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ANALYSIS OF FINITE-DIFFERENCE APPROXIMATION OF CODE "FLOW" IN PERFORMING DESIGN CALCULATIONS OF PARAMETERS IN THE PRIMARY AND SECONDARY CIRCUITS OF VVER REACTOR PLANT

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ABSTRACT *In code package "FLOW" energy and coolant hydrodynamics equations, neutron kinetics equations and equations of heat conduction in metalwork and fuel rods are solved together. Processes described by these equations are characterized by different time constants and during calculation different degree of time and space discretization is required. The objective of the work was demonstration of stability of numerical schemes used in "FLOW" program and its modules, assessment of error of numerical solution for nodalization. Initial differential equations in partial derivative of the law of conservation of mass, energy and momentum are reduced to the system of usual differential equations on the basis of spatial approximation with the use of integro-interpolation method. Approximation was performed by integrating the initial differential equations in partial derivatives within the limits of each elemental volume of division by spatial coordinate. The usage of integro-interpolation method provides fulfilment of laws of conservation on computational mesh as a whole and the second order of approximation by spatial coordinate. The usage of integro-interpolation method provides fulfilment of laws of conservation on computational mesh as a whole (conservatism of approximation method) and the second order of approximation by spatial coordinate. Effect of limiting error of integration in "FLOW" code on pressure values at the core outlet and mass discharge from the leak is not practically distinguished. Effect of these errors on the values of coolant flow rates at the core inlet and outlet becomes more noticeable. Still, maximum value of these differences doesn't exceed 7 % for flow rate at the core inlet. More substantial differences (of the order of 10–15 %) are observed in the course of calculated temperature of fuel rod cladding.*
Keywords: *software package; hydrodynamics equations; neutron kinetics equations; code "FLOW"; finite-difference approximation*

АНАЛІЗ КІНЦЕВО-РІЗНИЦЕВОЇ АПРОКСИМАЦІЇ КОДУ «FLOW» ПРИ ВИКОНАННІ ПРОЕКТНИХ РОЗРАХУНКІВ ПАРАМЕТРІВ ПЕРШОГО ТА ДРУГОГО КОНТУРІВ РЕАКТОРА ВВЕР

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АНОТАЦІЯ *У кодовому пакеті «FLOW» спільно розв'язуються рівняння гідродинаміки енергії та теплоносія, рівняння нейтронної кінетики та рівняння теплопровідності в металоконструкціях і твелах. Процеси, що описуються цими рівняннями, характеризуються різними постійними часу і під час розрахунку необхідний різний ступінь часової та просторової дискретизації. Метою роботи була демонстрація стійкості чисельних схем, що використовуються в програмі «FLOW» та її модулях, оцінка похибки числового розв'язку для нодалізації. Вихідні диференціальні рівняння в частинних похідних закону збереження маси, енергії та імпульсу зведено до системи звичайних диференціальних рівнянь на основі просторової апроксимації з використанням інтегроінтерполяційного методу. Апроксимацію проводили шляхом інтегрування вихідних диференціальних рівнянь у частинних похідних у межах кожного елементного об'єму ділення за просторовою координатою. Використання інтегроінтерполяційного методу забезпечує виконання законів збереження на розрахунковій сітці в цілому та другого порядку апроксимації за просторовою координатою. Використання інтегроінтерполяційного методу забезпечує виконання законів збереження на розрахунковій сітці в цілому (консервативність методу апроксимації) та другого порядку апроксимації за просторовою координатою. Вплив граничної похибки інтегрування в коді «FLOW» на значення тиску на виході з активної зони та масовий викид із витоку практично не розрізняється. Вплив цих похибок на значення витрат теплоносія на вході та виході з активної зони стає більш помітним. Однак максимальне значення цих відмінностей не перевищує 7 % для витрати на вході в активну зону. Більш суттєві відмінності (порядку 10–15 %) спостерігаються при розрахунковій температурі оболонок твелів.*
Ключові слова: *програмний комплекс; рівняння гідродинаміки; рівняння нейтронної кінетики; код «FLOW»; кінцево-різницева апроксимація*

