EURASIAN PHYSICAL TECHNICAL JOURNAL

2025, Volume 22, No.3 (53)

https://doi.org/10.31489/2025N3/66-74

Received: 12/05/2024 Revised: 23/05/2025 Accepted: 25/09/2025 Published online: 30/09/2025
Original Research Article

UDC 539.1; 621.039.573

DETERMINATION OF THE ENERGY RELEASE DISTRIBUTION AND TEMPERATURE IN THE IRT-4M NUCLEAR FUEL WHEN CHANGING THE CONFIGURATION OF THE CONTROL AND PROTECTION SYSTEM CHANNELS IN THE WWR-SM REACTOR

Baytelesov S.A., Tadjibaev D.P., Fayziyev T.B., Kungurov F.R., Alikulov Sh.A.

Institute of Nuclear Physics of Uzbekistan Academy of Sciences, Tashkent, Uzbekistan
*Corresponding author: baytel@inp.uz

Abstract. *The objective of this work is to determine the temperature distribution in the IRT-4M type fuel assembly with fuel 19.75% enriched in ^{235}U in the core of the WWR-SM reactor for the case of a square tube with rounded edges and a round hole in the center and the case of a round tube. In the case of installing a round tube inside the fuel assembly instead of a square tube with rounded edges and a round hole in the center, the volume of water in this space increases. On the one hand, this leads to improved heat removal, since the volume of cooling water increases, and on the other hand, an increase in the volume of water leads to an increase in thermal neutrons on this side of the fuel element, and this, in turn, leads to an increase in energy release. To determine these changes, we performed neutron-physical and thermal-hydraulic calculations for a channel with a square tube with rounded edges and a round hole in the center and for a round tube. It has been determined that replacing a square tube with rounded edges and a round hole in the center with a round tube as a guide for installing a compensating control rod will not affect the nuclear safety of the WWR-SM reactor operation.*

Keywords: fuel assembly, neutron-physical, thermal-hydraulic calculation, horizontal and vertical channels, reactor core, burnup.

1. Introduction

The results of neutron-physics and thermal-hydraulic calculations of a nuclear reactor core are among the key parameters that determine the nuclear safety of its operation. When developing new fuel, as well as when modifying the reactor core configuration or the geometry of structural elements, comprehensive neutron-physics and thermal-hydraulic calculations are a mandatory stage [1–6]. In global practice, continuous efforts are being made to improve the design of reactor cores for both research and power reactors. These efforts are accompanied by detailed calculations and analyses of neutron-physical and thermal-technical characteristics. Such studies are aimed at increasing fuel utilization efficiency, enhancing safety features, and expanding the application capabilities of reactors for both scientific and applied purposes.

In this study, neutron-physics and thermal-hydraulic calculations of the reactor core of the WWR-SM research reactor at the Institute of Nuclear Physics of the Academy of Sciences of the Republic of Uzbekistan (INP AS RUz) are presented. The calculations were carried out for a configuration loaded with 24 fuel assemblies (FAs) of the IRT-4M type containing low-enriched uranium (19.75% ^{235}U).

The reactor core uses six-tube IRT-4M type FAs. These assemblies contain fuel elements (FE) in the form of coaxial square-section tubular plates with a wall thickness of 1.6 mm and a length of 600 mm. The cladding of the fuel elements is made of SAV-1 alloy with a thickness of 0.45 mm (minimum 0.3 mm). Inside the six-tube FAs, either control and protection system (CPS) rods or experimental channels with diameters of 24–26 mm are installed. Using the REBUS computational code, the axial distribution of power density was determined for each fuel element in the most power-intensive FA. For this purpose, the height of the assembly was conditionally divided into 15 sections.

Thermal-physics calculations of research reactors uses a mathematical model of the type [7], and include the determination of temperature fields, heat fluxes, and other parameters related to heat transfer in the reactor core and auxiliary systems. These calculations are essential to ensure safe and efficient operation, as well as to optimize the reactor design and operating conditions [8]. The main objectives of thermal-hydraulic analysis are:

– to determine the distribution of heat fluxes and temperatures within the reactor core;

– to establish the maximum fuel temperature in order to confirm the absence of a melting risk in the most heavily loaded fuel elements; to evaluate the safety margin to the onset of critical heat flux.

