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Research Article

Calculations of research reactor thermal hydraulics based on VVER-440 fuel assamblies*

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Abstract

Having thoroughly analyzed the design features of VVER-type pressurized water reactors and VVR-type research reactors, the authors propose a design of a research reactor with low-enriched fuel based on deeply updated VVER-440 fuel assemblies. The research reactor is intended to solve a wide range of applied problems in nuclear physics, radiation chemistry, materials science, biology, and medicine. The calculated thermal hydraulics confirms the correctness of the fundamental approaches laid down in the reactor design.

An equivalent reactor core model in the form of a thick-walled cylinder was considered, and the radial power density distribution was obtained. According to the heat power level, five groups of FAs were identified. For each group, the coolant mass flow rate was calculated, which ensures alignment with the outlet coolant temperature.

The coolant flow regime was also estimated. It turned out that for the first row of FAs, the flow regime is in the transition region, while for the other rows the flow regime is laminar. A test by the $Gr{\cdot}Pr \geq 1{\cdot}10^5$ criterion showed its conformity (the calculated value was $1.96{\cdot}10^6$), indicating the transition to a viscous-gravitational regime. The FE surface overheating was calculated relative to the mixed coolant average temperature. The axial coolant flow temperature distribution is the same in all the FAs, the change in power is compensated by the corresponding change in the coolant flow. The maximum coolant overheating on the FE wall relative to the flow core is observed in the central FAs, reaching 31 °C, the boiling margin is about 15 °C.

The estimates showed a significant dynamic pressure margin during natural thermal-convective circulation. By calculation, the values of the FE surface overheating during the reactor normal operation were obtained. An approximately 15-degree surface overheating margin relative to the saturation curve is shown, which guarantees the absence of cavitation wear of the FE claddings. In general, the performed calculations confirmed the correctness of the approaches laid down in the reactor design and made it possible to specify the core thermal hydraulics necessary for further developing the concept.

Keywords

Research reactor, low-enriched fuel, natural circulation, long-term campaign, export potential, VVR, IRT, VVER-440

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1. Introduction

One of the main trends in national economies is the development of nuclear technology. This is due to a wide range of applied problems in nuclear physics, radiation chemistry, materials science, biology, and medicine. For providing initial personnel training and developing nuclear technologies, it seems most appropriate to use a pool-type pressurized water reactor (Bat' et al. 1985; Andrianov et al. 2012; Kuatbekov et al. 2014; Tretyakov et al. 2014). Relatively inexpensive facilities of this type have proven themselves in terms of both performance and safety. However, the existing research reactors (RR) were designed as specialized ones: the general trend was to force their characteristics in certain directions rather than to make them universal (Bat' et al. 1985). The use of highly-enriched fuel in existing RRs does not allow for technologies developed for power reactors, limiting the possibility of delivering RRs to developing countries, since in accordance with the IAEA requirements materials with a fissile isotope content above 20% are not transferable. In addition, the availability of refueling equipment and the high density of neutron fluxes in existing RRs provide the potential for accelerated accumulation of Pu-241 and its subsequent separation from spent fuel.

To promote a specialized multi-purpose technological reactor (Tretyakov et al. 2013; Boyko 2016) on the world market with the possibility of initial personnel training and carrying out both technological (Tarasov and Toporov 2018) and medical research and procedures (Abdullaeva et al. 2009; Novozhilova et al. 2017), an easy maintainable, highly reliable (with enhanced protection against possible personnel errors) and relatively inexpensive (both during construction and maintenance) nuclear reactor is needed, possibly with significantly reduced neutron flux density requirements.

To meet the above requirements, it is necessary at the design stage:

- to reduce the RR construction/operation costs;
- to reduce the RR decommissioning cost;
- to switch to cheap fuel (preferably regularly used in power reactors);
- to use low-enriched fuel (below 20%, as required by the IAEA);
- to ensure the high RR versatility (the possibility of conducting nuclear physics research and solving various technological problems up to the production of radioisotopes for medical purposes);
- to provide for a long-term non-reloading campaign (preferably for the entire RR life cycle);
- to ensure the RR compactness; and
- to make it feasible to locate the RR in any climatic zone, including arctic and arid conditions.

The authors consider the possibility of creating a research reactor with fuel assemblies (FAs) containing low-enriched fuel made in accordance with the technologies used for power reactors. Such a technical solution would ensure that at least the first three of the above basic requirements are met. A specific task is to estimate the research reactor thermal hydraulics based on upgraded VVER-440 fuel assemblies (Budov and Farafonov 1985; Dementiev 1990).

