Preliminary Core Design for Long life Small Modular LFR

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Content

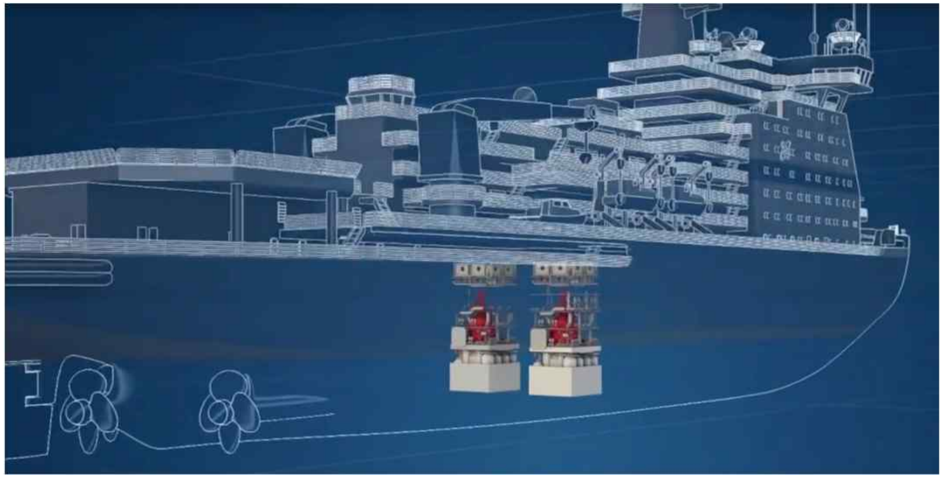
1. Introduction

1. Introduction (1/5)

- **Development of a conceptual design of long-cycle Small Modular Liquid-metal Fast Reactor (SMLFR) cooled by Lead-Bismuth Eutectic (LBE) for a power reactor.**
- **Main design criteria:**
  - **Transportable (ex. spent nuclear fuel (SNF) cask)**
    - **electricity demand in remote areas, off grid locations, icebreaker ship;**
  - **A long operating cycle (25~30 years without refueling) and small burnup reactivity swing.**
- **Computer codes for design and analyses**
  - **Fast reactor analysis code suite MC^2-3/TWODANT/REBUS-3 developed by Argonne National Laboratory (ANL).**
  - **Inhouse UNIST Monte Carlo code MCS**

1. Introduction (2/5)

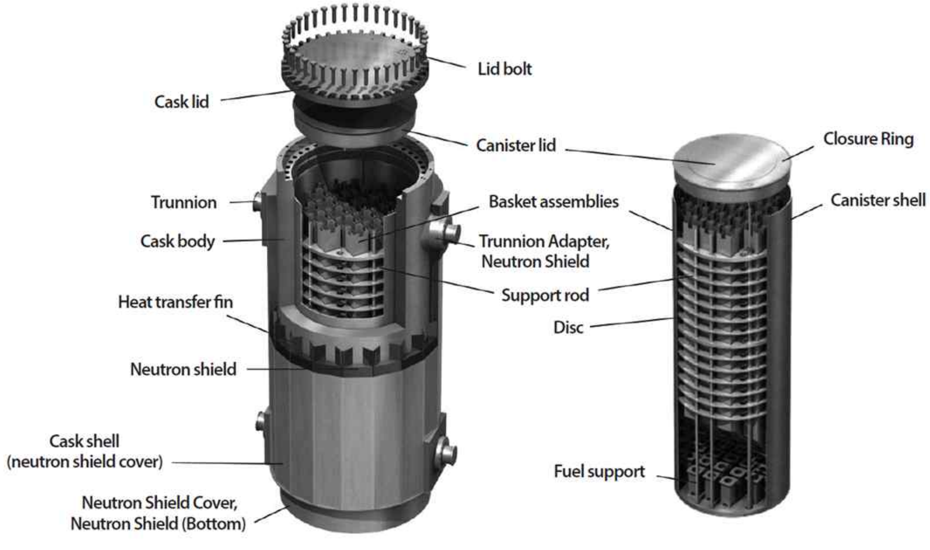
Icebreaker ship



- **LK-60 class icebreakers: powered by two RITM-200 integral pressurized water reactors (PWR), each rated at 175 MW thermal, and together delivering 60 MW (80,460 hp) to an electric motor propulsion system driving three shafts (Arktika - launched on 16 June 2016 at the Baltic Shipyard in St. Petersburg, Russia).**

Source: https://lynceans.org/all-posts/manufacturing-the-reactor-vessel-for-an-ritm-200-pwr-for-russias-new-lk-60-class-of-polar-icebreakers/

Spent Fuel Cask – MP187 [1]



(a) Cask (b) Canister

Item	Dual Purpose Metal Cask
Design basis fuel	
- Max. Burnup, MWd/MTU	45,000
- Min. Cooling period, year	10
- Max. Decay heat, kW/canister	16.8
Dimension, m	
- Canister	(O.D) 1.7 x (L) 4.9
- Cask	(O.D) 2.2 x (L) 5.3
Total weights, ton	104.7 (with a canister)
Cooling	Internal heat conduction & external natural convection

1. Ho-Seog Dho et al. "The Evaluation of Minimum Cooling Period for Loading of PWR Spent Nuclear Fuel of a Dual Purpose Metal Cask." *Journal of Nuclear Fuel Cycle and Waste Technology* 14.4 (2016): 411-422.

1. Introduction (3/5)

Core Design Requirements and Key Parameters

- **Fuel type: UN**
  - **Higher thermal conductivity, melting temperature and fissile density than oxide fuel;**
  - **^{235}U enrichment < 20w/o due to Korea's restriction on nuclear fuel.**
- **Coolant: LBE (Lead-Bismuth Eutectic)**
  - **Low melting point and high boiling temperature;**
  - **Outstanding capacity of heat transmission and neutronic features, and also it is chemically inert.**
- **Structure material: stabilized steel 15-15 Ti**
  - **Excellent thermal conductivity;**
  - **Irradiation and swelling resistance.**

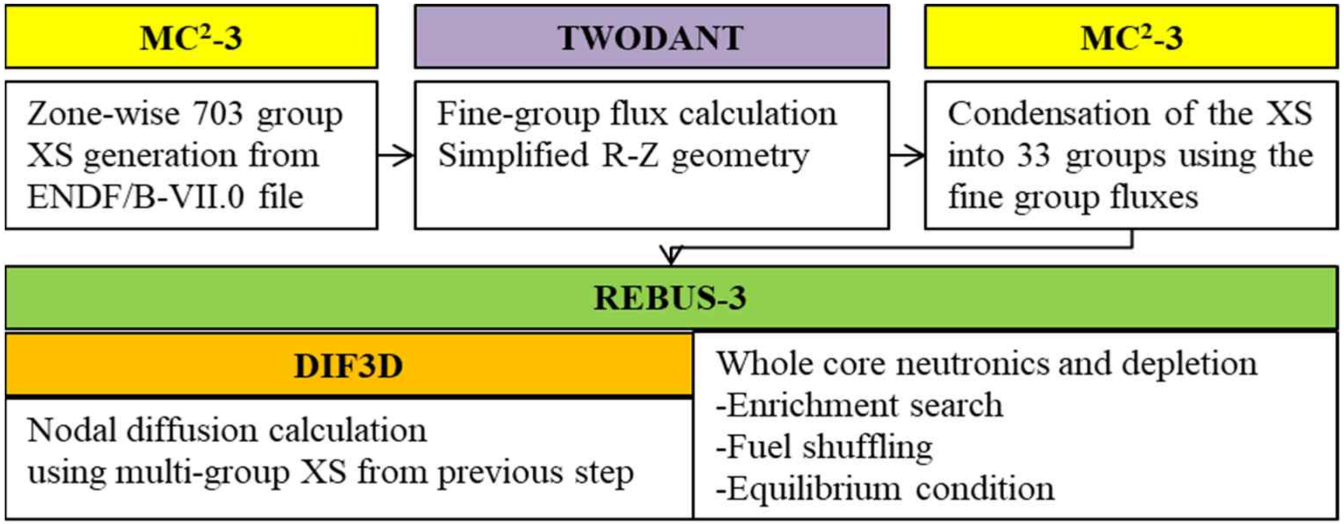
Parameter	Unit	Value
Thermal/Electric power	MW	37.5/15.0
Target cycle length	EFPY[a]	25~30
Assembly concept	-	Hexagonal
Fuel material	-	UN
- Smear density	%TD[b]	85
- Maximum ^{235}U enrichment	w/o	19.75
Clad material	-	15-15 Ti
Coolant	-	LBE
- Inlet/Outlet temperature	^{o}C	300/400
- Maximum coolant velocity	m/s	2.0
- Pressure	MPa	0.1

[a] *Effective Full Power Year*

[b] *Theoretical Density*

1. Introduction (4/5)

- **Fast reactor package code system, called Argonne Reactor Computation (ARC):**
  - **MC²-3 for multigroup cross-section generation;**
  - **TWODANT for the Boltzmann transport equation solution;**
  - **REBUS-3 for the analysis of fast reactor fuel cycles.**
- **Two steps analysis:**
  - **Step 1: generation of region-wise flux spectra with TWODANT/MC²-3 → broad-group cross-section condensation with MC²-3.**
  - **Step 2: REBUS-3 nodal diffusion calculation using 33-group cross-sections.**



1. Introduction (5/5)

- **Monte Carlo Code MCS [1]**
  - **Language: Fortran 2003**
  - **Purpose**
    - **Large Scale Reactor Analysis** with accelerated Monte Carlo simulation
    - University research: MC methodology development, advanced reactor design
  - **General 3-D geometry (CSG)**
  - **Nuclear Data**
    - ENDF-B/VII.0 and ENDF-B/VII.1
    - Continuous energy and multi-group
    - Double indexing method
  - **Physics**
    - Resonance upscattering (DBRC, FESK)
    - Probability table method
    - $S(\alpha, \beta)$
    - On the fly Doppler broadening
    - Cobra-TF/FRAPCON coupling
    - Photon transport
  - **Acceleration**
    - MOC and MC Hybrid solver
    - Modified power iteration
    - Wielandt method
    - CMFD
  - **Parallelism**
    - Parallel fission bank
  - **Depletion**
    - CRAM , MEM, Krylov Subspace
    - Hybrid depletion

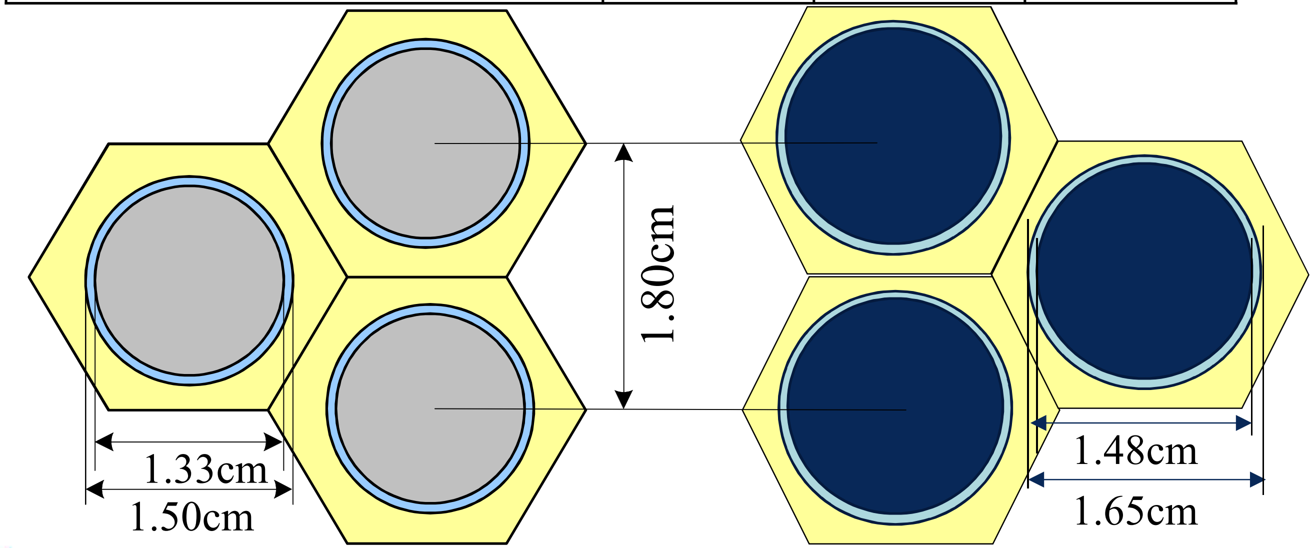
[1] J. Jang et al, "Validation of UNIST Monte Carlo Code MCS for Criticality Safety Analysis of PWR Spent Fuel Pool and Storage Cask," Annals of Nuclear Energy, 114: 495-509. (2018).

2. The Conceptual Core Design

2. The Conceptual Core Design (1/5)

Pin design parameters

Parameter	Fuel	Reflector	Control
Pin data			
- Pin material	UN	ZrO_2	B_4C
- Diameter, cm	1.5	1.65	1.5
- Cladding thickness, cm	0.085	0.085	0.085
- Pin pitch, cm	1.8	1.8	1.8
- Pin pitch-to-diameter ratio	1.2	-	-
Volume fraction at manufacture, %			
- Fuel/Reflector/Absorber	49.5	61.3	49.5
- Cladding	13.5	14.9	13.5
- Coolant	37.0	23.8	37.0



2. The Conceptual Core Design (2/5)

Sensitivity test cases

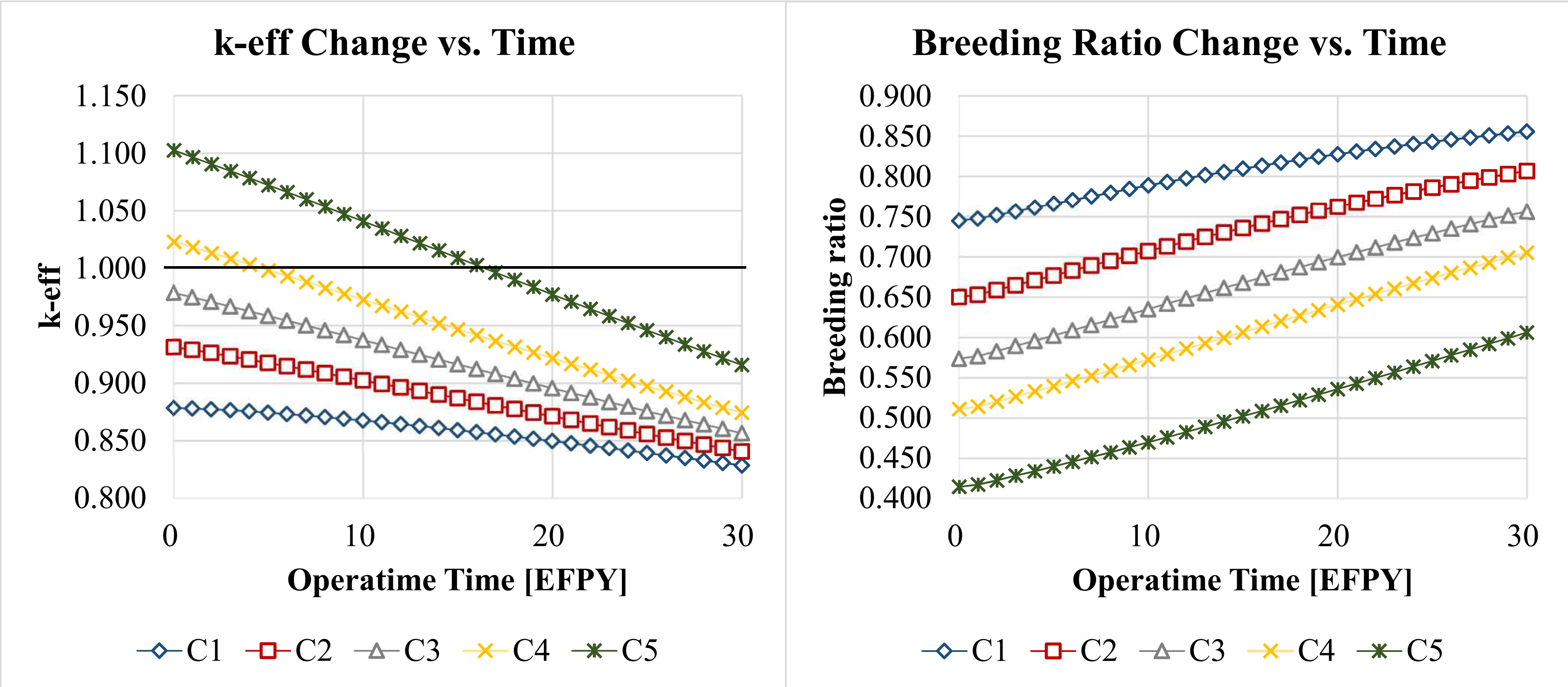
Core type	Equivalent core diameter, cm	Core height, cm	LEU enrichment, w/o
Type C			
- C1	80	120	12.0
- C2	80	120	13.5
- C3	80	120	15.0
- C4	80	120	16.5
- C5	80	120	19.5
Type E			
- E1	120	120	13.5
- E2	120	144	13.5
- E3	120	168	13.5
- E4	120	192	13.5
- E5	120	216	13.5

C1-C5: to analyze the breeding behavior of the core
E1-E5: to achieve to target cycle length

2. The Conceptual Core Design (3/5)

Sensitivity test on ^{235}U enrichment: Core height of 120 cm

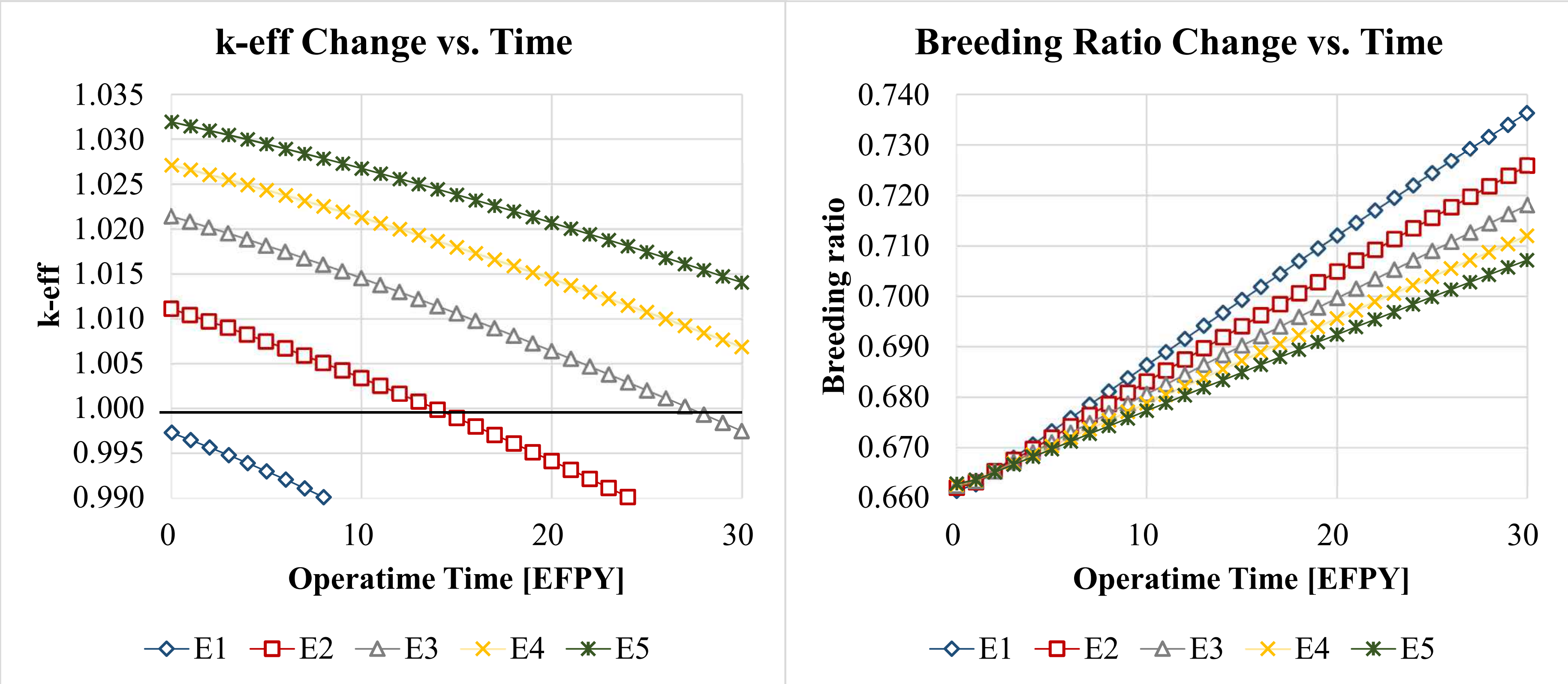
Type C (Diameter: 80 cm)	C1-12.0%	C2-13.5%	C3-15.0%	C4-16.5%	C5-19.5%
k_{eff} - BOC - EOC	 0.87850 0.82850	 0.93062 0.84129	 0.97863 0.85660	 1.02310 0.87433	 1.10291 0.91600
Cycle length, EFPY	0	0	0	4.66	16.44



2. The Conceptual Core Design (4/5)

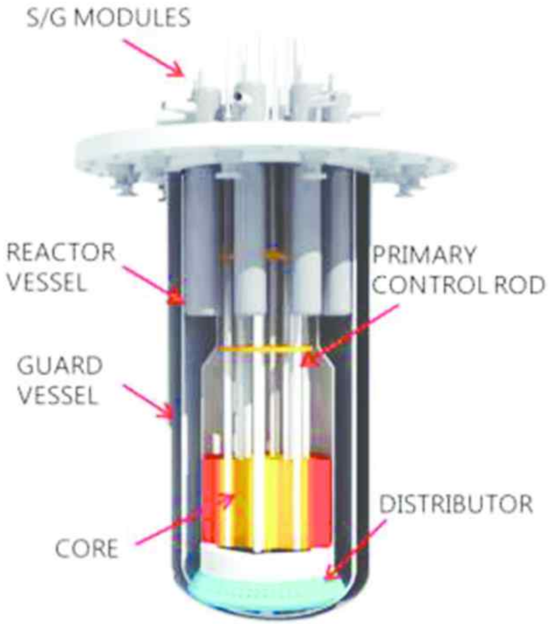
Sensitivity test on active core height (13.5% enrichment of ^{235}U)

Type E (Diameter: 120 cm)	E1-120cm	E2-144cm	E3-168cm	E4-192cm	E5-216cm
k_{eff} - BOC - EOC	 0.99725 0.96431	 1.01113 0.98378	 1.02145 0.99745	 1.02715 1.00685	 1.03200 1.01404
Cycle length, EFPY	0	13.82	27.23	38.47	>40



2. The Conceptual Core Design (5/5)

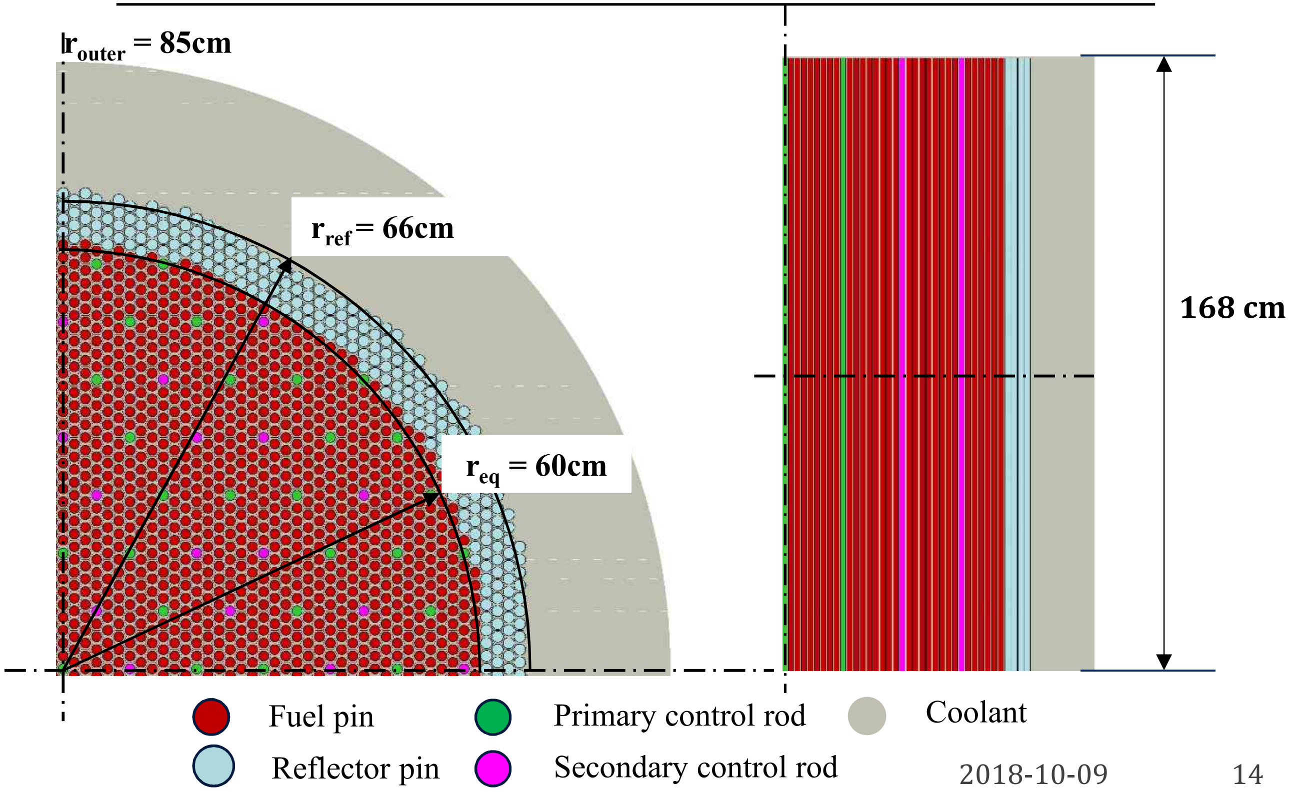
Optimized core design



Reactor module configuration of URANUS-40[1]

Parameters	Value
Core data	
- No. of fuel pin	3,780
- No. of control rod	155
- No. of reflector rod	948
Active core equivalent diameter, cm	120
Active core height, cm	168
Equivalent reflector thickness, cm	6

1. Yong-Hoon Shin et al. "ICONE23-2135 DESIGN STATUS OF SMALL MODULAR REACTOR COOLED BY LEAD-BISMUTH EUTECTIC NATURAL CIRCULATION: URANUS." In *The Proceedings of the International Conference on Nuclear Engineering (ICONE) 2015.23*, pp. _ICONE23-2. The Japan Society of Mechanical Engineers, 2015.

