Thermal-hydraulic and source term MELCOR studies for non-electrical applications HTGRs

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Why MELCOR 2.X for HTGRs?

1. GEMINI Plus project – H2020, starting point 2017, now finished. GEMINI https://gemini-initiative.com/geminiplus/

The research and development activities performed in the GEMINI+ project aimed to support the GEMINI Initiative. Over the course of 36 months, GEMINI+ partners worked together towards the demonstration of high temperature nuclear cogeneration with a High Temperature Gas-cooled Reactor (HTGR).

- The t-h calculations for Prismatic type 180 MWth HTGR configuration developed for the Polish industry, which screened the calculated maximum fuel temperatures (set acceptable level of 1600°C).

Development of Accident conditions:

- Depressurized Loss of Forced Circulation (DLOFC), 65 mm break in the primary system
- Pressurized Loss of Forced Circulation (PLOFC).

2. Small scale HTGR demonstrator pre-conceptual design national project GOSPOSTRATEG-HTR.

GOSPOSTRATEG-HTR project, which is an important vehicle for shaping the country's energy policy, allowing for the combination of the organizational potential of the state with the research and scientific capabilities of research institutions, supporting the coordination of preparation for the practical use of HTGR in the Polish economy.

- The t-h calculations plus exploration of the MELCOR 2.X capabilities of the source term evaluations for the HTGR.

The non-LWR modelling by U.S.NRC, which has a reflection in the series of documents, describing the Integrated Action Plan (IAP) for Advanced Reactors [32]. The most interesting from those documents, were the one deliberating on the analytical tools (called as a Strategy 2), among which the MELCOR code was described as a tool for severe accident progression, source term and consequence analysis evaluations.

GEMINI+ and pre-conceptual design of the research reactor

Design philosophy

- a significant number of industrial sites require process heat in Poland;
- HTGR can serve clean energy source;
- no commercial nuclear reactor constructed in Poland yet - one needs to gain some competence (human resources, industry, regulator, etc.);
- research reactor can serve building competence, research tasks, and also a small-scale demonstrator of HTGR technology for industrial applications;
- the design should combine features of the industrial reactor as planned by the GEMINI+ project and proven elements of the HTTR as a test reactor;
- it is expected to lead to a unique core and a reactor design matching specific Polish requirements in research, demonstration and applications;
- one of the objectives is that the power of the reactor should be as high as possible (order of 30-40 MWt) to maximize similarity with an industrial GEMINI+ type FOAK reactor design of 180 MWt power.

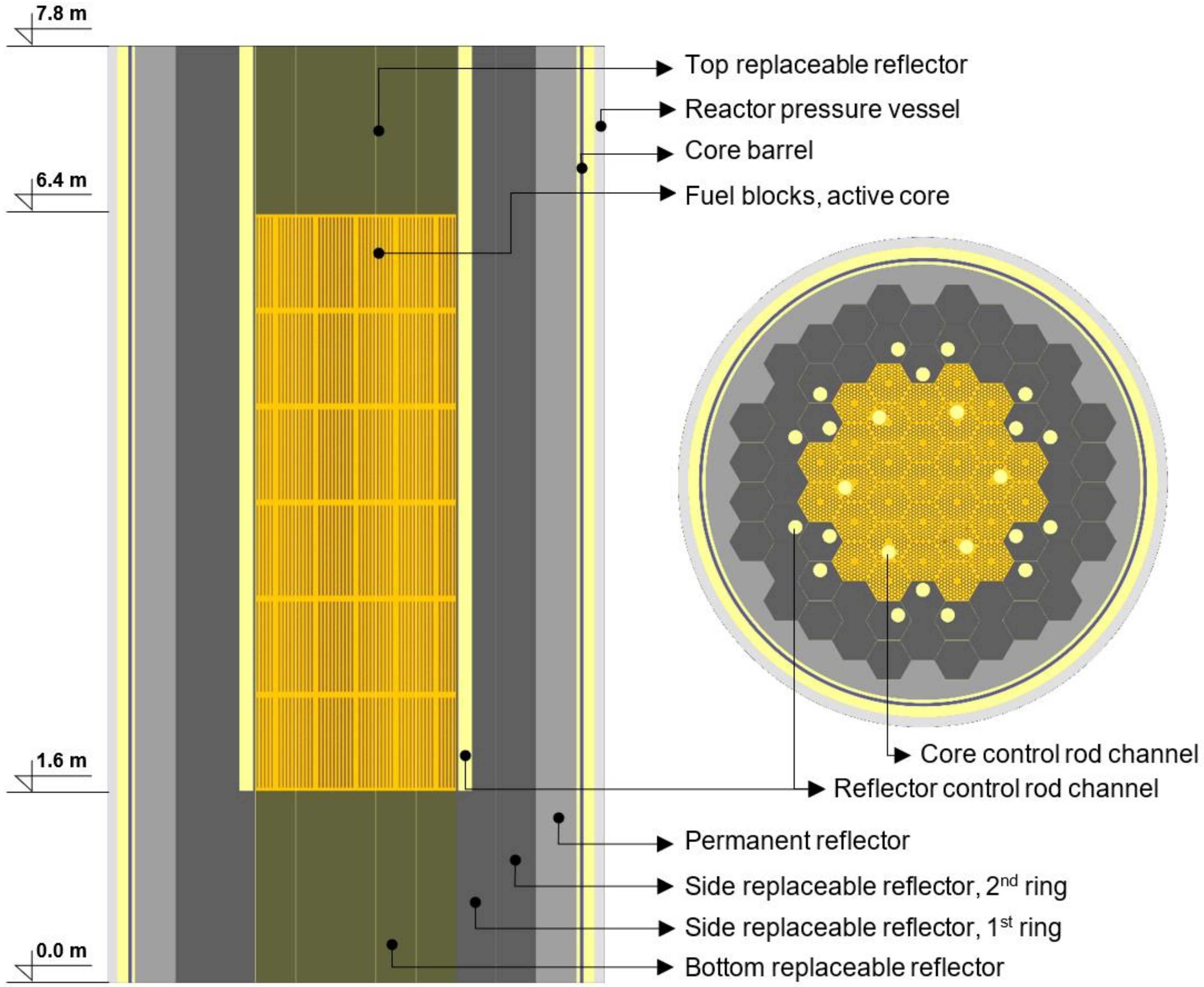

Fig. TeResa core configuration, Serpent model visualization. [4]

Reactor power conversion system

- **Power Conversion System**

Secondary system

- Main Steam Line delivers fresh steam from SG to Reboiler and Turbine
- Steam temperature before the Turbine is adjusted according to specification of TG
- Discharges from Turbine:
  - HP: HP regeneration and Deaerator
  - LP1: LP regeneration
  - LP2: District heating subsystem
- Some fresh steam can be taken to Deaerator
- Condensate from Reboiler (with reduction station), Condenser (with Condensate Pump) and regeneration is gathered in Deaerator
- Water from Deaerator is delivered through Feedwater Pump, HP Regeneration HX and FW Isolation/Control Valve to SG

- Tertiary system (Process system):
  - Water to Reboiler: return from process + Make-up system (Tanks, RO, etc.)
  - 3 sections of Reboiler: water heater, evaporator, steam super-heater
  - Process Steam parameters: t = 535 ^{0}C, p = 138 bar

- District heating system
  - Feeded with steam from LP discharge
  - Low pressure Intermediate Circuit (heat clutch) to improve isolation of district circuit water

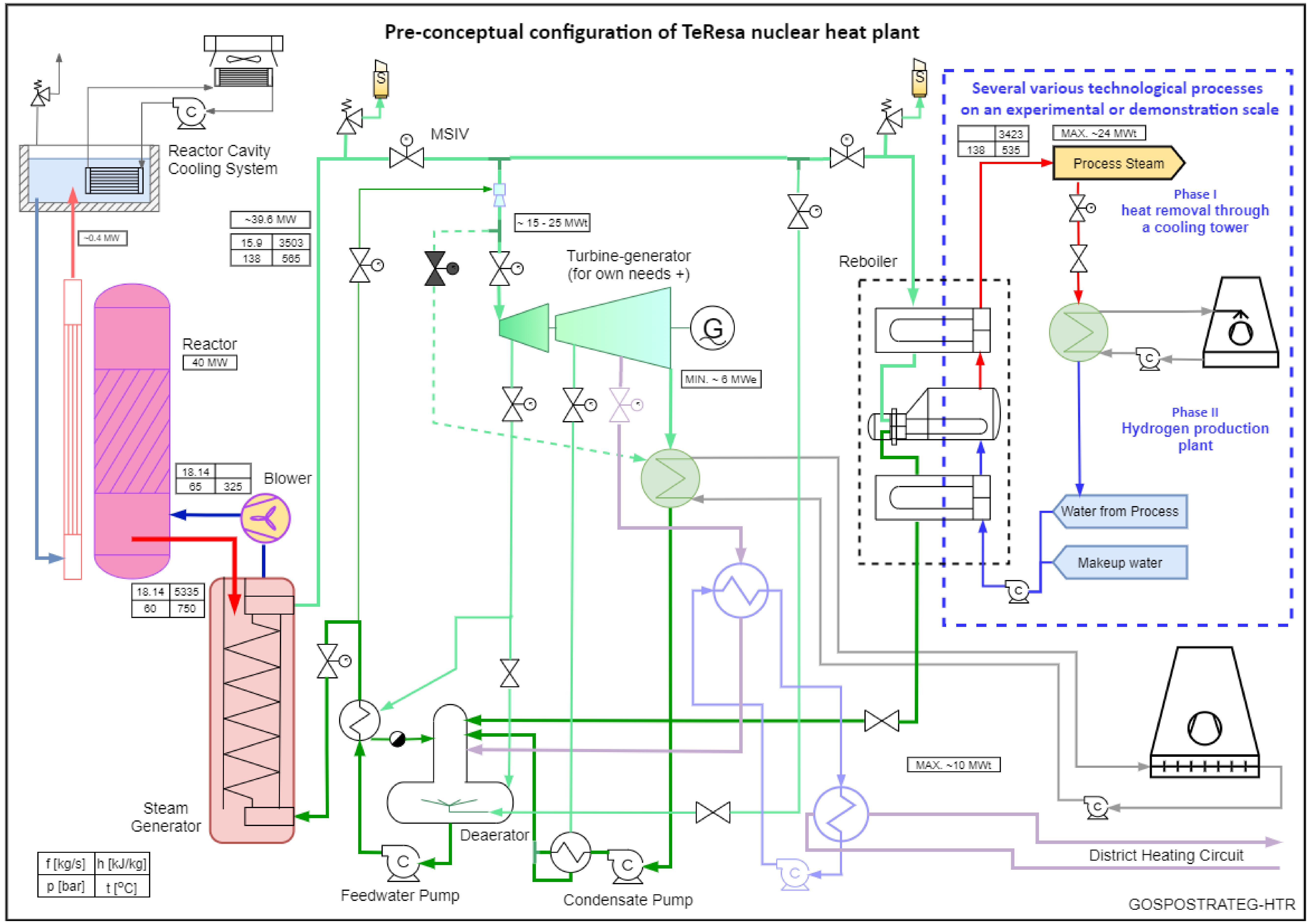

Fig. Pre-conceptual configuration of power conversion system for TeResa nuclear heat plant [4]

GEMINI+ and pre-conceptual design of the research reactor

- The GEMINI+ reactor specified for Polish end-users is designed as with thermal power of 165 MWth and 15MWth for internal power needs. The GEMINI+ system is conceived as a scaling down of the Framatome Steam Cycle – High Temperature Gas-Cooled Reactor (SC-HTGR) [31] and the European HTR-MODUL design [24]. GEMINI+ design, (compact configuration), power density of 6 MW/m3.
- The TeResa reactor is a pre-concept based on the GEMINI+ HTGR solution [1], which was built on the knowledge acquired in the past European R&D projects as well as existing HTGR designs, like GT-MHR, MHTGR, and SC-HTGR [2], power density of 2.4 MW/m3.
- GEMINI+ and pre-conceptual research reactor design is summarized in the Table 1.
- The data established and the design was one of the project GOSPOSTRATEG-HTR [3] milestones, which was later followed by the demonstration of the use of developed safety evaluation methodology for partial design safety considerations.

Tab. Basic pre-concept design data used for the analytical calculations. [4]

General Information		
	Unit	Value
Reactor thermal output (gross thermal power)	MWth	40 (GEMINI+ 180 MWt)
HTGR type	-	prismatic, block-type
Graphite block type	-	similar to GEMINI+
Graphite block height	cm	80
Graphite block hexagon flat-to-flat distance	cm	36
Graphite block material (fuel and reflector blocks)	-	NBG-17
Fuel	-	TRISO, 12 wt% enriched UO2
Active core height	cm	480 (880)
Active core effective radius	cm	212
RPV outer radius	cm	224.4
RPV material	-	SA508
Primary side		
Coolant type	-	helium
Coolant flow direction	-	downward flow pattern
Helium mass flow rate (at 100% power)	kg/s	18.14 (GEMINI+ 81.5)
Primary system pressure	MPa	6.0
Reactor vessel inlet coolant temperature	°C	325
Reactor vessel outlet coolant temperature	°C	750
Number of cooling loops	-	1
Steam generator type		once-through, helically coiled bundles
Secondary side		
Secondary side coolant	-	water
Main steam pressure (at Steam Generator (SG) outlet)	MPa	13.8
Main steam temperature (at SG outlet)	°C	540
Main steam mass flow rate (at 100% power)	kg/s	15.9 (GEMINI+ 251.0)
Feed water pressure (at SG inlet)	MPa	13.97
Feed water inlet temperature (at SG inlet)	°C	210
Feed water mass flow rate (at 100% power)	kg/s	15.9 (GEMINI+ 251.0)

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Reactor core pre-concept information

Industrial HTGR & Research reactor design

CORE ARRANGEMENT

- 31 fuel column, 3 rings around central fuel column with 2 mm gap between
- fuel column is a stack of 6 fuel blocks.
- active core height is RR pre-concept 4.8 m, GEMINI+ 8.8 m, and equivalent core diameter is 2.12 m.
- active core is surrounded by two rings of replaceable reflector and permanent reflector.

FUEL DESIGN

- TRISO fuel – a particulate fuel with ceramic multi-layer coatings surrounding UO_2 kernel.
- UO_2 kernels: enrichment 12% (U-235) but higher up to 19.75% may be considered.
- German reference fuel (HTR-MODULE) – material: UO_2/buffer/IPyC/SiC/OPyC,
- TRISO dispersed in a cylindrical graphite matrix (fuel compact, ⌀1.245 cm, 5 cm height) with packing fraction of 15%.
- reference design: a uniform enrichment over the core (one-zone) and once-through fuel cycle is considered.

BURNABLE POISONS

- reference design: europium oxide, Eu_2O_3 (atomic fractions: Eu-151 - 0.1912, Eu-152 - 0.2088, O - 0.6).
- material composition of a BP rod is uniform over the core
- number of BP rods per fuel block varies: 1 BP – peripheral core ring, 4 BP – control block, and 6 BP – other blocks.

REACTIVITY CONTROL SYSTEM (RCS)

- RCS comprises control rod system (CRS) and reserve shutdown system (RSS).
- 18 rod channels in the first ring of the side replaceable reflector and 6 rod channels in the active core.
- CRS uses boron carbide (B_4C) absorbers, length of the active part of control rod is 560 cm.
- reserve shutdown system (RSS) with boron carbide pellets is considered.

Performed calculations

Calculation campaign

Simulations performed in the area of:

- Analysis for the radioactive substances distribution in the HTGR circulation loop and radiation hazards under normal operating conditions. Core releases and effects of those hypothetical releases.
- Analysis of built-in safety features, safety systems, requirements for their operation in emergency situations and their classification and qualification for emergency conditions.
- Identification of initiating events and accident scenarios.
- Determination of the distribution and possible propagation processes of fission products in the HTGR reactor and their releases outside the HTGR reactor in situations of DBAs and severe accidents.

Field of Study	Code Name	Outline
Neutronics	Serpent 2 [11]	Serpent is a continuous-energy multi-purpose three-dimensional Monte Carlo particle transport code. It is in development at VTT Technical Research Centre of Finland since 2004.
	MCB [8]	MCB—Monte Carlo continuous energy burnup code is a general-purpose code used to calculate a nuclide density time evolution, including burnup and decay. Internally, MCB comprises MCNP code, which is used for transport calculations, and is coupled with thermal-hydraulic code POKE (thermohydraulic software)(Gulf General Atomic Incorporated, USA).
Thermal hydraulics	MELCOR 2.2 [12]	MELCOR is a fully integrated, engineering-level computer code developed by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, that models the progression of severe accidents in nuclear power plants.
	CATHARE2 [29]	CATHARE (Code for Analysis of Thermal hydraulics during an Accident of Reactor and safety Evaluation) is a two-phase thermal-hydraulic system used in pressurized water reactor safety analyses, the verification post-accidental operating procedures, and in research and development.
	ANSYS Fluent 2020 R1 [30]	Ansys FLUENT software contains the broad physical modeling capabilities needed to model flow, turbulence, heat transfer, and reactions for industrial applications.

Design neutronic investigations: MCB and POKE

Neutronic investigations

1. Core design considerations: carried out by the NCBJ project partners – AGH (Akademia Górniczo-Hutnicza) with the MCB [8] and POKE [9] codes (neutronic - thermal-hydraulic coupling) with control rods movement,
   - the fuel cycle length of 1250 days for the reference core configuration, that is uniform over the core fuel enrichment,
   - fuel packing factor, and
   - BP rod parameters.
2. The results of the neutronic calculations – core inventory in different cycle states, power distributions were input for the thermal-hydraulic and source term evaluations [10].

- The comparison of the Beginning of Cycle (BOC) axial power profile normalized to the core height of the GEMINI+ and TeResa cores - thermal response in the neutronic calculations.
- The additional cycle optimization calculations: burnable poison rods configuration able to ensure critical core state, with minimal control rod usage influenced the shape of the power profile.
- At the BOC, the core maximal power for the TeResa reactor was found to be closer to the bottom reflector component, due to the insertion of the control rod from the top by the Control Rod Drive Mechanism (CRDM).

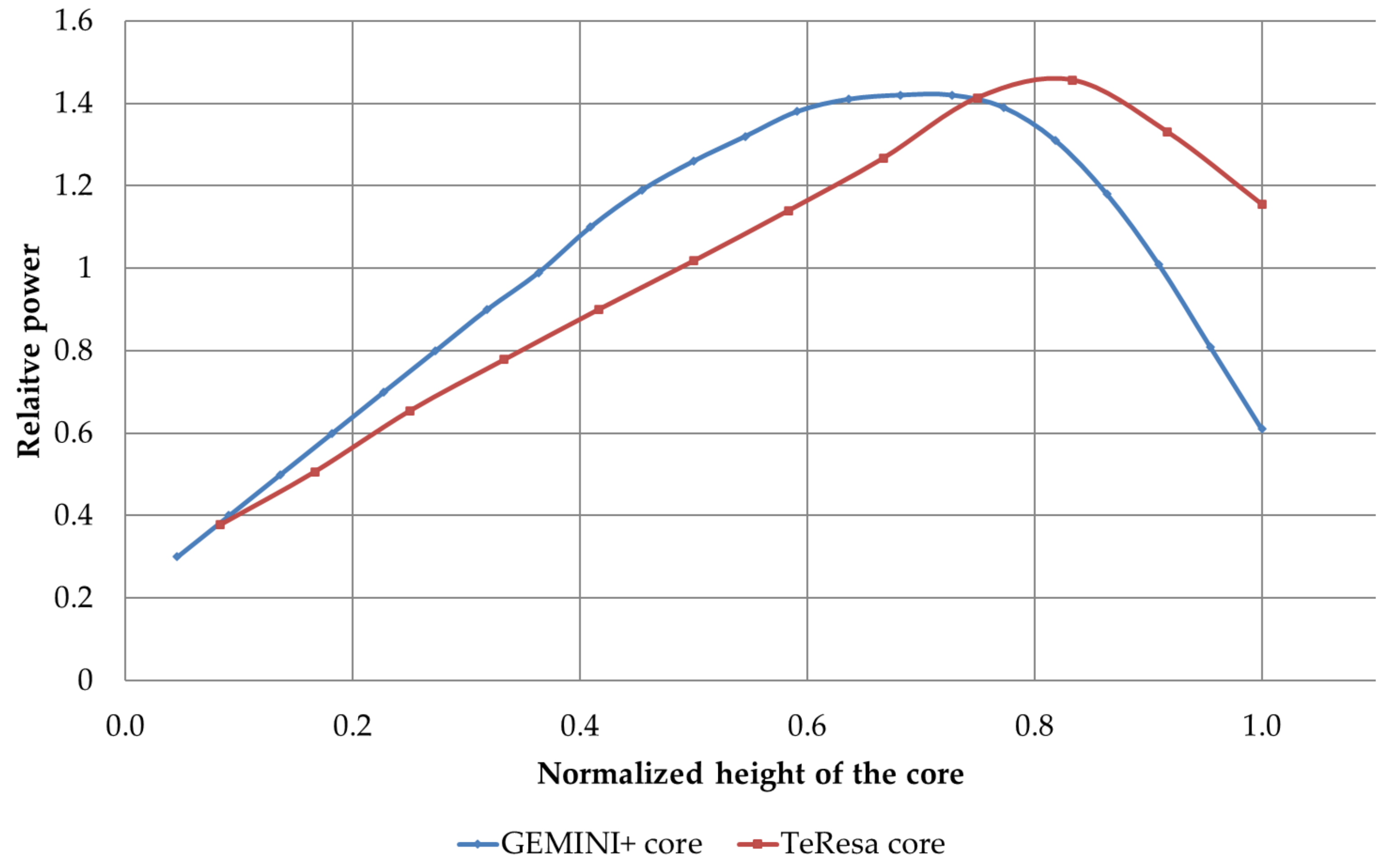


Fig. The comparison of the axial power distribution for GEMNI+ and research reactor core.

