Nigerian Journal of Physics (NJP)
ISSN online: 3027-0936
ISSN print: 1595-0611
DOI: https://doi.org/10.62292/njp.v35i3.2026.613
Volume 35(3), September 2026

Impact of ENDF/B-VII.1 and ENDF/B-VIII.0 Nuclear Data Libraries on NIRR-1 LEU Core Depletion using OpenMC

*[1]Gladys Ajibola Opashola, [1]Emmanuel Joseph and [2]Abdulsamad Asuku

[1]Department of Physics, Federal University Dutsin-Ma, Katsina State – Nigeria
[2]Centre for Energy Research and Training, Ahmadu Bello University, Zaria – Nigeria

*Corresponding Author's Email: g.ajibolaopashola@fudutsinma.edu.ng

ABSTRACT
The Nigeria Research Reactor-1 (NIRR-1), a 30 kW Miniature Neutron Source Reactor (MNSR), operates at low power with a highly thermalized neutron spectrum, leading to extremely low fuel burnup over extended lifetimes. Accurate depletion modeling is critical for predicting isotopic evolution, reactivity loss, core lifetime extension via beryllium shims, and safeguards compliance after the 2018 conversion from highly enriched uranium (HEU) to low-enriched uranium (LEU) fuel (~13% ^{235}U in UO_2). This study examines the sensitivity of NIRR-1 LEU core depletion predictions to ENDF/B-VII.1 (2011) and ENDF/B-VIII.0 (2018) nuclear data libraries using the open-source Monte Carlo code OpenMC. Parallel simulations over 10 effective full power days (EFPD) at nominal power kept all inputs identical except the nuclear data library. Results reveal strong convergence in this low-burnup thermal regime. U-235 depletion shows small oscillating differences (average absolute ~0.17%, max |0.32|%), U-238 consumption is negligible (<0.1% net variation), and Pu-239 buildup exhibits a modest systematic positive bias in ENDF/B-VIII.0 (~+0.22% average, up to +0.44% early). Burnup accumulation remains linear with relative differences <0.5% (average ~+0.27% higher in ENDF/B-VIII.0). These subtle effects, stemming from CIELO-project refinements (improved U-235 fission, U-238 resonances, Pu-239 fission/nu-bar), are far smaller than discrepancies in high-burnup light-water reactor benchmarks. Nuclear data library choice has negligible short-term impact on NIRR-1 operations, supporting ENDF/B-VIII.0 adoption for enhanced long-term accuracy without altering core management or safety margins. Extrapolation indicates potential 1–3% cumulative variances over the ~50–60-year LEU lifetime, warranting extended simulations and experimental validation (flux measurements, post-irradiation examination). The work highlights open-source Monte Carlo tools for high-fidelity analysis in resource-limited settings and advances lifetime predictions, non-proliferation safeguards, and capacity building for African research reactors.

Keywords:
NIRR-1,
MNSR,
Core depletion,
Nuclear data libraries,
ENDF/B-VIII.0,
OpenMC.

INTRODUCTION

Research reactors are essential facilities for scientific progress, serving as intense neutron sources for applications including neutron activation analysis (NAA), radioisotope production, materials testing under irradiation, and nuclear education and training. In contrast to power reactors, research reactors generally operate at low power levels with highly thermalized neutron spectra. This design enables high neutron fluxes at irradiation positions while resulting in very low fuel burnup even after many years of service.

The Nigeria Research Reactor-1 (NIRR-1), a Miniature Neutron Source Reactor (MNSR) designed by the China Institute of Atomic Energy, is installed at the Centre for Energy Research and Training (CERT), Ahmadu Bello University, Zaria. Commissioned in 2004, NIRR-1 initially operated with a highly enriched uranium (HEU) core containing 90.2% ^{235}U in UAl_4–Al dispersion fuel (Jonah et al., 2005). As a tank-in-pool type reactor, it delivers a nominal thermal power of 30–31 kW. The core consists of approximately 347–350 fuel pins arranged in concentric rings, moderated and cooled by light water, surrounded by beryllium reflectors (annular side and bottom reflectors plus top shim trays), and controlled by a single central cadmium rod (IAEA, 2005; IAEA, 2018; Ahmed et al., 2018). The facility supports NAA for geochemical and environmental studies, limited radioisotope production, and training in nuclear engineering within Nigeria. After achieving initial criticality with HEU fuel in February 2004, NIRR-1 was successfully converted to low-enriched uranium (LEU) fuel (≈13% ^{235}U in UO_2 pellets clad in Zircaloy-4) and reached second criticality on November 2, 2018 (Asuku, 2023).

This core conversion formed part of international non-proliferation efforts coordinated by the Reduced Enrichment for Research and Test Reactors (RERTR) program and the IAEA. The irradiated HEU was repatriated, and the new LEU core, manufactured in China, was implemented to reduce proliferation risks while maintaining essential performance parameters, including excess reactivity of approximately 4 mk and thermal neutron fluxes of ~10^{12} n/cm²/s in irradiation channels (International Atomic Energy Agency, 2018; Jonah et al., 2012; IAEA, 2019). NIRR-1 became the second MNSR outside China (after Ghana's GHARR-1 in 2017) to complete this transition.

