РОСАТОМ

ГОСУДАРСТВЕННАЯ КОРПОРАЦИЯ ПО АТОМНОЙ ЭНЕРГИИ «РОСАТОМ»

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Current state of development and justification of fast neutron reactor with lead coolant BREST-OD-300

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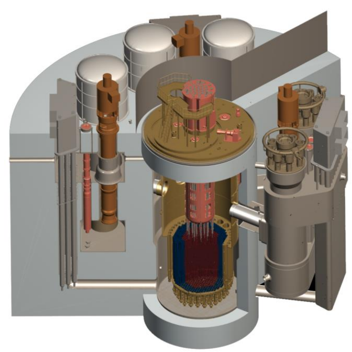
Target requirements for commercial reactors with lead coolant

Elimination of NPPs accidents requiring population evacuation, much less resettlement

Establishment of CNFC for full utilization of energy potential of natural raw uranium – <u>use of VVER Pu for starting loading</u>

Progressive approximation to radiation-equivalent (in relation to natural raw materials) RW disposal – <u>at the operating stage after development of fuel with MA</u>

BREST-OD-300



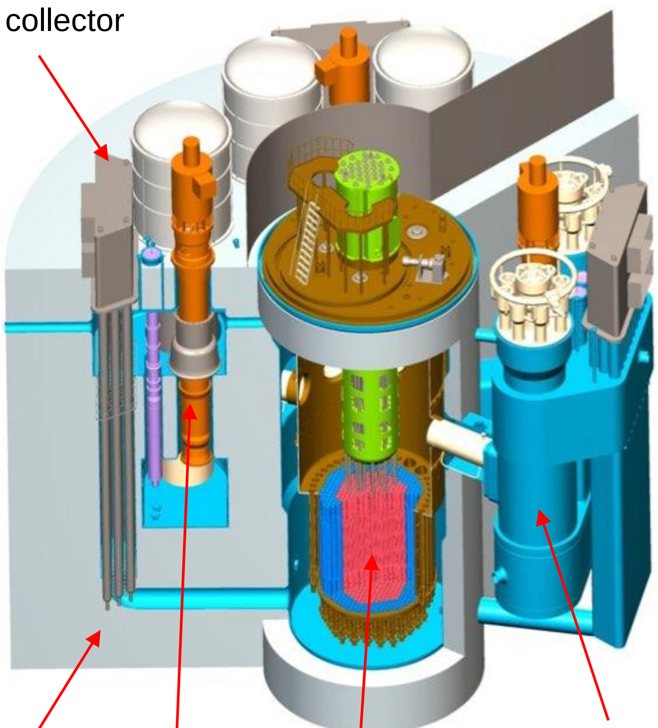
Commercial reactor

Marketability against other types of power generation - <u>demonstration of technology potential</u>

Technological enhancement to non-proliferation regime - <u>no blanket,</u>
<u>no Pu extraction during SNF reprocessing,</u>
<u>on-site NFC</u>

Design concept of BREST-OD-300

Preferential use for safety: neutron-physical and physico-chemical properties of fuel, coolant, materials, as well as design solutions that allow to fully realize these properties.



Mixed nitride fuel with high density and thermal conductivity, allows to ensure full reproduction of fuel in the core (core reproduction ratio ~ 1.05) and compensation of reactivity at fuel burnout.

Lead coolant with high boiling point, low activated, not entering into violent interaction with water and air in case of circuit depressurization.

Integral layout in combination with multi-layer metal-concrete vessel (no coolant escape to beyond the vessel) to exclude coolant losses.

No shutoff valves in the primary circuit – no circulation can be lost. A coolant circulation pattern with a free level difference – circulation is safely continued during loss of power.

Passive emergency cooling system with natural air circulation and heat removal to the atmosphere.

Specific indicators of BREST-OD-300

Thermal power, MW	700
Steam output, t/h, not less than	1500
Maximum neutron flux in the core, $cm^{-2}s^{-1}$	3.5×10^{15}
Fuel	(U-Pu)N
Fuel loading, t / FAs	20.8 / 169
Burnup at a point (annual refuelings): - for 6 % (~60 FAs), t - for 9 % (~35 FAs), t	 ~7.2 ~4.2
Maximum burnup, % h.a.	up to 10
Maximum damaging dose at the fuel cladding at a burnup of 9 % h.a., dpa	up to 140
Number of circulation loops	4

Maximum (hydrostatic) pressure of primary coolant, MPa	1.17
Cover gas pressure (argon) above the coolant level, MPa: - during normal operation - maximum	 ~ 0.104 0.2
Average mixed temperature of lead coolant in core inlet/outlet, °C	420/535
Secondary working medium	Water (steam)
Secondary coolant parameters (water-steam): - SG inlet pressure, MPa - SG outlet pressure, MPa - SG inlet temperature, °C - SG outlet temperature, °C	 18.5 17 505 340
Efficiency, %	43.5
Design life, years, not less than	30

Current status of BREST-OD-300 development

Research activity

Development activity

We are here
TRL 5 (6)

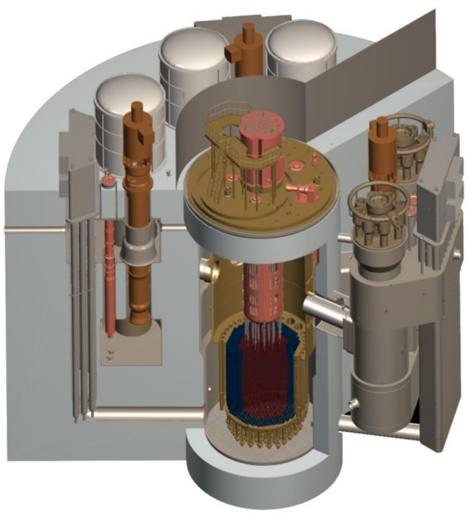
- Preliminary stages (technical proposal, conceptual design)
- Specification
- Engineering design (final solutions)
- Working design documentation, operational documentation, pilot model
- Correction of documentation (character O, O1)

Readiness for serial production or manufacturing of pilot product

- Preliminary research and solutions

- Mockup tests
- Technology development
- Verification of software tools

- Preliminary acceptance tests
- Development of federal rules, regulations and standards (if necessary)
- Construction licensing



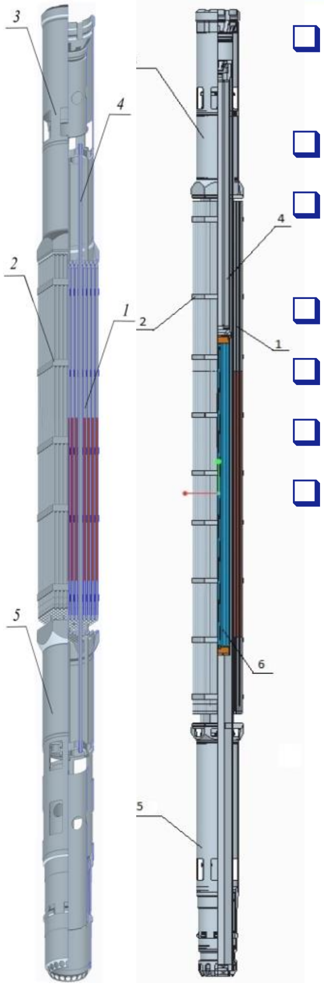
PDEC: stages of construction

Four stages for PDEC (Pilot and Demonstrational Energy Complex) construction and commissioning:

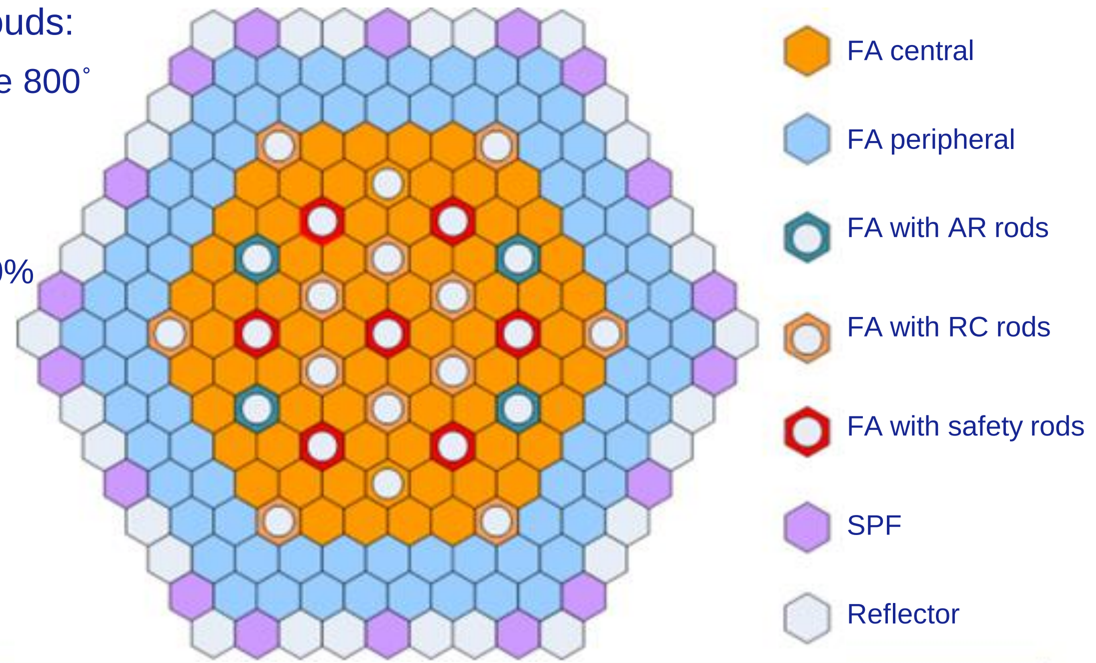
- Buildings and structures of the **fuel fabrication module (FFM)** (I stage)
- Buildings and structures of the **BREST-OD-300 power unit** (II stage)
- Buildings and structures of the **reprocessing module** (III stage)
- Turning of FFM for **fuel re-fabrication** from SNF reprocessing products (IV stage)

Reactor core

- Full fuel reproduction in the core – fuel density 12.3 g/cm^3; core reproduction coefficient (no blanket) is ~ 1.05; the ability to work with MA
- Small reactivity margin in the power range 30-100% ~ 0,65 β_{eff} (w/o Np-effect)
- With the passage of steam bubbles through the core, the maximum realizable reactivity effect $\Delta\rho_{max} < 0,5\ \beta_{eff}$
- Average core power density – 175 MWt/m^3
- Maximum linear power along the core – 420 Wt/cm
- Two radial sub-cores for regulation of radial power distribution and the coolant flow
- Hexagonal FAs without shrouds:
  - cladding temperature no more 800° at overlapping even 7 FAs coolant section
  - reduction in FA metal consumption by more than 30%
  - using of manufacturing experience of VVER FAs



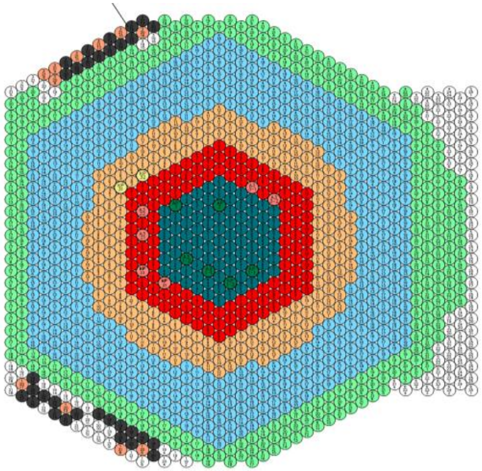
1 – Fuel pin; 2 – Spacer grid; 3 – FA head;
4 – Supporting tube; 5 – FA tail; 6 – Control rods



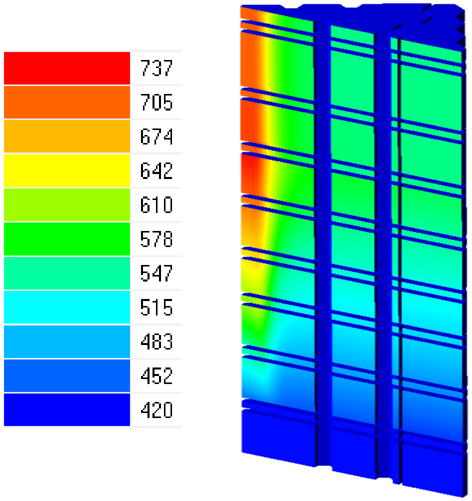
Core justification

- Heat transfer coefficient in a typical fuel bundle was experimentally obtained under conditions of LMC
- Experimentally obtained intercell mixing coefficients in bundles and on the border of fuel pin bundles
- Carried out neutronic experiments with nitride fuel at BFS, deviation of experimental criticality parameters from calculated – less than 0.2 % Δk/k
- Code validation was conducted on the basis of obtained data

LMC-stand

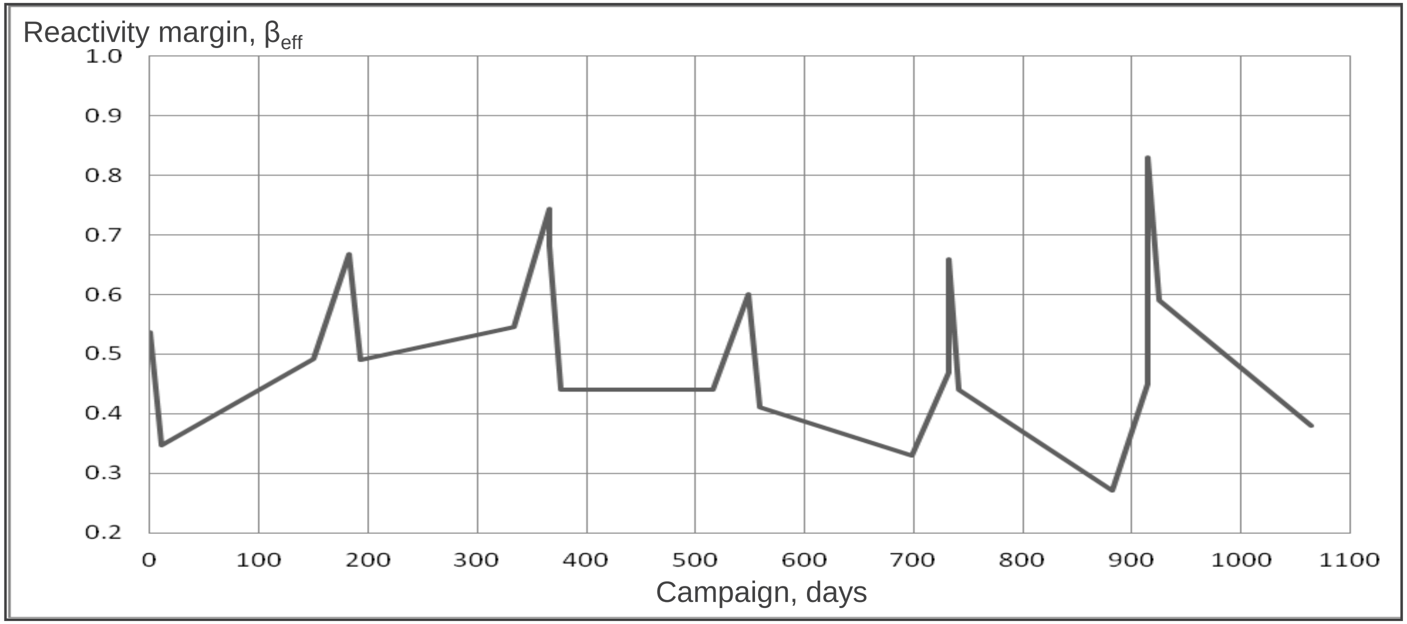


Benchmark model of BREST core. BFS-113 assembly with nitride



Surface temperature distribution of fuel rods in the core during the overlap of the 7 FAs

