$\rho = \frac{\{N_0(K_{eff}) - N_0\}}{N_0(K_{eff})} \quad \text{...(2.19)}$$

Where the N_0 in the numerator and denominator of equation (2.19) cancels out, the reactivity is then obtained as

$$\rho = \frac{\{(K_{eff}) - 1\}}{(K_{eff})} \quad \text{...(2.20)}$$

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

$$\rho = \delta K_{eff} = \frac{\{(K_{eff}) - 1\}}{(K_{eff})} \quad \text{...(2.21)}$$

Where $\delta K_{eff}$the deviation of a reactor from its critical state and K_{eff} is the effective multiplication factor. If the reactivity value in equation (2.21) above has a value of $\rho < 0$ 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 while 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).

2.8.1 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). A reliable method for measuring reactivity which is one of the most important problems to be ·solved in nuclear science and reactor physics due to various difficulties associated with its measurement has been established (Michiya and Hisahide, 2012). 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. Several factors (for example, fuel depletion, temperature, pressure, or poisons) can affect the reactivity of a reactor core. The reactivity coefficients are measurements used to quantify the effect that a variation in parameter (that is, increase in temperature, control rod insertion, increase in neutron poison) will have on the reactivity of the core, Reactivity coefficients are the amount that the reactivity will change for a given change in the parameter (that is, increase in temperature, control rod insertion, increase in neutron poison) (DOE, 1993).

2.8.2 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). The reactivity due to temperature feedback $\rho(t)$ is expressed as:

$$\rho(t) = \rho_0(t) + \alpha_m \Delta T_m(t) + \alpha_f \Delta T_f(t) \quad \ldots(2.22)$$

where $\rho_0(t)$ is the initial reactivity,

α_m and α_f are the moderator and fuel temperature coefficients (of reactivity)

ΔT_m and ΔT_f are the deviations of the spatially averaged moderator and fuel temperatures from the equilibrium temperatures respectively.

For the MNSR, the core temperature coefficient of reactivity α_T is given by the addition of the moderator and fuel temperature coefficients.

$$\alpha_T = \alpha_m + \alpha_f \quad \ldots(2.23)$$

The Reactivity Temperature Coefficient α_T is defined as the change in reactivity with respect to temperature and expressed as

$$\alpha_T = \frac{d\rho}{dT} \quad \ldots(2.24)$$

Where, reactivity (ρ) which is the fractional departure of a system from criticality is expressed as

$$\rho = \frac{\left((K_{eff}) - 1\right)}{(K_{eff})} \quad \ldots(2.25)$$

Differentiating equation (2.25) with respect to temperature (T) gives:

$$\alpha_T = \frac{d\rho}{dT} = \frac{dK_{eff}}{dT} \cdot \frac{d\rho}{dK_{eff}} = \frac{1}{K_{eff}} \frac{dK_{eff}}{dT} \quad \ldots(2.26)$$

The two dominant temperature coefficients are:

a) Moderator Temperature Coefficient

The moderator temperature coefficient of reactivity can be defined as the change in reactivity per degree change in moderator temperature. The semi-empirical relation amongst core inlet temperature, coolant temperature rise and power levels as obtained from simulation experiments on MNSR is expressed in the form (Shi, 1990):

$$\Delta T = (5.725 + 147.6H^{-2.64}) T_i^{-0.35} P^{(0.59+0.0091T_i)} \quad \ldots(2.27)$$

Where ΔT= Temperature difference between inlet and outlet orifices(℃)

H = Height of the inlet orifice (mm)

T_i = Inlet temperature(℃)

The designed inlet orifice of NIRR-1 was made to be 6 mm for safety and technical reasons (Yang, 1992). Therefore, substituting the value of H (0.90147) into Eq. (2.27) reduces the equation to:

$$\Delta T = 7.04 T_i^{-0.35} P^{(0.59+0.0091T_i)}$$

And we thus obtain the reactor power as

$$\mathrm{P} = \mathrm{Exp}\left[ln\left(\frac{\Delta T}{7.04 T_i^{-0.35}}\right)(0.59 + 0.0091 T_i)^{-1}\right] \quad \ldots(2.28)$$

For a fixed height of inlet orifice, it is expected from Eq. (2.29) that the reactor power varies linearly with coolant temperature rise (Ahmed *et al, 2006*).The magnitude and sign (+ or -) of the moderator temperature coefficient is primarily a function of the moderator-to-fuel ratio. If a reactor is under moderated, it will have a negative moderator temperature coefficient. If a reactor is over moderated, it will have a positive moderator temperature coefficient. A negative moderator temperature coefficient is desirable because of its self-regulating effect. For example, an increase in reactivity causes the reactor to produce more power. This raises the temperature of the core and adds negative reactivity, which slows down, or turns, the power rise (DOE, 1993).

b) Fuel Temperature Coefficient

Another temperature coefficient of reactivity, the fuel temperature coefficient, has a greater effect than the moderator temperature coefficient for some reactors. The fuel temperature coefficient is the change in reactivity per degree change in fuel temperature. This coefficient is also called the "prompt" temperature coefficient because an increase in reactor power causes an immediate change in fuel temperature. A negative fuel temperature coefficient is generally considered to be even more important than a negative moderator temperature coefficient because fuel temperature immediately increases following an

increase in reactor power. The time for heat to be transferred to the moderator is measured in seconds. In the event of a large positive reactivity insertion, the moderator temperature cannot turn the power rise for several seconds, whereas the fuel temperature coefficient starts adding negative reactivity immediately.

