Experimental R&D in Russia to justify Sodium Fast Reactors

Dr. Iuliia Kuzina

Leypunsky Institute for Physics and Power Engineering, Joint-Stock Company (IPPE JSC), Russia

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Meet the Presenter

Dr. Iuliia Kuzina is the Deputy Director General - Director of Nuclear Power Department in the State Scientific Centre of the Russian Federation – Leypunsky Institute for Physics and Power Engineering, Joint-Stock Company (IPPE JSC). Since 2000, she has been involved in studies of heat transfer in liquid metal coolants. Dr. Kuzina earned her PhD degree in 2003. In 2014, she became Head of the Laboratory for numerical and experimental studies of thermal hydraulics in loops with different types of coolants. Since 2016, Dr. Kuzina has held the position as Director of the Department in charge of SFR, LFR and light-water reactor design justification. Dr. Kuzina is a leader of computational and experimental work aimed at justification of safety and operability of fast reactor designs with liquid-metal coolants. In addition, she has been teaching at the university as a lecturer in thermohydraulic calculations since 2004. Dr. Kuzina is nominated as an expert from Rosatom in GIF LFR pSSC.

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Experimental R&D in Russia to Justify Sodium Fast Reactors Design

1. Sodium Fast Reactor Project

1.1 BN-type reactor systems

1 – intermediate heat exchanger

2, 3 – main and guard vessels

4 – support skirt

5 – pressure chamber

6 – core catcher

7 – core

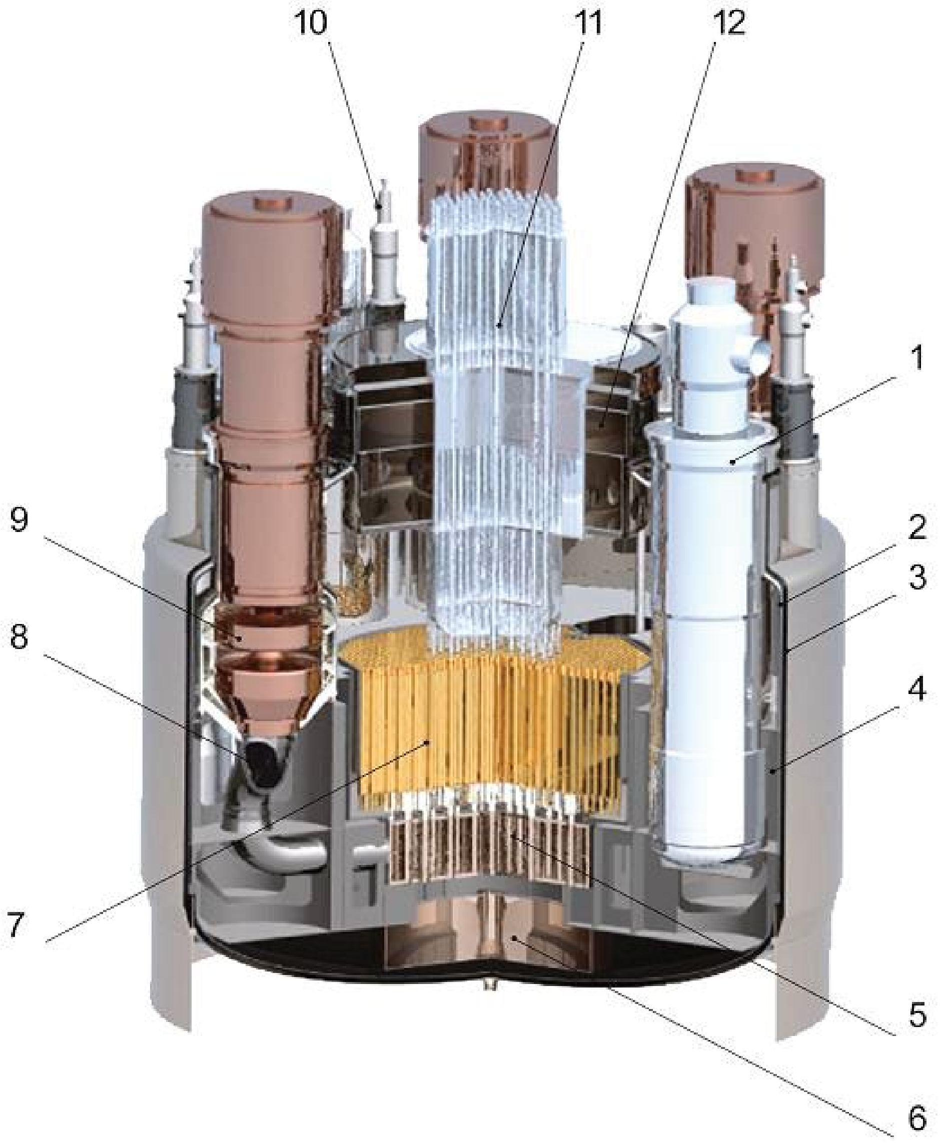
8 – pressure pipeline

9 – MCP-1 (primary main circulating pump)

10 – emergency heat exchanger

11 – CPS actuators

12 – rotating plugs



1.2 BN-type reactor systems

1 – reactor

2 – MCP-2 (secondary main circulating pump)

