Progress in the Development of Lead-cooled Fast Reactors

January 15, 2020
Innovative Nuclear System Workshop, Tokyo Tech, Tokyo, Japan

Laboratory for Advanced Nuclear Energy, Institute of Innovative Research

Toru Obara

Contents

- Features of lead cooled fast reactor
- Development status of GIF member states
- LBE cooled Stationary Wave Breed and Burn reactor

Features of lead-cooled fast reactor

- Types of coolant
  - Lead
    - Boiling point: 1737°C
    - Melting point: 327°C
  - Lead-bismuth eutectic (LBE)
    - Boiling point: 1670°C
    - Melting point: 124°C
- Chemical Reaction
  - Inert with air and water
- Neutronic feature
  - Hard neutron spectrum
  - Less neutron leakage

Features of lead-cooled fast reactor (continued)

- Thermal-hydraulics
  - Atmospheric pressure cooling
  - Natural circulation
- Points to keep in mind in the reactor development
  - Corrosion and erosion of steel
  - Polonium production in LBE
- International activity
  - One of the reactor types in Generation IV International forum (GIF)

Current LFR development status in GIF member states

- Japan – Fundamental study
  - Theoretical study
    - Breed and burn reactor concepts using LBE coolant in Tokyo Tech
  - Experimental study
    - Oxidation characteristics of lead-alloy coolants in air ingress accident
- EURATOM – Conceptual study
  - ALFRED
  - MYRRHA
- Russia – Licensing for constriction
  - BREST-OD-300 by ROSATOM

Current LFR development status in GIF member states (continued)

- Korea – Conceptual study
  - MINERVA
- USA – Conceptual study
  - Hydromine LFR-AS-200 and LFR-TL-X
  - Westinghouse LFR
  - Columbia Basin Consulting Group (CBCG) – relatively new start
- China – Conceptual and experimental study for ADS
  - CLEAR series

Report of LFR development status in GIF 25th LFR-pSSC meeting in October 2019, Bucharest, Romania

- Japan
- EU
- Russia
- Korea
- China

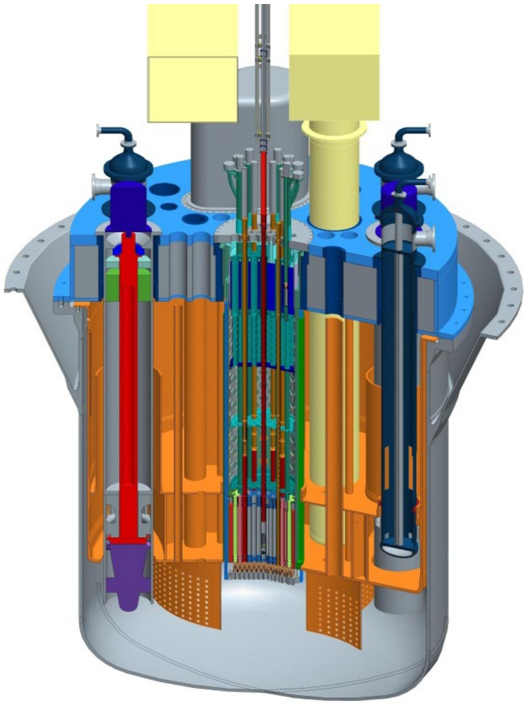
EURATOM Development Status

25^{th} LFR Prov. SSC Meeting

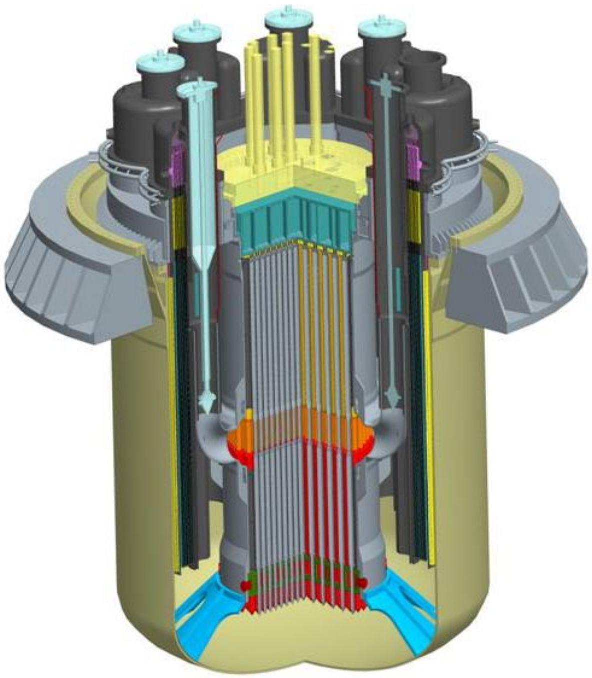
Alessandro Alemberti
Ansaldo Nucleare

Kamil Tuček
European Commission, Joint Research Center - EURATOM

Lead & LBE technology development in Europe
Presently two main projects (with many synergies):



MYRRHA (LBE)



ALFRED (LFR)

Euratom Research & Development related to LFRs

- **Main activities focused on: ALFRED and MYRRHA**
- **Euratom Horizon 2020 Collaborative Projects related to LFRs started in 2017:**
  - **GEMMA project running (materials for Gen IV / LFRs, 6.6 million EUR, ENEA – IT coordinated, started on 1 June 2017)**
  - **INSPYRE project running (fuel for FRs, 9.4 million EUR, CEA - FR, started on 1 September 2017)**
  - **M4F project running (materials for Gen IV and fusion, 6.5 million EUR, CIEMAT, started on 1 September 2017)**
- **Euratom 2019 project:**
  - **PIACE *passive safety freezing prevention in LFRs*, coordinated by ENEA, kick-off June 2019**
- **Running Euratom-US I-NERI PROJECT**
  Title: Small Modular Lead-cooled Fast Reactors in regional energy markets: safety, security, and economic assessments
- **Organization of the LFR-SFR workshop, June 2019 (Brasimone, IT)**
  Many synergies found, possibility to collaborate for future projects (probably from next call of 2020)

H2020 2019-2020 Euratom Fission Call

A budget of **€133.9 million** has been allocated to in total 17 topics. Topics related to reactor systems and specifically Gen-IV/LFRs are given below:

Topic	Title	Type
NFRP-01	Ageing phenomena of components and structures and operational issues	RIA
NFRP-02	Safety assessments for Long Term Operation (LTO) upgrades of Generation II and III reactors	RIA
NFRP-03	Safety margins determination for design basis-exceeding external hazards	RIA
NFRP-04	Innovation for Generation II and III reactors	IA
NFRP-05	**Support for safety research of Small Modular Reactors**	RIA
NFRP-06	**Safety Research and Innovation for advanced nuclear systems**	RIA
NFRP-07	**Safety Research and Innovation for Partitioning and/or Transmutation**	RIA
NFRP-08	**Towards joint European effort in area of nuclear materials**	CSA
NFRP-11	Advancing nuclear education	CSA
NFRP-16	Roadmap for use of Euratom access rights to Jules Horowitz Reactor experimental capacity	CSA
NFRP-17	Optimized use of European research reactors	CSA

Deadline for submission of proposals was: **25 September 2019**

Euratom Research & Development related to LFRs

ESNII supported Projects Proposals to the 2019 call

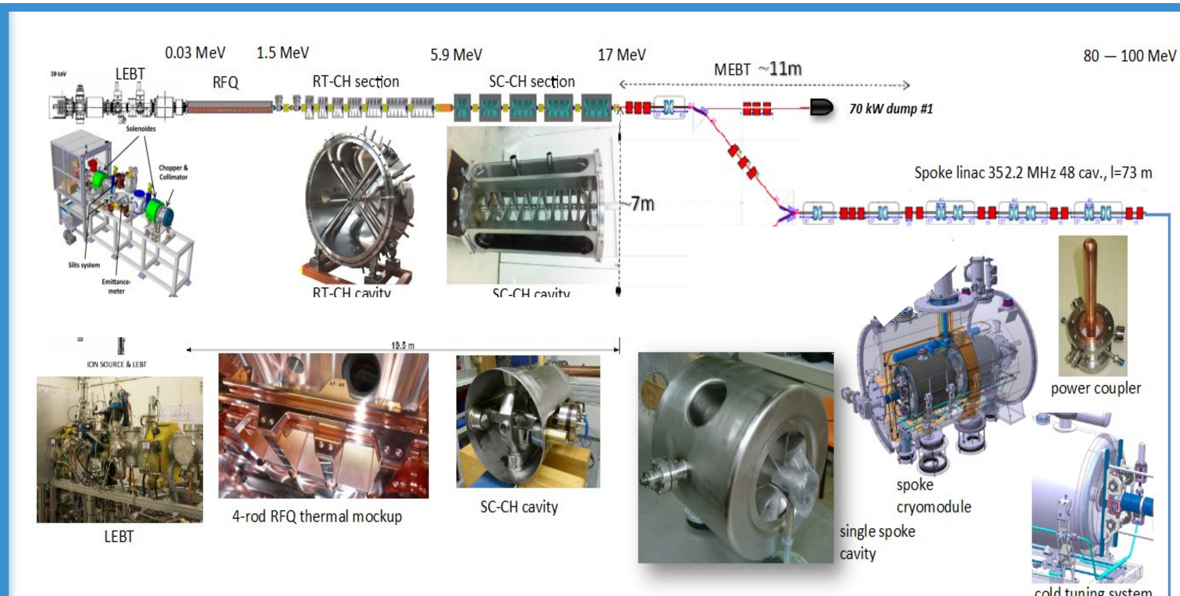
- **BONSAI -** Baseline fOr Nuclear Safety Assessment of Innovative SMR
  4 M€ - SFR, LFR, GFR, HTR, MSR investigated
- **PASCAL -** Proof of augmented safety conditions in advanced liquid-metal-cooled systems
  3.8M€ safety aspects and exp. of liquid metal systems
- **SAFEG -** Safety of GFRs through innovative materials, technologies, and processes
  3.8M€ safety aspects, materials and exp. for GFR
- **PUMMA -** Plutonium Management for More Agility
  3.8M€ - Federate the European community on MOX fuel
  Study Plutonium management in 4th generation reactors
- **PATRICIA -** 6.5 M€ Joint proposal between the Partitioning, Transmutation and Transmuter development communities
- **OrientM -** Establishing an European Joint Programme for Nuclear Materials
  1.1 M€ Coordinated Support Action – Vision Paper etc.
- **Open Access** Open access scheme, providing grants to users of the JRC research infrastructures, 750 k€

MYRRHA

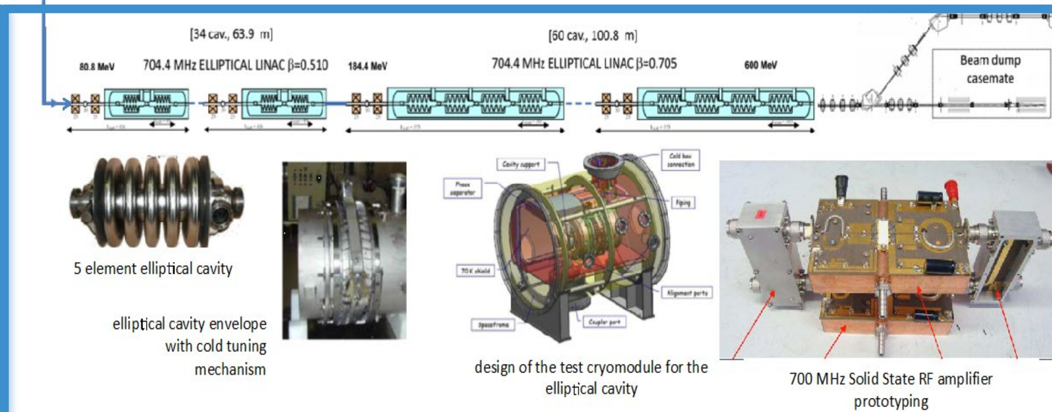
Benefits of phased approach:

- Reducing technical risk
- Spreading investment cost
- First R&D facility available in Mol by the end of 2026

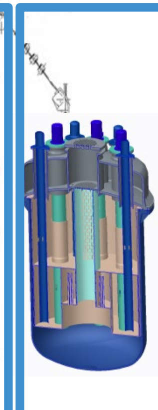
Phase 1 – 100 MeV



Phase 2 – 600 MeV



Phase 3 – Reactor



MYRRHA

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 period 2019 – 2038:
  - 287 MEUR investment (CapEx) for building of the MINERVA (Accelerator up to 100 MeV + proton target facility /PTF/) for the period 2019-2026
  - 115 MEUR for further design, R&D and Licensing of phases 2 (accelerator up to 600 MeV) & 3 (LBE-cooled sub-critical reactor) for the period 2019-2026
  - 156 MEUR for OpEx of the MINERVA for the period 2027-2038
- Belgium requests to establish an International non-profit organization (AISBL/IVZW) in charge of the MYRRHA facility, welcoming the international partners to participate

PIACE project started
PIACE : PASSIVE ISOLATION CONDENSER

Scope of the project: *to address the performances of an innovative Decay Heat Removal system (DHR) based on an isolation condenser which uses non-condensable gases to passively control the power removed from the primary system*

Final goal: to deliver a scaled-up conceptual design **validated for LFRs/ADSs and LWRs (PWR/BWR & PHWR) applications**

Innovation:
Passively delayed coolant freezing (LFR/ADS)
Passively smooth temperature transients (LWR)

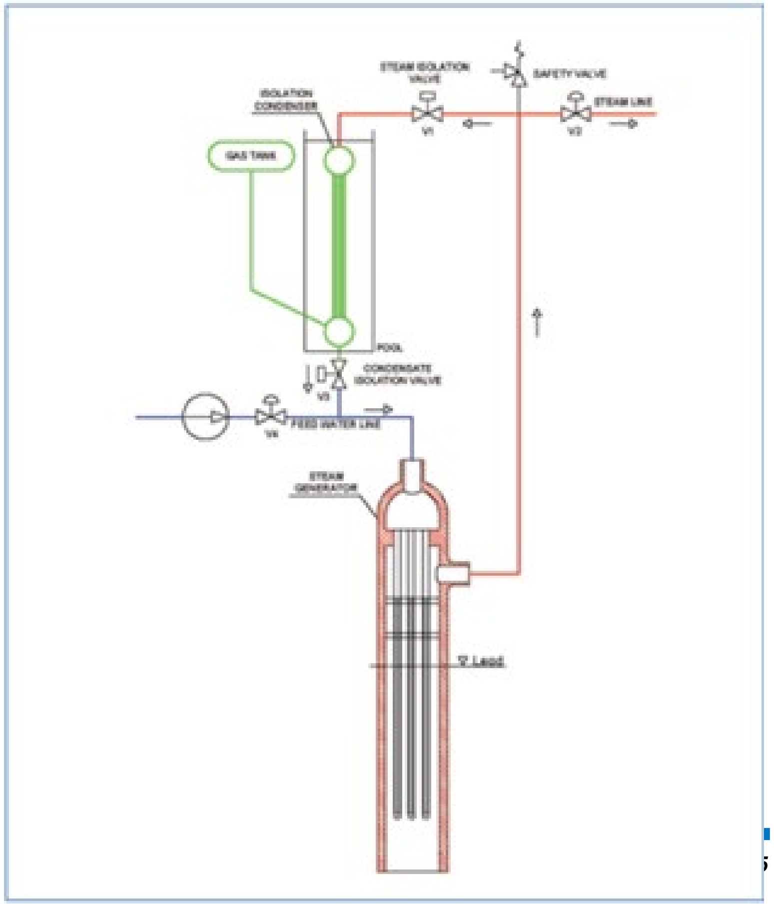
European Call: Nuclear Fission, Fusion and Radiation Protection Research (NFRP-2018)

Subsection: NFRP-2: Model development and safety assessments for Generation IV reactors

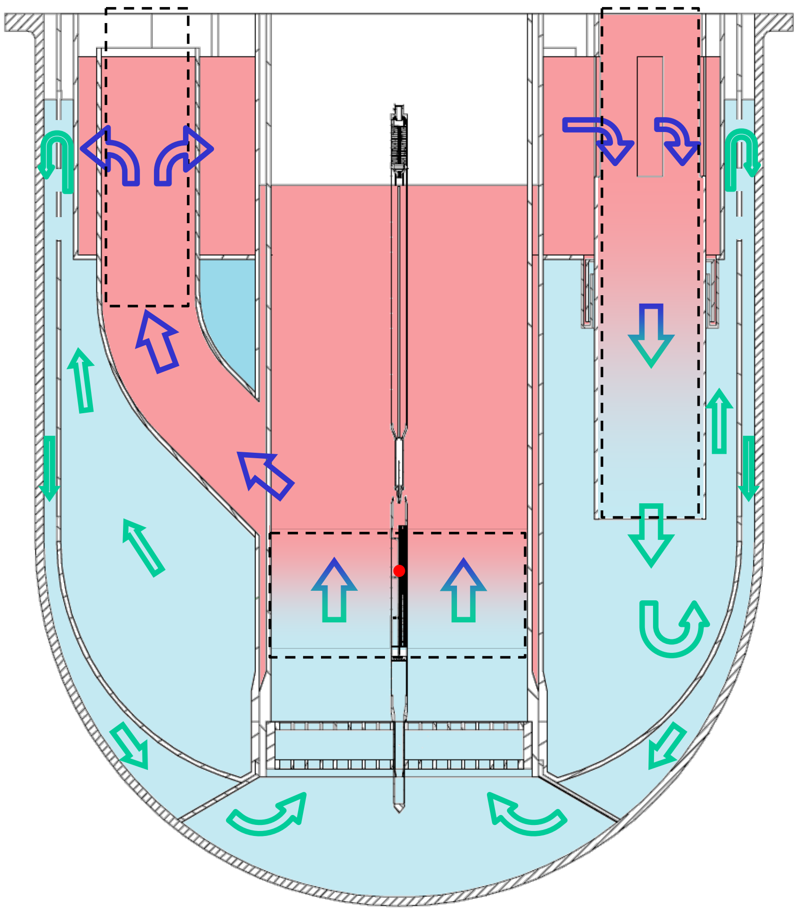
Duration: 36 Months (2019 – 2022)

Estimated Costs: 3,210,439.81 Euro

Requested EU Contribution: 2,247,229.76 Euro



ALFRED new design features



Main components: separated and extractable for out of vessel inspections & repair

Revised primary system configuration:
- separation of components based on safety classification
- minimization of T/H issues typical of FRs

"Cold" Lead temperature: limited to 400°C in the cold pool to ensure full compatibility with materials

Core: optimized and provided with a hot channel for qualification of future stages

Neutron irradiation: minimization of components subject to non-negligible neutron irradiation

Stages identification

	Stage 0 (Commissioning)	Stage 1 (Low Temp.)	Stage 2 (Medium Temp.)	Stage 3 (High Temp.)
Core inlet temperature (°C)	390	390	400	400
Core outlet temperature (°C)	390	430	480	520
Core thermal power (MW)	≈ 0	100	200	300

- ***Mitigation against lead corrosion at low / medium temperature:***
  - ***oxide layer working against the diffusion and loss of metal constituents***
  - ***moderate oxygen concentrations in the melt (~10^{-6}-10^{-8} wt.%)***
  - ***fairly stable for temperature below 450-480°C on austenitic stainless steels***
  - ***approximately uniform oxygen content in the whole coolant (sufficiently wide window: 10^{-7} wt.% plus/minus one order of magnitude).***
- ***Optimal oxygen concentration is a function of the minimum and maximum temperatures.***

PRO ALFRED project

Project title:

- ✓ *Preparatory activities for ALFRED infrastructure development in Romania*
  Focused on: feasibility studies for ELF and HELENA II, conceptual design for Hands-On, Meltin'Pot and Hub, licensing preparations, enhancing capacity (software tools, IT procurement), additional technical and economical evaluations, national E&T Programme definition, dissemination.

Funding:

- ✓ Romania national RDI budget in the frame of National Research, development and innovation plan 2015-2020, RDI Programme 5.5 for Generation IV reactor ALFRED.
- ✓ Competition call 2019, deadline for proposal July 2019
- ✓ Value: 2.5 million Euros

Status:

- ✓ Proposal submitted by RATEN as coordinator in partnership with University of Pitesti (UPIT), Polytechnical University of Bucharest (UPB), Institute for Physics and Nuclear Engineering (IFIN-HH), Institute for Economical Forecast (IPE) of Romania Academy.
- ✓ Evaluation steps closed;
- ✓ Contract negotiation just finished; signature expected before September 15, 2019.

Project duration: 15 months, September 2019 – November 2020.

ATHENA & ChemLab construction project

Project title:

- ✓ *"ALFRED – step 1, experimental research support infrastructure: ATHENA (Lead pool type facility) and ChemLab (Lead chemistry laboratory)"*, submitted by RATEN, Mioveni site

Funding:

- ✓ EU structural and cohesion funds for Romania, Competitiveness Operational Program
- ✓ Priority axis: Large RDI infrastructures for public research organizations and universities in less developed regions.
- ✓ Competition: 2018 call, deadline for proposal submission January 2019
- ✓ Value: 20 millions Euros

Status:

- ✓ Project Proposal submitted: January 2019
- ✓ Administrative and eligibility evaluation by the Intermediate Organism (OI) of Ministry for Research and Innovation closed: August 2019
- ✓ Final evaluation by the Management Authority (AM) of Ministry for European Funds: ongoing
- ✓ Expected contract signing: before end of 2019.

Project duration: 4 years, completion 2023

Thank you for your attention

РОСАТОМ

ГОСУДАРСТВЕННАЯ КОРПОРАЦИЯ ПО АТОМНОЙ ЭНЕРГИИ «РОСАТОМ»

НИКИЭТ им.Н.А.Доллежаля

Reactor BREST-OD-300: current state of development and justification

Andrei V. Moiseev (JSC NIKIET, Moscow, Russia)

25th LFR Prov. SSC Meeting

Ansaldo Nucleare S.p.A.,

Bd. Dacia, nr. 65, Ap. 2, Et.1

Sector 1 - 010407 Bucharest (Romania)

8-10 October 2019

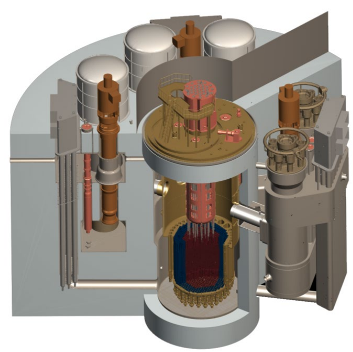
Target requirements for commercial reactors with lead coolant

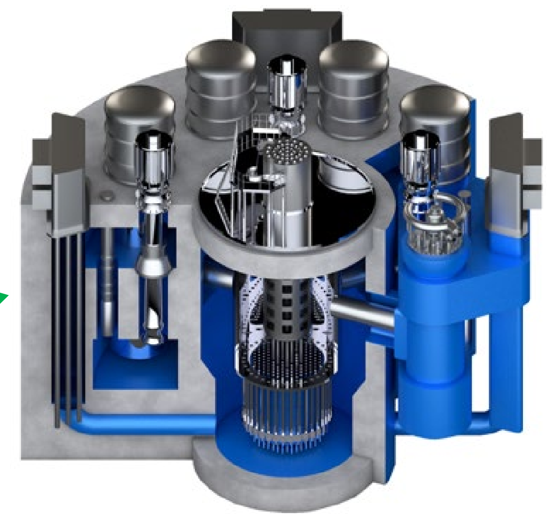
Elimination of NPPs accidents requiring population evacuation, much less resettlement

Commercial reactor

BREST-OD-300

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





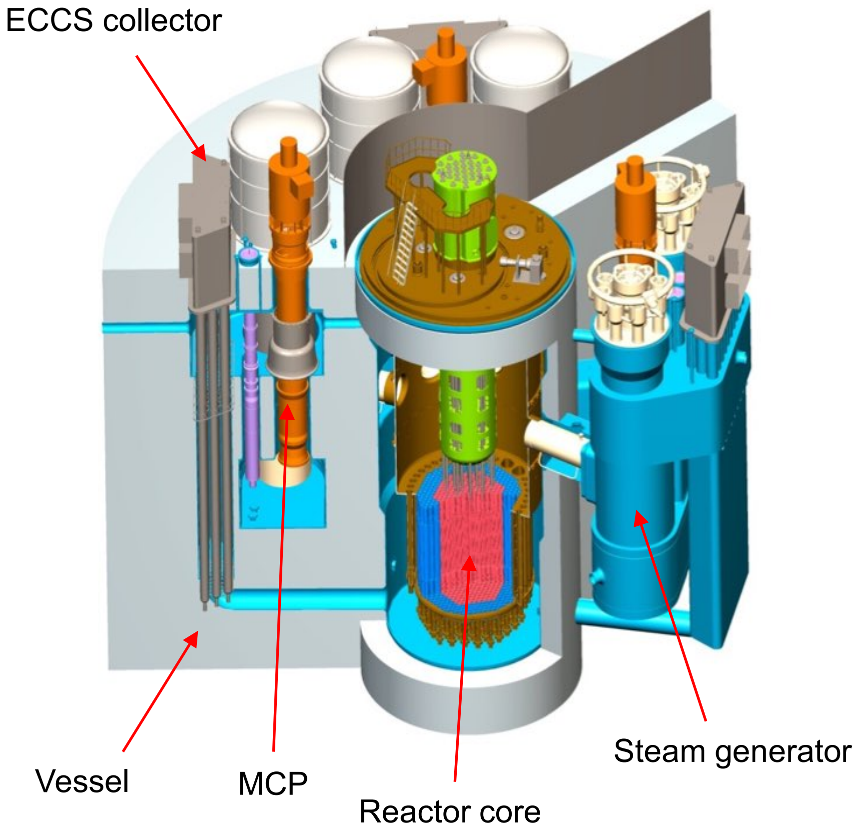
Progressive approximation to radiation-equivalent (in relation to natural raw materials) RW disposal – at the operating stage after development of fuel with MA

Marketability against other types of power generation - demonstration of technology potential

Technological enhancement to non-proliferation regime - no blanket, no Pu extraction during SNF reprocessing, on-site NFC, no uranium enrichment required

Design concept of BREST-OD-300

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



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

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

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

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

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

Specific indicators of BREST-OD-300

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

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

Current status of BREST-OD-300 development

We are here
TRL 5 (6)

Research activity

Development activity

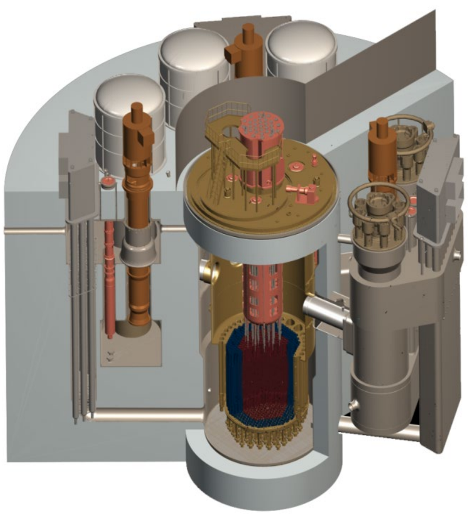
- Preliminary stages (technical proposal, conceptual design)
- Specification
- Engineering design (final solutions)
- Working design documentation, operational documentation, pilot model
- Correction of documentation (character O, O1)

Readiness for serial production or manufacturing of pilot product (elements of the BREST-OD-300 reactor)