2.8.3 Nuclear reaction rates, cross section and reactor power

Most of the neutrons in the NIRR-1 core are thermal neutrons due to its design. The rates of reactions induced by these neutrons are directly proportional to the

a. **thermal neutron flux**

Regarding the operation and use of nuclear research reactors, the problems of neutron detection and detectors are related to the problems of reactor control (measuring the power) and the experimental specification of the neutron field parameters (flux density, fluency and spectrum), (Kolros, 2008). The product nv, which has the units of (neutrons/cm^2-sec) and represents the total path length traveled per second by all the neutrons in 1cm^3, is called the neutron flux. Thus:

$$\phi = nv \qquad \ldots(2.29)$$

where

ϕ = neutron flux (neutrons/cm^2-sec)

n = neutron density (neutrons/cm^3)

v = neutron velocity (cm/sec)

The expression for neutron flux applies to any neutron energy. When applied to thermal neutrons the product is known as the thermal neutron flux. The term neutron flux in some

applications (for example, cross section measurement) is used as parallel beams of neutrons traveling in a single direction (DOE, 1993).

The Chinese Miniature Neutron Source Reactor (MNSR) built Nigeria Research Reactor -1 (NIRR-1) is a low power research reactor with maximum thermal neutron flux of 1 x 10^{12} n.cm^{-2}s^{-1} (Ahmed *et al.,* 2008). To measure the thermal neutron flux density, the miniature fission chambers, self-powered detectors, the activation detectors, the calorimeters, etc. are used. The proper detection material and the relationship between its absorption (activation) cross-sections and neutron energy are important (Kolros, 2008).

Thermal neutron flux values are very important parameters in the NIRR-1 core. These values can be used to determine a large variety of other parameters in the core which provide information both about the operational characteristics of the core and the safety of the reactor. These values can be used to predict the fission power of the reactor, as long as nuclear data parameters are accurately known (Ahmed et al., 2006). The equation that relates these two parameters is:

$$P = 3.1x10^{-10}\sum_f V_f \square \qquad \ldots(2.30)$$

where ϕ = average thermal neutron flux in the inner irradiation channel ($cm^{-2}s^{-1}$)

V_f = volume of the core = πr^2h (cm^3)

Core height (h) = 23cm

Core radius (r) = 11.5cm

$\sum_f$ = macroscopic fission cross-section of the core fuel = $1.013x10^{-2}cm^{-1}$

A highly enriched uranium (U-235) used as fuel for the NIRR-1 MNSR with a cylindrical core of cross-sectional area of $23cm^2$ makes it possible to reduce Eq. (2.30) to

a reactor power equation that is dependent only on thermal neutron flux as can be seen in Eq. (2.31).

$$P = 3.0x10^{-8}\square$$

…(2.31)

This prediction is quite accurate because at half power (15kW), the NIRR-1 neutron flux is $5 \times 10^{11} n/cm^2 s$. Thus, Eq. (2.31) indicates a linear relationship between the reactor power and its neutron flux. It also shows that one can predict the reactor power if the experimental neutron flux values are known.

b. **Macroscopic cross section for the reaction.**

The probabilities of different interactions are represented by their respective microscopic cross sections σ. The cross section can be thought of as an effective geometrical cross section encountered by a neutron (Philip, 2012). The larger the cross section, the higher the probability of the neutron hitting an atom. Generally, the cross sections decrease with higher neutron energies (Rinard, 1991). There are, however, important exceptions, such as resonance regions with large cross sections.

Neutron cross sections are defined separately for each type of reaction and isotope. For example,

σ_s = elastic scattering cross section

σ_i = inelastic scattering cross section

σ_a = absorption cross section

σ_γ = capture cross section

σ_f = fission cross section

The total cross section-i.e., the total probability that a reaction of any type will take place-is equal to the sum of all the σ's:

$$\sigma_{total} = \sigma_s + \sigma_i + \sigma_a + \sigma_\gamma + \sigma_f + \cdots \quad \ldots(2.32)$$

Neutron cross sections depend strongly on the energy of the neutron as well as on the atomic weight and atomic number of the target nucleus. The cross section σ (b) is called the microscopic cross section. Another form of the cross section, also frequently used, is the macroscopic cross section Σ (m^{-1}), defined by the equation

$$_i = N(nuclei/m^3)\, \sigma_i(\, m^2) \quad \ldots(2.33)$$

and having the following physical meaning:

$_i$ = Probability that an interaction of type *i* will take place per unit distance of travel of a neutron moving in a medium that has N nuclei/m^3 (Nicholas, 1995).

2.9 DESCRIPTION AND DESIGN OF NIRR-1

The difficulty of obtaining information about nuclear science and reactor physics from reactors operating at power has led to the development of more economical methods involving the use of zero-power critical assemblies and exponential assemblies (Weinberg and Wigner 1958, Thie, 1961, Kaplan, 1962,). A supplement to these are the "miniature" lattices for miniature neutron source reactors: MNSR (Peak et. al., 1962),The MNSR Nigeria Research Reactor -1 (NIRR-1) uses highly enriched U-235 fuel in an Al cladding. A tank is immersed in a large pool of water, and the core is, in turn, immersed in the tank-

an arrangement popularly called tank-in-pool type (Yang, 1992). Their maximum nominal power is ~ 30 kW which is equivalent to a thermal neutron flux of 10^{12} $n.cm^{-2}.s^{-1}$ in one of the inner irradiation channels (Balogun 2003, Ahmed 2004). For a typical MNSR, the central core consists of 347 fuel rods, with 4 tie rods and 3 dummy elements uniformly distributed on a total of ten concentric circles, each having a number of fuel rods ranging between 6 and 62 (Table 2.2) with a thick beryllium reflector (~ 10 cm) surrounding the core radially (Yang 1992).

