ANALYSIS OF NEUTRONICS SAFETY PARAMETERS AND CORE BURNUP LIFETIME OF BAEC TRIGA MARK-II RESEARCH REACTOR USING THE DETERMINISTIC TRIGAP AND TRIGLAV CODES

MD RAKIBUL HASAN

M.Sc. THESIS

DEPARTMENT OF NUCLEAR SCIENCE AND ENGINEERING

MILITARY INSTITUTE OF SCIENCE AND TECHBOLOGY

DHAKA, BANGLADESH

MARCH 2023

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A Thesis Submitted in Partial Fulfillment of the Requirements
for the Degree of Master of Science in Nuclear Science and Engineering

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M. Sc. Engineering Thesis

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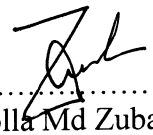

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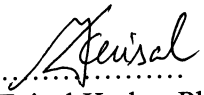

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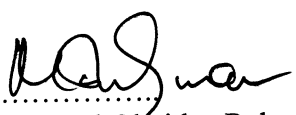

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DECLARATION

I hereby certify that the research reported in the thesis with the aforementioned title is entirely original work of mine and has never been submitted elsewhere for academic or other purposes. Additionally, I attest that the intellectual content of this thesis is entirely my own creation, and that I dutifully acknowledged and cited all sources used in its preparation in the reference section.

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ABSTRACT

Analysis of Neutronics Safety Parameters and Core Burnup Lifetime of BAEC TRIGA Mark-II Research Reactor Using the Deterministic TRIGAP and TRIGLAV Codes

BTRR has been operating science 1986 without any form of reloading or shuffling. On September 14, 1986, 50 low enriched uranium (LEU) fuel components were put into the original core, and it reached its first initial criticality. After that, 100 fuel elements were loaded and this configuration is known as the operational core. The goals of this study are (i) to analyze the neutronics core safety parameters of Initial as well as the functioning core of BTRR and (ii) to calculate individual fuel element burn-up as well as ring wise average burn-up at different burn-up conditions and core lifetime of the present low enrichment uranium (LEU) core configuration. To analyze the initial criticality experiment of BTRR, its initial critical TRIGA physical model has been developed and hence the initial effective multiplication factor, core excess reactivity has been calculated using deterministic code TRIGLAV and TRIGAP accordingly. Burnup calculations are predicated on the concept that while calculating the neutron density distribution, nuclide concentrations can be taken for granted to be constant. They are built on the neutron transportation calculation and the burnup equations, which are two fundamental equations in reactor physics. Individual fuel burnup as well as ring wise burnup calculation has been done by TRIGLAV code and it has been compared with the MVP-Burn code. Burnup has been calculated up to 1400 MWd and the 1400 MWd data has been compared with the result obtained from TRIGAP code.

To estimate the core life time, core excess reactivity has been considered and the calculated results are likened with the experimental obtained values from reactor operational data log book. The reactor may be operated safely for an additional 500 MWd days in accordance with the need of burnup and excess reactivity. This study will be helpful to formulate the most economic use of the fuel rod initially overloaded in the core. Additionally, the study provides insightful information on the behavior of the reactor and will guarantee improved reactor usage and operation in the future. Additionally, utilizing the same fuel components, this can provide insight into redesigning a better core configuration.

সারসংক্ষেপ

Analysis of Neutronics Safety Parameters and Core Burnup Lifetime of BAEC TRIGA Mark-II Research Reactor Using the Deterministic TRIGAP and TRIGLAV Codes

বাংলাদেশ পরমাণু শক্তি কমিশনের আওতাধীন ৩ মেগাওয়াট গবেষণা রিঅ্যাক্টরটি কোন প্রকার কোর রিলোডিং ছাড়াই ১৯৮৬ সাল হতে পরিচালিত হয়ে আসছে। ১৯৮৬ এর ১৪ সেপ্টেম্বর ৫০ টি LEU জ্বালানী উপাদান মূল কোরে স্থাপন করা হয়েছিল যা প্রাথমিক ক্রিটিকাল কোর হিসেবে পরিচিত। এর পরে, ১০০ টি জ্বালানী উপাদান লোড করা হয়েছিল এবং এই কনফিগারেশনটি অপারেশনাল কোর হিসাবে পরিচিত। এই গবেষণার লক্ষ্যগুলি হল (i) প্রাথমিক কোরের নিউট্রনিক্স কোর সেফটি প্যারামিটারের পাশাপাশি ৩ মেগাওয়াট TRIGA Mark-II রিসার্চ রিঅ্যাক্টরের অপারেশনাল কোর বিশ্লেষণ করা এবং (ii) পৃথক জ্বালানী উপাদান বার্ন-আপের পাশাপাশি রিং অনুসারে গণনা করা ও বিভিন্ন বার্ন-আপ পরিস্থিতিতে গড় বার্ন আপ নির্ণয় করা।

BAEC TRIGA চুল্লির প্রাথমিক ক্রিটিকালিটি পরীক্ষা বিশ্লেষণ করার জন্য, effective multiplication factor, core excess reactivity ইত্যাদি TRIGLAV এবং TRIGAP কোড ব্যবহার করে গণনা করা হয়েছে। বার্নআপ গণনাগুলি এই ধারণার উপর অনুমান করা হয় যে নিউট্রন ঘনত্বের বন্টন গণনা করার সময় নিউক্লাইড ঘনত্বকে ধ্রুবক হিসাবে গ্রহণ করা যেতে পারে। এগুলি নিউট্রন পরিবহন গণনা এবং বার্নআপ সমীকরণের উপর নির্মিত, যা রিঅ্যাক্টরট পদার্থবিদ্যার দুটি মৌলিক সমীকরণ। পৃথক জ্বালানী বার্নআপের পাশাপাশি রিং অনুসারে বার্নআপ গণনা TRIGLAV কোড দ্বারা করা হয়েছে এবং এটি MVP-বার্ন কোডের সাথে তুলনা করা হয়েছে। বার্নআপ 1400 MWd পর্যন্ত গণনা করা হয়েছে এবং 1400 MWd এর ডেটা TRIGAP কোড থেকে প্রাপ্ত ফলাফলের সাথে তুলনা করা হয়েছে।

মূল কোর জীবনকাল অনুমান করার জন্য, মূল কোরের এক্সেস রিএক্টিভিটি পরিমাপ করা হয়েছে এবং প্রাপ্ত ফলাফলগুলি রিএক্টরের অপারেশনাল ডেটা লগ বুক থেকে প্রাপ্ত পরীক্ষামূলক মানের সাথে তুলনা করা হয়েছে। গণনাকৃত মোট বার্নআপ, উষ্ণতম জ্বালানী সনাক্তকরণ, এবং এক্সেস রিএক্টিভিটি ডেটা অনুসারে বার্নআপ এবং এক্সেস রিএক্টিভিটি প্রয়োজন অনুসারে চুল্লিটি অতিরিক্ত 500 MWd এর জন্য নিরাপদে পরিচালিত হতে পারে।

এই অধ্যয়নটি প্রাথমিকভাবে কোরে লোড করা জ্বালানী রডের যথাযত অর্থনৈতিক ব্যবহার প্রণয়ন করতে সহায়ক হবে। উপরন্তু, একই জ্বালানী উপাদান ব্যবহার করে, কোর রিলোডিং বা কোর রিশাফল করে কোরের লাইফ টাইম বাড়ানোর জন্যও এই প্রবন্ধটি গুরুত্বপূর্ণ ভুমিকা রাখবে।

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LIST OF MAIN NOTATIONS

BAEC	Bangladesh Atomic Energy Commission
BTRR	BAEC TRIGA Research Reactor
TRIGA	Training Research, Isotopes, General Atomics
LEU	Low Enriched Uranium (20% weight & 19.7% enrichment)
HEU	High enriched Uranium
AERE	Atomic Energy Research Establishment
INST	Institute of Nuclear Science and Technology
IAEA	International Atomic Energy Agency
ST8	Standard fuel component with 8.5 weight percent U (20% enrichment),
STl2	Standard fuel component with 12 weight percent U (20% enrichment),
FLIP	Fuel component with 8.5 weight percent U (70% enrichment)

CHAPTER 1
INTRODUCTION

1.1 Background

The composition of nuclear fuel elements is continually changing in a running nuclear reactor. The initial nucleus divides into lighter nuclides during nuclear fission, producing secondary particles and energy. Nuclides can also change into other nuclides by spontaneous radioactive decay and other neutron-induced transmutation events. Even after the nuclear fuel has been removed from the reactor, the radioactive decay process still exists.

Predicting changes in the nuclear fuel's composition is crucial in many applications. For instance, changes in nuclide concentrations and how these changes are accounted for have a important impact on the safety and efficiency of a reactor core loading. This applies to both the loan of novel reactor designs and the optimization of reactor core loading in already-operational reactors. Additionally, it's critical to evaluate the used fuel's material degradation both immediately and later on. Predicting the nuclide concentrations at time increments of the order of thousands of years is necessary for final deposition applications. In reality, specific burnup calculation codes are used to assess the changes in nuclear fuel material characteristics.

Burnup calculations are predicated on the notion that while calculating the neutron density distribution, nuclide concentrations can be taken for granted to be constant. They are built on the neutron transportation equation and the burnup equations, which are two fundamental equations in reactor physics. In essence, the neutron transit equation balances the neutron density. In burnup calculations, the criticality equation a time-independent eigenvalue problem is modeled. In this case, the solution consists of the neutron density distribution and the multiplication factor, which describes the system's time dependency.

BTRR was activated on September 14, 1986 (salam et al., 2014) at AERE. The reactor was constructed with the goal of advancing a number of fields of fundamental nuclear research, workforce development, and radioisotope manufacturing for application in business, agriculture, and healthcare. The reactor is a 3000-kW graphite-reflected, light water-cooled unit designed for continuous

operation (thermal). The TRIGA reactor's demonstrated safety is a result of the U-ZrH fuel-moderator material's substantial quick negative temperature coefficient of reactivity. The objective of this study is to build the optimum fuel management strategy for the most effective use of the fuel elements initially loaded in the TRIGA core as well as to develop the applied technological know-how for burnup analysis. The two major objectives of in-core management are to (i) run the rated power in a safe and efficient way and (ii) achieve high fuel discharge burnup as envisioned in the design. This calls for ways for safely designing and implementing fuel reloading and/or reshuffling. The benchmark neutronic tests have only been done for the new core, using a variety of computer tools. (Bhuiyan et al., 2000; Huda et al., 2004). Knowing the current individual fuel burnup is crucial for the expensive core rearranging necessary to achieve optimal fuel utilization. Therefore, it is anticipated that this research will considerably advance the safe and cost-effective utilization of the TRIGA reactor. Four fundamental forms of information were used to conduct the in-core management study: (i) forecast for criticality (keff); (ii) predictions for power peaks; (iii) forecasts for neutron flux and power distributions; and (iv) projections for fuel element burnup. These have to do with how the core functions as a function of burnup, which has to do with how much energy each fuel element in the core generates. For core management, calculations of fuel burnup or evaporation in developing schemes were used.

In order to reach high fuel burnup and generally flat power distribution, a variety of fuel rod distribution techniques have been taken into consideration.

For this study, the TRIGLAV and TRIGAP computer code was utilized, which has already been used effectively for the analysis of the TRIGA research reactor (Ravnik et al., 1999). In this study, the following analyses were carried out: (i) Individual fuel burnup; (ii) ring-wise determination of individual fuel element burnup; (iii) calculating the lifespan of the current core configuration and (iv) formulating excess reactivity. This work used a two-dimensional TRIGLAV code mainly along with one-dimensional TRIGAP code to compute the core burnup lifetime of BTRR.

The estimated findings were contrasted with the three-dimensional MVP-BURN code and the Safety Evaluation Report's reference data (NUREG-1282). The calculated core burnup lifetime was found to have good agreement with the MVP-BURN result and reference data, indicating that the TRIGLAV and TRIGAP code simulates the TRIGA model properly and is appropriate for analysis of Core Burnup

life time experience

estimation of the LEU fresh core of BTRR. This study will also be useful in formulating the most cost-effective way to utilize the fuel rod that was first loaded in the core without rearranging and reloading. Additionally, this research will be helpful to improve the reactor core operation and usage within the safety margin.

1.2 Motivation of This Study

Since September 14, 1986, BTRR has been in operation. About 815 MWd of the reactor have been used thus far for isotope production, other studies, and training. It should be emphasized that the BTRR has been in operation since 1986 with no recent reloading or shifting of the fuel element in the core. To increase neutron flow and lengthen the lifespan of the reactor core, BAEC intends to rearrange or reload the nuclear fuel rods inside the core. It should be noted that no deterministic study of this nature has yet been carried out. For correct in-core nuclear fuel management, it is necessary to analyze neutronics core safety characteristics like effective multiplication factor (keff), control rod worth, excess reactivity of initial critical core, as well as operating core, of any research reactor as well as power reactor. In order to operate the reactor properly and satisfy regulatory criteria, it is also necessary to complete this study in order to receive a license from the nuclear regulatory authority. For computations involving research/power reactor neutronics, numerous complex and cultured computer codes have been created or modified for trivial and private computers.

In order to accomplish the aforementioned neutronics parameters using a well-validated computer code, a study has been conducted. Additionally, in order to accomplish the rearranging of the fuel rod within the core, the burn-up of the TRIGA fuel must be calculated. TRIGLAV is selected as the analysis's primary code from a variety of deterministic computer programs, and TRIGAP is used for data comparison.

1.3 Significance of the Study

This study will provide a fuel management strategy of BTRR during reshuffling or reloading in future. Identification of hottest as well as least hot fuel of each ring has been completed in this study. Reshuffling of hottest fuel with the least hot one may

increase the BTRR core lifetime which will be helpful for extended operation of BTRR

1.4 Research Questions

Goal of this study is to calculate the initial criticality of BTRR as well as the excess reactivity to validate the deterministic computer code TRIGLAV. After the validation this computer code has been used for the operational core study. The same process has been performed by another computer code TRIGAP. This is very helpful for data validation. After that core burnup calculation has been done accordingly.

1.5 Objectives of the Research

The main objectives of the research are:

- To analyze the neutronics core safety parameters of operational as well as initial core of 3 MW TRIGA Mark-II Research Reactor
- To calculate individual fuel element, burn-up as well as ring wise average burn-up at different burn-up conditions and core lifetime of the present low enrichment uranium (LEU) core configuration.

1.6 Structure of This Thesis

a) Chapter 1: This chapter includes a list of the study's objectives as well as some basic information on neutronics safety parameters and an examination of the burnup lifetime of the TRIGA-Mark 2 research reactor calculating methods.
b) Chapter 2: This chapter includes various related journals on the subject that have been evaluated and summarized.
c) Chapter 3: Calculation & Techniques are described in this chapter**.** This research has explained analysis of neutronics safety parameters which was done in RPED, INST, AERE, Saver Dhaka.
d) Chapter 4**:** A few outcomes were presented in a table and through observation.
e) Chapter 5**:** This section included a brief explanation of the findings and a comparison of the experimental findings to local, regional, and global standards.
f) Chapter 6**:** Conclusion remarks of the thesis

CHAPTER 2
LITERATURE REVIEW

2.1 Brief Description of TRIGA Mark-II Research Reactor

BTRR is the only operating nuclear research reactor in Bangladesh that achieves its educational and scientific objectives. The Bangladesh Atomic Energy Commission has been in charge of operating the TRIGA Mark-II research reactor since 1986. It has a maximum heat flux in the middle of the core of 8.1x1013 n/cm2/sec at full power of 3 MW. (BAEC). General Atomics, based in the city of San Diego, California, was this reactor's main supplier and contractor. It is a light water cooled, graphite reflecting reactor with a peak output of about 852 MW and continuously running at a steady state full power level. It is designed for routine pulsing with activity insertions of up to 1.4% dk/k ($2.00). The TRIGA reactor's demonstrated safety is a result of the U-ZrH fuel moderator material's significant rapid negative temperature coefficient of reactivity. Radiation levels are sufficiently low during pulsed or steady state operation that individuals can safely examine the core and experimental equipment.

The reactor's core can be naturally cooled for brief periods of time while being operated at power levels of up to 500 kW. The induced forced flow mode of operation ensure proper transmission of the reactor heat to the cooling tower for longer-term operation at low power or greater power. The water purification and cooling systems maintain low water conductivity, eliminate contaminants, preserve optical purity, and offer a way to dissipate reactor heat. Table 2.1 displays the reactor's primary design parameters.

Table 2.1: Principal Design Parameters of BTRR

Sl. No.	Principal Design Parameters	
1.	Reactor Type	TRIGA Mark-II
2.	Maximum Steady State power level	3 MW
3.	Maximum pulse	1.4% δk/k, $ 2.00, 852 MW
4.	Fuel moderator material	U-ZrH*

5.	Uranium Content	20 wt%
6.	Uranium Enrichment	19.7% ^{235} U
7.	Burnable Poison	0.47 wt% ^{166}Er and^{167}Er
8.	Shape Cylindrical	Cylindrical
9.	Overall Length of Fuel	38.1 cm (15 inch)
10.	Outside Diameter of the fuel	3.63 cm (1.43 inch)
11.	Cladding Material	Type 304 stainless steel
12.	Cladding thickness	0.051 cm
13.	Number of Fuel Element	100
14.	Total reactivity worth of control rods	10% δk/k
15.	Number of control rods	6

2.2 The Reactor Installation

2.2.1 Built in Safety

Due to its inherent safety, installation is possible in a structure with typical construction since a containment shell is not required. According to Fig. 2.1, the reactor is completely installed above ground.

2.2.2 Shielding

The reactor and the testing facilities are encircled by the concrete shield building in Fig. 2.2. The reactor core and reflector assembly are housed at the bottom of an aluminum tank that has a 2 m diameter and an 8.2 m depth. Above the core, there is a vertical water shield that is about 6.4 meters high. The core is shielded radially by at least 2.29 meters of concrete with a density of 2.75 grams per cubic centimeter, 45.7 centimeters of water, 5 centimeters of lead, and 19 centimeters of graphite reflector. (from outer to inner).

Fig. 2.1: BAEC 3 MW TRIGA Mark-II Research Reactor.

