ОДЦ УГР

ПРЕДПРИЯТИЕ ГОСКОРПОРАЦИИ «РОСАТОМ»

Joint Stock Company “Pilot & Demonstration Center for Decommissioning of Uranium-Graphite Nuclear Reactors”

Data Obtaining Approaches for Nuclear Graphite Characteristics to Decommission Uranium-Graphite Reactors

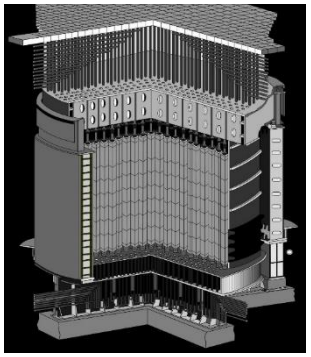
“PDC UGR”, JSC

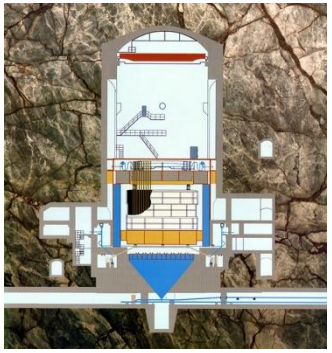
Alexandr Pavlyuk

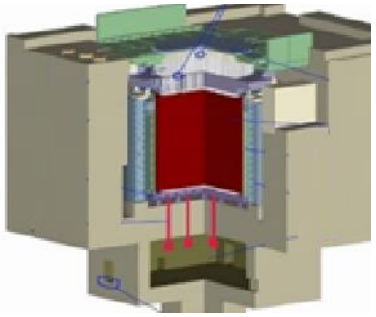
info@dnrc.ru

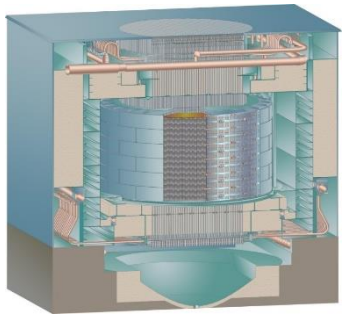
RUSSIAN URANIUM-GRAPHITE NUCLEAR REACTORS TYPES AND THEIR PICULIARITIES AFFECTING DECOMMISSIONING APROACH

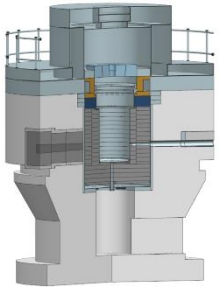


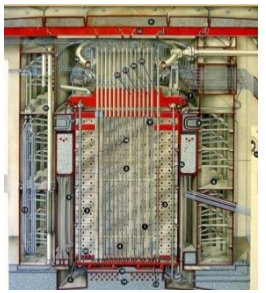






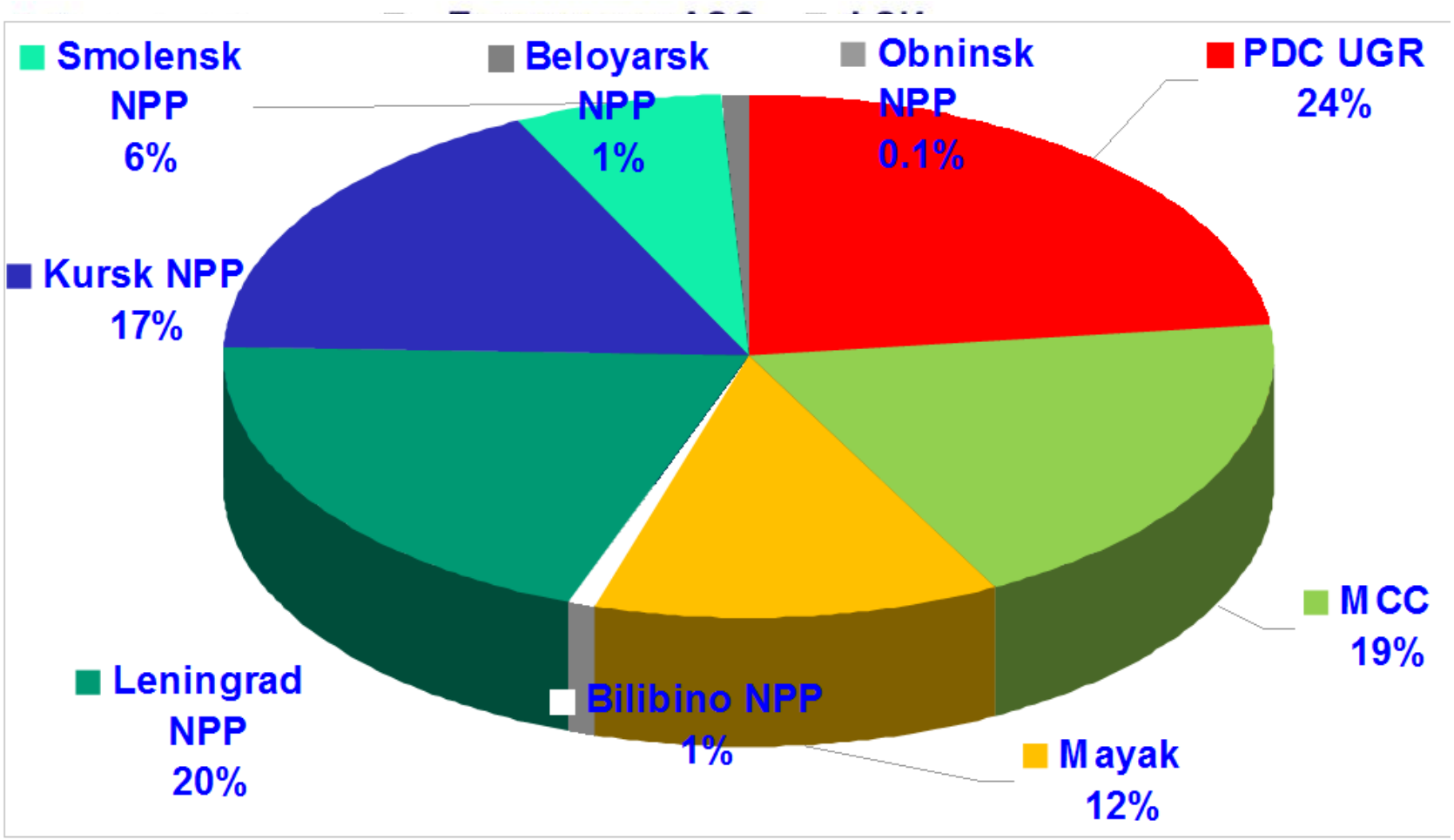






Russian UGRs of different types (AM, AMB, RBMK, ЭGP, PUGR, research TVR) vary in construction and location of reactor (deep-frozen soil, mountains, densely populated areas and etc.) and the core (above or under ground).

IRRADIATED GRAPHITE AMOUNT ACCUMULATED ON RUSSIAN SITES



Irradiated graphite amount accumulated on Russian sites, in %

Dorofeev A N , Komarov E A , Zakharova E V , Volkova A G , Linge I I , Utkin S S , Ivanov A Yu , Pavliuk A O , Kotlyarevskij S G On Reactor Graphite Disposal *Radioactive Waste*, 2019, no 2 (7), pp 18—30 DOI: 10 25283/2587-9707-2019-2-18-30 (In Russian)

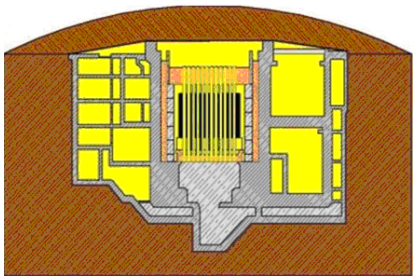
GRAPHITE WASTE MANAGEMENT OF RUSSIAN UGRs

Graphite waste management options

In situ entombment – PUGRs of PDC UGR and MCC

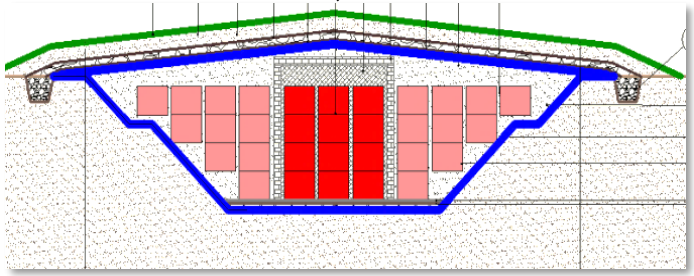
Dismantling - РВМК, АМВ, АМ, probably PUGR of Mayak

