CROSS-VERIFICATION OF ONE- AND THREE-DIMENSIONAL MODELS FOR VVER STEAM GENERATOR

K.S. Dolganov, A.V. Shishov

Nuclear Safety Institute of the Russian Academy of Science (IBRAE RAS), Moscow, Russia

Abstract

Within the framework of calculations of VVER accident scenarios several steam generator (SG) models for the SOCRAT integral code and the STAR CD code were developed at Nuclear Safety Institute of the Russian Academy of Science (IBRAE RAS). Cross-verification of the coolant flow over SG primary circuit was performed between these codes under rated operating conditions. The agreement of coolant velocities in tubes may be deemed as good, taking into account a large difference in the degree of nodalization of heat-exchanging tubes and collectors. Velocity field found by STAR CD calculations makes it possible to reduce the uncertainties of SG model built in the SOCRAT code.

The developed SG model for the SOCRAT code may be considered as acceptable for calculations of rated conditions. With respect to accident scenarios, the validation of one-dimensional models requires, in the strict sense, additional three-dimensional calculations of two-phase mixture and superheated steam flow, coupled to heat exchange with secondary circuit water. Such a development of three-dimensional SG model is planned for the next phases of work.

1. INTRODUCTION

The acting NPP safety standards postulate the need for reducing the conservatism in calculation methods, especially while analyzing beyond-design-basis accidents (BDBA). This requirement implies the need for developing finer models, which take into account both the real geometry of structures and the related specific processes.

The system one-dimensional codes (such as RELAP, ATHLET or CATHARE) are traditionally used to model the reactor transients and accidents. But there are issues where their capabilities are insufficient for an adequate simulation of all processes in a specific geometry of nuclear installations. For instance, the spatial distribution of parameters is substantially three-dimensional in some structures (such as lower and upper plenum, reactor vessel downcomer, core, containment volume etc.), which makes it necessary to use three-dimensional fine mesh models. Such models are successfully implemented in commercial CFD codes, which became a reliable tool of analysis in different industries. But as they are very resource expensive from the computational standpoint, their use for system analysis of nuclear installations is limited.

System codes are much more suitable for integral modeling of nuclear installation transients, and they still remain a basic tool for nuclear safety analysis. The use of CFD codes is recommended when a more detailed analysis of processes and events is needed in some local area, or when the investigated scenario is impossible for analysis with system code, due to its specific limits. The coupled use of system and CFD codes would possibly solve the problem, reducing conservatism and uncertainties of calculations and keeping computational costs at reasonable levels. However, the coupled use of system and CFD codes requires the solution of some specific problems, such as codes interfacing (Mahaffy, 2007). To give an example of R&D work in that domain, a project NEPTUNE developed by CEA/EDF aims at coupling different scale (system, component and CFD) thermal-hydraulics models in one multi-scale software platform for advanced two-phase flow calculations (Guelfi, 2005, 2007; Bestion, 2005). In particular, the investigations concern the coupling techniques and functionality.

An alternative way of system and CFD coupled use is cross-verification. The international experience of system codes use for analysis of different processes typical for nuclear installations reveals that the accuracy of calculations heavily depends on nodalization scheme and finally, on user skills (Allelein et al., 2007). Yet the direct experiments for complex geometry objects are often impossible, particularly as applied to accidents. That makes the verification studies very difficult and does not allow assessing if the model built in system code is correct. The cross-verification of CFD and system codes helps examining if the model is correctly built.

The highest possible accuracy of predicted physical-parameter fields in simulated components allows identifying the field peculiarities and taking them into account in one-dimensional models. Thus, the coupled use of one-dimensional system and three-dimensional CFD codes in form of cross-verification refines the system models (nodalization schemes) and keeps acceptable computational speed.

One should notice that the CFD models, when applied to specific problems of nuclear safety, still need a significant improvement. The modeling of such processes as water-steam mixture flow, heat and mass transfer between phases, noncondensable gases and aerosols behavior in coolant, counter-current flow etc. (Scheuerer, 2005) is not adequate enough. The main attention should be paid to modeling of components important for safety. Steam generator is one of these components, and all the mentioned processes are typical for its operation in rated conditions and accidents.
This article presents an example of a coupled use of system and CDF codes. The considered issue is the one-phase coolant flow in collectors and heat-exchanging tubes of VVER SG (PGV-1000 MKP). So far, only rated conditions of SG operation are considered, and cross-verification of accident scenarios is planned for the next phases of work.

2. SG ROLE IN SEVERE ACCIDENTS

When studying severe accidents progression, the main attention is given to processes of hydrogen release in containment volume, as well as energy and mass release from reactor vessel in corium catcher. These processes affect directly the integrity of the containment, which is the last barrier for an atmospheric release of radioactive fission products. Still a possibility remains of fission products release to the environment even when the last barrier integrity is preserved. The surfaces of SG heat-exchanging tubes and collectors form the primary circuit boundary. In case of multiple tubes rupture radioactive fission products are conducted by the coolant in the inter-tube volume of SG and are transported with steam through the secondary circuit. As the secondary circuit piping is located outside the containment, its non-hermeticity or possible additional failures lead to radioactive release to the environment. Thus, the radioactive materials can bypass the localizing safety systems.
Therefore the realistic estimate of SG tubes and collectors integrity in case of BDBA represents a substantial part of safety analysis and is closely tied with research on the possible mechanisms of the third and forth barriers damage. A correct estimate of SG tubes integrity could not be done without adequate simulation of processes and events in SG, particularly when it gravely affects the accident progression. There are scenarios where only balance models are sufficient, while other scenarios require more detailed models. For instance, the most frequent modeling of VVER-type SG tube bundle by three height layers turns out to be too coarse in case of accidents with SG rapid level drop. In that case a finer bundle partitioning by height is needed. The tubes nodalization is also required for a correct simulation of radioactive aerosols and fission products deposition onto heat-exchanging structures. The fine nodalization allows a better estimate of mass deposited on surface unit and hence, of value of the heat flux on the given area. Besides, the coolant flow reversal in some tubes of the bundle changes the distribution of fission products along tubes.
One should also emphasize that adequate modeling of the processes in SG secondary circuit is a rather complicated problem: this circuit has a complex flow regime of steam-water mixture, with multiple recirculation loops depending on different parameters. Yet it strongly affects the heat removal from the primary circuit. Under accident conditions not only a correct estimate of heat exchange between the circuits is required, but, especially, that of absolute values of physical parameters – mainly of pressure determining the operation of passive safety systems.

