THERMAL-HYDRAULIC CALCULATIONS FOR THE NEW INTEGRAL SMALL MODULAR REACTOR VVER-I WITH NATURAL CIRCULATION IN PRIMARY CIRCUIT

M.M. BEDRETDINOV
OKB Gidropress JSC, 21, Ordzhonikidze st., Podolsk, Moscow Region, 142103
Email: bedretdinov2013@yandex.ru

O.E. STEPANOV
OKB Gidropress JSC, 21, Ordzhonikidze st., Podolsk, Moscow Region, 142103
Email: olegmpei@yandex.ru

D.N. MOYSIN
OKB Gidropress JSC, 21, Ordzhonikidze st., Podolsk, Moscow Region, 142103
Email: daniilmoysin@mail.ru

M.A. BYKOV
OKB Gidropress JSC, 21, Ordzhonikidze st., Podolsk, Moscow Region, 142103
Email: bykov@grpress.podolsk.ru

Abstract

Initial constructive development of new small modular pressurized water reactor VVER-I with natural convection in primary circuit has been started in 2022 in JSC «GIDROPRESS». VVER-I thermohydrodynamics calculation results which were made using one-dimensional code KORSAR/GP are presented within this work. Results of provided calculations for normal operation mode of integral reactor VVER-I confirm results which were previously obtained using analytical methodology for reactor and steam generator general sizes estimation. Stability of circulation and coolant parameters of primary circuit are confirmed via the calculations. During normal operation the reactor core is cooled by means of natural circulation in non-boiling mode. For basic thermal power of 250 MW and 13.4 m height of reactor vessel temperature in outlet of the core is about 310 ºC, coolant mass flow through the core equals to 820 kg/s and the temperature of overheated steam in the outlet of steam generator is about 300 ºC. It is shown that the reactor has substantial potential for thermal power increasing right up to 400 MW without significant design changes.

1. INTRODUCTION

Thermohydrodynamics calculation results of new integral reactor VVER-I obtained using one-dimensional code KORSAR/GP [1] are presented within this work.

It is shown that developed design of reactor has a sufficient potential for thermal power increasing without significant constructive changes. Reactor design is provided with passive safety systems which can reliably transfer heat from the core during beyond design basis accidents with long-term station blackout and personnel inaction.

Results of this work can be used for the new VVER type reactor with natural coolant circulaton in primary circuit and passive safety systems design development.

2. VVER-I DESIGN DESCRIPTION

VVER-I is water-cooled reactor with natural circulation in primary circuit. Steam generator is integrated into reactor vessel and consists of 7400 vertical heat exchange U-tubes [2, 3]. Feed water is supplied into steam generator tubes and coolant of primary circuit flows through the area between steam generator tubes. The core consists of 91 shortened fuel assemblies with draft block under heating part which allows to ensure non-boiling circulation mode for the most powerful fuel assemblies.

VVER-I thermal power is chosen at level 250 MW with opportunity to force it till 400 MW without reactor significant constructive changes. Mass flow of feed water in steam generator is taken equal to 114 kg/s with temperature 184 ºC. Pressure amounts to 13 MPa and 3 MPa in the primary circuit and in the secondary circuit respectively. The temperature of overheated steam in the outlet of the steam generator is about 290 ºC. Pressurizer is moved outside the reactor vessel.

VVER- I reactor control system is close to control systems of operated nowadays high powered VVER reactors. Safety conception includes 4 levels of defense in-depth protection and 1-3 levels consists only on passive systems.

3. CALCULATION MODEL OF VVER-I REACTOR

Calculation model of VVER-I developed using one-dimensional code KORSAR/GP and the design of reactor [2] are shown in Fig. 1. The procedure for analyzing unsteady thermal-hydraulic processes in the KORSAR/GP code [1] is based on a fully (thermally and mechanically) nonequilibrium two-fluid model in the 1-D approximation. In the finite-difference approximation of the conservation equations, we use the method of control volumes and semi-implicit numerical scheme in time with automatic selection of the integration step subject to ensuring the preset accuracy of calculations.

