DETERMINATION OF IRRADIATED CONTROL ROD WORTH OF NIGERIAN RESEARCH REACTOR – 1 (NIRR – 1) USING THE OPERATIONAL DATA

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Department of Physics,
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DECLARATION

I declare that the work in the dissertation entitled "Determination of irradiated control rod worth of Nigeria research reactor – 1 (NIRR – 1) from the operational data" has been performed by me in the Department of Physics, Faculty of Science, Ahmadu Bello University Zaria, under the supervision of Dr.Yakubu Ibrahim Viva and Dr. Sadiq Umar.

The information derived from literature has been duly acknowledged in the text and a list of references provided. No part of this work was previously presented for another degree or diploma at any University.

---------- Date----------

Joseph, Istifanus Abaleni

CERTIFICATION

The dissertation entitled "Determination of irradiated control rod worth of Nigeria research reactor – 1 (NIRR – 1) from operational data" by Joseph, IstifanusAbaleni with Registration number M.Sc/Sci/13834/2011-2012/P15SCPY8067 meets the regulations governing the award of Masters of Science (MSc) in Ahmadu Bello University, Zaria, and is approved for its contribution to knowledge and literary presentation.

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ABSTRACT

An important parameter in the design and analysis of a nuclear reactor is the reactivity worth of the control rod which is a measure of the efficiency of the control rod to absorb excess reactivity. During reactor operation, the control rod worth is affected by factors such as fuel burnup, Xenon concentration, Samarium concentration and position of the control rod in the core. This study investigates and utilizes the operational data of the reactor to determine the control rod worth of Nigeria research reactor – 1 (NIRR – 1) after 7 years of operation. The reactor operational data were exported from the reactor computer system (logbook) to Microsoft excel spread sheet and were analyzed in terms of the neutron flux and burn up time. The initial number of atoms in the cadmium rod was calculated using the dimension of the control rod and equation 3.8 and is found to be **8.8705554×10^{46}/mole**. The final number of atoms in the cadmium rod was calculated using a decay equation (equation 3.5) to be **8.8346840×10^{46}/mole**. The fractional change in the number of atoms was estimated from the results and the control rod worth was determined to be **6.97mk**. The shutdown margin (SDM) and the safety reactivity factor (SRF) were also evaluated from the result and were found to be **3.08mk** and **1.79** respectively. Since the reactivity worth of control elements will change due to both the depletion of control absorber nuclei and changes in the core power distribution, control management analysis must continually monitor the worth of the control system component throughout core life in order to maintain both an adequate shutdown margin and sufficient operating maneuverability.

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CHAPTER ONE

INTRODUCTION

1.1 Introduction

Nuclear reactors are initially loaded with a significantly large amount of fuel than that required to achieve criticality, because the intrinsic multiplication factor of the core will change during operation due to fuel burn-up and fission products. Sufficient excess reactivity is also provided to compensate for negative reactivity feedback effects due to temperature and power defects reactivity (Dudersadt, 1976). Therefore the core loading or enrichment will be determined by the desire to build into the core enough excess reactivity to allow power operation for a predetermined period of time.

The basic purpose of the reactor control system is to provide a means for starting the reactor, that is, bringing the power output up to the desired level, for maintaining it at that level, and for shutting it down in the course of routine operations. As an adjunct to the control system, a power reactor has a protection system designed to shut the reactor down automatically in the event that potentially unsafe conditions should arise. Safety rods was also designed and proposed for MNSR facilities (Ibrahim et al, 2012). An essential requirement of the control system is that it must be capable of introducing enough negative reactivity to compensate for the built-in (positive) reactivity at initial startup of the reactor (Glasstone and Sesonke, 1967)

The power level of the reactor depends on the macroscopic fission cross-section and the neutron flux. Over a short time interval, the cross section remains essentially constant, although it may not have the same value at all locations in the core. Hence, the power level at any instant can be considered proportional to the neutron flux. Four general methods are possible for changing the neutron flux in a reactor: they involve temporary addition or removal of (1) fuel,

(2) moderator, (3) reflector, or (4) a neutron absorber (poison). Each of these methods or a combination of them has been used (or proposed) for reactor control, but the procedure most commonly employed, especially in power reactors, is the insertion or withdrawal of a material, such as boron or cadmium, having a large cross-section for the absorption of neutrons. The absorber and the fissile material may be regarded as competitors for neutrons: the larger the proportion absorbed by the control material, the smaller the fraction available for fission, and vice versa. The control elements are generally referred to as control 'rods' because they were originally, and often still are, long cylindrical rods (or combination of such rods). In power reactors they are in the reactor core: however, in some experimental reactors, the control rods are located in the reflector close to the core where the thermal neutron flux tends to be high. For NIRR-1, the single control rod is located in the centre.

Nuclear research and test reactors worldwide have been in operation for over 60 years, supporting nuclear science and technology development as well as providing an important role as a research tool in scientific fields such as medicine, agriculture, industry and basic research (Parrish et al., 2007). Research reactors comprise of a wide range of commercial nuclear reactors which are generally not used for power generation but for the primary purpose of producing neutron source for research.

Shutdown margin is normally defined as negative reactivity by which the reactor is subcritical if all control rods were fully inserted in the core except the most reactive one. By this requirement it is provided that the reactor can be made subcritical even if one of the control rods fails. The minimum limiting shutdown margin is zero which in principle is sufficient to shutdown the reactor. However, licensing authorities increase the limiting value for some conservativeness taking account uncertainties in control rod worth and excess reactivity

determination (Ravnic, 2000). Typically, for research reactors like NIRR-1, the shutdown margin value in the design is 0.5$ (1$=1=8.1mk), the available reactivity of the fresh core of NIRR-1 amounts to 4.97mk during the zero power experiment (SAR, 2005; Jonah et al., 2007). In order to bring back the core to its design shutdown limits of 0.5$, a cadmium poison of worth 1.2mk was inserted into one of its inner irradiation channels. The reactor has only one central control rod of integral reactivity worth of about 7.0mk that perform shim, safety and regulatory functions (Jonah et al, 2007; Qazi et al, 1996).

Because of burn-up of fission products, the core excess reactivity of NIRR-1 went down to about 2.8mk, which is not within the licensing condition. To bring the core excess reactivity within 3.0 – 4.0mk, the cadmium poison was raised bringing the core excess reactivity to 3.89mk about 0.12mk more than at the beginning. In this work, we will calculate the worth of the irradiated control rod to ascertain if it can still perform its safety function as required. The applied method for the calculations is introduced and the results are presented.

1.2 Nuclear Reactors

A nuclear reactor is a device to initiate and control a sustained nuclear chain reaction. In such a device, neutrons are used to induce nuclear fission reactions in heavy nuclei. These fission neutrons can be utilized to induce still further fission reactions, thereby inducing a chain of fission events. In a very narrow sense then, a nuclear reactor is simply a sufficiently large mass of appropriately fissile material (^{235}U or ^{239}Pu) in which such a controlled fission chain reaction can be sustained.

However a modern power reactor is a considerably more complex beast. It must not only contain a lattice of very carefully refined and fabricated nuclear fuel, but must as well provide

for cooling this fuel during the course of the chain reaction as fission energy is released, while maintaining the fuel in a very precise geometrical arrangement with appropriate structural materials. Furthermore, some mechanism must be provided to control the chain reaction, shield the surroundings of the reactor from intense nuclear radiation generated during the fission reactions, and provide for replacing nuclear fuel assemblies when the fission chain reaction has depleted their concentration of fissile nuclei. If the reactor is to produce power in a useful fashion, it must also be designed to operate both economically and safely.

Nuclear reactors have been used for over 50 years in a variety of applications. They are particularly valuable tools for nuclear research since they produce copious amounts of nuclear radiations, primarily in the form of neutrons and gamma rays (Duderstadt , 1976). Such radiation can be used to probe the microscopic structure and dynamics of matter. The radiation produced by reactors can also be used to transmute nuclei into artificial isotopes that can be used, for example, as radioactive tracers in industrial or medical applications. Small, compact reactors have been used for propulsion in submarines, ships, aircraft, and rocket vehicles. Indeed the present generation of light water reactors used in nuclear power plants are little more than the very big younger brothers of the propulsion reactors used in nuclear submarines.

The most significant application of nuclear fission reactors is in power plant. A nuclear power plant does not significantly differ from conventional fossil-fuelled plants: the reactor produces heat, which is used to generate steam, which is used to run turbines, which are connected to a generator that produces electricity. Such a description does not, however, tell the whole truth. The reactor is actually aware of everything that goes on within the system and all disturbances are inevitably reflected in the core neutronics. This is the crucial difference to conventional power plants. Another difference is that nuclear fuel is kept inside the reactor core

for a long time, usually for the duration of the operating cycle, which typically ranges from one to two years. The physical properties of the fuel change dramatically during the cycle and the reactor must be adjusted to cope with the changes. The boilers in conventional power plants are simply fed with a continuous stream of fuel and the reaction products are expelled at the same rate.

Nuclear reactors are classified according to variety of characteristic features such as fuel type, neutron energy at which greater part of all fissions occur, moderator material, arrangement and spatial disposition of fuel and moderator, purpose of the reactor and heat removal methods and coolant employed (Lamarsh, 1972).

1.2.1 **Thermal Reactors**

If the reactor core contains a considerably proportion of a moderator, the high energy of the fission neutrons will be rapidly decreased to the thermal region by scattering. Most of the fissions in such a reactor, called a thermal reactor, will then be caused by thermal (slow) neutrons, thermal reactors have the advantage over fast breeder reactors in greater flexibility of design, there is reasonable choice of both moderators and coolants, as well as of fuel materials. Depending on the nature of the fuel and moderator, a thermal reactor may be quite small or relatively large. In most commercial thermal reactors the fuel is either natural uranium (0.7% ^{235}U), with heavy water or graphite as the moderator, or uranium containing 90% of the fissile isotope with ordinary water as moderator for like the case of Nigerian Research Reactor 1. The fuel of a 2-4% fissile isotope reactor contains considerable proportion fertile ^{238}U (or ^{232}Th in few cases). During reactor operation, some of the fertile nuclides is converted into fissile ^{239}Pu (or ^{233}U) and could therefore be theoretically considered as thermal breeder reactor.

1.2.2 **Thermal Breeder Reactor**

Since there is a large probability that slow neutrons in a thermal reactor will be captured in non-fission reactions in fuel and structural materials, and because some neutrons are lost by escape, less than one neutron is available per fission, on the average, for the conversion of fertile into fissile nuclei. Hence, the quality of ^{239}Pu produced in a thermal reactor is usually less than the amount of ^{235}U consumed in maintaining the fission chain. Consequently, it is difficult to design thermal breeder based on ^{235}U (or ^{239}Pu) as the fuel. A thermal breeder of this type is theoretically possible by decreasing the loss of escaping neutrons in a large reactor and taking advantage of the fission of ^{238}U by the fast neutrons which are always present. This situation does not apply however, when ^{233}U is the fissile material, and so thermal breeders involving these nuclei, with ^{232}Th as the fertile species are possible.

1.2.3 **Fast Reactors**

Nearly all neutrons liberated in fission have high energies and so if the amount of element of low mass number in the reactor core and reflector is limited, the majority of the fission is produced by fast neutrons (Lewis, 2008). A nuclear reactor in which this is the case is called a fast reactor. The fuel materials for such reactor must contain a significant proportion of a fissile nuclides and some quantity of fertile material.

In fast reactors both the core and the reflector, called a blanket contain a fertile material which is converted into fissile specie by neutrons capture. Parasitic capture of fast neutrons is relatively low, and if the lost of neutrons by escape can be kept to a minimum; more than one neutron will be available, per fission, for the conversion of fertile into fissile nuclear in the core

and blanket. Under these circumstances it is possible for more fertile material to be produced by neutron capture than is consumed.

1.2.4 **Fast Breeder Reactors**

A reactor that is capable of producing more fissile materials by neutron capture than is consumed is called breeder. Power-breeder reactors which may play an increasingly important role in the future will be capable of producing power and at the same time generating more fissile material than they consume. In a true breeder, the fissile nuclide produced is the same as the one consumed, example, a fast reactor with ^{239}Pu as the fissile species ^{233}U and ^{232}Th as the fissile nuclides, respectively. Breeding of ^{233}U can also be achieved in reactors that do not depend on fast neutron fissions.

A fast reactor consuming ^{235}U and producing a large amount of ^{239}Pu (from ^{238}U) is a type of breeder but of less significance than a true breeder. The reserves of ^{235}U, the only fissile material existing in nature, are small. Hence, there may ultimately come a time when essentially all the available ^{235}U is exhausted. The further utilization of the remaining ^{238}U and ^{232}Th will then depends on the use of ^{239}Po or ^{233}U to maintain the fission chain. It is for this reason that the true breeder, which produces and consume these fissile nuclides, are special interest.

Classification based on moderator material used by thermal reactors include: graphite-moderated reactors, water moderated reactors (heavy-water reactors and light water reactors), light element-moderated reactors. Classification by coolant include: water cooled reactor (pressurized water reactors, boiling water reactor, pool-type reactor), liquid metal cooled reactor, gas - cooled reactors, molten salt reactors. Classification by use include: electricity, propulsion, production reactors for transmutation of elements (breeder reactors, creating various radioactive

isotopes, production of materials for nuclear weapons such as weapons-grade plutonium), providing a source of neutron radiation, research reactors.

1.3 Description of NIRR-1

NIRR-1 is a miniature Neutron Source Reactor (MNSR) designed by China Institute of Atomic Energy (Zhou Yongmao, 1986). It went critical on 3rd February, 2004 (Jonah et al., 2007) and since then it has been operational and utilized for Neutron Activation Analysis (NAA) applications (Jonah, 2005, 2006, 2008). It is a tank-in-pool type reactor having maximum authorized thermal power level of 31kW, in which light water is used as the moderator and coolant, and biological shielding. The fuel is 90.2% of ^{235}U, in the form of UAl, in Al alloy cladding. It has Beryllium metal as reflector and can operate for maximum of 4.5hours at full power (SAR, 2005). The core of the reactor is located at the center of the pool. It is 230mm x 230mm square cylinder (Jonah, 2007) having a total of 347 fuel pins with 3 Aluminum dummies in the fuel lattice (Yang, 1992). The length of the fuel element is 248mm, the active length being 230mm with 9mm Al alloy plug at each end (Jonah et al, 2007). The diameter of the fuel meat is 4.3mm and the ^{235}U loading in each fuel is 2.88g. The Al alloy clad is 0.6mm. At the centre of the core is one control rod which performs shim, safety and regulatory functions. The control rod is made up of Cadmium absorber 266mm long and 3.9mm in diameter with a stainless steel of 0.5mm thick as its cladding material. The control rod has a total length of 450mm and reactivity worth of 7.0mK. The reactor has ten irradiation channels comprising of five inner channels. The five channels including three of the outer channels have diameter of 22mm each while the two of the outer channels have a diameter of 34mm. The height of all the channels in the core region is 186.5mm.

