McSAFER Training Course

SMR Neutronics and Thermal Hydraulics

Heikki Suikkanen (LUT)

22.3.2022

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Course Practicalities

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Course overview

Tuesday - lecture room 7443 + Zoom

- Lectures by experts from the McSAFER project on various computational methodologies and tools applied for SMRs
- Lectures are streamed via Zoom and some lectures are also presented via remote connection

Wednesday - lecture room 7443

- Introduction and demonstration of the Kraken multi-physics framework applied for SMR safety analyses

Thursday - lecture room 7443 + LUT nuclear safety research laboratories

- Lecture on experimental investigations of SMRs with MOTEL test facility
- Laboratory activities in small groups

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Passing the course

The course is an actual post-graduate / continuing education course by LUT University

- You will get 2 ECTS credits by passing the course
    - Automatic for LUT students
    - Certificate can be obtained by external participants

What you need to do to pass the course?

- Participate on site (lecture day, Kraken day, lab day)
- After the contact teaching days
    - Complete short quizzes to be posted to the course digital learning environment
    - Return a short report on the laboratory exercise (each group will return one report)
- The course digital learning environment is Moodle. You will get login credentials.

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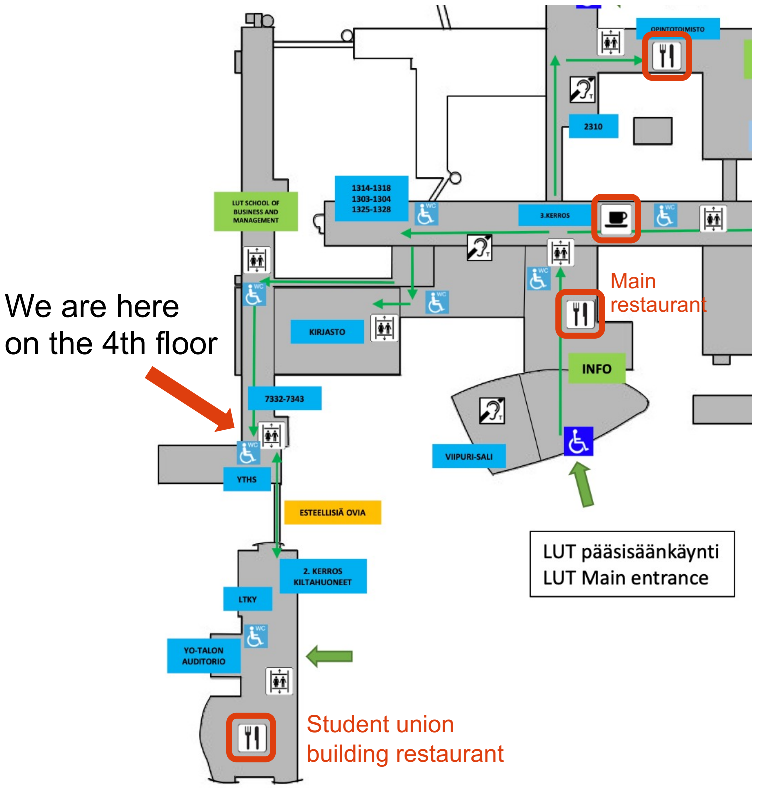
Breaks

- Coffee served on breaks on Tuesday and Wednesday
- Lunch can be enjoyed at everyones own expense at any of the student restaurants on the campus



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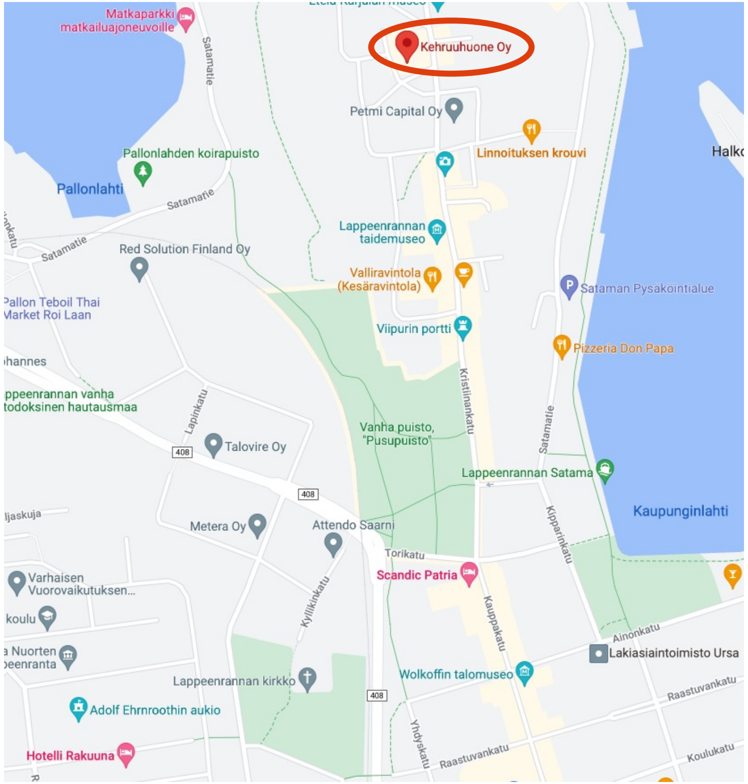
Dinner on Tuesday evening

- Place: Kehruuhuone restaurant in the “Fortress“ part of Lappeenranta
    - Kristiinankatu 20, 53900 Lappeenranta
- Time: Today at 19:00
- For everyone who expressed their interest to participate in advance!



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Brief Introduction of the McSAFER Project

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

McSAFER

High-Performance Advanced Methods and Experimental Investigations for the Safety Evaluation of Generic Small Modular Reactors (McSAFER)

- Euratom funded project
- Launched in 9/2020
- 3-year duration
- 13 partners
- Coordinated by KIT

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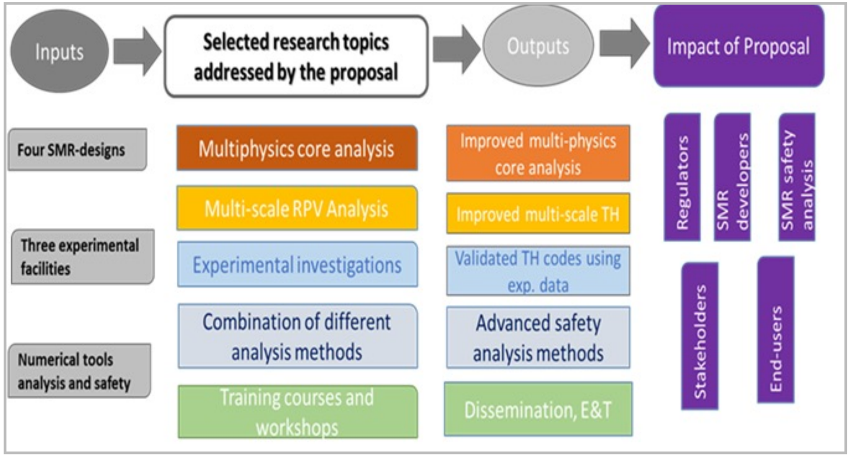
Technical goals

- Advance the safety research for SMR by combining experimental and analytical investigations (numerical simulations)
- Development, improvement of simulation tools for SMRs (safety evaluations)
- Validation of simulation tools with experimental data generated within McSAFER (COSMOS-H, MOTEL, HWAT)
- Application of simulation tools (traditional, advanced low-order and high-fidelity) to four SMR-designs
- Demonstrate advantages of advanced tools compared to legacy methods



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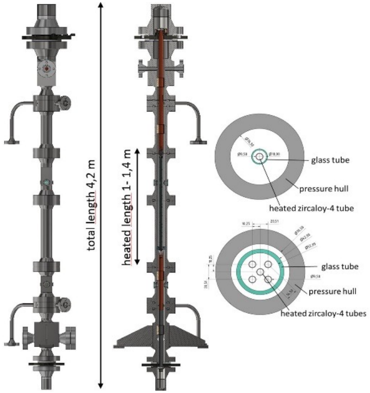
WP2: Key experimental investigations and validation

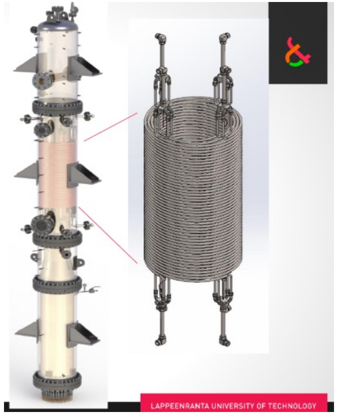
- Validation matrix: CFD, subchannel and system thermal hydraulics codes

COSMOS-H (KIT)



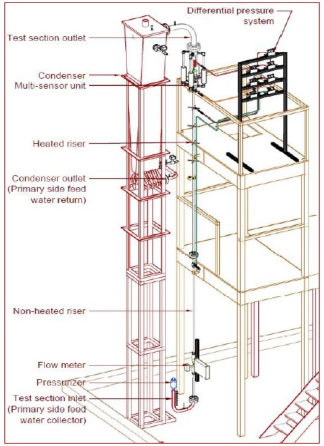
Codes	COSMOS-H		
	KIT	LUT	UJV
CFD	CFX	OpenFOAM	CFX
SubCH	SCF		VIPRE
SysTH	TRACE		RELAP3D

MOTEL (LUT)



Codes	MOTEL				
	KIT	LUT	UJV	UPM	TBL
CFD	CFX		FLUENT		
SubCH			VIPRE		COBRA-TF
SysTH		APROS		TRACE	

HWAT (KTH)



Codes	HWAT	
	KTH	UPM
CFD	OpenFOAM	
SubCH		
SysTH	GOTHIC/TRACE	TRACE

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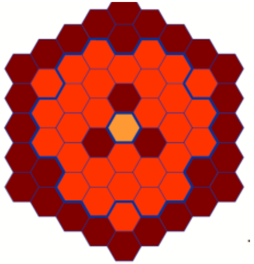
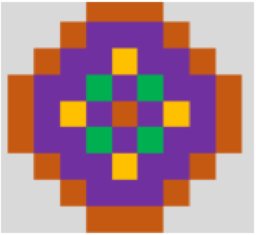
WP3: Core analysis with different methodologies

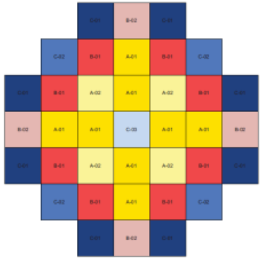


CAREM

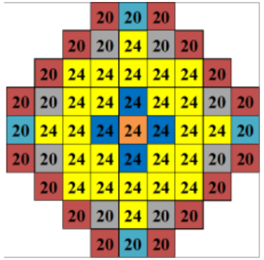


F-SMR

NuSCALE: High fidelity analysis
- Normal fuel
- ATF fuel



NuScale



KSMR

WP3: SMR core designs under investigation

Core analysis (Static /transient)

- Traditional with system code and point kinetics
  - **RELAP5, ATHLET, TRACE**
- 1D system code + 3D nodal diffusion
  - **TRACE/PANTHER**
  - **TRACE/PARCS**
- Low order transport + subchannel codes
  - **PARCS-SP3/Subchanflow (SMART)**
  - **APOLLO3/FLICA (F-SMR)**
  - **WIMS/ARTHUR (NuScale)**
  - **DYN3D-SP3/Subchanflow (NuScale)**
- High-fidelity MC + subch + TM codes
  - **SERPENT2/Subchanflow/TU**

WP3: Analysis methodologies

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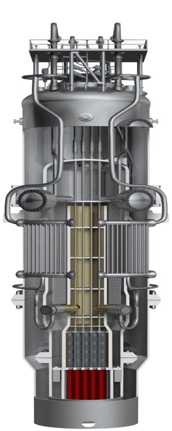
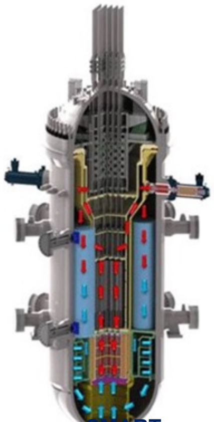
WP4: Multiscale RPV thermal hydraulic analysis



NuScale
(USA)



SMART
(Korea)

SMR: Designs to be analysed

Scenarios:

- **NuSCALE: Boron Dilution**
- **SMART: ATWS**

- 1D system TH code + PK
  - **TRACE**
  - **ATHLET**
  - **RELAP5-3D**
- 3D system TH-code + Subchannel code
  - **TRACE/Subchanflow**
  - **TRACE/ARTHUR**
- 3D system TH + CFD code
  - **TRACE/OpenFOAM**
  - **ATHLET/TrioCFD**
  - **ATHLET/FLUENT**

WP4: Safety Analysis methodologies

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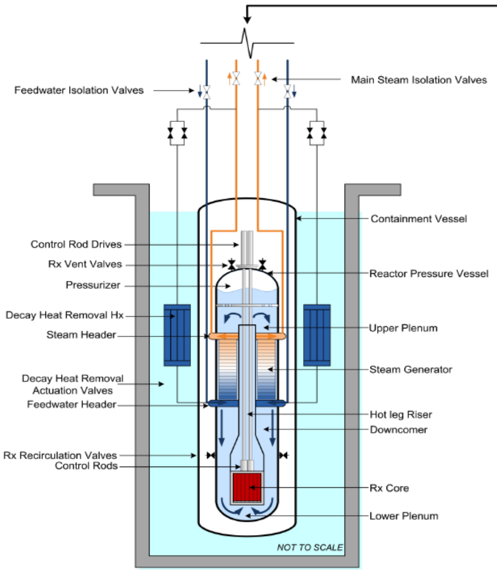
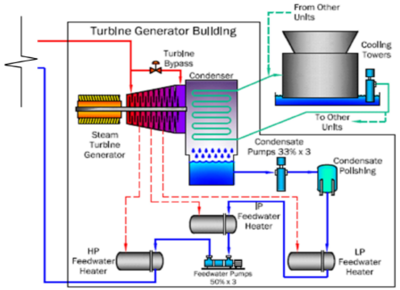
WP5: Multiscale/multiphysics SMR plant analysis





SMR Designs to be analysed: NuScale and SMART

Scenarios:

- **NuSCALE: SLB**
- **SMART: SLB**

- 1D system TH code + 3D nodal diffusion
  - **TRACE/PARCS (KIT)**
  - **TRACE/PANTHER (TRACTEBEL)**
  - **TRACE/ANTS (VTT)**
  - **ATHLET/DYN3D (HZDR, UJV)**
- 3D system TH-code + Subchannel code + 3D nodal diffusion
  - **TRACE/PARCS/SCF (KIT, UPM)**
  - **TRACE/WIMS/ARTHUR (JACOBS)**
- 3D system TH code + 3D nodal diffusion + CFD code
  - **TRACE/PARCS/OpenFOAM (KIT, UPM)**
  - **ATHLET/DYN3D/TrioCFD (HZDR)**
  - **TRACE/ANTS/OpenFOAM (VTT)**
  - **ATHLET/DYN3D/FLUENT(UJV)**

WP5: Safety Analysis methodologies

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McSAFER education and training, dissemination

- Training courses:
  - First training course on SMR Technologies, January 25-27, 2021: UPM
    - 194 online participants
  - **Second training course on neutronics and thermal hydraulics for SMR, Now: LUT**
  - MOOC course on Multiphysics simulations applied to SMR (March 2023), UPM
- Mobility program
  - 9 fellowships to be assigned for mobility of young researchers from partner organizations
  - See: https://mcsafer-h2020.eu/news-and-events/
  - Still available budget

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CAREM-like model with PARCS-SCF

KIT Model

KIT: J. Blanco, M. Garcia, L. Mercatali, V. Sanchez-Espinoza
CNEA: H. Lestani, M. Dalinger, A. Weir, R. P. Zalazar, E. Lopasso

22.03.2022

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Karlsruher Institut für Technologie

Comisión Nacional de Energía Atómica

Outline

1. Project Context
2. CAREM-like characteristics
3. Workflow & CAREM-like Model
4. Steady-state and transient scenarios
5. Conclusions/ Limitations / Future work

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Karlsruher Institut für Technologie

1. Context

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Context of CAREM modeling activities in McSAFER [1]

- High-Performance Advanced Methods and Experimental Investigations for the Safety Evaluation of Generic Small Modular Reactors [1]
- **Main objective:** Advancement of the safety research for Small Modular Reactors (SMR) by combining:
    - TH Experiments to investigate SMR-specific safety-relevant phenomena for code validation.
    - Advanced computational tools from previous European projects used for multi-physics analysis of different SMR design cores.
    - Conventional, low order and high fidelity numerical tools to study inherent safety features as well as how the safety function of core sub-criticality and core coolability under postulated design-basis accident-conditions is assured.
- Many university, R&D and industrial partners from different countries: KIT (DE), LUT (FI), CEA (FR), UJV (CZ), HZDR (DE), WOOD (UK), VTT (FI), JRC (BE), PE (DE), UPM (ES), TBL (BE), KTH (SW), CNEA (AR)

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Context of CAREM modeling activities in McSAFER [1]

- Different concepts of SMRs (LWRs quite advanced) [1]:
    - Smaller cores ~2 m active height with ~ 40 - 60 FAs (Steep flux gradients)
    - Powers 100 – 1000 MWth
    - Heterogeneous core loadings with profiled FAs and CRs (specially wo boron)
    - w/wo boron
    - Burnable poisons
    - Many systems integrated inside the RPV
    - Forced/natural convection
    - Passive safety systems
- → Defy nodal solvers (diffusion limit) with 1D TH (mixing) → Accuracy of local safety parameters?

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Context of CAREM modeling activities in McSAFER [1]

McSAFER

Accuracy of local safety parameters?

- Assess limits of **state-of-the-art** codes, i.e., coupled nodal neutronics with TH, at normal and transient conditions
  - Diffusion limit and mixing
  - Bundle feedback
  - Safety criteria: DNBr, max fuel temperature/enthalpy? Transients?
- Requires validation
  - Experiments for TH (fuel rod arrangements with helical HX) feasible
  - No experimental data available for neutronics parameters → verification → **Monte Carlo high fidelity** simulations (McSAFE) as reference
- Coupled high fidelity (Monte Carlo)
  - Less approximations
  - Local (pin-wise) feedback
  - Subchannel for the time being (CFD in the long-term)
  - High computational cost: HPC required (e.g. HoreKa)
- Alternatively, **advanced low order methods** (deterministic transport, e.g. SN, SPN, MOC,…)

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Context of CAREM modeling activities in McSAFER [1]

Accuracy of local safety parameters? → Working Package 3 (WP3) [1]

- 4 SMRs cores selected
- Multi-physics methodologies
  - Coupled **state-of-the-art** codes, i.e., coupled nodal neutronics with TH
  - Coupled **advanced low order methods** (deterministic transport, e.g. SN, SPN, MOC,…)
  - Coupled **high fidelity with Monte Carlo**
- Safety parameters during normal and off-normal conditions: steady-state and transient

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2. CAREM

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CAREM [2][3][4]

- CAREM stands for Central ARgentina de Elementos Modulares in Spanish.
- This LWR SMR was designed and developed by Argentina's Atomic Energy Comission (CNEA) and nuclear technology firm INVAP since 1984
- Currently under construction in Lima, Buenos Aires, Argentina.
    - Civil work started on February 8, 2014
    - 32 MWe prototype (CAREM 25) under construction: CNEA is working on the conceptual designed of the commercial CAREM of ~100-120 MWe

CAREM-25 Construction Site (Lima, Buenos Aires, Argentina) [2]

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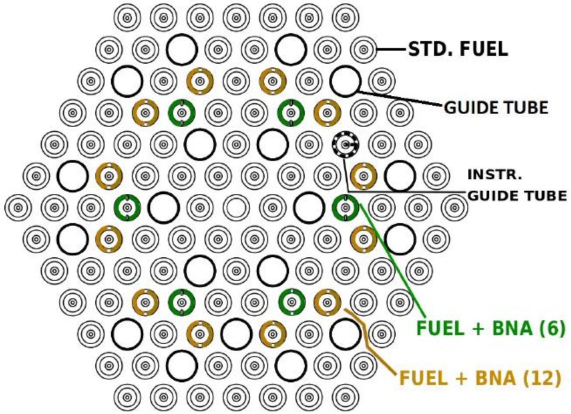
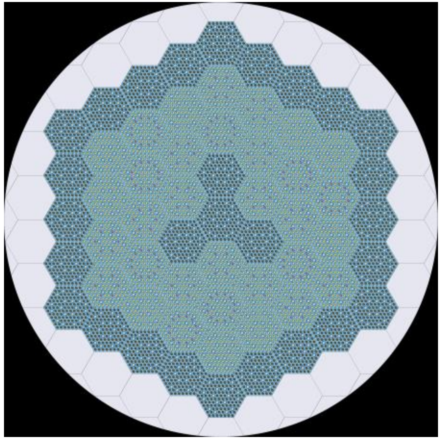
CAREM Design [5]

- RPV contains the helical once-through SGs, whole primary coolant, and CRDMs
- Natural circulation
- No boron during normal operation
- 61 Hexagonal Fuel Assemblies
  - ~1.4 m active height
  - 108 fuel rods (6-12 with Gd_2O_3 BP) with different enrichments
  - 18 CR guide tubes
  - 1 Instrumentation guide tube
- Rod Ejection Accident excluded by design. Innovative hydraulic control rod drive
- Low risk of large LOCA – absence of large orifices in primary system
- Passive safety systems



CAREM-like Fuel Assembly lattice (not actual FA) [6]



CAREM-like core (not actual core) [7]

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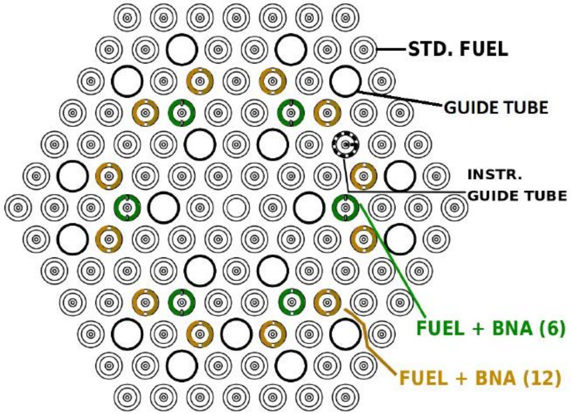
CAREM – Scenario [6]

- Steady-state
  - Hot Full Power
  - Equilibrium Xenon
  - No boron
  - Critical CRs position
- Transient
  - Reactivity Insertion Accident (RIA)
  - Cold front originated at the secondary side
  - Outlet pressure, inlet coolant temperature and mass flow values during 50 seconds



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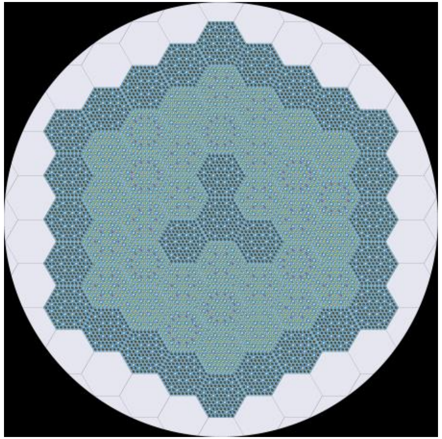

CAREM-like core (not actual core) [7]

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3. Workflow & Model

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Workflow & Model

NEUTRONICS

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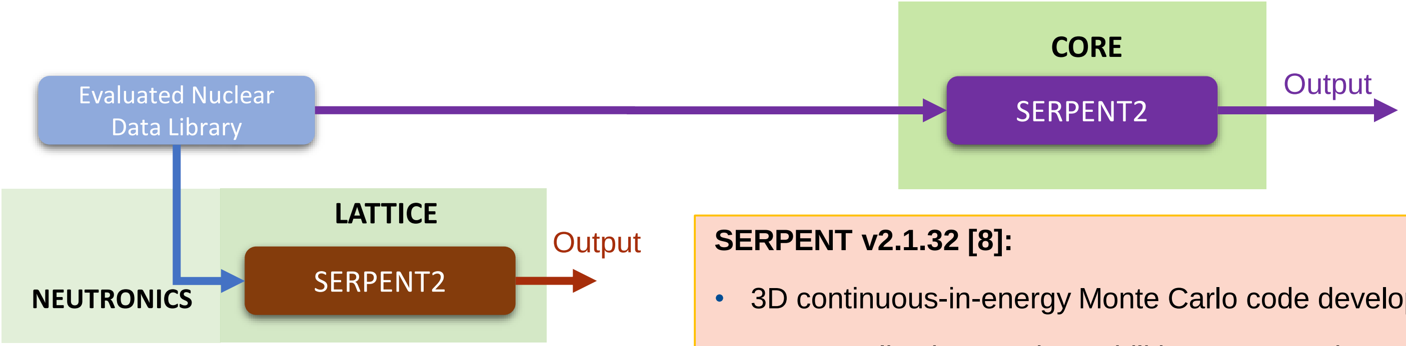
Workflow & Model: SERPENT2



SERPENT v2.1.32 [8]:

- 3D continuous-in-energy Monte Carlo code developed by VTT
- Many applications and capabilities, among others:
  - Condensation and homogenization (few groups XS generation)
  - Burnup
  - Criticality calculations
  - External Source/ Transient calculations
  - Neutron/ Photon transport (dose calculation)
  - Multi-physics coupled calculations

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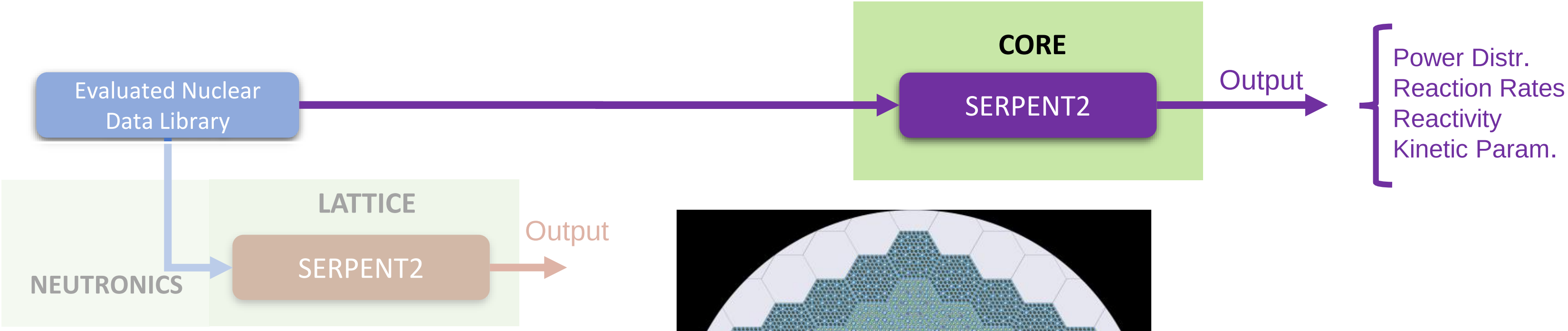
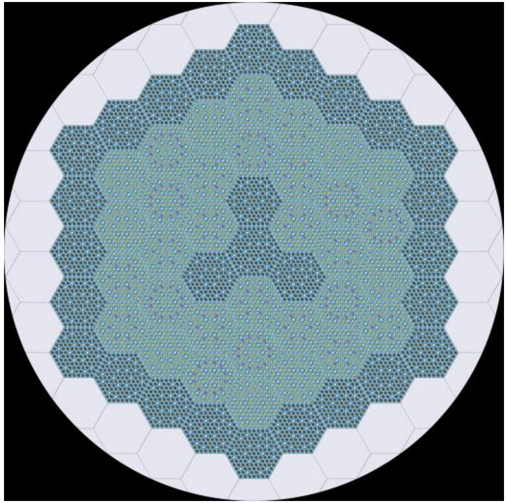
Workflow & Model: SERPENT2



Full core model:

- Serpent options:
  - Criticality/ Transient
  - Geometry/ Material Detail
  - Spatial/ Energy Continuum
  - Xenon Equilibrium
  - CZP/HZP/HFP
  - CRs positions



CAREM-like core (not actual core) [10]

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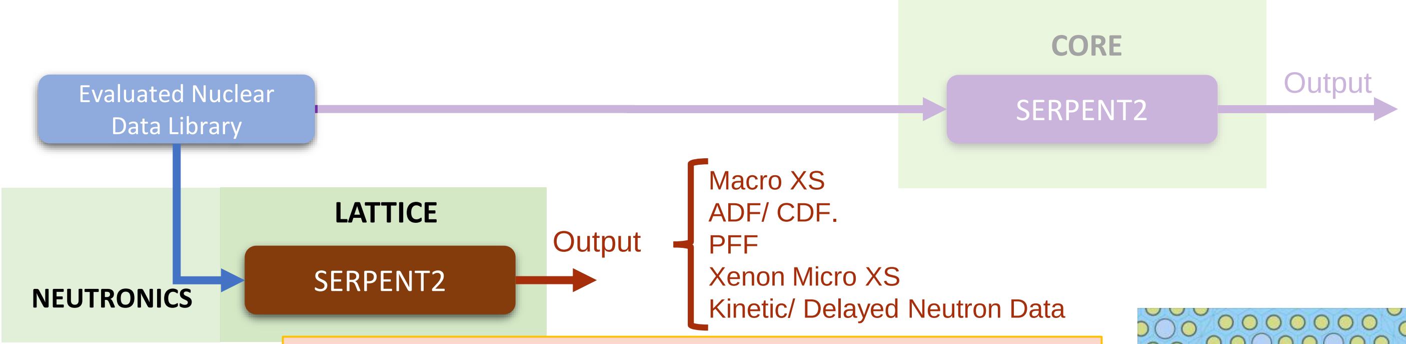
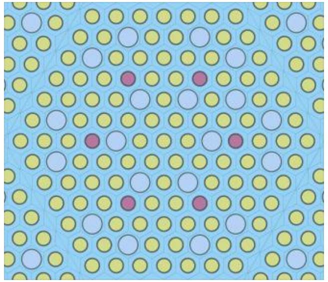
Workflow & Model: SERPENT2



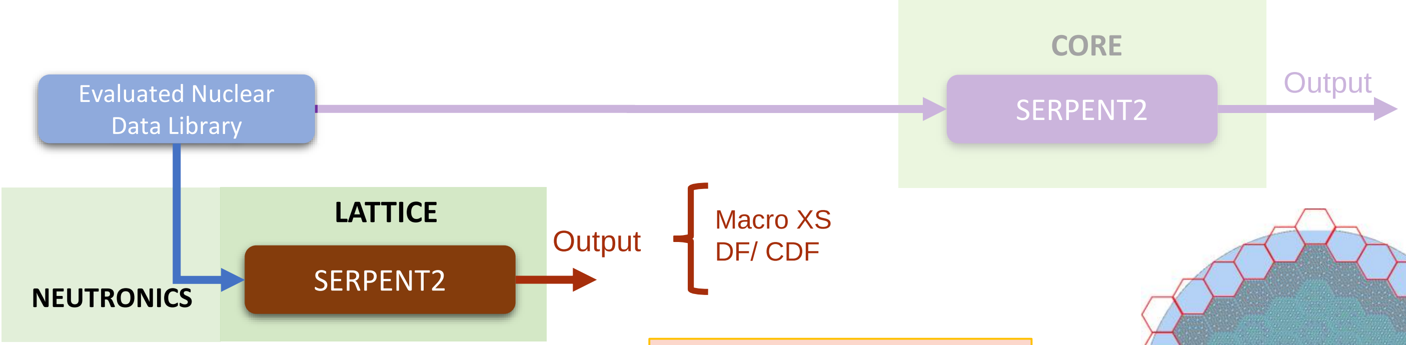
Few groups XS generation :

- Serpent options (VTT [9]):
  - Reflective BC
  - Spatial homogenization
  - Few groups energy structure (energy groups)
  - Leakage models (INF, B1, P1, FM CMM, …)
  - Discontinuity Factors (ADF, CDF, reflector DF)
  - Group-wise Form Functions (PFF) for Pin Power Reconstruction
  - Transport correction (reflector water)
  - Superimposed geometry (reflector DF)
  - Xenon micro XS



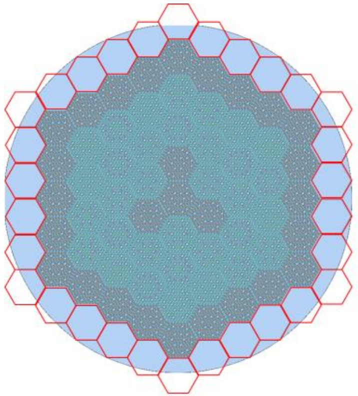
Heterogeneous FA with 6 BP pins [10]

Workflow & Model: SERPENT2



REFLECTOR

- Radial: 2D full core model with superimposed DFs
- Axial: 3D FA



Radial Reflector model [10]

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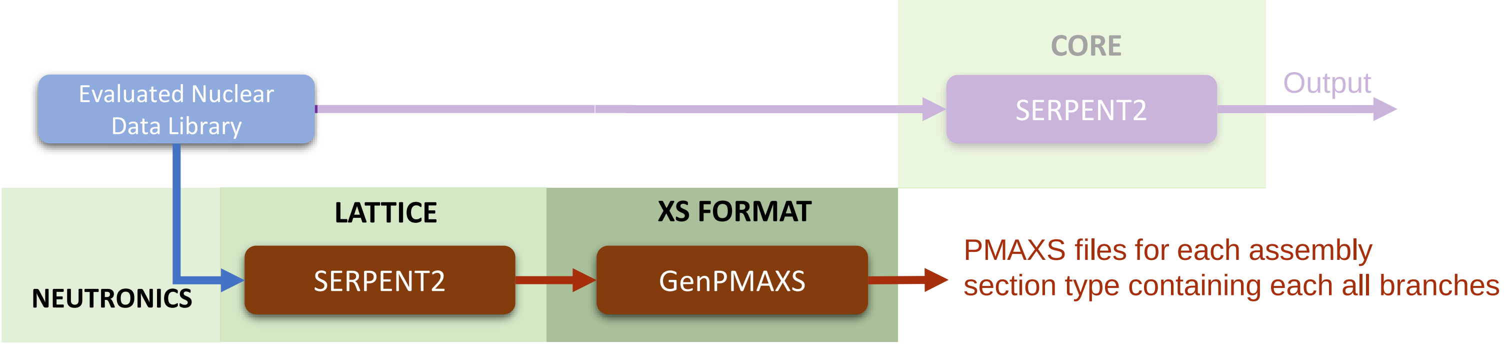
Workflow & Model: GenPMAXS



GenPMAXS v6.2/6.3 [11, 12]:

- Developed initially at Purdue University and currently at University of Michigan
- Provides the Generation of the **PMAXS files** from different codes formats (HELIOS, CASMOS, SERPENT2, …)
- The PMAXS (Purdue Macroscopic Cross Sections) files are required by PARCS nodal code

GenPMAXS v6.2/6.3 [11, 12]:

- It provides all information about the XS to be used for a given assembly/reflector configuration at different HIS, BU and TH conditions (**branches**)
  - Few groups XS as function of selected variables (Tf, Tc, CRs, Dc, Cb, Bu, His)
  - Discontinuity Factors (models for homogenous flux)
  - Group-wise Pin Form Functions
  - Fission spectrum (Chi)
  - Delayed neutrons data
  - Poisons yields and micro XS

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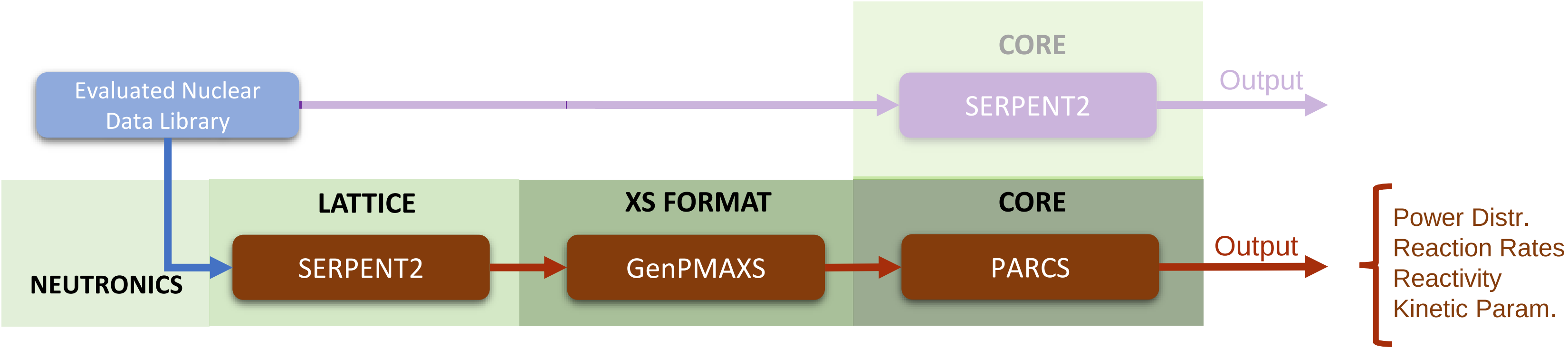
Workflow & Model: PARCS



PARCS v3.3.1 [13]:

- Purdue Advanced Reactor Core Simulator (PARCS) was developed initially at Purdue University and currently at University of Michigan.
- It was originally designed as neutronic nodal diffusion core solver but it includes now many capabilities:
  - Cartesian (FDM, ANM/NEM, NEMMG, FMDM), Hexagonal (TPEN), Cylindrical (FDM)
  - Multi-group solvers (NEM, TPEN, FDM, FMDM)
  - Eigenvalue calculations (criticality search, boron search, rod search)
  - Adjoint Kinetic parameters
  - Transient calculations
  - Diffusion and low–order transport solution
  - Equilibrium and transient Xe/Sm
  - Pin Power Reconstruction (PPR)
  - Internal TH model/ External coupling
  - Point Kinetics/ Quasi-static
  - Depletion
  - Etc.