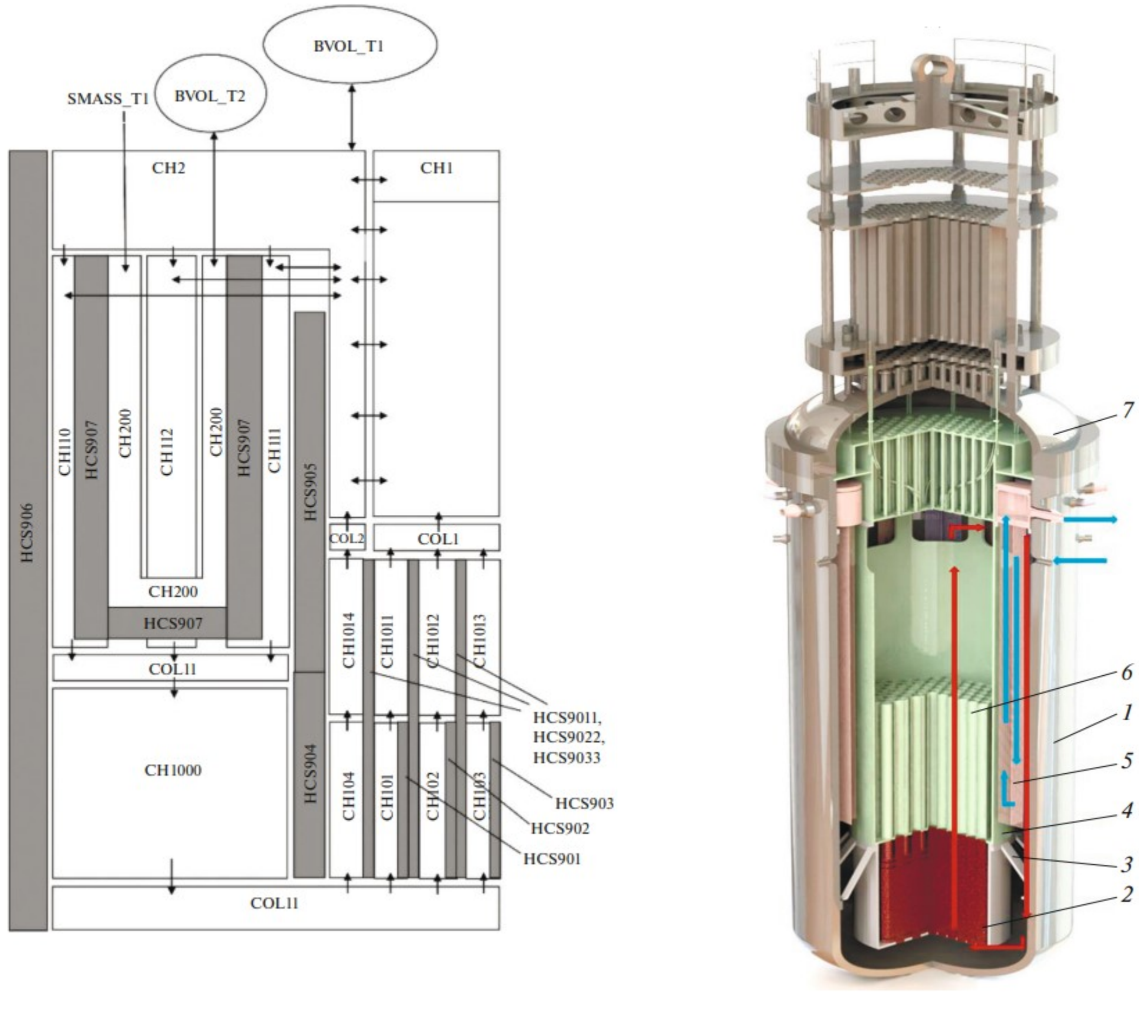

FIG.1. Calculation model (left) and design of reactor (right): 1- reactor pressure vessel; 2 - core; 3 - lower in-pile truss; 4 - barrel; 5- steam generator; 6 - draft tube unit; 7 - top head.