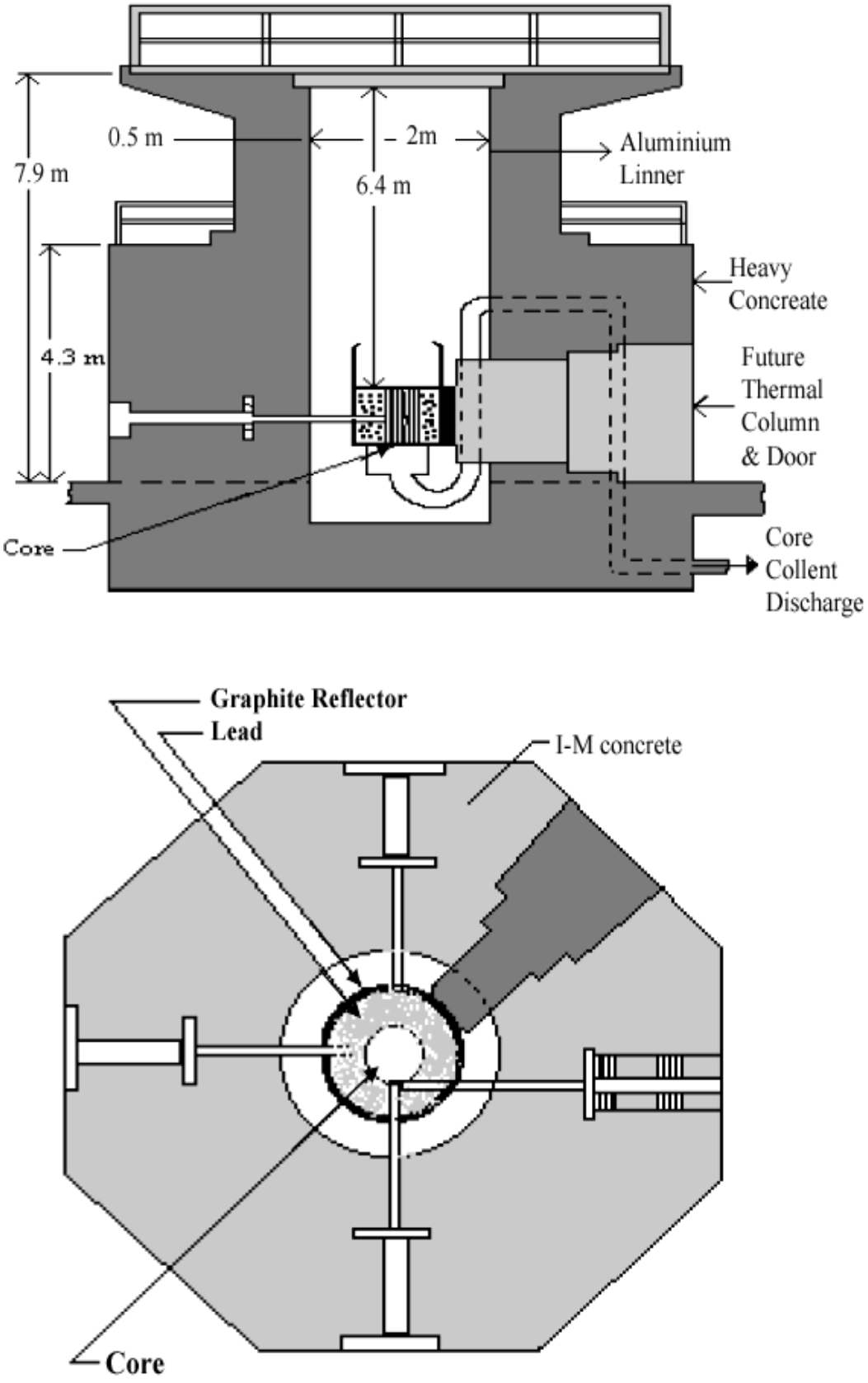


Fig. 2.2: Cutaway view of 3 MW TRIGA Mark-II reactor.

2.2.3 Fuel-Moderator Elements

Solid fuel components for TRIGA reactors are made by General Atomic (GA), and they evenly blend enriched uranium and ZrH moderator. A unique characteristic of these fuel-moderator elements is the prompt negative temperature coefficient of reactivity, which gives the reactor its inherent safety by automatically limiting the reactor power excursion. A lattice of cylindrical fuel moderator and graphite dummy pieces make up the reactor core. Two lengths of graphite are placed in the fuel element container, one above and one below the fuel, to serve as the top and bottom reflectors for the core. The core is encircled by a 19 cm thick graphite radial reflector, which is mounted on an aluminum platform at the tank's base. Except for the vicinity of the beam tubes and the thermal column structure, the graphite reflector is encircled by 5 cm of lead. About one-third of the core volume is taken up by water. The vertical water shield allows for constant physical access to and monitoring of the core. From the reflector assembly's outer face into the concrete shield structure, another graphite thermal column extends. The thermal column will be protected and given horizontal access through a hefty, track-mounted concrete door. The thermal column and door aperture will be blocked with concrete blocks for the current installation. Four beam ports run from the reactor assemblage to the outside face of the shield assembly through concrete and water.

2.2.4 Rotary Specimen Rack

A revolving specimen rack, housed in a well at the top of the graphite reflector, enables large-scale radioisotope production, activation analysis, and the irradiation of tiny specimens. The rotating specimen rack assembly consists of a ring-shaped, seal-welded aluminum shell and an aluminum rack positioned on certain bearings. The rack is supported by 41 uniformly spaced tubular aluminum containers that house the specimen container receptacles. With the exception of the receptacle in position one, which can only hold one specimen container, each container has an inner diameter of 30.5 mm and a altitude of 27.4 cm. This rack's positions are all exposed to fluxes of equivalent strength of neutrons.

2.2.5 Power Level

Six different types of control rods are used to regulate the TRIGA reactor's power output.:

i. One fuel follower regulating rod,
ii. Four fuel follower shim/safety rods, and
iii. One air follower transient rod.

The reactor described here is designed for continuous operation at a steady-state output level of 3000 kW and routine pulsing with reactivity insertions of up to 1.4% k/k ($2.00). (thermal). The maximum pulse reactivity insertions of 2.1% k/k ($ 3.00) are performed in a core without water holes in its central regions.

2.2.6 Prompt Negative Temperature Co-efficient of Reactivity

In the case of human error or mechanical failure, the TRIGA fuel possesses inherent qualities that will stop a nuclear accident. The rapid negative temperature coefficient inherent in the fuel and core design is the fundamental parameter that gives the TRIGA system a significant safety factor throughout steady state operation and under transient conditions. Due to the considerably reduced impact of unintentional reactivity changes on temperature and power, this self-actuating temperature coefficient offers a considerable deal of operational flexibility. These characteristics efficiently regulates massive, rapid, positive reactivity insertions. Thus, any abrupt rise in reactivity results in a spike in power, which instantly warms the fuel moderator material, causing the number of fissions to fall down due to changes in the neutron energy spectrum inside the fuel rod. In order to minimize the power increase, the reactor temperature automatically and immediately adjusts for the reactivity addition. Such control is inherent in the fuel of the TRIGA reactor and is independent of mechanical or electrical control mechanisms. This most important attribute is produced by the solid homogeneous alloy of uranium fuel and ZrH moderator that makes up the fuel components, also known as fuel moderator elements. The TRIGA LEU fuel's reactivity is decreased by the temperature hardened spectrum's interaction with a low energy resonance component. Thus, erbium is used in the TRIGA LEU fuel as a poison that can be burned and a chemical to boost the prompt negative temperature coefficient. Erbium has a double resonance at about 0.5 eV. Because the ^{235}U density in the fuel rod is around 2.5 times higher and because erbium is used, the ratio of the absorption probability to the neutron leakage probability is higher for

TRIGA LEU fuel than for ordinary TRIGA fuel. When the fuel moderator substance is heated, the neutron spectrum becomes more difficult and there is a greater chance that the neutrons will be caught by low energy resonances. Due to the increased intrinsic absorption with temperature, the reactivity diminishes as the fuel temperature rises. According to Fig. 2-3, which depicts the cold and hot neutron spectra as well as the energy-dependent absorption cross section for 167Er, when the fuel temperature increases, the neutron spectrum changes, pushing more thermal neutrons into the 167Er resonance (Antonopoulos et all.). Because a significant amount of the solid moderator and the fuel are thoroughly mixed, the temperature coefficient is quick. As a result, the temperatures of the solid moderator and fuel rise simultaneously, causing the temperature-dependent spectrum shift.

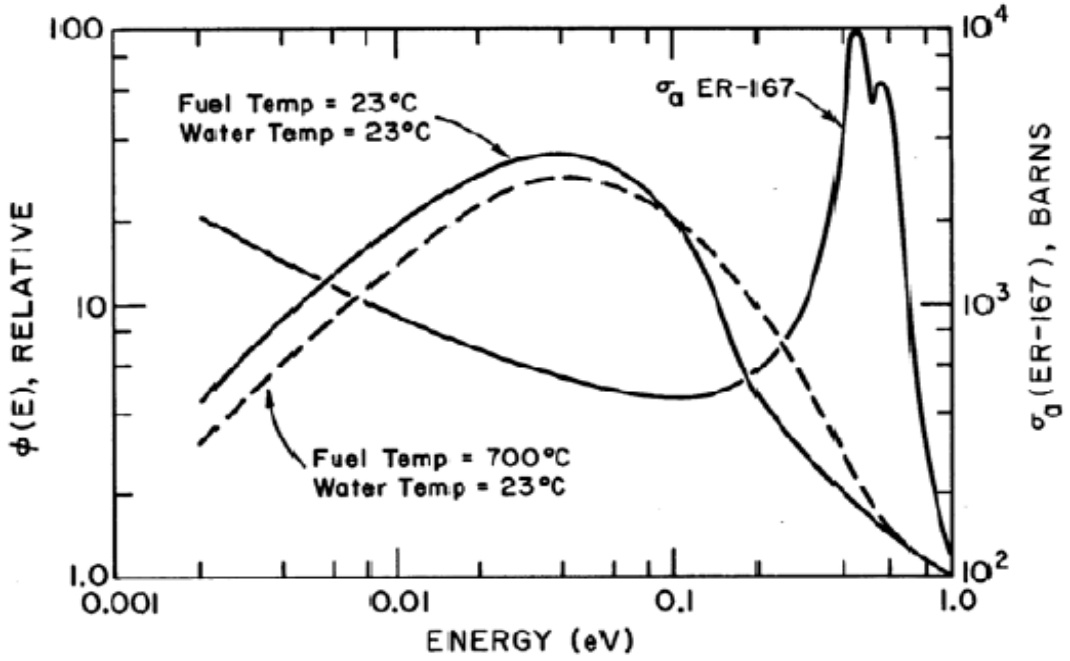


Fig. 2.3: Thermal Neutron Spectra vs. Fuel Temperature Relative to σ_a vs. Energy

The 167Er resonance receives an increasing number of thermal neutrons, which causes the temperature coefficient for the TRIGA LEU core to increase as a function of fuel temperature. The temperature coefficient of this erbium-containing TRIGA core depends on temperature, which is advantageous because it results in the least amount of reactivity loss when it reaches normal working temperatures. The quick negative temperature coefficient, which serves as a shutdown mechanism, rises significantly with any appreciable rise in the average core temperature. The current core of the TRIGA reactor will be the subject of our proposed study.

2.3 The Present Core Configuration

One hundred fuel elements are arranged in a concentric hexagonal pattern inside the TRIGA's core shroud. On October 9, 1986, during reactor start-up at full power operation, the reactor's current core configuration, as shown in cross-section, was reached. 2.4. Water serves as a moderator and coolant in the spaces between the rods when the components are stacked in seven concentric rings.

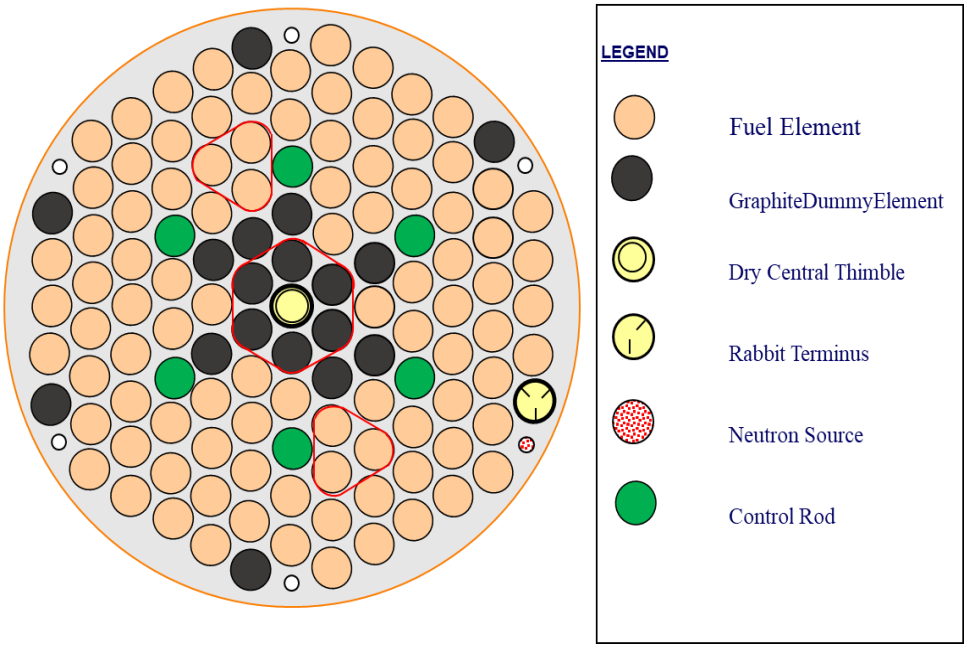


Fig. 2.4: Core configuration of BTRR

Six hexagonal bands with a total of 121 holes, each measuring 3.82 cm in diameter, are positioned around a center hole on the top grid plate. These holes house the control rods, graphite dummy elements, fuel moderators, and pneumatic tube. The center thimble can fit in the center hole, one of the 121 periodic groupings of holes. The other 95 single cells, which occupy the remaining 18 holes occupied by the graphite dummy elements, six control rods, and one pneumatic tube, are filled with the fuel-moderator material. The grid plate also has six source sites. The reactor core and reflector assembly are housed at the bottom of an aluminum tank that has a 2 m diameter and an 8.2 m depth. The water at the core, which is around 6.4 meters deep, serves as a vertical shield. The cutaway perspective of the TRIGA reactor core is shown in Fig. 2.2. The core is encircled by a 19 cm thick graphite radial reflector, as previously mentioned.

2.3.1 Characteristics of the Rings in the Core

There are nine rings total in the core arrangement: A, B, C, D, E, F, G, H, and I. They were each regarded as a distinct zone. The characteristics of these rings, which are concentric hexagonal rings of a hexagonal lattice, are listed in Table 2.2.

Table 2.2: Characteristics of different rings of BTRR

Rings	Characteristics of the rings	No. of rods
A	Central thimble	01
B	Graphite dummy elements	06
C	Fuel rods	05
	Graphite dummy elements	07
D	Fuel rods	12
	Control rods	06
E	Fuel rods	24
F	Fuel rods	30
G	Fuel rods	24
	Graphite dummy elements	05
	Source locations	06
	Pneumatic transfer tube	01
H	Water reflector	------
I	Graphite reflector	------

2.3.2 Fuel Moderator Elements

According to Fig. 2.5, the dynamic portion of each fuel modulator component measures about 3.63 cm in diameter and 38.1 cm in length (Salam et all.). The fuel is a solid, homogeneous combination of U-Er-ZrH alloy that contains about 0.47% by weight of erbium and about 20% by weight of U-235 that has been enhanced. The ratio of H to Zr atoms is roughly 1.6. To make hydriding easier, a tiny hole is drilled through the middle of the active fuel part; once hydriding is finished, a zirconium rod is put inside of this hole. Heliarc welding is used to create all closures on the 0.051 cm thick stainless-steel cans that surround each element. To act as the top and bottom reflectors for the core, two chunks of graphite are placed in the can, one above and one below the fuel. The can's two ends are joined with stainless steel

and fittings to give the fuel-moderator element a total length of 73.15 cm. In the event of human error, the fundamental features of U-ZrH TRIGA fuel will stop a nuclear accident. These characteristics result in a big, prompt negative temperature co-efficient of reactivity in the TRIGA reactor, which efficiently suppresses large, prompt positive reactivity insertions. Any abrupt reactivity addition results in an increase in power, which instantly warms the fuel moderator material, changing the fuel rod's energy spectrum by increasing the number of fissions. In order to minimize the power increase, the reactor temperature automatically and immediately adjusts for the reactivity addition.

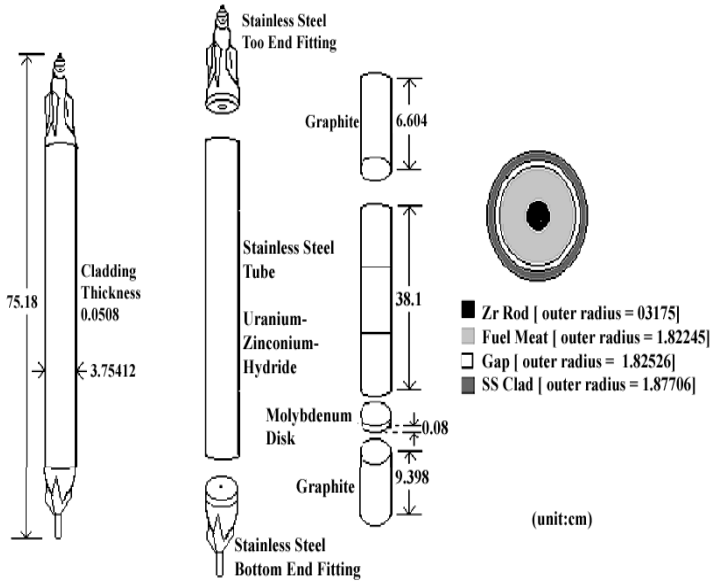


Fig. 2.5: BTRR Stainless Steel-Clad Fuel Element with Tri-flute and Fittings

2.3.3 Control Rods

The power level of the TRIGA reactor is controlled by using six control rods, including a regulating rod, a transient rod, and four shim/safety rods. These six rods consist of five fuel followers (LEU fuel) and one air follower (transient rod). The uppermost 16.51 cm section of the type 304 stainless steel tubes that serve as the regulating and shim/safety rods is an air void, while the next 38.1 cm of the tubes serve as the neutron absorber (boron carbide in solid form). A 38.1 cm length of U-ZrH fuel in the fuel-follower is located just beneath the neutron absorber. The rod's bottom portion is 16.5 cm air void. The top and bottom grid plates have holes with a diameter of 3.8 cm through which these rods travel. Fig. 2.6 depicts the precise

locations of the control rods (both in the withdrew and inserted states) (Salam et all.). The safety/transient rod is a solid aluminum tube measuring 3.18 cm in diameter and 93.35 cm long that is sealed and contains solid boron carbide as a neutron absorber. There is a follower part with air inside it below the absorber. The follower part is 53.02 cm long and the absorber section is 38.1 cm long. A perforated aluminum guide tube with the transient rod inside of it travels through the core.

