Processing and Validation of ENDF/B-VIII Nuclear Data Library Using NJOY21 and OpenMC for Modeling and Simulation of VVER-1200 reactor

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Abstract: Nuclear data plays one of the most important roles in calculation and modeling of nuclear reactor configuration using both stochastic and deterministic methods. Both Monte Carlo and deterministic reactor physics codes require nuclear data in a specific format in which they have been designed to work properly. This requirement creates the need of a data processing code to process, convert and validate raw nuclear data provided by ENDF, JEFF etc. for their specific use in reactor physics code like MCNP, OpenMC, Serpent, Monk, WIMSD5b etc. The purpose of this study is to process raw nuclear data files in order to validate the critical experimental benchmark using the Monte Carlo code OpenMC and NJOY21 in order to perform the criticality analysis of VVER-1200. Solid geometry of nuclear core for simulation was constructed using Python API of OpenMC. All the OpenMC calculations of k_{eff} values with processed ENDF/B-VIII nuclear data library will be contrasted with the values obtained through experimentation, which are available in the International Critically Safety Benchmark Evaluation Project. The objective of this study is to process the ENDF/B-VIII nuclear data library for producing an ACE-hdf5 format that works with OpenMC and successfully completes a few important benchmark scenarios. In order to reduce all mistakes resulting from approximations in geometry modeling or cross-section processing, neutronic calculations for the VVER-1200 reactor were then performed using the three-dimensional, continuous-energy OpenMC Monte Carlo code.

Keywords: OpenMC, VVER-1200, Criticality, safety, neutronic, NJOY21

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1. Introduction

Benchmarks are a necessary tool for the nuclear community to verify the availability of appropriate experimental data and for a variety of objectives. These details are essential to ensuring that modeling tools satisfy nuclear industry standards. In essence, their functions include verifying nuclear cross-section data and libraries, verifying nuclear codes and models, and utilizing benchmarks to instruct users through code comparison and benchmarking exercises. In the field of nuclear engineering, simulation tools are based on different analytical methods namely neutron transport and neutron diffusion theory. Neutron transport theory can be solved deterministically or simulated using probabilistic methods [1]. These theories have their own pros and cons. In order to operate a nuclear reactor safely and reliably, accurate simulations play a pivotal role. The design, licensing, and normal operation of nuclear power plants (NPPs), as well as the nuclear fuel cycle facilities for enrichment and material processing, fuel manufacturing, and transportation, depend on precise modeling and simulation predictions of nuclear energy systems. Neutron cross sections data, fission product yields, decay constants, γ–ray yields, branching ratios, delayed neutron precursors, and other data are used to infer the nuclear physics of these systems [2].

A broad range-purpose nuclear library must contain all of these data. Nuclear data, along with their uncertainties, serve as the cornerstone of physics for a variety of calculations. These include designs for nuclear reactor cores and fuel, safety parameter assessments, inventories of radionuclides and decay heat for accident investigations, analyses for biological radiation shielding, assessments of material damage in structures, evaluations of decay heat upon reactor shut-down, and material safeguards. The prime challenge that nuclear data program managers face is integrating multiple systems and their respective communities of experts while organizing the delivery of enhanced nuclear data files [3], [4].

The nuclear data library under evaluation has been independently developed in various countries [5], [6], [7]. In the field of nuclear data research [7], major projects such as ENDF United States), JEFF (NEA Data Bank), JENDL (Japan), CENDL (China), and ROSFOND (Russia) all work diligently to develop and publish libraries containing a wide range of isotopes. ENDF and JEFF nuclear data library, highly favored by the nuclear community, are renowned for providing dependable, evaluated data for all neutron reactions. A new version of these data libraries was released in2018: ENDF/B-VIII. 0 and JEFF 3. 3 in 2017 with a significant upgrade in nuclear data for a number of nuclides. The ENDF-6 formats can be found on the IAEA website. The aim of this study is to create continuous energy neutron data libraries using the most up-to-date nuclear data and analyze them through criticality benchmarking and criticality calculations of the VVER-1200 reactor. The nuclear data processing code NJOY218-9 is used to generate and process all radionuclides in the ACE format (A Compact ENDF format). Furthermore, S(α,β) thermal scattering is used to guarantee that the libraries are processed correctly [8], [9].

NJOY21 [9] was used in this study to process ENDF/B-VIII.0 nuclear data files and produce ACE-formatted cross-section libraries specifically designed for OpenMC simulations. The VVER-1200 reactor's high-fidelity modeling, including the incorporation of decentralized control capabilities, was made possible by these libraries. The modeling framework, for example, explicitly incorporates regional power feedback loops and distinct control rod banks, each with its own feedback of temperature and xenon concentration. This modular design reflects the decentralized nature of reactor operating by enabling the region-wise use of processed nuclear data. Furthermore, thermal feedback and boron control strategies were allocated to core quadrants independently, yielding spatially resolved neutron flux and power distribution data that further validated the decentralized methodology.

