CONCEPTUAL DESIGN OF AN INNOVATIVE MOLTEN METAL MICRO MODULAR REACTOR (4M)

BABAK ALIYARI [1], MOHAMAD RAHGHOSHAY [1], OMID NOORI-KALKHORAN [2]
[1] Dept. of Nuclear Engineering, Science and Research Branch, Islamic Azad University, Tehran, Iran
[2] Faculty of Nuclear Engineering, Shahid Beheshti University, Tehran, Iran
e-mail: babak.aliyari.edu@gmail.com

Abstract

In this paper, a thermal-neutronic conceptual design of a subcritical VSMR which core will be critical by use of a neutron generator and with modified inherent safety is presented. In this reactor, hereafter named ″4M″, the heat is generated by uranium fission being extracted by using liquid lead as primary coolant and helium flow for the secondary circuit. Helium removes the heat from the lead using heat exchangers, and the electricity is generated by a Brayton cycle. This study focuses on the thermal-neutronic design of the primary loop while secondary coolant and heat exchanger were considered only as boundary conditions. Thermal-neutronic parameters of the reactor core have been simulated using MCNPX 2.6 code and ANSYS software (CFX module). Simulation results confirmed the ability of the innovative 4M system to reach acceptable performances.

Keywords: *VSMR, Thermal-neutronic, simulation, Neutron Source*

Introduction

VSMRs (Very Small Modular Reactors) are a subcategory of SMRs that are defined as reactors with a power capacity of fewer than 10 MWe. In recent years, by considering the advantages of VSMRs (and their ability and flexibility in different areas), various innovative designs of this reactor type are under consideration. Several small modular nuclear reactors have been designed and developed for different purposes such as backup power, emergency power having the potential to fit in small electricity grids or autonomous operations. These reactors have lower capital and maintenance costs in comparison with large capacity plants and therefore, can minimize the financial risk to investors and maximize the profitability for industries in developing countries [1, 2].

Due to the importance of this kind of reactors, various design parameters and structures were considered by designers. The SSTAR is a concept of small lead cooled reactor without on-site refuelling developed by the Argonne National Laboratory (ANL) in the USA. SSTAR concept uses supercritical CO2 as the working media in the Brayton cycle and avoids intermediate heat transport system. It uses uranium with a maximum fissile fraction of 20% [3]. The thermal efficiency of SSTAR is about 40%, and the cycle length of this reactor was considered of about 20 years [4].

University of Science and Technology of China (USTC) developed SNRLFR-100 with the goal of achieving economic factors envisaged for Gen-IV of nuclear reactors. SNRLFR-100 is a fast SMR (Small Modular Reactor) that is cooled by natural circulation of lead in the primary loop and forced water-cooling system in the secondary one. The power of this reactor is of 100 MWth and the enrichment of the MOX fuel ranges from 16% to 24% of PuO_2 [5].

Also, a small lead-bismuth cooled reactor has been designed by Japan Atomic Energy Agency (JAEA). This is a homogenous reactor and its core consists of two regions. It uses nitride fuel (100% enriched 15N and 10.5-18.2% plutonium) and works for 30 years without refuelling. The power of this reactor is of 132 MWth (50 MWe), and primary and secondary cooling system are working based on natural circulation [6].

LFR-10 is under design and development by USTC with a thermal power of 10 MWth and uses different enrichment zone of UO2 as fuel. The core is cooled by natural circulation of LBE in the first loop, and pressurized water works as a coolant in the secondary loop [7].

Another concept, CLEAR-I (China LEAD-based Reactor), has been developed by the Institute of Nuclear Energy Safety Technology (INEST). CLEAR-I was designed for research platform as a fast reactor with LBE coolant and ADS (Accelerator Driven System). The thermal power of this reactor is about 10 MWth that is generated by UO2 as fuel with the enrichment of 19.75% [8].

In this paper, a new innovative VSMR, named ″4M″, was designed and simulated by use of MCNPX 2.6 [9] and ANSYS 18.2 CFX codes for neutronic and thermo-hydraulic calculations, respectively. Firstly, an innovative controllable neutron source was designed and simulated. Different materials for the neutron source were selected and tested (by simulation) with respect to the criteria of higher neutron efficiency and compatibility with the reactor core, the best of them being selected.

In the next step, the reactor core was studied, designed, and simulated by use of MCNPX 2.6 code. In this step, all the required neutronic parameters of reactor core were calculated and the multiplication factor and neutron fluxes in radial and axial directions were reported. Fig. 1 shows the 3D schematic of the 4M reactor and the layout of its components.

In this work, ANSYS CFX is used for thermal-hydraulic analysis and simulation of the reactor core. The generated heat in each mesh of the reactor core was extracted from MCNPX simulation results and used in ANSYS as a heat source. This reactor can produce energy in the amount of 8.5 MWth using metallic uranium pile as fuel with an enrichment of 7% of 235U. The reactor pile is cooled by lead in the primary loop by natural circulation while helium is used in the secondary loop to remove the heat. The electricity will be produced by the Brayton cycle. This reactor will be constructed as tank type being designed for almost 50 years of operation without refuelling.

