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Research Article

Neutronic peculiarities of the MTRR-SCW reactor as an experimental base for testing advanced light-water reactor technologies*

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Abstract

The nuclear power system has faced a challenging issue of significantly improving the characteristics of nuclear fuel breeding while maximizing the advantages of the technology of vessel-type pressurized water reactors used extensively in nuclear power. This is possible via switching to supercritical coolant parameters. An increase in production of fissile nuclides, as compared with traditional pressurized water reactors, is achieved by switching to a harder neutron spectrum due to reducing greatly the coolant density and using a dense fuel lattice. A necessary condition for the VVER-SKD design development is the establishment of an experimental base. A multipurpose test research reactor, MTRR-SCW, is the potential testing ground for the reactor technology, and for new structural and fuel materials and fuel rods. The paper presents the key characteristics of the MTRR-SCW reactor, as well as the potential MTRR-SCW applications at different stages of its operation (testing and research). A concept is proposed at the initial stage for the reactor phased rise to power, which will make it possible to justify the efficiency of the MTRR-SCW fuel with increased linear loads through experiments in the independent central loop channel. This concept also involves phased validation and study of the joint operation of the reactor plant and the steam turbine plant as part of the MTRR-SCW nuclear power plant. At the research stage of operation, safe operating limits need to be determined, and the choice of normal operating modes for the VVER-SKD power reactor justified, and experimental studies need to be undertaken to investigate the behavior of structural materials and fuel compositions as part of experimental fuel rods for the advanced light-water reactor cores with different neutron spectra. Long-term irradiation of experimental fuel rods is planned to be carried out in the MTRR-SCW's independent peripheral loop channel, and experimental simulation of emergency processes to be performed in the reactor's central loop channel. This paper deals with the issues to be addressed prior to starting the VVER-SKD power reactor design. Issues have been identified that can be fully or partially solved at effective facilities, as well as the applications for the MTRR-SCW prototype reactor. The paper presents the key characteristics of the MTRR-SCW reactor, and describes in detail the concept for the phased development of the research reactor capabilities and phased rise to power.

Keywords

VVER-SCP, MTRR-SCW, light-water reactor, supercritical coolant parameters, test reactor, research reactor

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Introduction

As part of implementing the 2019–2028 Sectoral R&D Subprogram entitled “Development of Technologies for the Tank-type Power Reactor with Supercritical Coolant Parameters (VVER-SKD)”, the Kurchatov Institute National Research Center, jointly with associate contractors, has developed a concept of the VVER-SKD power reactor with a fast neutron spectrum and light water coolant with supercritical parameters (Alekseev et al. 2023).

By now, as part of developing the VVER-SKD reactor plant concept and the VVER-SKD energy conversion system based on available databases and neutronic, thermal-hydraulic and strength investigations for available models, a range of experimental studies has been identified which are required to justify the reactor safety and stability, and the corrosion and radiation resistance of structural materials in the neutron field (Alekseev et al. 2023).

As a result of analyzing the obtained results, a number of objectives have been formulated and methods have been proposed for achieving these objectives, which should be experimentally confirmed and justified. An analysis of the domestic experimental base offered by research reactors and test benches has shown that only some of the studies required to justify the key provisions used in the VVER-SKD power reactor design can be undertaken at existing plants (Table 1).

To explore the VVER-SKD neutronics and verify neutronic codes, it has been proposed to undertake experimental studies at the BFS-1 critical test facility (Vnukov et al. 2023). In the course of preliminary calculations, the critical assembly configuration has been chosen which corresponds to the rated parameters of the VVER-SKD coolant using the traditional pellet simulation technology. Using a central insert with the heterogeneity properties and behavior similar to the VVER-SKD also allows simulating the key effects of the power reactor reactivity.

The selected thermal and hydraulic characteristics of the VVER-SKD NPP are expected to be studied and justified based on a multipurpose ex-core loop to include natural and forced SKD coolant circulation circuits, as well as two coupled circuits with forced circulation of the SKD coolant (Sedov et al. 2023b). Currently, an SKD coolant circulation circuit has been tested, the test results having confirmed the engineering and process solutions adopted at the development stage. The obtained results are used for planning ampoule and loop tests of cladding structural materials for the VVER-SKD reactor core.