All thermal-hydraulic parameters (heat fluxes, temperatures, coolant characteristics) were calculated both for average values over the reactor core and for the most heavily loaded fuel elements. For spatial resolution, calculations were performed at 15 points along the fuel assembly height — from the bottom (0 mm) to the top (600 mm) with a step of 40 mm. The distribution of thermal power in the reactor core was analyzed both for all loaded FAs and for individual elements within each assembly. Knowledge of the thermal loads acting on the FAs and their components is a prerequisite for the safe operation of research reactors.

Thus, the purpose of this study is to determine the distribution of temperature and power generation in IRT-4M type fuel assemblies with uranium enrichment of 19.75% ^{235}U in the WWR-SM reactor core in order to justify thermal-technical reliability and safe operation.

2. Calculation of energy release in the WWR-SM reactor core

2.1 WWR-SM Reactor

The WWR-SM research reactor at INP AS RUz uses water as both moderator and coolant, with a primary circuit flow rate of 1250 m³/h and maximum core temperature of 50°C. The reactor's maximum thermal power is 10 MW. The facility supports various research activities using horizontal and vertical neutron beam channels.

The reactor features:

– 9 horizontal channels for heavy nucleus fission physics, neutron physics, solid-state physics, and materials structure research, including a thermal column [9]

– 32 vertical channels for radioisotope production and other applied research tasks.

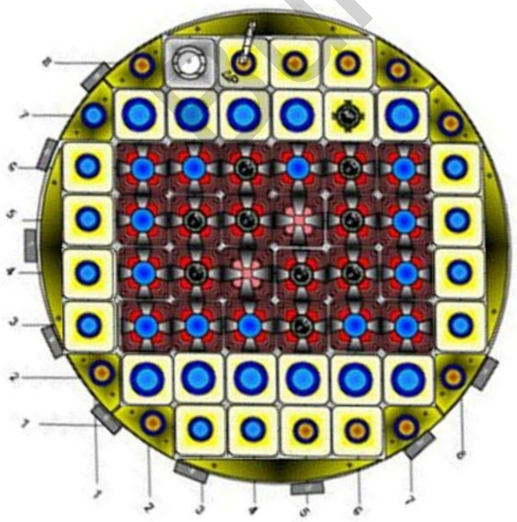


Fig. 1. Horizontal cross-section of the WWR-SM reactor core.

	6-tube fuel assembly (6-tube FA)
	8-tube fuel assembly (8-tube FA)
	6-tube FA with control rod drive mechanism (CRDM) or safety rod mechanism
	Beryllium reflector block with Be plug (Ø 44 mm)
	Horizontal dry channel
	9th dry irradiation channel
	Beryllium reflector block with channel (Ø 60 mm)
	Segmented Be reflector block with channel (Ø 44 mm)
	Beryllium reflector block with channel (Ø 44 mm)
	Automatic regulator mechanism in beryllium reflector block
	Lateral beryllium displacer

2.2 Neutron-Physics Calculations of the WWR-SM Reactor Core Loaded with 24 IRT-4M Fuel Assemblies

The calculation of power distribution and fuel burnup for the FAs was performed using the two-dimensional two-group code IRT-2D [10], developed by researchers at the Kurchatov Institute (Russia). The calculations were carried out under the condition of complete withdrawal of all control rods from the reactor core. Figure 2 shows the power distribution for the core configuration with 24 IRT-4M FAs, as calculated by the IRT-2D code [11].

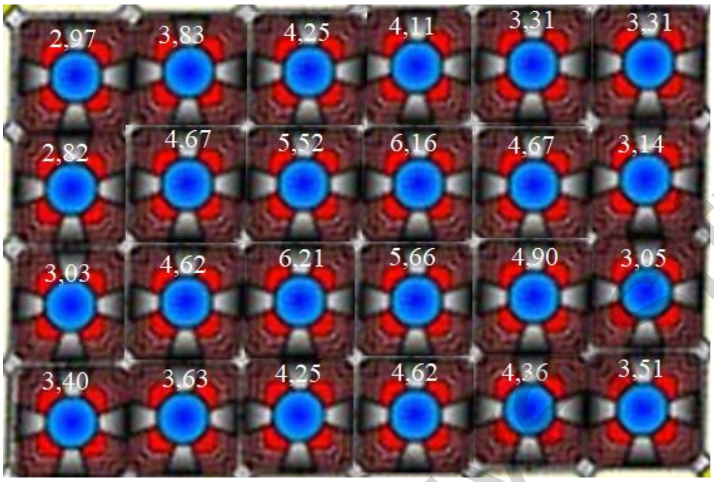


Fig.2. Thermal power distribution (%) for the 24-FA core configuration.

