Development and Validation of OpenMC Model for Criticality Safety Analysis of an MNSR-type Research Reactor

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ABSTRACT: The accurate prediction of criticality and neutronic safety parameters is essential for the safe operation, licensing, and continued utilization of research reactors. There is limited published work assessing the performance of the open-source Monte Carlo code OpenMC for compact-core Miniature Neutron Source Reactors (MNSR) such as the Nigeria Research Reactor-1 (NIRR-1). This study aimed to develop and validate a high-fidelity OpenMC model of the NIRR-1 for predicting key criticality and neutronic safety parameters. A detailed three-dimensional OpenMC model of the NIRR-1 was developed using the ENDF/B-VII.1 nuclear data library. The reactor geometry was explicitly modelled to include the active fuel pins, dummy pins, tie rods, control rod, beryllium reflectors, irradiation channels, fission chamber, reactor vessel, moderator, and pool water. Material compositions, geometry, simulation settings, and tally structures were implemented through the OpenMC Python API. Criticality calculations employed 100,000 neutron histories per batch over 100 batches, including 20 inactive batches to ensure source convergence. The validated model was used to determine the effective multiplication factor for fully withdrawn and fully inserted control rod positions, from which the core excess reactivity, control rod worth, shutdown margin, and effective delayed neutron fraction were calculated. The calculated core excess reactivity, control rod worth, shutdown margin, and effective delayed neutron fraction were 3.78 mk, 6.82 mk, 3.04 mk, and 7.99 mk, respectively. These values agreed with commissioning measurements within a deviation of less than 5% and satisfied the neutronic safety requirements for MNSRs. The study demonstrates that OpenMC provides an accurate and reliable open-source platform for predicting the neutronic safety characteristics of the NIRR-1 and supports its application in reactor safety analysis, operational support, and future studies of MNSR-type research reactors.

KEYWORDS: OpenMC, Nigeria Research Reactor-1, Neutronics, criticality, Validation, Miniature Neutron Source Reactor.

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I. INTRODUCTION

Monte Carlo–based neutron transport codes have been used for nuclear reactor physics calculations over the years as alternatives to deterministic codes. This is not unconnected with the need for accuracy and reliability in nuclear reactor simulations (Barcellos, 2021; Jang & Shim, 2023; Lee et al., 2021; Sanchez-Espinoza et al., 2021; Valentine et al., 2022; Wang et al., 2024). Monte Carlo–based neutron transport codes solves neutron transport stochastically (Cox et al., 2022; Deng et al., 2022; Guo et al., 2022; Lababsa et al., 2024, 2025). They have become very important tools in the design, optimization, safety analysis, and licensing of nuclear systems. Notable examples of Monte Carlo codes include Serpent, MCNP etc. However, these are proprietary codes which mean they have access/use limitations. This may have contributed to the slow progress in the development of competency and reproducibility of research data in reactor safety analysis especially for nuclear newcomer countries in the global south. This may have also heightened the entry barrier of scientists into the nuclear engineering profession in the education and training landscape of nuclear newcomer countries. To overcome this challenge, several open-source codes have been in development and use as alternatives to proprietary codes. The international Atomic Energy Agency (IAEA) has introduced the Open-source Nuclear Codes for Reactor Analysis (ONCORE) initiative to foster international

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collaboration framework for the development and application of open-source multi-physics simulation tools to support research, education and training (IAEA, 2025).

Among the several open-source reactor physics tools that exist include OpenMC. OpenMC is an open-source Monte Carlo neutron and photon transport code. OpenMC is capable of performing problems such as fixed source, subcritical multiplication and k-eigenvalue problems on models whose geometrical representation are built using either a constructive solid geometry (CSG) or Computer Aided Design (CAD) representation (Lababsa et al., 2024, 2025). A broard variety of physical quantities can be tallied and analyzed through a flexible and efficient tally system (Guo et al., 2022). OpenMC can also run in parallel mode using a hybrid MPI and OpenMP programming models (Romano and Forget, 2013, Romano et al., 2015). These are the features that make it particularly attractive for developing nations or nuclear newcomer countries, and academic institutions or nuclear research centres seeking affordable yet accurate reactor physics solutions as an alternative to proprietary reactor physics codes.

