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COMPARATIVE ANALYSIS OF TECHNICAL, ECONOMIC, WEIGHT, AND SIZE CHARACTERISTICS OF HORIZONTAL AND VERTICAL STEAM GENERATORS FOR 1000 MW NUCLEAR POWER PLANT UNITS

A comparative analysis of the technical characteristics and parameters of horizontal and vertical steam generators for reactor installations with water coolant in modern 1000 MW NPP power units has been conducted. It is noted that the peculiarities of the design schemes and constructions of steam generators in NPP power units with water coolants are significantly influenced by the strong dependence between the coolant temperature at the steam generator inlet and the pressure in the reactor circuit. The dependencies of coolant and working substance temperature changes and the amount of heat transferred in steam generator elements have been considered. The t-Q diagram of a steam generator with a water coolant featuring a non-boiling economizer, evaporator, and superheater is presented. The parameters of water coolants and working substances in nuclear power plant steam generators are provided. The t-Q diagram for steam generators without a superheater section is shown, because in modern 1000 MW nuclear power plant units with steam generators which use water as the working substance in the steam-turbine cycle, saturated steam without superheating is used. This is because minor superheating of the steam does not significantly increase the efficiency of the steam-turbine cycle but requires significant complexity in the design of the steam generators. It is noted that the correct choice of their structural schemes is crucial in the creation of steam generators with water coolants. It has been proven that the characteristics determining the structural schemes of steam generators with water coolants are: the scheme of washing the heat exchange surface with the coolant, the form of the heat exchange surface, the layout of steam generator elements, the principle of working substance movement, and others. The technical characteristics of a horizontal-type steam generator for 1000 MW nuclear power plant units are presented, which have proven to be quite effective in operation. However, their designs and characteristics limit the potential for further improvement of the technical and economic indicators of nuclear power plants. It is noted that improving the technical and economic efficiency of 1000 MW nuclear power plant units while simultaneously reducing capital costs for their construction is possible by increasing the unit capacity of powerful vertical-type steam generators installed on them. Compared to horizontal steam generators, vertical steam generators allow for a more rational layout of the primary circuit equipment in the reactor department, thus reducing the volume and cost of construction and installation works. The main dimensions and weight characteristics of possible designs of vertical steam generators for nuclear power plants with VVER-1000 reactors are provided. It is concluded that once-through vertical steam generators with spiral wound heat exchange tubes and water coolant in the first circuit tubes weigh approximately 1.5 to 2 times less than vertical steam generators with natural circulation. It is determined that once-through vertical steam generators with a hydraulic scheme involving the movement of the working substance in the tubes and the water coolant in the inter-tube space significantly lag behind once-through vertical steam generators with water coolant in the tubes in terms of dimensional and weight characteristics and also vertical steam generators with natural circulation in terms of weight. It is concluded that for VVER-1000 nuclear power plants, the most promising of all options are vertical once-through steam generators with the movement of the water coolant in spiral wound heat exchanger tube bundles. These steam generators occupy approximately four times less area in the reactor department than horizontal ones with the same steam production, which significantly reduces construction costs for nuclear power plants.

Keywords: nuclear power plant units, horizontal and vertical type steam generators, technical and weight characteristics, steam parameters, water coolants, working substances.

Introduction

Reliability, technological safety, energy efficiency, and energy and resource conservation are the primary strategic components of modern nuclear energy. A systematic analysis of technological processes, designs, and technical characteristics of current and future nuclear power reactors and steam generators of various types is crucial for meeting the high functional requirements of modern and future nuclear power plant units, which must comply with stringent safety criteria. Steam production at nuclear power plants using water coolants occurs in steam generators. In addition to the thermophysical and physicochemical processes typical of conventional heat exchange units, neutron-physical processes also occur in steam generators, defining their uniqueness and classifying them into a special category. Currently, both horizontal and vertical designs of single-shell steam generators are successfully used in modern nuclear power plants with water coolants in different countries. It is known that "Energoatom" plans to complete the construction of the 3rd and 4th power units with a capacity of 1000 MW using VVER reactors and horizontal steam generators at the Khmelnytskyi Nuclear Power Plant, and to construct four new 1000 MW power units using Westinghouse technology (AP 1000) with vertical steam generators at the Khmelnytskyi and South Ukraine Nuclear Power Plants. Consequently, a comparative analysis of the technical characteristics and parameters of horizontal and vertical steam generators for reactor installations with water coolants is relevant.