Main characteristics of some SMRs, VSMRs and of the 4M reactor are presented in Table 1.

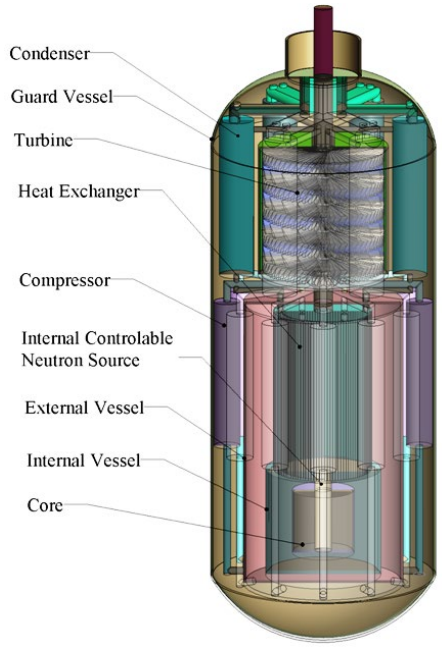


Fig. 1. Structure of designed ICNS and layout of particle sources

Table 1. Specifications of various SMRs, VSMRs and of the 4M reactor

	SSTAR	***SNRLFR-100***	***JAEA***	***LFR-10***	***CLEAR-I***	***4M***
Power (MWth)	45	100	132	10	10	8.5
Thermal efficiency (%)	43	NA	38	NA	NA	41
Fuel	U/PuN	MOX	PuN	UO2	UO2	Metallic U
Enrichment (%)	15-20	16-24	10.5-18.2	NA	19.75	7
Primary coolant	Lead/NC	Lead/NC*	LBE/NC	LBE	LBE/NC	Lead/NC
Secondary coolant	CO_2	Water/Steam FC**	Water/NC	Water/FC	Water/FC	Helium
Reactor type	Pool type	Pool type	Tank type	Tank type	Pool type	Tank Type

* Natural Circulation ** Forced Convection

4M Design Description

Small reactors have the capability to fulfil different requirements of the countries regarding the energy needs. They can produce electricity, heat, and power (combined or separated) for different purposes, with suitable price and sustainability for a long time.

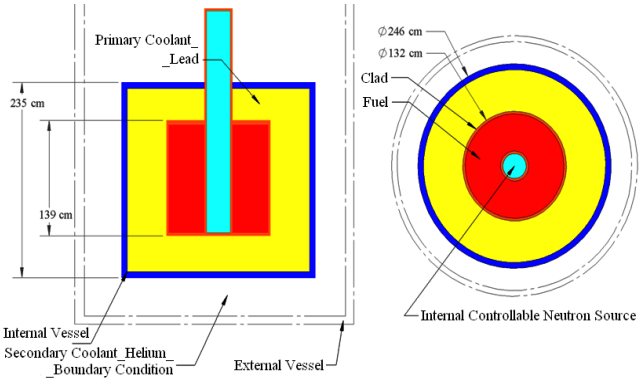


Fig. 2. 4M reactor core layout and sizing

Thus, 4M concept has been designed and simulated using MCNPX2.6 and ANSYS CFX (neutron source, core, and primary coolant system). The main design parameters of 4M reactor are shown in Table 2. Fig. 2 shows the core structure and sizing.

Table 2. Design parameters of 4M reactor

Reactor Type	Tank type
Power (MWth)	8.5
Core Power Density (kW/l)	5
Fuel type	Uranium metal
Enrichment	<7% by ^{235}U
Primary circulation type	Natural circulation
Primary coolant	Lead (Pb)
Secondary circulation type	forced convection
Secondary coolant	Helium
Estimated operation time	50 years
Energy Application	Electricity, District heating, Process steam
Design stage	Conceptual design

The neutron source

The three major goals considered in the design of the 4M reactor are safety, simplicity, and economics. Thus, 4M was designed as a subcritical reactor which criticality can be achieved by the use of Internal Controllable Neutron Source (ICNS). That is why it can propose a simpler design (no need of control rod and safety rod systems), lower construction cost, and a shorter construction period. This feature can improve the safety and controllability of 4M reactor. The first step consisted in the design of an ICNS unit (Fig. 3.), with the capability of producing an appropriate neutron spectrum for this reactor.

The neutron generation in ICNS starts by electron emission from wire source and tungsten filament as a result of thermo-ionic interaction and will continue by Bremsstrahlung interaction of electrons with Silver (e,γ). Photons hit the target and produce neutrons through gamma-neutron interaction (γ, n). ICNS can be subdivided in three particle sources; an electron source, a photon source, and a neutron source. The criticality and burn-up of reactor core will be controlled by the neutrons number that is generated by ICNS.