The Miniature Neutron Source Reactor (MNSR)**,** a class of low-power, tank-in-pool research reactors, represents an ideal test case for code benchmarking and validation. MNSR-type reactors, such as the Nigeria Research Reactor-1 (NIRR-1)**,** are widely used in neutron activation analysis, education and personnel training. Although several neutronics studies of MNSR-type reactors have been performed using various proprietary codes (Abrefah et al., 2010; Asuku et al., 2022, 2023; Khamis & Khattab, 1999, 2000; Khattab, 2005; Khattab & Khamis, 2004; Simon et al., 2021; Waqar et al., 2009), there are limited work in the open literature that has assessed the performance of OpenMC code for the reactors with compact cores such as the MNSR system. Jonah *et al*. (2009) developed an MCNP model of the NIRR-1 MNSR and conducted a feasibility study for conversion to LEU fuel using the MCNP computer code. They reported 12.45% as optimum enrichment of ^{235}U for UO_2 LEU fuel satisfying the operational safety limits of MNSRs. Similarly, Weetwa et al (2018) predicted neutronic and kinetic parameters of Ghana Research Reactor 1 (GHARR-1) after 19 years of operation using MCNP code.

Criticality safety parameters such as core excess reactivity (ρ_{ex}), control rod worth (CRW) and shutdown margins were estimated and found to demonstrate strong agreement with measured data. Ibikubke et al (2018) also evaluated the feasibility of NIRR-1 conversion to LEU using uranium-silicide candidate fuels. He found that the candidate fuels met the required ρ_{ex} of the NIRR-1. However, the authors estimated a reduction in the neutron flux in experimental position of approximately 10%. Recently, reactor physics analysis of the NIRR-1 has been performed using SCALE (Asuku et al., 2023) The study demonstrated the accuracy of SCALE for prediction of both neutronic safety and depletion parameters. Although studies such as verification of OpenMC for fast reactor physics analysis with China experimental fast reactor start-up tests (Guo, 2022) carried out validation assessment of OpenMC, the reactor type was much larger in size than the MNSR and its neutronic behavior is dominated by fast neutrons while that of the MNSR is essentially a thermal neutron reactor. Similarly, Lababsa et al., (2022, 2025), developed an OpenMC model and performed validation studies for a Materials Testing Reactor (MTR). Though the MTR studied in their work was a thermal reactor like the MNSR, it was much larger in size with a thermal power of 1 MW and thus its resonance self-shielding, spectrum shift and neutron leakage behavior varies from that of MNSRs such as the NIRR-1' The NIRR-1 has a maximum nominal thermal power output of 34 kW (Asuku et al.. 2023).

Validation of OpenMC code for MNSR-type reactors such as the NIRR-1 is necessary due to its compact geometry. MNSR cores exhibit strong neutron leakage (IAEA-TECDOC1844, 2012), significant resonance self-shielding (Khamis et al., 2004), and pronounced spectrum shifts with temperature and geometry (Zain et al., 2018), making their neutronic response highly sensitive to the underlying physics models and data processing methods. Hence it is imperative to develop and evaluate the performance of an OpenMC model of the MNSR (NIRR-1 as a case study) in order to contribute to the demonstration of the suitability of OpenMC for a wide range of reactors and support the broader application of OpenMC and reliability for safety analysis of MNSRs. OpenMC validation of MNSR neutronic parameters enhances confidence in key safety parameters and supports current and future needs such as High Enriched Uranium (HEU) – Low Enriched Uranium (LEU) conversion, regulatory verification, reproducibility, and research and development within the MNSR community.

In this context, the aim of the present study was to develop a high-fidelity 3D OpenMC model of a typical MNSR-type reactor using the NIRR-1 as a case study. Neutronic safety parameters such as effective multiplication factor k_{eff}, core excess reactivity p_{ex}, control rod worth (*CRW*), shutdown margin (*SDM*) and effective delayed neutron fraction β_{eff} were also estimated. The results were validated against experimental and literature data. The developed model will serve not only as a validation framework but also as a foundation for future multi-physics coupling**,** depletion studies**,** and sensitivity/uncertainty analysis**.** Beyond validating OpenMC against measured NIRR-1 data, this study provides a transparent and reproducible open-source benchmark model for MNSR-type reactors that can be easily coupled with other open-source tools for Multiphysics analysis, thereby supporting the objectives of the IAEA ONCORE initiative such as understanding the state-of-the-art in Multiphysics modelling to optimize design and safety of advanced nuclear reactors.