Purpose of the Work

The purpose of this article is to conduct a comparative analysis of the technical, economic, weight and size characteristics and parameters of horizontal and vertical steam generators for pressurized water reactor installations of nuclear power plant units with a capacity of 1000 MW, and to evaluate their impact on mass-dimensional indicators, installation features, and the reliability of power unit.

Main Material Presentation

The strong correlation between the inlet temperature of water coolants t_1' and their pressure P_1 in the reactor circuit significantly influences the design and construction of steam generators for 1000 MW nuclear power plant units using water coolants (an increase in pressure sharply increases the temperature and vice versa). For technical and economic reasons, the pressure of water coolants in reactor circuits is typically limited to 17-20 MPa [1–3].

In water-cooled reactors, boiling of water in the circuit (except for surface boiling) is not permitted. Thus, there is a certain underheating of the water coolant δt_{unh} to the saturation temperature t_{S1} corresponding to the pressure P_1 at the reactor outlet [4]. The temperature of the water coolant at the exit from the reactor (at the entrance to the steam generator) $t'_1 = t_{S1} - \delta t_{unh}$.The maximum possible temperature of the working fluid (steam) t''_2 at the steam generator outlet is determined by the temperature t'_1 and the temperature head Δt_{in} at the coolant inlet to the steam generator $t''_2 = t'_1 - \Delta t_{in}$. An important characteristic of the working fluid is the saturation temperature t_{S2} at a certain pressure P_2 in the evaporator, one of the steam generator elements. This value is determined by the coolant temperature t''_{1ev} and the temperature head Δt^{out}_{ev} at the evaporator outlet. Generally, the coolant cools sequentially through the superheater, evaporator, and economizer to reach the final temperature at the outlet t''_1, respectively by the values δt_{1sh} , δt_{1ev} , δt_{1ec} . The temperature of the heat carrier at the outlet of the evaporator $t''_{1ev} = t'_1 - \delta t_{1sh} - \delta t_{1ev}$, and the saturation temperature of the working fluid in it $t_{S2}(P_2) = t''_{1ev} - \delta t^{out}_{ev}$.

The change in coolant and working fluid temperatures and the amount of heat transferred in the steam generator elements is depicted in the t-Q diagram. The diagram plots characteristic temperatures for each steam generator element on the ordinate axis and the amount of transferred heat in the economizer Q_{ec}, evaporator Q_{ev}, and superheater Q_{sh} on the abscissa axis (Fig. 1).

Table 1 presents some values of water coolant and working fluid parameters in steam generators of nuclear power plants, obtained using a mathematical model of steam generators [4, 5], indicating that with the saturated steam pressure values generated 3,7÷8,8 МПа, the possible superheat is small, around 30°C.

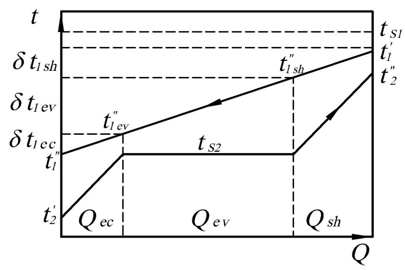

Fig 1 – *t-Q-* diagram of a steam generator with a water coolant with an economizer non-boiling type, evaporator and superheater

Table 1 – The value of the parameters of water coolants and the working substance in steam generators of nuclear power plants

Parameter	Value		
The pressure of the coolant at the entrance to the steam generator, MPa	10,0	14,0	20,0
The temperature of the coolant at the entrance to the steam generator, °C	284	310	340
Coolant temperature at the outlet of the evaporator °C	264	290	320
Temperature pressure at the outlet of the evaporator °C	20	20	20
Saturated vapor pressure, MPa	3,7	5,6	8,8
Temperature of saturated steam, °C	244	270	300
Possible output steam temperature from the steam generator, °C	274	300	330
Overheating of the steam is possible, °C	30	30	30