This work is innovative because it uses NJOY21 and OpenMC to process and validate the ENDF/B-VIII.0 nuclear data library, specifically for the VVER-1200 reactor, an advanced Generation III+ design that is presently being installed at the Rooppur Nuclear Power Plant in Bangladesh. With a focus on developing fundamental capabilities in Bangladesh, this work is one of the first localized attempts to benchmark ENDF/B-VIII.0 in the context of VVER-1200 modeling pertinent to South Asia, even though similar tools have been used internationally. Additionally, a comparison between the ENDF/B-VII and ENDF/B-VIII data libraries and a set of 12 ICSBEP criticality benchmarks is conducted here [10], [11]. To give a quantitative evaluation of library correctness, the performance metrics—k_{eff} deviation, χ^2, and mean absolute error—are assessed. Furthermore, the study examines how the reactivity and neutron spectrum of the VVER-1200 control rod design are affected by the absorber materials B_4C and $Dy_2O_3 TiO_2$. This is a topic of practical interest for operational safety and power flattening.

The overarching goal of this study is to support Bangladesh's nuclear energy aspirations by helping to build a verified, localized simulation infrastructure using OpenMC that can be expanded to full-core modeling, burnup analysis, shielding studies, and safety evaluations. To the best of our knowledge, this work is among the first to analyze and validate the ENDF/B-VIII.0 library especially for the VVER-1200 reactor type with implications for the Rooppur Nuclear Power Plant in Bangladesh, even though NJOY and OpenMC have been employed in other settings. This provides technical and regional detail that isn't often covered in the literature that is currently available. A thorough OpenMC model of the VVER-1200 reactor core geometry is included in the study; this model can be used as a basis for additional localized reactor study, such as burnup, shielding, and transient studies. In situations where such tools are still being developed or adopted, this modeling effort helps to create a dependable OpenMC simulation infrastructure.

A comprehensive overview of this study starts by: i) evaluating the processed libraries based on the critical benchmarks model outlined in the ICSBF handbook10; and ii) comparing these libraries with the VVER-1200 reactor to assess key neutronic parameters. The Department of Nuclear Science and Engineering at MIST has launched studies in reactor analysis [12]. This work describes neutronic calculations on a steady-state fresh core of the chosen PWR, the VVER 1200.

2. Methodology

The development and dissemination of nuclear data libraries, along with the assessment of code and data performance, is a worldwide endeavor. In this section, the methodology of nuclear data files generation and the benchmarking of the OpenMC code [13], [14], [15] is explained.

3. Nuclear Data File Generation

The Nuclear Data Processing Code, like NJOY, is extensively utilized for converting raw nuclear data from the Evaluated Nuclear Data Files (ENDF) format into formats that are beneficial for a variety of practical applications. These uses include nuclear waste management, nuclear medicine procedures, biological radiation shielding, criticality safety, fission and fusion reactor analysis, and more. NJOY is a flexible computer program that was used to convert evaluated nuclear data in the ENDF format into various libraries essential for criticality and shielding calculations. ENDF format libraries consist of computer-readable files containing nuclear data. These files describe several aspects such as nuclear cross sections, energy and angle distribution of reaction products, various nuclei produced during reactions, decay modes and product spectra of radioactive nuclei, as well as estimated errors in these data.

The NJOY21 nuclear data processing code adeptly processes the ENDF format, seamlessly converting it into an ACE format. They must be transformed into appropriate formats for various applications, like transportation or activation calculations, utilizing multi-group, point-wise, deterministic, or Monte Carlo methodologies. The newest version of NJOY is called NJOY21, representing a significant upgrade from its previous iteration, NJOY2016. The diagram in Fig. 1

shows the NJOY21 sequence for processing the most recent libraries in ACE format. Further information regarding the processing of nuclear data can be located elsewhere in references [8],[9].

In the NJOY21 processing code, MODER is utilized for converting between binary and ASCII modes, while RECONR is employed for reconstructing point-wise resonance cross-sections based on resonance parameters. The module's cross-section accuracy is approximately 0.5% (err=0. 005). BROADR is employed to generate Doppler-broadened and finely-pointed cross-sections. The precision of the cross-section in the module is around 0. 5% (err=0. 005). HEATR is employed for the creation of heat production cross-sections at specific points. GASPR is utilized for incorporating reactions that produce gas. THERMR is utilized for generating cross-sections for point-wise neutron scattering specifically in the range of thermal energy. PURR is utilized to generate tables of probability for incorporating the impacts of self-shielding into MCNP. ACER is employed for creating libraries in ACE format. In addition to the modules previously listed, there are various prominent modules that are utilized to validate the generated neutron cross-sections. ACER is particularly beneficial for conducting a range of consistency checks and generating multiple plot files.