II. THEORETICAL AND CONCEPTUAL FRAMEWORK

A. The NIRR-1

The NIRR-1 consists of the reactor core, annular and bottom beryllium reflectors, two fission chambers, one central control rod made of cadmium. The annular beryllium reflector contains five evenly distributed inner irradiation tubes installed within the beryllium annulus, five outer irradiation tubes are also installed outside the beryllium annulus (Jonah et al., 2009). The reactor vessel is a cylindrical aluminium alloy container, which has a diameter of 0.6 m and a height of 5.6 m. The reactor vessel is suspended in a pool of water made of stainless steel-lined reinforced concrete. The core consists of cylindrical fuel pins arranged in 10 concentric circles to form a fuel cage (Simon et al., 2021). The core is inside the annular beryllium reflector and sits on a bottom beryllium reflector plate.

The volume of the reactor vessel is 1.5 m^3. The active fuel pins are all 13 % enriched UO_2 fuel elements clad in Zircaloy-4 alloy. The concentric circles of the core at a pitch distance of 10.95 mm while the core is held in place by an upper and a lower grid plate fastened together by four tie rods and a central guide tube where the control rod fits for smooth insertion and withdrawal (Simon et al., 2021). In total, there are 350 lattice points in the fuel cage/core of which 335 are active fuel pins while the remaining 15 are dummy pins. The annular beryllium reflector and the bottom reflector are spaced to form the lower orifice, which controls water flow through the core (Asuku et al., 2023). However, the top shim tray of the core and annular beryllium reflector are spaced to form the upper orifice. The main specifications of the reactor are presented in table 1. Details of some key dimensions of the reactor is presented in table 2. However, the dimension of other components can be found in IAEA technical document IAEATECDOC1844 (2012) and CERT (2021).

B. OpenMC Code

OpenMC is an open-source Monte Carlo simulation code developed at the Massachusetts Institute of Technology (MIT) for performing calculations on arbitrary 1D, 2D or 3D geometries using continuous-energy cross-section data. It is a probabilistic tool designed to solve neutron and photon transport problems in various nuclear applications, including reactor physics, shielding analysis, and radiation protection. (Romano *et al.,* 2015).

The software provides a robust and extensible interface in Python and C/C++, simplifying preprocessing, postprocessing, multigroup cross-section generation, depletion analysis, Multiphysics coupling, geometry visualization, and tally processing. It utilizes HDF5-formatted cross-section data files and includes a Python-based data processing interface for converting and managing nuclear data formats like ENDF and ACE into HDF5. Parallel computation is enabled through hybrid MPI and OpenMP programming models (Romano *et al.,* 2020).

OpenMC accurately simulates nuclear reactions that produce secondary neutrons, such as (n, 2n), (n, 3n), fission, and inelastic scattering, following ACE-format energy and angular distribution laws. However, photon transport capabilities, including tracking photons from (n, γ) or fission reactions, are not yet implemented. Additionally, OpenMC employs collision, absorption, and track-length estimators for k_{eff}, combining them using a minimum-variance estimator derived from the covariance matrix of the individual estimators (Lababsa et al., 2024). Detailed OpenMC theoretical considerations for eigen value and other integral quantity can be found in OpenMC documentation (www.openmc.org).

III. MATERIALS AND METHOD

A. Openmc Deployment and Input Preparation

OpenMC was deployed using Docker Desktop on a Windows operating system. To do this, a Docker container was configured via PowerShell, enabling the installation of OpenMC along with its Python API and Jupyter Notebook for interactive simulation. The containerized environment ensured compatibility and reproducibility of the simulation workflow.

