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Joint Research Centre

Commissioning of the Liquid Lead Laboratory for mechanical testing at JRC Petten

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Heavy Liquid Metal Coolants in Nuclear Technologies (HLMC-2018)

Obninsk, Russia, 8–10 October 2018

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Joint Research Centre (JRC)

- The Joint Research Centre is the **European Commission’s science and knowledge service**
- Mission of JRC is to support EU policies with independent evidence throughout the whole policy cycle
- JRC hosts specialist laboratories and unique research facilities and is home to more than **2000 scientists** working to support EU policy
- The JRC has ten Directorates and is located across five EU Member States (Belgium, Germany, Italy, the Netherlands, and Spain)
- The JRC's nuclear work is conducted by the **Directorate G for Nuclear Safety and Security** and funded by the Euratom Research and Training Programme. JRC pursues research, knowledge management and training activities with an emphasis on nuclear safety and security

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Specialist laboratories and research facilities at JRC, incl.

High Flux Reactor (HFR) for fuel and material behaviour studies under irradiation

Actinide User Laboratory for handling of transuranium materials

AMALIA laboratory for studies of ageing of materials in LWR environments

Mechanical Performance Assessment laboratory for pre-normative research

High Performance Computing clusters for physics-based/multi-scale modelling

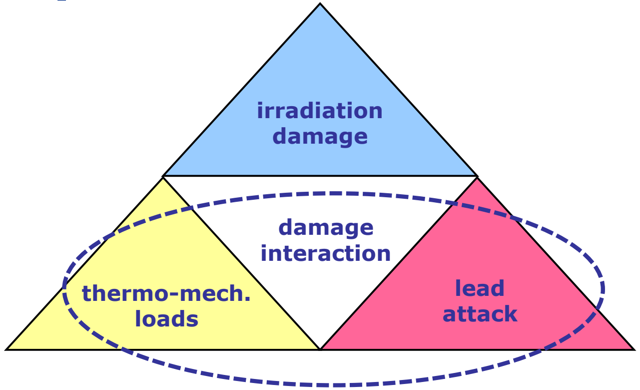
New LILLA facility for material testing in liquid lead

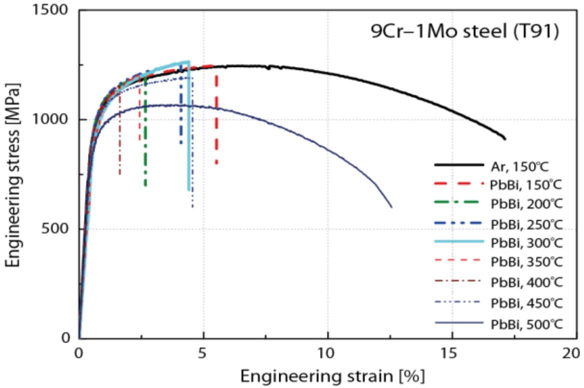
- **Objective:** contribution to assessment & improvement of future nuclear power generation
  - **Safety**
  - **Sustainability** (uranium utilization & nuclear waste management)
  - **Economy**

- The LILLA (LIquid Lead LAboratory) facility allows conducting **qualification tests** of candidate structural materials for future, **Generation-IV** lead-cooled fast reactors (**LFR**) in temperatures up to **650ºC**

- The LILLA facility is designed to perform mechanical tests in liquid lead with well-controllable parameters, in particular temperature, load, and oxygen content in **lead**

- Tests of the reliability of lead chemistry control systems and related components and instrumentation are also possible





B. Long, Z. Tong, F. Gröschel, Y. Dai, Journal of Nuclear Materials, 377 (2008) p. 219-224

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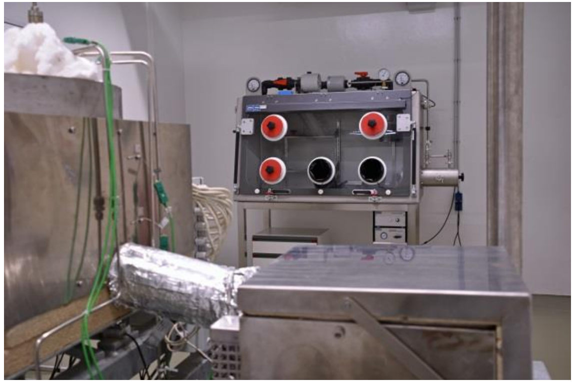
Laboratory space requirements for LILLA



- Temperature in the facility room: 20-24°C
- Air supply in the control room: 500 m^3/h
- Air supply in the facility room: 750 m^3/h
- Under pressure in the facility room: 5-10 Pa
- Room ventilation rate 5-8x of the room volume per hour
- Under pressure in the exhaust of the facility: 400-500 Pa
- Suction capacities:
  - Glove box: 10 m^3/h
  - Facility: 30-50 m^3/h
  - Facility room: 1 100 m^3/h
- HEPA filters
- Washable painting, self-closing door, ventilation alarm, hydrogen and oxygen sensors

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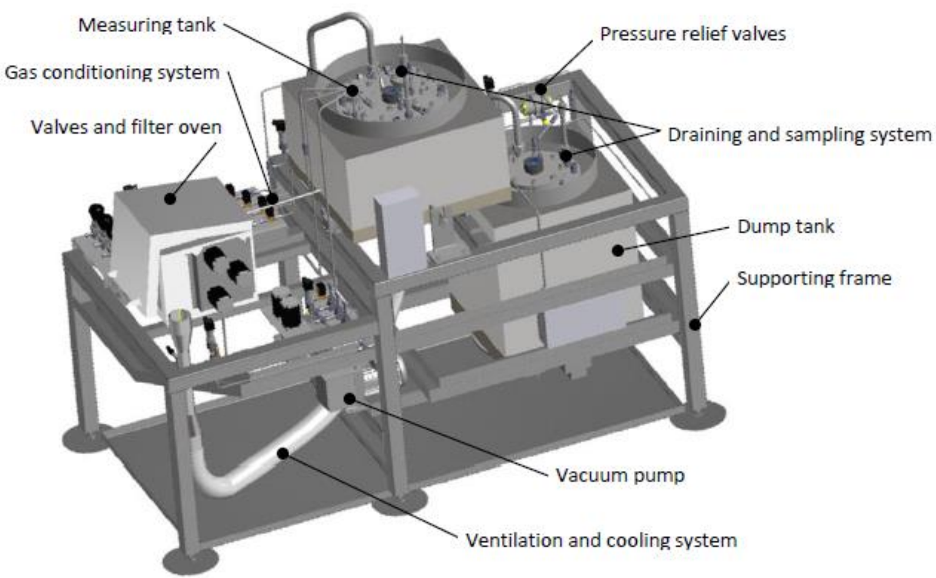
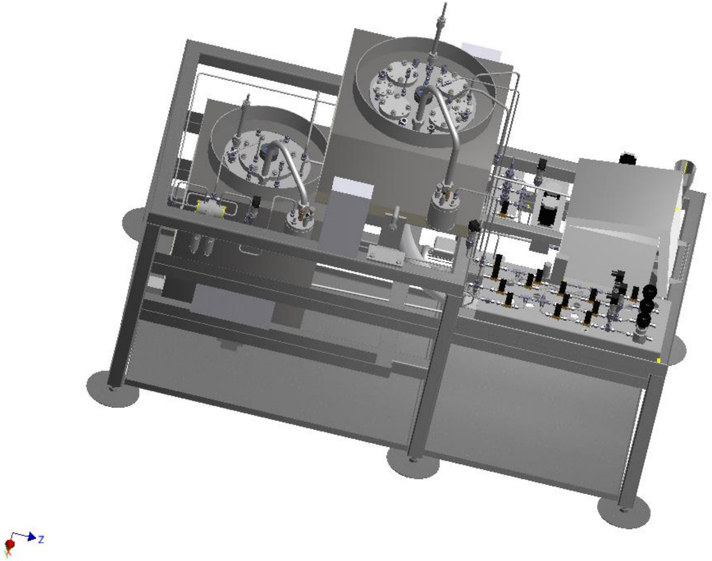
New LILLA facility for material testing in liquid lead





The facility consists of two cylindrical tanks, measuring tank and dump tank, and connecting piping to transport lead between the tanks as well as to deliver and extract gases to and from the facility, respectively.

The laboratory space has been commissioned and the LILLA facility delivered in July 2017.

Main performance characteristics of the LILLA facility

- Working temperatures in lead: up to 650ºC
- 4 test sections
- Lead inventory: ca. 27 l / 280 kg
- Structural material: AISI 316Ti
- Surfaces in contact with molten lead are protected by aluminium coating using pack cementation technology
- Possibility for active control of oxygen / gas composition, down to low oxygen concentrations (< 10^{-8} wt.%)
- Chemistry-controlling gas can be injected to cover gas space as well as directly to lead (below surface)
- Possibility for online sampling of lead composition during operation of the facility
- Possibility for filtering of gas and liquid lead

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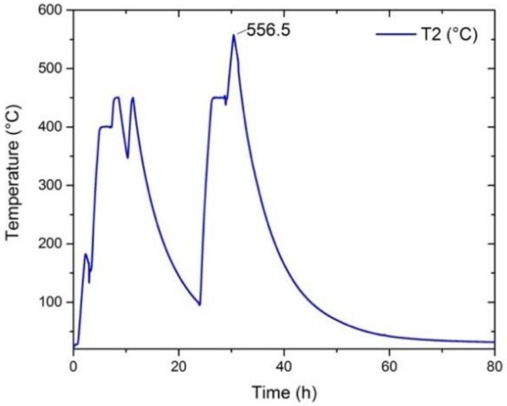
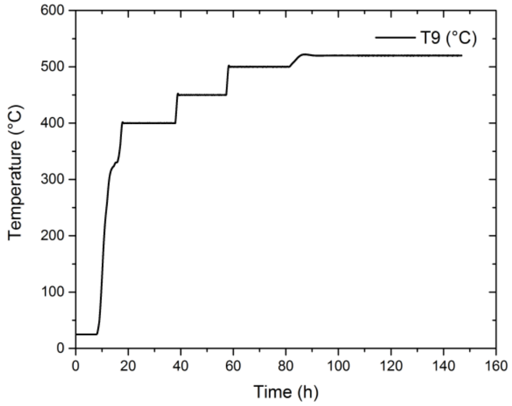
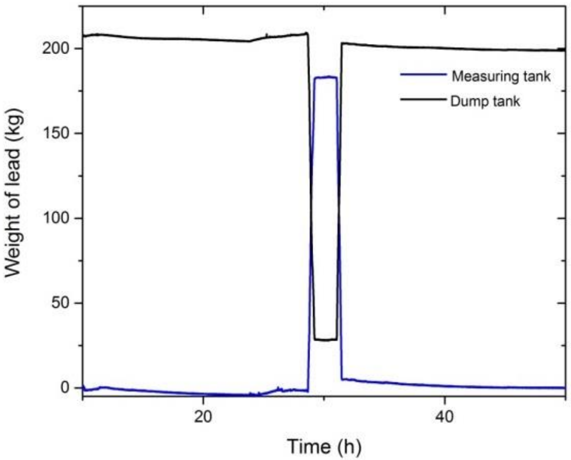
Acceptans test results



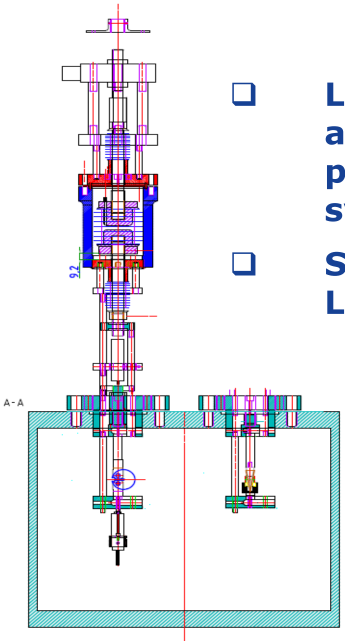




Test sections for the LILLA facility

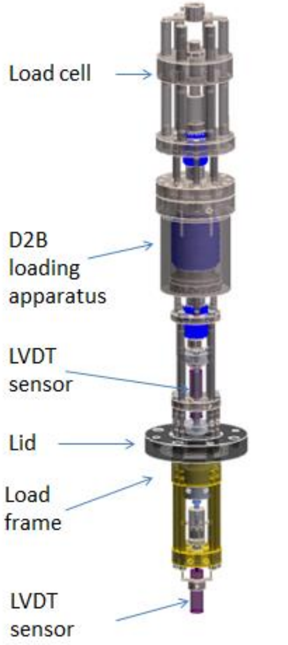
Test sections for the LILLA facility allow different types of tests:

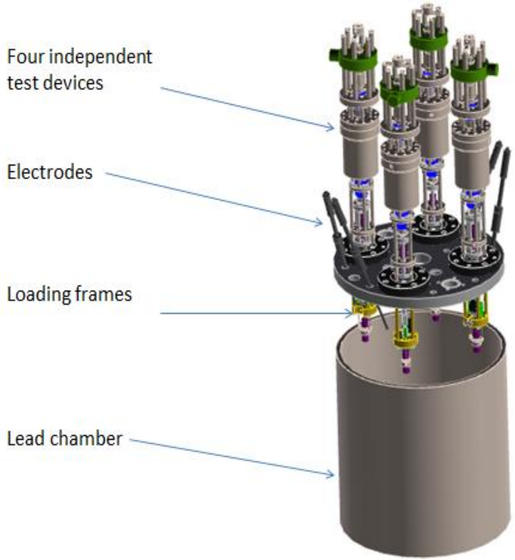
- Constant strain rate tensile (SSRT) tests
- Fracture Toughness Tests
- Crack Growth Rate tests
- Small Punch and Segmented Cone Mandrel Tests



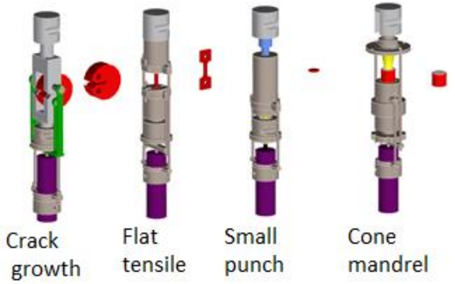
- ❑ **Load or displacement control is accomplished through unique pneumatic bellows-based system**
- ❑ **Strain feedback through the LVDT sensor**

Test section



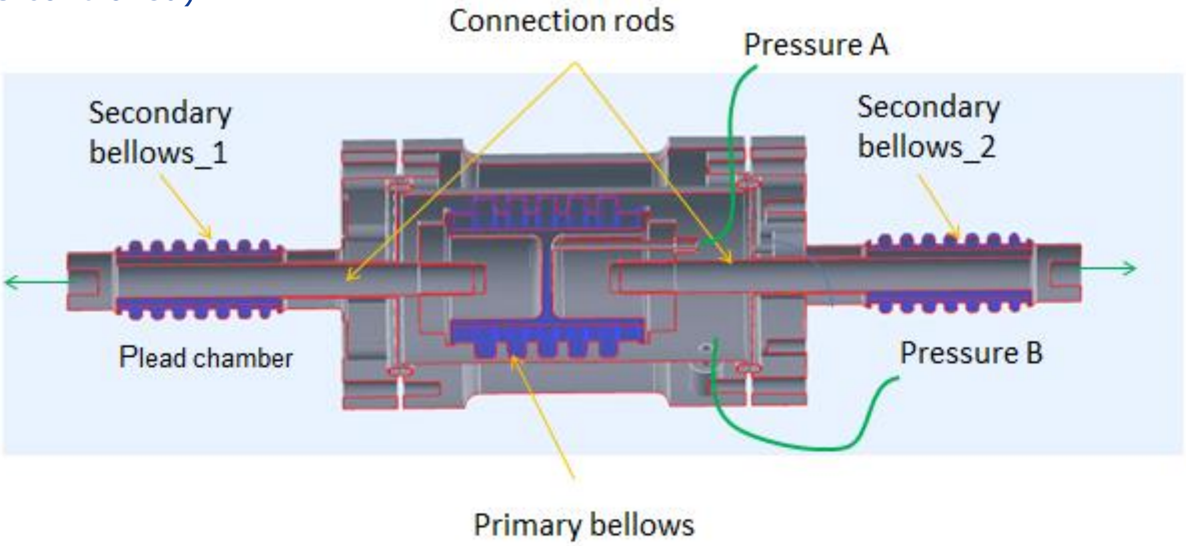


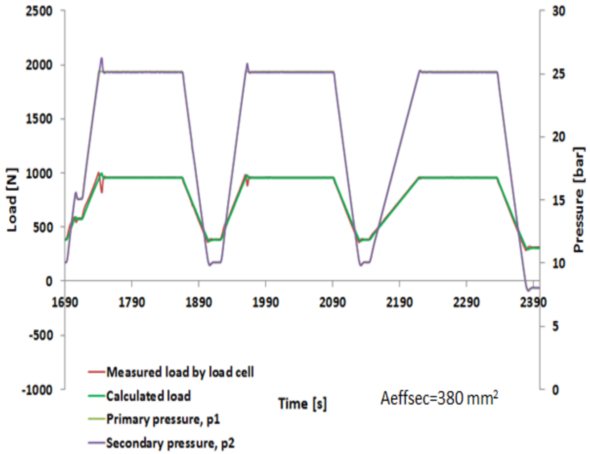
Detachable Fixings



The double2bellows loading apparatus

- Patented
- Direct load measurement by force sensor
- Primary bellows pressure p1 (PLC controlled)
- Secondary bellows pressure p2 (PLC controlled)
- Push load -> p1<p2
- Pull load -> p1>p2





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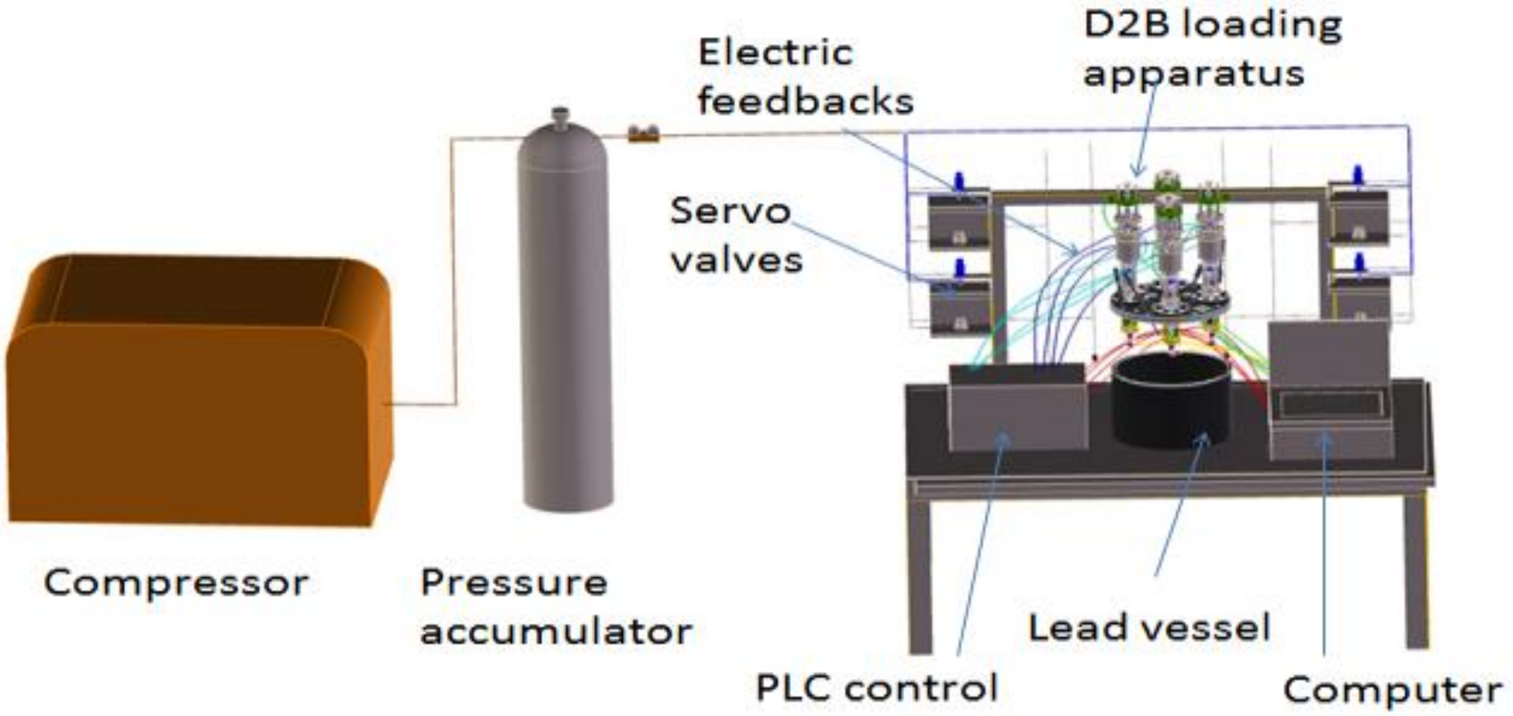
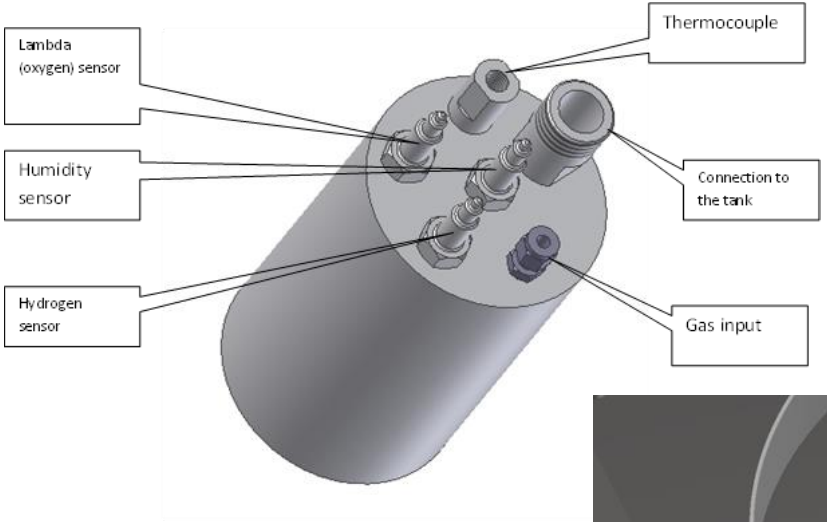
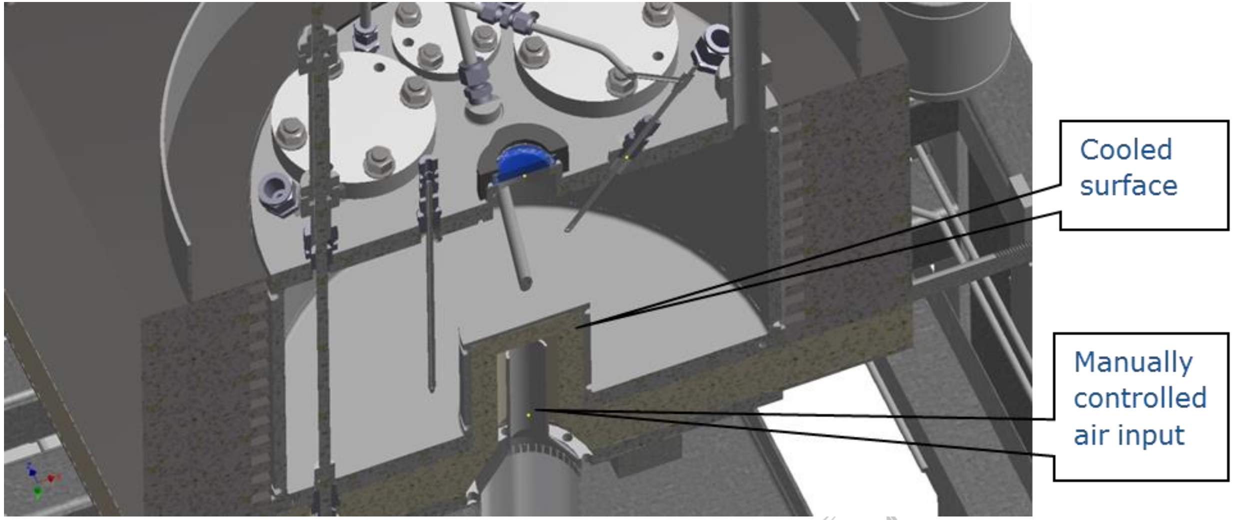
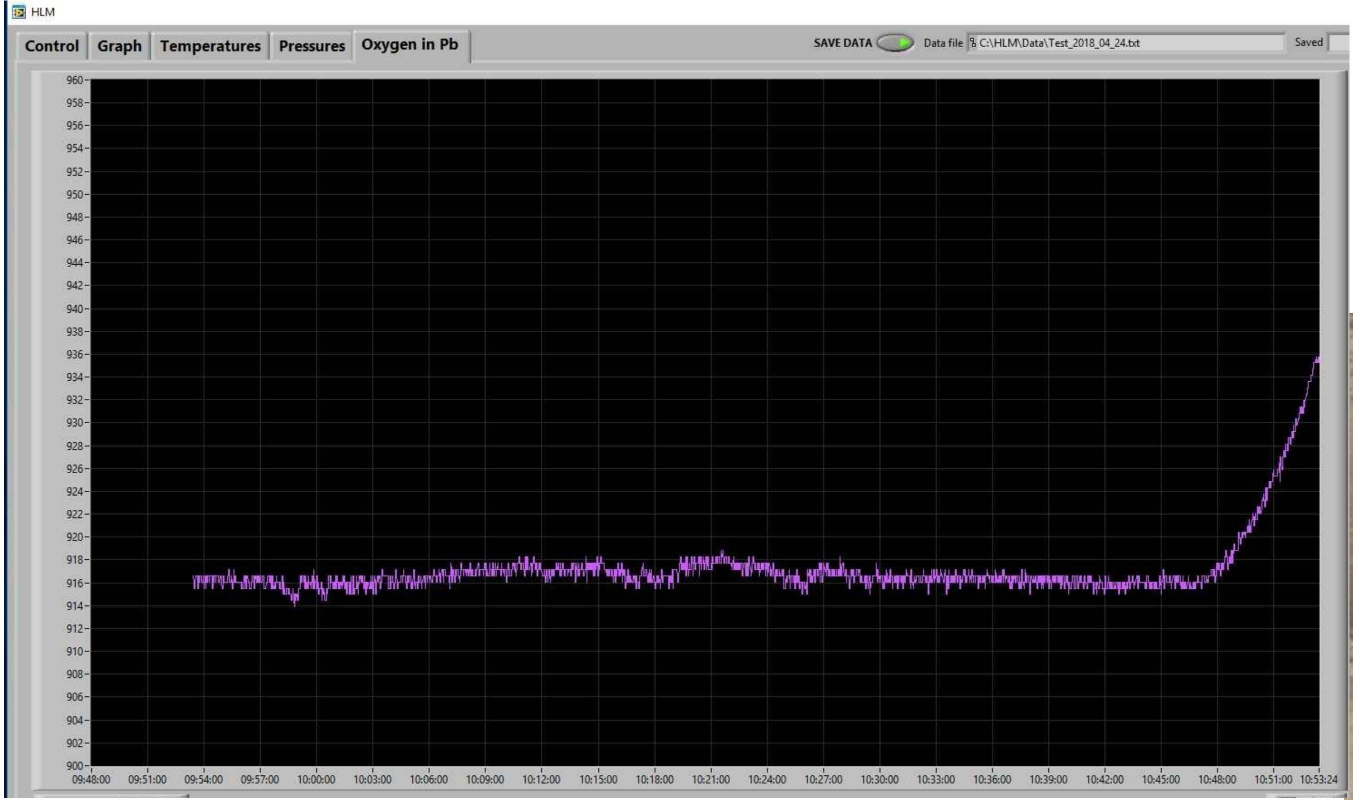
Principal scheme of the loading system



Installation of two Pt-air oxygen sensors


Lambda (oxygen) sensor
Thermocouple
Humidity sensor
Connection to the tank
Hydrogen sensor
Gas input


Cooled surface
Manually controlled air input



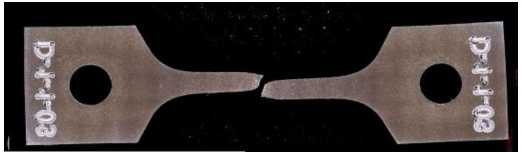
O_2 concentration in Pb at 450 °C ~ 10^{-7} wt.%

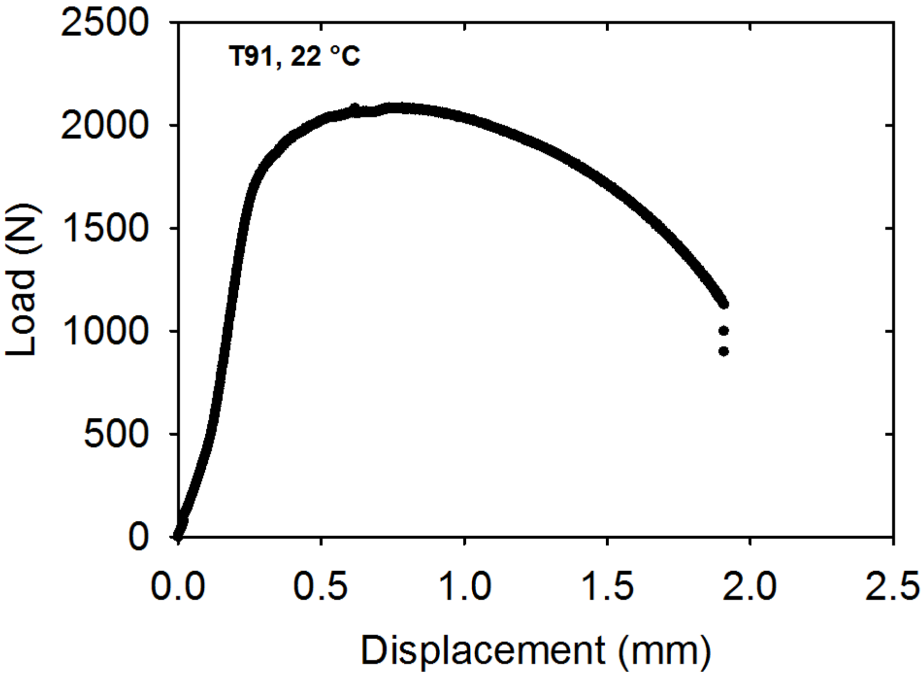
Integration of the Test sections T1 to the LILLA facility (03/2018)

