PROJECT OF A MULTIPURPOSE LEAD REACTOR WITH A HARD NEUTRON SPECTRUM

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**Abstract**

The article describes a conceptual design of a metal coolant reactor, which can be used to train young specialists, produce isotope products, develop an invention, etc. The concept of a low-power 0,5 MW and 25 MW thermal reactors with a hard neutron spectrum in the core is proposed. The reactor has the following feature: a sufficiently high neutron flux density 3,4·1015 n/(cм2·s) in the center of the core, high average neutron energy 0,86 МeV in the center of the core, as well as a high proportion of hard En>0,8 МeV, neutrons 35%. Extremely high design parameters of the reactor are achieved due to the small size of the core, DxH≈0,40х0,42 м2, innovative metal fuel made from an alloy of energetic plutonium with zirconium, Pu58%-Zr42%, and a heat carrier from a natural lead melt. The mass of power plutonium loaded into the reactor is 92 kg.

## INTRODUCTION

Modern large-scale nuclear energy emerged as a cumulative result of technical progress in the nuclear industry and the objective needs of the economy. The history of the development of nuclear energy is associated with an increase in the capacity of single units from 500 MW to 1500 MW and the creation of large nuclear power plants on their basis for about half a century. Modern nuclear power is characterized by the concentration of large reactor capacities in a relatively small number of nuclear power centers.

For a long time, nuclear submarines and nuclear surface ships remained the areas of application of small nuclear power. The only example of an albeit limited but highly effective commercial application of low-power nuclear plants is the Soviet and now Russian nuclear icebreaker fleet, as well as the floating nuclear power plant Akademik Lomonosov, which was commissioned in May 2020. Thus, it can be stated that all these years, small-scale power generation for commercial purposes has practically not developed. The world economy now needs, and will continue to need even more acutely, modern autonomous, reliable, environmentally friendly and cost-effective energy sources.

As an example of such sources for the purposes of electricity supply, heat supply, as well as for some technological needs, along with traditional and renewable energy sources, nuclear installations of low power may also be in demand.

A nuclear reactor cooled with liquid lead and using commercially developed uranium dioxide as a fuel with an enrichment of 19.7% in the 235U isotope. In order to safely handle the reactor, a relatively low thermal power of the reactor is assumed, 0.5 MW (a 25 MW reactor will be discussed in the next article). The possibility of accelerating the reactor on prompt neutrons is excluded due to the small reactivity margin, less effective fraction of delayed neutrons. The work is based on the experience at the Obninsk Institute of Atomic Energy (IATE) in the development of low-power reactors, for example, the MASTER IATE heat supply reactor with a capacity of up to 0.3 MW [3], as well as on the experience of the State Research Center of the Russian Federation - Physics and Power Engineering Institute» in the field of development of reactors with heavy liquid metal coolant for various purposes.

## GENERAL DESIGH SCHEME

An integral monoblock type arrangement was proposed, providing the placement of the inner core body together with reflectors, as well as the upper protective plug, heat exchangers, coolant technology devices that maintain the thermodynamic activity of oxygen dissolved in lead, thermal control sensors, in-vessel radiation shielding and a buffer tank over the free the level of molten lead for the considered reactor design [1, 2]. The reactor vessel is a robust steel cylindrical vessel with an elliptical bottom and a lid with holes for mounting and fastening the in-vessel structures.

It is assumed that the circulation of the coolant along the first circuit is carried out by natural convection. The choice of natural circulation for the primary circuit makes it possible to avoid the problem of developing and substantiating the operability of a circulation pump for pumping a heavy liquid metal coolant. Calculated estimates show that to ensure the natural circulation of the primary coolant, it is sufficient to have a lifting section with a height of ~ 3 meters above the core. This will provide a head of natural circulation of the order of 1400 Pa. At a coolant velocity of 0.1 m / s through the core, the hydraulic resistance of the core will be 480 Pa. If we assume that the total pressure loss along the loop is approximately three times higher than the pressure loss in the core, it can be concluded that natural circulation provides the required flow rate in the primary loop. The main elements of the proposed design of the reactor and their characteristics are given in table. 1.