The reactor core is a square cylinder of dimension 23cm by 23cm and it is surrounded by Be annulus on the sides and a Be plate at the bottom. A tray for shimming the reactor in the event of reactivity issue due to Sm poison sits on top of the core (figure 2.6). The fuel elements are pin type and are arranged in a bird cage, consisting of 350 lattices for the fuel pins (Azande, 2010). A single control rod made up of Cd material in stainless steel cladding moves centrally inside a guide tube located at the center of the core. The two core configuration sits in light water which serves as both coolant and moderator. The core is designed to be under-moderated with the number of $H/^{235}U$ ratios as 197 for the commercial MNSR (Jonah *et. al*, 2008).

Table 2.2: Technical Specifications of NRR-1 (Yang, 1992)

Parameters	Description
Reactor type	Tank-in-pool
Rated thermal power	30 Kw
Fuel	UA4 dispersed in Al
U-234 Enrichment	90.2%
Core shape	Cylinder
Core diameter	23.1 cm
Core height	23.0 cm
No. of fuel elements	344
Weight of U-235	998.116 g
No. of irradiation channels	10
Inner channels	5
Flux in inner channel	$1x10^{12}ncm^{-2}s^{-1}$
Flux in outer channel	$5x10^{11}ncm^{-2}s^{-1}$
Reactor cooling mode	Natural convection
Height of inlet orifice	6 mm
Height of outlet orifice	7.5 mm
Diameter of fuel meat	4.3 mm
Diameter of fuel element	5.5 mm
Excess reactivity	3.99 mk
Length of Cd control rod	230 mm

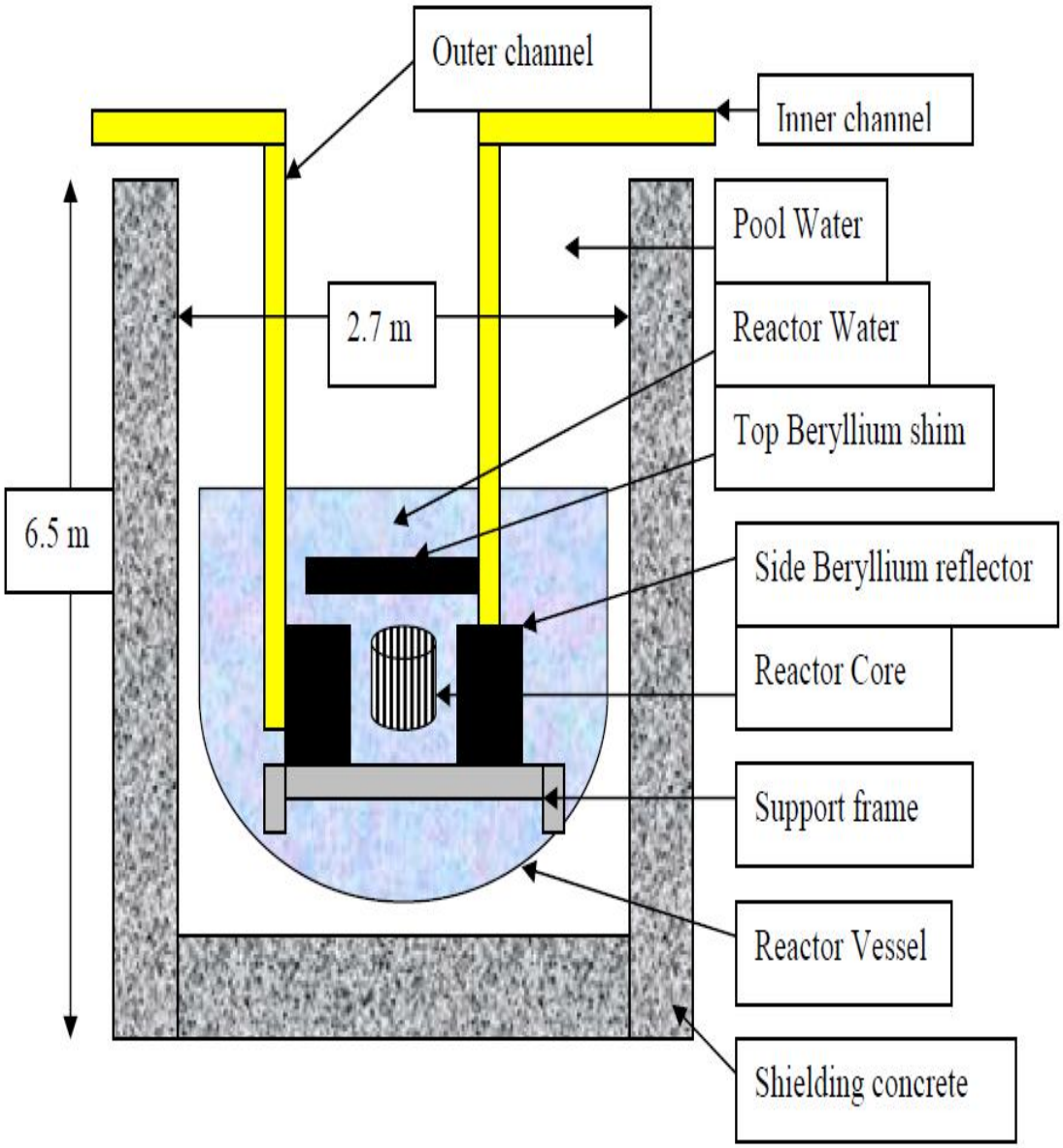


Fig. 2.6 Schematic diagram of MNSR-side view-(Ahmed, 2006)

