Applied Irradiation Research for Breeding Blanket and Plasma-Facing Materials in Fusion Systems

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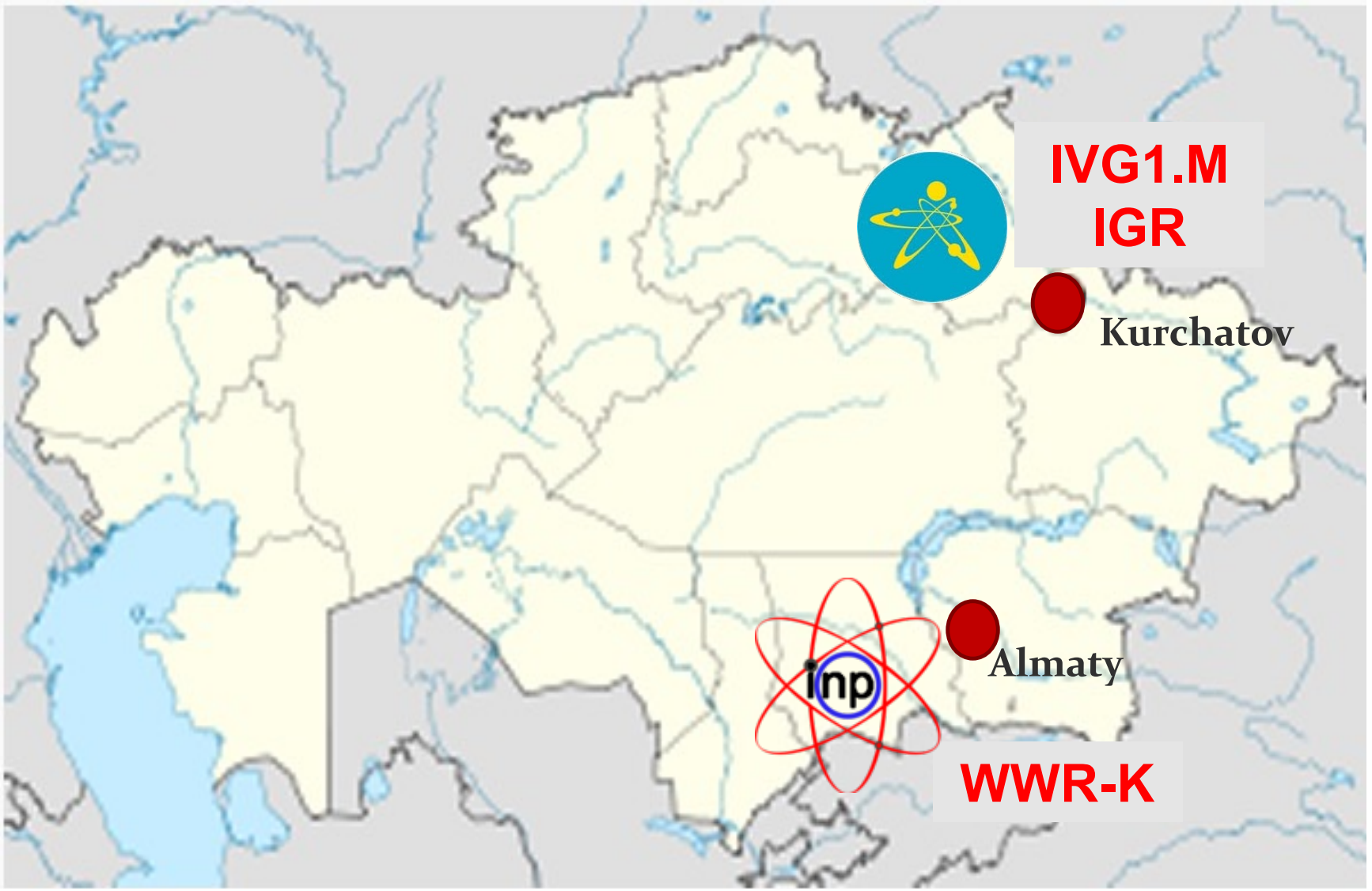
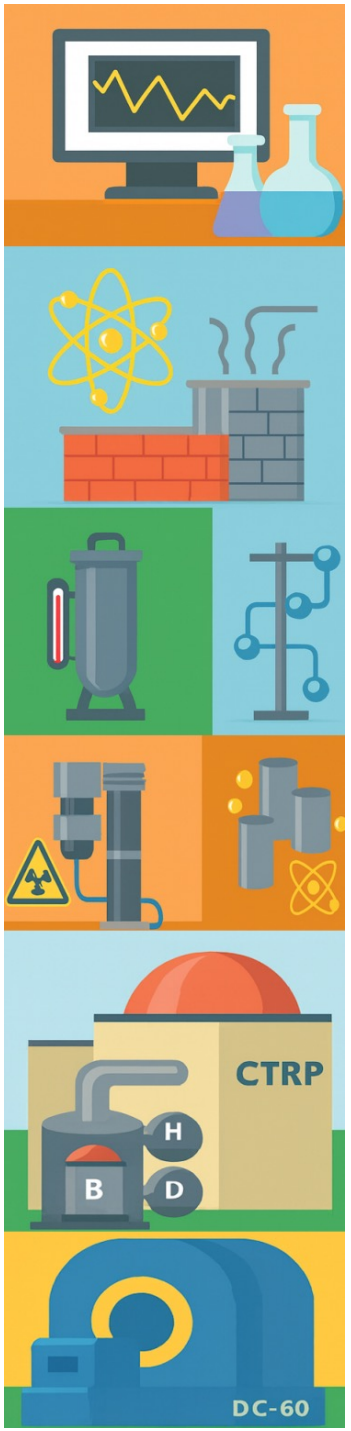
Support for fusion research in Kazakhstan



Main topics and projects in Kazakhstan

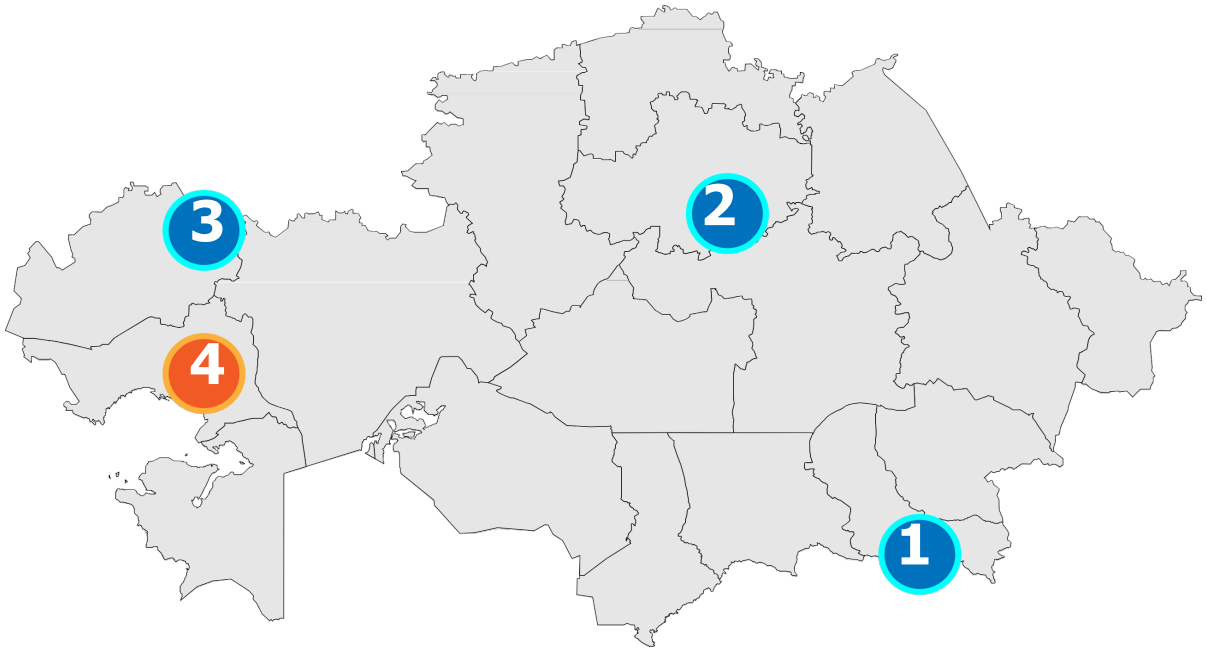
- Long-term reactor testing of Li_2TiO_3 lithium ceramics (including post-radiation thermodesorption experiments).
- Tritium generation and release from a lead-lithium eutectic under reactor irradiation conditions.
- Experiments on investigation of tritium and helium release from lithium; lead-lithium eutectic under reactor irradiation conditions.
- Experiments on hydrogen isotope sorption by lithium and lithium CPS under reactor irradiation conditions.
- Thermodesorption studies of irradiated beryllium of various trademarks.
- Study of the processes of interaction of hydrogen isotopes with FR structural materials under reactor irradiation conditions: tungsten, vanadium and its alloys, graphite, etc.
- Reactor testing of lithium CPS.
- Testing of radiation resistance of fibers and fiber optic sensors to study the behavior of sensors in ITER conditions.
- Irradiation of functional, structural and composite materials at cyclotron DC-60



Institute of Nuclear Physics

BRIEF REFERENCE

The main site of the Institute is located in Almaty. There are 2 branches of the Institute in Astana and Aksai cities.



1 Almaty
600 employees
7 basic facilities

2 Astana branch
50 employees
Heavy ions accelerator

3 Aksai branch
25 employees
LIRA facilities

4 Azgir
Expedition

INP staff in numbers

~700
employees

~150
young specialists

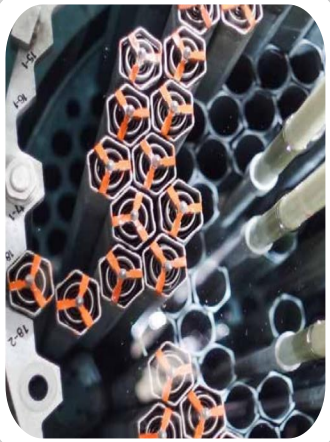
~74
specialists with academic degree

Basic Facilities

INSTITUTE OF NUCLEAR PHYSICS

Research reactor WWR-K

Critical Reactor Assembly



Electrons Accelerator ELV-4

Electrons Accelerator ILU-10

Cyclotron U-150M

Linear Accelerator UKP-2-1

Cyclotron Cyclone-30

Heavy ions acceleratorDC-60

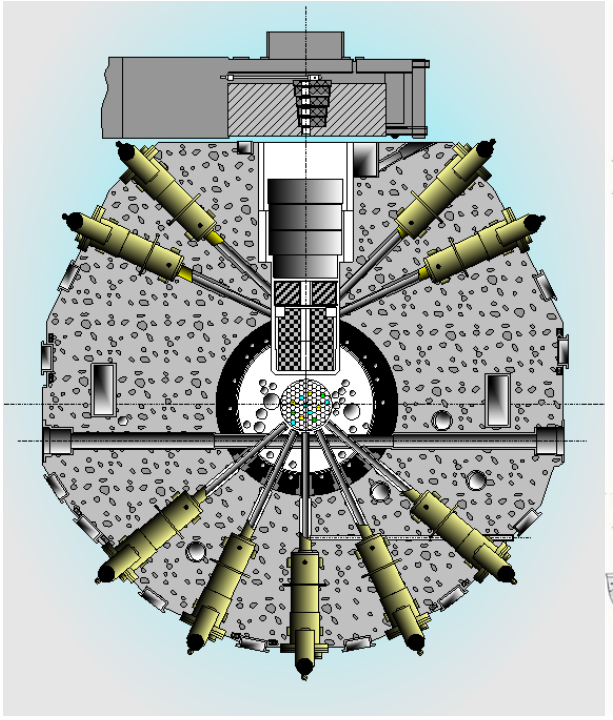
The WWR-K research reactor

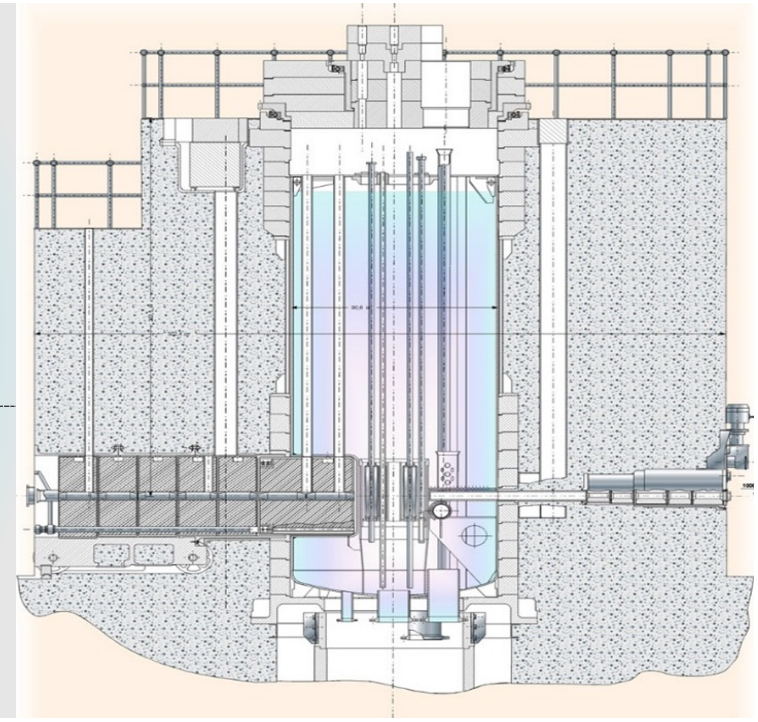
- Type: tank
- Thermal power: 6 MW
- Moderator: demineralized water
- Reflector: demineralized water and beryllium
- Coolant: demineralized water
- Pressure: atmospheric
- Type of coolant: forced
- Coolant circuit: two
- Core diameter: 720 mm
- Core height: 600 mm
- Fuel: UO2+Al matrix (LEU)
- Maximum of thermal/fast neutron flux: $2*10^{14}/8*10^{13}$ $cm^{-2}s^{-1}$

Irradiation positions:

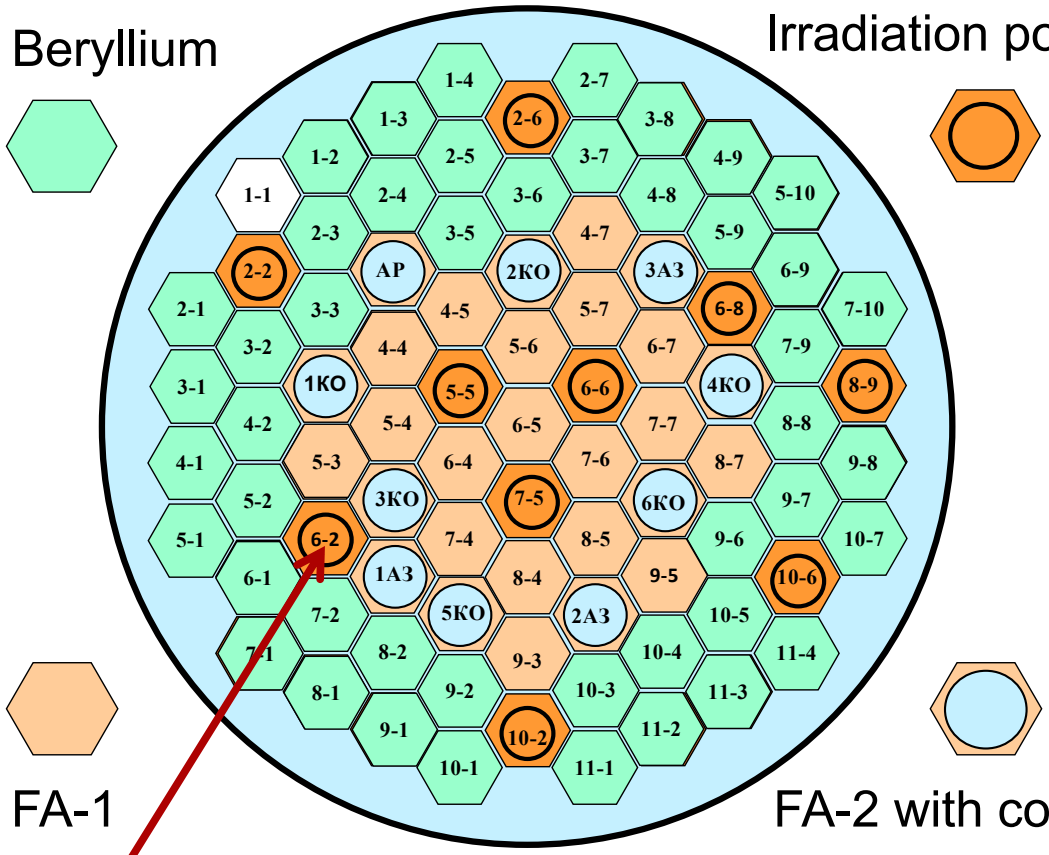
<u>Vertical experimental channels:</u>
Core: 8
Center – 3 (Ø60 mm)
Periphery – 5 (Ø60 mm)
Tank (reflector): 20
Ø200 mm — 2 channels
Ø100 mm - 13 channels
Ø70 mm - 5 channels

<u>Beam tubes:</u>
Radial: 9 (Ø100 mm and Ø60 mm)
Tangential: 1 (Ø192 mm)