Significant superheating of steam in steam generators with water coolants can be achieved at lower saturated steam pressures, which is not economically efficient for the nuclear power plant's steam-turbine cycle. Small superheating of steam does not significantly increase the steam-turbine cycle's efficiency but requires substantial complication of steam generator designs. Therefore, in modern 1000 MW nuclear power plant units with steam generators using water coolants, saturated steam without superheating is used as the working fluid of the steam-turbine cycle, and the t-Q diagram for these steam generators does not have a superheater (Fig. 2).

This diagram $Q_{nb.ec}$ and $Q_{b.ec}$ shows the quantities of heat transferred in the non-boiling and boiling sections of the economizer.

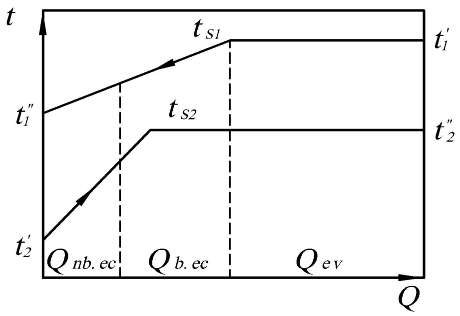

Fig 2 – *t-Q-* diagram of a steam generator with a water coolant of modern nuclear power plants

Correct selection of the design schemes of steam generators with water coolants is crucial in their creation.