2.9.1 NIRR-1 Operation and Safety Features

NIRR-1 is operated exclusively using two 500 kVA diesel power generators installed onsite. In addition to this, the control console, the micro-computer control system and the two rabbit systems are all connected to the power source through un-interrupted power sources for safety reasons. In the event of inadvertent release of total core excess reactivity, which may occur due to rod-stuck incident, reactor can be shut down by inserting Cd rabbits using either the rabbit systems or manually through a provision in NIRR-1, which allows for easy access to rector top. Reactor operators are licensed by the Nigerian Nuclear Regulatory Authority (NNRA) and are classified as Senior Operator (.Class A), Operators (Class B) and Trainee Operators (Class C). The qualifications, functions and training requirements of these reactor personnel are enumerated in the final safety analysis report. NIRR-1 was licensed by NNRA on June 01, 2004 to operate at maximum power of 31 kW for not more than 4½ hour per day, three days in a week (Tuesday, Wednesday and Thursday). The other two days (Monday and Friday) are for water purification and routine maintenance of the facility. From experience, during normal operation the reactor water temperature varies between 23.0 °C and 46.0 °C, whereas the temperature difference rises rapidly and attains a stable value due to the phenomenon known as 'insufficient natural circulation' A very important feature of NIRR-1 and similar facilities is the inherent safety due to limited core excess reactivity designed to be less than ½ β_{eff} and large negative total temperature coefficient of reactivity. This design feature of NIRR-1 limits the operation time to a maximum of 4½ hours at full power. Operational experience has revealed that for routine NAA works, it is sufficient to operate NIRR-1 at ½ power, which corresponds to a neutron flux of 5.0 x 10^{11} n/cm^2.s in the inner irradiation

channels. During steady state operation, the control rod is gradually withdrawn to compensate for reactivity loss due the large overall negative temperature coefficient of reactivity. Consequently, magnitude of the neutron flux remains constant during operation. The stability of the neutron flux is one of the hallmarks of NIRR-1 and similar facilities, which run on this same fuel loading for over 10 years. Since fuel is not modified at all for its entire core life time, the neutron spectrum in the irradiation channels does not change and the neutron flux is reproducible to within 1%. This allows convenient neutron activation analysis without the need to continually repeat the standardization measurements for all elements (Jonah *et. al*, 2008).

At low power, when fission rate is too low to generate much heat, addition of positive reactivity via control rod withdrawal produces an exponential rise (i.e a constant log rate increase). In power levels where there is significant heating, temperature changes in the fuel, moderator and coolant leads to changes in reactivity-an effect that occur quite quickly. At low power, the power will increase exponentially. However, as the temperature of the fuel increases, the reactivity will decrease and cause the rate of power increase to slow. Later, the reactivity will decrease even more as the light water (moderator) becomes warmer. The total decrease in reactivity will eventually be enough to cause the reactor to become subcritical and as such the power will reach a maximum and begin dropping. This behavior of the reactor is self-regulating since temperature changes induced by a power increase reduces reactivity and so prevents power from increasing indefinitely (CNSC, 2003).

The MNSR (NIRR-1) is designed to incorporate some inherent safety measures in order to prevent radiation leakage. The objective of its design is to achieve a small core structure with under- moderation, achieved by choosing highly enriched uranium as fuel

material with light water serving as both moderator and coolant (Yang 1992). The core structure is simplified by arranging the irradiation channels in a beryllium reflector (shown in fig. 2.1). This reduces the effect of flux distribution on the irradiation experiments due to movement of the central control rod (Balogun, 2003) thus, enhancing the performance of the reactor from both mechanical and computational physics points of view.

The two major provisions made in the design to increase the safety margin and to enhance the inherent safety of the reactors are (Ahmed *et. al*., 2006)

(i) The 197 ratio of hydrogen to Uranium 235 enlarges the degree of under-moderation process of the core and thereby increases the negative temperature coefficient and enhances the temperature feedback.

(ii) The ratio of the core height to diameter was optimized to 1.0 to take care of axial neutron leakage fraction, to increase the coupling of outlet and inlet flow, mixing of the water coolant and to enable part of the warm coolant to directly enter the core, to shorten the time of temperature feedback and to increase the worth of the top beryllium reflectors-used for the extension of the lifetime of the core.

CHAPTER THREE

METHODOLOGY

3.1 CONTROL ROD CALIBRATION BY POSITIVE PERIOD METHOD:

One of the most frequently used methods of control rod calibration is the positive period method. Period method is a method to measure a positive reactivity inserted into a core at critical state through positive reactor period (Misawa *et al*., 2010). It is an absolute method that permits the direct plotting of reactivity versus rod position. The reactivity of the system is related to the stable reactor period (time for power to change by the factor (e) through the "inhour equation" (Leslie, 1967). During measurement, the reactor is in supercritical operation mode. Control rod calibration has been performed with the positive period method and the comparison method in the modification works and outline of performance tests (Hidekini et. at., 1998).With the rod in different withdrawn positions, the reactor operated at low power and the reactor period was determined. Only the rod to be calibrated was in down position. If P (t) is the reactor power at time t and P(0) is the power at time 0s when starting the experiment, the following relation exists (Eq. 2.6):

P(t) = $P_0 e^{\frac{t}{T}}$

T = reactor period in (s)