TABLE 1. THE MAIN CHARACTERISTICS OF THE REACTOR. SAMPLE TABLE: ACCURACY OF NODAL AND CHARACTERISTIC METHODS

|  |  |  |
| --- | --- | --- |
| № | Characteristic | Value |
| 1 | Thermal power, MW | 0.5 |
| 2 | Number of fuel assemblies in core, number  | 7 |
| 3 | Number of fuel rods in fuel assemblies, number | 252 |
| 4 | Core diameter, mm | 716 |
| 5 | Core height, mm | 620 |
| 6 | Fuel element diameter along the ribs, mm | 13.9 |
| 7 | Fuel element diameter along the smooth part, mm | 12.7 |
| 8 | Fuel element lattice pitch, mm | 14.0 |
| 9 | Fuel | UO2 |
| 10 | Uranium enrichment U235, % | 19.7 |
| 11 | Fuel load, kg | 1176 |
| 12 | Loading uranium U235, kg | 205 |
| 13 | Core campaign, years | 20 |

The neutron-physical and thermal-hydraulic characteristics of the reactor are given in table. 2.

TABLE 2. THE NEUTRON-PHYSICAL THERMAL-HYDRAULIC CHARACTERISTICS OF THE REACTOR

|  |  |  |
| --- | --- | --- |
| № | Characteristic | Value |
| 1 | Coolant | Pb |
| 2 | Coolant volume in 1 circuit, m3 | ≈6.0 |
| 3 | Doppler effect of reactivity ∆K / K, 0K | -1·10-5 |
| 4 | Effective fraction of delayed neutrons, % | 0.720 |
| 5 | Lead temperature at the entrance / exit the core, 0С | 460/500 |
| 6 | Maximum fuel element cladding temperature, 0С | 526 |
| 7 | Average speed of the coolant in the core, m / s | 0.1 |
| 8 | Volume fraction of the coolant in the core,% | 26.0 |
| 9 | Average linear load on the fuel part of a fuel element, kW / m | 0.46 |
| 10 | Average density of energy release in the core volume, kW / L | 2.73 |
| 11 | Average neutron flux density in the core, n / cm2 • s | 1.6·1013 |
| 12 | Neutron flux density in the center of the core, n / cm2 • s | 2.4·1013 |

## CORE

The core consists of seven working cassettes, each of which is an assembly of 252 fuel elements. In the center of each cassette, a technological channel is organized for the control and protection system organs. The fuel element includes a fuel block made of fuel pellets with a diameter of 11.5 mm, the height of the fuel column is 620 mm, the average fuel density over the column is 10.5 g/cm3. There is a gap filled with helium between the fuel element cladding and the fuel pellet. The gap size of 0.05 mm is selected so that radiation swelling of the fuel during burnup does not lead to thermomechanical contact and interaction of the fuel with the cladding. The total height of the fuel element is 1200 mm, of which 400 mm is the compensation volume for collecting fission gas products.

The view of the reactor fuel cell is shown in Figure 1. Figure 2 shows a cartogram of a separate fuel assembly. A cross section of the reactor through the core is shown in Figure 3. Figure 4 shows a general view of the reactor.



FIG 1. Fuel cell cartogram: 1 - fuel; 2 - gap; 3 - shell with spacer ribs



FIG 2. Fuel assembly cartogram: 1 - fuel element; 2 - technological channel



FIG 3. Cross section of the reactor: 1 - reactor vessel; 2 - core shell; 3 - fuel assembly.



FIG 4. General view of the reactor: 1 - core; 2 - heat exchanger; 3 - drives of the control and protection system; 4 - shell; 5 - reactor vessel.

## CIRCULATION DIAGRAM OF THE PRIMARY COOLANT

The natural circulation of the primary coolant occurs in the following sequence.

 From the lower chamber, the lead coolant is fed into the core, passes through it from bottom to top, washing the fuel elements and taking heat from them, and heated to the common receiving (outlet) chamber above the core, then rises and enters 12 heat exchanger modules connected in parallel , where, moving from top to bottom, it washes the heat exchange tubes, giving off heat to the coolant of the second circuit, enters the lower chamber, from where it again enters the core.

A melt of a lead-bismuth eutectic or a steam-water mixture is considered as a coolant in the secondary circuit. The final selection of the secondary coolant will be made later.

## PASSIVE HEAT DISSIPATION SYSTEM

The passive heat removal system is designed to remove residual heat from the core and it is included in this project. The passive heat removal system includes pipelines and a chimney, which together form a natural air circulation loop in the gap between the walls of the vessel and the shaft, through which the passive removal of residual heat through the vessel and its release into the atmosphere is provided during normal and emergency cooling of the reactor.

