A Concept of VVER-SCP reactor with fast neutron spectrum and self-provision by secondary fuel

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Abstract. In recent few years RF Concern "Rosenergoatom has promoted R&D in support of designing of innovation VVER with supercritical parameters of water coolant (VVER-SCP). Main goals of VVER-SCP have been the followings: possibility of operation of reactor in a regime of self-provision by fuel in the closed cycle; energy efficiency of NPP should be not less than 40-42%. One of VVER-SCP concepts has been a variant of two-circuit NPP with fast reactor, cooled by light-water steam of supercritical pressure – SCPS-600 with electrical power of 600 MWe. Reactor SCPS-600 has a vessel with diameter like VVER-1000, but thicker wall of 350 mm. Combination of quite tight fuel lattice and SCP steam coolant (with the inlet/outlet reactor temperatures of 388oC and 500oC respectively and pressure of 24.5 MPa) allows realizing quite fast neutron spectrum in the core. The core is formed with three groups of Fuel Assemblies with different content of PuO_2: 16, 18.5 and 24% weight respectively. Butt and side blankets comprise dioxide of depleted uranium with content of 235U of 0.2%. Central zone of the core consists of the ThO_2 fertile fuel that provides appropriate void reactivity coefficient. With a load of 32.3 m.t. of heavy atoms the averaged burnup in reactor amounts 54.2 MW-days/kg_{ha}. The ratio of unloaded-to-loaded fissile atoms in reactor amounts 1.01 – 1.03 that make it possible to get the regime of self-provision of reactor by its own secondary fuel in the equilibrium closed fuel cycle. In the secondary circuit of the installation it is planned to use a quite compact supercritical steam turbine with intermediate steam separation without overheating. The steam going from Steam Generator to the turbine has pressure of 23.5 MPa and temperature of 480^{o}C. Net efficiency of the turbine installation amounts 42.5%. The paper considers effect of the design and technology solutions upon the main neutron-physics and thermal characteristics of the reactor. Special care is taken to ways of lowering a positive void reactivity effect by decreasing of parasitic neutron absorption in the core due to thinner cladding, use of new structure materials, like ferritic-martensitic steels and SiC, use of different variants of the spatial distribution of ThO_2 and solid moderators (like ZrH_2 and BeO) in the core.

Key Words: Supercritical pressure coolant, fast reactor, reactivity coefficients, fuel cycle.

1. Introduction

During a few years a cooperation of leading nuclear RF organizations, NRC “Kurchatov Institute”, OKB “GIDROPRESS”, SSC RF-IPPE and JSC “Atomenergoproekt”, supported by Rosenergoatom Concern OJSC, have been conducting R&D for elaboration of innovative VVER with supercritical pressure water coolant – VVER-SCP [1, 2]. These studies and developments have been directed for solution of three main problems: 1) getting competitive specific costs for NPP construction and operation; 2) providing appropriate expenditures for fabrication, handling, transport, storing and reprocessing of used nuclear fuel; 3) enhancing of Nuclear Installation safety up to the level, which allows avoiding necessity of evacuation of population even in the sequence of beyond design severe accident;

The main Generation IV system requirements have been demanded to the reactor concepts of VVER-SCP: sustainment development, economical efficiency, safety and reliability, non-proliferation and physical protection. Additional system requirements have been formulated as well:

- Adaptability to requirements of the Nuclear Power System with the closed nuclear fuel cycle (utilization of U-Pu and Th fuel of different isotopic content), maximum decrease of nature uranium consumption;

- Maximum possible utilization of mature technologies of light-water reactors with account of tendencies of their advancement;
- Multi-functionality: enlargement of power series, enhancing of market adaptability, extended power manoeuvring, possibility of multi- coproduction (district heat, water demineralization, chemistry, metallurgy and so on).
- Industrial construction and fabrication with account of leading world experience in reactor and other technologies.

The main challenge has been called for VVER-SCP: a creation of commercial power block with high power-conversion efficiency, which can be operated in closed fuel cycle with self-supplying by its own secondary fuel. The following goal indicators has been formulated:

- Effective utilization of fissile and fertile materials: getting the breeding ratio (BR)[1] up to ~ 1.0 – 1.05;
- Low specific load of fissile materials into the core (3 ÷ 5 mt /GWe);
- Shortened time of on-site fuel retention in spent fuel pool to 3 years;
- Increase of NPP power-conversion net efficiency up to 40 - 42 %;
- NPP construction time < 48 months;
- NPP payback time < 10 years;
- Reactor life ≥ 40 years.