PLC conrol for test

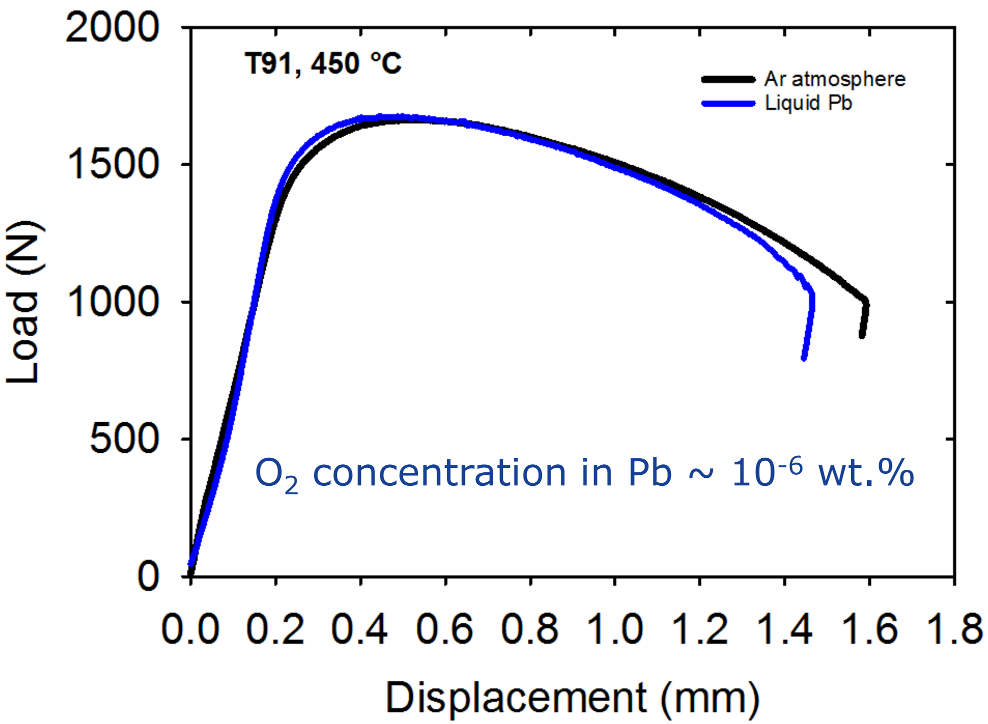
Liquid lead control

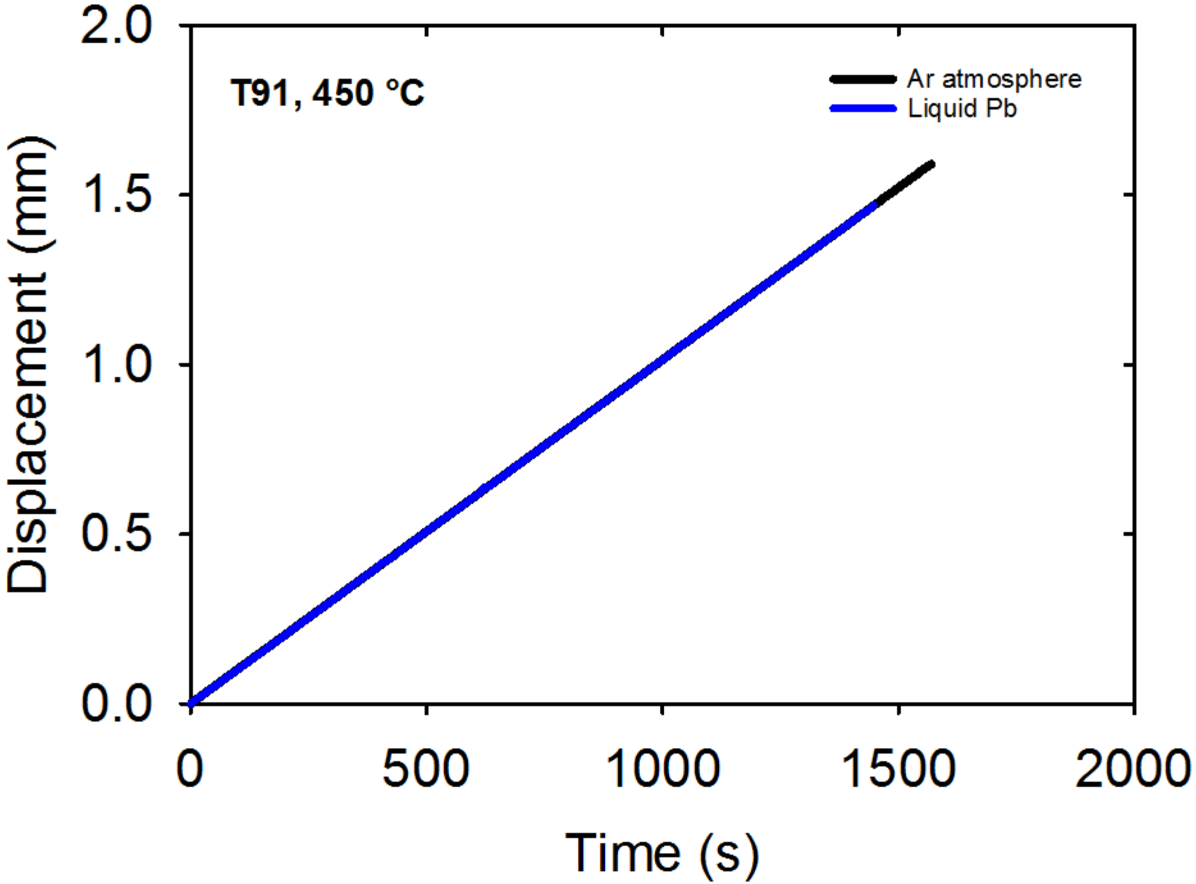
First SSRT tests at room temp. (01-02/2018)





First SSRT tests at elevated temperature in Ar and Liquid lead (04-05/2018)




T91, 450 °C
Ar atmosphere
Liquid Pb
Displacement (mm)
Time (s)
2.0
1.5
1.0
0.5
0.0
0
500
1000
1500
2000

LILLA facility project stakeholders

- Support to **Member States** in safety assessment and licensing of the European (SNETP/ESNII) LFR demonstrator **ALFRED** and the European Pb/Bi-cooled Technology Pilot Plant **MYRRHA**
- Support to European Committee for Standardization /**CEN**/: extension of the **RCC-MRx Code** for the design of mechanical components of heavy liquid metal (Pb, Pb/Bi)-cooled nuclear reactors
- **EERA Joint Program on Nuclear Materials (EERA)** Pilot Project WELLMET (Welds' manufacturing and characterization in heavy liquid metals)
- **Generation-IV International Forum (GIF)** LFR provisional System Steering Committee (SSC)
- **OECD/NEA** Expert Group on Liquid Metal Technology (EGLM)
- GEMMA (GEneration IV Materials MAturity) **H2020** Project

GEMMA: GEnIV Materials MAturity (2017-2021)

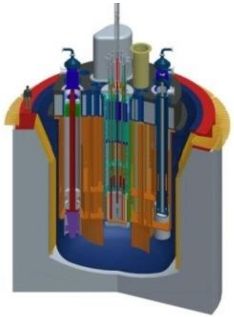
- **H2020 R&D collaborative project comprising 23 European and Korean partners, with a project budget of 6.6 M€**
- **Objective: qualification of structural materials and welded joints** for systems and components of ESNII Generation-IV demonstrators & prototypes foreseen in Europe: **ASTRID SFR, MYRRHA LFR, ALFRED LFR, ALLEGRO GFR**
- **The JRC contribution (51.4 person-months) to the project includes:**
  - coordination of WP2, incl. testing of mechanical properties in air, residual stress measurements in welds, and the **transformation of the generated experimental data to useful Design Rules in the RCC-MRx Design Code**
  - **in-situ mechanical tests of welded specimens in liquid lead** (WP4)
  - contribution to communication, dissemination, and data management in MatDB, in support to the **Horizon 2020 Open Data Pilot** allowing for verification and repeatability of project results (WP6)
  - **Total number of tests to be conducted by JRC: 134**

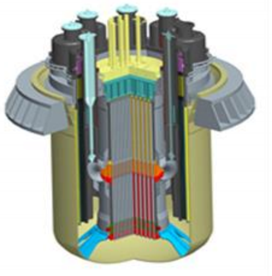
cen

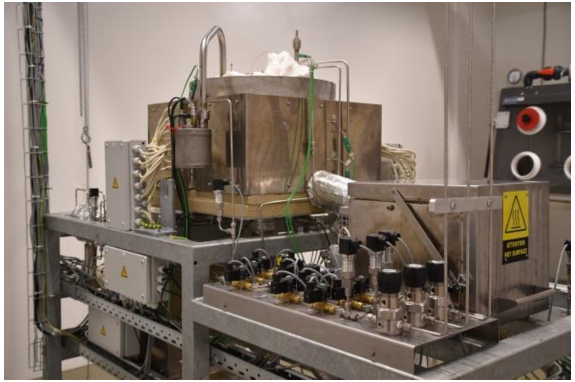
SNETP
Sustainable Nuclear Energy Technology Platform

ESNII
European Sustainable Nuclear Industrial Initiative

EERA
European Energy Research Alliance
JPNM
JOINT PROGRAMME ON NUCLEAR MATERIALS







Thank you for your attention!

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Safe Controlled Storage of SVBR-100 Spent Nuclear Fuel in the Extended-Range Future

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The Fifth Conference
"Heavy Liquid Metal Coolants in Nuclear Technologies" (HLMC-2018)
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Introduction (1)

- The design of SVBR-100 allows its operation without changes using different types of fuel both in an open nuclear fuel cycle (NFC) with postponed reprocessing of spent nuclear fuel (SNF) and in the closed NFC. The option for the fuel and NFC type is determined by economical expedience and technological mastering of the fuel and fuel cycle.
- It can be expected that at the nearest first stage of fast reactors (FR) implementation in the nuclear power (NP) in conditions of low costs of natural uranium and services on its enrichment, the operation of SVBR-100 with use of oxide uranium fuel in an open NFC with postponed reprocessing of SNF will be more economically expedient than that in the closed NFC despite the fact of significantly higher (approximately by a factor of 2 – 2.5) specific (per 1 kWh) consumption of natural uranium.

Introduction (2)

- Use of that fuel enriched in less than 20 % with postponed reprocessing of SNF is also the most expedient for the reasons of nonproliferation that is important in case of exporting those reactors to developing countries.
- When the costs of natural uranium and SNF storage increase much, changeover to the closed NFC will become economically expedient. At that point, reactors SVBR-100 will operate in a mode of fuel self-providing with a breeding ratio (BR) slightly exceeding one.
- However, operation of FRs with postponed reprocessing of SNF will require to solve the issue of safe controlled and economical storage of the SNF with retention of the opportunity to use it upon NFC closing.

1 SNF Maintenance Problems (1)

- **Up to nowadays the problem of NPP SNF maintenance has not been solved in full. That is a challenge to the current NP and a cause of concern for the certain part of population and national governments because of high long-lived radioactivity accumulated in the SNF. That fact and some other reasons, namely: safety, economics, nonproliferation, are hampering NP development. To hasten the process of finding the solution to that postponed problem, IAEA put forward the following fair principle: *"Radioactive waste shall be managed in such a way that will not impose undue burdens on future generations"*.**

- **Though, the principal scientific ways for finding the solution to that problem are clear in general terms, the practical realization of the highlighted problem requires performance of the corresponding R&D, demonstration of the entire cycle of SNF maintenance and harmonization of national strategy and policy in the area of nuclear fuel cycle in the world.**

1 SNF Maintenance Problems (2)

- That is conditioned by the fact that now the nuclear community does not possess the necessary knowledge, experience and opportunities for finding the safe and economically efficient solution to that complex problem. In the process of development of SNF maintenance methods, it should be accounted that at the existing technologies of SNF reprocessing, two first defense barriers are damaged, namely: fuel matrix and fuel element cladding. And the fuel, in which huge amount of radioactivity has been accumulated, is converted from a solid state into liquid one (or gas phase required for gas-fluoride technology) when radioactivity release into the environment is much possible.
- Moreover, in the process of SNF reprocessing it is more difficult to perform the account and control of nuclear fissile materials (NFM) that heightens the risk of their unauthorized proliferation.

2 SNF Maintenance Approaches

2.1 SNF is the NP Radioactive Waste

- **The SNF is considered as the NP radioactive waste that must be finally buried in deep geological formations, where SNF radioactive substances can be securely isolated for hundreds thousands years.**
- **During the last 30-40 years that approach has been taken up in the USA. The Yucca Mountain Nuclear Waste Repository was built on the site located in the mountains in the State of Nevada. That geological storage facility was provided for final burial of packages with spent fuel subassemblies (SFSA) after they had been cooled for many years in the in-plant SNF storages (SNFS) purposed to reduce radioactivity and corresponding heat release. The SFSAs should be located in special cans equipped by multi-barrier shielding.**
- **Currently the highlighted approach is conceded insufficient to the requirements of large-scale NP development, and implementation of that geological repository is stopped.**

2.2 SNF Controlled Storage (1)

- **Long (dozens of years) controlled storage of SNF in the in-plant or central SNFSs that now is realized almost in all countries. Currently that method of SNF maintenance is the cheapest one, meets the requirements of plutonium non-proliferation because in the SNFS the plutonium is under protection of strong gamma-irradiation of fission products that facilitates the account and control of the NFMs and leaves open the opportunity to use SNF in the closed NFC upon matured conditions in economics.**

2.2 SNF Controlled Storage (2)

- **Along with that when the issues of storage of power reactors SNF were considered at the IAEA Scientific Forum held within the frameworks of the 47th IAEA General Conference (Vienna, 2003), the certain countries expressed a desire to extend the SNF storage period up to 100 years and over. They explained that by a factor of delaying in implementation of the programs on SNF burial in geological repositories.**
- **In addition, significant extension of the SNF storage period in "dry" repositories makes possible to save the financial resources for construction of geological repositories. Absence of social agreement upon the issue of how to consider the SNF, namely: as a waste or fuel for the future NP, as well as lack of political will in activities on options for the sites for geological repositories and their construction, are resulting in the conclusion made by the certain countries concerning to expedience of extending of the SNF storage period in "dry" repositories.**

2.3 Organizing of the Closed NFC (1)

- **Organizing of the closed NFC with implementation of FRs and large-scale reprocessing of TR and FR SNF in a single NFC. Thus, the fission products are separated for their further immobilization and final isolation (really, radioactive wastes).**
- **The residual uranium and built plutonium are used for manufacturing of fresh fuel. At the same time, the task of noticeable reduction of amounts of stored SNF and unloading of repositories is solved.**

2.3 Organizing of the Closed NFC (2)

- **To solve the problem of MA burning, the different methods of nuclear transmutation of MA in fast critical or subcritical (accelerator driven) reactors are studied. In those reactors the long-lived MA are fissioned by fast neutrons and transmuted into relatively short-lived fission products. After required cooling in the controlled repository those fission products can be vitrified similar to fission products of uranium and plutonium and then safely buried in geological repositories.**
- **Realization of the highlighted approach needs implementation of the high number of FRs in the NP structure, which allow increasing approximately by a factor of hundred the efficiency of use of natural uranium energy potential as compared with that of TRs.**

3 Possible Duration of Temporal SNF Storage prior to Reprocessing (1)

- **At present it is not easy to determine the time when reprocessing of the SNF with recycling of plutonium and MA, separation of fission products and their final isolation will become economically expedient.**
- **That time will depend on specific consumption of natural uranium by existing TRs and its contribution into a fuel component of the electricity cost, forecast for resources of natural uranium and their dependence on the cost of natural uranium, escalation of natural uranium costs (now they are not increasing), outlooks for the paces of NP development in the current century, the cost of SNF storage and reprocessing, the cost of manufacturing of refabricated fuel, economical characteristics of thermal and fast reactors, the cost of final isolation of long-lived RAW.**

3 Possible Duration of Temporal SNF Storage prior to Reprocessing (2)

- For the purpose to make a preliminary decision for NFC variants for future large-scale NP, the comparative economic analysis of the open NFC with postponed SNF reprocessing and closed NFC for the USA conditions was performed in the Massachusetts Institute of Technology (MIT Reports). In those Reports it was revealed that up to the end of the current century the thermal reactors operating in the open NFC would not lose their competitiveness because of rise in the cost of natural uranium.
- However, as the commercial FRs operating in the closed NFC have not been implemented yet, there are much uncertainties in estimations of their economical characteristics. That point concerns both economical characteristics of FRs and economical parameters of processings of the closed NFC. All highlighted points will effect on forecasting determination of the time for economically expedient NFC closing. The same can be said about the assessment of economically available resources of natural uranium and paces of NP development.

3 Possible Duration of Temporal SNF Storage prior to Reprocessing (3)

- With due account of the fact that the total power capacity of SVBR-100 is 10 GW (100 reactors), operation of those reactors in the open NFC with postponed reprocessing, provided it is profitable, will require 280 thousands of tons of natural uranium.
That is why it can be expected that prior to the SNF has been involved in the closed NFC, duration of the period of storage of SVBR-100 SNF can take several decades.
- For that reason, development of the concept of sufficiently long controlled storage of SNF from SVBR-100 reactors is expedient.

4 The Concept of Safe Controlled Storage of SNF of SVBR-100 Reactors (1)

- **For the design of experimental-industrial power unit (EIPU) with RF SVBR-100 the variant of cassette-by-cassette extraction of the SFSA with their further placing in capsules with liquid lead was adopted. Then those capsules had to be stored in the in-plant storage facility with natural air cooling providing removal of residual heat from the SFSA.**
- **Along with that, experience of operating reactor facilities (RF) with lead-bismuth coolant (LBC) has revealed that it is possible to perform safe and quick refueling in short terms provided the whole core is replaced with use of the special refueling equipment set. At that point, cooling and long storing of SNF extracted from the reactor were realized in long storage tanks (LST) filled with liquid LBC, which was solidifying after that. Removal of residual heat was performed via the LST casing by natural circulation (NC) of the atmospheric air.**

4 The Concept of Safe Controlled Storage of SNF from SVBR-100 Reactors (2)

- By present, the obtained storage time without any signs of radioactivity release is about fifty (50) years. Such technology simplifies the technological process of refueling, shortens its duration, and makes possible diminishing of dimensions of the main building and reducing of its cost.
- However, when unloading in RF SVBR-100 is performed in a month from the moment of reactor shutdown, the residual heat is about 500 kW. For that reason, it is necessary to place the spent removable unit (SRU) that is a reactor basket with SFSA into the temporal storage tank (TST) filled with liquid LBC, in which there are conditions for NC and removal of heat via the tank casing to the water cooling system. After the residual heat has been decreased to the required level, the following two variants of SNF handling are possible:

4 The Concept of Safe Controlled Storage of SNF from SVBR-100 Reactors (3)

1) To transport the TPCs to the Mayak Production Association, where unloading of SRUs (baskets) from the TST is performed. Then they are dismantled for further reprocessing of SNF. That variant can be realized provided reprocessing of the SNF is organized at the Mayak Production Association on expiration of 10 years of SNF storing in the in-plant transport-package containers (ITPC) on the NPP site.

2) To transport the TPCs to the special site. At that site after the SRUs have been reloaded to the LSTs filled with quickly solidifying liquid lead, the LSTs are stored during the required time. On ten years of cooling the residual heat is about 25 kW and is easily removed by NC of the atmospheric air under the temperature of the LST wall being not more than 200 °C that provides solidifying of lead. Thus, the controlled storage of LSTs is performed during several decades till the time when SNF reprocessing and NFC closing are becoming economically expedient.

4 The Concept of Safe Controlled Storage of SNF from SVBR-100 Reactors (4)

- **With that storage, the four safety barriers are formed on the way of release of radioactive products into the environment, namely: fuel matrix, fuel element cladding, solid lead and steel casing of the LST. In addition to reduction of the cost, replacement of LBC by lead is providing more reliable protection of SRUs against extremal external effects due to the higher temperature of lead solidifying (327 °C). The additional shielding is a reinforced concrete hood, which walls are of a required thickness, that covers each LST.**
- **Separation of functions of SRU transportation in the expensive TPCs, which quantity is not large, and long storage of the LSTs protected against external effects by reinforced concrete hoods, which are comparatively cheap, can be more economically expedient than use of the large number of dual-purposed containers, the cost of each is about $2 million.**

4 The Concept of Safe Controlled Storage of SNF from SVBR-100 Reactors (5)

- That practice of temporal controlled storage of thermal reactors SNF in the reinforced concrete containers on the open site is realized in the USA (see Figure in the next slide). The cost of such storage heightens the cost of electricity only by 1-2 % upon the density of SNF storage being 0.5 tons of heavy metal per square meter.
- Upon that density of storage on the site, which square is 200 × 200 m^2, for safe controlled storage it is possible to place 13000 tons of the SVBR-100 SNF (about 1500 LSTs with spent removable parts). That corresponds to approximately 10 GWe of total power capacity of 100 year operating NPPs with reactors SVBR-100.

4 The Concept of Safe Controlled Storage of SNF from SVBR-100 Reactors (6)

SNF storage at the open site in the USA

Conclusion (1)

1. SNF maintenance is an NP postponed problem that has not been solved in full in any country in the world. It is possible to find the most complete solution to that problem when closing of the NFC is realized and the large number of FRs are implemented in the NP structure.
2. At present it is difficult to determine the real time when putting in operation of FRs operating in the closed NFC is becoming economically expedient.

   That period is determined by economically available resources of natural uranium and economical characteristics of FRs and processings of the closed NFC.

   For that reason, it is a universal practice of SNF storing in cooling water pools and then in “dry” storage facilities on the NPP sites or centralized repositories.

Conclusion (2)

3. For operated LBC cooled reactors the long storage of the SNF in long storage tanks filled with solidified LBC was successfully realized. Under such storage the four defense-in-depth barriers are formed on the way of radioactivity release into the environment, namely: fuel matrix, fuel element cladding, solid LBC and tank casing, which assure the high level of safety.
4. It is expedient to consider the similar solution for reactors SVBR-100, in which LBC is replaced by lead providing the higher level of protection due to its higher melting temperature. The time of such storage can reach several decades when realization of the closed NFC is becoming economically efficient.
   On the site, which square is 200 × 200 m^2, it is possible to place 13,000 tons of the SNF that corresponds to approximately 10 GWe of total power capacity of 100 year operating NPPs with reactors SVBR-100.

THANK YOU VERY MUCH

FOR YOUR ATTENTION

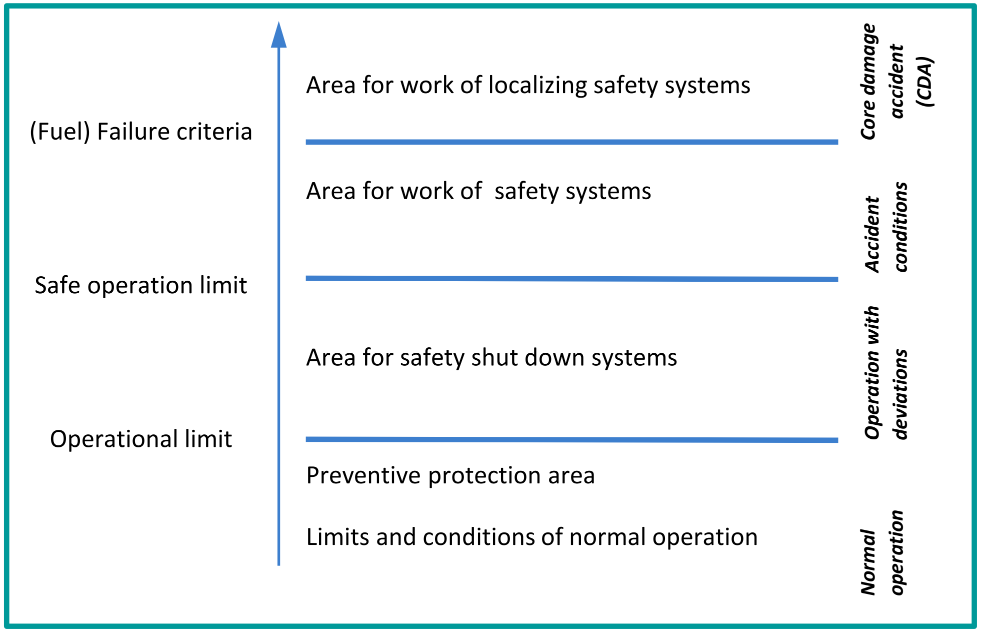
The main results of the design limits justification for the SVBR-100 reactor fuel rod cladding material

Petrochenko V.V., Grigoriev S.A., Dedul A.V., Komlev O.G., Kondaurov A.V.
JSC "AKME-engineering", Moscow, A.Dedul@svbr.org

HLMC – 2018, Obninsk, 8-10 October

FEATURES OF IDENTIFICATION OF SAFETY DESIGN LIMITS IN THE SVBR-100 PROJECT

A simplified scheme for the design limits in accordance with the requirements of the Russian safety rules and regulations

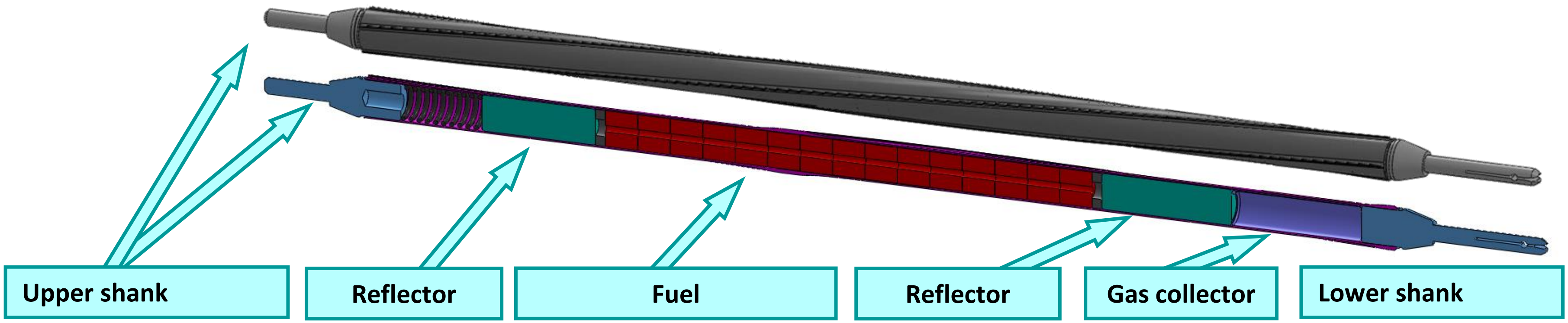


According to p. 1.1.2 «GENERAL SAFETY PROVISIONS OF THE NUCLEAR PLANTS» (NP-001-15):
«The general provisions set goals and basic criteria for the safety of nuclear power plants ... The scope of implementation of these principles and measures must comply with federal norms and rules in the field of the use of nuclear energy. **In the absence of the necessary legal acts, the proposed specific technical solutions are justified in accordance with the current level of development of science, technology and manufacturing.**»

For the most common types of NPPs, the limits for the fuel element damage and the requirements for the reactivity coefficients of the nuclear reactors are set in the Appendix to the Safety rules (NP-082-07). For new projects, which include the SVBR-100 reactor project, the main safety criteria, including the design limits, should be justified in the project (paragraph 1.1.2 of the GSP).

High boiling point, chemical inertness of the lead-bismuth coolant, ensuring safety, exclude a simple definition of the limiting process parameters that are dangerous in terms of the integrity of the safety barriers (fuel composition and fuel cladding). Nevertheless, such parameters, in accordance with generally accepted approaches to the safety justification, must be defined

OPERATION CONDITIONS OF SVBR-100 REACTOR CORE ELEMENTS (FUEL RODS)



Operation conditions for the SVBR-100 fuel rods

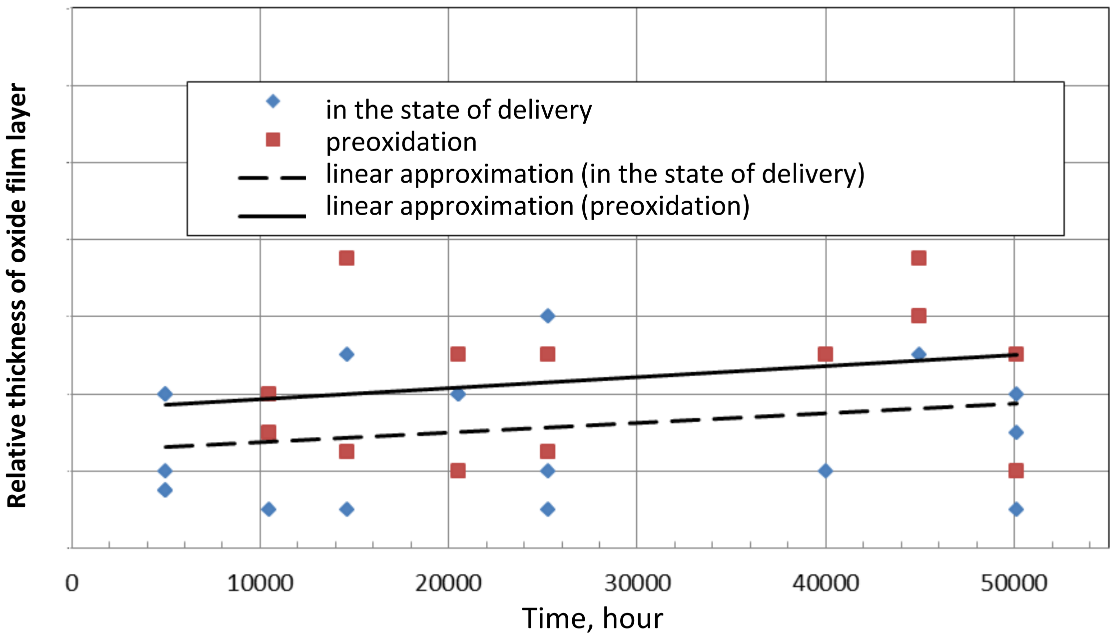
Characteric	Value
Coolant temperature at core inlet, ^{O}C	335^{+9}_{-28}
Coolant temperature at core outlet (fuel rod part), ^{O}C	485^{+14}_{-33}
Maximum (with uncertainties) cladding temperature, ^{O}C	620
Coolant velocity, m/s, no more than	2,5
Maximum linear load on fuel rod, Wt/sm	390
Neutron fluence with E>0,1 MeV for the cladding, maximum, 1/sm^2	$2{,}5\ 10^{23}$
Damage dose at cladding, dpa, maximum	97
Fuel burn-up, % h.a., maximum local	9,9
Core lifetime, eff. h.	50000
Fuel type	UO_2

Classical fuel rod of container type

Features:

- Cladding made from seamless cold-deformed pipe with spiral ribs
- Modified oxide fuel
- Molybdenum foil between fuel and cladding for increased maneurability

KEY RESULTS OF LONG-TERM RESOURCE TESTING OF CLADDINGS FROM EP-823 STEEL

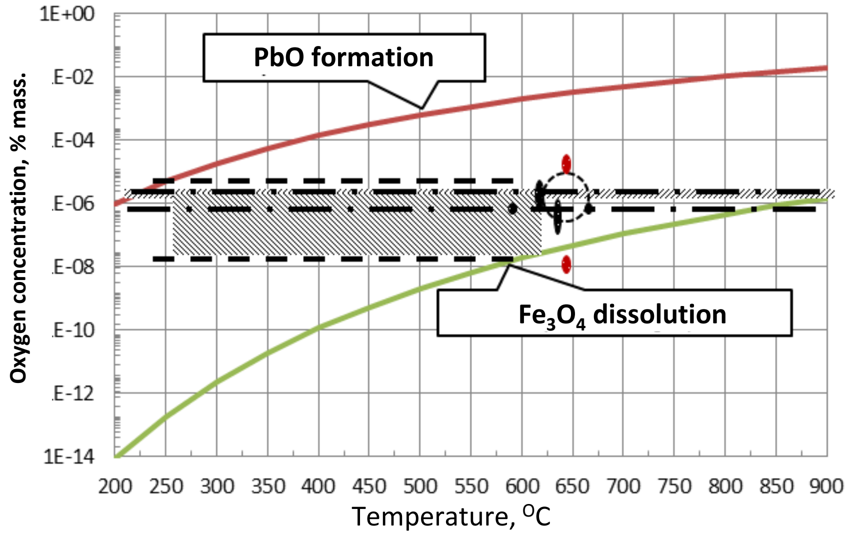


Bench corrosion testing of fuel rod models in the flow of coolant lead-bismuth was carried out on two identical loops SM-1 and CU-2M in the SSC RF-IPPE at the design values of the dissolved oxygen concentration and maximum circulation velocity of the alloy

- There is no effect of exposure in a lead-bismuth coolant on the mechanical properties of steel
- The mechanical properties of the shells vary within the limits typical for pipes made of EP-823 steel subjected to normal thermal aging at 600 ° C

- The total time for 12 campaigns of resource corrosion tests at temperature of 600 ° C was 50 135 hours, i.e. corresponded to the core lifetime provided for the SVBR-100 reactor
- As a result of corrosion tests, the claddings of fuel rod models made of EP823 steel (both in the delivery state and pre-oxidized according to the technological process previously adopted for fuel rods with lead-bismuth coolant) were not subjected to any corrosion-erosive damage.
- No liquid metal corrosion, no signs of development of crevice corrosion at the places of sample spacing in perforated grids of working areas, nor the progress of corrosion-erosion damages of spacing ribs of tubular samples
- The applied artificial defects with the complete removal of the "protective oxide film" after 25 269 hours effectively "self-healed" without the additional corrosion-erosive damages

TESTS OF SAMPLES IN CONDITIONS OF THE LEAD-BISMUTH CIRCUIT OXYGEN REGIME VIOLATION



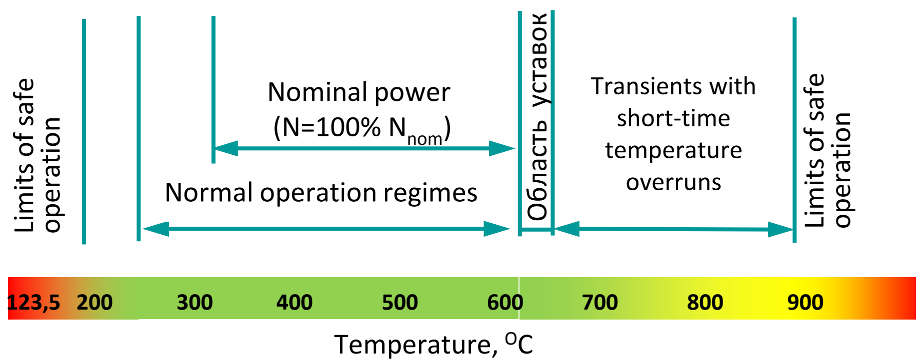
- - - - region of (thermodynamically) admissible values;

- · - · - - the area adopted for the parameters of the SVBR-100 (50 000 hours of successful testing)

- areas investigated on a time basis from 6000 to 25 000 hours

- areas studied on a time basis from 9860 to 29 530 hours under reactor loop testing conditions

- the test area for extreme values of oxygen concentration at a time base of about 500-600 hours



- The average oxidation rate of the metal was 0.14 mm / year at a time base of 635 hours. No any traces of liquid metal corrosion nor significant damage to the samples were not detected
- Local corrosion-erosion damages with a depth of 0.42-0.43 mm on the smooth part of the samples and up to 1.5-2 mm on the shanks of the models. The maximum rate of liquid metal corrosion was about 0.68 µm / h on a time base of 580 hours.