The proposed RR is a pool-type pressurized water reactor. The neutron spectrum is thermal. The coolant circulation occurs due to natural convection. To reduce the dose of ionizing radiation (produced by the decay of the short-lived nitrogen-19 isotope) on the pool surface, chippers are used that increase the coolant uplift time by more than three minutes. The coolant moves through the core upward; in order to prevent the fuel assemblies from levitating, they are equipped with an antilevitation safeguard. In the center of the reactor pool, a hexagonal cross-section basket made of stainless steel sheet is placed, in the lower part of which there is a core consisting of 36 fuel assemblies mounted on the support plate of the reactor basket in hexagonal packaging with a pitch of 146 mm. Of these, 30 are power fuel assemblies and six are assemblies with CPS absorbing rods. The main reactor characteristics are presented in Table 1.

Table 1. Main reactor characteristics.

Parameter	Value
Thermal power, W	$2.0 \cdot 10^6$
Core volume., m^3	0.734
Coolant temperature at the core inlet., °C	60.0
Average coolant heating in the core, °C	20.0
Average specific heat capacity c_p in the temperature range ΔT_r, kJ/(kg K)	4.22
Specific core power density, W/m^3	$2.724 \cdot 10^6$
Extrapolated additive to the core size, m	0.08
Radial power peaking factor	1.4
Axial power peaking factor	1.4
Total power peaking factor	1.96
Reactor tank volume, m^3	200

The fuel assemblies are deeply upgraded VVER-440 power fuel assemblies with lowest-enriched fuel (2.6% of the U-235 isotope), the fuel elements and fuel assembly sheath are made according to the technology adopted for VVER-440 with preserved cross-sectional dimensions and reduced length. Changes affect the heads of the fuel assemblies, the length of the fuel elements as well as the design of the spacer and support grids. The FA design is shown in Fig. 1. A fuel assembly contains a hexagonal tube sheath with upper and lower heads; the lower head is equipped with a circular groove under the collet of the glass of the core support plate and a locking mechanism that rigidly fixes the fuel assembly in the glass. The upper head is equipped with three spring-loaded plungers with a stroke slightly exceeding the gap between adjacent fuel assemblies for spacing relative to them. In the central tube of the fuel assembly there is a locking rod ending in a gripping head at the level of the upper head. In bottom nozzles and radial dimensions, the fuel elements