Special RW



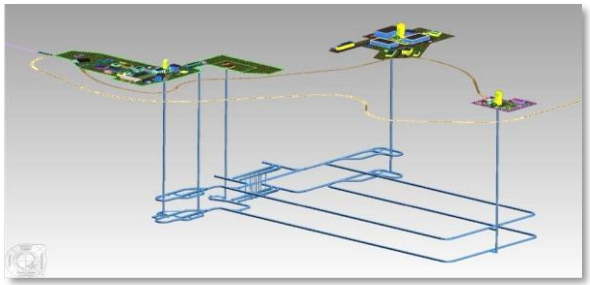
Special RW temporal storages

RW of class 3 and 4



RW surface storages

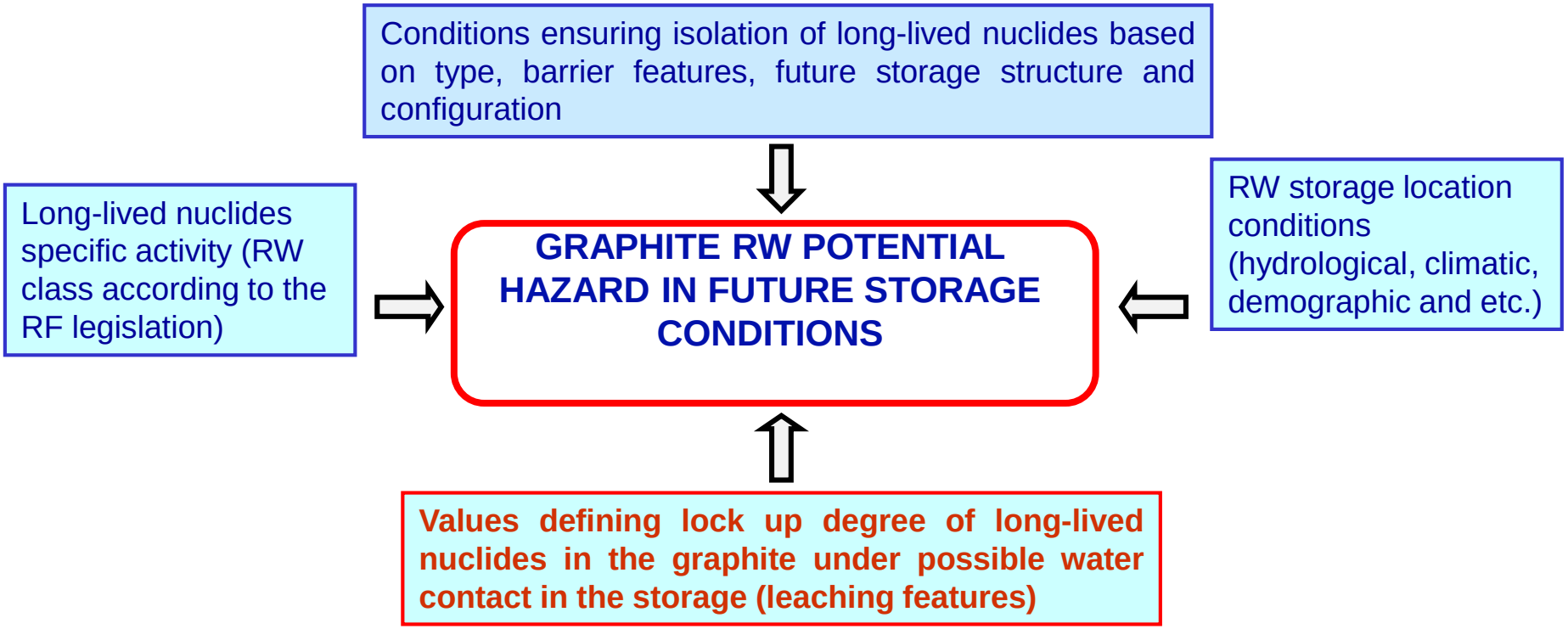
RW of class 1 and 2



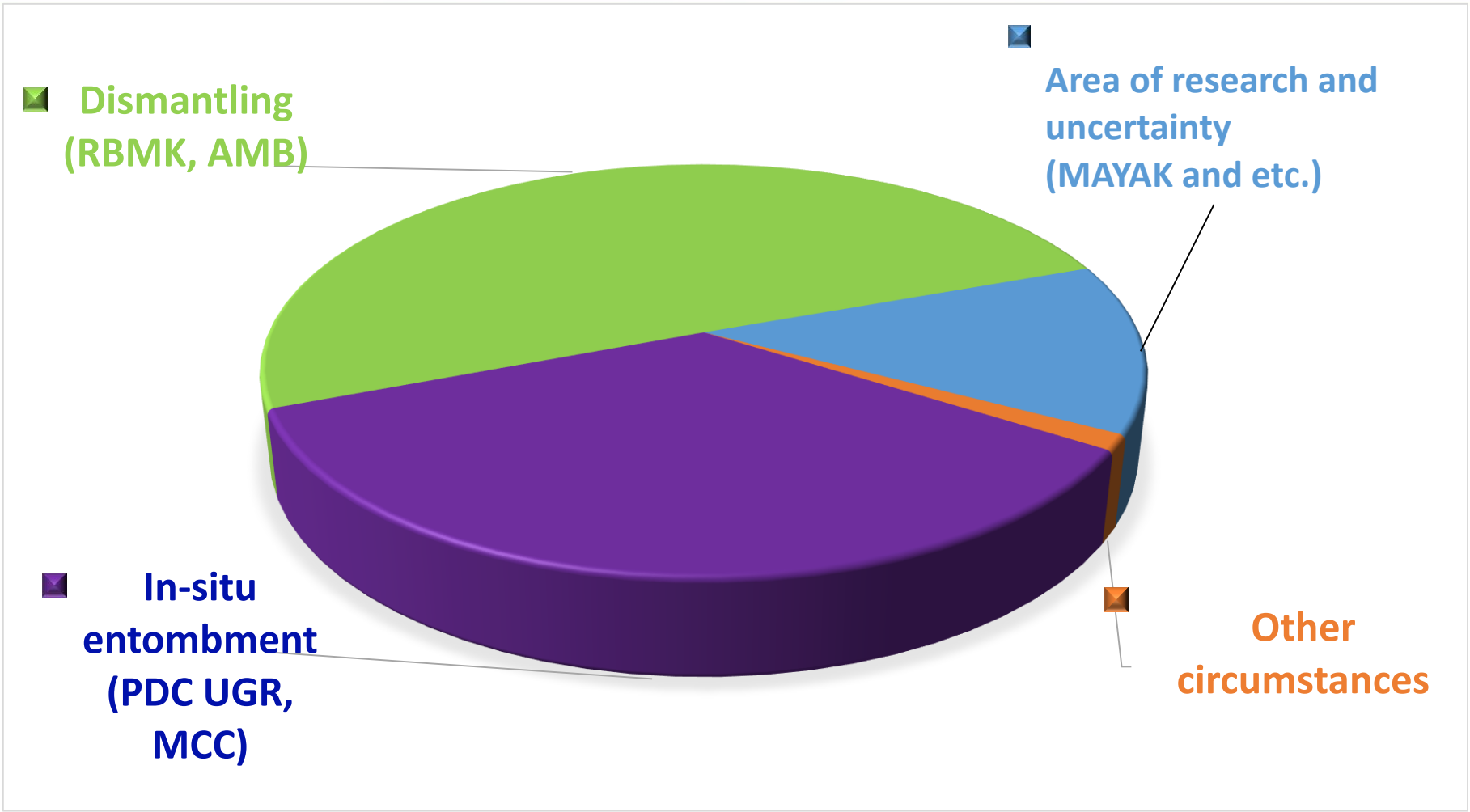
RW deep storages

IRRADIATED GRAPHITE CHARACTERISTICS AND RW STORAGE CONDITIONS, WHICH DEFINE GRAPHITE RW POTENTIAL HAZARD

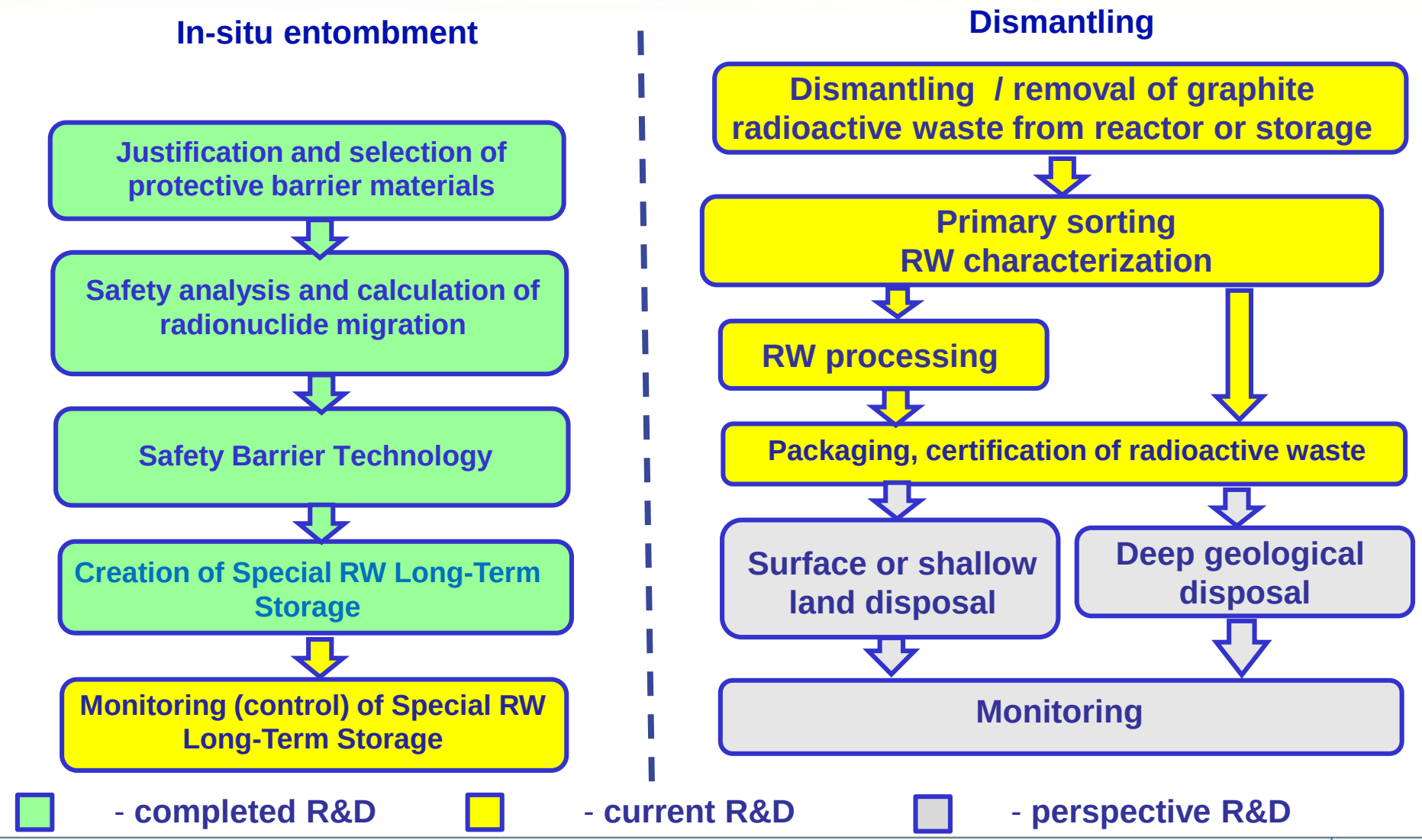
To justify any of the graphite RW management options it is necessary to assess the degree of their impact on the environment and public related to possible migration of long-lived nuclides contained in the graphite RW in conditions of the storage under design (potential hazard assessment)



THE PROPORTION OF IRRADIATED GRAPHITE FOR VARIOUS TREATMENT SCENARIOS



R&Ds FOR VARIOUS OPTIONS FOR IRRADIATED GRAPHITE RW MANAGEMENT



BASIC PARAMETERS DEFINING SAFETY OF DIFFERENT GRAPHITE RW MANAGEMENT OPTIONS

IN-SITU ENTOMBMENT

- Lock-up degree of radionuclides in graphite RW (formation of transportable radionuclide forms);
- Radionuclide activity (C-14 and Cl-36 long-lived radionuclides concentration, presence of nuclear fuel elements).

DISMANTLING

- radionuclides activity (Cs-137 and Co-60 dose forming radionuclides concentration; presence of nuclear fuel elements);
- graphite condition (cracks),
- physical and mechanical features;
- Wigner energy.

GRAPHITE RW ENTOMBMENT

- Lock-up degree of radionuclides in graphite RW (formation of transportable radionuclide forms);
- Radionuclide activity (C-14 and Cl-36 long-lived radionuclides concentration, presence of nuclear fuel elements).
- Wigner energy.

GRAPHITE RW STORAGE

- radionuclides activity (Cs-137 and Co-60 dose forming radionuclides concentration; presence of nuclear fuel elements);
- Lock-up degree of radionuclides in graphite RW;
- Wigner energy.

EI-2 DECOMMISSIONING PROJECT IMPLEMENTATION

Final shutdown, bringing the facility to nuclear safe state 1990-1993	CERS, R&Ds, development of a Decommissioning Concept 1993-2011	Project development, obtaining a license 2011-2012	Project implementation 2012-2015



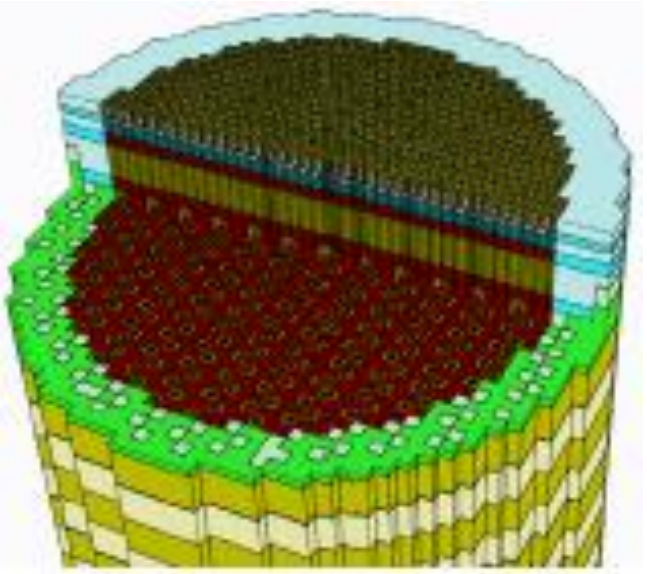
EI-2 Layout before Decommissioning project implementation
2012



Special RW Long-Term Storage
2015

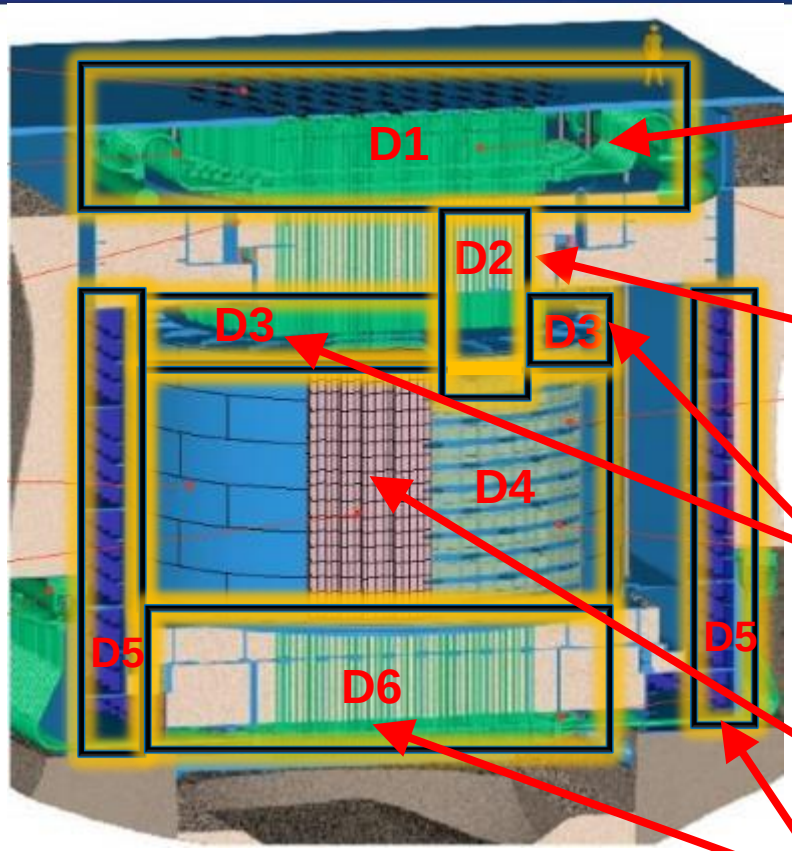
DISMANTLING

"Dismantling“ is conceptually defined as decommissioning method for uranium-graphite power reactors (UGR).



Objective of full research cycle is to test dismantling technology for UGR graphite stack during real decommissioning of a facility (shutdown UGR).