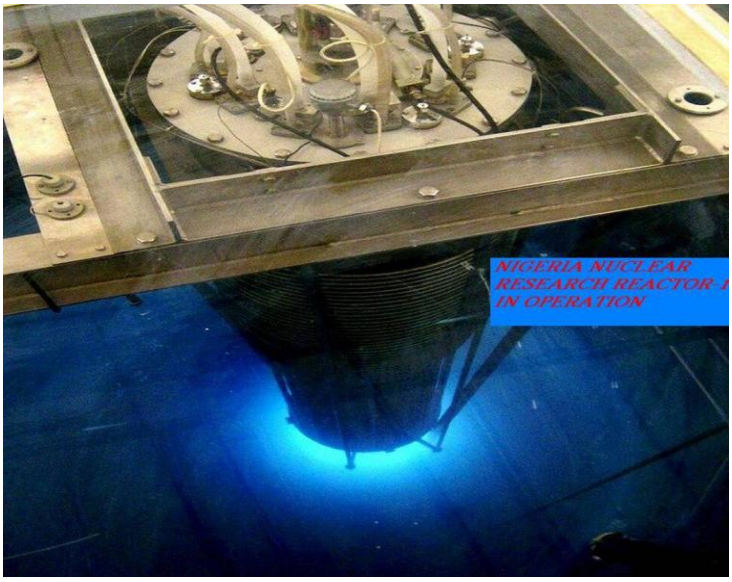

Fig 1.1 (Plate): Sectional view of NIRR-1

1.4 Control Rods

Control rods are an important reactor's component for maintaining the desired state of fission reactions within a nuclear reactor. They constitute a real-time control of the fission process, which is crucial for both keeping the fission chain reaction active and preventing it from accelerating beyond control. The nuclear fission chain reaction is the fundamental process by which nuclear reactors produce usable energy. Most commonly, Uranium ^{235}U is the fissionable material used in this chain reaction (as shown in Fig. 1), although the basic outline is applicable generally. In this process, a ^{235}U atom is struck by an incident neutron, causing the atom to fission into two smaller atoms (Krypton ^{92}K and Barium ^{141}B) and also release an average of 2.5

new neutrons. These new neutrons can then collide into more ^{235}U atoms, which undergo the same fission process, creating a chain reaction that releases substantial energy with each fission event.

Therefore, the key to sustaining the fission chain reaction is the amount of neutrons that propagate to the next generation of fissions. However, not all of the fission-produced neutrons trigger another fission event (some may simply escape the reactor or be absorbed by non-fissile isotopes, for example), so it is necessary to carefully engineer every parameter of the reactor to ensure that at least one neutron from each fission event is able to trigger another fission. Controls rods are one such adjustable parameter.

Control rods are usually used in control rod assemblies (typically 20 rods for a commercial Pressurized Water Reactor - PWR assembly) and inserted into guide tubes within a fuel element. A control rod is removed from or inserted into the central core of a nuclear reactor in order to increase or decrease the neutron flux, which describes the number of neutrons that split further uranium atoms. This in turn affects the thermal power, the amount of steam produced and hence the electricity generated.

The material used for the control rods varies depending on reactor design. Generally, the material selected should have a good absorption cross section for neutrons and have a long lifetime as an absorber (not burn out rapidly). The ability of a control rod to absorb neutrons can be adjusted during manufacture. A control rod that is referred to as a “black” absorber absorbs essentially all incident neutrons. A “grey” absorber absorbs only a part of them, while it takes more grey rods than black rods for a given reactivity effect, the grey rods are often preferred

because they cause a smaller depressions in the neutron flux and power in the vicinity of the rod. This leads to a flatter neutron flux profile and more even power distribution in the core.

If grey rods are desired, the amount of material with high absorption cross section that is loaded in the rod is limited. Material with a very high absorption cross section may not be desired for use in a control rod, because it will burn out rapidly due to its high absorption cross section. The same amount of reactivity worth can be achieved by manufacturing the control rod from material with a slightly lower cross section and by loading more of the material. This also results in a rod that does not burn out as rapidly.

Another factor in control rod material selection is that materials that resonantly absorb neutrons are often preferred to those that merely have high thermal neutron absorption cross sections. Resonance neutron absorbers absorb neutrons in the epithermal energy range. The path length travelled by the epithermal neutrons in a reactor is greater than the path length travelled by thermal neutrons. Therefore, a resonance absorber absorbs neutrons that have their last collision farther (on the average) from the control rod than a thermal absorber. This has the effect of making the area of influence around a resonance absorber larger than around a thermal absorber and is useful in maintaining a flatter flux profile.

Sufficient excess reactivity must be provided to compensate for negative reactivity feedback effects such as those represented by the temperature and power defects of reactivity. To compensate for this excess reactivity, it is necessary to introduce an amount of negative reactivity into the core which can be adjusted or control at will. This control reactivity can be used both to compensate for the excess reactivity necessary for long term core operation and also to adjust the power level of the reactor in order to bring the core to power, follow load demands, and shut the core down.

The control reactivity is most often present in the form of a strong neutron absorbers that can be inserted into or withdrawn from the core (although movable core reflector elements or fuel assemblies and coolant flow have also occasionally been used for reactivity control). The determination of the control reactivity requirements and the apportionment of control reactivity among various types of control elements is a very important aspect of nuclear reactor core design. There are several schemes used for introducing control absorption into a nuclear reactor core. One common method is to insert movable rods of absorbing material into the core. Such *movable control elements* not only can be used to adjust the core power, but because of their rapid response can also be used for scramming the reactor, as well as for shim and power shaping. Fixed absorbing materials are sometimes fabricated into the core with the intent that such absorption will gradually burn out along with the fuel. These burnable poisons are useful for extending the initial core lifetime of reactors (kalcheva and Koonen, 2007).

The grouping of such control rods, the determination of the control rod withdrawal sequence, and the coordination of movable control elements will change, due to both the depletion of the control absorber nuclei and changes in the core power distribution, control management analysis must continually monitor the worth of the control system components throughout core life in order to maintain both adequate shutdown margin and sufficient operating maneuverability.

Control rods often stand vertically within the core. In pressurized water reactors (PWRs) they are inserted from above, with the control rod drive mechanisms mounted on the reactor pressure vessel head. In Boiling Water Reactors (BWR), due to the necessity of a steam dryer above the core, this design requires insertion of the control rods from beneath. The control

rods are partially removed from the core to allow a chain reaction to occur. The number of control rods inserted and the distance to which they are inserted can be varied to control activity.

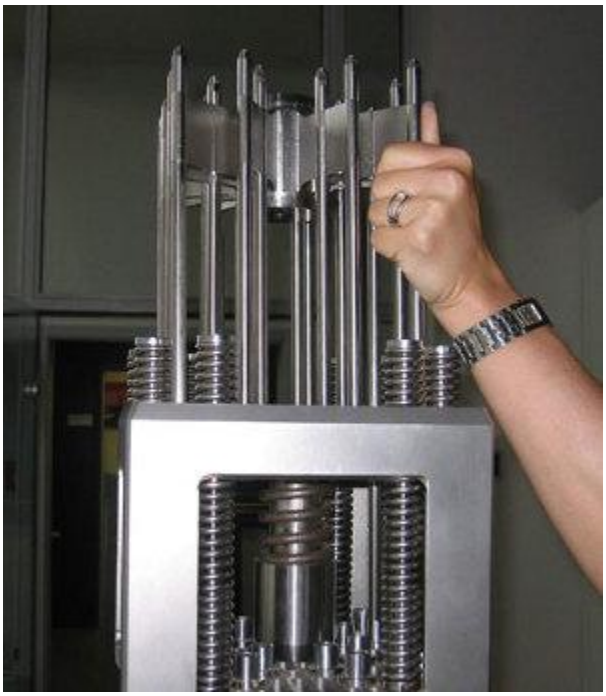

Fig 1.2: A cluster control rod assembly. The individual rods are attached by the *spider* at the top. Source:(Wikimedia Commons 2011)

Table 1.1: Dimension of Cadmium control rod of NIRR-1

Cd tube outer diameter = 3.9mm	Inner diameter = 2.0mm
Inside Cd tube: Al rod 2.0 X 266(mm)	Outside Cd tube: stainless steel outer diameter = 5mm Wall thickness = 0.5mm

Total length of Cd = 450mm	

1.4.1 **Types of Control Rods**

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

1.4.1.1 **Shim rods** – In a nuclear reactor, shim rods are used to compensate for the excess reactivity of a freshly loaded reactor, to influence the neutron flux distribution, for coarse control and/or to remove reactivity in relatively large amount. Shim rods are designed to compensate for the effects of burnup (that is energy production). Reactivity changes resulting from burnup can be large but, they occur slowly over periods of days to years, as compared with the seconds-to-minutes range over which safety actions and routine regulation take place. Therefore, shim rods may control a significant amount of reactivity but, they will work optimally only when constraints are imposed on their speed of movement. A common way in which shims are operated is by inserting or removing them as regulating rods reach the end of their most useful position range. When this happens, shim rods are moved so that the reactor rod can be reset. The functions of shim and safety rods are sometimes combined in rods that have low rates of withdrawal but that can be rapidly inserted. This is usually done when the effect of burnup decreases reactivity. The rods are partially inserted at the outset of operation. However, in the event that the system must be shut down as quickly as possible, the reactor operator may "Scram" the reactor, fully dropping the control rod into the core and immediately sending the reactor into a subcritical state. The amount of shim control required can be reduced by the use of a burnable "poison". This is a neutron-absorbing material, such as boron or gadolinium that burns off faster than the fissile material does over the lifetime of the core. At the beginning of operation, the inclusion of a burnable poison regulates the extra reactivity that has been built into

the fuel to compensate for the amount of fuel consumed. At the end of an operating period, the absorbing material is often completely transformed through neutron capture.

1.4.1.2 **Regulating rods** – Is a control rod that has a small reactivity equivalent and is used to make fine adjustments to the reactor reactivity and to maintain desired power or temperature. It usually moves more quickly than a shim rod and has a reactivity worth of less than a dollar. Regulating rods are deliberately designed to affect reactivity only by a small degree. It is assumed that at some time the rods might be totally withdrawn by mistake, and the idea is to keep the added reactivity in such cases well within sensible limits. A well-designed regulating rod will add so little reactivity when it is removed that the delayed neutrons will continue to control the rate of power increase (Encyclopaedia Britannica). A regulating rod is also a control rod intended to accomplish rapid, fine and sometimes continuous adjustment of the reactivity of a nuclear reactor (McGraw-Hill.,2003). Regulating rod is made of a substance that easily absorbs neutrons and is used to regulate the rate of nuclear fission in a nuclear reactor. The neutron balance, or reactivity of the reactor is the ratio between the number of neutrons released and the number of neutrons absorbed during fission per unit time and it is controlled by means of a regulating rod which is inserted into the core to reduce reactivity and power of the reactor and even in the complete termination of a chain reaction. The removal of the rod from the core results in an increase in the reactor's reactivity and a corresponding increase in its power. The position of the rod is also varied to compensate for operational changes in the reactivity of a nuclear reactor that result from, for example, a temperature change, a reduction in nuclear fuel, or an increased in the number of fragments of atomic nuclei that absorb neutrons.

1.4.1.3 **Safety rods** – Are control rods used to decrease the reactor reactivity in the case of emergencies. Safety rods have a large reactivity and their motion is very fast. Addition of a large amount of negative reactivity by rapidly inserting the safety rods is referred to as a "scram" or "trip" The most important function of the safety rod is to shut down the reactor, either when such a shutdown is scheduled or in case of a real or suspected emergency. These rods contain enough absorber to terminate a chain reaction under any conceivable condition. they are withdrawn before fuel is loaded and remain available in case a loading error require their action after the fuel is loaded, the rods are inserted, to be withdrawn again when the reactor is ready for operation. The mechanism by which they are moved is designed to be fail – sense in the sense that if there is a mechanical failure, the safety rods will fall by gravity or some other safe means into the core. In some cases, moreover, the safety rods have an automatic feature such as a fuse, which releases them by virtue of physical effects independent of electronic signals.

1.4.2 Scram Control

The reactor control system must be capable of shutting the reactor down under any credible operating conditions. Elements used for such scram control purposes must be capable of inserting negative reactivity very rapidly and must operate with an extremely high degree of reliability. Scram is an acronym for "safety control rod axe man". Webster defines scram as "A rapid emergency shutdown of a nuclear reactor". To most of us in the nuclear business this term means "to place the reactor in a safe condition". Scrams are usually activated or "tripped" by electronic means through some of the hundreds of safety sensors and systems of a modern nuclear reactor, but they still have a manual Scram. The term can always be found on any reactor

control desk, adjacent to a large red button labeled “Scram”. For MNSR, we have the single control rod performing scram function.

1.4.3 **Power Regulation**

Certain control elements are designed to compensate for small reactivity transients caused by changes in load demand, core temperature, and for power-level maneuvering. The regulating element is used to adjust the power level of the reactor and to compensate for short – term changes in reactivity.

1.4.4 **Shim Control**

Shim control elements are designed to cover the excess reactivity necessary to compensate for long-term fuel depletion and fission product buildup, as well as to shape the power distribution in the core in order to obtain better thermal performance and more uniform fuel burn up. The reactivity worth of such elements must be quite large to ensure adjustments over long time periods.