Input preparation in Jupyter Notebook began with the import of relevant libraries, including openmc, openmc.deplete, numpy as np, matplotlib.pyplot as plt, and matplotlib.image as mpimg. The ENDF/B-VII.1 library was employed for all neutron interaction data, providing the cross sections required for absorption, scattering, and fission processes (Brown *et al.,* 2019). The OpenMC cross section environment variable was used to link isotopic materials to the corresponding data. Other inputs, including geometry, materials, settings, and tallies, configurations, were also prepared using OpenMC's Python interface in Jupyter Notebook.

In modeling the NIRR-1 geometry, OpenMC's Python-based geometry API was utilized. Specifically, the openmc.Geometry and openmc.Cell classes were employed to define the spatial structure of the reactor in three dimensions (3-D). The modeling approach was adapted from in available literature data of the reactor NIRR-1 as documented by Asuku *et al.,* (2023).

1) Geometry Definition and Visualisation

The reactor geometry was modeled in 3D using OpenMC's Python API, capturing the essential radial components, fuel region (UO_2 pellet), cladding (Zr-4) to form a fuel pin region. Each region was constructed using basic surfaces (openmc.ZCylinder, openmc.Plane) and assigned to a Cell. These cells were organized in a Universe and passed to an openmc.Geometry object. The fuel pins were modeled in a radial pitch of ten (10) concentric radial rings. The central control rod, tie rods, and dummy pins were also defined using similar structure. The core was placed inside the reactor water using the "translation" command in OpenMC python API. The OpenMC model of the NIRR-1 core is presented in figure 1.

Regions outside the core such as annular and bottom beryllium reflector, irradiation channels, cadmium regulators, shim tray, reactor vessel, reactor pool water were also modeled in great details. Each region was visualized using the OpenMC "universe.plot" python class to ensure accurate geometrical representation of the reactor. The OpenMC models of the NIRR-1 with CR fully withdrawn (out) and with CR fully inserted (in) are presented in figures 2 and 3 respectively.

Table 1 Main Specifications of the NIRR-1

Parameter	Value
Type	Tank-in-pool
Nominal core power	34 kW
Coolant/Moderator	Deionised light water
Loading of ^{235}U in core	1350.40 g
Reflector	Metallic beryllium
Excess Reactivity - cold, clean	3.94 mk
Neutron flux at inner irradiation sites	1 × 10^12 n/cm²·s
Number of irradiation sites	10 (5 inner, 5 outer)
Control rod	1, Stainless steel clad, Cadmium absorber

Table 2 Geometric Dimension of some Key Components used in the Model.

Component	Parameter	Value
Inner channels	Number/size	5/Small (S)
	Inner radius (S)	1.10
	Outer radius (S)	1.25
	Length	528.5
Outer channels	Number/size	2/Large (L) + 3/S (same as inner channels)
	Inner radius (L)	1.70
	Outer radius (L)	1.85
	Height	540
Fission chambers	Number	2
	Inner radius	0.50
	Outer radius	0.65
Reactor vessel	Inner radius	30.0
	Outer radius	30.95
	Height	560
Reactor pool	Inner radius	135
	Outer radius	138
	Height	650

2) Material Specification

Material definitions were carried out using the openmc.Material() class, with isotopic compositions and mass density as detailed in literature (TECDOC1844, 2012; CERT, 2021). These materials were then assigned to the respective cells. The temperature for all materials was specified as 298K to accurately capture the p_{ex} of the reactor in line with experimental data.

3). Settings configuration and source distribution

OpenMC simulation parameters in this work were configured through the openmc.Settings() object. This included the number of particles, batches, inactive, source distribution, and run mode. Neutrons were initialized with a uniform spatial distribution across the fuel region, representing a steady-state fission source distributed throughout the core. The number of neutrons per generation was defined as 1,000,000 in the setting, out of which a minimum of 100 batch generations and 20 inactive generation were 'run' in to ensure enough neutron distribution in the core and reduce statistical error for source convergence. These initial generations, allowed the neutron population to reach steady-state conditions without tallying results, avoiding the bias of initial conditions (Moxon et al., 2016) as the data collection began after the inactive generations, with 100 batches generations providing sufficient data for reliable tally results (Moxon et al.,2016)..