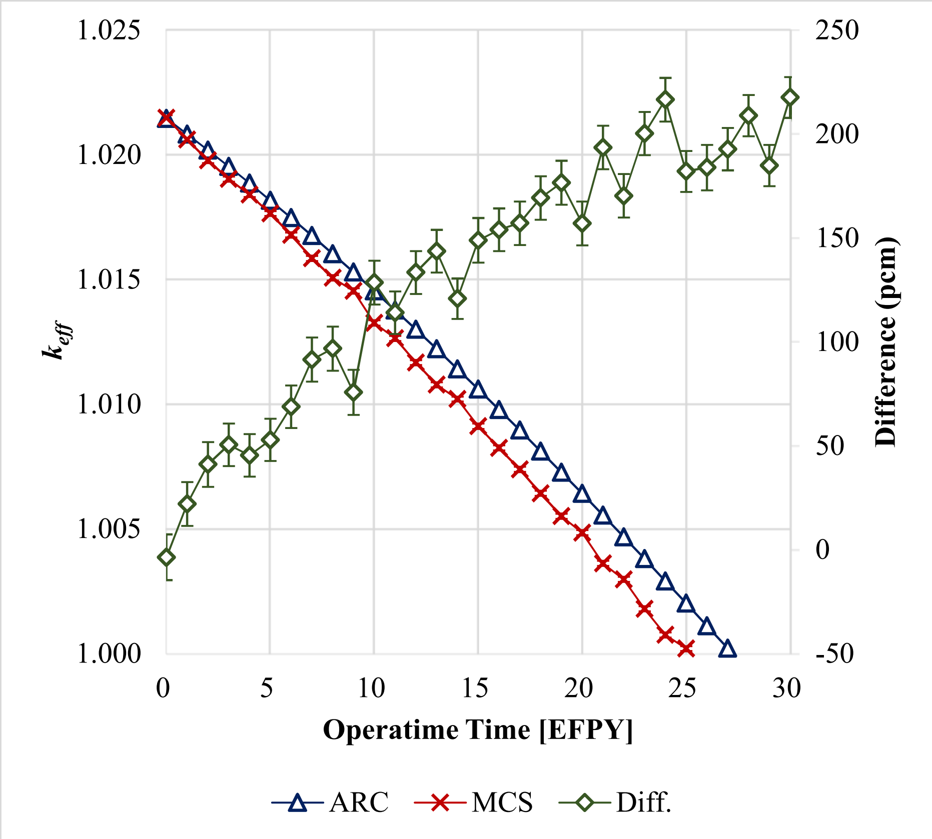


3. Performance analyses

3.1. Neutronic performance
3.2. Safety analysis

3.1. Neutronic performance (1/4)

Core multiplication factor, power and coolant parameters



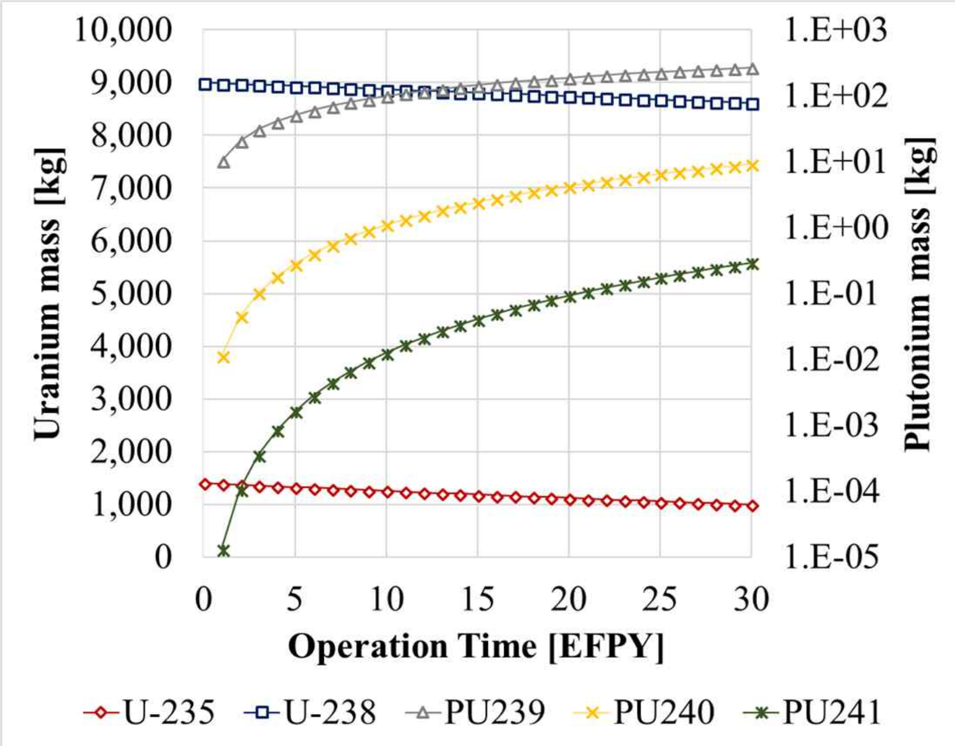
Parameter	Value
Cycle length, EFPYs	27.23
Excess reactivity, pcm	2,100
Average breeding ratio	0.70
Power	
- Average specific power density, MW/tHM	3.60
- Average power density, W/cm^3	20.24
- Linear power density, W/cm	59.05
Coolant (in active core region)	
- Total mass, kg	7,499.5
- Fluid velocity, m/s	0.580
- Flow rate, kg/s	2,589.5

- **>25 EFPYs operation time without refueling**
- **Low average power density due to its small size.**

3.1. Neutronic performance (2/4)

Change in heavy isotope mass over time

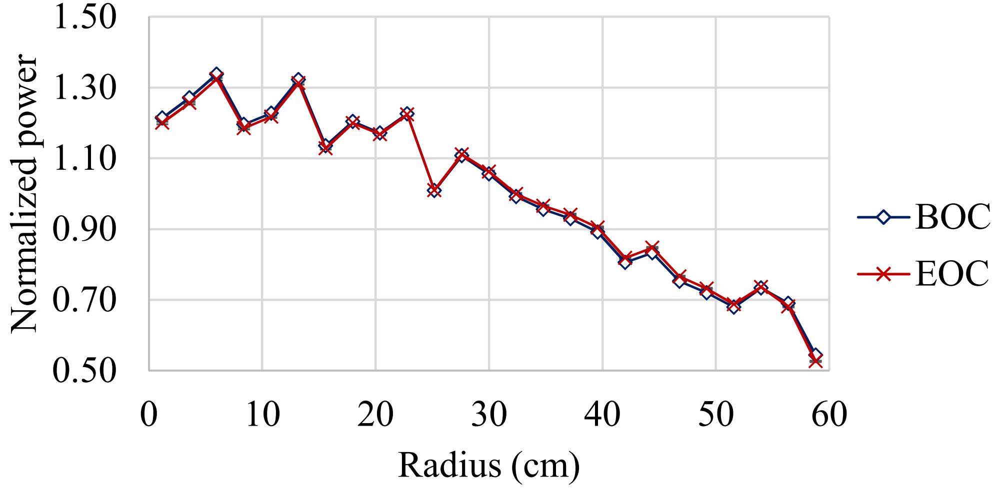
- **Loading heavy metal mass: 10.41 tons.**
- **During 30 EFPYs of depletion:**
  - **0.36 ton of ^{235}U and 0.33 ton of ^{238}U are consumed;**
  - **Low conversion rate of ^{238}U to ^{239}Pu → reactor breeding ratio < 1;**
  - **Average isotopic discharge burnup: 35.3 GWd/MTU;**



3.1. Neutronic performance (3/4)

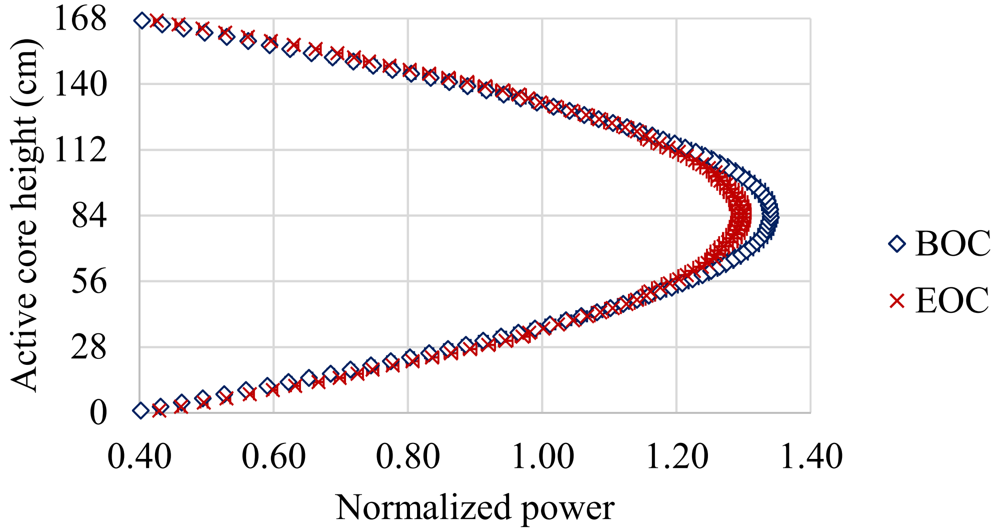
Normalized Power Distribution at BOC and EOC by MCS.

- **Movement of active core region from the center at BOC to the periphery at EOC, but slow transfer rate due to the small breeding ratio.**



Normalized Radial Power Distribution at BOC and EOC.

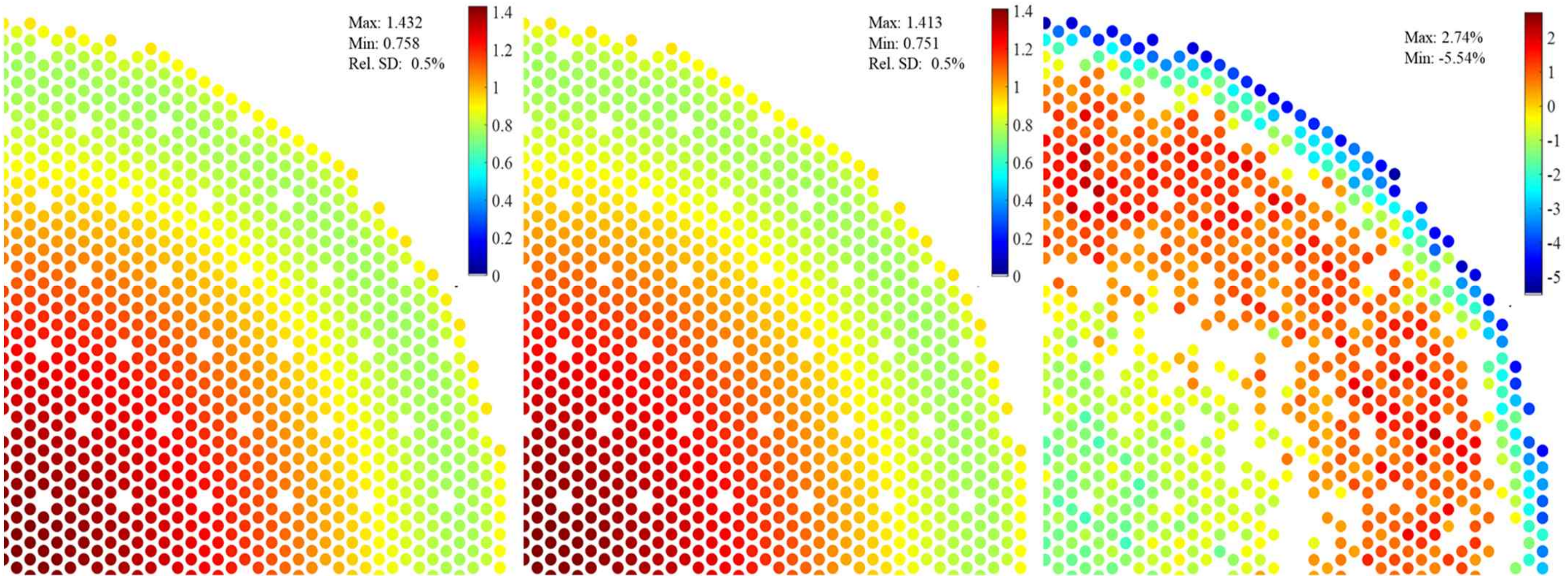
- **Axial power rate during the depletion (BOC → EOC):**
  - **Decreases at the center;**
  - **Increases at the bottom and the top of active core height.**



Normalized Axial Power Distribution at BOC and EOC.

3.1. Neutronic performance (4/4)

Normalized Power Distribution at BOC and EOC by MCS.



Normalized pin-wise power distributions at BOC (left) and EOC (middle) and the EOC power difference vs. BOC (right, %)

3.2. Safety analysis (1/1)

Reactivity feedback coefficients

- **Parameters of interest calculated by MCS:**
  - **Effective delayed neutron fraction (β_{eff});**
  - **Primary, secondary. and total control rod worth;**
  - **Fuel temperature coefficient (FTC): $FTC\left(\frac{pcm}{K}\right) = \frac{\rho_{T_2} - \rho_{T_1}}{T_2 - T_1}$**
  - **Coolant void reactivity (CVR): $CVR\ (pcm) = \rho_{void} - \rho_{base}$**
    - where ρ_{base} is the reactivity at normal condition and ρ_{void} is the calculated reactivity when removing 15% coolant amount from the core.

Parameter		BOC	EOC
Effective delayed neutron fraction (β_{eff}), pcm		753±27	636±15
Primary control rod worth, pcm		-4,595±14	-4,536±15
Secondary control rod worth, pcm		-3,264±15	-3,266±15
Total control rod worth, pcm		-7,527±15	-7,486±14
FTC, pcm/K	950K	-0.860±0.050	-0.818±0.049
	1,100K	-0.700±0.026	-0.682±0.024
CVR, pcm	15% void	-394±15	-305±15

4. Conclusions and perspectives

4. Conclusions and perspectives (1/2)

- **Conceptual design of an SMLFR core cooled by LBE which satisfies advanced reactor criteria.**
- **Achieved results with ARC deterministic code system MC^2-3/TWODANT/REBUS-3 and UNIST Monte Carlo code MCS**
  - **A small size core with uniform uranium nitride fuel (1.68 m high and 1.2 m in diameter) → transportable in a SNF cask and for icebreaker ship;**
  - **>25 EFPYs operation time at full power without refueling;**
  - **The primary and secondary control rod worth is capable of managing the excess reactivity.**
  - **The FTC and CVR are calculated to be clearly negative during the full lifetime.**

4. Conclusions and perspectives (2/2)

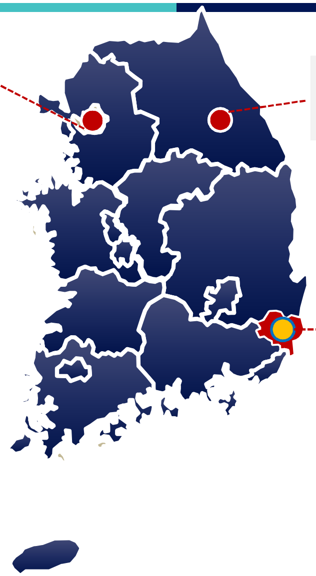
- **Limitations:**
  - **Focus only on the design of the reactor concept and the inherent safety;**
  - **Do not consider thermal-hydraulic feedback;**
  - **Do not represent a complete fuel cycle system.**
- **Future works:**
  - **More rigorous uncertainty evaluation: environment issues, the nuclear waste production and management, the economic concerns and the feasibilities of this SMLFR**

UNIST CORE

COmputational Reactor Physics & Experiment Lab in UNIST

- UNIST founded in 2009
- UNIST CORE founded in 2013

http://reactorcore.unist.ac.kr/



- Reactor physics code development
  - STREAM deterministic transport code
  - RAST-K nodal diffusion code
  - MCS Monte Carlo transport code
  - RXSP nuclear data processing code

UNIST campus

Shin Kori: 2 OPR-1000 & 4 APR-1400

Heavy Liquid Metal Coolants in Nuclear Technologies (HLMC-2018)
Obninsk, Russian Federation
October 08–10, 2018

国家电投
SPIC

Neutronics, Thermal-Hydraulics and Radiation Shielding Analysis of A 300 MWth LBE-cooled Fast Reactor Conceptual Design

Li Linsen, Yeoh Eing Yee, Wang Ziguan, Sun Canhui, Chen Yaodong, Li Yuquan
Advanced Nuclear Energy Technology Research Institute
State Power Investment Corporation Research Institute

2018.9

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About SPIC

June. 2015, State Power Investment Corporation (SPIC) was established through the merge of **China Power Investment Corporation** and **State Nuclear Power Technology Corporation.**

- SPIC is one of China's top five power generators.
- SPIC is one of China's three nuclear power developers and operators.
- SPIC is a Fortune Global 500 company.
- SPIC is committed to global businesses.

About SPICRI

State Power Investment Corporation Central Research Institute (SPICRI) was founded in September, 2015. The institute, with over 450 employees, is one of the five SPIC support platforms that for scientific innovation and a basement for technology talents.

Advanced Nuclear Energy

Software & Information

Advanced Material

Sustainable Energy

Hydrogen Energy

Power Plant Service

Big Data

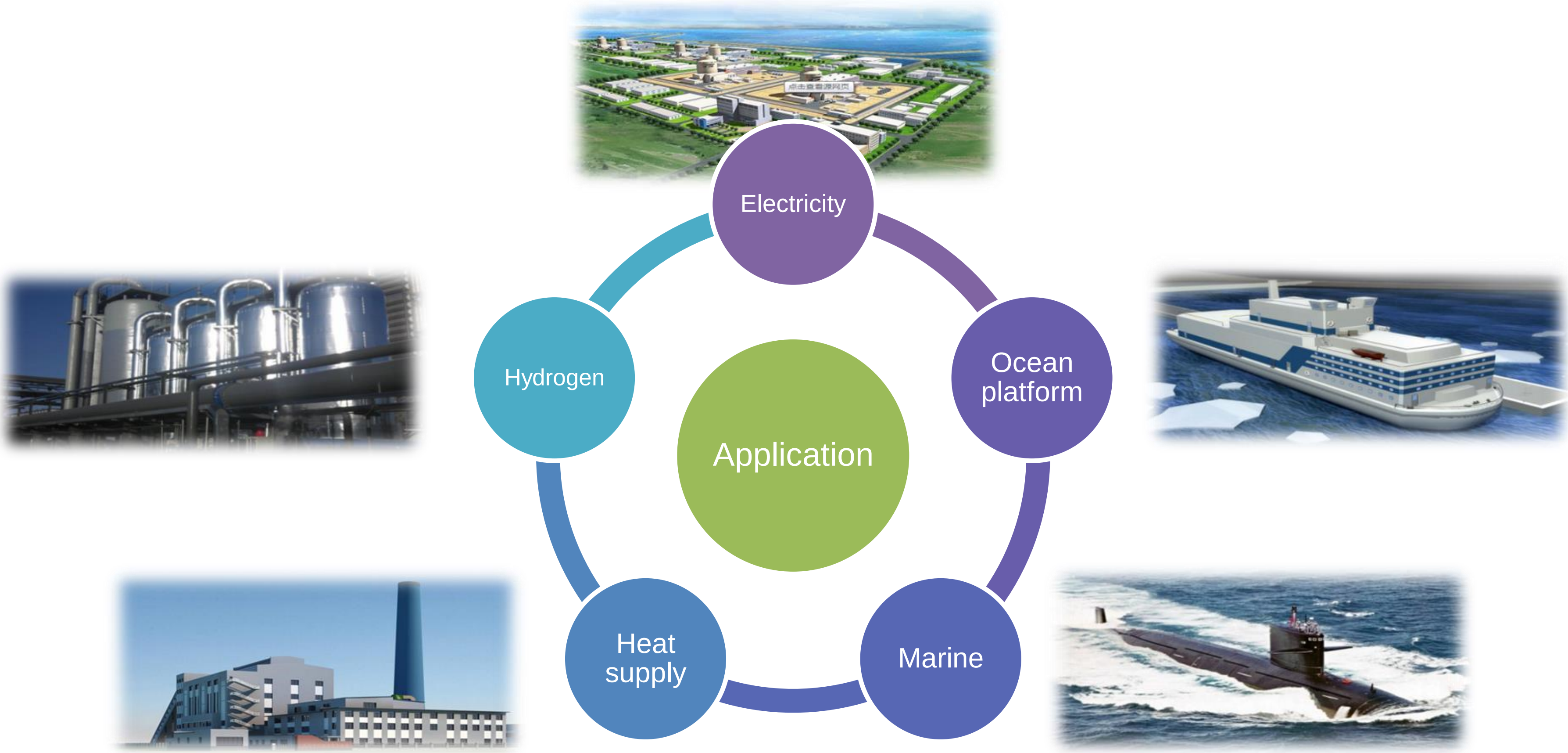
Strategy and Economy

Organization of SPICRI

1. Introduction

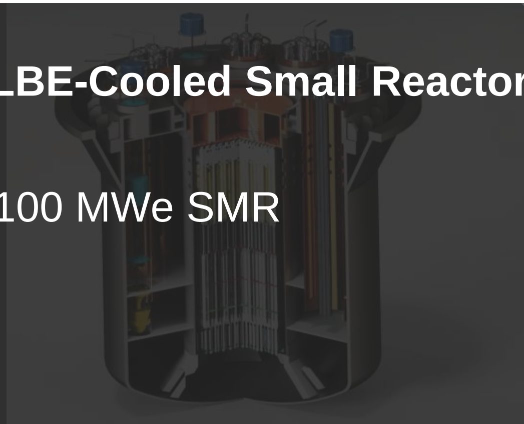
1.1 Applications of LFR

Lead-cooled Fast Reactor (LFR), as one of the six nuclear reactor technologies selected by the Generation IV International Forum (GIF), has become one of the most promising concepts and drawn attentions from the industry in China due to its wide application potential.



