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Research Article

Coupled modeling of neutronics and thermal-hydraulics processes in LFR under SG-leakage condition

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Abstract

The lead cooled reactor BREST-OD-300 is developing as a part of Russian federal project "PRORYV". Two- circuit scheme is used in the reactor for heat removal. An inherent risk of two- circuit reactor is the potential danger of water steam ingression in the core in the case of large leakage in steam generator initiated, for example, Steam Generator Tube Rupture (SGTR).

Reactor power and temperature response on vapor penetration to the core is studied, but pressurization effects are not in the purview of the paper.

The 3D multi-physics (neutronics + thermal-hydraulics) UNICO-2F code was developed for study of SGTR accident. The code calculates unsteady 3D space dependent distributions of coolant velocity, pressure and temperature, space distributions of vapor concentration and heat release density in the core and 3D temperature distributions in the fuel pins.

Guillotine rupture of one tube in Steam Generator (SG) is considered as initial event of the accident.

It is shown that even with the most conservative assumptions reactivity insertion due to vapor ingress in the core causes small increase of power in level and as a result maximum cladding temperature continue to stay well below safe operation design limit in the entire transient.

Hypothetical option of simultaneous tube rupture in few SG belonging to different loops is also analyzed. It is demonstrated that even in the case of simultaneous large leak in two SG the transient stays mild and temperature in the core after two small oscillations is stabilized at acceptable level.

In the long term the analysis confirmed the high level of reactor self-protection against SGTR accident.

Keywords

Lead cooled fast reactors, Nuclear safety, Accident, Steam generator leakage, Multi-component flows simulation

Introduction

The lead cooled 700 MW_{th} BREST-OD-300 reactor design is under development in the Russian Federation within the framework of "PRORYV" project. Two circuits are used for heat removal from the reactor, without intermediate heat transport system. A consequence of the elimination of such system is the need to address the potential risk of steam ingress into the core in case of large leak in the Steam Generator (SG).

The main concern is caused by the following two aspects of the initial failure:

a. possible loss of integrity of the primary circuit caused by internal pressure increase;
b. possible positive reactivity insertion because of steam penetration into the central section of the reactor core resulting in its excessive temperature increase.

Until now, attention has been mainly paid to the first one of the above two problems. The review Gang 2017 presents information about the progress in the area of mathematical and experimental simulation of phenomena of water or steam injection into some volume or the circuit filled with heavy metal coolant. Numerical studies of consequences of SG tube double-ended guillotine rupture are described in Bubelisa et al. 2013 as applied to the medium size European Lead Fast Reactor (ELFR) Frogheri et al. 2013. Calculations made using SIMMER-III code Kondo et al. 1992 have shown that pressure increase in this accident on ELFR could result in structural damage of the steam generator, however, more likely, it would be no threat for reactor structure.

The results of numerical studies of behavior of parameters of small size (10 MW_{th}) reactor in case of SG tube double-ended guillotine rupture as an initial event are presented in Gu et al. 2015. Studies were made using new Chinese Neutronics and Thermal-hydraulics coupled Code (NTC) Gu et al. 2015. Pressure change in the area of the tube rupture is analyzed, as well as the process of steam transport in the primary circuit to the reactor core. It was shown that although steam reached the reactor core, its volume fraction was within 0.1%.

Detailed numerical studies of the process of steam propagation through the primary circuit of the medium-size European lead cooled reactor ELSY were made using CFD code Jeltsov et al. 2018. Bubbles movement was simulated within the framework of Lagrangian approach. The steam amount entering the reactor core were evaluated for various leakage locations, steam bubble sizes and amounts of impurities in the coolant.

However, the overall goal of the reactor safety analysis is justification for the fact that under this accident conditions no safe operation limits specified for the reactor are exceeded, including max permissible temperature of the fuel element cladding. In order to reach this goal, the following procedures should be performed:

a. Accurate modelling of neutronics and thermal-hydraulics taking into account their coupling, and
b. 3D modelling of accident considering significant non-symmetry of spatial distribution of concentration of steam entering the reactor core and strong dependence of void reactivity effect on radius.

The 3D multi-physics (neutronics + thermal-hydraulics) UNICO-2F 3D code was developed for studying SGTR accident. The code is capable of calculating transient 3D spatial distributions of coolant velocity, pressure and temperature, steam concentration in the coolant and power density in the core, as well as fuel and cladding temperatures distribution over the core.