The calculations assumed that all 24 FAs in the core are of the six-tube type, with all control rods fully withdrawn from the core. This is because the control rod drive mechanisms can only be inserted into six-tube FAs. As seen in Figure 2, the most power-intensive FA with 6.212% power generation is located in the central part of the reactor core. Using the REBUS computational code developed at the Argonne National Laboratory in the United States, the axial distribution of power density was determined for each fuel element in the most power-intensive FA. For this purpose, the height of the assembly was conditionally divided into 15 sections. The axial power density distribution was determined for each six-tube fuel rod (FR) [12]. Figure 3 presents the axial power density distribution for each fuel rod in the most power-intensive FA (6.21%).

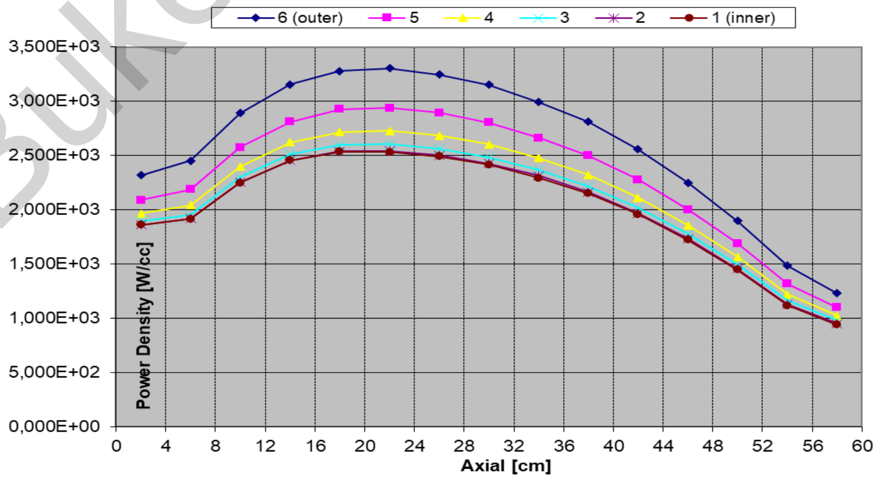
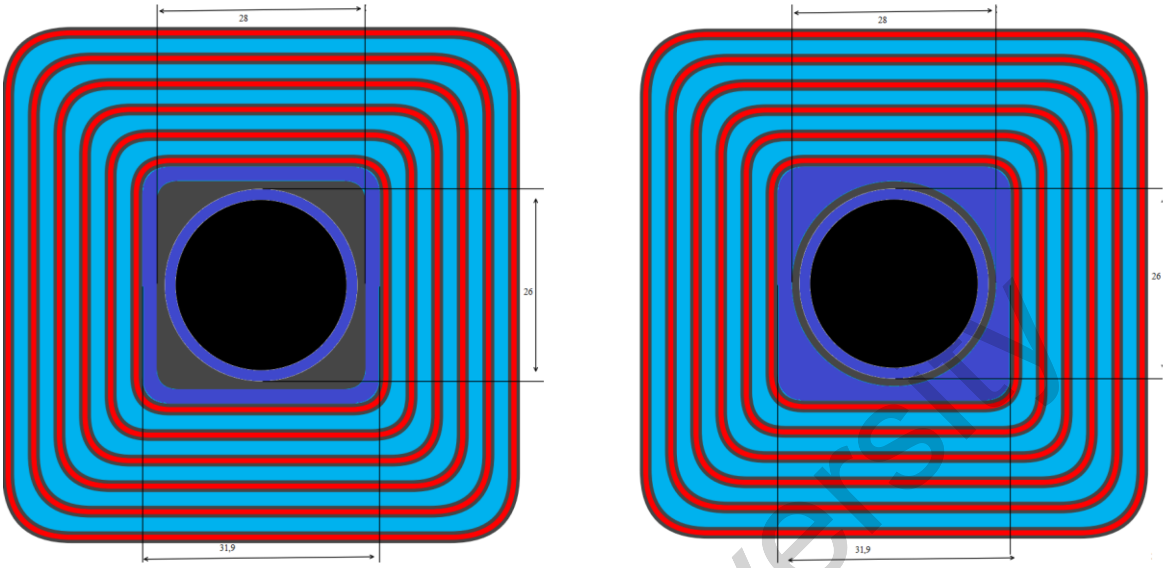