THE MAIN RESULTS OF DESTRUCTIVE TESTING OF FUEL ROD MODELS WITH OVERHEATING

The typical results of destructive testing of tubular samples made from EP-823 steel, loaded with internal gas pressure (end of fuel lifetime)

№ of sample	Testing temperature, °C	Time to damage, min
44	900	5,1
40	850	9,5
45	800	60
46	750	900

Typical view of samples after brake down tests (Sample №40 – t=850°C, σ=60 MPa, τ=9,5 min)

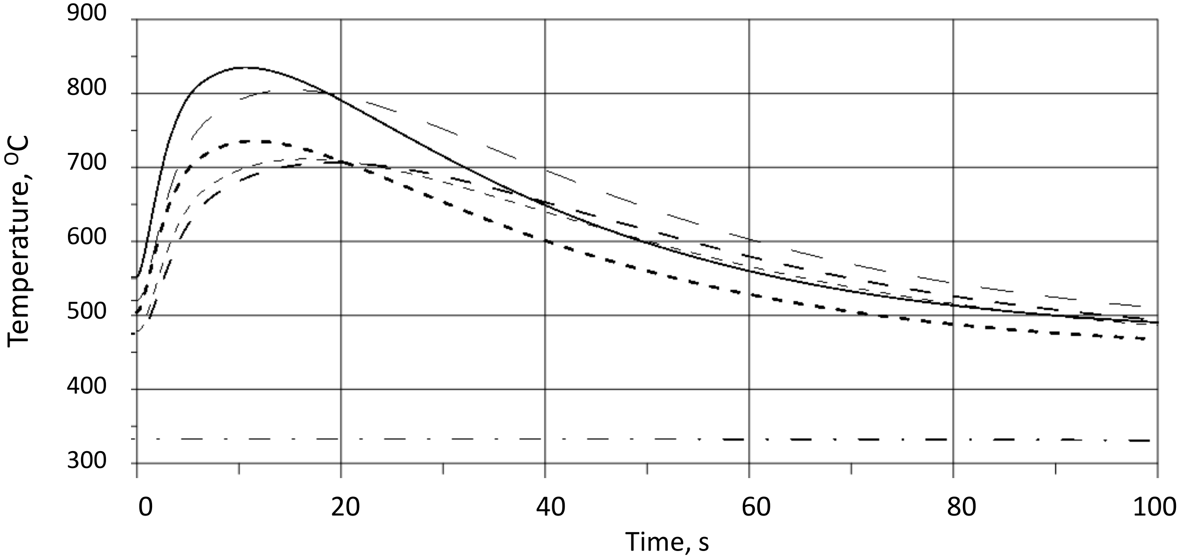
When simulating emergency overheating, depressurization always * occurred according to a typical scenario:

- **Significant plastic deformation (swelling and thinning of the cladding in the region of maximum temperatures)**
- **Longitudinal crack between ribs**
- **Crack opening and depressurization**

* *Note: once after significant deformation (before the formation of a longitudinal crack) - a crack appeared in the welding of the shank and tube (defect in the weld)*

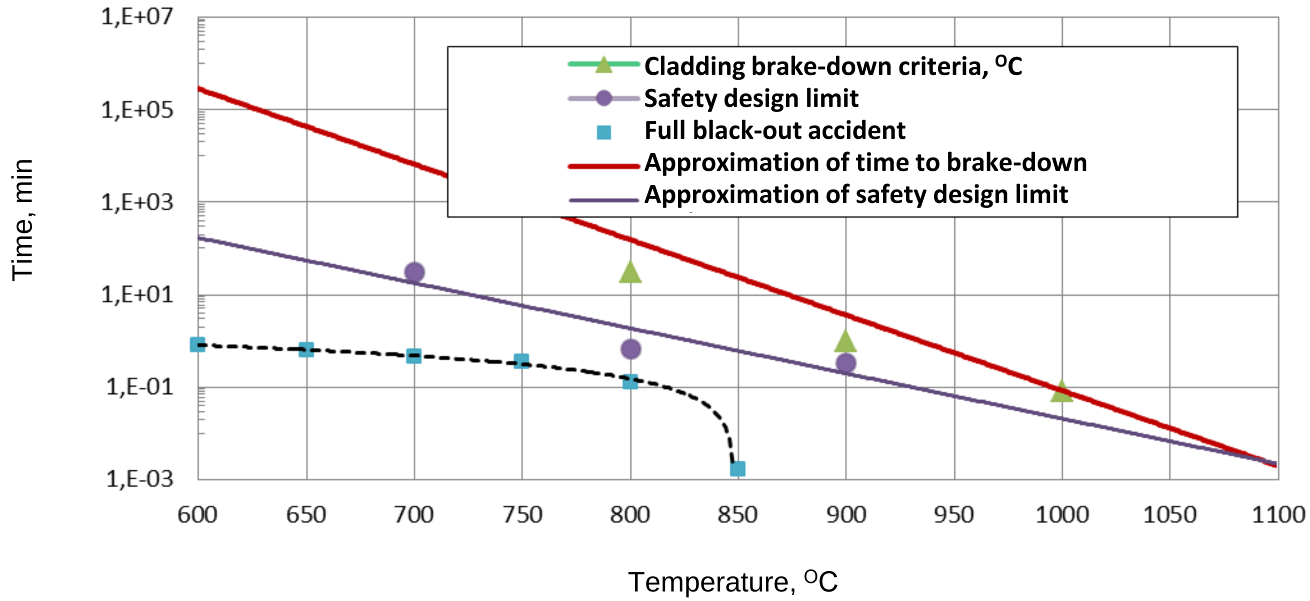
TYPICAL FEATURES OF "ACCIDENTAL" TRANSIENTS IN SVBR-100 REACTOR CORE

Changes in the representative temperatures of the core during the accident with the complete black-out of the NPP power unit with SVBR-100



- - - - - -	- temperature of the coolant at the outlet from the fuel part of the core
– – –	- coolant temperature at the outlet of the most loaded part of the core
–––	- «mixed» coolant temperature at the core outlet
– · – · –	- coolant temperature at the core inlet
———	- maximum cladding temperature of the most loaded fuel rod
- - - - - -	- maximum cladding temperature of the average loaded fuel rod

TEMPERATURE DEPENDENCES FOR SAFETY LIMITS AND LIMITING PARAMETERS FOR ACCIDENT WITH FULL NPP BLACKOUT



The temperature-time dependencies adopted in the SVBR-100 project for:

- Maximum core safety design limit and
- Safe operation limit

for the temperature of the fuel cladding

Based on the existing database for long-term strength and creep tests, an estimate was made for the maximum residual deformation of the cladding of the most intense fuel rod, which was approximately 0.24%, which apparently does not pose a danger to the performance of the fuel rod and the core as a whole after full black-out accident

CONCLUSION

- The proposed approach, namely, the definition of the maximum allowable temperature-time dependencies and / or calculated values of residual deformations of the fuel element cladding, to the formulation of key safety criteria for lead-bismuth cooled reactor plants seems acceptable for developers and quite convenient for practical use.

- Preliminary results of R & D fully confirmed the acceptability of the basic safety design criteria of the SVBR-100 reactor plant, which ensures a high level of safety

Thank you for attention !

Progress in LBE coolant chemistry technology towards realization of MYRRHA

Lim Jun, Gladinez K., Marino A., Prieto B. G., Rosseel K., Kennedy G., Aerts A.
Belgian Nuclear Research Centre (SCK•CEN)

HLMC-18, October 08 -10, 2018, IPPE, Obninsk, Russian Federation

Belgian Government decision on September 7, 2018

- Belgium decided to build a new large research infrastructure at Mol : MYRRHA
- Belgium allocated 558 M€ for the periode 2019 – 2038:
  - 287 MEUR investment (CapEx) for building MINERVA (Accelerator up 100 MeV + PTF) for 2019 - 2026
  - 115 MEUR for further design, R&D and Licensing for phases 2 (accelerator up to 600 MeV) & 3 (reactor) for 2019-2026.
  - 156 MEUR for OpEx of MINERVA for the periode 2027-2038
- Belgium requests to establish an International non-profit organization in charge of the MYRRHA facility for welcoming the international partners
- Belgium continue to mandate Secretary of State for Foreign Trade Mr Pieter De Crem for promoting MYRRHA and negotiating international partnerships

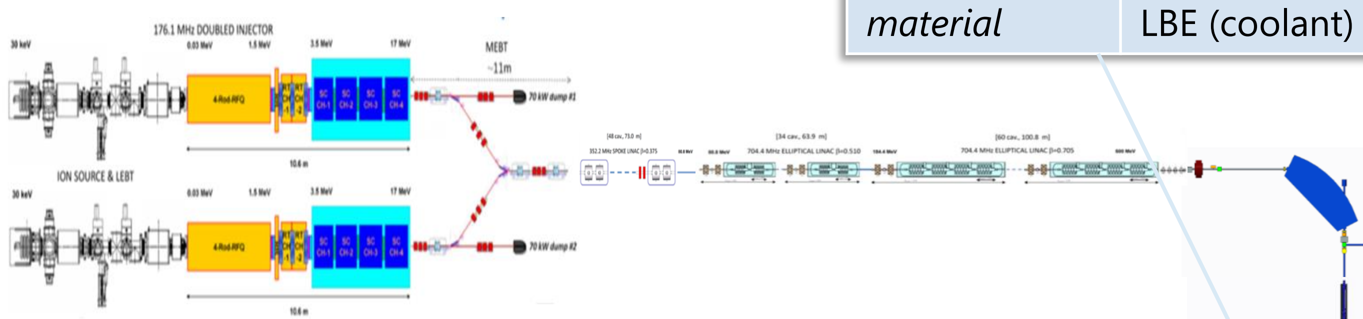
Key technical objective of the MYRRHA-project: an Accelerator Driven System

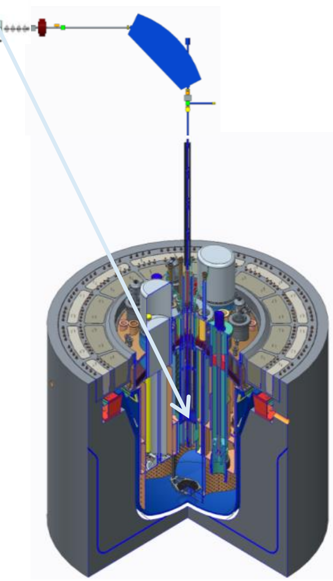
Construction of an Accelerator-Driven System (ADS) consisting of

- A 600 MeV – 2,5 mA to 4,0 mA proton linear accelerator
- A spallation target/source
- A lead-Bismuth Eutectic (LBE) cooled reactor able to operate in subcritical & critical mode

Accelerator	
particles	protons
beam energy	600 MeV
beam current	2.4 to 4 mA

Target	
main reaction	spallation
output	$2 \cdot 10^{17}$ n/s
material	LBE (coolant)



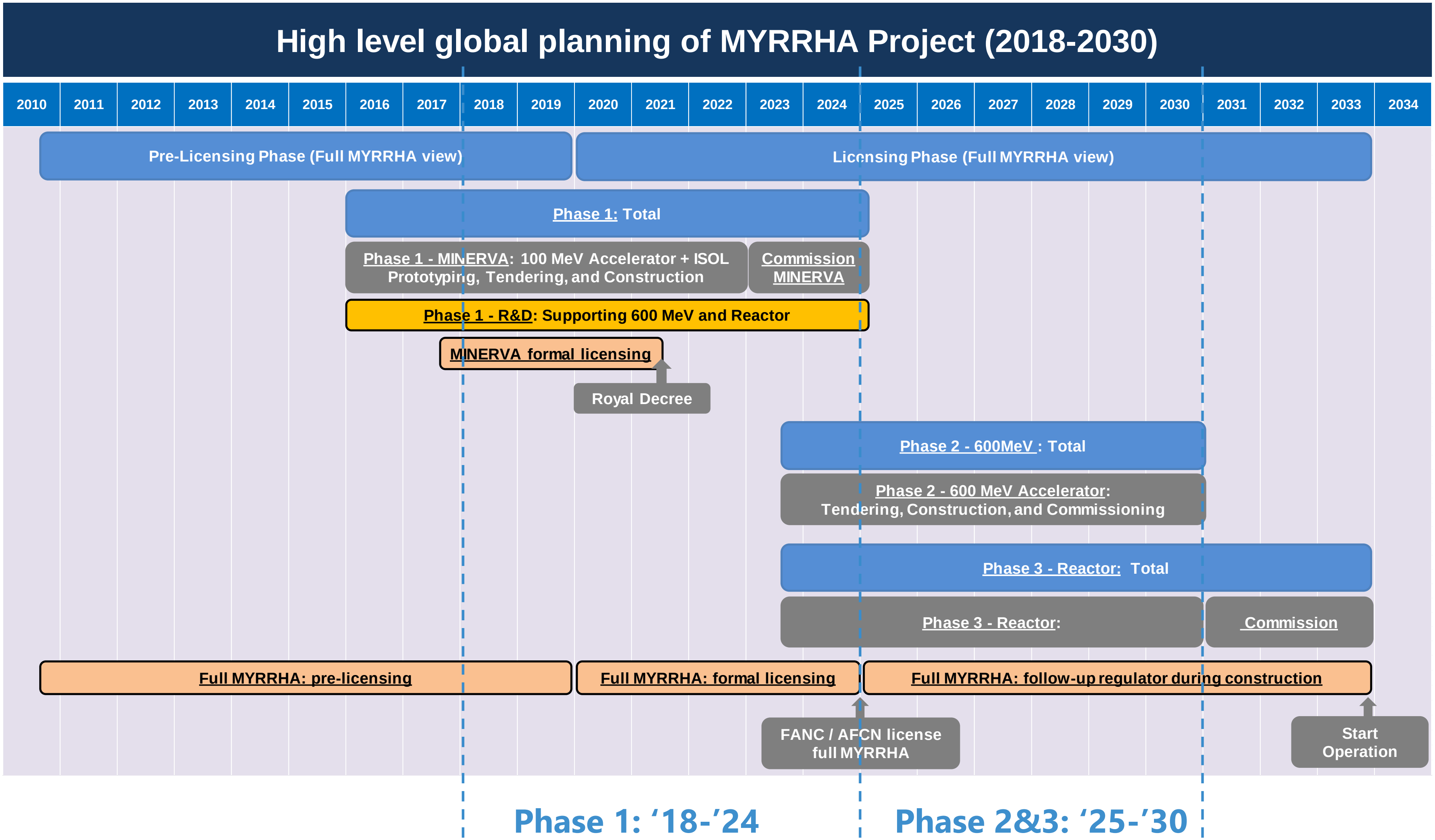


Reactor	
power	65 to 100 MW_{th}
k_{eff}	0,95
spectrum	fast
coolant	LBE

Source: SCK•CEN MYRRHA Project Team

Phased implementation plan MYRRHA Project (2018-2030)

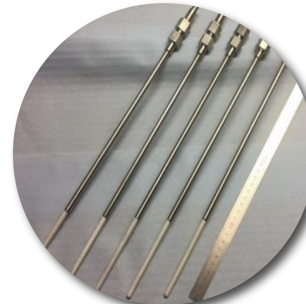
> Implementation High-Level overview



Source: SCK•CEN MYRRHA Project Team

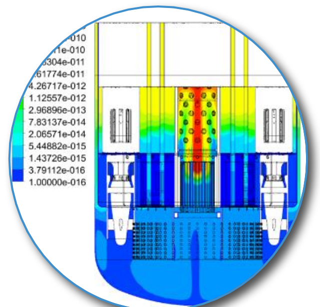
LBE coolant chemistry R&D for MYRRHA

Pb-O



Oxygen sensor

Corrosion products



Chemistry code

Oxygen control

Radio-nuclides

Purification

Outline

- MYRRHA
- **Pb-O Chemistry**
- Corrosion products chemistry
- Oxygen monitoring and control
- Coolant purification
- Po chemistry

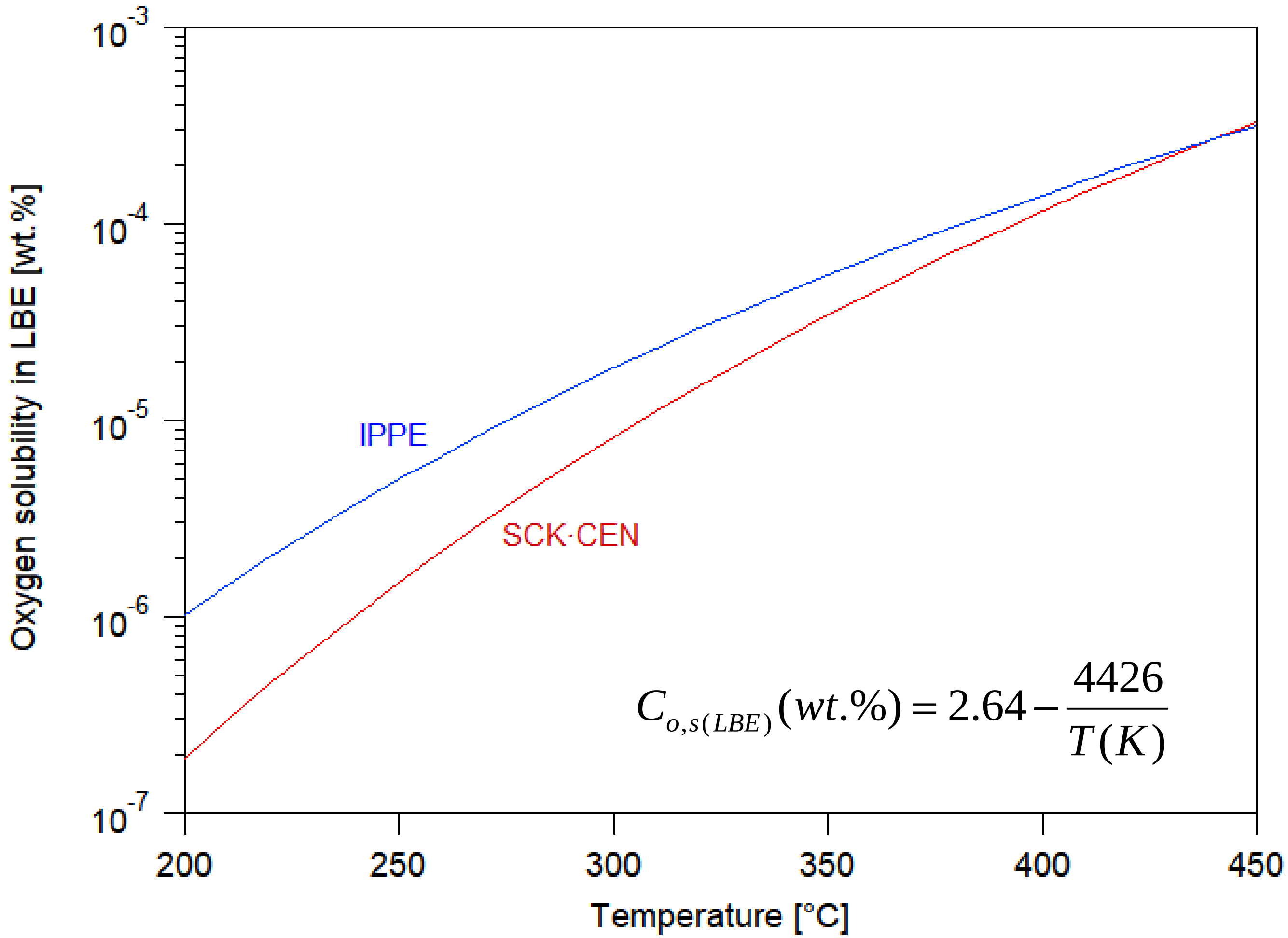
Pb (LBE)-O chemistry

- Oxygen solubility
- PbO metastable limit (supersolubility)
- Influence of suspended PbO (nano)particles on hydraulics

Source: [TBD]

Oxygen Solubility

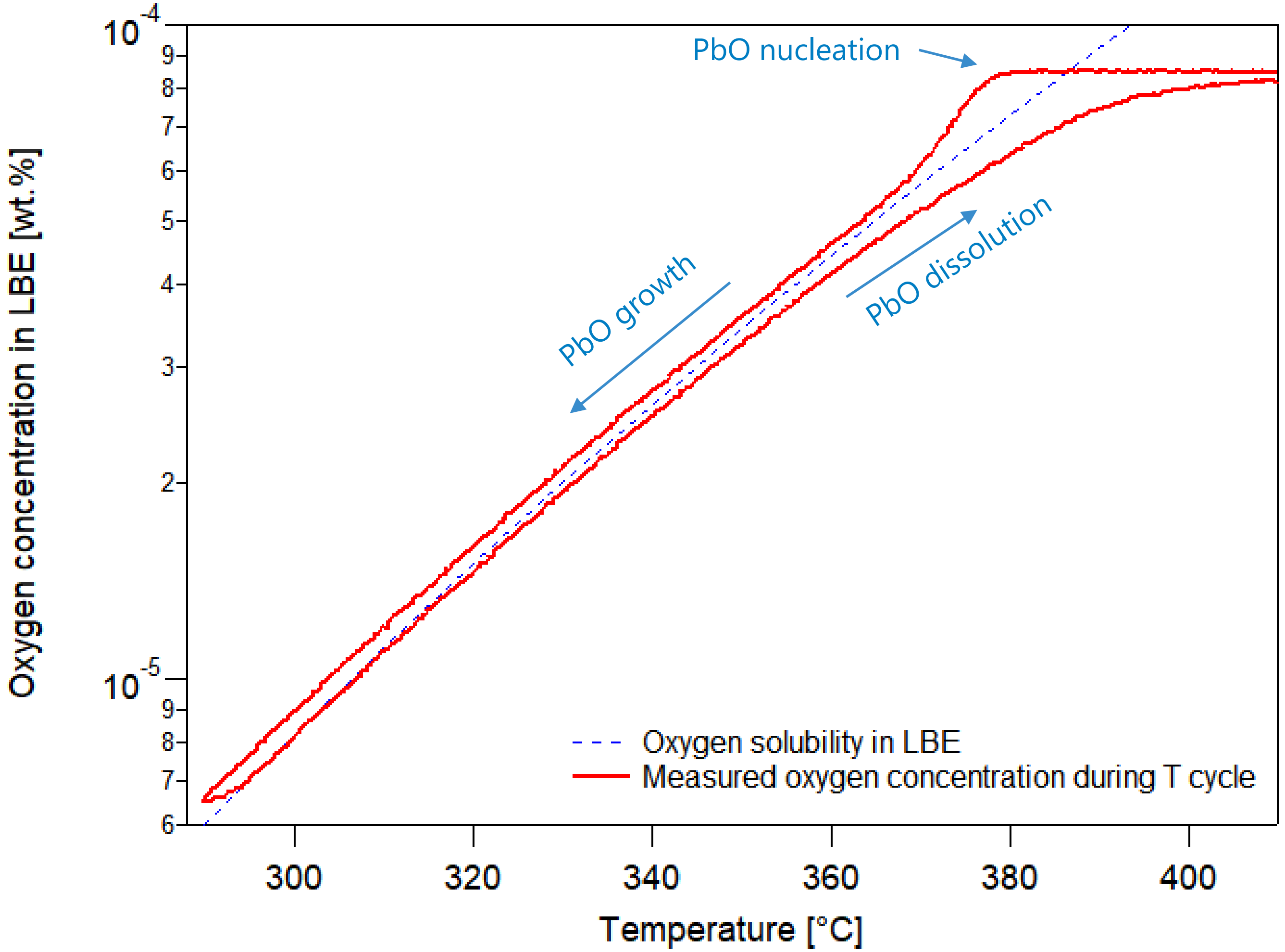
- Oxygen concentration in equilibrium of Pb and PbO in LBE



Source: [J. Lim et al, J. of the electrochemical society, 2017, 164, H743]

PbO metastable limit (supersolubility)

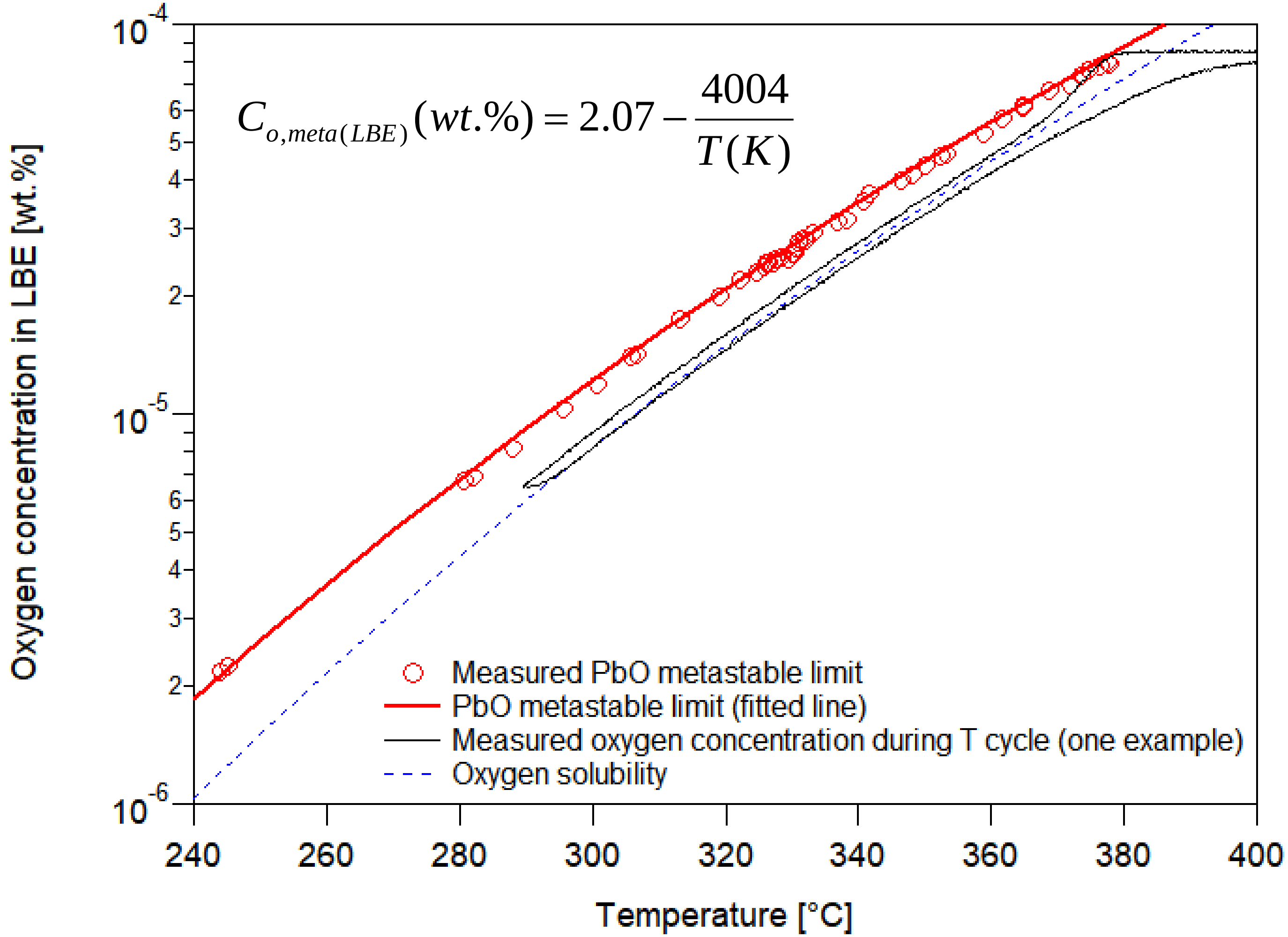
- Oversaturation required for the nucleation of PbO in LBE



Source: [K. Glaninez, et. al, Phys. Chem. Chem. Phys., 2017, 19, 27593]

PbO metastable limit (supersolubility)