To justify structural materials for the VVER-SKD reactor, a two-stage in-pile test procedure was proposed with experimental fuel elements irradiated initially in the BOR-60 fast reactor until the preset damaging dose level is reached, and transferred then into the irradiation device of the IR-8 thermal research reactor to be further irradiated in the supercritical-pressure water coolant (SKD coolant) environment (Blandinsky et al. 2022). As a result of the activities at the stage of planning a two-stage test of experimental fuel elements, experimental methods were developed, as well as programs for irradiations and pre-irradiation and post-irradiation examinations.

Many objectives cannot be however achieved without building the MTRR-SCW research reactor (Sedov et al. 2023a). The following can be identified among such objectives: conducting in-pile studies to justify the choice

Table 1. Experimental capabilities for achieving the VVER-SKD development and design objectives

Objective	Way to achieve
Achieving the required reactor core neutronic performance	Undertaking experimental studies at the BFS-1 critical facility (IPPE) using a central insert with uranium-plutonium fuel and light-water coolant (Vnukov et al. 2023
Achieving the required reactor core thermal-hydraulic performance	1. Exploring thermal-hydraulic processes in the vertical heated channel, as well as with the heated vertical cylindrical surface flown about as part of the VVER-SKD multipurpose ex-core loop (KI NRC) in conditions with natural and forced SKD coolant circulation (Sedov et al. 2023b).
	2. Exploring thermal-hydraulic processes with the electrically heated vertical dummy FA flown about as part of the CKTI heating test bench in conditions with natural and forced SKD coolant circulation
Irradiation of candidate cladding materials for experimental fuel elements taking into account the reproduction of the ratio between the damaging dose accumulation rate and burnup	Undertaking pre-irradiation tests and in-pile irradiation of gas-filled and flat KKM samples in the BOR-60 reactor (NIIAR) with subsequent additional irradiation in the SM-3 reactor (NIIAR) in the light-water SKD-coolant environment, as well as post-irradiation tests
Development of processes for monitoring SKD coolant modes and processes of the coolant interaction with structural materials	1. Development of the technology based on the VVER-SKD multipurpose ex-core loop (KI NRC)
	2. Development of the system for maintaining the quality of the SKD water coolant as part of the CKTI heating test bench
	3. Development, construction and operation of a reactor loop with the SKD water coolant (RPU-SKD) as part of the MIR.M1 research reactor (NIIAR) for irradiation (additional irradiation) of experimental fuel elements with uranium-plutonium fuel
Justifying the serviceability of the core fuel elements with MOX fuel and other advanced fuel types	Undertaking pre-irradiation tests and in-pile irradiation of experimental fuel elements with uranium and uranium-plutonium oxide fuel (NOU-1, NOU-2, MOX) in the BOR-60 reactor (NIIAR) with subsequent additional irradiation in the MIR.M1 reactor (NIIAR) and in the IR-8 reactor (KI NRC) in the SKD light-water coolant environment; conducting post-irradiation tests for irradiated experimental fuel elements
Justifying the key thermal equipment of the VVER-SKD NPP	Manufacturing and testing scale models of the VVER-SKD NPP’s key thermal equipment as part of the CKTI and EREC heating test benches

of operating conditions for a power reactor with a fast neutron spectrum using supercritical water as coolant, organizing and monitoring the design modes of reactor operation with supercritical coolant parameters, undertaking an integrated computational and experimental studies to obtain the required information for the development and verification of codes, testing new types of equipment for different process systems, instruments and control systems, and monitoring and diagnosing power reactors.

Operation of the MTRR-SCW reactor is expected to be in two stages. Stage 1 will be reactor operation in a test (experimental) mode, which consists in debugging all necessary systems and technologies, checking all nominal and transient modes and normal operation characteristics, determining the stability boundaries, and justifying recommendations for forming the required regulatory framework defined by specific features of the reactor design, the rector facility and the NPP.

Stage 2 is expected to be reactor operation as a research reactor using the capability for irradiating different types of fuel and structural materials in a fast and delayed neutron spectrum, as well as conducting in-pile experiments in independent reactor loops.

The paper considers the MTRR-SCW test operation stage, so a concept is proposed for the phased reactor rise to power.