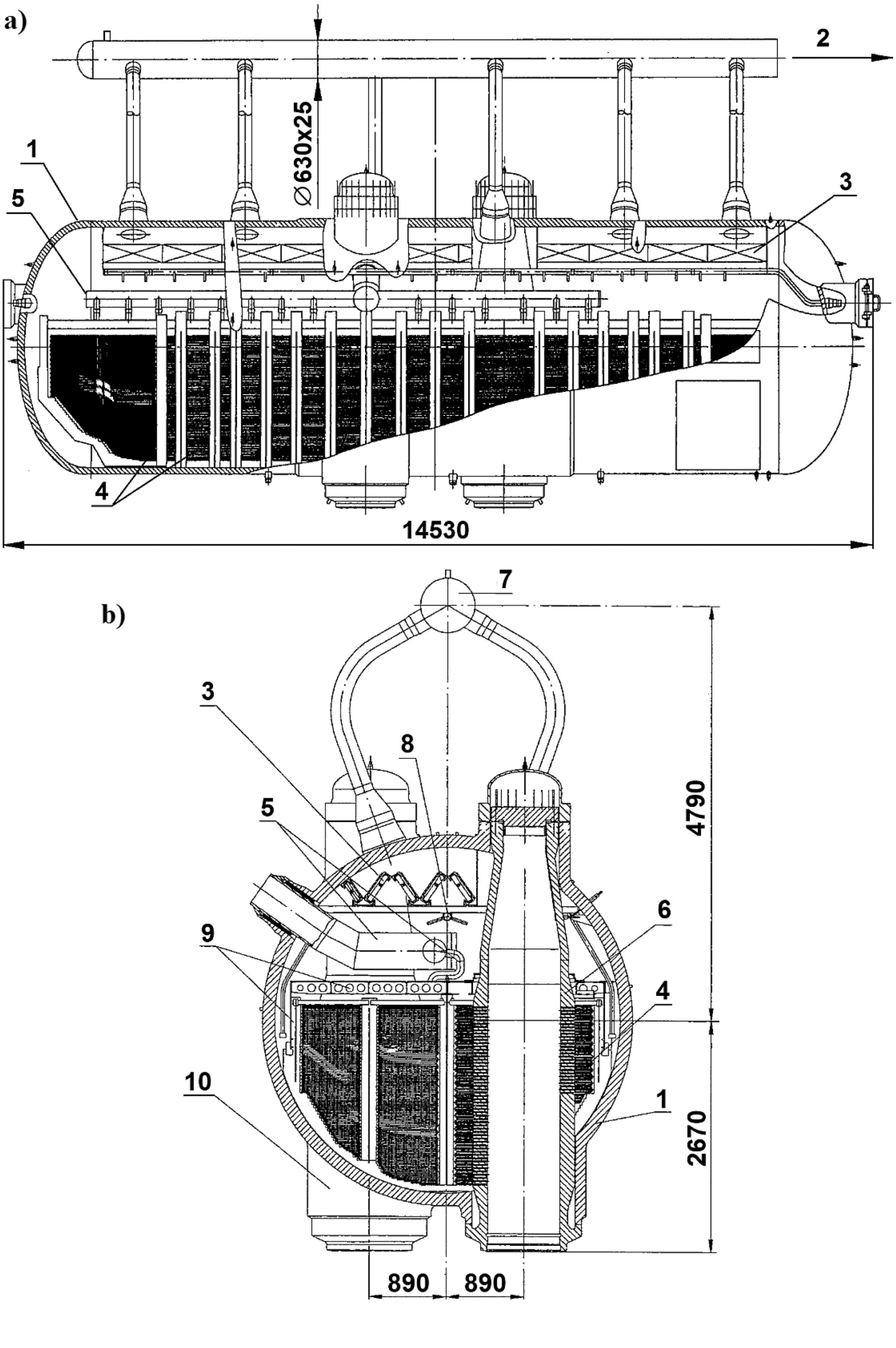

Fig. 3 – Horizontal type steam generator for VVER -1000:
a) – longitudinal section,
b) – cross-section,
1 – vessel; 2 – steam outlet; 3 – chevrons of the separator unit; 4 – heat exchange tube bundle; 5 – supply of feed water; 6 – outlet header; 7 – steam header; 8 – supply of auxiliary feed water; 9 – tube bundle support; 10 – inlet header [7]

Table 2 – Technical characteristics of the PGV-1000M steam generator [4, 5, 7]

Parameter	Value and permissible deviation
Thermal capacity, MW	750+53
Steam productivity, t/h	1470+103
Pressure of generated steam, MPa	6,3±0,2
Temperature of generated steam, °C	278±2
Feed water temperature, ºC	220±5
The temperature of the feed water when the high-pressure heaters are turned off, ºC	164±4
Emergency feed water temperature, ºC	5÷40
The pressure of the coolant of the first circuit at the entrance to the steam generator, MPa	15,7±0,3
The temperature of the coolant of the first circuit, C: - at the entrance - at the exit	 320±3,5 289±2
Nominal boiler water level, mm: - by a one-meter spirit level - on a four-meter level - on the "cold" end of the steam generator - on the "hot" end of the steam generator	 220÷320 270÷320 2250±50 2100±50
Resistance of the steam generator on the first circuit during the operation of the four main circulation pumps, MPa	0,125
Steam generator resistance along the steam-water path at nominal steam productivity, MPa	0,110
Humidity of steam at the exit from the steam generator, %	0,2
Consumption of flushing water, t/h: - continuous blowing - periodic blowing	 7,5 14,5
Maximum calculated pressure, MPa: - steam heat carrier, - which is generated	 17,6 7,8
Maximum calculated temperature °C: - steam heat carrier, - which is generated	 350 300
The wall temperature of the elements of the first and second circuits during hydrotests is not less than ºC	70
Pressure hydrotests for strength: - along the first contour, MPa - along the second contour, MPa	 24,5±0,2 10,8±0,1
Steam generator capacity, м3 : - along the first contour - along the second contour	 23,4 124,6
Weight of dry steam generator, kg	320000

Characteristics determining the design schemes include the coolant flow scheme over the heat exchange surface, the shape of the heat exchange surface, the layout of steam generator elements, and the principle of working fluid movement. During the design process, the selection and justification of each characteristic are consistently carried out [4]. Horizontal steam generators installed at nuclear power plants with VVER-1000 (Fig.3) reactors have proven to be very effective [6, 7]. Their technical characteristics are presented in Table 2. However, their designs and characteristics limit the possibilities for further improving the technical and economic indicators of nuclear power.