Nomenclature

C	specific heat capacity (J/kg/°K)
C_ρ	average volumetric heat capacity (J/(m^{3}K)
$\vec{G}$	average mass flux of mixture (kg/(m^{2}s)
G_v	mass flux of steam (kg/(m^{2}s)
G_l	mass flux of lead (kg/(m^{2}s)
I	node number in X direction (–)
J	node number in Y direction (–)
K	node number in Z direction (–)
P	mixture pressure (Pa)
q	heat release (W/m^3)
r_b	bubble radius (m)
T	mixture temperature (°C)
U	physical velocity of components (steam and liquid) (m/s)
u_b	rise rate of the bubble in stationary liquid lead (m/s)
X, Y, Z	Cartesian coordinates (m)

Greek letters

e	porosity (volume fraction occupied by lead and steam mixture), steam and lead volume ratio in mixture (–)
ρ	average density of mixture (kg/m^3)
σ	surface tension coefficient for lead (N/m^2)
$\widehat{\Lambda}$	tensor of friction coefficients (1/s)
$\widehat{\Psi}, \widehat{\Psi}_E$	tensors that are taken into account slipping of steam component in the mixture (–)

Subscripts:

b	bubble
l	lead (liquid)
v	steam
max	maximum

Widely used in the text and list of references

BREST-OD-300	Russian Lead Fast Reactor
CFD	Computational Fluid Dynamics
ELFR	European Lead Fast Reactor
LMR	Liquid Metal Reactor
MPPC	Main Pump Pressure Chamber
NTC	Neutronics and Thermal-hydraulics coupled Code
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIMMER-III	Two-dimensional, multiphase code, multicomponent, fluid-dynamics code
UNICO-2F	Russian code for analysis of transient behavior of power and temperature distributions in the LMR core in the case of SGTR
3D	three-dimensional

Some features of BREST-OD-300 reactor design

Heat is removed from BREST-OD-300 reactor core (Fig. 1) by four independent loops with two steam generators connected in parallel in each loop.

In case of SG leakage, steam from damaged tube first enters SG shell side, and then the generated bubbles carried away by the coolant flow are moving along the path “main pump suction chamber – main pump – Main Pump Pressure Chamber (MPPC) – inlet ring duct – reactor downcomer section – core diagrid – core”. Steam bubbles have an opportunity to separate to the reactor gas system moving to the core through the primary circuit parts that have free levels of the coolant. These are gas cavities in SG, MPPC and reactor downcomer section.

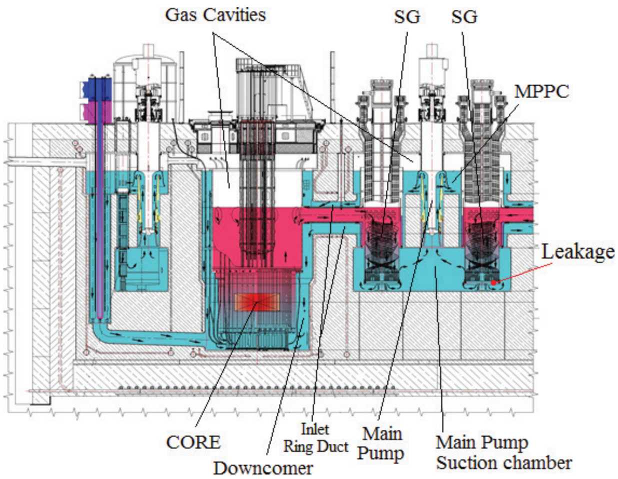


Figure 1. Lead flow in the primary circuit and decay heat removal system circuit.

Steam fraction supplied to the core inlet depends on several factors and, primarily, on the steam bubble size, since this parameter governs the effectiveness of bubbles separation into gas plenum of the reactor on their way to the reactor core. The initial bubble size is determined by the conditions in the rupture point, in particular, by the rupture configuration and actual pressure difference between the primary and the secondary circuits. Moreover, steam bubbles formed after SG tube rupture have different sizes, and bubble size spectrum depends on the outflow conditions. SG tube double-ended guillotine rupture is usually taken as an initial event for safety analysis.

In this case, steam flows out from two ends of ruptured tube and total flow rate of steam entering the primary circuit are determined by hydraulic resistance of the ruptured tube sections and the secondary circuit pressure at the inlet and outlet of the steam generator. The most conservative estimates made for the steam generator of BREST-OD-300 reactor plant give steam flow rate $G_v = 0.73$ kg/s.