3. SG MODELING

The thermohydraulic processes in VVER-type SG are analyzed in Russia using both one-dimensional system codes (RELAP, CATHARE, ATHLET, MELCOR etc.) and some three-dimensional codes (Trunov, 2003). A wide development of finite-volume models in last 20 years made it possible to use commercial CFD codes in nuclear safety, but still they are unable to replace the system codes.
To analyze the progression of severe accidents in VVER reactors, a best-estimate program complex SOCRAT is used (System Of Codes for Realistic Assessment of severe accidents, former RATEG/SVECHA – see details in Vasiliev, 2000; Bezlepkin et al, 2001; Bezlepkin, 2003). That complex adopts both the advantages of system approach to nuclear safety analysis and the last achievements in

simulation of separate components and processes. SOCRAT allows a coupled modeling of a wide range of thermohydraulic, physicochemical and thermomechanical effects at all stages of accident progression, from initial event up to corium release following the reactor vessel failure. The code models take into account the specific geometry of VVER reactors.
The verification studies of SOCRAT have been carried out over separate effects and using the data of integral experiments. They confirm the code capacities for adequate simulation of processes and effects characterizing the BDBA in VVER reactors. Using SOCRAT the relatively simple SG nodalization schemes can be built, which would adequately describe the dynamics of the main integral parameters in nominal conditions and accident scenarios.

3.1 SG MODEL FOR SOCRAT

SG of a VVER reactor represents a vessel heat exchanger with a horizontal tube bundle submerged in a water pool (fig. 1). A detailed description of construction and typical processes in VVER SG can be found in: Lukasevich et al., 2004, Trunov et al., 2001 and Trunov et al., 2003. While the general methodology for SG nodalization is not strictly fixed, several works help to get a rather full understanding of how that complex unit should be modeled (Haapalehto and Bestion, 1993; A.Moskalev et al., 1999; Nosatov, 2005; Gorchakov, 2006).
The choice of SG model should be made for every issue separately. While modeling SG as part of nuclear installation it is often needed to correctly simulate only SG inlet and outlet parameters. Any simplified SG model describing accurately heat transfer coefficients is actually sufficient for scenarios without rapid water level drop, when heat transfer is not strongly affected by level changes. More complicated models are required for scenarios with SG water level (SGWL) decrease: feed-water regulator failure, feed-water tubing break, steam line break etc.
Two factors cause the uncertainty: initial water inventory in SG and dependence between SGWL, heat source and heat transfer coefficient. Tube bundle longitudinal and height partitioning accounts for different heat fluxes in elements and allows describing heat transfer and water inventory in a more accurate way. In that case one can get a rather realistic estimate of integral parameters even from a simplified SG model having a small number of elements. So there are attempts to simulate multi-dimensional processes in SG by one-dimensional codes using several parallel channels. Such an approach has proven its efficiency for separate effects studies (V. Roginskaya, 2000). On the other hand, the accuracy of calculation schemes is limited by the uncertainty of correlations and material-property data, as well as by computational cost. The number of calculating nodes should not be redundant to enable computations under the model applied using available hardware-and-software tools. This statement is especially true with regard to long-term calculations of BDBA and severe accidents with typical progression time of about 24 hours.
The calculations confirm that SG model consisting of three height layers on the primary side and a recirculation channel (with one downcomer and one riser effective channels) on the secondary side simulates well enough the main processes which are typical for SG operation in normal conditions and during some accident scenarios.
Considering the experience gained in the use of system codes for SG modeling, their advantages and weak points, we were able to build a SG model for Russian code SOCRAT. The calculation scheme of SG is shown on fig. 2. The tube bundle is divided by height into three tube packets, each one representing one effective tube. So the area of heat-exchanging surface changes proportionally to water inventory drop during SGWL decrease. The number of tubes in each packet is defined using the design dependence between surface area, water-steam inventory and SGWL. The low packet includes 17 % of tubes, medium packet – 34 % of tubes and upper packet – 49 % of tubes. The forth volume (above upper packet) represents the volume between the upper range of tubes and a submerged perforated sheet. Due to a considerable non-uniformity of primary coolant temperature distribution along tubes the tubes (flow channels and tube walls) are divided into control volumes in longitudinal direction. Analysis of calculations that were performed before show that 6 volumes are enough (T. Haapalehto, D. Bestion, 1993; A. Nosatov et al., 1999; V. Roginskaya, 2000; V.N. Nosatov, 2005).
The secondary side of SG is represented by recirculation model: the integrity of inter-tube volume is described by one downcomer channel (including all channels and corridors where secondary fluid comes to the bottom) and one riser channel (including inter-tube channels where secondary fluid rises to the top). To ensure natural circulation of secondary steam-water mixture the channels are coupled. Each of these channels consists of 4 control volumes. Three control volumes of riser are coupled correspondingly to

three tube packets by boundary conditions. Beside the riser and downcomer, the secondary side is partitioned into the following volumes:

- feedwater supply zone (chamber SG_FW);
- low volume between SG vessel and the lowest range of tubes (channel SG_BOTTOM);
- submerged perforated sheet SPS (channel SG_SEP);
- volume above SPS – it models the separation of steam-water mixture (chamber SG_UP);
- upper steam volume (channel SG_TOP);
- steam line and steam collector (chamber SG_STEAM).