1.5 **Statement of Research Problem**

NIRR-1 has a single safety control rod made of cadmium absorber which has been operated almost a decade. Its efficiency to absorb excess reactivity may be affected by burnups in the interior of the core. Safety regulations require that the control rod must have a shutdown margin greater than 2.5mK and a safety reactivity factor greater than 1.5 (Balogun, 2003), so as a result of the long time burnups it is essential to re-evaluate whether the control rod meet up with these criteria (safety issue).

Recently, NIRR-1 was modified by increasing the excess reactivity (from 2.87mK to 3.89mK) more than the initial excess reactivity of 3.77mK due to licensing condition; because of this modification it is pertinent to evaluate the control rod to know whether the control rod can perform its safety functions.

1.6 Justification for the Study

(i) Reactor safety experts have recommended for the periodic assessment of control rods (Hong, 1999) which may be depleted as a result of burnups in the core interior.

(ii) Recommendation for safety assessment after reactor upgrade or modification (Pond et al., 1998)

(iii) NIRR-1 has one control rod as such its safety evaluation is very important.

1.7 Aim and Objectives

The aim of this work is to determine the irradiated control rod worth of NIRR-1.

The primary objective of this work is to:

(a) Determine the initial number of atoms of the Cd control rod

(b) Construct the operational history of NIRR-1

(c) Use mathematical model to calculate the fractional change in the number of atoms

(d) Calculate the present rod worth

(e) Calculate the Shut Down Margin(SDM) and Safety Reactivity Factor(SRF)

1.8 Scope of the research

There are several methods for the calculation of control rod worth. In this research, the reactor operational data generated from the daily operation of Nigeria research reactor – 1 (NIRR – 1) from February 2004 to March 2011 will be analyzed and utilized. The control rod specification was taken from the manufacturers' manual.

CHAPTER TWO

LITERATURE REVIEW

2.1 Interactions of Neutron with Matter

Nuclear reactors are based on self-sustaining fission chain reaction carried out by neutrons in fissile materials (Duderstadt, 1976). Neutrons have no charge and they can easily penetrate through electron cloud and coulomb potential barrier of the atomic nucleus. In a reactor neutrons are produced, slowed down and captured. Furthermore, energy is produced by the fission process, and, to a lesser extent, by radioactive decay. The most important nuclear quantities of a nucleus present in a reactor are capture, fission, and scattering cross sections (Duderstadt et al, 1976); Lewis et al, 1984). Cross section is the probability of interaction between a neutron and a target nucleus. This quantity depends on the target nucleus, the interaction type and the energy of the incident neutron.

2.1.1 Fission

This involves the formation of a compound nucleus after neutron capture. The compound excited state formed decays by energetically splitting the nucleus into two fragments.

$$^{235}_{92}U + ^{1}_{0}n \longrightarrow ^{235}_{92}U^{*} \longrightarrow \ ^{144}_{56}Ba + \ ^{89}_{36}Kr + 3^{1}_{0}n \qquad \text{----------} (2.1)$$

In addition to the prompt neutrons emitted at fission event, delayed neutrons are produced in the decay chains of fission product isotopes; an example of such a reaction is the decay fission product of ^{87}Br

$$^{87}_{35}Br \longrightarrow \ ^{87}_{36}Kr^{*} \longrightarrow ^{86}_{36}Kr + ^{1}_{0}n \qquad \text{----------} (2.2)$$

The average number of neutrons η emitted following the capture of a neutron by a fissile nucleus is a basic quantity for the possibility of establishing a chain reaction. It can be decomposed in two factors:

a. The probability that a capture leads to fission.

$$\frac{\sigma_f}{\sigma_f + \sigma_c} = \frac{1}{1+\varpi} \qquad \text{----------} (2.3)$$

where $\varpi = \frac{\sigma_c}{\sigma_f}$, $\sigma_{c\ and}\ \sigma_f$ denote the capture and fission cross sections respectively.

b. The mean number of neutrons emitted per fission.

Thus, for a single fissile nucleus we can write

$$\eta = \upsilon\left(\frac{1}{1+\varpi}\right) = \frac{\upsilon}{1+\varpi} \qquad \text{----------} (2.4)$$

where υ is the neutron speed. For mixture of isotopes, we have

$$\eta = \frac{\Sigma_j \Sigma_f^j}{\Sigma_j \Sigma_a^j} \qquad \text{----------} (2.5)$$

where $\sum_f^j$ and $\sum_a^j$ are the macroscopic fission and absorption cross sections respectively.

2.1.2 **Scattering**

The scattering cross section, either elastic σ_e(n,n) or in-elastic σ_{in}(n,n) controls the propagation of neutrons, in a reactor, the slowing down of the neutron following an elastic scattering is controlled by the atomic mass A of the nucleus. After the neutron is scattered an angle of θ in the centre of mass, the final laboratory energy E_f of the neutron of initial energy E_i is given by (Duderstadt, 1976)

$$E_f = \frac{E_i}{2}((1+\propto) + (1 - \propto)\cos\theta) \quad \text{----------(2.6)}$$

where $\propto = (\frac{A-1}{A+1})^2$. If the scattering in the center of mass is isotropic it follows that all final energies between E_i and $E_{i\propto}$ are equiprobable. Since, in this latter case, the neutron energy loss is proportional to its initial energy, it is convenient to measure energies in terms of lethargy $= \frac{E_i}{E}$ where E_i is some arbitrary initial energy (usually the average energy of fission neutrons), and E the actual neutron energy.

2.1.3 **Capture**

This is a reaction in which no secondary neutrons are emitted. The simplest capture reaction is the (n,γ) reaction. An example of such a reaction is the radiative neutron capture of ^{238}U.

$$^{238}_{92}U + ^{1}_{0}n \rightarrow ^{239}_{92}U + \gamma \quad \text{----------(2.7)}$$

Other capture reactions are proton emission (n,p) and $\propto$-particle emission (n,$\propto$). The (n,$\propto$) reaction is important in reactor control and a typical example is the one that occurs in Boron.

$$^{10}_{5}B + ^{1}_{0}n \rightarrow ^{11}_{6}B^{*} \rightarrow ^{7}_{3}Li + ^{4}_{2}He \quad \text{----------(2.8)}$$

2.2 **Cross Section**

A cross section is the effective area that governs the probability of some scattering or absorption event. Together with particle density and path length, it can be used to predict the total scattering probability via the Beer-Lambert law. In nuclear and particle Physics, the concept of a cross section is used to express the likelihood of interaction between particles.

When particles in a beam are thrown against a foil made of a certain substance, the cross section σ is a hypothetical area measure around the target particles of the substance (usually its atoms) that represents a surface. If a particle of the beam crosses this surface, there will be some kind of interaction. The term is derived from the purely classical picture of (a large number of) pint-like projectiles directed to an area that includes a solid target. Assuming that an interaction will occur (with 100% probability) if the projectile hits the solid, and not at all (0% probability) if it misses, the total interaction probability for the single projectile will be the ratio of the area of the section of the solid (the cross section, represented by σ) to the total target area.

This basic concept is then extended to the cases where the interaction probability in the targeted area assumes intermediate values – because the target itself is not homogeneous, or because the interaction is mediated by a non-uniform field. A particular case is scattering. In nuclear and particle physics, the concept of a neutron cross section is used to express the likelihood of interaction between an incident neutron and a target nucleus. In conjuction with the neutron flux, it enables the calculation of the reaction rate. The standard unit for measuring the cross section is the barn, which is equal to $10^{-28}m^2$ or $10^{-24}cm^2$.

The neutron cross section, and therefore the probability of an interaction, depends on:

The target type, the type of nuclear reaction, the incident particle energy, also called speed or temperature, and to a lesser extent, of its relative angle between the incident neutron and the target nuclide, the target nuclide temperature.

The likelihood of interaction between an incident neutron and a target nuclide, independent of the type of reaction, is expressed with the help of the total cross section σT. However, it may be useful to know if the incoming particle bounces off the target (and therefore continue travelling after the interaction) or disappears after the reaction. For that reason, the scattering and absorption cross section σS and σA are defined and the total cross section is simply the sum of the two partial cross sections:

$$\sigma T = \sigma S + \sigma A \quad \text{----------} \quad (2.9)$$

2.3 **Radioactive Decay**

Certain nuclei are unstable in the sense that they may spontaneously undergo a transformation into a different nuclide, usually accompanied by the emission of energetic particles. Such a spontaneous nuclear transformation is referred to as radioactive decay. The three most common types of radioactive decay found naturally occurring nuclides include alpha decay, in which a nucleus emits a helium nucleus ^{4}He; beta decay, which corresponds to the conversion of a neutron into a proton, generally accompanied by the emission of an electron and a neutrino; and gamma decay, the transition of a nucleus from one excited state to a lower excited state with the accompanying emission of a photon. However, other types of radioactive

decay are possible in a nuclear reactor since many unstable nuclides are produced in fission which does not occur in nature. For example, a certain nuclei of $^{87}Kr_{36}$ may decay by emitting a neutron.

The fundamental law describing decay is based on the experimental observation that the probability that a nucleus will decay in a given time interval is essentially constant, independent of the age of the nucleus or its environment, dependent only on the type of the nucleus itself. Hence the time rate of change of the number of original nuclei of a given type must be proportional to the number of nuclei present at that time. Let us call the proportional constant λ.

If dN nuclei decay during the time interval dt out of N nuclei present at time t, the fractional number of decayed nuclei over the total number of nuclei is dN/N, so that the decay probability per second is

Decay Probability Per sec = $\frac{dN/N}{dt}$ ---------- (2.10)

Experimental evidence has shown that this quantity has a constant value for a given nuclide and that it is the characteristic of this particular radioactive nuclide. It is called the decay constant of the nuclide and denoted by λ.

The fundamental law of radioactive decay can then be stated in the form

$$\frac{-dN}{N} = \lambda dt \quad \text{---------- (2.11)}$$

Here, the minus sign has been introduced to take account of the fact that dN represents a decrease in the number of nuclei present during the positive time interval dt. The above expression can be re-written in the slightly modified but equivalent form

$$\frac{dN}{dt} = -\lambda N \quad \text{---------- (2.12)}$$

It becomes clear that the number of nuclear decays per unit time, dN/dt, called the activity of the substance, is always proportional to the total number of nuclei, N, still present at time t.

If equation (2.11) is integrated between the times t=0 and t=t, the number of nuclei N, which remains after a time t has elapsed can be obtained in terms of the initial number N_o, which were present at time t=0

$$\int_{N_o}^{N} \frac{dN}{N} = -\int_0^t \lambda dt \quad \text{----------} \quad (2.13)$$

$$\log_e \frac{N}{N_o} = -\lambda t \quad \text{----------} \quad (2.14)$$

$$N = N_o \exp(-\lambda t) \quad \text{----------} \quad (2.15)$$

which shows that the number of surviving nuclei decreases exponentially.

Consequently, a rough estimation of the rod's divergence from its original state can be obtained by approximating the reduction dN_j of the j^{th} absorber's atom density N_j for an irradiation period dt, given by

$$dN_j = -N_j(\sum_{g=1}^{G} \sigma_{j,g}\Phi_g)dt \quad 2.16$$

whereΦ_g = Neutron flux in the energy group g

$\sigma_{j,g}$ = neutron absorption cross section of the absorber averaged over the energy group g.

2.4 **The Multiplication Factor**

In nuclear reactors the fission of a nucleus results from neutron absorption. This fission is accompanied with the emission of ν neutrons. These neutrons, in turn, may induce additional fissions, and thus, produce new neutrons. However, each neutron does not produce fission. It may be absorbed either in a non-fissile or in a fissile nucleus without fission (fission probability after neutron capture by a fissile nucleus is never 100%). A neutron created in a medium (which we first consider infinite) with fissile nucleus will, thus, give birth to infinite multiplication factor, K_∞ second generation neutrons. The number of neutrons generation is the result of a

neutron producing nuclear reaction which can be a fission or, more rarely, a (n, xn) reaction. The total number of neutrons following the apparition of a neutron in the multiplying medium will be (Glasstone and Sesonke, 1981):

$$n_{chain} = 1 + K_\infty + K_\infty^2 + \ldots\ldots + K_\infty^n \ldots\ldots = \frac{1}{1+K_\infty} \qquad \text{- (2.17)}$$

The total number of neutrons created in the medium per source neutron is simply $K_\infty n_{chain}$. One defines a neutronic "gain" as the ratio of the total number of neutrons (source + created) to the numberof source neutrons. This gain is then $\frac{1}{1-K_\infty}$. Since all neutrons are, ultimately, absorbed, the number of absorption reaction is, thus, $n_{reac} = n_{chain}$. For finite media one has to replace K_∞ by an effective multiplication factor,K_{eff}which is less than K_∞due to neutrons escaping from the system. If $K_{eff} > 1$, the reaction diverges, i.e. from one initial neutron one obtains a final number of neutrons going to infinity. A controlled divergence allows starting a reactor. When uncontrolled it leads to a criticality accident.The state of a fission chain reaction can be concisely summarized by the *effective multiplication factor*, *k*, which indicates the change in total number of fission events during successive generations of the chain reaction. It is defined as:

$$k = \frac{\text{Total number of fission events in a given generation}}{\text{total number of fission events in the previous generation}} \qquad \text{- - - - - (2.18)}$$

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

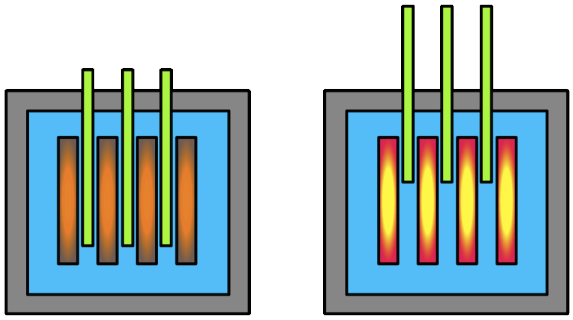

Fig 1.3: Representative diagram of control rod usage. The left image shows the control rods (green) inserted fully into the reactor core, putting the reactor in a subcritical state. In the right image, the control rods are removed, allowing more neutrons to accelerate the fission chain reaction and go supercritical. (Source: Wikimedia Commons 2011)

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

Control rods thereby find their use as an effective method for combating these time-dependent changes in reactors. Control rods are essentially a highly effective neutron-absorbing mechanical structure, which can be actively inserted or withdrawn from the reactor core while the fission process is occurring. By controlling the portion of the control rod that interacts with the fission reaction, the multiplication factor can be finely tuned to maintain reactor criticality (see Fig. 3). In addition, control rods can be used to intentionally make rapid changes to the reactor state (i.e.

turning the reactor on and off), especially as an emergency shut off feature by fully inserting the rods.When $K_{eff} = 1, one$ obtains a critical reactor. The possibility to keep precisely the condition $K_{eff} = 1$ is due to the presence of a small fraction of delayed neutrons which allow correction of deviation of the criticality coefficient K_{eff} from unity. If $K_{eff} < 1$, an incident neutron gives birth to a finite number of secondary neutrons. The medium is said to be multiplying and the multiplication factor is $\frac{1}{1-K_{eff}}$.