1. Introduction

1.2 Development and roadmap of BLESS



LBE-Cooled Small Reactor

100 MWe SMR

SMR

Modular Reactors

Commercial LFR

High Power, Lead-cooled

Multi-Use Design

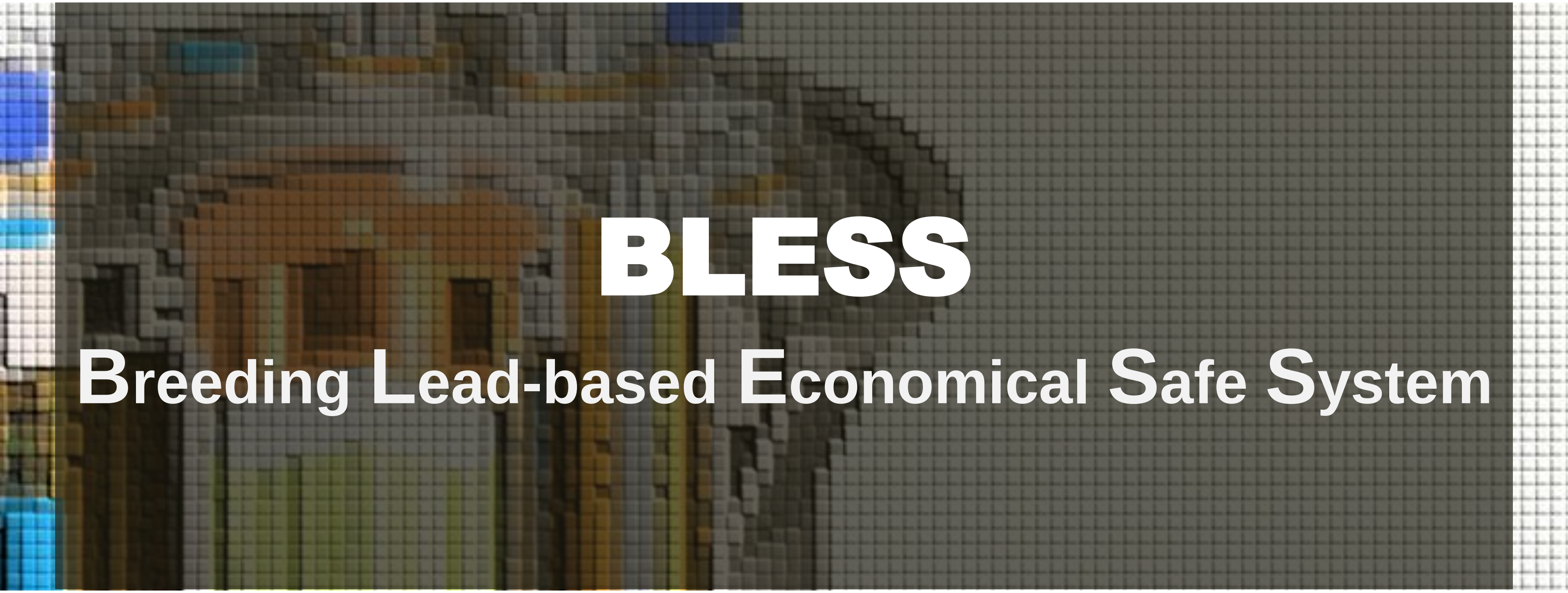
Ocean Platform

Transportable ……

1. Introduction

1.2 Development and roadmap of BLESS

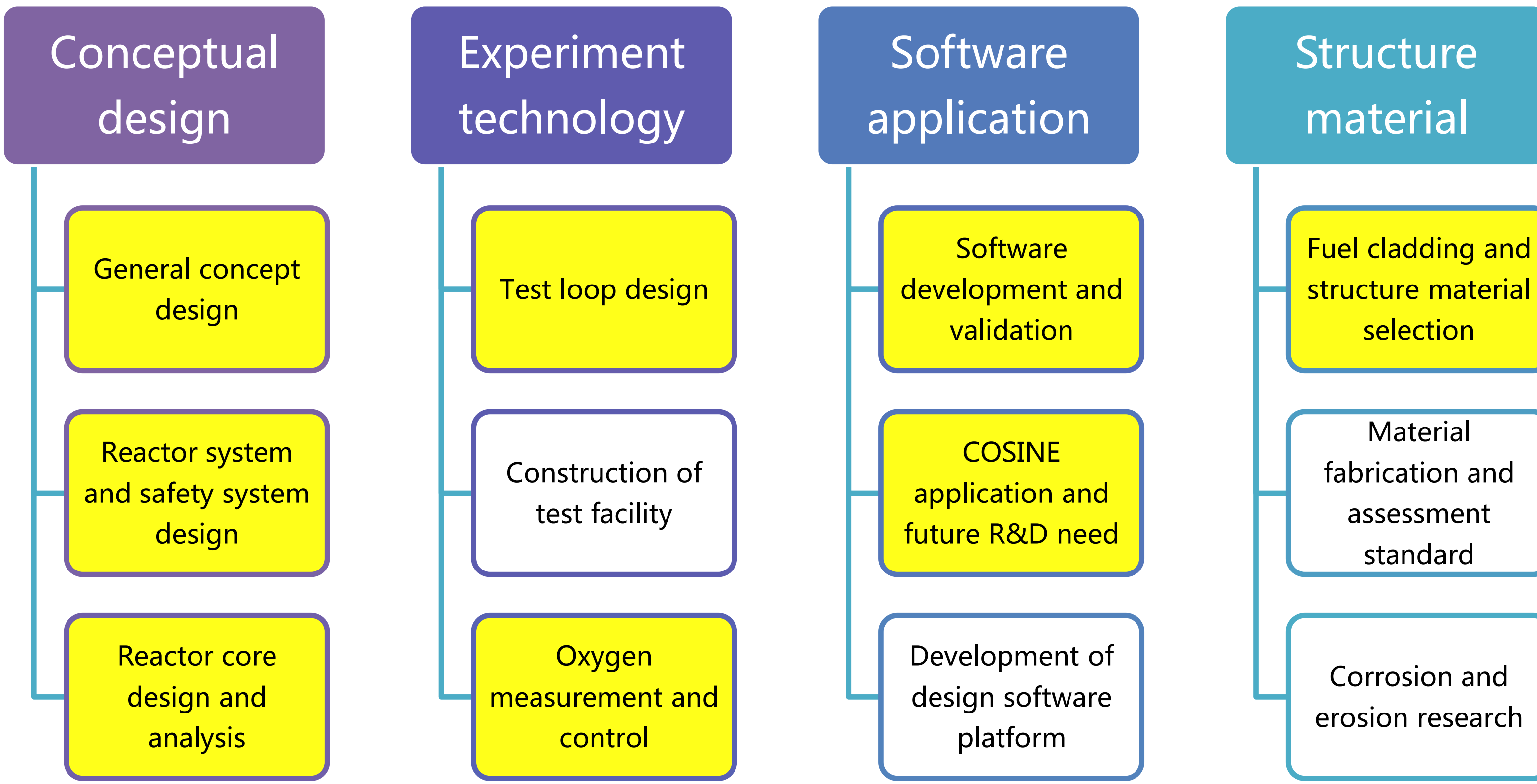
A LBE (Lead-Bismuth Eutectic)-cooled Fast Reactor - BLESS (Breeding Lead-based Economical and Safe System) has been proposed and developed by China State Power Investment Corporation Research Institute (SPICRI) and designed to meet the public demands of a safer, more economical and more environmental-friendly nuclear system.



1. Introduction

1.2 Development and roadmap of BLESS

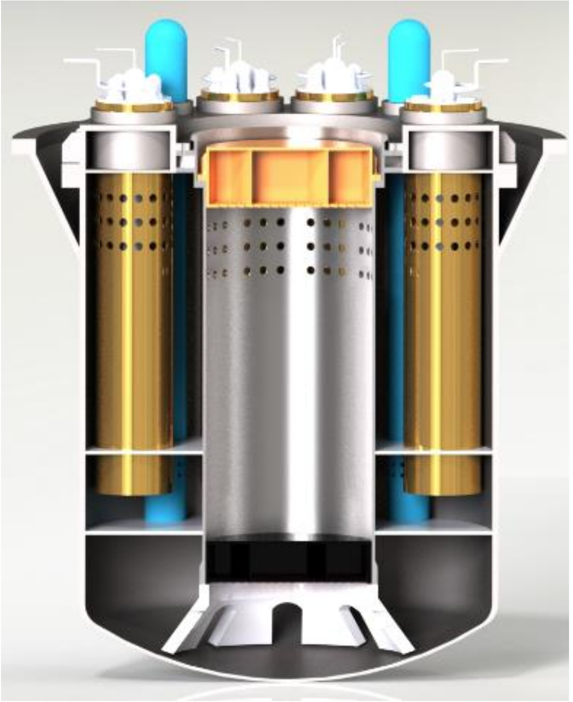
Recent Research Project funded by SPIC is focused on the BLESS-D concept, including research on conceptual design, structure material performance and software application.



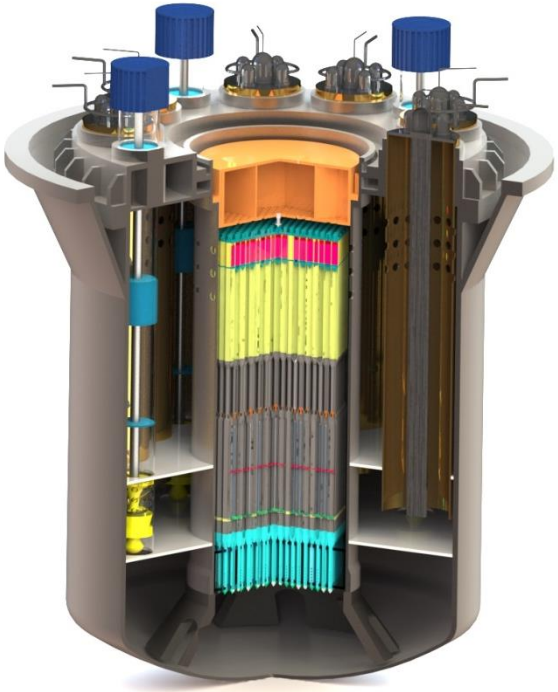
1. Introduction

1.3 Conceptual design of BLESS-D

In the roadmap among several proposed BLESS reactors, BLESS-D (BLESS-Demonstration) is a pool-type reactor cooled by LBE, which is devoted to demonstrate the technology of China LBE-cooled fast reactor. The thermal power of BLESS-D is 300 MW while the electric power is set at about 120 MW.



BLESS reactor vessel layout



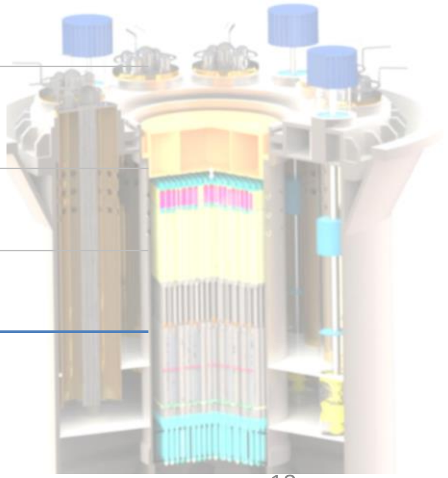
BLESS reactor design configuration

1. Introduction

1.3 Conceptual design of BLESS-D

BLESS-D Key Parameters

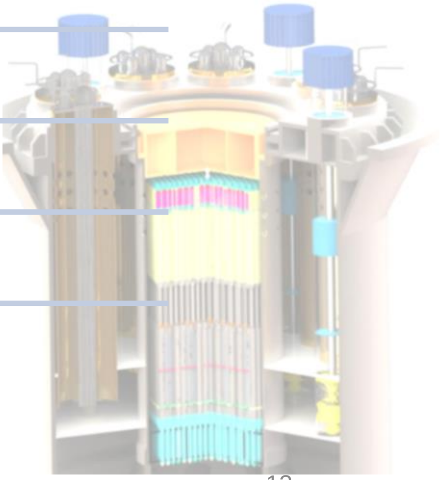
Thermal Power	300 MWt
Electrical Power	120 MWe
Fuel	UO_2
Enrichment	14%/16%/19.75%
Core height	70 cm
Vessel Diameter	2.5 m
Fuel Assembly	310
Fuel Rod in a Assembly	127
Coolant	LBE



1. Introduction

1.3 Conceptual design of BLESS-D

- Neutronics Design
- Reactor Shielding
- T-H Analysis
- SG Design & Analysis
- Primary Pump Design
- Primary System Design
- Secondary System Design
- Structure and Equipment Design and Analysis
- Safety System Design and Analysis



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2. Core neutronics analysis

☐ Objective

- Neutronics design in accordance with design criteria
- Research on the sensibility of geometry parameters on reactor performance
- Burn-up performance and breeding capability analysis
- Verification of key parameters calculation.

☐ Progress

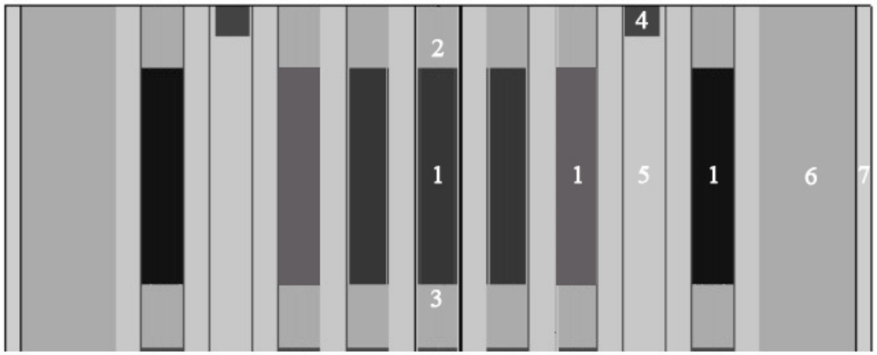
- ✓ A preliminary core design has been accomplished.
- ✓ Key parameters calculation completed: control rod worth, power distribution, burn-up calculation, reactivity coefficient.
- ✓ Verification calculation by the code RMC (Monte Carlo) has been done.
- ✓ Preliminary analysis on MOX fuel feasibility, Long-life waste disposal and breeding and transmutation capability.

2. Core neutronics analysis

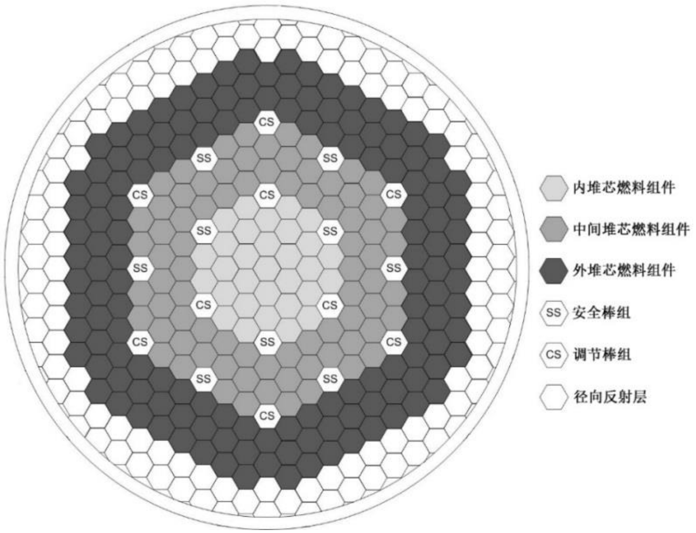
□ Analysis results

1. Core Layout

- Three Uranium enrichment (14%, 16%,19.75%) .
- With 200 mm reflector assemblies and 50 mm reactor vessel.
- Two Control rod systems, reactivity control (CS) and emergency shutdown (SS)



BLESS-D axial core layout



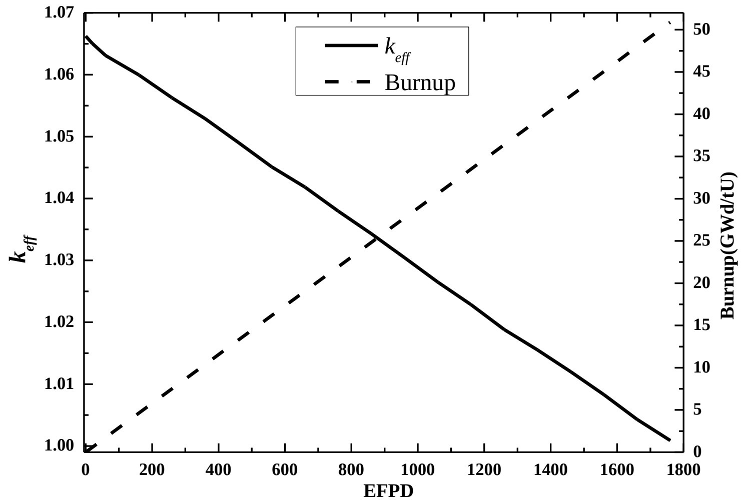
BLESS-D radical core layout

2. Core neutronics analysis

Analysis results

2. Burn-up calculation

- BOL k_{eff} **=1.06711(0.00013)** ;
- First cycle lifetime 1770 EFPD, or 5 EFPY
- Average burn-up **50.8 GWd/tU**



k_{eff} and burn-up against time

3. Reactivity coefficient

- All reactivity coefficients are negative for design criteria

	BOL	MOL	EOL
fuel temp. coefficient/(pcm/K)	-1.12 (0.04)	-1.13 (0.03)	-1.35 (0.05)
Coolant temp. coefficient/(pcm/K)	-0.44 (0.02)	-0.31 (0.02)	-0.12 (0.01)
Void coefficient/(pcm)	-3935 (21)	-3765 (18)	-3450 (16)

2. Core neutronics analysis

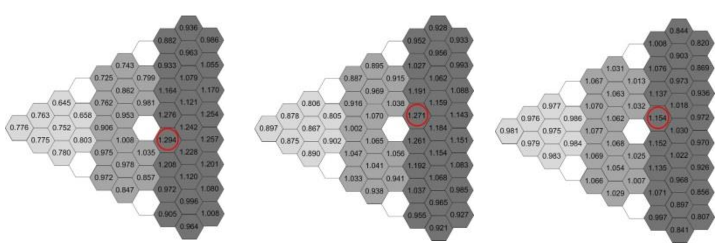
☐ Analysis results

4. Power distribution

- Power peak factor is acceptable. Core-level power distribution is well flattened and become better as the rising of burn-up depth

5. Control rod worth

- Reactivity control system (CS) works during normal operation status including reactor start, full power operation, etc. Emergency shutdown system (SS) normally extracted from the core and is inserted in an emergency shutdown status



Power distribution at BOL, MOL, EOL

CS+SS Reactivity worth and requirement

Parameters	Value/pcm
CS+SS total reactivity worth	13625
Total requirement	13024
Excessive reactivity at BOL	*6289*
Shutdown margin	*5263*
Worth of the most efficient CS Assembly	*754*
Worth of the most efficient SS Assembly	*718*

2. Core neutronics analysis

☐ Results

- Cycle lifetime is 5 EFPD with UO_2 fuel.
- All reactivity coefficients are negative, which comply with design criteria
- Power peak factor less than 1.294 in the duration of core cycle lifetime.
- Reactivity control systems meet the requirement.

☐ Future research

- Optimization of reactivity control system
- Optimization of core arrangement in accordance with T-H and shielding parameters
- Verification of key parameters with deterministic method code.

CONTENT

目录

3. Thermal-hydraulics analysis

❑ Objective

- Research on the thermal hydraulic analysis on reactor core using sub-channel code.
- Develop sub-channel code for BLESS and validate the code.

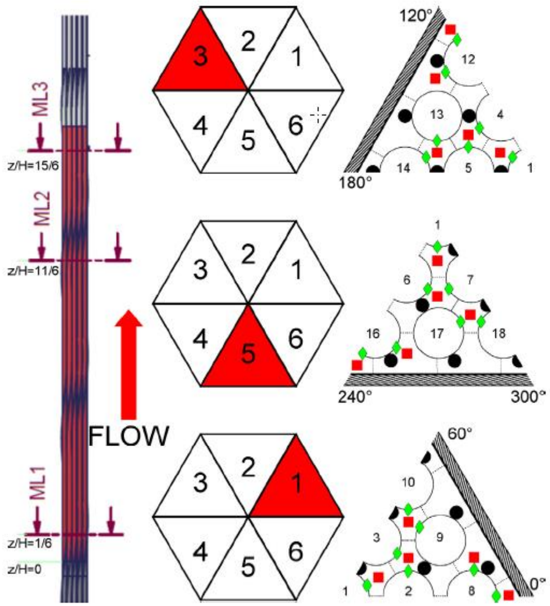
❑ Progress

- ✓ Develop sub-channel code for BLESS, and validate the code with KALLA experiment preliminary.
- ✓ Build the sub-channel analysis model, and complete the BLESS-D core thermal hydraulics simulation.
- ✓ Establish core T-H design limits of BLESS-D.

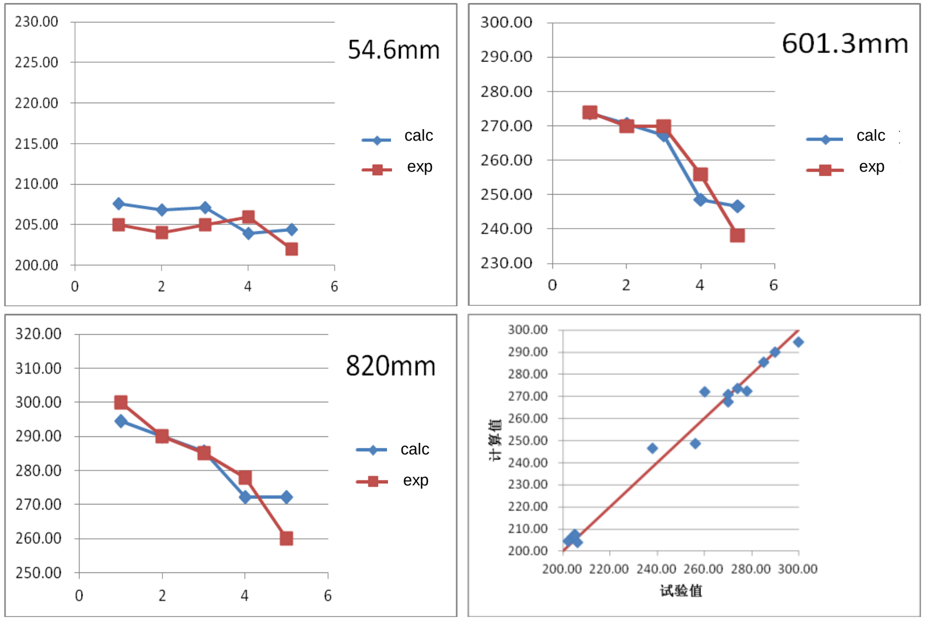
3. Thermal-hydraulics analysis

◻ Analysis results

1. Develop sub-channel code for BLESS. Validate the code with KALLA experiment.



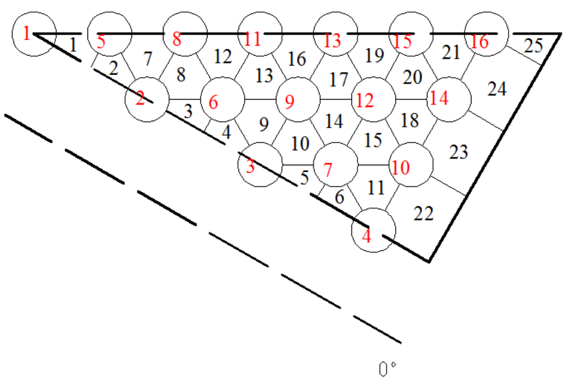
Results of KALLA experiment



Code validation

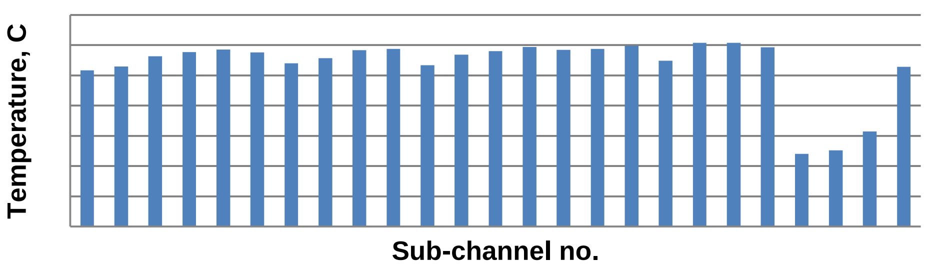
☐ Analysis results

2. Sub-channel analysis for BLESS-D core.

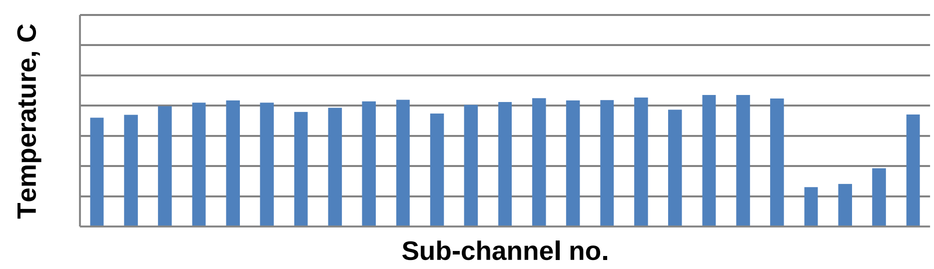


Inlet Temperature Ave., C	Outlet Temperature Max. , C	Cladding Temperature Max. , C	Fuel Temperature Max. , C	Coolant flow velocity Max., m/s
340	540.6	545.4	1055.9	1.2

Outlet temperature - hot channel



Outlet temperature - ave channel



3. Thermal-hydraulics analysis

□ Results

- According to the code validation preliminary, the sub-channel code could use for the conceptual analysis of BLESS-D.
- According to the results of sub-channel analysis of BLESS-D, the thermal hydraulic design parameter satisfied the design limits.

□ Future research

- Improvement in the sub-channel function and code validation further.
- Thermal hydraulic sensitivity analysis based on assembly design.
- Reactor thermal hydraulic analysis.

CONTENT

目录

4. Radiation shielding analysis

☐ Objective

- To evaluate the radiation damage of structural materials.
- To estimate the activation of coolant and structural materials.
- To calculate the neutron and gamma doses during reactor operation.
- Optimization of the shielding design.