In the result of conducted in R&D works the technical proposals have been developed to the designs of three variants of VVER-SCP. One of the developed concepts has referred to a fast reactor, cooled by light water SuperCritical-Pressure Steam: SCPS-600.

2. General information about SCPS-600 reactor concept

The NPP with SCPS-600 reactor has a two-circuit nuclear power installation with indirect power conversion system. The core of this reactor has a tight fuel lattice and pseudo-vapour coolant of 24.5 MPa entering with temperature of 390 ^{o}C and leaving with averaged temperature of 500 ^{o}C. Reactor is characterized by fast neutron spectrum and moderate volumetric power load. The secondary circuit has a turbine installation with supercritical steam at the entrance of 23.5 MPa and 480 ^{o}C. In the first circuit of the reactor installation the primary coolant is pumped through the reactor by the main circulation pump and then goes to the once-through steam generator, where the primary coolant transfers its heat to the secondary circuit. The primary circuit has an additional volume, which operates as pressurizer in start regimes under subcritical pressure condition and plays a part of a buffer under condition of supercritical pressure.

Reactor Installation of SCPS-600 has the following safety systems:

- Decay heat removal system;
- Emergency high pressure water supply to the core;
- Emergency passive core cooling system;
- Emergency heat removal from steam generator system;
- Emergency feed water supply system.

[1] Here and below we shall imply that Breeding Ratio is a ratio of the core unloaded-to-loaded fissile heavy atoms

Layout of the SCPS-600 reactor installation is shown in Fig. 1, and balance-of-plant scheme is presented in Fig. 2. Main characteristics of the reactor installation are given in Table 1.

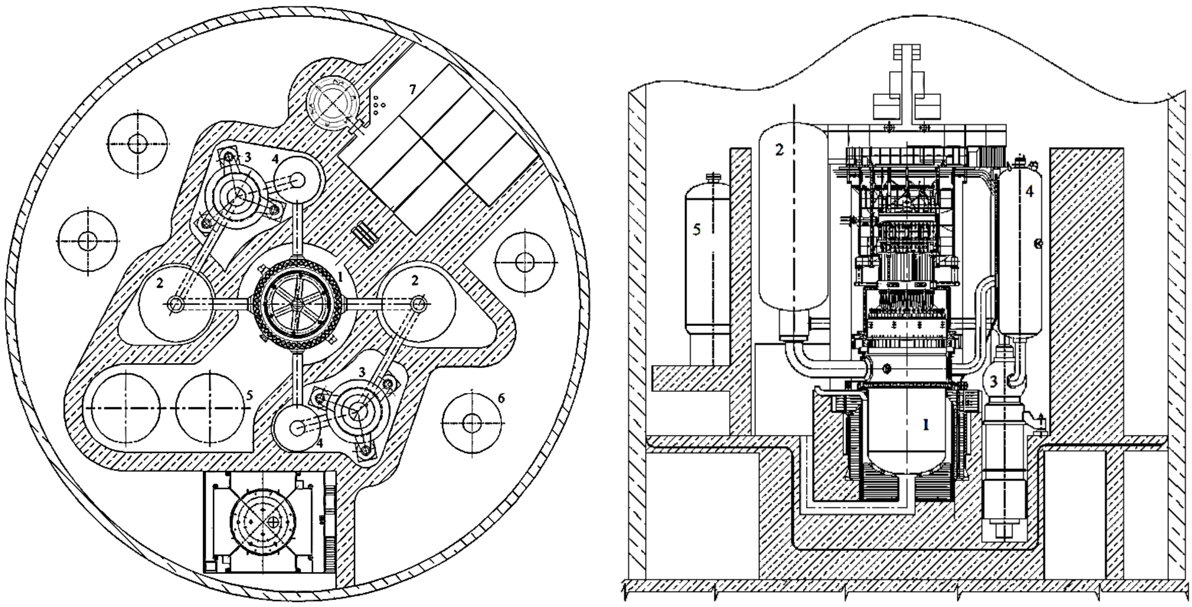


FIG. 1. Layout of SCPS-600 Reactor Installation: 1 – Reactor; 2- Steam Generator; 3 – Main Circulation Pump; 4 – Pressurizer – buffer; 5 – High-pressure emergency water accumulators; 6 – Low-pressure emergency water accumulators; 7 – Spent fuel pool

