Estimates of the modular underwater SNPP option

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***Abstract*: One of the most promising and growing areas in nuclear energy today is low-power nuclear power plants (LPNPs), what is caused by the inherent features of this area. But technical solutions used in the state of the art NPP designs do not allow to create a single ready-to-use fabricated module in transport dimensions, so it requires separate delivery of the reactor, turbine and other equipment and causes a significant amount of construction and installation work on site. This approach, due to «economies of scale», does not allow to create a cheap source of energy. The purpose of this work is to carry out calculations of an alternative NPP design based on the so-called «nuclear battery» approach or «Vinogradov scheme» – a single module that contain main equipment of the nuclear power plant (reactor, turbine, generator, heat regeneration system and condenser). There was considered a fast reactor with 125 MW thermal power with a lead coolant, parameters of the primary circuit are similar to BREST reactor (inlet/outlet coolant temperature – 420/540 ^{0}C) and supercritical Brayton cycle on carbon dioxide (initial/final pressure 20.0/8.0 MPa, initial/final temperature of the cycle 520/390 ^{0}C) with single-stage regeneration. Received results confirm the possibility of creating a single-module SMR with the following parameters: electrical power – 32 MW, efficiency – 25.6%, housing length of the module – 18.7 m, housing diameter 4.3 m, weight – 900-950 tons, fuel lifetime – 1080 days.**

Keywords—nuclear power plant, NPP, small nuclear power plant, SNPP, small modular reactor, SMR, Brayton cycle, supercritical cycle, carbon dioxide, CO2.

I. INTRODUCTION

Small nuclear power plants (SNPPs) and small modular reactors (SMR), based on which nuclear stations are created, today are one of the promising areas for the development of atomic engineering. In reviews [1, 2] and numerous articles on this topic, there are dozens of design options for nuclear power plants with different types of reactor plants. Interest in this area is driven by the desire to reduce the cost of nuclear electricity, which requires huge capital expenditures for the construction of high-power power plants [3], and the price per power unit in the case of SNPPs rises catastrophically due to «economies of scale».

In the paper [4] proposed four drivers for reducing the cost of SNPPs: modularity and factory production; simplification of design; standardization; harmonization. The target is to achieve cost of electricity (or LCOE) for SNPPs comparable to high-power nuclear power plants (Fig. 1).

An analysis of the proposed options for achieving specified goals and feasibility is far beyond the scope of this article. It is supposed, that modularity in most cases implies the manufacture of equipment in modules at the factory, but not SNPP transportation to the site of installation (single unit ready for launch). Wherein projects shouldn't be themselves just smaller copies of large nuclear power plants with possible improvements to the main equipment (integrated steam generators in the RITM and NuScale projects), but with inevitable installation and construction work.

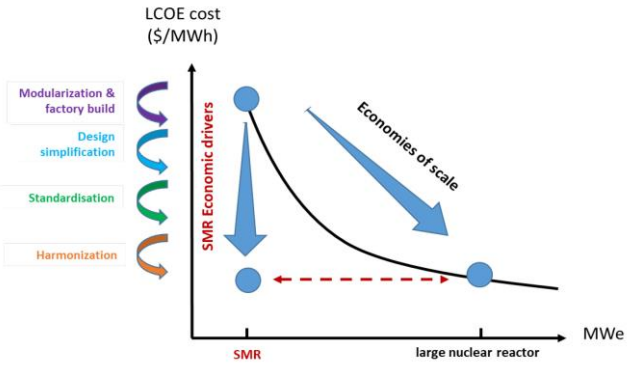


Figure 1. Ways to improve the economic indicators of ASMM [4].