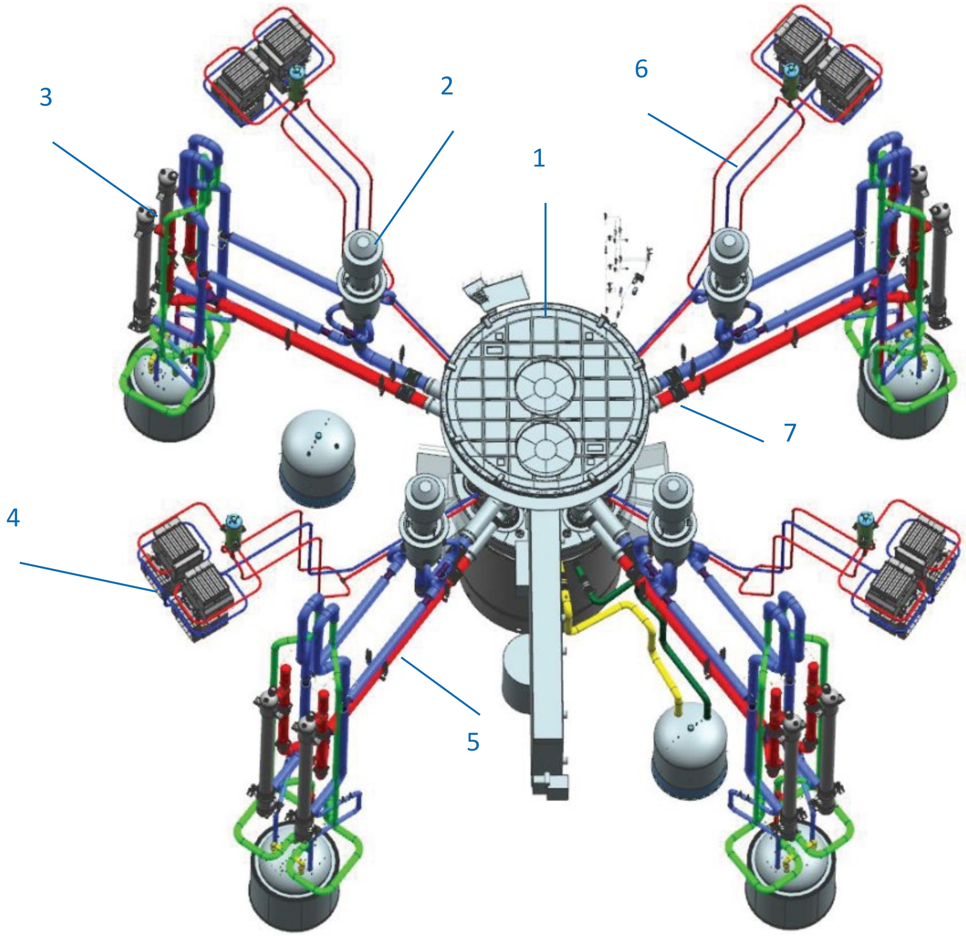
3 – steam generator

4 – air heat exchanger

5 – secondary circuit main pipeline

6 – DHRS pipelines

7 – expansion bellows



2. Core Justification

2.1 Core configuration

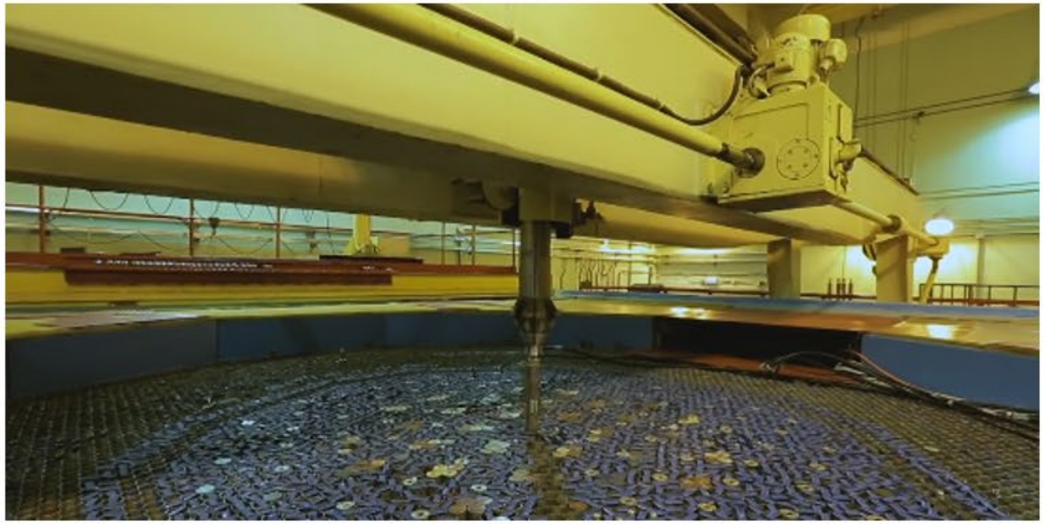
Calculated modeling of BFS loading

Core assembling at BFS

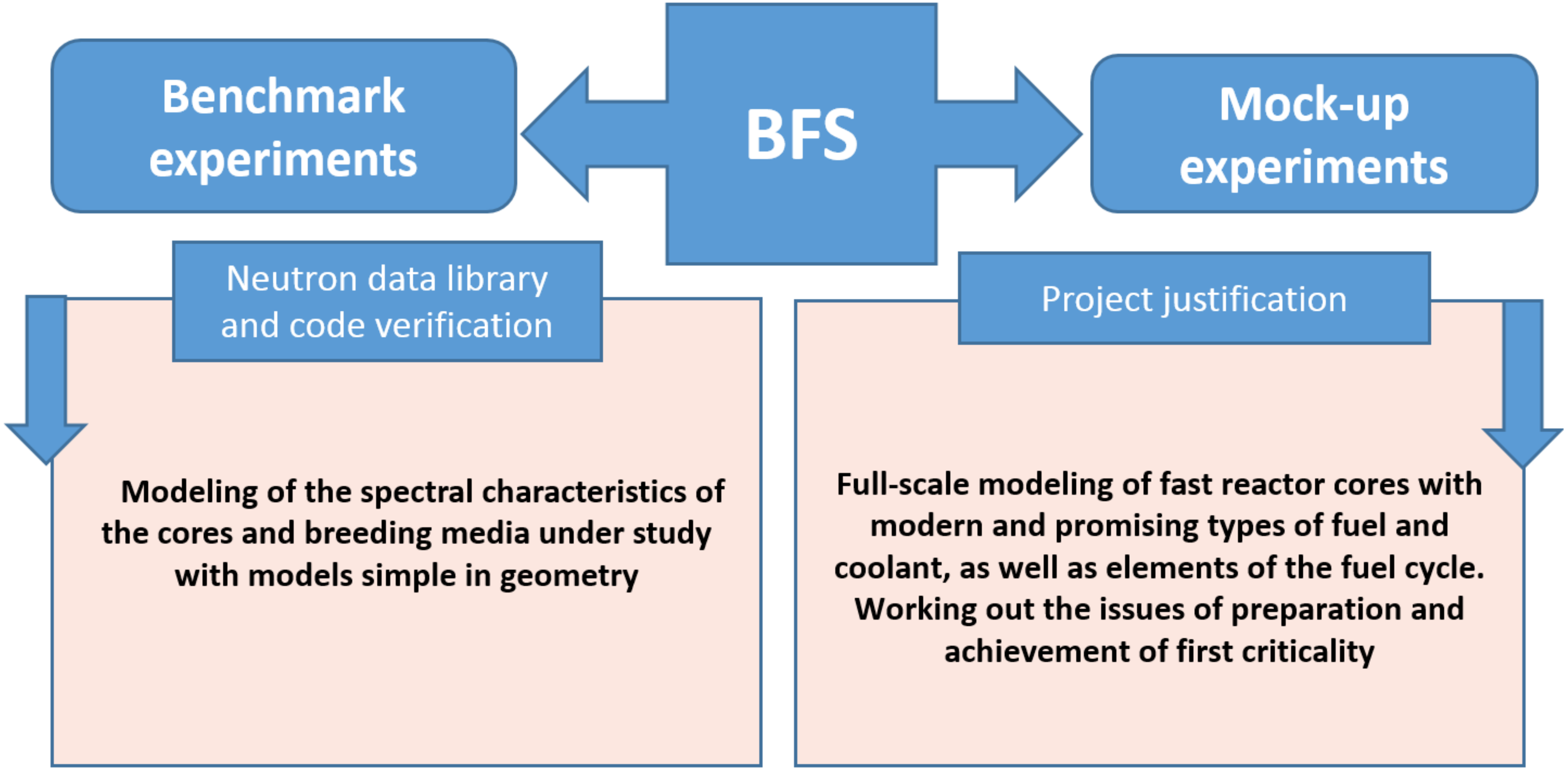
Experiments at BFS

Release of core basic design

Safety Analysis



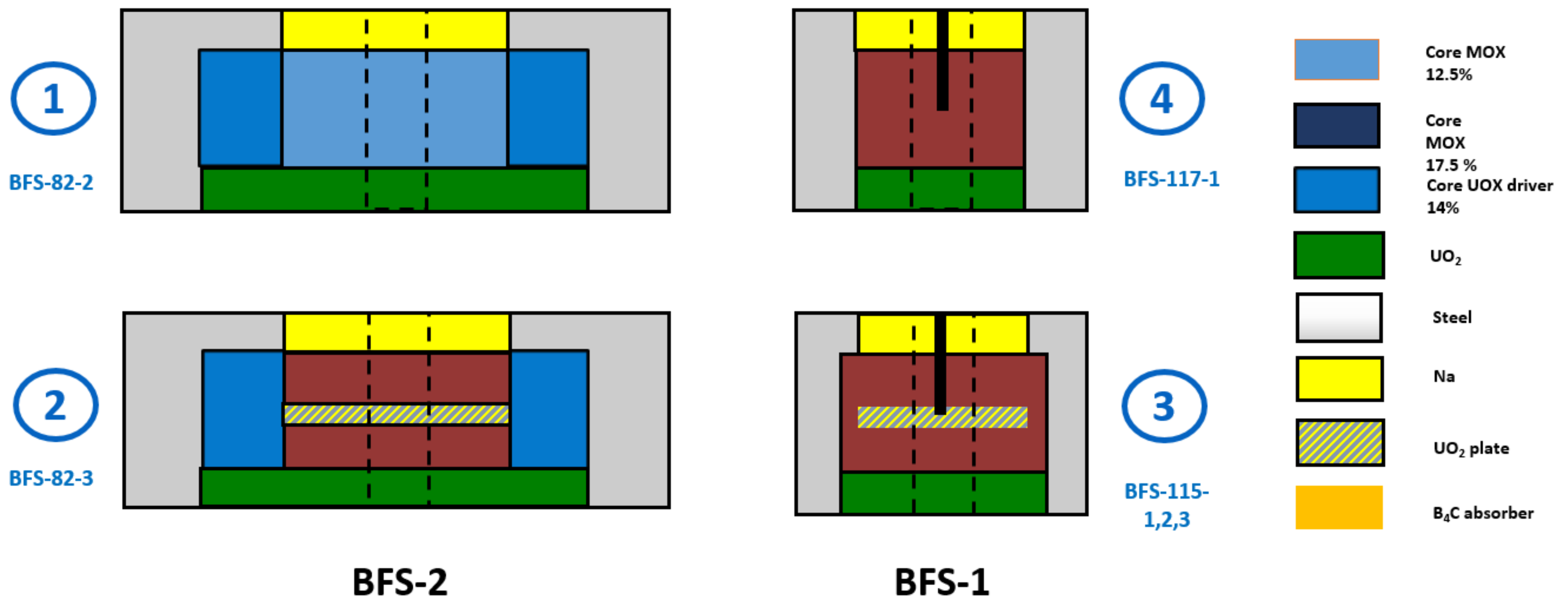
BFS critical facilities – potentiality to study neutronic characteristics for G4 reactor core justification



Cooperation with France (Implementing Agreement on Core Physics in the Field of Sodium-cooled Fast Reactors)

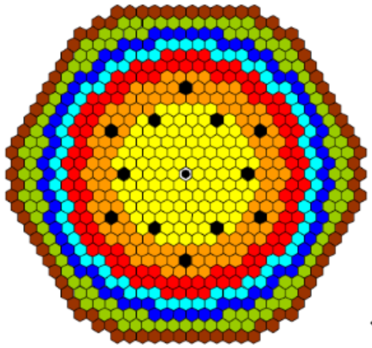
Joint experimental physics program in BFS

- Objective: Obtain experimental validation neutronic data for the design and safety assessment of innovative sodium-cooled fast reactor cores
- Four phases, 2012-2017 → **extended to 2021**

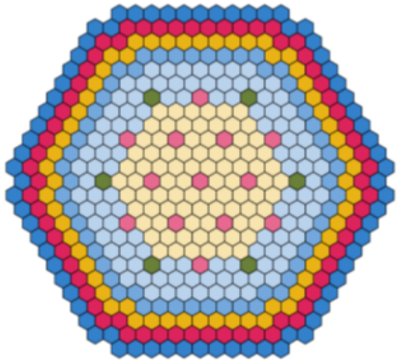


Cooperation with the Republic of Korea

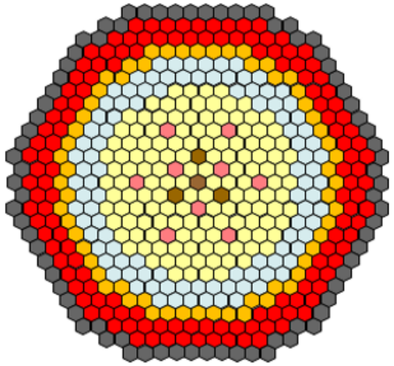
PGSFR – Prototype Gen-IV Sodium-cooled Fast Reactor



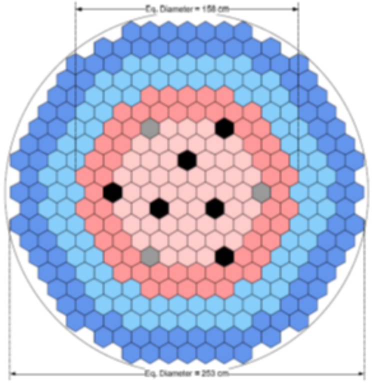
SFR-600
(KALIMER)
BFS(1)-73,75
1997-2000



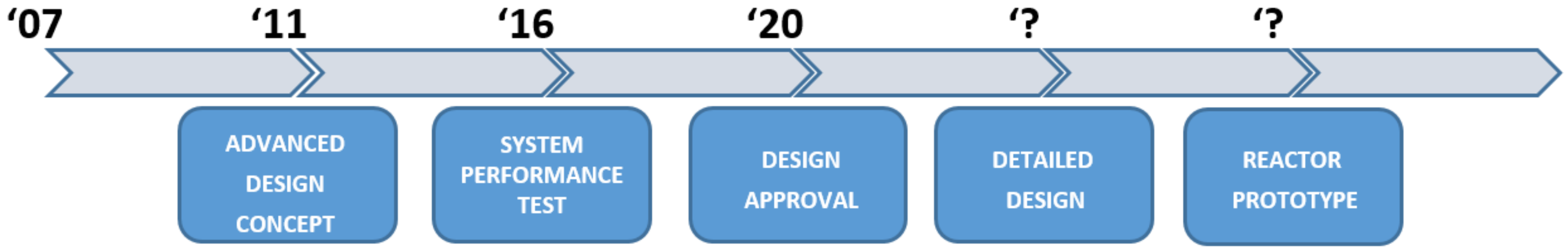
SFR-300
BFS(2)-76
2010-2011



SFR-100
BFS(1)-109
2012-2013



PGSFR-150
BFS(2)-84
2014-2016



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What are the benefits of international cooperation?

