TITLE PAGE

CORE LOADING REQUIREMENTS IN CONVERTING HEU MNSR CORE TO LEU USING WIMS AND CITATION CODES

BY

AZANDE TERUNGWA SYLVANUS

MSC/SCIEN/47810/2005-06

A RESEARCH WORK SUBMITTED TO THE POSTGRADUATE SCHOOL, AHMADU BELLO UNIVERSITY ZARIA, IN PARTIAL FULFILLMENT FOR THE AWARD OF MASTER OF SCIENCE DEGREE (M.SC. PHYSICS) PHYSICS DEPARTMENT, AHMADU BELLO UNIVERSITY ZARIA

October, 2010

DECLARATION

I declare that the work in this thesis entitled **Core loading requirements in converting HEU MNSR core to LEU using WIMS and CITATION CODES**, has been performed by me in the Department of Physics under the supervision of Dr. G.I. Balogun and Mr. A.S. Ajuji. The information derived from the literature has been duly acknowledged in the text and a list of references provided. No part of this thesis was previously presented for another degree or diploma at any University.

Azande Terungwa Sylvanus **signature:**
M.Sc./SCIEN/47810/2005-06 27th October, 2010

CERTIFICATION

This thesis entitled **CORE LOADING REQUIREMENTS IN CONVERTING HEU MNSR CORE TO LEU USING WIMS AND CITATION CODES** by **Azande Terungwa Sylvanus (MSC/SCIE/47810/2005-06)** meets the regulations governing the award of the degree of Master of Science Degree (M.Sc.) Physics of Ahmadu Bello University, and is approved for its contribution to knowledge and literary presentation.

…… Date: ……

Dr. G.I. Balogun

Chairman, Supervisory committee

…….. Date: ……

Mr. A.S. Ajuji

Member Supervisory Committee

…….. Date: ……

Dr. Nasiru Rabiu

Head of Department

…….. Date: ……

Dean, Postgraduate School

DEDICATION

This research work is dedicated to my mother, my wife ***Jane Mbahilen*** and ***Miss. Halima Balogun***, the daughter of my major supervisor, who was born on the eve of the commencement of this research work.

ACKNOWLEDGEMENT

The author gratefully acknowledges God, and those whose works are cited. Particular thanks are due to my Major Supervisor, *Dr. G.I. Balogun*, who through his constructive criticism and indefatigability I came to the limelight of this research work and similar thanks go to Mr. A.S. Ajuji, my supervisor. I appreciate *Mrs. M.N. Chiv* (my mother), *Jane* (my wife), and *Mrs. A. Gom* for their prayers, financial and moral supports through the course of this research.

I appreciate all the staff of Physics Department, Center for Energy Research and Training (CERT), and the Staff of Redemption College Zaria for their various contributions. Worthy of particular mention is *Dr. Y.A. Ahmed* of Reactor Engineering Section, CERT, whose works is not only cited but benevolently availed me of other materials. I also acknowledge *Miss. Nancy Lami* of CERT library, *Mrs. Chinyere Kalu* of the Nigerian College of Aviation Technology (NCAT) Zaria, *Mr. Joseph Nashakyaa* of Nigerian Institute of Transport Technology (NITT) Zaria, *Mr. Friday Yakubu* of Iya Abubakar Computer Center, Ahmadu Bello University Zaria, *Mr. Moses Agida*, a fellow student, *Mrs. Faga*, and many others too numerous to mention for their kind assistance throughout the period of this research.

Lastly, my thanks go to *Miss. Deborah Musa*, an out gone science student of Federal Government Girls' College (FGGC) Zaria, who assisted me in typesetting this research work. May the Almighty God richly reward you all a hundred fold.

ABSTRACT

The world over research reactors are being converted from highly enriched uranium (HEU) to low enriched uranium (LEU). The NIRR-1 falls in the category of HEUs, being 90% enriched in the fissile U-235. It is therefore desirable to convert it from this enrichment to about 20% in conformity with this global trend. The reason for this general trend is to make research reactor fuel as unattractive as possible to groups that may be interested in using such highly enriched reactor cores for non-peaceful purposes. In this work, we have developed a computational scheme that would theoretically achieve this objective as easily as possible. The scheme systematically reduces the enrichment from 90% (or any other initial values) to 20% and even below, in steps of 5% or any desired percentage variation. At each step, important neutronics parameters, especially safety related parameters are computed. Two fuel types are considered – UAl_4 and UO_2 – and the variation of safety reactivity factor (SRF) with enrichment is also discussed for each fuel type. It is shown that between 90% and 10% inclusive, the value of this parameter remains higher than 1.5. At 20% enrichment, SRF values of 1.595 and 1.827 are obtained for UAl_4 and UO_2 respectively, satisfying the IAEA requirement that this parameter should be greater than 1.5. The shutdown margin (SDM) at this enrichment is found to be 2.1780mk for UAl_4 and 2.9343mk for UO_2. The control rod curves give the characteristic s-shapes, with excess reactivity values in good agreement with the design specifications. Comparison of some of the results obtained in this work (e.g. mass loading) with those of MCNP calculations reported in the literature is very encouraging. It should be noted however that this theoretical investigation indicates a drastic review of the fuel loading of NIRR -1 core to achieve low enrichment.

TABLE OF CONTENTS

CHAPTER TWO

REVIEW OF RELATED LITERATURE

CHAPTER THREE

THEORITICAL CONSIDERATION AND CALCULATIONS

CHAPTER FOUR

CHAPTER FIVE

CONCLUSION AND RECOMMENDATIONS

LIST OF TABLES

LIST OF FIGURES

LIST OF GRAPHS

LIST OF APPENDICES

ABBREVIATIONS, DEFINITIONS, GLOSSARIES AND SYMBOLS

DOE: Division of Energy

eV: electron volt is the energy required by a unit (electronic) charge which has been accelerated through a potential of 1 volt. The electronic charge is 1.602×10^{-19}C; hence 1eV is equivalent to 1.602×10^{-19}J.

GTRI: Global Threat Reduction

HDBK: Handbook

HEU: Highly Enriched Uranium

HWR: Heavy Water Reactor

IAEA: International Atomic Energy Agency

INFCE: International Nuclear Fuel Cycle Evaluation

k_{eff} **:** Effective multiplication facto. Multiplying factor *k* of a reaction is the number of fission processes in a given generation compared with that in a previous generation.

LEU: Low Enriched Uranium

LWR: Light Water Reactor

MNSR: Miniature Neutron Source Reactor

NIRR-1: Nigeria Research Reactor-1

ONNP: Office of Naval Nuclear Propulsion

RERTR: Reduced Enrichment for Research and Test Reactors

SDM: Shutdown Margin

SDR: Spin Dependence Recombination

SRF: Safety Reactivity Factor

U-235: A fissile isotope of Uranium with mass number of 235

UAl_4: Reactor fuel made of Uranium-aluminium.

UO_2: Reactor fuel made of Uranium Oxide.

WGU: Weapon Grade Uranium.

WIMS: Winfrith's Improve Multi-group Scheme.

Fertile Materials: Materials that can undergo transmutation to become fissile materials.

Fissile materials: Materials that can undergo fission reaction.

Fission reaction: A nuclear reaction in which a heavy nuclide splits, by absorbing a neutron, to form light nuclei, two or more neutrons and energy.

Enrichment: Increasing the percentage of the fissile isotope U-235 in order to be able to sustain the chain reaction necessary in a nuclear reactor.

Criticality: the condition of a nuclear reactor which is just capable of sustaining a nuclear chain reaction.

Reactivity: the measure of ease of ignition from cold and of the relative rate of combustion under specified conditions in a nuclear reactor.

Scattering reactions: refer to the reactions in which the neutron remaining after interaction is generally moving in a different from that prior to the interaction. Scattering can be either elastic or inelastic. In elastic scattering, the energy exchange between the neutron and the nucleus is entirely kinetic in nature while in inelastic scattering; part of the kinetic energy of the neutron is transferred to the nucleus as internal (potential) energy.

Burn-up: The cumulative exposure of a nuclear fuel in a reactor.

Critical mass: **The minimum quantity of a material that is capable of sustaining a fission chain reaction.**

CHAPTER ONE

BACKGROUND TO THE STUDY

1.0 INTRODUCTION

The world over, research reactors are being converted from highly enriched uranium (HEU) to low enriched uranium (LEU). The NIRR-1 falls in the category of HEUs, being 90% enriched in the fissile U-235. It is therefore desirable to convert it from this enrichment to about 20% in conformity with this global trend. The reason for this general trend is to make research reactor fuel as unattractive as possible to groups that may be interested in using such highly enriched reactor cores for non-peaceful applications. In this work, we have developed a computational scheme that would theoretically achieve this objective. The scheme systematically reduces the enrichment from 90% (or any other initial values) to 20% and even below, in steps of 5% or any desired percentage variation. At each step, important neutronics parameters, especially safety related parameters are computed. Two fuel types are considered – UAl_4 and UO_2 – and the variation of safety reactivity factor (SRF) with enrichment is also discussed for each fuel type.

Some studies have been performed on the conversion of the core of the Syrian MNSR, which considered some fuel types like the dispersion ones in general (U-Alx-Al), besides the UO_2 fuels; (Khamis and Khattab, 1999; Albarhoum, 2004, 2005a, 2006). The general conclusion for the dispersion fuel types was that these fuels have low densities so that special configurations of the core should be used.

Another configuration has been considered in which a mixed fuel (some rods contain HEU fuel, and others containing LEU fuel) was employed; (Albarhoum, 2005b). In the case of the UO_2 fuel, different results were obtained (Khamis and Khattab, 1999; Matos and Lell, 2005). In a previous work of some colleges, a UO_2 fuel with 5.45 g content of U-235/fuel element was used; (Khamis and Khattab, 1999). The paper indicated a configuration in which only 199 fuel elements were necessary to have about 4.579 mk for the initial excess reactivity. In other works, (Matos and Lell, 2005); two types of UO_2 were considered: the UO_2 as a dispersed fuel, and a ceramic pellet fuel fabricated by Zircatec (CANADA). In some proposed models, the total number of fuel pins (Khamis and Khattab, 1999) and the core radius/height ratio (Matos and Lell, 2005) has been drastically changed. This brings about noticeable changes in the relative flux values for both inner and outer irradiation sites. The differences observed in their approach and methodology has prompted the desire to embark on comprehensive conversion studies of the NIRR-1 in order to convert the reactor from HEU to LEU (less than 20%) in conformity with the growing global trend.

In this work, HEU and LEU cores are analyzed using the present UAl_4 fuel and a potential LEU fuel (UO_2 clad in zircalloy). The existing HEU core was also analyzed to validate the reactor model used. A significant feature of this work is the preservation of the technical and geometric specification of the reactor so as to maintain the original designed thermal hydraulic of the reactor.

In the neutronic calculation work, the reactor core parameters were modified interactively using the thermal reactor neutronics code WIMS (Askew et al., 1966) and core calculation code CITATION (Fowler et al., 1989). However, experience has shown that the input requirements for applications of these codes can be demanding and difficult to setup (Albarhoum, 2008; Balogun, 2003a). To simplify the user interface to specific calculations like U-235 loading and fuel enrichment, a simple user interface was developed, where the major choices allow the user with the much simpler task of specifying basic geometrical, material and operational data and the program runs with ease.

This research work has been structured into five chapters. The first chapter introduces the need for the research work, discusses research reactors briefly with emphasis on Miniature Neutron Source Reactor (MNSR). The lattice code WIMS and core analysis code CITATION are also introduced. The aims and objectives, the scope and limitations as well as the significance and justification of the study are also highlighted in the first chapter. In Chapter two, the relevant literature that has been reviewed is presented with particular emphasis on those that touch on core conversion. Also reviewed are the basic principles underlying important calculations necessary at each step in core conversion studies in relation to the material composition of the core. Theories associated with WIMS and CITATION as well as the implementation of these codes is considered in Chapter three. Chapter four gives the results obtained as well as the discussion of the results. Conclusion and

recommendations are presented in the fifth Chapter. Cited works are then referenced.

1.1 RESEARCH REACTORS

A nuclear reactor is a device in which a controlled nuclear chain reaction is achievable. The main purpose of research reactors is to provide a neutron source for research and other purposes. Their output (neutron beams) can have different characteristics depending on use. Research reactors are simpler than power reactors and need far less fuel, and far less fission products build up as the fuel is used. West (1997) revealed that their fuel requires more highly enriched uranium, typically up to 20% U-235, although some older ones use 93% U-235. They also have a very high power density in the core, which requires special design features. (Krull, 2000) added that like power reactors, the core needs cooling, and usually a moderator is required to slow down the neutrons and enhance fission. As neutron production is their primary function, most research reactors also need a reflector to reduce neutron loss from the core. A common design is the pool type reactor where the core is a cluster of fuel elements sitting in a large pool of water. Among the fuel elements there are control rods and empty channels of experimental materials. Each element comprises several curved aluminium-clad fuel plates in a vertical box. The water moderates and also cools the reactor, and graphite or beryllium is generally used for the reflector, although other materials may also be used. Apertures to access the

neutron beams are set in the wall of the pool. Research reactors have a wide range of uses, including analysis and testing of materials and production of radioisotopes. Their compatibilities are applied in many fields, within the nuclear industry as well as in fusion research, environmental science, advanced materials development, drug design and nuclear medicine.

1.2 NIGERIA RESEARCH REACTOR – 1 (NIRR-1)

The Nigeria Research Reactor-1 (NIRR-1) is a Miniature Neutron Source Reactor (MNSR) designed by China Institute of Atomic Energy (CIAE); (Zhou, 1986). The reactor assembly comprise of a reactor core, beryllium reflector and a centrally located cadmium control rod. Figure 1.1 shows the longitudinal cross section through the reactor assembly.

The Nigerian Research Reactor-1 (NIRR-1) is located in Zaria, Kaduna State and operated by the Center for Energy Research and Training (CERT), Ahmadu Bello University (ABU). The reactor, which was critical for the first time on 03 February, 2005 is presently in its first fuel cycle; (Balogun and Jonah, 2004). It is specifically designed for use in neutron activation analysis (NAA), limited radioisotope production and also suitable for kinetics and control.

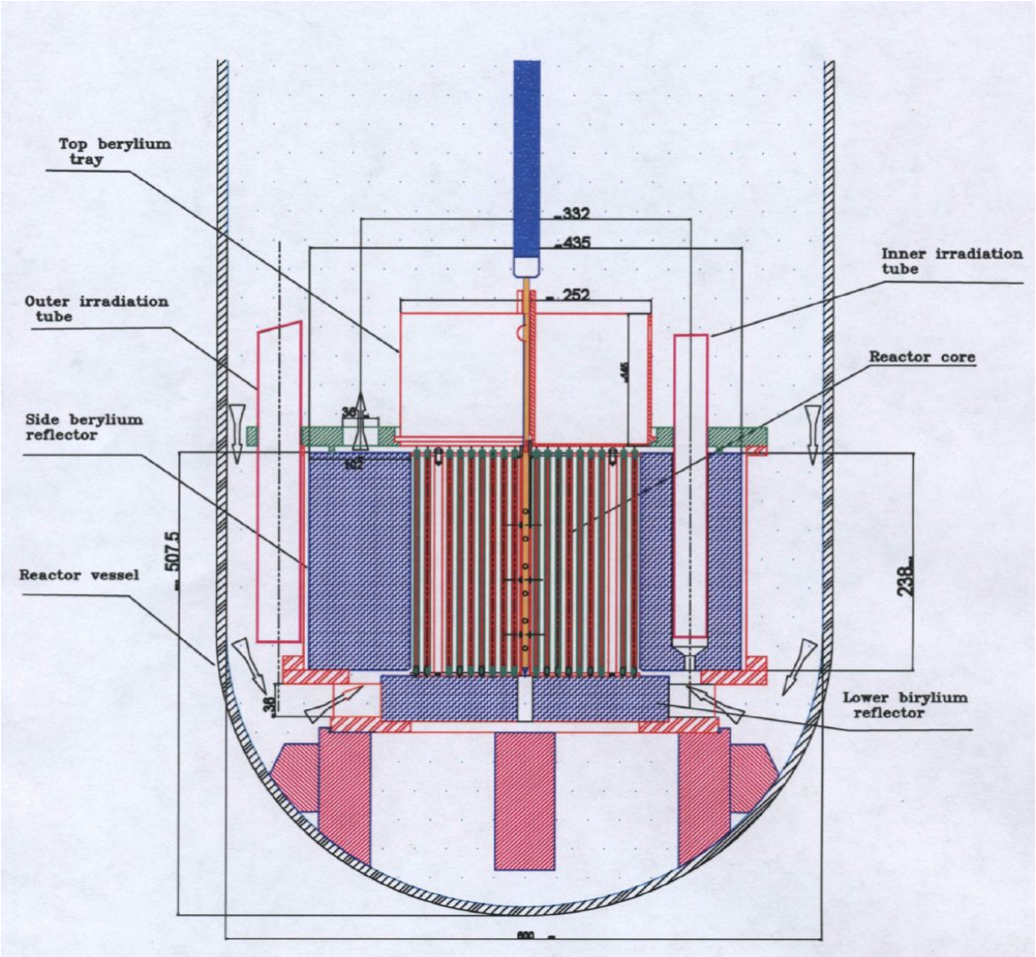


Figure 1.1: Schematic longitudinal section of NIRR-1 vessel with fuel elements, control rod, and beryllium reflector (all dimensions are in mm).

As is implicit in the name, it is designed mainly to serve as a neutron source. The core is a cylindrical fuel assembly containing 347 fuel elements. NIRR-1 has only one central control rod (figure 1.4) which performs the functions of safe startup, shutdown, power regulation and reactivity control of the reactor. The reactor core assembly, surrounded by beryllium reflectors, is located at the bottom of the

reactor vessel, which is suspended from the "I" beam structures that are embedded onto the reactor pool wall and can operate for a maximum of 4.5 hours at full power (i.e. equivalent to a thermal neutron flux of $1x10^{12}$ $n.cm^{-2}.s^{-1}$ in the inner irradiation channels); (Balogun and Jonah, 2004). Under these conditions, the reactor fuel loading can run for over ten years with a burn-up of less than one per cent.

The NIRR-1 is mainly divided into reactor complex, control system, auxiliary system, irradiation system, and water pool; (Jijin, 1996). The reactor complex consists of reactor core, beryllium reflectors, in-pile detector, moderator, coolant, and reactivity regulation as well as reactor vessel. The core is a 230mm by 230mm square cylinder with UAl_4 fuel clad in (303-Al) aluminium-alloy with a thickness of the clad material is 0.6mm as shown by figure 1.2.

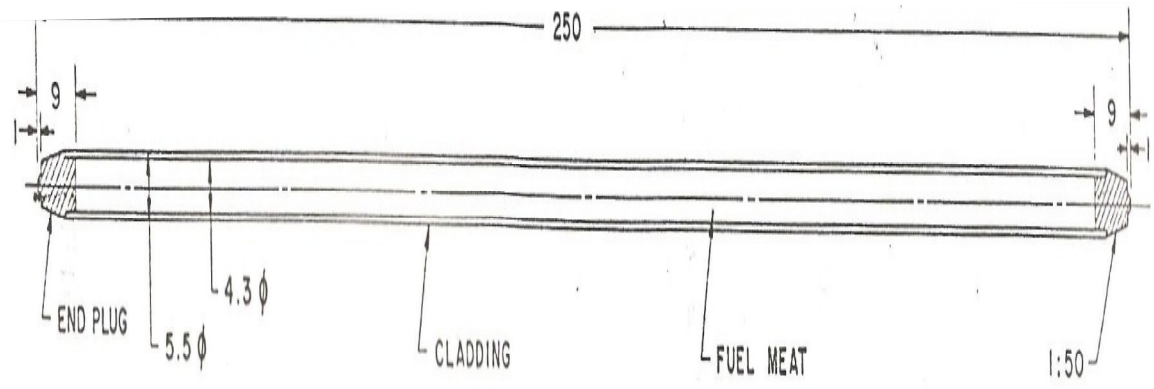


Figure 1.2: NIRR-1 fuel pin (dimensions are in mm)

The reactor is fueled with HEU fuel pins which are 90% enriched in U-235 with a total of 347 fuel pins and three dummy rods in the fuel lattice. The total fuel loading in the fresh core is about 1kg of U-235. Each fuel pin is 4.3mm in diameter and 248mm long with an active length of 230mm. The U-235 loading in each fuel meat is 2.88g which are secured by upper and lower grid plates to form a fuel cage.

The lower end of the fuel element is fixed to the lower grid plate by slightly conical self-locking fittings, while its upper end is free in the lattice of the upper grid plate to allow for expansion. Four out of the 355 lattices provided in the grid plates are used for fixing stainless steel tie rods to keep the fuel cage intact. Six other lattices are occupied by aluminium dummy pins. The central lattice is used for the central control rod guide tube, which is provided to facilitate the movement of the control rod. Figure 1.3 shows the core configuration. A thick beryllium reflector of about 10cm surrounds the core radially. The maximum nominal power of the reactor is about 30KW, the heat generated being removed by natural convection; (Jijin, 1996).

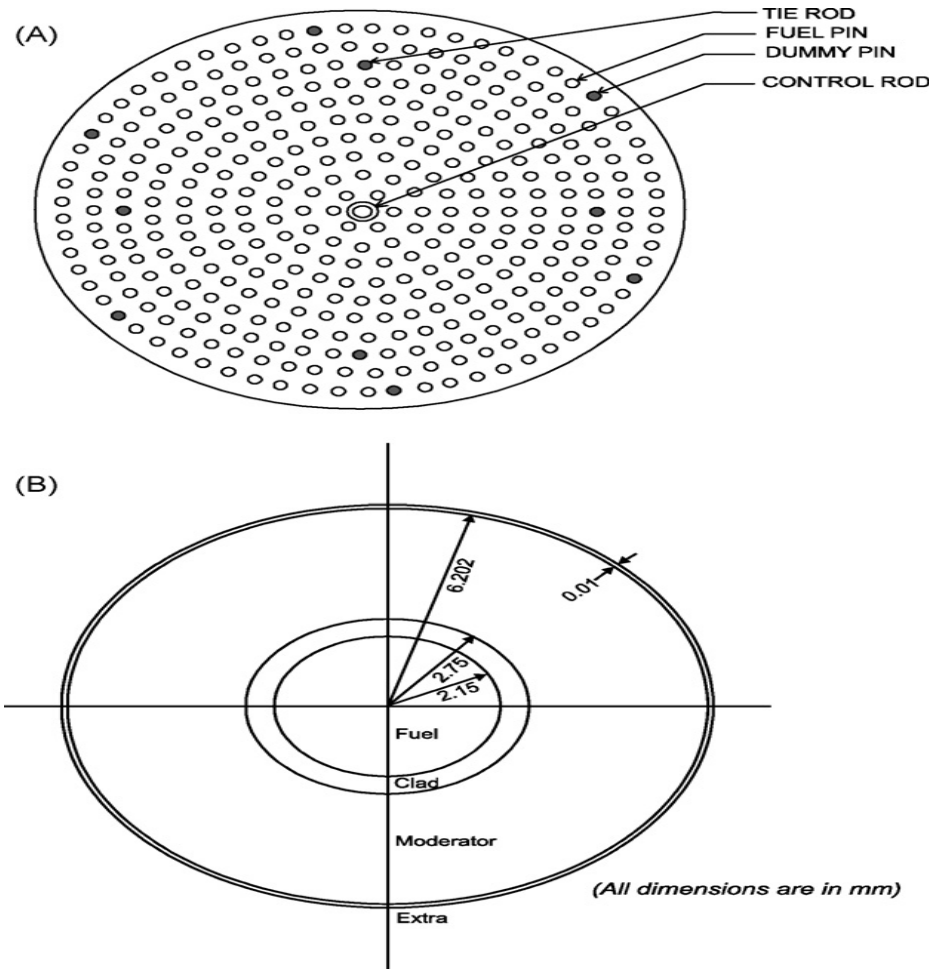


Figure 1.3: NIRR-1 Cross-sectional views of: (a) fuel pins, tie-rods and control-rod (top) and (b) fuel cell used for cross section generation (bottom).

The travel length of the rod in the core is 230mm. A weight of about 850g is attached to the control rod to facilitate its movement. The control rod moves through a hole in the top reflector shim tray having a travel length of about 230mm. Loss of power to its drive mechanism releases the rod to fall into the core under gravity; the total rod drop time from fully withdrawn position to fully inserted position is 26.5s.

A detailed description of the HEU core of NIRR-1 can be found in the Final Safety Analysis Report (FSAR); (SAR, 2005).

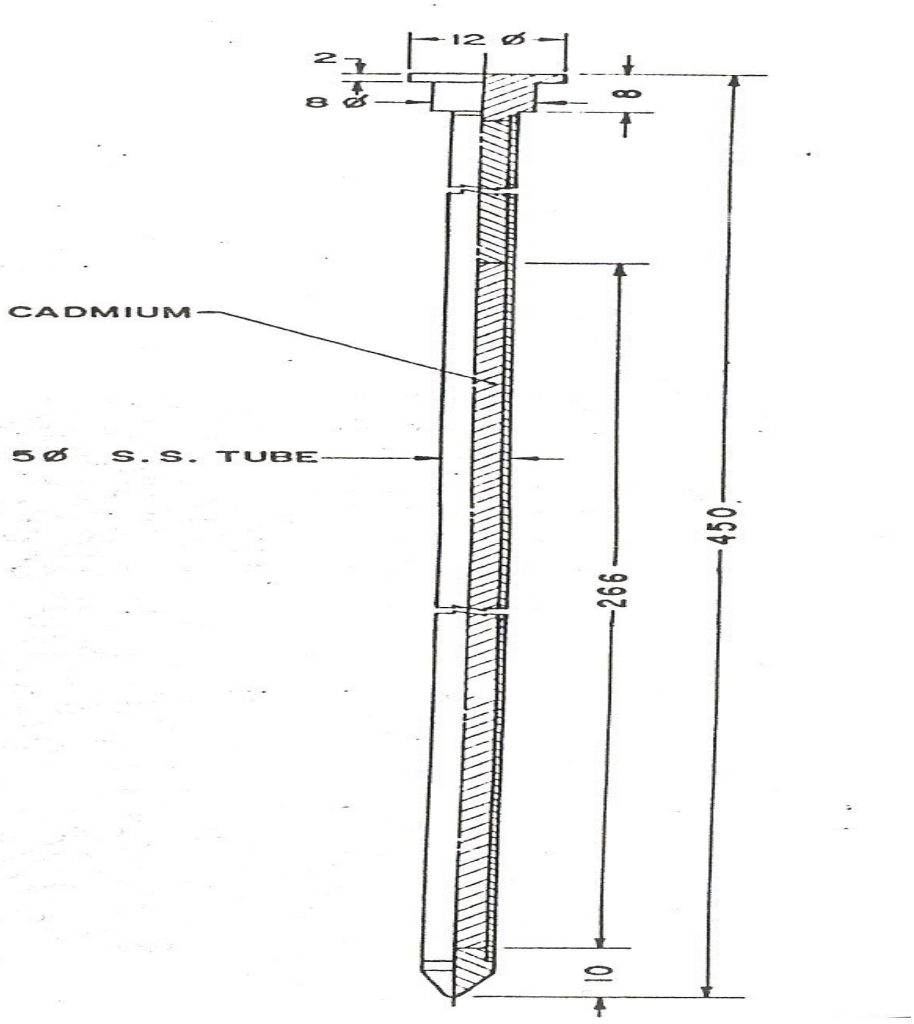


Figure 1.4: The Control Rod of Typical MNSR Reactor (dimensions in mm)

1.3 WINFRITH'S IMPROVED MULTI-GROUP SCHEME (WIMS)

The Winfrith's Improved Multi-group Scheme (WIMS) is a major lattice software package containing a wide range of lattice cell and burn-up methods for the design and development of all types of thermal reactors including experimental low power facilities as well as commercially operating power reactors. The lattice code WIMS (Askew et al, 1966; Halsall, 1998) was written to provide a comprehensive scheme of reactor lattice cell calculations applicable to a wide range of reactor types including both thermal and fast reactors. Halsall, (1998) added that the modular code system is designed for use by all levels of users' expertise to solve problems ranging from simple homogeneous systems through to the most complex 3-D whole core geometries. Newton and Hutton, (2002) emphasized that central to the aim of WIMS is to use a single physics model treatment of the resonance region that is capable of representing all types of reactors within a common framework. Askew et al, (1966) agreed that the aim has been to use fundamental nuclear data only and a noteworthy feature of the scheme is the ability to perform calculations to different degrees of approximation automatically at the discretion of the user. Newton and Hutton, (2002) emphasized that WIMS is still one of the few codes capable of dealing with graphite, heavy water and light water moderated systems as well as problems involving more than one moderator in the reactor.