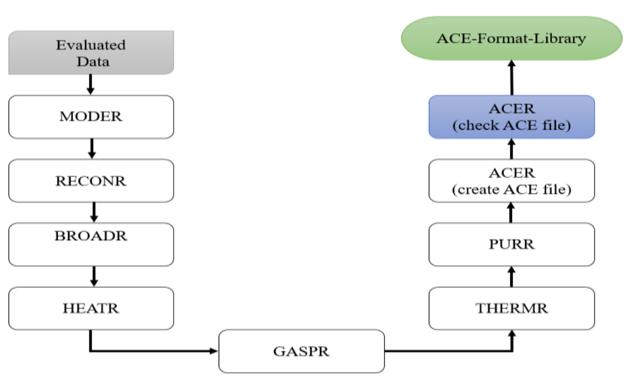

Figure 1. The nuclear data processing code NJOY21's flow chart.

Criticality benchmark problems

The OECD-NEA International Criticality Safety Benchmark Evaluation Project (ICSBEP) [10], [11] provided several of the criticality benchmarks used in this investigation. In order to promote computational validation of models, simulations, and nuclear data in support of criticality safety and reactor physics applications, the ICSBEP actively provides international review, preservation, and dissemination of essential benchmark data. The meaning of the ICSBEP terms is shown in Table 1. Among 119 benchmarks [11], twelve different benchmarks were considered to validate the ENDF/B VIII nuclear data library. For comparison purposes, ENDF/B VII nuclear data library is also used for criticality calculation. Main fissionable isotopes containing HEU, LEU, IEU and plutonium/uranium mixture systems were considered. The fissile substance in its physical state consists of metal (MET0, compound (COMP) and solution (SOL). In the case of the neutron spectrum, the spectrum consists of thermal (THERM), intermediate (INTER) and fast (FAST). Twelve benchmarks were modeled with Monte-Carlo code.

4. Simulation Code

In this research, the Monte Carlo OpenMC code was utilized [12], [13], [14]. The basic input and output structure of OpenMC code is shown in Fig. 2. The k_{eff} parameter calculations for criticality were carried out using two ENDFB/B libraries. 3000 iterations were conducted with a nominal size of the source of 10,000 particles per cycle (Fig. 3). To minimize statistical error and guarantee a homogeneous distribution of neutron sources, the first 100 cycles were skipped. Power iteration for OpenMC criticality calculation of the mean value of k is given in Fig. 3. Various aspects of the OpenMC code have been mentioned in Fig. 2.

Table 1. ICSBEP utilized abbreviations.

Abbreviation	Meaning
Fissile material	
HEU	High enriched uranium ($^{235}U \geq 60$ wt%
IEU	Intermediate or mixed enrichment uranium (60 wt%>^{235}U>10wt%)
LEU	Low enriched, natural, or depleted uranium ($^{235}U \leq 10$ wt%)
PU	Plutonium
Mix ^{233}U	Mixed uranium and plutonium Uranium ^{233}U systems
Physical form of fissile material	
MET	Metal
SOL	Solution
COMP	Compound system, e.g. lattice in water
Spectrum	
FAST	Fast system (≥50% of fissions above 100 keV)
THERM	Thermal system (≥50% of fissions below 0.625 eV)

Due to the evolving nature of the code, numerous changes have been made and continue to be implemented. The OpenMC code is readily accessible under an open source license, fostering greater collaboration within the nuclear community. Similar to MCNP [8], OpenMC monitors collision, absorption, and track-length estimators of k_{eff}. It subsequently computes a minimum variance combined estimator by utilizing the covariance matrix of these three single estimators. Moreover, the user is able to specify a mesh over which the Shannon entropy can be computed to confirm the convergence of the source distribution. OpenMC is a Monte Carlo particle transport code used for criticality and shielding calculations. It can be quite handy for creating 3D models through constructive solid geometry utilizing second order surfaces. OpenMC requires at least four input files: geometry, material, settings, and tallies files. The variety of issues such as the quantity of particles in the simulation, the number of batches and generations, and the convergence criterion are some of the elements to consider. The parameters are defined in the settings input file. The Tally input file allows for a wide range of tallies to be used in the calculation of flux, reaction rates, etc. During simulation. The Monte Carlo method is frequently utilized for solving neutron transport issues that emerge in reactor systems. The Monte Carlo method involves simulating particle histories to derive the average behavior of particles through the simulation of a large number of particle histories.

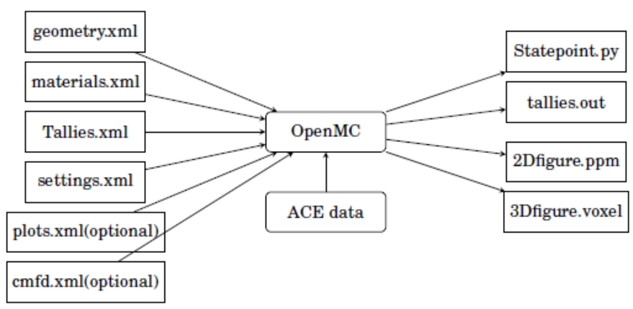

Fig. 2. OpenMC input output structure.

