DEVELOPMENT OF GRIF-SM — THE CODE FOR ANALYSIS OF BEYOND DESIGN BASIS ACCIDENTS IN SODIUM COOLED REACTORS

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Abstract.

GRIF-SM code was developed at the IPPE fast reactor department in 1992 for the analysis of transients in sodium cooled fast reactors under severe accident conditions. This code provides solution of transient hydrodynamics and heat transfer equations taking into account possibility of coolant boiling, fuel and steel melting, reactor kinetics and reactivity feedback due to variations of the core components temperature, density and dimensions. As a result of calculation, transient distribution of the coolant velocity and density was determined as well as temperatures of the fuel pins, reactor core and primary circuit as a whole.

Development of the code during further 6 years period was aimed at the modification of the models describing thermal hydraulic characteristics of the reactor, and in particular in detailed description of the sodium boiling process. The GRIF-SM code was carefully validated against FZK experimental data on steady state sodium boiling in the electrically heated tube; transient sodium boiling in the 7-pin bundle; transient sodium boiling in the 37-pin bundle under flow redaction simulating ULOF accident. To show the code capabilities some results of code application for beyond design basis accident analysis on BN-800- type reactor are presented.

1. INTRODUCTION.

In order to evaluate consequences of severe beyond design accidents resulting in the sodium boiling in the core, several codes, such as SAS-4A, FRAX-5 and PHYSYRAC were developed in the late 70-ies - early 80-ies. These codes were mainly applied for the analysis of two beyond design accidents, namely ULOF and UTOP accidents. Inevitable core disrupture in these accidents was assumed for these codes development, and the main task was to analyze dynamics of the core heating, sodium boiling onset, fuel elements melting and molten mass relocation processes in order to predict possible core damage. Indeed, for the designs well-known at that time it was impossible to avoid fuel element melting in severe beyond design accidents, in particular, in the ULOF accident, since considerable positive component of the sodium void reactivity effect was caused by the core designs applied.

GRIF-SM code was developed in Russia approximately ten years later for the similar application [1]. By this time certain changes occurred in the understanding of the reactor safety problem. Self-protection concept was pushed into the foreground, according to which reactor design should provide decay heat removal by the passive means and assure core integrity even in severe accidents. Within the framework of this concept new BN-800 reactor core design was proposed with the sodium plenum replacing upper radial blanket. Owing to this measure it became possible to achieve near zero integral value of the sodium void reactivity effect. Moreover, since the sodium boiling onset in case of accident would occur on the upper core boundary, where the sodium void reactivity effect value is negative, this gave grounds for the hope for appropriate heat removal from the core under sodium boiling conditions avoiding fast fuel elements melting.

Taking into account the above considerations, during the development of GRIF-SM code attention was paid rather to the detailed description of the reactor thermal hydraulics on the stage of sodium boiling in order to prove the possibility to prevent fuel elements melting, than to the analysis of consequences of the core disrupture and molten fuel and steel relocation. This change of priorities was the main point determining initial structure of GRIF-SM code and the trends of its further modification.

Therefore the key features of GRIF-SM code caused by the specified task are as follows:

- trend towards maximum possible degree of detail in the modeling of space characteristics of sodium boiling in the core in order to improve accuracy of sodium void reactivity effect evaluation;
- modeling of space distribution of thermal hydraulics parameters not only in the core but also in the reactor as a whole. This is required, first, in order to take into account core channels coupling and, second, for more correct evaluation of the coolant flow rate in the primary circuit in case of vapor release into the upper reactor plenum.

2. GRIF-SM CODE DISCRIPTION

Code consists of several modules. The block diagram of the program is shown on fig.1. In a block "primary sodium thermal hydraulics" space distributions of the following characteristics of sodium vapour flow are calculated: a velocity, pressure, enthalpy,

GRIF-SM code structure

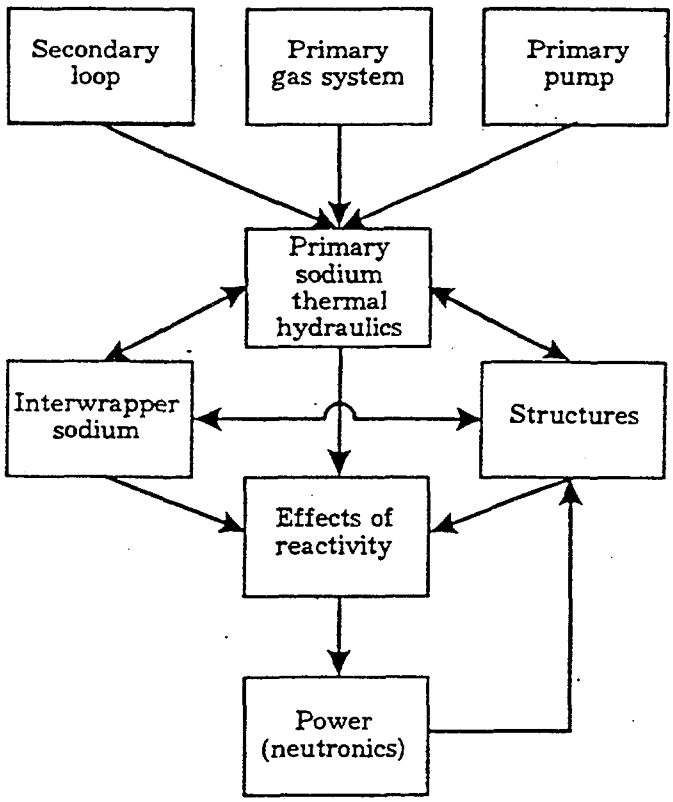


Fig.1

temperature, density and vapour quality for each step on a time. The system of equations of coolant thermal hydraulics includes:

continuity equation

$$\frac{\partial \rho}{\partial \tau} + \left(\vec{\nabla} \varepsilon \bar{u}\right) = G$$

equation of motion

$$\frac{\partial \bar{u}}{\partial \tau} + \frac{1}{\varepsilon}\left(\vec{\nabla} \varepsilon \frac{\hat{\psi}}{\rho} \bar{u}\right)\bar{u} = -\vec{\nabla} P - \hat{\Lambda}\bar{u} + \left(\vec{\nabla} \nu \vec{\nabla}\right)\bar{u} + \bar{F}$$

enthalpy equation

$$\varepsilon \frac{\partial \rho E}{\partial \tau} + \left(\vec{\nabla} \varepsilon \bar{u}\right) \frac{\psi}{\rho} E = \left(\vec{\nabla} \hat{\lambda} \vec{\nabla}\right) T + \varepsilon\left(\frac{\partial \rho}{\partial \tau} + \tilde{\psi}\left(\frac{\bar{u}}{\rho} \vec{\nabla}\right) P\right) + q_V$$

equations of state

$$\rho = \rho(P, E) \quad ; \quad T = T(P, E)$$

where main required variables are: u-mass velocity; P-pressure; ρ-density; E-enthalpy and coefficients: ψ,ψ,ψ- function of the slip ratio and sodium thermal hydraulic properties; ε-porosity; Λ-friction coefficient; λ-thermal conductivity; ν-viscosity.

The sodium boiling is described by slip-model of a two phase flow. The factors of friction and slip ratio for two phase flow are defined on Loccart-Martinelly correlation. There are taken into account in GRIF-SM code then main kinds of heat transfer in one and two phase flow: forced liquid and vapour sodium flow; boiling of under heated sodium; bubble boiling; film boiling; dryout (in case, when the size of a heat flux exceeds critical significance) and film condensation.

Dynamics of distribution of temperature in pins, control rods, subassembly wrappers, grids and other primary circuit elements, essentially influencing on a course of development of accident, is calculated in a block a "Structure".

The core is submitted by set of parallel channels. For each channel is calculated 2D distribution of temperature in a fuel and cladding for one pin in the channel. After the beginnings of fuel or clad melting the transient position of molten cavity is calculated in view of latent heat of melting. Model of thermal destruction of fuel pin is realized.

Two-dimensional temperature distribution is also calculated for SA wrapper as the distribution of temperature on thickness of a wrapper is important for the correct simulation of sodium vapour condensation in the top part of core subassemblies where heat release is small. Inter-wrapper sodium thermal hydraulics is simulated on base 1D model in a separate module.

The intermediate heat exchanger model is a part of the "Secondary loop" module. It includes equation of thermal conductivity for IXH tubes and 1D energy equation for the secondary sodium. The temperature and flow rate of secondary sodium on the IXH entrance versus time are specified.

The model of a primary gas system is intended for determination of changes in time of pressure in the reactor gas cavity, which is necessary for a module "Primary sodium thermal hydraulics" as a boundary condition. The primary gas system consists of several cavities, connected with one another, which can be partially filled in liquid and vapour sodium and inert gas. At severe accident sodium vapour can get in a gas system. The opportunity of its condensation at contact to "cold" designs is also considered.

The reactor power time dependence is calculated within the frame of "point kinetics" model with 6 groups of delayed neutrons.

During the reactivity calculations the following thermal effects are taken into account: sodium density reduction due to thermal expansion and boiling; axial and radial expansion of a core as a whole; fuel and clad material density changing; expansion of control rods and its drives; Doppler effect.

3. TESTING OF GRIF-SM CODE USING EXPERIMENTAL DATA

Results of testing of the code on KfK experimental data on sodium boiling in pipes, 7-pin and 37- pin bundles will be below also indicated.

3.1. Steady state sodium boiling in electrically heated pipe.

In this case in each of experiments the position of boiling point on height and pressure drop in the tubular test section were measured for different sodium velocity values varied in the range 0.5-5 m/s. The test tube had inner diameter 0.0066 m and with direct electrical heating, thermal power up to 300 W/cm^2 was available. As it follows from Fig.3 and Fig.4, in the total investigated range of coolant velocities on both parameters good agreement with experiment is observed.