According to the IAEA, today there are over 70 NPP projects at various stages of development. In addition to Russia, the USA, China, the Republic of Korea, France, Argentina, the Great Britain and other countries are implementing their reactor concepts. At the same time, in the options specified papers [1, 2, 4], most of the under consideration NPPs (RITM-200, Nuscale, etc.) are based on well-developed standard PWR solutions. Considering SNPP segment, it's supposed not to be obtaining cheap electrical energy, due to the large amount of capital cost and relatively low value of electrical energy production (negative economies of scale). Based on the above, it becomes obvious that a radical design changes are necessary, in particular when considering SNPP segment.

Vendors from different countries are developing their own technological solutions, looking for methods to optimize costs, design processes, licensing and construction. Thus, in papers [5, 6] the use of thermodynamic cycles with supercritical CO2 as a working fluid in the secondary circuit of a nuclear power plant was considered. In [5], different options of the basic thermal circuit were studied in the part of the secondary circuit, from the traditional Brayton cycle with regeneration, to more complex options: with partial cooling and with recompression of the working fluid. According to the authors [5], the calculated efficiency value (net) of the Brayton cycle with regeneration can reach 42.2% (at an initial cycle temperature of 550 ^{0}C), the maximum calculated efficiency value (net) was obtained for the Brayton cycle with recompression - 47.28% (with same initial temperature).

In papers [7, 8 and 9], use of supercritical CO2 cycles for fast reactors with liquid metal coolant was considered. In particular, in [7] a sodium-cooled reactor plant was considered, in [9] it was proposed to consider a fast lead-bismuth reactor (based on the russian SVBR-100 reactor) as one of the options under consideration. It was found that the vessel size required for the reactor, safety systems and power cycle could be significantly reduced (50–80%) compared to a traditional PWR. The lead-bismuth reactor provides compatibility with water and air, which leads to a simplified design with similar parameters compared to sodium-cooled

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fast reactors, but with worse thermal-hydraulic and corrosion characteristics.

The use of a fast reactor with lead coolant and supercritical CO2 in the secondary circuit was considered in papers [10, 11, 12] – in the so-called «Vinogradov scheme» - a modular single-case double-circuit plant, completely manufactured at the plant and delivered to the site of use as a ready-to-operate module. This module contains fast reactor with lead coolant, gas turbine on the supercritical carbon dioxide Brayton cycle (SCO2-Brayton cycle), initial heater, regenerator, cooler, electric generator and the necessary control and protection equipment [10].

The proposed approach: when, instead of the long-term construction of a large nuclear power plant project, several ready-to-operate NPPs («nuclear batteries») are supplied to the consumer from the factory, proclaimed by Russian (Vinogradov, [10, 11, 12]) and American (Jacopo Buongiorno at all, [8]) authors like a next step in nuclear energy. The manufacture of the « nuclear battery», its transportation, maintenance, and replacement are entirely the responsibility of the manufacturer. The consumer pays only for electricity.

The purpose of this work is to make calculations of the feasibility of the «nuclear battery» concept based on the «Vinogradov scheme» and to determine its main technical parameters. At the same time, in contrast to scheme, proposed by A.A. Vinogradov [10], removal of unused heat at the turbine outlet was considered in the cooler, and lead coolant parameters in the core were taken as in the BREST reactor: inlet/outlet temperatures – 420/540 ^{0}C (in contrast to proposed in paper [10] – at the level of 500/1100 ^{0}C).

II. The Main Conceptual Decisions Made in the «Vinogradov Scheme»

In this work, solutions proposed by A.A. Vinogradov in his works for small power nuclear power plants [10] are considered and partially used. NPP structural layout («Vinogradov scheme») is shown in Fig. 2.