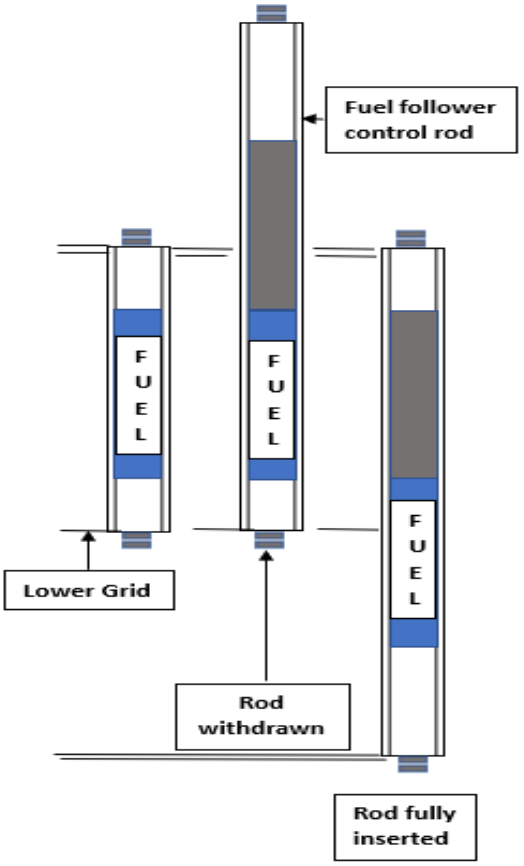


Fig. 2.6: Fuel Follower Control Rod with Withdrawn and Inserted Position

2.3.4 The Central Thimble

Samples can be exposed to radiation at the point of greatest flux in the reactor core using the central thimble, an aluminum tube with an internal diameter of 3.38 cm. The centre thimble was first filled with water, it may be noted here. The local scientists modified the design so that the top half of the thimble is dry and the bottom half is filled with water. This modification enables the direct exposure of dry samples to radiation on the central thimble, where the neutron flow is greater.

2.3.5 Graphite Dummy Elements

Graphite dummy elements must occupy all grid spaces left empty by the fuel-moderator elements to prevent coolant from bypassing them. With the exception of the top and bottom fittings, which snugly fit into the grid plates, they are the same size as the fuel moderator elements. Graphite makes up the whole interior of them.

2.3.6 Reflector Outside Core

A block of graphite in the form of a ring that radially encircles the core is the reflector. The block has a height of 52.7 cm, a diameter of 54.9 cm inside (below the rotational specimen rack), and a thickness of 19 cm. To lessen gamma heating in shield concrete, the graphite is encircled by a 5 cm thick lead layer. The faces of the beam apertures are not covered by the lead. A housing made of welded aluminum that is leak-tight houses the graphite and lead.

2.4 Experimental and Irradiation Facilities

There are numerous experimental facilities at the TRIGA reactor. Intense neutron and gamma fluxes can be produced using it for radioisotope manufacture, training, and research. The following is a list of the reactor's experimental and isotope production facilities:

- A circular well in the reflector assembly of the reactor houses a rotating specimen rack (Lazy Susan) that can hold 81 samples at once for activation analysis and isotope synthesis.
- The G ring of the core contains a pneumatically driven "rabbit" transfer system that enters the reactor core lattice and is used to create radioisotopes with extremely brief half-lives.
- A central cap that reaches the core lattice's center enables the extraction of a highly collimating of radiation or the introduction of small samples into the region of maximum flux.
- The water near the reflector where the samples are located may also expose them to radiation. For experimental purposes, the reactor features three radial and one tangential beam port.
- In-core irradiation equipment, such as hexagonal and triangular thermal column cuts.

2.5 Necessary Calculations

The neutrons in a reactor have a variety of energy and are not mono-energetic. Every type of energy has a unique flux and cross section. Again, even at a given energy, the position in the reactor affects the neutron flow at a particular moment. Furthermore, the initial geographical distribution of the fissile material could not be uniform, and it won't be after the reactor has been running for a while. The product of the macroscopic cross section and the neutron flux regulates the thermal power of a reactor with given active neutrons in a given energy range. Reactor calculations are necessary in this circumstance, and they can be accomplished by using the appropriate computer algorithms.

2.6 Reconsiderations of Previous works

An enormous amount of experimental data was gathered for safety parameters and core burnup estimates as soon as neutronics analysis became a focus. Numerous techniques, including the TRIGLAV and TRIGAP algorithms and thorough experiments on full-scale models, were used to collect this data. Conversely, theoretical inquiries came about at random. The phenomenological method attracted the most interest among all the theoretical investigations. Below is a summary of a few noteworthy examples of earlier works.

M.Q. Huda *et.al*. This study's main goal is to offer the TRIGA MARK II research reactor at AERE, Savar, with an ideal fuel management plan that is both efficient and effective. Criticality, power peaking, neutron flux, and burnup calculation were the four fundamental types of information that were calculated for the reactor and used in the core management research. The results of the burnup calculations for the TRIGA LEU fuel components are presented in this paper. Using the TRIGAP compute tool, the fuel component burnup for around twenty years of operation was computed. Radial geometry in one dimension is used in TRIGAP to conduct the computation. There is great agreement between the results of the TRIGAP calculations and those of the MVP-BURN and MCNP4C-ORIGEN2.1 computations. The longest core lifespan of the reactor is achieved by reshuffles at 20,000 MWh steps, which is 64,500 years. The study will enable improved reactor use and operation in the future and provide important knowledge about the behavior of the reactor.

M. Ranvik *et.al*. The paper shows the essential safety characteristics for research reactors are discussed, including temperature reactivity coefficient, power density peaking factors, and shutdown margin. The parameter's reactor physics explanations are provided, along with information on how to use them in safety assessments carried out as part of research reactor operation. Reactor computation is put up as a technique for determining them, presuming the utilization of freely accessible computer programs.

A. Persic *et.al*. discusses the TRIGLAV package, a software for the computational analysis of TRIGA Mark II reactors based on the four-group diffusion theory in 2D cylindrical dimensions and the lattice transport algorithm WIMSD for homogenizing unit cell cross sections. Four different types of fuel elements, as well as a variety of non-fuel elements including water, graphite, control rods, etc., are present in potential unit cells. Secondary parameters, such as fuel power and temperature distribution, xenon defect, etc., are examined in addition to the fundamental parameters, i.e., the multiplication factor and neutron flux distribution. There has been comparison between the software and the Monte Carlo code MCNP5. Several potential TRIGLAV application programs are also described in the article.

C. El Younoussi *et.al*. suggesting that the CENM TRIGA MARK II reactor is a part of the National Center for Energy, Sciences, and Nuclear Techniques (CNESTEN). It has a thermal column, a graphite reflector, four beam tubes, and is a standard 2 MW natural convection-cooled reactor. Numerous applications for the reactor can be found in a range of sectors, including industry, farming, medicine, training, and education. The current work has performed a computer analysis in the framework of the reactor's neutronic parameters investigation. A comprehensive MCNP model has been built that comprises all core components and supporting structures in order to calculate different core parameters. (The effective multiplication factor, reactivity experiments comprising control rods worth, excess reactivity and shutdown margin). Additional computations have been made to the neutron flux profiles at different locations in the reactor core. The cross sections are taken from the MCNP5 library, based on the ENDF/B-VII with continuous energy dependence, and treated specifically for thermal neutrons in light materials.

M.A Khattak *et.al* examining the 1982-operated 1MW Thermal TRIGA MARK II research reactor in Malaysia. The solid fuel components, such as uranium zirconium hydride, were loaded, and on June 28, 1982, it reached its initial criticality. That same year saw the start-up of the 1MW research reactor TRIGA MARK II. TRIGA's initiative aims is successfully implement several fields associated with radioisotope production, human resource development, and fundamental nuclear research. In this work, the TRIGLAV reactor physics computer program is used to evaluate the initial criticality of the TRIGA research reactor. This is why a simulation of the program's original core will be made, and the outcomes will be contrasted with the experimental results as detailed in the safety study's final report. (FSAR). To address the neutron diffusion issue, the TRIGLAV computer program use the finite differences method and fission density iteration. Four group time independent diffusion equations in a two-dimensional cylindrical shape make up the foundation of TRIGLAV. The fuel burn-up, reactor criticality, power and flux densities of the core, among other things, may all be determined using TRIGLAV.

T. Zagar *et.al* analyzes the significance of several scheming models for the precision of fuel element burnup. Calculations were performed using TRIGLAV, a program created in-house. Two separate unit-cell transport calculation routines (WIMS-D/4 and WIMS-D/5), two cross section libraries, two homogenization methods (FVH and EDH), and two reactor reflector unit-cell models were among the changes in the calculation parameters. Calculations of the neutron flux distribution, the burnup of the core and fuel elements, and the reactivity value of the fuel 0987l elements are provided and compared to experiments.

M.Ranvik *et.al* console several TRIGA reactors run on mixed fuels, including cunei and minor amounts of fuel with uranium contents or with drastically variable levels of enrichment. The utilization of irradiation facilities in or near the core is directly impacted by variances in the composition of the fuel material in both scenarios, which result in significant spatial and spectral variations of neutron flux in the reactor. They effect fuel use indirectly by having an impact on neutron economy. It is offered general guidelines for the best fuel utilization in a mixed core. They were created using extensive burn-up calculations for various core configurations and core management strategies, which were then validated by experimental data from the 250 kW TRIGA-Mark II reactor at the "J. Stefan" Institute in Ljubljana,

Yugoslavia. The mixed core uses two fuel element types: standard, 20% enriched, FLIP type 70% enriched. However, the results in their general form can also be applied to other TRIGA reactors and other types of mixed cores with comparable geometry and core composition.

M.A. Salam *et.al* Neutronic safety parameter measurement is essential while conducting a study to ensure the reactor's safety, efficient operation, and experimental investigation. Neutronic safety parameters of BRR were measured in this study, including control rod worth, core excess reactivity, loss of reactivity with power increases, power defect, reactivity coefficients, cooling effect on fuel temperature, and xenon poisoning. (BAEC). Each of these safety factors has a substantial impact on the reactor control system. Many of the reactor parameters that control the reactivity of the reactor are influenced by the temperature of the fuel, moderator, and coolant. The TRIGA fuel components offer a sizable rapid negative temperature coefficient of reactivity that enables safe reactor operation. With the help of the BTRR's digital instrumentation and control (I&C) system, the neutronic safety parameters were measured. The experimentally established neutronic properties were discovered to be within the safety limit, according to the Safety Analysis Report (SAR) of the BTRR. A library of experimental data will be made available for evaluating and improving reactor model simulations as a result of the current work, which will also help investigate how the reactor core's neutronic safety characteristics behave.

2.7 Summary of Review of Previous Research Works

From the reviewing the above literatures, a vast amount of knowledge has been accumulated regarding neutronic safety parameters and core burnup calculations. Information has been collected about deterministic code TRIGLAV and TRIGAP. In addition, with that reference data has been collected from this review work to compare with the calculated results. Finally, analysis has been to fulfill the objectives of the study.

CHAPTER 3
CALCULATION AND TECHNIQUE

3.1 Code Description

The neutronic safety parameters and core burnup lifetime of the BTRR LEU core were computed using the industry-standard two-dimensional diffusion computer programs TRIGLAV & TRIGAP with the required supporting data sets. The computer program TRIGAP was used to perform the burnup calculations.

For reactor computations of mixed cores in the TRIGA Mark II research reactor, the TRIGLAV software package is created. It can be used for criticality forecasts, power and flux distribution calculations, and burnup computations for fuel elements. The program is based on a two-dimensional, cylindric (r,), four group, time independent diffusion equation. The finite differences method is used to solve the diffusion equation while iterating the fission density. It is assumed that material constants are step functions of the local variables r and. The up-scattering of neutrons to higher energy groups is not disregarded. The program's geometry has been modified to fit the TRIGA Mark II reactor's cylindrical core. Every position of a fuel or non-fuel element in the core is regarded as a unit cell. The integrated transport program WIMS (Bowen et al., 1975) is used to calculate group constants for all unit cells. For each unit cell, the group constants are determined based on the geometry of the fuel or non-fuel elements, the material composition, the actual fuel element burnup, the temperature, the density and temperature of the water, the temperature of the cladding, and the xenon concentration. Two input files are required by TRIGLAV Both TRIGLAV.OUT and ELEM.OUT are written to the input files TRIGLAV.INP and ELEM.INP. It is optional to write flux output data to a unique file called TRIGA2D.FLU.

On the other hand, In particular for TRIGA type reactors, TRIGAP was created for research reactor computations. The software works under the assumption that the reactor's geometry is cylindrical, and problems are given with pertinent databases, primarily those containing information on the fuel's operation history and nuclear constants. The calculation's predicted accuracy is 0.5% for keff, 15% for power distribution and peaking factors, and 10% for fuel burnup, which produces numbers that are almost identical to those produced by typical power reactor programs.

(Mele and Ravnik, 1985; Ravnik et al., 1999). The foundation of TRIGAP is the two-group diffusion equation. (group boundary at 1 eVIt is resolved using the fission density iteration approach in the finite difference approximation. For the Bangladesh TRIGA MARK II research reactor, a database for the TRIGAP code was created. (Bhuiyan et al., 1992). The WIMS-D/4 code was used to develop the library (WIMS-D/4, 1983).

TRIGLAV Code

3.2.1 Brief Description of Neutron Diffusion Approximation in TRIGLAV

The 4-group time independent homogeneous diffusion equation in two-dimensional cylindrical (r,) geometry serves as the foundation for the TRIGLAV software package. The following formula is the diffusion equation for the energy group g:

$$-\nabla D^g \nabla \Phi^g + \sum_r^g \Phi^g = \left(\frac{1}{k}\right) \chi^g F + \sum_{g'=1, g' \neq g}^{4} \Sigma^{g' \to g} \Phi^{g'} \; ; \;\; g = 1, \dots \dots, 4, \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots \dots .. (1)$$

Were,

Φ^g = neutron flux

D^g = diffusion constant

$\sum_r^g$ = removal cross section; ($\sum_r^g = \sum_a^g + \sum_{g'=1,g' \neq g}^{4} \Sigma^{g' \to g} + D^g B_z^2$)

B_z^2 = axial geometrical buckling, user-defined on TRIGLAV input

$\Sigma^{g' \to g}$ = scattering cross section from group g'into group g

χ^g = part of fission spectrum in group g;

(Default TRIGLAV: $\chi_1 = 1, \chi_2 = \chi_3 = \chi_4 = 0$)

k = multiplication factor

F = fission density, which is defined as: $F = \sum_{g=1}^{4} \vartheta^g \sum_f^g \Phi^g$ (2)

Where $\vartheta^g$is number of fission neutrons in energy group g and $\sum_f^g$ = fission cross-section for group g. Neutron flux $\Phi^g(r, \vartheta)$, fission density $F(r, \vartheta)$, diffusion constant $D^g(r, \vartheta)$ and all cross sections $\sum_r^g(r, \vartheta)$, $\Sigma^{g' \to g}(r, \vartheta)$ and $\sum_f^g(r, \vartheta)$ are functions of local variables r and ϑ. The $\Phi^g(r_{boundary} r, \vartheta)$=0 boundary condition is imposed for all polar angles ϑ at the outer boundary of the reactor reflector. The finite difference method is used to solve the diffusion problem. Fission density

iterations are used to solve the finite difference equations. The inner iterations method is used to invert each group equation.

The TRIGA Mark II reactor core shape is modified to the two-dimensional difference mesh. The before demonstrated 7 fuel rings and graphite or water reflector make up the BTRR. Only the reflector ring is homogeneous; the other fuel rings are made up of unit cells. There is only one unit cell in the central "ring" of A. This design has 102 angular intervals that line up with unit cell borders. There must be 102 angular gaps for the mesh nodes to be homogenous. The radial dimension must include at least 8 intervals, while the minimum varies depending on the number of fuel rings.

Table 3.1: TRIGLAV code general physical model properties.

Geometry	Two dimensional cylindrical
Number of energy groups in diffusion calculation	4
Method of solution	Finite difference method fission density
Unit cell homogenization	Lattice cell program WIMS-D/4 in unit cell approximation
Number of energy groups in transport calculation	32

3.2.2 Unit cell homogenization

The homogenization of the material constants for each unit cell in the reactor core are made using the transport program WIMS. The core may consist of fuel and non-fuel unit cells. A fuel unit cell contains fuel element and corresponding volume of surrounding water, which is equal for all unit cells in the core. The WIMS model clearly treats the fuel rod and surrounding water in an unlimited array of identical unit cells (white boundary condition is imposed on the outer boundary of the unit cell). Currently, the program can treat four different fuel element types:

ST8 standard fuel component with 8.5 weight percent U,

STl2 standard fuel component with 12 weight percent U,

FLIP FLIP fuel component with 8.5 weight percent U, and LEU LEU fuel element with 20 weight percent U.

Average non-fuel unit cell containing non-fuel element (graphite element,

beryllium element, irradiation channel, control rods or just water) and surrounding water, cannot be treated in the unit cell approximation. Instead, a super-cell method is applied. A number of fuel rods surround the non-fuel cell in this model which has been illustrated in Fig.3.1. The following six non-fuel components are a part of the program:

ICl irradiation channel, void channel, IC2 irradiation channel, half void, half water, IC3 irradiation channel, filled with water and LW water only. GR graphite element (C in AI tube). BE beryllium element (Be in Al tube) (without tube). The program estimates unit cell values for the water and graphite reflector constants. W water reflector and G graphite reflector are noted.

