ALGERIAN JOURNAL OF ENGINEERING AND TECHNOLOGY

Algerian Journal of Engineering and Technology
ISSN: 2716-9227 (Print)
EISSN: 2716-9278 (Online)
DOI:
Mokhtari et al / vol. 10, n°2 - 2025

Original Article Open Access

Steady-State thermal-hydraulic and neutronic analysis of the NUR research reactor current configuration

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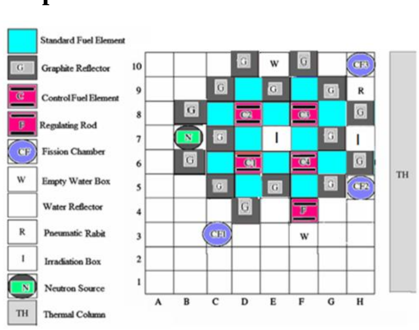
Article history: Received 01 September 2025, Revised 29 September 2025, Accepted 05 October 2025

ABSTRACT

Maintaining adherence to safety standards and limitations throughout a nuclear reactor's operational life is crucial to preventing incidents and accidents that could have detrimental effects on workers, the general public, and the environment. The integrity of the fuel, and specifically the first safety barrier (the fuel cladding), must be maintained in order to reduce the likelihood of an accident. Compliance with these limits ensures that even at the hottest point of the reactor core, the safety limits cannot be exceeded. This study provides a thermal-hydraulic and neutronic analysis of the currentconfiguration of the NUR research reactor under steady-state conditions. Its primary aim is to ensure that all the critical thermal-hydraulic parameters uphold margins below the safety limits. The neutronics calculations were performed using OpenMC code validated by the obtained results of WINS/CITVAP ,power density distribution and Power Peaking Factors (PPFs) were calculated. The PPFs for each channel in the core are obtained, and the hot one is then localized. These results were injected in thermal-hydraulic model established by PARET code, in which the core was divided into two regions (two parallel fuel platesand their associated cooling channels). The first region represents the hottest channel in the core, and the second one, the remainder part named the average channel. This model provides the evolution of the fuel, coolant, and cladding temperatures in the hot channel. The temperature profiles generated were compared to those of the asymptotic model of a previous work and those acquired by the TERMIC.1H code in order to validate the PARET model for the reactor core. Consequently, under steady-state conditions, the clad's maximum temperature remained well below the safety limit. The most significant obtained result is that the NUR research reactor can safely operate at a nominal power while staying within the thermal safety limitations.

Keywords: OpenMC; PARET; Clad temperature;NUR research reactor; Steady state; Critical Heat Flux Ratio CHFR; Safety limits.

Graphical abstract



Current configuration of the NUR Research Reactor

Recommended Citation
Mokhtari O, Radji L. Steady-State thermal-hydraulic and neutronic analysis of the NUR research reactor current configuration. *Alger. J. Eng. Technol. 2025, 10(2): 87-101*

1. Introduction

A nuclear reactor's safety analysis requires an understanding of thermal-hydraulics and neutronics phenomena; computation algorithms are needed to facilitate their quantification. Research reactors that are cooled and moderated by water are limited from the thermal point of view by critical phenomena leading to boiling crisis and damage in the fuel elements [1].Thermal-hydraulic assessments are classified into steady and transient states based on the reactor's operating mode. While the transient phases are used to simulate a reactor accident, the steady state represents the reactor's normal operation [2]**.**

The thermal-hydraulic objective of the design is to safely remove the heat generated in the fuel without producing excessive fuel temperatures or getting too close to the critical heat flux. Under steady-state operating conditions, local boiling should not be reached at any point in the reactor core [3].The main objective of this study is to ensure that all important thermal-hydraulic parameters keep margins well below the safety limits. Therefore, we determined the maximum fuel, cladding, and coolant temperatures, as well asdeparture from nucleate boiling ratio (DNBR) variation along of hot channel. These calculations are performed using PARET [4] thermal-hydrauliccode, with an inlet temperature of 40°C and an inlet pressure of 175kPa.

The OpenMC code was used to establish a three-dimensional geometric model of the current configuration of NUR reactor core. Developing an accurate physical model of the reactor requires validation. For this configuration, several neutron analysis studies were conducted [7, 8, 9]**.**The best way to validate a calculation code is to compare the simulation results with experimental ones [10] and/or by inter-comparison of two calculation codes results [7, 8]**.**This step demonstrated the validity of using the OpenMC code. The comparison focused on the K_{eff} value and the neutron flux, the main parameters affected by the model geometry.

The determination of the power density distribution and associated PPFs in nuclear reactors allows the location of the hottest point in the reactor core and thus the determination of the maximum fuel temperature and consequently the maximum temperature of the fuel cladding, which constitutes an important safety limit.The safety studies took into account two heat limitations in order to satisfy these safety requirements. The first one is the peak clad temperature, which is a direct indicator of the physical damage to the fuel plate. The second constraint accounts for the Critical Heat Flux (CHF), which characterizes the Departure from Nucleate Boiling (DNB) occurring at the surface of the fuel plate cladding [11, 12].

This work provides a steady-state thermal-hydraulic and safety analysis of the NUR research reactor. The neutron parameters determined by the OpenMC code are injected as Input into the PARET code in order to determine the hot channel temperatures (coolant, fuel, cladding and DNB) as well as the DNB ratio [13, 14].