The origin of WIMS can be traced back well over 45 years; as long ago as 1964, the first version of the WIMS code was being developed by Winfrith's reactor

physicists; (Askew et al, 1966). (Askew and Roth, 1982) stressed that because of its sound theoretical basis, and its free availability, the reference version of that early code, WIMSD, is still probably the most widely used lattice physics code in the world. Limitations of WIMSD, led to the development of the WIMSE modular scheme, which has been developed into the present sequence of codes that started with WIMS6 in 1992; (Halsall et al, 1994) followed by WIMS7 in 1996, and WIMS8 in 1998; (Halsall, 1996; 1998). The latest version (WIMS9A) has now been released and offers a range of new capabilities. In the early 1970's, a parallel development known as LWRWIMS was introduced for square assembly Light Water Reactors (LWR) geometries; this too was based on the modular concept of WIMSE. Halsall (1982) concluded that the features of LWRWIMS were unified and incorporated into the general WIMS code sequence from the issue of WIMS7 onwards. In addition, Halsall (1982) pointed out that a development of a companion ANSWERS code, the point in energy Monte Carlo code MONK8, has also been integrated into WIMS to run with the broad group cross-section data generated using WIMS.

The treatment of resonances in WIMS is based on the use of equivalence theorems; (Newton, 2004) which relate the library of accurately evaluated resonance integrals for each resonance absorber in each region to a particular heterogeneous problem; and to allow the effect of problem geometry to be exactly represented, WIMS uses sub-group theory which interprets the value of potential absorption cross sections that is appropriate to a specific problem; (Newton, 2004). The library

of homogeneous resonance integrals has been formed through the use of SDR (Spin Dependent Recombination) program; (Brissenden and Durston, 1965). The SDR program solves by numerical methods the slowing down problem associated with moderators homogeneously mixed with resonant isotopes and 12000 energy mesh points are used in this numerical solution; (Askew et al, 1966).

The resonance integral library data in WIMS refers to the calculations performed with the given resonant isotope in the presence of pure hydrogen as the moderator. The equivalent theorems employed take account of non-hydrogenous moderators which may be mixed with the fuel such as oxygen or carbon, and also include a first order correction for the interaction associated with the presence of several resonant isotopes; temperature effects are also included by providing in the library SDR results obtained from temperature broadened cross-section data, (Askew et al, 1966).

1.4 CITATION

CITATION (Fowler et al, 1989) is a core analysis code designed to solve problems using finite difference representation of the neutron diffusion theory treating up to three space dimensions with arbitrary group to group scattering. CITATION solves the neutron flux eigenvalue problem by direct iteration to determine the multiplication factor k, or the nuclide density required for a critical

system. Balogun (2003a) emphasized that only one Citation base input data need to be prepared from group constants generated by WIMS and any of these calculations (super- or sub-criticality test, control rod worth, core excess reactivity and shutdown margin, locating control rod's critical depth of insertion, control rod calibration, beryllium shims calibration, verification of some safety criteria, modifying core dimensions interactively, modifying beryllium reflectors dimensions interactively and calculating various temperature coefficients of reactivity) is carried out automatically, thereby eliminating the tedium and errors that usually attend manual data manipulation or regeneration. With this development, calculations that used to take days may now be more reliably done at most within hours.

1.5 STATEMENT OF THE PROBLEM

The international Reduced Enrichment for Research and Test Reactors (RERTR) program has recently advanced a cause for conversion of reactors with highly enriched uranium (HEU) to low enriched uranium (LEU). The reason for this global trend is to make test and research reactor fuel as unattractive as possible to groups that may be interested in using such highly enriched cores for non-peaceful uses. Miniature neutron source reactor (MNSR) is one of the many research reactors in the world that uses HEU as fuel. The Nigerian Research Reactor-1 (NIRR-1) falls in the category of HEUs, being 90% enriched in the fissile U-235.

1.6 JUSTIFICATION

There is now a broad international consensus about the importance and urgency of consolidating and reducing the stockpiles of nuclear-weapon materials located around the world. These measures are important because they lower proliferation risks due to the potential diversion or theft of nuclear materials. At the same time, reducing and eliminating excess stocks of these materials strengthens the non proliferation regime in general and also supports irreversible nuclear disarmament. Highly enriched uranium (HEU) has attracted particular public and political attention with its several characteristics that make it the material of choice for low-tech proliferators. It is relatively easy to handle and conceal due to its low level of radioactivity and, more importantly, only HEU can be used in the most basic weapon-design. There is now a broad international consensus (IAEA, 2002 §4.12) that this material be removed from the nuclear fuel cycle as soon as possible. Consequently, the world over, research reactors are being converted from HEU to LEU. The Nigerian Research Reactor-1 (NIRR-1) falls in the category of HEUs, being 90% enriched in $^{235}U_{92}$. There is therefore a need to convert this core to universally acceptable enrichment.

1.7 OBJECTIVES OF THE STUDY

The aims and objectives of this research work are:

I. To acquire practical skills needed as part of training for a nuclear reactor analyst/designer.

II. To investigate the fuel composition required to convert HEU MNSR core to LEU enrichment using WIMS and CITATION.

III. To maintain as far as possible the technical specifications and operating procedures of the present HEU core for the suggested LEU fuel.

IV. To automate the calculations of important parameters at each stage of the core conversion.

1.8 RESEARCH QUESTION

The efficacy of the scheme employed is demonstrated by automatically performing the following design calculations:

a. Ensure that the MNSR-type mockup is neither always super- nor always sub-critical.

b. Compute its control rod worth.

c. Estimate the excess reactivity and shutdown margin.

d. Locate the critical depth of insertion of the control rod.

e. Generate the integral rod curve data.

f. Calculate some safety criteria of this reactor type and

g. Modify core dimensions interactively amongst other calculations.

These calculations must be carried out at some intervals during the design of the reactor. The first two calculations are performed to answer two simple questions:

i. With the control rod fully withdrawn, is the mockup supercritical?

ii. When the control rod is fully inserted, is the mockup subcritical?

1.9 BASIC ASSUMPTIONS

For the design process to proceed, the answer to the above questions must be affirmative. This means that there exists a control rod depth of insertion (between fully inserted and fully withdrawn) for which the reactor is just critical. Failure of any of the two tests would require that the composition of the reactor under design be reviewed.

Having passed the tests, safety consideration requires that some safety criteria, including the safety reactivity factor (SRF) and shutdown margin (SDM) must also be established. These two parameters must satisfy some conditions according to a provision of the International Atomic Energy Agency (IAEA).

1.10 SIGNIFICANCE OF THE STUDY

This study facilitates the automation of the required calculations associated with reactor core conversion with huge savings in time and effort with the results that:

– It enables the expert reactor analysts/designers to carry out their duties more effectively in that:

• Errors associated with repetitive data input are eliminated

• Calculations that can take years may be done at most within hours.

– This also means that the non-expert (that is a beginner) reactor analyst/designer may embark on reactor design/analysis calculations with a minimum of effort.

This work is a measure to ensure that after the first fuel cycle, NIRR-1 should not be shutdown but should be able to adapt a new core (LEU). This will give a greater opportunity to many Nigerian researchers to utilize the reactor for a long time. Also the study shows that the principles and theories discussed in the classroom can practically be translated into workable systems. Furthermore the research imbibes the attitude of self-confidence in the researcher towards actualizing the spirit of independent research work.

1.11 SCOPE AND DELIMITATION OF THE STUDY

This is to investigate the down blending of surplus Highly Enriched Uranium (HEU), which has a greater than 20% concentration of U-235 or U-233 to Low Enriched Uranium (LEU), which has less than 20% concentration of U-235 to make it suitable for use in commercial nuclear fuel; maintaining as far as possible the technical specifications and operating procedures of the present highly enriched uranium core. Only two fuel matrices namely the present UAl_4 and a potential UO_2 fuels were considered. Regulatory policies and financial constraints are the major limitations of this research work.

CHAPTER TWO

REVIEW OF RELATED LITERATURE

2.0 INTRODUCTION

In this chapter, the relevant literature that has been reviewed is presented with particular emphasis on those that addressed the subject of core conversion from HEU to LEU. Nuclear reactor cores are made up of various components and these are briefly discussed. Also discussed are some of the important aspects of core conversion studies in relation to the material composition of the core.

2.1 URANIUM ENRICHMENT

Uranium, with chemical symbol U is one element that occurs in nature and has been the raw material for nuclear fuel. The property of uranium, particularly the uranium-235 isotope, important for nuclear power is its ability to fission, or split into two lighter fragments when bombarded with neutrons and releasing large amounts of energy in the process. Natural uranium occurs as a mixture of three different isotopes. These isotopes are uranium-234 (the highly radioactive trace component), uranium-235 (the only fissile material that occurs in nature in significant quantities), and uranium-238 (the most plentiful isotope; about 99.284

percent by weight in natural uranium but it is not fissile). However U-238 is fissionable and can thus be split by high energy neutrons, releasing large amounts of energy in the process.

Because of the presence of small quantities of U-235, natural uranium can sustain a chain reaction under certain conditions, and therefore can be used as a fuel in certain kinds of reactors, the so-called heavy water reactors. For the most common reactor type in use around the world today (the light water reactor LWR), which uses ordinary water as a coolant and moderator, the percentage of U-235 in the fuel must be made higher than the 0.7 percent found in natural uranium by a process called enrichment (i.e. increasing the percentage of the fissile isotope U-235) in order to be able to sustain the chain reaction necessary in a nuclear reactor; (McKinley and Balkany, 2004: Glaser, 2005b).

2.2 HIGH-ENRICHED URANIUM AND LOW-ENRICHED URANIUM

Two major uranium isotopes naturally occur in appreciable concentrations. These are U-238 and U-235, with natural isotopic fractions of 99.29% and 0.71% respectively. The relative concentration of these two isotopes, that is the enrichment or U-235 weight fraction of a given material stock, can be changed with a variety of isotope separation techniques exploiting physical effects to separate the species. As a matter of fact, the enrichment level determines the main characteristics of any

uranium composition both for reactor use as well as for weapon use; (Glaser, 2005a). U-235 displays a high probability of fission after neutron absorption throughout the entire energy range from thermal to fast neutron energies. On the other hand, U-238 is fissionable only above threshold energy of about 1MeV. Below this threshold energy, neutron capture dominates the total absorption cross section of this isotope; (Glaser, 2005).

Uranium containing at least 90 percent U-235 has been used to make nuclear weapons and uranium with this level of enrichment is called highly enriched uranium (HEU). The HEU fuel meant for research reactors is considered particularly vulnerable to diversion for use in nuclear weapons because it is generally less well-guarded, often located in cities or on university campuses.

The production of HEU began during the World War II within the U.S. *Manhattan project*. Although HEU was available at an early stage and employed in the nuclear weapon that destroyed Hiroshima on August 6, 1945, the production capacity was extremely low at that time; Glaser (2005a). Only after the World War II, the large U.S. enrichment facilities under construction were completed and these facilities were based on the gaseous diffusion process, in contrast to the calutrons used during the war; (Krass et al., 1983). The U.S. annual production rate of HEU and Plutonium peaked in the early 1960s at 80 metric tons and 60 metric tons respectively; (Albright et al., 1997).

An inventory of about 200-300 metric tons can be assumed to be absorbed in

deployed nuclear weapons Worldwide; (Glaser, 2004). Albright et al. (1997) agreed that indeed, most of the U.S. excess weapon-grade HEU is being placed in reserve for use in naval reactors. This stockpile is enough to fuel the entire U.S. nuclear-powered fleet for many decades; (ONNP, 1995) and should be well above 100 metric tons. HEU fuel continues to be used in about 150 nuclear-powered submarines and military surface vessels; (Chunyen and Von Hippel, 2001). Additionally on the civilian side, there are seven Russian nuclear icebreakers and cargo ships operated by *Murmansk shipping company* that consume about 500kg of HEU annually; (Bukharin, 2002). The total amount of HEU still present in the civilian nuclear fuel cycle, which includes fresh and irradiated but not yet shipped-back fuel, has been estimated to be approximately 50 metric tons; (Albright and Kramer, 2004). This material is largely stored as fuel elements in wet or dry storage at reactor sites or interim storage facilities. The most recent survey on spent fuel from research reactors, performed under the auspices of the IAEA and based on 210 out of about 550 reactors, listed 22,686 HEU and 40,184 LEU fuel elements stored worldwide; (Ritchie, 1998) and additional 32,932 assemblies were located in reactor cores.

The definition of *Low Enriched Uranium* (LEU) was first used by the U.S. Atomic Energy Commission in or prior to 1995. The same convention was later also adopted by the International Atomic Energy Agency (IAEA), which defines low-enriched uranium as “enriched uranium containing less than 20% of the isotope U-235”; (IAEA,2002 §4.12). The IAEA classifies LEU as a so-called *indirect use*

material, which in turn is defined as a nuclear material that cannot be used for "the manufacture of nuclear explosive devices without transmutation or further enrichment"; (IAEA, 2002, §4.25 and §4.26).

From a technical point of view, the choice of the LEU limit is to some extent arbitrary. Likewise, the adequacy of the conversion goal for research reactors just below that limit, usually 19.75%, is by no means obvious. Two factors are central in the process of defining the optimum enrichment level for research reactor fuel and from a nonproliferation perspective: the usability of the fresh or irradiated fuel and the concurrent and inevitable plutonium production in the fuel during irradiation; (IAEA, 1980b). Travelli, (1978) acknowledged that the proliferation resistance of nuclear fuels used in research and test reactors can be considerably improved by reducing their uranium enrichment to a value less than 20%, but significantly higher than natural to avoid excessive plutonium production. Similar arguments are used in the International Nuclear Fuel Cycle Evaluation (INFCE) documents; (IAEA, 1980a, Vol 8, section 12).

2.3 FERTILE MATERIAL AND FISSION CHAIN REACTION

All of the neutron absorption reactions that do not result in fission lead to the production of new nuclides through the process known as *transmutation*; (DOE-HDBK, 1993). These nuclides can, in turn, be transmuted again or may undergo

radioactive decay to produce still different nuclides. The nuclides that are produced by this process are referred to as transmutation products. Because several of the fissile nuclides do not exist in nature, they can only be produced by nuclear reactions (transmutation). The target nuclei for such reactions are said to be fertile. *Fertile materials* are materials that can undergo transmutation to become fissile materials; (DOE-HDBK, 1993). Figure 2.1 traces the transmutation mechanism by which two fertile nuclides, thorium-232 and uranium-238, produce uranium-233 and plutonium-239, respectively.

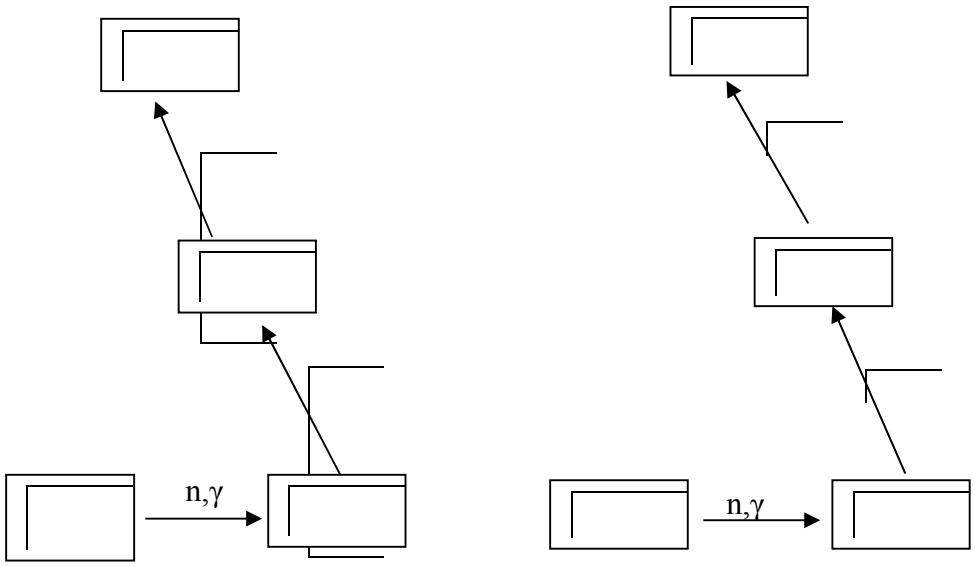


Figure 2.1: Conversion of Fertile Nuclides to Fissile Nuclides

If a reactor contains fertile material in addition to its fissile fuel, some new fuel will be produced as the original fuel is burned up. This is called *conversion*. Reactors that are specifically designed to produce fissionable fuel are called "breeder" reactors. In such reactors, the amount of fissionable fuel produced is

greater than the amount of fuel burn up. If less fuel is produced than used, the process is called conversion, and the reactor is termed a "converter"; (DOE-HDBK, 1993). The most important reaction in reactor physics is the fission chain reaction shown in the figure (2.2) below.

Although the heavy uranium isotope uranium-238 does not sustain a chain reaction, it can be converted into the fissile plutonium-239 by neutron capture followed by two levels of beta decay as shown in figure (2.1) above. Plutonium-239, like uranium-235, undergoes fission after the absorption of a neutron. Producing plutonium-239 in large quantities requires an intense source of neutrons; the source is provided by the controlled chain reaction in a nuclear reactor. This fission in U-238 occurs only for neutrons of energy above 1.1 MeV; and this has a comparatively minor effect on the sustaining of the cross sections at low energies; (Hunt, 1987).

Hunt (1987) concluded that the problem therefore is to slow the fission spectrum neutrons from their initial energies in the MeV range to energies of a fraction of an eV without losing too many of the neutrons in the ‘resonance trap’ of U-238. This is achieved through the process of moderation (repeated elastic collisions with light nuclei). Bennet (1972) pointed out that it would not be possible to sustain a chain reaction in a homogeneous assembly, that is, intimately mixed uranium fuel and moderator (say in powdered form) since losses of neutrons to the U-235 resonance trap would be too great.

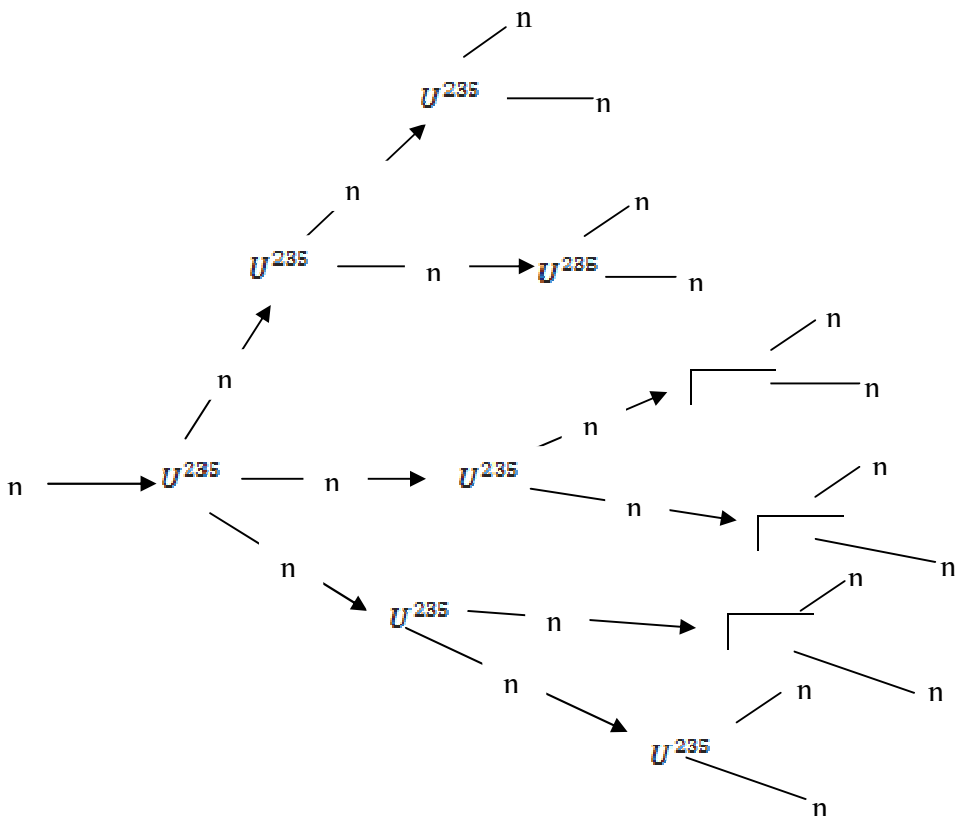


Figure 2.2: Fission chain reaction

A mathematical treatment of fission chain reaction is facilitated by considering the fission processes to be divided into 'generations'. Hunt (1987) agreed that this slightly artificial concept implies that a number of fission processes occur simultaneously in one generation; these provide the fission neutrons which are moderated and after some losses become the thermal neutrons which initiate the next generation of fission processes. To achieve a self-sustaining chain reaction based on a fuel containing a mixture of both isotopes, that is, based on a fuel of

given enrichment, only U-235 is immediately useful in a thermal spectrum present in a light-water cooled and moderated reactors. The situation is fundamentally different in a fast neutron spectrum, which is typical for fast reactors, but also relevant for nuclear weapons; (Glaser, 2005a). A fast chain reaction is readily achievable for very high U-235 fractions. The situation becomes more complex once U-238 is present in significant amounts. Glaser, (2005b) added that in the unresolved resonance region above 10 KeV, the added capture cross sections of U-235 and U-238 begin to compete with fission in U-238. As a result, the average number of neutrons released per absorption can be expected to decrease notably. Raising U-238 fraction in a material, simultaneously promotes the relevance of inelastic scattering in the configuration, which reduces the mean energy of the neutrons in a fast spectrum. Once a significant fraction of the neutron population reaches the resonance region of U-238, neutron capture starts to dominate all other processes and for natural uranium, no un-moderated critical mass exists. The consequence of these phenomena is that the critical mass of uranium increases sharply as the enrichment of the material decreases; (Glaser, 2005a).

2.4 NEUTRON MODERATION AND MODERATOR PROPERTIES

To slow-down fission neutrons from their initial energies without losing too many neutrons in the resonance trap of U-238 is done through moderation, which is

the repeated elastic collisions with light nuclei. Hunt (1980) pointed out that for many neutron induced reactions including nuclear fission in U-235 increase as the neutron energies are reduced. For many purposes, Hunt (1987) stated that it is adequate and convenient to assume that all thermal neutrons have energy of 0.025eV and a mean velocity of $2200ms^{-1}$, which corresponds to the mean energy and velocity of thermal vibrations at 293K ($20^{o}C$) temperature. Hunt (1987) also opined that like other thermal motions, they have a Maxwell-Boltzmann velocity distribution so long as the neutron capture cross section of the medium is small compared with the scattering cross section.

The number of neutrons in unit volume with velocity between v and $v + dv$ is, according to Hunt (1987), given by equation (2.01).

$$n(v)dv = 4\pi N\left(\frac{m}{2\pi kT}\right)^{\frac{2}{3}} e - \frac{mv^2}{2kT} v^2 dv \ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.01)$$

Where N is the total number of neutrons per unit volume, m is the neutron mass, T is absolute temperature of the moderating medium and k is the Boltzmann constant. The most probable speed and kinetic energy is given by equations (2.01) and (2.02) respectively.

$$v_p = \sqrt{\left(\frac{2kT}{m}\right)} \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.02)$$

$$^1/_2\, mv_p^2 = kT \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.03)$$

After moderation the neutrons diffuse through the moderator, preserving their thermal energy spectrum until captured by the moderator or material placed within the moderator, or lost by leakage at the moderator surface.

The number of neutrons that pass through a unit area perpendicular to the direction of motion per unit time is called neutron flux and denoted by φ. If n(v)dv neutrons per unit volume with speed in the range from v to v + vdv, then the flux is given by equation (2.04).

$$\varphi = \int v_n(v)dv \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.04)$$

The integration is over all speeds in the neutron spectrum. However, for neutrons of the same speed v, equation (2.05) holds.

$$\varphi = nv \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.05)$$

Hunt (1987) redefined neutron flux to be correctly applied to a beam of unidirectional neutrons as against its common use for the thermal neutrons moving in random directions as in a reactor. According to Hunt (1987), the correct term for neutrons moving in random directions is *neutron fluence*, that is, the number of neutrons per second incident on a small sphere divided by the area of the sphere. The flux of φ neutrons per unit area per second travelling as a parallel beam striking a target containing X nuclei per unit volume will be as in equation (2.06).

$$R_s = \sigma_s \varphi X \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.06)$$

But the number of molecules per unit volume is given by equation (2.07).

$$X = \frac{N_A \rho}{MW} \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.07)$$

Thus equation (2.06) becomes equation (2.08)

$$R_s = \sigma_s \varphi \frac{N_A \rho}{MW} \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.08)$$

Where N_A is Avogadro's number (N_A = 6.022 X 10^{23}). ρ is the macroscopic cross section and denoted by ∑. Hunt (1987) defined the microscopic cross section to be equal to the probability that an incident particle will interact per unit track length. Thus, equation (2.08) transforms to equation (2.09).

$$R_s = \sum \varphi \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.09)$$

The macroscopic cross section is the reciprocal of the mean free path λ_s, the average distance which a particle travels in the medium without interaction, as shown by equation (2.10).

$$\sum = \frac{1}{\lambda_s} \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.10)$$

The S.I. unit of flux is neutrons per square centimeters per second. A flux of neutrons of initial intensity φ_o will be attenuated exponentially according to equation (2.11).

$$\varphi_x = \varphi_o e^{-\Sigma x} \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.11)$$

Where x is the distance penetrated in the moderator.