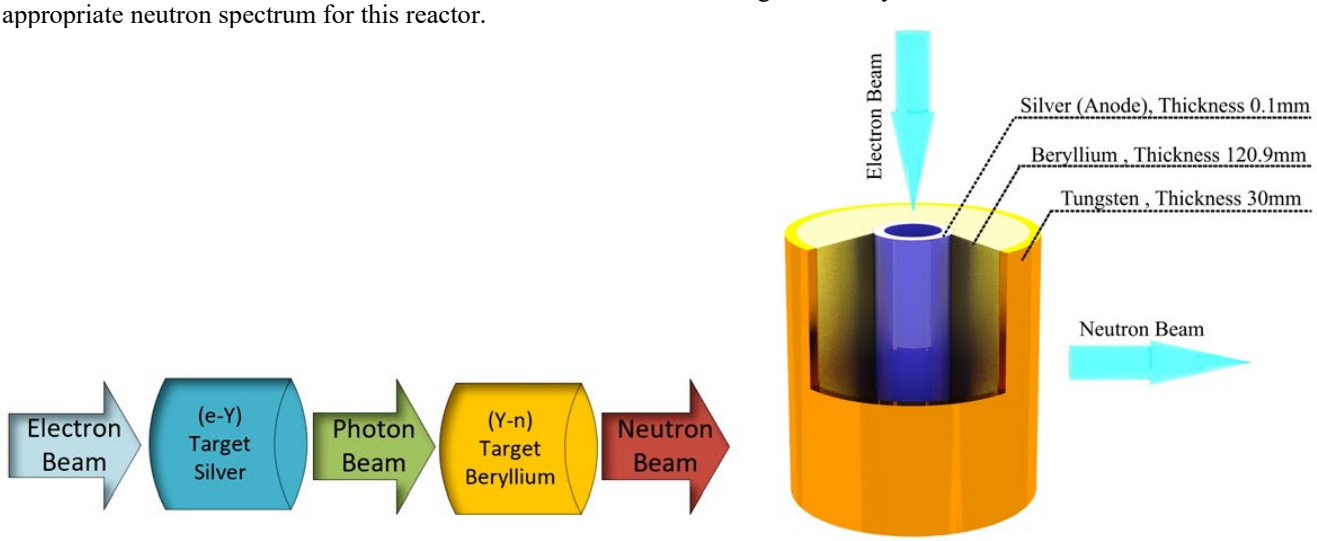


Fig. 3. Structure of designed ICNS (Left) and layout of particle sources (Right)

All of the ICNS interactions are simulated by MCNPX to estimate the neutron spectrum. This spectrum will be used in the next computational step as neutron source for core criticality estimations. According to the Bremsstrahlung phenomenon, emitted electrons are captured to photons. In Bremsstrahlung interaction, photon intensity will be driven by the Kramer rule [10].

$$I(\lambda)d\lambda = \frac{KiZ}{\lambda^2}\left(\frac{\lambda}{\lambda_{max}} - 1\right) \quad (1)$$

where K is the Kramer constant, λ is wave length, i is the tube current and Z is the atomic number of the anode material, $I(\lambda)$ describes the intensity density and $I(\lambda)d\lambda$ is the spectral density.

Reviewing the Kramer rule [11], Lead, Silver, Gold, Uranium, and Tungsten were chosen as candidates for photon source simulation. These five mentioned elements were simulated in MCNPX 2.6 and the number of photons and photons' spectrum were calculated.

Relevant simulations verified that both Tungsten and Silver are suitable as photon source in the first step, but in the end, Silver has been selected as a target for electrons due to the results and analysis of the simulations. Fig. 4 shows the simulation schematic in MCNPX 2.6. In this simulation, the electron point source (shown with '+' in Fig. 4) is considered in the centre of the Silver sphere (violet circle in Fig. 4).

72 detectors with 5-degree angular intervals have surrounded the source for precise angular calculation of the spectrum.

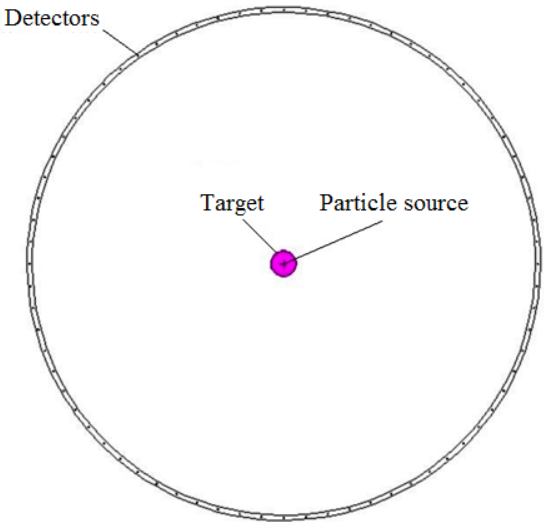


Fig. 4. The ICNS simulation schematic in MCNPX 2.6

The number of photons that are generated by Silver have been calculated and simulated for different electron source energies. The simulation parameters that are used in MCNPX 2.6 are presented in Table 3.