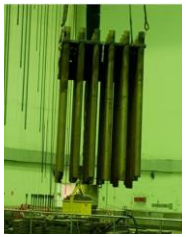
STAGES OF EXERCISING THE TECHNOLOGIES FOR DISMANTLING THE STRUCTURE OF THE PUGR







2017-2018

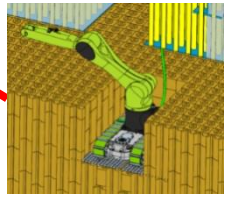





2018-2019




2019-2020



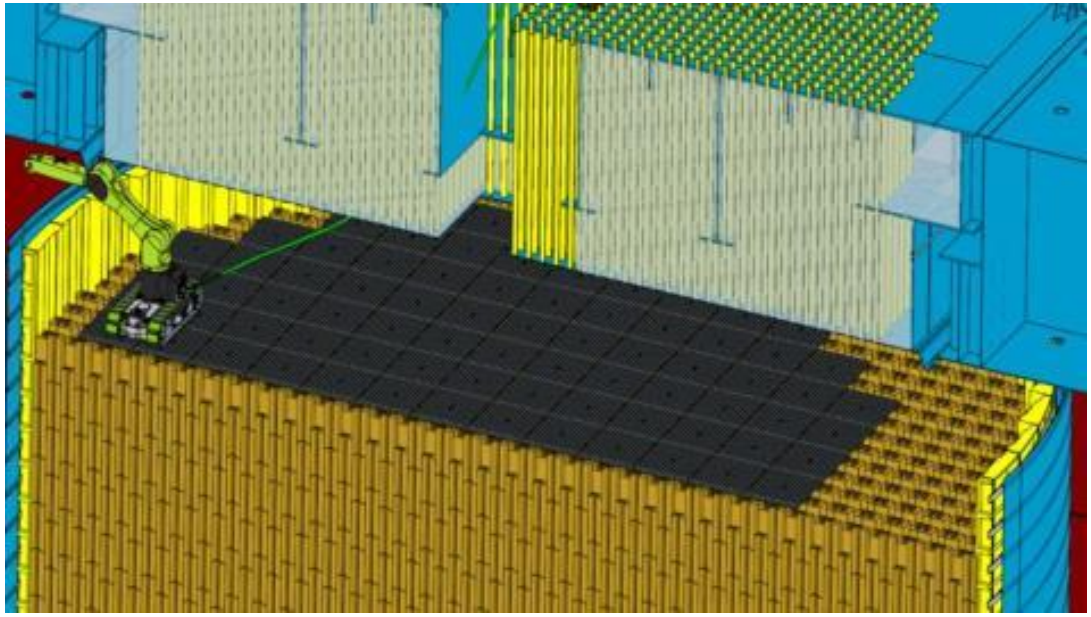

2020

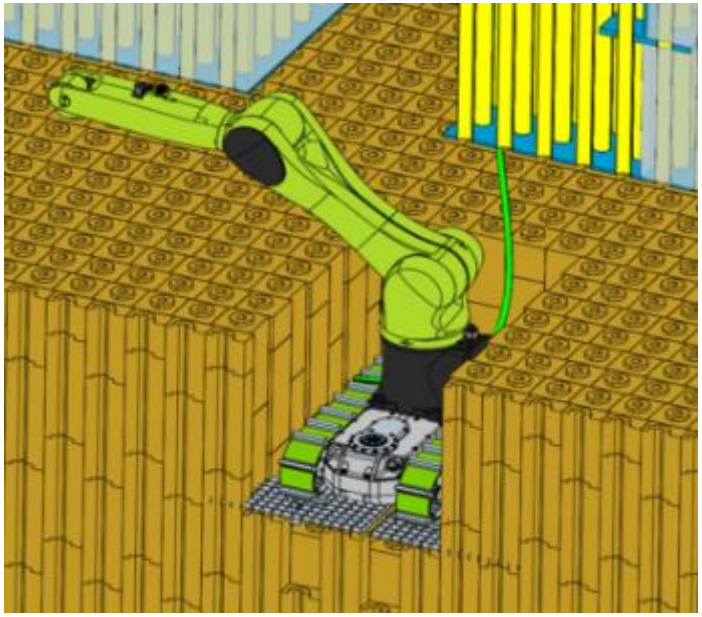
not earlier 2021

D1- upper communications;
D2- metal structures in the opening and upper layers of graphite blocks;
D3- roof and nitrogen collector;

D4 - graphite stack, bandages, casing;
D-5 - metal tanks of side protection (“L” scheme);
D-6 - lower metal structures (supporting “heel”, etc.).

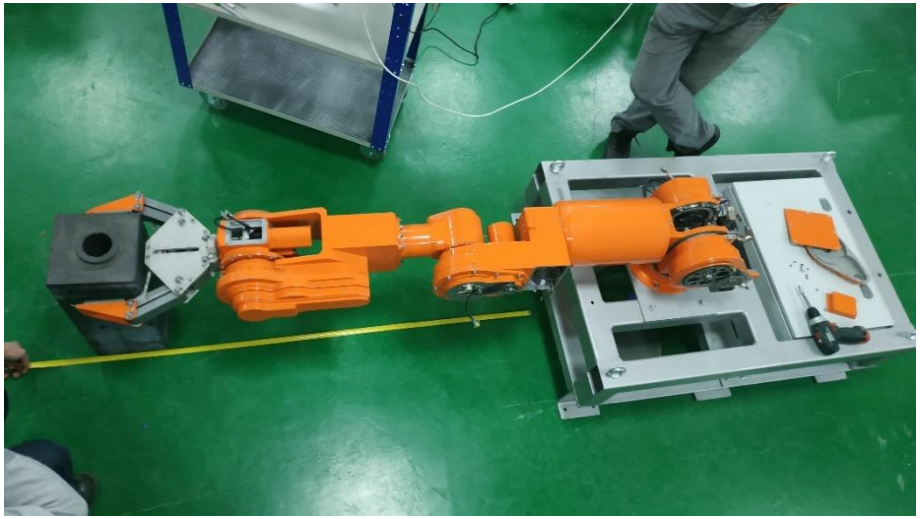
“PDC UGR” EXERCISING THE GRAPHITE STACK DISMANTLING TECHNOLOGY IN 2017-2018





Implementation of the dismantling technology using a robotic manipulator

TESTING BLOCK CAPTURE TECHNOLOGY USING THE ROBOTIC MANIPULATOR







PERFORMING OPERATIONS ON MOCK-UP MODELS

DEVELOPMENT OF TECHNOLOGICAL OPERATIONS USING A DEVICE TO REMOVE GRAPHITE BLOCKS.







Mock-up testing of capturing and lifting of graphite blocks

ADE-5 PUGR UPPER METAL STRUCTURES DISMANTLING TO EXERCISE RETRIEVAL



Initial condition



Partial demolition of floor plates



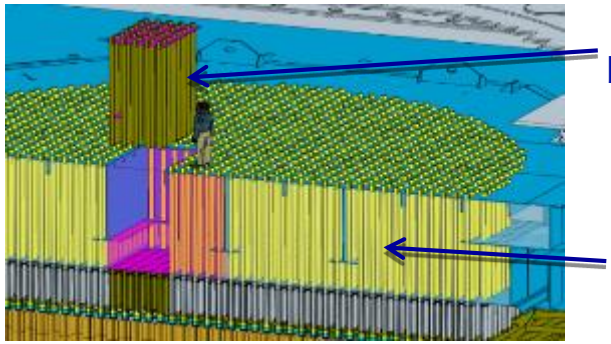


Partial demolition of tubes under floor plates

GRAPHITE STACK DISMANTLING TECHNOLOGY DEVELOPMENT

STAGES OF PRACTICAL WORKS ON PERFORMING AN OPENING IN UPPER BIOLOGICAL PROTECTION ("E" SCHEME PUGR ADE-5)

Sketch design of the opening



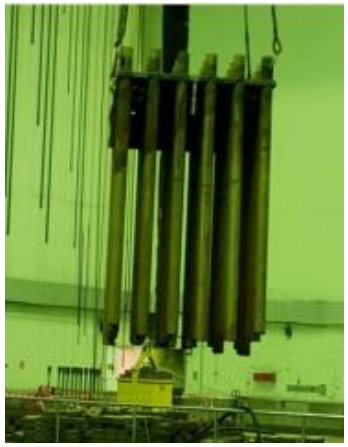
Extracting bunch of central pipes of "E" scheme

"E" scheme











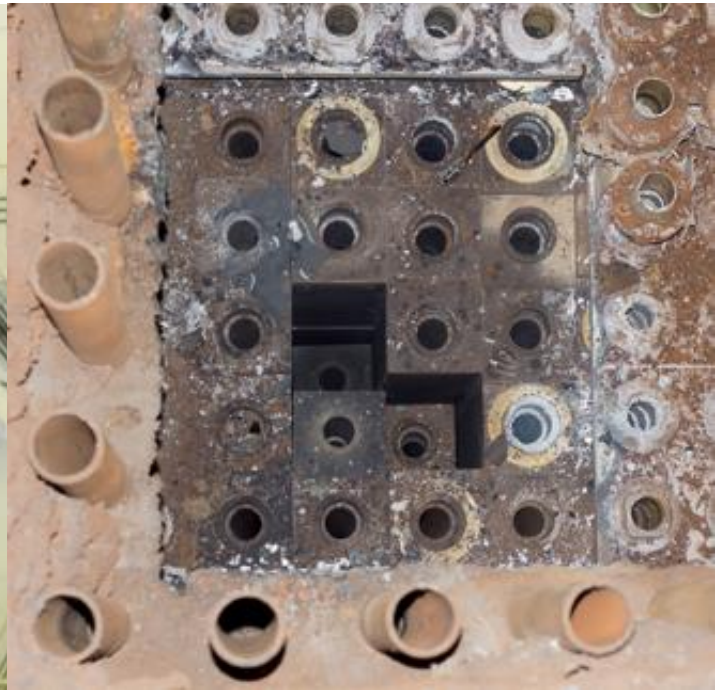


PERFORMANCE OF EXPERIMENTAL WORKS AT ADE-5 PUGR



Lifting a block through the opening





View of the opening after extraction of blocks

PERFORMANCE OF EXPERIMENTAL WORKS AT ADE-5 PUGR

PERFORMANCE OF EXPERIMENTAL WORKS ON RETRIEVAL OF GRAPHITE BLOCKS FROM GRAPHITE STACK

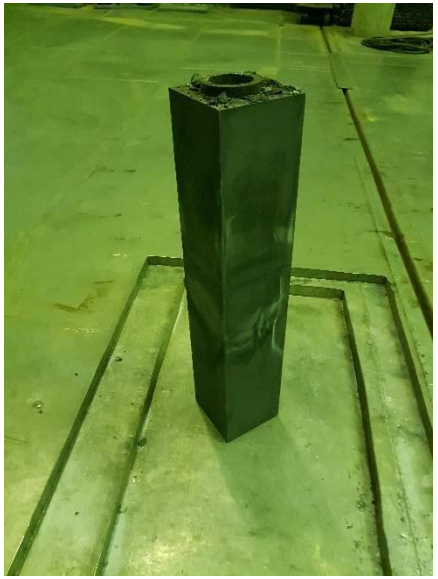
Retrieval of graphite blocks from the first and second levels of the graphite stack. The blocks were extracted through the opening one by one.



Lifting a block



View of the opening after extraction of a block



Extracted blocks

PERFORMANCE OF EXPERIMENTAL WORKS AT ADE-5 PUGR

PERFORMANCE OF EXPERIMENTAL WORKS ON EXTRACTION OF GRAPHITE BLOCKS FROM GRAPHITE STACK

Capturing, lifting the blocks were done without difficulty and remarks. The blocks were removed from the stack and installed in a container in Central Hall room for inspection and determination of radiation characteristics. Full inspection (from all sides including opening) of the removed graphite blocks showed no obvious signs of deformation and form changes.



Removed graphite blocks in the container



Determination of radiation characteristics of the removed graphite blocks

Testing of individual elements of electronic equipment under the influence of ionizing radiation in the metal structures of UGR





Test bench for testing individual elements of equipment under the influence of ionizing radiation in the metal structures of UGR

METHODS STANDARDIZATION

ОДЦ УГР
ПРЕДПРИЯТИЕ ГОСКОРПОРАЦИИ «РОСАТОМ»

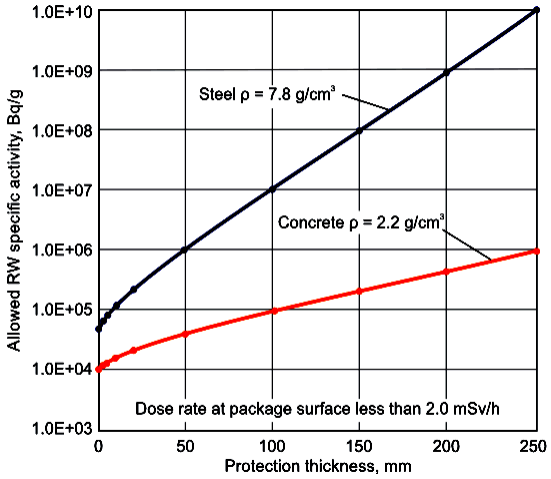
ВНИИНМ
имени А.А. Бочвара

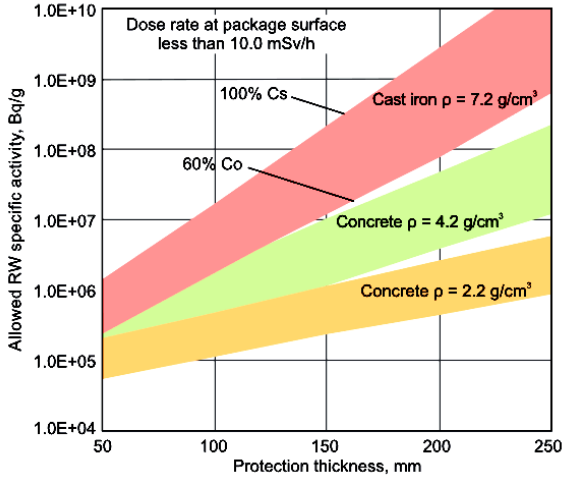
Standardization and Improvement of Graphite RW Characterization Methods for Graphite Blocks Sorting, Classification and Packaging

TASKS FOR IMPROVING METHODS FOR STUDYING GRAPHITE

Container certification	- Determination of container parameters taking into account the radiation characteristics of graphite radioactive waste; - Development of recommendations on the container accounting and control system, as well as the certification process.
Methods for graphite waste sorting and characterization (blocks, sleeves, etc.)	- Analysis of the possibility of determining nuclide vectors; - Development and testing of non-sampling methods for characterizing graphite blocks; - Development and testing of methods for assessing the content of fuel fragments' spillages in the extracted "incident" graphite blocks; - Development and testing of methods for sorting graphite elements during or after extraction; - Improving methods and equipment for characterizing blocks before extraction.
Analysis, improvement and certification of measure methods of graphite samples activity	- Certification of methods and development of standards; - Development of methods (calculated and / or experimental) for assessing activity and sample preparation to minimize the “nugget" effect; - Development of express methods for the determination of difficult to measure nuclides; - Production of standard specimen of irradiated graphite.