- Preliminary research and solutions

- Mockup tests
- Technology development
- Verification of software tools

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



PDEC: stages of construction

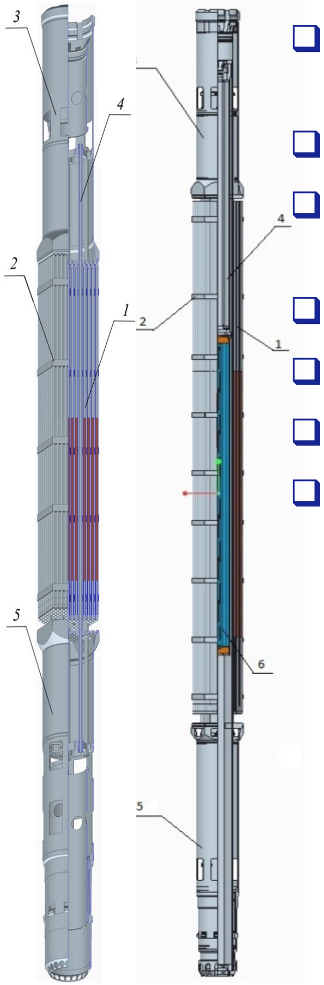
РОСАТОМ

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

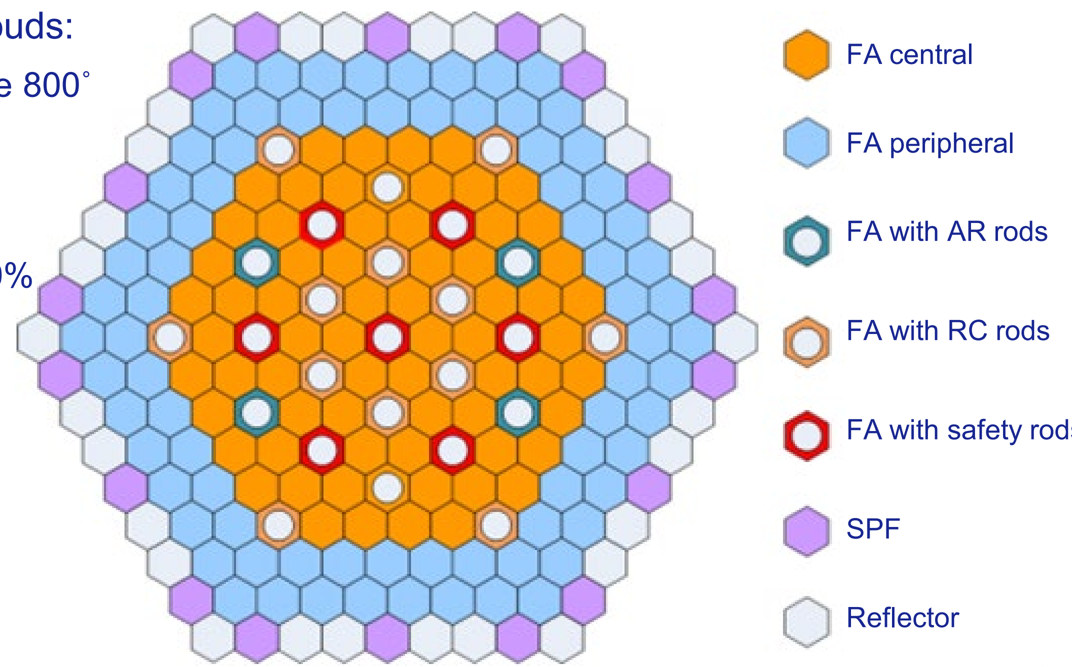
- Buildings and structures of the **fuel fabrication module (FFM)** (I stage)
- Buildings and structures of the **BREST-OD-300 power unit** (II stage)
- Buildings and structures of the **reprocessing module** (III stage)
- Turning of FFM for **fuel re-fabrication** from SNF reprocessing products (IV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

Reactor core

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



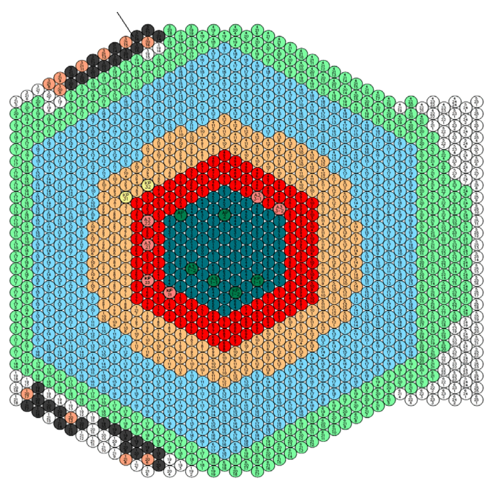
1 – Fuel pin; 2 – Spacer grid; 3 – FA head;
4 – Supporting tube; 5 – FA tail; 6 – Control rods



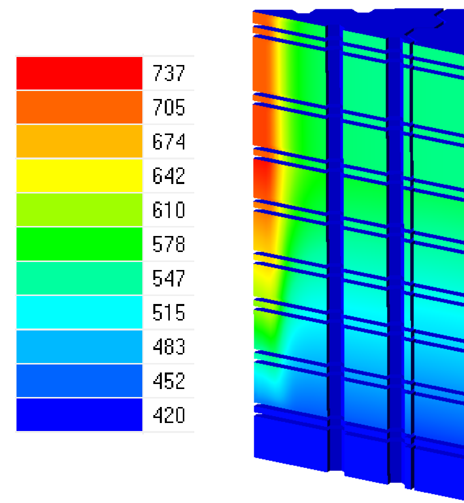
Core justification

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

LMC-stand

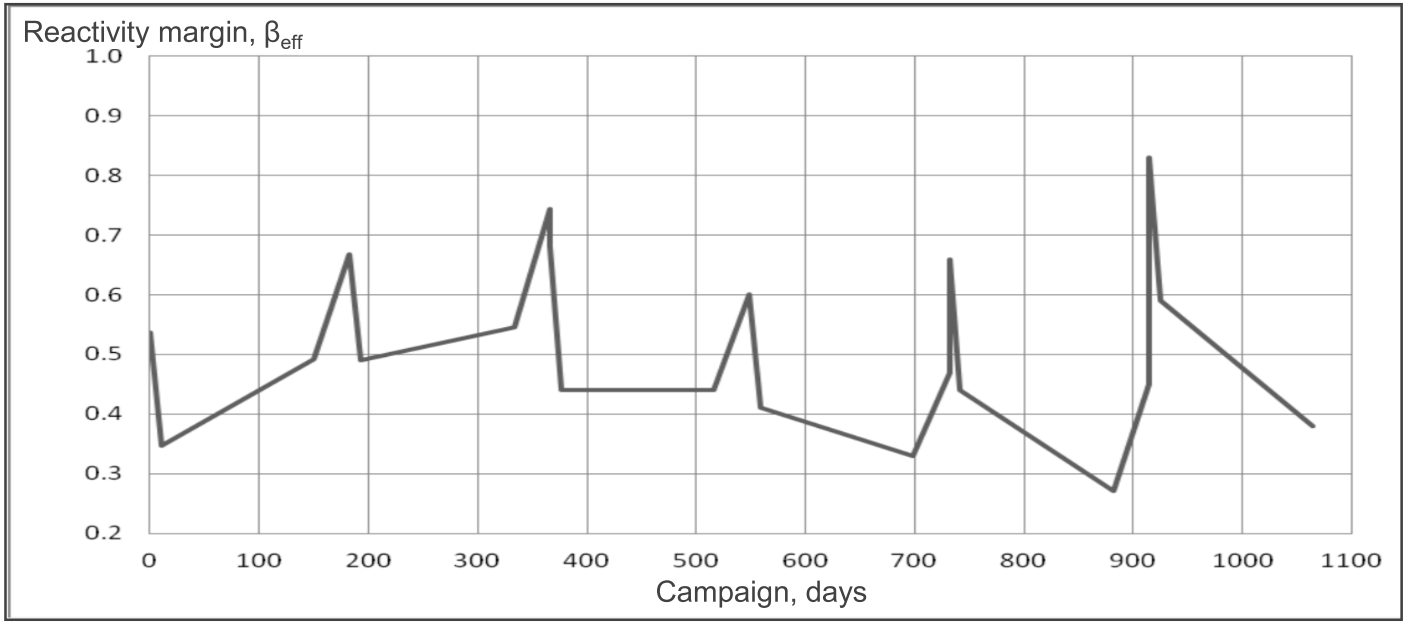


Benchmark model of BREST core. BFS-113 assembly with nitride



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

Neutron-physical characteristics of the core



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

Justification of core elements

Full-scale FA mock-up

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

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

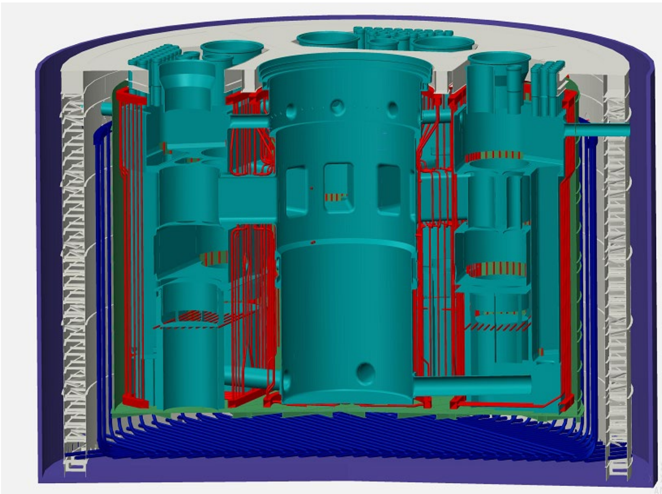
- Vibrometric and vibrostrength characteristics of FA mockups were obtained

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

- Reactor tests of the experimental FAs are on the way; the fuel pins of a discharged experimental FA are on the postreactor investigations.
Max. burn up - 7.4% h.a.

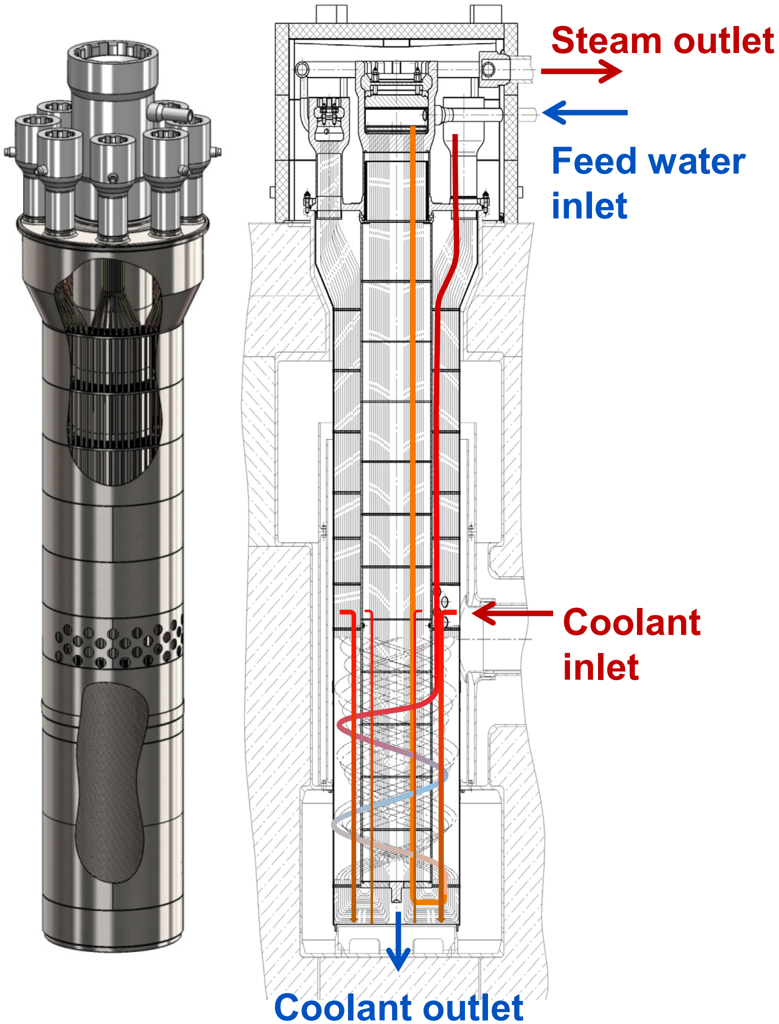
FA mock-up with a retort for testing

Reactor vessel



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

Steam generator (1/2)



- Required thermal hydraulic steam generator parameters have been substantiated. Stable operation limit has been determined – not less than 15 % of the flow rate
- Cumulative reports with material properties have been issued
- Structural integrity of SG elements has been computationally verified for all modes of operation
- Heat-exchange tubes have been put into manufacture
- Absence of dependant failure in case of one tube rupture has been experimentally demonstrated. With the passage of steam bubbles through the core, the maximum realizable reactivity effect $\Delta\rho_{max} < 0{,}5\ \beta_{eff}$
- Neutral water chemistry has been substantiated allowing for reduction of deposition formation during SG operation. A technology for heat-exchange SG tube cleaning has been developed

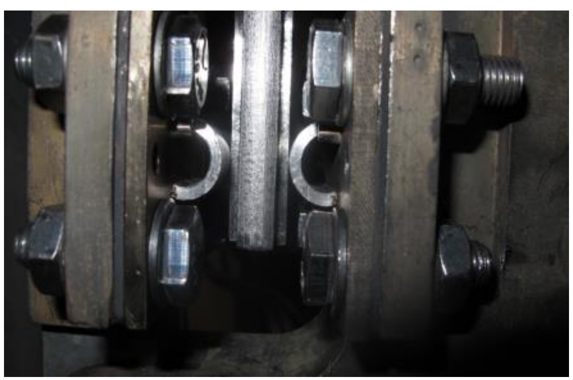
Steam generator (2/2)

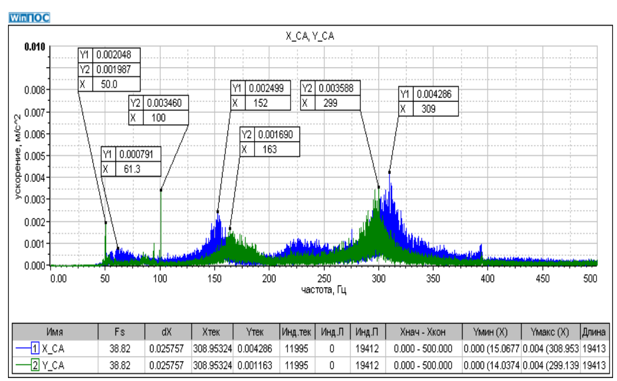
➢ Confirmed the fulfillment of thermal-cyclic strength conditions for heat-exchange tubes and welds to a tube sheet

➢ Experiments have been carried out to substantiate the increase in corrosion under the conditions of water, steam and lead coolant

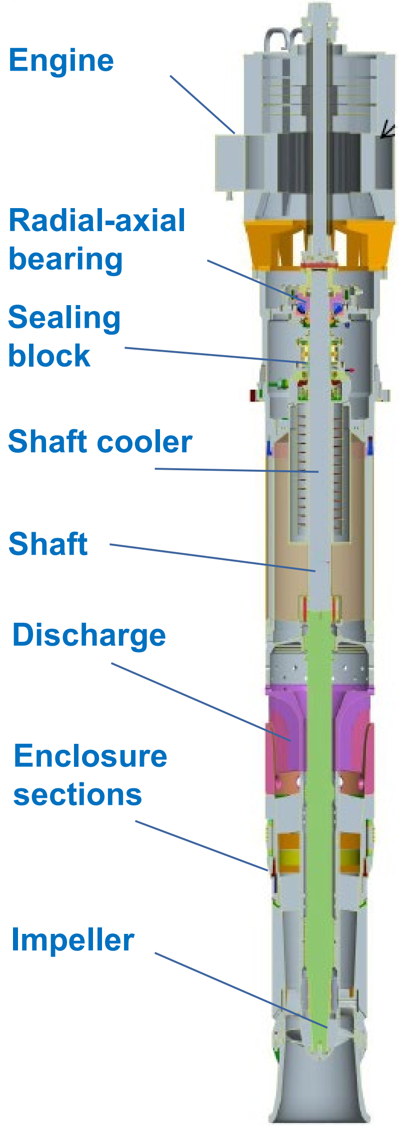
➢ Negligible influence of lead on creep rate in lead under loads typical for steam generator has been demonstrated

➢ A series of tribological tests on HET-spacer grid friction joints has been carried out. Physical and mechanical model has been developed. The calculations using the model has confirmed the 30-year life. A bench is developed for full-scale study of vibration characteristics

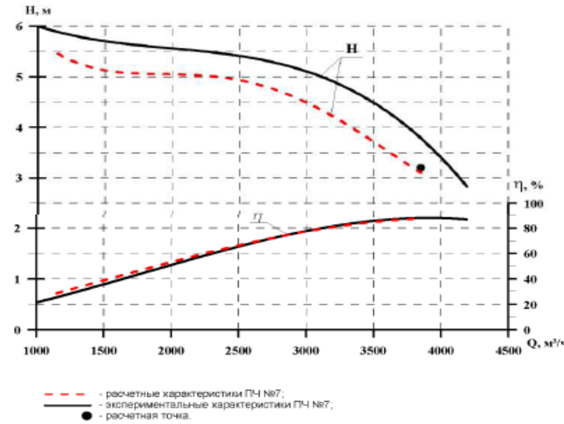




Main circulation pump



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



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

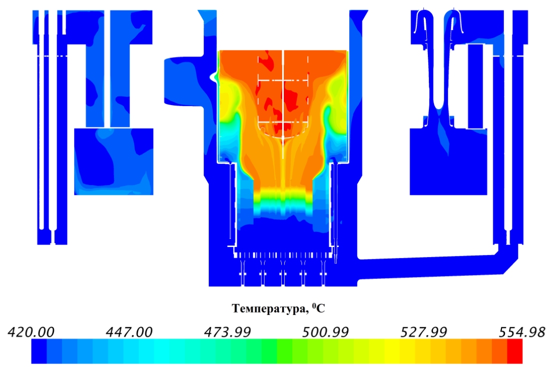
Other elements

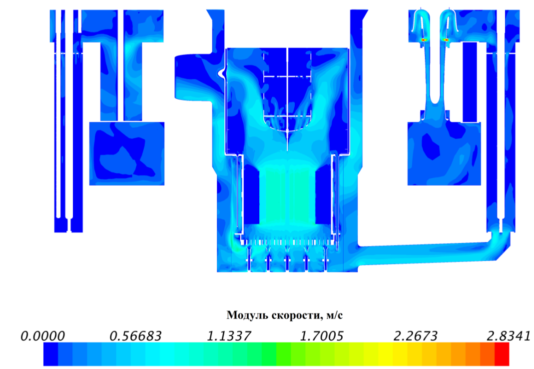
- CPS actuator prototype tests have been competed with positive outcome
- Primary converters of parameters of the primary circuit are manufactured and tested
- Engineering design of the reactor automated inspection and control system has been developed; a bench had been developed, which was used to demonstrate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

Research of output of FP and AP out of the coolant

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

Thermal hydraulic calculations justifying design solutions and safety





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

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

BREST-OD-300 safety requirements

The most conservative scenarios are considered:

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

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

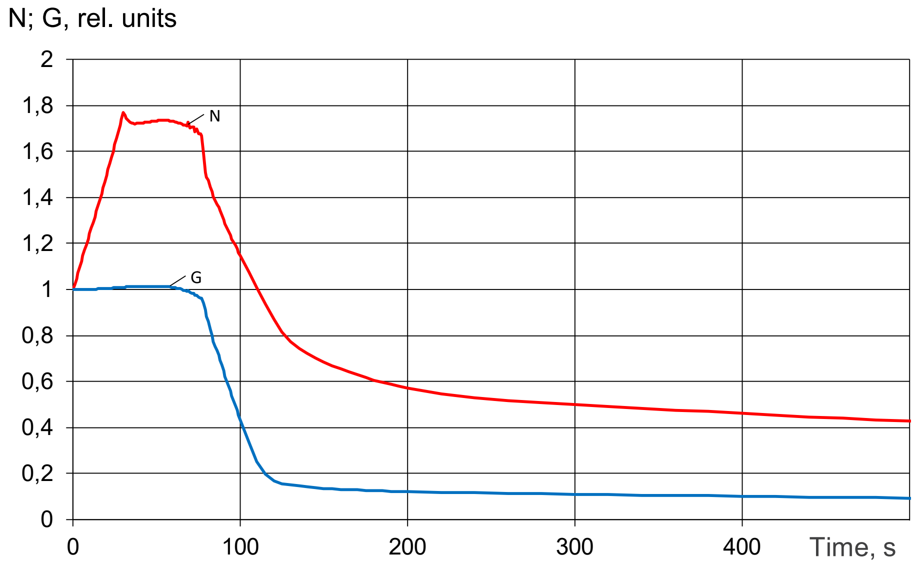
- Elimination of fuel (> 2700 °C) and cladding melting (~ 1500 °C).
- Elimination of coolant boiling (~ 1750 °C).
- Maintaining the integrity of the circulation loop.

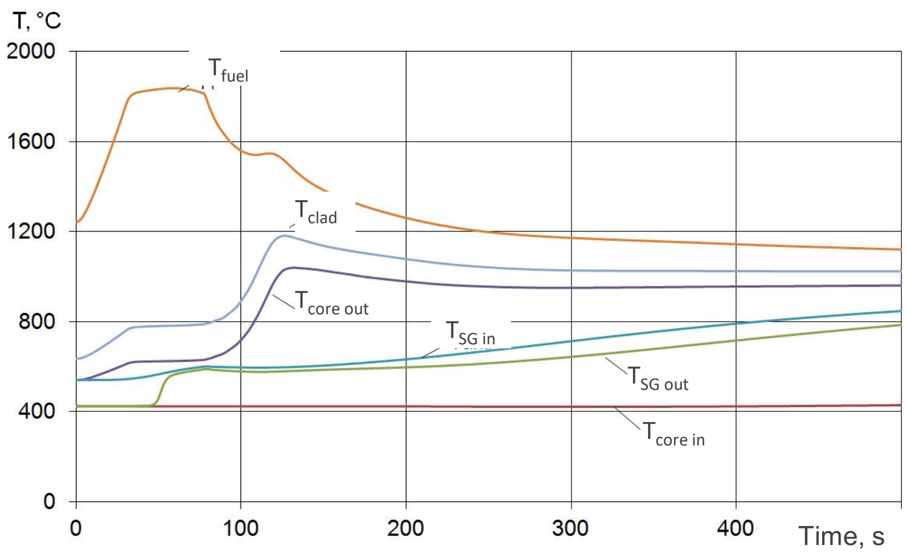
Limit of safe operation by temperature of fuel rod cladding: Maximum temperature of fuel rod cladding 740 °C – 1000 °C depending on the time

Requirements for off-site radioactivity emissions:

- Eliminate the need for evacuation and resettlement of the population.

Entering of the full reactivity margin at nominal power





Scenario:

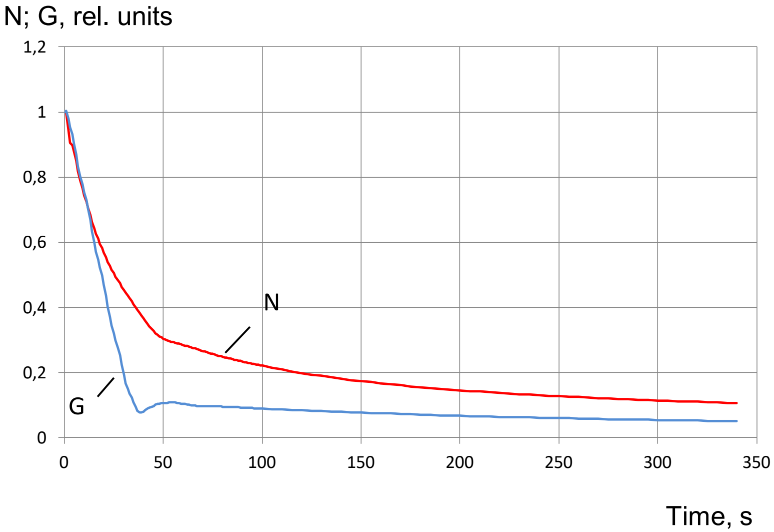
- initiating event – entering of the **full reactivity margin at nominal power 0.65 β_{eff}** with realistic design speed (in 30 seconds)
- active protection systems fail (total number of failures – 11, probability of realization $2.8*10^{-9}$)
- pump shutdown on setpoint temperature of the coolant inlet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

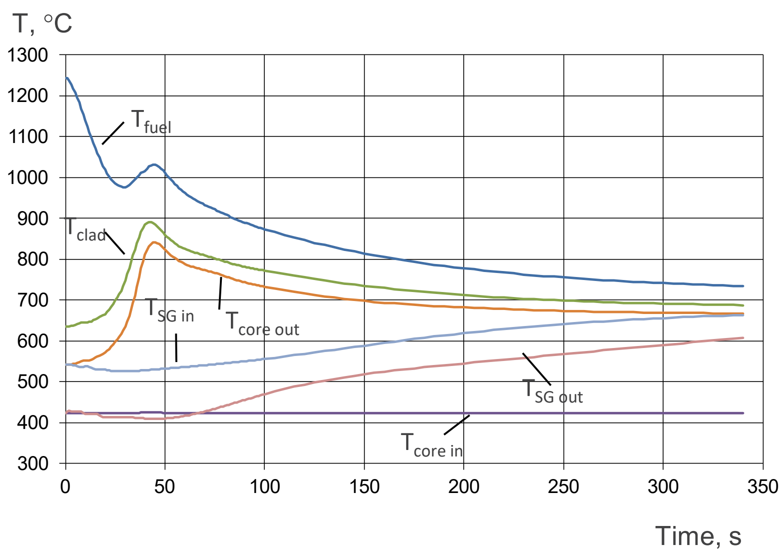
Outcome:

- melting of fuel and cladding, boiling of the coolant - does not occur (the void effect of reactivity is not realized), circulation circuit integrity ensured
- the level of possible damage of fuel rods by type of gas leakage – 7 %
- ***emissions into the atmosphere for all normalized radionuclides do not reach the reference level per day (no more than $6,14 \cdot 10^{10}$Bq)***

Probability of such a scenario $2.8*10^{-9}$

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





Scenario:

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

Outcome:

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

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

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

Regulatory acts and standardization

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

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

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

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

Prospects of lead-cooled reactors

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

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

ЭКСПЕРТНЫЙ СОВЕТ ПО АТТЕСТАЦИИ ПРОГРАММНЫХ СРЕДСТВ
ПРИ РОСТЕХНАДЗОРЕ

АТТЕСТАЦИОННЫЙ ПАСПОРТ ПРОГРАММНОГО СРЕДСТВА

Регистрационный номер 405 от 08 декабря 2016 года

Настоящий аттестационный паспорт устанавливает назначение и область применения программного средства

«MCU-BR с библиотекой констант MDBBR50»,

которые указаны в разделе 2 Приложения к настоящему аттестационному паспорту.

Аттестационный паспорт предоставлен

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

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

Срок действия аттестационного паспорта до 08 декабря 2026 года

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

ETSON | EUROPEAN TECHNICAL SAFETY ORGANISATIONS NETWORK

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

РОСАТОМ

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

РОСАТОМ

ТОПЛИВНАЯ КОМПАНИЯ РОСАТОМА
ТВЭЛ

НИКИЭТ им.Н.А.Доллежаля

ГНЦ РФ - ФЭИ

АТОМПРОЕКТ

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

Сибирский химический комбинат
ПРЕДПРИЯТИЕ ГОСКОРПОРАЦИИ РОСАТОМ

ЦКБМ

ЭЛЕМАШ
ПРЕДПРИЯТИЕ ГОСКОРПОРАЦИИ "РОСАТОМ"

НЗХК
ПРЕДПРИЯТИЕ ГОСКОРПОРАЦИИ «РОСАТОМ»

ИБРАЭ

ОАО "Машиностроительный завод "ЗИО-ПОДОЛЬСК"

ОАО "Инжиниринговая компания "ЗИОМАР"

КБСМ
КОНСТРУКТОРСКОЕ БЮРО СПЕЦИАЛЬНОГО МАШИНОСТРОЕНИЯ
АКЦИОНЕРНОЕ ОБЩЕСТВО

НИИАР

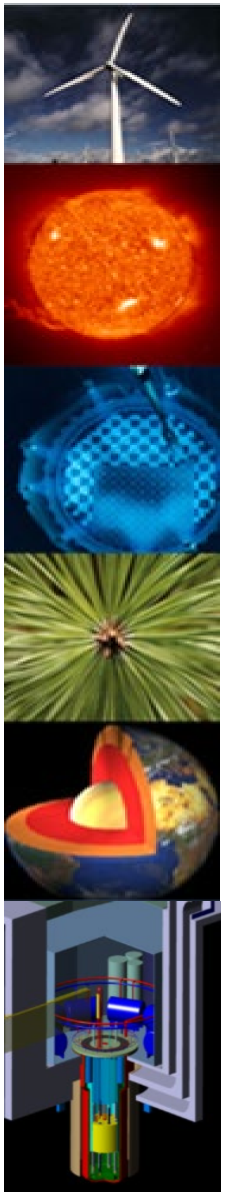
РОСЭНЕРГОАТОМ
БЕЛОЯРСКАЯ АЭС

ОКБМ АФРИКАНТОВ
ПРЕДПРИЯТИЕ ГОСКОРПОРАЦИИ «РОСАТОМ»

ИРМ
ИНСТИТУТ РЕАКТОРНЫХ МАТЕРИАЛОВ

Conclusion

1. Based on the concept of inherent safety, technical solutions have been developed.
2. The design approaches to the reactor equipment have been experimentally justified based on mockups of the equipment and its components and by means of computational justification with regard for the lead coolant effects; prototypes are being tested.
3. The reactor core design approaches have been validated by positive results of irradiation experiments, hydraulic and vibration experiments, analysis of neutronics using the certified codes.
4. The thermal-hydraulic calculations performed using CFD codes have shown that during anticipated operational occurrences with multiple failures superimposed, safe operation limits in terms of fuel and cladding temperatures are not exceeded and the localizing function of the reactor unit vessel is ensured.
5. The results of the radiation safety analysis have confirmed the implementation of target indicators, including no need for evacuation and resettlement of the public outside the site during anticipated operational occurrences with multiple failures.
6. The experiments formed the basis of the developed regulatory framework.
7. The BREST-OD-300 unit design received a positive conclusion of the Glavgosexpertiza, is in the process of licensing with Rostekhnadzor.
8. BREST-OD-300 is a pilot and demonstration power unit, it must complete R&D and produce results that cannot be obtained without creating it.