Introduction

"FLOW" code uses semi-rigid calculation model. All the structure of the primary circuit along the path of coolant flow is broken down into elemental volumes (cells), which are combined into interconnected structural elements. The number of structural elements is determined by the necessity for consideration of all

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circuit elements being under different operating conditions and simplification of calculation model as maximum as possible. The operating loop combines all operable loops. Such model is applied, if all operable loops are under similar conditions. The code allows to simulate up to four calculated loops being under different conditions (cooling water supply, power supply to reactor coolant pumps, connection of pressurizer and etc.) In any case the last calculated loop always corresponds to the emergency one and next to last calculated loop is used with "weight" equal to the number of remained loops, all of them being under similar conditions [1–5].

A degree of break-down of each structural element is determined by the objective of the calculation, but it is similar for all operable loops, geometrical and hydraulic characteristics of operable and emergency loops can be different.

In simulating the reactor pressure and collection chambers characteristics of each cell of these elements are assigned separately in connection with complexity of geometry of reactor chambers [1].

Cold pipelines can be included into any cell of RPC and hot ones - into any cell of RCC, respectively.

Pressurizer can be connected to any element of hot pipeline of the first operable or emergency loop.

In the emergency core cooling system, the following is singled out: passive unit - water accumulators being under gas pressure (maximum number of accumulators – four) and active unit - emergency injection pumps (maximum number – 12).

Both active and passive elements of ECCS can be connected to any cell of cold and hot pipelines of any loops and directly into reactor chambers.

Cells are connected with overflow lines which are characterized by pressure loss and sluggishness.

RCP sets can be included into any cell of cold pipelines of loops [1].

This calculation model allows to simulate leak from any element of the emergency loop or chambers of the reactor. In this case coolant leak either from one cell or, in case of double-ended outflowing, from two cells, being adjacent by the model, is provided for.

Each steam generator over the secondary circuit is represented by one special elemental volume wherein a calculated element of the varying volume is singled out. In this element coolant parameters above the steam-water mixture level are simulated [1, 3–5].

Steam lines are presented by one calculated volume connected with steam space of steam generators. Operation of feedwater supply system, turbine stop valves, steam dump valves, safety valves on steam generators, quick-acting shut-off valves is simulated.

The number of calculation volumes in the loops and chambers of the reactor is up to 100.

In the core there can be considered up to five channels, one of which simulates leaks (channels without fuel rods). The number of calculated volumes over channels height is up to 12, ten of them are over the heated part.

Integration step of thermohydraulic equation systems is limited by the length of minimum element of the calculational model, so it is necessary to take into account this factor in compiling the calculation model. By the same reason artificially excessive values (to have possibility to simulate the certain conditions) of pressure loss coefficients shall be avoided [1].

List of the main calculated parameters

The results of calculations made by code package "FLOW" are recorded into three files [1]:

- file in which in the process of calculation with the time step assigned by the user instantaneous values of all parameters required for restart are recorded successively. This file can be also used for visual display by KORR module of RP parameters (RP elements) during moments of time fixed in this file;
- file with tables of time variation of separate RP parameters. It is used by KORR module in plotting time variation of these parameters;
- file in which all input data assigned by the user and tables with all main calculated parameters in all elements of the calculational model are recorded in the form convenient for the user. Recording of tables with the results is performed in the process time interval determined by the user.

Basically, tables of time variation of any parameter (temperature, pressure, specific enthalpy, void fraction, coolant flow rate, temperatures of metalwork and fuel at all calculated points and etc.) calculated in code package «FLOW» can be recorded. Total number of recorded parameters shall not exceed 150, though this limitation can be easily changed in any direction. However, as a result of experience of using code package "FLOW" in performing design calculations for NPP with VVER a specific list of these parameters was mainly composed taking into account the requirements of guidelines [1,6-14].