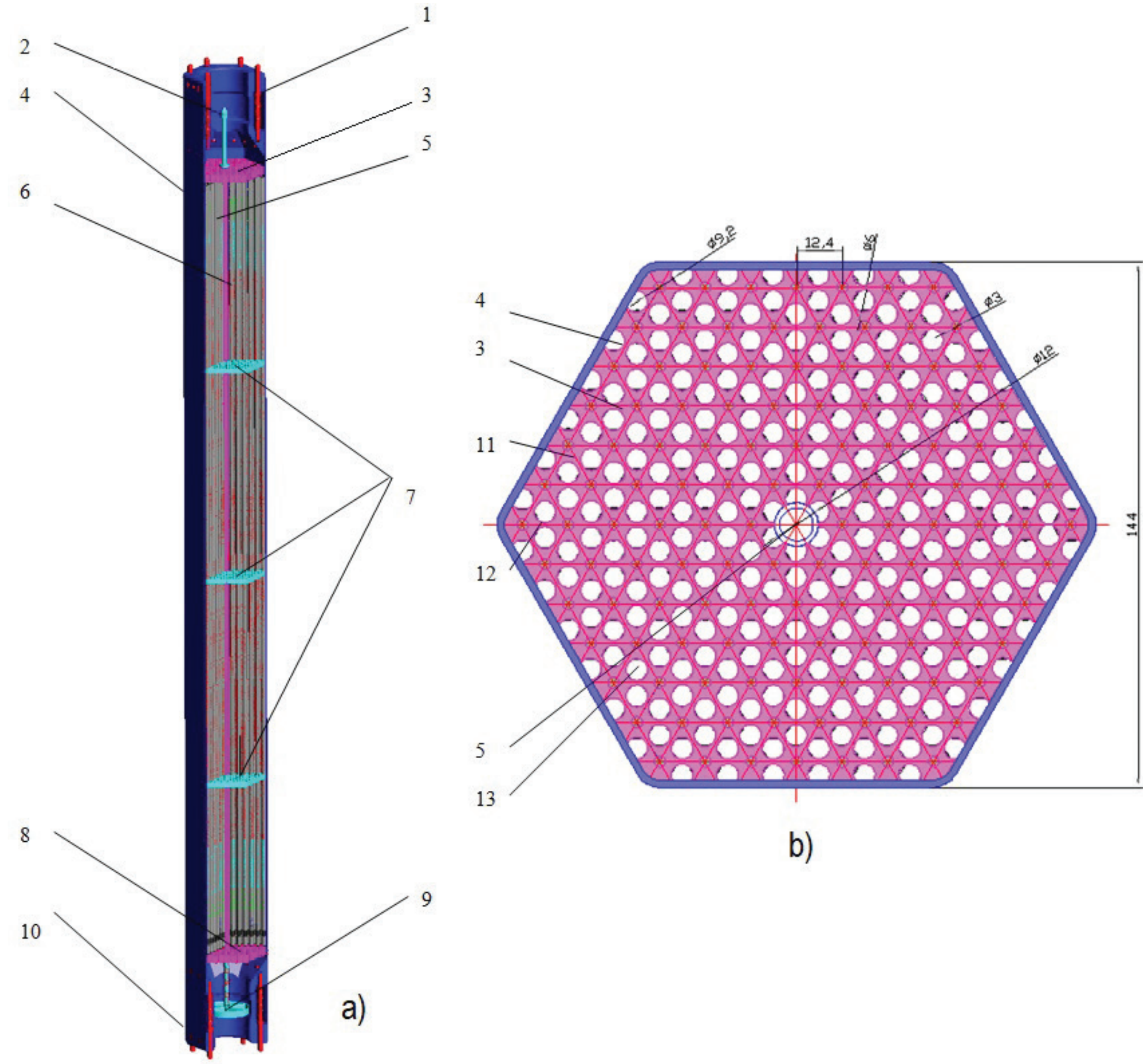

Figure 1. RR FA design: a) Vertical section; b) FA section above the upper sheet; 1. FA upper head; 2. FA collet stem ; 3. Upper tube sheet; 4. FA can; 5. Central pipe with a collet lock rod; 6. Fuel element; 7.Spacer grid; 8. Lower tube sheet; 9. Locking disk; 10 – FA lower head; 11. Coolant pass hole (D = 6 mm); 12. FE bottom nozzle (D = 3 mm); 13. FE cladding (D = 9.1 mm).

correspond to the standard ones for VVER-440. The fuel column length was reduced to 1075 mm, end beryllium reflectors were introduced, and the GFP collection cavities were increased due to the lack of backpressure from the coolant side.

The main characteristics of the fuel assemblies and fuel elements as well as the initial data for calculations are given in Table 2. The numerical values of the parameters correspond to the developed 3D model.

To evaluate the RR thermal characteristics, typical techniques described in (Petukhov 1967; Udaev 1973; Kirillov et al. 1990, 2010; Kutateladze 1990; Kirillov and Bogoslovskaya 2000; Petukhov et al. 2003) were used.

The physical parameters were taken from (Chirkin 1968; Physical Quantities 1991). The calculated RR integral characteristics are presented in Table 3.

The calculated estimates of the heat release in the FAs through the rows and the coolant flow rate through them

Table 2. Main FA/FE characteristics.

Parameter	Value
Core height., m	1.075
FE cylindrical surface length, m	1.60
Distance from the lower bottom nozzle to the fuel in the fuel element, m	0.275
Height of spacer grids, m	0.1
Number of spacer grids, pcs.	3
Number of support grids, pcs.	2
FA inlet length, mm	500
Inlet diffuser length, mm	60
FA inlet diameter, mm	96
FA outlet length, mm	125
Inlet diffuser length, mm	60
Distance of the FA centers from the core center	–
– first row, m	0.146
– second row, near, m	0.253
– second row, far, m	0.292
– third row, near, m	0.386
– third row, far, m	0.438

Table 3. Evaluated reactor integral characteristics.

Parameter	Value
Core equivalent diameter, outer, m	0.9325
Core equivalent diameter, inner, m	0.153
Core effective diameter, outer, m	1.0925
Core effective diameter, inner, m	0.153
Average core power density, W/m^3	2.800
Coolant mass flow rate, kg/s	23.87
Coolant volume flow rate at the core inlet., m^3/s	0.0243
Coolant circulation rate, 1/h	0.439
Total time of the coolant uplift to the pool surface, min	60.5

(provided that the temperature at the core outlet is equalized) are presented in Tab. 4. The axial linear power density distribution of the fuel rods for the groups of FAs is shown in Fig. 2a.

