NATURAL CONVECTION AS THE WAY OF HEAT REMOVAL FROM FAST REACTOR CORE AT COOLDOWN REGIMES

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Abstract

The problems of thermohydraulics in fast reactors at cooldown regimes at heat removal by natural convection are considered The results of experiments and calculations obtained in various countries in this area are presented. The special attention is given to heat removal through inter-assembly space in the core and also to problems of thermohydraulics in the upper plenum.

1. INTRODUCTION

The problems of natural and mixed convection in fuel assemblies of reactor cores represent a large practical significance from a point of view of reactors cooling in unnominal and emergency regimes of operation [1 - 3].

At natural convection in installation there are series of thermohydraulic phenomena which are not inherent to forced flow. So, at natural convection velocities and local directions of flow depend on distribution of temperature of the coolant, while at forced convection the velocity determines temperature distribution. The conditions with natural and mixed convection are characterized by thickening of hydrodynamics of a stream: there are reverse flows in fuel assemblies, owing to what there can be additional nonuniformities of temperature and hot spots, increase coefficients of a hydraulic resistance etc.

At the same time experimental and calculated researches of natural convection processes in a core and vessel of the fast reactor undertaken per last years, testify to positive influence of natural convection on processes of heat removal in conditions of fast reactor cooldown. In particular, concept of heat removal by natural convection originating in inter-assembly space of the nuclear reactor at use of let-down heat exchangers in a emergency cooldown system is successfully developed [4, 5].

As to calculated codes intended for estimations of temperature regimes with natural convection, they are verified on an experimental material and to the present time are advanced with reference to thermohydraulic processes in the upper plenum of the reactor, and in core fuel assemblies (inter-assembly space).

For the upper plenum of the reactor the following codes are most advanced: DYANA / ATTICA [6] - one-dimensional and two-dimensional code; TRIO [7], ASTEC [8], FLUTAN [9], AQUA [10] - three-dimensional codes developed in France, England, Germany, Japan; the two-dimensional NECTAR code [11] developed in IPPE etc.

The development of calculated codes for fuel assemblies of reactor cores is founded on two approaches: subchannel technique and model of a porous body. First from them is based to solution of the equations of impulse, mass and energy conservation in channels (cells) formed by adjacent fuel rods, second - on representation of a bundle of rods by an anisotropic porous medium with heat release distributed on volume (a quasi-homogeneous model).

In the report some information about developed in IPPE codes based on a subchannel technique (MID, TEMP, TEMP-M, MIF etc.) is given, and the brief outcomes of calculation obtained by these codes with allowance for heat interaction of core fuel assemblies (coolant leakage in inter-assembly space) are represented. It is necessary to consider enough explicitly

the methods of calculation of temperature fields in fuel assemblies of reactors at cooling through inter-assembly space in cooldown conditions, when approprINPE circulation circuits with natural convection appear (researches in ÎÊBÌ - Russia, in PNC - Japan; the calculated method based on models of a porous body is demonstrated in the latter case).

The problem on calculated research of thermohydraulics at natural convection in the reactor upper plenum conducted in IPPE by the NECTAR code is considered

Largely this report is devoted to exposition of experimental data on natural convection in fuel assemblies obtained in IPPE, as the base data for improvement of calculated codes.

2. EXPERIMENT AS A RESOURCE OF BASE CONSTANTS FOR IMPROVEMENT OF CALCULATED CODES UNDER THE ANALYSIS OF NATURAL CONVECTION PROCESSES IN FAST REACTORS

The evaluation of work of systems with natural convection is much more complicated than with forced flow, and the development of the calculated approaches is hampered because of a great many of parameters defining thermohydraulics in considered cases. Therefore experimental study of dynamics of temperature perturbations in systems with natural convection for concrete situations is necessary. By results of the analysis of base constants obtained in experiments, it is possible to update theoretical methods of the prognosis and evaluation of development of natural convection processes.

Thus it is necessary:

- To understand an essence of thermohydraulic phenomena accompanying natural convection processes;
- To develop a technique of effectiveness estimations of a emergency cooldown system of the fast reactor based on use natural convection; to generalize obtained experimental data used in these evaluations as empirical relations with the purpose of their further use in calculated codes.
- To select the basic direction in development of emergency cooldown systems, distinguished by an increased potency and safety at maintenance, i.e. to optimize a heat removal system connected with natural convection.
- To create a rigorous (high capacities) data base for development and debugging of calculated codes.

With the purpose to study operating regimes of nuclear power installation in conditions of natural circulation the reactor experiments will be carried out, for example [12 - 14]. The obtained data also are a test material for calculated codes of circulation circuit, collected of modules approximately circumscribing a conditions of heat and mass transfer in the circuit elements (core, mixing cameras, heat exchangers, pumps etc.). Thus the improvement of the calculated scheme for concrete arrangement of an equipment, for example [12] is fulfilled. The error of fulfilled calculations, certainly, is determined largely by error of exposition of hydrodynamics and heat transfer of the elements of a circuit, and also of selected common scheme of circulation. The heat and mass transfer processes in the elements of circulation circuit at natural convection conditions are determined by geometry of area, boundary conditions (availability and disposition of heat transfer surfaces), properties of the coolant, availability of free surfaces and other factors. The data retrieval for operation in NC conditions of such important component of the circuit , as the reactor core, is one of primal problems of the substantation of safety of nuclear power installation. The obtained data can also be used for a construction (improvement) of approximate models of calculation in separate modules of calculated codes at the analysis of operation of whole nuclear power installation.

The programs of researches of natural and mixed convection are realized in various centers of science of a series of countries, for example, in Grenoble (France) [15], in Richland (USA) [16], in Germany, England [17], PNC (Japan) [5] etc.

The solution of the task includes:

- Development of problems on modeling of processes with natural and mixed convection in assembly of cylindrical rods (fuel rods);
- Realization of experimental researches of thermohydraulics on models of fuel assemblies in conditions with natural and mixed convection;
- Development of a technique of calculation including mathematical statement with a solution of a boundary problem, system of constants (closing relations), method of numerical solution and creation of calculated code;
- Code testing by an experimental material;
- Realization of calculated researches with consequent generalization of outcomes;

The available work on thermohydraulics at natural convection in rod assemblies testify to two approaches in statement of experiments: the separate channel of rods bundle [18] is reproduced or the experiences are carried out on the whole bundle of rods [19].

Results of experimental researches obtained in IPPE on that and other technique are considered below. In the first case the cell of dense packing of smooth rods was reproduced, in the second case fuel assemblies of cores and shields of fast reactors BOR-60, BN-350, BN-600, BN-800 were simulated (the relative pitch of fuel assemblies simulators varied in an interval $1.04 < s / d < 1.24$).

At the first stage the development of the constants system for the solution of equations of thermohydraulics in regimes with natural and mixed convection in core fuel assemblies of reactors [20] was executed. The realization of experimental researches of temperature fields in model fuel assemblies in indicated modes ($Re \leq 2 \cdot 10^3$) is the second stage.

3. EXPERIMENTAL DATA OBTAINED ON MODEL FUEL ASSEMBLIES OF FAST REACTORS IN CONDITIONS OF COOLANT FLOW WITH LOW VELOCITIES AND AT NATURAL CONVECTION

3.1 Basic performances of model fuel assemblies

Model fuel assemblies of fast reactors consist from 37 fuel elements (simulators of fuel elements), located in a triangular lattice with a relative pitch $1.04 < s / d < 1.24$ and made in hexagonal cover. The assemblies with $s / d = 1.04; 1.062; 1.10; 1.15; 1.24$ were used. In the fig. 1. as an example the cross-section of the model fuel assembly of the BOR-60 reactor ($s / d = 1.10$) is shown. Model fuel assemblies of other reactors have a similar construction. The heating of the elements is carried out from nichromium heaters ensuring a constant heat flow on an interior surface of the elements and on length of energy release. The central, side and angular elements are made out rotary. On their surface are fixed till 12 Cr - Al thermocouples in covers from steel Õ18Í9Ò (diameter of a cover is 0.8 õ 0.5 mm, diameter of thermal electrodes is 0.2 mm), measuring the surface temperature of elements.

Coolant temperature is measured in collectors of models, and also in each cell at bundle outlet. The Cr - Al thermocouple in covers ∅ 0.8 õ 0.5 mm are used for it.

In the table I the geometric performances of model fuel assemblies of the BOR-60 reactor are represented.

In the table II the distances l from a beginning of energy release are given, on which the junctions of thermocouples for the same model are fixed.

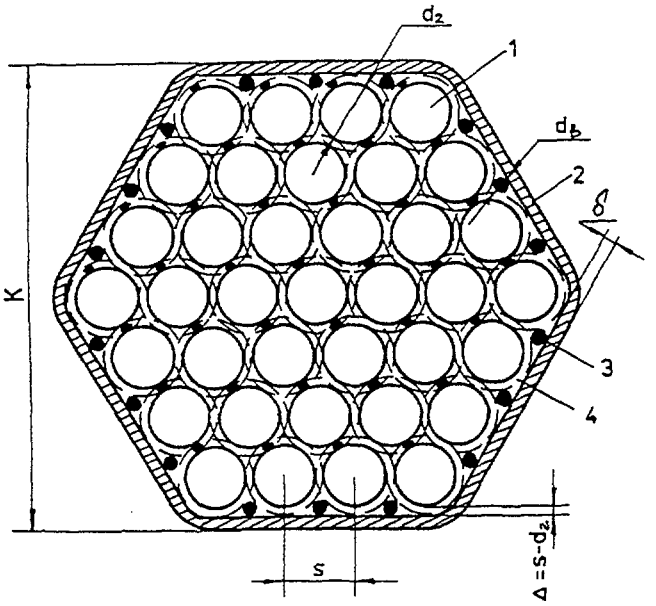


Fig.1. Cross section of the model fuel assembly BOR-60: 1 - simulator of fuel element; 2 - barrel; 3 - cylinder ; 4 - wire wrapper.

TABLE I. THE GEOMETRIC PERFORMANCES OF MODEL FUEL ASSEMBLY OF THE BOR-60 REACTOR

1	Name	Label	Dimen-sionality	Numerical value
1.	Exterior diameter of fuel element simulator clad	d = 2R	mm	17.4
2.	Interior diameter of fuel element simulator clad	d_1	mm	13.0
3.	Relative pitch of disposition of fuel element simulators	s / d	-	1.1
4.	Length of energy release	l_0	mm	900
5.	Distance between face lattices	L	mm	950
6.	Wire wrapper pitch	h	mm	317
7.	Diameter of the wrapping wire	$d_ï$	mm	1.5
8.	Interior size of the barrel "on a turn-key basis"	b	mm	119
9.	Thickness of a barrel wall	δ′	mm	2
10.	Diameter of cylinders	$d_â$	mm	4.5
11.	Area of cross-section of cells:			
	central	$\omega_ö$	mm^2	38.9
	side	$\omega_à$	mm^2	64.0
	angular	$\omega_ó$	mm^2	11.5
12.	The moistened perimeter of cells:			
	central	$Ð_ö$	mm	19.6
	side	$Ð_à$	mm	62.9
	angular	$Ð_ó$	mm	11.0
13.	Hydraulic diameter of cells:			
	central	$d_{ãö}$	mm	5.23
	side	$d_{ãà}$	mm	4.07
	angular	$d_{ãó}$	mm	4.2

TABLE II. THE DISTANCES FROM A BEGINNING OF ENERGY RELEASE OF ROTARY FUEL ELEMENT SIMULATORS, ON WHICH JUNCTIONS OF THERMOCOUPLES ARE FIXED ($D_Ã$ - HYDRAULIC DIAMETER OF CENTRAL CELLS)

[1] thermocouples of fuel element simulators			l	$l / d_ã$
angular	side	central	mm	($d_ã$=5.2 mm)
1	38	20	5	0.96
2	39	21	85	16.4
3	40	22	165	31.2
4	41	23	245	47.0
5	42	24	325	62.5
6	43	25	405	78.0
7	44	26	485	93.0
8	45	27	565	109
9	46	28	645	124
10	47	29	725	140
11	48	30	805	155
12	49	31	885	170

3.2. Experimental model

The experimental model is installed on a sodium or sodium - kalium circuit of a bench 6Á [21, 22].

Circulation of liquid metal (sodium, an sodium - kalium alloy: 22 % Na + 78 % K) is carried out by the electromagnetic pump of an alternating current by efficiency ~ 150 m^3/h (construction see in [22]).

The heat removal released in a model, is carried out in the heat exchanger metal - water with an intermedINPE cavity, completed helium or working under vacuum. A construction of the heat exchanger also see in [22].

The regulation of flow rate of liquid metal is carried out by the electromagnetic pump or valves (by silphon - alloy Na - K; with frozen compacting - sodium).

3.3. Technique of realization of experiments

In steady-state operation of the circuit the temperature fields of the characteristic elements (central and near a wall), and also coolant temperature in each cell of a bundle are measured. The turn of the elements in an interval of angles 0 - 360^0 through $\Delta\varphi$~5 - 10^0 is carried out.

The measurements are carried out with use of a measuring device IS-200 permitting to produce sequential pick up of information from 200 sensors.

In experiences a force of current, voltage and electrical power of each element of a model are measured. The measuring complex Ê-50 is used.

For smoothing loads of heaters the additional resistance including in a circuit of each heater is used.

In a table III the performances of experiences conducted on the BOR-60 model assembly are represented.

