# Neutronic Analysis of a core design based

# on AP300 model using OpenMC

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**Abstract**

This study presents an analysis of a light water Small Modular Reactor (SMR) design featuring a square-shaped fuel element. The core design is based on the Westinghouse UO₂ SMR, which operates with uranium oxide fuel enriched to less than 5%. To accurately simulate and analyze the reactor's performance, the study employed the open-source Monte Carlo code, OpenMC. This code was chosen for its ability to model complex reactor physics with high precision.

The primary focus of the study was on performing neutronics analyses of the core containing UO₂ fuel. Key reactor parameters were evaluated, including the effective multiplication factor (keff), which provides insight into the reactor's ability to sustain a nuclear chain reaction under various conditions. The radial neutron flux profile was also analyzed to understand the distribution of neutrons across the core and ensure that the reactor operates efficiently and safely.

Additionally, the study investigated the maximum-to-average flux ratio, a crucial parameter for assessing the thermal performance of the reactor. A high ratio could indicate potential hotspots, which need to be minimized for safe operation. The reactivity coefficients, including temperature and coolant density coefficients, were also characterized to evaluate the reactor's response to changes in operational conditions, such as variations in temperature and coolant flow.

The preliminary simulation results demonstrated strong performance for the SMR design, with values within expected ranges for a well-designed reactor. Globally, these results were further validated by comparing them with calculations obtained from the widely-used MCNP code, developed by Missouri S&T. The comparison showed good agreement between the two sets of calculations, reinforcing the reliability of the OpenMC simulations.

Overall, this study contributes valuable insights into the design and analysis of SMRs, showcasing the potential of open-source tools like OpenMC for reactor physics simulations. The findings support the continued development of SMR technology, which holds promise for enhancing the safety, efficiency, and scalability of nuclear power generation.

## INTRODUCTION

In this paper, we focus on a basic neutronic analysis of the Westinghouse AP300 Small Modular Reactor (SMR). SMRs are gaining attention as smaller, more adaptable nuclear fission reactors, offering economical, reliable, and clean power solutions for various applications, from industrial operations to remote sites. As research and development continue, SMRs have the potential to significantly reduce greenhouse gas emissions, meet growing energy demands, and support a more sustainable energy future. These reactors can generate low-carbon baseload power, meaning they don't rely entirely on fossil fuels to provide a steady and dependable energy source. Among the emerging SMR concepts, the AP300, which uses light water, is particularly noteworthy.

The primary goal of SMR development is to create innovative, safe, and economically viable small nuclear power plants. The AP300 is an integrated pressurized water reactor (PWR) with a square lattice-shaped core and an indirect steam cycle. Its design prioritizes safety while simplifying the construction process.

This project focused on analyzing light water SMRs with flexible fuel configurations. The core design, based on the Westinghouse UO₂ SMR with less than 5% enrichment, was developed using OpenMC, an open-source Monte Carlo-based code written in C++ and Python. Neutronics studies of a reference core with UO₂ fuel were conducted to determine key parameters, such as the radial neutron flux profile, maximum-to-average flux ratio, reactivity coefficient, and the critical boron content at the beginning of the reactor's life.

## methods and materials

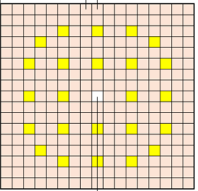
### Description of AP300

The American Small Modular Reactor (SMR) known as the AP300 combines innovative design elements with advanced engineering solutions to create a highly safe and competitive reactor. Developed by Westinghouse and inspired by the AP1000, the AP300 is an integral-type pressurized water reactor (PWR) that operates on an indirect steam cycle. Its unique design features include an integrated primary cooling system with pressurized natural circulation, in-vessel hydraulic control rod drive mechanisms, and passive safety systems.

#### Geometry

The core analyzed in this study is based on Westinghouse's SMR, an integral pressurized water reactor (PWR) with an active core height of 2.4 meters. The core consists of an 11 x 11 robust fuel assembly (RFA) layout, comprising 89 assemblies - 52 for fuel and 37 for control rod drive mechanisms. These components are housed within a core barrel and the reactor vessel. While the design elements were derived from the AP1000, the dimensions were adjusted for the SMR, resulting in a reactor vessel that is 3.5 meters in diameter and 24.7 meters in height [1].

The study focused on one type of fuel: uranium-oxide (UOX). The fuel assembly, as shown in Figure 1, follows a standard design with 264 fuel rod locations, 24 guide tube locations, and a central location for instrumentation, all in accordance with Westinghouse's design specifications.



