ISTC Project No. <3635>

< Scale experimental investigation of the thermal and structural integrity of the VVER pressure vessel Lower Head in severe accident >

Annual Project Technical Report

on the work performed from <Sept 01, 2008> to <Sept 01, 2009>

< State educational establishment of the higher education RF "Moscow Power Engineering Institute" (Technical university) - MPEI>

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Title of the Project:	" Scale experimental investigation of the thermal and structural integrity of the VVER pressure vessel Lower Head in severe accident"
Contracting Institute:	State educational establishment of the higher education RF "Moscow Power Engineering Institute" (Technical university) – MPEI
Participating Institutes:	- Federal State Unitary Enterprise "Experimental and Design Organization "GIDROPRESS"; - Lavrentyev Institute of Hydrodynamics of the Siberian Branch of the Russian Academy of Sciences
Commencement Date:	September 01, 2007
Duration:	36 months
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1. Brief description of the work plan: objective, expected results, technical approach

The objective of this project is the experimental and numerical study of the VVER LH reactor vessel scale models within transient thermal and its overpressure loading which correspond to realistic SA scenarios accompanied by the high-temperature heating, creep deformation of the reactor vessel.

In this context the project efforts are focused on the **following problems:**

1) the designing and construction of the test facility for test examinations of the VVER vessel scale models (up to ~1:5) on the conditions which correspond to SA in VVER. Experimental facility to be built includes: working space, scale model and its heater, control and experimental information gathering system (DAS), support systems (gas, water systems, video monitoring devices etc.);

2) manufacturing of the VVER LH reactor vessel scale models. Material and technology, as well as thermal treatment have to correspond the same conditions of the regular VVER vessels manufacturing;

3) pursuance of the material creep test experiments with samples from the VVER vessel steel on the time range 2-50 hours and temperature range 600 - 1200;

4) the carrying out the scale experiments with VVER vessel models on the high-temperature heat-up and creep deformation of the vessel;

5) the mathematical treatment and analysis of scale experiments, carrying out the numerical pre- and post-test structural analyses of scale experiments with vessel models by means of numerical codes for validation of the mathematical models implemented in these codes.

Expected results

In the course of fulfillment of the given project it is supposed to receive the following results:

1) experimental data on the creep behaviour, heat up and failure of the VVER-440 vessel scale models;

2) the material property creep test results of the 15Kh2NMFA vessel steel at the range of 600 to 1200 C and failure times from 2 to 50 hours.

Technical Approach and Methodology

1) **Task "À"**. The numerical simulations associated with choosing the accident scenarios and test conditions of the VVER-440 scale vessel models, will be carried out by means of the numerical codes. Pre-test simulations of the vessel models behaviour during scale experiments for the chosen SA scenarios will be carried out by means of the various codes at the project participants disposal. The mechanical characteristics of the vessel steel updated in view of the results received within the material property tests in the Task "D" will be taken into account in post-test simulations;

2) **Task "B"**. Development and manufacturing of the experimental test facility and supporting systems for the VVER-440 scale vessel models testing are associated with decision of a number of the constructional and technological problems. The heater construction thus will work on extreme temperature conditions (more than 1500 C) during the course of a few tens hours. Because of this, the serviceability preservation of the heater will be provided by using of the special materials (ceramics etc) and refractory alloys.

The separate attention is deserved with development and manufacturing of the VVER vessel scale models. It is supposed, that the vessel scale models will be made of the original VVER steel. Material and technology, as well as thermal treatment have to correspond the same conditions of the regular VVER vessels manufacturing. The inner diameter and wall thickness of the VVER-440 vessel scale model will be respectively ~750 mm and ~30 mm corresponding to a geometrical scale ~1:5.

For realization of the control by test facility and data gathering during the testing it is necessary to develop the control and data acquisition (DAS) systems. It is planned to use not less than 4 channels for control by the electrical, gas and water systems, and not less than 50 channels - for experimental data gathering (temperature, vessel displacements etc) during testing;

3) **Task "C"**. The study of the VVER-440 vessel behaviour in SA conditions through the experimental and numerical investigation of the vessel scale models behaviour (the thermal and structural analyses) within transient thermal and overpressure loads. The various heating

conditions, structure loading and cooling conditions of the model will be simulated in each of these tests. It is planned, the outer surface of the model will be cooled by air only.

It is planned, three scale experiments will be carried out within this experimental program. The peaked heat flux distribution on an internal surface of the scale model will be reproduced in each of these tests. Such heat flux profile corresponds to realistic SA scenarios accompanied by the formation of a stratified melt pool in the lower plenum of the VVER-440 reactor pressure vessel.

The following cases of the hot focus location on an inner surface of the vessel wall will be investigated:

- the heat flux peak is located on the inner surface of the cylindrical vessel wall and is above (~50 mm) the welded seam (joining the elliptical lower head and a cylindrical part of the vessel model);

- the maximum of the peaked flux is located on the elliptical bottom of the scale model. The polar angle of the heat flux peak lies in the range between 75 and 85^0. The bottom center of the lower head is 0^0.

The local thickness of the vessel wall will be measured before and after tests at a number of points on the outer vessel surface (at least the 30 locations) that will allow to synthesize a “thickness map” of the vessel wall.

More than the 30 thermocouples on the outer and more than 6 thermocouples on the inner vessel wall surfaces will be used to measure the wall temperature during testing. The monitoring of the vessel model deformation in course of the testing will be carried out by linear displacement transducers. It is planned the measurement of the vessel displacements will be carried out at the 10 locations. There are two displacement gauges at each location (one for horizontal displacement and one for vertical displacement).

The experimental data received as a result of the test fulfillment, will be used for carrying out the post-test simulations. The steel bars will be cut from the vessel model after testing for further manufacturing of the samples intended for the material property creep tests.

4) **Task "D"**. The considered project stage is concerned with determination of the creep characteristics and ordinary mechanical properties of the VVER vessel steel (15Kh2NMFA) in temperature range from 600 to 1200 C on the failure times from 2 to 50 hours.

The appropriate tests for determination of the ordinary mechanical properties of the VVER vessel steel will be carried out with tensile rate ~20 mm/min to avoid relaxation due to the creep during testing.

Within the framework of present project it is supposed to carry out the material property tests of this steel for two cases of the sample fabrication: the test specimens fabricated of the original VVER vessel steel (case #1) and manufactured of the vessel models after testing (case #2). The results of the 2-nd case tests will allow to estimate the influence of the prior heating and accumulative damage (as a result of the creep of the vessel model during the test) on the creep strength of the VVER vessel steel.

2. Technical progress during the 2-nd year

1) Task "A": The numerical calculations concerned with choosing of accident scenarios and test conditions of scale vessel models.

Terms of performance under the planned schedule: 1-7 quarters.

The following work has been carried out:

a) SA scenarios have been chosen;
b) pre-test calculations for the chosen SA scenarios by means of code SOKRAT have been carried out;
c) numerical simulation of thermal condition of the melt and the VVER-440 reactor vessel lower head by means of code NARAL have been carried out;

The work carried out within the 2-nd year corresponds to the planned work of the Project Working Plan.

Scope of carried out work: 100 %.

The main results received at the solving of tasks of this stage are presented in Appendix 1-1 (Sub-tasks “a” ..“d”) and Appendix 1-1; 2-1; 2-A-2.

2) Task "B"

Performance terms under the planned schedule: 1-7 quarters.

The following work on subtasks has been carried out:

Subtask B.1:
- Assembling of protective housing of experimental test facility has been finished. Assembling of auxiliary electric and water systems is carried out;

Scope of carried out work: 100 %;

Subtask B.2:
- a vessel model (1 piece) is at the stage of manufacturing.

Scope of carried out work: 75 %;

Subtask B.3:
- the heater of the vessel model has been made.

Scope of carried out work: 100 %;

Subtask B.4:
- DAS system of experimental test facility has been made.

Scope of carried out work: 100 %.

Subtask B.5
-Re-planning and repair of an experimental section and the administrative areas.

State:
Because of insufficiency of financial means, the means planned earlier for performance of this subtask were directed on solving of subtasks “A”-“G”.

Scope of carried out work: 8 %.

Activities on Task “B”, carried out within the 2-nd year, correspond as a whole to the planned work of the Project Working Plan.

The main problem now is manufacturing of the vessel models. Owing to developed situation with the prices for metal by the present moment, the means planned for manufacturing of 4 reactor vessel models, are not sufficient even for manufacturing of the 1st model. For solving of this problem efforts on search of missing financial assets are made both in Russia and abroad. In connection with such situation the delay in manufacturing of the vessel models is possible. The delay concerned with manufacturing of the vessel model may be from 4 till 7 months.

In case it will not be possible to find necessary financial assets for manufacturing of 3 models, number of tests (3 tests according to the Working Plan) will be reduced to 1-2 tests.

The main results received at the solving of tasks of this stage are presented in Appendix 3-1 ..Appendix 3-4.

Task 3 (“C”): “Carrying out of scale experiments on the VVER vessel models”

Work in this direction was not carried out, since terms of its performance under the planned schedule are 8-10 quarters.

Task 4 (“D”): “Determination of creep characteristics and short-term mechanical characteristics of VVER vessel steel (15Kh2NMFA) at temperature 600-1200 C on the time base of 2-50 hours”

Terms of carrying out under the planned schedule: 1-12 quarters.

Status of work:

Subtask D.1: “Manufacturing of specimens of VVER vessel steel”

The following work has been carried out in the given direction:
- specimens of the 3rd series have been fabricated.

Scope of carried out work: 100 %.

Subtask D.2: “Reports of high-temperature creep tests of VVER vessel steel specimens”

The following work has been carried out in the given direction:

- creep and short-term stretching tests of the VVER vessel steel specimens up to temperature 1050 C are carried out.

Scope of carried out work: 80 %.

The work carried out within the 2-nd year corresponds to the planned work of the Project Working Plan.

The main results received at the solving of tasks of this stage are presented in Appendix 4.

3. Technical progress during the year of reference

- compliance with tasks and milestones as described in the work plan
- achievements of the past year

4. Current technical status

- on schedule, behind, ahead
- refining next year schedule if necessary
- recommendation for changes of the work plan, if necessary

The work under the project is carried out according to the schedule of the Working plan. It is possible to assume a delay connected with manufacturing of the vessel models that will lead to delay of the vessel models tests from 4 till 7 months. Besides, in case of impossibility of manufacturing of planned number of models (4 pieces), the number of the models tests may be reduced from 3 to 1 or 2 tests.

5. Cooperation with foreign collaborators

- exchange of scientific material (information, computer codes and data, samples)
- signature of protocols (with short description)
- research carried out jointly
- trips to/from foreign collaborators
- workshops, topical meetings organized by the project team
- joint attendance to international conferences

For the reporting period joint discussions of the project tasks with foreign collaborators were carried out during thematic meetings:

1) 15th SEG SAM meeting, PSI, March, 2009
2) 16th SEG SAM meeting, IBRAE, Moscow, September, 2009.

6. Problems encountered and suggestions to remedy

The main problem is insufficiency of financial assets for manufacturing of the vessel models (4 pieces). Now manufacturing of only one vessel model is at the stage of manufacturing.

Search of additional sources of financing for manufacturing of vessel models is carried out both in Russia and abroad.

7. Perspectives of future developments of the research/technology developed

Attachment 1:	Illustrations attached to the main text (if)
Attachment 2:	Other Information, supplements to the main text
Attachment 3:	Abstracts of papers and reports published during the year of reference

Attachment 1:

Appendix 1-1

1. Choosing and justification of severe accident scenarios for VVER-440 RP

Abbreviations

ERC - emergency, regulating compensating assembly;
EMP - emergency makeup pump (primary circuit);
EFWP - emergency feed water pump (secondary circuit);
NPP - nuclear power plant;
BRU-A – steam dump valve to the atmosphere;
VVER - water-cooled water-moderated power reactor;
MCP - main coolant pipeline;
PORV - pilot-operated relief valve;
PRZ - pressurizer;
RPC – reactor pressure chamber;
LP – lower plenum;
DH – decay heat;
SG - steam generator;
NMP - normal makeup pump (primary circuit);
RP - reactor plant;
RCC - reactor collection chamber;
StV – stop valves;
SA – severe accident;
TG – turbine generator

1.1. General

Low probabilities of initiating events of the beyond the design basis accidents and still lower probabilities of the core melting are typical for the RP with reactors of VVER type. Availability of large amount of water in safety systems leads to that the process of accident after the beginning of the core melting can proceed under various scenarios. Thus, full meltdown of the core and the further formation of the melt pool on the bottom have negligible probabilities [1] that may be classified as improbable events. Scenarios with partial melting of the core should be considered as more probable ones.

It is known, that from a possible spectrum of accident modes in RP with VVER-440, leading to serious damage of the reactor core, accidents with the coolant leaks from the primary circuit are the most dangerous. Therefore, a number of SA scenarios in VVER-440 caused by primary circuit coolant leaks is considered and analyzed below.
The main objective of the conducted analysis was determination of such modes of SA course, at which sufficiently long maintenance of pressure in the primary circuit (at level from 0,3 to 1,8 MPa) in the course of accident is provided.

Recognizing that the main method of study of the RP behaviour at SA is the numerical simulation, the computer code SOKRAT/B1 [2] is used in the present research for carrying out the numerical analysis. The computer code SOKRAT code is used for through calculation of severe accident in the RP up to the reactor vessel failure. Brief description of the computer code SOKRAT is presented in Appendix A-2, brief description of the calculated VVER-440/230 model using the SOKRAT code is presented in Appendix A-3.

The main stages of calculation and analytical studies were as follows:

1) analysis and choosing of probable SA scenarios, at which the full or partial core disruption occurs leading to the long-term influence of corium on the VVER-440;

2) numerical simulation of SA process by means of the computer code SOKRAT.

1.2 Initial data. Choosing of SA scenarios in RP VVER-440/230

On the basis of VVER-440 safety analyses at the beyond the design basis accidents carried out in the nineties, it is possible to draw a conclusion, that disruption of the reactor core, moving of disruption products onto the reactor bottom, formation of corium, its heating and melting are most probable for the time interval up to 24 hours from the beginning of accident for initiating events concerned with the primary circuit coolant leaks. From all spectrum of probable primary circuit leaks, partial disruption of the reactor core and long-term (more than 10 hours) in-core accident progress are most probable to be realized at leaks, the equivalent diameter of which is less than Dnom 200. Realization of long-term development of the reactor core degradation at leaks, which are more than Dnom 200, is possible at operation of the safety systems providing supply of sufficient water amount into the reactor within acceptable time intervals.

It should be expected, that taking into account features of the VVER-440 RP construction, SA scenarios are probable at which partial degradation of the reactor core and long-term development of the in-core processes as well as long-term keeping of corium on the reactor bottom are possible.

In some cases, in the course of SA in VVER it is possible to restore the water supply into the reactor from the normal or emergency makeup pumps. At that, the steam, generated at evaporation of the makeup water, leads to rise in the reactor pressure to the value at which the amount of generated steam will be equal to the amount of steam entrained into leak and condensed on the equipment components.

The results of the calculation analysis of some SA scenarios in VVER-440 are presented and considered below. The Power Unit 1 of Kola NPP has been chosen as the reactor plant for which the calculation and analytical analysis was carried out. At analysis carrying out nominal parameters and characteristics of this RP equipment were accepted. The main initial data are presented in tables 1.1-1.3.

The accident scenario with the initiating event of coolant leak from MCP at the reactor inlet at Dnom less than 200 and ECCS failure is considered as a base one. At such accident restoration of water supply into the reactor from the normal or emergency makeup pumps is possible. For this accident scenario the values of makeup flow rate and leak size were estimated at which the required pressure level in the vessel is assured.

Earlier, the parametric calculations, as a result of which the dependences between the leak size and the pressure value in the reactor vessel at SA were determined, were carried out for determination and making the initial data key parameters more exact (necessary for carrying out the thermal-hydraulic calculations and SA development process) by means of an engineering procedure In the given engineering approach values of specific critical steam flow rate (at X=1), accepted according to the procedure by V.V. Fisenko, are used depending on the relation of nozzle length to the leak diameter (l/d) (table 1.4). Values of critical outflow rates at mass steam-content X=1 for a considered range of the leak sizes (from Dnom 38 to Dnom 120) are presented in table 1.5 and in fig. 1.1. Values of critical steam flow rate are received without consideration of probable condensation of steam on the reactor equipment elements. Realization of such accident scenario is possible when the primary circuit normal makeup pumps and the primary circuit emergency makeup pumps will be used as possible sources of the primary circuit makeup.

Calculations for the following variants of initiating events, on which NPP blackout is superimposed, were carried out by means of SOKRAT code:

Variant No. 1
Partial break of MCP cold leg within the range of Dnom 32 with further restoration of water supply to reactor from two normal makeup pumps with the flow rate of 3,28 kg/s.

Variant No. 2
Partial break of MCP cold leg within the range of Dnom 75 with further restoration of water supply to reactor from two normal makeup pumps with the flow rate of 3,28 kg/s.

Variant No.3
Partial break of MCP cold leg within the range of Dnom 120 with further restoration of water supply to reactor from one emergency makeup pump with the flow rate 16.4 kg/s.

1.3 The results of SA calculations in RP VVER-440/230 by means of the computer code SOKRAT

Earlier (see the Annual report for 2008 under the project №3635) calculations of the accident scenario *«Partial break of MCP cold leg within the range of Dnom 32 with further restoration of water supply to reactor from normal makeup pump with the flow rate of 1,64 kg/s»* have been executed. Chronological sequence of events in this mode is presented in table 1.6 (further as the **Variant "A"**). The basic results of the given calculation presented in table 1.6 and are necessary for an opportunity of comparison to the results received in the present research for variants № 1-3 mentioned above.

1.3.1 Variant No. 1. The results of calculations by means of the computer code SOKRAT «Partial break of MCP cold leg within the range of Dnom 32 with further restoration of water supply to reactor from two primary circuit normal makeup pumps with the flow rate of 3,28 kg/s»

The considered variant differs from the Variant "A" only in flow rate of water supplied from two primary circuit normal makeup pumps. In order to analyze influence of emergency water flow rate on course of accident scenario calculations were carried out with water supply from two primary circuit normal makeup pumps with flow rate 3,28 kg/s in 7200 s.

The main characteristics of accident are presented in Tab. 1.7 where for the analysis the results of considered earlier in the absence of feed (further as the Variant "Base"). From the table it is evident, that character of the core disruption is different.

The analysis of the data presented in Table 1.7 allows to conclude the following:

- integrity of the vessel remains within not less than 4 and 5.5 hours after the beginning of SA in case of water supply into reactor from makeup pumps with flow rate of 1,64 and 3,28 kg/s, respectively;

- destruction of the vessel at availability of feeding is caused by that only approximately 50% of thermal energy released in the melt goes to the coolant in lower plenum and intensity of convective heat removal from the RP external vessel wall is insufficient for ensuring necessary heat removal;

- it is necessary to consider a less conservative variant of the calculation diagram of the core disruption and arrival of the melt in the bottom part of the reactor vessel. In the presented calculations it was postulated, that disruption of the core occurs at a time.

1.3.2 Variant No. 2. «Break of MCP cold leg within the range of Dnom 75 with further restoration of water supply to reactor from primary circuit normal makeup pumps with the flow rate of 3,28 kg/s»

The scenario with break of MCP cold leg at the reactor inlet with leak Dnom 75 was considered. Blackout of the power unit is superimposed on initiating event. It is supposed, that in 6100 seconds after initiating event the emergency water with flow rate 3,28 kg/s and temperature 60 °C is supplied to the reactor downcomer. Main characteristics of accident Dnom 75 are presented in Table 1.8. Chronological sequence of events is presented in Table 1.9. Dynamics of RP main parameters including time of change over to steady-state mode 1000 s is presented in fig. 1.2-1.10.

Increase in the leak size from Dnom 32 to Dnom 75 (variants №2 and №3) under the considered scenario of water supply from two makeup pumps (3,28 kg/s) reduced time of the reactor vessel integrity keeping from 25600 s to 20000 s.

1.3.3 Variant No. 3.

« Partial break of MCP cold leg within the range of Dnom 120 with further restoration of water supply to reactor from one emergency makeup pump with the flow rate 16.4 kg/s»

Calculation of accident with break of the cold leg within Dnom 120 and water supply from one emergency makeup pump after drying of the core to the inlet nozzle area according to code SOKRAT/B1 was carried out. Chronological sequence of events in the given mode is presented in table 1.10.

Results of calculation in the form of plots of change of the reactor plant main parameters are presented in figures 1.11–1.17. Change of parameters taking into account time of static setting 1000 s is presented in figures 1.12-1.16.

Change of RP capacity from the beginning of initiating event is presented in figure 1.11. By the moment of the beginning water supply from ECCS pump the temperature in the area was 1000 - 1200 K. Calculations have shown, that destruction of all fuel assembly channels, moving of the melt in the CRA area and lower plenum and interaction with the reactor vessel occurs approximately at 4550th s after initiating event. Destruction of the vessel wall occurs approximately in 7500 s after initiating event. By this moment pressure in the vessel is 0,37 MPa.

Comparison of key events of the accidents for variants № 1-3 is presented in table 1.11.

The increase in the leak size in comparison with earlier considered Dnom 75 to Dnom 120 under the considered scenario of water supply from ECCS pump has reduced time of reactor vessel integrity keeping from 20000 s to 7500 s. Increase in the leak size from Dnom 32 to Dnom 75 (variants №1 and №2) under the considered scenario of water supply from two makeup pumps (3,28 kg/s) reduced time of the reactor vessel integrity keeping from 25600 s to 20000 s. At increase in equivalent diameter of the leak to Dnom 120 and water supply from one emergency pump after drying of the core time of the reactor vessel integrity keeping reduces to 7500 s.

Literature

1. V.M. Berkovich, A.B. Malyshev, Yu.V. Shvyryaev. Building of NPP power units with VVER of new generation. Teploenergetika, No.11, 2003.
2. Leonid Bolshov, Valery Strizov. Proceedings of ICAPP' 06 Reno, NV USA, June 4-8, 2006, Paper 6439

Table 1.1

Main initial data of the VVER-440

Parameters	Nominal value
Reactor power, MW Maximum linear fuel rod load Decay heat Initial position of working members group	1375 325 W/cm MS ISO 10645-92, presented in table 2 1,75 m
Pressure in RCC	12,26 MPa
Temperature of the loop cold leg	268 C
Maximum linear fuel rod load, W/cm Level in PRZ, m Steam generator: - pressure - mass of water - temperature of feed water - nominal flow rate of feed water	325 5,12 m 4,7 MPa 29 tons 223°C 125 kg/s
Coolant flow rate through: - reactor - core - operating fuel assembly - ERC	 8851,0 kg/s 8022,5 kg/s 24,46 kg/s 34,37 kg/s

Table 1.2

Power of decay heat (MS ISO 10645-92)

Parameter	**Value**							
Time, s	0,1	10,0	20,0	30,0	50,0	100,0	200,0	300,0
Power, %	6,33	4,70	4,248	3,975	3,634	3,19	2,800	2,534

Continuation of table 1.2

Parameter	**Value**							
Time, s	500,0	750,0	1000,0	1500,0	2000,0	2500,0	3000,0	5000,0
Power, %	2,358	2,166	2,010	1,809	1,665	1,557	1,472	1,259

Continuation of table 1.2

Parameter	**Value**

Time, s	10000,0	15000,0	20000,0	36000,0
Power, %	1,020	0,917	0,852	0,730

Table 1.3

Characteristics of steam dump devices

Parameters	Nominal value
Opening / closing pressure of control SG PORV	5,52 / 5,1 MPa
Opening / closing pressure of operating SG PORV	5,63 / 5,1 MPa
SG PORV capacity (at opening pressure)	69,5 kg/s
Time of SG PORV opening / closing	2,0 / 2,0 s
Opening / closing pressure of BRU-A	5,2 / 4,905 MPa
Pressure of BRU-A regulation	5,2 MPa
BRU-A capacity (at opening pressure)	55,6 kg/s
Time of BRU-A opening / closing	12,0 / 12,0 s

Table 1.4

Values of specific critical steam flow rate (at X=1) depending on the relation of nozzle length to the leak diameter (l/d)

P, MPa	L|d	G, t/m^2 *c
1,0	0,5	2,42
	6	1,44
1,5	0,5	3,6
	6	2,145

Table 1.5

Values of critical outflow rates at mass steam-content X=1 for the leak sizes (from Dnom 38 to Dnom 120) and pressure 1.0 and 1.5 MPa

Dnom, mm	38	50	80	85	90	100	120	L|d
Cross-section, m^2 *10^{-3}	1,13	1,96	5,024	5,67	6,36	7,85	11,3	
G, kg/s (1,0 MPa)	2,7	4,75	12,16	13,7	15,38	18,997	27,36	**0,5**
	1,63	2,83	7,23	8,2	9,0	11,3	16,28	**6**
	4,08	7,065	18,1	20,4	22,9	28,26	40,7	**0,5**

G, kg/s (1,5 MPa)	4,08	7,065	18,1	20,4	22,9	28,26	40,7	**0,5**
	2,43	4,21	10,8	12,2	13,6	16,8	24,2	**6**

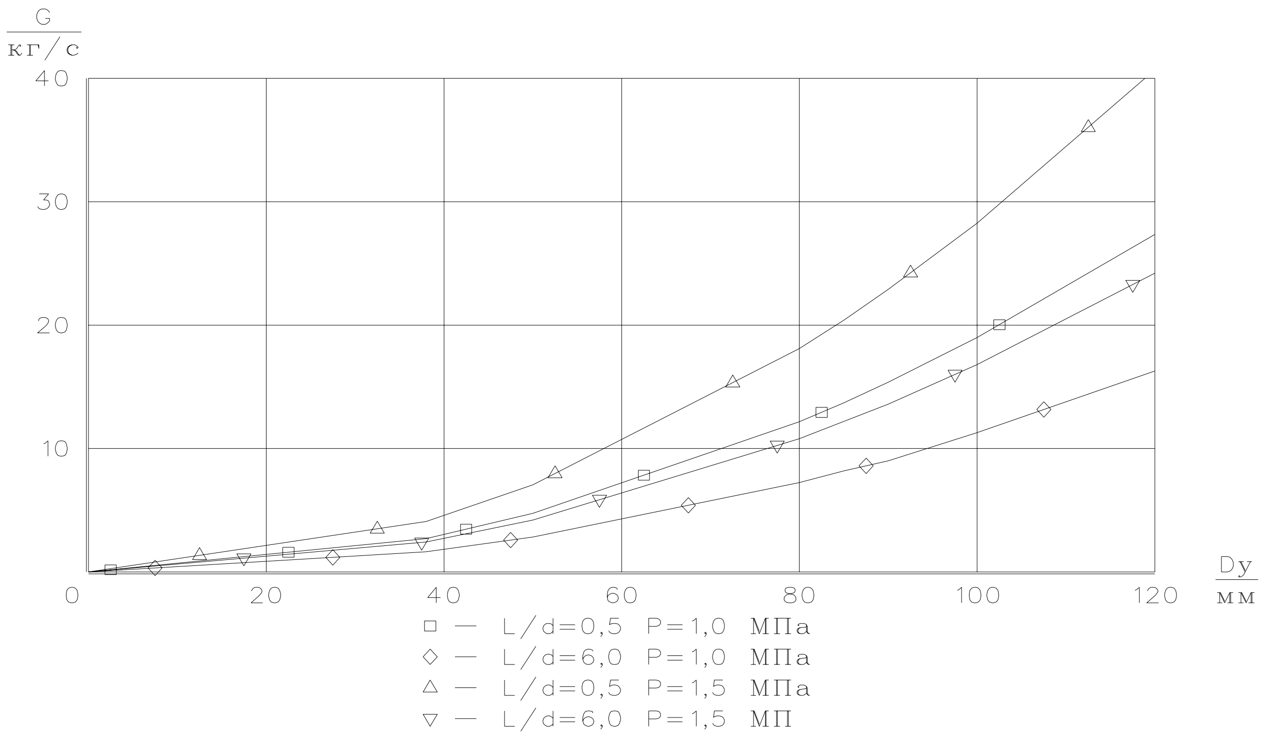


Figure 1.1 – Dependences of critical steam flow rate on the leak size (Dnom) and the nozzle sizes (L/d) for pressure of 1,0 MPa and 1,5 MPa