One more limitation of steam flow rate is related to limited conveyance capacity of MPPC due to inevitable and drastic bubble coalescence after reaching critical value of steam fraction. This critical value corresponds to packing factor of the balls when the surfaces of adjacent balls will inevitably contact to each other. Maximum packing factor 0.74046 (Gausse) corresponds to the case when the balls are placed in the angles of triangular pyramid. However, the realization of such regular and dense structure in two-phase flow looks extremely improbable. More realistic is the flow structure when the bubbles are positioned in the angles of cube. The maximum packing factor for the case is 0.52. As evaluation shows the conveyance capacity of MPPC of BREST-OD-300 reactor is limited by value – 0.58кg/c

The possibility of simultaneous failure of SG tubes in more than one loop can be considered theoretically, however, it’s clear that in view of extremely low probability of such initial event it should be treated as hypothetical.

It should be noted that the least favorable consequences would be expected not as a result of the guillotine tube rupture but in case of appearance of the long narrow crack in the tube wall, since in the latter case there is high probability of formation of the small size bubbles, which can be hardly separated. In the course of their movement the bubbles can break down or, vice versa, merge forming larger bubbles depending on the flow mode and separate into the reactor gas plena.

Analytically one can assume the possibility of chain mechanism causing rupture of many SG tubes, when a tube adjacent to that already failed is broken by the jet coming from the breach. However, it should be taken into account that impossibility of a single tube rupture escalation to multiple ruptures of the tube bundle has been confirmed experimentally for conditions of BREST-OD-300 steam generator Abramov et al. 2014.

The primary circuit of BREST-OD-300 reactor was intentionally designed to assure to max extent the separation of the steam into the cover gas reactor system though gas-lewd interface, namely the cavities with free lead levels are arranged. Separation goes sequentially in the following three elements: steam generator itself, main pump pressure chamber and reactor downcomer.

While moving in liquid lead, steam bubbles would either break down into smaller bubbles or, vice versa, coalesce into larger bubbles, depending on flow characteristics, ultimately reaching the reactor gas plenum.

Many publications have been devoted to determination of raising velocity of bubbles in liquid. In particular, relationships worked out by Peables and Garber are considered the most universal Yollis 1972. These relationships are also recommended for studying cases with significant density difference between liquid and gas. Similar relationships obtained on the basis of criterial analysis were proposed by Berdnikov (Fig. 2) (Berdnikov and Levin 1977).

The primary circuit of BREST-OD-300 reactor was intentionally designed to assure to max extent separation of the steam into the cover gas though free surfaces while it passes through MPPC.

It follows from presented data that rise rate rapidly decreases with the decrease of bubble size and it becomes low with respect to typical flow velocity in the primary circuit elements (1 – 2 m/s) already for the bubbles of less

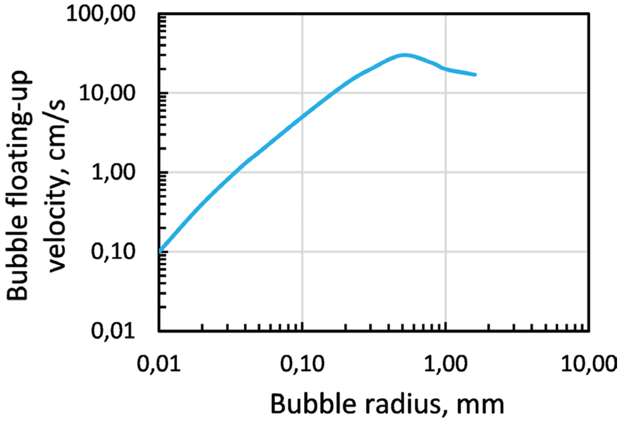


Figure 2. Gas bubble raising velocity u_b versus bubble radius r_b in liquid lead and liquid steel (according to Berdnikov and Levin 1977).

than 10^{-4} m diameter. Hence it appears that in conservative analysis rise rate should be neglected, as well as the possibility of steam separation in the course of its transport from tube rupture point to the reactor core.

UNICO-2F code and numerical model

The in-house UNICO-2F was developed for analysis of transient behavior of power and temperature distributions in the LMR core. Its structure is shown on Fig. 3.

The code consists of three main modules, their functions being described below.

