Comprehensive Neutronic Assessment of the VSC-24 Cask for Potential Application in Dry Storage of VVER-1200 Spent Fuel

KAZI LABIBA RAHMAN AURNA, ROSHLAN RAHMAN DIPTO,
ABDUS SATTAR MOLLAH*

Department of Nuclear Science and Engineering
Military Institute of Science and Technology, Mirpur, Dhaka-1216
BANGLADESH
Corresponding author

Abstract: Spent nuclear fuels discharged from reactors are first stored in cooling pools, but due to limited pool capacity, dry storage systems are increasingly employed as an intermediate and long-term management solution Nuclear dry cask storage has high safety and it is capable of holding large volume of waste at low cost, and can be both stored and transported. The dry type storage cask VSC-24 for spent fuel of VVER type reactor was analysed with neutron transport calculations in this work. Criticality calculations were conducted with the Monte Carlo transport code OpenMC 0.15.2 and an H5 continuous energy cross-section library derived from ENDF/B-VIII.0 was implemented. Comparison to available literature was made for validation purposes in both spatial and energy space for the calculated neutron flux spectra. The effect of the thickness of the concrete outer layer of cask was also studied varying concrete thickness. The results provide a comprehensive guidance for shielding design, long term behaviour of storage and contribute to the safety assessment of dry cask storage. The Rooppur Nuclear Power Plant (RNPP) or other regional centralized facilities could use the VSC-24 cask for temporary dry storage. Its evaluation facilitates cross-compatibility tests between Russian and Western storage technologies and aids in the creation of a regulatory framework for the management of wasted fuel in Bangladesh. Furthermore, this study strengthens local expertise in the processing and storage of spent fuel by facilitating knowledge transfer and capacity building in nuclear engineering safety analysis.

Key-Words: - Nuclear, Dry Cask, Neutronics, Monte Carlo, OpenMC, Neutron, Shielding.

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1 Introduction

The storage pools at the reactor site house the radioactive spent fuels that are removed from the reactor at the conclusion of the planned cycles. Dry storage is commonly used for final disposal as the space in the pool is limited. While the dry storage technique has been used mainly in countries like the US, Canada, Russia and China, some studies have also been incorporated in South Korea, France and Germany. Reactors like VVER-400 fuels are stored in dry storage barrels with the Castor 440/84 type in the Czech Republic [1]. The fuels are eligible to be transferred and stored into the steel dry casks after having been kept in the spent fuel pool for a minimum of one year. As the fuel burns up, nuclear reactor fuels usually become less reactive. In essence, the drop in reactivity is caused by a rise in the concentration of fission products that absorb neutrons and a decrease in the concentration of fissile nuclides. Burn-up credit, which is currently of great interest in the field of nuclear criticality safety research, typically entails crediting fuel burn-up in order to account for this reactivity decrease for criticality safety evaluations and control of spent fuel.

After the removal from the water, this cask is filled with a filling gas and sealed using bolts or welding in order to enhance the heat transfer or conveyance developed within. Afterwards, they are inserted into a second cask section which will be of shield type and normally of concrete [2]. To reduce fuel's hazardous environmental effects, the storage time to a final disposal site. Besides, numerical tools should assist in safety assessment and provide long term performance predictions. The most accurate approaches in these predictions are Monte Carlo programs using automatic variant methods of decrease [3]. The level of radiation for the containers of the packaged nuclear materials in the case of storage may differ based on the standards of the concerned regulatory agencies. The standards for the containers in the case of storage resemble those for the containers in the case of transport in regard to the standards of Belgium [4]. Bangladesh has completed the building of the first unit of its initial nuclear power station, the Rooppur Nuclear Power station (RNPP), which is on the verge of commissioning. This facility is home to a Generation-III+ pressurized water reactor (PWR) with advanced passive safety features, notably the Russian-designed VVER-1200 variant. Managing spent nuclear fuel (SNF) safely and effectively has emerged as a major engineering and regulatory concern as nuclear energy grows globally. After pool cooling, the VVER-1200, a Generation III+ pressurized water reactor (PWR), produces large amounts of high-burnup spent fuel that needs strong dry storage systems. Spent fuel is stored dry in a variety of casks, including VSC-24, VSC-32, TN-24, TN-32, TN-40, HI-STORM 100, CASTOR® and CONSTOR® [5–13]. Originally developed for PWR assembly in the United States, the VSC-24 (Ventilated Storage Cask) system has shown decades of operational dependability. Fig. 1 shows projections for the discharge, reprocessing, and storage of spent fuel [11].