Table 1.6
Var. "A": Dnom 32 with water supply from 1 feed water pump with flow rate of 1,64 kg/s

Chronological sequence of events. Calculation by means of the computer code SOKRAT

Time, s*	Event
1000,0	Initiating event – leak in partial break of MCP at the reactor inlet nozzle within the range of Dnom 32
1000,0	Blackout of NPP. Due to NPP blackout the following equipment does not operate: - heaters and PRZ injection; - the primary circuit normal makeup system; - the secondary circuit feed pumps; - BRU-K RCPs of the 1st and 4th loop change over to their own coastdown
1000,0	Failure of diesel-generators to start. As a result the primary and the secondary circuit emergency makeup pumps do not operate
1000,0	Actuation of EP-1
1000,0	Closing of the stop valves of both TGs
~ 1005,0	Start of BRU-A periodic operation
1360,0	Beginning of coolant level decrease in the core
~1900-4100	Stabilization of coolant level within 1,85- 2,00 m
4500,0 (75min)	Decrease in coolant level below the core top, start of the fuel rods heating
5850,0	Maximum temperature of fuel rod cladding surfaces reaches the value of 1200 K – beginning of the developed steam-zirconium reaction
6180,0	Total drying of the core
8200,0	Beginning of water supply from 1 feed water pump with flow rate of 1,64 kg/s
10000,0	Beginning of the core disruption
13000,0	Beginning of the reactor vessel melting
15000,0	Through melting of the RP vessel
15000,0	End of calculation. Pressure in the primary circuit decreases below 1,5 MPa

*Time of the events beginning is accepted from the beginning of calculation including stationary state of 1000 s

Table 1.7
Comparison of key events of SA scenarios in RP with VVER-440

No. of TA variant	Var. "Base" (Dy 32 mm, absence of feed)	Var. "A" (Dy 32 mm + water supply, 1,64 kg/s)	Var. "1" (Dy 32 mm + water supply, 3,28 kg/s)
Water flow rate from NMP, kg/s	0	1,64	3,28
Destruction of operating assemblies of I-II years of the fuel life-time	X	X	X
Operating assemblies of the III-IV years of the fuel life-time	O	X	O
Operating assemblies of the 5th year of the fuel life-time	X	O	X
CRA	O	O	O*
Time** of the core fall, s	9000	9000	9000
Precipitated mass of fuel, t	14	28	20
Pressure in I pressurizer, MPa	0,4	1,0	1,6…1,8
Mass of water in lower plenum at the moment of the vessel destruction, kg	1400	6500	9600
The moment of the beginning of the vessel destruction, s	17500	13000	21000
The moment of through destruction of the vessel, s	20500	15000	25600

X- there is a destruction
O – integrity remains
* - 80% of fuel assemblies of control rod assembly were filled in by the moment of the beginning of the RP vessel through melting
** - time is counted from initiating event

Table 1.8 Main characteristics of accident Dnom 75

Water flow rate from NMP, kg/s	2x1,64
Operating assemblies of I-II years of the fuel life-time	X
Operating assemblies of the III-IV years of the fuel life-time	O
Operating assemblies of the 5th year of the fuel life-time	X
CRA	O*
Time** of the core fall, s	6682
Fallen out mass of fuel, t	19,8
Pressure in I circuit (MPa) since the moment when the temperature of the vessel wall has reached 1000 °C, before vessel destruction	2,4…1,8
Mass of water in lower plenum at the moment of the vessel destruction, kg	7700

* - 80% of fuel assemblies of control rod assembly were filled in by the moment of the beginning of the RP vessel through melting
** - time is counted from initiating event

Table 1.9
Var. №2
"Leak in partial break of MCP at the reactor inlet nozzle within the range of Dnom 75"

Chronological sequence of events (time is counted from initiating event)

Time, s	Event
0,0	Initiating event – leak in partial break of MCP at the reactor inlet nozzle within the range of Dnom 75
0,0	Blackout of NPP. Due to NPP blackout the following equipment does not operate: - heaters and PRZ injection; - the primary circuit normal makeup system; - the secondary circuit feed pumps; - BRU-K RCPs coastdown
0,0	Failure of diesel generators to start. As a result the primary and the secondary circuit emergency makeup pumps do not operate
0,0	Actuation of EP-1
~3250,0	Decrease in coolant level below the core top, start of the fuel rods heating
~4800,0	Total drying of the core
~5000,0	Maximum temperature of fuel rod cladding surfaces reaches the value of 1000-1200 K – beginning of the developed steam-zirconium reaction
6682,0	Time of the core fall, s
6100,0	Beginning of water supply from two normal makeup pumps
16100,0	Beginning of the reactor vessel destruction
20000,0	Through melting of the vessel, s
27000,0	End of calculation. Pressure in the primary circuit decreases below 1,0 MPa

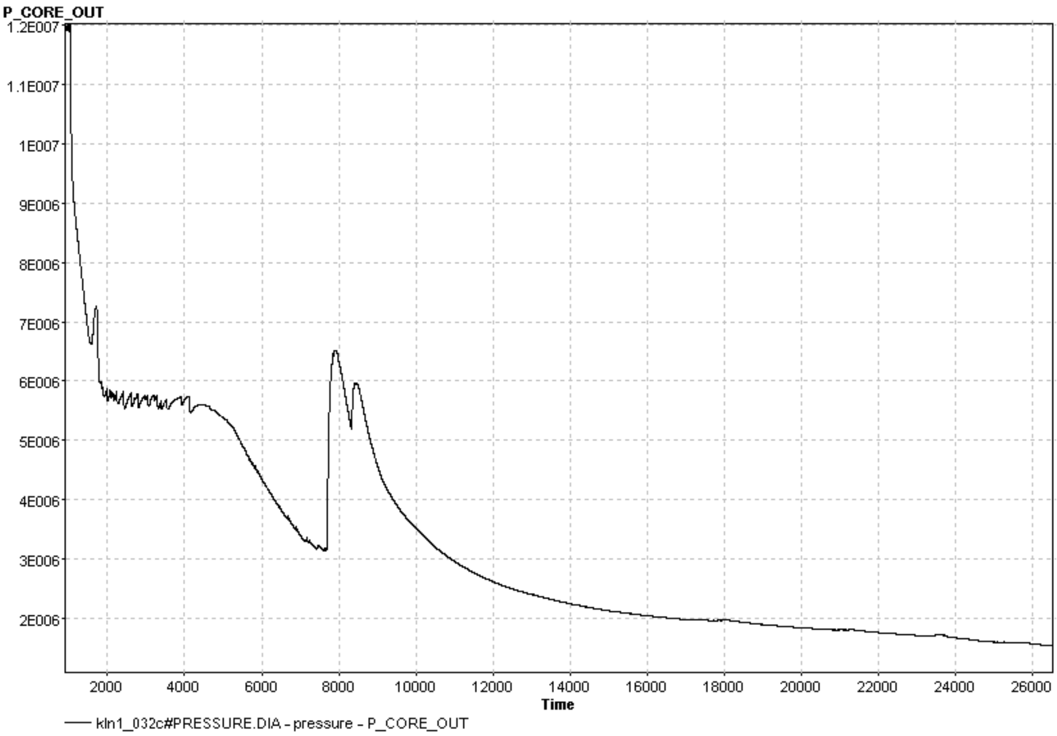


Fig. 1.2 – Var.№2. Dynamics of pressure in the primary circuit

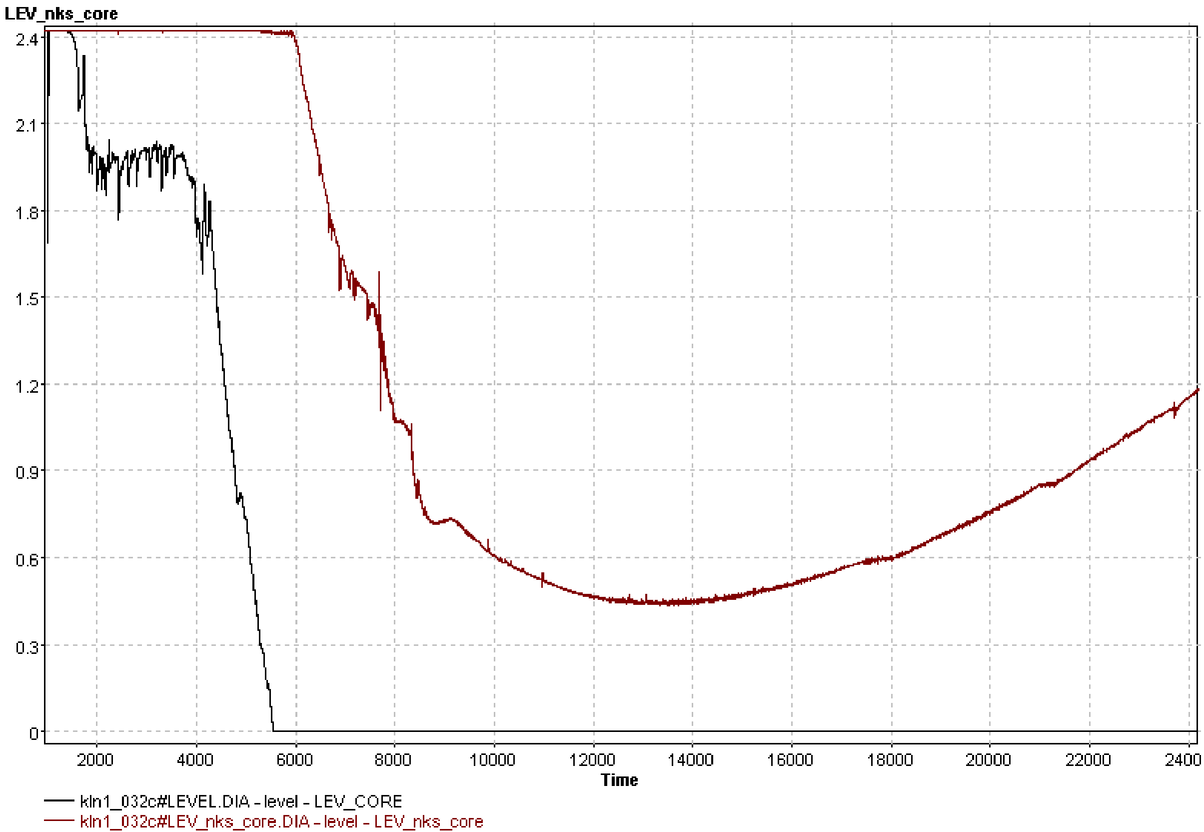


Fig. 1.3 – Var.№2. Change of the coolant level in the core (operating assembly area and CRA area)

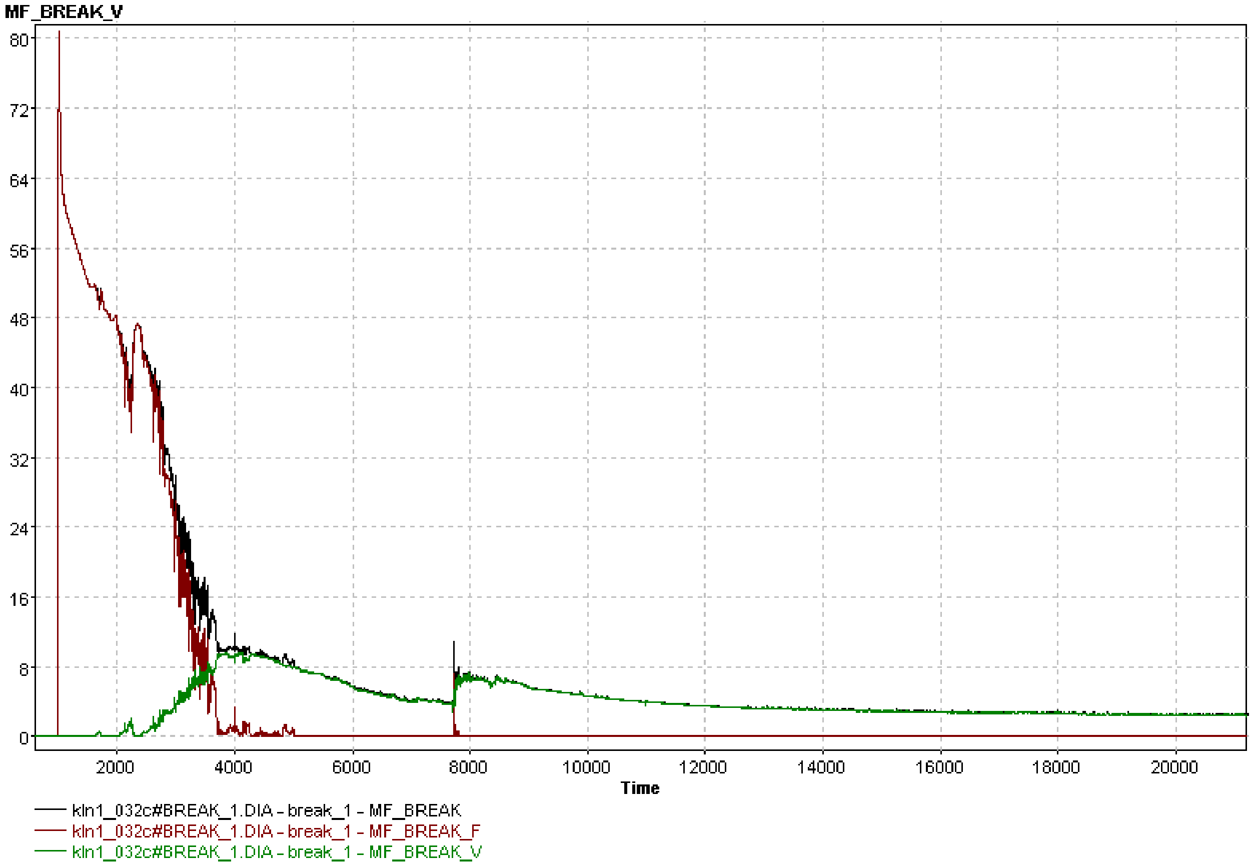


Fig. 1.4 – Var.№2. Dynamics of leak flow rate

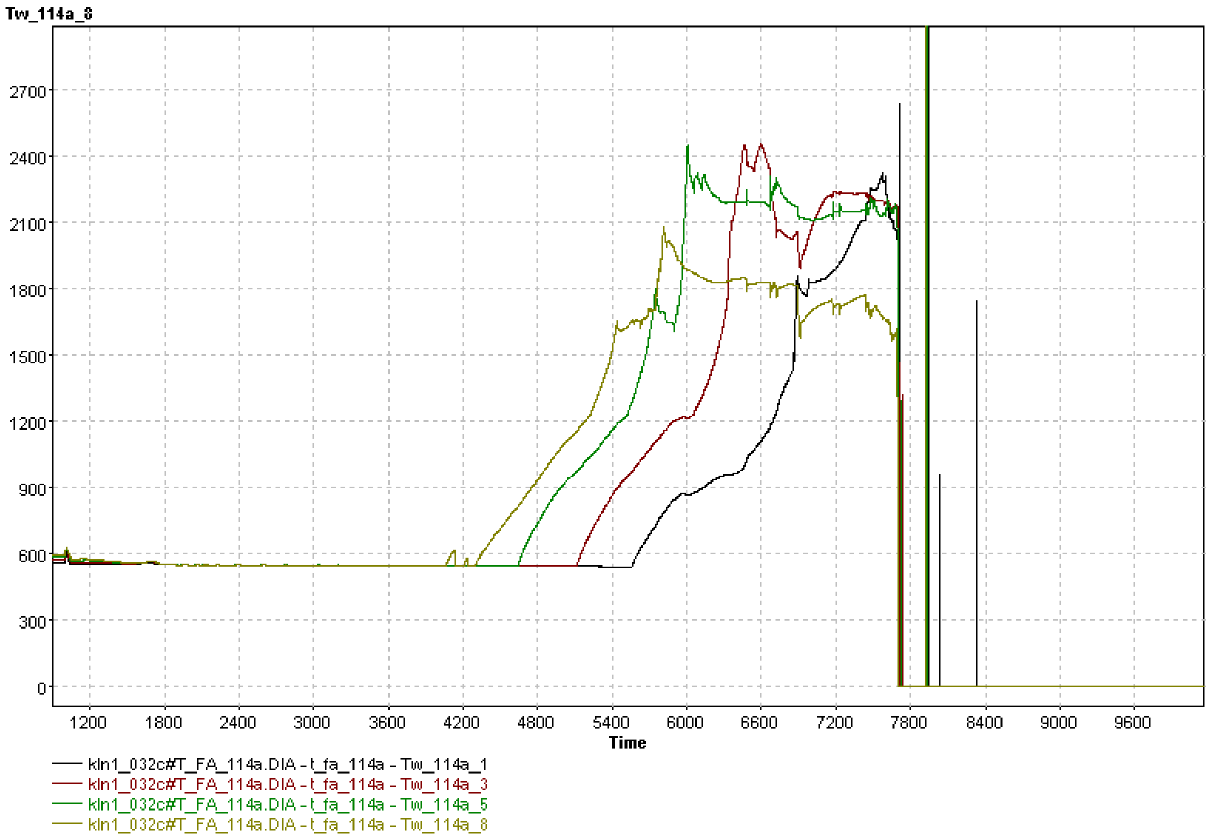


Fig. 1.5 – Var.№2. Change of temperature of fuel rod claddings in operating assembly (OA) of I-II years of the fuel life-time

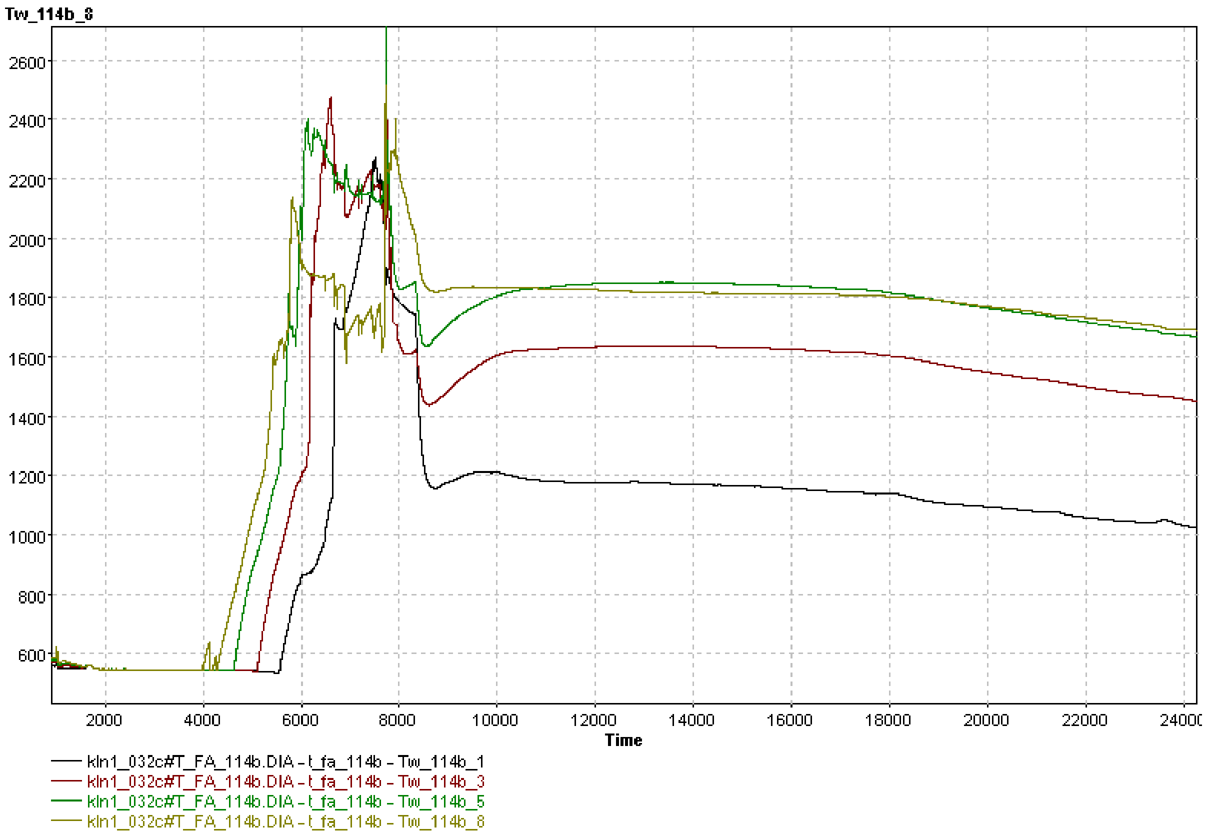


Fig. 1.6 – Var.№2. Change of temperature of fuel rod claddings in operating assembly (OA) of III–IV years of the fuel life-time

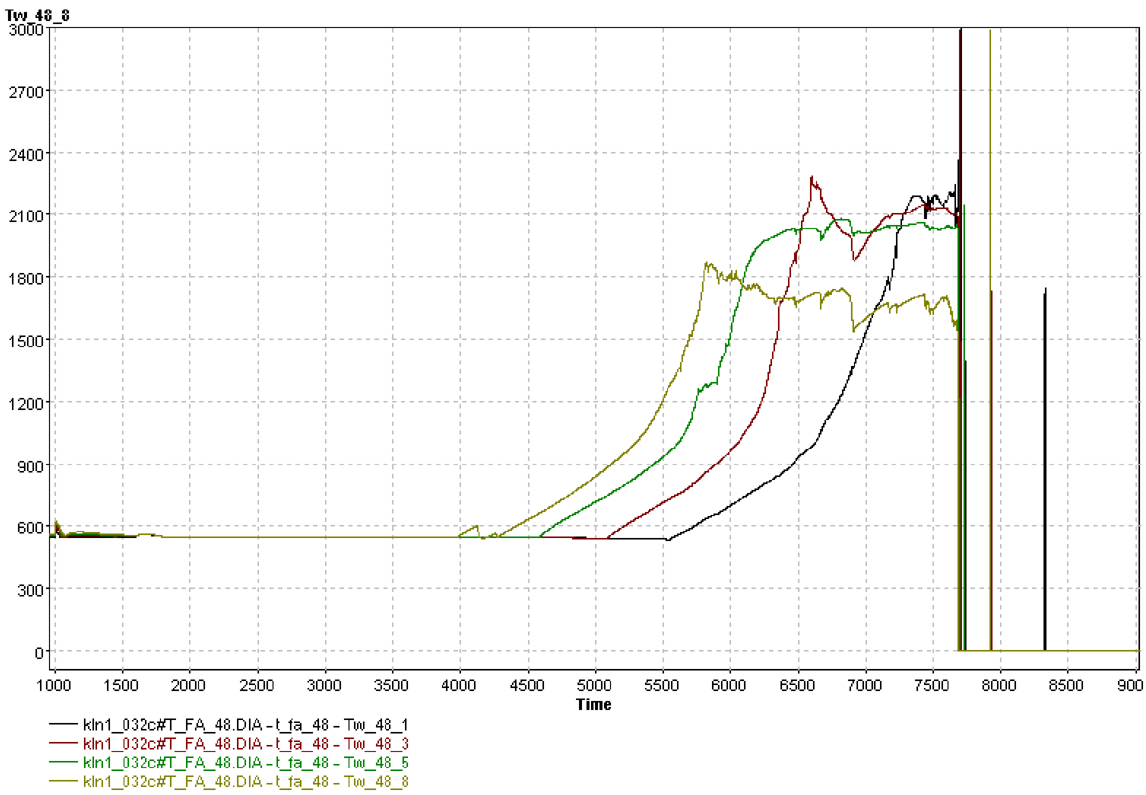


Fig. 1.7 – Var.№2. Change of temperature of fuel rod claddings in operating assembly (OA) of the 5th year of the fuel life-time

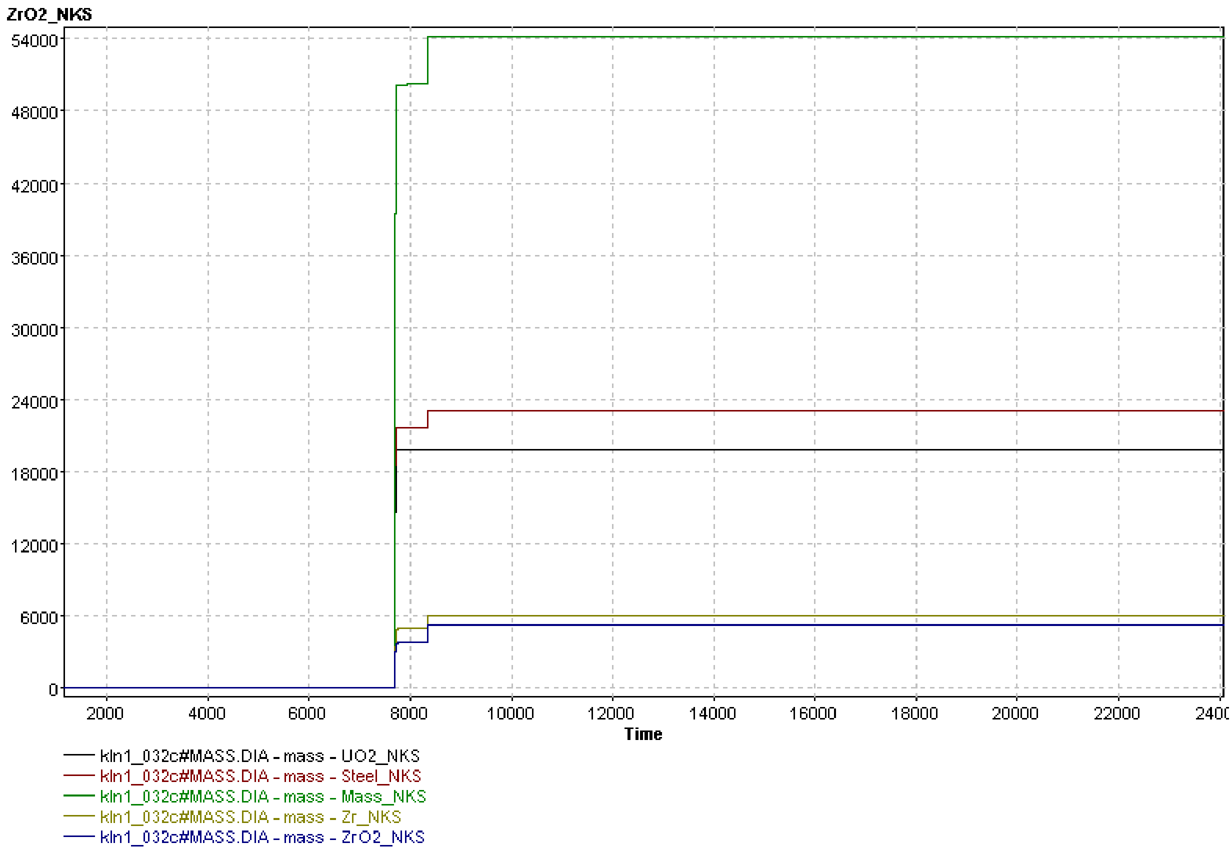


Fig. 1.8 – Var.№2. Accumulation of the melt mass in the lower plenum

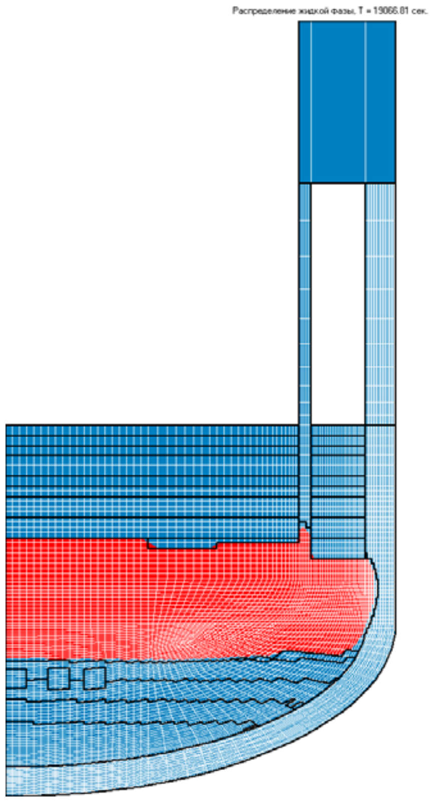


Fig. 1.9 – Var.№2. Location of liquid mass of the melt before the reactor vessel through melting by the moment of T = 19066 s from beginning of IE

