ROSATOM

VII International Forum ATOMEXPO

atomexpo 2016

State atomic energy corporation “Rosatom”

PROJECT «PRORYV» (Breakthrough)

Results achieved and further development of closed nuclear fuel cycle technologies (Federal Target Program “Nuclear Power Technologies of the New Generation” and the concept of the Federal Target program “Nuclear Power Technologies of the New Generation-2”)»

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Nuclear power in the world today

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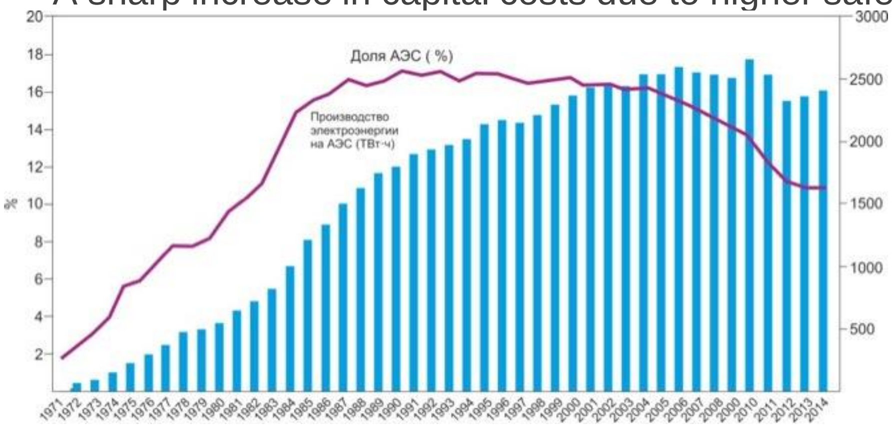
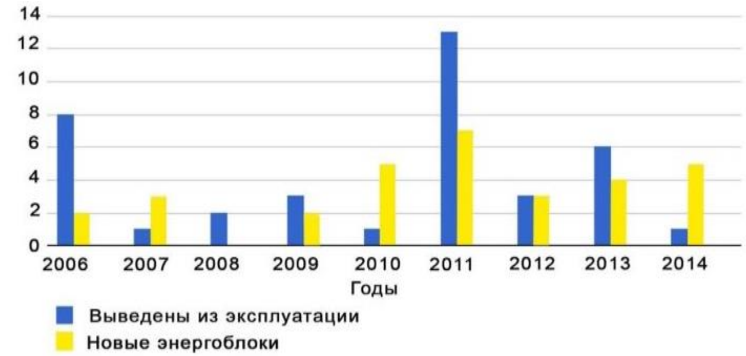
- In the second half of the 20th century the rapid growth of power production from NPPs in the 80s was superseded by a period of stagnation, followed by a decline in the early 2000s (from 18% to 10% of overall share)
- NPP unit decommissioning exceeds new nuclear build, which is surpassed by alternative energy in terms of scale and commissioning (about 30-50 GWt annually for wind power alone)
- 6 severe accidents over the past 60 years. Fukushima – damage > 100 billion dollars. Several countries reject nuclear energy development (Germany, Switzerland, Belgium, Italy)
- A sharp increase in capital costs due to higher safety requirements





The irradiated nuclear fuel (INF) problem transitioned from being temporarily set aside to requiring immediate response (increased interest in closed NFC, INF storage pool capacity overload, buildup of nuclear materials targeted for reprocessing)

- Global INF inventory reached 400 000 t, annually increases by 10-12 kt/year, approximately 120 000 t reprocessed
- Enhanced requirements for INF storage in terms of timelines and risks**

***The EPA requires the DOE to prove that the Yucca Mountain site can safely store INF, taking into account the aftermath of possible earthquakes, volcanic eruptions, climate change and cask corrosion for a period of 10 000 to 1 000 000 years*

Project «PRORYV» objectives

- Eliminating the threat of severe accidents, requiring evacuation of local population
- Complete utilization of uranium resources
- Gradual advance towards radiation-equivalent (in relation to the initial natural raw material) radwaste disposal
- Reinforcing non-proliferation through technological means
- Providing competitive nuclear power in comparison to the cost of energy of other generation technologies

Chernobyl, Fukushima – a pattern or random occurance?

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1955 EBR-1 45% of the fuel melted (FR)

1957 Windscale, graphite fire

1957 Chelyabinsk – 70, Kyshtym accident

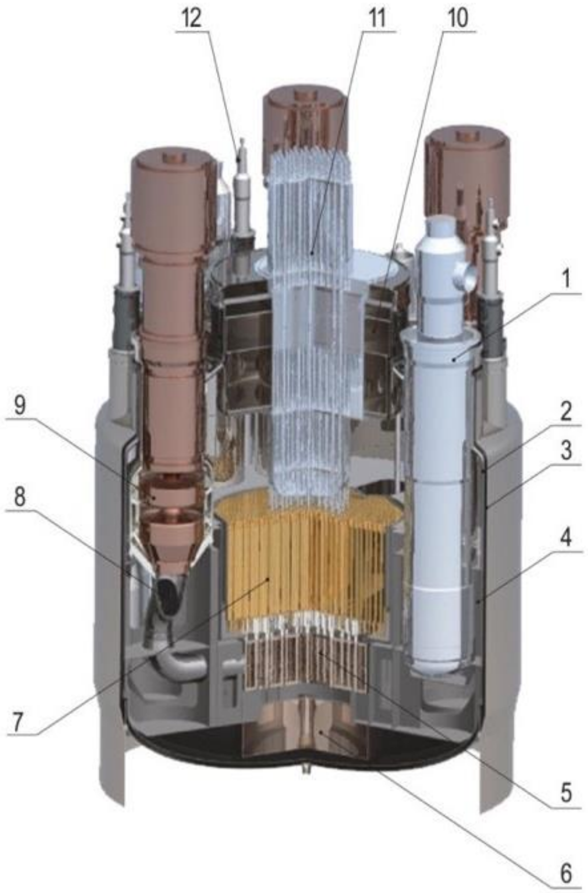
1979 TMI – fuel melted on Unit 2

1986 Chernobyl NPP – prompt critical excursion

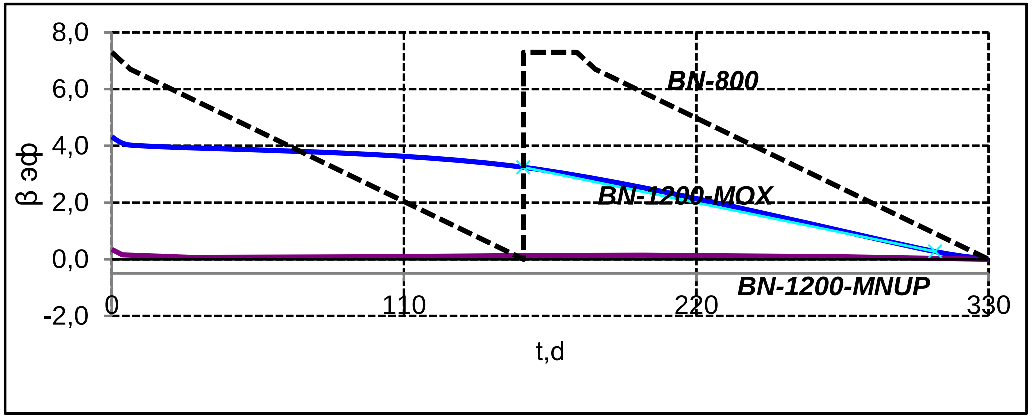
2011 Fukushima – fuel melted in three units and INF storage area

Project «PRORYV»: eliminating loss of cooling and prompt-neutron reactor power excursion accidents

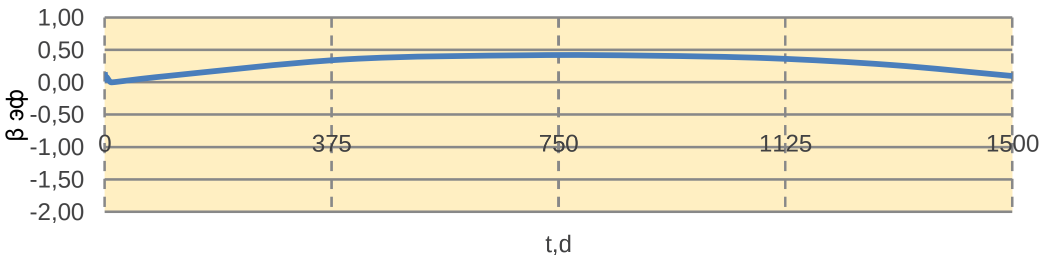
Integral reactor design – allows localization of coolant leaks in the reactor vessel and eliminates core uncovery. Thus, severe accidents leading to the evacuation of local population are excluded.