2. Core Description

The NUR research reactor is an open-pool light water reactor with a nominal power of 1 MW and a thermal neutron flux of about 10^{13} (n/cm².s). The reactor core consists of an arrangement of MTR (Material Testing Reactor) fuel elements and graphite reflector blocks, allowing flexible configurations sitting on a grid located inside a pool made of stainless steel. This one is filled with demineralized light water used as a coolant (flows downward), moderator, and reflector [15]. The vessel is surrounded by a thick octagonal structure made of heavy concrete, used as radial biological shielding.

Fig 1 depicts the NUR reactor's current configuration (referred to as x-1) as it is modeled in this work. It consists of a neutron source, 15 graphite reflector elements, 3 irradiation boxes, 3fission chambers, 2 water boxes, 12 Standard Fuel Elements (SFE), and 5 Control Fuel Elements (CFE).

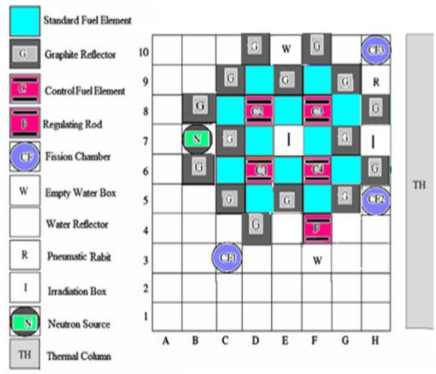


Fig1. Current configuration of the NUR Research Reactor [8]

3. Materials and Methods

3.1. Neutronic analysis

3.1.1. OpenMC code

As a complicated physical system, a nuclear reactor core is modeled by solving the Boltzmann transport equation, which controls how neutrons behave in materials. Therefore, in order to solve this equation, a certain number of techniques have been developed, and incorporated into computer algorithms named calculation codes. Nuclear reactor simulations and analyses are increasingly relying on OpenMC. Developed in Python and C++, this software can be considered as a robust open-source tool. It allows for the accurate modeling of complex geometries. Thanks to its support for parallel computing, OpenMC enables both workstations and powerful computing clusters to perform calculations quickly [16]. It is capable of performing fixed source, k-eigenvalue, and subcritical multiplication calculations on models built using either a constructive solid geometry representation. OpenMC supports both continuous-energy and multigroup transport. The continuous-energy particle interaction data is based on a native HDF5 format that can be generated from ACE files produced by NJOY. In this work the cross-section library was derived from the JEFF-32 dataset sourced from the Nuclear Energy Agency's (NEA) website.

The Computational Reactor Physics Group (CRPG) at the Massachusetts Institute of Technology (MIT) developed the OpenMC Monte Carlo code in the year 2011. The primary aim of the development was focused on criticality calculation that is necessary for nuclear reactor simulation [6].

For modeling complex geometric objects, OpenMC uses a constructive solid geometry representation. Closed volumes, or cells, can be represented as the intersection of multiple half-spaces. Each half-space is in turn defined as the positive or negative side of a plane or quadratic surface; it provides structures that facilitate the modeling of a two or three dimensional grid made up of quadrilaterals [5]. In addition, it has a flexible tally system that enables users to obtain physical results of interest. Combinations of filters and scores define tallies [6].

The free gas approximation will not adequately describe the scattering kinematics for thermal neutrons scattering from bound molecules like hydrogen or deuterium in water, graphite, beryllium, etc.; instead, $S(\alpha, \beta)$ scattering law data must be employed. The $S(\alpha, \beta)$ data are given on ACE files separate from the normal nuclide data[17].

In the case of a nuclear reactor model, neutrons are especially important because they are the particles that induce fission . Knowing the behavior of neutrons allows one to determine how often and where fission occurs. The amount of energy released is then directly proportional to the fission reaction rate since most heat is produced by fission. By simulating many neutrons (millions or billions), it is possible to determine the average behavior of these neutrons (or the behavior of the energy produced, or any other quantity one is interested in) very accurately.

When you build and install OpenMC, you will have an OpenMC executable on your system. When you run it, the first thing it will do is look for a set of XML files that describe the model you want to simulate. Three of these files are required (materials.xml, geometry.xml, settings.xml) and another three are optional (tallies.xml, plots.xml, Input Files using OpenMC's Python API).If you have a moderating material in your model like water or graphite, you should assign thermal scattering data S(α,β) using the Material.add_salpha_beta() method [5,6].

In this work the OpenMC is installed on Linux-x86_64. On Processor An Intel® Core™ i7 (16 Go of RAM), which met the simulation's requirements. The operating system is Ubuntu 22.04 LTS. Simulation Software: OpenMC, version 0.14.

3.1.2.OpenMC model validation:

To model the NUR reactor core using OpenMC code a mesh grid was superposed over the geometry of the core. The SFE and CFE were modeled using the concept of repeated structure in the OpenMC code, which means that the concept of universe is exploited. A fuel plate with half of the adjacent cooling channel on both sides was taken as the universe. The universe is composed of three cells in the case of the NUR reactor, the first cell represents the active part, the second represents the aluminum cladding, and the third represents the coolant, which is water. The universe is then placed in a grid (lattice) with the dimension of a fuel element, as can be seen in Fig 2 obtained with OpenMC-Plotter.