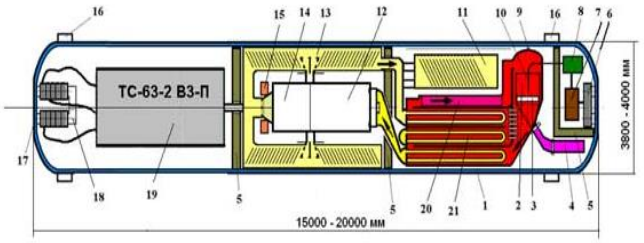


Figure 2. NPP structural layout («Vinogradov scheme», [10]). Designations: 1 – sealed housing; 2 – reactor core; 3 – grids for dropping nuclear fuel from the core; 4 – containers for dumped fuel (so called «coffins»); 5 – thermal partition; 6 – ultrasonic control equipment; 7 – unit of control; 8 – control and safety system; 9 – gas valve; 10 – CPS rod; 11 – heat exchanger No. 1 (for initial heating); 12 – gas turbine; 13 – main heat exchanger No. 2; 14 – compressor; 15 – compressor electric engine; 16 – support rings; 17 – magnetically transparent vessel wall; 18 – high-voltage bushing-type transformer; 19 – turbogenerator; 20 – liquid metal circulation pump (1st circuit of the reactor); 21 – heat exchanger of the 1st circuit.

The scheme assumes the use of a fast reactor with a lead coolant in the primary circuit, and carbon dioxide (CO2) with supercritical pressure at all points of the thermal circuit as the working fluid. The main equipment of the nuclear power plant is assumed to be located in a cylindrical housing. The transmission of electrical energy, as well as the reception and transmission of control signals, is supposed to be carried out in a "non-contact" way through the housing wall. The use of spherical nuclear fuel with an original design (with a hydrodynamically transparent shell) [11, 13] (Fig. 3) reduces the resistance to coolant movement.

To increase safety, it was envisaged to use a specially designed active core, which provides fuel collection in the so-called «coffins» (special containers in which the fuel is in obviously subcritical state and is cooled due to the natural circulation of lead and heat release through the nuclear power plant body) in emergency situation.

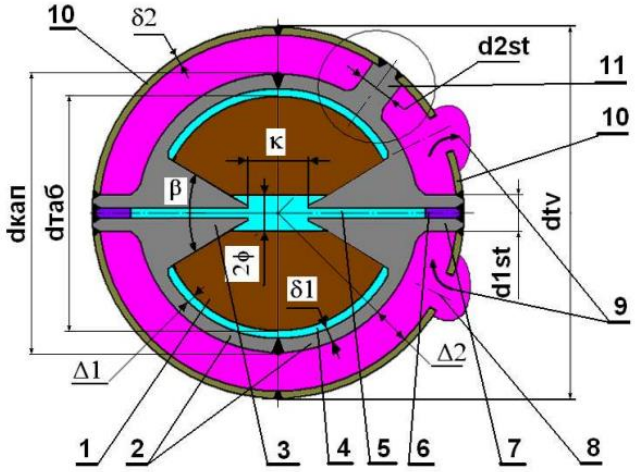


Figure 3. Spherical nuclear fuel with a hydrodynamically transparent shell [11, 13]. Designations: 1 – nuclear fuel pellet; 2 – inner shell of fuel element; 3 – protrusion; 4 – gap for fuel swelling; 5 – gaps with heat-conducting gas; 6 – end-caps; 7 – heat-eliminating parts; 8 – axis of a hole; 9 - holes for coolant passage; 10 - outer shell of fuel element; 11 – heat-eliminating parts.

III. Calculation Estimates of a Modular SNPP on the Brighton SCO2 Cycle

As part of this work, some calculations were made regarding the possibility of creating one of the options of so-called «Vinogradov scheme» [10].

This paper introduces a double-circuit modular SNPP with fast neutron reactor, thermal power of which is 125 MW (thermal), and high-temperature lead coolant (liquid metal) in the 1st circuit.

In terms of working fluid of the secondary circuit it is proposed to chose a supercritical CO2 gas (S-CO2 Brayton cycle) with the simplest one-stage regeneration. The principal thermal diagram and T-S diagram of the cycle are presented in Figure 4 [6]. This cycle option was chosen to simplify the nuclear power plant and reduce the number of heat exchangers, sacrificing a higher value of cycle efficiency in favor of a compact installation).