Accurate core depletion (burnup) analysis is critical for research reactors such as NIRR-1. It predicts the time-dependent evolution of fuel composition, reactivity decline, buildup of actinides and fission products, and overall core lifetime. Owing to its intermittent, low-power operation (typically 2–3 hours per day), NIRR-1 experiences extremely slow burnup, with less than 1% ^{235}U depletion expected over 10–15 years. Dominant effects include modest ^{235}U consumption (~7–8 g over the core lifetime), trace Pu-239 production (~0.05 g), and transient or equilibrium poisoning by fission products such as ^{135}Xe (short-term) and ^{149}Sm (long-term) (Alhassan & Boafo, 2016; Dawahra et al., 2018). Reliable depletion calculations support safe core-life extension through beryllium shim addition, safeguards compliance, and optimized facility utilization.

High-fidelity Monte Carlo neutron transport codes (MCNP, Serpent, OpenMC) have become preferred tools for depletion studies in research reactors. These codes excel at modeling complex three-dimensional geometries and utilizing continuous-energy cross sections, outperforming traditional deterministic methods (WIMS/CITATION) in many cases (Fensin et al., 2006; Romano et al., 2015; Alhassan & Boafo, 2016). By coupling neutron transport with Bateman-equation solvers, they track nuclide transmutation over time and reveal sensitivities that are particularly important in low-burnup systems where small reactivity changes can be operationally significant. Nuclear data libraries which were processed from ENDF files via tools such as NJOY, provide the fundamental microscopic cross sections, fission yields, decay data, and thermal scattering laws that drive these simulations (MacFarlane & Kahler, 2010).

Throughout operation, fission depletes ^{235}U while neutron capture on ^{238}U initiates chains that produce transuranic isotopes such as ^{239}Pu. These isotopic changes influence core reactivity, neutron spectrum, and burnup behaviour (Spriggs, 2005). Differences between successive evaluated nuclear data libraries can therefore produce meaningful variations in predicted inventories and performance parameters, especially for key actinides (^{235}U, ^{238}U, ^{239}Pu). The shift from ENDF/B-VII.1 (2011) to ENDF/B-VIII.0 (2018) introduced CIELO-project improvements in major actinide evaluations, thermal scattering laws, and resonance parameters. While many updates enhanced accuracy, some comparisons have shown regressions in thermal-spectrum depletion benchmarks, such as slightly higher ^{235}U absorption leading to underpredicted reactivity loss (Chadwick et al., 2011; Brown et al., 2018; Trkov et al., 2019; Kim & Wieselquist, 2021; Park et al., 2019). Most published sensitivity studies address high-burnup power reactors (e.g., PWRs with differences of 500–1000 pcm), leaving the impact on low-burnup thermal research reactors like MNSRs relatively underexplored (Vu, 2024).

Given the importance of precise neutronic characterization for safety analysis, operational planning, and long-term core management, particularly following HEU-to-LEU conversion, understanding the sensitivity of depletion results to the choice of nuclear data library is essential for facilities such as NIRR-1 (Ojovan & Lee, 2019).

The present study fills this gap by quantitatively assessing the influence of ENDF/B-VII.1 versus ENDF/B-VIII.0 nuclear data libraries on the depletion behaviour of the NIRR-1 LEU core, using the open-source Monte Carlo code OpenMC. The work aims to improve confidence in lifetime predictions, safety margins, and the validation of modern nuclear data libraries for thermal-spectrum research reactors in Africa.

Theoretical Analysis

Reactor core depletion analysis describes the time-dependent transmutation of nuclides through neutron interactions and radioactive decay. The governing equations are the Bateman equations (Bateman, 1910; Duderstadt & Hamilton, 1976):

$$\frac{dN_i}{dt} = \sum_{j\neq i}\left(\sigma_{j\rightarrow i}\phi N_j + \lambda_{j\rightarrow i}N_j\right) - (\sigma_{i,\mathrm{abs}}\phi + \lambda_i)N_i \quad (1)$$

where N_i is the number density of isotope i, $\sigma_{j\rightarrow i}$ represents production cross-sections (e.g., capture or fission yield), ϕ is the neutron flux, λ are decay constants, and $\sigma_{i,\mathrm{abs}}$ is the total absorption cross-section. In low-burnup thermal systems like NIRR-1, dominant processes include fission and capture on ^{235}U, and (n,γ) capture on ^{238}U leading to the ^{239}Pu chain (Spriggs, 2005; Dawahra et al., 2018):

$^{238}U(n,\gamma)^{239}U \rightarrow \beta^- \rightarrow {}^{239}Np \rightarrow \beta^- \rightarrow {}^{239}Pu$

Burnup (BU) is defined as the cumulative energy released per unit initial heavy-metal mass (Stanculescu, 2003):

$$BU = \frac{P \cdot t}{M_{\mathrm{HM}}} \text{(MWd/kgU)} \quad (2)$$

where $P$is thermal power and $M_{\mathrm{HM}}$is the heavy-metal mass. For NIRR-1 (30 kW, thermal flux ~10^{12} n/cm²/s), burnup accumulates slowly (<1% ^{235}U depletion over decades), amplifying the importance of small reaction-rate variations (Yahaya et al., 2017; Abdulaziz et al., 2021).

