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Lattice Calculations and Power Distribution for Nigeria Research Reactor-1 (NIRR-1) using Serpent Code

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Abstract: *Nigerian Research Reactor (NIRR-1) is a tank-in-pool Miniature Neutron Source (MNRS) reactor type that originally uses 90.3% enriched UAl_4-Al which is a highly enriched uranium fuel as the source of neutron. The recent conversion of this reactor core to low-enriched uranium with 13% enriched UO_2 may significantly affect many neutronics behavior in the core and hence power output. Hence, the need to investigate the likely changes in the neutronic parameters that the fuel replacement may cause. This study investigates the static neutronic behavior in the reactor core lattice arrangement and reactor power distribution using "Serpent" Monte Carlo reactor Physics burnup simulation code module computations which include multi-group diffusion and flux-power reactivity equation for k_{eff} eigenvalue and diffusion solution respectively were carried out to determine the effect of lattice geometry and mesh on the power distribution in the core. The result shows that the power distribution is slightly affected by the macroscopic group constant, heat transfer, fluid temperature, coolant density and fission reaction stability. The computational results of power distribution in arrayed fuel core and power distribution per neutron energy generated in the reactor indicate a uniform distribution. The result is important in determining the effect of fuel change and core configuration and in the overall operational safety of the reactor.*

Keywords: NIRR-1, Lattice calculation, power distribution, fuel, Serpent code

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1.0 Introduction

Nigerian Research Reactor- 1 (NIRR-1) is a miniature neutron source reactor (MNSR) type with a nominal thermal power of 30 kW under steady-state conditions (Jonah *et al*, 2012). It is designed by the China Institute of Atomic Energy (CIAE). The reactor achieved its first criticality on 3rd February 2004 and has been utilized for Neutron Activation Analysis (NAA) and limited radioisotope production for over 15 years (Balogun *et al*., 2005, Jonah *et al*., 2005, Jonah *et al.*, 2006). It is filled with about 1.4 kg low enriched Uranium (LEU) and has a limited core excess reactivity of 3.94 mK measured during onsite zero power and criticality experiments (SAR, 2021). The reactor can operate at full power for a maximum of 3 hours mainly due to its large negative temperature feedback effects. Its design was tanked in the pool with 347 active fuels 3 dummy elements and 4 tie rods arranged in 11 concentric circular arrays as its core. The reactor core is surrounded by a beryllium reflector of 10 cm thickness. The major work of the beryllium reflector is to reduce the reactivity interference of irradiated samples and neutrons (Jonah, 2011). The inner and outer irradiation channels of NIRR-1 are located in and just after the beryllium reflectors (Aliyu *et al.*, 2017). As built fuel of NIRR-1 is 13% enriched UO_2 clad in zircology.

Neutron thermalization/moderation is achieved by the light water as a coolant and moderator. Detailed description of NIRR-1 facility has been reported elsewhere (Ahmed *et al.*, 2002, Balogun, 2003, Jonah *et al*., 2006). Heat removal from the core of this reactor is achieved by water recirculation and natural convection. The coolant flow is at transitional state from laminar to turbulent. The characteristic of the coolant circulation was ascertained in the study by (Guo, 1983, Shi, 1990) and thermal hydraulics simulation tests (Zhang, 1984).

Previous literature (Ahmed *et al.*, 2008; Balogun, 2003) indicate that fuel power distribution in the fuel lattice depends on the types and properties of the fuel material, control rod responses and reactivity compensation. It was also noted that reactor core assembly affects power distribution under the influence of moderator type, metallic reflectors and low critical mass (Ahmed *et al.*, 2011; Yahaya *et al.*, 2017). The reactor criticality and power output is determined by the effective multiplication factor (k_{eff}). In the analysis of reactivity-initiated accidents (Musa, 2023), it is noted that the NIRR-1 core becomes prompt critical when the inserted reactivity is larger than the delayed neutron fraction. This effect caused energy deposition in the core to rise rapidly until the negative fuel temperature feedback terminated the rise in the energy within a tenth few of a few seconds. The energy produced increased at an exponential rate with a corresponding incease in the reactor power (Musa, 2023).