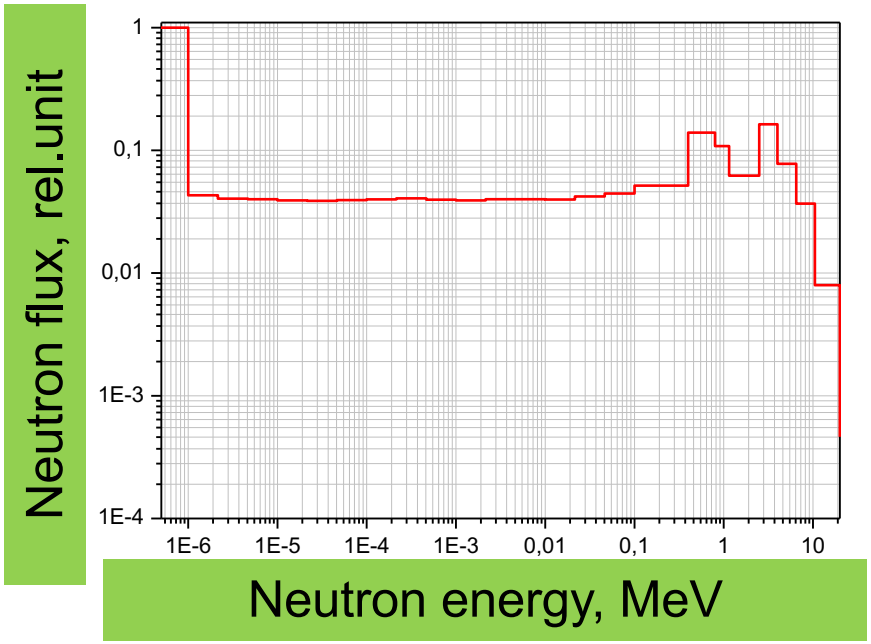




The WWR-K research reactor

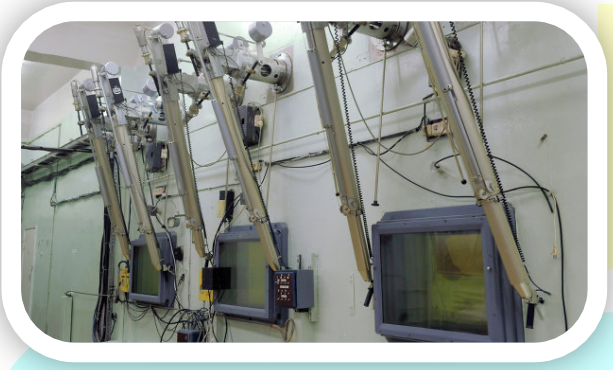


Neutron spectrum in core



Channel type	Location	Diameter, mm	Max neutron flux, $cm^{-2}s^{-1}$	
			E_n<0.625 eV	E_n>0.1 MeV
Neutron fluxes Unit: $cm^{-2}s^{-1}$				
Vertical	in core center	60	~$2.0\cdot10^{14}$	~$7.5\cdot10^{13}$
	in core periphery	**60**	**~$3\text{-}8.0\cdot10^{13}$**	**~$1.4\cdot10^{13}$**
	in reactor tank	200	~$9.0\cdot10^{11}$	~$1.5\cdot10^{11}$
		100	~$3.3\cdot10^{11}$	~$2.0\cdot10^{11}$
		70	~10^{9}	~$6.8\cdot10^{8}$
Radial beam	from reactor tank	60 and 100	~$2.0\cdot10^{8}$	~10^{8}
	from thermal-column recess	100	~$2.0\cdot10^{7}$	~10^{7}
Tangent beam	from reactor tank	200	~$2.0\cdot10^{11}$	~10^{11}

Experimental Facilities



«Two kind of hot cells:

- ❑ concrete shielding (5 cells)
- ❑ steel shielding (4 cells)

Hydraulic transfer system

Utilized to load and unload of irradiation ampoule to/from the core during reactor operation

Critical test-bench (assembly)

Zero power reactor
Maximum thermal power: 100 W
Reflector: deionized water and/or beryllium
Moderator: deionized water
Fuel composition: UO_2+Al;
Enrichment in U-235: 19.7 % (since 2012)
Max. thermal neutron flux density: 10^9 cm^{-2} s^{-1}
Diameters of experimental channels: 65, 96 and 140 mm.
.

Pneumatic transfer system

Installed at beam tube #3, allows transfer of ampoule with a sample from laboratory room to the core and return back for gamma spectrometry measurement (neutron activation analysis)

Gas-vacuum loop facility

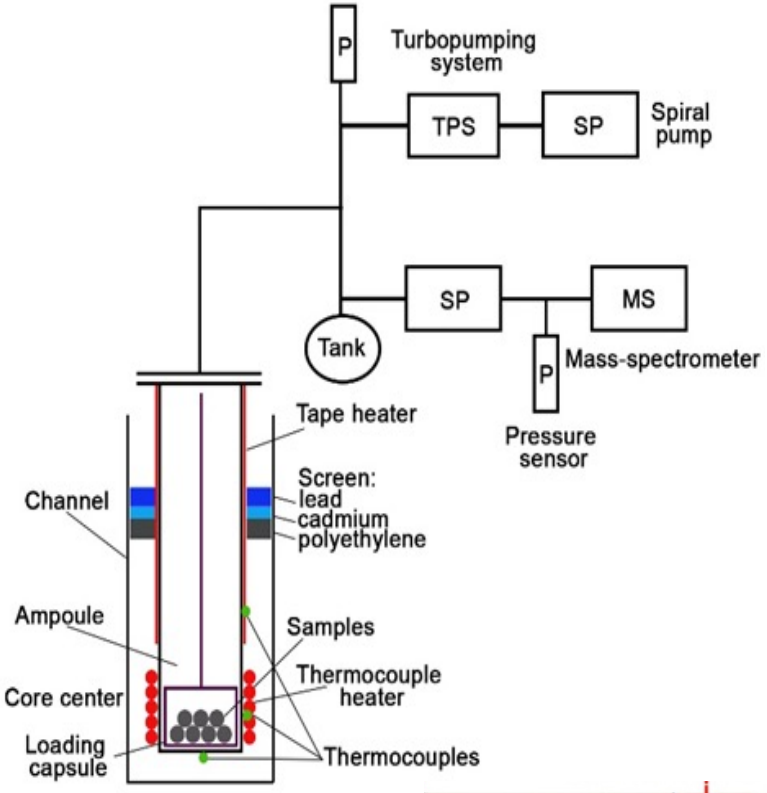
Gas-vacuum Loop Facility is designed to create a vacuum or gas environment in a sample during in-pile test

Facility of neutron radiography and tomography:

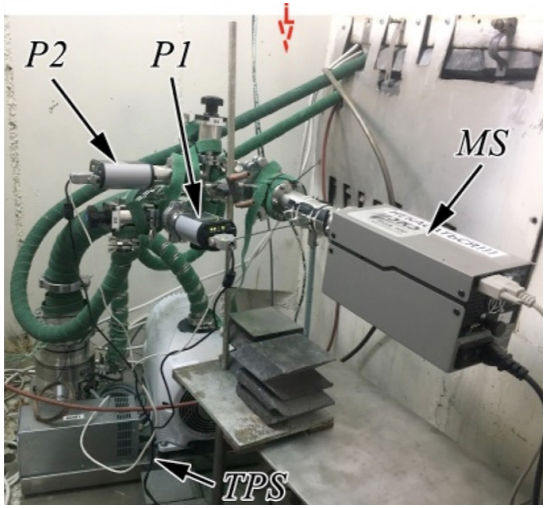
Designed for the study of objects by non-destructive methods

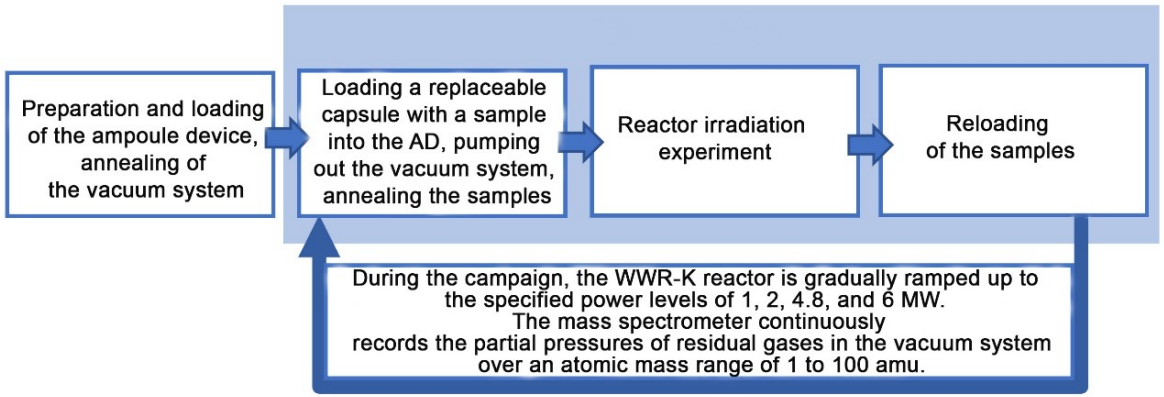
Experimental Part

EXPERIMENTAL FACILITY FOR STUDYING GAS RELEASE FROM MATERIALS OF A NUCLEAR REACTOR (NR) AND A FUSION REACTOR (FR) AT THE WWR-K REACTOR

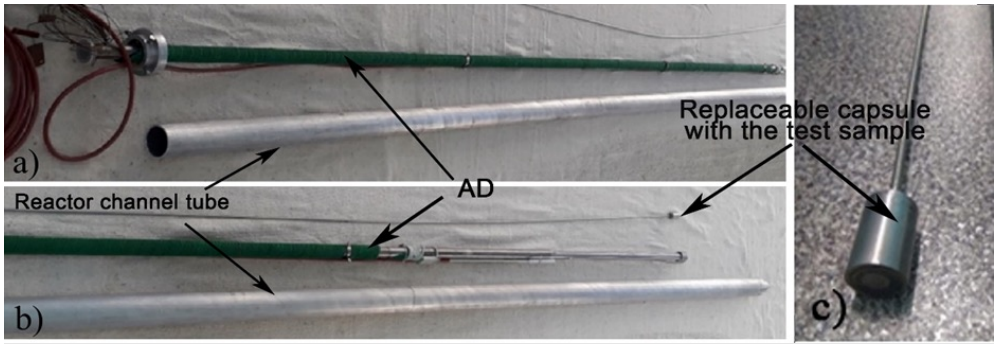


The CIRRA (Complex of In-Reactor gas Release Analysis) Facility





Scheme of vacuum extraction method



View of the irradiation device:
a) – the upper part of the AD and a tube of the reactor channel,
b) – capsule with guide wire, bottom part of the AD and a tube of the reactor channel;
c) capsule enlarged

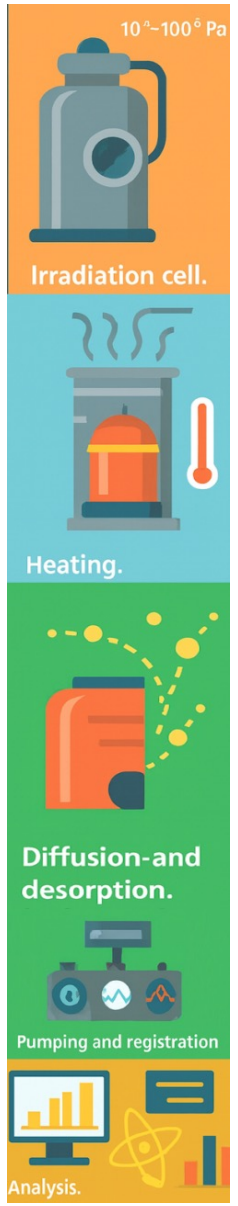
Irradiation ampoule device

The vacuum extraction method of tritium from lithium ceramics under irradiation conditions

The **vacuum extraction method** of tritium under irradiation involves **the release** of tritium, which is formed **directly in the** lithium ceramic **sample** through the nuclear reaction $^{6}Li(n,\alpha)T$, from the material **by heating it in vacuum** conditions while pumping with the experimental setup.

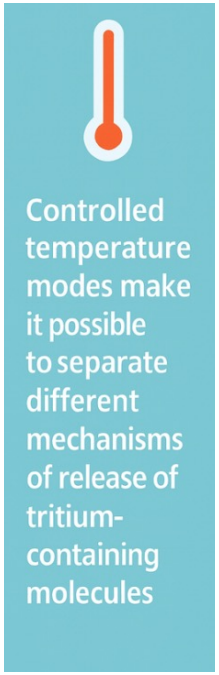
Main principles of the method:

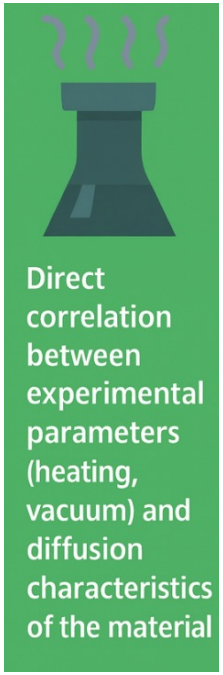
- **Irradiation cell.** The sample is placed in a sealed chamber – an irradiation cell, where the pressure is reduced to about 10^{-4}–10^{-6} Pa.
- **Heating.** The irradiation cell is placed in an ampoule device, which is then loaded into the reactor channel. The ampoule is subsequently heated (usually from several hundred to 1000 °C) according to a predetermined temperature profile.
- **Diffusion and desorption.** Under the influence of high temperature and low pressure, gas molecules (tritium compounds and other gases) are released from the material structure.
- **Pumping and registration.** The released gases are pumped out by a turbomolecular pump and registered using mass spectrometers and/or ionization chambers.
- **Analysis.** From the gas release profile (dependence of gas flux on temperature and/or time), the concentration, release form, diffusion coefficients, and activation energies are determined.



Advantages of the method

- **High sensitivity** to low concentrations of gases (including tritium).
- **Controlled temperature modes** make it possible to separate different mechanisms of release of tritium-containing molecules.
- **Direct correlation** between experimental parameters (heating, vacuum) and diffusion characteristics of the material.
- **Ability to vary reactor power during the experiment.**
- **Weak influence of the gas phase on tritium release processes from the ceramic surface.** Since there is a low concentration of water vapor and other gases in the sample chamber, their effect on the surface processes of tritium release is minimal.



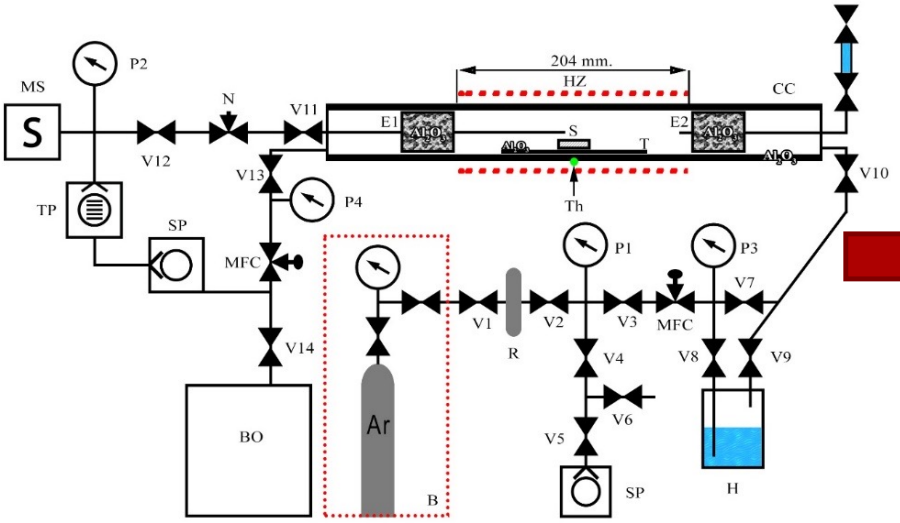


Application in tritium studies:

Application in tritium studies:

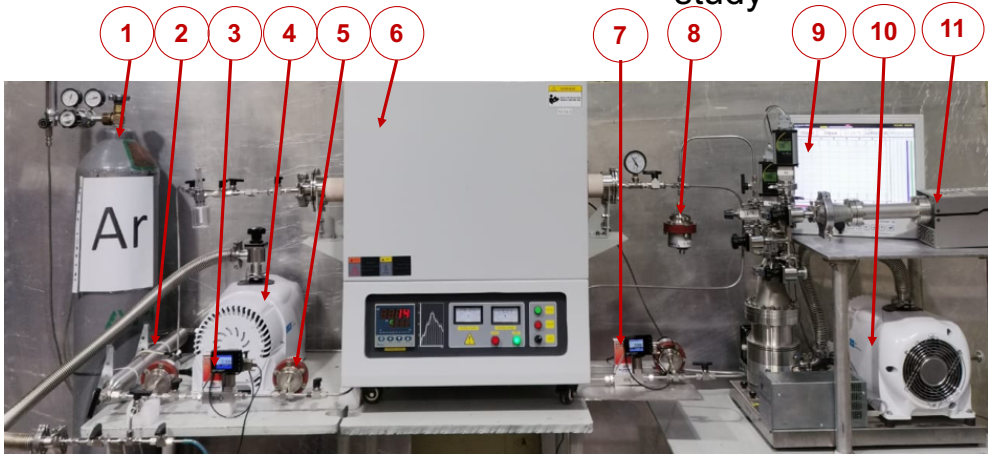
- Allows quantitative evaluation of **the tritium diffusion coefficient** in the ceramic matrix, since the rate of tritium release under vacuum is directly related to its migration through the volume and surfaces of the granules.
- Makes it possible to determine tritium transfer parameters (diffusion coefficients, desorption coefficients, etc.) by analyzing tritium release curves.
- Used for **validation of numerical calculation** models by comparing experimental and theoretical parameters.