Key characteristics of the MTRR-SCW

Based on earlier studies by Lapin and Blandinsky 2023, the smallest possible core volume was selected, with which it is possible to simulate the key neutronic and thermal-hydraulic characteristics of the VVER-SKD power reactor. This volume is 500 liters. In the MTRR-SCW reactor core it is possible to install an irradiation device in the central independent loop channel in order to determine the safe operating limits for the VVER-SKD fuel elements and to simulate emergency processes. An irradiation device will be also installed in the peripheral independent loop channel for conducting studies to justify the designs of fuel elements and FAs for different concepts of advanced light-water reactors (Fig. 1).

The key characteristics of the MTRR-SCW presented in Table 2 have been determined for the selected core volume taking into account the safety requirements for research reactors and peculiarities of their operation.

The resonant absorber of gadolinium oxide (Gd_2O_3) in the amount of 0.5% is homogeneously arranged in fuel to bring the reactor to a subcritical state with its core filled with high-density coolant.

To reduce the coefficient of non-uniformity and the linear load for the most heated fuel element, the MTRR-SCW uses three-zone leveling of the power density field with a different content of plutonium in fuel: the core's central part contains fuel with a plutonium content of

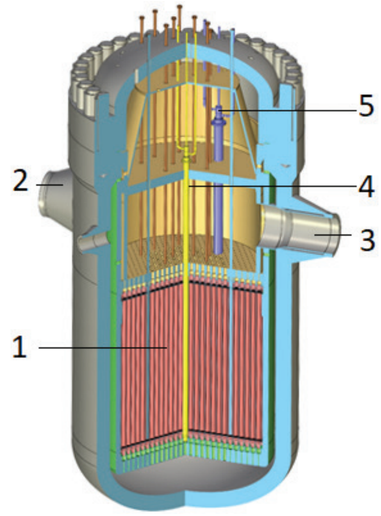

Figure 1. MTRR-SCW reactor: 1 – reactor core; 2 – inlet nozzle; 3 – outlet nozzle; 4 – central loop channel; 5 – peripheral loop channel.

22%, the middle part contains fuel with a content of 27%, and there is fuel with a plutonium content of 35% on the periphery. In this case, it is possible to achieve a non-uniformity coefficient of 1.17 in fresh fuel. As fuel burns up, the core radial power density peaking factor decreases.

A variety of the side reflector designs have been studied to achieve acceptable power density peaking factor values, and to reduce the fast neutron fluence and the reactor vessel damaging dose. The best reflector FA option is that with assemblies made of a fertile material. However, when a fertile material used in the reflector, the neutron flux density decreases due to the power density redistribution. The lateral blanket assemblies of seven steel rods are therefore considered. Since there should be a large number of experimental and irradiation channels in a test reactor, it is reasonable to use a large number of reflector assemblies. Irradiation devices can be installed in place of such channels. In addition, a large number of the reflector assemblies will allow, if required, increasing the core diameter by increasing the number of FAs and reducing the number of the reflector assemblies. Based on this, a core with a diameter of 0.97 m, surrounded by six rows of the reflector assemblies, is considered as the base option for the SKD reactor. Such number of assemblies will make it possible to have a fluence and damaging dose margin for the reactor

Table 2. Key characteristics of the MTRR-SCW research reactor and the VVER-SKD power reactor

Characteristic	VVER-SKD	MTRR-SCW
Power, MWth	1250	Up to 100
FA width across flats, mm	144.6	62.4
Number of fuel elements in FAs	200	36
Design life, years	60	30
Coolant temperature (inlet/outlet), °C	405/520	410/475
Core coolant pressure, MPa	27.5	28
Fuel cycle duration, days	330	90
Number of fuel cycles	2	5
Fuel burnup (average), MW·day/kg	35	25