Neutron-physical characteristics of the core



Parameter	Value, β_{eff} (% $\Delta k/k$)
Reactivity margin on power	0.65 (0.24)
Temperature and power effect of reactivity	1.21 (0.44)
Maximum reactivity margin (“cold” state)	1.85 (0.68)
Worth of the Passive Feedback System (PFS)	0.63 (0.23)
Worth of all Safety Rods (SR)	7.4 (2.7)
Worth of group: 4 AR + 14 CR	13.9 (5.12)

Justification of core elements

Full-scale FA mock-up

- FA mockups (all types) were produced in the industrial conditions

- Strength characteristics were obtained for FA components and FA mockups as a whole

- Vibrometric and vibrostrength characteristics of FA mockups were obtained

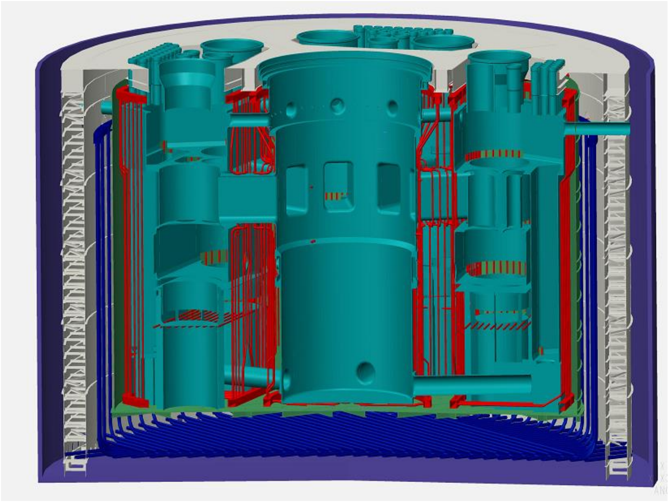
- Hydraulic characteristics of FA mockups were obtained (in water and lead flow)

- Reactor tests of the experimental FAs in BN-600 are on the way (8 experimental FAs); the fuel pins of a discharged experimental FA are on the postreactor investigations.
Max. burn up - 7.4% h.a.
All fuel rods retained tightness (over 400 fuel elements).

- Reactor tests in BOR-60 are on the way (7 experimental FAs)

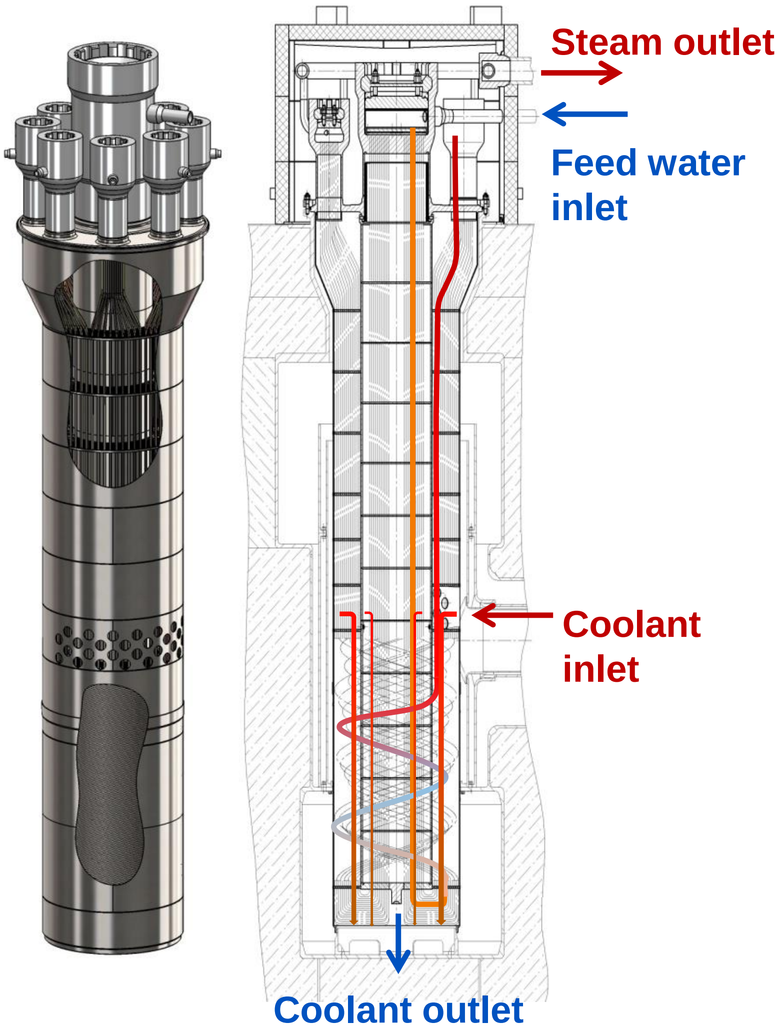
FA mock-up with a retort for testing

Reactor vessel



- Properties of the high-temperature concretes have been experimentally obtained at operating temperatures and under irradiation; chemical inertness of the coolant with respect to concrete has been shown
- Cumulative report on metallic structural materials including welds has been issued; the materials have been put into manufacture
- Codes for thermal and structural integrity-related tasks have been validated
- Localizing function of concrete has been confirmed computationally using mockups
- The assembly and installation procedure has been conceptually developed

Steam generator (1/2)



➢ Required thermal hydraulic steam generator parameters have been substantiated. Stable operation limit has been determined – not less than 15 % of the flow rate

➢ Cumulative reports with material properties have been issued

➢ Structural integrity of SG elements has been computationally verified for all modes of operation

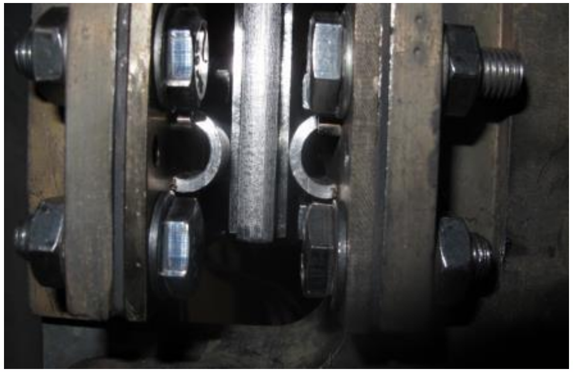
➢ Heat-exchange tubes have been put into manufacture

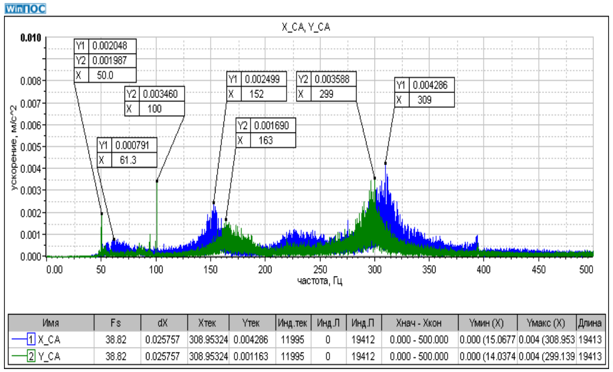
➢ Absence of dependant failure in case of one tube rupture has been experimentally demonstrated. With the passage of steam bubbles through the core, the maximum realizable reactivity effect $\Delta\rho_{max} < 0,5\ \beta_{eff}$

➢ Neutral water chemistry has been substantiated allowing for reduction of deposition formation during SG operation. A technology for heat-exchange SG tube cleaning has been developed