TABLE III. THE CHARACTERISTICS OF EXPERIMENTAL MODES

№	Velocity $\overline{w}$ m/s	Flow rate G m^3/h	Temperature at assembly outlet $t_{вых}$ ^0C	Heating of the coolant Δt ^0C	Power of the simulator of fuel element $N_{им}$ kW	Thermal Flow $\overline{q}$ $kcal/m^2h$	The Reynolds number* Re	The Peclet number* Pe
1	0,0279	0,335	285	135	0,41	7160	294,9	1,75
2	0,0416	0,497	241	91	0,41	7160	439,7	2,60
3	0,0576	0,688	214	64	0,40	7000	608,8	3,60
4	0,0848	1,014	199	48,9	0,45	7870	896,3	5,30
5	0,112	1,339	187	37	0,45	7870	1183,7	7,0
6	0,140	1,673	180	30	0,45	7870	1479,7	8,75
7	0,154	1,84	177	27	0,45	7870	1627,6	9,6
8	0,168	2,00	174,7	24,8	0,45	7870	1775,6	10,5
9	0,197	2,35	171	21,1	0,45	7870	2082,1	12,3

* - The Reynolds and Peclet numbers are calculated on mean speed of the coolant on cross-section of assembly (u) and hydraulic diameter of central cells ($d_{äö}$)

3.4. Results of experiments and their analysis

3.4.1. Some effects in model fuel assemblies of reactors connected to natural convection

Assuming that maximum nonuniformity of temperature of a wall on a perimeter of fuel elements near a wall $\Delta T_W^{max} = \frac{t_w^{max} - t_w^{min}}{\overline{qR}} \lambda_f$ (the fig. 2) is determined by a difference of heating of the coolant in approprINPE channels, and making simple balance relations for adjacent streams of the coolant in channels in the assumption, that between streams the fluid is not intermixed and does not exchange a heat, we shall receive an inversely proportional assotiation between a difference of heating of the coolant in channels and the Peclet number:

$$\frac{\Delta t_{III} - \Delta t_{II}}{\overline{qR}} \lambda_f = \pi l_i \left(\frac{d_{rIII}}{Pe_{III} \omega_{III}} - \frac{d_{rII}}{Pe_{II} \omega_{II}} \right) = \frac{\pi l_i}{Pe} \left(b \frac{d_{rIII}}{\omega_{III}} - c \frac{d_{rII}}{\omega_{II}} \right) \quad (1)$$

Here Δt_{III} and Δt_{II} - average heating of the coolant in central (III) and near-a-wall (II) channels of fuel elements assembly (it is supposed, that in channels II the coolant is not heated up on comparison with channels III) l_i - current distance from a beginning of energy release; d_{aIII}, d_{aII}, ω_{III}, ω_{II} - hydraulic diameters and cross-sectional areas of channels III and II; $Pe_{III} = \frac{w_{III} d_{rIII}}{a}$, $Pe_{II} = \frac{w_{II} d_{rII}}{a}$, $Pe = \frac{\overline{w} d_{rIII}}{a}$ - the Peclet numbers calculated at mean speeds in channels III and II and at mean speed on section of fuel assembly ($\overline{w}$); b and c - factors connecting speeds w_{III}, w_{II} and $\overline{w}$.

The increase of nonuniformity of temperature near-a-wall fuel elements with decreasing Pe can not be boundless because of heat transfer between channels, which is proportional to difference of heating in channels. In the field of low speeds the natural convection promoting intermixing of fluid between channels and reducing temperature

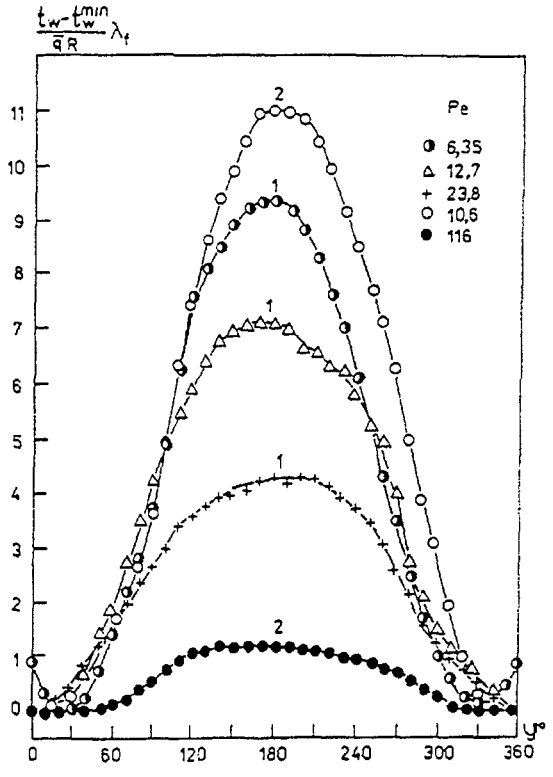


Fig. 2. The temperature fields of near-a-wall wire-wrapped (1) and smooth (2) simulators of fuel elements of the model fuel assembly BOR-60.

nonuniformity is exhibited. It determines peaks in regularities $\Delta T_w^{max} = f(Pe, Re)$ at some Peclet (Reynolds) numbers - fig. 3, 4 [23 - 25].

Position and size of peaks depend on geometry of a channel, physical properties of the coolant, the distances from a beginning of energy release and other factors. At fixed geometry of a channel the numerical values of peaks are about identical on various distances from a beginning of energy release, but the displacement of peaks to area of increased Ðå (fig. 3) happens at increasing $l/d_э$. Decreasing of peaks at the end of a zone of energy release (fig. 4) is probably also.

The influence of natural convection has an effect for character of behavior of temperature nonuniformities of fuel elements on length of energy release (fig. 5). For example, in model fuel assembly of the BOR-60 reactor the temperature nonuniformity of fuel elements is increased linearly on length of energy release at large Peclet numbers (Ðå ≥100); in the field of small Ðå the tendency to stabilization is exhibited, at some Ðå the stabilization comes, and at the very small Peclet regularities lose monotone character because of influence of natural convection.

Let's consider the data on distribution of coolant heating in channels of model fuel assembly at small Peclet numbers. P.A. Ushakov has shown, that [26]: if there is no natural convection, with decreasing of the Reynolds number the temperature nonuniformity of fluid on radius of assembly should be aimed to constant value characteristic to the given kind of fluid and given geometry of assembly. Really, by assuming, that two fluid flows with flow rates v_{III} è v_{II} ($v_{III} < v_{II}$) flow past, not mixing up among themselves, in central (III) and side (II) channels of fuel assembly, and heat transfer exists between channels by means of heat conduction of fluid, we shall receive distribution of mean temperatures of flows on length z as:

$$t_{f\,III} - t_{f\,II} = c_1\left(1 - e^{-c_2\frac{z}{d_{r\,III}Pe}}\right), \qquad (2)$$

where $c_1 = \frac{qh}{\lambda}\left(\frac{1-\beta}{1+\beta}\right)$, $c_2 = (1+\beta)\frac{4d_{r\,III}}{h}$, $\beta = \frac{v_{III}}{v_{II}}$, $Pe = \frac{\overline{w}d_{r\,III}}{a}$; h - the characteristic size in a plane, perpendicular axis z.

It is visible that at Ðå → ∞ the difference of temperature $t_{f\,III} - t_{f\,II}$ aims to zero, and at decreasing the Peclet (Pe → 0) - to a constant $\tilde{n}_1$.

From results of experience follows that in the field of small Reynolds numbers (Peclet) with decreasing Re in assembly equalization of temperature of fluid happens(fig. 6), that is connected to development of natural convection. In moved apart assembly of rods (s / d = 1.24) conditions for development of natural convection are better, than in tight assembly (s / d = 1.062; 1.15) - the equalization happens at bigger Re (smaller heating). The origin of peaks in tight assembly is possible only in area very small Re: in model assembly of shields of reactors (s / d = 1.04; 1.062) peaks are not observed even at very small Reynolds numbers.

In assembly of wire-wrapped rods a condition for a development of natural convection it is worse, than in assembly of "smooth" rods (the part of cross section of assembly is overlapped by edges), - the equalization happens at smaller Re (bigger heating).

With decreasing of the Prantl number peaks displace in area of smaller Reynolds numbers (see data for sodium and alloy of a sodium - kalium on the fig. 6).

In a regime of transient current (1000≤Re≤4000) the redistribution of the flow rate of the coolant on cross-section of fuel elements assembly can happen, if the channels of assembly have various geometry [25 - 27]. It is stipulated by that the flow in such channels is with various Reynolds numbers. By assuming, that in central ("ö") channels of fuel assembly

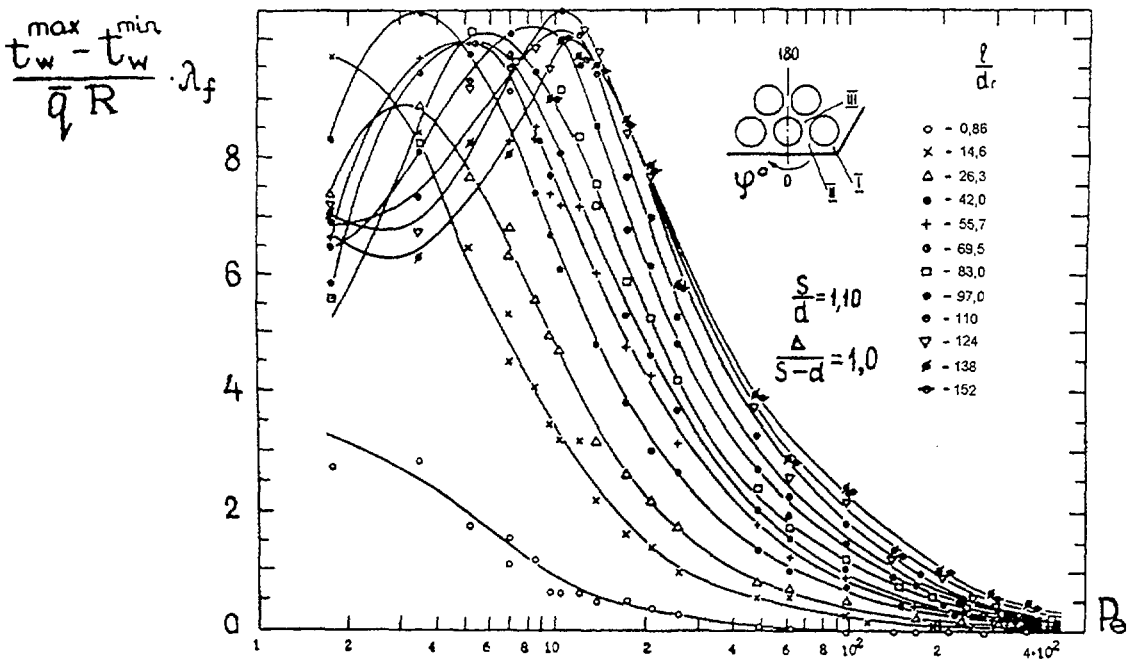


Fig. 3. Change of maximum nonuniformities of temperature of the lateral fuel element simulator of the model fuel assembly BOR-60 from the Peclet number.

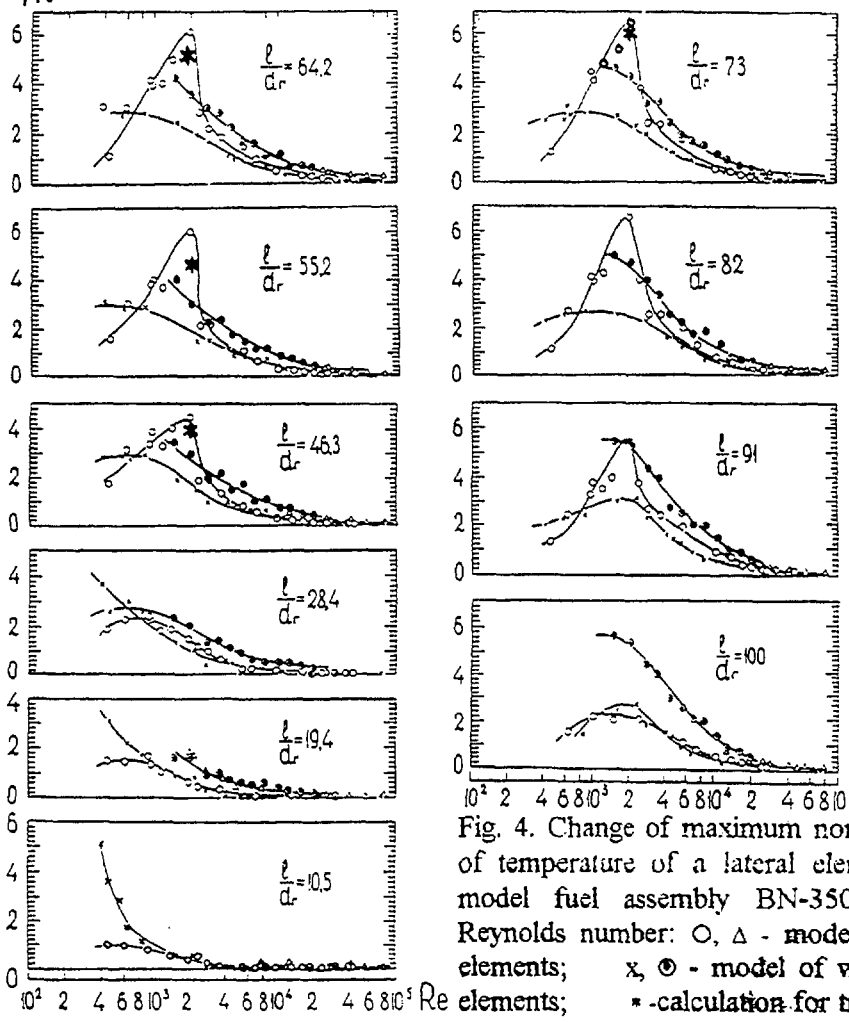


Fig. 4. Change of maximum nonuniformities of temperature of a lateral element of the model fuel assembly BN-350 from the Reynolds number: ○, △ - model of smooth elements; ×, ⊙ - model of wire-wrapped elements; * -calculation for transient flow regime.

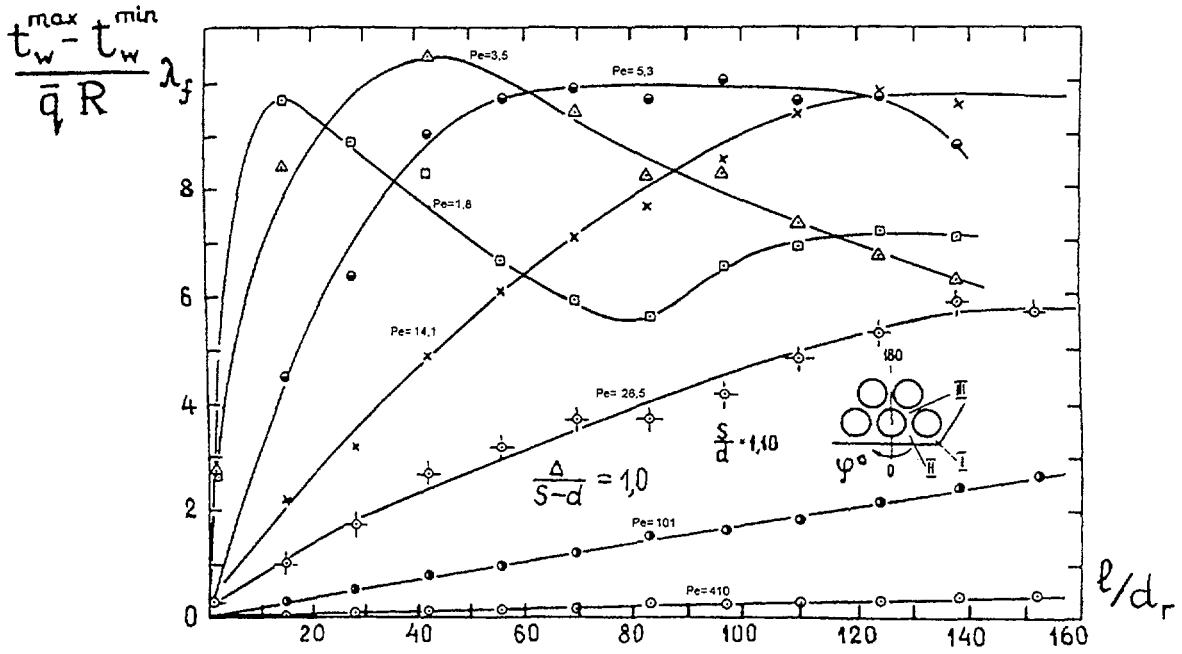


Fig. 5. Change of maximum nonuniformities of temperature on length of the near-a-wall smooth simulator of the fuel element of the model fuel assembly BOR-60.