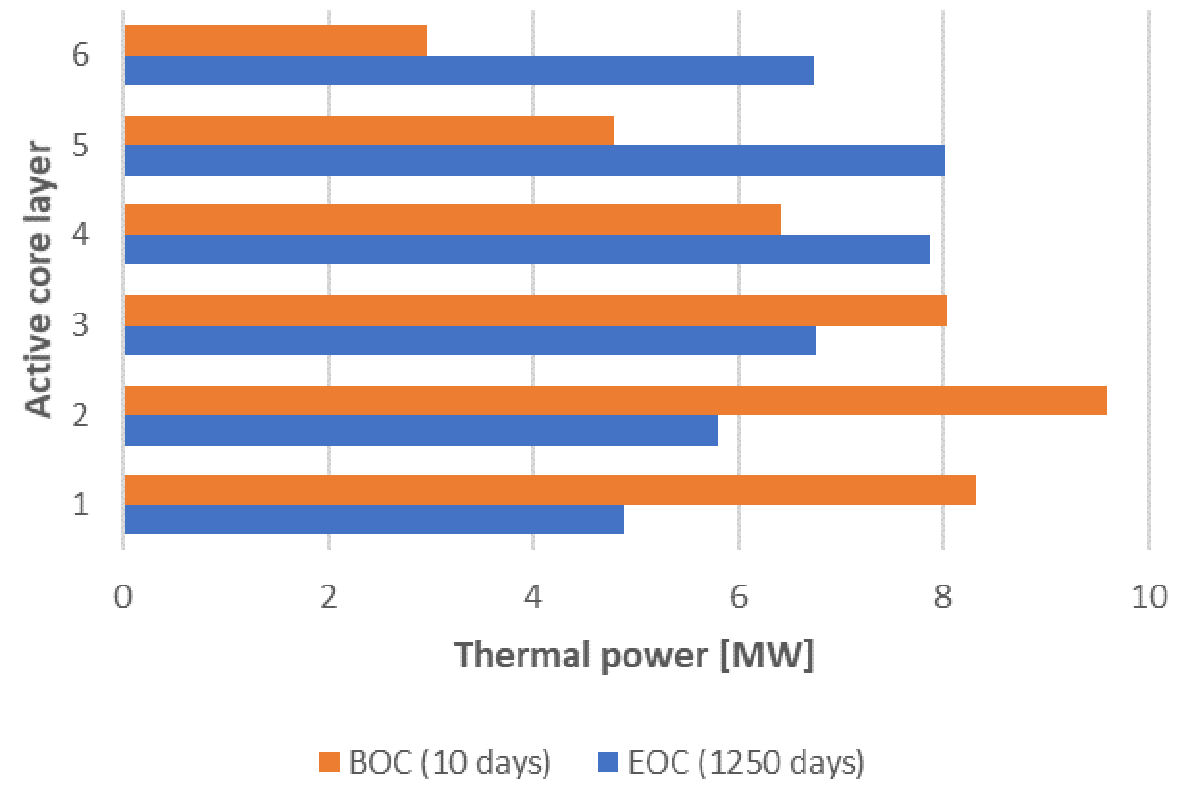


Fig. BOC and EOC axial power profile comparison for research reactor.

Neutronic investigations: Serpent 2

- Capability to perform automated depletion zone division for materials in burnup/activation calculation:
  - division of the geometry into sub-volumes which during depletion are considered as a separate ones.
  - the differences in local flux and spectrum are taken into account,
  - used to divide the fuel material, burnable poison rods, as well as the graphite in fuel and reflector blocks.
- constant power operation mode for 550 days both for BP and fuel with fuel matrix, one full cycle for replaceable reflector and permanent reflector zone.
- Capability of the universal multi-physics interface - used to converge both power and temperature distributions at BOL with unrodded core by means of data exchange between Serpent 2 and MELCOR 2.2.14.
  - the power distribution affects the temperature profile, which in turn influences the power distribution through changes in the effective neutron cross sections.
  - this capability allows to map temperature field over the whole geometry and assigns temperature to different materials.
  - four iterations were sufficient to get converged power and temperature distributions.

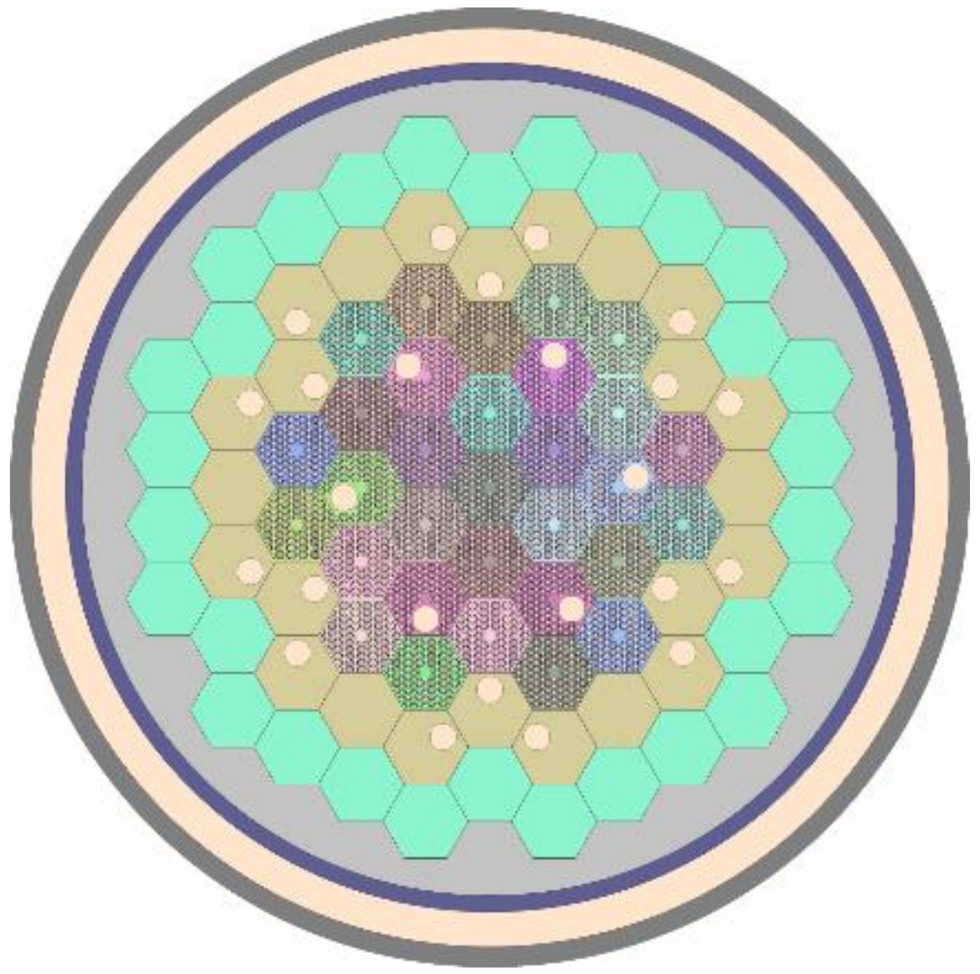


Fig. Example of automatic material division in Serpent model. Division of graphite into material zones in blocks. Different colors in graphite blocks mean different material in depletion sequence.

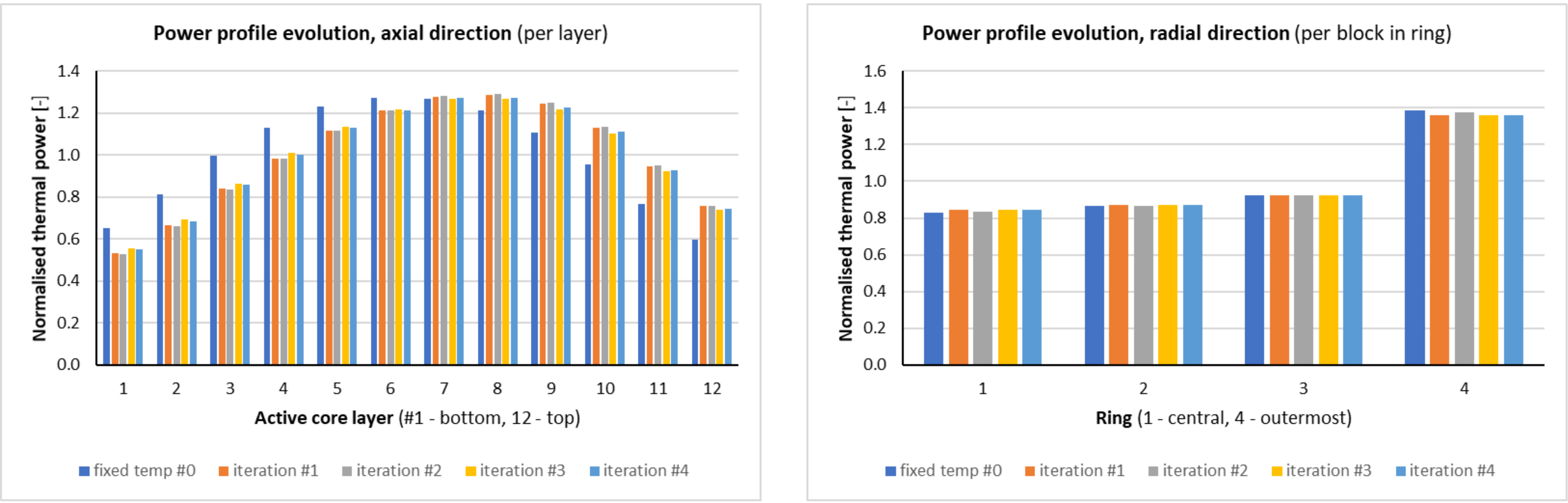


Fig. Successive iterations of the axial and radial power distributions, BOL.

Thermal-hydraulic investigations: MELCOR code

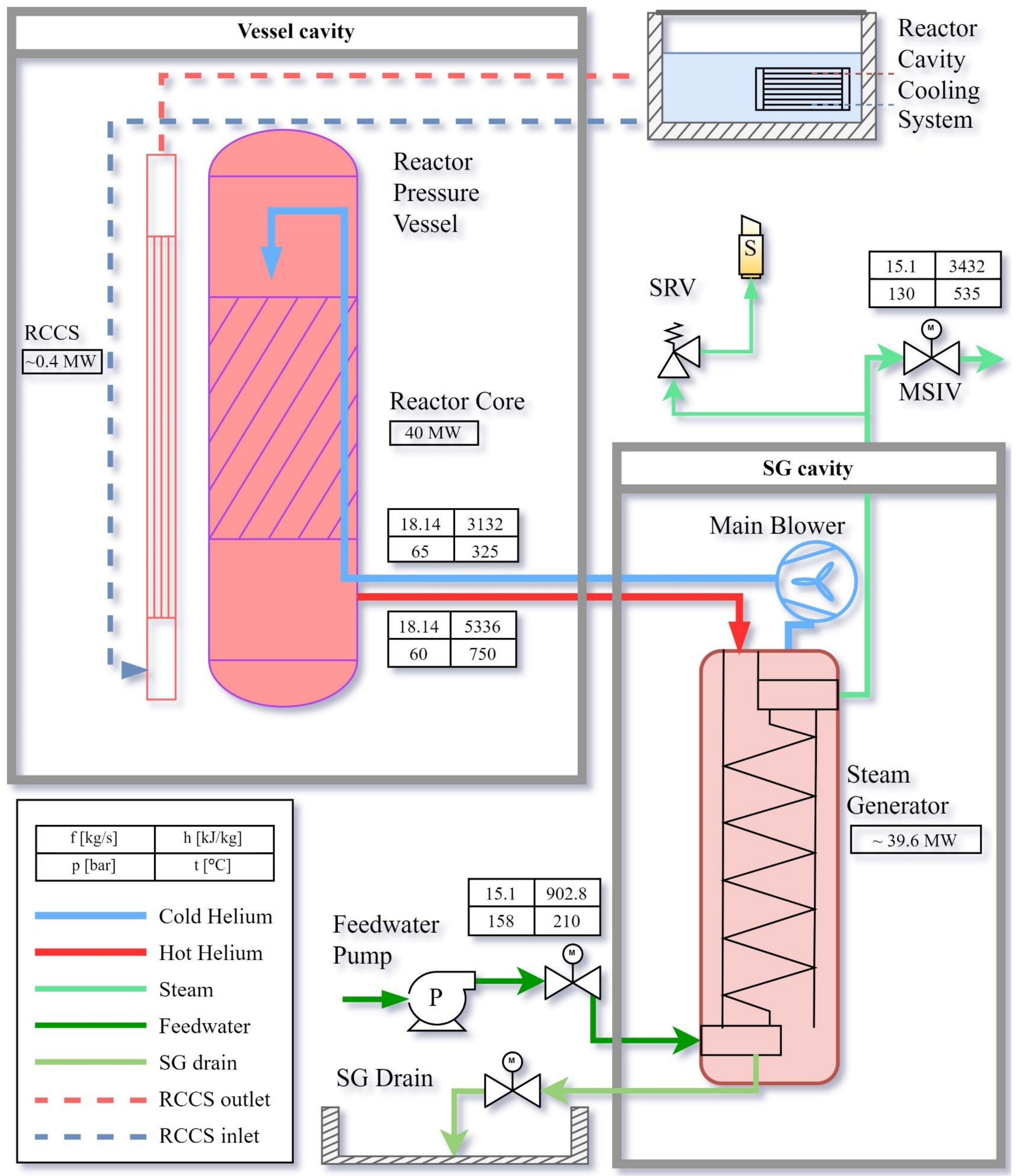


Fig. The system overview analyzed in MELCOR 2.X code

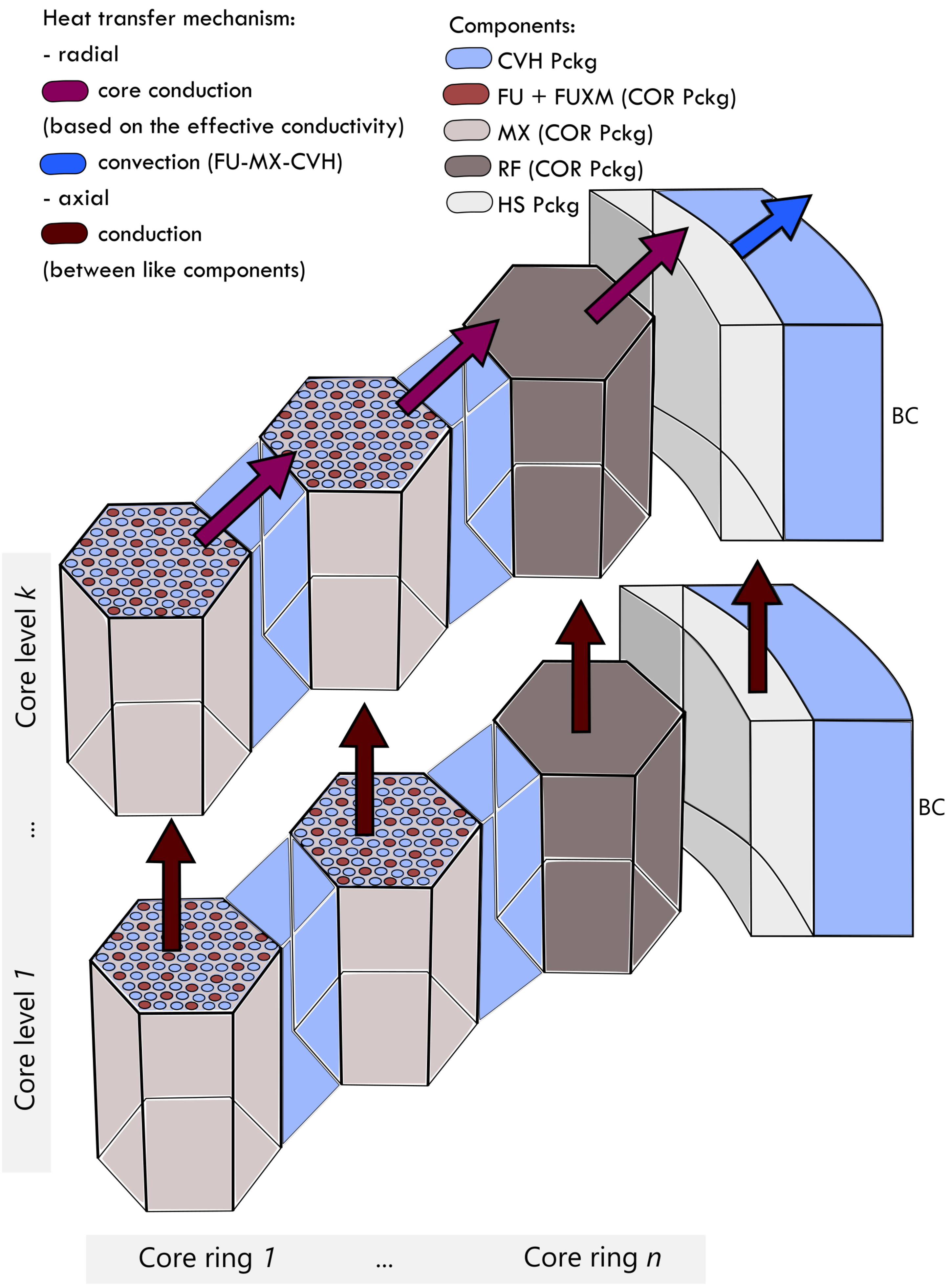


Fig. MELCOR code core region computational domain for the COR package

Fig. (most left) - depicts the scheme of the main components analyzed under the system evaluations with the use of MELCOR code.

The Fig. (left) - two directions of heat transfer in the core (COR package by the MELCOR code) - radially and axially.

For the HTGRs, especially the PMR, the conduction heat transfer mechanism plays crucial role:

Radial conduction (internal and between blocks) and convection are distinguished. In fact, one arrow may suggest an exchange only between blocks, we will also detail it.