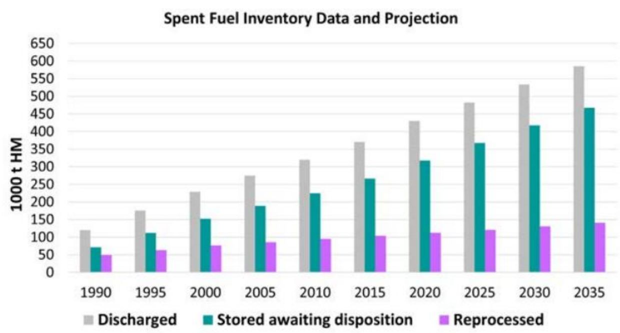


Fig. 1: Data and projections for the inventory of used nuclear spent fuel.

Its use with Russian-designed VVER fuel hasn't been thoroughly investigated, though. Through thorough Monte Carlo simulations and comparative research, this study seeks to assess the VSC-24's neutronic appropriateness for storing VVER-1200 spent fuel assemblies. Although the Rooppur Nuclear Power Plant (RNPP) in Bangladesh uses VVER-1200 reactors, local dry storage infrastructure is still being developed. Examining current cask technologies, like the VSC-24, can offer an affordable and reliable basis for temporary spent fuel management. By carrying out this research, evidence-based regulatory decision-making is supported and VVER-specific neutronic behavior inside alternate cask systems is better understood. In addition to giving comparative subcriticality and basket thickness data, benchmark results for non-US licensing studies, and a methodology for cross-reactor dry storage feasibility assessments, this study offers the first comprehensive neutronic analysis of modifying the VSC-24 cask for VVER-1200 spent fuel. At the international national conference, a preliminary work for VSC-24 was presented [14].

In this case, it should be noted that the neutronics simulation described below relates to the VSC-24 storage container that can be used for transport and storage of spent nuclear fuels which will be generated through the operation of a VVER type reactor has had been investigated. Results of this study were obtained using the current OpenMC 0152 Monte Carlo transport code [5] and H5 type continuous energy cross-section library based on ENDF/B-VIII.0 evaluated data were used in criticality calculations. Criticality calculation was done both directly and with a six factor formula. The spectrum of neutron flux both energy and spatial, were generated and compared with the literature. In addition, the effect of changes in the thickness of the concrete section, which is the outermost layer of the cask, on the reactivity was also investigated.

A research question has been established as follows: When filled with VVER-1200 spent fuel assemblies, can the VSC-24 cask remain subcritical ($k_{eff} < 0.95$)?

2 Methodologies

2.1 Material Definition

The Fresh and Spent fuel composition used in this model is listed in Table 1. The isotopic quantity and composition of the spent fuel are influenced by multiple factors, including fuel type, enrichment, initial fuel arrangement, loading and reactor operational history. In this study, the modeled VVER-1200 spent fuel cask is fueled with uniformly enriched fresh UO2 pellets and spent fuel isotopes (Petrovskiy et al., 2020).

Table 1. Fresh and Spent fuel composition.