The flow part of the core is modeled by parallel channels CH101-103, heat sources in form of fuel assemblies are modeled by heat elements HCS901-903. Draft tubes under heating part of fuel assemblies are presented as heat structures HCS9011, HCS 9022, HCS 9033. Channel CH104 simulates bypass of the core and channel CH1014 simulates area under bypass of the core. The flow part of draft block is presented by channels CH1011-1013. Heat structures HCS904 and HCS905 modulate core enclosure and in-hull shaft respectively. Reactor vessel is presented by heat structure HCS906. Chambers COL1, COL2 and channels CH1, CH2 simulate the inner area of the reactor in-shaft. In primary circuit channel CH110 modulate the flow area of steam generator cold part (where the feed water is supplied to the secondary circuit), channel CH111 modulate the flow area of steam generator hot part (where overheated steam is generated in the secondary circuit), channel CH112 modulate the flow area between channels CH110 and CH111. Chamber COL11, channel CH1000 and chamber COL111 modulate the flow part of the reactor between steam generator and inlet of the core. Channel CH200 simulates the tract of steam generator. Heat element HCS907 modulates tube bundle of steam generator. The pressurizer is presented by element BVOL_T1, and the boundary condition for steam generator is presented by element BVOL_T2. The feed water supply to steam generator is modulated by element SMASS_T1.

4. CALCULATION RESULTS

When the reactor thermal power is equal to 250 MW coolant temperature amounts to 262 °C at the inlet and to 310 °C at the outlet of the reactor core. The temperature of overheated steam at the outlet of steam generator reaches 295 °C. Coolant mass flow in primary circuit stabilizes at value 993 kg/s. Average velocity of the coolant in the reactor core is about 1 m/s.

Calculation results obtained using KORSAR/GP code compared with analythical method are presented in Table 1.

TABLE 1: CALCULATION RESULTS

Parameter	Calculation method	
	Analythical method [3]	KORSAR/GP code
Primary circuit mass flow, kg/s	922	993
Outlet temperature of the core, °C	310	310
Inlet temperature of the core, °C	258	262
Overheated steam temperature, °C	285	295

Earlier obtained results using analythical method [3] show some difference with KORSAR/GP code results because of conservative approach of analythical method with using safety factors which lead to reduction of mass flow in primary circuit, increasing temperature difference at the inlet and at the outlet of the reactor core and also to decreasing overheated steam temperature.

Coolant velocity distribution results in steam generator interpipe area of the primary circuit are presented in Fig. 2. Obtained results indicate non-uniform coolant velocity distribution between cold and hot parts of steam generator. This effect is turned out because overheating section of steam generator begins from the steam generator pipes bending. That is why most part of mass flow goes through the cold part of steam generator.

Cross-links which connect cold and hot parts of steam generator in primary circuit were removed from calculation model after above mentioned effect was detected. This decision allowed to make coolant velocity distribution much more uniform (Fig. 3).

Steam volume fraction distribution along steam generator tubes is presented in Fig. 4. Removement of cross-links allows to shorten overheating section of steam generator, makes temperature at the inlet of the core lower (249 °C) and also leads to temperature of overheated steam increasing till 299 ºC.

In the construction this effect can be reached by each part of steam generator (cold and hot) separation from each other. One of possible variation of packing tube bundle of each part of steam generator in special cases is demonstrated in Fig. 5.

