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Modeling and Analysis of Neutron Flux of a VVER 1200 Reactor Core Using Monte Carlo Code OpenMC

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Abstract
Nuclear reactors and their associated facilities are complex systems that require accurate modeling and analysis to ensure safety, security, and efficient operation. This study aims to model and analyze the neutron flux of a generic VVER 1200 reactor core using the Monte Carlo code OpenMC. For the purpose of this analysis, a generic model of the VVER 1200 reactor core was developed using an OpenMC simulation environment. Furthermore, using this model, the neutron flux spectrum inside the reactor core was evaluated, from which neutron fluence was calculated accordingly. This study analyzed the effect of neutron flux on the developed model of the VVER 1200 reactor core. The research results reveal a close relationship between neutron flux and reactor power, whereby an escalation in neutron flux leads to a proportional escalation in reactor power output, indicating their interdependence. Additionally, the results highlight a strong correlation between neutron fluence and neutron flux, where an elevation in neutron flux causes a corresponding rise in neutron fluence, which is a significant contributor to the increase in the ductile-to-brittle transition temperature of the RPV wall. In this study, the VVER 1200 reactor core model was simulated and after 60 years of operation; the fluence, which is the total neutron flux received by the reactor vessel, was extracted from the evaluated neutron flux spectrum, and its value was found to be at the design limit of the $10^{19}$ scale. So, it is important to maintain the neutron flux at a level that provides sufficient power output while ensuring the RPV's safety and longevity. The results also show that OpenMC is an effective tool for simulating the neutron flux distribution in the VVER 1200 reactor core. The study
findings provide valuable insights into the behavior of the reactor core, which can help improve the design and operation of nuclear reactors for safer and more efficient nuclear energy production.

Keywords: Monte Carlo code OpenMC, VVER 1200, reactor pressure vessel, neutron flux

1. Introduction
The VVER 1200 (or “Water Water Energy Reactor 1200”), originated in Russia, is a generation (III+) pressurized light water reactor. It has a thermal power output of 3,200 MW and uses enriched UO$_2$ fuel. The VVER 1200 features several advanced active and passive safety systems, which are developed to ensure the safe shutdown of the reactor in the event of an accident. It also has several design features that are intended to make it more resistant to earthquakes and other external hazards [1]. However, the neutron flux is a critical aspect of the nuclear reactor core, and understanding its behavior and effect is vital for the risk-free and efficient functioning of nuclear reactors. Also, the neutron flux provides a valuable perspective into the behavior of neutrons and their interactions with materials, which can be applied in a broad spectrum of nuclear applications, including power generation, radiation therapy, and materials testing. Additionally, a strong correlation exists between neutron fluence and neutron flux: an elevation in neutron flux causes a corresponding rise in neutron fluence, which is a significant contributor to the increase in ductile to brittle transition temperature (DBTT) [2–3]. The neutron fluence is explicitly stated as the total number of neutrons per unit area that passes through a material. Alternatively, neutron flux is the rate at which neutrons pass through a unit area over a unit of time. In nuclear materials, neutron fluence causes atomic displacement, leading to the accumulation of radiation damage and causing a significant increase in the DBTT, which is a critical parameter that determines the brittle fracture behavior of materials [4–5].

Study shows that nuclear facilities are the target for extremists, and so the nuclear safety and security of nuclear materials are vital. These facilities are becoming more eye-catching for present-day extremists or other enthused groups such as national or international terrorists [6–7]. Thus, these facilities have security and safety concerns related to the nuclear materials. One such safety concern involves the effect of neutron flux in a reactor core is significant, which includes influencing the rate of nuclear reactions and irradiation embrittlement of the reactor pressure vessel (RPV) wall, which affects DBTT, affecting fuel depletion, generating heat, and contributing to the overall behavior and safety of the reactor [8–9]. Several types of code can calculate neutron flux profiles such as Monte Carlo codes, which use random sampling techniques to simulate neutron behavior in a system; diffusion codes, which use the diffusion equation to calculate neutron flux profiles; transport codes, which use the Boltzmann transport equation to calculate neutron behavior; and hybrid codes, which combine aspects of both diffusion and transport codes to balance accuracy and computational efficiency. The choice of code will depend on the level of accuracy required and the available computational resources [10]. Monte Carlo codes are often preferred over other types of codes for neutron transport calculations because Monte Carlo codes can model
complex geometries and materials with great flexibility, and these codes are capable of providing highly accurate results because they simulate the behavior of individual neutrons and consider all possible interactions with materials in the system. This feature can be especially important for nuclear applications in which even small errors can have significant consequences.