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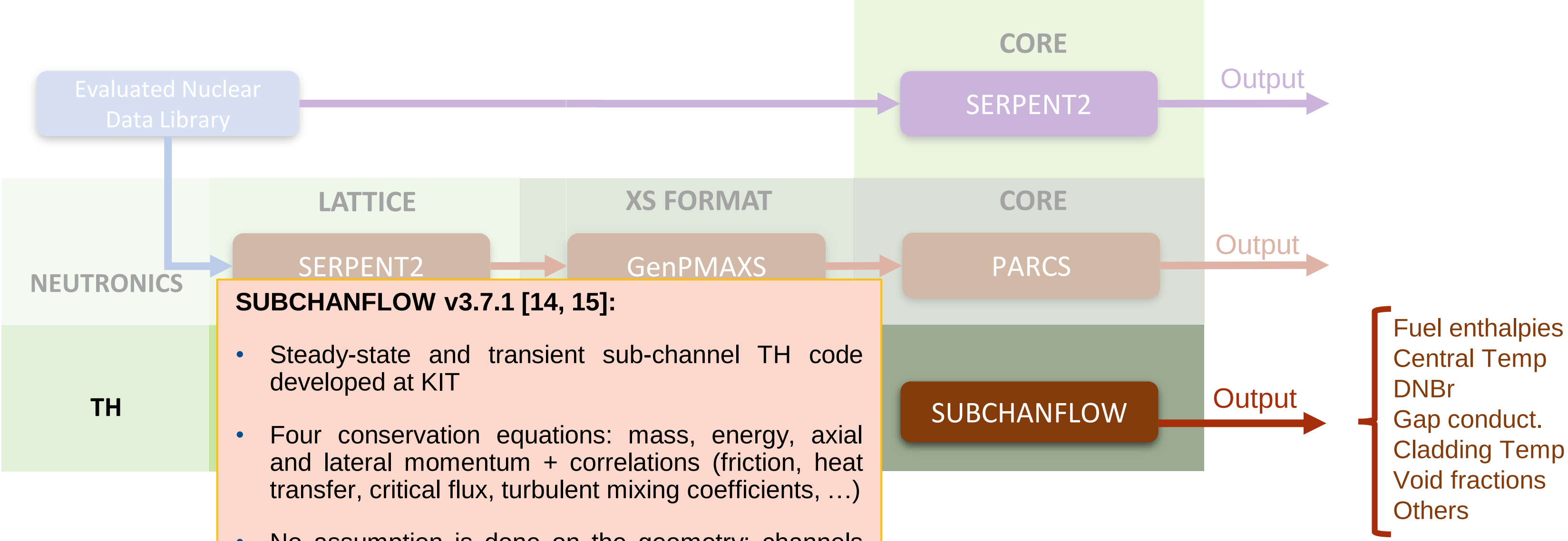
Workflow & Model: SUBCHANFLOW



SUBCHANFLOW v3.7.1 [14, 15]:

- Steady-state and transient sub-channel TH code developed at KIT
- Four conservation equations: mass, energy, axial and lateral momentum + correlations (friction, heat transfer, critical flux, turbulent mixing coefficients, …)
- No assumption is done on the geometry: channels and rods are defined with hydraulics parameters and connectivity tables
- Safety parameter calculation

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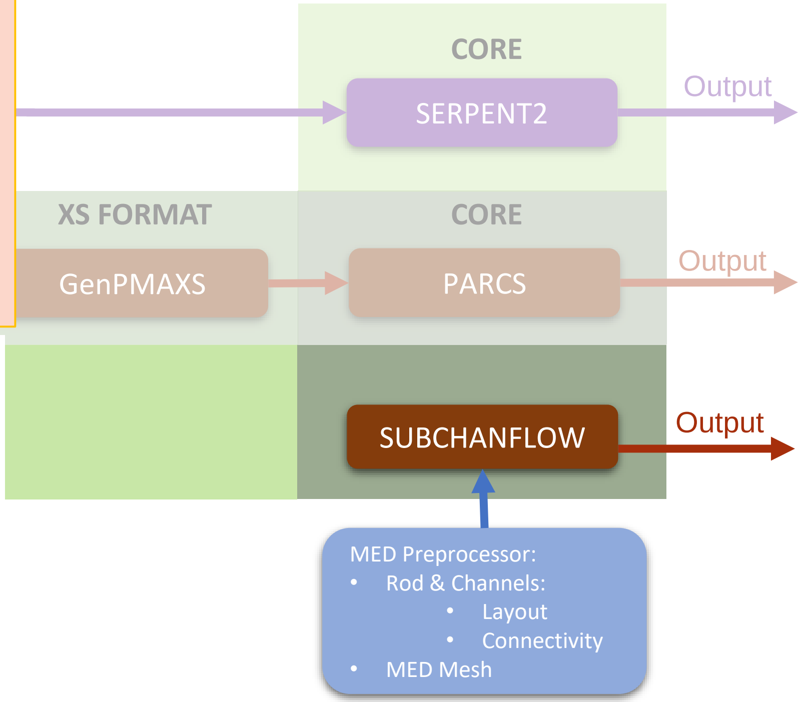
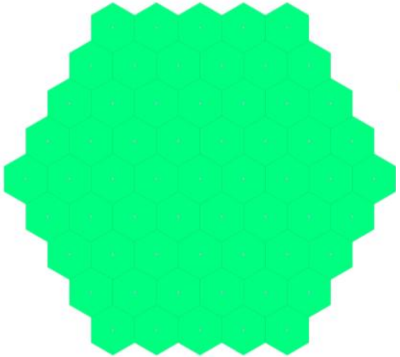
Workflow & Model: Preprocessor

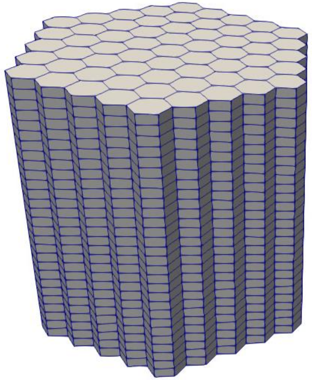
MED Preprocessor [16]:

- Developed at KIT by M. Garcia
- Generates files for couplings (PARCS-SCF, SERPENT2-SCF)
  - Rod/channels layout/connectivity
  - MED mesh [17]
- For PARCS-SCF:
  - Rod equivalent model: average thermal behavior
  - 1 channel with one rod per node + axial discretization





CAREM-like preprocessor plot



CAREM-like MED mesh

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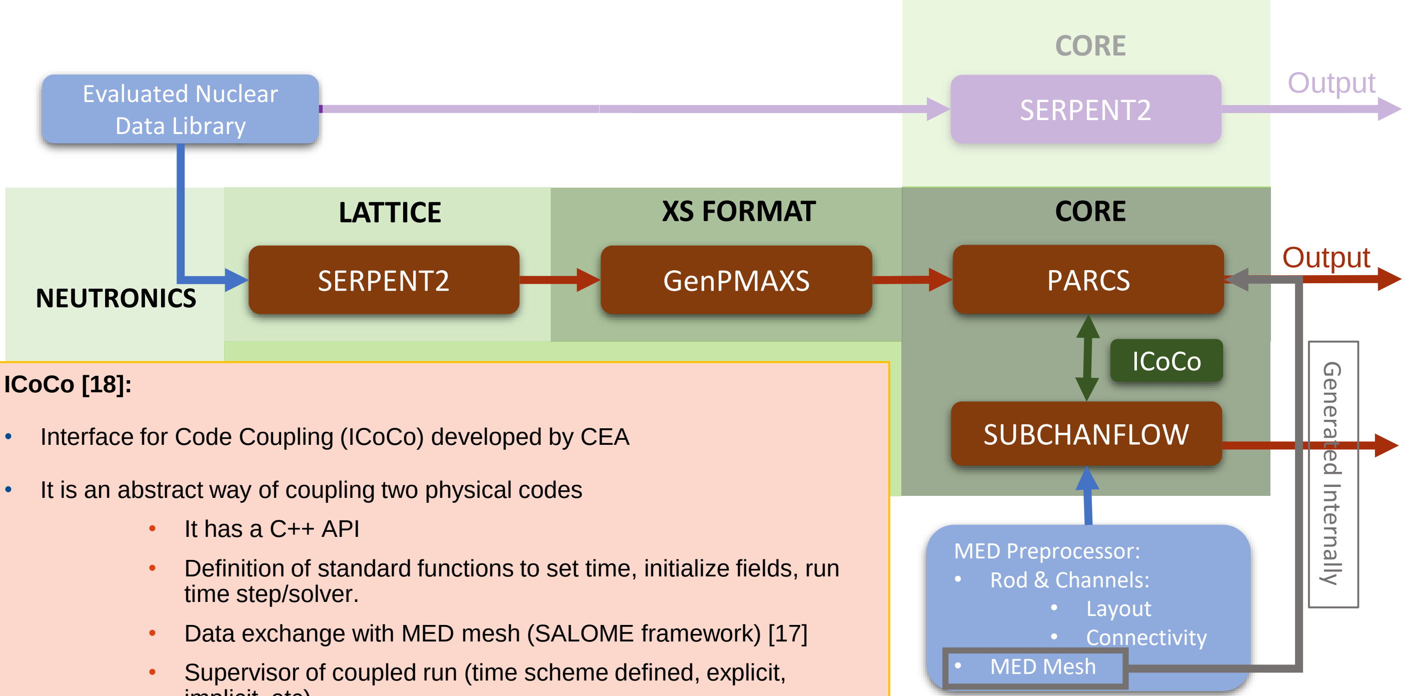
Workflow & Model: PARCS-SCF ICoCo



ICoCo [18]:

- Interface for Code Coupling (ICoCo) developed by CEA
- It is an abstract way of coupling two physical codes
  - It has a C++ API
  - Definition of standard functions to set time, initialize fields, run time step/solver.
  - Data exchange with MED mesh (SALOME framework) [17]
  - Supervisor of coupled run (time scheme defined, explicit, implicit, etc)

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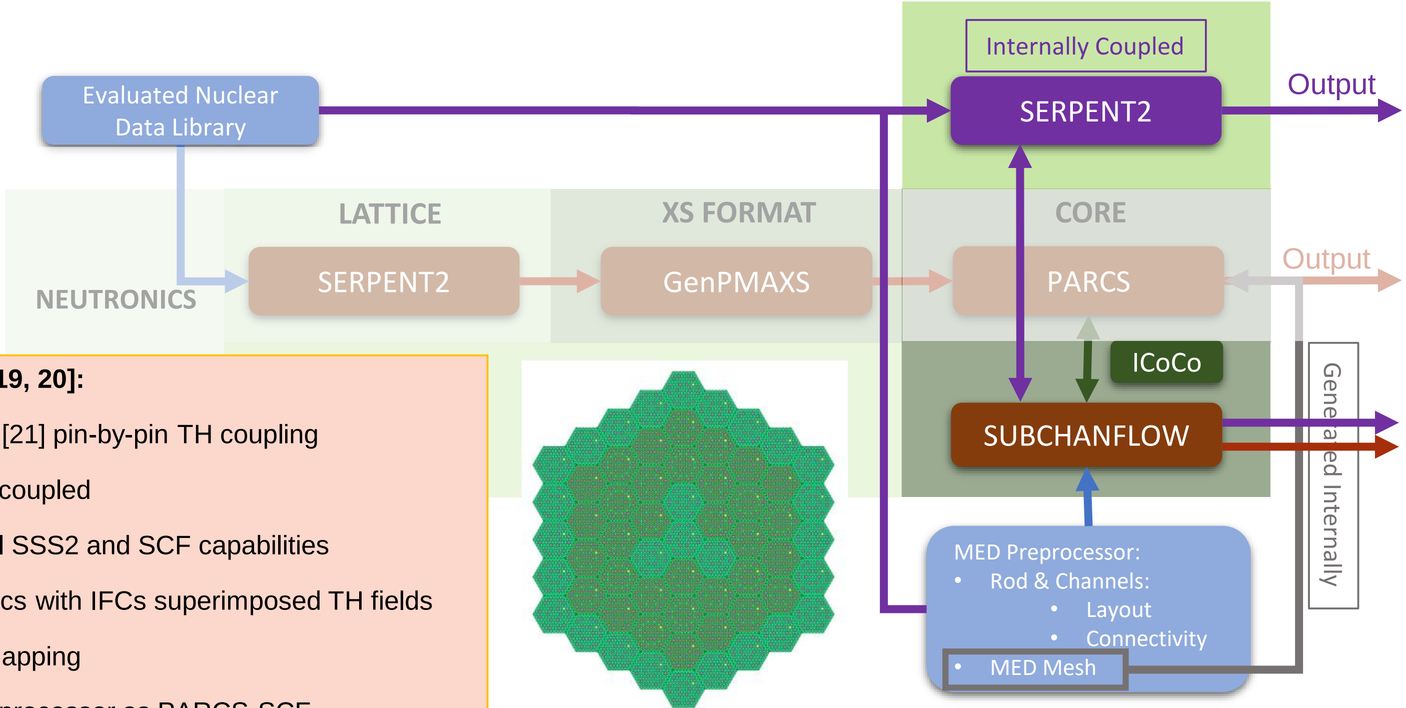
Workflow & Model: SSS2-SCF



SSS2-SCF [19, 20]:

- MCSAFE [21] pin-by-pin TH coupling
- Internally coupled
- Inherits all SSS2 and SCF capabilities
- Multiphysics with IFCs superimposed TH fields
- Internal mapping
- Same preprocessor as PARCS-SCF

Pin-wise CAREM-like preprocessor plot

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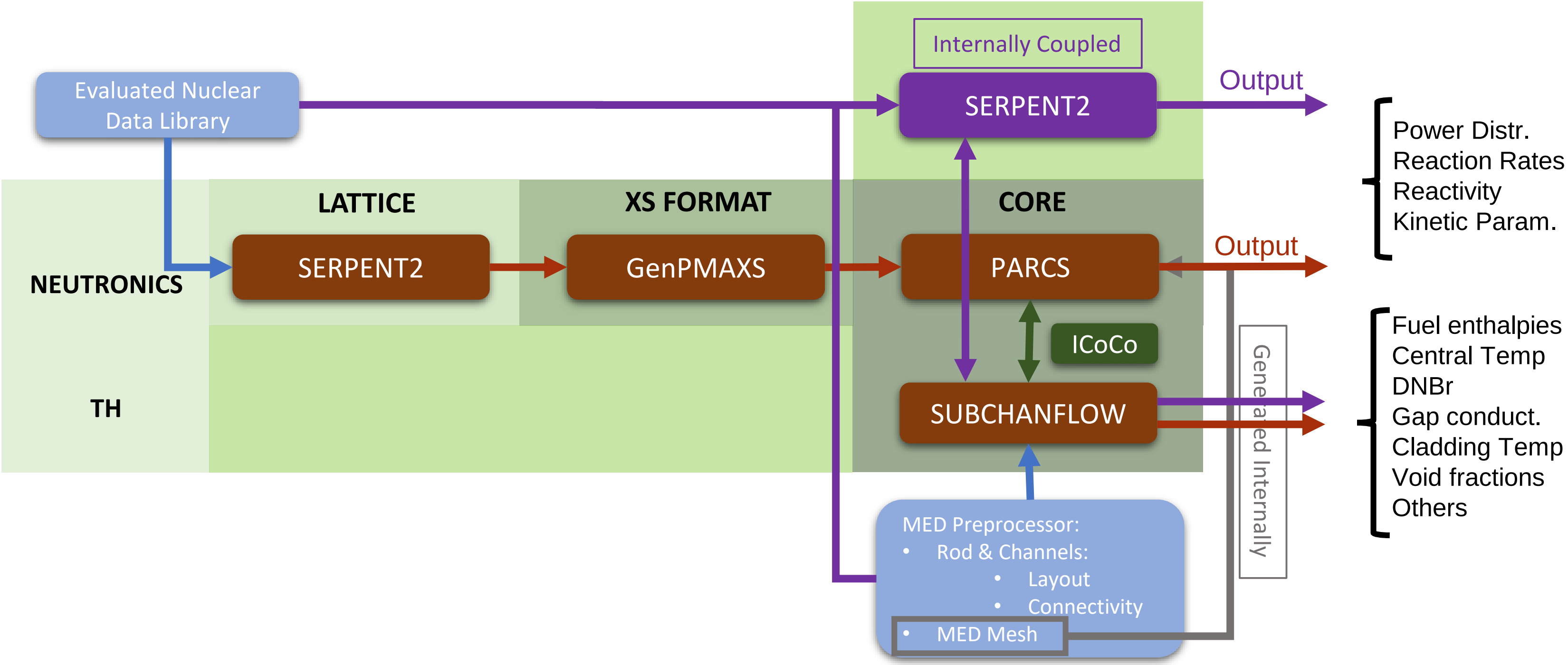
Workflow & Model: Summary SERPENT/PARCS-SCF



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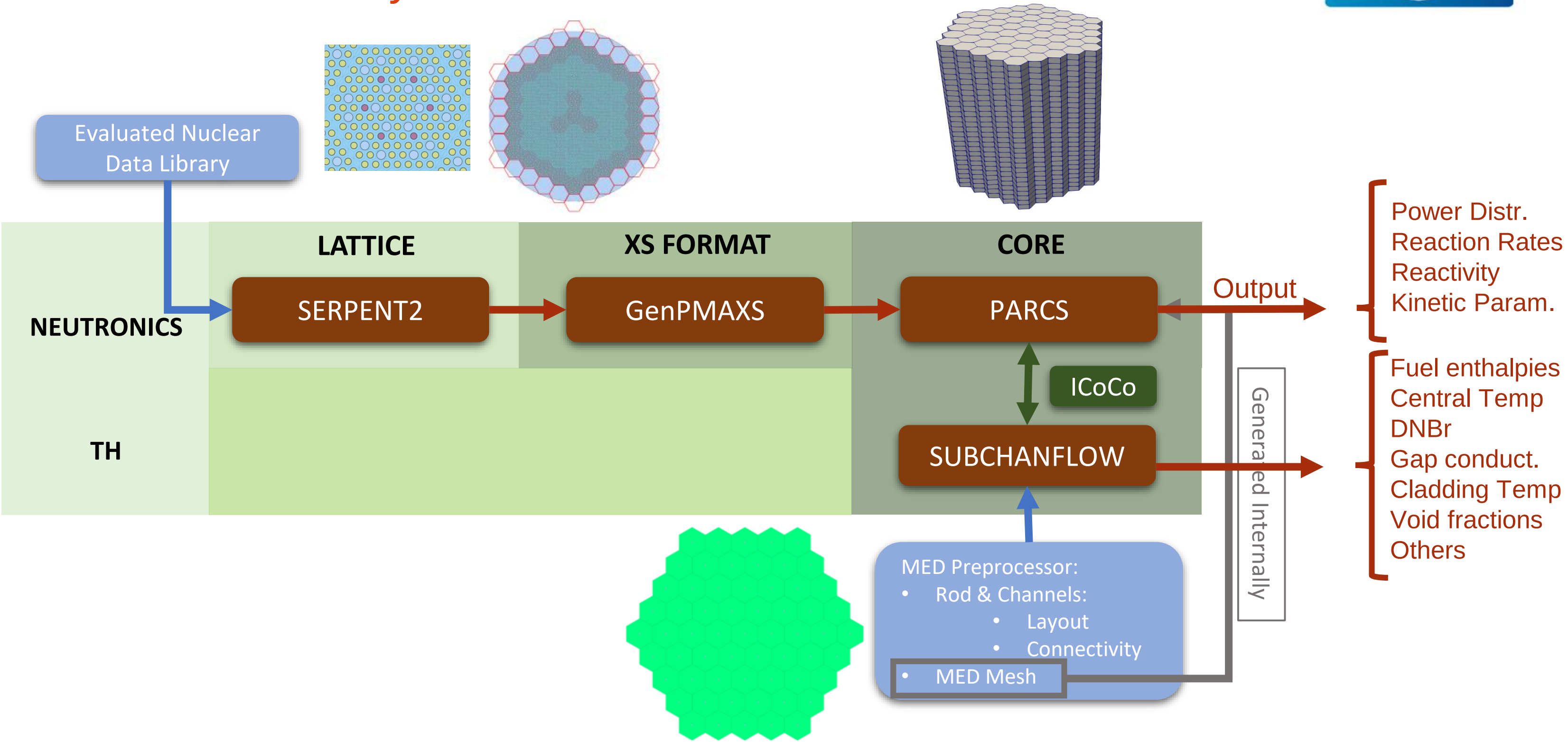
Workflow & Model: Summary PARCS-SCF model



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4. Results

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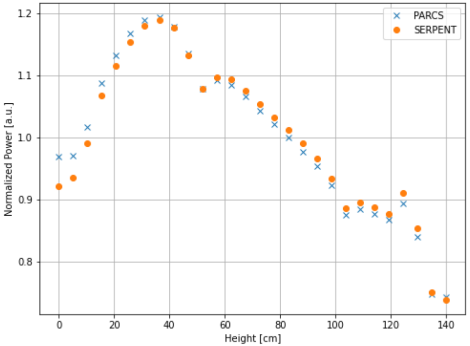
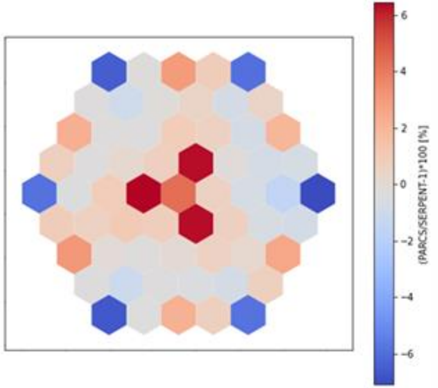
Comparison SERPENT2 / PARCS-ICoCo

PARCS-ICoCo -SERPENT2 table for critical HFP configuration (rods inserted).

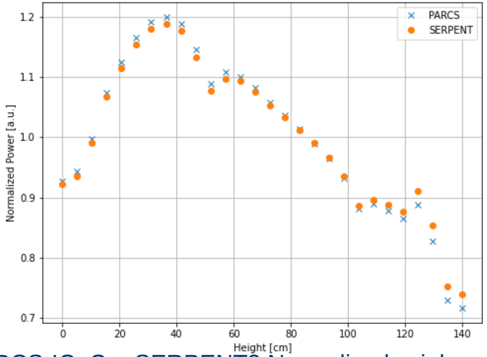
EG	Leakage Model	Delta Rho	Axial Power		FA Power	
		[pcm]	RMS [%]	MAX [%]	RMS [%]	MAX [%]
2	CMM	57	1.68	5.10	2.68	7.08
2	INF	-291	2.49	6.36	2.30	6.17
2	INF+CMM	-668	2.70	6.35	2.49	7.13
4	CMM	56	1.33	3.06	2.92	7.60
4	INF	66	0.93	2.11	2.65	6.91
4	INF+CMM	-227	0.75	1.75	2.65	7.22
8	CMM	84	1.17	3.06	2.74	7.09
8	INF	193	0.97	2.46	2.65	6.59
8	INF+CMM	-82	0.79	2.09	2.58	6.88



PARCS-ICoCo -SERPENT2 Normalized axial power comparison for critical HFP configuration using FM CMM leakage model, DFs , equilibrium Xe and TRC with 2 energy groups.



PARCS-ICoCo -SERPENT2 axially integrated FA normalized radial power relative difference using FM CMM leakage model, DFs, equilibrium Xe and TRC with 2 groups.



PARCS-ICoCo -SERPENT2 Normalized axial power comparison for critical HFP configuration using FM CMM leakage model, DFs , equilibrium Xe and TRC with 4 energy groups.

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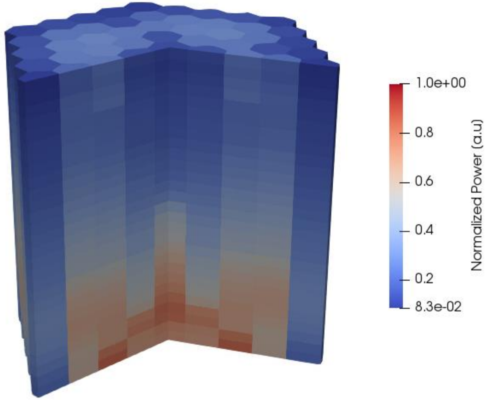
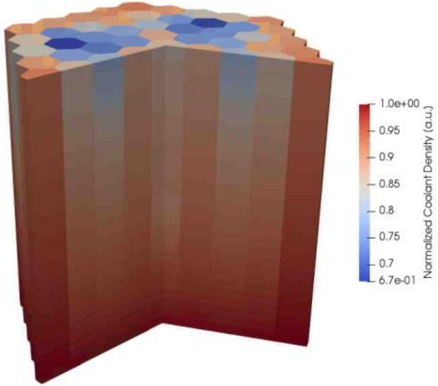
Steady-state PARCS-SCF

- Hot Full Power (HFP)
- Equilibrium Xenon
- No boron
- Critical CRs position

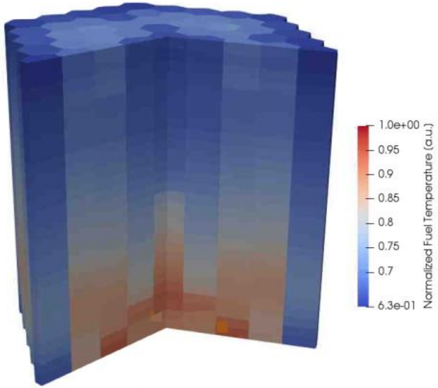
Normalized Power [a.u]



Normalized Coolant Density [a.u.]



Normalized Fuel Temperature [a.u.]



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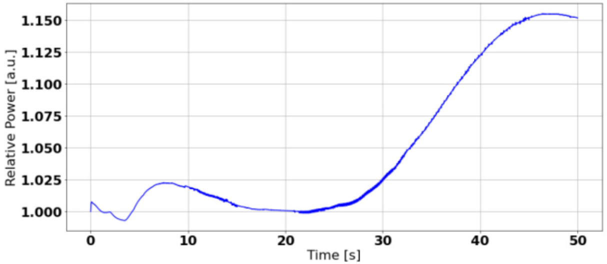
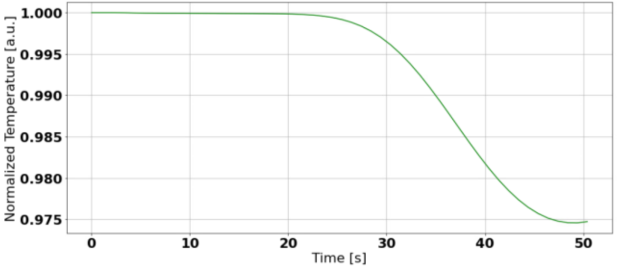
Transient PARCS-SCF

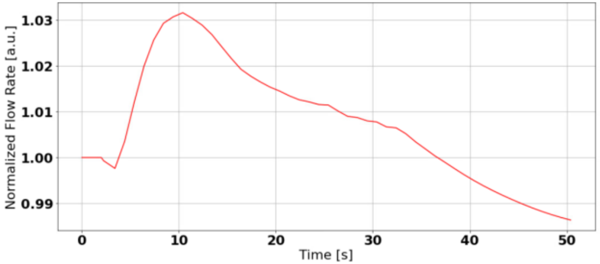
- Steady-state
  - Hot Full Power (HFP)
  - Equilibrium Xenon
  - No boron
  - Critical CRs position
- Transient
  - Cold front originated in the secondary
  - Outlet pressure, inlet coolant temperature and mass flow values during 50 seconds
  - Transient Xenon

Normalized Power [a.u.]

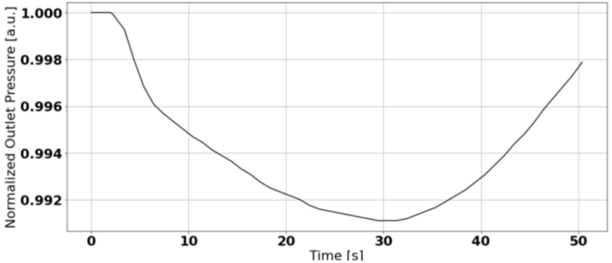


Normalized Inlet Temperature [a.u.]





Normalized Flow Rate [a.u.]



Normalized Outlet Pressure [a.u.]

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5. Conclusions/ Limitations / Future work

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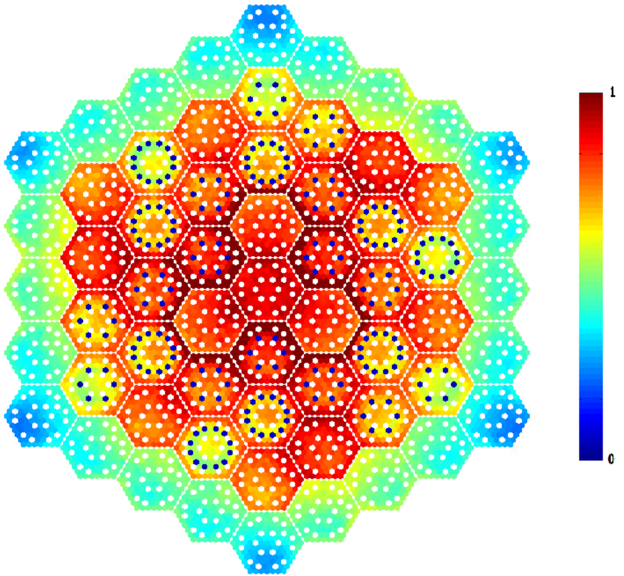
Conclusions/ Limitations / Future work

- Good agreement (Reactivity, power distribution) between SERPENT2 and PARCS-ICoCo model taking into account calculation time
- Influence of number of energy groups, DFs, leakage models, TRC
- TPEN homogeneous solution needed for DFs calculation
- Corner Discontinuity Factors are used by PARCS-ICoCo to improve TPEN solution → Convergence issues
    - ZDF for axial heterogeneity?
- Pin Power Reconstruction not available for hexagonal solution → useful to compare local parameters
- SERPENT2-SCF model will be used to compare at least Steady-state conditions → High computation demand → HPC needed



CAREM-like SERPENT2-SCF model (not actual core) at HFP ARO conditions

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Thank you for your attention!