2.5 **Control of Reactivity**

To operate a nuclear reactor, we want most of the time to keep K_{eff}= 1 (ρ = 0) so that we will have a steady reactor.When we want to reduce power or shut the reactor down, we need ways to make K_{eff}< 1 (i.e., insert a negative reactivity). This is usually done by inserting rods or devices made of strong neutron absorbers such as boron, cadmium, or gadolinium (Duderstadt et al, 1976).

When we are starting the reactor up and want to increase power, we need to make K_{eff} slightly greater than 1 (i.e., insert positive reactivity) for a short time.

The safe and stable reactor operation can be guaranteed implementing a proper reactivity control. This task includes several aspects: (i) the mechanical systems, such as control elements, that provide adjustment of the power level and shut down the reactor; (ii) the processes of fuel depletion and fission product poisoning that require positive counteraction; (iii) "built-in" a factor, such as burnable poisons, that can ensure a long – term control reactivity.

2.6 **Excess Reactivity**

The core reactivity present with all control elements withdrawn from the core is named excess reactivity, ρ_{ex}. $\rho_{ex}$is a function of both time (due to fuel depletion and nuclide transmutation) and temperature (due to reactivity feedback). Larger values of ρ_{ex} may generally imply longer corelifetimes, but at the expense of larger control requirements and poorer neutron economy (since with more control reactivity in the core, there will be more neutron absorption).

2.7 **Shutdown Margin**

The negative reactivity of the core present when all control elements have been fully inserted to achieve minimum core multiplication is called shutdown margin. The shutdown margin ρ_{sm} is afunction of time and temperature. For example, the shutdown margin for a core loaded with fresh fuel at ambient temperature in which no depletion or fission product buildup has occurred will be quite different from the shutdown margin characterizing a core that has been operating at power for some time. Typically shutdown margins are chosen such that the core multiplication is below critical. The shutdown margin not only characterizes the core multiplication in its shutdown state, but is also related to the rate at which the reactor power level may be reduced in an emergency shutdown or "scram." The fractional power level decrease achieved immediately after control insertion is given approximately by:

$$\frac{P_o}{P_1}=\frac{\beta-\rho_o}{\beta-\rho_1}=\frac{\beta}{\rho_{sm}+\beta} \qquad \text{----------} \quad (2.19)$$

where: P_0 – power before control rod insertion; P1 – power after control rod insertion;

β -effective fraction of the delayed neutrons; ρ_{sm}- shutdown margin.

2.8 **Control Rod Worth**

The rod worth $\Delta\rho$ is defined as the difference between the excess reactivity and the minimum reactivity when all control elements are fully inserted.

That is, $\Delta\rho = \rho_{ex} + \rho_{sm}$ ---------- (2.20)

where ρ_{ex} is the excess reactivity

ρ_{sm} is the shutdown margin

2.9 Overview of Absorbing Materials Used For Nuclear Control

Chemical elements with a sufficiently high capture cross section for neutrons include silver, indium and cadmium. Other elements that can be used include boron, cobalt, hafnium, gadolinium, and europium. Because these elements have different capture cross sections for neutrons of varying energies the compositions of the control rods must be designed for the neutron spectrum of the reactor it is supposed to control. Light water reactors (BWR, PWR) operate with "thermal" neutrons, breeder reactors with "fast" neutrons.

2.9.1 Cadmium – Thermal Absorber

Cadmium is a soft, malleable, ductile, toxic, bluish-white bivalent metal which can be easily cut with a knife. It is similar in many respects to zinc but reacts to form more complex compounds. Naturally occurring cadmium is composed of 8 isotopes. For two of them, natural radioactivity was observed, and three others are predicted to be radioactive but their decays were never observed, due to extremely long half-life times. The two natural radioactive isotopes are ^{113}Cd (beta decay, half-life is 7.7 X 10^{15} years) and ^{116}Cd (two-neutrino double beta decay, half-life is2.9 x 10^{19}years). The other three are ^{106}Cd, ^{108}Cd (double electron capture), and ^{114}Cd (double beta decay); only lower limits on their half-life times have been set. At least three

isotopes -^{110}Cd, ^{111}Cd, and ^{112}Cd - are absolutely stable. Among the isotopes absent in the natural cadmium, the most long-lived are ^{109}Cd with a half-life of 462.6 days, and ^{115}Cd with a half-life of 53.46 hours. All of the remaining radioactive isotopes have half-lives that are less than 2.5hours and the majority of these have half-lives that are less than 5 minutes. This element also has 8 known metal states with the most stable being ^{113m}Cd (t½ 14.1 years), ^{115m}Cd (t½ 44.6 days) and ^{117m}Cd (t½ 3.36 hours).

Cadmium has one dominant isotope - ^{113}Cd, which has big thermal absorption cross section ~10^4-10^5barns. All of the Cd – isotopes including ^{113}Cd have resonance (~10^3- 10^4 barns) in the epithermal region: 10^{-5} MeV - 10^{-2} MeV.

2.9.2 **Hafnium – Thermal and Resonant Absorber**

Hafnium is a shiny silvery, ductile metal that is corrosion resistant and chemically similar to zirconium. Hafnium is estimated to make up about 0.00058% of the Earth's upper crust by weight. It is found combined in natural zirconium compounds but it does not exist as a free element in nature. Minerals that contain zirconium, such as alvite (Hf, Th, Zr) SiO_4H_2O, thortveitite and zircon ($ZrSiO_4$), usually contain between 1 and 5% hafnium. Hafnium and zirconium have nearly identical chemistry, which makes the two difficult to separate. About half of all hafnium metal manufactured is produced as a by-product of zirconium refinement. This is done through reducing hafnium (IV) chloride with magnesium or sodium in the Kroll process. The most important isotopes are ^{177}Hf, ^{178}Hf and ^{176}Hf. The main isotope which is burning out during irradiation is ^{177}Hf. By contrast with Cd, the Hf- isotopes have their absorption mainly in the epithermal region (10^{-6}-10^{-2} MeV), which is characterized with many resonances ~10^3 – 10^5 barns, the thermal cross section varies as ~ 10^3 barns.

2.9.3 **Boron (Boron Carbide)**

Boron carbide (chemical formula B_4C) is an extremely hard ceramic material used in tank arm or,bullet-proof vests, and numerous industrial applications. With a hardness of 9.3 on the moths scale, it is the fifth hardest material known behind boron nitride, diamond, ultra hard fullerite, and aggregated diamond nanorods. Discovered in the 19th century as a by-product of reactions involving metal borides, it was not until the 1930s that the material was studied scientifically. Its ability to absorb neutrons without forming long lived radionuclides makes the material attractive as an absorbent for neutron radiation arising in nuclear power plants. Nuclear applications of boron carbide include shielding, control rod and shut down pellets.

2.9.4 **Gadolinium (Gd_2O_3)**

Gadolinium is a silvery white, malleable and ductile rare earth metal with a metallic luster. It crystallizes in hexagonal, close-packed alpha form at room temperature; when heated to 1508 K, it transforms into its beta form, which has a body-centered cubic structure. Unlike other rare earth elements, gadolinium is relatively stable in dry air; however, it tarnishes quickly in moist air and forms a loosely adhering oxide that spalls off and exposes more surfaces to oxidation. Gadolinium reacts slowly with water and is soluble in dilute acid. Gadolinium has the highest thermal neutron capture cross-section of any (known) element, 49,000 barns, but it also has a fast burn-out rate, limiting its usefulness as a nuclear control rod material. Gadolinium is used in nuclear marine propulsion systems as a burnable poison. The gadolinium slows the initial reaction rate, but as it decays other neutron poisons accumulate, allowing for long-running cores.

Naturally occurring gadolinium is composed of 5 stable isotopes, ^{154}Gd, ^{155}Gd, ^{156}Gd,^{157}Gd and ^{158}Gd, and 2 radioisotopes, ^{152}Gd and ^{160}Gd, with ^{158}Gd being the most abundant(24.84% natural abundance). 30 radioisotopes have been characterized with the most stable been ^{160}Gd with a half-life of more than 1.3×10^{21} years (the decay is not observed, only the lower limit on the half-life is known), alpha-decaying ^{152}Gd with a half-life of 1.08×10^{14}years, and ^{150}Gd with a half-life of 1.79×10^{6} years. All of the remaining radioactive isotopes have half lives that are less than 74.7 years, and the majority of these have half-lives that are less than 24.6seconds. This element also has 4 meta states with the most stable being ^{143m}Gd (t½ 110 seconds),^{145m}Gd (t½ 85 seconds) and ^{141m}Gd (t½ 24.5 seconds).The primary decay mode before the most abundant stable isotope, ^{158}Gd, is electron capture and the primary mode after is beta minus decay. The primary decay products before^{158}Gd are element Eu (Europium) isotopes and the primary products after are element Tb (Terbium) isotopes. The Gd isotopes are characterized with high thermal cross section (^{155}Gd, ^{157}Gd~105 barns) and also epithermal, ~ 10^{4} barns.

2.9.5 **Europium (Eu_2O_3)**

Europium is never found in nature as a free element; however, there are many minerals containing europium, with the most important sources being bastnasite and monazite. Europium has also been identified in the spectra of the sun and certain stars. Relative depletion or enrichment of europium in minerals relative to other rare earth elements is known as the europium anomaly. Naturally occurring europium is composed of 2 stable isotopes, ^{151}Eu and ^{153}Eu, with^{153}Eu being the most abundant (52.2% natural abundance). 35 radioisotopes have been characterized, with the most stable being ^{150}Eu with a half-life of 36.9 years, ^{152}Eu with a half

life of 13.516 years, and ^{154}Eu with a half-life of 8.593 years. All of the remaining radioactive isotopes have half-lives that are less than 4.7612 years, and the majority of these have half-lives that are less than 12.2 seconds. This element also has 8 metal states, with the most stable being^{150m}Eu (t½ 12.8 hours), ^{152m}1Eu (t½ 9.3116 hours) and ^{152m}2Eu (t½ 96 minutes).

The primary decay mode before the most abundant stable isotope, ^{153}Eu, is electron capture and the primary mode after is beta minus decay. The primary decay products before^{153}Eu are element Sm (samarium) isotopes and the primary products after are element Gd (gadolinium) isotopes. All Eu – isotopes have very big thermal cross section (~ 10^5 barns) and also epithermal cross section (~ 10^4 barns).

2.10 **Depletion / Burnup Analysis**

The study of the interaction of the core power distribution with the time – dependent production of depletion of nuclei in the core is known as depletion or burnup analysis. Burnup analysis is closely related to the topic of nuclei fuel management in which one tries to optimize the fuel loading, arrangement, and reloading in order to achieve the most economical power generation within the design constraints. During reactor operation the fuel composition will change as fissile isotopes are consumed and fissile products are produced. The nuclear designer must monitor these processes over core life in an effort to ascertain fuel composition and reactivity as a function of energy removal. This requires studying the depletion and production chains for the principal isotopes (e.g ^{235}U – ^{238}U or ^{233}U – ^{232}Th) coupled with the equations determining the neutron flux in the core. Fission products with large capture cross sections and fission yields namely ^{135}Xe and ^{195}Sm are usually treated explicitly; the remaining fission products are lumped into one or several groups, each characterized by an effective cross section.

Furthermore, any nuclei with a short half – life are omitted from the burnup calculation. The calculation of the core multiplication and power distribution must be made times over operating lifetime of the core composition changes. The fuel burnup or lifetime can be limited by either reactivity considerations (i.e the multiplication of the core drops too low for further power production) or by limitation on materials stability under high radiation influences.

2.11 **Energy Dependence on Neutron Cross Sections**

Recall that neutrons cross sections depend not only on the nature of the target nucleus but also on the energy of the interacting neutron. It is convenient to classify neutrons that are involved in nuclear reactions according to the general behavior of the various cross sections and to divide the neutron energies into several regions to take account of these general trends.

1. A high-energy region which comprises neutron energies between 10MeV to 0.1MeV. Neutrons within this range will be called "FAST NEUTRONS".
2. Energies between 0.1MeV and 1000eV. Neutrons within this range will be termed "INTERMEDIATE NEUTRONS".
3. Energies between 1000eV and 1eV. Neutrons within this range will be called "EPITHERMAL NEUTRONS".
4. Energies between 1eV and less. Neutrons within this range will be called "THERMAL OR SLOW NEUTRONS".

Table 2.1 Neutron Cross-Sections for Cadmium Isotopes (Allison Gicking, 2012)

Cadmium Isotopes	Thermal (Barns)	Epithermal (Barns)	Product Isotopes
Cd-106	1.00	-	Cd-107
Cd-108	1.10	4.29	Cd-109

Cd-110	0.10	2.00	Cd-111m
Cd-114	0.036	3.16	Cd-115m
Cd-114	0.22	8.7	Cd-115
Cd-116	0.05	-	Cd-117
Cd-116	0.027	0.422	Cd-117m

1Barn= $10^{-24}cm^2$(Lamarsh, 1972)

2.12 **Review of Previous Work**

Research reactors have been adequately utilized for research and development. These include cross section measurements (Jonah et al, 2008), Neutron Activation Analysis (Jonah et al, 2005, 2006), radioisotope production, Neutron therapy etc. Due to the desire to improve the utilization of research reactors, improve safety of reactors (power and research reactors) and nuclear proliferation issues especially with reactors using enriched ^{235}U, several works have been done in the area of reactor safety, reactor design and benchmark studies for conversion from High Enriched Uranium (HEU) to Low Enriched Uranium(LEU).