Spent fuel Isotope	Relative Fraction (%)	Fresh Fuel
235-U	0.9	

238-U	97.35	UO_2 fresh, 4.3 % enriched
237-U	0.6	
239-Pu	0.54	
240-Pu	0.25	
241-Pu	0.15	
242-Pu	0.08	
237-Np	0.6	
241-Am	0.2	
243-Am	0.2	

2.2 VSC-24 OpenMC Model

Sierra Nuclear Corporation is a manufacturer of the VSC-24 [9, 11, 15]. The containers have been upgraded to hold 24 VVER-1000 hexahedral fuel assemblies. The VSC24 system has a 50-year lifespan in mind. Fuel assemblies known as MSBs (Multi-Assembly Sealed Baskets), which are also depicted in Figure 2, are supported by a cylindrical box inside the barrel [15]. This basket is positioned inside a VCC (Ventilated Concrete Cask), a cylindrical ring made of reinforced concrete with an outer diameter of 3.35 meters and a total height of 5.3 meters. The concrete cask is transported on and off the heavy haul transfer trailer using the MSB Transfer Cask (MTC), another component of the cask. Fig. 3 depicts the cask's component [9]. Table 2 displays several significant geometrical parameters [9]. The cask has two cover sections: a structural lid and a shielding lid. In addition to the steel protective part and the RX-277 neutron shielding material, there is a 76 mm steel disc structural cover. The cask is filled with helium gas at atmospheric pressure prior to the coverings being attached in order to improve heat transmission within.

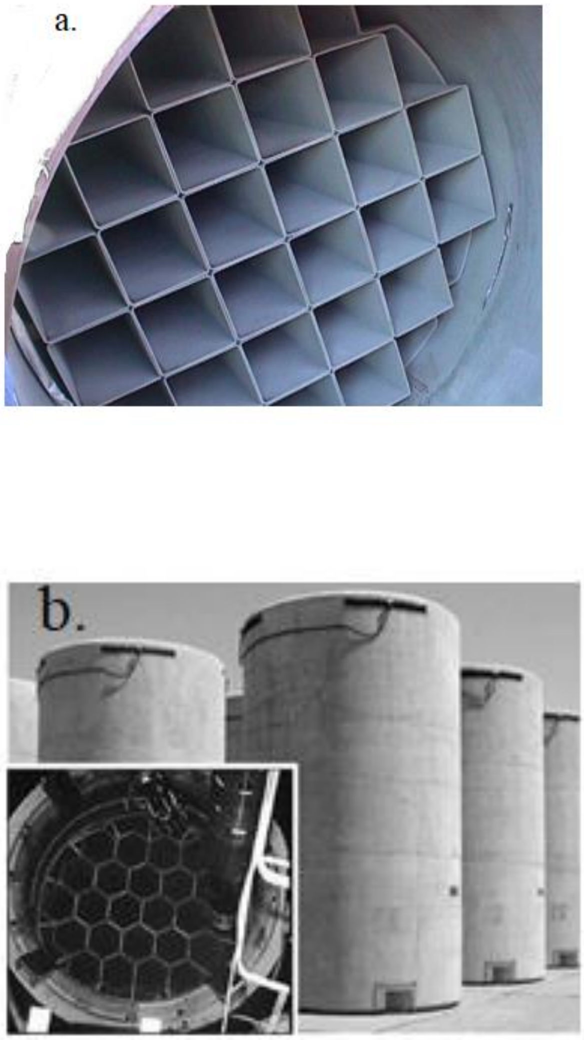


Fig. 2: Dry storage container VSC-24 PWR square lattice (a) VVER-1000 hexagonal lattice (b).

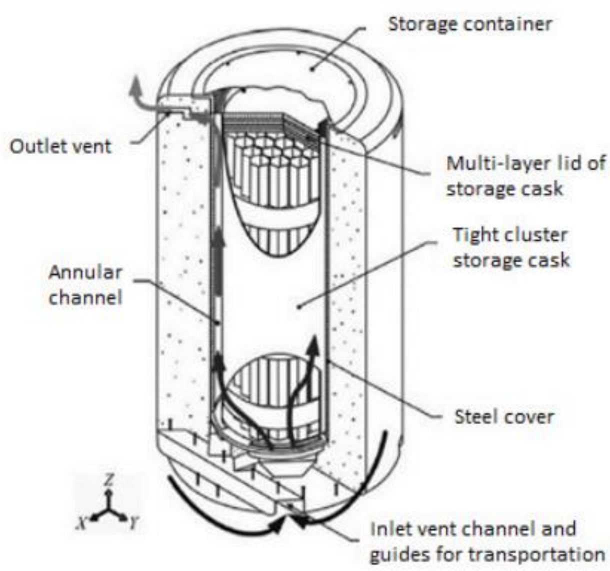


Fig. 3: Component of the VSC-24 Cask.