Feed water supply is simulated using quasi-channel SG_FW_IN and boundary SG_MFW. The SG vessel wall is represented by a heat element including thermal isolation to account for heat losses.

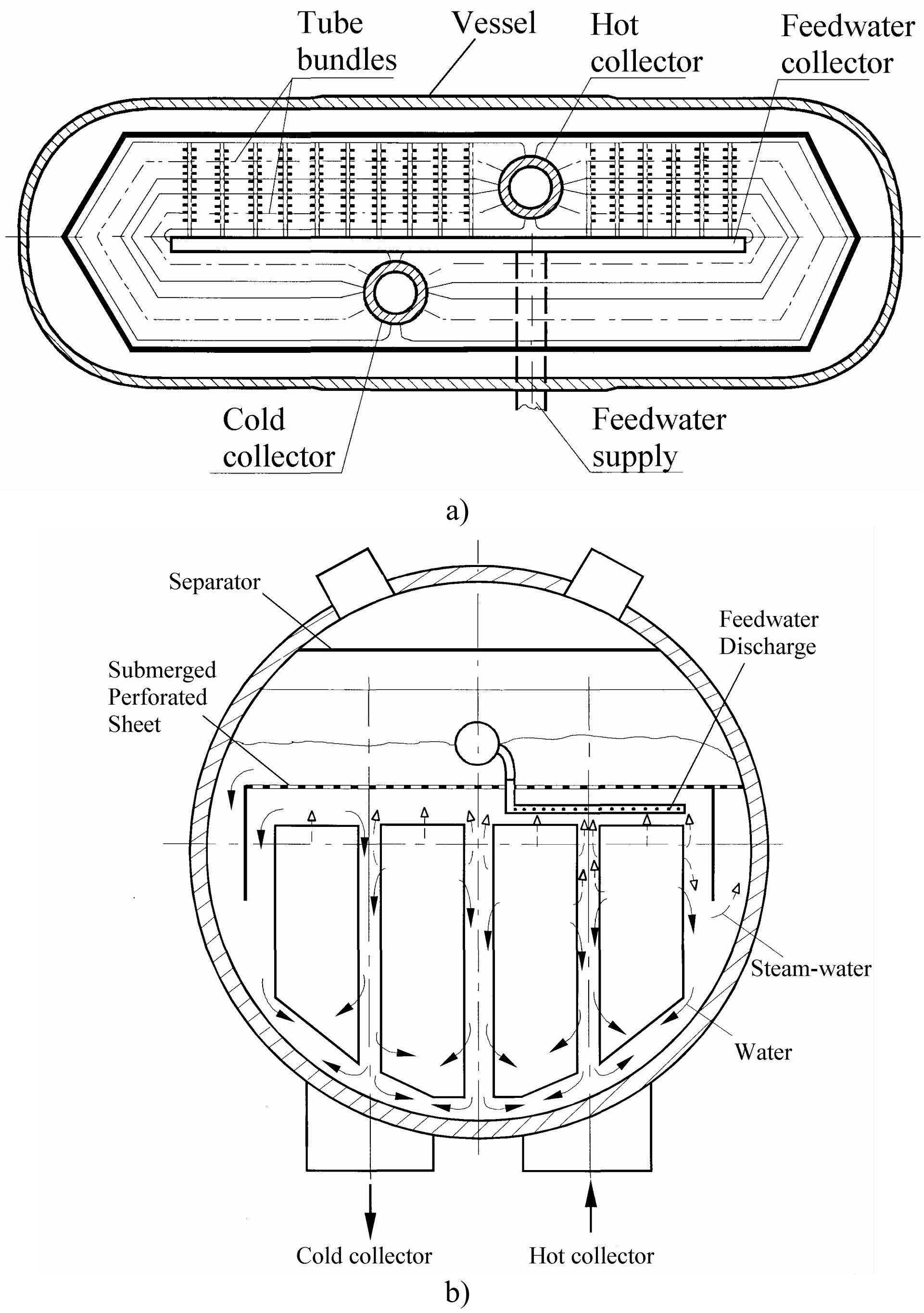

Fig. 1 General view of PGV-1000MKP: a) – above view; b) – cross-sectional view

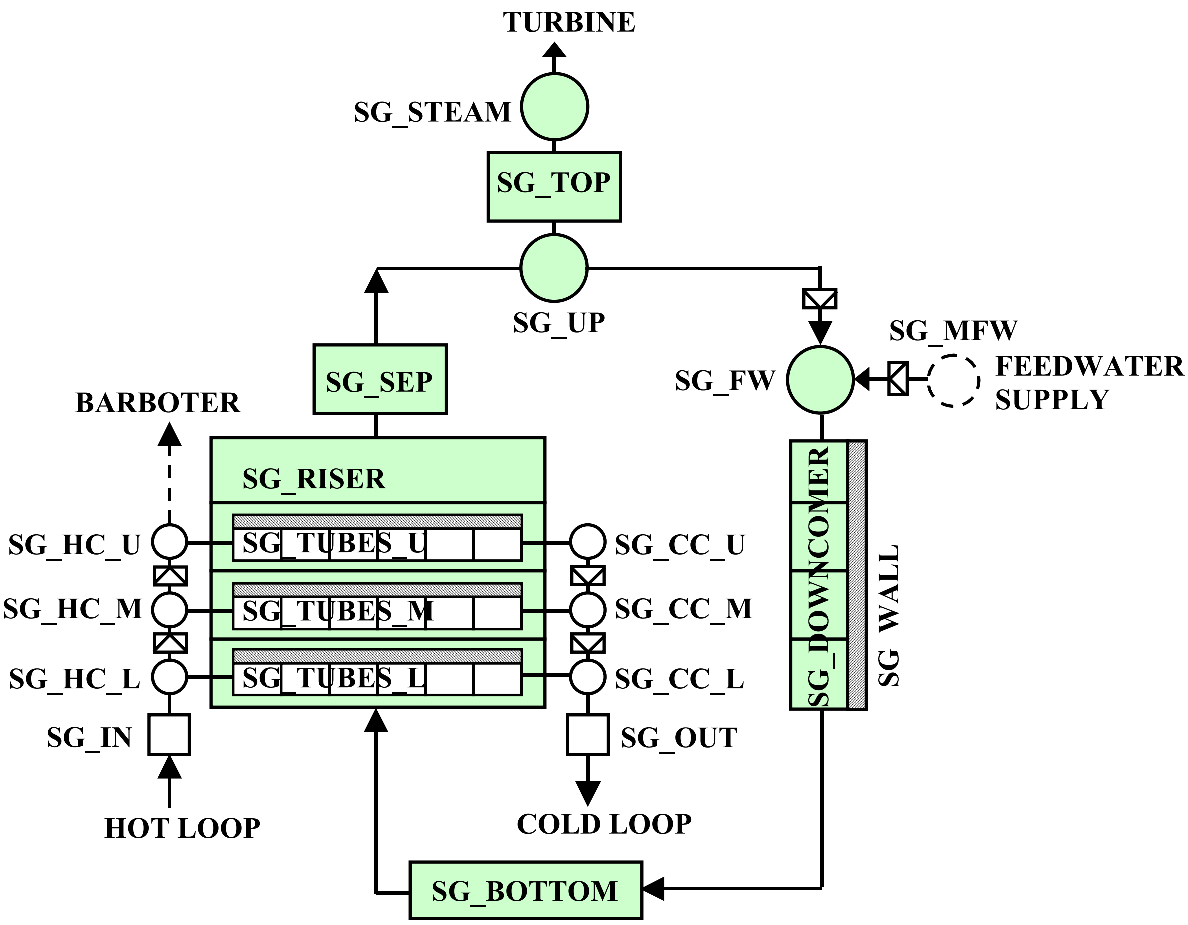

Fig. 2 Nodalization scheme for VVER SG

The SG model for SOCRAT is suitable for analysis of normal operating conditions and accident scenarios without substantial decrease of SGWL. However further model development on the basis of three-dimensional parameters distribution is required to make a full and more accurate analysis of all SG operating conditions. That distribution was calculated using a specially built three-dimensional SG model for CFD code (STAR CD).

3.2 SG MODEL FOR STAR CD

According to the features of tube assembly in the bundle, in each horizontal range all tubes have different length. The difference between the inner and outer tubes comes up to several meters. That results in different resistance of channels and accordingly, in different velocities and flow rates in tubes, depending on their place in the bundle. Besides, the shift of collectors along SG longitudinal axis produces the non-uniformity in feedwater distribution over the tube bundle near hot and cold collector. Therefore the primary water is cooled down differently, the difference coming up to several degrees at cold collector inlet. Besides, normally the up-to-date calculations made with one-dimensional codes do not consider the changes accumulated during SG operation (tubes failure, deposition of corrosion particles in tubes, cracking). For example, the plugging of failed SG tubes affects the flow rate distribution across the other tubes.