There are various Monte Carlo neutronic codes, but a few notable codes are MCNP (Monte Carlo N Particle Transport) [11], KENO [12], TRIPOLI [13], Serpent [14], OpenMC [15], and others. In particular, the OpenMC [16] is an open-source particle transport simulation code that is used to model the behavior of neutrons and other particles as they move through materials. In OpenMC, neutrons are tracked through the geometry of the system being studied, and their interactions with materials are modeled using nuclear data. One advantage of OpenMC is its flexibility and modularity, which allows users to customize the simulation code to fit their specific needs. Additionally, the open-source nature of the code means that it can be freely accessed and modified by users and developers around the world, allowing for a collaborative approach to simulation development and improvement. The primary objective of this task is to develop a model of the generic VVER 1200 reactor full core, including the RPV, using an OpenMC simulation code with the use of the ENDF/B-VII.1 nuclear data library that has been effectively integrated and to evaluate and analyze the effects of neutron flux and fluence into the developed model of the generic VVER 1200 reactor core. Thus, the ENDF/B-VII.1 data library is a comprehensively evaluated collection of nuclear data, providing all essential cross section data required for conducting neutronic analyses. This library comprises nuclear data pertaining to 423 nuclides as of the 2012 edition of ENDF/B-VII.1 [16–19].

Table 1. Fundamental specifications for the fuel assembly (FAs) of the generic VVER 1200 reactor core [17–21].

<table>
<thead>
<tr>
<th>Fuel Assembly (FAs)</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>FA Shape</td>
<td>Hexagonal</td>
</tr>
<tr>
<td>Number of fuel assemblies in the core</td>
<td>163</td>
</tr>
<tr>
<td>The gap between fuel assemblies</td>
<td>34 cm</td>
</tr>
<tr>
<td>Number of fuel elements in an FA</td>
<td>312</td>
</tr>
<tr>
<td>The edge length of the hexagonal prism</td>
<td>13.65 cm</td>
</tr>
</tbody>
</table>

Table 2. Fundamental specifications for the fuel elements (FEs) of the generic VVER 1200 reactor core [17–21].

<table>
<thead>
<tr>
<th>Fuel Element (FEs)</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Type</td>
<td>UO₂</td>
</tr>
<tr>
<td></td>
<td>UO₂ + Gd₂O₃</td>
</tr>
<tr>
<td>Enrichment</td>
<td>Up to 4.5 wt % of $^{235}$U</td>
</tr>
<tr>
<td>Mass fraction of Gd₂O₃ in fuel</td>
<td>5.0 to 8.0 wt % of Gd₂O₃</td>
</tr>
<tr>
<td>Shape of FE</td>
<td>Cylindrical</td>
</tr>
<tr>
<td>Fuel height</td>
<td>373 cm</td>
</tr>
<tr>
<td>The pitch between fuel elements</td>
<td>1.275 cm</td>
</tr>
<tr>
<td>Fuel pellet diameter</td>
<td>0.76 cm</td>
</tr>
<tr>
<td>Central hole diameter</td>
<td>0.12 cm</td>
</tr>
<tr>
<td>Clad material</td>
<td>E-110 alloy</td>
</tr>
<tr>
<td>Clad density</td>
<td>6.515 g/cm³</td>
</tr>
</tbody>
</table>
2. Theoretical Model and Reactor Core