EXPERIMENTAL SETUP for post-irradiation TDS (thermo desorption experiments)



weighting

microstructure study



Appearance of the experimental setup:
1 – gas balloon, 2 – receiver, 3, 7 – mass flow controller, 4 – dry pump, 5, 8 – pressure sensor, 6 – electrical furnace, 9 - multi-channel technological recorder, 10 – pumping station, 11 – residual gas analyzer

Results of reactor experiments with lithium ceramics in a vacuum

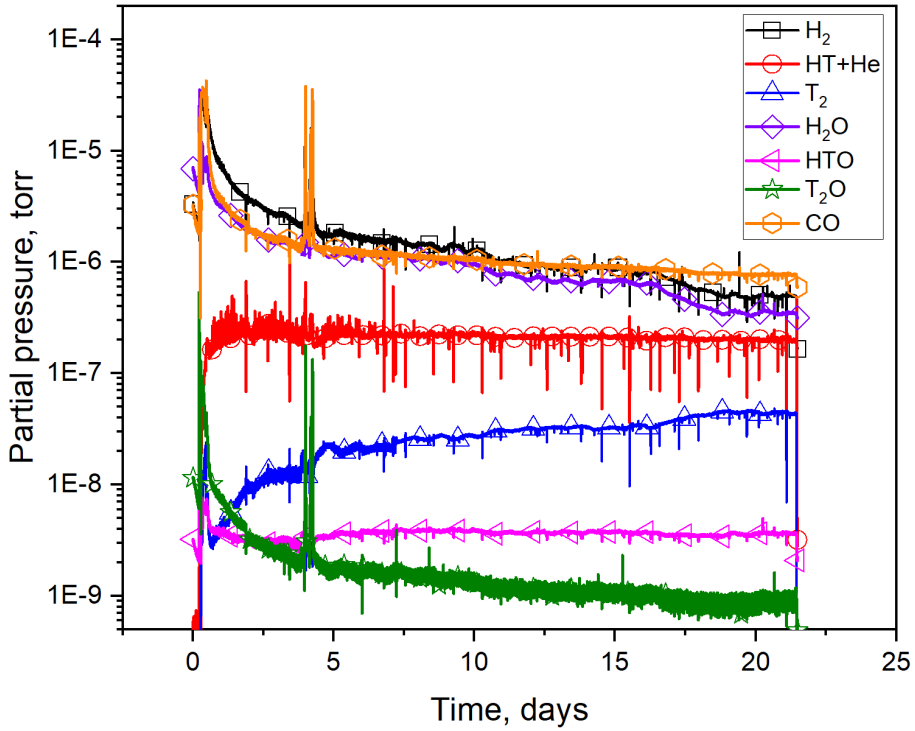
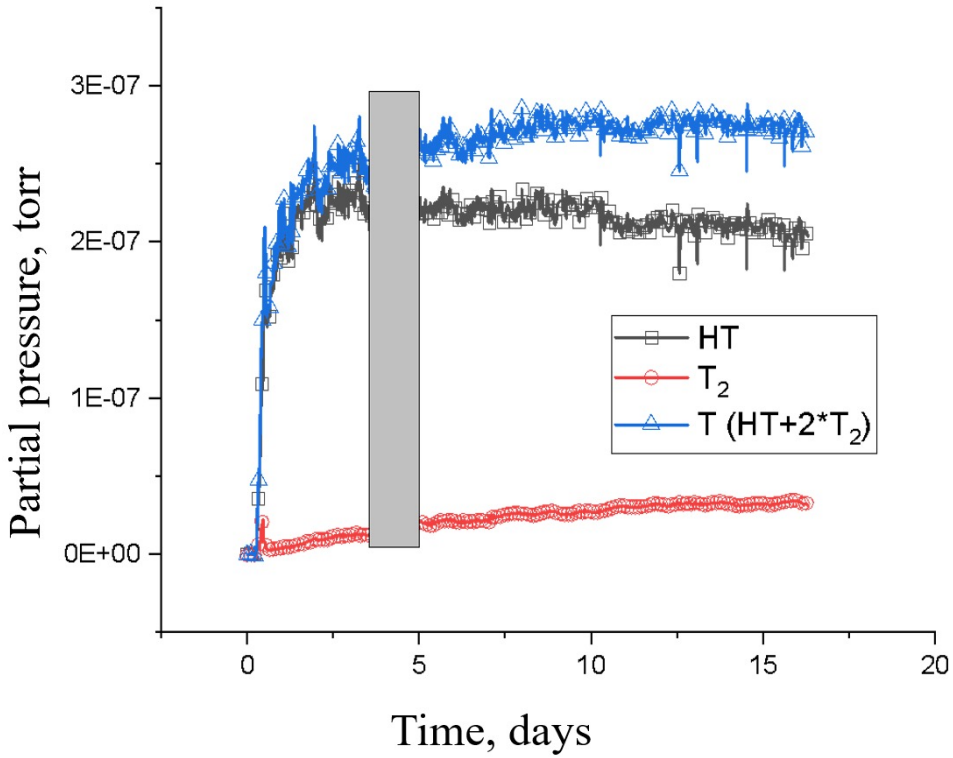


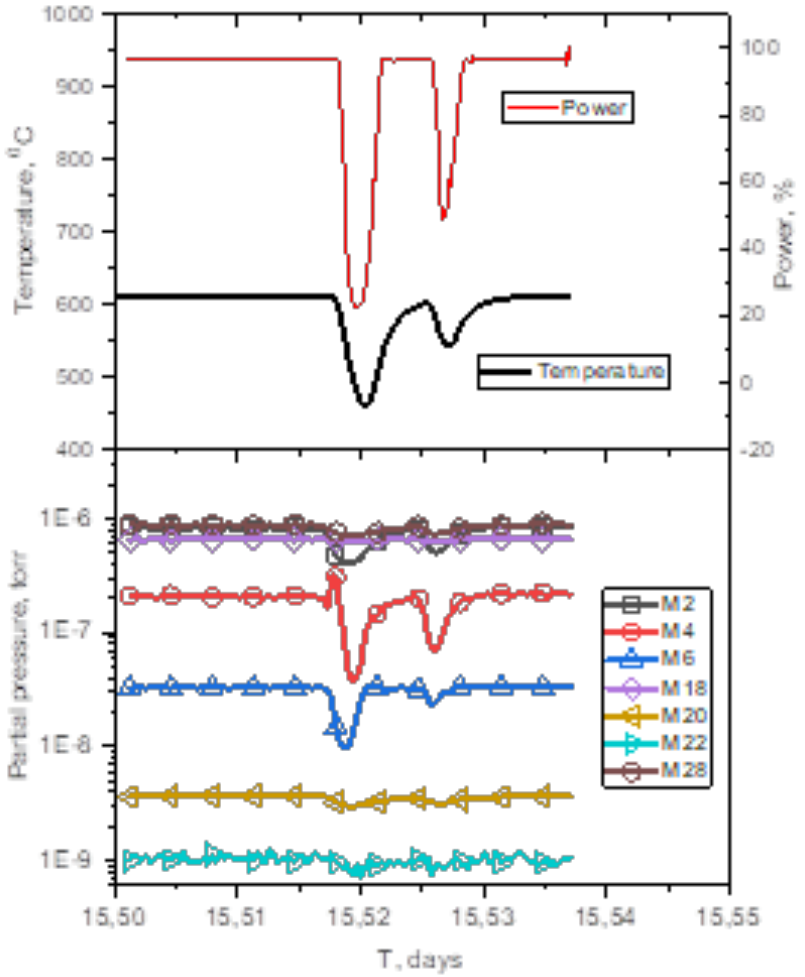
Diagram of the reactor experiment with lithium ceramics performed using vacuum extraction method



Tritium release in the experiment

Reactor experiments

Method for estimating the activation energy of tritium diffusion in ceramics during reactor irradiation



Changes in reactor power lead to changes in sample temperature and correspondingly to changes in tritium fluxes HT (M4) and T_2 (M6)

We propose a procedure for estimating the E_D activation energy of tritium diffusion in ceramics, which is based on the following statements:

- at quasi-equilibrium tritium ($HT+T_2$) release from ceramics, the flux from one pebble can be represented

$$J = D \cdot S \cdot \frac{\partial C(r)}{\partial r}\Big|_{r=R_0}, \qquad D = D_0 \exp\left\{-\frac{E_D}{R \cdot T}\right\}$$

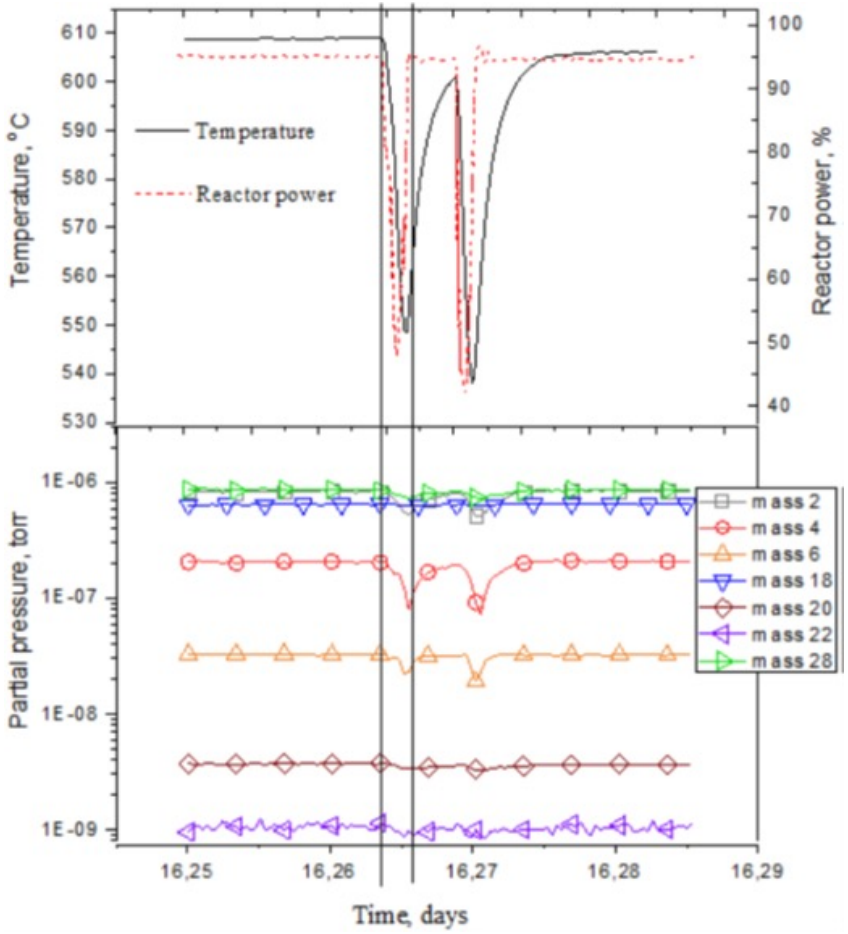
where J is the flux of tritium from the sample; D is the diffusion coefficient, S is the effective area of the pebble; R is the radius of the pebble, C(r) is the concentration of tritium in the sample.

- We assume that for a short time of reactor power change the tritium concentration gradient on the pebble surface can be considered constant

- then, respectively, the change in tritium flux from the sample is due to the change in the diffusion coefficient as a result of a change in sample temperature

Reactor experiments

Method for estimating the activation energy of tritium diffusion in ceramics during reactor irradiation



Selected sector of the experiment associated with changes in reactor power (at 16th day of irradiation) for calculations

It is necessary to plot the dependences of the logarithm of the flux on the reverse temperature in the form:

$$\log\left(J\left(\frac{1}{T}\right)\right) = A + \frac{B}{T},$$

And then linearize them, it is possible to determine the activation energy of diffusion by the formula:

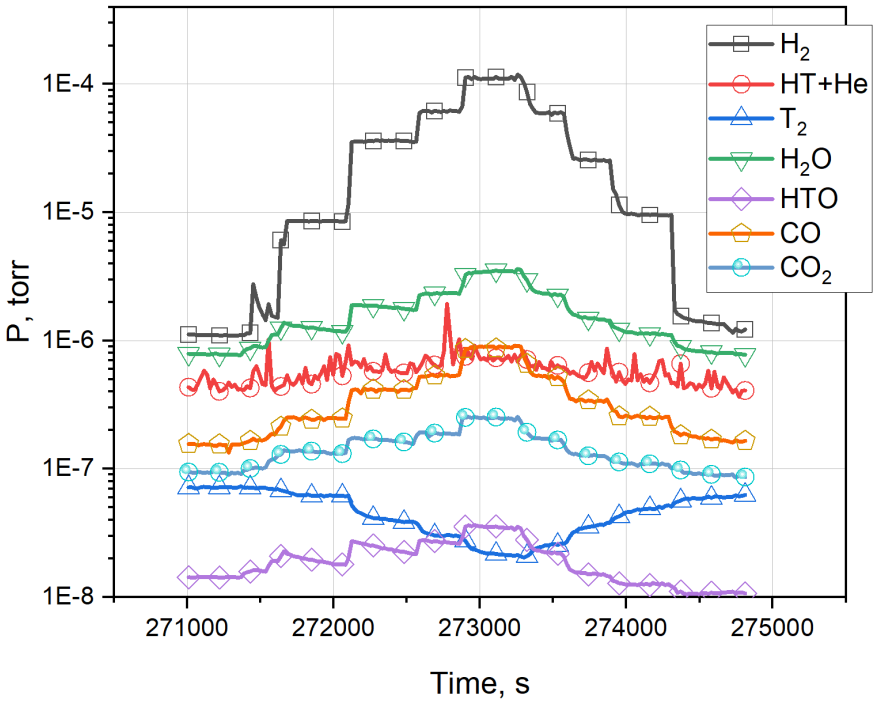
$$E_D = -B \cdot R / \log(e),$$

In this case the coefficients A and B are equal to:

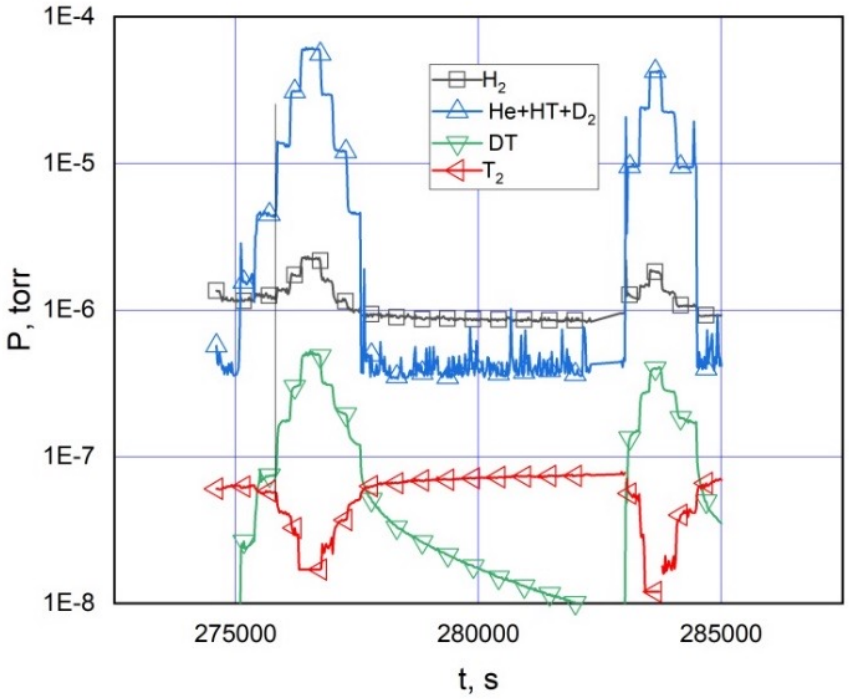
$$A = \log\left(S \cdot D_0 \cdot \frac{\partial C(r)}{\partial r}\Big|_{r=R_0}\right)$$

$$B = -E_D \cdot \log(e)/R$$

Results of reactor experiments with lithium ceramics with gas injection



Change of partial pressures of gases in the chamber of the facility at different hydrogen flows into the ampoule during irradiation time



Change of partial pressures of gases in the facility chamber at different deuterium flows into the ampoule

Features of the experiments – peak release of helium from lithium ceramics

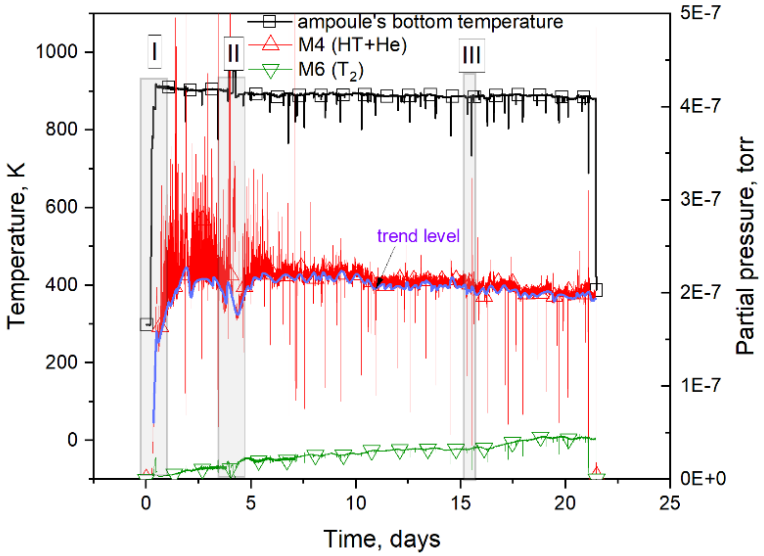
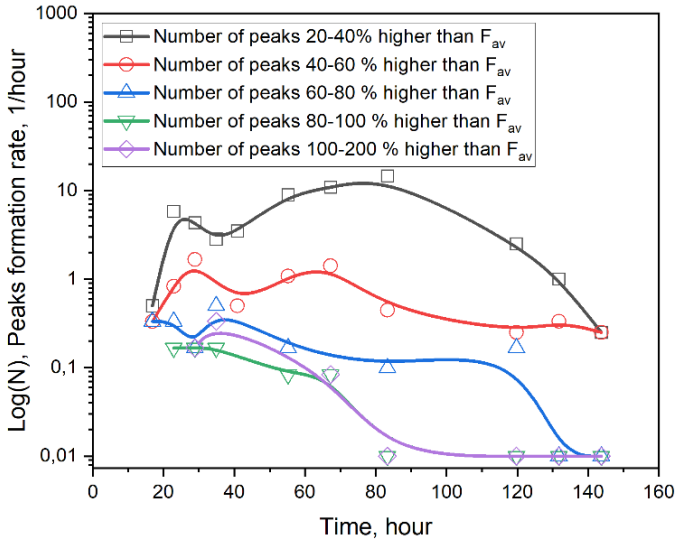
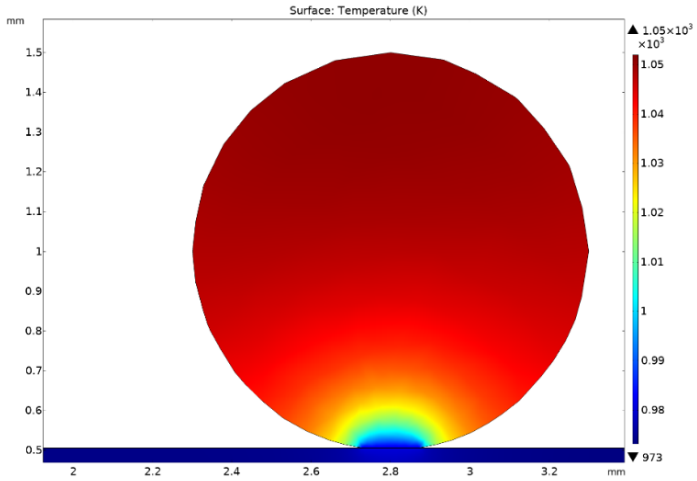