High-density fuel of equilibrium composition – eliminates reactivity induced accidents



BREST reactor reactivity margin with MNUP fuel



Reactor testing of experimental assemblies with MNUP fuel

BN-600 reactor

11 experimental fuel assemblies loaded for irradiation (over 400 fuel elements)

4 experimental fuel assemblies loaded for irradiation in 2016 (12-15)

Irradiation of 4 experimental fuel assemblies was completed

All assemblies maintained hermetic integrity

BOR-60 reactor

10 dismountable experimental fuel assemblies installed for irradiation, of which:

Irradiation of 1 experimental fuel assembly was completed (OU-1).

OU-5 was unloaded due to destruction of the fuel elements on account of overheating

MIR reactor

An instrumental fuel assembly consisting of 7 fuel elements was loaded for irradiation with in-core monitoring of fuel center temperature, fuel rod pressure and fuel stack elongation.

Post-irradiation experimentation results of combined experimental fuel assembly KETVS-1 БN-600 (with 5.5% h.a. burnup)

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- **Maximum cladding deformation in nitride fuel assemblies is lower than in oxide fuel**
- **Gas release from MNUP fuel is 3 times lower than from oxide fuel. Higher helium concentrations were observed in MNUP fuel due to (n, α)-reactions on nitrogen**
- **Initial post-irradiation experiments showed** an absence of fuel mass-transfer, distance-wire rupture, fuel element interaction of the bundle and the presence of local gaps between the cladding and pellets.
- A comparison of the experimentation data with post-test and project (pre-test) calculations was performed in relation to:
  - Profilometry and fuel assembly and stack elongation;
  - Gas release.

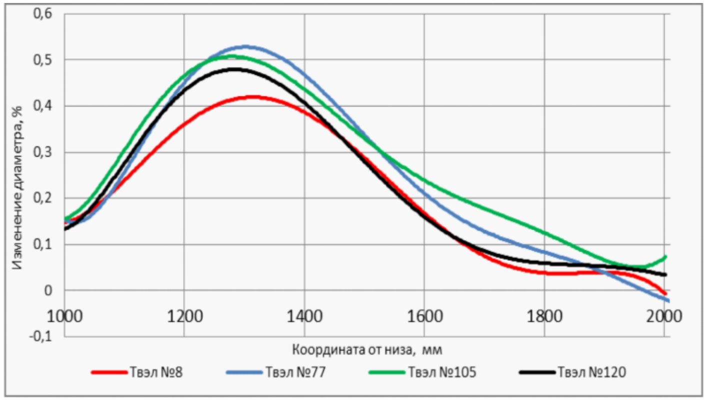


Рисунок 32-140 – Профилограммы 4-х оксидных твэлов - соседей твэлов со СНУП топливом КЭТВС-1

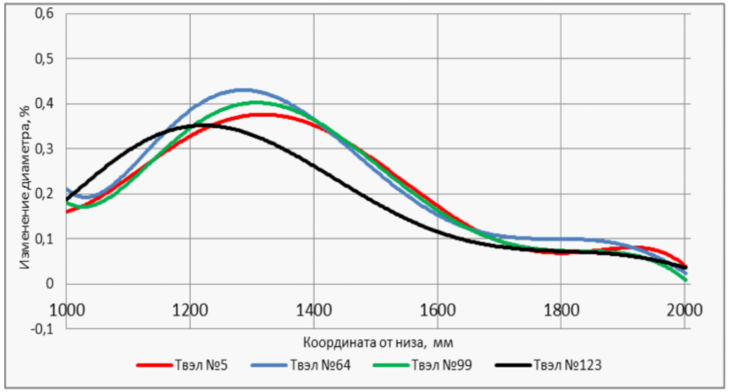


Рисунок 32-141 – Профилограммы 4- твэлов со СНУП топливом

The figures show smoothed profile diagrams of nitride and oxide KETVS-1 fuel assemblies

Comparison results

- Conservative parameters were factored in the nuclear design calculations based on accepted fuel assembly performance criteria
- Changes in MNUP KETVS-1 fuel assembly cladding diameter resulted from, for the most part, the swelling of steel.

FNR INF reprocessing technologies

Parameter	U-Pu (Np) purification coefficient from FP		Actinide extraction Pu (Am)		INF cooling before reprocessing	
	FACT	Potential	FACT	Potential	FACT	Potential
Pyro	10^3	10^6	97 % (95 %)*	99,9% (99,9 %)	1 year	1 year
Gas-fluoride tech	10^{4-6}	10^7	-	99,9% (99,9 %)	-	1 year
Hydro	10^7	10^7	99,9 % (99,9 %)*	99,9% (99,9 %)	4 years	3 years
Pyro + hydro	-	10^7	-	99,9% (99,9 %)	-	1 year
Gas-fluoride + hydro	-	10^7 **	-	99,9% (99,9 %) **	-	1 year

PH-process – combined FR INF reprocessing technology

A combined reprocessing approach allows:

- Reprocessing of high burnup INF with low a cooling period
- Reinforcing the non-proliferation regime
- FM losses during reprocessing at ≤ 0,1 %
- Recycling products, suitable for nuclear fuel fabrication
- Maintaining low volumes of HLW
- Am and Cm extraction and separating Am from Cm

Choosing the coolant

- **A lead coolant is chemically inert with air** with the oxide layer blocking any further development of the reaction.
- **A lead coolant is chemically inert with water** and completely excludes any explosive levels of hydrogen generation in the circuit.
- **A high boiling point and high thermal capacity** eliminates the possibility of accidents related to the boiling crisis phenomenon and removes the issue of a positive void coefficient of reactivity
- Lower **moderating efficiency** of the lead coolant heavy nuclei compared to sodium light nuclei
  - solves the positive void coefficient of reactivity problem;
  - allows using a wider fuel element array in the reactor core, which in turn:
    - removes the limitations for **natural circulation** intensity, which plays a key role in eliminating the possibility of accidents due to loss of cooling
    - allows for significant reduction in power needed for providing coolant flow
- **Retention of fission products** (iodine, caesium and others., except noble gasses) reduces the possibility and impact of radioactive materials escaping into the environment.
- The absence of bismuth (compared to a lead-bismuth coolant) **eliminates the problem of polonium generation** (Po-210).
- A greater abundance of lead in the earth's crust and **lower cost** when compared to bismuth.