One way to enhance the technical and economic efficiency of power plants is to increase the unit capacity of the equipment installed, including steam generators, while reducing capital construction costs [6]. For 1000 MW nuclear power plant units with VVER-type reactors, this can be achieved by using powerful vertical-type steam generators. Compared to horizontal steam generators, vertical ones allow more efficient arrangement of first-circuit equipment in the reactor section, thereby reducing construction and installation costs [6, 7].

The main dimensions and mass characteristics of possible vertical steam generator designs for 1000 MW nuclear power plant units with VVER-type reactors (Fig.4) are presented in Table 3. The analysis of this data allows for certain conclusions.

The main drawbacks of powerful vertical single-shell steam generators with water coolants and natural circulation are complex construction, large masses, and significant overall.

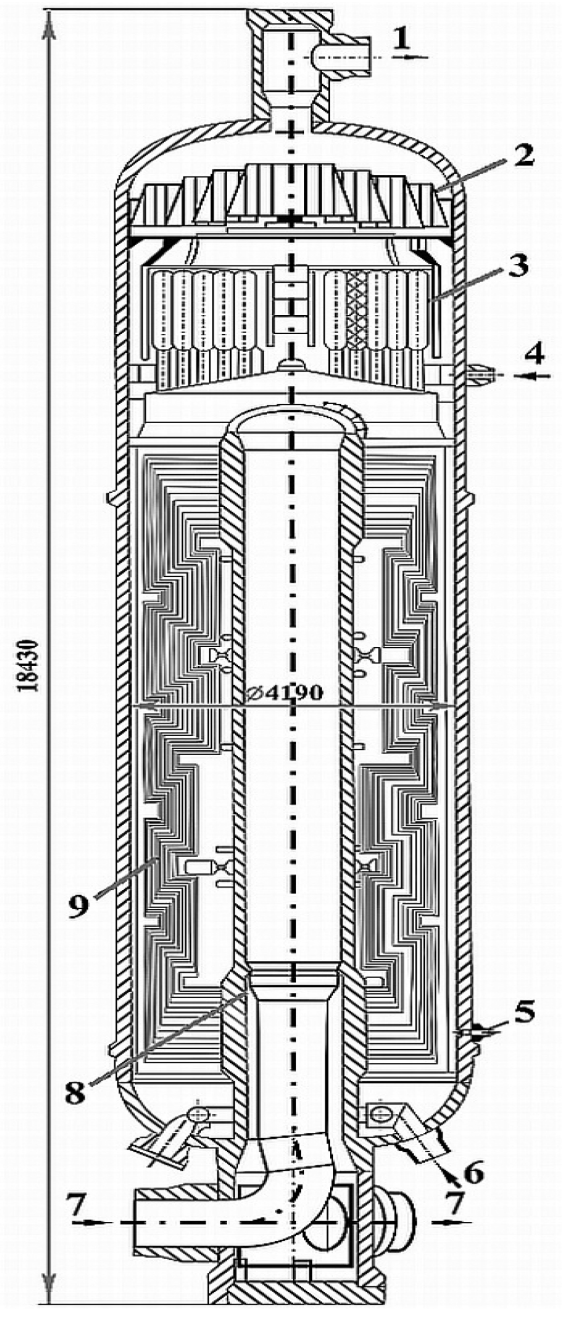

Fig. 4 – Steam generator with header tube fixation
1 – steam; 2, 3 – chevron and swirl-vane separators; 4 – supply of auxiliary feed water;
5 – continuous blow down nozzle; 6 – feed water; 7 – coolant; 8 – header; 9 – tube bundle platen [7]

These drawbacks are largely eliminated in vertical once-through steam generators with water coolants (Fig.5). The transition to a once-through scheme and the abandonment of separation devices significantly simplifies the design, improves mass and dimensional characteristics, and facilitates transportation.

Vertical once-through steam generators with helically wound heat exchange tubes and water coolants in the first circuit tubes weigh approximately 1.5–2 times less than vertical steam generators with natural circulation.