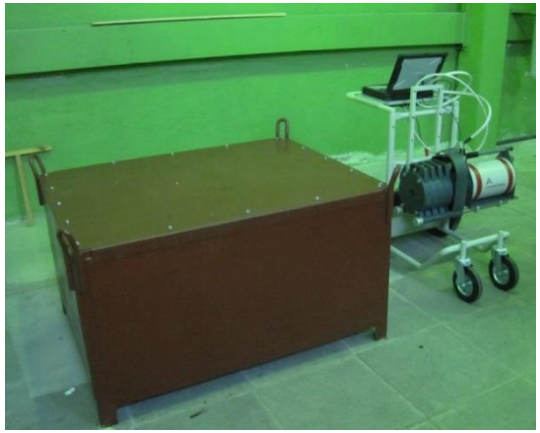
SORTING AND CHARACTERIZATION OF GRAPHITE BY GAMMA RADIATION INTENSITY







NZK-150-1,5P



Collection, conditioning, storage

- **Limit of annual effective execss exposure dose for the personnel**

  20 mSv/year
  (50 mSv/year)

NRB-99/2009

Transportation for storage/disposal

- **Limit of maximum dose rate value**

  Up to 10 mSv/h at package surface

  0.1 mSv/h at the distance of 2 m from the vehicle surface

SanPiN 2.6.1.1281-03
NP-053-16

Disposal

- **Limit of absorbed dose rate at the package surface**

  2 class undefined
  3 class – 10 mGy/h
  4 class – 2 mGy/h

NP-093-13

Determination of thickness of radiation shielded cask body for storage

Radchenko M.V., Kormilitsyna L. A., Matyunin J. I., Mogulyan V. G. Multi-purpose Packages for Radioactive Waste // Radioactive waste. — 2017. — No 1. — pp. 52—59.

SORTING AND CHARACTERIZATION OF GRAPHITE BLOCKS BY GAMMA RADIATION INTENSITY



	0,4	0,6	1,1	
0,3	0,5	0,9	1,8	1,5
0,2	0,3	2,1	**10**	**3,04**
0,5	0,6	0,9	1,4	0,63
	0,3	0,4	0,7	

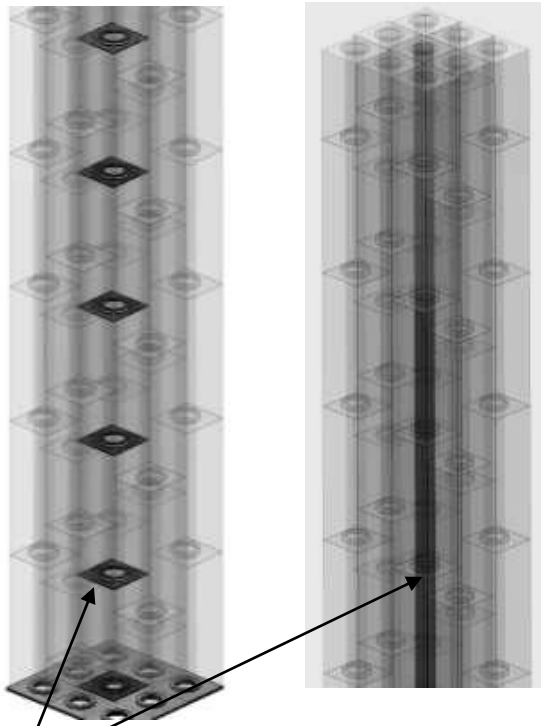
Packing wall

Cartogram of gamma radiation intensity from MK-1.36 packaging with graphite blocks (distance ~ 10 cm),rel.

According to the requirements for NZK containers, the maximum equivalent dose rate on the surface should not exceed 2.0 mSv/h and at a distance of 1 m - not more than 0.1 mSv/h. The problem is solved by placing “highly active” blocks in the central part of the package or in other packaging.

Characteristics of graphite blocks

Irradiated reactor graphite may contain fuel fragments that have been deposited into the pile as a result of incidents, followed by the spread of fission products and actinides throughout the pile.



Graphite pile areas with fuel spills



Windscale Graphite Pile Cells with Horizontal Fuel Spills (UK)

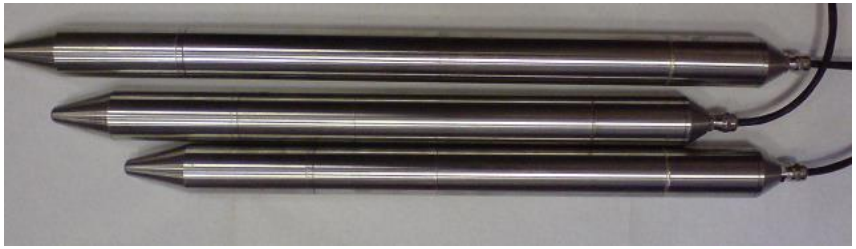
DETERMINATION OF ISOTOPE COMPOSITION OF IRRADIATED GRAPHITE SAMPLES

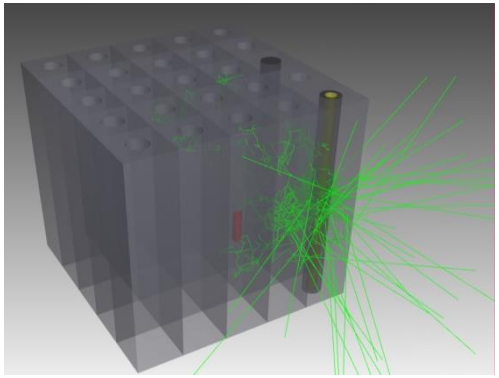
Radionuclides	Technique
Hard-to-measure H-3, C-14, Cl-36	- Radiochemistry & Liquid scintillation spectroscopy ;
Nuclear dangerous U-235, Pu-239 and etc.	- Neutron Activation Analysis - Active and passive neutron control
Gamma-emitting: ^{60}Co, ^{133}Ba, ^{134,137}Cs, ^{152,154,155}Eu	- γ- spectrometry with HPGe-detector.
Alpha-emitting: ^{239,240}Pu, ^{241,243}Am, ^{244}Cm	- γ/X- spectrometry with LEGe-detectors; - Radiochemistry and α-spectrometry with Si(Au)-detectors.

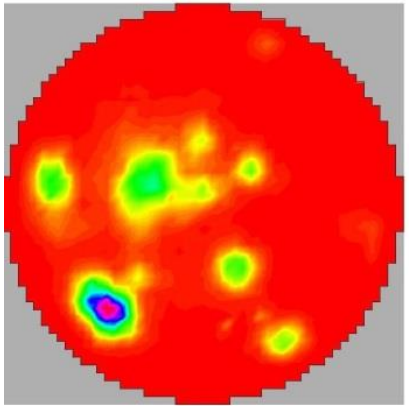
Application Of Passive Control Method For Fission Materials In Graphite Stack

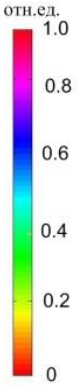
Software & analytical system during testing at the UGR

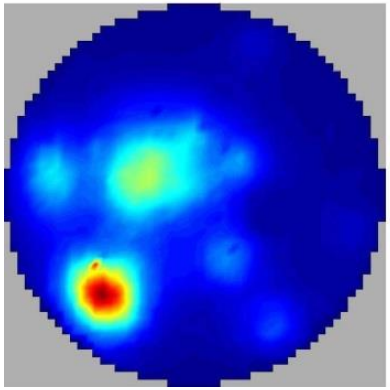


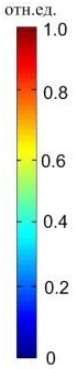


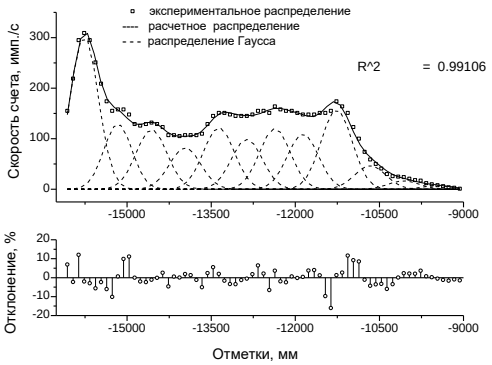








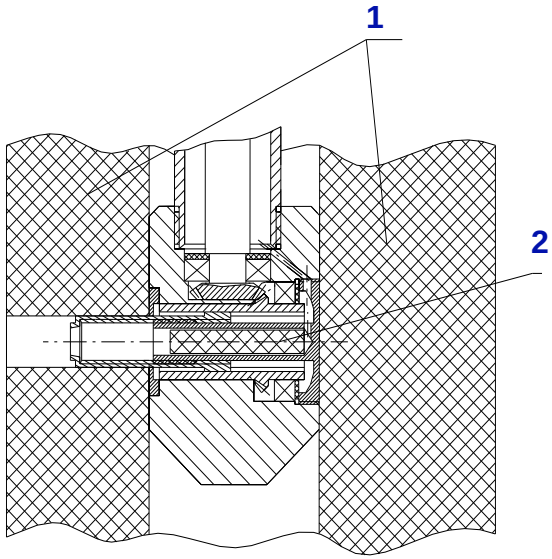


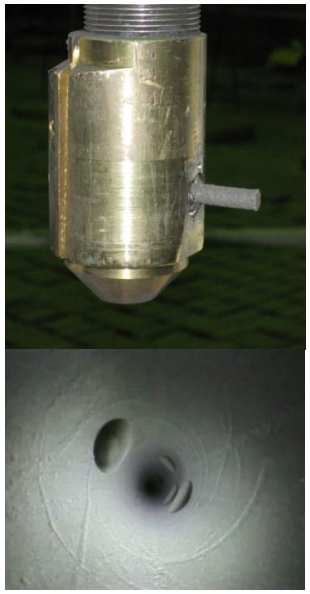


Distributions of gamma and neutron field in graphite stack and an analysis algorithm for determining the parameters determination of the amount of fuel fragments' spillages (Patent RF 2 649 656)

DETERMINATION OF THE RADIONUCLIDE CONTENT OF SAMPLES OF IRRADIATED GRAPHITE

To analyze graphite stack there were sampling devices elaborated which allowed obtaining samples from I-1, EI-2, ADE-3, ADE-4, ADE-5, AI, AV-1, AV-2, AV-3, unit №2 of Leningrad NPP reactor and rings of Kursk NPP.





1 – Graphite block
2 – Graphite sample
3 – Holes in graphite block after horizontal sampling

Device for horizontal sampling of graphite from UGR without graphite blocks removal (Patent RU 2378510)



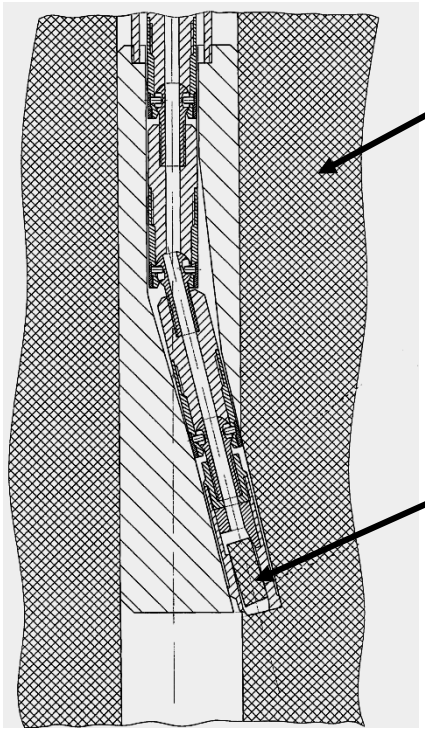


The selected block graphite of the RBMK-1000 reactor of the LNPP power unit No. 2: block fragment, graphite crumb

SAMPLING EQUIPMENT FOR REFLECTOR GRAPHITE BLOCKS



Cell view after angled drill out of graphite sample from EI-2



Device for angled drill out of samples from graphite stack and reflector without graphite block retrieval (RF Patent 2454536)

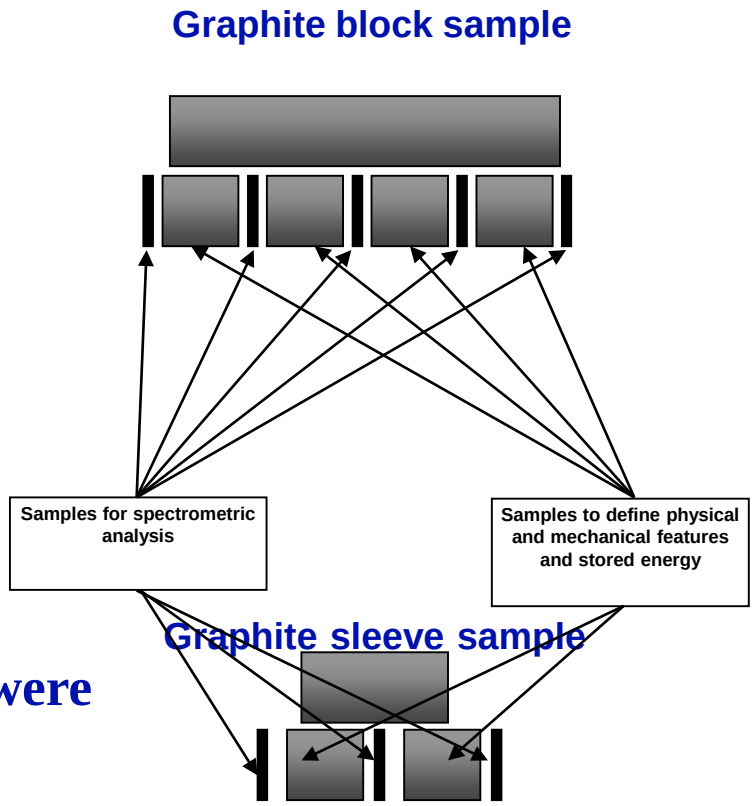
GRAPHITE SAMPLING DATA AND LAYOUT

Number of samples from "PDC UGR" graphite stack elements (pcs.).