More complex schemes of the Brayton carbon dioxide cycles were also considered (with intermediate superheating, with intermediate cooling, with partial cooling, with recompression) with different parameters of working fluid [5, 6]. Their use leads to an increase of efficiency, but, in turn, leads to an increase in the number of units of heat exchange equipment, dimensions and weight of the installation. Calculations for more complex cycle options was not carried out in this work.

The design diagram of the nuclear power plant is presented in Figure 2 (so-called «Vinogradov scheme»). On the right side of the nuclear power plant one can see the nuclear reactor

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and primary circuit equipment, biological protection elements (for simplicity, not indicated on the design diagram), containers for fuel dumping (so-called «coffins») and control equipment. A nuclear reactor is geometrically a hexagonal prism, the frame of which is «sheathed» with sheets of steel with small thickness (the pressure of the liquid metal coolant is assumed to be 1 MPa), which includes a cylindrical basket (placed conventionally in the center) with a spherical fuel and reactor internals.

On the left side of the nuclear power plant there is an electric generator, a gas cooler and equipment for organizing the circulation of the cooling medium.

The operation diagram of the nuclear power plant is as follows (Fig. 4). The compressor receives recirculated CO2, which is sent for heating in the regenerative heater after compression. The heated flow of the working medium enters the reactor to further increase the temperature. Next, the supercritical working fluid is sent to a gas turbine, which drives the electric generator. After expansion in the turbine the working fluid enters the regenerative heater to transfer heat to the working fluid compressed in the compressor. Then, before being supplied to the compressor, the «cooled» CO2 stream is directed to cooler, where additional cooling of the working fluid occurs.

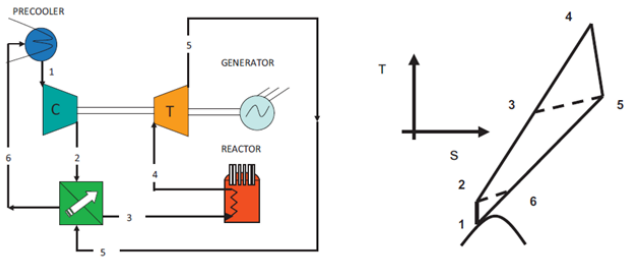


Figure 4. Schematic thermal diagram of a SNPP and T-S - cycle diagram [6]. Designations: 1 – outlet from the gas cooler / inlet to the compressor; 2 – output from the compressor / input to the regenerative unit (according to the heated medium); 3 – exit from the regenerative unit (through the heated medium) / entrance to the primary heater; 4 – output from the primary heater / inlet to the gas turbine; 5 – exit from the gas turbine / entrance to the regenerative unit (according to the heating medium); 6 – exit from the regenerative unit (through the heating medium) / entrance to the gas cooler.

The initial data for the calculations are presented in Table 1.

TABLE 1. INITIAL DATA FOR CALCULATION

Parameter	Meaning:
Reactor thermal power Qt , MW	125
Fuel element type	Spherical type
Fuel type	(U- Pu)N
Fuel density, g/cm3	14.32
Fuel element cladding	EP823
Reactor coolant (RC)	Lead
RC temperature at the inlet/outlet of the core, ^{0}C	420/540
Working fluid (WF)	SCO2
WF temperature at the heater inlet/outlet, ^{0}C	340/520
WF temperature at the cooler outlet, ^{0}C	35
Pressure WF at the entrance to the core, MPa	20
Pressure WF at the inlet to the cooler, MPa	8
Cooling water temperature, ^{0}C	15

The parameters of the reactor coolant were adopted as in the BREST reactor plant: inlet/outlet temperatures – 420/540 ^{0}C [14] (i.e. very conservative, in contrast to those proposed by A.A. Vinogradov in [10] at the level of 500/1100 ^{0}C).

(U-Pu)N-fuel [15], which has sufficiently high thermophysical characteristics and meets safety requirements (density higher than the density of reactor coolant), was considered as nuclear fuel. EP823 steel [16], which is resistant to lead at high temperatures, was chosen as the fuel cladding material.