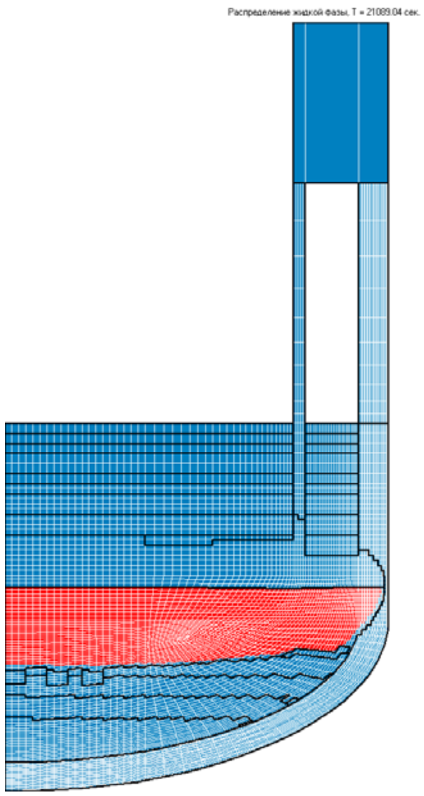


Fig. 1.10 – Var.№2. Location of liquid mass of the melt at the reactor vessel through melting by the moment of T = 21089 s from beginning of IE

Table 1.10

Var.3
«Leak in partial break of MCP at the reactor inlet nozzle within the range of Dnom 120»

Chronological sequence of events

Time, s	Event
0,0	Initiating event – leak in partial break of MCP at the reactor inlet nozzle within the range of Dnom 120
0,0	Blackout of NPP. Due to NPP blackout the following equipment does not operate: - heaters and PRZ injection; - the primary circuit normal makeup system; - the secondary circuit feed pumps; - BRU-K RCPs coastdown
0,0	Failure of diesel-generators to start. As a result the primary and the secondary circuit emergency makeup pumps do not operate
0,0	Actuation of EP-1
260,0	Decrease in coolant level below the core top, start of the fuel rods heating
~500,0	Total drying of the core
800,0	Maximum temperature of fuel rod cladding surfaces reaches the value of 1000-1200 K – beginning of the developed steam-zirconium reaction
800,0	Beginning of water supply from 1 primary circuit emergency makeup pump (~ 16,4 kg/s)
4550,0	Destruction of the core operating assembly, beginning of corium interaction with reactor vessel
7500,0	Destruction of RP vessel. Pressure at the moment of destruction in reactor vessel 0,37 MPa
20000,0	End of calculation

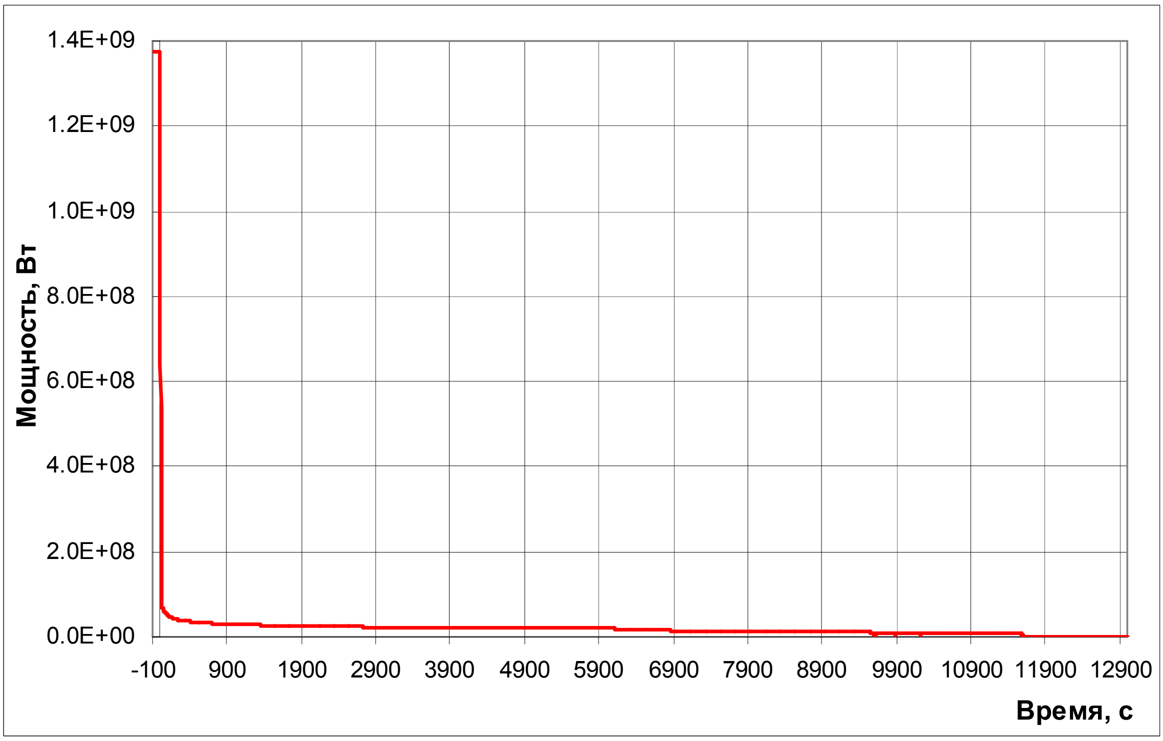


Fig. 1.11 – Var.№3.   Change of RP capacity

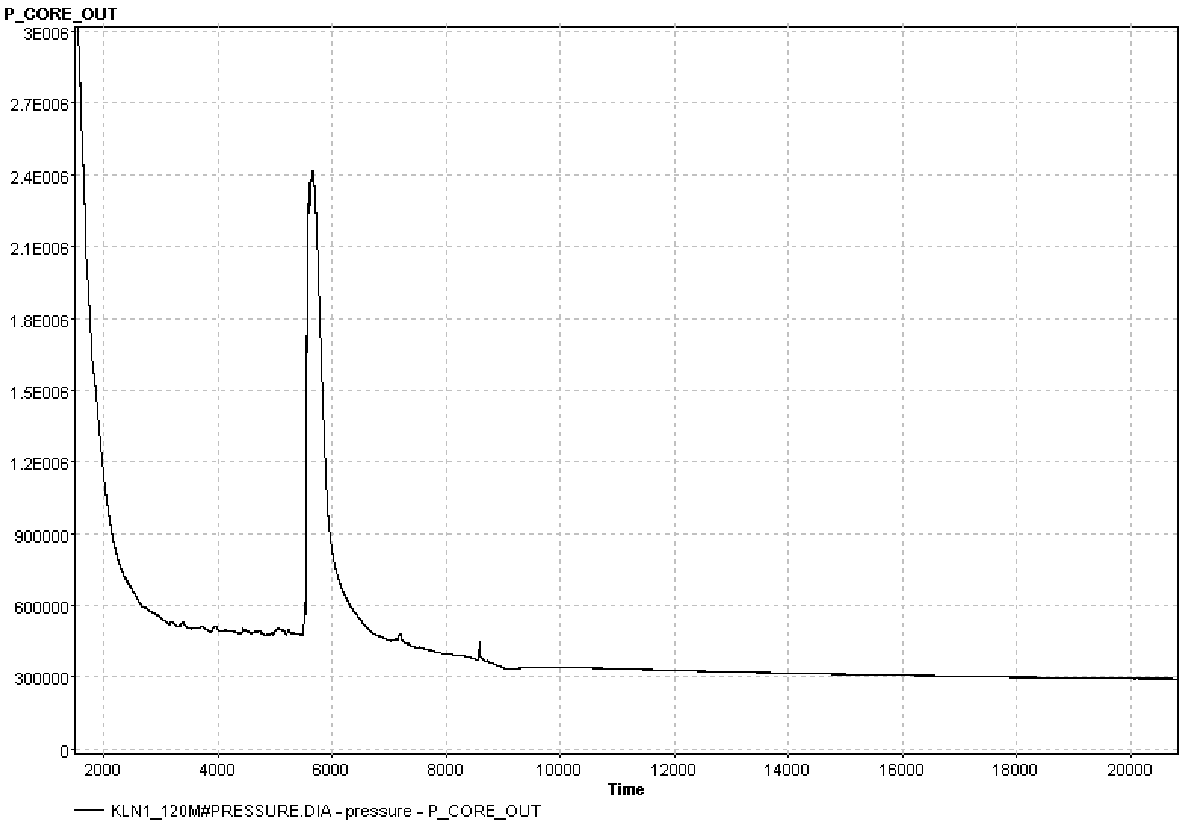
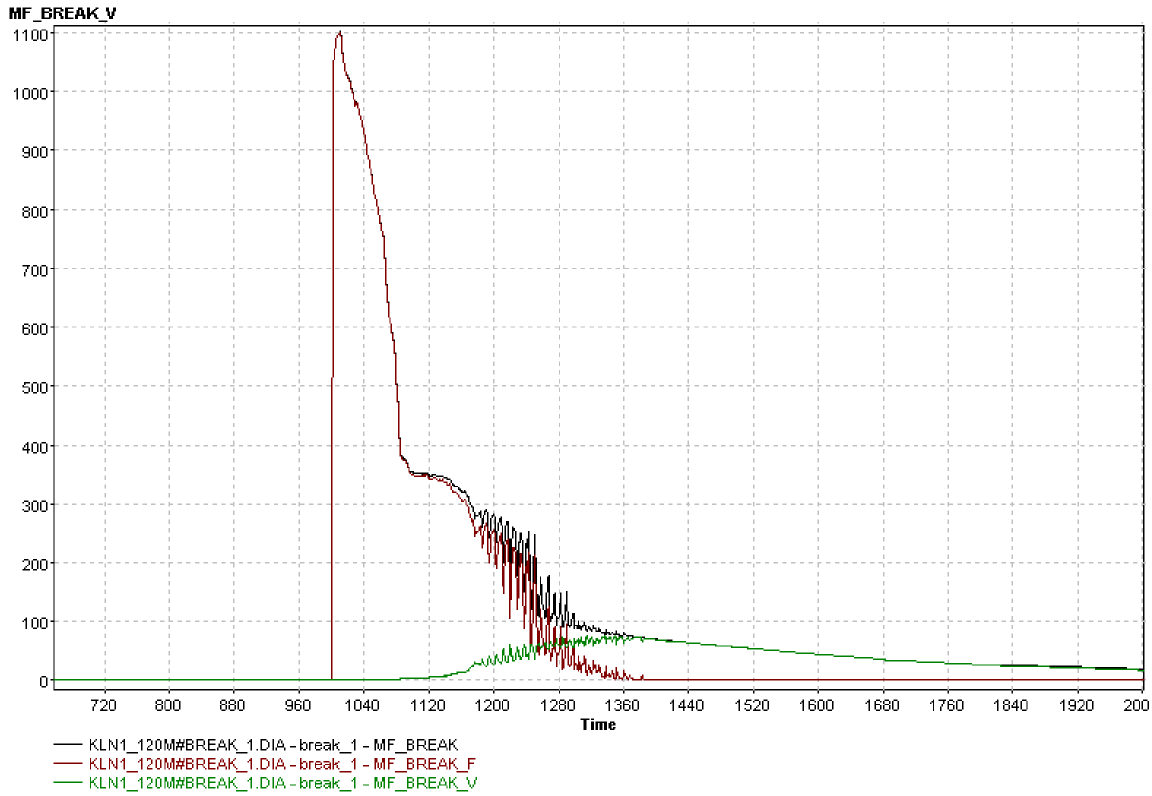
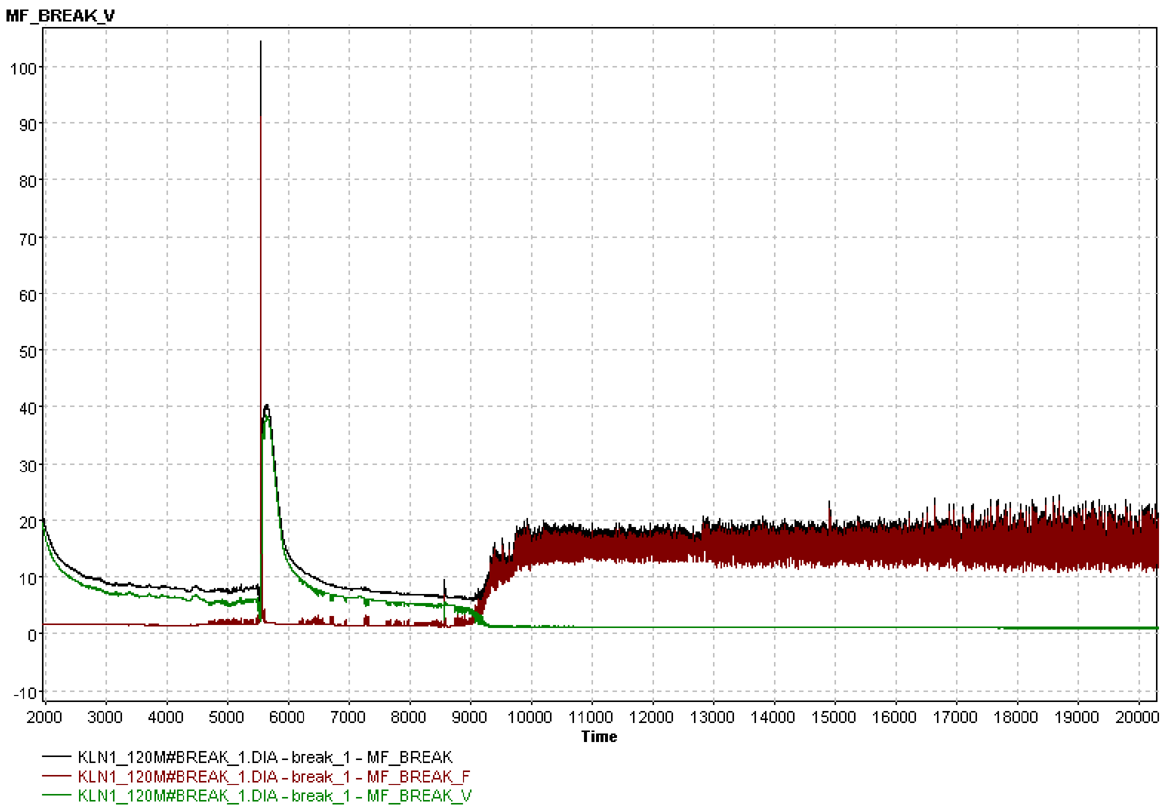


Fig. 1.12 – Var.№3. Dynamics of pressure in the primary circuit



___ Total ___ Water ___ Steam

Fig. 1.13 – Var.№3. Dynamics of leak flow rate



___- Total ___ Water ___ Steam

Fig. 1.14 – Var.№3. Dynamics of leak flow rate

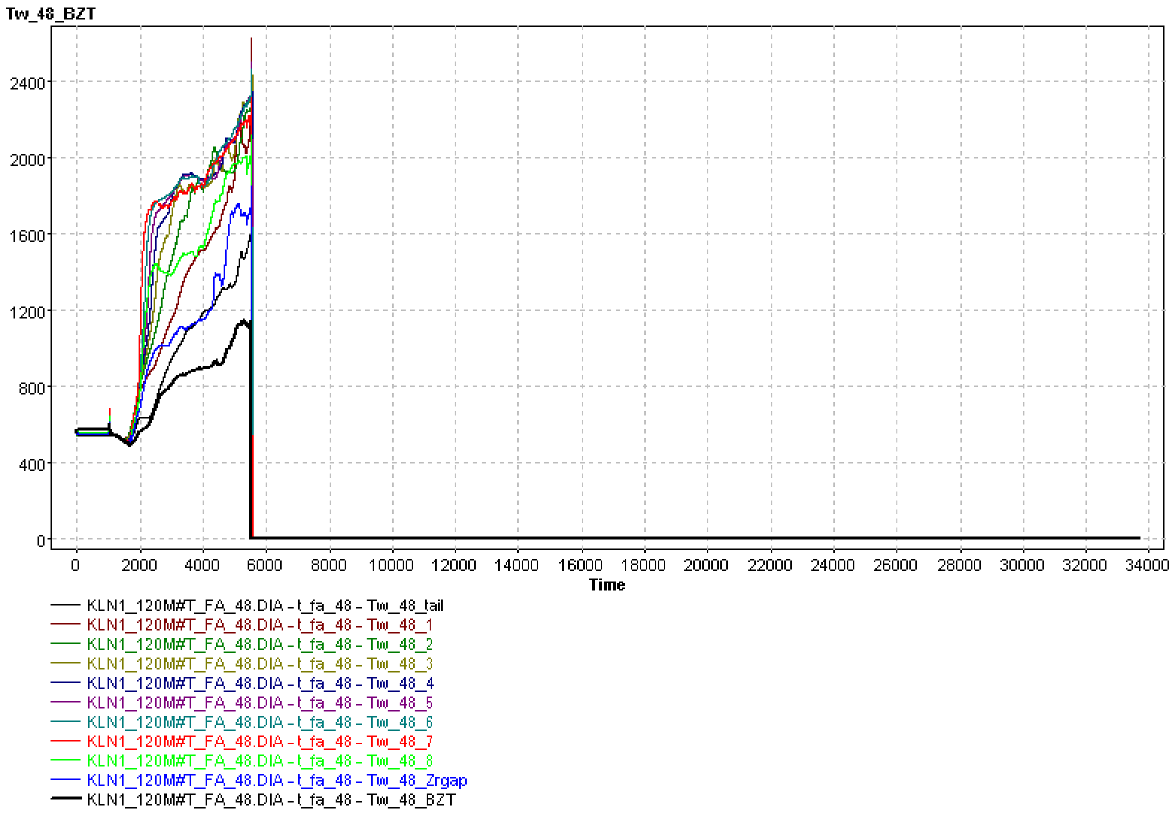


Fig. 1.15 – Var.№3. Change of temperature of fuel rod claddings in operating assembly (OA) of I-II years of the fuel life-time

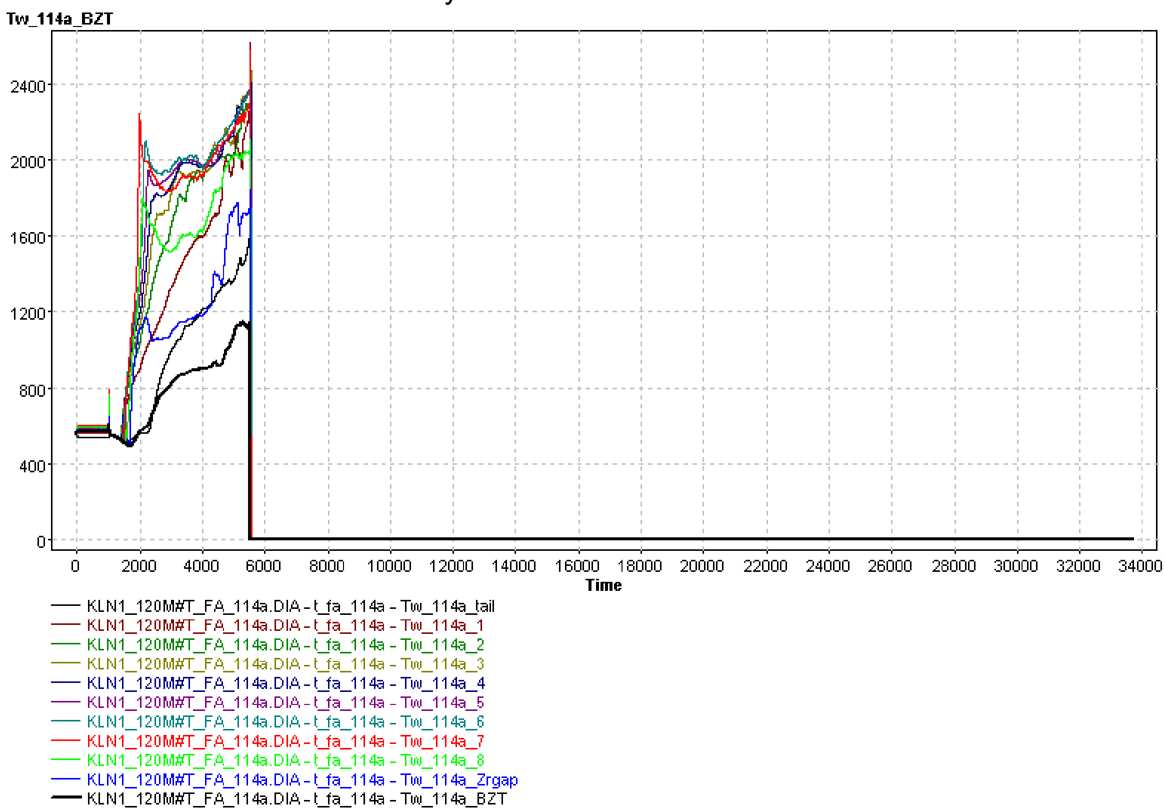


Fig. 1.16 – Var.№3. Change of temperature of fuel rod claddings in operating assembly (OA) of III–IV years of the fuel life-time

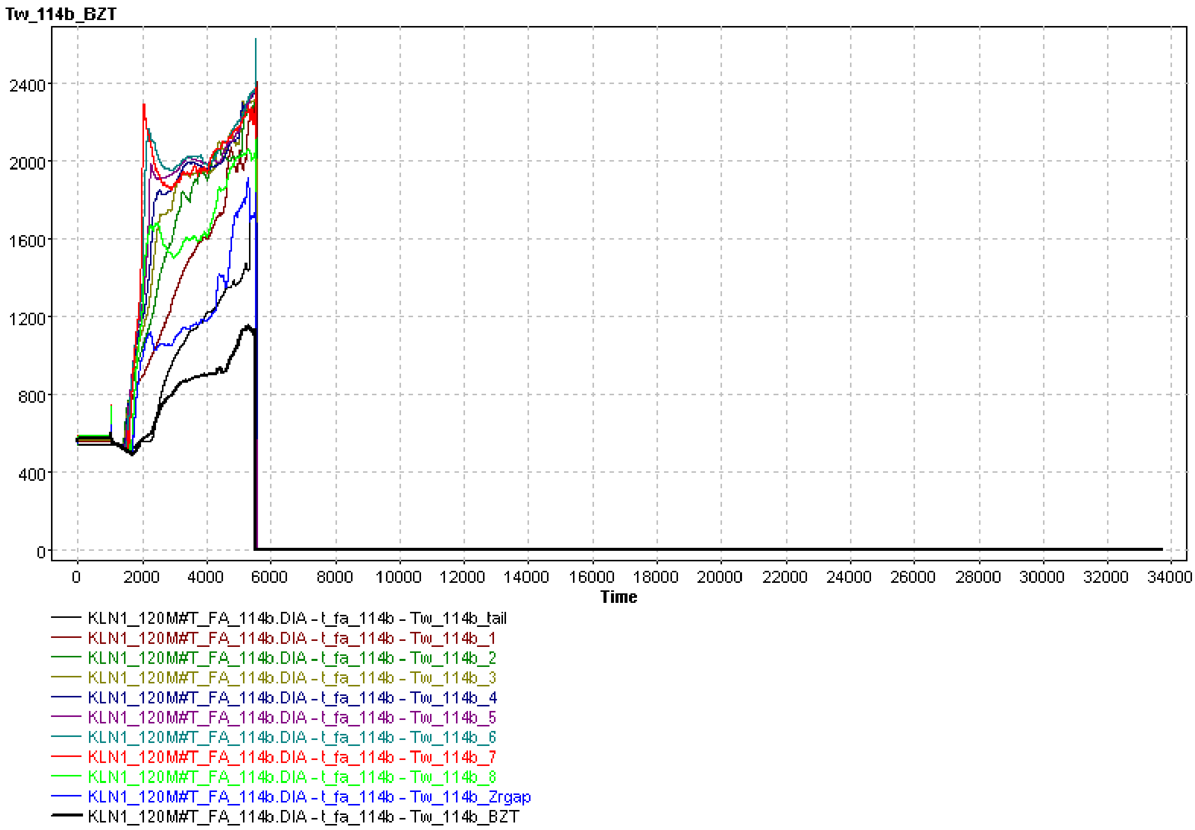


Fig. 1.17 – Var.№3. Change of temperature of fuel rod claddings in operating assembly (OA) of the 5th year of the fuel life-time

Table 1.11

Comparison of key events of the accidents for variants “A” and №1-3

No. of variant	**Var. “A” (Dnom 32)**	**Var. №1 (Dnom 32)**	**Var. №2 (Dnom 75)**	**Var. №3 (Dnom 120)**
Water flow rate from the pump, kg/s	1,64	3,28	3,28	16,4
Operating assemblies of the I-II years of the fuel life-time	X	X	X	
Operating assemblies of the III-IV years of the fuel life-time	X	O	O	
Operating assemblies of the 5th year of the fuel life-time	O	X	X	
CRA	O	O*	O*	
Time** of the core fall, s	9000	9000	6682	4550
Fallen out mass of fuel, t	28	20	19,8	
Pressure in I pressurizer, MPa	1,0	1,6…1,8	2,4…1,8	2,4…0,3
Mass of water in lower plenum at the moment of the vessel destruction, kg	6500	9600	7700	
The moment of the vessel destruction beginning, s	**13000**	**21000**	**16100**	
The moment of through destruction of the vessel, s	**15000**	**25600**	**20000**	**7500**

Appendix 2-A-2

Brief description of the VVER-440-230 calculated model using the computer code SOKRAT

Calculated model of the power unit includes a reactor with the reactor vessel internals, three primary circuit circulation loops (one emergency loop with the pressurizer (PRZ), one equivalent not emergency loop, uniting loops 2 and 3, as well as one equivalent not emergency loop, uniting loops 4, 5 and 6), steam generators (SG), pressurizer system, safety systems. Connection with the turbine generator and the condensate-feeding path is realized by means of boundary conditions.

Calculated model of reactor

The reactor model consists of the inlet nozzle area, the downcomer, the lower plenum, area of the reactor core barrel bottom and the ERC protective tubes, the core, the upper plenum, the outlet nozzle area. Nodalization of the reactor is shown in figure P-1.

The inlet nozzle area is simulated by one chamber. The chamber is connected with the cold pipeline of each loop by horizontal connection, and also with the RPC downcomer by vertical connection, and through bypass section – with the outlet nozzle area which is also simulated by one chamber. The RPC downcomer is presented by three elements: VESSEL_DOWN1, VESSEL_DOWN2, VESSEL_DOWN3. The channel VESSEL_DOWN1 simulates the circular channel between the vessel and the reactor core barrel from the cold nozzle area to the level of the bottom plate of takeout basket, VESSEL_DOWN2 – the area between the reactor core barrel and the reactor vessel from the bottom plate to the lower part of the reactor core barrel bottom, VESSEL_DOWN3 – a part of the area under the lower lattice of the reactor core barrel bottom.

The chamber VESSEL_LP, channel VESSEL_LP_NKS and chamber VESSEL_NKS cover the volume between the vessel elliptic bottom and the lower lattice of the reactor core barrel bottom. Connection between the chamber VESSEL_NKS and the channels NKS_CORE, NKS_ARK considers availability of perforation holes in the lower lattice of the reactor core barrel bottom by means of hydraulic resistance coefficient. The volume inside the reactor core barrel up to supporting plate is presented by parallel channels NKS_ARK and NKS_CORE. The first channel corresponds to the coolant volume in the ERC protective pipes, the second one – to the coolant volume among pipes. The chambers PLATE_ARK and PLATE_CORE simulate the coolant volume in the supporting plate apertures.