*FIG. 1. Layout of of 17 x 17 fuel assembly.*

The fuel rod consists of a cylindrical pellet encased in a Zircaloy-clad tube. A small gap exists between the fuel stack and the cladding to accommodate fuel swelling caused by fission products, preventing potential damage to the cladding. This gap is filled with helium gas to improve heat conduction from the fuel to the cladding. The guide tubes within the fuel assembly provide space for the insertion of the Rod Cluster Control Assembly (RCCA), which is a spider-like assembly that holds evenly spaced control rods. These control rods are made of either Silver-Indium-Cadmium or Boron Carbide, depending on the type of fuel used.

The detailed specifications for the fuel rod, cladding, structural components, control rods, and burnable poisons (both discrete and integral) were obtained from the Consortium for Advanced Simulation of Light Water Reactors (CASL) VERA core physics benchmark specifications [2] and are presented in Tables 1, 2, and 3.

TABLE 1. FUEL ROD GEOMETRY SPECIFICATIONS

|  |  |  |
| --- | --- | --- |
| Details | Material | Dimensions |
| Pellet Radius | UO2 | 0.4095 cm |
| Pitch |  | 1.26 cm |
| Clad Inner radius | Zirc-Alloy | 0.4177 cm |
| Clad Outer radius | 0.475 cm |
| Fuel stack Height |  | 244 cm |
| Rod Height |  | 305 cm |

TABLE 2. RCCA GEOMETRY SPECIFICATIONS

|  |  |  |  |
| --- | --- | --- | --- |
| Details | Material | Dimensions | Remarks |
| Poison Radius |  | 0.525 cm | For Ag-In Cd Rods: |
| Poison heigh |  | 290-300 cm | 80% Ag, 15% In and |
| Step size | Ag-In-Cd/B4C | 1.5875 cm | 5% Cd and |
| Numbers of steps | 183 Nos | (ρ = 10.2 g/cc) |

TABLE 3. PRESSURE VESSEL GEOMETRY SPECIFICATIONS

|  |  |  |
| --- | --- | --- |
| Details | Material | Dimensions |
| Vessel Inner radius | SS316 | 142.7 cm |
| Vessel Outer radius |  | 147.7 cm |

#### Uranium-oxide Fuel assembly

The uranium oxide (UOX) fuel used in this study was low-enriched uranium (LEU) with an enrichment level of less than 5% and a density of 10.36 g/cm³, equivalent to 95% of the theoretical density. The arrangement of fuel assemblies and burnable poisons within the core followed a radial loading pattern. The core was divided into three regions, each with different fuel enrichment levels: the central and intermediate regions contained fuel enriched to 2.35% and 3.4%, respectively, while the outer peripheral region used fuel enriched to 4.45%. This varying enrichment pattern helped achieve a more uniform radial power distribution, contributing to a favorable overall power profile within the core.

For the analysis, only the U-235 and U-238 isotopes were considered. The composition of these isotopes for each enrichment level was determined and is presented in Table 4.

TABLE 4. COMPOSITION OF URANIUM-OXIDE

|  |  |  |  |  |
| --- | --- | --- | --- | --- |
| Nuclide | w/o (2.5%) | w/o (3.4%) | w/o (4.5%) | Remarks |
| U235 | 2.071 | 2.997 | 3.922 |  |
| U238 | 86.076 | 85.150 | 84.225 | ρ = 10.36 g/cc |
| O16 | 11.853 | 11.853 | 11.853 |  |

### OpenMC modeling

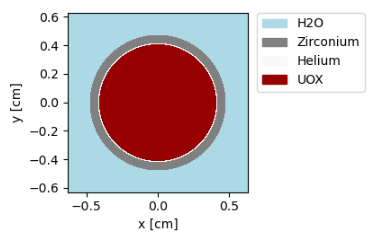
The development process of OpenMC emphasizes transparency, as it is an open-source transport code with fully visible source code. This transparency is particularly beneficial for new users, allowing them to better understand and engage with the software. The development branch of OpenMC is crucial for accessing the latest features, as it incorporates ongoing updates before they are officially released. While static releases of the code are available, some developments needed for this research were implemented using version 0.14.0.

In collaborative projects, maintaining consistency in code versions can be challenging. From a validation perspective, it is essential to ensure that each revision does not affect the underlying physics or create compatibility issues with previously completed simulations. Serpent, another transport code, similarly lacks an official release and is distributed in a manner similar to MCNP, with static versions. Given that OpenMC is still under active development, these factors should be carefully considered [3,4].