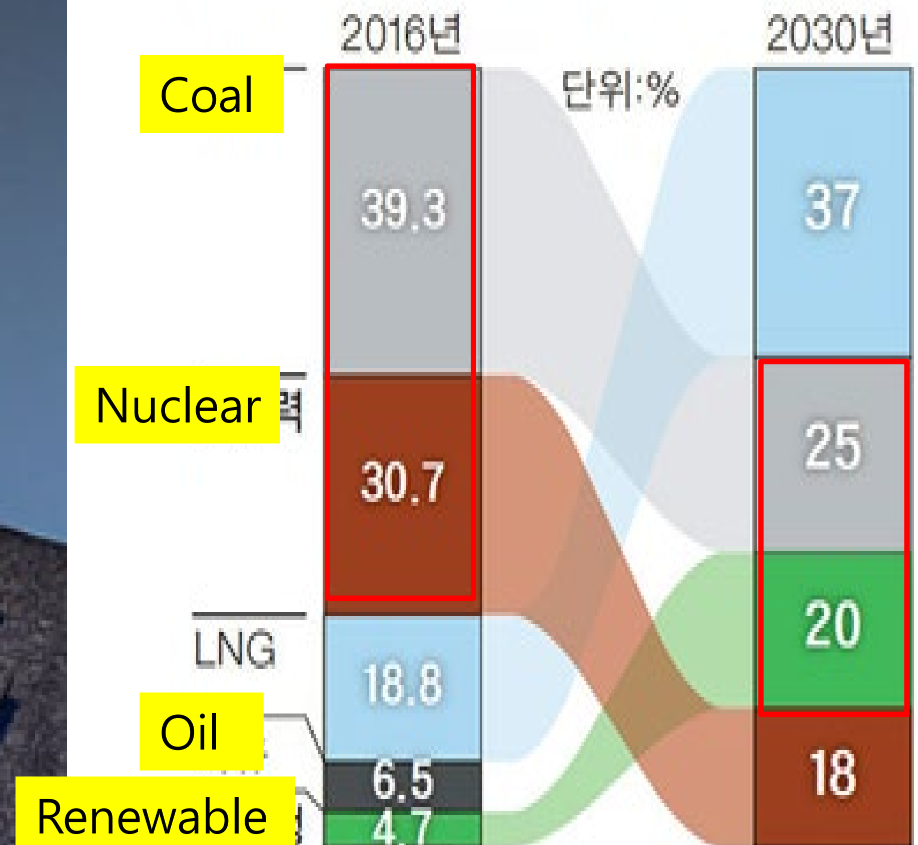
National Status on LFR Development: ROK

Il Soon Hwang
Professor Emeritus, SNU
Chair Professor, UNIST

I. Nuclear Phase-out
II. Spent Nuclear Fuels
III. Micro Modular Reactor
IV. Summary

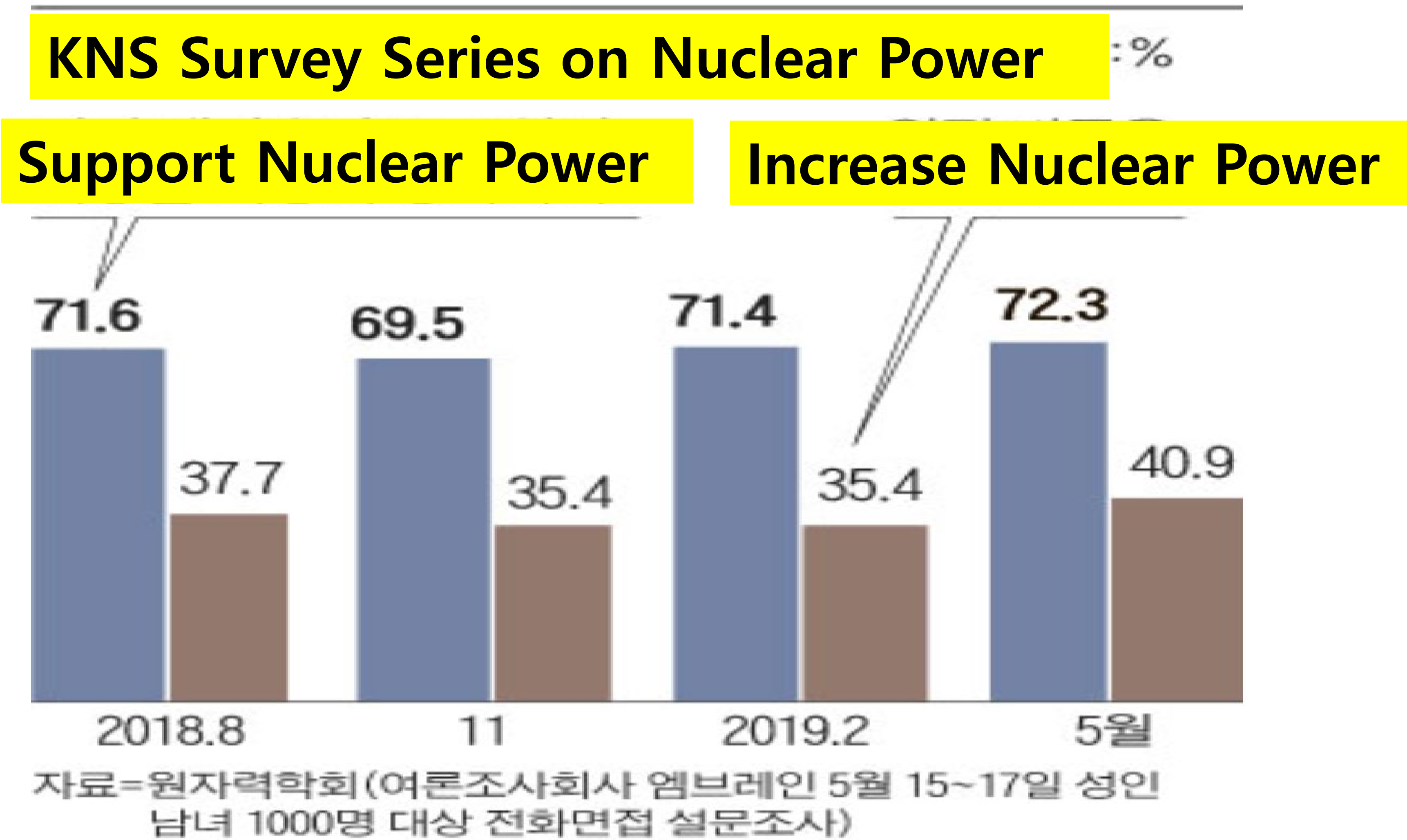
I. Nuclear Phase-out: Moon's Administration

Renewable 4X & LNG 2X Up



From Moon's Election Campaign

I. Nuclear Phase-out: Korean Public Support>2/3

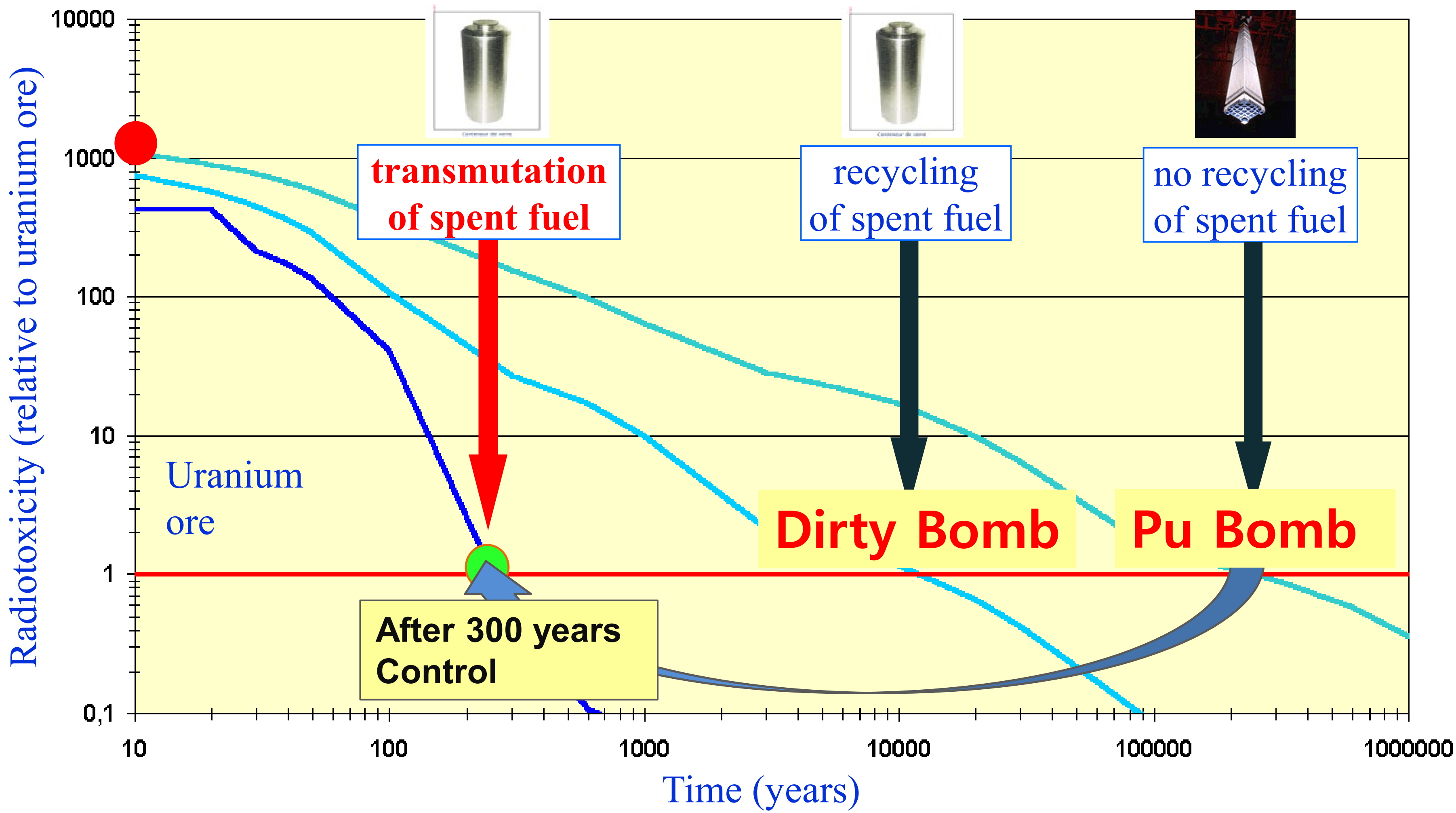


I. Nuclear Phase-out to SMR & MMR

문미옥 과학기술정보통신부 차관이 축사를 전했다. | 사진: 김경채

TerraPower-20190926-UNIST

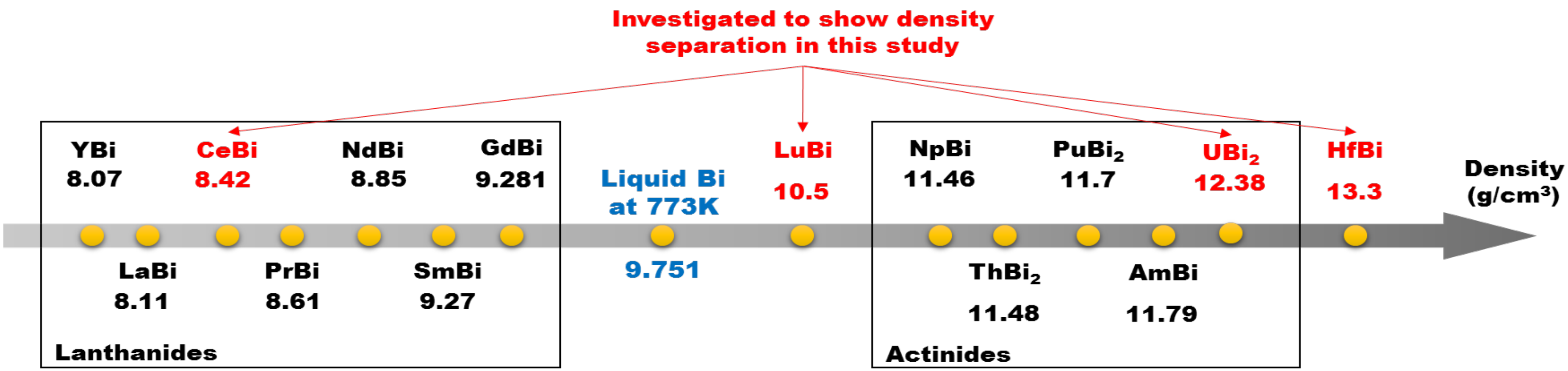
II. SNF Burning by Pyro & SFR



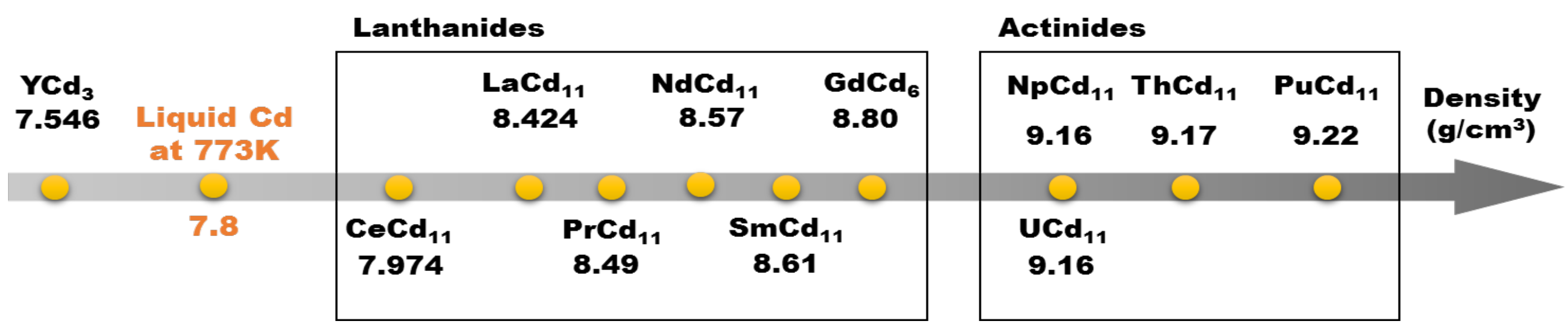
Prof. H.A. Abderrahim

Pyroprocess (KAERI) & PyroGreen (Academia)

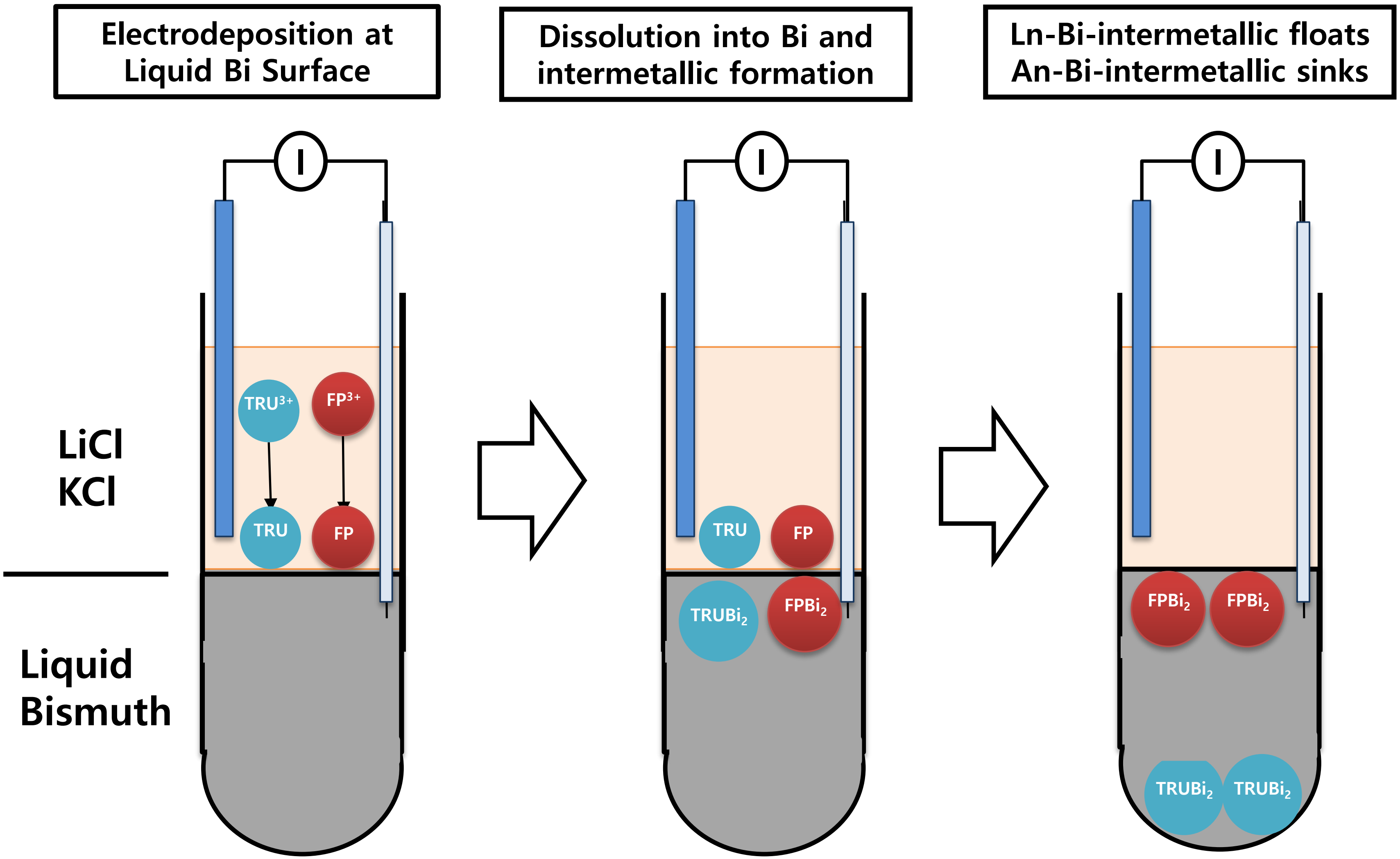
Densities of Bi, Actinides and Lanthanides(FP)



Densities of Cd, Actinides and Lanthanides(FP)



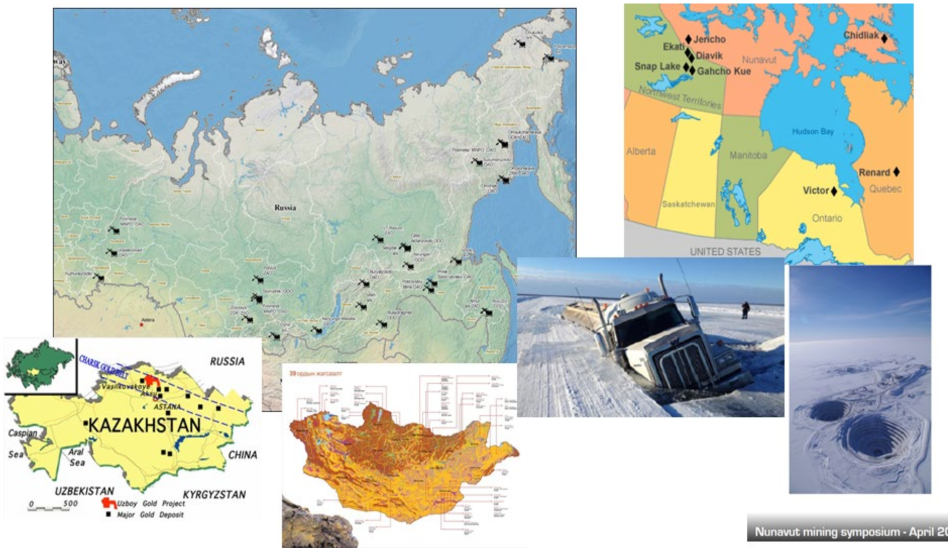
PyroGreen to achieve DF Goal & Eliminate HLW

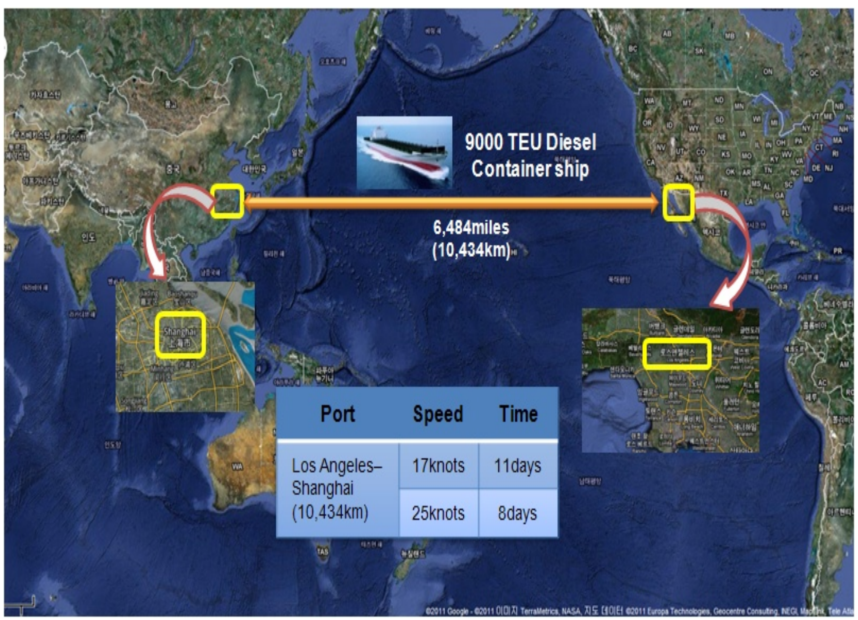


III. SMR’s for Marine Application

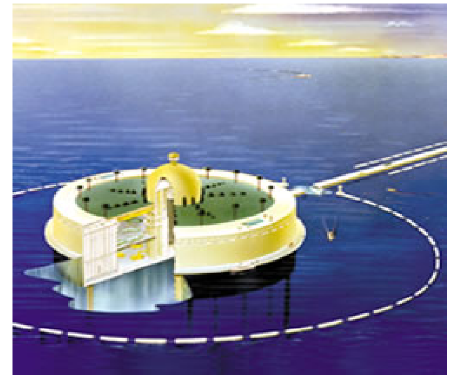
- Yamal & Gyda LNG Shipping to East Kamtchatka by Nuclear Icebreaker

III. SMR & MMR



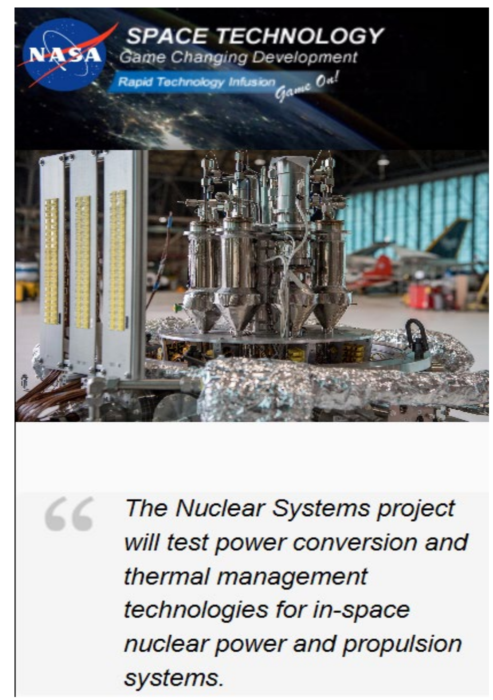


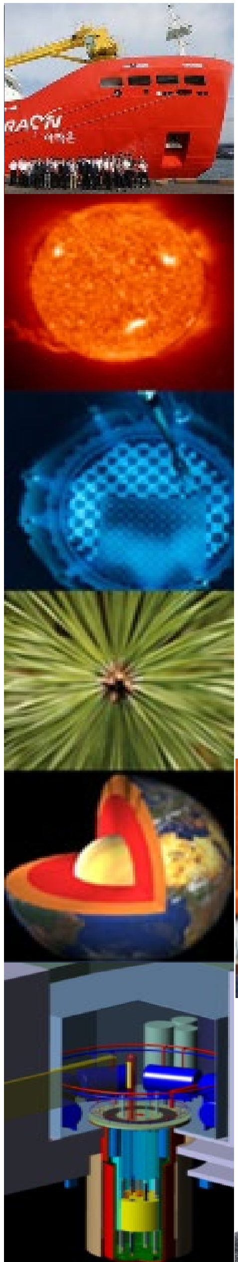
Port	Speed	Time
Los Angeles–Shanghai (10,434km)	17knots	11days
	25knots	8days



Nuclear Desalination and District Heating

Artist's rendering of a proposed island "nuplex" using nuclear energy to generate electricity and desalt seawater for irrigating crops -ORNL