The listed factors cannot be taken into account accurately in one-dimensional system codes. Therefore the development of three-dimensional models is important for a more detailed study of the processes, which are typical for both primary and secondary side of SG.
As a result, two three-dimensional CFD models were developed in IBRAE, accounting for design features of collectors and tube bundle (size, geometry, elevations etc.). They include the hydrodynamic volumes bounded by the surfaces of hot and cold collectors and all interconnecting heat-exchanging tubes and do not include the SG secondary side. The sketch of calculation region is shown on fig. 3.

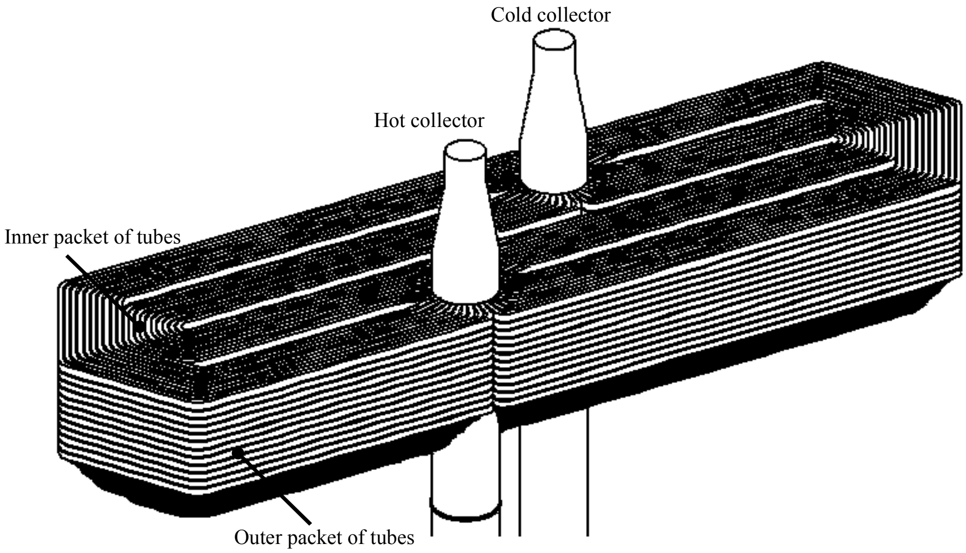

Fig. 3 Sketch of the calculation region of SG in STAR CD

In the first (base) model the heat exchanging tubes are grouped in 738 blocks, every block being considered as a one-dimensional porous structure. Each block includes 4 to 18 heat exchanging tubes. The number of volumes in the block cross-section corresponds to the number of heat exchanging tubes in each block (fig. 4). The second SG model is finer and describes *each* heat exchanging tube as an individual porous body (fig. 5). The mesh dimension in that case is 3.5 million control volumes. The collector regions in both models are meshed with volumes, that is, the coolant flow in collectors is strictly 3D (fig. 6). A k-ω turbulence model was used in all calculations. Additional data can be found in A.V. Shishov's article «Development of 3D tubing model for a VVER steam generator» dedicated to these models and published in the present collection of articles.