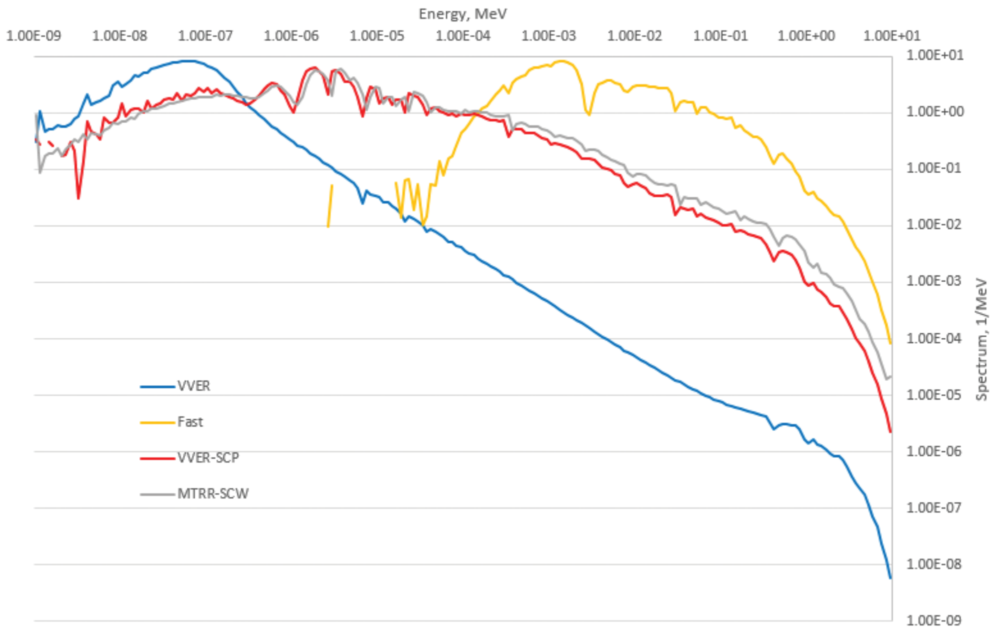

Figure 2. Neutron spectrum.

vessel and internals, change the reactor core configuration by increasing the core dimensions, and install a sufficient number of irradiation devices in the reflector.

The MTRR-SCW and VVER-SKD reactors have a similar axial structure of fuel elements with the same outer diameters and fuel rod pitch, in which MOX fuel is used. Both reactors have close operating parameters (pressure, density, mass velocity, average coolant temperature, cladding and fuel temperature in fuel elements), and a close fast resonance neutron spectrum (Fig. 2).

To ensure safe operation of the reactor and the possibility for monitoring its reactivity in the process of operation, the reactor will include CPS control rods. There will be an independent loop channel in the core center, so it was taken into account that having control rods in the immediate vicinity of the channel will lead to a decrease in the flow density in the loop device, and lead potentially to a major high uncertainty in prediction and subsequent simulation of the loop channel neutronic performance. In addition, ampoule and experimental devices will be installed in the core's central part where the flow density is the highest. For this reason, the neutronic performance needs to stable in the course of the fuel cycle and provide the highest possible rate of the fluence rise and damaging dose accumulation. The scram rods and the rods that make up for temperature and density effects as the reactor reaches the rated power level are withdrawn in full from the core and do not have major effect on neutronic performance. On the other hand, there is a need for ensuring high efficiency of these rod groups, which can be achieved thanks to their accommodation in the central part of the core at quite a distance from the central loop channel. The shim rods for burnup compensation are in the core at the beginning of the fuel cycle and are withdrawn as fuel burns out and reactivity drops, so their effect on the distribution of neutronic characteristics will be substantial. This is the case with automatic regulators as well. They should be therefore installed on the core periphery. A map of the CPS rod arrangement in the MTRR-SCW reactor is presented in Fig. 3.

The proposed configuration of the control rod arrangement meets all requirements for control rods of research reactors (Lapin et al. 2024). The scram rods, without the most efficient one, render the reactor subcritical. The CPS shim CRs make it possible to control temperature and density effects in any reactor state and, along with AC rods, bring the reactor to a subcritical state at a level of $k < 0.99$.