III. Micro Nuclear Energy Research & Verification Arena (MINERVA)

UNIST, USN, KHU, Moojin Ltd, KAIST, SNU, KINGS

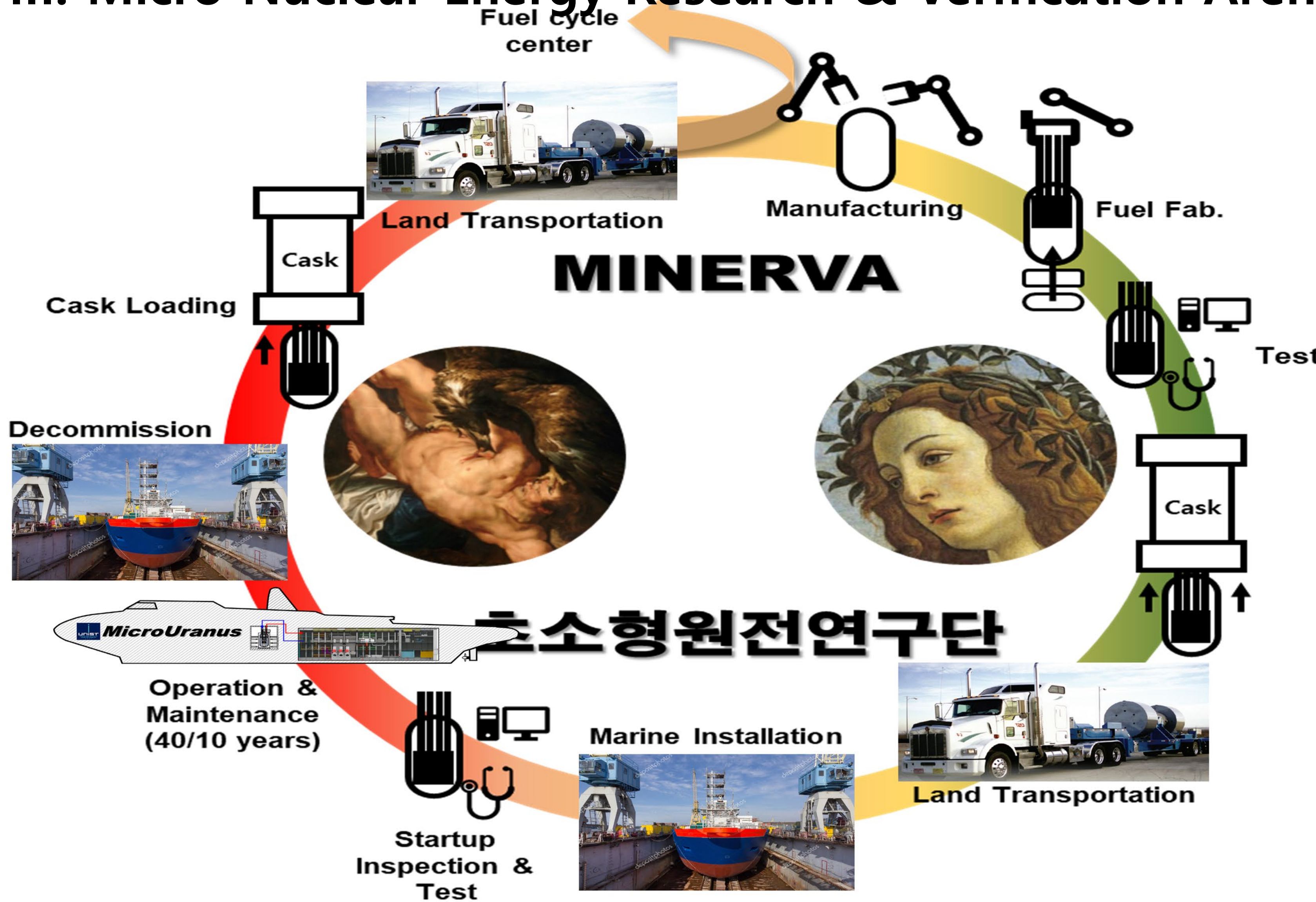
2019 ~ 2022

Korea National Research Foundation

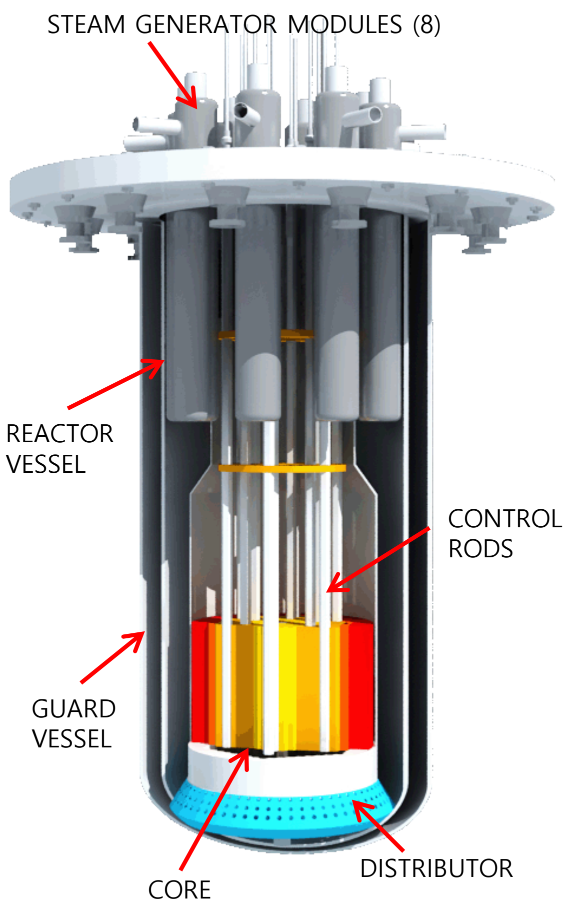
III. Micro Reactor

- UK NUVIA for DECC in 2016
  - Power < 30 MWe
- Breakthrough Institute in 2018
  - Power < 10 MWt
- US DOD Roadmap, 2018
  - Power< 10 MWe
- Westinghouse eVinci in 2019
  - Land transportability of a completed reactor
- MicroUranus in 2019
  - Land transportability using Shipping Cask

III. Micro Nuclear Energy Research & Verification Arena



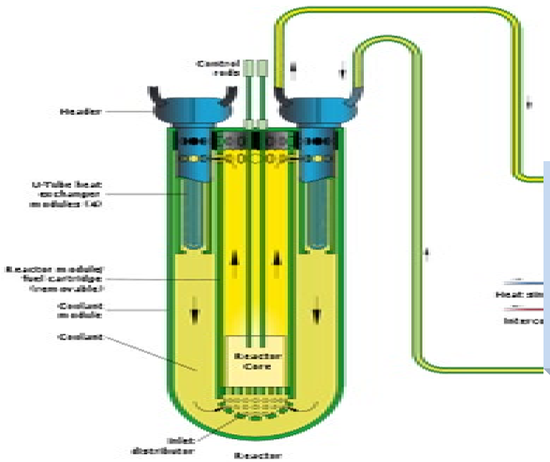
III. URANUS: LBE-cooled Passive SMR Design



- Thermal-hydraulic design parameters in normal operation

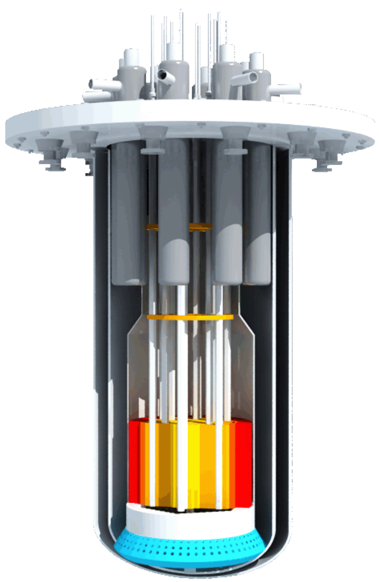
System	Parameter	Value
Primary	Average core power density	19.42 MW/m^3
	Average linear heat density	8.57 kW/m
	Reactor coolant flow rate	4928 kg/s
	LBE Coolant temperature	300/440 °C
	Peak fuel centerline temperature	756 °C
	Peak cladding OD temperature	470 °C
	Thermal center height difference	4.91 m
	Total pressure drop	9132 Pa
	Average flow velocity at core outlet	0.29 m/s
	Guard vessel outer diameter	4.45 m
	Core power during normal operation	100 MW_{th} (~40 MW_e)
Secondary	Number of units	8 E/A
	Feedwater/steam temperature	277/393 °C (superheated)
	Pressure	8.01 MPa
	Steam flow rate in normal operation	52.68 kg/s
	Estimated ideal efficiency*	39.7%

III. Micro Nuclear Energy: Lead Fast Reactor

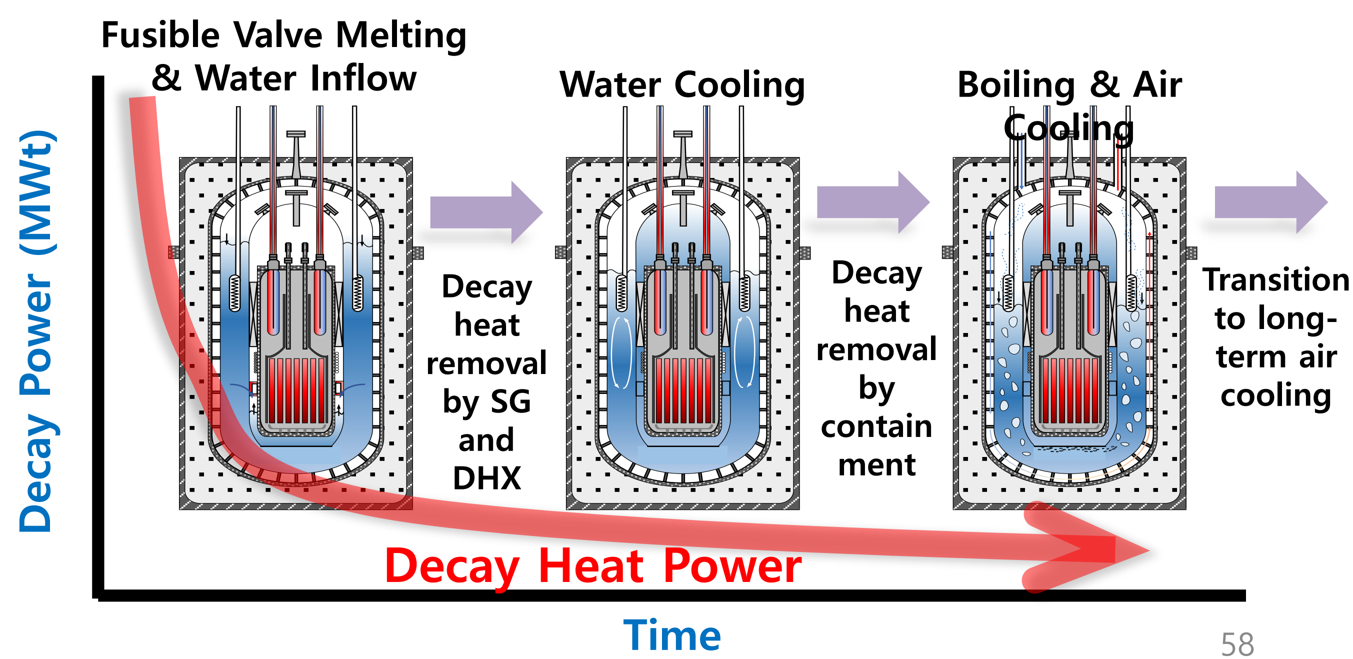


Risk of PEACE

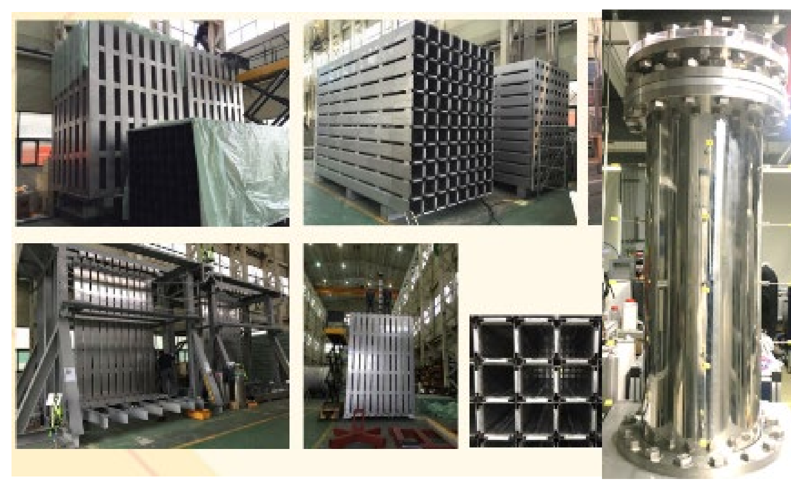
- Chemically Stale / Absorb Radioactivity / Boiling at 1,670 C
  - Coolant Leak / Radiation Leak
  - Hydrogen / Steam Explosion
- Meltdown & Spread / Emergency Freezing
  - Melt-down / Recriticality
  - Solidification / Source-term



MicroUranus
(MINERVA)



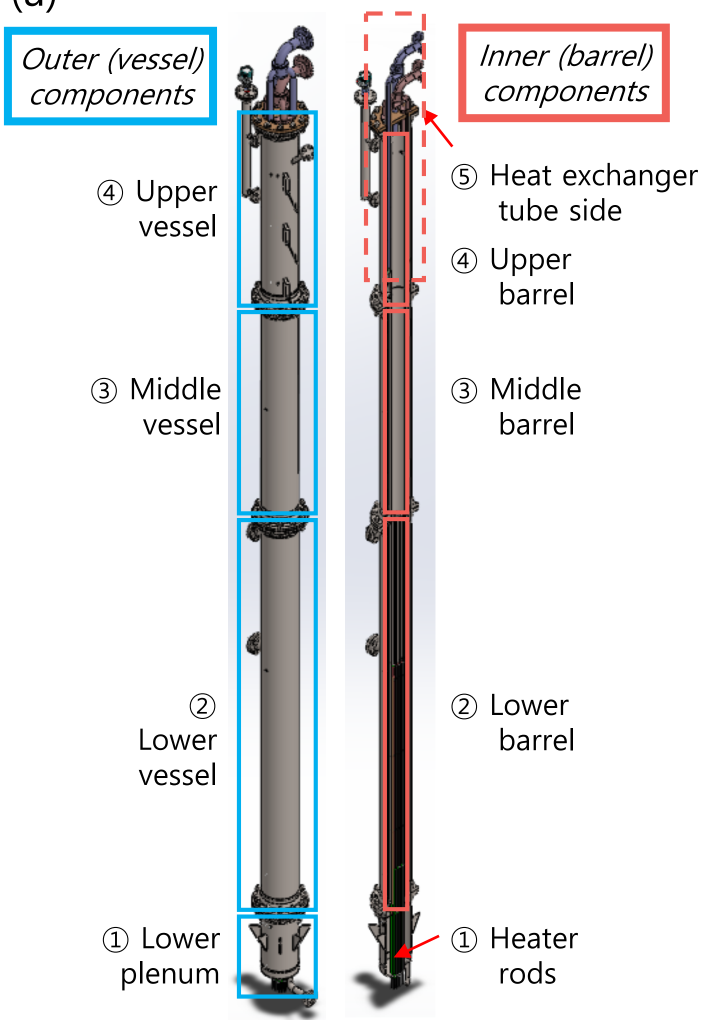
III. Micro Nuclear Energy Verification Arena (MINERVA)



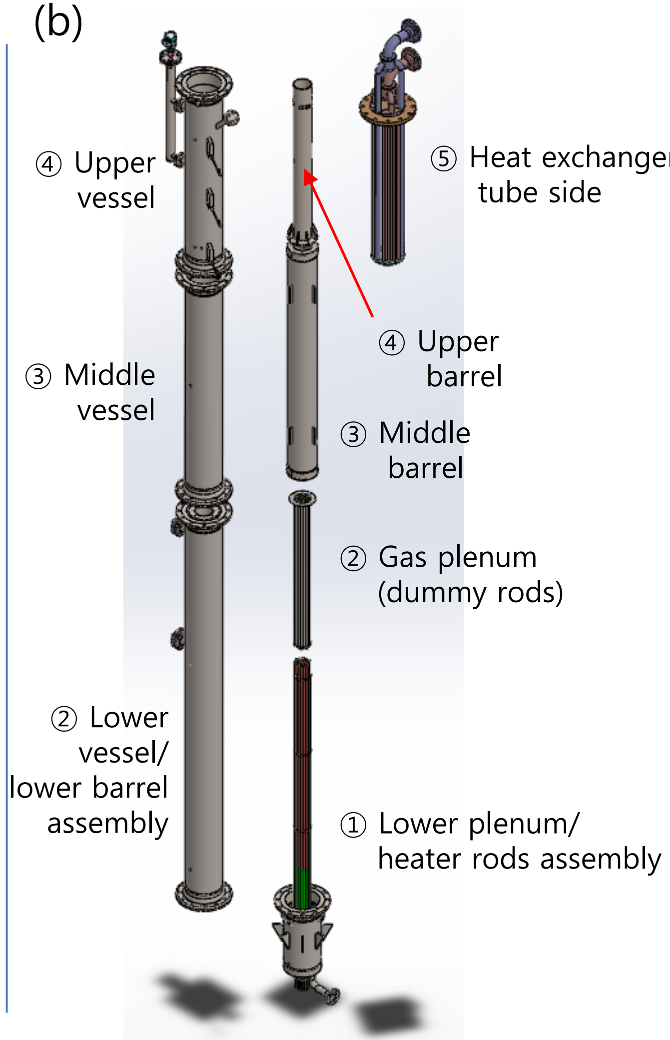
III. Pool-type URANUS integral test facility: PILLAR

- PILLAR design in detail

(a)



(b)

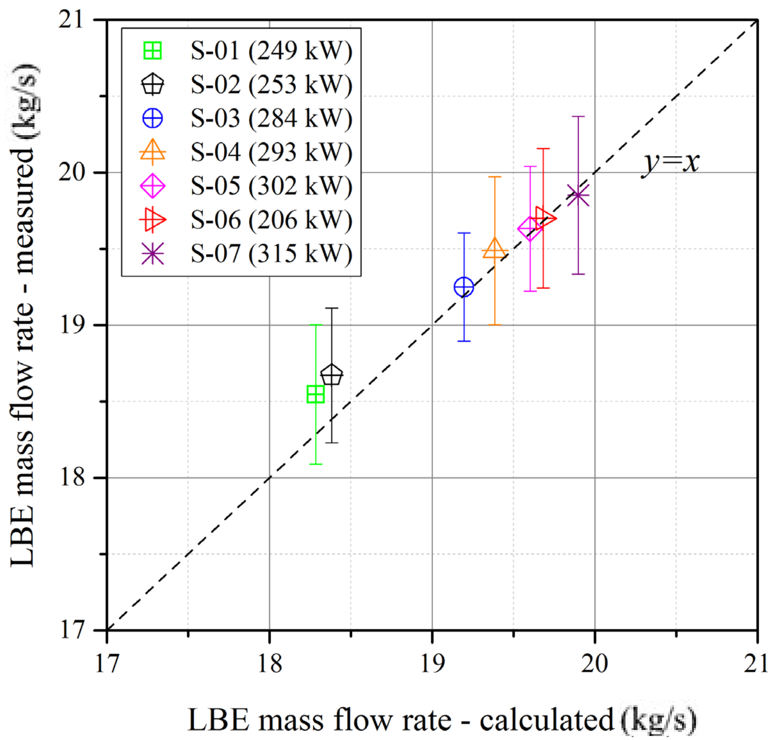


(c)

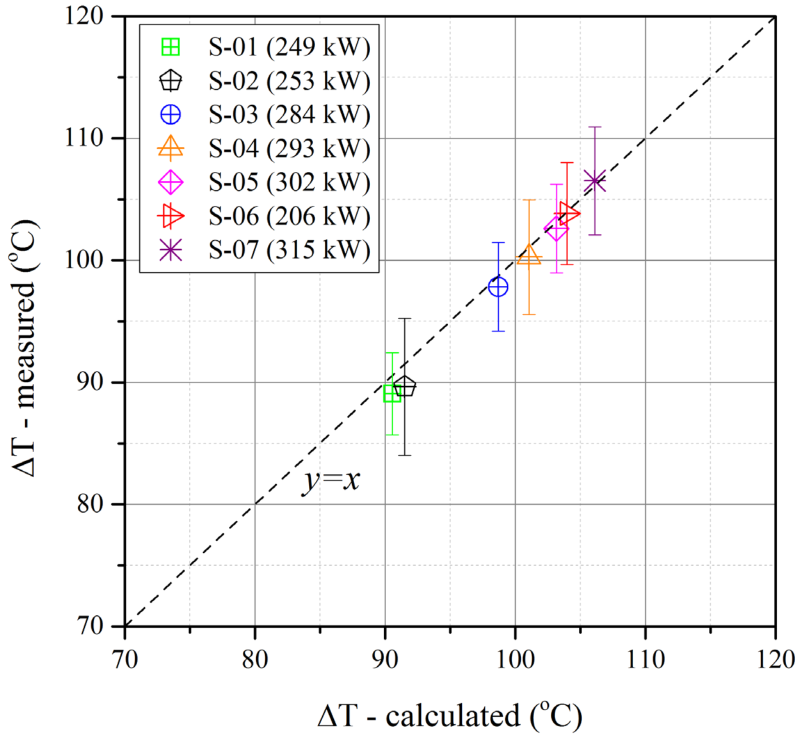
III. Code Validation by PILLAR Experiments

1D STH code validation by integral pool natural circulation test results

- Modeling results



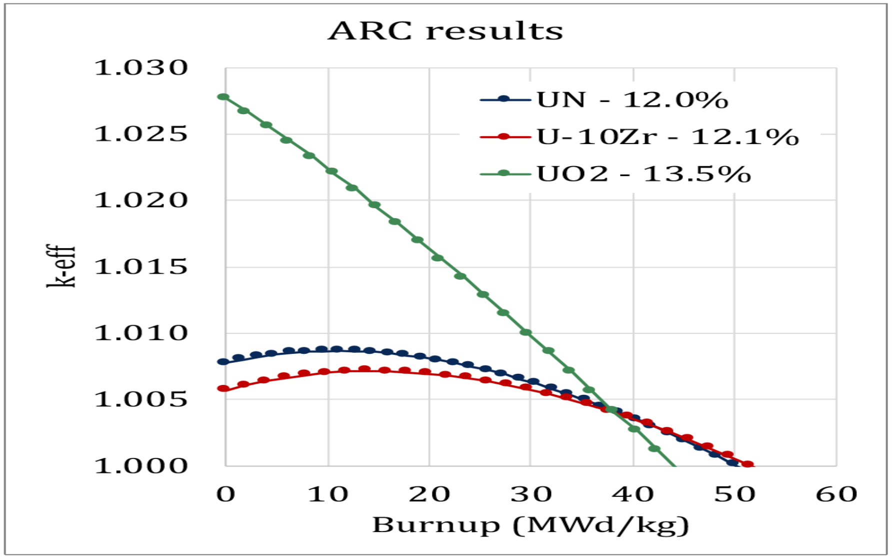
▲ Measured mass flow vs. calculated mass flow

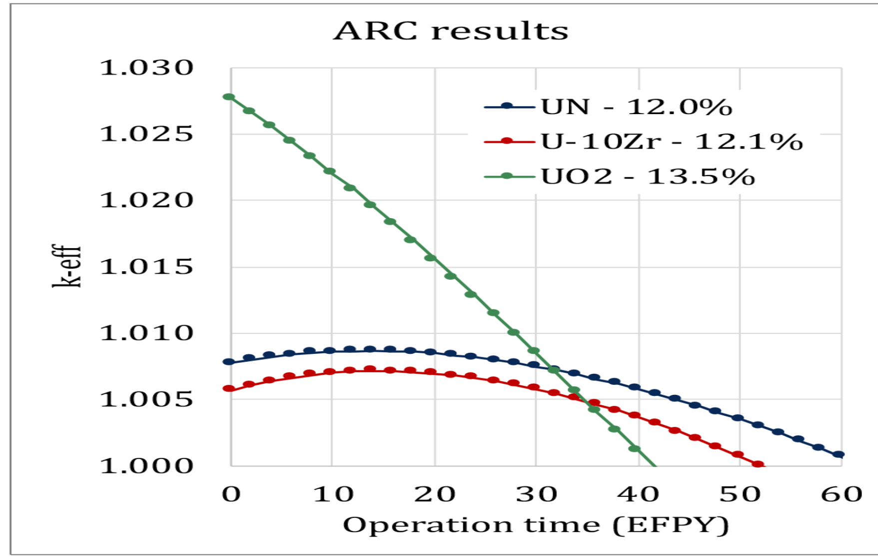


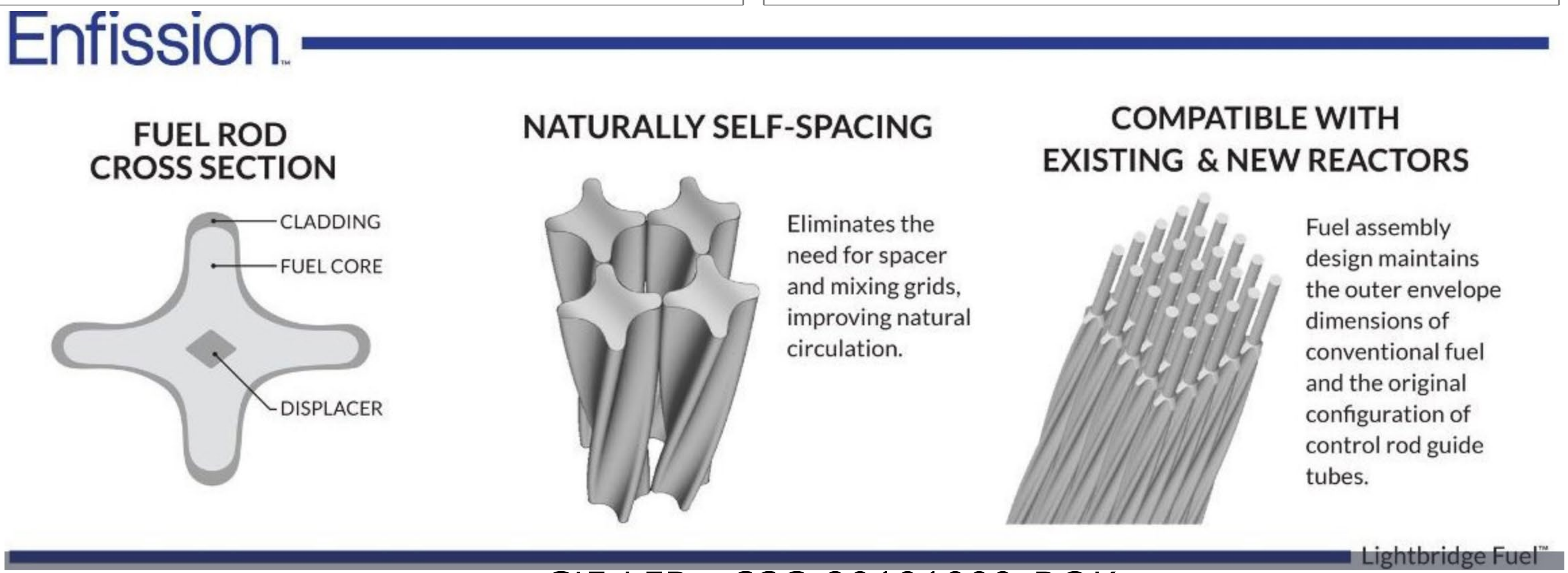
▲ Measured temp. difference vs. calculated temp. difference

MARS-LBE simulation showed good agreement within max. 3% deviations in mass flow rate and temperature difference between core inlet and outlet, respectively

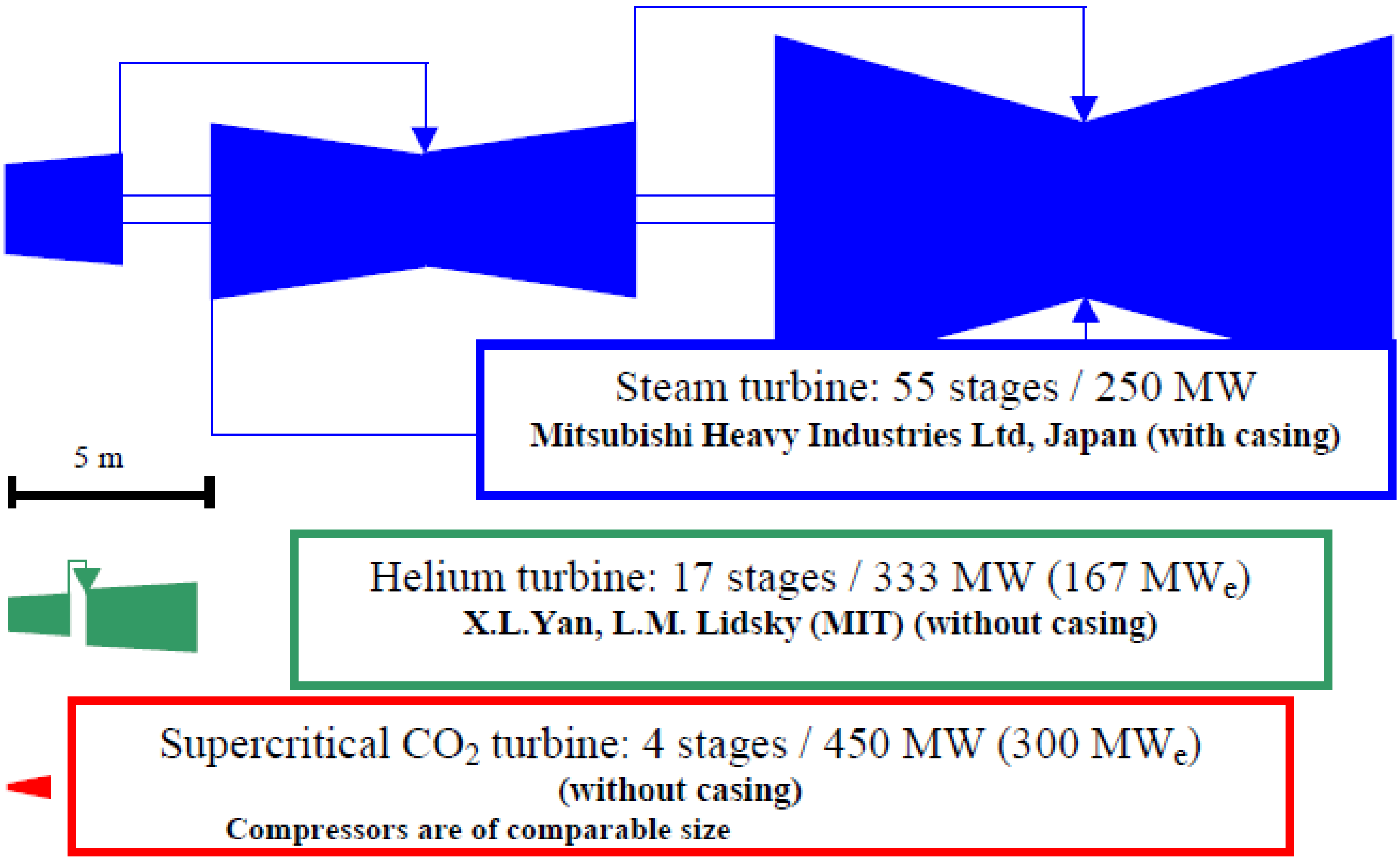
Fuel Option Study: UO2, UN, U-Zr







BOP System Option Study



IV. ROK Status: Summary

- ROK Government's Energy Transition Policy
  - SMR & MMR as New Focus
- Spent Nuclear Fuel – Real Impediment
  - Public Consensus Process Again
  - Transmutation by Pyroprocess & SFR: KAERI
  - PyroGreen: Academia
- Micro Reactor for Marine Applications
  - MicroUranus from URANUS
  - 10~20 MWe Capacity for 40 Year Non-refueling
  - Fuel Option Study
  - BOP System Option Study
  - PIRT in progress

USA Status on LFR Development

Craig F. Smith
Research Professor
Naval Postgraduate School, USA

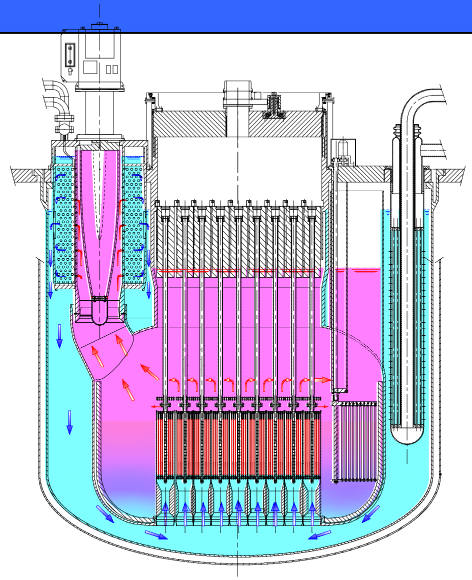
GIF – LFR pSSC

Office of Ansaldo Nucleare
Bucharest, Romania
October 7, 2019

Overview of Activities

- The SSTAR system remains a legacy system, little additional work being done since completion of its conceptual design: point of discussion
- Continuing recent industrial developments include two ongoing LFR initiatives and a third new start:
  - Hydromine LFR-AS-200 and LFR-TL-X
  - Westinghouse LFR
  - Columbia Basin Consulting Group (CBCG) – relatively new start
- Additionally, an ongoing EU-US INERI project is considering the possible role of a small LFR in powering an assured microgrid.