Fig.3. Axial power density distribution along the height of each fuel rod.

As shown in Figure 3, the minimum power density occurs in the inner fuel rod. Replacing the oval-square tube with a circular one alters the power density distribution in the inner tube, which may lead to increased temperatures.

Figure 4 presents:
(a) Horizontal cross-section of the IRT-4M fuel assembly with an oval-square central tube
(b) Cross-section with a circular central tube
Modified flow areas for water passage in the central part of the FA are highlighted in blue.

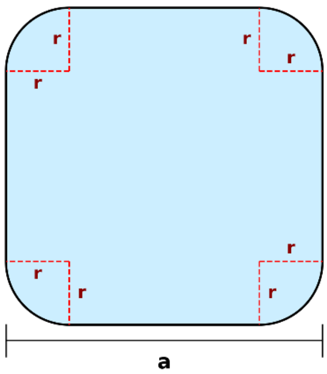


a) b)

Fig.4. Horizontal cross-section of IRT-4M - 6-tube fuel assembly with oval-square central tube for control and safety rods fuel assembly: (a) with oval-square central tube and (b) with circular central tube.

The area calculations were performed using the following formulas:

$$S = S_{square(a)} - 4\,S_{square(r)} + S_{circle}$$
$$S_{square(a)} = a^2$$
$$S_{square(r)} = \; r^2$$
$$S_{circle} = \pi r^2$$
$$S = a^2 - 4\,r^2 + \pi r^2$$
$$S = a^2 - (4-\pi) r^2$$



For the inner sixth fuel rod (FR): Radius: 2.9 mm
Rounded-square cross-sectional area: 1010.38 mm²
For the oval-square tube: Radius: 2.45 mm
Rounded-square cross-sectional area: 778.84 mm²
Central flow channel characteristics: Radius: 14 mm, Area: 615.44 mm²
Area difference compared to oval-square configuration: 163.4 mm²

The replacement of oval-square tubes with circular tubes within the fuel assembly (FA) increases the water volume in this region. This modification provides dual effects:

1. Enhanced heat removal due to greater coolant volume
2. Increased thermal neutron flux density near the fuel rod (FR), leading to:

– Higher power density
– Subsequent temperature elevation

Analysis Methodology:

– Computational study of power density redistribution following oval-square to circular channel conversion
– Axial power distribution analysis with 15 axial segments (4 cm each) along the FR height
– Complete results documented in Table

There are fresh fuel assemblies in the central parts of the core, and burnt-out fuel assemblies are on the periphery (Figure 1). The calculation of energy release and burnup in the fuel assembly was carried out using the two-dimensional and two-group program IRT-2D [10], developed by the National Research Center (NRC) "Kurchatov Institute", Russia. The table shows the results of calculations for reactor operation at 10 MW power. In case of installation of a round tube inside the FA instead of a square tube with rounded edges and a round hole in the center, the volume of water in this space increases. On the one hand, this leads to improvement of heat removal, since the volume of cooling water increases, and on the other hand, the increase in the volume of water leads to an increase in thermal neutrons on this side of the fuel element, and this, in turn, leads to an increase in energy release.

To take these changes into account, we calculated the energy release when replacing the channel with a square tube with rounded edges and a round hole in the center with a round tube. The energy release distribution by the height of the FE was calculated. For this, the FE was divided by height into 15 parts, each part having a length of 4 cm. The results of the calculations are presented in Table 1. As can be seen from Table 1, when replacing a square tube with rounded edges and a round hole in the center with a round tube, the energy release in the FA increases by an average of 5.5%.