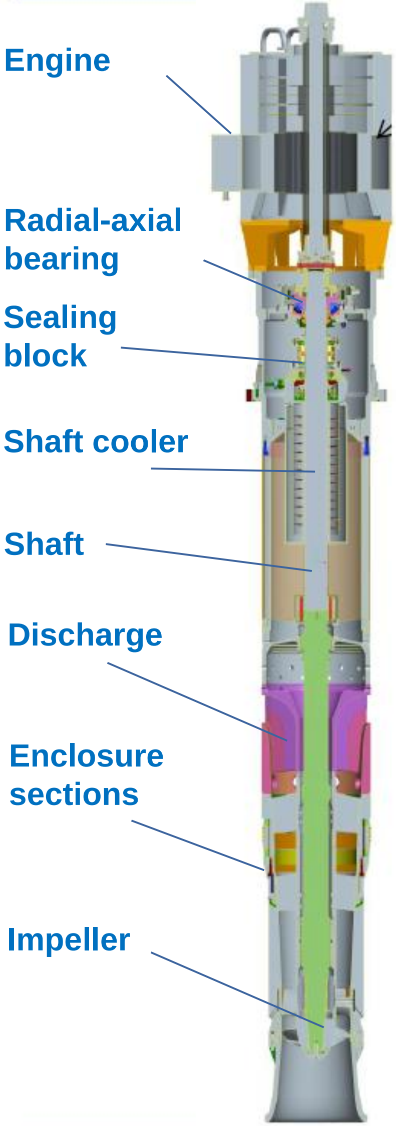
Steam generator (2/2)

- Confirmed the fulfillment of thermal-cyclic strength conditions for heat-exchange tubes and welds to a tube sheet
- Experiments have been carried out to substantiate the increase in corrosion under the conditions of water, steam and lead coolant
- Negligible influence of lead on creep rate in lead under loads typical for steam generator has been demonstrated
- A series of tribological tests on HET-spacer grid friction joints has been carried out. Physical and mechanical model has been developed. The calculations using the model has confirmed the 30-year life. A bench is developed for full-scale study of vibration characteristics

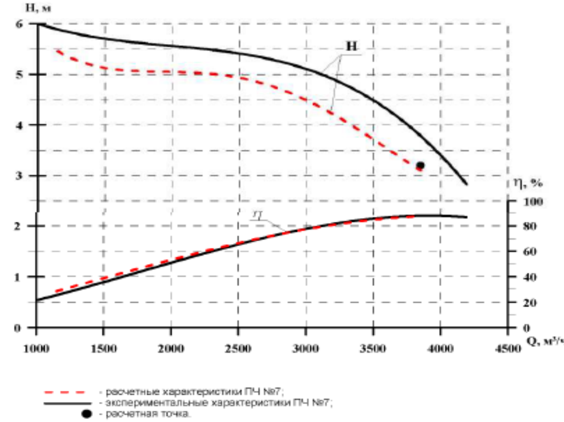




Main circulation pump



- ➢ Flow part has been optimized using the medium-scale water and lead test benches
- ➢ Obtained required head-flow curve ensuring pump operation within 30-100% range



- ➢ Bottom radial bearing has been developed and manufactured
- ➢ Endurance tests of the bearing using lead are conducted (with 30% of the design life achieved)
- ➢ End gas seal has been designed. Manufacture for bench testing has been started
- ➢ Manufacture of a pump prototype has been started
- ➢ Preparation is carried out to build a full-scale bench for MCP testing in lead
- ➢ Strength calculations have been carried out based on structural material properties from the cumulative reports

Other elements

- CPS actuator prototype tests have been competed with positive outcome
- Primary converters of parameters of the primary circuit are manufactured and tested
- Engineering design of the reactor automated inspection and control system has been developed; a bench had been developed, which was used to demonstrate operation stability CPS channel regulators during various transients
- Designed and tested steam generator safety system fittings
- Endurance testing of coolant quality system components are conducted
- Testing of the elements of the fuel element cladding control system is completed.

Research of output of FP and AP out of the coolant

- Two out-of-reactor loop units and one in-reactor loop unit were established
- Data on outputs of ^{210}Po, ^{131}I, ^{115m}Cd, ^{110m}Ag, ^{123m}Te, ^{210}Hg, ^{124}Sb were obtained
- Experimental data on mass transfer of gaseous (Kr, Xe) and volatile (I) fission products from the nitride fuel into the gas environment (He) for the code verification were obtained
- Experiments are going on…

Thermal hydraulic calculations justifying design solutions and safety

Distribution of temperature and velocity modulus in vertical section intersecting the axis of one of the MCPs (1^{st} second of loss of power supply transient)

- Computational justification of circulation in the primary circuit has been carried out using the 3D codes
- Calculations have been performed both for normal operation and anticipated operational occurrences
- In general, the 3D calculations show that the anticipated operational occurrence calculations performed using the channel analysis codes give conservative (higher) temperatures
- Validation of 3D codes is nearing completion

BREST-OD-300 safety requirements (1/2)

The most conservative scenarios are considered:

- Unauthorized entering of the full positive reactivity margin.
- Violation of the heat sink from the core at complete electric breakdown.

Target requirements for the BREST-OD-300 to the consequences of scenarios:

- Elimination of fuel and cladding melting.
- Elimination of coolant boiling.
- Maintaining the integrity of the circulation loop.

Requirements for off-site radioactivity emissions:

- Eliminate the need for protective measures such as evacuation and resettlement of the population.

BREST-OD-300: specific safety criteria (2/2)

Limit of safe operation by temperature of fuel rod cladding:

Maximum temperature of fuel rod cladding 740 °C – 1000 °C depending on the time

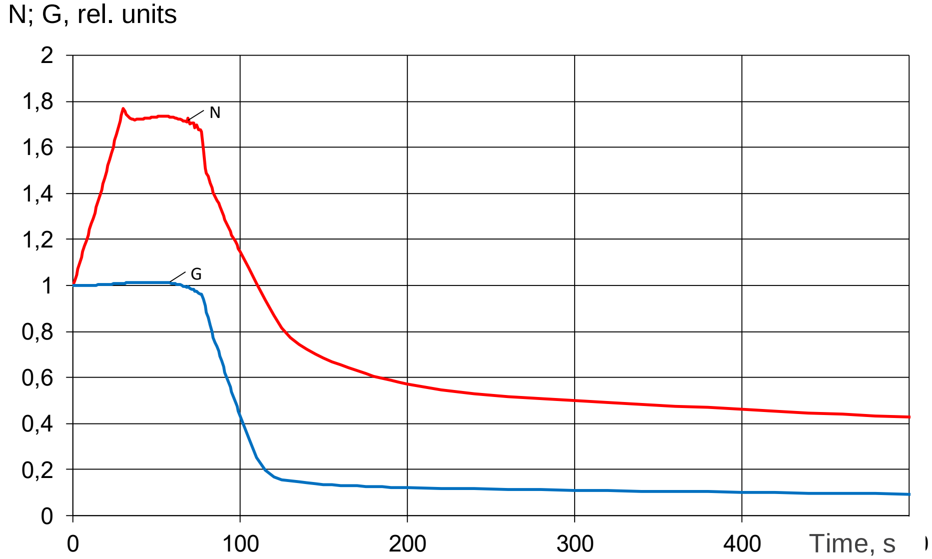
Reference data:

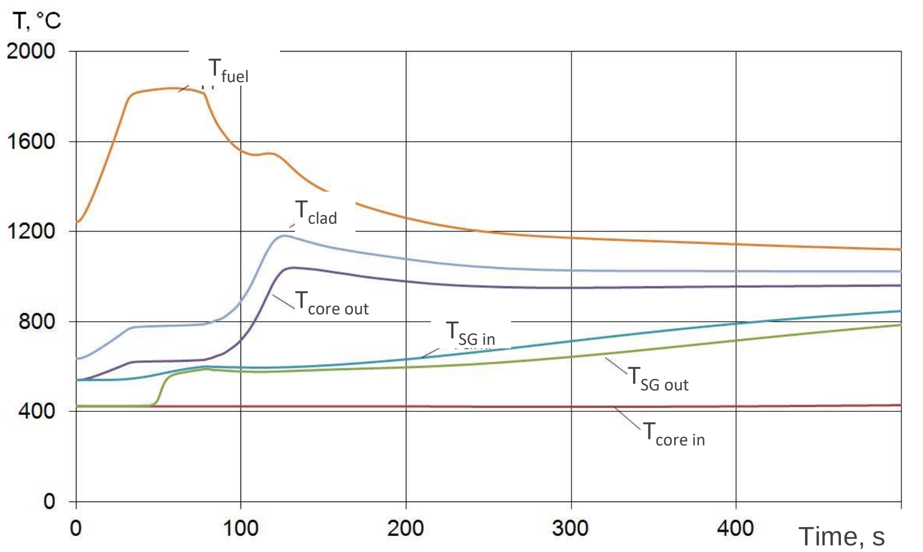
Fuel melting point > 2700 °C

Cladding melting point ~ 1500 °C

Boiling point of lead 1750 °C

Entering of the full reactivity margin at nominal power (1/4)





Scenario:

- initiating event – entering of the **full reactivity margin at nominal power 0.65 β_{eff}** with realistic design speed (in 30 seconds)
- active protection systems fail (total number of failures – 11, probability of realization $2.8 \cdot 10^{-9}$)
- pump shutdown on setpoint temperature of the coolant inlet at SG – 520 °C
- as a consequence - loss of forced cooling of the core and transition to Pb natural circulation
- passive feedback system (PFS) works due to the decrease in coolant pressure (input negative reactivity – 0.63 β_{eff})

Calculation results (peak values):

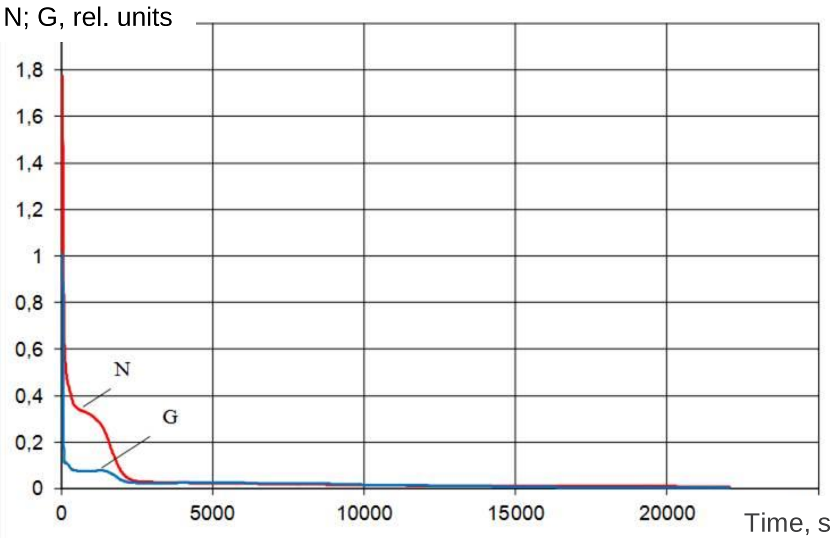
- Maximum reactor power – 1.8 N_{nom}
- Maximum fuel temperature ~ 1850 °C
- Maximum cladding temperature ~ 1200 °C

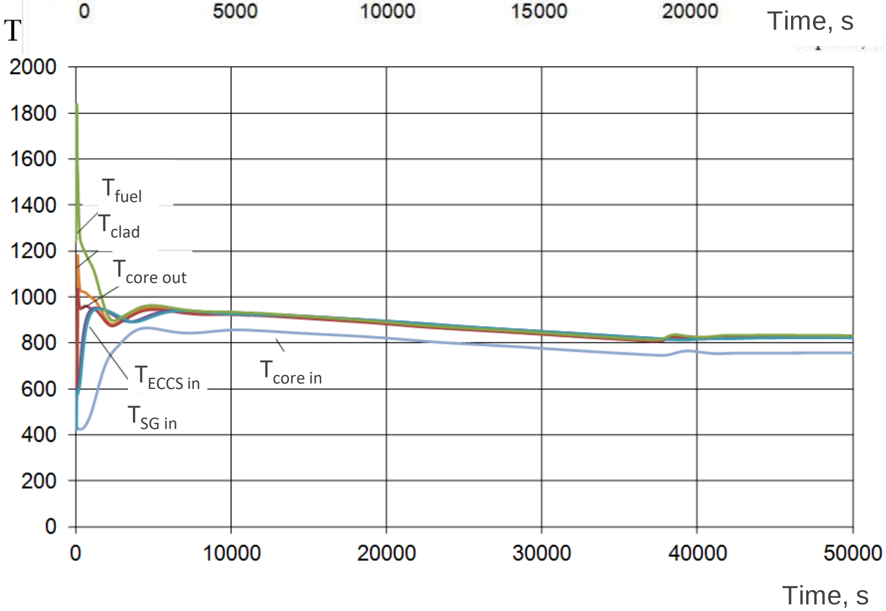
Entering of the full reactivity margin at nominal power (2/4)

Event	Actuation set point	Failures	Consequences
1) Actuation of Fast Controlled Power Reduction-4	Mismatch of power and coolant head level	**Failure**	Increase of the core power and temperature
2) Actuation of EPR	110 % Nnom	**Failure**	Increase of the core power and temperature
3) Actuation of EPR	115 % Nset	**Failure**	Increase of the core power and temperature
4) EPR	Mismatch of power and coolant head level	**Failure**	Increase of the core power and temperature
5) Actuation of Fast Controlled Power Reduction-4	580 °C at the reactor central core outlet	**Failure**	Increase of the core power and temperature
6) EP	120 % Nnom	**Failure**	Increase of the core power and temperature
7) EP	600 °C at the reactor central core outlet	**Failure**	Increase of the core power and temperature
8) Actuation of SG safety device, isolation steam valves and feedwater isolation valves, shutdown of feedwater pump-2	620 °C at the reactor central core outlet		Increase of the core power and temperature, termination of heat removal from the primary circuit
9) EPR	Signal for closure of all feedwater isolation valves	**Failure**	Increase of the core power and temperature
10) MCP shutdown	520 °C at SG outlet		Transition to natural circulation, decrease in heat removal from the core, insertion of negative reactivity by the PFBS
11) EPR	More than 2 RCPs tripped	**Failure**	Heating of the core and the primary circuit
12) Normal cooldown system	430 °C at the ECCS outlet	**Failure**	Primary circuit heating
13) ECCS	450 °C at the ECCS outlet	**Failure of two ECCS loops**	Residual power removal

Probability of implementation of the scenario: entering of the full reactivity margin when applying multiple failures of systems and equipment – $2.8 \cdot 10^{-9}$

Entering of the full reactivity margin at nominal power (3/4)
Steady-state values





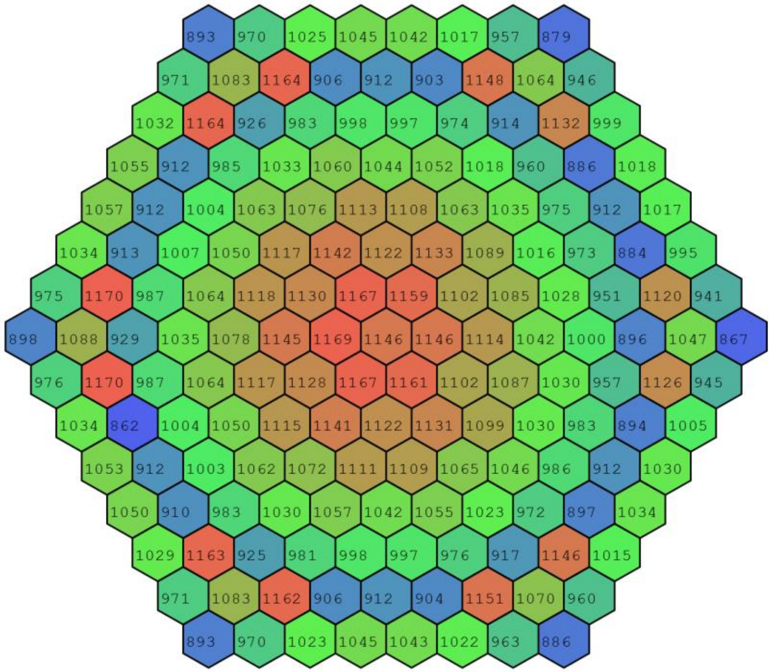
Scenario development - shutdown (power reduction) of the reactor due to:

- temperature equalization of fuel, construction materials and coolant – increase in average temperatures and Doppler effect
- temperature reactivity effect due to expansion of the core with increasing coolant temperature at the core inlet

Results of calculations of steady-state values:

- maximum fuel temperature ~ 950 - 800 °C
- maximum cladding temperature ~ 950 - 800 °C

Entering of the full reactivity margin at nominal power (4/4)



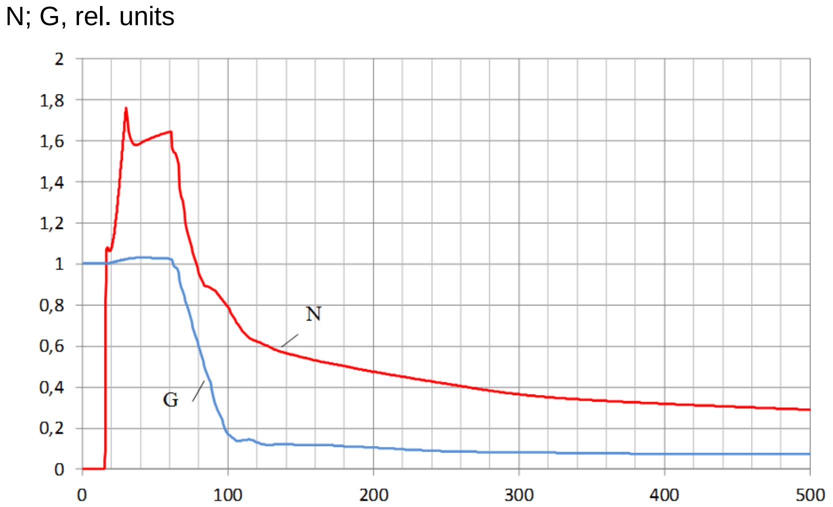
Map of maximum temperature distribution of fuel cladding (at 125 s)

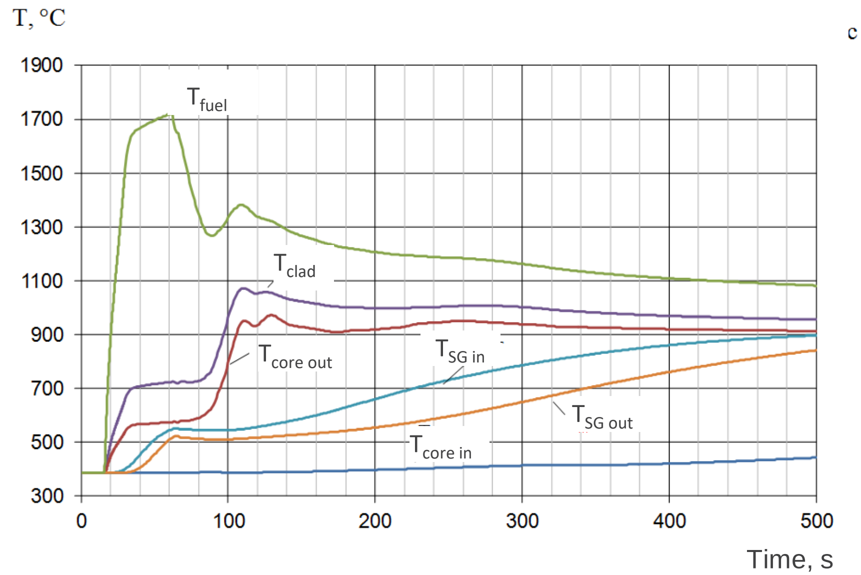
Outcome:

- melting of fuel and cladding, boiling of the coolant - does not occur
- circulation circuit integrity ensured
- in the part of FAs (distribution - in fig), the maximum design limit on the cladding temperature is exceeded
- the level of possible damage of fuel rods by type of gas leakage – 7 %
- **emissions into the atmosphere for all normalized radionuclides do not reach the reference level per day**

Nuclide	Emission, Bq / day	Reference level, Bq / day
Cs-134	$1.9*10^{6}$	$2.5*10^{6}$
Cs-137	$3.6*10^{6}$	$5.6*10^{6}$
I-131	$5.2*10^{6}$	$5.0*10^{7}$
Kr-85	$7.3*10^{8}$	
Xe-133	$4.6*10^{10}$	
Xe-133m	$9.6*10^{8}$	$1.9*10^{12}$
Xe-135	$1.3*10^{10}$	
Sum Kr+Xe	$6.1*10^{10}$	

Entering of the full reactivity margin at “cold” state





Scenario:

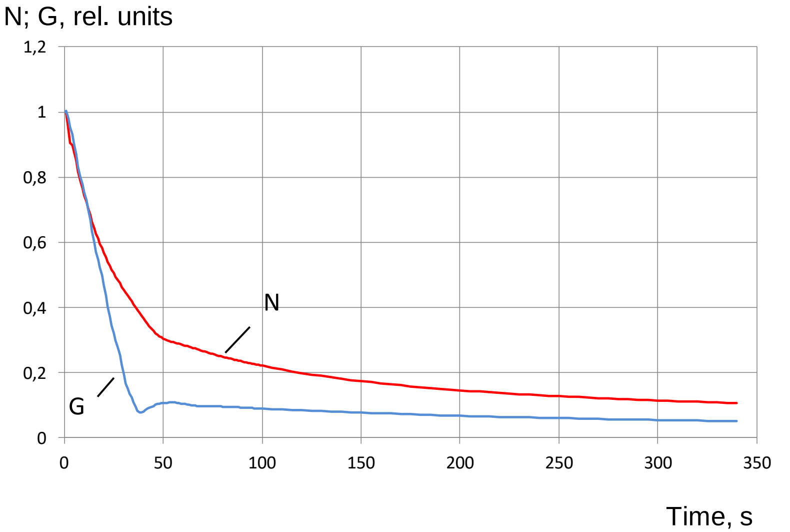
- initiating event – entering of the **full reactivity margin at “cold” state 1.85 β_{eff}** with realistic design speed
- at the initial stage feedbacks do not appear due to low power level (~ 20 s)
- further development of the scenario as a whole repeats the previous

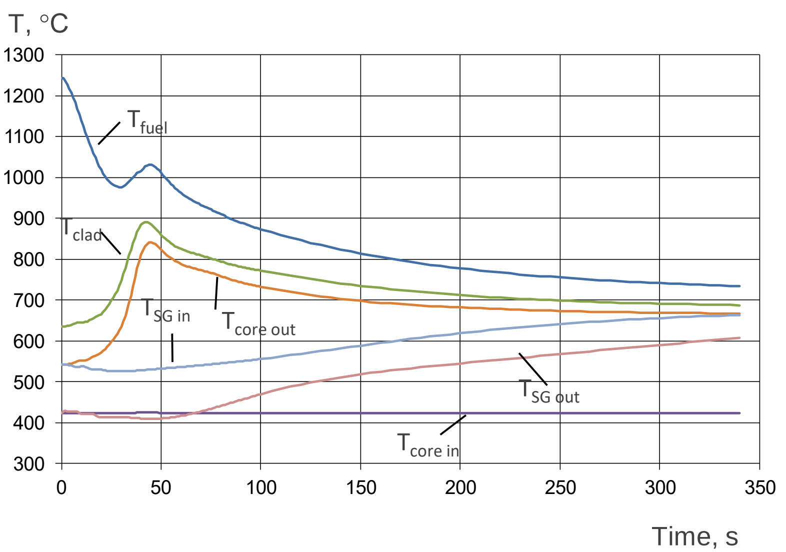
Calculation results (peak values):

- Maximum reactor power – 1.75 N_{nom}
- Maximum fuel temperature ~ 1700 °C
- Maximum cladding temperature ~ 1070 °C

Consequences in steady state are similar to the previous scenario, there is no excess of emissions above the established limits.