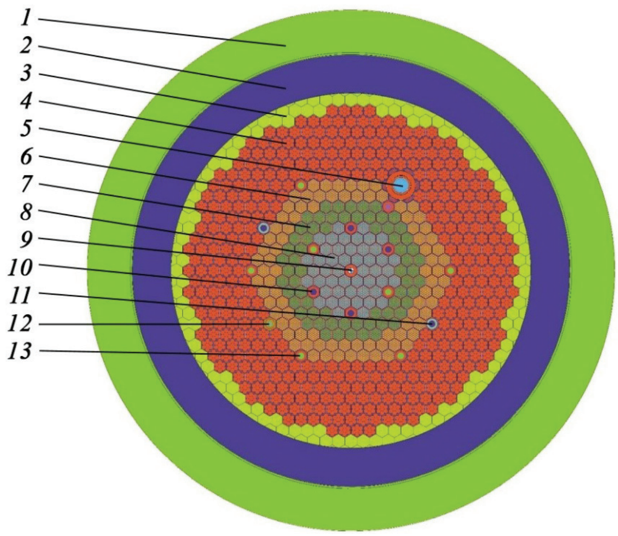

Figure 3. MTRR-SCW core map: 1 – reactor vessel; 2 – coolant in the downcomer section; 3 – reactor pit and core baffle; 4 – changeable reflector cartridges; 5 – independent reactor loop CRs in PLC; 6 – Core FAs with high plutonium content; 7 – Core FAs with average plutonium content; 8 – Core FAs with low plutonium content; 9 – independent reactor loop CRs in CLC; 10 – scram rod channels; 11 – cartridges with automatic control rods; 12 – ampoule irradiation device; 13 – cartridges with shim rods.

Specific to the MTRR-SCW reactor is a highly variable neutron spectrum. In the nominal mode of operation with a low coolant density, there is a fast resonance spectrum, with which there is no major effect from highly absorbing fission products (samarium-149). In the process of bringing the reactor to a long-term outage state in the core, a thermal spectrum occurs, with which the negative contribution of samarium-149 to reactivity increases greatly, this having a favorable effect on the reactor safety. The EoC effect of samarium-149 is 600 pcm.

The reactivity coefficient values in conditions with the rated reactor power are given in Table 3.

Table 3. Reactivity factors at rated reactor power

Reactivity coefficient	Value
Delayed neutron fraction	0.0037
Doppler reactivity coefficient, 1/K	$-1.73 \cdot 10^{-5}$
Density reactivity coefficient, $1/cm^3/g$	$-1.52 \cdot 10^{-2}$
Power reactivity coefficient, 1/MW	$-1.36 \cdot 10^{-4}$

Test stage of operation

Stage 1 shall consist in the MTRR-SCW operation as a test reactor and is expected to support the basis laid down in the reactor design, and to justify the VVER-SKD power reactor designs. An extensive experience exists in operating supercritical water facilities in conventional thermal power. However, unlike the reactor core, the specific heat flux through the heated surface in the thermal power plant SKD boilers is much less important. It is proposed therefore that reactor power will be assimilated at stage 1 on a step-by-step basis to determine permissible heat fluxes. The following concept of phased power assimilation is proposed, which includes

- phased increase in reactor power and core power density until the target parameters of the VVER-SKD power reactor are reached (a maximum power density of 270 to 300 W/cm), accompanied by justification of each further increase in reactor power with obtaining positive results of the MTRR-SCW fuel element testing in the central loop channel with higher loads than in the event of fuel elements in the core driver;
- justifying the serviceability and achieving the design performance of primary and auxiliary equipment of the reactor facility and the nuclear power plant based on the MTRR-SCW reactor equipped with scale models of the reactor facility and VVER-SKD NPP key equipment.

Due to the lack of a reference basis for operation of plants with the power density parameters corresponding to the VVER-SKD power reactor, the key objective of the phased MTRR-SCW rise to power is a step-by-step increase in linear loads in the core fuel elements, which is preceded by justification for the serviceability of experimental fuel elements with the same fuel in the central independent core loop. The MTRR-SCW is thus capable to use only its experimental capabilities to justify the reactor rise to the maximum design power.

