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SERPENT CODE POINT KINETIC PARAMETERS ANALYSIS: A NEXUS FOR NIGERIA RESEARCH REACTOR – 1 CORE CONVERSION USING ENRICHED URANIUM DIOXIDE FUEL

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ABSTRACT

The International Atomic Energy Agency (IAEA) requires that all test and research reactors operating on Higher Enriched Uranium (HEU) should be converted to Low Enriched Uranium (LEU) for safety and security purposes. Nigeria having a Miniature Neutron Source Reactor (MNSR) has been long interested in fuel technological research not just to develop the area but also to meet with resolution on the nuclear treaty set out by the global nuclear regulatory body. In this study, reactor kinetic parameters such as effective delayed neutron fractions, prompt neutron lifetime as well as mean neutron generation time were analysed for Nigerian Research Reactor-1 (NIRR-1). Serpent Monte Carlo code 1.17 is used in the analysis. For delayed neutron parameters determination, we used fission probability iteration under one averaged generation time and neutron population rate. The calculated values for delayed neutron were recorded as analogue prompt and implicit prompt neutron lifetime, reproduction time and emission time are in the order of 3×10^{-7} (s), in agreement with the calculated data from the nuclear data libraries and some literature. The result for delay neutron fraction and other time-based parameters support the fuel core conversion for NIRR-1. The computational and pictorial results obtained from Serpent code simulation described well the transient behavior of the delayed neutron in this reactor. The analytical results also spelled out the relevance and compatibility of low enriched uranium dioxide fuel over higher enriched type. The result of this study conforms with other results obtained from similar reactors but with different Monte Carlo codes and with higher enriched uranium.

Keywords: Delayed Neutron, NIRR-1, Point Kinetic Parameters, Serpent Code, Uranium dioxide

INTRODUCTION

The delayed neutrons are the kinematic neutron in any system where neutrons are present. Determination of the neutron kinetic parameters is important in reactor physics calculations because of its roles in reactivity analysis, vacuum fractions and Doppler Effect analysis as well as in reactor control and safety analysis (Heba, 2021). Reactor kinetic parameters are the effective delayed neutron fraction, prompt neutron lifetime and neutron generation time. The effective fraction of delay neutron plays a key role in the control of reactivity value of control bars, Doppler effects, vacuum fractions and in the control of many safety parameters in reactor (Ott and Neuhold, 1985). It is important that reactor have delay neutrons. Without delay neutrons, the reactor power will increase to a higher magnitude and within a short time that significant damage may results (Svetozar *et al*, 2008). The effective delay neutron fraction (β_{eff}) determines the time-dependent response of the nuclear reactor. A smaller value of β_{eff} indicates that a larger fraction of the fission neutrons appears as the prompt neutrons, hence the kinetic response of the reactor is faster. On the other hand, a larger value of β_{eff} indicates a small fraction of fission neutrons appears as the prompt neurons and the reactor core has a lower response (Gehin *et al*, 2004). The prompt neutron lifetime is a measure of the time taken for changes in the reactor core multiplication factor to affect the neutron population. It is related to the neutron generation time and has an impact on the time scale of the core response to reactivity changes (Heba, 2021). Neutron lifetime is defined as the average time between the emission of a fission neutron and its absorption in the active part of the reactor core (Dan, 2010). Another important parameter that characterizes the time behaviour of neutrons is prompt neutron generation/reproduction time. This parameter is important in determining the dynamic response of a nuclear reactor (Hetrick, 1997, Salawu, 2020). It is the mean time required for one generation of neutrons to produce another generation of prompt neutrons or precursors due to fission process (Lamarsh and Baratta, 2001).

A nuclear reactor falls into subcritical level when its often-used multiplication factor decreases below the standard value. Thereafter, it leads to a sudden change in the external source strength and perhaps the reactivity of the reactor. As this change occurred, the neutron population is automatically forced to move to a new station. The determination of these point kinetic parameters is considered the best solution for analysing the criticality condition of a reactor. Moreover, point kinetic parameters are used for the prediction of reactor's power. These parameters are directly related to the reactor's control rod insertion and reactivity. Any deviation noted in the solutions of point kinetic parameters is a result of the sudden change in flux shape (Jonah *et al*, 2012; Azande, 2011, Ibrahim, 2015). Similarly, when the neutron source transient trip data was over predicted for negative insertion, it would lead to wrong results in point kinetic parameters (Bergenas and Scheinman, 2008; Azande *et al.*, 2010).