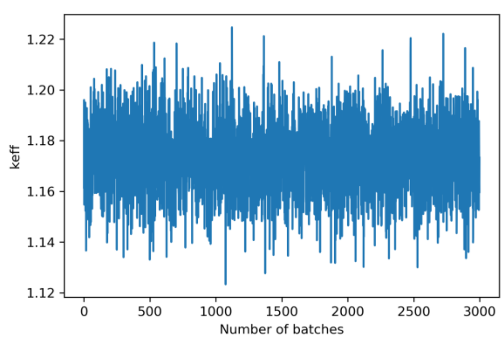

Fig. 3. Power iteration for Monte Carlo critical calculation of mean value of k_{eff}.

5. Simulation of VVER-1200

In 2011, the Russian state nuclear corporation, Rosatom, signed an intergovernmental agreement for the construction of two 1200 MWe nuclear reactors at Rooppur. These facilities are intended for the Bangladesh Atomic Energy Commission (BAEC). The government is in the process of establishing two modern VVER-1200 reactors with advanced safety measures at the Rooppur Nuclear Power Plant in Ishwardi, Pabna, Bangladesh to meet the Sustainable Development Goals (SDGs) for a green environment from nuclear energy [16], [17]. This plant, situated in a region with a population of 160 million, aims to generate 9% of its electricity from nuclear power. By the middle of the next decade, it also plans to lessen its reliance on fossil fuels.

The initial Rooppur NPP is part of the newest Generation III+ iteration of Russian VVER (Water-cooled Water-moderated Power Reactor) technology, specifically the AES-2006 (VVER-1200, V-392M) technology, incorporating site-specific safety enhancements. The VVER-1200 reactor technology is an evolution of the VVER-1000 reactor system. The latest updates in VVER-1200 technical characteristics, stemming from VVER-1000 technology, include [18], [19], [20, [21]:

- By increasing the RPV wall's height from 10,897 mm to 11,185 mm, inner diameter from 4150 mm to 4250 mm, and wall thickness (core shell) from 192.5 mm to 197.5 mm, the radiation exposure on the RPV wall is significantly reduced. There is an increase in fuel length from 3530 mm to 3730 mm. Reactor power increases as a result of this modification, although the total number of fuel assemblies (FAs) in the reactor core remains constant at 163. Some technical parameters of VVER-1200 are depicted in Table 2.

Table 2. Technical parameters of VVER-1200.

Fuel rod pitch (cm):1.275
FA height (cm): 353
Hole diameter (cm): 0.15
Fuel pellet diameter (cm): 0.757
Cladding inner diameter (cm): 0.773
Cladding outer diameter (cm): 0.910
Fuel pellet material: UO_2
Fuel pellet density (g/cm^3): 10.4-10.7
Cladding material Alloy: Zr+1%Nb
Control rod diameter (cm): 0.7
Control rod clad outer diameter (cm): 0.82
Control rod material: B_4C & $Dy_2O_3TiO_2$
Control rod density (g/cm^3): 1.8 & 5.1
Clad material Steel: 06x18H10T
Control rod length (cm): 350

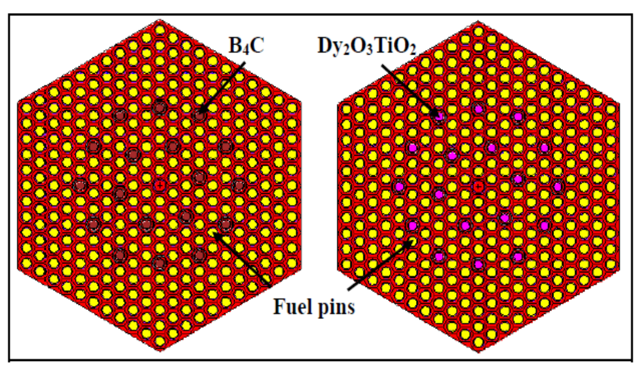

Fig. 4. Fully inserted two different cluster rods in fuel assemblies of VVER-1200.

- In the VVER-1000 reactor, there are 61 control rods which are divided into 10 groups or banks. On the other hand, the VVER-1200 design can accommodate up to 121 control rods that are divided into 12 groups. The absorbing materials used are B_4C and $Dy_2O_3TiO_2$, with only B_4C employed in VVER-1000.

The effect on the value of multiplication factor of both absorbing materials was tested in a fuel assembly lattice with fully inserted control rods and non-leakage neutrons system as shown in Fig. 4 [21]. Table 3 shows the main features of the VVER-1200 reactor. Acknowledging Bangladesh's intention to engage an overseas vendor or vendors for the supply of nuclear plants, it is advisable to plan for a certain level of localization in the manufacturing and operation processes. One task that can be actively pursued in localization involves conducting reactor analysis.

Table 3. Technical Characteristics of Rooppur NPP reactor Unit 1 and Unit 2.