Nuclear data libraries supply the microscopic cross-sections, fission yields, decay constants, and thermal scattering laws needed to solve the Bateman system. ENDF/B-VII.1 (Chadwick et al., 2011) and ENDF/B-VIII.0 (Brown et al., 2018) differ mainly due to CIELO-project improvements: refined ^{235}U thermal fission and resonance parameters, updated ^{238}U capture resonances, and enhanced ^{239}Pu fission/nu-bar data. These updates can shift thermal-region reaction rates by 1–3%, influencing actinide inventories and reactivity evolution (Kim & Wieselquist, 2021; Park et al., 2019).

Monte Carlo neutron transport solves the time-independent Boltzmann equation stochastically (Lewis & Miller, 1984; Romano et al., 2015):

$$\mathbf{\Omega} \cdot \nabla\psi(\mathbf{r}, E, \mathbf{\Omega}) + \Sigma_t(\mathbf{r}, E)\psi = \int \Sigma_s(\mathbf{r}, E' \rightarrow E, \mathbf{\Omega}' \rightarrow \mathbf{\Omega})\psi \, dE' d\mathbf{\Omega}' + \frac{\chi(E)}{4\pi} \nu\Sigma_f \psi + Q \quad (3)$$

OpenMC couples continuous-energy transport (using NJOY-processed pointwise data; MacFarlane & Kahler, 2010) with a predictor-corrector Chebyshev Rational Approximation Method (CRAM) solver for the Bateman equations (Pusa & Leppänen, 2010; Romano & Forget, 2013). This high-fidelity approach accurately models NIRR-1's 3D geometry (347 fuel pins, Be reflectors, light-water moderation) and spatial flux variations without multigroup approximations common in deterministic codes like WIMS/CITATION (Yahaya et al., 2017).

In this work, identical model inputs (geometry, power history, depletion steps) isolate nuclear-data effects. Theoretical expectations indicate that ENDF/B-VIII.0 refinements produce minor systematic shifts in U-235 retention, U-238 capture, and Pu-239 production, precisely the quantities evaluated here. The framework confirms that Monte Carlo depletion with modern ENDF libraries provides the most reliable method for low-burnup thermal research reactors (Romano et al., 2015; Brown et al., 2018).

MATERIALS AND METHODS

Materials

Computational Tool

The open-source Monte Carlo neutron transport code OpenMC was chosen as the primary simulation platform. OpenMC supports continuous-energy neutron transport, includes integrated depletion and burnup modules based on the predictor-corrector Chebyshev Rational Approximation Method (CRAM) for solving Bateman equations, and is fully compatible with processed ENDF-formatted nuclear data libraries (Romano & Forget, 2013; Romano et al., 2015). Its Python application programming interface (API) facilitates automated post-processing of results, making it well-suited for comparative depletion studies.

Methods

Research Design

This study adopted a computational experimental approach, a standard methodology in reactor physics for quantifying sensitivities and uncertainties arising from nuclear data variations (Shen et al., 2019; Brown et al., 2018). The workflow consisted of: (i) constructing a detailed three-dimensional model of the NIRR-1 LEU core, (ii) conducting parallel depletion simulations using two distinct nuclear data libraries (ENDF/B-VII.1 and ENDF/B-VIII.0), and (iii) performing quantitative comparisons of key outputs, primarily isotopic inventories (U-235, U-238, Pu-239) and burnup metrics so as to isolate the effects of the nuclear data libraries.

NIRR-1 Core Model Development

The Nigeria Research Reactor-1 (NIRR-1) is a 30 kW Miniature Neutron Source Reactor (MNSR) located at the Centre for Energy Research and Training (CERT), Ahmadu Bello University, Zaria, Nigeria. Following its 2018 conversion, the core uses low-enriched uranium (LEU) fuel enriched to approximately 13% ^{235}U in UO_2 form (International Atomic Energy Agency, 2018; Asuku, 2023).

A high-fidelity 3D model of the Nigerian Research Reactor-1 (NIRR-1) low-enriched uranium (LEU) core was implemented in the OpenMC using a Cartesian coordinate system, with the core center defined as the origin (0,0,0). The model explicitly represents all major components, including: 347 fuel pins plus three aluminum dummy elements, the central cadmium control rod, light-water moderator and coolant, grid plates, beryllium reflectors (side, bottom, and top shim trays), inner and outer irradiation channels, reactivity regulators, fission chambers, startup (slant) guide tube, temperature measurement devices, aluminum support structures, reactor vessel, pool, and stainless-steel liner (Jonah et al., 2005a; Snoj et al., 2012; Romano et al., 2015).