The typical responses of delayed neutron and its related factors are solved using point kinetic equation. The calculation of point kinetic parameters by Serpent code is based on iterated fission probability (IFP) (Stacey, 2018; Stacey, 2007). The related factors include reactivity (ρ), neutron power, multiplication factor (k_{eff}), reactor core lifetimes and periodic movement of control rod. The kinetics parameters are observed to be arbitrary because they depended on the weighting functions and time has no physical meaning in the relation. The approximated kinetic parameters of reactors are assumed under constant flux shape, space-time dependence, energy time dependence, first-order perturbation theory for reactivity calculation, and inhomogeneous balance equation for independent neutron source reaction (Leppänen *et al.*, 2014).

This study focused on the determination of five-point kinetic parameters for Nigerian Miniature Neutron Source Reactor by simulating three of its sections, which are lattice, core assembly and pin-cell in line with the previous studies by Jonah *et al* (2009a); Jonah (2011); Mengjiao *et al.*, (2017). The parameters of interest here include estimated analogue prompt neutron lifetime, reproduction time, implicit prompt neutron lifetime and reproduction time and then the average delayed neutron life/emission time.

Serpent 1.1.7 model description

"Serpent" is a three-dimensional software used in determining the point kinetics parameters of nuclear reactors. It is a Monte Carlo-based code that has been specifically designed for reactor physics burn-up simulation and other reactor physics applications. The code always started by generating the homogenized multi-group constants for which the point kinetic parameters calculations would be done (Bradley *et al.*, 2007; Jonah and Ahmed, 2016; Odoi and Gbadago, 2017). The code is capable of simulating and predicting what is happening internally in the reactor, like burn up in the pin-cell and in the reactor core within a reasonable time as was the case for other deterministic lattice codes. The software stood alone as an application to simulate entirely the fuel depletion or core assembly burn-up of any given reactor within the shortest possible time. The Message Passing Interface (MPI) was installed to help in reducing the overall calculation time for it adopted the parallelization mode (Ghasabyan, 2013; Jaakko, 2013; Leppänen *et al.*, 2015). For the Serpent code to run successfully, an open-source library on Graphics Data (GD) is normally installed on the computer to be used for the analysis (Jonah, Ibikunle and Li, 2009b; Leppänen and Viitanen, 2012). Though, it was reported that these parameters can functionally be determined with a compiled source code without the use of GD and MPI (Briesmeister, 2000; Jaakko, 2013). Nevertheless, a standard Graphic C-Compiler (GCC) was incorporated because the Serpent 1.1.7 source code needed it to be built successfully. By default, the GCC was not part of the necessary conformant for the analysis but it was the most standard and fastens open-source compiler that can support the analysis compared to Borland C, Turbo C and DJ's GNU Programming Platform (DJGPP) for Intel and above, IBM PC compatible that supports DOS operating system.

The Serpent code has tallies for setting up the integral reaction rates. The code has a combined tracking routines method with which all lattice physics calculations could be done efficiently. The two tracking routines were surface-to-surface (a conventional ray-tracing) and the Woodcock delta-tracking method. Serpent has the advantage of using two methods in solving depletion equations (Bateman) during pin cell burn-up calculation. The two methods are Transmutation Trajectory Analysis (TTA) and Chebyshev Rational Approximation Method (CRAM). In the present study, the CRAM was used based on its matrix exponential solution (Leppänen, 2009a; Jaakko, 2013; Korkmaz and Agar, 2014; Herrero *et al.*, 2016; Kępisty *et al.*, 2017). Furthermore, the cross-section data libraries: radioactive decay and fission yield and also thermal scattering libraries are all used by the Serpent. These libraries are code-named as JEF-2.2, JEFF-3.1, JEFF3.1.1, ENDF/B-VI.8 and ENDF/B-VII. The Serpent code is highly reconstructed software that all the begotten cross-section information from the ACE format data libraries could be used diligently. The code can speed up all the calculations including those that involved a single unionised energy grid (Leppänen, 2009b; Jaakko, 2013). The code considered and merely pivoted all physical interactions on kinematics classical collision and reaction laws of evaluated nuclear data of fission. Standard physical interactions include decay constants, homogenized reaction cross-sections, infinite multiplication factors, effective multiplication factors, assembly discontinuity factors, delayed neutron fractions, diffusion coefficients, scattering matrices, and precursor group. The code used geometry and mesh descriptions follows that of the standard Monte Carlo approach on the bases of the reactor's cells, surfaces, and universes (NISA, 2011). Serpent code could be used for other purposes such as educational training, group constant generation, demonstration, validation of other deterministic codes, and fuel cycle studies (Goorley *et al*, 2003; Ramos *et al*, 2004; NISA, 2011).

