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Thermal-hydraulic analysis of new generation TVS-2M fuel in VVER-1000 reactor using porous media approach method

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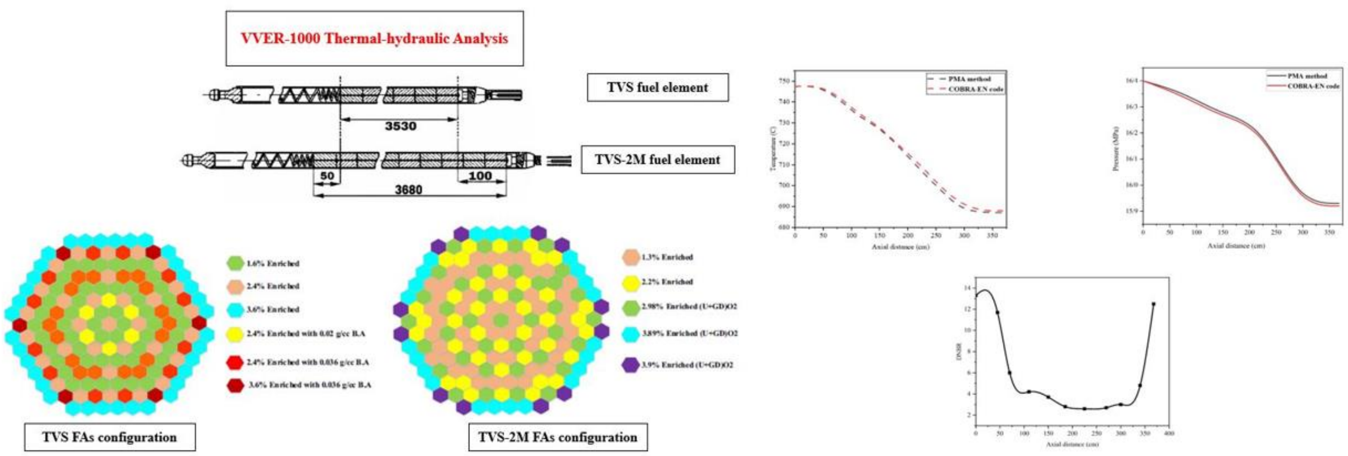
Abstract

The TVS-2M Russian Fuel Assembly (FA) is a newly suggested fuel type that incorporates gadolinium oxide in different concentrations, combined with enriched UO2 at different levels of U-235 enrichment. TVS-2M FA is designed to improve the safety and efficiency of reactor core. In the Porous Media Approach (PMA) method, the porous medium is generally described by parameters such as permeability, porosity, and the tortuosity of the flow paths, which are utilized to represent the resistance to fluid flow and the material's heat transfer properties. The main aim of present paper is to model the hottest FA of Bushehr Nuclear Power Plant (BNPP) reactor core using the COBRA-EN code and PMA method to evaluate the thermal-hydraulic parameters such as fuel rod temperature, coolant temperature and density, Departure from Nucleate Boiling Ratio (DNBR), and Critical Heat Flux (CHF) in different axial and radial nodes. In such manner, each FA is modeled and broken down into a network of simplified regions. Although detailed geometries are specified, the thermal-hydraulic parameters are computed for each component. Results showed, the average core coolant temperature ranges from 29°C 1 to 322°C. Meanwhile, the coolant temperature in the hottest FA reaches about 328 C. Moreover, the fuel rod temperature varies from about 310°C to 930°C in the hottest FA. On the other hand, the average coolant density into the core changes from 742 kg/m³ to 675 Kg/m^3.