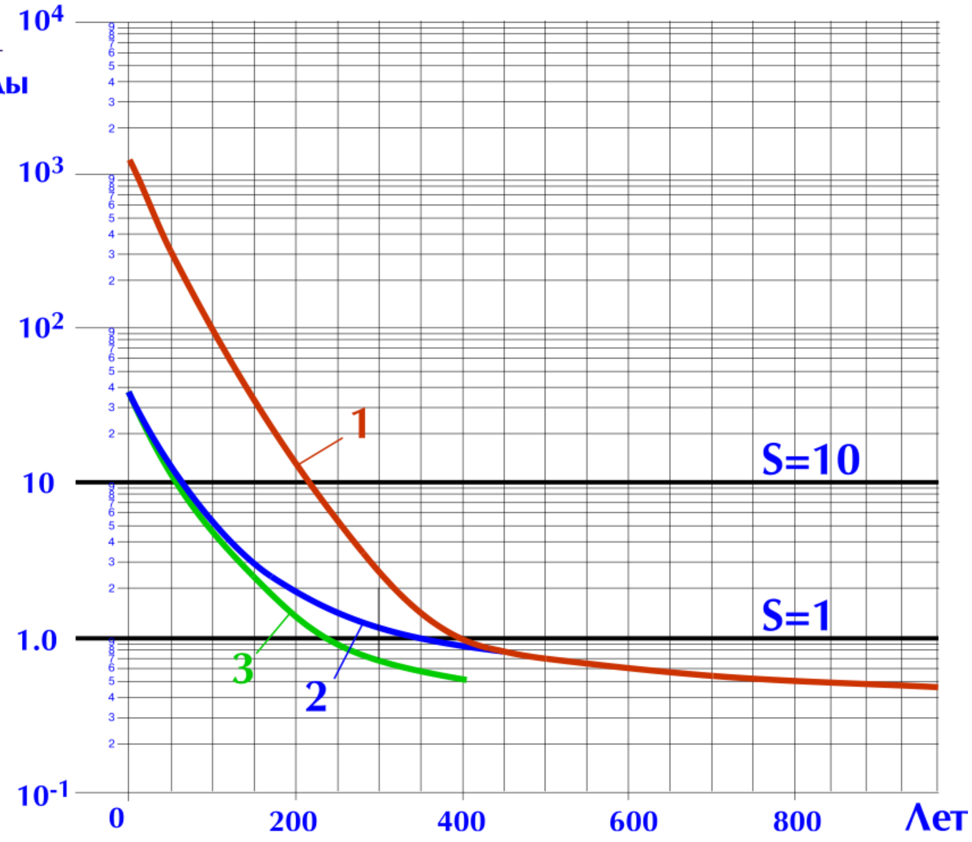
Radiation-equivalent approach to radioactive waste management

- Reprocessing INF for transferring Pu, MA (U, Pu, Am, Np, Cm) and long-lived FP (Tc, I) for transmutation in fast reactors
- Interim storage of HLW before final disposal in approximately 150-300 years in order to lower their biological hazard by a factor of 100
- Combined extraction of radium and thorium together with natural uranium for subsequent transmutation in fast reactors

Radiation-equivalent radwaste и natural uranium

$$S = \frac{A^{\text{р.-экв.}}_{\Delta\text{ВАО}}}{A_{\text{U руды}}}$$

Кривая	Доля, идущая в отходы, %				
	Sr	Cs	U	Pu	MA (Np+Am+Cm)
1	10	15	0,05	0,1	0,1
2	0,1	1	0,05	0,1	0,1
3	0,1	1	0,01	0,01	0,1



Radiation balance excluding (S=1) and including (S=10) nuclide migration, depending on the length of long-term controlled decay storage of long-lived HLW

Technologically enhancing the nuclear non-proliferation regime

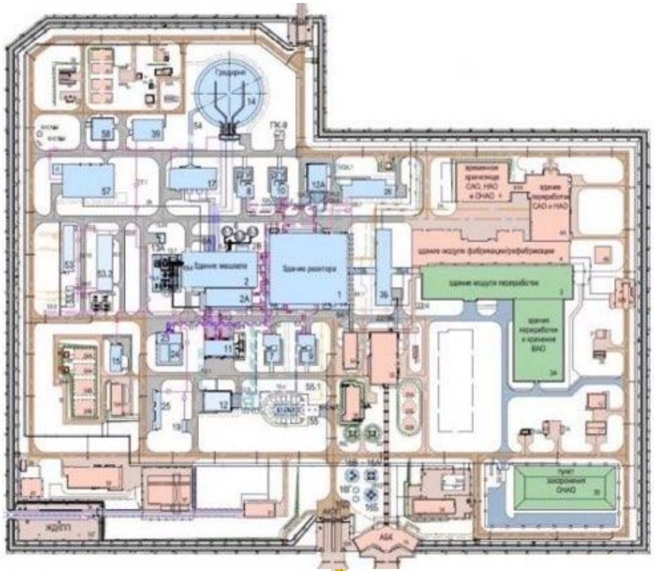
Excluded from the nuclear fuel cycle:

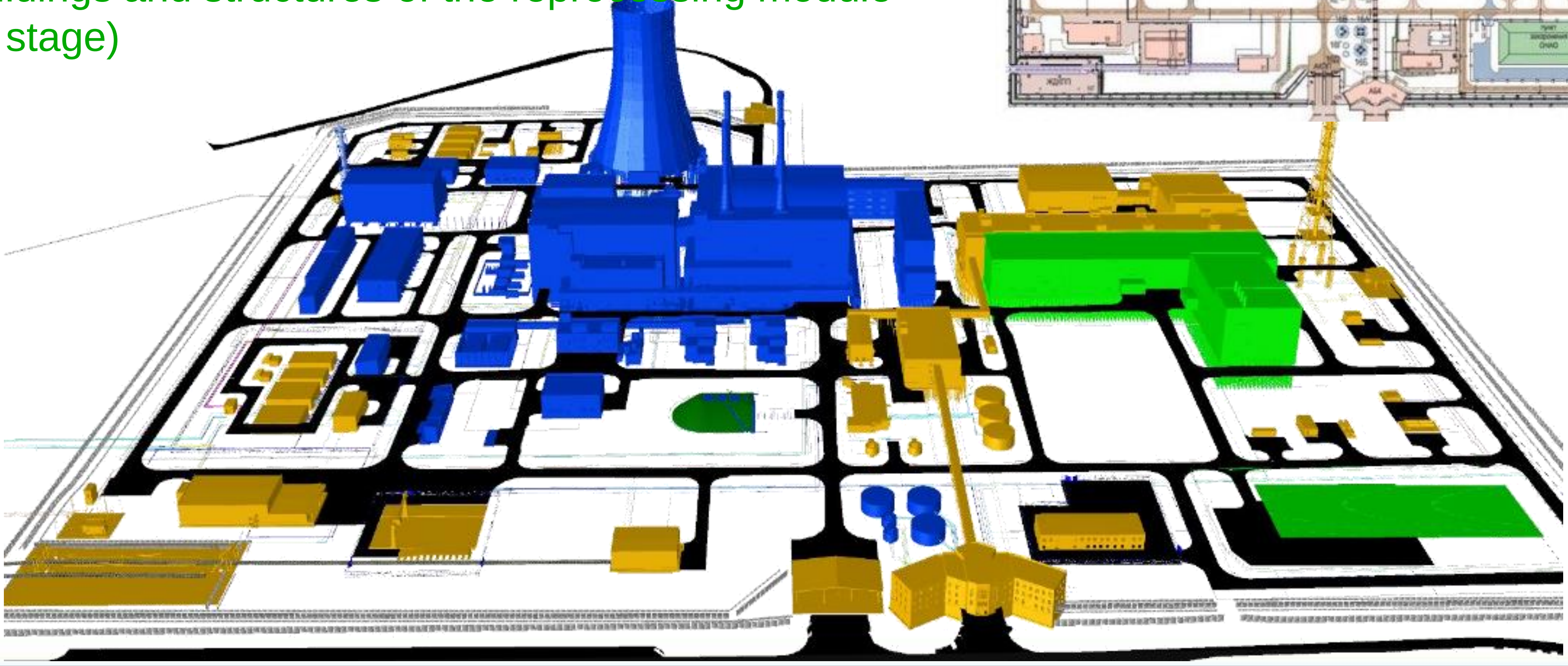
- uranium isotopes separation (enrichment)
- Blanket for pure plutonium production
- Pu separation and/or 233U, 235U from INF
- Long-term INF storage
- Storage for separated plutonium
- Conventional transport streams for nuclear materials