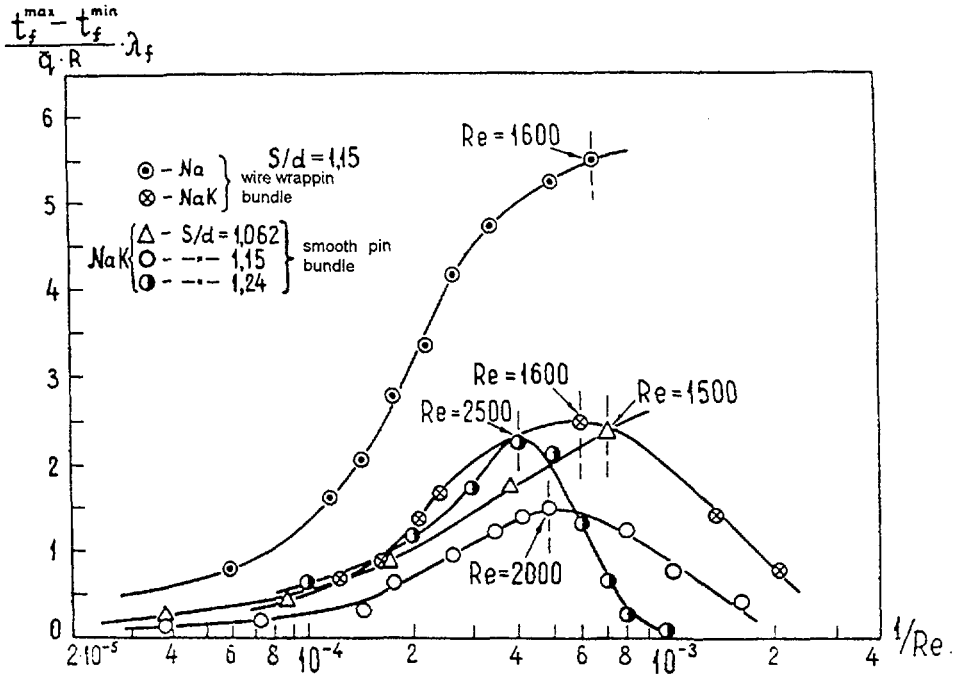


Fig. 6. Change of maximum nonuniformities of temperature of fluid in cross-sections of a model fuel assembly in a function 1 / Re.

turbulent current has a place, and in circumferential (" ï ") - laminar at general transient current in fuel assembly, we shall receive a nude fact, that the redistribution of speed is connected to a measure of difference of hydraulic resistance at turbulent ("ò") and laminar ("ë") currents of the coolant:

$$w_п = w_ц \sqrt{\frac{d_{гп}}{d_{гц}}} \cdot \sqrt{\frac{\xi_{цт}}{\xi_{пл}}} \qquad (3)$$

Above mentioned calls additional nonuniformity of temperature in fuel assemblies. For example at model fuel assembly outlet of the BN-350 reactor temperature of fluid varies in such manner that the maximum heating of fluid are observed not in center of assembly but on some distance from center (fig. 7). Temperature of fuel elements located near to a cover of fuel assembly exceeds temperature of central fuel elements. These fuel assemblies have the greatest nonuniformity of temperature on a perimeter also, as fuel elements are under influence of the greatest gradients of temperature in fluid.

The calculation of distribution of flow rate of the coolant on channels of assembly in transient area of current by using a hypothesis of isobaric cross-section (Δp=idem) should take into account various regimes of flow in adjacent channels, while at a homogeneous regime of flow on cross-section of assembly usually relations are correct:

$w \sim d_э^{0.714}$ for a turbulent mode of flow,

$w \sim d_э^2 / A$ for a laminar mode of flow.

À - the coefficient dependent on the form of a channel.

Evaluations of temperature nonuniformities of fuel elements and fluid in fuel assembly of fast reactors in regimes with small Peclet numbers the including regimes with natural convection, is necessary to make with use of the above-stated graphic relations. The marked effects connected to natural convection, should be taken into account at development of approprINPE computational codes.

3.4.2. Analysis of effects connected to natural convection in fuel assembly of reactors, with the help of the modified Grashof criterion

At Pr << 1 (liquid metal coolant) significant influence of a molecular thermal conduction at natural convection is spreaded far from area of a near-a-wall stratum. In this case at the analysis of experimental data it is expedient to use a criterion $GrPr^2$ [28]. This criterion is recommended by Kutateladze S.S. for natural convection of liquid metals from the fact of significant influence of molecular heat transfer.

By entering for our case the modified Grashof number

$$Gr^* = \frac{g\beta q_w l^4}{\lambda_f \nu} \qquad (4)$$

(here g - acceleration of gravity, β- coefficient of the volumetric extension, q_w - density of heat flow on a surface of the fuel element (of the simulator), l - axial coordinate counted from a beginning of energy release; λ_f, ν - thermal conductivity and kinematic viscosity of the coolant accordingly), Matjuhin N.M. has made attempt to analyze with the help of criterion Gr^*Pr^2 experimental data explained in section 2.4.1.

In the fig. 8. the local values Gr^*Pr^2 found on q_w and l in twelve cuts on length of a zone of energy release (see approprINPE values $l / d_э$ in tables II) are shown. Two zones are chosen there: zone A - range of a modification Gr^*Pr^2 in experiences of the authors [19] and zone Á - range of a modification Gr^*Pr^2 in experiences of the authors [18].

With use of local values of a parameter Gr^*Pr^2 the relations ($\Delta T_w^{max} = f(Gr^*Pr^2)$) for wire-wrapped (fig. 9) and smooth (fig. 10) elements of model assembly BOR-60 are constructed at various values of the Peclet number.

In relations $\Delta T_w^{max} = f(Gr^*Pr^2)$ the following singularities take place:

- At Đå < 100 for wire-wrapped side elements (fig. 9) and Đå < 10 for the smooth side elements (fig. 10) two ranges of modification ΔT_w^{max} in function of growth Gr^*Pr^2 are observed - at first the magnification ΔT_w^{max} up to some "limiting" value, characteristic for fixed Đå, and then - falling ΔT_w^{max} at large values of Gr^*Pr^2 parameter;
- In accordance with magnification of the Peclet number a displacement of a "limiting" value ΔT_w^{max} to area of bigger values Gr^*Pr^2 happens; and at Đå = 370 for the wire-wrapped elements (fig. 9) and Đå = 26.5; 100 for the smooth elements (fig. 10) limiting values ΔT_w^{max} are not reached in an investigated interval of modification of the Gr^*Pr^2 parameter.

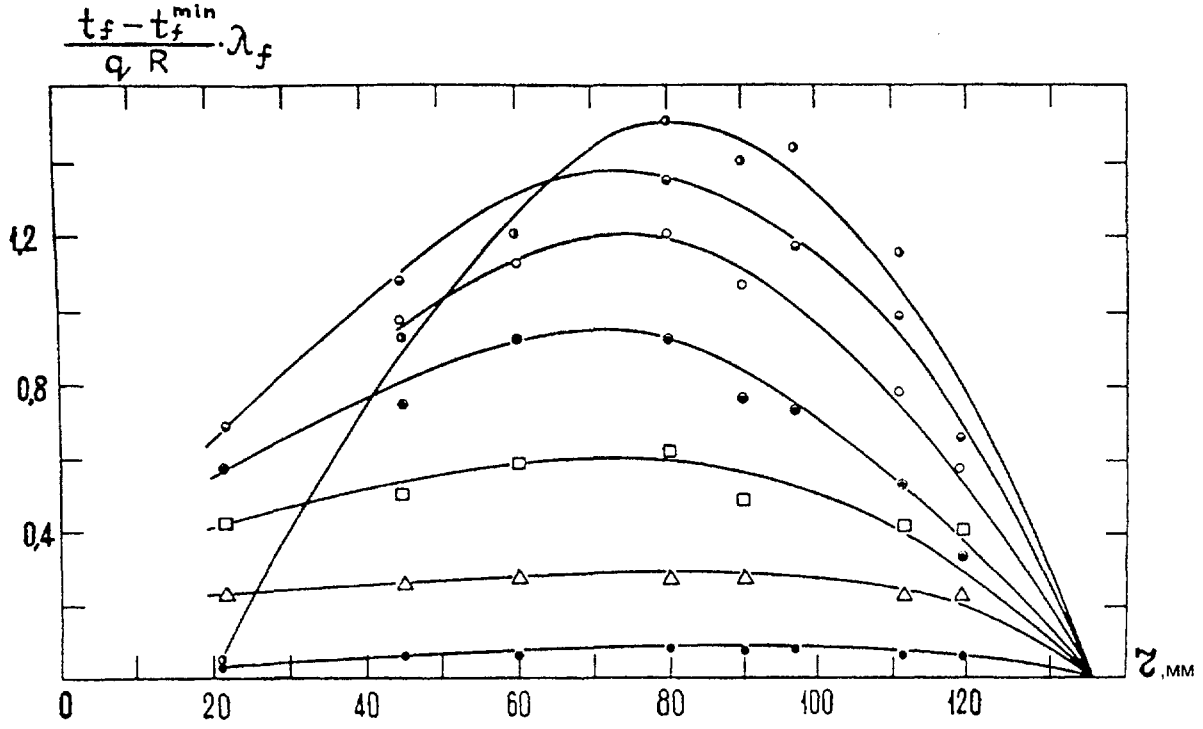


Fig. 7. Radial distributions of fluid temperature in a model fuel assembly BN-350: ○, △, □, ○, ○, ○, ○ - Pe = 568, 179, 141, 104, 80, 67, 53 accordingly.

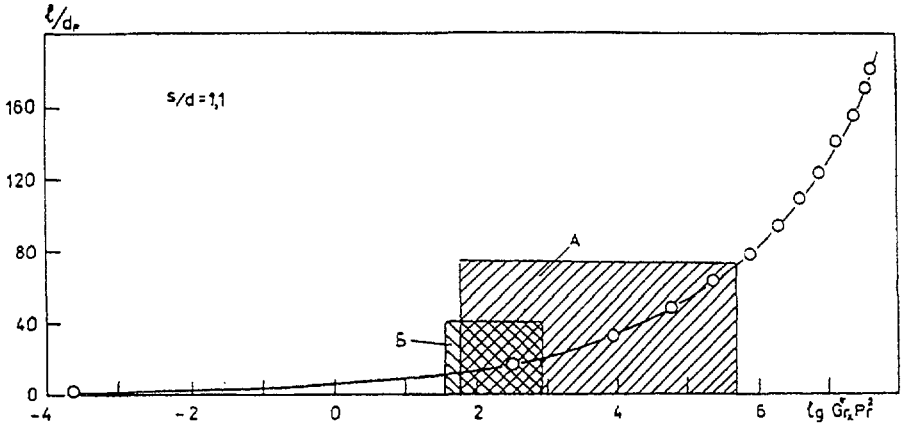


Fig. 8. Values Gr^*Pr^2 on length of a segment of energy release of the fuel element simulator of the model fuel assembly BOR-60.

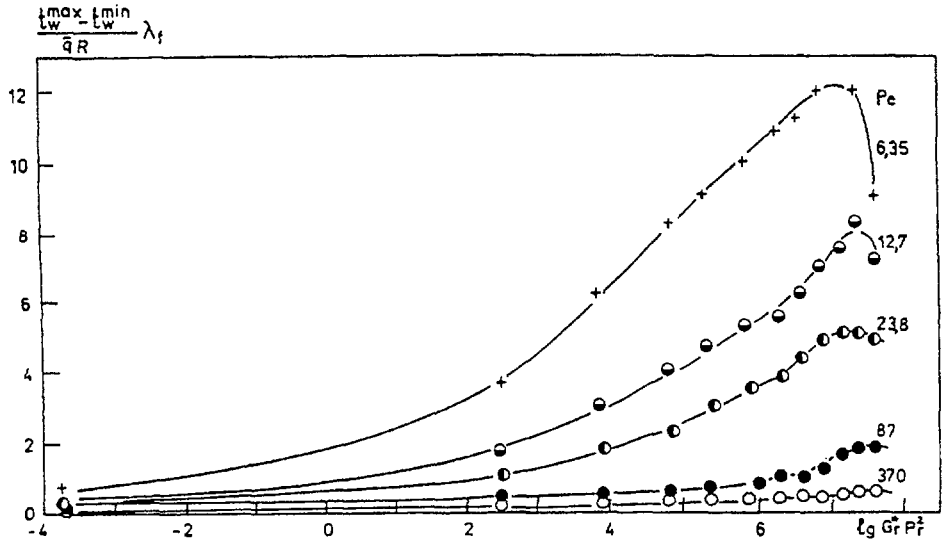


Fig. 9. Relation of maximum temperature nonuniformity of the near-a-wall simulator of fuel element of the model fuel assembly BOR-60 from Gr^*Pr^2.

– The limiting values ΔT_w^{max} are about identical to the smooth side elements for various numbers Ðå and they are $\Delta T_w^{max} \approx 10$ (fig. 10).

4. ABOUT SOME RESULTS OF RESEARCHES OF HEAT TRANSFER TO AN ALLOY SODIUM-KALIUM IN A CELL OF DENSE PACKING OF RODS AT MIXED CONVECTION IN IPPE [18]

In light of the above-stated data for natural convection in model fuel assembly of reactors we shall mark some results of thermal researches obtained by V.I.Koljaskin, L.Ê. Êudrjavtceva and P.À.Ushakov at natural convection in a cell of dense packing of rods and treated in function of the $GrPr^2$ parameter (for a defining size in the Grashof criterion the hydraulic diameter of the channel $d_ã$ is adopted, for a defining difference of temperatures - temperature pressure wall - fluid). These results correspond with the above-stated data in some measure.