Graphical Abstract



1. Introduction

In recent years, Russia has advanced a new generation of nuclear fuels for cutting-edge reactors, backed by extensive research and development. Since 1964, over 69 VVER reactors have been constructed globally, with 60 currently in operation. To enhance the efficiency of these VVER reactors, the Russian fuel manufacturer TVEL, in collaboration with research institutions like Kurchatov, has developed new generations of TVS fuels. The latest TVS fuels, such as TVS-2M, offer significant improvements over the current standard fuel, including higher physical and chemical resistance, extended useful life, longer active fuel lengths, and greater fissile material content [1-2]. These fuels also allow for extending the reactor operating cycle from 300 days to 12 to 18 months or increasing the total reactor power from 1000 MW to 1200 MW. Additional benefits include prolonged overall operational time, enhanced long-term reactor performance, improved fuel efficiency, the substitution of the burnable absorber CrB_2+Al with Gd_2O_3, and reduced operational costs. The VVER-1000 is a Russian-designed pressurized light water reactor in BNPP, boasting a thermal power output of 3000 MW and an electric power output of 1000 MW. This reactor utilizes light water as both a moderator and a coolant. In the primary circuit, the light water circulates through the FAs, absorbing heat and reaching high temperatures without boiling due to the high pressure maintained within the system [3-4].

Thermal-hydraulic analysis refers to the study of the behavior and performance of the coolant and heat transfer mechanisms for both thermal and fluid dynamics aspects that plays an important role to ensure the safety and efficient operation of the system. There are two main techniques; core analysis and subchannel analysis for the core thermal-hydraulic parameters evaluation. Core analysis involves examining the thermal-hydraulic behavior of the entire reactor core, which consists of FAs, coolant, and other reactor components. It provides an integrated view of the thermal performance, focusing on core-wide conditions, such as average temperature, power distribution, and coolant flow. While, the subchannel analysis focuses on smaller, localized regions within the reactor core and is particularly important in assessing the behavior of coolant in the individual fuel rod bundles. These subchannels, also known as flow channels, are the spaces between adjacent fuel rods or FAs where the coolant flows. Various techniques are employed in reactor core thermohydraulic analysis, ranging from advanced Computational Fluid Dynamics (CFD) simulations, which offer in-depth insights into fluid flow and heat transfer, to more streamlined system codes such as RELAP5 and TRACE, which model the reactor's overall performance during transient events and safety situations. Additionally, subchannel codes such as COBRA and VIPRE are commonly called to analyze the specific thermal-hydraulic characteristics within individual FAs [5-7].

In PMA method, the porous or perforated structure is characterized by its volume-averaged properties within a geometrically defined porous region. In this continuous and uniform porous region, resistance source terms are used as closure terms in the momentum equation. These source terms reflect the macroscopic impact of the porous structure on the fluid flow. The PMA method is a recognized approach for analyzing reactor core fuel assemblies. It introduces the technique of establishing conservation equations using the concept of porosity within the control volume. This method ensures high accuracy in solving momentum equations for single-phase fluids. However, the PMA method does not assess fuel temperature gradients. [8-9]. This approach is particularly useful for analyzing the thermal-hydraulic behavior of the reactor core and can be coupled with neutronic codes for comprehensive simulations.

Although several reports and papers on various technical aspects of TVS-2M fuels have been studied in recent years, their thermal-hydraulic facets did not discuss in details. Zare Ganjaroodi et al, in 2024, analyzed the new generation TVS-2M FAs in VVER-1000 reactor using three nuclear simulation codes; WIMS & CITATION, MCNPX, and SuperMC [10]. Afzali et al, in 2022, employed a multi-layer neural network to identify the optimal TVS-2M fuel arrangement, capable of calculating the burnup for various fuel configurations. By using core parameter data, they trained the neural network to establish a real-time prediction system for burnup and actinide concentration for different fuel assembly arrangements [11]. Zahedi Yeganeh and Ansarifar developed a new Small Modular Nuclear Reactor (SMR) core, incorporating modifications to the FA type, fuel enrichment, and gadolinium concentration in the FAs. These changes aim to enhance and improve the NuScale nuclear reactor, which is a typical PWR-SMR [12]. Furthermore, the PMA method has been explored in various studies for core modeling. Notably, in 2024, Zare Ganjaroodi et al. modeled an advanced Small Modular Reactor (SMR) core, specifically the CAREM-25 reactor, which operates using natural circulation without the need for pumps. Their research aimed to investigate the thermal-hydraulic parameters of the reactor core. This study highlights the potential advantages of using the PMA method for improving the accuracy and efficiency of core thermal-hydraulic analysis in SMR designs. [13].