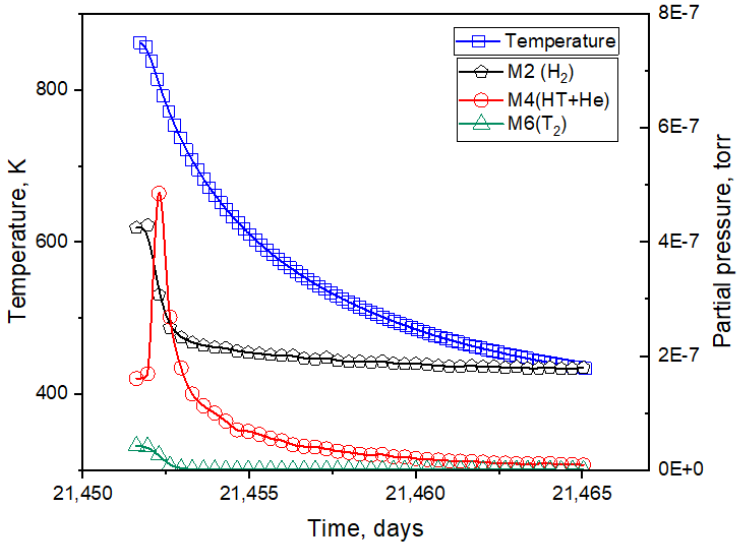
Diagram of the reactor experiment regarding the irradiation of lithium ceramics



Time dependence of the rate of helium peak release from lithium ceramics



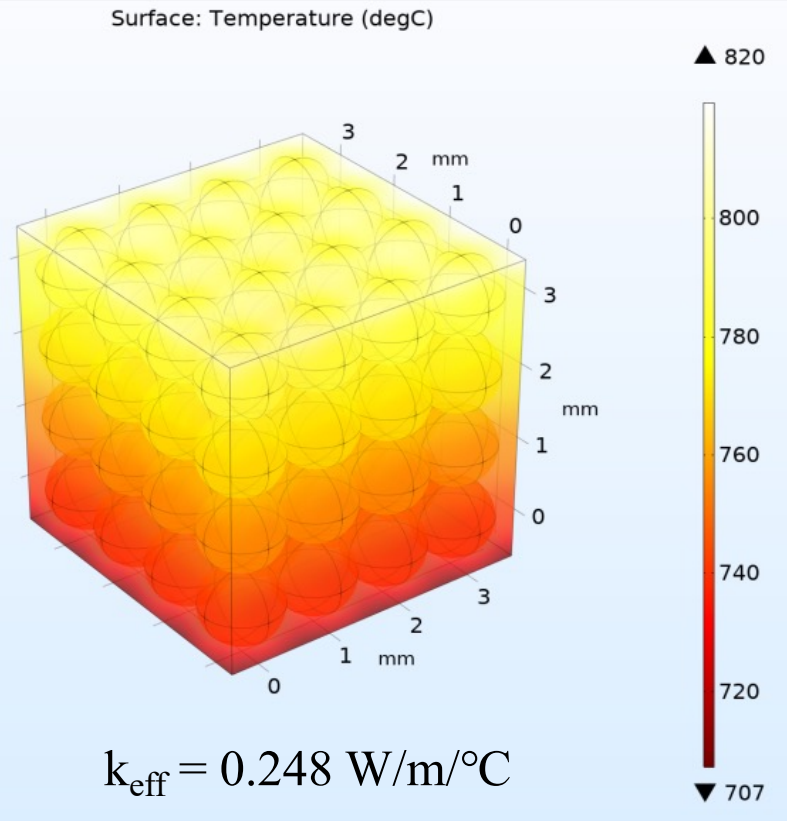
Equilibrium temperature field along the pebble of lithium metatitanate at the bottom of the capsule during irradiation



Enlarged area of the experiment section during a reactor stoppage

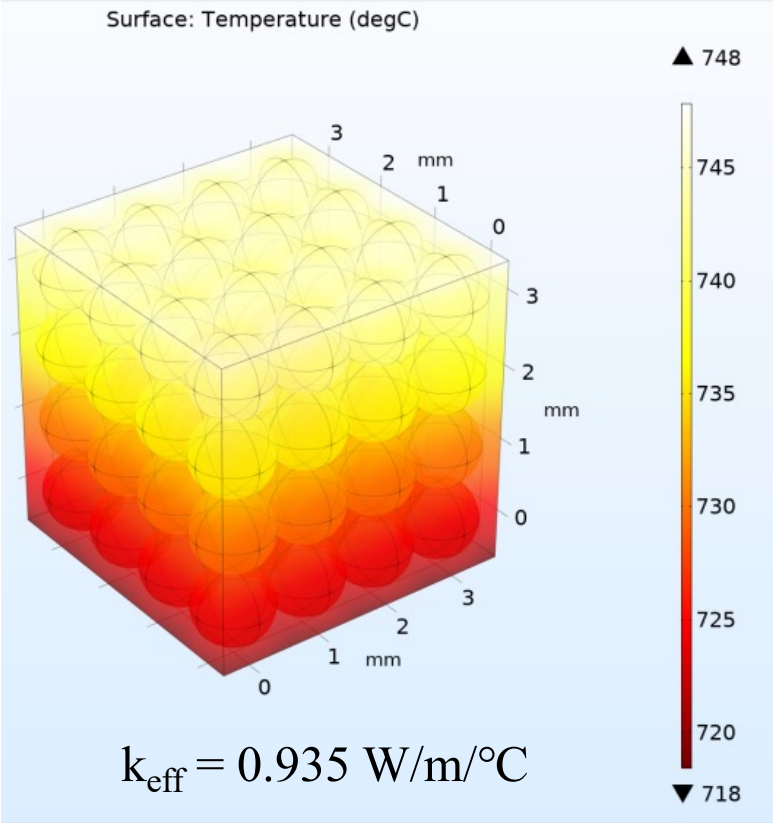
Calculation of temperature fields in a lithium ceramic pebble bed

10 Pa He (Vacuum extraction)



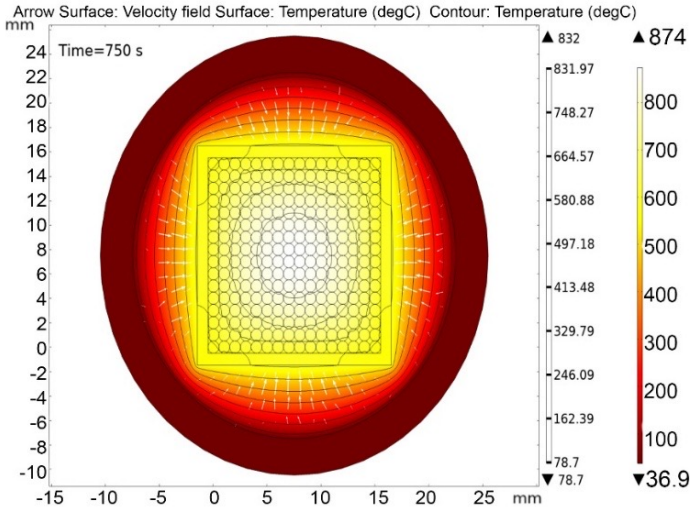
Stationary temperature distribution over the pebble bed at a helium pressure of 10 Pa

100 kPa He

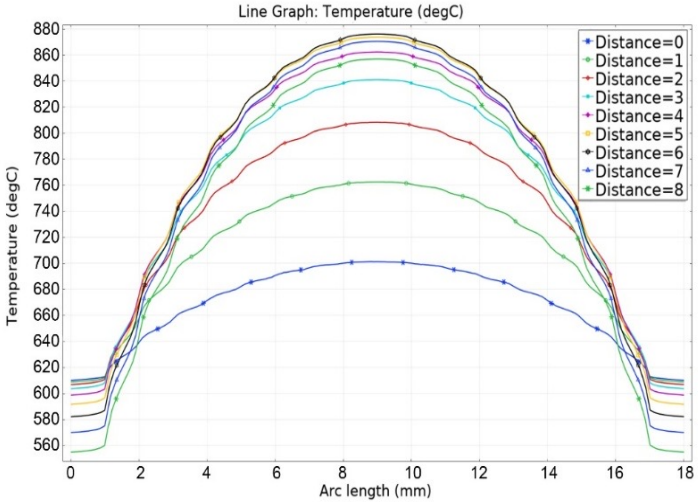


Stationary temperature distribution over the pebble bed at a helium pressure of 100 kPa

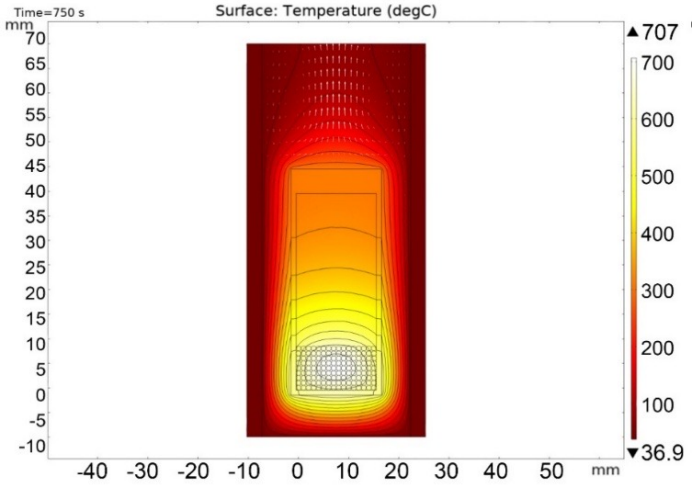
Calculation of temperature fields in a lithium ceramic pebble bed



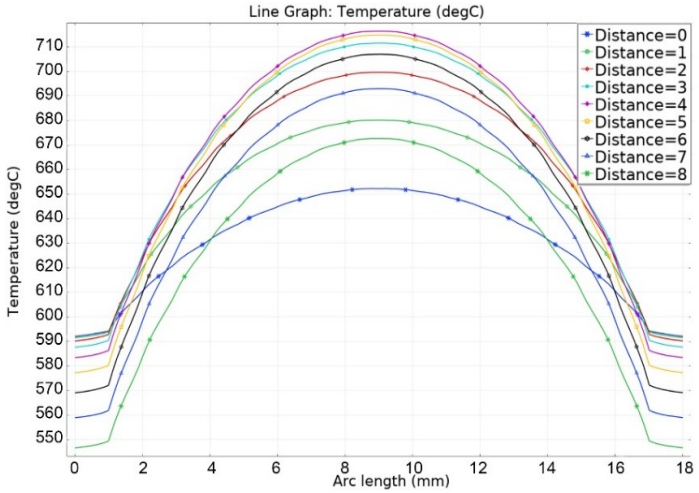
Final temperature distribution over the ampoule device and pebble bed at time 750 s (cross section through the 5-th layer of pebbles) at 10 Pa helium pressure



Temperature distribution over the layers of the pebble bed (0 - bottom layer, 8 - top) and across the capsule walls at 10 Pa of helium pressure



Final temperature distribution over the ampoule device and pebble bed at time 750 s at 100 kPa of helium pressure



Temperature distribution over the layers of the pebble bed (0 - bottom layer, 8 - top) and along the capsule walls at 100 kPa of helium pressure

Comparison between vacuum extraction method and purge gas method

Vacuum extraction		Purge gas method	
+	-	+	-
❑ the ability to fully control the gas phase composition in the experimental chamber, which significantly facilitates the analysis of possible reactions on the ceramic surface associated with the release of both tritium-containing molecules and helium; ❑ the ability to conduct experiments to study the release of tritium from lithium ceramics with a given change in the gas composition in the experimental chamber; ❑ high sensitivity and almost complete lack of delay in registration of changes in the composition of the gas phase (including tritium-containing molecules and helium) when changing various experimental parameters (such as sample temperature; reactor power, etc.).	❑ it is established that there are some restrictions on the lower limit of the investigated temperatures of ceramics, which is due to the low thermal conductivity of the pebble layer; ❑ when the samples are placed in several layers in the gravel layer, a noticeable temperature gradient occurs, which is difficult to measure and often can only be estimated by calculation; ❑ it is better to use high-resolution mass spectrometers in the region of 4 masses to separate the signal for molecules HT and helium.	❑ experiment better simulates the conditions of real operation of ceramic pebble samples; ❑ minor gradient of pebble bed during reactor experiments.	❑ more complex methodical preparation of experiments; ❑ It is difficult to estimate the composition of the gas medium and specifically tritium molecules; ❑ as the gas passes to the tritium measurement system, the signal changes due to the diffusion of tritium molecules in the carrier gas, i.e. there is a certain difficulty in determining the real release rate from the samples with the signal (for areas where noticeable changes in release processes occur).

Conclusions

•**An experimental method of tritium vacuum extraction was developed and tested**, enabling *in situ* registration of gas release from lithium ceramics under neutron irradiation at the CIRRA facility of the WWR-K reactor in vacuum conditions (10^{-4}–10^{-6} Pa).

•**The method demonstrated high sensitivity to gas release dynamics:** impulsive helium emissions and stable tritium release mainly in the HT and T_2 molecular form were recorded, reflecting diffusion and surface reactions during sample heating.

•**It was established that the thermal conductivity of the ceramic pebble bed significantly decreases** under vacuum (10 Pa) compared with helium atmosphere (100 kPa) - from *0.935 W/m·°C* to *0.248 W/m·°C* - requiring precise in-capsule temperature control to prevent overheating above 900 °C.

•**The tritium release form changes over time:** the fraction of HT molecules decreases from ~85 % at the beginning of irradiation to ~70 % after 21 days, due to hydrogen depletion and redistribution of tritium between the ceramic bulk and surface.

•**The vacuum extraction technique proved effective for calibrating tritium diffusion and desorption models** in lithium ceramics and for assessing gas-removal efficiency in breeder blanket modules of future fusion reactors.

It is better to use high-resolution mass spectrometers in the region of 4th mass to separate the signal for molecules HT and helium.

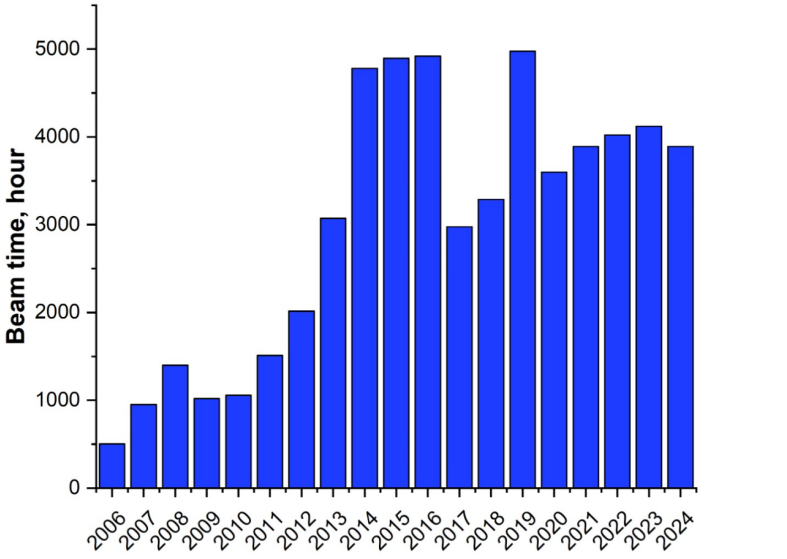
In 2025, the Institute of Nuclear Physics acquired a high-resolution quadrupole mass spectrometer. Installation and assembly work have been completed, and testing is planned in the near future.