The volume inside the baffle on the core height breaks into three vertical parallel channels CORE_127, CORE_132 and CORE_54. In each channel fuel part divided into 11 cells on height is allocated. Element CORE_132 simulates the core volume with 132 assemblies (fuel assemblies of ERC and RC) with of the 3rd and the 4th years of the fuel life-time, element CORE_127 simulates the volume with 127 assemblies of the 1st and the 2nd year of the fuel life-time, element CORE_54 – the volume with 54 assemblies of the 5th year of fuel life-time. Element CORE_BYPASS simulates a leaking of the coolant from the bottom reactor volume into the top one bypassing the core: the channels inside the baffle, gaps between the core and the baffle, the baffle and the reactor core barrel, assemblies-shields. The channel CORE_ARK simulates a part of the core near to absorbing ERC extension pieces.

The reactor collection chamber is simulated by five thermal-hydraulic elements. The elements VESSEL_VKS_L, VESSEL_VKS_L_M, VESSEL_VKS_M simulate the coolant volume between the bottom and the top plate of the protective tube unit (PTU), element VESSEL_VKS_U simulates the volume between the PTU top plate and the reactor cover. The chamber VESSEL_VKS_M represents the volume inside the reactor core barrel with the center of mass located at the inlet nozzle level. The volume between the reactor core barrel and the reactor vessel at the same level is presented by one equivalent chamber connected, on the one hand, with volume in the reactor core barrel VESSEL_VKS_M by quasichannel VESSEL_VKS_M_OUTLET according to the flow section of the reactor core barrel perforation holes, and on the other hand, with the hot legs of circulation loops.

Metal constructions and the reactor core elements are presented by two types of elements: the elements processed by the module RATEG and the elements processed by the module SVECHA. Thermal elements are represented in cylindrical geometry. Structures of a complex form are also presented by cylindrical elements pursuant to criteria of mass and heat exchange surface conservation; the parameter of thermal elements multiplicity is applied for keeping of this conformity.

All assemblies, except for assemblies-shields, are divided into three groups (CORE_FA54, CORE_FA127, CORE_FA132) according to the hydrodynamic breakdown and radial profile of energy release in the core. The thermal element CORE_FA127 is set by an element of SVECHA of F_ROD type with multiplicity of 126*127 and corresponds in the construction to one fuel rod. Axial breakdown of an element (11 cells) corresponds to the breakdown of the adjoining hydraulic element CORE_127, radial breakdown is uniform, into seven cells. Thermal elements CORE_FA132 and CORE_FA54 are set similarly, with multiplicity of 126*132 and 126*54 respectively. Energy release is set in the layers containing the material “UO2” by means of formula sensors. Fuel assembly shrouds are described by the thermal elements SHROUD_FA127, SHROUD_FA132 and SHROUD_FA54 respectively.

The thermal element CONTROL_ROD (element of SVECHA of S_ROD type) simulates absorbing ERC extension pieces. Element CORE_BAFFLE (an element of SVECHA of BAFFLE type) simulates the baffle. The thermal shield is simulated by the element HEAL_SHIELD. Element BZT_TUBES_LOW (an element of SVECHA of SHEATH_TUBES type) simulates PTU pipes located between the bottom and middle plates. Element BZT_PERF_SHROUD (an element of SVECHA of BAFFLE type) simulates the PTU perforated shroud.

Thermal elements of the core and the core baffle are connected among themselves by radiant heat exchange.

Thermal elements BARREL_ARK, BARREL_CORE and BARREL_BZT (an element of SVECHA of BAFFLE type) simulate the in-core reactor barrel within ERC protective pipes, the core and the protective tube unit, respectively.

The bottoms of the reactor core barrel and the reactor vessel are set in module GEFEST.

Thermal elements RPV_WALL_ARK, RPV_WALL_CORE, RPV_WALL_BZT, RPV_WALL_U (RATEG elements) simulate the reactor vessel. For outer surfaces of these elements the temperature of medium is set from the module ANGAR as a boundary condition.

The other reactor metal constructions as well as circulation loop pipelines and SG heat exchange surfaces are also simulated by the elements of RATEG module.

Calculated model of the reactor coolant circuit

The calculation diagram of emergency loop with pressurizer is shown in figure P-2. It includes a hot leg, SG, a cold leg with RCP. Separate sections of pipelines are allocated in chambers – for connection of the active ECCS, the PRZ surge pipeline, leak nodes. Pipeline walls are simulated by the thermal elements HOT_WALL_1 and COLD_WALL_1. On external borders of pipelines the temperature is set corresponding to the temperature in the protective casing.

The other loops differ from emergency one negligibly.

Calculated model of the primary circuit SG

The following main elements are included in the steam generator model: a pressure (hot) collector, heat-exchange pipes, a cold collector and valves of emergency gas removal system (EGRS).

SG1 hot collector consists of the channel SG_IN_1 and chambers HC_L_1, HC_M_1 and HC_U_1. Configuration of the cold collector is similar to configuration of the hot one. The hydraulic channels SG_TUBES_L_1, SG_TUBES_M_1, SG_TUBES_U_1 and corresponding thermal elements are three packages into which the heat-exchange pipe bundle breaks on height. The diameter of pipes is $\varnothing 16\times 1.5$ and the length is 9,418 m. Multiplicity of thermal and hydraulic elements is set taking into account distribution of heat-exchange surface on the SG height.

Calculated model of RCP

Operation of RCP is simulated by means of a special pump model (element PUMP, Type=3). At the stage of stationary state setting, the elements of REGULATOR type are used for receiving in loops of the required coolant flow rate. When changing over to calculation of transient and emergency modes, angular velocity is set by means of the table.

Calculated model of the pressurizer system

The calculation diagram is shown in figure P-2. The surge pipeline SURGE_LINE is presented by three channels and it is connected to the chamber HOT4_1 of the loop No. 1 hot leg. The PRZ consists of hydraulic elements PRZ_I, PRZ_M, PRZ_U. Power of PRZ tubular electric heaters (thermal element PRZ_HEATER) is switched on and switched off according to the pressure regulation program in PRZ. The units PRZ TEH are presented by a variable power transducer. The PRZ vessel wall is presented by the thermal element PRZ_WALL. Zero length channels IPU_PRZ1 and IPU_PRZ2 with the valves simulating operation of PRZ PORV are connected to the PRZ steam volume PRZ_U. The flow area of the valves is set according to the steam and water capacity, taking into account the critical outflow model.

Calculated model of the secondary circuit

The secondary circuit model is limited by the steam generator model (figure P-2).
The secondary circuit steam generator is presented by the model with recirculation, with allocation of riser and downcomer.

Geometrical volumes and heat-exchange surface are distributed on height according to the design dependences of the heat exchange surface area and the volume of steam-water mixture on the steam-water mixture level in the SG.

The steam generator secondary circuit (on example of loop No.1) is divided into nine zones on height and presented by seven volumes which simulate the following:

- SG_FW_1 – feed water supply zone;
- SG_DC_1 – the downcomer, joining the channels which are free from heat-exchange tubes and which adjoin the SG vessel inner surface, and inter-packet corridors;

– SG_BOTTOM_1 – SG bottom volume between the lower vessel forming part and the pipe bundle lower row;

– SG_RISER_1 – the riser which includes heat-exchange surface, breakdown of three lower cells corresponds to the primary circuit breakdown;

– SG_SEP – volume between the pipe bundle upper rows and the dipped perforated sheet;

– SG_UP – volume over the dipped perforated sheet;

– SG_TOP_1 – the top steam volume.

Feed water is supplied through the quasichannel with boundary condition SG_MFW_1. Steam discharge pipes and the steam collector are presented by one channel SG_STEAM_1 with the equivalent area of cross section.

SG vessel wall is presented by the thermal element SG_WALL_1
The calculation diagram of the steam line system includes the boundary condition, simulating steam dump onto the turbine, and valves SG PORV, BRU-A, BRU-K.

Brief description of the simulated events and the calculated diagram of the module GEFEST

The volume of empty space from the supporting lattice level to the vessel bottom exceeds 15 m^3 and water will be there till the reactor core disruption. In case of the central melting of the core supporting lattice, the melt went out through the broken lattice gets into the water and it is split into particles of about several millimeters size. The core barrel bottom has 1640 through apertures of 40 mm diameter which are evenly distributed on the core barrel bottom lower plate. It is supposed, that the formed small particles of the hardened melt pass for the mass part through the core barrel bottom and get onto the vessel in the course of the core melting.

A similar situation may occur at the lateral melting of the reactor wall if the core hardened material blockades do not form in the top part of the downcomer. At such accident scenario the corium arrived onto the bottom plate warms up under the influence of residual energy release, evaporates water, and then melts. Water evaporation over the corium surface can be considered by setting of boundary conditions of convection type at corresponding borders of considered area.

The subarea volume between the core barrel bottom and the vessel is about 9 m^3 that in recalculation on average density of corium is about 60 tons. The amount of the core material arrived in the vessel lower head can be more than a specified value, therefore the reactor core barrel bottom will interact with the melt upon termination of the core corium receipt and melting and will fail.

Based on the SA scenarios described above, it is possible to make the following diagram for calculation of the thermal condition of the melt and the reactor vessel lower head:

- calculation is supposed as one-stage, as there is no necessity for simulation of the material moving from the reactor core barrel onto the vessel bottom;
- for consideration of thermal destruction of the reactor core barrel bottom part by means of module GEFEST, the core barrel is placed a little bit lower, than it is actually. It is possible owing to that fact that the reactor core barrel bottom part obligatory melts for the considered accident scenario.

The calculation grid of final elements, used at carrying out of numerical calculations by means of the computer code GEFEST, has 4903 points and 4672 elements. It is supposed, that the reactor vessel outer surface is cooled by air. The condition of the vessel thermal destruction (thermal criterion) is reaching of the VVER vessel steel melting temperature (~1500) on the vessel outer surface.

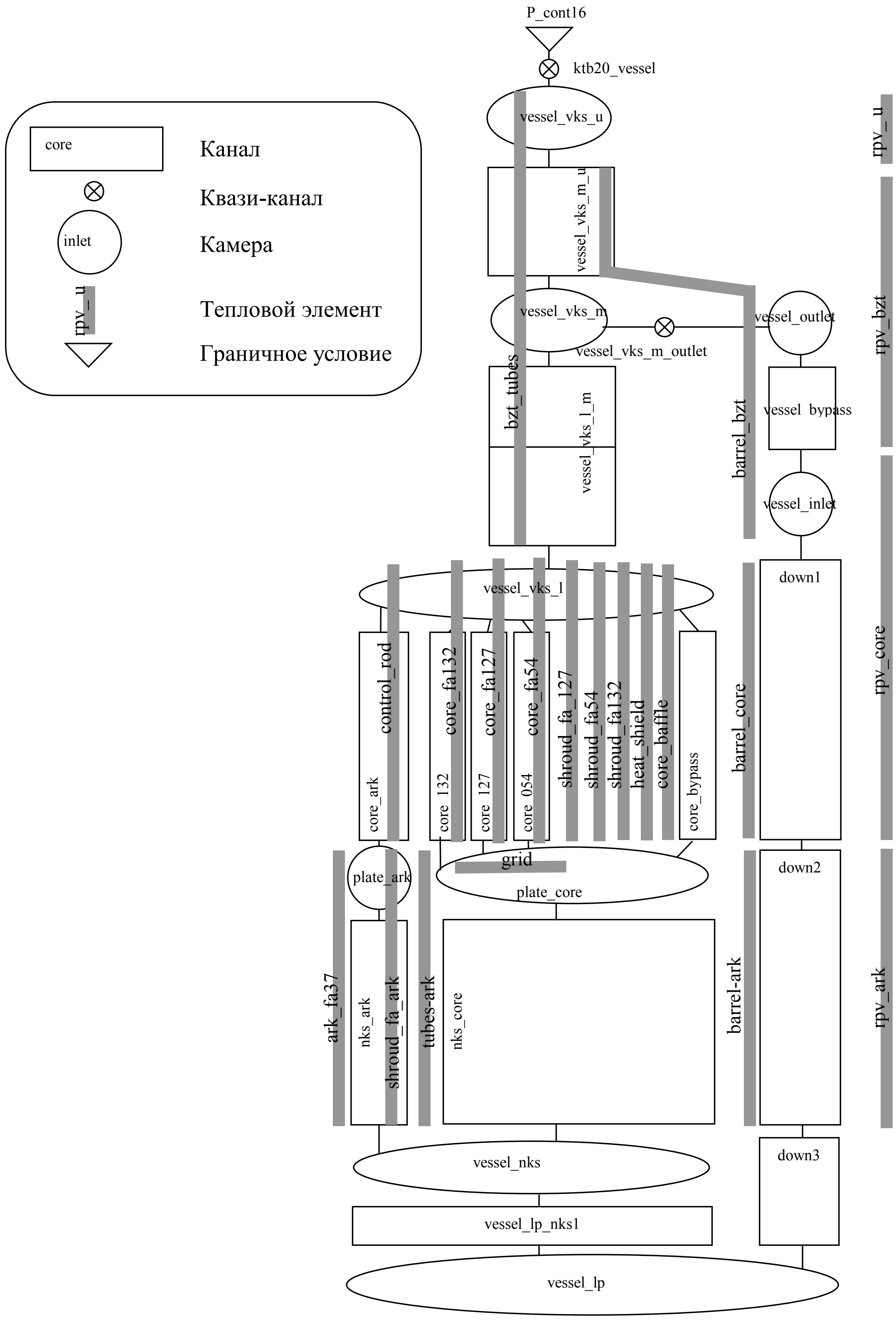


Fig. P-I – Nodalization of VVER-440/230

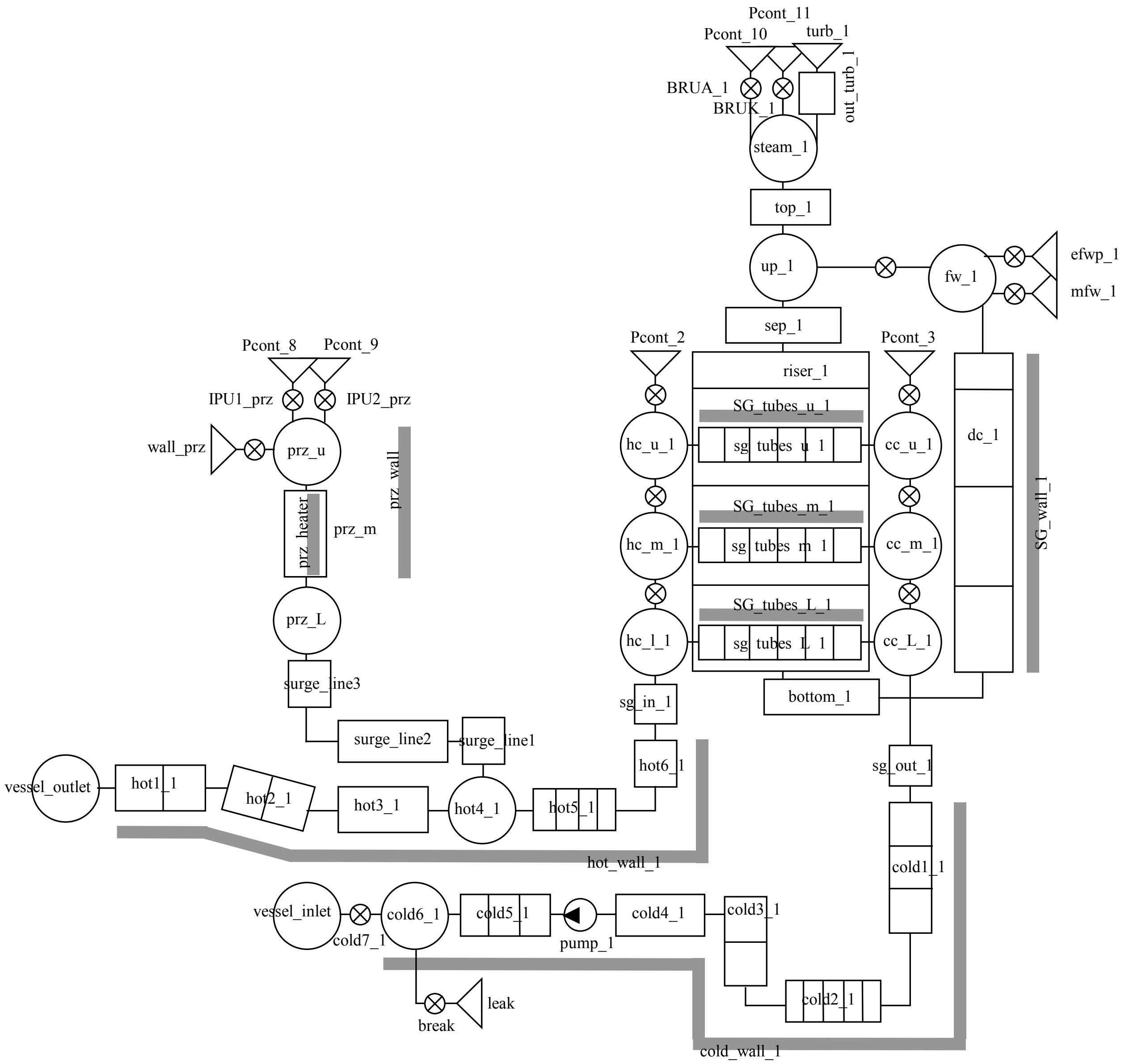


Fig. P-2 – Nodalization of VVER-440/230 primary circuit (on the example of emergency loop)

Appendix 2-1

2. Simulation of thermal condition of the melt and the VVER-440 reactor vessel by the code NARAL

Simulation of thermal condition of the melt and the VVER-440 reactor vessel lower head by means of code NARAL had the following objectives:

a) obtaining of an estimation of value and character of thermal load distribution on the vessel wall internal surface, that is important due to adequacy of the thermal load reproduction in the course of scale experiments;

b) study of conservative approaches and models application limits at simulation of thermal processes in the system “melt – reactor vessel wall”.

The computer code NARAL code is used for thermal condition of the melt and the reactor vessel lower head. Brief description of the computer code NARAL is presented in Appendix 2-A-2.

2.1 Initial data and the analysis of results

One of the basic features of code NARAL for simulation of the melting processes in corium, representing a multicomponent system, is a pseudo-homogeneous model of binary system realised in it. The model of binary system is formulated for the single-phase medium possessing effective properties of a mixture.

Basic feature of the suggested model is refusal of solving the movement equations and taking into account of turbulent convection in a liquid phase by effective heat conductivity and diffusion coefficients including molecular and turbulent components. Earlier a similar approach was used by a number of researchers with reference to pure substances at solving only a heat conduction equation without consideration of the melt components transfer.

A calculation case when the melt composition includes 7 tons of UO2 was considered in the course of simulation. Such composition of corium corresponds to earlier considered calculation case No.4 at carrying out of parametric calculations by means of code GEFEST (see the Annual report for 2008 under the project №3635).

During numerical simulation it was supposed, that ehere is no stratification in the melt.
Numerical simulation of the studied system thermal condition was carried out for two values of the energy release density (initial) in corium: 0,66 and 0,29 MW/m3.

Two calculation cases of a studied system were considered for each of these values:

Calculation case "A":

- a system "the melt – reactor vessel wall" is studied;

Calculation case "B":

- thermal condition of the melting pool is studied with specifying of corresponding boundary conditions at the border of the melt interaction with the vessel wall surface.

2.2 Results

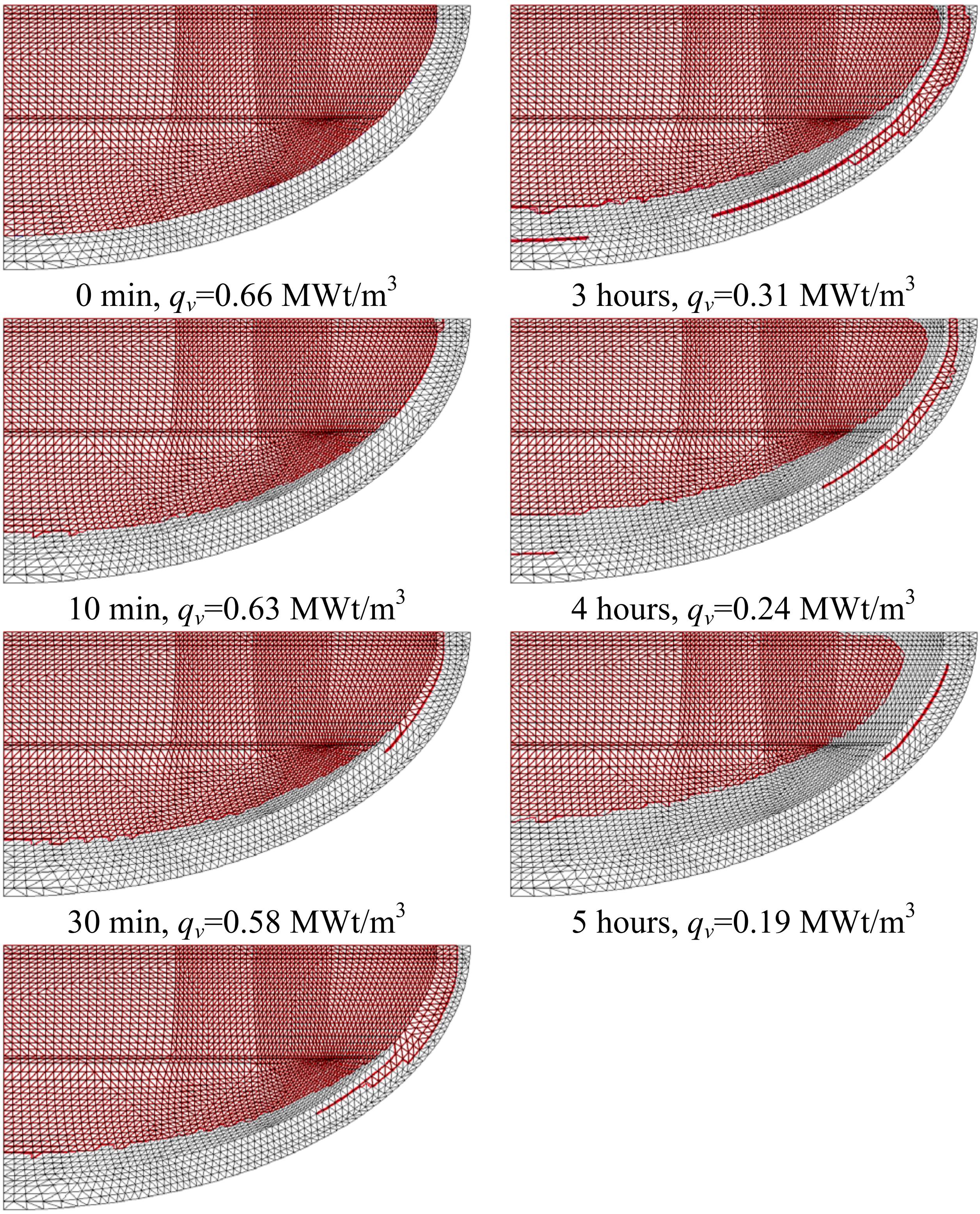
Some results of the carried out calculations are presented in fig. 2.1-2.4. It is necessary to note some features of investigated processes course:

1) Melting of the vessel wall under a formed crust of corium is observed. The most intensive melting of the vessel is observed in the top part of the vessel bottom. In a ground part of the bottom also there is a vessel melting area;

2) In the course of time these melting areas reduce owing to reduction of energy release level in the melt;

3) Maximum value of the thermal flow rate from the melt on the reactor vessel does not exceed 0.7 and 0.26 MW/m2 in case of volume energy release equal to 0.66 0.29 MW/m^3, respectively (fig. 2.3-2.4);

4) Use of the conservative approach (determination of heat flux on the vessel wall from the melt without consideration of the adjoint thermal problem "the melt – vessel wall") can lead to the underestimated heat flux values on a vessel wall (curves on fig. 2.3-2.4).



0 min, q_v=0.66 MWt/m^3

3 hours, q_v=0.31 MWt/m^3

10 min, q_v=0.63 MWt/m^3

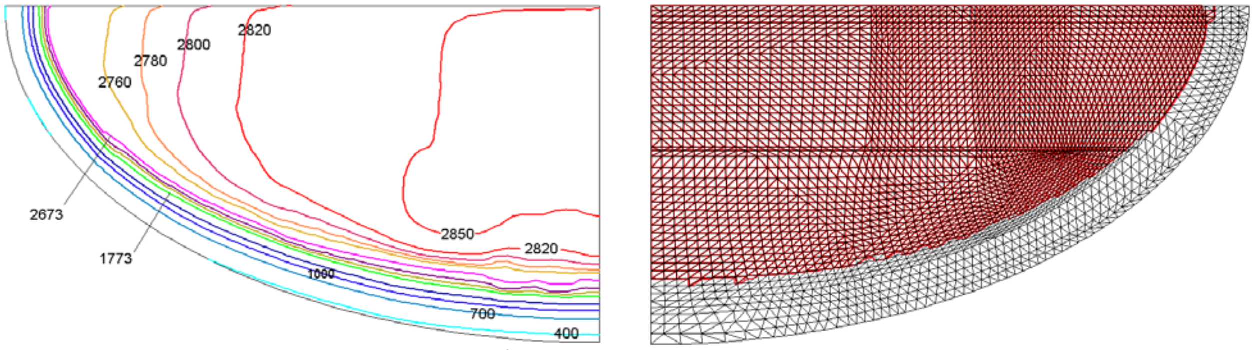
4 hours, q_v=0.24 MWt/m^3

30 min, q_v=0.58 MWt/m^3

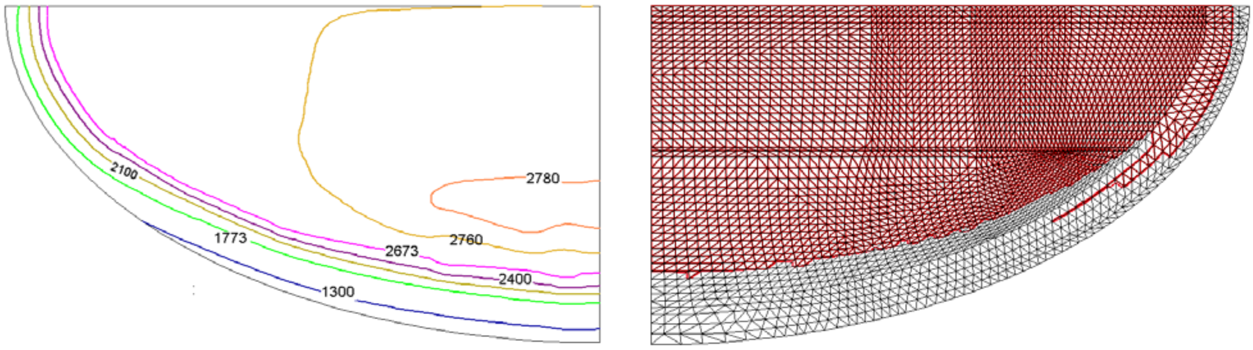
5 hours, q_v=0.19 MWt/m^3

1,5 hours, q_v=0.45 MWt/m^3

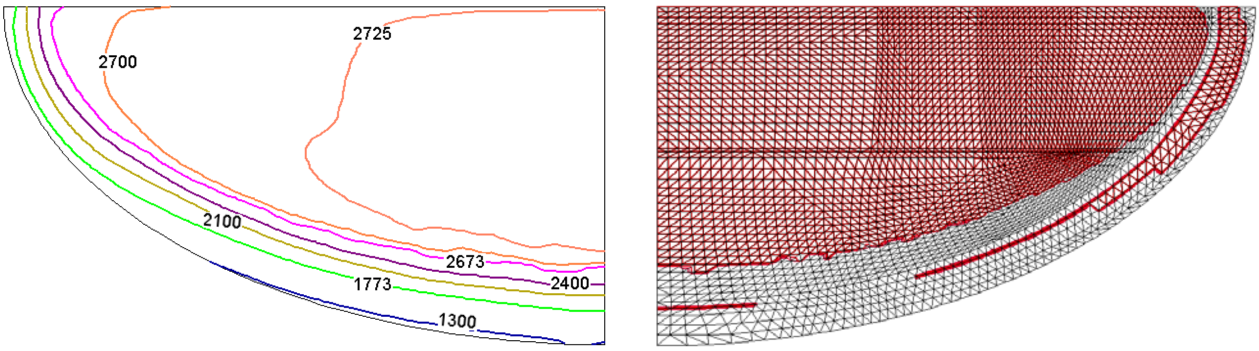
Fig. 2.1 - Liquid mass fraction distribution. Air cooling of the vessel. Initial q_v=0.66 MWt/m^3



