3-D THERMAL HYDRAULIC ANALYSIS OF TRANSIENT HEAT REMOVAL FROM FAST REACTOR CORE USING IMMERSION COOLERS

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Abstract

For advanced fast reactors (EFR, BN-600M, BN-1600, CEFR) the special complementary loop is envisaged in order to ensure the decay heat removal from the core in the case of LOF accidents. This complementary loop includes immersion coolers that are located in the hot reactor plenum. To analyze the transient process in the reactor when immersion coolers come into operation one needs to involve 3-D thermal hydraulics code. Furthermore sometimes the problem becomes more complicated due to necessity of simulation of the thermal hydraulics processes into the core inter-wrapper space. For example on BN-600M and CEFR reactors it is supposed to ensure the effective removal of decay heat from core subassemblies by specially arranged internal circulation circuit: "inter-wrapper space".

For thermal hydraulics analysis of the transients in the core and in the whole reactor including hot plenum with immersion coolers and considering heat and mass exchange between the main sodium flow and sodium that moves in the inter-wrapper space the code GRIFIC (the version of GRIF code family) was developed in IPPE.

GRIFIC code was tested on experimental data obtained on RAMONA rig under conditions simulating decay heat removal of a reactor with the use of immersion coolers. Comparison has been made of calculated and experimental result, such as integral characteristics (flow rate through the core and water temperature at the core inlet and outlet) and the local temperatures (at thermocouple location) as well.

In order to show the capabilities of the code some results of the transient analysis of heat removal from the core of BN-600M - type reactor under loss-of-flow accident are presented.

1. INTRODUTION.

It is proposed that additionally to the normal heat removal route via the steam plant, the advanced fast reactors will be equipped with the Direct Heat Removal System (DRS). This system is intended for removal of decay heat from the core in the case of loss of station service power. Each DRS circuit includes an immersion cooler (IC)), installed in the reactor hot plenum and an air cooler. These two coolers are connected via intermediate sodium loop. The transportation of the heat from the core to immersion cooler occurs both by free convection of the sodium in the hot plenum of reactor and by circulation via primary loop. The energy that can be transported this way per time unit depends on reactor design and sometimes can be essentially limited due to stratification of the sodium in the hot plenum. In order to intensify the heat removal from the core subassemblies for advanced Russian fast reactor BN-600M and Chinese experimental fast reactor CEFR the innovative core design was proposed. In these projects the "cold" sodium from IC outlet is partly directed into subassembly inter-space via specially arranged gaps and holes in the reactor internal structures forming additional heat removal circuit.

For thermal hydraulic analysis of the transients in the core and in the whole reactor including immersion coolers and considering heat and mass exchange between the main sodium flow and sodium flowing in the inter-wrapper space GRIFIC code was developed.

2. GRIFIC CODE DISCRIPTION

The code contains the following modules:

- 3D thermal hydraulic model for calculation of sodium velocity, pressure and temperature in the primary circuit;
- 3D model for simulation of inter-wrapper sodium thermal hydraulics;
- set of 1D, 2D and 3D models for calculations of temperature distributions in "impermable" elements (pins, wrappers, etc.);
- 1D thermal hydraulic model of intermediate heat exchanger and immersion cooler; primary pump model.

For simulation of the main sodium flow in the reactor the GRIFIC code is solving a system of heat-and-mass exchange equations in an approximation of a model of viscous incompressible flow in a porous body, taking into account the buoyancy effect in Boussinesq approximation in the cylindrical coordinate system. This system of equations (1-5) in the divergent form is presented bellow:

$$\frac{\partial(\varepsilon U)}{\partial z}+\frac{1}{r}\cdot\frac{\partial(r\varepsilon V)}{\partial r}+\frac{1}{r}\cdot\frac{\partial(\varepsilon W)}{\partial\varphi}=J \tag{1}$$

$$\frac{\partial U}{\partial\tau}+\frac{1}{\varepsilon}\frac{\partial(U\varepsilon U)}{\partial z}+\frac{1}{\varepsilon r}\frac{\partial(rV\varepsilon U)}{\partial r}+\frac{1}{\varepsilon r}\frac{\partial(W\varepsilon U)}{\partial\varphi}= \tag{2}$$

$$=-\frac{1}{\rho}\cdot\frac{\partial P}{\partial z}+\nu\Delta U-\Lambda_z|\bar{\omega}|U-\frac{\rho(T)}{\rho_T}g+\frac{\Delta P_N}{L_N}+J\left\{\frac{U_J}{U}\right\}$$

$$\frac{\partial V}{\partial\tau}+\frac{1}{\varepsilon}\frac{\partial(U\varepsilon V)}{\partial z}+\frac{1}{\varepsilon r}\frac{\partial(rV\varepsilon V)}{\partial r}+\frac{1}{\varepsilon r}\frac{\partial(W\varepsilon V)}{\partial\varphi}-\frac{W^2}{r}= \tag{3}$$

$$=-\frac{1}{\rho}\frac{\partial P}{\partial r}+\nu\Delta V-\Lambda_r|\bar{\omega}|V+J\left\{\frac{V_J}{V}\right\}$$

$$\frac{\partial W}{\partial\tau}+\frac{1}{\varepsilon}\frac{\partial(U\varepsilon W)}{\partial z}+\frac{1}{\varepsilon r}\frac{\partial(rV\varepsilon W)}{\partial r}+\frac{1}{\varepsilon r}\frac{\partial(W\varepsilon W)}{\partial\varphi}+\frac{VW}{r}= \tag{4}$$

$$=-\frac{1}{\rho r}\frac{\partial P}{\partial\varphi}+\nu\Delta W-\Lambda_\varphi|\bar{\omega}|W+J\left\{\frac{W_J}{W}\right\}$$

$$C_p\rho\frac{\partial T}{\partial\tau}+\left(C_p\rho\right)\cdot\left[\frac{\partial(\varepsilon UT)}{\partial z}+\frac{1}{r}\cdot\frac{\partial(r\varepsilon VT)}{\partial r}+\frac{1}{r}\cdot\frac{\partial(\varepsilon WT)}{\partial\varphi}\right]= \tag{5}$$

$$=\frac{\partial}{\partial z}\lambda_z\frac{\partial T}{\partial z}+\frac{1}{r}\frac{\partial}{\partial r}\left[r\lambda_r\frac{\partial T}{\partial r}\right]+\frac{1}{r^2}\frac{\partial}{\partial\varphi}\left[\lambda_\varphi\frac{\partial T}{\partial\varphi}\right]+$$

$$+\sum_n\alpha_n\frac{\Pi_n}{S_n}(\Theta_n+T)_v+J\left(C_p\rho\right)\cdot\left\{\frac{T_J}{T}\right\},$$

where $\quad |\bar{\omega}| = \sqrt{V^2 + U^2 + W^2}$

U, V, W - velocity components in axial (*z*), radial (*r*) and azimuth (φ) directions; *P* - pressure; *T* - temperature of sodium.

Porosity of medium for coolant ε is a function of coordinates and porous medium resistance coefficients Λ_z, Λ_r and Λ_φ may depend on flow parameters. The thermal hydraulic properties - effective kinematic viscosity ν, coolant density ρ and specific heat capacity C_P are the functions of sodium temperature T. Effective conductivity coefficients of porous medium can be different for different directions. The sum $\sum_n \alpha_n \cdot \frac{\Pi_n}{S_n} \cdot (\Theta_n + T)$, takes into account heat exchange with the structures coming in contact with the primary coolant and Θ_n are the surface temperatures of these structures.

The system of governing equations for inter-wrapper sodium is the same as (1)-(5), but it is solved only for subdomain where the inter-wrapper sodium presents. The mass source J is not equal to zero only on the boundaries of this subdomain where mass exchange between the main sodium flow and inter-wrapper space takes place. The source term J is expressed as a function of pressure difference:

$$J = A + B \cdot \left[P_J(r, z, \varphi) - P(r, z, \varphi)\right] \qquad (6)$$

where $P_J(r,z,\varphi)$ and $P(r,z,\varphi)$ - pressure distributions respectively in the inter-wrapper space and in the main sodium flow.