SVIR – calculation of velocity, pressure and temperature of coolant within computational domain, which, in this case, includes reactor downcomer section and reactor core, and temperature of structural elements, fuel and cladding (heat and mass transfer equations set is solved to approximation of the model of viscous ideal liquid flowing in porous body in Cartesian or cylindrical coordinate system; there is a model of thermal conductivity of multilayer cylindrical element bounded by the coolant flow, which can be used for evaluation of 3D temperature distribution in the fuel elements).

DENS_V – calculation of steam concentration within computational domain.

MAGTDP – calculation of reactor core power profile taking into account steam concentration variable (MAG neutronics module was designed on the basis of similarly-named code Suslov and Babanakov 1996, which was meant to solve steady state and transient neutron transport equations within the framework of diffusion approximation using parameterized constants method).

When formulating two-phase thermal hydraulics equations set and related two-phase flow pattern diagram the following simplifying assumptions are used:

- Only bubble flow pattern is simulated;
- The possibility of bubbles fragmentation and merging is not taken into account;
- It is assumed that steam and lead are in thermally-equilibrium state;
- Steam properties are described by perfect gas law;
- Compression work contribution to the energy balance is neglected;
- Steam phase momentum equation is represented in simplified form on the basis of correlation for bubbles rise rate in the resting liquid lead (in particular, possible effect of added mass on dynamics is not taken into account);

It is generally clear that such an approach is applicable, first, to relatively slow processes and, second, when steam volume fraction in lead is still not too big.

Taking into account the above listed simplifications the following equation set describing movement of liquid metal and steam phases mixture in nuclear reactor in porous body approximation has been stated:

Mass balance equation for mixture:

$$\frac{\partial \rho}{\partial \tau}+\frac{1}{\varepsilon}(\vec{\nabla}\varepsilon\vec{G})=0 \qquad (1)$$

Momentum equation for mixture:

$$\frac{\partial \vec{G}}{\partial \tau}+\frac{1}{\varepsilon}\left(\vec{\nabla}\widehat{\Psi}\frac{\varepsilon}{\rho}\vec{G}\right)\vec{G}=-\vec{\nabla}P-\widehat{\Lambda}\vec{G}+\vec{F} \qquad (2)$$

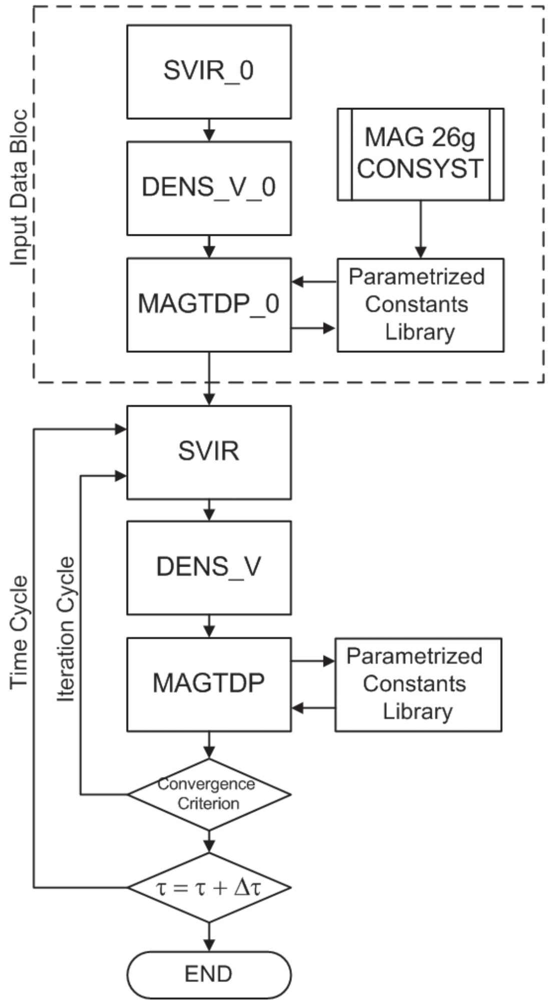


Figure 3. General structure of UNICO-2F code.