- **Test capabilities.** New test procedures are developed and mastered and the available ones are modified at the state-of-the-art level.
- **Calculation analysis of the test and its estimation.** It is performed with engagement of the most advanced computer codes and neutron data (ENDF\B, JEF, JENDL etc.) It makes it possible to both improve the national neutron data library (ROSFOND), and present data for the libraries of international partners.
- **Maximum work efficiency.** By concluding contracts, the stakeholders generate a research program and, as a result, exclusively receive new experimental information.
- **Public domain data.** Due to international collaboration, dozens of experiments performed at the BFS were subject to thorough expert review and included into the ICSBEP and IRPhEP Handbooks.

2.2. Justification of core safety

Results obtained experimentally:

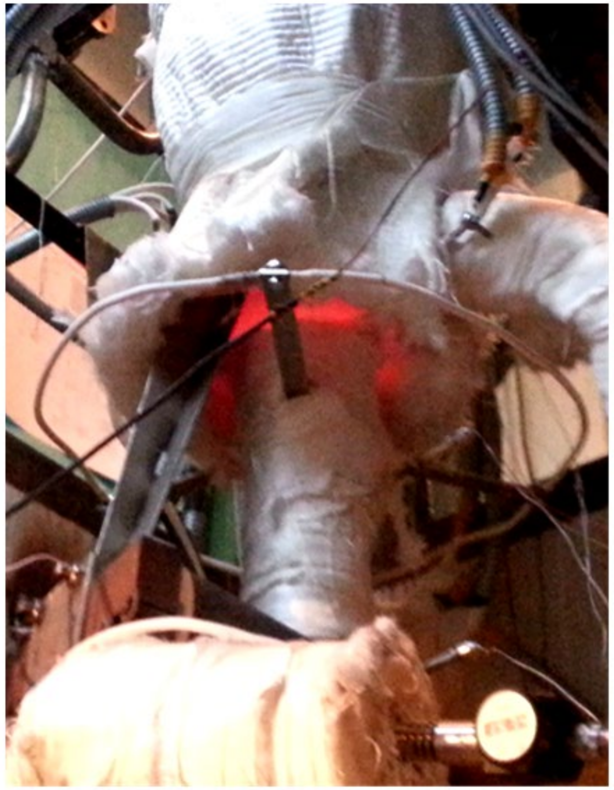
- form the basis for validation of computer codes
- rule out excessive conservatism

Research into high temperature processes

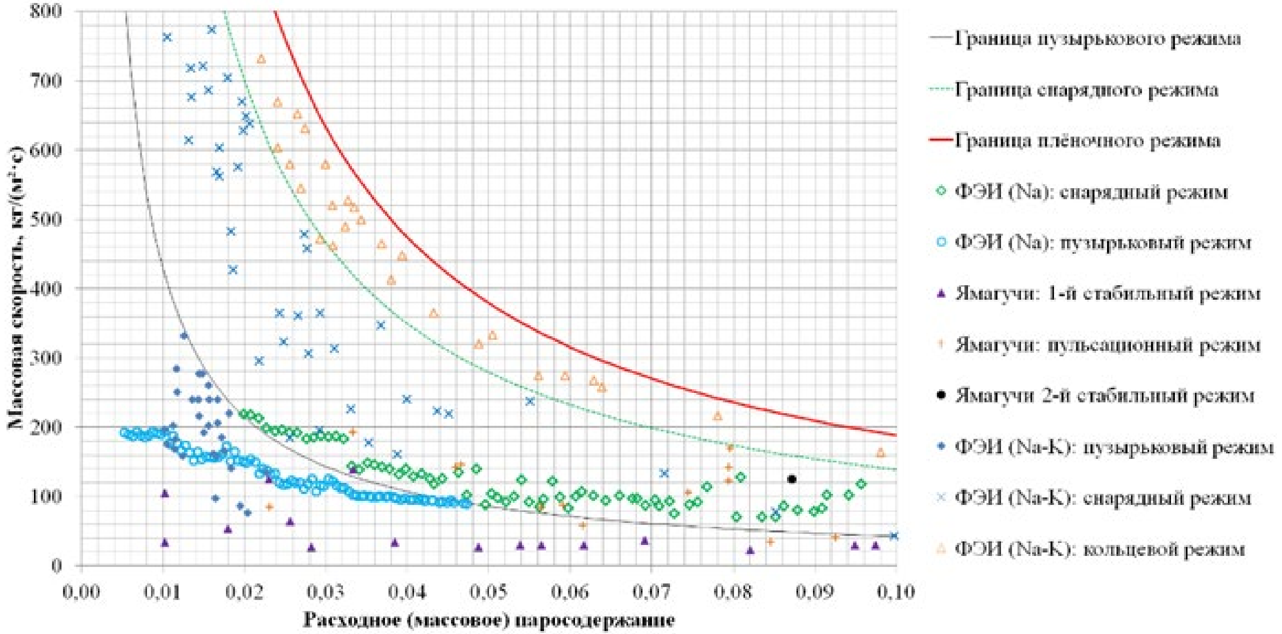
Reactor accidents and transients without core destruction.

Validation of severe accident codes – critical part of Gen-IV reactor safety analysis

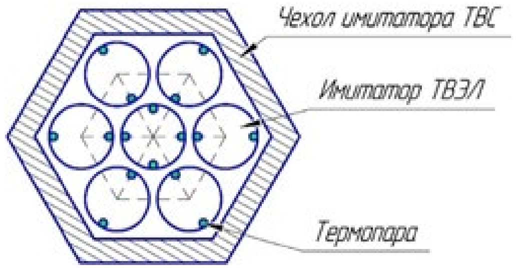
Research into sodium boiling in the fuel assembly model of the fast reactor core (1/2)



Test model in the course of the experiment



Flow regime chart of two-phase liquid-metal coolant flow in FA



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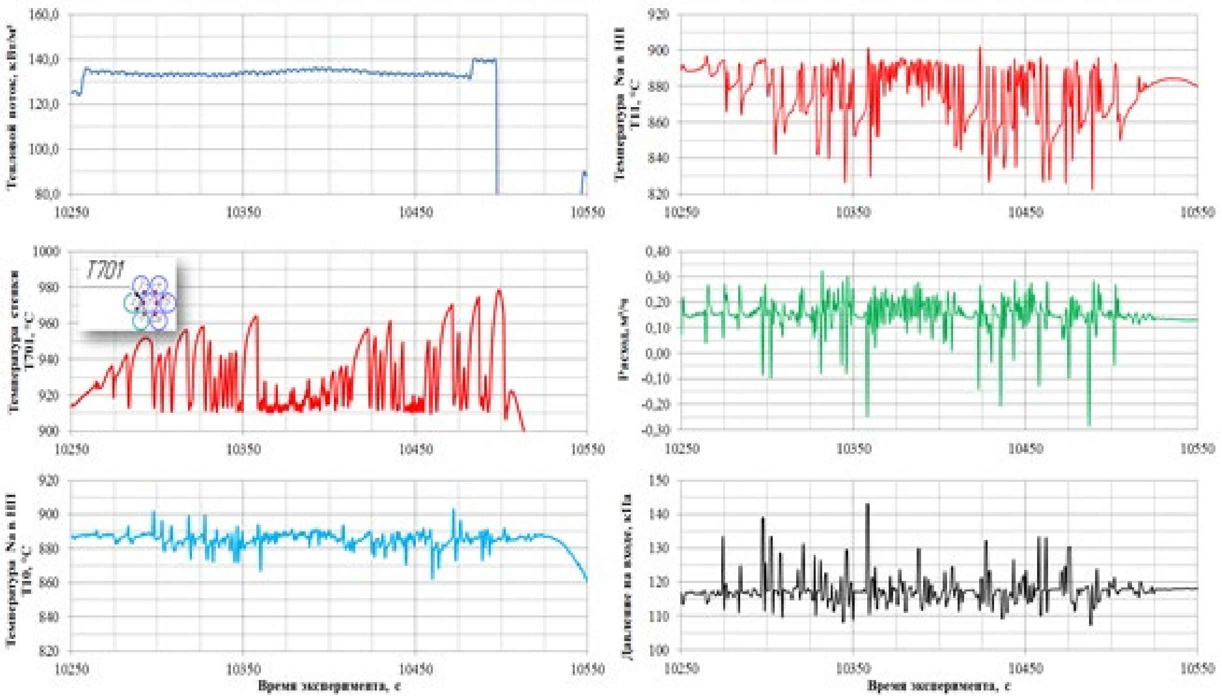
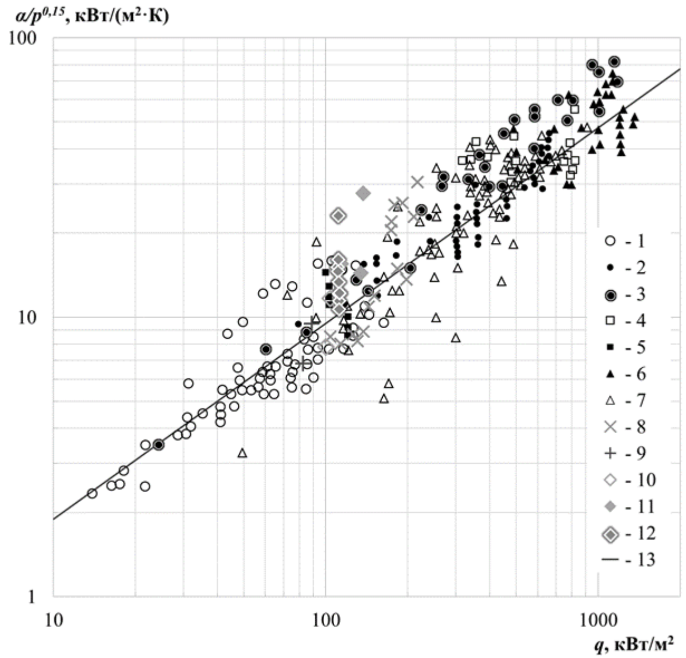
Research into sodium boiling in the fuel assembly model of the fast reactor core (1/2)