The system of equations (1-5) is solved numerically by the iterative finite-difference method [1].

3. CODE VALIDATION

The capability of the code to simulate correctly natural convection phenomena in the complicated design similar to pool-type reactor with immersion coolers were tested on the RAMONA rig.

RAMONA (reactor model designed for natural convection studies) - is a thermal hydraulic rig geometry similar to that of the pool- type fast reactor, intended for studying transient processes taking place in the reactor on transition phase from nominal steady state operating conditions to decay heat removal conditions with the use of IC [2].

Main characteristics of the RAMONA rig (Fig. 1):

The core is simulated by 8 circular channels formed by 9 heater rings;

IHXs - 8 straight tube counter flow type heat exchangers;

ICs - 4 straight tube counter flow type heat exchanges;

PPs - 4 centrifugal pumps;

ACS - a hollow cylinder with permeable walls;

Coolant - water.

The rig height is 0.58m. The rig radius - 0.5m.

In the region of pump positioning a constant head is assumed ensuring a coolant flow rate through the core of G=0.84kg/s under nominal conditions.

The core power under rated conditions is N=30kW. Heat removal is carried out only through IHXs.

The experiment scenario implies the following sequence of events [3]:

At the initial time (0s) a reduction of heaters capacity to 1 kW takes place. Simultaneously, reduction of coolant flow rate through the core is started. On the 120^{th}s pumps are shutdown and water flow rate through the core is fully determined by natural convection. Secondary coolant flow rate through IHXs is reduced to 0 by 18s. On the 240^{th}s

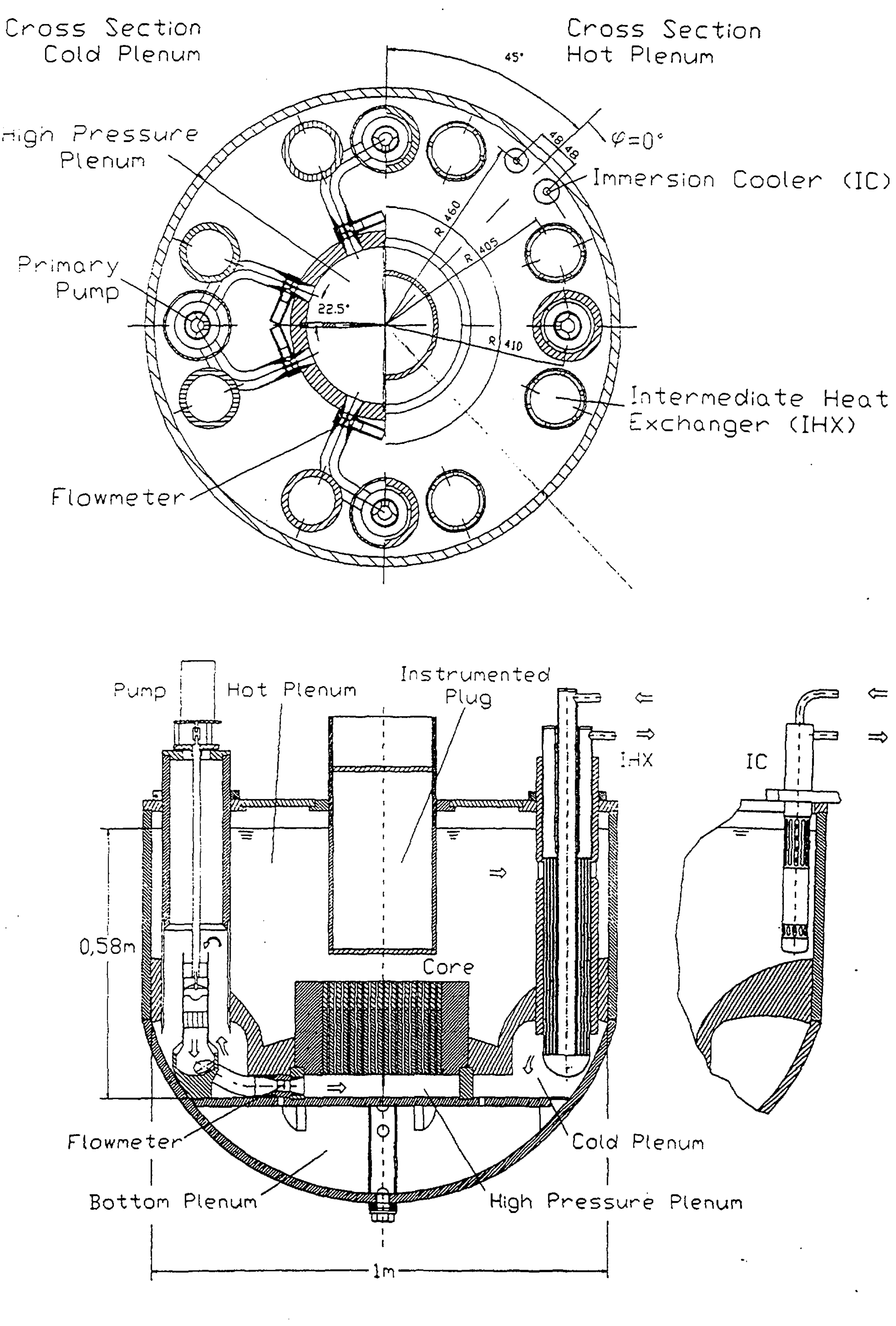


Fig. 1 **RAMONA Transient Model**

ICs are switched on. In so doing changing of water flow rate and inlet temperature through the secondary loop of IC simulates air heat exchanger operation.

The calculation was performed for the 4 initial hours of transient. Three-dimensional calculations model represents a cylindrical sector with an angle of 90° and comprises all the main components of the facility:
one forth of the core and of ACS, two IHXs, one IC and one pump.

Fig.2 illustrates the calculated region. The numerals denote:
3 ⇒ the core;
4, 5 ⇒ above core structure;
2, 6 ⇒ support structure;
7, 8, 9 ⇒ IC;
13 ⇒ pump;
14 ⇒ pump piping;
10 ⇒ dummy region simulating the vessel shape.

Dash-dot lines show thermocouple trees arrangement, a reading of the height of thermocouples positioning (Fig.9-12) is taken from the bottom of the cold plenum.

The calculations of one steady state and one transient (BENCHMARK-CASE 1, [3]) conditions was carried out. Calculated velocity and temperature fields for vertical section passing through the IC are shown in Fig.3,4. In Fig.5-8 the comparison of the obtained data with the experimental result is made with regard to integral characteristics : water flow rate through the core, average temperatures at its inlet and outlet. The calculation represents correctly all peculiarities of flow rate dynamics under these conditions during the whole 4 hour period. The calculated flow rate deviation from the experimental data is mainly within +10÷-20%. An agreement for "average temperatures at the core inlet and outlet" is even better (Figs.7-8) and the difference from the experiment does not exceed 1°C for the most part of the transient process.

In Figs.9-12 the dynamics of local temperature values in different rig components at points where thermocouples are positioned is shown. In Fig.9 temperatures in the hot plenum (HT1.*) are presented. The calculations give an overestimated temperature value - by about 1°C in the upper part and by 2°C in the lower one..

The most significant difference between calculation and experimental result for the hot plenum consists in different intensity of heat exchange rates for the coolant in the annular cavity formed by the support ring "2" and item "6" and the rest of the hot plenum volume located above. As it can be seen from Fig.10, in the experiment coolant contained within the cavity is heated slower resulting in the more abrupt temperature rise on the upper boundary of the cavity. Similar picture is also observed in the cold plenum (Fig.11) where temperature drop along the height is higher than that obtained by calculation. However, dynamics of the average flow temperature on the cold section of the stream is simulated by the calculation to a much better accuracy (Fig.12).