Multicomponent model of clad melting

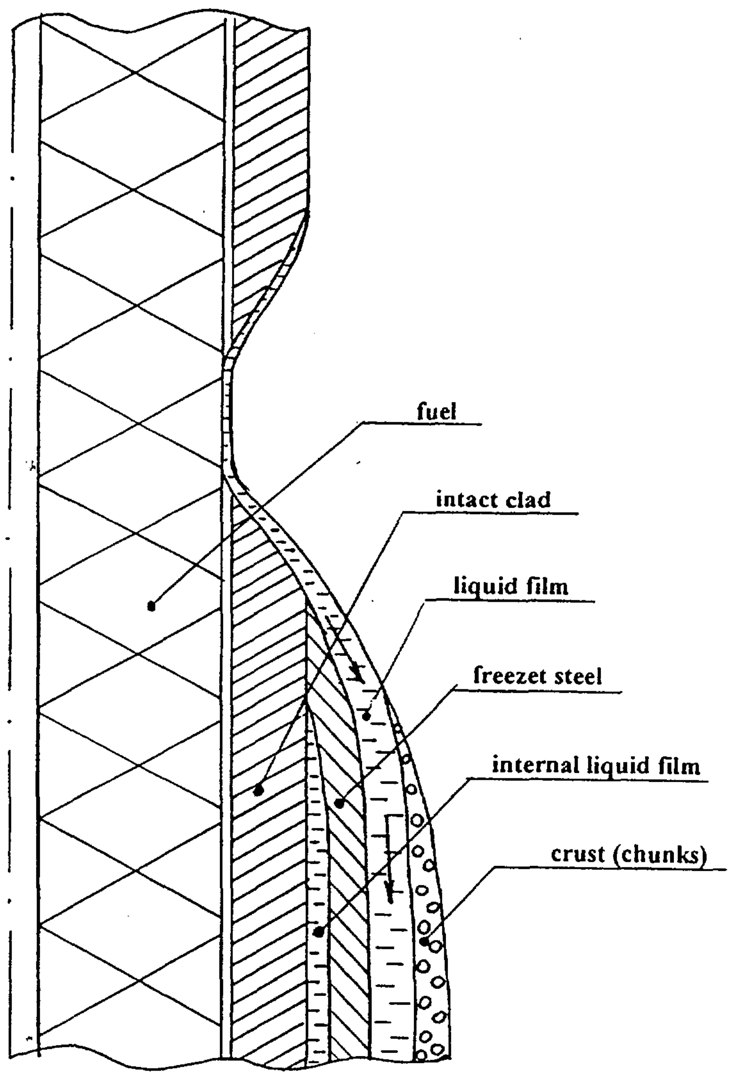
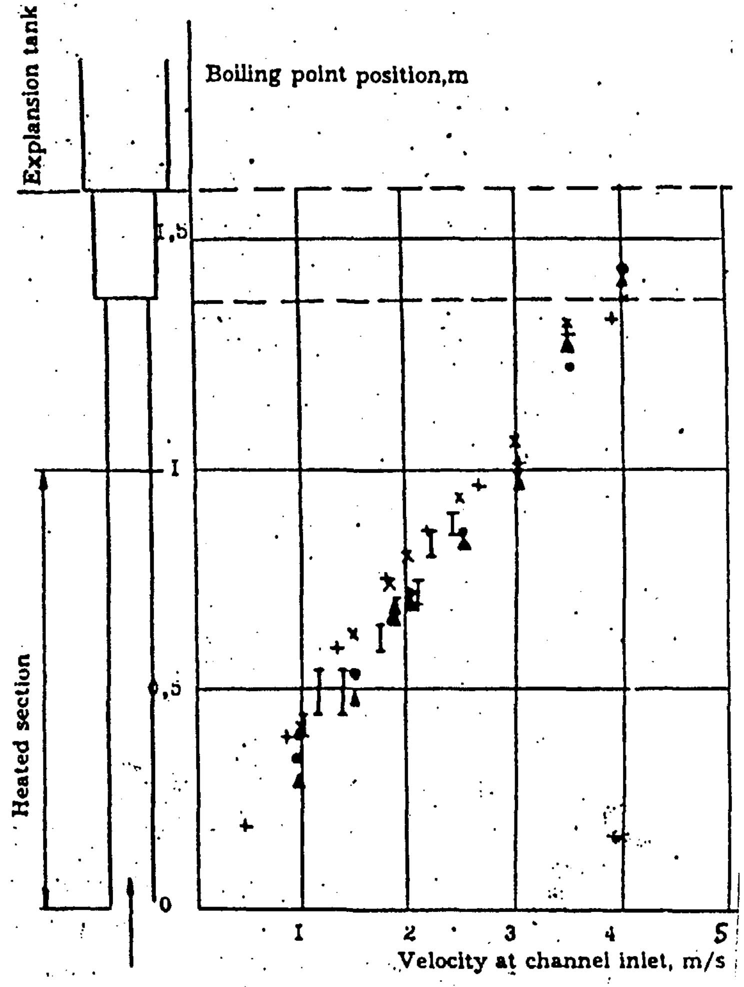


Fig 2

Boiling point position as a function of sodium velocity at the experimental channel inlet.

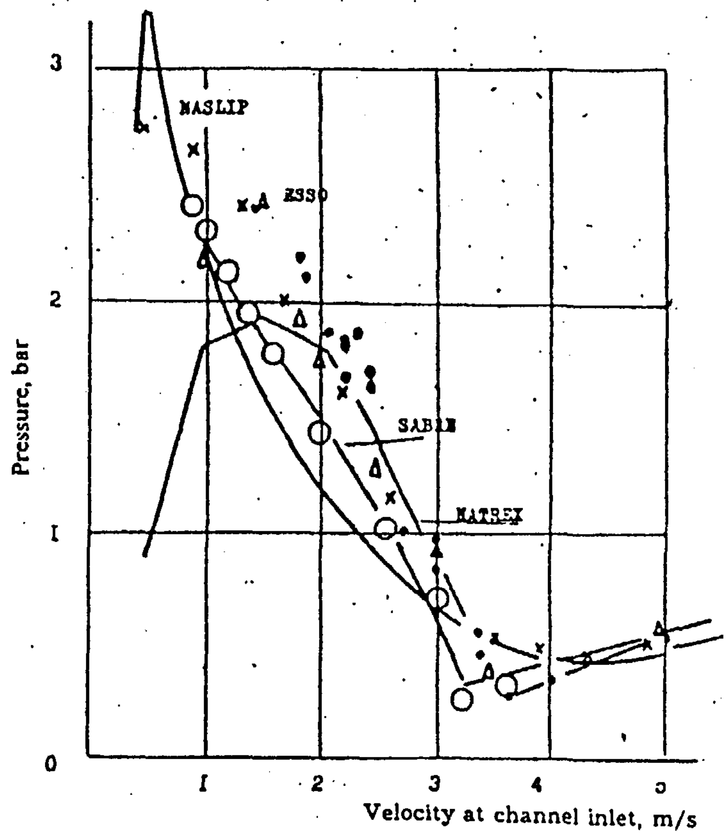


I -experiment ▲ -ESSO ● -NATREX

X -SABRE + -GRIF-SM

Fig.3.

Pressure drop between the points P11 and P14 as a function of sodium velocity at the channel inlet.



• -experiment X -GRIF-SM

O -NASLIP Δ -ESSO

Fig.4.

3.2. Dynamics of sodium boiling in 7-rod bundle.

The testing section was 7-pin assembly, put in a hexagonal wrapper (Fig. 5). Heated part of the bundle had a length of 0.6 m. Heaters power is constant and equal to 114 kW. During experiment the sodium flow rate was linearly reduced from 100 % to 30 % for 10 seconds and then remained constant (Fig. 6). Sodium temperature and volumetric vapour quality in various cross-sections on height were measured. In accordance with flow rate reduction sodium temperature at the outlet a heated part of the bundle increases (Fig.7). The boiling in experiment begins in 9 seconds after beginning of transient in the unheated part of bundle. In accordance with GRIF-SM calculations the boiling begins in the same point, but with small delay near 0.15s. The process of moving of the lower boundary of vapour bubble is equally well calculated by both codes, as to the upper boundary, as it is visible from Fig.8. GRIF-SM gives a little overestimated velocity of its movement in comparison with experiment, while COMMIX-2 underestimates it.

3.3. Modelling of ULOF accident on 37-rod assembly.

Main geometrical characteristics of test assembly were close to those of the SNR-300 reactor core subassembly (Fig.9). The pin simulators of 0.006 m diameter were located in hexagonal wrapper with a pitch of 0.0079 m. The real distribution of power rating over height

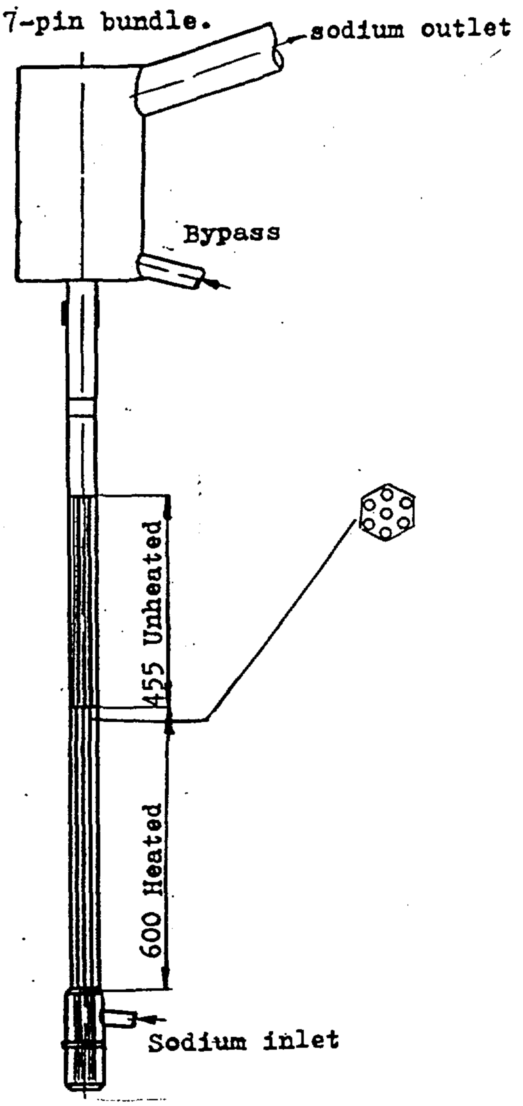


Fig.5.