Variation of the central simulator wall temperature (T701), coolant temperature in the sodium plenum and the flow rate during sodium boiling in the natural convection mode in the heat flux range from 120 to 135 kW/m2

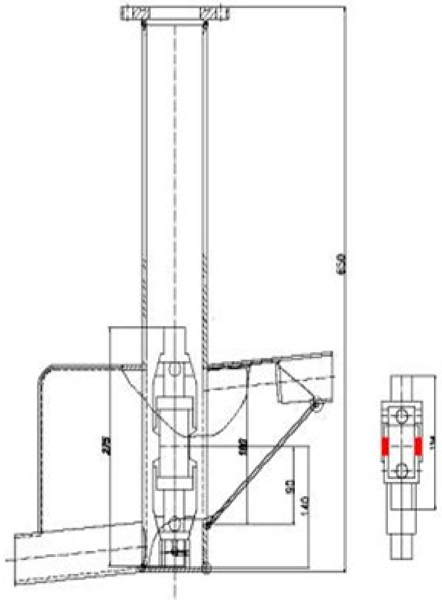


Comparison of various authors’ experimental data on heat transfer during liquid metal boiling in pipes with the IPPE data for fuel assemblies

Research into Passive Safety System (PSS)

- The passive safety system has an actuating component. This component of various designs is currently being tested
- Endurance tests are carried out prior to research into actuation processes
- Temperature increase rate under emergency conditions is high, up to 25 degrees per second
- High temperature and its increase rates impose serious requirements on characteristics and quality of the facility and equipment

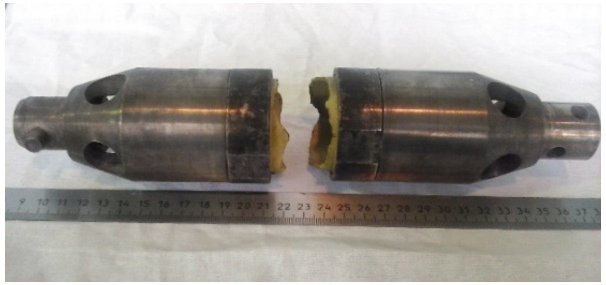
High temperature section of 6B facility used for studying components of passive safety system of fast reactors



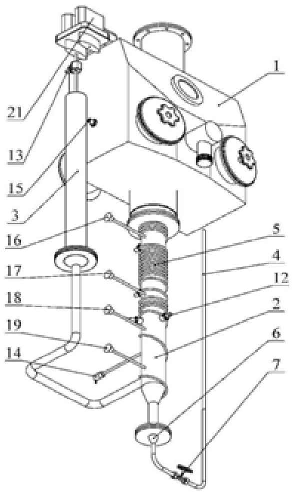
Location of the element in the test chamber

Test chamber

The test section allows studying samples of PSS-T elements and defining conditions, and time of their destruction in hot sodium flow (operating temperature is 720°C, flow rate is 1 m/s)



Liquid Metal Facility "PLUTON" (1/3)



Facility "Pluton" is designed for investigation of ULOF accidents through the use of corium simulators including uranium-containing ones (material movement in the course of fuel melting in FA models of various geometries, fuel heat-to-mechanical energy conversion factors, etc.)

Measured parameters: coolant power, temperature, flow rate, pressure

Scope of studies:

- ✓ *Studies on the characteristics of simulated corium-sodium interaction processes as dependent on the input parameters of the experiments (initial sub-cooling of sodium, melt mass outflow parameters, m_{Na}/m_{corium} mass ratio, etc.)*
- ✓ *Study on the fuel cladding failure*
- ✓ *Study of the reactor structural materials' behavior in case of their contact with corium*
- ✓ *Study in justification of in-vessel corium traps*

Liquid Metal Facility “PLUTON” (2/3)

Distribution of materials along the height of FA models and plugging of their flow sections in the course of simulating an uncontrolled loss-of-coolant accident in BN reactors

Fragments of claddings with drop-like impregnation of steel and iron were present in the area of global degradation.

Rod bundle

The end outlet openings of the test bundle were completely plugged with the steel melt, the drainage openings on the edges of the wrapper tube were partially plugged.

When the reaction chamber was opened, and the 19-rod assembly was removed and dismantled, it was found that sodium was present on the walls of the reaction chamber in the form of a deposit mixed with the products of its thermal interaction with the corium simulator. The assembly wrapper tube was burnt through in some places, on one of the edges the lower header was partially melted.

The initial configuration of the rod bundle remained the same in the outer layer, in its upper part. The inner layer of the simulators retained the initial configuration in the middle part.

The bottom half of the assembly was an area of global cladding degradation with multiple fractures, with obvious signs of cladding material melting, the presence of longitudinal and transverse cracks on the cladding fragments left.

Liquid Metal Facility “PLUTON” (3/3)

Experimental study of corium behavior at the boundary between corium and reactor structures (material wear, skull formation, physical-chemical parameters of the boundary area)

Phase stratification of the melt into metallic (Na) and ceramic (fuel) phases is shown experimentally. The chemical composition of the near-wall layer in the areas of phase localization corresponds to the main component of the phase.

The influence of thermal interaction of corium with nitrogen, which simulates the vapor phase of sodium, is considered. Melt ejections from the interaction area due to the gas phase expansion were experimentally estimated.

Material wear and deformations of specimens are recorded for the categories subject to melt shock impact only.

The results obtained make it possible to improve the high-temperature part of the COREMELT code (core melting)

Corium simulator melt stratification into metallic (a) and ceramic (b) phases

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2.3 R&D in Justification of New Fuel

In BN-1200, it is supposed to combine new technical solutions for the fuel element

- Fuel element cladding:
  - Austenitic steel EK164-ID c.d. at the initial stage
  - Improved steels of ferritic-martensitic class at the following stages
- Two fuel types:
  - MOX
  - New fuel type – mixed nitride uranium-plutonium

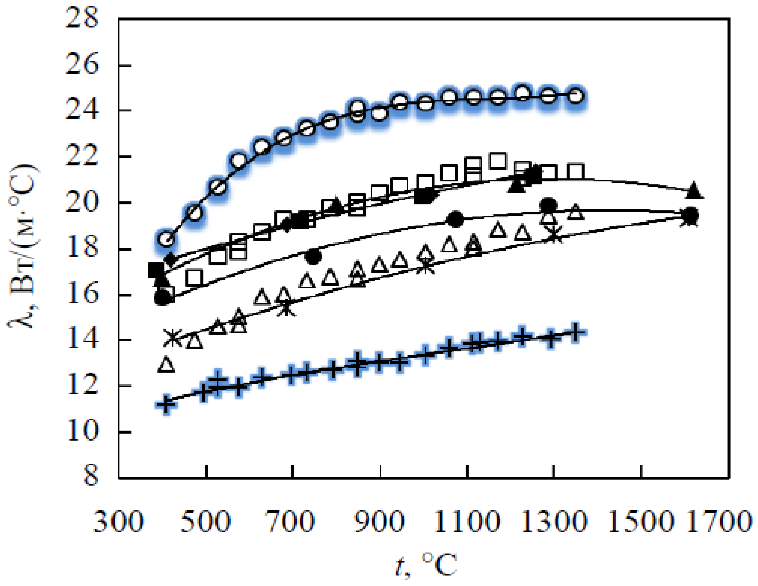
Irradiation tests are carried out at fast neutron reactors

- BOR-60
- BN-600

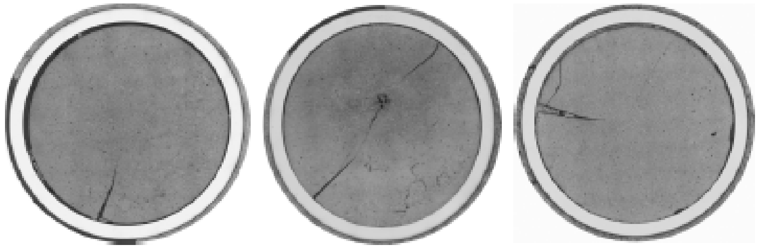
Reached fuel burnup fraction – 7.6 % ha and damaging dose – 96 dpa

Comprehensive program for computational and experimental justification of fuel elements with nitride fuel

- Pre-reactor experimental studies of the properties of nitride fuel and cladding material
- Testing of experimental fuel elements with MNUP fuel in BOR-60 and BN-600 reactors
- Post-reactor studies of irradiated fuel elements
- Development of methods, codes and criteria for substantiating the operability of fuel elements
- Improvement of fuel elements with MNUP fuel, development and optimization of technologies for their manufacture