Nuclear reactor cores are hexagonal or rectangular lattices of fuel pins assemblies, where each assembly is itself a lattice of fuel, control, and instrumentation pins, surrounded by moderator material that moderates fission neutron energy and carries away heat (Timothy and Jain, 2012). Fuel pins are normally cylindrical and contain uranium-based fuel, absorbing material for controlling the nuclear chain reaction, or instrumentation. Assemblies vary from one reactor to another depending on the degree of enrichment in the fuel material, type and function of the control rod, or other parameters (Timothy and Jain, 2012).

The study presented is focused on the simulation of NIRR-1 lattice geometry and mesh to determine their effects on power distribution. A low-enriched uranium fuel (UO_2) is used in the study. The SERPENT model employed in the study used actual reactor and fuel design data to simulate the scenarios. The obtained results are benchmarked using the NEA nuclear data. The following section of this paper describes the theory of fission chain reactions applied to nuclear reactors, the code used in the simulation and the calculation setup. The results obtained for the power distribution are presented and discussed in the last section.

2.0 Theory of fission chain reaction

Nuclear reactors work on the principle of fission reactions. Fission reaction is a chain reaction induced by neutrons in heavy nuclei like Uranium which results in the production of lighter nuclides and more neutrons. The reaction is accompanied by the release of energy of about 200 MeV per event. The neutrons emitted by the fission nuclei can be used to induce more fission reactions that may result in inducing a chain of fission events. The chain reaction can quantitatively be described in terms of multiplication factor (k) which is defined as the ratio of number of fission neutrons in one generation to the number of fission neutrons in the preceding generation. The k for theoretical infinite reactor is expressed in Equation 1 as (John and Anthony, 2001; James and Louis, 1976)

$$k_{\infty} = \frac{n_{FG}}{n_{FPG}} \quad (1)$$

where n_{FG} is the number of fission neutron in generation , n_{FPG} is the number of fission neutron in preceding generation.

If k is greater than 1, it means that the fission neutrons are increasing from one generation to another and the energy released by the fission reaction is increasing with time. The reactor is said to operate at the supercritical level. When k is less than 1, the fission neutrons decrease with time and the reaction is said to be subcritical. However, if k equates to 1, then the reaction proceeds at a constant rate which means that energy is released at a steady level and the reaction is said to be critical (James and Louis, 1976). Nuclear reactors are designed to operate at a critical level. This is achieved by choosing a particular reactor material composition and configuration, and then readjusting the k value until is unity (John and Anthony, 2001).

Equation 1 will lead to oversimplification when considering a finite reactor, in which some of the neutrons are lost from the system through leakage or absorbed in non-productive capture. Therefore, an effective multiplication factor is proposed to describe the state of criticality. The effective multiplication factor (k_{eff}) is expressed in Equation 2 (John and Anthony, 2001; Shannon, 2018):

$$k_{eff} = \frac{P(t)}{L(t)} \quad (2)$$

where $P(t)$ is the rate of neutron production in the reactor and $L(t)$ is the rate of neutron loss (absorbed and captured) in the reactor. The production and the loss rates will change with time due to fuel consumption.

The neutron lifetime (l) is defined by Equation 3.

$$l \equiv \frac{N(t)}{L(t)} \quad (3)$$

where $N(t)$ is the total neutron population in the reactor at a time t and $L(t)$ is the rate of neutron loss. Equation 3 is important in studying the neutron population behaviour in a reactor (John and Anthony, 2001). The neutron population at any time t, assuming an initial number of neutrons (N_0) in the reactor at time $t = 0$ is given by Equation 4.

$$N(t) = N_0 \exp\left[\left(\frac{k-1}{l}\right) t\right] \quad (4)$$

Equation 4 indicates that the growth or decay of the neutron population in a nuclear reactor is an exponential growth law. The reactor period T which is expressed in Equation 5 (John and Anthony, 2001).

$$T = \frac{l}{k-1} \quad (5)$$

Equation 5 shows that as the multiplication factor reaches unity, the reactor period T approaches infinity which corresponds to reactor power or time-independent neutron population. The value of k will determine whether the reaction is controllable or not and hence the power output.