Geometric dimensions and material compositions were sourced from the final Safety Analysis Report (SAR, 2005) and validated against prior MCNP-based models and experimental benchmarks (e.g., criticality, flux distributions) reported in the literature. A representative mid-plane (x–y) view of the core geometry, as visualized in compatible tools, is shown in Figure 1.

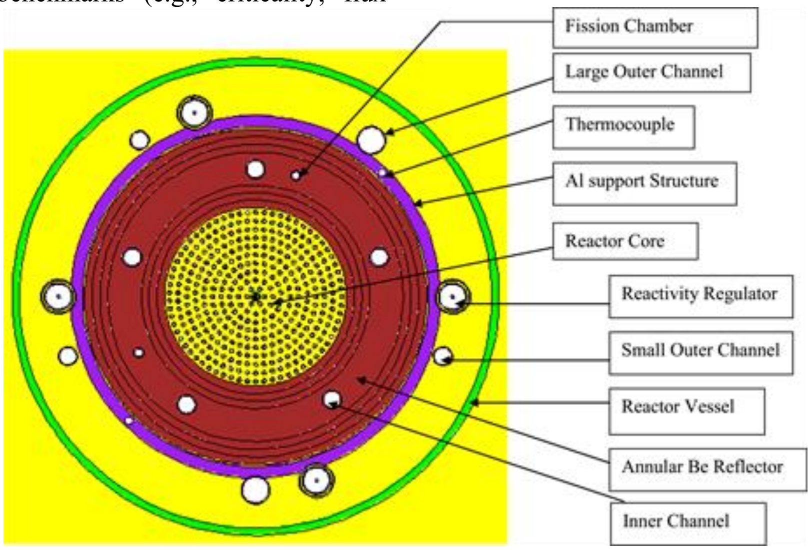

Figure 1: Geometric representation of the NIRR-1 core mid-plane (x–y view) in the Monte Carlo model

Nuclear Data Libraries

Two evaluated nuclear data libraries were employed in this study:

i. ENDF/B-VII.1 (released 2011), featuring improved neutron cross sections, fission product yields, and thermal scattering laws for key isotopes (Chadwick et al., 2011).

ii. ENDF/B-VIII.0 (released 2018), incorporating CIELO-project advancements in major actinide evaluations (U-235, U-238, Pu-239), resonance parameters, and fission yields (Brown et al., 2018).

The original ENDF files were processed with NJOY to produce continuous-energy pointwise cross-section libraries compatible with OpenMC.

Monte Carlo Depletion Simulations

Depletion calculations were performed using OpenMC (version 0.13.3) (Romano et al., 2015). Simulations maintained a constant thermal power of 30 kW, employing the predictor-corrector CRAM integration scheme for time advancement. Key Monte Carlo parameters were particles per cycle (100,000), inactive cycles (20) and active cycles (80). Depletion was simulated over equivalent operational periods up to 10 days (with potential extension as needed for trend analysis), capturing time-dependent isotopic evolution. Two identical simulations were executed, one using ENDF/B-VII.1 and the other ENDF/B-VIII.0, with all other inputs (geometry, power history, depletion chain) held constant.

Data Analysis and Quantification of Differences

Results from the two libraries were compared via percentage differences in isotopic masses (U-235 depletion, U-238 inventory, Pu-239 buildup) and cumulative burnup. The relative difference was calculated as:

$$\%\,\Delta = \frac{R_{VIII.0} - R_{VII.1}}{R_{VII.1}} \times 100 \qquad (4)$$

where $R_{VIII.0}$ and $R_{VII.1}$ denote results from ENDF/B-VIII.0 and ENDF/B-VII.1, respectively.

Simulation Procedure

The analysis followed these steps:

i. Model verification: Benchmark the OpenMC model by comparing calculated k_{eff}, neutron flux distributions, and power peaking against published experimental data and prior deterministic/Monte Carlo results (e.g., Serpent lattice or MCNP validations).

ii. Baseline depletion run: Perform full-core depletion using ENDF/B-VII.1.

iii. Library comparison run: Repeat the simulation identically but with ENDF/B-VIII.0 (using identically processed data).

iv. Data extraction: Record time-dependent values for U-235 mass, U-238 mass, Pu-239 buildup, total burnup (MWd/kgU or equivalent), and reactivity trends.

Validation, Benchmarking, and Uncertainty Considerations

Where feasible, predicted inventories and burnup were cross-checked against independent benchmarks, including prior WIMS/CITATION or SCALE-based depletion studies for NIRR-1 conversion and operation. Central-value comparisons form the core of this work; however, both libraries include covariance data that could support future uncertainty propagation analyses (e.g., via stochastic sampling or sensitivity methods; see references in Brown et al., 2018, and related depletion uncertainty literature).