The working fluid pressure at the inlet of gas turbine was determined to be 20 MPa. To exclude the transition of the working fluid to the saturation zone during cooling in the gas cooler, the pressure in the gas cooler was taken at the level of 8.0 MPa, and an additional limitation was introduced on the sCO2 cooling temperature in the cooler - the outlet temperature was not less than 35 ^{0}C. The water parameters in the cooler are assumed to be at the following level: inlet temperature – 15 ^{0}C, pressure – atmospheric.

As part of the study, the thermal circuit of the nuclear power plant was calculated and the value of the net cycle efficiency was determined, thermal-hydraulic calculations of the reactor and heat exchangers, calculations of the gas turbine and compressor were carried out, and the weight and size characteristics of the main equipment and the nuclear power plant housing were determined.

Calculations of the thermal circuit, gas turbine and compressor were carried out according to the methods and recommendations [17, 18 and 19]. The axial layout of the turbine and compressor was adopted. The gas turbine, compressor and electric generator are located on the same shaft. The temperature of the working fluid at the inlet to the gas turbine was 520 ^{0}C, at the outlet – about 390 ^{0}C.

Thermal-hydraulic calculations of the active core (AC) were carried out according to the methods and recommendations [20], data on resistance coefficients were taken according to the recommendations [21].

The reactor active core (AC) is a cylindrical basket of fairly compact dimensions (diameter – 0.954 m, height – 0.35 m) with a spherical fuel and lead coolant. The calculation of the reactor core (core) was carried out according to the methods of [22] in the Sintez program. It was assumed that the core would be divided into 3 subzones with different contents of fissile nuclides and the presence of a lead reflector with 0.5 m thick.

Mixed nitride U-Pu fuel was considered as fuel. In accordance with the recommendations given in the source [15], EP823 steel [16] was selected as a structural material for the shells of fuel elements in the reactor with a lead coolant (with BREST parameters) and the temperature of the outer surface not higher than 560 ^{0}C (taking into account the erosive effect of the reactor coolant) during normal operation. Then mechanical properties of fuel elements shells do not change upon contact with lead coolant, but the coolant circulation rate in the core is additionally limited.

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To solve the problem of high hydraulic resistance during the flow of reactor coolant in the core fuel pebble, the source [11, 13] proposed the use of a hydrodynamically transparent shell for spherical fuel elements (Figure 3). This task is optimization, due to the fact that, on the one hand, it is necessary to reduce hydraulic losses along the path of the coolant and reduce the costs of pumping it, on the other hand, by placing spherical fuel elements at a distance from each other («sparse» fuel lattice), it is necessary to use more highly enriched fuel (in this work it is not carried out).

Calculations of heat exchange equipment were carried out according to classical methods and recommendations [23, 24 and 25]. The material of the tubes of the primary heat exchanger (heater) is 10Х15Н9С3Б1-ІІІ according to the recommendations [16], tubes 16x1.5 mm. The material of the regenerator tubes is steel X12MF, tubes 16x1.5mm. The material of the cooler tubes is titanium alloy VT1-1, tubes 16x0.5mm.

In the calculations, all heat exchangers were considered as counterflow devices with a higher pressure medium in the tubes. The layout of the heat exchangers inside a cylindrical housing required the use of complex multi-pass schemes (for example, the regenerator is supposed to be made in the form of a ring heat exchanger, inside which a turbine and a compressor are located), which are not considered in this work.

The CO2 temperature at the heater inlet was taken to be no lower than 340 ^{0}C (at least 10 ^{0}C higher than the melting point of lead).

The thermophysical properties of the reactor coolant were taken according to the data of [26], and the working fluid - according to the data of [27].