The reactivity, ρ which is the measure of the deviation of the reactor from the critical level is expressed as Equation 6 (Shannon, 2018).

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \quad (6)$$

A positive reactivity indicates a supercritical reactor a zero reactivity indicates a critical reactor, and a negative reactivity indicates a subcritical reactor (Shannon, 2018). The temperature coefficients of reactivity α_T

which characterises the reactivity feedback due to variation in temperature T of the core is given by Equation 7 (James and Louis, 1976).

$$\alpha_T = \sum_j \alpha_j \equiv \sum_j \frac{\partial \rho}{\partial T_j} \quad (7)$$

The above equation describes the effect of temperature feedback on reactor stability. For reactors with a positive α_T, the increase in temperature would cause an increase in reactivity (ρ) and hence the power output. In this case, the reactor would be unstable with respect to the variation in temperature. However, if the reactor possesses negative α_T, then it is clear that an increase in temperature would cause a decrease in reactivity and hence a decrease in the reactor power which tends to stabilise the reactor power (James and Louis, 1976). Thermal process in the core are characterised by different time behaviours. It is observed that the temperature response changes in the reactor is rapid to any change in power level but it takes an appreciable amount of time to transfer this thermal energy to the coolant which means the temperature response of the coolant is much slower compared to that of core.

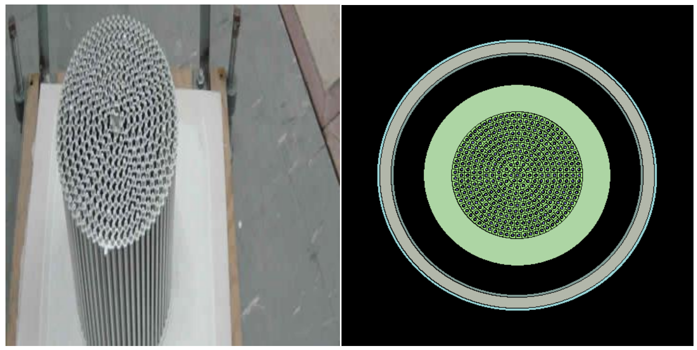
3. 0 Materials and Methods

3.1 Simulation condition set-up

Serpent code was used to calculate single assembly lattice physics data and homogenized few-group cross-section data that are used in the study. It is a Monte Carlo transport code designed for a reactor core calculation. It was written in an ANSI-C programming language and operated on a Linux environment. The source code of this software was compiled to runs using GNU Make utility files. Whereby, all execution commands or operational assignments for running a nuclear reactor simulation programs used to request a higher computational system of the highest order capacity. Thereafter, it produced a graphical output solution of the simulation with an open-source GD graphics library. Finally, the study used 'sss' as an executable command to runs the input file that were successfully compiled. All output for this nuclear reactor analysis, modelling and simulation were extracted using an OCTAVE platform. Technically, this code performed all of the intended neutronics analysis. The code required only to know the location of the input file that contained all the necessary information about NIRR-1 specification. These specifications include the description of the specific geometry, cladding, fuel materials, and source neutron population of the reactor. The choice of the cross-section library was necessary for all calculations to be completed noted in the present study and earlier study by (Leppänen, 2009). The input to the generic reactor was done on a 10 cm scale. The geometry simulation was made on a 1m grid cell, which is a typical node size for SERPENT usage. Detailed description, formulation theories and calculations of SERPENT are described elsewhere (Leppnen, 2016; Leppnen *et al.*, 2013; Leppnen and Isotalo, 2012).

3.2 Programming aspect of the simulation

The fuels of the Nigeria Research reactor are arranged in a circular form of eleven concentric cycles as shown in Figure 1 (a). The fuel elements are all enriched uranium alloy extrusion clad with aluminum. They are arranged in ten multi-concentric circle layers at a pitch distance of 10.95 mm. The analysis the original design of the NIRR-1 reactor was adopted in designing our inputs program and different approaches in simulating the reactor was followed for consideration and justification purpose. These elemental components used as the nuclear fuel models for core conversion were selected from the software. The core of the reactor with 350 lattice positions and 374 fuel elements were modelled as accurately as possible in the SERPENT code as shown in Figure 1 (b). The size of the computational domain for the reactor core was 230 mm × 230 mm × 230 mm which gives a total number of 12167000 cell control volume and this is the typical size of the NIRR-1 reactor core. The model is used to simulate input and output data for analysis.