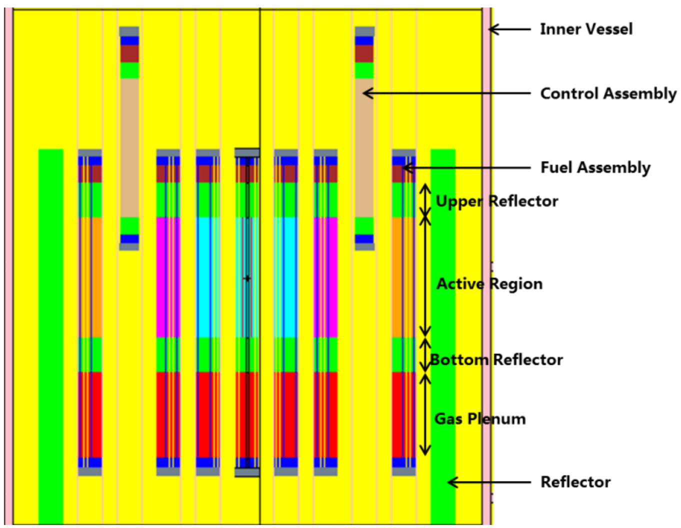
☐ Progress

- ✓ Reactor configuration was modelled by using Monte Carlo code.
- ✓ DPA values in some key components of the BLESS-D are computed.
- ✓ Isotope inventories of coolant at various irradiation time steps are evaluated.

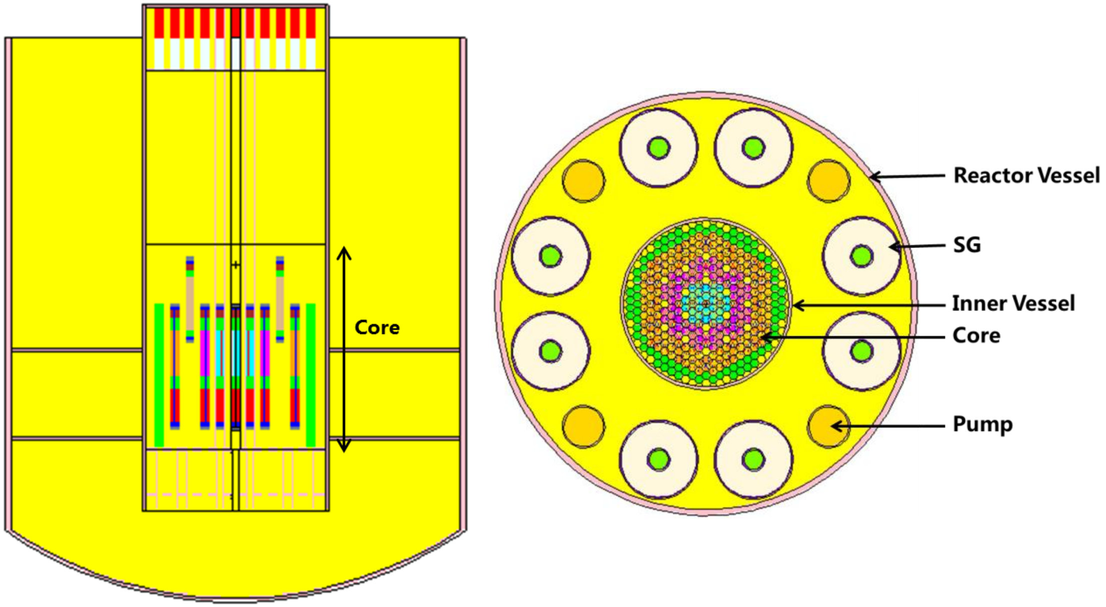
4. Radiation shielding analysis

◻ Analysis results

1. Reactor configuration of BLESS-D.



Model of active core

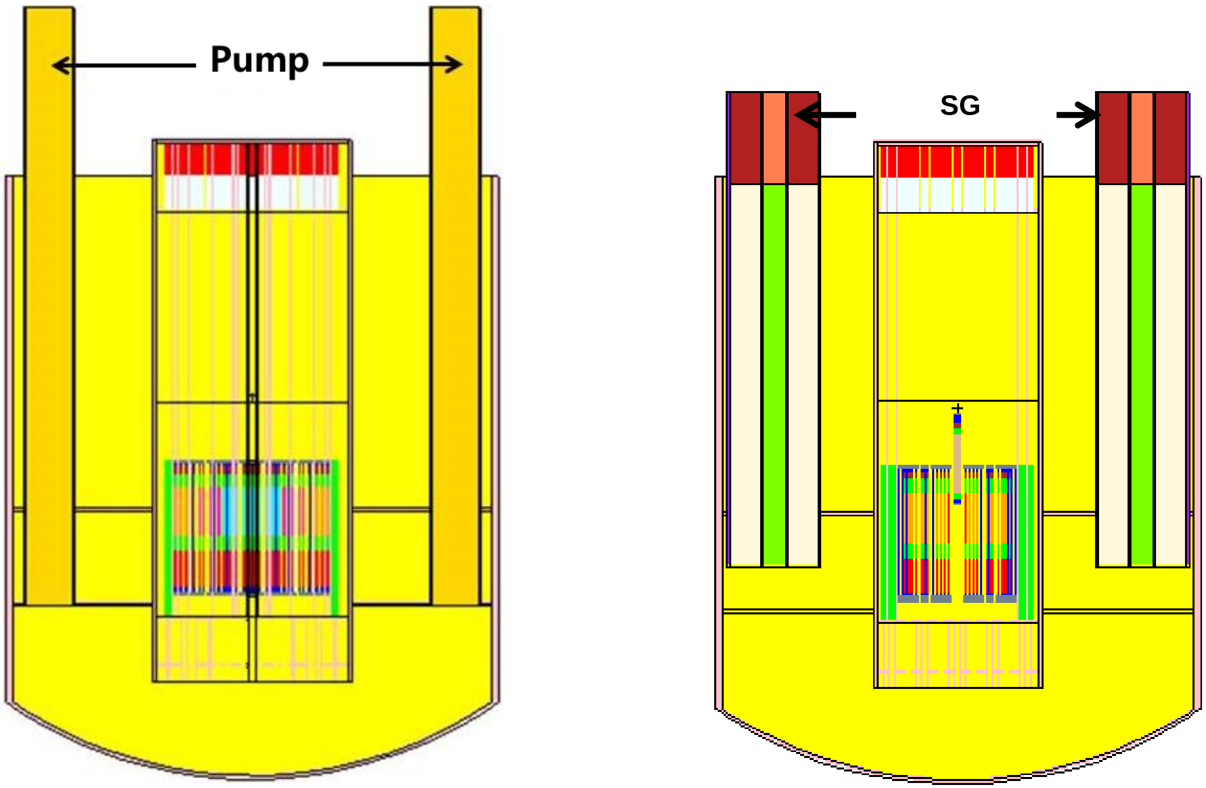


Model of BLESS-D reactor

4. Radiation shielding analysis

◻ Analysis results

1. Reactor configuration of BLESS-D



Vertical section of pumps and SGs models

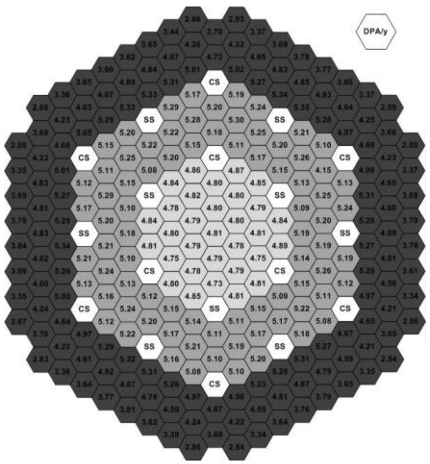
4. Radiation shielding analysis

☐ Analysis results

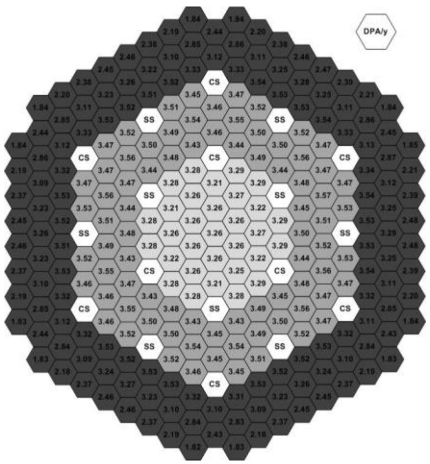
2. DPA values in some key components

- Fuel Cladding
  - ✓ The highest DPA value of the fuel pin cladding is **5.35 dpa/y.**

- Assembly Shroud
  - ✓ The highest DPA value of the assembly shroud is **3.57 dpa/y.**



DPA values in the fuel pin claddings of the central fuel assembly



DPA values in the assembly shroud

4. Radiation shielding analysis

☐ Analysis results

2. DPA values in some key components

- Inner Vessel

DPA values in the inner vessel

Segment No.	Distance from the bottom（cm）	DPA/s	DPA/y	Region
1	20	3.11E-11	9.82E-04	Gas plenum
2	40	9.04E-11	2.85E-03	
3	60	2.34E-10	7.37E-03	
4	80	7.14E-10	2.25E-02	
5	100	2.50E-09	7.90E-02	Bottom reflector
6	120	5.73E-09	1.81E-01	Active region
7	140	7.54E-09	2.38E-01	
8	160	6.72E-09	2.12E-01	
9	180	3.81E-09	1.20E-01	
10	200	1.21E-09	3.83E-02	Upper reflector
11	220	3.73E-10	1.18E-02	Plug
12	240	1.94E-10	6.12E-03	

✓ The segment with highest DPA value occurs at the active region of the core which has the value of **0.24 dpa/y.**

4. Radiation shielding analysis

◻ Analysis results

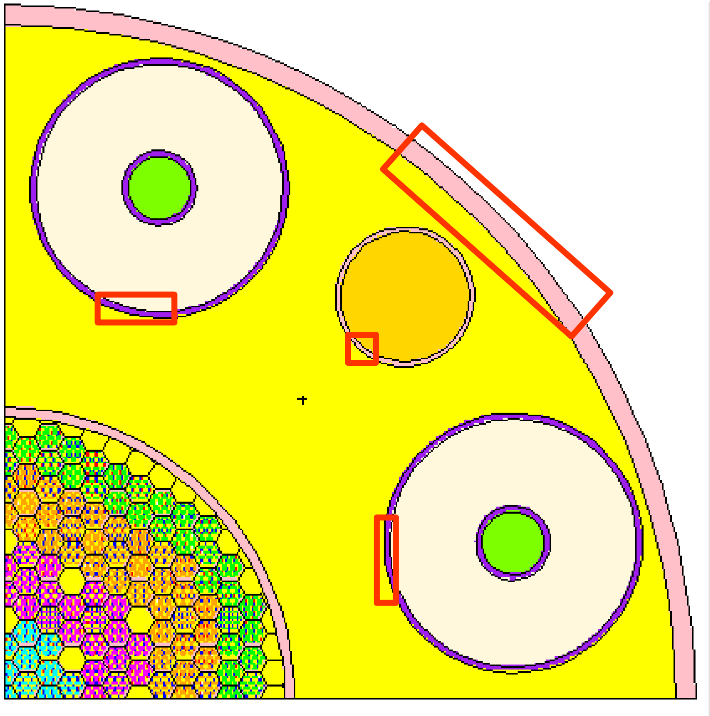
2. DPA values in some key components

- SG, Pump and Reactor Vessel

DPA values in main components

Main Components	Max. DPA/s	Max. DPA/y
Pump casing	2.69E-10	8.48E-03
#1 SG casing	2.30E-09	7.25E-02
#2 SG casing	2.38E-09	7.51E-02
Reactor Vessel	6.16E-13	1.94E-05

✓ The segment with highest DPA value in these main components occur at the active region of the core.



4. Radiation shielding analysis

☐ Analysis results

3. Coolant activation

- Activation of coolant outside the inner vessel

Isotope inventories of coolant at various irradiation time steps (per kg of coolant)

1 year Irradiation		10 years Irradiation		20 years Irradiation	
Nuclide	**Concentrations (g)**	**Nuclide**	**Concentrations (g)**	**Nuclide**	**Concentrations (g)**
Bi209	5.50E+02	**Bi209**	5.50E+02	**Bi209**	5.50E+02
Pb208	2.37E+02	**Pb208**	2.37E+02	**Pb208**	2.37E+02
Pb206	1.08E+02	**Pb206**	1.08E+02	**Pb206**	1.08E+02
Pb207	9.93E+01	**Pb207**	9.93E+01	**Pb207**	9.93E+01
Pb204	6.20E+00	**Pb204**	6.20E+00	**Pb204**	6.20E+00
Cu63	1.81E-02	**Cu63**	1.81E-02	**Cu63**	1.81E-02
In115	1.34E-02	**Ag107**	1.29E-02	**He4**	1.81E-02
Ag107	1.31E-02	**In115**	1.04E-02	**Ag107**	1.27E-02
Ag109	1.22E-02	**Ag109**	1.04E-02	**Bi210m**	8.92E-03
Cu65	8.35E-03	**He4**	8.79E-03	**Ag109**	8.69E-03
Sn120	1.94E-03	**Cu65**	8.35E-03	**Cu65**	8.35E-03
Sn118	1.42E-03	**Bi210m**	4.46E-03	**In115**	7.89E-03
Ni58	1.34E-03	**Sn116**	2.76E-03	**Sn116**	4.20E-03
Fe56	1.29E-03	**Cd110**	2.29E-03	**Cd110**	3.97E-03
Sn116	1.05E-03	**Sn120**	1.94E-03	**Pb205**	3.80E-03
Cd114	6.74E-04	**Pb205**	1.90E-03	**Sn120**	1.94E-03
In113	6.04E-04	**Sn118**	1.42E-03	**Sn118**	1.42E-03
Ni60	5.35E-04	**Ni58**	1.34E-03	**Ni58**	1.34E-03
Cd112	5.28E-04	**Fe56**	1.29E-03	**Fe56**	1.29E-03
Sn119	5.08E-04	**Cd114**	8.38E-04	**Cd114**	8.90E-04
Rest	4.43E-03	**Rest**	5.16E-03	**Rest**	5.37E-03

4. Radiation shielding analysis

☐ Analysis results

3. Coolant activation

- Activation of coolant inside fuel assembly with highest neutron flux

Isotope inventories of coolant at various irradiation time steps (per kg of coolant)

1 year Irradiation		10 years Irradiation		20 years Irradiation	
Nuclide	**Concentrations (g)**	**Nuclide**	**Concentrations (g)**	**Nuclide**	**Concentrations (g)**
Bi209	5.50E+02	**Bi209**	5.50E+02	**Bi209**	5.49E+02
Pb208	2.37E+02	**Pb208**	2.37E+02	**Pb208**	2.37E+02
Pb206	1.08E+02	**Pb206**	1.08E+02	**Pb206**	1.08E+02
Pb207	9.93E+01	**Pb207**	9.93E+01	**Pb207**	9.94E+01
Pb204	6.20E+00	**Pb204**	6.16E+00	**Pb204**	6.11E+00
Cu63	1.81E-02	**He4**	2.14E-01	**He4**	4.40E-01
Ag107	1.30E-02	**Bi210m**	1.07E-01	**Bi210m**	2.15E-01
In115	1.27E-02	**Pb205**	4.74E-02	**Pb205**	9.42E-02
He4	1.20E-02	**Cu63**	1.80E-02	**Bi208**	2.16E-02
Ag109	1.15E-02	**Po210**	1.24E-02	**Cu63**	1.78E-02
Bi210m	1.08E-02	**Ag107**	1.14E-02	**Po210**	1.24E-02
Po210	1.02E-02	**Bi208**	1.09E-02	**Ag107**	9.89E-03
Cu65	8.35E-03	**Cu65**	8.31E-03	**Cd110**	9.39E-03
Pb205	4.77E-03	**Cd110**	6.75E-03	**Cu65**	8.28E-03
Sn120	1.94E-03	**In115**	6.07E-03	**Sn116**	7.02E-03
Sn116	1.46E-03	**Ag109**	5.70E-03	**Cd108**	2.95E-03
Sn118	1.42E-03	**Sn116**	5.20E-03	**Ag109**	2.68E-03
Ni58	1.34E-03	**Sn120**	1.95E-03	**In115**	2.68E-03
Fe56	1.29E-03	**Cd108**	1.64E-03	**Sn120**	1.95E-03
Cd110	1.18E-03	**Sn118**	1.42E-03	**Sn118**	1.43E-03
Rest	6.99E-03	**Rest**	8.98E-03	**Rest**	9.94E-03

4. Radiation shielding analysis

☐ Results

- The preliminary damage evaluations of some key components in BLESS-D show that the damages are under the design limits for 18 years reactor life.
- The isotope inventories of coolant at various irradiation time steps are calculated preliminarily to estimate the activation of coolant and structural materials

☐ Future Research

- Optimization of BLESS-D shielding design according to neutron and gamma dose distribution.
- Source term assessment of BLESS-D.

CONTENT

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5. Conclusions

1

- Neutronics analysis was performed by a Monte-Carlo code RMC. Parameters of core design including reactor size, fuel assembly scale, fuel pitch, fuel enrichment, reactivity coefficient, control-assembly arrangement and burn-up performance was calculated and analyzed.

2

- Property and some models in the sub-channel code are discussed and adapted for LBE-cooled fast reactor. The analysis method could be used in the preliminary evaluation and analysis for LBE-cooled reactor.

3

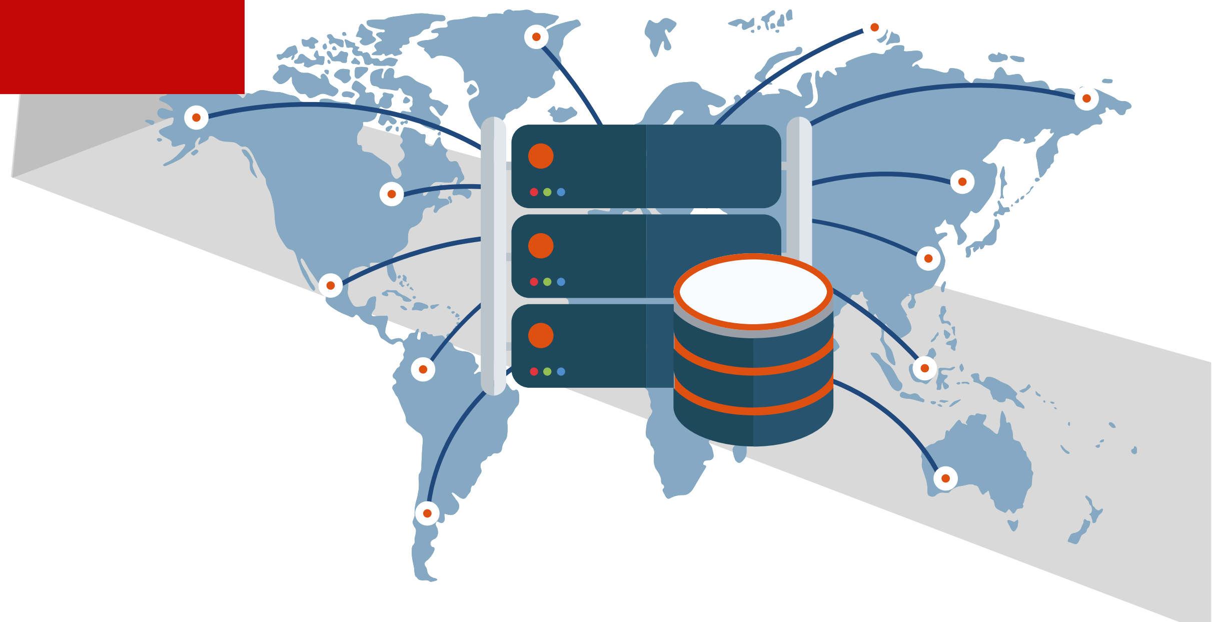
- The main components considered in this study include the fuel cladding, the inner vessel, the pump, the Steam Generators (SGs) and the reactor tank. The DPA rates for main components are calculated and presented using the Monte Carlo transport method.

4

- According to the preliminary neutronics, Thermal-Hydraulics and radiation shielding analysis, a 300 MWth LBE-cooled fast reactor core conceptual design has been developed and proposed, further design optimization and more analysis will be executed in the future research.

国家电投集团中央研究院
SPIC CENTRAL RESEARCH INSTITUTE

谢谢 Xiexie!
Спасибо!



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FIFTH CONFERENCE ON HEAVY LIQUID METAL COOLANTS
IN NUCLEAR TECHNOLOGIES

国家电力投资集团有限公司
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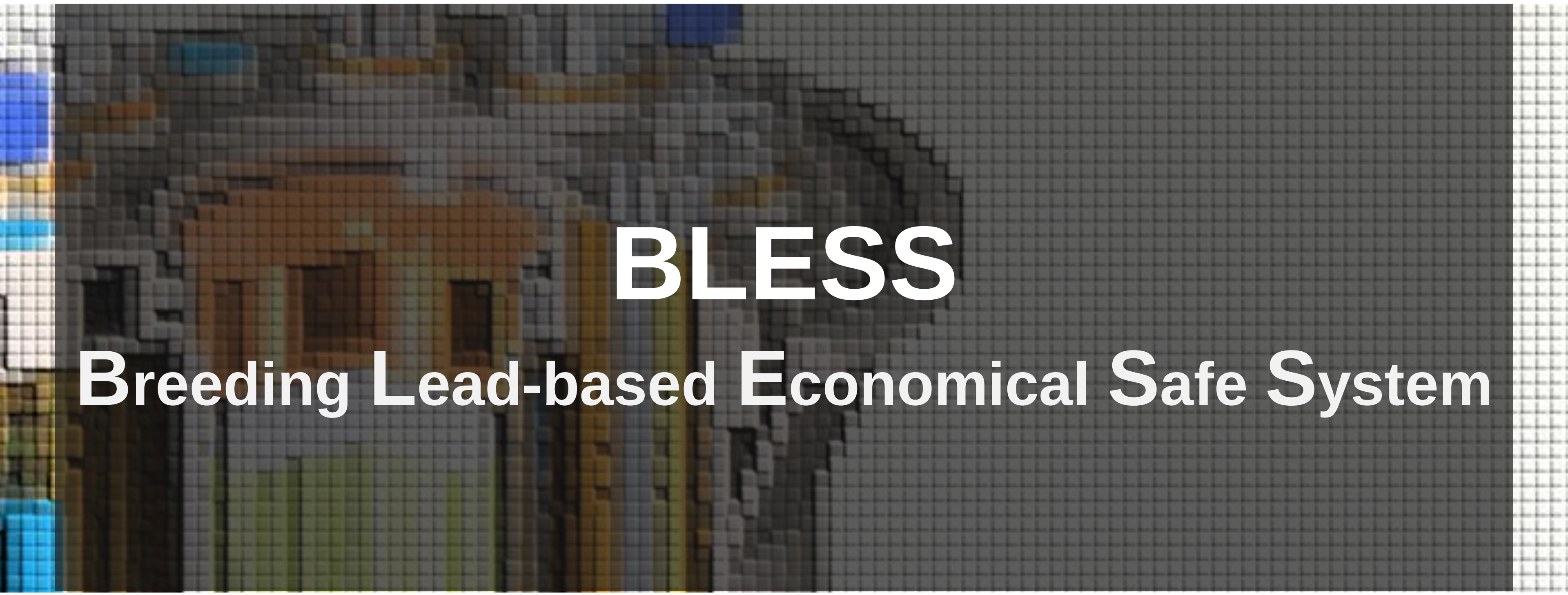
LBE-Cooled Fast Reactor BLESS: Conceptual Core Design

Ziguan Wang, Linsen Li, Yeoh Eing Yee, Yaodong Chen

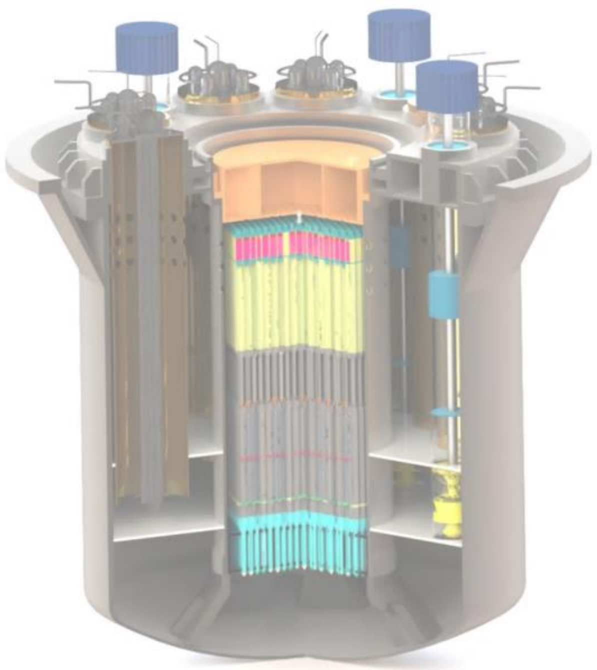
SPIC Research Institute, Beijing, CHINA

2018.10.09


BLESS
Breeding Lead-based Economical Safe System

Main R&D Progress - Key Parameters

Thermal Power	300 MWt
Electrical Power	120 MWe
Core height	70 cm
Core Diameter	250 cm
Fuel	UO_2
Fuel Assembly	310
Fuel Rod in a Assembly	127
Coolant	Lead-bismuth eutectic

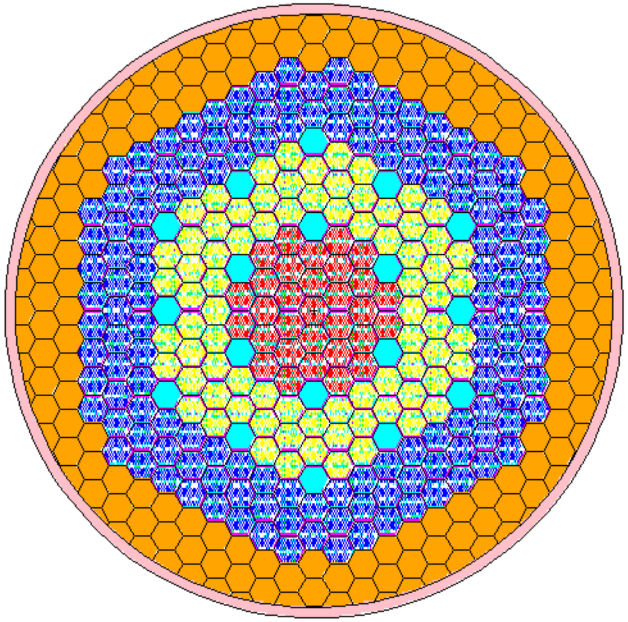


BLESS-D Conceptual Design

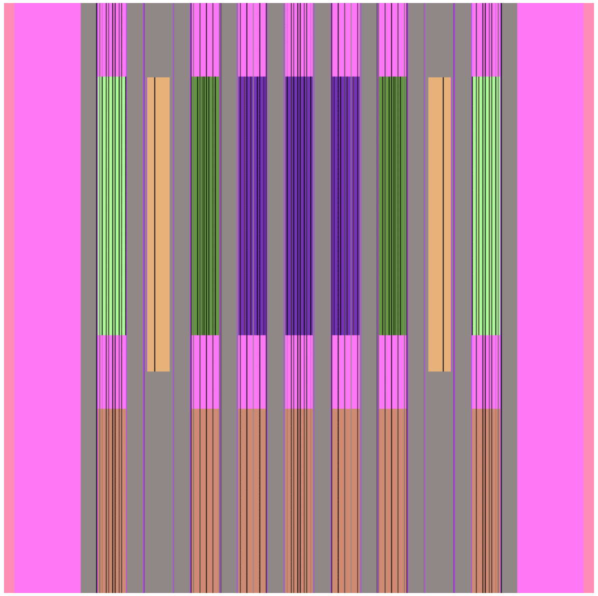
- **Neutronics Design**
- Reactor Shielding
- T-H Analysis
- SG Design & Analysis
- Primary Pump Selection
- Primary Cycle Design
- Secondary Cycle Design
- Equipment and Structure Analysis
- Safety Analysis

Main R&D Progress - Tools

A Monte Carlo code RMC(Reactor Monte Carlo) is used in the neutronics design.