Energy equation for mixture:

$$\frac{\partial(C_pT)}{\partial\tau}+\frac{1}{\varepsilon}\left(\vec{\nabla}\widehat{\Psi}_E C_p\frac{\varepsilon}{\rho}\vec{G}\right)T=q \quad (3)$$

Steam mass balance equation:

$$\frac{\partial\rho_v\varepsilon_v}{\partial\tau}+\frac{1}{\varepsilon}\vec{\nabla}(\varepsilon\varepsilon_v\rho_v\vec{U}_v)=0 \quad (4)$$

Simplified momentum equation for steam:

$$\vec{U}_v=\vec{U}_l+u_b(r_b)\frac{\vec{\nabla}P}{\left|\vec{\nabla}P_0\right|}, \quad (5)$$

where P_0 is pressure distribution in the reactor for normal conditions (without steam injection).

Equation of volume fraction balance:

$$\varepsilon_v+\varepsilon_l=1 \quad (6)$$

Equation of state

for lead: $\rho_l=\rho_l(T)$, $C_l = const$
for steam: $\rho_v=\rho_v(T,P_b)$, $C_v = const$

where pressure in the bubble P_b is composed of circumjacent lead pressure P and addition caused by surface tension effect $P_b = P + 2\sigma/r_b$

In the equations, average density ρ, average volumetric heat capacity $C\rho$ and average mass velocity $\vec{G}$ of mixture are calculated by the following relationships:

$$\rho=\rho_v\varepsilon_v+\rho_l\varepsilon_l \quad (7)$$

$$\vec{G}=\rho_v\varepsilon_v\vec{U}_v+\rho_l\varepsilon_l\vec{U}_l \quad (8)$$

$$C\rho=C_v\rho_v\varepsilon_v+C_l\rho_l\varepsilon_l \quad (9)$$

Heat conductivity equation is solved for calculation of 3D fuel and cladding temperature in the pins. Heat transfer coefficient on clad outer surface is calculating taking into account deterioration of heat transfer due to decrease of the heat capacity.

One-phase 3D thermal-hydraulics module of the code UNICO-2F named GRIF was validated against a few in-pile reactor tests including analyses of Sodium Natural Convection in the Upper Plenum of the MONJU Reactor Vessel Ohira et al. 2013 and international benchmark on the natural convection test in Phenix reactor Tenchine et al. 2013.

Computational domain simulating the central part of BREST-OD-300 reactor (Fig. 4) is covered by non-uniform Cartesian difference mesh having 39×75×58 nodes in X, Z, and Y (I, K, and J) directions and has the size 6,0×9,5×6,0 m correspondingly. The domain includes in itself: downcomer, inlet header, core and upper plenum.

Conditions at the inlet of the reactor downcomer section are used as input data. These include flow rate and temperature of lead and steam, as well as steam bubble size (it is assumed that all incoming bubbles have the same given size, which then changes in the process of their movement depending on external temperature and pressure).

Reactor parameters behaviour in case of SG large leakage

Scenario reference option

The following assumption were put into the base of the reference scenario:

1. Double-side guillotine rupture of one SG tube is assumed as the initial event of accident.
2. Failure occurs when reactor is operating on rated power.

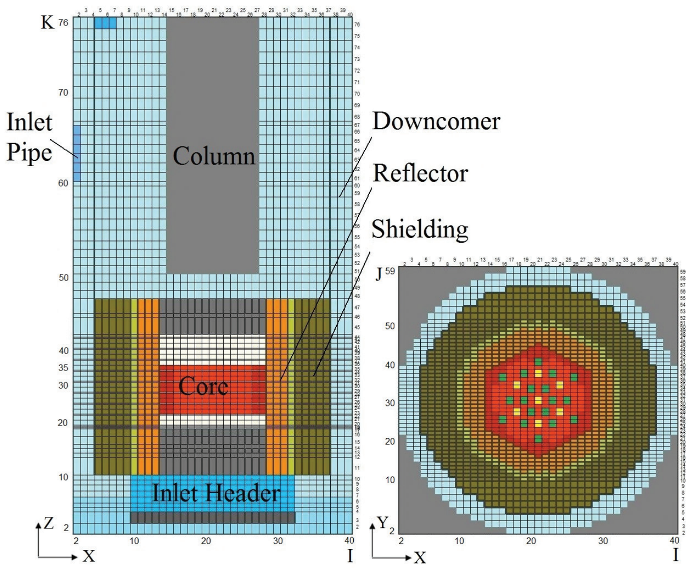


Figure 4. Vertical (**a**) and horizontal (**b**) sections of computational domain.