If P_0 is the initial power, then after doubling time t,

$$P(t) = 2P_0 \qquad \ldots(3.1)$$

And eqn. (3.1) becomes

$$2P_0 = P_0\, e^{\frac{t}{T}}$$

and $T = t/\ln(2) = 1.4427t$

Thus the period T is given by

$$T = 1.4427t_{av} \quad \ldots(3.2)$$

The uncertainty in periods were calculated using (Agustin et al, 2009)

$$U_T = \frac{\sigma_t}{\sqrt{N}} \quad \ldots(3,3)$$

where σ_t= uncertainty in doubling time

N = number of doubling time values.

where t_{av} = average time (doubling time)

while the uncertainty in integral reactivities were obtained using the formula (Agustin et al, 2009)

$$\sigma(\rho_{int}) = \sqrt{\sum_{i=1}^{N} \sigma^2(\rho_{int,i})} \quad \ldots(3.4)$$

Where $\sigma(\rho_{int})$ is the uncertainty in Integral reactivity

Normally one measures reactivities by the positive period method by waiting a sufficient length of time after a reactivity insertion for transient terms to die out so that the time-dependent flux approaches a pure exponential from which the asymptotic period is determined. The Inhour equation then relates the asymptotic period to the reactivity. If the temperature of the moderator and fuel changes during the transient, however, the time-dependent flux never approaches the shape of a pure exponential (Bretscher, 1987). During the exercise, reactivity value of the control rod was determined by measuring the stable reactor period. The physical background of the evaluation method is the INHOUR equation (eq.2.7). Result expected from the exercise is the integral reactivity curve (total reactivity).

As described above, the reactivity equation (Inhour equation) provides the relationship between reactor period, *T*, and the reactivity ρ. The period can be determined by noting the time required for a power reading system (such as recorder) to change by a factor of 2. This reading was started after a while and repeated a few times to ensure that the reactor is on a steady period and that the initial transient change caused by movement of the rod and prompt neutrons has disappeared. From these readings the periods were easily determined, and hence the rod worth.

The materials needed to perform this experiment include:

- Stop watch (chronometer)
- Control console

3.2 EXPERIMENTAL PROCEDURE

1. The control rod calibration data was used.
2. The reactor was brought critical at low power ($5 \times 10^9 n/cm^2 s^{-1}$). Record of the critical power and the corresponding control rod position was made.
3. The control rod was then withdrawn from the critical position by 10 mm.
4. Using an appropriate stopwatch, the time it took for the reactor power to be multiplied by 2 (power doubling time) was measured for each withdrawal position and the value of the doubling time was recorded.
5. Step (4) was repeated three times to obtain three values of doubling time.
6. An average doubling time (t_{av}) for each rod withdrawal position was determined in seconds.
7. Thus the period was calculated using the relation $T=1.4427t_{av}$

8. Using the period value and the Inhour data available in the control room, the corresponding reactivity (mK) for each rod withdrawal position was determined and their values recorded.
9. The reactor was brought back to the original critical position at low power level.
10. Steps 2 to 8 were repeated until the control rod is withdrawn to 177mm. For every withdrawal, it was ensured that the control rod was first brought to critical power before withdrawing to the next step of additional 10mm.
11. To achieve the best possible step withdrawal as possible, the control rod was rapidly withdrawn while ensuring that the desired control rod position mark was not exceeded.
12. Using the Excel data sheet a plot of the curve of Integral reactivity (mK) versus rod position was obtained.

CHAPTER FOUR

RESULTS

4.1 CONTROL ROD CALIBRATION RESULT

Data of control rod positions and doubling times were obtained from the calibration experiment. The procedures for obtaining this data have been explained in the previous chapter. The tables below show the data obtained from the experiment.

Table 4.1 Values of Rod position, doubling time, average doubling time, Period, Uncertainty in Period, Reactivity, Integral Reactivity, uncertainty in Reactivity and Integral Reactivity.

Rod position (mm)	t_1(s)	t_2(s)	t_3(s)	t_{av}(s)	T(s)	U_T (s)	ρ (mk)	ρ_{int} (mk)	$\sigma(\rho)$ (mk)	$\sigma_{int}(\rho)$ (mk)
107	-	-	-	-	-	-	-	-	-	-
117	192.68	159.08	174.85	175.54	253.25	9.71	0.41	0.41	0.01	0.01
127	61.59	51.17	46.20	52.99	76.45	4.53	0.65	1.06	0.04	0.04
137	34.oo	28.84	17.65	29.42	42.44	2.49	0,52	1.58	0.03	0.05
147	22.88	17.65	15.53	18.69	26.96	2.18	0.50	2.08	0.05	0.07
157	16.45	12.47	11.33	13.42	19.36	1.55	0.37	2.45	0.04	0.08
167	14.00	8.85	5.51	9.45	13.63	2.47	0.55	3.00	0.29	0.30
177	11.01	7.27	4.40	7.56	10.91	1.91	0.34	3.34	0.17	0.11

CHAPTER FIVE

DISCUSSION

5.1 CONTROL ROD CALIBRATION

The control rod calibration for the length of the rod (107mm – 177mm) at 10mm step increment was achieved by observing the effect of a movement of the control rod on the power level ($5 \times 10^9 n/cm^2 s^{-1}$ critical power) of the NIRR-1. This was done at low power so that there was little effect on reactivity due to temperature variation.