Parameter	Value
Reactor nominal thermal power	3200 MW
Maximum utilization factor	Over 90%
Operation mode	Base load
Service life	At least 60 years
Maximum linear heat flux	420 W/cm
Time of fuel operation cycle	4 to 5 years
Period between refueling	12 months

This analysis encompasses criticality, burnup, shielding, and accident analysis of the reactor. The advancement of expertise in this field will contribute to enhancing both the economic and safety aspects involved in the construction and operation of a nuclear reactor. With this perspective in mind, a neutronic analysis of the VVER 1000 reactor was commenced. The analysis was conducted using OpenMC for the cold zero power state at the start of the cycle with the specifications acquired from the publicly available literature. The fuel assembly (FA) displayed in Figure 4 consists of 312 fuel rods, 18 guide tubes, and one instrumentation tube. The fuel pin of the VVER-1000 reactor boasts a unique structure that sets it apart from other PWRs. As seen in Figure 5 [12], [21], its fuel pellet features a center hole that is filled with helium. By lowering center temperatures and creating a free volume into which any fission gas that is released can expand, it lowers internal pressure. The material of the fuel pin is uranium dioxide (UO_2).

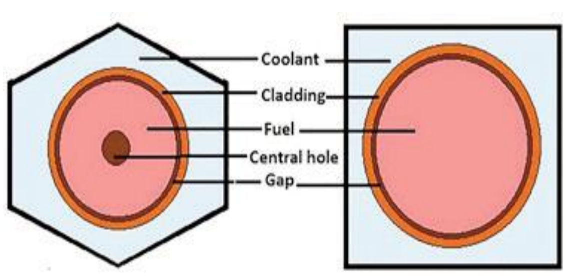

Fig. 5. Fuel pellet of VVER-1200.

The VVER-1200 is an improvement over the VVER-1000 in terms of plant performance and safety. In actuality, the VVER-1000 and VVER-1200 fuel assemblies have the same architecture. The fuel column height, which is 3730 mm for the VVER-1200 and 3530 mm for the VVER-1000, is the only difference between both. The fuel assembly model used for VVER-1000 under consideration is TVS-2 and for VVER-1200 is TVS-AES-2006. One of the major differences between VVER-1000 and VVER-1200 models is the cluster regulating system used. The new design of the cluster regulating system is totally different in number and absorbing material used than in VVER-1000. The 61 control rods in the VVER-1000 are arranged in 10 groups, but the 121 control rods in the VVER-1200 are arranged in 12 groups. In the case of nominal power, one group is a working group which is maintained in the regulation range of the core height, other groups are in the top position. There is only one absorbing material in VVER-1000 which is B_4C, while in VVER-1200 the rods are designed to have two absorbing materials; B_4C and $Dy_2O_3TiO_2$ as in Fig. 4. A horizontal cross-section OpenMC model of the VVER-1200 assembly is shown in Figure 6. The VVER-1200's fuel components, including all gaps and Zirconium alloy E110 [E110 is a zirconium–1% niobium (Zr-1Nb) alloy] cladding, are precisely modeled.

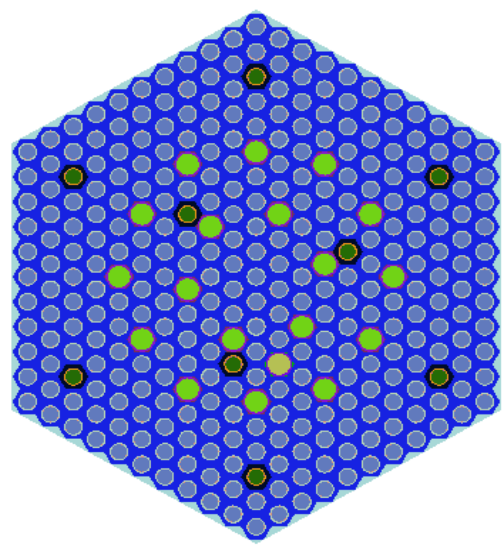

Fig. 6a. Horizontal cross-section OpenMC model of the VVER-1200 assembly.

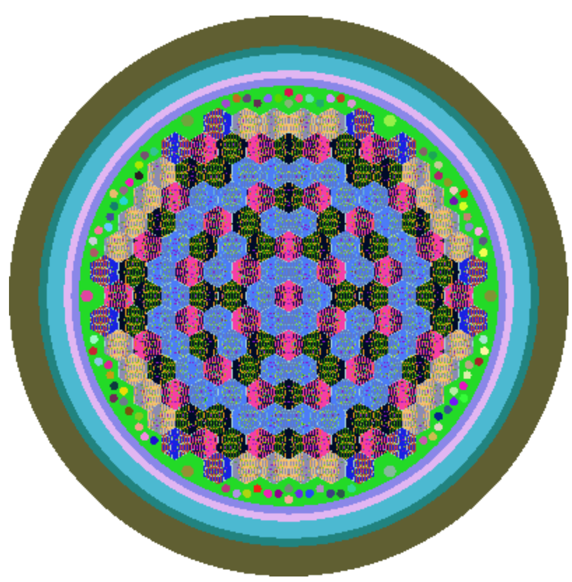

Fig. 6b. Top view of OpenMC model of the VVER-1200 core.