RMC Model: Radical



RMC Model: Axial

Main R&D Progress - Fuel

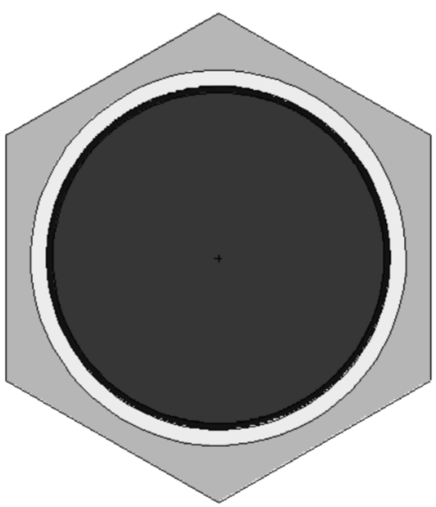
UO_2

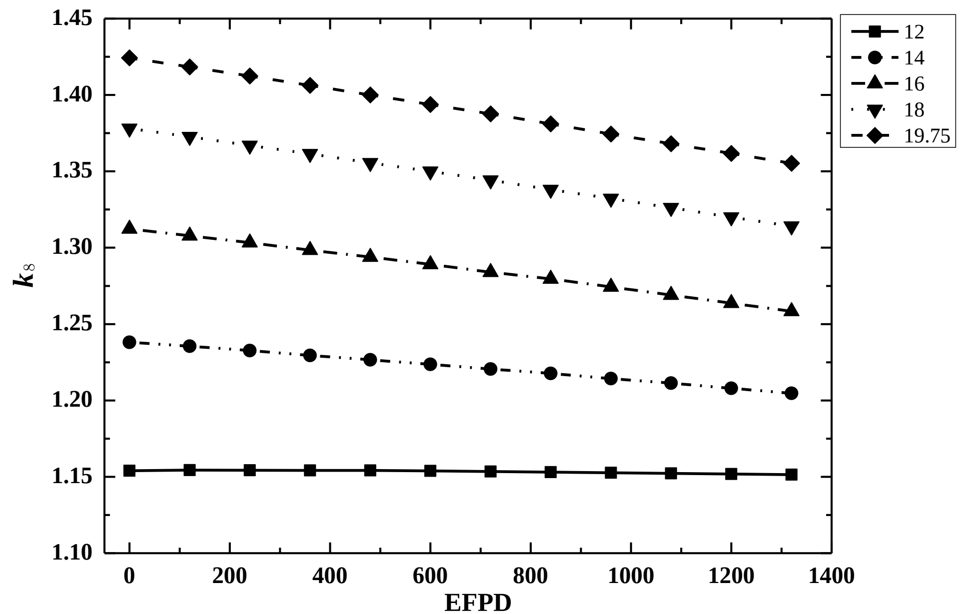
MOX

UO_2

Main R&D Progress - Key Parameters

- k_{inf} of a total reflected boundary cell was calculated to investigate the optimistic enrichment of UO_2.
- In the optimistic solution, a larger excessive reactivity, a larger breeding ratio and a more flattened power distribution is preferred.
- The ^{235}U enrichment 14%, 16% 19.75% was chosen in a 3 regions core layout.



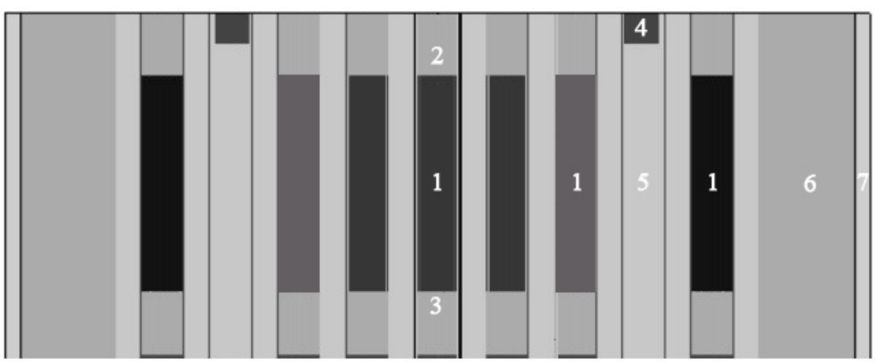


k_{inf} against burn-up with various enrichment

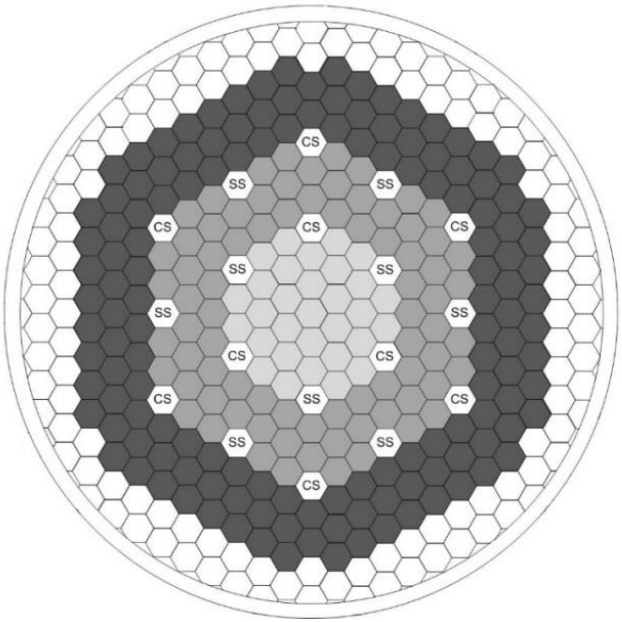
Main R&D Progress - Key Parameters

Core layout

- 3 fuel regions, with uranium enrichment 14%, 16%,19.75%.
- With 200 mm thick reflector assemblies and 50 mm thick reactor vessel.
- 2 Control rod systems, one for reactivity control (CS) and one for emergency shutdown (SS)



BLESS-D axial core layout

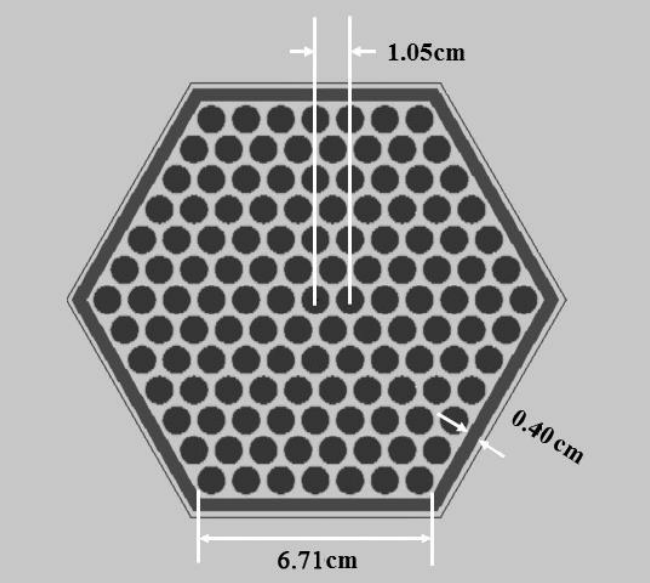


BLESS-D radical core layout

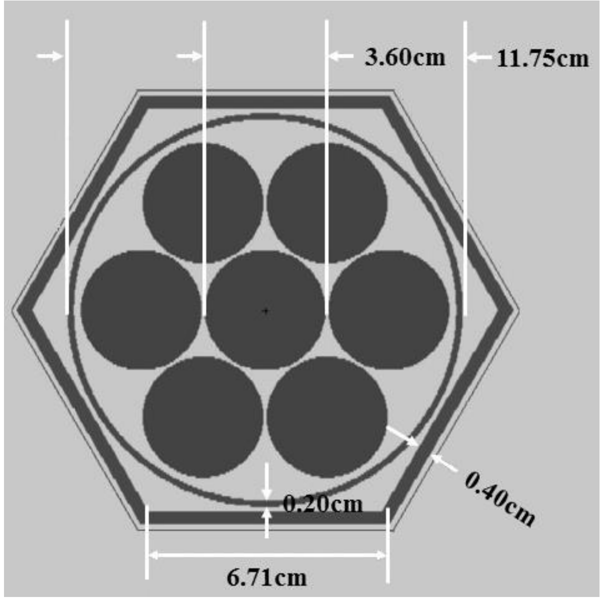
Main R&D Progress - Key Parameters

Core Layout – Fuel Pin and Assembly

- Hexagonal fuel assembly wrapping 127 fuel pin
- Hexagonal control rod assembly with 7 absorber (B_4C) rod, same layout for both control and shutdown system



Single fuel assembly layout

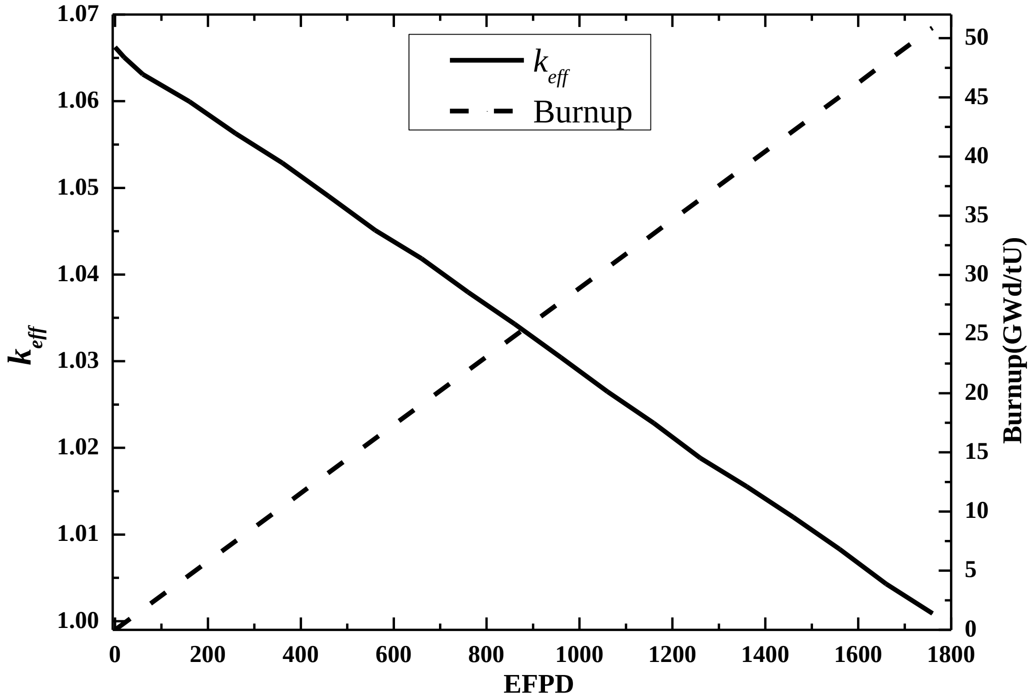


Control rod assembly layout

Main R&D Progress – Preliminary results

Burn-up calculation

- BOL k_{eff} **=1.06711(0.00013)** ;
- First cycle lifetime 1770 EFPD or 5 EFPY
- Average burn-up **50.8 GWd/MtU**



k_{eff} and burn-up against time

Main R&D Progress - Preliminary results

Reactivity coefficient

- All reactivity coefficients are negative during the whole cycle life-time.
- The coolant temperature coefficient is getting worse as the rising of burn-up depth.

	BOL	MOL	EOL
Fuel temp. coefficient/(pcm/K)	-1.12 (0.04)	-1.13 (0.03)	-1.35 (0.05)
Coolant temp. coefficient/(pcm/K)	-0.44 (0.02)	-0.31 (0.02)	-0.12 (0.01)
Void coefficient/(pcm)	-3935 (21)	-3765 (18)	-3450 (16)

Main R&D Progress - Preliminary results

Control rod worth

Reactivity control system (CS) works during normal operation status including reactor start, full power operation, etc. Emergency shutdown system (SS) normally extracted from the core and is inserted in an emergency shutdown status.

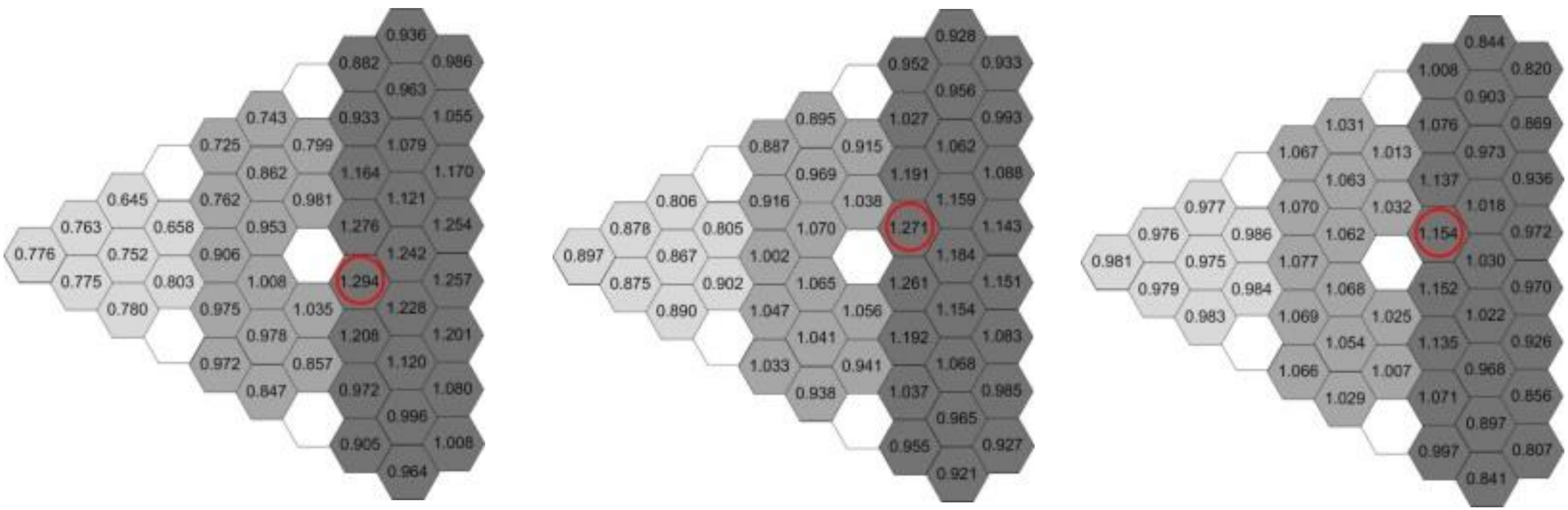
CS+SS Reactivity worth and requirement

Parameters	Value/pcm
CS+SS total reactivity worth	13625
Total requirement	13024
Excessive reactivity at BOL	*6289*
Shutdown margin	*5263*
Worth of the most efficient CS Assembly	*754*
Worth of the most efficient SS Assembly	*718*

Main R&D Progress - Preliminary results

Power distribution

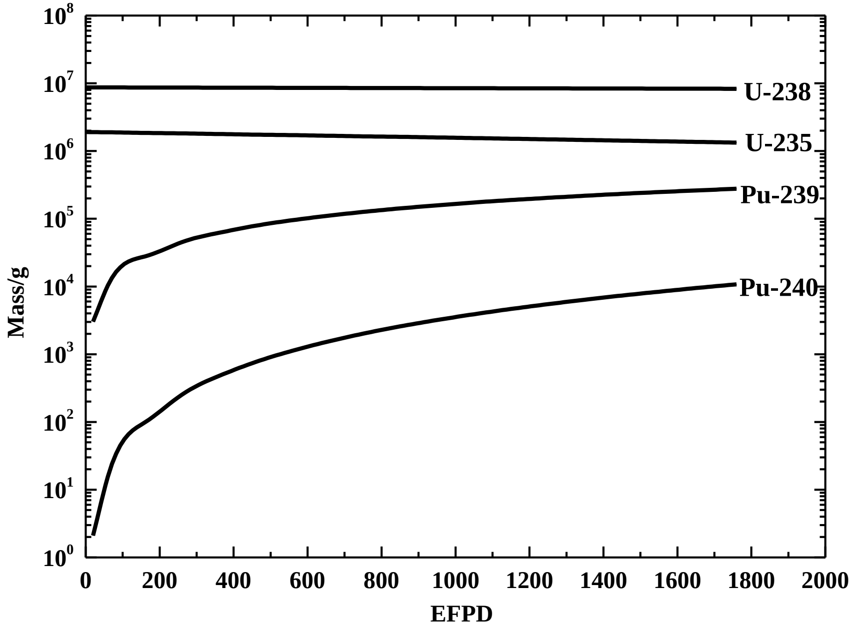
- Power peak factor is 1.294, 1.271, 1.154 at BOL, MOL, and EOL.
- Core-level power distribution is well flattened and become better as the rising of burn-up depth



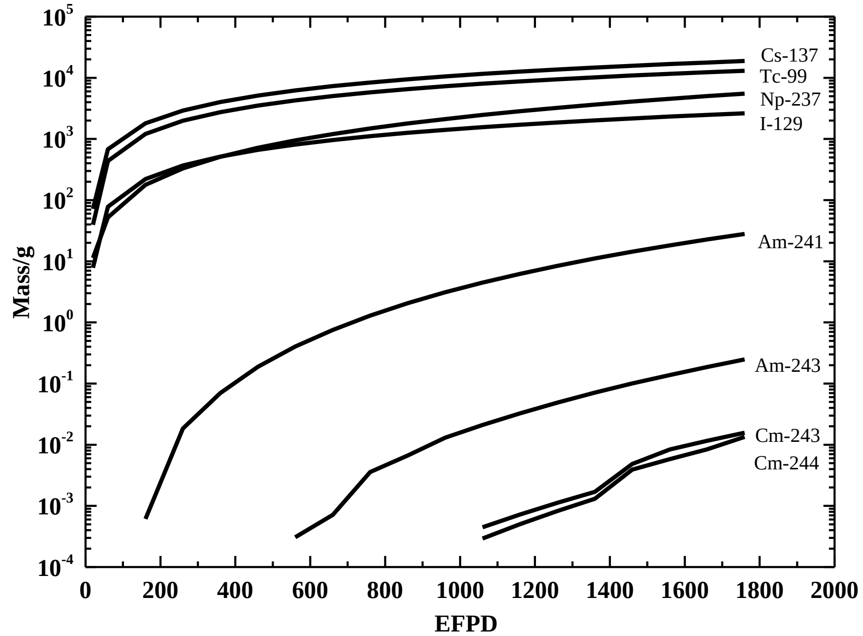
Power distribution at BOL, MOL, EOL

Main R&D Progress - Preliminary results

Accumulated production of minor actinides and long-lived fission products against burn-up.