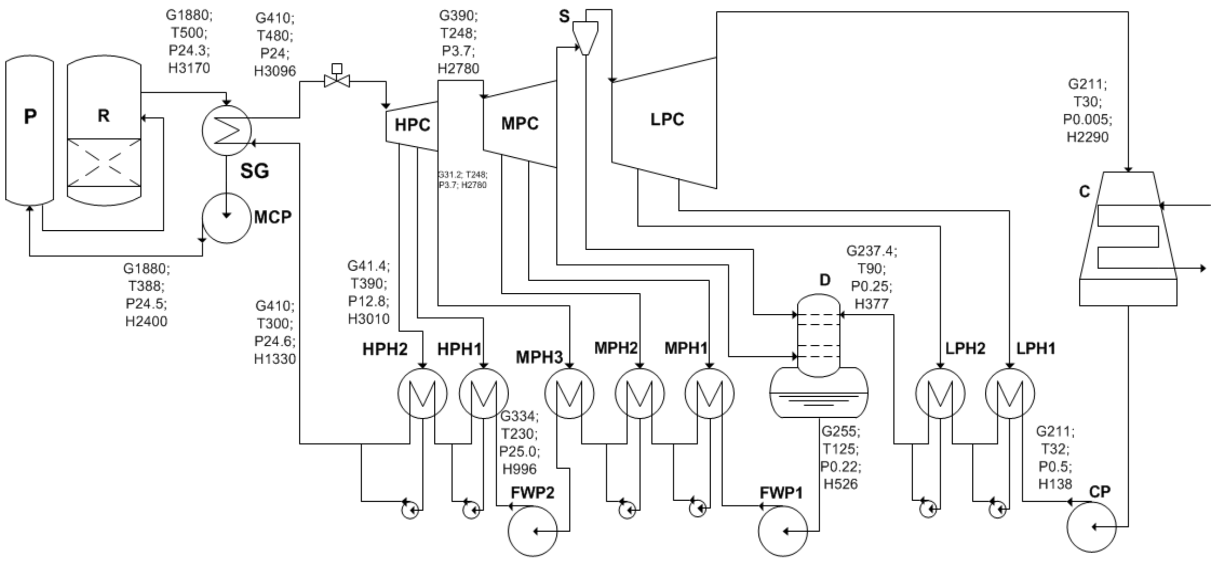


FIG. 2: Balance-of-plant scheme of SCPS-600 Nuclear Power Installation: R – reactor; P – pressurizer - buffer, SG – steam generator; MCP – main circulation pump; C – condenser; LPH – low-pressure heaters; D – deaerator; MPH – mid-pressure heaters; HPH – high-pressure heaters; CP – condensate pump; FWP – feed-water pumps; HPC, MPC and LPC – high-pressure, mid-pressure and low-pressure turbine cylinders; S – separator.

TABLE I: MAIN CHARACTERISTICS OF SCPS-600 REACTOR

Name of characteristic	**Value**
Type of reactor	Vessel-type
Thermal / electric reactor power, *MW*	1430 / 600
Net NPP efficiency	40.2%
Primary coolant	supercritical pseudo vapour
Primary equipment layout	Block-loop
Number of the primary loops	2
Type of the primary coolant circulation	Forced
Neutron spectrum in the core	Fast
Nominal primary coolant flow rate, *kg/s*	1880
Nominal / design-limited primary coolant pressure, *MPa*	24.8 / 27
Coolant temperature at the reactor inlet / outlet, *°C*	390 / 500
Overall reactor sizes (outer diameter / height / thickness), *mm*	4535 / 7500 / 320
Reactor life time, *years*	60

3. SCPS-600 reactor core

In the basic variant of SCPS-600 design the core is formed with fuel assemblies (FA) differed by content of PuO_2 in their fuel (16, 18.5 and 24 % weight), residing time and the kind of reactivity control (see Fig. 3). Butt and side blankets comprise dioxide of depleted uranium with content of ^{235}U of 0.2%. The main core design characteristics are presented in Table 2.