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F-SMR core analysis

Anthime FARDA (CEA)

22/03/22

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F-SMR core analysis

- The F-SMR core
- Cold water insertion transient
- Calculation scheme used
- Transient analysis

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The F-SMR core

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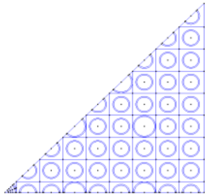
The F-SMR core

- Not a real industrial reactor
    - Core designed by genetic algorithms
- No identified link to the system
    - Boundary conditions are not representative
    - Simplifications can be made on the core design
- Some basic specifications :
    - Use of standard PWR assemblies
    - Boron free core
        - need of poisoning
        - High number of control rods
        - Important control rod insertion
    - Very heterogeneous core



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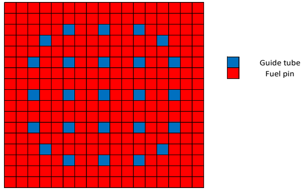
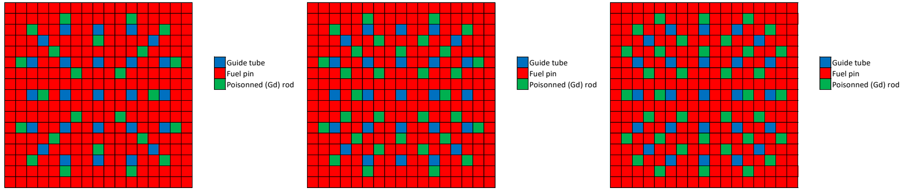
The F-SMR core

- Single batch loading
  - 54 types of assemblies
  - Axially heterogeneous
    - Different ^{235}U enrichment
    - 4 different positions for poisonned rods :
      - No Gd pin
      - 24 Gd pins
      - 32 Gd pins
      - 36 Gd pins
    - Different Gd content in each pin
  - All fresh assemblies at BOC





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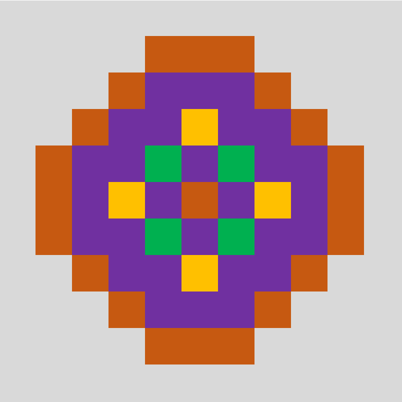
The F-SMR core

- Core loading
  - 57 assemblies with ¼ core symetry



Core top

Core bottom

enr	nbGd	TGd
4,95	0	0
4,95	0	0
4,95	0	0
4,95	0	0
4,95	0	0
4,95	0	0
4,95	0	0
4,95	0	0

enr	nbGd	TGd
4,95	0	0
4,95	32	10
4,95	32	10
4,95	32	10
4,95	32	10
4,95	32	10
4,95	32	10
4,95	32	10

enr	nbGd	TGd
4,95	24	6
4,95	24	6
4,95	24	6
3,50	24	6
3,50	24	6
3,50	24	6
3,50	24	6
3,50	24	6

enr	nbGd	TGd
4,70	36	10
4,70	36	10
4,70	36	10
4,70	36	10
4,70	36	10
4,70	36	10
4,70	36	10
4,30	36	8

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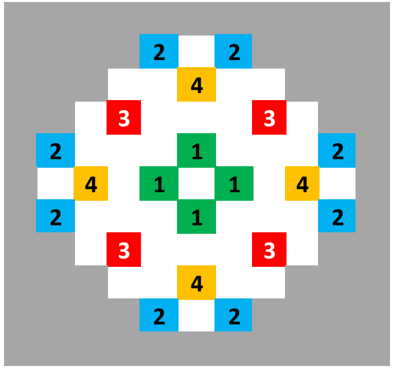
The F-SMR core



- Reactivity management of the core is assured by control rods
  - Every assembly has a control rod
    - Insertion from the top
  - CR are divided into 5 groups
    - Group 1
    - Group 2
    - Group 3
    - Group 4
    - Safety CR

  - Insertion of a group of CR starts after previous one has reached the bottom of the core

  - Falling time in case of SCRAM is set to 1 s

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The F-SMR core

- Core thermal hydraulics

Parameter	Nominal operation value
Core power	300 MWth
System pressure (core outlet)	15 MPa
Core flow rate	2145.8 kg/m²/s
Core inlet temperature	553 K

- Some reactivity coefficients

Reactivity coefficient	rod position	coefficient (pcm/°C)
moderator coefficient	ARO	-49,07
	ARI	-97,42
	Critical	-60,01
Doppler coefficient	ARO	-2,32
	ARI	-3,15
	Critical	-2,53

rod insertion	keff	reactivity (pcm)
ARO	1,08582	7904
ARI	0,74765	-33752
G1 to G4 in	0,99825	-175

rod inserted	Rod worth
All	41656
G1 to G4	8079

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Cold water insertion transient

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Cold water insertion transient

- Negative moderator coefficient ensure reactor safety
    - But…

- Very highly negative moderator coefficient
    - Cold water insertion leads to important reactivity insertion
    - Resulting in power increase
    - Leading to SCRAM…

- The transient is modelled as a modification of temperature in core inlet
    - Inlet temperature drops from 280° to 150°C
    - Reactivity insertion is around 6300 pcm

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Cold water insertion transient

- Transient timeline
  - At t<0: The core is at BOC in critical state.
  - At t=0: Inlet water temperature drops in 0.1 s from 280°C to 150°C
  - SCRAM is activated when the reactor power reaches 105% Pnom with a delay of 0.01 s.
  - Control rods used for core reactivity control are fully inserted at t=0.55 s.
  - Safety control rods are fully inserted at t=1.15 s. Core is stabilized and subcritical before the complete insertion of control safety control rods.

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Calculation scheme

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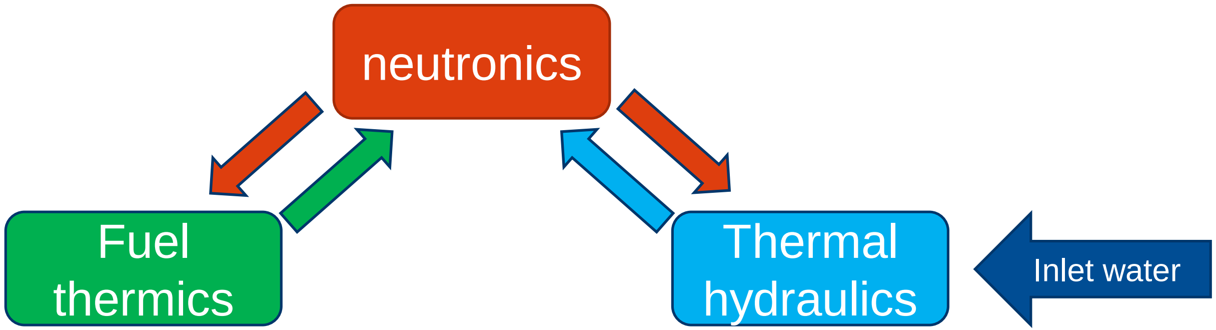
Calculation scheme

- Cold water insertion transient only implies core physics :
    - Neutronics
    - Core thermalhydraulics
    - Fuel thermics



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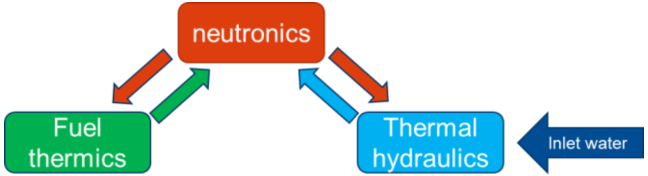
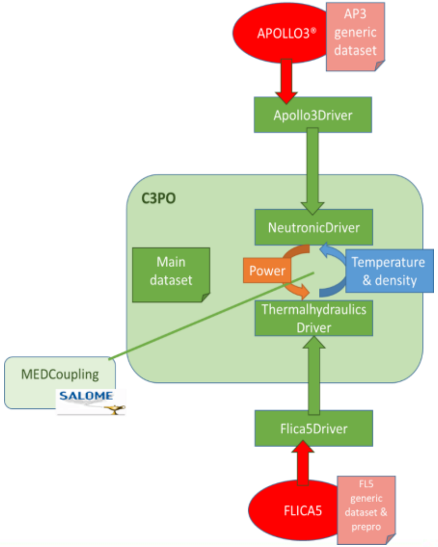
Calculation scheme

- CEA current codes for reactor analysis are
  - Neutronics : APOLLO3®
  - Core thermalhydraulics : FLICA5
  - Fuel thermics : FLICA5



- Coupling between codes is performed through an ICoCo interface via the C3PO framework
  - Each code exhibits a python ICoCo Interface
  - For each code, a PhysicsDriver is written into C3PO
  - Coupling takes the form of a C3PO script in python
  - Exchanges between codes can be
    - Scalar values getValue()/setValue()
    - Field exganges uses MEDCoupling tools : getMEDField()/setMEDField()
  - Treatment of the values happens in the exchanger class.



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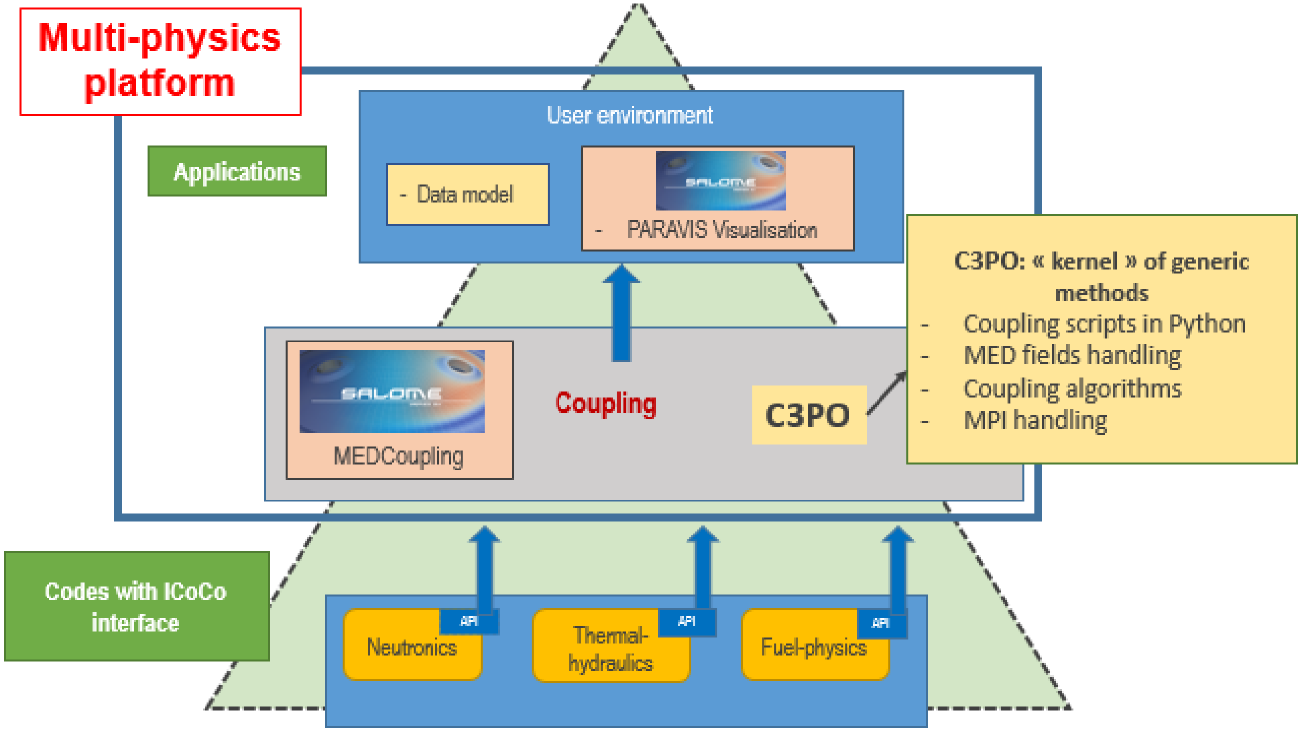
Calculation scheme



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Calculation scheme

- A coupling is written is the coupler class
    - Variation of inlet temperature
    - SCRAM and modification of control rod positions
    - Field and data exchanges
    - March in time

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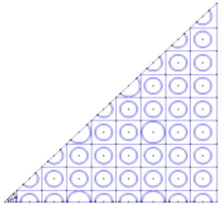
Calculation scheme

- **FOCUS on the neutronics**
- APOLLO3® 2 steps calculation
  - 2D Lattice APOLLO3® leads to the production of a MPO
    - 281 groups energy mesh PIJ
    - 30 groups MOC calculation
    - Condensation to 2 energy groups
    - Homogenization at the assembly level



  - Equivalence SPH
    - 281 groups energy mesh
  - 3D Core calculation is performed by the MINOS solver of APOLLO3®
    - 2 groups energy mesh
    - 4 meshes per assembly

Fuel temperature [°C]	Moderator temperature [°C]	Moderator density [g/cm^3]	Burnup [MWd/t]
10	10	1.00670	0
20	20	1.00504	1007
30	30	1.00215	2013
280	125	0.94239	4026
650	200	0.86904	8051
800	280	0.76464	12075
1200	311	0.70246	16098
2700	324	0.66808	20121
	344	0.60225	30143
	356	0.48857	40193
			49768
			59811
			69852
			79891
			89930

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Calculation scheme

- **FOCUS on the neutronics**
- Validation at the cell scale against a reference TRIPOLI-4® calculation for each type of 2D assembly

Table 9: Summary of the comparison on the pin-by-pin fission rates.

2D Assembly	Max σ T4	Min (A2-T4)/T4	Max (A2-T4)/T4	RMS (A2-T4)/T4
1	0.08%	-0.45%	0.43%	0.22%
2	0.15%	-1.46%	0.54%	0.45%
3	0.16%	-1.34%	0.73%	0.49%
4	0.16%	-1.92%	0.73%	0.54%
5	0.17%	-1.25%	0.92%	0.51%
6	0.17%	-1.43%	0.90%	0.49%

Table 10: Summary of the comparison of pin-by-pin absorption rates.

2D Assembly	Max σ T4	Min (A2-T4)/T4	Max (A2-T4)/T4	RMS (A2-T4)/T4
1	0.07%	-0.52%	0.87%	0.24%
2	0.08%	-0.97%	0.70%	0.40%
3	0.08%	-0.95%	1.04%	0.44%
4	0.08%	-0.98%	0.78%	0.45%
5	0.09%	-0.92%	1.03%	0.40%
6	0.09%	-0.95%	1.19%	0.41%

2D-assembly	Enr [%]	nbGd [-]	TGd [-]	TRIPOLI-4 k_{eff} [-] (±σ [pcm])	APOLLO3 k_{eff} [-]	Reactivity difference [pcm]
1	4.95	0	0	1.43844	1.43844	0
2	4.95	24	6	1.16806	1.17028	162
3	3.5	24	6	1.06470	1.06619	131
4	4.95	32	10	1.07072	1.07333	227
5	4.7	36	10	1.01577	1.0175	167
6	4.3	36	8	1.00048	1.00212	164

0.4%	0.0%	0.0%	0.2%	0.0%	-0.2%	-0.2%	-0.2%	0.2%
0.1%	0.2%	0.2%	0.3%	-0.1%	-0.2%	0.4%	-0.2%	-0.2%
	-0.2%	0.1%		0.0%	0.0%	0.1%	-0.5%	-0.2%
0.2%	0.0%	0.4%	0.3%	0.0%		0.0%	-0.2%	-0.2%
-0.3%	0.2%	-0.2%	0.2%	0.4%	0.0%	0.0%	-0.1%	0.0%
	-0.2%	0.0%		0.2%	0.3%		0.3%	0.2%
-0.2%	0.2%	0.3%	0.0%	-0.2%	0.4%	0.1%	0.2%	0.0%
-0.2%	0.2%	0.2%	-0.2%	0.2%	0.0%	-0.2%	0.2%	0.0%
	-0.2%	-0.2%		0.4%	0.2%		0.1%	-0.4%

0.5%	0.3%	0.2%	0.3%	0.2%	0.1%	0.1%	0.1%	0.9%
0.1%	0.1%	0.1%	0.1%	-0.1%	-0.2%	0.2%	-0.2%	0.1%
-0.4%	-0.2%	-0.1%	-0.4%	-0.1%	-0.1%	0.0%	-0.3%	0.1%
0.1%	0.0%	0.2%	0.0%	-0.1%	-0.5%	-0.1%	-0.3%	0.1%
-0.3%	0.0%	-0.1%	0.0%	0.2%	0.0%	-0.1%	-0.2%	0.2%
-0.4%	-0.3%	-0.1%	-0.4%	0.0%	0.0%	-0.4%	0.1%	0.3%
-0.3%	0.1%	0.3%	-0.1%	-0.2%	0.2%	-0.2%	0.1%	0.3%
-0.2%	0.1%	0.1%	-0.2%	0.1%	-0.1%	-0.2%	0.1%	0.3%
-0.4%	-0.3%	-0.4%	-0.4%	0.1%	0.0%	-0.4%	-0.1%	-0.1%

Figure 1: Assembly 1 (northeast quarter), relative discrepancies on the fission (left) and absorption (right) reaction rates between TRIPOLI-4® and APOLLO3® simulations.

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Transient analysis

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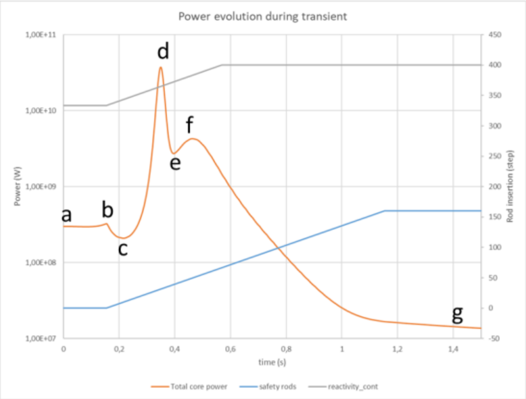
Transient analysis

- Transient timeline
  - At t<0: The core is at BOC in critical state.
  - At t=0: Inlet water temperature drops in 0.1 s from 280°C to 150°C
  - SCRAM is activated when the reactor power reaches 105% Pnom with a delay of 0.01 s.
  - Control rods used for core reactivity control are fully inserted at t=0.55 s.
  - Safety control rods are fully inserted at t=1.15 s. Core is stabilized and subcritical before the complete insertion of control safety control rods.



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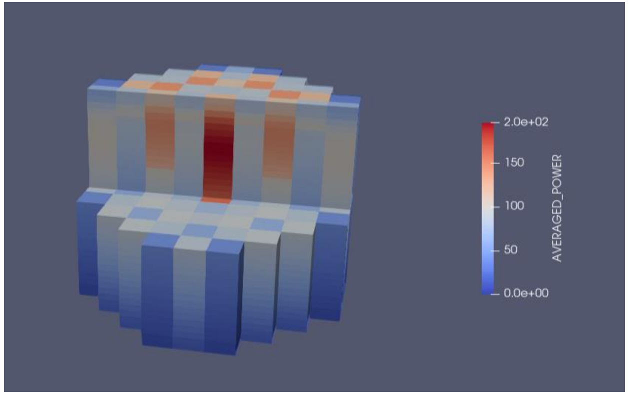
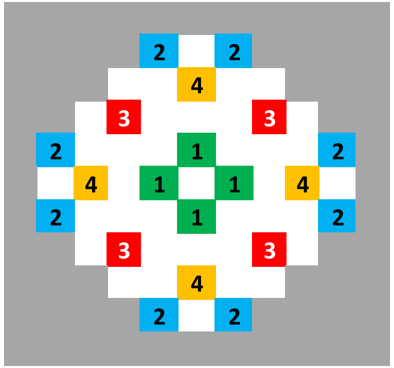
Transient analysis

- for t<0 s, the core is in critical state

- Criticality is obtain by fixing the position of control rods
    - Group 1, 2 and 3 are fully inserted
    - Group 4 is inserted at 33 steps





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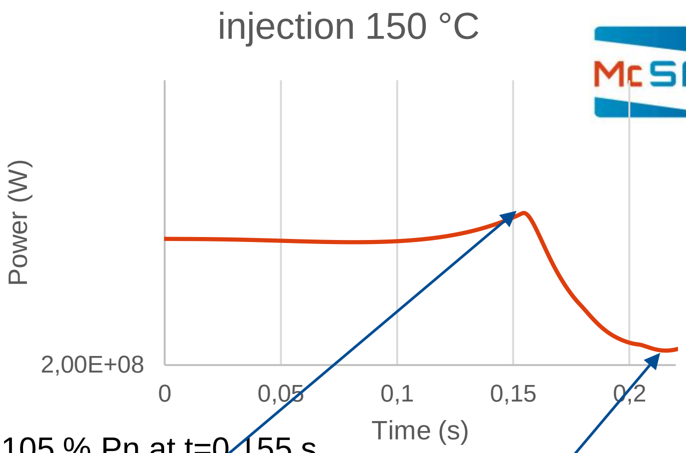
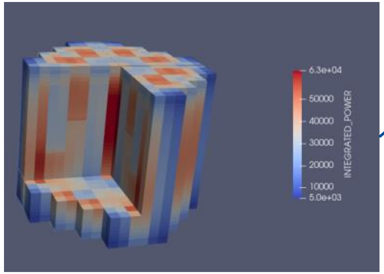
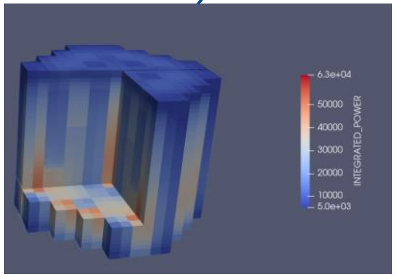
Transient analysis

- At t=0 s, water inlet temperature drops from 280°C to 150°C
- Reactivity increases leading to power increase
- SCRAM is activated on a high power signal, when power reaches 105 % Pn at t=0.155 s
  - For conservative reasons, a delay of 0.01 s is applied before effective control rod insertion
- During the first instant of the CR insertion, power decreases







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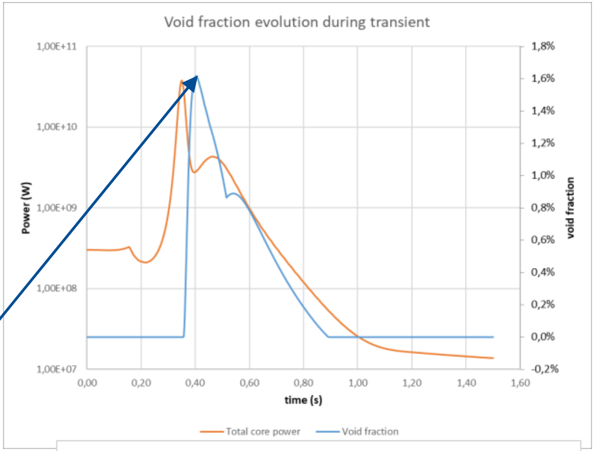
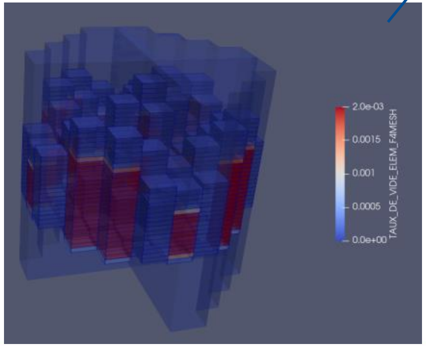
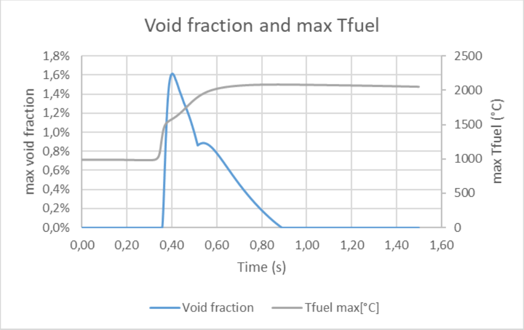
Transient analysis

- As cold water front advances, counter-reaction has a higher contribution
- Fuel temperature increases to 2000 °C leading to an important Doppler effect
- Higher temperature lead to a small boiling limiting the local moderation
- A second power increase is started







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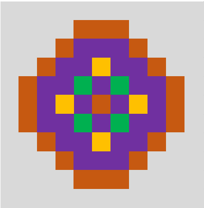
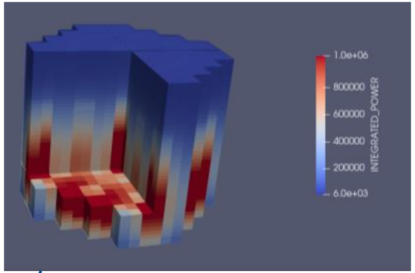
Transient analysis

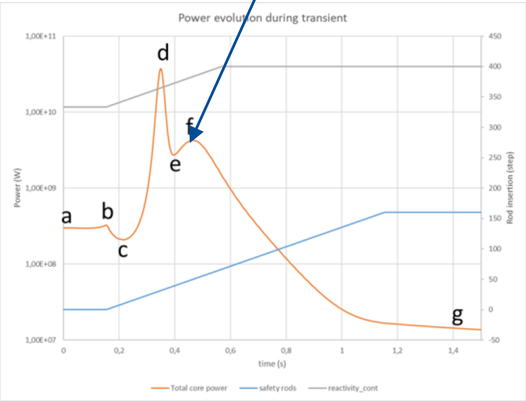




- Control rod insertion progresses
    - Reactivity management rods advances
    - Safety control rods arrive in the higher efficiency area

- Most of the power is located in the lower part of the core

- Fuel temperature is also higher in this area
    - Especially in non poisoned assemblies (central and periphery)



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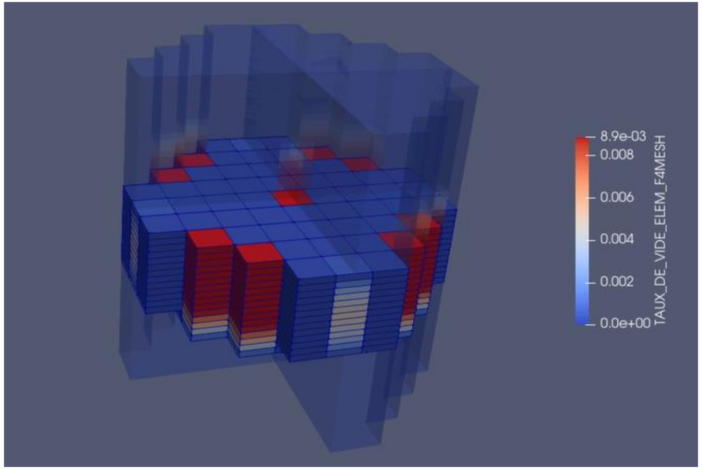
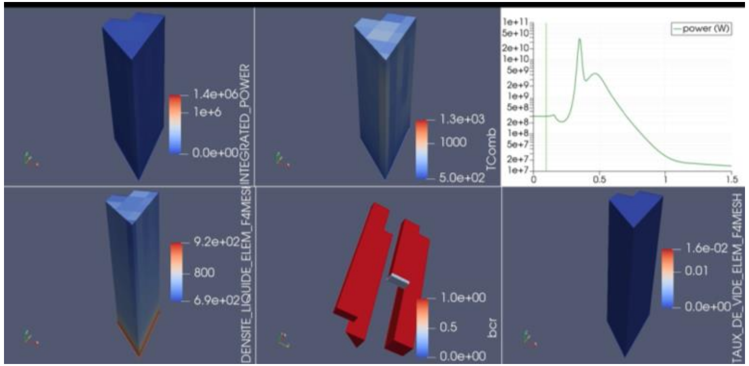
Transient analysis

- Videos illustrating the transient
    - **Progression of void fraction**



    - **Summary of the transient**



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Thank you for your attention !

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NuScale analysis with TRACE

2nd McSAFER Training Course

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Cesar QUERAL / cesar.queral@upm.es

Universidad Politécnica de Madrid, UPM

March 22-24.2022 in Lappeenranta, Finland

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Outline

- Introduction. Main Characteristics of Small Modular Reactors
- SMR modeling challenges with system codes
- Experimental facilities related to SMR
- Application: NuScale Power Module
  - NuScale Power Module. Fundamentals
  - TRACE modeling capabilities applied to SMRs
  - Modeling NuScale with TRACE
  - Transient analyses performed within the McSAFER project
- Final remarks

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Introduction.
Main Characteristics of Small Modular Reactors

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Introduction. Main characteristics of SMRs

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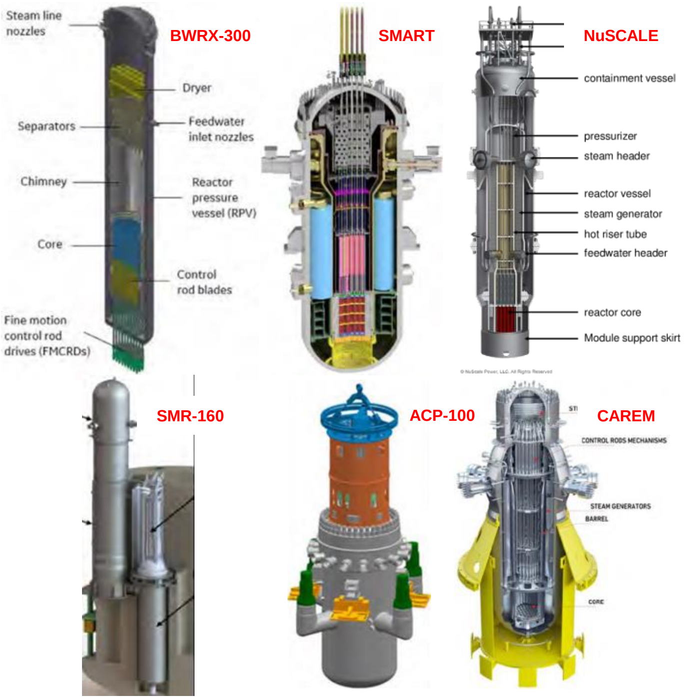
The main characteristics of the SMRs can be summarized as follows:

- **Integral** design for the RPV (RCS is integrated within the RPV)
- **Forced convection** (SMART, ACP-100, mPower,) or **Natural circulation** (NuScale, CAREM, SMR-160) for establishing the flow to cool the core.
- Different concepts of **helically coiled SGs** (NuScale, SMART, CAREM).
- Certain designs such as CAREM, SMART, SMR-160, F-SMR… consider the possibility of deploying a **boron free core** design. Others are operated using a certain amount of **Boron diluted** in the RCS (W-SMR, NuScale, SMART, ACP-100)
- Most of the SMR designs relies on **passive safety systems** to deal with **accidental conditions** and with the **decay heat removal**.