5.2 CONTROL ROD POSITION AND REACTOR PERIOD

The result in Table 4.1 shows the behavior of the NIRR-1 power doubling time with increasing rod control rod position. It can be seen from the table that the doubling time decreases with increasing rod position until the rod is withdrawn to 177 mm position. The average values of the doubling time were used in the calculation of the reactor period in the table. When the control rod was withdrawn by a step of 10mm (the step withdrawal here was an approximation of an essentially ramp withdrawal) to a predetermined level, the neutron flux in the core began to buildup. This could be attributed to the fact that the rod withdrawal resulted in an increase the thermal fuel non-leakage factor (*f*) and consequently in fission cross-section of the core. This made the reactor slightly supercritical as expected from theoretical prediction (DOE, 1993). A similar trend exists between the reactor period and control rod position as seen in the table in that the period is inversely proportional to the control rod position. The values of the period in the table were therefore used to obtain Figure 5.1. The figure indicated an exponential drop in the reactor period as the control rod

moves further away from the critical position. This is in agreement with theoretical considerations where the rod is presumed to serve as reactivity insertion or removal tool. The Table also shows the uncertainties related to the calculated periods with values found to be within the predicted experimental margins (Agustin *et al.*, 2009). The results obtained shows that the period (*T*) is related to the doubling time. This is also an agreement with Leslie, 1967, which says a long reactor period corresponds to a slow power rise. It is thus an indication that the smaller the value of *T*, the more rapid the change in reactor power and vice versa.

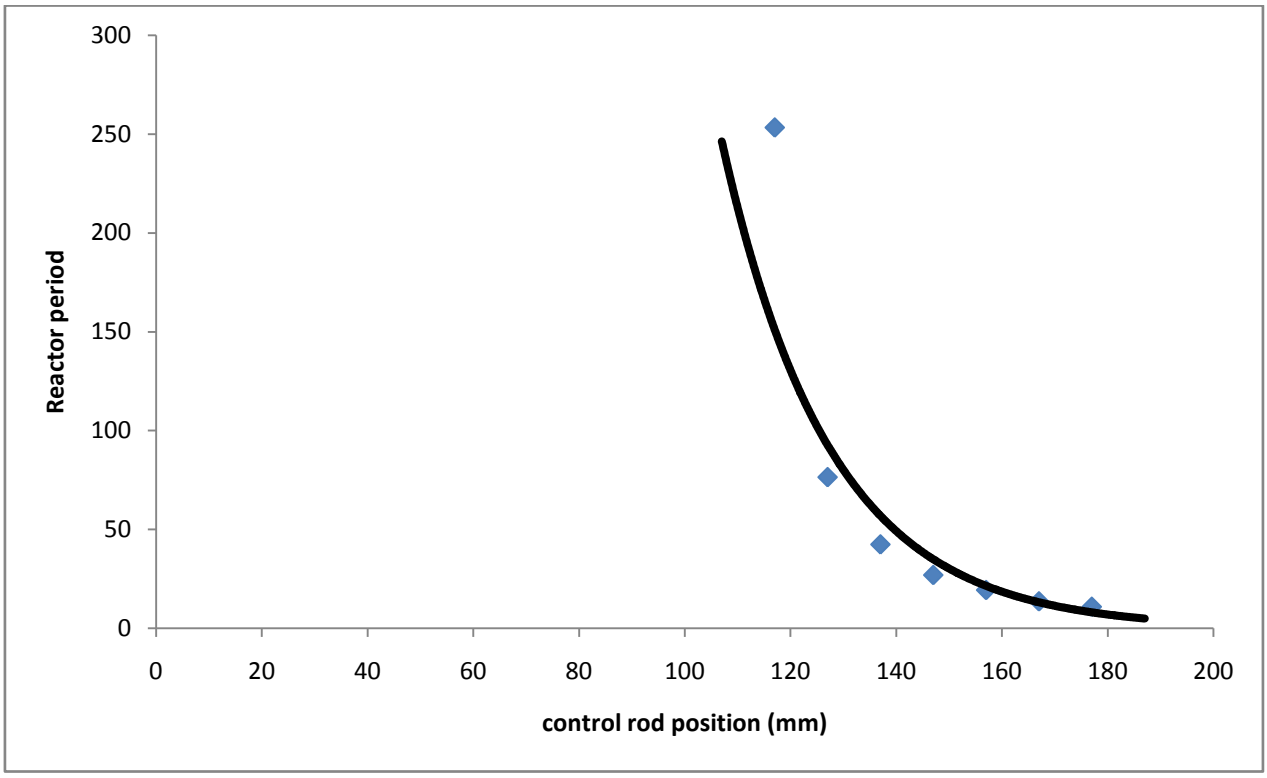


Fig.5.1: Relationship between reactor period and rod position

5.3 THE INTEGRAL ROD WORTH CURVE

The reactor periods obtained in Table 4.1 were used in the Inhour equation for MNSRs proposed by Zheng 1993 (equation 2.17a) to obtain the reactivities and the integral reactivities. The values of the integral reactivities were plotted against withdrawal positions to obtain Fig.5.2. The figure was compared with the s-shape integral curve as earlier obtained for NIRR-1 during the zero power experiments in China (Fig. 5.3). The two figures (5.2 and 5.3) were found to be in agreement with each other and with theoretical prediction (Zheng, 1993). The exception is that the integral rod worth of (3.34 ± 0.11)mk obtained here was for 107mm to 177mm length and not for the full withdrawal length. The limitation to the length set in this work was for safety and precautionary reasons. The safety analysis report (SAR, 2011) of NIRR-1 does not allow the introduction of all the excess reactivity (3.77)mk of the reactor. Therefore, an extrapolation of the upper end (177-230)mm yielded the perfect s-shape curve in Fig. 5.3. The (3.34 ± 0.11)mk integral rod worth obtained was for the (107=177)mm whereas the 7mk total rod worth obtained by the Chinese manufacturers of MNSR (Zheng 1993) was for the full withdrawal length (0 – 230mm). This shows that our integral rod worth curve is in line with the values obtained for the NIRR-1 calibrated rod during the zero power experiments in China (Fig. 5.3). It is also slightly above the 3.0mk obtained for the same calibration length for Pakistan MNSR measured at Zero-Power critical facility (Zheng, 1993). The increase can be attributed to the slight increase in power of NIRR-1 (31kW) compared to the Pakistani 30kW. The result showed that the length of the rod calibrated accounted for a larger part (47.8%) of the total rod worth and thus the most reactive portion of the control rod.