a. A Brief Overview of the Core of the VVER 1200 Reactor

The VVER is a type of pressurized water reactor that uses a combination of light water and H$_3$BO$_3$ for cooling and moderating functions. The VVER 1200 can produce 3200 MWth of thermal energy, which in turn generates 1200 MWe of electrical power. The fuel assemblies (FAs) of VVER 1200 are designed to generate heat without exceeding the maximum limits of fuel rod damage. Each FA contains 312 fuel rods, and the core is...
encased by several layers, including the core baffle, core barrel, water reflector, cladding material, and core vessel. The framework of the FA comprises 18 guide channels and one measuring channel. Reactor power regulation and control are achieved through two systems that adjust reactivity: (1) the control and protection system and (2) the boron regulation system. The control and protection system includes up to 121 rod cluster control assemblies divided into 12 banks, which are used to suppress the chain reaction quickly and maintain power at a desired level. The boron regulation system is used for small changes in reactivity by adjusting H$_3$BO$_3$ concentration in the primary circuit water. The fuel assemblies come in six different types with varying enrichments, numbers of fuel pins with different enrichments, and different numbers of pins with a weight percentage of the burnable neutron absorber Gd$_2$O$_3$ ranging from 5% to 8% [17–18].

b. Core Model and Analysis Procedures

The models were created using Python (Python 3.7) code in Jupyter Notebook and represented in OpenMC. Initially, a single type of fuel rod was developed and placed in a hexagonal pattern with a spacing of 1.275 cm, forming an assembly pitch of 23.6 cm. Boolean operations were used to define different zones within the cells. The generic VVER 1200 reactor core geometry was constructed with a hexagonal lattice and two planes perpendicular to the z-axis, both using reflecting boundary conditions, effectively producing an infinite geometry in the z-axis.

The OpenMC process involves making certain assumptions:

- Pin-by-pin model
- The boundary condition is set to be reflective in every direction (x, y, and z)
- A finite boundary is applied in the z-axis, and it uses reflective boundary conditions
- Cross section data library: ENDF/B-VII.1
- 4000 batches with 100 inactive batches and 100,000 particles in each batch

The OpenMC code provides three different methods to compute the $k$-eigenvalue: the track-length estimator (Equation 1), the collision estimator (Equation 2), and the absorption estimator (Equation 3). These methods are mathematically expressed through the following:

\[
k_{\text{track-length}} = \frac{\Sigma_{\text{all flights}} w_j d_j v \Sigma_f}{W}, \quad (1)
\]

\[
k_{\text{Collision}} = \frac{\Sigma_{\text{all collisions}} w_j \left(\frac{\nu \Sigma_f}{\Sigma_t}\right)}{W}, \quad \text{and} \quad (2)
\]

\[
k_{\text{track-length}} = \frac{\Sigma_{\text{all absorption}} w_j \left(\frac{\nu \Sigma_f}{\Sigma_a}\right)}{W}. \quad (3)
\]
The OpenMC Monte Carlo code uses the combined collision estimator, track length estimator, and absorption estimator to calculate the average $k_{inf}$ value. This value is computed based on several parameters, including the total weight starting each generation (or batch) denoted by $w$, the pre-collision weight of the particle as it enters event $j$ denoted by $w_j$, the length of the $j$th trajectory denoted by $d_j$, and macroscopic neutron production cross section ($\nu\Sigma_f$) and absorption cross section ($\Sigma_a$). By using these values, the OpenMC code can accurately estimate the average $k_{inf}$ value for the VVER 1200 reactor core [19].

3. Results and Discussion
Initially, six different types of fuel assembly geometries (Figure 2) were created using the OpenMC simulation environment for the generic VVER 1200 reactor. These assemblies, along with the core baffle, core barrel, water channels, water reflector, cladding material, and RPV, were used to develop the full core model of the reactor (Figure 3). Subsequently, the thermal neutron flux (Figure 4) and fast neutron flux (Figure 5) spectra were evaluated. The analysis of these spectra revealed that the neutron flux in the reactor core is high, but the neutron flux in the RPV is comparatively lower. Specifically, the neutron flux in the core was on the order of the $10^{14}$ scale, whereas the neutron flux on the RPV wall was on the order of the $10^{10}$ scale. Additionally, Figure 8 demonstrates a linear relationship between the neutron flux in the core and the neutron flux on the RPV wall. The investigation of radial neutron flux distributions indicated that the fast neutron flux (Figure 5) in the peripheral region is higher than at the center of the core. High neutron flux levels are desirable in a nuclear reactor because they promote fission, which is vital for power generation. However, excessive neutron flux can lead to material degradation and increase the risk of nuclear accidents. In the developed VVER 1200 reactor core model under normal operating conditions, Figure 7 illustrates that reactor power increases linearly with neutron flux. After 60 years of operation, the neutron fluence was extracted from the evaluated neutron flux spectrum. As depicted in Figure 9, the neutron fluence on the RPV wall exhibited a linear increase with neutron flux and reached a value on the order of the $10^{19}$ scale, which corresponds to the design limit [22]. Neutron fluence significantly affects RPV embrittlement in a nuclear reactor. Higher neutron fluence at the RPV wall raises the DBTT, resulting in increased brittleness of the material [1–5]. Therefore, it is important to determine the optimal value of neutron flux, and a balance must be struck between the desired power output and the acceptable level of RPV embrittlement. This balance requires careful monitoring and control of the neutron flux levels, as well as a continuous assessment of the RPV’s condition.
Figure 1. Flow diagram of the simulation (theory and methodology—OpenMC Stable) [15].
FAs, type Z13, fuel pin enrichment 1.3%