Experiments with materials of fusion reactors under accelerator conditions

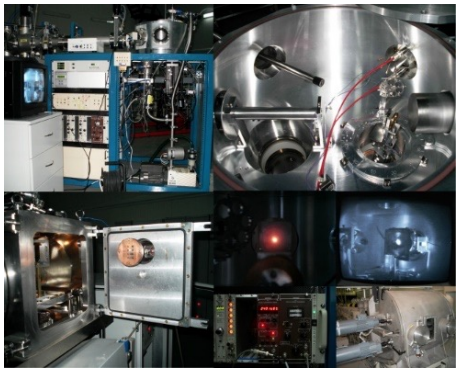
Cyclotron DC-60

Parameters:

Ions	**He, C, O, N, Ne. Ar, Kr, Xe, Hf, Ge, Fe, Cr, Ni**
Energy	**0.4 ÷ 1.75 MeV/nucl**
Low-energy	**20 keV/charge**
Current	**1 ÷ 400 nA**
Fluence	**10^{11} – 10^{18} ion/cm^2 0.1-100 dpa**

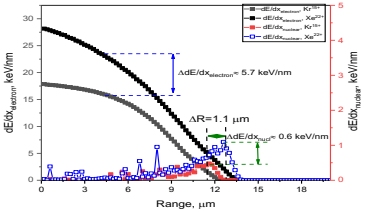


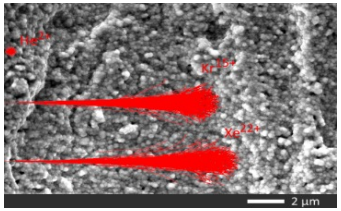
Beam operating of the DC-60 cyclotron for the period 2006 - 2024

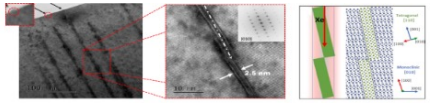


Modeling radiation damage that simulates real operating conditions

- *Irradiation with heavy ions Kr, Xe simulating the impact of fission fragments*
- *Irradiation with low-energy ions simulating gas swelling processes*



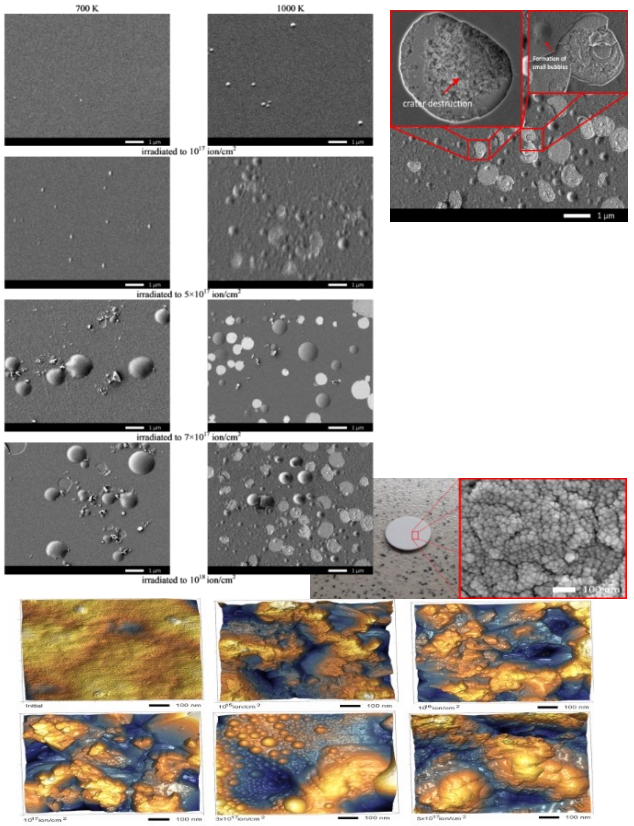


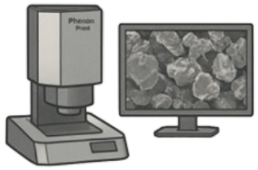


Modeling of ceramic destruction processes associated with the accumulation of radiation damage, simulating real operating conditions

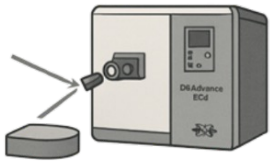
Heavy ion accelerator DC – 60

Modeling of gas swelling processes associated with helium accumulation in the surface layers of lithium ceramics

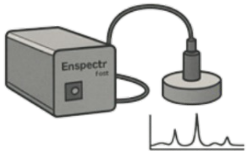




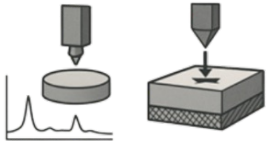
Scanning Electron Microscopy



X-ray Diffraction
Structural Parameters



Phase Composition



Mechanical Testing
Hardness of Ceramics

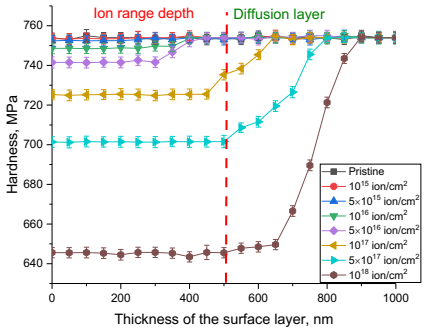
Comprehensive characterization of radiation damage, including methods for determining strength and thermophysical parameters, as well as visualization of defects

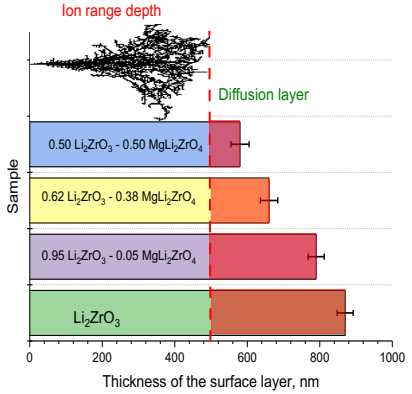
Available methods for characterization of different materials irradiation at DC-60 cyclotron

Research in the field of blanket materials using accelerator technologies

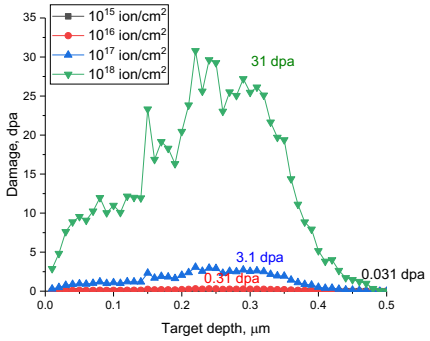
Opportunity:

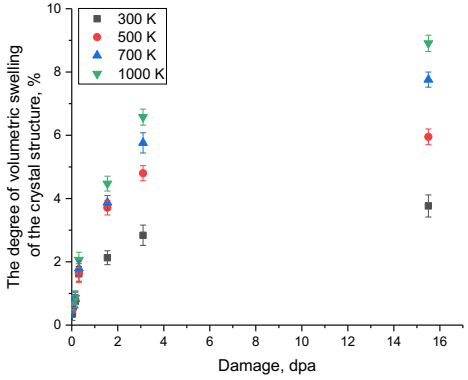
- Modeling of helium swelling processes of near-surface layers of lithium-containing ceramics (including the possibility of irradiation at high temperatures up to 1000 K)
- High-dose irradiation of samples with low-energy He^{2+} ions (40 keV) with doses up to 50 – 60 dpa (fluence of the order of 10^{18} cm^{-2})
- Simulation of the most approximate conditions for the accumulation of radiation damage during high-dose irradiation, including high-temperature exposure



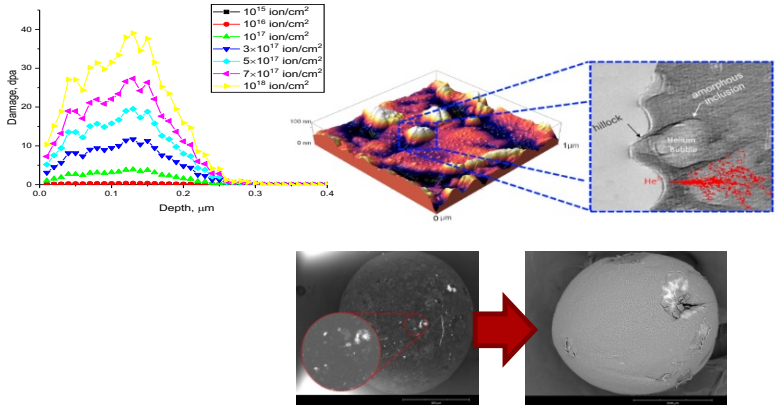


Study of the mechanisms of destruction of surface layers and diffusion of ions into the depths of samples under high-dose irradiation





Analysis of the destruction of the surface layer of lithium-containing ceramics depending on the irradiation temperature

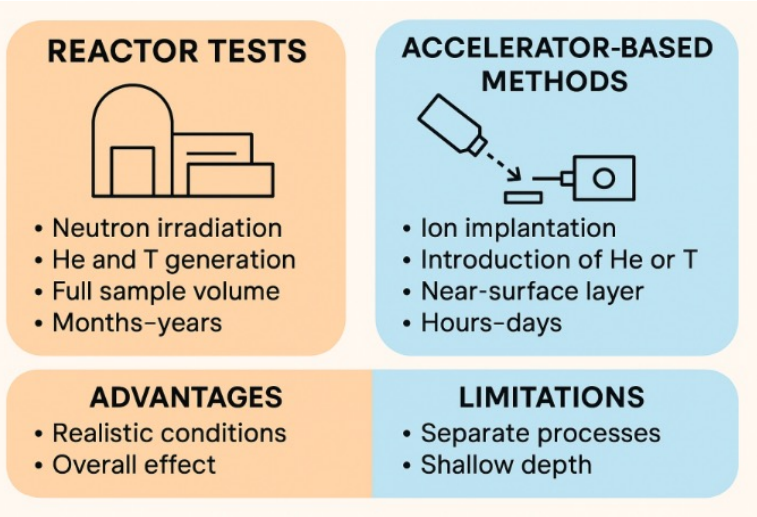


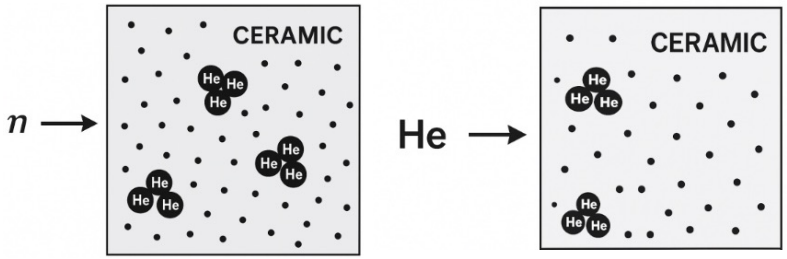
Visualization of radiation damage in high-dose irradiation

Comparison of Reactor Tests and Accelerator Technologies for Modeling Gas Swelling

Comparison of reactor tests and the possibilities of using accelerator technologies for modeling gas swelling processes.

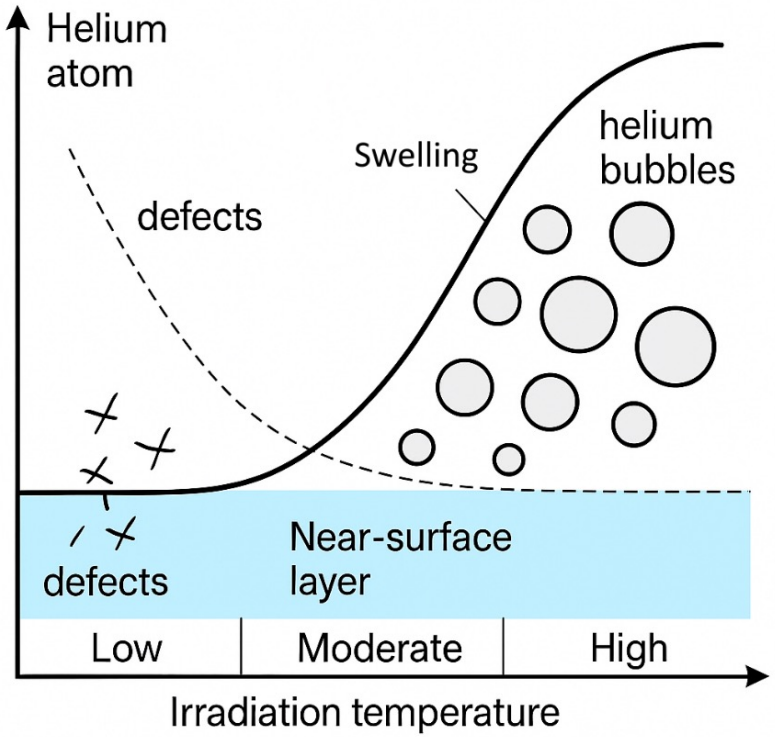
Criterion	Reactor Tests	Accelerator Technologies
Radiation source	Neutrons (broad spectrum)	Ions (monoenergetic beams)
Reaction products	He + T simultaneously	Separate introduction of He or H/T
Research scale	Entire sample volume	Near-surface layer (up to 0.2 - 1 μm)
Experiment duration	Months–years	Hours–days
Cost	Very high	Significantly lower
Parameter control	Limited	High precision
Realism of conditions	Maximally close to reactor environment	Simulation of individual processes



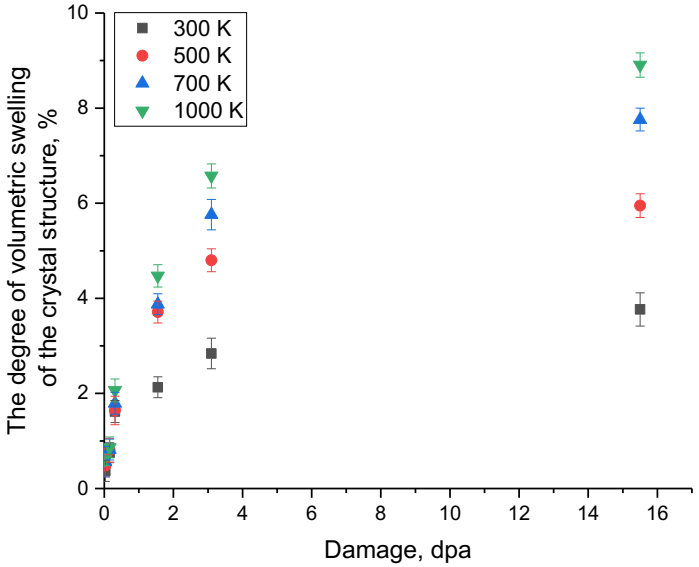


Modeling of processes of volumetric swelling

The aim of the experiments is to determine the role of irradiation temperature on the degradation rate of the surface layer of lithium-containing ceramics depending on the degree of damage.



Scheme of the relationship between helium accumulation in the near-surface layer and irradiation temperature

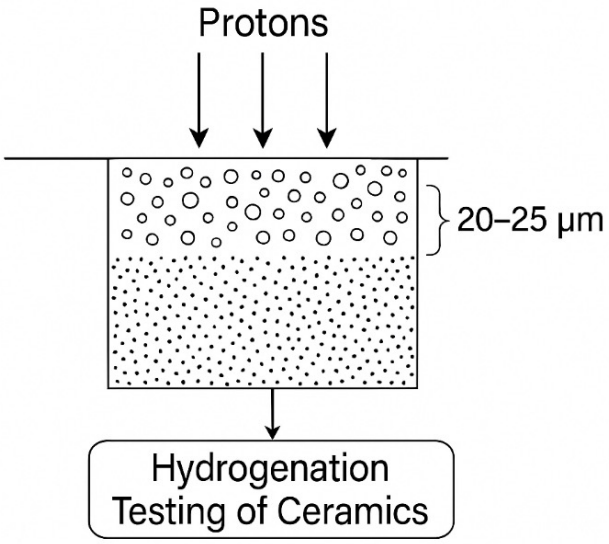


Analysis of the destruction of the surface layer of lithium-containing ceramics depending on the irradiation temperature

Irradiation capabilities at the UKP-2-1 accelerator

Main features

- *Irradiation with protons with energy from 0.5 to 1.5 MeV*
- *Modeling of hydrogenation processes in near-surface layers at a depth of about 10 - 20 μm*



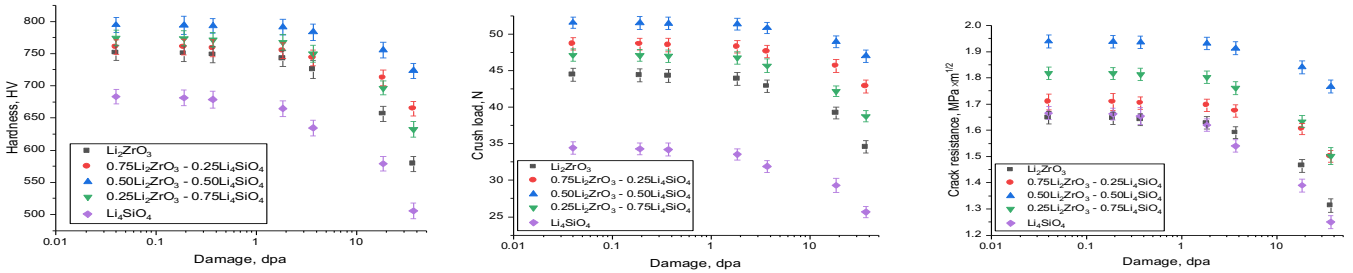
Technical capabilities of UKP-2-1 provide accelerated ion beams from hydrogen to plutonium with energies in the range from 300 keV to 4 MeV and beam currents on the target from a few nanoamperes to tens of microamperes. Due to a number of technical improvements an energy spread in the proton beam of ~ 100 eV at ion energy of 1.5 MeV was achieved.