Temperature dependence of MNUP fuel thermal conductivity


Macrostructure of fuel and fuel element cladding in various cross-sections

3. Justification of In-Vessel Primary Circuit Equipment

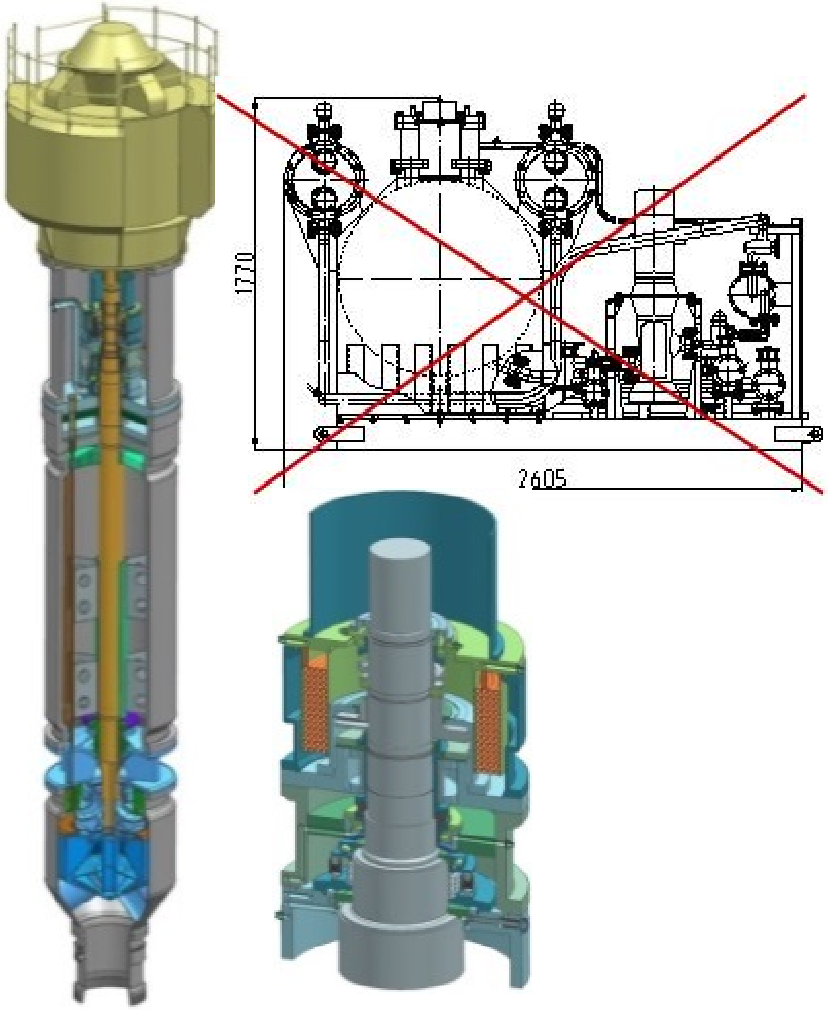
3.1 Main Circulation Pump of the Primary Circuit

Design - a radically new rotor turning gear lubrication system having nothing comparable in BN-type reactors

Lube oil supply system – downsized

- An oil bath lubrication system was applied
- Many elements of the system were removed, thus reducing the mass of this unit

A full-scale mockup of the upper bearing unit has been created and is being tested



MCP-1 with an oil bath lubrication system without a lube oil unit

Primary Circuit Pump Testing

The equipment that has reference samples and requires no justification at high temperatures undergoes water tests.

Components that require liquid metal and high temperatures are tested at sodium test facilities.

Shaft seal mockups undergo water tests.

Testing of a model flow meter for MCP-1 at the sodium test facility IRS-M.

Calibration characteristics of the electromagnetic and vortex channels were obtained, the basic reduced error in the flow rate measurement was determined.

MCP-1 shaft seal mockup

3.2 Emergency Heat Exchanger Testing

- A completely new design of a heat exchanger with a check valve
- The valve opens once a year during reloading, once every ~330 days
- High sodium temperatures up to 550 degrees Celsius
- Probability of the check valve welding with the heat exchanger

Performance capability of this component was confirmed under conditions close to real

Check valve (sphere) after endurance tests

3.3 R&D in justification of a plugging meter to measure impurities in sodium

The design of a small-sized plugging meter to measure impurities in sodium and a system for its non-dismantling periodic verification have been developed and are being tested at the sodium test facility.

Small size of the equipment allows it to accommodate a plugging meter inside the vessel of BN-type fast reactors .

The reduction in size has been achieved by combining the functions of a pump and flow meter in one device.

3.4 Study of a coolant flow in the course of decay heat removal

Two test facilities were used to study coolant flows inside the reactor vessel. The working fluid of both test facilities is water.

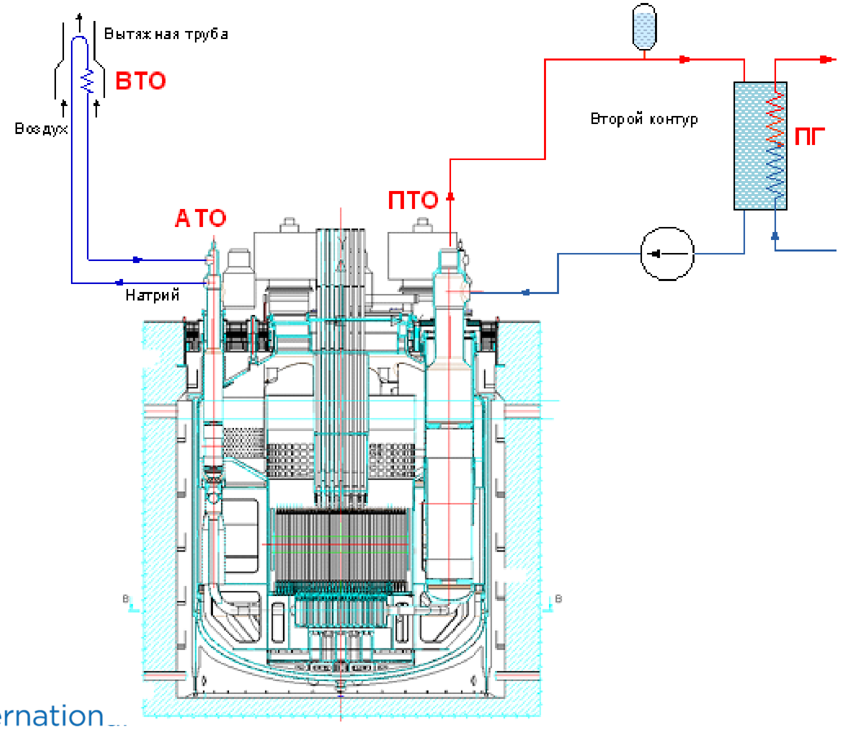
TISEY – simulates a sector of the BN-1200 reactor

V-200 –simulates a BN-1200 configuration

BN-350 decay heat removal is carried out through the main heat removal system, but with the use of auxiliary electromagnetic pumps

BN-600, BN-800 – the DHRS is on the secondary circuit

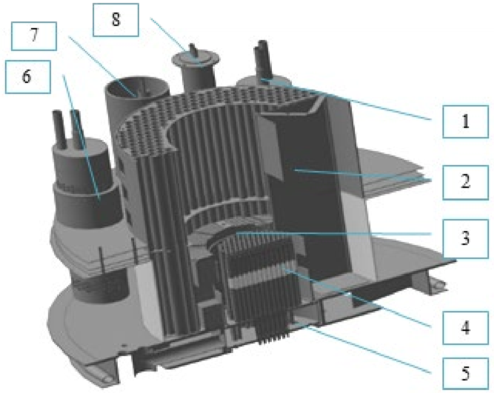
BN-1200 – three-circuit four-loop EHRS of a "sodium-sodium-air" type with straight-tube emergency heat exchangers



At the IPPE, thermo-hydraulic studies were performed with the water model of the primary loop at the scale of ~ 1:10, for a pool-type fast liquid metal cooled reactor.