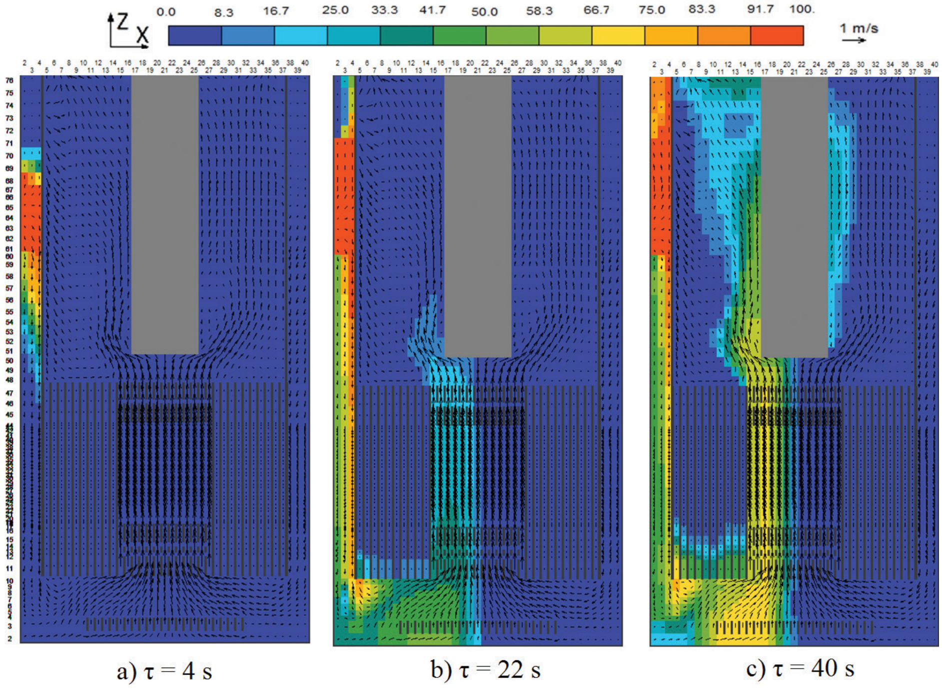


Figure 5. Distribution of steam velocity and relative concentration in section J = 22.

3. Total failure of safety system takes place (control rods position does not change. and the main pumps are still in operation) and power behavior is only determined by temperature and void reactivity effects.
4. The flow rate of steam injecting through the rupture remains constant over time and equal to 0.73 kg/s. (The decrease of steam flow rate through the leakage is expected due to pressure build-up in the zone of steam injection at the beginning of the transient and due to secondary pressure decrease when SG isolation is anticipated to start transient termination. Both those effects are conservatively neglected.).
5. The tube rupture is located at the most unfavorable position - at the bottom of SG tube bundle and therefore it is assumed that whole injected steam is entrained by lead flow to the main pump. (The separation along this path is neglected).
6. Due to steam separation in MPPC the steam flow rate entering reactor downcomer section becomes equal to 0.58 kg/s. Steam is only supplied to downcomer through the nozzles of failed loop (two nozzles out of eight in case of one failed loop).
7. Diameter of incoming steam bubbles was varied during the parametric study in the range 0.00001–0.001 m.

All those assumptions are conservative.

Reactor transient for reference scenario

Results of analysis of steam velocity and concentration patterns behavior for the reference option are shown in Fig. 5. Steam concentration distribution is presented in relative units, i.e., steam concentration in the inlet nozzle is taken equal to 100%.

There are two specific stages of the process of steam propagation through the reactor. The first stage starts when the steam enters downcomer section of the reactor. Then steam-lead mixture "plume" moves downward in the downcomer section and on 22-nd second the steam enters the left part of the core. This stage is characterized by strongly pronounced azimuthal non-symmetry of steam distribution in the downcomer and in the core. As a result, the first sudden change of steam content occurs in the core (Fig. 6) resulting in the change of reactor power and temperature (Fig. 7).