Table 3. Simulation parameters for calculating photons characteristics for different electron source energies

Target radius (cm)	10
Target shape	sphere
Energy of electron (MeV)	15; 10; 5; 1
Number of particles	2.0 x 10^7
Maximum Error (%)	0.11

Fig. 5 shows the result of these simulations as photon's spectrum. As can be seen, by increasing the energy of the electrons (voltage), photon's spectra keep their shape but spectrum peak is shifted to higher energies and probabilities. The growing rate of photon probability versus electron's energy is also shown separately in Fig. 6 for better comparison. This figure proves that the number of photons is changing almost linear versus the energy of the electron source.

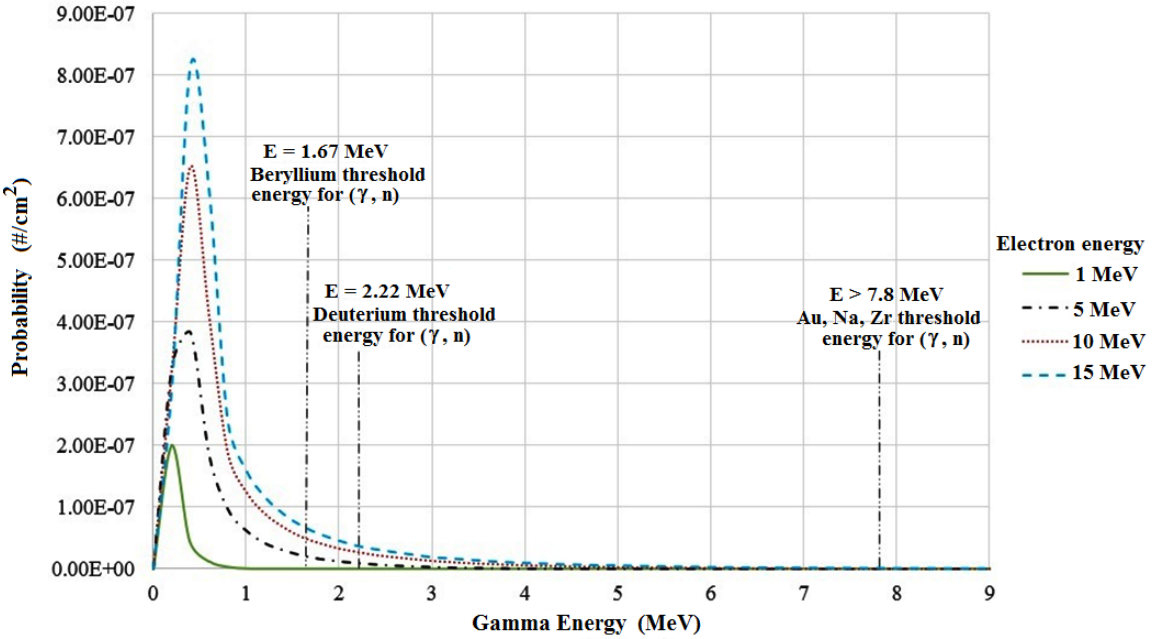


Fig. 5. Photon's spectrum of Silver and values of (γ, n) threshold energies

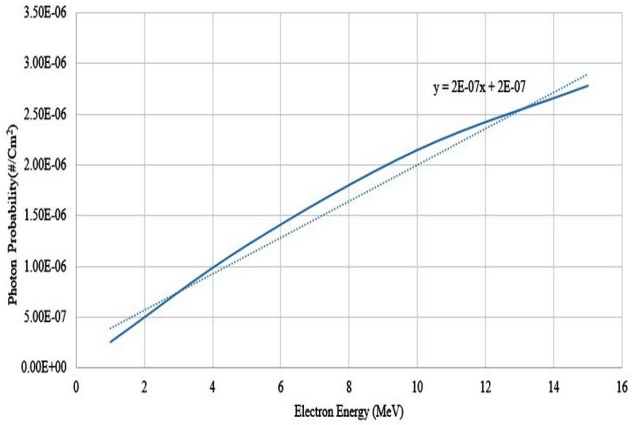


Fig. 6. The photon probability versus electron's energy

In the last section of ICNS design, neutrons are produced by gamma-neutron interaction. Different targets were studied as neutron source. These selected targets are Beryllium, Gold, Sodium, Zirconium and Deuterium. All of the comparisons between these elements as neutron targets have been done based on the photons spectra obtained in the previous step (see Fig. 5).