RESULTS AND DISCUSSION

The results of the depletion simulations for the NIRR-1 LEU core over 10 effective full power days (EFPD) at 30 kW, comparing ENDF/B-VII.1 and ENDF/B-VIII.0 using OpenMC are well presented here. All inventories are normalized (likely per representative fuel pin or lattice cell) for detailed analysis, as full-core totals scale accordingly (e.g., initial core U-235 ≈1406 g, U-238 ≈9036 g post-conversion; Simon et al., 2021; Asuku, 2023), and differences are equally quantified.

Results

Impact on U-235 Inventory

Table 1 shows U-235 inventory evolution. Initial values differ slightly (+0.30% in ENDF/B-VIII.0), likely due to cross-section processing and thermal scattering updates (Brown et al., 2018). Both libraries predict near-linear depletion (~0.43–0.44 g over 10 EFPD normalized), consistent with low burnup in MNSRs (~0.03–0.04 g/EFPD scaled; Yahaya et al., 2017). Percentage differences oscillate (-0.32% to +0.30%; average absolute ~0.17%), reflecting CIELO revisions to U-235 fission/capture in ENDF/B-VIII.0, which cause minor rate variations in thermal spectra (Kim & Wieselquist, 2021; Park et al., 2019). These are negligible short-term (<0.4%) compared to high-burnup PWRs (up to 5–10%).

Table 1: U-235 inventory variation over 10 days (g) and percentage differences.

Time (days)	ENDF/B-VII.1 (g)	ENDF/B-VIII.0 (g)	Difference (%)
0.0	4.047757	4.060000	+0.30
1.0	4.004163	4.008167	+0.10
2.0	3.960591	3.948000	-0.32
3.0	3.917061	3.922000	+0.13
4.0	3.873585	3.865000	-0.22
5.0	3.830170	3.831000	+0.02
6.0	3.786825	3.780000	-0.18
7.0	3.743554	3.740000	-0.10
8.0	3.700363	3.710000	+0.26
9.0	3.657251	3.660000	+0.08
10.0	3.614221	3.620000	+0.16

Figure 2 visualizes near-overlapping depletion curves, confirming minimal library impact short-term but potential 1–2% long-term variance affecting lifetime estimates (~200–300 EFPD baseline).

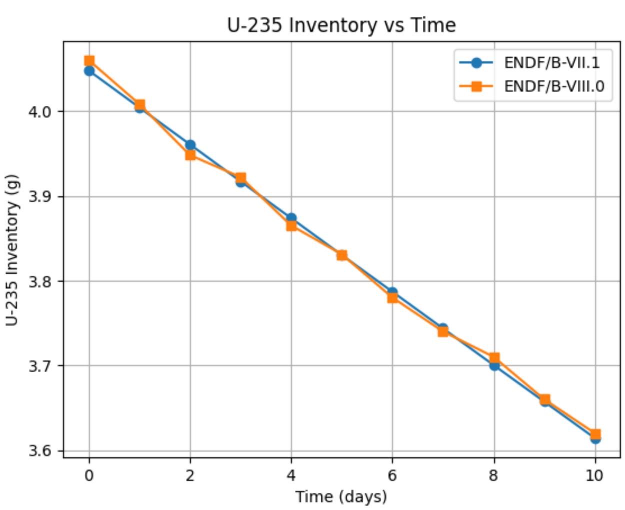

Figure 2: Comparison of U-235 inventory depletion over 10 EFPD using ENDF/B-VII.1 and ENDF/B-VIII.0

Impact on U-238 and Pu-239 Inventories

Table 2 indicates minor U-238 depletion (~0.04–0.05 g normalized over 10 EFPD), driven by (n,γ) capture. Differences are small (average absolute ~0.02–0.17%), with ENDF/B-VIII.0 showing slight faster depletion in some steps due to CIELO-adjusted resonances (Brown et al., 2018).

Table 2: U-238 inventory variation over 10 days (g) and percentage differences

Time (days)	U-238 Inventory ENDFVII.1 (g)	U-238 Inventory ENDFVIII.0 (g)	Difference (%)
0.0	26.948719	26.960000	+0.04
1.0	26.944453	26.950000	+0.02
2.0	26.940123	26.932000	-0.03
3.0	26.935761	26.938000	+0.01
4.0	26.931375	26.930000	-0.01
5.0	26.926967	26.929000	+0.01
6.0	26.922537	26.915000	-0.03
7.0	26.918082	26.920000	+0.01
8.0	26.913609	26.910000	-0.01
9.0	26.909110	26.900000	-0.03
10.0	26.904597	26.905000	+0.00

Table 3 shows Pu-239 buildup (~0.024 g normalized), from U-238 capture chain. ENDF/B-VIII.0 predicts consistently higher (+0.19–0.44%; average ~+0.22%), due to refined Pu-239 fission/nu-bar and U-238 capture data enhancing survival (Kim & Wieselquist, 2021; Park et al., 2019). This aligns with thermal benchmark trends (2–10% higher transuranics at low burnup; Choi et al., 2021).