Table 1. Energy release distribution by the height of the internal FE (60 cm is the upper part of the fuel element).

Height of FE, cm	Round tube, kW	Round hole, kW	The difference, %
60	2.79	2.60	6.81
56	2.96	2.76	6.76
52	3.61	3.45	4.43
48	4.39	4.19	4.56
44	4.97	4.74	4.63
40	5.59	5.33	4.65
36	6.17	5.87	4.86
32	6.6	6.27	5.00
28	7.03	6.66	5.26
24	7.24	6.83	5.66
20	7.13	6.73	5.61
16	6.73	6.37	5.35
12	6.01	5.65	5.99
8	5.12	4.81	6.05
4	4.69	4.40	6.18
Total	81.03	76.46	5.64

*Are given here: 1- Energy release in the center of FA with round tube, 2- Energy release in the center of FA with square tube with rounded edges and a round hole in the center, 3- The difference in energy output between square and round pipes.

3. Thermohydraulic calculations

For thermal hydraulic calculations, the ASTRA code [13] developed by NRC “Kurchatov Institute”, Russia, was used. Thermophysical parameters for calculations were taken from [13,14].

The thermal hydraulic calculations were performed for two cases: 1 – the case with a square tube with rounded edges and a round hole in the center and 2 – the case with a round tube.

Figure 5 shows a diagram of one side of the outer fuel element of the IRT-4M type fuel assembly. Water passes through the gaps in the fuel assembly from top to bottom.

The following notations are used in the figure:

TW1 - outer channel coolant temperature, °C

T1 - outer coolant/clad interface temperature, °C

T01 - outer clad/fuel meat interface temperature, °C

TM - maximum fuel meat temperature, °C

T02 - inner clad/fuel meat interface temperature, ℃
T2 - inner clad/coolant interface temperature, ℃
TW2 - inner channel coolant temperature, ℃
All dimensions are given in millimeters.

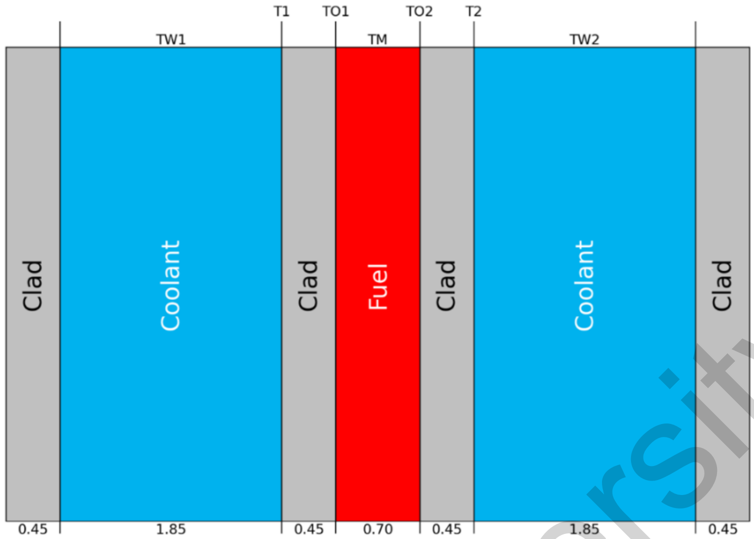


Fig.5. Schematic diagram of the fuel element side.

Table 2 shows the most heat-stressed side of the fuel element in cell 4-5 of the core, where a square tube with rounded edges and a round hole in the center is located in the center of the FA.

Table 2. Results of calculations using the ASTRA code of the most heat-stressed side of the fuel element in cell 4-5 of the cores, in the center of the FA, where there is a square tube with rounded edges and a round hole in the center.