The coolant flow regime was also estimated. It turned out that for the first row of FAs, the flow regime lies in the transition region, while for the others, it is laminar. A test by the $Gr \cdot Pr \geq 1 \cdot 10^5$ criterion showed its conformity (the calculated value was $1.96 \cdot 10^6$), which indicates the transition to a viscous-gravitational regime. Calculations were made of the FE surface overheating relative to the coolant average transitional temperature. The axial coolant flow temperature distribution is the same in all the FAs, the change in power is compensated by the corresponding change in the coolant flow. Fig. 2b shows the temperatures of the coolant and the FE surfaces in the five groups of FAs. The maximum overheating of 31 °C is observed in the central FAs; the boiling margin is about 15 °C.

An analysis of the design features of VVER-type pressurized water reactors and BWR-type research reactors allowed us to propose the following design of a research reactor with low-enriched fuel based on deeply updated VVER-440 fuel assemblies.

An equivalent model of the reactor core in the form of a thick-walled cylinder was considered, and the radial heat density distribution was obtained. According to the level of heat power, five groups of FAs were identified. For each group, the coolant mass flow rate was calculated, which ensures alignment with the outlet coolant temperature.

The estimates showed a significant margin of dynamic pressure during natural thermal-convective circulation. By calculation, the values of the FE surface overheating during reactor normal were obtained. An approximately 15-degree surface overheating margin relative to the saturation curve is shown, which guarantees the absence of cavitation wear of the FE claddings.

The performed calculations confirmed the correctness of the approaches laid down in the reactor design and made it possible to specify the core thermal hydraulics necessary for further developing the concept.

Table 4. Estimated heat release and flow rate in the FAs.

Distribution of FAs in groups	Distance from the core center, m	Number of FAs in a group, pcs.	FA thermal power kW	Coolant mass flow rate, kg/s	Coolant average rate, m/s
First row	0.146	6	87.55	1.045	0.112
Second row, near	0.253	6	71.35	0.851	0.091
Second row, far	0.292	6	63.18	0.754	0.081
Third row, near	0.386	6	40.49	0.483	0.052
Third row, far	0.438	12	27.05	0.323	0.034

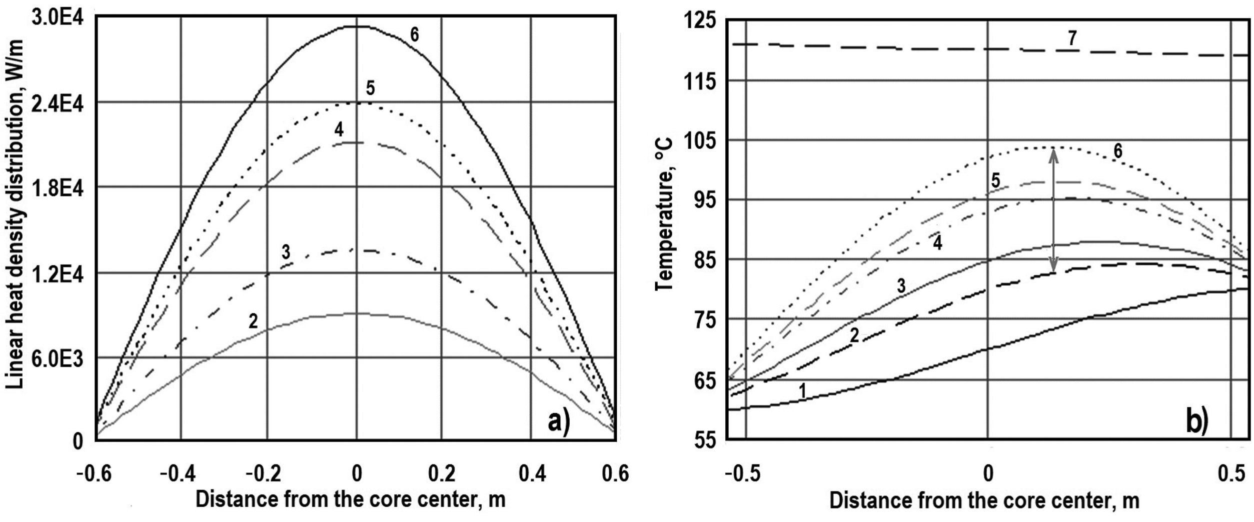

Figure 2. The calculated values of the core thermophysical parameters: a) Linear heat density; b) Coolant/FE surface temperatures; 1. Coolant; 2. FA first row; 3. FA second row, near; 4. FA second row, far; 5. FA third row, near TBC; 6. FA third row, far TBC; 7. Saturation temperature.