Table 3 – The main dimensions and mass-dimensional characteristics of vertical steam generators with a water coolant for nuclear power plants with reactors of the VVER-1000 type [7]

Parameter	Steam generators with natural circulation		Once-through steam generators		
	single hull	with a removable separator	with coolant in spirally twisted tubes	with the coolant in the pipe space	with coolant in straight tubes
Standard size of tubes of heat exchange surfaces, mm	12×1,2	12×1,2	12×1,2	12×1,2	14×1,4
The total number of tubes of heat exchange surfaces, mm	34950	33120	31100	28400	20800
Internal diameter of the steam generator body, m	5,85	$3{,}910^3$	3,6	3,75	3,8
Weight of the steam generator, kg	900000	900000	450000	930000	500000
Specific mass of the steam generator (per unit of electric power produced by the power unit), kg/MW	$1{,}80\cdot10^3$	$1{,}82\cdot10^3$	$0{,}9\cdot10^3$	$1{,}86\cdot10^3$	1,00·

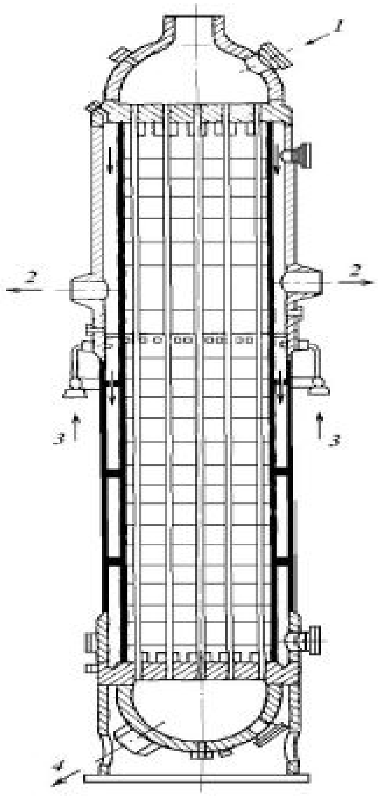

Fig. 5 – Once-through vertical steam generator with water coolant in direct heat exchange tubes:
1 – coolant input; 2 – outlet steam; 3 – feed water input;
4 – coolant outlet [7]

In vertical once-through steam generators with helically wound heat exchange tubes, the heat exchange surface is made of two helically wound bundles connected to the central collector.

The bundles are included in parallel for the water coolant flow inside the tubes and in series for the working fluid movement, which flows from bottom to top in the inter-tube space. A special system of internal collector devices organizes the coolant movement so that heat exchange in both bundles follows a counterflow scheme. To ensure equal diameters of the helically wound heat exchange bundles, the coolant flow rates in the upper and lower bundles are the same, each accounting for 50% of the total flow rate. Vertical once-through steam generators with the working fluid moving in the tubes and the water coolant in the inter-tube space are significantly inferior in overall and mass characteristics to those with water coolants in the tubes, and in mass also to vertical steam generators with natural circulation.

To ensure the safe operation of 1000 MW nuclear power plant units with vertical steam generators where water coolants move in the inter-tube space, systematic control over the condition of the internal surfaces of the steam generator shells is necessary. The design of the flange joint, allowing for disassembly for systematic control, complicates the steam generators' design and reduces their reliability. Additionally, lifting the covers of steam generator shells with collectors and heat exchange bundles requires a crane with a lifting capacity of at least 500 tons and sufficient space in the main building, increasing both the building height and installation costs.

Vertical once-through steam generators with the water coolant moving in straight heat exchange tubes and flat tube sheets are comparable in overall dimensions and height to those with helically wound tubes, but they have a larger internal diameter of the shell. The principal design scheme of these steam generators is mainly determined by the chosen method of self-compensation for thermal expansions of the tubes and shell.

A comparative analysis of the main mass and dimensional characteristics of possible vertical steam generator designs for VVER-1000 nuclear power plants, considering manufacturing, installation, and operating conditions, indicates that the most promising option is the vertical once-through steam generator with helically wound heat exchange tube bundles.