Main limitations

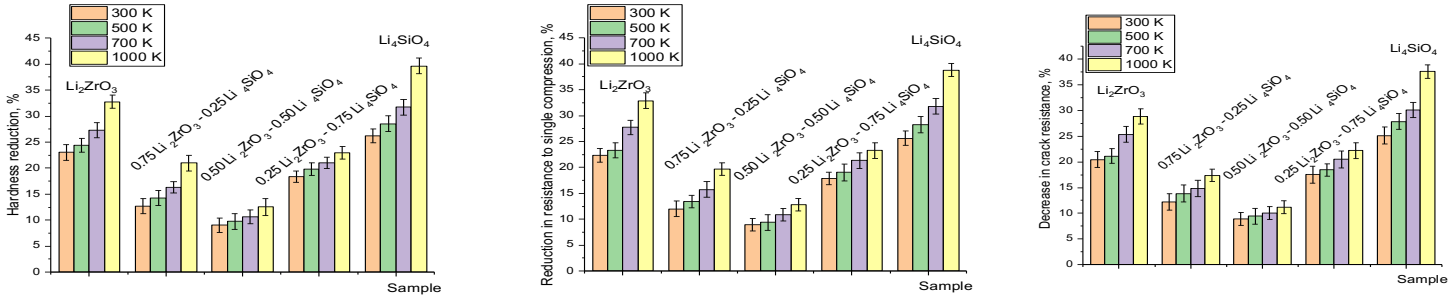
- *Low degree of accumulation of atomic displacements (no more than 15 dpa for lithium ceramics)*

Irradiation capabilities at the UKP-2-1 accelerator

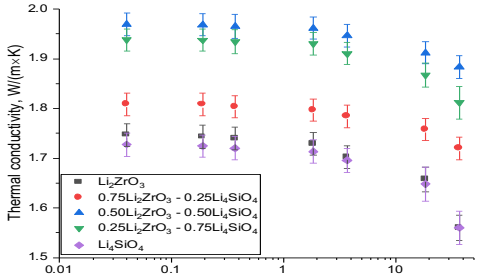
Study of hydrogen embrittlement processes of lithium-containing ceramics and the kinetics of degradation of structural, thermal conductivity and strength properties depending on the irradiation dose and the concentration of implanted hydrogen



Results of evaluation of changes in strength parameters of $(1-x)Li_2ZrO_3 – xLi_4SiO_4$ ceramics exposed to proton irradiation



Results of changes in the softening factors of the main strength parameters of samples depending on the irradiation temperature



Results of changes in the thermal conductivity coefficient of the studied ceramics depending on the magnitude of atomic displacements

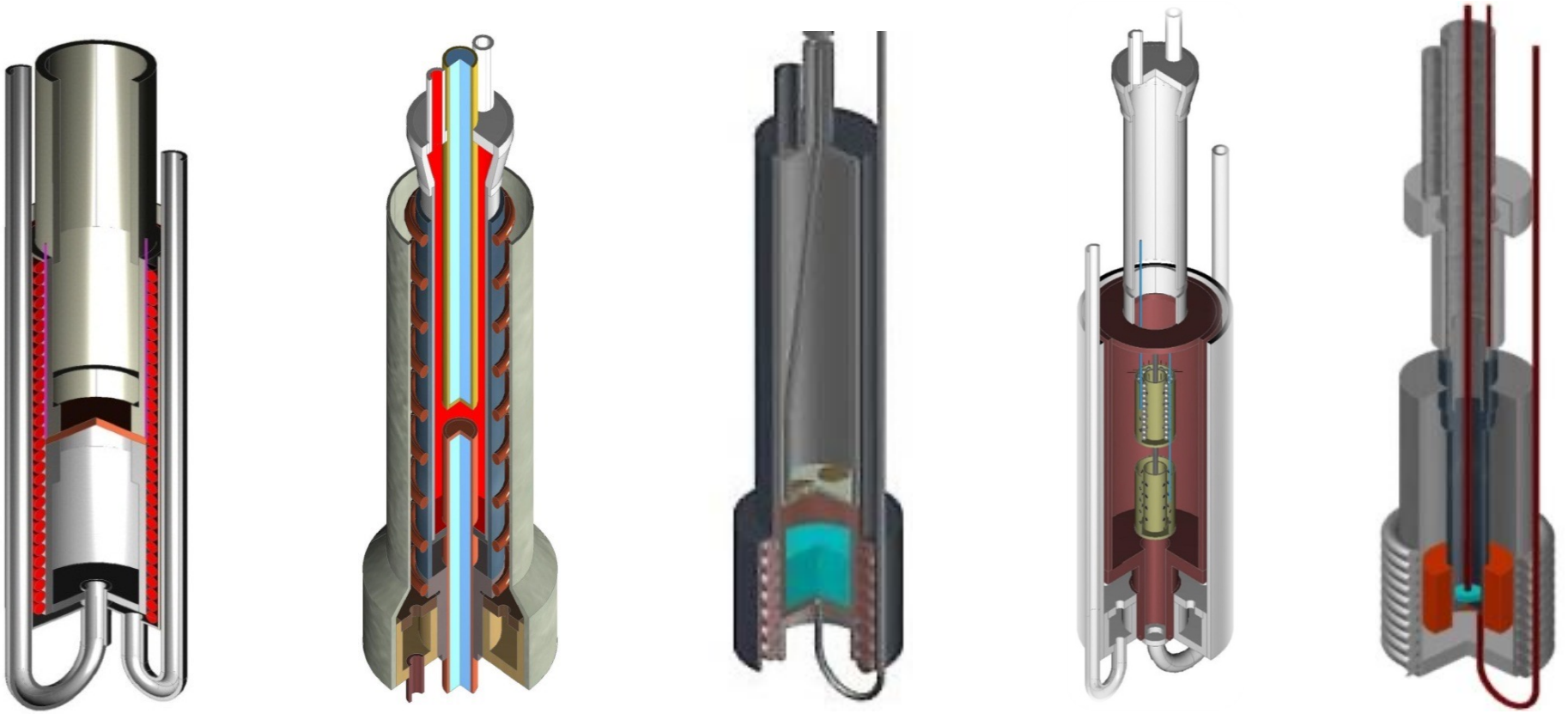
Tests of structural and functional materials of fusion reactors under reactor irradiation conditions

Experiments for materials science

During many years, the following materials were studied in out-of-pile and in-pile experiments: beryllium of various grades (production of RK and USA); graphite's, including FP-479 (Germany), which is used as coating in KTM vacuum chamber; molybdenum; tungsten of various grades made in Germany (project CRP IAEA); stainless steels; low activated alloys and steels considered as the candidate structural materials for fusion reactors.

#	Material	Experimental conditions	Main results
1	Graphite – RGT, FP-479	T ~ 300-1800 K Total fluence – $3.2 \cdot 10^{18}$ n/cm^2	Temperature dependences of diffusion, permeability and solubility coefficients
2	Tungsten	T ~ 300-1400 K Reactor power – 6 MW	Temperature dependences of diffusion and solubility coefficients
3	Stainless steels – SS316IG, MANET, F82H, 08Ch18N10T, 07Ch16N6	T ~ 300-900 K Reactor power - 6 MW Hydrogen isotopes pressure - 10^{-10} torr	Temperature dependences of diffusion, permeability and solubility coefficients, capture constant and traps concentration
4	Beryllium – DV-56, TShG-200, etc.	T ~ 300-1500 K Reactor power - 6 MW	Temperature dependences of diffusion coefficients
5	Vanadium and alloys – V4Cr4Ti, V10Cr5Ti	T ~ 300-1000 K Reactor power – 1-6 MW	Temperature dependences of diffusion, permeability and solubility coefficients. Temperature of phase transition
6	Li, Li_2TiO_3, Li17Pb83, lithium CPS	T ~ 400-1100 K Reactor power – 0.5-6 MW	Temperature dependences of diffusion, capture constants, traps concentration and tritium release rates
7	Aluminized and silicon coatings on stainless steels	T ~ 300-900 K Reactor power - 6 MW Medium – Li17Pb83	PRF – permeability reduction factor of coatings, temperature dependence of permeability
8	CuCrZr alloy	T ~ 300-900 K Reactor power - 6 MW	Temperature dependences of diffusion, permeability and solubility coefficients

Experimental irradiation ampoule devices



Reactor Ampoule Devices (AD):
a – AD for experiments on hydrogen permeability (V, V4Cr4Ti, 12Ch18N10T, SS316IG, Ni, CuCrZr, Cu); b – AD for experiments on tritium permeation through the sample with/without coating (MANET, F82H); c – AD for experiments on tritium generation and release from lithium-containing materials (Li, Pb83Li17, lithium CPS); d – AD for experiments on energy release from lithium-containing materials (Li2TiO3); e – AD for experiments on thermal desorption (Be, lithium CPS)

Investigated materials

Materials	Vanadium V	Vanadium alloy V4Cr4Ti	Stainless steel 12Cr18Ni10Ti, MANET, F82H (with / without coating), SS316IG (ITER Grade), Ni	CuCrZr alloy and Cu	Lithium-containing materials Ceramics Li_2TiO_3 (96% ^6Li), lead-lithium eutectic $Pb_{83}Li_{17}$, lithium and lithium CPS	Beryllium TV-56,	Tungsten W
Temperature ^0C	from 20 up to 700	from 20 up to 700	from 300 up to 700	from 400 up to 700	from 300 up to 1000	from 50 up to 900	from 700 up to 1000
Isotopes	H, D	H, D	H, D, T	H, D	H, D,T	H,D	D

The main parameters of interaction of isotopes with these materials (such as: diffusion, permeability and solubility coefficients), including in the process of reactor irradiation were obtained in the experiments

IVG.1M research reactor

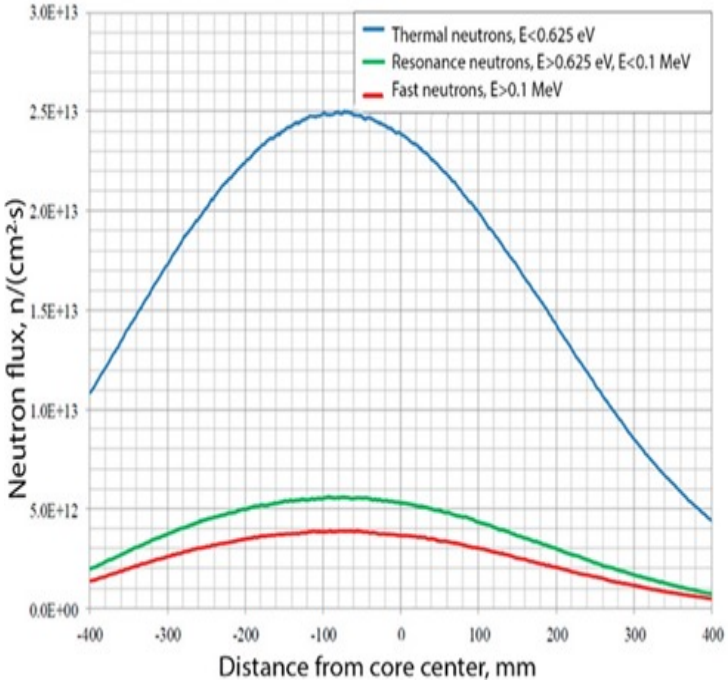
The IVG.1M control panel

TECHNICAL PARAMETERS OF THE IVG.1M REACTOR

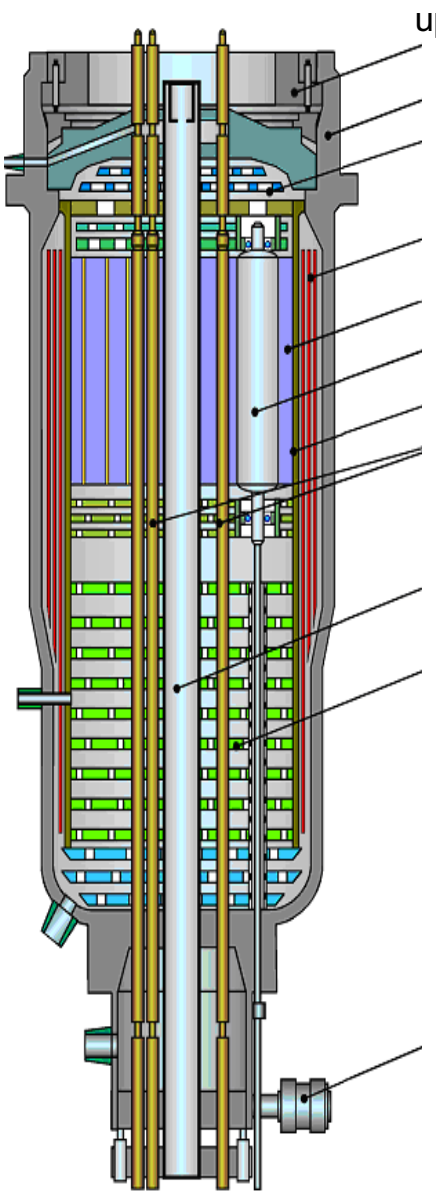
Thermal Power	6 MW
Core effective diameter	548 mm
Core height	800 mm
Thermal neutron flux density	
1.4×10^{14} n/cm^2·s	

Reactor view

Upper part of the IVG.1M reactor (Up-reactor room)

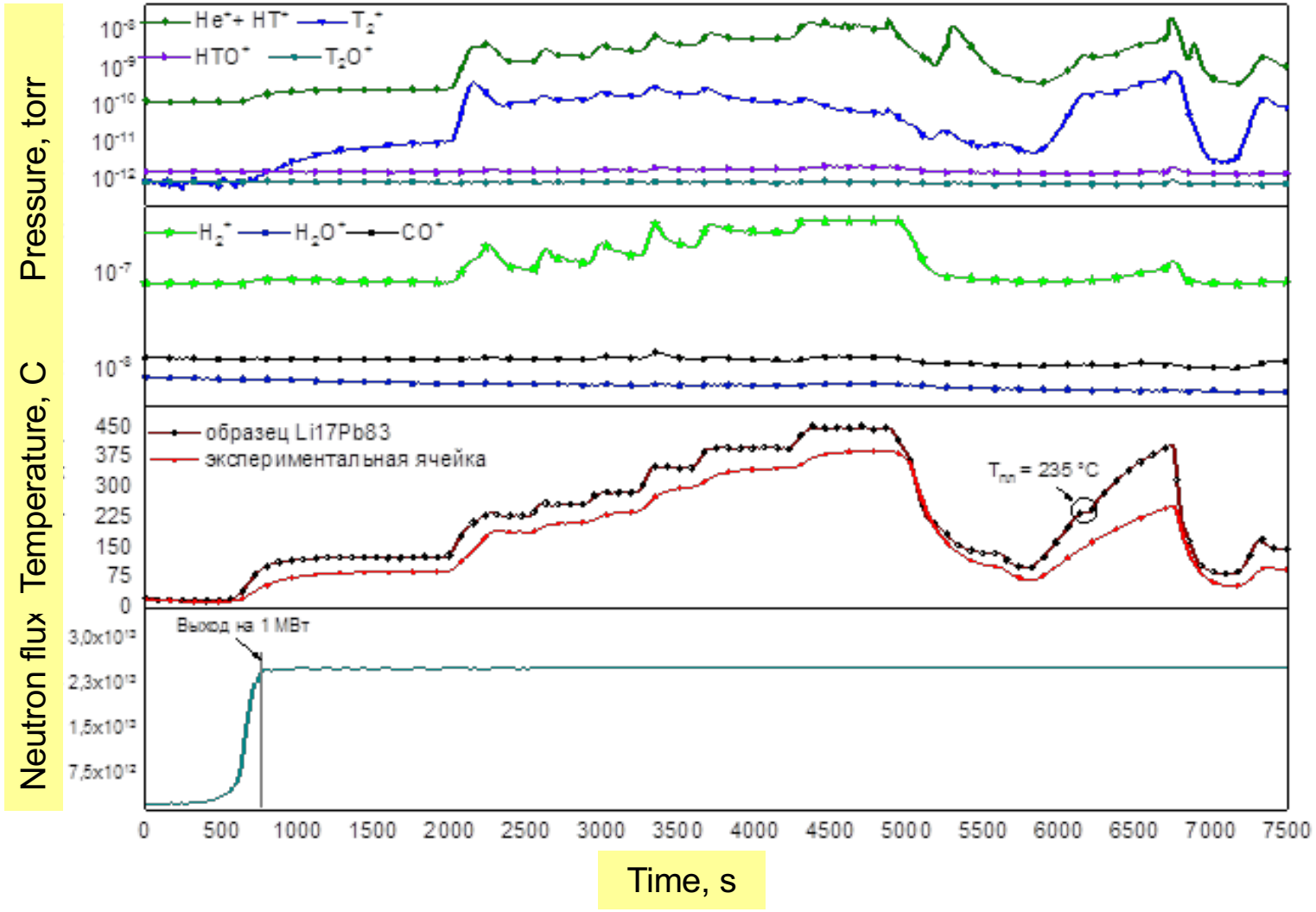


Neutron flux distribution by height of the IVG.1M experimental channel



Scheme of a vertical section of the IVG.1M reactor

In-situ experiment results at IVG1.M

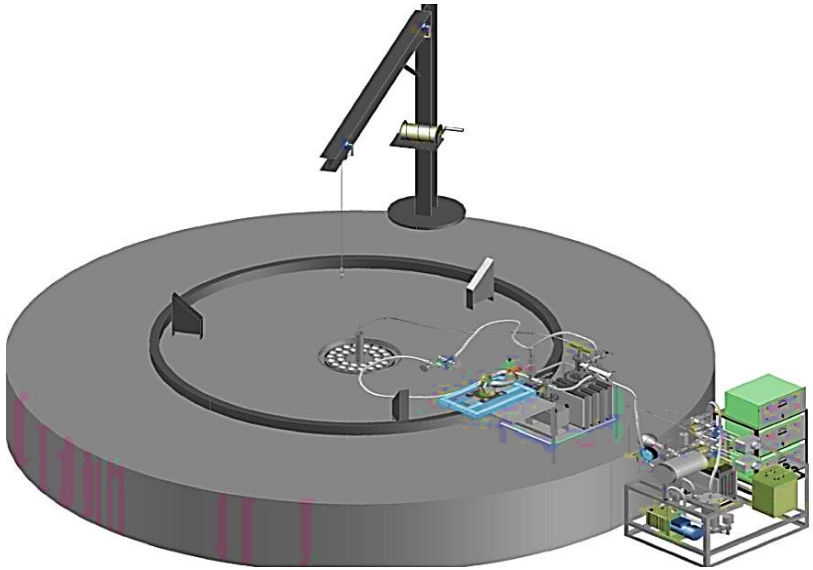


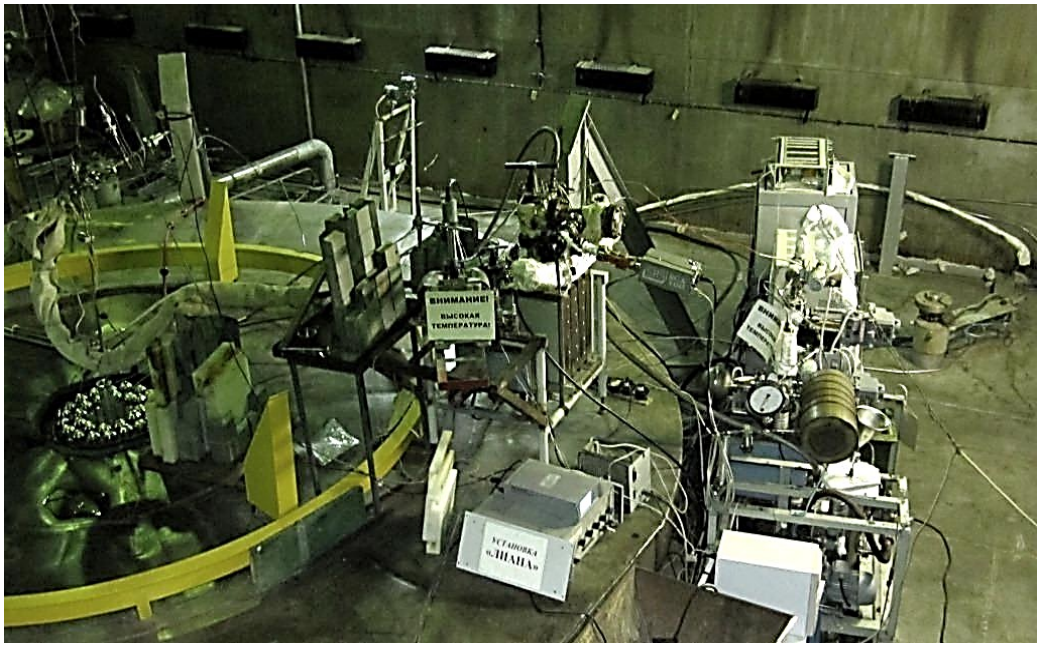
LIANA experimental test bench

The LIANA test bench is designed to study the interaction of hydrogen isotopes with nuclear reactors' (NR) and fusion reactors' (FR) materials during neutron irradiation at the reactor IVG.1M at different temperatures and gas pressures.