Table 3: Pu-239 inventory buildup over 10 days (g) and percentage differences

Time (days)	Pu-239 Inventory ENDFVII.1 (g)	Pu-239 Inventory ENDFVIII.0 (g)	Difference (%)
0.0	0.000000	0.000000	+0.00
1.0	0.000488	0.000490	+0.41
2.0	0.001812	0.001820	+0.44
3.0	0.003743	0.003750	+0.19
4.0	0.006107	0.006120	+0.21
5.0	0.0087717	0.008790	+0.22
6.0	0.011639	0.011660	+0.18
7.0	0.014638	0.014670	+0.22
8.0	0.017715	0.017750	+0.20
9.0	0.020830	0.020870	+0.19
10.0	0.023954	0.024000	+0.19

Figure 3 and 4 illustrate tight overlap for U-238 (slight faster depletion in VIII.0) and systematic higher Pu-239 in VIII.0, with implications for long-term safeguards (~5–10% more Pu-239 lifetime total).

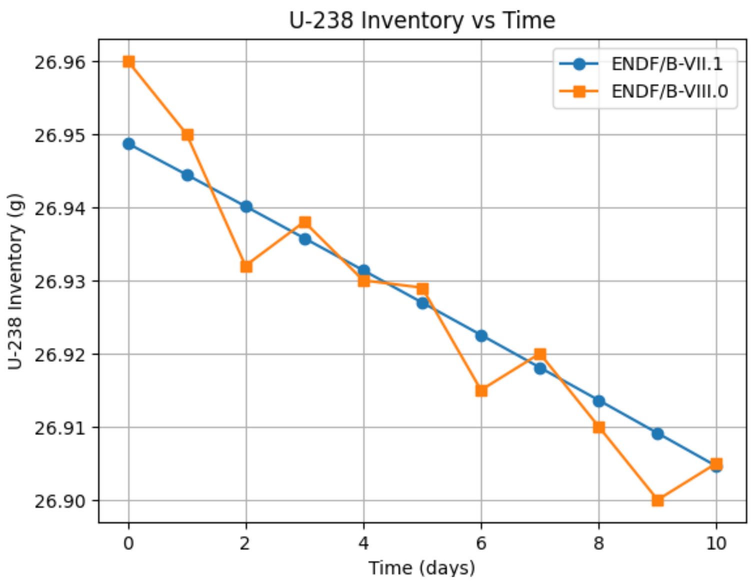

Figure 3: Comparison of U-238 inventory depletion over 10 EFPD

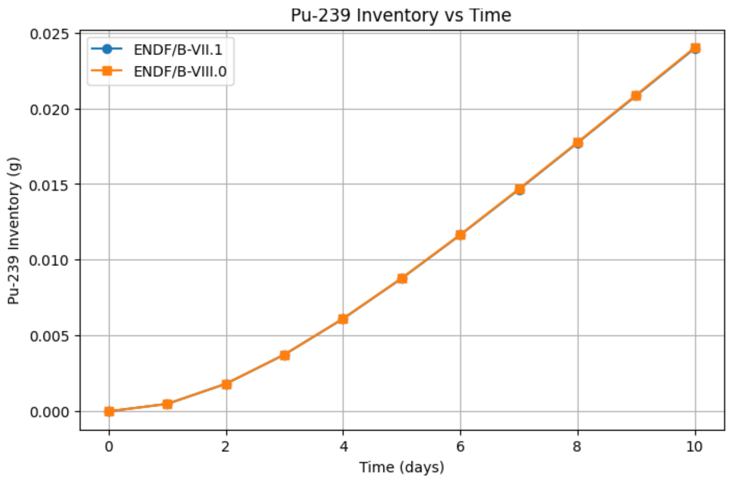

Figure 4: Comparison of Pu-239 inventory buildup over 10 EFPD

Impact on Burnup Predictions

Table 4 shows cumulative burnup (MWd, normalized/total equivalent). Linear accumulation (~0.012 MWd/EFPD) reaches ~0.124 MWd at 10 EFPD, consistent with low MNSR burnup (<0.2 GWd/t lifetime; Balami et al., 2025). ENDF/B-VIII.0 predicts slightly higher (+0.16–0.45%; average ~+0.27%), tied to minor actinide shifts.