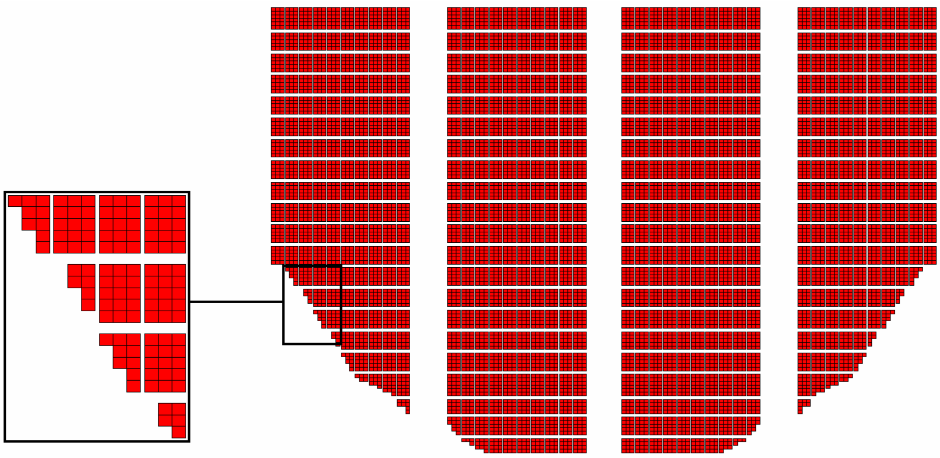

Fig. 4 Mesh in the tube bundle cross-section (heat exchanging tubes are grouped in blocks)

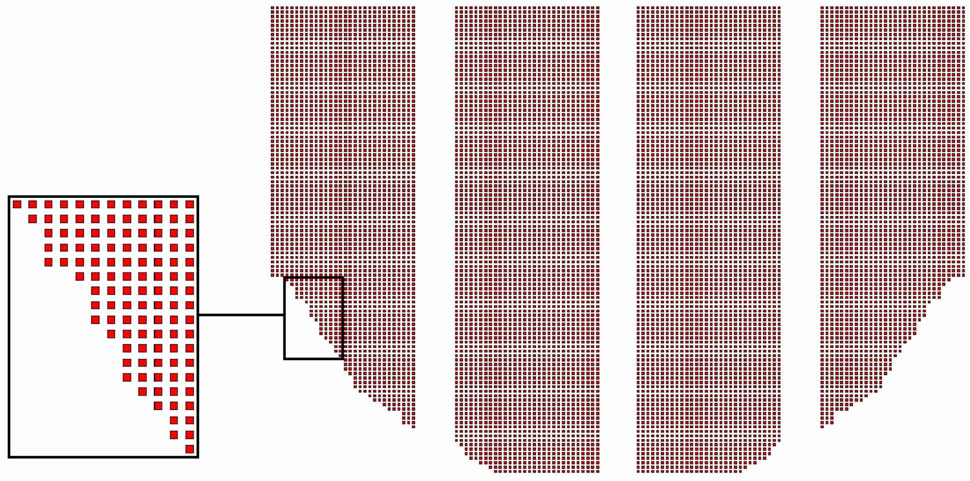

Fig. 5 Mesh in the tube bundle cross-section (heat exchanging tubes are presented individually)

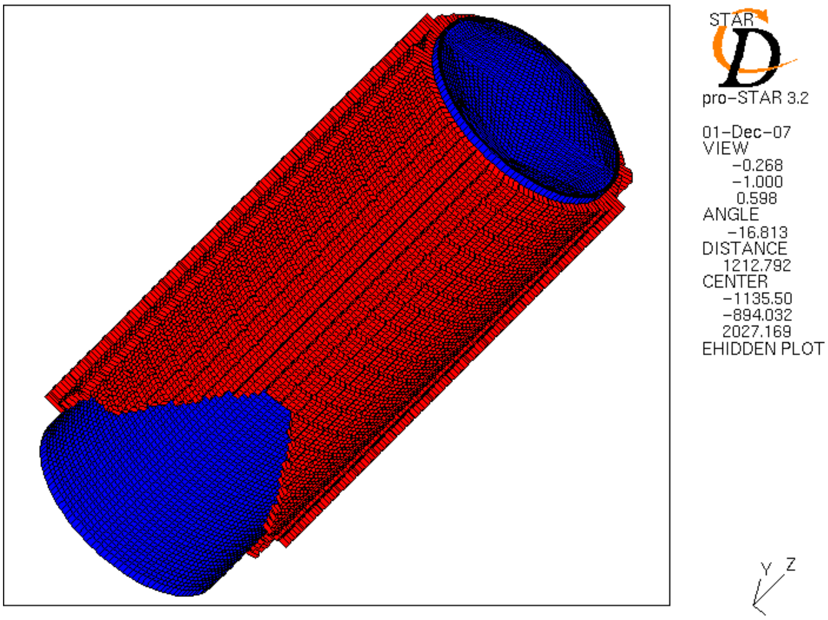

Fig. 6 Meshing of the perforated part of collector

4. RESULTS

Using SOCRAT and STAR CD SG models, velocity values in heat-exchanging tubes and collectors were found at nominal operating conditions. Figure 7 shows the coolant filtering velocity field in the middle transverse A-A section of the tube bundle shown in figure 5. To get a velocity value, the filtering velocity should be divided by porosity (being 0.33 for the base SG model). The velocities in packets were compared to the values found with SOCRAT (Table 1). We can conclude that velocity values in low, medium and upper tube packets found with SOCRAT fall within a range of values found by STAR CD three-dimensional calculations.