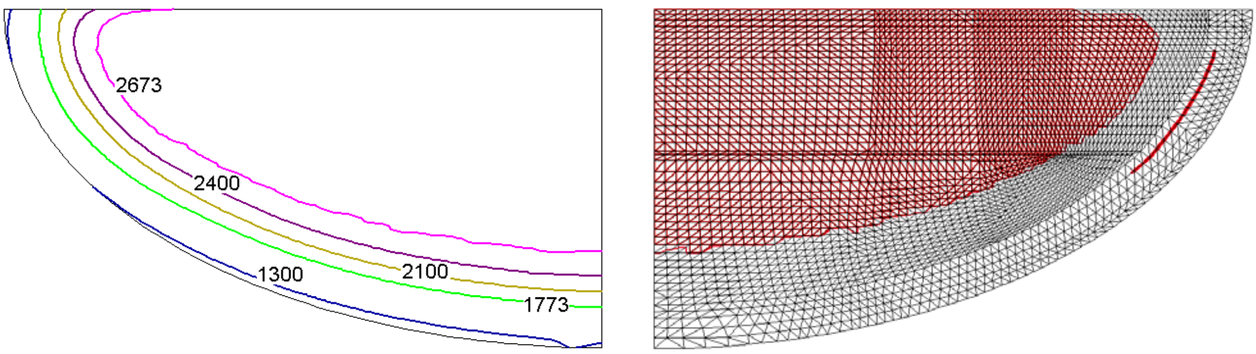
10 min, q_v=0.63 MWt/m^3



1,5 hours, q_v=0.45 MWt/m^3

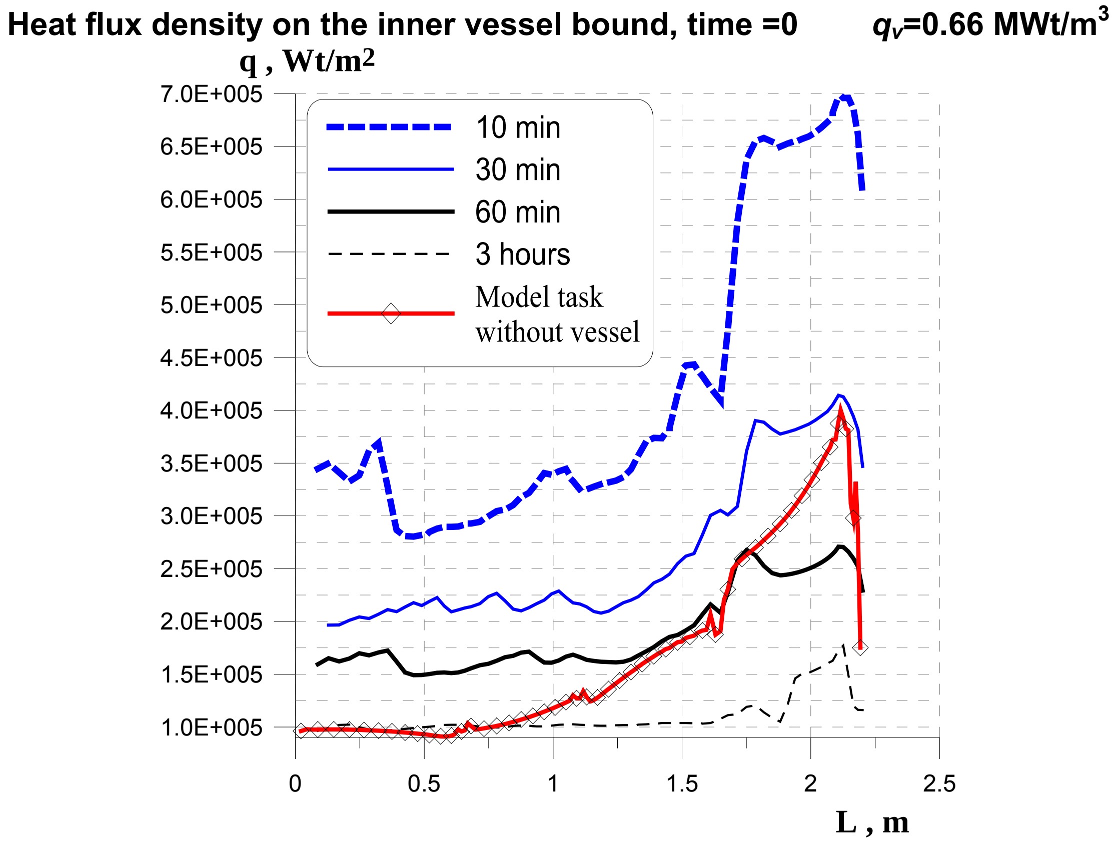


3 hours, q_v=0.31 MWt/m^3



5 hours, q_v=0.19 MWt/m^3

Fig. 2.2 – Temperate (K) and liquid mass fraction distribution (corium as eutectic material, melting at *T*=2673 K). Initial q_v=0.66 MWt/m^3



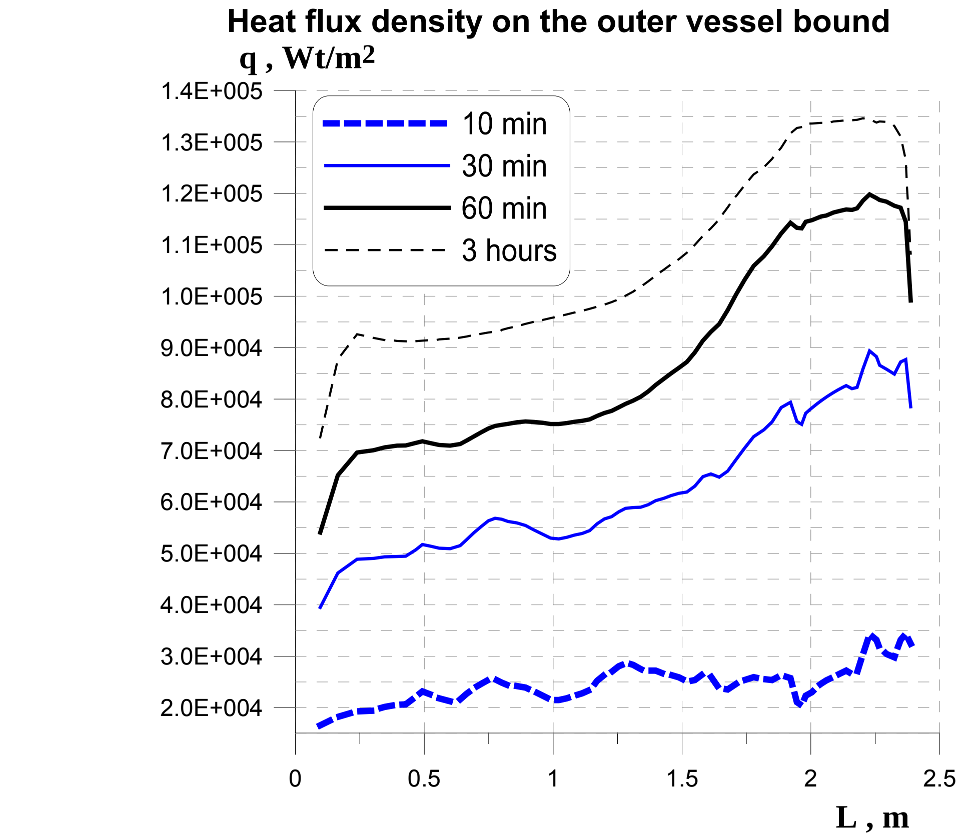
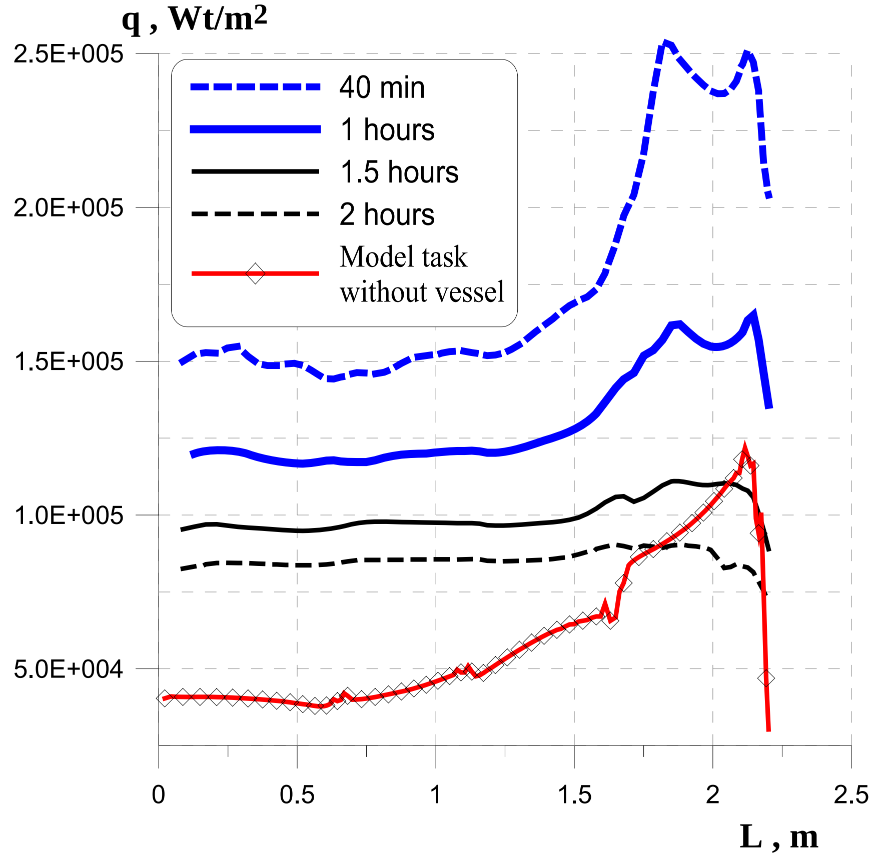


Fig. 2.3 – Heat flux density on the evolvent length of the RPV wall (L). Initial q_v=0.66 MWt/m^3

Heat flux density on the inner vessel bound, time =0 q_v**=0.29 MWt/m**^{3}



Heat flux density on the outer vessel bound

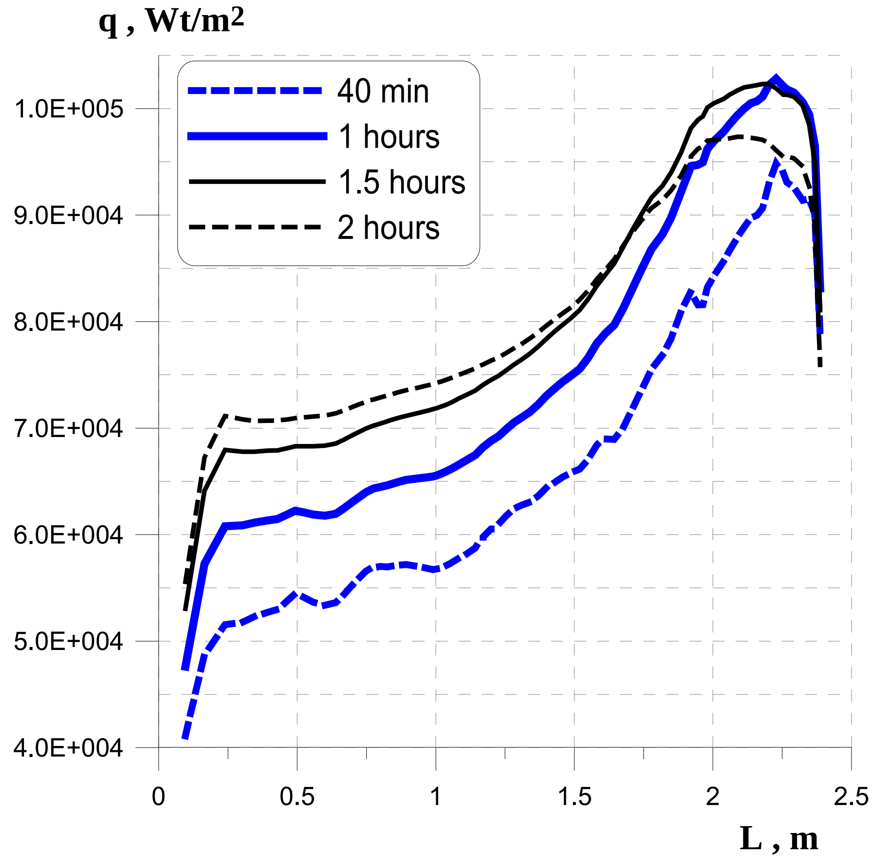


Fig. 2.4 – Heat flux density on the evolvent length of the RPV wall (L). Initial q_v**=0.29 MWt/m**^{3}

Appendix 2-A-2

P.1 METHOD OF APPROXIMATE CALCULATION OF HEAT EXCHANGE IN A PROBLEM OF CORIUM MELT RETENTION

P.1.1 Introduction

Progress of severe accident in the nuclear power rector of VVER type after disruption of the core and filling of the vessel lower head with corium melt is characterized by a difficult interaction of practically all known kinds of heat-and-mass transfer: volumetric thermal emission owing to nuclear and chemical reactions, free convection in the corium pool, heat exchange at phase transitions of multicomponent corium, constructions and reactor walls, heat transfer by radiation over the pool with melt, thermal conductivity through the vessel walls and lining material layers, boiling of cooling water on the vessel external surface. All these processes are interconnected and character of their interaction changes essentially with time. As a result, the problem of heat transfer turned out to be non-stationary, interfaced and essentially nonlinear.

Within the framework of the specified themes, the work devoted to development of computer code NARAL for analysis thermo-hydraulic processes with melting in the pool of multicomponent melt in the bottom chamber of the reactor vessel was carried out earlier. The numerical models checked by many test calculations enable carrying out detailed estimations of convective heat exchange processes of the core melt in the reactor vessel lower head with high degree of reliability. The further development of the code moves on a way of its integration with programs for carrying out strength calculation in the interfaced statement. The essential restrictive factor in this direction is a long period of time of the detailed calculations carrying out, caused by complexity of solving the equations of convective turbulent transfer for which estimation different variants of k - ε - turbulence model are usually used. At the same time, in the investigated problem as the basic practical results the estimation of a condition of the vessel and heat flows through cooled surfaces, instead of a hydrodynamic picture of currents in the melt pool is interesting first of all.

In the present section, a computer model is presented for approximate description of a temperature field in multicomponent melt and for calculation of the reactor vessel melting taking into account the most essential physical processes.

P.1.2 Mathematical model

At making a computer model intended for receiving practical results for the real designs, the aim to consider the interfaced character of a problem was pursued, based on rather simple, but sufficiently reliable models of heat transfer in the melt pool and in the reactor vessel.

The developed model should consider both influence of turbulent natural convection in a liquid core, and possible redistribution of thermal load owing to non-uniformity of the hardened layer of corium on the borders of the melt pool. Besides at traditional quasi-stationary statement, some effects, which are possible at change of a thickness of lining material layer as a result of phase transitions (melting/hardening) in multicomponent substance can be missed, and, thus, the problem should be solved in non-stationary statement.

Mathematical simulation of processes of multicomponent systems melting became possible only recently with appearance of new techniques, allowing to consider features of phase transition in such systems. Unlike one-component systems, there is a real area of melting in the multicomponent systems limited to lines of solidus and liquidus and, in some cases, to eutectic line. Depending on concentration of components, which can vary owing to convective and diffusion mechanisms of transfer, temperatures of solidus and liquidus have character distributed in time and space. It is supposed, that the corium melt consists basically of uranium oxide UO_2 and zirconium oxide ZrO_2.

Pseudo-homogeneous model of binary system which is formulated for the single-phase environment possessing effective properties of a mixture suggested and realized within the scope of two-dimensional "final-element" computer code NARAL/FEM was accepted as a basis of mathematical model. When lowering convective terms and refusing the solving of motion equations, it is possible to present the mathematical formulation of suggested model as follows:

Energy equation

$$\frac{\partial}{\partial \tau}\left(\rho\, h\right) = \nabla\left(\frac{\lambda_{eff}}{C_{ps}}\nabla h\right) + S_h + q_{\text{v}} \ , \qquad (2.1)$$

Equation of diffusion of one of the melt components

$$\frac{\partial}{\partial \tau}\left(\rho\, c\right) = \nabla\left(\rho f_l D_l \nabla c\right) + S_c \ , \qquad (2.2)$$

where

$$\lambda_{eff} = \lambda + \lambda_t$$

$$S_h = -\frac{\partial}{\partial \tau}\left(\rho\, f_l\, \delta H\right),$$

$$\delta H = \int_0^T \left(C_{pl} - C_{ps}\right) dT + L\ ,$$

$$S_c = \nabla\left(\rho f_l D_l \nabla (c_l - c)\right).$$

The subscript *l* concerns a liquid phase, *s* - a solid phase, *L* - latent heat of melting, q_v - capacity of residual volume heat emission, C_p - thermal capacity at constant pressure, λ_{eff} - effective coefficient of thermal conductivity, D_l – effective coefficient of diffusion, f_l (*T, c*) - mass fraction of a liquid phase depending on temperature *T* and component concentration *c*.

The presented equations are written down in terms of pseudoenthalpy of a solid phase

$$h = \int_0^T C_{ps} dT$$

and averaged mass concentration *c* of binary melt component. Function *h* is continuous in the whole area, and the equation (2.1) is analogous to usual equation of thermal conductivity except a source summand S_h which considers melting/hardening processes in control volume. Solving of the diffusion equation allows to calculate correctly the specified processes in multicomponent systems on the basis of the real phase diagram with solidus and liquidus temperatures depending on concentration of the component. For calculation of a mass fraction of a liquid phase f_l (*T, c*) iterative procedure of equation solving (2.1) - (2,2) is used.

Basic feature of suggested model is refusal of the motion equation solving and taking into account of turbulent convection in a liquid phase through effective coefficients of thermal conductivity and diffusion including molecular and turbulent components. Earlier a similar approach was used with reference to pure substances at solving of only thermal conductivity equation without consideration of the melt components carrying over.

A component of effective thermal conductivity λ_t considering turbulent convection is determined based on the method of free convective heat-and-mass transfer in heat-generated liquid at large Rayleigh (Richardson) numbers when the thermogravitational mechanism of turbulence generation dominates.

Verification of the program complex NARAL/FEM was carried out on a wide range of test problems of thermal conductivity and convection at high and low numbers *Ra* for simple and complex geometry. Results of calculations conform well both with experimental data and calculation data of other authors. Testing of operability, realized in a code of effective thermal conductivity model, seems also to be necessary.

P.1.3 A comparative analysis of the model

For check of the model operability, comparative calculation with reference to the experimental data received on Finnish installation COPO was carried out. In experiment, water in the closed cavity 0.1 m thick is used for imitation of molten corium; source of heat distributed uniform is kept for simulation of a residual thermal emission; the bottom, lateral and top surfaces of experimental cell are cooled at constant temperature. Geometrical characteristics of installation were chosen in scale 1:2 to the real sizes of the VVER-440 reactor bottom chamber that has allowed to reach number Ra=10^{14}-10^{15} in a series of experiments.

For simulation of experiment, in the present paper the energy equation was solved in the Cartesian coordinate system with an uniform source of volume heat emission:

$$\frac{\partial\left(\rho C_{\mathrm{p}}\, T\right)}{\partial \tau}=\left[\frac{\partial}{\partial x}\left(\lambda_{eff}\frac{\partial T}{\partial x}\right)+\frac{\partial}{\partial y}\left(\lambda_{eff}\frac{\partial T}{\partial y}\right)\right]+q_{v}\,. \qquad (2.3)$$

On borders of calculated area, the boundary first kind condition with the temperature of cooling realized in experiment is set. So long as quasi-stationary thermal mode is the most interesting, specification of initial approximations of temperatures is not so essentially.

Results of calculation for experimental series fh1 to which Ra=1.3·10^{15} corresponds are presented in tab. P1.1. The values specified in the table were received after achievement of quasi-stationary condition of heat exchange process. In this paper unlike the experiment, the maximum temperature drop in the area Δ*T* was much more (95K against 14K). The hottest area is the flow core with reference to which it is not necessary to expect from calculation correct temperature distributions. At the same time, the basic temperature drop occurs in a thin border area. Distribution of temperatures in this area determines value of local thermal load on the construction vessel that is of the greatest interest.

Table P1.1
Comparison of experimental and calculated data (series fh1)

Source of data	ΔT, K	$\bar{q}_{top}$	$\bar{q}_{side}$	$\bar{q}_{bot}$
		kW/m^2		
Experiment COPO	14.0	15.2	6.2	2.9
k-ε-model	11.5	3.5	10.0	9.0
modified k-ε-model	15.1	13.8	6.5	3.9
model of effective thermal conductivity	95.0	19.4	7.2	2.8

It is necessary to note, that alternative calculations of this problem on grids which are different regarding character of junction crowding to borders of area showed sensitivity of temperature field results to value of distances to the wall from nearby junctions: the more roughly is the wall grid, the lower is a maximum temperature drop in the area. It, probably, may be explained by the structure of the formula for value λ_t, depending on distance to the wall and leading on different grids to different thermal resistances in the border area. At the same time, the variation within reasonable limits of grid junction crowding value to the area borders has practically not affected values of received heat flows.

Distribution of heat flow on the bottom and vertical borders of calculated area, on length of generating line, counted from the central lower point of the elliptic bottom, received at calculation on effective thermal conductivity model is shown in fig. P1.1. Point of transition of the vessel surface elliptic generating line to the vertical wall is marked with dotted line at $l \approx 1.1$m.

Experimental and calculated values of heat flow integrated density on elliptic generating line of calculated area $\bar{q}_{bot}$, lateral vertical $\bar{q}_{side}$ and top $\bar{q}_{top}$ borders are presented in tab. P1.1. The estimations of heat flows received in detailed calculation on the basis of “low Reynolds” k - ε model of turbulence are presented here too for comparison. It is evident, that use of the last one considerably overrates a heat flow on the bottom and lateral surfaces (3 and 1.5 times respectively) and underrates 4 times on the top wall of installation. Considering shortcomings of

k - ε- models which were developed initially for description of currents in pipes, have introduced into the Chen model with reference to free convection at numbers $Ra > 10^{14}$ correction functions to account dependence of transfer coefficients on conditions of temperature and density stratification. In particular, function of dependence of turbulent Prandtl number on Richardson number $Pr_T = f(Ri)$ was used. It is evident from table P1.1, that use of k-ε model modified in such a way improves essentially results of calculation.

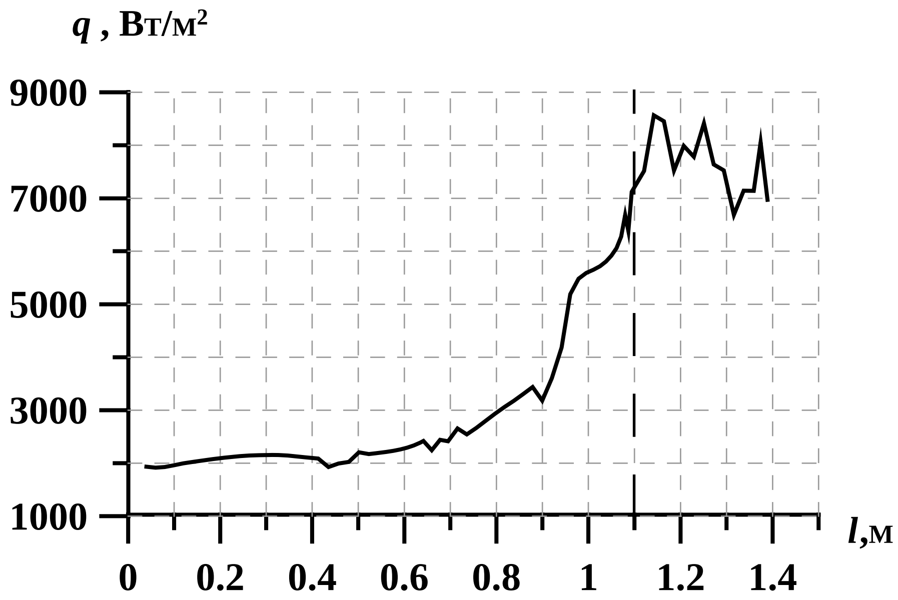


Fig. P1.1 - Heat flux on length of the calculated area generating line (bottom and vertical borders)

As follows from table P1.1, the calculation of thermal loading carried out in the present paper according to the model of effective thermal conductivity conforms sufficiently well to the experimental data and calculation on the basis of modified k - ε model. Values of heat flow density on the bottom wall coincide practically with experimental values, and on lateral and top borders of area they exceed experimental values by 16 and 28 % respectively, providing more conservative estimations of thermal loading.

P1.4 Conclusions

1) Testing of computer model for approximate analysis of heat exchange in the core melt pool on the reactor vessel bottom in the conditions of severe accident is presented. Simulation of experimental data received during study of strongly developed free turbulent convection at the known Finnish installation COPO was carried out. It was shown, that results of calculations by means of code NARAL/FEM conform well both to experimental data, and to calculated data of

other authors received within the framework of detailed simulation by means of differential k - ε turbulence models.

The suggested approach, combining a model of effective thermal conductivity and the equation of component concentration transfer, differs from developed earlier ones:
- in simplicity of realization without solving the equations of motion and turbulent characteristics transfer (for example, with reference to k - ε -model of turbulent transfer, kinetic energy of turbulence and dissipation of turbulent energy);
- in correct account of turbulence at high Rayleigh numbers;
- in more exact calculation of processes of oxide crusts formation and melting in multicomponent corium melt;
- in consideration interconnection of thermal processes in the melt pool and in the reactor vessel;
- in realization of models within the scope of final element method that allows to apply them successfully to any complex geometry and to adapt easier the technique at carrying out interfaced thermal and strength calculations;
- in carrying out calculations in two-dimensional statement.

2) As a whole, taking into account complications of calculations using differential models of the turbulence, the considered approach of the account turbulent convection using effective thermal conductivity model is reasonable at carrying out calculations of a thermal condition of nuclear power plants constructions filled with the core melt.

3) The considered method of heat exchange calculation can be used for analysis a thermal condition of corium pool with reference to VVER-type reactor vessel in the conditions of severe accident.

Appendix 3-1

3. Task “B”: development and manufacturing of the experimental test facility and supporting systems for the VVER scale vessel models testing

3.1 General description and technical features of the experimental test facility IVRSA-VVER

Experimental test facility IVRSA-VVER (Investigation of the Vessel Rupture during Severe Accident on the VVERs) is intended for thermomechanical tests of scale models of the VVER reactor vessels (PWR and similar to it) in the conditions simulating a stage of severe accident (SA) in reactor installations with VVER, when the molten fragments of the core accumulate in the lower head of the reactor vessel (LH of the RPV).

The objective of the models testing is study of thermal and mechanical features of behaviour and failure of the vessel in the conditions of high-temperature creep of the RPV steel.

Heating of tested models should be carried out by means of the special internal electric heater placed inside the vessel model. The maximal electric power of the heater will reach 200 kW. Further the increase of electric power up to 400 kW is planned.

Heating of the model is carried out on the inner surface of the vessel owing to thermal radiation from a heater surface («a dry variant»), or by means of the melt simulator. At this stage of investigations it is supposed, that heating of the vessel model will be carried out owing to thermal radiation.

Power loading of the vessel model is carried out owing to internal overpressure, created by the gas medium (argon, etc.).