The thermal-hydraulic study of the reactor core is vital for ensuring operational safety, optimizing reactor performance, and meeting regulatory requirements, while also providing insights into the reactor's behavior under both normal and accident conditions. Modeling the reactor core is typically performed at key design and safety points to analyze the behavior of the most sensitive components and nodes during accident scenarios. In this study, the hottest FA of VVER-1000 reactor core in BNPP is modeled using the PMA method and COBRA-EN code to calculate the core's thermal-hydraulic parameters. The hottest FA is determined based on the FA Power Peaking Factors (PPF) identified in our previous studies.

2. Material and Method

Table 1. List of symbols and abbreviations.

#	Symbol	Abbreviation	Unit
1	Temp	Temperature	C
2	P	Pressure	Mpa
3	L	Length	cm
4	η	Power	MW
5	A	Flow/coolant area	m^2
6	T	Time	sec
7	D	Diameter	cm
8	ρ	Density	gr/cm^3
9	m	Mass	Kg
10	h	Height	cm
11	q	Thermal flux	j/m^2
12	G	Coolant mass flow rate	m^3/h
13	Pr	Critical Pressure Ratio (System Reference Pressure / Critical Pressure)	-
14	h	Enthalpy	j/Kg
15	h_{in}	Inlet enthalpy	j/Kg
16	h_{fg}	Evaporation enthalpy	j/Kg

2.1. The VVER-1000 reactor core

The Bushehr VVER-1000 reactor core, an advanced Russian-designed pressurized water reactor (PWR), boasts a sophisticated and robust engineering framework. It utilizes hexagonal FAs that are packed with approximately 126 tons of uranium dioxide fuel, enriched to about 4%. This configuration allows the reactor to achieve a substantial thermal power output of 3000 MW and an impressive electric power output of 1000 MW (Fig. 1). The coolant flow rate through the reactor core is approximately 21,000 tons per hour and the inlet coolant temperature is around 290°C which will increase to about 325°C as it passes through the core [1-2]. The VVER-1000 design parameters are presented in Table 2. Geometrically, the new TVS-2M FAs are nearly identical to their predecessors (TVS). The primary difference lies in the active fuel length, which has increased from 353 cm to 368 cm. Additionally, there is a notable structural variation in the fuel rod grids pacers. The height of the grids has been increased by 10 mm, resulting in a reduction in their number from 15 in the TVS FAs to 13 in the new design. Furthermore, at the inlet of the FAs, Anti-Debris Filters (ADF) have been implemented to prevent potential debris from entering the fuel area and causing damage to the fuel rods. This enhancement contributes to the overall efficiency and reliability of the reactor's operation [14-15]. The TVS-2M and TVS FAs are listed in Table 3.

Table 2. The VVER-1000 reactor design parameters [10].

Parameter	Value
Coolant flowrate through the reactor during operation of four RCP sets, m^3/h, nominal	84800
Coolant pressure at the core outlet	15.7
Coolant temperature at the reactor outlet	321
Coolant heating in the reactor	30
Pressure differential in the reactor	0.381
Number of loops, pes.	4
Fuel height in the core in cold state	3.53
Equivalent diameter of the core	3.16
Flow area of the core	4.14
Pitch between FAs	0.236
Time of CPS AR drop	1.2 ... 4
Number of FA in the core	163
Mass of fuel in FA	489.8
Nominal loading of reactor on UO_2	79840
Assigned service life of vessel, years	40
Diameter of the vessel cylindrical part in the core area	4.535