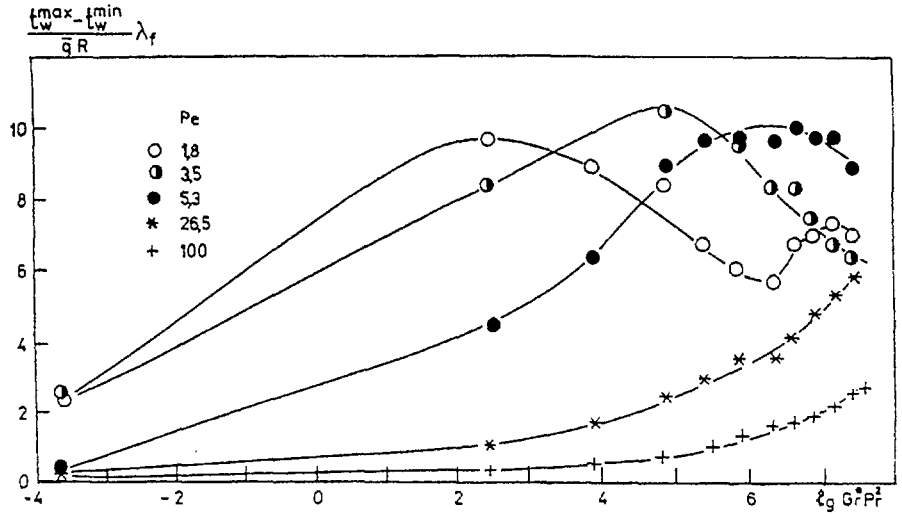


Fig. 10. Relation of maximum temperature nonuniformity of the near-a-wall simulator of fuel element of the model fuel assembly BOR-60 from Gr^*Pr^2.

So the dimensionless nonuniformity of temperature of a wall of the channel $\Delta T_w^{max}=(t_w^{maax}- t_w^{min})\lambda_f / \bar{q}R$ decreases with magnification $GrPr^2$ at a parallel mixed convection on the average on 20 %, at counter - on 35 % (fig. 11), and the temperature fields in a coolant flow react to natural convection a little more. This fact corresponds to zones of ΔT^{max} diminution in the fig. 9, 10 in experiences with model fuel assembly of reactors.

The significant growth of heat transfer is obtained at parallel mixed convection with magnification $GrPr^2$ at numbers Ðå < 100 (fig. 12). At Ðå > 100 influences of natural convection sharply decrease, and at Ðå > 400, as the authors [18] mark, becomes commensurable with an exactitude of experiments.

5. NATURAL CONVECTION IN INTER-ASSEMBLY GAPS OF FAST REACTORS CORES

5.1.Calculated researches in OKBM [4]

At power failure of installation in the fast reactor the stopping of pumps of the first and second loops happens, the channels of system of emergency reactor cooldown are

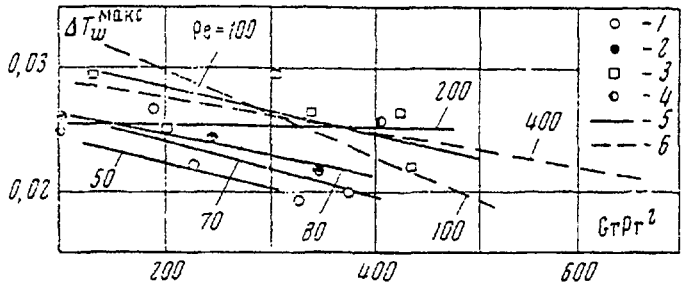


Fig. 11. Influence of natural convection to dimensionless temperature nonuniformity on a perimeter of a channel. Pe : *1* - 70; *2* - 80; *3* - 100; *4* - 190; *5* and *6* - accordingly for parallel and counter mixed convection.

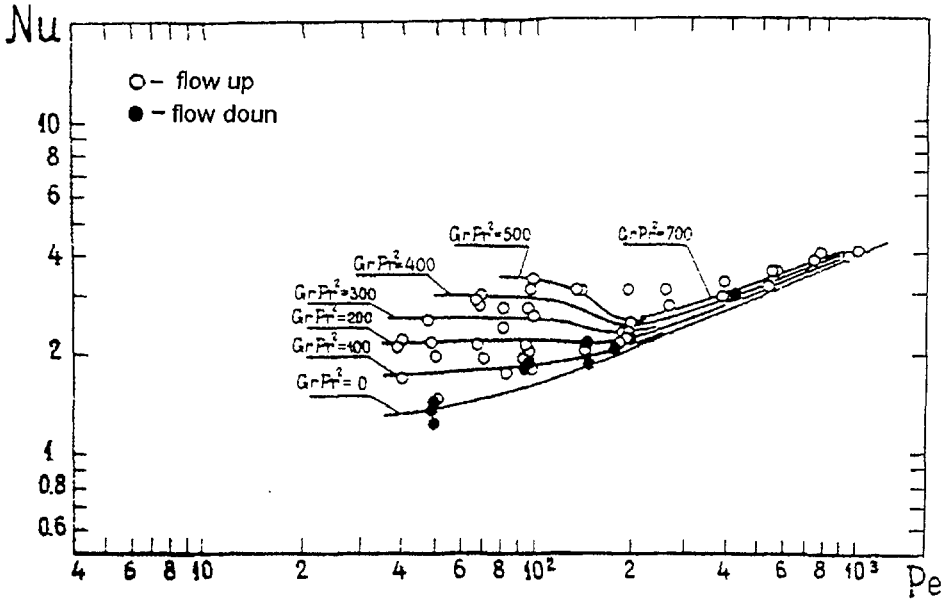


Fig. 12. Heat transfer in a channel of densely-packed bundle of rods.

included and the "cold" sodium with outlet temperature from let-down heat exchanger $t_{âûõ} \approx 350^0$Ñ (for system of emergency reactor cooldown with let-down heat exchangers for the project of the upgraded fast reactor BN-600Ì [4]) merges in the bottom of "hot" cavity of the plenum and begins to fill in inter-assembly space. Sodium in inter-assembly space is heated up because of higher temperatures inside fuel assembly (especially in case of small flow rate inside fuel assembly), a circuit of natural circulation appears: the let-down heat exchanger inlet - the let-down heat exchanger outlet - bottom of "hot" plenum - inter-assembly space - space between heads of fuel assembly and the above core construction - the heat exchanger inlet.

The less flow rate of sodium inside fuel assembly, the bigger role the flow rate in inter-assembly space plays in relation to temperature field inside fuel assembly, cooling sodium and fuel elements of fuel assembly. The moment can occur, when the coolant flow rate inside fuel assembly will become less, than in inter-assembly space, then the role of natural circulation through inter-assembly space will be defining.

If to consider a limiting case of blockage of the coolant input to fuel assembly, inside fuel assembly the regime of U-figurative flow called in the literature "thermosiphon" (fig. 13) will take place. This regime was analysed and calculated by V.À.Sobolev, according to which brief exposition of physics of a considered appearance is given below.

If there is heat release in fuel assembly and leakage of sodium in inter-assembly space the gradient of temperature on a radius of fuel assembly (in a center of fuel assembly temperature is higher, than on a rim) is created, owing to what there is a pressure of natural convection: the sodium in a center of fuel assembly floats up, leaving through orifices in the head of fuel assembly, and to it place a more cold sodium through the bottom of windows of the head of fuel assembly goes(from inter-assembly space, from a let-down heat exchanger after rise of a level "cold" sodium up to heads of fuel assembly; the flows mix up), being lowered along walls of fuel assembly. The lowered pole of sodium is warmed up from a hotter flow in a center of fuel assembly. However heating is not too great because of good heat transfer through a wall of fuel assembly and a big sodium flow rate in inter-assembly space , exceeding the sodium flow rate in a "thermosiphon" circuit($G_{ïîï} > G_{òðèñ}$).

The calculations are carried out at reviewing by one fuel assembly of core with allowance for flow rate of natural circulation in inter-assembly gaps: the part of fuel assembly on a height from a beginning of energy release zone to absorber rods bundle outlet is selected, the cut of a chosen part of fuel assembly is devided into 12 parts in a transversal plane; the task includes 619 elements (sodium, a cover of a fuel element, fuel cores, a cover of an absorber rod, cores of absorber rods). The task of searching of temperature field is solved at specific heating rate, component 2 % from nominal, both specific temperature and sodium flow rate on inter-assembly space inlet.

5.2. Researches in IPPE

In IPPE calculated and experimental researches of temperature fields with reference to fuel assemblies of fast reactors under nonadiabatic conditions on covers of fuel assemblies (model assemblies), stipulated by leakage of the coolant in inter-assembly space, including at natural convection were carried out [26, 29]. The subchannel technique of thermohydraulic calculation of fuel assemblies (groups of interacting fuel assemblies by means of inter-assembly space) was developed, calculated codes was verified by means of obtained experimental material.

5.2.1. Subchannel technique and calculated codes

The subchannel technique of calculation of temperature regimes of fuel assemblies is published in a series of papers, in particular in [29, 30], therefore there is no necessity in it

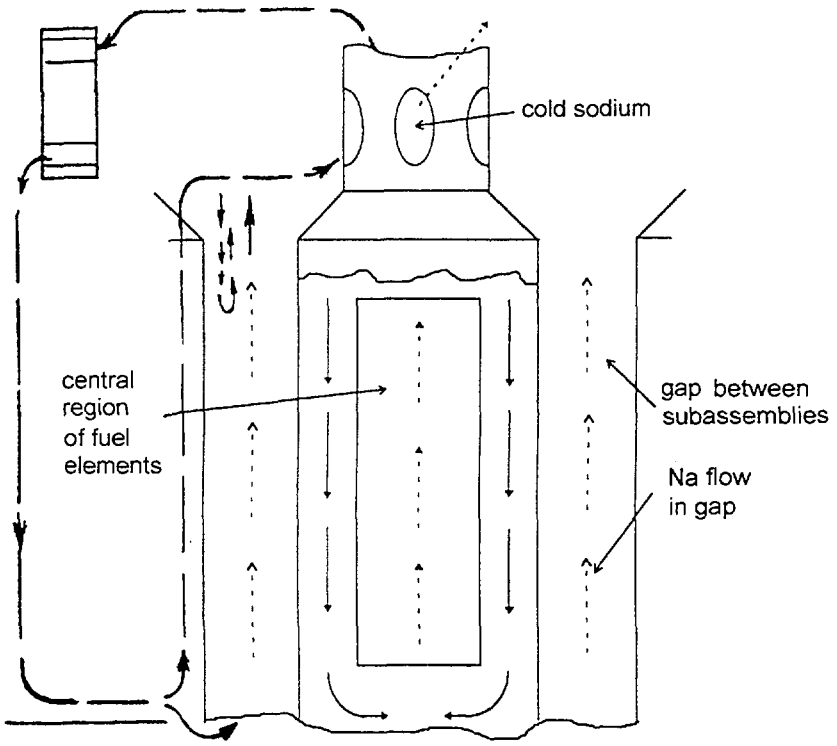


Fig. 13. The scheme of "thermosyphon" flow at blockage of fuel assembly inlet of the fast reactor.

reviewing. Let's mark only, that the whole population of channels in fuel assembly and in inter-assembly space (fig. 14, 15) is considered at an approprINPE solution of the balance equations. Such channels are: interior (fig. 14, á); interior, contiguous to peripheral (fig. 14, â); side, angular; channels in gaps between fuel assemblies (fig. 15). In assembly the interchannel exchange happens because of wire wrapper (fig. 16) and of other mechanisms of transfer.

As an example we shall consider an energy balance equation for channels in inter-assembly gaps (index ξ) in case of the stabilized flow of the coolant at wire-wrapped fuel elements by indirectional wire wrapper within the framework of integral model of interchannel exchange:

$$\frac{d}{dZ}\left(W_\xi T_\xi \Omega_\xi\right) = \sum_{i=1}^{2} \mu_{i\xi}^{T} L(T_i - T_\xi) + \sum_{i=1}^{2} \frac{\delta_{\text{заз}} / s}{Pe}\left(T_i - T_\xi\right) \qquad (5)$$

Here $Z = \frac{z}{L}$, $W_\xi = \frac{w_\xi}{\overline{w}}$, $T_\xi = \frac{t_\xi c_p \rho \overline{w} d_{r0}}{4\overline{q}L}$, $T_i = \frac{t_i c_p \rho \overline{w} d_{r0}}{4\overline{q}L}$, $\Omega_\xi = \frac{\omega_\xi}{\omega_0}$; $Pe = \frac{\overline{w} d_{r0}}{a}$;

z - longitudinal coordinate, L - length of energy release, w_ξ - speed in a channel of an inter-assembly gap, $\overline{w}$ - mean on section of fuel assembly speed of the coolant, $d_{ā0}$ - hydraulic diameter of central channels, $\overline{q}$ - mean heat flow from a surface of fuel elements; ω_ξ- cross-sectional area of a central channel, $\delta_{çàç}$ - width of an inter-assembly gap, s - distance center to center of fuel elements; $ñ_ô$, ρ, à and - thermal capacity, density, temperature conductivity of the coolant.

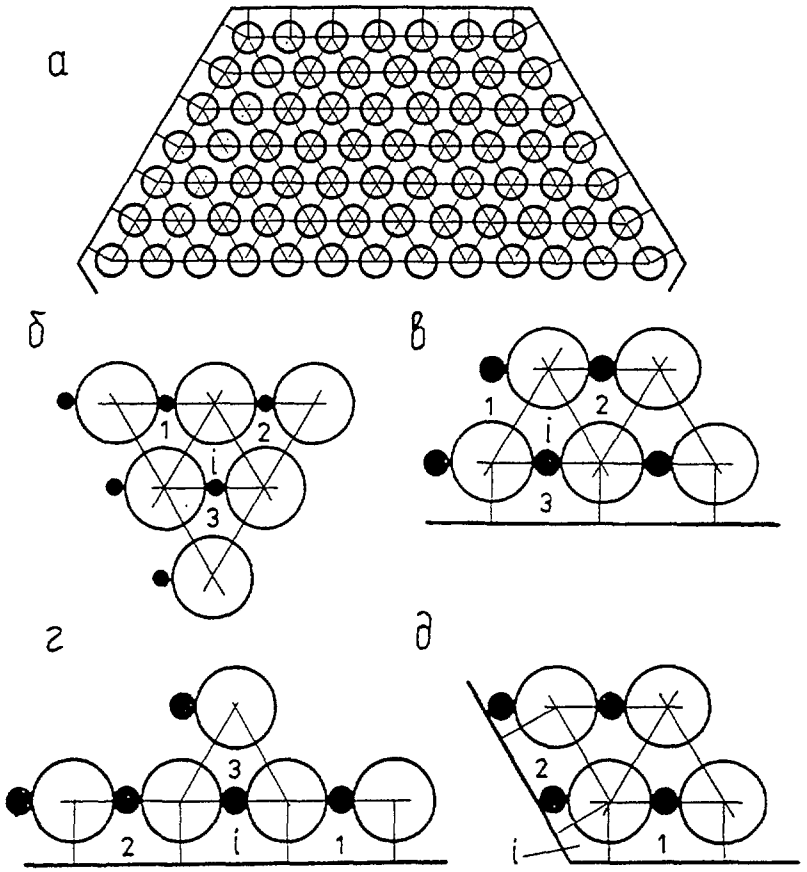


Fig. 14. The scheme of breakdown of cross section of hexagonal fuel assembly (а) on characteristic channel banks (б, в, г, д).