6. Stability Analysis of the Simulation System

The stability and dependability of the findings across different iterations and modeling settings is a crucial component of Monte Carlo reactor simulations. The following steps were done to examine the simulation system's stability in order to guarantee the validity of our conclusions. After a 100-cycle idle period to allow for source convergence, the OpenMC simulations were executed with 3000 active cycles and 10,000 particles each cycle. Fission source geographical distribution was tracked using Shannon entropy calculations. The source distribution attained statistical stability, as seen by the convergence of entropy profiles across simulated batches. The k_{eff} statistical uncertainty stayed within ±0.00016, suggesting a solution that was well-converged. Both the ENDF/B-VII and ENDF/B-VIII nuclear data libraries were used for the simulations. The system response is consistent and not too sensitive to modest fluctuations in cross-section data, as evidenced by the k_{eff} values (Table 5), which varied slightly between the two but remained within acceptable boundaries. This implies numerical stability while using NJOY21 to process input libraries and then using them in OpenMC.

7. Results and Discussion

The ENDF/B-VIII nuclear data has been processed with NJOY21 code. The results of cross-sections of ENDF/B-VIII along with ENDF/B-VII are shown in Figs. 7-8 for U-235 and U-238, respectively. It is evident from Figs. 7 and 8 that the cross-sections at resonance energy levels are well resolved in case of ENDF/B-VIII compared to ENDF/B-VI. Typical information displayed on the screen at the conclusion of the OpenMC simulation is displayed in Table IV. The HDF5 output file would also contain all of the data that was printed to standard output.

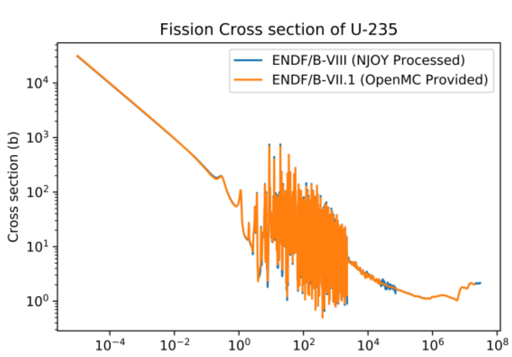

Fig. 7. Fission cross-sections of U-235.

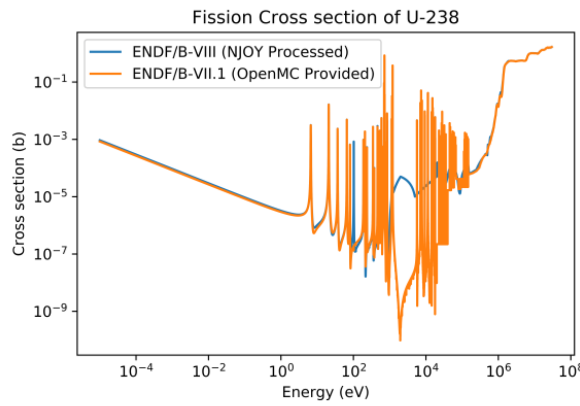

Fig. 8. Fission cross-sections of U-238.

Table 4. Summary of standard output of OpenMC code.

```
=========>TIMING STATISTICS    <==========
Total time for initialization    = 2.8074e+00 seconds
Reading cross sections           = 2.7922e+00 seconds
Total time in simulation         = 1.5504e+03 seconds
Time in transport only           = 1.5479e+03 seconds
Time in inactive batches         = 2.1927e+01 seconds
Time in active batches           = 1.5285e+03 seconds
Time synchronizing fission bank = 1.4598e+00 seconds
Sampling source sites            = 1.2370e+00 seconds
SEND/RECV source sites           = 2.1848e-01 seconds
Time accumulating tallies        = 1.7358e-03 seconds
Total time for finalization      = 4.0960e-06 seconds
Total time elapsed               = 1.5532e+03 seconds
Calculation Rate (inactive)      = 22803.4 particles/second
Calculation Rate (active)        = 19300.6 particles/second

=============>RESULTS    <=============
k-effective (Collision)     = 1.17332 +/- 0.00022
k-effective (Track-length)  = 1.17330 +/- 0.00026
k-effective (Absorption)    = 1.17321 +/- 0.00019
Combined k-effective        = 1.17325 +/- 0.00016
Leakage Fraction            = 0.00000 +/- 0.00000
```

Table 5 shows that the value of k-infinity in the case of B_4C control rod is lower than the case of $Dy_2O_3TiO_2$. This is due to the fact that B-10 has a higher absorption cross section than Dy-164 as it can be noticed in Fig. 9 obtained using ENDFPLO. This justifies the use of 121 rod clusters in VVER-1200 as compared to 61 rod clusters in VVER-1000. The increase in the number of rod clusters and the use of a material with lower absorption cross section but higher scattering cross section will lead to a smaller depression in the neutron flux and the power in the vicinity of the fuel rod. This is due to the fact that the rods have a gray absorber material. This results in a flatter neutron spectrum profile (Fig. 10) and better power distribution in the core. Obtained neutron spectrum agrees well with neutron spectrum by other codes.