TW1	T1	T01	TM	TO2	T2	TW2	QF1	QF2	SC1	SC2
45.0	54.6	54.8	55.0	54.9	54.9	45.0	91.	85.	7.54	7.28
45.8	57.7	58.1	58.3	58.1	58.1	45.2	116.	112.	5.72	5.55
46.7	62.3	62.8	63.0	62.8	62.8	45.5	154.	153.	4.27	4.15
48.0	66.2	66.7	67.0	66.7	66.7	46.0	183.	187.	3.52	3.43
49.5	69.9	70.5	70.8	70.5	70.5	46.4	210.	221.	3.02	2.95
51.1	72.9	73.6	73.9	73.6	73.6	47.0	228.	247.	2.70	2.65
52.9	75.3	76.1	76.4	76.0	76.0	47.6	237.	266.	2.49	2.45
54.6	75.4	76.1	76.4	76.0	76.0	48.2	221.	261.	2.47	2.45
56.2	73.4	74.0	74.2	73.8	73.8	48.8	183.	233.	2.61	2.61
57.5	70.8	71.2	71.4	71.1	71.1	49.3	139.	201.	2.84	2.86
58.6	70.8	71.2	71.3	71.0	71.0	49.8	128.	197.	2.83	2.87

The following designations are used:
QF1 - outer surface heat flow, kW/m*m
QF2 - inner surface heat flow, kW/m*m
SC1 - ONB margin on temperature for outer surface
SC2 - ONB margin on temperature for inner surface

Table 3 shows the most heat-stressed side of the fuel element in cell 4-5 of the core, where a round tube is located in the center of the FA. Table 4 shows the most heat-stressed parts of all 6 FEs of the FA. As can be seen from the table, when replacing a square tube with rounded edges and a round hole in the center with a round tube, the water temperature increases by about 2%.

Table 3. Results of calculations using the ASTRA code for the most heat-stressed side of the FE in cell 4-5 of the core, in the center of the FA where a round tube is located.

TW1	T1	T01	TM	TO2	T2	TW2	QF1	QF2	SC1	SC2
45.0	56.0	56.4	56.5	56.4	56.4	45.0	106	99	6.57	6.35
45.8	58.5	58.8	59.0	58.9	58.9	45.3	122	119	5.42	5.26
46.9	63.1	63.6	63.8	63.6	63.6	45.6	162	160	4.09	3.98
48.2	67.2	67.7	68.0	67.7	67.7	46.0	191	196	3.37	3.29
49.7	71.1	71.7	72.0	71.7	71.7	46.5	220	233	2.89	2.83
51.4	74.3	75.0	75.4	75.0	75.0	47.1	241	260	2.58	2.53
53.3	77.0	77.8	78.1	77.7	77.7	47.8	251	282	2.37	2.33
55.2	77.0	77.8	78.1	77.7	77.7	48.4	233	277	2.35	2.33
56.8	75.1	75.7	75.9	75.5	75.5	49.0	194	248	2.48	2.48
58.2	72.2	72.6	72.8	72.5	72.5	49.6	148	214	2.70	2.73
59.4	72.3	72.7	72.9	72.6	72.6	50.1	137	210	2.67	2.71

Table 4. Results of calculations using the ASTRA code for the most heat-stressed parts of all 6 FEs of the FA

# of FE tube	Round tube, °C	Square tube with rounded edges and a round hole in the center, °C	Difference in thermal stress, %
1	78.7	77	2.16
2	78.4	76.6	2.30
3	78.5	76.7	2.29
4	78.7	77	2.16
5	79	77.3	2.15
6	77.8	76.1	2.19

According to the performed calculations, the cladding surface temperatures are significantly lower than the values at which cladding damage may occur, and even lower than the temperature at which vapor bubbles begin to form in the coolant (124 °C, see [15]).

For IRT-4M nuclear fuel, the onset of nucleate boiling (ONB) safety factor according to the Bergles–Rohsenow correlation [16] should not be less than 1.3. According to the calculation results, when replacing the square tube with a circular one, this factor exceeds 2 for both the outer and inner surfaces.

In the case of installing a circular tube inside the fuel assembly instead of a square tube with rounded edges and a central circular hole, the volume of water in this region increases. On the one hand, this enhances heat removal due to the larger amount of coolant. On the other hand, the increased water volume leads to a higher fraction of thermal neutrons near the fuel rod surface, which in turn results in an increase in power generation. Calculations show that the power in the fuel elements rises by approximately 2%.