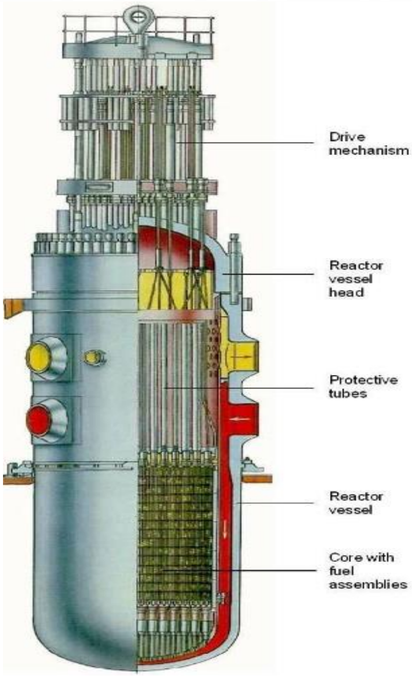


Fig. 1. VVER-1000 reactor core scheme.

Table 3. The TVS and TVS-2M fuel technical parameters [10].

Parameter	Value	
	AFA	**TVS-2M**
Core thermal power	3000	3000
Numbers of FAs	163	163
Geometry	Hexagonal	Hexagonal
Number of fuel rods per assembly	331	331
Measuring tube	1	1
Fuel assembly overall length	45700	45700
Fuel assembly overall width	23.5	23.5
Fuel rod length	383.6	398.8
Active length	353	368
Fuel rod outside diameter	0.91	0.91
Pellet length	0.9-1.2	0.9-1.2
Pellet outside diameter	0.78	0.76
Fuel density	10.4-10.7	10.4-10.7
Cladding material	Zircalloy-4 +1%Nb	Zircalloy-4 +1%Nb
Spacer grid material	Zircalloy-4 +1%Nb	Zircalloy-4 +1%Nb

2.2. *The Porous Media Approach*

The PMA method is a widely recognized technique used to assess the FAs within a reactor core. This method is employed to formulate conservation equations by incorporating the concept of porosity within the control volume. Through this approach, the momentum equations are solved with a high degree of precision for a single-phase fluid. However, the PMA method is not typically used to calculate temperature gradients within the fuel, meaning it does not directly address the variations in fuel temperature [8-10].

2.3. The COBRA-EN code

The COBRA-EN thermal-hydraulic analysis is performed on a series of parallel channels formed by cylindrical fuel rods and open gaps. The axial direction is aligned with the channels, extending from the flow inlet to the outlet. To solve the differential equations governing the flow, the channels are divided into axial segments by planes perpendicular to the z-axis, though these segments may not be evenly spaced. The volumes defined by these axial planes and the lateral boundaries of the channels form a three-dimensional grid of computational cells (control volumes) used for the mass, energy, and momentum balance equations. The COBRA-EN code features two modes for solving the mass, energy, and momentum equations: the three-equation mode and the four-equation mode. In the three-equation mode, these equations are solved for the properties of a fluid mixture, considering the fluid as a combination of liquid and vapor. In the four-equation mode, the mass continuity equations for liquid and vapor are solved separately, while the other equations are handled in the three-equation mode. The separate treatment of mass continuity in the four-equation mode results in more accurate outcomes [16].

In this study, the core hottest FA was first identified based on the highest power generated in each assembly [10]. Next, using the PMA method and COBRA-EN code under identical conditions (same boundary and initial conditions), the core was modeled to calculate the thermo-hydraulic parameters by solving the coupled conservation equations. Specifically, two sets of material porosity equations and mathematical formulations are used to develop the system of equations for the porosity environment. The conversion equations in PMA method, which include mass, momentum, and energy equations for fluid flow, are solved using subchannel analysis and are consistent with earlier research. Ultimately, the equations derived from the PMA method are discretized using the finite volume and lattice methods. As a result, the resulting system of nonlinear algebraic equations is solved through numerical iteration techniques [13]. The PMA method flowchart is illustrated in Fig. 2.

The neutronic results from our previous study showed that,

The peak power density at the center of the core with the TVS-2M FA has decreased, causing the power density distribution to flatten. Additionally, the thermal power of the hottest FA decreases as well, with its value dropping from 22.4 MW in the TVS fuel to 21.5 MW in the TVS-2M FA configuration. Although the number of fuel rods increases from 311 in the TVS FA to 312 in the TVS-2M FA, the reduction in power is primarily due to the presence of the consumable poison Gd2O3 in the central regions of the reactor core. The absorption cross-section for gadolinium is several times greater than that for boron absorption [10].