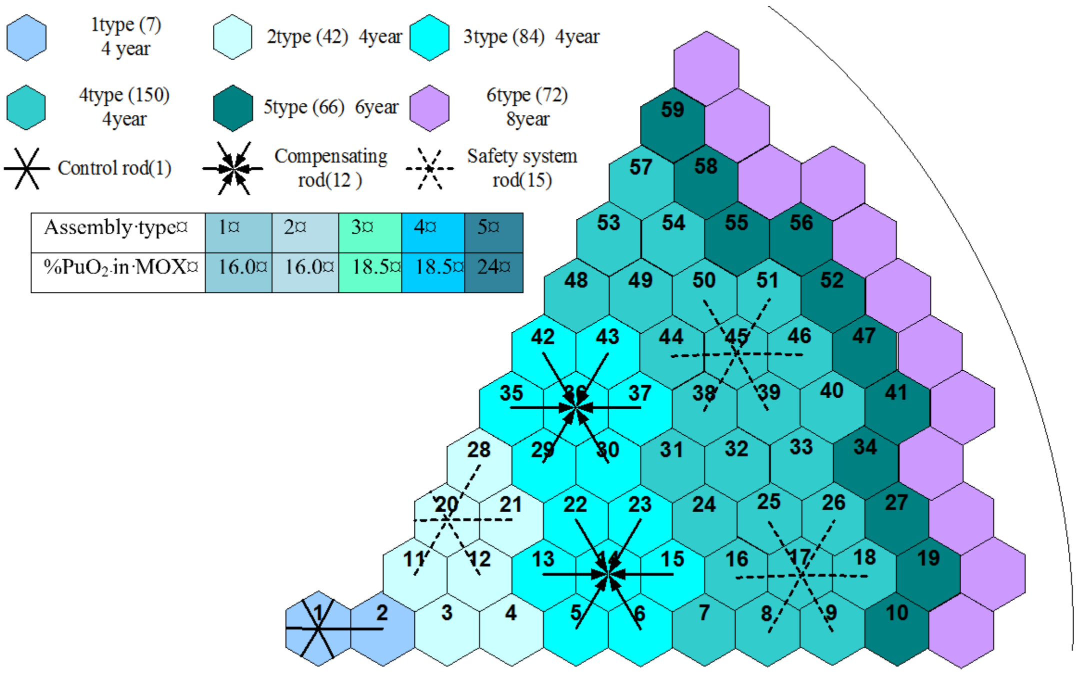


FIG. 3. SCPS-600 core loading lay-out.

TABLE II: CORE DESIGN CHARACTERISTICS

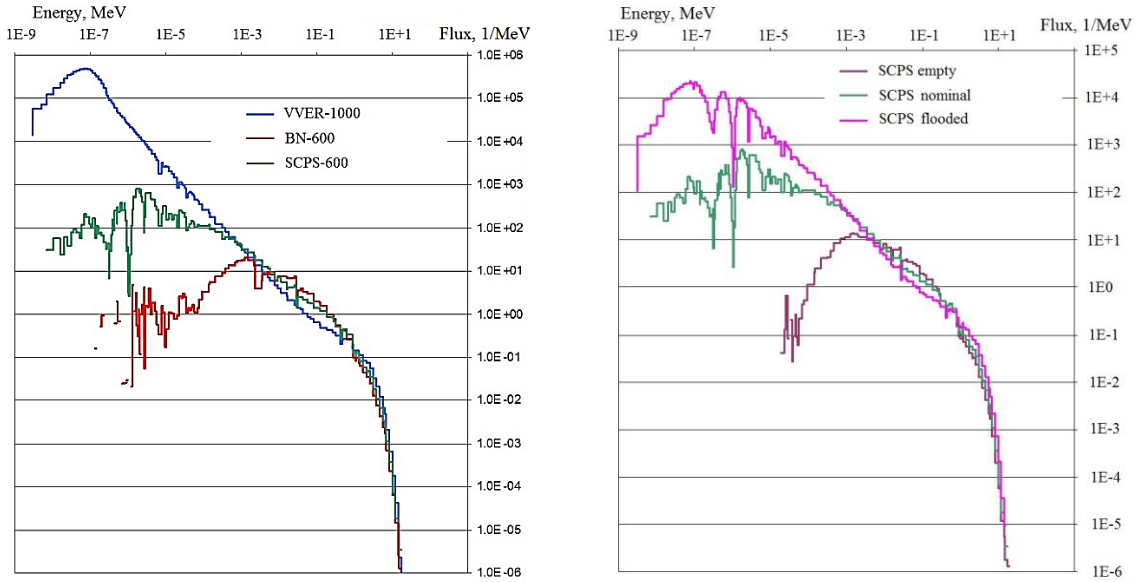
Number of Fuel Assemblies in the core	349
Numbers of FA of side blanket	72
Core height (without butt blankets), *mm*	1700
Axial thickness of blankets (central / upper / lower), *mm*	300 / 150 / 150
Equivalent core diameter, *mm* (without side blanket / with side blanket)	3000 / 3200
Core/Fissile part -averaged volumetric power load, *kW / l*	160 / 210
Core/Fissile part -averaged linear power load, *W / cm*	150 / 200
FA lattice pitch, *mm*	146.6
FA across flat size, *mm*	144.6
Duct thickness, *mm*	2
Number of safety system clusters/rods in cluster	28/42
Compensating and safety system rod material	B_4C (80% ^{10}B)
Number of fuel rods in FA of core/side blanket	199 / 169
Number of cluster rod channels in the core FA	6
Fuel rod bundle pitch in FA of core/side blanket, *mm*	9.49 / 10.67
Fissile core height, *cm*	150
Axial blanket height upper/lower, *cm*	25 / 25
Fuel pin outer diameter/cladding thickness, *mm* - core FA - side blanket FA	 8.4 / 0.6 11.25 / 0.7
FA structure materials	analog of MA-956