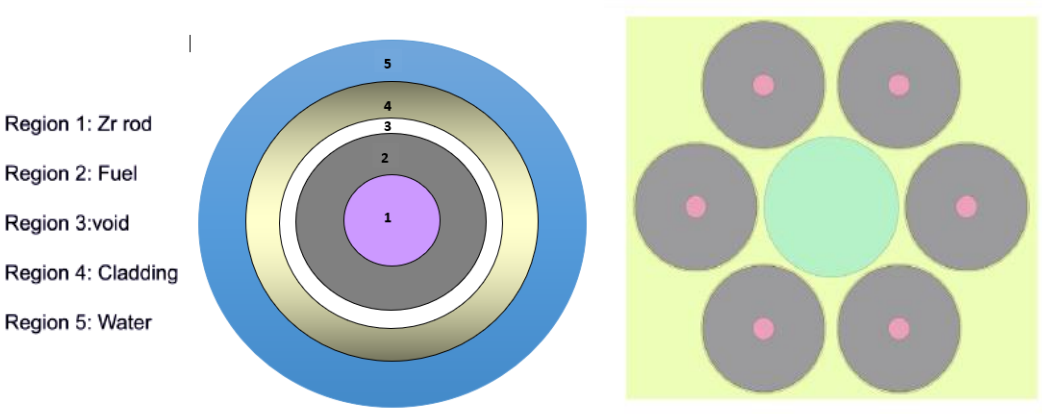


Fig. 3.1: Fuel and non-fuel model in WIMS-D/4

The flux and transmission lines are used to determine the burnup increment for each component in a burnup step. Automatically calculated burnup increases are written to file ELEM.OUT along with the burn up that was read from file ELEM.INP. Specific power per element (in kW) times burnup time (in days) equals burnup increment. To compute the specific power per element, power distribution and input reactor power are used.

3.2.3 Subroutines

The TRIGLAV software package is run using the batch process TRIGLAV (file TRIGLAV. BAT). It runs every program subroutine and makes changes to temporary files. The directory where the batch program TRIGLAV and the input

files TRIGLAV.INP and ELEM.INP are placed must be the same as the directory where all other executable subroutines can be found. After the programming software has executed, all statistical analysis is stored to the directory containing the input files. The batch procedure also adds entries to the TRIGLAV.LOG log file. The flowchart for the TRIGLAV software package is shown in Figure 3.1.

3.2.3.1 Subroutine TRSTART

The subroutine TRSTART reads input file TRIGLAV.INP and prepares temporary input file TRIGA2D. INP for cross section and diffusion calculations.

3.2.3.2 Subroutine WITRIG

In subroutine WITRIG the input files for the program WIMS are prepared for each fuel and non-fuel unit cell in reactor core.

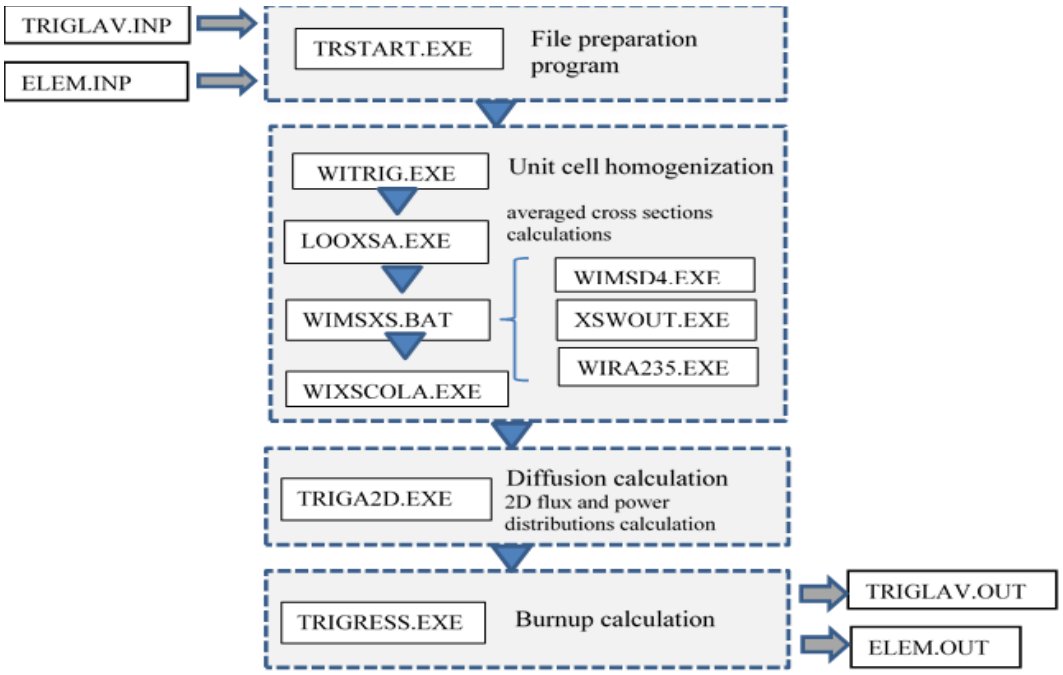


Fig. 3.2: Calculation scheme of the TRIGLAV code system

3.2.3.3 Subroutine LOOXA

The subroutine LOOXA prepares the WIMSXS.BAT batch procedure.

3.2.3.4 Subroutine WIMSXS

Subroutine WIMSXS runs the WIMSD4 program for all unit cells with input files

named WIMlnnn. WIN. When WIMS outputs are prepared, the subroutine XSWOUT reads each WIMS output, performs the group condensation from 32 energy groups into 4 groups and writes four group material constants to WIMlnnn.XSC files.

3.2.3.5 Subroutine WIXSCOLA

The subroutine WIXSCOLA combines all WIMlnnn. XSC cross section files into one file and prepares the input cross section file with name TRIGA2D.XS which is cross section input file for subroutine TRIGA2D.

3.2.3.6 Subroutine TRIGA2D

TRIGA2D is an impartial two-dimensional multigroup diffusion code. It is used as a subroutine in TRIGLAV program. All input data come through TRIGA2D. INP file (geometry of reactor core, mesh points, number of groups, convergence criteria) and TRIGA2D.XS file (calculated cross sections, see subroutine WITRIG). The result of calculations of subroutine TRIGA2D are power and flux distributions (for all four groups) and criticality calculation (multiplication factor). Power distribution is written to file TRIGA2D.P, all other results are temporarily written to file TRIGA2D. OUT and are later rewritten to file TRIGLAV. OUT. TRIGA2D can optionally write all four flux distributions to special file TRI GA2D.FLU.

3.2.3.7 Subroutine TRIGRES

The burnup option in WIMS is used to calculate the unit cell cross sections at the specified burnup. Fuel element burnup (BU1 in [MWd]) is indicated on the input. To accommodate the recommended burnup value BU1, it is divided into n intervals, each 1MWd in size, and a reminder of the proper size (interval n + 1).

$BU1 = nb+\gamma,$

$\gamma = BU1 \bmod b$

b= 1MWd

After that Subroutine TRIGRES calculates the burnup increment of each element if required. Reactor power P and burnup time step Δt is taken from TRIGA2D. INP temporary file. Element power (P_{el}) values are calculated from fission density distribution F (r, ϑ) stored in TRIGA2D. P file. Element power is normalized as follows.

$P_{el}=\beta p\ c/v \int(F(r,v)dV)$……(3)

Burnup increments are then calculated as follows

$$\Delta BU\_(1,el)=P_{el}\ \Delta t \text{……. (4)}$$

Burnup values in percent ($\Delta BU\_(2,el)$) are determined for each fuel element according to relations presented in equation 5 which are calculated using WIMS code and built into the code.

$$Y=0.00+1.31x-2.63e\text{-}3x^2 + 5.20e\text{-}6x^3 \text{……... (5)}$$

Burnup increments (BU (1, el) for all fuel components in the reactor core, in [MWd] and [%]) are the results of TRIGRES. They are automatically included in the element burnup in ELEM.INP so that the updated element burnup in ELEM.OUT is available. All of the elements' burnup and power are recorded in the output file TRIGRES.OUT, and they are later rewritten in the final output file TRIGLAV.OUT.

3.3 Data files

3.3.1 File TRIGLAV.INP

The TRIGLAV.INP file is used to input all autonomous input data defining reactor operation (power, loading pattern, etc.). The first set of input data consists of information, geometric specifications, and parameters for the diffusion equation solution. The information regarding the reactor core's core loading pattern is written in the second part. While all of the data can be provided in any format, as well as core loading pattern must be provided in the sequence and in specific format. All data in TRIGLAV. INP file are read in lines after the characteristic keyword.

All keywords begin with $*.

$* TRIGLAV This must be first card in input file, next two lines are reserved for any

kind of comments and will appear also on output file.

$* FLAGS After this card follow print control flags for cross sections, results of inner iterations and group flux distribution printout in output file TRIGLAV.OUT or in special file TRIGA2D. FLU. 1 is entered for printout of data, 0 is entered if no printout is wanted.

- 1st flag controls cross sections printout in TRIGLAV.OUT.
- 2nd flag controls inner iterations data printout on file TRIGLAV.OUT.

• 3rd flag controls group flux distribution printout on file TRIGLAV.OUT.

• 4th flag controls group flux circulation printout on file TRIGA2D.FLU.

$* ITERATIONS Allowed number of inner and outer iterations for diffusion calculation is read after this card.

• number in 1st line stands for allowed number of inner iterations.

• number in 2nd line stands for allowed number of outer iterations.

$* CONVERGENCE Convergence criteria for diffusion calculation.

• number in 1st line stands for convergence criterion for flux distribution inner iterations (parameter (1), recommended value 0.0000001.

• 2nd number stands for convergence criterion for fission density outer iterations (parameter (2), recommended value 0.00l.

• 3rd number stands for convergence criterion of multiplication factor outer iterations (parameter (3), recommended value 0.00001.

$* BUCKLING Squared axial buckling equal for all four groups in [cm^{-2}].

$* POWER Total thermal power of the reactor in [kW].

$* TWATER Temperature of water [OK] (parameter Twater).

$* XENON Flag for xenon condition, if 1 is entered Xe will be in equilibrium, if 0 is entered there will be no Xe.

$* BURNUP Time step in [days] for burnup increment calculation (parameter ~t).

$* RINGS Number of fuel rings in reactor core is read after this card. This number specifies geometry of TRIGA Mark II reactor core used in calculation.

• For reactors with 7 fuel rings and with graphite or water reflector (fuel rings A to G and reflector) number 7 must be entered, Then the following core geometry is assumed:

- reactor core height (fissionable material) =38.1cm

- ring radii: A=2.3cm, B=6.1cm, C=l0.lcm, D=14.1cm, E=18.1cm, F=22.1cm, G=26.1cm

- reactor core radius (fissionable material) =26.1 cm

- reflector outer radius (boundary condition <I> = 0) =58.5cm.

$* MESH This parameter determines type of finite differences mesh used in calculation. There are six mesh densities by default.

1 - very coarse mesh (8/9x102), 8 or 9 (depending on the number of rings) intervals in radial direction and 102 intervals in angular direction. This is minimum number of intervals possible. (There is one radial interval per fuel ring and two in reflector.)

Not recommended.

2 - coarse mesh (54/60x102), 54 or 60 intervals in radial direction and 102 intervals in angular direction.

3 - normal mesh (90/100x102), 90 or 100 intervals in radial direction and 102 intervals in angular direction. Recommended for burnup problems.

11 - coarse approximately equidistant mesh (27/29x156), 27 or 29 intervals in radial direction and 156 intervals in angular direction. Each interval in radial direction is ~2cm wide and each interval in angular direction is ~2.3° (~0.04rad) wide.

12 - normal approximately equidistant mesh (81/87x156), 81 or 87 intervals in radial direction and 156 intervals in angular direction. Each interval in radial direction is ~0.67cm wide and each interval in angular direction is ~2.3° (~0.04rad) wide.

13 - fine approximately equidistant mesh (90/100x156), 90 or 100 intervals in radial direction and 156 intervals in angular direction. Each interval in radial direction is ~0.34cm wide in core and ~1.3cm wide in reflector. Intervals in angular direction are ~2.3° (~0.04rad) wide.

$* LOADING Data in this part must follow the predefined format (3X,6(A4, lX, A4, lX)). This is known as the core loading pattern.

The parts that make up the core loading pattern may be either fuel- or non-fuel-related. In the event that a fuel element occupies a location, two fuel element identifiers are written for each slot. The element's location (ring-l-position number, for example, B-06) is the first identifier in the pair, and the fuel element identification number is the second identifier (id). If a non-fuel element occupies a position, the location and the element's characteristic name are written in pairs. Following distinctive names have been used to describe several categories of non-fuel elements: GR graphite element (C in Al tube), BE beryllium element (Be in an aluminum tube), IC1 Void irradiation channel, IC2 Half void irradiation channel, IC3 Wet irradiation channel, and LW water only (without tube). In the reactor model, reflector is always the last ring. Type of reflector is the final piece of information in the core loading pattern. There are two different kinds of reflectors: W stands for water, while G stands for graphite.

3.3.2 File ELEM.INP

Data of the fuel elements used in calculations can be found in the file ELEM. INP. This file must contain every fuel element specified by the LOADING command (there can be also elements that are currently not used in reactor). A message is written to file WITRIG.PRI and the program is terminated if an element specified by the LOADING command cannot be located in the file ELEM. INP. The first two lines are for comments and will show up in the output file as well. Data on the following lines should be entered using the following format (A4, lX, A4, lX, 7FlO.2) (nine columns):

Table 3.2 File ELEM.INP

Column	Parameter	Description
1	id	fuel element dentification number
2	type	element type
3	m(U)	mass of U in [g]
4	e	enrichment in [%]
5	m(^{166}Er)	mass of ^{166}Er in [g]
6	m(^{167}Er)	mass of ^{167}Er in [g]
7	BU1	element burnup in [MWd]
8	BU2	element burnup in [%]
9		last burnup increments in [MWd]

Identification codes may be chosen at random (e.g., four numbers). It is advised to utilize the fuel element identification numbers under which they are listed in the documentation. The most recent burnup increment is also arbitrary because it is output data that the TRIGRES function writes to this file.

3.3.3 File TRIGLAV.OUT

All output data generated during the calculation are contained in the output file TRIGLAV.OUT. Two components of the output are separated. The first section provides the input parameters, and the second section is the calculation's output. The printing of all data is self-explanatory.

-Printout of input parameters:

- parameters for general input:

-file names,

- remarks and

-Print the control flags.

-Physical characteristics include the number of groups utilized in the computation,

- the bounds of the groups' energies (lst=fast group, 4th=thermal group),

-spectrum of fission,

-axial buckling measured in [cm^{-2}],

- the reactor's heat output in kW, - the water's temperature in OK,

- burnup interval in [days] and xenon condition.

- Information on the core geometry includes the number of fuel rings, their outside radii, the outside radii of water- or graphite-based reflectors, the number of unit cells in a ring, and the type of reflector.

Numerical parameters, the number of allowed iterations, the need for convergence, the type of finite difference mesh utilized, and finally, the number of intervals in radial and angular direction employed in the computation.

- The core loading pattern.

- Cross sections for all unit cells in core are written (if print flag for cross sections is 1) as follows: element number, position, type of element, group number, diffusion constant, absorption cross section, fission yield and scattering matrix.

• After control printout all output data are printed in the following format:

- Solutions of diffusion equation (outer, inner iterations and multiplication factor monitoring).

Results of outer iterations are printed in lines beginning with OUT:

column description

1 OUT: outer iteration,

2 outer iteration number,

3 fission density normalization factors,

4 maximum fission density difference from previous iteration,

5 maximum relative fission density difference of previous iteration,

6 location of maximum fission density difference in radial direction

7 maximum fission density difference in angular direction.

If print flag for inner iterations equals 1 then inner iterations result are printed in lines beginning with I: (for each group in one line) column description ;

1 I: inner iteration,
2 number of inner iterations for group g,
3 group number g,
4 location of group flux maximum (radial)(j),
5 location of group flux maximum (angular)(i),
6 locations of maximum difference of group flux (radial)(j),
7 locations of maximum difference of group flux (angular)(i),
8 maximum group flux,
9 maximum difference of group flux and
10 maximum relative difference of group flux.

Pointwise fission density values are printed (if print flag for flux is 1). In the first block the mesh radii are printed. In the next blocks the pointwise power density values (in cm^{-3}) at the angular directions indicated on the left and at the radius indicated in the first block is printed.

The pointwise group flux distribution is written, if print flag for flux is 1, in the same format as fission density values. In the first block the mesh radii are printed. In the next blocks the pointwise group flux values (in[cm-2s-1]) at the angular directions indicated on the left and at the radius indicated in the first block are printed. This is repeated for each group.

- Time needed for calculation (CPU elapsed time).
- Power per element in [kW] is printed together with element number, location, characteristic name, type and power per element in [kW].
- Fuel elements of particular type (fuel and non-fuel) together with them burnup increments in [MWd].
- Fuel elements, arranged according to their identification numbers. For each fuel element power in [kW], initial burnup, burnup increment and final burnup in [MWd] are printed.
- Total core burnup increments in [MWd].
- Average fuel element burnup in [MWd] and average fuel element burnup increment in [MWd].

3.3.4 File ELEM.OUT

The ELEM.OUT file has updated burnup statistics for the fuel elements used in the calculation. It and the ELEM. IMP file are identical save from the updated burnup

information for elements used in the most recent calculation. For each fuel component, it includes the same kind of information. Comments from ELEM. IMP are in the file's first two lines. These are the lines that come next:

Column Description

1 Serial number of the fuel elements

2 Type of the elements

3 Masses of U in g

4 Enrichments in %

5 Masses of ^{166}Er in g

6 Masses of ^{167}Er in g

7 Updated fuel rod burnups in MWd

8 updated fuel rod burnups in %

9 element burnups increment in MWd

The output format utilized is (A4, lX, A4, lX,4FlO.2, F10.3, F10.2, F10.3). Only the last three columns of data are updated; the first six columns of data are rewritten from the input file. The ELEM. OUT file needs to be renamed to the ELEM. IMP file for the subsequent burnup cycle calculations.

3.3.5 TRIGAP Code

The code TRIGAP was up dated for reactor calculation analysis for safe operation and specially the fuel management strategy of a TRIGA reactor. The application is,

- The criticality calculations are modified to integral experiments when the code receives the relevant data base, which essentially consists of the nuclear constants and the fuel operating history (burn-up). The reactor is shaped like a cylinder. If these conditions are satisfied, the program can be used for criticality forecasts, power peaking predictions, calculations and data logging for fuel element burn-up, in-core fuel management, and fuel usage improvement. The computations' expected accuracy is 0.5% for absolute criticality, 15% for power distributions and peaking, and 10% for fuel element burn-up—pretty much the same numbers as those demanded by power codes.
- The TRIGAP program was initially made for the IBM-PC. There is also a VAX version available. On an IBM-PC, a typical runtime is 8 minutes. Hard disks are not required.

The program consists of the main program TRIGAP, libraries and input:

TRIGAP.LIB – effective group constants

ELEM.DAT – fuel element data

TRIGAP.INP – input description

The structure of the main program is presented on flow-chart. For detailed description of the geometry and physical model of the reactor see CEBIS code manual. Both other subroutines use the same geometry and physical model. Detailed description of the subroutines is shown below and illustrated at Fig. 3.3.