The experimental test facility consists of a protective casing, an auxiliary premise for placing of auxiliary systems, system of the model heating, water and gas systems, video monitoring system, and also data acquisition system (DAS) (fig. 3.1.1, 3.1.2). The sketch of the test facility basic schematic diagram is represented in fig. 3.1.3.

Dimensions of tested models have the following parameters:
- diameter: 300-1000 mm;
- height: up to 2500 mm;
- weight: up to 2800 kg.

The weight of the experimental test facility is ~13 tons, and dimensions are ~4900x4800x4700 mm.

Characteristic features of tests to be carried out at the given experimental test facility, are:

a) high temperatures of the investigated construction heating: up to 1900ºC;
b) value of radial and longitudinal deformations of the vessel: up to 150 mm;
c) duration of tests: 0.5 - 70 hours;
d) maximal overpressure: up to 8 MPa.

The DAS system of the experimental test facility should provide:

- a required mode of the tested model heating owing to control of electric power supplied to the model's heater;
- control of the overpressure level in the vessel model owing to admission/discharge in it of the gas medium from available cylinders;
- controllable level of the cooling water supplied to the cooling systems of the heater and the top part of the model construction.

The following parameters shall be registered in the course of tests in a continuous mode:

a) temperatures of the vessel outer surface:
b) temperatures of the vessel inner surface:
c) movings of the model outer surface:
d) parameters of auxiliary systems condition (electric, gas, water systems):

Construction of the experimental test facility consists of the basic protective casing and auxiliary housing. The protective casing (housing) is a welded metal construction in which lateral panels and shutters of the cover can be opened in the course of the experiments carrying out (see Fig. 3.1.2).

Auxiliary systems of the experimental test facility (gas, electric, water systems) are located in the auxiliary housing. The electric system consists of two dry-type power transformers (OSU-100) and devices controlling them. Electric load from transformers is transferred to the model's heater by means of aluminium conductor run and flexible copper current-carrying arrangements (fig. 3.1.4).

Now the construction of the basic protective and auxiliary housings has been completed and assembling of auxiliary (gas, electric and water) systems is carried out.

It is planned, that Creation of experimental test facility will be completed by October 2009.

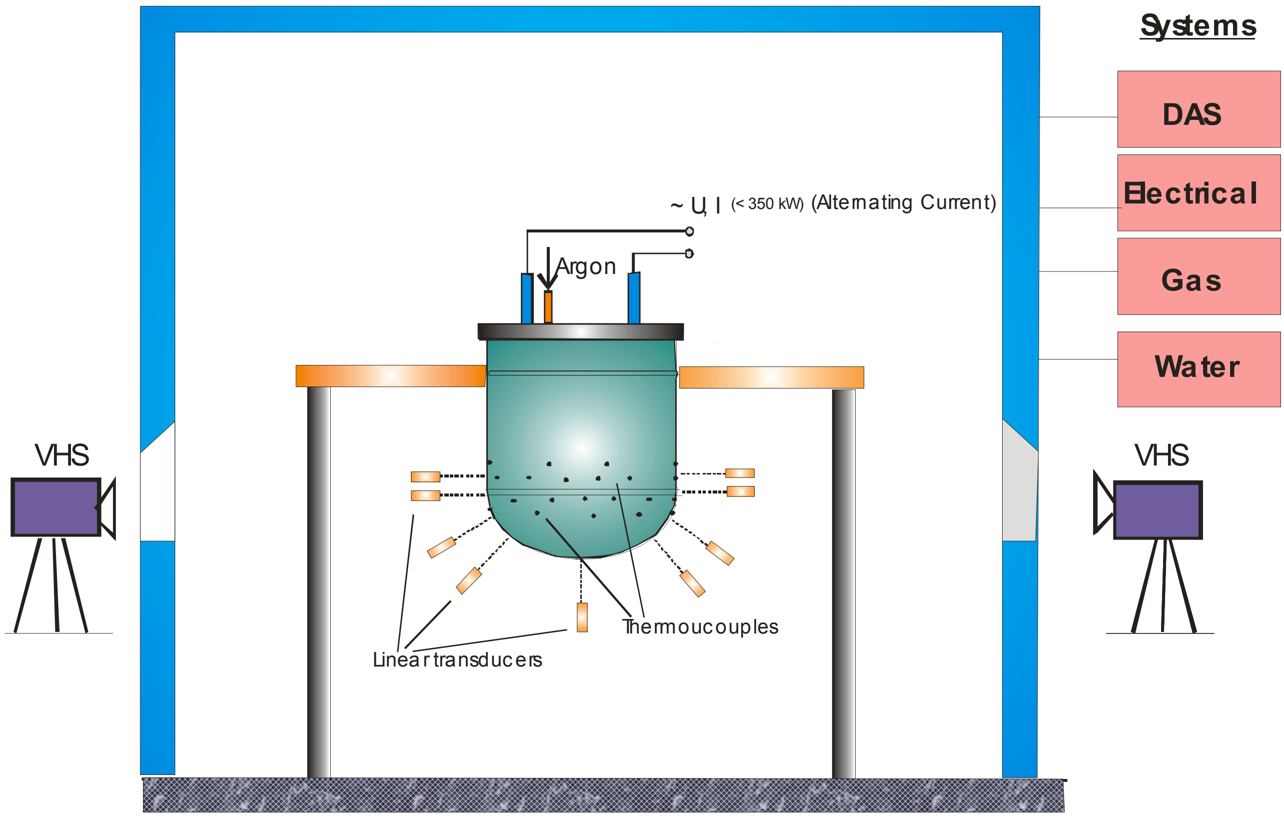


Fig. 3.1.1 - The sketch of the IVRSA test facility and auxiliary systems

Fig. 3.1.2 - the protective casing and auxiliary housing of the test facility

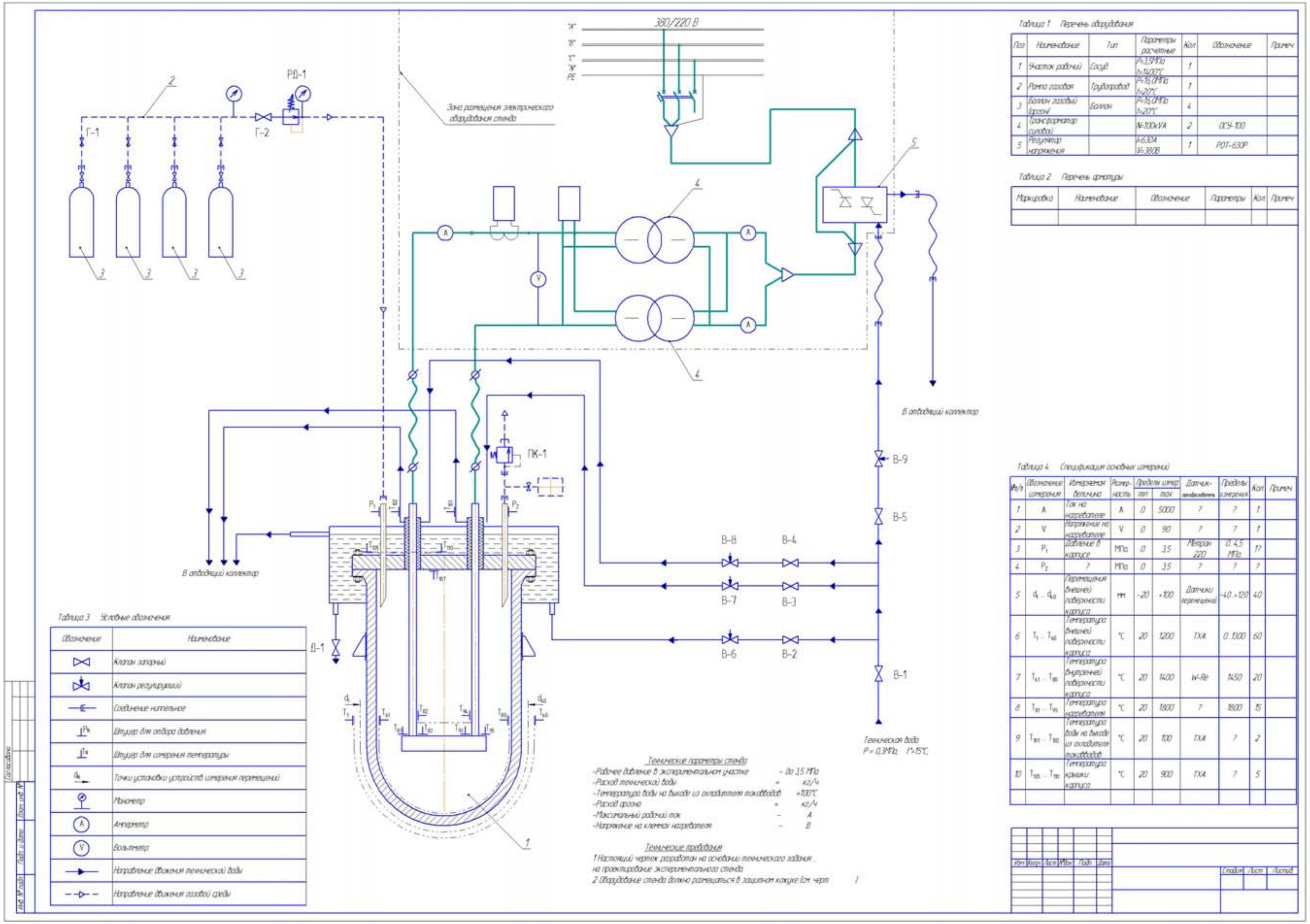


Fig. 3.1.3 - The sketch of the IVRSA-VVER test facility

Fig. 3.1.4 - The power transformers installed in the housing and arrangements for their control

Appendix 3-2

3.2 General description and technical features of automated scientific data acquisition and control system (DAS) of IVRSA-VVER test facility

Object of automation and control is:

- the experimental test facility IVRSA-VVER (Investigation of the Vessel Rupture during Severe Accident on the VVERs).

This experimental test facility is meant for the scale vessel models thermomechanical tests under conditions simulating the severe accident (SA) stage in reactor installations with VVER, when the inner surface of the vessel is exposed to integrated thermal-powered impact of molten core fragments under overpressure in the vessel.

The heating of the tested models shall be performed by means of a special 400 kW electrical heater, placed inside the vessel model. Overpressure in the vessel model shall be provided with gas medium supply with required parameters into the inside cavity of the model.

The following parameters shall be registered in the course of tests in the continuous mode (sampling rate no less than 20 s^{-1}):

a) temperature of the vessel outer surface:

- number of channels: up to 120;
- temperature range: 20-1350^{o} C;

b) temperature of the vessel inner surface:

- number of channels: up to 50;
- temperature range: 20-1800^{o} C;

c) displacements of the model outer surface:

- number of channels: up to 60;
- range: -30-120 mm;

d) overpressure value in the model and auxiliary systems:

- number of channels: 2;
- range: up to 7 MPa;

e) parameters of auxiliary systems status (electric, gas, water systems):

- number of channels: up to 6;

f) control system of the experimental test facility shall provide:

- a required mode of the tested model heating owing to control of supplied electric power to the model's heater;
- control of the overpressure level in the vessel model owing to gaseous medium inlet/outlet from available cylinders;
- required level of cooling water supply to the heater's cooling systems and the model construction top part.

Accuracy of measuring channels shall be 0,2% or better.

DAS has a hierarchical structure composed of the lower and upper levels:

– hardware of lower level (distributed data acquisition stations, sensors, secondary transducers, “dry contacts” of technological equipment status etc.);
– hardware of upper level (experimenter's automated workstation (AWS), communication means).
– hardware of lower level (distributed data acquisition stations, sensors, secondary transducers, “dry contacts” of technological equipment status etc.);
– hardware of upper level (experimenter's automated workstation (AWS), communication means).

Communication among the hardware of upper and lower levels is performed via Ethernet at 10/100 Mbit/s exchange speed.

DAS and control system shall survive under conditions of essential electromagnetic interference and ensure the required accuracy of parameters measurement at great length of communication lines with sensors (up to 100 m).

Measuring and control channels of DAS and control system shall have galvanic uncoupling. The system construction shall ensure simplicity of connection and disconnection of communication lines and power supply for its portability.

DAS is developed based on modern soft- and hardware and technologies ensuring the acquisition, processing, transmission, archiving and presentation of operational data. Suggested program-engineering solutions were given on the basis of non-full initial data and information scope on the automation object and can be of a non-full character.

DAS has a hierarchical structure composed of the lower and upper levels. Structural diagram of the DAS is represented in fig. 3.2.1.

DAS consists of the following levels:

- hardware of lower level (distributed data acquisition stations, sensors, secondary transducers, “dry contacts” of technological equipment status etc.);
- hardware of upper level (AWS, communication means).

Communication among the hardware of upper and lower levels is performed through Ethernet at 10/100 Mbit/s exchange speed.

Number of input/output channels is represented in Table 3.2.1.

A suggested soft- and hardware complex consists of distributed data acquisition stations ADAM-5000/TCP of Advantech Company.
Modules ADAM-5017 is used as analog input modules.
Modules ADAM-5018 is used as a thermocouples input modules.
Modules ADAM-5024 is used as an analog output modules.
Modules ADAM-5055S and ADAM-5060 are used for input and output of discrete signals.

Table 3.2.1. Number of DAS channels

Type of channel	Number
AI thermocouples AI current/voltage	126 48
AO	4
DI	8
DO solid-state	8
DO relay	6

By the present moment, DAS system manufacturing has been completed.

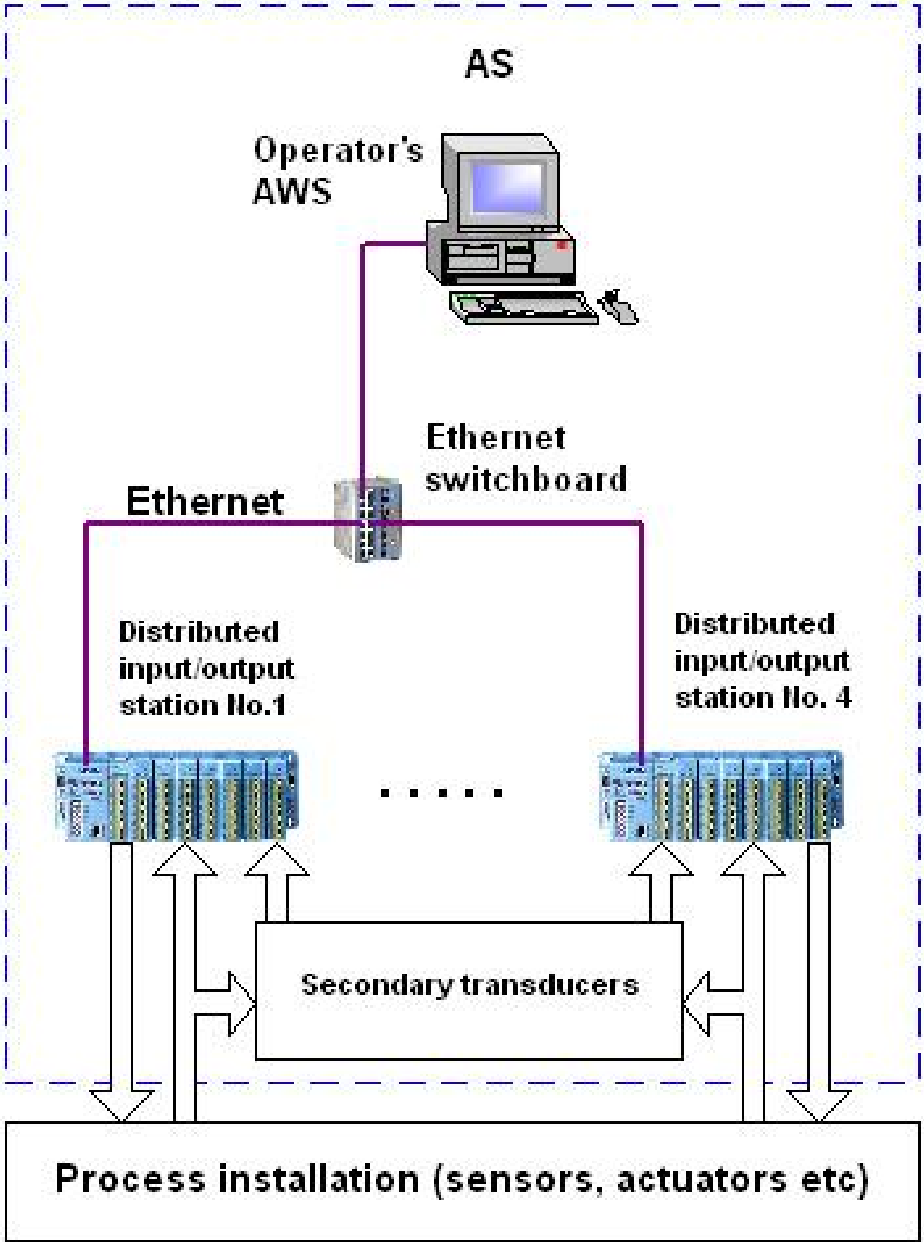


Fig. 3.2.1 - Structural diagram of DAS

Appendix 3-3

3.3 Task "B": Experimental VVER-440 vessel scale model

The object of investigations in the present project is the model of the VVER-440 reactor vessel lower head, made on scale ~1:5. At that, the material and technological features of scale models manufacturing should correspond as much as possible to the manufacturing technique of original VVER vessels.
First of all, the following is meant herein:

a) the material of the vessel model should be of the original VVER vessel steel;

b) the technique of manufacturing of the vessel model components (manufacturing of forgings for steel bars of cylindrical course and the bottom), their welding and heat treatment should correspond to standard technique of the VVER vessels manufacturing.

The construction of the vessel model represented in fig. 3.3.1 has been developed as an initial variant (variant "No. 1"). In this construction the cylindrical course, the bottom and the flange of the vessel model have been made of the VVER vessel steel. As initial steel bars for the model components (course, bottom and flange) forgings of this steel are used. On top of the vessel there is a cover in which passing channels for current-carrying arrangements and communications (the thermocouples, gas supplies, etc.) are located.

For hermetic sealing of a flange joint between the vessel and the cover using of the expanded graphite or a nickel gasket is supposed. In order to maintain a required mode of cooling of the cover and the top part of the model it was decided to use water cooling (fig. 3.3.1).

A construction of the vessel model, presented in fig. 3.3.1, was developed as a base variant.
In connection with economic conditions which have developed in Russia in the autumn 2008-winter 2009, manufacturing of the given model construction has been suspended. Works on continuation of this model construction manufacturing can be begun not earlier than autumn of 2009, or can be shifted for later terms.

In addition, under the program of investigations manufacturing of 4 vessel models was planned, but owing to a drastic rise of prices for the vessel steel and cost of its processing at manufacturing of models, a problem concerned with performance of this stage of work arose.

To solve this problem efforts as on search of additional financial resources necessary for manufacturing of the scale VVER vessel models and to develop a simplified constructions of the model have been made.

Such simplified construction of the model would allow to reduce cost of manufacturing both owing to reduction of the vessel steel weight, and owing to the construction embodiment. As a result of the carried out work variants of the vessel model presented in fig. 3.3.2-3.3.3 have been developed.

Unlike a basic construction (fig. 3.3.1) in these constructions the VVER vessel steel is used only for those components of the construction (cylindrical course, the bottom) which will be investigated during tests. Less critical components of the construction (the flange, the bottom) are supposed to be made of a cheaper steel (Steel 20, 22K).
Besides, the vertical sizes of the model variants No. 2 - No. 4 are much less than corresponding sizes of construction No. 1, that also reduces total weight of the VVER vessel steel in the model construction. Use of this type of constructions will allow to reduce cost of the vessel models manufacturing more than by 40 % in comparison with a basic construction (fig. 3.3.1).

Because continuation of work on manufacturing of this vessel model construction has been suspended, considerable efforts were made to solve this problem.

In particular, it was decided to start development and manufacturing of a simplified scale model construction (Var. N4, fig. 3.3.3). It was decided to use a flat bottom in this construction to simplify and reduce the price of the model manufacturing. This decision is justified because it is planned in the scheduled experiment to study heating and deformation of the model cylindrical part (course) (fig. 3.3.4). In this vessel model the VVER steel (15Kh2NMFA) is used for the cylindrical course of the vessel. The bottom of model will be made of the Steel 20 or 22K.

As a result of the carried out work it was possible to find additional sources of financing within Russia for manufacturing of **one** scale vessel model (Var. No.4) (fig. 3.3.3). By now preliminary work has been carried out and manufacturing of the first vessel model has been started.

The billet from 15Kh2NMFA steel is used for the manufacturing of the vessel model (fig. 3.3.5-3.3.7). After mechanical treatment and welding of the vessel model course with the bottom, the thermal and final mechanical treatments will be carried out.

Planned date of the model manufacturing completion: 10-th quarter (the April, 2010). Therefore, in the given direction of work the time delay at least for 6 months is expected.

Now search of necessary financial assets for manufacturing of the 2nd and the 3rd vessel models is carried out. The search is carried out both inside, and outside of Russia.

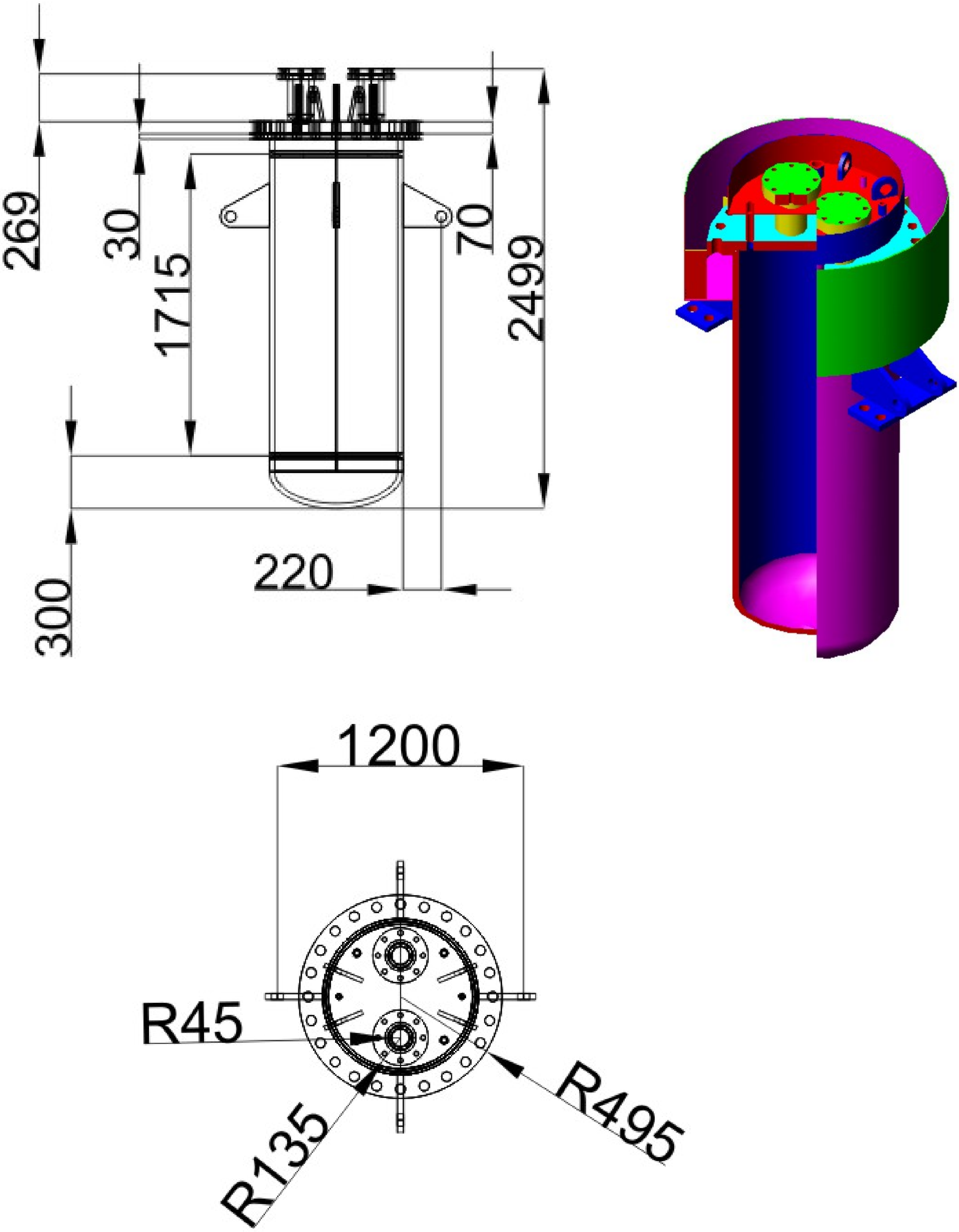


Fig. 3.3.1 - Variant No1 of the VVER vessel model construction (Variant “Base”)