References

- Abdullaeva GA, Koblik YuN, Kulabdullaev GA (2009) Use of the VVR-SM Reactor for the Development of the Neutron Capture Therapy Method in Uzbekistan. Proceedings of the Russian Academy of Sciences. Series: Physics 73(4): 540–544. [in Russian]
- Andrianov AA, Voropaev AI, Korovin YuA, Murogov VM (2012) Nuclear Technology: History, State, Future. NIYaU MIFI Publ., Moscow, 180 pp. [in Russian]
- Bat' GA, Kochenov AS, Kabanov LP (1985) Research Nuclear Reactors. Energoatomizdat Publ., Moscow, 280 pp. [in Russian]
- Boyko AA (2016) The Political Aspects of the Development of the Nuclear Energy Market in the Countries of South Asia. NSG Factor in the Promotion of Nuclear Energy in the Region. Bulletin of Volgograd State University. Ser. 4. Istoriya. Regionovedenie. Mezhdunarodnye otnosheniya [History. Regional Studies. International Relationships] 1 (37): 100–109. [in Russian] https://doi.org/10.15688/jvolsu4.2016.1.11
- Budov VM, Farafonov VA (1985) Designing AES' Basic Equipment. Energoatomizdat Publ., Moscow, 264 pp. [in Russian]
- Chirkin VS (1968) Thermophysical Properties of Nuclear Technique Materials. Handbook. Energoatomizdat Publ., Moscow, 484 pp. [in Russian]
- Dementiev BA (1990) Nuclear Power Reactors. Energoatomizdat Publ., Moscow, 352 pp. [in Russian]
- Kirillov PL, Bobkov VP, Zhukov AV, Yuriev YuS (2010) Handbook for Thermohydraulic Calculations in Nuclear Power. Atomenergoizdat Publ., Moscow, 770 pp. [in Russian]
- Kirillov PL, Bogoslovskaya GP (2000) Heat and Mass Transfer in Nuclear Power Plants. Energoatomizdat Publ., Moscow, 456 pp. [in Russian]
- Kirillov PL, Yuriev YuS, Bobkov VP (1990) Handbook for Thermohydraulic Calculations (Nuclear Reactors, Heat Exchangers, Steam Generators). Energoatomizdat Publ., Moscow, 360 pp. [in Russian]
- Kuatbekov RP, Kravtsova OA, Nikel KA (2014) Proposals for the New Generation of University Reactors. Proc. of the 3rd International Conference "Innovative Projects and Technologies of Nuclear Energy". JSC "NIKIET", Moscow, 210–217. [in Russian]
- Kutateladze SS (1990) Heat Transfer and Hydraulical Resistance. Reference book. Energoatomizdat Publ., Moscow, 367 pp. [in Russian]
- Novozhilova OO, Meluzov AG, Ivanova NL (2017) Analysis of the sse of nuclear reactors in medical practice. Proc. of NNSTU 4(119): 108–113. [in Russian]
- Petukhov BS (1967) Heat Transfer and Resistance under Luminary Liquid Fluency in Tubes. Energiya Publ., Moscow, 411 pp. [in Russian]
- Petukhov BS, Genin LG, Kovalev SA, Solovyev SL (2003) Heat Transfer in Nuclear Power Plants. MEI Publ., Moscow, 548 pp. [in Russian]
- Physical Quantities: Reference (1991) Ed. Grigoryeva IS, Meilikhova EZ. Energoatomizdat Publ., Moscow, 1232 pp. [in Russian]
- Tarasov VA, Toporov YuG (2018) The Optimal Distribution of the Irradiation Resource of a Research Reactor in Large-Scale Production of Radionuclides. Col. papers of JSC SSC RIAR 1: 26–33. [in Russian]
- Tretyakov IT, Romanova NV, Sokolov SA, Trushkin VI, Kuatbekov RP, Kravtsova OA, Osipovich SV, Nikel KA, Radayev AI, Goryachikh AV (2014) New Projects for Research Reactors. Proc. of the 3rd International Conference "Innovative Projects and Technologies of Nuclear Energy". JSC "NIKIET", 168–177. [in Russian]
- Tretyakov IT, Sokolov SA, Trushkin VI, Kuatbekov RP, Kravtsova OA, Osipovich SV, Nikel KA, Goryachikh AV, Radaev AI, Kulakov AE (2013) Development of Projects for Perspective Basin Research Reactors. VANT. Ser. Obespechenie Bezopasnosti AES [NPP safety] 33: 103–110. [in Russian]
- Udaev BN (1973) Heat Transfer. Vysshaya Shkola Publ., Moscow, 360 pp. [in Russian]