THERMAL POWER CALIBRATION OF NIGERIA RESEARCH REACTOR-1 BY CALORIMETRIC AND HEAT BALANCE METHODS

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BY

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DECLARATION

I declare that the work in this thesis entitled **"Thermal Power Calibration of Nigeria Research Reactor-1 (NIRR-1) by the Calorimetric and Heat Balance Methods"** has been carried out by me in the Department of Physics, Ahmadu Bello University, Zaria. 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.

_____ _____

Sunday Arome AGBO **Date**

CERTIFICATION

This thesis entitled **"THERMAL POWER CALIBRATION OF NIGERIA RESEARCH REACTOR-1 (NIRR-1) BY CALORIMETRIC AND HEAT BALANCE METHODS"** by Sunday Arome AGBO meets the regulations governing the award of the degree of Masters of Science (Nuclear Physics) of the Ahmadu Bello University, and is approved for its contribution to knowledge and literary presentation.

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DEDICATION

To

the memory of my father, Shaibu Agbo

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I am eternally indebted to my supervisors, Dr. Y.A. Ahmed (Deputy Dean Student Affairs, ABU, Zaria and Senior Research fellow, Nuclear Science and Engineering Section, Centre for Energy Research and Training, Ahmadu Bello University, Zaria) and Prof. I.O.B. Ewa (Former Minister of Science and Technology, Federal Republic of Nigeria and Professor at Centre for Energy Research and Training, Ahmadu Bello University, Zaria) for their sincere guidance and tireless support throughout my work. Their patience, critiques and detailed explanations of concepts have really helped me in completing this research work.

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ABSTRACT

This research work presents the results of the thermal power calibration of Nigeria Research Reactor-1 (NIRR-1), a low power Miniature Neutron Source Reactor (MNSR), using calorimetric and heat balance methods. The calibration was performed at two different power levels: half power (15kW) and full power (30kW). The calorimetric method involved operating the reactor at a constant power and isolating coiling system to avoid cooling of the pool water. For this method, the reactor pool temperature rise with time was measured, the reactor heat capacity constant was calculated and the heat losses from the reactor pool to the environment were evaluated. The power was then evaluated as a function of temperature-rise rate. The total thermal power is the sum of heat losses from the reactor pool to the environment and the power calculated as a function of temperature-rise rate. The heat balance method consisted of the steady state energy balance of the cooling loop of the reactor. For this method, the measurement of the inlet temperature, outlet temperature and temperature difference were carried out and also flow-rate of the coolant passing through the core was determined. The heat losses were evaluated and the values added to the power calculated as a function of flow rate and temperature difference gives the total thermal power. The Average water temperature rise during these experiments were: (29.5 to 32.0)^{o}C and (27.0 to 29.6)^{o}C for calibration at half and full power levels respectively .The thermal power obtained by the calorimetric method at half and full power were: (15.8±0.7) kW and (30.2±1.5) kW respectively. For heat balance method, the values of the thermal power obtained at half and full power were: (15.2±1.2) kW and (30.7±2.5) kW respectively. It was observed that the calorimetric method is more accurate with deviations of only 4% and 5% for calibrations at half and full power respectively. It is therefore recommended that the calorimetric method should be adopted for routine thermal power calibration of NIRR-1.

Table of Contents

	Title Page	i
	Declaration	iv
	Certification	v
	Dedication	vi
	Acknowledgment	vii
	Abstract	viii
	Table of contents	ix
	List of Figures	xii
	List of Tables	xiii
CHAPTER ONE (Introduction)		
1.1	Introduction	1
1.2	Aim and Objectives	5
1.3	Statement of Research Problem	6
1.4	Justification	6
1.5	Previous work	7
CHAPTER TWO (Theory)		
2.1	Cross Section and Neutron Flux	12
	2.1.1 Cross Section	12
	2.1.2 Neutron Flux	13

2.1.3 Reaction Rates 15

2.2 Research Reactors 15

2.2.1 NIRR-1 Description 17

2.2.2 NIRR-1 Design 21

2.2.3 Safety Features of NIRR-1 24

2.3 Reactor Power 25

2.3.1 Reactor Power Level 25

2.3.2 Reactor Power Calculation 26

2.3.3 Relationship between Neutron Flux and Reactor Power 27

2.2.4 Reactor Power Calibration 27

a) Calorimetric(slope) method of reactor Power Calibration 28

b) Thermal balance (ballistic) method of reactor power calibration 30

2.4 Dependency of NIRR –Reactor Power on Moderator Temperature 32

2.5 Transient Characteristics of NIRR-1 Reactor 34

2.6 Reactivity Coefficients 35

2.6.1 Moderator Temperature Coefficient 35

2.6.2 Fuel Temperature Coefficient 39

2.7. Reactor Dynamics 40

2.8 Dynamic Feedback Characteristic of NIRR-1 41

2.9 Heat Transfer Phenomena in NIRR-1 42

2.9.1 Energy Equations 43

2.9.2 Heat losses 44

(a) Heat Lost through Conduction 44

(b) Heat Losses by Evaporation. 48

(C) Heat Losses by Convection. 49

CHAPTER THREE (Material and Method)

3.1 Materials and Methods 53

3.1.1 Calorimetric (slope) Reactor power Calibration 53

3.1.3 Calorimetric Method Calibration Procedure 53

3.2 Heat Balance (Ballistic) Reactor Power Calibration 55

3.2.1 Heat Balance Method Calibration Procedure 55

3.3 Instrumentation 57

CHAPTER FOUR (Results)

4.1 Time and Temperature Data for Heat Balance Method 61

4.2 Time and Temperature Data for Calorimetric Method 65

CHAPTER FIVE (Discussion)

5.1 Calorimetric (slope) method calibration 72

5.2 Heat balance (Ballistic) method calibration 73

5.3 Calorimetric method uncertainty 75

5.4 Heat balance method uncertainty 75

CHAPTER SIX (Conclusion and Recommendation)

6.1 Conclusion 77

6.2 Recommendation 78

References 79

List of Figures

Figure 2.1: Schematic diagram of the coolant flow pattern of NIRR-1 19

Figure 2.2: Schematic diagram of MNSR 20

Figure 2.3: Thermal circuit diagram for NIRR-1 52

Figure 3.1: A layout core configuration showing the various components. 59

Figure 4.1: Temperature rising rates during calorimetric power calibration procedure 71

List of tables

Table 2.1: Technical Specifications of NRR-1 22

Table 2.2: Design Parameters of NIRR-1 Fuel 23

Table 2.3: Variables Affecting NIRR-1 Heat Conduction 51

Table 4.1: Time and Temperature Data for Heat balance method at 15kW 60

Table 4.2: Calculated parameters for Heat balance method at 15kW 61

Table 4.3: Time and Temperature Data for Heat balance method at 30kW 62

Table 4.4: Calculated parameters for Heat balance method at 30kW 63

Table 4.5: Time and Temperature Data for Calorimetric method at 15kW 64

Table 4.6: Calculated parameters for Calorimetric method at 15kW 65

Table 4.7: Time and Temperature Data for Calorimetric method at 30kW 66

Table 4.8: Calculated parameters for Calorimetric method at 30kW 67

Table 4.9: Comparison of Uncertainties 69

CHAPTER ONE

INTRODUCTION

1.1 INTRODUCTION

Fuel burn-up was shown to be linearly dependent on the reactor thermal power (Podvratnik, 2011). It is therefore obvious that the reactor thermal power calibration is very important for precise fuel burn-up calculation. The reactor power can then be determined from measuring the absolute thermal neutron flux distribution across the core in horizontal and vertical planes (Musa *et al.,* 2012). It has also been established that flux distributions can be measured with activation of cadmium covered and bare foils irradiated at steady reactor power (Souza, 2002). But Shaw (1969) demonstrated that this method consumes a lot of time and is not accurate. It can therefore be said that the foil activation method is most suited for zero power reactors and seldom applied to bigger reactors. In the case of high power reactors in which a temperature rise across the core is produced and measured then a heat balance method is the most common and accurate method of determining the power output of the core (Mesquita *et al.,* 2007).

Accurate reactor thermal power calibration is important for: safe monitoring and evaluation of reactor dynamic behavior, determination of fuel burn-up and normalization of neutron fluxes and dose rate. (Mesquita *et al*., 2007; 2009; 2011, Podvratnik, 2011). Power excursion of any reactor is of great concern to reactor physicist for safe operation reasons. As power is related to the neutrons population and to the mass of fissile material present, its measurement is essential to the safe control and operation of the reactor as well as the reliability of the research reactor (DOE, 1993, Podvratnik, 2011). It therefore became imperative to undertake power measurements and calibration from time to time to establish

the stability of the reactor and evaluate its fuel burn-up. Reactor power calibration helps in maintaining and improving safety (Beamex 2014). A lot of literatures show how safety is based upon never exceeding established operation limits such as reactor power (Lopez, 1961, Zagar *et al*., 1999, Mesquita *et al*., 2007, 2011, Beamex 2014). A byproduct of improved calibration is an improvement in safety. (Beamex 2014).

As the reactor is operated, energy is released through the fission process. The majority of this energy appears as energy carried by fission fragments, gamma rays, neutrons and beta particles emitted (Jevremovic, 2008). When these particles interact with the surrounding materials, heat is produced. This process heats up the fuel meat and starts the chain of heat transfer The heat released from nuclear fuel is transferred to the coolant through the heat conduction of fuel and cladding, and the heat convection between cladding outer surface coolant. Then, the heat is carried out of the reactor core by the coolant.

The thermal-hydraulic design of Nigeria Research Reactor-1 (NIRR-1), intended as with no reactivity insertion, is closely dependent on the structural design of the reactor. The core is cooled by natural convection which is established through the heat generated by fission occurring in the core. The reactor coolant is drawn through the inlet orifice by natural convection flow through the channels within the fuel elements. The coolant moves up through the core and exits through the core outlet orifice at temperatureT_2 to the upper part of the tank where the temperature isT_3. The coolant inside the core passes through the aperture surrounding the upper ends of the fuel. New colder coolants is substituting the hotter one in the core causing the coolant in the downcomer (with temperature T_1) to move

downwards and maintain its temperature so that the coolant enters the reactor core at temperature T_1. In the simplified model, the velocity of the coolant in the downcomer is supposed to be equal to the velocity of the coolant in the core. Heat transfer from core to the reflector is by conduction and convection and also heat transfer from the tank to the pool (T_0) across the wall of the tank is again by conduction and convection mechanisms

While there has been a great deal of research on the behavior of NIRR-1 power resulting from changes in core-coolant temperature (Ahmed *et a*l., 2008), this study focuses on the heat transfer in the system as a whole. The fuel rods in the NIRR-1 core heat the surrounding water which flows through the fuel channels then up and over the rods in a natural convection manner. The heated water then can either evaporate from the open top of the reactor pool or the heat is transferred to the environment. Transferring the heat to the environment can be either through the top surface which is open to the air or through the aluminum tank wall, concrete barrier and outer steel tank wall. Currently, through one of these two methods is the only way the pool water is cooled back to ambient temperature.

SAR, 2005 pointed out that the decay heat in NIRR-1 is removed by natural circulation of the reactor vessel water in a manner similar to that during operation. It has been established in SAR, 2005 that a number of thermal hydraulic tests and calculations, especially from the dynamic experiments, have shown that the natural circulation of the prototype MNSR (which is comparable to NIRR-1) has the following characteristics:

i) Negative Feedback Effect:

This occurs when the temperature difference between the inlet and outlet coolant of MNSR increases; the floating force and circulating head will increase to make the volume flow rate and flow velocity high, which in turn will limit the rise in temperature.

ii) Insufficient Circulation:

Because of the small size of the core, the distance from the inlet orifice to the outlet orifice is small. The water, after being heated in the core goes out through the upper part of the core. Part of the water does not get sufficiently cooled before it sinks down, resulting in part of outlet water being carried back into the core due to siphoning effect. This direct re-circulation of the part of hot water causes a rise of the inlet water temperature. This phenomenon is called insufficient circulation, which speeds up the rise of the coolant temperature in the core and shorten the function time of the temperature effect. It is thus not possible to cause the inlet water temperature to rise in such a short time by heating the core only, but by the coupling action between the inlet and the outlet coolant. Consequently this offers some benefit to the reactor safety

For safe operation of NIRR-1, it is imperative to measure its power accurately. There are different methods reported for measuring the reactor power such as thermal hydraulic methods (Ahmed *et.al,* 2006; 2008), neutron flux measurement (Ahmed *et. al.*, 2008; Musa *et al.,* 2012), Cherenkov radiation intensity measurement (Arkani *et al.,* 2009) and ^{16}N gamma activity (Hamid *et al.*, 2011).

The neutron flux method of measuring reactor power can be handled through the foil activation method, calorimetric (slope) method and heat balance method. Earlier, power measurement has been conducted using the foil activation method (Musa *et. al.*, 2012). The present work therefore, to explored the calorimetric (slope) and heat balance methods which were hitherto not used to measure and calibrate NIRR-1 power.

1.2 AIM AND OBJECTIVES

The aim of this research work is to use calorimetric and heat balance methods in the thermal power calibration of Nigeria Research Reactor-1 (NIRR-1).

The following objectives would be achieved.