Knowledge of the efficiency of a control rod to absorb excess reactivity in a nuclear reactor i.e., knowledge of its reactivity worth is important from many points of view. These include the analysis and the assessment of the shutdown margin of new core configurations (upgrading, conversion, refueling, etc.) as well as several operational needs, such as calibration of the control rod worth can be assessed either experimentally or theoretically through utilization of neutronic codes.

Nuclear reactor safety is a subject of major interest in the field of nuclear science/engineering. Thus, several works have been reported concerning the assessment of a

control rod worth. Varvayanni et al (2009) calculated the control rod worth of the Greek Research Reactor (GRR-1) using deterministic and stochastic methods. The capability of the deterministic code system to reliably calculate a control rod worth was examined with respect to various options adopted for its application. GRR-1 has five (5) safety rods, the most significant depletion was found to be that of the Cd113 isotope which was found to be reduced to a fraction of 0.03 of its nominal value for rod1 and 0.09 for rod4 , 0.90 for rod2 and 0.97 for rod3, the total rod worth was found to be dropped by about 6.5%. The rod worths were classified as rod2=76.8rod1, rod3=75.9rod1, rod4=78.8rod1 and rod5=73.3rod1. For both simulations the nominal densities of the rods absorbers are used. The Monte Carlo calculations produced very reasonable results for the control rod's worth, compared satisfactorily with measurements and within an expected range of discrepancies if the absorber's depletion is taken into account. The deterministic results appear to be quite sensitive to the presence of the other rods and the consequent shadowing effect. The integral rod worth is produced more satisfactorily when the rod displacement take place relatively away from the other rods, while a gradual underestimation occurs when the rod is found within the area of influence of a neighboring strong rod. In another related work,Varvayanni (2009) estimated the irradiated control rod worth of the Greek research reactor with five neutron absorbers. They suggested a method for predicting the accurate assessment of its absorbing capacity in a new core environment, that is, based on measurements of its integral worth conducted during the rod's operation in its former core environment, and on corresponding calculations performed using the nominal absorber concentration in the rod. An incremental determination of the neutron absorption macroscopic cross section in user-defined horizontal layers of the rod was performed, the calculated total rod worth for a progressively varying total macroscopic absorption cross section of an effective absorber using the above

methodology almost estimates the discrepancy with measurements compared to the computations made with the absorption cross section corresponding to the nominal absorber densities. This makes the rod worth assessment in case of planned core modifications, more realistic.

In another work by Fadaei (2009) the control rod worth for VVER-1000 (Vodo-Vodyanoi Energetichesky in Russian or WWER – Water-Water Energetic Reactor) nuclear reactor was calculated using WIMS (Winfrith Improved Multigroup Scheme) and CITATION codes. Dynamic analysis for the control rod worth calculation was performed by preparing a software with FORTRAN (Formula Translator) program for coupling neutronic codes and comparing between the results and reference, it was discerned that manner of fuel assembly modeling and transport equation solving have important effect on calculation result. They showed clearly that with increasing anisotropic distribution of rods in fuel assembly due to insertion of more control rods into the core, the effect increased. Another similar work was done by Titouche (2003) using strong absorber nuclear data for diffusion codes to calculate the control rod worth of NUR (Nuclear Uranium Reactor) reactor. The calculation procedure starts with a reference flux and reaction rate distribution calculation in the control rod. Then equivalent diffusion parameters were applied to all situations in which the control rod was used, the fewgroup parameters are limited to the absorber itself and diffusion theory is assumed to become valid immediately outside the rod. The transport code ANISN (Is a one dimensional discrete ordinates transport code system with Anistropic scattering which solves the 1-D multigroup time independent transport equation by method of discrete ordinates) from MTR-Pc package was used to perform transport calculations, α and β coefficients were determined using the calculated currents fluxes. The lattice code WIMSD/4 were used to generate fewgroup cell cross section data and diffusion coefficients, the diffusion theory worth calculations using this method was found to be in good

agreement with experimental measurements. The control rod worth for five (5) safety rods were calculated as 5.9mk, 5.5mk, 3.9mk, 4.3mk and 0.98mk However, use of Monte Carlo code such as MCNP – Monte Carlo N-Particle transport code was recommended.

In order to calculate some neutronic parameters, Jonah, 2007 calculated the cold core excess reactivity of the Nigeria research reactor (NIRR-1) in his work "Monte Carlo simulation of core Physics parameters of the Nigeria research reactor (NIRR-1)".The cold core excess reactivity was calculated to be (4.76 ±0.07) mk, the SDM and control rod worth was obtained using the MCNP as 4.07 & 7.61mk respectively which was in good agreement with measured data. MCNP deck was used to perform the calibrations of the control rod worth as well as the worth of top Be shims, the calculated data for the shim and control rod were used to plot the graph of the integral worth of the control rod data against the rod position in the active fuel region. In another related work, Ahmed et al (2010) examined the effects of core excess reactivity and coolant average temperature on maximum operable time of NIRR-1 miniature neutron source reactor, using the zero power experimental results a graph of measured cold core excess reactivity versus control rod critical depth of insertion for NIRR-1 was plotted with 3.77mk excess reactivity in 2004 with control rod position of 11.8cm. By January 2009, the rod position was found to be 8.8cm which by evaluation resulted to an excess reactivity of 2.80mk, this result is less than the reactor's initial excess reactivity of 3.77mk in 2004 and far less than the internationally accepted value of 3.5 – 4.0 mk for MNSR like NIRR-1 (Zhu, 1990). It therefore means that if the cadmium sheet inserted into one of the unconnected inner irradiation channel is pulled out, its excess reactivity of 1.16mk will be introduced into the system. This will bring NIRR-1 to the required 3.5 – 4.0 mk excess reactivity margin.

In an effort to reduce the blockage probability in the interiors of safety rods due to the continuous movement of the rods up and down in their sites, a new safety shim was designed by selecting absorber thinner and the cladding thicker at the same time (Shoushtari et al, 2009), using Monte Carlo simulation of a research reactor with nominal power of 7MW. They employed detailed model of the reactor core including standard and control elements, reflectors, irradiation channels, control rods, reactor pool and thermal column. The following physical parameters of reactor core were calculated for the present low enriched uranium (LEU) core: core reactivity, control rod worth, thermal and epithermal neutron flux distributions, shutdown margin and delayed neutron fraction. Reduction of unfavorable effects of blockage probability of control safety rod in their interiors because of not enough space in their sites, and lack of suitable capabilities to fabricate very thin plates for control safety rod cladding was also studied. It was discovered that making the absorber rod thinner and control safety rod cladding thicker by introducing a better blackness absorbing material and a new stainless steel alloy respectively are ways to reduce the effects of the unfavorable blockage probability of the control safety rod. The total control rod worth was calculated using the expression

$$\text{CR worth } \% = \frac{K_{eff(out)} - K_{eff(in)}}{K_{eff(out)}} \quad \text{----------} \quad (2.21)$$

$K_{eff(out)} and K_{eff(in)} are the multiplication factors when the control rods are fully out and in the core.$

The total control rod worth was calculated as 14.11% with control rod thickness of 2.6mm and fully inserted..These results indicate that the new introduced absorber rod makes a better situation for core to reach critical or sub critical position and suitable for the present reactor core. So it is feasible to make the new absorber rod thinner so that an adequate blackness (equal to the blackness of the present rods) for shutdown is attained. In a similar work, Kachelva (2008) used improved Monte Carlo-Pertubation method to estimate the control rod worth in a research

reactor. perturbation method was used to obtain the equation for the relative efficiency of control rod insertion. A series of coefficients describing the axial absorption profile was used to correct the equation for a composite rod having a complicated burnup irradiation history. These coefficients were derived from the macroscopic absorption cross-section obtained from detailed Monte Carlo calculations by MCNPX2.6F of the axial burnup profile during control rod life. This method suits well for the estimation of the control rod worth for the simple case of a fresh rod which is composed of one single absorber material, but it is too approximate for a composite or burnup rod. Sembiring et al (2008) in his work"Validation of the Monte Carlo code MVP on the first criticality of Indonesian multipurpose reactor", they investigated that the movable control rods (absorber blades) were modeled as their exact geometry and dimensions. Consequently, 60cm water layer above the core had to be included in the calculation to provide enough space for the absorber blades when a control rod was fully withdrawn. Approximately 30cm water layers were included below the core bottom support and around the beryllium block and element reflectors.

The total control rod worth was evaluated based on the experimental results; the total control rod worth was 17.80% ΔK/K. The value was obtained by summation of single control rod worth with shim rod compensation method.

To assess the effect of fuel burnup on control rod, Savva (2010), two different approaches a deterministic and stochastic one were used for the determination of the rods worth dependence on the fuel burnup level and Xe concentration level in a conceptual symmetric reactor core based on the MTR (Materials testing reactor) fuel assemblies used in Greek research reactor. For the deterministic approach the neutronics code system composed by the SCALE (Standardized Computer Analyses for Licensing Evaluation) modules NITAWL and XSDRN and the diffusion

code CITATION were used, while for the stochastic one the Monte Carlo code TRIPOLI (Is a 3D general purpose continuous energy Monte carlo transport code) was applied. Their result shows that when the Xe is present in the core, the rod's worth is significantly reduced while the rod worth variation with increasing burnup depends on the rod's position in the core grid. The rod worth obtained with the use of the Monte Carlo code is higher than the one obtained from the deterministic code. In another work, Boafo (2013) assessed the effect of fuel burnup on control rod worth for HEU and LEU cores of Ghana Research Reactor-1 (GHARR-1). In their study, an assessment was made of the effect of fuel burnup on the control rod worth for GHARR-1 by simulating its MCNP5 model for fresh and irradiated cores. Firstly, fuel burnup was calculated for both HEU and LEU cores using the deterministic code BURNPRO, the results were then used to modify the MCNP5 model of GHARR-1. The modified deck was then stimulated in order to calculate the control rod worth as well as determine the thermal neutron flux in both inner and outer irradiation sites. The results showed that the control rod worth increased from 6.4mk to 7.44mk for the HEU after 1.16% burnup of ^{235}U. In the case of the LEU however, there was a reduction from 7.61mk to 6.98mk after 0.72% burnup, this reduction in the control rod worth can be attributed to the presence of ^{238}U responsible for the capture of neutrons in the resonance region. The resultant effect of the resonance capture by ^{238}U is reduced thermal neutron flux which leads to reduced control rod worth. Another work was by Taryo (2003) using Verified WIMS-D4 and CITATION codes. In the case of control rod interaction in a conceptual MTR –type reactor equipped with 2 absorber blades and 2 pairs of absorber blades, the result indicated that the shadowing effects in the MTR-type reactor vary depending on the extent of the absorber blades insertion into the reactor core and the effect is certainly more dominant for the case of 2 pairs of absorber blades rather than 2 absorber blades. In their analysis, RSG-GAS

containing 250g ^{235}U of U_3O_8 -Alx fuels has been employed. All calculation results were carried out using the combination of WIMS-D4 and CITATION-3D codes. The reactivity worth of rods G6, F8, F5, E9, D4, C8, C5 and B7 are 1.18%, 1.41%, 1.44%, 1.24%, 1.31%, 1.31%, 1.39% and 1.13% (ΔK/K) respectively. It can be estimated that the total worth of all the collective RSG-GAS rods is 14.328% ΔK/K, while the total worth of all single rods is 10.393% ΔK/K. The interaction term with all RSG-Gas rods fully down is 37.90% representing an anti-shadowing effect. The value 37.90% indicated that the reactivity worth of each rod augments 37.90% over it's the original reactivity worth. In a related work by Asuku (2015) on the application of positive period method in the calibration and determination of intergral worth of NIRR-1, the control rod was calibrated from 107mm to 177mm withdrawal length at a low flux of 5 x 10^9ncm^{-2}s. Doubling times measurements were carried out in conjunction with the 'inhour' data to obtain reactor periods and their corresponding reactivities. The integral rod worth of (3.34 ± 0.11) mk obtained was compared with the in-built full rod length total reactivity worth of 7mk and was found to have accounted for 47.8% of it, this result is an indication that 10mm withdrawal length is approximately 0.488mk average reactivity insertion and the length used here is the most active portion of the control rod. It also shows that only 30% of the control rod length is required to achieve the characteristic curve and also to achieve shutdown using the positive period method. This has satisfied the safety requirements of MNSR for shutdown margin. In another related work by Babangida (2016) on the Burn-up calculations of NIRR-1 and the estimation of the core life time expectancy after 10years using the codes WIMS and CITATION. The burn-up analysis carried out have indicated that the excess reactivity of NIRR-1 follows a linear decreasing trend having 216 Effective Full Power Days (EFPD) operations and the reactivity worth of top beryllium shim data plates was calculated to be 19.072mk. the result of depletion analysis for

NIRR-1 core shows that 7.99g ± 0.0008g of U-235 was consumed for the period of 12 years of operating time. The production of the build-up of Pu-239 was found to be 0.0347g ± 0.00434g. The core life time estimated in this research was found to be 30.33 years. This is in good agreement with the literature.

2.13 **Burnup**

Nuclear fuel is removed from a reactor every few years when it can no longer economically keep a chain reaction going. This "spent" fuel remains radioactive and must be managed. At first, it goes into a pool onsite for cooling and storage. These casks are specially designed to contain the radioactivity and allow hot spent fuel to cool further.

To understand "burnup," it helps to know more about the uranium that fuels a reactor. Before it is made into fuel, uranium is processed to increase the concentration of atoms that can split in a controlled chain reaction in the reactor. The atoms release energy as they split. This energy produces the heat that is turned into electricity. In general, the higher the concentration of those atoms, the longer the fuel can sustain a chain reaction. And the longer the fuel remains in the reactor, the higher the burnup.

In other words, burnup is a way to measure how much uranium is burned in the reactor. It is the amount of energy produced by the uranium. In nuclear power technology, burnup (also known as fuel utilization) is a measure of how much energy is extracted from a primary nuclear fuel source. It is measured both as the fraction of fuel atoms that underwent fission in %FIMA (fissions per initial metal atom) and as the actual energy released per mass of initial fuel in gigawatt-days/metric tonof heavy metal (GWd/tHM), or gigawatt-days per metric ton of uranium (GWd/MTU). Average burnup, around 35 GWd/MTU two decades ago, is over

45 GWd/MTU today. Utilities now are able to get more power out of their fuel before replacing it. This means they can operate longer between refueling outages. It also means they use less fuel.