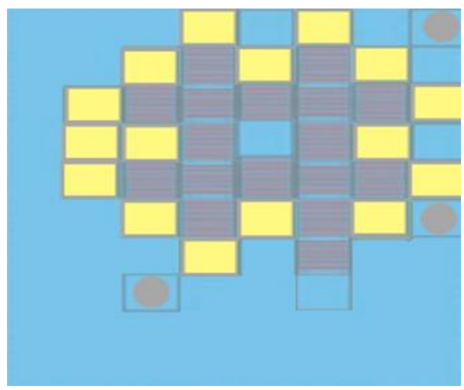


Fig 2. OpenMC Model of the NUR reactor core

The MC (Monte Carlo) model developed in this work includes the most of essential components of the reactor core needed for neutronics calculations using the OpenMC code. The irradiation boxes and the water boxes, which are of the same sections as the fuel element, are modeled by aluminum parallelepipeds filled with water. The fission chamber is modeled by an aluminum cylinder. The modeling of the control fuel element is carried out in 2 stages: first the element is modeled with the 14 fuel plates in the same way as for the standard element, and then the universe of the absorber plate is included in the grid of the control element. In this work, the control rod absorber plate modeling was not taken into consideration, meaning that the comparison is limited to results from fully withdrawn control rods.

First, and before starting to simulate the current configuration, we need to validate the model established with OpenMC. The OpenMC calculation must be consistent with previous published research and/or experimental values. This step will demonstrate the validity of using Open-Source code and will allow us to verify the generated results. The comparison will focus on the k_{eff} effective multiplication factor (or the core excess reactivity ρ) and the flux distribution, since those are mostly affected by the model [18].

To validate the established OpenMC model of the NUR reactor core, the simulated values of excess reactivity, thermal neutron flux in the central irradiation box as well as the radial distribution of thermal neutron flux in the reactor core are compared with the results of previous works calculated using a deterministic computer code namely WIMS/CITVAP[7,8].In this work, the OpenMC was run with 3000 batches, 50 inactive batches to be skipped, and 10000 particles in each batch for the criticality calculations. This resulted in a statistical uncertainty (σstat) of 18 pcm at 1σ.

For the flux calculations, OpenMC was run with 3000 batches, 100 inactive batches to be skipped and 20000 particles in each batch. The statistical error ranges from 2% at the core center to 8 % at its periphery due to the relatively low number of neutrons compared to the center, for the fuel element furthest from the core, the statistical error is about 5%.

The excess reactivity ρ is defined as the positive reactivity in the core obtained when all control rods are completely extracted. It was calculated using the following equation:

$$\rho = \frac{k_{eff}-1}{k_{eff}} \qquad (1)$$

The calculated OpenMC core excess reactivity for the fresh fuel and for the beginning of life BOL [15] of current configuration are compared with experimental results[10], and those of WIMS/CITVAP [7, 8]. Results are shown in Table 1. In the comparison of the core excess reactivity, all the control rods are set at totally withdrawn from the reactor core.

Table 1. Comparison of excess reactivity obtained by OpenMC, WIMS/CITVAP [7, 8] and measurements[10]

Parameters	**Core excess reactivity ρ (pcm)**
Fresh fuel	
This work (OpenMC)	4720.16
WIMS/CITVAP[7]	4848.00
WIMS/CITVAP[8]	4681.00
Dif. (OpenMC/(WIMS/CITVAP[7]),%)	-2.64
Dif. (OpenMC /(WIMS/CITVAP[8]),%)	0.83
BOL[15]	
This work (OpenMC)	3085.75
WIMS/CITVAP [8]	3028.00
Experimental [10]	3192.00
Dif. (OpenMC/(WIMS/CITVAP[8]), %)	1.91
Dif. (OpenMC/Exp, %)	-3.33

Dif = (Calculated-Measured)/Measured x 100

The core excess reactivity value calculated by OpenMC for NUR reactor current configuration deviates in the fresh fuel case from the WIMS/CITVAP code results [7] by about 128 pcm (2.64%) and [8] by about 39 pcm (0.83%). In the case of BOL [15] our results deviates from the measured value by about 107pcm (3.33%) and from WIMS/CITVAP codecalculation[8] by 57.75 pcm (1.91%) as indicated in Table 1. Compared with the different results obtained for similar studies, these results are satisfactory and it can be concluded that there is a good agreement between the values calculated by OpenMC, the experimental values and those obtained by WIMS/CITVAP code in previous works.

OpenMC code was used to calculate the axial and radial distributions of thermal (Energies below 0.625 eV) neutron flux and power density for current configuration, for this purpose, the regions that contain fissile material must be discretized into several cells, flux and kappa-fission scores were used. The kappa-fission score is the rate of production of recoverable energy per fission. The recoverable energy is defined as the kinetic energy of the fission product, the kinetic energies of the prompt and delayed neutrons, the total energies of the prompt and delayed γ-rays, and the total energy released by the delayed β-particles. If a very rigorous approach is desired, a coupled neutron-photon simulation must be performed and the heating score calculated.

OpenMC used a combination of filters, including Regular Mesh for spatial discretization and Energy Filter for energy ranges, to compute the distribution of neutron flux and fission power density with "flux" and "kappa-fission " scores. The origin (X = 0, Y = 0), is defined at the center of the central irradiation box E7.

Since the neutron fluxes in the OpenMC output file are normalized per neutron source and averaged over the computed volume, a normalization factor is applied to obtain absolute fluxes that are comparable to the measurements.