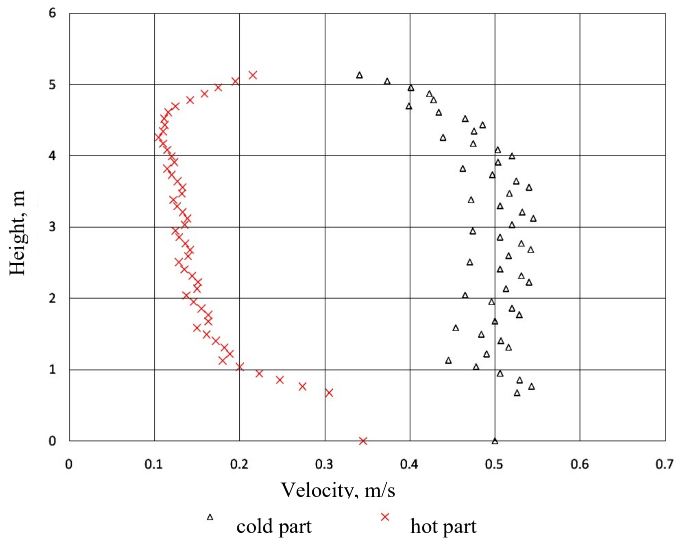

FIG. 2. Interpipe area coolant velocity distribution

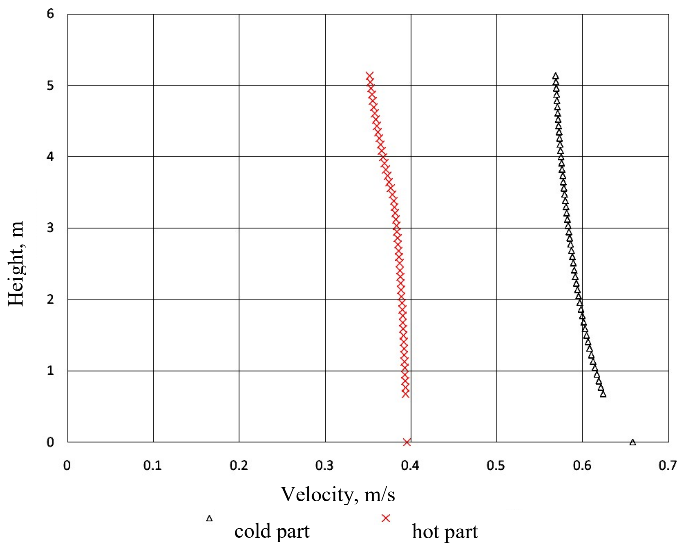

FIG.3. Interpipe area coolant velocity distribution

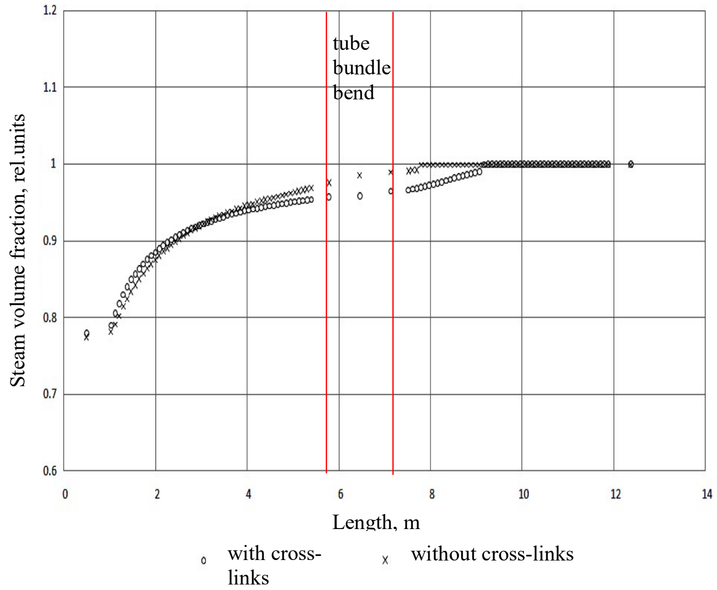

FIG. 4. Steam volume fraction distribution

To demonstrate that developed design of reactor has a sufficient potential for thermal power increasing without significant constructive changes two additional calculations fow thermal power 300 and 400 MW were made using KORSAR/GP code. Constructive changes were related to steam generator tube bundle increasing and consequently the height of all circulation contour. Steam generator power increasing was provided by proportional feed water mass flow increasing. Obtained calculation results are presented in Table 2. The results show that it is necessary to increase the height of reactor till 15.2 m to force thermal power to 400 MW.

Heat flux from fuel rods cladding is equal to 0.5 MW/m^2 when thermal power of the core amounts to 400 MW. This value is much lower than critical heat flux for pressure 13 MPa which reaches 1.3 MW/m^2.

Position of VVER-I among similar small modular reactors with natural circulation of coolant in primary circuit from the point of view of reactor height dependence from reactor thermal power is shown in Fig. 6. The height of small modular reactors begins from 4 m for relatively small reactor ELENA and reaches 17 m for Japanese reactor project IMR thermal power of which is close to thermal power of VVER-440. With the exception of NuScale reactor which stands out possibly because of safety hull application, VVER-I with nominal thermal power 250 MW and also with forced thermal power 300 and 400 MW quite well fits worldwide trend of such type reactors development.