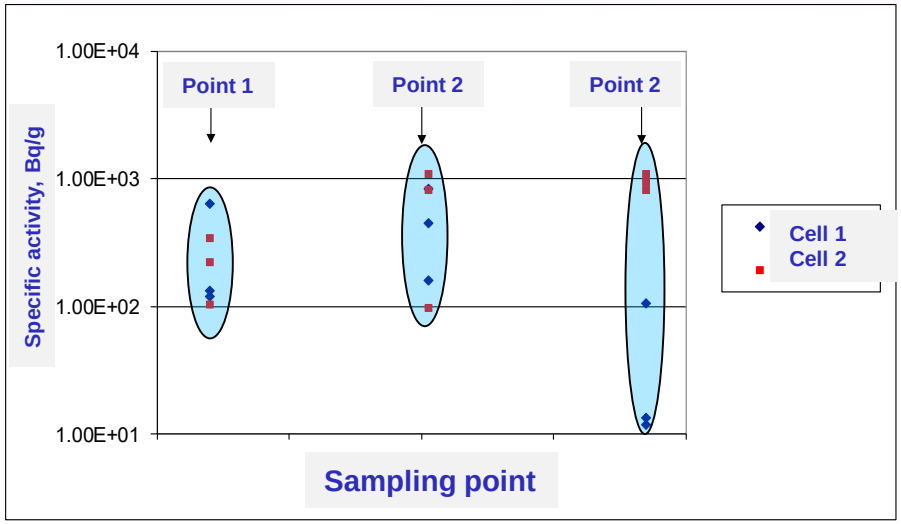
Elements	*Reactor*				
	I-1	*EI-2*	*ADE-3*	*ADE-4*	*ADE*
Graphite blocks	147	73	100	20	30
Graphite sleeves	150	150	91	12	12
Reflector elements	18	16	-	-	-

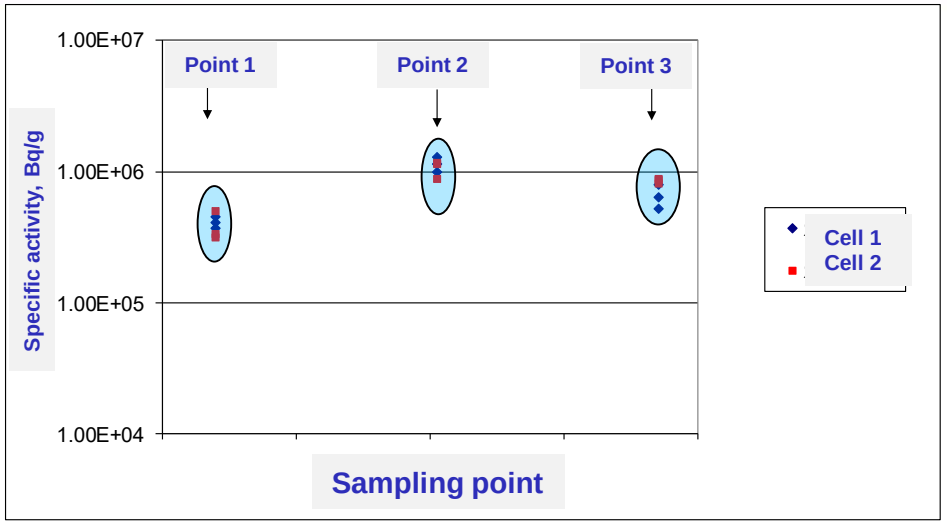
More than 2000 irradiated graphite samples were done and used in research

Samples cutting

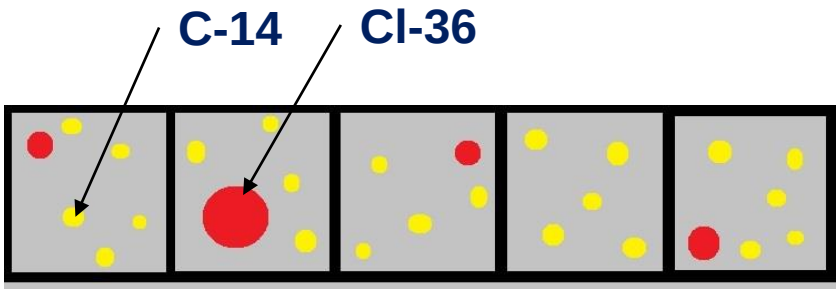


PROBLEMS OF MEASURING THE ACTIVITY OF DIFFICULT-TO -MEASURE NUCLIDES (CI-36 AND OTHERS). "NUGGET" EFFECT





Comparison of the non-uniform distribution of Cl-36 relative to C-14 in graphite samples taken from different points

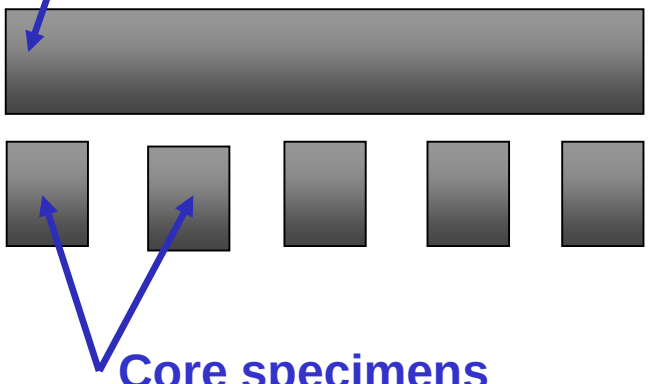


Graphite Samples

Significant non-uniformity of the distribution of the Cl-36 radionuclide in graphite leads to non-representative sampling and errors in estimates (the "Nugget" effect, etc.)

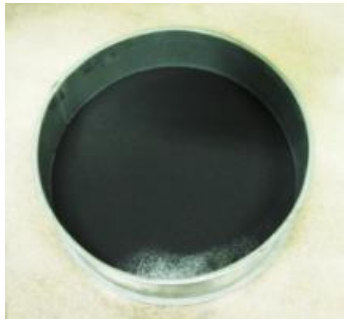
OPPORTUNITIES FOR REDUCING INFLUENCE "NUGGET" EFFECT IN GRAPHITE

Through-the-thickness specimen of block



Core specimens

Drilling and / or grinding



Ground specimen

Estimated by calculation

By calculation

$$\frac{s^2}{a_L^2} = \frac{1-a_L}{a_L}\left(\frac{1}{Masse_e} - \frac{1}{Masse_L}\right)\tilde{K}$$

The formula of Pierre Ji (EDF, France) or others.

Assessment of fundamental uncertainty related to sampling by calculation

RESULTS OF INTERNATIONAL INTERLABORATORY COMPARISONS ON DETERMINATION OF RADIONUCLIDES IN GRAPHITE SAMPLES

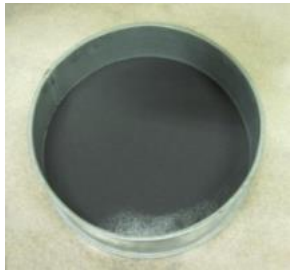
Nuclide	Laboratory 1	Laboratory 2	Laboratory 3
$^{239}Pu+^{240}Pu$	9,0±1,0	1,55±0,47	1,53±0,38
^{238}Pu	7,3±1,0	0,42±0,17	0,27±0,07
^{241}Am		2,1±0,6	3,1±0,4
^{3}H	$(9,2±1,0)\cdot10^{4}$	$(4,5±0,5)\cdot10^{4}$	$(5,6±1,1)\cdot10^{4}$
^{14}C	3400±400	6700±700	7500±1500
^{36}Cl	19±2	6,9±2,1	-

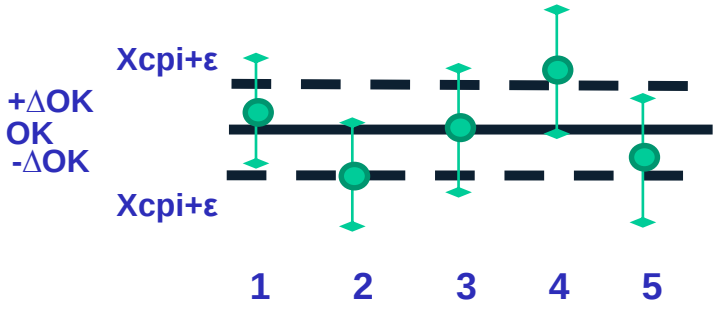
Graphite sample analysis results (Bq/g) obtained in laboratories in France, Germany and the Russian Federation. In each laboratory, measurements were made of several samples of graphite, the table shows the average values

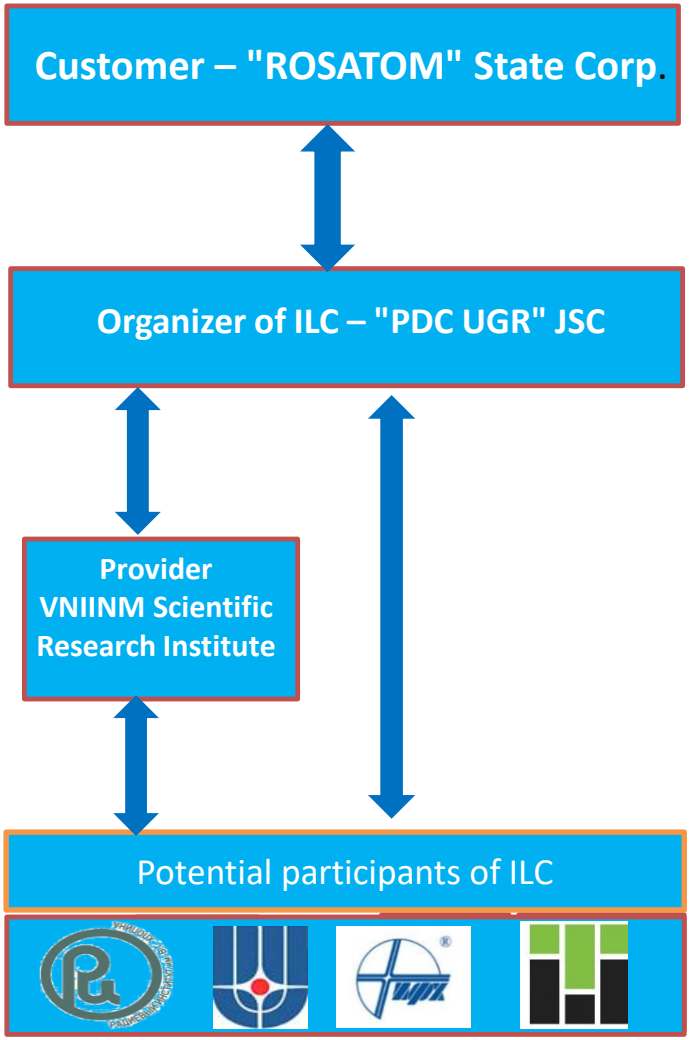
ANALYSIS & IMPROVEMENT OF THE ANALYTICAL BASIS OF MEASUREMENT OF HARD-TO-DETERMINE NUCLIDES IN THE IRRADIATED GRAPHITE (2017-2018)

Purpose of inter-laboratory comparisons (ILC):

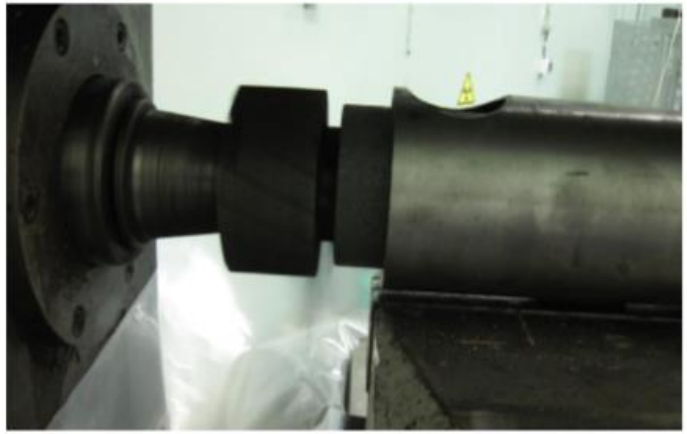
- assessment of the methodological base state in the Russian Federation for performing measurements of hard-to-determine nuclides in graphite radioactive waste during decommissioning of uranium-graphite reactor (RBMK, PUGR and others).
- Identification of problems in laboratories and implementation of corrective actions to ensure the uniformity of measurements;
- preparation and certification of batches of graphite in powder form with the activity of hard-to-determine nuclides (^{14}C, ^{36}Cl and others) uniformly distributed over the volume for their further application in improving the methodological base and performing experimental research.