In fig. 6 presents the design of a vertical steam generator for a 1000 MW NPP power unit (AP 1000) using Westinghouse technology, which in its technical characteristics is close to the design of a vertical steam generator for VVER-1000 and has the same advantages compared to a horizontal steam generator.

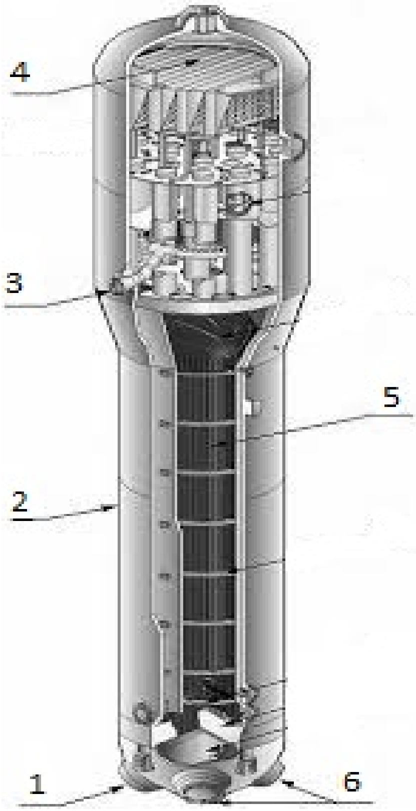

Fig. 6 – Steam generator for nuclear power plants with AP-1000:
1 – coolant input collector; 2 – frame; 3 -– feed water distribution collector; 4 – separator; 5 – heat transfer surface;
6 – coolant outlet collector [7].

Conclusions

Currently, in various countries, both horizontal and vertical designs of single-shell steam generators are successfully used in modern 1000 MW nuclear power plant units with water coolants. Both designs are quite similar in their technical characteristics and reliability indicators. However, vertical steam generators occupy approximately four times less area in the reactor section compared to horizontal ones with the same steam production capacity, which significantly reduces construction costs for nuclear power plants.

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ПОРІВНЯЛЬНИЙ АНАЛІЗ ТЕХНІКО-ЕКОНОМІЧНИХ І МАСОГАБАРИТНИХ ХАРАКТЕРИСТИК ГОРИЗОНТАЛЬНИХ ТА ВЕРТИКАЛЬНИХ ПАРОГЕНЕРАТОРІВ ЕНЕРГОБЛОКІВ АЕС ПОТУЖНІСТЮ 1000 МВТ

Виконано порівняльний аналіз технічних характеристик і параметрів горизонтальних і вертикальних парогенераторів для реакторних установок з водним теплоносієм сучасних енергоблоків АЕС потужністю 1000 МВт. Зазначено, що на особливості конструктивних схем і конструкцій парогенераторів енергоблоків АЕС з водними теплоносіями великий вплив робить існування сильної залежності між температурою цих теплоносіїв на вході в парогенератор та їх тиском в контурі реактора. Відмічено, що конструкції і характеристики горизонтальних парогенераторів енергоблоків АЕС потужністю 1000 МВт обмежують можливості подальшого підвищення техніко-економічних показників. Визначено, що прямотечійні вертикальні парогенератори з гідравлічною схемою, що передбачає рух робочої речовини в трубках, а водного теплоносія – в міжтрубному просторі, за своїми габаритними і масовими характеристиками значно поступаються прямотечійним вертикальним парогенераторам з водним теплоносієм в трубках, а за масою – також і вертикальним парогенераторам з природною циркуляцією. Визначено, що для АЕС з ВВЕР-1000 найбільш перспективними зі всіх варіантів є вертикальні прямотечійні парогенератори з рухом водного теплоносія в спіральновитих теплообмінних трубних пучках. Ці парогенератори займають площу в реакторному відділенні приблизно в 4 рази меншу, ніж горизонтальні такої ж паропродуктивності, що істотно знижує витрати на будівництво АЕС.

Ключові слова: енергоблоки АЕС, парогенератори горизонтального і вертикального типів, технічні і масогабаритні характеристики, параметри пари, водні теплоносії, робочі речовини.