Scheme diffusion may be one of the causes of that more intensive heat exchange is calculated in free volumes.

4. HEAT REMOVAL PROBLEM CALCULATION FOR BN-600M – TYPE REACTOR

The vertical section of *BN-600M – type* reactor passing through IC is schematically represented on Fig. 13. The arrows show the expected pattern of sodium flow in the reactor after primary pump shutdown and immersion coolers coming into operation. Sodium from hot plenum passes through IC tube bundle, discharges through outlet windows and goes downwards. Then it enters rod shielding, steel annular shielding and annular gap forming

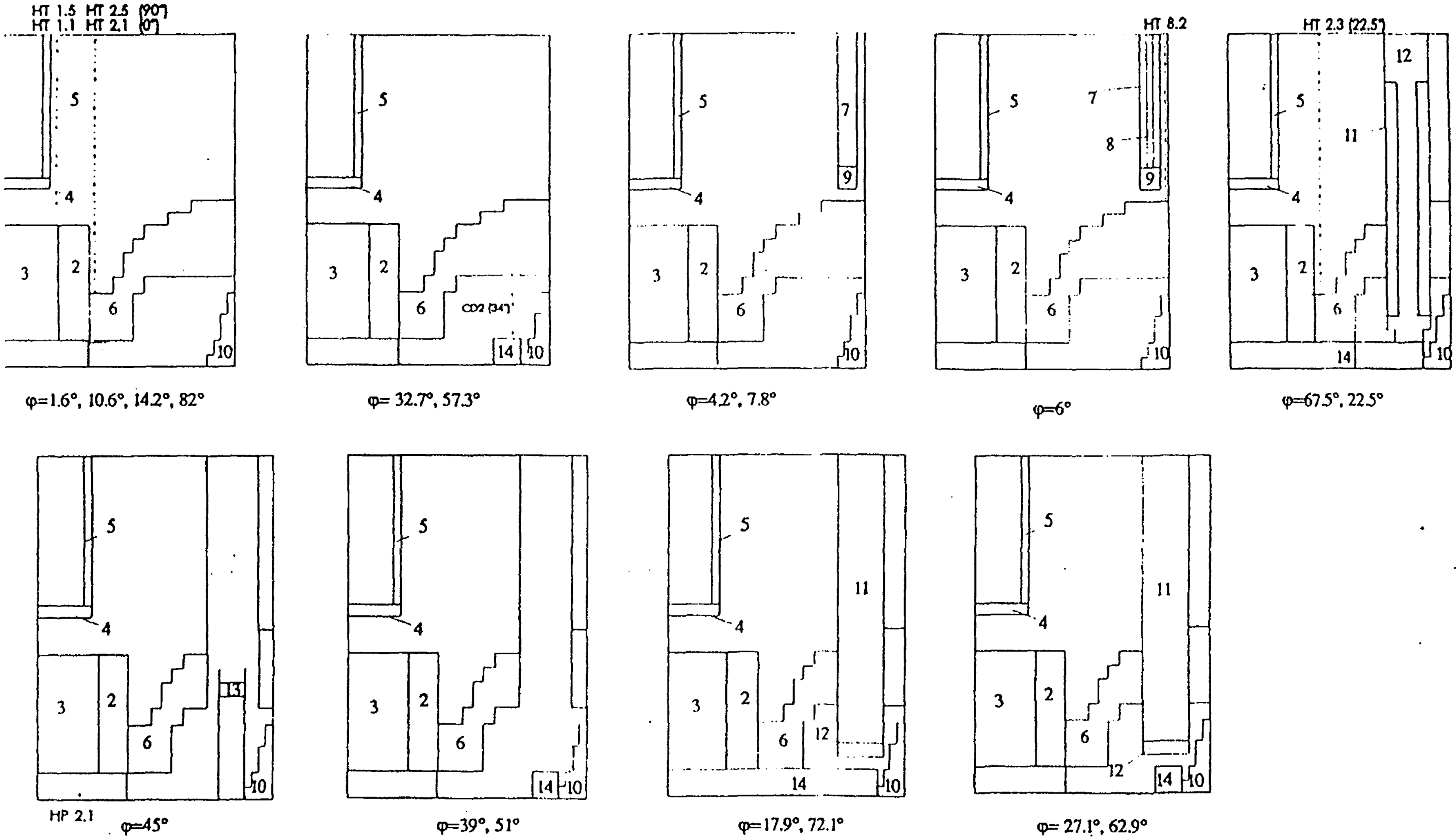
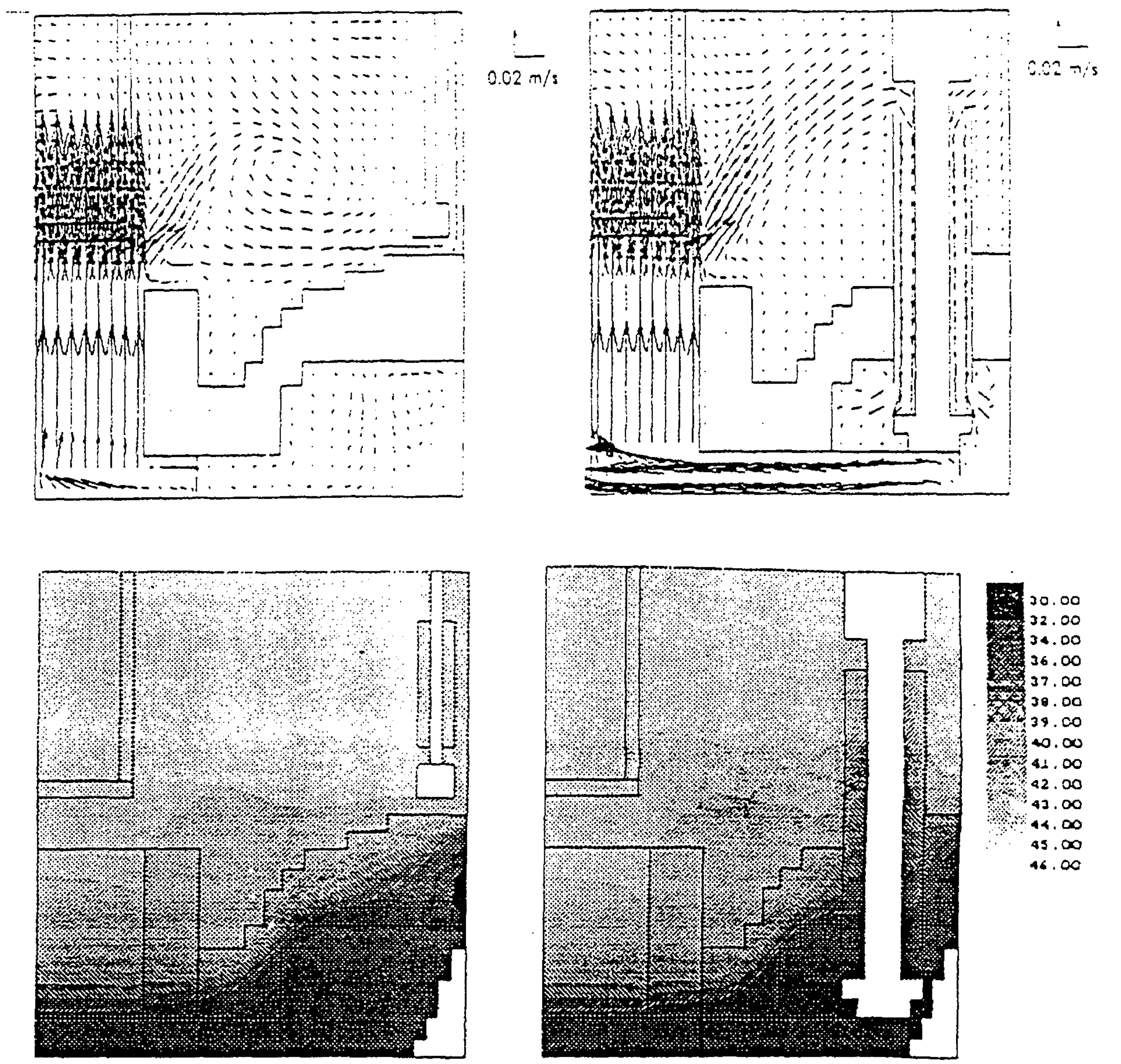


Fig.2 Calculational region.

three parallel regressive flows. And then through the holes in the shielding and structures "cold" sodium is directed into inter-subassembly space where it cools core during its upward movement. The GRIFIC calculations showed that the flow distribution described above is established in the reactor in a few ten seconds after beginning of the accident. Sodium temperature in the "hottest" point reaches its maximum value and then gradually decreases (Fig. 14).