In the second stage (after first 50 seconds of transient), the downcomer is gradually filled with steam-lead mixture, because it also starts floating up and fills in steps the whole downcomer volume. Steam accumulated in the downcomer section is taken up by the lead flowing from the other (intact) loops and brought to the reactor core. From this moment (approximately after 40-th second) the

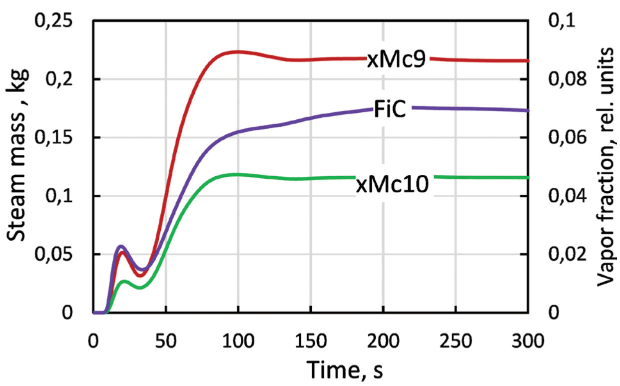


Figure 6. Steam mass (xMc9) and max local vapor fraction (FiC) in central section of the core and steam mass (xMc10) in peripheral section.

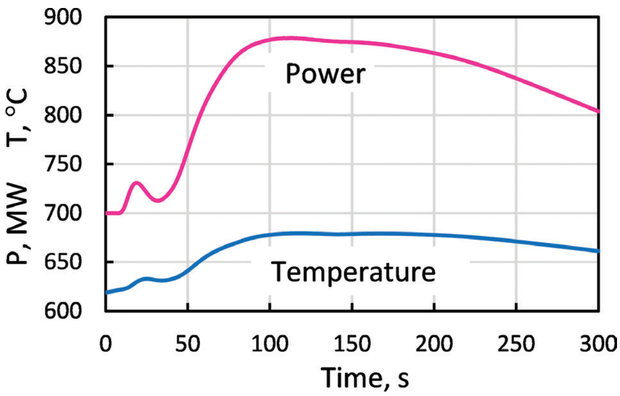


Figure 7. Behavior of reactor power and max cladding temperature.

amount of steam entering the core starts increasing again, and now pattern of steam supply to the core is azimuthally symmetric. The second stage is finished by steady state onset (Figs 6, 7), when the amount of steam entering reactor through the inlet nozzles becomes equal to that leaving the reactor.

Max reactor power achieved during the transient exceeds rated value by 25%, and max fuel element cladding temperature is 680 °C, this being significantly lower than the corresponding safe operation limit Moreover, safe operation limit in terms of max fuel element cladding temperature (it is allowed to expose cladding to 900 °C no longer than for 10 minutes) is not reached.

On the effect of steam bubble size

Flow pattern in the downcomer section and effectiveness of steam separation to the gas plenum of the reactor depends on the size of bubbles. Large bubbles are floating up immediately after their arrival from the inlet nozzles forming upward flow plume, and entering reactor gas plenum. Small bubbles are brought downward by the flow and they reach the reactor core mostly in its periphery. Ratio of numbers of separated steam bubbles and those brought to the reactor core strongly depends on the bubble size (Fig. 8).

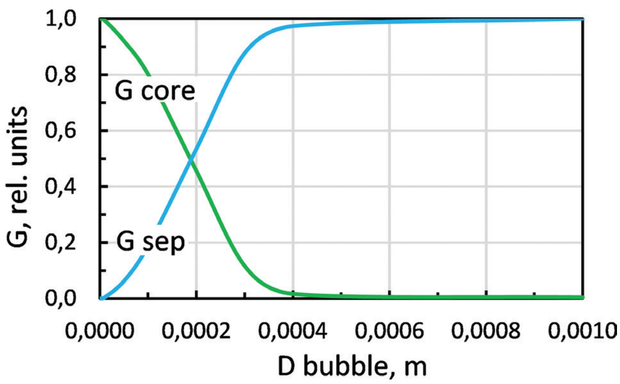


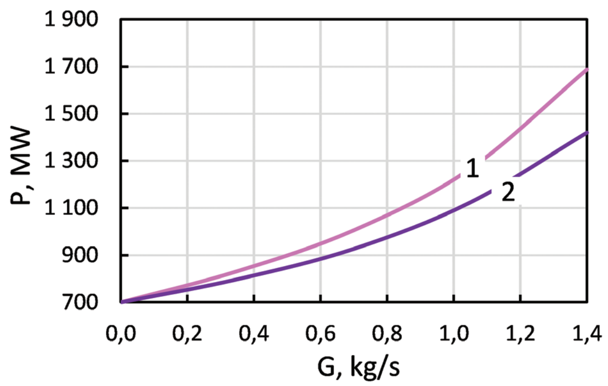
Figure 8. Relative flow rates of steam separated in the downcomer section (Gsep) and that entering the reactor core (Gcore) as a function of steam bubble diameter.