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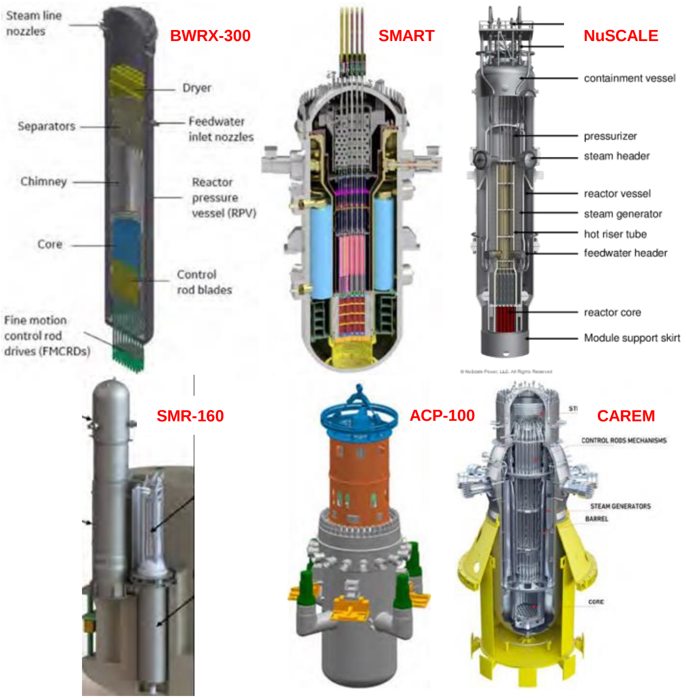
Introduction. Main characteristics of SMRs

As a direct consequence of the peculiarities of the SMR designs, certain events are considered to be precluded such as:

- RPV integral designs eliminate the possibility to have a **LBLOCA**
- **LOFA** (Loss Of Flow Accidents) events are precluded due to the elimination of pumps in the RCS (Natural circulation)
- **RIA** due to a rod ejection event is avoided because of the new internal control rod drive mechanism designs (F-SMR, CAREM or mPower)



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SMR modeling challenges with system codes

SMR modeling challenges with system codes

- **Typical SMR phenomena** challenging system codes can be summarized as:
  - Possibility of having no preferential flow direction in certain regions because of the integral RPV design
  - Components with special geometries (tube bank arrangement of SG tubes, flow mixing devices…)
  - Natural circulation, flow mixing and oscillatory flows
  - Passive safety systems

- Remarkable **thermal-hydraulic issues** in **system codes**:
  - 3D fine meshes of complex geometries and flow mixing due to turbulence modeling
  - Specific heat transfer and friction coefficients are needed for helical SGs and other complex geometries
  - Stability boundaries must be analyzed in several LWR-SMR designs
  - Passive DHRS based on condensation heat transfer and flow-maps interrelationship
  - Influence of non-condensable gases on the condensation process
  - Comprehensive experimental database for code validation

- **Multi-physics and multi-scale tools** may be used to improve the accuracy of the calculations

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Experimental facilities related to SMR

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Experimental facilities for code validation

SMR Design	Experimental facility	Tests
NuScale	MASLWR → NIST → NIST-1	Large scale with integral-effects data for SBLOCAs, long-term core cooling, and high-pressure condensation data.
	GEST facility in SIET SIET TF-2	Validating inner/outer surface heat transfer and pressure drop models for helical SGs
	Stern Laboratories	NuScale CHF testing
	AREVA Richland Test Facility (RTF)	NuScale Fuels testing
	AREVA KATHY loop	NuScale CHF testing
CAREM	CAPCN	Integral Test Facility: 1:1 in height and pressure. Natural Circulation and Self-pressurization facility constructed and operated to study the thermo-hydraulic dynamics in conditions similar to CAREM-25 operational states.
	Low Pressure Facility	Characterizing friction losses and flow induced vibration.
	TH LAB IPPE	Thermal limits and CHF
	CNEA-CAB CHF Freon Facility	Thermal limits and CHF

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Experimental facilities for code validation

SMR Design	Experimental facility	Tests
SMART	VISTA-ITL; SMART-ITL/FESTA	Integral Test Facilities
	SWAT; FTHEL; SCOP (Separate Effects Test Facilities)	SWAT: SMART ECC Water Asymmetric Two-phase choking test (ECC bypass flow) FTHEL: Freon Thermal Hydraulic Experimental Loop (CHF test) SCOP: SMART COre flow distribution and Pressure drop test
BWRX-300	Tests from facilities for ESBWR	NRC approved for ESBWR design
W-SMR	Results from AP600 and AP1000 facility tests	ADS (Automatic Depressurization System) tests (Milano, Italy) CMT (Core Makeup Tank) tests PRHR tests (Pittsburgh, USA) SPES-2 tests (Piacenza, Italy) APEX tests (Corvallis, USA) UCB tube condensation facility Westinghouse CMT test facility
ACP-100	-	Passive emergency core cooling system testing facility Fuel assembly critical heat flux CMT and passive residual heat removal system Passive containment heat removal testing facility
mPower	MIST IST	Multiloop Integral System Test (MIST) Integrated System Test (IST) facility (scaled facility of the B&W mPower reactor)

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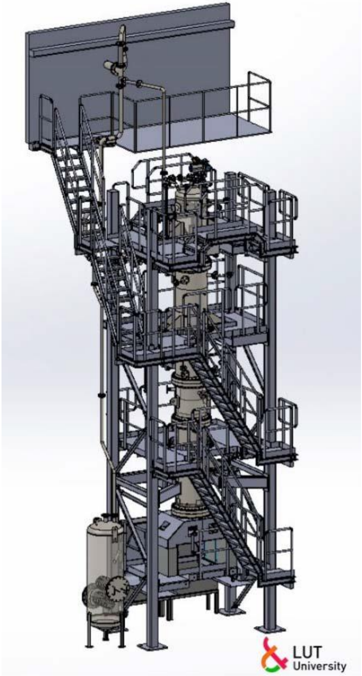
Experimental facilities for code validation. McSAFER Project

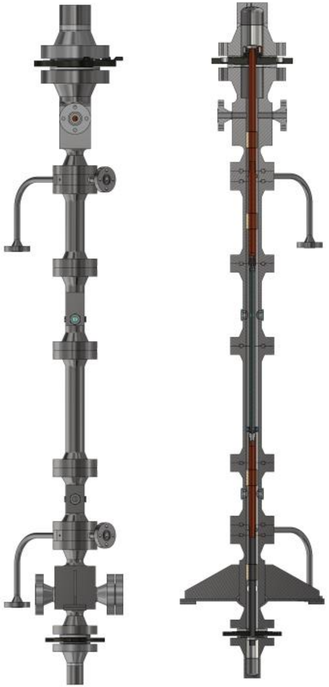
MOTEL (LUT)

Integral Test Facility
(similar to NuScale)



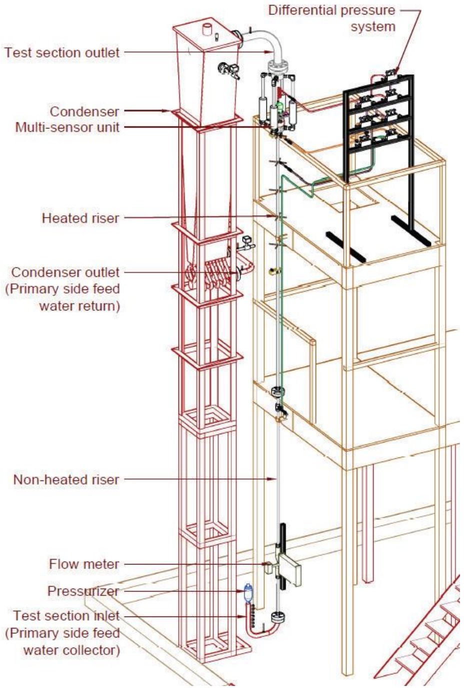
COSMOS-H (KIT)

Separate Effect Test Facility
(CHF)



HWAT (KTH)

Separate Effect Test Facility
(CHF and condensation)



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Application: NuScale Power Module

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Application: NuScale Power Module

NPM Fundamentals

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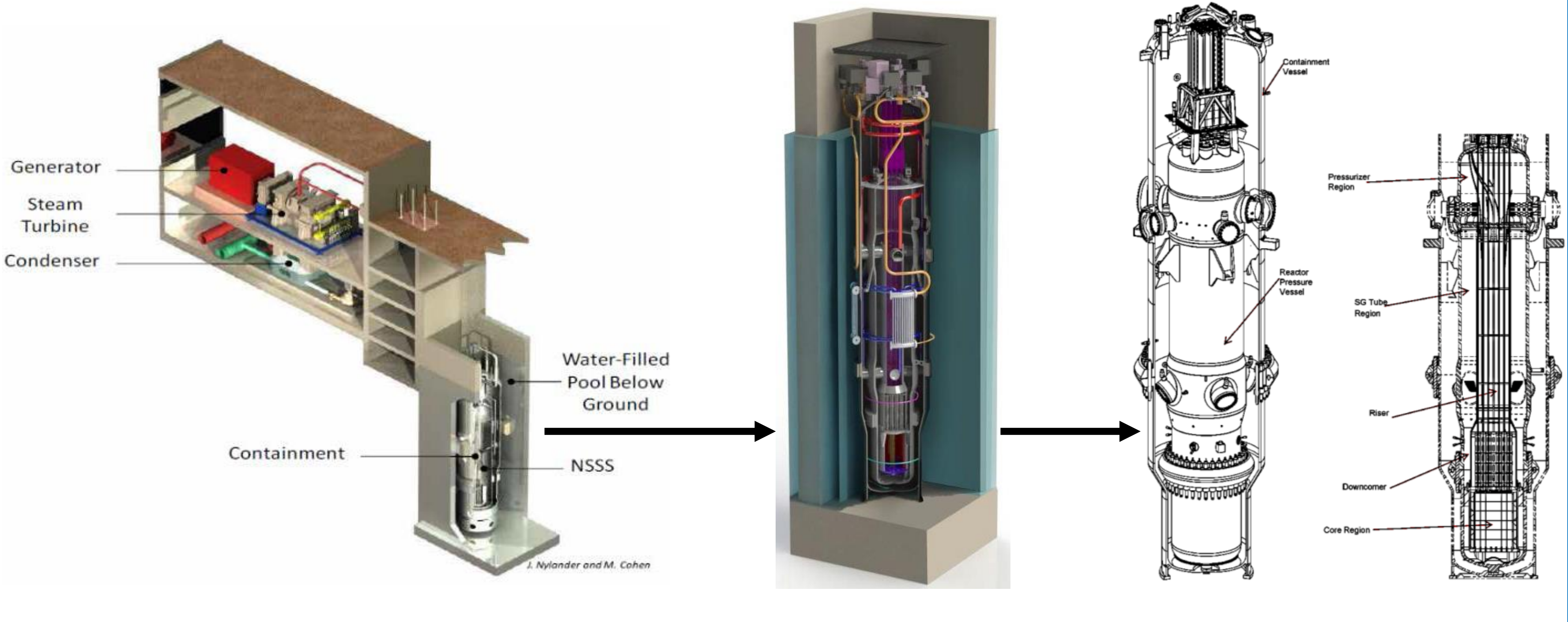
NPM Fundamentals. General Arrangement



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NPM Fundamentals. RCS description

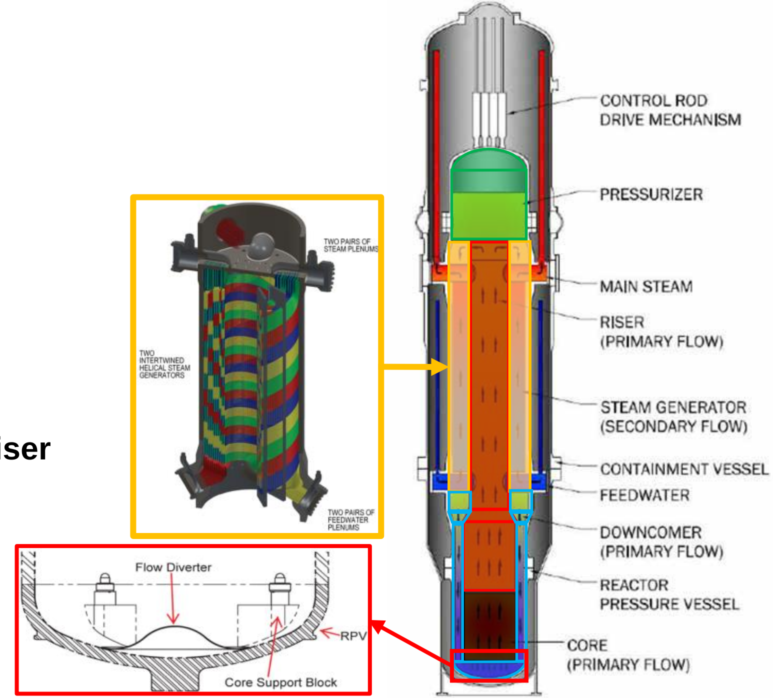
- **Small core**:
  - Power thermal rate = **160 MWt**
  - **37** PWR-Fuel Assemblies (FAs developed by AREVA)
  - Square matrix of **17x17**
  - **2 meters** of **active height**
  - **M5** alloy for the cladding

- **Hot leg**: central riser (lower, transition, and upper regions)

- **SGs**: Vertical helically coiled SG:
  - **2 SGs** with **2 trains** per each SG
  - **SG tubes** are **intertwined** and located **surrounding the riser**
  - **345** tubes **per train**

- **Cold leg**: **Downcomer** and **Lower Plenum (Flow Diverter)**

- **PZR**: **integrated** at the **RPV top head** region



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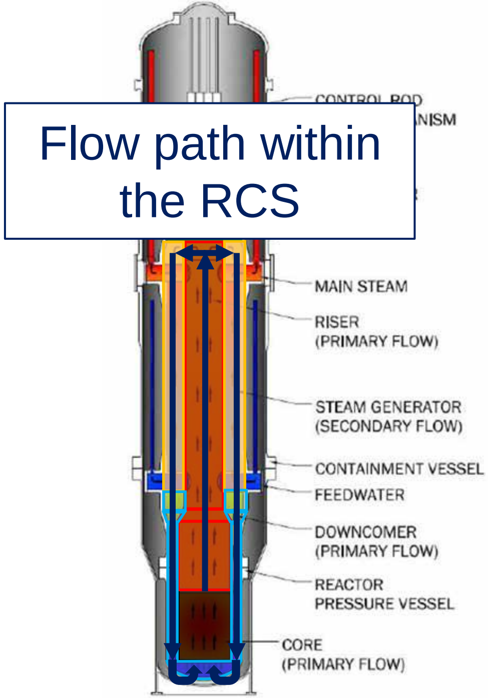

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NPM Fundamentals. Secondary Side

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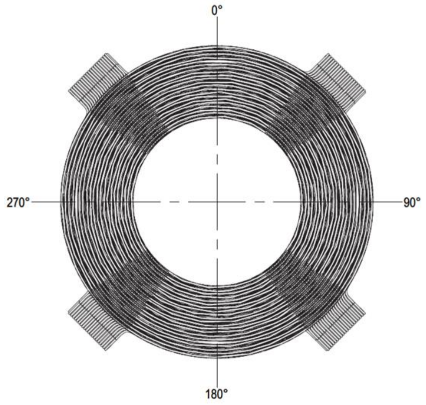
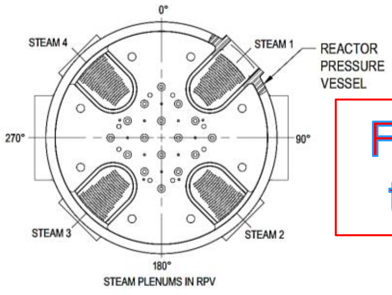
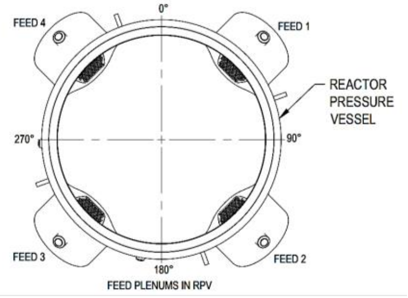
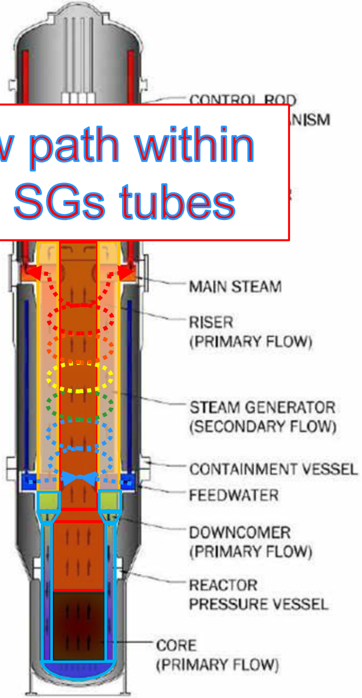
- **Helically-shaped tubes in the Steam Generators**.
- **Fluid flows upwards** through the inner region of the tubes.
- **21 columns** of tubes (rings) in the **radial direction**.
- Inlet and Outlet of the tubes are **offset 90 degrees**.
- **Different number of tubes** at different radial regions.









Flow path within the SGs tubes

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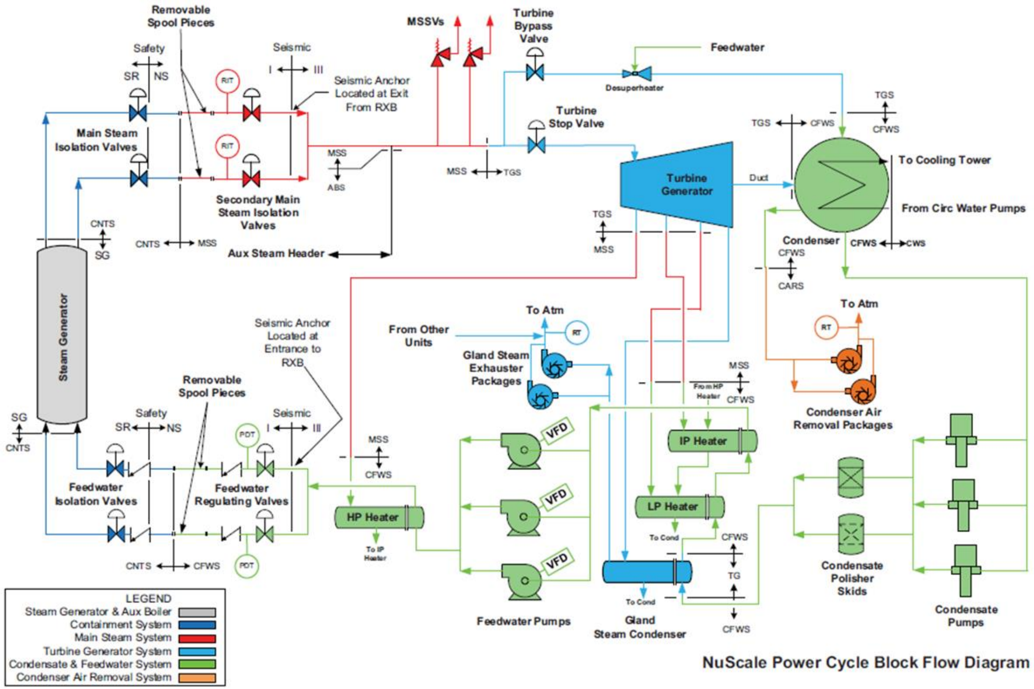
NPM Fundamentals. Secondary side

- Apart from the SGs, the secondary side in every NPM is very similar to the one found in a conventional PWR.

- An additional set of isolation valves is mounted at each steam/feedwater line respectively (2 MSIVs and 2 FWIVs per line).

- One out of three FW pumps is in standby during normal operation, but it starts automatically in case of trip of a running FW pump.



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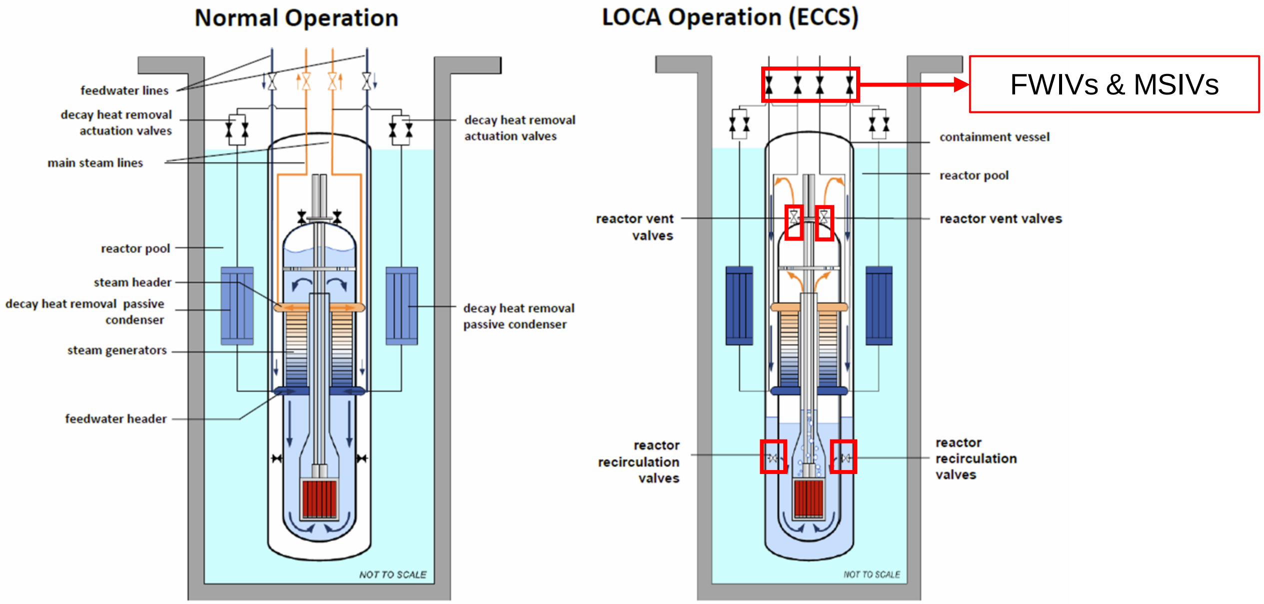
NPM Fundamentals. Safety Systems

- **Emergency Core Cooling System:** ECCS is mainly designed to deal with **LOCA** transients



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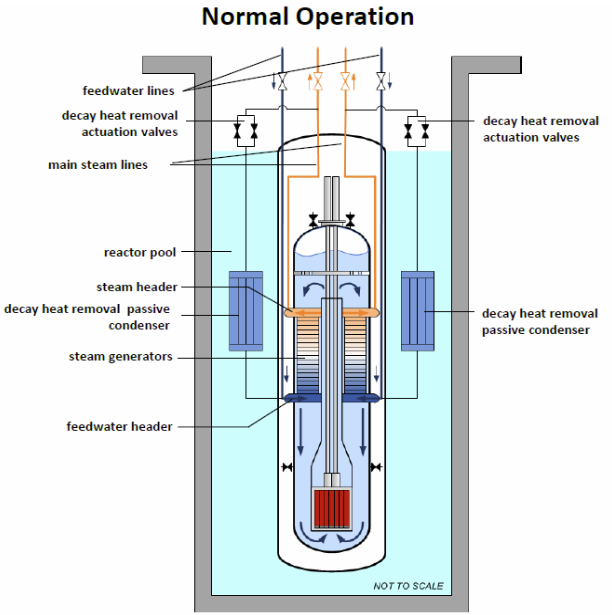
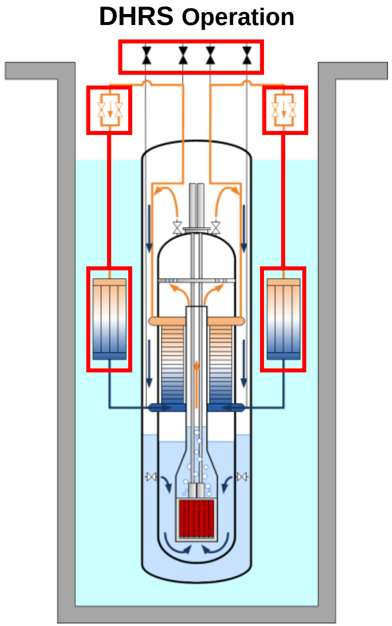
NPM Fundamentals. Safety Systems

- **Decay Heat Removal System:** (DHRS is devoted to remove the Decay Heat under **Non-LOCA conditions**)





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Application: NuScale Power Module

TRACE capabilities regarding SMRs

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TRACE capabilities regarding SMRs

- **TRACE** code includes the following models needed to simulate SMRs:

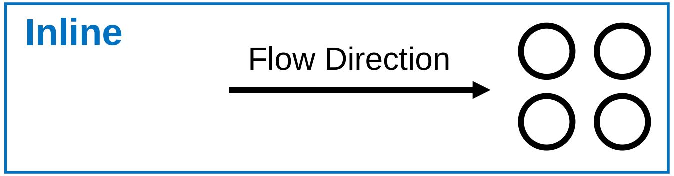
  - Dedicated options to model the **heat transfer** and **friction factor** for **helically coiled SGs**
    - **Heat transfer** correlations for **helical pipes** in the **inner surface**
    - **Zukauskas heat transfer** correlations for **cross flow** in the **outer surface of the tubes** in a tube bank.
    - **Friction factor** correlations **inside** the **helically coiled pipe** as a function of the Dean number.
    - **Zukauskas friction factor** correlations for **cross flow** through a tube bank (**outer surface of tubes**).
  - **Dryout** conditions **inside** the **helical tubes**.
  - **Condensation** heat transfer models.
  - Effect of **non-condensable gases** on **condensation** heat transfer.
  - **High-order numerical methods** to solve the **spatial differences** along with a **semi-implicit method** for the **time integration** is available in **TRACE** (relevant for **Boron Dilution or Flow Mixing transients**)

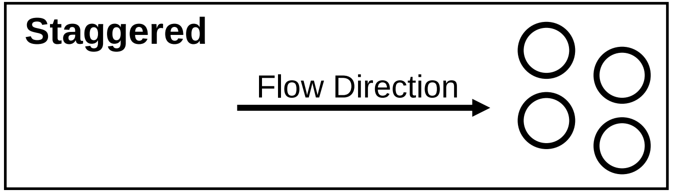
This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

TRACE capabilities regarding SMRs. Helical SGs in TRACE

1) 'Tube bank Crossflow' PIPE or VESSEL (1D/3D):

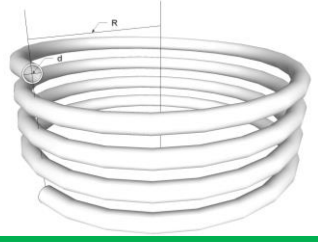
- Recommended option for the modeling of the primary side of the helical SGs.
- Two available arrangement options for the tubes:
  - Inline
  - Staggered
- Additional data must be input:
  - Longitudinal/Transversal-pitch of the tube bank.
  - Tube rows at each axial cell





2) 'Curved pipe' PIPE (1D):

- Recommended option for the modeling of the Secondary side of the helical SGs.
- Friction factor as function of the Dean number
- Inside Tube diameter/coil diameter ratio must be input



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Application: NuScale Power Module

Modeling NuScale using TRACE

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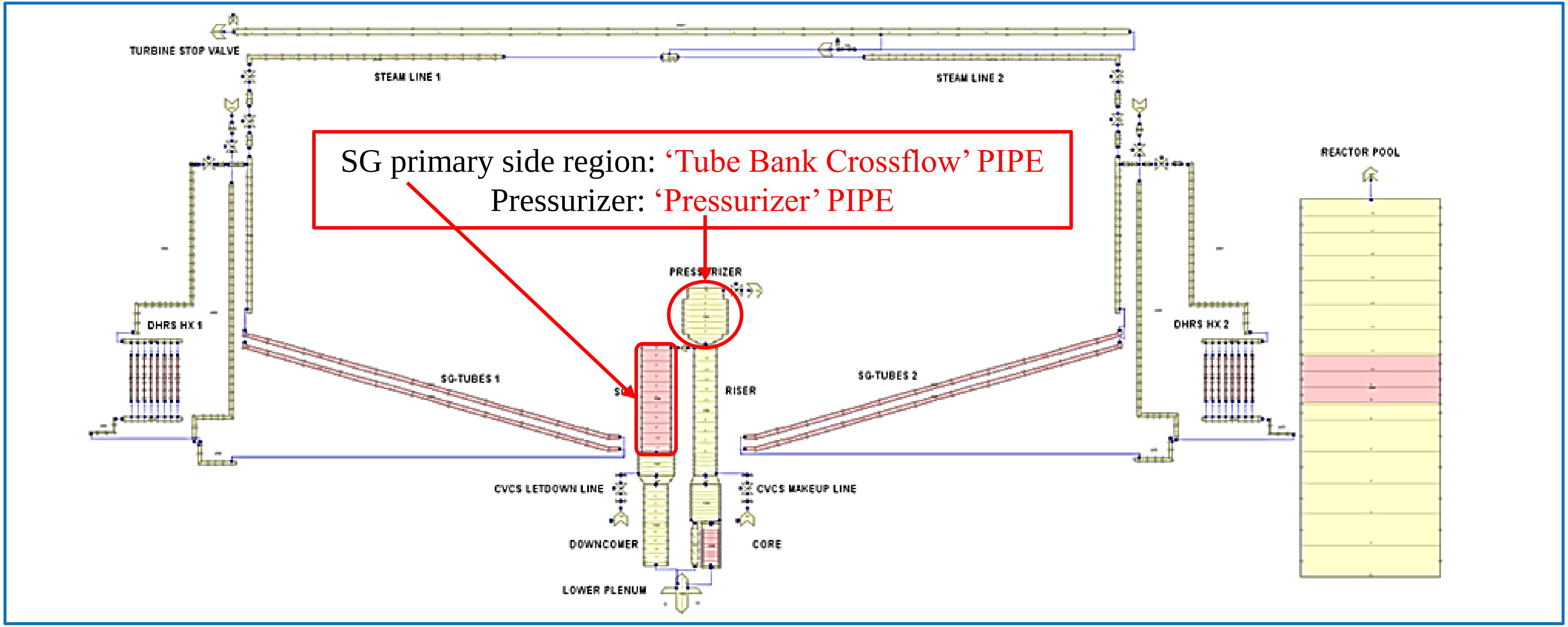
Modeling NuScale using TRACE. UPM 1D approach



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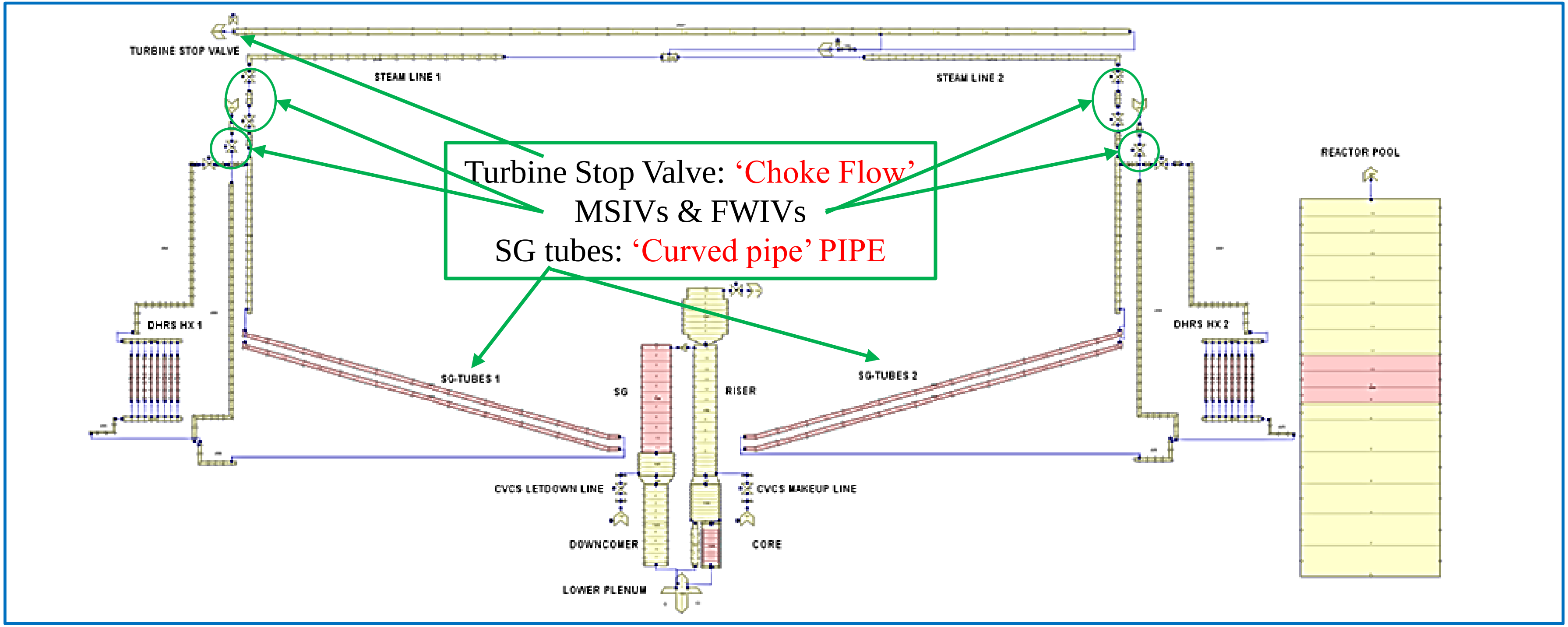
Modeling NuScale using TRACE. UPM 1D approach

McSAFER



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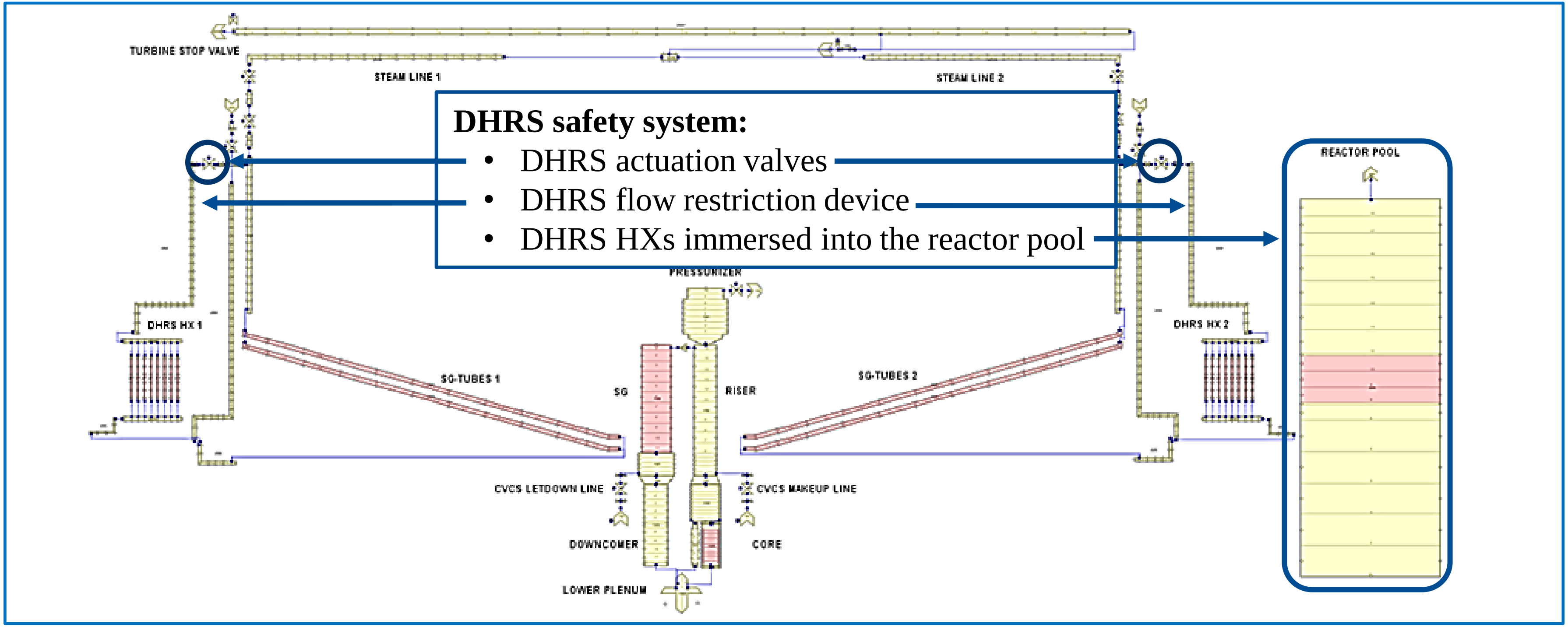
Modeling NuScale using TRACE. UPM 1D approach