4. Specifics of SCPS-600 neutron physics

SCPS-600 has a fast neutron spectrum, but it is softer than in typical sodium fast reactor like BN-600 (see Fig. 4-a). On one hand, this results in less fluence of fast neutrons for the reactor vessel and structure materials, than in the sodium fast reactors at the same value of fuel burn-up, but on the other hand, the softer spectrum is a reason of bigger void reactivity effect [3].

In the situation with loss of coolant the neutron spectrum is shifted significantly into the fast area (Fig. 4-b), that makes fissile plutonium more effective. So, decrease of the coolant density accompanies with increasing k_{eff} (see Fig. 5). Special problem is increasing of positivity of void effect during burn-up, because of accumulation of fission products, better absorbing the neutrons in the thermal-resonant part of the spectrum.

In basic variant of the core design the void reactivity effect (VRE) amounts +5.9$ in the begging of cycle (BOC) and +9.6$ in the end of cycle (EOC). Such a level of VRE is inappropriately big from point of view of ensuring reactor characteristics in the frame of safety limits.



a) b)

FIG. 5. Neutron spectra: a) – comparison of nominal spectra in different reactors: VVER-1000, fast BN-600 and SCPS-600; b) – neutron spectra of SCPS in nominal state, without the coolant (empty) and filled with water (flooded).

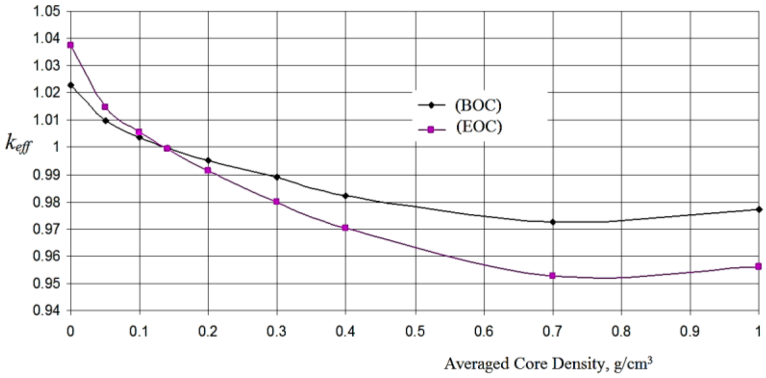


FIG. 5. Evolution of k_{eff} – density dependence from the beginning to the end of fuel cycle for the core with MOX-fuel

The series of calculational studies has been devoted to the investigation of ways to diminish the void effect. Thus introduction of solid moderators, like BeO and ZrH_2 into the core fuel allowed decreasing the positive void effect by 2 – 3 $. The reverse side of such a measure is decreasing the breeding ratio in the reactor from 1.0 in the basic core design to 0.95 - 0.96 in the core with integrated moderators. As well, the required plutonium content has proved by 10% higher than in the basic core design for getting the same burn-up.

Embedding of 40-cm central axial blanket (see Fig. 6) with depleted UO_2 into the reactor core together with integration of 10% ZrH_2 in the MOX fuel has diminished VRE to +1.2$ in BOC and to +3.8$ in EOC not changing the breeding in reactor.

An effective measure to decrease the void effect is introduction of thorium into the core. In one variant of the core design, a central axial zone has been introduced in the middle part of the core as it is shown in Fig. 6. This 30-cm zone has comprised ThO_2 fertile fuel. With such an insertion the value of VRE has decreased to +3.2 $ in BOC and to +3.6 $ in EOC without the usage of integrated moderators. A benefit of this variant is a stability of VRE during the fuel cycle. As well, the breeding ratio has increased up to 1.02 and content of plutonium in the core fuel assemblies has proved by 10% less than in the basic variant with MOX fuel. Note, the effectiveness of the control rods is not considerably affected by insertion of the central axial blanket, because of fast neutron spectrum provides a quite big neutron importance in the core central part.