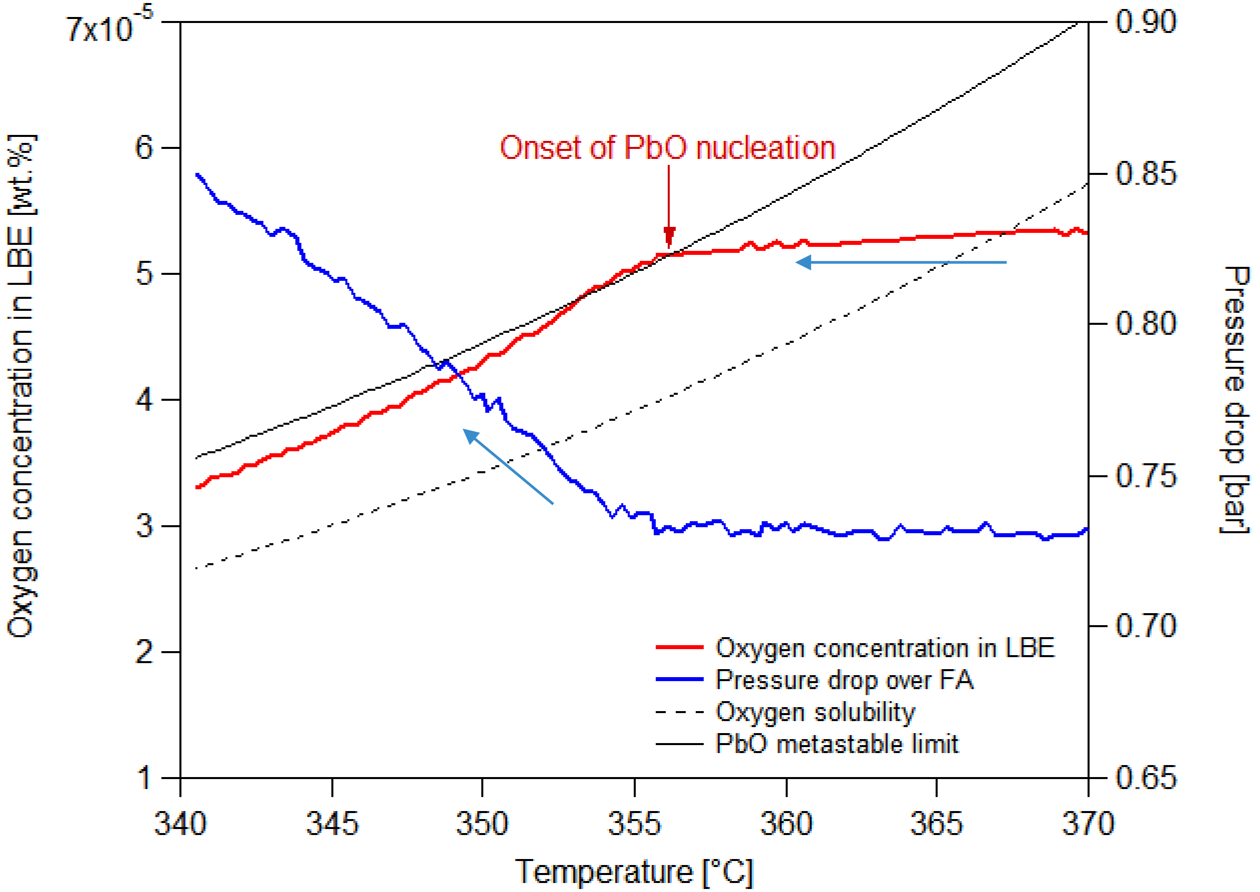
- Measured by thermal cycle test

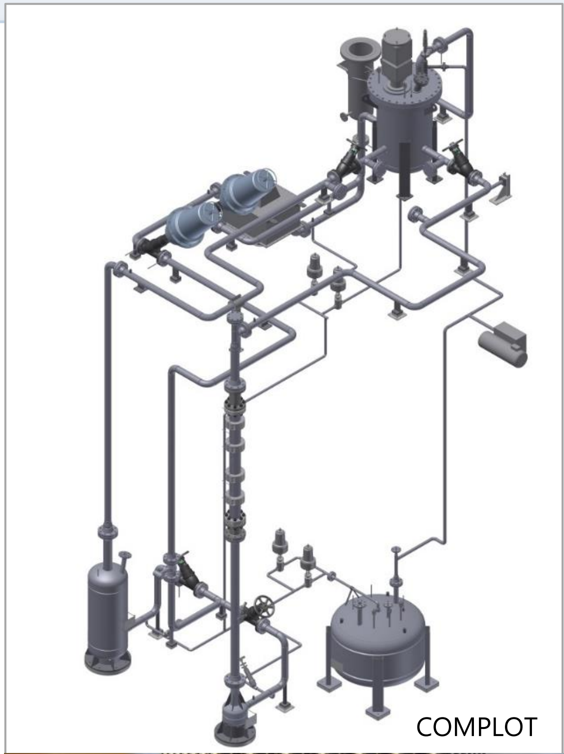


Source: [K. Glaninez, et. al, Phys. Chem. Chem. Phys., 2017, 19, 27593]

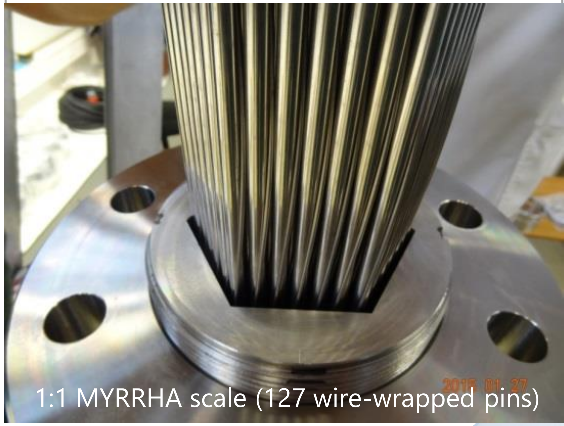
Influence of PbO on hydraulics

- Influence of suspended PbO (nano)particles on fuel assembly pressure loss





COMPLOT



1:1 MYRRHA scale (127 wire-wrapped pins)

Source: [Experimental results in COMPLOT at SCK•CEN, 2018]

Outline

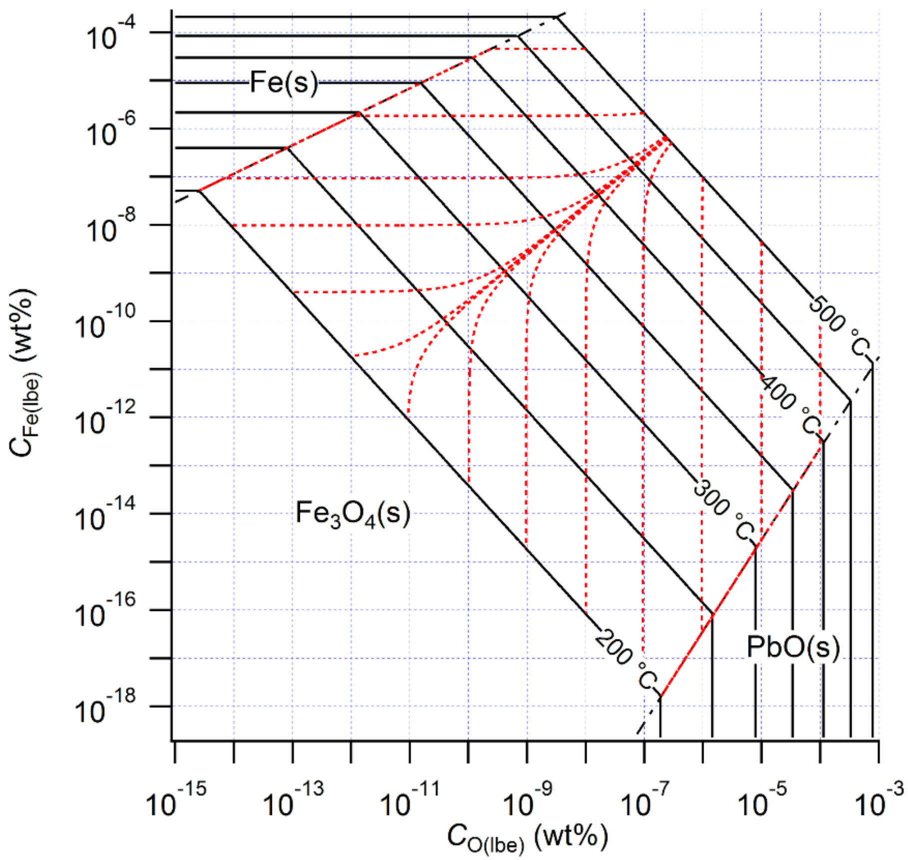
- MYRRHA
- Pb-O chemistry
- **Corrosion products chemistry**
- Oxygen monitoring and control
- Coolant purification
- Po chemistry

Corrosion Products Chemistry

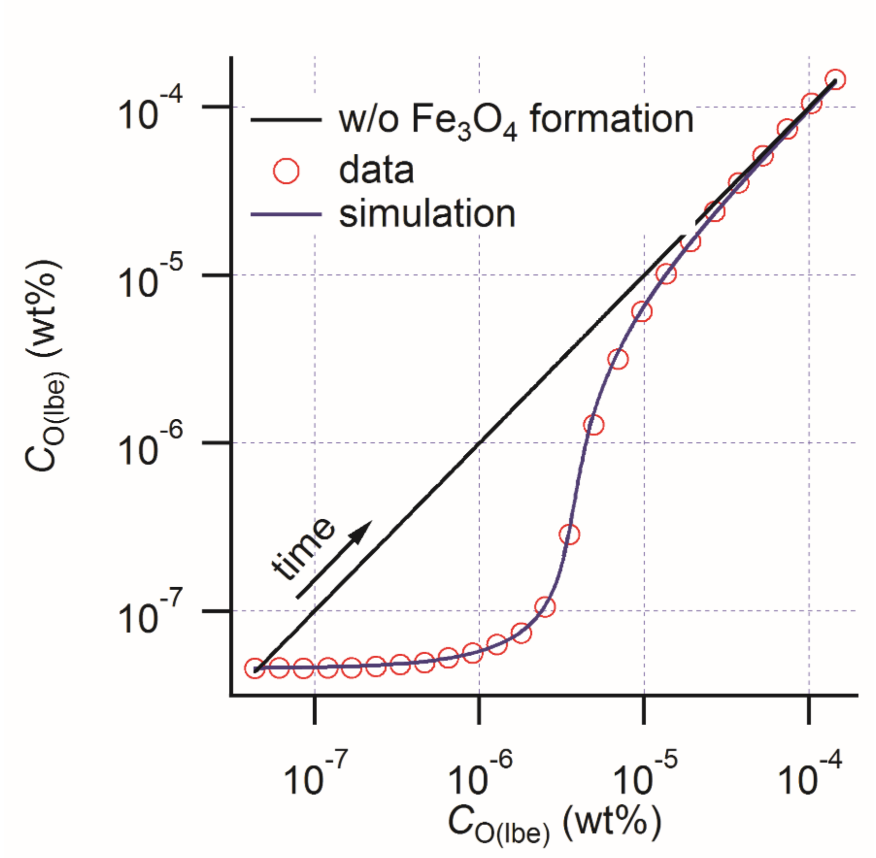
- Fe, Cr, Ni and its oxide are major corrosion products
- Fe-O interaction
- Ni-Fe-O interaction

Fe-O interaction in LBE

- Fe(lbe)-O(lbe) equilibrium diagram



- Dissolved oxygen change during oxygen addition by EOP

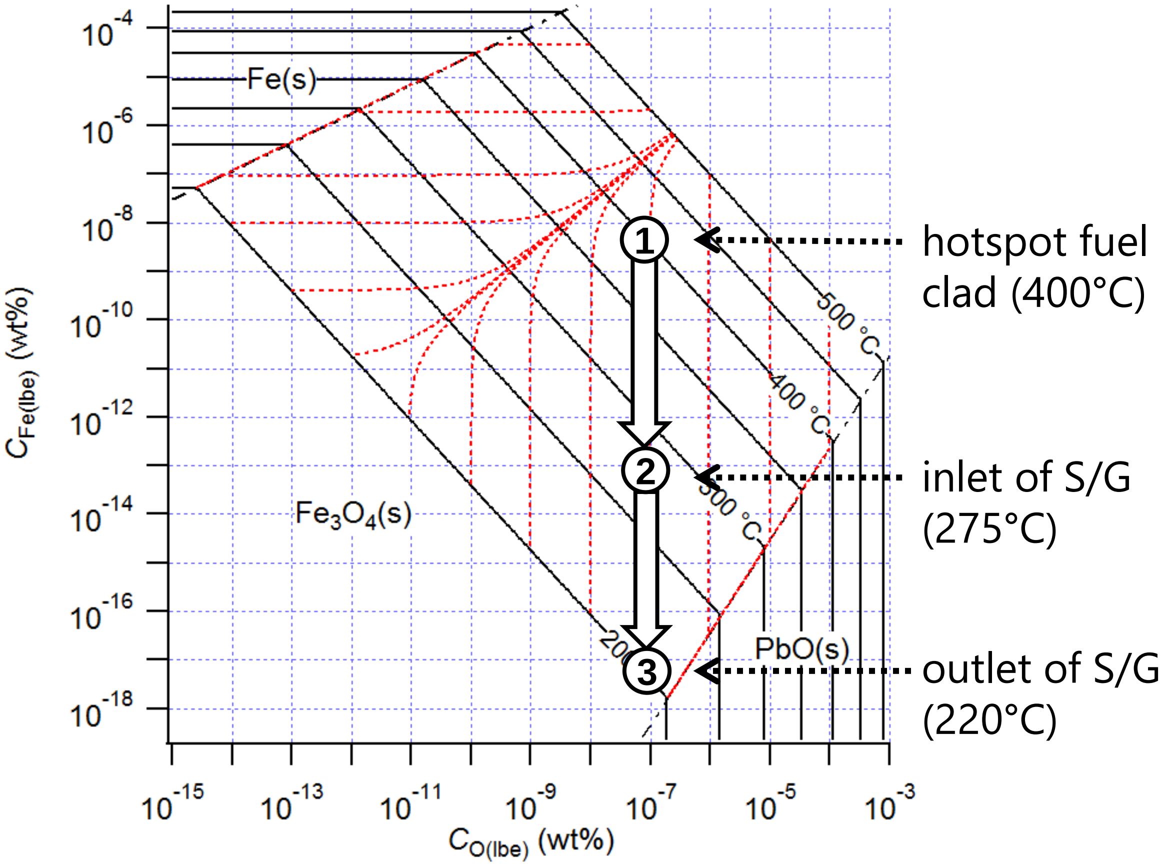


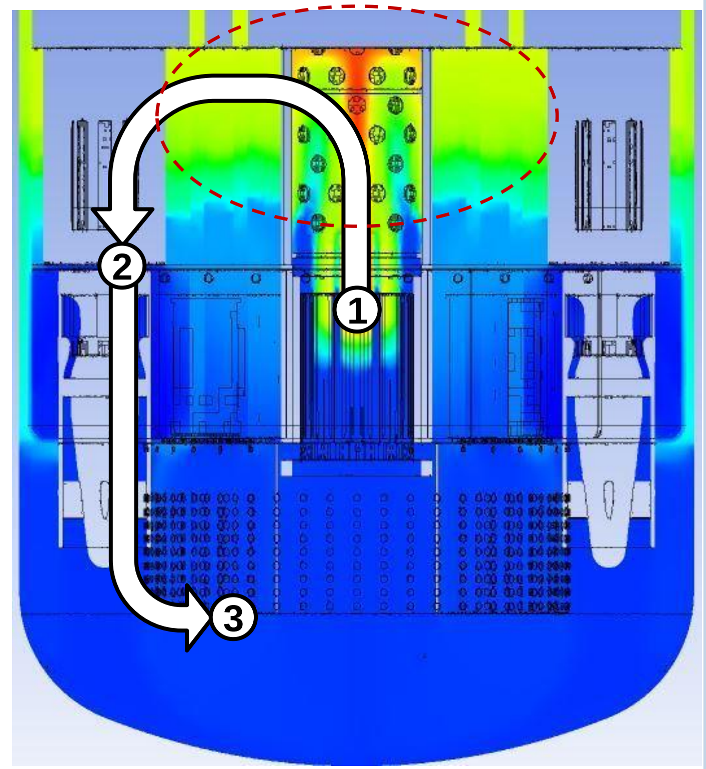
Source: A. Aerts, et. al, Phys. Chem. Chem. Phys., 2016, 18, 19526

Fe-O interaction in primary coolant of MYRRHA

- Corrosion: Fe release
- Oxidation in the coolant to magnetite
- Expected formation in MYRRHA:

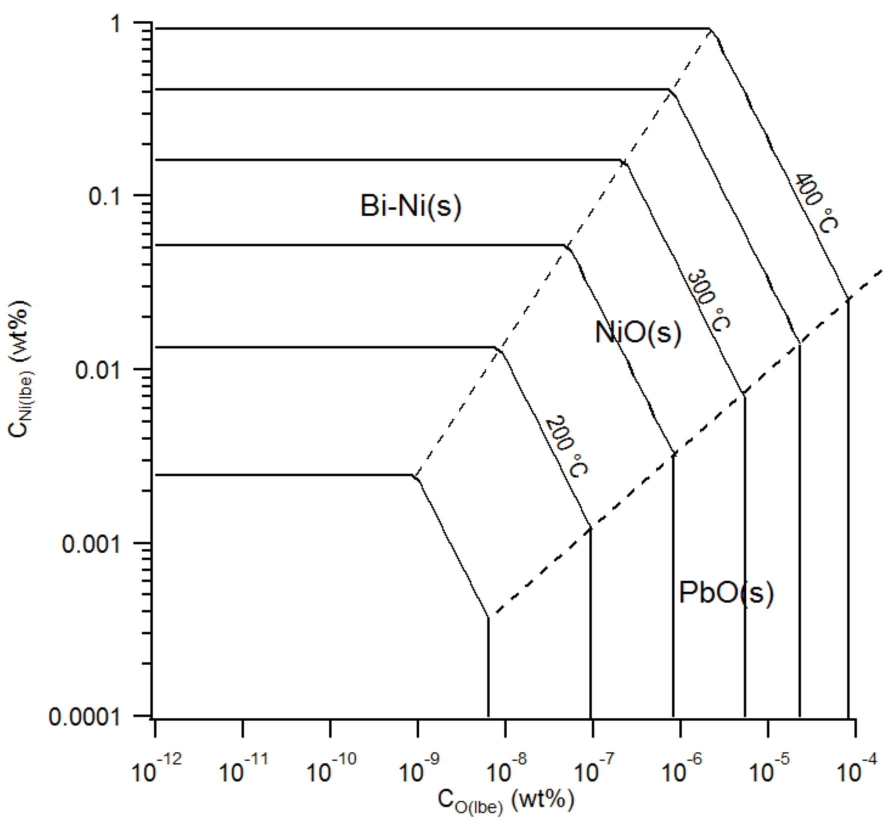
Nucleation and growth of Fe_3O_4 is likely to occur in upper plenum, and not in S/G.



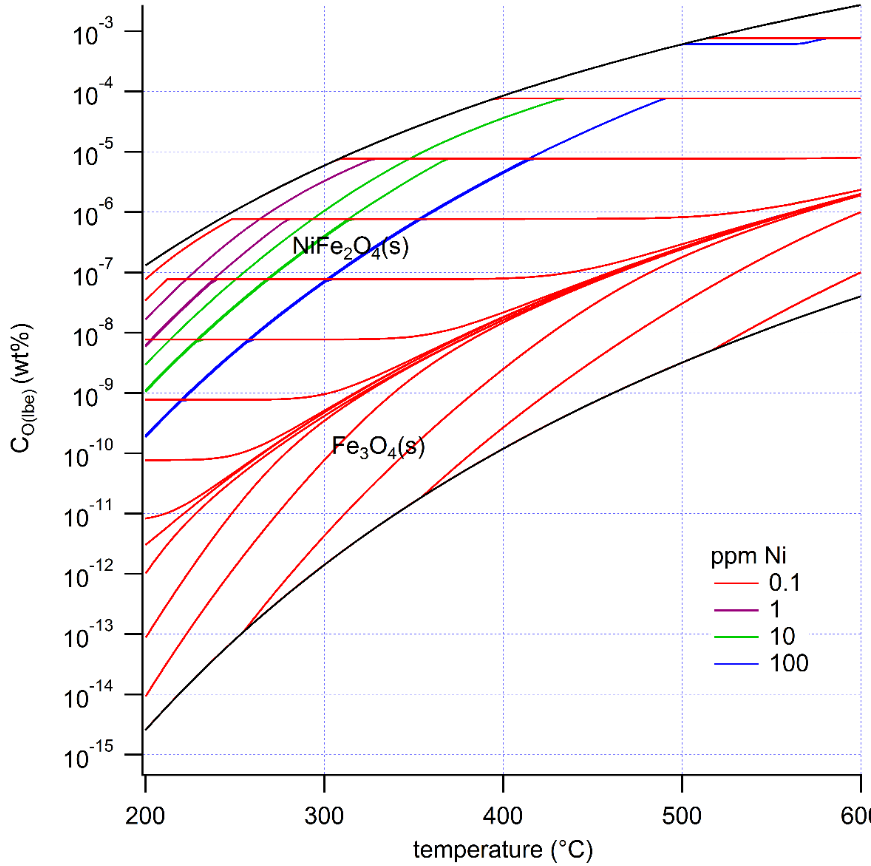


Fe-Ni-O interaction in LBE

- Ni(lbe)-O(lbe) equilibrium diagram



- Fe(lbe)-Ni(lbe)-O(lbe) equilibrium diagram

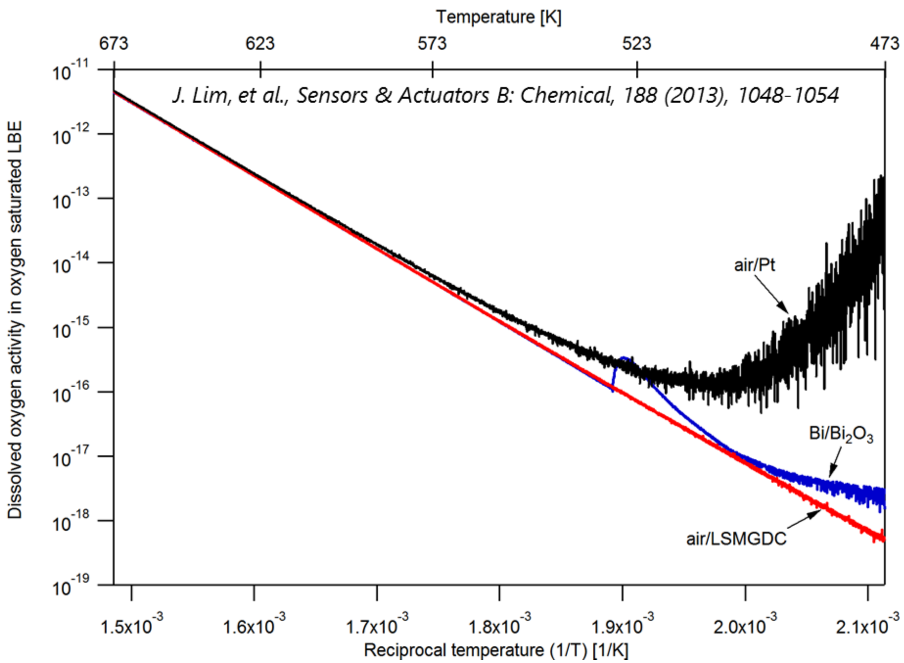


Outline

- MYRRHA
- Pb-O Chemistry
- Corrosion products chemistry
- **Oxygen monitoring and control**
- Coolant purification
- Po chemistry

Oxygen sensors @SCK·CEN

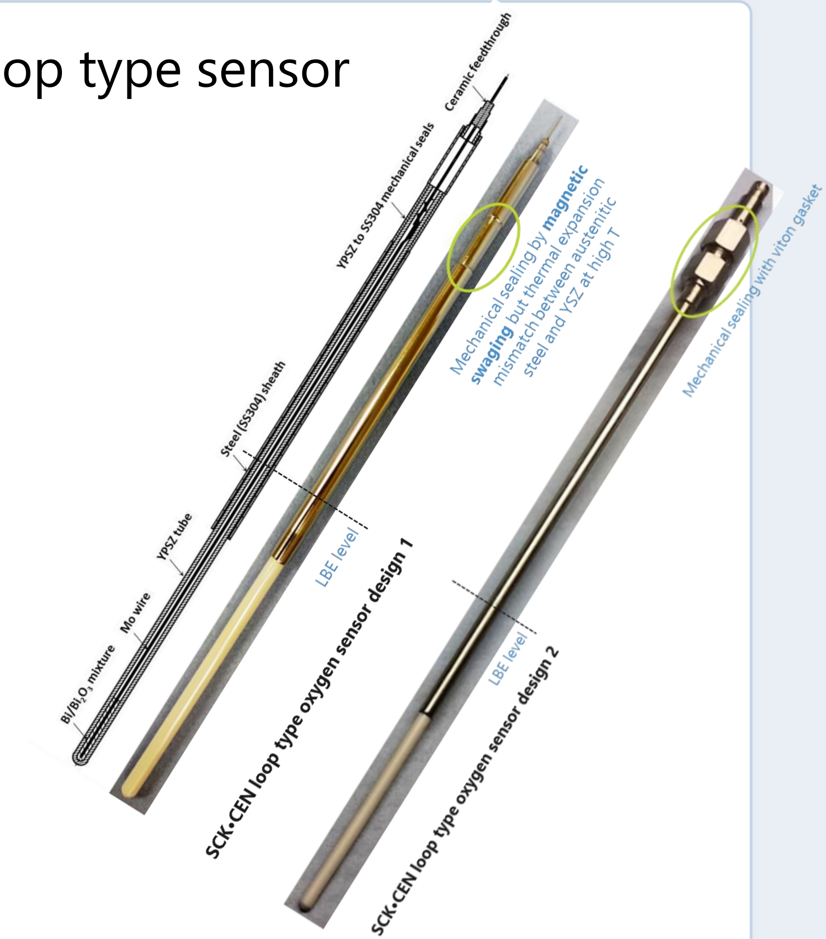
- Minimum operating temperature : reference electrode



- Pool type sensor

LBE level

- Loop type sensor



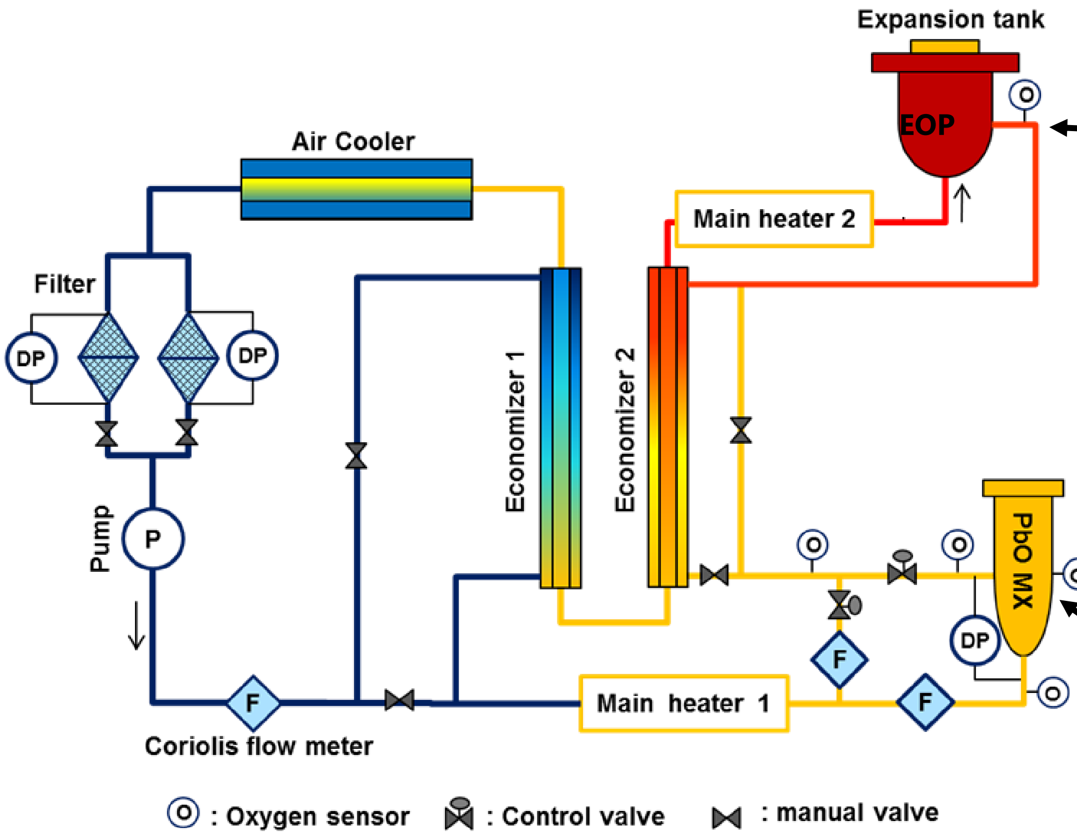
Ceramic to metal joint
by mechanical sealing using graphite gasket

Oxygen control : Sink and Source

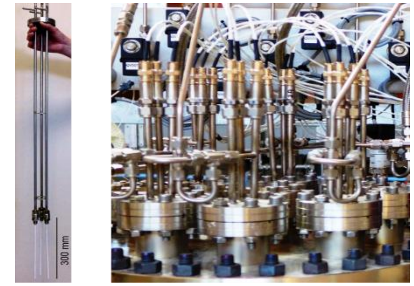
	technology	reactions	SCK-CEN test facility
	Gas control	$O(lbe) + H_2(g) \rightarrow H_2O(g)$ $O_2(g) \rightarrow 2O\ (lbe)$ $H_2O \rightarrow O\ (lbe) + H_2(g)$	Helios 3 pool (bubbling) CRAFT corrosion loop (cover gas)
	PbO mass exchanger (PbO MX)	$PbO(s) \rightarrow Pb(lbe) + O(lbe)$	MEXICO chemistry loop
	Electrochemical oxygen pump (EOP)	$2O^{2-}(lbe) \rightarrow O_2\ (g) + 4e^-$ $O_2\ (g) + 4e^- \rightarrow 2O^{2-}(lbe)$	MEXICO chemistry loop
	Cold trap	$Pb(lbe) + O(lbe) \rightarrow PbO(s)$	LiLIPuTTeR loop

Oxygen control : MEXICO LBE chemistry loop

- 7 tons of LBE

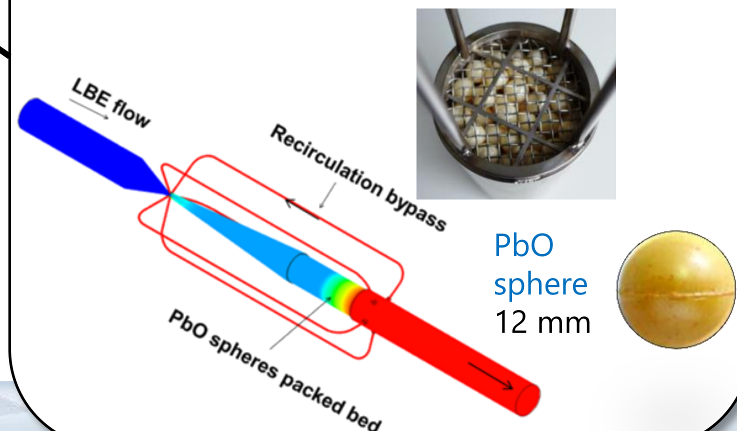


Electrochemical oxygen pump (EOP)