Table 2: VSC-24 cask parameters.

Parameter	**Value**
Fuel assembly	
Shape	hexahedron
Number of fuel rods	312
Length	3.837 m
Storage cask	
Num. of spent fuel assemblies	24
Height	4.973 m
Diameter	1.715 m
Inner medium	Helium
Pressure inside	1 atm
Container	
Height	5.809 m
Diameter	3.378 m
Width of annular channel	0.070 m

The VSC-24 cask model was developed using the final safety analysis report's geometric measurements [15-23]. The materials are used to determine the color of the axial and radial section views in Figures 4 and 5. The fuel bundles are housed in a hexahedral basket construction in the model's innermost region. It is specified that each fuel bundle must include uniformly clothed, fuel, air, and helium gas components.

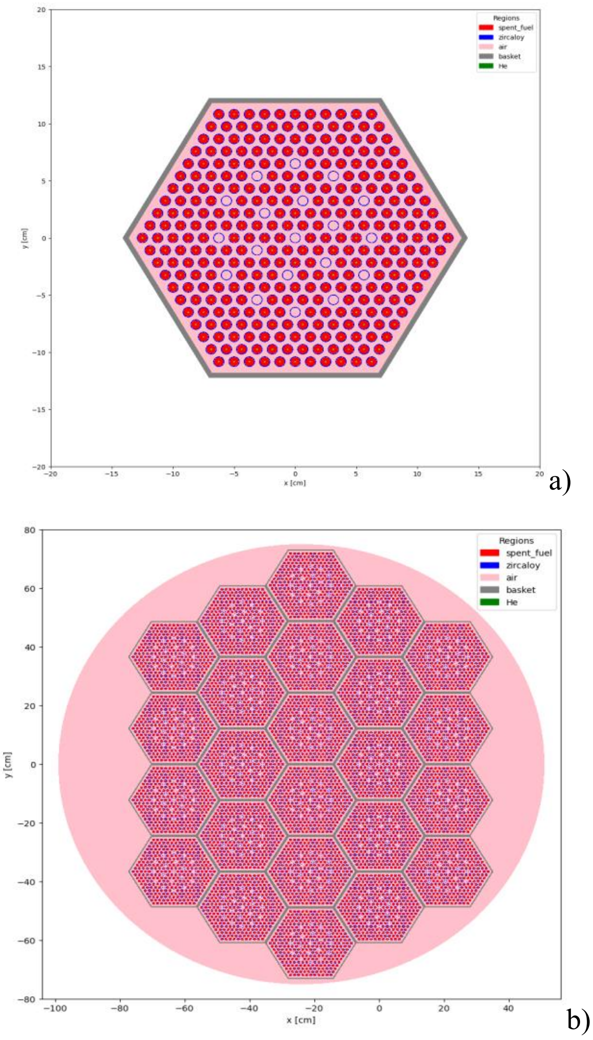
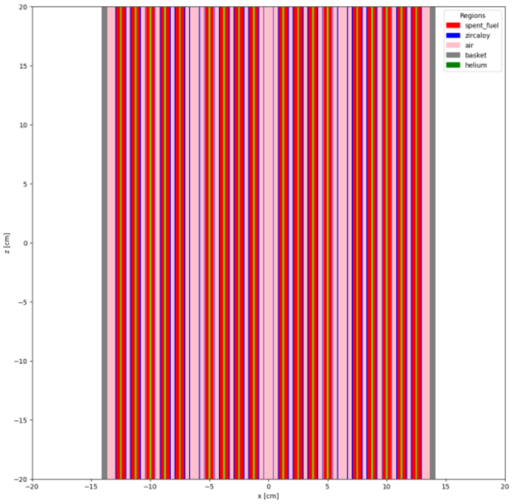
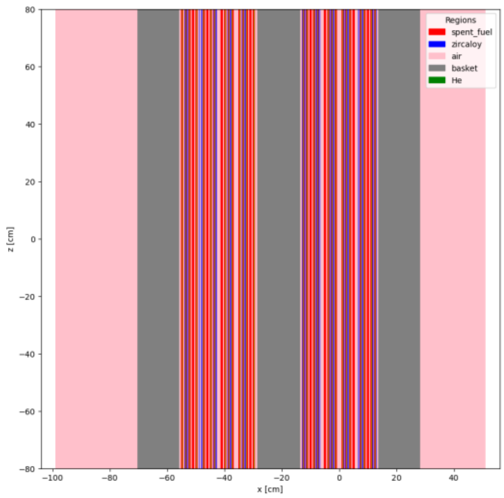