The following formula was used to determine the normalization factor S for a reactor core configuration:

$$\phi\left[\frac{\mathrm{neutron}}{\mathrm{cm^2.s}}\right] = \mathrm{S}\,\phi_{\mathrm{OpenMC}} \qquad (2)$$

With

$$S = \frac{P\,nu}{Q\,k_{eff}} \qquad (3)$$

- ϕ :Absolute flux
- ϕ_{OpenMC} :Flux calculated by OpenMC(neutron.cm/source)
- P :Reactor power (J/s)
- Q = 200 MeV or 3.2x10-^{11} (J/fission)for U-235
- nu: Number of neutrons per fission(neutrons/fission)
- k_{eff} :Effective multiplication factor [neutrons/source]

Due to unavailability of the experimental neutron flux distribution of the NUR reactor core, An intercomparison between the results presented in this work and those obtained by WIMS/CITVAP code [7] was conducted in order to validate the OpenMC code.

The OpenMC and previous works [7] (using WIMS/CITVAP code) results for the radial neutron flux distribution are plotted and compared in Fig 3.

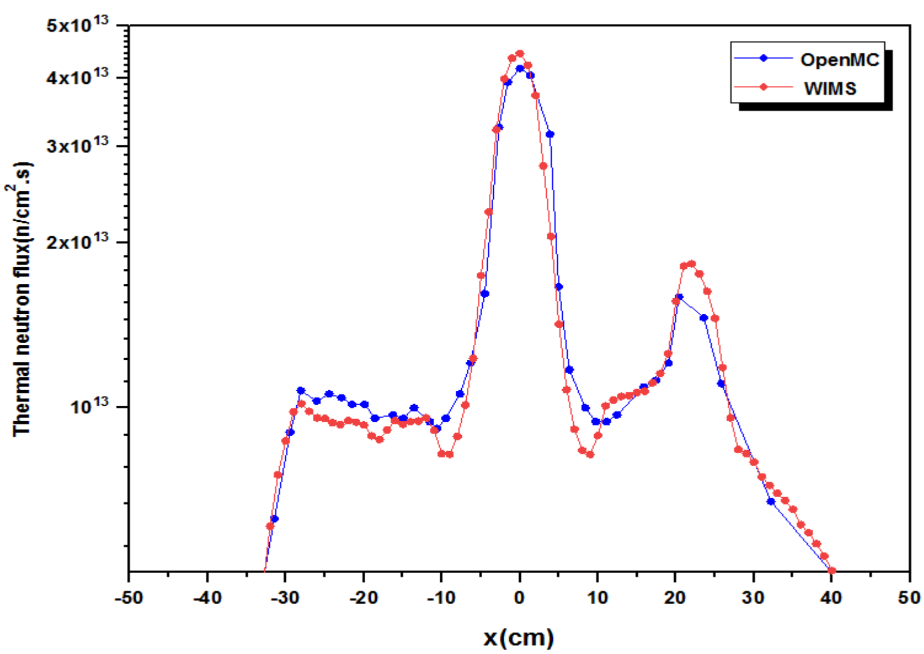


Fig 3.Thermal neutron flux at mid-plane along the x-axis (row 7)in the NUR reactor core

As illustrated in Fig 3, our computed OpenMC results show an overestimation with respect to the WIMS/CITVAP code [7] values. This issue has also been noted in previous research and has been linked to the distinction between the two codes; the stochastic OpenMC computational code and the deterministic code WIMS/CITVAP [7]. Therefore; the difference between the results is likely due to the disparities between the established models.

From the same figure we can deduce that the thermal neutron flux shows a significant peak in the E7 position (central irradiation box) due to the presence of water (neutron moderating element). At this location, according to OpenMC, the maximum thermal neutron flux calculated in the NUR reactor core is 4.44 E+13 (n/cm².s). This value agrees well with the measured value of 3.98 E+13 (n/cm².s) [10] and that calculated by WIMS/CITVAP code [7] in which the maximum thermal flux value is 4.17E+13 (n/cm².s).

Results of this study show that the OpenMC model established for the current configuration is quite accurate and validated against experimental values and deterministic calculations.

3.1.3. Power density distribution and PPFs calculation

In the M.C. calculations of the power density distribution, we assumed that power density is proportional to the fission density; which implies that all recoverable fission energy is deposited at the fission point [28].

Since the kappa-fission score in the OpenMC is normalized per neutron source and averaged over the computed volume, a normalization factor is applied to obtain the power density values.

The following formula was used to determine the normalization factor C for a reactor core configuration:

$$\text{Power density}\left(\frac{\mathrm{W}}{\mathrm{cm}^3}\right) = \mathrm{C} * (\text{Kappa} - \text{fission})_{\text{OpenMC}} \quad (4)$$

With

$$\mathrm{C} = \frac{\mathrm{P\,nu}}{\mathrm{V.Q}*\mathrm{k_{eff}}} \quad (5)$$

- P : Reactor power (Watts)
- Q = 200 MeV for U-235
- nu: number of neutrons per fission (neutrons/fission)
- k_{eff} :effective multiplication factor [neutrons/source]
- V : volume of interest (cm^3)

The power density distribution was performed by calculating the rate of fission energy deposition in each discrete unit cells of a detailed three-dimensional geometry of NUR core reactor. The maximum power density is located in the fuel element E6. For the rest of the calculations, this element will constitute the hot channel.

In order to determine the axial distribution of the power density in the NUR reactor core, only the hottest fuel element (E6) is considered. The average power density in each plate of this element is calculated. Once the hottest plate, is located, a calculation of the axial power density is carried out, which at the same time represents the axial distribution of the power density of the reactor core. The fuel plate in question is divided into 20 segments in the axial direction, the power densities were calculated in each segment. Fig 4 gives the axial distribution of thermal power density in the hottest channel.