To estimate the efficiency of the core cooling by organised sodium flow in the inter-wrapper space the same accident - (LOF) - was analysed using GRIFIC code, for the case.

Ramona Benchmark (Case 1)
Velocity and Temperature Distributions

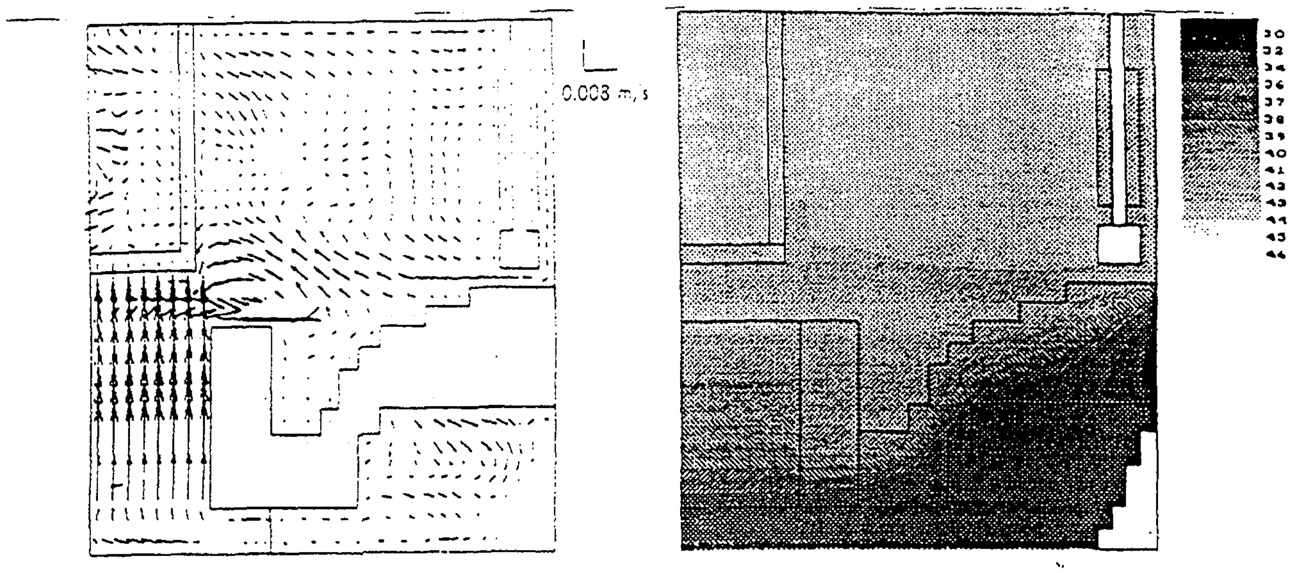


φ =6°   φ =22.5°

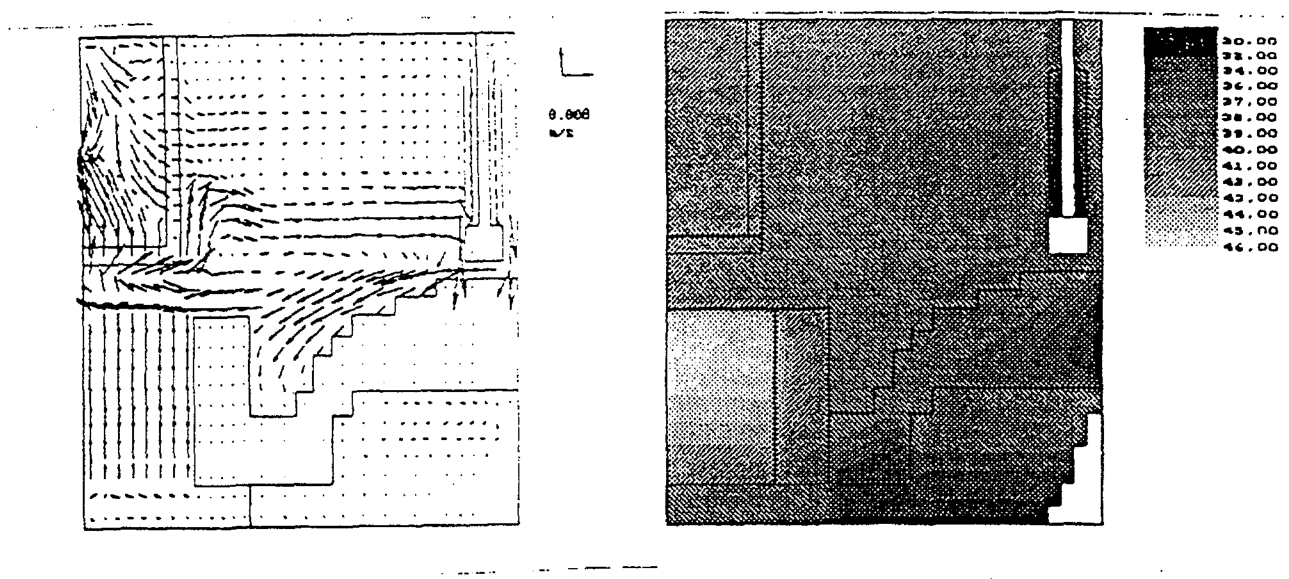
Fig.3

When the sodium velocity in the inter-wrapper space is supposed to be equal zero. The maximum sodium temperature histories for these both cases are presented on (Fig 14). One can conclude from the figure, that sodium circulation in the inter-wrapper space effects essentially heat removal from the core.