1 To analyze the accuracy of calorimetric and heat balance methods in NIRR-1 thermal power calibration.

2 To Calibrate the thermal power of NIRR-1 using calorimetric and heat balance methods

3 To evaluate the dependency of power as a function of temperature-rise rate.

4 To Determine NIRR-1 power using flow rate and coolant temperature difference.

5 To compare between the results obtained and flux parameters on Console to determine best method.

1.3 STATEMENT OF RESEARCH PROBLEM

The most important factor in reactor control is the precise information about reactor power at any moment. Accurate reactor power measurement is required not only for safety purposes but also by research reactor users for experiments. The knowledge of online reactor power level has been a problem to MNSR reactor operators. The Integrated Safety Assessment for Research Reactor (INSARR) mission 2009 has recommended for power display in NIRR-1 control console to have redundant systems or components that perform the same safety function by incorporating into the systems or components different attributes, such as principles of operation, operating conditions and manufacturers for reactor safety and international standard.

The need for accurate power level determination is a safety criteria and operational limiting condition. In the operational limits and conditions (OLCs), we have the power limit which said that the power should not be more than 20% of the full power (SAR, 2005). However, no power is displayed in NIRR-1 control system. Research reactors are often involved with experiments which require an accurate determination of the reactor power. Normalization of experimental measurements cannot be possible unless the statistical variation of the reactor power is known.

1.4 JUSTIFICATION

Thermal power calibrations of low power research reactors (up to 1MW) are normally performed during the initial startup and their results are used for many years (Mesquita *et*

al., 2007; Ahmed *et al.*, 2008; 2011). However, heat capacity usually change when experimental installations are made, changes in the collimator or other mechanical modifications like the case of NIRR-1 installation of cadmium-lined irradiation channel *(*Ahmed *et al.*, 2013). Consequently, the need to calibrate the system with the present installations in and around the core is very crucial.

From the time when NIRR-1 was commissioned in 2004 till date attempts were made to measure its flux stability using Activation foils method (Musa, *et al.*, 2012,),and also to make certain its flux variation (Ahmed, *et al.*, 2008), maximum operable time (Ahmed et. al., 2011). No comprehensive effort was made to calibrate NIRR-1 power despite its usefulness in utilization and experimental work. Available literature (Mesquita *et al.*, 2007; 2009; 2011) showed that slope and heat balance methods are less cumbersome and straight forward with good results. This work explored these methods in the calibration of Nigeria Research Reactor-1 (NIRR-1) power. And recommend ways to adopt one of the methods for routine calibration of NIRR-1 power.

1.5 PREVIOUS WORK

The need to use more than one method to calibrate the power of a reactor became very important as ascertained by (Shortall *et al*., 1954) who used three independent methods (Gold Foil Activation, Fission Product Determination and Calorimetric Methods) in the calibration of CR&D water boiler reactor. They observed that though the three independent methods gave answer to within 13 percent, the accuracy of Fission Product Determination Method was estimated to be within $\pm$ 10 to 15 %. They reported that the calorimetric

method just like its counterparts is the most inherently accurate, straightforward and presumably the simplest method for measuring the power of CR&D water boiler reactor.

The behavior of reactor power level and flux with changes in core-coolant temperature for a MNSR facility was investigated earlier by (Ahmed *et al.*, 2008) at two power levels using NIRR-1. The method used in the measurements was a thermal hydraulic by exploiting core coolant temperature values to predict the reactor power. Their results show that the semi-empirical relationship based on thermal hydraulics and neutronics parameters could be used to predict the reactor's operating power. Also, the data measured indicated that the reactor's operating power level can be estimated from the preset thermal neutron flux value. The measured data shows that there is a strong dependence of the reactor power on coolant temperature in agreement with the design of MNSR. In an effort to extend the earlier work to other methods available for measurements and calibration of NIRR-1 power, we used the calorimetric (slope) and heat balance methods.

Much earlier works like (Lopez 1961) calibrated the power of Ksutmil reactor by noise analysis. He determined the zero power reactor transfer function via frequencies ranging from 2cps to 1000cps. He also determined the break frequency by a least squares fit of a function of the form $A+{}^{B}/_{(\alpha^2+\omega^2)}$. The value obtained was 19.3 ± 0.6 was said to compare favorably with the value obtained by other methods. Knowledge of transfer function was used in the noise analysis to calculate detector efficiency and charge transferred per neutron absorbed and therefore calibrated the reactor power at several levels. He reported that at low power levels, the effects of instrument noise was important

enough and overrides the pile noise while at high powers, feedback temperature effects are present and that the calculated power level was compared with the power indicated by the linear recorder of the reactor console, the results agreed within 20%. He finally concluded that the determination of zero power reactor transfer function is possible although it involves a considerable amount of effort and that the determination of reactor power by noise analysis seems rather cumbersome and probably of no application to power and research reactors.

Recently, (Podvratnik, 2011) preformed calculations to support absolute thermal power calibration of Slovenian TRIGA Mark II reactor. He found that thermal power calibration with an absolutely calibrated fission chamber proves to be viable. He reported that knowing the accurate mass of the uranium contained inside the fission chamber and accurate calibration of the fission chamber is crucial in correct thermal power calibration. This informs the earlier use of absolute method in the determination of NIRR-1 power (Musa *et al.,* 2012). The work which we seek to extend here is by utilizing the calorimetric and heat balance methods in the calibration of the NIRR-1 reactor power.

Recently, (Ahmed, *et al.,* 2011) measured the available excess reactivity of NIRR-1 using the method of control rod critical depth of insertion and a method in an effort to measure the reactor's power and obtain its maximum operable times. Their result shows that there is a strong correlation between the reactor's operating time and its coolant temperature with the core excess reactivity. At 3.77mk excess reactivity it was observed that for a full-power flux with the control rod position unlimited, the reactor operated for five continues hours. At half-power and under the same excess reactivity condition, the reactor reaches 8 h

before the temperature effects sets-in. However, re-measurements done in 2009 shows that excess reactivity of the reactor has reduced to 2.80 mk, the operable time at full flux dropped to 3.5 h while that of half-power became 7 h. They also observed that the reactor's fluence depletion over the period under review was directly related to the excess reactivity reduction. They therefore conclude that for a clean core excess reactivity of 3.77 mk, 5 and 8h are the optimum times for the operation of miniature neutron source reactors under the two flux conditions (full and half power, respectively). As demonstrated in the study, a deviation from these optimum times is an indication of a drift in the reactor's core excess reactivity and hence the need to add beryllium shims to compensate for the lost. The present work therefore ascertains if this need exist using calorimetric (slope) and heat balance methods.

(Mesquita *et al.*, 2011) calibrated the thermal power of the IPR-R1 TRIGA reactor by the calorimetric and the heat balance methods. In calorimetric method, they operated the reactor at a constant power, with primary cooling system switched off. They recorded the rate of temperature rise of the water and calculated reactor power as a function of the temperature-rise rate and the system heat capacity constant. They reported that the heat balance procedure consists of the steady-state energy balance of the primary cooling loop of the reactor. In this method, they measured inlet and outlet temperatures and the water flow in the primary cooling loop. The heat transferred through the primary loop was added to the heat leakage from the reactor pool. They finally concluded that the calorimetric method calibration presented a large uncertainty and that the main source of error was the determination of the heat content of the system, due to a large uncertainty in the volume of the water in the system and a lack of homogenization of the water temperature. They

therefore adopted the heat balance calibration in the primary loop as the standard procedure for calibrating the power of the IPR-R1 TRIGA nuclear reactor.

Recently, (Musa *et al.*, 2012) determined the radial and axial neutron flux distribution in irradiation channel of the NIRR-1 using foil activation technique. This attempt is to characterize the newly installed cadmium-lined irradiation channel in the reactor. The results obtained showed that the installed cadmium line did not affect NIRR-1 flux stability. As an extension of the work, we ventured here into thermal power calibration of NIRR-1 by calorimetric and heat balance method.

From the foregoing, it became evident that using more than one method to calibrate NIRR-1 power is the most preferred option. This if done, will give room for analyses of the accurate method and will guarantee reliable results. And that is the purpose of this research work. The present work therefore uses calorimetric (slope) and heat balance methods in the calibration of NIRR-1 power.

CHAPTER TWO

THEORECTICAL CONSIDERATIONS

2.1 NEUCLEAR CROSS SECTIONS AND NEUTRON FLUX

To determine the frequency of neutron interactions, it is necessary to describe the availability of neutrons to cause interaction and the probability of a neutron interacting with material. The availability of neutrons and the probability of interaction are quantified respectively by the neutron flux and nuclear cross section. (DOE, 1993).

2.1.1 Cross Section

The probability of a neutron interacting with a nucleus for a particular reaction is dependent upon not only the kind of nucleus involved, but also the energy of the neutron. Consequently, the absorption of a thermal neutron in most materials is much more probable than the absorption of a fast neutron. Also, the probability of interaction will vary depending upon the type of reaction involved. (Foster, 1977)

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

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

square centimeter as being "as big as a barn" when applied to nuclear processes. The name has persisted and microscopic cross sections are expressed in terms of barns. The relationship between barns and cm^2 is shown below.

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

Whether a neutron will interact with a certain volume of material depends not only on the microscopic cross section of the individual nuclei but also on the number of nuclei within that volume. Therefore, it is necessary to define another kind of cross section known as the macroscopic cross section (∑). The macroscopic cross section∑ is the probability of a given reaction occurring per unit travel of the neutron. It is related to the microscopic cross section (σ) by the relationship shown below (DOE, 1993)

$$\sum = N\sigma \qquad (2.1)$$

where:

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

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

σ = microscopic cross-section (cm^2)

2.1.2 Neutron Flux

Macroscopic cross sections for neutron reactions with materials determine the probability of one neutron undergoing a specific reaction per centimeter of travel through that material.(DOE, 1993) If one wants to determine how many reactions will actually occur, it is necessary to know how many neutrons are traveling through the material and how many centimeters they travel each second. It is convenient to consider the number of neutrons

existing in one cubic centimeter at any one instant and the total distance they travel each second while in that cubic centimeter. The number of neutrons existing in a cm^3 of material at any instant is called neutron density and is represented by the symbol n with units of neutrons/cm^3. The total distance these neutrons can travel each second will be determined by their velocity. A good way of defining neutron flux (ϕ) is to consider it to be the total path length covered by all neutrons in one cubic centimeter during one second. Mathematically, this is the equation (DOE, 1993):

$$\phi = nv \qquad (2.2)$$

ϕ= neutron flux (neutrons/cm^2-sec)

n= neutron density (neutrons/cm^3)

v = neutron velocity (cm/sec)

The term neutron flux in some applications (for example, cross section measurement) is used as parallel beams of neutrons traveling in a single direction. The *intensity* (I) of a neutron beam is the product of the neutron density times the average neutron velocity. The directional beam intensity is equal to the number of neutrons per unit area and time (neutrons/cm^2-sec) falling on a surface perpendicular to the direction of the beam. One can think of the neutron flux in a reactor as being comprised of many neutron beams traveling in various directions. Then, the neutron flux becomes the scalar sum of these directional flux intensities (added as numbers and not vectors), that is, $\Phi = I_1 + I_2 + I_3 + \ldots I_n$.

Since the atoms in a reactor do not interact preferentially with neutrons from any particular direction, all of these directional beams contribute to the total rate of reaction. In reality, at a given point within a reactor, neutrons will be traveling in all directions. (DOE, 1993)

2.1.3 Reaction Rates

If the total path length of all the neutrons in a cubic centimeter in a second is known, (neutron flux (ϕ)), and if the probability of having an interaction per centimeter path length is also known(macroscopic cross section ($\sum$)), multiply them together to get the number of interactions taking place in that cubic centimeter in one second. This value is known as the reaction rate and is denoted by the symbol R. The reaction rate can be calculated by the equation shown below (DOE, 1993):

$$R = \phi \sum \quad (2.3)$$

where R= reaction rate (reactions/sec)

ϕ is the neutron flux (neutrons/cm^2-sec)

$\sum$= macroscopic cross section (/cm)

But $\sum = N\sigma$

$$R = \phi N\sigma \quad (2.4)$$

2.2 RESEARCH REACTORS

Devices that are designed so that the fission chain reaction can proceed in a controlled manner are called nuclear reactors. In a reactor, this control is accomplished by varying the value of multiplication factor k, which can be done by the person operating the reactor. (Foster, 1977). To increase the power being produced by a reactor, the operator increases k to a value greater than unity so that the reactor becomes supercritical. When the desired power level has been reached, he returns the reactor to critical by adjusting the value of k specified power level.

To reduce power or shut the reactor down, the operator mainly reduces k, making the reactor sub critical. As a result, the power output of the system decreases. Nuclear bombs and explosives cannot be controlled in this way, and these devices are not referred to as reactors. To make a reactor critical, or otherwise to adjust the value of k, it is necessary to balance the rate at which neutrons are produced within the reactor with the rate at which they disappear. Neutrons can disappear in two ways as the result of absorption in some type of nuclear reactor, or by escaping from the surface of the reactor. (Lamarsh, 1982).