New configuration of LFR-AS-200



Compact primary system < 1m^3/MWe !!

(~ 4 times less than SPX1, 2-3 times less than integrated PWRs)

- Elimination of components no longer needed
- Innovative components
- Reversal of traditional engineering solutions

Compact reactor building

- Compact primary system
- No intermediate loops
- No risk of LOCA

Resistance to seismic loads

Short reactor vessel: only 6,2 m

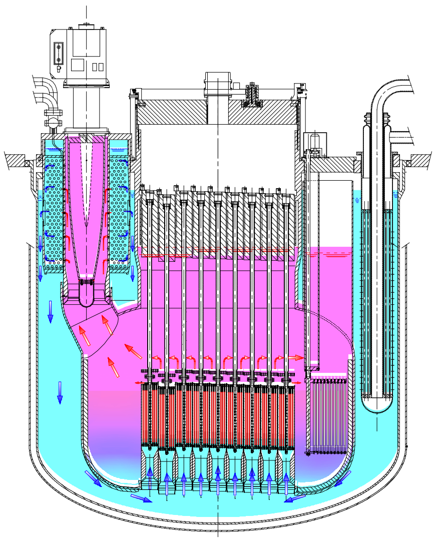
Safety

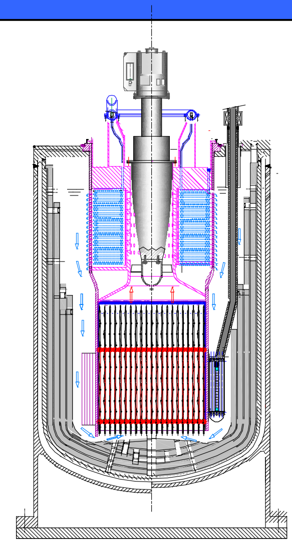
New steam generator to minimize the effect of the steam generator tube rupture accident

Main design parameters of LFR-AS-200

Core power (MWth)	480
Electrical power (MWe)	200
Core inlet/outlet T (°C)	420/530
Primary loop pressure loss (bar)	1,3
Secondary cycle	Superheated steam
Turbine inlet pressure (bar)	180
Feed water /steam temperature (°C)	340/500

From the LFR-AS-200 to the LFR-TL-X





vSMR
LFR
Basic
Concept

LFR-AS-200 has six steam generators

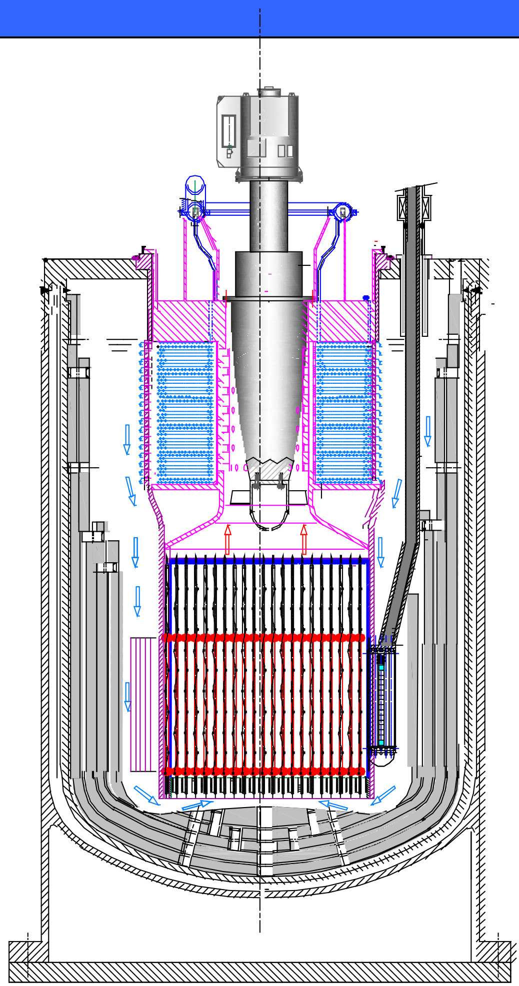
LFR-AS-200: AS stands for Amphora-Shaped, referring to the shape of the inner vessel, and 200 is the electrical power in MW

LFR-TL-X: TL stands for Transportable Long-lived core and X its power, ranging from 5 to 60 MWe or more, depending on the application

- **The LFR-TL is a transportable reactor with long-life core; this allows the design of a sealed reactor with elimination of fuel handling, to be carried out elsewhere**
- **The single Pump-Spiral-Tube Steam Generator assembly is co-axial with the reactor vessel in a Matryoshka-type configuration**

The LFR-TL-5 Conceptual Design

- **The upper part of the inner vessel, which supports the core, contains the STSG that, in turn, contains the circulation pump.**
- **Control and shut-down rods are located outside the core.**
- **The LFR-5 could be deployed in the near-term, owing to the lower operating temperature and the use of already-qualified materials.**
- **The LFR-TL-5 is able to operate continuously for 15 years.**
- **The LFR-TL-5 is fueled UO_2 with enrichment below 20% (19.75%).**
- **The LFR-TL-5 is intended for sites without interconnected grids, mines, and islands.**
- **Structural support of the LFR-5 from the bottom is proposed only for ship propulsion (to be confirmed). The use of an LFR-TL reactor for commercial ships requires revision of the rated power and other additional technical options.**



Merchant ship propulsion: opportunity for micro-reactors

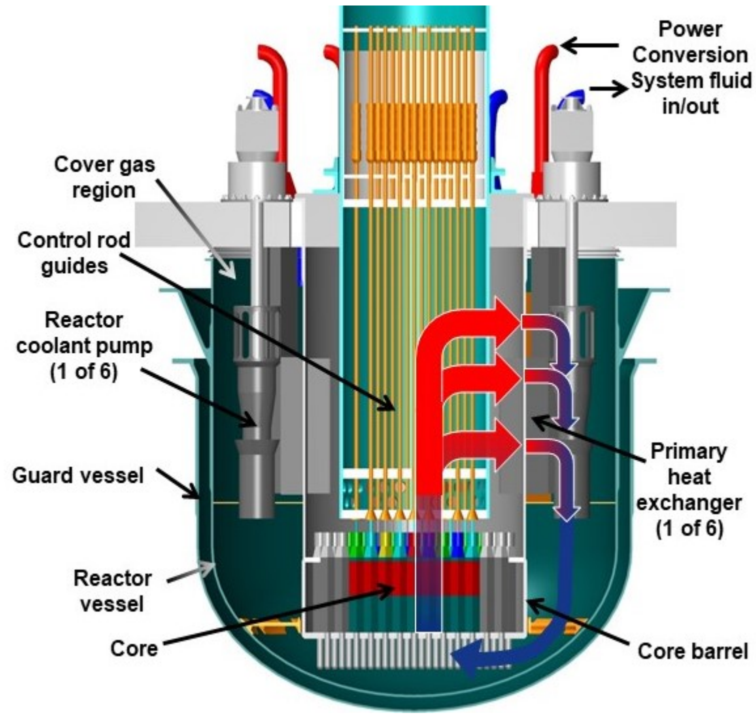
Merchant ship nuclear propulsion is on the agenda for commercial shipping because of pressure to decarbonize.

Additionally, the decline of polar sea ice over the last few decades indicates that the North Polar region may be open to regular marine traffic by the middle of the century. Soot from oil burning will be deposited on the snow and ice (Femenia, 2008). The presence of large numbers of hydrocarbon burning vessels in the region may lead to substantial additional ice loss. (Arctic Marine Shipping Assessment, 2009; Strategies for the Success of Nuclear Powered Commercial Shipping By B. Haas).

Desired Feature / Characteristic	*HNE LFR-TL-X and LFR-AS-200*	*Implications for shipping*
Passive safety/simplified design	LFR systems feature a high level of passive safety • relative chemical inertness of lead • high boiling temperature • ability to operate at atmospheric conditions • favorable thermo-hydraulic properties that support natural circulation heat removal.	Simplified design and operation result in high reliability, low maintenance operations over very long time frames.
Safety in the event of catastrophic occurrence	In the event of catastrophic occurrence, core becomes encapsulated in frozen lead coolant and prevents release of significant radioactivity.	Eliminates the potential for environmental dispersion of contaminants in the event of otherwise catastrophic conditions.
Compactness	LFR designs are very compact.	Reduced volume frees up cargo capacity.
Very Long refueling interval/sealed system	These systems can operate for 15-20 years without refueling.	Eliminates service outages/routing restrictions for refueling; eliminates safety issues with fuel transfer operations.

Westinghouse's Lead Fast Reactor

Aims at economic competitiveness, even in the most challenging global markets, through a simple and robust design, passive safety and lifecycle requirements embedded in the design from the early design phase



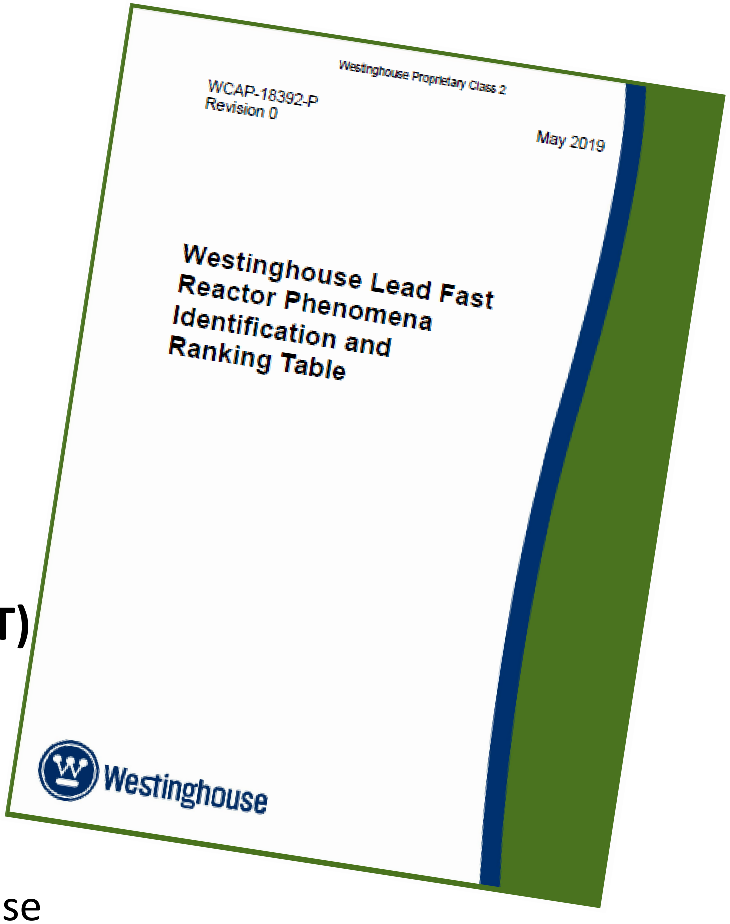
- ➢ 950 MWt (~450 MWe) reactor, to be developed starting with a lower-power prototype unit for technology demonstration
- ➢ Hybrid, micro-channel type heat exchangers to reduce vessel size/weight
- ➢ Thermal energy storage system to provide load-following with minimum variations in core thermal power
- ➢ Supercritical CO_2 power conversion system with air as the ultimate heat sink
- ➢ Oxide fuel and lead T<550°C for prototype unit. Advanced fuel and higher temperatures sought past demonstration phase

LFR development activities in Westinghouse

- **Development activities selected to most effectively de-risk the program, as to increase public/private investors' confidence in technical and economic viability**

- **Development activities span from being technology-agnostic to being technology-specific:**
  - Program foundation activities
  - Economic analysis
  - Licensing activities
  - Applied R&D, design, analysis and testing

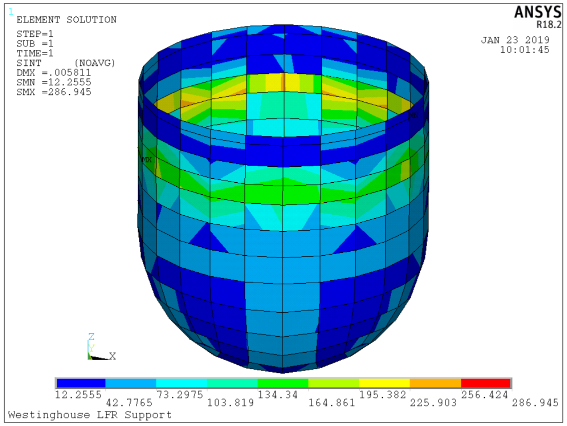
- **Phenomena Identification and Ranking Table (PIRT) performed to inform plan for testing and modeling&simulation**
  - PIRT panel: Argonne National Laboratory, Ansaldo Nucleare, ENEA, Fauske&Associates, and Westinghouse



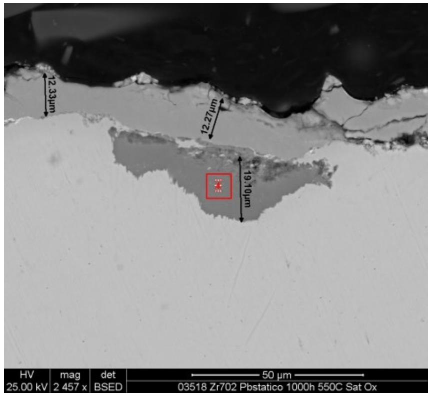
Westinghouse's updates on LFR technology development since March 2019 GIF LFR meeting

- Continuing plant design, informed primarily by safety analysis and economics
- Exploring long-life core design options
- Developed fuel failure model for LFR in SAS4A, through collaboration with Argonne National Laboratory
- Enhancing SAS4A capabilities for Passive Heat Removal System modeling, through coupling with GOTHIC
- Completed materials corrosion testing at 550°C (at ENEA). Testing at 700°is ongoing
- Setting up liquid metal embrittlement testing facility in Westinghouse, to support joint program with Oak Ridge National Laboratory on LFR materials
- Continuing collaboration with DOE for the development of Pb cartridge for the Versatile Test Reactor

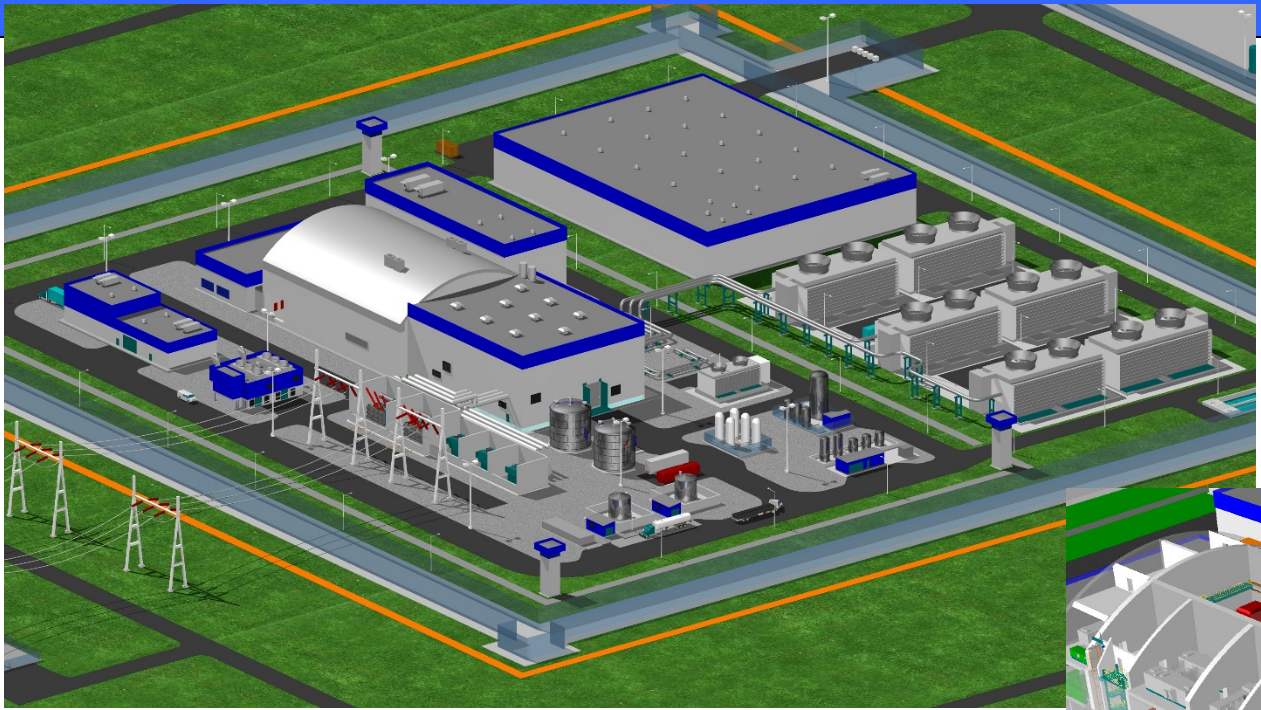
Seismic assessment for LFR vessel supports



Micrograph from corrosion testing (Courtesy of ENEA)

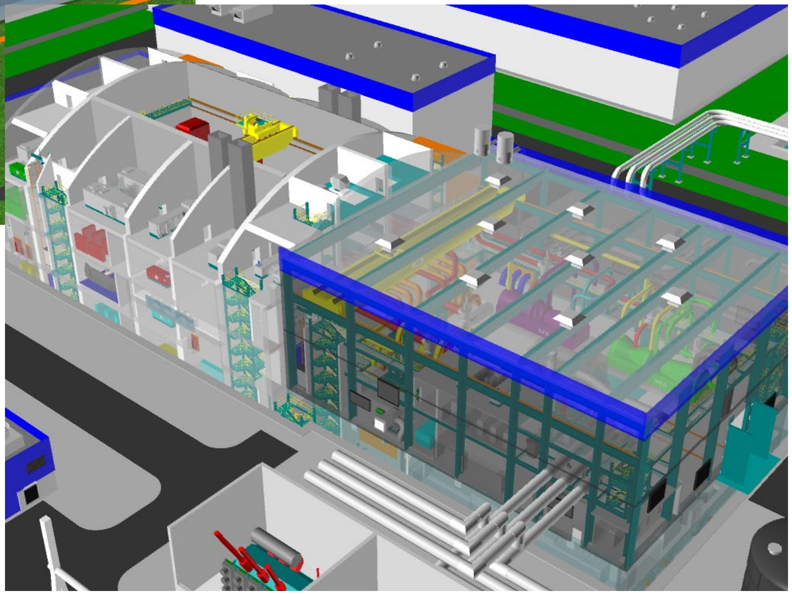


Westinghouse's updates on LFR technology development since March GIF meeting (cont.)



Representative plant layout for the Westinghouse LFR

Nuclear Island and Turbine Building



Columbia Basin Consulting Group (CBCG)
Lead Bismuth Fast Reactor w/Grid Scale Battery

CBCG is taking an Integrated Approach to Clean-Energy Production with a Competitive, Nuclear Plant Design and Load-Following via an Integrated Grid-Scale Battery Concept

Both the Nuclear Plant and the Grid-Battery are New Designs by CBCG – when paired as an integrated facility, demand load-fluctuations are accommodated by the battery, the nuclear plant remains at baseload operations.

The Nuclear Plant is based on a Lead-Bismuth Eutectic Coolant in a Fast Reactor Spectrum

CBCG's core expertise is advanced reactor systems development and operations. CBCG has utilized the US experience in the liquid-metal (sodium) fast reactors and the operations advantages of a lead-bismuth coolant to develop a very competitive design concept for a Small Modular, Lead-Bismuth Cooled, Fast Reactor plant.

Initial Efforts focused on Licensing and Regulatory Requirements

CBCG, with a DOE "GAIN" program Voucher, secured the services of Pacific Northwest National Laboratory to address these questions. The joint study concluded the technology was Licensable under current NRC rules.

A second DOE "GAIN" Voucher evaluated the Containment Building requirements. The joint study with the National Laboratory, concluded that leak-tightness requirements were reduced with Polonium mitigation.

CBCG is developing of a Polonium mitigation system to reduce containment building requirements and offsite release potentials by eliminating the principal radiological release hazard associated with this technology.

Columbia Basin Consulting Group (CBCG)
Lead Bismuth Fast Reactor w/Grid Scale Battery

Early Approach

CBCG initiated a collaboration with AKME-engineering to license their technology for the US market. However, world events frustrated this collaboration and CBCG proceeded to develop a new design.

Development Objectives and Progress

CBCG has been awarded a DOE Advanced Reactors Development Grant which has focused efforts on the nuclear system configuration and the nuclear island concept design. CBCG's concept is a "Loop-Design", scalable at 100MWe and 250MWe levels. At these power levels, the primary components are suitable for factory fabrication and shipping. The balance of plant components are pre-engineered systems from several vendors.

CBCG's is designing the plant for a 60 year life with an extended fuel cycle of 7-10 years, using uranium-oxide fuel. The nuclear plant design is suitable for multi-mission objectives, including electricity production, thermal energy production, desalination, etc. CBCG is also exploring alternative power conversion cycles.

CBCG Grid Scale Battery

Initially developed as an adjunct to the CBCG's nuclear plant concept, the Battery is suitable for applications in Renewable Energy. The design offers a reduced cost, extended life, and ease of fabrication and transport. The configuration is sufficiently flexible to be adaptable for multiple deployment applications.

US-EU INERI project

US-EU INERI project: *Small Modular Lead-cooled Fast Reactors in Regional Energy Markets: Safety, Security, and Economic Assessments*

Craig F. Smith (US) and Kamil Tucek (EU), PIs

- Purpose: To assess the feasibility and deployment potential of a highly secure lead-cooled SMR-based electric power grid that:
  - ➢ Is robust and resilient against external and internal threats
  - ➢ Provides load-following characteristics for integration of intermittent renewable energy sources
- In other words:
  - ➢ Investigate the feasibility and assess the potential deployment of
    - ✓ Small Modular Lead-cooled Fast Reactors (LFR)
    - ✓ In regional energy markets and for insular applications
  - ➢ Assess integration of small (load following) LFRs into the energy mix
  - ➢ Assess effectiveness:
    - ✓ Balancing out intermittent energy sources and demands
    - ✓ Supporting the capability for polygeneration
    - ✓ Enhancing energy security through increased robustness and resilience against external and internal threats
  - ➢ Consider safety, regulatory and preliminary economic aspects

US-EU INERI project - continued

- INERI Project is ongoing
- Focus is on the lead-cooled SMR nuclear power source since reliability, resilience and recovery of other main grid components receives much attention elsewhere
- Project efforts include:
  - Kickoff meeting with wide participation by all partners (hosted by Euratom / JRC)
  - Program review meetings held at Oak Ridge in July 2017, Ispra, Italy in July 2018, and LLNL in June, 2019
  - Preliminary consideration of desired features of an SMR for such application (CPT James Bowen, NPS)
  - Cost-benefit evaluation targeted on the SMR-assured grid at US Navy installations. (LT Jeffry Asch and LCDR Gregory Brant, NPS)
  - Survey of relevant approaches and data from Euratom Framework Programme projects (Euratom / JRC)
  - IP and confidentiality rights related to INERI and Euratom project deliverables clarified (Euratom / JRC)
  - Presentation of financial risks study presented at IAEA Fast SMR technical meeting in Milan, September 2019
  - Project status meetings held in October 2018 and May 2019
  - Final project workshop being planned for early 2020

Some final comments

- Current US government-sponsored LFR efforts remain limited; however, there is continuing DOE interest in LFR technology, mainly as a backup option to the SFR
- Some significant industrial activities to be noted – Hydromine, Westinghouse and Columbia Basin Consulting Group
- US researchers are continuing to maintain LFR options through national lab, university and industry projects.