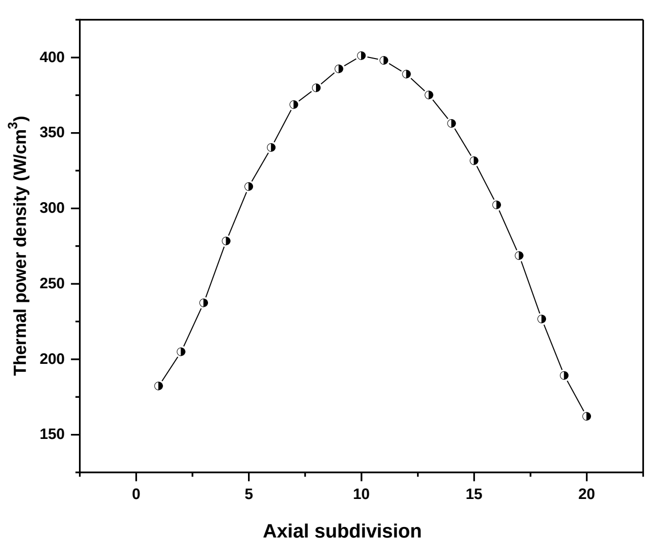


Fig 4.Axial power density distribution through the hot channel calculated ByOpenMC in the NUR reactor core

This figure shows the axial distribution of the thermal power density in the active part of the hottest channel calculated by OpeMC. According to this, the maximum thermal power density occurs practically in the center of the fuel plate, and the lowest at the top and bottom of it due to leaks.

In this study, the radial Fr and axial Fz peaking factors were multiplied to determine the total power peaking factor Ft. The radial power peaking factor Fr, is taken to be the ratio of the radial average power density in a fuel plate to the radial average power density of all the fuel plates in the core [29]. The axial power peaking factor is defined as the ratio of the maximal axial power density to the average axial power density in the hottest fuel plate [29, 30].

According to the PPFs definitions given above, the computed axial PPF, Fz is 1.31 and the computed radial PPF, Fr is 2.29. Consequently, the total PPF, F_T assumes a value of 2.99.

3.2. Thermal-hydraulic analysis

Research reactors can be analyzed under steady state and transient conditions using the coupled thermal hydrodynamic and point kinetics capability of the computer code PARET [4].One to four regions can be used to represent the core in this code. Originally developed at the Idaho National Engineering Laboratory for power reactor conditions (Obenchain, 1969), this code was modified by Argonne National Laboratory (ANL) [2, 4, 19] and is especially well-suited for plate type research reactor safety assessments. With use of point kinetics, one-dimensional hydrodynamics, and on-dimensional heat transfer, PARET is essentially a linked neutronic-hydrodynamic heat transfer algorithm [20]. Several reactors across the world use it for safety evaluations for both normal and non-destructive reactivity events and those caused by coolant flow loss [2, 11]. The code computes many parameters as a function of time and space, including the following: fuel, cladding, and coolant temperatures; enthalpy, reactor power, critical heat flux ratio; pressure drop and heat transfer coefficient along each channel [20, 21,22, 23].

Input of PARET includes physical dimensions and geometry of the reactor core, fluid flow parameters; initial system pressure; thermal properties of the fuel element materials, delayed neutron information, power density distribution , reactivity coefficients, initial power level..etc.The output file contains all information produced by the code and requested by the user including: coolant temperature, mass velocity, void fraction, fuel center temperature, fuel surface temperature, clad surface temperature, surface heat flux, burnout ratio, local and total pressure drop across channels,it contain data for a given channel at every time step[20, 24].

In order to predict steady-state thermal-hydraulic behavior of current configuration of NUR reactor, PARET code, a channel-type computer program, was used to conduct the study. The two-channel model approach was used, which consists of the hottest channel with the highest power density; the other core channels are taken as average channel. Each region may have different input parameters like power peaking factors, coolant velocity, and hydraulic parameters. In the axial direction, the two regions were subdivided into twenty equal sections.

Heat transfer in the fuel plates is determined from the solution of one-dimensional conduction equation in the radial direction. A variety of heat transport correlations are supported by this code. The Onset of Nucleate Boiling (ONB) is determined using the Bergles and Rohsenow [25] correlation. The Forgan-Whittle correlation calculate the heat flux at the onset of flow instability [26]. For the CHF (Critical Heat Flux) calculation, the Mirshak correlation [27] is selected since it is appropriate for low-pressure plate-type research reactors.

When nucleate boiling occurs, bubbles of vapor that form on the surface of the clad are collapsed by the surrounding liquid and both liquid and vapor are in contact with the surface. If the bubbles coalesce and blanket the surface with vapor or the surface dries out, then a condition, referred to as “departure from nucleate boiling” (DNB) and sometimes called “burn-out”, has occurred. The heat flux at which this occurs is called the “critical heat flux” (CHF) [24]. For water-cooled reactors, the deviation from nucleate boiling ratio, or DNBR, is a crucial design parameter that serves as an operational safety limit.

The DNBR evaluation is traditionally performed by thermal-hydraulics analysis code. The general practice is to estimate the critical heat flux using empirical correlations that have been validated by the experiment.