Ramona Benchmark (Case 1)
Velocity and Temperature Distributions



t=50s



Time=3500s

Fig.4

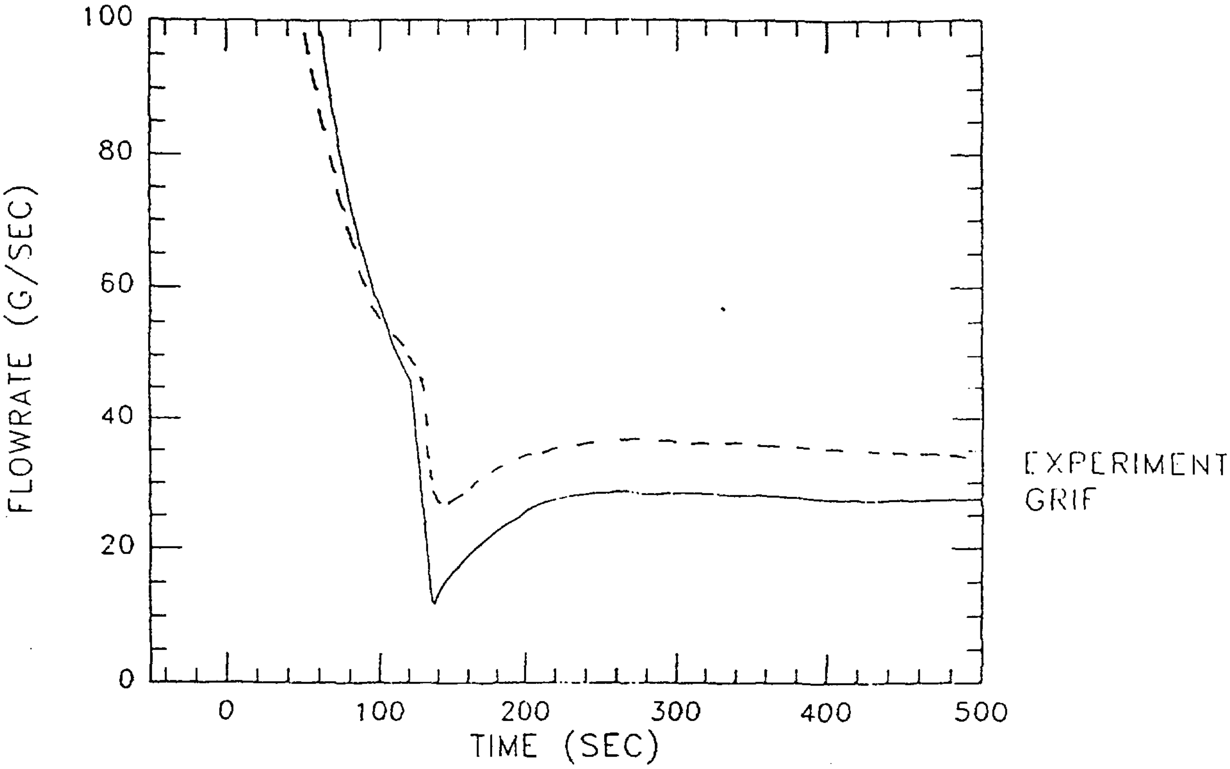


Fig.5 CORE FLOW RATE (0-500SEC)
(BENCHMARK-CASE1)

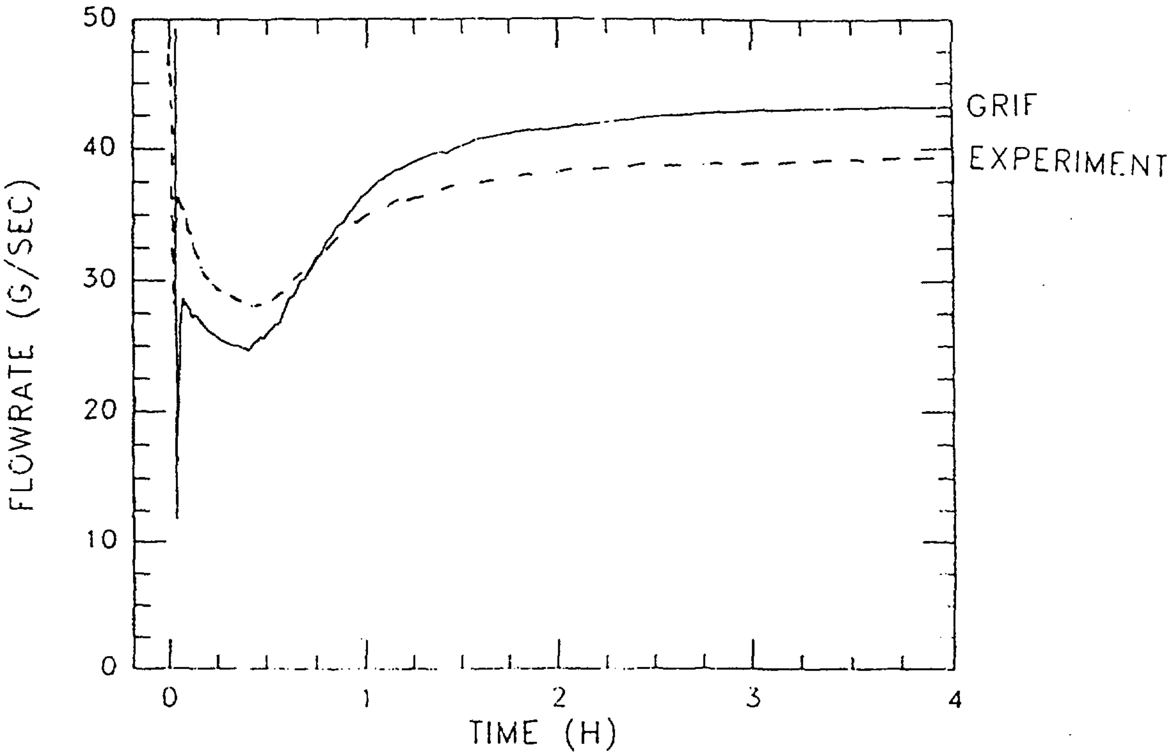


Fig.6 CORE FLOW RATE (0-4H)
(BENCHMARK-CASE1)

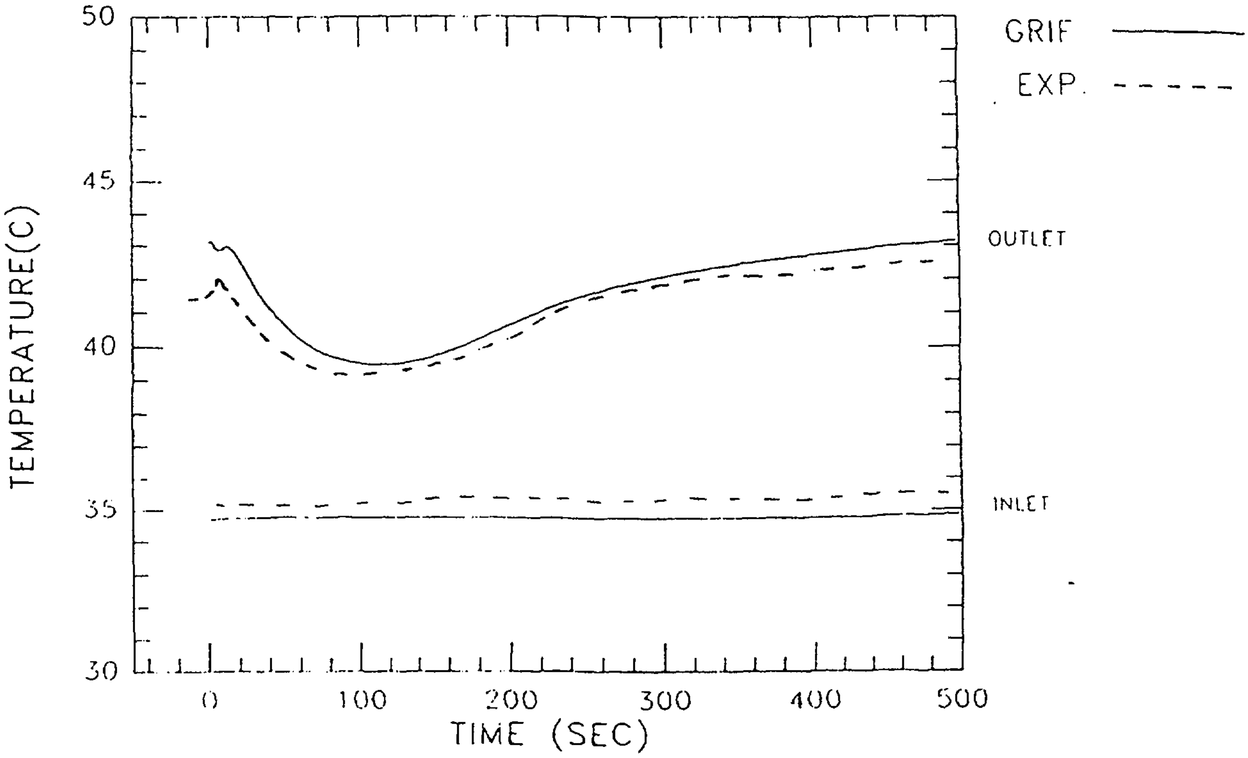


Fig.7 CORE INLET AND OUTLET TEMPERATURES (0-500S) (BENCHMARK-CASE1)