Internal:

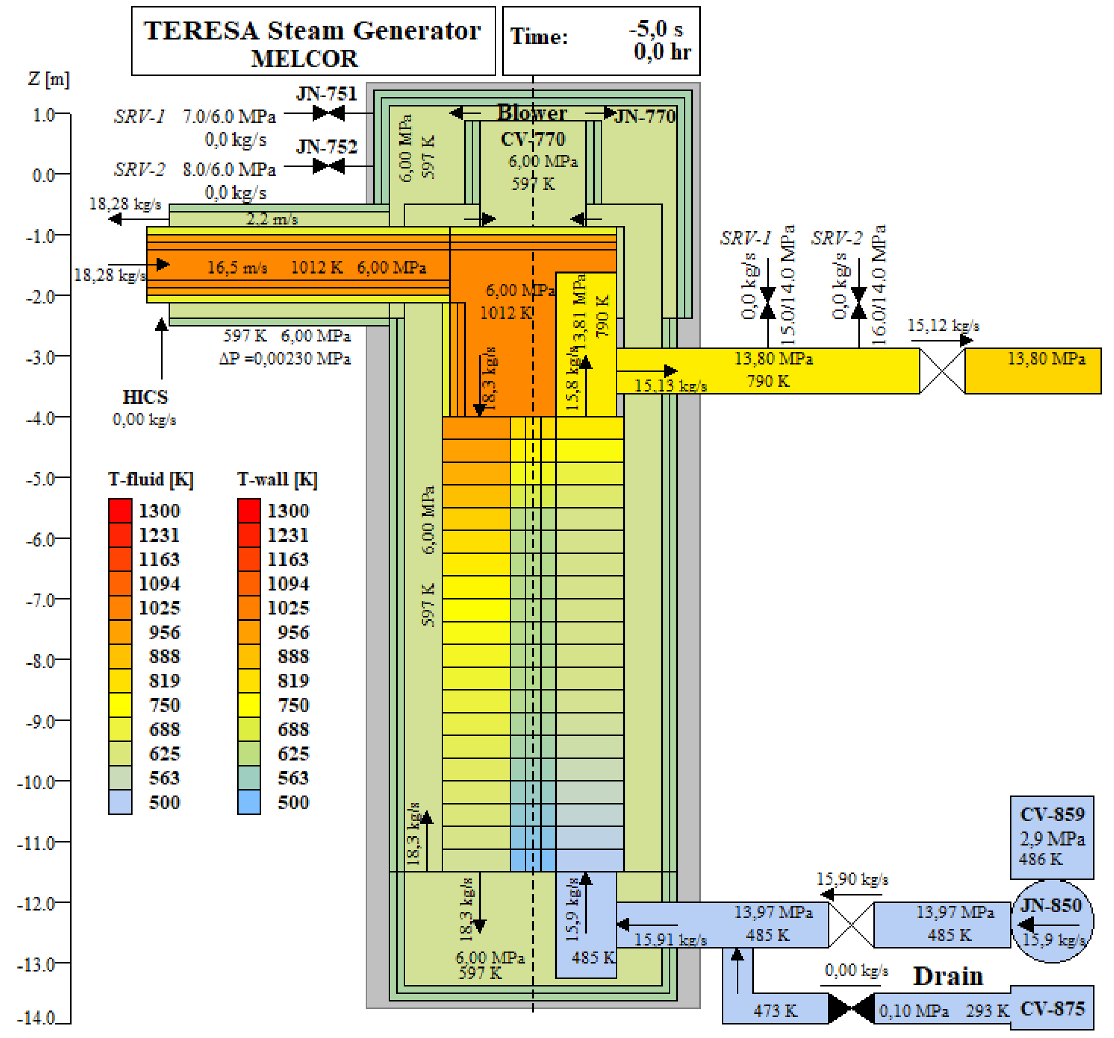
- Using a cylindrical geometry - fuel/graphite block,
- thickness of the block is accounted for - the temperature distribution is not linear, calculated from the solution of the heat conduction equation in the space of 1D cylindrical variables and presented as a logarithmic distribution (steady state equation of heat diffusion in the form of the Laplace equation)

Between blocks (between cells):

- In MELCOR, a radial conduction model dedicated to HTGR, which includes the effective conductivity of graphite blocks by applying the Tanaka-Chisaka correlation for a continuous solid medium.
- The expression takes into account the discontinuity of the domain (the presence of flow channels treated as domain pores), which can significantly affect the conduction in the block.

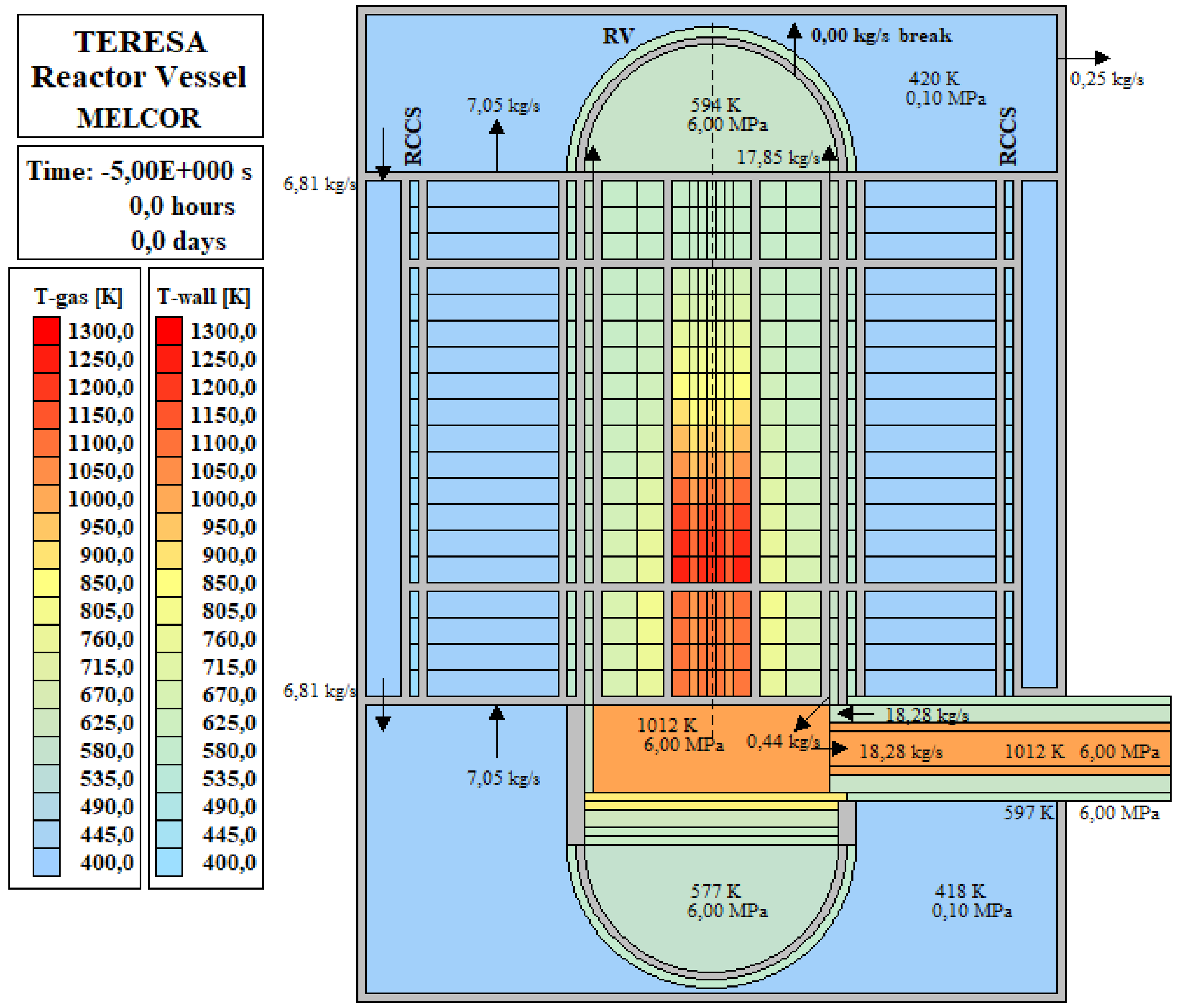
Thermal-hydraulic investigations (GEMINI+ and pre-conceptual research HTGR)

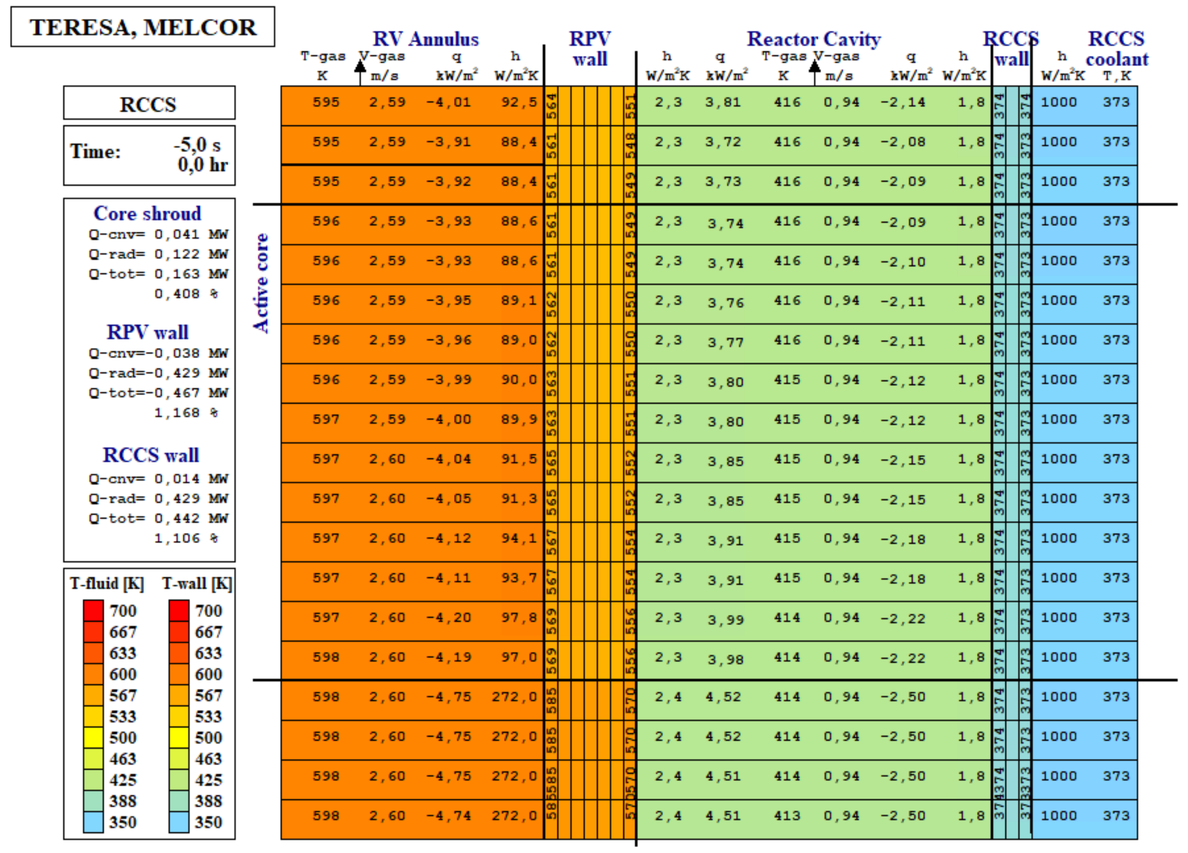
1. Used tools: MELCOR 2.2 is a fully integrated, engineering level computer code that models the progression of severe accidents in nuclear power plants.
   - ❑ For HTGR the MELCOR framework is adopted and code is using specific modules – HTGR dedicated. It was extensively used during GEMINI Plus project [13], [14] and during the investigations it was found the version 2.2.14 had some limitations (updated to more recent version of the code 2.2.21 advised).
2. The developed MELCOR 2.2 model consist of the primary and secondary side of the pre-conceptual research reactor. The main modelled components and plant systems are as follows:
   - ❑ Reactor Pressure Vessel (RPV) with core.
   - ❑ Steam Generator (SG) and helium Blower.
   - ❑ Reactor Cavity.
   - ❑ Safety Systems.
   - ❑ Reactor Core Cooling System in the Reactor Cavity.



Tab. MELCOR 2.2 code model qualification table at “steady-state” level. [4]

Parameters	Design Value	Simulation MELCOR 2.2	Relative Error
Reactor power (MW_{th})	40.0	40.0	0.000%
Helium pressure of primary loop (MPa)	6.0	6.0	0.000%
Helium mass flow rate (kg/s)	18.14	18.14	0.000%
Helium RPV Inlet temperature (K)	598.0	598.0	0.000%
Helium RPV Outlet temperature (K)	1023.0	1015.0	−0.782%
Main feed-water temperature (K)	483.0	482.5	0.000%
Main steam temperature (K)	813.0	794.0	−3.519%
Main steam pressure (MPa)	13.8	13.8	0.000%
Feed-water flow rate for steam generator (kg/s)	15.9	15.95	0.314%





Thermal-hydraulic investigations (GEMINI+ and pre-conceptual research HTGR)

Thermal-hydraulic investigations

- MELCOR 2.1 and SPECTRA (NRG developed t-h code) comparison for the GEMINI+ system.
- Same nodalization scheme – user effect elimination.

DLOFC sceario analysis:

- some differences in the calculation results for both codes, although the results are coherent comparing them between the assumptions (BE and C). The differences in the maximum fuel temperatures are 146 K in the BE case and 191 K in the conservative case
- At time of accident of around 28 h for both assumptions (BE and C) for the SPECTRA code the established ratio of power extracted from the RPV by convection to radiation was around 0.205, while for the MELCOR code it was 0.1235. This could be influenced by both different heat transfer correlations for the convective heat exchange used in the codes, or different RPV wall temperatures evaluations.