Research reactors are simpler than power reactors and need far less fuel, and far less fission products build up as the fuel is used. (West, 1997) revealed that research reactor fuel requires more highly enriched Uranium, typically up to 20% U-235, although some older ones use 93% U-235. They also have high power density in the core, which requires special design features. (Krull, 2000) added that like power reactors, the core needs cooling, and usually a moderator is required to slow down the neutrons and enhanced fission. As neutron production is their primary function, most research reactors also need a reflector to reduce neutron loss from the core. A common design is the pool type reactor where the core is a cluster of fuel elements sitting in a large pool of water. Among the fuel elements there are control rods and empty channels of experimental materials. Each element comprises several curved aluminum-clad fuel plates in a vertical box. The water moderates and also cools the reactor, and graphite or beryllium is generally used for the reflectors, although other materials may also be used. Apertures to access the neutron beams are set in the wall of the pool. Research reactors have a wide range of uses, including analysis and testing of materials and production of radioisotopes. Their compatibilities are applied in many fields,

within the nuclear industry as well as in fission research, environmental science, advanced materials developments, drug design and nuclear medicine.

2.2.1 Nigeria Research Reactor-1 (NIRR-1) Description

NIRR-1 is a Miniature Neutron Source Reactor (MNSR) designed by China Institute of Atomic Energy (CIAE) (Zhou, 1985). First criticality was achieved on 03 February 2004 and has been operated safely (Jonah *et al.,* 2005, and 2006). This reactor however, is specifically designed for use in neutron activation analysis (NAA) and limited radioisotope production. (Balogun *et al.,* 2005). The research reactor (NIRR-1) has a tank-in-pool structural configuration and a nominal thermal power rating of 31 kW. The current core of the reactor is a 230 x 230 mm square cylinder and fueled by U-Al_4 enriched to 90% in Al-alloy cladding. It has a total number of 347 fuel pins and three Al dummies in the fuel lattice. The length of the fuel element is

248 mm, the active length being 230 mm with 9 mm Al-alloy plug at each end. (Jonah *et al.,* 2006) The diameter of the fuel meat is 4.3 mm and the ^{235}U loading in each fuel element is about 2.88 grams. The cladding is Al-alloy, whose thickness is 0.6 mm.

The reactor (NIRR-1) has only one control rod serving as shim rod, regulation rod as well as safety rod. 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) NIRR-1 is operated using the control console, the microcomputer control system and the two rabbit systems all connected to a power source and during normal operation the reactor water temperature varies between 23^0C and 46^0C. However, the temperature difference rises rapidly and attains a stable value due to 'insufficient natural circulation' phenomenon.

The overall length of the control rod is 450 mm in length.

One basic feature of NIRR-1 is the inherent safety due to limited core excess reactivity With a built-in clean cold core excess reactivity of 3.77 mk measured during the on-site zero-power and criticality experiments, the reactor can operate for a maximum of 4.5 hours at full power, mainly due to the large negative temperature feedback effects. Under these conditions, with the same fuel loading, the reactor can run for over ten years with a burn-up of <1% (Jonah *et al.*, 2007) An MCNP model of the reactor fueled with HEU has been developed and benchmarked by measured data obtained during the onsite zero power and power rising experiments (SAR 2005).

The core of the NIRR-1 is located 4.7m under water close to the bottom of the light water reactor vessel. The quantity of water is 1.5m^3 in the vessel, which serves the purpose of radiation shielding, moderation and as primary heat transfer medium. The water-filled reactor vessel is in turn immersed in a water-filled pool 30m^3.The cold water is drawn through the inlet orifice as shown in the figure 2.1. The water flow past the hot fuel elements and comes out through the core outlet orifice. The hot water rises to mix with the large volume of water in the reactor vessel and to the cooling coil. Heat passes through the walls of the container to the pool water. A diagrammatic representation of the heat transfer mechanism of NIRR-1 is represented in figure 2.1

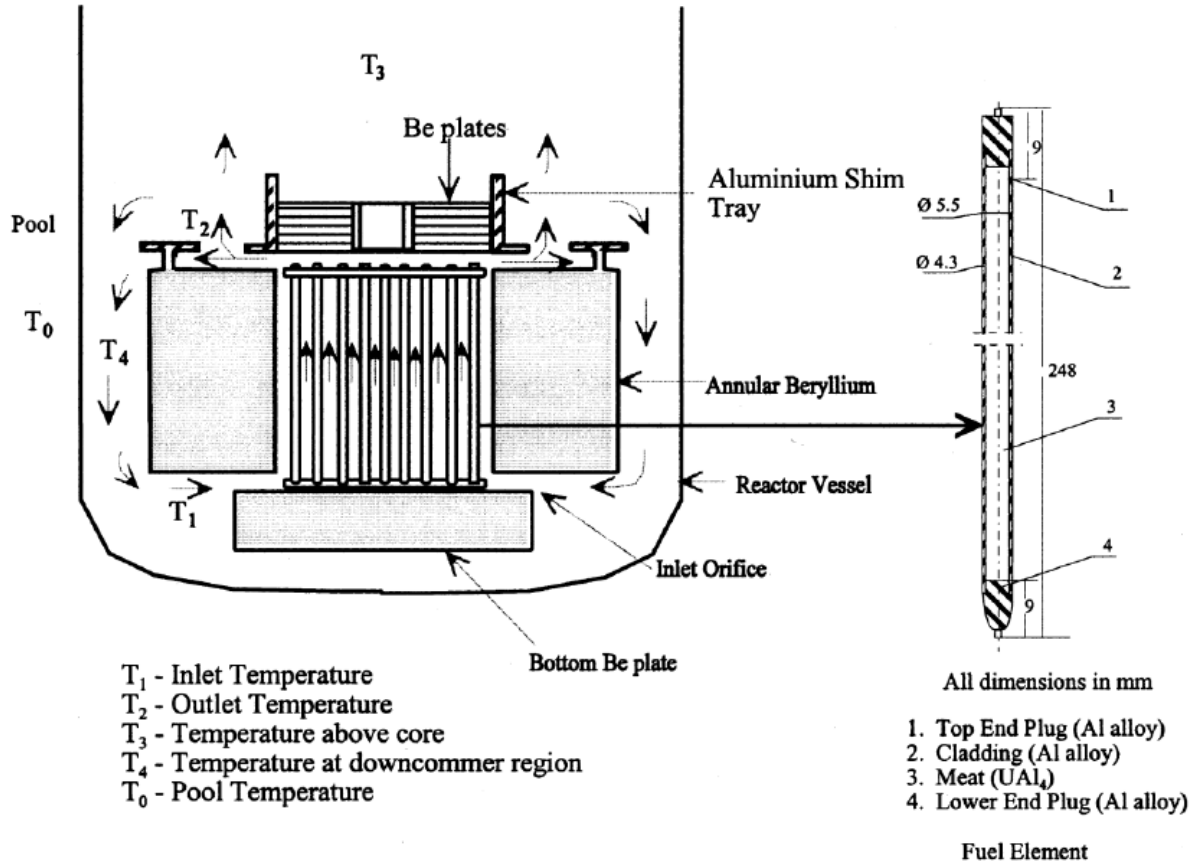


Fig 2.1: Schematic diagram of the coolant flow pattern. (SAR, 2005)

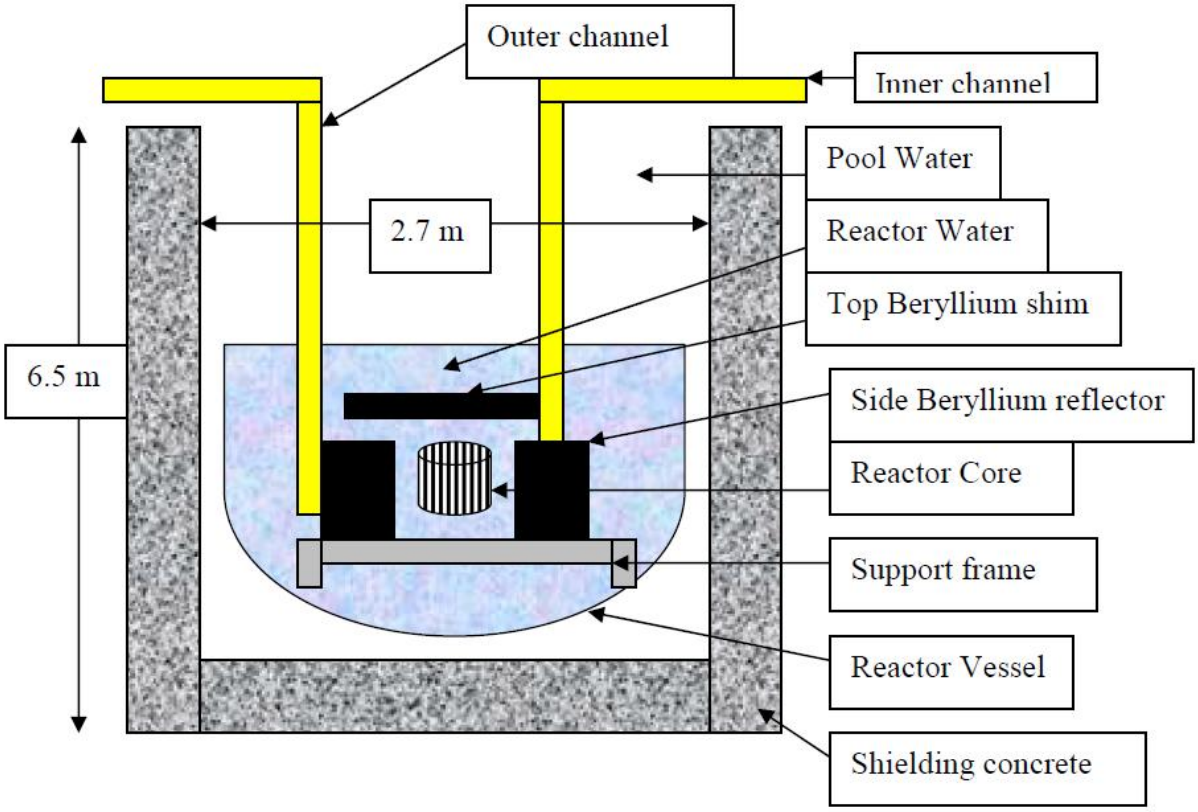


Fig2.2: Schematic diagram of MNSR (Ahmed *et al*, 2011)

2.2.2 Nigeria Research Reactor-1 (NIRR-1) Design.

The 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 commonly called tank-in-pool type (Yang, 1992). Their maximum nominal power is ~ 30 kW which is equivalent to a thermal neutron flux of $1x10^{12}$ $n.cm^{-2}.s^{-1}$ in any of the 5 inner irradiation channels (Balogun, 2005; Ahmed *et al.,* 2006). 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 2.1) with a thick beryllium reflector (~ 10 cm) surrounding the core radially (Yang, 1992). The NIRR-1 has one central control rod made of cadmium and cladded with stainless steel. The rod performs shim, safety and regulatory control functions. This makes the reactor safe and compact enabling it to be located in densely populated area without harm to the public (Yang, 1992).

Table 2.1: Technical Specifications of NRR-1 (Yang, 1992).

Parameters	Description
Reactor type	Tank-in-pool
Rated thermal power	30 kW
Fuel	UAl_4 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	$1x10^{12} n cm^{-2} s^{-1}$
Flux in outer channel	$5x10^{11} n cm^{-2} s^{-1}$
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

Table2.2: Design Parameters of NIRR-1 Fuel (Yang, 1992)

Description	Rate
Rate of thermal power	30kW
Maximum heat flux at fuel element surface	$3.56x10^3 KW/m^2$
Inlet coolant temperature	30^0C
Outlet coolant temperature	54^0C
U^{234}Content/fuel element	2.88g
Coolant Velocity	15mm/s
Enrichment factor	90.2%
Uranium Weight fraction/fuel	27.5%
Fuel density	3.36203g/cm
Volume porosity	3%
H/U ratio	197

2.2.3 Safety Features of Nigeria Research Reactor-1 (NIRR-1)

The NIRR-1 is designed to include some inherent safety measures in order to prevent radiation leakage. The objective of its design is to achieve a minute 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. This reduces the effect of flux distribution on the irradiation experiments due to movement of the central control rod (Balogun, 2004), thus, enhancing the performance of the reactor from both mechanical and computational physics points of view.

Two major provisions were made in the design to increase the safety margin and to enhance the inherent safety of the reactors as mentioned below:

(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.

Consequently, an appropriate choice of inlet flow orifice is required for the core thermal hydraulics design to yield the desired rise in temperature for ensuring an inherent safety of the reactor. The design is such that the temperature should not be too high to allow the

occurrence of boiling in the core and cause a large temperature effect that will significantly reduce the excess reactivity for the compensation of Xenon poisoning.

2.3. Reactor Power

Neutron power is essentially the fission rate multiplied by the prompt energy release per fission. We cannot measure this directly, but we can monitor the flux and the average neutron flux in the core is proportional to the overall fission rate.

Reactor thermal power (often shortened to thermal power) is the reactor's rate of heat energy production. Thermal power takes account of nuclear decay heating and of conventional heat (pump heat and heat losses to the environment).