Project «PDEC». General layout of the site

Four stages for PDEC construction and commissioning:

- Buildings and structures of the fuel fabrication module and start-up refabrication complex (stage I and IV)
- Buildings and structures of the BREST-OD-300 reactor facility (II stage)
- Buildings and structures of the reprocessing module (III stage)

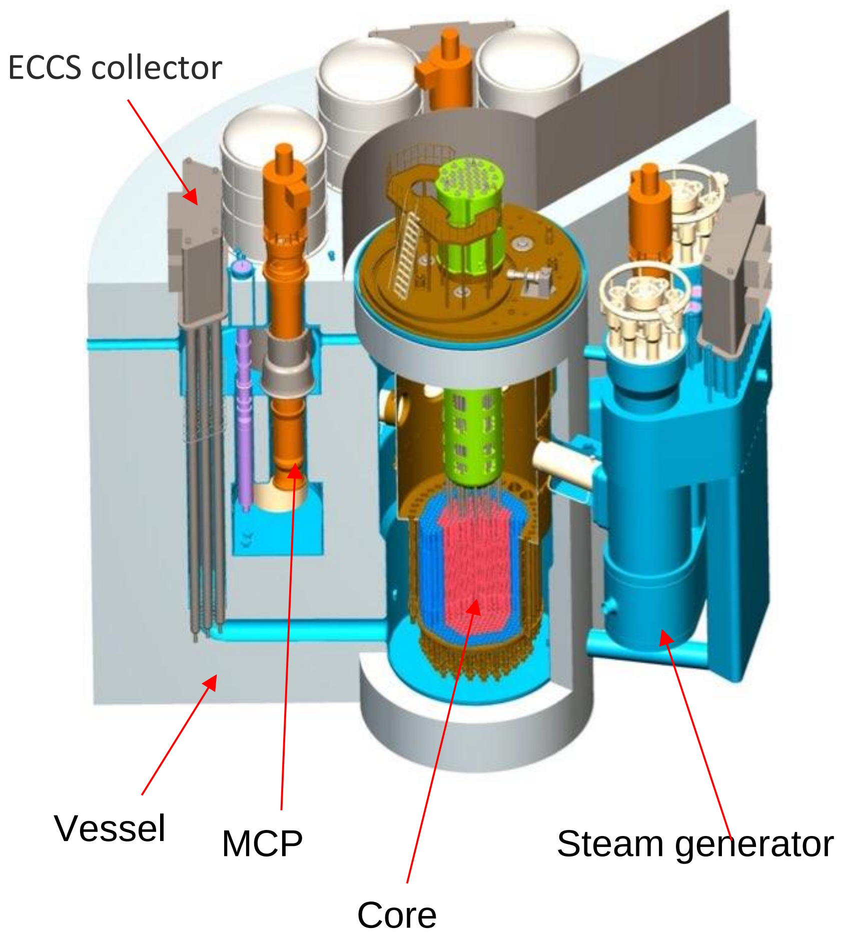




PDEC main parameters

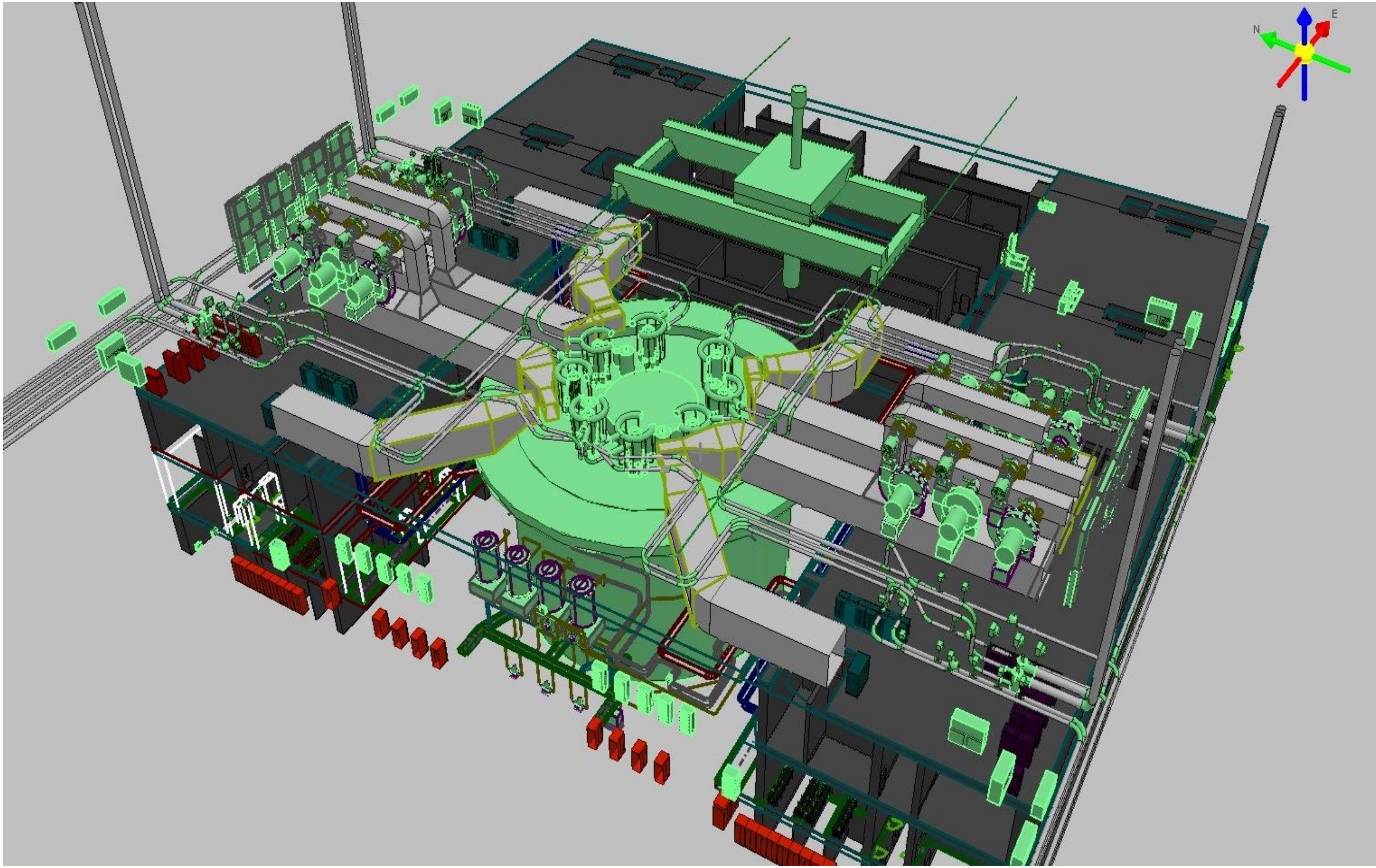
Installed electrical capacity of unit	300 MW
Type of fuel	MNUP
MNUP fuel fabrication and refabrication production capacity	14 t/year
BREST-OD-300 INF reprocessing production capacity	5 t/year