As well as having a large energy transfer per collision, a good moderator should have a high cross section for elastic scattering, and the product $\sigma_s\xi$ is a better criterion of moderator performance than ξ alone; (Hunt, 1987). ξ is defined as the average logarithmic energy decrement per collision given by equation (2.12).

$$\xi = \ln\left(\frac{E_{before}}{E_{after}}\right) \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.12)$$

Hunt (1987) added that a good moderator should also have a high density of scattering nuclei per unit volume, and this is reflected in the definition of slowing-down power, which is equal to ξ/λ_s where λ_s is the mean free path. The slowing-down power, S_d, is given by equation (2.13).

$$S_d = \frac{N_A \rho \sigma_s \xi}{MW} \ (cm^{-1}) \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.13)$$

A further important parameter according to Hunt (1987) for the design of moderators is the slowing-down length of the neutrons during the moderating process, which is the net vector or "crow's flight" distance r_s travelled. The slowing-down length, according to Hunt (1987), determines the thickness of moderator required for thermalisation. After moderation to thermal energies, the neutrons diffuse through the moderator with random thermal motion, and the diffusion length

L is the net vector distance that the neutrons travel before being captured by the moderating material. The total net vector distance travelled by the neutrons between entering the moderator as fast neutrons and final capture by the moderator as thermal neutrons, is called the migration length; (Hunt, 1987). The migration length, denoted by μ, is related to the diffusion length L and the 'crow flight distance r_s as given by equation (2.14)

$$\mu = \sqrt{(L^2 + r_s^2)} \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.14)$$

Hunt (1987) said that the neutron diffusion length is inversely related to the capture cross section of the moderating material for thermal neutrons and should be as long as possible, since clearing in any application of the moderator it is undesirable for too many of the thermal neutrons to be captured by the moderator; this can be particularly important in some nuclear reactor designs. The moderator ratio $\xi\sigma_s/\sigma_c$ can be used as a figure of merit for a moderator since it involves the fraction of neutrons moderated to those captured; (Hunt, 1987).

Other properties of a good moderator according to Hunt (1987) are the slowing-down time which is the mean time which the moderating process takes, while the mean time for which the neutrons exist as thermal neutrons before capture by the moderator is the diffusion time; the sum of these is referred to as the neutron lifetime. Hunt (1987) pointed out that in the presence of materials which absorb neutrons strongly such as nuclear fuel assemblies; both the diffusion length and the

diffusion time are very much shorter. Hunt (1987) said that in most respects, water is the ideal moderator except for its high cross section for capture of slow neutrons by the H(n,γ)D reaction. This prevents water from being used as a moderator in natural uranium reactors which it is necessary to minimize neutron loss in order to maintain the fission chain reaction. Although the high capture cross section of ordinary (light) water is too high to permit its use in natural uranium reactors, it can be used in slightly enriched reactors where the need for neutron economy is not so great; (Hunt, 1987). Hunt (1987) concluded that graphite or heavy water must be used in this case although their moderating characteristics are otherwise inferior.

2.5 CROSS SECTIONS

To determine the frequency of neutron interactions, it is necessary to describe the availability of neutrons to cause interaction and the probability of a neutron interacting with material. The availability of neutrons and the probability of interaction are quantified by the neutron flux and nuclear cross section. Fission neutrons are born with an average energy of about 2 MeV and these fast neutrons interact with the reactor core materials in various absorption and scattering reactions; (DOE-HDBK, 93). Collisions that result in scattering are useful in slowing neutrons to thermal energies. Thermal neutrons may be absorbed by fissile nuclei to produce more fission or be absorbed in fertile material for conversion to fissionable fuel. The absorption of neutrons in structural components, coolant, and other non-

fuel materials results in the removal of neutrons without fulfilling any useful purpose; (DOE-HDBK, 93).

Cross section is a measure of the probability of an interaction such as fission or ionization occurring between two elementary particles. The cross sections refer to the probability that a specific event takes place, the capture cross sections referring to the probability that a particle is captured by the target nucleus while reaction cross sections refer to the probability that a specific complete reaction will take place; (Glaser, 2005a). The probability of a neutron interacting with a nucleus for a particular reaction is dependent upon not only the kind of nucleus involved, but also the energy of the neutron; (DOE-HDBK, 93). Accordingly, the absorption of a thermal neutron in most materials is much more probable than the absorption of a fast neutron and also, the probability of interaction will vary depending upon the type of reaction involved. The probability of a particular reaction occurring between a neutron and a nucleus is called the *microscopic cross section* (σ) of the nucleus for the particular reaction. The *macroscopic cross section* is the probability of a given reaction occurring per unit travel of the neutron. This cross section will vary with the energy of the neutron.

The cross sections values σ are derived from both experimental data as well as from theoretical models, if direct experimental measurement is impossible; (Glaser, 2005a). Classically, the nucleus is represented by an imaginary disc of cross sectional area σ. If there are N such discs per unit volume in a target of

thickness *t*, the number of discs per unit area of target Nt will occupy a fraction $Nt\sigma$ of the total target area. If the target is irradiated with n_o number of incident particles, by the simple law of probability there should be $n_o Nt\sigma$ collisions. Hunt (1987) defined reaction probability as being proportional to the cross section of the imaginary disc given by equation (2.15)

$$\sigma = \frac{number\ of\ events}{n_o Nt} \quad \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.15)$$

Because of the classical analogy, it is known as the cross section of nuclear reaction. The microscopic cross section may also be regarded as the effective area the nucleus presents to the neutron for the particular reaction. That is the larger the effective area, the greater the probability for reaction. Because the microscopic cross section is an area, it is expressed in units of area, or square centimeters, and are measured in barns per nucleus (1barn = 10^{-24} cm^2 = 10^{-28} m^2); (DOE-HDBK, 93).

If $\partial\sigma$ refers to the probability that a particle will be emitted within a small solid angle $\partial\Omega$ inclined at an angle of θ_e to the direction of the incident particle, then by definition the cross section per unit solid angle, $\frac{\partial\sigma}{\partial\Omega}$, is the differential cross section at the emission angle θ_e given by

$$\int_0^{4\pi} \frac{\partial y}{\partial \Omega} d\Omega = \sigma \quad \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.16)$$

Cross sections are affected by factors such as the reaction kinetics, the potential barrier for charged incident particles, resonance effects and shell structure. Cross sections are summarized and organized in the so-called "evaluated nuclear data files or libraries"; (CSEWG, 2001). To use these evaluated data libraries with specific neutronics codes, such as WIMS, the data has to be adequately formatted to suit the requirements of the code and of the specific problem to be solved. To safely and efficiently operate a nuclear reactor it is necessary to predict the probability that a particular absorption or scattering reaction will occur. Once these probabilities are known, if the availability of neutrons can be determined, then the rate at which these nuclear reactions take place can be predicted; (DOE-HDBK, 93).

2.6 MULTI-GROUP ENERGY SCHEME

Neutrons have a wide energy spectrum, ranging from a fraction of an eV to a few MeV. The cross sections vary over decades in this range so it can hardly be expected that one group approximation be very accurate. One group theory does not come close to predicting the buckling, even if the cross sections are varied within their experimental error; (Garland, 2005). The two-group model can be further improved by using energy-averaged cross sections obtained by a comprehensive cell code that employs a detailed energy structure. This work is detailed in (Garland, 1975) but the main point to note is the inadequacy of the one-group model or even the two-group model since the appropriate cross sections are not explicitly available and since these low order models do not come close to capturing the energy

structure. Accurate cell calculations are done typically with up to 150 neutron energy groups to obtain cell-averaged cross sections and then, few-group approximations are used for the full core calculation based on the cell averaged cross sections; (Garland, 2005). Few-group calculations can be successfully done but only if they are backed up by detailed multi-group cell calculations. To form the multi-group neutron diffusion equations the energy range for the neutrons is first divide up into groups; (Garland, 2005).

2.7 CRITICAL MASS, CRITICALITY AND REACTIVITY

Critical mass is the minimum amount of a given fissionable material necessary to achieve a self-sustaining nuclear chain reaction under specified conditions. This implies that critical mass corresponds to precisely the amount of material in a defined configuration that is required to maintain a self-sustaining neutron chain reaction. Glaser (2005b) agreed that the critical mass represents a reasonable amount of material required to construct a nuclear weapon or explosive device using material of a given composition. It depends on several factors, including the kind of fissionable material used, its concentration and purity, and the composition of geometry of the surrounding reaction system. The critical mass of uranium increases sharply with decreasing enrichment. Most importantly; Glaser (2005a) emphasized that a low critical mass simplifies the assembly process of the final super-critical configuration, a process that is extremely time-critical and

requires extreme acceleration of the previously sub-critical components. This means that a low critical mass also reduces the amount of material that has to be produced, diverted, or otherwise acquired.

Criticality refers to the condition of a nuclear reactor which is just capable of sustaining a nuclear chain reaction. Gilphin (1982) defined reactivity as the measure of ease of ignition from cold and of the relative rate of combustion under specified conditions. (Hunt, 1987; Garland, 2004) defined a multiplying factor *k* of a reaction as the number of fission processes in a given generation compared with that in the previous generation or, in mathematical terms,

$$k = \frac{fissions\ in\ generation\ (x+1)}{fissions\ in\ generation\ x} \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.17)$$

The primary objective of nuclear operations is to safely operate at constant power for the most part; this implies the need for control of the nuclear fission reaction; (Garland, 2004). Equation (2.17) shows clearly that if the value of *k* is greater than unity, then the number of fission processes is growing from generation to generation and the state of such a reactor assembly is called supercritical. If *k* is equal to unity, the fission rate is constant, that is, the fission chain is just continuing at a steady state and the assembly is then called critical; in a subcritical assembly, *k* is less than unity, the fission rate is falling and the chain reaction will rapidly die away; (Hunt, 1987). It is unfortunate that the name 'critical' is associated with steady state (k=1);

when a reactor is critical, it is a status quo situation but does not imply that there is a crisis; (Garland, 2004).

Hunt (1987) defined the reactivity Δk of a system by equation (2.18) as

$$\Delta k = \frac{k-1}{k} \quad \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.18)$$

And if Δk is positive, the assembly is supercritical. $\Delta k = 0$ corresponds to the critical condition and negative Δk to the subcritical condition. The size of the assembly for which k_{eff} is equal to unity, that is, when a sustained chain reaction just becomes possible, is called the critical size of the assembly; (Hunt, 1987). Neutron leakage, and with it critical size, may be reduced by surrounding the reactor core (part of a nuclear reactor containing the fissile material) with additional moderating material which acts as a reflector, returning some of the escaping neutrons to the reactor assembly; (Hunt, 1987). Gilphin (1982) defined a reflector as a layer of material surrounding the core of a nuclear reactor, whose purpose is to reduce the escape of neutrons by means of scattering processes which result in the return of many of the neutrons to the core. A reflector material should have a high neutron-scattering cross section and a low neutron capture cross section. Reflectors have the added advantage that escaping fast neutrons are moderated before being reflected back into the assembly and both reflectors and the reflected thermal neutrons augment the neutron flux near the reactor surface, producing a more uniform flux throughout the reactor volume; (Hunt, 1987). In designing a commercial reactor, a just critical

assembly would not be very useful, since allowance must be made for the loss of fissile materials during operation, for the build-up of fission product poisons and for the control of the reactor. These all require that some excess reactivity must be designed into the initial reactor core.

The central problem of a criticality calculation is the determination of the effective multiplication factor, k_{eff} of the system. k_{eff} of exactly one, less than one, or greater than one corresponds to criticality, sub-criticality, and super-criticality, respectively. By introducing the infinite multiplication factor, k_∞ for the extension, and therefore without neutron leakage, Glaser (2005a) said that one can also express the k-values in terms of the effective macroscopic cross sections of the system.

$$k_\infty = \frac{production}{absorption} = \frac{\bar{\nu}(\sum f)}{(\sum a)} \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.19)$$

And

$$k_{eff} = \frac{production}{absorption + leakage} = P_{nl} k_\infty \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.20)$$

These expressions for k_∞ and k_{eff} appear deceptively simple as the fundamental quantities, that is the effective macroscopic cross sections of the entire system and the non-leakage probability P_{nl}, are of course unknown; and their determination would be equivalent to the solution of the fundamental transport equations. Glaser (2005a) added that diffusion theory can be used to derive formulae for k_∞ and k_{eff}

that are valid under certain conditions and for specified simple geometries and most generally, k_{eff} can be written in an integral form as:

$$k_{eff} = \int_V \int_0^{\infty} \int_E \int_\Omega v\Sigma_{fis}\,\phi d\Omega dE dt dv \cdot \left[\int_V \int_0^{\infty} \int_E \int_\Omega \Sigma_{abs}\,\phi + (\Delta . \vec{j}) d\Omega dE dt dv\right]^{-1} \quad \ldots (2.21)$$

The fundamental theoretical problem of calculating the k_{eff} of any nuclear system, which is not exactly critical, is population control. Specifically, in a super-critical system, an exponential growth of the neutron population has to be prevented cycle-by-cycle of the neutron population, that is, without biasing the real k_{eff} of the system; (Cullen et.al. 2003).

2.8 NUCLEAR FUEL ASSEMBLIES

Bukharin (2002) described nuclear fuel assemblies for research reactors to be typically plates or cylinders of uranium-aluminium alloy (U-Al) clad with pure aluminium. These fuels are different from the ceramic UO_2 pellets enclosed in Zircalloy cladding used in power reactors. For use in heavy water reactors (HWR) as well as light water reactors (LWR), the uranium is needed in the form of UO_2; in Magnox reactors (graphite moderated, gas cooled), the uranium is needed in the form of uranium metal. Hunt (1987) concord that any natural uranium reactor design must be heterogeneous, that is, consist of uranium fuel in the form of rods separated by comparatively large volumes of moderating material. The design of a plate-type fuel compared to typical pin-type fuels is fundamentally different with

respect to the retention of fission gases released during irradiation. In the case of pin-type fuels, fission gases are mainly accommodated by gaps between the fuel pellets and the cylindrical fuel cladding. Glaser, (2005a) added that in plate-type fuels, cladding and fuel zone (meat) are bonded, and a different mechanism has to be provided to accommodate the gaseous fission products and to support the swelling of the fuel.

Any uranium composition with U-235 content of at most 20% (low enriched uranium LEU) is classified as direct-use material, while uranium used in nuclear weapons is typically enriched to more than 90% (weapon grade uranium WGU). In spite of these facts and based upon data published in the open literature, it is nevertheless difficult to assess the net strategic value of a given uranium stock of intermediate enrichment, that is between 20% and 90%. The question of enrichment was a major focus of the UN- sponsored International Nuclear Fuel Cycle Evaluation in 1980 and it concluded that enrichment should be reduced to not more than 20% U-235. The nuclear weapon states currently consider progress in nuclear disarmament of secondary relevance, arguing that the proliferation of nuclear terrorism and the proliferation of nuclear weapons pose the most serious threat to global security. Nevertheless, it is widely recognized that a revitalization of the nuclear disarmament process is essential to reduce the demand for these weapons and a precondition to prevent proliferation of nuclear weapons in the long term; Glaser, (2004). Most recently, this requirement has been re-emphasized in the final

report of the U.N. high level panel on threats, challenges, and change; (United Nations, 2004)- and the fundamental bargain underlying the nonproliferation treaty.

In spite of the current stand still in nuclear disarmament, there are fortunately important related areas where the objectives and interests of the international community clearly coincide. In particular, there is now a broad international consensus about the importance and urgency of consolidating and reducing the stockpiles of nuclear-weapon materials located around the world; Glaser, (2005b). These measures are important because they lower proliferation risks due to the potential diversion or theft of nuclear materials. At the same time, reducing and eliminating excess stocks of these materials strengthens the nonproliferation regime in general and also supports irreversible nuclear disarmament.

The current civilian HEU stockpile has been estimated to be about 50 metric tones, which is much less than the inventory reserved for military purposes, but still enough for several thousand nuclear weapon or exclusive devices. Glaser, (2005a) agreed that virtually all of the civilian weapon-grade uranium is associated with the present or former use in HEU-fueled research reactors. During the 1960s, a large number of research reactors started to use HEU and, as a result, almost 50 countries received HEU fuel to fuel their facilities. Many HEU-fueled reactors have been shut down or converted to LEU fuel but more than 100 reactors worldwide still use HEU in their cores. 78 HEU-fueled reactors have been excluded from the Global Threat

Reduction (GTRI) Conversion Program scope for a variety of reasons, including (1) classification as defense related facilities, (2) location in countries that currently do not collaborate with the United States on reactor conversion programs, and (3) requirements for very specialized LEU fuel which would be too costly and time consuming to develop; (Staples and Butler, 2007). Since the inception of the Conversion Program, 48 of the 129 reactors have been converted to LEU fuel or have shutdown prior to conversion.

Glaser, (2006) stated two-pronged approach required to eliminating HEU from civilian nuclear fuel cycle. First, legacy materials stemming from the former use in research reactors which are usually still being stored at the respective reactor sites worldwide, have to be consolidated and, ultimately, down-blended to low enrichment. Second, the remaining operational HEU-fueled research reactors in the world have to be converted to LEU to eliminate the demand for fresh HEU. The latter has significantly dropped since the creation of the Reduced Enrichment for Research and Test Reactors (RERTR) Program in 1978, but still amounts to about one metric tone per year; (Glaser, 2004).

Major national and international programs have been launched recently to carry out a complete global 'cleanout' of HEU and other high risk materials. Similarly, for the first time, the conversion of all research reactors worldwide has been defined as an explicit objective with a target date for completion within a decade. However, while the global 'cleanout' of HEU is an uncontroversial

undertaking, the conversion of research reactors is a more complex technical and administrative process, because the interests of research reactor operators and users are involved; Glaser, (2005a). Potential criteria to guide a conversion process include minimum reactor modification, minimum changes in operational characteristics and neutron flux values, minimum licensing problems, and minimum fuel cycle costs; (IAEA, 1980b).

From a technical perspective, HEU fuel is always superior to LEU fuel due to higher concentration of fissile U-235 and the parasitic absorption in U-238 – both characteristics that however also explain the weapon-usability of HEU. To overcome this disadvantage of LEU fuels, the development of high density fuels for research reactors began within the framework of the RERTR program, raising effective uranium densities in the fuel several fold compared to the initial 1.0 – 1.5 g/cc that were achievable until the 1980s; Glaser, (2005a). The availability of these fuels is a prerequisite for meeting most of the technical and economic criteria relevant in a conversion process.

The fact that HEU-fueled reactors have been and are still operated in about 50 countries in the world, has lead to broad geopolitical distribution of the material in fresh and irradiated forms, while fabrication, transportation, and long-term interim storage of this fuel creates additional proliferation risks. Originally, all HEU was exclusively produced for military purposes and those stocks that have been available for civilian applications stem from excess military production capacities.

Huge quantities of HEU are in existence today, while the possibility of renewed production of this material for military purposes, in particular by the gas centrifuge, received considerable attention in the years 2003 and 2004; (Glaser, 2004a).

2.9 CLASSES OF NUCLEAR FUEL

The design of a plate-type fuel compared to typical pin-type fuels is fundamentally different with respect to the retention of fission gases released during irradiation. In the case of pin-type fuels, fission gases are mainly accommodated by gaps between the fuel pellets and the cylindrical fuel cladding. In plate-type fuels, cladding and fuel zone (meat) are bonded, and a different mechanism has to be provided to accommodate the gaseous fission products and to support swelling of the fuel. The main type of designs available nowadays briefly introduced below with aluminium-based dispersion fuels been by far the most common type in use, though monolithic fuel may become an important alternative in the future.

a) CERAMIC FUEL

The most common ceramic fuel is uranium-dioxide (UO_2), which is primarily used in commercial power reactors. In research reactors, ceramic fuel are less common and less favorable due to poor thermal conductivity of UO_2 and difficulty in fabricating plates with this material. One exception to this rule is the so-called caramel fuel, which was developed by the French CEA in the 1970s;

(Schwartz 1978).

Caramel fuel consists of a large number of small uranium-oxide pastilles (caramels), each about 2x2 cm^2 and with a thickness of 1.5 – 4.0 mm. A layer of these pastilles is arranged on a rectangular surface, separated by a spacer grid. This fuel region is covered with two Zircalloy plates to form the fuel plate itself. Such a design according to Glaser (2005a) enables the fission gases released during irradiation of the fuel to be accommodated in the regions between the fuel pastilles. Caramel fuels were produced with low enrichment, usually below 10%, and used in French research reactors. Caramel fuel was developed originally for use in propulsion reactors of French submarines; (Schwartz, 1978).

b) DISPERSION-TYPE FUEL

Dispersion fuel consists of small fuel particles embedded into a non-fissile and ideally inert matrix material, which guarantees adequate irradiation behavior of the fuel. The fundamental idea of dispersion fuel is to isolate the fuel particles such that the fission gases released during irradiation can be accommodated locally within the matrix; (Glaser, 2005a). Consequently, no additional gap or void between the fuel and the cladding is needed to contain fission products or to support swelling of the fuel. The effective thermal conductivity of dispersion fuel can be very high if the matrix material is properly chosen. Glaser (2005b) wrote that two different fuel-matrix permutations are conceivable within the class of dispersion fuel. These are metallic (aluminium-based dispersion fuels) and ceramic (non aluminium-based

dispersion fuels) matrices. In practice, only fuel with metallic matrices has been extensively used in research reactors and other reactors; (Hofman and Snelgrove, 1994). Several dispersants are suitable and have been used in aluminium-based dispersion fuels with the volume fraction of dispersant in the matrix being limited to 30 – 50% so as to generate stable irradiation behavior of the fuel.

Research reactors may be operated at relatively low temperatures and metallic components in the fuel are therefore acceptable; (Glaser, 2005b). Aluminium is a favored matrix material because it is both widely available and characterized by a very high thermal conductivity. Early dispersion type fuel used with HEU since the 1960s were primarily based on UAl_x and U_3O_8 dispersants which achieve effective uranium densities of up to 1.5 g/cc. Efforts to make high-density fuel available, according to Glaser (2005b), that could be used with LEU lead to the development of fuel using U_3Si_2 and uranium-molybdenum alloys as dispersants.

In cermets fuel, the ceramic fuel particles are dispersed in a metallic matrix. The most common combination is UO_2 in Zircalloy or stainless steel. The rationale of using cermets, according to Glaser (2005b), is to overcome shortcomings of the pure ceramic (UO_2) fuel: cermets provide highly reliable fuel elements performance. Cermets have been used with HEU in military propulsion reactors (Hofman and Snelgrove, 1994), where aluminium-based dispersion fuels cannot be used because of the high operating temperatures in these PWR-type reactors.

Dispersion-type fuels contain a uranium compound dispersed in an aluminium matrix. The volume balance in the fuel meat is given in (Glaser, 2005a) by the sum of the fractional volumes occupied by the fuel particles, the matrix material, and the residual void,

$$V_{total} = V_{fuel} + V_{matrix} + V_{void} = 1.0 \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.22)$$

Once the target uranium density in the fuel or the volume fraction of the dispersant in the meat is specified, the porosity of the fuel must be estimated. Glaser, (2005a) defined the porosity as the fractional volume of void in the meat and, in general, increases with increasing volume fraction of the fuel particles in the aluminium matrix. The porosity originates from an imperfect flow of matrix aluminium around the fuel particles during production of the fuel plate and is required to accommodate fission gas released during irradiation of the fuel. Even though porosity is essentially irrelevant in the context of neutronics calculations, (Glaser, 2005a), typical porosity values have been used to adjust the number densities, that is, to reduce the volume of the matrix material, for the various fuel-types correspondingly.

$$V_{void} \approx 0.072V_{fuel} - 0.275V_{fuel}^2 + 1.320V_{fuel}^3 \ldots\ldots\ldots\ldots\ldots\ldots (2.23)$$

For$V_{fuel} \leq 0.5$. This cubic fit has been obtained based on experimental data obtained for uranium slilcide fuels; (Matos and Snelgrove, 1992). Nevertheless, this expression has also been used for other dispersion-type fuels and provides useful

estimates for the porosity to be expected; (Bretcher and Matos, 1996).

c) MONOLITHIC FUELS

Glaser, (2005a) wrote that using uranium metal directly as a fuel material would obviously maximize the density of fissile material in the fuel. Furthermore, metals are characterized by high thermal conductivities, leading to a flat temperature gradient and a low centerline temperature in the fuel. Unfortunately, metallic uranium displays poor irradiation behavior and is subject to phase changes between 300K and its melting point, making it unsuitable for use in power reactors. The most promising candidate material is a uranium-molybdenum alloy with 5-15% molybdenum content and a performance optimum at approximately 10 wt%; (Sesonske and Yevick, 1966), and uranium-molybdenum fuels were used in some prototype fast reactors in the 1960s.

In the years 2000-2002, problems emerged with dispersion fuels using uranium-molybdenum alloy as dispersant. During the analysis of these problems, micro-plate containing U-Mo (10) alloy were irradiated under research reactor conditions; unexpectedly, the fuel behaved extremely well up to a very high burn up of 70-80% $^{235}U_{92}$ showing locally contained fission bubbles; (Hofman and Meyer, 2002). Nowadays, monolithic fuels represent the most promising candidate for the next generation of high-density fuels for research reactors; (Glaser, 2005a).

2.10 CONTROL ROD WORTH AND S-SHAPE CURVE

The control rod worth is the amount of reactivity that corresponds to full withdrawal or insertion of the control rod from or into the reactor. It is the reactivity control ability of the control rod. The control rod worth is determined by experiments; (Jiajuan, 1993). There are many measurement methods, amongst which the dynamic methods are most widely used. Some of these methods include power pulse neutron source method, source deviation method, rod drop method, and period method. Amongst these methods, the rod drop method does not need special devices, the instrument is safer and it is suitable for engineering reactors.

The measurement of subcritical margin with the rod drop method or source deviation method is based on the same principles. The correlation of subcritical margin (or reactivity) ρ with neutron counting in the core is derived from point kinetic equations through Laplace transforms, which is written as;

$$\rho^{-} = \frac{n_o \left[\Lambda + \Upsilon_1 \sum_{i=1}^{16} \frac{\beta_i}{\lambda_i} + \gamma_2 \sum_{j=1}^{16} \frac{\beta_i}{\lambda_i} F_j\right]}{\int_0^{\infty} n(t)\,dt} \quad \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.24)$$

Where ρ^{-} is the reactor sub-criticality, n_o is neutron counting at steady state in the core, Λ is the neutron generation time, Υ_1 is the effective factor of fission delayed neutron fraction, Υ_2 is the effective factor of photo-neutron fraction (it is very small for MNSR), $\lambda_i$and $\beta_i$are the decay constant and fraction of delayed neutrons in

group i, λ_j and β_j are the decay constant and delayed neutrons in group j, $F_j = 1 - e^{-\lambda_j T_o}$ is the accumulation factor for delayed photo-neutron in group j, T_o is the steady operation time of the reactor under certain power level, and $\int_0^\infty n(t)dt$ is the net integral neutron counting from the time of the rod drop to the time when neutrons decay to background level and

$$\int_0^\infty n(t)dt = n_s - n_d T_s \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.25)$$

Where n_s is the total integrated neutrons counting after the rod drop, n_d is the background neutron counting, and T_s is the time taken for the integrity counting.