Beryllium is selected as an appropriate choice for neutron production, the interaction being simulated by means of MCNPX 2.6 in the conditions presented in Table 4 (the same layout shown in Fig. 4 was considered for simulation, Beryllium being the ″target″ whiles the photons are the ″particle source″).

Table 4. Simulation parameters of Beryllium as neutron source in MCNPX

Target radius (cm)	10
Target shape	sphere
Energy of photons	Photon spectrum emitted by Ag
Number of particles	2.0 x 10^7
Maximum Error (%)	0.086

Fig. 7 shows the neutron-energy spectrum resulted from Beryllium interaction with gamma particle. As presented in Fig. 7, most of the neutrons energies are in the range of ″intermediate energies″ [12] with a maximum around 0.392 MeV. This ICNS neutron spectrum will be used in the next section as neutron generator source for core criticality. It is important to note that the core neutron spectrum differs from this spectrum.

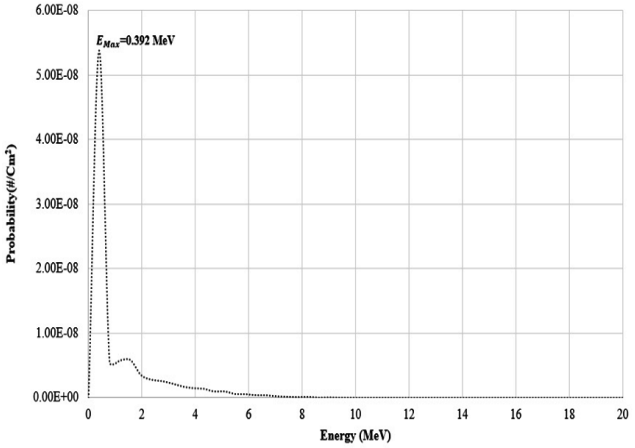


Fig. 7. The neutron-energy spectrum resulted from Beryllium interaction with gamma particles

To validate the capability of the designed ICNS to produce appropriate neutron flux for core criticality, the results of the simulations have been qualitatively compared with the experimental neutron source that was designed and tested at the University of Craiova [13]. Their experimental neutron source uses a linear electron source, while lead, tungsten and uranium have been tested as targets for photon production. Similar to designed ICNS, Beryllium was used as a target for neutron generation (gamma-neutron interaction) in their experiments. Table 5 shows the number of neutrons that are produced by experimental ICNS versus the energy of electrons that are emitted from the central filament [13].

Table 5. Neutron generation due to various electron energies tested at University of Craiova

Number of neutrons (n/cm² sec mA)	***Electron energy (MeV)***
2.4 x 10^5	3 MeV
2.4 x 10^7	7 MeV
1.0 x 10^{12}	15 MeV

Calculation of the core parameters

The core multiplication factor is one of the main parameters in the design and study of nuclear reactors, and in the present work it has been calculated using MCNPX 2.6 code. The neutron spectrum generated by ICNS has been considered in these calculations, while without ICNS it is not possible to achieve core criticality.

Four different types of uranium fuel were studied, namely: (i) natural UO_2, (ii) natural Uranium metal, (iii) 7% enriched UO_2, and (iv) 7% enriched metallic Uranium.

Table 6 shows the parameters that were used in MCNPX to simulate the reactor core. The core layout (schematic of the input file for MCNPX) was shown in Fig. 2, while the multiplication factor values are presented in Table 7. It can be seen that the neutron multiplication factor for metallic Uranium as fuel (considering the ICNS neutron spectrum) is about 1.007 and this value is suitable for a sustainable reaction chain in the reactor.

Table 6. Parameters of MCNPX for simulation of the reactor core

Number of source histories per cycle	20000
Initial guess for keff.	1
Number of cycles to be skipped before beginning tally accumulation	10
Number of calculation cycles	100
Number of nodes in mesh (x, y, z)	40x40x43 = 68800

Table 7. Values of multiplication factor for the considered fuels

	U metal (0.7%)	***U metal (7%)***	***UO2 (0.7%)***	***UO2 (7%)***
Without ICNS	K ≈ 0.43	K ≈ 0.998	K ≈ 0.36	K ≈ 0.90
With ICNS	K ≈ 0.43	K ≈ 1.007 ± 0.008	K ≈ 0.36	K ≈ 0.92

Reactor neutron spectrum can affect some of the main characteristics of the reactor such as cycle length, burn up, fission products, power, etc., thus, representing one of the main parameters during the conceptual design stage. Average neutron-energy spectrum and radial and axial neutron flux were calculated for the reactor core. Fig. 8 shows the average neutron-energy spectrum in the reactor core which is dominated by neutrons with energies around 0.196 MeV.