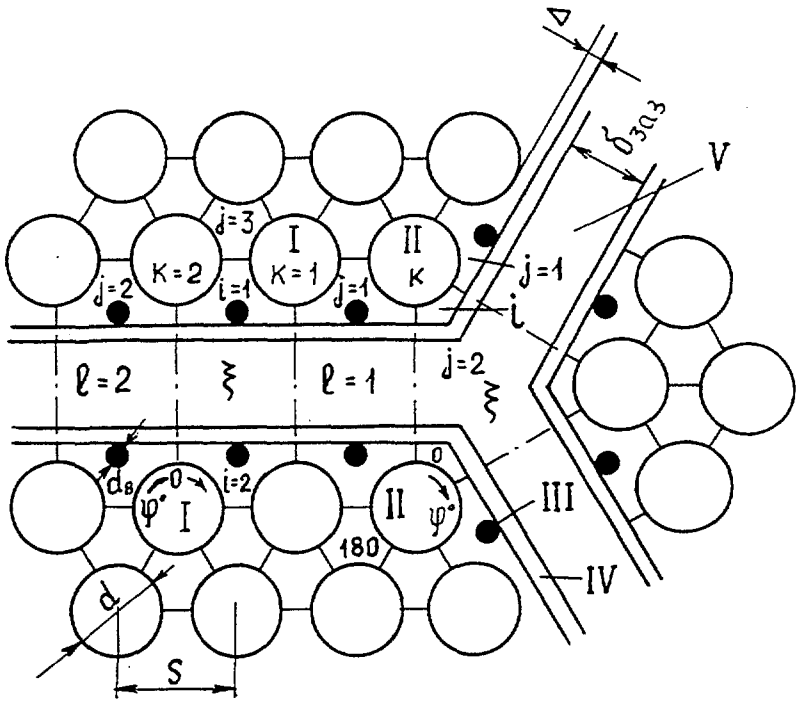


Fig. 15. The scheme of fuel elements arrangement in circumferential zones of adjacent fuel assemblies and breakdown of zones on characteristic channels: I - lateral fuel element, II - angular fuel element, III - cylinder , IV - cover of fuel assembly, V - inter-assembly gap.

The coefficient of interchannel exchange by heat through a cover of fuel assembly, $м^{-1}$, is determined by relation:

$$\mu_{i\xi} = \frac{8}{\pi} \frac{k_{i\xi} s / d}{\lambda_f Pe}, \qquad (6)$$

where the heat transfer coefficient through a cover of fuel assembly is calculated with allowance for of heat conduction of a cover and coefficients of heat transfer in channels of a near-a-wall zone of fuel assembly and inter-assembly space (about coefficients of heat transfer in different categories of channels of fuel assemblies more in detail see in [29]).

The set of equations (5) is decided together with similar equations for other categories of fuel assembly channels at definition of a system of interchannel exchange constants in fuel assembly. The system of these constants is completed in experiments and is explained in [31, 32].

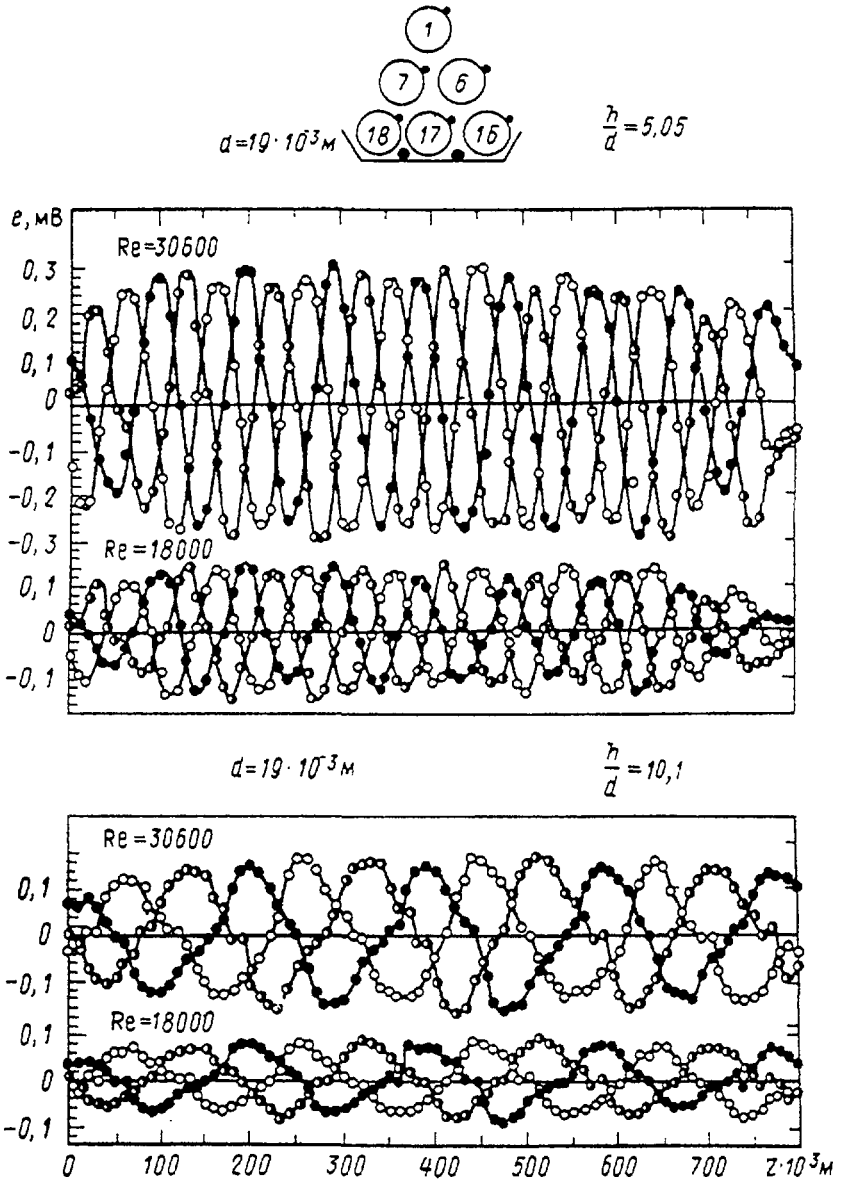


Fig. 16. Change cross-sectional leakage of the coolant (in mW of the sensor) on length of gaps in lattices of wire-wrapped fuel elements of a type "an edge on an cover".

With use of the subchannel technique are developed the following computational thermohydraulic program: GID, TEMP, TEMP-M, MIF, ÌID (non-stationary case). Their detailed definition see in [26, 32].

5.2.2. Results of calculation, comparison to experiment

In general case the effect of non-adiabatic boundary conditions on covers of fuel assembly to temperature field is determined by geometry of fuel assembly, intensity of interchannel exchange in fuel assembly and heat transfer between fuel assemblies, size inter-assembly leakage of the coolant, and also character of energy release field.

The thermal exchange between fuel assemblies can increase or reduce temperature non-uniformity of circumferential fuel elements depending on character of fuel assemblies interaction.

The solution of a set of energy balance equations for group interacting fuel assemblies of the reactor BN-600 with irregular energy release in cross section shows, that the increase of heating of the coolant in adjacent fuel assemblies has a place in the field of low energy release at small leakage of the coolant in gaps between fuel assemblies ($g_{ïð} \le 2\%$) together with some drop of a peak level of coolant heating in area with high energy release. The drop of coolant heating in circumferential areas of fuel assembly (fig. 17) happens because of leakage increase of the coolant in inter-assembly gaps.

At low intensity of heat transfer through a cover of fuel assembly ($\mu^{\dot{O}}_{i\xi} < 1\ i^{-1}$) or at small coolant leakage ($g_{ïð} << 1\%$) heat transfer to the inter-assembly gap unsignificantly influences on a temperature field of fuel assembly. At high intensity of inter-assembly heat transfer ($\mu^{\dot{O}}_{i\xi} > 2\ i^{-1}$) and at large coolant leakage in an inter-assembly gap ($g_{ïð} > 1\%$) the drop of coolant heating on a rim of fuel assembly appears essential. It considerably increases non-uniformity of temperature on a perimeter of circumferential fuel elements. At the same time the heat transfer through a cover of fuel assembly acts on a temperature field of maximum three - four numbers of fuel elements, contiguous to a cover, and can render essential effect on

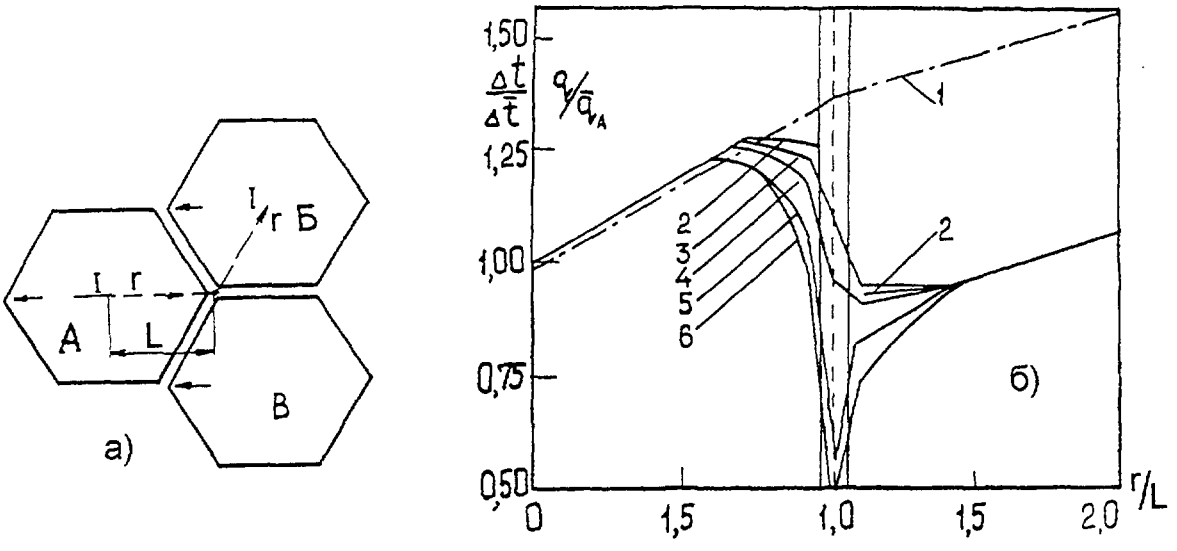


Fig. 17. Distribution of energy release (1) and heating of the coolant in cross section of two interacting fuel assemblies: 1 - calculation disregarding of inter-cartridge heat transfer; 2 - 6 - calculation with allowance for of inter-cartridge heat transfer at the coolant flow rate in inter-cartridge gaps, component 0; 0,02; 0,04 of coolant flow rate through fuel assembly and equal ∞ accordingly; а) - arrangement of adjacent fuel assemblies; б) - results of calculation in a direction r.

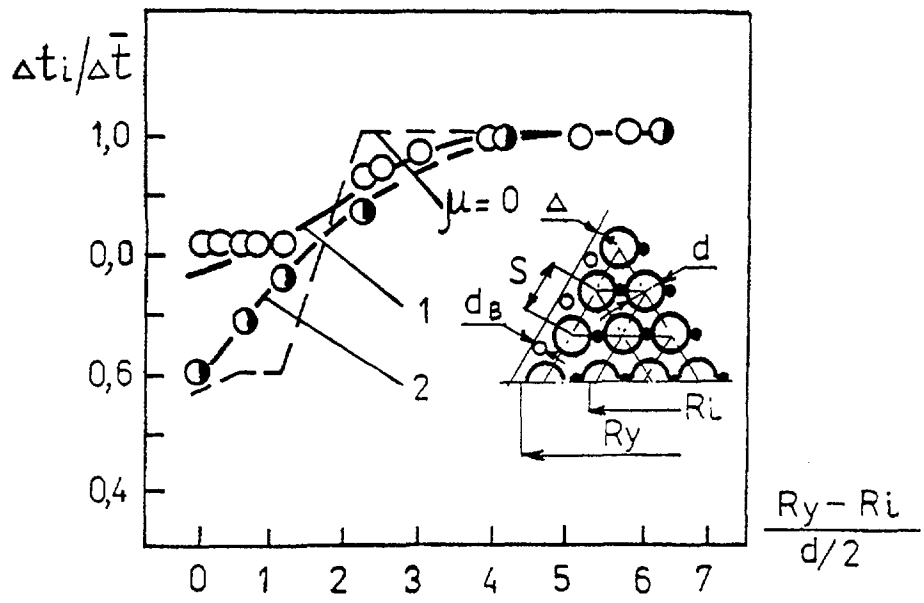


Fig. 18. Distribution of coolant heating on a radius of model fuel assembly of the BOR-60 reactor under adiabatic conditions on a cover (1) and on a radius of thermometric package of fuel elements of the reactor in conditions inter-assembly leakage, equal 4% (2); ○, ◑ - experimental data [23, 24] and [33] accordingly; - - - - - - calculation disregarding of interchannel heat transfer.

a level of temperature of fuel assembly with small number of elements. The experimental data obtained on a thermometric bundle of fuel elements of the BOR-60 reactor [33] (fig. 18) and on the assembly of 19 simulators of fuel elements, simulating the fuel assembly of the BN-350 reactor (fig. 19) [29] testify to it.

5.3. Researches in Japan

The Japanese experts [5, 34 - 36] are engaged in problem of decay heat removal from cores of fast reactors through inter-assembly space of fuel assemblies in conditions of reactors cooldown per the last years.

5.3.1.Experimental researches

The problem on experimental researches of an indicated problem in Japan leaves for frameworks of the given report. Let's indicate only, that in Japan the extensive program of researches [34] is developed, the unique experimental base (fig. 20, 21) is created, series of experiments are conducted by using sodium as the coolant. For familiarity of the reader with some obtained results it is possible to recommend papers [5, 36].