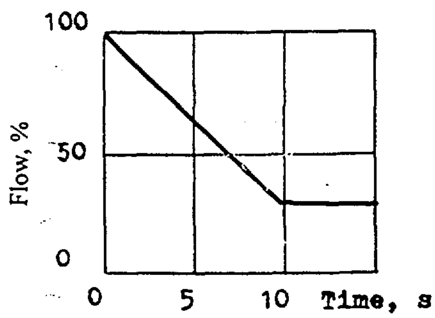


Fig.6.

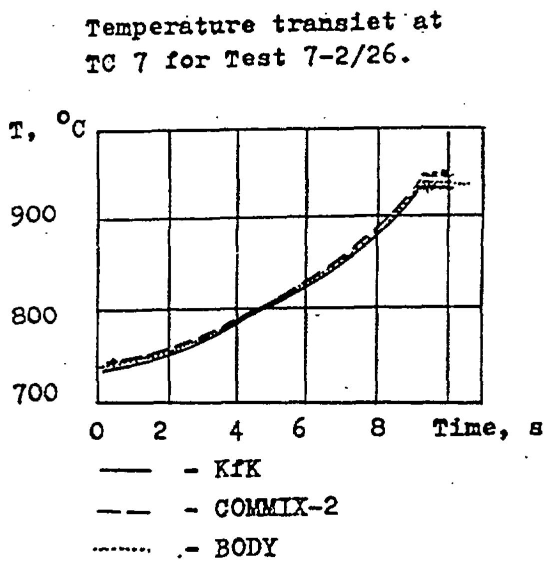
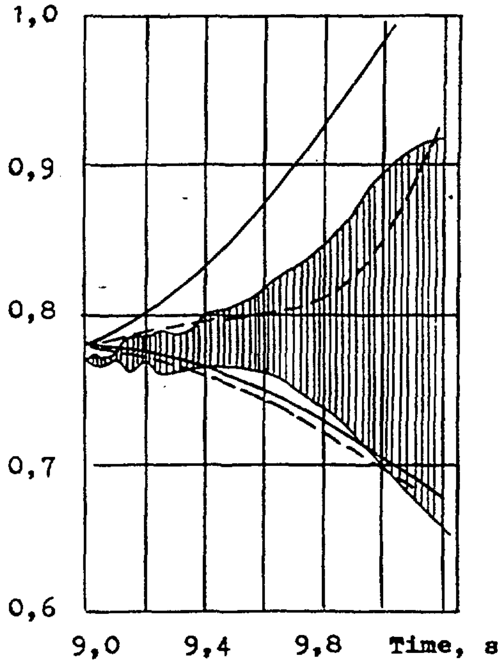


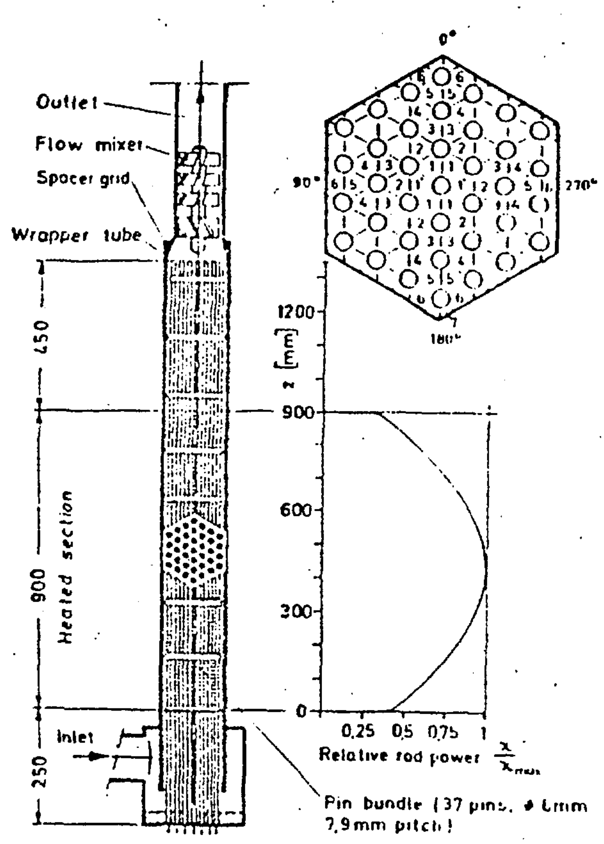
Fig.7.

Transiet axial limits of
disposal sodium vapor.



- KfK
- COMMIX-2
- BODY

Fig.8.



Schematic diagram of the KNS 37 pin bundle

Fig.9.

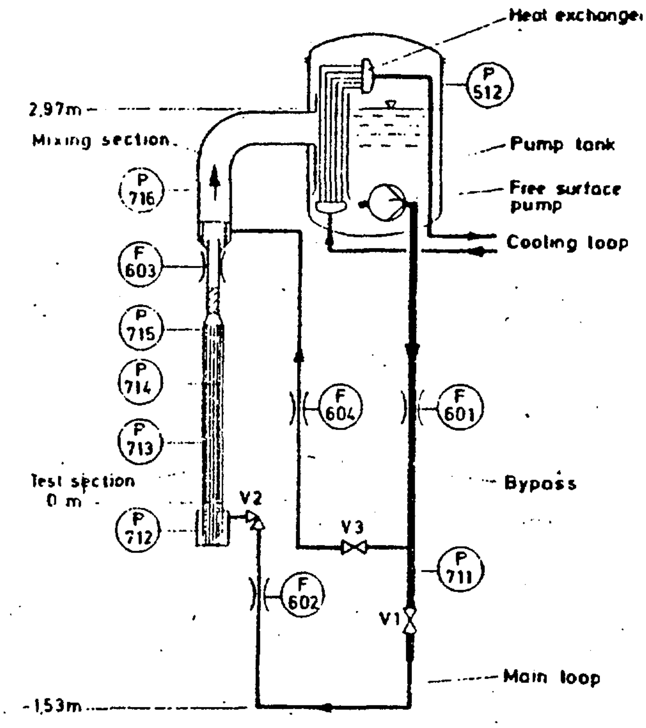
was taken into account. Maximum linear power of one element was 320 W/cm. The experiment consisted of decrease of the coolant flow rate through assembly according to the law, close to hyperbolic. The power remained constant up to the moment, when there was threat of thermal destruction of simulators. During experiment, pressure and sodium flow rate in various elements of a rig (Fig.10), as well as temperature and vapour quality in various points of assembly were measured. The obtained data permit to look after the movement of fronts of boiling either at the height of assembly, or at its cross section.

Experimental results were calculated using 2D version of the GRIF-SM code. The whole assembly was simulated in calculations, including inlet device, rod bundle and outlet flow mixer. The results of comparison of GRIF-SM calculations with experimental data are given in Fig.11-13.

There is a good agreement between calculated inlet flow rate and experimental results (Fig.11), but the flow reverse occurs approximately tenth of second later in calculations. Another difference is that GRIF-SM calculation gives the oscillations of flow rate after the boiling onset. As it follows from Fig.12 and Fig13, GRIF-SM simulates in appropriate way the vapour generation and boiling fronts spreading till the moment of power termination. The behaviour after power termination is similar to that of the peripheral channels but cooling of sodium vapour in the top part of central channels seems to be underestimated in the calculations.

3.4. ULOF accident simulation on BN-800 type reactor.

It ought to consider as a verification activity the participation of GRIF-SM code in comparative calculations of ULOF-accident on BN-800-type reactor in which also the codes



Simplified flow sheet of the KNS sodium boiling loop

Fig.10.