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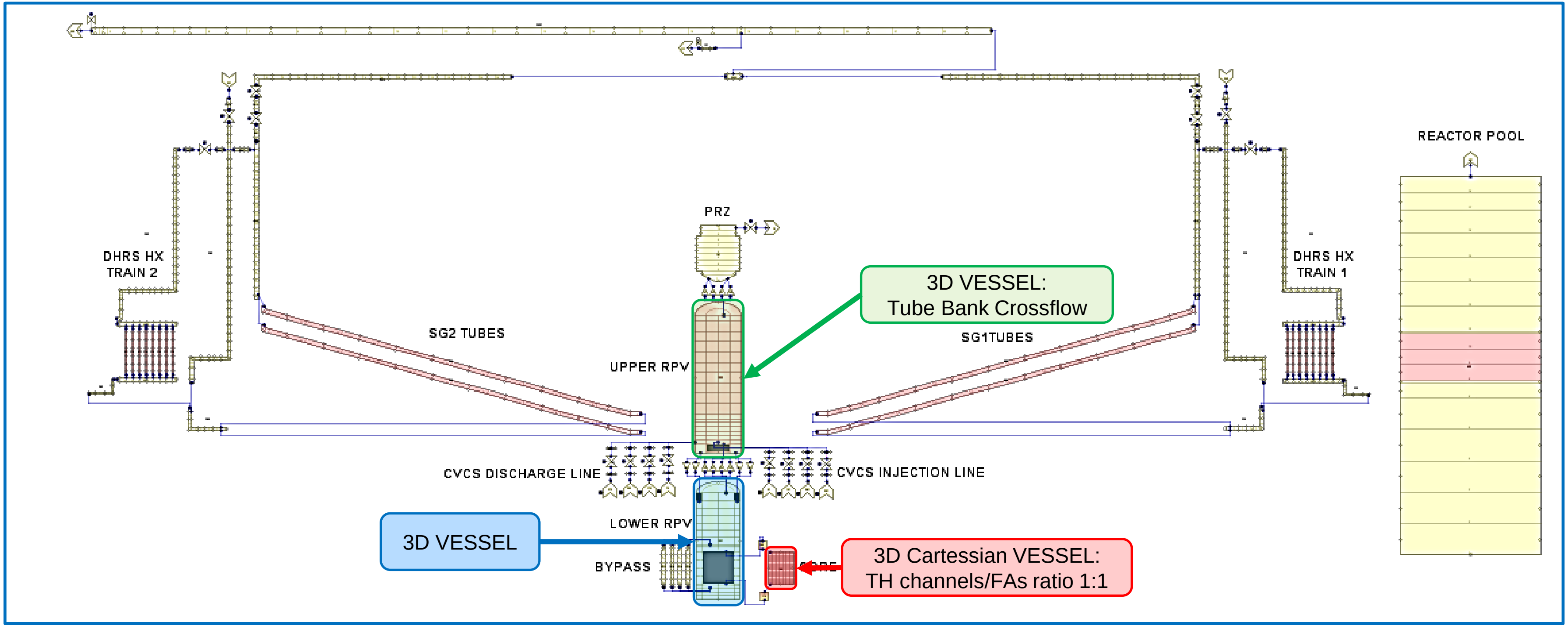
Modeling NuScale using TRACE. UPM 1D/3D approach



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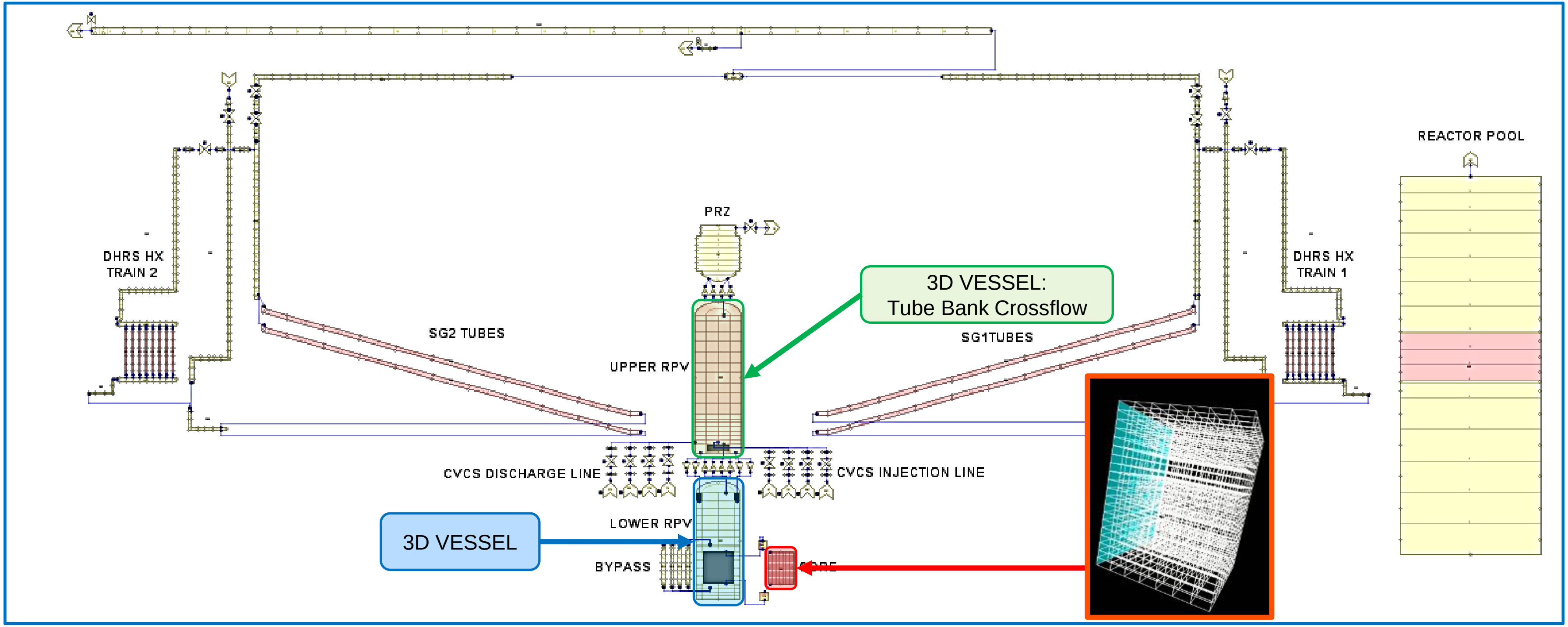
Modeling NuScale using TRACE. UPM 1D/3D approach



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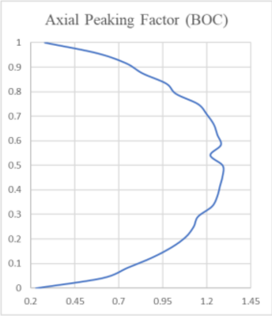
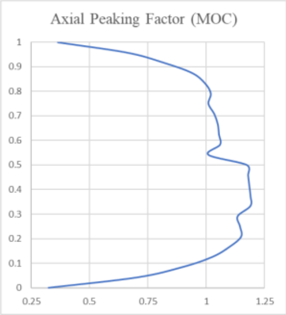
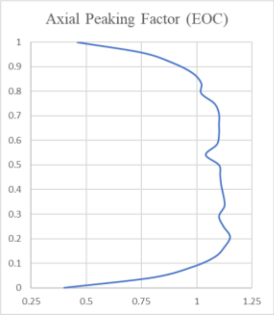
Modeling NuScale using TRACE. Core modeling

TRACE Point Kinetics model:

Axial Power Profiles (BOC, MOC, EOC) (1D/3D)







- Reactivity feedbacks are considered: PKs/PARCS
- Decay Heat correlation

Radial Power Distribution (BOC, MOC, EOC) (3D)

BOC

		0.895	0.911	0.888		
	1.033	1.137	0.957	1.135	1.033	
0.888	1.135	1.054	0.967	1.054	1.136	0.895
0.911	0.957	0.967	1.091	0.967	0.957	0.911
0.895	1.136	1.054	0.967	1.054	1.135	0.888
	1.033	1.135	0.957	1.137	1.033	
		0.888	0.911	0.895		

MOC

		0.887	0.907	0.883		
	1.053	1.119	0.965	1.118	1.052	
0.883	1.118	1.057	0.983	1.057	1.118	0.887
0.907	0.965	0.983	1.112	0.983	0.965	0.907
0.887	1.118	1.057	0.983	1.057	1.118	0.883
	1.052	1.118	0.965	1.119	1.053	
		0.883	0.907	0.887		

EOC

		0.886	0.915	0.883		
	1.029	1.105	0.980	1.105	1.029	
0.883	1.105	1.064	1.003	1.064	1.105	0.886
0.915	0.980	1.003	1.123	1.003	0.980	0.915
0.886	1.105	1.064	1.003	1.064	1.105	0.883
	1.029	1.105	0.980	1.105	1.029	
		0.883	0.915	0.886		

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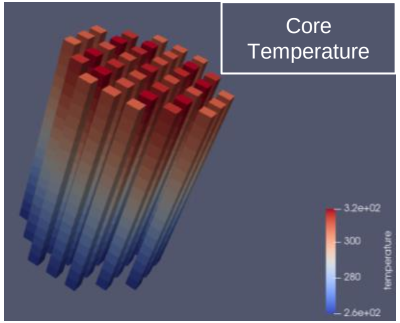
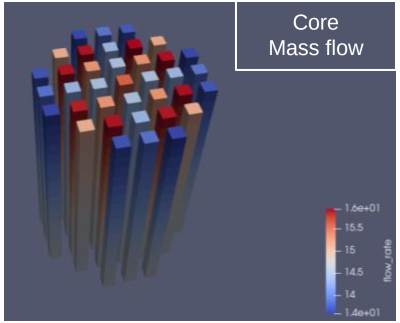
Modeling NuScale using TRACE. Multiphysics and Multi-scale tools

Multi-scale approach: 3D TRACE/SCF

- A **SCF** core model has been developed by UPM within the McSAFER Project for the **TRACE/SCF** tool performed at KIT.

Results of the **SCF** core model





Multi-physics approach: TRACE/PARCS

- A **PARCS** core model is currently under development as a result of the collaboration between VTT and UPM research groups.
- UPM group has been provided with the **SERPENT-XS** by VTT group.
- **TRACE/PARCS** calculation results will be done soon.

Multi-physics approach: TRACE/OpenFoam/PARCS

- Calculation of the flow mixing in the downcomer and in the lower plenum is expected to be improved with respect to **TRACE/PARCS** simulations.
- **TRACE/OpenFoam/PARCS** coupling scheme developed by KIT.
- This tool will be used in SMART SMR calculations

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Application: NuScale Power Module

Transient Analyses performed within McSAFER project

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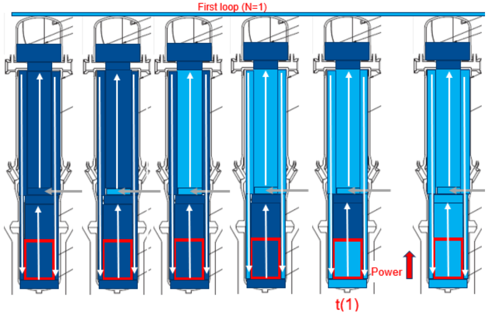
Modeling NuScale using TRACE. Boron Dilution

Boron Dilution analyses have been done in the WP4 with 1D and 3D coarse TH meshes:

- **Boron Dilution sequence brief description:**

1. Unborated water is injected in the RCS by means of the malfunction of the CVCS.
2. A boron dilution front is formed and starts to turning around the RCS loop over an over.

--- Borated Water --- Unborated Water



- **Two cases at 100% of power (160 MWt):**
  1. **Constant Power** (Time to loss the SDM)
     - Boron concentration evolution
  2. **Point kinetics** (Best-Estimate)
     - Axial Power Profile (BOC)
     - Radial Power Distribution (BOC)
     - Reactivity coefficients (BOC)
     - Decay Heat Correlation = ANS-94

- **Van Leer method with flux limiters** to solve **spatial differences** and **semi-implicit method** for **time integration** have been selected.
  - **Avoiding numerical diffusion**

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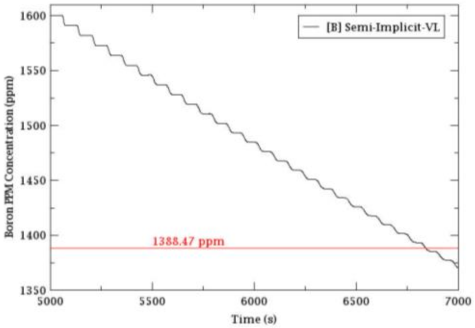
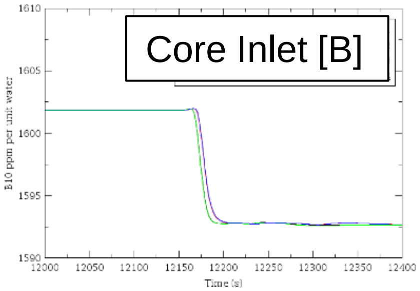
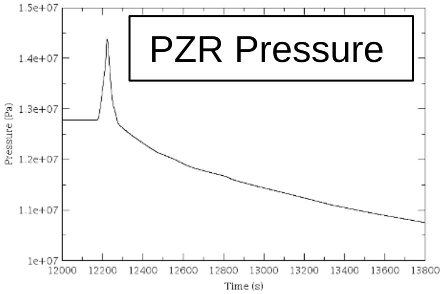
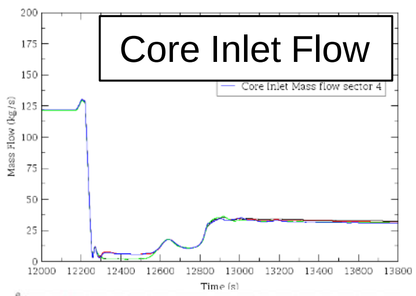
Modeling NuScale using TRACE. Boron Dilution

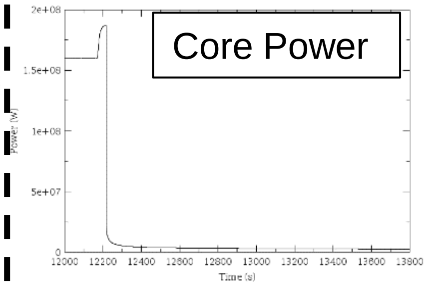
Constant Power results:	Best Estimate case results (3D model)

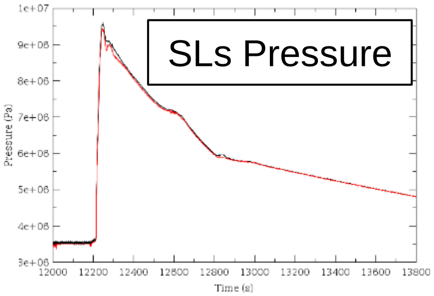


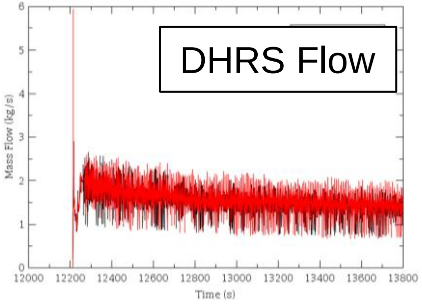












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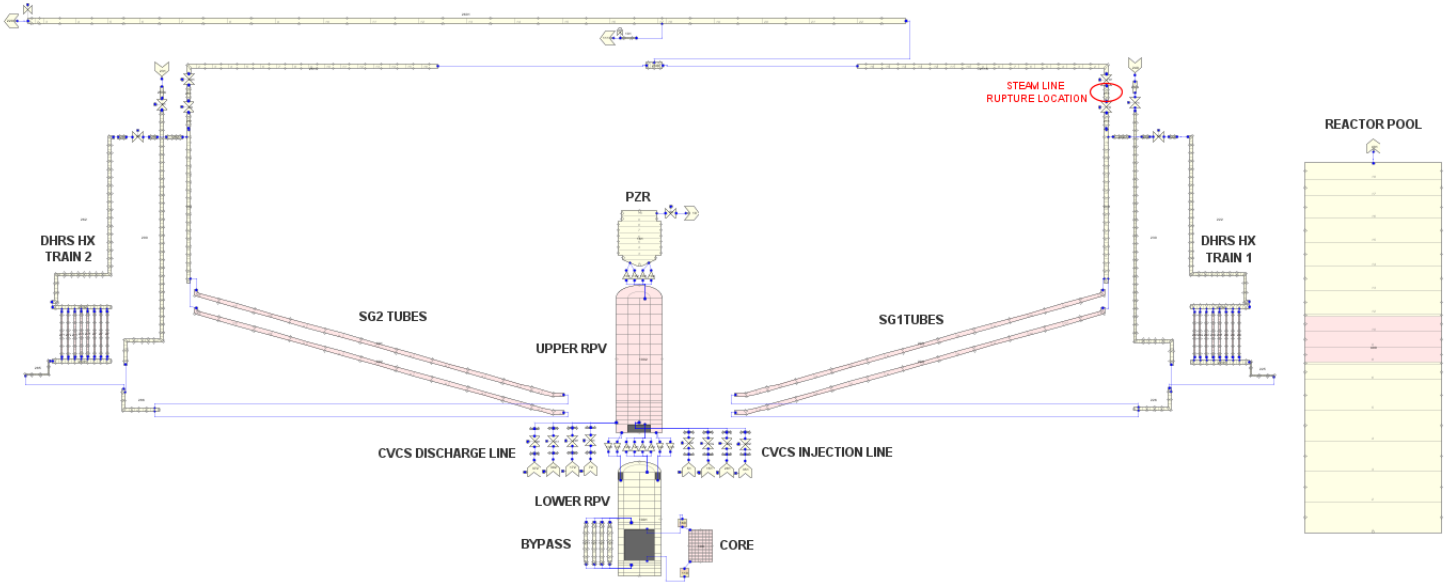
Modeling NuScale using TRACE. Future works

Steam Line Break simulations will be performed soon for the WP5 calculations by means of several multiphysics tools such as TRACE/PARCS or TRACE/OpenFoam/PARCS for NuScale design:



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Final remarks

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Final remarks

- Heat transfer and friction factor models are already available in TRACE for the modeling of both sides of helically coiled SGs. A dryout model in the inside of the helical tubes is also implemented.

- The application of a 3D modeling approach or a multi-scale tool (System Code + CFD) is needed to simulate the peculiarities in the RPV design of the SMRs. CFD codes could play an important role in the modeling of the flow mixing by turbulence and complex geometries

- Condensation models could be also improved to increase the accurate of TH simulations in which passive safety systems must be considered.

- The application of multiphysics tools is more than justified to model SMRs achieving a 3D description of the core power (nodal diffusion codes) coupled with a sufficiently good TH modeling (System Codes + CFD).

- Summarizing, modeling of NPM using system codes allow to consider most of the TH phenomena in which SMRs are involved (apart from turbulence and 3D fine meshes for complex geometries).

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Thanks for your attention

Contact details:
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CFD Analysis of NuScale

Ladislav Vyskocil, UJV Rez, a. s., Czech Republic

March 22, 2022

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

– Introduction to the Computational Fluid Dynamics

– Example Application of CFD to NuScale

– Coupling of CFD Code with System Code

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Introduction to the Computational Fluid Dynamics

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What is CFD?

Fluid dynamics is the science of fluid motion.

- Experimental fluid dynamics.

- Theoretical fluid dynamics.

- Computational fluid dynamics (CFD).

Computational fluid dynamics (CFD) is the science of predicting fluid flow, heat transfer, mass transfer, and other related phenomena by solving the mathematical equations which govern these processes.

• CFD analyses can be used in:

- Design of new products, equipment, machine…

- Redesign or solving problems of existing equipment

• CFD analysis complements experiments and reduces the effort required in the laboratory.

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CFD Codes

Examples of commercially available general purpose CFD codes:

- Ansys Fluent
- Ansys CFX
- STAR-CCM+ (Siemens)

Examples of free, open-source codes:

- code_saturne
- TrioCFD
- OpenFOAM

Specialized CFD codes

- neptune_cfd (multiphase flows in nuclear reactors)
- FUN3D (aeronautics)

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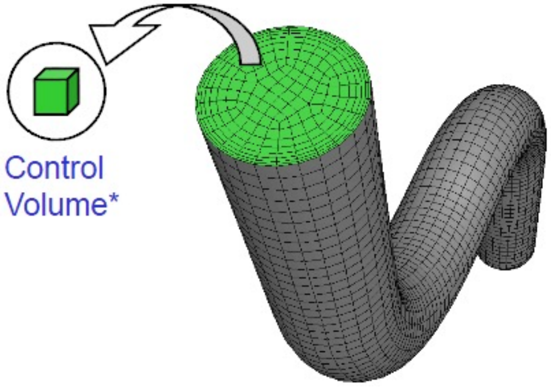
How the CFD works

Finite Volume Method (FVM)

- Computational domain is discretized into a finite set of control volumes (cells).
- The discretized domain is called the “grid” or the “mesh”.
- General conservation (transport) equations for mass, momentum, energy, etc., are discretized into algebraic equations.
- All algebraic equations are solved to obtain the pressure, velocities, temperature and other variables in every cell in the domain.



$$\underbrace{\frac{\partial}{\partial t}\int_V \rho\phi\, dV}_{\text{Unsteady}} + \underbrace{\oint_A \rho\phi \mathbf{V}\cdot d\mathbf{A}}_{\text{Convection}} = \underbrace{\oint_A \Gamma_\phi \nabla\phi \cdot d\mathbf{A}}_{\text{Diffusion}} + \underbrace{\int_V S_\phi\, dV}_{\text{Generation}}$$

Equation	Variable
Continuity	1
X momentum	*u*
Y momentum	*v*
Z momentum	*w*
Energy	*h*

Example: Computational mesh with 1 million cells and 7 variables in flow field: p, u, v, w, h, tke, epsilon. CFD code solves 7 million algebraic equations for 7 million unknowns in every time step.

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Procedure of CFD Analysis

- Define goals
- Identify computational domain
- Identify physical phenomena
- Pre-processing
  - geometry of computational domain
  - computational mesh
  - select suitable models of physical phenomena
  - adjust boundary and initial conditions
  - adjust solver
- Run the code, compute solution
- Post-processing
  - examine results

modify domain, modelling approach…

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Computational Mesh

- The mesh is a discrete representation of the geometry of the problem.
- The mesh has cells grouped into boundary zones where boundary conditionss are applied.

The density and quality of the mesh has a significant impact on:

- Rate of convergence (or even lack of convergence).
- Solution accuracy.
- CPU time required.

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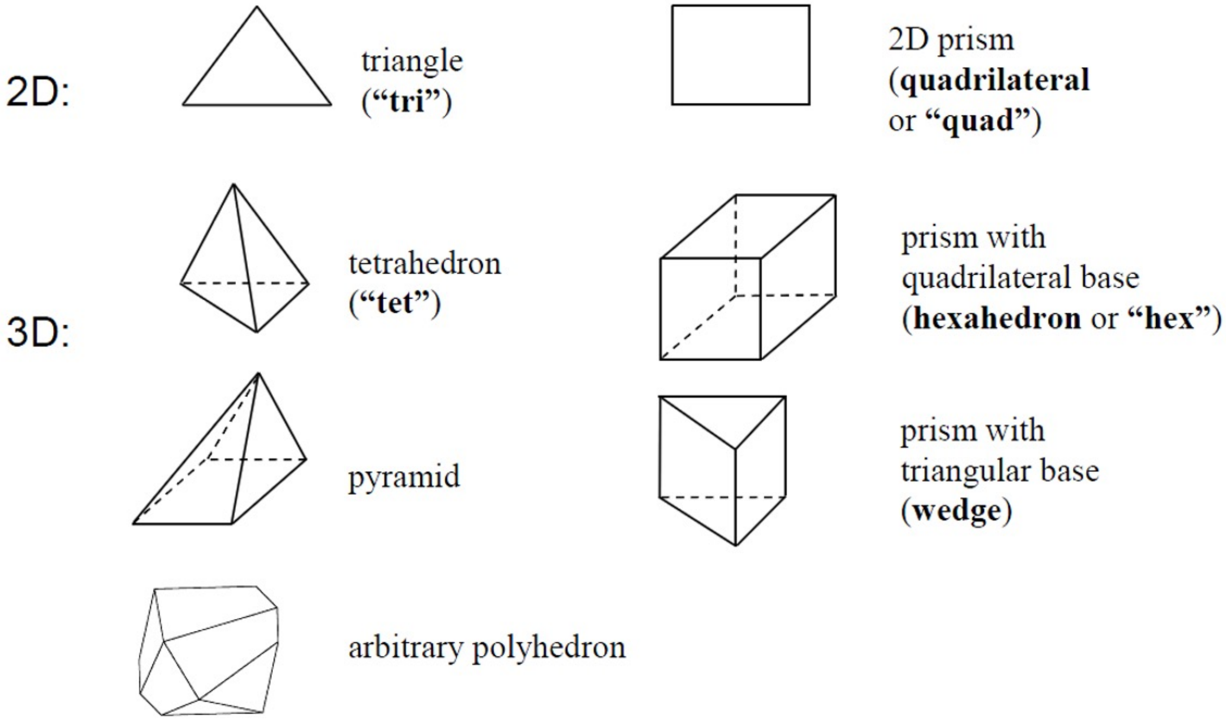
Computational Mesh

Types of cells that can be used in Fluent code



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Mesh Quality

For the same cell count, hexahedral meshes will give more accurate solutions, especially if the grid lines are aligned with the flow.

- The mesh density should be high enough to capture all relevant flow features.
- The mesh adjacent to the wall should be fine enough to resolve the boundary layer flow.

- Three measures of mesh quality:
  - Skewness.
  - Smoothness (change in size).
  - Aspect ratio.

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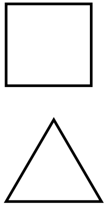
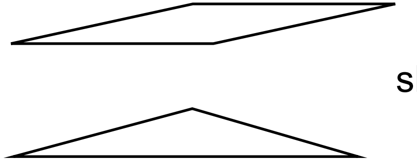
Mesh Quality



optimal cells



skewed cells

Skewness

Cells should not be skewed.

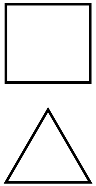
Smoothness (change in size).

Change in size should be gradual (smooth).

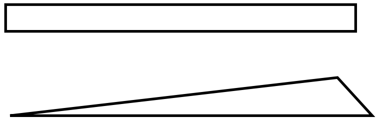
smooth change in cell size

large jump in cell size

Aspect ratio is ratio of longest edge length to shortest edge length.
Equal to 1 (ideal) for an equilateral triangle or a square.



ideal aspect ratio



high aspect ratio cells

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Turbulence Models

All flows become unstable above a certain Reynolds number.

- At low Reynolds numbers flows are laminar.
- For high Reynolds numbers flows are turbulent.
- The transition occurs anywhere between 2000 and 1E6, depending on the flow.
- For laminar flow problems, flows can be solved using the instantaneous conservation equations (continuity equation, momentum equations, energy equation).
- For turbulent flows, the computational effort involved in solving those equations for all time and length scales is prohibitive.
- An engineering approach is to calculate time-averaged flow fields for turbulent flows.

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Turbulence Models

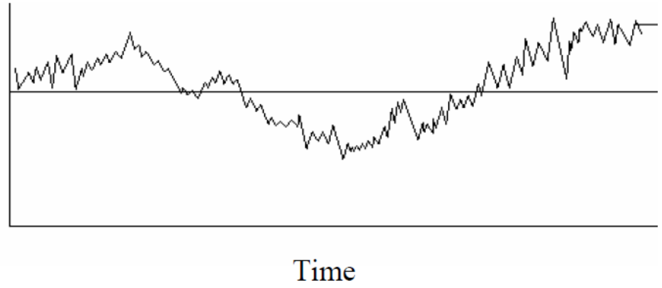
Turbulence: Unsteady, aperiodic motion in which all three velocity components fluctuate, mixing matter, momentum, and energy. Pressure and temperature also fluctuate.

Velocity and pressure decomposition into mean and fluctuating parts:

$$\mathbf{u} = \mathbf{U} + \mathbf{u}'$$

$$p = P + p'$$

$U_i(t) \equiv U_i + u_i(t).$



Turbulent kinetic energy

$$k = \frac{1}{2}\left(\overline{u'^2} + \overline{v'^2} + \overline{w'^2}\right)$$

Continuity equation is valid for both instantaneous velocities and time averaged velocities:

$$div\ \mathbf{u} = 0$$

time average: $\overline{div\ \mathbf{u}} = div\ \mathbf{U} = 0$

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Turbulence Models

Navier-Stokes momentum equations are valid for instantaneous velocities and pressure.

Time average of the N-S momentum equation results in the **Reynolds equations**.

$$x-momentum: \quad \frac{\partial(\rho U)}{\partial t} + div(\rho U\mathbf{U}) = -\frac{\partial P}{\partial x} + div(\mu\ grad\ U) + S_{Mx}$$
$$+\left[-\frac{\partial(\rho\overline{u'^2})}{\partial x} - \frac{\partial(\rho\overline{u'v'})}{\partial y} - \frac{\partial(\rho\overline{u'w'})}{\partial z}\right]$$

Reynolds equations contain an additional stress tensor. These are called the Reynolds stresses.

$$\boldsymbol{\tau} = \begin{pmatrix} \tau_{xx} & \tau_{xy} & \tau_{xz} \\ \tau_{yx} & \tau_{yy} & \tau_{yz} \\ \tau_{zx} & \tau_{zy} & \tau_{zz} \end{pmatrix} = \begin{pmatrix} -\rho\overline{u'^2} & -\rho\overline{u'v'} & -\rho\overline{u'w'} \\ -\rho\overline{u'v'} & -\rho\overline{v'^2} & -\rho\overline{v'w'} \\ -\rho\overline{u'w'} & -\rho\overline{v'w'} & -\rho\overline{w'^2} \end{pmatrix}$$

…6 additional unknowns in the momentum equations

In turbulent flow, the Reynolds stresses are usually large compared to the viscous stresses.

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Turbulence Models

- The time averaged equations contain 6 additional unknowns in the momentum equations.
- Similarly, additional unknowns appear in the time averaged scalar equations (e.g. energy equation).
- Turbulent flows are usually quite complex, and there are no simple formulae for these additional terms.
- The main task of turbulence modeling is to develop computational procedures for accurate prediction of the Reynolds stresses and the scalar transport terms.

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Turbulence Models

Examples of turbulence models based on Reynolds Averaged Navier-Stokes (RANS) equations (time averaged):

- One equation model: Spalart-Almaras.
- Two equation models: k-ε models, k-ω models
- Seven equation model: Reynolds stress model.

The number of equations denotes the number of additional PDEs that are being solved.

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Heat Transfer

Conduction: diffusion of heat due to temperature gradients.

Convection: heat is carried away by moving fluid. The flow can either be caused by external influences, forced convection; or by buoyancy forces, natural convection. Convective heat transfer is tightly coupled to the fluid flow solution.

Conjugate heat transfer: conduction of heat through solids coupled with convective heat transfer in a fluid.

Radiation: transfer of energy by electromagnetic waves between surfaces with different temperatures, separated by a medium transparent to the radiation.

CFD code can solve all these modes of heat transfer.

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Common Boundary Conditions in Incompressible Flow

Inlet Boundary Conditions

- *Pressure Inlet:* specified static pressure
- *Velocity Inlet:* specified components of velocity vector
- *Mass Flow Inlet:* specified mass flow rate and its direction

For all these BC types: specified temperature (or other scalar), turbulence variables (e.g. turbulence intensity and hydraulic diameter)

Outlet Boundary Conditions

- *Pressure Outlet:* specified static pressure
- *Outlet Vent:* specified pressure loss coefficient and ambient static pressure
- *Mass Flow Outlet:* specified mass flow rate and its direction
- *Outflow*: zero normal gradient for all flow variables except pressure

For all these BC types: specified temperature (or other scalar) and turbulence variables that will be used in the case of reversed flow at the outlet.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Common Boundary Conditions in Incompressible Flow

Inlet and Outlet Combinations

- Pressure inlet and pressure outlet
- Velocity inlet and pressure outlet (or outlet vent)
- Mass flow inlet and pressure outlet (or outlet vent)

Multiple inlets and outlets

- Multiple velocity inlets and multiple pressure outlets
- Multiple velocity inlets and multiple outflows with flow rate weighting
- Multiple mass flow inlets and outlets and at least one pressure outlet or outlet vent

Boundary location should be selected so that flow either goes in or out on the whole BC surface at given time.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Common Boundary Conditions in Incompressible Flow

Wall

- No-slip condition is enforced in viscous flow at wall: tangential fluid velocity equal to wall velocity, zero normal velocity.
- Alternatively, wall shear stress can be specified.
- Thermal boundary conditions:
  - specified wall temperature
  - specified wall heat flux
  - specified ambient temperature and heat transfer coefficient

In case of modelling conduction in solid wall, thermal BC is specified at outside surface of solid wall.

Wall roughness can be defined for turbulent flows.

Wall shear stress and heat transfer is calculated based on local flow field.

Velocity can be assigned to wall (moving wall).

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Common Boundary Conditions in Incompressible Flow

Symmetry

Symmetry reduces computational effort; only symmetric part of the domain is modelled.

Flow field and geometry of the domain must be symmetric:

- Zero normal velocity at symmetry plane.
- Zero normal fluxes of all variables at symmetry plane.
- No inputs required.

Periodic boundaries

Used when geometry of domain and expected flow field are periodically repeating.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Source Terms

Porous Media

Porous zone is a special type of fluid zone with pressure loss in flow determined from user inputs of resistance coefficients.

Porous media can be used to model flow through:

– Perforated plates

– Tube banks (reactor core)

Energy source term can be specified in a fluid zone or its part. This way, heat generation in a rector core can be prescribed.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Material Properties

For the fluid material, these properties need to be specified:

- Density
- Viscosity
- Heat capacity
- Thermal conductivity
- Other properties depending on the used model (for example diffusion coefficients for simulation of species mixture)

For the solid material, these properties need to be specified:

– Heat capacity

– Thermal conductivity

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Validation

Verification - the process of determining that a model implementation accurately represents the developer's conceptual description of the model and the solution to the model;

Validation - the process of determining the degree to which a model is an accurate representation of the real world from the perspective of the intended uses of the model.