Table 4: Burnup accumulation over 10 days (MWd) and percentage differences

Time (days)	Burnup ENDFVII.1 (MWd)	Burnup ENDFVIII.0 (MWd)	Difference (%)
0.0	0.00000	0.00000	+0.00
1.0	0.01240	0.01242	+0.16
2.0	0.02490	0.02496	+0.24
3.0	0.03730	0.03739	+0.24
4.0	0.04970	0. .04982	+0.24
5.0	0.06210	0. 06226	+0.26
6.0	0.07450	0.07470	+0.27
7.0	0.08680	0.08705	+0.30
8.0	0.09920	0.09953	+0.33
9.0	0.11150	0.11194	+0.39
10.0	0.12380	0.12435	+0.45

Figure 5 confirms near-identical linear trends, with negligible short-term difference (<0.5%).

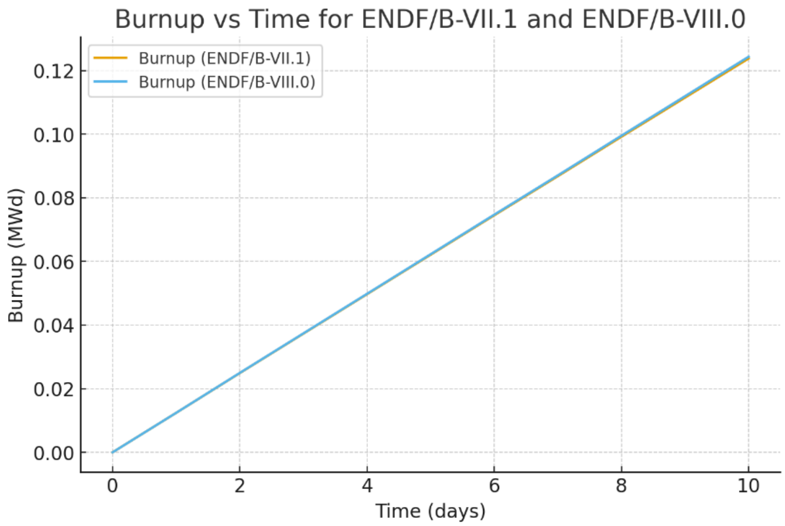

Figure 5: Comparison of burnup accumulation over 10 EFPD.

Discussion

Results indicate convergence between ENDF/B-VII.1 and ENDF/B-VIII.0 for NIRR-1's low-burnup thermal regime: U-235 depletion and burnup show <0.5% short-term differences, while Pu-239 buildup is systematically ~0.2–0.4% higher in ENDF/B-VIII.0 due to CIELO updates (Brown et al., 2018). These are far smaller than in high-burnup LWRs (reactivity underestimation ~400–1000 pcm; Park et al., 2019; Kim & Wieselquist, 2021). For MNSRs, ENDF/B-VIII.0 offers refined accuracy without operational changes, supporting extended lifetime (~56 years LEU vs. ~50 HEU; Simon et al., 2021) and safeguards (low Pu buildup). It should be noted that extended simulations and experimental validation (flux measurements, post-irradiation exams) are recommended for long-term biases (Abdulaziz et al., 2021; Yahaya et al., 2017). This advances open-source Monte Carlo applications for African research reactors under non-proliferation efforts.

CONCLUSION

This study assessed the impact of ENDF/B-VII.1 and ENDF/B-VIII.0 nuclear data libraries on the depletion behaviour of the Nigeria Research Reactor-1 (NIRR-1) low-enriched uranium (LEU) core using the open-source Monte Carlo code OpenMC. Depletion simulations were performed over 10 effective full power days (EFPD) at 30 kW, focusing on actinide inventories (U-235 depletion, U-238 consumption, Pu-239 buildup) and burnup in a low-burnup thermal spectrum typical of Miniature Neutron Source Reactors (MNSRs). Results showed strong agreement between the libraries in this short-term, low-power regime. U-235 depletion displayed small oscillating differences (average absolute ~0.17%, maximum |0.32|%), with no dominant systematic bias. U-238 consumption was minimal and nearly identical (<0.1% net difference), while Pu-239 buildup exhibited a modest positive bias in ENDF/B-VIII.0 (~+0.22% average, up to +0.44% early). Burnup accumulation remained linear with negligible relative differences (<0.5%, averaging ~+0.27% higher in ENDF/B-VIII.0). These subtle effects, driven by CIELO-project refinements in U-235 fission, U-238 resonances, and Pu-239 fission/nu-bar data, are far smaller than discrepancies in high-burnup PWR benchmarks as stated. The differences fall within Monte Carlo statistical uncertainties and operational tolerances, supporting the adoption of ENDF/B-VIII.0 for improved long-term fidelity without requiring changes to core management, beryllium shim strategies, or safety margins. The study demonstrates the value of open-source tools like OpenMC for accessible, high-fidelity depletion analysis in resource-constrained settings, contributing to accurate lifetime predictions (~50–60 years or 200–900 EFPD), excess reactivity control (~4 mk initial), and safeguards compliance following HEU-to-LEU conversion. It advances nuclear data sensitivity quantification for African research reactors and aligns with IAEA/RERTR non-proliferation goals. Future work should extend simulations to full core lifetime, incorporate uncertainty propagation using ENDF/B-VIII.0 covariances, and pursue experimental validation (flux mapping, post-irradiation examination). Similar studies on other MNSRs would further validate modern library performance in low-burnup thermal systems. ENDF/B-VIII.0 proves reliable for NIRR-1 depletion modeling, with short- to medium-term differences too small to affect operations, while reinforcing the safe, extended utility of Nigeria's only research reactor through updated nuclear data.