Main elements and technical characteristics of the BREST-OD-300 reactor



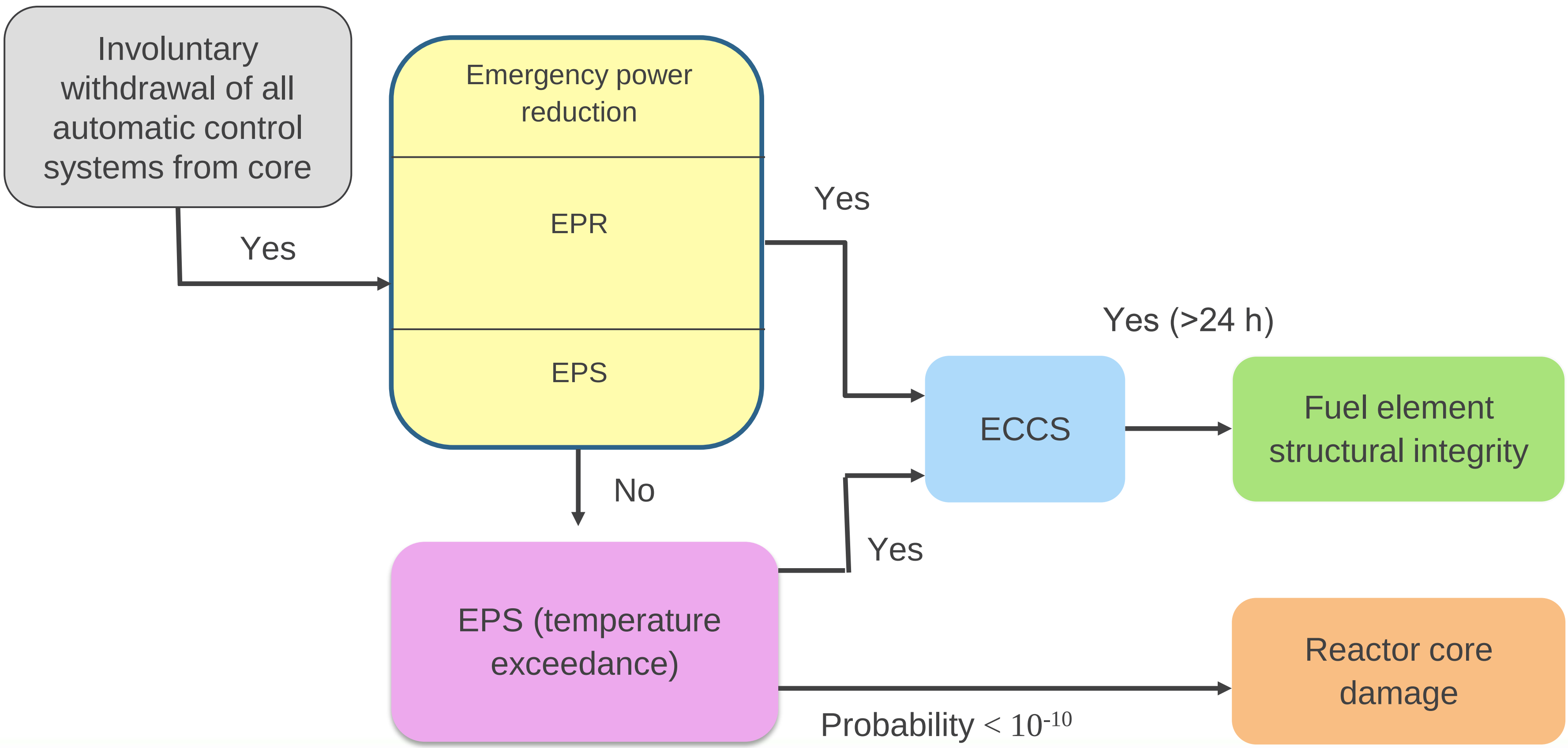
Thermal power, MW	**700**
Number of loops	**4**
Primary circuit coolant	**Lead**
Maximum (hydrostatic) coolant pressure in primary circuit, MPa	**1,17**
Average core inlet/outlet temperature, °C	**420/535**
FA in active zone	**169**
Fuel load, t	**20,6**
Electric power, MW	**300**
SG inlet/outlet water/steam temperature, °C	**340/505**
SG outlet pressure, MPa	**17**
Rate of steam production, t/h	**1500**
Unit net efficiency, %	**43**

BREST-OD-300 central hall and major equipment cross section view



NOF scenario for exceptional conditions: introduction of full reactivity margin

Event initiation group	Safety functions and expected impact		
	Emergency reactor shutdown and maintaining its subcritical state	Emergency removal of residual heat from fuel elem to Pb and atmosphere	Expected impact



Fast reactor safety

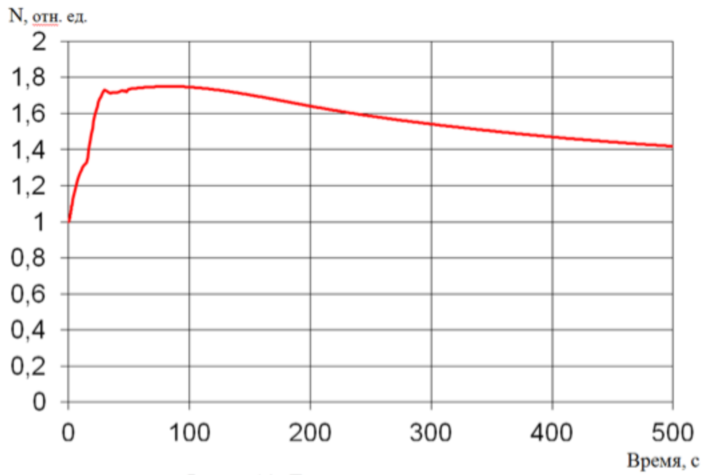


Рисунок 4.1 - Поведение мощности реактора

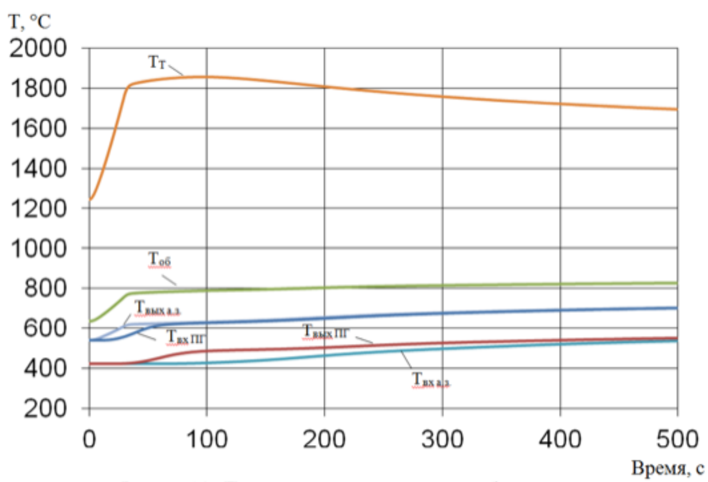


Рисунок 4.2 - Поведение температуры топлива, оболочки твэл, свинцового теплоносителя на входе и выходе а.з. и ПГ

The fundamental characteristics of fast reactors allow them to potentially provide a high level of nuclear safety, enabling power operations with a small operating reactivity margin during transient reactor processes and deviation in fuel composition (absence of strong absorbers, iodine wells etc..)