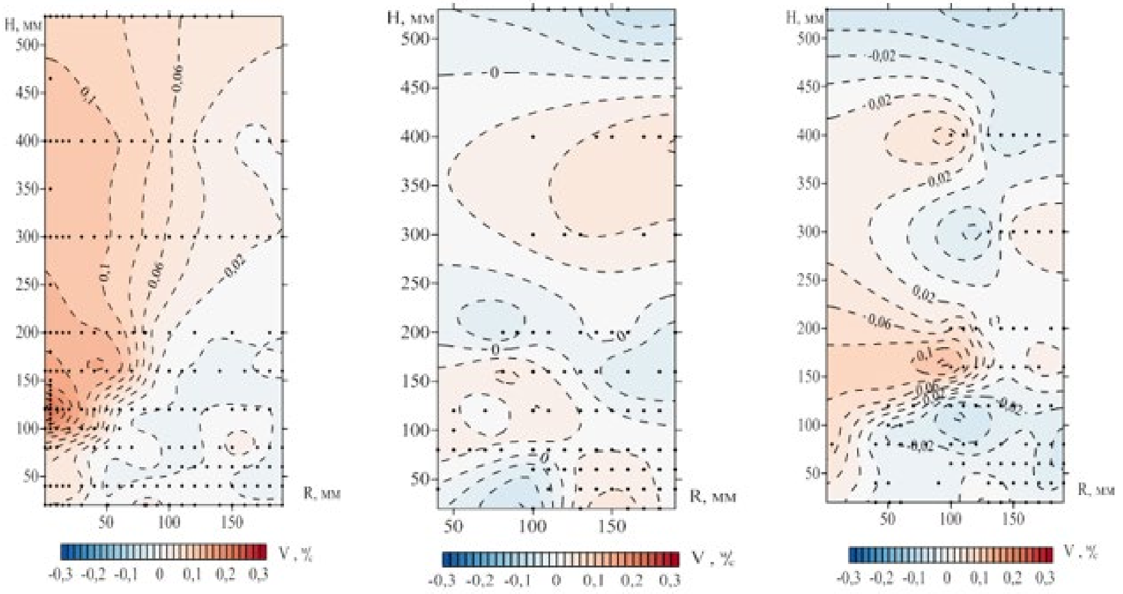
General view of the model

In-vessel model equipment



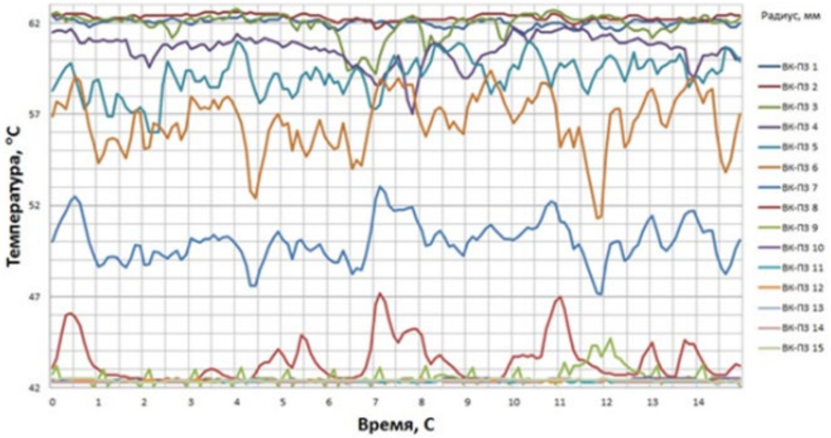
1, 6 – intermediate heat exchangers
2 – elevator baffle
3 – elements of in-vessel protection
4 – core (FA simulators)
5 – pressure chamber
7 – MCP-1 simulator
8 – autonomous heat exchangers

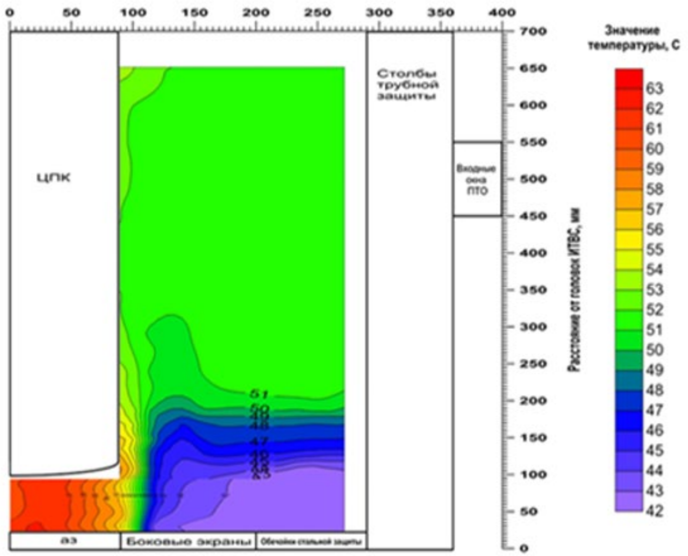
Measurement results obtained at that model (1/2)



a) b) c)

Coolant velocity fields in the upper chamber of the reactor: vertical component (a), radial component (b), azimuthal component (c)

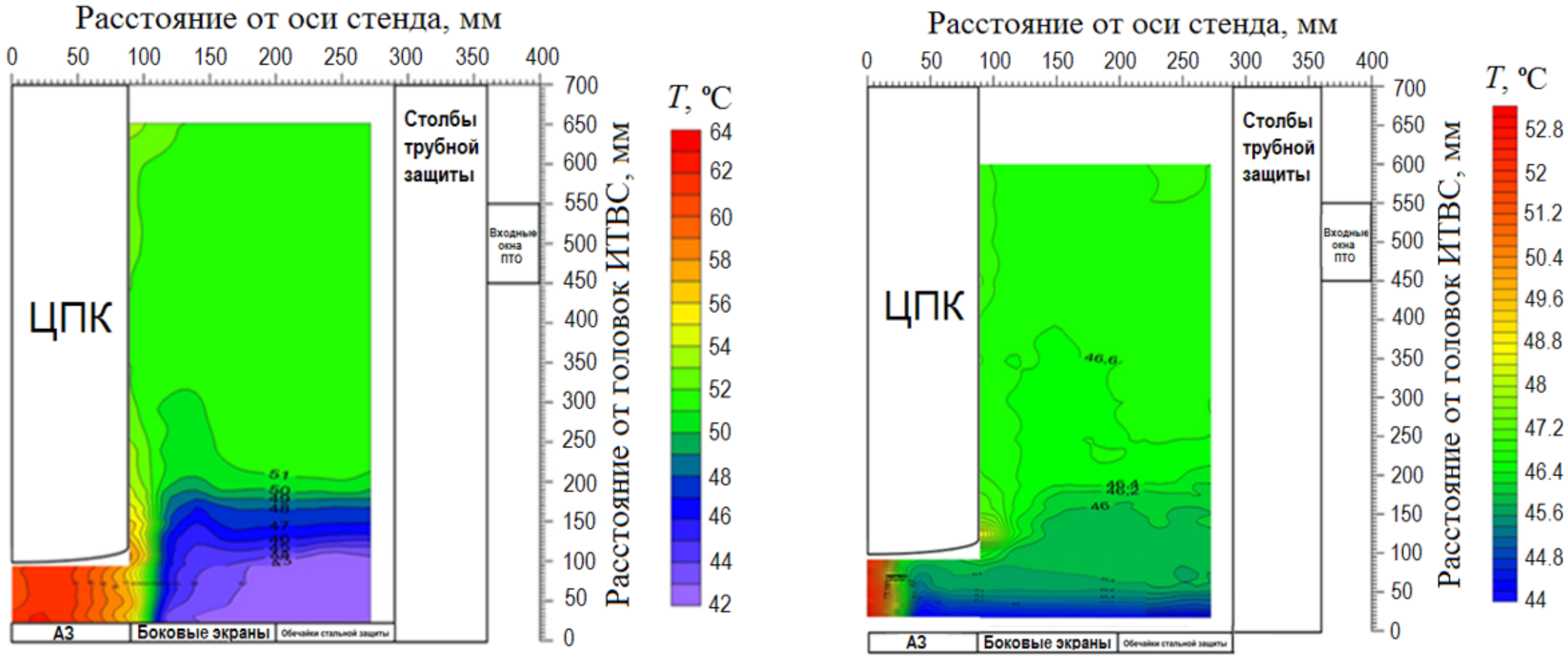




Average coolant temperature distribution along the height of the upper chamber

Measurement results obtained at that model (2/2)

The comparison was made of the charts showing thermohydraulic processes in the reactor vessel under the nominal conditions and the steady-state natural convection regime



Average coolant temperature in the upper chamber under nominal conditions and under natural circulation conditions

The steady-state natural circulation conditions are characterized by significantly lower temperature gradients in the vertical direction above the radial blankets, as compared to the nominal conditions.

4. Justification of the Secondary Circuit Equipment

4.1 Testing of the Secondary Pump

The leak-proof pump of the secondary circuit is installed directly on the intermediate heat exchanger (of a new design, with a synchronous motor)

- Testing of a bearing (with a full-scale axle bearing model)
- Testing of the individual components of the pump (a sealed motor, a thin-wall shield)
- Testing of a full-scale axial/radial bearing model (1:3) of the secondary pump.

The tests are conducted with the parameters similar to the operating ones, both in sodium and water.

4.2 Testing of Expansion Bellows

The BN-1200 design envisages expansion bellows installed in the main pipelines and steam generator modules.

The tests cover a few steps:

- Heating, cooling (melting and solidification)
- Flexibility of bellows in different planes
- Examination after all the test cycles completed

Expansion bellows confirmed their operability at the preset parameters: cyclic pressure loading (up to 35 cycles, at 528 ± 3°C, up to 0.64 ± 0.02 MPa), their axis deviation of 6° in various directions.

4.3 Testing of Steam Generator Systems

The scope of R&D work on BN-1200 steam generators includes a great number of tests:

- Research into the processes in heat tubes with single-tube and multi-tube models
- Development of the assembling process for a straight-tube steam generator of a long length
- Development of water-to-sodium leak detection systems
- Testing of rupture (bursting) disks

Thermo-Hydraulic Studies in Justification of Large-Modular Steam Generator

The experiments were performed with a single-tube model at the "SPRUT" test facility with the aim to justify the design parameters of a new design of a large-modular reactor steam generator where within one shell the processes of steam evaporation and superheating take place, at the modes of 12.5% and 75% of the nominal sodium flow rate, with real parameters in the sodium loop and high-pressure water loop.

Heat transfer parameters in the modes of sodium flow rate of 75%

Parameter	Mode 1	Mode 2	Mode 3
Pressure, MPa	12.71	12.55	14.14
Water mass velocity, kg/(m^2·s)	955	964	981
Sodium temperature at x = 0, °C	365	363	375
Sodium temperature at $x = x_{cr}$	399	399	406
Critical (boundary) steam quality	0.381	0.402	0.346
Critical density of the heat flux, MW/m^2 (experiment / skeleton tables)	0.66/0.67	0.63/0.64	0.762/0.78
Distance to the CHF area from the heating onset, m	7.3	6.9	6.6

The experimental CHF data obtained demonstrate a satisfactory agreement with the data of skeleton tables on calculated CHF in tubes

A significant effect of water pressure on both the critical (boundary) steam quality and the value of heat flux density was observed.
With pressure growth, the value of heat flux density goes up and the value of critical (boundary) quality goes down.