Verification: Does the code solve the equations correctly?

Validation: Does the code solve the correct equations?

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Example of Validation

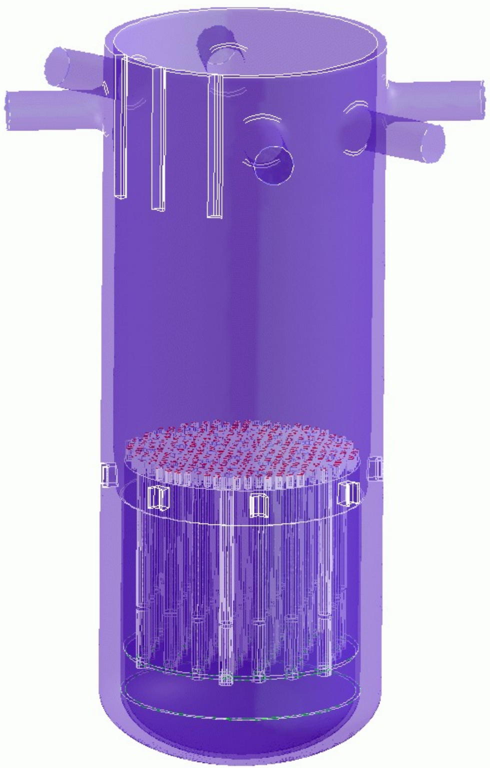
Mixing in VVER-440 reactor downcomer and lower plenum

Measurements in the nuclear power plant: one of six primary loops is subcooled.

Mixing occurs as coolant travels through the downcomer and lower plenum.

Temperatures at the core inlet are measured.

Given the mass flow rates and temperatures at reactor inlet CFD code Fluent can calculate the temperature field at the core inlet.



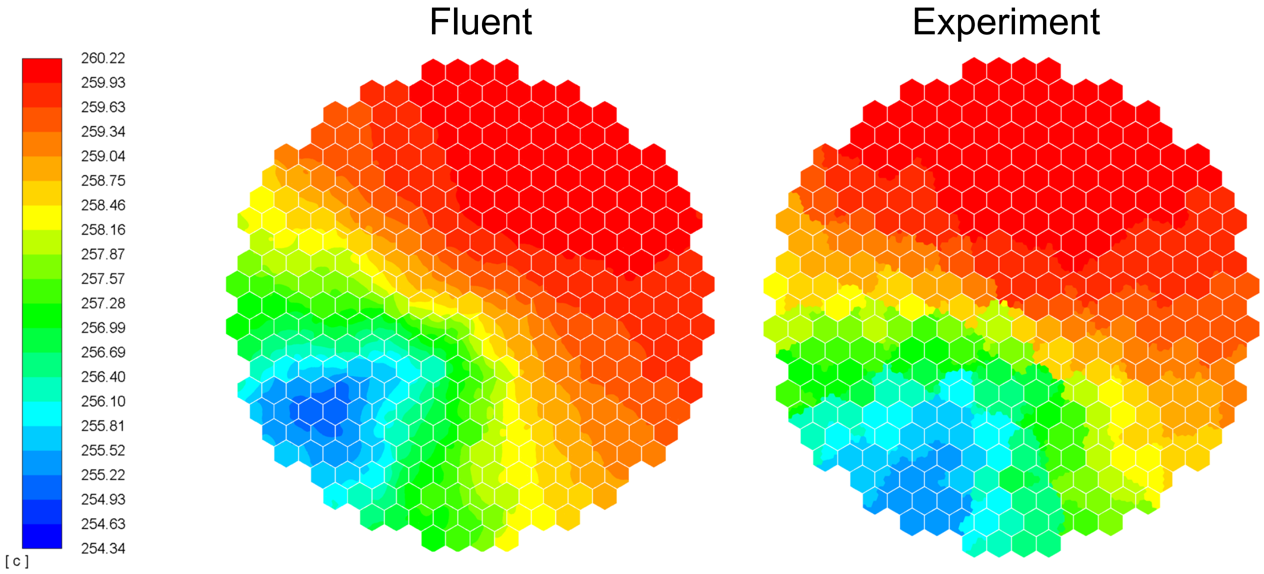
Computational domain for CFD simulation

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Example of Validation

Mixing in VVER-440 reactor downcomer and lower plenum

Fluent code underestimated mixing. The results are on the safe side from the nuclear safety point of view.



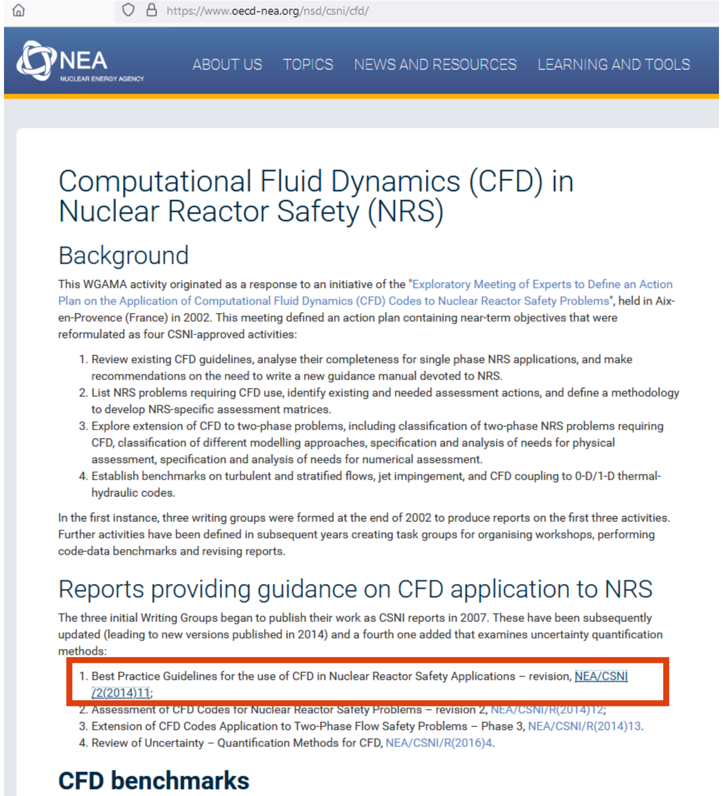
CFD simulation was performed by Martin Kratochvil, UJV Rez.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Best Practice Guidelines

http://www.oecd-nea.org/nsd/csni/cfd/

Best Practice Guidelines for the use of CFD in Nuclear Reactor Safety Applications – revision


https://www.oecd-nea.org/nsd/csni/cfd/

NEA NUCLEAR ENERGY AGENCY

ABOUT US TOPICS NEWS AND RESOURCES LEARNING AND TOOLS

Computational Fluid Dynamics (CFD) in Nuclear Reactor Safety (NRS)

Background

This WGAMA activity originated as a response to an initiative of the "Exploratory Meeting of Experts to Define an Action Plan on the Application of Computational Fluid Dynamics (CFD) Codes to Nuclear Reactor Safety Problems", held in Aix-en-Provence (France) in 2002. This meeting defined an action plan containing near-term objectives that were reformulated as four CSNI-approved activities:

1. Review existing CFD guidelines, analyse their completeness for single phase NRS applications, and make recommendations on the need to write a new guidance manual devoted to NRS.
2. List NRS problems requiring CFD use, identify existing and needed assessment actions, and define a methodology to develop NRS-specific assessment matrices.
3. Explore extension of CFD to two-phase problems, including classification of two-phase NRS problems requiring CFD, classification of different modelling approaches, specification and analysis of needs for physical assessment, specification and analysis of needs for numerical assessment.
4. Establish benchmarks on turbulent and stratified flows, jet impingement, and CFD coupling to 0-D/1-D thermal-hydraulic codes.

In the first instance, three writing groups were formed at the end of 2002 to produce reports on the first three activities. Further activities have been defined in subsequent years creating task groups for organising workshops, performing code-data benchmarks and revising reports.

Reports providing guidance on CFD application to NRS

The three initial Writing Groups began to publish their work as CSNI reports in 2007. These have been subsequently updated (leading to new versions published in 2014) and a fourth one added that examines uncertainty quantification methods:

1. Best Practice Guidelines for the use of CFD in Nuclear Reactor Safety Applications – revision, NEA/CSNI/R(2014)11;
2. Assessment of CFD Codes for Nuclear Reactor Safety Problems – revision 2, NEA/CSNI/R(2014)12;
3. Extension of CFD Codes Application to Two-Phase Flow Safety Problems – Phase 3, NEA/CSNI/R(2014)13.
4. Review of Uncertainty – Quantification Methods for CFD, NEA/CSNI/R(2016)4.

CFD benchmarks

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Example Application of CFD to NuScale

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Simulation of Coolant Mixing during MSLB Accident

- Goal of the simulation is to predict the evolution of temperature field at the core inlet during Main-Steam Line Break (MSLB) Accident
- CFD simulation is performed in Ansys Fluent 15 code.
- Initial and boundary conditions for CFD simulation are used from the simulation in system thermal hydraulic code RELAP5-3D. This simulation was performed by Dr. Marek Bencik (UJV Rez).
- Sequence of events

Time [s]	Event	Comment
0	Guilotine break of MSL1 outside RPV	
2,05	SCRAM - Low Main Steam Pressure (< 2,07 MPa)	CRs start to move
2,05	ESFAS - Low Main Steam Pressure (< 2,07 MPa)	Secondary side isolation
7,05	MSIVs and FWIVs fully closed	Valve closing time = 5 s
12,70	ESFAS - High Main Steam Pressure (> 5,52 MPa)	SL 2 ⇒ DHRS activated - not simulated in calculation
200	End of calculation	

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Simulation of Coolant Mixing during MSLB Accident

NuScale nodalization for RELAP5-3D

- Reactor model is split into 4 segments around the perimeter.

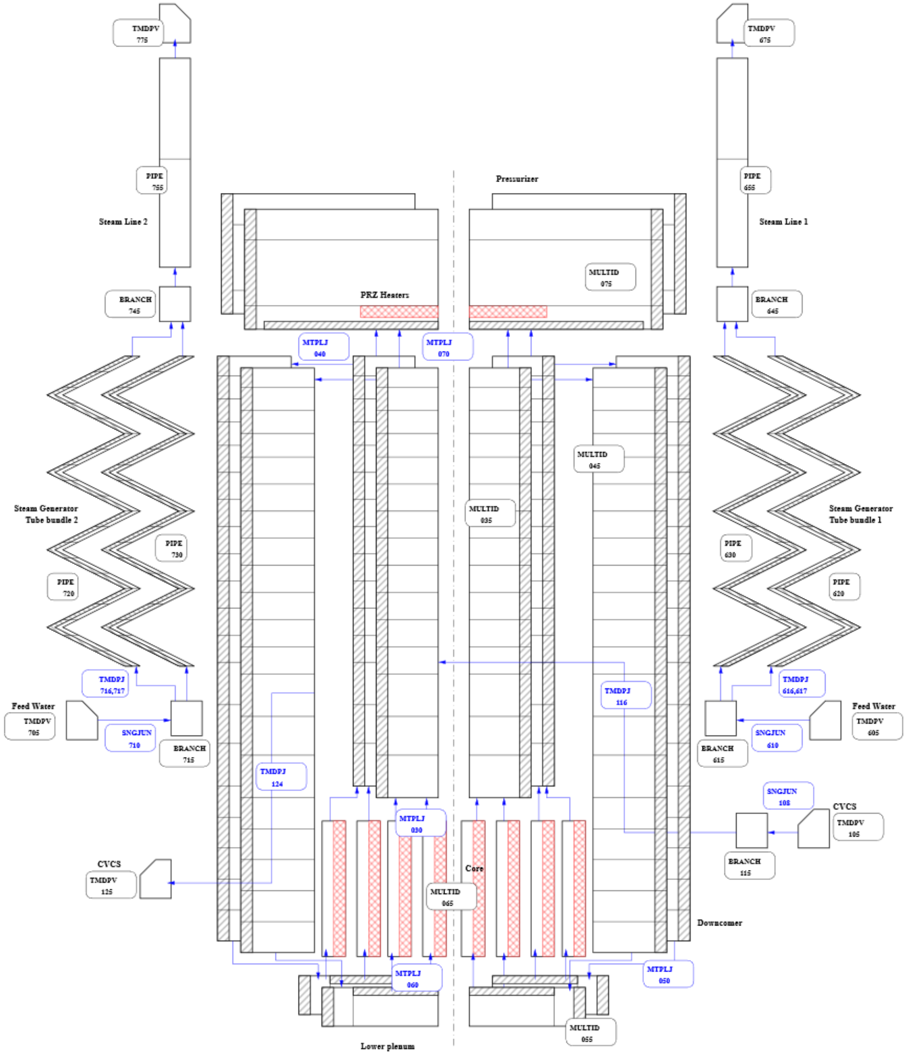
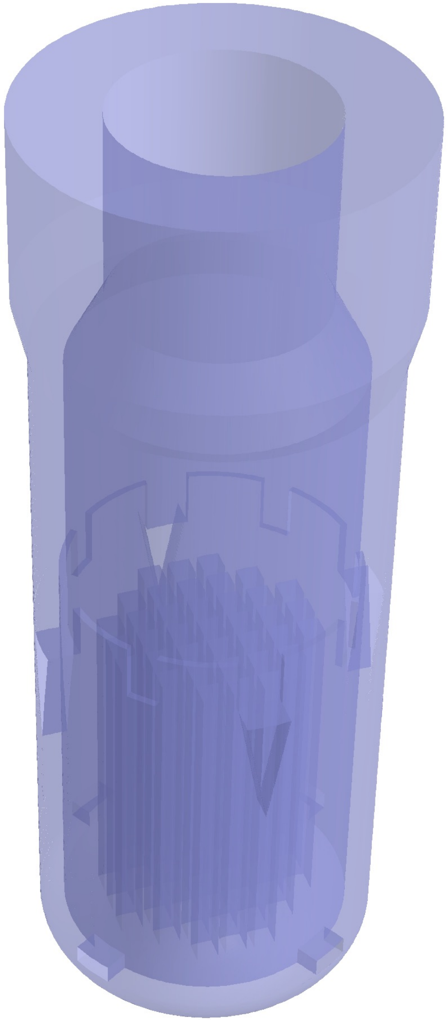


Figure used from: McSAFER Deliverable D4.4: Analysis of NuScale plant with 3D system code and intercomparing between codes

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Simulation of Coolant Mixing during MSLB Accident

- Computational domain for CFD simulation covers downcomer, lower plenum and core.
- Simplified spacers in downcomer are included in the downcomer
- Inlet into computational domain is placed at the steam generator outlet.
- Outlet from the domain is placed at the core outlet.



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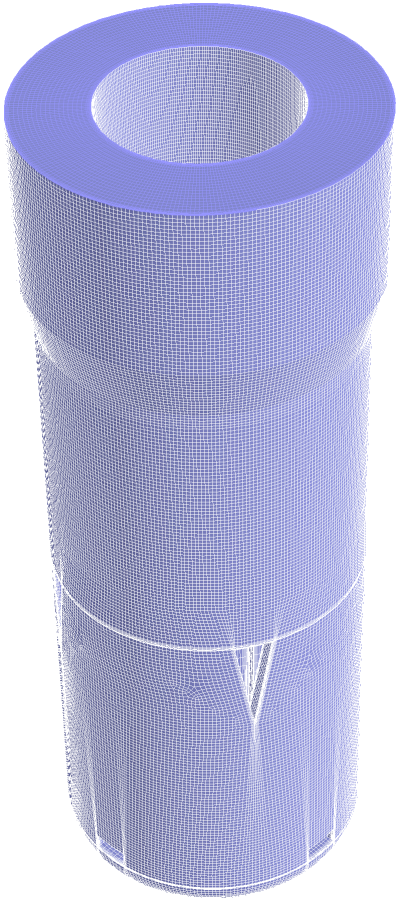
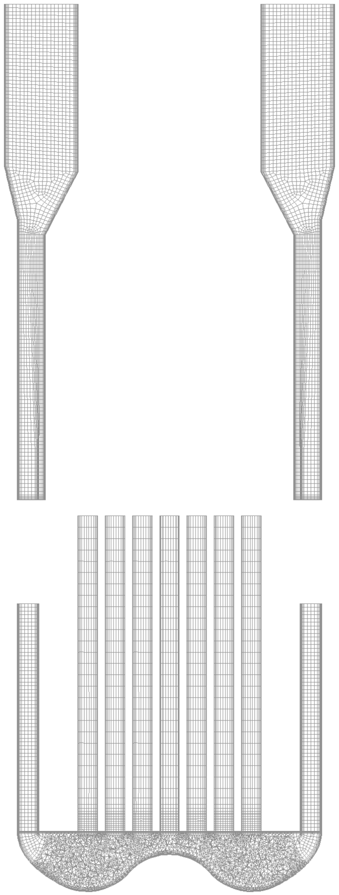
Simulation of Coolant Mixing during MSLB Accident

- Computational domain and mesh was created in Ansys Gambit code.
- Computational mesh for CFD simulation contains 2.44M cells.
- Hexahedral cells are used in downcomer and in the core.
- Tetrahedral cells are used in lower plenum.





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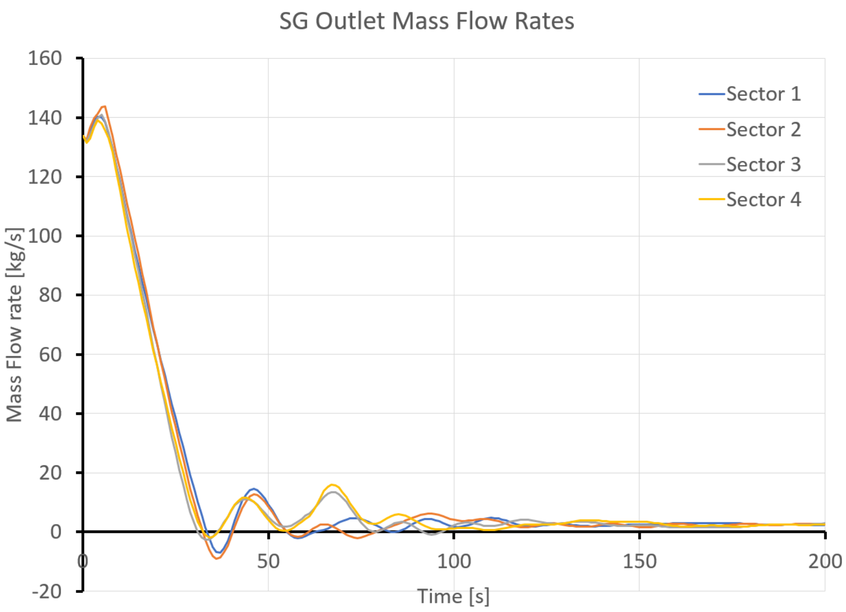
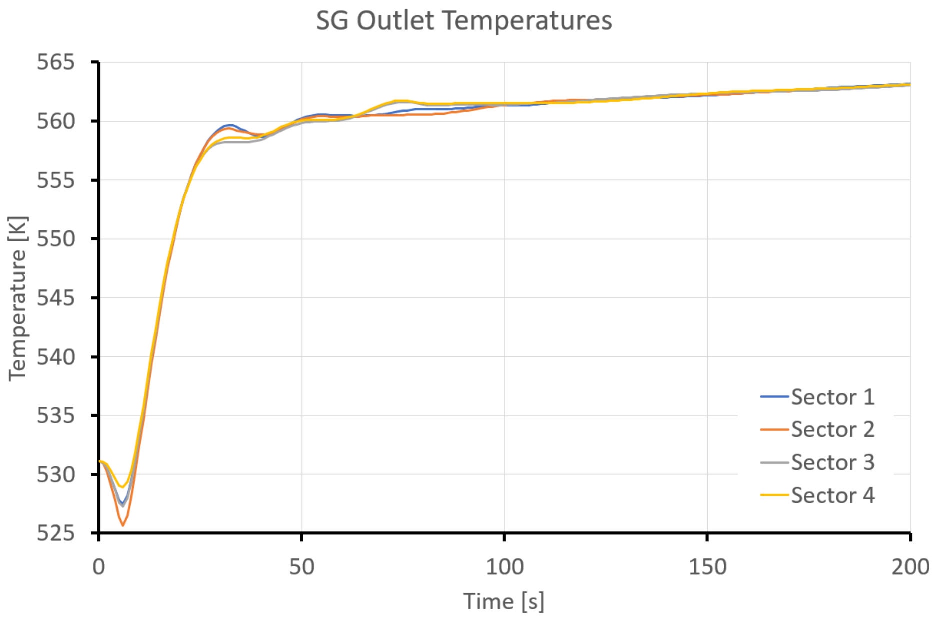
Simulation of Coolant Mixing during MSLB Accident

- Inlet boundary conditions are used from RELAP5-3D code:





This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Simulation of Coolant Mixing during MSLB Accident

McSAFER

- Wall boundary conditions: adiabatic walls, conjugate wall heat transfer is neglected.
- Outlet boundary conditions are used from RELAP5-3D code. Mass flow rates are specified for 4 sectors. Middle fuel assembly (No.37) is used as a constant pressure boundary condition.
- Initial conditions are calculated in Fluent code as a steady state for the boundary conditions at time = 0 s in RELAP5-3D code.

		18	26	27		
	16	17	23	24	25	
13	14	15	19	20	21	22
10	11	12	37	34	35	36
6	7	8	9	31	32	33
	3	4	5	29	30	
		1	2	28		

Core sectors and numbering of fuel assemblies

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Simulation of Coolant Mixing during MSLB Accident

Fluent assumptions and solver settings

- Realizable k-epsilon turbulence model with full buoyancy effects, standard wall functions.
- Second order upwind schemes for discretization of convective terms in all solved equations, body force weighted pressure interpolation scheme.
- Water physical properties (density, dynamic viscosity, isobaric heat capacity and thermal conductivity) are specified as piecewise-linear functions of temperature at pressure at core inlet in RELAP5-3D at time 0 s.
- Time step 0.01 s, first order implicit discretization in time.

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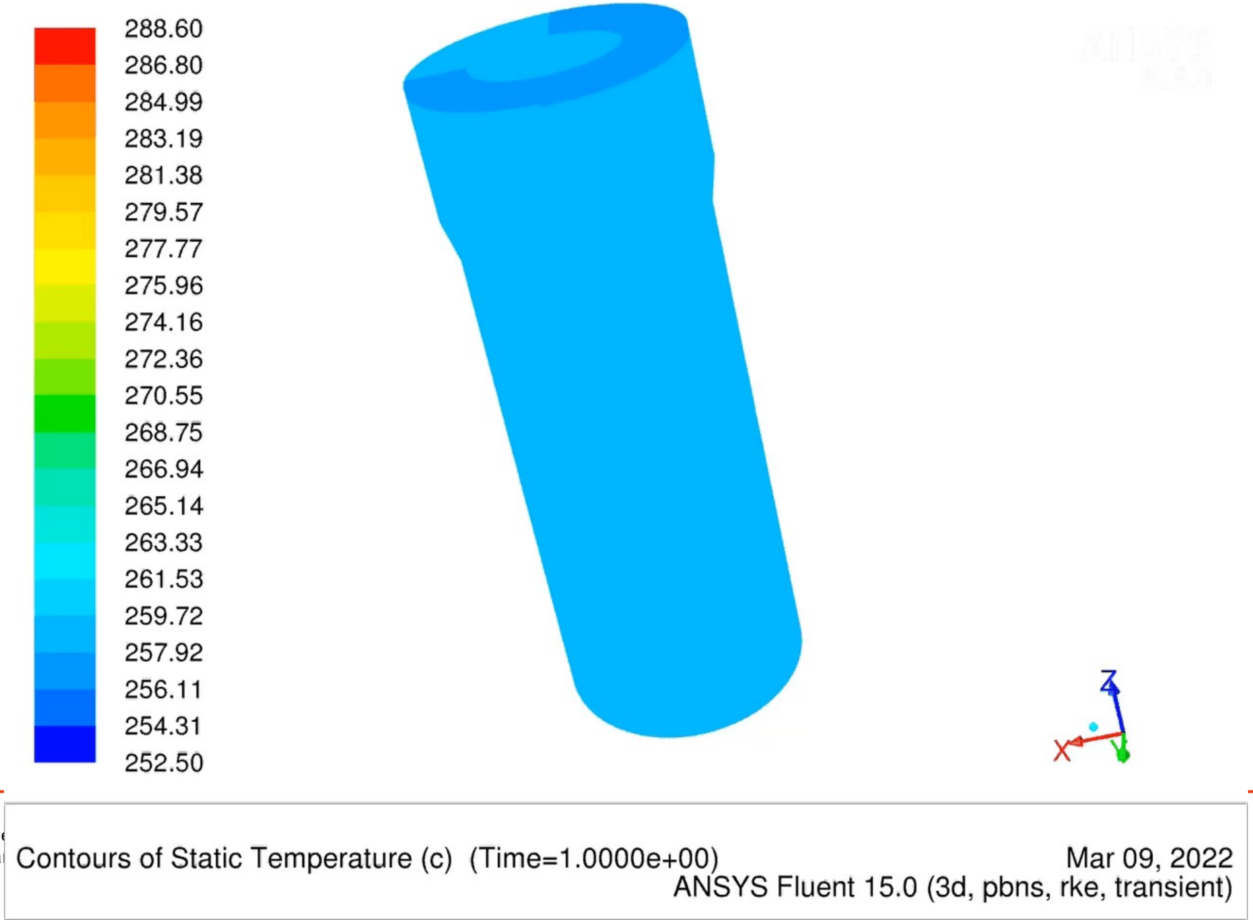
Simulation of Coolant Mixing during MSLB Accident

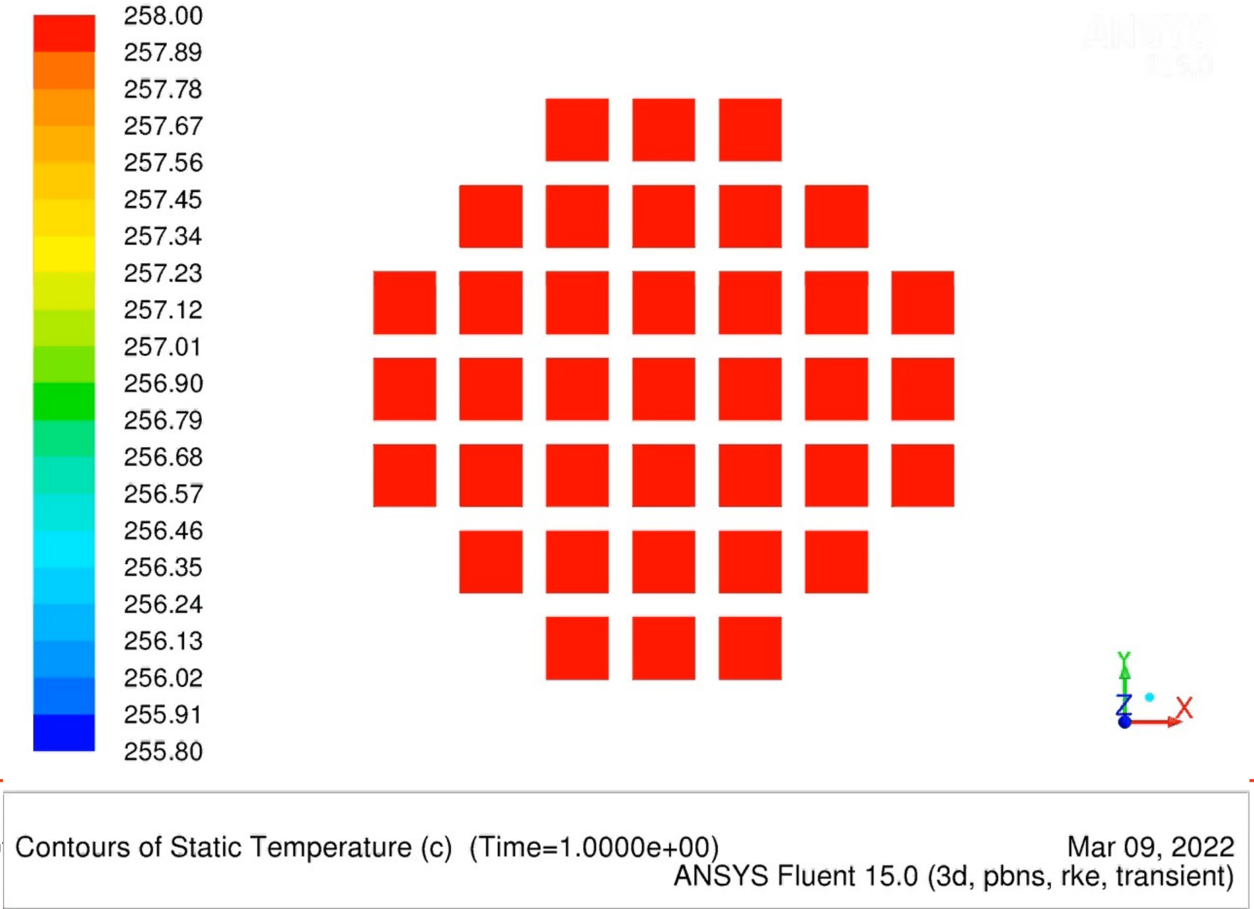
Calculated evolution of temperature field in downcomer [°C]



This project has received funding from the
training programme 2019-2020 under gra

Simulation of Coolant Mixing during MSLB Accident

Calculated evolution of temperature field [°C] at core inlet

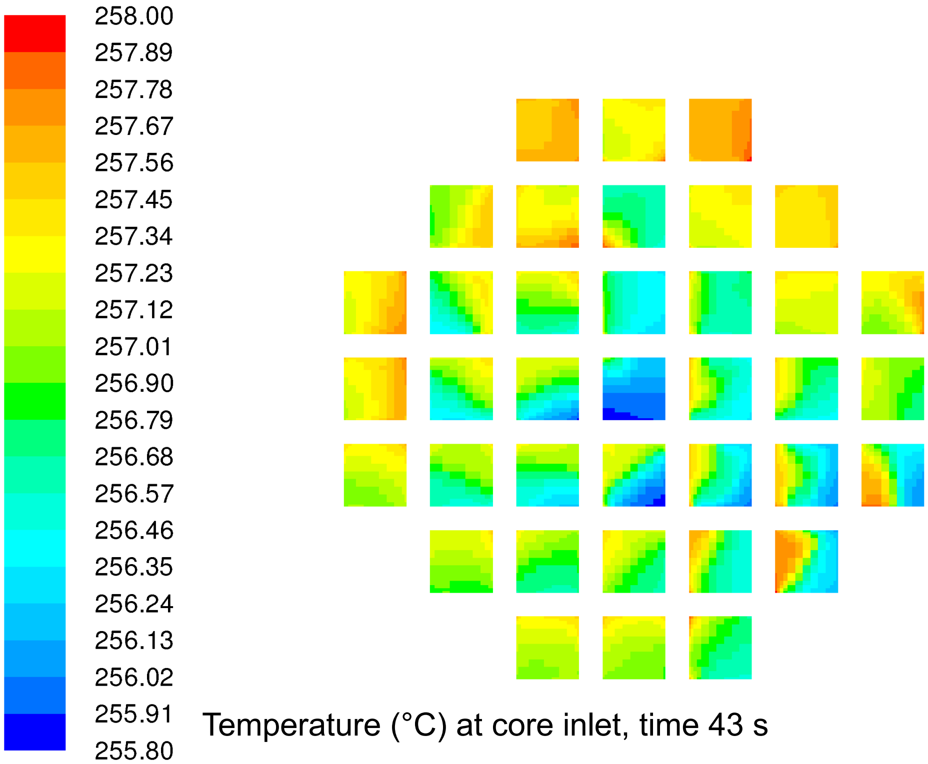


This project has received funding from the
training programme 2019-2020 under gran

Simulation of Coolant Mixing during MSLB Accident

Minimum temperature at the core inlet is achieved at time = 43 s.

Difference between maximum and minimum temperature at the core inlet is 2.1 °C.



Temperature (°C) at core inlet, time 43 s

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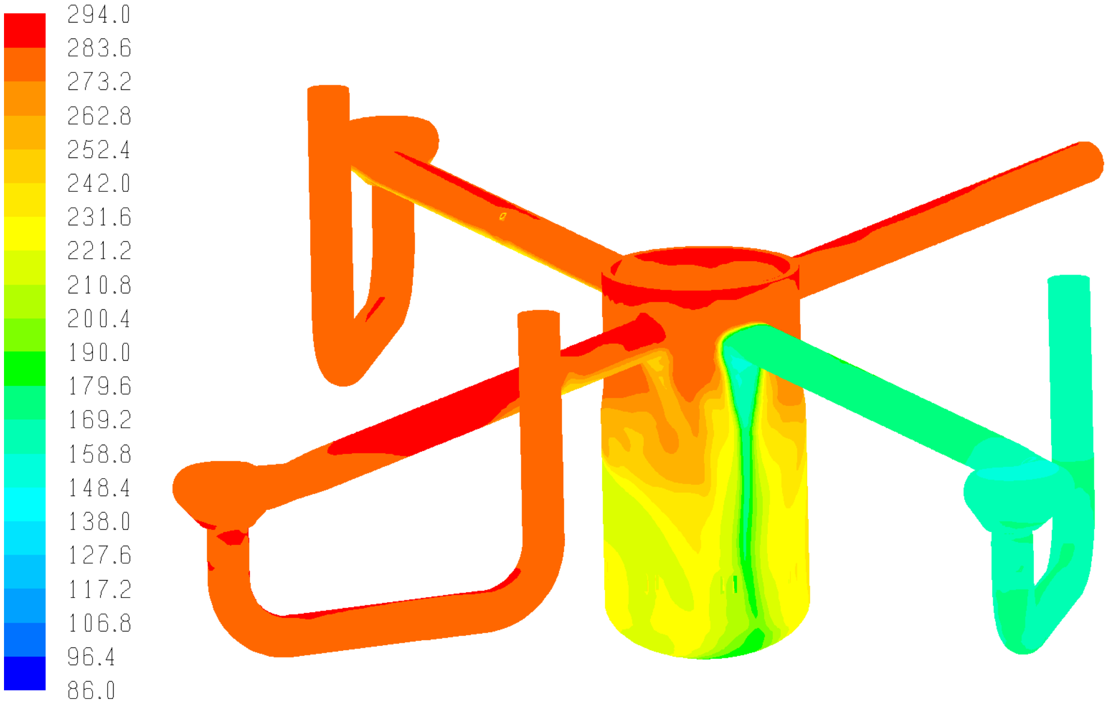
Example of MSLB Accident in VVER-1000 Reactor

Coolant mixing during MSLB scenario in large reactor looks completely different.