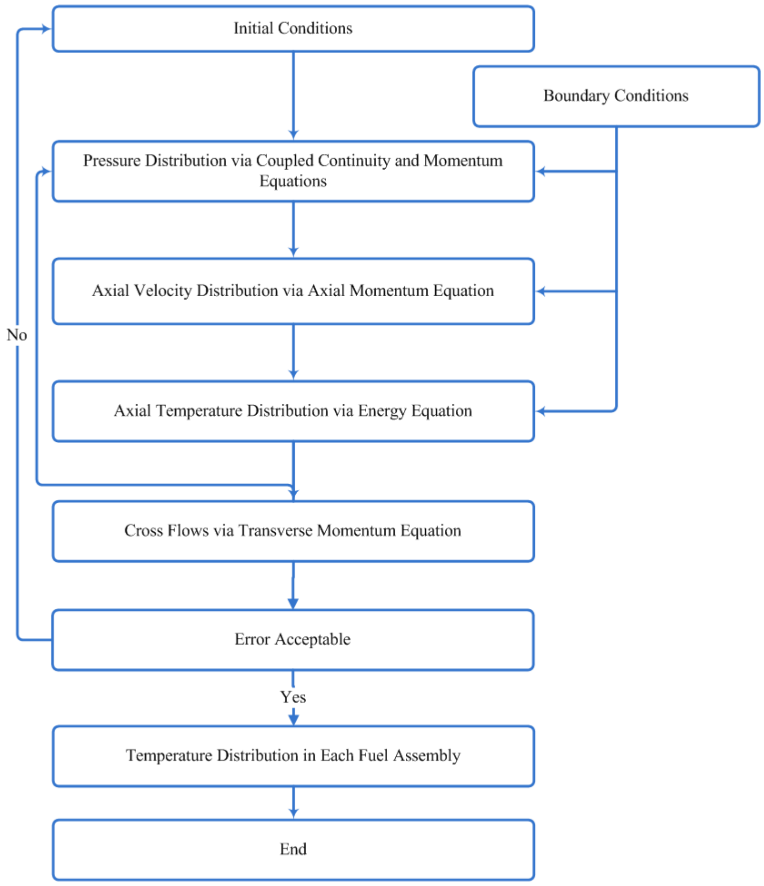


Fig. 2. The PMA method flowchart.

3. Results and Discussion

Results showed, the typical core coolant temperature varies between 291°C and 322°C. However, in the hottest FA, the coolant temperature can peak at around 328°C (Figs. 3 and 4). This difference highlights the thermal variations within the system, with certain areas experiencing higher temperatures due to localized heat generation. This temperature disparity is a critical factor in reactor operation and safety management, necessitating precise monitoring and control to ensure optimal performance and prevent overheating. The average density of the coolant as it enters the reactor core fluctuates between 742 kg/m^3 and 675 kg/m^3. This variation is influenced by several factors, including temperature changes, pressure conditions, and the specific dynamics of the reactor's operation. Diligent oversight and regulation of coolant density are crucial to avoid any possible complications associated with coolant flow rate and the stability of the system. The axial coolant density changes in the hottest FA are demonstrated in Fig. 5. Eventually, the results analysis from the two methods in this study shows an acceptable agreement of the calculations.

Advances in Energy Science and Technology
Vol. 1, No. 1|1 June 2025
AEST

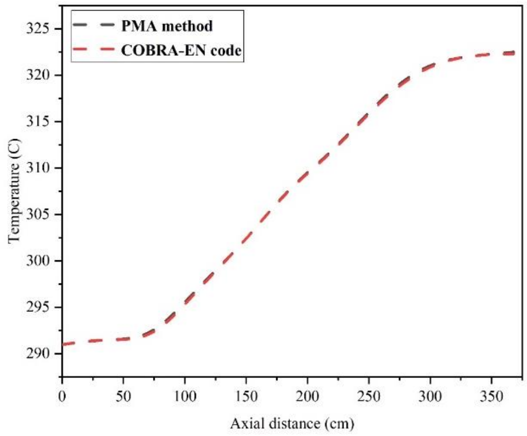


Fig. 3. Axial coolant average temperature in the reactor core.