For calculation of critical heat flux, Mirshak correlation [27] is used in this work. The correlation is given as:

$$q_c^{"} = 151.(1 + 0.1198\,U)(1 + 0.00914\,\Delta Tsub)(1 + 0.19\,P) \quad (6)$$

Where

$q_c^{"}$ = Critical heat flux(W/cm^2)
U = Coolant velocity (m/s)
ΔTsub = Exit water sub-cooling (°C)
P = Pressure (bars absolute).

The DNBR is the ratio of the limiting heat flux for the occurrence of DNB to the actual local heat flux of the fuel plate. A DNBR value of at least 1.2 is generally specified for research reactors in each core channel at maximum operating power.This ratio is determined by [23]:

$$DNBR = \frac{q_c^{"}}{q_{actual}^{"}} \quad (7)$$

$q_c^{"}$:Critical Heat Flux

$q_{actual}^{"}$:Actual heat flux

The ratio between the predicted heat flux and the actual operating heat flux is called the departure from nucleate boiling ratio (DNBR). This ratio changes across the length of the fuel plate and reaches a minimum value called MDNBR, which is one of the important thermal-hydraulic safety parameter.

According to the core model assumed in PARET, in this study, the hottest channel is divided axially into 20 equal sections in order to comply with the axial distribution of power density determined by the neutronic code OpenMC.

The determination of the axial temperature distribution using the PARET calculation code is done as follows:

1. The reactor core is divided into two channels. One channel represents the hottest fuel plate, while the rest represent average fuel plates.

2. A number of discrete nodes are used to calculate the axial length of each channel, as seen in studies that use the maximum number of axial nodes allowed in PARET.

3.The model takes into consideration the axial power distribution, which is frequently user-defined by the neutronic calculations, and adds heat transfer correlation.

A set of temperature profiles for the coolant, cladding, and fuel along the axial direction is the result, illustrating the temperature variations from one end of the fuel plate to the other.The flow chart of the steady-state neutronic and thermal-hydraulic calculations for the NUR reactor core is presented in Fig 5.

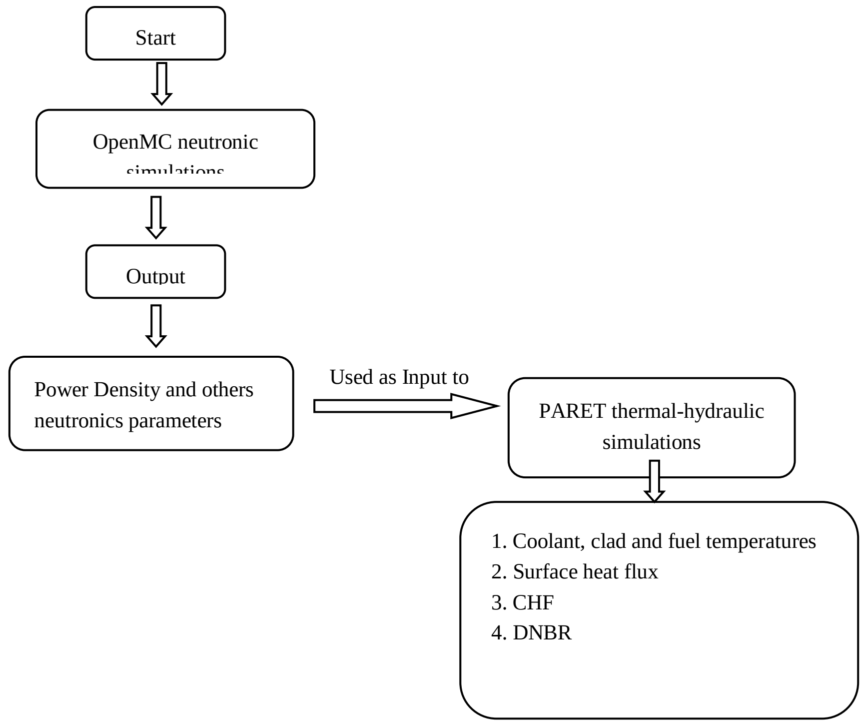


Fig 5. Steady state calculation flow chart of the NUR reactor core.

4. Results and Discussions

The OpenMC code is used in this work to compute the power density and power peaking factors for the NUR research reactor's current configuration. The results obtained give the PPFs for each fuel plate of the reactor core, and the hot channel is subsequently determined. Once the hottest channel is located, a thermal-hydraulic model using PARET code is applied. This model illustrates the temperature evolution of coolant, clad and the fuel. The obtained results are compared with those of previous research.

4.1. Axial temperature distributions

The objective of the thermal-hydraulic model is to determine the thermal safety margin and to ensure that the fuel cladding integrity is maintained during steady state, as well as during abnormal conditions at full power. For PPFs obtained by the neutronic code OpenMC, coolant and clad temperatures evolutions in the hot channel are provided by the thermal-hydraulic model.

Temperature profiles of coolant, clad, fuel, and DNB of the NUR research reactor operating at full power in the hottest channel are shown in Fig 6. The coolant is light water with an inlet temperature of 40°C with a downward flow.