FAs, type Z33Z9, fuel pin enrichment 3.3%, contains 9 fuel pins with fuel enrichment of 2.4% and Gd₂O₃ concentration equal to 8%

FAs, type Z40, fuel pin enrichment 4.0%

FAs, type Z44B2, fuel pin enrichment 4.4%, contains 12 fuel pins with fuel enrichment of 3.6% and Gd₂O₃ concentration equal to 5%

FAs, type Z33Z2, fuel pin enrichment 3.3%, contains 12 fuel pins with fuel enrichment of 2.4% and Gd₂O₃ concentration equal to 8%

FAs, type Z24, Fuel pin enrichment 2.4%

Figure 2. Radial view of the different types of fuel assemblies in the VVER 1200 reactor.
Figure 3. Radial view of a generic VVER 1200 reactor core.
Figure 4. Radial view of thermal neutron flux spectrum of a generic VVER 1200 reactor core.

Figure 5. Radial view of fast neutron flux spectrum of a generic VVER 1200 reactor core.
In Figure 6, the unit for flux (n-cm/eV-src) refers to the number of neutrons that pass through a unit area (cm\(^2\)) in a given amount of time per unit energy (eV) per unit solid angle (src). This unit is commonly used to represent the neutron flux in neutron transport simulations.

Figure 6. Neutron flux distribution as a function of energy range of a generic VVER 1200 reactor core.

Figure 7. Comparison of reactor power and core neutron flux of a generic VVER 1200 reactor.

DOI:
In Figure 7, for obtaining the reactor power output from the neutron flux, the following assumptions [23] and parameters are typically taken into consideration:

- Macroscopic fission cross section 0.1 cm\(^{-1}\)
- Energy released per fission is 200 MeV

![Figure 8. Comparison of RPV and core neutron flux of a generic VVER 1200 reactor model.](image)

![Figure 9. Neutron flux vs. neutron fluence on the RPV wall of a generic VVER 1200 reactor core.](image)
4. Conclusion
This study focused on modeling and analyzing the neutron flux of a VVER 1200 reactor core using the OpenMC code. The results extracted from the simulations show that OpenMC can effectively simulate the neutron flux distribution in the reactor core. The study also analyzed the effect of the neutron flux spectrum on the core and neutron fluence on the RPV wall. The analysis performed shows that the neutron flux significantly affected the RPV wall. Therefore, although the neutron flux is an important parameter for reactor performance, it must be managed carefully to ensure that it does not exceed the RPV fluence limit and cause embrittlement of the RPV wall. However, specific conclusions from the study are the following:

- The optimal value of neutron flux in a nuclear reactor strikes a balance between the desired power output and the acceptable level of RPV embrittlement. This balance requires careful monitoring and control of neutron flux levels, as well as a continuous assessment of the RPV’s condition to ensure the safe and reliable operation of the reactor.
- The study highlights the Monte Carlo code OpenMC, which proved to be an effective tool for simulating the neutron flux distribution in the VVER 1200 reactor core.
- Overall, the results obtained from this research study provide valuable insights into the behavior of the VVER 1200 reactor core under normal operating stages.

5. References


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