If the rod drops at a state with sub-criticality ρ_1 and the sub-criticality measured after the rod drop is ρ_2, then the rod worth $\Delta\rho$ will be

$$\Delta\rho = \rho_2 - \rho_1 \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2.26)$$

The control rod worth refers to the amount of reactivity which is absorbed by fully inserted control rod. But for an actual reactor under operation, it is necessary to know the reactivity equivalence corresponding to the unit distance in the core that the control rod traveled; (Jiajuan, 1993). That is the so-called S-shape curve of the control rod worth. The s-shape curve is obtained by experimental measurements, usually on the zero power critical facility. According to Jiajuan (1993), the measurement carried out by inserting the control rod to be measured into the core and reactivity is measured by period method. At the beginning, the reactor is at

critical state, thus $\rho_0 = 0$, the rod is then withdrawn by a length Δh_1 and the reactivity ρ_1 is measured. The process is repeated at different withdrawal lengths until the rod is fully withdrawn from the core and the curve in figure 2.3 is obtained.

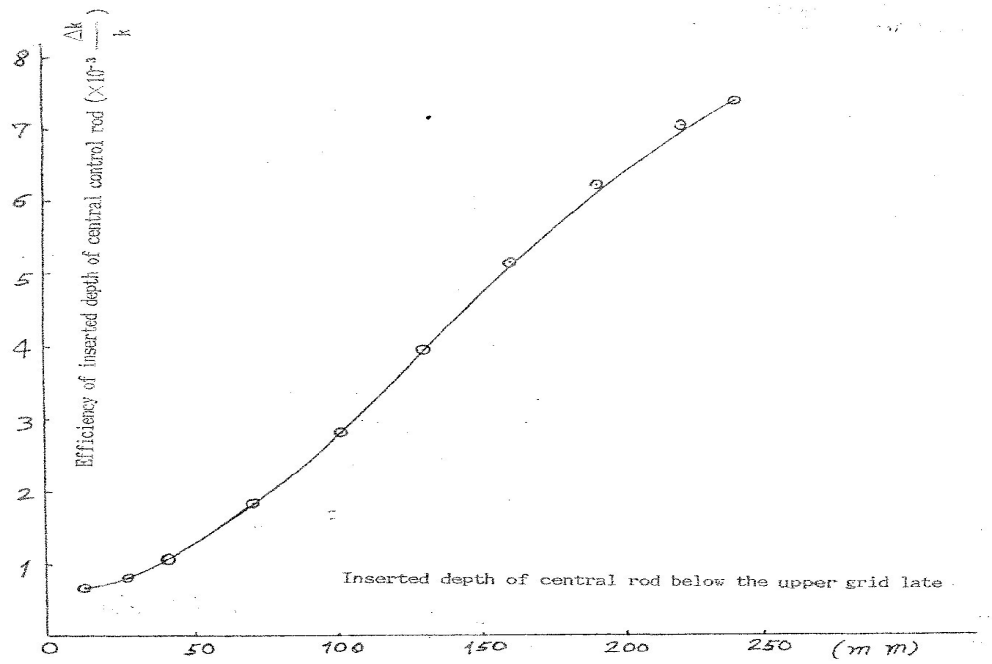


Figure 2.3: S-shape curve of the control rod for Pakistan MNSR at zero-power critical facility; Extract from the training Material for Nigeria MNSR, China Institute of Atomic Energy, (1993).

2.11 THE CONTROL ROD AND REACTIVITY REGULATION

Some materials have very high neutron absorption cross section, which can be regarded as *black body* for neutrons, while some other materials are regarded as

grey body due to their relevant small absorption cross section. The control rod for different purposes is made of different absorption materials. The safety rod, for instance, is mainly for shutting down the reactor to ensure safety, thus should have high coefficient and be made of a black body material like cadmium or boron, while the regulation rods or shim rods are for starting up the reactor or power regulation or compensation of the reactivity losses, thus their coefficients should not be too high and are usually made of grey body materials such as boron-steel. The control rod of MNSR is made of cadmium, a black body material.

If there are N_0 neutrons in the preceding generation, then there are $(N_0 k_{eff})$ neutrons in the present generation. The numerical change in neutron population is $(N_0 k_{eff} - N_0)$. The gain or loss in neutron population, $(N_0 k_{eff} - N_0)$, expressed as a fraction of the present generation $(N_0 k_{eff})$, is given in (DOE-HDBK, 93) as;

$$\rho = \frac{N_0 k_{eff} - N_0}{N_0 k_{eff}} \qquad (2.27)$$

This relationship represents the fractional change in neutron population per generation and is referred to as *reactivity* (ρ). Cancelling out the term No from the numerator and denominator, the reactivity is determined as shown in the equation below.

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \qquad (2.28)$$

From equation (2.28) it may be seen that ρ may be positive, zero, or negative, depending upon the value of k_{eff}. The larger the absolute value of reactivity in the reactor core, the further the reactor is from criticality and it may be convenient to think of reactivity as a measure of a reactor's departure from criticality; (DOE-HDBK, 93).

Reactivity is a dimensionless number. It is simply a ratio of two quantities that are dimensionless. The value of reactivity is often a small decimal value. In order to make this value easier to express, artificial units are defined. By definition, the value for reactivity that results directly from the calculation of equation (2.28) is in units of $\Delta k/k$. Alternative units for reactivity are $\%\Delta k/k$ and pcm (percent millirho); (DOE-HDBK, 93). Another unit of reactivity that is used at some reactors is equivalent to 10^{-4} $\Delta k/k$. This unit of reactivity does not have a unique name. Special units for reactivity that do have unique names are dollars and cents; (DOE-HDBK, 93). The amount of reactivity (Δ) in a reactor core determines what the neutron population, and consequently the reactor power, is doing at any given time. The reactivity can be affected by many factors (for example, fuel depletion, temperature, pressure, or poisons).

To quantify the effect that a variation in parameter (that is, increase in temperature, control rod insertion, increase in neutron poison) will have on the reactivity of the core, *reactivity coefficients* are used. Reactivity coefficients are the amount that the reactivity will change for a given change in the parameter; (DOE-

HDBK, 93). For instance, an increase in moderator temperature will cause a decrease in the reactivity of the core. The amount of reactivity change per degree change in the moderator temperature is the moderator temperature coefficient. Reactivity coefficients are generally symbolized by Δx, where x represents some variable reactor parameter that affects reactivity. The definition of a reactivity coefficient in equation format is given in (DOE-HDBK, 93);

$$\alpha x = \frac{\Delta \rho}{\Delta x} \qquad (2.29)$$

If the parameter *x* increases and positive reactivity is added, then Δx is positive. If the parameter *x* increases and negative reactivity is added, then Δx is negative. *Reactivity defects* (Δρ) are the total reactivity change caused by a variation in a parameter. Reactivity defects can be determined by multiplying the change in the parameter by the average value of the reactivity coefficient for that parameter; (DOE-HDBK, 93).

The exact effect of control rods on reactivity can be determined experimentally. For example, a control rod can be withdrawn in small increments, such as 0.5 inch, and the change in reactivity can be determined following each increment of withdrawal. By plotting the resulting reactivity versus the rod position, a graph similar to Figure 2.3 is obtained. The graph depicts integral control rod worth over the full range of withdrawal. The *integral control rod worth* is the total reactivity worth of the rod at that particular degree of withdrawal and is usually

defined to be the greatest when the rod is fully withdrawn; (DOE-HDBK, 93). The slope of the curve ($\Delta\rho/\Delta x$), and therefore the amount of reactivity inserted per unit of withdrawal, is greatest when the control rod is midway out of the core. This occurs because the area of greatest neutron flux is near the center of the core; therefore, the amount of change in neutron absorption is greatest in this area; (DOE-HDBK, 93). If the slope of the curve for integral rod worth in Figure 2.3 is taken, the result is a value for rate of change of control rod worth as a function of control rod position. The *Differential control rod worth* is the reactivity change per unit movement of a rod and is normally expressed as Δ/inch, Δk/k per inch, or pcm/inch. The integral rod worth at a given withdrawal is merely the summation of the entire differential rod worth up to that point of withdrawal. It is also the area under the differential rod worth curve at any given withdrawal position; (DOE-HDBK, 93).

The reactivity of the reactor will gradually reduce with operation time. Hence, if a reactor were designed with effective multiplication factor k_{eff} equal to unity, the reactor would not be able to operate normally. The designed k_{eff} being greater than one implies that the design reactivity shall be greater than zero. The designed reactivity above zero is called excess reactivity ρ_{ex}, which is stored in some way at the beginning of operation and then gradually released when needed. Utilization of regulators can also store some ρ_{ex}. The ρ_{ex} is achieved by inserting additional fuel elements into the core, (Jiajuan, 1993). Initially sufficient excess reactivity must be designed into the reactor core to allow for the build-up of equilibrium levels of reactor poisons and for all the fall in reactivity due to U-235

burn-up during a reasonable operating time of the fuel elements since this will not be fully compensated for by Pu-239 build-up in the long term. Hunt (1987) agreed that typically a natural uranium reactor core would be designed to have a reactivity of about +0.05 with the initial charge of unused fuel and for slightly enriched reactors approaching +0.10 because of the smaller Pu-239 build-up and higher level of fission products during operation. This means that the reactor assembly must be appreciably above the minimum critical size.

The reactivity of the assembly is controlled by using rods of neutron-absorbing material, usually cadmium or boron, which have capture cross-section of 20,000b for ^{113}Cd $(n,\gamma)^{114}Cd$ reaction and 4000b for the $^{10}B(n,\gamma)^{11}B$ reaction; (Glasstone, 1956). Hunt (1987) added that the control rods are automatically raised or lowered in the reactor assembly, thus making the appropriate small adjustments in Δk to maintain the reactor power at a predetermined level or to increase or reduce it as required. The control of reactors is possible by the fact that not all the neutrons are produced at fission. Jonah (2007) explained that the energy for β-decay for very far from stability goes up to 10MeV and above; in such cases, excited energy will be so high that there is a probability for a neutron to be emitted rather than β and this neutron is called β-delayed neutron. Jonah (2007) concluded that the β-delayed neutron gives enough time to control reaction rate in nuclear fission and the process is critical in the working of a nuclear reactor. Hunt (1987) agreed that there are some 50 delayed neutron emitters, but reactor control can be explained in terms of six such emitters with half-lives in the range 54-0.18s. Assuming that Δk_{prompt} is

0.01; the rate of change in neutron population, according to Hunt (1987), is therefore given by equation (2.30).

$$\frac{dn}{dt} = n\,\frac{\Delta k_{prompt}}{l} \qquad (2.30)$$

The quantity l is the time between one generation of fission processes and the next. This is essentially the time required to moderate the fission neutrons plus the time for which thermal neutrons exist before being captured by U-235 or lost in one of the competitive processes. Hunt (1987) gave the prompt neutron lifetime in graphite to be about 10^{-3}s and 10^{-4}s in water.

Reactor control rods are quite heavy and cannot be move at a very high speed, so that a rector above criticality due to prompt neutrons alone would be difficult if not impossible to control; (Hunt, 1987). Including 0.65% of delayed neutrons in (2.30), we get equation (2.31).

$$\frac{dn}{dt} = n\,\frac{\Delta k_{prompt}}{l} + \sum \lambda_d C_d \qquad (2.31)$$

λ_d is some mean value of decay constant for the delayed neutrons and C_d is their concentration. If Δk_{prompt} is made slightly negative, the rate of neutron increase, $\frac{dn}{dt}$, will depend mainly on λ_d, the reciprocal of the average mean life of the delayed neutron emitters which is of the order of 13s; Hunt (1987); the time constant of the reactor therefore becomes tens of seconds rather than milliseconds and control become comparatively easy. Hunt (1987) gave the necessary condition for this that

Δk_{prompt} must be negative, that is, the assembly must be subcritical if prompt neutrons alone are considered, and this implies that the total reactivity must be less than 0.0065.

Prompt criticality is therefore to be avoided in an operational reactor, and the built-in excess reactivity in a fresh un-poisoned core must be neutralized by the insertion of control rods which are gradually withdrawn as fission products poisons build up, and as the reactivity changes due to depletion of the U-235 content of the initial fuel; more rapid changes in reactivity, to start the reactor after shut-down or to change the reactor power or to compensate for short-term power variations, are achieved by more rapid movement of the control rods, but the relatively long time constant of a reactor in its normal non-prompt critical condition makes this a relatively simple process; the time constant of a reactor increases as its prompt reactivity becomes more negative since, under these conditions, criticality becomes increasingly dependent on the longer-lived neutron emitters; (Hunt, 1987). Reactors may be shut down rapidly in emergencies by the insertion of additional neutron-absorbing material in the form of rods, spheres or solutions.

As the reactor operates, ρ_{ex} reduces. The full power operation time from the time a new core is used to the time when k_{eff} reduces to unity is called core lifetime. Jiajuan, (1993) highlighted two factors governing the core lifetime of the MNSR as the quantity of ρ_{ex} and the content of fuel element damage. The excess reactivity ρ_{ex} of the MNSR is usually chosen to be (3.5~4.0) mk, which ensures a core lifetime of

10 years. The core of MNSR is surrounded with metallic beryllium as reflector. At the beginning of the core life, the upper beryllium reflector on the top of the core is intentionally put away to let the part of the core neutrons escape from the top, which makes the ρ_{ex} small enough to meet safety requirements. As the operation time accumulates, ρ_{ex} of the original core decreases to 2.5~3.0 mk depending on the initial cold state ρ_{ex}, a certain thickness of upper beryllium plate is added to its predetermined position to reduce the neutron leakage, thus increasing the ρ_{ex} enabling the reactor to maintain another period of steady state operation. Another upper beryllium plate is once again added when the ρ_{ex} reduces to a certain value. In this way, the 10 years lifetime is guaranteed. But at a certain beryllium thickness, the reactivity gain will be saturated.

2.12 SIMPLIFIED SCHEME OF WIMS AND CITATION CODES

The computer codes WIMS-D4; Askew et al (1966) and CITATION; Fawler (1987) are a pair of versatile reactor design tools but are not ‘user friendly’; (Balogun, 2003a). The input requirements for some of the applications of the thermal reactor neutronics code WIMS-D4 can be demanding to setup. The code is both powerful and flexible enabling a wide range of thermal reactor calculations to be to be performed. However, for standard calculations there would be some benefit to be gained from a simple user interface, where the major choices leave the user

with the much simpler task of specifying basic geometrical, material and operational data. The provision of sensible defaults and ability to select from pre-checked libraries of such data would further aid WIMS user; (Balogun, 2003a).

CITATION is a package designed to provide the user with framework for easily setting up WIMS models and performing sequences of calculations. The aim is for CITATION to provide a menu of basic application options for the user to select from. For each distinct type of fuel element, WIMS is asked to perform say 30 burn-up cycles through about a dozen code modules; (Balogun, 2003a). In addition, at most of these cycles, cross section data are created for up to 20 variants of the standard operating conditions such as temperature changes, coolant density changes, the decay of Xe-135 due to a shut-down, or the insertion of control rods, making in all about 600 cycles of input to WIMS.

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

CHAPTER THREE

THEORITICAL CONSIDERATION AND CALCULATIONS

3.10 INTRODUCTION

WIMS and CITATION are deterministic neutron transport codes that calculate neutron flux distributions and values of k-infinity or k-effective for an extensive range of reactor cell types. To carryout the calculation, it has to solve a mathematical form of the transport equation and some approximations are required in order to solve the equation efficiently. The most important of the approximations is the use of energy groups. Neutron cross sections are complicated functions of energy and it is necessary to average them in some way over discrete energy ranges. This process is performed prior to the start of WIMS and CITATION calculations and results in a nuclear data library which is read from and but never written to by WIMS and CITATION. The averaging is performed so that the reaction rate in a given energy range or group is preserved, because the solution is a function of reaction rates. In WIMS-D4 the total energy space is divided into four regions; fast, resonance, epithermal, and thermal regions.

For cross-sections that exhibit resonance type behavior, the cross sections can change by many orders of magnitude. In this situation, the flux is strongly dependent on the particular problem and the assumption of a fixed functional form

is not adequate. In order to produce a library that can be used on any thermal reactor, cross sections are generated for a number of different forms for the flux. This is carried out by solving the neutron slow down equations for a simple homogeneous mixture of a single resonance isotope and a fictitious nuclide that has the scattering properties of hydrogen. For WIMS-D4, the slowing down equations is solved by the SDR code. By varying the ratio of the resonance absorbing to the scattering isotope the amount of flux depression at the resonance energies varies, giving different values for the group averaged cross sections. These tabulated values are used in the treatment of resonance shielding. The essence of the resonance treatments in WIMSD is to relate more complex problems, in particular heterogeneous problems containing mixtures of resonance and moderating nuclides, to the homogeneous data stored in the WIMS library.

3.11 HOMOGENEOUS RESONANCE THEORY

Approximations to the nuclide reaction rates calculated in SDR can be obtained by equating the neutron collision rate to the neutron slowing down density and using the expression for the probability of scattering to higher lethargy; Glasstone and Edlund (1952)

$$\Sigma_t(U)\phi(U) = q(U) \qquad (3.01)$$

$\Sigma_t(U) = the\ macroscopic\ total\ cross - sections, \phi(U) = flux,$

$q(U) = theneutron\ slowing\ down\ density$

In the narrow resonance approximation, the average logarithmic energy per collision is assumed much larger than the width of the resonances. In this case the contribution of the slowing down density from integration over the resonances can be neglected. The scattering cross-sections outside the resonances are approximately constant and equal to the potential scatter cross-sections. The slowing down density is given by;

$$q(U) = \frac{\Sigma_{pot}}{\xi} \qquad (3.02)$$

Σ_{pot} = Macroscopic potential scatter cross section, ξ =system average logarithmic energy decrement per collision.

In the wide resonance approximation, any logarithmic energy loss in a collision is assumed negligible relative to the width of the resonance. All scatter is effectively self scatter neither adding nor removing neutrons. In this case, the slowing down density is simply given by the scattering rate;

$$q(U) = \Sigma_s(U)\phi(U) \qquad (3.03)$$

Σ_s = the macroscopic scatter cross sections.

For light nuclei the narrow resonance approximation holds well. For heavy nuclei whether the narrow resonance approximation or the wide resonance approximation is appropriate depends on the individual characteristics of the resonance. Generally, the narrow resonance approximation holds at high resonance energies where resonances tend to be narrow and close together and there is a tendency towards the wide resonance approximation at lower resonance energy where resonances are wide and well spaced or separated.

In practice, a mixture of the narrow and wide resonance approximations is required. This is achieved by using the intermediate resonance approximation proposed by Goldstein and Cohen (1962). The intermediate resonance parameter λ is introduced such that if $\lambda = 1$, then the narrow resonance approximation applies. This gives the general slowing down density as;

$$q(U) = \frac{\lambda \Sigma_{pot}}{\xi} + (1-\lambda)\Sigma_s(U)\phi(U) \qquad (3.04)$$

Equating this to the collision rate per unit lethargy in a homogeneous geometry with only a single spatial region gives

$$\Sigma_t(U)\phi(U) = \frac{\lambda \Sigma_{pot}}{\xi} + (1-\lambda)\Sigma_s(U)\phi(U) \qquad (3.05)$$

As the total cross-section is just the sum of the absorption and scatter cross-sections, this can be rearranged to give

$$\phi(U) = \frac{\lambda\Sigma_{pot}}{\bar{\xi} + (\lambda\Sigma_s(U) + \Sigma_a(U))} \qquad (3.06)$$

Introducing a resonance scatter cross-section, that is zero in the absence of resonance scatter defined by

$$\Sigma_{rs}(U) = \lambda(\Sigma_s(U) - \Sigma_{pot}) \qquad (3.07)$$

and also define

$$\Sigma_p = \lambda\Sigma_{pot} \qquad (3.08)$$

gives

$$\phi(U) = \frac{\Sigma_p}{\bar{\xi}(\Sigma_p(U) + \Sigma_{rs}(U) + \Sigma_a(U))} \qquad (3.09)$$

At lethargies higher than the resonance lethargy, the flux will be reduced due to captures in the resonance. This has an effect on the out scatter cross-sections for higher mass nuclides. A correction to out scatter cross-section is included in WIMS, based on either a fine group calculation or an evaluation of the resonance escape probability. By definition,

$$\Sigma_a(U) = \sum_i N_i \sigma_a^i(U) \qquad (3.10)$$

$$\lambda = \frac{\sum_i N_i \lambda_i \sigma_s^i(U)}{\sum_i N_i \sigma_s^i(U)} \qquad (3.11)$$

$$\Sigma_p = \sum_i N_i \lambda_i \sigma_{pot}^i \qquad (3.12)$$

$$\bar{\xi} = \frac{\sum_i N_i \xi_i \sigma_s^i(U)}{\sum_i N_i \sigma_s^i(U)} \qquad (3.13)$$

$$\sigma_{rs}^i(U) = \lambda_i \left(\sigma_s^i(U) - \sigma_{pot}^i\right) \qquad (3.14)$$

$$\Sigma_{rs}(U) = \sum_i N_i \lambda_i \left(\sigma_s^i(U) - \sigma_{pot}^i(U)\right) \qquad (3.15)$$

$$\sigma_p^i = \frac{\Sigma_i}{N_i} \qquad (3.16)$$

For a single resonance nuclide the broad group average flux can be derived as follows;

$$\bar{\phi} = \int \phi(U) dU = \int \frac{\Sigma_p}{\bar{\xi}\left(\Sigma_p^i + \Sigma_{rs}^i(U) + \Sigma_a^i(U)\right)} dU \qquad (3.17)$$

Dividing this by the number density of the resonance nuclide and substituting the above expressions for macroscopic cross-sections gives;

$$\bar{\phi} = \frac{1}{\bar{\xi}} \int \frac{\sigma_p^i(U)}{\left(\sigma_p^i + \sigma_{rs}^i(U) + \sigma_a^i(U)\right)} dU \qquad (3.18)$$

Rearranging by adding and subtracting $\sigma_{rs}^i + \sigma_a^i$ to the numerator of the integrand, the broad group flux can be related to the resonance integrals stored on the library.

$$\bar{\phi} = \frac{1}{\xi}\left[\int du - \int \frac{\sigma_{rs}^{i}(U) + \sigma_{a}^{i}(U)}{\left(\sigma_{p}^{i} + \sigma_{rs}^{i}(U) + \sigma_{a}^{i}(U)\right)} dU\right] \qquad (3.19)$$

The library tabulation of resonance integrals for nuclide i, per unit lethargy, for absorption and scatter as a function of both temperature and the amount of scattering per atom of nuclide i are defined to be;

$$I_{a}^{i}(\sigma_{p}) = \frac{1}{\delta U}\int \frac{\sigma_{a}(U)\sigma_{p}}{\sigma_{p} + \sigma_{rs}(U) + \sigma_{a}(U)} dU \qquad (3.20)$$

and

$$I_{rs}^{i}(\sigma_{p}) = \frac{1}{\delta U}\int \frac{\sigma_{is}(U)\sigma_{p}}{\sigma_{p} + \sigma_{rs}(U) + \sigma_{a}(U)} dU \qquad (3.21)$$

Noting that σ_p is assumed to be lethargy independent then;

$$\bar{\phi} = \frac{\delta U}{\xi}\left[1 - \left[\frac{\sigma_{rs}^{i}(\sigma_{p}^{i})}{\sigma_{p}^{i}} + \frac{I_{a}^{i}(\sigma_{p}^{i})}{\sigma_{p}^{i}}\right]\right] \qquad (3.22)$$

The microscopic broad group average cross-sections for nuclide i resonance reaction X are given by;

$$\bar{\sigma}_{x}^{i} = \frac{\int \sigma_{x}^{i}(U)\phi(U)dU}{\int \phi(U)dU} \qquad (3.23)$$

Assuming the form of the flux from above then;

$$\bar{\phi}.\bar{\sigma}_x^i = \int \frac{\sigma_x^i(U)\Sigma_p}{\bar{\xi}\left(\Sigma_p+\Sigma_{rs}(U)+\Sigma_a(U)\right)} dU \qquad (3.24)$$

Following a procedure similar to that for the broad group average flux gives;

$$\sigma_x^i = \frac{I_x^i\left(\sigma_p^i\right)}{1-\left[\frac{I_{rs}^i\left(\sigma_p^i\right)}{\sigma_p^i}+\frac{I_a^i\left(\sigma_p^i\right)}{\sigma_p^i}\right]} \qquad (3.25)$$

In practice, there is an interaction effect between the different resonance nuclides in the problem. The flux depression caused by one nuclide can affect the resonance shielding of another nuclide. In WIMS-D4 two models are used to treat this problem. In the higher energy part of the resonance region there are many narrow resonances. Here the resonances are assumed to interact in a random manner, and the expressions for the flux derived above are modified to take into account the contribution to the flux depression from other nuclides. In the lower energy portion of the resonance region, a detailed slowing down calculation is performed within WIMS-D4. This slowing down calculation generates correction factors defined as;

$$\frac{interacted\ cross\_sections}{(uninteracted\ cross\_sections)}$$

These correction factors are used to model the interaction between the various nuclides in the calculation.

Out-scatter or removal from a broad group is affected by the presence of resonances within the group. Heavier nuclides may only be able to scatter neutrons out of the broad group from the higher lethargies of the group where the flux may be depressed or enhanced by resonances. In the higher energy region, a correction for this effect is calculated on a nuclide by nuclide basis. This requires the calculation of the resonance escape probability p, and is known as the $f(P)$ factor.