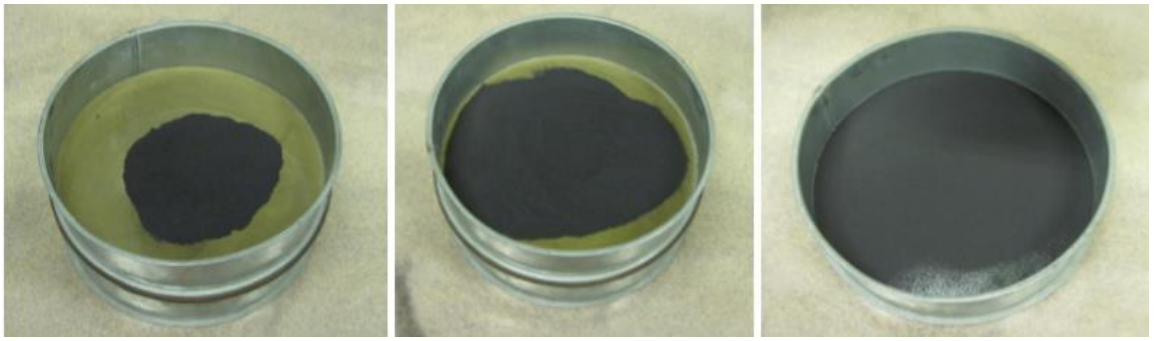




PREPARATION OF GRAPHITE BATCHES

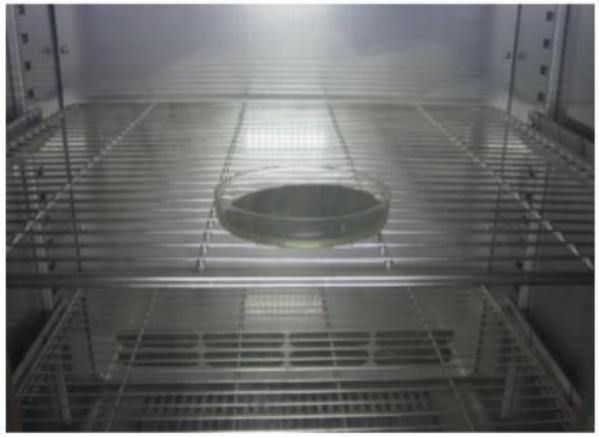


Rough grinding



Process of sifting graphite weighed portion (left and center); Graphite weighed portion after sifting (right)





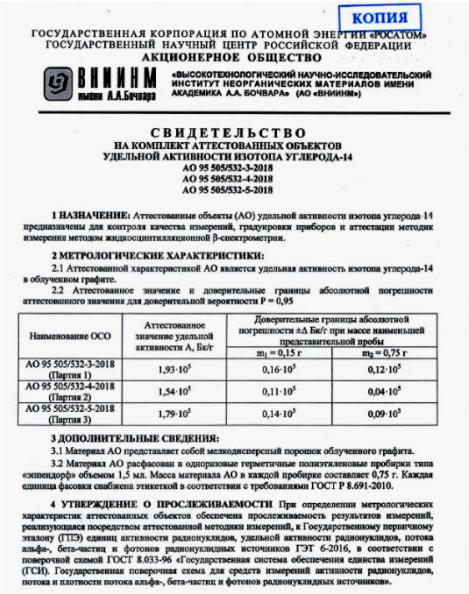
Drying



Samples after mixing

ANALYSIS AND IMPROVEMENT OF ANALYTICAL BASIS FOR DIFFICULT TO MEASURE NUCLIDES IN IRRADIATED GRAPHITE

- an assessment of the methodological support of organizations performing measurements of difficult-to-determine radionuclides in irradiated graphite was performed;
- a comparative analysis of the results was carried out, the most effective measurement methods were determined;
- recommendations were developed to improve the quality of measurements and standardization;
- samples of irradiated graphite were certified for the development of methods for characterizing graphite and performing experimental work



КОПИЯ

ГОСУДАРСТВЕННАЯ КОРПОРАЦИЯ ПО АТОМНОЙ ЭНЕРГИИ «РОСАТОМ»
ГОСУДАРСТВЕННЫЙ НАУЧНЫЙ ЦЕНТР РОССИЙСКОЙ ФЕДЕРАЦИИ
АКЦИОНЕРНОЕ ОБЩЕСТВО
ВНИИНМ имени А.А.Бочвара
«ВЫСОКОТЕХНОЛОГИЧЕСКИЙ НАУЧНО-ИССЛЕДОВАТЕЛЬСКИЙ ИНСТИТУТ НЕОРГАНИЧЕСКИХ МАТЕРИАЛОВ ИМЕНИ АКАДЕМИКА А.А. БОЧВАРА» (АО «ВНИИНМ»)

СВИДЕТЕЛЬСТВО
НА КОМПЛЕКТ АТТЕСТОВАННЫХ ОБЪЕКТОВ
УДЕЛЬНОЙ АКТИВНОСТИ ИЗОТОПА УГЛЕРОДА-14
АО 95 505/532-3-2018
АО 95 505/532-4-2018
АО 95 505/532-5-2018

1 НАЗНАЧЕНИЕ: Аттестованные объекты (АО) удельной активности изотопа углерода-14 предназначены для контроля качества измерений, градуировки приборов и аттестации методик измерения методом жидкостной сцинтилляционной β-спектрометрии.

2 МЕТРОЛОГИЧЕСКИЕ ХАРАКТЕРИСТИКИ:

2.1 Аттестованной характеристикой АО является удельная активность изотопа углерода-14 в облученном графите.

2.2 Аттестованное значение и доверительные границы абсолютной погрешности аттестованного значения для доверительной вероятности P = 0,95

Наименование ОСО	Аттестованное значение удельной активности A, Бк/г	Доверительные границы абсолютной погрешности ±Δ Бк/г при массе навески представительной пробы	
		m_1 = 0,15 г	m_2 = 0,75 г
АО 95 505/532-3-2018 (Партия 1)	$1{,}93 \cdot 10^5$	$0{,}16 \cdot 10^5$	$0{,}12 \cdot 10^5$
АО 95 505/532-4-2018 (Партия 2)	$1{,}54 \cdot 10^5$	$0{,}11 \cdot 10^5$	$0{,}04 \cdot 10^5$
АО 95 505/532-5-2018 (Партия 3)	$1{,}79 \cdot 10^2$	$0{,}14 \cdot 10^2$	$0{,}09 \cdot 10^2$

3 ДОПОЛНИТЕЛЬНЫЕ СВЕДЕНИЯ:

3.1 Материал АО представляет собой мелкодисперсный порошок облученного графита.

3.2 Материал АО расфасован в одноразовые герметичные полиэтиленовые пробирки типа «эппендорф» объемом 1,5 мл. Масса материала АО в каждой пробирке составляет 0,75 г. Каждая единица фасовки снабжена этикеткой в соответствии с требованиями ГОСТ Р 8.691-2010.

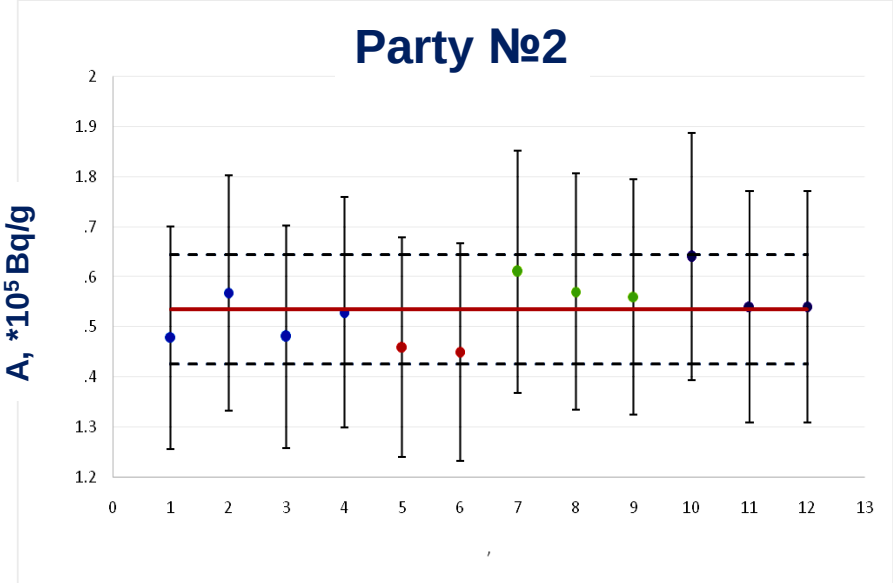
4 УТВЕРЖДЕНИЕ О ПРОСЛЕЖИВАЕМОСТИ При определении метрологических характеристик аттестованных объектов обеспечена прослеживаемость результатов измерений, реализующаяся посредством аттестованной методики измерений, к Государственному первичному эталону (ГПЭ) единиц активности радионуклидов, удельной активности радионуклидов, потока альфа-, бета-частиц и фотонов радионуклидных источников ГЭТ 6-2016, в соответствии с поверочной схемой ГОСТ 8.033-96 «Государственная система обеспечения единства измерений (ГСИ). Государственная поверочная схема для средств измерений активности радионуклидов, потока и плотности потока альфа-, бета-частиц и фотонов радионуклидных источников».

Radionuclides	IPhE RAS	NRC KI	RI	TPU
Beta-emitters: $^{3}H, ^{14}C, ^{36}Cl, ^{63}Ni, ^{90}Sr$	$^{14}C, ^{36}Cl, ^{90}Sr$	$^{14}C, ^{3}H, ^{90}Sr$	^{14}C	^{14}C
Alpha emitters: $^{239,240}Pu, ^{241,243}Am, ^{244}Cm, ^{235,238}U$	$^{239,240}Pu, ^{241,243}Am, ^{244}Cm, ^{235,238}U$	$^{238}, ^{239+240}, ^{241}Pu, ^{241,243}Am, ^{235,238}U$	$^{241,243}Am$	-

RESULTS OF DETERMINING THE CONTENT OF C-14 IN THE COURSE OF INTERLABORATORY COMPARISONS

The results of the calculation of the evaluation criteria (K1 and Z-index) of the quality of the laboratory when determining C-14 in Party No. 2

№	Xi	Δ	K1	Z
1	1,47912	0,221868	0,254115	0,951561
2	1,5678	0,23517	0,137347	0,545148
3	1,48035	0,222053	0,248365	0,930802
4	1,5288	0,22932	0,029217	0,11308
5	1,46	0,219	0,344749	1,274262
6	1,45	0,2175	0,393103	1,443038
7	1,61	0,2415	0,308489	1,257384
8	1,57	0,2355	0,146497	0,582278
9	1,56	0,234	0,104701	0,413502
10	1,64	0,246	0,424797	1,763713
11	1,54	0,231	0,019481	0,075949
12	1,54	0,231	0,019481	0,075949
СКО	0,05925			

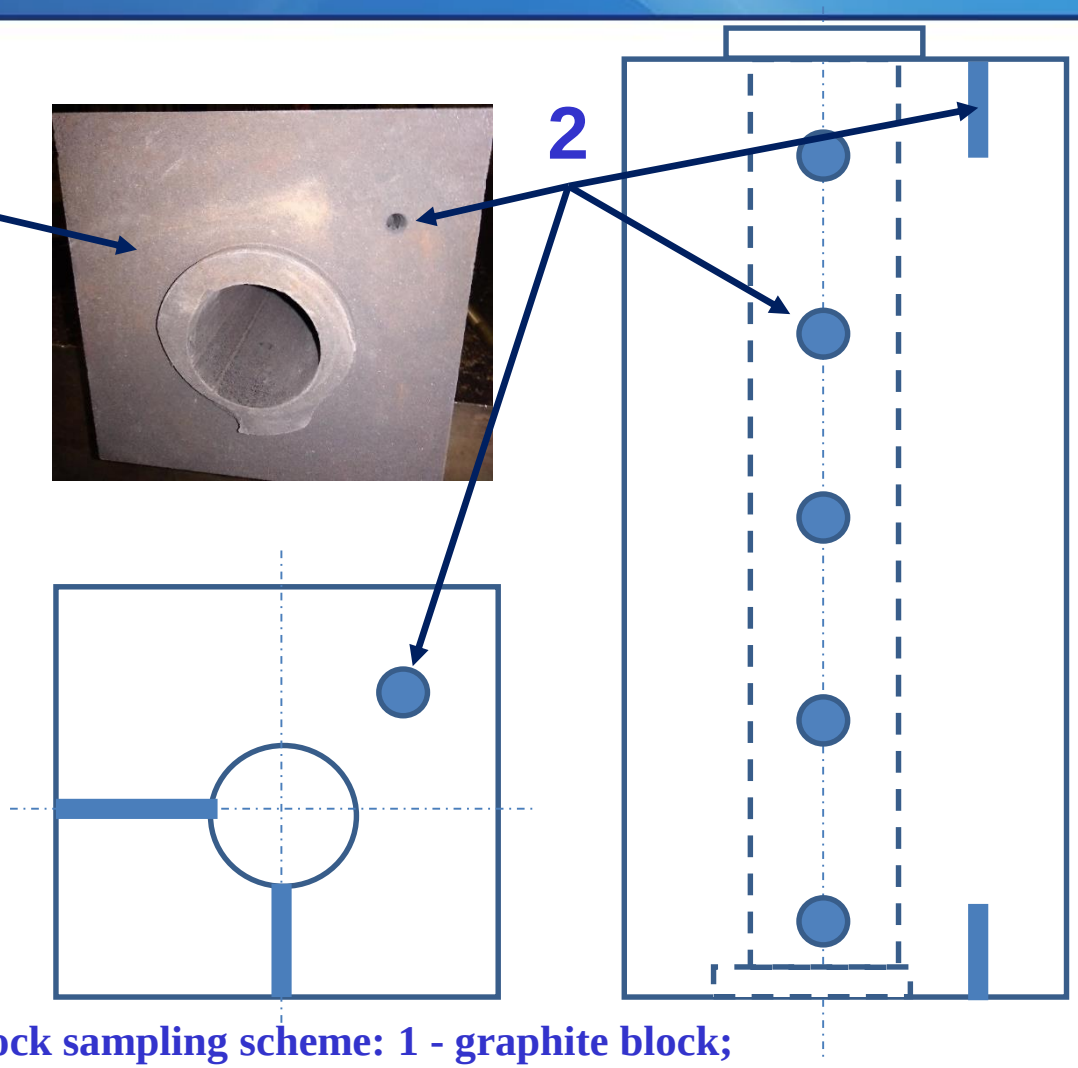


The results of the determination of the content of C-14 by participants in the inter-laboratory comparisons in Party No. 2, the certified value of the specific activity – $(1,536 \pm 0,10)*10^5$ Bq/g

SELECTION OF GRAPHITE SAMPLES FROM THE EXTRACTED PUGR BLOCK



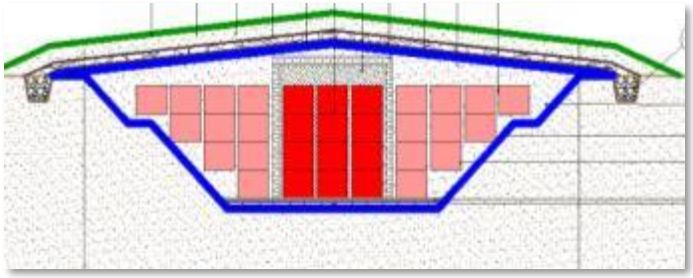




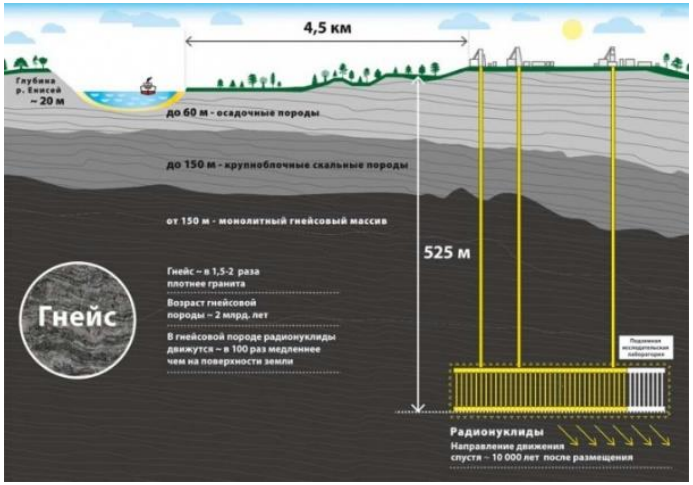
Block sampling scheme: 1 - graphite block; 2 — points for sampling graphite.