It can generally be said that the result in this work is an indication that 10mm withdrawal length is approximately 0.488mk average reactivity insertion and the length

used here is the most reactive portion of the control rod. The 3.29% uncertainty for the integral rod worth obtained was within experimental margins (Agustin *et al.*, 2009). The integral rod worth curve obtained in fig.5.2 also shows that only 30% of the control rod length is required to achieve the characteristic curve using the positive period method.

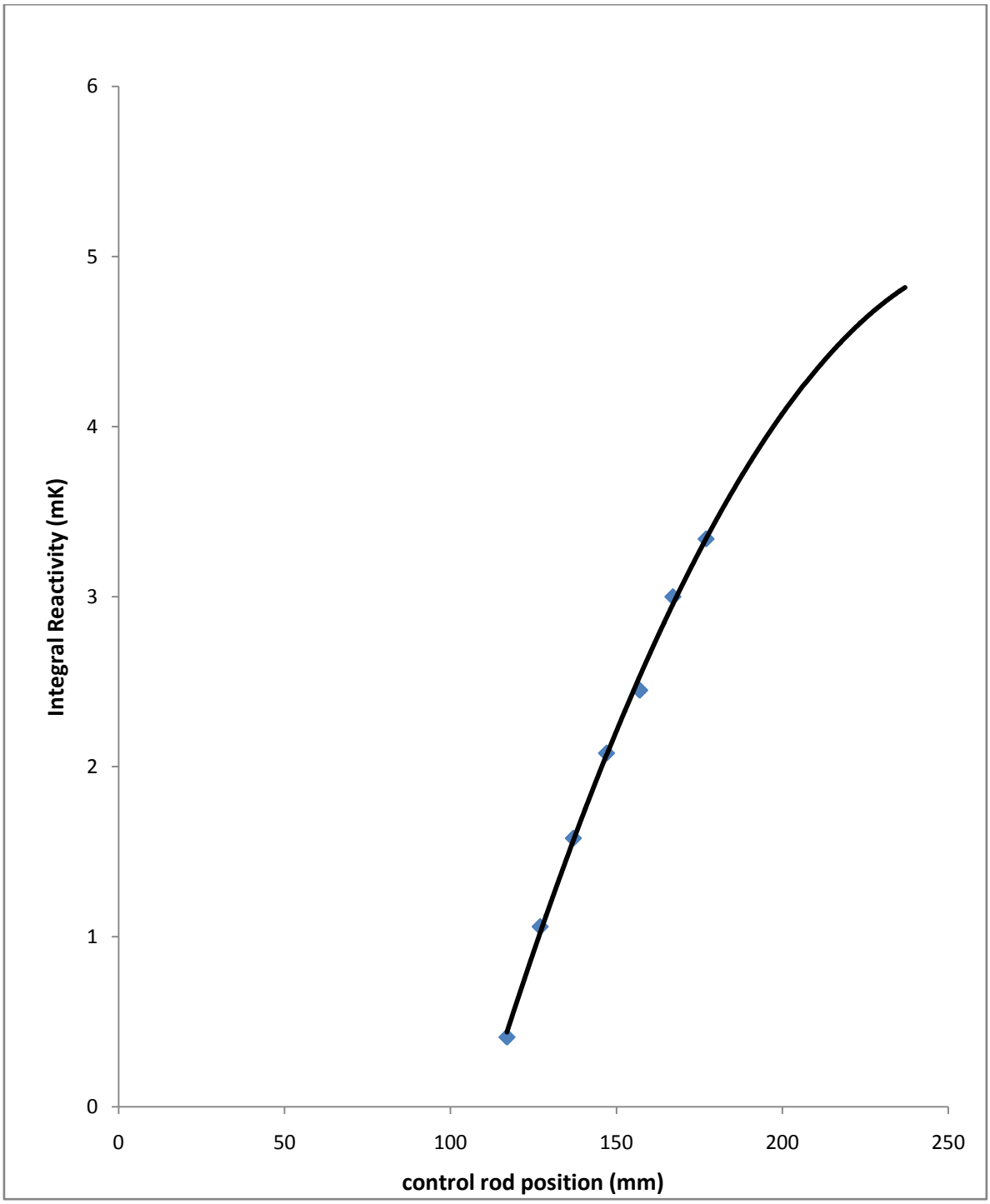


Fig.5.2: Integral rod worth curve of the control rod

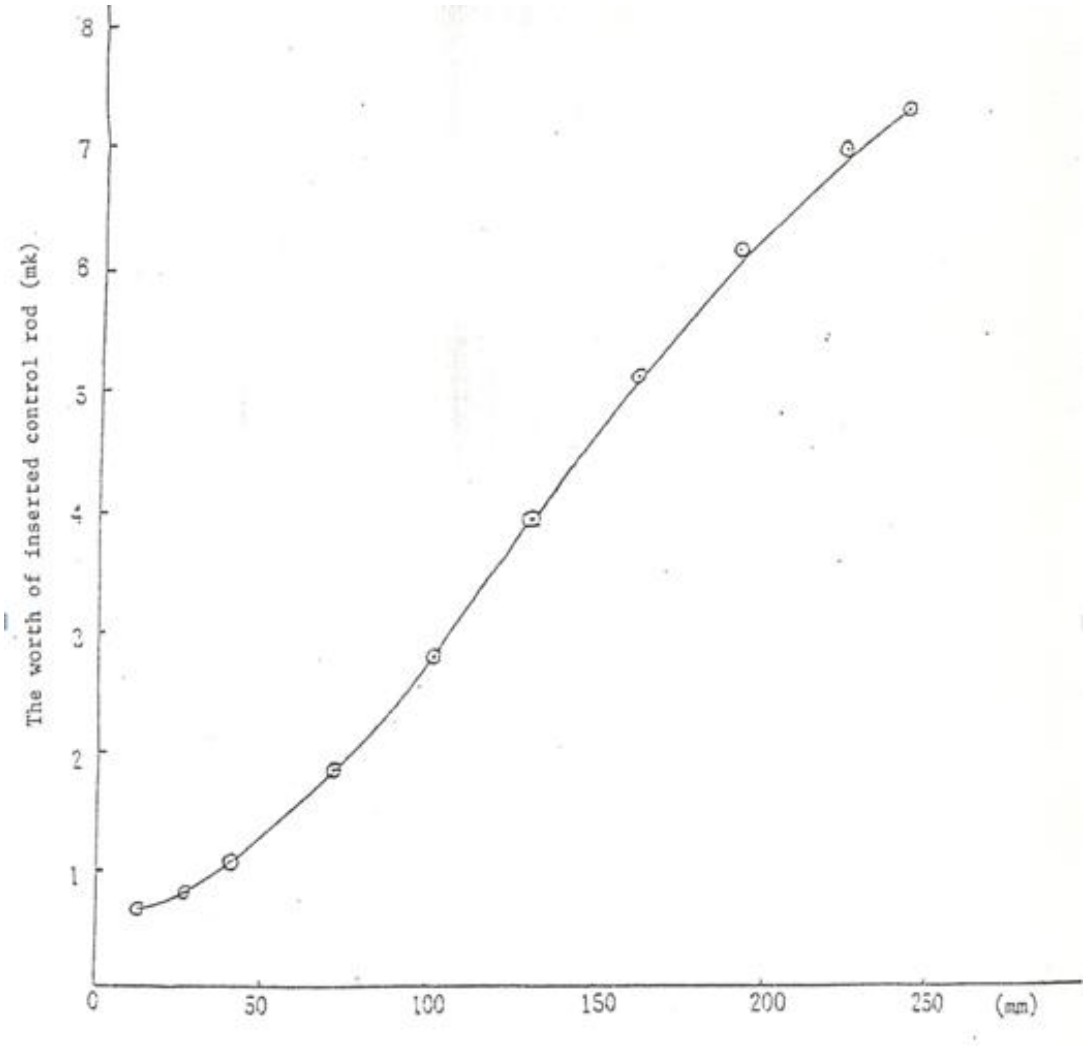


Fig. 5.3: NIRR-1 Integral Curve during Zero Power Experiments in China (Zheng, 1993)

CHAPTER SIX

CONCLUSION AND RECOMMENDATION

6.1 CONCLUSION

The Positive Period Method was used in this work to calibrate and determine the integral worth of NIRR-1 control rod. The method was found to be suitable to the core configuration and control rod mechanism of NIRR-1 MNSR. The control rod was calibrated from 107mm to 177mm withdrawal length at a low flux of 5×10^9 ncm^2s^{-1}. The behavior of the reactor power doubling time with increasing rod control rod position shows that the latter decreases with increasing rod position. The reactor periods obtained were used in the Inhour equation for MNSRs to obtain the reactivities and the integral reactivities. The values of the integral reactivities were plotted against withdrawal positions to obtain a figure that compared well with the s-shape integral curve earlier obtained for NIRR-1 during the zero power experiments in China. Our result shows that 10mm withdrawal length is approximately 0.56mk reactivity insertion and the length used here is the most reactive portion