In the lower energy region the fine group calculation is again used to calculate the effect. The correction factors are defined as the ratio of the best estimate result taken from the fine group calculation, to a calculation which represents the SDR treatment used to generate the library data. Therefore, for resonance nuclides the $f(P)$ is defined as;

$$\frac{(broad\ group\ average\ removal\ cross\_section\ in\ the\ presence\ of\ all\ nuclides)}{(broad\ group\ average\ removal\ cross\ section\ in\ the\ absence\ of\ all\ other\ resonant\ nuclides)}$$

To derive the $f(P)$ correction for the non-resonant nuclides an additional calculation is carried out in the absence of all absorption which gives $f(P)$ correction as;

$$\frac{(broad\ group\ average\ removal\ cross\_section\ in\ the\ presence\ of\ all\ nuclides)}{(broad\ group\ average\ removal\ cross\_section\ in\ the\ absence\ of\ all\ absorptions)}$$

3.12 NUCLEAR DATA LIBRARY

The resonance integral and other quantities needed to evaluate the effective broad group average cross-sections using the formulae developed above are read by WIMS from the United Kingdom Atomic Energy Agency (UKAEA) nuclear data library generated using the SDR code. In this context, SDR solves the neutron slowing down equation for a simple homogeneous mixture of a single resonance absorbing isotope and a fictitious nuclide which has the scattering properties of hydrogen. Apart from data based on the SDR code, the cross-sections in the nuclear data file have been group averaged through use of the GALAXY code; Bell et al (1964), assuming a $\frac{1}{E}$ spectrum in the resonance region, and a spectrum typical of the under moderated water system in the fast region. The GALAXY solution is in essence a continuous energy solution which can be used to average cross sections or resonance integrals over broad group energy structure specified by the user. Each execution of GALAXY provides energy group data for a specific isotope, temperature and ratio of the resonance absorbing isotope to the fictitious hydrogen like scattering nuclide (sigma-p value). The nuclear data library is therefore assembled from many executions of the GALAXY code to give a full temperature and sigma-p parameterization. In the case of non-resonant nuclides, with no temperature dependant thermal cross sections, only a single execution of the GALAXY is required in which the flux is assumed to have the shape of the fission spectrum at high energy, to be of the form $1/_{energy}$ in the resonance range and to have Maxwellian thermal energy dependence. The ratio of the resonance isotope to the fictitious hydrogen like scattering nuclide, the sigma-p value, is given in the

input to GALAXY by specifying the energy independence potential scattering cross-section due to the fictitious nuclide, σ_o. GALAXY takes the number density of the resonance nuclide to be unity and so the potential scattering cross-section in the calculation, per atom of resonance isotope, is;

$$\sigma_p^i = \sigma_o + \lambda_i \sigma_{pot}^i \tag{3.26}$$

The definition of the scattering, per atom of the resonant isotope, in a homogeneous mixture of the resonance isotope and hydrogen, is given by;

$$\sigma_p^i = \frac{N_H}{N_i} = \sigma_{pot}^H + \lambda_i \sigma_{pot}^i \tag{3.27}$$

Comparing (3.26) and (3.27) shows that the input value of σ_o specifies the ratio of the hydrogen like nuclide to the resonant isotope. WIMS-D4 takes resonance integrals from GALAXY for four reactions, absorption, fission, resonance scatter and removal. These integrals are formed from the broad group average cross sections derived in the SDR execution;

$$I_a^i(\sigma_p) = \frac{1}{\delta U}\int \frac{\sigma_{rs}^i(U)\sigma_p}{\sigma_p + \sigma_{rs}^i(U) + \sigma_a^i(U)}\, dU = \frac{\bar{\sigma}_a^i \sigma_p}{\sigma_p + \bar{\sigma}_{rs}^i + \bar{\sigma}_a^i} \tag{3.28}$$

$$I_f^i(\sigma_p) = \frac{1}{\delta U}\int \frac{\sigma_f^i(U)\sigma_p}{\sigma_p + \sigma_{rs}^i(U) + \sigma_a^i(U)}\, dU = \frac{\bar{\sigma}_f^i \sigma_p}{\sigma_p + \bar{\sigma}_{rs}^i + \bar{\sigma}_a^i} \tag{3.29}$$

$$I_{rs}^i(\sigma_p) = \frac{1}{\delta U}\int \frac{\sigma_{rs}^i(U)\sigma_p}{\sigma_p + \sigma_{rs}^i(U) + \sigma_a^i(U)}\, dU = \frac{\bar{\sigma}_{rs}^i \sigma_p}{\sigma_p + \bar{\sigma}_{rs}^i + \bar{\sigma}_a^i} \tag{3.30}$$

$$I_r^i(\sigma_p) = \frac{1}{\delta U}\int \frac{\sigma_r^i(U)\sigma_p}{\sigma_p+\sigma_{rs}^i(U)+\sigma_a^i(U)}\,dU = \frac{\bar{\sigma}_r^i\sigma_p}{\sigma_p+\bar{\sigma}_{rs}^i+\bar{\sigma}_a^i} \tag{3.31}$$

It is recalled that the definition of the resonance scattering cross section is;

$$\sigma_{rs}^i(U) = \lambda_i\left(\sigma_s^i(U) - \sigma_{pot}^i\right) \tag{3.32}$$

The bar over the cross section denotes the group average value. If a simple mixture of a resonance isotope and hydrogen is considered then it can be verified that the use of these definitions, when substituted into the equation for the broad group average cross section given above, exactly recovers the SDR broad group average cross section. For instance, to recover the group averaged absorption cross section;

$$\frac{1}{\sigma_p^i}\left(I_a^i + I_{rs}^i\right) = \frac{\bar{\sigma}_a^i + \bar{\sigma}_{rs}^i}{\bar{\sigma}_p^i + \bar{\sigma}_{rs}^i + \bar{\sigma}_a^i} \tag{3.33}$$

$$1 - \frac{1}{\sigma_p}\left(I_a^i + I_{rs}^i\right) = \frac{\sigma_p}{\bar{\sigma}_p^i + \bar{\sigma}_{rs}^i + \bar{\sigma}_a^i} = \frac{I_a^i}{\bar{\sigma}_a^i} \tag{3.34}$$

Which gives,

$$\bar{\sigma}_a^i = \frac{I_a^i}{1 - \frac{1}{\sigma_p}\left(I_a^i + I_{rs}^i\right)} \tag{3.35}$$

as required.

3.13 EQUIVALENCE THEORY

The purpose of the equivalence treatment is to form a relationship between heterogeneous and homogeneous geometry problems. The heterogeneous problem can then be solved by the solution of one or more homogeneous problems. The effect of the heterogeneous geometry is to provide additional paths for neutrons, outside the fuel, to avoid collisions in the fuel resonances and thereby act as an effective increase in the neutron scattering cross section relative to a homogeneous geometry, (Newton, 2004). The heterogeneous system is approximated by a two region system consisting of a region containing resonance nuclides, denoted by f, and a region containing moderator, denoted by M. the neutron balance for the system can be written as;

$$\Sigma_t^f(U)V_f\phi_f(U) = \psi_f(U)V_f P_{ff} + \psi_m(U)V_m P_{mf} \qquad (3.36)$$

Where

$\phi_f(U)$ = fuel flux, $\Sigma_t(U)$ = fuel total cross section, V_f = fuel volume, $\psi_f(U)$ = slowing down density in the fuel, P_{ff} = fuel to fuel first flight collision probability, V_m = moderator volume, $\psi_m(U)$ = slowing down density in moderator, P_{mf} = moderator to fuel first flight collision probability.

In the fuel region the intermediate resonance approximation is used for the slowing down density. This approximation is a mixture of the narrow resonance approximation in which neutrons are assumed to suffer an energy loss per scattering collision much greater than the width of the resonance and the wide resonance approximation in which neutrons are assumed to suffer a negligible energy loss per scattering collision. In the narrow resonance approximation the approximation the contribution to the slowing down density from integration over the resonance is neglected. Outside the resonances the scatter cross-section is assumed constant and can be replaced by the potential scattering cross-section. The flux solution for this problem is a constant and assuming a normalization of the flux to unity, the slowing down density is simply given by;

$$\psi(U) = \Sigma_{pot} \tag{3.37}$$

In the wide resonance approximation there is no energy loss in a scattering event, all scatter is assumed self scatter, the slowing down density is constant and;

$$\psi(U) = \Sigma_s(U)\phi(U) \tag{3.38}$$

The intermediate resonance approximation combines these two approximations using the Goldstein and Cohen intermediate resonance parameter λ to give the slowing down density in the fuel as;

$$\psi_f(U) = \lambda\Sigma^f_{pot} + (1-\lambda)\Sigma^f_s(U)\phi(U) \tag{3.39}$$

In the moderator there are no resonances and the narrow resonance approximation is used. Also the absorption in the moderator is negligible and the potential scatter cross section can be replaced by the total cross section;

$$\psi_m(U) = \Sigma_t^m(U) \tag{3.40}$$

This gives

$$\left(\Sigma_t^f(U) - P_{ff}(1-\lambda)\Sigma_s^f(U)\right)V_f\phi_f(U) = \lambda\Sigma_{pot}^f V_f P_{ff} + V_m P_{mf}\Sigma_t^m(U) \tag{3.41}$$

The region-wise flat flux approximation is used to relate the 'moderator to fuel' first flight collision probability to the 'fuel to fuel' first flight collision probability. From reciprocity;

$$V_m P_{mf}\Sigma_t^m(U) = V_f\Sigma_t^f(U)P_{fm} \tag{3.42}$$

Also, given that all neutrons collide in the system;

$$P_{fm} + P_{ff} = 1 \tag{3.43}$$

Therefore,

$$\Sigma_t^f(U) - P_{ff}(1-\lambda)\Sigma_s^f(U)\phi_f(U) = \lambda\Sigma_{pot}^f P_{ff} + \Sigma_t^f(U)(1-P_{ff}) \tag{3.44}$$

The rational approximation, first introduced by Wigner, is now introduced for the form of the fuel to fuel first flight collision probability;

$$P_{ff} = \frac{\Sigma_t^f(U)}{\Sigma_t^f(U)+\Sigma_e} \tag{3.45}$$

Where

$\Sigma_e = \frac{S_f}{4V_f} = (mean\_chord)^{-1}, \phi_f(U) = \frac{\left(\Sigma_p^f+\Sigma_e\right)}{\left(\Sigma_p^f+\Sigma_e\right)+\Sigma_{rs}^f(U)+\Sigma_a^f(U)}$ in which the following relations have been used;

$$\Sigma_t^f(U) = \Sigma_a^f(U) + \Sigma_s^f(U),\ \lambda\Sigma_{pot}^f = \Sigma_p^f,\ \lambda\Sigma_s^f(U) = \Sigma_{rs}^f(U) + \Sigma_p^f.$$

This form for the flux is the same as that found for homogeneous geometry but with potential scatter cross section enhanced by the escape cross section establishing the equivalence relation.

In the case where the total fuel to fuel flight collision probability can be expressed in single rational form, the heterogeneous geometry can be represented by an equivalent heterogeneous geometry in which the potential scattering cross section is augmented by the escape cross section.

The single rational approximation to the fuel to fuel first flight collision probability can be improved by the introduction of the bell factor a, such that;

$$P_{ff} = \frac{\Sigma_t^f(U)}{\Sigma_t^f(U) + a\Sigma_e} \tag{3.46}$$

It is also sometimes useful to employ a double rational approximation to the fuel to fuel first flight collision probability of the form;

$$P_{ff} = \frac{\beta\Sigma_t^f(U)}{\Sigma_t^f(U)+\alpha_1\Sigma_e} + \frac{(1-\beta)\Sigma_t^f(U)}{\Sigma_t^f(U)+\alpha_2\Sigma_e} \tag{3.47}$$

In this case, a strict equivalence relation can not be derived. However, this approximation is not too gross if this relation is only used to establish the multi-term equivalence between the homogeneous systems. In this case the flux is assumed to be of the form;

$$\phi_f = \sum_j \frac{\beta_j(\Sigma_p^f+\alpha_j\Sigma_e)}{(\Sigma_p^f+\alpha_j\Sigma_e)+\Sigma_{rs}^f(U)+\Sigma_a^f(U)} \tag{3.48}$$

The full equivalence relation can be stated as;

In the case where total fuel to fuel first flight collision probability can be expressed as the sum of rational expressions, the heterogonous geometry can be represented by the sum of a set of weighted homogenous geometries, weights β_j in which the potential scattering cross section is augmented respectively by $\alpha_j\Sigma_e$.

Mathematically, Askew et al, (1966) gave the heterogeneous resonance integral as;

$$I_x^i(het) = \sum_j \beta_j I_x^i\left(\frac{\Sigma_p+\alpha_j\Sigma_e}{N_i\psi_i}\right) \tag{3.49}$$

3.14 INFINITE CELL ARRAYS

The objective here is to evaluate the heterogeneous value of potential scattering cross section (sigma-p) to use in the evaluation of the homogeneous equivalent resonance integrals for an array of pin cells. It follows from above derivations that $\Sigma_p^{het} = \Sigma_p + \beta\Sigma_e$ where $\Sigma_p = \sum_i N_i \sigma_p^i$ and $\Sigma_e = \frac{S_f}{4V_f}$ where V_f = fuel volume, S_f = fuel surface area, N_i = nuclide volume density for nuclide *i* in the fuel region and σ_i = microscopic potential scatter of nuclide *i*. The unknown value is alpha, which originates in the expression for the system fuel to fuel first flight collision probability which in terms of broad group average values is given by;

$$P = \frac{\Sigma_t^f}{\Sigma_t^f + \alpha\Sigma_e} \qquad (3.50)$$

To derive alpha for an infinite array of cells an average unit cell is considered which consists of a fuel region f and a series of non-fuelled region j and has an outer surface S_b; (Newton, 2004). The following first flight probabilities are defined for the average cell;

P_{ff} = the probability of neutrons born uniformly and isotropically in the fuel to suffer their first collision in he fuel, P_{fb} = the probability of those neutrons to reach the cell boundary S_b un-collided, P_{bf} = the probability of neutrons entering the cell through S_b with a cosine distribution with respect to the normal on S_b, to suffer their first collision in the fuel, P_{bb} = the probability of those neutrons to traverse the cell

un-collided, t_{fb} = the probability of neutrons leaving the surface of the fuel S_f with a cosine distribution, to reach S_b un-collided, t_{bf} = the probability of neutrons entering with a cosine distribution through S_b to reach S_f un-collided.

From these average cell probabilities the system fuel self-collision probability, of an infinite array of pins, can be defined by following neutrons from cell until they collide;

$$P = P_{ff} + P_{fb}P_{bf} + P_{fb}P_{bb}P_{bf} + P_{fb}P_{bb}P_{bb}P_{bf} + \cdots \quad (3.51)$$

giving

$$P = P_{ff} + \frac{P_{fb}P_{bf}}{(1 - P_{bb})} \quad (3.52)$$

From reciprocity,

$$P_{bf} = \frac{4V_f\Sigma_t^f}{S_b}P_{fb} \quad (3.53)$$

Regardless of the number of regions between S_f and S_b;

$$P_{bf} = t_{fb}(1 - P_{ff}) \quad (3.54)$$

Therefore,

$$P = P_{ff} + \frac{\frac{S_f \Sigma_t^f}{S_b \Sigma_e} t_{fb}^{\,2} (1 - P_{ff})^2}{(1 - P_{bb})} \qquad (3.55)$$

To proceed further, an expression is needed for the first flight cell blackness$(1 - P_{bb})$. This can be calculated from its three contributions:

1. Collisions of neutrons that have never reached the fuel = $\sum_{j \neq f} P_{bj}$
2. Collisions inside the fuel = $t_{bf} \gamma_f$ where γ is the fuel first flight blackness
3. Collisions of neutrons that have transverse the fuel = $t_{bf}(1 - \gamma_f)(1 - t_{fb})$

Using the reciprocity relation $S_f t_{fb} = S_b t_{bf}$ and that

$$\gamma_f = \frac{4V_f \Sigma_t^f}{S_f}(1 - P_{ff}) = \frac{\Sigma_t^f}{\Sigma_e}(1 - P_{ff}) \qquad (3.56)$$

Giving

$$(1 - P_{ff}) = \frac{\Sigma_t^f (1 - P_{ff})^2}{\Sigma_t^f (1 - P_{ff}) + \frac{S_b \Sigma_e}{S_f t_{fb}^{\,2}} \left[\sum_{j \neq f} P_{bj} + t_{bf}(1 - t_{fb}) \right]} \qquad (3.57)$$

It should be noted that the right hand term in the denominator of the quotient is independent of the fuel cross section and can therefore be evaluated numerically. Setting $= \frac{S_b \Sigma_e}{S_f t_{fb}^{\,2}} \left[\sum_{j \neq f} P_{bj} + t_{bf}(1 - t_{fb}) \right]$, the expression for the system fuel self-collision probability can be written as;

$$P = P_{ff} + \frac{\Sigma_t^f (1 - P_{ff})^2}{\Sigma_t^f (1 - P_{ff}) + X\Sigma_e} \tag{3.58}$$

Introducing the Dancoff factor γ as the reduction in the fuel escape probability for an array of pin-cells relative to the fuel escape probabilities for an isolated pin-cell: the evaluation being carried out in the limit of black fuel; Pershagen and Carlvik (1956),

$$\gamma_f = \lim_{\Sigma_t^f \to \infty} \left\{ \frac{\Sigma_t^f}{\Sigma_e} (1 - P_{ff}) \right\} = 1 \tag{3.59}$$

Rewriting the expression for the system fuel self-collision probability as

$$\frac{1 - P}{1 - P_{ff}} = 1 + \frac{\frac{\Sigma_t^f}{\Sigma_e}(1 - P_{ff})}{\frac{\Sigma_t^f}{\Sigma_e}(1 - P_{ff}) + X} \tag{3.60}$$

and utilizing the expressions for γ and γ_f in the limit of black fuel gives

$$\gamma = 1 + \frac{1}{1 + X} \; or \; X = \frac{\gamma}{1 - \gamma} \tag{3.61}$$

The rational approximation is now introduced for both the cell and system fuel self-collision probabilities such that, $P_{ff} = \frac{\Sigma_t^f}{\Sigma_t^f + \alpha\Sigma_e}, P = \frac{\Sigma_t^f}{\Sigma_t^f + \alpha\Sigma_e}$ therefore,

$$\frac{\Sigma_t^f}{\Sigma_t^f + \alpha\Sigma_e} = \frac{\Sigma_t^f}{\Sigma_t^f + \alpha\Sigma_e} + \frac{\Sigma_t^f (1 - P_{ff})^2}{\Sigma_t^f (1 - P_{ff}) + \frac{\gamma}{1 - \gamma}\Sigma_e} \tag{3.62}$$

After some algebra it can be shown that,

$$\alpha = \frac{a}{1 + \frac{a(1-\gamma)}{\gamma}} \tag{3.63}$$

The Dancoff factor is evaluated by numerical integration for infinite array of pin-cells arranged on either a square or triangular lattice for PWR or VVER type lattice. The Dancoff factors for these regular lattices may be adjusted to give a first order correction for irregularities in the lattice, for instance water gaps between assemblies or water filled guide tubes.

WIMSD can also calculate the Dancoff factor for cluster geometries and plate bundles. For these geometries effective values of Σ_p are determined for both a finite and an infinite lattice. For cluster geometries, the resonance shielding is greater in the center of the cluster than on the outside where the pins are only shielded on one side. An approximation correction for this spatial effect is introduced by using infinite lattice values of Σ_p for all pins except the outer ring of pins. The resonance integral of the outer ring of pins I_o is then set such that as to preserve the resonance integral of the average cluster pin I_c.

$$I_o = I_I + (I_c + I_I)\frac{M}{M_o} \tag{3.64}$$

Where I_I is the infinite lattice resonance integral, M is the number of pins in the cluster and M_o is the number of pins outside ring.

3.15 SUB-GROUP THEORY

The sub-group method in WIMS was developed in order to be able to extend the resonance treatment to more complex geometries, originally the double heterogeneity of the DRAGON HTR fuel elements, but more recently, a wide range of complex systems, including submissions of fuel pins; (Newton, 2004). As in the equivalence treatment, the sub-group method must yield broad group effective cross sections that can be used in standard coarse group, coarse mesh transport solution. Ideally, effective absorption, transport, scattering and fission cross sections are required for all nuclides that are affected by resonance shielding; also these cross sections must be derivable from the data in the WIMS library, and in particular, from the existing tabulations of resonance integrals, which contain data as a function of library broad group, and temperature; (Newton, 2004).

3.16 REACTOR DESIGN AND CORE MODELS

The reactor is formed of the core, annulus and bottom reflectors, tank wall, irradiation sites, upper grid, shim tray, and the control rod. Other less important

components were eliminated. A detailed design and geometry of NIRR- 1 core has been published in (Jonah et al., 2006; Balogun, 2003b; Ahmed et al., 2002; Jijin, 1996). A summary of the design parameters of NIRR-1 is shown in table 3.1. Since the approach to the calculations of a new fuel would require principally the quantity of uranium and the total number of fuel elements, it would be convenient to adopt a model of the reactor (and in particular of the core) that considers the conservation of matter in the core rather than a very detailed model in which the single pins are to be described. A very simple model of the reactor has been constructed (Figure 3.1).

The same number of neutron groups and the same limits of these groups, which pertain to the previous models (Albarhoum, 2005a) are adopted here. This model run faster than the model described in (Albarhoum, 2005a) for the diminished number of components forming the reactor. The saving in the running time would be about 40% (this saving refers to the total saving which comprises the time of running WIMSD4 for the cell calculations, plus the time of running CITATION for the core calculations).

Table 3.1: Summary of the design parameters of NIRR-1

Parameter	Value
Reactor type	Tank-in-Pool (MNSR)
Nominal power (kW)	30
Fuel material (enrichment %)	UAl4-Al (~90.2%)
Core geometry	Cubic cylindrical
No. of fuel pins	344
Fuel pin active length (mm)	230
Fuel pin dimensions	
(a)Diameter (mm)	4.3
(b) Total length (mm)	248
Clad material	Al-303
Clad thickness (mm)	0.6
Number of control rods	1
Control rod material	Cadmium
Control rod diameter including S.S. tube (mm)	50
Control rod active length (mm)	230
Reflector material	Metallic Beryllium
Reflector thickness (radial, mm)	100
Reflector thickness (bottom, mm)	500
Reflector thickness (top; added as shim, mm)	0–120

3.17 MATERIALS AND IMPLEMENTATION

Using the above described model for the Nigerian MNSR (NIRR-1) the following parameters would be found (see Tables 4.1 and 4.2) for the actual reactor using HEU fuel. There are 3 dummy elements made of aluminum having the same external diameter of the fuel rods plus other 4 tie rods connecting the upper and lower grids. They are assumed to be made of aluminum too.

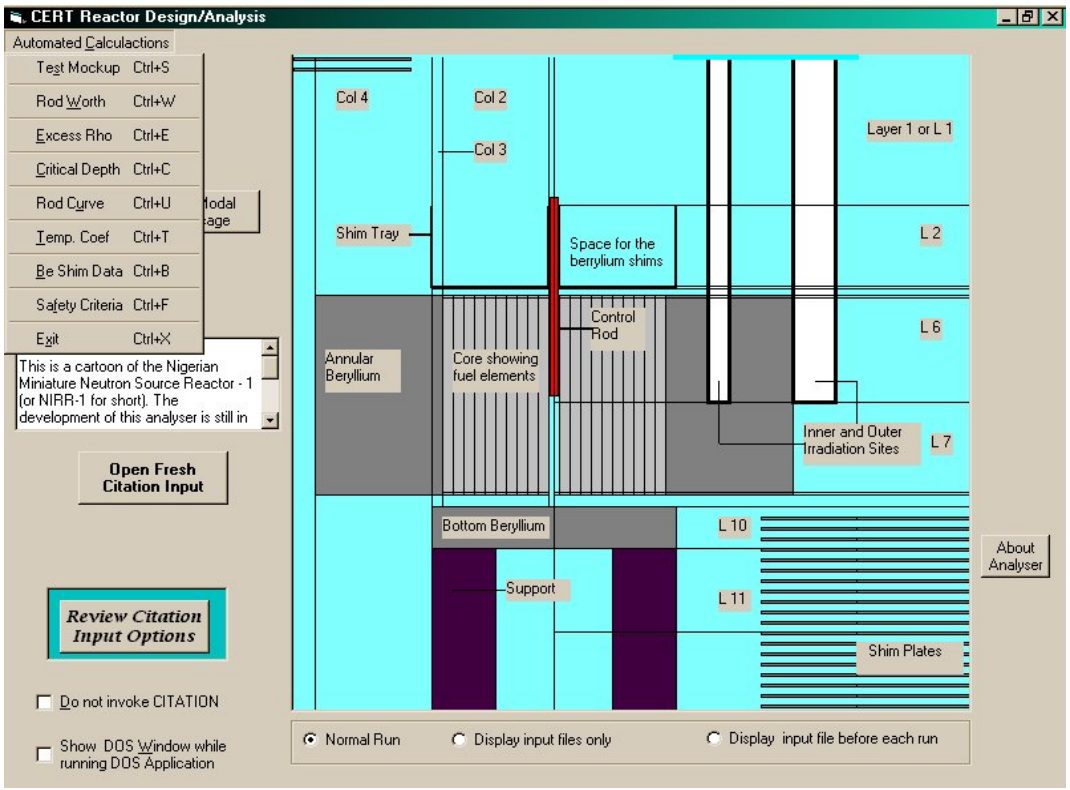


Figure 3.1: A vertical cross section of NIRR-1 and menu of various neutronics analysis that the user may carry out automatically

The process of preparing WIMS input files (appendix A), generating their output files, identifying and exporting the group constants required for CITATION base input files (see appendix B), formatting the input files such that CITATION would understand is quit cumbersome, time consuming, and prone to errors. It is estimated that an expert reactor analyst/designer would require not less than twenty-four hours to achieve this while a beginner would take days to obtain a result for a single calculation say calibration of the control rod at a given temperature. Another disadvantage of the manual procedure is that once a failure (result convergence or power supply) occurs or there is need to review any parameter, say temperature, one would have to start all over from scratch. For instance the MNSR operates within the temperature range of 20^{o}C and 80^{o}C. If the control rod is to be calibrated manually within this temperature range, say at intervals of 1^{o}C, group constants have to be generated for each new temperature. If there is need to alter the interval, say to 0.5^{o}C or 2^{o}C, then the whole process has to be carried out from the scratch for any of the intervals independently. And each time there is an error, the process has to start afresh.

This work has taken care of all these tedium by allowing the user to modify input options as desired and at any interval without initial procedures as long as one CITATION base input file has been prepared. The desired calculations can be carried out by clicking the button for such a calculation. In the case of power failure or a pause at user's discretion, the scheme automatically saves the last file it worked with so that whenever the code is invoked, the user has the option to resume the suspended calculation from the point where it stopped. In a situation where there is no convergence, the scheme adjusts at least one input parameter as predetermined by the user until a convergence is arrived at. The adjustment can be upwards or downwards depending on

which direction favors convergence as deemed by the scheme. Hence, reactor calculations that usually take several days can now be done within hours.

To model the neutronics of NIRR-1, a simplified scheme of WIMS and CITATION (Balogun, 2003a) system of computer codes were employed. The two codes were interfaced via a dozen visual BASIC codes (see appendices C and D). WIMSD (Halsall, 1980; Askew et al, 1966) uses 69 group multi region integral transport theory to solve the neutron transport equation for the lattice cells. The group collapsed cross sections were obtained from WIMSD and were used as input for CITATION (Fowler et al, 1971), which uses finite difference scheme to solve the neutron diffusion equation from one up to three dimensions.