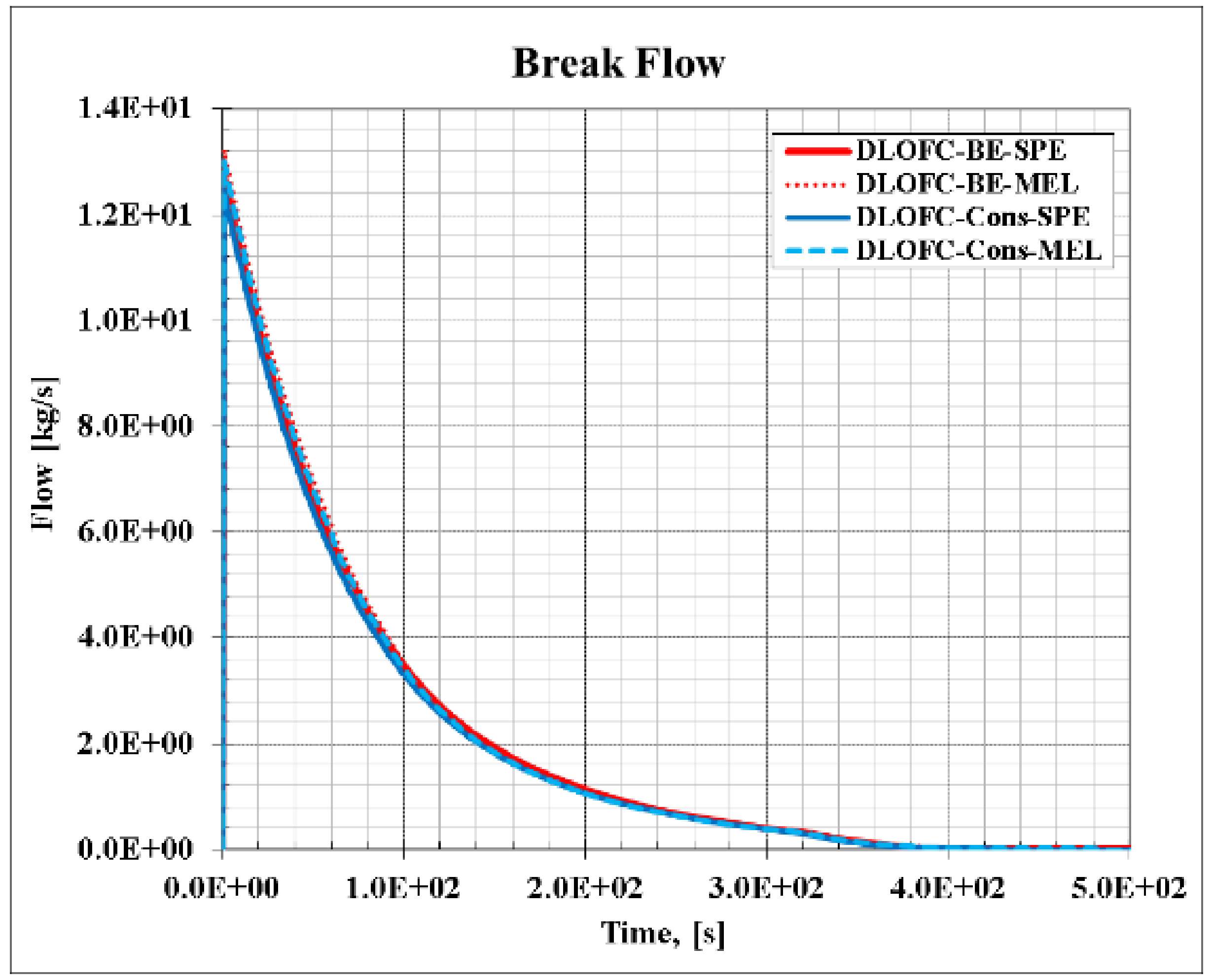


Fig. DLOFC break flow

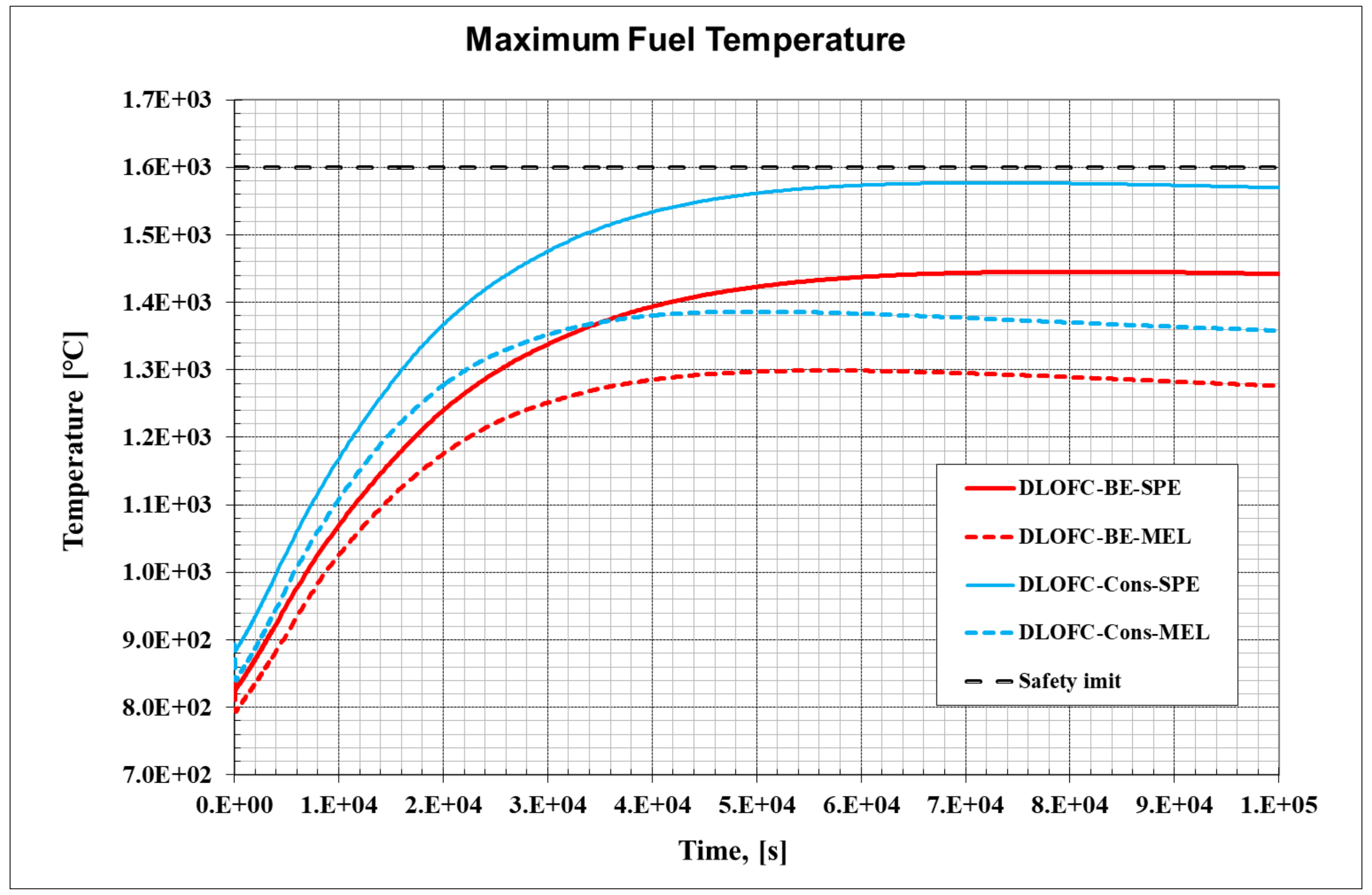


Fig. DLOFC maximum fuel temperatures

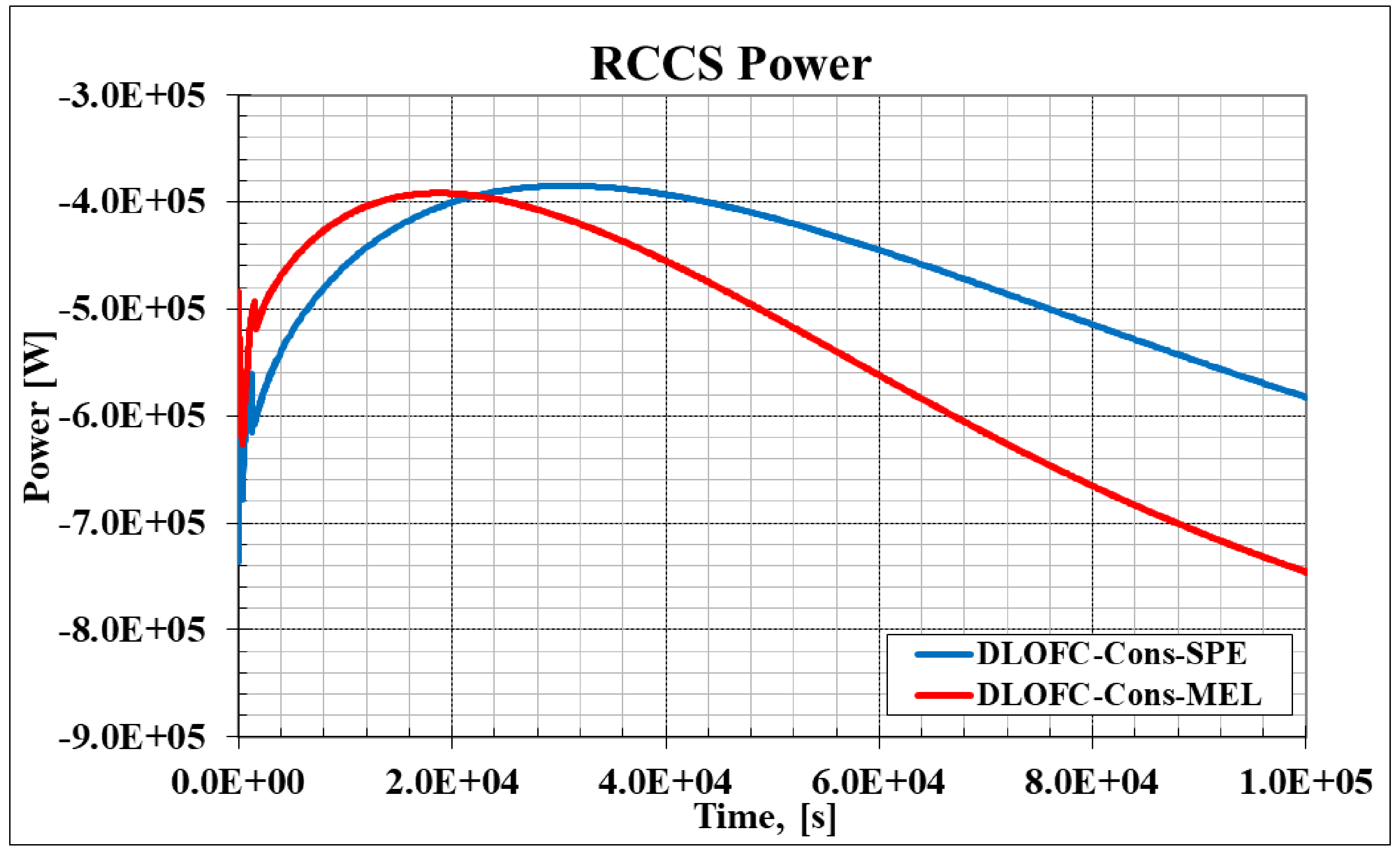


Fig. DLOFC (C) RCCS power

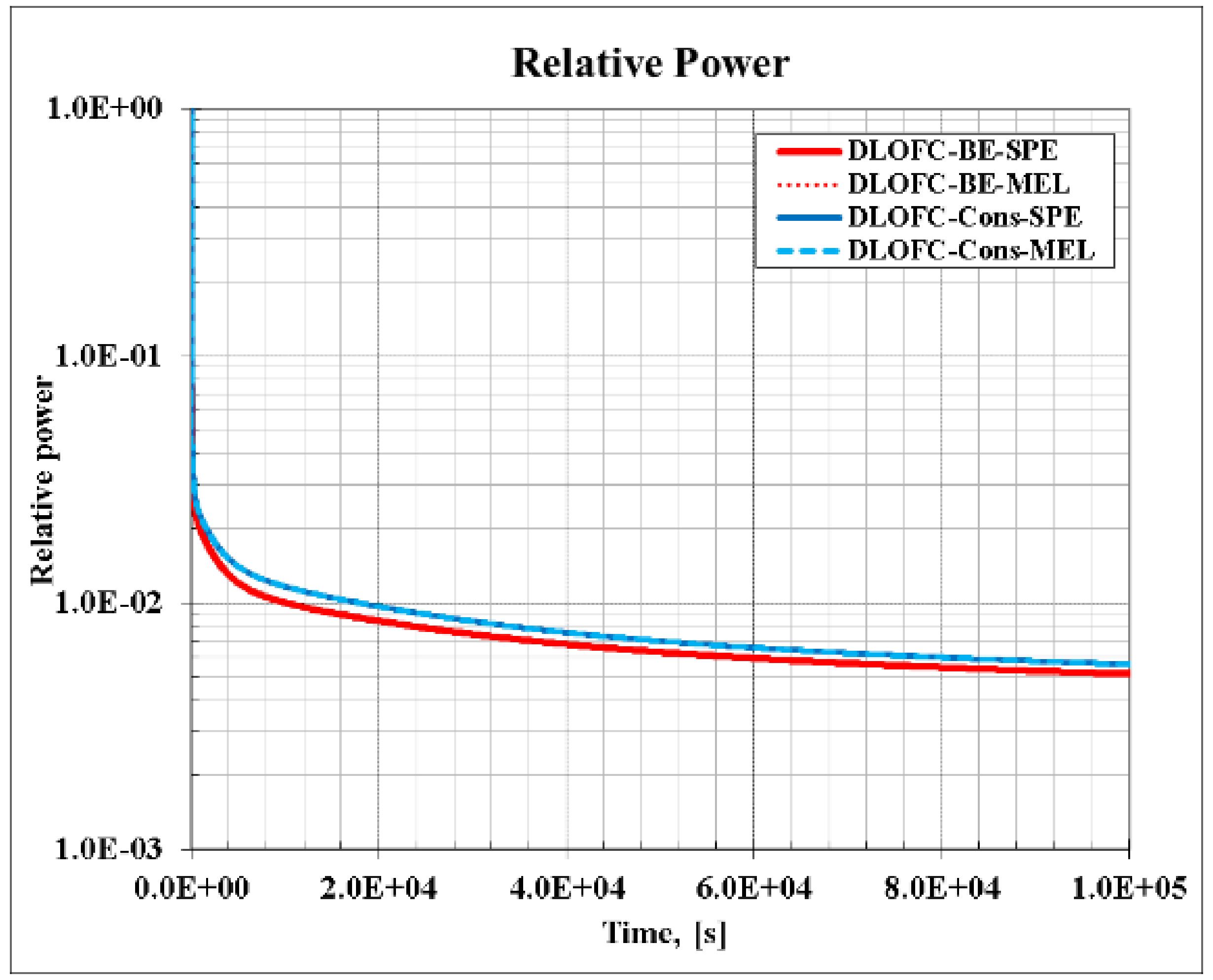


Fig. DLOFC relative power – long term

Thermal-hydraulic investigations (GEMINI+ Air Ingress scenario)

- MELCOR and SPECTRA calculations of the DBA air ingress scenario were performed for single and double flow path break declaration. [34]
- A very good agreement of the break flow and the primary pressure is observed during the blowdown phase for both single and double flow path option. The primary system volumes are very similar in both models, so the depressurization proceeds in a very similar way.
- short term velocity is very good. In conclusion, a very good agreement is observed in the short term, during the blowdown period.
- The MELCOR calculated velocity show large oscillations. It is seen that the gas velocity in the break oscillates around zero with values between roughly −2 m/s and +3 m/s.
- oscillatory flow behavior that could not be eliminated in both MELCOR runs.

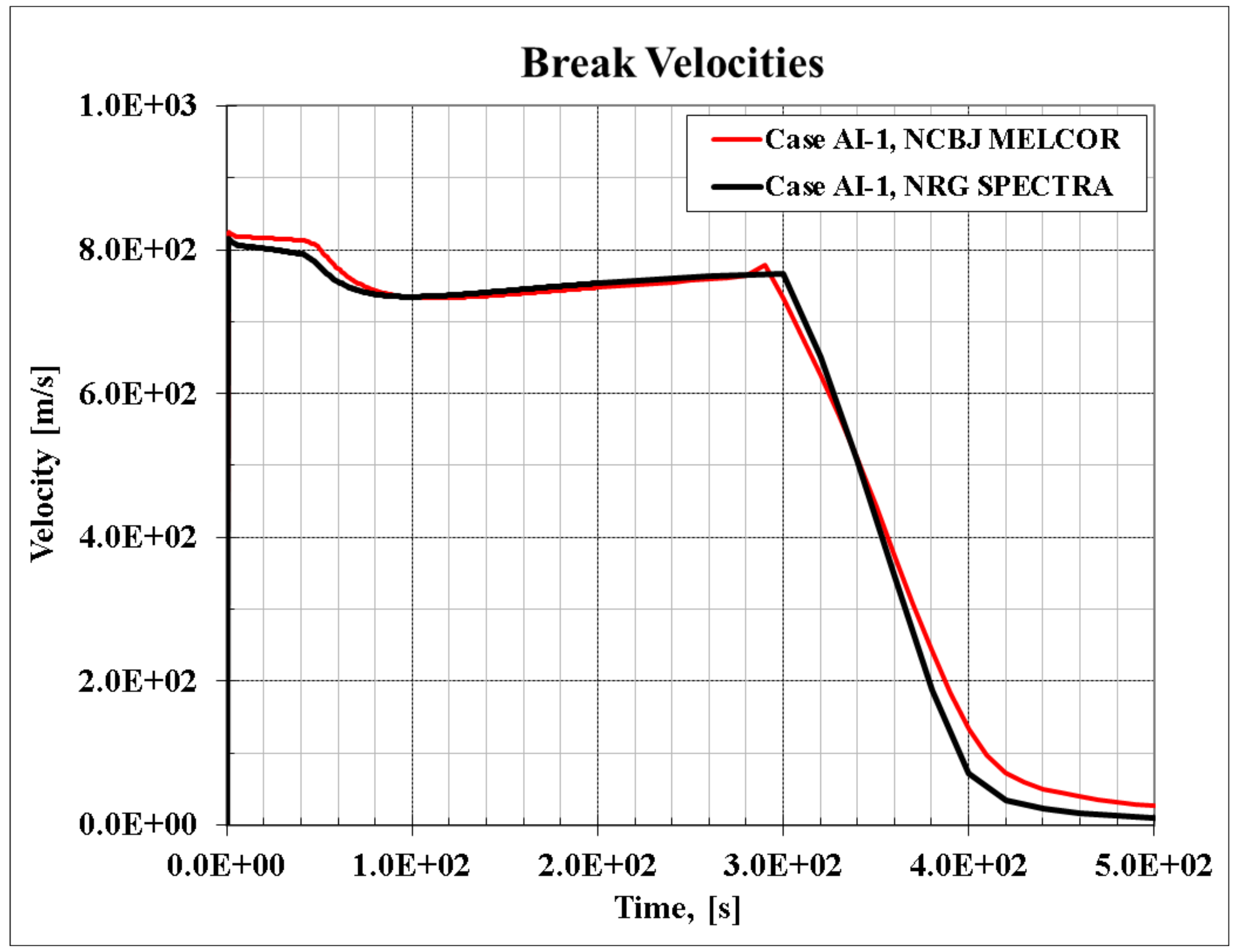


Fig. Break velocity, short term - DBA AI, break as single flow path

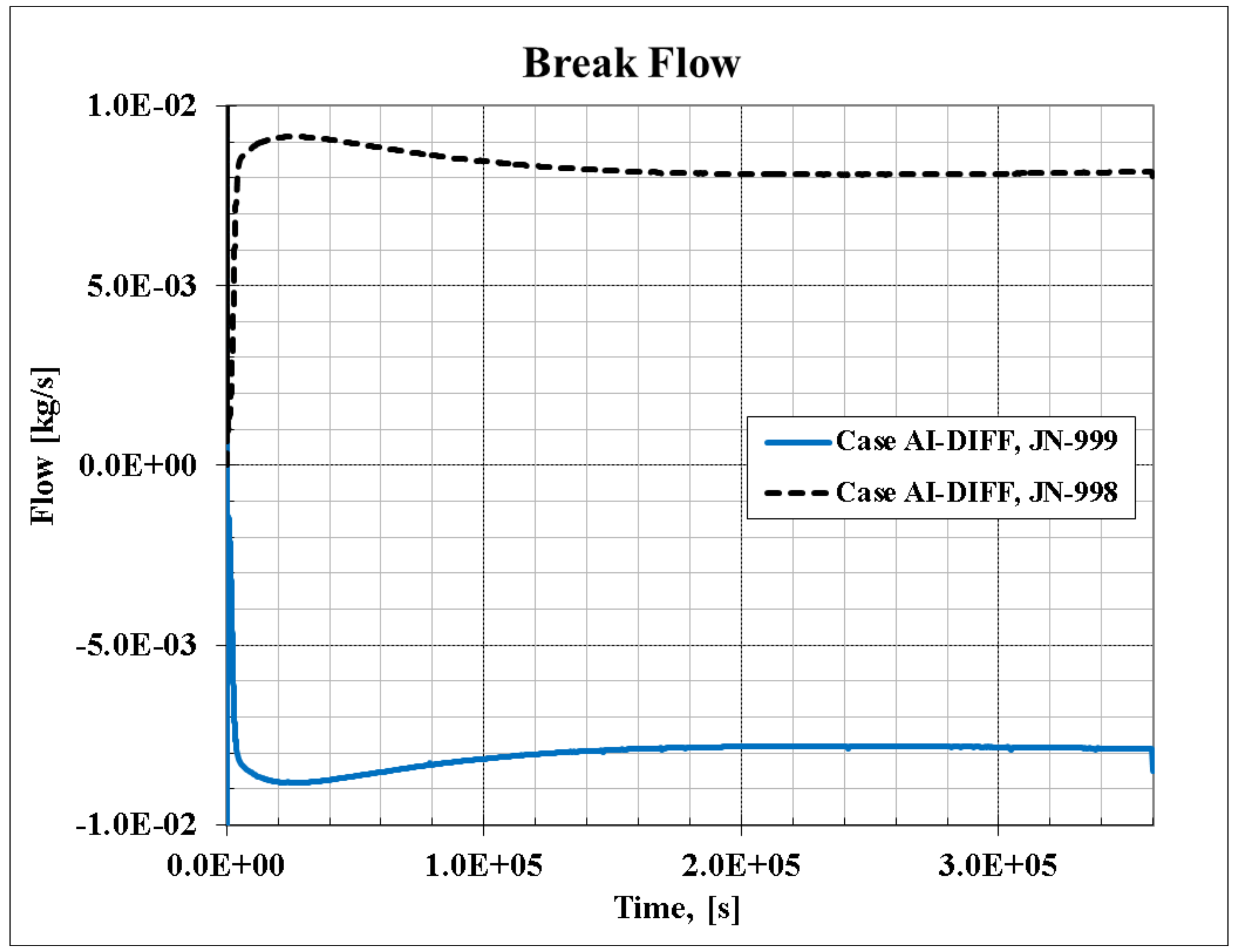


Fig. SPECTRA mass flow in the break, long term.

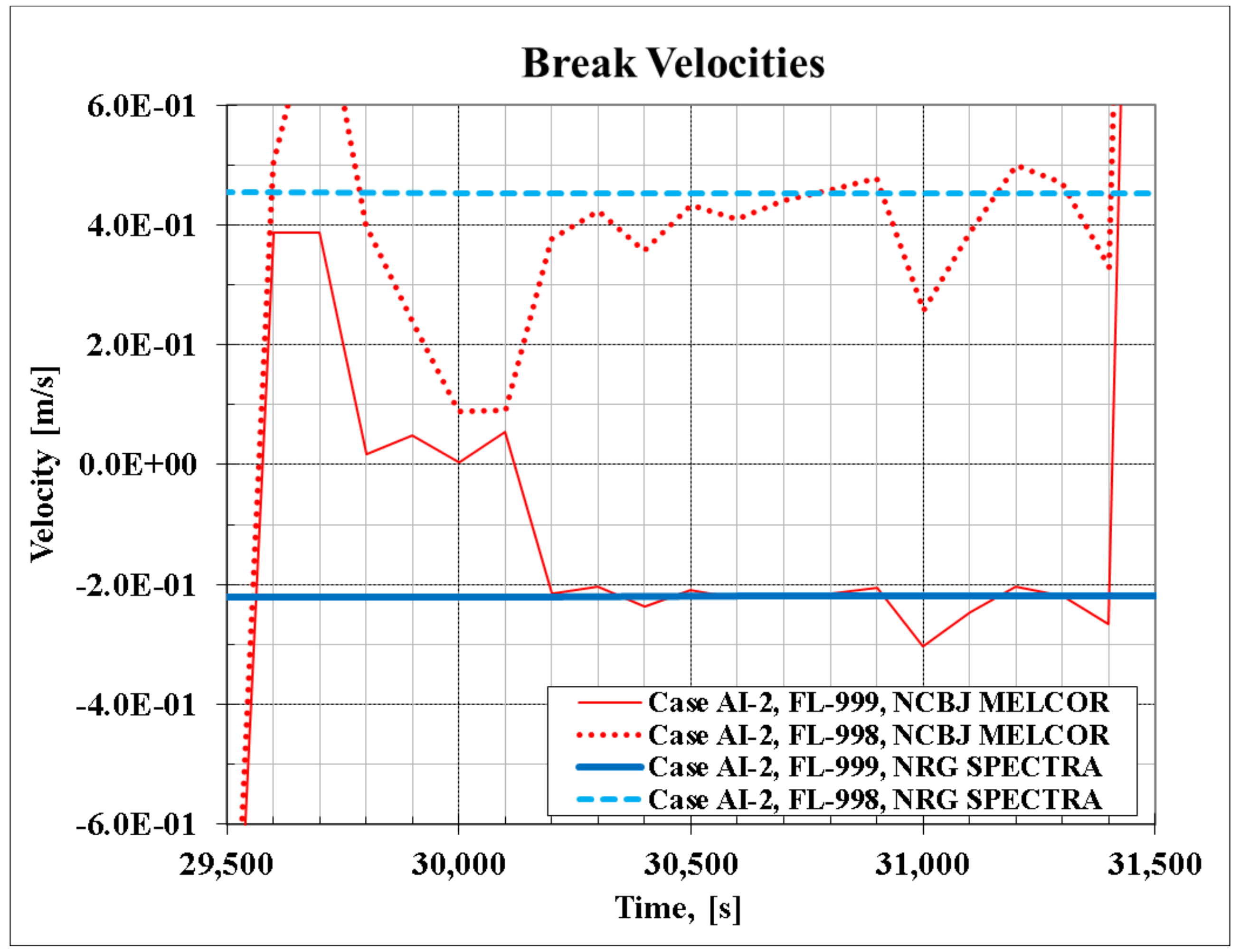


Fig. Break velocity - DBA AI, break as double flow path, time ~30,000 s.

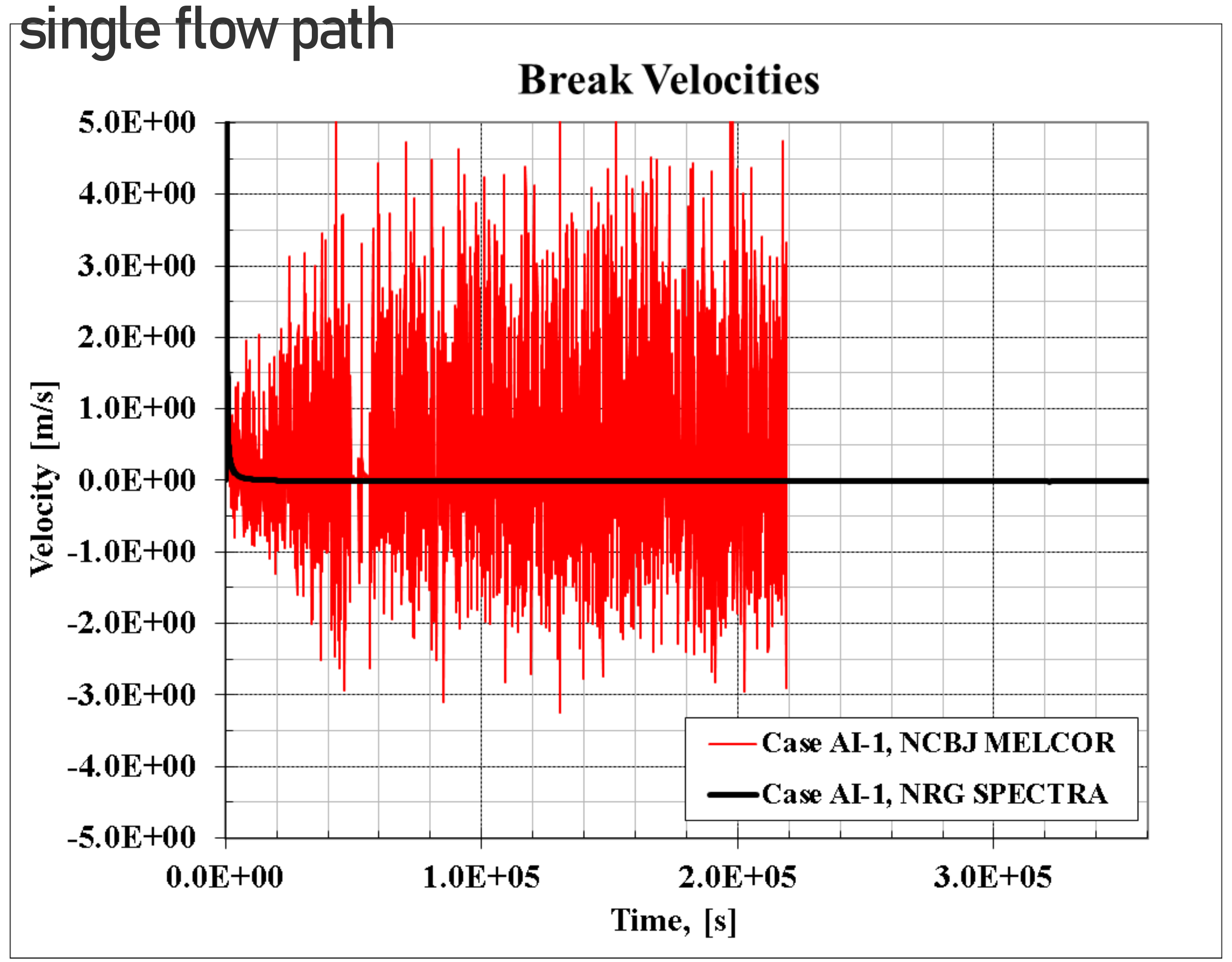


Fig. Break velocity, long term - DBA AI, break as single flow path.