of the control rod. The work established that routine calibration of the MNSR control rod via positive period method is possible, and satisfied all safety requirements in that the rod contains enough worth to bring the reactor to a shutdown (shutdown margin = 2.5mk, SAR, 2011). This work also observed that although the Positive period method is a relatively accurate method of reacivity determination (Leslie, 1967), there are limitations to its experimental procedures and consequently accuracy for MNSRs with a single central control rod. The problems associated with positive period for MNSRs as observed in this research are:

i. The reactor cannot (without adding or removing fuel elements) cannot be made critical with a full insertion of the control rod thus limiting the length of the control rod that can be calibrated.

ii. Step withdrawal cannot be achieved because the control rod drive mechanism of MNSR does not support step withdrawal by design and as such the stepwise rod withdrawal is essentially a rapid Ramp withdrawal.

6.2 RECOMMENDATION

a. The positive period method should be exploited in the routine calibration of NIRR-1 in conjunction with other methods.

b. The interval of length calibrated in this work should be exploited in reactivity studies.

c. Other methods should be exploited in calibrating the full length of the control rod.

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APPENDIX

(Appendix 1)

Parameters of delayed Potoneutron in beryllium

Group	$\lambda_i\ (s^{-1})$	$\beta_j(\times 10^{-5})$
1	6.24×10^{-7}	0.057
2	2.48×10^{-6}	0.038
3	1.59×10^{-5}	0.260
4	6.20×10^{-5}	3.200
5	2.67×10^{-4}	0.360
6	7.42×10^{-4}	3.680
7	3.60×10^{-3}	1.850
8	8.85×10^{-3}	3.660
9	2.26×10^{-2}	2.070

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