At the initial operation stage, MTRR-SCW must be licensed for a capacity of up to 10 MW. Heat fluxes from the fuel element surfaces will be of the same order as in the SKD boilers of fossil fuel fired thermal power plants. The driver zone is a neutron source at this stage. The central loop channel is used to justify the serviceability of fuel elements with such fuel with increased power density parameters in the neutron spectrum that is close to the driver zone and with fuel elements cooled by light-water SKD coolant. The increased power density in the CLC experimental fuel elements channel (compared to the driver zone fuel elements) is achieved through increasing the uranium enrichment with the same plutonium content in order to retain the structural and chemical composition. Thus, the possibility is justified for operating the driver fuel elements at higher linear load values, which will allow obtaining, on a step-by-step basis, the MTRR-SCW operating license at up to the design power of 100 MW and reaching the power density parameters specific to the VVER-SKD power reactor. Noteworthy, the MTRR-SCW reactor operation at different power levels allows varying the coolant flow through the core, as well as varying the coolant heating in the core, with the coolant pressure not being a control parameter and kept at a level of 28 MPa in the process of operation at different power levels (Table 4). Notwithstanding this, the spectral and other neutronic characteristics required for reproducing the performance of the VVER-SKD power reactor are defined only by the core volume and fuel composition and do not depend on the power level at which the MTRR-SCW will operate in each particular experiment.

Table 4. Performance of the MTRR-SCW with a volume of 500 l in the process of operation at different power levels

Parameter	MTRR-SCW operating stages		
Power, MW	100	30	10
Coolant flow through reactor, kg/s	249	74.7–249	24.9–249
Change in core coolant temperature ($\Delta T = T_{outlet} - T_{inlet}$), °C	47.0	47.8–9.4	47.8–2.56
Capture reaction rate to fission rate ratio with Pu-239 (α) in central FA	0.40	0.41–0.43	0.41–0.44
Plutonium content in fuel,%	29	29	29
Maximum fuel temperature, °C 1360	1360	701	550

It was taken into account that the reactor operation at power levels other than nominal levels will lead to a criticality change. This is largely due to a growth in the average core coolant density with a decrease in the power level, as well as a decrease in fuel temperature. The reactor is assumed to be critical when the MTRR-SCW operates at a power of 100 MW, and there is a slightly excessive supercriticality (0.52% and 0.7% respectively) during operation at reduced power levels (30 and 10 MW) at the expense of the reactivity released from the Doppler effect as the fuel temperature decreases. Such excessive supercriticality can be fully made up for by the regular control rods.