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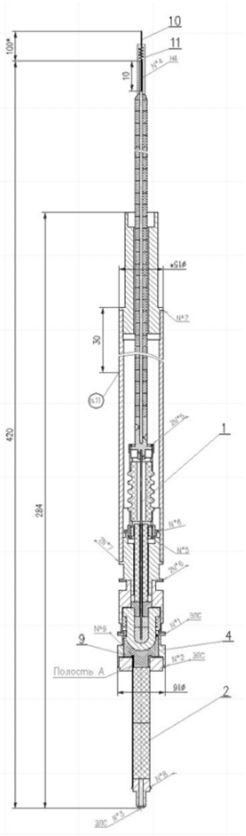
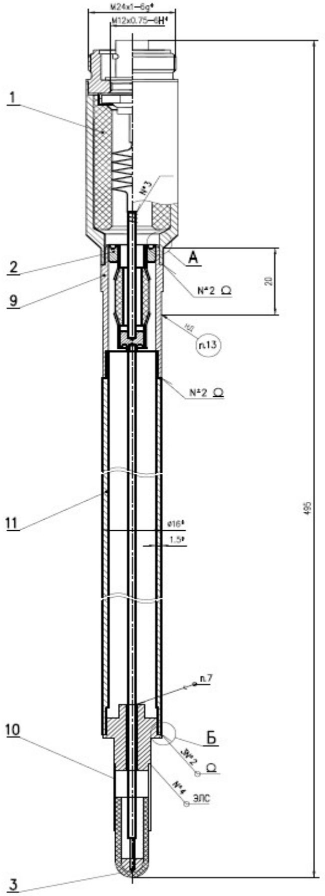
4.2 A small-water-to-sodium-leak detection system is designed on the basis of solid-electrolyte sensors SKV-N and SKK-N that monitor hydrogen and oxygen contents in sodium

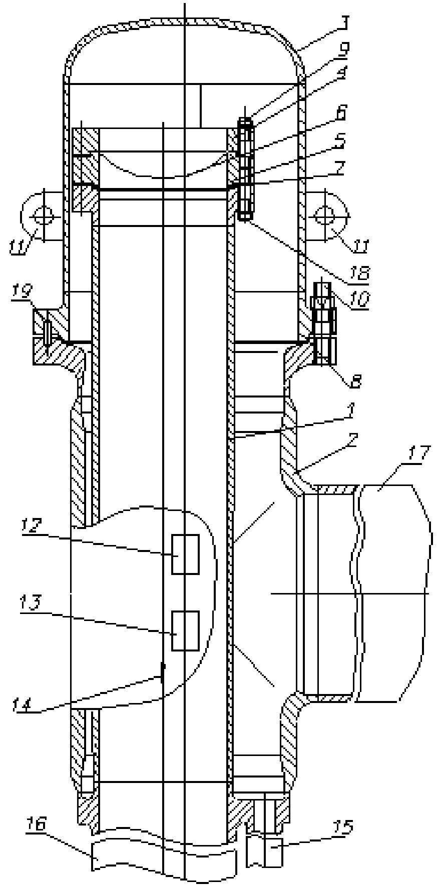


SKV-N



SKK-N

4.3 Steam generator overpressure passive safety system is designed on the basis of pressure safety devices in sodium and gas.

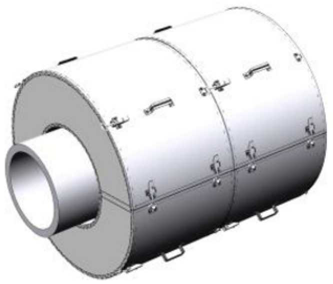


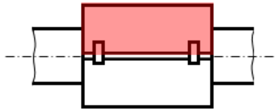
1 – insert;
2 – housing;
3 – cap;
4 – top flange;
5 – intermediate flange;
6 – rupture disk;
7 – cone spacer $D_{ВН}$ = 210 mm;
8 – cone spacer $D_{ВН}$=370 mm;
9 – pin M16x1.5;
10 – pin M22;
11 – cap slinging elements;
12 – a plate with rupture disk characteristics;
13 – name plate;
14 – flow direction;
15 – sodium drainage nozzle;
16 –inlet nozzle;
17 – discharge nozzle;
18 – pin M16;
19 – location pin

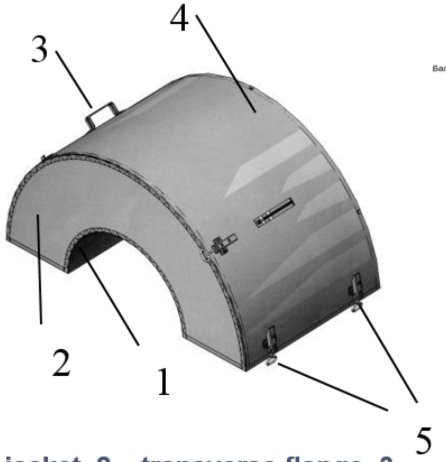
Structural design of the first-of-a-kind sample UPM-200

4.4 Sodium Leak Detection with the Use of Fast-Removable Thermal Insulation as a Protective Jacket in the Secondary Circuit (1/2)

Fast-removable thermal insulation (FRTI) design

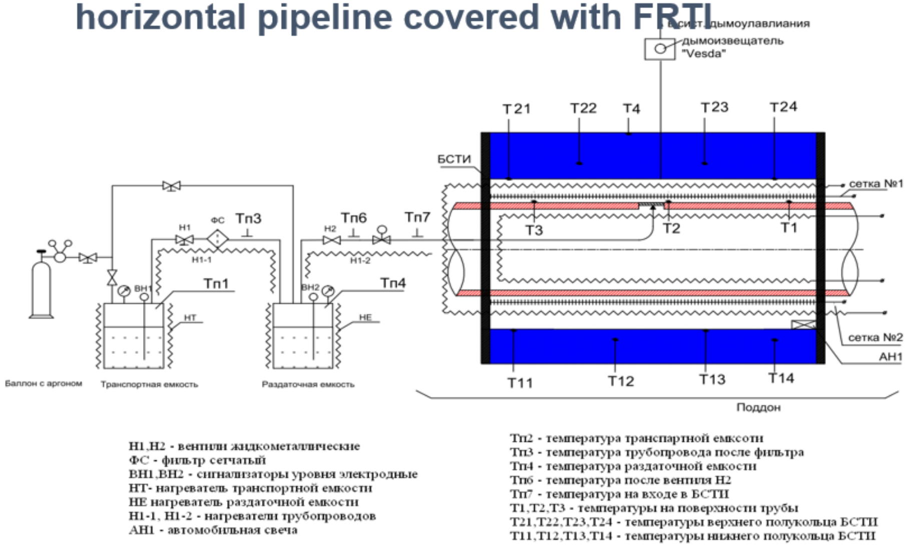






1 – inner jacket, 2 – transverse flange, 3 – handles,
4 – outer jacket, 5 – locks

Process diagram of the test section with a horizontal pipeline covered with FRTI



Six experiments were conducted:

– three experiments with a horizontal pipe, ∅ 325×12 mm, with different sodium leak rates through a flaw with a diameter of 2 mm,

– three experiments with a vertical pipe, ∅ 325×12 mm, with the same parameters.

4.4 Sodium Leak Detection with the Use of Fast-Removable Thermal Insulation as a Protective Jacket in the Secondary Circuit (2/2)

The view of FRTI in place after experiment No.1

Steady-state conditions:
– sodium temperature in the dispensing chamber – 500 °C;
– distributing pipeline temperature – 500 °C;
– argon pressure in the gas cavity of dispensing chamber – 0.2 MPa;
– temperature of the pipe, 325×12 mm – 447 °C.

The use of contact sensors for sodium leak detection (wire and meshed detectors) demonstrated its effectiveness in terms of sodium leak detection at early stages, with the capability of sodium leak localization.

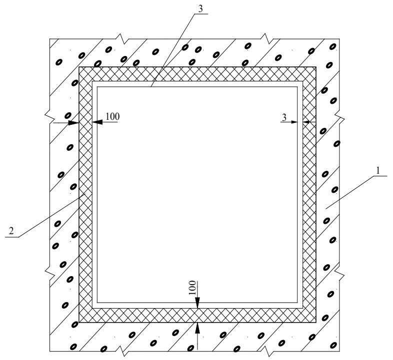
4.5 Test Research into Sodium Fires

Test procedures:

- sodium, the sodium pipeline and test section are heated to the preset temperature;
- the required pressure is achieved by gas supply to the tank with sodium;
- The valve is opened and sodium goes to the test section, under thermal insulation, with a specified flow rate.

Different types of thermal insulation require experimental justification.
The conclusions based on the experimental results:

- no sodium spray fire at the preset leak rates;
- confirmation of the concept "leak before break";
- performance testing of the sodium leak and fire detectors;
- verification of computer codes.



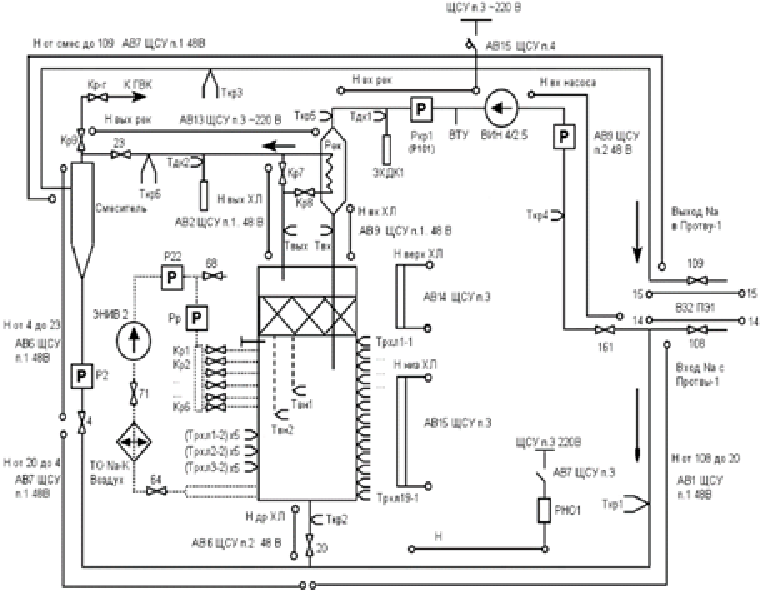
1 – reinforced concrete wall;
2 – thermal insulation; 3 – steel lining