#### Pencil model

The fundamental component of nuclear fuel is the pencil fuel, which consists of small cylindrical pellets enclosed in a longer cylindrical cladding. The active length of the fuel element refers to the effective length of fissile material where fission can occur. In the present case study, the fuel rod model is a long cylindrical pin with a length of 2.4 meters and a diameter of 8.19 mm. This fuel rod contains uranium oxide (UO2) pellets and is surrounded by zirconium alloy cladding with a gap, either air or helium gas, between the pellets and the cladding. The Python scripts in the following code cells demonstrate how to create the pencil, accounting for various fuel rod configurations, such as pencil fuel, gap, cladding, and surrounding water.

At last, we can achieve the illustration depicted in Fig. 2 below. The red inner disk is a visual representation of the fuel pellet's cross-section, while the light grey ring represents the cylindrical cladding. The white area between the fuel and the cladding signifies the gaseous space, which permits the thermal expansion of the fuel when it heats up during reactor operation.



*FIG. 2. Pencil model with cladding using OpenMC.*

The thimble guide and instrumentation tube are both defined similarly, except the thimble guide has a larger inner radius to facilitate the insertion of the control rod, which has nearly the same diameter as fuel rod one. The inner volume of the thimble guide is filled with water instead of fuel, as the control rods are completely withdrawn from the fuel assembly in this scenario.

#### Square lattice fuel assembly model

The fuel assembly (FA) features a group of fuel rods enclosed in cladding and arranged in a square lattice pattern. A moderating substance envelops the fuel rods within each assembly. In the case of the AP300 reactor, light water is utilized to moderate neutrons and as a cooling agent within the primary vessel of the reactor. The Fuel Assembly will also contain control rods and instrumentation thimbles, left empty in this instance (filled with water). Fig. 3 displays a square fuel assembly scheme with 264 square fuel elements, including 24 control rods (green) and one thimble guide for instrumentation (yellow).

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Description générée automatiquement

*FIG. 3. Fuel assembly model using OpenMC*

The core used for the analysis was based on a Westinghouse’s SMR, an integral PWR with an active core height of 2.4 meter (~8 feet). Fig 4 presents the cross sectional view of the assembly layout for the SMR core also indicating the location of the control rod drive mechanisms.

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*FIG. 4. SMR Core model using OpenMC*

### Numerical calculation using OpenMC

After accurately defining the physical and geometric models, it will be important to configure the simulation's characteristics. OpenMC defaults to using neutrons as the transported particle and conducting the eigen-source calculation. In this study, we will employ OpenMC for criticality calculations and also score tallies for educational and code utilization purposes **[5-7]**. The default calculation in OpenMC is the criticality calculation, which requires defining the source and particle population for generation using the Monte Carlo method.

The OpenMC simulation utilized a neutron population of up to 50000 neutrons per cycle (batch). This necessitated approximately 80 minutes of Time Machine Calculation (TMC) for each simulation conducted on the Google Colab platform. The cross section for fuel, cladding, moderator and vessel were from ENDF/B-VII libraries at 600K temperature.

## PRELIMiNARY RESULTS

### Neutronic parameters

In Table 5, the prompt multiplication factor (kp) was found to be 1.1881 with a margin of error of ±0.00025. The effective delayed neutron fraction (βeff) was determined to be 0.0068. Additionally, the delayed neutron fraction can be utilized to compute the average neutron generation time, indicating the rate at which power can increase during regular reactor operation. The shutdown margin, also known as the control rod worth, represents the negative reactivity needed for the reactor to become subcritical from its current state. The control rod worth for a typical shutdown condition (i.e., at 600K) was established as 0.163.

Additionally, from Table 5, the moderator reactivity coefficient is -1.45E-03 δk/°C. The negative value indicates that the reactor is under-moderated. This coefficient depends on the fuel-to-moderator ratio. It's important for the reactor to have a negative reactivity coefficient as this helps it self-regulate. If there's a positive reactivity, it will raise core temperatures, but the large negative feedback reactivity introduced will help control the power and safely shut down the reactor.

Moreover, the reactivity coefficient for fuel was determined to be -2.24E-05 δk/°C. The negative value indicates that any positive reactivity will cause the fuel temperature to increase rapidly, ultimately providing a negative feedback reactivity that ensures the safety of the reactor. A negative fuel reactivity coefficient is preferred over a moderator reactivity coefficient because in the event of a positive reactivity insertion, the fuel's negative feedback response is faster than that of the moderator. Thus, the fuel reactivity coefficient is also known as the prompt reactivity coefficient. Additionally, a greater fuel reactivity coefficient value can result in the Doppler Effect or Doppler Broadening phenomenon, particularly at higher fuel temperatures.