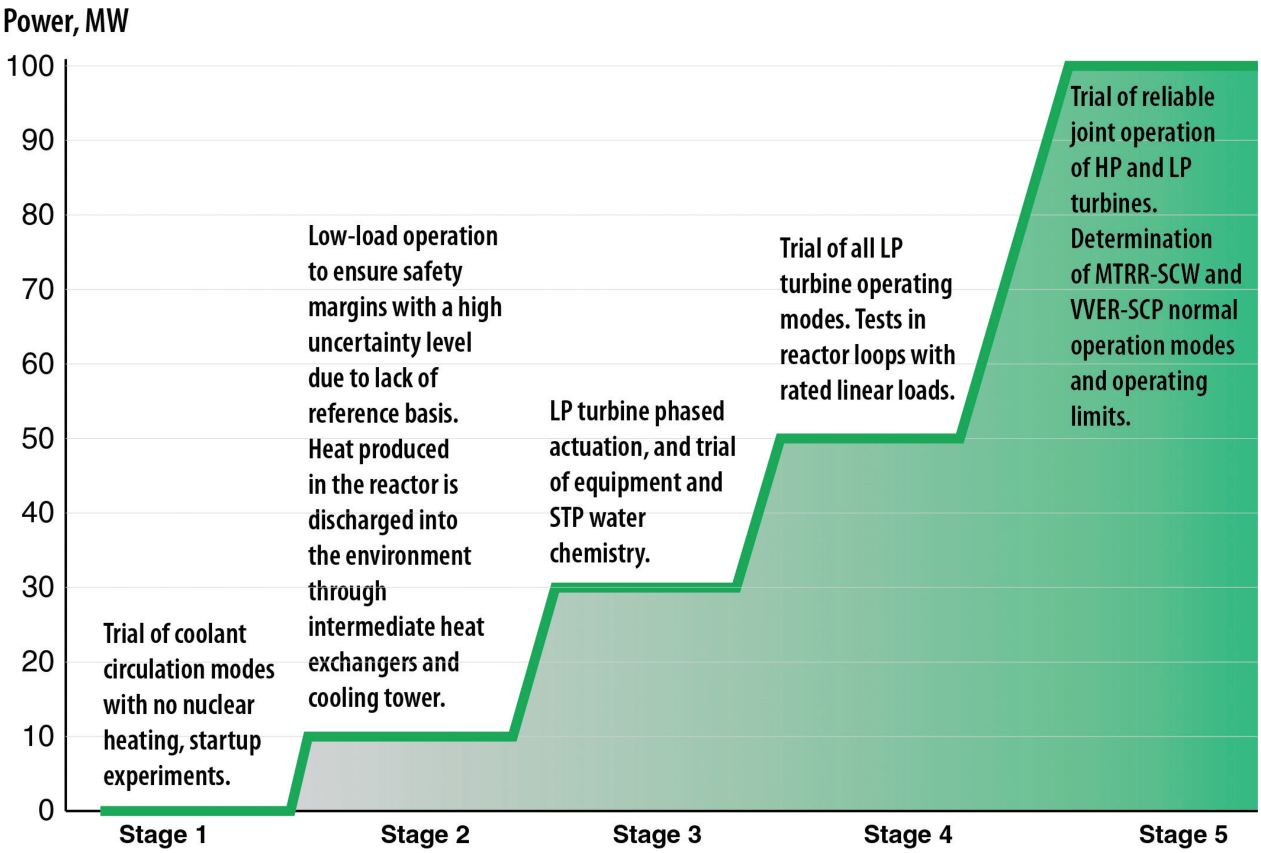

Figure 4. Phased MTRR-SCW rise to power.

The concept of a step-by-step justification for the MTRR-SCW power increase suggests a delayed step-by-step actuation of the steam turbine plant for joint operation with the reactor facility (Fig. 4).

At the initial stage, the heat produced in the reactor will be discharged into the environment through intermediate heat exchangers and the cooling tower (see Fig. 4). After the successful test stage of reactor operation with reduced linear loads and the reactor equipment testing, the next investigation stage will be proceeded to.

When the reactor operates at a power of 30 MW, similar studies are conducted to investigate operation with increased linear loads in the central loop facility, and the initial stage of the investigation program is undertaken for joint operation of the reactor facility and the steam turbine plant as part of the MTRR-SCW NPP. At this stage, as operation is started, generated heat will continue to be discharged into the environment, and the low-pressure turbine will be actuated then and the steam turbine plant (STP) components will be tested and debugged at 30% of the STP rated power (100 MW). From that time, the cooling towers have the functions of removing the STP HP and LP waste heat and as the emergency heat removal system (see Fig. 4).

After that, the reactor rises gradually to 50 and 100 MW. As part of these stages, all LP turbine operating modes will be tried, and subsequently those of the HP turbine and joint operation of the HP and LP STPs will be also tested. After the reactor and NPP operation is tried at the maximum power (100 MW), either only the LP STP or the jointly operating LP and HP STPs can be used in future in the reactor thermal power range of 30 to 100 MW.

When the reactor operates at a power of 10 to 100 MW, experimental studies are undertaken at the initial stage to explore the processes which are specific to the VVER-SKD power reactor. Normal operation limits are specified, and the reactor normal operation modes are determined. Experimental studies are conducted further for technologies of different SKD coolant circulation modes in the neutron field conditions, which will make it possible to solve the problems of not only the VVER-SKD reactor design, but also of other reactor designs with SKD water coolants. Experiments can be conducted in the central loop facility with the phase transition of the coolant as it flows through the reactor core. The peripheral loop channel installed in place of seven FAs will make it possible to organize simulation of the SKD reactor processes with a two-way coolant circulation pattern.

The reliability and safety of the MTRR-SCW operation is achieved at the expense of safety margins with respect to the maximum power level built in advance in the rector design. In the event of a step-by-step change from one reduced design power, that has already been justified, to another (higher) design power, required justifications are developed in accordance with the design documentation, taking into account the experience of operation in the previous reduced power mode. This ensures the following:

- continuity of studies – after experience is accumulated in low-power operation, no new reactor is required to be built and the core fuel is replaced in the same vessel, which ensures that there are no time delays for construction of the reactor, the reactor facility and the NPP, as well as of the building and auxiliary engineering systems for the next power range to continue testing;
- unity of measuring methods and tools, and continuity of measurements, which makes the results of measurements undertaken on one (or same-type) equipment more comparable and representative;
- continuous accumulation of information on radiation damage to the reactor vessel and internals, which is ensured by the fact that witness samples are inserted from the very start of the reactor operation;
- no need for using multiple industrial sites since the entire experimental program is concentrated at one site.