We shall stay on a computational technique used recently by the Japanese experts for the analysis of experimental data in more detail.

5.3.2. Computational researches [35]

The numerical method of the analysis is based on use of approximation of a porous body [37]. The equations of mass, moment and energy conservation for flow in porous environment are noted in the final differential form (non-stationary case) for internal and external channels of assembly. It is supposed that the full diffusion of momentum

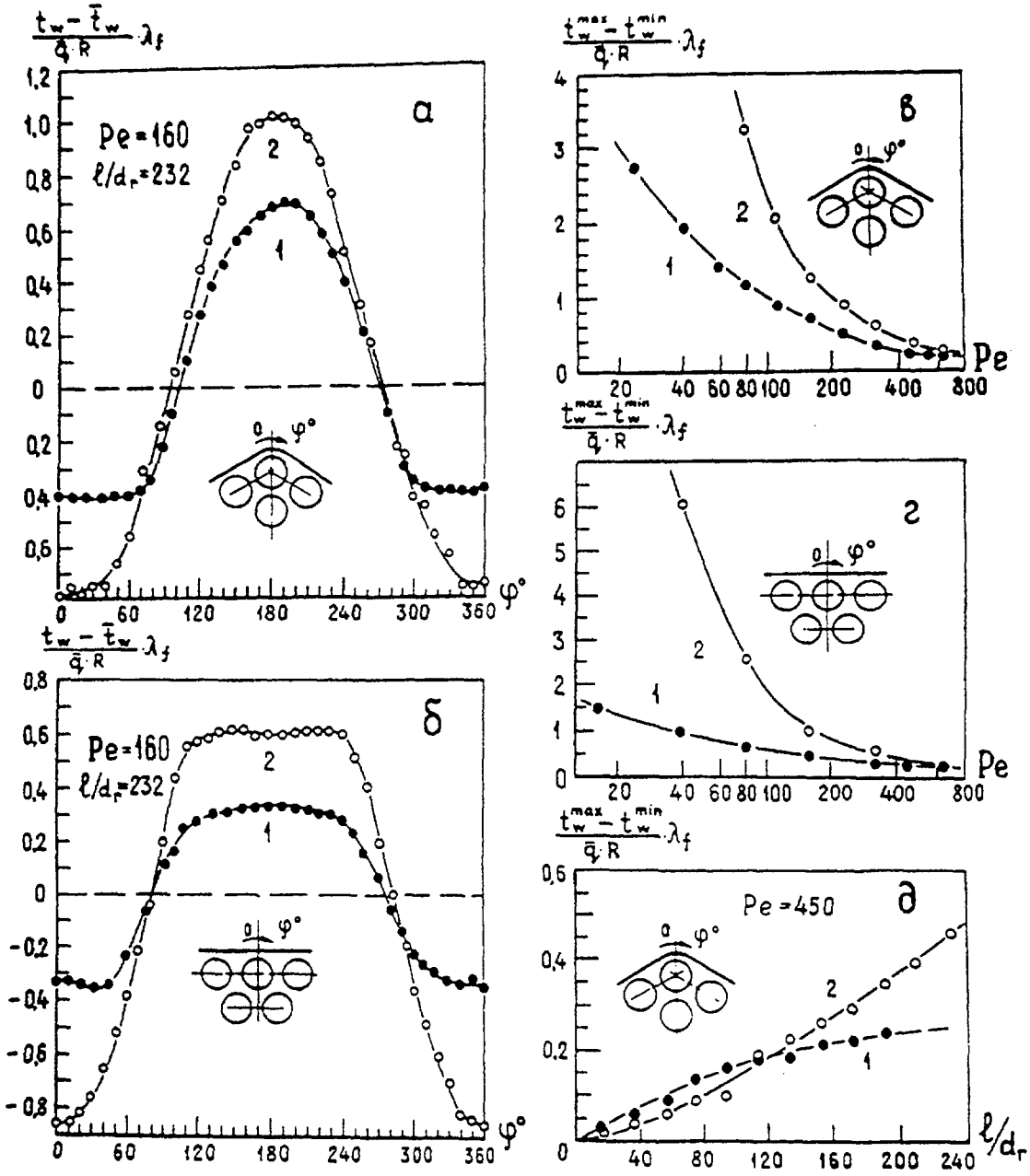


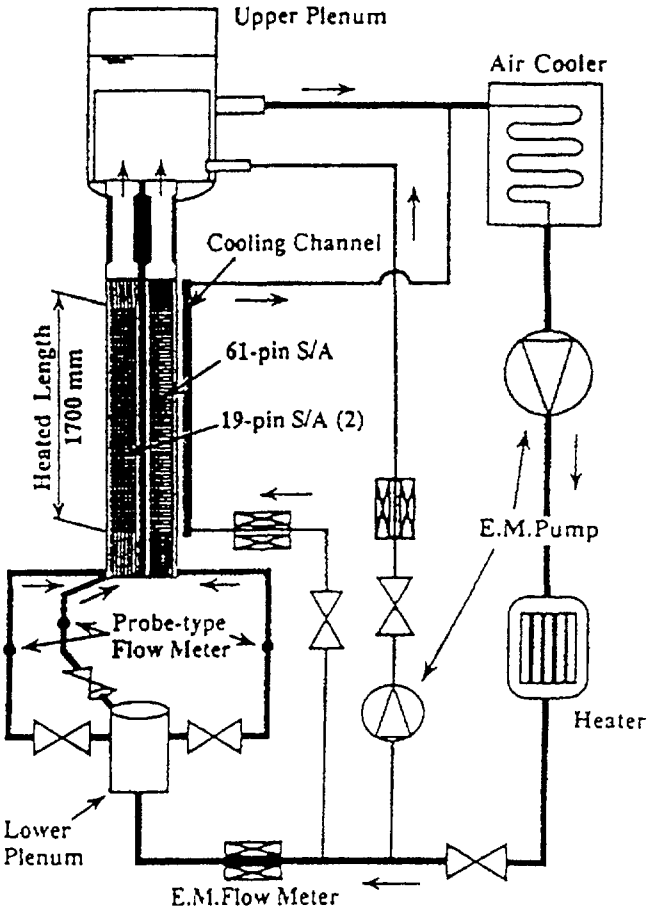
Fig. 19. Temperature fields of angular (а) and lateral (б) simulators of fuel elements of the model fuel assembly of the BN-350 reactor at adiabatic (1), non-adiabatic (2) conditions on a cover of assembly and regularity of change of maximum nonuniformities of temperature from the Peclet number (в, г) and from length of energy release (д).

$$\tilde{A}_m = \varepsilon_e + \varepsilon_M + k\nu \tag{7}$$

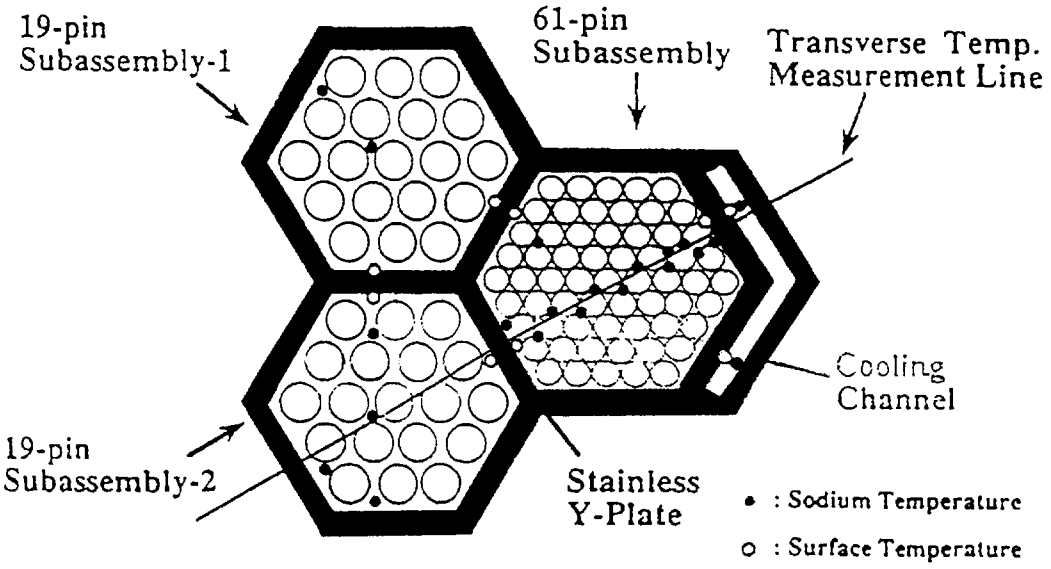
and energy

$$\tilde{A}_e = Pr_t(\varepsilon_e + \varepsilon_M) + k\frac{\lambda}{\rho c_p} \tag{8}$$

consist from heat transfer by means of wire wrapper (ε_e), thermal vortexes (ε_M) and molecular transfer ($\nu, \frac{\lambda}{\rho c_p}$).The coefficients of turbulent intermixing undertake from S.K.Cheng and

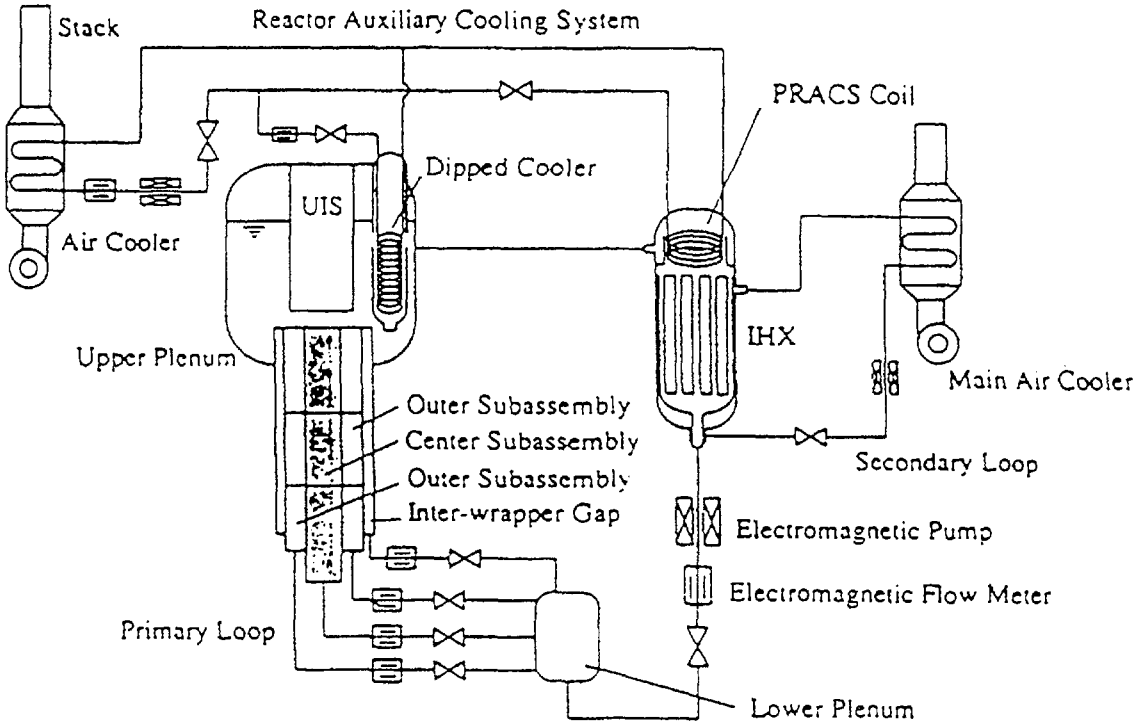


Flow Diagram of the CFR Test Section installed in the CCTL

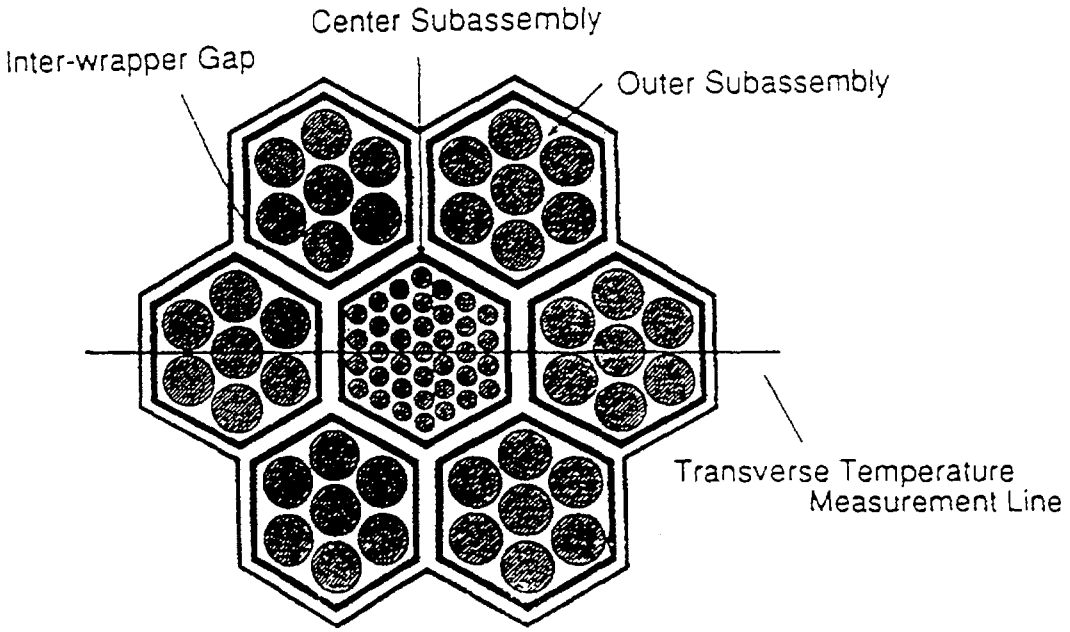


Cross Section View of CCTL-CFR Subassemblies

Fig. 20. CCTL-CFR experiment.



Flow Diagram of PLANDTL-DHX



Cross Section View of PLANDTL-DHX Subassemblies

Fig. 21. PLANDTL-DHX experiment.

N.E.Todreas [38, 39], and coefficients of exchange by means of wire wrapper - from [40, 41].

The turbulent Prantl number is supposed equal 1. The coefficient k, taking into account geometric factor in molecular exchange, is calculated from Cheng and Todreas [39]:

$$k = 0.66\left(\frac{P}{D}\right)\left(\frac{C}{D}\right)^{-0.3} \tag{9}$$

Here P - distance center to center of rods, D - diameter of rods, C - gap between adjacent rods. It is supposed, that the relation (9) is fair and for wire-wrapped rods.