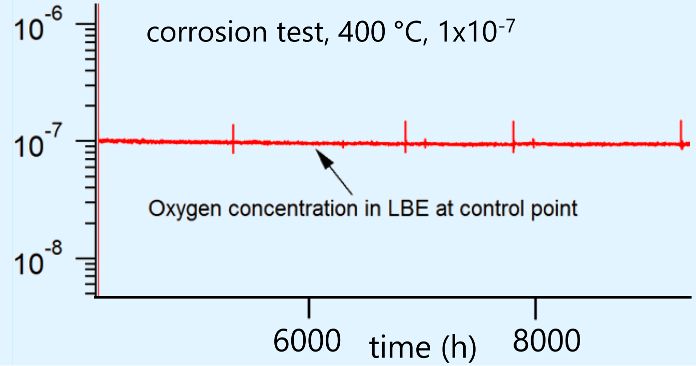
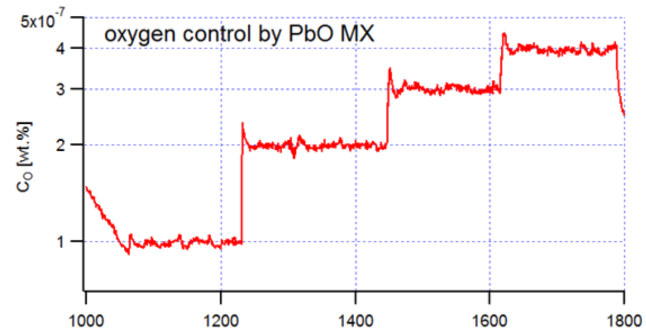
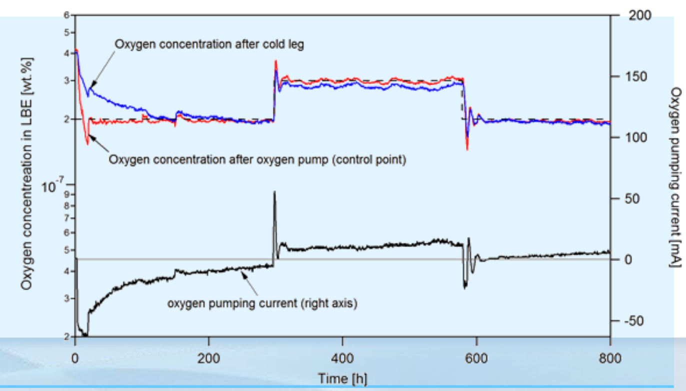


5 modules, 15 pumps in operation

PbO mass exchanger (PbO MX)



Oxygen control

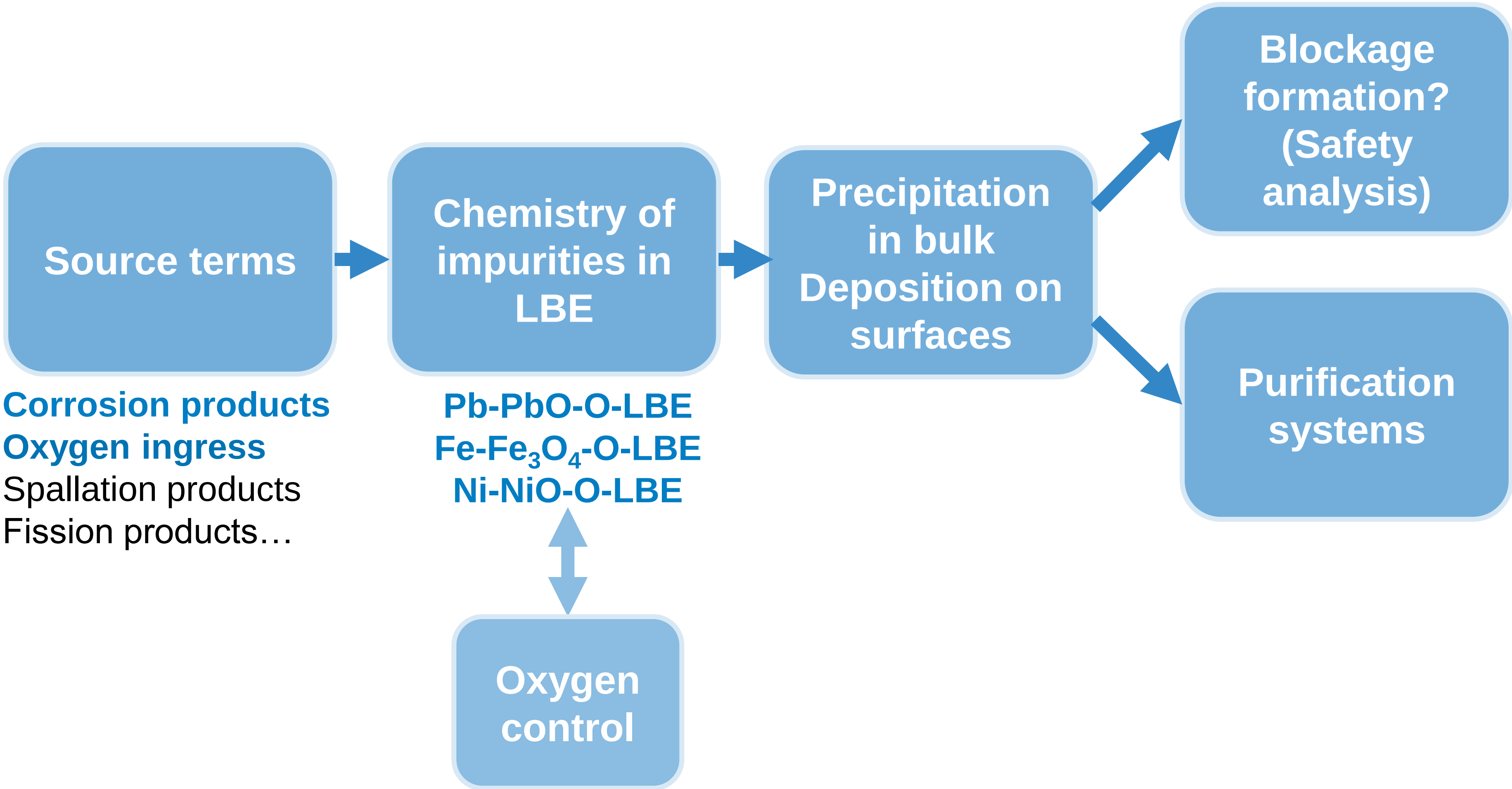
	technology	SCK-CEN test facility	example
	Gas control	Helios 3 pool (bubbling) CRAFT corrosion loop (cover gas)	
	PbO mass exchanger (PbO MX)	MEXICO chemistry loop	
	Electrochemical oxygen pump (EOP)	MEXICO chemistry loop	

Source: J. Lim et al, J. Nucl. Sci. Technol., 54 (2017), 131-137

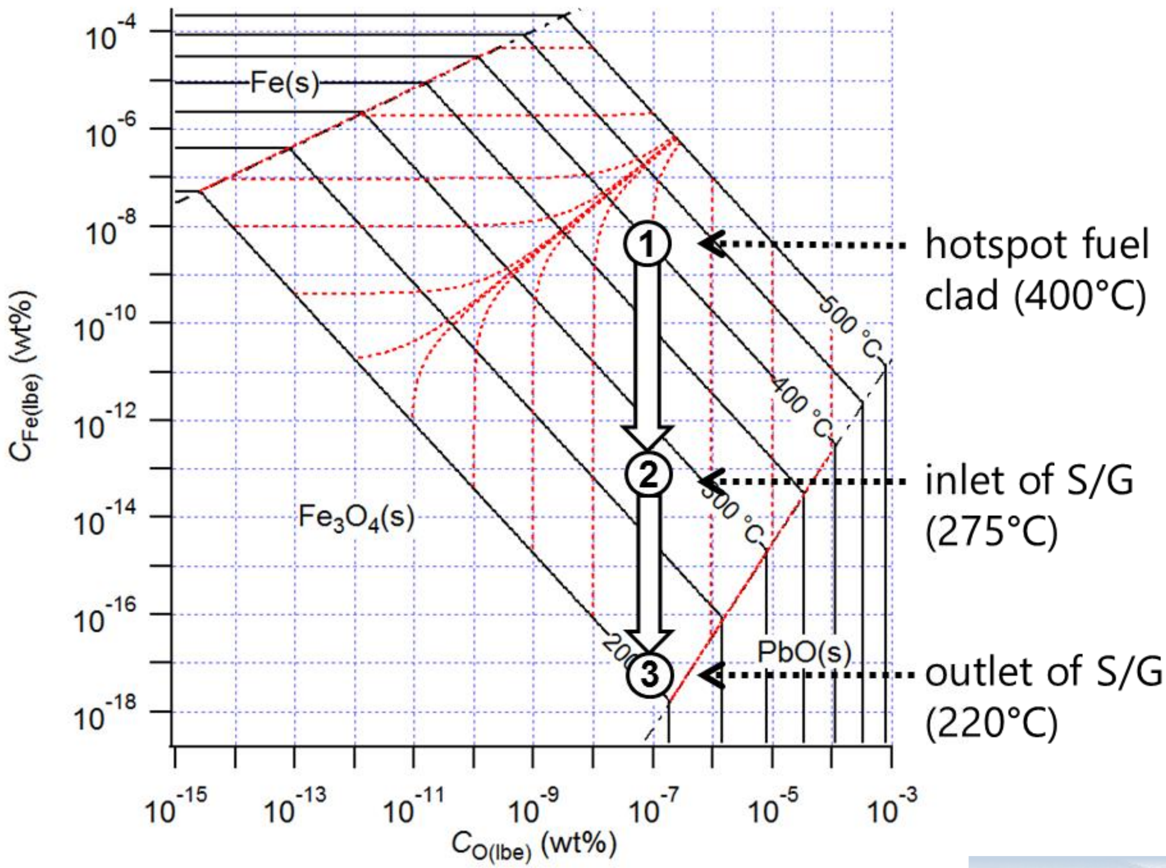
Outline

- MYRRHA
- Pb-O Chemistry
- Corrosion products chemistry
- Oxygen monitoring and control
- **Coolant purification**
- Po chemistry

Coolant purification



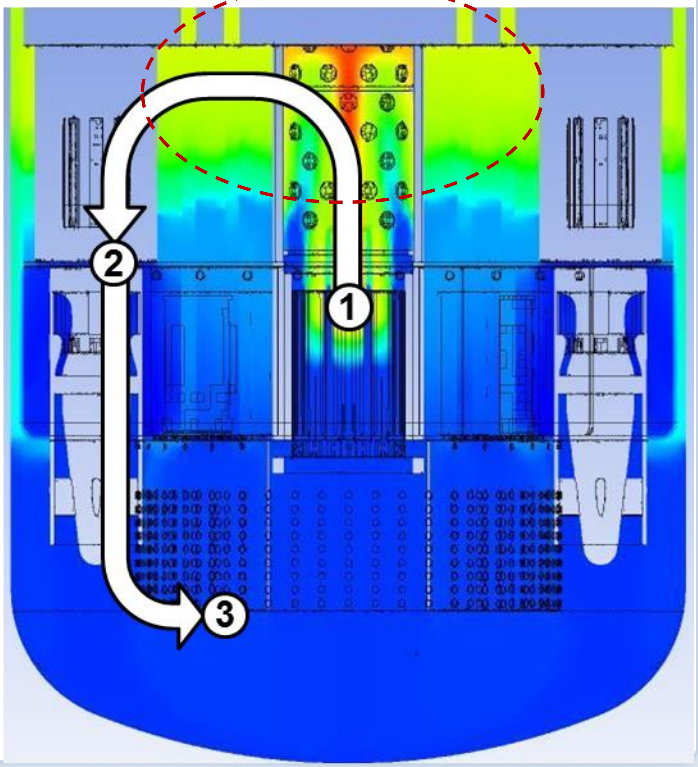
Coolant purification : iron/magnetite



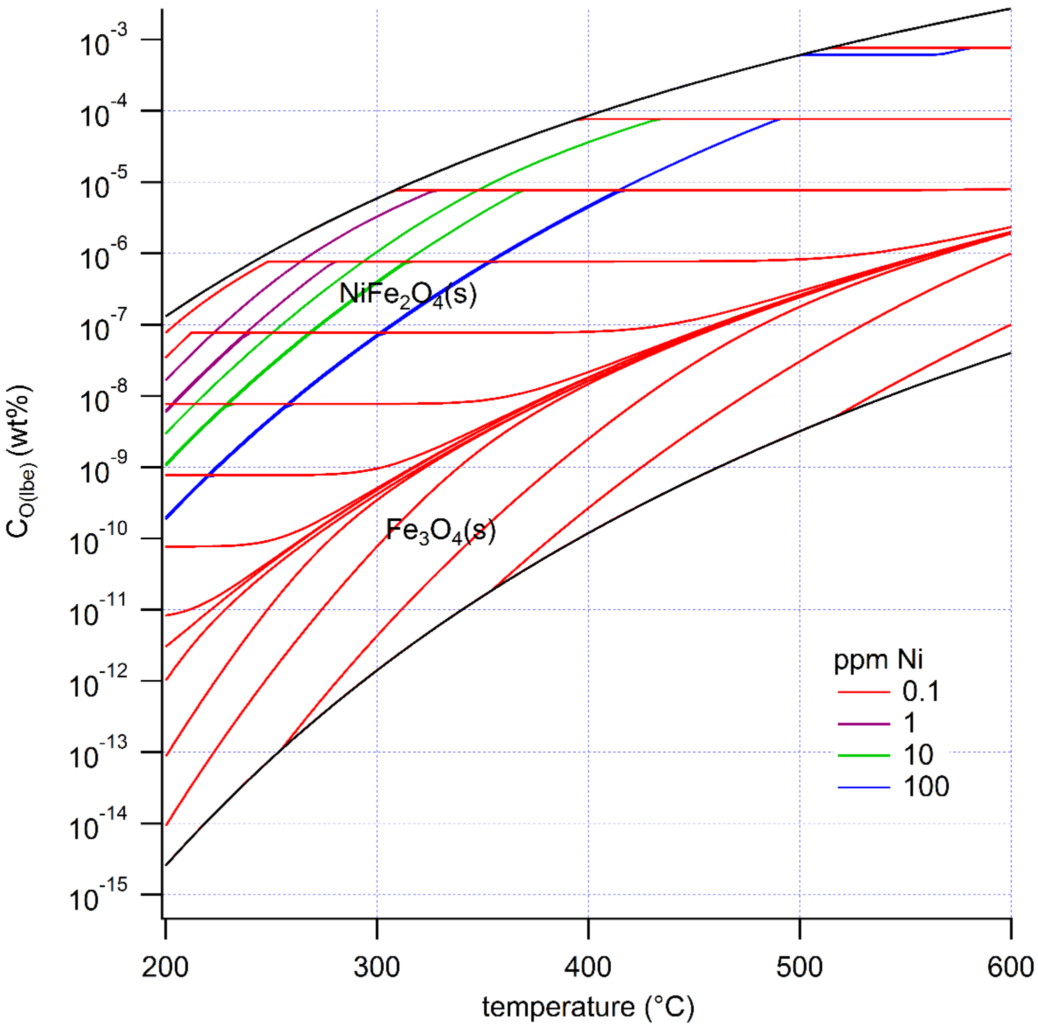
The nucleation and growth of Fe_3O_4 is most likely to occur in upper plenum.

→ Modeling of Fe_3O_4 particle transport

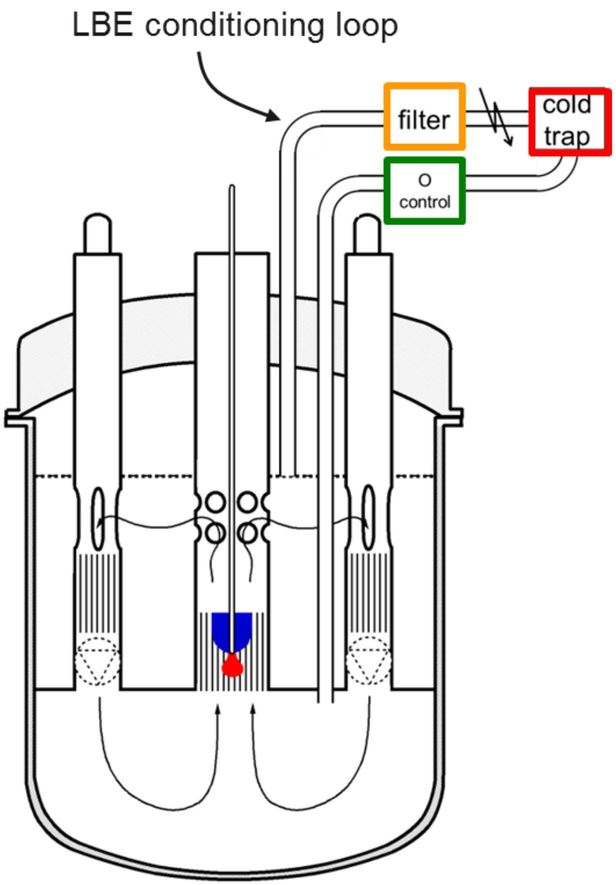
→Cleaning free surface



Chemistry of corrosion product : Ni and its oxides



Trapping by precipitating Fe_2NiO_4 in **cold trap**



Outline

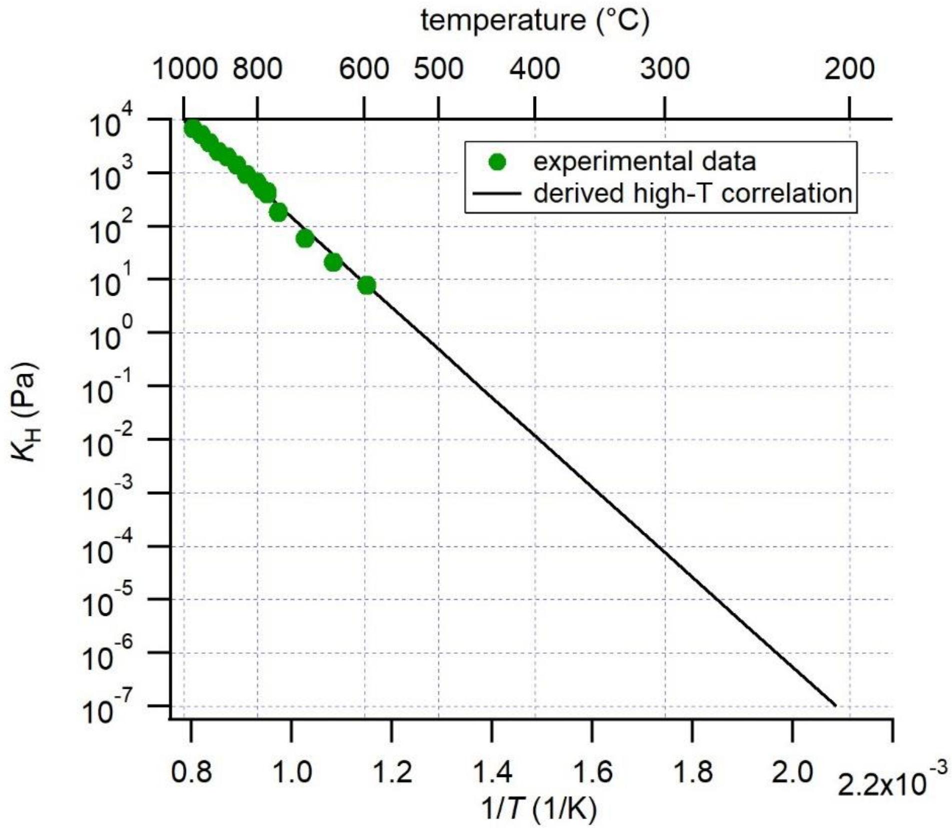
- MYRRHA
- Pb-O Chemistry
- Corrosion products chemistry
- Oxygen monitoring and control
- Coolant purification
- **Po Chemistry**

Po evaporation from LBE

- High-temperature Po evaporation
- Normal operation temperature Po evaporation
  - surface effects
  - carrier gas composition

High-temperature Po evaporation

- Po evaporation above 600 °C under Ar, Ar/5%H_2 and Ar/2%H_2O

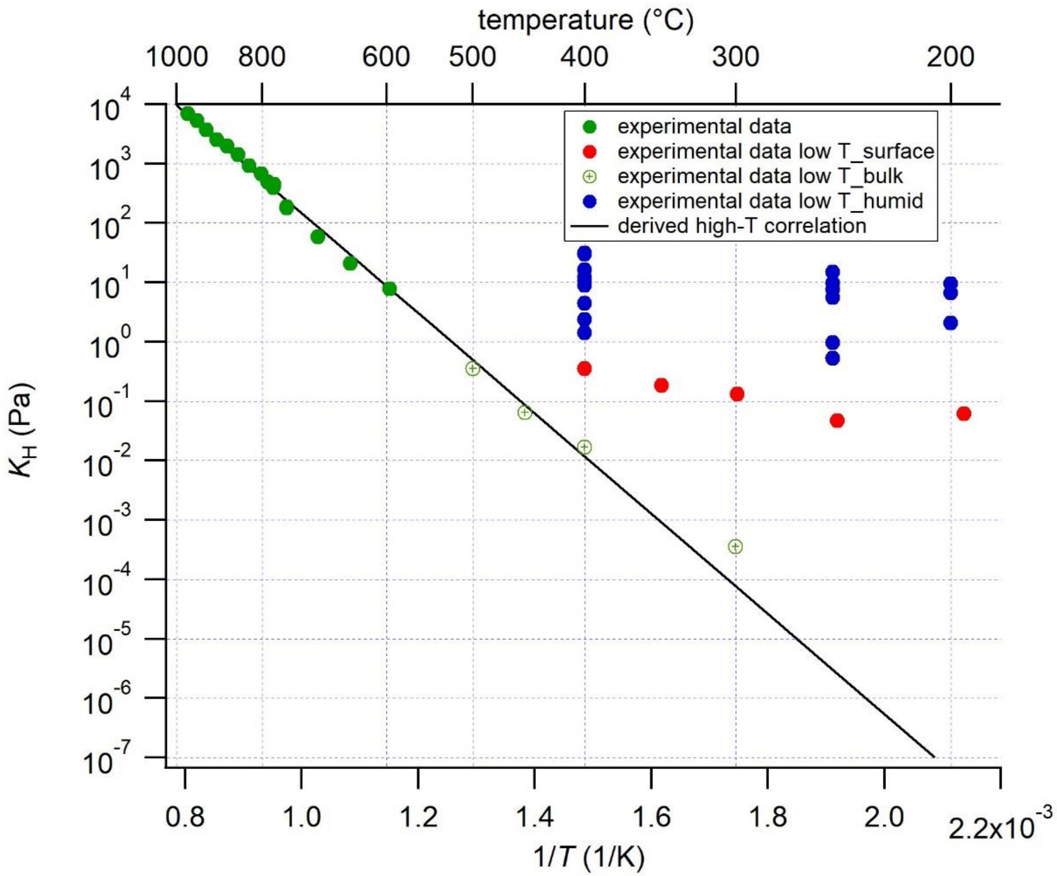


- Not sensitive to gaseous atmosphere composition
- No oxide layer on surface LBE sample (high oxygen solubility)

Source: *[B.G. Prieto, et al. Radiochimica Acta, 2017, 102, 1083–1091]*

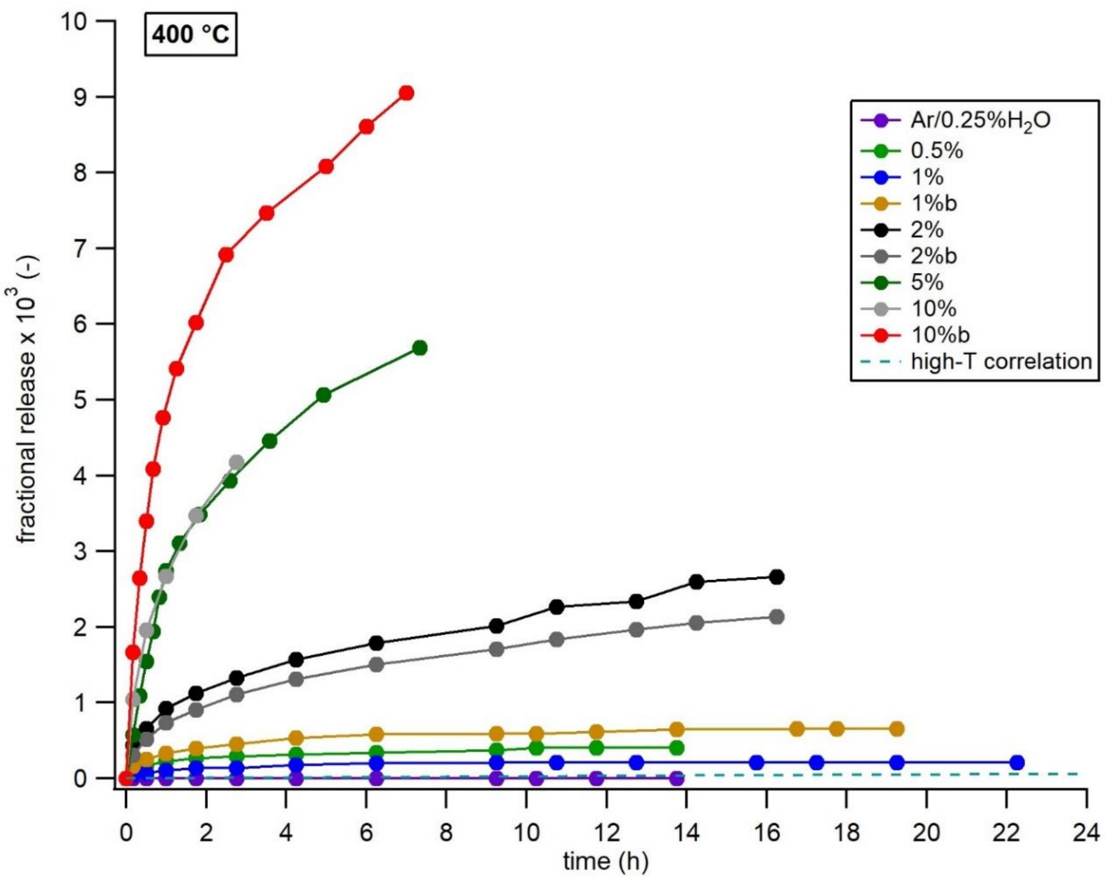
Normal operation temperature Po evaporation: carrier gas composition

- Po evaporation < 600 °C under Ar/x%H_2O with x = [0,…,10]
- Enhanced Po and volatility of released Po species in humid atmospheres



Normal operation temperature Po evaporation: carrier gas composition

- Po evaporation < 600 °C under $Ar/x\%H_2O$ with x = [0,...,10]
- Fractional Po release increases with moisture content in the carrier gas



Source: [TBD]

Summary

- Oxygen solubility and supersolubility were measured to predict PbO nucleation, growth and transport.
- The influence of suspended PbO particles on hydraulics has been studied.
- A scientific basis was established for the behavior of impurities (corrosion products, …) in LBE, which guides the development of coolant purification systems
- Accurate and precise oxygen sensors in a wide temperature range down to 200°C were developed. Oxygen control was accomplished on both lab and pilot scale.
- It was found that Po release increases with moisture content in the carrier gas.

MYRRHA

...?