Conclusions

To solve the problems of justifying experimentally the evolution of the VVER-SKD effort, a concept is proposed for a nuclear power plant based on the MTRR-SCW reactor, the operation of which is divided into two stages: a test stage and a research stage.

Preliminary, it was shown with the use of design studies that, regardless of the power level at which the reactor operates, the selected core volume of 500 liters changes slightly the spectral performance, and the resultant excessive reactivity can be fully made up for by the CPS control rods.

A concept is proposed at the initial stage for the phased rise to power from the minimum control power level to 100 MW to ensure that the adopted MTRR-SCW designs are safely justified.

Due to the lack of a reference basis for the MTRR-SCW core fuel element operation with the power density parameters specific to the VVER-SKD power reactor, serviceability of the MTRR-SCW core fuel elements in conditions of a gradual increase in power density should be tested and justified in the central loop channel. Stable joint operation of the reactor facility and the steam turbine plant should be justified at the test stage of the MTRR-SCW study.

The construction of the MTRR-SCW will therefore allow forming a domestic experimental base for a broad range of safety justification experiments and for testing the NPP key components, as well as for confirming experimentally the fundamentals for the concept of advanced light-water reactors.

References

- Alekseev PN, Kovalishin AA, Sedov AA, Andrianova EA, Blandinsky VYu, Kolesov VV, Kotov YaA, Nevinitsa VA, Pustovalov SB, Simonov SS, Subbotin SA, Fomichenko PA, Fonarev BI, Chistov AS, Shmelev AN (2023) Development of VVER technology based on power reactors with a light-water coolant of supercritical parameters. VANT. Ser. Physics of Nuclear Reactors 1: 48–63. [In Russian]
- Blandinsky VYu, Kolesov VV, Nevinitsa VA, Fomichenko PA, Sedov AA, Frolov AA, Pustovalov SB, Shchurovskaya MV, Simonov SS, Pesnya YuE, Trofimchuk VV, Nasonov VA, Zhemkov IYu (2022) Features of changing the isotopic composition of experimental reactor fuel rods of the VVER-SKD type under sequential irradiation in the fast and thermal neutron spectrum. VANT. Ser. Physics of Nuclear Reactors 2: 90–96. [In Russian]
- Lapin AS, Blandinsky VYu (2023) Justification for choosing the power of a multipurpose test research reactor with supercritical parameters of a light-water coolant. Safety of NPPs and Personnel Training: Abstracts. Obninsk, October 26–27, 2023, Obninsk. Optima-Press LLC Publ., 78–79. [In Russian]
- Lapin AS, Blandinsky VYu, Nevinitsa VA, Fomichenko PA, Volkov YuN (2024) MTRR-SCW nuclear safety concept. Global nuclear safety 15(2): 5–19. https://doi.org/10.26583/gns-2024-02-01 [In Russian]
- Sedov AA, Blandinsky VYu, Kotov YaA, Kuzenkova DS, Lapin AS, Nevinitsa VA, Pustovalov SB, Stepanov AS, Subbotin SA, Fomichenko PA, Fonarev BI, Frolov AA (2023a) The concept of a nuclear power unit on the basis of a low-power multipurpose test research reactor with supercritical light-water coolant. Thermal Power Engineering 70(5): 323–338. https://doi.org/10.1134/S0040601523050063
- Sedov AA, Polyakov PS, Pustovalov SB, Parshin VV, Simonov SS (2023b) The main results of the trial tests of a closed circuit with natural circulation of a light-water coolant under supercritical pressure. Thermal Power Engineering 70(3): 223–235. https://doi.org/10.1134/S0040601523030072
- Vnukov RA, Kotov YaA, Nevinitsa VA, Kolesov VV, Blandinsky VYu, Lapin AS, Kuzenkova DS, Fomichenko PA, Mikhailov GM, Korobeynikova LV, Buryevsky IV, Stogov VYu, Shaginyan RA, Eliseev VA, Gulevich AV (2023) Evaluation of the possibility of performing an experiment for VVER-SKD zones at the BFS facility. VANT. Ser. Nuclear Reactor Constants 1: 117–129. [In Russian]