## SOLIDWORKS SOFTWARE OVERVIEW

The computer-aided design system SolidWorks was created for use on a personal computer in the Microsoft Windows operating environment [4].

SolidWorks uses the principle of 3D solid and surface parametric design, which allows the designer to create solid parts and assemble assemblies in the form of 3D electronic models, from which 2D drawings and specifications are created in accordance with the requirements of technical documents.

Three-dimensional modeling of products gives a lot of advantages over traditional two-dimensional design, for example, the elimination of errors in the assembly of a product at the design stage, the creation of an electronic model of a control program part for processing on a CNC machine. With the help of SolidWorks software, you can see the future product from all sides in volume and give it a realistic display in accordance with the selected material for a preliminary design assessment.

A SolidWorks 3D part is a combination of 3D primitives. Most of the elements are based on a flat sketch from which a basic 3D object is created.

The 3D model of the reactor core is shown in Figure 5.



FIG 5. 3D model of the reactor core in the SolidWorks software.

## SERPENT SOFTWARE OVERVIEW

SERPENT is a software package (SP) developed at the VTT Technical Research Center of Finland, which is designed to determine the neutron-physical characteristics of systems containing nuclear fissile materials. This SP implements the Monte Carlo method and does not have independent modules, it allows calculations of various characteristics of the system, such as:

• infinite and effective neutron multiplication factors;

• nuclide composition and activity of nuclear fuel;

• residual heat generation;

• multi-group constants;

• activation of structural elements;

• the speed of various reactions.

In the SERPENT SP, you can define the geometry of complex elements, limited by surfaces of no higher than the second order, by combining which it is possible to accurately model the geometry of any two- or three-dimensional system. Also, it is possible to simplify the definition of the most common surfaces, such as spheres, cylinders, hexagonal prisms, etc., as well as the ability to create arrays of elements located on a square / triangular lattice, simulate a spherical load of fuel rods, and set the location of circular clusters characteristic for RBMK and CANDU reactors.

Setting geometry, material composition, calculation parameters and other input data is carried out using one or several text files and does not require a specific sequence of data setting, which makes it easier to use the SERPENT SP.

The computational model represents 1/6 of the reactor fuel assembly due to considerations of the power of the computational complex and is shown in Figure 6. The computational model for computations in SolidWorks is a cover in which the fuel rods with fuel are placed. The peculiarity of the calculation is that you need to put plugs in the holes.

Figures 6 and 7 show the design models made using the SERPENT SP [5].



FIG 6. Design model in SolidWorks software.



FIG 7. Calculation model of the reactor in the SERPENT software package.

## CALCULATION RESULTS

The calculation results in SolidWorks are presented in Figures 8, 9 and 10 with the core height equal to 780 mm. The initial data were taken from Tables 1 and 2, and the energy release of one fuel element was additionally calculated (q = 9985 W/m2). The initial velocity was 0.1 m/s and the initial temperature was 733 0K.

|  |  |  |
| --- | --- | --- |
| 0 | 0 | 0 |
| FIG 8. Pressure distribution at a height of 780 mm. | FIG 9. Temperature distribution at a height of 780 mm. | FiIG 10. Velocity distribution at a height of 780 mm.  |

You can see from these figures that when the coolant speed is 0.1 m / s, the temperatures have increased significantly compared to the previous calculation. The solution to reduce the temperature was to increase the speed of the coolant, started from 0.2 m / s, and still there was a high temperature, the next calculation was already at a speed of 0.3 m / s, and again there was a big warm-up. And only at a speed of 0.4 m / s did we get temperatures approximately the same as shown in Table 2.

# CONCLUSIONS

All tasks were completed, namely:

* analyze the concept of the reactor plant;
* designed a computational model of the reactor;
* checked the required options with SolidWorks.

The advantages of fast reactors are obvious. They are also recognized abroad, so today a new market is opening up for Russia as a world leader in this field, to which it can supply technologies and high-tech equipment. At the same time, there is a risk that China or India will lose in the race of "fast" technologies, therefore, it is necessary to accelerate the development of this direction in the Russian nuclear power industry, relying on the best world practices.

It should be noted that unresolved issues remain, such as the correct assessment of the uneven distribution of the neutron flux over the reactor core and others. These issues will be considered carefully in the next phase of the work.

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