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JOINT STOCK COMPANY
"D.V. EFREMOV INSTITUTE OF ELECTROPHYSICAL APPARATUS"
(JSC «NIIEFA»)

РОСАТОМ

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Laser separation of lead isotopes by means of selective photochemical reactions for nuclear-energy facilities of the next generation

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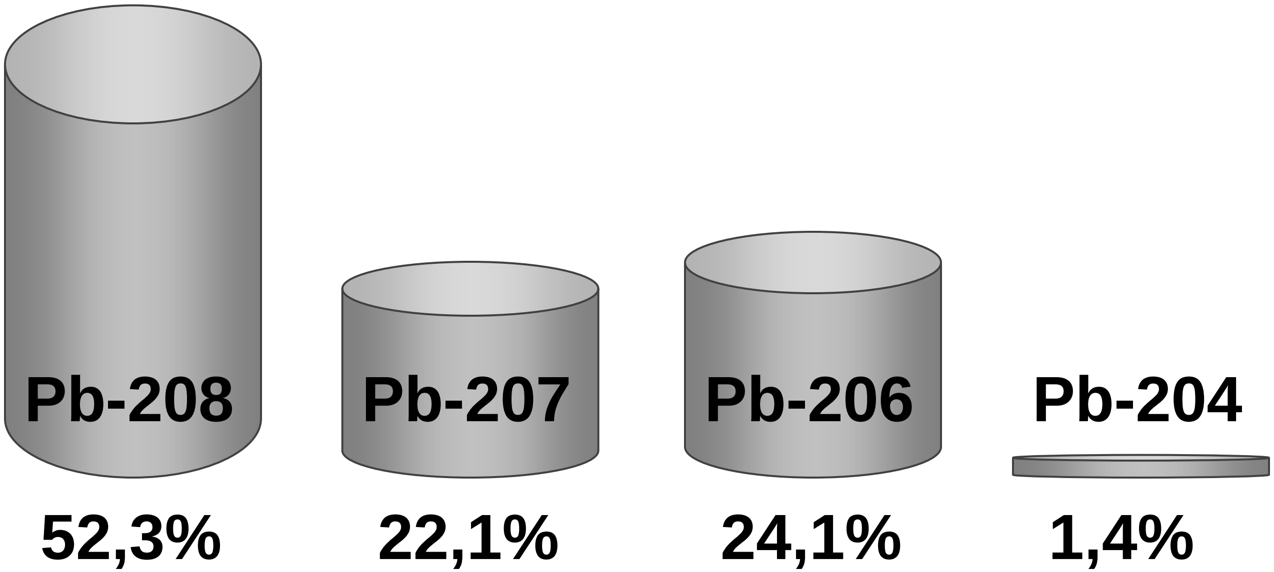
ТЖМТ – 2018
г. Обнинск

Isotopic composition

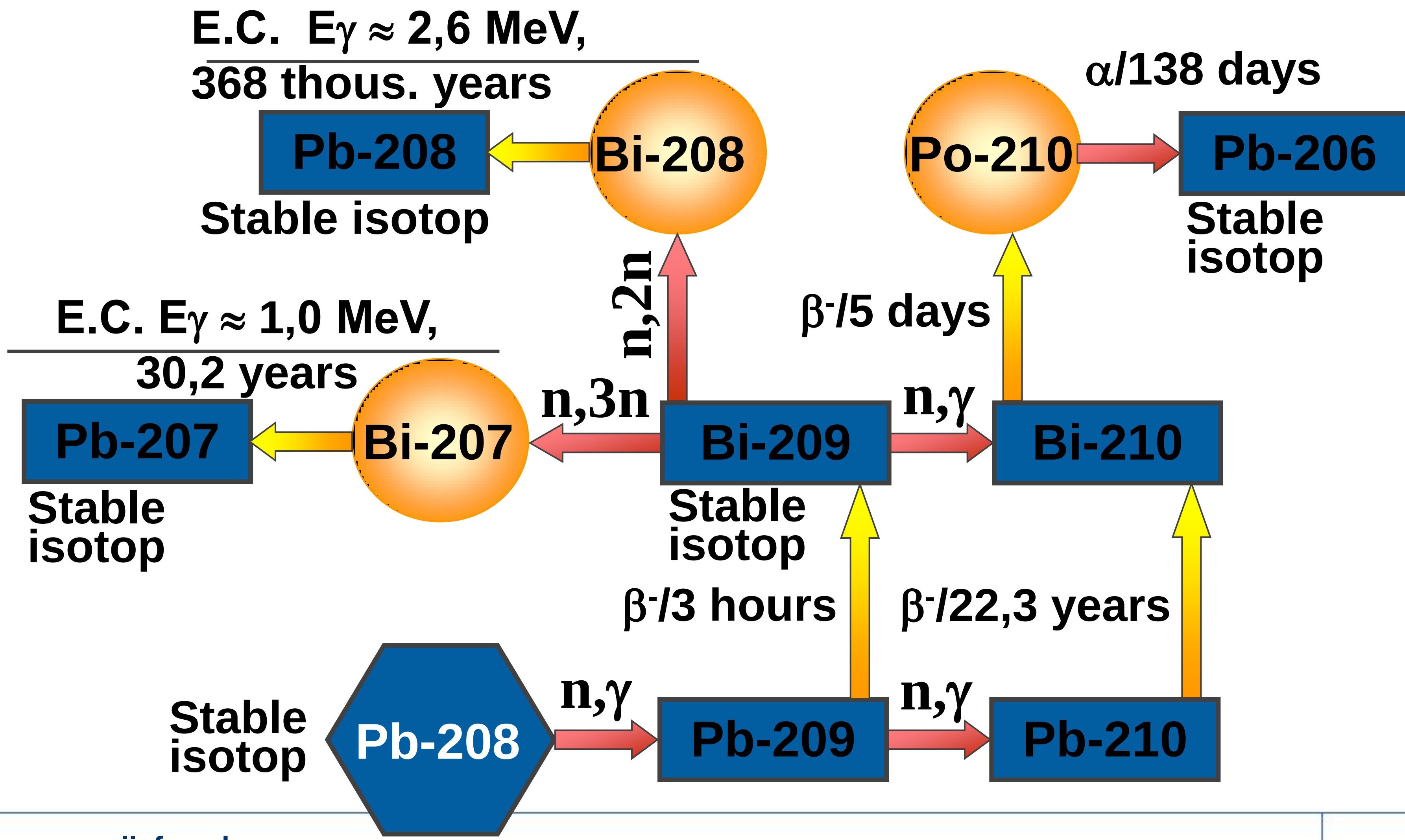
Long-time irradiation in PRF neutron spectrum results in accumulation of extremely dangerous alpha and gamma radionuclides, including polonium Po- 210 and bismuth Bi - 207. Its generation in the coolant is connected to the large contents (52.3 %) of lead isotope Pb-208 in natural lead. The computation of the applicant shows, that coolant, highly enriched with lead isotope - 206 (95%-96%), accumulates much less (in 103-104 times) radionuclides of polonium and bismuth, as against natural lead

Isotope	204	206	207	208
Natural lead	1.4%	24.1%	22.1%	52.3%
Product	**0.1%**	**95.0%**	**1.5%**	**3.4%**

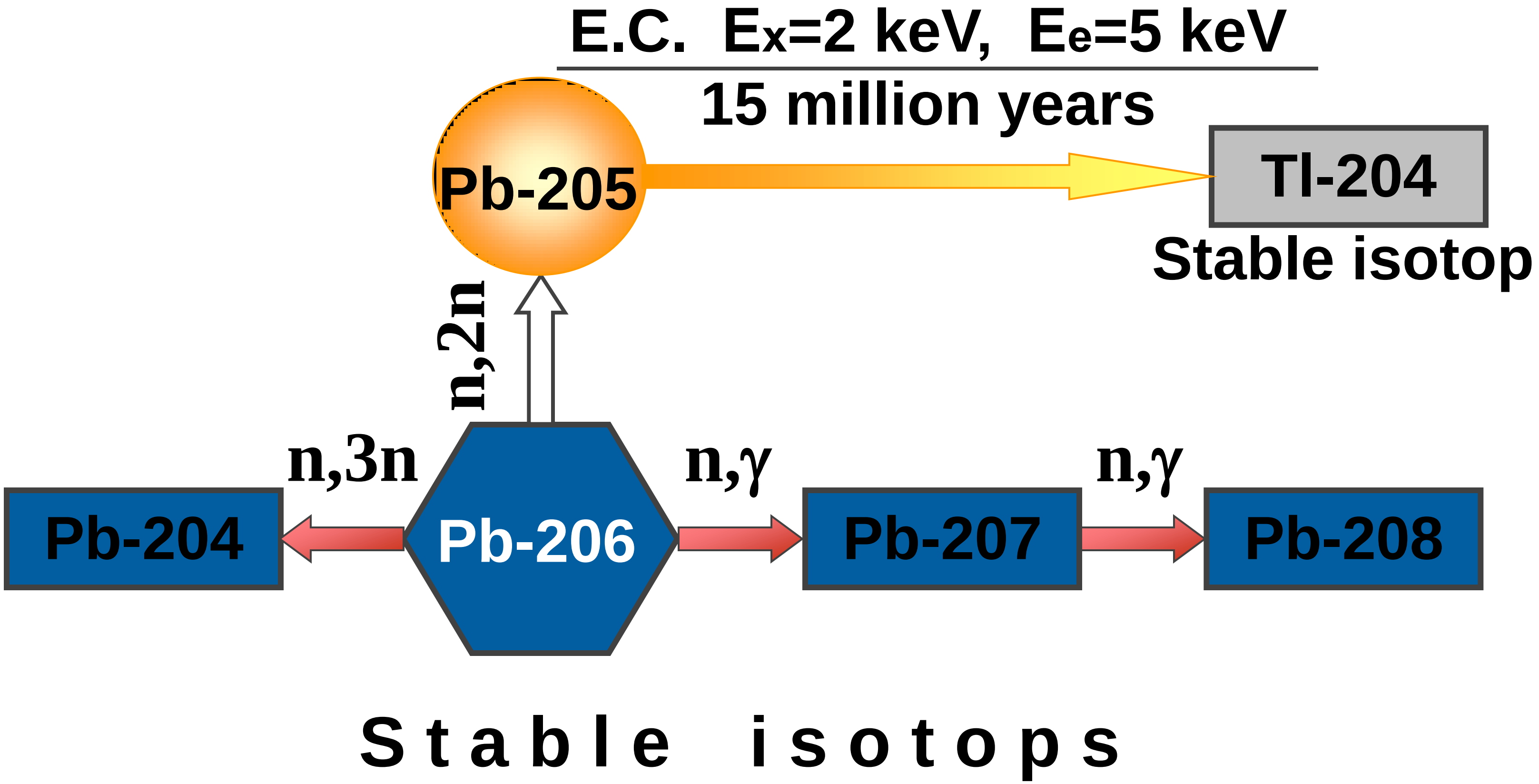
Pb-nat isotopical contents



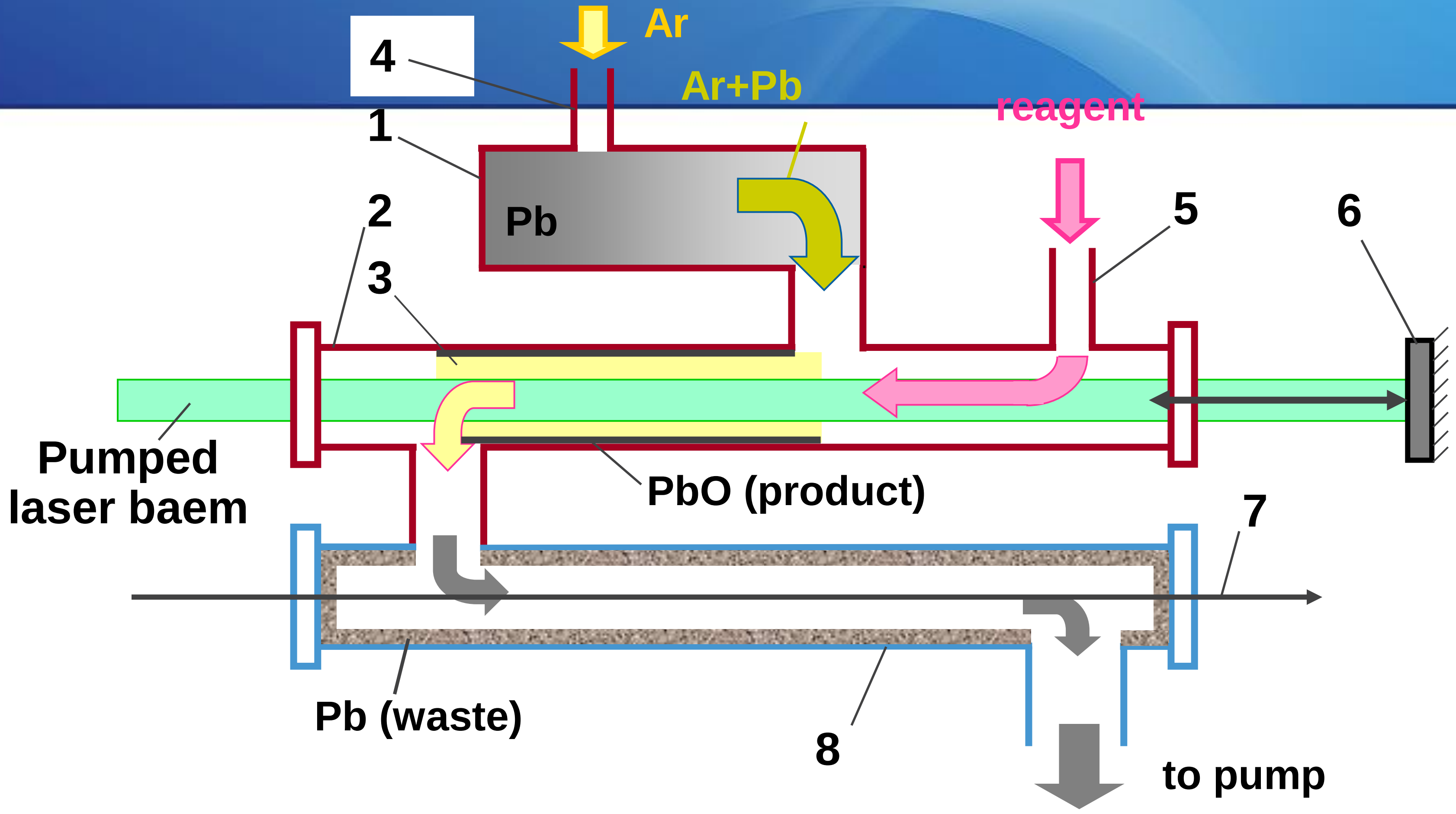
Accumulation of radionuclides Bi-207, Bi-208 and Po-210 in natural lead during its irradiation in fast reactor



Accumulation of radionuclide Pb-205 in lead isotop Pb-206 during its irradiation in fast reactor



Separation circuit of Pb isotopes



1- Pb reservoir, 2 - interaction chamber 3- interaction zone, 4 - Ar supply, 5 - reagent gas supply, 6 - reciprocal mirror, 7 - probing laser radiation, 8 - collector chamber

Optical setup scheme

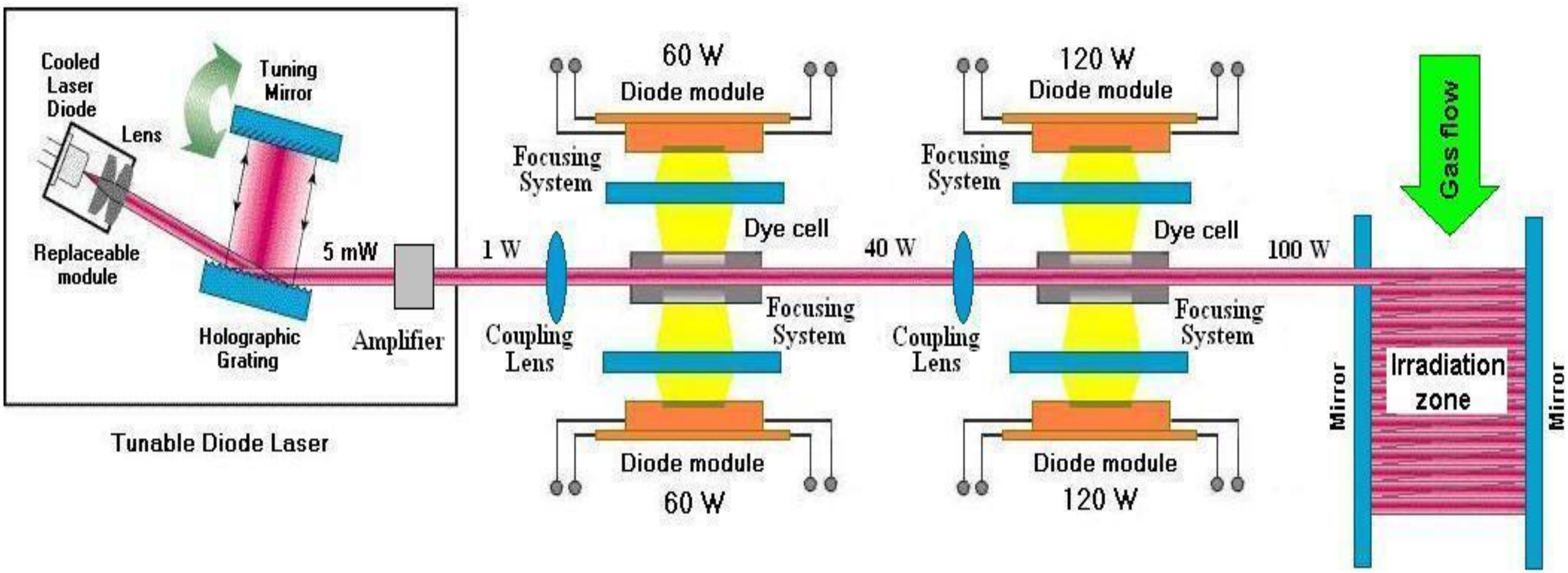
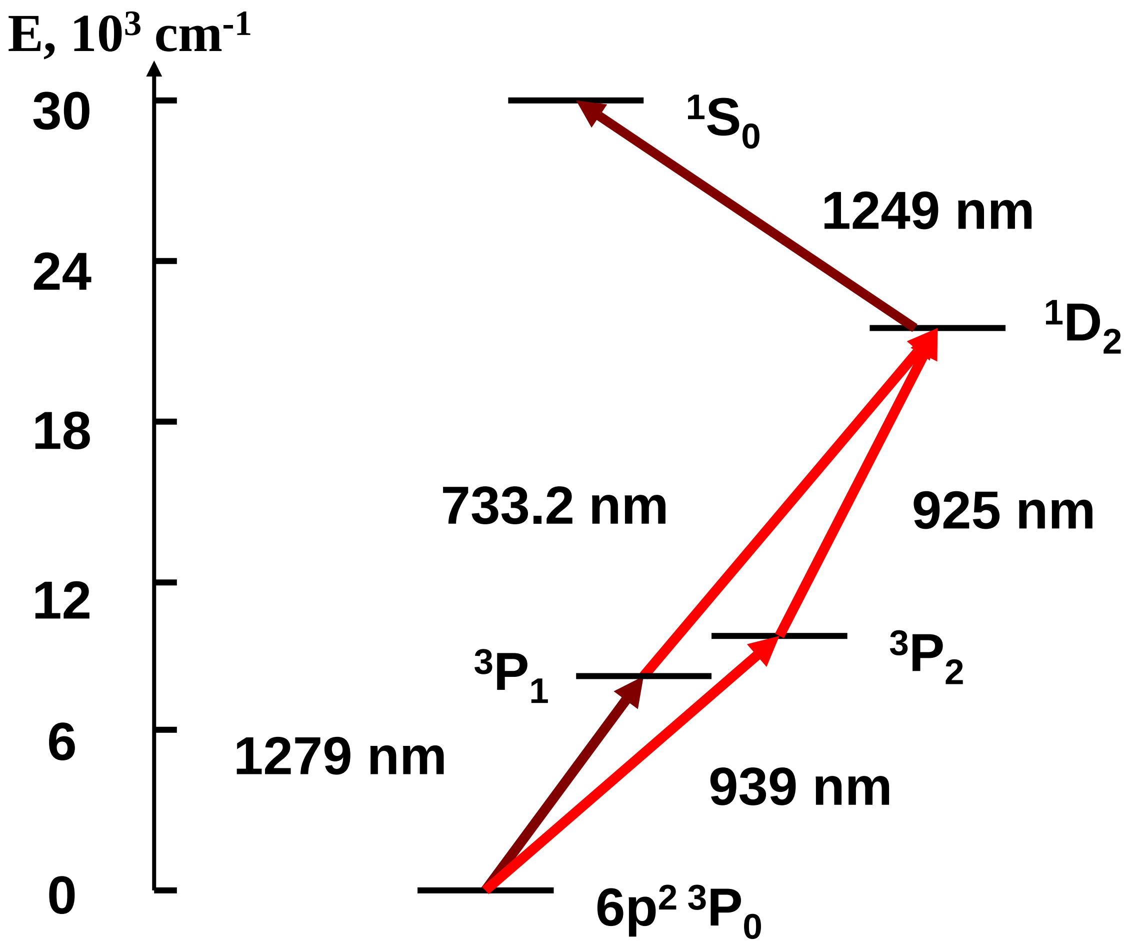
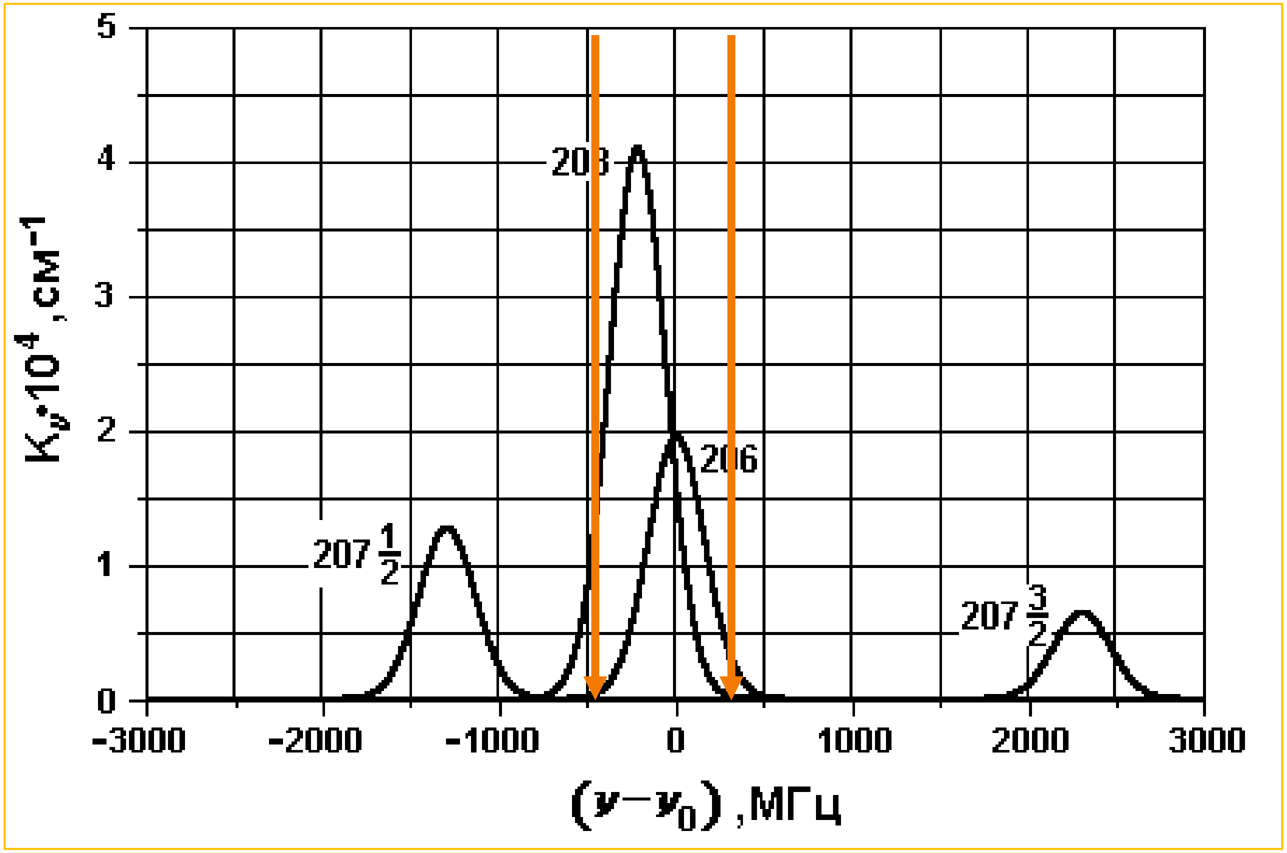


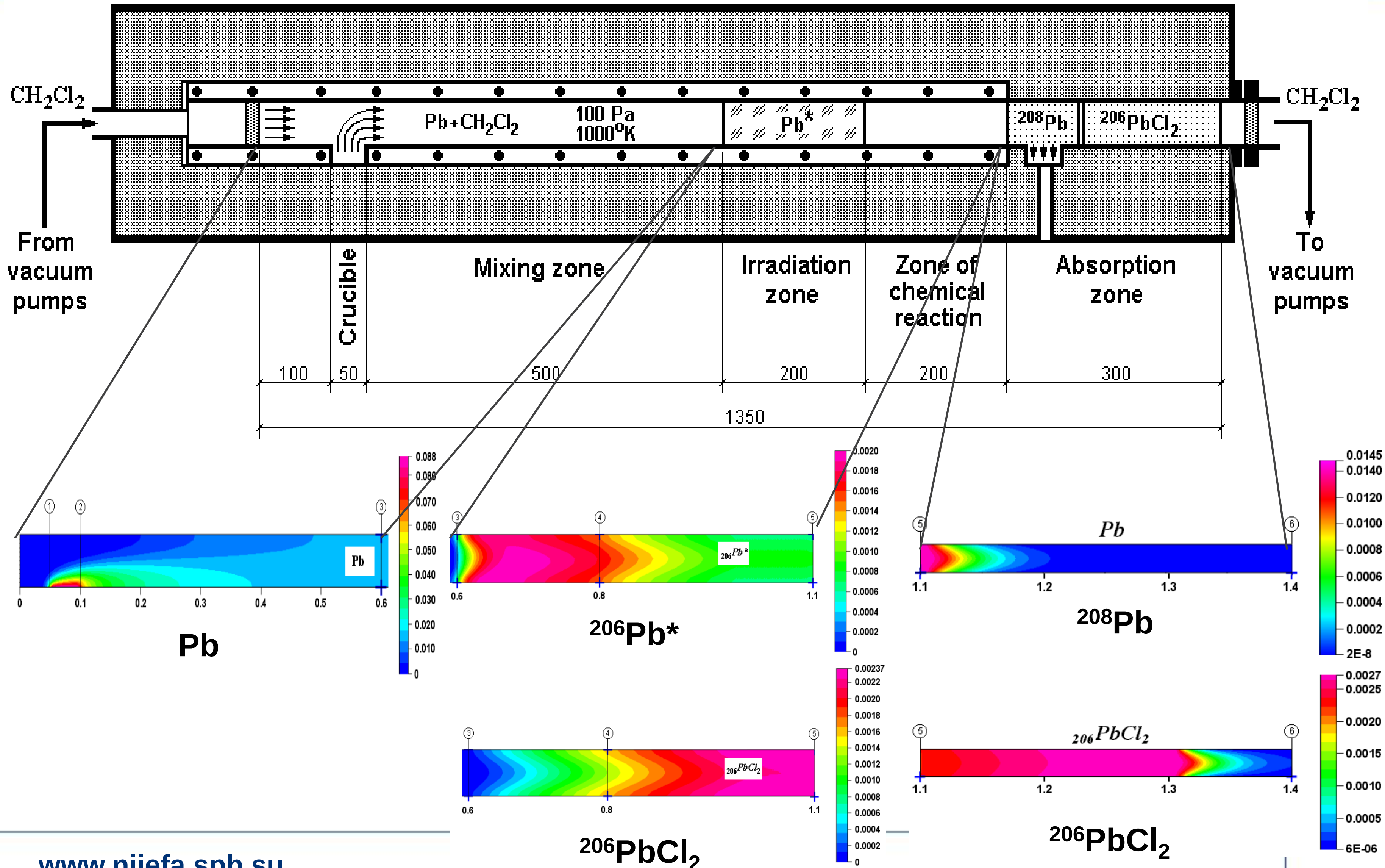
Diagram of the lower operating levels of Pb configuration np2



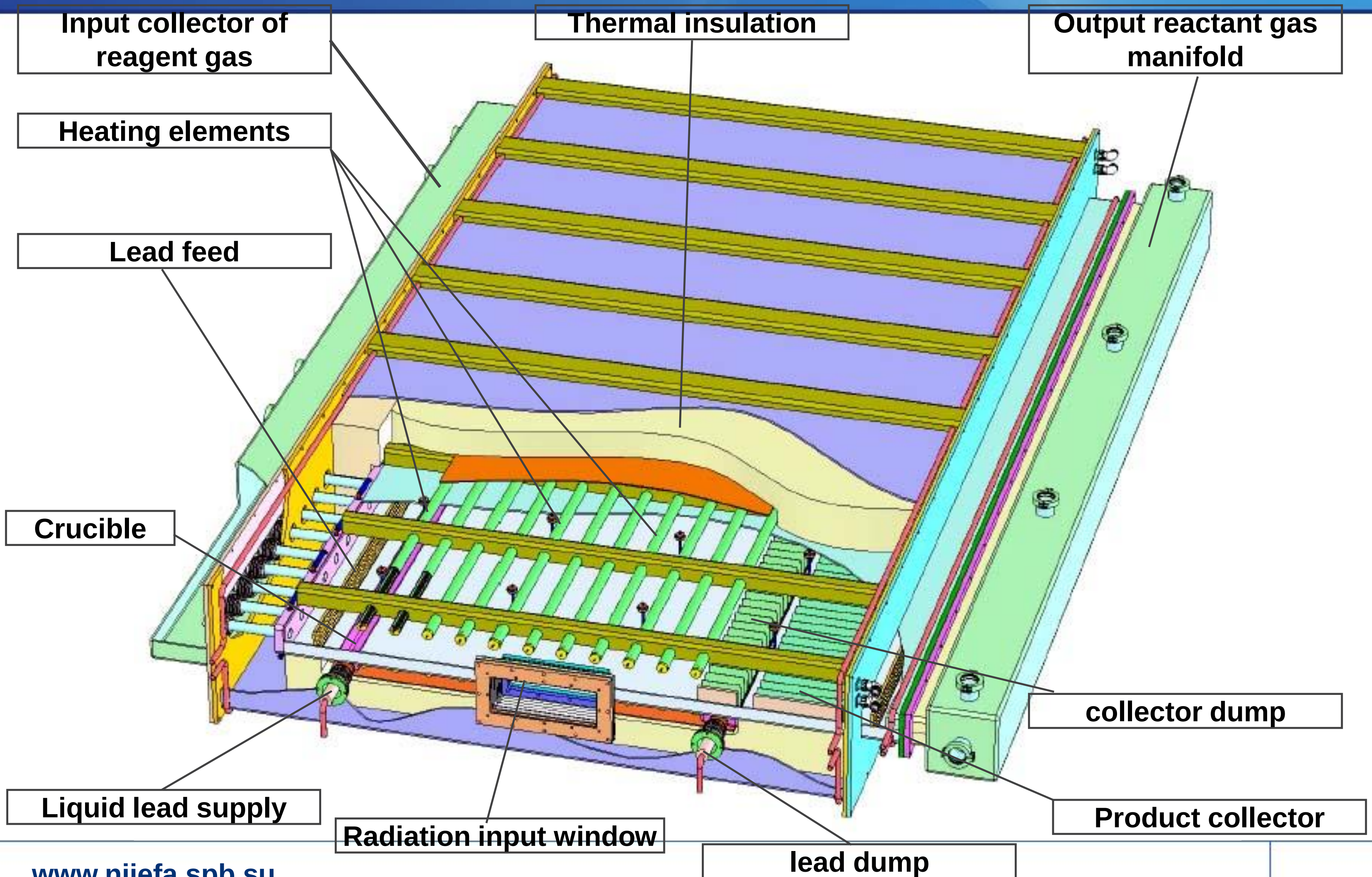
Lead isotope absorption coefficient contours