Code "FLOW" provide for recording tables of the following fixed parameters [1]:

1) nodal values of time;

2) values of pressure differential on the core;

3), 4) values of pressure at core inlet and outlet, respectively;

5), 6) values of specific enthalpy at core inlet and outlet, respectively;

7), 8) values of boric acid concentration at core inlet and outlet, respectively;

9) values of relative power in fuel;

10) pressure in the pressurizer;

11)…13) values of water volume (liquid phase) in the primary circuit, reactor and the reactor pressure chamber, respectively;

14), 15) values of mass and energy fluxes of the coolant flowing through break (breaks) in the primary circuit;

16) values of heat flux minimum departure from nucleate boiling in the core;

17) values of fuel maximum temperature;

18) values of total cooling water flow from ECCS;

19), 20) values of coolant flow rate at core inlet and outlet;

21) values of heat flow from fuel rod claddings to the coolant;

22), 23), 24) values of average density of the coolant, average temperature of fuel and maximum temperature of fuel rod claddings in the core, respectively.

Parameters 3) – 9) are also intended for using as boundary conditions during analysis of thermal-and-hydraulic situation in separate assemblies (channels) of the core by code.

Further in evaluating the calculation results the values of parameters, which are usually used in substantiating safety of RP with VVER, will be taken as the main ones [1]:

- coolant pressure in the primary circuit;
- coolant temperature in the secondary circuit;
- coolant temperature at the core outlet;
- coolant temperature at the reactor inlet;
- - minimum value of heat flux DNB;
- maximum value of fuel rod claddings temperature;
- maximum value of fuel temperature;
- maximum value of depth of fuel rod claddings oxidation;
- quantity of oxidized zirconium in the core;
- reactivity in the core;
- mass discharge from the primary circuit;
- energy ejection from the primary circuit.

In substantiating the methods, the following problems can be chosen:

- analysis and substantiation of numerical schemes stability, sensitivity to input data and determination of error in numerical solution of the system of equations;
- check of programs capability in the field of application;
- programs testing by solving special problems and by comparison with calculation results of the other programs;
- comparison of calculation results with the results of experiments.

In this section the results of solution of the first problem – substantiation of numerical schemes stability, sensitivity to input data and determination of error in numerical solution of the system of equation – are presented [1].

Verification of codes by the results of experiments is given in [10].

In code package "FLOW" energy and coolant hydrodynamics equations, neutron kinetics equations and equations of heat conduction in metalwork and fuel rods are solved together. Processes described by these equations are characterized by different time constants and during calculation different degree of time and space discretization is required.

So, hydrodynamics equations taking into account coolant compressibility ("FLOW" code) with the use of explicit numerical schemes impose restriction on integration time step [1]

$$\Delta\tau \le \frac{l}{|c|+|w|},\ \Delta\tau\sim 0{,}001\text{ s.} \qquad (1)$$

Equation of motion for incompressible coolant – restriction

$$\Delta\tau \le \frac{l}{\xi|w|},\ \Delta\tau\sim 0{,}01\text{ s.} \qquad (2)$$

Energy and mass equations - respectively restrictions

$$\Delta\tau \le \frac{M}{\sum|G|},\ \Delta\tau\sim 0{,}1\text{ s.} \qquad (3)$$

$$\Delta\tau \le \frac{U}{\sum|G\cdot i|+|Q|},\ \Delta\tau\sim 0{,}1\text{ s.} \qquad (4)$$

Equations of heat conduction –

$$\Delta\tau \le \frac{\delta^2\cdot\rho\cdot c_{\text{P}}}{\lambda},\ \Delta\tau\sim 0{,}1\text{ s.} \qquad (5)$$

Neutron kinetics equations –

$$\Delta\tau \le \frac{L}{\beta},\ \Delta\tau\sim 0{,}0001\text{ s.} \qquad (6)$$

Equations use the following symbols given in Table 1 [1].