BREST-OD-300. Complete electric breakdown of the unit (1/2)





Scenario:

- the greatest deterioration of the conditions of heat removal from the reactor core - total blackout
- shutting down four MCPs and stopping the supply of feed water when operating at the nominal power
- removal of residual energy is carried out by two of the four ECCS loops (postulated failure of two emergency decay heat removal loops, postulated failure of normal cooling system)

Result:

Maximum cladding temperature of the most loaded fuel element during ~ 45 s exceed 800 °C and achieves ~ 890 °C

Melting of fuel rod cladding and fuel does not occur. The integrity of the circuit is ensured.

FP output from the reactor unit for the first day does not exceed a reference level of emissions per day during normal operation.

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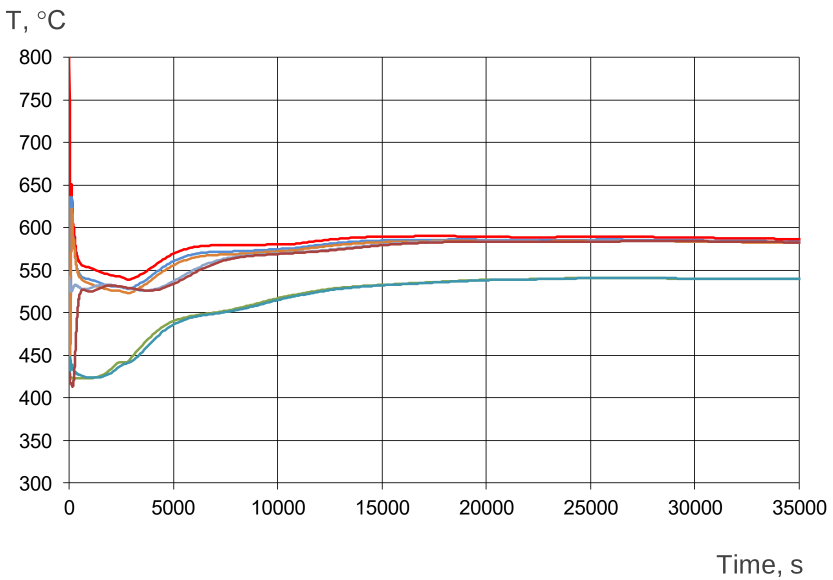
BREST-OD-300. Complete electric breakdown of the unit (2/2)



Behavior of fuel temperature, fuel cladding, lead coolant at the inlet and outlet of core and SG with prolonged cooldown.

Decay heat from the reactor is diverted by the operation of only two ECCS loops.

Result :

Melting of fuel rod cladding and fuel does not occur. The integrity of the circuit is ensured.

FP output from the reactor unit for the first day does not exceed a reference level of emissions per day during normal operation.

Key safety conclusions for the core

The core of BREST-OD-300 with core reproduction coefficient ~ 1 provides maximum reactivity margin < $0.65\beta_{eff}$ when operating at 30-100% N_{nom} power (there is no reactivity margin for reactor runaway on prompt neutrons), in the "cold" state the maximum reactivity margin (MRM) ~ 1.85 β_{eff}

At such levels of the MRM, even its complete implementation with the deterministic failure of active systems of action on reactivity:

- does not lead to melting of fuel element cladding and fuel, boiling of the coolant
- circulation circuit integrity is ensured
- emissions into the atmosphere do not reach the reference level per day
- transformation of the initial event into a nuclear accident is excluded

Complete electric breakdown of the unit also does not lead to melting of fuel element cladding and fuel, boiling of the coolant. Long-term cooldown is carried out using a passive ECCS system with natural circulation in the 1st circuit Pb coolant.

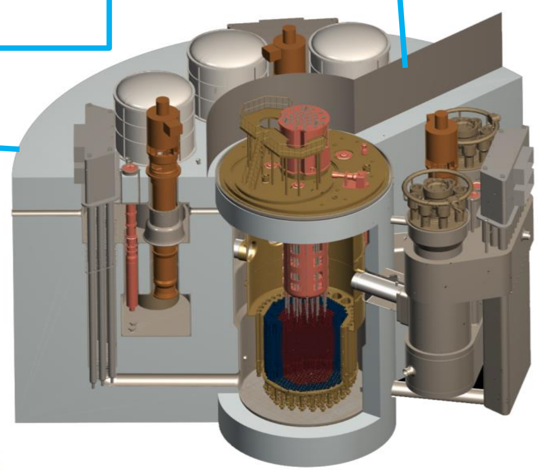
The probability of core damage (without melting of fuel) for internal reasons does not exceed $9{\cdot}10^{-9}$ 1/year, that allows to provide an acceptable level of safety in the development of energy at the reactors of BREST type.

Radiation safety justification (1/2)

Emission from reactor 4.3·10 E8 Bq for the first day operational occurrences accompanied by multiple failures for a scenario with full reactivity insertion

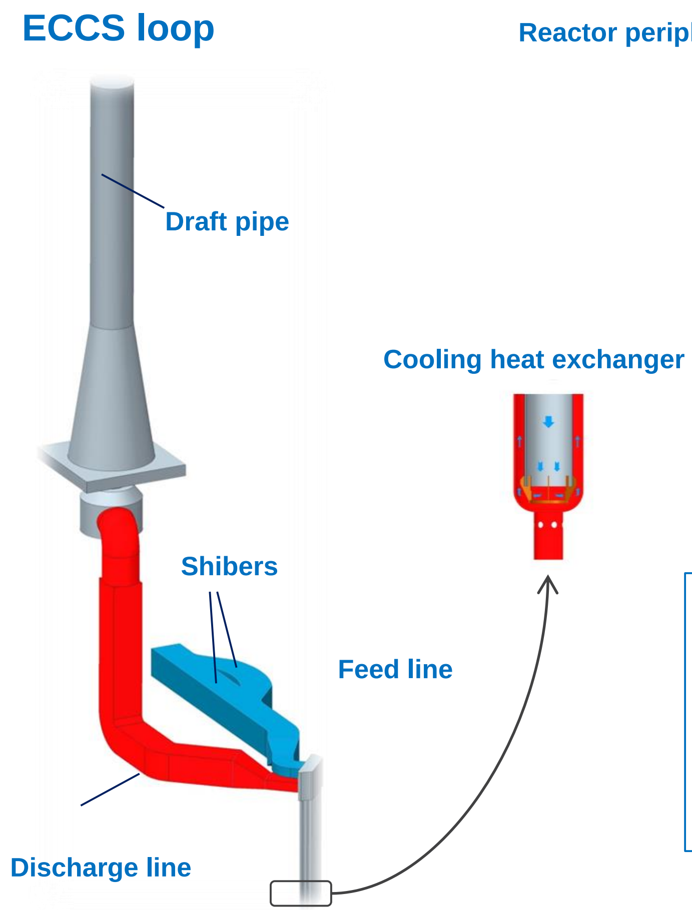
At the level of natural background radiation - less then 0,1 mkSv/h