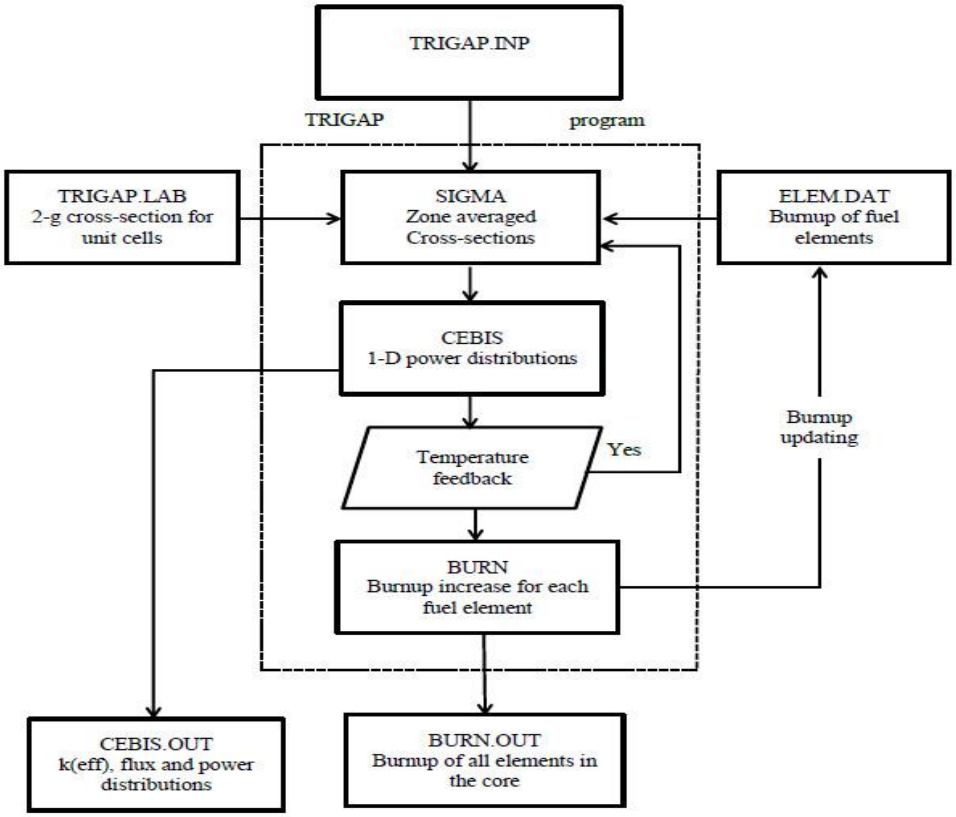


Fig. 3.3: Flow chart of the program TRIGAP

3.4 Subroutines and Libraries

3.4.1 Subroutine SIGMA

The effective 2-g cross-sections for all types of unit cells (fuel and nonfuel), which are kept in library TRIGAP.LIB as a function of burn-up, are used by SIGMA to determine the ring-smeared two-group diffusion constants. To determine the cross-sections for each fuel element in the core that correspond to the proper burn-up, SIGMA consults the library. The burn-up of each fuel component is monitored in

ELEM.DAT. The cross-sections of the effective unit-cells are computed using linear interpolation if the element burns away in between two steps in the library. The unit-cell cross-section is not expected to extrapolate. If an element's burn-up exceeds the library-defined threshold, program termination and a message are both recorded on CEBIS.OUT. The collection has the unit-cell cross sections for burn-up at maximum power, equilibrium Xe, and Sm conditions. For different operating scenarios, the cross-sections are produced using the adjustments listed below.Linear temperature correction (power correction)

(a) Xenon correction

(b) Samarium correction

3.4.2 Linear Temperature Correction (power correction)

Fuel and coolant temperatures are thought to change linearly with change of power. Only at one (nominal) power are the cross-sections for various fuel kinds on non-fuel unit cells tabulated. (temperature). The following modification is made if the fuel element power differs from nominal (for example, if the reactor power or radial and azimuthal power fluctuations are different).

$$\Delta\Sigma^{\mathrm{p}}(\mathrm{p},\tau) = \Delta^{\mathrm{p}}(\tau).\left(1 - {}^{\mathrm{p}}/_{\mathrm{p}^0}\right) \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (6)$$

Were,

$\Delta\Sigma^{\mathrm{p}}(\mathrm{p},\tau)$ = power correction to 2-g diffusion constants for unit cells $(\mathrm{D}_1, \mathrm{D}_2, \Sigma_{a1}, \Sigma_{a2}, \Sigma_{12}, \upsilon_1\Sigma_{\mathrm{f1}}, \upsilon_2\Sigma_{\mathrm{f2}})$

p = power of the element

p^0 = nominal power of the element

$\Delta^{\mathrm{p}}(\tau)$ = difference between cross-section at nominal and zero power

τ = burnup

It relate the power correction with burn-up. Δ^{p} has been calculated for two fixed burn-up values (τ_o, τ_1) and linearly interpolate :

$$\Delta^{p}(\tau) = \Delta^{p}(\tau_0) + \frac{\Delta^{p}(\tau_1) - \Delta^{p}(\tau_0)}{(\tau_1 - \tau_0)}(\tau - \tau_0) \ldots\ldots\ldots (7)$$

Values for p are included in the TRIGAP.LAB whereas SIGMA and p are included as independent variables for each fuel component. To prepare the x-section in SIGMA, TRIGAP.INP needs input an anticipated power distribution. (e.g., average power for all elements). When the power density reaches the temperature (or power) feedback, the application will continue this process for the chosen number

of times. In all applications in the real world, two iterations are sufficient.

3.4.3 Xenon Correction

For each fuel element, the Xenon correction is determined in the same manner as temperature:

$$\Delta\sum^{x}(p,\tau) = \Delta^{x}(\tau).(1 - f(p))\text{……..(8)}$$

$$\Delta^{x}(\tau) = \Delta^{x}(\tau_0) + \frac{\Delta^{x}(\tau_1)-\Delta^{x}(\tau_0)}{\tau_1-\tau_0}(\tau - \tau_0)\text{……..(9)}$$

$$f(p) = \frac{1+c}{1+c.{}^{p}/_{p_0}}.\frac{p}{p_0}\text{…….(10)}$$

Where:

$\Delta\sum^{x}(p,\tau)$ = Xenon adjustments to unit-cell 2-g diffusion constants

$\Delta^{x}(\tau_0)$ = Difference between full power, equal to Xe, and zero power

$\Delta^{x}(\tau_1)$ = Cross-sections interval at full power, equal to Xe

p = element power (kW).

p_0 = nominal element power (kW).

$f(p)$ = For a given power, it is a function proportional to the equilibrium Xe concentration.

c = Constant

It is tallied for all categories of fuel components in TRIGAP.LIB. The same goes for x (_1) and x (0).

The equilibrium xenon at that power is always taken into account when Xenon correction is calculated. There is a chance to calculate the reactor core at power and without the Xe condition (a parameter for each input data element).

3.4.4 Samarium Correction

Samarium adjustment is done for each element independently, just like temperature and xenon correction. It considers the impact of Sm saturation following a prolonged duration of shut-down. It was incorporated into the plan for exceptional situations that might occur after some fuel elements' reactors have been inactive for a while (for example: partially burned materials that were stored before to implantation for a sufficient amount of time to attain equilibrium samarium).

$\Delta\sum^{s}(\tau) = \Delta^{s}(\tau).FLAG$

For Δ^{s}:

$$\Delta^s(\tau) = \Delta^s(\tau_0) + \frac{\Delta^s(\tau_1)-\Delta^s(\tau_0)}{\tau_1-\tau_0}(\tau-\tau_0) \ldots\ldots\ldots\ldots\ldots\ldots\ldots (11)$$

List of symbols:

$\Delta\sum^s$ = samarium correction

FLAG ,

0 = equilibrium Sm

1 = peak Sm

$\Delta^s(\tau_0)$ = Conditions with saturated Sm after shut-down.

τ = burnup of the element.

3.4.5 Homogenization

Xenon and Samarium corrections are calculated using the formula

$$\sum(p,\tau) = \sum(p_0,\tau) + \Delta\sum^p(p,\tau) + \Delta\sum^x(p,\tau) + \Delta\sum^s(\tau) \ldots\ldots\ldots (12)$$

Ring averaged cross-section are calculated by assuming that same volume V_i is occupied by each unit-cells :

$$\langle\sum_g\rangle = \frac{\sum_i V_i\sum_{i,g}}{\sum_i V_i} \qquad g = 1,2$$

$$\langle\frac{1}{D_g}\rangle = \frac{\sum_i {V_i}/{D_{i,g}}}{\sum_i V_i} \qquad g = 1,2$$

3.4.6 Subroutine CEBIS

CEBIS is a standalone one-dimensional two-group diffusion code in its original form. The Manual of this program contains a thorough description. A subroutine called CEBIS is utilized in the TRIGAP software. Typically, it is decreased. Only cylindrical geometry may be employed with it, and the adjoin diffusion equation cannot be resolved. For CEBIS, no external input is required. With the exception of the graphs of the power and flux distributions, everything in the independent version of CEBIS is optional. Ring averaged power distributions from CEBIS can be used to determine the corresponding power, Xe, and Sm adjustments. Corrected cross-sections are fed into CEBIS as input, a new power distribution is generated, and this process is recurrent until two CEBIS power distributions become equal. The software is not designed with the convergence requirement in mind. Experience has shown that for small, compact cores like TRIGA, two iterations are

sufficient. If there are additional iterations, only the most recent iteration's output is displayed on CEBIS.OUT. Following the iteration process, the burn-up computation will be carried out by the program.

3.4.7 Subroutine BURN

The burn-up of every fuel rod in the current step is calculated by the subroutine BURN using the total energy (measured in MWh) that the reactor and core power distributions have produced.

Distributions of power are sourced from CEBIS. In heterogeneous rings with various fuel constituents, azimuthal power variations are considered even if they have no bearing on the ring's average power. With the aid of a 2D diffusion algorithm and R-geometry, power correction factors for mixed rings were determined beforehand.

Increments of burn-up for each fuel rod in the reactor are the end result of BURN. In order to ensure that ELEM.DAT contains an updated burn-up, they are automatically added to the elements' burn-up. There is also a separate output file for BURN that records burn-ups for each zone for each element in the core in MWh and in% of U^{235} along with the element's identification number. At the conclusion, elements are rewritten with their burn-up in descending order, which is helpful for computations involving in-core fuel management.

3.5 TRIGAP.LIB

TRIGAP program's database contains the TRIGAP library (= TRIGAP.LIB) and the element history (= ELEM.DAT). For LEU fuel elements that were either 20 w/o of U-235 or 8.5 w/o of U-235 with 20% enrichment of U-235 and burnable poison Er, the library was developed at AERE in Savar at 3 MW.

It contains the functioning cross-sections of the TRIGA reactor's fuel and nonfuel unit cells. The volume is the same for each unit-cell. Nonfuel unit cells have a rod made of a non-fissile substance (such as graphite, beryllium, void, or water) that is encircled by water. A fuel rod and water are found in fuel unit cells. All unit-cell cross-sections were calculated using the 18-group transport approximation and the WIMS-S transport code. For unit-cells free of fissile material, a supercool approximation (central, nonfuel rod, surrounded by six fuel rods) had to be used; ultimately, only cross-sections over the central nonfuel unit-cell were taken into

consideration. The cross-sections are calculated at a different nominal power for each type of fuel unit-cell and tabulated according to the amount of U-235 that is burnt, ranging from 0% to 35% for FLIPS and 50% for LEU and standard components. Any unit-cells that don't have fuel in them are treated as if they did. The library's information is arranged as follows:

Number of types of elements, masses of elements for all types and coefficients (for three types of fuel) for conversion of burn-up from MWd /tU to % U-235 (format il0), format (6e12.5), format (3e14.6)

- Burn-up (in % U-235) and identification number of step, at which cross-sections for FLIP, LEU and standard fuel elements are calculated (35 steps for FLIPS, 25 steps for LEU and standard F.E.)
- Cross section for FLIPS (at 35 different burn-up steps), LEU (25 step) and standard elements (25 steps). At each burn-up step the cross-sections are written in the following order:

$$D_1 \quad \sum_{a1} \quad \sum_{12} \quad \nu_1 \quad \sum_{f1} \quad {}^{\nu_1}/_{c_f}$$

$$D_2 \quad \sum_{a2} \quad \nu_2 \quad \sum_{f2} \quad {}^{\nu_2}/_{c_f}$$

- Cross-sections of non-fissile unit-cells in the following order: water-channel, void-channel, graphite element, beryllium element, reflector format
- Nominal power for all three types of fuel format (3f10.6)
- Cross-sections at nominal power are followed by different coefficient for corrections of cross-sections to other conditions:
- burn-up for all three types of fuel at which temperature, xenon and samarium corrections are tabulated, format (3f10.6)
- constants for xenon corrections for all three types of fuel format (3f10.6)
- temperature, xenon and samarium corrections format (5e14.6)

The identification number for the Bangladesh Library is as follows:

1. LEU FUEL FOLOWER
2. LEU FUEL
3. STANDARD FUEL
4. TRANSIENT ROD
5. WATER ONLY (UNIT CELL)

6. VOID CHANNEL
7. GRAPHITE ELEMENT
8. DETECTOR
9. IRRADIATION WATER CHANNEL (THIMBLE+WATER)
10. CENTRAL THIMBLE ($\frac{1}{2} \times VOID + \frac{1}{2} \times WATER$)
11. WATER REFLECTOR
12. GRAPHITE REFLECTOR

This library has been created from WIMS 69 group library. But it is a modified library which contains the cross section of ZrH. This is very important for the TRIGA Reactor Analysis as the cross section of a free hydrogen and hydrogen in zirconium hydride differs significantly.

3.6 ELEM.DAT

For each TRIGA reactor, an ELEM.DAT file must be created depending on the reactor's operating history. For each element, it should include the following information:

- The element's identification number;
- The element's type
- The element's past

3.7 TRIGAP.INP

All independent input data are entered through the TRIGAP.INP file, into TRIGAP. All the requirements for solving the diffusion equation are specified in the first general section of the input data. The identification numbers and flags for the xenon and samarium corrections for each core component are written in the second half. The ring-averaged power densities must be defined at the conclusion. All input data must be entered in the format and order specified. The first two cards are set aside for comments of any kind and will also appear in the CEBIS.OUT file. The subsequent 16 cards (numbers 3 through 18) are shown in textual format.

Table 3.3: TRIGAP input data

Card no	Parameter	Description
3	core radius, R_C	radius of the reactor core

		(fissionable material) in cm
4	reactor radius, R_R	outer radius of the reflector (cm), where $\varphi = 0$ boundary condition is imposed
5	core height, H_C	height of the reactor core (fissionable material) in cm
6	reactor height, H_R	height of the whole reactor (cm), including top and bottom reflectors, where on outer boundaries $\varphi = 0$ boundary condition is imposed
7	P_{therm}	total thermal power of the reactor in kW
8	N_Z	number of homogeneous zones into which the reactor is divided in radial direction **Note:** $1 \leq N \leq 20$ (default DIMENSION) **Note:** In TRIGAP zones, containing fissionable material must be identical to rings, as they are defined for TRIGA 250 kW reactor. Reflector region can be split in more than one zone.
9	N_I	Number of finite differences intervals, in which R_R is divided. Note: to get negligible discretization error, N_I must be greater than reactor radius divided by the

			minimal thermal neutron diffusion length L, $L_{min} = \min_{i=1,N_Z} \sqrt{\frac{D_{2,i}}{\Sigma_{2,i}}}$ Note: $N_I \leq 200$ (default dimension)
10		ε_1	convergence criterion for multiplication factor, recommended 0.00001
11		ε_2	convergence criterion for fission density, recommended 0.0001
12		N_{IT}	allowed number of fission density iterations, recommended $10 \leq N_{IT} \leq 200$
13		B_{Z1}^2	axial buckling in first group (cm^{-2})
14		B_{Z2}^2	axial buckling in second group (cm^{-2}) Note: for first approximation geometric B_Z^2 can be used for buckling
15	Igraf		If igraf=1, graphs of fluxes and power will be plotted, if igraf=0 there will be no plots on CEBIS output
16		N_s	N_s controls the printout of the results, i.e., every N_s-th point value of fluxes will be printed out of N_I Note: $1 \leq N_s \leq N_I$

17	burnup calculation flag	If 1 is entered burnup of all elements in the core is calculated after diffusion calculation from the energy produced by the core in the current step. if 0 is entered only diffusion equation is solved and no burnup calculation is performed
18	iter	number of temperature / power iterations; If zone-averaged power densities are known when first running the program, one iteration will suffice, otherwise at least two iterations are necessary
19 format (2f12.4)	E_{acc}, E_{curr}	E_{acc} = accumulated burnup = total energy produced by the core in MWh before the burnup step (if not known could be set to zero) E_{curr} = energy produced in current burnup step in MWh
20 format (8f8.4)	a(iz), iz=1, N_Z	zone outer radii including reflector (Homogeneous zones radios)
21+N_Z format (8i4)	nmax(iz), iz=1, N_Z	number of locations (fuel and nonfuel) in rings starting from the center Note: For reflector zones enter 1.
21+N_{Z+1}	C_1, C_2, C_3	coefficients for azimuthal

format (3f10.4)	power corrections for all three types of fuel elements (flip, leu and standard). C_i is the ratio between fuel element power and power, averaged over all fuel elements in the ring. Typical values are: $C_1 = 1.4, C_2 = 1, C_3 = 0.6$

Follow the information about core loading pattern and operation conditions. Data for all fuel elements are read from TRIGAP.INP through two DO loops:

```
   DO *              IZ=1, NZ
   READ (**)         IN (N, IZ), IXE (N, IZ), N=1, NMAX
** FORMAT           (6(I8,212))
*  CONTINUE
```

Outer loop counts zones (rings) from 1 to number of zones N_Z = NZ. Implicit loop counts positions in zone (ring) IZ from 1 to maximal number of positions NMAX= Nmax.

IN = identification number is occupied of fuel element,

IXE = flag for Xe correction

IXE =	0, f.e. is xenon free (Fresh fuel element or not operating a longer period, app. 50 hrs.)
	1, Xe is in equilibrium (Continuous operation)

ISM = flag for Sm correction

ISM =	0, Sm at normal operation (Continuous operation)
	1, Sm is in saturation after shut down (Not operating a longer period, app. 2 months)

For other cases:

IN = identification number of the non-fuel element or material (3.6 File ELEM.DAT)

IXE = ISM = 0

Numbering of the positions within a ring is arbitrary for fuel or non-fuel elements (positions).