Fig. 4: VSC-24 Cask OpenMC model (a. Radial View of Assembly b.Whole Cask)



a)



b)

Fig. 5: VSC-24 Cask OpenMC model (a.Axial View of Assembly b. Whole Cask)

In our analysis, for fresh fuel loaded cask assessment, the modeled VVER-1200 core is fueled with uniformly enriched UO2 pellets, enriched to 4.95 wt.% 235U across all its 163 fuel assemblies [8]. Spent fuel composition was taken for 45 GWd/tU burnup of PWR reactor with 10 years cooling time [6]. Cross section data ENDF/B-VIII.0 neutron and photon libraries were used in OpenMC calculations. Fig. 6 displays an OpenMC model flow diagram.

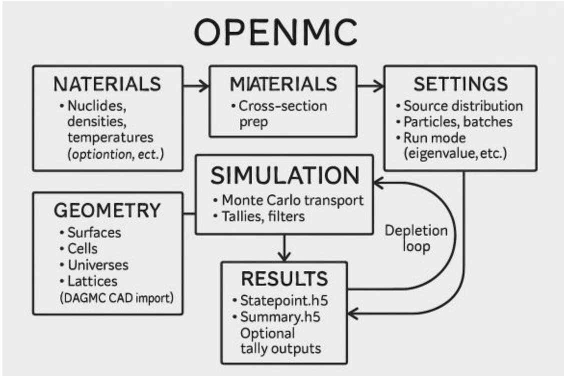


Fig. 6: OpenMC process flow diagram.

3 Results

3.1 Criticality Calculation

The criticality calculation using the OpenMC code and Continuous energy cross section libraries based on ENDF/B-VIII.0 of the VSC-24 cask, which consists of 24 fuel assemblies filled with helium gas and containing spent fuel with average 1-2 % enrichment, the final result of k_{eff} value was 0.37735 +/- 0.00053 and UO_2 fuels with an average of 4% enrichment k_{eff} value was 0.67651 +/- 0.00399 which are below the criticality limit. OpenMC calculations were carried out with 2E+5 particles and 30 inactive and 270 active cycles. This result is acceptable in terms of safety criteria (<0.95). Table 3 presents a comparison between two cases,

Table 3: Results of OpenMC calculations of keff

Cask	Fuel	k_{eff}
Helium filled	fresh, 4.3 % enriched	0.67651 +/- 0.00399
Helium filled	used, 45 GWd/tU burnup	0.37735 +/- 0.00053

3.2 Reaction Rates

The reaction rates occurring in different materials are plotted in figure 7. The Colors blue, orange, green, Red and purple represent fission, scatter, absorption, nu-fission and total interaction respectively. In X-axis material 1,2,3,4 and 5 denote spent fuel, Air, basket, Helium and Zircaloy respectively. From the graph we can see, around 10^9 fission reactions are taking place in the spent fuel, scattering reaction is more dominant in spent fuel. Total interaction is significantly lower in Helium. As usual nu-fission was only found in spent fuel.