RESEARCH OF PROCESSES OF LEACHING LONG-TERM NUCLIDES FROM IRRADIATED GRAPHITE UNDER DIFFERENT CONDITIONS OF BURNING. DEVELOPMENT OF THE STANDARD OF ROSATOM STATE CORPORATION. (2017-2018)

When choosing the optimal strategy for decommissioning uranium - graphite reactors, special attention is paid to the resistance of graphite to the potential impact of the aquatic environment and the ability to retain long-lived radionuclides, activation products, Cl-36 and C-14 actinides and fission products. The probability of interaction with groundwater and, accordingly, the rate of leaching of radionuclides is taken into account when justifying the safety of both near-surface and deep burial of graphite radioactive waste.



Near-surface disposal of radioactive waste



Deep geological burial

ON PERFORMANCE OF COMPREHENSIVE PROGRAM OF MANAGEMENT WITH IRRADIATED GRAPHITE OF UGR

DIRECTION: RW management / RW disposal

TOPIC: “Study of the leaching of long-lived nuclides from irradiated graphite under different conditions of disposal. Development of the standard of ROSATOM State Corporation for determining the leaching rate of long-lived nuclides from graphite radioactive waste »

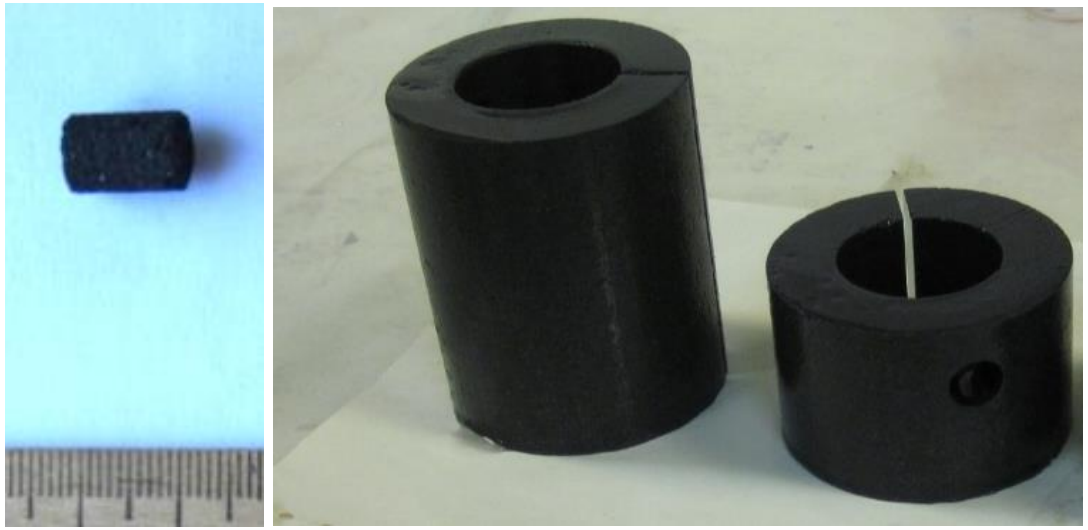
OBJECTIVES: Determination of conditions affecting the rate of leaching of nuclides from graphite RAW. Development of a draft Rosatom standard for determining the rate of leaching of long-lived nuclides from graphite radioactive waste.

TASKS:

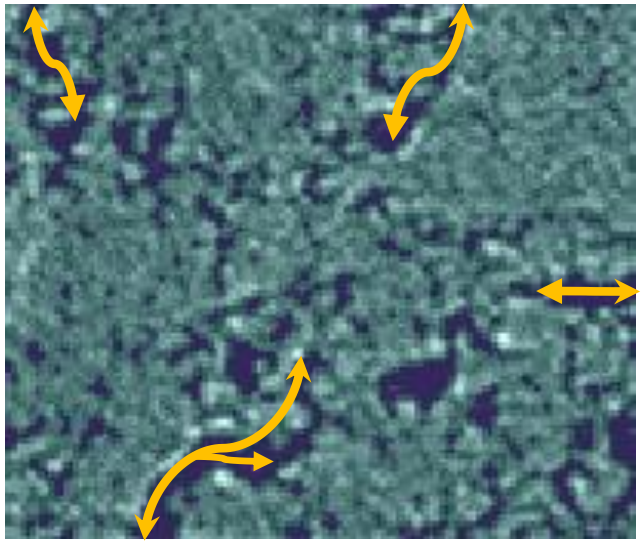
- a comparative analysis of methods for assessing the leaching rate;
- determination of conditions affecting the rate of leaching of radionuclides from graphite radioactive waste;
- determination of the optimal characteristics of equipment for performing measurements;
- development of a draft standard for determining the rate of leaching from graphite radioactive waste.

RELEVANCE OF THE DEVELOPMENT OF THE SPECIAL STANDARD FOR THE DETERMINATION OF CHARACTERISTICS OF LEACHING FROM GRAPHITE RW

A significant difference between graphite radioactive waste from other types of radioactive waste is the developed porous structure. Even for volumes characteristic of small samples, the condition $F_{pore} >> F_{geom}$ is fulfilled due to the developed porous structure of the material (volume fraction of open porosity is up to 22%, average open pore diameter is 10-100 μm).



Samples of irradiated graphite: 1 (left), types 2 and 3 (right)

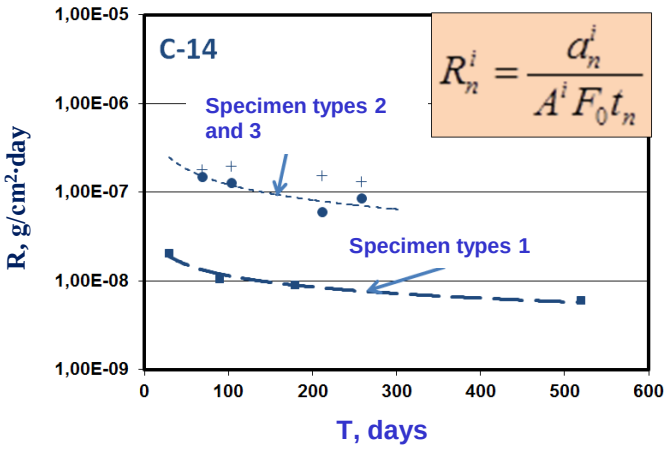


The way nuclides exit through the pores

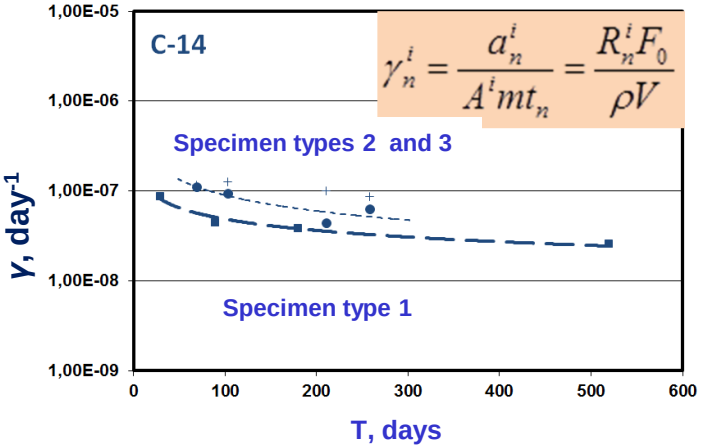
For 1 cm³ the sample area F_{geom} = 6 cm², and the pore area F_{pore} > 100 cm²

RESEARCH OF PROCESSES OF LEACHING LONG-LASTING NUCLIDES FROM IRRADIATED GRAPHITE UNDER DIFFERENT CONDITIONS OF BURNING

The change in the indicators of the process of leaching C-14 from irradiated graphite over time for samples 1, 2 and 3:



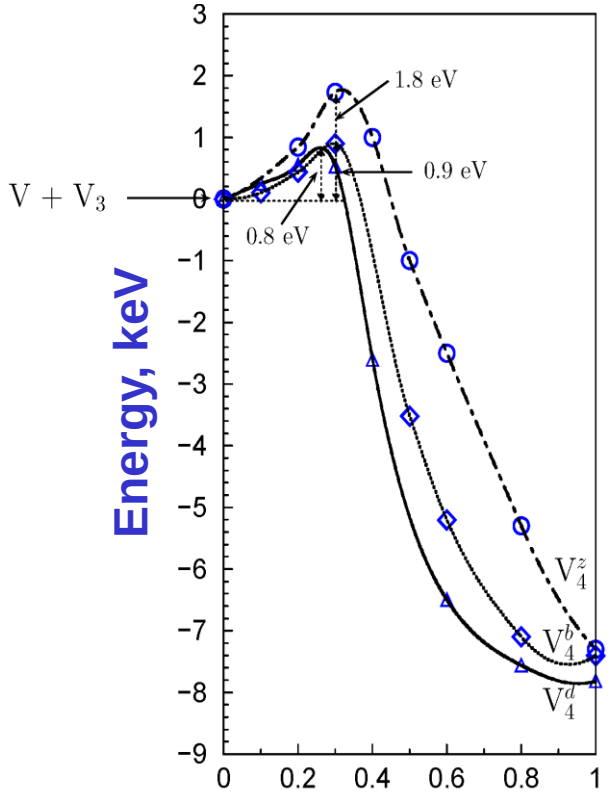
The leaching rate of the i-th radionuclide, R^i_n



The proportion of leached i-th radionuclide, γ^i_n

a^i_n - the activity or amount of the *i-th* radionuclide released into the liquid during the *n-th* leaching period, Bq; A^i - the specific activity of the *i-th* radionuclide in graphite, Bq/g; *m* - the mass of the whole sample, fragment, sample (in the case of crushed material), cm^2; ρ - the graphite density of the whole sample, fragment, g/cm^3; *V* - the volume of the whole sample, fragment, cm^3; t_n - the duration of the *n-th* leaching period, days; F_0 - geometric surface area of the sample or fragment, cm^2

ANALYSIS OF THERMAL EFFECTS UNDER WIGNER ENERGY RELEASE



Wigner energy annealing diagram

Wigner energy release rate

$$\frac{dH(E,T)}{dT} = -\frac{v}{a}(H(E,T)\exp(-E/kT),$$

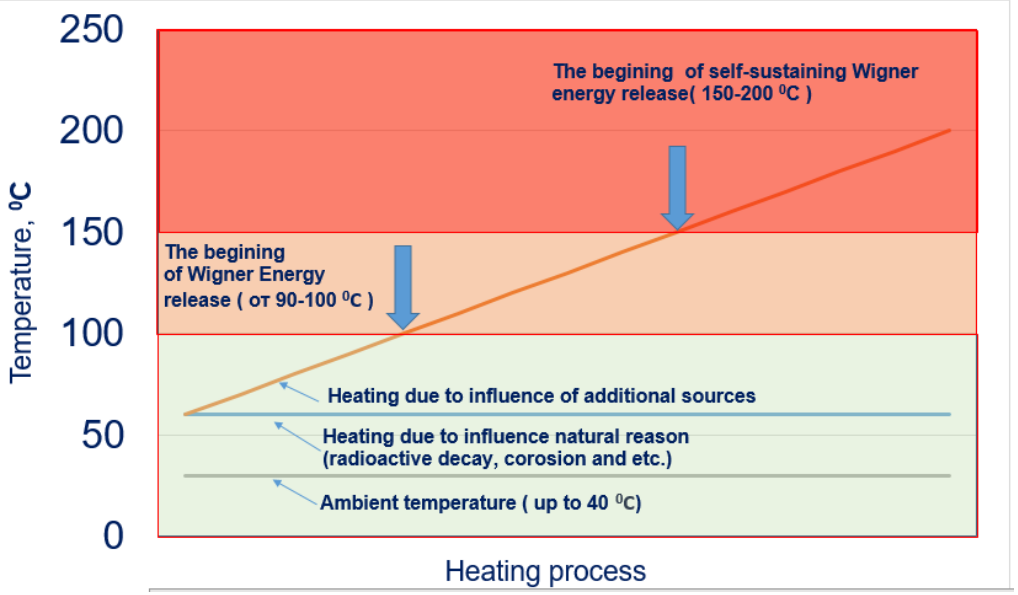
***dH (E,T)/dT*- rate of release of stored energy;**
***E* – activation energy;**
***k* – Boltzmann constant;**
***T* – annealing temperature;**
***a* – heating rate;**
***S(E,T)* – stored energy**

The temperature at which energy is released depends on the activation energy E.

It was confirmed that, in general, the precipitation begins when graphite is heated from ~ 100 °C and higher. In very rare cases (for research reactors), the onset of release at 70 °C is possible.