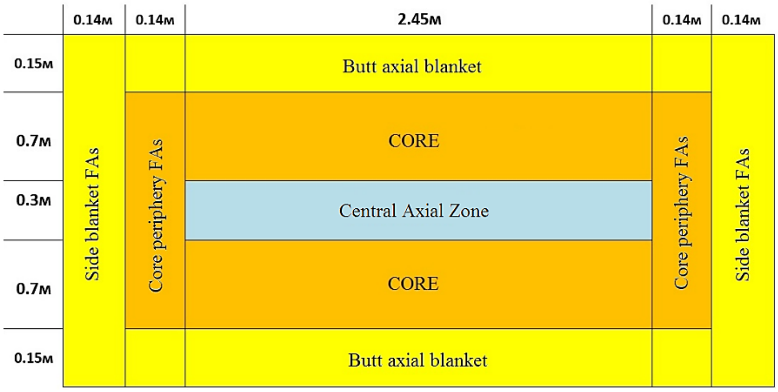


FIG. 6. Axial cross-section of the core with ThO_2 central axial zone.

In equilibrium closed fuel cycle with the own (SCPS-600) fuel a certain quantity of minor actinides and ^3U are produced in the fuel and believed to be return back into the core after reprocessing. In this case the value of VRE slightly increases in BOC (up to +3.5 $) and practically is not changed to the end of cycle (+3.6 $). Breeding ratio has increased up to 1.05 (see Table 3).

Note, the further addition of thorium in the core can not only decrease even more the void reactivity effect, but also unfortunately shift the flood reactivity effect from negative to positive area. In such a case the reactor can become more dangerous not in situation with loss of the coolant, but in flooding the core with liquid water.

Another quite effective way of diminishing VRE is application of low neutron-absorptive structure materials. Fig. 7 demonstrates an example of safe behavior of dependence of k_{eff} on the coolant density in the hypothetical case of usage of composite SiC/SiC as structure material in the reactor core.

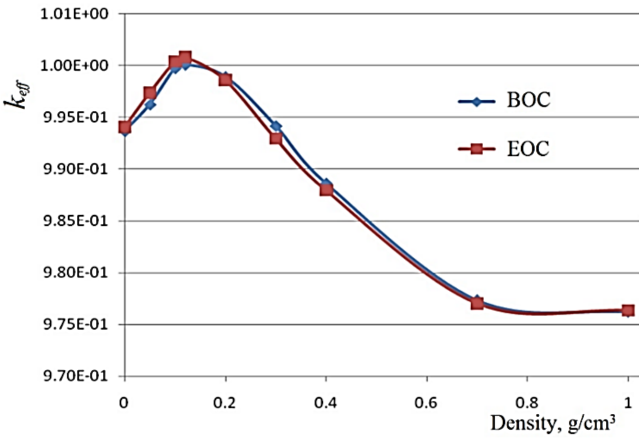


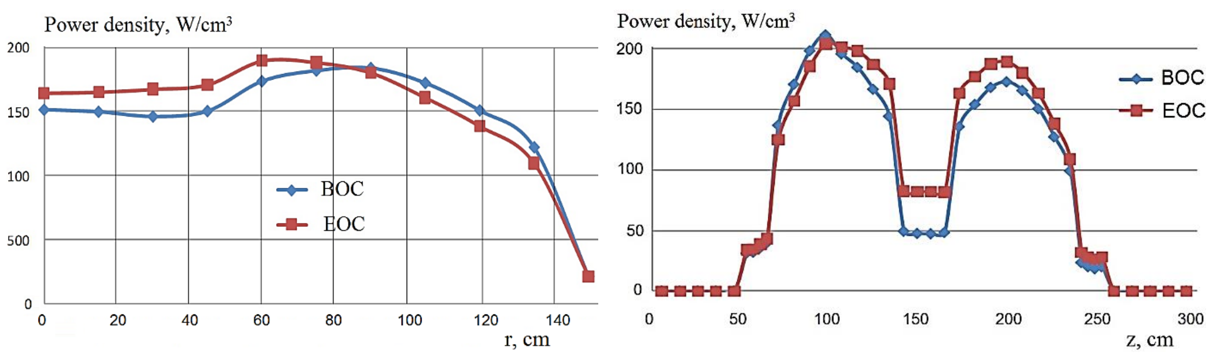
FIG. 7. Dependence of k_{eff} on density for the core with $(Pu\text{-}^8U\text{-}^3U)O_2$ in fissile part, ThO_2 in central insertion and SiC/SiC structure materials.

Characteristics of the reactor closed fuel cycle and reactivity coefficients are given in Table 3 for the core, comprising $(Pu\text{-}^8U\text{-}^3U)O_2$ in fissile part and central axial fertile zone with ThO_2. Two variants of the core structure materials are compared: analog of alloy MA-956 and composite SiC/SiC. Usage of Fe-Cr alloy increases positivity of VRE. However, the net power reactivity coefficient is negative in the all presented core variants (see Table 3).