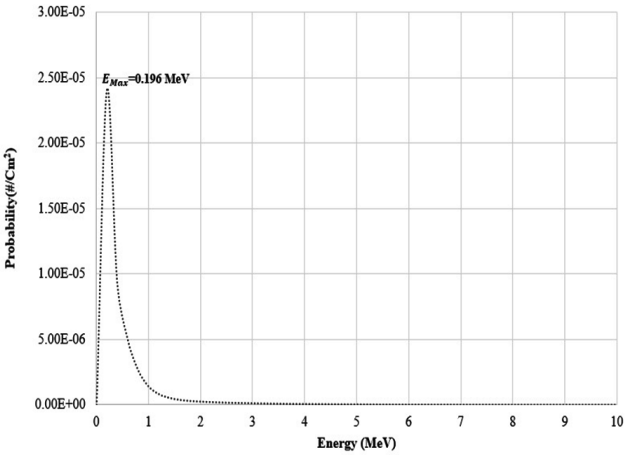


Fig. 8. Average neutron-energy spectrum in 4M reactor core

Neutron fluxes in the reactor core are calculated at different axial and radial positions. For this purpose, the reactor core and its adjacent coolant were nodalized with fine meshes. Figs 9 and 10 illustrate the reactor core radial and axial flux profiles, respectively. Both figures reveal almost symmetric behavior, which is to be expected. As the pile has annular geometry and the ICNS is located in the centre of the pile, two peaks can be seen in Fig. 9, while the flux decreases in the ICNS region. The maximum axial flux (see Fig. 10) is reached almost in the middle plate of the reactor core.

Flux (Probability) contour of the reactor core and primary coolant (lead) are shown in Fig. 11 from (a) side view and (b) top view, and can confirm the results illustrated in Figs. 9 and 10.

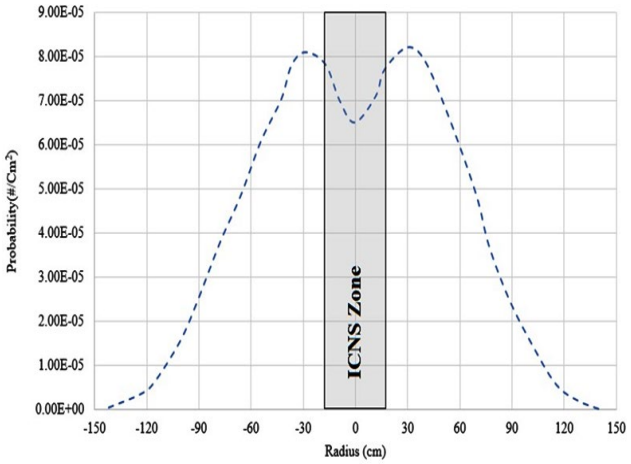


Fig. 9. Radial flux profile of 4M reactor core

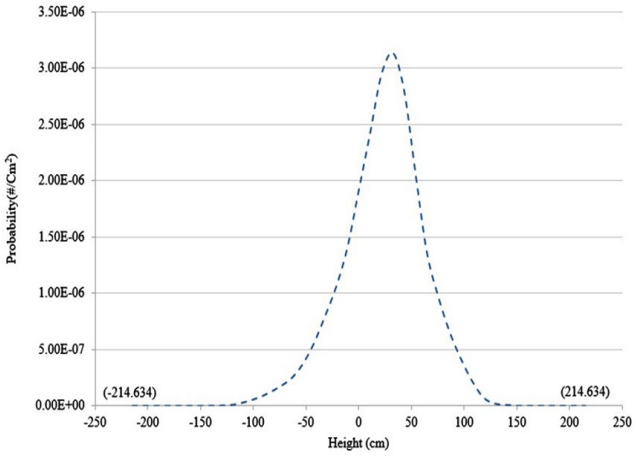


Fig. 10. Axial flux profile of 4M reactor core

As can be seen in Fig. 11, the maximum flux value (dark-red colour) is reached in the pile body (fuel). As ICNS is located in the centre of the pile, the flux in this region is lower than the one in the pile body (red colour). As Lead has been selected as the primary coolant, it can also play the role of shielding for the reactor core (gamma radiation and neutrons).