Here is an example of “Main steam line break at hot zero power” in VVER-1000 reactor simulated in Fluent code for the assessment of pressurized thermal shock.



VVER-1000, SLB1 scenario: wall temperatures at time 200 s

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

NuScale Mixing Test

Mixing scalar test defined by Alexander Grahn (HZDR)

Standalone simulation with prescribed boundary conditions:

mass flow: 587 kg/s

pressure: 12.8 MPa

water properties at 258 °C

1/4 of steam generator outlet is subcooled

Simulation in Fluent

Calculated flow field is not steady.

After initialization with mixing scalar = 1, 30 s of transient was performed with mixing scalar = 0 at ¼ of outlet.

After that, another 30 s was simulated with temporal averaging of variables.

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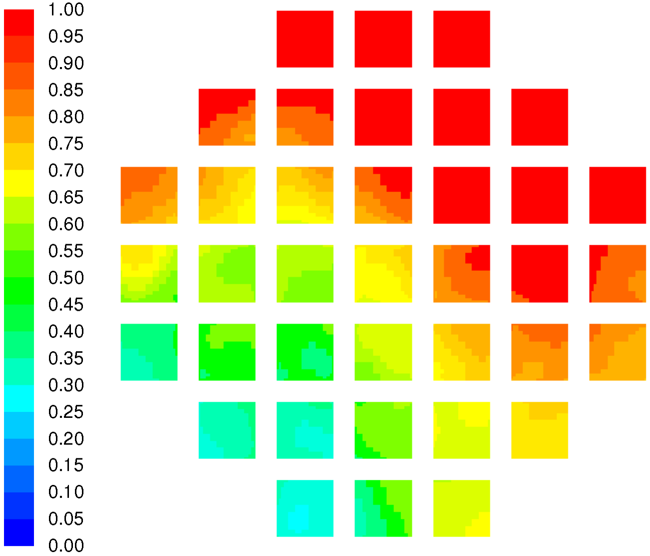
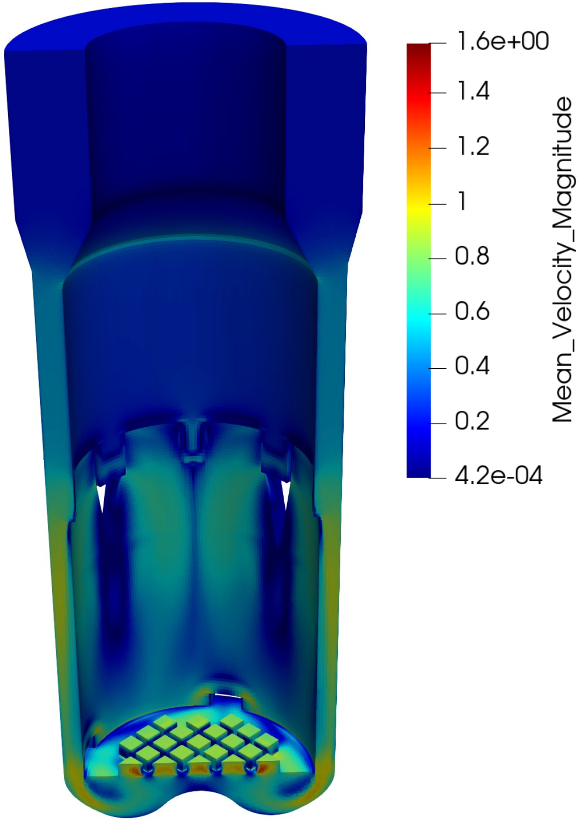
NuScale Mixing Test

Mixing scalar test
Coolant mixing at the core inlet is not completed.



Mixing scalar at core inlet (time average over 30 s)



Velocity in cell centers (time average over 30 s)

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Coupling of CFD Code with System Code

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Coupling of CFD Code with System Code

System code

- One-dimensional approach, coarse mesh
- Fast simulation (small CPU time)
- Not always satisfactory for simulation of 3D phenomena

CFD code

- Three-dimensional approach, fine mesh
- Demanding on CPU time
- More accurate simulation of 3D effects (e.g. mixing in downcomer)

Coupling

- Entire nuclear power plant is simulated by the system code.
- At the same time, the selected components are simulated by CFD code.

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Coupling of CFD Code with System Code

Coupling of codes is needed in simulations of scenarios with feedbacks between the codes.

For example:

- Temperature field at the core inlet calculated with CFD code influences solution calculated by system / neutron kinetic code.
- Pressure differences in CFD domain influence the mass flows in system code.
- Temperature field calculated in Fluent code influences operator actions in system code.

Coupling interface between Ansys Fluent and Athlet/Dyn3D was developed at UJV Rez.

Method used: Coupling of overlapped domains, explicit in time.

Vyskocil, L., Macek, J.: Coupling CFD code with system code and neutron kinetic code. Nuclear Engineering and Design 279 (2014), pp. 210-218. DOI: 10.1016/j.nucengdes.2014.02.011

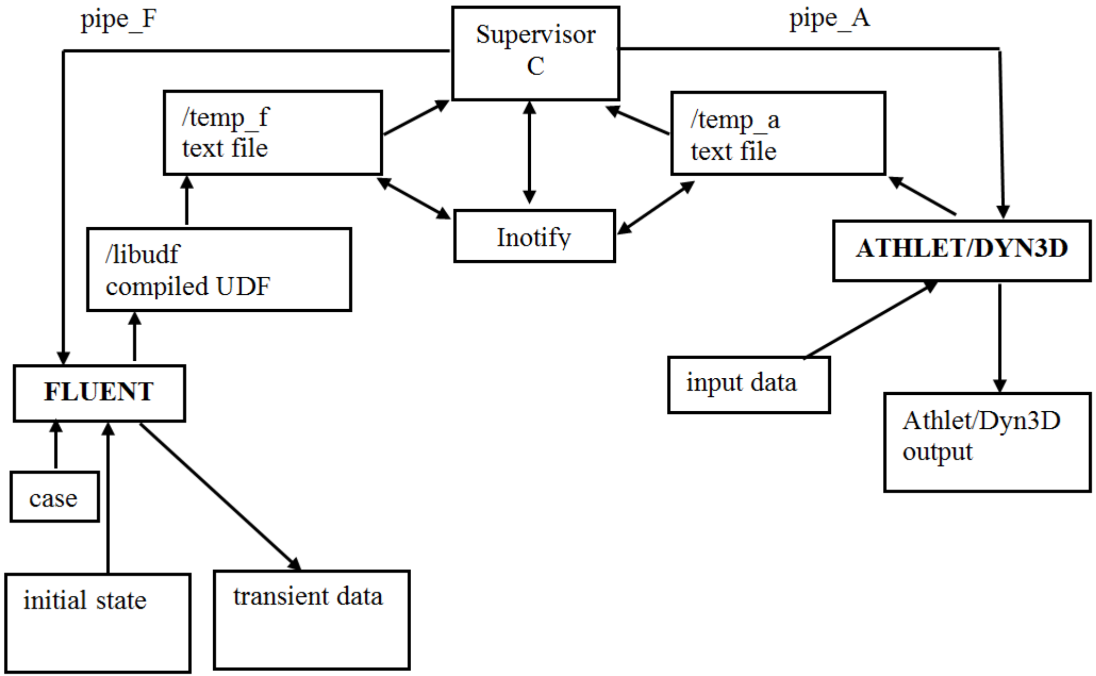
This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Architecture of coupled system

Fluent code: no modifications, without GUI, parallel run is possible

Athlet code: modified subroutines ATRANS.F and DQPOLY.F for data exchange

Dyn3D code: internally coupled with Athlet, no direct communication between Fluent and Dyn3D



UDF = user-defined function

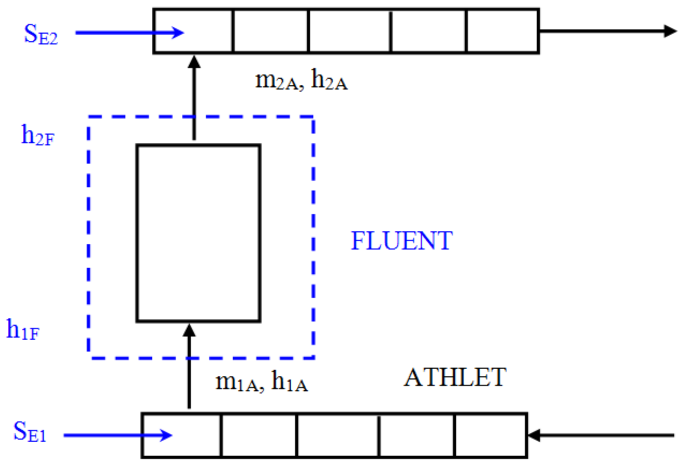
pipe = unidirectional inter-process communication channel

Inotify = Linux kernel subsystem which watches changes in file system and reports them to applications

C = C programming language

Exchange of variables between system TH code and CFD code

Temperature feedback from Fluent code into Athlet code:



m[kg/s] = mass flow
h[J/kg] = enthalpy

Energy sources S_{E1} and S_{E2} given to Athlet:

$$S_{E1} = |\min(m_1, 0)| \cdot (h_{1F} - h_{1A}) \quad [W]$$

$$S_{E2} = \max(m_2, 0) \cdot (h_{2F} - h_{2A}) \quad [W]$$

Flow reversal is possible.

Transformation between Athlet enthalpy and Fluent pseudo-temperature:

$$T_F = \frac{h_A - h_{ref}}{c_{P,F}} + T_{ref} \;,\; c_{P,F} = const$$

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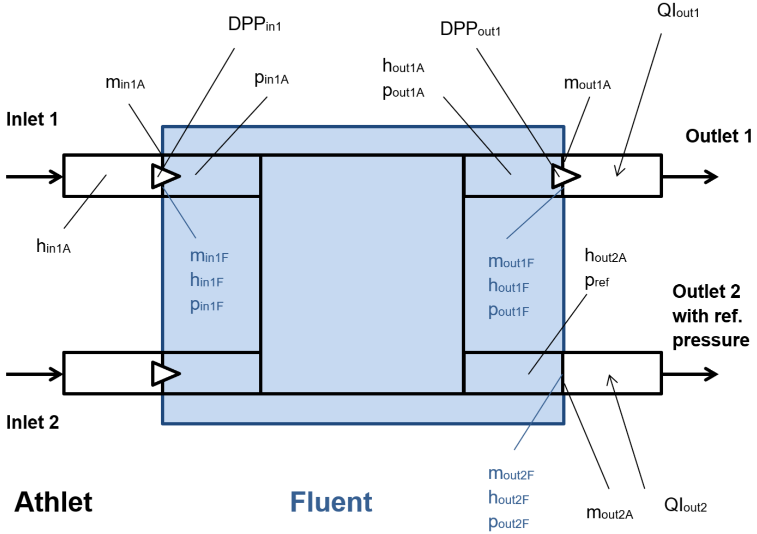
Exchange of variables between system TH code and CFD code

Temperature and pressure feedback from Fluent code into Athlet code:



Transfer of pressure differences from Fluent into Athlet:

$$DPP_{in1} = (p_{in1A} - p_{ref}) - p_{in1F} \quad [Pa]$$

$$DPP_{out1} = p_{out1F} - (p_{out1A} - p_{ref}) \quad [Pa]$$

m [kg/s] = mass flow
h [J/kg] = enthalpy
p [Pa] = pressure

QI [W] = energy source term in Athlet control volume
DPP [Pa] = pressure jump in Ahlet junction

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Real case: VVER-1000, SDA opening at 20% power

So far, the coupling has not been used for the simulation of NuScale. Athlet model of NuScale reactor is not completed yet.

Here is an example of application of coupling the real scenario at VVER-1000. Only temperature feedback is considered.

Opening of steam dump to the atmosphere at 20% of nominal power

- extensive steam release from affected MSL with subsequent decrease of steam temperature and pressure
- decrease of coolant temperature in attached primary loop
- non-uniform temperature field at the reactor core inlet
- non-uniform power distribution in the core
- temperature field at the core outlet was measured in the test

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Real case: VVER-1000, SDA opening at 20% power

Computational domains

Athlet: primary and secondary circuit

Dyn3D: reactor core

Fluent (2 domains): downcomer and lower plenum; upper plenum

Dr. Jan Hadek (UJV Rez) created the input deck for internally coupled Athlet/Dyn3D codes and processed the experimental data.

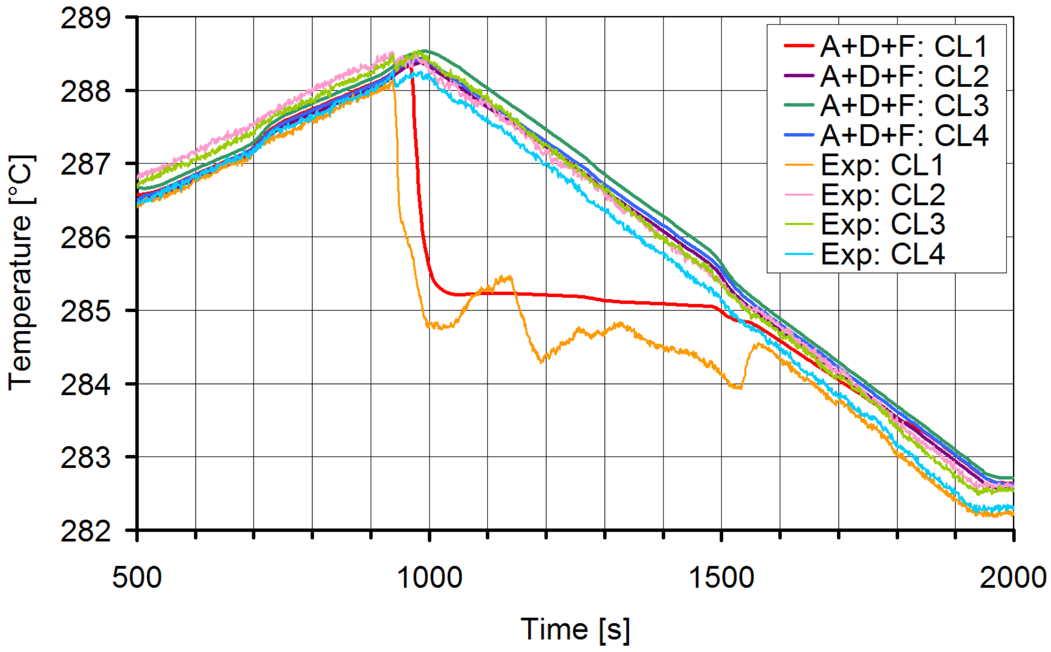
Simulation of this test by Dyn3D/Athlet without CFD code

Hadek J., Meca R., Macek J.: Validation of Thermal-Hydraulic Computing Model of VVER-1000 Temelin NPP for Coupled DYN3D/ATHLET Codes, ICONE19-43570, Japan, May 16-19, 2011

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Real case: VVER-1000, SDA opening at 20% power

Temperatures at reactor inlets



Exp = experiment

A+D+F = simulation by coupled Athlet/Dyn3D+Fluent

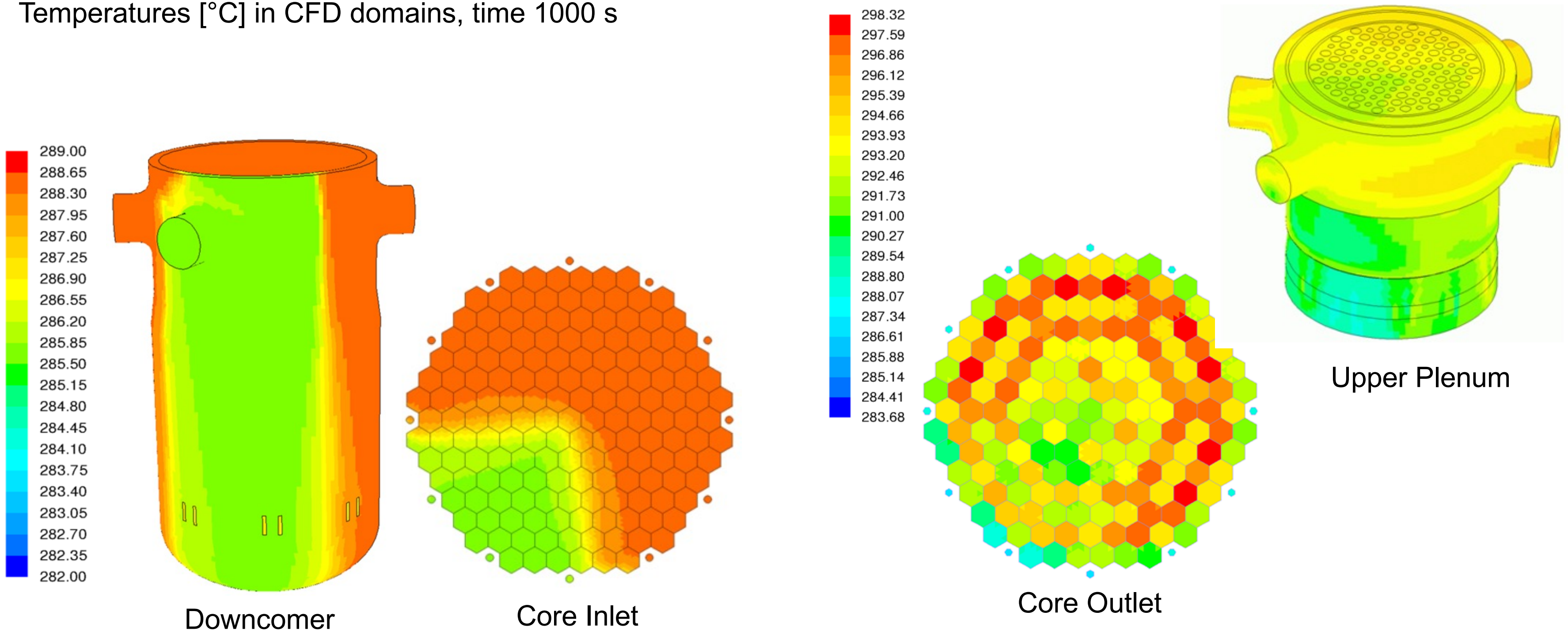
CL1 = cold leg1

Mass flow rates in loops are almost constant during the transient - approx. 4400 kg/s in each loop, variation less than 2%.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Real case: VVER-1000, SDA opening at 20% power

Temperatures [°C] in CFD domains, time 1000 s



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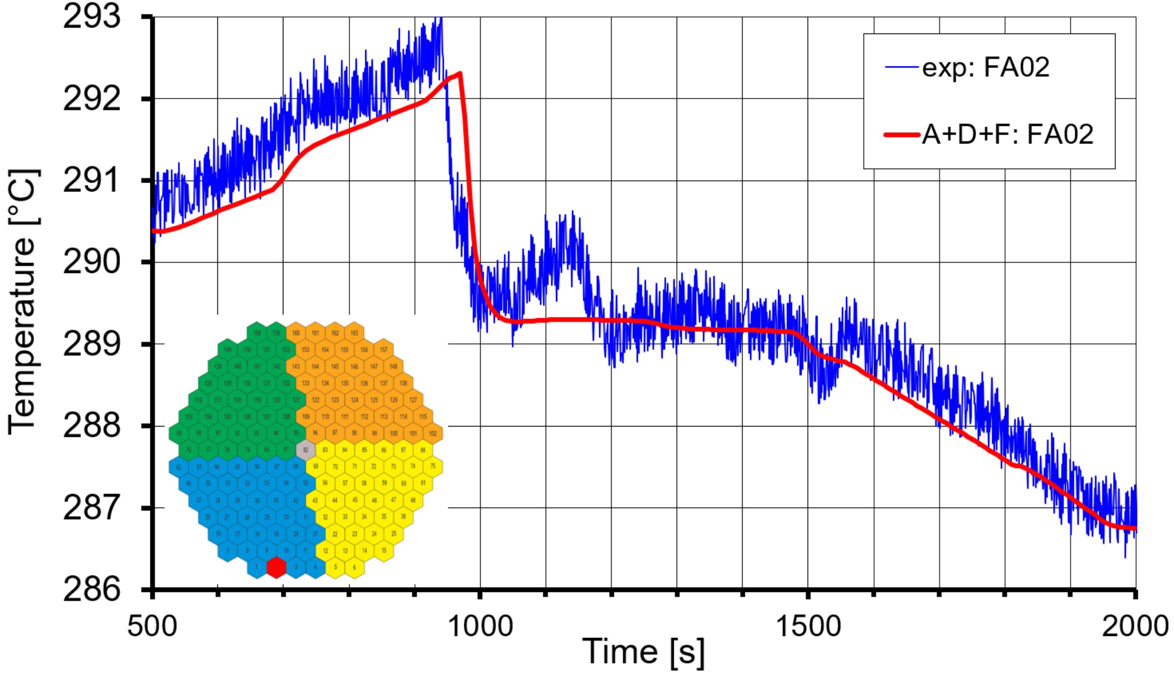
Real case: VVER-1000, SDA opening at 20% power

Temperatures at core outlet



Fuel assembly in subcooled sector

Exp = experiment
A+D+F = coupled Athlet/Dyn3D+Fluent
CL1 = cold leg1

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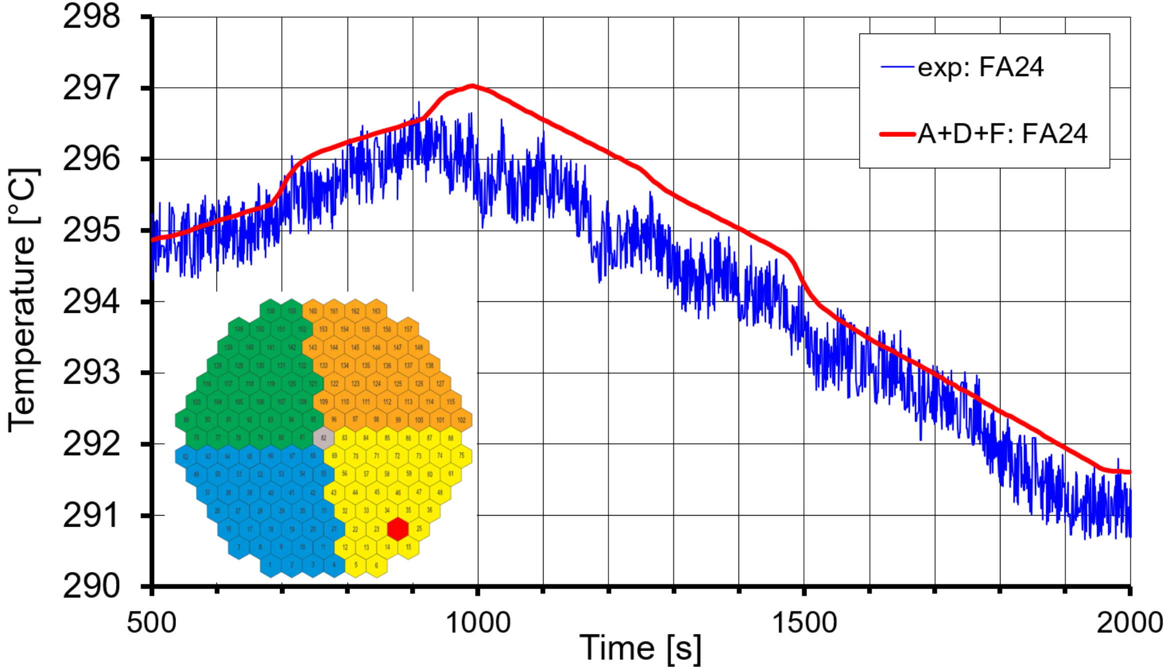
Real case: VVER-1000, SDA opening at 20% power

Temperatures at core outlet



Fuel assembly outside of the subcooled sector

Exp = experiment

A+D+F = coupled Athlet/Dyn3D+Fluent

CL1 = cold leg1

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Conclusions

- CFD code Fluent can be used for simulations of coolant mixing in NuScale reactor.
- MSLB scenario was simulated in Fluent code based on data calculated with system TH code RELAP5-3D.
- In this scenario, maximum calculated temperature difference at the core inlet is relatively small, 2.1 °C.
- Mixing scalar test revealed that unevenness of temperature distribution at the steam generator outlet during normal operation will reach the core. The mixing of coolant on its way from SG outlet to core inlet was not completed.
- CFD code Fluent can be coupled with Athlet system TH code. This method has not been applied to NuScale yet.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Acknowledgements

- Introduction to CFD is mostly based on “Lectures on Applied Computational Fluid Dynamics” by Dr. André Bakker from Ansys Inc.
- Dr. Jan Hadek and Dr. Radim Meca from UJV Rez provided invaluable help with development of coupled system ATHLET/DYN3D+Fluent.
- Dr. Marek Bencik from UJV Rez performed MSLB accident simulation in RELAP5-3D code and provided data for the CFD simulation.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Multiscale analysis of SMART

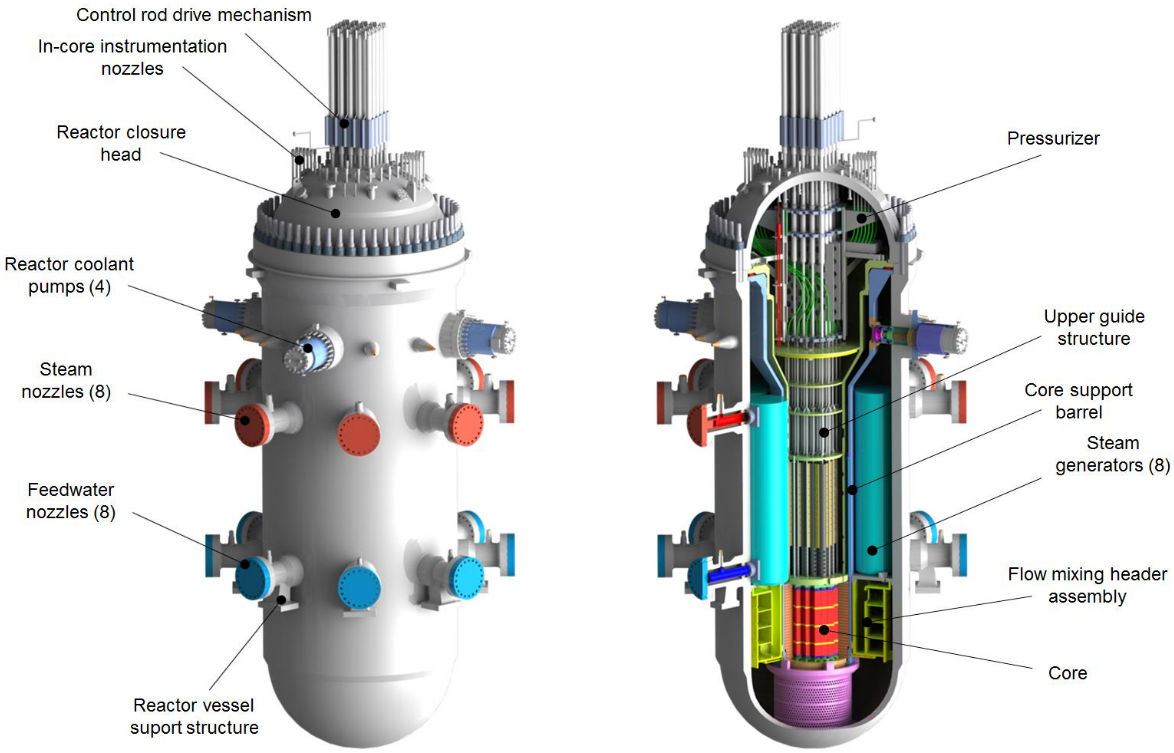
McSAFER Training Course on SMR Neutronics and Thermal Hydraulics

Manuel García – KIT (remotely)

22.03.2022

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

The SMART reactor



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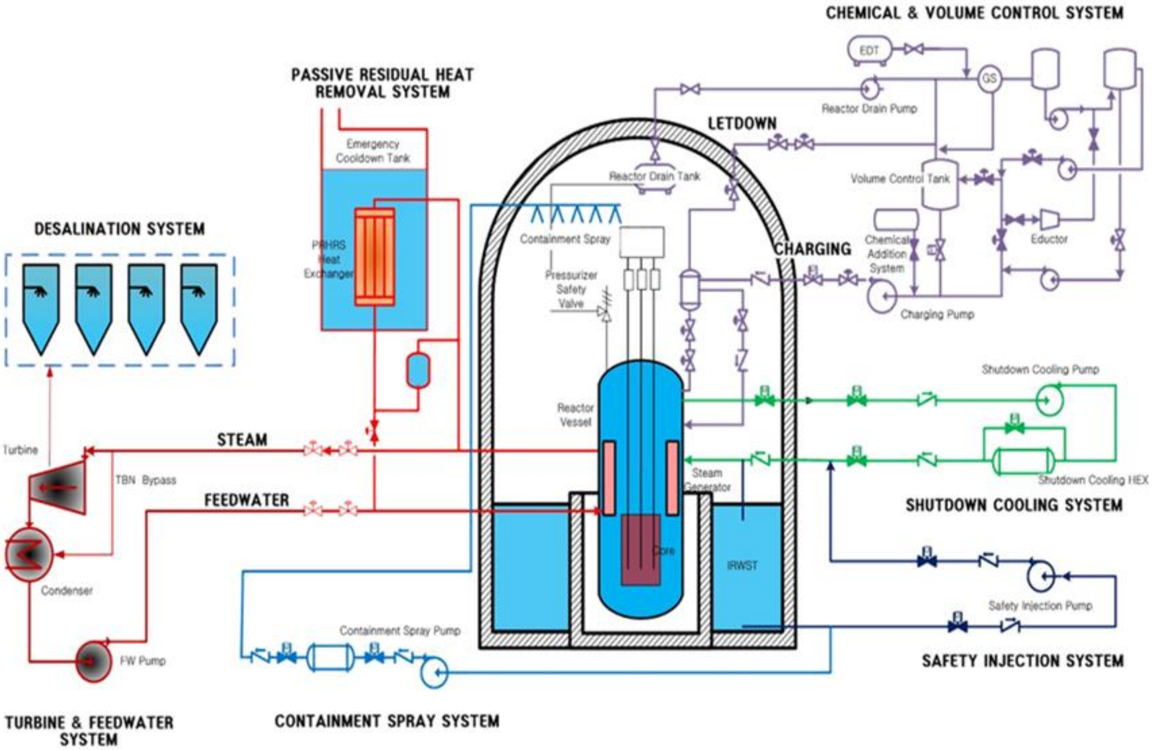
The SMART reactor



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Introduction

- Motivation:
  - Enhance modelling capabilities with respect to standard plant simulations.
  - Improve the physical description where system codes may not be appropriate.
  - Develop a suitable model for the SMART reactor:
    - Subchannel approach for the core.
    - CFD solution for the downcomer, flow-mixing header assemblies and lower plenum.
    - Standard analysis for the rest of the plant (primary and secondary loops, pressurizer, safety systems, etc).
- Calculation codes:
  - PARCS: diffusion-based nodal-level neutronics (US NRC).
  - TRACE: system thermalhydraulics (US NRC).
  - SUBCHANFLOW (SCF): subchannel and nodal-level thermalhydraulics (KIT).
  - OpenFOAM: general-purpose CFD library (OpenCFD Ltd, free software).

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General methodology

- Coupling approach for thermalhydraulics:
  - Domain decomposition:
    - No computational domains overlap.
    - Plant, subchannel and CFD models coupled through boundary conditions.
  - Implementation:
    - ICoCo-based object-oriented design.
    - Mesh-based feedback exchange.
- Neutronic-thermalhydraulic coupling for the core:
  - Standard feedback loop (power, temperatures and densities).
  - PARCS used as power source for TRACE or SCF.

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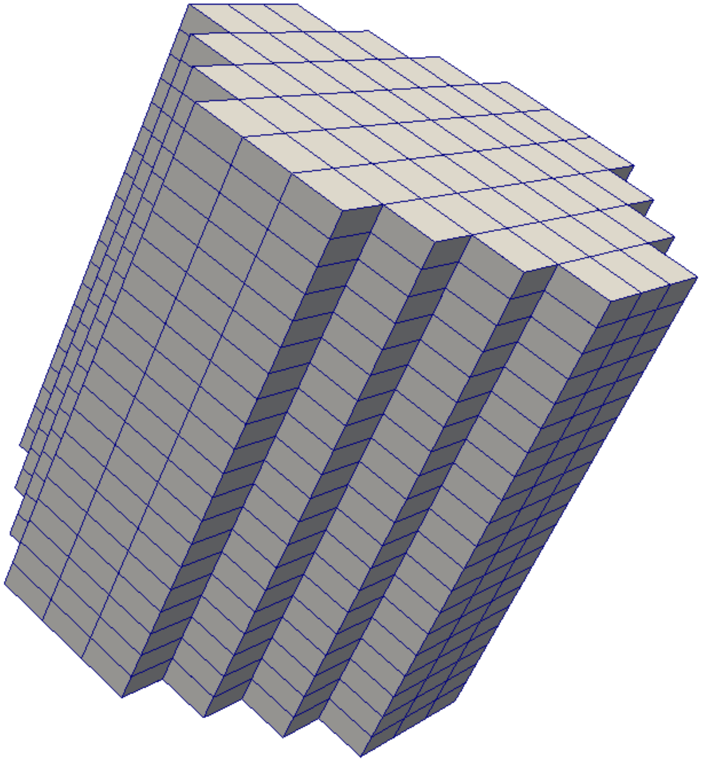
PARCS model

- Standard nodal-level core model:
  - One node per fuel-assembly.
  - PMAX cross-sections.
  - Beginning of Cycle (BOC) materials.
  - Doppler temperature and moderator density and temperature feedback.
- Neutronic-thermalhydraulic coupling:
  - SCF or TRACE densities and temperatures.
  - Power distribution for SCF or TRACE.