Thermal-hydraulic investigations – DBA (1/3)

- The model was used to simulate the following accidents (from the Design Basis Accidents list established under the GEMINI Plus project [7]):
  - DLOFC - loss of the system integrity with loss of forced flow of the coolant - assumption of the best estimate (BE).
  - DLOFC (without RCCS system) - loss the system integrity with loss of forced flow of coolant - best estimate assumption - without heat removal by RCCS.
  - PLOFC - loss of forced flow of coolant while maintaining integrity of the system - assumption of the best estimate.
  - Water Ingress - loss of tightness of the primary and secondary sides inside the steam generator - assumption of the best approximations.

The Best Estimate calculations is based on the following assumptions:

- Reactor power: 100%.
- Nominal temperature at the inlet to the pressure vessel.
- Nominal temperature at the outlet to the pressure vessel.
- Emission factor between RV / RCCS: 0.9.
- Natural convection (multiplier), default: 1.0.
- Default decay heat curve – calculated by the SPECTRA and Serpent exchange in GEMINI Plus project [13].

Tab. Core maximal temperatures for the pre-conceptual RR core at BOL state for DLOFC (BE), PLOFC and WI accident scenarios.

Accidental scenario	Maximum fuel temperature	Active core axial position from the bottom	Active core radial position	Time of occurrence
-	[K]	[m]	[-]	[s]
DLOFC	1261.0	~1.0	central column	25 400
PLOFC	1050.0	~4.2	central column	10 000
WI	1185.0	~0.2	4^{th} ring	2 500

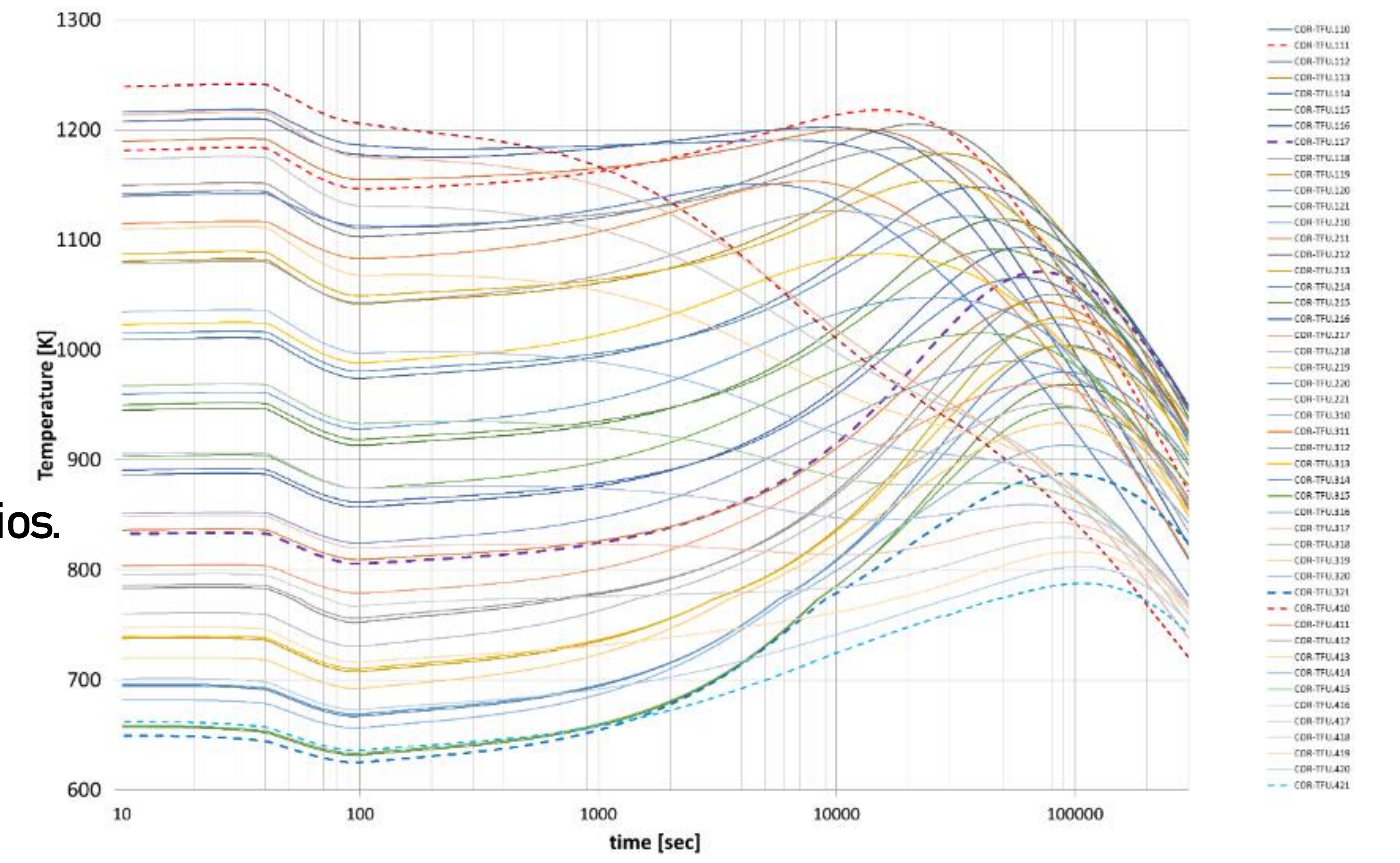


FIg. Time evolution of fuel temperatures for all core computational cells in DLOFC accident scenario - logarithmic scale for time.

Thermal-hydraulic investigations – DBA (2/3)

DLOFC - performed for BE . Loss of forced coolant flow with 65 mm rupture (equivalent diameter) on the Reactor Pressure Vessel head cover, assuming BE. The timeline of the accident event is as follows:

- t = 0 s - pipe break.
  - Coolant flow decreases, pressure drops, core outlet temperature increases.
  - The reactor power is slowly decreasing due to the negative temperature coefficient.
- t = 40 s - SCRAM signal appears due to activation of one of 4 logic signals (pressure difference, core outlet temperature, low primary/secondary flow ratio, high core power)
- t = 41 s - SCRAM
  - Control rods drop in 10s providing -10 $ negative reactivity insertion.
- t = 60 s - stopping the main feedwater pumps and FW isolation,
- t = 60 s - primary blower stops (impeller rotation is assumed to drop to zero in 30 s),
  - Long-term heat is removed by the RCCS system, cooling the reactor cavity and the pressure vessel.

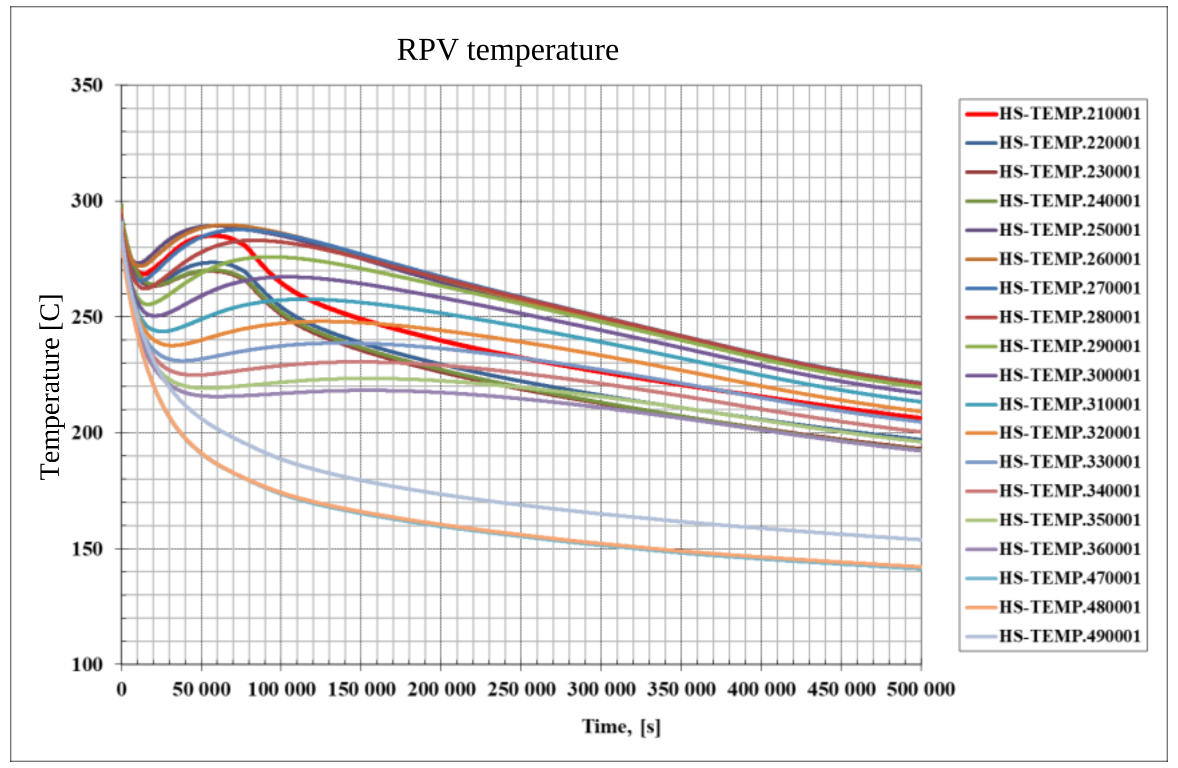


Fig. Temperature in the RPV layers during DLOFC accident.

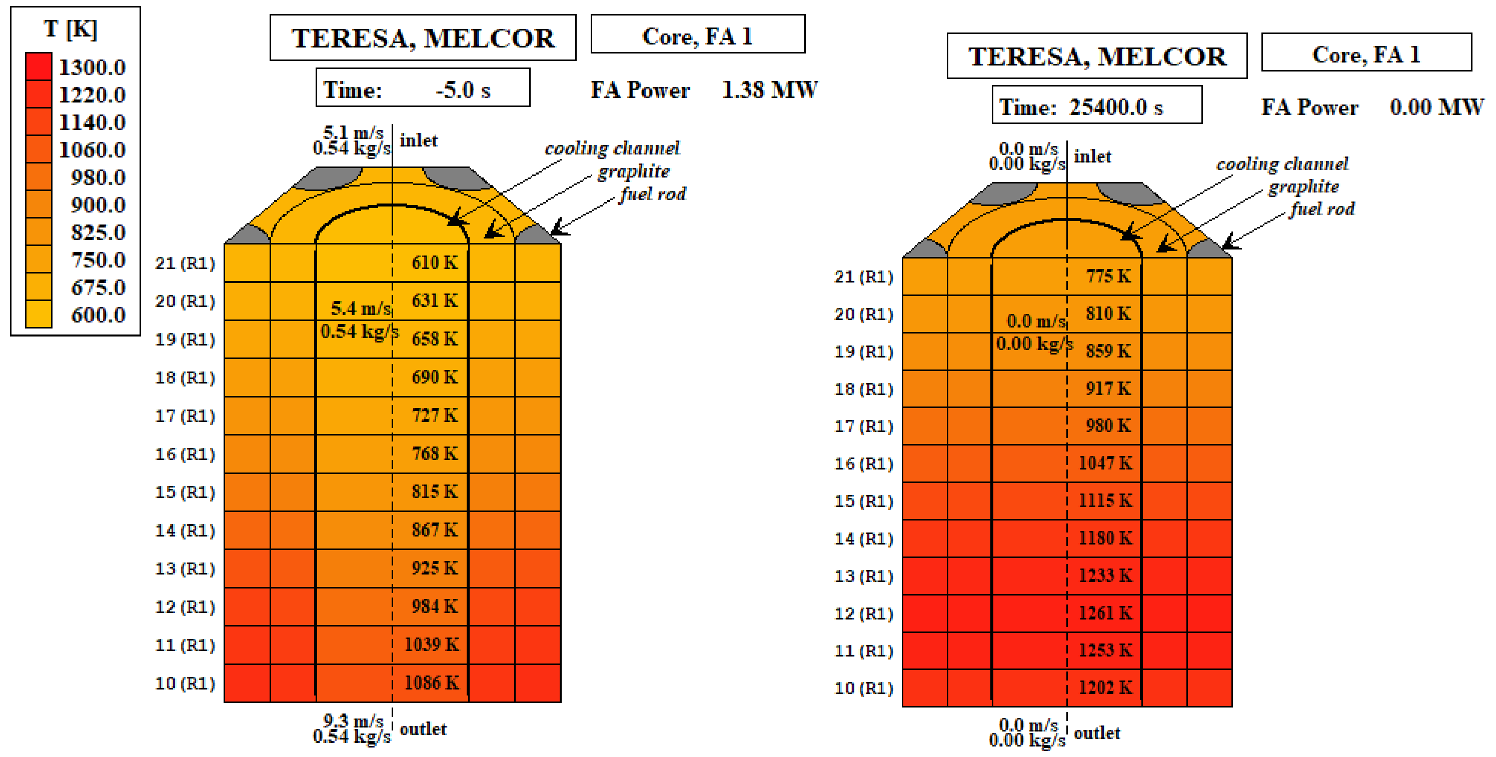


Fig. Temperature in the cooling channel of central block of the core. Left: steady state (5 s before accident initialization); Right: DLOFC conditions, Tmax at t=25 400 s. (BOL case).

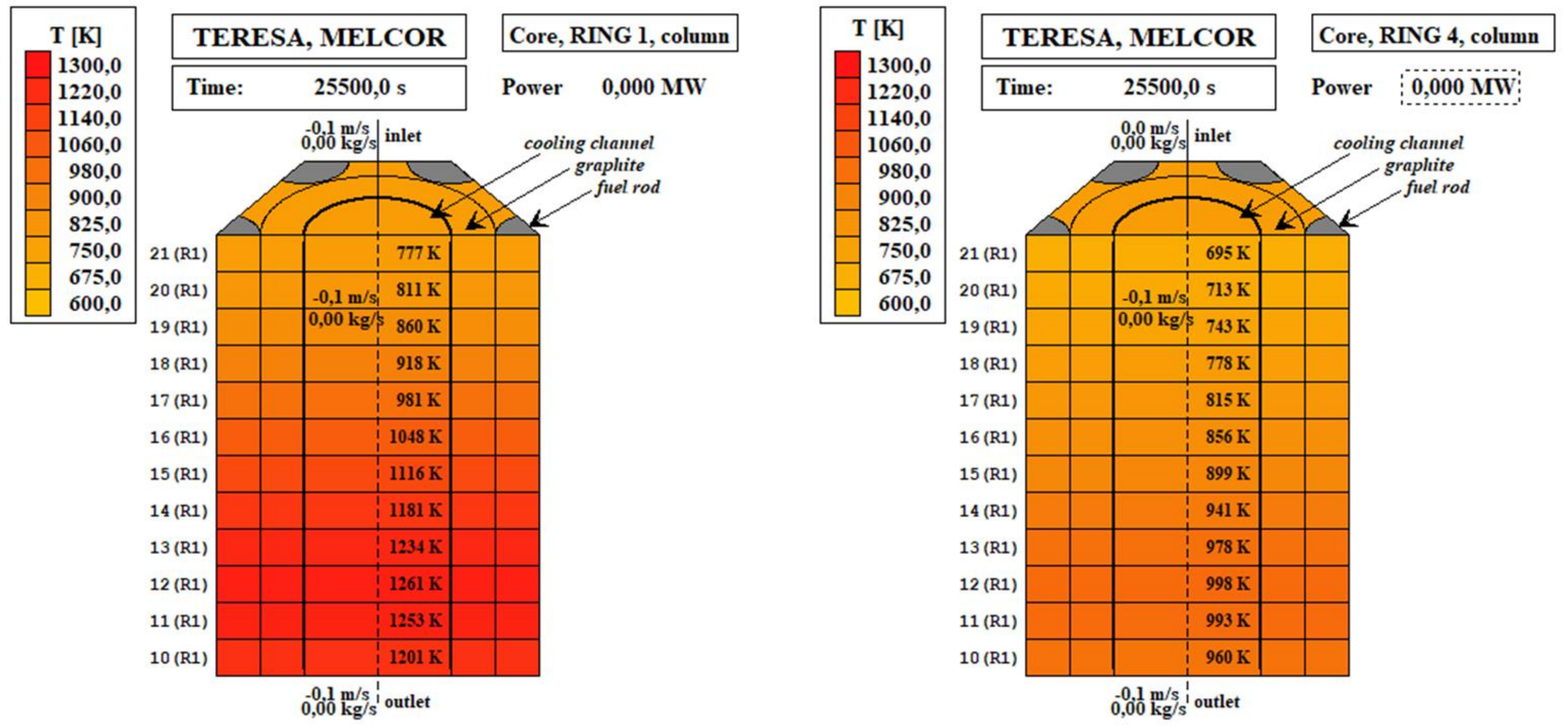
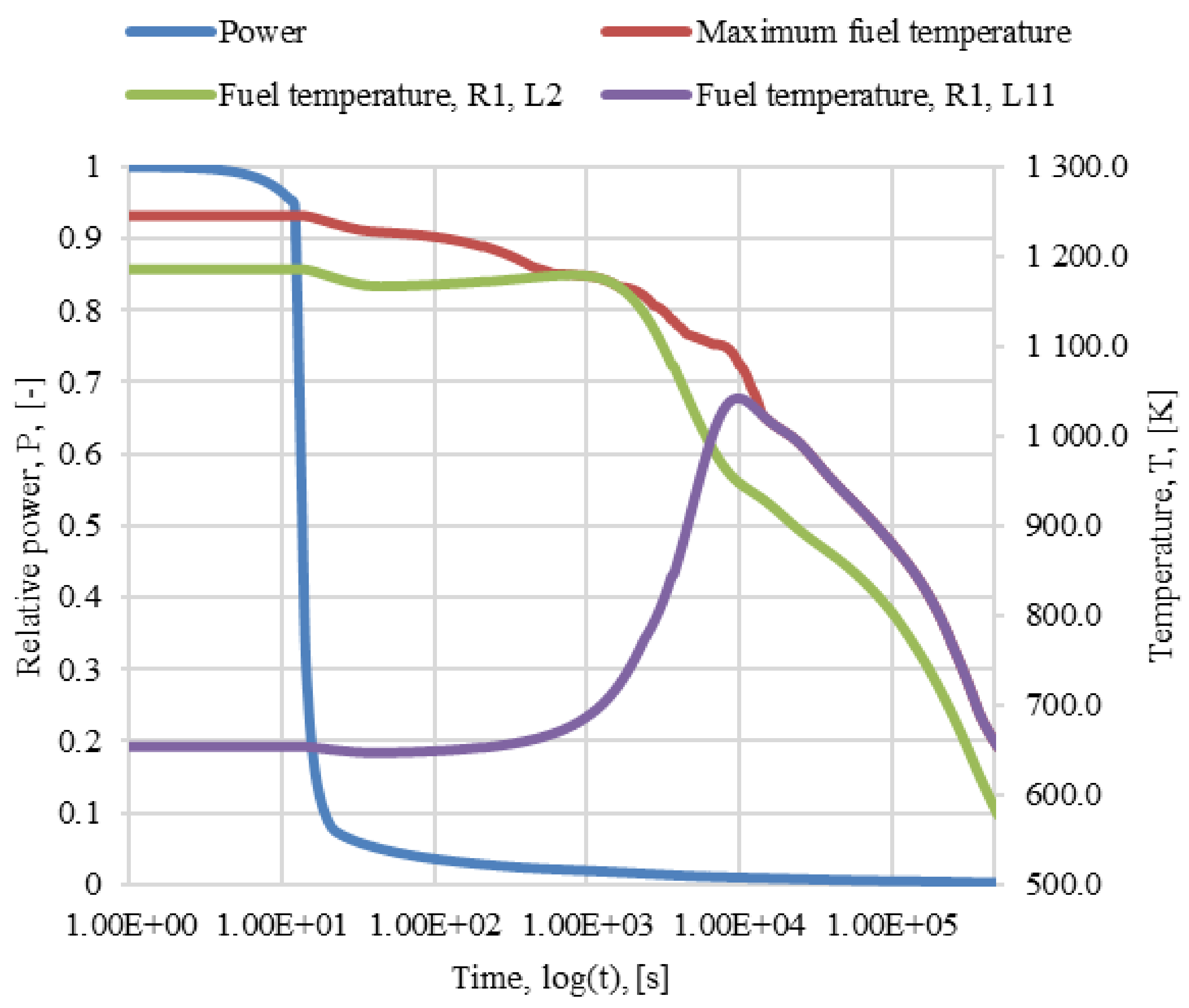


Fig. Temperature in the cooling channel of central block and block in 4th ring at T_{max} time.