4. Conclusion

The performed neutron-physics and thermal-hydraulic calculations of the WWR-SM research reactor core with a loading of 24 IRT-4M type fuel assemblies containing low-enriched uranium (19.75% ^{235}U) made it possible to obtain the axial power density distribution for each fuel element and to determine the key thermal-technical parameters. It was shown that the maximum values of fuel temperature and heat flux in the most power-intensive fuel elements do not exceed the permissible design limits, thereby confirming the existence of sufficient margins for nuclear safety and thermal reliability. The calculation results confirm the correctness of the chosen reactor core configuration and applied design solutions. The obtained data can be used for optimizing reactor operating regimes, improving fuel utilization efficiency, as well as for further analysis of the feasibility of introducing new fuel types and conducting experimental studies in the reactor core.

Thus, the conducted study provides scientifically substantiated confirmation of the safe and reliable operation of the WWR-SM reactor with IRT-4M type fuel assemblies and forms the basis for further work aimed at improving its fuel cycle and design characteristics.

Conflict of interest statement
The authors declare that they have no conflict of interest in relation to this research, whether financial, personal, authorship or otherwise, that could affect the research and its results presented in this paper.

Credit author statement
Baytelesov S.A.: Writing - Original Draft**; Fayziev T.B:** Development of Facility and Measurements**; Kungurov F.R.:** Review & Editing**; Alikulov Sh.A.:** Conceptualization, Data Curation**; Tadjibaev D.P.:** Methodology, Investigation**.**
The final manuscript was read and approved by all authors.

References

1. Honghao Yu, Jiejin Cai, Sihong He, Xuezhong Li. (2021) Analysis of neutron physics and thermal hydraulics for fuel assembly of small modular reactor loaded with ATFs. *Annals of Nuclear Energy,* 152, 107957. https://doi.org/10. 1016/j.anucene.2020.107957

2. Malmir H., Vosoughi N., Zahedinejad E. (2010) Development of a 2-D 2-group neutron noise simulator for hexagonal geometries. *Annals of Nuclear Energy*, 37(8), 1089 – 1100. https://doi:10.1016/j.anucene.2010.04.007

3. Sidi-Ali K., Medouri E.M., Ailem D., Mazidi S. (2023) Neutronic calculations and thermalhydraulic application using CFD for the nuclear research reactor NUR at steady state mode. *Progress in Nuclear Energy,* 159, 104640. https://doi.org/10.1016/j.pnucene.2023.104640

4. Alzaben Y, Sanchez-Espinoza V.H., Stieglitz R. (2019) Analysis of a control rod ejection accident in a boron-free small modular reactor with coupled neutronics/thermal-hydraulics code. *Annals of Nuclear Energy,* 134, 114 – 124. https://doi.org/10.1016/j.anucene.2019.06.009

5. Erfaninia Ali, Hedayat Afshin, Mirvakili S.M. (2017) Nematollahi M.R. Neutronic-thermal hydraulic coupling analysis of the fuel channel of a new generation of the small modular pressurized water reactor including hexagonal and square fuel assemblies using MCNP and CFX. *Progress in Nuclear Energy*, 98, 213 – 227. https://doi.org/10.1016/ j.pnucene.2017.03.025

6. Xiaobei Xu, Zhouyu Liu, Hongchun Wu, Liangzhi Cao. (2022) Neutronics/thermal-hydraulics/fuel-performance coupling for light water reactors and its application to accident tolerant fuel. *Annals of Nuclear Energy,* 166, 108809. https://doi.org/10.1016/j.anucene.2021.108809

7. Yesbayev A.N., Yessenbayeva G.A.,Ramazanov M.I. (2019) Investigation of the model for the essentially loaded heat equation. *Eurasian Physical Technical Journal*, 16 , 1(31). https://doi.org/10.31489/2019No1/113-120

8. Shaimerdenov A.A., Shamanin I.V., Pribaturin N.A., Gizatulin Sh.Kh., Koltochnik S.N., Chekushina L.V. (2018) Thermophysical Justification for Conducting Lifetime Tests of Experimental Fuel Assemblies WWR-KN in the WWR-K Reactor. *Intern. Scientific Journal for Alternative Energy and Ecology,* https://doi.org/10.15518/isjaee.2018.10-12.023-033