Research methods

- hydrogen permeability (HP) method;
- gas absorption method;
- gas degassing, thermodesorption method (TDS).

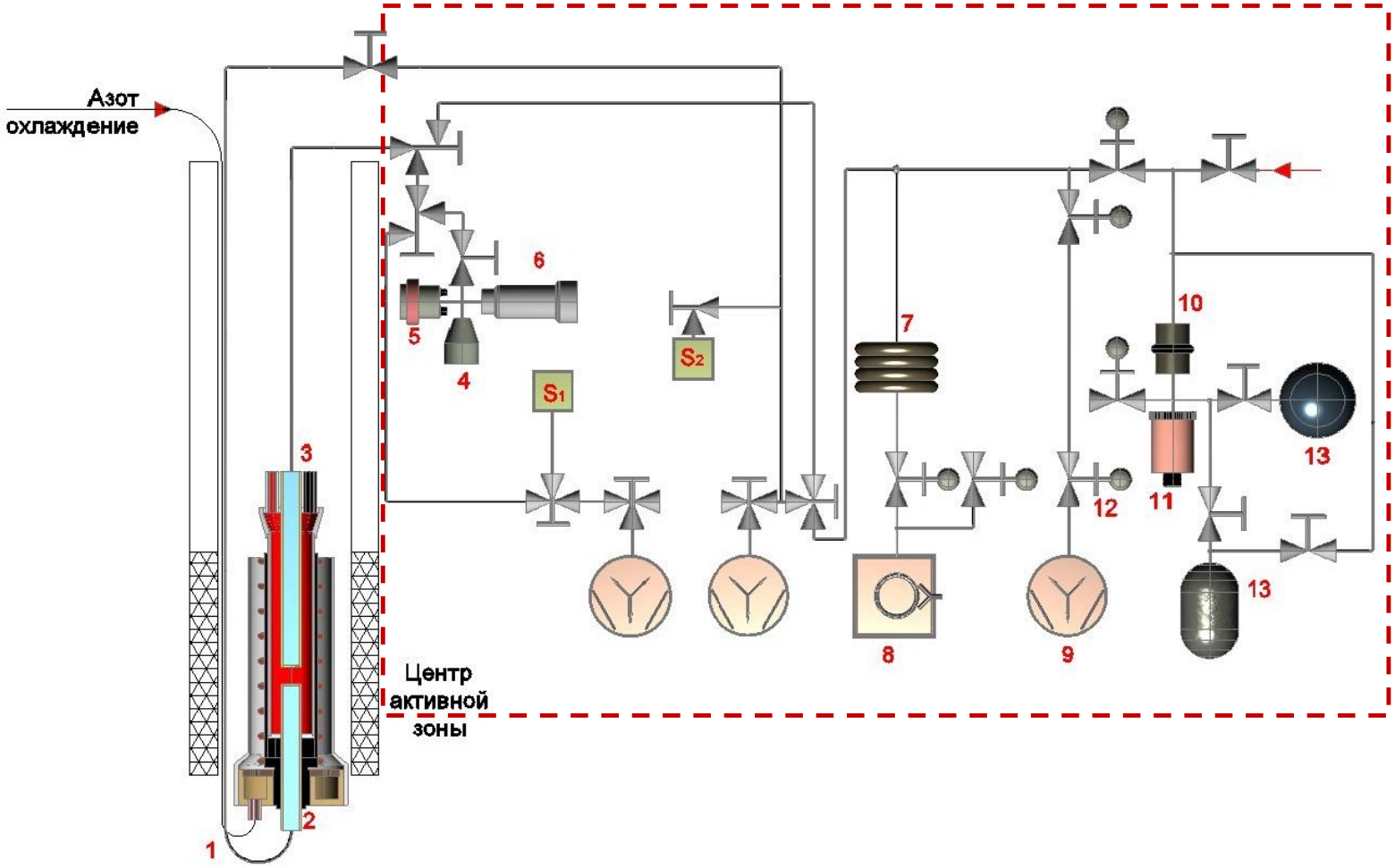




Technical characteristics of the bench

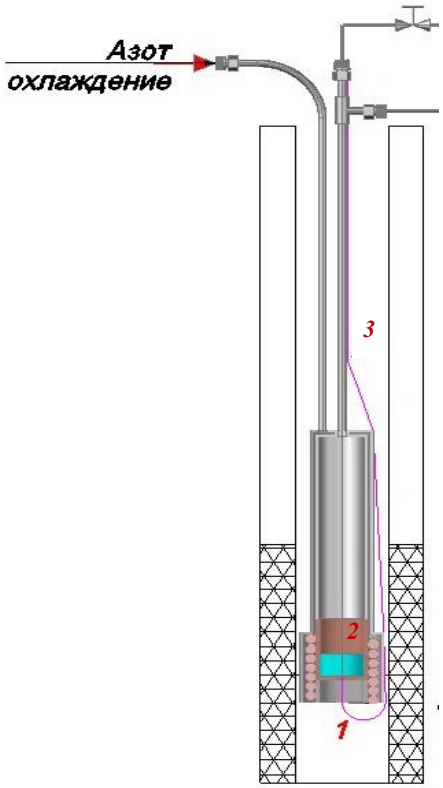
Temperature range of the research	293...1027 K
Hydrogen isotope pressure range on the inlet side of the sample	10^2 ...10^6 Pa
Pressure range of hydrogen isotopes in the measuring path	10^{-4}...10^{-7} Pa

Schematic diagram of the LIANA test bench in experiments with two tube samples

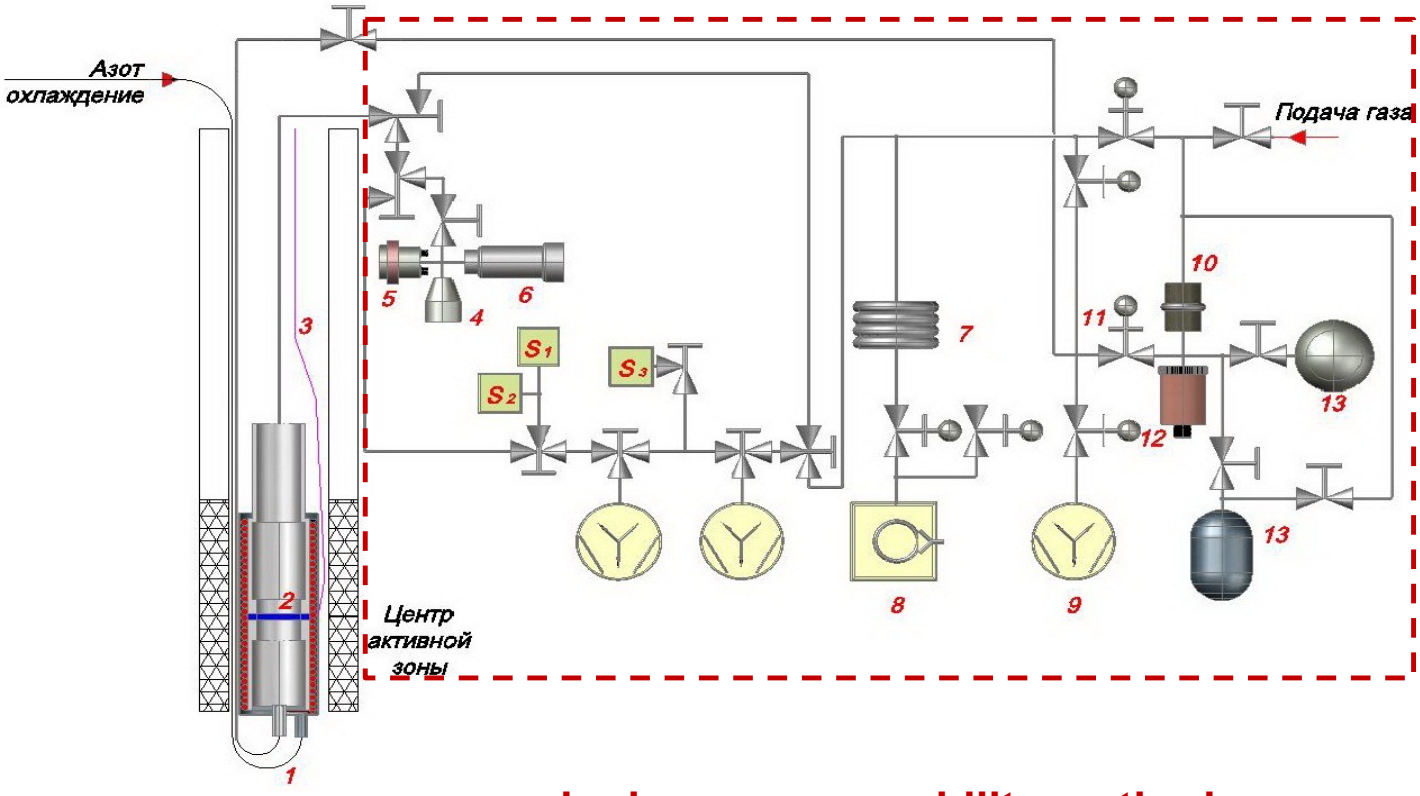


1 - ampoule device with samples; 2,3 - investigated sample; 4 - ConvecTorr pressure sensor; 5 - ICD-100 pressure sensor; 6 - ASG 2 pressure sensor; 7 - nitrogen trap; 8 - fore vacuum pump; 9 - high-vacuum pump; 10 - hydrogen filter; 11 - PMDG pressure sensor; 12 - vacuum valve; 13 - tank for spectrally pure hydrogen; S1, S2, - quadrupole mass spectrometers ;

Schematic diagram of the LIANA test bench in experiments by HP and gas absorption methods



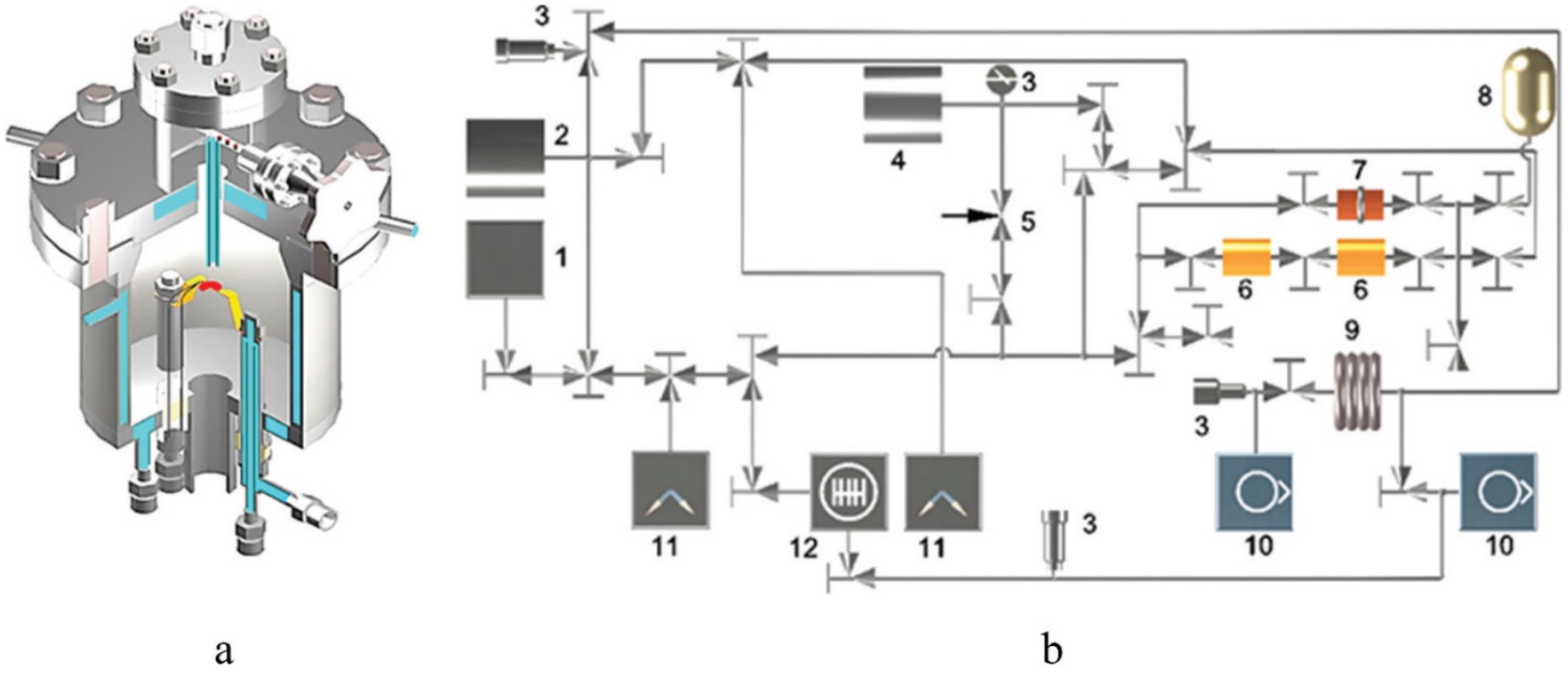
gas absorption
and thermal desorption methods



hydrogen permeability method

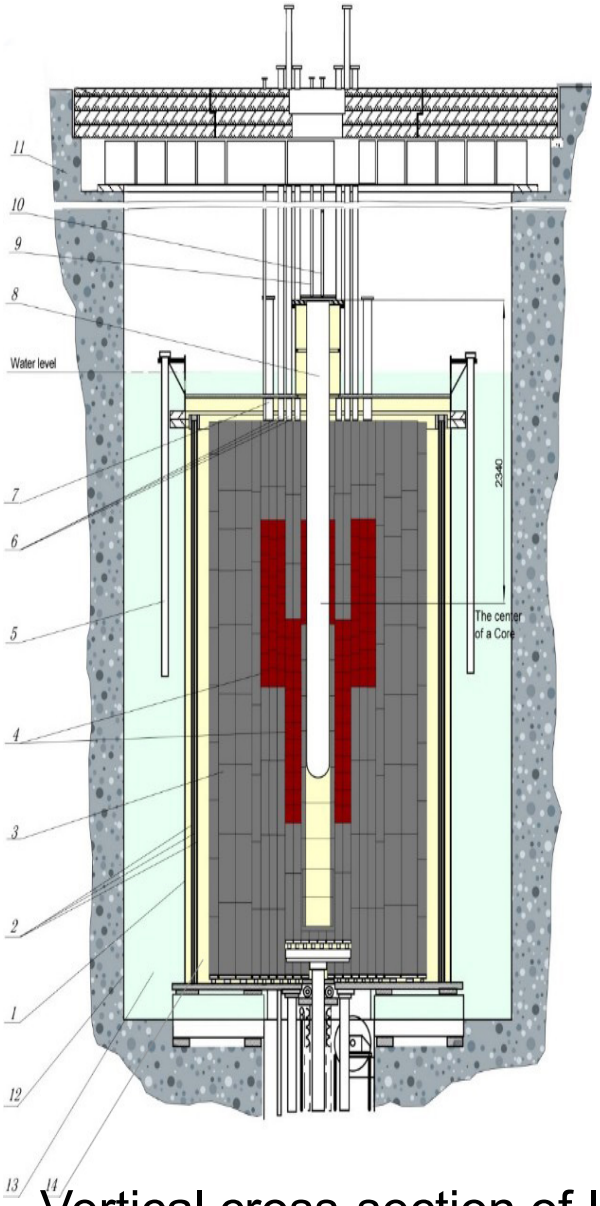
1 - ampoule device with samples; 2,3 - investigated sample; 4 - ConvecTorr pressure sensor; 5 - ICD-100 pressure sensor; 6 - ASG 2 pressure sensor; 7 - nitrogen trap; 8 - fore vacuum pump; 9 - high-vacuum pump; 10 - hydrogen filter; 11 - PMDG pressure sensor; 12 - vacuum valve; 13 - tank for spectrally pure hydrogen; S1, S2, S3, - quadrupole mass spectrometers, RGA-100

Thermodesorption facility "VIKA" for studying the interaction of hydrogen isotopes with materials



The VIKA experimental facility: (a) vacuum chamber and (b) scheme of vacuum system, where 1 = mass spectrometer; 2 = vacuum chamber; 3 = pressure sensor; 4 = additional changeable chamber; 5 = leak valve; 6 = gas purification system; 7 = filter; 8 = tank; 9 = nitrogen trap; 10 = forvacuum pump; 11 = Penning-type pump; and 12 = turbomolecular pump