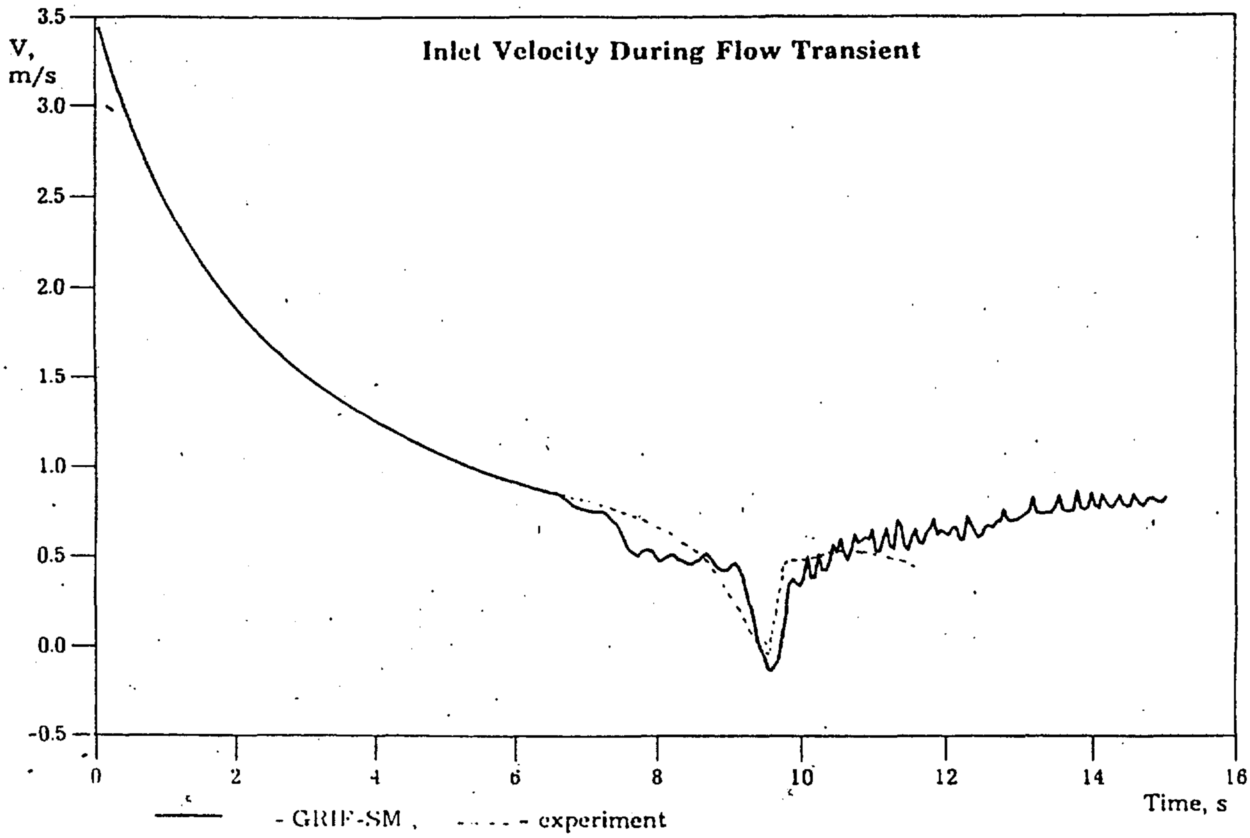
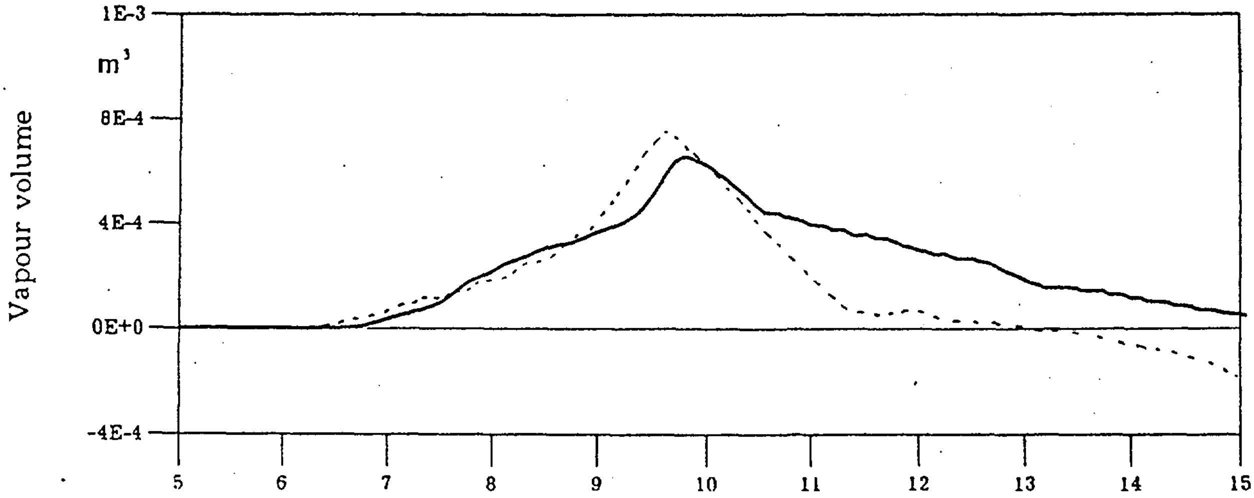


Fig.11.

Vapor volumes as a function of time



- GRIF-SM
- Experiment

Fig.12.

Axial development of the boiling boundary

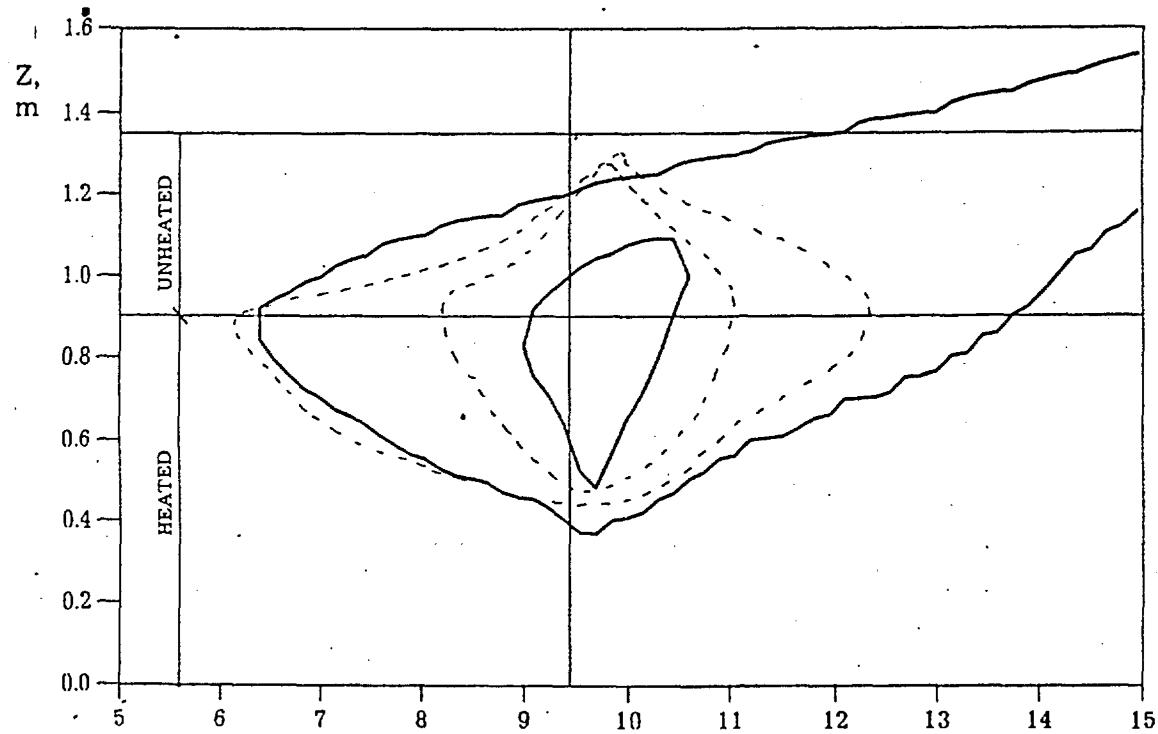


Fig.18 - GRIF-SM
- Experiment

Fig.13.

SAS-4 A (Germany, FZK), FRAX-5 (Great Britain, AEA) and PHYSIRAC (France, CEA) participated. Comparison has shown, that all of them have given rather similar results.

4. TRENDS OF THE CODE MODIFICATION

Sodium worth in fast reactors strongly depends on space coordinates, and zone of positive sodium void reactivity effect values is kept in the central part of the core. Therefore it is rather important to provide as high as possible accuracy in description of sodium density distribution over the core radius. For this purpose, the number of channels consisting the core has been increased to 30. This makes significant improvement of the calculation accuracy.

The experiments have shown, that in fast reactors considerable heterogeneity of the SA radial temperature profile can occur. Usually, owing to the heat transfer to the sodium flowing outside SA, the coolant temperature in the peripheral channels of the SA is lower than that in the central channels. As a result of this, sodium boiling would not start at once over the whole SA cross section area, but first in the SA centre with its further gradual propagation to the periphery. In order to simulate both general (over the reactor core) and local (over SA) distribution of thermal hydraulic parameters a new version of GRIF-SM code was developed which was called BOS (Boiling Subassembly) [2].

In the typical fast reactor ~ 10% of the core flow cross section relates to the inter-wrapper sodium. Its temperature and hence its density during the transient may be significantly different from those of sodium flowing inside fuel subassemblies.

In order to make proper evaluation of distribution of the inter-wrapper sodium temperatures the following points should be taken into account:

(1) possibility of lateral flow of inter-wrapper sodium in addition to the axial flow;

(2) possibility of its boiling onset in sufficiently long duration processes.

This was made for the modified version of the code, i.e. when the inter-wrapper sodium thermal hydraulics is described within the framework of two-dimensional two-phase model.

TABLE I. DEVELOPMENT HISTORY OF GRIF-SM CODE

Module	Initial version	Updated versions
1. Primary sodium thermal hydraulics.	4 channel presentation of the core.	30 channel presentation of the core. 2D-simulation of velocity and temperature distribution inside subassembly (BOS code).
2. Inter-wrapper sodium	1D and 1 phase thermal hydraulic model.	2D and 2 phase thermal hydraulic model.
3. Structures.	The set of 1D-3D models for temperature calculations in fuel pins, SA wrappers. Simplified 'thermal' model of pin melting.	Multicomponent model of pin melting and molten steel relocation (CANDLE module).
4. Neutronics.	"Point kinetics" with 6 group of delayed neutrons.	3D space time kinetics (GVA-code-{GRIF-SM+VOLNA}).

If the assumptions concerning accident scenario are too conservative, conditions for the sodium burn-out appear resulting in the fuel element melting which occurs very soon. In order to make analysis of such accidents within the framework of the modified code version, CANDLE code module has been developed [3], which is capable of evaluating molten steel relocation within the core.

The main characteristics of the model used as a basis for CANDLE module are as follows:

(1) 5-component simulation of pin clad melting:
  - intact clad,
  - liquid steel film,
  - frozen steel,
  - crust,
  - *internal liquid film.*

(2) 1-D thermal hydraulics models for every component.
(3) Components a), c), e), are fixed, components b) and d) are moving together.
(4) Molten fuel is fixed.
(5) Intact fuel stubs are fixed.