Fission power is the name given to the heat generated because of nuclear processes in the fuel. This includes wasted heat, such as that generated in the moderator and shielding. It does not include any conventional heating. The operating license places an upper limit on fission power, which is enforced by regulating the reactor thermal power

2.3.1 Reactor Power Level

A change in reactor power level can result in a change in reactivity if the power level change results in a change in system temperature (Jacobs, 1960). The power level at which the reactor is producing enough energy to make up for the energy lost to ambient is commonly referred to as the point of adding heat. If a reactor is operating well below the point of adding heat, then variations in power level produce no measurable variations in temperature. The inherent stability and power turning ability of a negative temperature

coefficient are ineffective below the point of adding heat. If a power excursion is initiated from a very low power level, power will continue to rise unchecked until the point of adding heat is reached, and the subsequent temperature rise adds negative reactivity to slow, and turn, the rise of reactor power. In this region, reactor safety is provided by automatic reactor shutdown systems and operator action. (DOE, 1993)

2.3.2 Reactor Power Calculation

To get the total number of reactions occurring in a core per unit time, one needs to multiply the reaction rate per unit volume by the total volume of the core. If the amount of energy involved in each reaction were known, it would be possible to find the rate of energy release (power) due to a certain reaction (DOE, 1993). In a reactor where the average energy per fission is 200 MeV, it is possible to determine the number of fissions per second that are necessary to produce one Watt of power using the following conversion factors.

1 fission = 200 MeV

1 MeV = 1.602×10^{-6} ergs

1 erg. = 1×10^{-7} Watt-sec

$$1\text{Watt}\left(\frac{1erg}{1x10^{-7}Watt-sec}\right)\left(\frac{1MeV}{1.602X10^{-6}erg}\right)\left(\frac{1fission}{200MeV}\right) = 3.12x10^{10}\ \frac{fissions}{Watt-sec}$$

This is equivalent to stating that 3.12×10^{10} fission release 1Watt-second of energy.

The power released in a reactor can be calculated based on Equation (2.3). Multiplying the reaction rate by the volume of the reactor results in the total fission rate for the entire reactor. Dividing by the number of fissions per Watt-sec results in the power

released by fission in the reactor in units of watts. This relationship is shown mathematically in Equation (2.5) below (DOE, 1993):

$$P = \frac{\phi_{th}\Sigma_f \, V}{3.12\times10^{10}\frac{fission}{Watt-sec}} \tag{2.5}$$

where P= power

ϕ_{th}= thermal neutron flux ($neutrons/cm^2$-sec)

$\sum_f$= macroscopic cross section for fission (cm^{-1})

V= volume of core (cm^3)

2.3.3 Relation between Neutron Flux and Reactor Power

In an operating reactor the volume of the reactor is constant. Over a relatively short period of time (days or weeks), the number density of the fuel atoms is also relatively constant. Since the atom density and microscopic cross section are constant, the macroscopic cross section must also be constant. Examining Equation (2.5), it is obvious that if the reactor volume and macroscopic cross section are constant, then the reactor power and the neutron flux are directly proportional. This is true for day-to-day operation. The neutron flux for a given power level will increase very slowly over a period of months due to the burn-up of the fuel and resulting decrease in atom density and macroscopic cross section (DOE, 1993).

2.3.4 Reactor Power Calibration

The thermal power of a water-cooled reactor can be determined by measuring the increase in temperature of the water during a given time interval. The electrical power of a power reactor can be calculated from the thermal power via multiplication with its efficiency which is between 32% and 42% (Eugene, 2003)

Temperature fluctuations cannot be avoided at the cause of reactor operation. However, any fluctuation in temperature influences the reactivity. The mean energy of the thermal neutrons increases with increasing temperature and this has its feedback on the neutron absorption because the neutron cross sections depend on the neutron energy (nuclear temperature coefficient). Further, the mean free path and the non-leakage factor depend on the density and then on temperature (density temperature coefficient).The nuclear temperature coefficient results mainly from the energy dependence of absorption and scattering cross sections.

The most obvious need for an accurate determination of the reactor power level is that of satisfying the safety criteria established for a given system. (Bullock, 1965).The methods to be investigated in this research work are the calorimetric (slope) and thermal balance (ballistic) methods

a, Calorimetric (Slope) Method of Power Calibration

Although power monitoring of a research reactor is performed by a means of nuclear instrumentation, its calibration is done by thermal procedures (Mesquita *et al.*, 2007, 2011). The thermal power of a research reactor cooled by water can be determined by measuring the increase of water temperature over a given period of time (Bullock, 1965; Zagar *et al.*, 1999). In this method, the temperature-rise ($\Delta T/\Delta t$) is determined and the reactor power as a function of temperature-rise rate is calculated. The slope method of power calibration depends on heat transfer effects, natural convection of water, heat loss and similar effects

The basic formulation is the following (Zagar, 1999; Mesquita *et al.*, 2007, 2011):

$$Q = \frac{\Delta T}{\Delta t} K \quad (2.6)$$

Where Q is the power and K is the experimentally determined heat capacity constant given by:

$$K = \rho V_w . C_p \quad (2.7)$$

Where ρ is the water density, V_w is the water volume and c_p is the water specific heat capacity. The reactor pool heat capacity constant can be calculated as well. In the first approximation, one can assume that the reactor pool temperature is constant throughout the pool and neglect all heat losses from the pool. One can approximate the reactor pool as an insulated "point" system. So the reactor heat capacity K can be simply calculated from pool volume V_w.

The power level is usually calculated by taking the slope of the graph of temperature against time and multiplying the slope by the pool constant of the reactor.

This method is the most inherently accurate, straightforward and presumably the simplest method for calibrating reactor power (shortall, 1954, Bullock, 1965, Mesquita *et al.*, 2007). Though it has its limitation like difficulty in the determination of uncertainty associated with it can be in the range of several percent (Streit *et al.*, 2005). This method is viable for reactors having less heat loss. Bullock, (1965).

The calorimetric method uncertainty

From the calorimetric method equation given by (2.76), the power uncertainty can be calculated considering the following uncertainties of: the water density (ρ), the water

volume in the pool (V_w), the estimated water specific heat capacity (c_p) as function of temperature, the water pool temperature rise (ΔT) and the time interval (Δt).

The power uncertainty U_q is given by the following equation (Mesquita *et al*., 2011)

$$U_q^{'}=\left(\left(\frac{\partial q_{u_\rho}}{\partial \rho}\right)^2+\left(\frac{\partial q_{u_{c_\rho}}}{\partial c_\rho}\right)^2+\left(\frac{\partial q_{u_{v_w}}}{\partial V_w}\right)^2+\left(\frac{\partial q_{u_T}}{\partial T}\right)^2+\left(\frac{\partial q_{u_t}}{\partial t}\right)^2\right)^{\frac{1}{2}} \quad (2.8a)$$

where U_ρ, U_{c_ρ},U_{v_w}, U_T and U_t are the consolidated uncertainties of the independent primary variables: ρ, c_p, V_w, T and t. Solving the differential partial equation gives the following relative uncertainty expression for the value for the thermal power:

$$\frac{U_q^{'}}{q}=\left(\left(\frac{U_\rho}{\rho}\right)^2+\left(\frac{U_{C_\rho}}{C_\rho}\right)^2+\left(\frac{U_{v_w}}{V_w}\right)^2+\left(\frac{U_T}{T}\right)^2+\left(\frac{U_t}{t}\right)^2\right)^{\frac{1}{2}} \quad (2.8b)$$

b Power Calibration by Heat Balance Method

In this method, the reactor core is normally cooled by natural convection of deionized light water in the reactor pool. Pool water temperature depends on the reactor power, as well as on the external temperature, which affects heat dissipation in the cooling tower. The total power is determined by the thermal balance of cooling water passing through the core added to the calculated heat losses (Mesquita *et al.,* 2011).

The inlet and outlet temperatures can be measured by thermocouples positioned at the inlet and outlet orifices . The flow-rate can be measured by an orifice plate and a differential pressure transmitter. But in the case of NIRR-1 reactor which does not have primary and

secondary loop, the flow rate is determined via the following basic formula (Jevremovic, 2008):

$\dot{m}$= Temperature difference between inlet and outlet orifices(℃)

H = Height of the inlet orifice (mm)

T_i = Inlet Temperature in ℃

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 (6 mm) into Equation. (2.11) reduces the equation to:

$$\Delta T = 7.04 T_i^{-0.35} P^{(0.59+0.0019T_i)} \quad (2.12)$$

The reactor power is thus given by

$$P = \text{Exp}\left[Ln\left(\frac{\Delta T}{7.04T_i^{-0.35}}\right)(0.59 + 0.0019T_i)^{-1}\right] \quad (2.13)$$

From Eq. (2.13), for a fixed height of inlet orifice it is expected that the reactor power varies linearly with coolant temperature rise. Ahmed *et al*, 2008 performed measurements to verify the theoretical predictions of reactor power and flux parameters that result from changes in core inlet temperature (T_i) and the temperature difference between the coolant inlet and outlet (ΔT) in the Nigeria Research Reactor-1 (NIRR-1), which is a miniature neutron source reactor (MNSR). The data they obtained confirmed that there is a strong dependence of the reactor power on coolant temperature in agreement with the design of MNSR. They reported that the experimental parameters were found to be in good agreement with data obtained using a semi-empirical relationship between the reactor power, flux parameters, core inlet temperature, and the coolant temperature rise. The

relationship was therefore used to predict the power level of NIRR-1 from its neutron flux parameters to which it has been found to be proportional.

Apart from the above method of determining the reactor power via the thermal hydraulic parameters, the neutronic parameters(neutron flux values) could as well be exploited 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:

$$P = 3.1x10^{-10}\sum_f V_f \phi \quad (2.14)$$

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 cylindrical core with a cross-sectional area of $23cm^2$ and highly enriched uranium (U-235) was used as fuel for the Nigeria MNSR. The above parameters make it possible to reduce Eq. (2.14) to only a flux dependent reactor power equation as shown in Eq. (2.15).

$$P = 3.0x10^{-8}\phi \quad (2.15)$$

Eq. (2.15) reveals a linear relationship between the reactor power and its neutron flux. It shows that the reactor power could also be predicted from the experimental neutron flux values.

At a flux of$5x10^{11}cm^2s$ the results show a fairly constant reactor power with coolant temperature rise, T, It can also be observed that the reactor power oscillates near its reported full thermal power of 31kW at a neutron flux of $1x10^{12}cm^2s$.

2.5 TRANSIENT CHARACTERISTICS OF NIRR-1

According to the design requirements, the NIRR-1 is a reactor with natural circulation cooling, self-stable by negative reactivity feedback. Therefore, the demonstration of its safety features during the positive reactivity insertion transient is very important. Theoretically, the kinetic transient is a physics thermo hydraulics-coupled problem, besides the neutronics response. That the moderator mass, momentum and energy conservation and various hydraulic flow resistances should be taken into account.

The NIRR-1 is cooled by natural circulation (i.e. thermal siphon) to remove the fission energy in the core, the cool water is sucked up through the inlet orifice formed between the side Be annulus and the bottom Be plate and flowing upward to wash the fuel elements. After being heated, the water flows out of the core through the outlet orifice formed between the top Be shim tray and the side Be annulus. The heat that the warm water carries then is conducted to the pool water through the wall of upper and lower sections of the reactor vessels. Finally, the heat is dispersed into the environment

2.6 REACTIVITY COEFFICIENTS

Changes in the physical properties of the materials in the reactor will result in changes in the reactivity. Reactivity coefficients are useful in quantifying the reactivity change that

will occur due to the change in a physical property such as the temperature of the moderator or fuel.

2.6.1 Moderator Temperature Coefficient

The temperature coefficient of reactivity is a very important safety parameter of research reactors. Many parameters that determine the multiplication factor depend on temperature. As a result, a change in temperature leads to a change in multiplication factor (k) and alters reactivity of the system (Lamarsh, 1982) A negative temperature coefficient of reactivity is desirable since it tends to counteract the effects of transient temperature changes during reactor operation. In MNSR reactors such as the NIRR-1 the moderator is the hydrogen that is mixed with the fuel itself. If the fuel temperature increases when the control rods are suddenly removed, the neutrons inside the hydrogen-containing fuel rod become warmer than the neutrons outside in the cold water. These warmer neutrons inside the fuel cause less fissioning in the fuel and escape into the surrounding water. The end result is that the reactor automatically reduces the power within a few thousandths of a second, faster than any engineered device can operate. The inherent safety of the NIRR-1 reactor arises from the prompt negative temperature reactivity coefficient, whose measured value effectively limits the power when excess reactivity is suddenly inserted. The prompt temperature coefficient refers only to fuel temperature, and the overall temperature coefficient of the reactor refers to the change in the total core temperature.

Prompt and inherent reactivity feedbacks are considered important factors for research reactor safety (Akaho *et al.,* 2002). For the MNSR such as the NIRR-1 reactor, inherent safety features are provided in their design since they are built with a negative Reactivity

Temperature Coefficient (RTC). The mechanisms that affect reactivity in NIRR-1 reactor are the fuel and moderator expansion, void production among others. The reactivity due to temperature feedback$\rho(t)$ is expressed as (Akaho *et al.*, 2002)

$$\rho(t) = \rho_o(\mathrm{t}) + \alpha_M \Delta T_M(\mathrm{t}) + \alpha_F \Delta T_F(\mathrm{t}) \tag{2.16}$$

where $\rho_o(\mathrm{t})$ is the initial reactivity, α_M and α_F are the moderator and fuel temperature coefficients of reactivity respectively. ΔT_M and ΔT_F and the deviation spatially averaged moderator and fuel temperatures from the equilibrium temperatures respectively. For MNSR such as NIRR-1 reactor, the core temperature of reactivity α_T is given by the addition of moderator and fuel temperature coefficient (Mirza, 1996, Akaho *et al.*, 2003)

$$\alpha_T = \alpha_F + \alpha_M \tag{2.17}$$

The extent to which the reactivity is affected is affected by changes in temperature is described in terms of the temperature coefficient of reactivity, denoted as α_T.It is defined as the change in reactivity with respect to temperature and is expressed as (Erradi *et al.*, 2003)

$$\alpha_T = \frac{d\rho}{dT} \tag{2.18}$$

where, reactivity which is the fractional departure of a system from criticality is expressed as (Lamarsh, 1982):

$$\rho = \frac{k-1}{k} \tag{2.19}$$

where, k is the multiplication number.