Table 1 – The symbols for the equations in this work

Symbols	Clarifications
c	Isoentropic sound velocity in coolant
c_p	Metal heat conduction
G	Coolant flow rate in junction (summation for all junctions being adjacent the given calculated element)
$G\cdot i$	Enthalpy flow in junction (summation for all junctions being adjacent the given calculated element)
L	Prompt neutron life
l	Spatial coordinate step (length of nodes)
M	Coolant mass in the node
Q	Thermal flow to coolant in the considered node
U	Internal energy of coolant in the node
β	Total fraction of delayed neutrons
δ	Thickness of calculated layer (spatial coordinate step in metal
λ	Metal heat conduction
ρ	Metal density
w	Coolant velocity
ξ	Pressure loss coefficient
$\Delta\tau$	Time integration step

It is clear from the equations that values of typical time for different processes vary within the wide range (from 0,1 to 0,0001 s) and solution of the system of equations required combination of different numerical schemes [1].

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For example, the last restriction is stricter and it required the usage of the special method of integration. The first restriction also resulted in introduction into code "FLOW" of subroutines implementing algorithm of implicit method for solution of the system of hydrodynamics equations in the primary circuit together with the usage of explicit scheme.

The objective of the work

The objective of the work is demonstration of stability of numerical schemes used in "FLOW" program and its modules, assessment of error of numerical solution for nodalization.

Substantiation of finite-difference approximation of code "FLOW"

Initial differential equations in partial derivative of the law of conservation of mass, energy and momentum are reduced to the system of usual differential equations on the basis of spatial approximation with the use of integro-interpolation method. Approximation was performed by integrating the initial differential equations in partial derivatives within the limits of each elemental volume of division by spatial coordinate [1].

The usage of integro-interpolation method provides fulfilment of laws of conservation on computational mesh as a whole (conservatism of approximation method) and the second order of approximation by spatial coordinate.

The figure represents cells V_i, V_j, V_k, V_1 with centres of mass Z_i, Z_j, Z_k, Z_1, respectively, Arrows designate directions of coolant movement under steady-state conditions. Variation of flow area and local pressure loss are considered to be concentrated at the points of cells junction (in figure $\overline{Z}_i$, $\overline{Z}_j$, $\overline{Z}_k$, $\overline{Z}_l$). l_i – length of i-element in direction of axis Z_1; H_i – length of projection of i-element to the vertical; G_j – coolant flow rate through junction at point $\overline{Z}_i$; G_i – coolant flow rate at point Z_i – in the centre of mass of cell V_i.

The systems of usual differential equations, obtained as a result of spatial approximation of Cauchy problem, are integrated with the use of any of the following methods: special method based on the usage of the implicit scheme in combination with Neuton method; Euler-Cauchy method of the second order of accuracy with automatical choice of integration step.

Calculation of the accident conditions with break of the main coolant pipeline (MCP) at the reactor inlet by "FLOW" program was carried out with the use of calculational nodalization models. Nodalization (four-loop) corresponds to break-down of the primary circuit which is recommended for the design projects. The necessity for performance of calculation of a variant with three cooling loops is caused by: requirements of testing the algorithms of both control programs and subroutines of package "FLOW" as well as requirements in large number of calculational sections in substantiating finite difference approximation [1–5].

Curves, reflecting variation of desired variables, obtained in calculations with the use of four-loop nodalization model are given in Fig. 1–3, in this case the limiting error of integration (ε=0,0005, or 0,00025, or 0,0001), controlling the actual choice of time step is a parameter. Calculation of the transient with MCP break at the reactor inlet is carried out before reaching the steady-state conditions of cooling in the core which takes place after 140 s (if to trace the transient by pressure – Fig. 1 and by fuel temperature – Fig. 3), or after 100 s (by maximum claddings temperature, Fig. 3) [1].

As is well seen in Fig. 1 and 2 effect of limiting error of integration in "FLOW" code on pressure values at the core outlet and mass discharge from the leak is not practically distinguished. Effect of these errors on the values of coolant flow rates at the core inlet and outlet becomes more noticeable. Still, maximum value of these differences doesn't exceed 7 % for flow rate at the core inlet. More substantial differences (of the order of 10–15 %) are observed in the course of calculated temperature of fuel rod cladding, Fig. 3 [1].