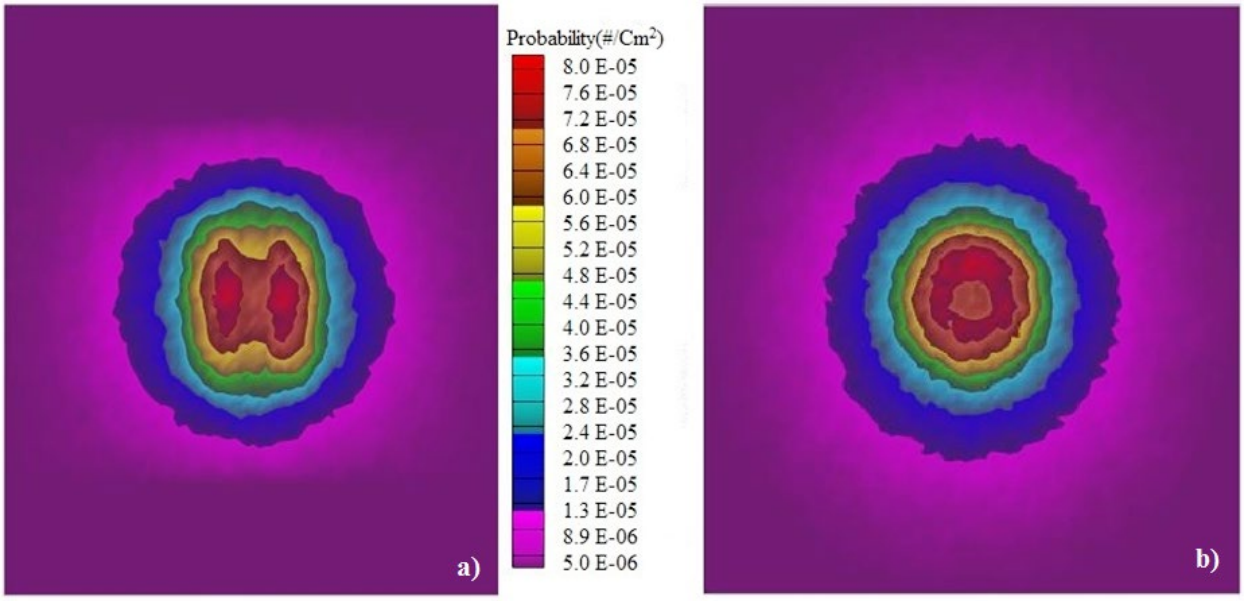


Fig. 11. Flux contour of reactor core: a) side view; b) top view

Core flux changes during reactor operation due to burnup will be compensated by altering the neutron source intensity. Specifying the exact amount of this alteration needs detailed burnup calculation that is out of this article scope but will be presented in future papers.

Thermal-hydraulic simulation of the primary system

The safety improvements of a nuclear system require a comprehensive investigation of the whole power plant from neutronic, thermal-hydraulic and their coupling point of view. It should be considered that the main parameter that limits the power value of a nuclear power plant is the ability of heat removal or in other words the thermal-hydraulic design.

In this section, the core and the primary coolant system of the 4M reactor are simulated using ANSYS 18.2 CFX, and the temperature profiles of the fuel, clad, primary coolant, and vessel are calculated. Secondary coolant parameters are considered as boundary conditions for the primary coolant simulation and therefore no results are presented.

The physical properties of Uranium, Tungsten and Lead used in the simulations are presented in Table 8. As the density gradient is the main force of natural circulation, Lead density is considered as function of temperature in ANSYS by use of User Defined Function (UDF).

Temperature contour of the reactor core is presented in Fig. 12. It can be seen that the maximum temperature of the reactor core is reached in the fuel around the ICNS (close to inner diameter of fuel annulus), its value being around 1200 K, which ensure an appropriate safety margin relative to the melting point of the fuel (around 1400 K). The fuel clad (Tungsten) and the primary coolant (Lead) have their maximum temperature around 1000 K.

Fig. 13 shows the velocity contour in the reactor core from two different views. It can be seen that the heat is transferred from fuel (heat source) to coolant resulting in the reduction of coolant density, which starts moving toward the heat sink to exchange the heat with the Helium (secondary coolant-heat sink).

The whole energy generated in fuel and transferred by internal vessel is removed by Helium in the side and upper part of the internal vessel. The circulation contour of the Lead between fuel pile (heat source) and internal vessel can be seen in the Fig. 13. The maximum velocity of coolant is around 40 cm/s at the pile outer surface, that is justifiable by comparison to other reactors of the same category [7, 15, 16].