The burnup level affects the fuel's temperature, radioactivity and physical makeup. It is important to review spent fuel cask designs because each system has limits on temperature and radioactivity. How hot and how radioactive spent fuel is depends on burnup, as well as the fuel's initial makeup and conditions in the core. All these factors must be taken into account in designing and approving dry storage and transport systems for spent fuel.

Nuclear fuel is encased in metal cladding. In the reactor, this cladding reacts with cooling water. The reaction forms oxide on the outside (similar to rust) and releases hydrogen. These processes begin slowly, then start to accelerate as the fuel reaches burnup of 45 GWd/MTU. Anything higher is considered high burnup. But in reality there is no sharp line between low and high burnup. It is a continuum. That means the difference between fuel burned to 45 GWd/MTU and 46 or 47 GWd/MTU can be very small.

When spent fuel is placed in a dry storage system and the water is removed, the temperature of the fuel increases and the makeup of the cladding can change. This change can result in the fuel cladding becoming less "ductile," or pliable, as it cools. It was also once thought that cladding of higher burnup fuel could become brittle enough to create a safety concern. Research now shows that while it may become less ductile, safety of the public will not be impacted for the systems (Walters, 1998)

The actual fuel may be any actinide that can support a chain reaction, including uranium, plutonium, and more exotic transuranic fuels. This fuel content is often referred to as the *heavy metal* to distinguish it from other metals present in the fuel, such as those used

for cladding. The heavy metal is typically present as either metal or oxide, but other compounds such as carbides or other salts are possible.

In a power station, high fuel burnup is desirable for:

i. Reducing downtime for refueling
ii. Reducing the number of fresh nuclear fuel elements required and spent nuclear fuel elements generated while producing a given amount of energy
iii. Reducing the potential for diversion of plutonium from spent fuel for use in nuclear weapons

It is also desirable that burnup should be as uniform as possible both within individual fuel elements and from one element to another within a fuel charge. In reactors with online refuelling, fuel elements can be repositioned during operation to help achieve this. In reactors without this facility, fine positioning of control rods to balance reactivity within the core, and repositioning of remaining fuel during shutdowns in which only part of the fuel charge is replaced may be used.

In once-through nuclear fuel cycles such as are currently in use in much of the world, used fuel elements are disposed of whole as high level nuclear waste, and the remaining uranium and plutonium content is lost. Higher burnup allows more of the fissile ^{235}U and of the plutonium bred from the ^{238}U to be utilised, reducing the uranium requirements of the fuel cycle.

In once-through nuclear fuel cycles, higher burnup reduces the number of elements that need to be buried. However, short-term heat emission, one deep geological repositorylimiting factor, is predominantly from medium-lived fission products, particularly ^{137}Cs (30.08 year half life) and ^{90}Sr (28.9 year half life). As there are proportionately more of these in high-burnup fuel, the heat generated by the spent fuel is roughly constant for a given amount of energy generated.

used fuel (*Etienne Parent,2003)*

2.14 Burnup Measurement

Burnup is simple: if 5% of the initial heavy metal atoms have undergone fission, the burnup is 5%. In reactor operations, this percentage is difficult to measure, so the alternative definition is preferred. This can be computed by multiplying the thermal power of the plant by the time of operation and dividing by the mass of the initial fuel loading. For example, if a 3000 MW thermal (equivalent to 1000 MW electric) plant uses 24 tonnes of enriched uranium (tU) and operates at full power for 1 year, the average burnup of the fuel is

$$\frac{3000\ \text{MW x 365 days}}{24\ \text{metric tonnes}} = 45.63\ \text{GWd/t} \qquad \text{----------} \quad (2.22)$$

or 45,625 MWd/tHM (where HM stands for heavy metal, meaning actinides like thorium, uranium, plutonium).Converting between percent and energy/mass requires knowledge of κ, the thermal energy released per fission event. A typical value is 193.7 MeV (3.1×10^{-11} J) of thermal energy per fission. With this value, the maximum burnup of 100%, which includes fissioning not just fissile content but also the other fissionable nuclides, is equivalent to about 909 GWd/t. Nuclear engineers often use this to roughly approximate 10% burnup as just less than 100 GWd/t.

2.15 Proliferation

Burnup is one of the key factors determining the isotopic composition of spent nuclear fuel, the others being its initial composition and the neutron spectrum of the reactor. Very low fuel burnup is essential for the production of weapons-grade plutonium for nuclear weapons, in order to produce plutonium that is predominantly ^{239}Pu with the smallest possible proportion of ^{240}Pu and ^{242}Pu.

Plutonium and other transuranic isotopes are produced from uranium by neutron absorption during reactor operation. While it is possible in principle to remove plutonium from used fuel and divert it to weapons usage, in practice there are formidable obstacles to doing so. First, fission products must be removed. Second, plutonium must be separated from other actinides. Third, fissionable isotopes of plutonium must be separated from non-fissionable isotopes, which is more difficult than separating fissionable from non-fissionable isotopes of uranium, not least because the mass difference is one atomic unit instead of three. All processes require operation on strongly radioactive materials. Since there are many simpler ways to make nuclear weapons, nobody has constructed weapons from used civilian electric power reactor fuel, and it is likely that nobody ever will do so. Furthermore, most plutonium produced during operation is fissioned. To the extent that fuel is reprocessed on-site, as proposed for the Integral Fast Reactor, opportunities for diversion are further limited. Therefore, production of plutonium during civilian electric power reactor operation is not a significant problem.

The choice of the research work is necessitated due to the fact that all the review literatures with respect to NIRR-1, routine work on the reactor and other maintenance carried out from 2004 to 2007 do not re-evaluate the control rod worth. Moreso, reactor experts have recommended a periodic assessment of the control rod worth to ascertain the integrity of the control rod to perform its intended safety function.

CHAPTER THREE

MATERIALS AND METHOD

3.1 Methodology

There are several methods for assessing the depletion of a reactor control rod. These include Deterministic and Stochastic method, WIMS and CITATION Codes, Strong Absorber nuclear data for diffusion codes, Improved Monte Carlo Pertubation method, Monte Carlo Codes, e.t.c. In this work we reconstructed the operational history of NIRR-1 including different grid positions occupied by the rod and varying insertions during operation. Typically, NIRR-1 is operated at three different power levels and the reactor is operated three times a week for 48 weeks in a year. The reactor is shut down for about four weeks in a year for annual maintenance.

During the reactor operation the control panels are observed. Safety regulations regarding reactor operations include watching the entrance and exit from the reactor hall for unauthorized persons, observing or listening for alarms and announcing over the loudspeaker changes in reactor operations are also observed.

3.2 Operational History of NIRR-1

NIRR-1 is currently operated at different power level and fluxes as shown below

- Full power(31Kw) and $1x10^{12}n/cm^2s$
- Half power(15.5Kw) and $5x10^{11}n/cm^2s$
- Low power(6Kw) and $3x10^{9}n/cm^2s$

The reactor has also been used for either short irradiations (6 hours) or long irradiations (3 hours). The short irradiations are carried out for the determination of short lived radionuclides whiles the long irradiations for long lived radionuclides.

Table 3.1 Summary of NIRR-1 operational regime

NIRR-1	Weekly	Yearly(48 weeks)
Operation	3 times	144 times
Periodic maintenance	1	48 times
Annual maintenance	-	4 weeks

3.3 Data Collection

The data were retrieved from the reactor computer system which was originally designed to record and store operational information. The retrieved data is in the form of a log book containing the date of the operation and all physics parameters. It includes the time (hr) of operation, neutron flux (n/cm^{2}s), rod height (mm), temperature (^{o}C for inlet, outlet and pool), reactivity (mK) and the radiation (uSv/Hr) as shown in table 3.2.

3.4 Data Organization, Analysis and Presentation

The reactor physics data were retrieved and transferred to an excel spread sheet and analyzed. It was noticed that there were times when the reactor misbehaved, in such times the data are not coherent with known previous pattern. These operating regimes were sorted out and re-organized.

The analysis is in the form of collecting all the fluxes for each operation and the corresponding time for each operation from when the reactor became critical up to the period when the reactor was modified was considered (3rd Feb. 2004 to 30th March 2011). The

modification involved the removal of cadmium poison (acting as shim) of worth 1.2mK. Addition of about 1.2mK to the reactivity constitutes a major modification to the NIRR-1.

3.5 **Determination of Fractional Change in Cadmium Atom**

The absorber's depletion during a time period ΔT, can be estimated by integrating equation (2.16) above, assuming that during ΔT, the neutron flux in the vicinity of the rod is constant.

$$\int_{N_j^o}^{N^{\Delta T}} \frac{dN_j}{N_j} = -\left(\sum_{g=1}^{G} \sigma_{jg}\Phi_g\right)\Delta T \qquad \text{----------(3.1)}$$

Integrating LHS of (3.1) gives

$$\log_e \frac{N^{\Delta T}}{N_j^o} = -\left(\sum_{g=1}^{G} \sigma_{jg}\Phi_g\right)\Delta T \qquad \text{----------(3.2)}$$

Taking the exponential of equation (3.2) gives

$$\frac{N^{\Delta T}}{N_j^o} = \exp\left(-\sum_{g=1}^{G} \sigma_{jg}\Phi_g\right)\Delta T \qquad \text{----------(3.3)}$$

and rearranging equation (3.3) gives

$$N^{\Delta T} = N_j^o \exp\left(-\sum_{g=1}^{G} \sigma_{jg}\Phi_g\right)\Delta T \qquad \text{----------(3.4)}$$

The final atom density of the absorber thus becomes

$$N_j^{\Delta T} = N_j^0 \exp\left[-\left(\sum_{g=1}^{G} \sigma_{j,g}\Phi_g\right)\Delta T\right] \qquad \text{----------(3.5)}$$

where N_j^0 = Absorber's atom density at the beginning of time, step ΔT

$N_j^{\Delta T}$ = Final atom density.

The excel spread sheet was constructed in-line with equation 3.5 in order to calculate the final atom density of the cadmium absorber. The equation using the table evaluates every single operation at any time t (sec) to determine how the number of atoms keeps reducing due to burn-ups up to a time Δt when the reactor was modified. The final number of atoms of the cadmium

after this period of operation is calculated using equation (3.5). The percentage change in the original number of atoms in the cadmium absorber is calculated using simple mathematical or empirical comparison.

$$\left(\frac{N_j^0 - N_j^{\Delta T}}{N_j^o}\right) * 100\% \qquad \text{----------(3.6)}$$

This represent the percentage decrease in control rod worth from the original value or the deviation from the initial rod worth.

3.6 Determination of Control Rod Worth of NIRR-1

The control rod worth is usually calculated using equation (2.19) above

That is, $\Delta\rho = \rho_{ex} + \rho_{sm}$

where $\Delta\rho$ is the control rod worth

ρ_{ex} is the excess reactivity

ρ_{sm} is the shutdown margin

3.7 Calculation of SDM & SRF

The shutdown margin (SDM) is obtained from equation (2.19) as

SDM = control rod worth – core excess reactivity = $\rho_{cr} - \rho_{ex}$

The safety reactivity factor (SRF) is obtained using the relation below

$$\textbf{SRF} = \frac{\text{control rod worth}}{\text{core excess reactivity}} = \frac{\rho_{cr}}{\rho_{ex}} \qquad \text{----------(3.7)}$$

3.9 Initial Number of Atoms in Cadmium Absorber (N_j^o)

The initial number of atoms of the cadmium absorber was obtained using technical control rod data as given by the manufacturer (SAR, 2005). The number of atoms for cadmium can be calculated using

$$N = \frac{mN_A}{A} \quad \text{----------} \quad (3.8)$$

where m = mass of cadmium

N_A = Avogadro's number

A = Atomic weight of cadmium

CHAPTER FOUR

RESULTS AND DISCUSSION

This chapter shows all our calculations starting from the initial number of atoms in the cadmium absorber, final number of atoms in the cadmium absorber, fractional change in the number of atoms (%), present control rod worth, shut down margin and safety reactivity factor.

4.1 Initial Number of Atoms in Cadmium Absorber (N_j^o)

The initial number of atoms of the cadmium absorber was obtained using technical control rod data as given by the manufacturer (SAR, 2005). The number of atoms for cadmium can be calculated using

$$N = \frac{mN_A}{A} \quad \text{----------} \quad (3.8)$$

where m = mass of cadmium

N_A = Avogadro's number

A = Atomic weight of cadmium

A = 112.4amu (1amu = 1.661 X 10^{-24}gram)

Density (Cd) = 8.65gcm^{-3}

Recall, that density = $\frac{Mass}{Volume}$

Also, volume (cylinder) = $\pi r^2 h$ (r = radius of cadmium rod and h = height of cd rod)

So that V = 3.142 * $(\frac{0.39}{2})^2$* 26.6

V = 3.1776cm^3

Mass = Density * Volume

Mass = 8.65gcm^{-3} * 3.1776cm^3

m = 27.486g

Therefore, number of moles N = $\frac{mN_A}{A}$

$$N = \frac{27.486 * 6.022140857\ X\ 10^{23}}{1.866\ X\ 10^{-22}}$$

N_j^o = 8.8705554 X10^{46}/ Mole

4.2 Final Number of Atoms in Cadmium

Using equation (3.5), the final absorber's atom density at the end of time ΔT, can be calculated as analyzed in the excel spread sheet and is found to be

$N_j^{\Delta T} = N_j^0 \exp\left[-(\sum_{g=1}^{G} \sigma_{j,g}\Phi_g)\Delta T\right]$

$N_j^{\Delta T}$ = 8.83468403754561 X 10^{46}/ Mole

4.3 Fractional Change in Number of Atoms (%)

The percentage (%) fractional change in the number of cadmium atoms was calculated using equation 3.6 as

$$\left(\frac{N_j^0 - N_j^{\Delta T}}{N_j^o}\right) * 100\% = \left(\frac{8.8705554\ x\ 10^{46} - 8.8346840\ x\ 10^{46}}{8.8705554\ x\ 10^{46}}\right) * 100\% = 0.40$$

This implies that approximately 0.40 of the initial control rod worth (7.0mk) has been lost due to burn-up representing 0.03mk.