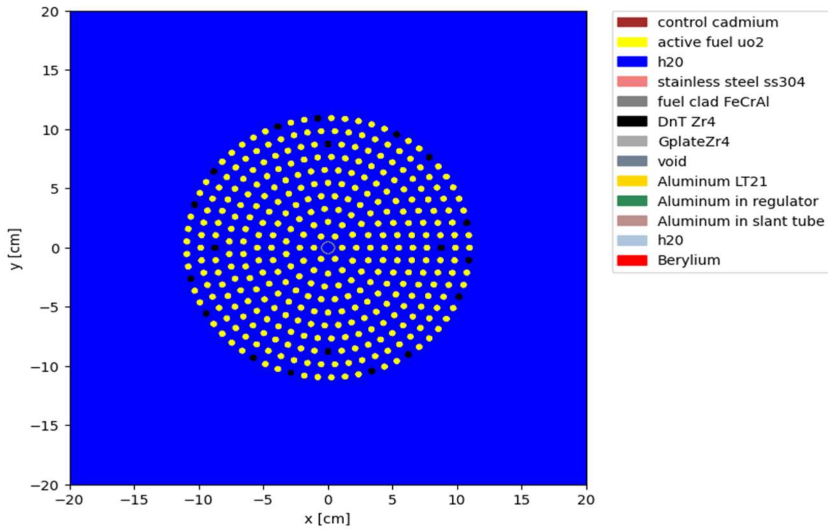

Figure 1 OpenMC model of the NIRR-1 core (X-Y plane)

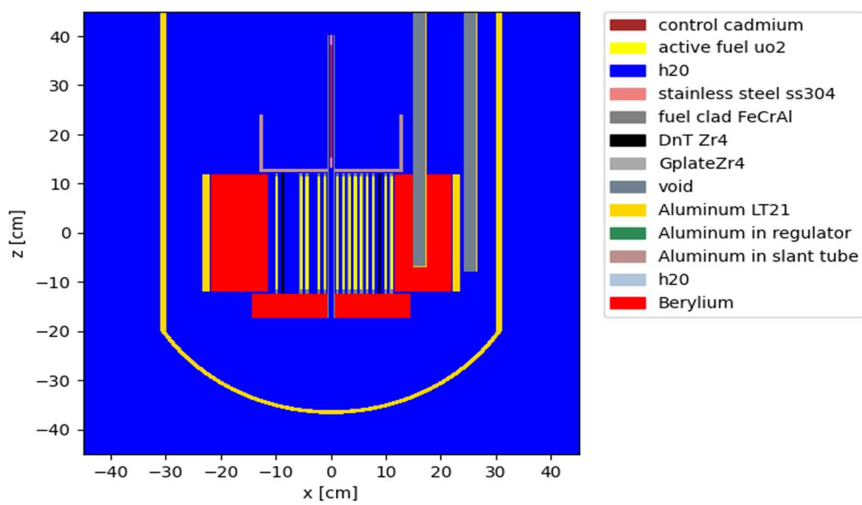

Figure 2 OpenMC model of the NIRR-1 with CR out (X-Z plane)

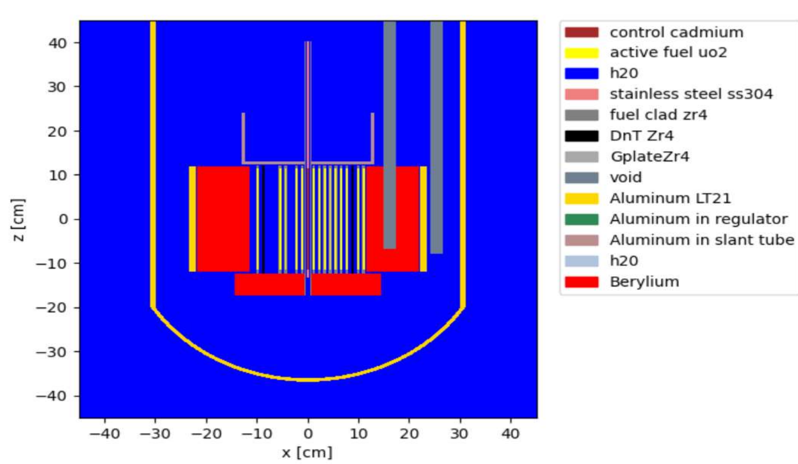

Figure 3 OpenMC model of the NIRR-1 with CR in (X-Z plane)

4) Execution/calculation

(a) Eigenvalue (k_{eff}) calculation

The eigenvalue mode in OpenMC was employed to evaluate the neutron multiplication factor k_{eff} for all configurations of the reactor.