Table 1. Relative (numeric to average design ratio) coolant velocity ($v_{num} / v_{design}^{av}$) in tube bundle

Packet of tube bundle	SOCRAT	STAR CD	
	Simple model	Min	Max
Low	1,01	0,95	1,20
Medium	1,04	0,80	1,14
Upper	1,00	0,77	1,01

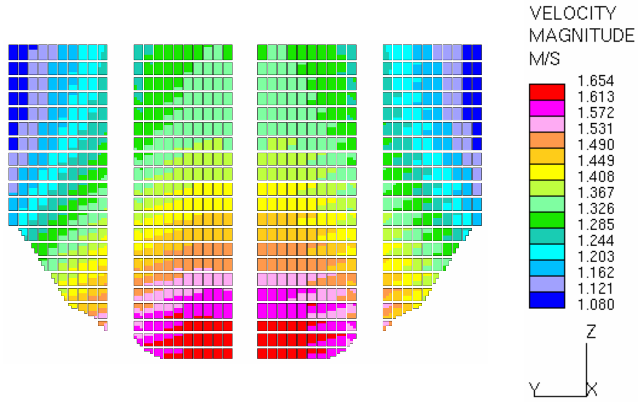

Fig. 7 Coolant filtering velocity field inside the tube bundle (UD-type discretization) of the base SG model

However, a more detailed comparison reveals the difference in velocity distribution across tube packets. In three-dimension problem the maximal values correspond to low packet tubes and decrease from low to upper tubes. In SOCRAT calculations a distribution of velocity by height is relatively regular (maximum-to-minimum ratio being 1.03). Such a discrepancy is explained by the fact that unlike three-dimension model, the hydraulic resistance values in all tubes of SOCRAT model are equal (all tubes being straight and having the same length). Thus, three-dimension calculations allow defining the correct values of hydraulic resistance for its use in model of one-dimensional codes.
The comparison of results reveals as well the substantial spread of velocities across tubes of the same packet in three-dimensional problem: 25 % in tubes of the low packet, 34 % in medium packet, 24 % in upper packet. It means that using three-packet bundle partitioning and one-dimensional channels introduces the uncertainty of about 30 % in velocity (or flow rate) field in each packet. Remind that this is right for normal operation of SG. A possible way to reduce that uncertainty is to simulate all tubes inside each packet with more than one effective tube.
To estimate the effect of nodalization scheme, a finer model of SG was developed for SOCRAT code. Tube bundle was presented with 5 packets by height (instead of 3 as it was before) and the tubes were divided by length in 8 control volumes (6 before). The comparison of results got from calculations with SOCRAT and STAR CD are presented in Table 2.

Table 2. Relative (numeric to average design ratio) coolant velocity ($v_{num} / v^{av}_{design}$) in tube bundle

Packet of tube bundle (from bottom to the top)	SOCRAT	STAR CD	
	Detailed model	Min	Max
1	1.23	1.04	1.20
2	1.10	0.92	1.06
3	0.97	0.83	1.01
4	0.92	0.80	0.98
5	0.89	0.77	0.94

The velocity distribution by tube packets became more realistic, with higher values in lower tubes, while the velocities in some packets are out of range calculated by STAR CD.

In order to confirm the applicability of a porous body model for the considered issues a comparison of numerical results and analytical solutions for a special model issue was made. A pressure drop between hot and cold collector and velocity in different tubes of the bundle were chosen as target variables. Four typical configurations of tubes were considered (fig. 8). For each type of tube a pressure drop was calculated and compared to an analytical value. The discrepancy of target variable values did not exceed 6 % that proves the possibility of use of porous body model in complex issues of coolant flow in SG primary circuit.

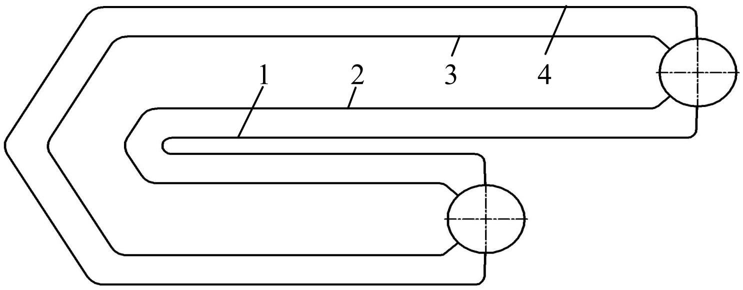

Fig. 8 Different configurations of tubes considered in numerical/analytical comparison

Besides, the base SG model (with tubes gathered in porous blocks) was tested using first- (UD-type) and second-order (MARS-type) discretization schemes. The pressure drop between collectors in the first order scheme is 0,1391 MPa while in the second order scheme it is 0,1377 MPa. The discrepancy in velocities in tubes is also relatively small and is shown on fig. 9-10 and in Table 3.