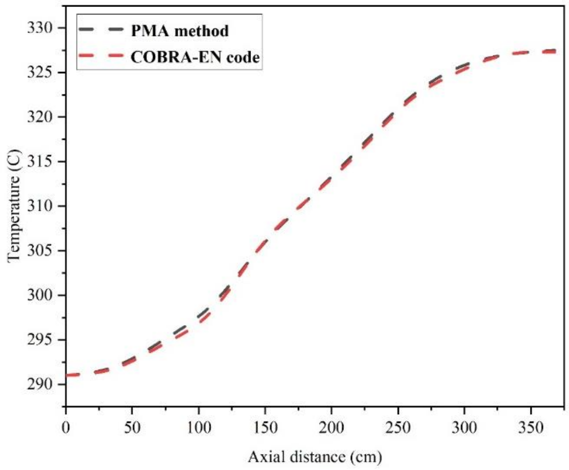


Fig. 4. Axial coolant temperature in the hottest FA.

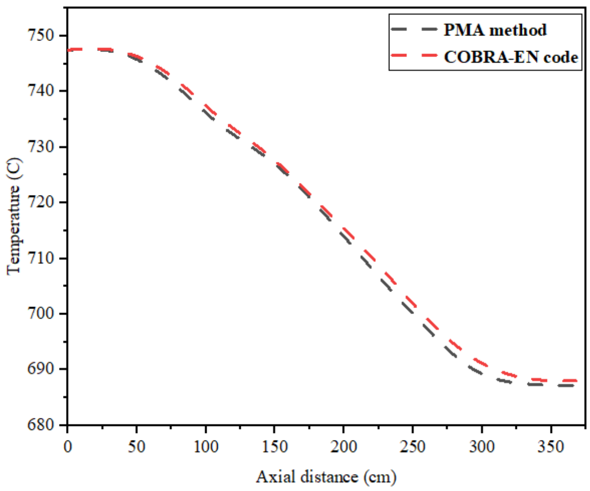


Fig. 5. Axial coolant density in the hottest FA.

Due to the use of Gd_2O_3 as burnable poisons in TVS-2M fuel types, the power peak in the middle of the core has decreased, making the power distribution tend towards flattening. This result leads to a more balanced axial temperature distribution in the reactor core, such that the coolant temperature distribution in the hottest TVS-2M FAs is lower compared to TVS, and the axial coolant temperature distribution in the coldest TVS-2M configuration is higher compared to TVS arrangement. In TVS-2M FAs, the presence of ADF inlet filters and the resulting resistance to incoming flow, along with a 15 cm increase in fuel length, as well as changes in the number and size of the FA spacer grids with a 10 cm increase in height and a reduction in the number from 15 in standard fuels to 13 has caused the fluid pressure drop along the FAs to be around 0.05 MPa. Meanwhile, in TVS FAs, the pressure drops along the fuel channels is approximately 0.048 MPa. Fig. 6. showed the pressure drop into the core with TVS-2M FAs configuration.

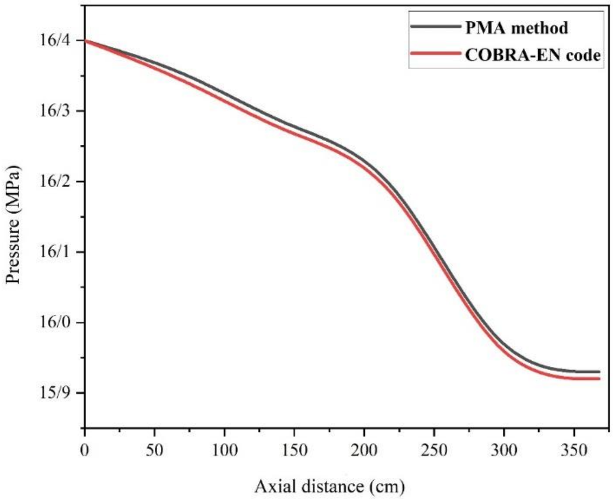


Fig. 6. Axial distribution of coolant pressure variation.