Simulation Condition set-up

For the easy running of the code in this study, a single and several input files were used for the interaction between the user and the code itself. While various output files were used in describing the Serpent 1.1.7 code running format, a line interface command was used in sending all running and executing commands. The main and optional input-files names are used as general syntax commands for executing all calculations in the analysis. Generally, the input file of the code contained a standard text file that usually described the nature of the input. Then the inputs were divided into several main files for easy referral. The available input file options used in carrying this simulation include run, run in parallel mode, calculate, generate, disperse the pebble or particles that were generated randomly and distributed, check the calculated volume used in random points, test the geometry of the cell, surface and universe using the tracked neutron that was randomly sampled, track, replay, exit for quitting the on-going running simulation from using the previous calculated random number seed and then print version information. A replay input file option is also provided that will force the code to use again the same random number seed as it was in the previous run. The seed was taken from system time and written in a separate seed file and or set manually for further use.

The geometry and its plotter were debugged using an input file named geometry test option. The code used to track the sampled neutron randomly across the geometry and checked whether the neutrons were within the correct cells or not and correctly defined or not in the input programme for the simulation. The code can spot the input errors using a similar input file option. The input file option for checking the volume was used for verifying and ensuring whether the volumes used were correct for all neutronic parameterized calculations. For statistical accuracy required in the result of this study, a large random point of at least 1,000,000 was used. The particle data, packing fractions, volume, and dimensions of the reactor lattice cells, geometries, and pin cells were correctly defined in the code. Similarly, the command 'fed' was used to calculate the cell's volume for the simple lattice geometries of NIRR-1 whereas a cell that has complicated geometries, their volumes were calculated manually.

The running time of the Serpent code varied depending on which parameter of the physical interaction is calculated. In a two-dimensional problem like infinite-lattice calculations, which involved 3 million neutrons, the simulation time is about 5 to 20 minutes using a single-processor of 3 GHz on a PC Workstation. Secondly, a completed running time for assembly burnup calculation of 40 steps with predictor-corrector calculation on a Light Water Reactor with actinide and fission product nuclides of more than 250 and 65 separate depletion zones and 3 million neutron histories per transport cycle was found to be 15 hours on the same computer of a single-processor of 3 GHz on a PC workstation (Goorley *et al*, 2003; Ramos, Ferrer *et al*, 2004; Meggitt, 2006; Fensin *et al*, 2009; Jonah *et al*, 2009b; NISA, 2011; Jaakko, 2013). The present study used Linux-based operating system which was installed on CORE i5 DELL personal computer (PC). Macintosh, Mac OSX operating systems and UNIX workstations were all used as recommended by the software developers.

ANALYSIS RESULTS

The point kinetic parametric results obtained from the suggested low enriched uranium fuel analysis done on Nigerian MNSR obtained from Serpent code simulation are presented in Table 1. The parameters are due to kinematic neutron behaviours of fuel components. These behaviours were observed in three regions of the reactor such as lattice, core assembly, and in pin-cell regions. These parameters described the transient behaviour of neutron in NIRR-1 when operating within the limited range of multiplication facto

Table: Point kinetic parameters for Nigerian MNSR fuel analysis

Parameter (time-dependent in seconds)	Lattice	Core assembly	Pin-cell
Analog estimated prompt neutron lifetime	3.25×10^{-7}	3.36×10^{-7}	3.00×10^{-9}
Analog estimated reproduction time	3.14×10^{-7}	3.30×10^{-7}	3.16×10^{-9}
Implicit estimated prompt neutron lifetime	3.25×10^{-7}	3.37×10^{-7}	4.72×10^{-8}
Implicit estimated reproduction time	3.14×10^{-7}	3.30×10^{-7}	4.97×10^{-8}
Average delayed neutron emission time	1.08×10^{1}	1.04×10^{1}	0.94×10^{1}

In Table 2, the results of five-point kinetic parameters at the lattice region of the NIRR-1 obtained using four different nuclear data libraries are presented. The libraries were ENDFB7, ENDFB68, JEF22 and JEFF31.

Table 2: Point kinetic parameters of NIRR-1 obtained using different libraries

Parameter (time dependent in seconds)	ENDFB	ENDFB68	JEF22	JEFF31
Analog estimated prompt neutron lifetime	3.26×10^{-7}	3.23×10^{-7}	3.33×10^{-7}	3.24×10^{-7}
Analog estimated reproduction time	3.14×10^{-7}	3.12×10^{-7}	3.16×10^{-7}	3.15×10^{-7}
Implicit estimated prompt neutron lifetime	3.26×10^{-7}	3.24×10^{-7}	3.33×10^{-7}	3.24×10^{-7}
Implicit estimated reproduction time	3.14×10^{-7}	3.13×10^{-7}	3.17×10^{-8}	3.24×10^{-7}
Average delayed neutron emission time	1.04×10^{1}	1.05×10^{1}	1.12×10^{1}	1.29×10^{-7}