Differentiating equation (2.19) gives

$$\alpha_T = \frac{1}{k^2}\frac{dk}{dT} \tag{2.20}$$

In all cases of interest k is close to unity and so equation (2.20) can be written approximately as

$$\alpha_T = \frac{1}{k}\frac{dk}{dT} \qquad (2.21)$$

Equation (2.19) is more convenient than equation (2.18) for calculation purpose and is often taken as the definition of α_T. According to equation (2.21), α_T is equal to the fractional change in k per change in temperature and has units of (degrees)$^{-1}$

The response of reactor to a change in temperature depends on the algebraic sign of coefficient of reactivity á$_T$.

Consider a situation when α_T is positive. Since k is always positive, $\frac{dk}{dT}$ is also positive, which means that an increase in T leads to an increase in k and so on. Now, suppose for some reason the temperature of the reactor increases. This increase the value of k, which in turn leads to an increase in the power level of the reactor giving rise to a further increase in the temperature, another increase in k and so on. Thus, with α_T positive, an increase in temperature leads to ever increase in temperature and power until the reactor either shutdown by outside intervention or it melts down.

However, suppose the reactor temperature decreased, if α_T is positive a decrease in T leads to a decrease in k. This reduces the temperature, giving rise to a further decrease in k, and so on, until the reactor eventually shutdown. Thus, if $\alpha_T > 0$, an increase in T leads to meltdown, a decrease in T to shutdown-in the absence of external intervention.

The situation is quite different when α_T is negative. In this case, $\frac{dk}{dT}$ is negative, and an increase in T gives a decrease in k. Now an increase in reactor temperature leads to decrease in reactor power, which tends to decrease the temperature and return the reactor to its original state. Furthermore, a decrease in T results in an increase in k, so that if T goes down the power goes up and again the reactor tends to return to its original state. Clearly, a reactor with positive α_T is inherently unstable to changes in temperature, whereas a reactor having a negative α_T is inherently stable. (Lamarsh, 1982)

It should be noted that the temperature ordinarily does not change uniformly throughout a reactor. For instance, an increase in reactor power is reflected first by a rise in the temperature of the fuel since this is the region where power is generated. In a thermal reactor, the coolant and moderator temperatures do not increase until heat has been transferred from the fuel to these regions. It is important in discussing temperature coefficient to specify the component whose temperature is undergoing change. Thus, the fuel temperature coefficient is defined as the fractional change in k per unit change in fuel temperature. The moderator temperature coefficient is the fractional change in k per unit change in moderator temperature. (Larmash, 1982)

Because the temperature of the fuel reacts instantly to changes in reactor power, the fuel temperature coefficient is also called prompt temperature coefficient, denotes as α_{prompt} (lamarsh, 1982) The value of α_{prompt} determines the first response of a reactor to changes in either fuel temperature or reactor power. For this reason, α_{prompt} is the most important temperature coefficient insofar as reactor safety is concerned. In practice, control

of a MNSR reactor such as the NIRR-1 reactor is enhanced by several negative feedback mechanisms relying on physical phenomena

2.6.2 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 (DOE, 1993). 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.

2.7 REACTOR DYNAMICS.

When power changes in a reactor are large enough to influence the value of the reactivity, the transient behavior of the reactor is termed as the reactor dynamics. As can be expected, the influence of power on the reactivity has to be quantified in order to properly describe the dynamic behavior of the reactor. (Alhassan, 2009)

During operation of a reactor the energy released due to nuclear fission is deposited to the coolant. The resulting temperature distribution in the fuel and coolant (in BWRs even the void fraction distribution) is a subject of the thermal-hydraulic analysis of the reactor. The

temperature distribution, which in a general case is a function of both the time and the location, is influencing the values of microscopic cross-sections for various nuclear reactions caused by neutrons. As a result the reactivity will depend on the temperature changes.

From a practical point of view it is important to know what the influence of temperature on reactivity is and how it will influence the operation of the nuclear reactor. In general the following two cases can be considered:

- Reactivity increases with the temperature: in this case the increasing reactivity will cause the increase of the reactor power, which, in turn, will cause the increase of temperature, etc. That means in this case the reactor will be inherently unstable.
- Reactivity decreases with temperature: in this case the decreasing reactivity will cause the decrease of the reactor power which will be followed by the decrease of the temperature, and so on. Clearly the reactor will be inherently stable in such a case.

The conclusion is that reactors should be constructed in such a way which assures the decreasing reactivity in function of temperature. This can be achieved by using proper materials and a proper nuclear reactor configuration. The reactivity changes with temperature because the reactivity depends on macroscopic cross sections, which themselves involve the atomic number densities of materials in the core, (Alhassan, 2009)

2.8 DYNAMIC FEEDBACK CHARACTERISTICS OF NIRR-1

NIRR-1 has the following characteristic. If the output power increases, the reactivity becomes negative and the reactor tends to decrease the output power. If the output power decreases, the reactivity becomes positive and the reactor tends to increase the output power. In other words, the output of the NIRR-1 reactor exhibits negative feedback. The magnitude of the feedback changes over a wide range by the design of the reactor. The reactor fuel contains fissile material and fertile material. The fertile material has large resonance absorption. Because of this large absorption, the neutron spectrum (neutron flux) has a sharp dip at this energy. Therefore, the number of absorbed neutrons decreases compared to a flat neutron spectrum. This is called self-shielding. If the output power and temperature increase, the thermal motion of fuel will intensify. Thus, when a neutron in motion with a constant energy *E* collides with a fuel nucleus, the energy should be modified by the thermal motion of the nucleus. This means that the resonance of the cross-section widens, while the area stays constant. Thus, the peak of the cross-section decreases, the self-shielding effect is reduced, and absorption increases. This is a negative feedback for U-238 capture resonance

In practice, water-moderated reactors such as the NIRR-1 are designed with a moderator-to-fuel ratio so that the reactor is operated in an under moderated condition. The reason that some reactors are designed to be under moderated is if the reactor were over moderated, an increase in temperature would decrease the moderator to fuel ratio (N^m/N^u) due to the expansion of the water as its density became lower. This decrease in N^m/N^u would be a positive reactivity addition, increasing effective multiplication factor (k_{eff}) and further raising power and temperature in a dangerous cycle. If the reactor is under moderated, the

same increase in temperature results in the addition of negative reactivity, and the reactor becomes more self-regulating.

2.9 Heat Transfer Phenomena in NIRR-1

As the reactor is operated, energy is released through the fission process. The majority of this energy appears as energy carried by fission fragments, gamma rays, neutrons and beta particles emitted (Jevremovic, 2008). When these particles interact with the surrounding materials, heat is produced. This process heats up the fuel meat and starts the chain of heat transfer

The heat released from nuclear fuel is transferred to the coolant through the heat conduction of fuel and cladding, and the heat convection between cladding outer surface coolant. Then, the heat is carried out of the reactor core by the coolant.

While there has been a great deal of research on the behavior of NIRR-1 power resulting from changes in core-coolant temperature (Ahmed *et a*l., 2008), this study focuses on the heat transfer in the system as a whole. The fuel rods in the NIRR-1 core heat the surrounding water which flows through the fuel channels then up and over the rods in a natural convection manner. The heated water then can either evaporate from the open top of the reactor pool or the heat is transferred to the environment. Transferring the heat to the environment can be either through the top surface which is open to the air or through the aluminum tank wall, concrete barrier and outer steel tank wall. Currently, through one of these two methods is the only way the pool water is cooled back to ambient temperature.

2.9.1 Energy Equations

In order to determine the heat transfer into the reactor pool from the operating core, the reactor is first considered to be a closed system with a control volume surrounding the core and water. The energy equation is then:

$$E_{in} - E_{out} = \Delta E_{System} \quad (2.22)$$

Where E_{in} is energy into the system (J)

E_{out} is energy out of the system (J)

ΔE_{System} is the energy balance of the system (J)

This expands to:

$$Q - W = \Delta U + \Delta KE + \Delta PE \quad (2.23)$$

As work added into the system (W), kinetic energy (KE) and potential energy (PE) are not applicable in this instance only heat transfer (Q) and internal energy (U), which has been expanded remain, the equation then becomes

$$Q_{in} = (m_t C_P \Delta T) + m_e(\Delta H_V + C_P \Delta T). \quad (2.24)$$

where Q_{in} is heat (J)

m_t is the mass of water in the reactor tank (kg), assumed to be full

C_p is the specific heat of water (4,183.2 J/kg K, at 20°C) ΔT is the temperature difference from the starting and ending conditions, in °C.

M_e is mass of water evaporated in (kg)

ΔHV is latent heat of vaporization for water at 20°C (2,453.5 x103 J/kg)

Using this equation it is possible to estimate the overall average temperature increase of the reactor tank water.

2.9.2 Heat losses from the reactor pool to the environment.

The core of the Nigeria Research Reactor one (NIRR-1) is placed below the room floor in the bottom of a cylindrical pool. The pool is designed in accordance with industrial building standards. Its inside diameter is 2.7m, depth below ground is about 6.5 m and the wall thickness is 0.4 m. The pool is made of reinforced concrete and lined with stainless steel whose cylindrical part is 3 mm thick and base 5 mm. The concrete wall of the pool is 400 mm thick, while the bottom lining rests on 500 mm concrete. The reactor pool transfers heat to the environment by conduction, convection and evaporation.

(a) Heat loss by conduction

The heat generated in the fuel of NIRR-1 is transferred to the water through natural convection, from the water in the reactor core to the reinforced concrete lined with stainless steel by conduction and is finally cooled by the ambient air through convection. To gain a complete understanding of the heat loss from (NIRR-1) by conduction, it was necessary to investigate the conduction process and gauge its significance. In order to model the heat from the core to outside surface, a thermal circuit of the reactor was created following the convections set forth by (Incropera *et al.*, 2007)

Firstly, Fourier law of heat transfer is expressed in cylindrical form as:

$$q_r = \mathrm{k}(2\eth rL)\frac{dT}{dr} \tag{2.25}$$

where q_r is heat rate (W)

k is thermal conductivity (W/m-K)

r is radial distance (m)

L is cylindrical length (m)

T is temperature (K)

Applying the general solution to equation (2.25) and using the temperatures of the inner and outer surface as boundary conditions creates an expression for the heat transfer rate (H_t).

$$H_t = \frac{2\eth Lk(T_{s,1} - T_{s,2})}{ln\frac{r_2}{r_1}} \quad (2.26)$$

In order to calculate the total thermal resistance using the thermal circuit model, the material properties and dimensions were separated out of equation (2.26). Equation (2.27) is the cylindrical thermal resistance for conduction:

$$R_{cond} = \frac{In\frac{r_2}{r_1}}{2\eth Lk} \quad (2.27)$$

where R is thermal resistance (K/W)

The insulating materials in NIRR-1 are arrayed in a serial configuration so the thermal circuit takes the form as shown in Figure 2.3 with Equation (2.28). In this form the heat transfer to the outside surface of the steel tank is given by knowing the temperature of the core (in contact with the water) and temperature of the outer stainless steel tank.

$$q_r = \frac{T_{core} - T_{ss}}{\left(\frac{1}{2\eth r_{core} L_{core} h}\right) + 2\left(\frac{In\frac{r_{ss,2}}{r_{ss,1}}}{2\eth k_{ss} L_{ss}}\right) + \left(\frac{In\frac{r_{conc,2}}{r_{conc,1}}}{2\eth k_{conc} L_{conc}}\right)} \quad (2.28)$$

where

T_{core} = Temperature of the core (in contact with H_2O)

T_{ss}= Temperature of the outer stainless steel

r_{core} = Core radius

L_{core} = Core height

h = Heat transfer coefficient of water

$r_{conc,1}$ = Inner radius of concrete layer

$r_{conc,2}$ = Outer radius of concrete layer

k_{conc} = Thermal conductivity of concrete

L_{conc} = Height of concrete layer

$r_{SS,1}$ = Inner radius of stainless steel tank

$r_{SS,2}$ = Outer radius of stainless steel tank

k_{SS} = Thermal conductivity of stainless steel

L_{SS} = Height of stainless steel tank

In order to use Eq (2.28) and find the heat loss through the tank walls the convective heat transfer coefficient (h) of the tank water was first calculated. The first step in this process is the calculation of the dimensionless Grashof number, representing a measure of the ratio of the buoyancy forces to the viscous forces acting on the fluid (Incropera *et al*, 2007)

$$Gr = \frac{g\beta(T_{core}-T_o)L_{core}^3}{v^2} \qquad (2.29)$$

where Gr is Grashof number, g is gravitational acceleration ($9.8 m/s^2$), β is volumetric thermal expansion coefficient of water (207.71×10^{-6} K^{-1}, at 20°C) (Cengel and Boles, 2008), T_o is the bulk pool temperature, V is the kinetic viscosity of water ($1.0058 \times 10^{-6} m^2/s$, at 20^0C) (Cengel and Boles, 2008), L is the characteristic length of the heat transfer surface, equivalent to 0.9 times the diameter of the pool or (2.43m).