Relative location of the model components is shown in Fig.2.

Another trend of the code modification concerns development of algorithms of joint solution of thermal hydraulics and space neutronic kinetics equations. Within the framework of point kinetics, components of the core reactivity are usually represented as algebraic functions (which are often linear) of material concentration values, coefficients in these relationships being considered constant during accident. Actually, if significant disturbance of the initial configuration takes place, i.e. vapor bubbles are formed and steel and fuel melt is relocated within the core, then material worth become strongly dependent on changing configuration. This effect is necessary to take into account when determining either amplitude or form factors of the power density values. Code package called GVA implements joint space and time analysis of neutronic and thermal hydraulic reactor characteristics. This package is actually a combination of two codes, namely GRIF-SM and VOLNA, the latter being neutronic code capable of solving equations of space and time neutron transfer using quasi-stationary method [4].

5. APPLICATIONS

Below results of analysis of some beyond design accidents are given with reference to the BN-800 reactor having near zero value of integral sodium void reactivity effect. There are three radial enrichment zones in the reactor core, and the upper axial blanket is replaced by the sodium plenum.

Calculation area simulating the reactor is shown in Fig.14. The core is represented by 10 parallel annular channels, each of them being divided into three subchannels having different fuel burnup values.

5.1. ULOF accident

After the main pumps are shut down, sodium flow in the primary and secondary circuits decreases according to the near hyperbolic law. Core sodium and fuel element cladding temperatures increase, whereas the fuel temperature decreases because of the reactor power reduction. As a result of this, negative contribution is given to the reactivity value by the sodium density effect, radial core expansion and control rod drive bars thermal expansion (Fig. 15). Doppler effect and axial core expansion give positive components of the reactivity. The total temperature reactivity effect remains negative, and the reactor power is decreased.

However, on the 38th second sodium boiling starts in the 5/1-st channel of the medium enrichment zone.

Calculational region simulating BN-800 type reactor geometry

k															
29	25	25	25	25	25	25	25	25	25	25	25	25	28	25	25
28	25	25	25	25	25	25	25	25	25	25	25	25	28	28	37
27	25	25	25	25	51	51	51	51	51	51	51	51	28	28	37
26	25	25	25	25	51	51	51	51	51	51	51	51	28	28	30
25	25	25	25	25	51	51	51	51	51	51	51	51	27	27	30
24	51	51	51	51	51	51	51	51	51	51	51	51	27	27	30
23	51	51	51	51	51	51	51	51	51	51	51	51	27	27	30
22	33	34	35	36	24	24	24	24	24	24	24	24	27	27	30
21	45	46	47	48	49	49	49	49	49	49	52	69	27	27	30
20	40	41	42	43	44	44	44	44	44	44	52	68	27	27	30
19	40	41	42	43	44	44	44	44	44	44	52	68	27	27	30
18	40	41	42	43	44	44	44	44	44	44	52	68	27	27	30
17	18	19	20	21	22	22	22	22	22	22	52	32	27	27	30
16	13	14	15	16	17	17	17	23	23	23	52	39	27	27	30
15	13	14	15	16	17	17	17	23	23	23	52	39	27	27	30
14	13	14	15	16	17	17	17	23	23	23	52	39	27	27	30
13	13	14	15	16	17	17	17	23	23	23	52	39	27	27	30
12	13	14	15	16	17	17	17	23	23	23	52	39	27	27	30
11	13	14	15	16	17	17	17	23	23	23	52	39	27	27	30
10	13	14	15	16	17	17	17	23	23	23	52	39	27	27	30
9	13	14	15	16	17	17	17	23	23	23	52	39	27	27	30
8	13	14	15	16	17	17	17	23	23	23	52	39	27	27	30
7	13	14	15	16	17	17	17	23	23	23	52	39	27	27	30
6	13	14	15	16	17	17	17	23	23	23	52	39	27	27	30
5	13	14	15	16	17	17	17	23	23	23	52	39	27	27	30
4	53	54	55	56	57	57	57	57	57	57	52	57	27	27	30
3	63	64	65	66	67	67	67	67	67	67	67	67	27	27	30
2	2	3	4	5	6	7	8	9	10	11	12	31	27	27	29
1	1	1	1	1	1	1	1	1	1	1	1	1	26	26	29
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15 i

1 -lower inlet header;
2,3,4,5,6,7,8,9,10,11,12,31 -subassembly inlets
13,14,15,16 -low enrichment zone;
17 -middle enrichment zone;
23 -high enrichment zone;
18,19,20,21,22,32 -pin tail zone;
24,33,34,35,36 -subassembly heads and outlet windows;
45,46,47,48,49 -upper axial shield;
40,41,42,43,44 -sodium layer;
53,54,55,56,57 -bottom axial blanket;
63,64,65,66,67 -pin gas volumes;
52 -radial blanket;
39,68,69 -inner radial shield and storage;
51 -upper outlet plenum;
26 -entrance region
29 -pump and downcomer;
27 -intermediate radial shield;
28 -rod shield;
30, 37 -intermediate heat exchanger;
25 -upper column and other impermeable elements;

Thick solid lines - impermeable boundaries
Double think solid lines - the boundaries between zones with different enrichment

i-radial number of unit
k-axial number of unit

Fig.14.

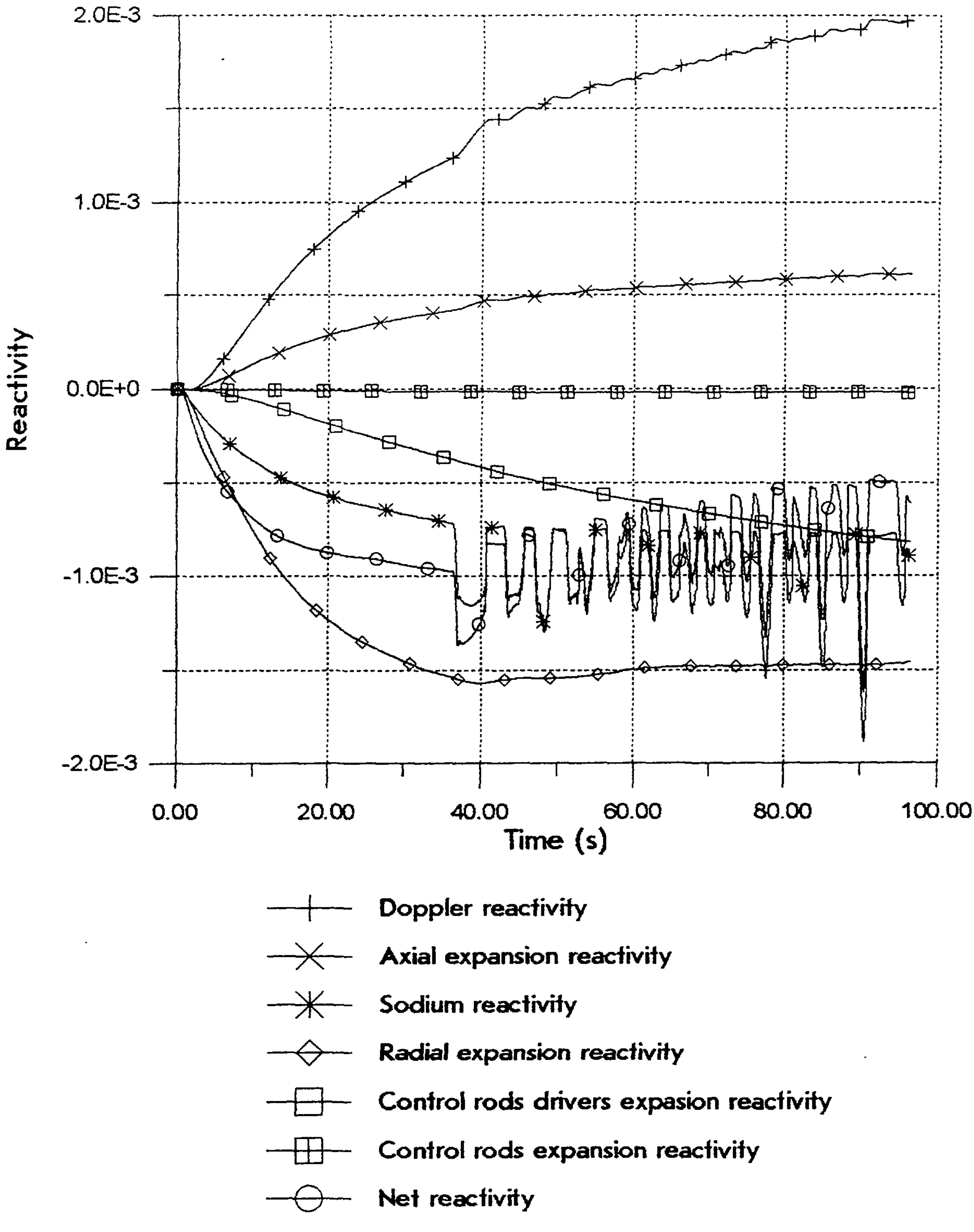


Fig.15. **Reactivity versus time.**

The sodium-vapour mixture is spreading into the sodium volume and the upper axial blanket causing negative effects of reactivity, power decrease and cessation of boiling. In the process an increase of power and sodium boiling take place again. And again sodium vapour movement into the sodium volume and the upper axial blanket results in the negative effects of reactivity, power decrease and cessation of boiling (Fig. 18). In the reactor, self-sustained oscillations of power set up with periodic coolant boiling. Reactor power in this case is oscillating and gradually decreases (Fig. 16).

There are oscillations of the outlet sodium temperature, which does not exceed saturation temperature. Slight oscillations of maximum fuel temperature also occur (Fig. 17), this temperature approaching sodium boiling point. Thus, neither fuel nor cladding melting takes place.