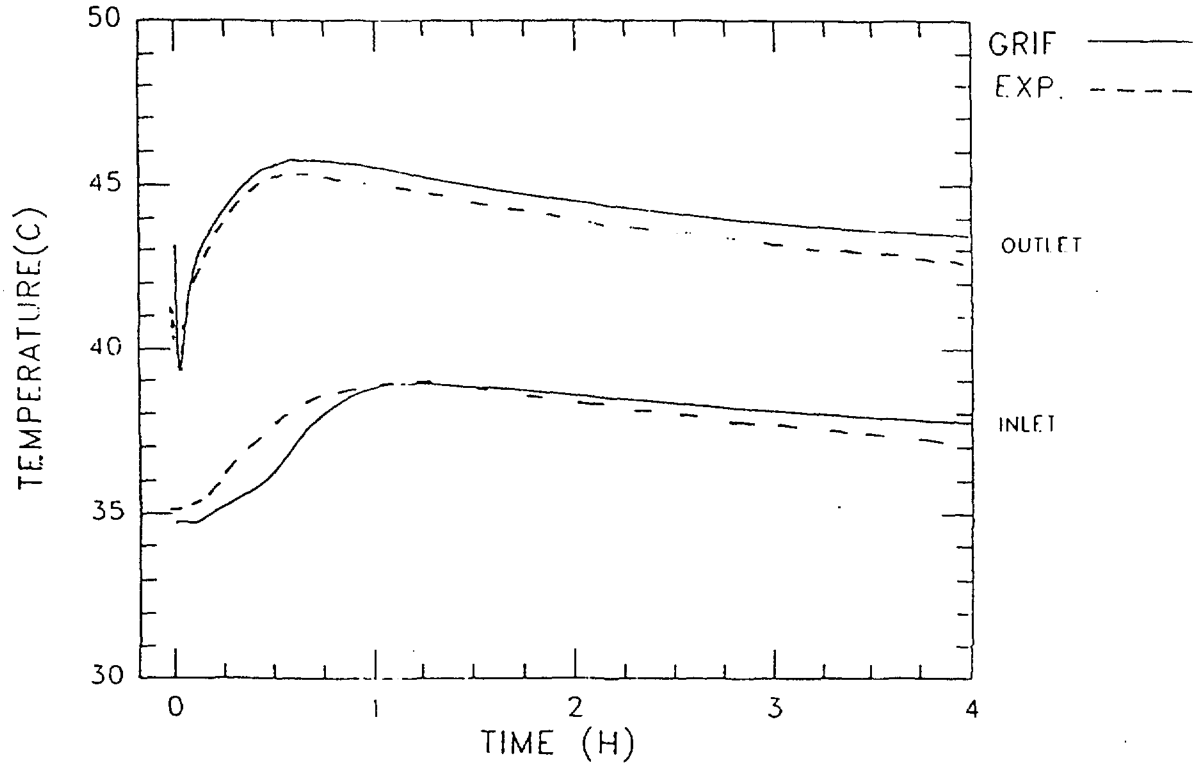


Fig.8 CORE INLET AND OUTLET TEMPERATURES (0-4H) (BENCHMARK-CASE1)

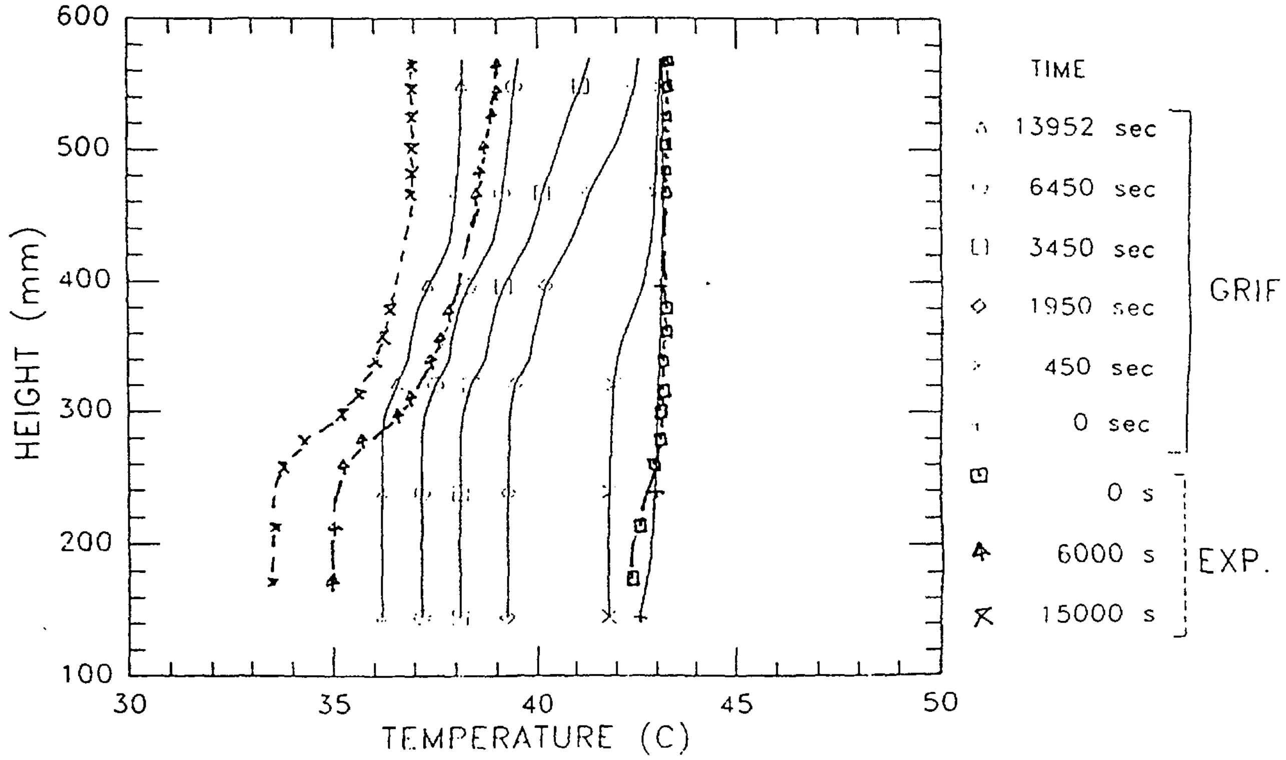


Fig.9 HOT PLENUM AXIAL DIRECTION TEMP. DISTRIBUTION (CASE1,HT23)