To generate the group constants (Tayyab et al., 2004) for fuel region, control rod, control rod follower, reflectors and other non-fueled regions, unit cell calculations based on Wigner-Seitz (Iqbal et al., 2007; Lamarsh, 1966) cell modeling was used. The lattice structure of the core was represented in the form of a super cell (Figure 3.2), which comprises of a fuel, cladding, moderator, dummy and a structure zone. Hence this unit fuel cell is representative of the entire core modeled in WIMS. The central region of cell contains fuel material. Around the fuel meat there is aluminum cladding. Third annulus contains water as moderator while the outermost region is for the structural material of core, which includes aluminum dummy pins and tie rods of the fuel cage.

The 69 energy group WIMSD library was collapsed to produce a four group self shielded cross-section data set. Based on energy limit for neutron up scattering, unresolved resonances and inelastic scattering, the energy group structure with boundaries (Table 3.2) were chosen. Group constants generated by lattice cell calculations were used to represent different core regions in CITATION. Macroscopic cross-sections were generated using WIMSD while CITATION was used for core modeling.

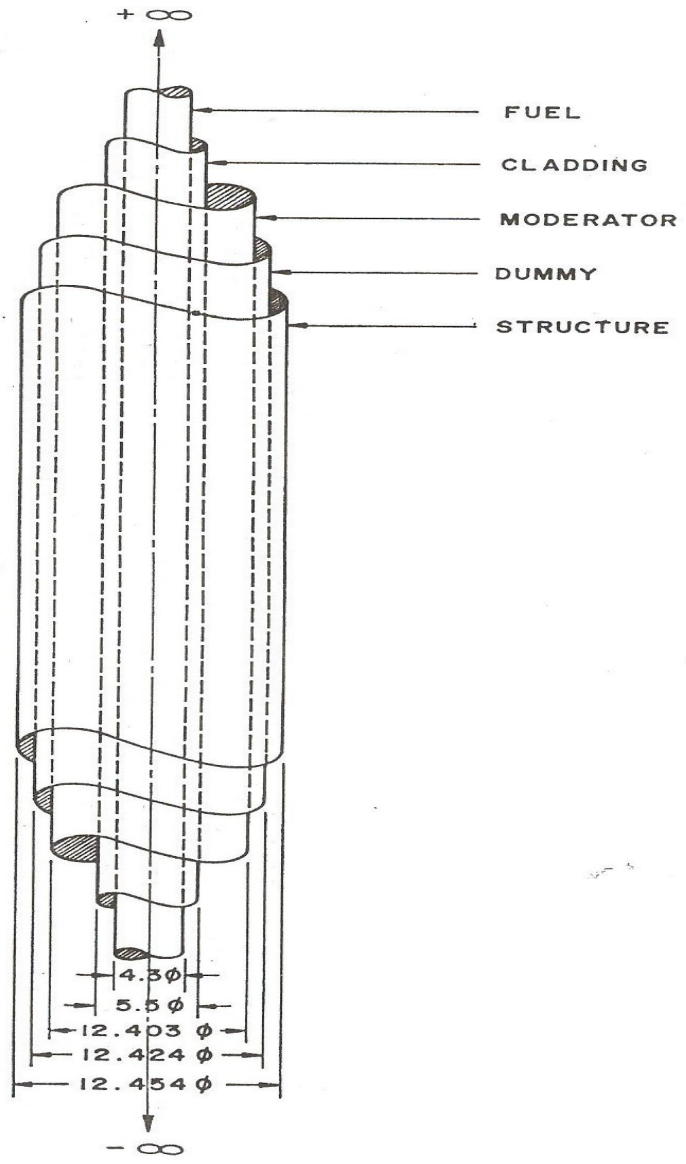


Figure 3.2: Supper Cell

Both HEU and LEU cores were analyzed employing the same cell model because both cores contain same number of fuel pins with same dimensions. Only the contents of fuel and clad material were changed while analyzing the LEU fuelled core. These macroscopic cross-sections have been employed in CITATION. As NIRR-1core is of square cylindrical shape, it's modeling in r-z geometry of CITATION code was achieved by conserving the total area of core and beryllium reflector (figure: 3.3).

Table 3.2: Details of the energy group structure used in WIMS-D4 for condensation in the transport theory-based group constant generation

Group number	Energy (eV)	
	Upper limit	Lower limit
1	1×10^{7}	1.3530×10^{6}
2	0.8210×10^{6}	9.1180×10^{3}
3	5.530×10^{3}	0.780
4	0.625	0.180
5	0.140	5.0×10^{-3}

In this regard, effort has been made to simulate the actual system as closely as possible. On the same principle irradiation sites, fission chambers and control rod guide tube have been modeled.

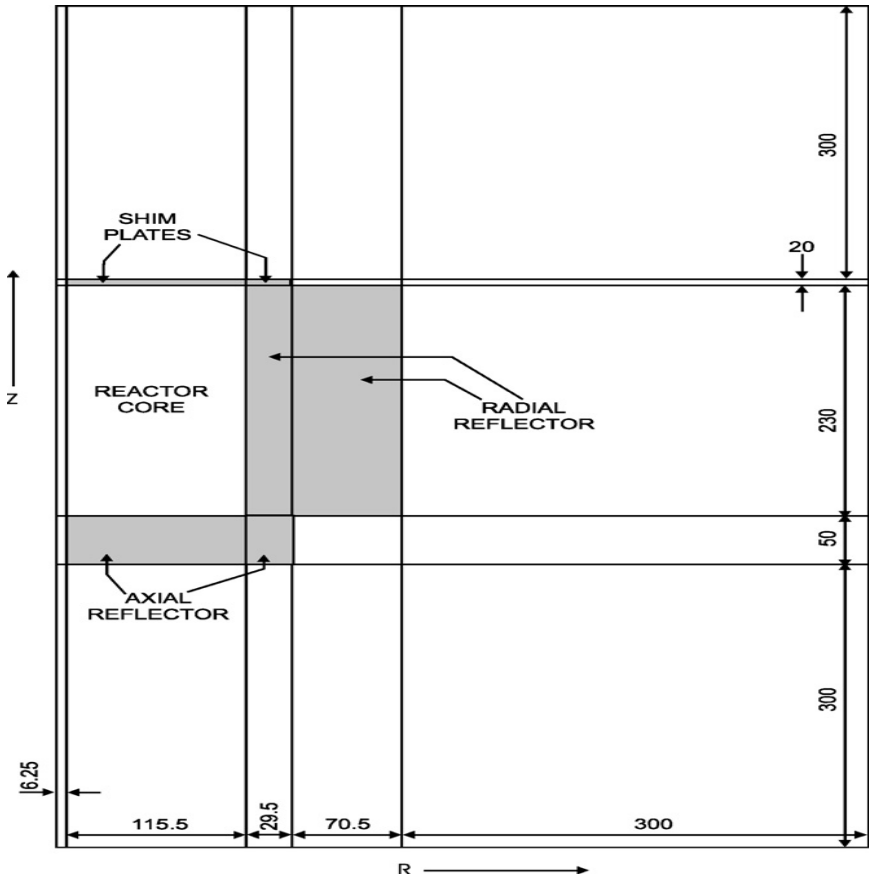


Figure: 3.3: Details of r-z model used for neutronics analysis of NIRR-1.

On invocation, the code prompts the user to supply some information from which it is able to locate the control rod exactly by automatically modifying the CITATION input data

before proceeding with each CITATION run and the user has the option to change this input file to his/her CITATION input file. On invocation, the code displays a vertical cross-section of a typical Miniature Neutron Source Reactor (MNSR), showing the reactor core, the central control rod, the annular and bottom beryllium reflectors, the shim tray, as well as the inner and outer irradiation sites as depicted by figure 3.1 above:

A pull-down menu called 'Automated Calculations' containing "Test Mockup", " Rod Worth", "Excess Rho", "Critical Depth", "Rod Curve", "Be Shim Data", "Temp. Coef" and "Safety Criteria" is also presented to the user. When any of these calculations is started, DOS runs in the back ground by default except elected by the user to have the DOS window displayed while the DOS application is running. Text boxes allowing the user to make changes to the default values that are applicable are displayed upon invocation of the code as shown by figure 3.4. The values that can be changed by the user include:

- CITATION input file
- The integer value representing the group constant data for the control rod as provided in section 008 of the CITATION input file
- The length of the control rod
- The columns occupied by the annular beryllium reflector and the thickness of annular beryllium reflector
- The columns occupied by the bottom beryllium reflector and the thickness of bottom beryllium reflector

The movement of the control rod from location to location is reflected on the monitor as well as a rotating moon with the message "CITATION is running" to warn the user from

interrupting CITATION while it is running. At the end of each CITATION run, the scheme examines the output and extracts the calculated k_{eff}.

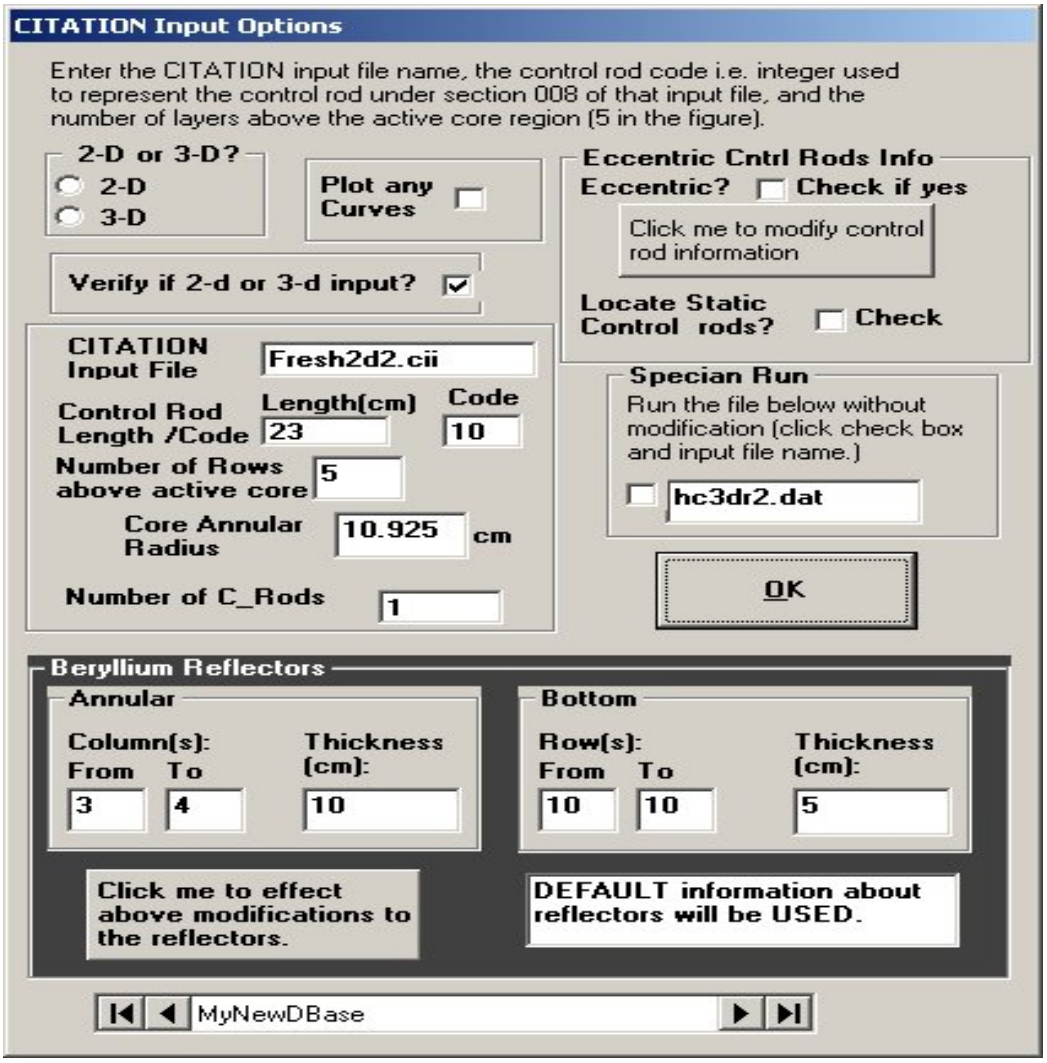


Figure 3.4: Form allowing the user to interactively modify dimensions of major core components and specify other parameters

3.18 ENRICHMENT

Two options are available to change enrichment which are either by fixing the mass loading of U-235 in the core and varying the enrichment in a specified steps of reduction or increment or by fixing the desired enrichment and varying the mass loading of U-235 in the core (see Figure 3.5). This is accomplished by checking or un-checking a box provided for this purpose. When the box provided is checked, a desired enrichment is predefined and the interval for changing mass loading of U-235 in the core is specified. Each time a calculation is not successful, the mass loading is adjusted either upwards or downwards depending on which direction favors convergence and the number density of U-235 is re-evaluated before proceeding with the main calculation. When un-checked, the desired mass loading is predefined and the steps of increase or decrease in enrichment are specified.

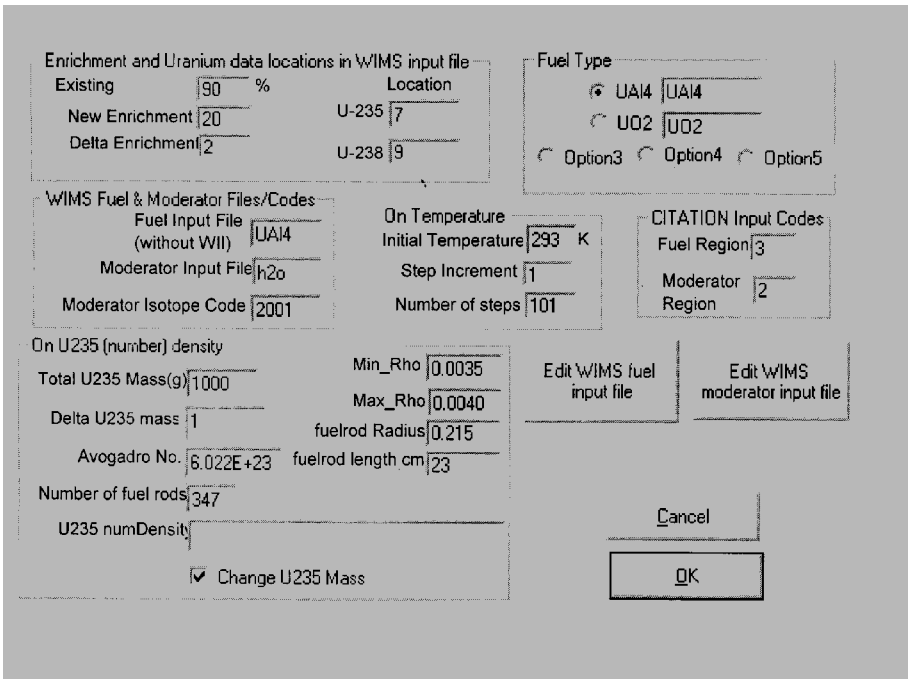


Figure 3.5: Enrichment and uranium data location in WIMS and CITATION input files

Each time a calculation is not successful, the core enrichment is adjusted either upwards or downwards depending on which direction favors convergence and the number density of U-235 is re-evaluated before proceeding with the main calculation. This number density is evaluated as shown below;

$$volume\ of\ fuel\ pin = \pi r^2 l\ or\ \frac{1}{4}\pi d^2 l \qquad (3.65)$$

$$Where\ the\ radius\ of\ one\ fuel\ meat = \frac{fuel\ meat\ diameter}{2} = \frac{4.3}{2}$$
$$= 2.15\ mm$$

$$and\ length\ of\ fuel\ meat = 230\ mm\ ;$$

$$\therefore volume\ of\ one\ fuel\ meat\ V_{fp} = \pi \times (2.15)^2 \times 230 \times 10^{-9}$$
$$= 3.340 \times 10^{-6} mm^3 = 3.340 cm^3\ U^{235}$$

$$Avogadro's\ number\ of\ U^{235}\ atomweighs\ 235 grams \qquad (3.66)$$

$$1\ atom\ of\ U^{235}\ weighs\ \frac{235}{Avogadro's\ number}\ grams \tag{3.67}$$

$$n\ atom\ of\ U^{235}\ weighs\ \frac{235n}{Avogadro's\ number}\ grams \tag{3.68}$$

$$Number\ of\ fuel\ pins\ (347)\ contain\ 1000\ grams\ of\ U^{235} \tag{3.69}$$

$$1\ fuel\ pin\ contains\ \frac{1000}{number\ of\ fuel\ pins}\ grams\ of\ U^{235} \tag{3.70}$$

This is the weight of U^{235} in one fuel pin.

$$Number\ of\ U^{235}\ in\ 1\ fuel\ pin = \frac{U^{235}\ weight\ in\ one\ fuel\ pin}{weight\ of\ one\ U^{235}\ atom} \tag{3.71}$$

$$U^{235}\ number\ density = \frac{Number\ of\ U^{235}\ in\ fuel\ pins}{volume\ of\ fuel\ pins} \tag{3.72}$$

$$i.e\ U^{235}\ number\ density = \frac{2.881844N_A}{235 \times 3.340} = 2.21104 \times 10^{21}$$

$$Uranium\ concentration\ in\ Al = 27.5\ weight\ \%;$$

$$Al\ concentration\ in\ alloy = 72.5\ weight\ \%$$

$$Al\ weight\ in\ one\ fuel\ pin = \frac{72.5}{27.5} \times 2.881844 = 7.5976g;$$

$$U^{238}number\ density\ = 2.4257 \times 10^{20};$$

$$Al\ number\ density = \frac{7.5976N_A}{27 \times 3.340} = 5.0735 \times 10^{22};$$

$$pure\ Zr\ number\ density = \frac{6.506N_A}{91.224} = 4.29483 \times 10^{22}$$

These equations are coded into the scheme such that once the mass loading of U-235 in the core is known; the scheme can automatically determine the number density. Each time that there is a change in mass loading of U-235; a new number density is evaluated before proceeding to carry out other calculations.

3.18 TEST MOCKUP

When this menu item is clicked, two tests are performed. First, the control rod is fully withdrawn and the effective multiplication factor k_{eff} is calculated. If a value greater than unity is returned, the message "Reactor has passed first test" is displayed and the second test follows. Otherwise, "Reactor has failed first test" is displayed. The control rod is then fully inserted and again CITATION is invoked to compute k_{eff} whose value is expected to be less than unity. Again, the appropriate message is displayed as to whether the reactor has passed or failed the second test. Failure of any of these two tests clearly indicates that the reactor that has been put together is unreasonable hence the analyst/designer must go back to the drawing board. Otherwise, if the design passes both tests, then the design process can be carried forward and other tests may follow.

3.19 ROD WORTH

Control rods are essentially black to thermal neutrons and cause drastic thermal flux depression when inserted. On clicking this menu, CITATION input data is modified such that the control rod is fully inserted and the value of $k_{eff} = k_{in}$ is noted; next the control rod is fully withdrawn and again the value of $k_{eff} = k_{out}$ is noted. The control rod worth is then returned as the difference $k_{out} - k_{in}$.

3.20 EXCESS RHO AND CRITICAL DEPTH

On clicking this item, k_{out} is again calculated as for rod worth and the excess reactivity,ρ_{ex}, is obtained through $\rho_{ex} = k_{out} - 1$. The values k_{eff} for fully inserted and fully withdrawn control rod are calculated to ensure that between these extremes a critical depth of insertion where $k_{eff} = 1$ exists. Satisfied that this is the case, the scheme then proceeds to locate the critical depth systematically. The control rod is exactly half inserted by modifying CITATION input appropriately to reflect this fact. CITATION is then run and the value of k_{eff} is again obtained. If $|k_{eff} - 1|$ is greater than the tolerance set by the user and $k_{eff} > 1$ then the control rod is relocated such that it moves from half inserted to three-quarter inserted and CITATION is run again. If on the other hand $|k_{eff} - 1|$ is less than the tolerance set by the user and $k_{eff} < 1$ then the control rod is relocated such that it moves from half inserted to one-fourth inserted and CITATION is run again. On the next pass, the control rod is moved one-eighth of the core length up or down from the preceding position according to whether the new calculated value of k_{eff} is greater or lees than unity. This process continues until the depth of insertion converges to the critical depth of insertion.

3.21 ROD CURVE

This is associated with the generation of the integral rod curve data. When clicked, the user is prompted to confirm the excel path, to change temperature options desired to carryout this calculation, to elect whether the calculation should be done just in the core or in the whole of the reactor, and to supply withdrawal step in centimeters (cm). After the user has supplied the necessary information and clicks OK, CITATION input data is modified to reflect full control rod insertion and the corresponding k_{eff} is obtained. The control rod is then withdrawn in steps

chosen by the user. CITATION runs after each withdrawal until the control rod is fully withdrawn and a table of k_{eff} versus depth of insertion of the control rod is then produced. From this table, the graph of integral rod worth against depth of insertion is plotted automatically if Microsoft Excel (Windows'97 or later versions that are DOS compatible) is installed on the user's computer.

3.22 BERYLLIUM SHIM DATA

To generate the data for the beryllium shim thickness against corresponding reactivity insertion, this button is clicked. An option form appears which allows the user to easily modify the thickness of each beryllium shim, the row that each of the shims should occupy if different from the default row, and the state of the control rod that the user wishes to generate the beryllium shim data. On clicking OK, the calculations proceed automatically.

3.23 TEMPERATURE COEFFICIENT

The most important physical quantity that is related to the inherent safety of the reactor is the temperature coefficient of reactivity. Reactivity coefficients; the differential changes in reactivity produced by differential changes in reactor core conditions; are useful in evaluating the response of the core to external disturbance. On clicking this button, a form (Figure 3.6) availing the user various options appears. Amongst these are eight text boxes for the user to input or modify.

i. WIMS Input File (fuel region) - this is where the user specifies the name of the WIMS fuel region filename, that is the file from which the fuel region group constants may be generated.

ii. Moderator Region - the name of the WIMS input file from which the moderator region group constants may be generated is specified here.

iii. Initial Temperature - this is where the starting temperature for the calculation of the temperature coefficient of reactivity is specified.

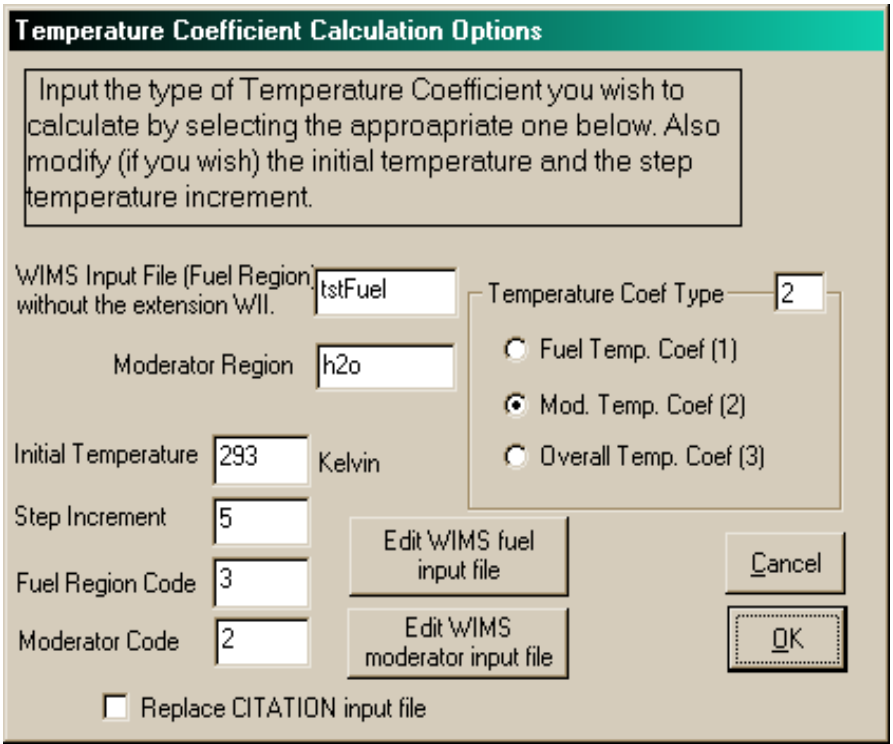
iv. Step Temperature Increment (Delta_T)

v. Number of Steps – the number of steps to be taken is specified here.

vi. Temp Interval- this allows for specification of the temperature interval.

vii. Fuel Region Code- the integer used to represent the fuel region in CITATION input is entered here. This is necessary so that the scheme can identify which group constants to replace in the user's CITATION input after each modification and re-calculation of group constants by WIMS.

viii. Moderator Region Code - the integer used to represent the moderator region in the



CITATION input is specified here as in (vii) above.

Figure 3.6: Form for temperature coefficient of reactivity options

This form provides options for specifying the type of temperature coefficient the user intends to compute (fuel, moderator, or overall temperature coefficient). Available also are options to edit or examine both WIMS input files for the moderator region and the fuel region. When satisfied with the options, the user clicks OK to automatically start the required temperature coefficient of reactivity calculation. The result is displayed indicating whether it is positive or negative.

3.24 SAFETY CRITERIA

The most significant safety parameter related to core physics analysis is the shutdown margin (SDM). This parameter is obtained by subtracting the positive core excess reactivity required to overcome xenon poisoning, fuel depletion, and the power defect from the total control rod reactivity worth. Usually technical specification require that the SDM be of at least $30\% \frac{\Delta k}{k}$. Any difference between the estimated SDM and the limiting value represents excess reactivity available for experiments. When this button is clicked, immediately the code inserts the control rod into the reactor, putting it in the fully withdrawn position by modifying the input data appropriately before CITATION is invoked. At the next pass the control rod is located at fully inserted position.

From the two calculated k_{eff} values, the safety reactivity factor (SRF) and the shutdown margin (SDM) are obtained as follows:

$$SRF = \frac{control\ rod\ worth}{core\ excess\ reactivity} = \frac{\rho_{cr}}{\rho_{ex}} \qquad (3.73)$$

And

$$SDM = \rho_{cr} - \rho_{ex} \qquad (3.74)$$

where $\rho_{cr} = \frac{k_{out} - k_{in}}{k_{out} k_{in}}$.

CHAPTER FOUR

RESULTS AND DISCUSSION

A core loading requirements in converting HEU MNSR (NIRR-1) core to LEU in steps of 5% using WIMS and CITATION was carried out. At each step of the conversion process the following important calculations, some of which have a bearing on safety, were performed:

a. Testing the Mockup;

b. computing built-in excess reactivity;

c. calculating the control rod worth;

d. locating the critical depth of insertion of the control rod;

e. calibration the control rod and;

f. determining safety reactivity factor;

g. Shutdown margin.

This is because the results of these calculations are liable to change with change in the core composition and where they have a bearing on safety, they have to be maintained within acceptable limits.

In this work, the computational scheme developed systematically reduces the enrichment from 90% to 20% and even below, in steps of 5%. At each step, important neutronics parameters, especially safety related parameters (k_{eff}, SRF, SDM, critical depth of insertion, control rod worth, and critical depth of insertion) are computed. Two fuel types are considered: UAl_4 and UO_2 and the variation of safety reactivity factor (SRF) with enrichment, control rod curve and shim curve are discussed. All the calculations were done automatically without manually opening WIMS or CITATION input files after preparing the base data.