This result from (2.29) was multiplied by the Prandtl number which is 7 for water at 20°C to obtain the Rayleigh number, a measure of the magnitude of buoyancy and viscous forces in the water.

The heated, upward-facing, flat plate correlation was used to model the reactor core. This correlation calculated the ratio of conductive to convective heat transfer known as the Nusselt number (Nu). Based on the calculated Rayleigh number and the chosen correlation the following relation was used (Incropera *et al,* 2007):

$$Nu=0.15Ra_l^{\frac{1}{3}} \tag{2.30}$$

The heat transfer coefficient (h) was then calculated through their relationship derived from Newton's law of cooling (Zagar *et al.,* 1999, Incropera *et al,* 2007,)

$$\text{Nu} = \frac{hL}{k} \tag{2.31}$$

where k is thermal conductivity of water at 20^0C = $5.984\text{x}10^{-3}$W/m-K (Cengel and Boles, 2008)

(b) Heat loss by Evaporation

The heat losses due to the evaporation in the upper surface of the reactor pool were calculated by the following equation (Holman, 2002):

$$q_e = \dot{u}ë \tag{2.32}$$

where ë is the difference between the specific enthalpy of saturated water and the specific enthalpy of saturated steam at the wet-bulb temperature of the air in the reactor room, and $\dot{u}$ is the rate of mass transfer from the pool to the air , given by the equation:

$$\dot{u} = h_d A\ ñ_{air}(C_{sat} - C_\infty) \tag{2.33}$$

where A is the area of the upper surface of the reactor pool, $ñ_{air}$ is the air density, C_{sat} is the vapor concentration at saturation conditions for the air at the reactor room temperature, C_∞ is the vapor concentration in the air in the reactor room and h_d is the mass−transfer coefficient given by the following equation:

$$h_d = \frac{h_c}{ñ_{air} C_{ñ_{air}}} \left(\frac{P_r}{Sc}\right)^{2/3} \tag{2.34}$$

where P_r is the Prandtl number (0.713 for the air at 20 ºC), *Sc* is the Schmidt number (0.612 for water vapor diffusing in the air at 20 ºC), $C ñ_{air}$ is the heat capacity of the air, h_c is the convection heat transfer coefficient, obtained in terms Nusselt number Nu (equation 2.31).

where *k* is the thermal conductivity in the air (0.0257W/m-K) (Cengel and Boles, 2008), the Nusselt number was obtained from:

$$\text{Nu} = 0.14(Gr\ x\ Pr)^{1/3} \tag{2.35}$$

Gr is the Grashof number which is expressed as:

$$Gr = \frac{g\beta(T_{sur}-T_{\infty})L^3}{v^2} \quad (2.36)$$

g is gravitational acceleration (9.8m/s^2), β is volumetric thermal expansion coefficient of air,

V= kinetic viscosity of air (15.11X$10^{-6}$$m^2$/s at 20^0C) (Cengel and Boles, 2008), T_{sur} is the water pool temperature as the surface, T_∞ is the air temperature in the reactor room.

(c) Heat loss by Convection.

When the air temperature T_{air} is higher than bulk pool temperature T_o then thermal conditions above pool are stable and there is no convection. In this case the heat transfer from air to water occurs only by convection through air, which can be neglected. But in the opposite case the heat transfer occurs from the pool to the open air by natural convection. The rate of heat convection from the pool surface to the air was estimated according to Newton's law (Incropera, *et al*, 2007):

$$q'' = h.(T_o - T_{air}) \quad (2.37)$$

The average convection heat transfer coefficient h, can be expressed in terms of Nusselt number (equation 2.31)

k is the thermal conductivity of air (0.0257W/m-K) (Cengel and Boles, 2008)

The Nusselt number for horizontal surface is:

$$Nu=0.15Ra_l^{\frac{1}{3}} \quad (2.38)$$

Rayleigh number is product of Grashof number and Prandtl number and can be expressed as

$$R_{a_l} = \frac{g\beta(T_o - T_{air})L^3}{á v} \quad (2.39)$$

where g is the acceleration due to gravity (9.8m/s), β is the volumetric thermal expansion coefficient of the air ($3.4x10^{-3}K^{-1}$ at 20^0C) (Cengel and Boles, 2008). T_o is bulk pool temperature, T_{air} is the air temperature in the reactor room and v is the kinematic viscosity of the air ($15.11X10^{-6}m^2/s$) and á is air thermal diffusivity ($1.9x10^{-5}m^2/s$) (Cengel and Boles, 2008)

Table 2.3: Variables Affecting NIRR-1 Heat Conduction (SAR, 2005)

Variables (Units)	Description	Value
r_{core}	Core radius	0.12m
L_{core}	Core height	0.23m
h	Heat transfer coefficient of water	0.55W/m^2-K
$r_{conc,1}$	Inner radius of concrete layer	0.4m
$r_{conc,2}$	Outer radius of concrete layer	0.5m
k_{conc}	Thermal conductivity of concrete	0.8m
L_{conc}	Height of concrete layer	6.5m
$r_{SS,1}$	Inner radius of SS tank	3mm
$r_{SS,2}$	Outer radius of SS tank	5mm
k_{SS}	Thermal conductivity of SS	14.9 W/m-K
L_{SS}	Height of SS tank	6.5m

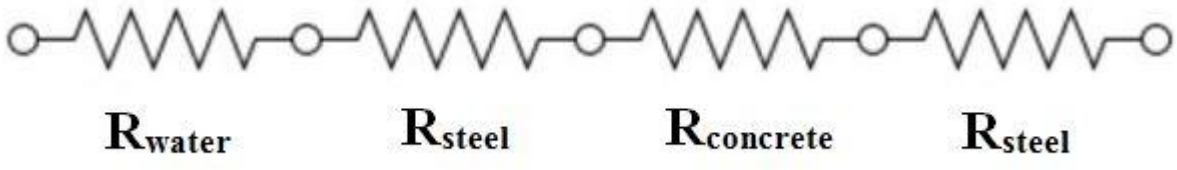


Fig 2.3: Thermal circuit diagram for NIRR-1.

CHAPTER THREE

METHODOLOGY

3.1 Materials and Methods.

3.1.1 Calorimetric (slope) Method of Reactor Thermal Power Calibration

Although power monitoring of research reactor is performed by means of nuclear instrumentation, its calibration is done by thermal process. The thermal power of a research reactor cooled by water can be determined by measuring the increase of water temperature over a given period of time.

The following materials were used to perform the experiments:

a. Temperature detectors: to measure the pool and reactor water temperature.

b. Stop watch: to record the time.

c. Recording data sheet

3.1.2 Calorimetric Method Calibration Procedure at half power

- It was ensured that all other sources of heat addition to the pool water, specifically the pool lights, were turned off.
- Bulk pool temperature was measured and recorded.
- Temperature of air in the reactor room was measured and reactor
- The reactor power was increased to 15kW.
- Power level was obtained using the neutron flux.
- The power level was stabilized at 15 kW.
- High precision temperature detector (thermocouple) was employed to measure pool temperature at 12 minutes intervals for about 6h 20min.

- Pool temperature increase on temperature detector was recorded during 6h 20min.
- Temperature versus time was recorded for every 12 minutes
- Recording of the temperature was stopped at the end of 6:20h.
- The heat losses were calculated

The procedure in 3.1.2 was repeated for experiment performed at 30kW. Though for experiment performed at this power level, the reactor operated for 3h 50min.

DATA EVALUATION (SLOPE)

a, The power level was calculated by slope method from the temperature changes.

b, Excel was used to plot the graph (linear least- squares fit data was used)

The basic formulation was the following:

$$Q = \frac{\Delta T}{\Delta t} K$$

where

Q is the thermal power, $\frac{\Delta T}{\Delta t}$ is taken from the calibration curve after fitting the results

K is the experimentally determined heat capacity constant given by:

$$K = \rho V_w C_p$$

where ρ is the water density, V_w is the water volume and C_p is the water specific heat capacity

3.2 Heat Balance (Ballistic) Method of Power Calibration

The heat balance (ballistic) methodology for the thermal power calibration consists of the measurement of the power dissipated in the core added to the calculated heat losses from the reactor pool.

The following materials were used to perform the experiments:

a. Temperature detectors: to measure the inlet and outlet temperatures of the water
b. Stop watch: to record the time

3.2.1 Calibration procedure at half power (15kW)

- **T**he reactor was operated at a preset flux of $5x10^{11}n/cm^2s$ (this is to make the reactor to operate at a power of half of its expected installed capacity) for 1h (for the purpose of short irradiation) with the cooling system off. Afterwards the cooling system was switched on and the reactor operated for 2h.
- During the 5h period, the inlet and outlet temperatures as well as the temperature difference were recorded for every 20 minutes.
- The flow-rate corresponding to each of the interval was obtained during the 5h period.

3.2.2 Calibration procedure at full power (30kW)

- The reactor neutron flux was preset at its licensed full power value of $1x10^{12}ncm^{-2}s^{-1}$ and was operated for 2h (for the purpose of short irradiation) with the cooling system off. Afterwards the cooling system was switched on and the reactor operator for over 3h.
- During the 3h period, the inlet and outlet temperatures as well as the temperature difference were recorded for every 5 minutes.

- The flow-rate corresponding to each of the interval was obtained during the 3h period.

- **DATA EVALUATION (HEAT BALACE)**

The heat generation rate in a nuclear reactor core is directly proportional to the fission rate of the fuel and the thermal neutron flux present. On a straight thermodynamic basis, this same heat generation rate is also related to the coolant temperature difference across the core and the flow-rate of the coolant passing through the core. The equation relating these parameters is given by (Mesquita *et al.*, 2007, 2011):

$$Q = C_p \dot{m} \Delta T$$

where Q = Thermal power (heat generation rate), $\dot{m}$ is the flow-rate of the coolant passing through the core, ΔT also expressed as $(T_{out} - T_{in})$ is the coolant temperature difference, T_{out} is the outlet temperature, T_{in} is the inlet temperature and C_p is the specific heat capacity of the coolant passing through the core.

The flow-rate can be measured by an orifice plate and a differential pressure transmitter. But in the case of Nigeria Research Reactor-1 (NIRR-1) which does not have an installed device for measuring the flow rate, the flow rate can be determined via the following basic formula (Jevremovic, 2008):

$$\dot{m} = \frac{ñv}{\Delta t}$$

where ñ is the density of the coolant, v is the volume of the coolant passing through the core and Δt is change in time. It can also be determined indirectly from the thermal balance along the core using measurements of the water inlet and outlet temperatures (Mesquita and

Rezende, 2005). The equation that relates the coolant temperature and density of NIRR-1 is given by ((Mansir *et al.*, 2012) :

$$y = 2E\text{-}05X^3 - 0.006X^2 + 0.0233X + 999.97$$

where y = density, X = temperature

3.3 Instrumentation

A thermocouple was positioned inside the pool to measure the water pool temperature. High precision temperature detectors (thermocouples) were used to measure the inlet and outlet temperature. One of them was positioned at the outside of the side beryllium annulus near the core inlet orifice to measure the inlet temperature. The other was positioned at the upper part of the side beryllium annulus near the core outlet orifice to measure the outlet temperature. The combination of these two pairs of thermocouples monitored the temperature difference of the reactor coolant. The locations of the thermocouples are shown in Fig 3.1. A thermocouple was positioned just above the pool surface to measure the air temperature in the reactor room.

A miniature fission chambers made using stainless steel walls and electrodes, with the operating voltage that varies from about 50 to 300 V was employed as neutron flux detector. Walls of the chambers are lined with highly enriched uranium to enhance the ionization current with argon as a common choice for the chamber fill gas used at a pressure of several atmospheres. There are two small vertical holes each 10 mm in diameter and 190 mm deep on the side annular beryllium reflector on the same circle as the inner irradiation sites (i.e. on a circle of radius 165 mm and at angles of 144^0 to each other). A current-type miniature fission chamber (LB 1120) is in each hole to monitor neutron flux at

each irradiation site and provide control signals. The locations of the neutron flux detectors are shown in Fig. 3.1. The sensitivity, linearity, follow-up features and lifetime of the selected miniature fission chambers are adequate to meet the requirements of NIRR-1 control.