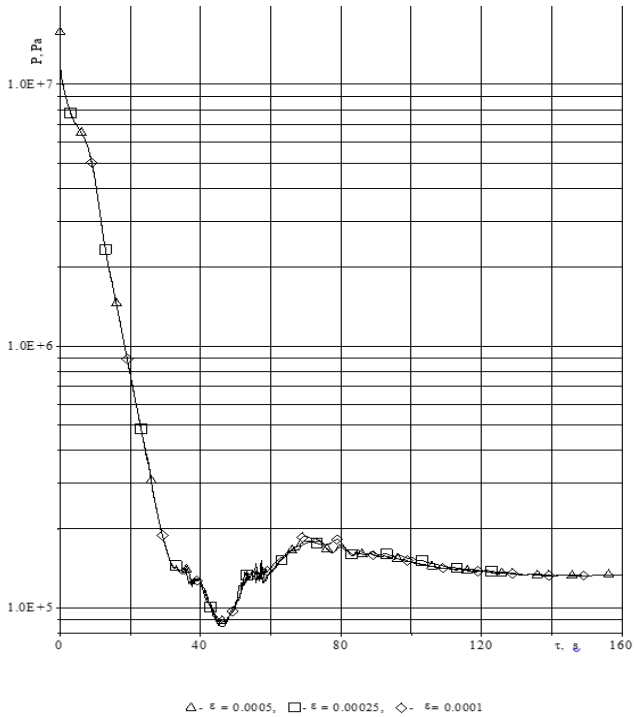

Fig. 1 – Effect of limiting error of integration on pressure at the core outlet

Effect of spatial discretization on the behaviour of the most sensitive parameters such as temperatures in the centre of fuel and temperatures of fuel rod cladding at different elevations of the core is shown in Fig. 4 and 5 [1]. In general, equidistance shall be noted in the course of curves of fuel and cladding temperatures during the whole transient, that shows stability and convergence of algorithm of calculation. Available variations of calculated temperatures of small amplitude are caused by its crude approximation with the use of crude calculation mesh and differences, connected with it, don't exceed 100 ^{0}C, that is equal to not more than 10 % within the

given range of temperatures variation. The general trend is some overestimation of temperatures during calculations using more crude mesh: as a rule, all curves (with a smaller number of sections) go higher. The considered accident conditions with MCP break result in variation of the desired parameters within the wide range and are characterized by large non-linearities resulting in high requirements for the used algorithm of calculation.

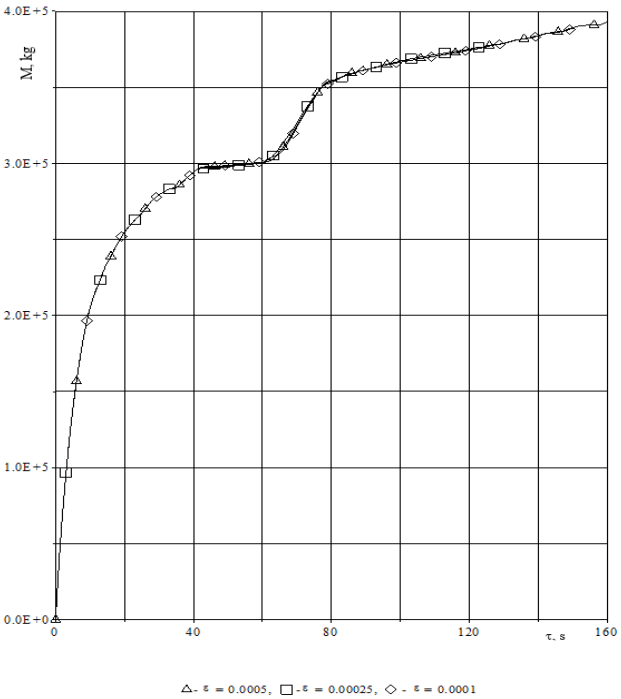

Fig. 2 – Effect of limiting error of integration on the calculated mass discharge from leaks