5.2. ULOF accident analysis taking into account parameters distribution over the subassembly cross section.

ULOF accident version with the deliberately increased severity has been considered with reference to the BN-800 reactor. In this analysis, two negative components of temperature reactivity effect were neglected, namely core radial expansion and control rod drive bar expansion components. The calculation was made using GRIF-SM code initial version and BOS code version.

The difference between these methods is more significant after the boiling start (Fig.19). Owing to the non-uniformity of SA temperature profile boiling starts earlier, namely: 13.18 s (BOS) and 17.66 s (GRIF-SM). As it was observed before, sodium boiling is initiated at the core outlet in the SA central area of the "hottest" channel 5/1. Then the boiling starts successively in 1/1, 2/1, 3/1, 4/1, 7/1 and 5/2 channels. Along with the increase of the number of boiling channels vapour bubble growth in the SA is observed. Fig.20 shows spatial distribution of mass velocity for all channels where the boiling has started by this time. It is obvious that on this stage the boiling occurs only in the upper part of the most subassemblies (where the sodium void reactivity effect is negative), However the height of the boiling area in 5/1 channel is 42% of the core height, and hence the boiling front has penetrated into the area of positive sodium void reactivity effect.

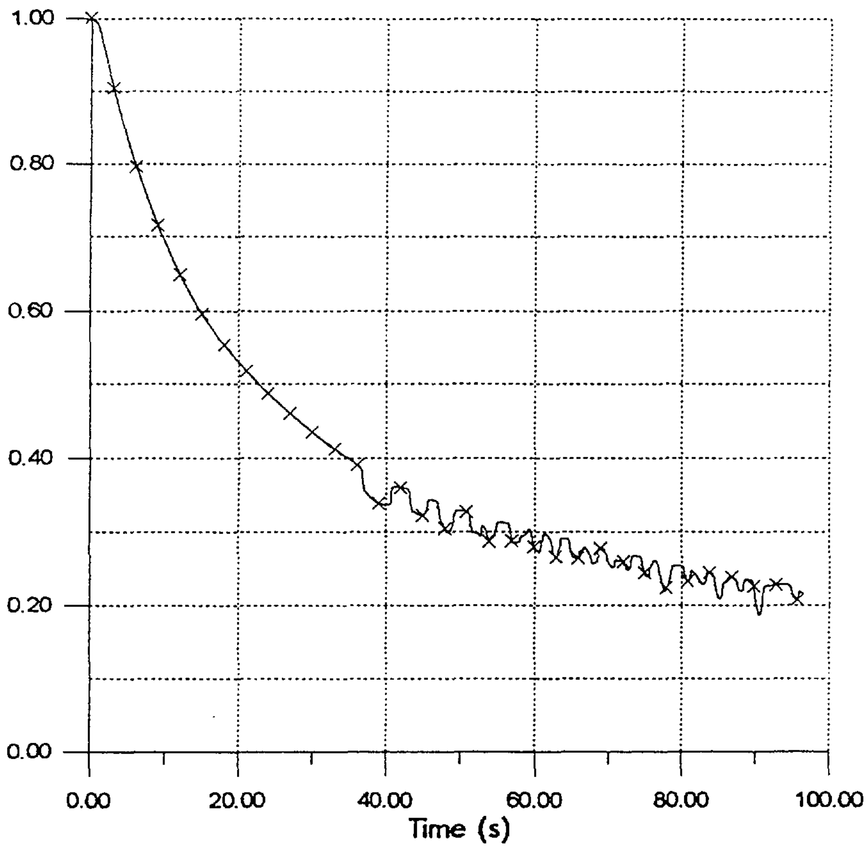


Fig.16. **Relative reactor power.**

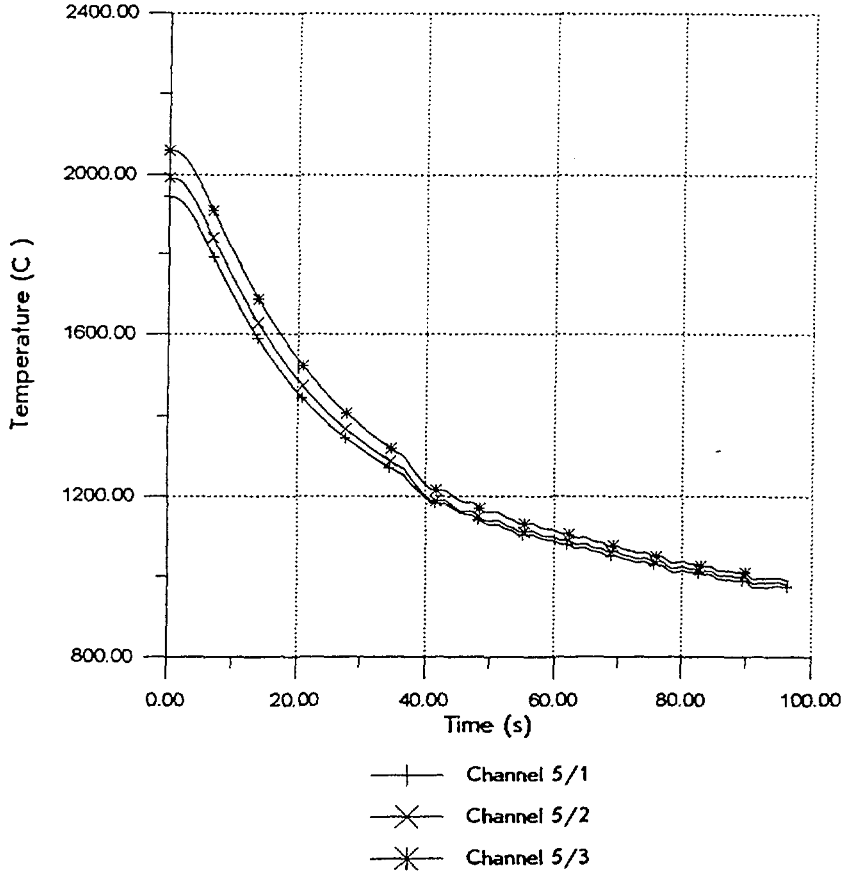


Fig.17. Fuel inner surface temperature in the central cross section of the core.

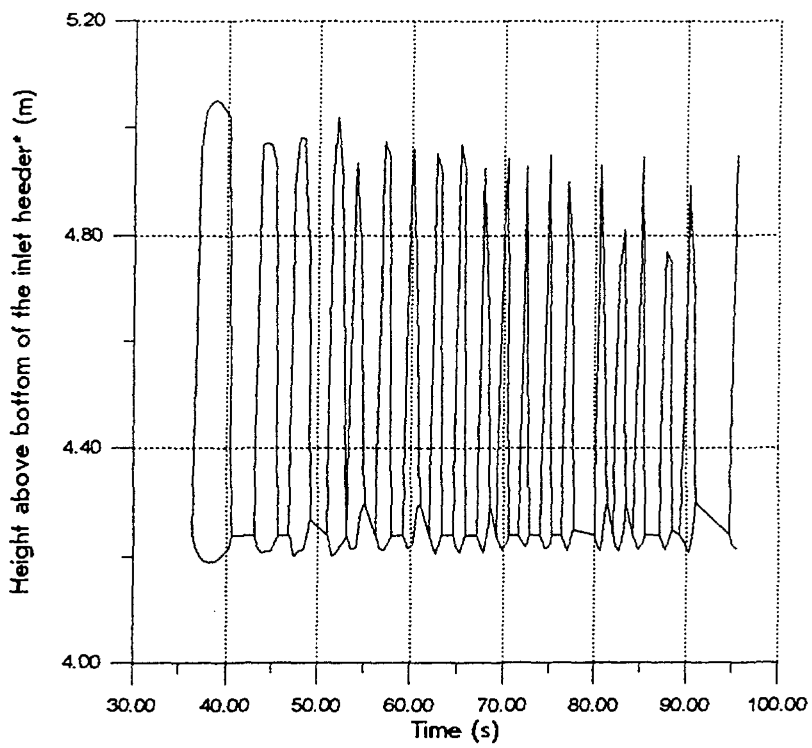


Fig.18. . Channel 5: Boiling front.

*) The height of the fissile zone is 4.24 m.

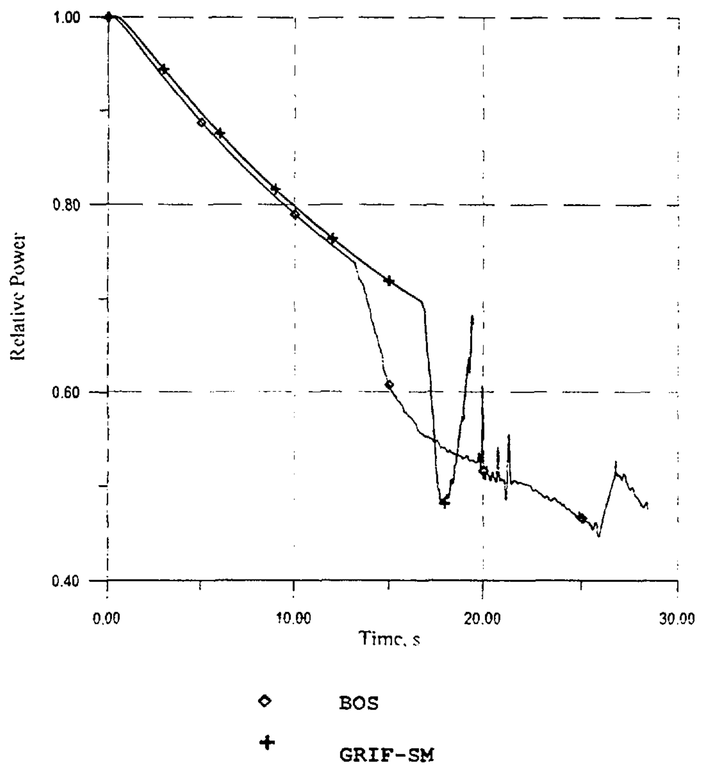


Fig.19. . ULOF accident on BN-800 like reactor. Radial effect is neglected.