TABLE III: NEUTRON-PHYSICS CHARACTERISTICS FOR THE CORE WITH (Pu-^{8}U-^{3}U)O_2 AND ThO_2 CENTRAL INSERTION

Characteristics	**Structure materials**	
	MA-956 analog	**SiC/SiC**
Average fuel burnup in stationary cycle, MWt·days/kg: - core - central axial zone - axial breeding zone - radial breeding zone	 54.4 17.7 3.8 6.8	 53.5 20.3 4.2 7.5
Maximum fuel burnup in stationary cycle, MWt·days/kg	79.3	74.9
Annual fuel feeding, mt/year - core - central breeding zone - axial breeding zone - radial breeding zone	 8.08 1.47 1.91 3.10	 8.08 1.47 1.91 3.10
Margin for burn-up, $	3.95	3.99
$BR = \frac{\sum(m_{U233}+m_{U235}+m_{Pu239}+m_{Pu241})_{output}}{\sum(m_{U233}+m_{U235}+m_{Pu239}+m_{241})_{input}}$ - average breeding ratio - core breeding ratio - central zone breeding ratio - axial blanket breeding ratio - radial blanket breeding ratio	 1.02 0.88 0.05 0.04 0.05	 1.05 0.90 0.05 0.06 0.04
Reactivity coefficients: - delayed neutron fraction (BOC/EOC) - VRE (BOC/EOC) - Doppler reactivity coefficient (BOC/EOC), 1/K - density reactivity coefficient (BOC/EOC), cm^3/g - power reactivity coefficient (BOC/EOC), 1/MWe - maximum after-trip Xe load, mk	 0.0038 / 0.0037 +3.5/+3.6 (-2.1/-2.3) E-05 (-4.5/-6.0)·E-02 (-8.9/-8.5)·E-06 1.5	 0.0038 / 0.0037 -1.95/-1.48 (-2.6/-2.6)·E-05 (+2.3/+2.5)·E-02 (-2.1/-1.9)·E-05 1.6

Fortunately, a quite stable-in-cycle core power distribution can be organized in reactor by zoning of plutonium content by groups as it is shown in Fig. 3. An evolutions of the radial and axial power distributions are presented in Fig. 8 for the core design with (Pu-^{8}U-^{3}U)O_2 fuel, central axial ThO_2 zone and Fe-Cr-alloy structure materials.



a) b)

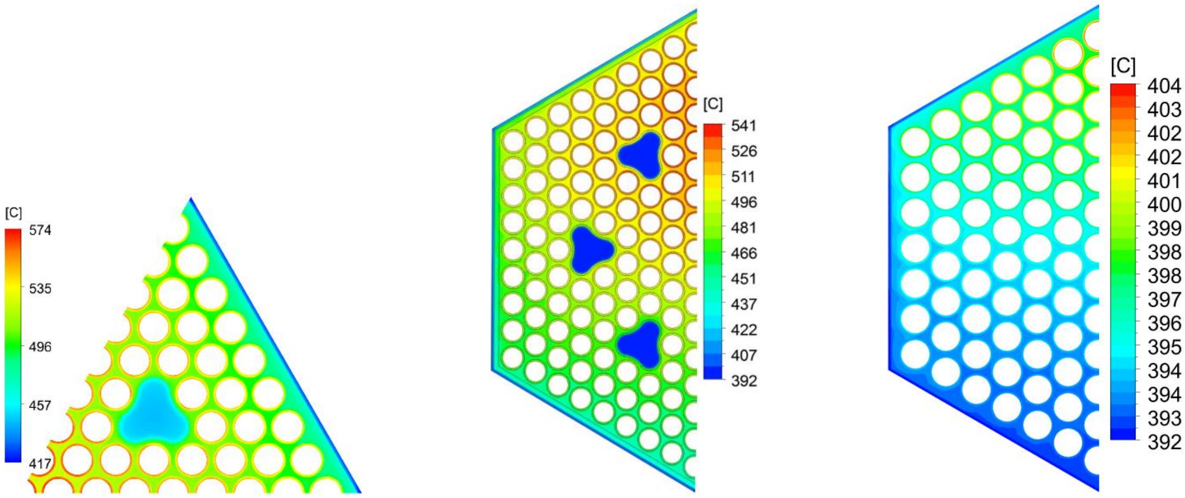
FIG. 8 Radial (a) and axial (b) power distributions in the core with (Pu-^{8}U-^{3}U)O_2 fuel, central axial ThO_2 zone and Fe-Cr-alloy structure materials