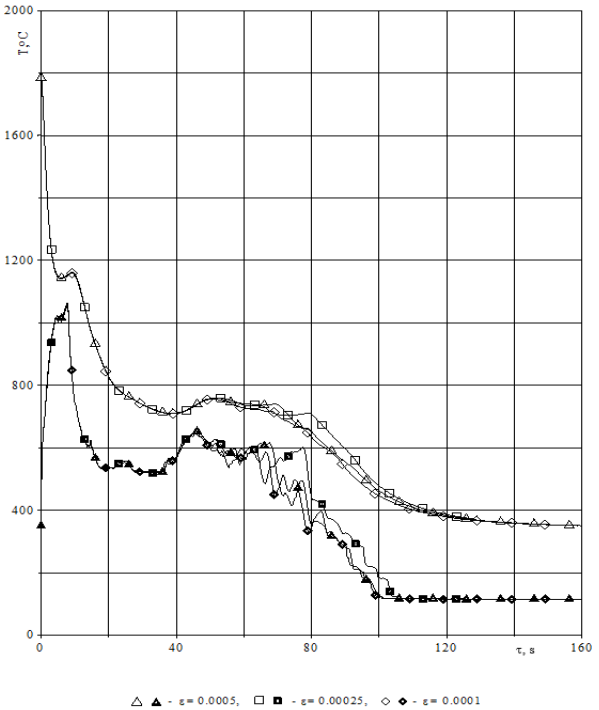

Fig. 3 – Effect of limiting error of integration on maximum calculated temperatures of fuel and fuel rod cladding

The presented results show quite adequate behaviour of the desired variables, stability and convergence of the obtained results in decreasing parameters of spatial-time discretization of the calculation mesh.

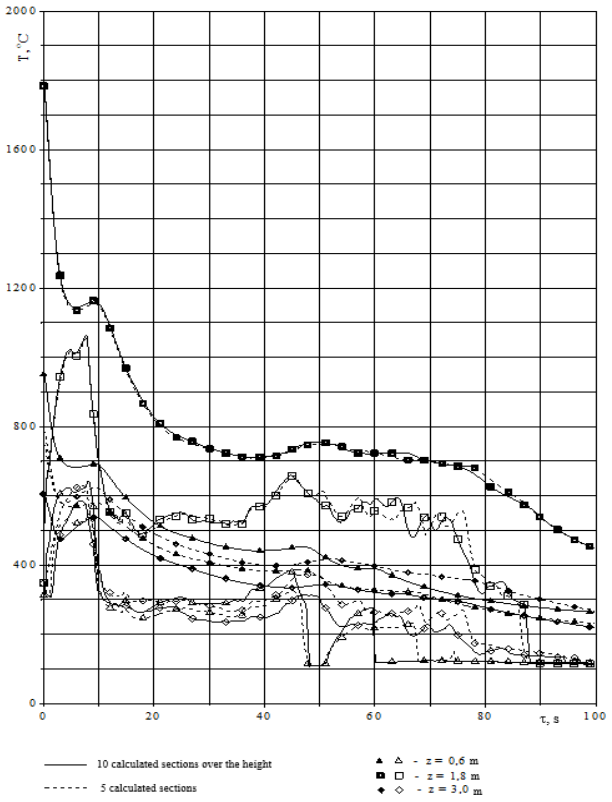

Fig. 4 – Temperatures of fuel in the centre and fuel rod cladding in different sections over the height