4.4 Calculation of Present Control Rod Worth of NIRR-1 after burn-up

Recall that the initial control rod worth is 7.0mK (Jonah, 2007). From the fractional change in number of atoms which represent 0.03mk, it implies that the present control rod worth after burn-up is obtained as:

$7.0mk - 0.03mk$ = 6.97mk

4.5 Calculation of Shut Down Margin (SDM) and Safety Reactivity Factor (SRF)

The shutdown margin is calculated using equation 3.8. Recall that the present core excess reactivity of NIRR-1 (section 1.5) to be 3.89mk and the present control rod worth after burn-up, thus the present shutdown margin is

SDM = present control rod worth – Present core excess reactivity

SDM = 6.97 – 3.89 = 3.08mK

and, SRF = $\frac{Control\ rod\ worth}{Core\ excess\ reactivity}$

SRF = $\frac{6.97}{3.89}$

SRF = 1.79

Table 4.1
Summary of results for the Nigerian Research Reactor-1 (NIRR-1)

Parameter	Calculated value	Initial value
Initial number of atoms in Cadmium	8.8705554×10^{46}/mole	-
Final number of atoms in cadmium	8.8346840×10^{46}/mole	-
Fractional change in number of atoms	0.40%	-
Control rod worth	6.97mk	7.0mk
Shutdown margin	3.08mk	3.5mk
Safety reactivity factor	1.79	calculated from initial value

As presented in table 4, the fractional change in the number of atoms in the cadmium control is approximately 0.40 of its original value representing 0.03mk: This means that the control rod lost 0.03mK due to burn-up during reactor operation for the period under review.

The present control rod worth after burn-up was found to be 6.97mk. This value shows a marginal decrease of the original control rod which is expected of low power reactors like our MNSR.

Also, the shutdown margin and safety reactivity factor was calculated to be 3.08mk and 1.79 respectively. This value of SDM & SRF passed safety consideration that requires the value to be greater than 2.5mk and 1.5mk respectively satisfying the International Atomic Energy Agency (IAEA) provision. This means that the control rod in NIRR-1 can still perform its intended safety function after the upgrade.

CHAPTER FIVE

SUMMARY, CONCLUSION AND RECOMMENDATION

5.1 Summary and Conclusion

In addition to constructing the operational data of the reactor from inception in 2004 to when a modification was conducted on the reactor in 2011, the following were also evaluated. The initial number of atoms of the cadmium control rod was calculated to be 8.8705554 x 10^{46}/mole. The final number of atoms of the cadmium control rod was calculated to be 8.8346840 x 10^{46}/mole. The fractional change in the number of atoms of the cadmium control rod was calculated to be approximately 0.40 amounting to 0.03mK of the initial control rod worth. The present control rod worth was evaluated to be 6.97mk. This value compares closely to the 7.0mk value obtained from the off-site zero power critical experiment performed in

December 1997 at the China institute of atomic energy (CIAE) and 7.1mk obtained by Balogun in 2003. The shutdown margin (SDM) and the safety reactivity factor (SRF) was calculated to be 3.08mk and 1.79 respectively and compares well with the value obtained from the off-site zero power critical experiment performed in December 1997 at the CIAE, Beijing (Balogun 2003) satisfying the IAEA requirement that this parameter be greater than 2.5mk and 1.5 respectively (Balogun, 2003).

The results from our findings compares very well with recent works by Asuku et al., (2015) and Babangida et al., (2016). This confirmed that the reactor has satisfied all safety and regulatory requirements both by the Nigerian Nuclear Regulatory Authority (NNRA) and the International Atomic Energy Agency (IAEA).

5.2 Recommendation

Based on our result, we recommend the following;

1. The present control rod in NIRR-1 reactor can still perform its intended safety functions however, we recommend that some other methods like Stochastic method, WIMS and CITATION Codes, Strong Absorber nuclear data for diffusion codes, Improved Monte Carlo pertubation method, Monte Carlo codes be use to re-evaluate the control rod worth in subsequent assessment.

2. Based on our calculations, we recommend a periodic review safety assessment of the control rod between 5-10 years to ascertain whether the control rod can still perform its safety functions.

REFRENCES

Ahmed, Y.A., Ewa, I.O.B., Umar, I.M., 2002. Effective Resonance Energy and Non-Idealityof Epithermal Neutron Flux Distribution in Neutron Activation Analysis. Nigerian Journal of Physics 14 (1), 82–85.

Ahmed, Y.A., Ewa, I.O.B., Umar, I.M., 2006. Variations in Nuclear Data and its Impact on INAAJournal of Applied Sciences 6 (8), 1692–1697 (Asian Network for Scientific Information, ISSN 1812-5654).

Ahmed, Y.A., Mansir, I.B., Dewu, B.B.M., 2013. Installation of permanent cadmium-lined channel as a means for increasing epithermal NAA capabilities of miniature neutron sourcereactors.Nuclear Engineering and Design 263 (2013) 70– 76

Aoyama, T., Sekine, T., Maeda, S., Yoshida, A., Maeda, Y., Suzuki, S., Takeda, T., 2007. Core performance Tests for the JOYO MK-III upgrade. Nuclear Engineering and Design 237, 353-368.

Balogun, G.I., Jonah, S.A., Ahmed, Y.A., Sa'adu, N., 2004. Results of On-site Zero power and Criticality Experiments for the Nigeria Research Reactor-1, Internal, Report CERT/NIRR-1/ZP/01.

Bell, G., Glastone, S., 1970. Nuclear Reactor Theory, Litton Educational Publishing, Inc.

Bretscher, M.M., 1984. Blackness Coefficients, Effective Diffusion Parameters, and Control Rod worth for Thermal Reactors, ANL/RERTR/TM-5, 1984.

Bretscher, M., 1997. Computing Control Rod Worths in Thermal Research Reactors. ANL/RERTR/TM-29.

Duderstadt, J.J., Hamilton, L.J., 1976. Nuclear Reactor Analysis. John Wiley & Sons, New York.

Etienne, P., 2003. “Nuclear fuel cycles for mid-century Deployment” (PDF). MIT.p.81.

Fadaei, A.H., Setayeshi, S., 2009. Control Rod Worth calculation For VVER-1000 Nuclear Reactor Using WIMS and CITATION Codes. Progress in Nuclear Engineering 51, 184-191.

Faghihi, F., Fadaie, A.H., Sayareh, R., 2007. Reactivity Coefficients Simulation of the Iranian VVER-1000 Nuclear Reactor Using WIMS and CITATION codes. Progress in Nuclear Energy 49, 68-78.

Foster, A.R., Wright, R.L., 1997. Basic Nuclear Engineering, 3rd Edition, Allyn & Bacon, Inc.

Fowler, T.B., Vondy, D.R., Guningram, G.W., 1971. Nuclear Reactor Core Analysis Code: CITATION. Oak Ridge National Labaratory, ORNL-TM-2496, Rev. 2.

Glasstone, S., Edlund, M.C., 1956. The Elements of Nuclear Reactor Theory, Van Nostrand, New York.

Hong, L.P., 1999. Depletion Analysis on the Control Rod Absorber of RSG Gas Oxide and Silicide Fuel Cores. Atom Indonesia 25(11).

Hosoya, T., Kato, T., Murayama, Y., 2007. Investigation of JRR-3 Control Rod Worth Changed with Burn-up of Follower Elements. In: 13th International Congress of Radiation Research, San Francisco, California, 08.07.07 – 12.07.07.<http://www-pub. iaea.org/ MTCD/publications/PDF/P1360_ICRR_2007_CD/Papers/T.% 20Hosoya.pdf>

IAEA-TECDOC-233, 1989. Research Reactor Core Conversion from the use of Highly Enriched Uranium to the use of Low Enriched Uranium Fuels. Guidebook.

IAEA-TECDOC-643, 1992. Research Reactor Core Conversion from the use of Highly Enriched

Uranium to the use of Low Enriched Uranium Fuels. Guidebook.

Ibrahim, Y.V., 2012. Monte Carlo Simulation of Additional Safety Control Rod for Commercial MNSR to Enhance Safety. Annals of Nuclear Energy 2012.

Itagaki, M., Miyoshi, Y., Gakuhari, K., Okada, N., Sakai, T., 1993. Control Rod Interference Effects Observed during Reactor Physics experiments with Nuclear Ship "MUTSU". Jornal of Nuclear Science and Technology 30(5), 465-476.

Jacobs, A.M., Kline, D.E., Remick, F.J., 1960. Basic Principles of Nuclear Science and Reactors, Van Nostrand Company, Inc.

Jonah, S.A., Liaw, J.R., Matos, J.E., 2007. Monte Carlo simulation of core physics parameters of the Nigeria research reactor-1 (NIRR-1). Annals of Nuclear Energy 34, 953–957.

Jonah, S.A., Umar, I.M., Oladipo, M.O.A., Balogun, G.I., Adeyemu, D.J., 2006. Standardization of NIRR-1 irradiation and counting facilities for instrumental neutron activation analysis. Applied Radiation and Isotopes 64, 818–822.

Jonah, S.A., Ibrahim, Y.V., Akaho, E.H.K., 2008. The determination of reactor spectrum-averaged cross-sections in miniature neutron source reactor facility. Applied Radiation and Isotopes 66 (10), 1377–1380.

Kalcheva, S., Koonen, E., 2008. Improved Monte Carlo-Pertubation Method for Estimation of Control Rod Worths in A Research Reactor, International Conference on the Physics of Reactors "Nuclear Power: A Sustainable Resource", Casino-Kursaal Conference Center, Interlaken, Switzerland, September 14-19, 2008.

Kalcheva, S., Koonen, E., 2007. Optimized Control Rods of the Belgium Research reactor(BR2). Open Report SCK.CEN-BLG-1054.

Lamarsh, J.R., 1972. Introduction to Nuclear Reactor Theory, Addison-Wesley Company.

LiemPeng, H.L.P., Taswanda, T., TagorMalem, S., Sekimoto, H., Yoshitaka, N., 2002. Study on the Control Rod Interaction effect on RSG Gas Multipurpose Reactor(MPR-30). Annals of Nuclear Science Energy 29, 701- 716.

Musa, Y., Ahmed, Y.A., Yamusa, Y.A., Ewa, I.O.B., 2011. Determination of radial and axial neutron flux distribution in irradiation channel of NIRR-1 using foil activation technique, Annals of Nuclear Energy 50 (2012) 50–55.

Qazi, M.K., Akaho, E.H.K., Maakuu, B.T., Anim-Sampong, S., (1996). Nuclear Core Design Analysis of Ghana Research Reactor-1, NNRI Technical Report, GAEC-NNRI-RT-35.

Ravnic, M., (2000). Determination of Research Reactor Safety Parameters by Reactor Calculations. Lecture given at the Workshop on Nuclear Data and Nuclear Reactors: Physics, Design and Safety held at Trieste, Italy, 13th March – 14th April.

Saji, E., Takahashi, A., Uchida, S., Suzuki, K., 1986. Control Rod Worth in High Conversion PWR. Journal of Nuclear Science and Technology 23(8), 745-751.

SAR, 2005. Final Safety Report of Nigerian Research Reactor-1. CERT Technical Report-CERT/NIRR-1/FSAR-01.

Savva, P., Varvayanni, M., Catsaros, N., (2010). Dependence of Control Rod on Fuel Burn-up, Nuclear Engineering and Design 241(2011) 492-497.

Salvatores, M., 1992. Computational/Experimental Trends of Control Rod Worth in Large Fast Reactor Decoupled Cores. NEA / NSC / DOC (93) 10, CEA, Directorate for Nuclear Reactors, Cadarache, France.

Schindler, G.M., 1958. “On the Efficiency of A Concentric Cutt-Off Rod of A Thermal Reactor as a Function of the Inserted Length of the Rod”, J. Nuclear Energy, Vol.8, pp.18 to 32. Pergamon Press Ltd, London.

Shimazu, Y., Okazaki, K., Tsuji, M., 2006. Feasibility Study for Evaluation of Control Rod Worth in Pressurized Water Reactors using Neutron Count Rate during a Control Rod Drop Testing. Journal of Nuclear Science and Technology 43(8) 919-923.

Stievenart, P., Erkes, P., 1958. Determination of Reactor Transients and Time Variation of Core Material Concentration and Excess Reactivity byGraphical Methods, A/Conf. 15/P/1896, 2d International Conference on Peaceful Uses of Atomic Energy, Geneva, 1958.

Titouche, W., Zidi, T., Meftah, B., 2003. Strong Absorber nuclear Data for Diffusion Codes Calculations: Control Rod Worth. Centre de ReherecheNucleaire de Birine Ain-Oussera, Algeria.

Usachev, L.N., Bobkov, Y., 1972. Planning on Optimum Set of Microscopic Experiments and Evaluations to obtain a given Accuracy in Reactor Parameter calculations. INDC ECP-19U. IAEA International Nuclear Data Committee.

Varvayanni, M., Savva, P., Catsaros, N., Antonopoulos-Domis, M., (2009). Homogeneous Zones Definition in deterministic Codes and effect on the Computed Neutronics Parameters. Annals of Nuclear Energy 36, 567-574.

Varvayanni, M., Savva, P., Catsaros, N., 2009. Control Rod Worth Calculation Using Deterministic and Stochastic Methods. Annals of Nuclear Energy 36(2009) 1718-1725.

Varvayanni, M., Catsaros, N., Antonopoulos-Domis, M., 2009. Estimation of Irradiated Control Rod Worth. Annals of Nuclear Energy 36 (2009) 1706-1710.

Walters, L.C., 1998. “Thirty years of fuels and materials information from EBR-II”. Journal of Nuclear Materials (Elsevier) 270:39-48

Williams, M.M.R., 2004. Uncertainties in Control Rod Worth in A Damaged Reactor Core. Annals of Nuclear Energy 31, 1073-1081.