Inner nuclear fuel breeding eliminates the potential risk of prompt-neutron reactor power excursion.

Project research work shows that realizing this concept is feasible.

A case study using BREST-OD-300 as an example showed that uncontrollable power growth with full introduction of reactivity margin is blocked at a level of $1.4N_{НОМ}$

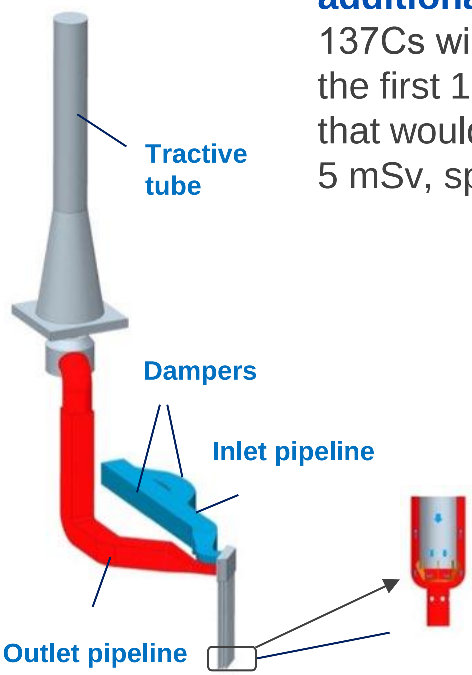
Furthermore, fuel pin cladding temperature does not exceed 815 °C, melting of the fuel elements is not possible

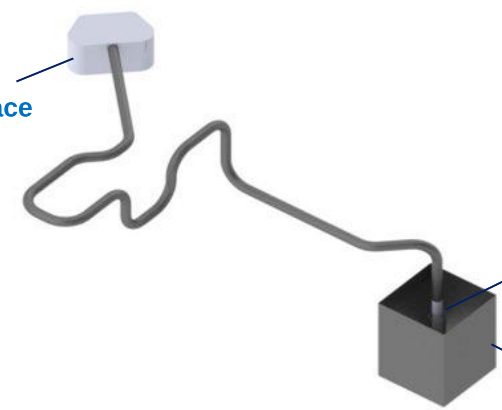
Radiation impact

Under the scenario where introduction of operating reactivity margin takes place, FP escape for the first days does not exceed 4,3*10E8 Bk (does not exceed the daily controlled level of emissions under standard operation)

ECCS loop

If a steam generator rupture and a steam generator containment system failure are **additionally factored** in, daily emissions of inert radioactive gases, 131I, 134Cs and 137Cs will constitute 3.4·1015 Bk, 5.7·108 Bk, 2.8·108 Bk и 3.7·108 Bk respectively. In the first 10 days after the accident, the maximum dose of external exposure to radiation that would have taken place on the site would not exceeded 1.5 mSv (does not exceed 5 mSv, special safety measures not required)





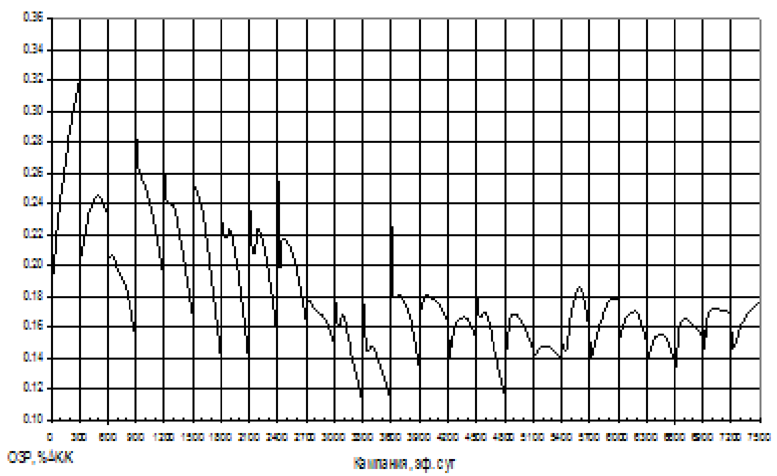
Thanks to the design characteristics of the reactor with a lead coolant, a small reactivity margin, passive safety features, the total probability of reactor core damage from all exceptional conditions in the span of 24 h does not exceed **$9 \cdot 10^{-9}$ 1/year.**

Reactor core

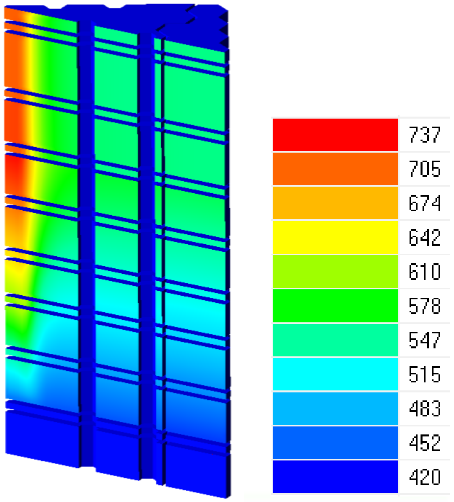
Base principles of an equilibrium reactor core were confirmed:

- maximum reactivity margin at rated power - **$0{,}4\beta_{eff}$ ($0{,}65\ \beta_{eff}$)** with operating measurements and initial error compensation taken into account;
- Attaining an equilibrium state in a closed NFC with MA transmutation;
- Stability of the neutron field: relative variation of power throughout refueling intervals of the central zone **FA < 1 % and peripheral zone FA < 3 %**, maximum linear power for central zone FA fuel elements - **420 watt/cm and for peripheral zone FA - 340 watt/cm**;

The physical concept of equilibrium state operation will be experimentally confirmed and verified on the pilot production facility with a closed NFC.

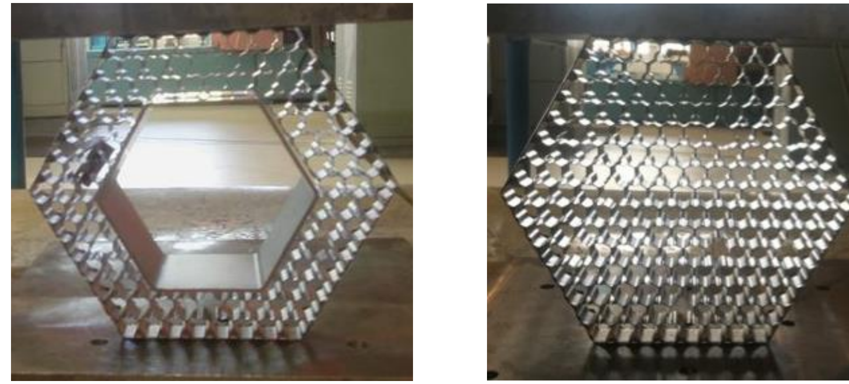


Fore cores with jacket-free FA, overlapping of the flow cross-section of 7 central FA does not result in exceeding the safety operation limits in relation cladding temperature



Reactor core materials justification

- Production prototypes completed for all types of fuel assemblies (FA)
- Structural performance characteristics obtained for FA elements and prototype units
- Obtained vibrometric and vibration resistance characteristics for central zone prototype FA
- Hydraulic characteristics obtained for central zone prototype FA and FA with CPS (on water)
- Reactor tests are underway for experimental FA in BN-600 (11 exp. FA) and in BOR-60 (10 FA)