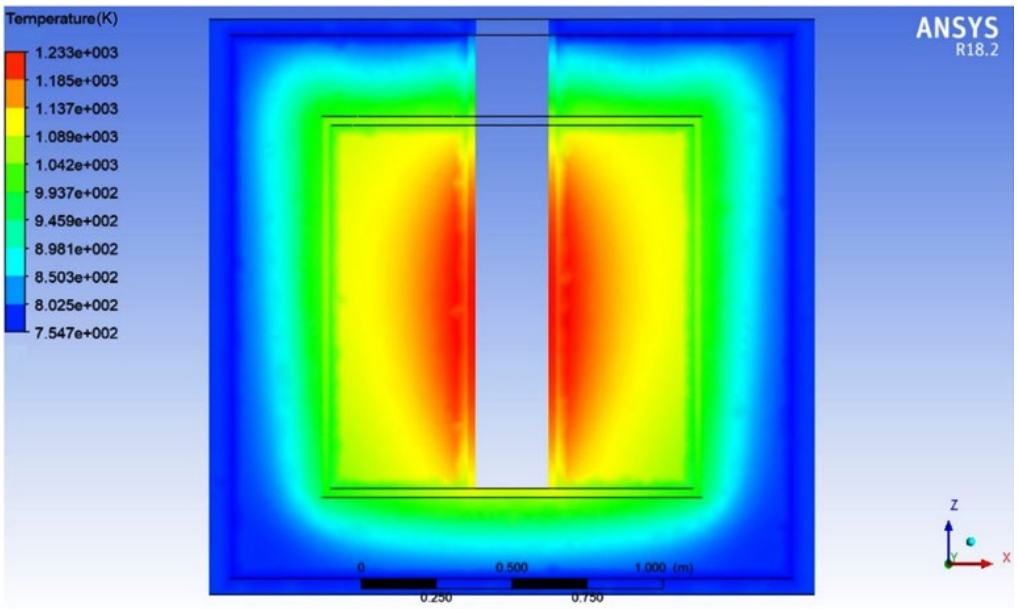


Fig. 12. Temperature contour of 4M reactor core simulated by ANSYS CFX 18.2

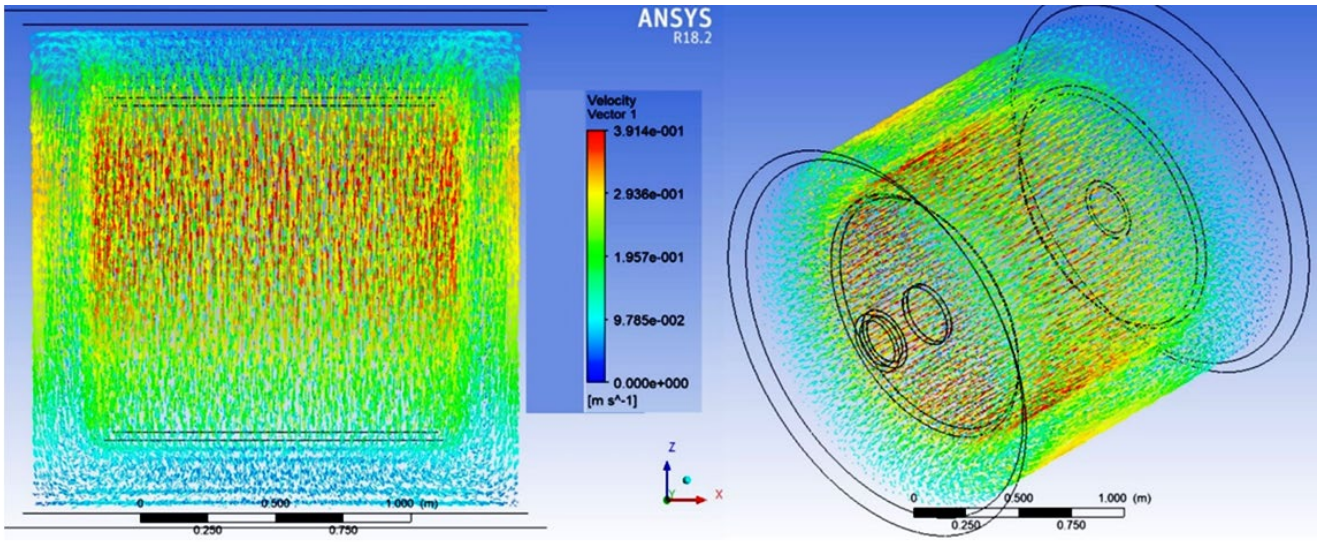


Fig. 13. Velocity contour of 4M reactor core simulated by ANSYS CFX 18.2

Conclusions

Small modular reactors have the ability and flexibility to work in a wide range of situations and locations with higher energy production that are not comparable with the other energy sources.

In present paper, after reviewing the VSMRs that are under study and design around the world, a new VSMR design with innovative characteristics was presented. The reactor was considered as long-life subcritical using enriched 7% U^{235} metallic. The criticality can be achieved by the use of Internal Controllable Neutron Source (ICNS) that represents one of the design features of this reactor making it safer and more controllable. In addition, the use of ICNS can avoid the need of control rods.

Comprehensive simulation of ICNS as well as thermal-neutronic parameters evaluations was carried out in the study using MCNPX 2.6 code and ANSYS 18.2 software.

The neutronic simulation of ICNS confirms the ability of the 4M reactor to generate intermediate neutron spectrum that can increase the number of fissions and the power density of the reactor.

The multiplication factor characterizing the 4M reactor in the presence of ICNS is 1.007, with a 0.7% error. Thermal-hydraulic simulation of the reactor core by using ANSYS software can prove the ability of core cooling by the Lead natural circulation. Moreover, selecting Lead as primary coolant, it can play an extra role as shielding for the reactor core.

The results of the present simulations indicate that 4M reactor has a good capacity for further research and experiments.

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