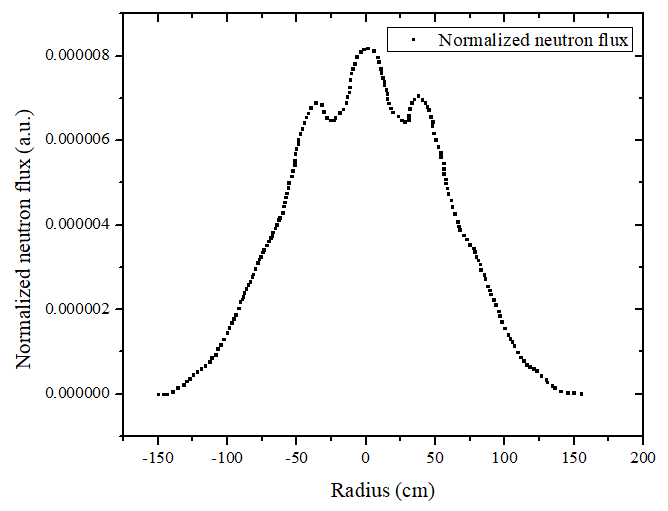
TABLE 5. NEUTRONIC PARAMETERS OF AP300

|  |  |  |
| --- | --- | --- |
| Details | Our study | DSOUZA **[8]** |
| Effective multiplication factor | 1.1881±0.00025 | 1.11945 ±0.00014 |
| Maximum to average flux ratio | 2.672 | 2.807 |
| Delayed neutron fraction | 0.0068 | 0.00671 |
| Control rod worth | 0.1630 | 0.1715 |
| Fuel coefficient of reactivity (δk/°C) | -2.24E-05 | -2.58E-05 |
| Moderator coefficient of reactivity (δk/°C) | -1.45E-03 | -1.57E-03 |

### Normalized radial neutron flux

It is evident from the illustration that the flux profile is evenly distributed throughout the core in both calculations. This uniform distribution results from the consistent loading of the fuel and the burnable absorber, which guarantees predictable core temperatures and consistent fuel depletion during reactor operation. The high source of heat generation is indicated by the maximum neutron flux at the central fuel assembly. Additionally, the surrounding area, especially the 2.35 enriched UO2 fuel assemblies, also displayed higher flux, as shown in the figure.

The flux in the cores was shaped or flattened using a combination of reflector conditions (light water) and zoning (variation in fuel concentration or poison loading). Although zoning had a high impact on neutron economy due to its high neutron absorption cross section, it proved to be the more effective method. Both MCNP simulation and our study showed a maximum to average flux ratio above 1, at 2.807 and 2.537, respectively, indicating that maximum heat generation rate is more significant than the average neutron flux. The UO2-fueled core had a greater maximum to average flux ratio than a standard reactor (2.3 value), based on the energy safely carried by the coolant.



*FIG. 4. Normalized radial neutron flux*

## conclusion

In this research, the study is focused on the conceptual design and the neutronic characteristics of SMR model based on AP300 fuel design. The 3-D model was carried out using OpenMC code owing to assess keff, neutron energy spectra and spatial neutron flux distributions. The main conclusions conducted by these calculations are.

(1) ENDF-B/VII, the newer cross section data library, overestimates the keff in comparison to use DSOUZA results about 1.1881±0.00025. (2) Maximum to average flux ratio 2.672, Delayed neutron fraction 0.0068, Control rod worth 0.1630, Fuel coefficient of reactivity -2.24E-05 δk/°C), Moderator coefficient of reactivity -1.45E-03 δk/°C. (3) As long of the radial distance and at different levels, the neutron flux keeps symmetrical shapes. This study presents a preliminary result that should be improved and compared with other benchmarks.

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He obtained in 2006 him PhD in Nuclear Engineering at EVRY VAL D’ESSONNE University (as part of University of Paris-Saclay) and occurred at CEA/Saclay in France. after that a post-doctoral in criticality safety assessment at IRSN-Fontenay aux Roses.

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Currently He participates in CRP: Economic Appraisal of Small Modular Reactor (SMR) Projects: Methodologies and Applications, Project title: Economic and financial case study on HIGH TEMPERATURE GAS COOLED SMR in Tunisia.

Also, He is involved in the research area of performance analysis and reliability by applying Markov approach to dynamic reliability modelling of Safety Critical Systems (SCS) and digital instrumentation and control systems for NPP/SMR assessments.