After the last card describing the core loading pattern the cards specifying ring averaged power densities must follow:

format (e 16.6)	p(iz), iz =1, N_Z	ring averaged power densities (kW/cm^3). If power densities are not known, core averaged power densities can be used for all zones and number of iterations (iter) must be set to at least 2.

3.8 Output Description

TRIGAP program output consists of CEBIS.OUT file and BURN.OUT file. CEBIS.OUT file contain the output data like effective multiplication factor, flux data etc. At the same time BURN.OUT contains the data of burnup of individual fuel elements, power per fuel element etc.

3.8.1 CEBIS.OUT:

The only output of CEBIS is printed output. It is split into two sections. All general input parameters are first provided in an easily understandable manner. Then, all zones' group constants are printed. The direct equation's answer makes up the second portion. The iteration procedure's most significant outcomes are printed first. The point-wise distributions are printed following that. The distributions of flow and power are then plotted.

3.8.2 BURN.OUT

Information about BURN.OUT is as follows:

The information on the burn-up of each element in the core is printed after the total cumulative energy produced by the core. The identification number of the (fuel)

element is printed for each zone along with the amount of initial U-235 burned up in MWh and percent. The thermal power of the fuel element during the current burn-up step is also printed. The list of fuel ingredients is then shown in descending order of U-235 burn-up percentage. The final piece of data is the average burn-up for all fuel element types.

Chapter 4
RESULT AND OBSERVATION

4.1 Brief Description of Analyzed Neutronics Parameters

The two-dimensional TRIGLAV code was used to investigate the most significant neutronics parameter for BTRR. The following were the examined variables:

1. Effective Multiplication factor
2. Core Excess reactivity
3. Individual Fuel element burn-up and Ring wise burn-up
4. Core burnup lifetime

4.1.1 Effective Multiplication Factor

Because it is crucial to comprehending the chain reaction's progression in the reactor core, the multiplication factor must be determined. Calculating neutron flux is necessary for this factor's determination. As a result, it is a crucial fundamental parameter for nuclear reactors, and the value of this parameter depends on the material composition, the core's geometry, and the nuclear data library. To accurately simulate the neutron life cycle in a real, constrained reactor, it is essential to take into account neutrons that leak out. A multiplication factor that takes leakage into account is called the Effective Multiplication Factor (keff), which gauges the expansion or contraction of the reactor's neutron cloud. The ratio is the product of the amount of neutrons produced by fission in one generation and the number of neutrons lost to absorption and leakage in the generation preceding it. This element can be expressed numerically as follows:

$$k_{eff} = \frac{Neutron\ production\ from\ fission\ in\ one\ generation}{Neutron\ absorption\ + Neutron\ leakage;\ \ in\ the\ preceding\ generation}$$

The neutron density is neither increasing nor decreasing in a self-sustaining chain reaction of fissions, hence the value of k_{eff} is equal to 1. The critical condition is the state in which the neutron chain reaction is self-sustaining and the neutron population is neither growing nor shrinking. The reactor is referred to as supercritical if the neutron production outweighs the neutron absorption and leakage. Keff is greater than one in a supercritical reactor, and the neutron flow

rises with each generation. On the other side, the reactor is referred to as subcritical if the neutron generation is smaller than the neutron absorption and leakage. Keff is smaller than one in a subcritical reactor, and the flux falls with each generation.

4.1.2 Core Excess Reactivity

The core excess reactivity built into a reactor to overcome the effect of fuel burn-up and the buildup of toxic fission-products while it is operating. It is greater than the quantity required to reach criticality. Reactivity is a gauge of how far a reactor has deviated from critical. It is a useful idea to forecast how the neutron population of a reactor will change over time and is related to the value of keff. When keff is not equal to 1, the power level, neutron density, etc., are continually changing. The core excess reactivity is denoted by $\rho\ (\$)$ and it is explained as below equation.

$$\boldsymbol{\rho}\ (\$) = \frac{\boldsymbol{k_{eff}-1}}{\boldsymbol{k_{eff}\beta}} \qquad \ldots\ldots\ldots\ldots\ldots\ (13)$$

Where, to translate the unit from k/k to dollars or cents, β with a value of 0.007. This equation was used to determine the excess reactivity in this study.

4.1.3 Individual Fuel element burn-up and Ring wise burn-up

The burnup estimations and their interpretation were the main focus of this work. According to definition, the burnup is the quantity of energy per unit mass of fuel, typically represented in units of MWd or kWh if you choose. In other words, the thermal energy produced in the reactor during that time is equal to the reactor power (assumed constant) times the irradiation time. Let's formulate this as follows:

Burnup = P × T / M,

where P is the thermal power of the reactor (the electrical power is typically about one-third of this), T is the irradiation time (the actual time spent inside the reactor is obviously no good as a measurement because fuel replacement and maintenance also require time), and M is the total mass of the fuel. If we use the actual T of the fuel element and, if necessary, the actual M's in the event that the fuel mass has changed, the burnup can effectively explain the condition of a single fuel element or even a fuel rod. In burnup calculations, it is described as a time-independent eigenvalue problem known as the criticality equation, in which case the solution

consists of a distribution of neutron densities and a multiplication factor that identifies the time dependency of the system.

4.1.4 Core Burnup Lifetime:

Core Burnup lifetime is referred to be the lifetime of a core that will maintain criticality or critical state within safety margin without rearranging or reloading the fuel element in the first loaded core. In other words, how long the core will function until it needs to reload or rearrange the fuel element in the first loaded core for criticality.

CHAPTER 5
DISCUSSION ON RESULT AND RELEVANCE

5.1 Analysis of Initial criticality of BTRR

The 2-D neutron diffusion coding system TRIGLAV was used to perform the initial criticality analysis of the BTRR. The same study has been completed using TRIGAP code. By employing the inverse multiplication method, this code was used to forecast the amount of fuel elements needed to reach its first criticality. In order to load BTRR core, fuel rods were added one at a time against the multiplication factor.

5.1.1 Prediction of Critical Buckling Value

The BAEC TRIGA reactor's initial critical core, whose core configuration is depicted in Fig. 5.2, was utilized to examine the reactor's criticality. Predicting the core's buckling value is the first stage, and it would be simpler to determine the original reactor core's criticality. By adjusting the value of buckling in the core, the value of the multiplication factor may be shown. The buckling value was first set at 0.005001 and decreased until it reached the critical limit. The values of the buckling and multiplication factors are provided in Table 5.1, and its curve is depicted in Fig. 5.1.

Table 5.1 The values of buckling and multiplication factor, k_{eff}

Buckling value (no. of neutrons per cm^2)	Multiplication factor, k_{eff} (Calculated)
0.005001	0.9887090
0.004901	0.9908721
0.004801	0.9930440
0.004701	0.9952244
0.004601	0.9974171
0.004501	0.9996161
0.004424	1.0013161

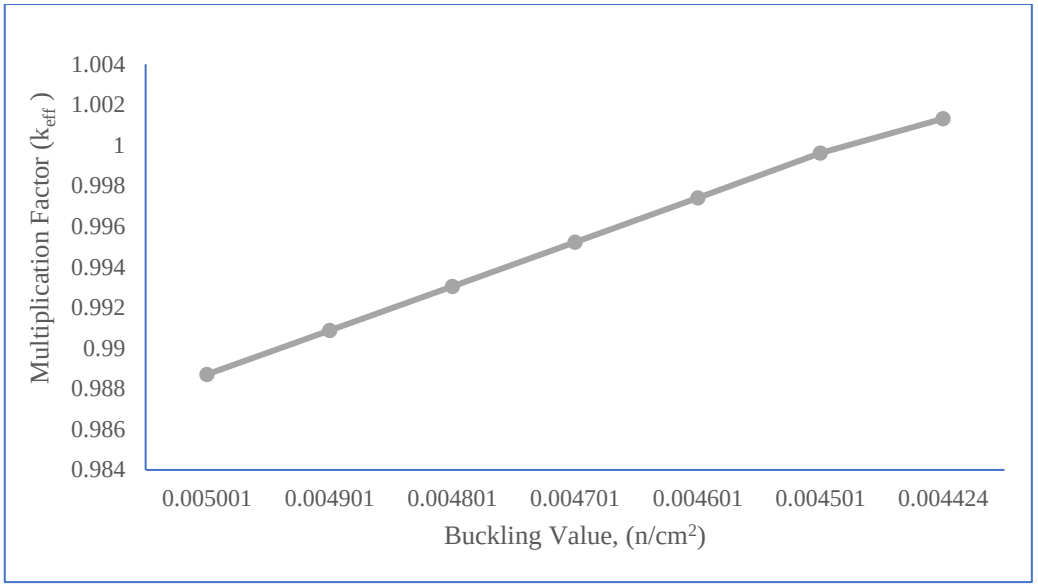


Fig. 5.1: Change of Multiplication Factor with respect of buckling values

The curve between a multiplication factor and a buckling value is depicted in Fig. 5.1. It is possible to infer from the graph that as the buckling value rises, the multiplication factor falls. The critical buckling value employed in this study is 0.004424 against the critical multiplication factor.

5.1.2 Assessment of Number of Fuel Elements Required for Initial Criticality

The same critical buckling value, 0.004424, is applied for each fuel element put in the core once the prediction of buckling value attained its criticality. The TRIGA LEU fuel is made up of burnable toxic erbium, zirconium hydride, and 20% (wt) uranium enriched to 19.7% of 235U. Starting with the first fuel element fed into the core, the LEU core's multiplication factor is often noted.

Table 5.2: Number of fuel elements vs. multiplication factor

Number of fuel elements	Multiplication factor, k_{eff}
1	0.2515029
6	0.5854959
12	0.7253804
18	0.8078631
24	0.8622152
30	0.9049892
36	0.9388538
42	0.9695445

48	0.9930919
50	1.0013161

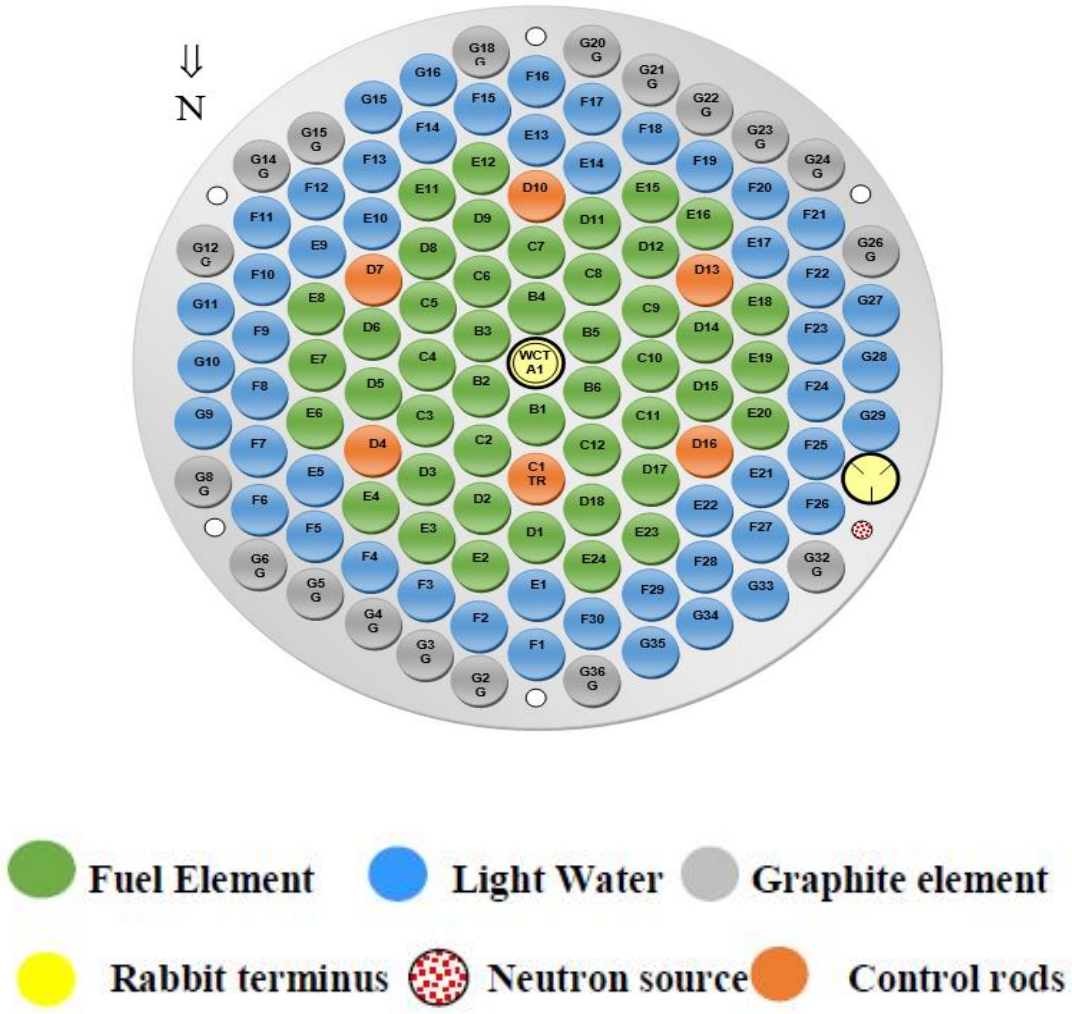


Fig. 5.2: Assessment of number of fuel elements required for initial criticality

The amount of fuel elements required to construct the reactor core in order for it to reach its first criticality is then noted. The graph of Table 5.2's fuel element count vs multiplication factor is shown in Fig. 5.2.

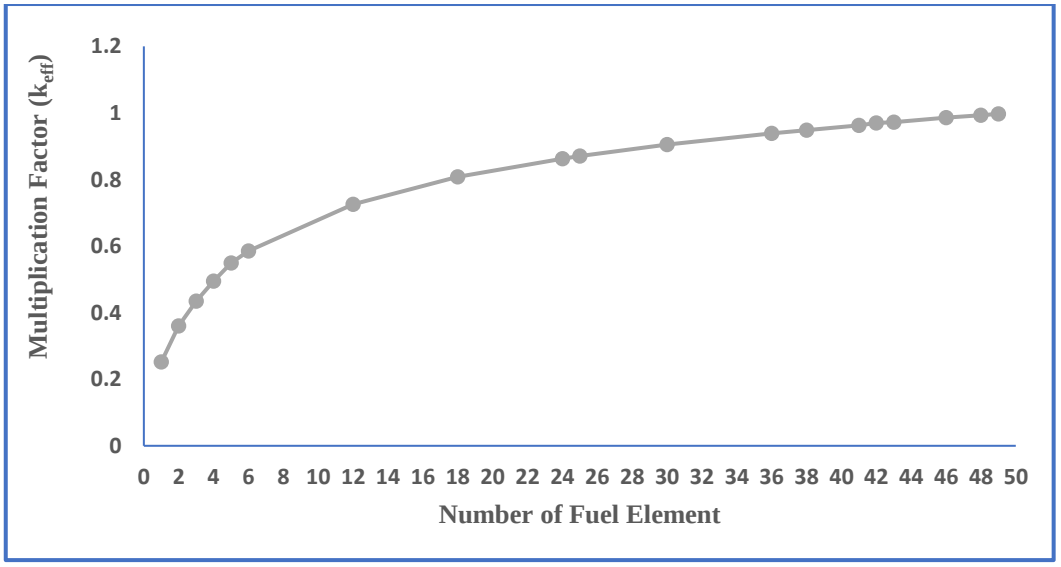


Fig. 5.3: Change of multiplication factor with fuel rod insertion

According to Table 5.2, the core becomes critical when it has 50 fuel elements loaded, and Fig. 6 shows that the multiplication factor rises as the number of fuel elements grows. While the experimental value of keff is 1.0012746, the computed value by the TRIGLAV algorithm is 1.0013161 (0.0042%). Because it affects the geometry of the core, the composition of the materials, and the nuclear data library, it is a key fundamental parameter of nuclear reactors. As a result, whereas its measured excess reactivity is 0.19, the excess reactivity calculated by the TRIGLAV code is 0.1962. Table 5.3 displays the calculated k_{eff} and excess reactivity values after the experiment.

Table 5.3: Comparison of the results (k_{eff} and excess reactivity)

Methods	k_{eff}	Excess Reactivity ($)
Experiment	1.0012746	0.190
TRIGLAV Code	1.0013161 (0.0042%) *	0.196 (3.16 %)
TRIGAP Code	1.001308(0.0033%)	0.195 (2.63%)

*Error in %= (C-E/E) X 100

From Table 5.3, it may conclude that the calculated values of k_{eff} and excess reactivity show a reasonable agreement with experiment. Besides, the % deviations are within the acceptable limits.

5.2 Estimation of Critical Mass

The smallest quantity of fissile material (U-235) required for an ongoing nuclear fission chain reaction is known as the critical mass. The nuclear properties of a fissionable substance—more particularly, its nuclear fission cross-section—as well as its density, enrichment, shape, temperature, purity, and environment—all affect the critical mass of that substance. When 50 fuel elements are placed into the original core, the core becomes critical. U-235 has a total mass of 4.794 kg in 50 fuel components. Therefore, 4.794 kg of U-235 constitute the initial critical mass of the BAEC TRIGA Mark II Research Reactor.

5.3 Validation of the TRIGLAV Code against Other Sources

In order to verify and validate the TRIGLAV code, the findings from the experiment and results from other computer programs were compared. The output of TRIGLAV yielded a multiplication factor value of 1.0013161 (0.0042%). Another simulation was run using the 1-D deterministic neutron diffusion code TRIGAP [10] to confirm the results. The multiplication factor of the starting core produced using the TRIGAP code is 1.001308 (0.0034%), but its experimental value is 1.0012746 , according to this simulation.

Table 5.4: Number of fuel elements vs. multiplication factor

Fuel elements Count	Multiplication factor, k_{eff} (TRIGLAV)	Multiplication factor, k_{eff} (TRIGAP)	Multiplication factor, k_{eff} (Experimental Data from log book of 1986)
1	0.2515029	0.2987022	-
2	0.3600676	0.4317676	-
3	0.4348652	0.5045977	-
4	0.4946223	0.5485955	-
5	0.5490102	0.5764301	0
6	0.5854959	0.5942008	-
12	0.7253804	0.7566236	-
18	0.8078631	0.8809288	-
24	0.8622152	0.9123074	-

25	0.8702981	0.9196712	0.47
30	0.9049892	0.9236573	-
36	0.9388538	0.9335319	0.841
38	0.9479641	0.9488616	0.863
41	0.9625924	0.9668239	0.911
42	0.9695445	0.9735357	-
43	0.9719812	0.9785291	0.939
46	0.9856201	0.9900121	0.919
48	0.9930919	0.9962517	-
49	0.997153	0.9982153	0.991
50	1.0013161	1.001308	1.0012

The multiplication factor for TRIGLAV, TRIGAP, and experimental data for analysis of the initial critical core of the BAEC TRIGA reactor are shown in Fig. 5.4 along with the number of fuel elements.