(a) **(b)**

Figure 1: Models developed for NIRR-1 (a) Original NIRR-1 array fuel core (b)Serpent 1.1.7 NIRR-1 lattice in arrayed fuel core

4.0 Results and Discussions

The power distribution in arrayed fuel core of the reactor is presented in Figure 2. The figure indicates a uniform neutron flux in the lattice region and hence uniform power distribution. It should be noted that in each nuclear reactor, there is a direct proportionality between the neutron flux and the reactor thermal power. It is therefore essential to maintained a balance between the rate of heat generation and removal to prevent the temperature rise and power surge which if not checked could lead to the melting of the fuel and result in the failure of reactor core and other structural materials. The heat removal in the reactor is maintained by reactor safety systems including control rod and coolant. In reactor physics, the thermal hydraulics of nuclear reactors described the concept involving the coupling of heat transfer and fluid dynamics to accomplish the desired heat removal rate from the core under both normal operation and accidental conditions. It should be mentioned here that the analysis indicates a slight change in power distribution with the change in macroscopic group constant, coolant density and temperature, heat transfer, fission reaction stability and the response of the control rod. In most cases, there is a direct relationship between these parameters and the core power distribution and output in line to theoretical prediction. The response of the control rod was observed to be directly related to reactor start-up, shutdown, reactivity compensation and power regulation.

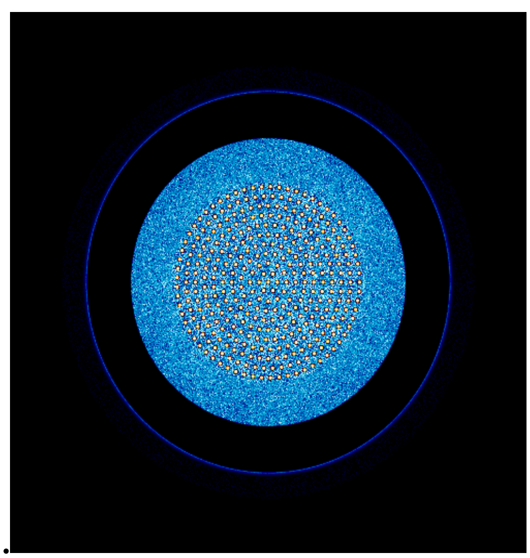


Fig. 2: NIRR-1 power distribution in arrayed fuel core

In Fig.3, a spectrum of the power distribution per neutron energy generated by the NIRR-1 reactor with the low-enriched uranium fuel is depicted. The figure showed an interesting and good distribution of power for all the lattices in agreement with the power distribution shown in Figure 2. The central dark blue colour region of the spectrum indicated that the power in the core region is evenly distributed. Even though a control rod is within this region and is essentially responsible for controlling the power and the flux in a symmetrical form. The results indicate that the reactor work normally with the criticality label, which is numerically one. The subsequent variations in colour that it to say as the intensity of the blue colour reduces or becomes pale indicated the radial movement of the neutron in the lattices. Moreover, the spectrum shows that the insertion of the control rod into the core centre of the NIRR-1 reactor and its desertion out of it was excellent. In addition, the spectrum shows that the reactor can accept to use or operate with un-irradiated low enriched uranium fuel for it started up long period operation and shut down as well. The spectrum gave a good view and remained the same under all ranges of neutron energy which implies that the reactor can efficiently work even at very low neutron populations. By good choice of neutron source number and subcritical multiplication process, the nuclear instrument would be able to monitor and detect the neutron populations throughout the reactor's operational period. Therefore, this was done here in this study. Indeed, in this analysis neutron source installation was done appropriately which is why the power distribution was successfully achieved and controlled. The uncontrollable reactor power distribution would raise the temperature of the reactor and also lead to a criticality problem that cannot be detected or controlled with just rod withdrawal.