Comparison the coolant temperature presented that, due to the reduction of the thermal power of the hottest FA from 22.4 MW to 21.5 MW, the temperature difference in the hottest channel of the two types of fuel bundles is about 2°C. However, this causes changes in other coolant properties such as density. Due to the higher temperature of the fluid in the TVS fuel, the coolant density in this fuel bundle will be lower than that in the TVS-2M fuel. Additionally, considering the same flow rate at the boundary conditions for both fuel bundles, the fluid velocity in the TVS-2M fuel is slightly lower than in the TVS fuel. Given the temperature difference in the fluid of the two types of fuel bundles, the radial temperature distribution in the center of the hottest FA rods is also affected by this temperature difference, showing a similar behavior with slight differences. Moreover, the COBRA-EN code and PMA method results are very close to each other which indicates the high accuracy of the calculations using PMA method. Fig. 7. showed the average radial temperature distribution in the hottest FA.

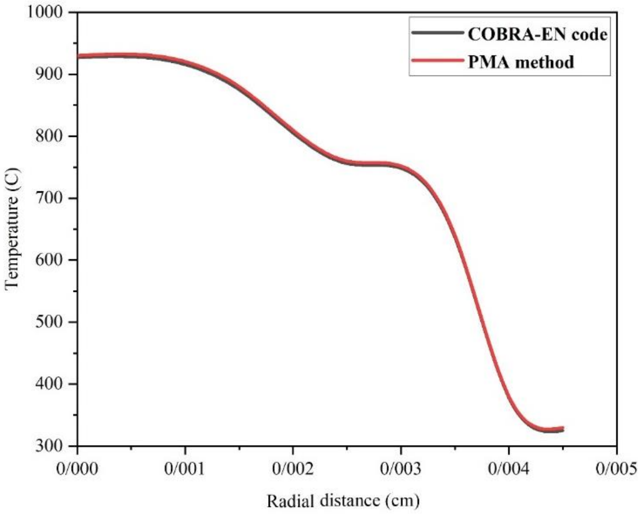


Fig. 7. Average radial temperature distribution in the hottest FA.

One of the important safety parameters of nuclear reactor cores is the Departure from Nucleate Boiling Ratio (DNBR) parameter, which is considered a significant safety constraint. The Critical Heat Flux (CHF) is calculated using the EPRI correlations, which are also employed in the COBRA-EN code [16-17]. Hence, to calculate the CHF or the DNBR, the following correlations are called.

$$q''_{CHF} = \frac{1}{0.0036} \frac{AF_A - X_{in}}{CF_C F_g F_{nu} + \left(\frac{h - h_{in}}{0.0036 q'' h_{fg}}\right)} \tag{1}$$

F_A, F_C, F_g, and F_{nu} are selected parameters used to correct the CHF for various conditions, and their value is typically 1.0 under standard operating conditions. Corrections related to the fuel holding grid spacers, C_g, and in this case, F_g is adjusted as follows:

$$F_g = 1.3 - 0.3C_g \tag{2}$$

The correction for the non-uniform axial heat flux at the axial level Z is written as bellow:

$$F_{nu} = 1.0 + \frac{Y - 1}{1 + 0.0036G}$$
$$Y = \frac{\int_0^z q''(z)dz}{q''(z)z} \tag{3}$$

In a uniform axial heat flux, the value of Y is equal to 1.0. Finally, DNBR is obtained through the following relationship. Figs. 8 and 9 pictured the CHF and DNBR axial distribution in the hottest FA by COBRA-EN code.