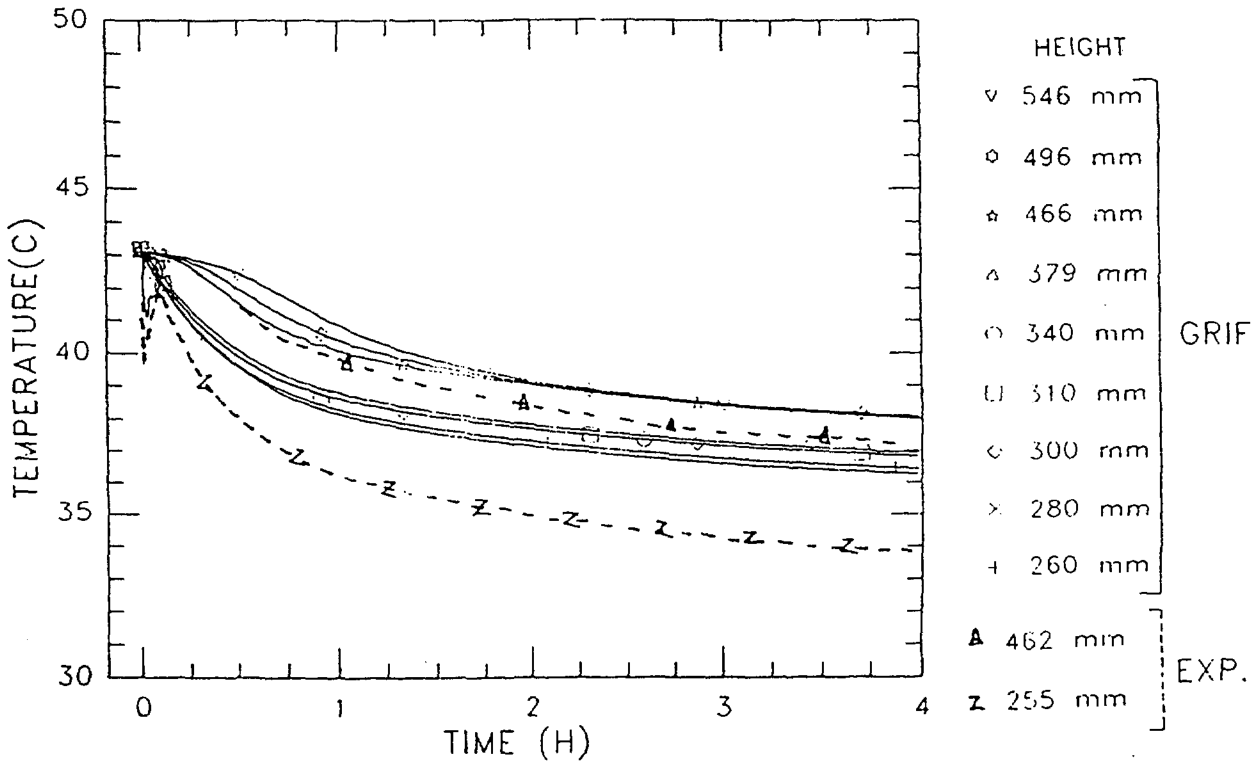


Fig.10 TEMPERATURE COURCES OF HOT PLENUM HT1-1 TREE (CASE1, 0-4H)

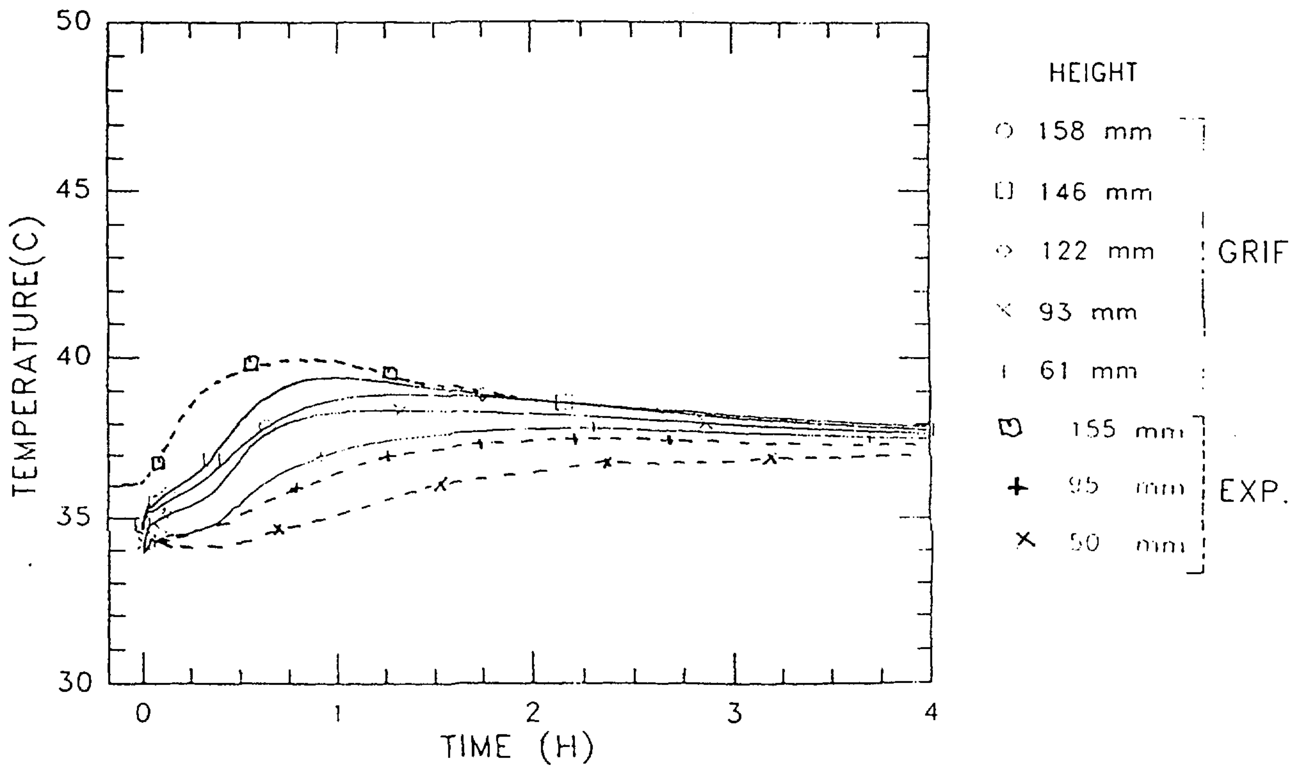


Fig.11. TEMPERATURE COURCES OF COLD PLENUM CD2 TREE (CASE1, 0-4H)

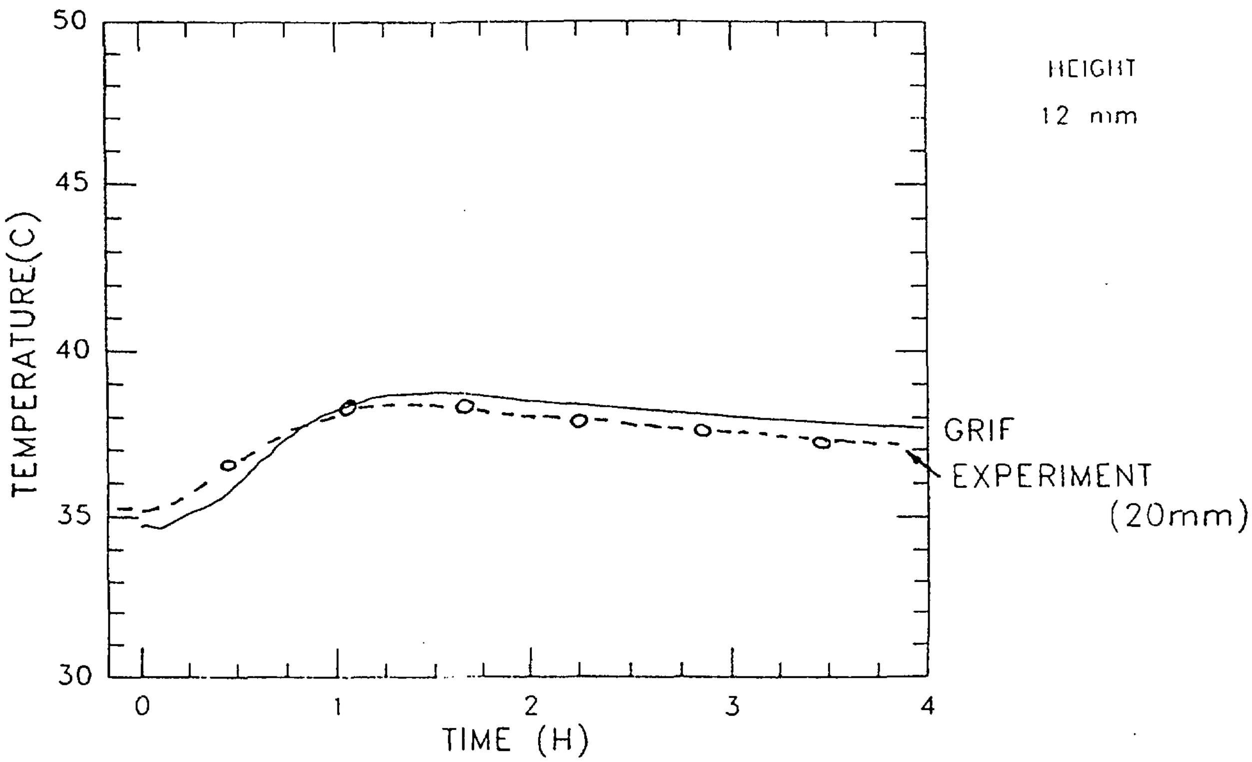


Fig.12 TEMPERATURE COURCES OF HIGH PRESSURE PLENUM HP2-1 TREE (CASE1, 0-4H)