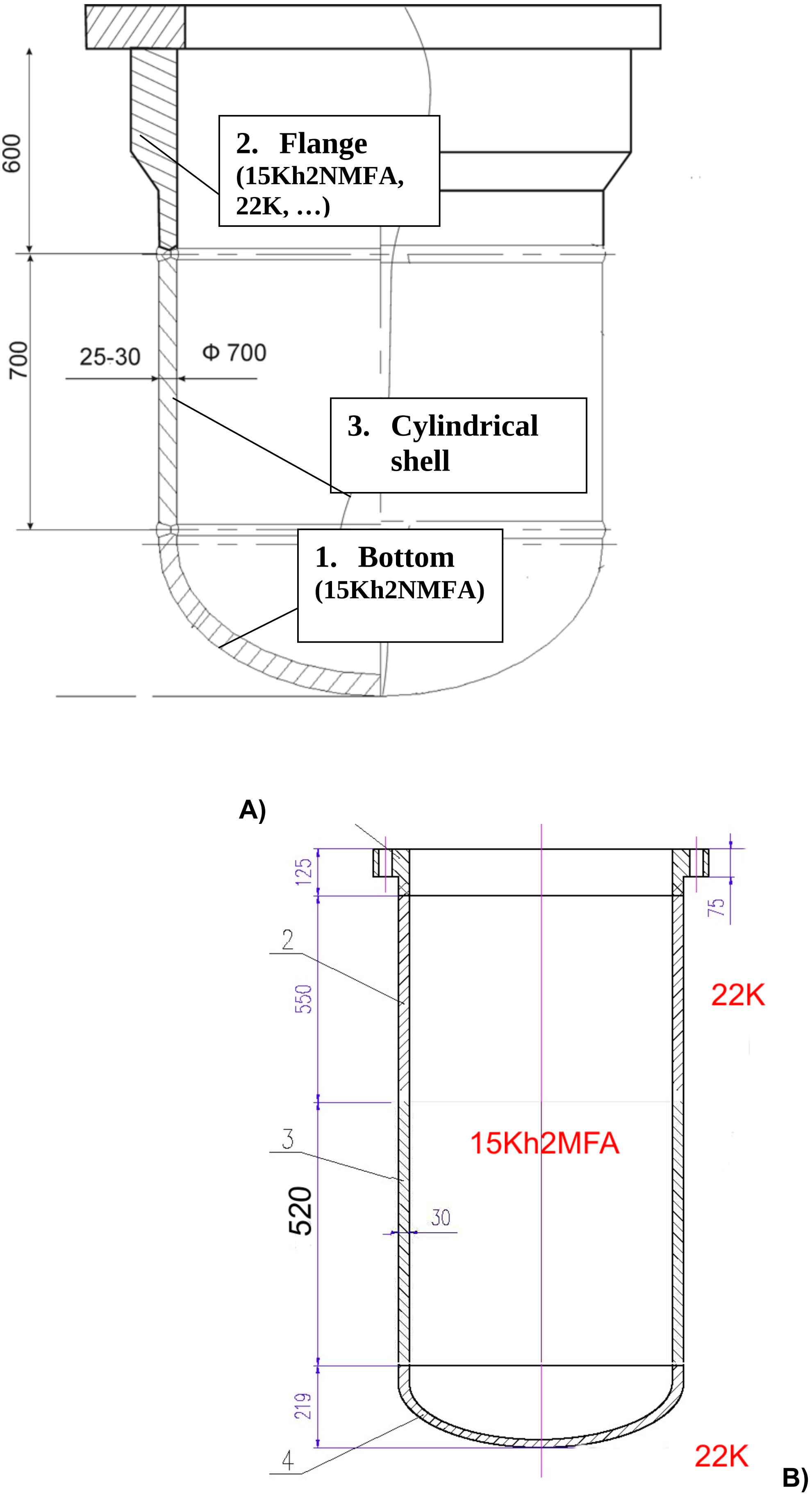


Fig. 3.3.2 – Variants «2» (A) and «3» (B) of the VVER vessel model construction

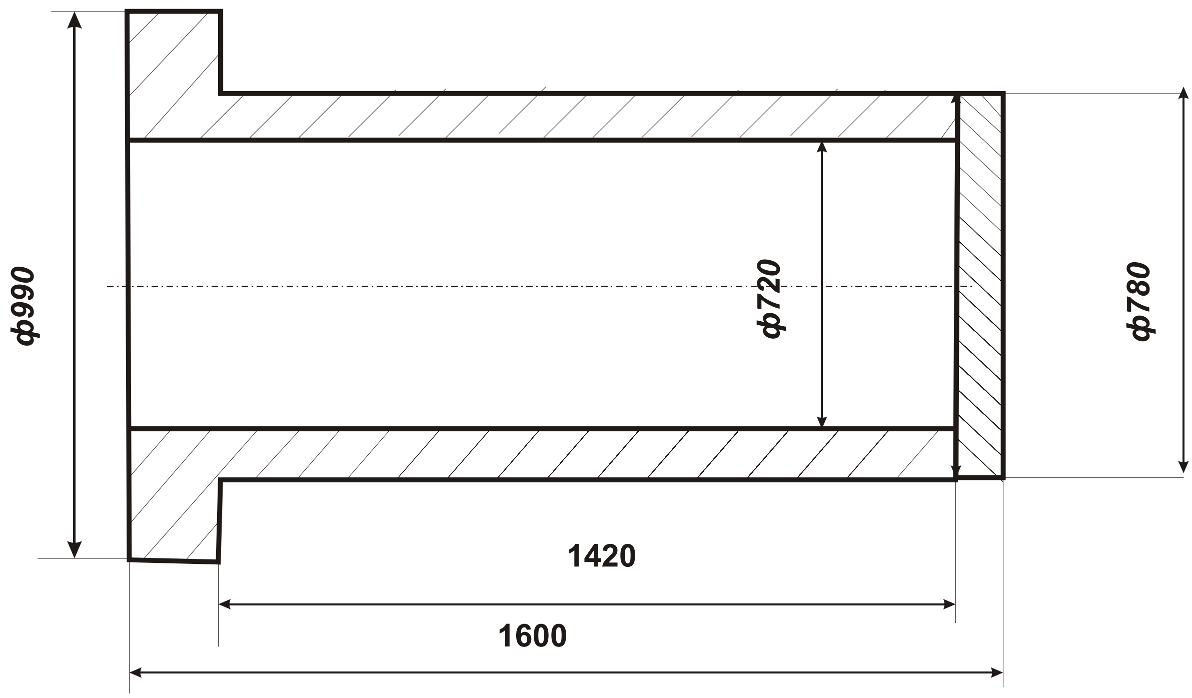


Fig. 3.3.3 – Variants «4» of the VVER vessel model construction

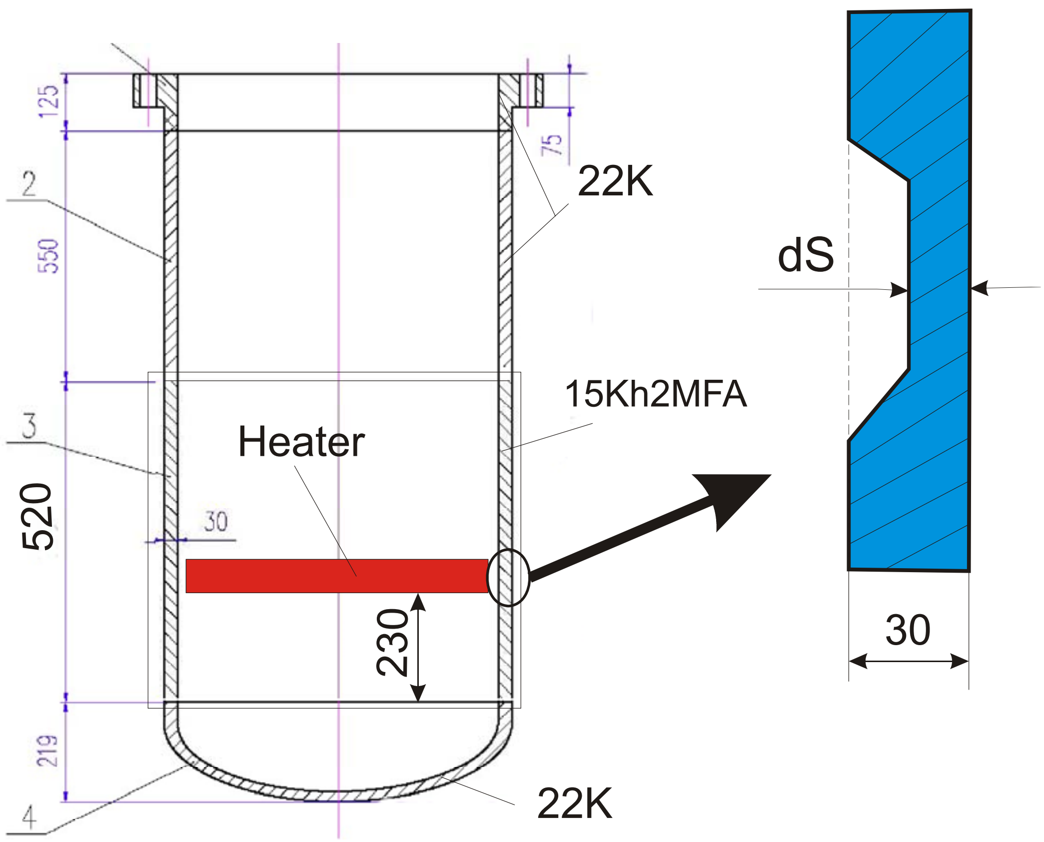


Fig. 3.3.4 – Scheme of heating of the vessel model cylindrical part

Fig. 3.3.5 – unloading of the billet from the 15Kh2NMFA steel

Fig. 3.3.6 – Preparation and machining of the vessel models billet

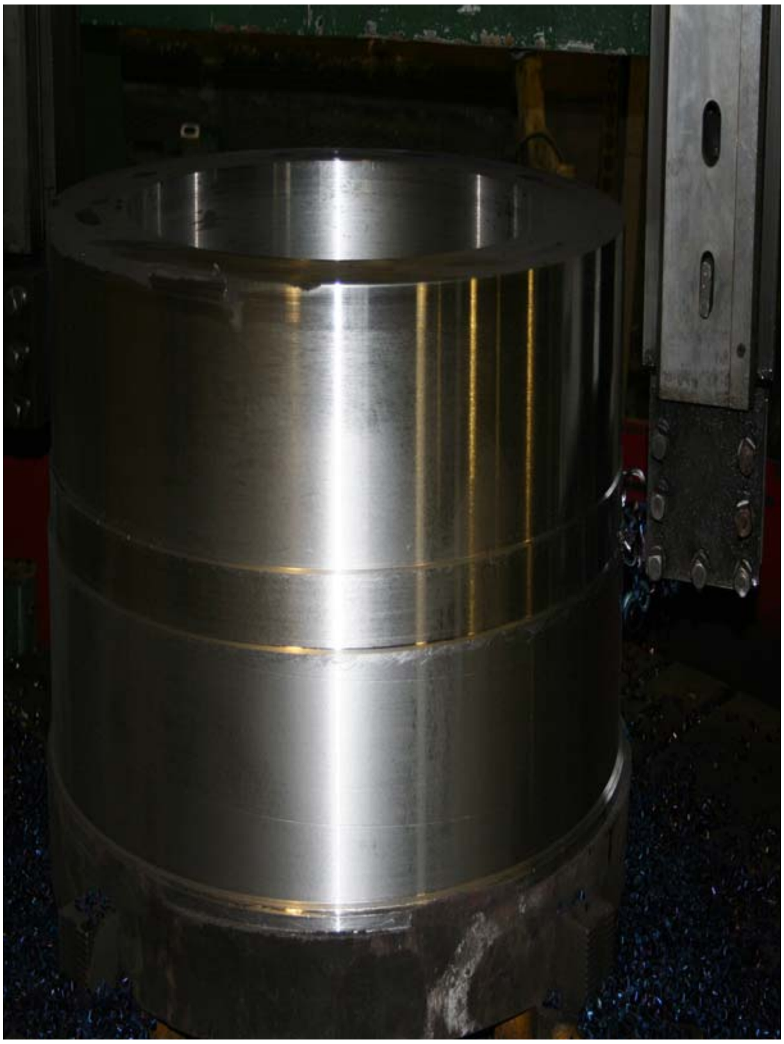


Fig. 3.3.7 – Preparation and machining of the vessel models from the billet

Appendix 3-4

3.4 Task "B": Heater of the IVRSA-VVER test facility (4)

For heating of experimental vessel models it is planned to use a electrical ring heater (represented in fig. 3.4.1). At a similar circuit of heating, a local heating of the vessel is realized, which corresponds to the case of the melt stratification in SA, when a layer of overheated steel is formed in the upper part of the melt pool. Such stratified structure of the melt pool leads to the intense heating of the vessel in the area of contact of the steel melt level with the vessel wall (fig. 3.4.2).

For the heating it is planned to use 2 power transformers with the total capacity up to 200 kW which have already been assembled with corresponding control systems in auxiliary housing. Further, if necessary, it will be possible to assemble additional power transformers with the total capacity up to 400 kW. Working medium inside the model vessel is argon.

Several perspective materials are considered as heating elements:

- spiral heating elements of molybdenum or tungsten. The heater is produced as of spirals bundle which were made from a molybdenic or tungsten wire;
- graphite heating element in the form of a ring;
- tungsten-molybdenic heater.

At the present time, the construction and the main parameters of the electrical heater heater are determined and the main elements and nodes of the construction are produced. The main nodes of the heater construction are represented in fig. 3.4.3-3.4.5.

The electrical heater consist of the following constructive elements:

- spiral heaters made of tungsten (molybdenum) wire of 1 mm diameter (fig. 3.4.3);
- water-cooled current-carrying arrangements, which consist of an upper (fig. 3.4.4) and a lower (fig. 3.4.5) parts.

Final assembly of the heater will be carried out at final installation and assembling of the vessel scale model.

3.4.1 Testing of the heating elements at small-scale experimental test facility

In order to choose the type, construction and thermal and physical parameters of the heating elements to be used in the heater during reactor vessel models testing, the small-scale experimental test facility was developed (Fig. 3.4.6).

This construction is a cylindrical vessel with diameter and height of 220 and 1200 mm, respectively. Thickness of the vessel wall in the working area is 9 mm. The heating elements to be tested are placed inside the vessel by means of special cooled current-carrying arrangements.

Gas medium filling the inside vessel cavity of the test facility during test is argon.

The brief description of this installation is presented in ISTC annual report under the project №3635 (2008).

For the purpose of check of heating elements operability which will be used in the heater of the vessel model, the tests of the 1-st and 2-nd series of spiral heaters have been carried out. Spiral heaters were made of molybdenum (1-st series) and tungsten (2-nd series) wire of 1 mm diameter.

Besides, objectives of the carried out tests was the following:

- determination of electrophysical properties of heating elements at high temperatures;
- determination of heat flow density on the wall of experimental test facility;
- life tests of heating elements.

To obtain thermal and physical and electrophysical parameters of heating elements, thermocouples (W-Re) were fastened on the outer of cylindrical course of the test facility, as well as on its inner surface and on spiral heating elements. Some fragments of tests are presented in fig. 3.4.7, 3.4.8.

In these photos (fig. 3.4.7) the conditions corresponding to various levels of electric power (P) supplied to the heater are presented (Max P < 14 kW).

At that, value of the heat flux on the inner wall of the vessel model of about 0.37 MW/m2 has been reached. The further increase of supplied electric power was unreasonable owing to that it was impossible to provide corresponding heat removal from the outer surface of the heater model by air.

At carrying out of life tests of heating elements, depressurization of cooled current-carrying arrangements has occurred and water has got to internal volume of the device. It has led to melting and destruction of heating elements (fig. 3.4.8.).

The main conclusions from the carried out tests:

1) heating elements of molybdenum provide a necessary mode of heating (heat flux on the vessel model wall) which should be realized at test of scale vessel models;

2) constructive improvement of junctions of experimental test facility is required for tests of heating elements.

At present, the following activities have been carried out or are at the stage of performance:

a) improvement of cooled current-carrying arrangements has been carried out;

b) the external casing for providing of water cooling of the vessel model is manufactured that will allow to carry out tests at higher electric power supplied to the heater;

c) tests of heating elements of tungsten are under preparation;

d) tests of heating elements of a lamellar design of molybdenum alloy are under preparation.

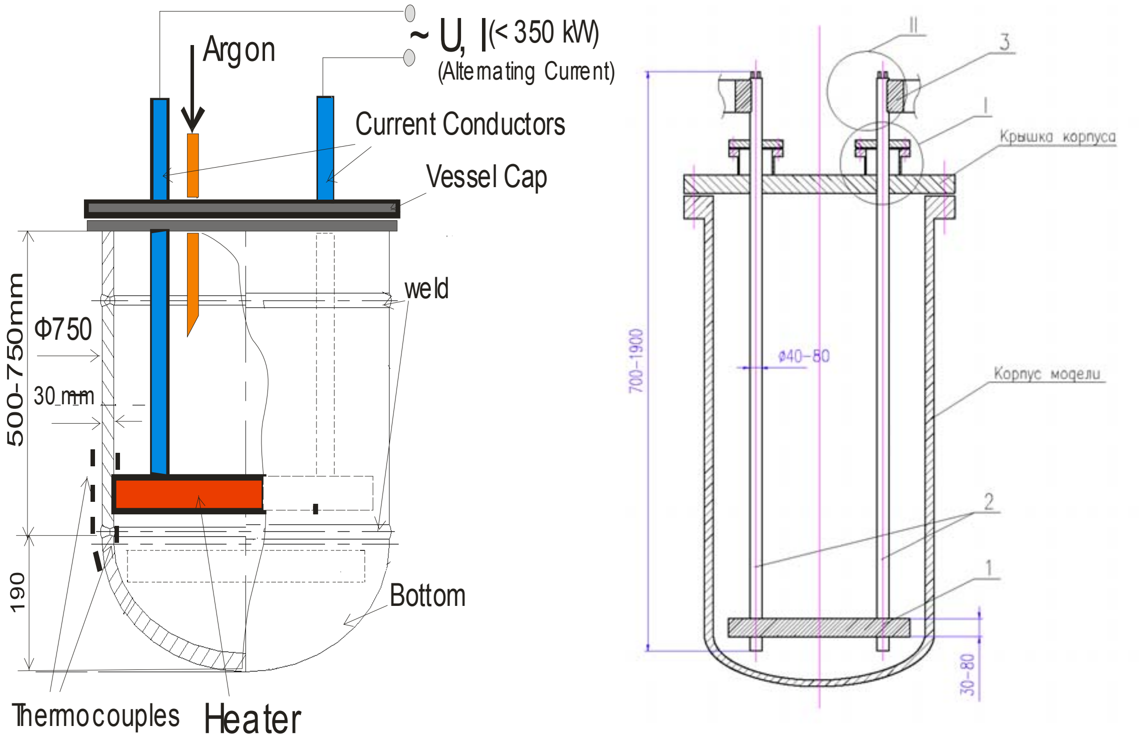


Fig. 3.4.1– The sketch of a vessel model heater

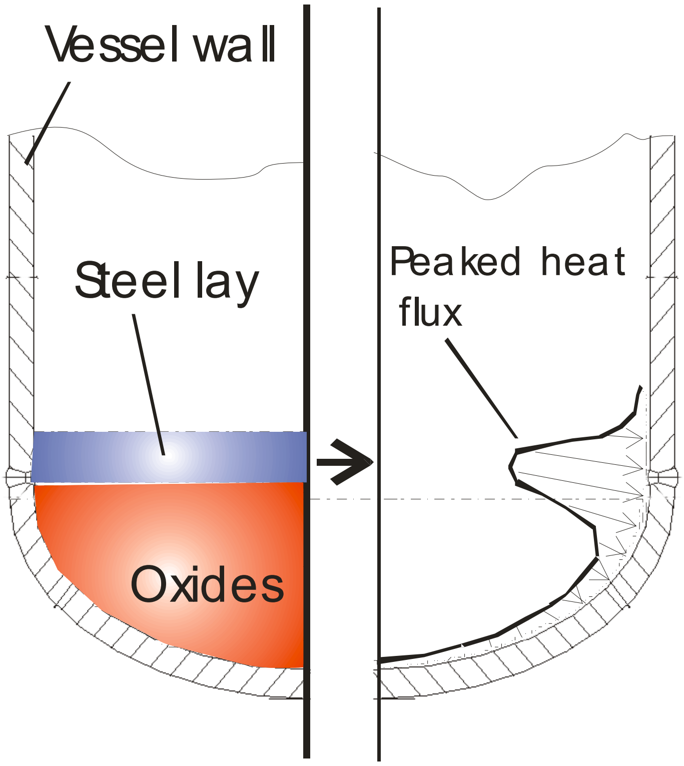


Fig. 3.4.2– The scheme of the heat flux distribution on a inner surface of the RPV vessel wall during severe accident

Fig. 3.4.3 – Spiral heaters

Fig. 3.4.4 – The upper part of the heater current-carrying arrangement

Fig. 3.4.5 – The lower part of the heater current-carrying arrangement

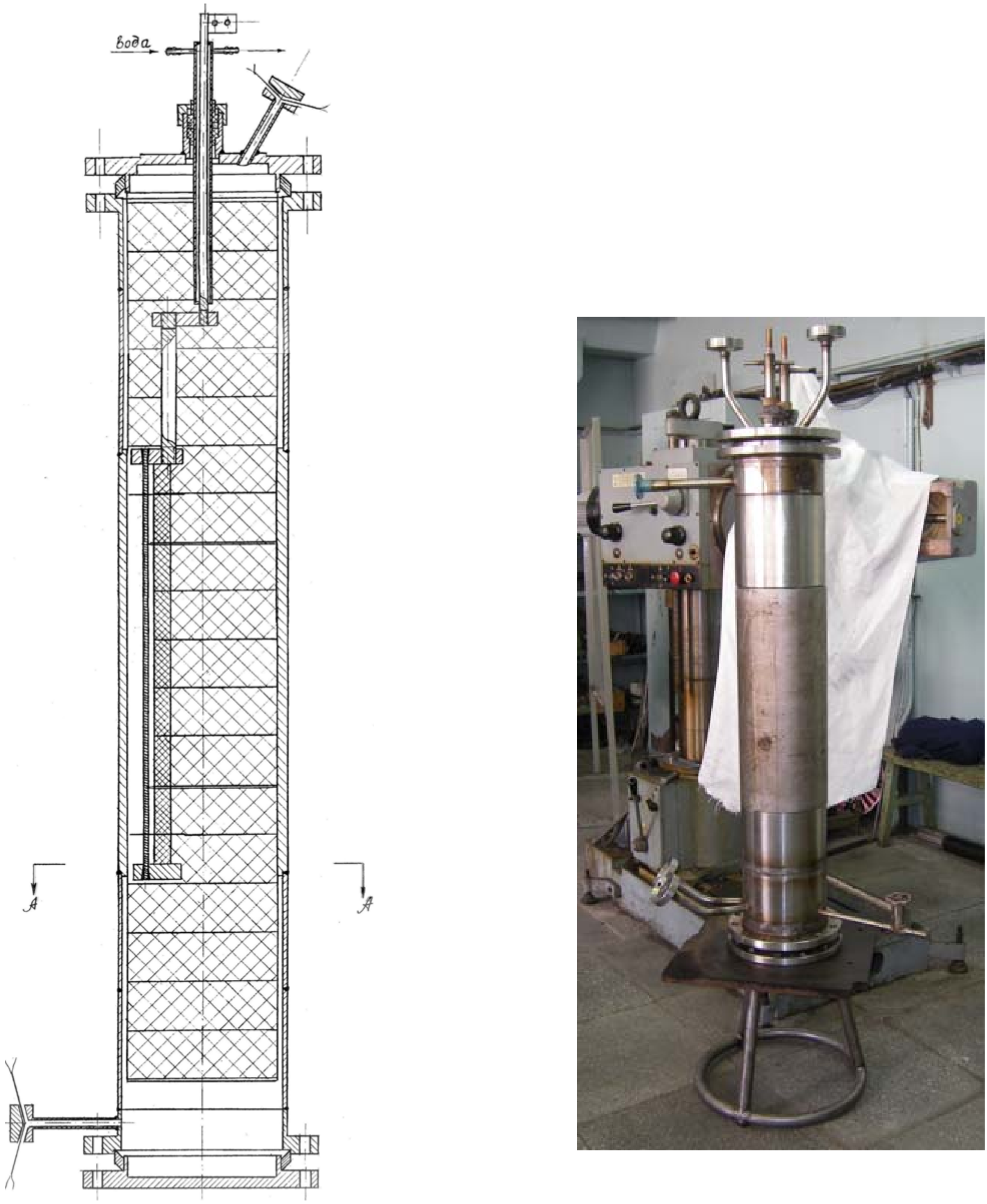


Fig. 3.4.6 –Small-scale experimental test facility

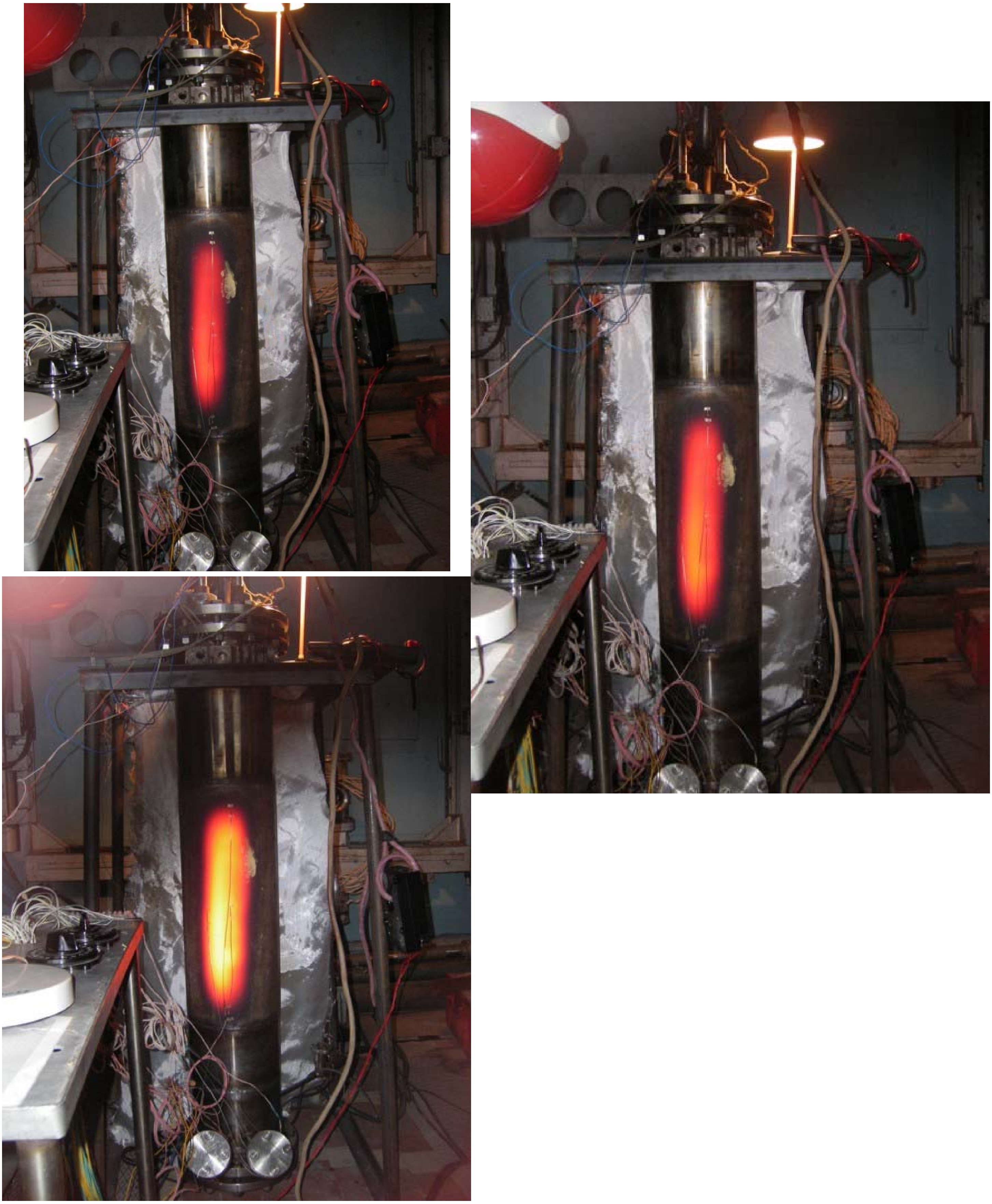


Fig. 3.4.7 – Fragments of the heater element testing

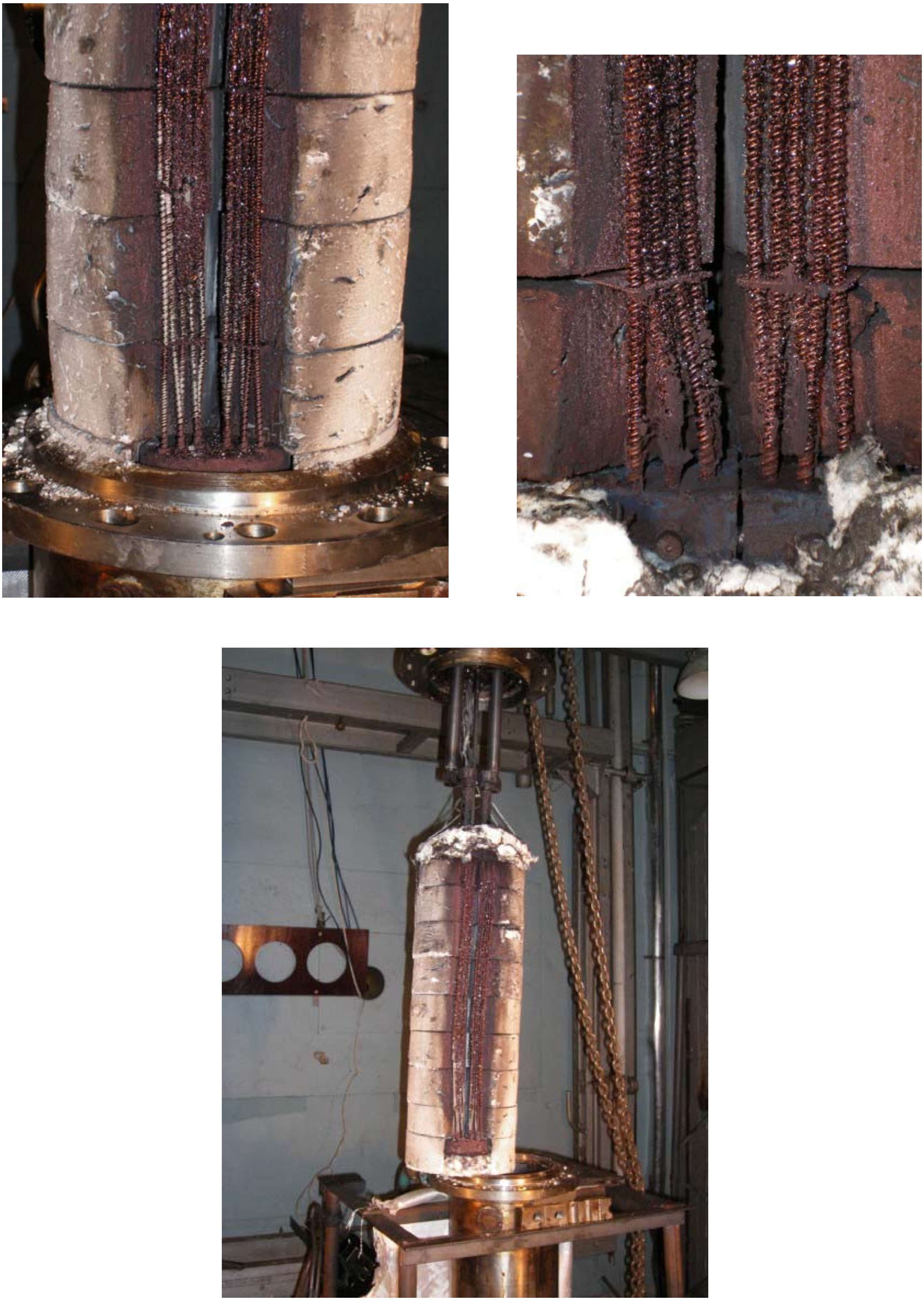


Fig. 3.4.8 – State of the small scale experimental test facility after tests

Appendix 4

4. Problem “D”: Determination of creep parameters and short-term mechanical characteristics of the VVER vessel steel 15Kh2NMFA

4.1 Initial data

A vessel steel 15Kh2NMFA (TS 108-765-78) of which the lower course of the VVER vessel No. 1152.02.70.000 (number of fusion 107210), number of forgings 5015 has been made was the material which mechanical properties were subject to definition in process of material property studies.