The design of safety enclosure for the rooms with sodium

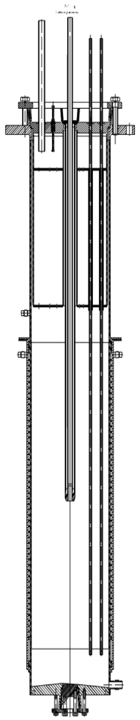
5. On Some Issues of Sodium Coolant Technology

5.1 Testing of a cold trap model at the sodium test facility

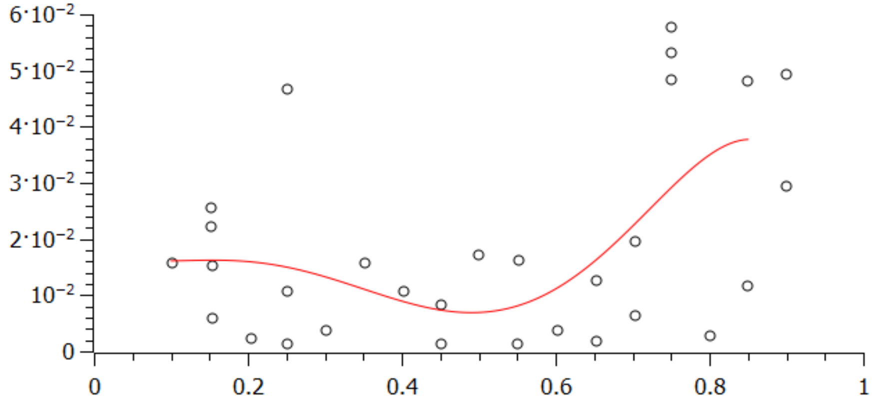
Schematic diagram of the cold trap model connection at the "Protva-1" test facility



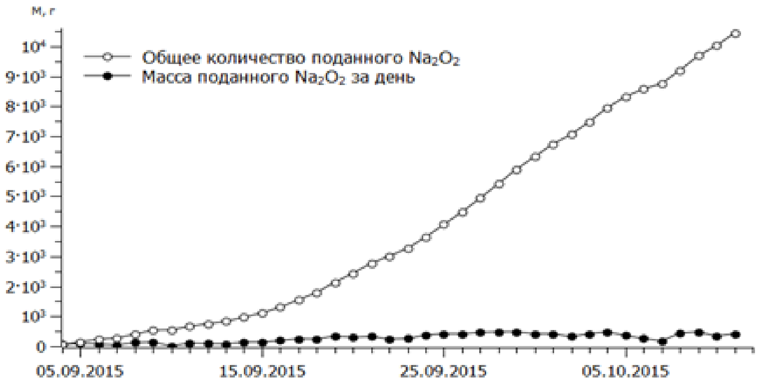
Cold trap model schematic



Distribution of deposits along the trap model length was determined

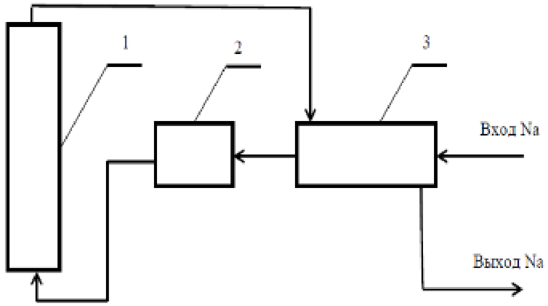


The graph of impurity supply to the sodium circuit and the amount of sodium peroxide supplied



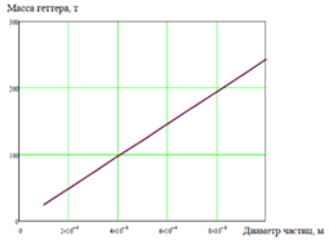
Recommendations were developed to improve the design and operating modes of cold traps.

5.2 Development of a combined system of primary sodium purification from oxygen, with the use of getters, sorbents and filters



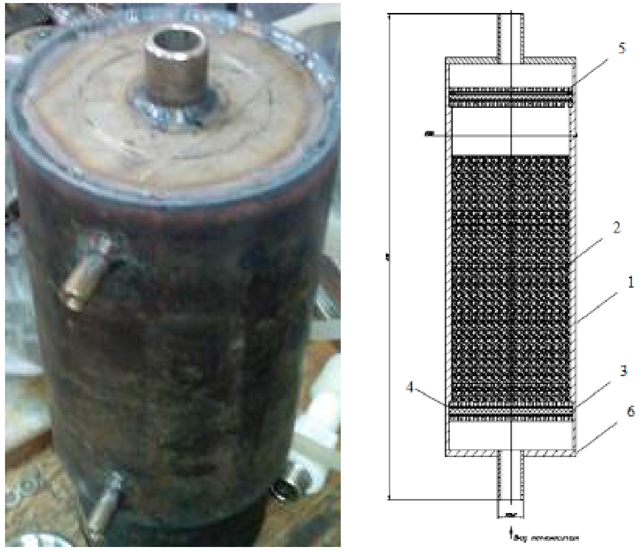
1 – горячая ловушка; 2 – нагреватель; 3 – рекуператор

Hot trap connection diagram used in the reactor outage (standby) conditions

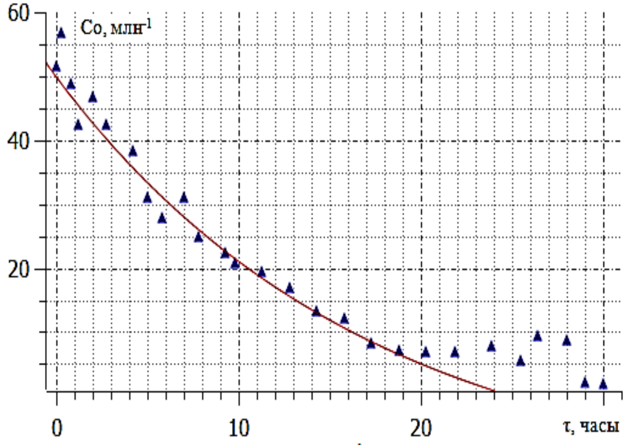


Zirconium powder consumption in a fast reactor for 60 years, depending on the getter particle size, tons

$$m_{\text{гет}} = \frac{(C_0 - C)\cdot \rho_з \cdot d \cdot m_{\text{Na}}}{6A\exp\left(-\frac{B}{T}\right)\cdot \tau^n}$$

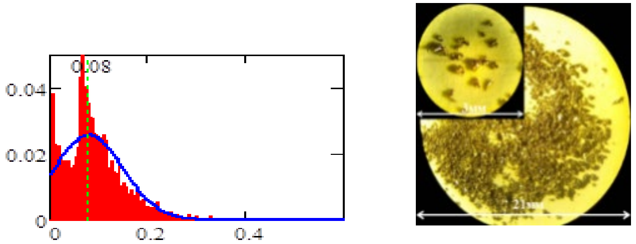


Getter purification module, 6 kg of zirconium powder



Variation of the oxygen concentration in sodium at the "PROTVA-1" test facility in the course of purification with zirconium getter, T=550°C

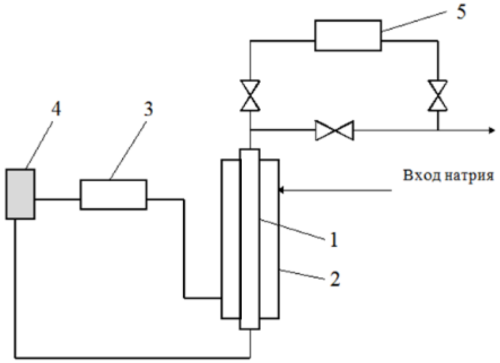
$$C(\tau) = C_0 - \frac{SK\tau^n}{G_{Na}} + \frac{Q_{O_2}\tau}{G_{Na}}$$



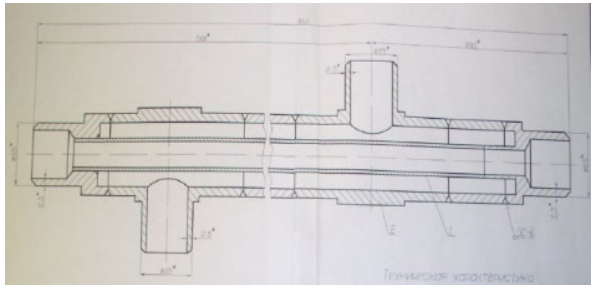
Determination of the dispersed getter powder composition

5.3 Research into Mass Transfer of Steel Corrosion Products in Sodium (up to 780 ^{0}C)

Corrosion product source

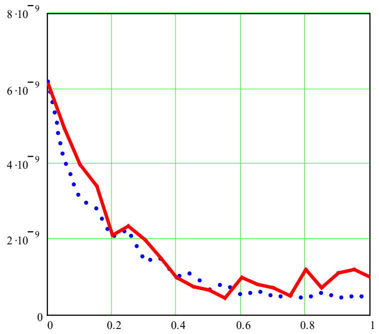


Suspension filter

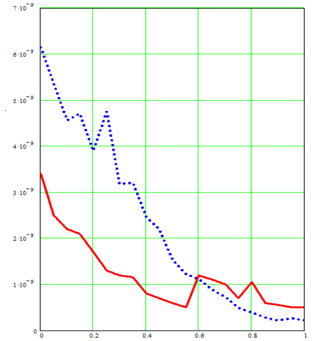


Recuperator with a mass transfer tube

Parts of the mass transfer tube after cutting



Comparison of experiments on chromium transfer in sodium with different hydrogen content



Hydrogen impact on nickel mass transfer in sodium

Thank you for your attention

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Upcoming Webinars

Date	Title	Presenter
28 October 2021	Metal Fuel for Prototype Generation-IV SFR : Design, Fabrication and Qualification	Dr. Chan Bock Lee, KAERI, Republic of Korea
18 November 2021	Geometry Design and Transient Simulation of a Heat Pipe Micro Reactor	Dr. Jun Wang, University Of Wisconsin, Madison, USA
15 December 2021	Development of an austenitic/martensitic gradient steel by additive manufacturing	Dr. Flore Villaret, EDF, France