Institute of Nuclear Energy Safety Technology, CAS · FDS Team

发展先进核能科技，让人类生活更美好！

www.fds.org.cn

National Status on LFR development in China

Contributed jointly by FDS Team

Institute of Nuclear Energy Safety Technology (INEST)

Chinese Academy of Sciences (CAS)

www.fds.org.cn

25th GIF LFR pSSC meeting, Bucharest, 8th Oct. 2019

The Signature by INEST of the LFR MOU will be Hold in 48th PG meeting (18 OCT), Weihai, China

GEN IV International Forum

Version, 29 Aug 2019

48th Policy Group Meeting – Draft Agenda - Day 1
18 Oct 2019 – Haiyue Jianguo Hotel（海悦建国酒店）, 4F Ball room, Weihai, China

08:30	Light Refreshments	
09:00	**Closed Session** • GIF governance, approval of interim positions • Cooperation with private sectors (Small Gr report) • EG Charge for the SIAP • Activity directions of TFs • Schedule for Turkish bid • Future schedule of PG	PG members
10:50	**Break**	
11:00	**Open Session**	
11:00	Remarks and Announcements	Mr. Kamide
11:10	Report from the SCWR SSC	Mr. Huang
11:40	Report from the LFR SSC	Mr. Alemberti
12:10	Signature by INEST of the LFR MOU	China

Domestic GIF LFR Co-ordination Meeting

✓ Purpose

- To share the LFR related information
- To establish a domestic co-ordination group to support GIF LFR related work

✓ Schedule

- Nov. or Dec. 2019

✓ Participants

- CAS , CGN , CNNC, SPIC, universities…

Contents

China's Plan on Nuclear Energy (Plan up to 2020)

- **Nuclear power plant in China (by August, 2019)**
  - *45 reactors (~ 45.9 GWe) in operation*
  - *13 reactors (~16.6GWe) under construction*

~4%

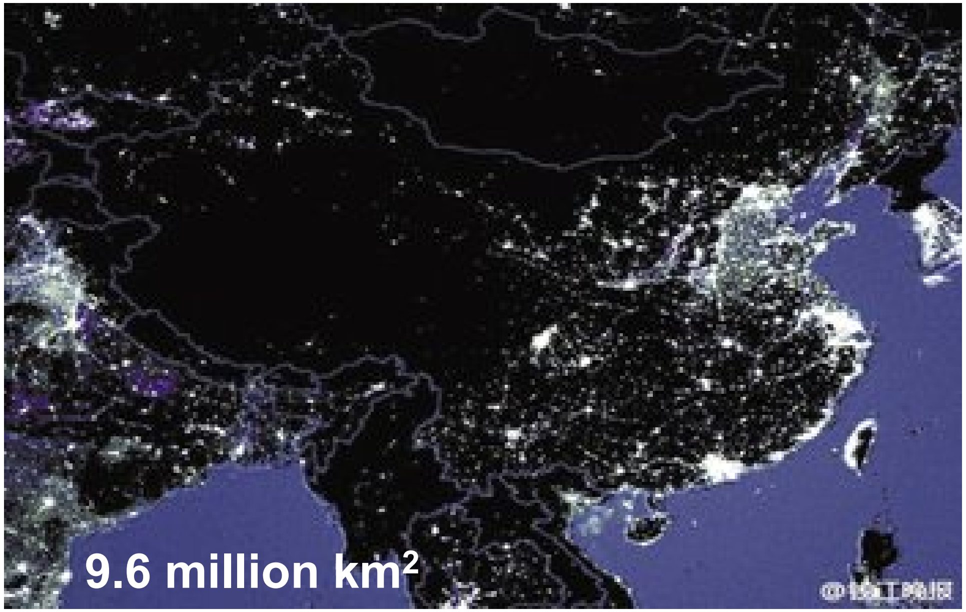
- **National plan of developing nuclear energy before 2020**
  - *58 GWe in operation*
  - *30 GWe under construction*

~5%

- **National plan for nuclear and radiation safety before 2020**
  - *More R&D are required to enhance nuclear safety, especially in the basic research of nuclear safety*
  - *~79.8 billion RMB investment plan (~13 billion US $)*

Small-scale Energy Supply Demand in China

- **Remote area**: ~12% of land area is desert without electricity supply
- **Offshore unit**: ~1/4 of gas and oil reserve in the sea, waiting for exploitation
- **Distributed power supply**: independent industry, with wind or solar energy
- **Emergency electricity supply**: Natural disaster in some provinces



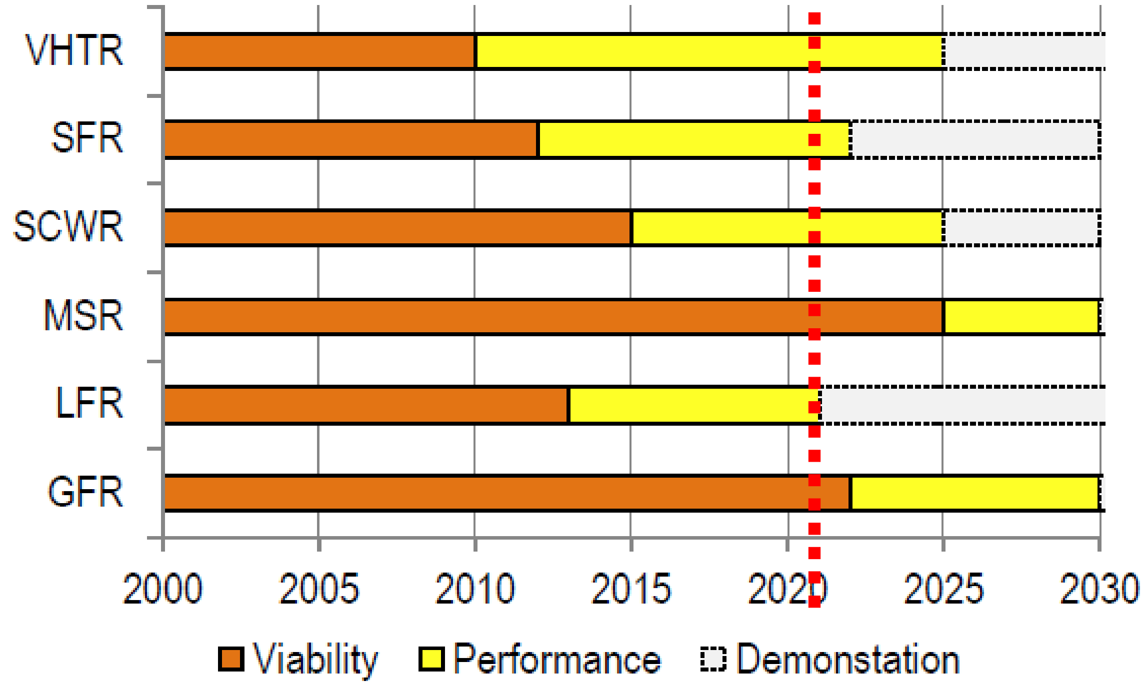
High Requirements

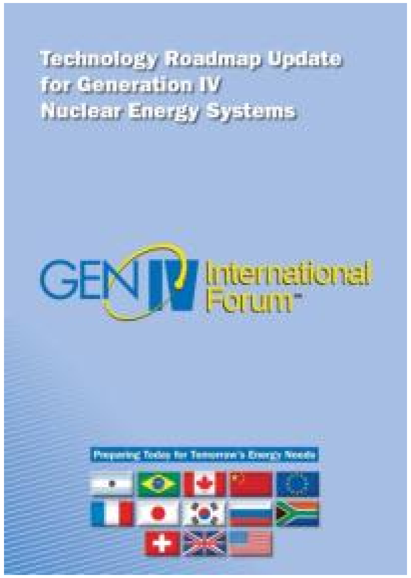
- ☐ Independent
- ☐ Flexibility
- ☐ Sustainability
- ☐ Economical

Latest Roadmap of Generation IV Reactors

—— GIF organization evaluated in 2014

LFR is expected to be the first Generation-IV nuclear system to achieve industry demonstration and commercial application





Characteristics of Lead-based Reactor

Safety advantages

- **Neutronics：** Negative coefficient, Floating core debris
- **Thermal-hydraulics:** Low pressure, no LOCA, natural circulation
- **Chemistry:** Chemical inertial, no reaction with water and air, no hydrogen explosion

Sustainability advantages

Low neutron absorption, Low moderation, enable sustaining hard neutron spectrum

→

High efficiency in fuel utilization

Burning long-lived, high-level actinide wastes

Key Technologies Shared among Various Lead-based Coolants

- Three major types of Lead-based coolants

Lead	High power conversion coefficient →Electricity application for Gen-IV LFR
Lead Bismuth	Low operation temperature →Early application for ADS, etc..
Lead Lithium	Tritium breeding →Long-term application for fusion

- ☐ Similar properties, key technologies shared with others
- ☐ Can be applied for both critical and sub-critical systems

Major China Lead-based Reactor Program

1980s-1990s	2000s	2010s
"863" program "973" program	ITER International/Domestic Research Program	Strategic Priority Research Program of CAS National Major Science and Technology infrastructure

- **National High-Tech. Project : Fusion-fission hybrid reactor**
  INEST/FDS in charge of Lead-based hybrid reactor
- **ITER Project : Fusion reactor**
  INEST/FDS in charge of China lead-based liquid blanket
- **Strategic Priority Research Program of CAS : ADS system**
  INEST/FDS in charge of lead-based sub-critical reactor
- **China Lead-based Mini-Reactor (CLEAR-M)**
  Supported by national/local government project and Industry investment

~30 years lead-based reactor R&D experiences

Y. WU，FDS Team. Design and R&D Progress of China Lead-based Reactor for ADS Research Facility，Engineering，2016, 2(1):124-131.

Contents

I. Strategy and Plan

II. Reactor Design Study

- CLEAR-M (Mini-reactor for energy production)
- CLEAR-A (ADS system for multi-purposes)
- FDS series concepts (fusion reactors)

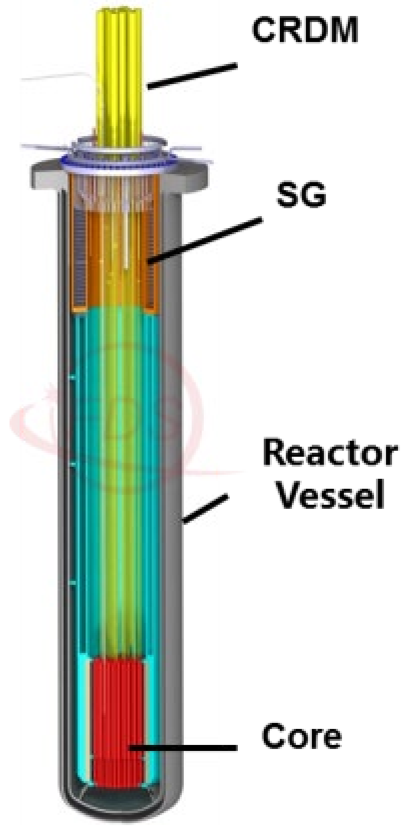
III. R&D Progress

IV. Summary

Lead-based Mini-Reactor CLEAR-M10

- **Small modular**
  - Easy to transport and install
- **Inherent safety**
  - No severe accident
- **Long refueling period**
  - Better economy

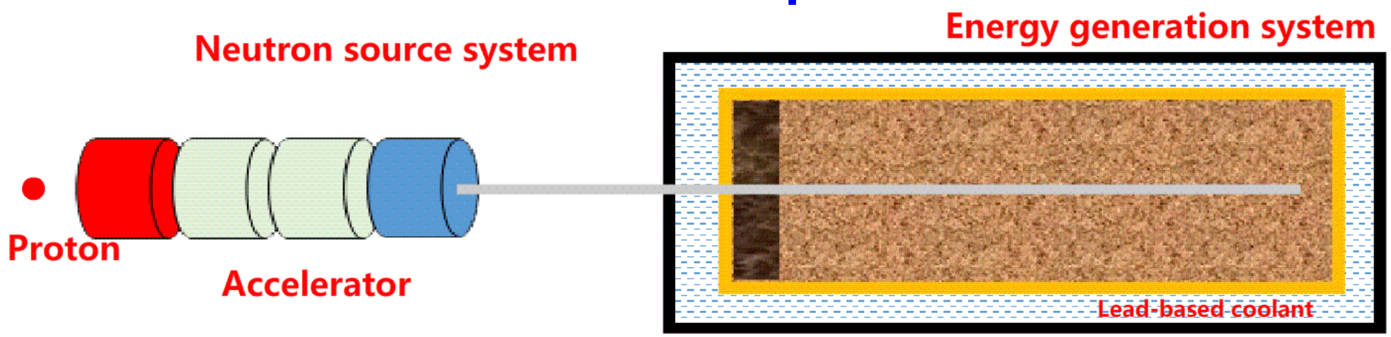
Parameter	Values
Thermal power	35MWth
Electrical power	14MWe 10MWe+17MWt
Fuel	Ave.18.5% UO2



Experiment reactor CLEAR-M10a → **Demonstration reactor CLEAR-M10**

Advanced External Neutron Source Driven Nuclear Energy System (CLEAR-A)

Neutron source + Subcritical Operation+ Lead-based reactor

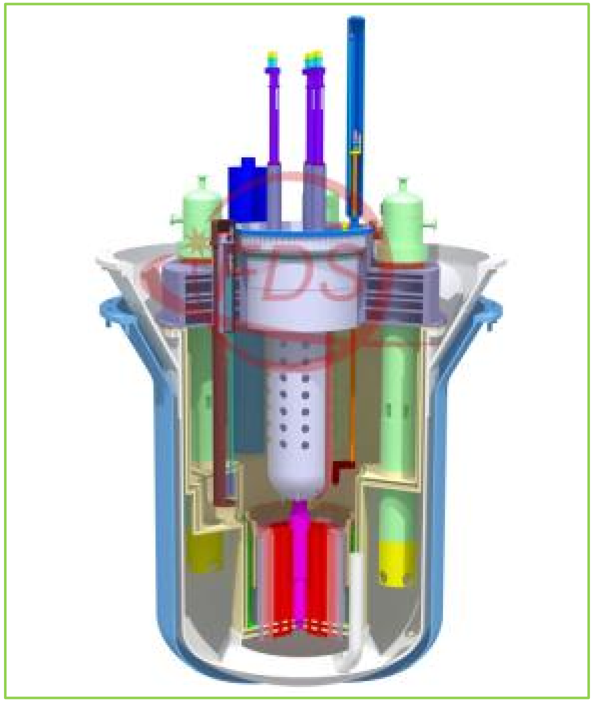


- **Inherent safety**
  - Subcritical operation
  - Lead as coolant
- **Sustainable resources**
  - Depleted U or Th as fuel with breeding reaction
  - More than 10 years refueling period
- **Less nuclear waste**
  - TRU transmutation with high energy neutrons
  - Only FP remove from spent fuel

		A10 Experimental	A100 DEMO	A1000 DEMO
Accelerator	Type	**Cyclotron**	**Cyclotron**	**Cyclotron**
	Proton Energy /MeV	900	900	900
	Current /mA	1-10	~10	7×~10
Target	Material	LBE	Lead	Lead
Reactor	Power	**10MWt**	**100MWe**	**1000MWe**
	Coolant	LBE	Lead	Lead

Subcritical Experimental System : CLEAR-I/A10

- **Objective:** subcritical lead-based reactor for CLEAR-A
- **Design status:** the detailed conceptual design has been done

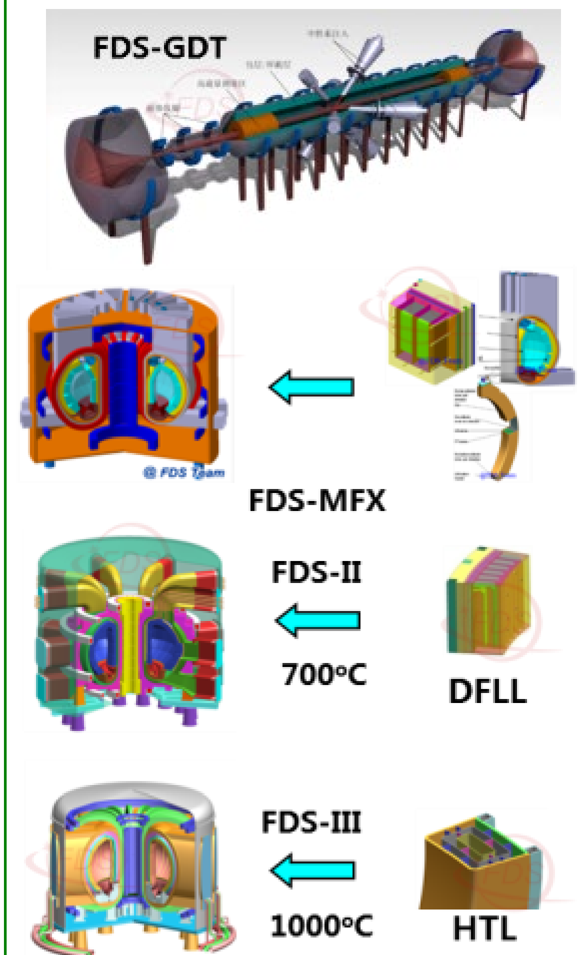


Parameter	CLEAR-I	CLEAR-A10
Thermal power	10MW	10MW
Accelerator	**Linac (250MeV/10mA)**	**Cyclotron (900Mev/1mA)**
Fuel (enrichment)	UO_2(19.75%)	UO_2(19.75%)

The CLEAR-A10 design is based on the CLEAR-I. CLEAR-A10 and CLEAR-I have the common main design parameters and main reactor systems, except the main coolant and the accelerator.

FDS Series Fusion Reactor & Blanket Concepts Development

- **Non-tokamak-based Fusion Reactor**
  - **FDS-GDT: D-T fusion based on GDT mirror**
    - Fusion materials and components test; Fusion-fission hybrid reactor driver
  - **FDS-D3He: D-3He aneutronic fusion**
    - Inherently Safe, Clean, Economic fusion power; Fusion Space Propulsion
- **Tokamak-based Fusion Reactor**
  - **Fusion TEST Reactor**
    - **FDS-MFX:** Multi-Functional eXperimental Reactor (abbreviated as MFX)
    - **CFETR:** Chinese Fusion Engineering Testing Reactor (Liquid PbLi Blanket)
  - **Fusion DEMO Reactor**
    - **C-DEMO:** Chinese DEMO Reactor (energy production, fuel breeding, multiplication)
    - **FDS-SFB:** Fusion Reactor for Spent Fuel Burner (early application)
  - **Fusion POWER Plant**
    - **FDS-II:** Fusion Power Reactor (high-efficiency electricity generation)
    - **FDS-III:** High Temperature Fusion Reactor (hydrogen production)
    - **FDS-ST:** Spherical Tokamak-based Reactor (economical)

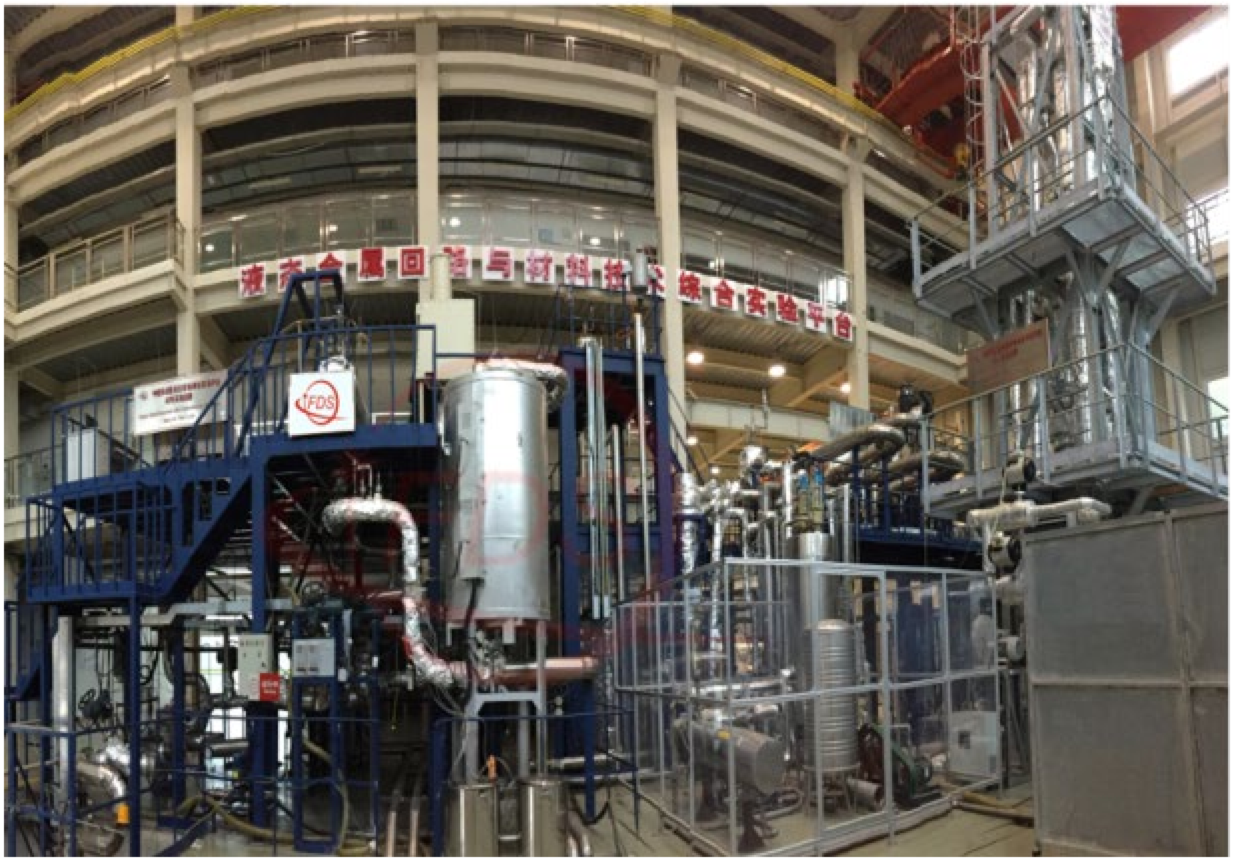


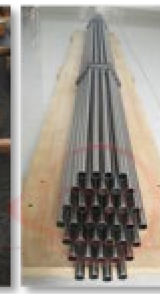
Contents

Key Technologies

- Coolant Technology
- Key Components

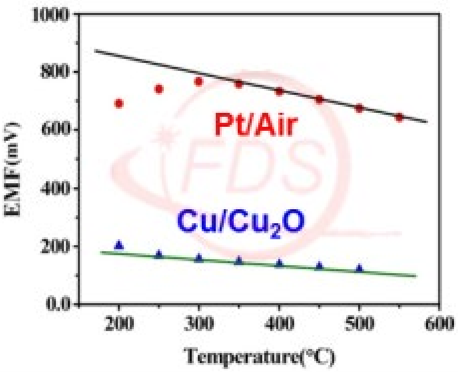
- Materials and Fuel
- Operation and Control

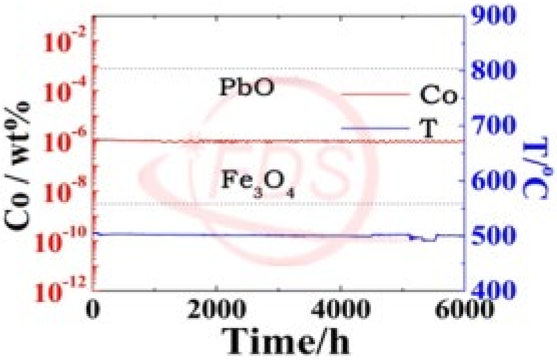




Key Technology Ⅰ: Coolant Technology

- **Oxygen sensor**
  - Pt/air， Cu/Cu_2O
- **Oxygen control**
  - Gas phase and solid phase
- **On line purification technologies of LBE alloy**
- **Preparation of high purify LBE (impurity <50ppm)**





Key Technology II: Key Components for Reactor

1:1 scale prototype components, tested under lead alloy condition

Refueling system

Main Vessel

Fuel assembly simulator

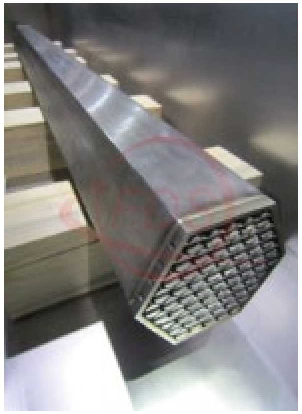
Control rod driven system

Heat exchange

HLM pump

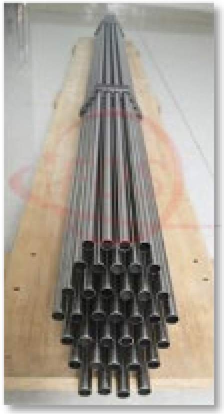
Key Technology III : Materials and Fuel

- **Fabrication of full-size prototype fuel assembly**
- **China LEad-based reactor fuel claDding: CLED**
  - High mechanical properties by increasing the Ti/C ratio
  - Good compatibility with LBE **(>10000h)**

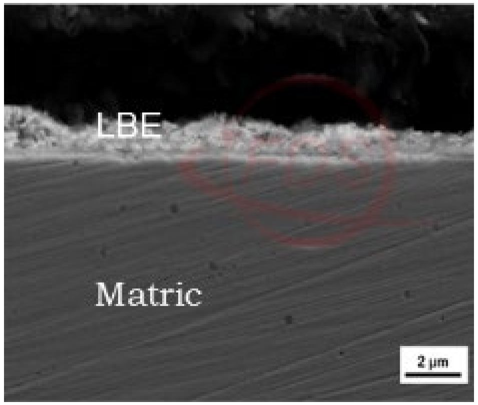


Full-size prototype fuel assembly with UO2

Forging Bar



Cladding tube



4000h corrosion test at LBE (10^{-6}wt% oxygen concentration)

Key Technology IV: Reactor Operation and Control

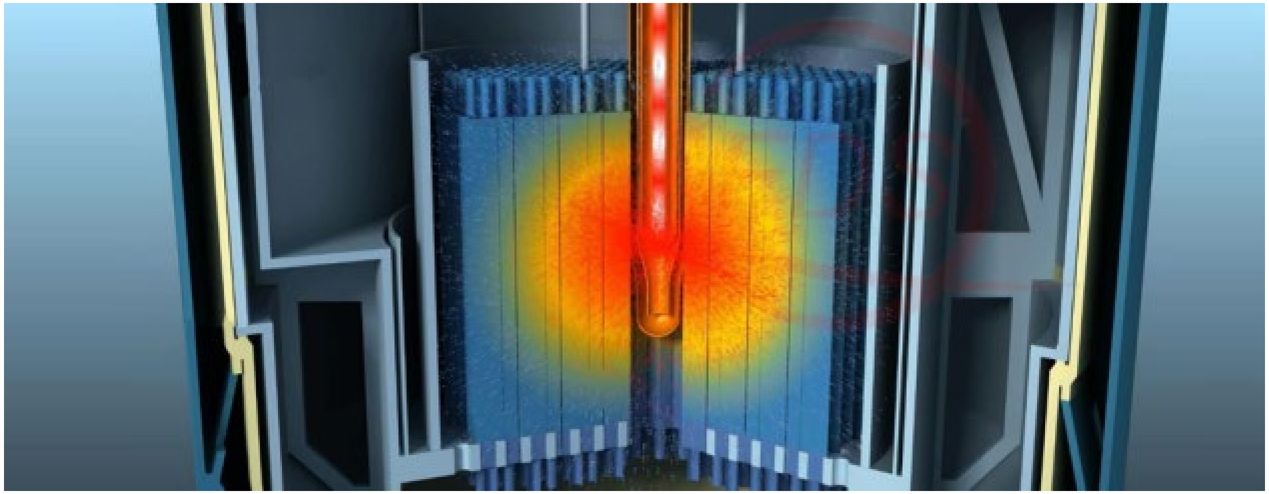
Lead-based reactor full scope simulator

- Reactor operator training for whole life cycle
- Reactor safety assessment for DBA and BDBA accidents

Digital Simulation Reactor
——Virtual Lead-based Reactor CLEAR-V

Multi-physics integrated simulation by ~30 codes

- ❑ Independent intellectual property rights (>1000 person-year manpower)
- ❑ Applied in 90+ nations, 1000+ institutions
- ❑ ITER project neutronics reference code
  (International thermonuclear experimental Reactor)



Yican Wu. Multi-functional Neutronics Calculation Methodology and Program for Nuclear Design and Radiation Safety Evaluation. Fusion Science and Technology, 2018(74):321-329.