REFERENCES

Abdulaziz, M., Simon, J., Ibrahim, Y. V., & Asuku, A. (2021). Comparative analysis of core life-time for the NIRR-1 HEU and LEU cores. *Progress in Nuclear Energy*, 139, 103830. https://doi.org/10.1016/j.pnucene.2021.103830

Asuku, A. (2023). Application of SCALE code for reactor physics analysis of the as-built Nigeria miniature neutron source reactor operating with low enriched uranium fuel. *Nuclear Engineering and Design*, 405, 112215. https://doi.org/10.1016/j.nucengdes.2023.112215

Brown, D. A., Chadwick, M. B., Capote, R., Kahler, A. C., Trkov, A., Herman, M. W., Sonzogni, A. A., Danon, Y., Carlson, A. D., Dunn, M., Smith, D. L., Hale, G. M., Arbanas, G., Arcilla, R., Bates, C. R., Beck, B., Becker, B., Brown, F., Casperson, R. J., Conlin, J., ... Zhu, Y. (2018). ENDF/B-VIII.0: The 8th major release of the nuclear reaction data library with CIELO-project cross sections, new standards and thermal scattering data. *Nuclear Data Sheets*, 148, 1–142. https://doi.org/10.1016/j.nds.2018.02.001

Chadwick, M. B., Herman, M., Obložinský, P., Dunn, M. E., Danon, Y., Kahler, A. C., Smith, D. L., Pritychenko, B., Arbanas, G., Arcilla, R., Brewer, R., Brown, D. A., Capote, R., Carlson, A. D., Cho, Y. S., Derrien, H., Guber, K., Hale, G. M., Hoblit, S., Holloway, S., ... Young, P. G. (2011). ENDF/B-VII.1 Nuclear Data for Science and Technology: Cross sections, covariances, fission product yields and decay data. *Nuclear Data Sheets*, 112(12), 2887–2996. https://doi.org/10.1016/j.nds.2011.11.002

Jonah, S. A., Jibiri, N. N., Funtua, I. I., & Mallam, S. P. (2005a). The Nigerian Research Reactor-1 (NIRR-1): Design, characteristics and applications. *Journal of Radioanalytical and Nuclear Chemistry*, 266(1), 83–88.

Jonah, S.A.; Balogun, G.I; Umar, I.M.; Mayaki, M.C. (2005). Neutron Spectrum parameters in irradiation channels of the Nigeria Research Reactor – 1 (NIRR – 1) for k_0 – NAA standardization. J. Radioanal. Nucl.Chem..266 (1). 83 – 88.

Kim, K. S., & Wieselquist, W. A. (2021). Reactivity underestimation of ENDF/B-VIII.0 compared with ENDF/B-VII.1 for the pressurized water reactor depletion analysis. *Transactions of the American Nuclear Society*, 124, 520–523. (Also available via OSTI: https://www.osti.gov/servlets/purl/1808386)

Park, H. J., Choi, S., & Shim, H. J. (2019). Comparison of ENDF/B-VIII.0 and ENDF/B-VII.1 in criticality, depletion benchmark, and uncertainty analyses by McCARD. *Annals of Nuclear Energy*, 131, 443–459. https://doi.org/10.1016/j.anucene.2019.04.012

Paul K. Romano, Andrew R. Siegel, Benoit Forget, & Kord Smith (2015). OpenMC: A state-of-the-art Monte Carlo code for research and development. *Annals of Nuclear Energy*, 82, 90–97.

Romano, P. K., & Forget, B. (2013). The OpenMC Monte Carlo particle transport code. *Annals of Nuclear Energy*, 51, 274–281. https://doi.org/10.1016/j.anucene.2012.06.040

Romano, P. K., Horelik, N. E., Herman, B. R., Nelson, A. G., & Forget, B. (2015). OpenMC: A state-of-the-art Monte Carlo code for research and development. *Annals of Nuclear Energy*, 82, 90–97. https://doi.org/10.1016/j.anucene.2014.07.054

Snoj, L., Zerovnik, G., & Ravnik, M. (2012). Computational analysis of the Nigerian Research Reactor-1 (NIRR-1) core with Monte Carlo methods. *Annals of Nuclear Energy*, 46, 71–79.

Yahaya, B., Ahmed, Y. A., Balogun, G. I., & Agbo, S. A. (2017). Estimating NIRR-1 burn-up and core life time expectancy using the codes WIMS and CITATION. *Results in Physics*, 7, 196–202. https://doi.org/10.1016/j.rinp.2016.12.013