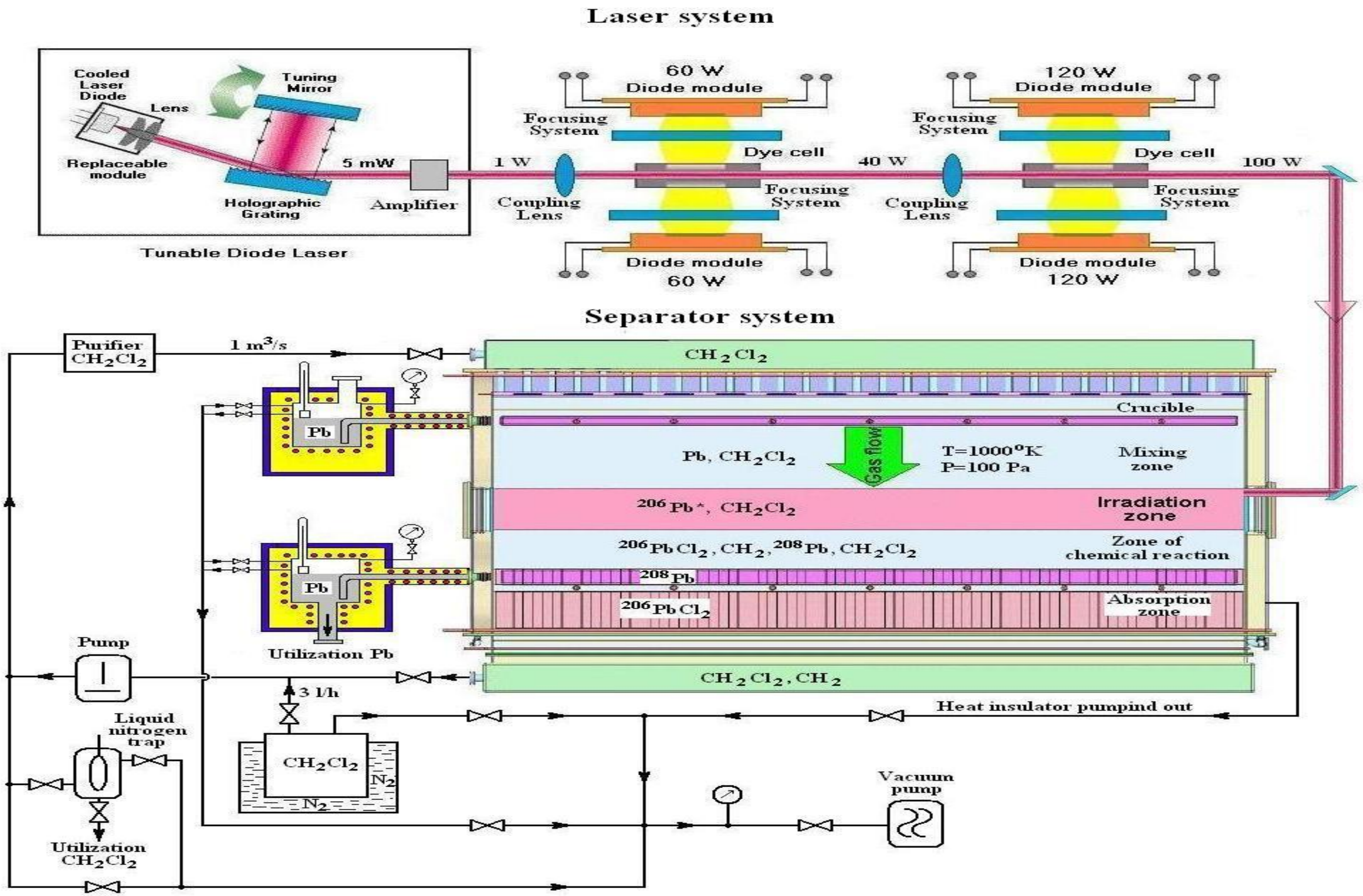
Separator system



Separator system design model



Блок схема установки



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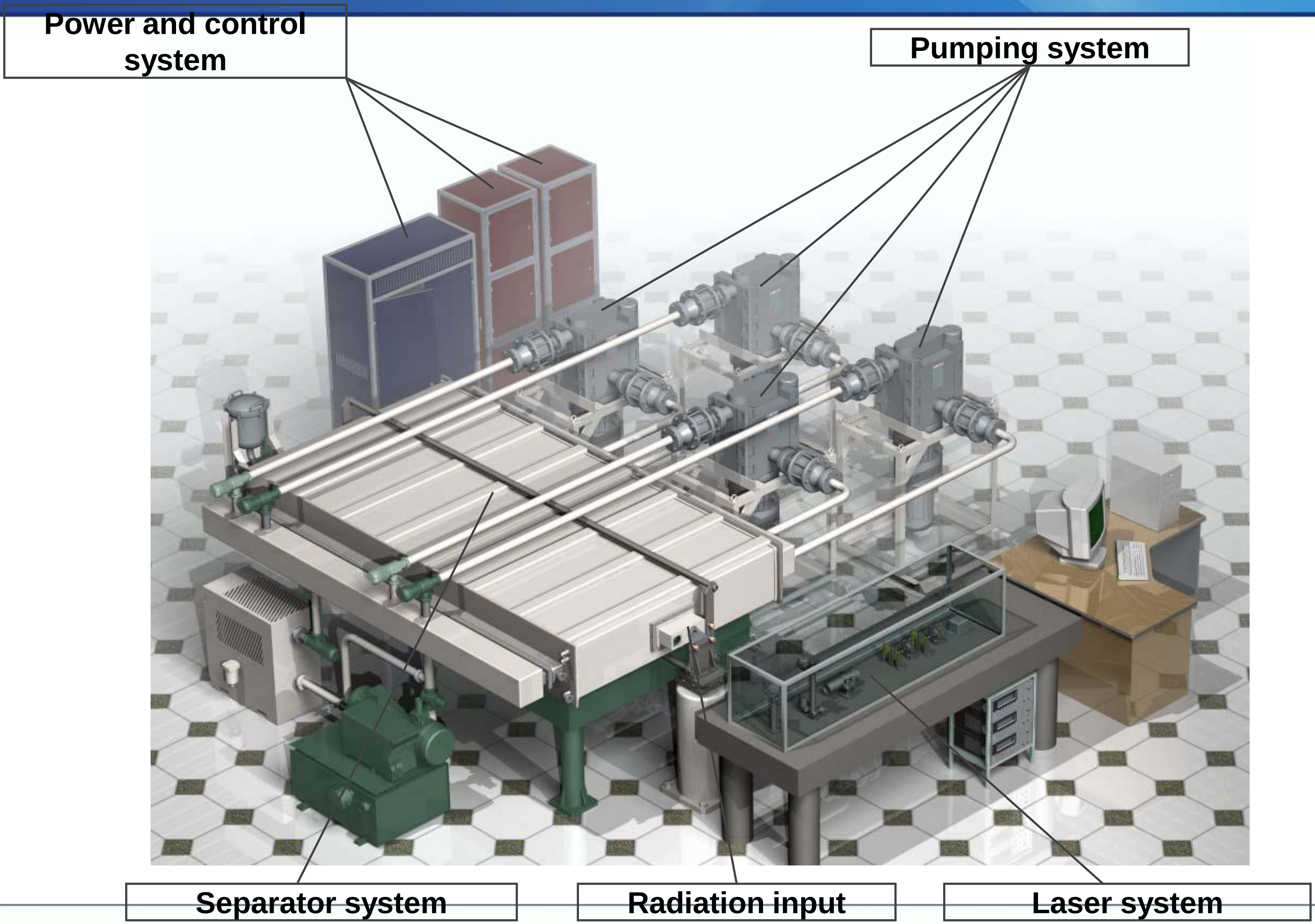
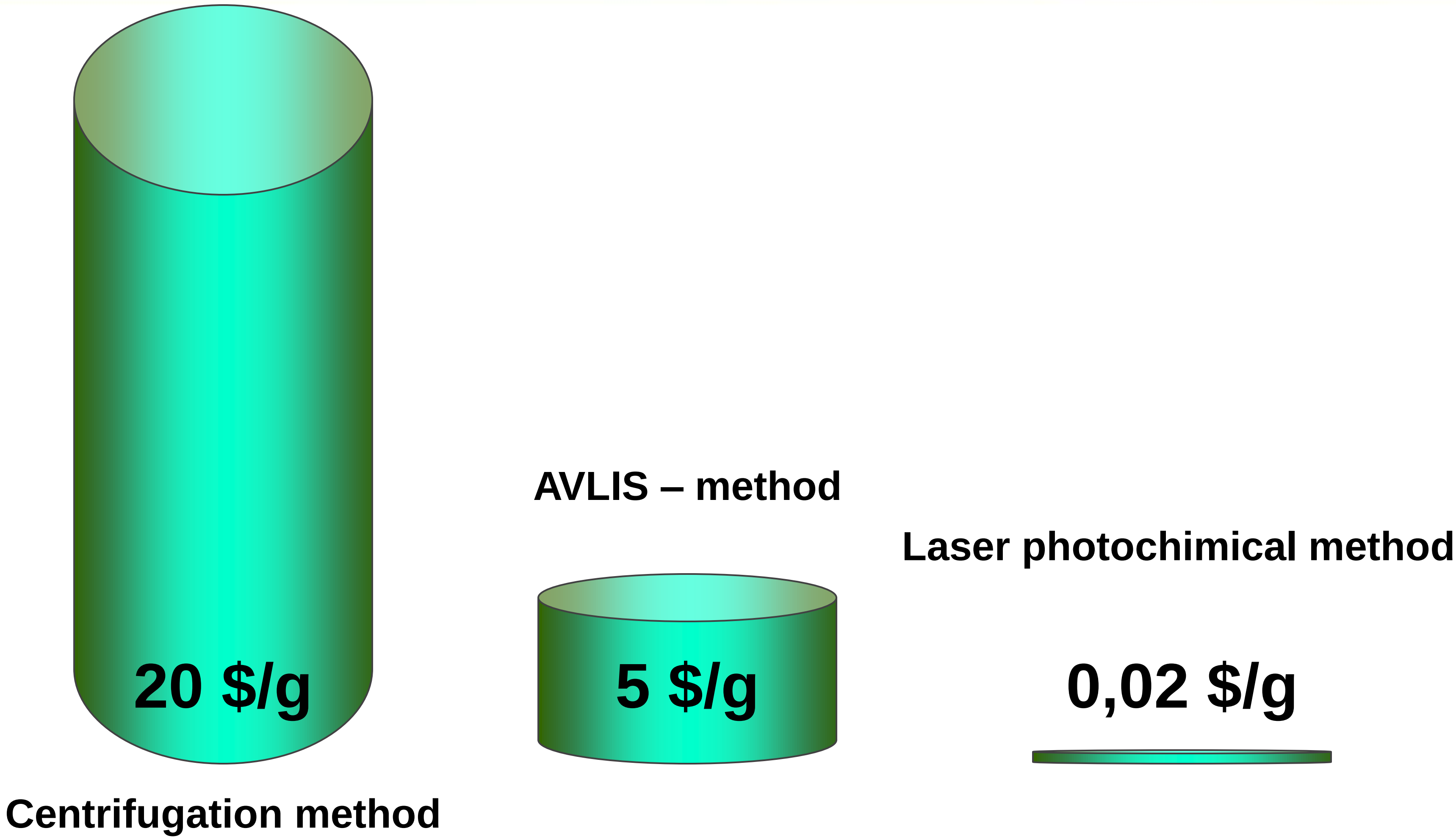
General view of the laser system



Cost of producing Pb-206



PROJECT

INVESTIGATION OF PROCESSES OF HIGH - PERFORMANCE LASER SEPARATION OF LEAD ISOTOPES BY SELECTIVE FOR DEVELOPMENT OF ENVIRONMENTALLY CLEAN PERSPECTIVE POWER REACTOR FACILITIES

Project participants:

**Federal State Unitarian Enterprise “D.V.Efremov Scientific Research Institute of Electrophysical Apparatus”,
Scientific Technical Center MICROTECHNOLOGIES (FSUE "NIIEFA" STC MIT)**

Institute of Semiconductor Physics, Siberian Division, Russian Academy of Sciences.

A.I. Leypunsky Institute for Physics and Power Engineering

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ROSATOM

Joint Stock Company "State Scientific Centre of the Russian Federation – Institute for Physics and Power Engineering named after A.I. Leypunsky"

STATE ATOMIC ENERGY CORPORATION "ROSATOM"

Heavy Liquid Metal Coolants in Nuclear Technologies (HLMC-2018)

The effect of gas bubbles on the erosion of steel in lead

Alekseev V.V., Orlova E.A.

Introduction

The most significant factor leading to the erosion of steel in lead is the hydrodynamic effect of the coolant flow. Erosion failure of structural materials in a heavy coolant is rapidly developing and occurs in places with increased coolant velocities, in the field of hydrodynamic disturbances, and so on. Protective coatings formed on steels based on oxide technology are vulnerable from the point of view of erosion wear. In the presence of gas bubbles in the coolant, erosion occurs at the lowest flow rates, and the erosive effect itself has a cavitation character. In the conditions of cavitation action of gas bubbles on the oxide coating, breaches of its integrity can be created, as a result of which an intensive influence of lead on steel occurs in damaged areas. The experimental and calculated data obtained make it possible to estimate the process parameters under which a cavitation damage of the oxide coating occurs on the steel surface in the flowing lead.

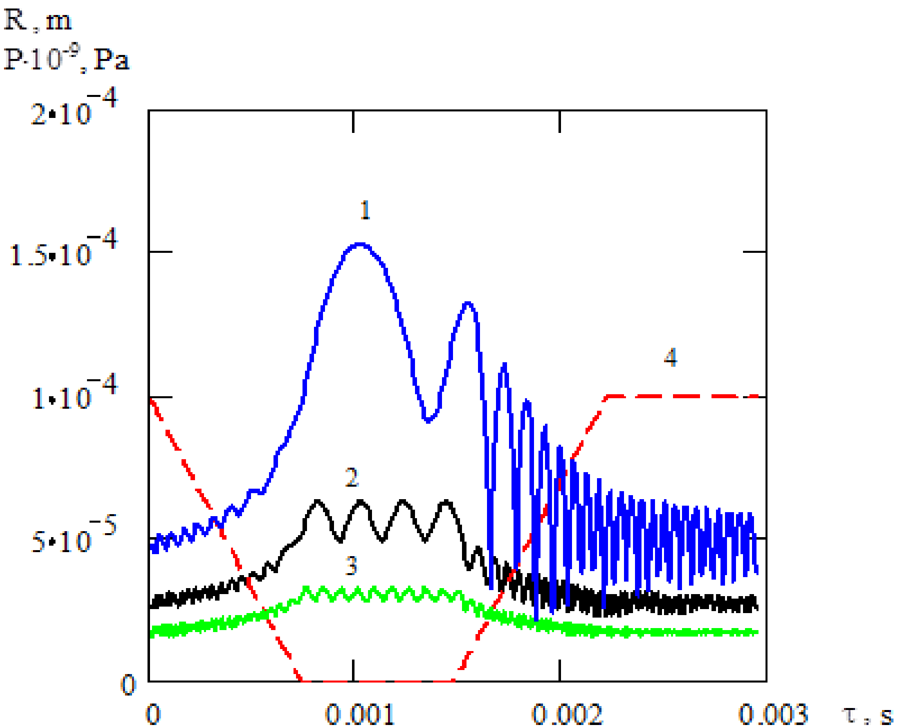
Processes during gas cavitation in a lead coolant

The equation of the bubble wall motion

$$RR'' + \frac{3}{2}R'^2 = \frac{p' - p_\infty - \frac{2\sigma + 4\mu R'}{R}}{\rho}$$

where R, R', R'' – radius, velocity and acceleration of the bubble wall;
p' – pressure in the gas bubble;
p_∞– pressure in lead;
σ – surface tension;
μ– viscosity of the liquid;
ρ – density of the liquid.

Processes during gas cavitation in a lead coolant



Change in the radius of gas bubbles in lead for a time (R_0: 1 – 50 µm; 2 –30 µm; 3 –20 µm) with the pressure change in the liquid (curve 4); $\frac{dP}{d\tau} = 14\ 10^7\ Pa/s$

Processes during gas cavitation in a lead coolant

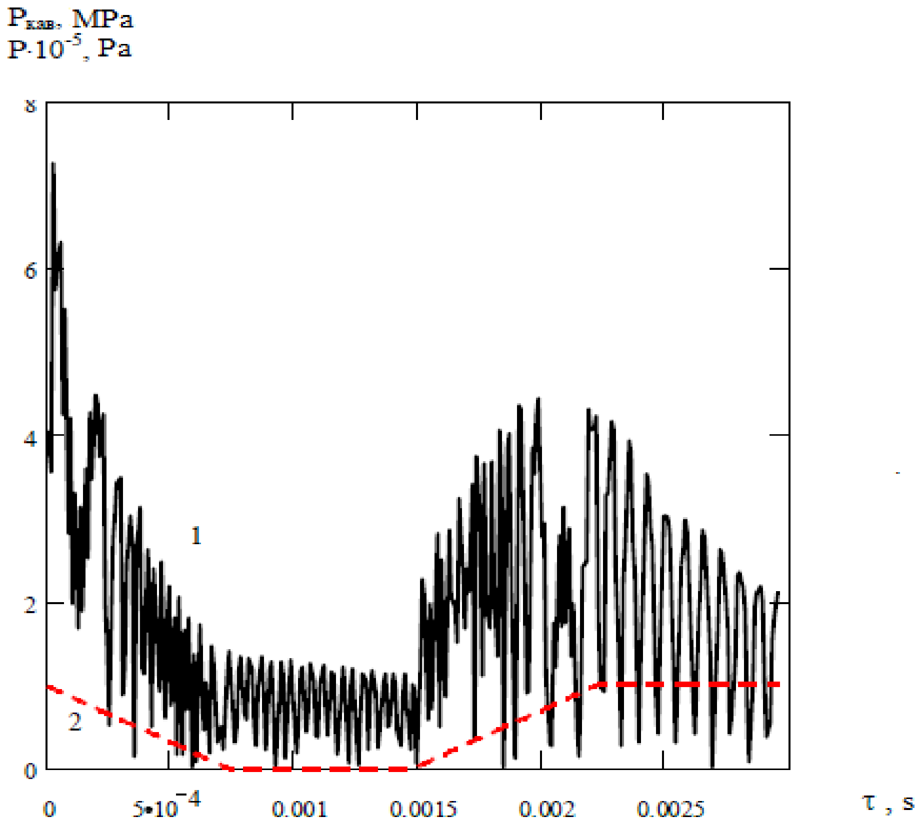
The hydraulic shock pressure

$$P_{\text{гу}} = \frac{\rho c u_0 \rho_s c_s}{\rho c + \rho_s c_s}$$

where c –sound speed;
u_0 – velocity of the jet.
Index s refers to the material of the wall, about which the jet strikes.

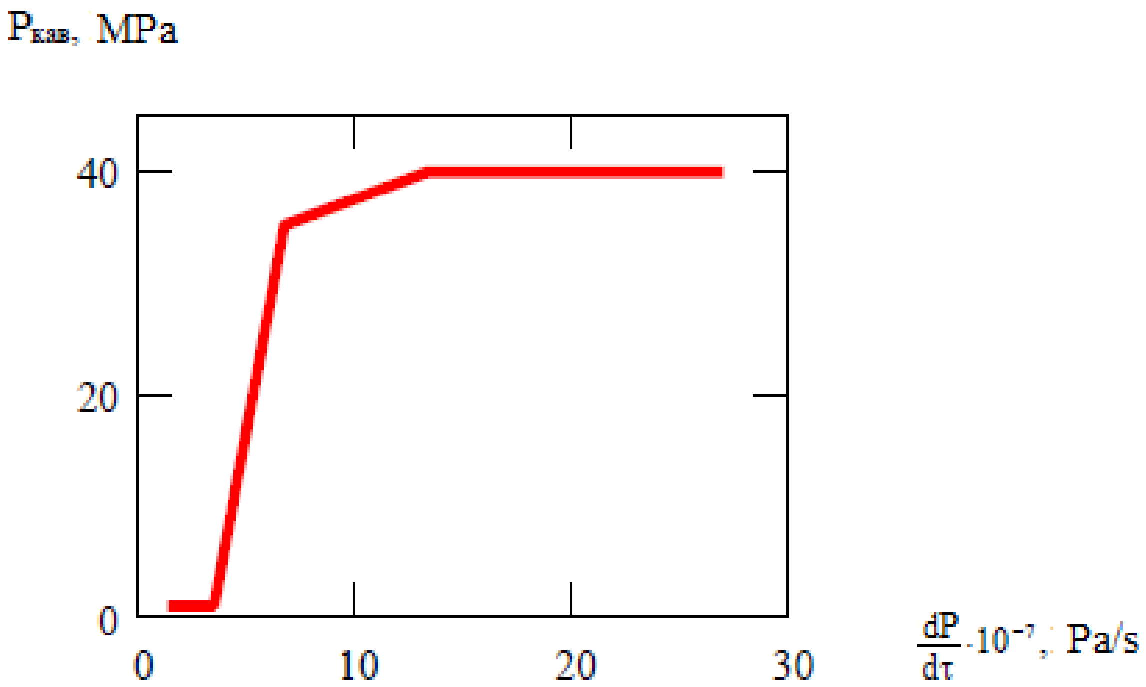
This shock pressure is significantly higher than the decelerating pressure in the jet

Processes during gas cavitation in a lead coolant



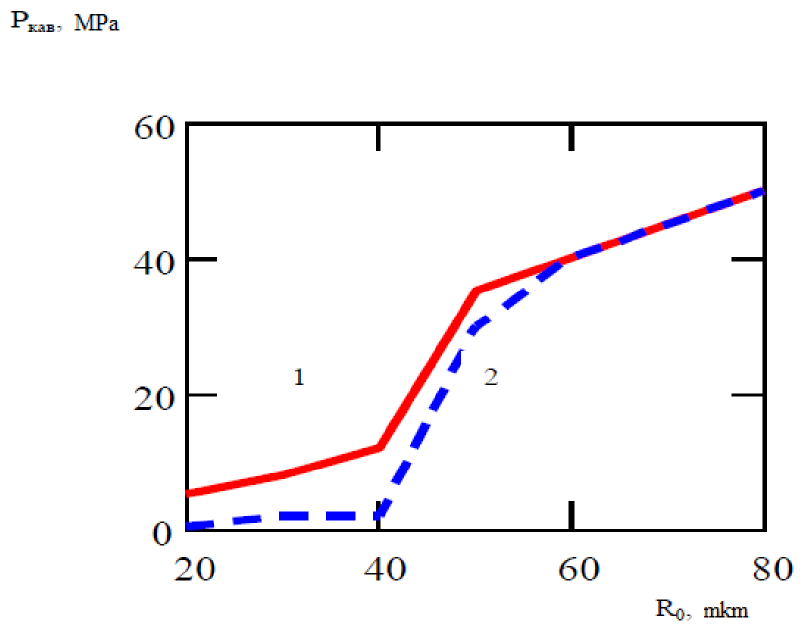
Pressure developed by lead jets on the wall for a bubble R_0 = 20 μm:
1- cavitation pressure; 2- pressure in lead

Processes during gas cavitation in a lead coolant



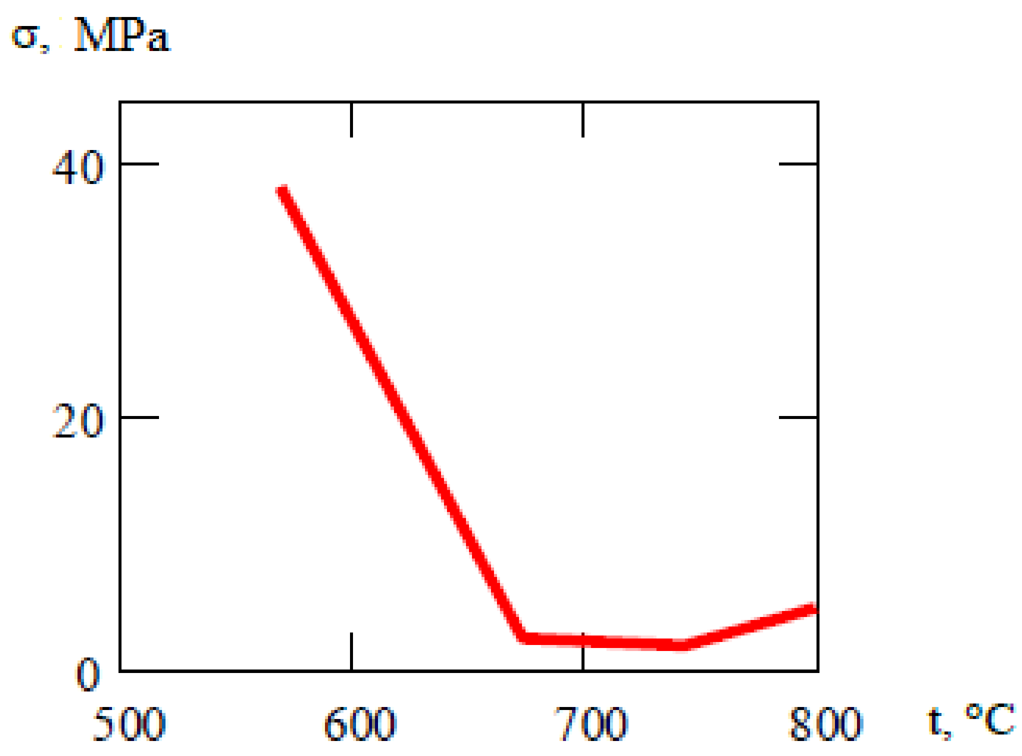
Dependence of the impact pressure of cavitation jets on the rate of pressure change in the lead for R_0=50 μm

Processes during gas cavitation in a lead coolant



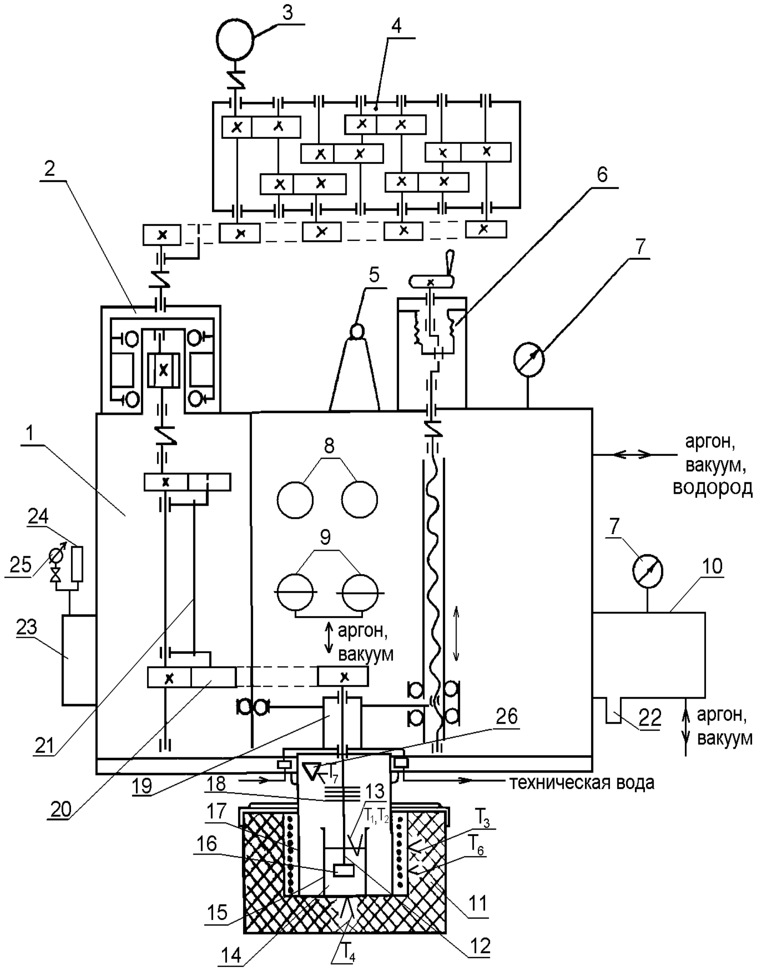
Dependence of the impact pressure of cavitation jets on the size of gas bubbles:
1 – $dPd\tau=14\ 10^7\ Pa/s$; 2 – $dPd\tau=7\ 10^7\ Pa/s$

Processes during gas cavitation in a lead coolant



Dependence of the strength of iron oxide on temperature

Experimental installation



Experimental installation

A view of the location of the cavitation zones behind the protrusions on the erosion samples after tests in lead at 1200 rpm and 650^{0}C

Model of cavitation caverns formation

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The concentration of bubbles in liquid lead (Cb)

The volume of gas flowing into the cavity per unit of time before the start of bubble expansion is described by expression

$$\mathbf{V_0 = C_b\, S\, u,}$$

where S –the cross-sectional area of the liquid flow, from which the gas bubbles are separated due to the centrifugal effect;

u – the lead velocity in front of the protrusion.

After the expansion of the bubbles, the volume of gas entering the same amount of liquid per unit time will be

$$\mathbf{V_p = C_b\, S\, u\, K_p^{\,3},}$$

where $K_p = R_p/R_0$ – the degree of expansion of the bubble.

If the average lifetime of the bubbles from entering the cavern before the collapse is denoted by τ_{cx}, then the volume of the cavity is expressed as $V_к = V_p\tau_{cx}$.

Otherwise the volume of the cavity can be expressed as $V_к = S_к L_к$, where $L_к$ – the length of the cavity behind the protrusion; $S_к$ – the cavity cross-sectional area.

A joint solution of the above relations gives an expression for estimating the concentration of bubbles in lead

$$\mathbf{C_b = L_к\, S_к./\ (\tau_{cx}\, u\, S\, K_p^{\,3})}$$

Model of the gas bubbles distribution on the height of the lead column

ROSATOM

A model is proposed for the distribution of the bubble concentration in lead by the height of the crucible. The mathematical description of the process in a stationary mode is based on:

$$D\frac{dc}{dy} = wc$$

where c – the concentration of bubbles in the liquid;
D – the coefficient of turbulent diffusion of bubbles in lead;
y – the coordinate on the height of the lead column;
w – the rate of rise of the bubbles.
The radius of the gas bubbles generated in the coolant is calculated by Weber number

$$c = c_b^{max} \exp\left(\frac{w(y-h)}{D}\right)$$

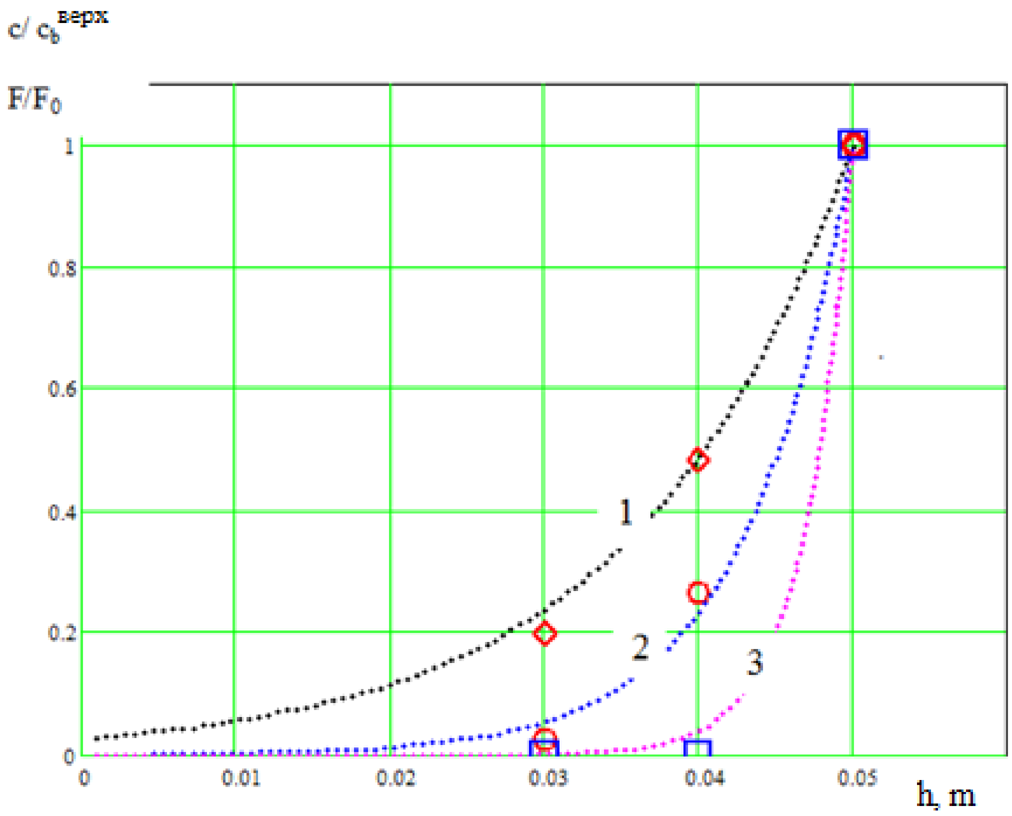
where h – the height of the lead column in the crucible;
c_b^{max} – the gas concentration at the surface of the lead mirror.
$D = 6{\cdot}10^{-5}$ m^2/s. The rate of bubble rise is estimated by the Todes-Rosenbaum formula.
.