FDS Software

I. Physics & Engineering Calculation

- SuperMC — Super Multi-functional Calculation Program for Nuclear Design and Safety Evaluation
- NTC — Neutronics-Thermohydraulics Coupled Simulation Program
- MTC — Magnetic-Thermohydraulics Coupled Simulation Program
- TAS — Tritium Analysis Program for Fusion System
- HENDL — Hybrid Evaluated Nuclear Data Library
- FusionDB — Fusion Database Management System

II. Interactive Design and Optimization

- RiskA — Reliability and Probabilistic Safety Assessment Program
- RiskAngel/TQRM — Risk Monitor for Nuclear Power Plant
- RiskBase — Database Management System for Reliability Analysis
- SYSCODE — Fusion System Design and Economical Assessment Program
- KylinRay — Accurate Radiotherapy System

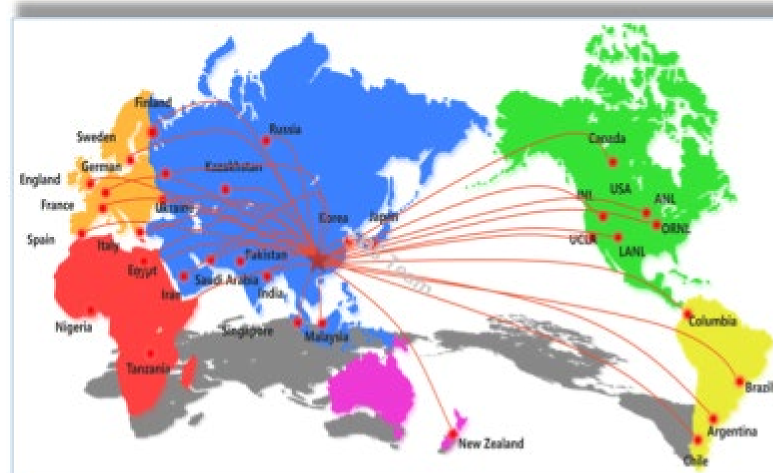
III. Multi-process Integrated Comprehensive Simulation

- Virtual4DS — Virtual Nuclear Power Plant in Digital Society Environment
- VisualBUS — Digital Nuclear Reactor
- CLEAR-V — Virtual Lead-based Nuclear Reactor
- Fusion-V — Virtual Fusion Reactor
- CROSS — Informatization Collaboration Platform for Scientific Research
- NCloud — Nuclear Cloud Platform
- NBigData — Nuclear Big Data Platform

SuperMC: Super Multi-functional Calculation Program for Nuclear Design and Safety Evaluation

- CAD-based accurate automatic modeling for complex irregular geometry
- Feature-accelerated high-efficiency calculation (whole process coupling)
- Multi-style visualized analysis
- Collaborative nuclear and multi-physics design based on cloud computing

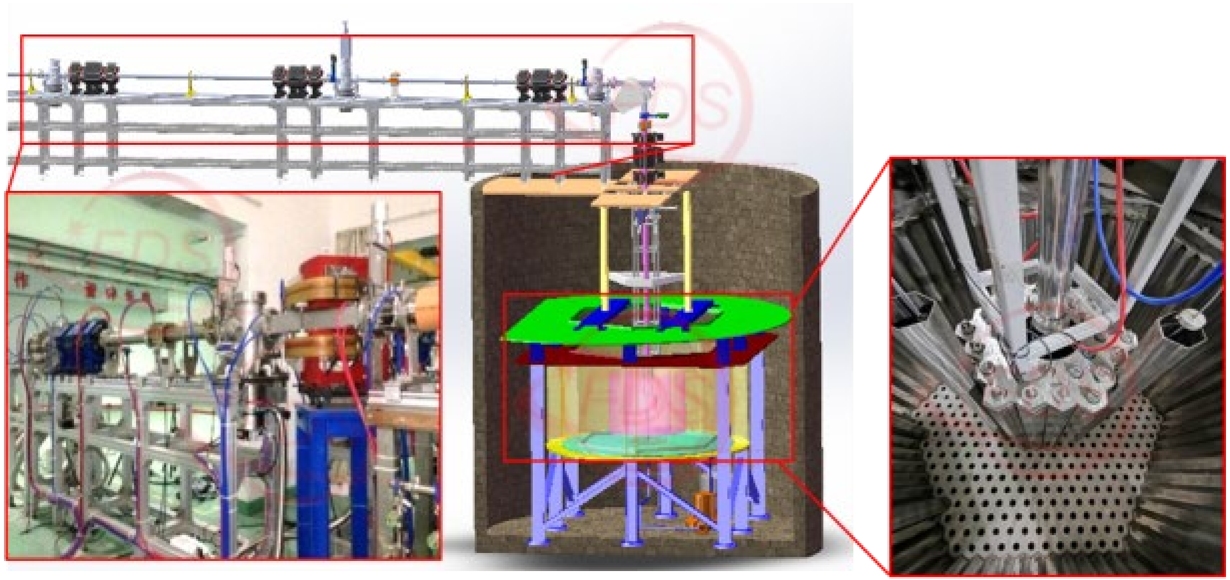
- Widely used in 60+ countries and 40+ mega-projects
- Selected as the reference code by ITER, and supported to build ITER 3D basic neutronics models
- Widely distributed by OECD/NEA Data Bank (IAEA1437)



Neutron Physics Test Reactor

—— Lead-based Zero Power Critical/Subcritical Reactor CLEAR-0

- Validation of the neutronics analysis method, code and database for both critical and subcritical
- Provide experimental data to support the licensing of CLEAR

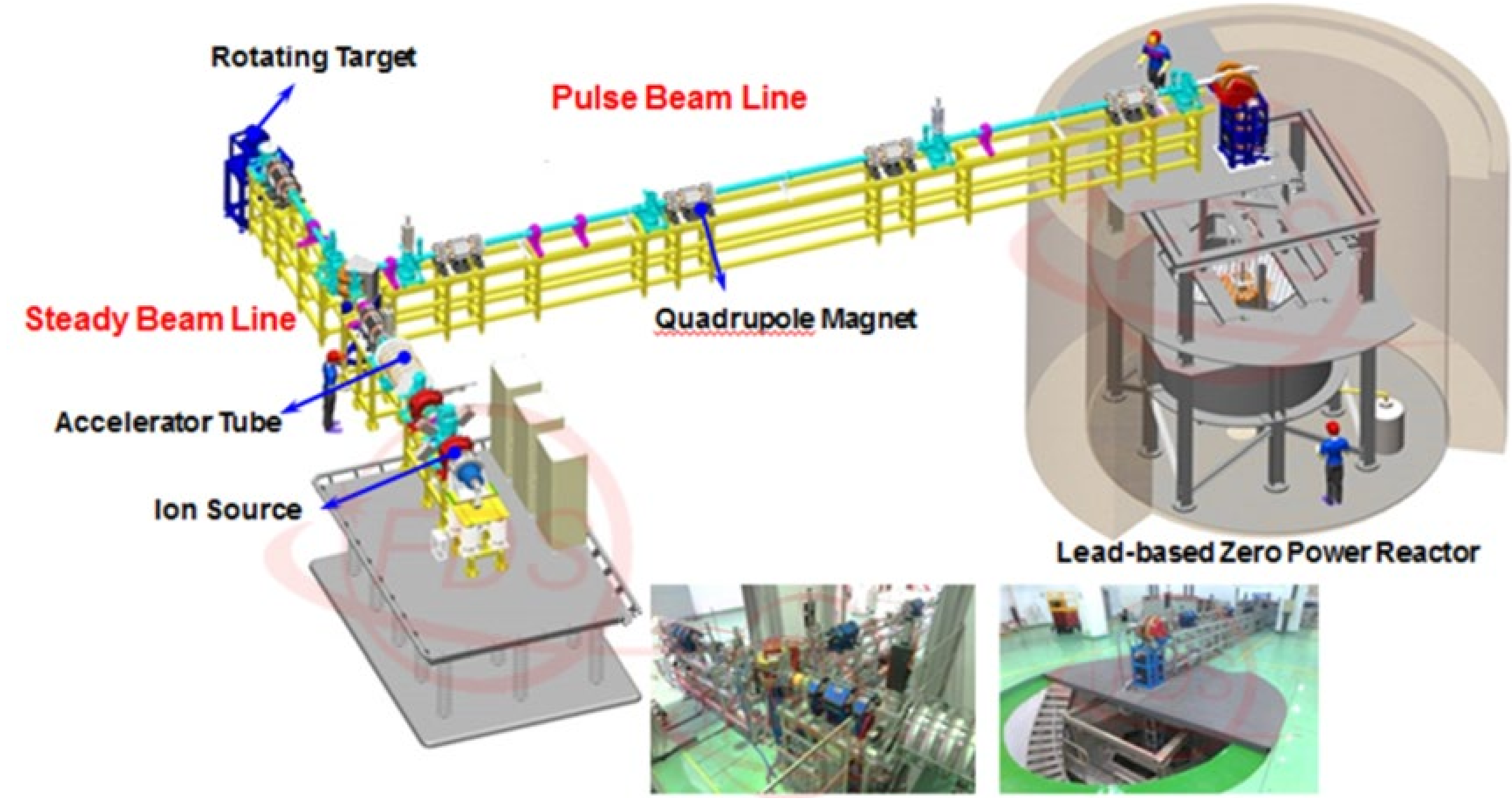


Accelerator
DT neutron source
(HINEG)

PbBi fast reactor

Reactor core

HINEG-I: D-T Fusion Neutron Generator Coupling with Lead-based Zero Power Fission Reactor

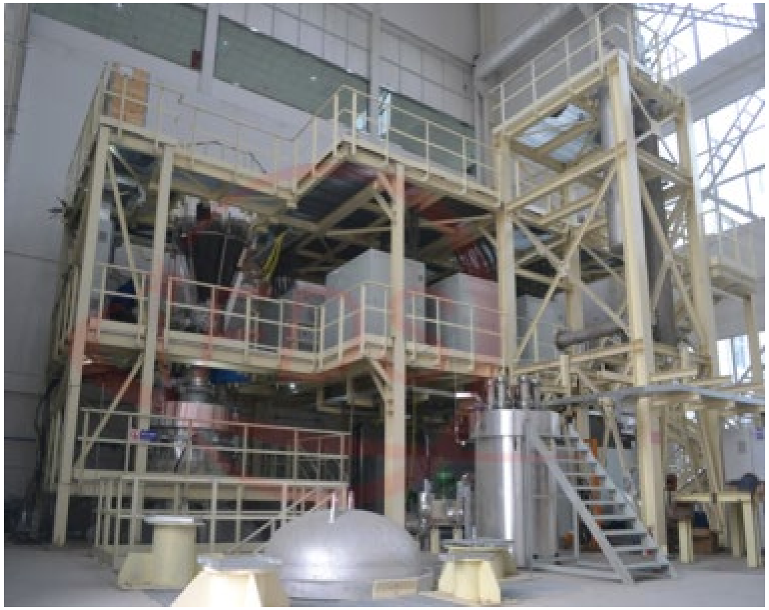


Fusion neutrons with yield up to 6.4×10^{12}n/s

Engineering Validation Facility

——Integrated Non-nuclear Test Facility CLEAR-S

- ❑ Reactor prototype components validation
- ❑ 1:1 height, 1:2.5 diameter to CLEAR-I



Y. Wu, FDS Team. CLEAR-S: An Integrated Non-nuclear Test Facility for China Lead-based Research Reactor. International Journal of Energy Research, 2016, 40(14):1951-1956.

Further Implementation Activities

- ✓ **Industrial park for lead-based reactor**
  - laboratory under construction
- ✓ **China Industry Innovation Alliance of Lead-based Reactors (CIIALER)**
  - president member INEST/FDS Team, over 100 enterprises
- ✓ **International Co-operative Alliance for Small LEad-based Fast Reactors (CASLER)**
  - chair INEST/FDS Team, over 20 members

IAEA CRP Proposed

—— CLEAR-S Pool LOF Benchmarking

☐ General descriptions

- Performed components testing and system commissioning (Forced & natural circulation) has been finished
- The coming tests (LOH+LOF) will be conducted end of 2019
- First proposed in 51st TWG-FR meeting (with 16 members interested ENEA, CEA, IKET, ANL, IPPE, SCK, JAEA, PSI, KAERI, CIAE......)

☐ Proposed programs (HLM pool LOF)

1. **Steady state thermal-hydraulic experiments**
   - *300-385 °C with >2MW full power*
2. **Transient: PLOF**
   - *FC → NC*
   - *Full power → 5% full power*
   - *PHX → Dip cooler (installed in cold pool*
   - *5% full power → 1% full power*
   - *Dip cooler → RVACS*

T-H codes benchmarking

- Sub codes
- System codes
- CFD codes
- Coupling code

The consultant meeting is planned to be hold in 2020

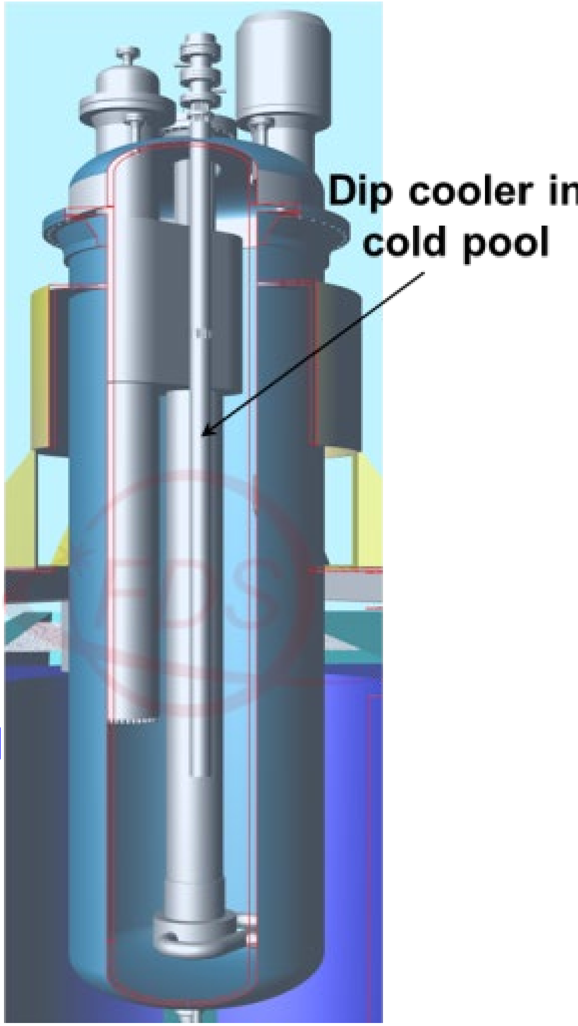
CLEAR-S HLM Pool Facility

- 2.5 MW mock-up pool type facility with lead-bismuth eutectic coolant
- General aims including qualifying prototypical components, pool thermal hydraulics and TH code V&V
- Performed components testing and system commissioning (forced & natural circulation) for >2MW has already been done
- The coming tests (LOF) will be conducted in Oct. 2019

CLEAR-S is an available platform for pool-type HLM reactor T-H benchmark

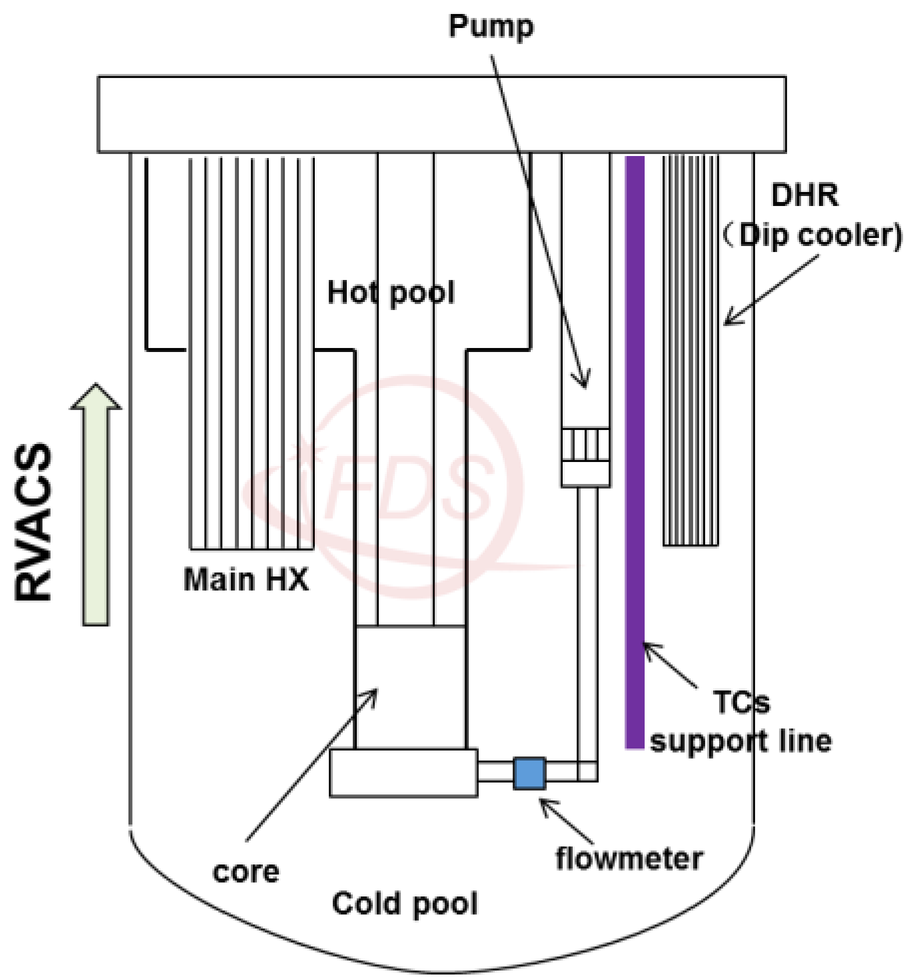
CLEAR-S main features

- Outside Diameter of MV 2000 mm
- Height of MV 6500 mm
- Material AISI 316L
- Max LBE Inventory 240 tons
- Power Size 2.5 MW
- Temperature Range 200 to 550 ºC
- Nom Mass Flow-Rate 220 kg/s
- DHR system Dip Cooler & RVACS (reactor vessel air cooling system)
- Secondary Side High pressure water up to 10MPa



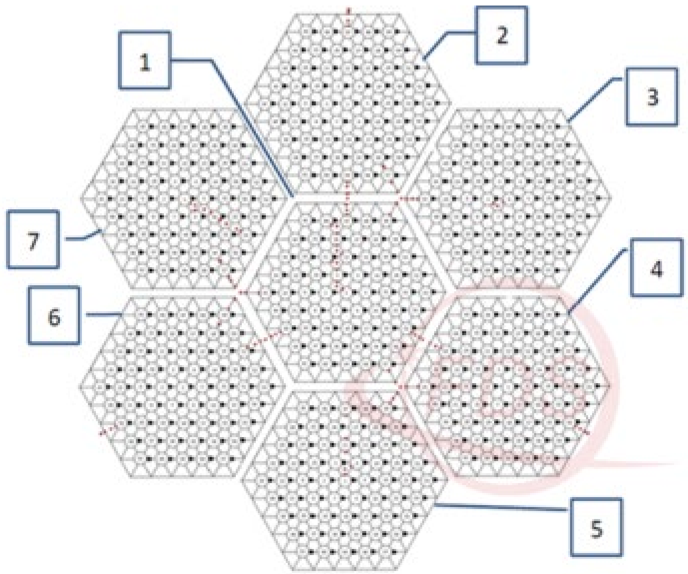
Measurement Points Configuration

- 1 venturi flowmeter
- 24 pressure sensors
- 374 TCs in pool (arranged in TCs support lines)
- 145 TCs in HX and DHR
- 57 TCs in internals and pump
- 285 TCs in core



Core simulator

• NO. of FAs	7
• NO. of rods	61 x 7
• Spacer type	Wire spacer
• Total heating power	2.5 MW
• Gap of inter-wrap flow	3.5mm

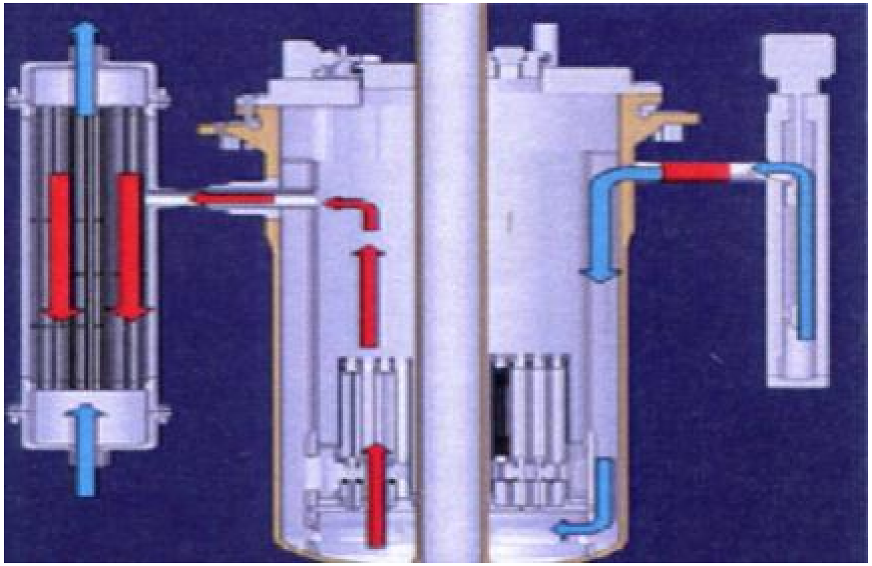


Total 285 TCs set in core simulator

Other LFR Activities in China

- **CiADS**
  - ✓ Launched by CAS (IMP)
  - ✓ The design and R&D of granular windowless target ADS system)

different options are being discussed



CiADS reactor

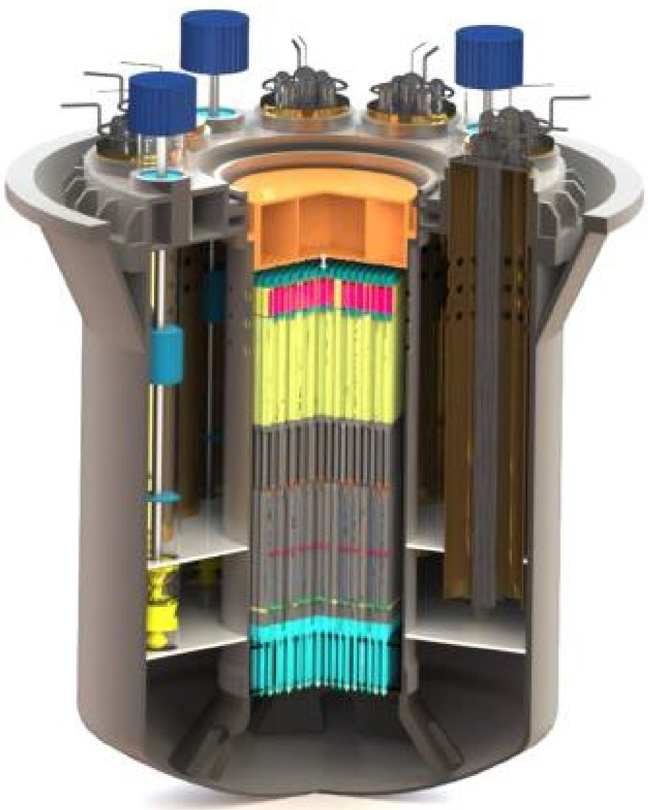
Prototype of granular target

Other LFR Activities in China

- **CGN**
  - ✓ CAS-CGN collaboration on CiADS in past few years
  - ✓ CLFR series reactors conceptual design
  - ✓ R&D: ODS steel, CLRASS cladding material, coolant processing, oxygen control
- **CNNC**
  - ✓ LBE corrosion loop operation for 168h
- **SPIC**
  - ✓ BLESS reactor conceptual design and related R&D

SPIC – Concept Design

Concept design of 100 MWe LFR BLESS



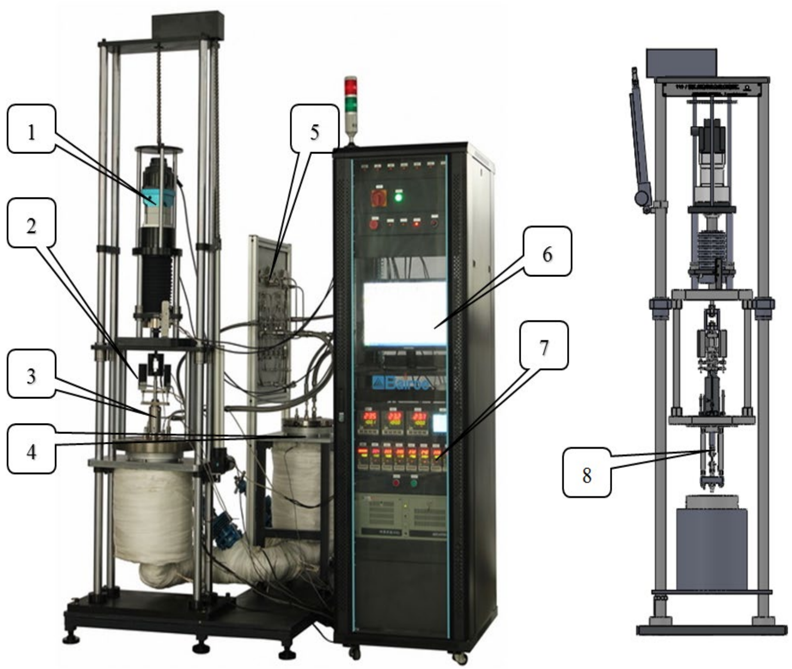
BLESS-D concept design

BLESS-D main design parameters

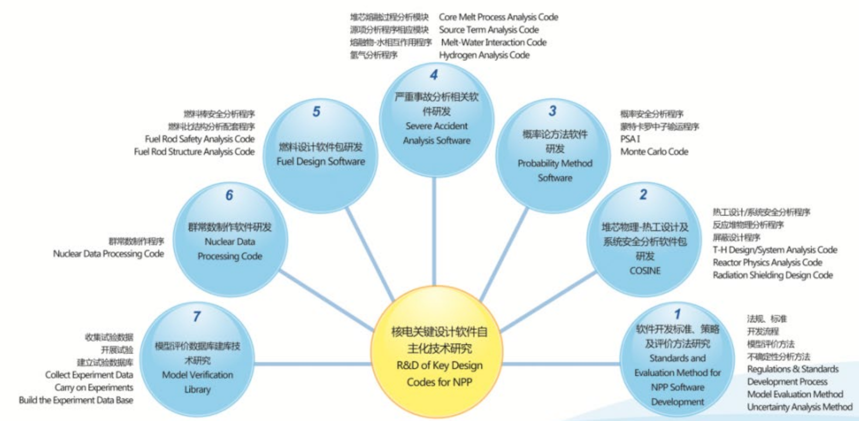
Thermal Power	300 MWt
Electric Power	120 MWe
Fuel	UO_2
Enrichment	14%/16%/19.75%
Active zone height	70 cm
Active zone diameter	242.2 cm
Core vessel diameter	256 cm
No. of Fuel assemblies	247
No. of Fuel rods per FA	127
Fuel rod diameter	9.29 mm
Linear power density	116 W/cm
Coolant	LBE
Core inlet temp.	340 ℃
Core outlet temp.	490 ℃
Operation pressure	0.1 MPa

SPIC – R&D Activity

- ✓ T-H experiment technology and facilities detailed design
- ✓ Corrosion and erosion performance research of Martensite and Austenite steel in liquid LBE environment are studied
- ✓ Application of COSINE code system and its future development for LFR R&D



Static and stressed corrosion test facility



COSINE code system

Contents

Summary

1. Lead-based reactor has many attractive features and may play an important role in the future energy supply, including a bridge role in the transition period from fission energy to fusion energy.

2. China has launched LFR/ADS engineering project in 2011. The engineering design and R&D activities for CLEAR series as well as the following fusion/hybrid reactor programs are going on in order to finish the construction of the first engineering experimental system.

3. Wider and deeper international collaboration is encouraged.

Thanks for Your Attention!