It is marked, that the role of a thermal conduction is insignificant on a comparison with intermixing through thermal curls in conditions of convective mixing, by means of wire wrappers and redistribution of flow in the field of low values of Reynolds numbers.

The inter-assembly heat transfer is taken into account. It is marked, that if it is great, it can be the reason of large temperature gradients across a bundle and even of an inverse flow.

The Grashof number according to Engel etc. is introduced [42].

$$Gr_{\Delta T} = \frac{g\beta \dfrac{\Delta T_b\big|_0^L}{L} D_{eb}^4}{\nu^2}, \tag{10}$$

where D_{eb} - hydraulic diameter of a bundle of rods, L - length of energy release, $\Delta\dot{O}_b$ - difference of mean-mixing (index "b") temperatures between two levels.

As it is visible from (10) $Gr_{\Delta T}$ is determined from average longitudinal temperature gradient in limits of length of energy release.

As in an ensemble of assemblies with inter-assembly heat transfer the modification of an axial gradient of temperature happens, the modified Grashof criterion is used

$$Gr^*_{\Delta T} = \frac{g\beta \dfrac{\Delta T_b}{\Delta z}\Big|_k D_{eb}^4}{\nu^2}, \tag{11}$$

where k - number of an axial knot of a calculated grid.

Local on an axes $Gr^*_{\Delta T}$ numbers are used in a calculated code AQUA.

The resistance coefficients for longitudinal flow are calculated from relation of Cheng and Todreas [38]. The resistance in transversal direction is supposed much smaller, than in longitudinal.

The threshold function eliminating effect of a diffusion through thermal curls at the Richardson number $Ri \le 0.1$ is introduced into reviewing, about what the data of [39, 43] testify.

This function varies from 1 up to 0 and looks as follows (fig. 22):

$$fx_{pl} = \max\left[0.1 - 10^{\{-6(Ri-0.1)\}}\right] \tag{12}$$

$$Ri = \frac{Gr}{Re_i^2} \tag{13}$$

$$Gr = \frac{g\beta\Delta T_b\big|_0^L D_{eb}^3}{\nu^2} = Gr_{\Delta T}\frac{L}{D_{eb}} \tag{14}$$

It is supposed, that the threshold on Richardson (Ri) is applicable as for local and for average conditions, i.e.

$$\Delta T_b\big|_0^L = \frac{\Delta T_b}{\Delta z}\Big|_k \cdot L \tag{15}$$

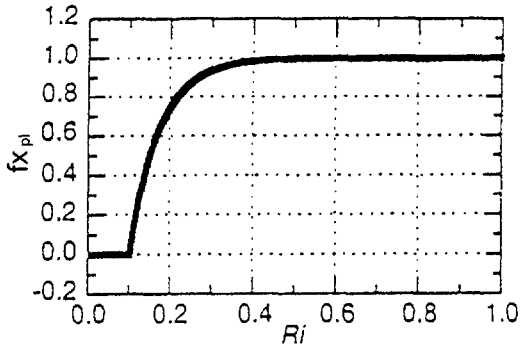


Fig. 22. A curve of a threshold function for diffusion at mixing of thermal vortexes.

$$Gr = Gr^*_{\Delta T} \frac{L}{D_{eb}} \tag{16}$$

The relations for coefficients of interchanging are obtained for the following intervals of a modification of defining parameters: $170 \le Gr_{\Delta T} \le 650$; $520 \le Re_b \le 4400$; $0.6 \le Gr_{\Delta T} / Re_b \le 0.73$; $1.08 \le P / D \le 1.25$.

Here $Re_b = \frac{v_b D_{eb}}{\nu}$ - average on a bundle the Reynolds number.

The calculated grid is selected according to Ro and Todreas [37]. On a vertical the assembly is devided into 29 knots at overall length of 2825 mm. The interval between knots on length of energy release makes 82.5 mm or 87.5 mm. The non-stationary calculations are fulfilled at use of a more small-sized grid with a half distance between knots on a vertical, i.e. 41.25 mm or 43.75 mm.

The coefficients of heat transfer inside a cover of assembly, and also on an interior surface of a cover are calculated on a relation Subbotin V.I.

$$Nu = 5.0 + 0.025\ Pe^{0.8} \tag{17}$$

The sodium in an inter-assembly gap is supposed fixed (the first stage of calculations); the one-dimensional heat conduction equation in perpendicular direction to a surface of a cover is decided.

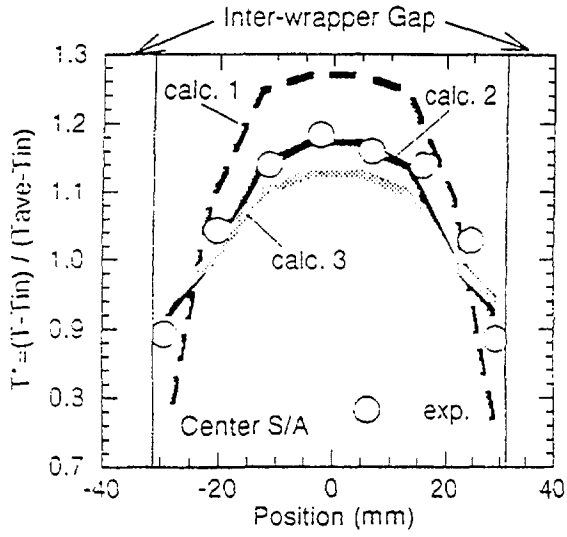
The numerical method is used according to [44]. The characteristic results of calculation of relative heating of the coolant on a cut of model assembly are represented in the fig. 23.

The rather in-depth account of methodical side of research [35] in the report is connected to importance of an analyzable problem and with necessity of comparison of results of this work with earlier obtained data in IPPE on a considered problem [26, 29, 31, 32].

6. THERMOHYDRAULICS IN THE UPPER PLENUM OF THE FAST REACTOR IN COOLDOWN CONDITIONS

6.1 Processes defining thermohydraulics at natural convection in the upper plenum of the reactor

Creation of thermohydraulic codes of various assigning, various volume and content, passes already more than 20 years. The calculated codes, their architecture and content are determined mainly by population of problems, which are before the scientists and designers investigating nominal and abnormal processes of existing nuclear reactors. Per last years it is problems connected to safety of nuclear reactors.



Calculation	The factor of intermixing	Threshold function
1	is not used	is not used
2	is used	is used
3	is used	is not used

$Re_B = 6912$, $Gr_{\Delta T} = 913$, $Gr = 2{,}27 \cdot 10^6$, $Ri = 0{,}005$

Fig. 23. A normalized temperature profiles on the upper of a heated zone at initial Condition-1.

The basic place among thermohydraulic processes takes heat transfer in bundles of rods of a core and in the upper plenum of the reactor. In a core it is possible to mark researches of temperature fields in a condition with blockage of a part of cross section of fuel assembly, role of stringers and other researches partially considered above.

One from major problems of thermohydraulics of the upper plenum of the fast reactor, especially in the cooldown process, is the definition of velocity and temperature fields for various variants of removal of thermal power. These temperature and high-speed fields depend on phenomenon of temperature striping which have a place in liquid metal reactors. Let's mark some from them: thermal stratification, thermal striping, core - plenum interaction, free surface sloshing.

6.1.1. Thermal stratification

Thermal stratification phenomenon appears in large volume of upper plenum after scram due to relatively cold temperature fluid being continuously supplied with low velocity into the upper plenum at decay heat power level. The interface between hot (upper) and cold (lower) fluid regions in the upper plenum will oscillate slowly and largely under certain conditions. Such a large oscillatory motion of the stratification interface, which has a large temperature difference, causes severe thermal stress to the reactor vessel and reactor components.

6.1.2. Thermal striping

Thermal striping phenomena characterized by random temperature fluctuations inevitably occur in the mixing regions of hot and cold coolant, e.g., the area between the core outlet and the upper core structure. In the region around the core outlet, the interaction of cold sodium flowing out of a control rods assembly and hot sodium flowing out of adjacent

fuel assemblies causes the thermal striping and might give high-cycle fatigue to the in-vessel components located in this region. Therefore it is necessary to evaluate influenced area and characteristics of the random thermal process and to protect the corresponding components from high-cycle fatigue.

6.1.3. Free surface sloshing

In the heat transport system of liquid-metal reactors, free surfaces exist for absorbing the coolant thermal expansion when the temperature rises. As it gives large temperature difference to attached structures, the free surface must be kept stable not to cause sloshing and rapid shrinkage during various operating conditions.

6.1.4. Core - plenum interaction

From the viewpoint of passive safety, it is desirable that the decay heat could be removed only by natural circulation when the pony motor could not be operated. During the operation of cooling system, relatively cold temperature fluid from the heat exchangers directly immersed in a hot plenum flows down to the core outlet region and may repeat to penetrate into some subassemblies under a certain natural circulation condition. This phenomenon, which is called the core-plenum interaction, is probable to influence not only structure material but also the natural circulation head which determines core flow rate and therefore affects the core coolability.

6.2 Codes used for natural convection research and some results of researches

Let's mark some system codes used for natural convection research and partially mentioned in introduction. It is codes AQUA, TRIO-VF, FLUTAN, ASTEC, SSC-L, DINUS-3, NECTAR. The SSC-L code deals with various accidents and transients such as natural circulation. But in the given code for the upper plenum the two-dimensional model is used. It is not enough of it, since the natural convection has a three-dimensional character. With the help of the ASTEC code thermohydraulics of installation RAMONA was simulated. The comparison of computed and experimental results shows that following the pump rundown transient the core inlet and outlet temperatures, and hence the core flow rate, recover faster than the experimental data show. The similar code on the given problem is the German FLUTAN code, which also allows to analyze thermohydraulic phenomena including a system of passive cooling of the reactor. The French TRIO-VF code are used to compute the characteristics of temperature fluctuations occurring in a tee mixing hot and cold sodium pipe flows. In this case in the given code the model of a turbulence LES (modeling of large curls) is applied.

But the most characteristic and universal code for natural convection research, under the judgement of the authors, is the Japanese code AQUA.

A general purpose multi-dimentional thermohydraulic analysis code AQUA was developed at PNC (Japan) and was improved in order to evaluate thermal stratification phenomena including large oscillatory motions of the stratification interface. AQUA is based on the finite difference method and is characterized by the incorporation of advanced turbulence models, higher-order discretization schemes for the convection terms, treatment of time-dependent terms with second-order accuracy.

AQUA is applied to simulate both the water and sodium experiments. To accurately model the buoyancy effects, two kinds of turbulence models were implemented to AQUA: the $k - \varepsilon$ model with Pr_t being dependent on the local Richardson number and the algebraic stress model (ASM). Model ASM gives good results for sodium experiments.

For the evaluation of the temperature fluctuation intensity, $k - \varepsilon - \theta$ model and the Reynolds stress model (RSM) as well as ASM were incorporated into AQUA. ASM is the

most suitable turbulence model because it shows the better prediction accuracy and uses much less computational time.

Similar AQUA code developed in Japan is the DINUS-3 code. The direct simulation code DINUS-3 is the code without any turbulence models. The basic purpose of the code is to investigate temperature fluctuations in the core outlet region and in the upper plenum of fast reactors.

As it was marked above, in IPPE the two-dimensional NECTAR code is developed. Let's present as an example the results calculations of process of decay heat removal by natural convection which are carried out on this code. The scheme of installation is shown in the fig. 24. The standard set of Navier-Stokes and energy equations was decided by a finite difference method of Patankar.

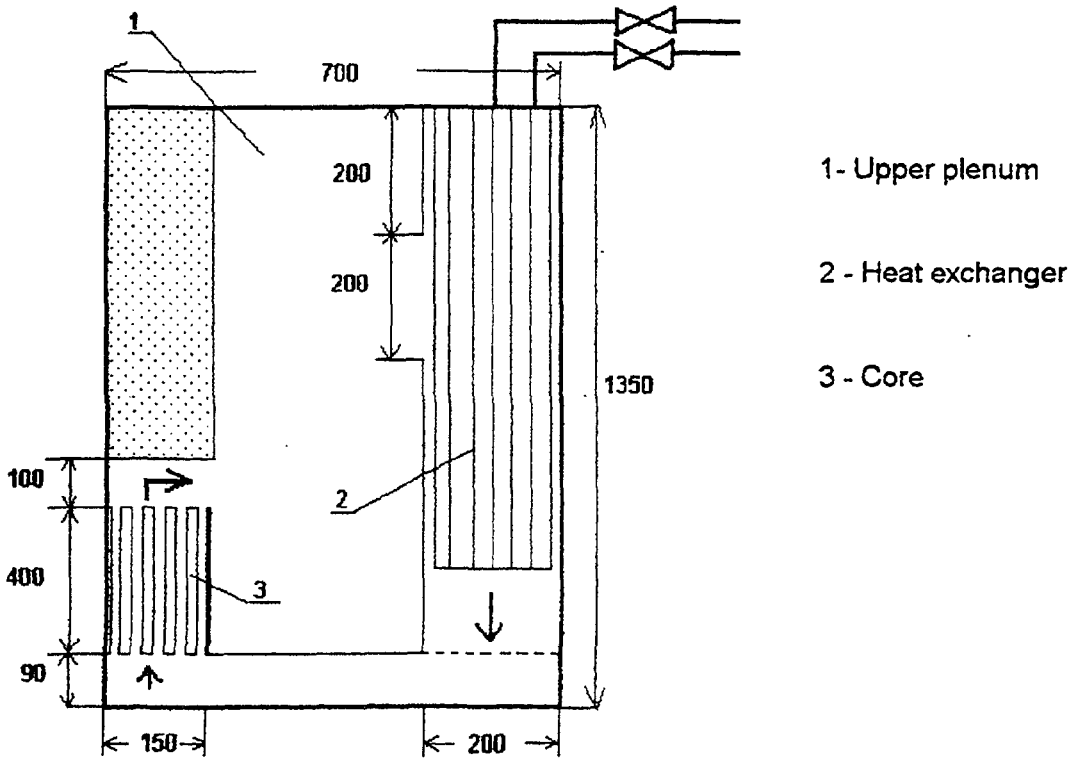


Fig. 24. The scheme of the installation for calculations by the NECTAR code.

In the fig. 25 the modification of temperature of the coolant (sodium) at core outlet in time is shown. The thermal power, Q, assigned as a parameter, varied from 0.1 kW up to 10 kW. For all values Q there is a growth of temperature for all considered slice of time. These results have a preliminary character; for reliable simulation of system of emergency cooldown the in-depth study of influence of a thermal power Q, geometry of a model, performances of the heat exchanger is necessary.