Thermal-hydraulic investigations – DBA (3/3)



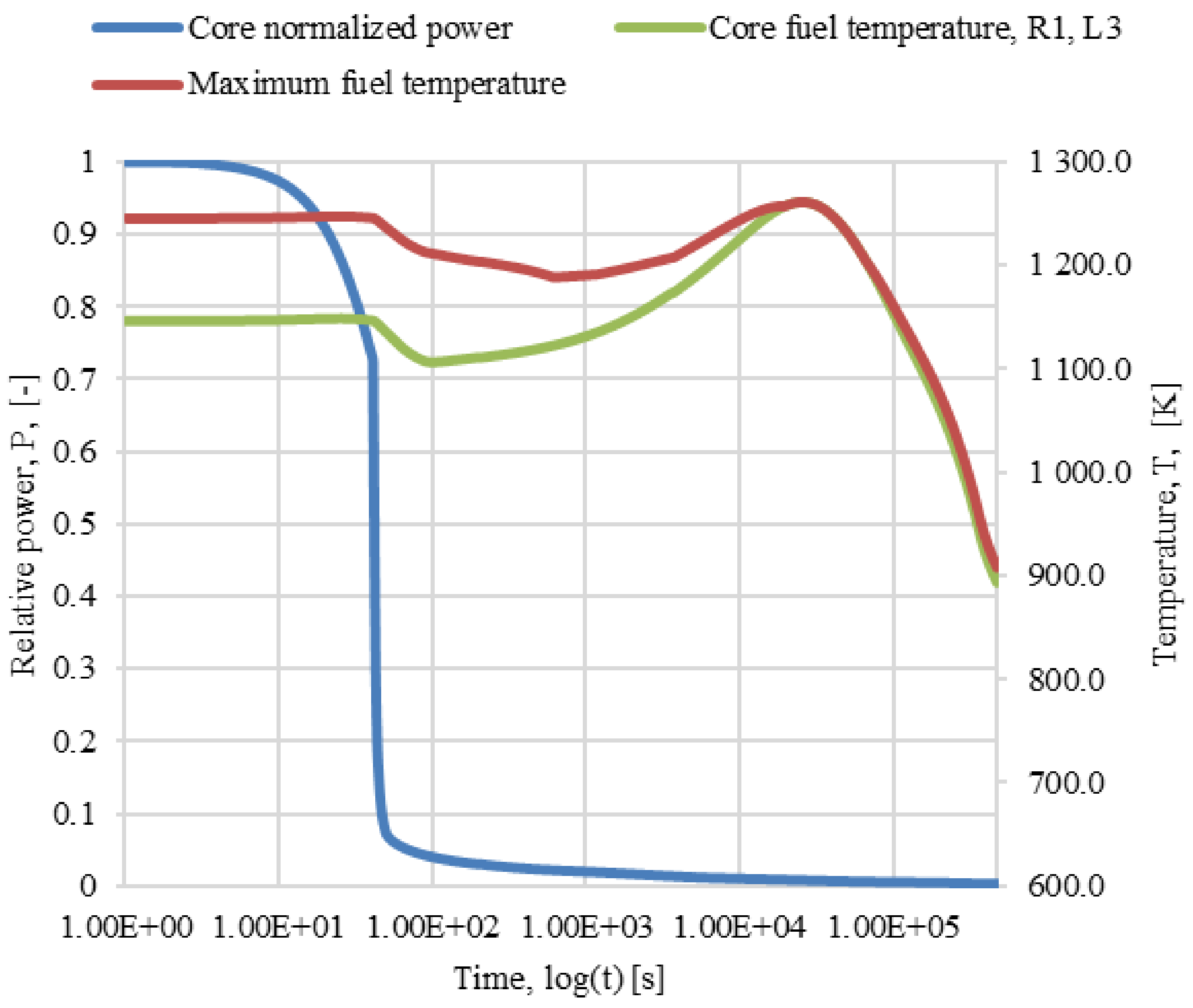


Fig. DLOFC BE (left) and PLOFC (right) accident core normalized power (P, [-]) and fuel maximum temperatures (T_f, [K]).

- The maximum temperatures for each scenario did not exceed the 1600 °C (the temperature limit of increased fission products (FP) release rate).
- Depending on the scenario the maximum temperature position for the Beginning of Life (BOL) power distribution core was found in various positions and accident times.
- Most challenging scenario DLOFC – is characterized by highest temperatures slightly exceeding the steady state ones for the BOL core – low power density (~2.36 MW/m3)

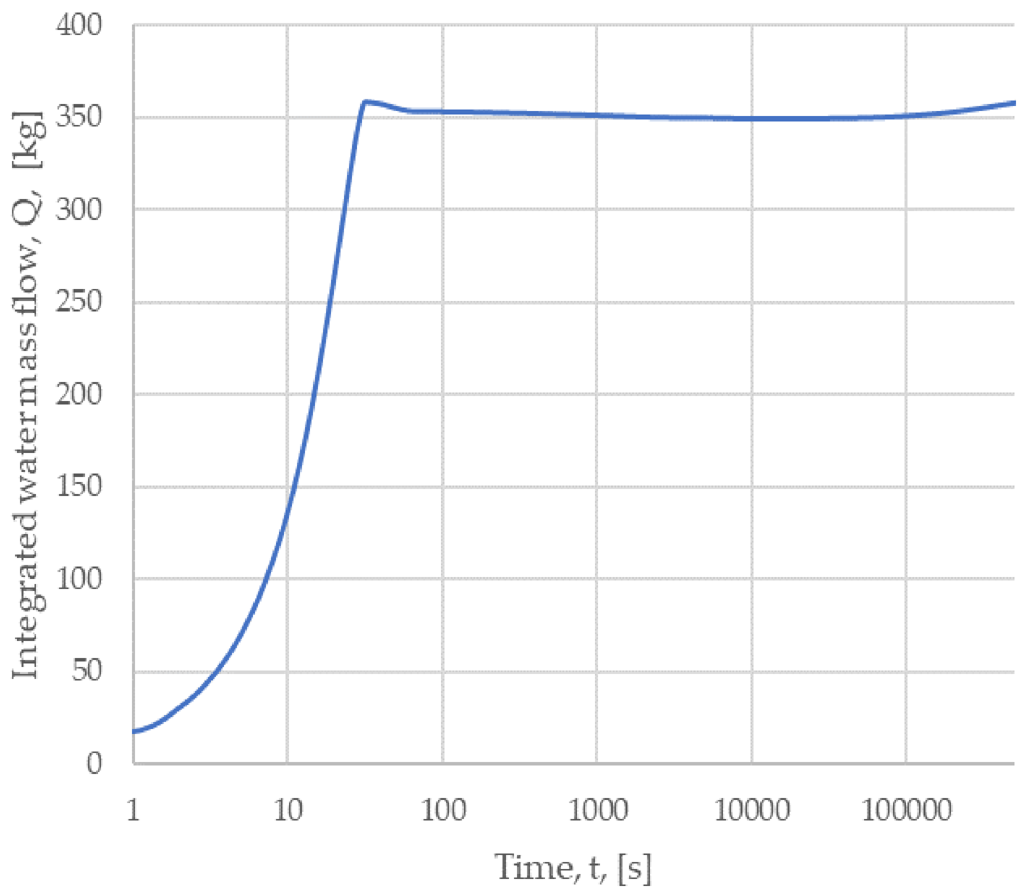


Fig. Water Ingress Accident integrated break water mass flow for 500 000s period.

Water Ingress:

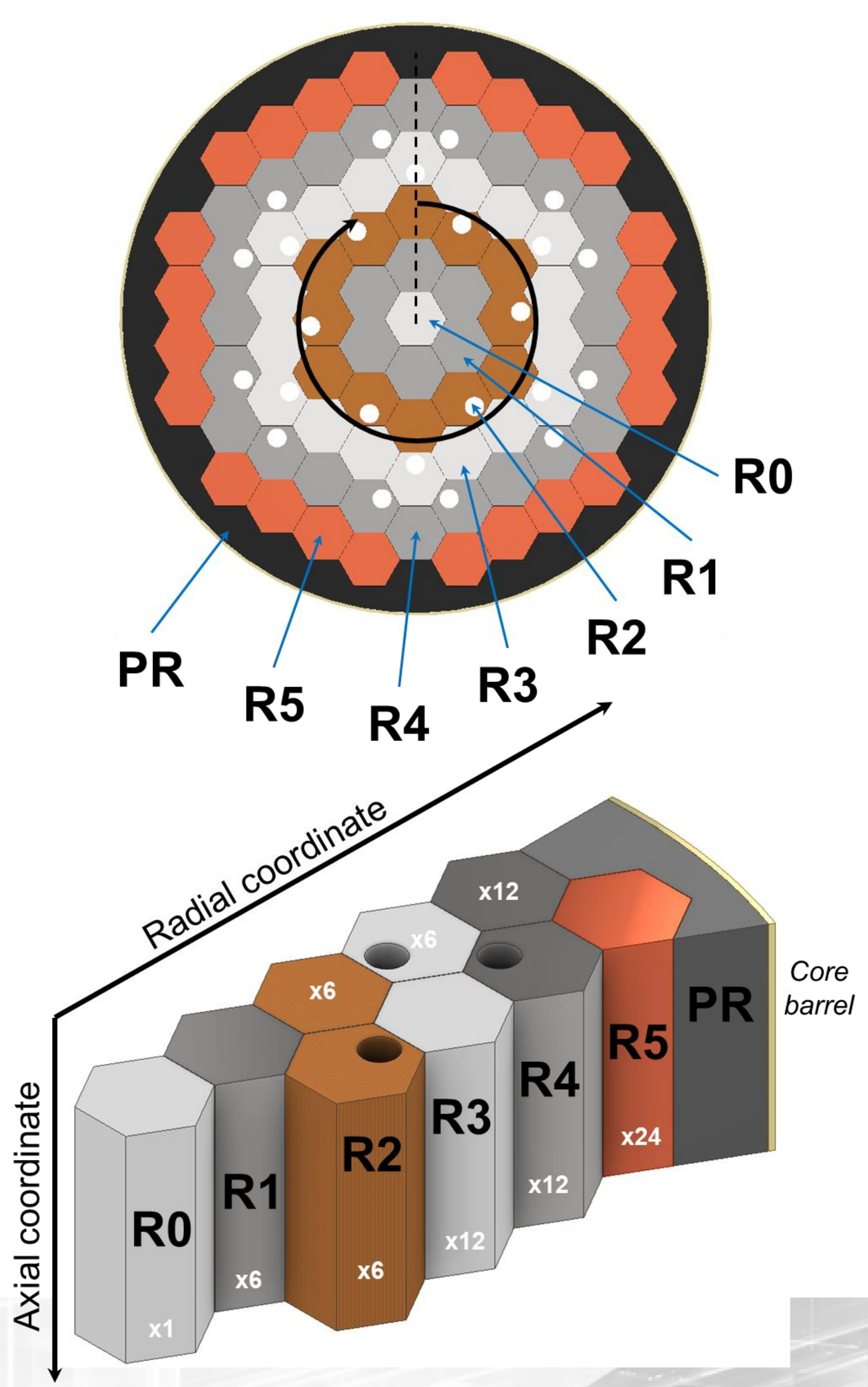
- the relatively low mass of water: break placement (top of the SG) and performance of the SG drain system,
- the fluid in the form of vapor,
- Reactor Protection System (RPS) - the SG drain actuation.
- system is actuated by the signal of the moisture detection in the Helium Purification System (HPS) line,
- In the analyzed scenario, the RPS with the drain system played significant role, and emptied the SG immediately after the signal detection, 15 seconds into the accident the water ingress was stopped.

Code benchmarking

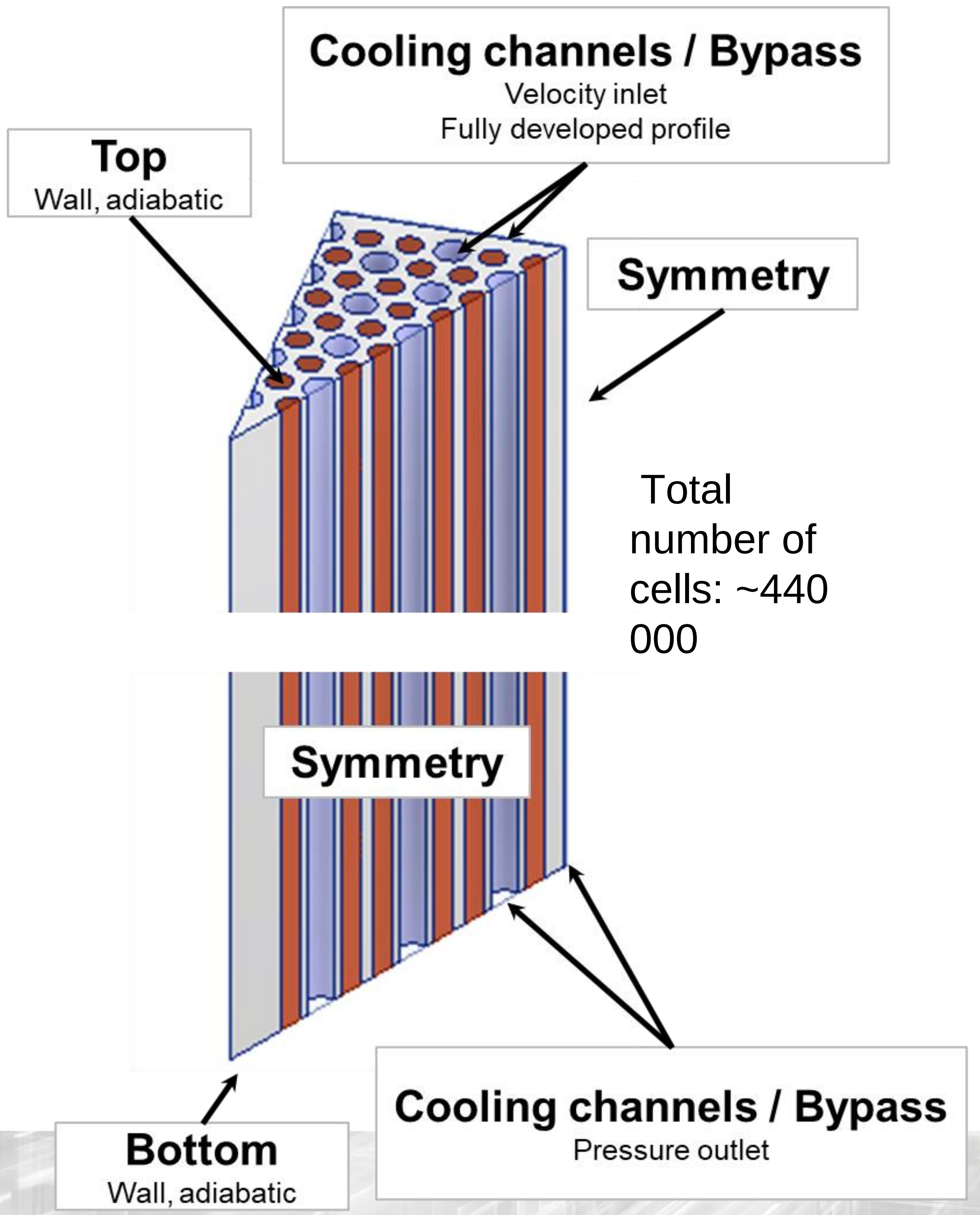
Thermal-hydraulic local vs. system approach – code benchamarking (1/3)

Sebastian Gurgul, Elżbieta Fornalik-Wajs, AGH University of Science and Technology, Faculty of Energy and Fuels, Department of Fundamental Research in Energy Engineering

2.5D thermo-hydraulic model

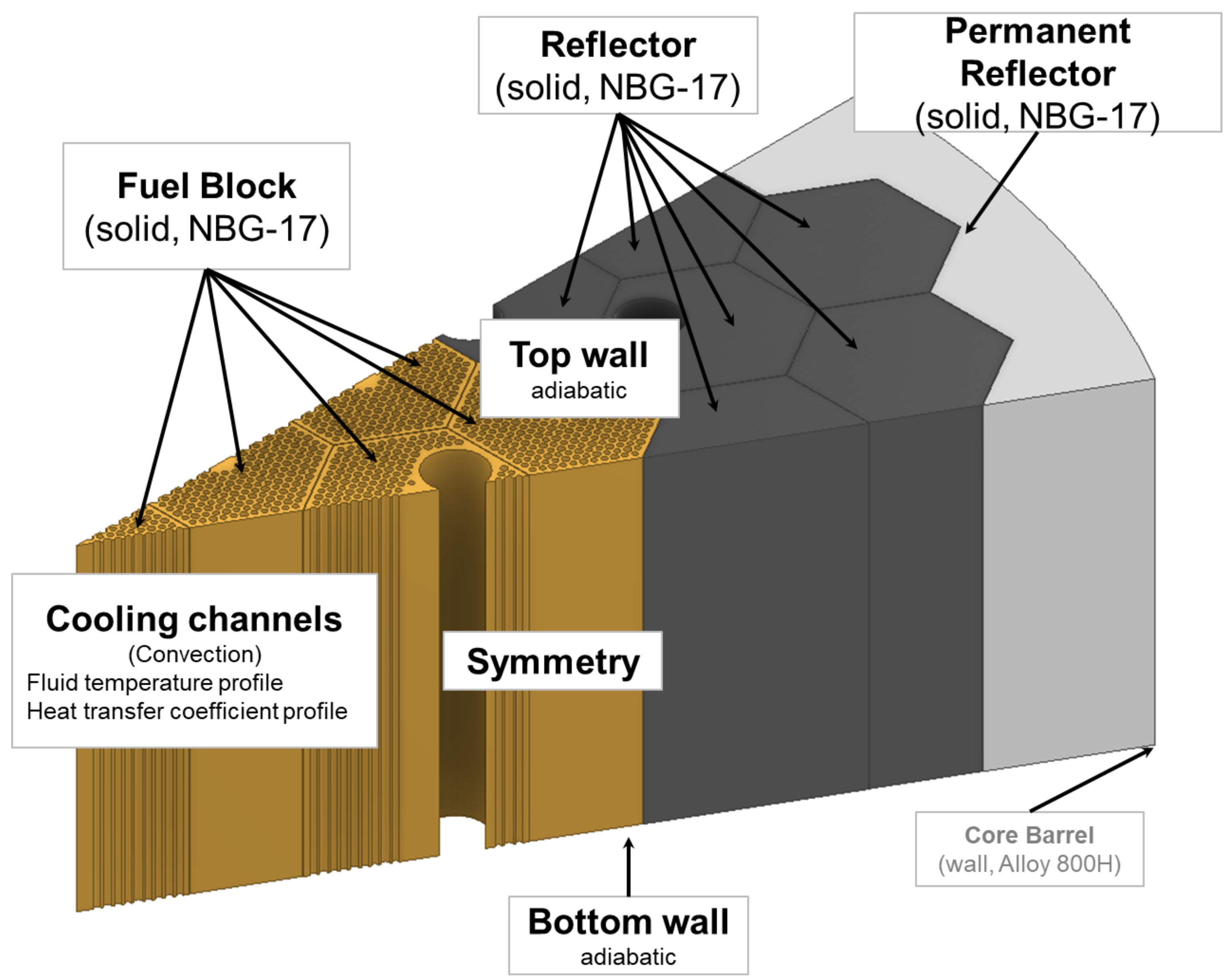


3D 1/12 Fuel Block Fluent model



3D 1/12 HTR core Fluent model

Total number of cells: ~2.5M



Thermal-hydraulic local vs. system approach – code benchamarking (2/3)