ULOF accident on BN-800 like reactor.
BOS - calculation.

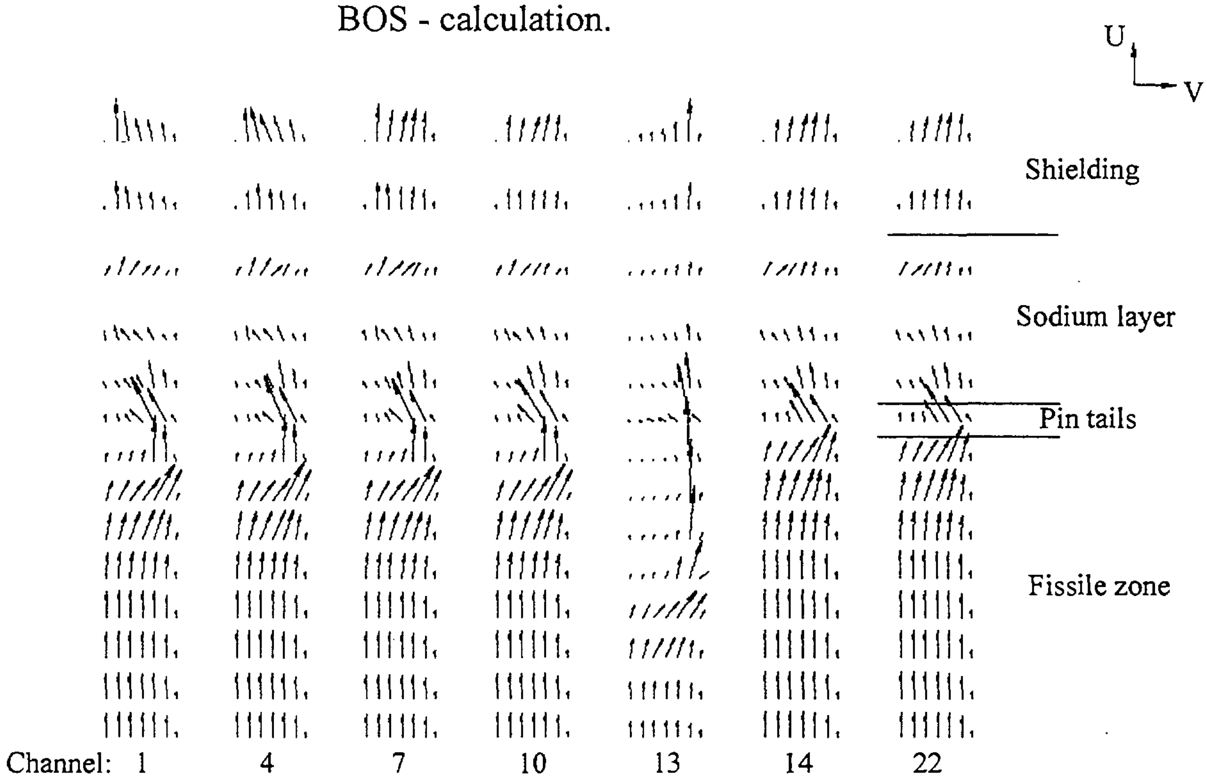


Fig.20. Mass flow rate distribution on subassembies (U=U, V=V*10) , [kg/(s·m·m)].
Time = 22 s.

It would be interesting to compare the results on reactor power for the boiling stage obtained using GRIF-SM code (i.e. taking into account parameters distribution within the subassembly) and BOS code. In the GRIF-SM evaluations, the following scenario was realized: sodium boiling was initiated on the upper boundary of the core, and therefore additional negative reactivity was first inserted causing power decrease. However, according to the essence of one-dimensional model of SA sodium boiling, taken for GRIF-SM code, onset of boiling means blockage of the total SA cross section area by the sodium vapour bubble. SA pressure drop decreases too fast, and sodium flow decreases causing in its turn increase of the vapour generation rate. Intensive expansion of the vapour bubble results in the vapour penetration into the area of positive sodium void reactivity effect and power increase.

Entirely different picture is obtained when using BOS code. Vapour bubble is formed first in the central part of SA cross section, where it is passed over by the single-phase coolant flow (Fig.20). As a result, SA pressure drop increases more slow, the boiling process is more stable, and the rate of the reactor power decrease is lower.

Thus, the conclusion can be made on that from the standpoint of boiling stability one-dimensional boiling model gives deliberately conservative results.

However, in spite of more favorable course of the accident dry-out conditions are achieved in 5/1 channel by 20^{th} second, followed by fast increase of cladding temperature.

5.3. Accident with closing of three check valves of the primary pump.

It has been assumed that in the initial state the reactor is operated at the rated power. By the beyond design basis accident scenario, closing of thee check valves takes place despite inhibitory interlocking in their control schemes preventing closing of more than one check valve. When reactor operates on three heat removal loops all the check valves are in the open state. In case of a valve stem break the valve is kept in the open state by the flow. Closing of the check valve is made only automatically by the "loop disconnection" signal under plant operation conditions on three loops. According to calculation, the probability of closing of three check valves of the primary pumps during time interval between the two nearest planned maintenance (3000 hours) is $2.3*10^{-7}$. The main contribution to this probability is made by a general-cause failure of control system elements. After closing of check valves and a decrease of sodium flow rate through the reactor the safety system operates. The reactor is brought to a subcritical state. However, sodium circulation though the primary pipe line – "the core - the upper mixing plenum - intermediate heat exchangers - primary pump - check valves -feeding header - core diagrid" - fully ceases.

In the reactor, complicated circulation sodium currents arise including the core, the mixing plenum and the upper part of intermediate heat exchangers. In some fuel subassemblies of the core downward motion of sodium takes place, in other ones-upward motion. In the process the upward and downward motion zones in the core are moving, sodium motion in subassemblies changing its direction. Sodium in the core boils up, boiling zones periodically displacing in accordance with the movement of downward and upward motion zones. The temperature of sodium reaches the saturation temperature from time to time (Fig.21). Due to decrease of decay heat release sodium boiling fully ceases in 400 second from the beginning of the accident.

In the course of this accident analysis, advantages of the approach taken as the basis for GRIF-SM code were demonstrated, i.e. that all the main components of the reactor, such as core, inlet and outlet plenums and intermediate heat exchanger are simulated within the framework of one calculation area. In this case the core decay heat is removed by the internal flow circuit formed to incorporate all the above mentioned components.

Uncontrolled closing of three check valves of the primary circuit.

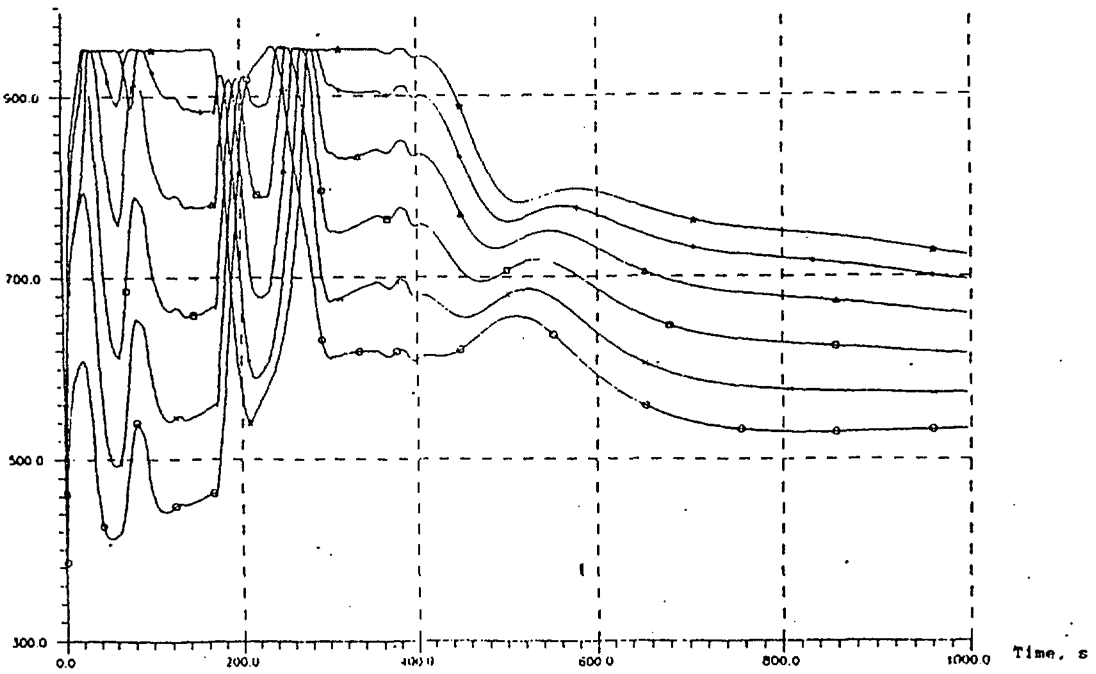


Fig.21. -sodium temperature in LEZ1 in various sections of the core height with a pitch of 0.2 H_{core}

BN-800 REACTOR
ULOF+UTOP ACCIDENT.