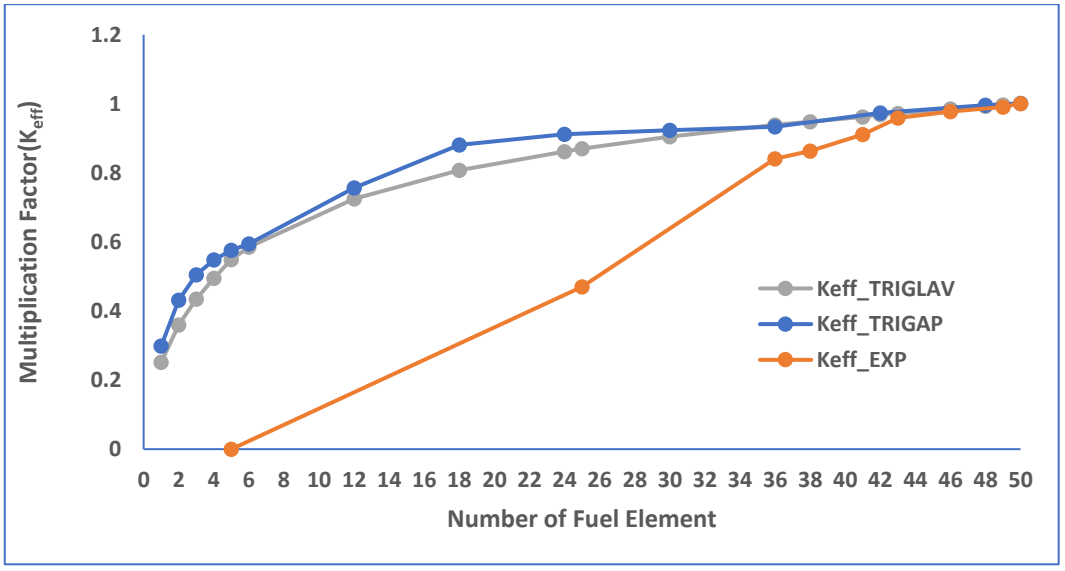


Fig. 5.4: Comparison of the multiplication factor, k_{eff}, and the number of fuel elements for the experimental data and the TRIGLAV and TRIGAP models.

The 1-D neutron diffusion code TRIGAP, which is specifically created for computations involving the 3 MW BAEC TRIGA Research Reactor, and the TRIGLAV code are in good agreement despite having distinct methods and

approaches for calculating the multiplication factor. The curves produced from the experimental data differ significantly from the TRIGLAV-calculated values in the subcritical zone. Although it took the same number of fuel elements for criticality to be reached, the multiplication factor produced by TRIGLAV is still thought to agree with TRIGAP. The very low k_{eff} value in the subcritical zone is the cause of the significant disparity between the TRIGLAV results and experimental results. This might be caused by the detector's relatively low signal-to-noise ratio. In addition, there is a disparity between the TRIGAP and TRIGLAV codes in the subcritical region because of the implications of the physical model with 1-D and 2-D simulations, although they are both quite consistent in the critical region.

The TRIGLAV computer code is validated as all three curves become consistent and agree with one another in the vicinity of the critical region. The neutron flow rises when more fuel components are introduced to the core, and as a result, the multiplication factor rises as well. The hypothesis is supported.

5.4 Operational Core Analysis:

In order to establish the operational core configuration, 100 fuel elements were therefore loaded in step; hence, the multiplication factor and the excess reactivity were calculated. BTRR operational core has been shown in figure 5.5.

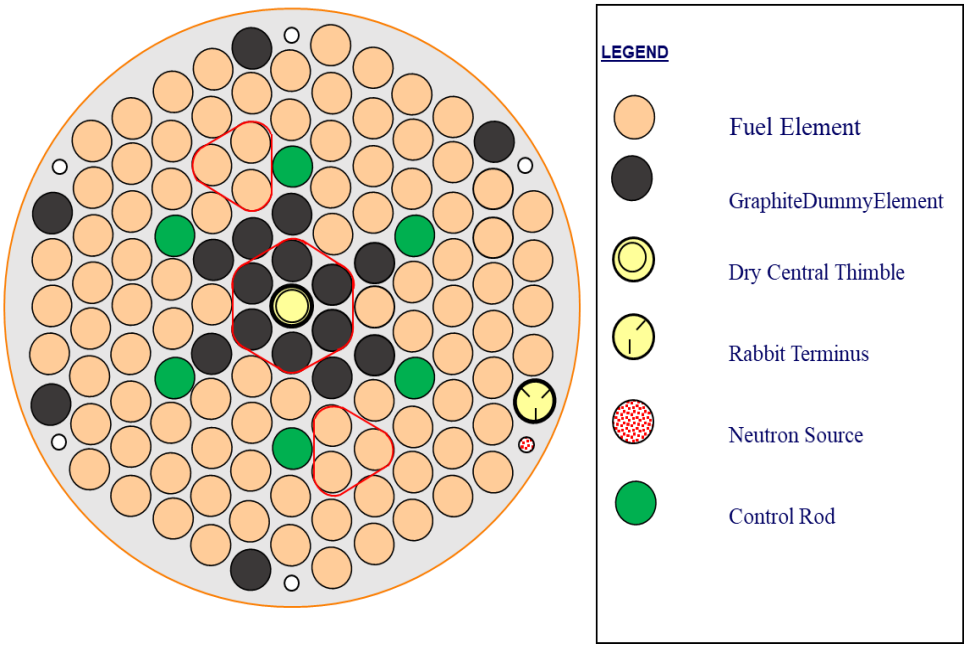


Fig. 5.5: BTRR operational core

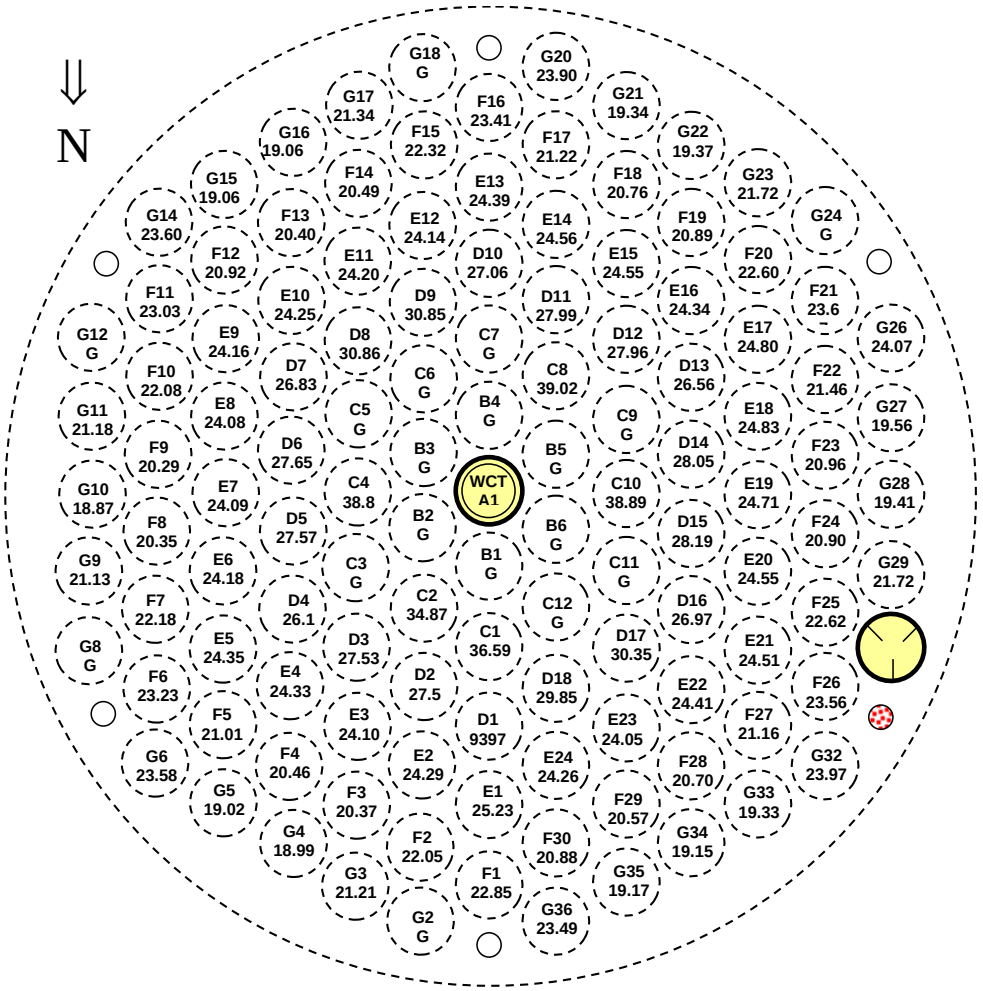
Table 5.5: Comparison of results (k_{eff} and excess reactivity)

Methods	k_{eff}	Excess Reactivity ($)
Experiment	1.077459	10.27
TRIGLAV Code	1.0744693(0.28%)	10.34 (0.68 %)
TRIGAP Code	1.0743390(0.29%)	10.32 (0.49%)

*Error in %= (C-E/E) X 100

From Table 5.5, it may conclude that the contained values of k_{eff} and excess reactivity show a reasonable agreement with the experiment. Besides, the % deviations are within acceptable limits.

In addition, the power per fuel element of the BTRR operational core which has been calculated using TRIGLAV code has been shown in Fig. 5.6.



*kW is used as unit of Power/fuel element (Fresh Core , 0 burnup condition, 2.4 MW power)

Fig. 5.6: Power per fuel element data

5.5 Burn-up Calculation

Steps of Burn-up Calculation

- Obtain the total operational history including MWd and average Power
- Run TRIGLAV for the operational core with average power and total accumulated operation time in unit of days of [change POWER and BURNUP in input file]. use the buckling value obtained from previous calculation and xenon set to equilibrium
- This will provide a file named TRILAV.OUT containing the current neutronics parameters of the core, and the ELEM.OUT, containing data of fuel burnup
- Next is to determine how long or how much-accumulated burnup the core can achieved until the reactivity become zero or k_{eff}=1.0 with full power and nominal water temperature during operation
- Copy the original ELEM.INP (fresh fuel data) to another folder. change the name of ELEM.OUT file obtained from 3rd step to ELEM.INP In TRIGLAV.INP change the power to nominal full power, water temp, TWATER to nominal temp during full power operation, XENON set as 1, equilibrium, BURNUP is your dependent variable, change this value many times from the shortest to the longest tenure until k_{eff}=1.0

Table 5.6: TRIGLAV burnup information at an 800 MWd burn up

	Hottest fuel				Least hot fuel			
Ring	Fuel Location	Power (kW)	Burn-up (%)	Burn up (MWd)	Fuel Location	Power (kW)	Burn-up (%)	Burn up (MWd)
C	C-08	38.95	16.573	13.763	C-02	34.82	15.879	12.304
D	D-08	30.81	13.038	10.887	D-04	26.04	11.962	9.202
E	E-01	25.23	10.293	8.915	E-20	24.54	9.897	8.671
F	F-21	23.67	9.788	8.363	F-09	20.31	9.604	7.176
G	G-20	22.10	9.086	8.455	G-28	19.43	9.048	6.865

Table 5.6 identifies the individual hottest fuel and the least hot fuel element of each ring at 800 MWd burnup. The fuel housed at C-08 position is the hottest fuel element of C ring having 16.573% burnup whereas the fuel housed in C-02 position is identified as the least hot fuel with 15.879% burnup. Consequently, other ring fuels

has also been identified.

Table 5.7: Data for each ring's TRIGLAV burnup at 1400 MWd burn up

	Hot fuel				Least hot fuel			
Ring	Fuel Location	Power (kW)	Burn-up (%)	Burn up (MWd)	Fuel Location	Power (kW)	Burn-up (%)	Burn up (MWd)
C	C-08	36.37	28.981	24.83	C-02	33.45	27.76	23.003
D	D-08	30.08	22.803	19.43	D-04	25.38	15.57	17.77
E	E-01	25.33	17.275	16.90	E-20	24.67	15.40	15.97
F	F-21	23.73	17.112	16.76	F-24	21.28	14.93	13.52
G	G-20	23.76	15.897	14.56	G-28	19.78	14.45	13.10

Table 5.7 identifies the individual hottest fuel and the least hot fuel element of each ring at 1400 MWd burnup. From the data it is clear that the fuel housed in C-08 position is the hottest fuel of C ring with 28.981% burnup and the fuel element of C-02 is the least hot one. For D ring it is D-08 and D-04 respectively. This information will be very helpful for any sort of reloading or re shuffling of BTRR core which may ensure optimum use of existing TRIGA fuel. The BTRR fuel burnup was computed using 50-Megawatt Day (MWd) intervals up to a burnup of 200 MWd , and the burnup step after that was taken as 100 MWd up to a burnup of 1400 MWd . To precisely track changes in burnup parameters, smaller burnup steps are first taken. Figures 5.7, 5.8, 5.9 and 5.10 show the average burnup by ring at 700, 800, 1200 and 1400 MWd, respectively. Dry Central Thimble (DCT) is present in Ring A, while graphite dummy element is present in Ring B. These two circles are therefore absent from the figures. These graphs compare the TRIGLAV burnup computation to the MVP-BURN code. The outcomes of these codes, which are given above, show a very good agreement with one another.