DISCUSSIONS

The calculated values for point kinetic parameters of NIRR-1 nuclear reactor obtained from the Serpent code simulation (Table 1) under one averaged generation is compared with the calculated value of the delayed neutrons (l_d) obtained from the same reactor but using different neutron data libraries (Table 2). It can be seen clearly that the two data closely agree with each other signifying the preference of low enriched uranium dioxide (UO_2) as fuel for NIRR-1 over the higher enriched uranium fuel. The data also indicates that the reactor core conversion is operable and feasible. The calculated NIRR-1 point kinetics parameters results obtained from this analysis indicated that the presence of associated delay neutrons in the system of NIRR-1 agrees with the reactor manufacturer's expectation. Indeed, the delayed neutron obtained eventually affected the typical responses of the kinematic neutrons of NIRR-1 in relation to its neutron lifetime, periodic movement of the control rod, reactivity (ρ), effective multiplication factor (k_{eff}) and neutron power. Even though the reactivity factor was used in reactor power prediction and underestimation when the reactor criticality was uncontrollably raised to a higher level or over predicted during negative insertion in the transient trip of the neutron source. The assumed separable time-dependent variables and constant flux shape was used for the approximation of the calculated point kinetic parameters of NIRR-1, which agreed and validated with approximated first-order perturbation theory. The first-order perturbation theory used separable time-dependent variables for reactivity calculations and balancing the inhomogeneous neutron reaction equation when using an independent neutron source. Since the system of the Nigeria research reactor attained its criticality when effective multiplication factor was at a lower range and the results obtained from the present study is in good agreement with various verified nuclear data libraries, hence, an indication of good precision of Serpent code simulation on the neutron behaviour of NIRR-1. The numerical values of the prompt neutron lifetime and generation time at the lattice and core assembly shown in Table 1 are in the order of 10^{-7} s. However, the values of these parameters at pin cell are in the range of 4.97×10^{-8} s to 3.16×10^{-9} s. The computed neutron lifetime in the present study is lower than that reported by Heba (2021) which is in the range of 1.3×10^{-8} to 9.6×10^{-9}. In addition, the observed neutron reproduction time of 3.14×10^{-7}s at both lattice and core assembly region of the reactor is below the value of 4.403×10^{-5}, $4.565\times10^{-5,}$ and 50.33×10^{-5} reported by Muhammad and Majid (2008); Farhan (2010) and Woodruff and Deen (1994) respectively. In the study by Housiadas (2000), a mean neutron generation time of 5.76×10^{-5} s is reported for the Greece Research Reactor (GRR-1). It was shown in the study by Luka *et al.* (2008) that the mean neutron reproduction time decreases with increasing fuel enrichment. In their study, the mean reproduction time varies from 2.8×10^{-5} s to 4.8×10^{-5} s for 30 wt. % or 8.5 wt. % fuel, respectively. The large difference between the results of the present study and that reported in the literature arises from differences in neutron nuclear data used and assumptions used in the software model. The prompt neutron lifetime mainly depends on the amount of enrichment of ^{235}U in the fuel as it is approximately inversely proportional to the average absorption cross-section of the fuel as shown earlier in the literature (Luka *et al.,* 2008). The effective delayed neutron fraction strongly depends on the core size and is less dependent on the fuel types and enrichment (Bretscher 1997, Luka *et al.* 2008). Therefore, there is a need for improvement of calculation methods for kinetic parameters and improvement of nuclear data.

The computed neutron kinetic parameters obtained from four different nuclear data libraries is presented in Table 2. It is seen clearly that the computed data for the four libraries agree closely with each other indicating only a minimal deviation from the mean. The analysis shows a deviation of 0.005 for ENDFB7, 0.035 for ENDFB68, -0.065 for JEF22, and 0.025 for JEFF31 respectively. Though the kinetic parameters result obtained in this study have an order of 10^{-7}, which is bit lower than the theoretical value of 10^{-4}, the presented data is within the general value reported in many literatures. The arbitrariness and insignificant deviation we have in the values of the calculated point kinetic parameters of NIRR-1 has no physical meaning because the parameters depend on the weighting functions, which is responsible for causing such deviation and sudden change in flux shape experimentally.

CONCLUSION

This study presents the numerical results of five-point kinetic parameters for NIRR-1 MNSR obtained with the Serpent simulation code. The calculated time-dependent point kinetic parameters are in the order of 3×10^{-7} (s), in agreement with the relevant literature. The analysis results indicate the suitability and compatibility of using low enriched uranium (UO_2) as fuel for NIRR-1 instead of its former higher enrichment uranium fuel. The finding of this study is useful in modelling the transient behaviour of the delayed neutron of research reactors especially the one with similar characteristics to NIRR-1 for core conversion technology.

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