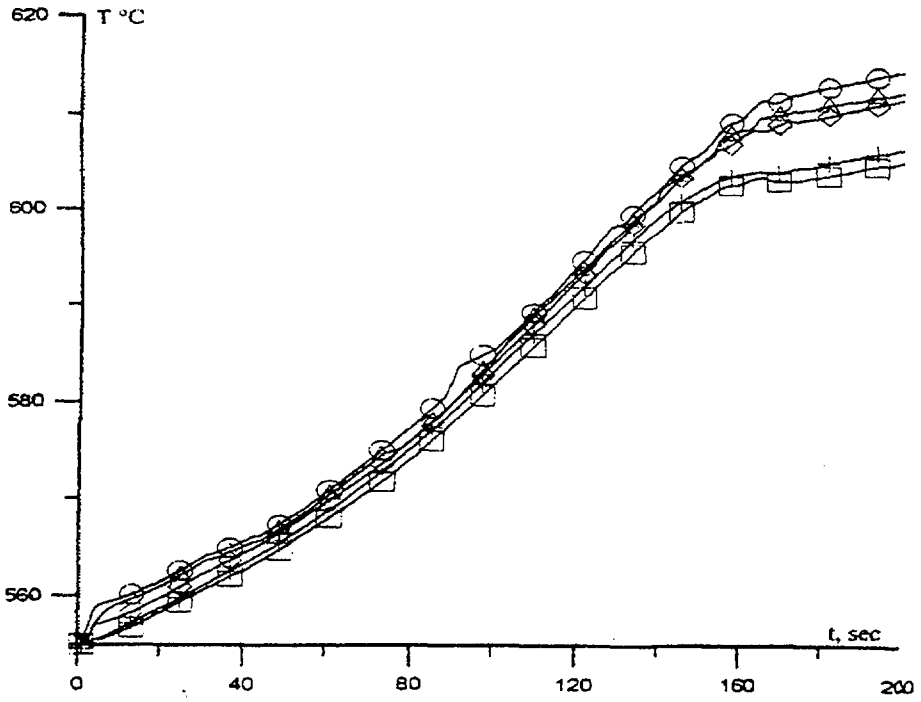


Fig. 25. Change of the coolant temperature (sodium) at outlet core in time at various powers: □ - 0.1 kW, + - 1 kW, ◊ - 2 kW, Δ - 4 kW , ○ - 10 kW.

REFERENCES

[1] IAEA / IWGFR / 88, Specialists Meeting on "Evaluation of Decay Heat Removal by Natural Convection", 22-23 February 1993 // Oarai Engineering Center, PNC, Japan.

[2] ZHUKOV A.V., SOROKIN A.P., KUZINA J.A Removal of Residual Heat Natural Convection at Emergency Cooldown Fast Reactors: the Concepts and Results of Researches The report on V international conferences "Safety NPS and training of personnel", OINPE, Obninsk, 2-5 February 1998. The theses of the reports // OINPE, Obninsk, 1998.

[3] ZHUKOV A.V., SOROKIN A.P., KUZINA J.A. Results of Decay Heat Removal Researches for Fast Reactors on Problems of Emergency Reactor Cooldown (Review of Studies) // Meeting 4-th working groups on fast reactors technology, IPPE, June 15-20, 1998, Obninsk, Russia.

[4] THE AGREEMENT ON FAST REACTOR BETWEEN COMMISSARIAT ON AN ATOMIC ENERGY ON FRANCE AND MINATOM OF RUSSIA ON 1994-1995.

[5] SOBOLEV V.A., VORONOV V.N., KUZAVKOV N.G. Analysis of Decay Heat Removal from the Core Using Passive Cooling System // Sodium Cooled Fast Reactor Safety, International Topical Meeting, Obninsk, Russia, October 3 - 7, 1994, Proceedings, v. 4, p.p. 6-12 - 6-23.

[6] KAMIDE H., IEDA Y., TODA S., ISOZAKI T., SUGAWARA S. Multi-bundle Sodium Experiments for Thermohydraulics in Core Subassemblies during Natural Circulation Decay Heat Removal Operation // IAEA / IWGFR / 88, Specialists Meeting on "Evaluation of Decay Heat Removal by Natural Convection", 22-23 February 1993, Oarai Engineering Center, PNC, Japan.

[7] ERTEL V., REINDERS R. Development and Application of Thermohydraulic System Code DYANA/ATTICA // Proc. Intern. Conf. On Fast Reactor Safety, Juersy, GB, 1986.

[8] GRAND D. et al. The Dimensional Computations of Thermalhydraulic Phenomena in Reactor Vessel // Advanced in Mathematics, Computations and Reactor Physics, Int. Meeting, Pittsburg, USA, April 28 - May 2, 1991.

[9] GEORGEOURA S.E., KEETON J.A. A three Dimentional Simulation of a Pool Type Fast Reactor Model using the ASTEC Code // NURETH-5, Sept. 21 - 24, 1992, Salt Lake City, USA.

[10] BORGWALDT H., BAUMANN W., WILLERDING G. FLUTAN, Input Specification (related to family of COMMIX codes) // KfK 5010, Kernforschungszentrum Karlsruhe, Germany, May 1992.

[11] MAEKAVA I., Numerical diffusion in single-phase multidimensional thermal-hydraulic analysis // Nucl. Engrg. Des. 120 (1990), 323.

[12] KUZINA J.A., SOROKIN A.P., UKHOV V.A. The Status of Thermohydraulic Codes for Reactors on Fast Neutrons / Meeting 4-th working groups on fast reactors technology, IPPE, June 15-20, 1998, Obninsk, Russia.

[13] THE AGREEMENT ON FAST REACTOR BETWEEN COMMISSARIAT ON AN ATOMIC ENERGY ON FRANCE AND MINATOM OF RUSSIA ON 1996-1997.

[14] MITENKOV F. M., BAGDASAROV YU.YE., BUKSHA YU.Ê. et al. Engineering methods of the analysis of modes with natural circulation in installations of a type BN // Atomic Energy, 1987, v. 62, № 3, p. 147 - 152.

[15] YAMAGUCHI A., et al. Plant-wide Thermal Hydraulic Analysis of Natural Circulation Test at JOYO with MK-Ï Irradiation Core // NURETH-4, v.1, p. 398 - 404.

[16] SATON K., et al. Study of Decay Heat Removal by Natural Circulation // NURETH-4, v. 1, p. 378 - 383.

[17] AUBRY S., LAMBERT M. Thermal-Hydraulic Study of LMFBR Fuel Assemblies under Accident Cooling Conditions during Internal Storage or Handling // Science and Technology of Fast Reactor Safety, BNES, London, 1986, p. 219, 223.

[18] FORT J.A., et al. Investigation of Flow Recirculation in Rod Bundle during Natural Convection Flow and Power Transient // Second International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Santa Barbara, California, USA, Jan. 11 - 14, 1983.

[19] WYDLER P., SIGG B., DUTTON P. Studies of Decay Heat Removal by Natural Convection using the SONACO Sodium-Cooled 37-Pin Bundle // Science and Technology of Fast Reactor Safety, BNES, London, 1986, p. 185 - 190.

[20] KOLYASKIN V.I., KUDRYAVTSEV L.K., USHAKOV P.A. Heat Transfer to an Alloy Sodium -êàëèé in a Cell of Dense Packing of Cores at Mixed Convection ÒÂÒ, ò.11, ¹ 4, 1973, p. 781 - 787.

[21] DATTON, UELTY. Experimental Research Natural Convection about a Lattice of the in Regular Intervals Heated up Vertical Cores at Small Prandtl Numbers // Heat Transfer, ¹ 3, 1975, p. 57 - 63.

[22] BOGATIRJOV I.L. et al. System of Constants for Subchanals Thermogidraulics of Account of Modes the Operations Subassemblies Reactors with Natural and Mixed Convection // Preprint IPPE-2238, Obninsk, 1992.

[23] THE CATALOGUE OF STANDS, REACTORS OF ZERO CAPACITY AND OTHER EXPERIMENTAL INSTALLATIONS // Published by CEC, CSTC 1-2, 1978, p. 20 - 22.

[24] ZHUKOV A.V., SOROKIN A.P., SVIRIDENKO YE.YA., HUDASKO V.V. Experimental and Settlement Modeling Thermogidraulic Heat Transfer of Devices NPP. Model Assembly, Gauges, Technique. Liquid Metal the Stand. The Manual for the Students of Courses of Improvement of Qualification and Special Faculty // ÎINPE, Obninsk, 1992 ã., p. 59 - 76.

[25] SUBBOTIN V.I., ZHUKOV A.V., PASHEK M. et al. Experimental Study on Model of the Operating Temperature Regimes of the Fuel Elements of a Bor-60 Reactor // Preprint IPPE - 137, Obninsk, 1969.

[26] SUBBOTIN V.I., ZHUKOV A.V., PASHEK M. et al. Experimental Study on Model of the Operating Temperature Regimes of the Fuel Elements of a Bor-60 Reactor // Heat Transfer, Soviet Research, 1971, v.3, ¹ 5, p. 15 - 26.

[27] ZHUKOV A.V., KUDRYAVTSEVA L.K., SVIRIDENKO YE.YA. et al. Experimental Research on Models of Temperature Heat Removel Elements // Liquid Metals, Ì., Atomizdat, 1967, p. 170 - 194.

[28] ZHUKOV A.V., SOROKIN A.P., KIRILLOV P.L. et al. The Methodical Instructions and Recommendations on Thermogidraulic to Account of Fast Reactors Core // RTM–1604.008-88, Obninsk, Published by DSTI, IPPE, 1989.

[29] SUBBOTIN V.I., USHAKOV P.A., ZHUKOV A.V. et al Temperature Fields of Fuel Elements in Fast Reactors Core with Liquid Metal Cooling // Atomic Energy, 1967, v.22, № 5, p. 372–378.

[30] KUTATELADZE S.S. Bases of the Theory of Heat Exchange // "Nauka", Novosibirsk, 1970.

[31] ZHUKOV A.V., KIRILLOV P.L., MATJUKIN N.M., SOROKIN A.P., et al Thermogidraulic Account in Fuel Subassemblies Fast Reactors with Liquid Metal Cooling // Ì: Energoatomizdat, 1985.

[32] ZHUKOV A.V., SOROKIN A.P., USHAKOV P.A. et al Thermogidraulic Account in Fuel Subassemblies Fast Reactors with Liquid Metal Cooling // Atomic Energy, 1981, v.51, p. 307 - 311.

[33] ZHUKOV A.V., SOROKIN A.P., MATJUKIN N.M. Interchannel Exchange in Fuel Subassemblies Fast Reactors; Theoretical Bases and Physics of Process // Ì: Energoatomizdat, 1989

[34] ZHUKOV A.V., SOROKIN A.P., MATJUKIN N.M. Interchannel Exchange in Fuel Subassemblies Fast Reactors: the Settlement Programs and Practical // Ì: Energoatomizdat, 1991.

[35] GRYAZEV V.M., ASEEV N.A., MARKIN S.A et al. Calculated-experimental researches Heat of physics and hydrodynamics of packages of an reactor core Bor-60 // In Heat physics and hydrodynamics of an core and generators pairfor of fast reactors. Prag, DSTI, Atomic Energy Commette, 1978, v.1, p. 182 - 209.

[36] IEDA Y., KAMIDE H., OHSHIMA H., SUGAWARA S., NINOKATA H. Strategy of Experimental Studies in PNC on Natural Convection Decay Heat Removal // IAEA / IWGFR / 88, Specialists Meeting on "Evaluation of Decay Heat Removal by Natural Convection", 22-23 February 1993, Oarai Engineering Center, PNC, Japan.

[37] NISHIMURA M., KAMIDE H., HAYASHI K., MOMOI K.: Inter-Subassembly Heat Transfer During Natural Circulation Decay Heat Removal - Experimental Transient Behavior from Forced to Natural Circulation and its Multi-dimensional Analysis with Mixing Model // Eighth International Topical Meeting on Nuclear Reactor Thermal-Hydraulic, Kioto, Japan, September 30 - October 4, 1997., Proceeding, vol.2, p. 903 - 913.

[38] KAMIDE H., HAYASHI K., MOMOI K. Experimental Study of Core Thermogydraulics in Fast Reactors During Transition from Forced to Natural Circulations - Influence of Inter-Wrapped Flow // NURETH-8, New Horizonns in Nuclear Reactor Thermal-Hydraulics, Eighth International Topical Meeting on Nuclear Reactor Thermal-Hydraulic, Kyoto, Japan, September 30 - October 4, 1997., Proceeding, vol.2, p. 922 - 931.

[39] RO T.S., TODREAS N.E. Porous Body Analysis of Vertical Rod Bundles under Mixed Convection Conditions // Ph.D.Thesis, Department of Nuclear Engineering, M.I.T, (1983).

[40] CHENG S.K., TODREAS N.E. Hydrodynamic Models and Correlation for Bare and Ware-Wrapped Hexagonal Rod Bundles // Subchannel Friction Factor and Mixing Parameters", Nucl. Engrg. And Des., 92, 227 - 251, 1986.

[41] CHENG S.K., TODREAS N.E. Energy Transfer Mechanism under Mixed Convection Conditions in LMFBR Wire-Wrapped Bundles // Proc. Of the 3-rd Int. Topical Mtg. On Reactor Thermal Hydraulics, Newport, USA, p 16 - E1, Oct.1985.

[42] NINOKATA H., EFTHIMIADIS A., TODREAS N.E. Distributed Resistance Modeling of Wire-Wrapped Rod Bundles // Nuclear Engireeng and Design, 1987, v. 104, p. 93 - 102.

[43] GEORGE T.L., BASEHORE K.L., et al., COBRA-WC: a Version of COBRA for Single-Phase Multiassembly Thermal Hydraulic Transient Analysis // PNL-3259, Battelle Pacific North West Laboratory, 1980.

[44] ENGEL F.S., MARKLEY R.A., BISHOP A.A. The Effect of Radial Heat Flux Gradients and Flow Regimes on the Peak Sodium Temperature Rise in Wire Wrapped Bundles // Proc. Of NUREG / CP-0034, vol. 2, 566 - 582, New York, Sept, 1982.

[45] BATES J.M., KHAN E.U. Investigation of Combined Free and Forced Convection in a 2 x 6 Rod Bundle During Controlled Flow Transients // AIChE Symposium Series, Heat Transfer, Orlando, 1980.

[46] DOMANUS H.M. A New Implicit Numerical Solution Scheme in the COMMIX-1A Computer Program // ANL 83-64, NUREG/CR-3435, 1983.

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