A feature (and additional complication) of the thermal scheme is the «behavior» of carbon dioxide at parameters close to the saturation line. In this region, CO2 gets fairly large values of isobaric heat capacity (at some temperature values a resonant character is even observed); with increasing parameters of the working fluid, the value of isobaric heat capacity decreases significantly (several times), while the rate of decrease of the value noticeably decreases. This «behavior» of carbon dioxide leads to significant heating of the working fluid in the compressor unit (during compression) and amounted to about 95-100 ^{0}C.

To use the heat of the working fluid at the outlet of the gas turbine a recuperator was considered. The recuperator (or regenerative heater) is organized in the form of a ring heat exchanger, which is located around the gas turbine unit, it allows considering more closely-grouped placement of nuclear power plant equipment. The temperature of the heating CO2 at the outlet of the recuperator is determined by the heating of the working fluid in the compressor (during compression) and the temperature pressure in the heat exchanger and is about 135-140 ^{0}C. Thus, the role of the regenerative heater in this scheme becomes obvious, the thermal power of which is significant, and under certain conditions becomes greater than the thermal power of the «main» heater (an additional feature of the thermal circuit).

One of the disadvantages of the «simplified» thermal scheme appears in the peculiarities of regenerative heating of the working fluid: the value of the isobaric heat capacity of compressed (up to 200 bar) CO2 at the inlet to the recuperator (which should be heated) significantly exceeds the value of the isobaric heat capacity of CO2 at the exhaust from the gas turbines (the pressure of which is not more than 75 bar). This is an inevitable source for «irreversibility» of the cycle and heat losses.

To estimate the dimensions of the turbogenerator, a T-32-2V3 turbogenerator with a power of 32 MW (rotation speed 3000 rpm) was taken as a prototype with dimensions: generator length - 8100 mm, generator height - 3000 mm [28].

The main calculation results are presented in Table 2.

The calculated value of electric power (net) was 32 MW (net efficiency 25.6 %).

TABLE 2 . MAIN CALCULATION RESULTS

Parameter name	Meaning
Electric power (net), MW	32.0
Thermal circuit efficiency (net), %	25.6
Core dimensions, diameter/height, m	0.95 /0.35
Reactor coolant flow rate, kg/s	7065
Working fluid (sCO2) flow rate, kg/s	558.3
Number of sub- zones in active core	3
Pu concentration by subzone , %	13.1; 18.5; 21.8
Conversion factor	0.464
Core peaking factor	1.152
Duration of fuel campaign, days	1080
Heater surface area, m2	1256
Regenerator surface area, m2	5513
Cooler surface area, m2	1451
Overall dimensions of the nuclear power plant housing (diameter x length), m	4.3 x 18.7
Module weight, t	900-950

Layout calculations are far beyond the scope of this article. According to calculated estimates, overall dimensions module were: diameter - 4.3 m, length - 18.7 m. According to estimates, the mass of such assembled nuclear power plant, excluding the mass of the primary circuit coolant and nuclear fuel, was at least 900-950 tons.

The main results were obtained and published as part of the work [29-32].

IV. CONCLUSION

1. The calculation shows the possibility of creating a modular SNPP with supercritical CO2 Brayton cycle with a net efficiency of about 25.6% in dimensions (estimates): length - no less than 18.7 m, diameter - no more than 4.3 m, weight - 900-950t, campaign – 1080 days.

2. It is necessary to emphasize the extremely conservative nature of the estimates, which allows us to consider the obtained estimates as reliable upper limits of technical parameters, in particular: options for optimizing the regeneration system were not considered; the sealed housing was considered to have the same thickness along its entire length; methods for intensifying heat transfer were not considered; modern options for radiation protection (the use of multilayer, multi-component protection options), etc. were not considered.

3. The obtained estimates show the prospects for continuing work in this direction, since the considered option

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allows us to provide the power of the floating nuclear power plant "Ak. Lomonosov" with two such modules.

4. One of the options that can radically improve the parameters of SNPPs is single-circuit options, which, in the case of micro-power, can be considered promising, allowing delivery of the module to the consumer by ground transport.

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