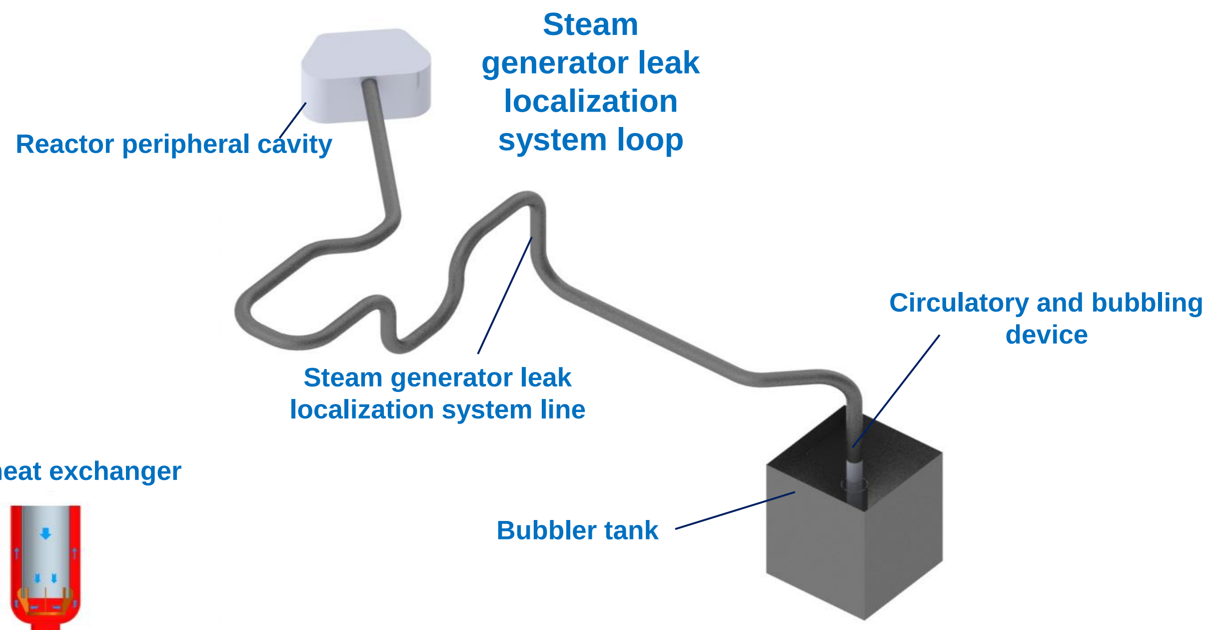
1.5 mSv for initial 10 days at imposing of depressurization of the SG and SG break localization system



Activation of removable in-vessel equipment is virtually absent (apart from that located in the central cavity).

Radiation safety justification (2/2)





Thanks to design solutions adopted in the reactor, the use of lead coolant, provision of small reactivity margin and passive safety systems, the total probability of the core damage from all initiating events based on 24 h does not exceed 9·10E-9 1/year.

Regulatory acts and standardization

- The reactor design is based on the requirements of the existing federal norms and regulations for NPPs: general regulations for safety assurance (OPB), nuclear safety rules (PBYa), radiation safety standards (NRB), etc.

- To provide development of innovative NPPs, the processes of the development of design, new federal norms and regulations and advanced software tools (computational codes) are almost parallel

- Rules for arranging and safe operation and the corresponding documentation (welding, control regulations, etc.), vessel strength calculations norms are specific

- Currently, activities in collaboration with Rostekhnadzor are ongoing on approval and implementation of the fundamental norms and regulations based on which reactor is developed

Prospects of lead-cooled reactors

- ➢ In the process of BREST-OD-300 development, **technologies** are developed, experiments are conducted, and innovative engineering solutions for equipment development are found which are useful for scaling as well
- ➢ **Computational codes** used for BREST-OD-300 can also be applied in the analysis of similar reactors, or with small adjustments in case of changes of new reactor parameters
- ➢ **Regulatory system** including the one that was updated based on the lessons learned after going through the BREST-OD-300 life-cycle stages can be applied to other reactors
- ➢ Inherent qualities can be used for other reactors **in a relatively wide power range**

Федеральная служба
по экологическому, технологическому и атомному надзору
(Ростехнадзор)

ЭКСПЕРТНЫЙ СОВЕТ ПО АТТЕСТАЦИИ ПРОГРАММНЫХ СРЕДСТВ
ПРИ РОСТЕХНАДЗОРЕ

АТТЕСТАЦИОННЫЙ ПАСПОРТ ПРОГРАММНОГО СРЕДСТВА

Регистрационный номер 405 от 08 декабря 2016 года

Настоящий аттестационный паспорт устанавливает назначение и область применения программного средства

«MCU-BR с библиотекой констант MDBBR50»,

которые указаны в разделе 2 Приложения к настоящему аттестационному паспорту.

Аттестационный паспорт предоставлен

Акционерному обществу «Ордена Ленина Научно-исследовательский и конструкторский институт энерготехники им. Н.А. Доллежаля» (АО «НИКИЭТ»),
юридический адрес: 107140, Россия, г. Москва, улица Малая Красносельская, д. 2/8.

Настоящий аттестационный паспорт действует при соблюдении условий Приложения, являющегося его неотъемлемой частью.

Срок действия аттестационного паспорта до 08 декабря 2026 года

Председатель экспертного Совета
по аттестации программных средств
при Ростехнадзоре, к.т.н. С.Н. Богдан

ETSON | EUROPEAN TECHNICAL SAFETY ORGANISATIONS NETWORK

НИКИЭТ им.Н.А.Доллежаля

Key organizations involved in the development of the BREST-OD-300

РОСАТОМ

ПРОРЫВ
ИННОВАЦИОННО-ТЕХНОЛОГИЧЕСКИЙ ЦЕНТР ПРОЕКТА ГОСУДАРСТВЕННОЙ КОРПОРАЦИИ ПО АТОМНОЙ ЭНЕРГИИ «РОСАТОМ»

РОСАТОМ

ТОПЛИВНАЯ КОМПАНИЯ РОСАТОМА
ТВЭЛ

НИКИЭТ им.Н.А.Доллежаля

ГНЦ РФ - ФЭИ

АТОМПРОЕКТ

ВНИИНМ имени А.А.Бочвара
ПРЕДПРИЯТИЕ ГОСКОРПОРАЦИИ «РОСАТОМ»

Сибирский химический комбинат
ПРЕДПРИЯТИЕ ГОСКОРПОРАЦИИ РОСАТОМ

ЦКБМ

ЭЛЕМАШ
ПРЕДПРИЯТИЕ ГОСКОРПОРАЦИИ “РОСАТОМ”

НЗХК
ПРЕДПРИЯТИЕ ГОСКОРПОРАЦИИ «РОСАТОМ»

ИБРАЭ

ОАО “Машиностроительный завод "ЗИО-ПОДОЛЬСК"”

ОАО “Инжиниринговая компания "ЗИОМАР"”

КБСМ
КОНСТРУКТОРСКОЕ БЮРО СПЕЦИАЛЬНОГО МАШИНОСТРОЕНИЯ
АКЦИОНЕРНОЕ ОБЩЕСТВО

НИИАР

РОСЭНЕРГОАТОМ
БЕЛОЯРСКАЯ АЭС

ОКБМ АФРИКАНТОВ
ПРЕДПРИЯТИЕ ГОСКОРПОРАЦИИ «РОСАТОМ»

ИРМ
ИНСТИТУТ РЕАКТОРНЫХ МАТЕРИАЛОВ

Conclusion

1. Based on the concept of inherent safety, technical solutions have been developed.
2. The design approaches to the reactor equipment have been experimentally justified based on mockups of the equipment and its components and by means of computational justification with regard for the lead coolant effects; prototypes are being tested.
3. The reactor core design approaches have been validated by positive results of irradiation experiments, hydraulic and vibration experiments, analysis of neutronics using the certified codes.
4. The thermal-hydraulic calculations performed using CFD codes have shown that during anticipated operational occurrences with multiple failures superimposed, safe operation limits in terms of fuel and cladding temperatures are not exceeded and the localizing function of the reactor unit vessel is ensured.
5. The results of the radiation safety analysis have confirmed the implementation of target indicators, including no need for evacuation and resettlement of the public outside the site during anticipated operational occurrences with multiple failures.
6. The experiments formed the basis of the developed regulatory framework.
7. The BREST-OD-300 unit design is in the process of licensing with Rostekhnadzor.