(b) p_{ex} calculation

In order to estimate the p_{ex} of the reactor, the k_{eff} of the reactor was first calculated with the control rod fully withdrawn. The resulting k_{eff} was then substituted into equation 1 (Simon et al, 2021)

$$\rho_{ex} = {(k_{\text{eff}} - 1)}/{k_{\text{eff}}} \tag{1}$$

(c) Control Rod Worth (CRW)

The CRW of the reactor was obtained by calculating two neutron multiplication factors. One with the CR fully withdrawn k_{eff} and the other with the CR fully inserted. k_{eff^*}. To estimate the k_{eff^*} the geometry of the reactor was modified to reflect a fully inserted CR as shown in figure 2. The CRW was then calculated using equation (2) (Asuku et al, 2023)

$$CRW = \frac{k_{\text{eff}} - k_{\text{eff}^*}}{k_{\text{eff}} \times k_{\text{eff}^*}} \tag{2}$$

(d) Shutdown Margin (SDM)

The *SDM* is the difference between the *CRW and* ρ_{ex} as presented in equation (3) (Ibikunle et al, 2018).

$$SDM = CRW - \rho_{ex} \tag{3}$$

(e) Effective delayed neutron fraction β_{eff}

To obtain the β_{eff} the of the reactor was initially calculated using both delayed neutron and prompt neutrons. This is the default setting in OpenMC. However, since there was no direct method for estimating it the k-ratio method (Autin et al, 2022; Mweetwa et al., 2018) was used. In this method, the multiplication factor with prompt and delayed neutron (k_{eff^d}) was first calculated. Then the settings cell of the OpenMC code was modified to exempt delayed neutrons from the next calculation i.e multiplication factor with only prompt neutrons (k_{eff^p}) i.e suppressing the delayed neutrons in the effective multiplication calculations through muting the delayed neutrons by setting the "*create_delayed_neutrons command*" to False (Romero-Barrientos et al., 2022, Darian, 2025). Equation (4) was used to estimate the β_{eff} (Autin et al, 2022; Romero-Barrientos et al., 2022; Darian, 2025).

$$\beta_{eff} = {(k_{\text{eff}^d} - k_{\text{eff}^p})}/{k_{\text{eff}^p}} \tag{4}$$

where

k_{eff^p} = effective multiplication factor with only prompt neutrons

k_{eff^d} = effective multiplication factor with both prompt and delayed neutron

IV. RESULTS AND DISCUSSION

A. Calculated ρ_{ex}

Figure 4 shows the Shannon entropy evolution for the NIRR-1 OpenMC model. The Shannon entropy evolution was necessary to confirm the adequacy of the 20 inactive generations in the particle history used in the calculations in this work. As can be observed from the figure, the entropy decreased rapidly during the initial batches and reached a stable plateau at approximately 9.668 after about 10 batches. The absence of any systematic drift thereafter confirms adequate convergence of the fission source distribution. Since convergence was achieved before the completion of the 20 inactive generations, the subsequent active generations were deemed suitable for estimating neutronic safety parameters in this work.