On the Effect of Hydrogen Contribution

Effect of vapor penetration to the reactor core has two main components:

- void reactivity effect caused by lead displacement. This effect is positive for central zones of the core, and it is positive in the core of BREST-OD-300 reactor in case of lead uniform displacement from the whole core;
- reactivity effect related to the presence of hydrogen. An increased capture on hydrogen and decreased number of neutrons per fission due to moderation provide negative hydrogen effect on reactivity as direct calculations using both Monte-Carlo and diffusion theory show. This effect is negative and its magnitude increases with the increase of hydrogen content.

It can be seen from comparison of curves in Fig. 9 that positive void reactivity effect caused by lead displacement is partly compensated by the negative effect from hydrogen presence, and it is hydrogen presence that slows down power build-up rate approximately on 30%. Coolant temperature heat-up also decreases, as well as max value of the fuel element cladding temperature.



1 – spectrum degradation isn't taken into account
2 – all reactivity components are taken into account

Figure 9. Max power value achieved during the transient versus steam flow rate at the core inlet and depending on reactivity effect components taken into account.

Hypothetic Accident: Large Leaks Occurring Simultaneously in Several SG

It is clear that the probability of large leakages occurring simultaneously in two or more loops is extremely low, and common cause failure is excluded by the independence of each primary loop. Nevertheless, for the sake of deep insight into the accidental processes and reactor self-protection margins numerical studies were carried out on the accident with simultaneous SG tube double-ended guillotine ruptures in two or more loops as an initial event. It is also conservatively supposed that all steam bubbles generated in the leakage location are transported to the core inlet without separation.

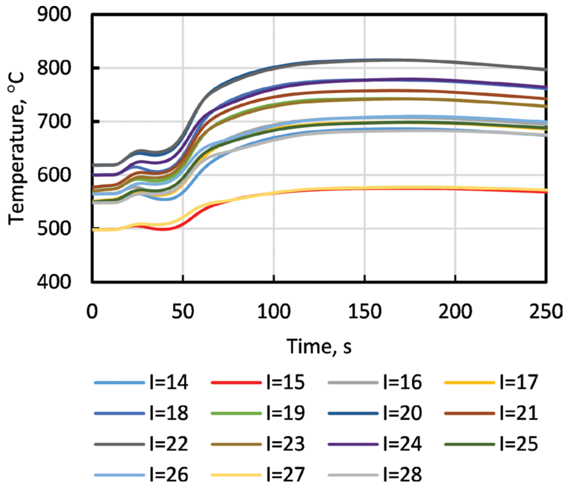


Figure 10. Behavior of fuel element cladding temperature in some fuel assemblies in the outlet core cross section (K = 35, J = 29), Gv = 0.73*2 kg/s.

It has been found that in case of a sudden emergence of large leaks simultaneously in two adjacent SGs transient nature is the same and, as usual, after two fluctuations core power and temperature become stable (Fig. 10).

Moreover, safe operation limit in terms of max fuel element cladding temperature (to expose cladding to 900 °C no longer than for 10 minutes) is not exceeded (Table 1).

Table 1. Max values of parameters depending on number of failed loops

Number of failed loops	1	2
Steam flow rate, G*v* kg/s	0.73	1.46
Steam mass xMc9, kg	0.26	0.52
Steam mass xMc10, kg	0.14	0.28
Max power, MW_{th}	927	1415
Max cladding temperature, °C	680	815

Conclusions

In the context of analysis of accident caused by large leak in the steam generator in two-circuit lead cooled reactor plant, it has been demonstrated that 2F 3D UNICO-2F code is capable of modeling core thermo-hydraulics and neutronics taking into account their mutual influence.

Numerical studies have been performed showing high level of self-protection of BREST-OD-300 reactor in case of SG leak, even with the most conservative assumptions (negligibly small gas bubble raising velocity due to small specified diameter of incoming steam bubbles, neglecting of possible decrease of vapor flow rate through the breakup during the transient, total failure of safety system). Max cladding temperature in the entire transient is well below safe operation design limit, and transient is ended with safe and stable reactor condition. Moreover, even in hypothetic accident with simultaneous tube ruptures in two steam generators in two different loops safety limit on clad temperature is not exceeded and therefore and core saves its integrity.

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