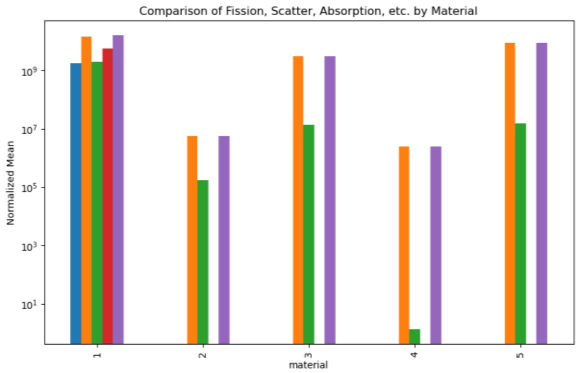


Fig. 7: Interaction Rates in Spent Fuel Cask Materials.

3.3 Neutron Energy Spectrum

Typical neutron flux quantity was found to be around 10^4 orders of magnitude. Neutrons with 1 MeV energy were most abundant in the cask which is in good agreement with literature.

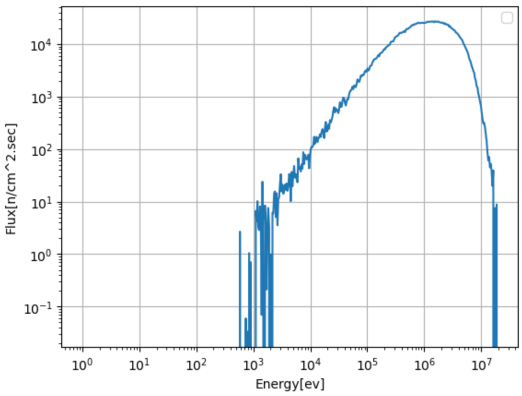


Fig. 8. Neutron Flux Spectrum.

Also using the Multigroup structure of Openmc, neutron flux was plotted as shown in Figs. 8 and 9. The energy groups were divided into 500 groups and likewise XMAS-172 and SHEM-361 were plotted. “XMAS-172” designed for LWR analysis, and “SHEM-361” is designed for LWR analysis to eliminate self-shielding calculations of thermal resonances. No significant difference was found between SHEM-361 and XMAS-172 indicating self-shielding plays no vital role in the case of our model.

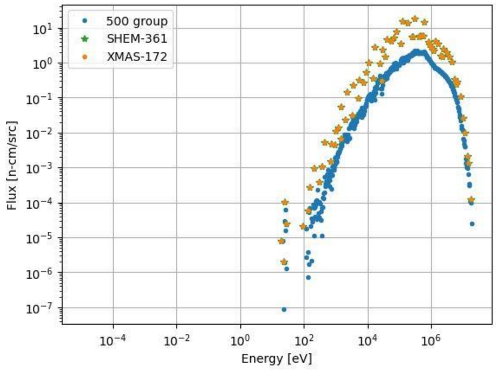


Fig. 9: Comparison between SHEM-300, XMAS-172 and 500 Groups fluxes.

For watt spectrum the governing equation is,

$$p(E')dE' = ce^{\frac{-E'}{\alpha(E)}} sinh\, sinh \sqrt{b(E')dE'}\, dE' \quad (1)$$

Here a and b are the parameters for the distribution and are given as the function of incoming energy of the neutron. They are interpolated on the incoming energy grid by a specific law of interpolation.

3.4 Search for Optimum Basket Thickness

This plot (Fig. 10) illustrates the relationship between basket thickness and the effective multiplication factor (k_{eff}). k_{eff} gradually increases from roughly 0.36 to 0.46 as the basket thickness increases from 0.15 cm to 3 cm. According to the trend, thicker baskets improve neutron moderation or reflection, which boosts k_{eff} and the neutron economy [22].

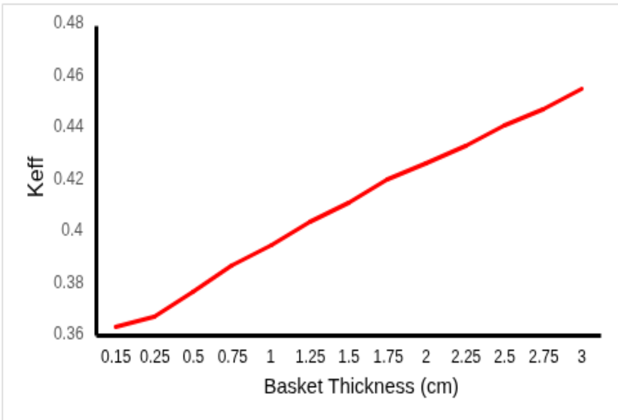


Fig. 10: k_{eff} Change with Basket thickness around the assembly.