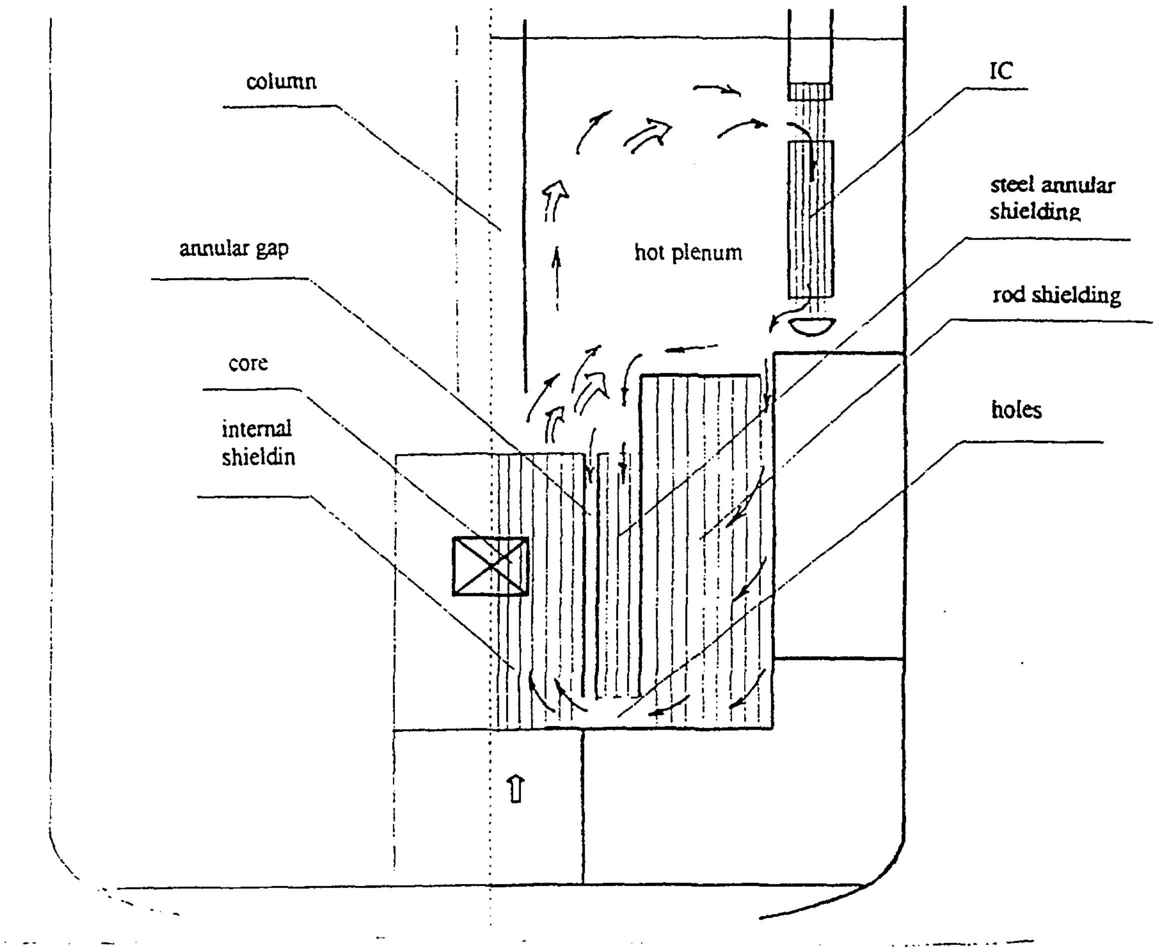


Fig.13. Schematic flow scheme of decay heat removal for BN-600M – type reactor.

⇨ - main sodium flow
→ - inter-wrapper sodium flow

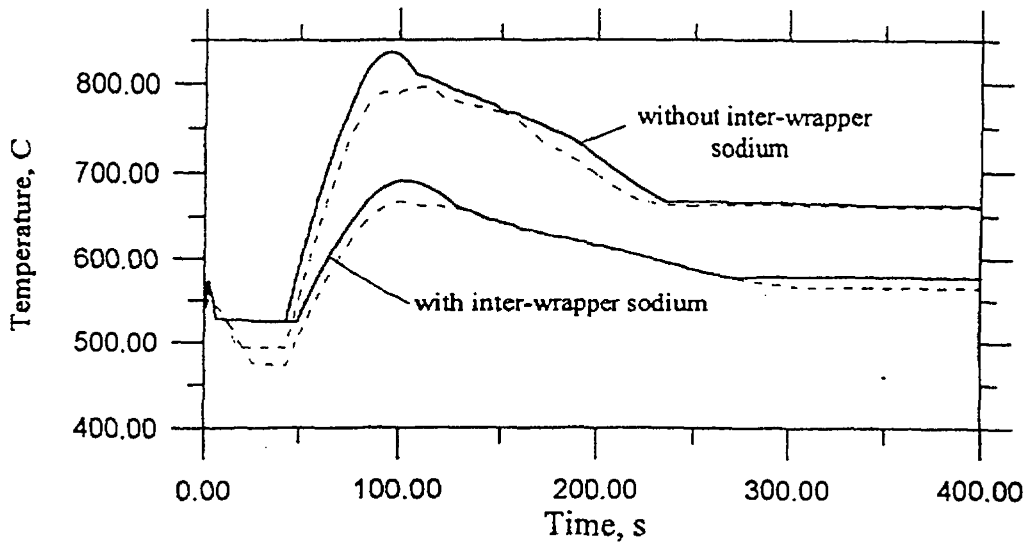


Fig. 14. Maximum sodium temperature history during LOF-accident.
— main sodium
- - inter-wrapper sodium

5.CONCLUSION

On the basis of the GRIFIC code testing it can be stated that for the pool-type plants :

- the GRIFIC code assures an adequate accuracy of integral characteristics (such as flow rate through the core, its inlet and outlet temperatures);
- it allows to calculate correctly local temperature values for the most components of the flow section;
- to improve the accuracy of the modelling one can recommend to enhance the order of finite difference scheme for approximation of convective terms.

NOMENCLATURE

U, V, W - velocity components in axial (z), radial (r) and azimuth (φ) directions;
ε - porosity of medium for coolant;
ν - effective kinematic viscosity;
ρ_T - coolant density;
Λ_z - porous medium resistance coefficient in axial,
Λ_r - radial,
Λ_φ - azimuth directions;
P - pressure;
T - temperature;
C_P - specific heat capacity of coolant;
λ_z - effective conductivity coefficient of porous medium in axial,
λ_r - radial,
λ_φ - azimuth directions;
Π - perimeter;
S - longitudinal section area;
α - heat transfer coefficient;
ΔP_N – primary pump head;
ΔL - primary pump dimension.

REFERENCES

[1] LEONCHUK Ì., CHVETSOV I., CHVETSOVA L. Calculation of Space-dependent Heat and Mass Exchange in (R,Φ,Z) – geometry. Preprint IPPE-1977., Obninsk 1989.(in Russian).

[2] WEINBERG D., KAMIDE H., MARTEN K., HOFFMAN H. Thermohydraulic Investigations on the Transition from Forced Nominal to Natural Circulation DHR Operation Conditions in the Reactor Model Ramona. Description of a Benchmark Problem. Karlsruhe, 1990.

[3] WEINBERG D., OSHIMA H., MARTEN K., HOFFMAN H. Thermohydraulic Investigations on the Transition from Forced Nominal to Natural Circulation DHR Operation Conditions in the Reactor Model Ramona. Experimental results of Case 1 and Case 2. Karlsruhe, 1991.