Mass of U-235、U-238、Pu-239、Pu-240 against Burn-up



Mass of LLFP against Burn-up

Main R&D Progress - Preliminary results

Nuclide	Half-life/a	BLESS-D/(g/tHM)	PWR/(g/tHM)[18]	Ratio of BLESS over PWR
^{239}Pu	2.41×10^{4}	18699.9	5470.21	3.42
^{237}Np	2.14×10^{6}	272.87	436.73	0.62
^{240}Pu	6550.32	463.52	2231.29	0.21
^{241}Am	432.58	0.64	296.13	0.00216
^{243}Am	7370.24	2.53×10^{-3}	83.80	0.0000302
^{243}Cm	28.50	0.00	0.43	0.00
^{244}Cm	18.12	0.00	21.91	0.00
^{245}Cm	8500.00	0.00	0.80	0.00
^{99}Tc	2.07×10^{5}	855.13	841.12	1.02
^{129}I	1.57×10^{7}	167.38	229.31	0.73

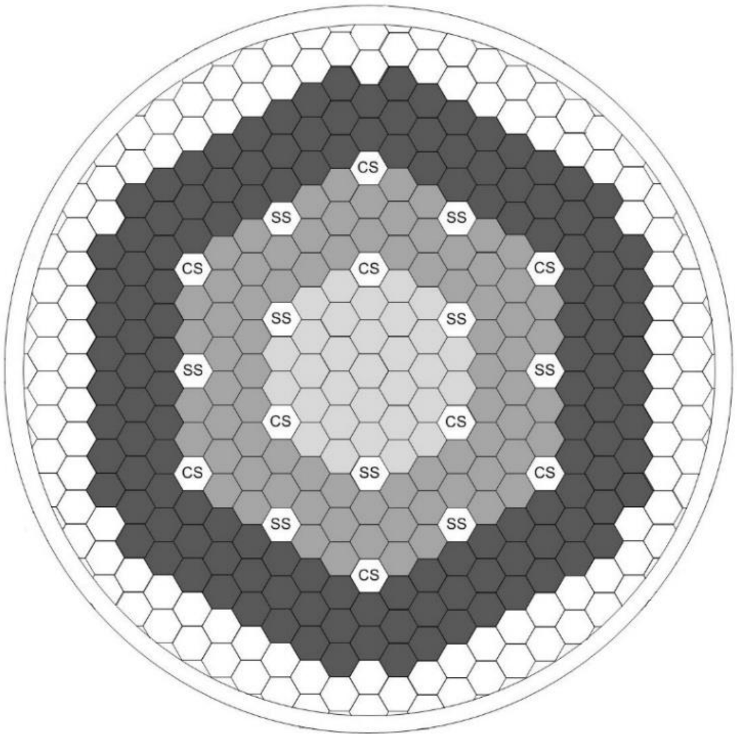
MOX

Main R&D Progress - Preliminary results

- Same core layout
- Implement of MOX (UO_2+PuO_2) fuel
- Minor Actinides (MA) were added to analyze transmutation capability
- PuO_2 enrichement 16%、18%、22%

U, Pu isotope fraction in MOX fuel

Isotope	Fraction/%	Isotope	Fraction/%
^{238}Pu	2.332	^{234}U	0.003
^{239}Pu	56.873	^{235}U	0.404
^{240}Pu	26.997	^{236}U	0.010
^{241}Pu	6.104	^{238}U	99.583
^{242}Pu	7.693		

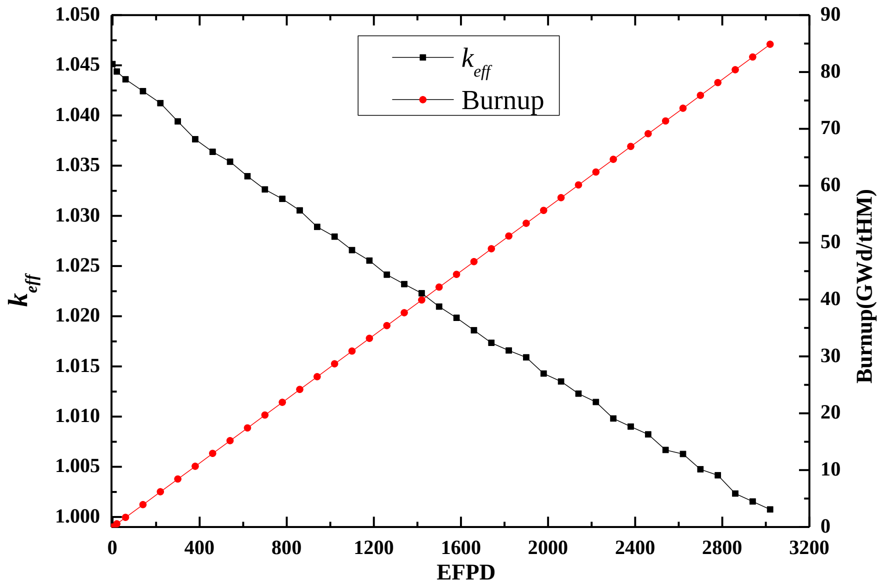


Main R&D Progress - Preliminary results

- Addition of MA will reduce the β_{eff} and significantly influence the safety margin.
- As a consequence, the fraction of overall MA in the fuel should be **5% at most.**
- MA composition is listed below.

isotope	Fraction/%
Np-237	56.20
Am-241	26.40
Am-243	12.00
Cm-243	0.03
Cm-244	5.11
Cm-245	0.26

Main R&D Progress - Preliminary results



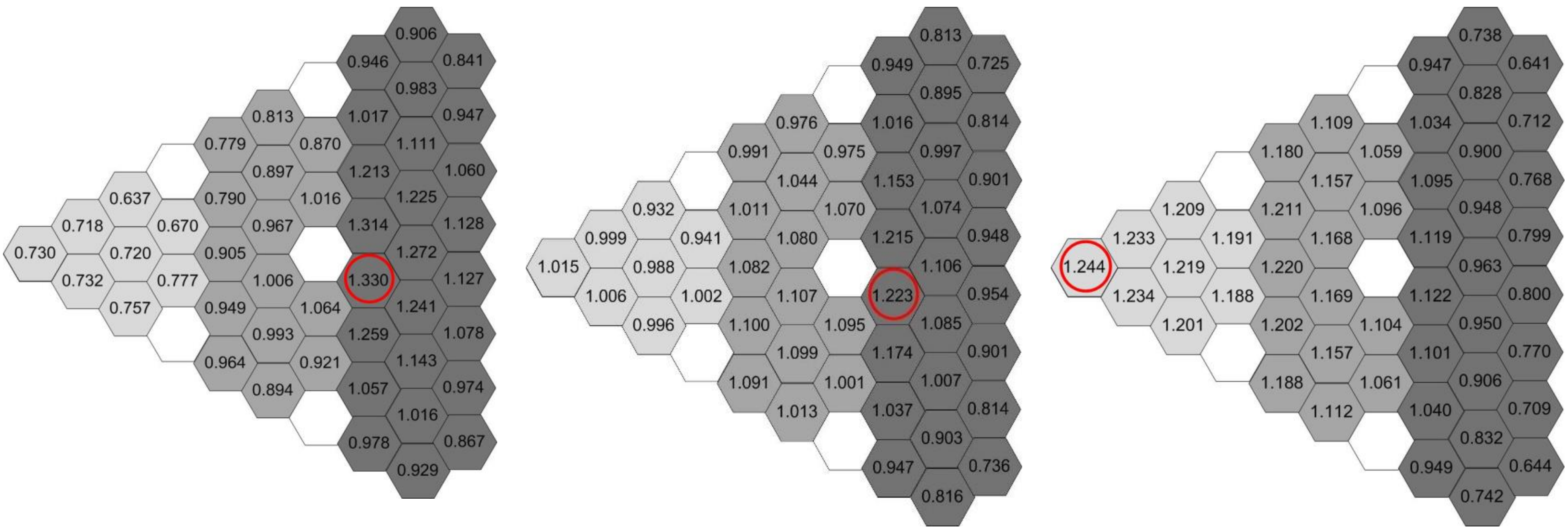
- BOL k_{eff} =1.04512(0.00013)；
- First cycle lifetime 3000EFPD，or 8.2EFPY
- Average burn-up 84.8 GWd/tHM；

Main R&D Progress - Preliminary results

- All reactivity coefficients are negative during the whole cycle life-time.
- The coolant temperature coefficient is getting worse as the rising of burn-up depth.
- Due to the addition of MA, the reactivity coefficient safety margin is getting worse compare to the UO_2 case.

	BOL	MOL	EOL
Fuel temp. coefficient/(pcm/K)	-0.876 (0.03)	-0.874 (0.04)	-0.911(0.03)
Coolant temp. coefficient/(pcm/K)	-0.204 (0.02)	-0.119 (0.01)	-0.07 (0.01)
Void coefficient/(pcm)	-1559(12)	-1302(14)	-657(8)

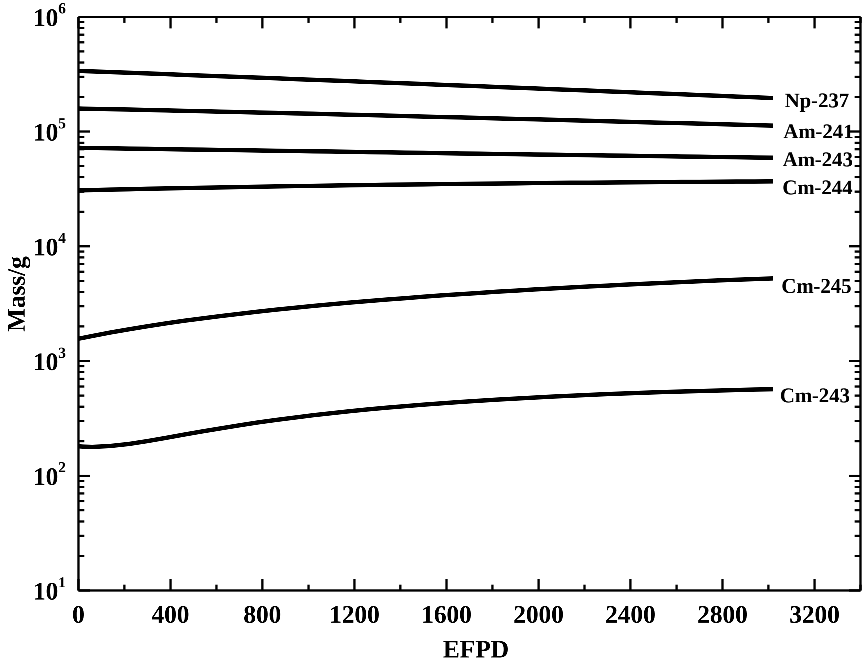
Main R&D Progress - Preliminary results



Power distribution at BOL, MOL, EOL

- Power peak factor is 1.330, 1.223, 1.244 at BOL, MOL, and EOL.
- Core-level power distribution is well flattened and become better as the rising of burn-up depth

Main R&D Progress - Preliminary results



Mass of MA (Np-237, Am-241 , Am-243 , Cm-243 , Cm-244 , Cm-245) against burn-up

isotope	BOL Mass/g	EOL Mass/g
^{237}Np	3.38×10^5	1.96×10^5
^{241}Am	1.59×10^5	1.13×10^5
^{243}Am	7.22×10^4	5.92×10^4
^{243}Cm	180.50	568.80
^{244}Cm	3.07×10^4	3.68×10^4
^{245}Cm	1.56×10^3	5.26×10^3
TOTAL	6.02×10^5	4.11×10^5

Conclusion

Results

- ✓ A preliminary core design with UO_2 and MOX fuel has been accomplished. Key parameters calculation completed: control rod worth, power distribution, burn-up calculation, reactivity coefficient.
- ✓ Preliminary analysis on Long-life waste disposal and breeding and transmutation capability.

Plans

- ☐ Revise of reactivity control system.
- ☐ Revise of core arrangement in accordance with T-H and shielding parameters.
- ☐ Verification of key parameters with deterministic method code.

СПАСИБО

День веселья, верь, настанет.

A day of fun, believe, will come.

—— Алекса́ндр Серге́евич Пу́шкин

Aleksandr Sergeyevich Pushkin

Numerical modelling of Lead-Cooled Fast Reactor using the MCB code

9 October 2018, Heavy Liquid Metal Coolants in Nuclear Technologies, Obninsk, Russian Federation

AKADEMIA GÓRNICZO-HUTNICZA IM. STANISŁAWA STASZICA W KRAKOWIE
AGH UNIVERSITY OF SCIENCE AND TECHNOLOGY

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Mikołaj Oettingen,
Przemysław Stanisz

Scope of the study

Information on how nuclides transmute in the fuel cycle can:
1) improve the modeling of the fuel cycle
2) help in the verification of nuclear data

A neutron physics study for a lead-cooled reactor	→	Application part ELFR core modeling (European Lead Fast Reactor).
The evolution of nuclear fuel using a trajectory for folding time intervals	→	A new calculation method for transmutation representation

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Software

The Monte Carlo Continuous Energy Burn-up Code

MCB = MCNP + TTA

MCNP Monte Carlo N-Particle Transport Code
The code provides results such reaction rates and heating per nuclide for each burnable zone. Results are obtained from the Monte Carlo simulation in order to provide neutron distribution and spectrum.

TTA Transmutation Trajectory Analysis
The code provides evolution of the nuclides concentration through numerical solution based on the oryginal Bateman solution.

MCB has features:
- an integrated burn-up calculation code (calculations are integrated in one code)
- deals with the complexity of the burn-up process (i.e predictor-corrector method)
- deals withthe complexity of the fuel cycle process (i.e automatization shuffling and reloading of fuel)
- calculations include continuous energy representation of cross-section, spatial effects of full core reactor model and
- nuclide production in all possible reaction or decay channels.

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Hardware - Prometheus

2.4 Pflops

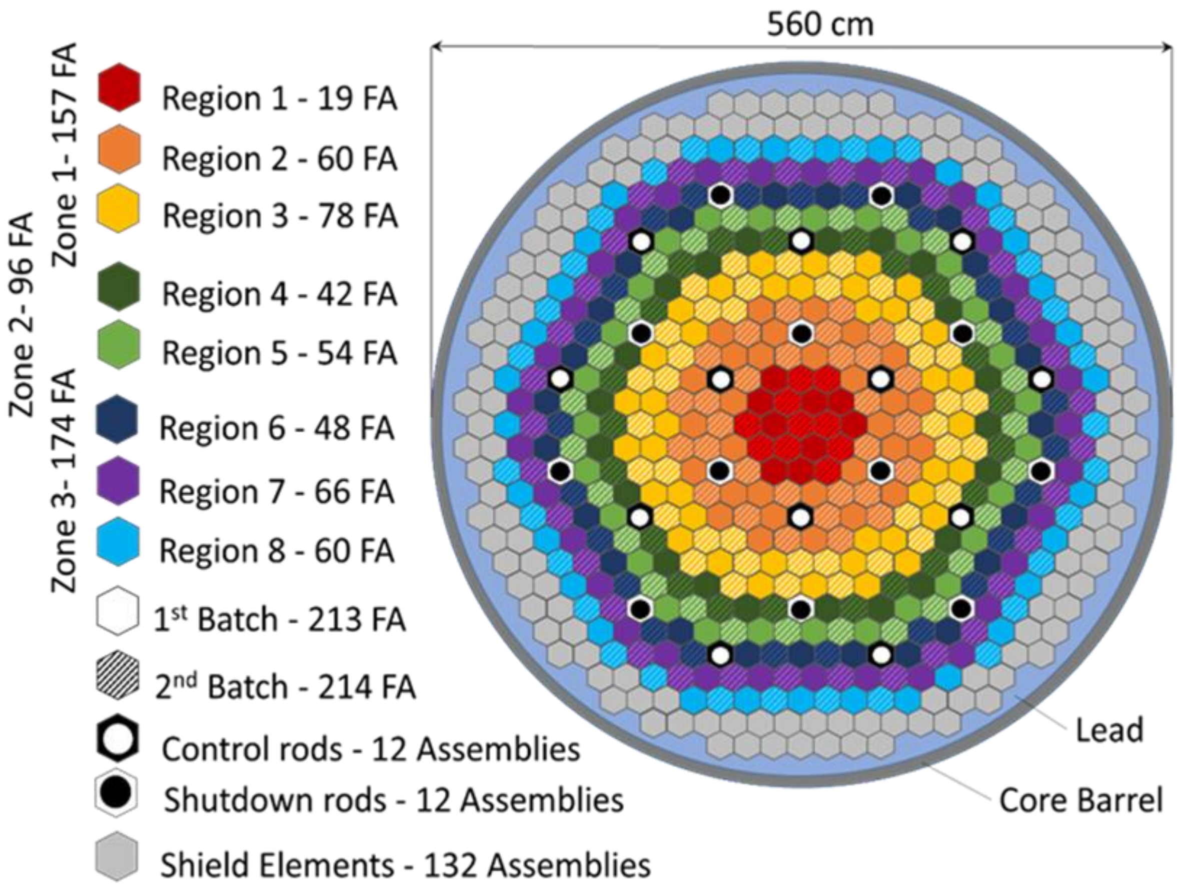
56000 cores

279 TB RAM

10 PB storage

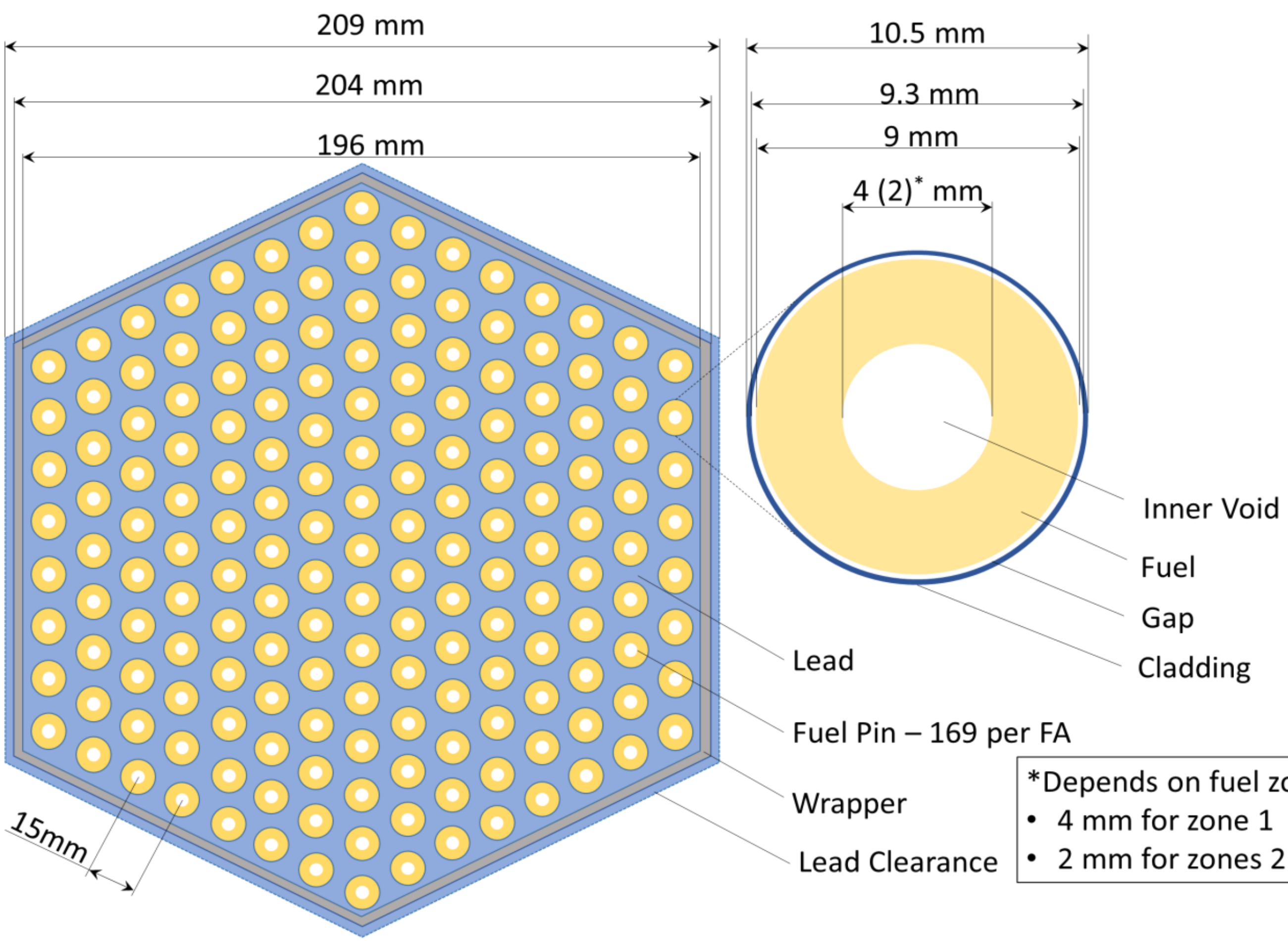
TOP500® SUPERCOMPUTER SITES

Numerical Model – European Lead Fast Reactor



Power [MW_{th}/MW_e]	1500/600
Primary coolant	Lead
Inlet temperature [°C]	400
Outlet temperature [°C]	480
Coolant velocity [m/s]	1.53
Fuel	MOX+MA
Pu enrichment [% wt.]	~18
Fuel irradiation time [days]	2x900
Cooling before recycling [years]	7.5
Barrel diameter [cm]	560
Barrel thickness [cm]	5
FA geometry	Hexagonal
Active height [cm]	140
Number of FA	427
Number of FA per zone 1/2/3	157/96/174
Number of pins per FA	169
FA pitch [mm]	209

Numerical Model – European Lead Fast Reactor



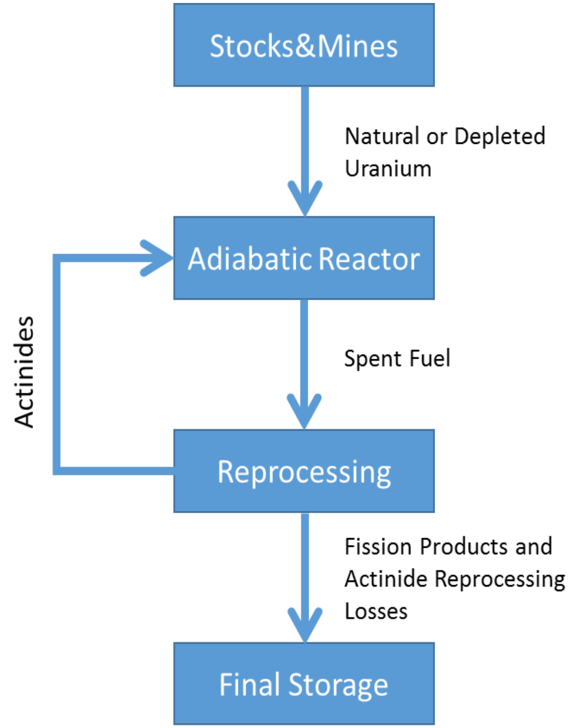
*Depends on fuel zone:
- 4 mm for zone 1
- 2 mm for zones 2 and 3

Fuel pin pitch [mm]	15
Pellet outer diameter [mm]	9
Pellet inner diameter [mm]	4(2)
Fuel pin clad thickness[mm]	0.6
Fuel pin gap thickness [mm]	0.15
Number of CR	12
Number of SR	12
CR(SR) pellet diameter [mm]	14
Number of shield assemblies	132
SA rod diameter [mm]	14
CR(SR, SA) clad diameter [mm]	0.7

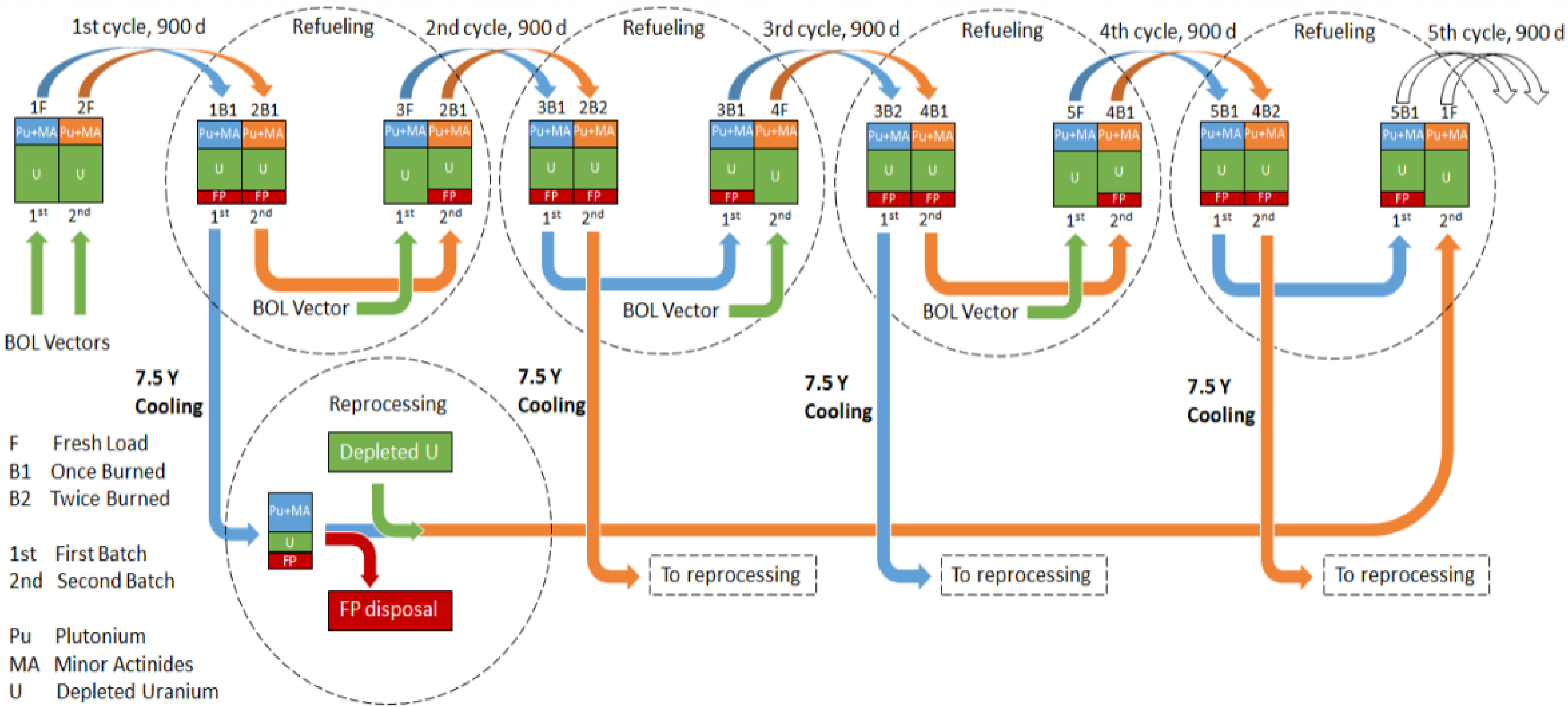
Numerical Model – Equilibrium fuel composition

Isotope	BOL [g]	EOL [g]	Isotope	BOL [g]	EOL [g]	Isotope	BOL [g]	EOL [g]
U	2.14E7	2.00E7/ 2.15E7*	^{240}Pu	1.30E6	1.69E6	^{242}Cm	0	3.77E1
^{234}U	6.42E2	5.71E4	^{241}Pu	2.94E5	1.18E5	^{243}Cm	2.39E2	5.35E2
^{235}U	8.64E4	2.23E4	^{242}Pu	3.71E5	2.07E5	^{244}Cm	1.09E4	3.06E4
^{236}U	2.14E3	3.57E4	^{244}Pu	0	3.19E0	^{245}Cm	4.14E3	1.06E4
^{238}U	2.13E7	1.99E7/ 2.14E7*	Am	3.29E5	2.93E5	^{246}Cm	3.23E2	5.63E3
^{237}Np	1.36E4	2.65E4	^{241}Am	2.70E5	2.08E5	^{247}Cm	6.05E0	9.73E2
Pu	4.82E6	4.80E6	^{242m}Am	9.11E2	1.55E4	^{248}Cm	4.93E-1	3.49E2
^{238}Pu	1.13E5	1.16E5	^{243}Am	5.79E4	6.96E4	HM	2.66E7	2.52E7/ 2.67E7*
^{239}Pu	2.74E6	2.67E6	Cm	1.53E4	4.87E4			

BOL: Beginning of Life (fresh fuel); EOL: End of Life (after 124.2 years); HM: Heavy Metal; *before and after admixing of depleted uranium.