The specimens fabricated for the tests had circular cross-section of 8 mm diameter and working length of 45 mm. The specimens were made in two variants: without a thread part and with a thread part for fastening in pinchers of the testing machine (fig. 4.1).

4.2 Procedure of tests carrying out. The basic results of tests

Heating of specimens was carried out in furnace space (the air environment or argon) with duration of heating up to going out at a set mode from 45 till 60 minutes. The temperature was kept in the course of tests with 2 ^{0}C accuracy.

The creep tests of the investigated steel in a temperature range 750-1050 ^{0}C were carried out on the time base up to 50 hours. Creep experiments were carried out in the protective argon medium at constant level of loading.

The results of the carried out creep tests of the investigated steel 15Kh2NMFA at different values of temperature and stress are represented in fig. 4.2-4.8. Tests were carried out, as a rule, in the protective medium of argon, but some modes of tests were carried out in the air medium.

It is necessary to note some features of investigated steel behaviour in the studied range of tests:

1) absence of protective medium leads to considerable hardening of material and slowing-down of deformation process at creep of this steel. It is connected, first of all, with

formation of hard oxide film on the tested specimens surface;

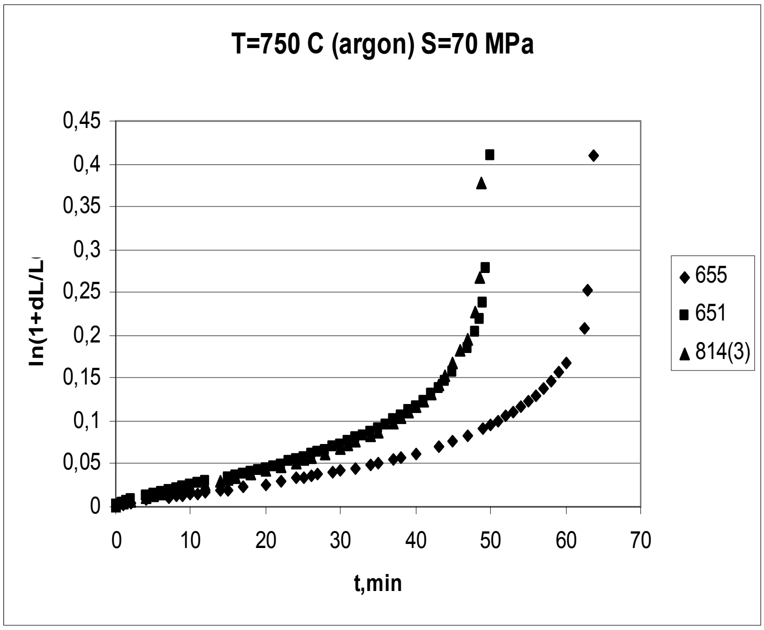
2) with increase of the process duration, deformation at the moment of the specimens destruction considerably increases.

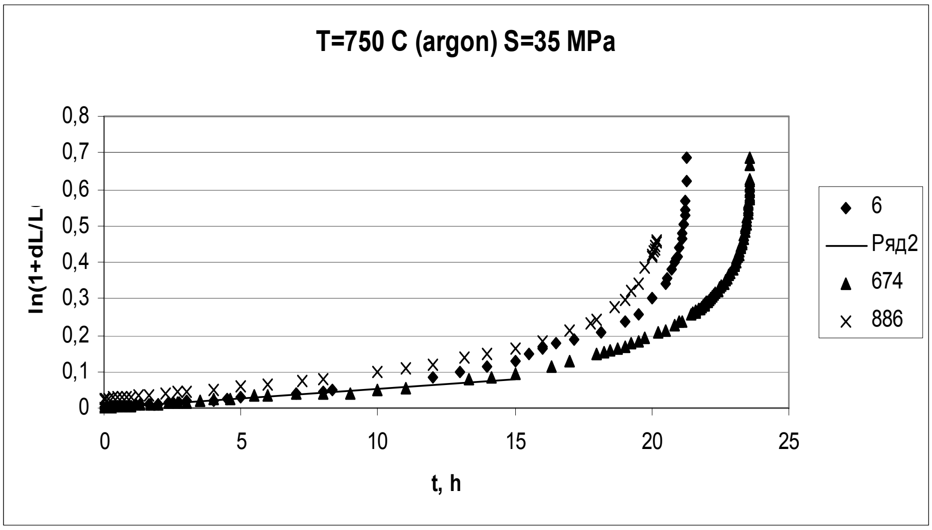
Now preparation of the additional series of creep tests at temperatures above 850 C in vacuum is performed.

The main results of these tests are presented in the below mentioned diagrams.

Fig. 4.1 – Specimens for mechanical tests of vessel steel

T= 750C





$n = \mathbf{4{,}17}$

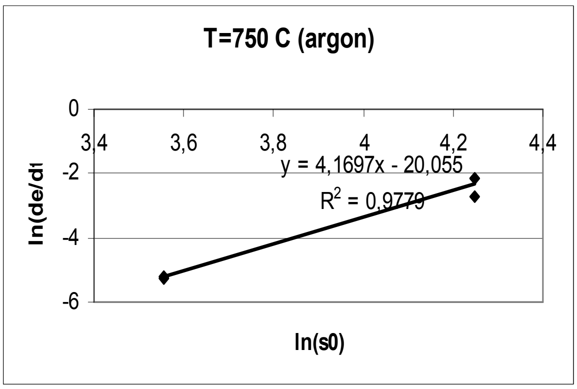
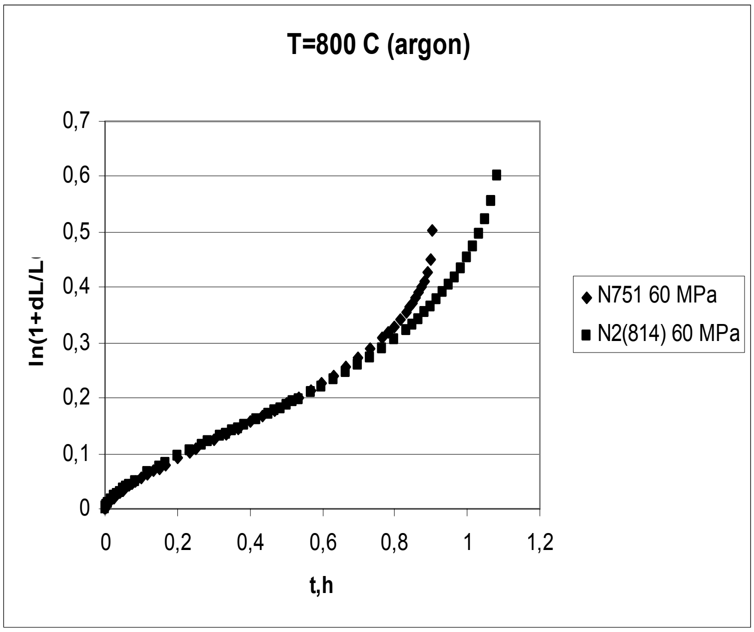
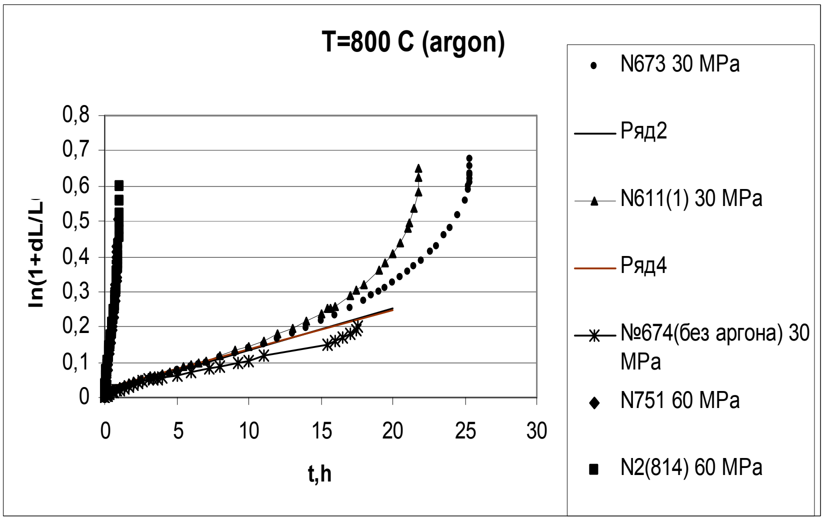


Fig. 4.2 – Diagrams of creep deformation at 750 ^{0}C

This work is supported financially by <Funding Party> and performed under the contract to the International Science and Technology Center (ISTC), Moscow.

T=800C





$n = \mathbf{4{,}837}$

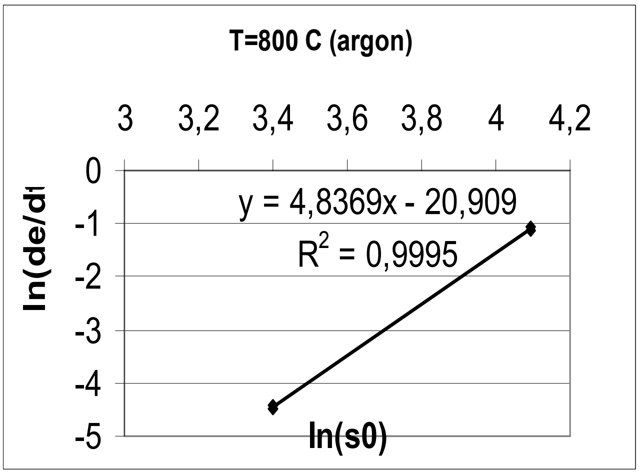
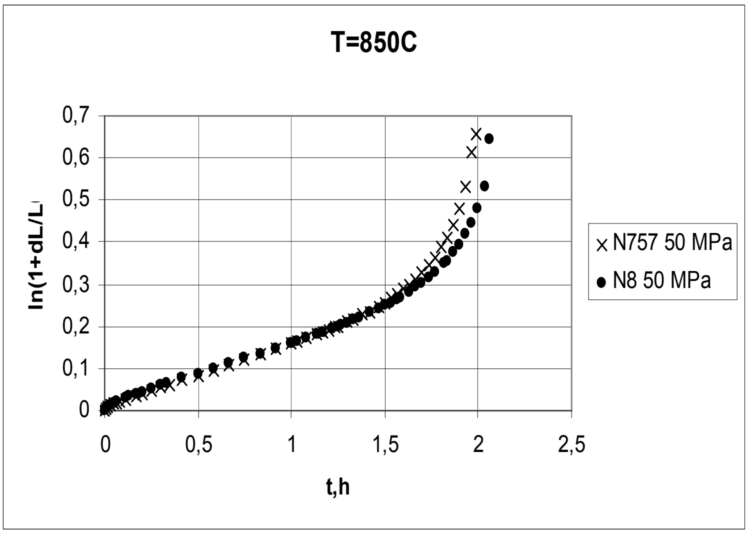
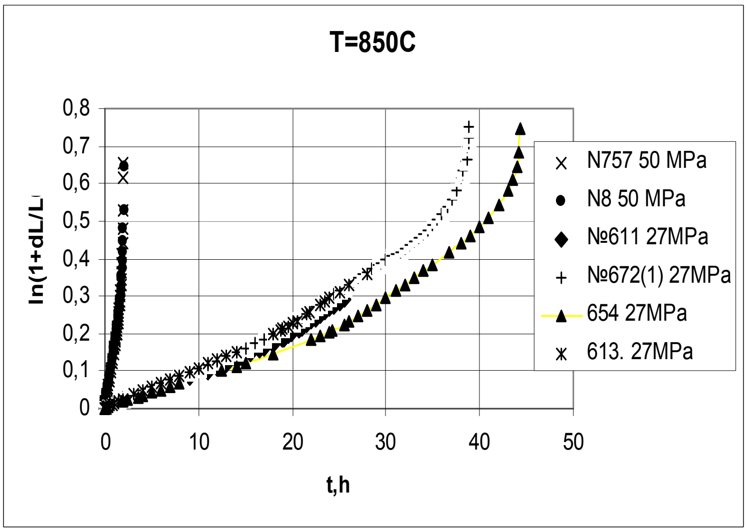


Fig. 4.3 – Diagrams of creep deformation at 800 ^{0}C

T=850C





$n = \mathbf{4{,}319}$

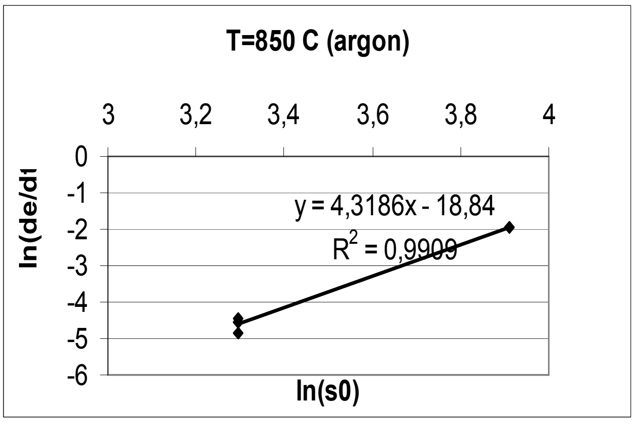
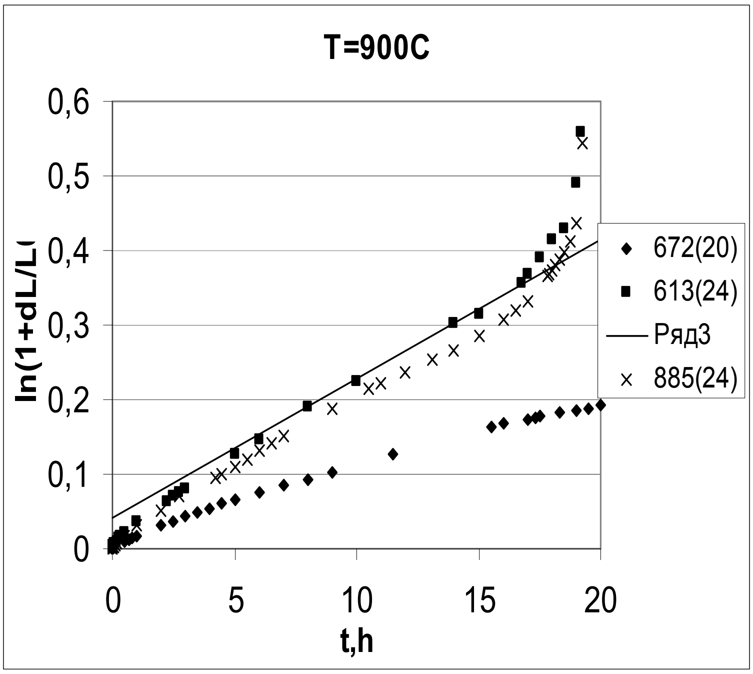
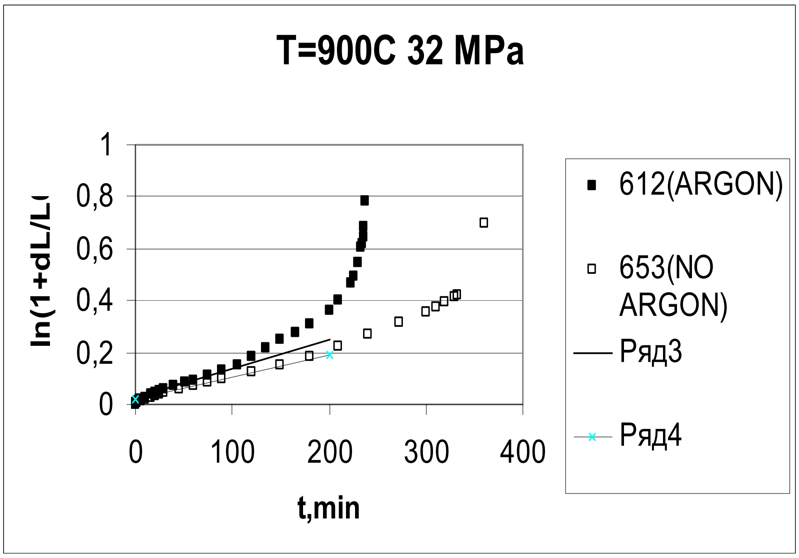


Fig. 4.4 – Diagrams of creep deformation at 850 ^{0}C

T=900°C





$n = \mathbf{4{,}826}$

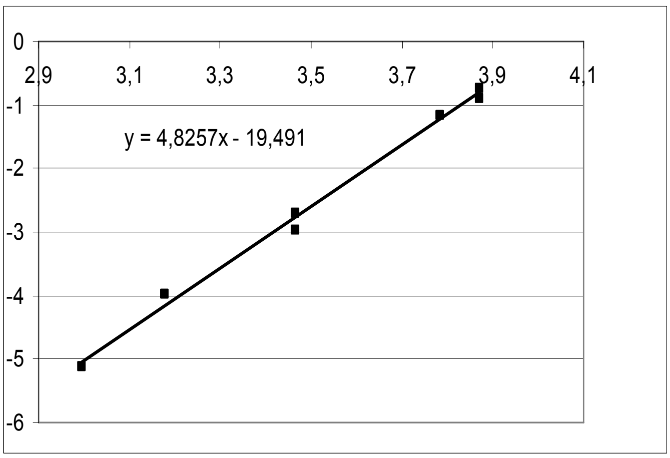
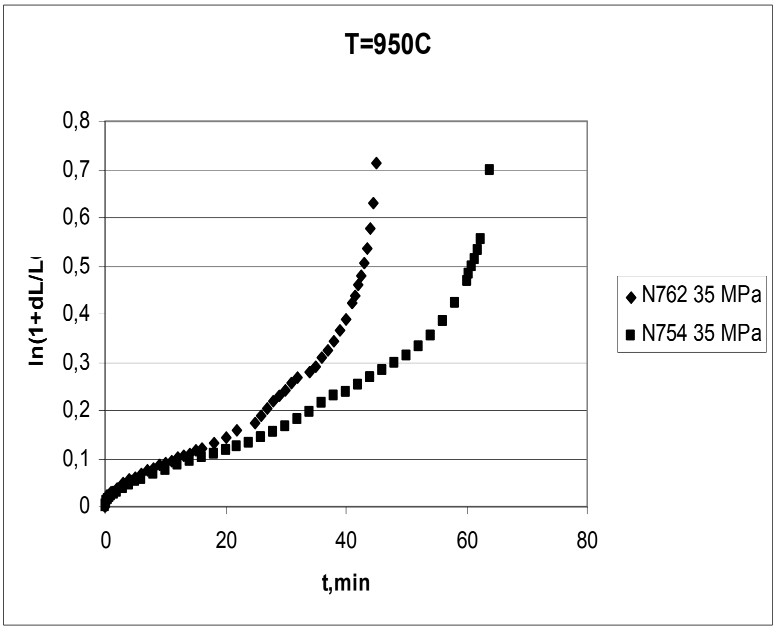
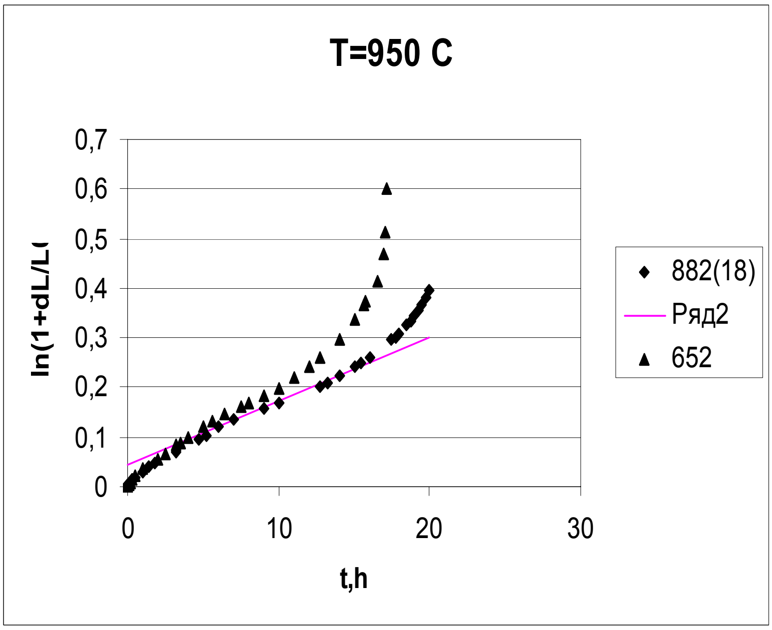


Fig. 4.5 – Diagrams of creep deformation at 900 ^{0}C

T=950^{o}C





$n = \mathbf{4,476}$

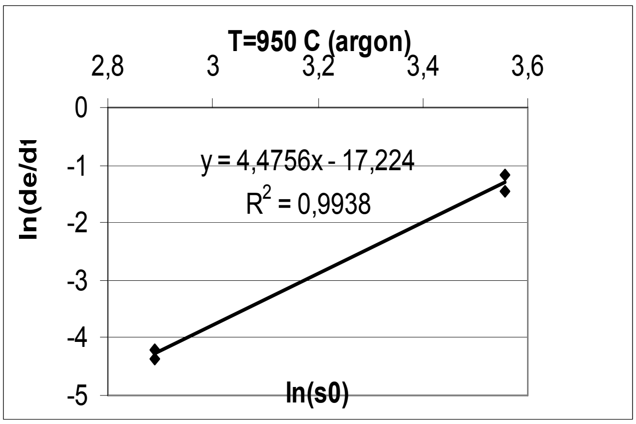
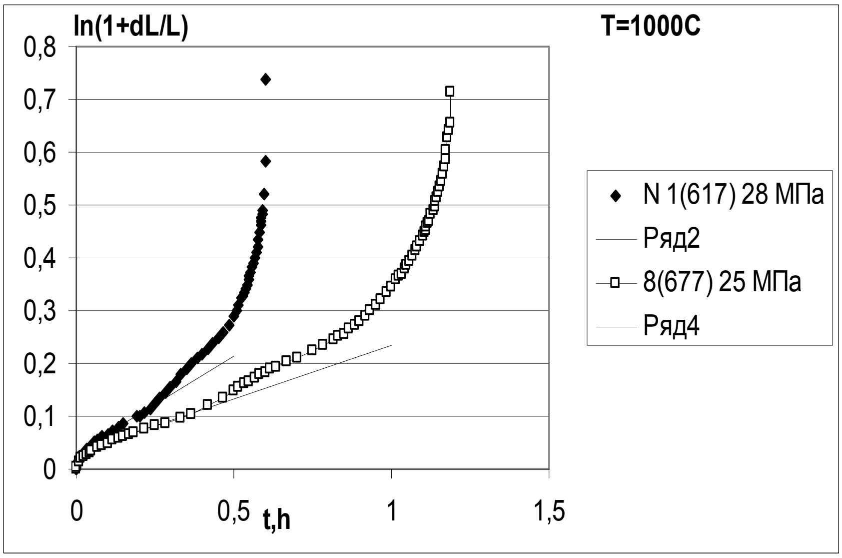


Fig. 4.6 – Diagrams of creep deformation at 950 ^{0}C

T=1000°C



$n = \mathbf{5{,}16\ 3}$

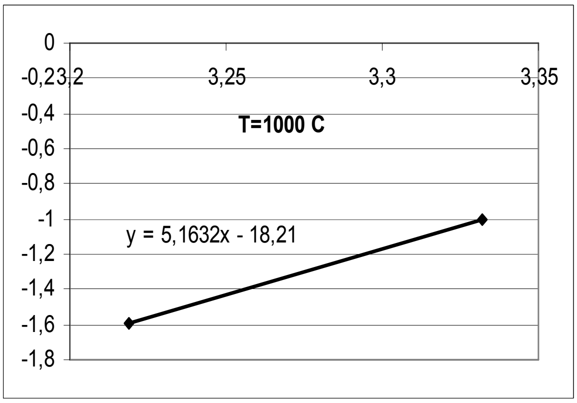
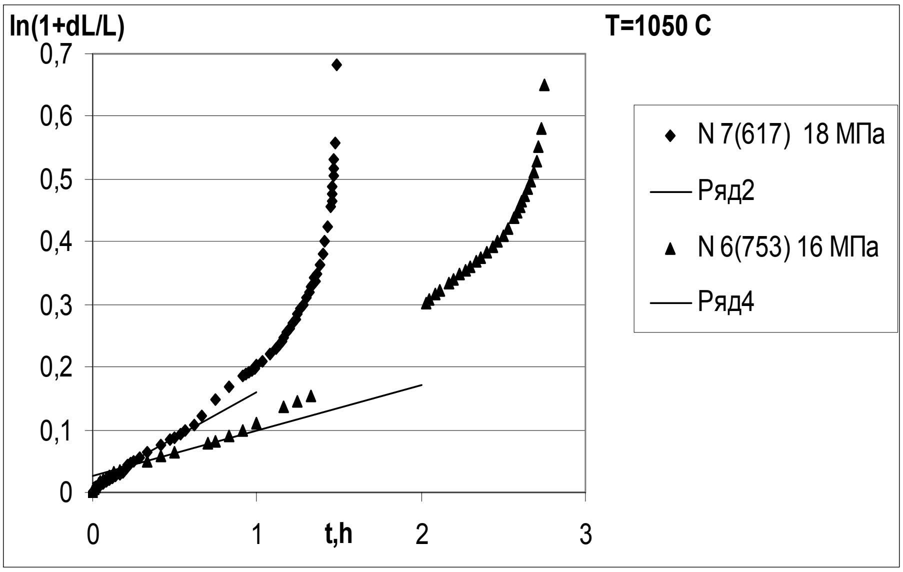


Fig. 4.7 – Diagrams of creep deformation at 1000 °C

T=1050°C



$n = $ **5,893**

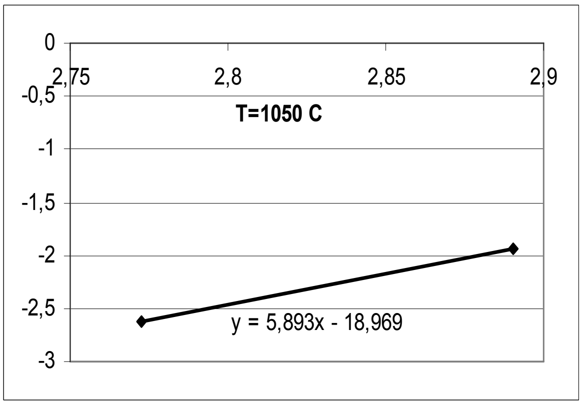


Fig. 4.8 – Diagrams of creep deformation at 1050 °C