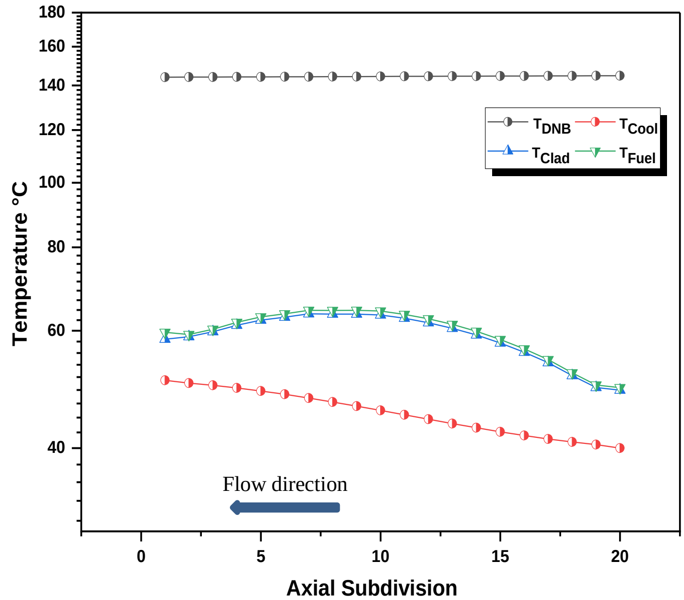


Fig 6. Hot channel axial temperatures distributions at 1MW of the NUR reactor core

According to the computation results, under normal operating conditions at full nominal power of 1 MW, the maximum fuel cladding temperature of hot channel was about 63.58°C, remaining below the temperature limit imposed by the manufacturer (90°C) to avoid the increase of the corrosion rate of the clad.

The maximum temperature of the coolant is 50.56°C, which is lower than the ONB temperature (Onset of Nucleate Boiling) set at 118°C. The ONB is taken as a warning in steady-state conditions, as it does not actually correspond to any critical event (allowing the avoidance of the progressive formation of a vapor film between the clad and the coolant, which degrades the heat transfer process). And it is much lower than the DNB temperature of 144°C calculated by PARET code.

Our results were compared to those of previous work based on the asymptotic methods [31]; the comparison was made for the maximum temperatures of the clad and coolant with the results obtained by using the probabilistic method for the calculation of neutronic parameters. Since the neutronic parameters used as input for the PARET thermal-hydraulic code in our work come from a Monte Carlo calculation (OpenMC code), a comparison with the probabilistic approach was performed.

Table 2 presents a comparison between the obtained results from this study and those of a previous work on the maximal temperatures of coolant and clad.

Table 2. Comparison of PARET and asymptotic methods [31] results of the maximal temperatures the coolant and the clad for the NUR reactor core

Parameters	**Maximum coolant temperature °C**	**Maximum clad temperature °C**
This work (PARET)	50.56	63.58
asymptotic methods [31]	53.17	69.44
Dif. (PARET/ asymptotic methods [31], %)	5.16	9.2

These results show that the maximum temperatures obtained by the two models present a difference of 9.2% for the clad temperature and 5.16% for the coolant temperature. This difference is due to the fact that the first results are based on analytical solutions of the chosen equations (asymptotic methods) [31], while we used the PARET code with its own integrated equations.

4.4. DNBR calculation

The departure from nucleate boiling ratio (DNBR) is the ratio of the predicted heat flux and the actual operating heat flux.This ratio is one of the crucial thermal-hydraulic safety parameters that must be assessed for the safe operation of nuclear reactors.It varies throughout the length of the fuel plate and reaches a minimum value known as MDNBR. The hot channel axial DNBR distributions for the NUR research reactor at full power of 1MW, as determined by PARET, is shown in Figure 7.

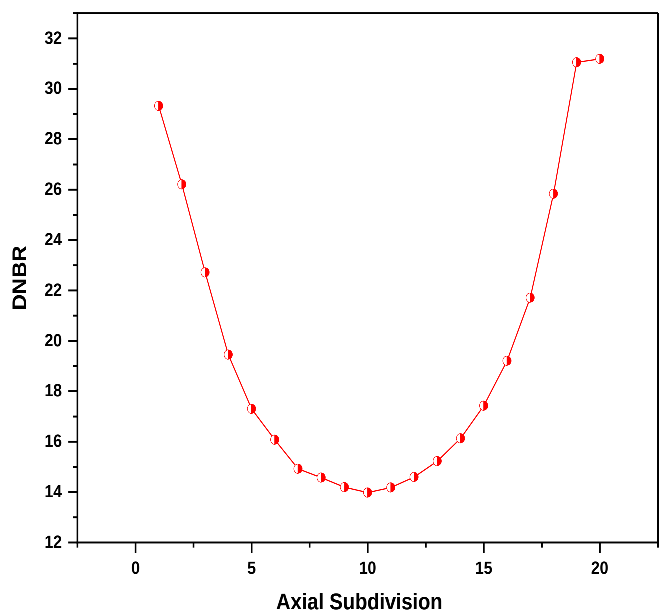


Fig 7. Hot channel axial DNBR at full power for the NUR reactor calculated by PARET.

Fig 7 shows that, for a given set of operating parameters, the DNBR varies along the flow channel. At the coolant inlet, it is 31.18, indicating a low heat flux, and a corresponding high critical heat flux of 192.72 W/cm². The system is therefore well within the safety limits. As the fuel temperature increases, the DNBR decreases to a minimum of 13.98 at the 11th subdivision around the axial center of the fuel plate, a value considered to be well outside the safety limit for this parameter. The minimum DNBR is well above the recommended value of 1.2. Thus, the NUR research reactor can operate at full power in the current configuration without worrying about the Departure from Nucleate Boiling (DNB).

For calculating DNBR, critical heat flux CHF needs to be evaluated. The heat fluxes are determined by the correlations mentioned above.

Fig 8 shows the evolution of Critical Heat Flux, Surface heat flux and DNBR along the hot channel.