POSSIBLE REASONS FOR BEGINNING WIGNER ENERGY RELEASE AT DIFFERENT STAGES OF IRRADIATED GRAPHITE HANDLING

When analyzing the probability of the beginning of Wigner energy release for various stages of irradiated graphite management, it is necessary to take into account the initial and additional heating of graphite RW.



Reasons for heating graphite waste:

- heat sources in graphite radioactive waste (high content of fission products and actinides) (up to 60 ^{0}C);

- biological activity of organic waste;

- corrosion processes.

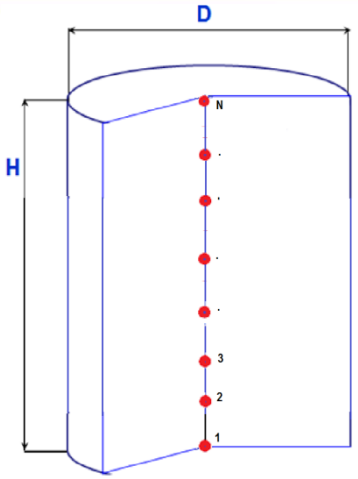
Reasons for augmented heating of graphite waste:

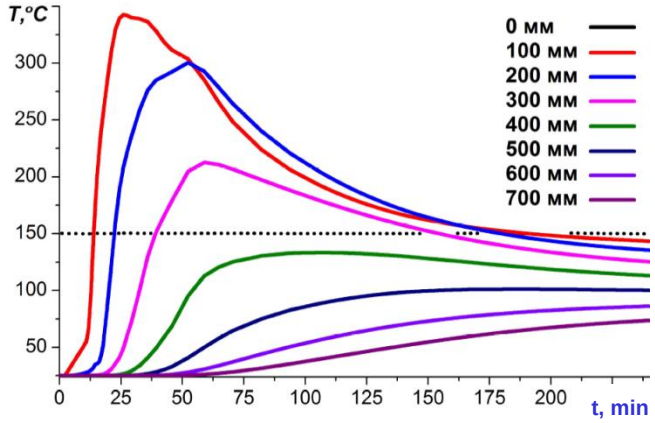
- ingress and ignition of flammable liquids, etc .;

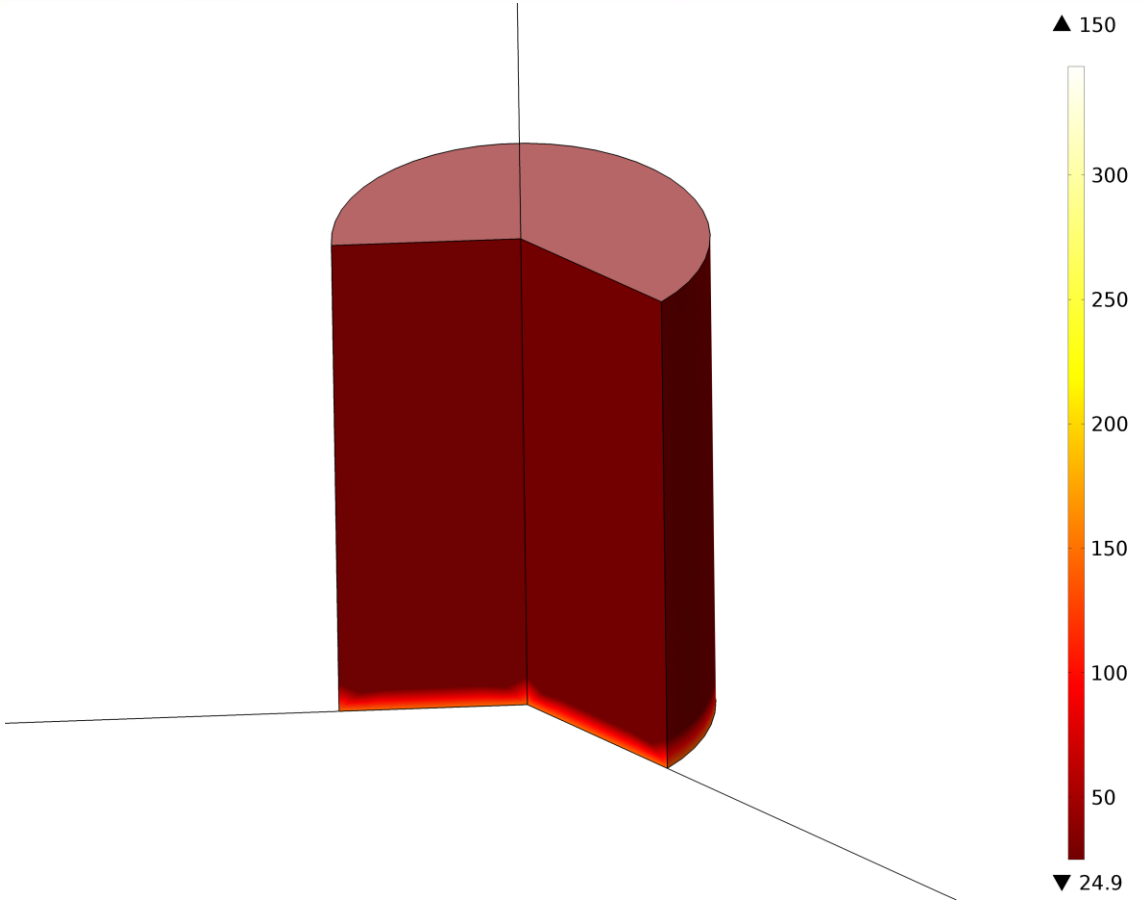
- ingress of heated particles or fragments when using thermal cutting methods during dismantling;

- concrete pouring (temperature of liquid concrete ~ 70 ^{0}C)

ANALYSIS OF THERMAL EFFECTS AS A RESULT OF HEAT GENERATION IN GRAPHITE



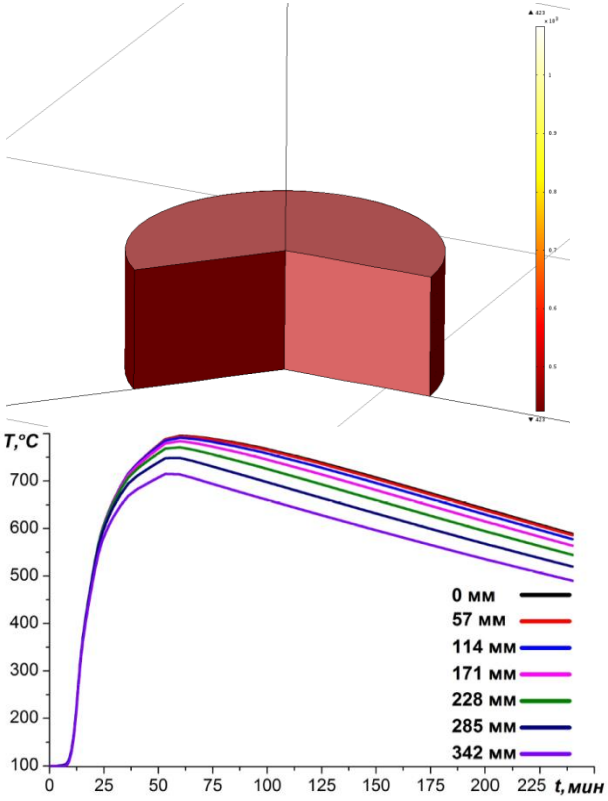




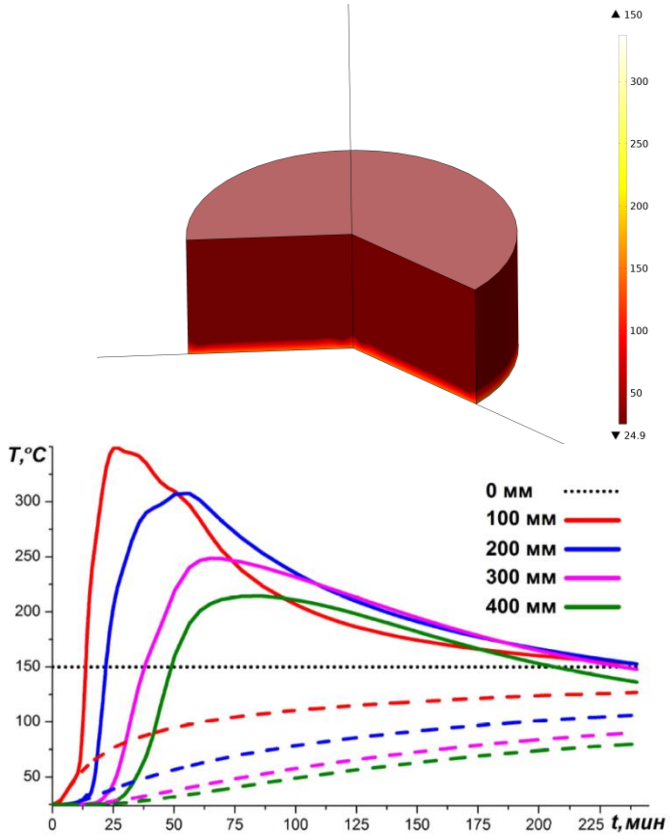
Time curves for temperature increase at points along the graphite cylinder axis inside a standard container (d=560 mm, h=800 mm, V=200 liters) during generation of Wigner energy with the generation spectrum specific for graphite irradiation temperature of 20-50 ^{0}C (bottom end temperature is maintained at the level of 150 ^{0}C).

ANALYSIS OF THERMAL EFFECTS AS A RESULT OF HEAT GENERATION IN GRAPHITE

Time curves for temperature increase at points along the graphite cylinder axis (d=1000 mm, h=400 mm) during generation of Wigner energy with the generation spectrum specific for graphite irradiation temperature of 20-50 ^{0}C

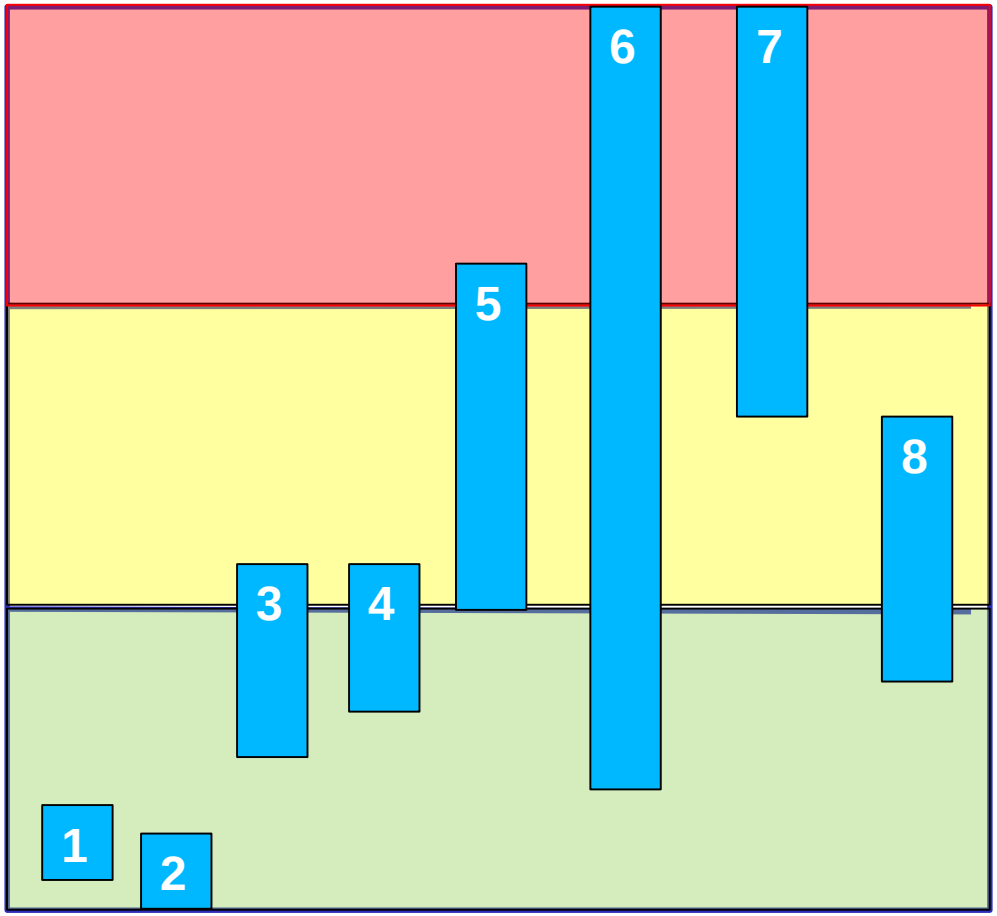


At starting point the temperature of graphite does not exceed 100 ^{0}C.



Bottom end temperature is maintained at the level of 150 ^{0}C.

COMPARATIVE PROBABILITY ANALYSIS OF WIGNER ENERGY BEGINNING FOR VARIOUS STAGES OF IRRADIATED GRAPHITE TREATMENT



1 - In-situ entombment;
2 - Storage of recycled (with thermal annealing) waste in containers;
3 - Storage of non-recycled waste in containers;
4 - Exposure before dismantling;
5 - Dismantling of graphite pile;
6 - Storage in non-conditioned form in storage facilities (nuclear legacy);
7 - Deep geological disposal with HLW (temperature from 90 0C and more).
8 - Pouring with pre-heated concrete.

High-level
Medium level
Low level

CONCLUSION

“PDC UGR” JSC has accumulated a lot of unique experience and has created the experimental base which can be a basis for foundation at “PDC UGR” of International Irradiated Graphite Center.

Currently “PDC UGR” is paying a lot of attention to development and standardization of methods for obtaining data on PUGR, RBMK and etc. graphite features, which allow to perform analysis of safety issues for different variants to treat irradiated graphite (in situ entombment, dismantling and etc.).