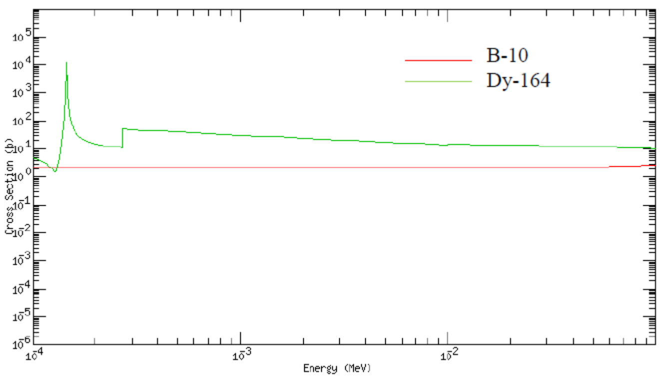

Fig. 9. Absorption cross section of B-10 and Dy-164 isotopes.

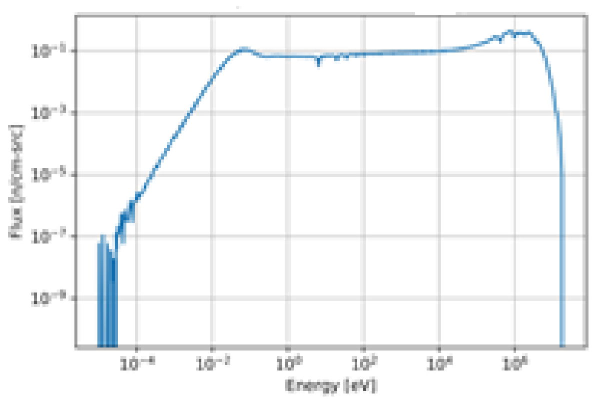

Fig. 10. Neutron energy spectrum profile.

Table 5. Values of k_{inf} corresponding to two different cluster rod systems.

Model	k-infinite	Standard deviation
Fully inserted B_4C control rods	1.01674	0.00007
Fully inserted $Dy_2O_3TiO_3$ control rods	1.09130	0.00007

The results of OpenMC k_{eff} calculations with ENDF/B-VI and ENDF/B-VIII data libraries for the some benchmarks are given in Table 6. Tables 7 and 8 provide the average value of uncertainty for each of the benchmark systems. The results obtained from OpenMC code are comparable with the benchmark values.

Table 6. OpenMC calculations of k_{eff} values with two data libraries and benchmark k_{eff}.

Case Name	Benchmark keff	ENDF/B-VI	ENDF/B-VIII
HEU-MET-FAST-001	1.00000 ± 0.00100	0.99782 ± 0.00014	1.00005 ± 0.00009
HEU-MET-INTER-006-case-1	0.99770 ± 0.00080	0.98623 ± 0.00012	0.98936 ± 0.00015
HEU-SOL-THERM-013-case-1	1.00120 ± 0.00260	0.99944 ± 0.00013	0.99838 ± 0.00018
IEU-MET-FAST-001-case-1	0.99890 ± 0.00100	0.99753 ± 0.00010	0.99898 ± 0.00018
IEU-MET-FAST-002	1.00000 ± 0.00300	1.00337 ± 0.00012	0.99711 ± 0.00011
IEU-MET-FAST-003-case-2	1.00000 ± 0.00170	0.99912 ± 0.00010	0.99981 ± 0.00010
IEU-MET-FAST-004-case-2	1.00000 ± 0.00300	1.00305 ± 0.00010	1.00394 ± 0.00011
LEU-SOL-THERM-001	0.99910 ± 0.00290	1.01007 ± 0.00012	1.00191 ± 0.00013
PU-MET-FAST-001	1.00000 ± 0.00200	0.99777 ± 0.00010	0.99907 ± 0.00012
U233-MET-FAST-002-case-1	1.00000 ± 0.00100	0.997611 ± 0.00010	1.00023 ± 0.00015
U233-SOL-INTER-001-case-1	1.00000 ± 0.00430	0.96096 ± 0.00015	0.98221 ± 0.00017
U233-SOL-THERM-001-case-1	1.00000 ± 0.00310	0.99842 ± 0.00016	0.99936 ± 0.00011

The χ^2 and ⟨|Δ|⟩ metrics can be calculated using the following formulas [3]:

$$\chi^2=\sum \frac{(k_{cal}-k_{exp})/\delta k_{exp})^2}{n} \quad \ldots \quad (1)$$

$$\langle|\Delta|\rangle=\sum |k_{cal}-k_{exp}|/n \quad \ldots. (2)$$

With the exception of a few benchmark situations, as shown in Table 6, the ENDF/B-VIII.0 yields better agreement with the experimental in most cases than ENDF/B VII. However, for the remaining circumstances, a similar behavior between the two libraries is emphasized.

Table 7. The χ^2 between calculated and benchmark k_{eff}.

Type of benchmarks	ENDF-VI	ENDF/B-VIII
HEU benchmarks	4.535	2.533
IEU benchmarks	2.590	1.340
LEU benchmarks	2.549	2.586
Pu benchmarks	3.166	3.310
^{233}U benchmarks	2.665	1.623

Table 8. The average difference between the computed and benchmark k_{eff} in pcm (⟨|Δ|⟩).