Table 4.1: Results for UAl_4 fuel from low to high enrichments in steps of 5

Enrichment (%)	U-235 core loading (g)	K_{eff} value	S.R.F	Rod worth (mk)	S.D.M (mk)	Excess Reactivity (mk)	Critical Depth (cm)	K_{eff} at Critical Depth
05.00	2505	1.00352	1.239	4.36	0.84	3.52	18.69	1.000033
10.00	1448	1.00354	1.588	5.63	2.08	3.56	13.66	1.000013
15.00	1274	1.00353	1.696	5.99	2.46	3.53	12.94	0.999993
20.00	1201	1.00386	1.595	6.16	2.18	3.86	13.66	0.999985
25.00	1155	1.00360	1.747	6.28	2.69	3.60	12.58	0.999976
30.00	1125	1.00351	1.809	6.35	2.84	3.51	12.22	1.000000
35.00	1107	1.00399	1.605	6.40	2.41	3.99	13.66	0.999953
40.00	1090	1.00393	1.639	6.44	2.51	3.93	13.30	0.999983
45.00	1074	1.00430	1.833	6.49	2.95	3.54	12.22	0.999950
50.00	1063	1.00365	1.784	6.52	2.86	3.65	12.22	1.000047
55.00	1052	1.00361	1.812	6.54	2.93	3.61	12.22	0.999990
60.00	1043	1.00364	1.803	6.57	2.92	3.64	12.22	1.000008
65.00	1034	1.00351	1.876	6.59	3.08	3.51	11.86	1.000024
70.00	1027	1.00359	1.843	6.61	3.02	3.59	12.22	0.999928
75.00	1023	1.00405	1.634	6.62	2.57	4.05	13.30	0.999989
80.00	1012	1.00409	1.862	6.65	3.08	3.57	11.86	1.000043

85.00	1004	1.00352	1.894	6.67	3.15	3.52	11.50	1.000087
90.00	998	1.00375	1.627	6.69	2.94	3.75	12.22	1.000046

Table 4.2: Results for UO_2 fuel from low to high enrichments in steps of 5

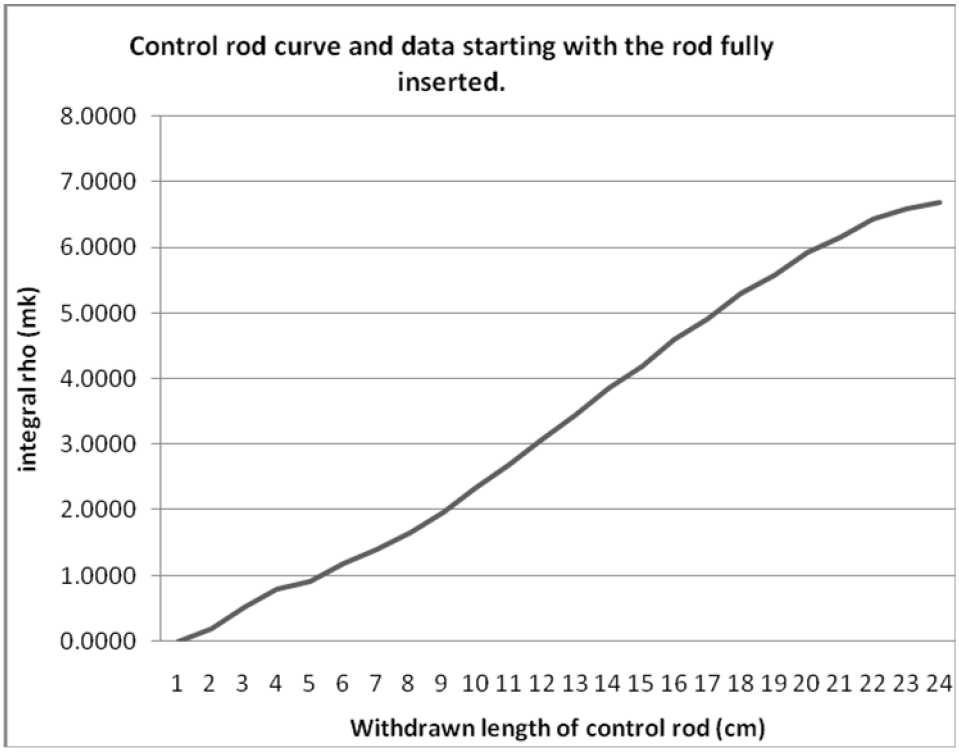
Enrichment (%)	U-235 core loading (g)	K_{eff} value	S.R.F	Rod worth (mk)	S.D.M (mk)	Excess Reactivity (mk)	Critical Depth (cm)	K_{eff} at Critical Depth
05.00	2505	1.00352	1.239	4.36	0.84	3.52	18.69	1.000033
10.00	1389	1.00352	1.666	5.87	2.35	3.52	12.94	1.000069
15.00	1216	1.00355	1.769	6.27	2.73	3.55	12.22	1.000080
20.00	1141	1.00355	1.827	6.48	2.93	3.55	12.22	0.999969
25.00	1098	1.00354	1.867	6.60	3.07	3.54	11.86	1.000052
30.00	1070	1.00362	1.847	6.68	3.06	3.62	12.22	0.999926
35.00	1049	1.00359	1.881	6.75	3.16	3.59 3.57	11.86	1.000021
40.00	1033	1.00357	1.898	6.78	3.21	3.54	11.50	1.000086
45.00	1020	1.00354	1.929	6.83	3.29	3.63 3.67	11.50	1.000034
50.00	1009	1.00363	1.892	6.86	3.23	3.64	11.86	0.999994
55.00	999	1.00367	1.877	6.89	3.22	3.60 3.62	11.86	1.000020
60.00	990	1.00364	1.902	6.92	3.28	3.52	11.50	1.000076
65.00	982	1.00360	1.927	6.94	3.34	3.53 3.58	11.50	1.000030
70.00	975	1.00362	1.927	6.97	3.35	3.56	11.50	1.000030
75.00	968	1.00352	1.984	6.98	3.46		11.50	0.999927
80.00	960	1.00353	1.984	7.01	3.47		11.50	0.999924

85.00	953	1.00358	1.965	7.03	3.45		11.50	0.999954
90.00	946	1.00356	1.983	7.06	3.50		11.50	0.999922

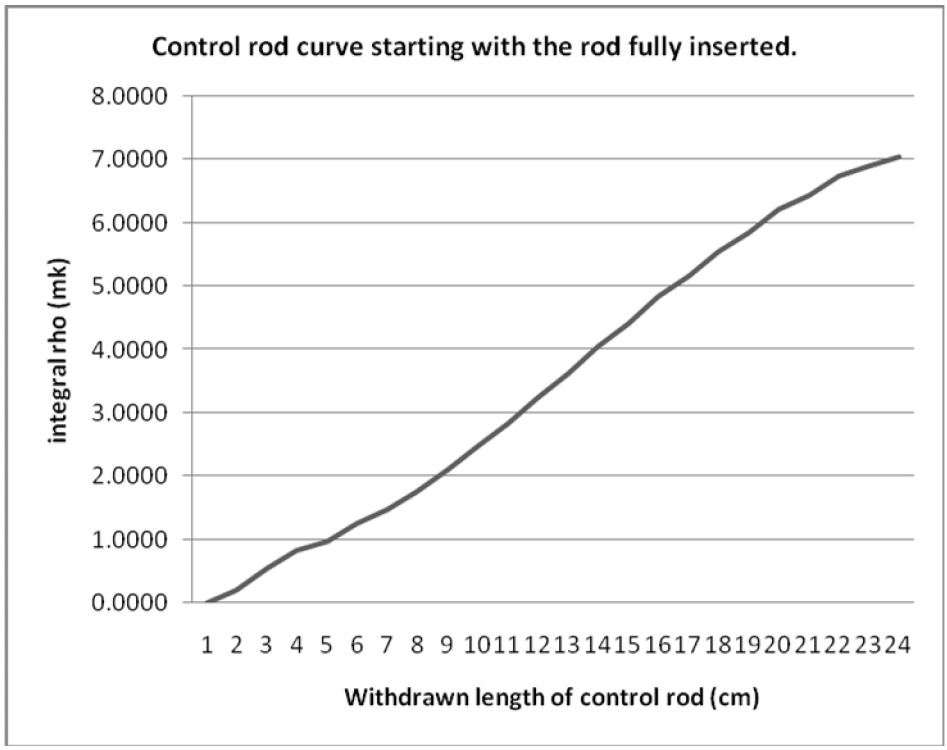
The excess reactivity of UAl_4 was found to be 3.57mk, 3.52mk, and 3.86mk at 10%, 15% and 20% enrichments respectively. UO_2 gave 3.53mk at 10%, and 3.55mk for both 15% and 20%. This is in good agreement with the design specification of 3.5 – 4.0 mk for MNSR as reported by (Jonah et al., 2007; Chengzen, 1993; the First Safety Analysis Report (FSAR, 2005)) and HEU values reported by (Balogun, 2003). The results from tables 4.1 and 4.2 show that the SDM values are within the range of 2.0 and 3.5 mk which is in good agreement with the SDM specification for the MNSR. On safety criteria, the Safety Reactivity Factor (SRF) of NIRR-1 should be greater than 1.5 as reported in Balogun (2003a). This implies that the greater the ratio of the control rod worth to the core excess reactivity is, the better and safer the reactor would be. This would give room for a wider shutdown margin enabling the reactor to be gradually shutdown when in operation. The results obtained for UAl_4 and UO_2 at LEU compares favorable with HEU values given in Balogun (2003).

When the value of multiplication factor k_{eff} is greater than unity, the reactor assembly is said to be supercritical and when the factor is less than unity, it is called a subcritical assembly. For a fission chain reaction to be just continuing at a steady state, k_{eff} has to be equal to unity. It is then expected that when the control rod (23 cm long) is fully withdrawn, the supercritical condition should be attained while sub-criticality should be attained when the control rod is fully inserted. It is desirable that criticality should be attained about midway between full insertion and full withdrawal of the control rod so as to allow for easy control and regulation of reactivity. The results obtained show critical depths which approximately satisfied this expectation.

The control rod worth is the amount of reactivity that corresponds to full withdrawal or insertion of the control rod from or into the reactor. It is the reactivity control ability of the control rod. Under actual reactor operation, it is necessary to know the reactivity equivalence corresponding to the unit distance in the core that the control rod travels. From the NIRR-1 zero power experiments, an s-shaped curve of the control rod worth is required. The results obtained give the so-called s-curve as needed (graphs 4.1 and 4.2). The results at 20% and 90% were compared and the comparison shows quite a remarkable agreement with control rod curves having similar characteristics for both fuel types.

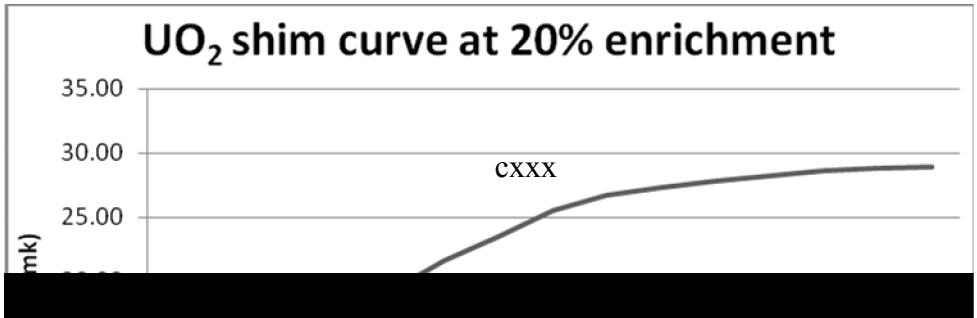


Graph 4.1: Control rod curve for UAl_4 at 20% enrichment

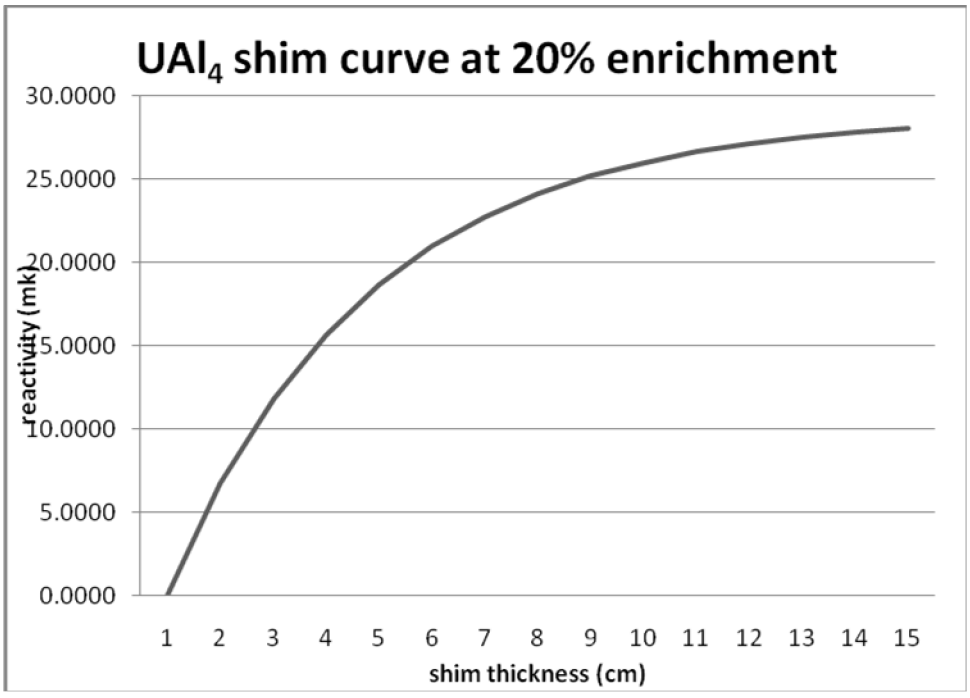


Graph 4.2: Control rod curve for UO_2 at 20% enrichment

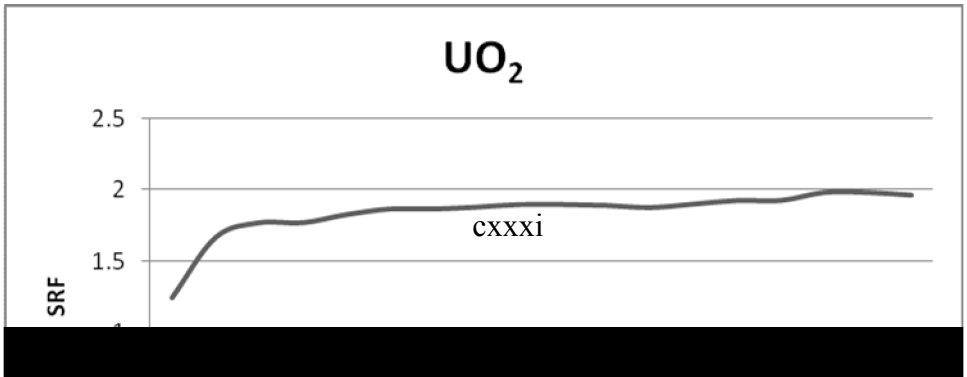
The plots in graphs 4.3 and 4.4 show that beyond about 15 cm shim thickness, the reactivity insertion is already saturated. The variation of SRF with enrichment for UO_2 (graph 4.5) indicates that SRF increases steadily with increase in enrichment while that of UAl_4 (graph 4.6) did not show this characteristic rather oscillates between 1.5 and 2.0. This behavior of UAl_4 is due to the compact nature of the core, which was designed to cause insufficient thermal circulation of coolant in the core; (Ahmed et.al, 2008). However, both at LEU and HEU, the variations are within 1.5 and 2.0.



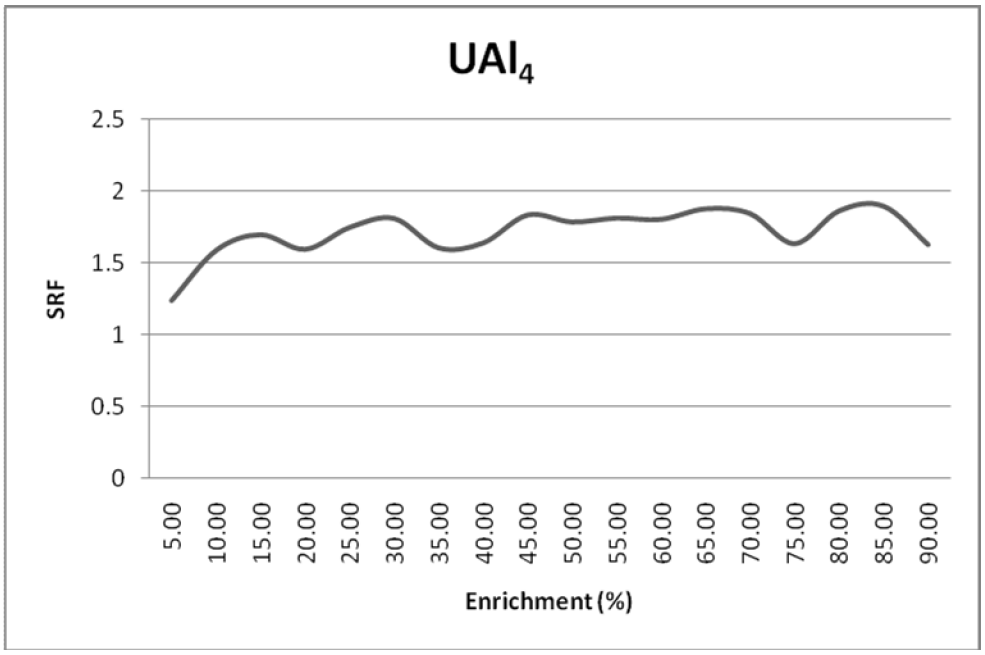
Graph 4.3: A Plot of reactivity against shim thickness for UO_2 at 20% enrichment



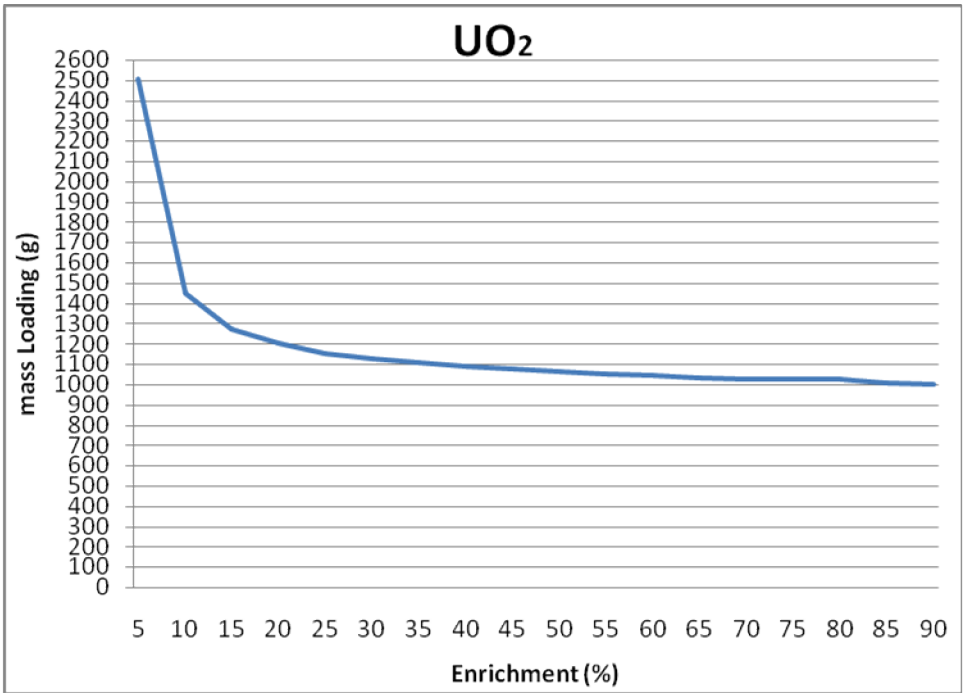
Graph 4.4: A plot of reactivity against shim thickness for UAl_4 at 20% enrichment



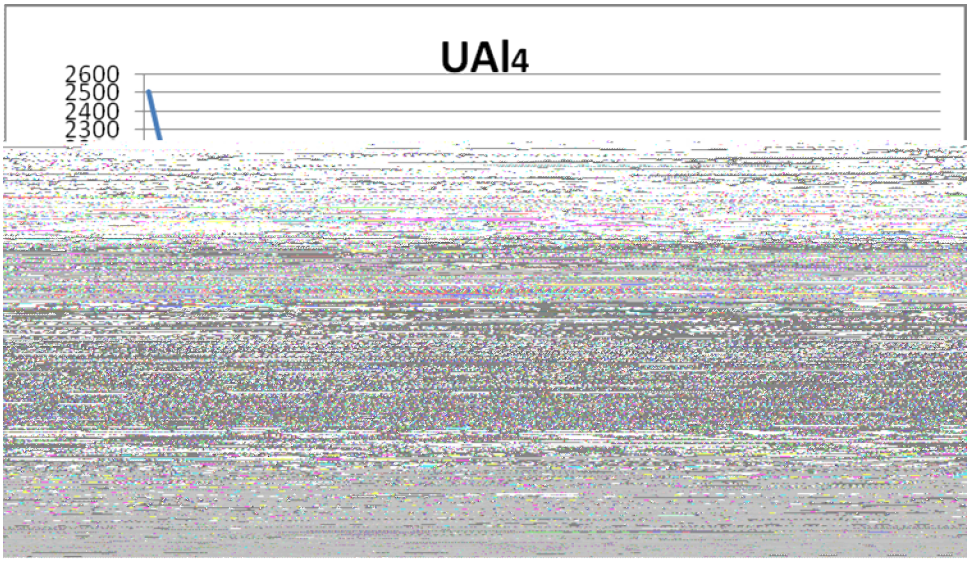
Graph 4.5: A plot of safety reactivity factor against enrichment for UO_2



Graph 4.6: A plot of safety reactivity factor against enrichment for UAl_4



Graph 4.7: plot of mass loading against enrichment for UO_2



Graph 4.8: plot of mass loading against enrichment for UAl_4

The mass of uranium increases sharply as enrichment of the material decreases. The plots in graphs 4.7 and 4.8 display this behavior. The mass of uranium drops sharply from 2505g at 5% enrichment for both fuel types down to 998g and 946g enrichment at 90% enrichment for UAl_4 and UO_2 fuel types respectively.

Table 4.3: Comparison of WIMS and MCNP values of LEU for UO_2 Fuel (varying mass)

Clad mat/ thickness (mm)	WIMS values	M.C.N.P values			U-235 core loading (g)	Enrichmen (%)	k_{eff} value

4.3	Zr/0.6	1286	12.45	1.00354	1270	12.45	1.00476
4.3	Zr/0.6	1144	19.75	1.00358	1349	19.75	1.00475

Table 4.4: Comparison of WIMS and MCNP values of LEU for UO_2 Fuel (varying enrichment)

Meat diameter (mm)	Clad mat/ thickness (mm)	WIMS values			M.C.N.P. values		
		U-235 core loading (g)	Enrichment (%)	k_{eff} value	U-235 Core loading (g)	Enrichment (%)	k_{eff} value
4.3	Zr/0.6	1270	12.95	1.00388	1270	12.45	1.00476
4.3	Zr/0.6	1349	10.95	1.00424	1349	19.75	1.00475

UAl_4 were not available for comparison.

The LEU results for UO_2 fuel type were compared with MCNP values (Jonah et. al, 2006); (tables 4.3 and 4.4) and other researchers cited in literature (Waqar et al, 2008; Sampong et al, 2006; Khamis and Khattab, 1999) as shown by table 4.5. The agreement was significantly encouraging. However, MCNP values for UAl_4 were not available for comparison.

By fixing the fuel enrichment of UO_2 at 12.45% and 19.75% as given by MCNP, the mass loadings of U-235 obtained were 1286g and 1144g against 1270g and 1349g of the MCNP respectively. The effective multiplication factors given by the scheme were 1.00354 and 1.00352 respectively as shown in table 4.3 above. Similarly, by varying the enrichment and keeping the mass constant, the mass of 1270g gave 12.95% while 1349g gave 10.95% with effective multiplication factors of 1.00388 and 1.00424 respectively as depicted by table 4.4 above.

Table 4.5: Comparison of results obtained on HEU to LEU conversion for other MNSR with this work

Works	Fuel type	Enrichment (%)	Excess reactivity (mk)
UO_2	20.00	4.5790	□ Sampong et al (2006)

UO_2	11.20 12.60 19.75	3.7616 3.9799 3.5762	

It has been shown that between 90% and 10% inclusive, the values of these parameters are within the acceptable values illustrated by the results above except at the enrichment of 5% and the mass loading obtained for each enrichment can serve as a reference or starting point (mass) for future studies using the Monte Carlo code. At the enrichment of 5% and below, the operation of the MNSR can be considered unrealistic because too high proportion of the fission neutrons is lost in non-fission reactions.

There are several fuel matrices such as UO_2, UAl_4, U_3Si_2, U-Mo, and U_3O_8-Al amongst others, however, only the first two fuel matrices were considered. A proposal has been made in the scheme to accommodate such other fuel matrices in future studies.

CHAPTER FIVE
CONCLUSIONS AND RECOMMENDATIONS

5.3 CONCLUSIONS

WIMS and CITATION are powerful schemes that can tremendously facilitate attempts at modifying the MNSR. This work shows that to convert NIRR-1 from HEU to LEU using WIMS and CITATION, the core loading of U-235 has to be increased from 998g to 1201g for UAl_4 fuel while for UO_2 fuel the loading has to be increased from 946g to 1141g of U-235. The SRF values of 1.595 for UAl_4 and 1.827 for UO_2 at 20%. The variation of safety reactivity factor (SRF) with enrichment for UO_2 indicates that SRF increases steadily with increase in enrichment while that of UAl_4 did not show this characteristic indicating the compact nature of NIRR-1 core. The shutdown margin (SDM) at this enrichment is 2.18mk for UAl_4 and 2.93mk for UO_2. The control rod curves give the so-called s-curve as needed, with excess reactivity values in good agreement with the NIRR-1 design specifications. The control rod curves for UAl_4 and UO_2 curves at 20% were compared with that of UAl_4 at 90% reported in Balogun (2003a). This preliminary investigation results on conversion to LEU using the deterministic codes WIMS and CITATION indicate a drastic review of the material composition of the MNSR core to achieve enrichment of about 20%. A remarkable aspect of this work is that it brings about savings in time.