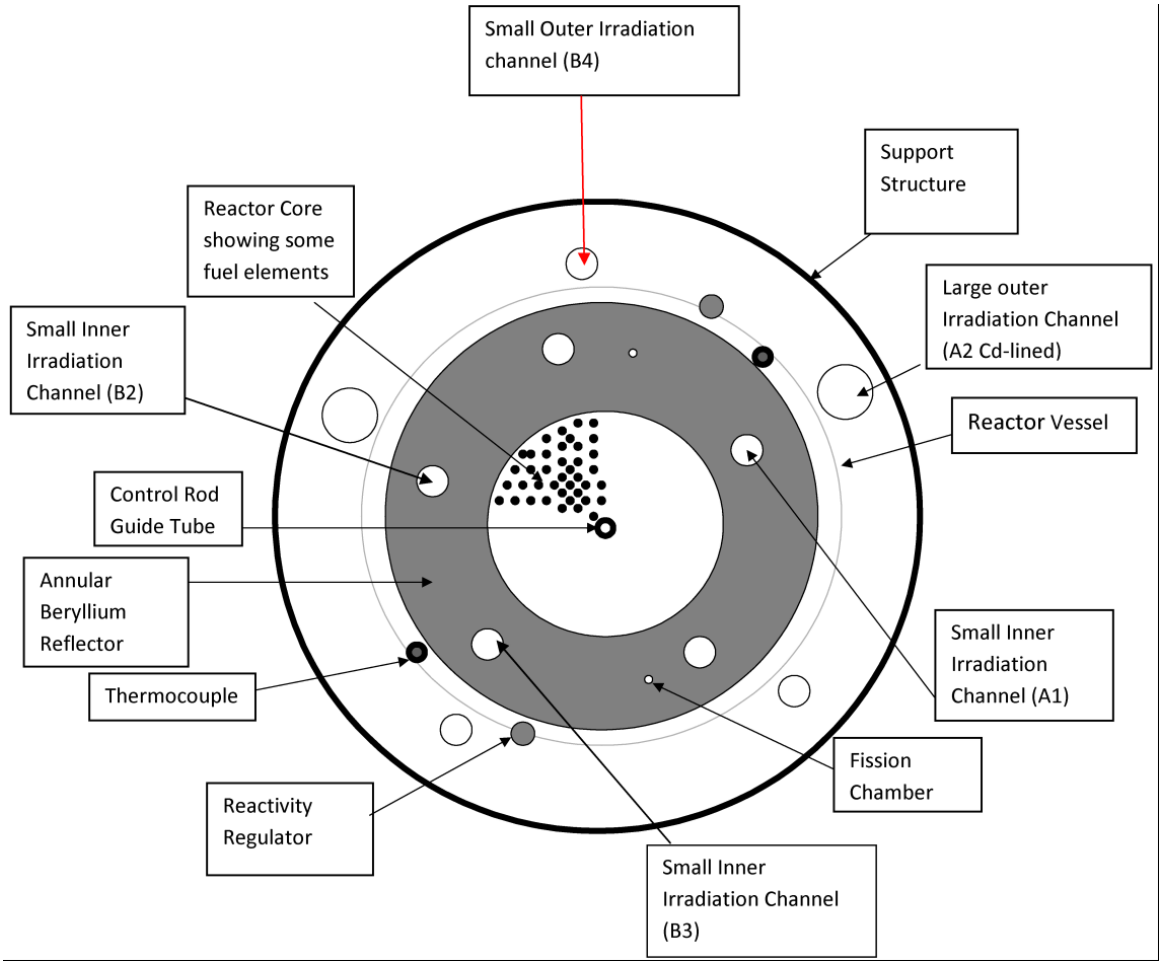


Figure 3.1: A layout core configuration showing the various components.

CHAPTER FOUR

RESULTS

The data recorded during the experiments for heat balance method of NIRR-1 power calibration at half and full power levels are presented in tables 4.1 and 4.3 respectively. The reactor parameters calculated using data obtained from table 4.1 are presented in table 4.2. The reactor parameters calculated using data obtained from table 4.3 are presented in table 4.4.

The experimental data recorded during the calorimetric method of NIRR-1 power calibration at half and full power levels are presented in table 4.5 and 4.7. The reactor parameters evaluated using the data obtained from table 4.5 are presented in table 4.6. The reactor parameters calculated using data obtained from table 4.7 are presented in table 4.8.

Table 4.1: Data obtained for calibration at half power (15kW) by heat balance method.

Time (h)	T_{in} (℃)	T_{out} (℃)	ΔT (℃)	ñ(kg/m^3)	$\dot{m}$ (kg/s)
11:00	32.9	45.8	12.9	999.32	0.277
11:20	33.0	46.0	12.8	999.33	0.279
11:40	33.9	46.3	13.0	999.30	0.275
12:00	34.0	46.3	12.3	999.39	0.290
12:20	34.2	46.4	12.2	999.40	0.293
14	34.3	46.6	12.3	999.39	0.290
13:00	34.4	46.9	12.5	999.36	0.286
13:20	34.7	46.8	12.1	999.41	0.295
13:40	34.8	47.3	12.5	999.36	0.286
14:00	34.8	46.9	12.1	999.41	0.295
14:20	34.9	47.3	12.4	999.37	0.288
14:40	35.4	47.3	11.9	999.43	0.300
15:00	35.5	47.6	12.1	999.41	0.295
15:20	35.5	47.6	12.1	999.42	0.295
15:40	35.6	47.8	12.2	999.40	0.293
16:00	35.5	47.8	12.3	999.39	0.290

Table 4.2: Calculated parameters from data obtained from table 4.1.

Parameter	Value
Average coolant flow rate	(0.2892 ± 0.0071)kg/s
Average inlet temperature	$(34.59 \pm 0.85)^0$C
Average outlet temperature	$(46.92 \pm 0.64)^0$C
Average ΔT	$(12.36 \pm 0.31)^0$C
Power dissipated	14.9kW
Total heat losses from the reactor pool	0.26kW
Total reactor Power	15.2kW
Standard deviation of the readings (S_q)	$\pm$0.01kW
Uncertainty	1.2kW($\pm 7.9^o/_o$)

Table 4.3: Data obtained for calibration at full power (30kW) by heat balance method

Time (hr)	T_{in}(℃)	T_{out}(℃)	ΔT (℃)	ñ(kg/m^3)	$\dot{m}$ (kg/s)
10:10	35.4	53.9	18.5	998.47	0.389
10:30	35.4	55.0	19.6	999.27	0.367
10:50	36.1	55.0	18.9	998.40	0.380
11:10	36.7	55.4	18.7	998.44	0.384
11:30	36.6	56.0	19.4	998.31	0.371
11:50	36.3	56.1	19.8	998.23	0.363
12:10	37.3	56.2	18.9	998.40	0.380
12:30	36.9	56.8	19.9	998.22	0.361
12:50	37.1	55.5	18.4	998.49	0.391
13:10	37.4	57.2	19.8	998.23	0.363
13:30	37.7	57.2	19.5	998.29	0.369
13:50	37.9	57.2	19.3	998.33	0.372

Table 4.4: Calculated parameters from data obtained from table 4.3

Parameter	Value
Average coolant flow rate	(0.3742 ± 0.0104)kg/s
Average inlet temperature	$(36.73 \pm 0.82)^0$C
Average outlet temperature	$(55.96 \pm 1.04)^0$C
Average ΔT	$(19.23 \pm 0.53)^0$C
Power dissipated	30.2kW
Total heat losses from the reactor pool	0.54kW
Total reactor Power	30.7kW
Standard deviation of the readings (S_q)	$\pm$0.03kW
Uncertainty	2.5kW($\pm 8.1^o/_o$)

Table 4.5: Time and temperature data for calorimetric method calibration at half power (15kW)

Time(hr)	Neutron flux ($ncm^{-2}s^{-1}$) (Measured)	Neutron flux ($ncm^{-2}s^{-1}$) Preset	Temperature Inlet (℃)	Temperature Outlet (℃)	Temperature (℃) Pool
0.0	1.66E+08	5.00E+11	27.4	27.5	29.5
0.2	5.51E+11	5.00E+11	2	40.5	29.3
0.4	5.58E+11	5.00E+11	29.6	42.2	29.3
0.6	5.62E+11	5.00E+11	30.6	43.4	29.4
0.8	5.56E+11	5.00E+11	31.8	4.0	29.4
1.0	5.66E+11	5.00E+11	32.1	44.3	29.6
1.2	5.62E+11	5.00E+11	32.7	45.0	29.7
1.4	5.63E+11	5.00E+11	33.1	45.4	29.9
1.6	5.64E+11	5.00E+11	33.5	45.7	29.9
1.8	5.70E+11	5.00E+11	33.8	45.9	30.0
2.0	5.61E+11	5.00E+11	34.5	46.2	30.0
2.2	5.67E+11	5.00E+11	34.4	46.5	30.2
2.4	5.65E+11	5.00E+11	35.1	46.7	30.4
2.6	5.75E+11	5.00E+11	34.4	46.7	30.4
2.8	5.65E+11	5.00E+11	34.7	46.8	30.6
3.0	5.69E+11	5.00E+11	35.2	47.1	30.6
3.2	5.66E+11	5.00E+11	35.0	47.2	30.8
3.4	5.69E+11	5.00E+11	35.2	47.1	30.7
3.6	5.73E+11	5.00E+11	35.3	46.8	30.8
3.8	5.65E+11	5.00E+11	35.4	47.1	30.9
4.0	5.75E+11	5.00E+11	35.1	47.6	31.1
4.2	5.74E+11	5.00E+11	35.5	47.5	31.1
4.4	5.65E+11	5.00E+11	35.8	47.1	31.2
4.6	5.68E+11	5.00E+11	35.7	47.4	31.2
4.8	5.79E+11	5.00E+11	35.4	47.4	31.3
5.0	5.71E+11	5.00E+11	34.7	47.8	31.5
5.2	5.64E+11	5.00E+11	35.7	47.9	31.6
5.4	5.73E+11	5.00E+11	35.6	48.0	31.6
5.6	5.75E+11	5.00E+11	35.7	47.5	31.6
5.8	5.71E+11	5.00E+11	36.1	48.3	31.8
6.0	5.65E+11	5.00E+11	36.3	48.5	31.9
6.2	5.63E+11	5.00E+11	35.8	47.6	32.0

Table 4.6: Calorimetric method results at half power

Parameter	Value
Temperature -rise rate $\left(\frac{\Delta T}{\Delta t}\right)$	0.45 (^{o}C/h)
Average water temperature rise	(29.5 to 32.0)^{o}C
Average pool temperature	(30.60$\pm$0.84) ^{o}C
Power dissipated	15.1kW
Thermal losses from the reactor pool	0.7kW
Total reactor Power	15.8kW
Uncertainty	0.7kW($\pm 4^{o}/_{o}$)

Table 4.7: Time and temperature data for calorimetric method calibration at full power (30kW)

Time(hr)	Neutron flux ($ncm^{-2}s^{-1}$) (Measured)	Neutron flux ($ncm^{-2}s^{-1}$) Preset	Temperature Inlet (℃)	Temperature Outlet (℃)	Temperature (℃) Pool
0.0	1.36E+08	5.00+11	25.4	25.5	27.0
0.2	1.11E+12	1.00E+12	26.7	46.1	26.8
0.4	1.10E+12	1.00E+12	29.9	48.8	26.9
0.6	1.11E+12	1.00E+12	31.9	50.1	26.7
0.8	1.12E+12	1.00E+12	33.5	51.9	27.1
1.0	1.13E+12	1.00E+12	34.1	53.0	27.5
1.2	1.12E+12	1.00E+12	35.6	54.0	27.7
1.4	1.13E+12	1.00E+12	34.9	54.3	27.7
1.6	1.14E+12	1.00E+12	36.5	54.2	28.0
1.8	1.14E+12	1.00E+12	36.8	55.4	28.1
2.0	1.14E+12	1.00E+12	36.7	55.4	28.2
2.2	1.14E+12	1.00E+12	36.7	55.5	28.4
2.4	1.14E+12	1.00E+12	37.2	55.7	28.6
2.6	1.15E+12	1.00E+12	38.5	56.0	28.8
2.8	1.13E+12	1.00E+12	37.3	56.6	29.0
3.0	1.15E+12	1.00E+12	36.9	56.9	29.2
3.2	1.14E+12	1.00E+12	38	57.0	29.4
3.4	1.14E+12	1.00E+12	38.7	57.2	29.6
3.5	1.01E+12	1.00E+12	37.9	57.2	29.6

Table 4.8: Calorimetric method results at full power

Parameter	Value
Temperature -rise rate$\left(\frac{\Delta T}{\Delta t}\right)$	0.86 (^{o}C/h)
Average water temperature rise	(27.0 to 29.6)^{o}C
Average pool temperature	(28.12$\pm$0.97) ^{o}C
Power dissipated	29.2kW
Thermal losses from the reactor pool	1kW
Total reactor Power	30.2kW
Uncertainty	1.5kW($\pm5^{o}/_{o}$)

Table 4.9: Comparison between the uncertainties values obtained in this work and other calibrations

Reactor name	**Calibration method**	**Uncertainty value (%)**	**Reference**
TRIGA	Calorimetric	15	Zagar *et al.,* 1999
TRIGA	Calorimetric	21	Mesquita *et al.,* 2011
NIRR-1	Calorimetric	5	This work
NIRR-1	Calorimetric	4	This work
TRIGA	Heat Balance	5.7	Mesquita *et al.,* 2007
TRIGA	Heat Balance	5.9	Mesquita *et al.,* 2011
NIRR-1	Heat Balance	8.1	This work
NIRR-1	Heat Balance	7.9	This work