Type of benchmarks	ENDF-VI	ENDF/B-VIII
HEU benchmarks	306.925	253.050
IEU benchmarks	172.012	144.133
LEU benchmarks	284.678	273.723
Pu benchmarks	325.67	298.997
^{233}U benchmarks	219.556	188.167

The results of k_{eff} of VVER-1200 (initial homogeneous core) with OpenMC-0.11.0 were found to be 1.211187 (-0.07687) for ENDF/B-VIII and 1.21489 (-0.07989) for ENDF/B-VI, respectively. The k_{eff} results between two nuclear data libraries are in good agreement. Based on this preliminary study, OpenMC can be used for modeling and simulation of full core subject to availability of actual technical data from the ROSATOM/BAEC.

Despite the fact that this study concentrated on nuclear data processing and conventional Monte Carlo simulation, computational intelligence techniques could be applied to expand the methodology. Future developments of this study might combine computational intelligence methods like fuzzy logic and neural networks with conventional Monte Carlo-based simulations. Machine learning models, for instance, might greatly speed up parametric investigations by acting as stand-ins to forecast keff or flow profiles under various input conditions. Similarly, control strategies could incorporate fuzzy rule-based systems to handle reactor operation uncertainty. A promising route to intelligent, real-time reactor modeling and control is provided by such hybrid approaches. Such hybrid approaches could greatly improve sophisticated reactor systems like the VVER-1200's modeling, control, and real-time decision-making capabilities.

The present study has used the OpenMC Monte Carlo code in conjunction with the most recent version of the NJOY21 processing code to perform benchmark computations for the recently released ENDF/B-VIII data library. All things considered, it can be concluded that ENDF/B-VIII performs better than competing libraries in a few particular energy zones. The ENDF/B-VIII cross-section libraries, which include ENDF/B-VI, have statistically identical coefficients and standard deviations (σ). Nevertheless, there are minor differences in the k_{eff} values. To support a wide range of OpenMC applications, the ENDF/B library has been processed in ACE format at different temperatures. Procedures for quality assurance have been put in place to guarantee the processed ENDF file's integrity. After a thorough quality assurance process that involved: i) ACE file verification, ii) compiling NJOY21 warnings and messages, and iii) applying a wide range of criticality benchmarks to give a broad evaluation of the ENDF/B-VIII nuclear data library's overall performance, the processed data were judged acceptable. Together with NJOY21 and OpenMC, the new library's performance in regard to the criticality benchmarks shows good agreement with the criticality data and benchmark values for vver-1200. To create a comprehensive core model of vver-1200 using OpenMc and MCNP codes, more research is being conducted. Benchmark calculations have been performed for the latter release of the ENDF/B VIII information library within the show consideration using the OpenMC Monte Carlo code and the most recent version of the NJOY21 preparation code. By comparing with the other libraries, it can be inferred that ENDF/B VIII is advanced in a few instances for each specific vitality locale. Measurably ambiguous are the standard deviation (σ) and coefficients among the ENDF/B-VIII cross-section libraries that include ENDF/B-VI. Be that as it may, a few differences can be observed within the values for k_{eff}. QA methods have been connected to ensure the quality of the ENDF prepared record. The handled information were judged to be worthy concurring to a broad QA method:

i) checking Pro records, ii) compiling notices and messages from NJOY21, iii) utilizing an extended suite of criticality benchmarks giving a common sign of the in general execution of ENDF/B-VIII atomic information library. The execution of the modern criticality benchmarks library in conjunction with NJOY21 and OpenMC, appears to be great for utilizing the benchmark values and criticality information of VVER-1200.

8. Conclusions

In this work, the ENDF/B-VIII nuclear data library is successfully processed using NJOY21 and integrated into OpenMC for high-fidelity neutron transport modeling of the VVER-1200 reactor. The effectiveness and dependability of ENDF/B-VIII in contemporary Monte Carlo reactor simulations were confirmed by the simulations' exact replication of important reactor physics characteristics as the effective multiplication factor (k_{eff}). The results highlight the significance of regularly updating nuclear data libraries and simulation toolchains in regulatory, design, and operational contexts, going beyond technical validation. Safety margins, fuel cycle optimization, and radiation protection regulations are all directly impacted by the accuracy of reactor core calculations. Therefore, it is advised that nuclear utilities and national regulatory bodies give ENDF/B-VIII or later libraries top priority in safety evaluations and licensing frameworks, particularly for Generation III+ reactors such as the VVER-1200.

ENDF/B-VIII.0 data processed by NJOY21 and OpenMC can be used to further improve the simulation workflow through the use of computational intelligence approaches. To reproduce intricate simulation findings, neural networks can be used as surrogate models, allowing for quick predictions over a large design space. Fuzzy logic systems can incorporate linguistic or uncertain elements to aid with reactor control decision-making. The creation of predictive maintenance frameworks, decentralized control optimization, and real-time reactor monitoring for VVER-1200 and related reactor designs is made possible by these integrations.

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