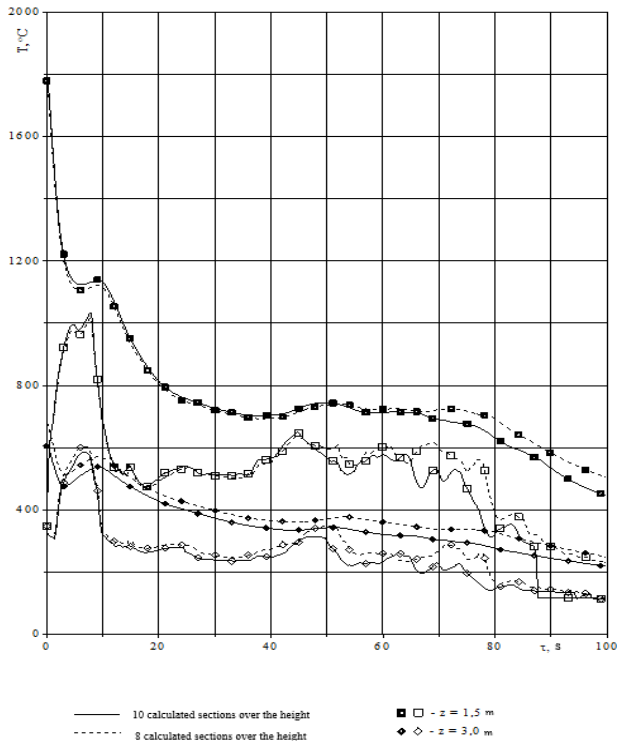

Fig. 5 – Temperatures of fuel in the centre and fuel rod claddings in the central and upper sections Kr=1,74

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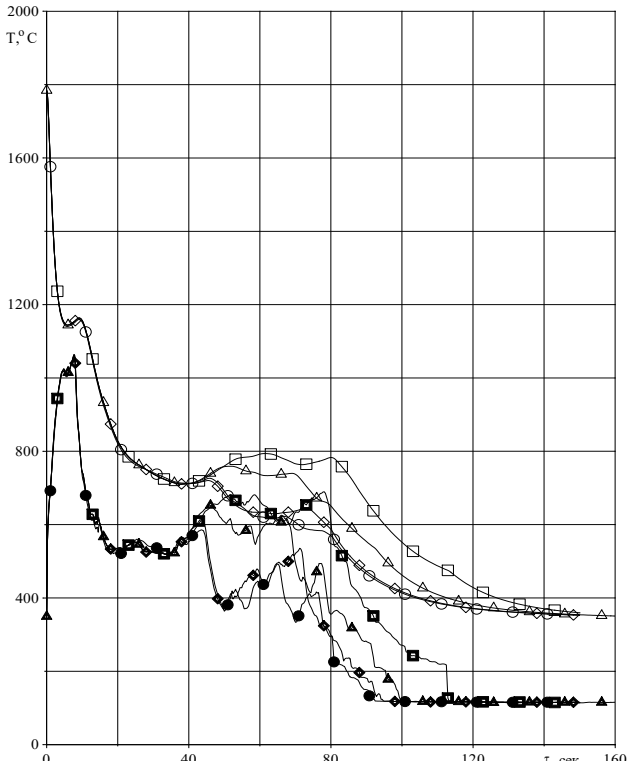
The next series of calculations showed the effect of nodalization of the primary circuit and the model of water supply by ECCS pumps on the calculation results of these conditions. The design (four-loop) variant of calculation of the accident conditions with MCP break at the reactor inlet shown above was taken as the initial one. The core was simulated by 10 sections and error of integration was accepted to be equal to ε = 0,0005. Transfer to three-loop variant of nodalization model of calculation with different, alternative variants of water supply from ECCS pump as well as with increased number of calculated sections in loops has been considered.

Fig. 6 demonstrates behaviour of the same calculated parameters that are mentioned above. In this case the main design variant with water supply by one pump of ECCS to the second loop and by one pump to the emergency loop practically coincides with a version of supply by one pump (equally) to the second and third loops and by one pump to the emergency loop, but as filling of serviceable loops by water from ECCS pumps occurs slower, the process of the core cooling is slightly protracted, see Fig. 6.