$$DNBR = \frac{q''_{CHF}(nu)}{q''(z)} \tag{4}$$

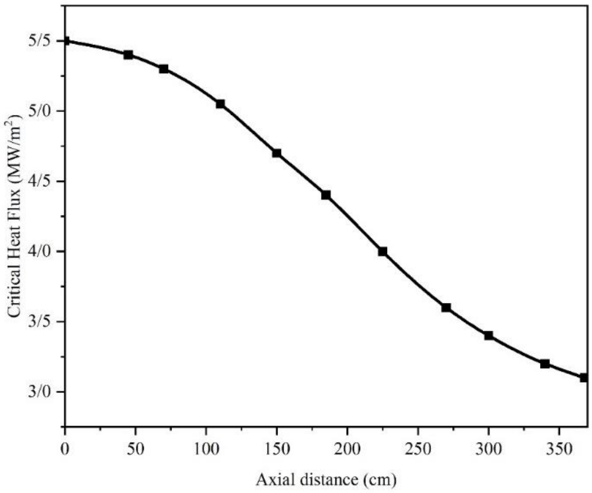


Fig. 8. CHF axial distribution in the hottest FA by COBRA-EN code.

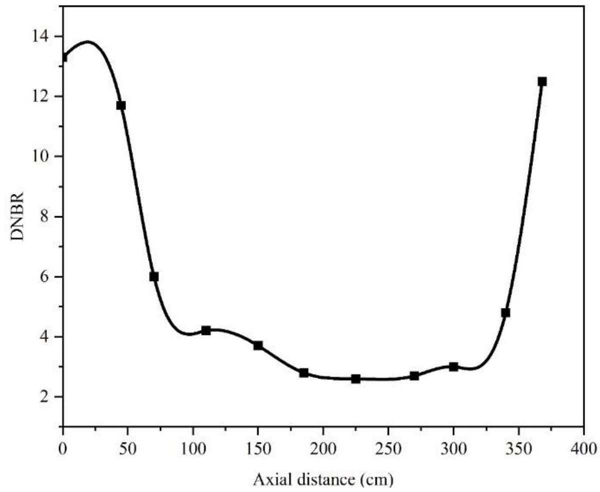


Fig. 9. DNBR axial distribution in the hottest FA by COBRA-EN code.

The minimum value of DNBR in the hottest TVS-2M FA occurs at a height of 2.208 m (DNBR=2.54). According to the BNPP safety analysis report, the minimum DNBR should not be less than 1.66. Hence, the use of TVS-2M fuels will maintain the reactor's safety margin.

4. Conclusion

The primary objective of this paper is to model the hottest FA of BNPP reactor core using the PMA method. The goal is to assess key thermal-hydraulic parameters, including fuel rod temperature, coolant temperature and density, as well as other relevant variables, across various axial and radial zones within the reactor core. Hence, the geometry is simplified and model the medium as a porous material to facilitates the calculation of thermal and fluid dynamic properties that affect reactor performance. Results presented that, the average core coolant temperature varies between 291°C and 322°C. In the hottest FA, the coolant temperature can reach approximately 328°C. Meanwhile, the average coolant density entering the core ranges from 742 kg/m³ to 675 kg/m³. Additionally, in this hottest FA, the fuel rod temperature ranges from roughly 310°C to 930°C. The CHF axial distribution in the hottest FA by the COBRA-EN code showed that its maximum and minimum values vary axially from 5.5 MW/m^2 to near 3 MW/m^2. Moreover, the comparison of the results of the COBRA-

EN code and the PMA method indicates an acceptable accuracy of the calculations. For further studies, the core safety assessment during normal operation and accident scenarios with TVS-2M FA configuration and TVS-2M fuel mechanical analysis are recommended.

Ethical Consideration

The authors of the article certify that all ethical principles related to research has been completely met.

Conflicts of Interest

The authors declared that they have no conflicts of interest in this paper. Also, we declare the following financial interests that represents a conflict of interest in connection with the research works submitted.

Data availability

No data was used for the research described in the article.

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