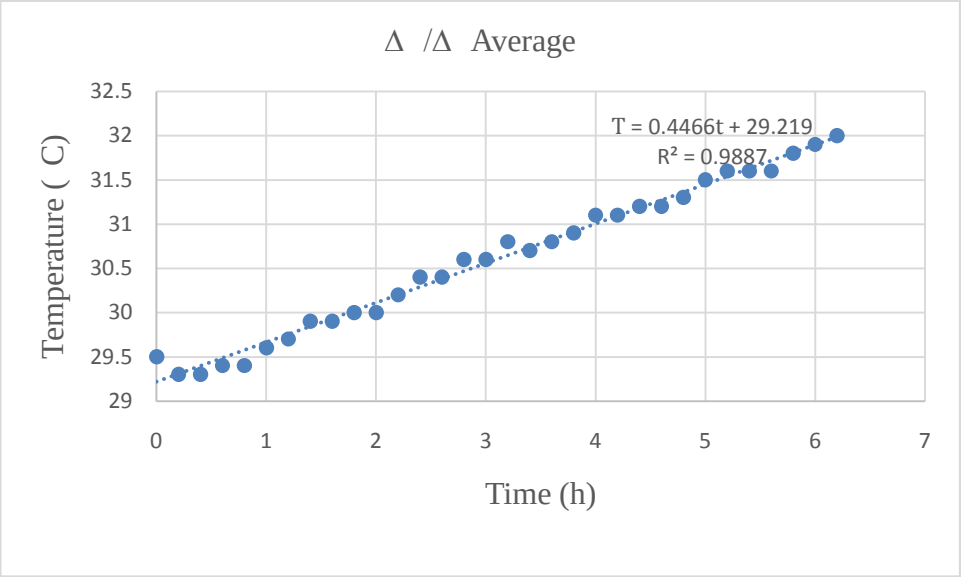


Fig 4.1: Average temperature rising rates during calorimetric power calibration procedure at (15kW)

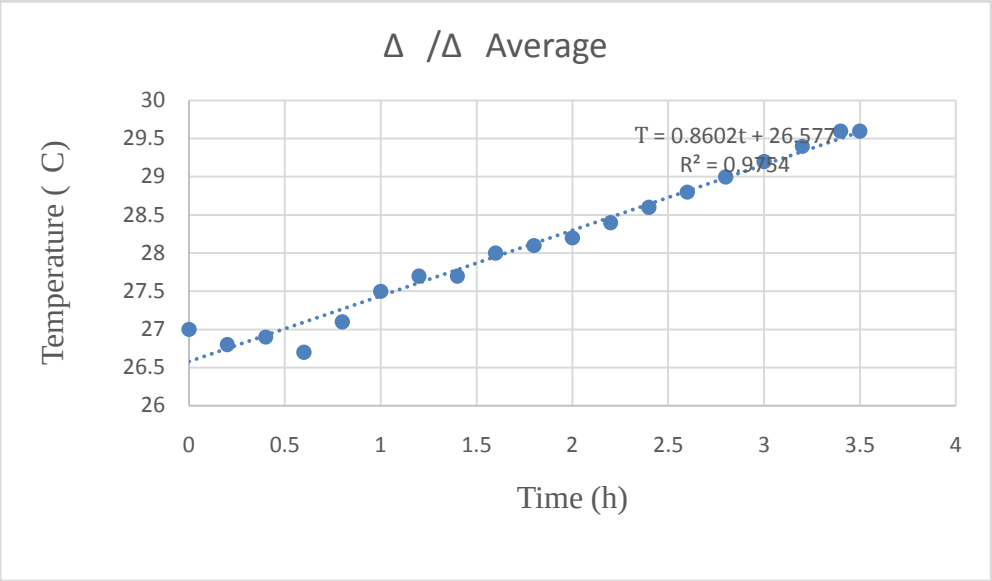


Fig 4.2: Average temperature rising rates during calorimetric power calibration procedure at (30kW)

CHAPTER FIVE

Discussion

5.1 The calorimetric method calibration

Reactor pool heat capacity constant was calculated. In the first approximation it was assumed that the reactor pool temperatures is constant throughout the pool and neglect all heat losses from the pool. The reactor pool was approximated as an insulated "point" system, the reactor pool was treated as insulated so the reactor heat capacity K was simply calculated from wet pool volume using Equation (2.7). All thermo physical properties of water in this calculation were evaluated at 20^{o}C. The estimated volume of water during the experiment was (29.0 ± 1.0) m^3. The heat capacity calculated was K= 33.5 kWh/^{o}C. The point pool approximation was employed and this approximation was reported to be acceptable for low calibration powers (eg 20kW) (Zagar *et al.,* 1999). Report from (Zagar *et al.,* 1999) revealed that with low calibration power the water natural convection is great to promote mixing in the pool as such there was no need for external stirring.

For calibration at half power, the reactor operated at 15 kW for over 6hr and the chiller system was isolated to avoid cooling of the pool water. Temperature versus time was recorded for every 12 minutes. The measurement of punctual temperatures and time were done with considerable accuracy and precision. There was a steady rise of pool temperature with time as can be seen in table 4.5. there was also a steady rise in the inlet and outlet temperature with time. The average inlet and outlet temperatures were found to be 34.0 ˚C and 45.8 ˚C respectively. The average pool temperature was calculated and was found to be 30.6 ˚C. The average water temperature rising rate and its fitted equation are shown in Fig 5.1. Table 4.5 presents the result of the experiment while Table 4.6 presents some

parameters calculated in this period using data obtained from table 4.5. The power dissipated was calculated using Equation (2.6). The thermal power obtained by this method was 15.8 kW (±4 %). The heat losses calculated using Equations (9), (13) and (18) was found to be 0.7kW. The uncertainty associated with this method as calculated using Equation (2.8b) was ±4 %.

The reactor neutron flux was preset at its licensed full power value of $1x10^{12}ncm^{-2}s^{-1}$ and the reactor operated for 3h 50min with the cooling system (chiller) switched off. During this period, the reactor pool temperature-rise with time was recorded for every 12 minutes. The measurement of punctual temperatures and time were done with considerable accuracy and precision. A steady rise in pool temperature was also observed as can be seen in table 4.7. The average water rising rate and its fitted equation for calibration at this power level are shown in Fig 4.2. Table 4.7 presents the results of the experiment. Some parameters calculated in this period are presented in Table 4.8. The heat losses calculated was found to be 1kW. The value of the thermal power ($30.2kW(\pm 5^{o}/_{o})$) obtained compare well with 29.5kW calculated from fitting formula based on the stimulation test data during initial startup as reported in the Nigeria Research Reactor -1 (NIRR-1) Safety Analysis Report, (2005)

5.2 Heat balance method calibration

For calibration at half power, the reactor was critical at 15 kW indicated on the control console and microcomputer system. The reactor operated during a period of 5 hours. The power dissipated was monitored during the whole test period. The inlet and outlet temperatures were measured and recorded for every 20 minutes and the coolant temperature

for each of the interval was determined. The results as presented in Table 4.1 show a steady rise in inlet and outlet temperatures with time. This is due to the compact nature of the core, which was designed to cause insufficient thermal circulation of coolant in the core. The coolant temperature was observed to be fairly constant with time having an average value of 12.36^{0}C. As can be seen in Table 4.1, the measured temperatures were observed to be stable for over 40 minutes. The value of the average core flow-rate obtained (0.2892 ± 0.0071) kg/s is in good agreement with the value obtained by (Jonah *et al.*, 2012) using PLTEMP/ANL code version 4.1. The heat losses from the reactor pool at this power level was 0.26kW. The thermal power obtained was 15.2kW($\pm 7.9^{o}/_{o}$). Table 4.1 presents the results of the experiment performed at this power level while Table 4.2 presents some parameters calculated from the result of the experiment.

For calibration at full power (30 kW), The reactor neutron flux was preset at its licensed full power value of $1x10^{12}ncm^{-2}s^{-1}$ and was operated for over 2h with the cooling system switched. As can be seen in table 4.3, the measured temperatures were observed to be stable for 40minutes. The value of the average core flow-rate obtained was (0.3742 ± 0.0104) kg/s. The value of the thermal power (30.7kW($\pm 8.1^{o}/_{o}$)) obtained in this work is in good agreement with 29.5kW calculated from fitting formula based on the stimulation test data during initial startup as reported in the Nigeria Research Reactor -1 (NIRR-1) Safety Analysis Report, (2005). Table 4.3 presents the results of the experiment performed at full power while Table 4.4 presents some parameters calculated from the result of the experiment.

5.3 Calorimetric method uncertainty

The power dissipated was calculated using Equation (2.6). Then, the power uncertainty was calculated considering the uncertainties of: the water density (ρ), the water volume in the pool (V_w), the estimated water specific heat capacity (c_p) as function of temperature, the water pool temperature rise (ΔT) and the time interval (Δt).

The power uncertainty U_q is given by the following equation (Figliola and Beasely, 1991, Holman, 1998, Mesquita *et al.*, 2009, 2011):

$$U_q^{'}=\left(\left(\frac{\partial q_{u_\rho}}{\partial \rho}\right)^2+\left(\frac{\partial q_{u_{c_\rho}}}{\partial c_\rho}\right)^2+\left(\frac{\partial q_{u_{V_w}}}{\partial V_w}\right)^2+\left(\frac{\partial q_{u_T}}{\partial T}\right)^2+\left(\frac{\partial q_{u_t}}{\partial t}\right)^2\right)^{\frac{1}{2}}$$

where U_ρ, U_{c_ρ},U_{v_w}, U_T and U_t are the consolidated uncertainties of the independent primary variables: ρ, c_p, V_w, T and t. Solving Equation (5.1) gives the following expression for the relative uncertainty value for the thermal power:

$$\frac{U_q^{'}}{q}=\left(\left(\frac{U_\rho}{\rho}\right)^2+\left(\frac{U_{C_\rho}}{C_\rho}\right)^2+\left(\frac{U_{v_w}}{V_w}\right)^2+\left(\frac{U_T}{T}\right)^2+\left(\frac{U_t}{t}\right)^2\right)^{\frac{1}{2}}$$

6.4 Heat balance method uncertainty

The power dissipated was calculated using Equation (2.10). Then, the power uncertainty was calculated considering the uncertainties of: flow-rate, ($\dot{m}$), the inlet and outlet temperatures and the specific heat capacity (C_p) of the coolant. The power uncertainty $U_q^{'}$ is given by the following equation (Figliola and Beasely, 1991, Holman, 1998, Mesquita *et al.*, 2009, 2011):

$$U_q^{'}=\left(\left(\frac{\partial q_{u_{\dot{m}}}}{\partial \dot{m}}\right)^2+\left(\frac{\partial q_{u_{c_\rho}}}{\partial c_\rho}\right)^2+\left(\frac{\partial q_{u_{T_{in}}}}{\partial T_{in}}\right)^2+\left(\frac{\partial q_{u_{T_{out}}}}{\partial T_{out}}\right)^2\right)^{\frac{1}{2}}$$

where $U_{\dot{m}}$, U_{c_ρ}, $U_{T_{in}}$, and $U_{T_{out}}$ are the consolidated uncertainties of the independent primary variables: mass flow-rate ($\dot{m}$) , specific heat capacity (C_p), inlet temperature (T_{in}) *and outlet temperature (T_{out}).*

$$\frac{U_q'}{q} = \left(\left(\frac{U_{\dot{m}}}{\dot{m}} \right)^2 + \left(\frac{U_{C_\rho}}{C_\rho} \right)^2 + \left(\frac{U_{T_{out}}}{T_{out} - T_{in}} \right)^2 + \left(\frac{U_{T_{out}}}{T_{out} - T_{in}} \right)^2 \right)^{\frac{1}{2}}$$

Adding the standard deviation of readings (S_q) of the average power to equation 2.10b, the uncertainty can then be expressed as:

$$\frac{U_q}{q} = \left(\left(\frac{U_q'}{q} \right)^2 + \left(\frac{S_q}{q} \right)^2 \right)^{\frac{1}{2}}$$

The range within the region of uncertainty in these experiments agrees with other nuclear reactor calibrations (Zagar *et al.*, 1999, Mesquita *et al.*, 2007, Mesquita *et al.*, 2011) as shown in Table 4.9.

CHAPTER SIX

CONCLUSION AND RECOMMENDATION

6.1 Conclusion

The knowledge of the reactor thermal power calibration is important for neutron flux and burn-up determination. Burn-up is linearly dependent on the thermal power of the reactor and its accuracy is crucial for the determination of the mass of burned ^{235}U, fission products and decay heat power generation.

The thermal power of Nigeria Research Reactor-1 (NIRR-1) has been calibrated. The results show that the heat balance method presented a larger uncertainty than the calorimetric method. The thermal power obtained by the calorimetric method were: (15.8±0.7) kW and (30.2±1.5) kW for calibrations at half and full power levels respectively. For heat balance method, the thermal power obtained were: (15.2±1.2) kW and (30.7±2.5) kW for calibrations at half ad full power levels respectively.

The value of the thermal power (30.7kW($\pm 8.1^{o}/_{o}$)) and (30.2kW($\pm 5^{o}/_{o}$)) obtained at full power for heat balance and calorimetric methods respectively in this work compare well with 29.5kW calculated from fitting formula based on the stimulation test data during initial startup as reported in the Nigeria Research Reactor -1 (NIRR-1) Safety Analysis Report, (2005). The range within the region of uncertainty in these experiments agrees with other nuclear reactor calibrations (Zagar *et al.,* 1999, Mesquita *et al.,* 2007, Mesquita *et al.,* 2011).

6.2 Recommendation

We recommend that:

(1) Much Longer irradiations (above 6hr, at lower flux) should be carried out and the measured thermal power be compared with the results obtained in this work to ascertain NIRR-1 stability.

(2) Calorimetric method of thermal power calibration described in this work should be adopted for routine calibration of NIRR-1 power.

(3) A stirrer should be installed in NIRR-1 pool to homogenize the temperature of the pool as is the case with power reactors. Although, the low temperature variations in NIRR-1 presumes a stirrer as rather unnecessary. Nevertheless, the investigation of its contribution is useful in power reactor studies.

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