△ - *four loops water supply by ECCS pump into the second and emergency loops, 16 (17) sect. τ, s - in each loop;*
□ - *the same, but ECCS water, supplied into the second loop, is supplied into the second and third loops;*
◇ - *three loops, ECCS water supply into the second and emergency loops;*
○ - *the same, but 24 sections – in each loop (in emergency loop – 25).*

Fig. 6 – Maximum temperatures of fuel and fuel rods cladding

In case of three-loop nodalization model of calculation with increased water supply from ECCS to the second (two pumps, loop has "weight" 2) loop and by one pump to the emergency loop (symbol - ◊) the process of core cooling takes place more actively. In the variant with the same scheme of water supply by ECCS pumps, but with the increased number of sections by a factor of one and a half in the loops (symbol - O) cooldown takes place, practically for the same time, but there are noticeable differences in behaviour of temperature curves, Fig. 6. These differences reach 20 %, but they are not fundamental as they appear in the process of fuel rods cooling at values of cladding temperatures being about two times less than maximum values. At the stage of fuel rod heating-up and in the area of maximum values of temperatures (the first 25 s of the process) differences in temperatures don't exceed 1 %. In general, comparison of the main and two last variants show good convergence of the results with the increased number of calculated sections in the loops.

Analysis of stability by input data

Numerical stability by the input data is provided by conservatism of the used difference schemes, stability of the used numerical schemes and calculated processes (natural mechanism of self-regulation), continuity and differentiability of the right parts of differential equations. Exceptions are equations simulating operation of the systems of protection and interlockings where breaks are possible in the right parts of equations due to assumptions of "instantaneous" opening of some valves (for example, on the pressurizer), "instantaneous" connection of separate systems (for example, systems of standard make-up of the primary circuit) [1]. Presence of such breaks decreases an order of approximation of differential equations by a difference scheme and reaching the conditions for actuation of such systems, naturally, qualitatively affect the calculation results and it is apparently not difficult to invent such conditions when little variation of any input parameter, for example, pressure in the primary circuit under nominal conditions will result in qualitative variation of solution due to either opening or failure to open of the safety valve on the pressurizer.

However, such factors as actuation of the mentioned systems of protection and interlocking, which qualitatively affect the processes in the plant, are the subject of special study in the project and in choosing the conditions for their actuation much higher values of tolerance and conservatism than errors in calculations of parameters given in [1] are provided for.

Conclusions

The results of solution substantiation of numerical schemes stability, sensitivity to input data and determination of error in numerical solution of the system of equation are presented.

In code package "FLOW" energy and coolant hydrodynamics equations, neutron kinetics equations and equations of heat conduction in metalwork and fuel rods

are solved together. Processes described by these equations are characterized by different time constants and during calculation different degree of time and space discretization is required.

The stability of numerical schemes used in "FLOW" program and its modules, assessment of error of numerical solution for nodalization, is demonstrated. Initial differential equations in partial derivative of the law of conservation of mass, energy and momentum are reduced to the system of usual differential equations on the basis of spatial approximation with the use of integro-interpolation method. Approximation was performed by integrating the initial differential equations in partial derivatives within the limits of each elemental volume of division by spatial coordinate.

The usage of integro-interpolation method provides fulfilment of laws of conservation on computational mesh as a whole (conservatism of approximation method) and the second order of approximation by spatial coordinate.

Effect of limiting error of integration in "FLOW" code on pressure values at the core outlet and mass discharge from the leak is not practically distinguished. Effect of these errors on the values of coolant flow rates at the core inlet and outlet becomes more noticeable. Still, maximum value of these differences doesn't exceed 7 % for flow rate at the core inlet. More substantial differences (of the order of 10–15 %) are observed in the course of calculated temperature of fuel rod cladding.

List of abbreviations

DNB - departure from nucleate boiling
ECCS - emergency core cooling system;
MCP - main coolant pipeline;
NPP - nuclear power plant;
RCC - reactor collection chamber;
RP - reactor plant;
RPC - reactor pressure chamber;
VVER - water-cooled water-moderated power reactor.

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