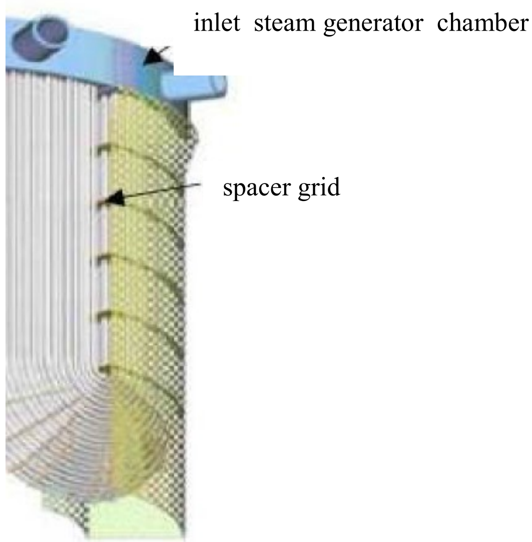

FIG. 5. Steam generator tube bundle case

TABLE 2. CALCULATION RESULTS

Parameter	Thermal power, MW		
	250	300	400
Reactor height, m	13.4	14.4	15.9
Feed water mass flow, kg/s	114	137	185
Primary circuit mass flow, kg/s	820	905	1064
Outlet temperature of the core, °C	309	310	316
Inlet temperature of the core, °C	249	247	244
Overheated steam temperature, °C	299	297	291

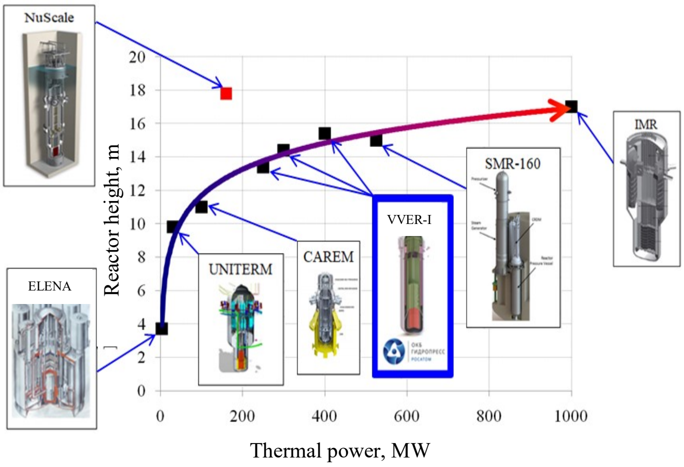

FIG. 6. Reactor height versus thermal power

5. CONCLUSION

Calculation results obtained using KORSAR/GP code confirm correctness of earlier obtained results using analythical method [3].

The reliability of core cooling is substantiated, a non-boiling circulation mode in the most powered fuel assemblies is achieved due to draft block under the reactor core implementation.

Packing each part of steam generator tube bundle in special cases allows to make coolant velocity distribution in steam generator interpipe area more uniform. Due to this solution lower temperature at the inlet of the reactor core and higher temperature of overheated steam at the outlet of steam generator can be reached.

It is shown that developed design of reactor has a sufficient potential for thermal power increasing without significant constructive changes till 400 MW by increasing height of circulation contour on 2.5 meter.

The project is developed in compliance with relevant safety assurance standards, both Russian requirements and international guides, in the field of nuclear power use. Only passive safety systems and safety elements are used to manage design basis and beyond design basis accidents. The assumed structure of integration of safety systems is subject for further development at subsequent design stages.

REFERENCES

[1] YU. G. DRAGUNOV, M.A. BYKOV, V.A. VASILENKO, YU.A. MIGROV. Experience with introduction and development of the KORSAR computer code for substantiating the safety of NPSs with type VVER reactors / Thermal Engineering. 2006.

[2] https://www.atomic-energy.ru/news/2023/02/09/132648.

[3] M.M. BEDRETDINOV, M.A. BYKOV, O.E. STEPANOV, R.M. SLEDKOV. Thermal-Hydraulic Calculations for the New Project of Small Modular Reactor VVER-I, «XXXIX Siberian thermophysical workshop», 28 – 31.08.2023, Novosibirsk