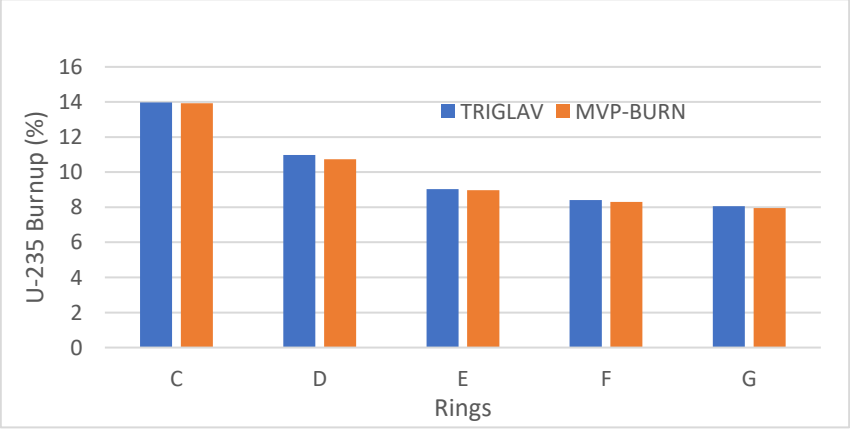


Fig. 5.7: Core burnup data comparison at 700 MWd

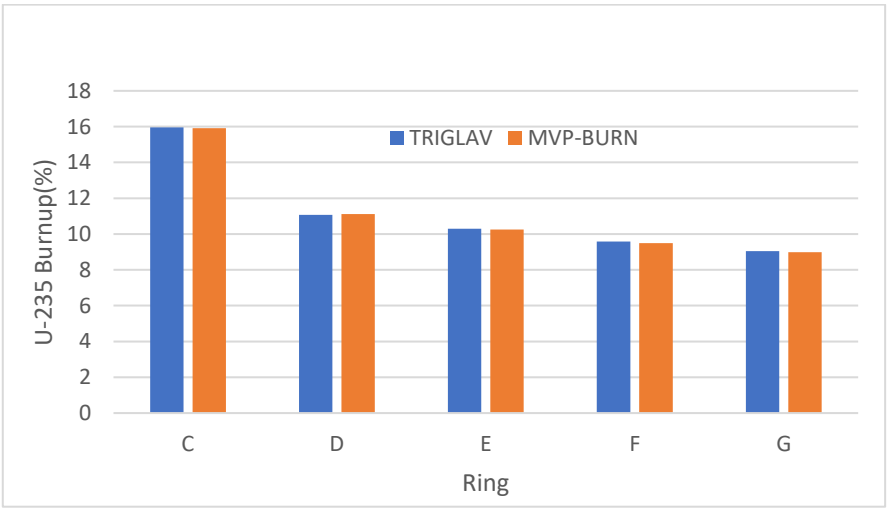


Fig. 5.8: Core burnup data comparison at 800 MWd

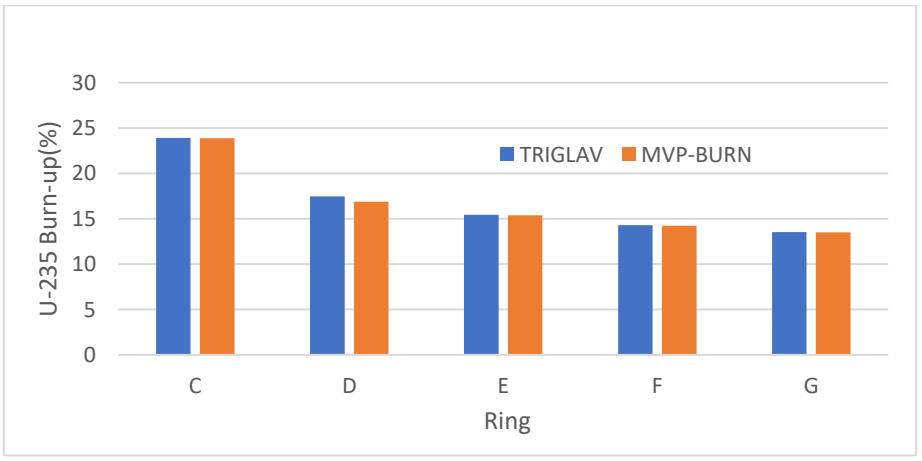


Fig. 5.9: Core burnup data comparison at 1200 MWd

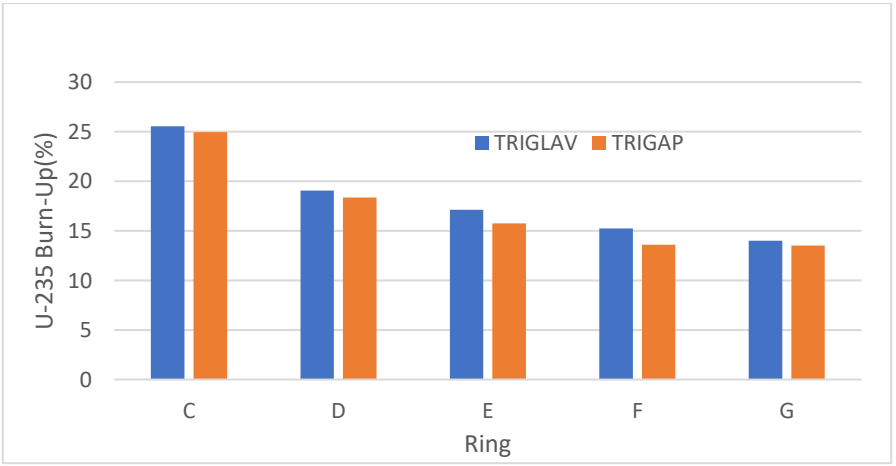


Fig. 5.10: Core burnup data comparison at 1400 MWd

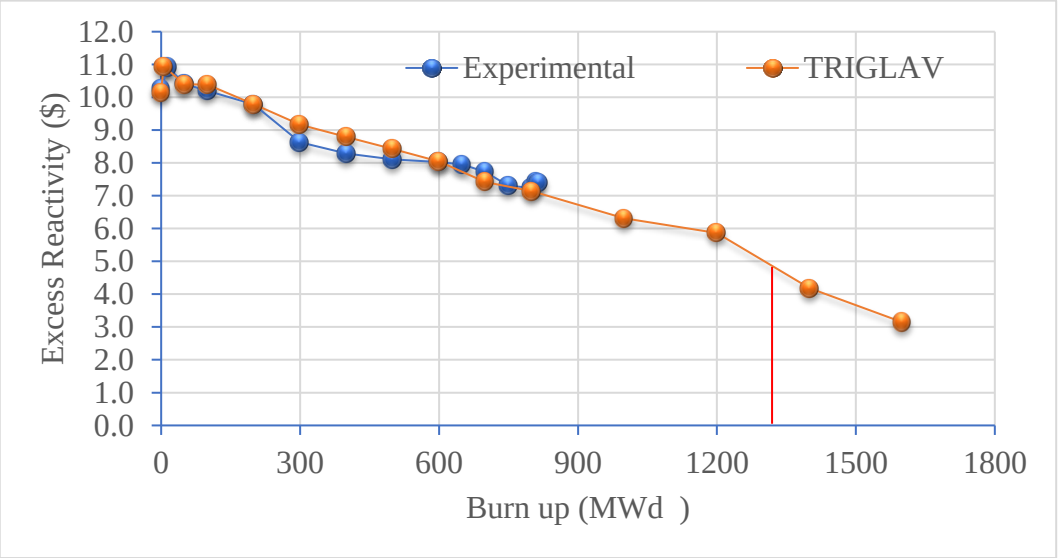


Fig. 5.11: Core excess reactivity comparison

Fig. 5.11 shows the actual excess reactivity data from the operation logbook plotted with TRIGLAV results. Due to the core's conversion to Dry Central Tube (DCT) in 1988 from Wet Central Tube (WCT), initial data displays a zigzag pattern (Salam et al., 2016). This switch resulted in an increase in the fresh core excess reactivity from 10.27 to 10.94 dollars. The fuel is burned up, reducing the extra reactivity. The TRIGLAV value and the real excess reactivity data from the experimental results are nicely aligned. The real excess reactivity at an 800 MWD burn up condition is $7.4, but the TRIGLAV value is $7.14. The TRIGLAV result differs by 3.51% from the real excess reactivity. Radioactive fission fragments are accrued inner side of the fuel cladding element during the reactor operation. Among them a major part of the fission products is gaseous. With the increment of fuel burnup, the fission gas pressure is also increased which cause maximum yield stress on the cladding elements. The maximum allowed individual fuel burnup for TRIGA fuel is around 50% (Lyric et al., 2013). According to the TRIGLAV analysis the maximum ring wise burnup is remain within the safer limit even on 1400 MWD burnup.

5.6 Analysis of Core Burnup Lifetime

The core burnup lifetime of the 3 MW TRIGA Mark II Research Reactor was computed as part of this study's secondary goal, and it was done using the two-dimensional diffusion code TRIGLAV at the current highest permitted power level of 2.4 MW. The following significant neutronic characteristics of the in-core fuel

management of the 3 MW TRIGA LEU core are also explained in this paper in addition to the core burnup duration. Utilizing the TRIGAP algorithm, the core burnup lifetimes under various power conditions were computed. Table 5.8 lists the determined values for core burnup lifetimes using various techniques.

Table 5.8: Comparison of core burnup lifespan (MWd)

TRIGLAV Code	TRIGAP Code	MVP-BURN Code	NUREG-1282 as Reference
1300	1250	1200	1200

The calculated findings were contrasted with the reference data from the Safety Evaluation Report and the MVP-BURN three-dimensional code. (NUREG-1282). Core excess reactivity has been used to predict the fuel life. According to TRIGLAV, the increased reactivity in the present core design will cost $5 after 1300 MWD burnup. Since the existing core configuration only allows for a minimum of $5 worth of extra reactivity, the BTRR can be operated at roughly 1300 MWD while operating in a critical situation.

Analysis on core life time estimation has been completed based on previous usage history. BTRR has been operated for Radio-Isotope (RI) production as well as to perform different research works from 1987. RI production was continued up to 2008 and then it became suspended due to an incident occurred in DCT. After that from 2009 BTRR is mainly operated for various nuclear research only.

Table 5.9: Yearly Burn-up for Radio-Isotope production and various experiments

Year	MWd
2003	11.29167
2004	30.8375
2005	76.2600
2006	125.8833
2007	97.12667
2008	81.70083

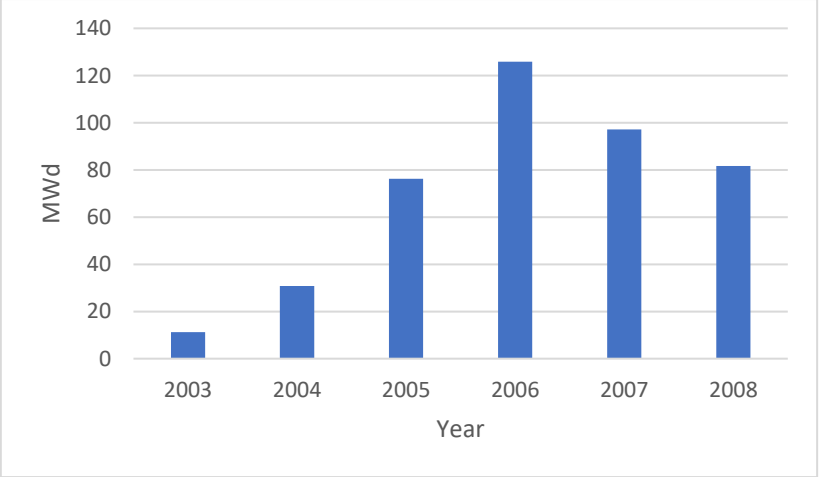


Fig. 5.12: Yearly Burn-up data from 2003 to 2008

Fig. 5.12 shows the burn-up from 2003 to 2008. During this tenure, BTRR was operated to perform various nuclear research and Radio-Isotope production. At this time total 423.10 MWd burn-up has been occurred with a yearly average of 70.51667 MWd . From the analysis it is found that with the current core configuration it is possible to operate the reactor up to 1300 MWd . If BTRR intend to operate in this frequency then within next 7 years core reloading will be mandatory.

Table 5.10: Yearly Burn-up for various neutronic experiments only

Year	MWd
2010	25.31
2011	9.09
2012	15.22
2013	21.10
2014	19.37
2015	19.48
2016	10.95
2017	18.39
2018	6.31
2019	4.55
2020	2.45
2021	0.19
2022	0.74

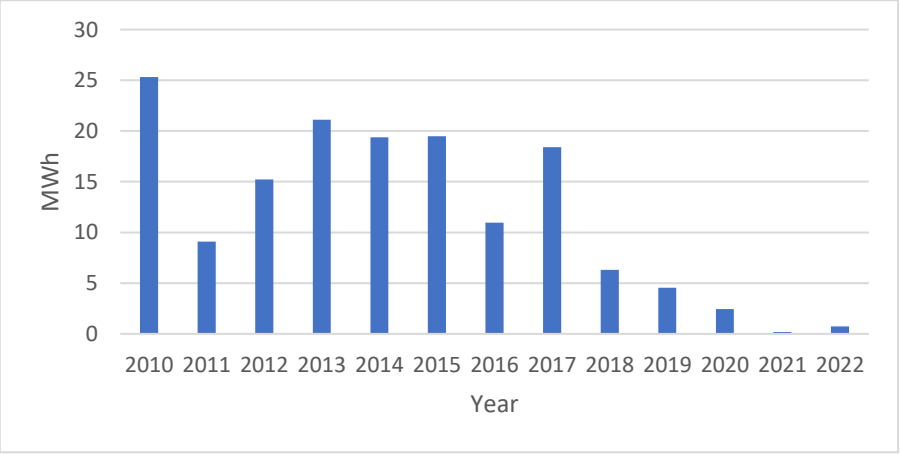


Fig. 5.13: Yearly Burn-up data from 2010 to 2022

Fig. 5.13 shows the burn-up data from 2010 to 2022. During this period, BTRR was operated to perform various nuclear research only. At this time total 153.20 MWd burn-up has been occurred. From 2018, modernization work has been started which reduce the operation hour of BTRR. From 2010 to 2017, the average annual burn-up was 17.369 MWd .

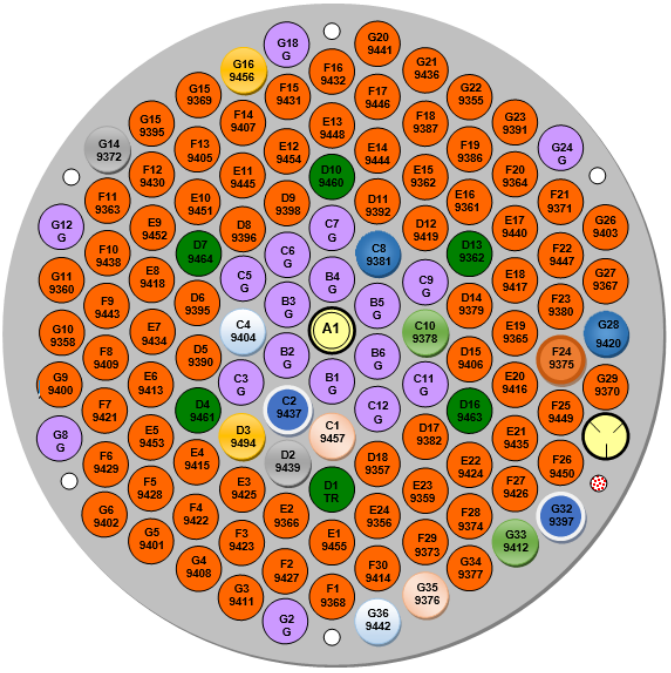


Fig. 5.14: BTRR core reshuffle pattern.

It can be stated that if BTRR intend to operate in this frequency then it is possible to operate the reactor for the next 20 years. It is possible to operate the reactor until the core excess reactivity fall below the value of about 5$ which is needed to

override the Cold-to-Hot reactivity change according to SAR. According to the analysis, BTRR can be operated up to 1300 MWD in total without reshuffling/refueling the core. Then the 1300 MWD burnt core has been reshuffled by replacing the maximum burnt fuel element with the minimum burnt fuel element, then the second maximum burnt with the second minimum and so on (for example C2 has been interchanged with G32, C4 has been reshuffled with G36 etc., illustrated in Fig. 5.14.). In this reshuffle scheme, about 0.6$ core excess reactivity was regained which corresponding core life has been stated in table 5.11.

Table 5.11: Change of life time with the reshuffle core

Core name	Estimated core life
Current Core	1300 MWd
Reshuffled Core	1370 MWd

Such reshuffle may increase the life time of BTRR slightly (approximately 70 MWd in addition).

5.7 Change of control rod position

BTRR has six control rods having 15-inch boron carabid layer which actually plays as the neutron absorbing element. These 15 inches has been divided in to 999 steps for easy understanding of reactor operator regarding the present position of the control rod. Table shows the control rod position during full power operation at zero burn-up as well as at 800 MWd burn-up condition.

Table 5.12: variation of the control rod positions due to loss of excess reactivity

Burnup (MWd)	Transient	Shim-1	Shim-2	Shim-3	Shim-4	Regulating
0	497	492	492	488	492	502
800	542	547	547	549	551	555

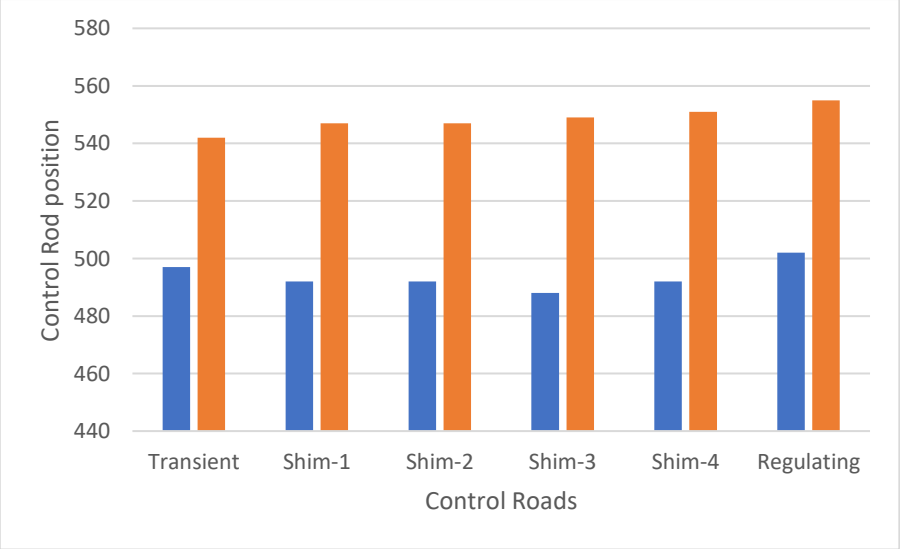


Fig. 5.15: Control rod position in different burn-up condition

The variation of the control rod positions is occurred due to loss of excess reactivity with respect of burnup which has been shown in Table 5.12 and illustrated at Fig. 5.15.

5.8 Summary of the discussion

A computational model of the initial critical core of the TRIGA Mark-II research reactor in Bangladesh is developed using a combination of the computer programs WIMS-D/4 and TRIGLAV. This concept is supported by an essential experiment on the first initial core configuration and uses LEU fresh fuels. The criticality experiment demonstrates that following core loading with fuel elements of the 50-LEU type, the initial core achieves initial criticality. In addition, operational core analysis has also performed. The projected values of the multiplication factor by the TRIGLAV & TRIGAP codes, which are in good agreement with the experimental data for the initial critical experiment, demonstrate that the TRIGA physical model is accurate enough to replicate the initial critical experiment, excess reactivity, burn-up calculation etc.. This analysis will be essential for enhancing the safety and economical use of the 3 MW TRIGA Mark-II research reactor at AERE, Savar, Dhaka, Bangladesh.

CHAPTER 6
CONCLUSION AND RECOMMENDATION

6.1 Conclusions

A computational model of the initial critical core of the TRIGA Mark-II research reactor in Bangladesh is developed using a combination of the computer programs WIMS-D/4 and TRIGLAV. This concept is supported by an essential experiment on the first initial core configuration and uses LEU fresh fuels. The criticality experiment demonstrates that following core loading with fuel elements of the 50-LEU type, the initial core achieves initial criticality. The projected values of the multiplication factor by the TRIGLAV code, which are in good agreement with the experimental data for the initial critical experiment, demonstrate that the TRIGA physical model is accurate enough to replicate the initial critical experiment and excess reactivity. Due to the BTRR's aging, an expert group conducted a periodic safety review, which determined that the BTRR can only operate at a maximum output of 2.4 MW. For the core excess reactivity, the operational core has been simulated using the TRIGLAV algorithm, and estimates for individual fuel burnup and ring-wise burnup have been made. When comparing the computed burnup result to the burnup result of the MVP-BURN function, a very good alignment can be found. The estimated result has been compared with the genuine protracted operational data in the scenario of core excess reactivity. The difference between the estimated additional reactivity and the real data using the TRIGLAV code is only 3.51%. The expected calculation result shows that the reactor can run at full power for up to up to 1300 MWd burnup. This burn-up calculation has some non-linearity, although it has a very good alignment with MVP-BURN code. The lack of TRIGA fuel is currently a key operational limitation for the BTRR. The reactor can be used for its intended function in full force, according to the projected burnup. As the burned core has not yet been reshuffled or reloaded, the computed burnup data can be used to do so. This code can be utilized to teach students about core management and demonstrate it to them, both of which will significantly aid in the development of human resources in the field of nuclear science and technology.

6.2 Recommendations for future works

- Prepare a specific fuel reloading pattern for BTRR core.

- Neutron flux calculation may be done which may help the reactor user group to compare on different experiments.
- BTRR core reloading pattern study with 9 fresh fuel, need to be completed for effective life time extension.

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APPENDIX-A

Journal Publications

Hasan, MR, Khan, MJH & Mollah, AS, 2023, Calculation of LEU fuel burnup and core life time estimation of BAEC TRIGA Research Reactor Using 2D-TRIGLAV Code', The International Journal of Science and Technoledge, Vol 11 Issue 1, p.08-13, January 2023.

Conferences Publications

Hasan, MR, Khan, MJH & Mollah, AS, 2022, 'Analysis of Initial Criticality and Neutronic Parameters of Operational Core of 3 MW TRIGA Mark-II Research Reactor Using TRIGLAV Deterministic Code', International Conference on Energy and Power (ICEP) , PID-10, p.006, MIST, Mirpur , Bangladesh, 11-13 December.

Hasan, MR, Khan, MJH & Mollah, AS, 2022, 'Analysis of Criticality of Operational Core of BAEC TRIGA Research Reactor using Deterministic Code TRIGLAV', International Conference on Physics for Sustainable Development and Technology (ICPSDT-2022), NHP-03, p.67, CUET, Chattogram, Bangladesh, 22- 23 January.

Hasan, MR, Khan, MJH & Mollah, AS, 2023, Calculation of Individual Fuel Element Burnup and Detection of Hottest Fuel Element of BAEC TRIGA Research Reactor Using TRIGLAV Code, National Conference on Physics, 9-11 March.