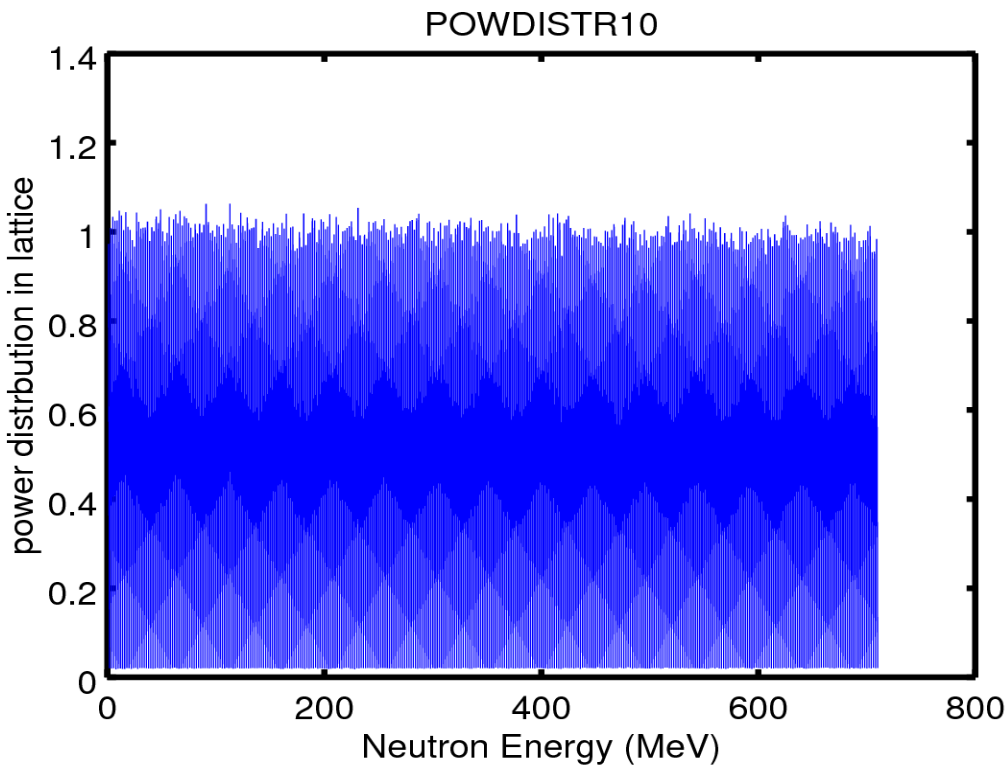


Fig. 3: Spectrum of the power distribution per neutron energy

The outline border between each two colours, grid boxes and pixel meshes is referred as the waiting period between every time the rods were incrementally withdrawn. At each waiting period, the neutron population were increased through subcritical multiplication. On this basis and with a sufficient magnitude of the source neutron with less than one value of the $K_{eff,}$ the chain reaction is automatically sustained even if are not self-sustainable. Therefore, the leaked out and neutron lost through absorption were fully compensated. Moreover, the NIRR-1 was designed as a thermal reactor, which used water as its moderator and reflector as well but it caused no effect on this chosen fuel material chemically. Predictably, the power distribution shows uniformity in temperature, fuel depletion good neutron flux shaping and choice of poisons.

4.0 Conclusion

The NIRR-1 reactor core lattice was modelled and simulated using Monte Carlo neutron transport code called Serpent. The analysis shows that there is a uniform power distribution across the core of reactor with the same symmetrical effect. The results of power distribution in arrayed fuel core and spectrum of the power distribution per neutron energy agrees closely with one another by showing a uniform power distribution. Indeed, the output of the reactor operation was reasonable even with low energy hence neutron population agrees with the choice of our neutron source and the operable multiplication factor.

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Compliance with Ethical Standards Declarations

The authors declare that they have no conflict of interest.

Data availability

All data used in this study will be readily available to the public.

Consent for publication

Not Applicable

Availability of data and materials

The publisher has the right to make the data Public.

Competing interests

The authors declared no conflict of interest.

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Authors' contributions

All the authors participated equally in all sections of the study.