5. Thermal hydraulics aspects

An increment of the coolant temperature in the reactor core is not very big (390 – 500 ^{o}C). There is quite big value of volumetric heat capacity in this pseudo-vapor area at the pressure of 24.5 MPa, which is comparable with that one of liquid sodium. So, having the mild value of the core volumetric power production (210 kW/ l in the core fissile part) the averaged value of the coolant mass flux in the FAs is proved mild too (1200 kg/m^2/s). Given this condition, the hydraulic resistance does not exceed 0.3 MPa and total pressure loss in the primary circuit is about 0.6 MPa. Primary main circulation pump (MCP) has practically the same pressure head and volumetric flow rate as MCP of reactor VVER-1000, but, unlike VVER, it pumps a dense pseudo-vapor not liquid water. SCPS-600 has two MCPs, so their overall electric self-consumption is proved 20% less than in VVER-1000, which has four MCPs.

Stable radial power distribution gives a possibility to design a chart of orifices at the inlet of the core FA, which allows getting the necessary flow distribution among Fuel Assemblies, minimizing the coolant and structure temperatures in the core with account of power distribution drift during the fuel cycle.

Fortunately, a heat transfer in the area of nominal coolant parameters is close to the one-phase heat transfer, described by formula by Dittus-Boelter [4]. This simplifies the task of calculation of temperatures of the coolant and fuel rod claddings in the Fuel Assemblies, especially in the area of hot spots. Fig. 9 presents CFD-calculated temperature distributions in hot spot cross-sections of the core Fuel Assemblies.



a) b) c)

FIG. 9. Temperatures in hot-spot cross-sections of Fuel Assemblies:
a) – central FA, b) – periphery FA, c) – FA of side blanket

Fig. 9 demonstrates, that fuel rod cladding temperatures nowhere exceed 574 ^{o}C. Account of uncertainty factors gives an upper temperature limit for claddings of 650 ^{o}C in nominal reactor operation. Such a temperature level is believed to be appropriate for ensuring reliable operation of the Fe-Cr-Al alloy.

6. Thermal hydraulic and neutronic reactor stability

Studies of thermal hydraulic stability has shown a big margin to onset of inter-channel instability of the flows in the core Fuel Assemblies and to the limit of overall primary circuit flow instability.

Neutronic reactor stability has also quite enough margin to the safe limit. Fig. 10 shows the limits of area of reactor stability in coordinates of unitless Doppler and density reactivity coefficients: $-\frac{(T_{fuel}-T_{coolant})}{\beta_{eff}}\frac{\partial k_{eff}}{\partial T_{fuel}}$ and $-\frac{\gamma}{\beta_{eff}}\frac{\partial k_{eff}}{\partial \gamma}$. Drifts of nominal points of reactor operation during the fuel cycle for two variants of the core loads are indicated at the picture. One can see from Fig. 10 that nevertheless nominal point of operations is drifting to the instability boundary, the considerable margin of stability remains all over the fuel cycle.

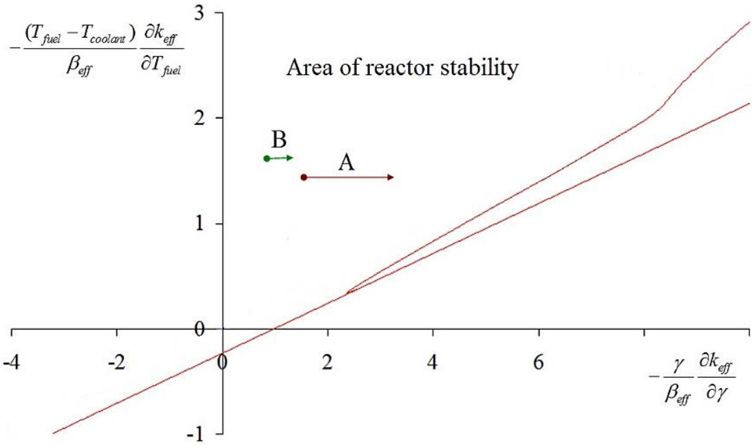


FIG. 10. Area of reactor stability and drift of nominal points of reactor operation: A – core with MOX fuel; B – the core with (Pu-^{8}U-^{3}U)O_2 in fissile part and central axial fertile zone with ThO_2

7. Conclusion

A brief description of SCPS-600 reactor concept, presented in this paper, has addressed to only some aspects of neutron physics, thermal hydraulics and reactor stability. The studies done in the result of R&D [1, 2] has shown a possibility of creation of fast supercritical steam reactor, which can operate in regime of self-provision of reactor by its own secondary fuel in the equilibrium closed fuel cycle.

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