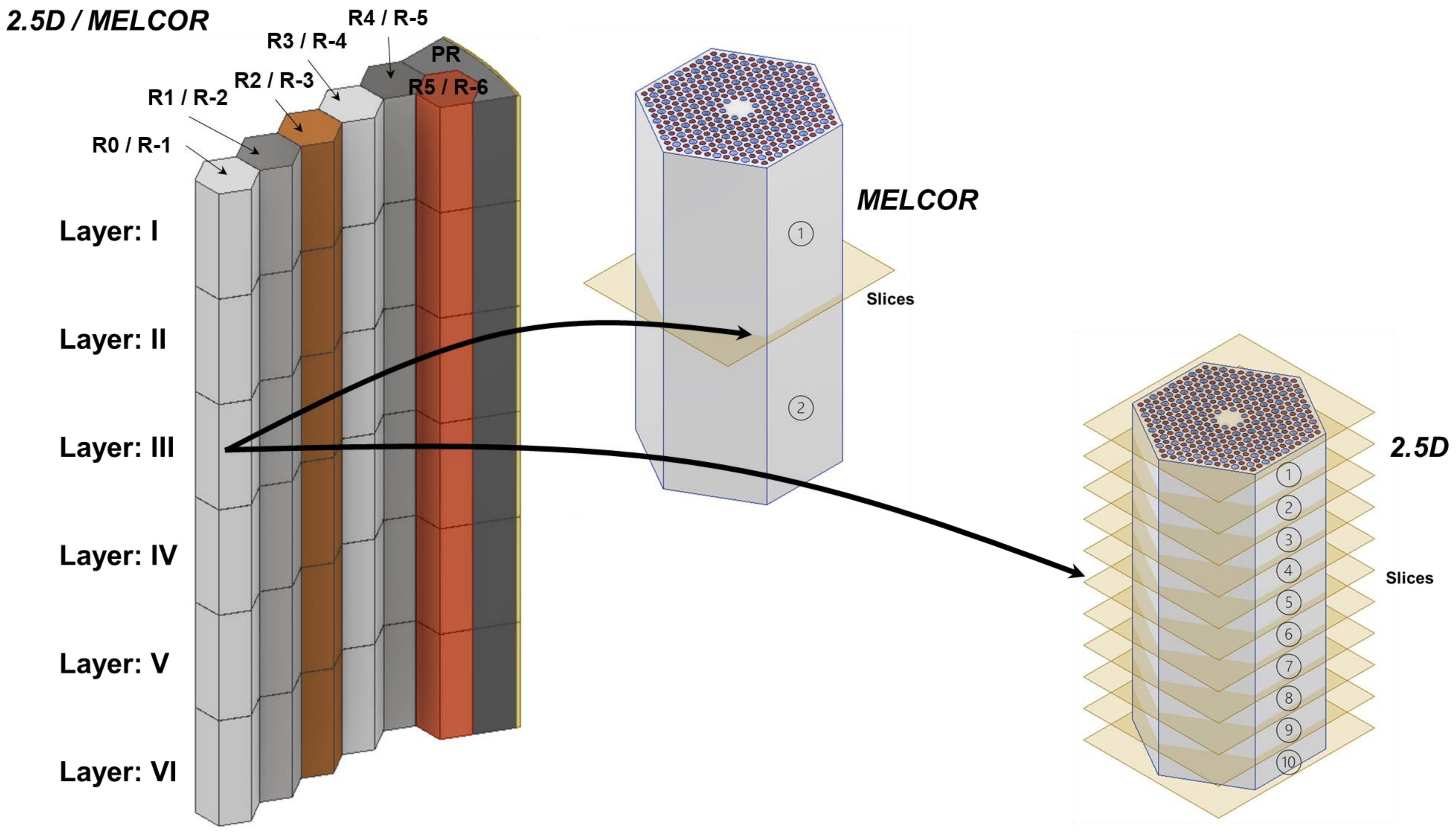


Fig. 2.5D results averaging. Adaptation of axial division of CFD calculation to the division of MELCOR

Tab. Comparison of the results from the 2.5D and MELCOR models

		Fuel				Bypass					
		Ring-1	Ring-2	Ring-3	Ring-4	BP-1	BP-2	BP-3	BP-4	BP-5	BP-6
T, K	MELCOR	1046	1064	1040	1131	597	597	597	597	824	702
	2.5D	1034	1038	1054	1074	1068	1081	1098	929	729	650
m, kg/s	MELCOR	0.5	3.1	5.7	6.0	0.1	0.2	0.4	0.5	0.5	0.5
	2.5D	0.515	3.088	5.631	6.175	0.087	0.219	0.350	0.481	0.525	0.438
m, %	MELCOR	3.10	17.80	32.80	34.40	0.50	1.30	2.00	2.80	2.60	2.70
	2.5D	2.94	17.63	32.16	35.27	0.50	1.25	2.00	2.75	3.00	2.50

- Comparison between the results delivered by these **two approaches shows good agreement**, except in the areas, in which some assumptions (i.e. temperature of helium in bypass) were applied
- Realization of the project with these two methodologies leads to the **knowledge of limitations of CFD and MELCOR models**, therefore they can be considered as **complimentary tools**
- **They can supplement each other** with the missed or assumed data and validate themselves in the case of lack of experimental data.

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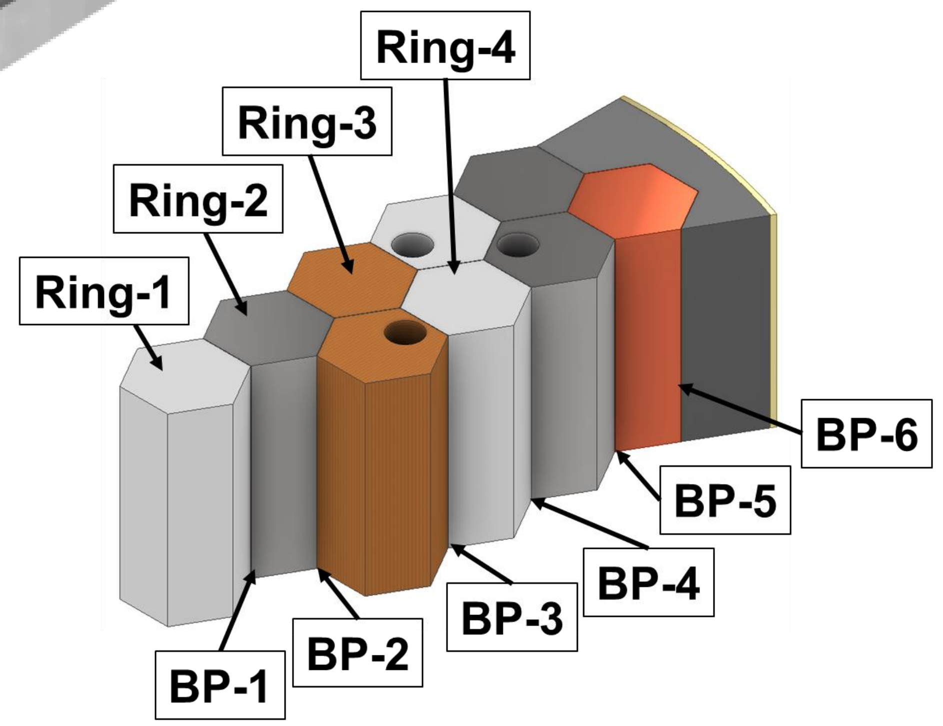
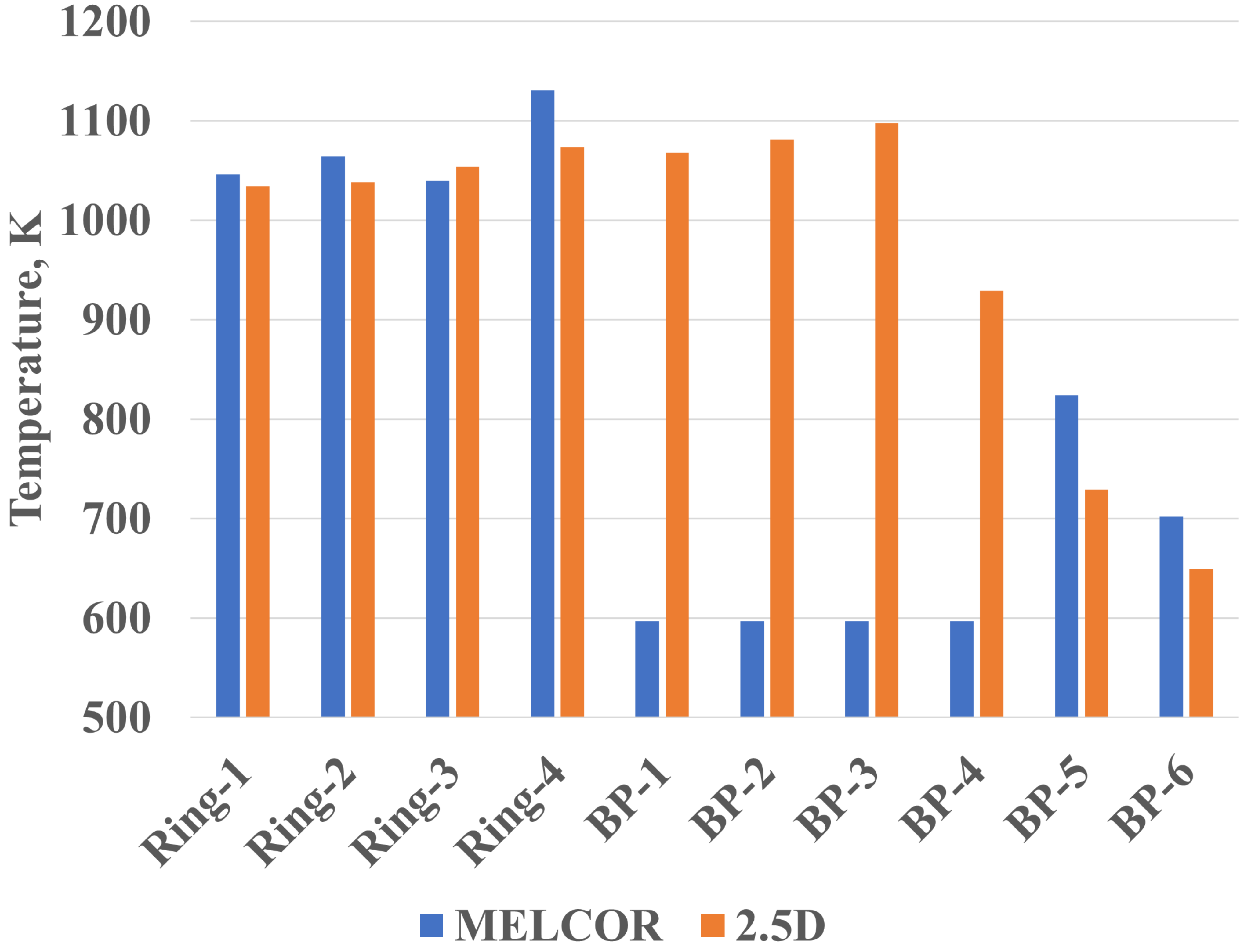
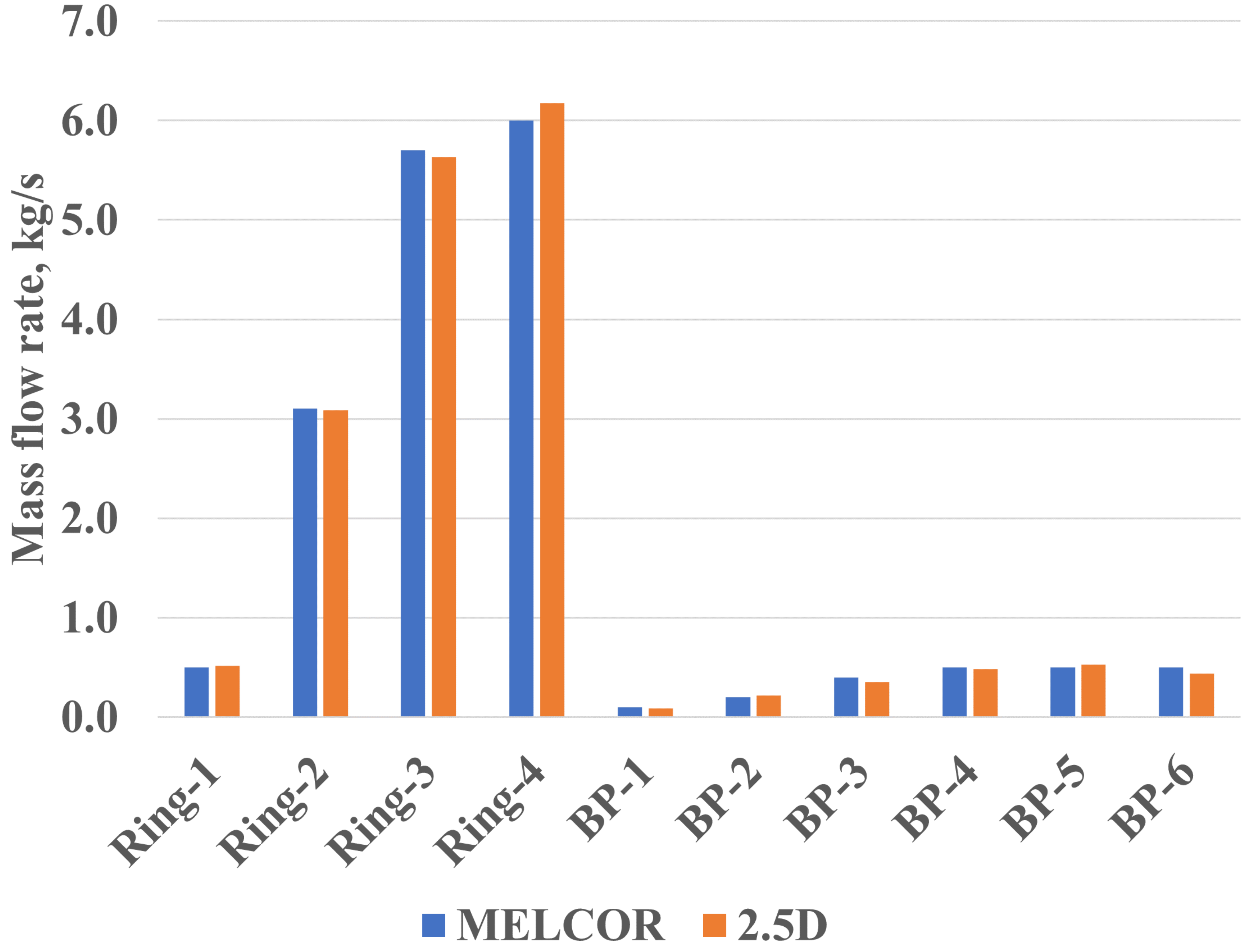
Thermal-hydraulic local vs. system approach – code benchamarking (3/3)







	Fluent (1/12 fuel block model)	MELCOR
Mass weighted average temperature values at the helium outlet	1031K	1046K

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Source term calculations

Source term evaluation

- The vast majority of fission products in HTGR (prismatic) reactors are retained in the TRISO fuel structure.
- The retention of the fission products in the TRISO fuel is made possible by the carbon coatings surrounding the UO_2.
- In the event that fission products migrate or form in the vicinity of the fuel, most of the material is retained in the primary circuit or in interconnected systems.

Normal operation releases

- Source term analysis for the normal operation were executed using simple analytical methods and extensive literature review.
- Releases were calculated for the whole plant, including the retention in the building and use of the ventilation systems.
- The core inventory, fuel release rates, attenuation factors, and other factors related to the transport of radioactive isotopes in the graphite matrix and coolant were selected from the HTGR-related literature and rescaled to the research reactor design.

Accidental releases

- Simulations were prepared with the MELCOR 2.2.14 computer code [12].
- The Depressurized Loss of Forced Cooling (DLOFC) accident, which is typically considered for HTGRs.
- The core inventory was estimated at the EOC (End-of-Cycle) core state on the basis of neutronics calculations with MCB code performed by AGH [10].
- The main focus in the project was put on fuel releases, primary circuit and containment radionuclide retention were not analysed (evaluated on the literature basis).

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Accident source term calculations – MELCOR

Source term evaluation

- The basic model of radionuclide release includes simulations of the diffusion process in subsequent layers of the TRISO fuel and the graphite matrix of the fuel compact and the graphite block. Diffusion coefficients taken from [33]
- The equations are solved by the COR package are partial differential equations for diffusion time dependent for transient and time independent for steady state with sources.

$$\chi \frac{\partial C}{\partial t} = \frac{1}{r^n} \frac{\partial}{\partial r}\left(r^n D \frac{\partial C}{\partial r}\right) - \lambda C + \beta$$

- 1st term (left hand side) responsible for the change in concentration of the examined group of radionuclides over time. This term for steady state is zero.
- 1st term (right side), responsible for the change of concentration in space, i.e. the space term (or diffusion term) depending on the geometry and the diffusion coefficient.
- 2nd negative source for a group of radionuclides and is responsible for radioactive decays. This term is proportional to the concentration and half-life of the group.
- 3rd term represents the source resulting from the production of radionuclides in the fission process. Major differences between the 2.2.14 and 2.2.18 MELCOR versions.
- The compatibility between the packages is preserved and there is no room for mistakes resulting from the lack of consistency between the input data to the DIF and RN1 and DCH models – v.2.2.18

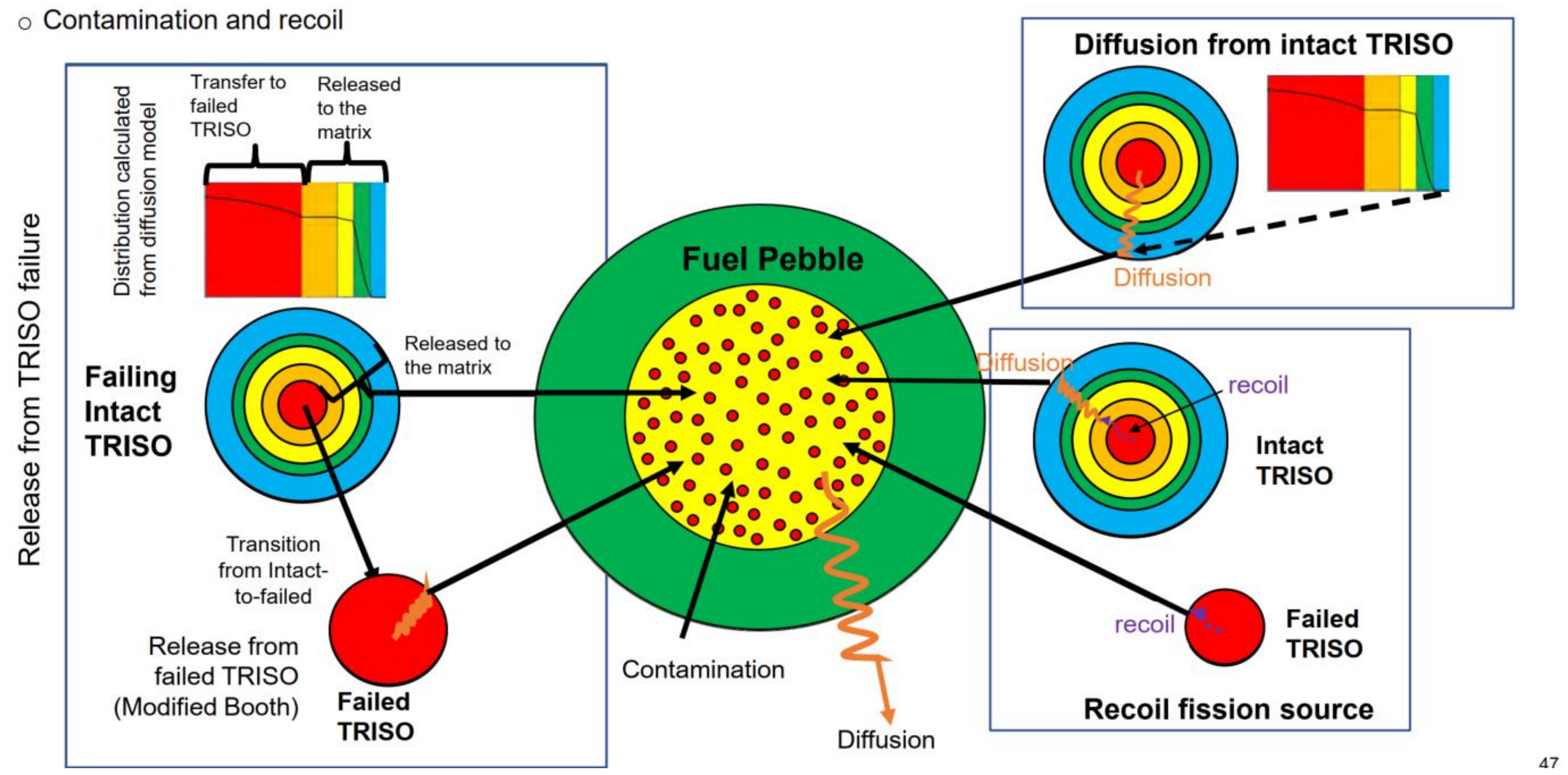


Fig. Basic sources of radionuclides and release mechanisms considered by MELCOR. The scheme is for ball fuel, but the mechanisms are analogous to compacts in prismatic reactors. The yellow area for prismatic reactors corresponds to compact, the green area is graphite blocks.