5.4 RECOMMENDATIONS

UO_2 fuel type is therefore recommended for use in NIRR-1 at low enrichment based on the results obtained (The SRF of 1.827 for UO_2 against 1.595 for UAl_4, showing that UO_2 gives a wider room for shutdown margin than UAl_4. The variation of safety reactivity factor (SRF) with enrichment for UO_2 indicates that SRF increases steadily with increase in enrichment while that of UAl_4 did not show this characteristic. Also UO_2 has a higher shutdown margin (SDM) of 2.934mk against 2.178mk for UAl_4 at LEU) and for the fact that UO_2 fuel kernel is a small-scale, safe, proliferation-resistant fuel that can be used in a variety of applications.

Considering the properties of zirconium (cladding material for UO_2) such as specific gravity, melting and boiling points, zirconium is especially more suitable in water-moderated reactors because of its low neutron absorption cross-section, excellent corrosion resistance at moderately elevated temperatures, strength, ductility, and ease of fabrication. Whereas, the fuel element essentially comprising rods of UAl_4 provided with a sheath (clad) of aluminium or an aluminium alloy contains impurities (Si, Ti, Ge, Zr, Sn, Pb, In, Tl, Fe, Nb, Ga and inert gas argon are used as an additional components) inherent to the manufacturing process. The fuel element cladding of the UAl_4 fuel of MNSR is made of aluminium alloy No. 303 and the elements are always submerged in water all the time. The impurities are expected to be in the water even though the reactor and pool waters are continuously purified. Therefore, pitting corrosion is expected, which may make it possible to damage the fuel element cladding if one batch of fuel elements operate for a long time. More so, UAl_4 fuel matrix is hard, brittle and can only last for about 10 years.

Furthermore, considering the results of WIMS and CITATION in two dimensions geometry research reactor, there is the need to extend this research work to three dimensions as in power reactor (PWR) geometry.

REFERENCES

1. Ahmed, Y.A.; Ewa, I.O.B.; Umar, I.M. (2002). Effective resonance energy and non-ideality of epithermal neutron flux distribution in neutron activation analysis. Nigerian Journal of Physics 14 (1), 82–85.

2. Albarhoum M. (2008). Automation of the modeling and some neutronics calculations of the Syrian miniature neutron source reactors. Annals of Nuclear Energy 35, 1760–1763

3. Albarhoum M. (2004). Core Configuration of the Syrian reduced enrichment fuel MNSR. Proceedings of the 2004 International Meeting on Reduced Enrichment for Research and Test Reactors, Vienna, Austria, November 7-12.

4. Albarhoum M. (2005a). The use of UAlx-Al reduced enrichment fuel in a well reflected MNSR. Proceedings of the 2005 International Meeting on Reduced Enrichment for Research and Test Reactors, Boston, USA, November 6-11.

5. Albarhoum M. (2005b): A 3-D Neutronics Model for the Calibration of the Control Rod of the Syrian MNSR. Progress in Nuclear Energy, 46, No. 2, pp. 159-164.

6. Albarhoum M. (2006). Mixed Fuel versus Low Enriched Fuel in the Syrian MNSR. Proceedings of the 2006 International Meeting on Reduced Enrichment for Research and Test Reactors, Cape Town, South Africa, October 29- November 2.

7. Albarhoum, M., (2008). Automation of the modeling and some neutronic calculations of the Syrian miniature neutron source reactor. Annals of Nuclear Energy 35, 1760–1763.

8. Albright, D., Berkhout, F., and Walker, W. (1997). Plutonium and High Enriched Uranium 1996. *World Inventories, Capabilities and Policies*. Stockholm International Peace Research Institute (SIPRI), Oxford University Press.

9. Albright, D. and Kramer, K. (2004). Stockpiles Still Growing. *Bulletin of the Atomic Scientists*, pages 14 – 16, November/December.

10. Askew J.R, Fayer F.J. and Kemshell P.B. (1966). *A General Description of Lattice Code WIMSD*. Journal of the British Nuclear Energy Society.

11. Balogun G.I. (2003a). Automating some analysis and design calculations of miniature neutron source reactor at CERT (I) Annals of Nuclear Energy 30, 81-92.

12. Balogun G.I. (2003b). On safety related calculations associated with miniature reactor core conversion studies, J. Appld. Sci & Tech. 8, 8-13.

13. Balogun G.I., Jonah S.A. (2004). *Results on zero power and criticality experiments for the Nigeria Research Reactor-1*, internal report CERT/NIRR-1/ZP/01.

14. Bell, G.I. (1959). Theory of effective cross sections. LA-2322.

15. Brissenden, R.J. and Durston, C (1965). The calculation of neutron spectra in the Doppler region. Conference on application of computing methods to reactor problems. ANL, 7050. p. 51.

16. Cullen, D.E., Clouse, C.J., Procassini, R., Little, R.C. (2003). *Static and Dynamic Criticality: Are They Different?* UCRL-TR-201506, Lawrence Livermore Laboratory.

17. DOE-HDBK-1019/1-93. DOE fundamentals handbook nuclear physics and reactor theory Vol. 1 of 2, January 1993.

18. Enrichment for Research and Test Reactors, Boston, USA, November 6-11, 2005.

19. Fowler T.B.: *CITATION-PC README File* (February 1996)

20. Fowler T.B., Vondy D.R., Cunnigham G.M., Nuclear reactor analysis code CITATION. ORNL-TM-2496, (1989).

21. Frost, B.R.T. (1982). *Nuclear Fuel Elements.* Pergamon Press, Oxford.

22. Garland, Wm. J.: Engineering Physics 4D3/6D3 Nuclear Reactor Systems Analysis (Reactor Physics). McMaster University, Hamilton, Ontario, Canada

23. Glaser, A. (2004). Beyond A.Q. Khan-The Gas Centrifuge, Nuclear Weapon Proliferation, and the NPT Regime. *INESAP Information Bulletin, No.23,* pages 50-54.

24. Glaser, A. (2005a). Neutronics Calculations Relevant to the Conversion of Research Reactors to Low-Enriched Fuel, Dissertation, Department of Physics, Darmstadt University of Technology.

25. Glaser, A. (2005b). "About the Enrichment Limit for Research Reactor Conversion: Why 20%?" *Proceedings of the 27th RERTR Meeting*, Boston, USA, November 6-10.

26. Glaser, A. (2006). "On the Proliferation Potential of Uranium Fuel for Research Reactors at Various Enrichment Levels", Science and Global Security, 14, p. 1–24.

27. Glasstone, S.: *Principle of reactor engineering*. Macmillan

28. Glasstone S. and Edlun M.C.: *The elements of nuclear reactor theory*. D Van Nostrand Company Inc.

29. Gilpin A.: *Dictionary of Energy Technology*, Butterworth Scientific Arbor Science.

30. Halsall M.J. (1980). *A summary of WIMSD4 input options*. AEEW-M137, UKAEA.

31. Halsall M.J. (1992). *LWRWIMS, A light water reactor lattice calculations*. AEEW-R1498.

32. Hofman, G.L.; Meyer, M.K. (2002). "Progress in Irradiation Performance of Experimental Uranium-Molybdenum Dispersion Fuels", In: *Proceedings of the 24th RERTR Meeting*, San Carlos de Bariloche, Argentina, 3-8 November.

33. Hofman, G.L.; Snelgrove, J.L.; Hayes, S.L.; Meyern, M.K; Clark, C.R.: "Progress in Post-Irradiation Examination and Analysis of Low-Enriched U-Mo Research Reactor Fuels", In: *Transactions of the 7th RRFM Meeting*, March 9–12, 2003, Aix-en-Provence, France, pp. 43–49.

34. Hofman, G.L. and Snelgrov, J.L.: *Dispersion Fuels*. Chapter 2 of volume 10 A (Nuclear Materials, part 1, edited by B.R.T. Frost) In R.W. Cahn, P. Haasen, and E.J. Kramer (eds): Materials Science and Technology. VCH Verlagsgesellschaft mbH, Weinheim, Germany, 1994.

35. Hofman, G.L.; Yeon, S.K.; Ho, J.R.; Finlay, M.R. (2006). "Results of Low Eriched U-Mo Dispersion Fuel mini-plates from Irradiation Tests, RERTR-6", In: *Proceedings of the 28th RERTR Meeting*, Cape Town, South Africa, October 29 - November 2.

36. Hunt S.E. (1980). *Fission, fusion and energy crisis,* 2nd ed. Pergamum.

37. Hunt S.E. (1987). *Nuclear Physics for Engineers and Scientists*. Ellis Horwood limited, Chichester.

38. Hutton J.L. (2000). *New capabilities of WIMS code* paper presented at international conference on advances in reactor physics and mathematics and computations into the next millennium. Pittsburgh, Physor.

39. IAEA. *Safeguards Glossary. 2001 Edition. International Nuclear Verification Series, No.3*. International Atomic Energy Agency, Vienna, 2002a.

40. IAEA. *International Nuclear Cycle Evaluation (INFCE), volume 1-9*. International Atomic Energy Agency, Vienna, 2002b.

41. IAEA. *Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels Guidebook.* IAEA-TECDOC-233, International Atomic Energy Agency, Vienna, 1980b.

42. IAEA. *Research Reactors in the World. Reference Data Series No.3, September 2000 Edition.* International Atomic Energy Agency, Vienna, 2002c.

43. IAEA. *THE Applications of research Reactors.* IAEA-TECDOC-1234, International Atomic Energy Agency, Vienna, 2002d.

44. Iqbal, M., Muhammad, A., Mahmood, T., Ahmed, N., (2007). On comparison of experimental and calculated neutron energy flux spectra at miniature neutron source reactor (MNSR). Ann. Nucl. Energy.

45. Jijin G. (1993). *General description of Nigeria Miniature Neutron Source Reactor, Training Material for Nigeria MNSR,* Book 1, China Institute of Atomic Energy.

46. Jiajuan S. (1993). *General description of Nigeria Miniature Neutron Source Reactor, Training Material for Nigeria MNSR,* Book 2, China Institute of Atomic Energy.

47. 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.

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

49. Khamis I.; Khattab K.: Lowering the enrichment of the Syrian miniature neutron source reactor. Annals of Nuclear Energy 26, 1999, P. 1031-1036.

50. Matos J. and Lell R.M.(2005). Feasibility study of Potential LEU Fuels for a Generic MNSR Reactor. Proceedings of the 2005 International Meeting on Reduced Enrichment.

51. Newton T.D. (2004). *The Development of modern design and reference core neutronics methods for PBMR,* Serco Assurance.

52. Newton T.D. and Hutton J.L. (2002). *The next generation WIMS code: WIMS9*, Serco Assurance.

53. ONNP. *Report on the Use of Low Enriched Uranium in Naval Nuclear Propulsion.*

Technical Information for the Director, Office of Naval Nuclear Propulsion, to the U.S. Congress, 1995.

54. Pershagen, B. and Carlvik, I. (1956). Calculation of the disadvantange factor for the moderator in a cylindrical lattice cell. A.B. Atomenergie, AEF-69.

55. Pistner, C.; Liebert, W.; Fujara, F. (2006). Neutronics calculations on the impact of burnable poisons to safety and non-proliferation aspects of inert matrix fuel, Journal of Nuclear Materials; 352, 268-275.

56. Ritchie, I.G. (1998). Growing Dimensions- Spent Fuel Management at Research Reactors. *IAEA Bulletin*, 40(1).

57. Safety Analysis Report (SAR) for the Syrian Miniature Neutron Source Reactor, China Institute of Atomic Energy, 1993, China.

58. Sampong, S., Maakuu, B., Akaho, Andam, A., Liaw, J., Matos, J., (2006). Progress in the neutronics core conversion (HEU-LEU) analysis of Ghana Research Reactor-I. In: Proceedings of the 28th International Meeting on RERTR Program, Cape Town, October 29–November 2.

59. Sangren, W.C. (1960). *Digital Computers and Nuclear Reactor Calculations.* John Wiley & Sons Inc., New York.

60. Schwartz, J.P. (1978). *Uranium Dioxide Caramel Fuel-An Alternative Fuel Cycle for Research and Test Reactors.* Communication presented at the International Conference on Nuclear Non-proliferation and safeguards. Atomic Industrial Forum, New York, October 22-25.

61. Taubman C.J. (1975). *The WIMS 69 group library tape166259*. SGHW Development Division AEEW-M1324, Dorchester.

62. Tayyab M,; Pervez S. and Iqbal M.: Neutronics analysis for core conversion (HEU–LEU) of Pakistan research reactor-2 (PARR-2) Annals of Nuclear Energy 35 (2008) 1440–1446

63. Travelli, A. (1978). The U.S. Reduced Enrichment Research and Test Reactor (RERTR) Program. In *Proceedings of the 1978 International Meeting on Reduced Enrichment for Research and Test Reactors (RERTR),* Pages 3-15, Argonne National Laboratory, Argonne, Illinois, November 9-10.

64. United Nations. (2004). *A More Secure World: Our Shared Responsibility.* Report of the Secretary General's High-Level Panel on Threats, Challenges, and Change: New York.

65. Von Hippel, F. (2004). A comprehensive Approach to Elimination of Highly-Enriched Uranium from All Nuclear-Reactor Fuel Cycles. *Science and Global Security.*

66. Waqar S, Mirza S.M., Mirza N.M., Asad T. (2008). A comparative neutronics study of the standard HEU core and various potential LEU alternatives for a typical MNSR system. Nuclear Engineering and Design 238, p. 2302–2307

Appendix A: WIMS Input File

```
CELL 6
SEQUENCE 1
NGROUP 4,4
NMESH 40,40
NREGION 5,0,5
NMATERIAL 5
PREOUT
INITIATE
FEWGROUP 5,15,45,69
ANNULUS 1,0.2150,1
ANNULUS 2,0.275, 2
*ANNULUS 3,0.62015,3
*ANNULUS 4,0.62120,4
*ANNULUS 5,0.6227,5
ANNULUS 3, 0.617344105,3
ANNULUS 4, 0.6216,4
ANNULUS 5, 0.6252,5
*Sylvanus File
MATERIAL 1,-1,293.0,1,235.2,2.211E-03,238.2,2.3804E-04,27,$
5.0735E-02
*MATERIAL 1,-1,293.0,1,235.2,2.0189E-03,238.2,2.3804E-04,27,$
*MATERIAL 1,-1,293.0,1,235.2,2.2189E-03,238.2,2.3804E-04,27,$
*5.7186E-02
MATERIAL 2,-1,293.0,2,27,6.0244E-02
MATERIAL 3,-1,293.0,3,2001,6.673971E-02,16,3.336985E-02
MATERIAL 4,-1,293.0,3,27,6.0244E-02
MATERIAL 5,-1,293.0,3,1056,0.087549
MESH 15,5,10,5,5
BEGINC
PARTITION 5,15,45,69
THERMAL 1
NOBUCLING
BEGINC
```

Appendix B: CITATION Input File

```
** CITATION ** INPUT DATA FOR NIRR-1 RESEARCH REACTOR
TWO DIMENSIONAL R-Z GEOMETRY
001
 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 1 0 4 0 0
 1 0 1 1 1 1 0 0 1 1 0 1 1 1 1 0 0 0 0 0 0 0 0 0
550                                   60   60
```

```
 1.5
003
  0  0  0  0  7  0  0  0  0  0  1  0  0  0  0  0
       .0001      0.0001                0.0001
 .4692        0.4692      0.3        1.0
004
  8  0.625   10  10.925   5  3.000    4  7.000   14  30.000   4  10.000

 12  19.000   3  1.000   13  20.500   3  0.200    3  0.400    3  0.900
 12  19.000   5  4.000    3  0.400    3  0.500    5  5.000   18  30.000
005
  2  2  2  2  2  2
 10  2  2  2  2  2
 10  2  2  2  2  2
 10  5  2  2  2  2
 10  2  2  2  2  2
 10  6  2  2  2  2
  2  3  4  4  2  2
  2  3  4  4  2  2
  7  6  4  4  2  2
  7  6  2  2  2  2
  2  4  4  2  2  2
  2  2  2  2  2  2
008
  4  3  1
    10     1 2.41207E+00 7.88399E-03 6.31941E-20
  8.9558E-02  6.3211E-02  2.1822E-04  0.0000E+00
    10     2 1.53515E+00 4.27834E-03 1.79990E-20
  0.0000E+00  2.2360E-01  4.6835E-02  4.4852E-06
    10     3 2.28844E+00 1.41589E-02 1.49671E-21
  0.0000E+00  0.0000E+00  2.8710E-01  3.7316E-02
    10     4 2.31641E+00 3.79598E-01 2.86356E-24
  0.0000E+00  0.0000E+00  5.7308E-03  4.4936E-01
     2     1 2.31474E+00 5.23423E-04 2.90232E-12
  3.7445E-02  1.0538E-01  6.5618E-04  0.0000E+00
     2     2 7.82476E-01 2.18910E-07 2.50896E-12
  0.0000E+00  2.7873E-01  1.4726E-01  1.4398E-05
     2     3 5.59551E-01 9.82805E-04 2.50711E-11
  0.0000E+00  0.0000E+00  4.4785E-01  1.4688E-01
     2     4 1.52079E-01 1.90763E-02 3.61806E-10
  0.0000E+00  0.0000E+00  7.0056E-05  2.1727E+00
     3     1 2.32558E+00 1.47593E-03 2.66566E-03
  5.2963E-02  8.9361E-02  5.1691E-04  0.0000E+00
     3     2 9.63609E-01 1.12225E-03 1.37500E-03
  0.0000E+00  2.6672E-01  1.1384E-01  1.1103E-05
     3     3 8.90534E-01 1.70465E-02 1.54382E-02
```

0.0000E+00 0.0000E+00 4.0750E-01 1.0729E-01
3 4 4.74696E-01 1.22994E-01 2.16378E-01
0.0000E+00 0.0000E+00 4.7748E-04 1.4966E+00
4 1 1.46108E+00-6.74210E-03 2.53996E-12
1.6395E-01 7.0933E-02 1.2198E-16 0.0000E+00
4 2 5.78656E-01 1.55444E-07 1.62893E-12
0.0000E+00 5.4857E-01 2.7478E-02 0.0000E+00
4 3 4.83681E-01 3.44555E-05 1.48119E-11
0.0000E+00 0.0000E+00 6.7205E-01 1.7071E-02
4 4 4.35424E-01 6.61386E-04 2.62700E-10
0.0000E+00 0.0000E+00 1.1841E-05 7.6486E-01
5 1 2.75003E+00 1.65147E-04 1.25887E-12
9.9553E-02 2.1492E-02 6.1437E-17 0.0000E+00
5 2 2.83584E+00 2.98368E-04 1.25616E-12
0.0000E+00 1.1603E-01 1.2154E-03 0.0000E+00
5 3 4.02073E+00 9.03633E-04 1.06141E-11
0.0000E+00 0.0000E+00 8.1715E-02 2.8526E-04
5 4 3.82302E+00 4.73577E-03 9.24719E-11
0.0000E+00 0.0000E+00 4.7915E-04 8.1976E-02
6 1 2.45647E+00 4.69608E-04 1.39361E-08
4.9364E-02 8.7307E-02 5.2105E-04 0.0000E+00
6 2 1.08610E+00 3.61085E-05 1.20967E-08
0.0000E+00 2.5165E-01 1.1644E-01 1.1366E-05
6 3 1.27043E+00 1.09886E-03 1.58643E-07
0.0000E+00 0.0000E+00 3.7653E-01 1.1698E-01
6 4 8.36959E-01 1.82978E-02 3.09132E-06
0.0000E+00 0.0000E+00 7.0252E-05 1.7409E+00
7 1 2.52241E+00 3.78575E-04 7.37550E-13
6.0937E-02 7.2772E-02 4.0621E-04 0.0000E+00
7 2 1.45448E+00 8.11154E-05 2.21958E-13
0.0000E+00 2.1782E-01 7.8261E-02 7.6137E-06
7 3 2.25705E+00 1.01120E-03 4.17746E-13
0.0000E+00 0.0000E+00 2.6807E-01 7.4881E-02
7 4 1.79423E+00 1.53062E-02 7.82986E-12
0.0000E+00 0.0000E+00 6.3911E-05 1.1280E+00
0
7.452795E-012.545717E-011.488919E-040.000000E+00
999

My own fuel for NIRR-1 follows

3 1 1.88301E+00 1.01817E-03 1.09486E-03
3.0825E-01 1.8712E-01 6.3132E-04 0.0000E+00
3 2 8.03203E-01 8.00045E-04 1.32638E-03
0.0000E+00 6.1492E-01 1.3281E-01 1.2224E-05
3 3 7.98725E-01 1.30986E-02 1.43315E-02

```
 0.0000E+00  0.0000E+00  6.9401E-01  1.1784E-01
    3     4 4.40473E-01 1.22843E-01 1.87736E-01
 0.0000E+00  0.0000E+00  6.5989E-04  1.7909E+00
```

Appendix C: A program to read and save fluxes in each energy group (developed by Azande et al., 2009)

```
FUNCTION SUBGRPGIS0()
 DIM GRPCOUNTER
 GRPCOUNTER = 0

'NOW READ AND SAVE FLUXES IN EACH ENERGY GROUP
'*****************************************
DIM SPECIALSUBGRPJJ AS BOOLEAN
DIM SPECIALSUBGRPII AS BOOLEAN

DIM NUMOFSUBGRPSJJ
DIM NUMOFSUBGRPSII
DIM NUMINLASTSUBGRPJJ
DIM NUMINLASTSUBGRPII
DIM WHOLESUBGRPSJJ
DIM WHOLESUBGRPSII
NUMOFSUBGRPSJJ = HORNUMFLUXPTS \ 11
NUMOFSUBGRPSII = VERTNUMFLUXPTS \ 50
WHOLESUBGRPSJJ = NUMOFSUBGRPSJJ ' I.E. IF NUM OF HOR FLUX POINTS > 11
WHOLESUBGRPSII = NUMOFSUBGRPSII ' I.E. IF NUM OF VERT FLUX POINTS > 50

  IF HORNUMFLUXPTS MOD 11 THEN
    SPECIALSUBGRPJJ = TRUE
    NUMINLASTSUBGRPJJ = HORNUMFLUXPTS MOD 11
    NUMOFSUBGRPSJJ = NUMOFSUBGRPSJJ + 1
  END IF

  IF VERTNUMFLUXPTS MOD 50 THEN
    SPECIALSUBGRPII = TRUE
    NUMINLASTSUBGRPII = VERTNUMFLUXPTS MOD 50
    NUMOFSUBGRPSII = NUMOFSUBGRPSII + 1
  END IF

DIM NUMINSUBGRPII
  NUMINSUBGRPII = 50
 DIM INTH, INTI, INTJ, INTL, INTM, INTN
 DIM INTJJ, INTII
```

```
 OPEN "SUMFLUX.DAT" FOR OUTPUT AS #10
 DIM II, JJ, NUMOFGRPS
  JJ = HORNUMFLUXPTS + 1: II = VERTNUMFLUXPTS + 1: NUMOFGRPS = 4
 REDIM PRESERVE GRPFLUXES(NUMOFGRPS, II, JJ)
 REDIM PRESERVE SUMOFFLUXES(NUMOFGRPS, II, JJ)

 ' ***************  " GROUP  1 FLUX" *********************
 '*******************12345678901234************************
 DIM GRPNUM AS INTEGER ' ENERGY GROUP NUMBER
 DIM MYFILENUM
 MYFILENUM = 2
DIM NUMPERLINE: NUMPERLINE = 11
IF WHOLESUBGRPSJJ = 0 THEN
   DO WHILE GRPCOUNTER <= NUMOFGRPS AND NOT EOF(1) 'NOT
FOUNDFLUXVALU
     LINE INPUT #1, NEWLINE
     IF MID(NEWLINE, 1, 6) = " GROUP" AND MID(NEWLINE, 11, 4) = "FLUX" THEN

       GRPNUM = CINT(MID(NEWLINE, 7, 3))
       IF GRPNUM <> 0 THEN CLOSE MYFILENUM
       MYFILENUM = "#" + CSTR(GRPNUM + 1)
       GRPCOUNTER = GRPCOUNTER + 1
       LINE INPUT #1, NEWLINE   'SKIP 1 LINES
       LINE INPUT #1, NEWLINE  'READ ROW FLUX INDEX

       IF WHOLESUBGRPSII = 0 THEN
         DIM INTK, NEWLINE2
         FOR INTK = 1 TO VERTNUMFLUXPTS
           LINE INPUT #1, NEWLINE
           NEWLINE2 = RTRIM(NEWLINE)
           FOR INTJ = 1 TO HORNUMFLUXPTS
              GRPFLUXES(GRPCOUNTER, INTK, INTJ) = CDBL(MID(NEWLINE2, INTJ * 11
- 4, 9))
           NEXT INTJ
         NEXT INTK
       END IF

       IF WHOLESUBGRPSII <> 0 AND SPECIALSUBGRPII = FALSE THEN
         FOR INTK = 1 TO VERTNUMFLUXPTS
           LINE INPUT #1, NEWLINE
           NEWLINE2 = RTRIM(NEWLINE)
           DIM JK
           FOR JK = 1 TO HORNUMFLUXPTS
              GRPFLUXES(GRPCOUNTER, INTK, INTJ) = CDBL(MID(NEWLINE2, INTJ * 11
- 4, 9))
           NEXT JK
```

```
          IF INTK MOD 50 = 0 THEN
            LINE INPUT #1, NEWLINE: LINE INPUT #1, NEWLINE
          END IF
        NEXT INTK
      END IF

      IF WHOLESUBGRPSII <> 0 AND SPECIALSUBGRPII = TRUE THEN
        FOR INTK = 1 TO VERTNUMFLUXPTS
          LINE INPUT #1, NEWLINE
          NEWLINE2 = RTRIM(NEWLINE)
          FOR JK = 1 TO HORNUMFLUXPTS
            GRPFLUXES(GRPCOUNTER, INTK, INTJ) = CDBL(MID(NEWLINE2, INTJ * 11 - 4, 9))
        NEXT JK
        IF INTK MOD 50 = 0 THEN
          LINE INPUT #1, NEWLINE: LINE INPUT #1, NEWLINE
        END IF
      NEXT INTK
    END IF

    END IF
  LOOP
 END IF
END FUNCTION
```

AppendixD: A program to calculate number density (developed by Azande et al., 2009)

```
If getCorrectU235Mass Then
  If keff < setK_eff1 Then
    frmCoreConversion.U235Mass.Text = frmCoreConversion.U235Mass.Text * 1 + _
       frmCoreConversion.DeltaU235Mass.Text * 1
  ElseIf keff > setK_eff2 Then
    frmCoreConversion.U235Mass.Text = frmCoreConversion.U235Mass.Text * 1 - _
       frmCoreConversion.DeltaU235Mass.Text * 1
  End If
  tstMkUpData(8) = frmCoreConversion.U235Mass
  frmCoreConversion.U235NumDensity.Text = 1E-24 * _
    frmCoreConversion.U235Mass.Text / _
    frmCoreConversion.numOfFuelRods.Text _
    * frmCoreConversion.Avogadro.Text / _
    (235# * 3.1415926536 * frmCoreConversion.FuelrodRadius.Text ^ 2 * _
       frmCoreConversion.fuelrodlength.Text)
    CitRunCounter = CitRunCounter + 0
End If
```