The gas concentration at the surface of the lead mirror is related to the concentration at the level of the upper sample by the relation

$$c_b^{max} = c_b^{верх} /(1- \exp(w(h^1 - h)/D))$$

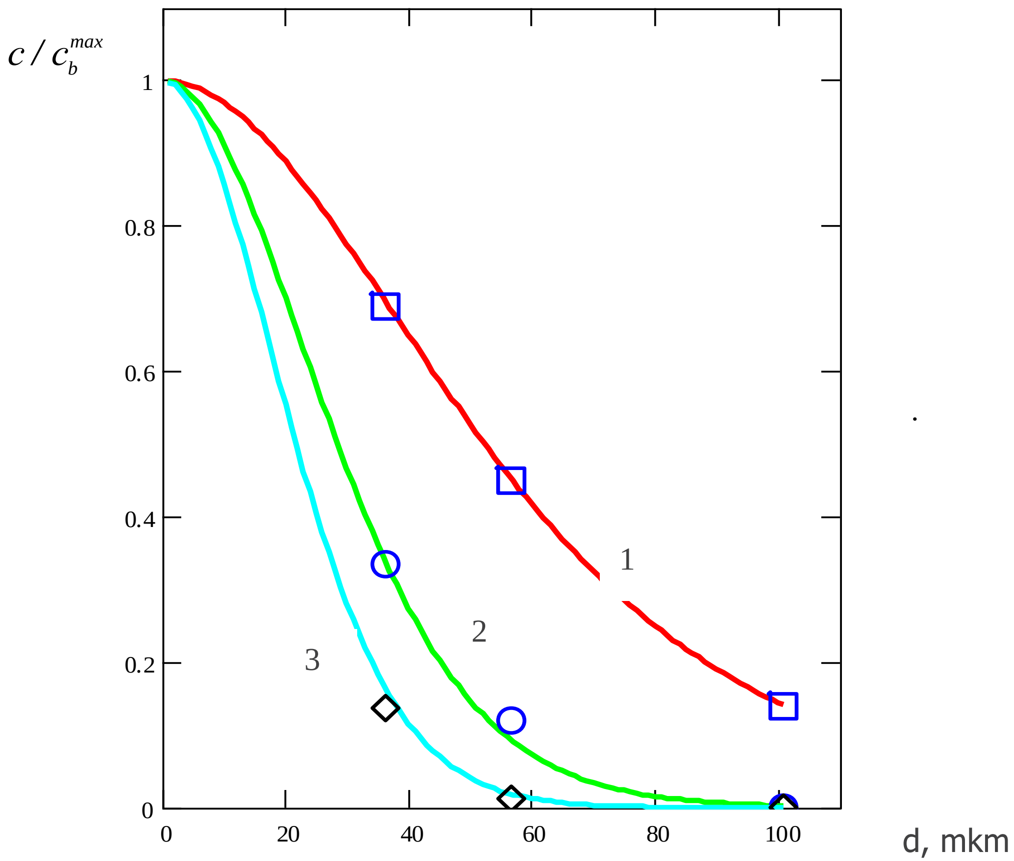
where h^1 – the height of the lead column at the level of the upper sample;
$c_b^{верх}$ – the bubble concentration at the level of the upper sample.

Results of calculations



Distribution of the relative concentration of bubbles in lead by the crucible height at different rotational speed of the cylinder with the samples (curves):
1 - 1500 rpm; 2 - 1200 rpm; 3 - 900 rpm.
The symbols denote the relative values of the erosion areas of the samples in units F/F_o, where F_o is the area of erosion damage of the upper sample

Results of calculations



The relative concentration of gas in lead for different bubble sizes:
solid lines - concentration at sample levels: 1 - 50 mm; 2 - 40 mm; 3 - 30 mm; symbols – estimates using experimental data

Results of calculations

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The parameters of cavitation process at T = 650 ^{0}C

Number of revolutions of the cylinder, 1/min	u, m/s	R_0, μm	$P_{кав}$, MPa	K_P	$L_к^{верх}$, m	c_b^{max}, m^3/m^3
900	1,22	50,5	40	3,1	$3{\cdot}10^{-3}$	0,1
1200	1,62	28,4	8	2,0	$8{\cdot}10^{-3}$	0,07
1500	2,03	18,2	4	1,7	$37{\cdot}10^{-3}$	0,12

Thank you for your attention!

РОСАТОМ

ГОСУДАРСТВЕННАЯ КОРПОРАЦИЯ ПО АТОМНОЙ ЭНЕРГИИ «РОСАТОМ»

Heavy Liquid Metal Coolant Flow Measurement

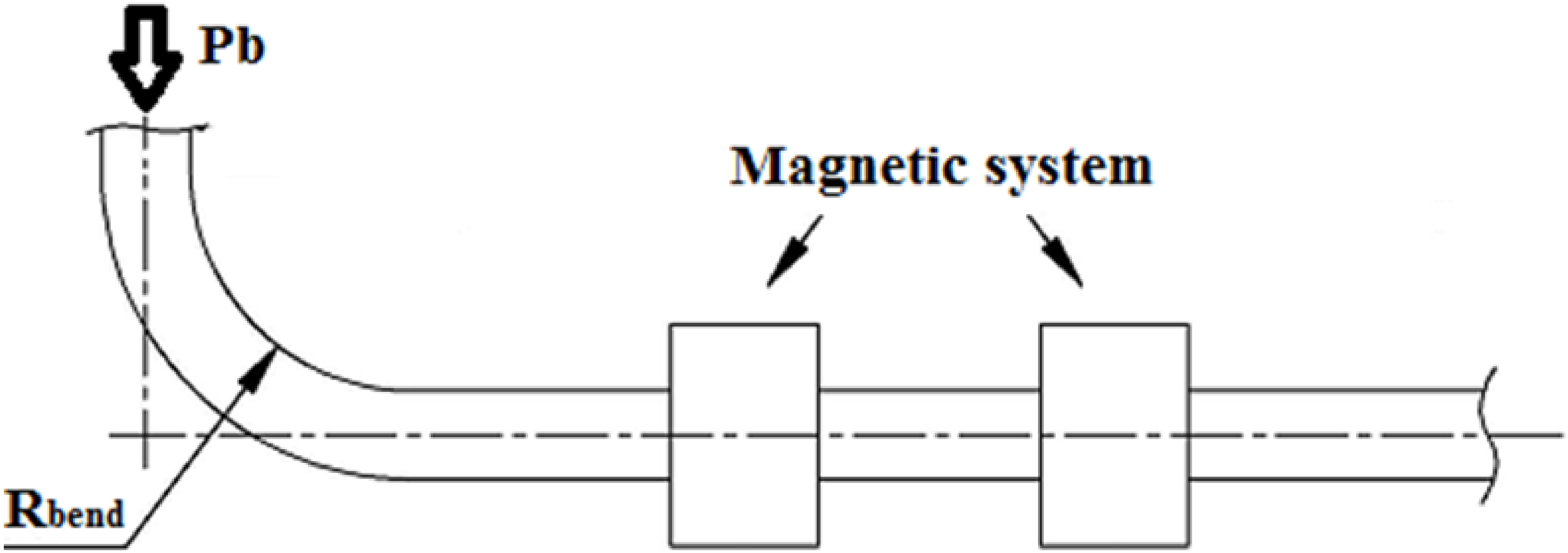
Kebadze B.V., Lagutin A.A., Shurupov V.A., Generalov E.V., Fomin A.N.
(JSC "SSC RF - IPPE", Obninsk)

Problems in measuring lead flowrate

Technical difficulties arise in measuring lead flow with conventional flow meters:

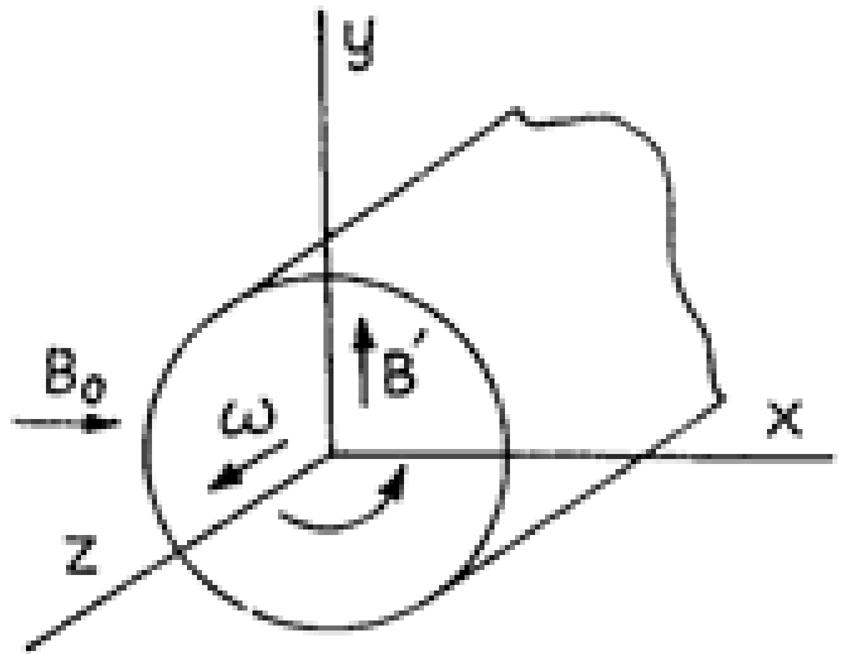
- Contact resistance at the "liquid metal – pipeline wall" boundary;
- High operating temperatures (up to 600°C);
- Certificated calibration facilities for lead flowrates $G \geq 100$ m^3/h are not available.

Measuring section of lead rig pipeline for non-contact correlation flow meter (NCCF)



Directly measured by NCCF parameter is the transport time of hydrodynamic inhomogeneities (vortices) between two measuring positions [1, 2].

The generation of the NCCF signal



The source of the signal for NCCF – fluctuations of induced magnetic field,

B_0 – a constant magnetic field (MF)

ω - vorticity

B′ - fluctuations of induced

The advantages of contactless correlation flowmeters (NCCF)

Characteristics of NCCF does not depend on:

- conductivity and temperature of liquid metal,
- the presence of impurities,
- contact resistance at the boundary «liquid metal-pipeline wall»,
- aging of the magnetic system.

Installation of transducers on the measuring section of the pipeline does not require welding.

Operation conditions and components of the NCCF system

The parameters of coolant and measuring section of lead rig pipe:

- flow range 0÷72 m^3/h;
- temperature range 200÷550 °C;
- pipeline diameter Dn92.

NCCF system components:

- measuring section of lead rig pipeline including the bend as vortices generator;
- primary converters (magnetic systems with receiving coils);
- secondary instrumentation.

NCCF primary transducers on measuring section of lead rig pipe

The components of instrumentations

Subsystem of normalizing converters:
• noise amplifiers,
• thermocouple signals converting modules,
• analog signals converters to standard 4-20 mA DC signals.

The digital processing unit:
• industrial computer,
• the ADC Board,
• SOFTWARE.

The principle of NCCF graduation

$$G_{real} = k_c G_{cor}$$

$$G_{cor} = V/\tau$$

V - the volume between two measuring positions,

τ - the average time of vortices transport.

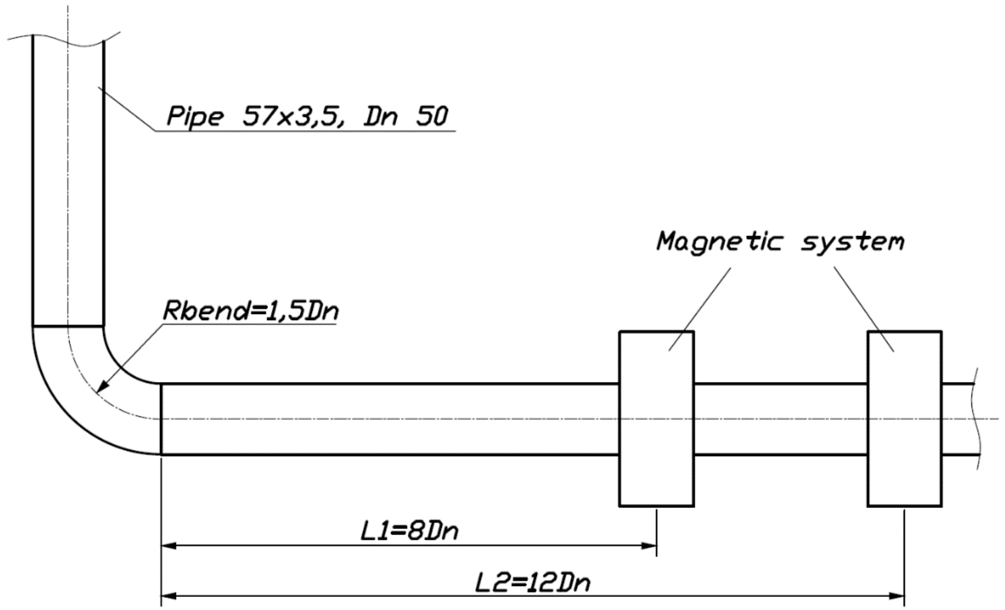
The purpose of graduation - to determine the coefficient k_c.

The object of the graduation – NCCF system with measuring section of the pipeline including the bend to be installed in lead rig.

The calibration tool - sodium calibration facility IRS-M, certified for flowrate up to 100 m^3/h.

Measuring section of pipeline with standard bend

The vortex generator - standard steep bend, R_{bend}= 1,5Dn
Diameter of measuring section Dn50
The range of flowrate 5 - 70 m^3/h.
Position of the first transducer relative to bend, range: 3 Dn-12 Dn.
The distance between the measuring positions: 3Dn; 4Dn



The results of the NCCF calibration

The dependence of the correction factor k_c from the Reynolds number (a - R_{bend} ~ 3,5 Dn; b - R_{bend} = 1,5 Dn)

$Re \cdot 10^{-4}$	6,68	8,68	10,84	13,13	15,50	16,73	21,98	27,88	38,33	44,29	54,64	65,26	76,62	86,38
k_c	0,917	0,926	0,933	0,934	0,936	0,939	0,948	0,947	0,953	0,952	0,957	0,961	0,963	0,966

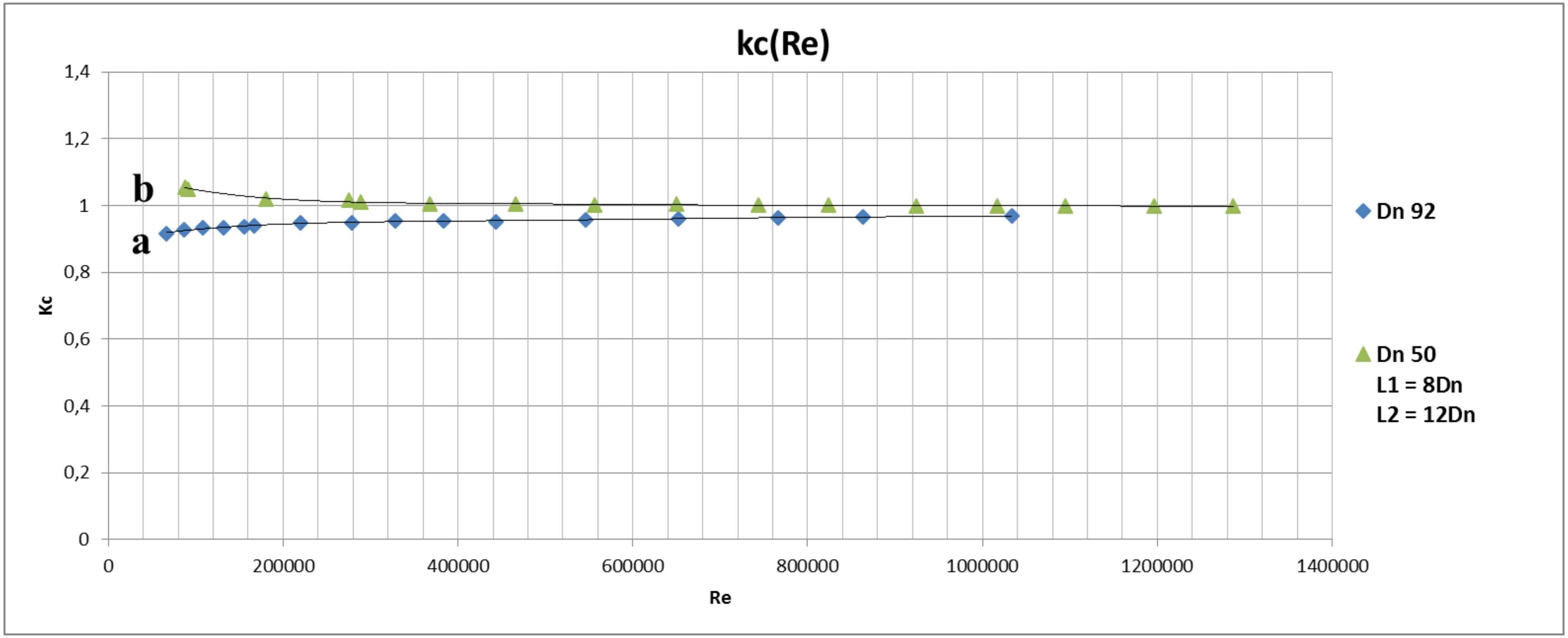
a

$Re \cdot 10^{-4}$	9,09	18,06	28,84	36,82	46,64	55,67	64,96	74,39	82,33	92,37	101,6	109,4	119,6	128,7
k_c	1,050	1,019	1,009	1,005	1,004	1,003	1,003	1,001	1,000	0,998	0,999	0,999	0,998	0,998

b

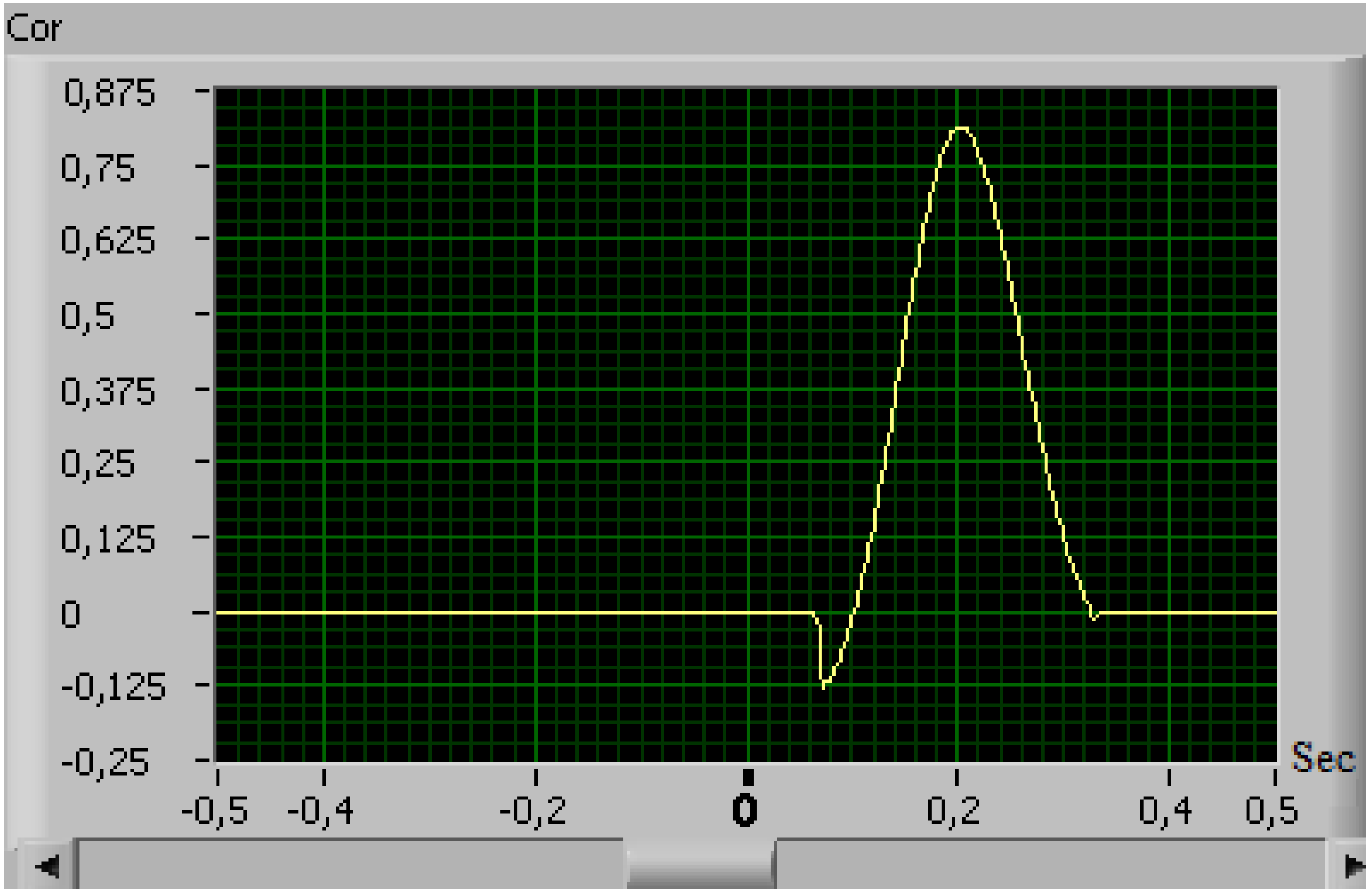
During the lead flowrate measurement, approximated dependence $k_c = k_c(Re)$ was used.

Correction factors for smooth (a) and standard steep (b) bend



Note:
Standard steep bend R = 1,5 Dn, GOST 17375-2001

Measurements on the lead rig



Correlation function, lead flowrate ~ 30 m³/h

Ability of NCCF certification for G > 100 m³/h

The basis of NCCF certification for lead flowrate $G > 100$ m^3/h is **the physical modeling** of compliance with the geometric and hydrodynamic (Re = idem) similarity.

Extended flowrate range of IRS-M facility - 385 m^3/h.

The facility is pre-certified for this range.

Example of calibration ability based on modeling:

The sodium flow rate $G = 385$ m^3/h through the model measuring section of the pipeline Dn 100 provides calibration of the NCCF system intended for lead flowrate range **up to 600 m^3/h** with measuring section of pipe Dn 300.

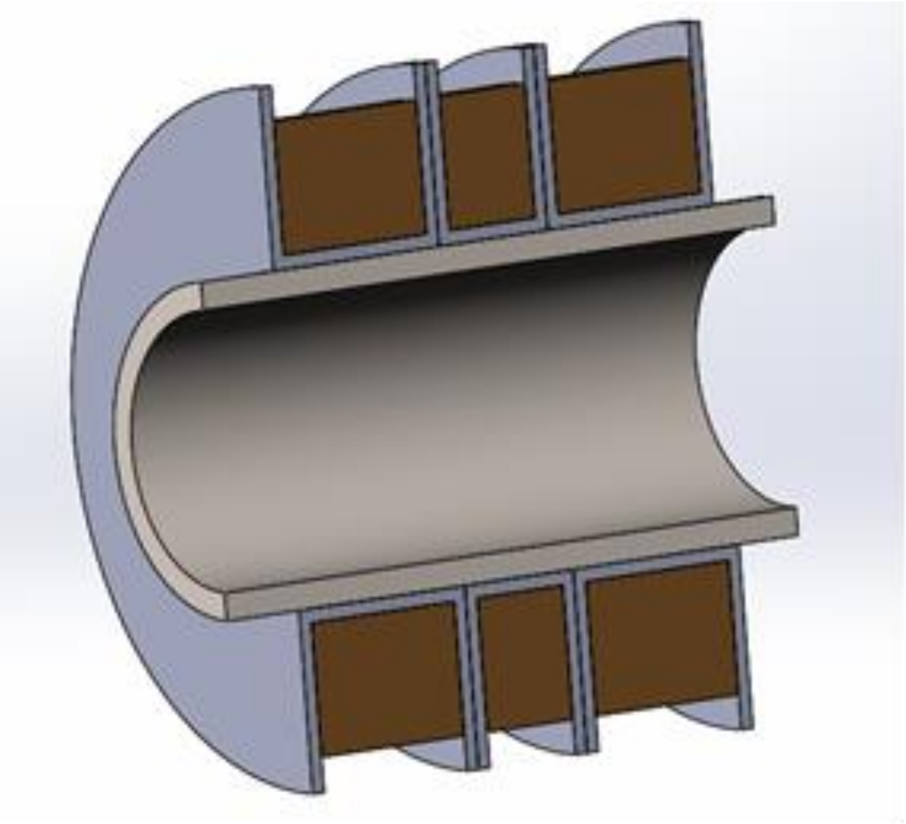
Eddy current flowmeter

A sample of eddy current flow meter with two excitation coils and one receiving coil was developed and pre-tested at the IRS-M installation.

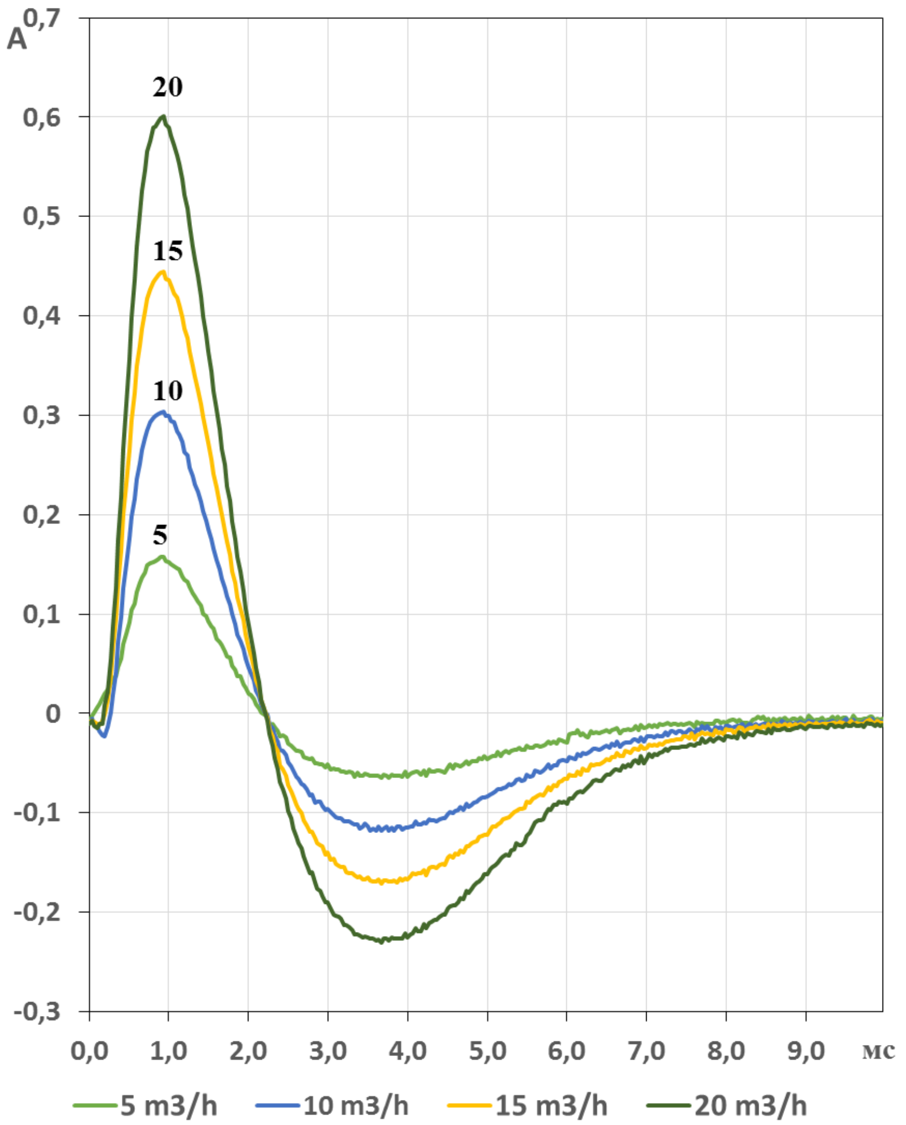
Excitation coil is powered by the pulses of (0,1 - 5 mc) duration.

The combination of mounted in series on the HMLC loop contactless correlation and eddy current flowmeters allows to extend the range of flowrate measurements. The use of NCCF system enables reliable in situ verification.

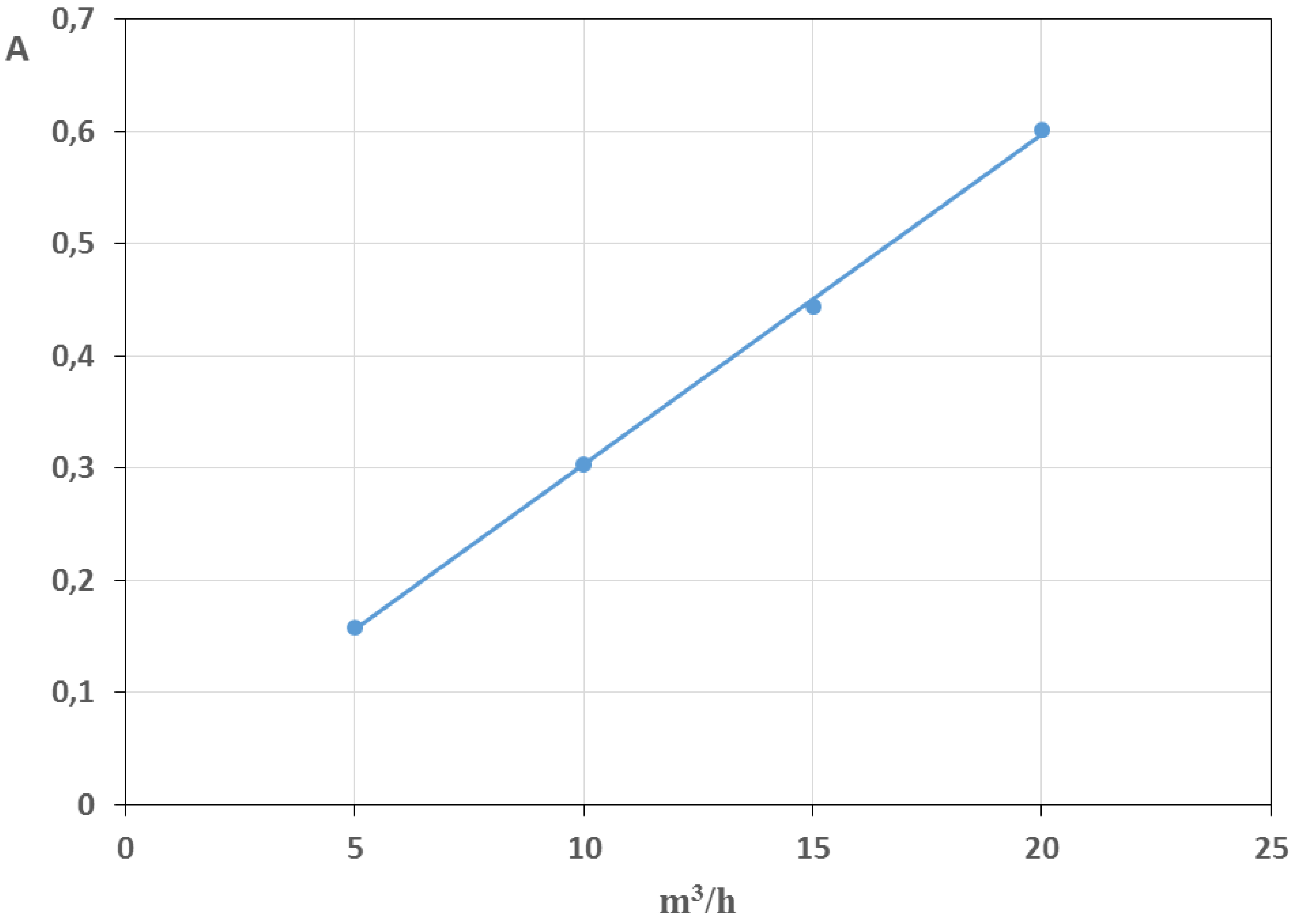
Primary converter of eddy current flow meter



The output signal of the eddy current flow meter



Signal amplitude dependence on the flowrate



Conclusion

- The measurement system based on non-contact correlation flowmeters has been developed, manufactured and tested at the lead coolant rig.
- The technique of calibration for NCCF intended to lead flowrate measurement on the sodium verification facility, using geometric and dynamic similarity (Re = idem) is offered.
- Calibration characteristics for several types of vortex source are experimentally determined (R_{bend} ~ 3,5 Dn; R_{bend} = 1,5 Dn).
- A sample of an eddy current flowmeter with pulse power supply was tested.

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Thank you for the listening