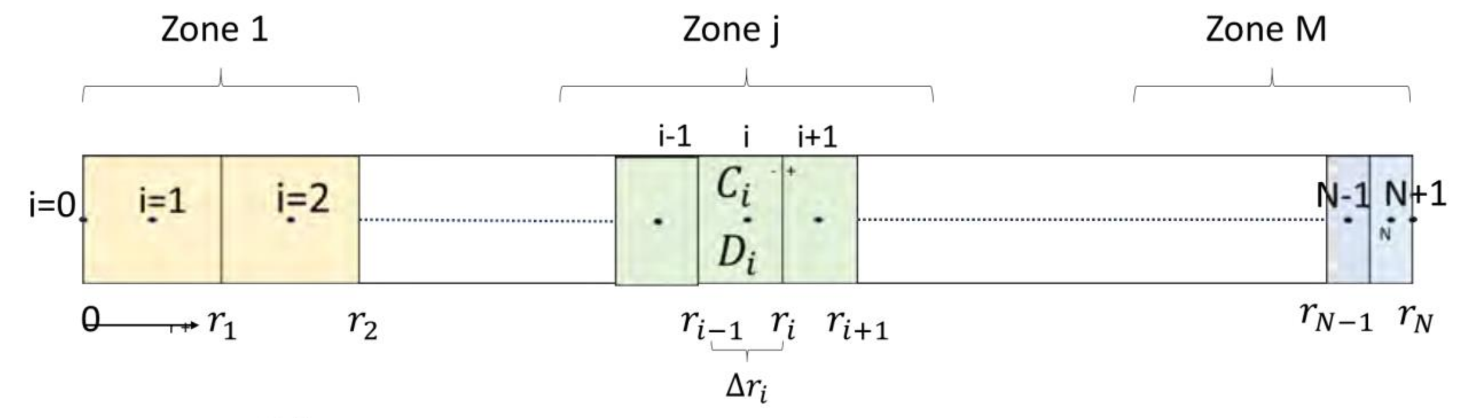


Fig. Division of the calculation area for the diffusion equation solver in MELCOR

Accident source term calculations (1/2)

Calculations for five main radionuclide groups:

- XE (Xenon),
- CS (Caesium),
- BA (Barium/Strontium),
- I (Iodine),
- AG (Silver).

The calculations are performed in four stages described below (MELCOR 2.2) [25]:

1. Stage 0: Calculation of the normal reactor operation and establishment of the thermal steady state.
2. Stage 1: Calculation of the radionuclide releases (especially through diffusion) from the fuel during normal operation of the reactor. Calculations are performed for many years of the core steady-state operation.
3. Stage 2: Calculations of radionuclides transport in the primary circuit during normal operation.
4. Stage 3: Calculations of transients along with the release of radionuclides from the fuel and other processes that radionuclides undergo.

The total mass of the significant groups is ~14 kg for EOC.

Tab. Pre-conceptual design core composition for radiologically significant radionuclide groups. Masses for the End Of the Cycle.

CLASS	STATE NAME	EOC [KG]
1	XE	8.42
2	CS	4.95
3	BA	3.89
4	I2	6.59
5	TE	1.21
6	RU	4.28
7	MO	6.23
8	CE	21.46
9	LA	18.16
10	UO2 (U)	713.65
11	CD	0.05
12	AG	0.11
13	BO2	3.23E-18

Accident source term calculations (2/2)

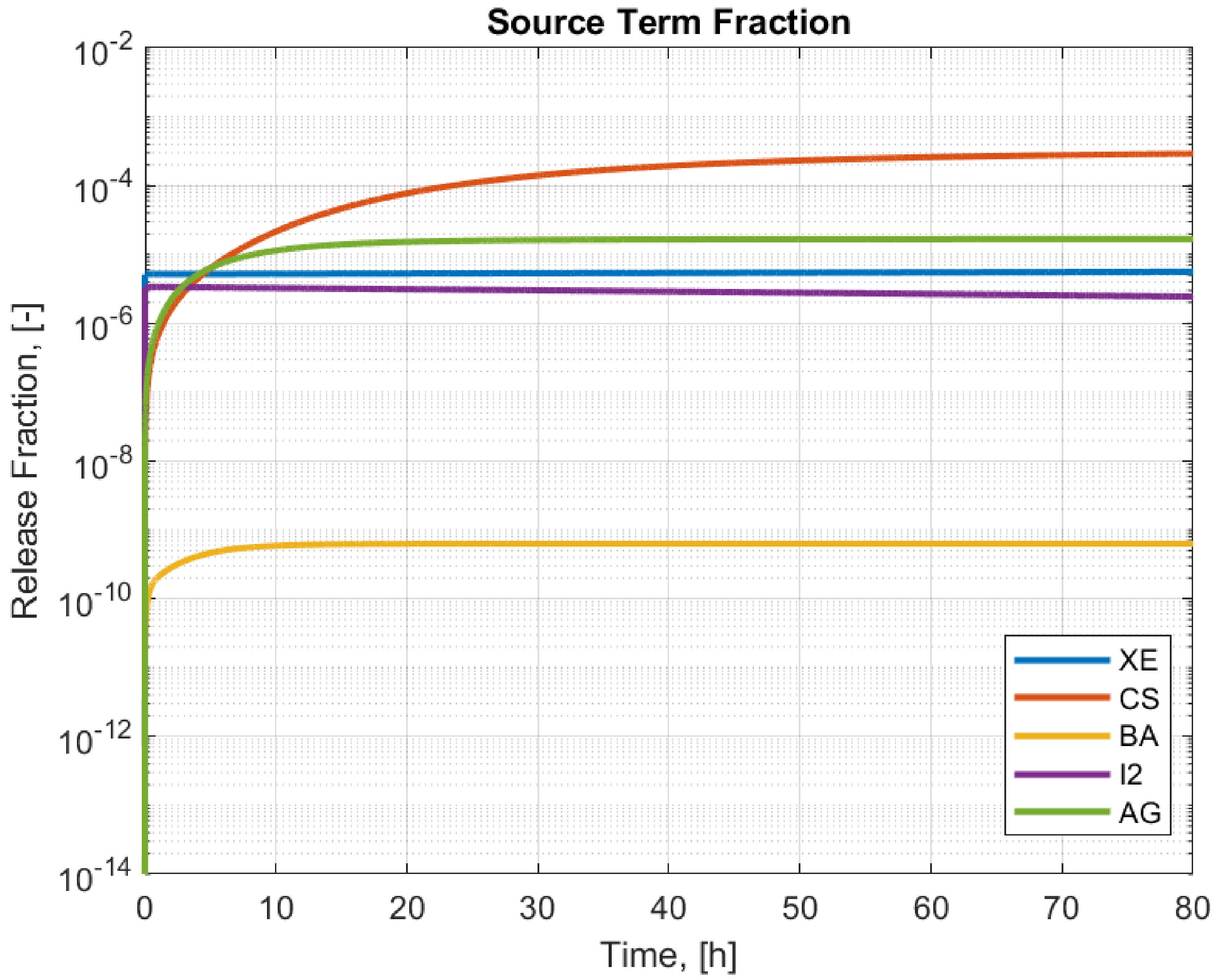
Source term evaluation

- the initial fraction of damaged particles - 10^{-5} ,
- a temperature-dependent damage curve based on the AVR curve - conservative and default for MELCOR 2.2.,
- no heavy metal contamination of the fuel,
- no initial SiC damage, and
- no recoil of fission products,
- five radiologically significant isotopes: Cs-137, I-131, Xe-135, Sr-90 and Ag-110.

RESULTS:

1. For longer time scale (hours) - releases from TRISO fuel are driven by the fuel temperature - diffusion process, damage accumulation process in the fuel.
   - the temperature rises relative to the steady state temperature - change of diffusion coefficient and fuel damage.
   - the caesium group (CS)- release of caesium occurs during the period of fuel temperatures rise.
   - silver group (AG) and the strontium group (BA) - releases increased over time up to a peak temperature (approximately 25,000 seconds)
2. DLOFC accident, the highest release expressed in the fraction of the initial core mass was observed for the caesium group. The model predicts the release of about 2×10-4 caesium fraction.

Fig. Radionuclide releases from fuel to the primary coolant system expressed as initial core inventory fraction. Results for the pre-conceptual research HTGR reactor during the DLOFC accident.



Releases of caesium - relatively high in comparison to expected releases for prismatic HTGRs [17]. Similar caesium fuel releases were observed also in Sandia National Laboratories MELCOR simulations of the PBMR-400 pebble-bed design [28].
Recommendation to rebuild the model with the new versions 2.2.18 or 2.2.21.

Summary

Conclusions

- Multiple codes were used for the demonstration of the methodology of the preliminary safety assessment of the research reactor.
- The neutronic investigations (MCB-POKE, Serpent 2-MELCOR 2.2.14) were performed coupled with thermal-hydraulic calculations, giving the more realistic and detailed results of the fuel cycle calculations for the purpose of the safety evaluations.
- Thermal-hydraulic evaluations using MELCOR 2.2.14 showed large margins for the fuel and RPV temperatures for the pre-conceptual research reactor design.
  - The modeling strengths and issues were identified of prismatic HTGRs modeling.
- Source term evaluations, both by analytical modeling and computer simulations were performed.
- The analytical calculations showed some differences for the releases of Iodine and metallic isotopes depending on the used methodology.
- Simulations done in MELCOR 2.2. are the promising method for the safety assessment of the HTGRs, although some concerns remain.
  - The major recommendation is to use the most recent code version (MELCOR 2.2.18 or 2.2.21)

REFERENCES

1 LINDLEY B, DAVIES M, DOMINIQUE H, STEMPNIEWICZ M, DE GEUS EAR, KUIJPER J, BRINKMAN G, VANVOR D., Final description an justification of GEMINI + system. Wood; 2020.
2 KUGELER K, ZHANG Z., Modular High-temperature Gas-cooled Reactor Power Plant. Berlin: Springer, Tsinghua University Press; 2019.
3 NATIONAL CENTRE FOR NUCLEAR RESEARCH, GOSPOSTRATEG-HTR project website. [Internet]. Otwock 2021 [cited March 2022]. Available from: http://gohtr.pl/.
4 SKRZYPEK E, MUSZYŃSKI D, SKRZYPEK M, DARNOWSKI P, MALESA J, BOETTCHER A, DĄBROWSKI M.P., Pre-Conceptual Design of the Research High-Temperature Gas-Cooled Reactor TeResa for Non-Electrical Applications. Energies. 2022.
5 Act of Parliament of 29 November 2000 on the Atomic law, with later amendments. [Internet]. [cited 2022 March]. Available from: http://isap.sejm.gov.pl/isap.nsf/download.xsp/WDU20010030018/O/D20010018.pdf.
6 IAEA, Safety Standards, Safety of Nuclear Power Plants: Design, Specific Safety Requirements No. SSR-2/1 (Rev. 1). Vienna: IAEA; 2016.
7 BRINKMANN G, VANVOR D, JUNG A., Final GEMINI + Safety Options Report. European Union's Horizon 2020 research and innovation programme; 2020.
8 CETNAR J, GUDOWSKI W, WALLENIUS J., MCB: A Continuous Energy Monte Carlo Burnup Simulation Code. In: EUR 18898 EN, OECD/NEA; 1999. p. 523.
9 PFEIFFER W, MALEK G, LUND K., POKE A Gas-Cooled Reactor Flow and Thermal Analysis Code. Gulf General Atomics Inc.; 1970. GA-10226.
10 CETNAR J, FORNALIK-WAJS E, DOMAŃSKA G, KOPEĆ M, OETTINGEN M, GAJDA P, GURGUL S., Development studies of the core configuration along with the experimental program and numerical calculations supporting safety analyzes,. Kraków: NCBJ internal report; 2021.
11 LEPPÄNEN J, PUSA M, VIITANEN T, VALTAVIRTA V., The Serpent Monte Carlo Code: Status, Development and Applications in 2013. Annals of Nuclear Energy. 2015,82:142-150.
12 HUMPHRIES LL., MELCOR Computer Code Manuals, Vol. 1: User's Guide, Vol. 2:Reference Manuals, Version 2.2.14959. Albuquerque: Sandia National Laboratories; 2019.
13 SKRZYPEK M, SKRZYPEK E, STEMPNIEWICZ M, MALESA J., Study on the DLOFC accident of the GEMINI+ conceptual design of HTGR reactor with MELCOR and SPECTRA. In: Proceedings of HTR 2021; 2021; Virtual Conference, Indonesia.
14 STEMPNIEWICZ M., GEMINI+ D1.15: Comparison of Air Ingress Calculations and Recommendations for Future Applications. Arnhem: NRG; 2020.
15 UDIYANI PM, KUNTJORO S., Estimation Of Routine Discharge Of Radionuclides On Power Reactor Experimental RDE. Urania. 2017 February:1-68.
16 IDAHO NATIONAL LABORATORY, HTGR Mechanistic Source Terms White Paper. Idaho Falls, Idaho: Idaho National Laboratory, NGNP; 2010.
17 PETTI D.A, HOBBINS R.R, LOWRY P., GOUGAR H., Representative Source Terms and the Influence of Reactor Attributes on Functional Containment in Modular High-Temperature Gas-Cooled Reactors. Nuclear Technology. 2013:181-197.
18 CAO J, ZHANG L, XIE F, XIA B, TSZ TANG LAM S., Source Term Study on Tritium in HTR-PM: Theoretical Calculations and Experimental Design. Science and Technology of Nuclear Installations. 2017July.
19 VERFONFDERN K, SUMITA J, UEAT S, SAWA K., Modeling of fuel performance and fission product release behavior during HTTR normal operation (a comparative study of the FZJ and JAERI modeling approach). Japan Atomic Energy Research Institute; 2001.
20 YUANZHONG L, JIANZHU C., Fission Product Release and Its Environment Impact for Normal Reactor Operations and for Relevant Accidents. Nuclear Engineering and Design. 2002 March 11:81-90.
21 KUIJPER J, MUSZYŃSKI D., Core design calculations of GEMINI + system. GEMINI+; 2021.
22 FÜTTERER M.A, D'AGATA E, RAEPSAET X., Is Tritium an Issue for High Temperature Reactors? In: Proceedings of HTR 2014; 2014; Weihai, China.
23 CHEN H, LI C, XING H, FANG C., The R&D of HTR-STAC Program Package: Source Term Analysis Codes for Pebble-Bed High-Temperature Gas-Cooled Reactor. Hindawi Science and Technology of Nuclear Installations. 2018:9.
24 KUGELER K, NABIELEK H, BUCKTHORPE D., The High Temperature Gas-cooled Reactor, Safety considerations of the (V)HTR-Modul. Luxembourg: Publications Office of the European Union; 2017.
25 SANDIA NATIONAL LABORATORY, Mechanistic Source Term Considerations for Advanced Non-LWRs, Revision 1, SAND2021-1668. Sandia National Laboratory; 2021.
26 SANDIA NATIONAL LABORATORY, Preliminary Radioisotope Screening for Off-site Consequence Assessment of Advanced Non-LWR Systems, SAND2021-11703. Sandia National Laboratory; 2021.
27 CETNAR J, FORNALIK-WAJS E, DOMAŃSKA G, KOPEĆ M, OETTINGEN M, GAJDA P, GURGUL S., Opracowanie konfiguracji rdzenia wraz z programem eksperymentalnym i wykonanie obliczeń numerycznych wspomagających analizy bezpieczeństwa. Analizy stanów awaryjnych w celu określenia parametrów fizycznych i eksploatacyjnych reaktora badawczego HTGR.. Kraków: Akademia Górniczo-Hutnicza; 2022.
28 SANDIA NATIONAL LABORATORY, SCALE/MELCOR Non-LWR Source Term Demonstration Project - High-Temperature Gas-Cooled Reactor. In: Meeting for US NRC; 2021.
29 CATHARE. 2 V25_3mode9.1 Code: General Description; CEA: Paris, France, 2019.
30 ANSYS. Ansys Manual Guide, Release 2021 R1; ANSYS: Canonsburg, PA, USA, 2021.
31 L. Lommers, F. Shahrokhi, F. Southworth i J. Mayer III, „ AREVA Modular Steam Cycle–High Temperature Gas-Cooled Reactor Development Progress," w HTR2014: 7. International Topical Meeting on High Temperature Reactor Technology, Weihai (China), 2014.
32 U.S.NRC, NRC Non-Light Water Reactor Near-Term Implementation Action Plans, ML 17165A069, July 2017
33 SCALE/MELCOR Non-LWR Source Term Demonstration Project – High-Temperature Gas-Cooled Reactor, July 2021.
34 M.M. Stempniewicz, E.A.R. de Geus, F. Roelofs, M. Skrzypek, E. Skrzypek, J. Malesa, P. Vacha, J. Pokorny, N. Kohtz, C. Pohl, Analysis of air ingress scenario in GEMINI+ plant, Nuclear Engineering and Design, Volume 399, 2022, 112030, ISSN 0029-5493,

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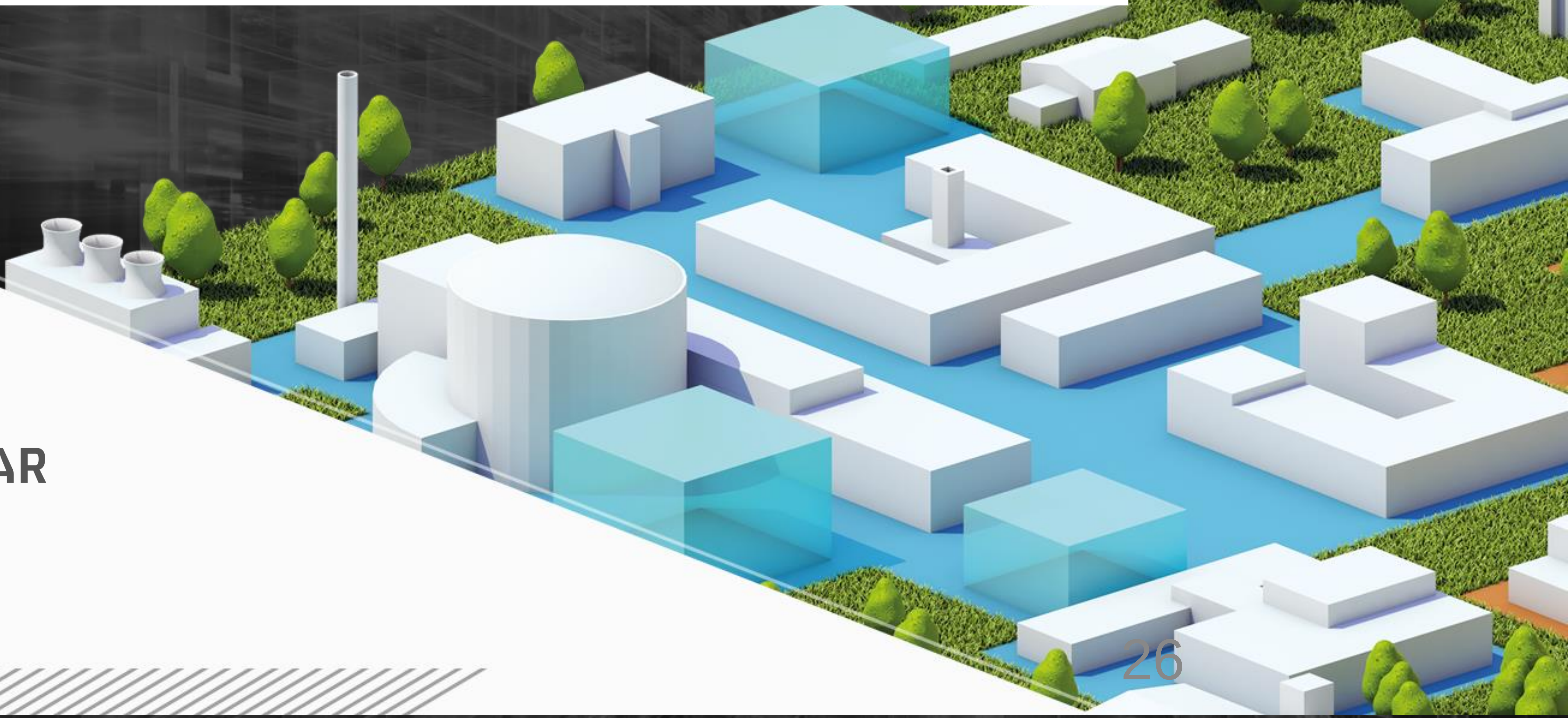

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