The modelling of adiabatic fuel cycle

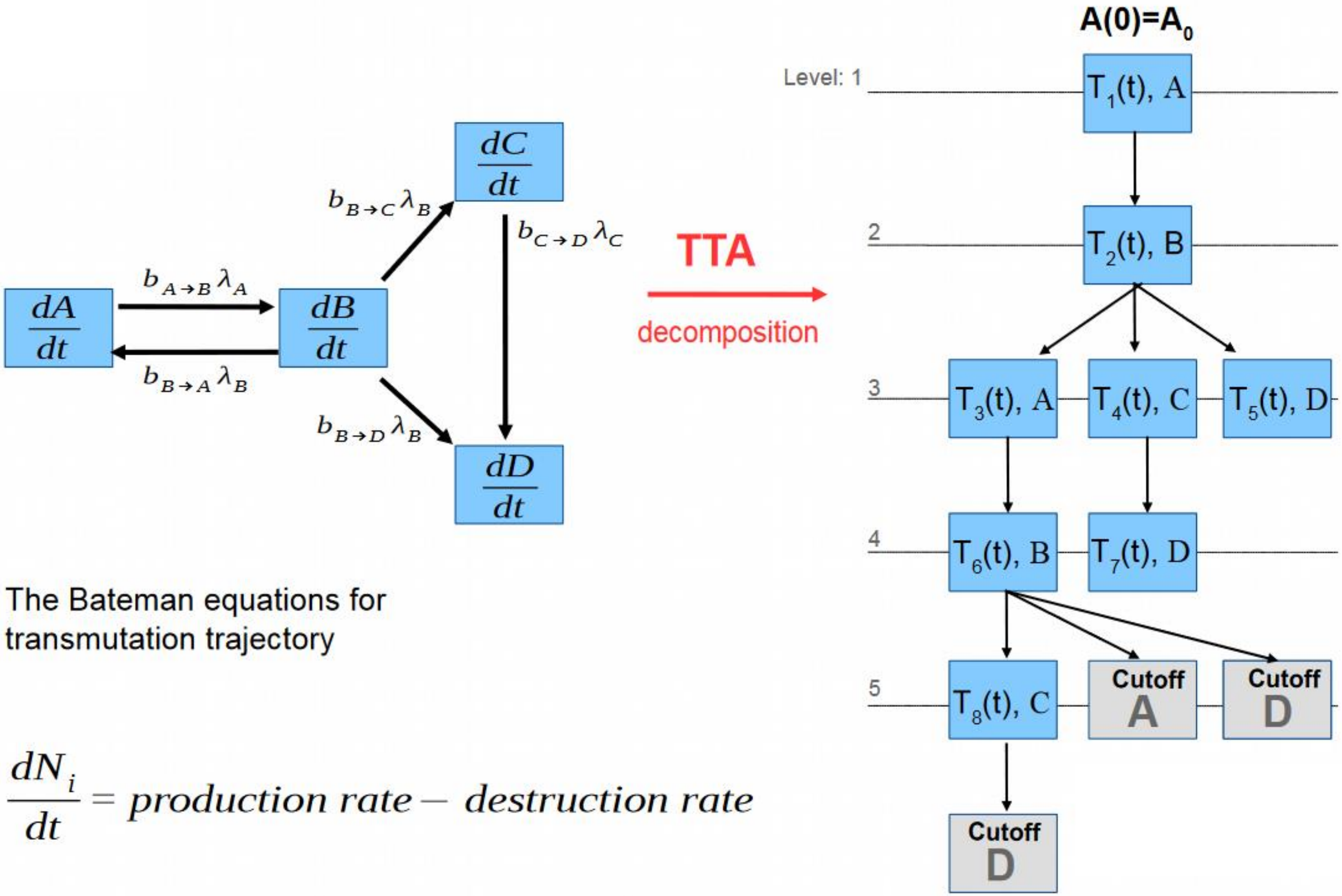


F Fresh Load
B1 Once Burned
B2 Twice Burned

1st First Batch
2nd Second Batch

Pu Plutonium
MA Minor Actinides
U Depleted Uranium
FP Fission Products
Example:
4B2 4th Load Twice Burned

Transmutation Trajectory Analysis



The Bateman equations for transmutation trajectory

$$\frac{dN_i}{dt} = production\ rate - destruction\ rate$$

Transition and Passage functions

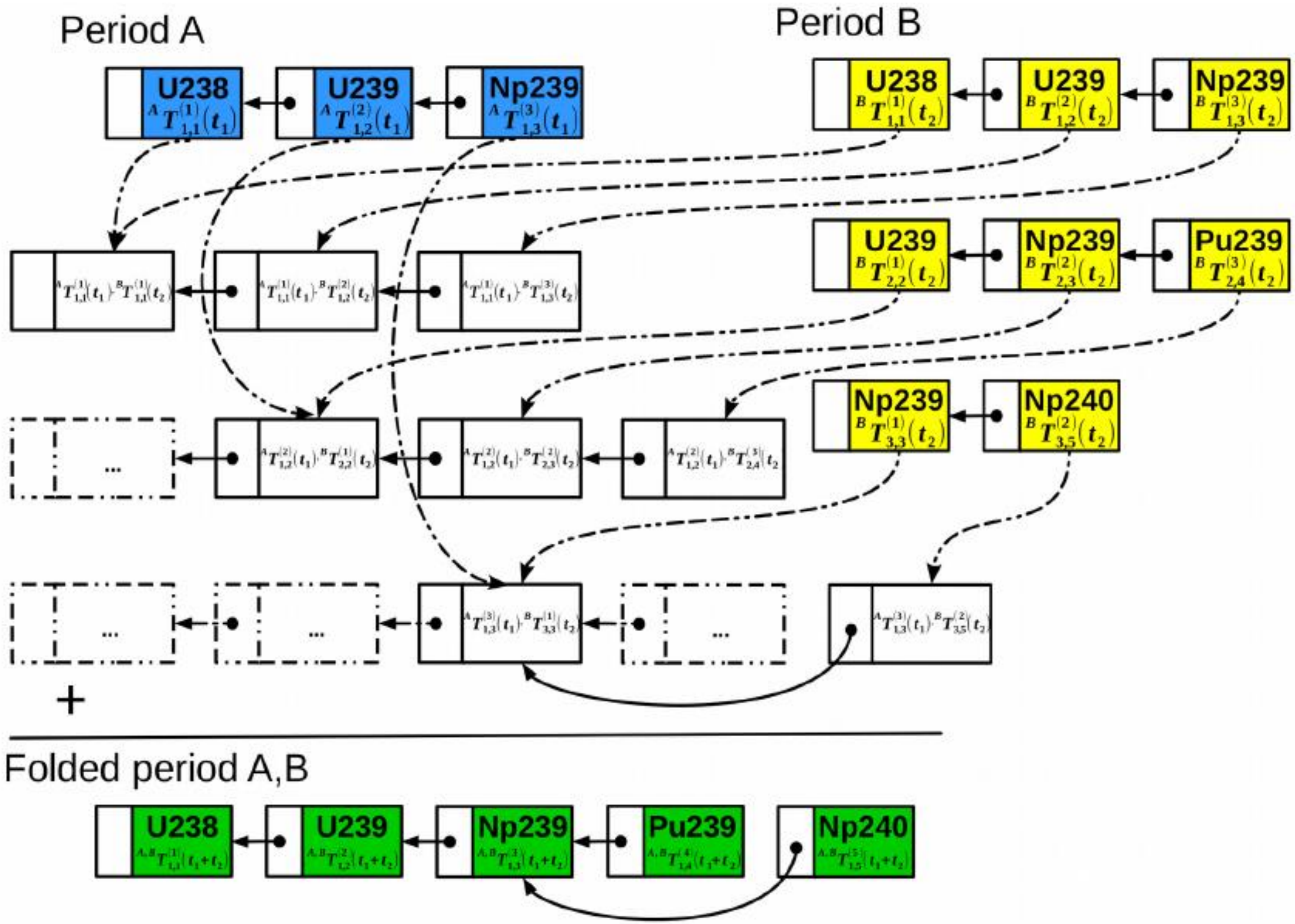
The trajectory transition calculate the number density that goes from initial nuclide to the formed nuclide for a given time t.

$$T_n(t) = N_n(t)/N_1(0)$$

Passage is defined as the total removal rate in the considered trajectory or a fraction of the nuclides in a chain that passed beyond n nuclide and is assigned (or not) to following nuclides in the chain for the considered period.

$$P_n(t) = I_n(t)/N_1(0)$$

Trajectory Period Folding Method



The Trajectory Period Folding Method

Folded trajectories have been applied in the following procedures:

1) The origin of the fuel.

What amounts of final composition are formed from the initial nuclides;

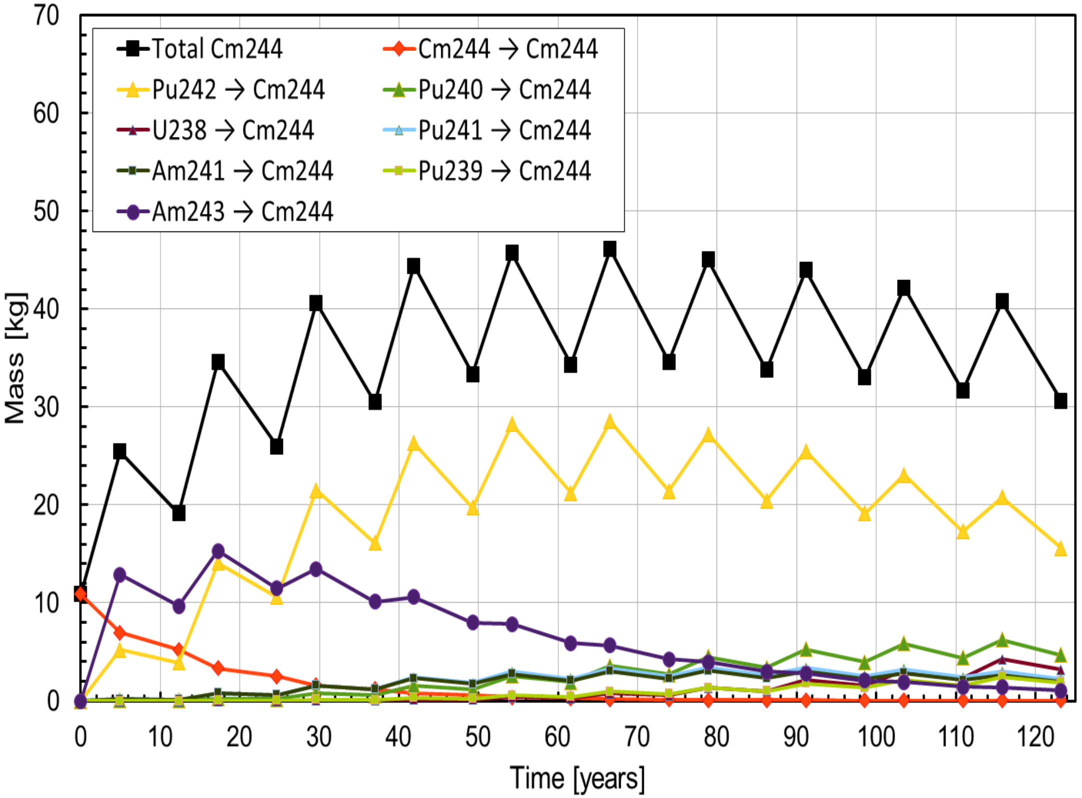
2) Trajectory evolution.

Which are the most common routes for transmutation over time;

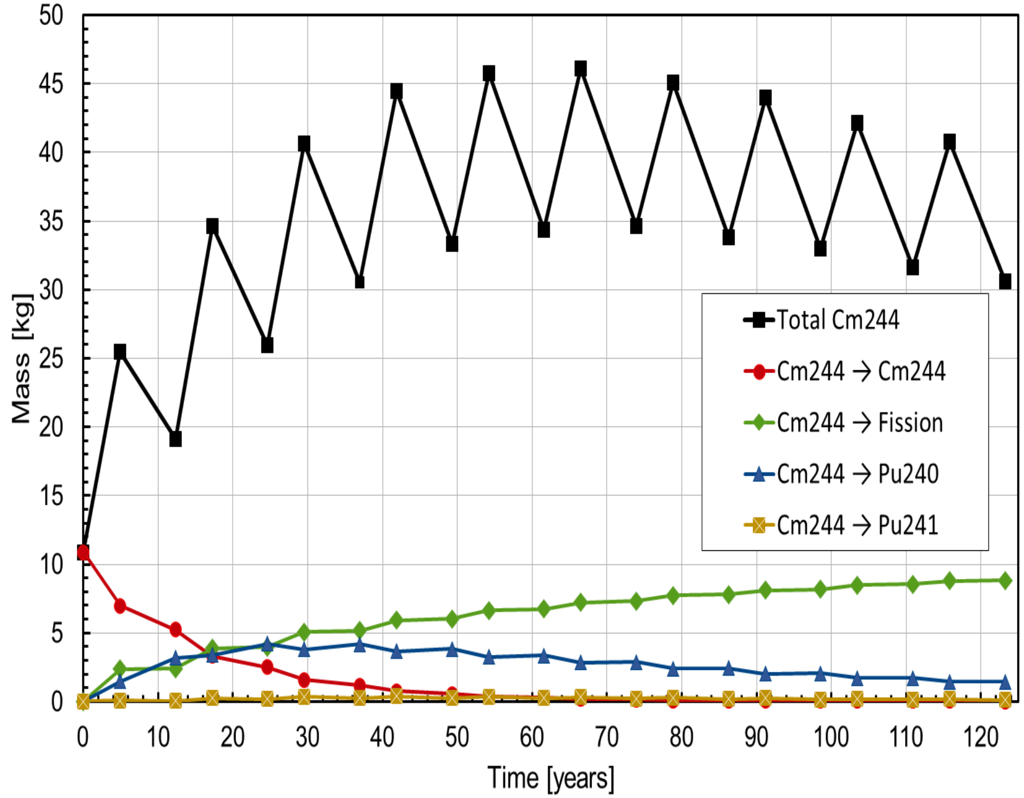
3) The most frequent reactions.

Which trajectories are the most sensitive for the cross-sections;

The origin of the fuel – Cm244

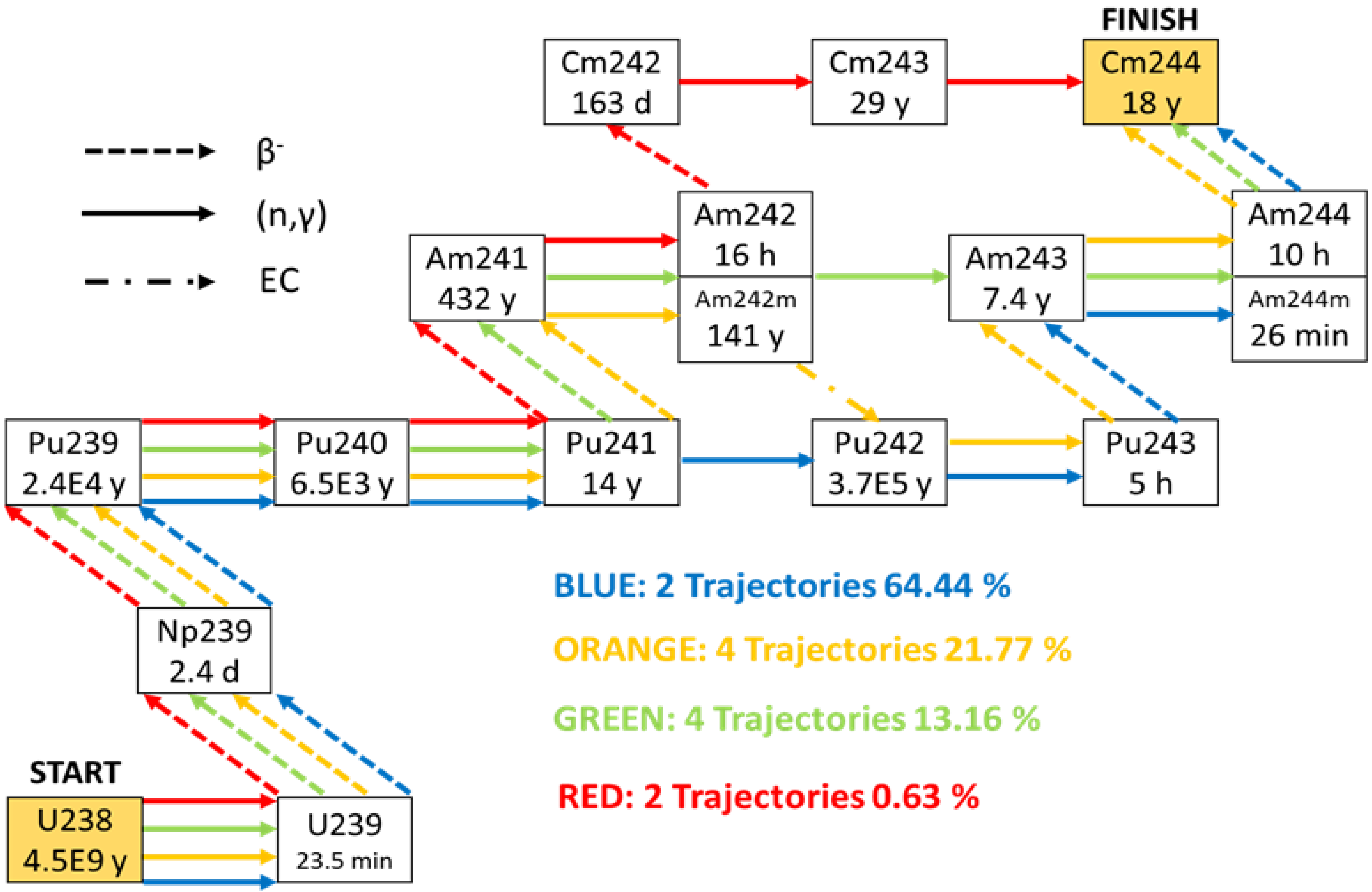


Batch mass flow of ^{244}Cm with source production from the initial fuel.



Batch mass flow of ^{244}Cm with source destruction from the initial fuel.

Trajectory evolution – Cm244



BLUE: 2 Trajectories 64.44 %

ORANGE: 4 Trajectories 21.77 %

GREEN: 4 Trajectories 13.16 %

RED: 2 Trajectories 0.63 %

The most frequent reactions – ^{244}Cm

Time [years]	$^{243}Am(n,\gamma)$	$^{242}Pu(n,\gamma)$	$^{240}Pu(n,\gamma)$	$^{241}Pu(n,\gamma)$	$^{241}Am(n,\gamma)$	$^{239}Pu(n,\gamma)$	$^{238}U(n,\gamma)$	$^{242m}Am(n,\gamma)$	$^{242}Cm(n,\gamma)$	$^{243}Cm(n,\gamma)$	$^{242}Am(n,\gamma)$
12,3	100,00%	28,88%	0,14%	0,13%	0,01%	0,00%	0,00%	0,00%	0,00%	0,00%	0,00%
24,6	99,99%	48,18%	0,79%	0,65%	0,14%	0,02%	0,02%	0,06%	0,01%	0,01%	0,00%
37,0	99,97%	61,98%	2,17%	1,65%	0,53%	0,09%	0,09%	0,24%	0,03%	0,03%	0,00%
49,3	99,96%	71,81%	4,33%	3,07%	1,26%	0,30%	0,30%	0,55%	0,04%	0,04%	0,00%
61,6	99,94%	78,76%	7,19%	4,86%	2,34%	0,72%	0,72%	0,97%	0,06%	0,06%	0,00%
73,9	99,93%	83,68%	10,70%	6,95%	3,75%	1,44%	1,44%	1,47%	0,07%	0,07%	0,00%
86,2	99,92%	87,10%	14,70%	9,27%	5,43%	2,53%	2,53%	2,01%	0,08%	0,08%	0,00%
98,6	99,90%	89,47%	19,11%	11,77%	7,34%	4,05%	4,05%	2,55%	0,10%	0,10%	0,01%
110,9	99,90%	91,07%	23,76%	14,36%	9,40%	6,00%	6,00%	3,08%	0,10%	0,10%	0,01%
123,2	99,89%	92,12%	28,60%	17,03%	11,57%	8,41%	8,41%	3,59%	0,11%	0,11%	0,01%

Reaction participation in the total ^{244}Cm production

Conclusions

- the focal point of the work was the practical application of the the MCB code to the modeling of ELFR reactor in the adiabatic equlibrium state

- TTA and period folded trajectory method can indicate trajectories with the highest transition participating to the MA inventory,

- nuclear reactions belonging to chosen trajectories can be quantitatively characterized to allow for the identification of the most efficient one in production of crucial nuclides,

- the standard solution for the time evolution of nuclide concentrations governed by a set of first-order differential equation may be use to represent the isotope mass evolution as an integrated function for the entire irradiation period including multi-cycling reloading scheme,

- improvement of nuclear cross sections should focus on those nuclides which are present in trajectories and participate in production of the gateways nuclides or safety related nuclides

References

AGH University of Science and Technology
Faculty of Physics and Applied Computer Science

Doctoral thesis

Przemysław Stanisz

Lead cooled reactor neutronic study towards verification of nuclear data and modelling methodology for nuclear transmutations

Available online at www.sciencedirect.com
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General solution of Bateman equations for nuclear transmutations

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Monte Carlo modeling of Lead-Cooled Fast Reactor in adiabatic equilibrium state

CrossMark

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Mikolaj Oettingen
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Tomasz Mirowski

THE MCB CODE FOR NUMERICAL MODELING OF FOURTH GENERATION NUCLEAR REACTORS

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Благодарю за внимание!

Thank you for your attention!

Alexandrov Research Institute of Technology

The State Atomic Energy Corporation «ROSATOM»

USING THE KORSAR COMPUTER CODE FOR MODELING THERMAL-HYDRAULIC PROCESSES WITH LIQUID-METAL COOLANTS

D.V. Benediktov, A.V. Vakarin, S.N. Volkova,
I.G. Danilov, Yu.A. Migrov

V International Scientific Conference «Heavy Liquid-Metal Coolants in Nuclear Technologies *(HLMC -2018)*»
October 08–10, 2018
Obninsk

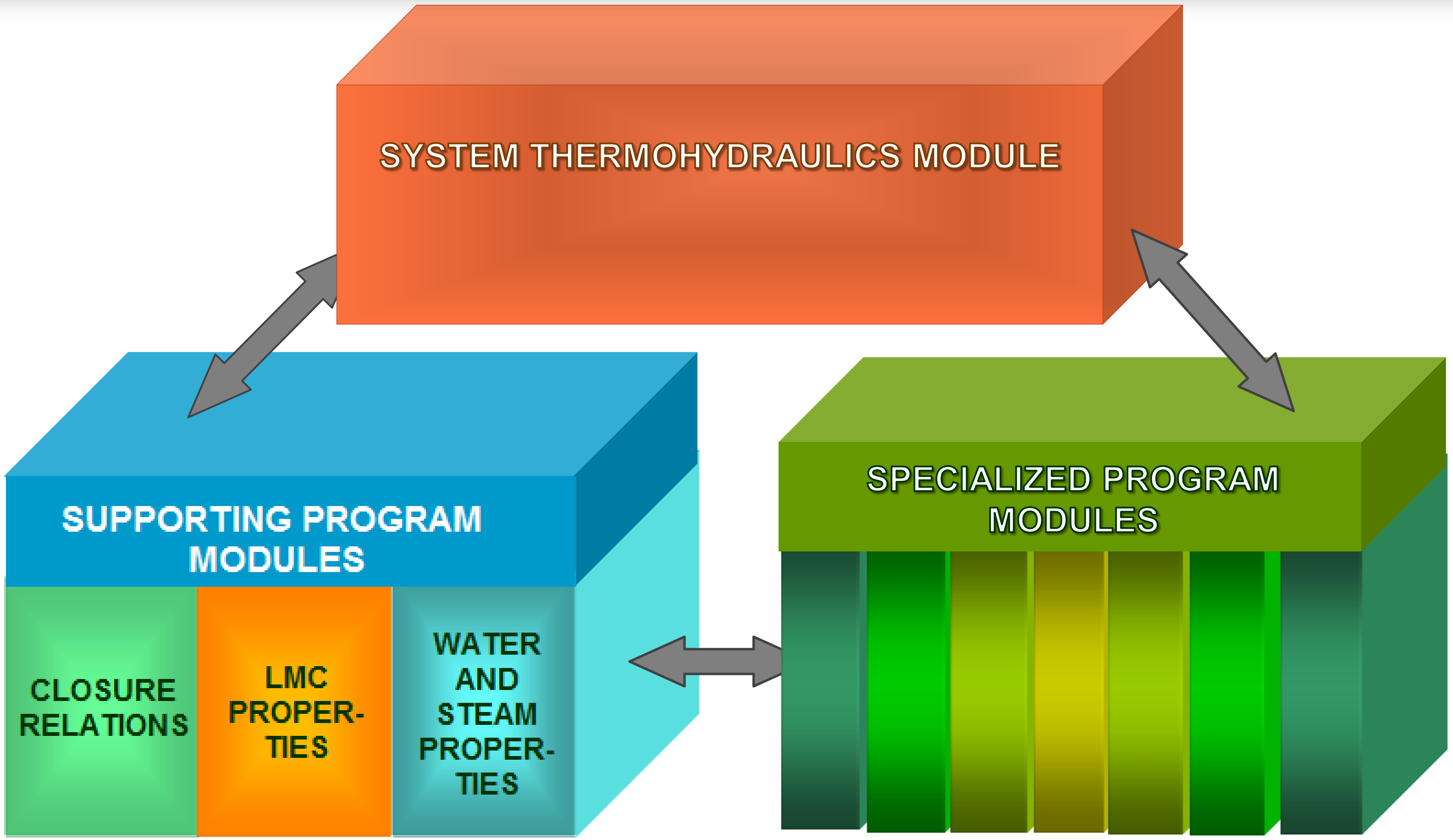
Goals and objectives

Main goal is development of thermal-hydraulic computer code intended for numerical modeling of dynamic processes at advanced NPPs with heavy liquid metal coolants (LMC).