9. Yusupov D.D., Abdullaev R.A. Research Nuclear Facility of Uzbekistan. [in Russian] Available at: file:///D:/Statya_2023_2024_2025/Statya_2025_Eura_Asian/71.pdf

10. Arkhangelsky N.V. (1985) *IRTNOW program for two-dimensional calculation of neutron flux densities in a two-group diffusion approximation*. Description of the application, Moscow, IAE. Available at: https://rusneb.ru/catalog/ 000200_000018_rc_682884/

11.Alikulov Sh.A., Baytelesov S.A., Boltaboev A.F., Osmanov B.S., Salikhbaev U.S. (2014) Experimental Studies of Spent Fuel Burn-up in the WWR-SM Reactor. *Nuclear Engineering and Design,* 277, 163 – 165. https://doi.org/10.1016/j.nucengdes.2014.06.020.

12.Baitelesov S.A., Dosimbaev A.A., Kungurov F.R., Salikhbaev U.S. (2008) Neutron-physical and thermal-hydraulic calculations of VVR-SM fuel assemblies from highly and low-enriched uranium. *Atomic Energy*, 104, 5. Available at: https://j-atomicenergy.ru/index.php/ae/article/view/1719

13.Taliev A.V. (2006) ASTRA modernized program for calculation of the thermal mode of the fuel assemblies of research reactors with tubular coaxial fuel elements. Available at: https://inis.iaea.org/records/m4cyv-40x35/

14.Baytelesov S.A., Kungurov F.R., Yuldashev B.S. (2020) Thermal-Hydraulic Calculations of the WWR-SM Research Reactor. *Nuclear Physics and Atomic Energy*, 21(2), 152–156. https://doi.org/10.15407/jnpae2020.02.152

15.Woodruff W.L., Warinner D.K., Matos J.E. (1992) Consequence Analysis Method. IAEA-TecDoc 643, *Research Reactor Core Conversion Guidebook. 2, Analysis, Appendices D; Radiological Consequence Analysis*. RERTR Program, Argonne National Laboratory, Argonne, Illinois, United States of America. Available at: https://www-pub.iaea.org/MTCD/Publications/PDF/te_643v2_prn.pdf

16.Li Dong Huang (2009) Evaluation Of Onset Of Nucleate Boiling Models. *ECI International Conference on Boiling Heat Transfer*. Available at: https://inis.iaea.org/records/c8grd-f8556

AUTHORS' INFORMATION

Baytelesov, Sapar Akimovich - Doctor (Sci.), Professor, Head of laboratory, Institute of Nuclear Physics, Uzbekistan Academy of Sciences, Ulugbek, Tashkent, Uzbekistan; Scopus Author iD: 9740002000; https://orcid.org/0000-0003-3926-9579; baytel@inp.uz

Kungurov, Fakhrulla Rakhmatullayevich – Doctor (Sci.), Senior researcher, Deputy Director; Institute of Nuclear Physics, Uzbekistan Academy of Sciences, Ulugbek, Tashkent, Uzbekistan; Scopus Author iD: 15021139600; https://orcid.org/0000-0003-4359-3523; fkungurov@inp.uz

Ajikulov, Sherali Abdusalomovich - PhD, Senior Researcher, Institute of Nuclear Physics, Uzbekistan Academy of Sciences, Ulugbek, Tashkent, Uzbekistan; Scopus Author iD: 55815707200; https://orcid.org/0000-0003-4028-6501; alikulov@inp.uz

Tadjibaev, Diyer Pulatovich – PhD, Senior researcher, Institute of Nuclear Physics, Uzbekistan Academy of Sciences, Ulugbek, Tashkent, Uzbekistan Scopus Author iD: 35305693000; https://orcid.org/0009-0001-9782-8079; tadjibaev@inp.uz

Fayziyev Temur Bahriddin o'g'li – PhD student, Institute of Nuclear Physics, Uzbekistan Academy of Sciences, Ulugbek, Tashkent, Uzbekistan; Scopus Author iD: 58427407800; https://orcid.org/0000-0002-8824-5488; temurfayziyev@outlook.com