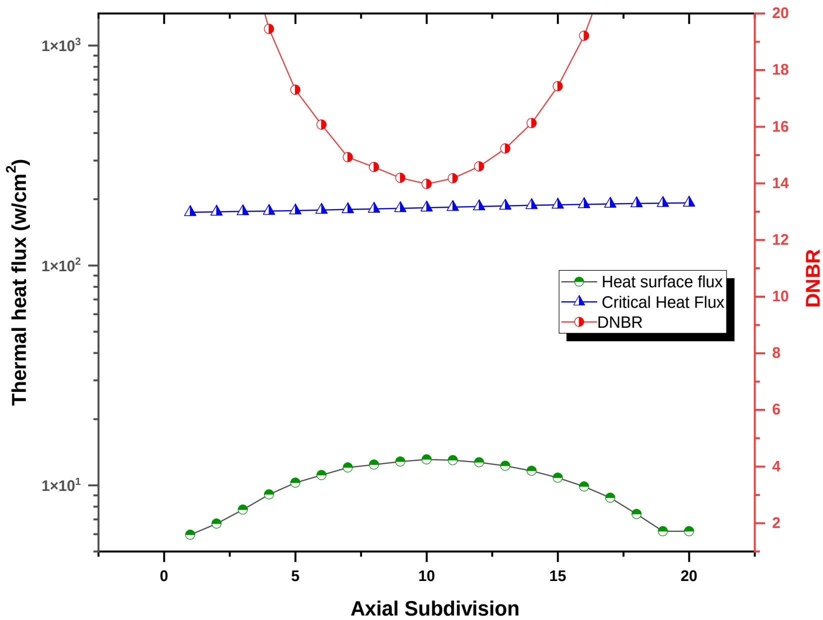


Fig 8. Evolution of Critical Heat Flux, Surface heat flux and DNBR along the hot channel.

Fig 8 shows that the critical heat flux gradually drops from a maximum value of 192.72 W⁄cm2 at the channel entry to 174.57 W⁄cm^2 at the outlet. while the heat flux curve at the surface of the clad takes a bell shape with a maximum of 13.10 W/cm^2, then gradually decreases towards the outlet (due to the use of the Mirshak correlation [27] which is function of ΔTsub which decreases).

The temperature profiles generated were compared to those acquired by the TERMIC.1H code [32] in order to validate the PARET model for the reactor core. TERMIC.1H code was developed to perform the thermal design of research reactors, provides information about heat flux for a given maximum wall temperature, ONB, and DNB. The maximum temperatures obtained using TERMIC.1H code in this study are compared to those of PARET code and the result of the asymptotic methods [31] in table 3.

Table 3. Comparison of PARET TERMIC.1H and asymptotic methods [31] results of the maximal temperatures of coolant and the clad for the NUR reactor core

Parameters	**Maximum coolant temperature °C**	**Maximum clad temperature °C**
This work (PARET)	50.56	63.58
This work (TERMIC.1H)	51.40	67.40
asymptotic methods [31]	53.17	69.44
Dif. (PARET/asymptotic methods [31], %)	5.10	9.2
Dif. (TERMIC.1H/asymptotic methods [31], %)	3.44	3.10
Dif. (TERMIC.1H/ PARET, %)	1.66	5.90

Using the TERMIC.1H code, the coolant's and the clad's maximum temperatures were found to be 51.4°C and 67.4°C, respectively. From table 3 we can show that the relative difference between these two models (TERMIC.1H and asymptotic methods [31]) is 3.44% and 3.1% for the maximum temperatures of coolant and clad, respectively. This is a very satisfactory difference (less than 5%). However, the values of the asymptotic methods [31] remain more conservative than those obtained by TERMIC-1H code. The difference is due to the use of other varieties of correlations for the calculation of temperatures**.** TERMIC-1H considers the exit sub-cooling (ΔTsub) in the Mirshak correlation [27] an independent variable, and this greatly overestimates the DNB margin.

Results for the maximal temperature showed good agreement between the TERMIC-1H and PARET codes. The relative difference between their results is 1.66% for the maximum coolant temperature and 5.9% for the maximum clad temperature. The results of the asymptotic methods [31] for the clad temperature are 3% higher than those of TERMIC-1H and 9% higher than those of PARET. The comparison shows that the results of the asymptotic methods [31] are more conservative than those of TERMIC-1H and PARET.

5. Conclusions

In this work, neutron calculations were performed using the OpenMC code and compared with those of the deterministic WIMS/CITVAP code. This comparative study yielded good agreement for the criticality calculation and the axial and radial flux distributions. The results obtained allowed us to conclude that the OpenMC model is validated for neutron calculations of the current configuration of the NUR reactor. The power density distribution and Power Peaking Factors (PPFs) calculated were used as input for the thermal-hydraulic PARET code to determine the fuel cladding temperature, which is an important safety limit.

The maximal temperatures generated for clad and coolant are compared to those acquired by the TERMIC.1H code and asymptotic methods in order to validate the established PARET. Comparison between the different codes results shows that the asymptotic methods are more conservative than the TERMIC.1H and PARET codes. This analysis also ensures that all safety related thermal-hydraulic parameters are within the steady-state thermal design limits. Therefore, it is concluded that the NUR reactor can operate in its current configuration safely at its nominal power of 1 MW while respecting the thermal designand safety limits.

Ethical Statement

This study does not contain any studies with human or animal subjects performed by any of the authors.

Conflict of Interest

The authors declare that they have no conflict of interest.

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