IGR research reactor

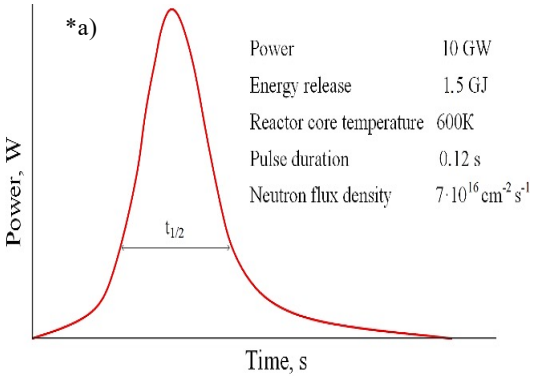


1 – casing;
2 – lateral shield (three shells);
3 – reflector;
4 – non-moveable and movable parts of the reactor core;
5 – ionization chamber channel;
6 – channel of the control rods;
7 – lateral experimental channel;
8 – central experimental channel;
9 – channel of physical measurements;
10 – channel of thermocouples;
11 – biological shield;
12 – water tank;
13 – cavity of cooling water (blue color);
14 – helium medium (yellow color)

Vertical cross-section of IGR*

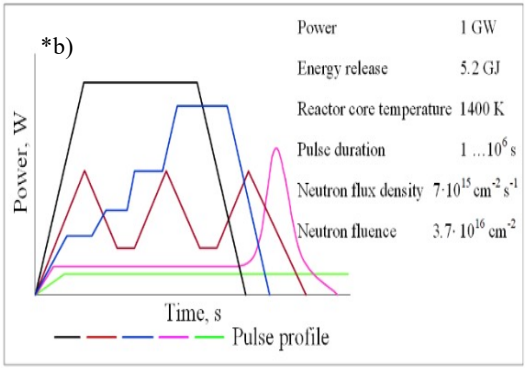
TECHNICAL PARAMETERS OF THE IGR

Thermal neutron flux density	7×10^{16} n/cm^{2}s
Thermal neutron fluence	3.7×10^{16} n/cm^2
Fast neutron fluence	1.1×10^{15} n/cm^2
Axial coefficient of thermal neutron flux irregularity	1.15
Axial coefficient of fast neutron flux irregularity	1.24
Minimum half-width of impulse	0.12 s



"Burst" mode

IGR Reactor Operation Modes

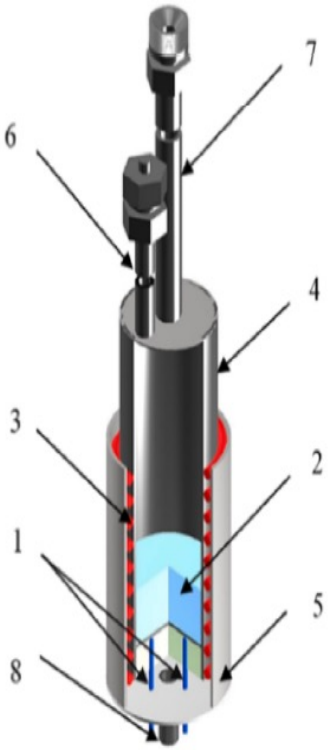


"Pulse" mode

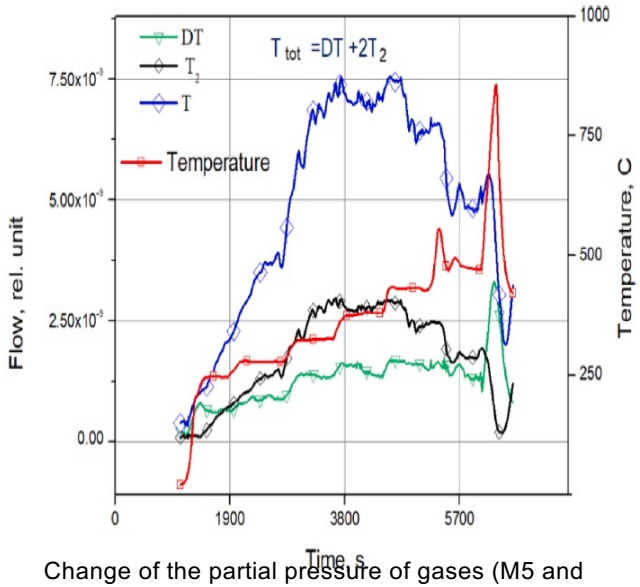
* V. Vityuk, A. Vurim. Method for determining the energy parameters in pulse reactor experiments. *Annals of Nuclear Energy*, Vol. 127, 2019.

*V.K. Tskhe et. al. The method of the reactivity margin calculation required for the IGR reactor start-up in the "Pulse" mode. *Annals of Nuclear Energy*, Vol. 168, 108875, 2022

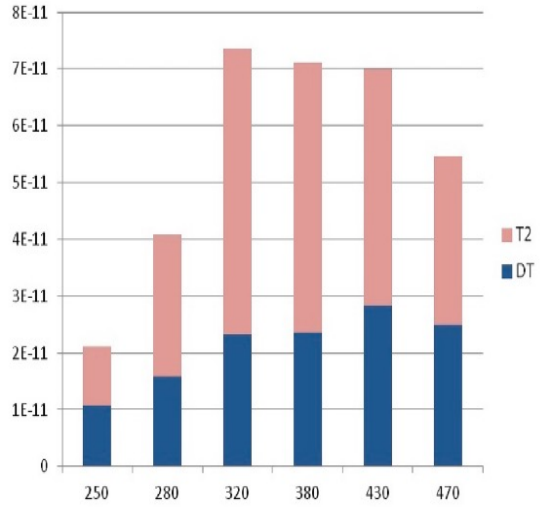
Study of lithium-containing materials (ceramics, eutectics) of fusion reactor blanket and the promotion of lithium technologies for protection of plasma-facing materials



Reactor ampoule device: 1 – thermocouples; 2 – lead–lithium eutectics; 3 – heater; 4 – body of ampoule device; 5 – cooling case; 6 – fitting for connection of filling device; 7 – fitting for AD degassing and connection with mass-spectrometer; 8 – nitrogen supply pipe



Change of the partial pressure of gases (M5 and M6) from lead-lithium eutectic Li15.7Pb in the presence of deuterium over the sample.



Absolute values of gas flow from the eutectic (mol/s)

The processes that occur in a lithium eutectics sample under the conditions of irradiation and deuterium supply are as follows:

1. Tritium and helium generation as a result of nuclear reactions:

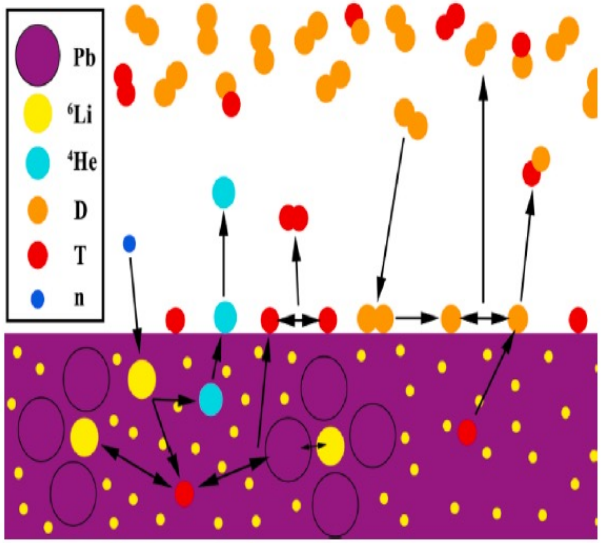
$^{6}Li + n \rightarrow T + {}^{4}He + 4{,}79\ MeV$ (1)

$^{7}Li + n \rightarrow T + {}^{4}He + n - 2{,}47\ MeV$ (2)

2. Capture and release of tritium by an unblocked lithium atom:

$Li + D \leftrightarrow LiD;\ Li + T \leftrightarrow LiT.$ (3)

3. Adsorption of deuterium molecules on the surface of lithium eutectics.
4. Dissociation of deuterium molecules into atoms.
5. Dissolution of deuterium atoms from the surface into the volume is assumed to be negligible since solubility of hydrogen isotopes in eutectics is rather low.
6. Exit to the surface from the near-surface layer of tritium and helium atoms (generated in the bulk of the sample).
7. Non– activation desorption of helium atoms from the surface.
8. Association of hydrogen isotopes atoms and desorption of D2, DT, T2 molecules from the surface of lithium eutectics.

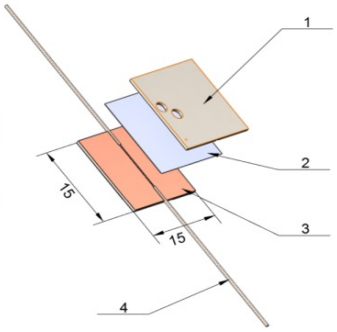


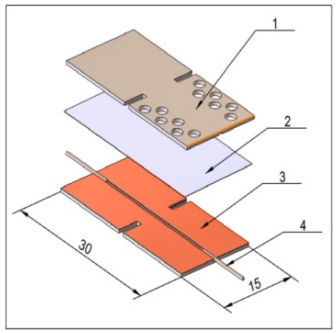
General scheme of the processes in a lithium eutectics sample during irradiation experiments

Tazhibayeva I, et al. Reactor studies of tritium release from lead-lithium eutectic Li15.7Pb with deuterium over the sample *Nuclear Materials and Energy 25 (2020) 100868*

Testing of radiation resistance of fibers and fiber optic sensors to study the behavior of sensors in ITER

The reactor tests conducted at IVG1.M and WWR-K reactors, comparative studies of the radiation resistance of optical fibers (WFs) and sensors based on fiber Bragg gratings (FBGs) in the process of mixed gamma neutron reactor irradiation in the near infrared range. Radiation-resistant fibers produced by FORC-Photonics (RF) with various coatings (copper, high-temperature polyimide, acrylate and aluminum) are also used for testing.

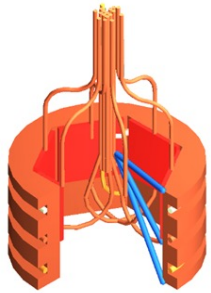




FBG-based sensors designs

Photographs of optical fibers coated with polyimide, acrylate, aluminum, and copper (from left to right).

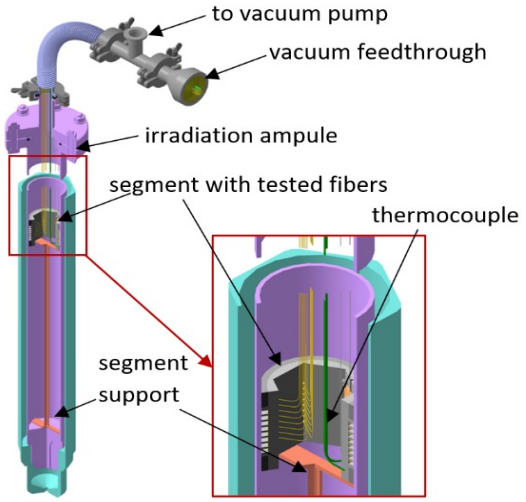
3D view of ampule device



3D view of cylindrical segment with a FBG-based sensors

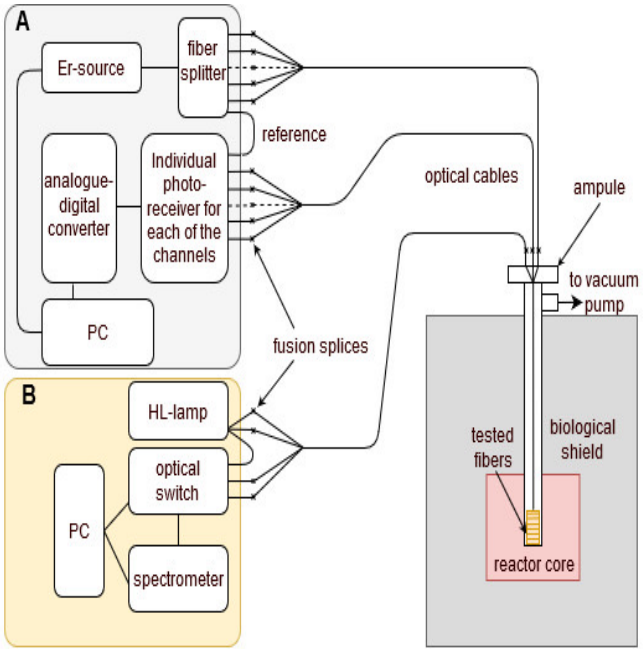
Ampoule device for studies OF at ITER conditions

- D'_γ~ 103 Gy/s
- D_γ~ 109Gy
- $\varphi(>0,1\ eV)\sim 10^{13} n/cm^2 s$
- $\Phi(>0,1\ eV) \sim 10^{20} n/cm^2$
- $T < 350\ ^\circ C$
- $P \sim 10^{-6} Pa$

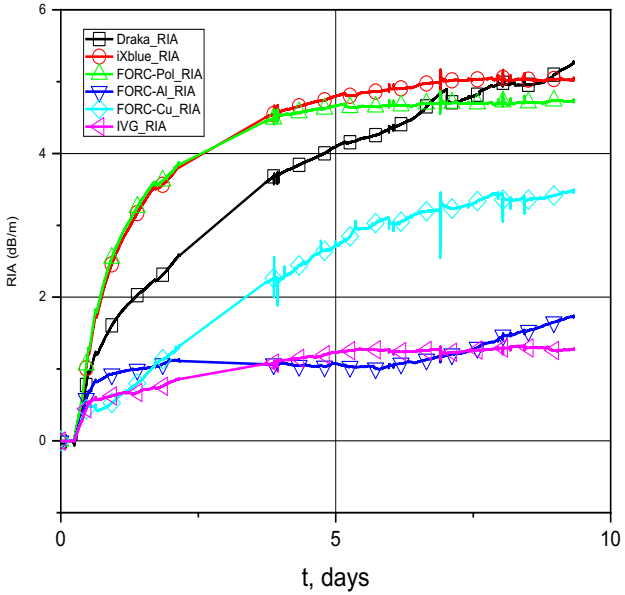


Testing of radiation resistance of fibers and fiber optic sensors to study the behavior of sensors in ITER

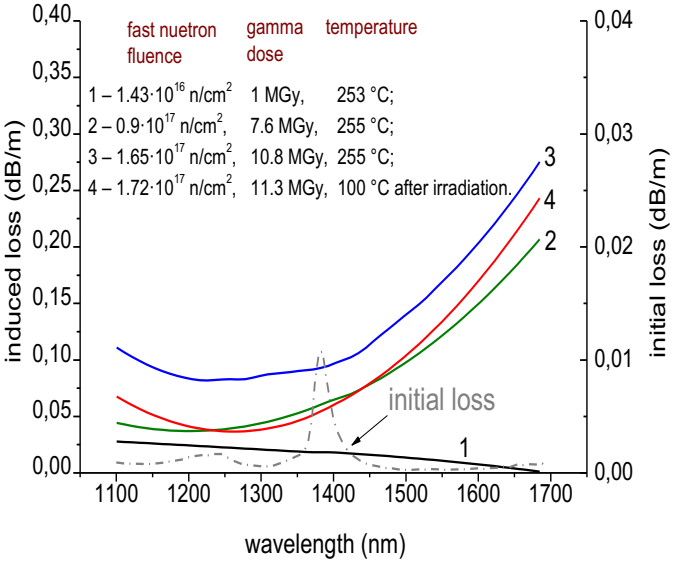
The reactor experiments included the following: measurements of radiation-induced attenuations in optical fibers, registration of the time dependence of the change in the spectrum of transmitted light from the selected source, measurements of the shift of the resonant wavelength and its amplitude in the temperature sensors.



Schematic diagram of the experiment on testing of optical fibers and optical fiber sensors on reactors IVG.1M and WWR-K



Dependence of changes in radiation-induced attenuations for different fibers on the irradiation time.

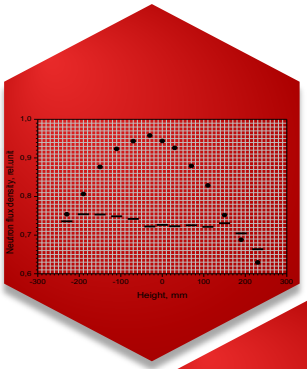


Spectrum of initial and radiation-induced optical losses in optical fibers with an undoped silica core and fluorinated cladding in a protective polyimide coating under reactor irradiation at a fast neutron flux of 2.4 10^{13} n / cm^2s and a dose rate of 1.57 kGy / s.

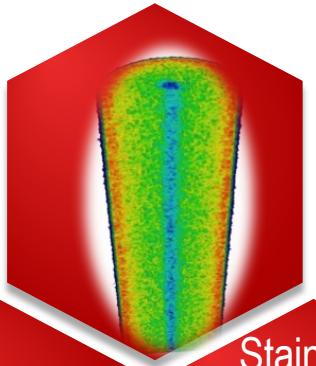
The total accumulated fluence for fast neutrons during the experiment was 1.76×10^{24} m^{-2}, the temperature of the samples was 185-206 °C. Based on the test results, the effect of reactor irradiation on the parameters of optical fibers and fiber-optic temperature sensors was evaluated, which will allow to select the optimal material for use in the ITER reactor.

An important result of this study was the identification of irreversible destruction of polyimide and acrylate coatings in the process of reactor irradiation due to the combined effect of high neutron fluence, vacuum, and increased irradiation temperature.

Main previous publications on related topics



CPS-systems, plasma-facing materials irradiation



Optical fibers testing in fission reactor conditions

Fusion materials irradiation in reactors

Beryllides irradiation and corrosion testing

Stainless steel, tungsten irradiation in fission readtor

TDS and other PIE studies after irradiation

In-situ reactor experiments

Permeability experiments under reactor irradiation conditions

Fusion materials irradiation in DC-60 cyclotron

Thank you for your attention