Table 3. Maximum and minimum of velocities in different discretization schemes

Velocity in tubes, m/s	First order UD discretization	Second order MARS discretization
Maximal	4,99	5,03
Minimal	3,24	3,23

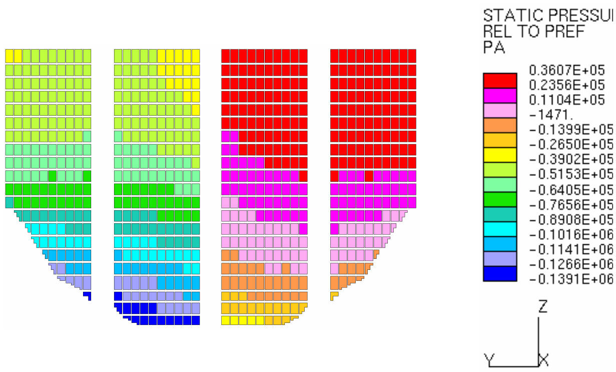

Fig. 9 Relative static coolant pressure field inside the tube bundle (UD-type discretization)

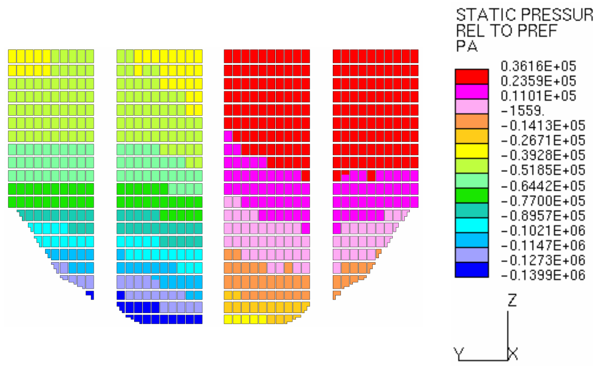

Fig. 10 Relative static coolant pressure field inside the tube bundle (MARS-type discretization)

The difference in results for both discretization schemes did not exceed 5 %.

In order to assess the effect of numerical grid on the velocity distribution a base CFD model of SG was compared to a finer SG model where *each* tube was represented by a porous body. The calculation result for the detailed model is shown on fig. 11. The maximal value is 4,8 m/s, minimal 3,32 m/s. Thus, mesh refinement leads to result change not more than 5 %.

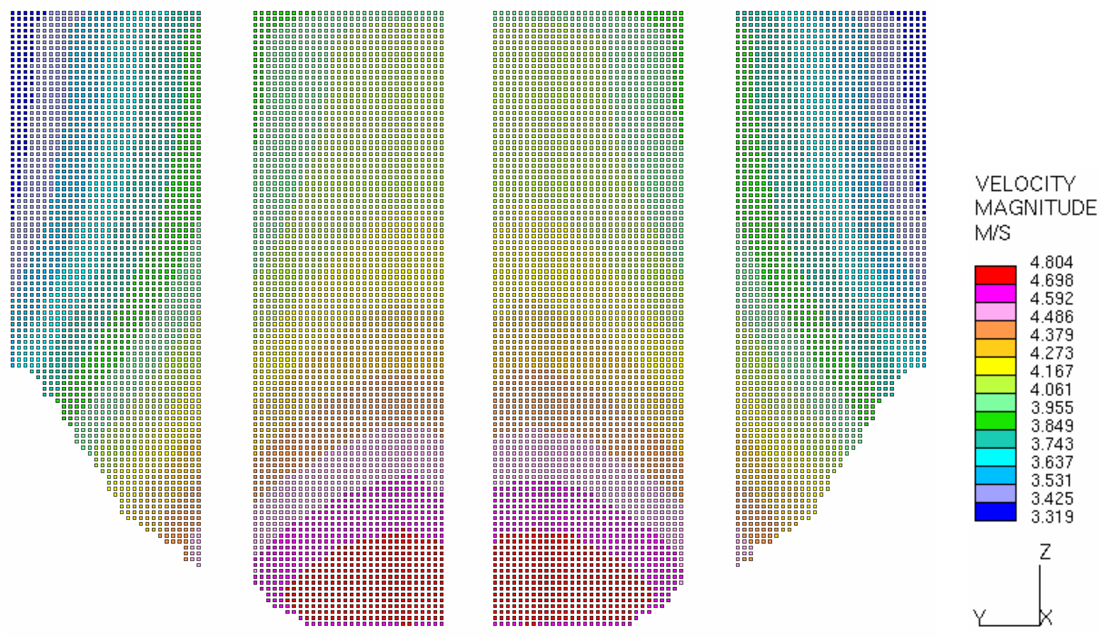

Fig. 11 Coolant velocity field inside the tube bundle (UD-type discretization)

The comparison to experiments or plant data was not made because of the absence of such data at the moment. Separate measurements were made to study the circulation of secondary coolant, but in the issues considered here the SG secondary circuit is not modeled.
The in-depth analysis of uncertainties is planned for the next stages of that project, when a model of heat-exchange through collector and tube walls will be built in addition to a hydraulic one. The uncertainty parameters are chosen as follows: the boundary conditions of heat exchange with secondary circuit; coolant temperature at SG inlet; coolant flow rate in tubes. The uncertainty in boundary conditions is defined by the non-uniform feed-water distribution and, correspondingly, by different temperature of secondary coolant in SG. The coolant temperature variation at SG inlet can be caused by different conditions of coolant flow in reactor loops and by plant operation features. At last, the plugging of tubes during SG operation changes the flow distribution in tube bundle. It depends on the number and the location of tubes plugged. The additional impact can be caused by the deposition of corrosion products affecting the roughness of tube surfaces.