Main objectives:

- Development of technology platform for computer code;
- Selection and validation of physico-mathematical models and numerical methods;
- Software implementation of mathematical models and software testing;
- Integrated verification of computer code;
- Expertise and certification.

KORSAR/LMC



Functional modules

Characteristics of KORSAR/LMC specialized version

1. The module to calculate the thermal-physical properties of lead-bismuth eutectic is developed and software-implemented for the following application range:

   - temperature: 398.15÷1944К;

   - pressure: 0÷10MPa.

   Sources of properties:

   - Sobolev V. Database of thermophysical properties of liquid metal coolants for GEN-IV. Scientific Report SCK-CEN-BLG-1069, 2010.

   - Morita K. at al. Thermodynamic properties of lead-bismuth eutectic for use in reactor safety analysis. ICONE13-50813. 13-th Int. Conf. on Nuclear Engineering, Benjing, China, May 16-20, 2005.

2. Coolant energy equation includes an additional term describing conductive heat transfer.

Characteristics of KORSAR/LMC specialized version

3. **The characteristic of used coolant is introduced (key: met)**

   0 – water, 1 – LMC

   Interaction of circuits with different coolants is provided via heat-conducting structure.

4. **Simplified flow-pattern and heat-transfer maps are used for LMC. LMC freezing and boiling modes are not considered.**

   Friction factors for single-phase flow are calculated using dependences for water coolant.

   Heat-transfer coefficients are calculated accounting for the flow path geometry:

   - for round pipe

$$\mathrm{Nu} = 5 + 0{,}025 \cdot \mathrm{Pe}^{0{,}8} \qquad \mathrm{Pe}<4000 \qquad 0{,}004 \le \mathrm{Pr} \le 0{,}04$$

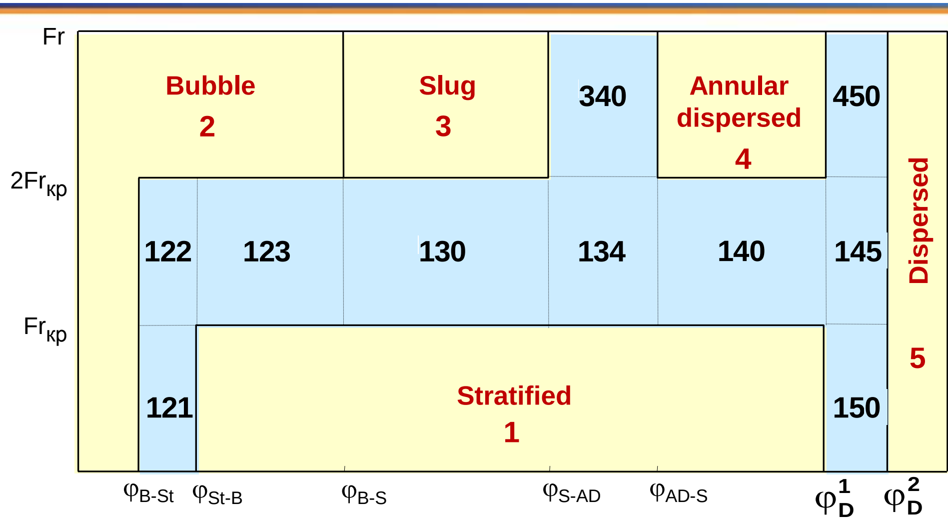
   - for rod bundles:

$$\mathrm{Nu} = 7{,}55 \cdot x - 20 \cdot x^{-13} + 0{,}041 / x^2 \cdot \mathrm{Pe}^{0{,}56+0{,}19 \cdot x} \qquad 1{,}3 \le x \le 2 \qquad \mathrm{Pe}<4000$$

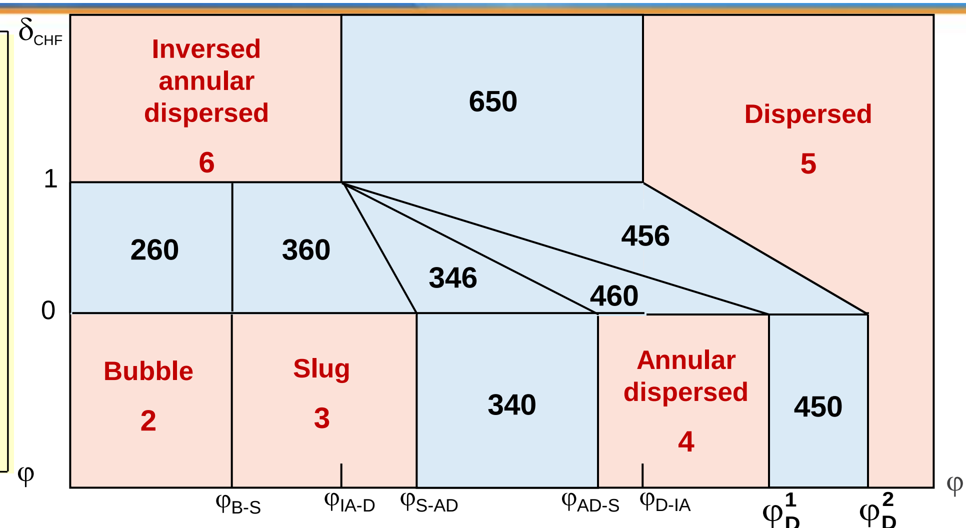
$$\mathrm{Nu} = 0{,}047 \cdot (1 - \exp(-3{,}8 \cdot (x-1))) \cdot (\mathrm{Pe}^{0{,}77} + 250) \qquad 1{,}1 \le x \le 1{,}95 \qquad 30<\mathrm{Pe}<5000$$

FLOW-PATTERN MAPS

Horizontal channel



Vertical channel



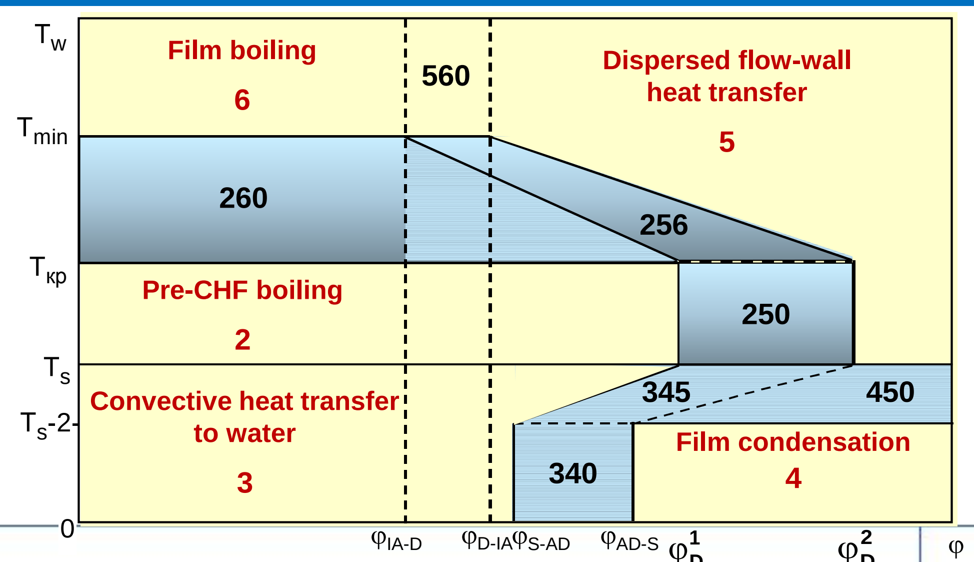
Heat transfer mode map

φ – void fraction

$$Fr = \frac{\left| j_V - 1{,}5 j_f \big/ X^{0{,}85} \right|}{\sqrt{gD}} \cdot \sqrt{\frac{\rho_V}{\rho_f - \rho_V}}$$

$$X = \left(\frac{j_f}{j_V}\right)^{0{,}9} \cdot \left(\frac{\rho_f}{\rho_V}\right)^{0{,}4} \cdot \left(\frac{\mu_f}{\mu_V}\right)^{0{,}1}$$ – Lokkart-Martinelli parameter

$$\delta_{CHF} = \frac{T_W - T_{кр}}{T_{min} - T_{кр}}$$

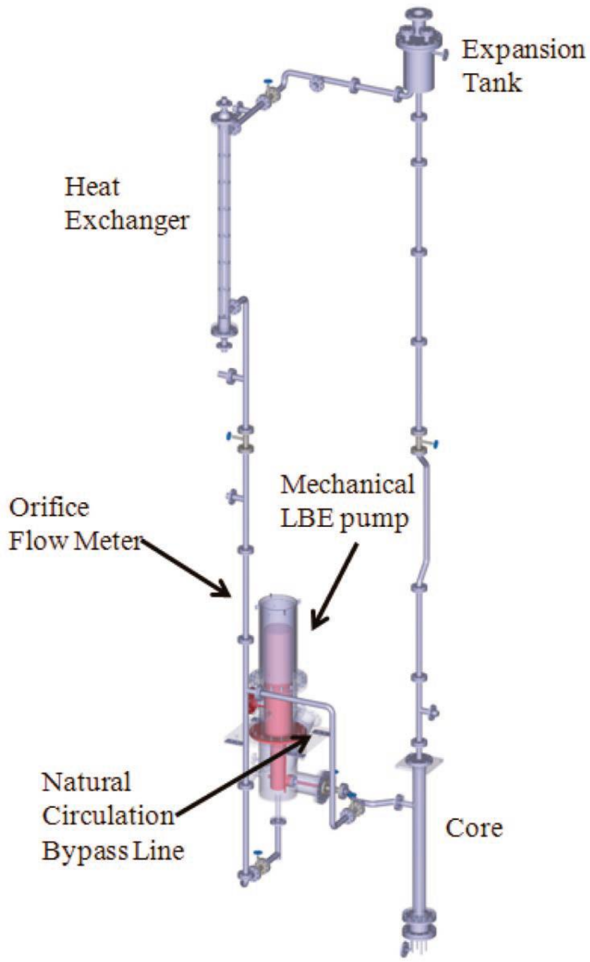


Characteristics of KORSAR/LMC specialized version

5. **The module for gas pressurizer computation (ACCUM module) is improved via using LMC as a coolant. Gas dilution and transfer is not considered.**

6. **The pressure field for LMC is calculated using single-matrix method. Good pressure matrix condition allows solving simultaneous equations for pressure for two coolants without modification of subprogram of numerical computation. To assure the stability of method, compressibility is restricted by the lower limit of ~10^{-9}.**

KORSAR/LMC verification
HELIOS test facility (South Korea)



HELIOS test facility is a single-loop facility intended to investigate natural and forced circulation of lead-bismuth coolant.

Main components:

-**core simulator;**

-**expansion vessel;**

-**intermediate heat exchanger;**

-**by-pass for coolant purification;**

-**by-pass for natural circulation;**

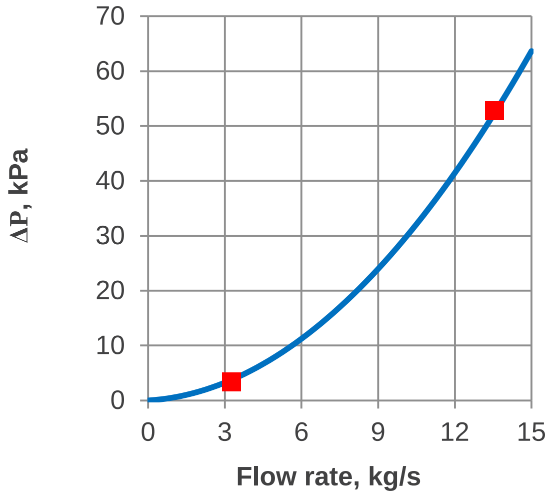
-**pump;**

-**drainage tank.**

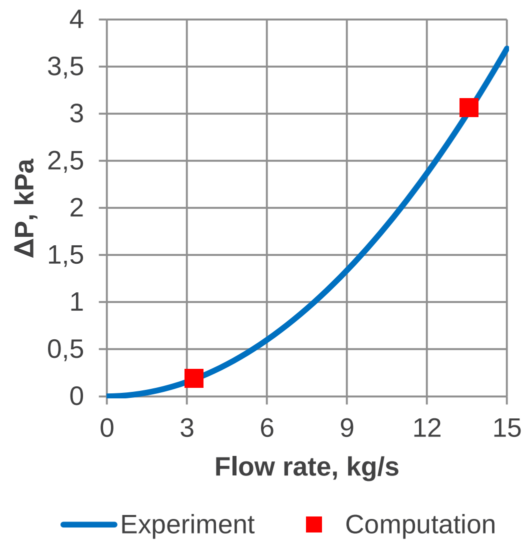
KORSAR/LMC verification
HELIOS test facility (South Korea)

The experiments, used as a benchmark, have resulted in hydraulic pressure losses at core simulator, valve, and measuring orifice.

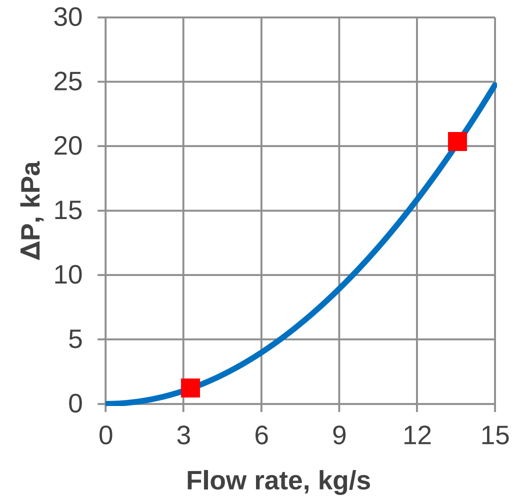
Experimental values of pressure losses were obtained for two flow rate values (13.57 kg/s and 3.27 kg/s) which were used for hydraulic characteristics of each pressure loss location.



Core simulator

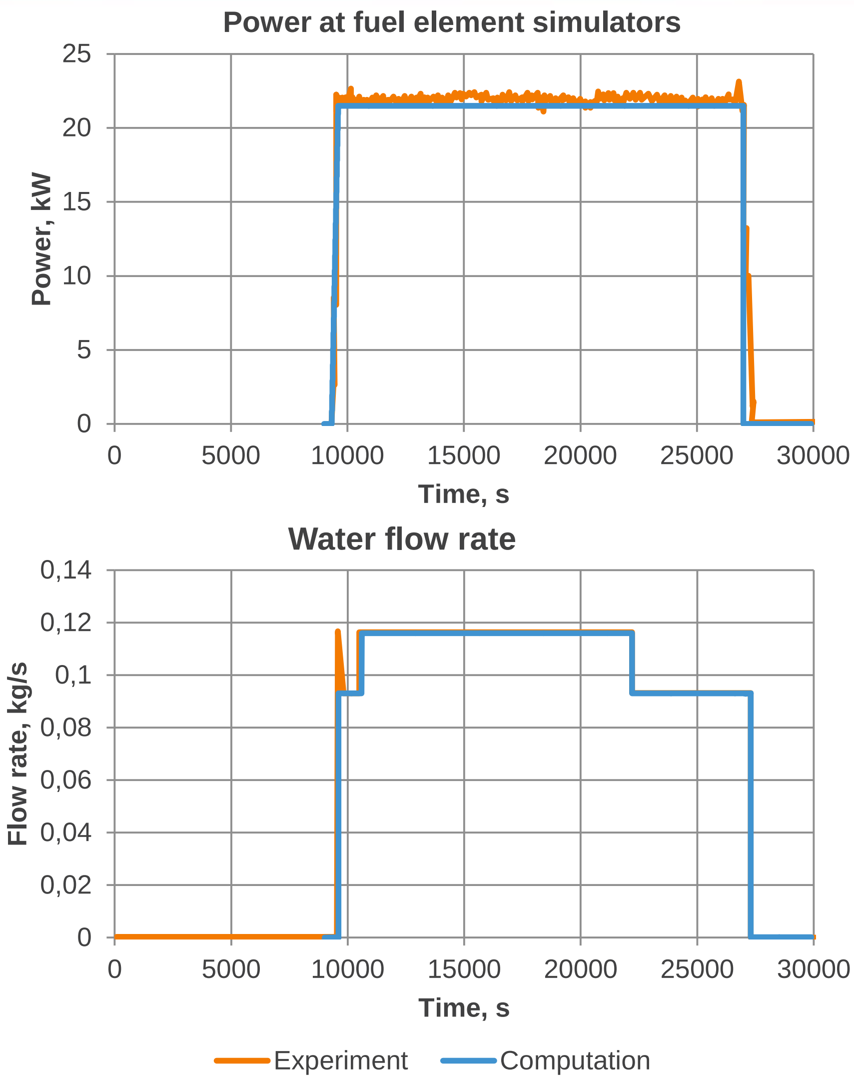


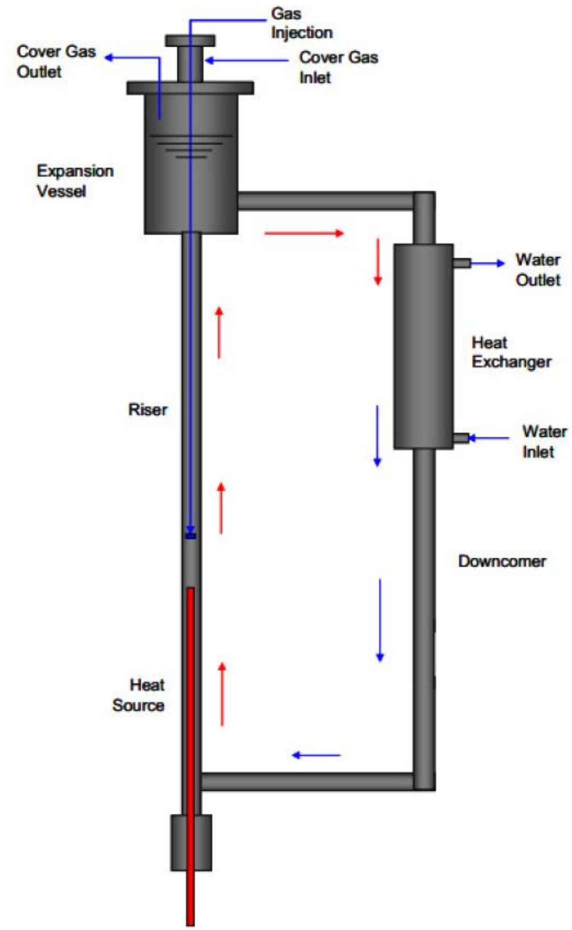
Valve

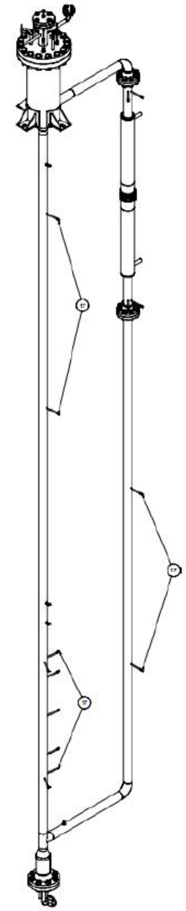


Orifice

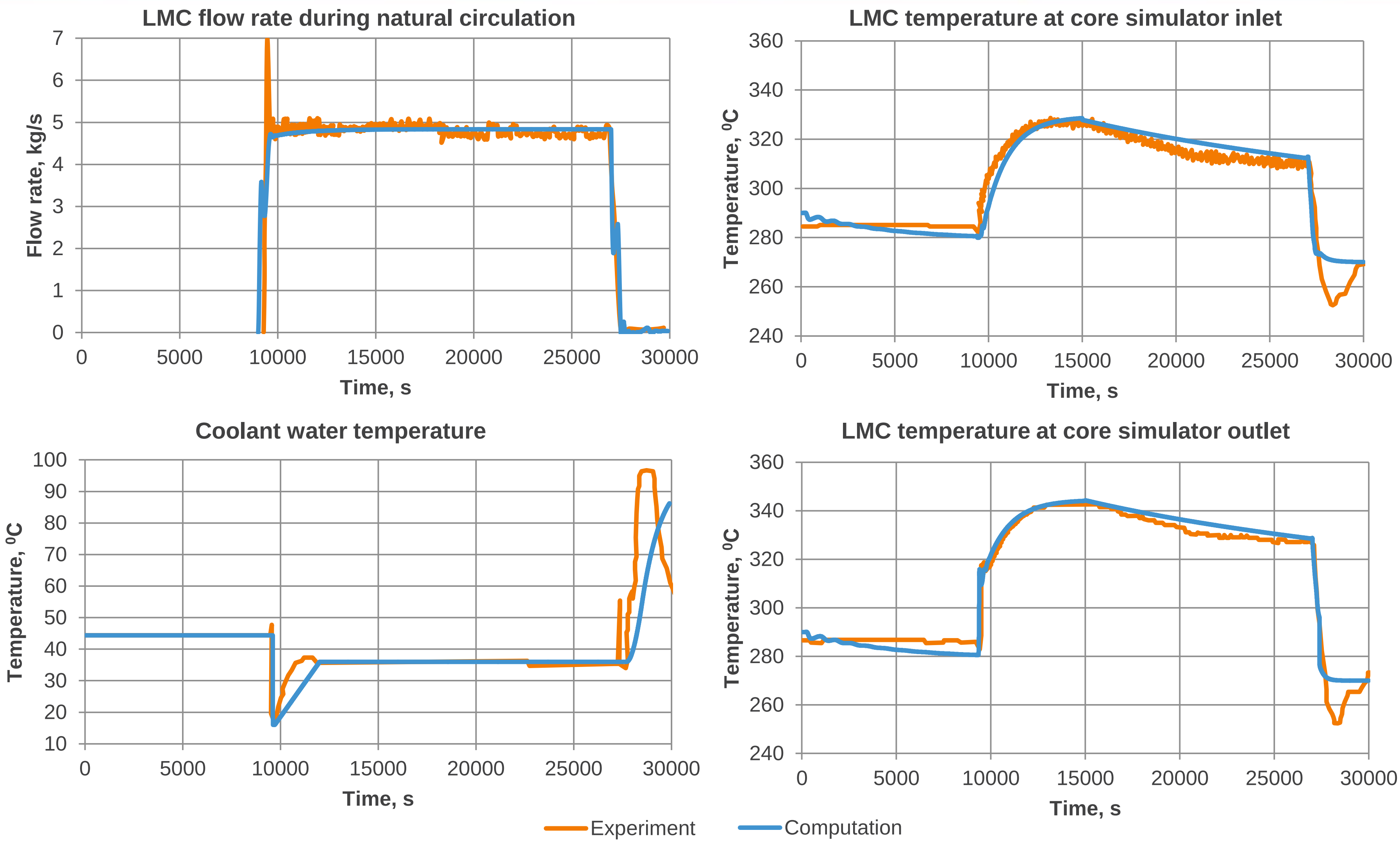
KORSAR/LMC verification
NACIE test facility (Italy)







KORSAR/LMC verification NACIE test facility (Italy)



Conclusion

1. The first version of KORSAR/LMC was developed to enable simultaneous performance of reactor circuits (without hydrodynamic and mass-exchange interaction) with two coolants: water and lead-bismuth eutectic.

2. The numerical simulation of experimental modes at the integrated test facilities has shown the availability of KORSAR/LMC computer code for modeling of thermohydraulic.

3. For certification of KORSAR/LMC computer code it is required to develop the verification matrix containing information about simulation of separate physical phenomena at integrated or local test facilities.

4. To enhance the potential of the code in modeling the interloop leaks it is required to develop the relevant computational models using the results of experimental study of water steam/lead-bismuth eutectic interaction.

Thank you for attention!