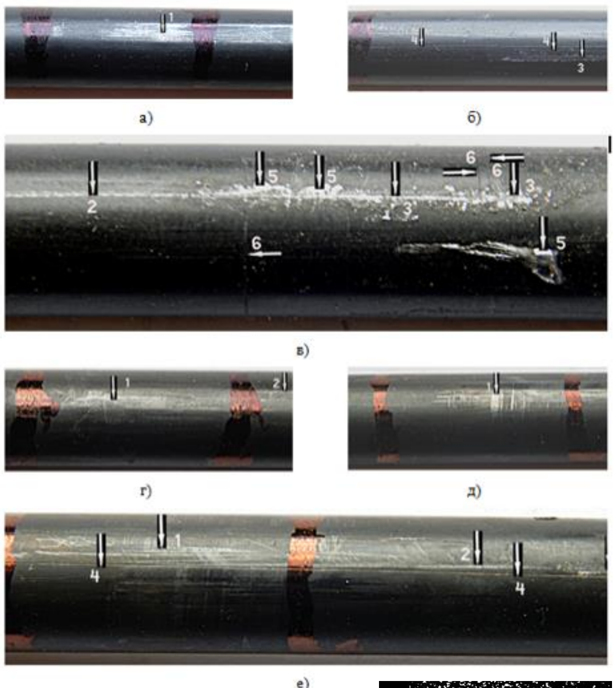


FA seismic resistance testing

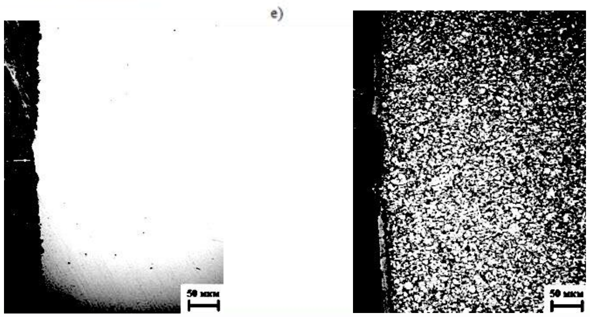
Mechanical testing of prototype and framework of peripheral zone FA and FA with CPS

Reactor core materials justification

Corrosion testing of two prototypes - 540C and 450C for 2500 h
Contact traces under spacer grid comparable with oxide layer width (less than 20 µm)



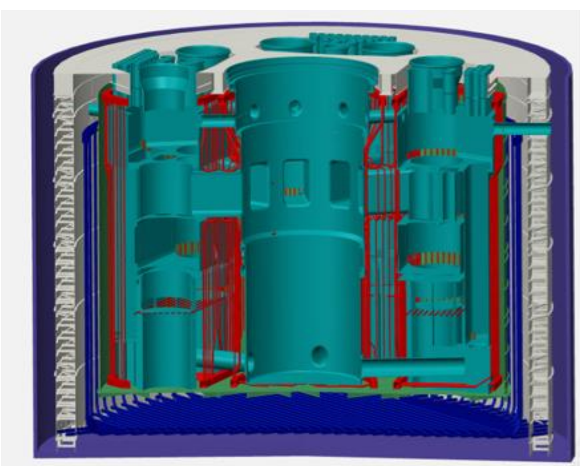
External view of the prototype and surface of the fuel element imitator after testing in lead coolant flow (1 – light line, 2 – light strip, 3 – gray line, 4 – scratch mark, 5 – Pb fragments, 6 – cross over scratch mark) а,б,в) in lower spacer grid zone; г,д) in middle spacer grid zone; е) in upper spacer grid zone



Cladding microstructure before and after corrosion testing

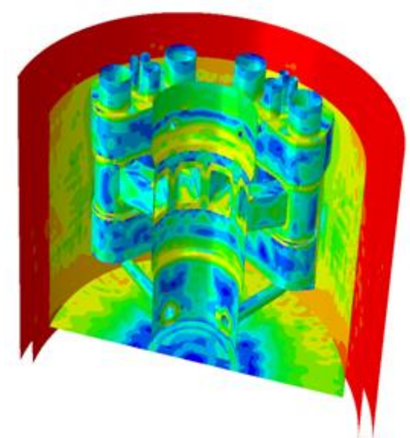
Vessel

- An integral design ensures that loss -of-coolant probability is no higher than 9,7E-10 1/year.
- The vessel performs a localizing function.
- No chemical interaction of lead with concrete is featured, penetration depth does not exceed 0.5 mm
- Heating regimes were tested on a full-scale model of the central part of the vessel , experiments with drying technology and gas release were conducted
- Technical specs for the bimetal shell (first circuit boundary) ЭП302-09Г2С were developed, 5 sheets were made;
- Technical specs for heat-resistant concrete were developed, their operating range physical characteristics obtained
- Research was completed on radiation resistance for total metal service life of the shell (0.12 dpa) and concrete (increased hardening by 24%, reduction of thermal conductivity by 8%, reduction of the coefficient of linear expansion by 10%). Sufficiency of characteristics is verified.









Steam generator

- Mono-metal corrosion resistant tubes in water and lead, no welded junctions across body
- Coiled heat-transfer section
- Extended throttle, which maintain hydrodynamic stability and limit consumption in event of tube containment failure

Module thermal power , MW	90
Height, m	12,1
Immersed section diameter, m	2,0
Mass, t	69
Rate of steam generation, kg/s	52,40
Service life, years	30

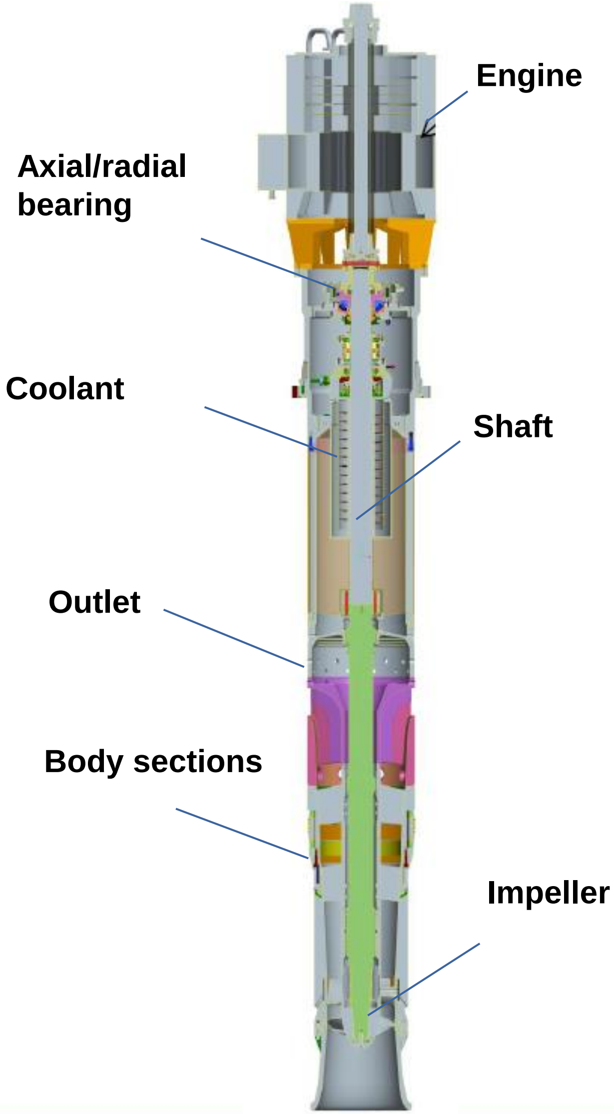
Steam generator

- Heat-transfer coefficients were determined using an 18-tubed model with a lead coolant
- Thermohydraulic stability limits in start-up modes were determined with/without throttling
- It was experimentally determined that it is impossible for a single tube rupture to develop into multiple tube rupture
- A method for high-temperature rinsing was developed and experimentally proven
- A new material for steam generator tubes was developed, tubes over 35m in length were obtained for the first time in Russia