3.5 Spatial Neutron Flux Distribution

Figure 11 shows the spatial distribution of normalized neutron flux. The flux steadily drops towards the edges (lighter shades) from its maximum in the centre (dark purple). The data show that the predicted flux peaks at the core center because of the higher fission density, while the flux decreases close to the boundaries where leakage takes place.

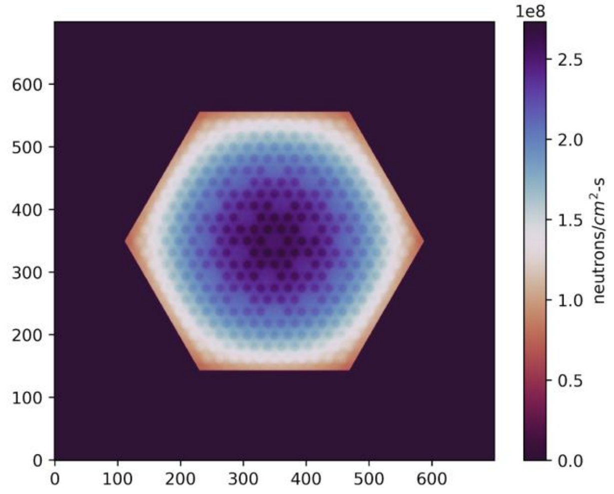


Fig.11: Spatial Flux distribution in one Assembly

4 Validation

The criticality result was validated with CSAS4 Code which was studied with similar material [7] and the neutron flux value was compared and found compatible [8]. The criticality found in reference 7 was found to be 0.324 which is near the keff value we found in our study and under the standard limit. The flux level found in reference 8 was found to be on average in order of 10^3-10^4 n/cm^{2}s^{-1} at top, bottom and side of TN-32 cask which is also found to be the range in our case.

5 Conclusion

The criticality calculation and neutronics analysis of the VSC-24 type storage casks were examined with both fresh fuel and spent fuel. The effective multiplication factor k_{eff} for helium filled spent fuel storage cask is calculated using the OpenMC Monte Carlo code package. Calculation was made for fresh (4.3% enriched) and spent (45 GWd/tU) fuel which composition was taken from the literature. Capacity of the reinforced concrete cask was 24 fuel assemblies placed inside a stainless steel basket. The level of criticality was below the safety margin which is below 0.95 for both cases. The flux spectrum for the cask was determined which was compatible with the literature. This study can be used as a preliminary analysis to determine the capacity of dry casks that can be used in the transport and storage of VVER type reactor fuels. Additional calculations showed that to comply with the keff < 0.95 request in the most conservative case (helium filled cask without burnup credit), the thickness of the stainless-steel basket structure should be 1.5 cm.

The effective multiplication factor (k_{eff}) stays below 0.95 when the VSC-24 cask is fully loaded with VVER-1200 spent fuel assemblies, according to simulation. This answers the study objective by confirming that the cask remains safely subcritical under typical storage settings, guaranteeing compliance with regulatory restrictions and proving dependable neutron safety for dry storage operations.

The Rooppur Nuclear Power Plant (RNPP) or other regional centralized facilities may be able to use the VSC-24 cask for temporary dry storage. Its assessment facilitates cross-compatibility studies between Russian and Western storage technologies and aids in the creation of a regulatory framework for the management of wasted fuel in Bangladesh. Additionally, this initiative strengthens local skills in handling and storing spent fuel by facilitating knowledge transfer and capacity building in nuclear engineering safety analysis.

Thermal and structural behavior during long-term storage, including decay heat and radiation-induced material deterioration, should be included in the future study.

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