APPENDIX

Table 3.2Data retrieved from the computer system of the reactor

Log_1_	Dec_9_44_2	11							
Time	NeutronFl	ux(n/cm⁻²/s)	Rod Height	Tempe	Ratu	re(⁻C)	Reactivity	Radia	tion(uS
(Hr)	Measured	Preset	(mm)	Inlet	Outl	et pool	(mK)	R.Top P	uriColu
0	2.72E+08	1.00E+12	82	26	26.2	28.2	-0.151	0	0
0.2	5.65E+11	1.00E+12	82	27.4	39.1	28.5	0.042	14.8	2.97
0.4	5.64E+11	1.00E+12	82	28.2	40.6	28.4	0.029	13.93	2.97
0.6	5.70E+11	1.00E+12	82	29.6	41.9	28.4	0.04	14.73	2.03
0.8	5.69E+11	1.00E+12	82	30.9	43	28.6	0.02	17.83	2.67
1	5.74E+11	1.00E+12	82	31.4	43.6	28.6	0.039	15.67	1.53
1.2	5.69E+11	1.00E+12	82	31.9	43.7	28.7	0.011	14.2	2.13
1.4	5.70E+11	1.00E+12	82	32.3	44.2	28.8	0.016	15	2.43
1.6	5.70E+11	1.00E+12	82	32.7	44.9	28.8	0.004	15.33	2.07
1.8	5.75E+11	1.00E+12	82	32.8	44.9	28.9	0.005	13.33	2.23
2	5.72E+11	1.00E+12	82	32.9	44.9	29	0.008	15.53	2.17
2.2	5.78E+11	1.00E+12	82	32.7	45.4	29.1	0.022	16.27	2.43
2.4	5.70E+11	1.00E+12	82	33.2	45.6	29.1	-0.015	15.67	3.23
2.6	5.79E+11	1.00E+12	82	33.7	45.5	29.2	0.013	16.5	3.97
2.8	5.77E+11	1.00E+12	82	33.4	45.7	29.2	0.019	15.47	3.03
3	5.74E+11	1.00E+12	82	33.8	45.8	29.3	0.01	15.93	3
3.2	5.76E+11	1.00E+12	82	33.5	45.7	29.5	0.011	16.3	3.87
3.4	5.77E+11	1.00E+12	83	34.2	45.8	29.6	0.006	18.5	3.7
3.6	5.79E+11	1.00E+12	82	33.6	46.2	29.6	0.004	14.87	1.83
3.8	5.78E+11	1.00E+12	82	33.9	46.3	29.7	0.017	15.93	3
4	5.74E+11	1.00E+12	82	34.6	46.2	29.8	0.002	15.53	2.47
4.2	5.73E+11	1.00E+12	82	34.1	45.9	29.9	0.003	16.07	3.37
4.4	5.78E+11	1.00E+12	82	34.2	46.4	29.9	0.022	17	2.97
4.6	5.74E+11	1.00E+12	82	34.6	46.3	29.9	-0.001	16.67	3.27
4.8	5.74E+11	1.00E+12	82	34.7	46.5	30.1	0.01	16.07	2.9
5	5.76E+11	1.00E+12	82	34.4	46.3	30	0.018	18.87	5.23
5.2	5.79E+11	1.00E+12	82	34.2	46.6	30.2	0.015	16.47	2.27
5.4	5.80E+11	1.00E+12	82	35	46.9	30.3	0.02	17.07	3.77
5.6	5.74E+11	1.00E+12	83	33.8	46.8	30.3	-0.014	18.87	5.13
5.8	5.79E+11	1.00E+12	82	34.3	46.5	30.3	0.011	17.13	3
6	5.78E+11	1.00E+12	82	35.4	46.6	30.5	0.019	16.73	4
6.2	5.80E+11	1.00E+12	82	34.5	46.8	30.4	0.02	15.43	2.87
6.4	5.79E+11	1.00E+12	82	35	47.2	30.7	0.013	17.13	3.4
6.5	1.22E+10	1.00E+12	82	34.4	37	30.6	-3.862	3.6	2.93

Table 3.3Operation data at full power (31Kw and $1x10^{12}n/cm^2s$)

Log_1_	Dec_10_10_	2009								
Time	NeutronFl	ux(n/cm²s)	Rod Height	Tempe	Ratu	re(⁻C)	Reactivity	Radia R.Top	tion(uS	v/Hr) mn
(Hr)	Measured	Preset	(mm)	Inlet	Outl	et pool	(mK)	P	uriColu	Hall
0	2.43E+08	5.00E+11	82	24	24.3	25.4	0.154	0	0	0
0.2	5.48E+11	5.00E+11	82	24.8	37.7	25.7	0.033	8.8	3.93	0.27
0.4	5.49E+11	5.00E+11	82	26	38.8	25.5	0	6.63	4.4	0
0.6	5.53E+11	5.00E+11	82	27.3	40.2	25.5	0.013	9.47	4.2	0.27
0.8	5.60E+11	5.00E+11	82	28.3	40.5	25.5	0.012	6.57	2.5	0
1	5.62E+11	5.00E+11	82	28.6	40.9	25.7	0.033	10.9	3.2	0.53
1.2	5.62E+11	5.00E+11	82	29.3	41	25.6	0.032	9.9	3.3	0.37
1.4	5.59E+11	5.00E+11	82	30.2	41.8	25.8	0.014	10.17	3.13	0
1.6	5.61E+11	5.00E+11	82	29.8	42.3	25.9	-0.001	12.43	2.9	0.4
1.8	5.64E+11	5.00E+11	82	30.3	42.5	26.2	0.039	13.67	3.4	0
2	5.63E+11	5.00E+11	82	30	42.9	26.1	0.019	15.97	3.9	0
2.2	5.65E+11	5.00E+11	82	30.7	43.1	26.4	0.004	13.73	3.5	0.27
2.4	5.64E+11	5.00E+11	82	31.8	43.9	26.5	0.04	16.17	3.1	0
2.6	5.67E+11	5.00E+11	82	31	43.5	26.7	0.017	13.67	3	0
2.8	5.68E+11	5.00E+11	82	31.3	43.4	26.7	0.031	13.17	2.7	0
3	5.63E+11	5.00E+11	82	31.8	43.8	26.7	0.006	16.2	2.1	0.47
3.2	5.66E+11	5.00E+11	82	31.7	43.9	27	0.017	14.83	3.6	0.27
3.4	5.63E+11	5.00E+11	82	31.6	44.2	26.9	0.008	15.93	3.3	0.43
3.6	5.67E+11	5.00E+11	82	31.9	44.1	27	0.007	13.23	3	0.43
3.8	5.66E+11	5.00E+11	82	32	44.6	27.2	0.001	12.27	3.2	0.27
4	5.66E+11	5.00E+11	82	32	44.4	27.3	0.023	14.57	3.2	0.3
4.2	5.59E+11	5.00E+11	82	31.6	43.7	27.2	-0.011	12.83	3.3	0.47
4.4	5.68E+11	5.00E+11	82	32.5	44.6	27.4	-0.004	14.1	2.5	0
4.6	5.62E+11	5.00E+11	82	32	44.5	27.4	-0.002	17.7	3.5	0.67
4.8	5.67E+11	5.00E+11	82	32.4	44.5	27.6	0.016	15.4	3.9	0.23
5	5.67E+11	5.00E+11	82	32.3	45	27.7	0.007	13	3.2	0.67
5.2	5.64E+11	5.00E+11	83	32	44.5	27.7	0.004	13.57	3.6	0.2
5.4	5.64E+11	5.00E+11	82	32.6	44.8	27.9	0.001	14.73	2.9	0.6
5.6	5.68E+11	5.00E+11	82	32.2	44.6	27.9	0.025	14.5	3.5	0.57
5.8	5.70E+11	5.00E+11	82	32.8	44.9	27.9	0.004	17.8	3.8	0.1
6	5.70E+11	5.00E+11	82	32.7	45.1	28	-0.005	15.97	2.7	0.2
6.2	5.62E+11	5.00E+11	82	32.7	44.5	27.9	-0.03	14.33	2.5	0.2
6.4	1.62E+11	5.00E+11	82	32.8	42.5	28.1	-4.502	11.07	3.2	0.2
6.4	1.58E+10	5.00E+11	82	32.7	35.3	28.2	-3.849	2.87	3.53	0.87

Table 3.4Operationaldata at half power (15.5Kw and $5x10^{11}n/cm^2s$)

Log_12	_Mar_9_42_	2009								
Time	NeutronFl	ux(n/cm⁄s)	Rod Height	Tempe	Ratu	re(⁻C)	Reactivity	Radia R.Top	tion(uS	v/Hr) mn
(Hr)	Measured	Preset	(mm)	Inlet	Outl	et pool	(mK)	P	uriColu	Hall
0	8.73E+07	5.00E+11	82	23.9	23.9	25.1	-2.902	0	0	0
0.2	1.08E+12	5.00E+11	82	24.3	43.8	25.7	0.046	24.13	1.27	0.53
0.4	1.10E+12	5.00E+11	82	27.1	46.7	25.8	0.011	24.37	2.4	0.13
0.6	1.11E+12	5.00E+11	82	29.4	48.7	25.9	0.01	15.1	2.3	0.2
0.8	1.12E+12	5.00E+11	82	31.1	50.1	26	0.017	24.33	1.8	0.9
1	1.12E+12	5.00E+11	82	32.3	51.3	26.3	-0.006	25.4	2.7	0.43
1.2	1.13E+12	5.00E+11	82	33	52	26.5	0.049	25.33	3.3	0.4
1.4	1.13E+12	5.00E+11	82	34.3	53.1	26.5	0.015	29.03	3.57	0.3
1.6	1.13E+12	5.00E+11	82	34.9	53.6	26.8	0	27.77	2	0.4
1.8	1.12E+12	5.00E+11	82	35.9	53.9	27	0.022	27.4	3.17	0.6
2	1.12E+12	5.00E+11	82	35.4	54.6	27.1	-0.003	23.23	1.77	0.3
2.2	1.14E+12	5.00E+11	82	36.9	54.7	27.4	0.028	28.23	1.87	0.4
2.4	1.13E+12	5.00E+11	82	36.8	55.2	27.8	-0.03	31.57	2.4	0.8
2.6	1.13E+12	5.00E+11	82	37.6	55.2	27.7	0.005	31.1	3	0.53
2.8	1.15E+12	5.00E+11	83	36.4	55.6	27.8	0.021	33.9	1.57	0.7
3	1.14E+12	5.00E+11	82	36.6	55.3	28.1	-0.003	33.43	2.63	0.8
3.2	1.14E+12	5.00E+11	82	38.4	55.8	28.4	-0.002	28.93	1.6	0.53
3.4	1.15E+12	5.00E+11	83	37.6	55.5	28.3	0.04	28	3.77	0.5
3.6	1.12E+12	5.00E+11	83	37.1	55.8	28.6	0.036	31.57	3.33	0.83
3.8	1.15E+12	5.00E+11	82	37.3	56.2	28.7	0.03	29.6	2.87	0.3
4	1.15E+12	5.00E+11	82	37.5	56.2	29.1	0.018	30.83	2.87	1.3
4.1	2.27E+10	5.00E+11	82	36.8	40.5	29.1	-4.557	5.83	0.93	0.2

Table 3.5 Operational data at low power (6Kw and $3x10^9 n/cm^2 s$)

Log_1	8_Oct_10_44_	2010							
Time	NeutronFlu	x(n/cm⁄s)	Rod Hei	ght Tem	Peratur	e(⁻C)	Reactivit	y Radiation(uS	v/Hr) mn
(Hr)	Measured	Preset	(mm)	Inle	t Outle	t pool	(mK)	R.TopPuriColu	Hall
0	1.11E+09	5.00E+11	82	25.9	26.2	27.8	-0.117	0.00 0.00	0
0.2	1.22E+09	5.00E+11	82	25.6	25.9	27.7	0.014	0.10 8.43	0.1
0.4	1.30E+09	5.00E+11	82	25.9	26	27.7	0.072	0.30 10.47	0.2
0.6	1.34E+09	5.00E+11	82	25.6	26.1	27.6	0.014	0.20 9.57	0.3
0.8	1.36E+09	5.00E+11	82	25.6	26.1	27.8	-0.026	0.10 10.10	0.6
1	1.37E+09	5.00E+11	82	26	26.3	28	-0.096	0.20 8.53	0.3
1.2	1.39E+09	5.00E+11	82	25.8	26	27.8	-0.023	0.30 8.93	0.2
1.4	1.39E+09	5.00E+11	82	25.9	26	27.8	-0.046	0.00 10.73	0.1
1.6	1.41E+09	5.00E+11	82	25.7	25.9	27.7	0.024	0.00 10.63	0.4
1.8	1.40E+09	5.00E+11	82	26.2	26.2	27.7	-0.022	0.30 12.03	0.1
2	1.41E+09	5.00E+11	82	25.9	26.1	27.8	0.082	0.30 11.77	0.2
2.2	1.39E+09	5.00E+11	82	26.1	26.2	27.9	-0.069	0.60 11.07	0.2
2.4	1.38E+09	5.00E+11	82	26	26.3	27.8	-0.039	0.10 14.10	0.5
2.6	1.40E+09	5.00E+11	82	26.1	26.2	27.9	0.005	0.10 15.47	0.1
2.8	1.38E+09	5.00E+11	82	25.9	26.2	27.9	-0.052	0.50 18.17	0.3
3	1.39E+09	5.00E+11	82	26.1	26.3	27.9	-0.035	0.30 18.77	0.2
3.6	1.42E+09	5.00E+11	82	26.1	26.3	27.7	0.049	0.20 14.90	0.6
3.8	1.40E+09	5.00E+11	82	26	26.3	27.9	0.002	0.20 13.23	0.6
4	1.40E+09	5.00E+11	82	26.4	26.6	28.1	0.07	0.40 16.07	0.3
4.2	1.38E+09	5.00E+11	82	26	26.3	27.8	0.007	0.10 15.40	0.1
4.4	1.39E+09	5.00E+11	82	26.4	26.5	28.1	0.041	0.30 17.10	0.8
4.6	1.37E+09	5.00E+11	82	25.9	26.1	27.8	0.005	0.50 18.43	0.3
4.8	1.36E+09	5.00E+11	82	25.9	26.2	27.7	-0.027	0.30 16.13	0.3
5	1.35E+09	5.00E+11	82	26	26.2	27.8	-0.075	0.20 18.40	0.3
5.2	1.37E+09	5.00E+11	82	26.2	26.3	28.1	0.028	0.20 16.40	0.3
5.4	1.36E+09	5.00E+11	82	26.1	26.4	27.8	-0.051	0.20 15.40	0.73
5.7	1.38E+09	5.00E+11	82	26.1	26.4	27.8	0.042	0.00 21.50	0.4