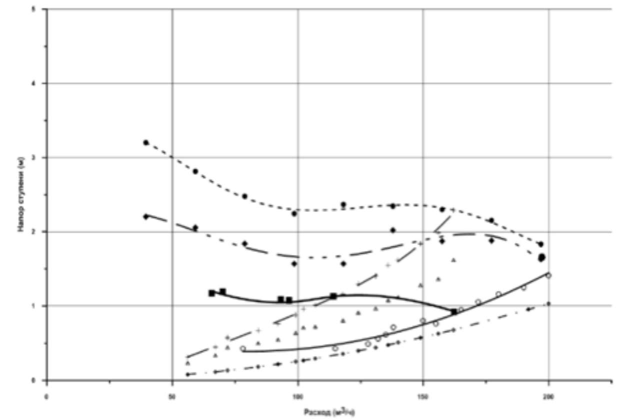

MCP



Height without/with drive, m	9,4/12,7
Diameter inside FR vessel, m	1,2
Removable part mass, t	32,3
Nominal feed (consumption under nominal temperature of the pumped fluid), m3/h, (kg/s)	3850 (11300)
Nominal working pressure, m	2,8
MCP Power, kW, not more than	650
Positive suction head, m	2
Assigned service life (with part maintenance), years	30

MCP

- A medium-scale work site and MCP prototype was developed (lead)
- Output performance characteristics were obtained at the liquid end (lead) at an 80% level of the required amount (limitation of test facility)
- Hydrostatic bearing working capacity was confirmed at the medium-scale level (over 300 start-stop cycles)
- Liquid end (water) output performance was optimized, consumption, pressure and net positive suction head was determined
- A mock-up unit was constructed for testing the full-size lower radial bearing, preparations are being made for conducting tests with lead



Structural materials

ЭП302-Ш (in-vessel components)

- Mechanical characteristics in operational range with damaging factors taken into account were determined for most of the intermediate products.
- Corrosion allowance for lead was determined based on experimentation results (10000 h to 50000 h).
- Mechanical characteristics were obtained for a damaging dose twice higher than in the design spec.

ЭП302М-Ш (steam generator, new material)

- Mechanical characteristics in operational range were determined with most of the damaging factors taken into account.
- Corrosion allowance for lead was determined based on experimentation results (7000 - 15000 h). Experiments are ongoing.

ЭП823-Ш (reactor core components)

- Experiments have been conducted for determining specific operating times in oxygen modes with deviation from standard mode.

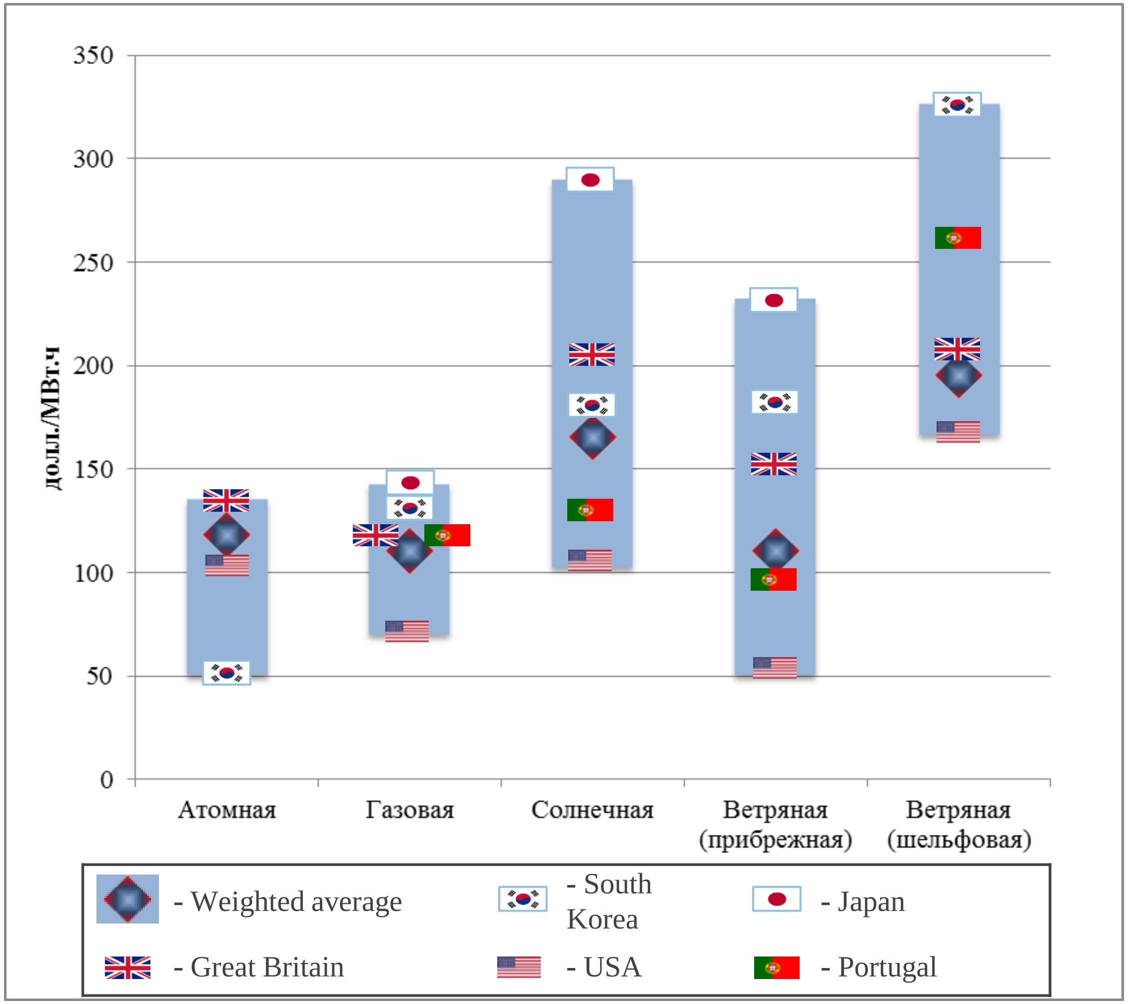
General view of PDEC

FRM construction

PDEC construction

Competitiveness of modern power generation technologies

LCOE for new powerplants projects (commission date - 2020)



- Competitiveness of renewables very much depends on the region
- Several renewables projects are currently under development in Russia
- Combined cycle plants will continue to be the main competitor for nuclear, along with emerging renewable power

Sources: Projected Costs of Generating Electricity – 2015 Edition. International Energy Agency (IEA), Nuclear Energy Agency (NEA), 2015, ставка дисконтирования 10%

Westinghouse proposes LFR project

- According to Pezze (Vice President, Global Technology Development and Chief Technology Officer today) all types of reactor were considered, whether cooled by gas, various metals, and even molten salts. Safety of each design was the key consideration, but economic viability (without which none could be built) was also a guiding consideration. (She noted that "the team did even look at LWR or Light Water Reactor designs for this study as well.")
  The study, including some 15 or 16 criteria appropriately weighted, resulted in a rather clear winner - the lead cooled fast reactor or LFR

- It also said the reactor's load-following capabilities "would further support the increased use of renewable energy sources"

- Westinghouse president and CEO Danny Roderick said, "Westinghouse's vision is to be the first to innovate the next technology, and we believe an LFR plant will be the next advanced reactor technology to be deployed

- Westinghouse submitted its project proposal for advanced reactor concepts that can be demonstrated in the 2035 timeframe

Source http://www.world-nuclear-news.org

THANK YOU
FOR YOUR ATTENTION!