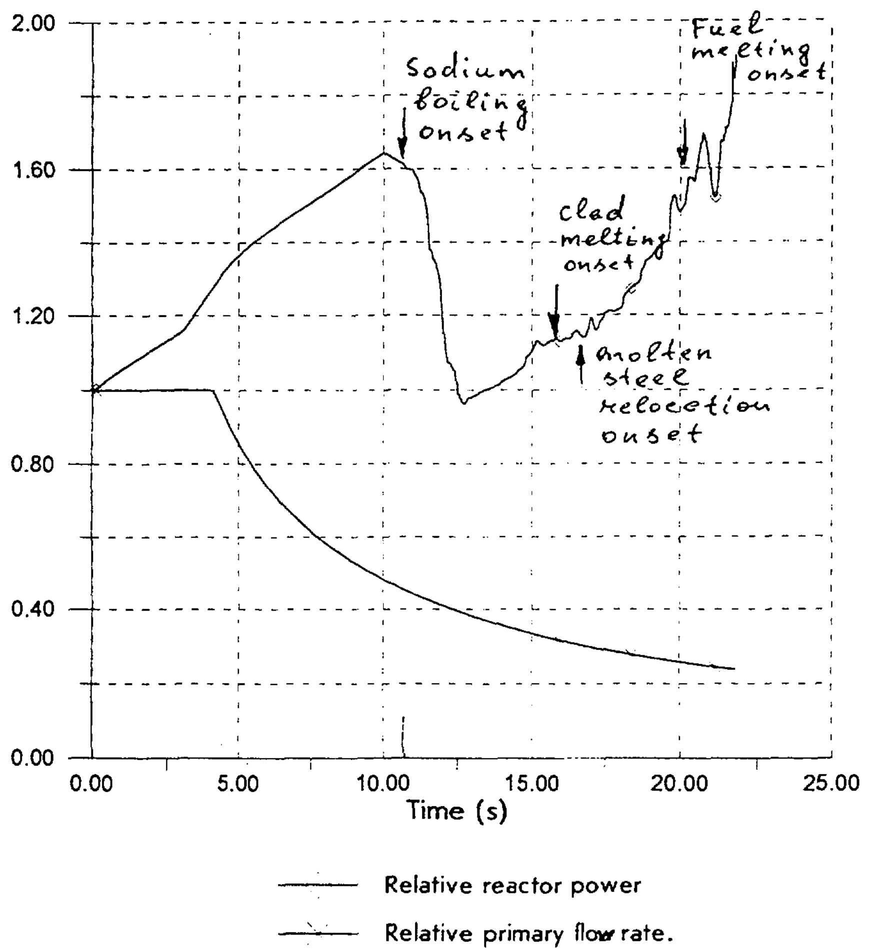


Fig.22.

5.4. UTOP + ULOF combined accident.

Incredible version of the UTOP accident with the primary pump speed decrease was considered. The accident starts with the unauthorized withdrawal from the core of one control rod followed by the withdrawal of 6 shim rods, all this being caused by several postulated failures. The analysis of the accident has been made using modified version of GRIF-SM code together with CANDLE module.

Sequential withdrawal of control rod and shim rods results in the power increase (Fig. 22). Decrease of the primary pump speed begins 4 seconds after the accident start. Sodium boiling onset on 11th second leads to the abrupt insert of negative reactivity (Fig. 23), however power level although decreased still remains sufficiently high, so 5 more seconds later melting of the fuel element cladding and molten steel relocation start. Melting takes

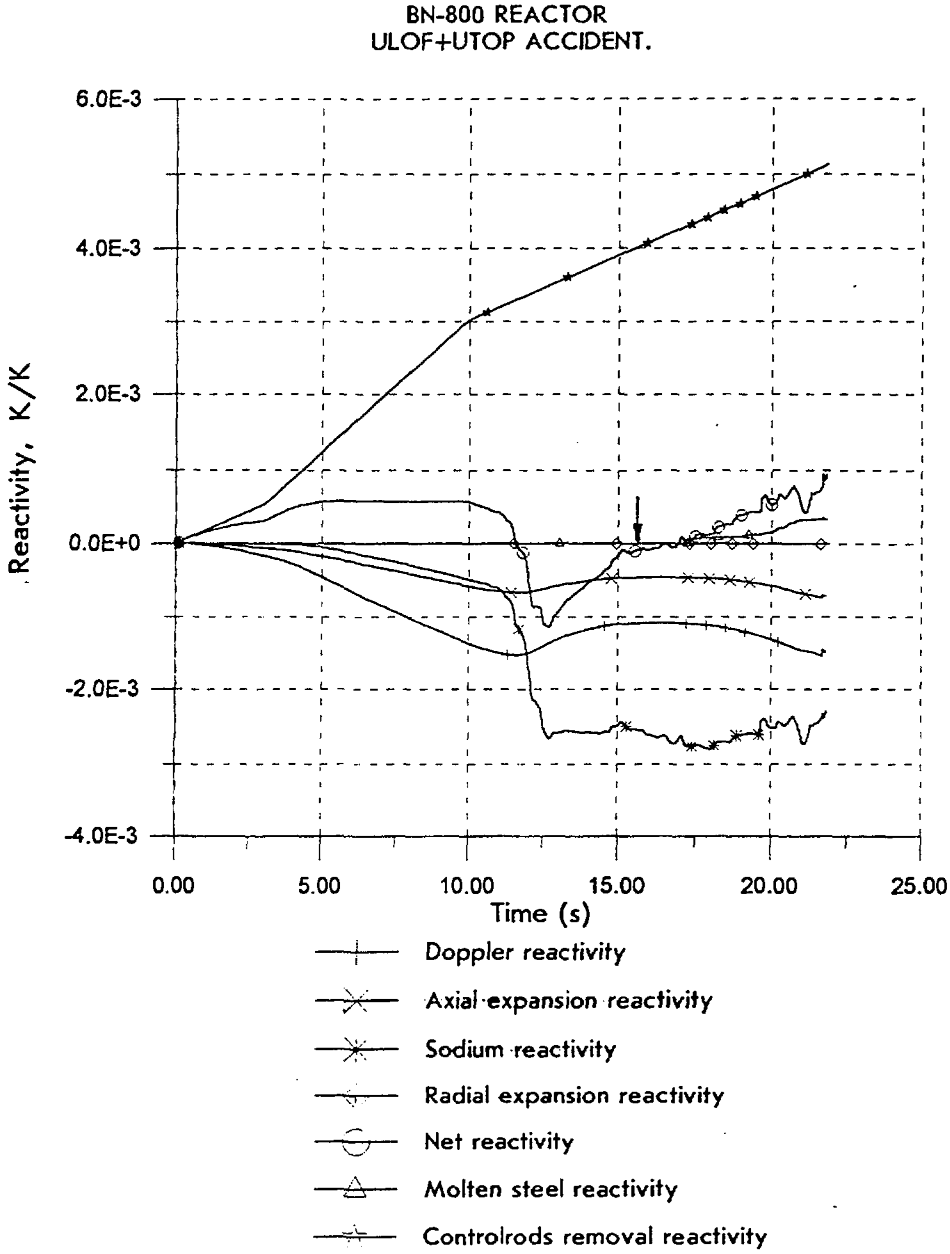


Fig.23. Reactivity versus time.

place under conditions when considerable sodium flow in the core is still maintained. This results in that sodium vapor flowing at high speed entrains the molten steel, which moves upwards as liquid film along the fuel elements, its solidification causing gradual plugging of the flow cross section area (Fig. 24). Steel relocation from the core central area to its upper boundary causes positive reactivity insertion and reactor power increase, thus resulting in the fuel melting.

BN- 800 REACTOR
ULOF+UTOP ACCIDENT.

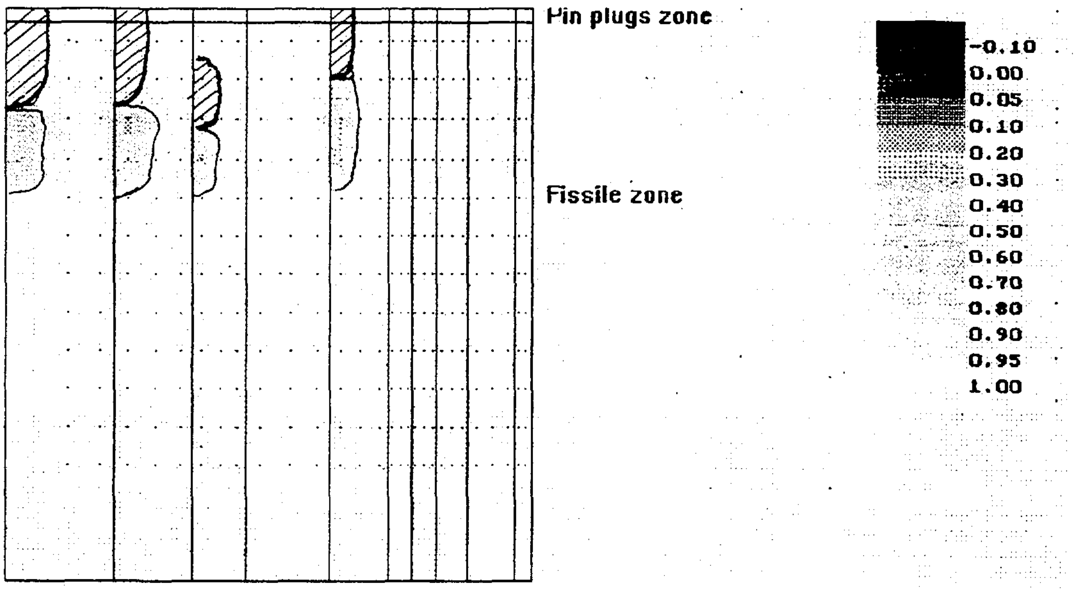


Fig. 24 Positions of "melted zones" and "removed steel zones in the core. Time=21s.

6. CONCLUDING REMARKS.

During the last eight years the program GRIF-SM is heavily used for the substantiation of nuclear safety of Russian fast reactors. For reactor BN-800 following accidents were studied: uncontrolled compensating rods removal; unprotected loss of flow; uncontrolled closing of primary reverse valves. Code was validated against experimental data on sodium boiling in electrically heated tubes and bundles. So it was shown, that GRIF-SM code is a universal computing tool capable to simulate emergency processes in fast reactor at the various initial conditions, when the sodium boiling is possible.

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