The need for further development of one-dimensional SG models supposes the adaptation of CFD codes to simulation of accident scenarios that lead to SGWL decrease, followed by the voiding and heating up of the upper part of tube bundle.
At the next stage of CFD SG models development it is necessary to account for presence of non-condensible gases, radioactive aerosols and fission products in coolant. A correct description of flow in tubes would allow the calculation of temperature field and stress-strain state in tubes (in case of severe accidents when deposition of heat generating fission products becomes important) in a more realistic way as compared to up-to-date calculation models.
Using three-dimensional models one could study the possibility of local reversion or stagnation of water-steam flow in some tubes or in all tubes of the bundle. Yet using the only one-dimensional codes makes such investigations embarrassing.
The SG tubes can be optimally grouped with respect to non-uniformity in magnitude/direction of coolant flow through the bundle. The groups having the close values of flow direction and magnitude can then be described using a small number of effective tubes. Thus the models used in system code become more adequate and their simplicity is kept not affecting the calculation cost.
Cross-verification of results obtained from CFD and system codes helps to decrease considerably the conservatism of one-dimensional models and at the same time to select the priority steps toward the CFD models development.

5. CONCLUSIONS

Importance and effectiveness of a coupled use of CFD and system codes in nuclear safety analysis is shown. In the framework of three-dimensional approach a model of PGV-1000MKP SG primary circuit is build for CFD code STAR CD. A SG model for system code SOCRAT is built and justified. Using STAR CD and SOCRAT codes hydrodynamic calculations of coolant flow through collectors and tube bundle in normal operating conditions are performed. Cross-verification of velocity field in tube bundle has shown a good agreement of results obtained from SOCRAT and STAR CD calculations.
Further development of three-dimension SG model for STAR CD is planed. It will allow cross-verification of calculations of different accident scenarios important for SG tubes integrity.

REFERENCES

H.J. Allelein et al. International Standard Problem ISP-47 on Containment Thermal Hydraulics – Final Report, NEA/CSNI, 2007.

A.A. Gorchakov, O.V. Kuvshinova, “Modeling of Thermohydraulic Transients in VVER-1000 Steam Generators with Improved Version of CORSAR Code”, *7th Int. Seminar on Horizontal Steam Generators*, Moscow, Russia, 2006.

T. Haapalehto, D. Bestion “Horizontal steam generator modeling with CATHARE: validation on several nodalization schemes on plant data”, *2nd Int. Seminar of Horizontal Steam Generator modeling*, Lappeenranta, Finland, 1993.

B.I. Lukasevich, N.B. Trunov, Yu.G. Dragunov, S.E. Davidenko *VVER reactor plant steam generators for nuclear power plants*, Akademkniga, Moscow, Russia, 2004.

J. Mahaffy et al. Best Practice Guidelines for the use of CFD in Nuclear Reactor Safety Applications, NEA/CSNI, 2007.

V.N .Nosatov *Modelling of VVER-type reactors accidents*, PhD thesis, Moscow, Russia, 2005.

M. Scheuerer et al. “Evaluation of Computational Fluid Dynamics Methods for Reactor Safety Analysis” – Final Synthesis Report, ECORA, 2005.

N.B. Trunov, S.A. Logvinov, Yu.G. Dragunov Hydrodynamical and thermochemical processes in VVER steam generators, Energoatomizdat, Moscow, Russia, 2001.

N.B. Trunov, O.I. Melikhov, V.I. Melikhov, Yu.V. Parfenov “Analysis of Thermal Hydraulics and Solubale Impurity Distribution in Horizontal Steam Generator PGV-1000 with STEG Code” *Proc. of 11th Int. Conf. on Nucl. Eng.*, Tokyo, Japan, April 20-23, 2003.

V. Roginskaya “Steady-State Distribution of Void Fraction and Water-steam Mixture Velocity in a Secondary Circuit of Steam Generator PGV-1000”, RRC KI, Moscow, Russia, 2000.

A.Moskalev, S.Pylev, V.Roginskaya et al. “ATHLET calculations for comparative analysis of WWER SG models and test calculations for station blackout”, NSI RRC KI Report No 90-12/1-4-99, Moscow, Russia, 1999.

A.D. Vasiliev, A.E. Kiselev, G.V. Kobelev “Integral best estimate code RATEG/SVECHA: structure, verification and preliminary results of modeling of severe accident during in-vessel stage at nuclear power plant with VVER-1000-type reactor”, *Proc. of conf. “Safety aspects of nuclear power plant with VVER”*, Saint-Petersburg, Russia, October 12-15, 2000, P. 35-42.

V.V. Bezlepkin, V.O. Kuchtevich, A.D. Vasiliev et al. “The status of development of code RATEG/SVECHA/HEFEST for modeling of core degradation processes at severe accidents”, *Proc. of 2nd Russian conf. “Safety Ensurance of NPP’s with VVER”*, Podolsk, Russia, November 19-23, 2001.

V.V. Bezlepkin, V.O. Kuchtevich, Vasiliev A.D. et al. “Severe accidents numerical computation analysis using Russian code RATEG/SVECHA/HEFEST”, *Proc. of 3rd Russian scientific and technical conf. “Safety Insurance of NPP’s with VVER”*, Podolsk, Russia, May 26-30, 2003. V. 6, p.128-139.

Methodology STAR CD: Computational Dynamics Limited, 2005.

A. Guelfi, M. Boucker, J.M. Hérard et al. “A New Multi-Scale Platform for Advanced Nuclear Thermal-Hydraulics Status and Prospects of the NEPTUNE Project”, *NURETH-11*, Avignon, France, October 2-6, 2005.

D. Bestion, A. Guelfi “Multiscale analysis of nuclear reactors thermal-hydraulics – the NEPTUNE project”, *La Houille Blanche*, № 5, 2005.
A. Guelfi, D. Bestion, M. Boucker et al. “NEPTUNE: A New Software Platform for Advanced Nuclear Thermal Hydraulics”, *Nuclear Science and Engineering*, July 2007, V. 156, · № 3, · P. 281-324.