The k_{eff} for the different simulated conditions of the developed reactor model in this work is presented in table 3. As expected, the k_{eff} and k_{eff^*} of the reactor were critical and subcritical respectively, suggesting desired performance in response to effect of perturbations on the reactor using the CR movement/position.. This is because withdrawal and insertion of the CR implied withdrawal and insertion of the cadmium material which has a high absorption cross section for thermal neutrons (Asuku et al, 2016). In this regard the developed model performed as anticipated in line with literature (Asuku et al, 2016; Ibikunle et al, 2018). However, to quantify the accuracy of the model, the ρ_{ex} was estimated. ρ_{ex} is considered the baseline criticality safety parameter of nuclear fission reactors (Simon et al, 2021). The ρ_{ex} in this work was estimated to be 3.78 mk. For the pupose of validation, this value was compoared with experimental value of 3.98 mk obtained during commissioning experiment of the reactor as presented in table 4. The bias between the OpenMC result and experiment was found to be 4.6%. This slight deviation implies a strong agreement between calculated value and experiment in line with similar findings by Asuku et al, (2023; Jonah et al, (2009). The reason for this slight deviation may be attributed to the slightly lower predicted k_{eff} which may be as a result of nuclear data updates such as capture cross sections for nuclides such ^{235}U, and ^{238}U in ENDFVII.1 (Austin et al, 2022). The calculated ρ_{ex} also compares strongly with the SCALE calculated value of 3.82 mk (Asuku et al, 2023) and matches the value of 3.79 mk reported by Simon et al, (2021). These findings demonstrate that the developed OpenMC model of the NIRR-1 in this work can be used to estimate the ρ_{ex} of the reactor within a margin of < 5% deviation from measurement. The absolute difference of 0.19 mk between calculated and measured ρ_{ex} corresponds to approximately 1.6 standard deviations of the Monte Carlo uncertainty and is therefore not statistically significant at the 95% confidence level.

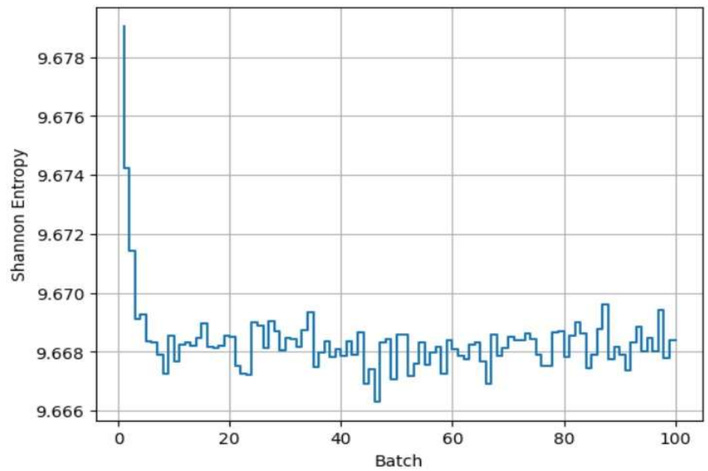

Figure 4 Shannon entropy evolution demonstrating source convergence of the OpenMC NIRR-1 model.

Table 3 Calculated Multiplication Factors

Parameter	Value
k_{eff}	1.00379 ± 0.00012
k_{eff^*}	0.99696 ± 0.00011
k_{eff^d}	1.00379 ± 0.00012
k_{eff^p}	0.99577 ± 0.00011

Table 4 Calculated ρ_{ex}

ρ_{ex}	Value (mk)
This work	3.78 ± 0.12
SCALE calculated (Asuku et al, 2023)	3.80
SCALE calculated (Simon et al, 2021)	3.78
Measured (CERT, 2021)	3.97

B. *Evaluation of CRW and SDM Parameters*

CRW and *SDM* are important neutronic safety parameters in all nuclear fission reactors. They provide insights on the safety of the reactor during normal operation and emergency conditions. The calculated *CRW* in the present study was estimated to be 6.82 ± 0.16 mk. However, the *SDM* was calculated to be 3.04 ± 0.20 mk as shown in figure 5. Together, the values of *CRW* and *SDM* imply that an insertion of the CR will add a negative reactivity of – 3.78mk (6.82 – 3.04 mk) into the reactor, resulting in a decrease in neutron population over successive generations (subcriticalstate of the reactor otherwise known as shutdown). This is because of the cadmium meat of the CR which has a high absorption cross section for thermal neutrons. Although the design *CRW* of the NIRR-1 is 7.0 mk, the reported measured value of the *CRW* is 6.49 mk (CERT, 2021) while that of the *SDM* is 2.55 mk. The regulatory safety requirement for *SDM* for the NIRR-1 is > 2.5 mk (CERT, 2021). Hence, implying that the NIRR-1 is operating safely within regulatory limits. A comparison of the *CRW* and *SDM* is also presented in figure 5. The deviations of the predicted *CRW* and *SDM* in this work from measured result were found to be 2.6% and 0.7% respectively, indicating a strong agreement between calculated result and measured value. The slight difference could be perturbation assumptions in the OpenMC code or the cross section data of the CR materials used in the simulation. The *CRW* and *SDM* in this work also compare well with SCALE result in literature (Asuku et al, 2023). This demonstrates OpenMC performance in accurately estimating the *CRW* and *SDM* of the NIRR-1.