Website: www.fds.org.cn

E-mail: contact@fds.org.cn

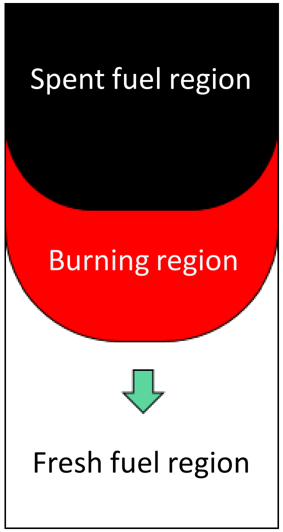
LBE cooled Stational Wave Breed and Burn reactor

Background

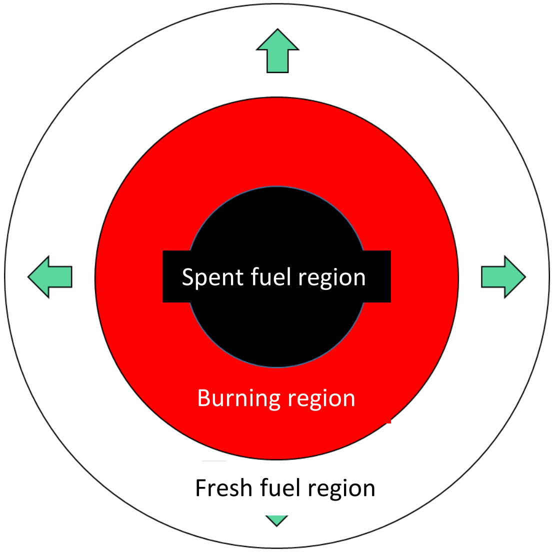
- Once through fuel cycle fast reactor
  - Fuel
    - Natural uranium or depleted uranium
  - Burnup characteristics
    - Fertile is converted into fissile in the reactor
    - Fissile is consumed by fissions
  - Fuel cycle
    - Once through
    - Enrichment facility nor Reprocessing facility is not needed.
- Lead alloy coolant in fast reactor (Lead or Lead-bismuth eutectic)
  - Hard neutron spectrum
  - Low neutron absorption
  - Low neutron leakage
  - Chemically inert with air or water

Background (continued)

- Study for once thorough fuel cycle fast reactors
  - Breed and Burn reactor
  - TWR
  - CANDLE
- Movement of burning wave
  - Axial direction
    - CANDLE
  - Radial direction (or horizontal direction)
  - No wave (random shuffling)



Axial movement of burning wave



Radial movement of burning wave

Background (continued)

- Advantage and disadvantage of each concept
  - In axial movement of burning wave
    - Reloading of a part of core
  - In radial movement of burning wave
    - Loading of low multiplication factor fuel assembly at the core center where neutron importance is large
    - Change of radial power density profile in operation
  - In random shuffling
    - Difference of power density between neighboring fuel assemblies
- One of the ideas of solutions (Rotational fuel shuffling)
  - Loading fresh fuel at the edge, approach to the center and move out to the outer edge
    - High infinite multiplication factor fuel at high neutron importance region
    - Close multiplication factors of neighboring fuel assemblies

Concept of rotational fuel shuffling scheme

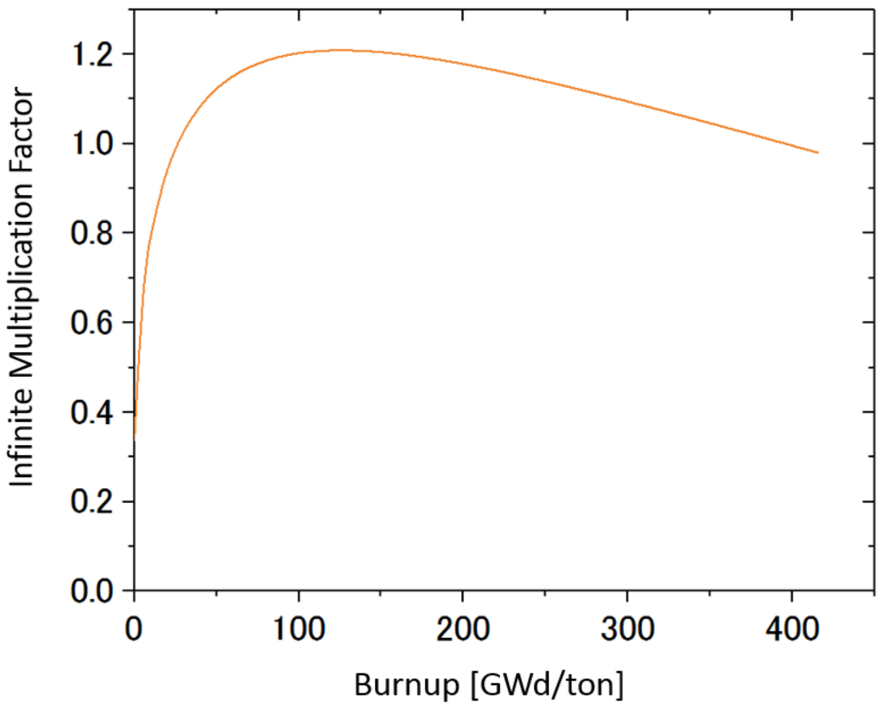
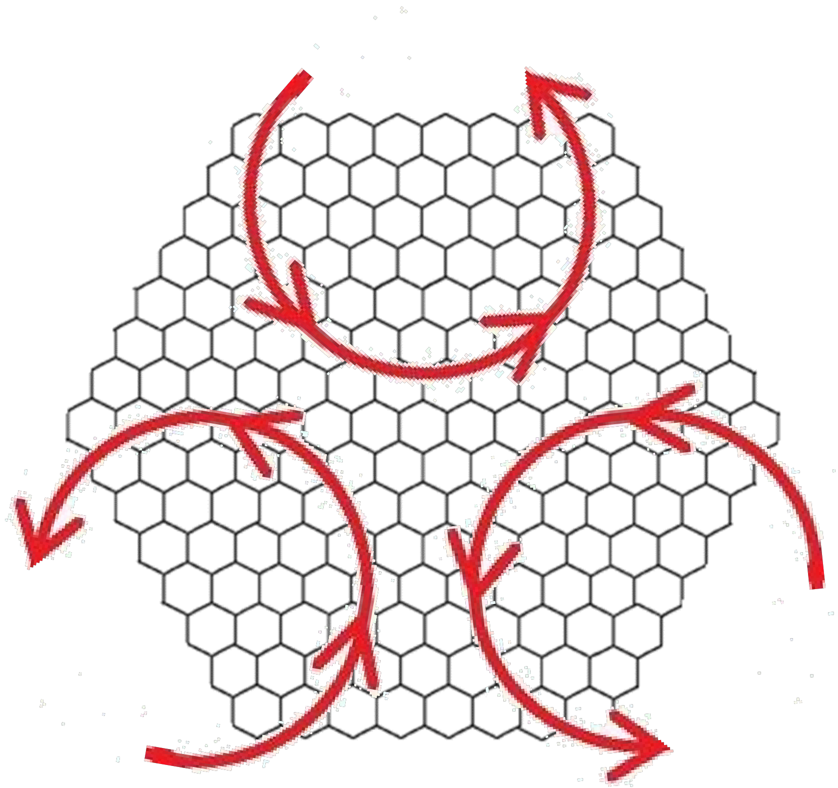


Fig. Change of infinite multiplication factor along burnup



Expected advantage of the rotational fuel shuffling

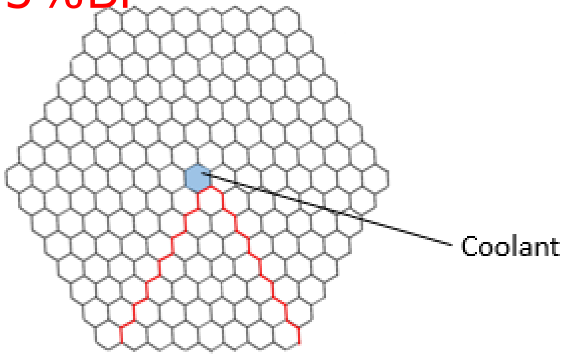
- High infinite multiplication factor fuel at high neutron importance region
  - Easy to make critical
- Close multiplication factors of neighboring fuel assemblies
  - Continuous power density profile
- Stable nuclide density distribution in the core
  - Stable power density profile

Purpose of study

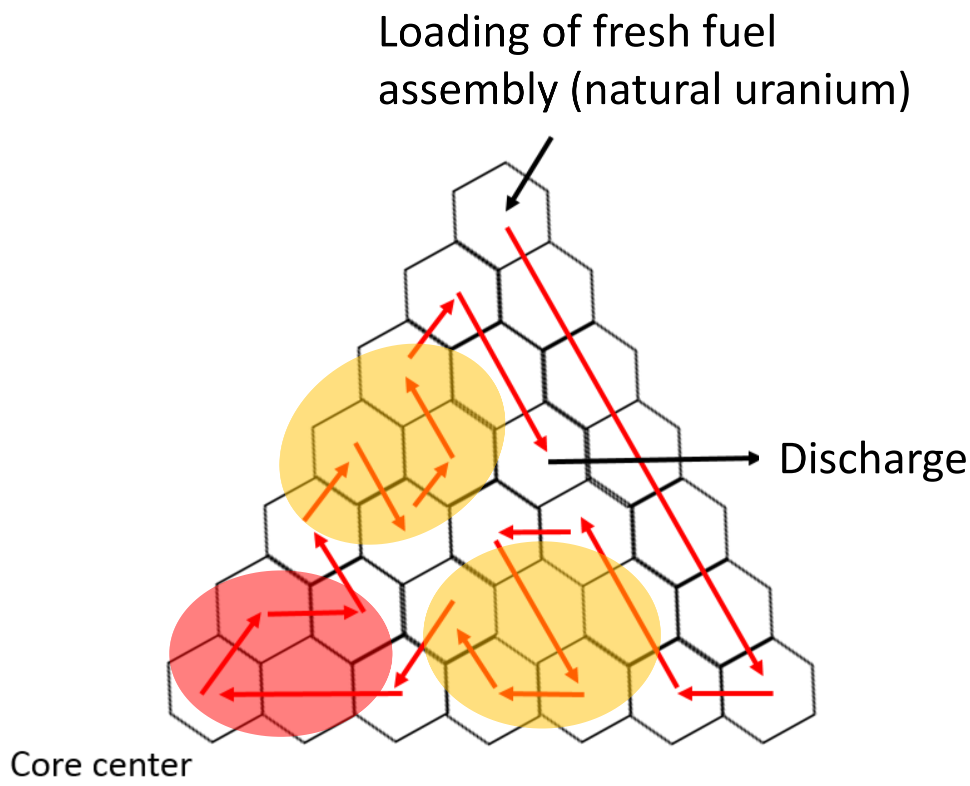
- To show the possibility of once-through fuel cycle fast reactor that has enough reactivity in operation, stable power density profile, and small power difference between the neighboring fuel assemblies by the rotational fuel shuffling

Core design

Thermal Power	740 MW
Core height	220.0 cm
Core equivalent radius	123.4 cm
Fuel	Nat. U+10%Zr
Fuel Cladding material	Ideal 9Cr-ODS
Coolant	44.5%Pb+55.5%Bi
Fuel radius	4.5mm
Fuel smear density	75%
Fuel pin radius	5.1 mm
Fuel pin pitch	10.8 mm
Number of fuel assemblies	168 + 1 coolant channel
Fuel temperature	800 K
Coolant temperature	700 K



Shuffling pattern in trial



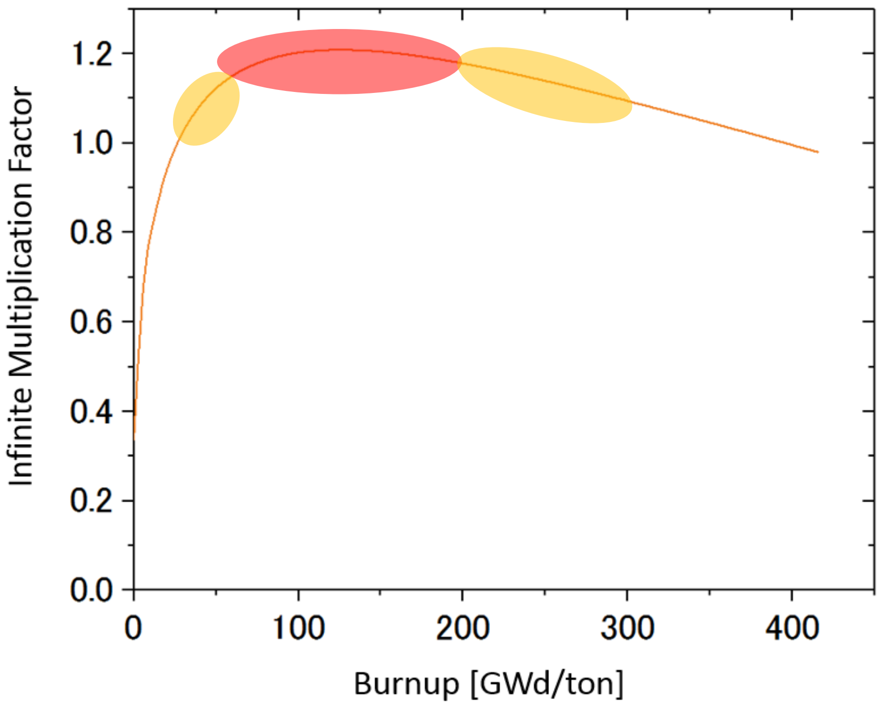


Fig. Change of the infinite multiplication factor of a fuel assembly along burnup

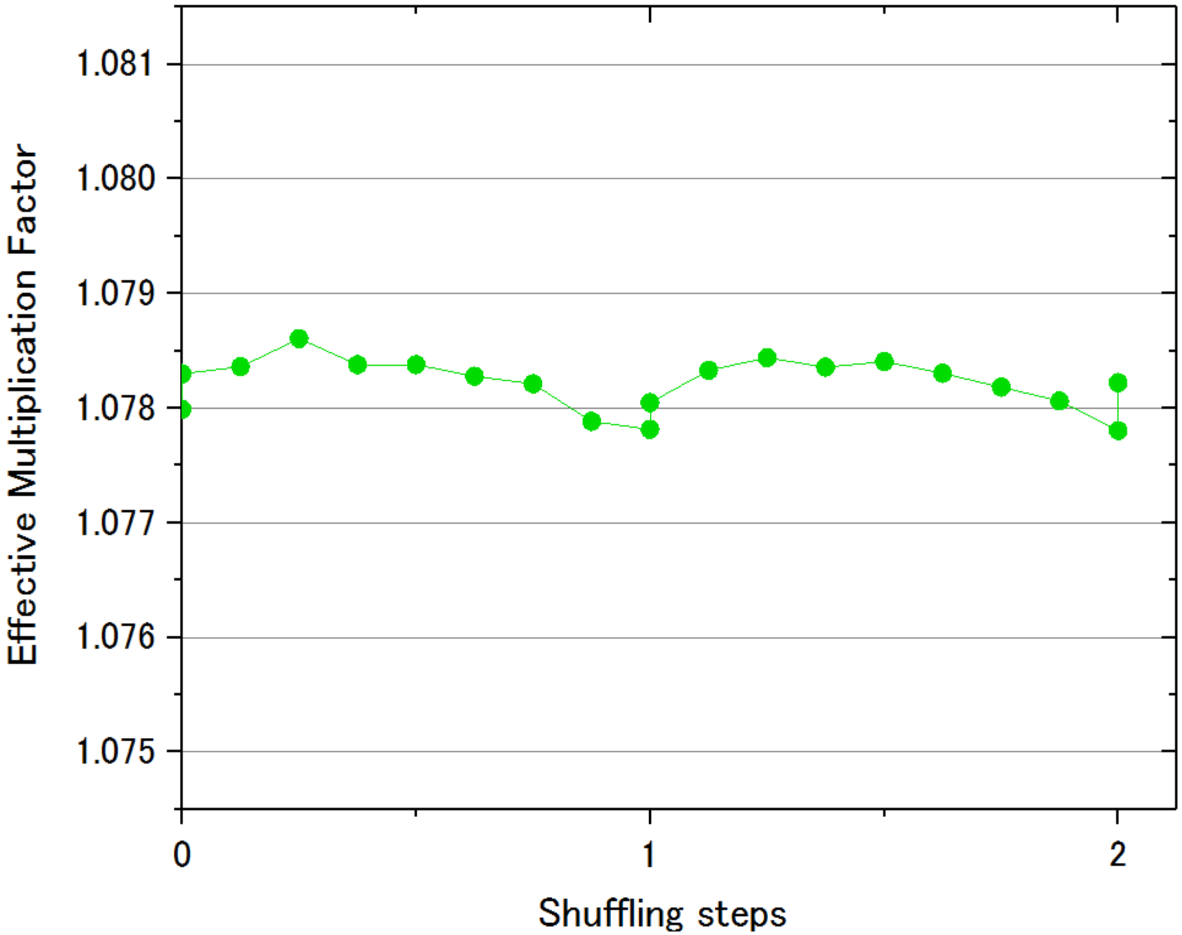
- Analysis
  - For 1/6 core
- Shuffling interval
  - 1000days

Analysis method

- Codes and library
  - Continues energy Monte Carlo code MVP2.0 and MVP-BURN with JENDL-4.0 nuclear data library
  - Additional original code for fuel assembly shuffling
- Condition in Monte Carlo and burnup calculation
  - Analysis for 1/6 core
  - 87 tally regions
  - 10,000 neutron histories per batch
  - 50 skipped batches
  - 50 active bathes
  - neutron transport calculation interval in burnup calculation
    - 125 days
- Calculation procedure
  - Begin the burnup calculation and shuffling from the natural uranium loaded initial core
  - Repeat the shuffling and burnup calculation until the burnup characteristics becomes equilibrium

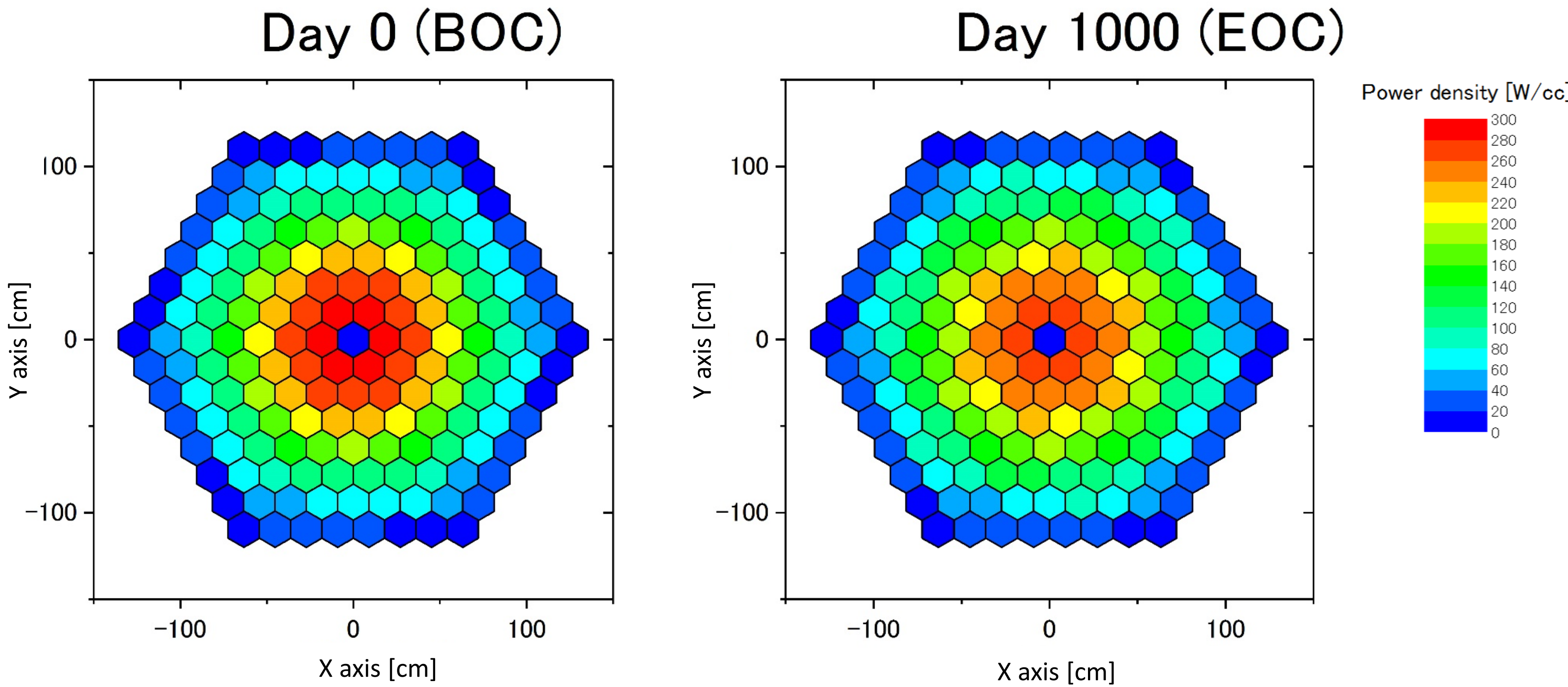
Results of analysis

Change of effective multiplication factor in the equilibrium condition



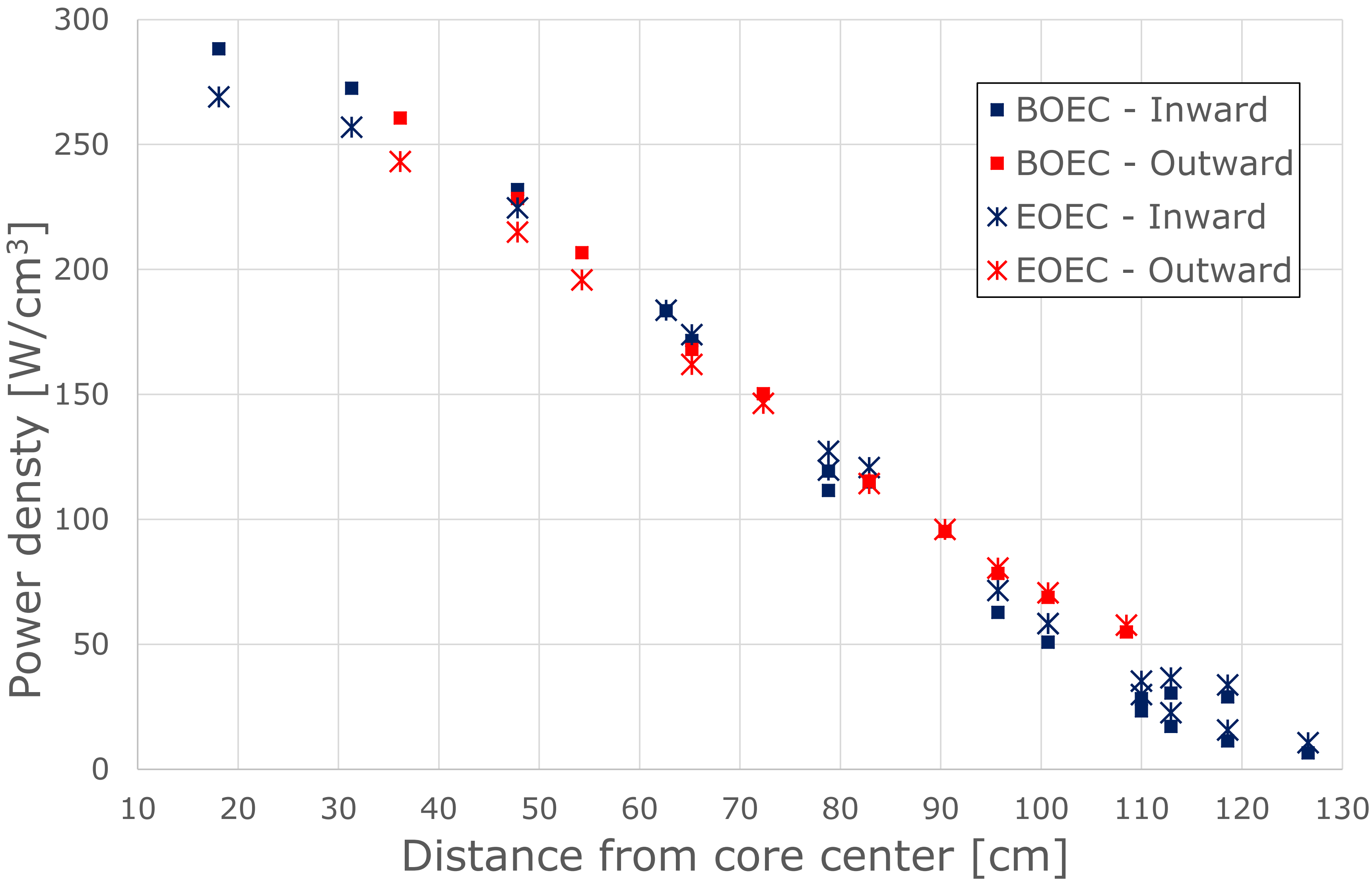
Average discharge burnup: 300 GWd/t

Average power density of each assembly at BOC and EOC

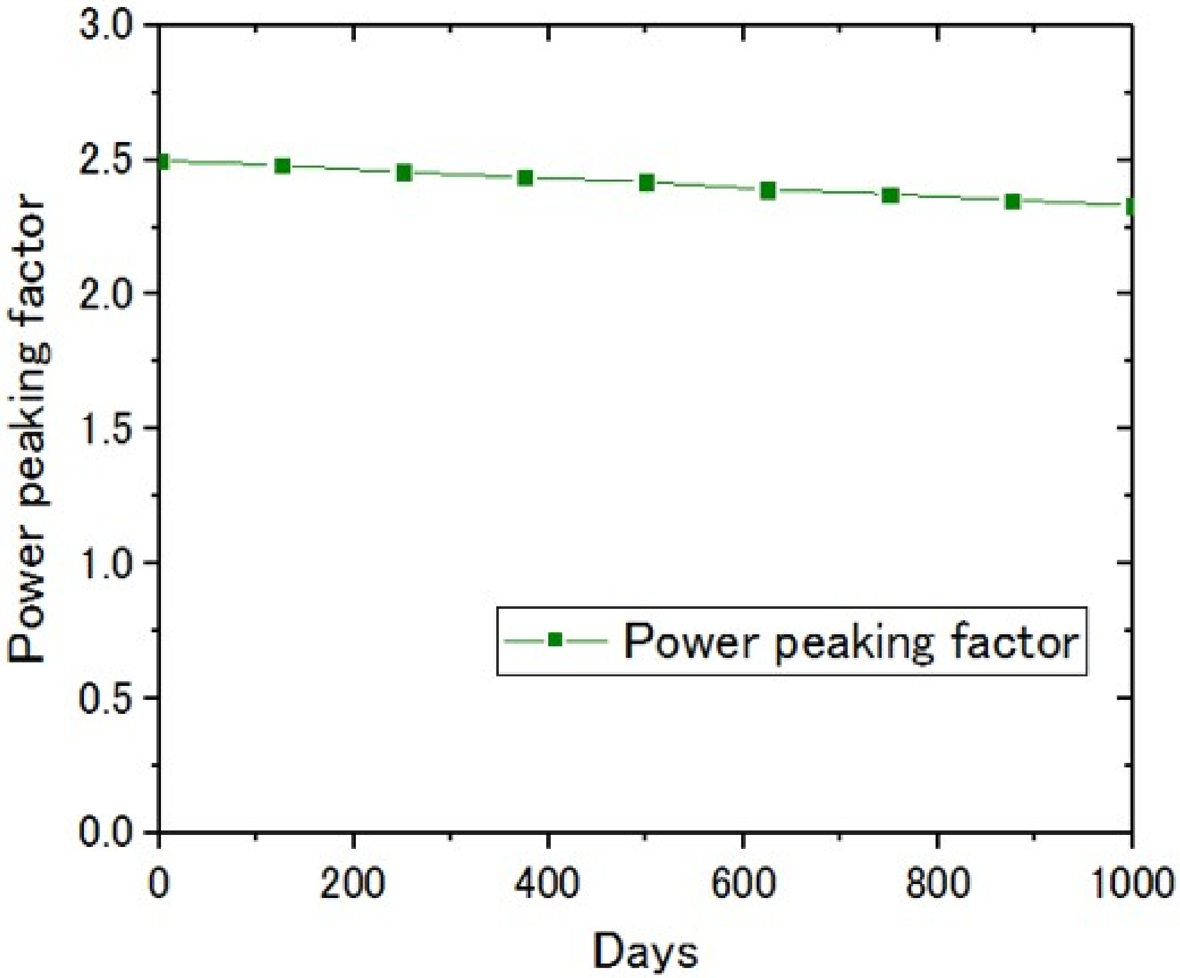


BOC: Beginning Of the Cycle (one shuffling step)
EOC: End Of the Cycle (one shuffling step)

Radial power density profile at BOC and EOC in equilibrium condition



Change of power peaking factor for radial direction in a shuffling interval



Conclusions

- The concept of LBE cooled once through fuel cycle fast reactor with rotational fuel shuffling scheme was shown.
- The reactor can achieve equilibrium condition and be critical. The reactor power density profile can be stable in the equilibrium condition.
- It means the concept has a possibility to design an once thorough fuel cycle fast reactor by solving the issues which the conventional concepts have.
- This preliminary analysis was to discuss in the equilibrium condition. The study for the optimum initial core will be a future work.
- The integrity of fuel cladding is another issue to be solved.

Summary

- Features of lead cooled fast reactor
  - Several attractive features by the use of lead or LBE coolant
- Development stages of GIF member states
  - The constriction of BREST-OD-300 in Russia can be started after the completion of licensing procedure.
- LFR study in Tokyo Tech
  - Stationary Wave Breed and Burn Reactor with LBE coolant

Tokyo Tech

Thank You