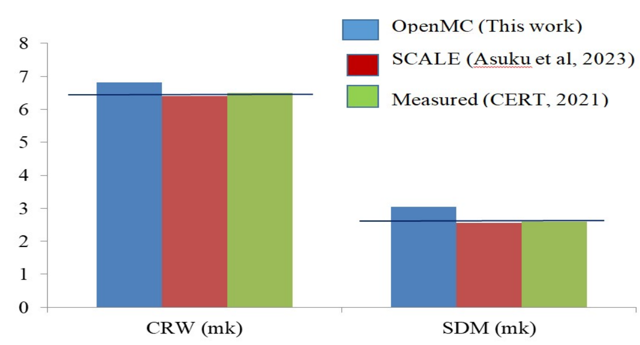

Figure 5 Comparison of CRW and SDM results with Measured Data and Literature

C. Calculated β_{eff}

Table 5 presents the result of the calculated β_{eff} in this work. The β_{eff} was estimated to be 0.007990 equivalent to 7.99 mk. This value is approximately twice (2 times) the value of the predicted ρ_{ex}. It implies that the NIRR-1 cannot undergo a prompt critical under normal operation condition. This is because the condition for prompt criticality is that

$\rho_{ex} > 1/2\, \beta_{eff}$ (Mweetwa et al., 2018). This further confirms the safety of the NIRR-1 in line with MNSR regulatory operational limits $\rho_{ex} \leq \frac{1}{2}\beta_{eff}$ (TECDOC1844, 2012). A comparison of the calculated β_{eff} in this work to measured data of 0.008014 equivalent nt to 8.014 mk reveals a deviation of approximately 0.2% indicating a strong agreement between calculated and measured value. This further demonstrates OpenMC capability in accurate estimation of the NIRR-1. However, the slight deviation could be attributed to the number of particle history used for the simulation.

Table 5 Estimated β_{eff}

$\boldsymbol{\beta_{eff}}$	**Value**
This work	0.007990
Measured (CERT, 2021)	0.008014

V. CONCLUSION

In *this* work, a new model of the NIRR-1 was developed using open-source reactor physics computational tool OpenMC. Criticality/neutronic safety parameters such as k_{eff}, p_{ex}, *CRW*, *SDM* and β_{eff} were also estimated using the developed model. The findings in this suggest that the aforementioned safety parameters of the NIRR-1 can be reliably predicted within a deviation of < 5% margin with reference to measured data. The calculated safety parameters in this work satisfy the operational neutronic safety requirement of MNSRs. Consequently, this work confirms the safety of the NIRR-1 and demonstrates the accuracy of the newly developed OpenMC model of the NIRR-1 in predicting key neutronic safety parameters. Beyond demonstrating the predictive accuracy of OpenMC for MNSR safety analysis, this work establishes a validated open-source modeling framework that can serve as a reference benchmark for future reactor physics studies, educational activities. The model and developed and validated in this work is required in multi-physics developments such as coupled thermal-hydraulics analysis involving MNSR-type reactors. The availability of a transparent and reproducible OpenMC model supports open-science practices, facilitates independent verification of reactor safety analyses, and contributes directly to the IAEA ONCORE initiative by expanding the portfolio of validated open-source reactor models available to the international community, particularly for nuclear newcomer countries.

AUTHOR CONTRIBUTIONS

A.Asuku: Conceptualization, Methodology, results interpretation, Initial draft of manuscript; **Y.V. Ibrahim**: Methodology and supervision; **S.A. Jonah**: Supervision and team leadership; **J. Simon**: Manuscript writing and Results interpretation; **B. Yahaya**: Manuscript review and editing; **B.D. Jatau**: Manuscript review; **D. Sani**: Manuscript review and Methodology; **I. B. Kachalla**: Manuscript review and Methodology; **B. Bulus**: Manuscript review and Methodology; **A. Sadiq**: Manuscript review.

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