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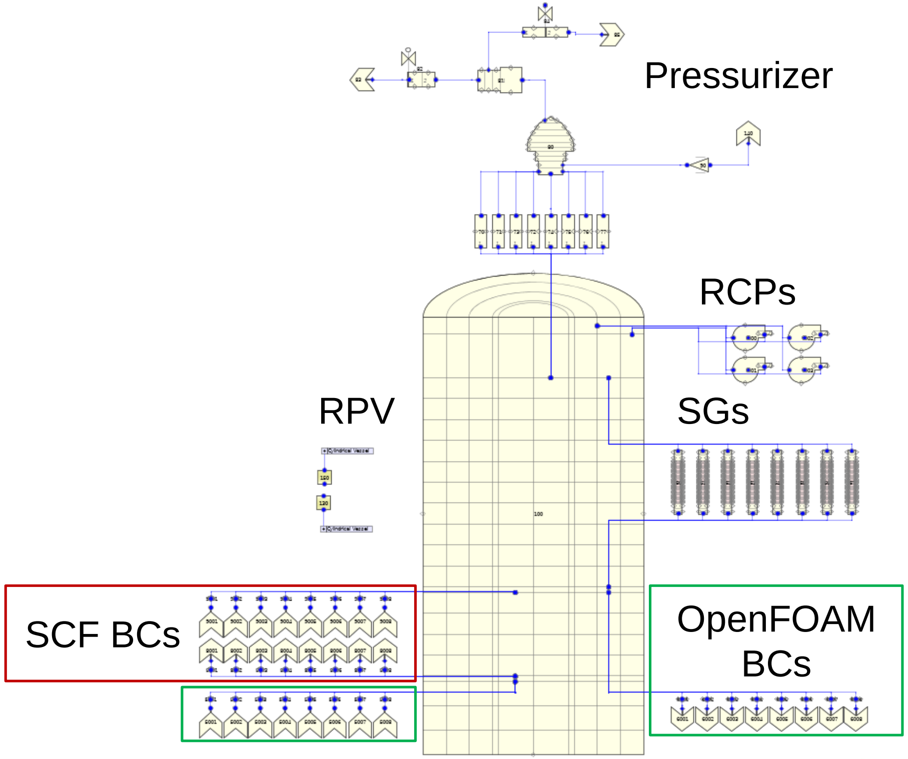
TRACE primary system

- Primary cooling system model:
  - Reactor Pressure Vessel (RPV).
  - Reactor Coolant Pumps (RCPs).
  - Steam Generators (SGs).
  - Downcomer, flow-mixing header assemblies and lower plenum (OpenFOAM subdomain).
  - Reactor core (SCF subdomain).
  - Upper plenum and core riser.
  - Pressurizer and relief valves.
- Neutronic-thermalhydraulic coupling:
  - PARCS power as heat source.
  - Densities and temperatures for PARCS.



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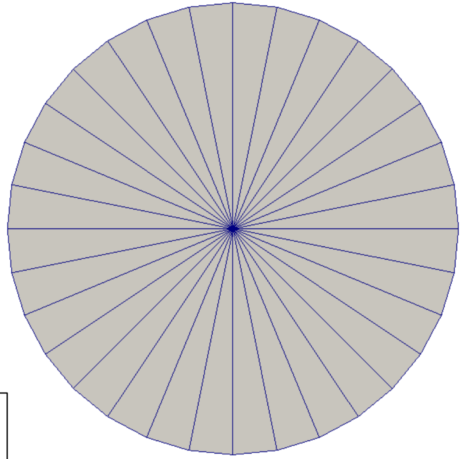
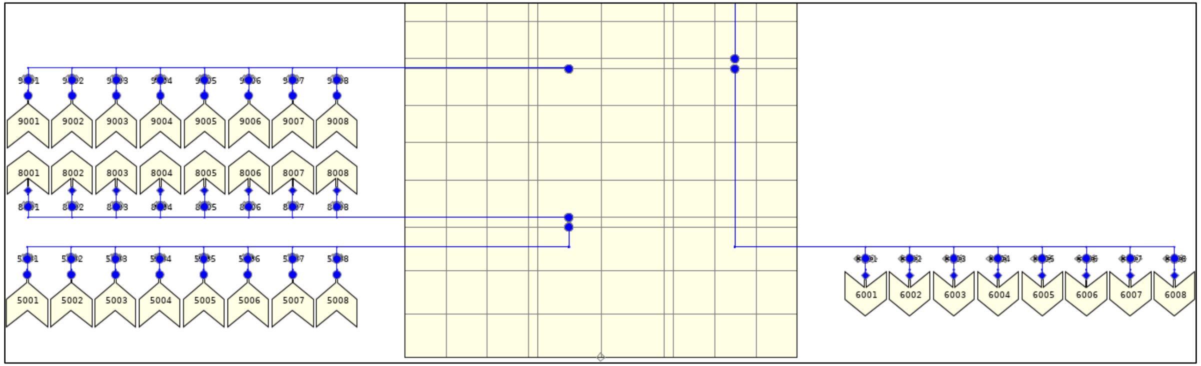
TRACE SCF core boundary conditions

- Boundary conditions:
  - Inlet pressure (breaks).
  - Outlet temperature and mass flow rate (fills).
- Output boundary variables:
  - Outlet pressure (fills).
  - Inlet temperature and mass flow rate (breaks).





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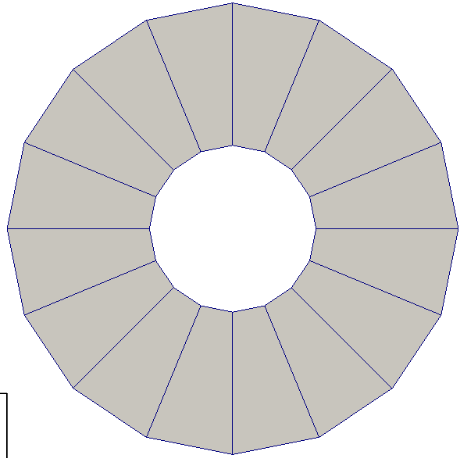
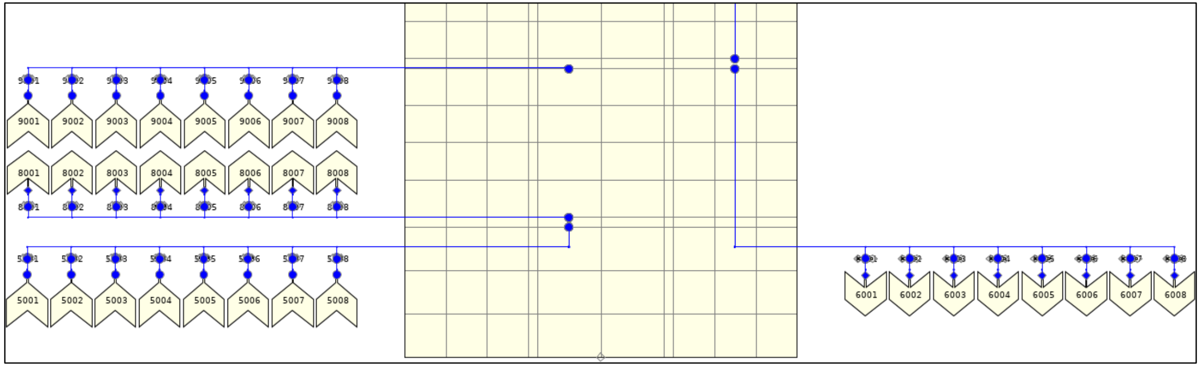
TRACE OpenFOAM boundary conditions



- Boundary conditions:
  - Downcomer pressure (breaks).
  - Core inlet temperature and mass flow rate (fills).
- Output boundary variables:
  - Core inlet pressure (fills).
  - Downcomer temperature and mass flow rate (breaks).



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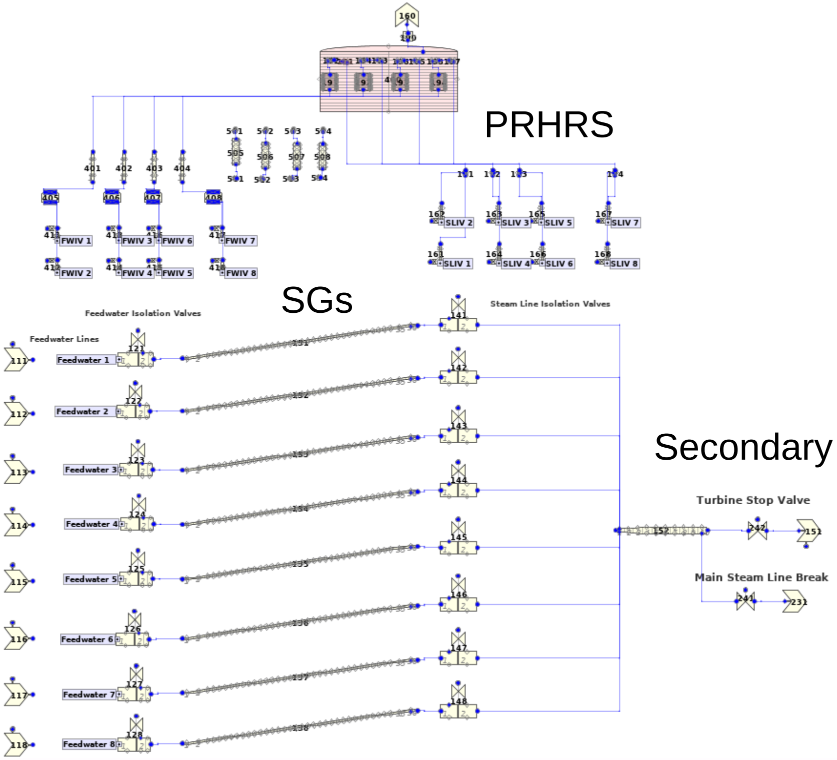
TRACE secondary system

- Secondary cooling system model:
  - Feedwater lines.
  - Steam Generators (SGs).
  - Steam lines.
  - Passive Residual Heat Removal System (PRHRS).
- Reactor trip:
  - Calculated as part of the TRACE control logic.
  - Primary- and secondary-side variables are considered.
  - The SCRAM signal is transferred to PARCS, if applicable.



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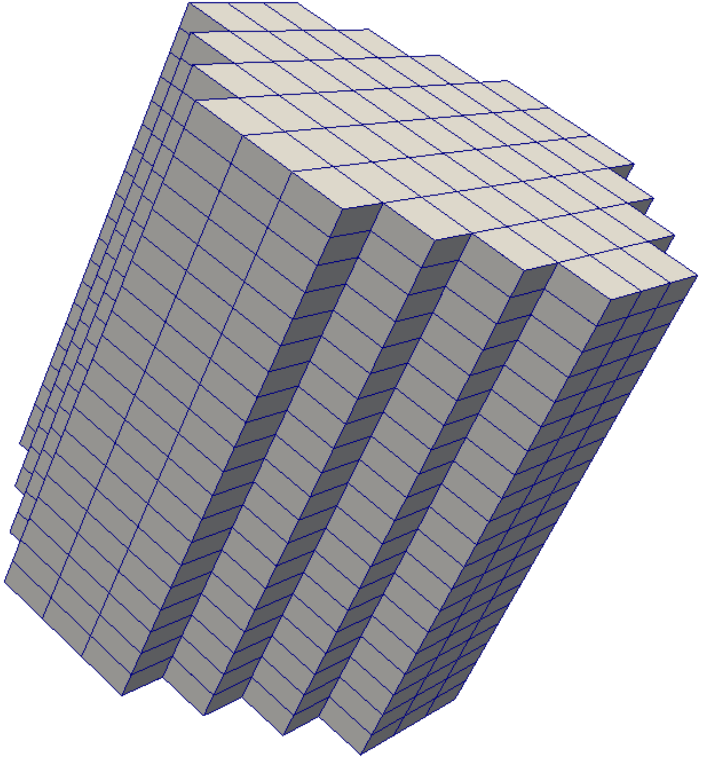
SCF model

- Standard nodal-level core model:
  - One channel per fuel-assembly.
  - One average fuel rod per fuel-assembly.
- Neutronic-thermalhydraulic coupling:
  - PARCS power as heat source.
  - Densities and temperatures for PARCS.
- Boundary conditions:
  - Outlet pressure (from TRACE).
  - Inlet temperature and mass flow rate (from TRACE or OpenFOAM).
- Output boundary variables:
  - Inlet pressure (to TRACE or OpenFOAM).
  - Outlet temperature and mass flow rate (to TRACE).



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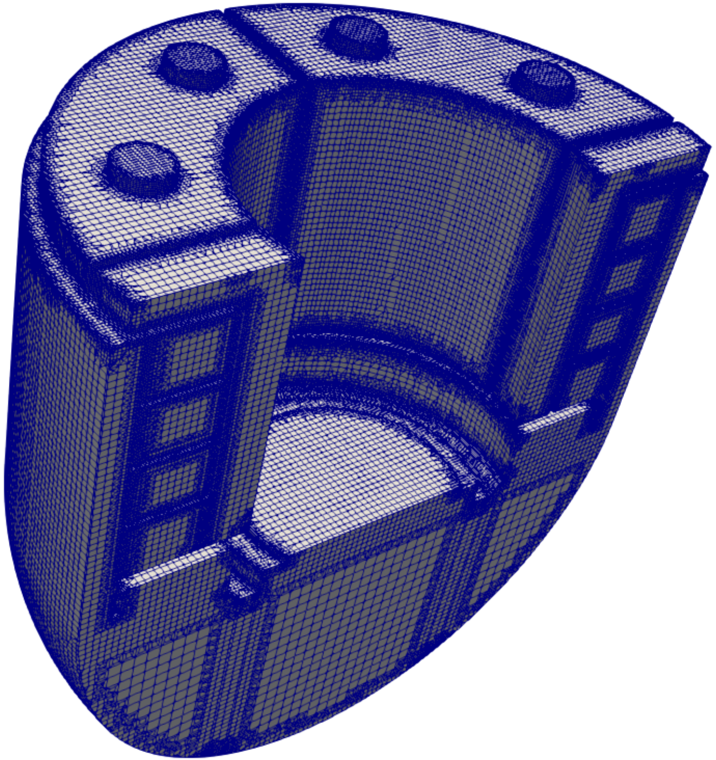
OpenFOAM model

- OpenFOAM solver:
  - PIMPLE solution algorithm.
  - Transient calculation.
  - Incompressible, turbulent flow.
  - RANS k-ε turbulence model.
- SMART model:
  - Flow inlet at the steam-generator outlet.
  - Explicit flow-mixing header assembly geometry.
  - Porous-media approach for the lower plate and the inlet fuel-assembly nozzles.
  - Flow outlet at the core inlet.



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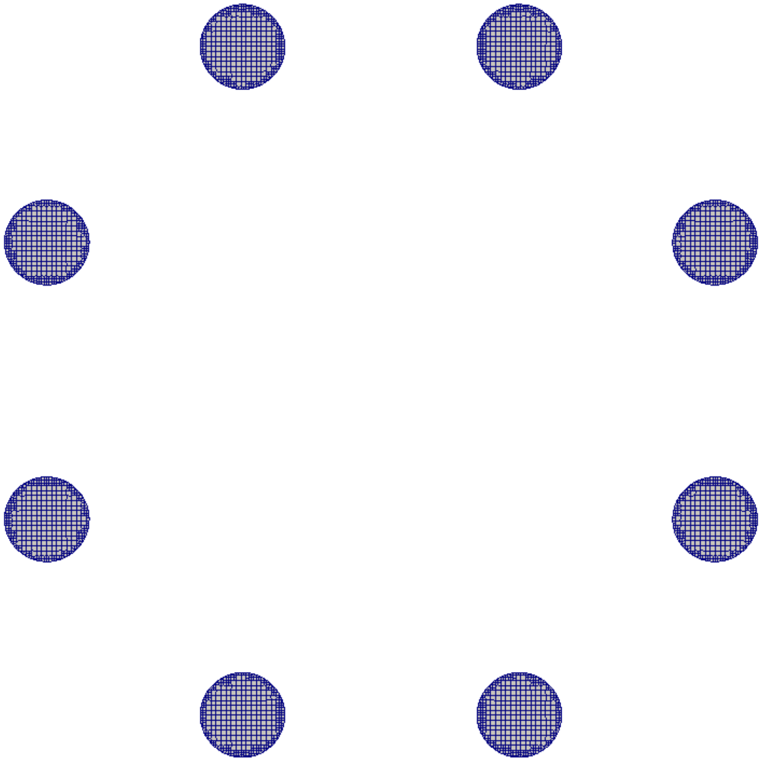
OpenFOAM downcomer inlet

- Boundary conditions (flow inlet):
  - Mass flow rate (from TRACE).
  - Temperature (from TRACE).
  - Zero-gradient pressure.
- Output boundary variables:
  - Pressure (to TRACE).



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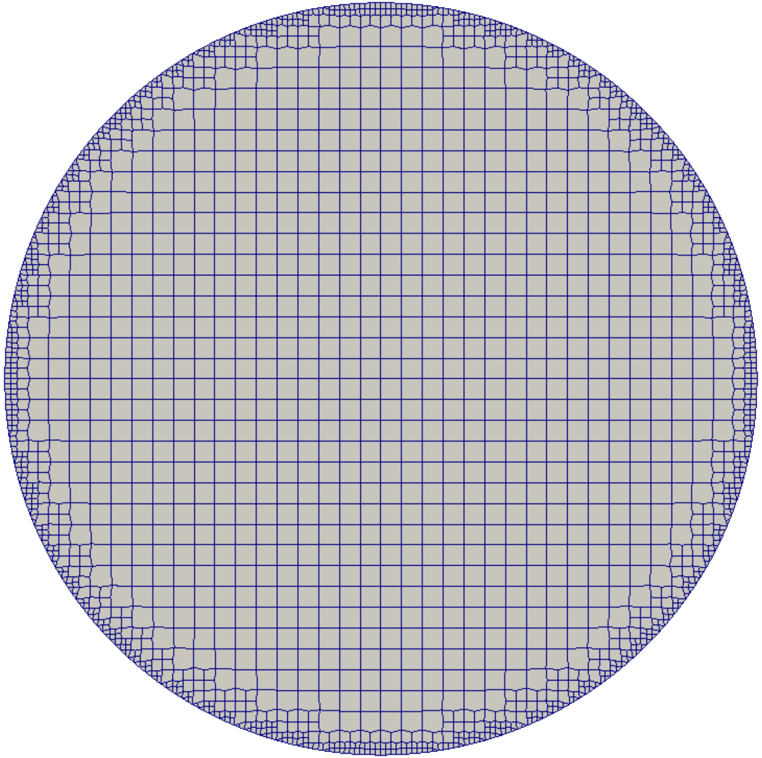
OpenFOAM lower-plenum outlet

- Boundary conditions (flow outlet):
  - Zero-gradient mass flow rate.
  - Zero-gradient temperature.
  - Pressure (from TRACE or SCF).
- Output boundary variables:
  - Mass flow rate (to TRACE or SCF).
  - Temperature (to TRACE or SCF).



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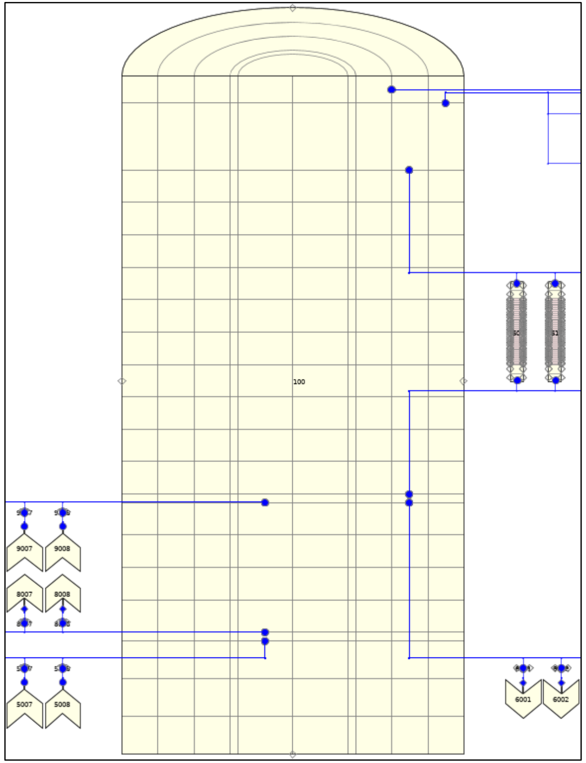
Coupling scheme

- Pressure-velocity coupling:
  - Pressure boundary conditions:
    - Fixed at outlet (calculated by another code).
    - Zero-gradient at inlet (feedback to another code).
    - Boundary conditions travel in the direction opposite to the flow (?).
  - Velocity (mass flow rate) boundary conditions:
    - Fixed at inlet (calculated by another code).
    - Zero-gradient at outlet (feedback to another code).
    - Boundary conditions travel in the direction of the flow (?).
  - Pressure instabilities can occur (and often do) in multiscale systems using domain decomposition.
- Temperature coupling:
  - Analogous to the mass flow rate.



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Mass and energy conservation

- Mass conservation:
  - Ensured exchanging mass flow rate instead of velocity.
  - The mass balance is right even if densities vary between codes.
  - Velocities and densities may be discontinuous across domains.
- Energy conservation:
  - Not ensured exchanging temperatures instead of (e.g.) enthalpy.
  - If the h(p, T) functions vary between codes the energy balance is not exactly satisfied.
  - Temperature is continuous across domains.

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Multiscale analysis in the McSAFER project

- Coupled calculation systems:
    - PARCS-TRACE.
    - PARCS-SCF-TRACE.
    - PARCS-TRACE-OpenFOAM.
    - PARCS-SCF-TRACE-OpenFOAM.
    - A single SMART model for each code is maintained and used for all cases.
- Transient scenarios for SMART:
    - Anticipated Transient Without Scram (ATWS).
    - Main Steam Line Break (MSLB).

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PARCS-SCF-TRACE ATWS simulation

- Main transient parameters:
  - Initial plant state: nominal operating conditions at 330 MW.
  - Initiating event: Loss of Feedwater Flow Accident (LOFA) at t = 100 s.
  - There is a 5.6 s delay in the closure of the feedwater isolation valves and activation of the PRHRS.
  - The first shutdown system (SCRAM) fails.
  - The initial 500 s of the transient are simulated, after which it is assumed that the reactor can be tripped manually.

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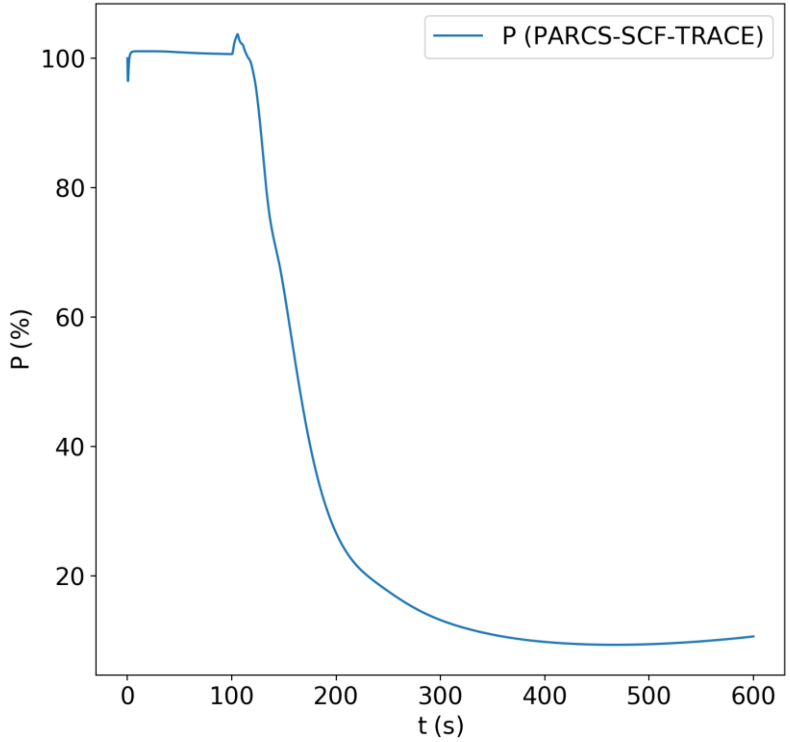
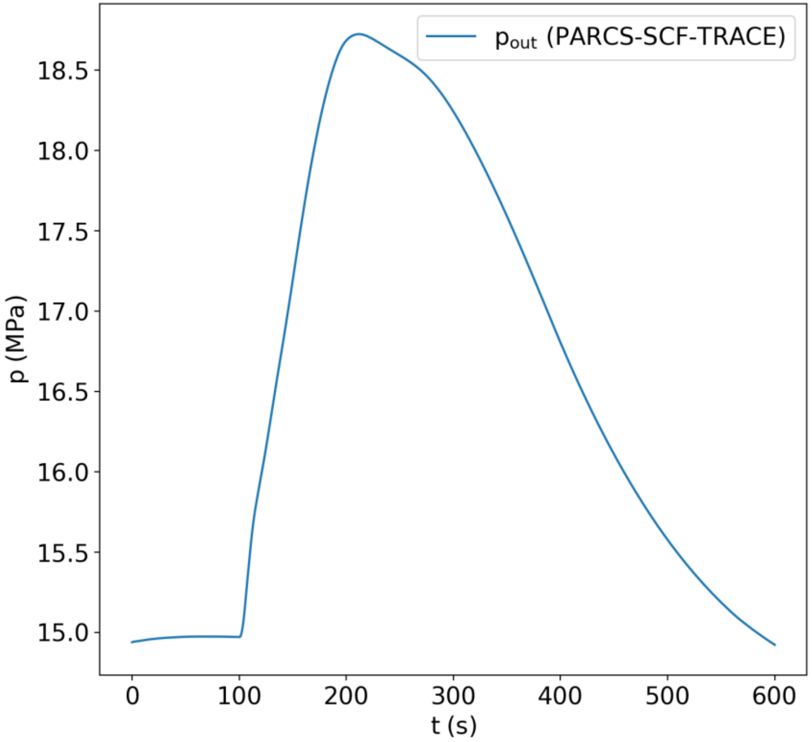
PARCS-SCF-TRACE ATWS simulation





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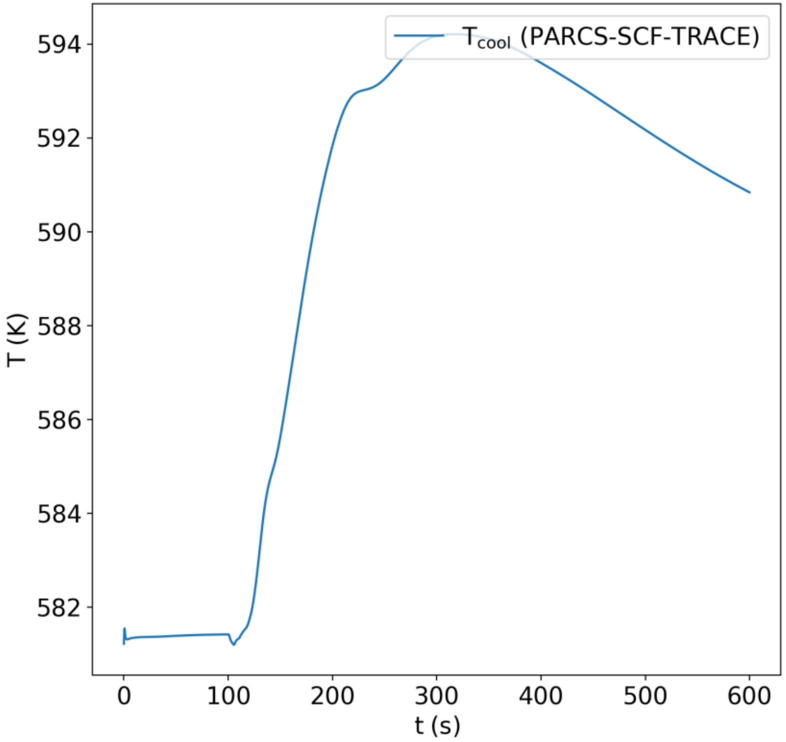

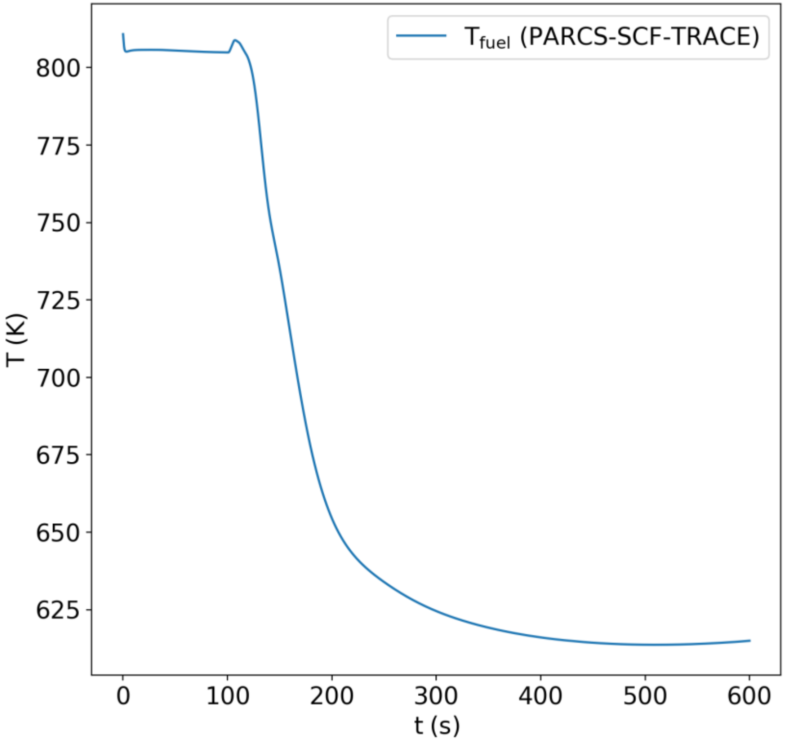

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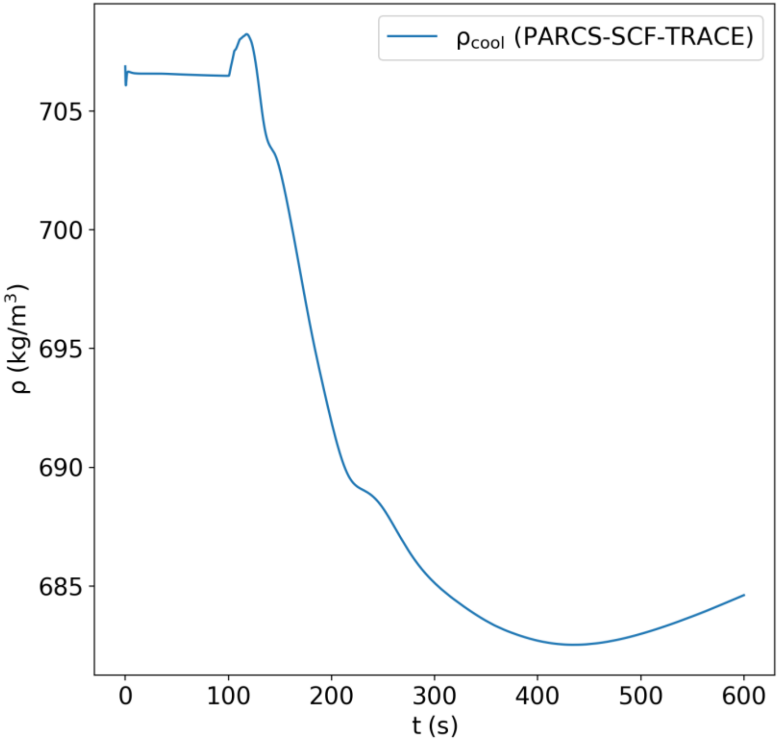
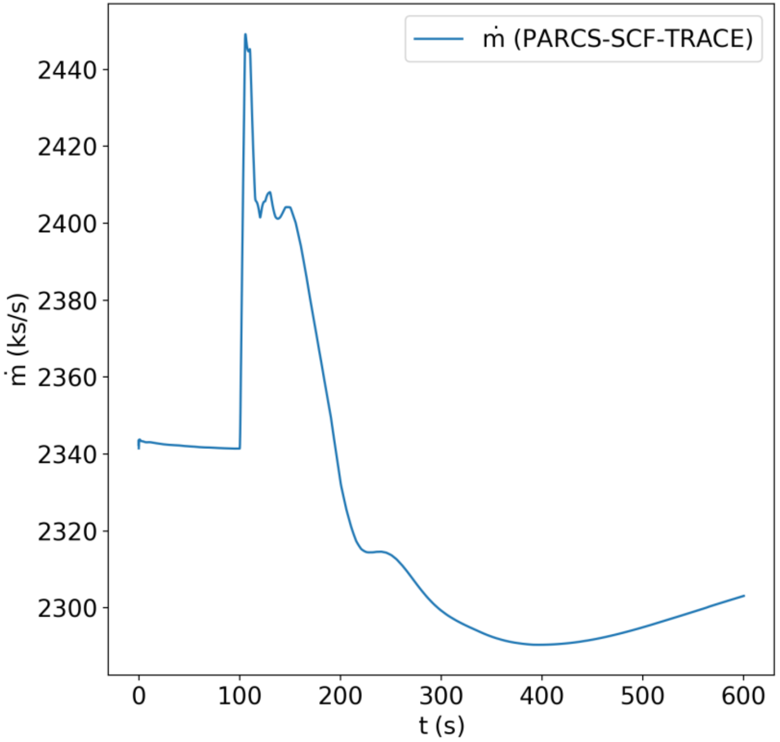
PARCS-SCF-TRACE ATWS simulation





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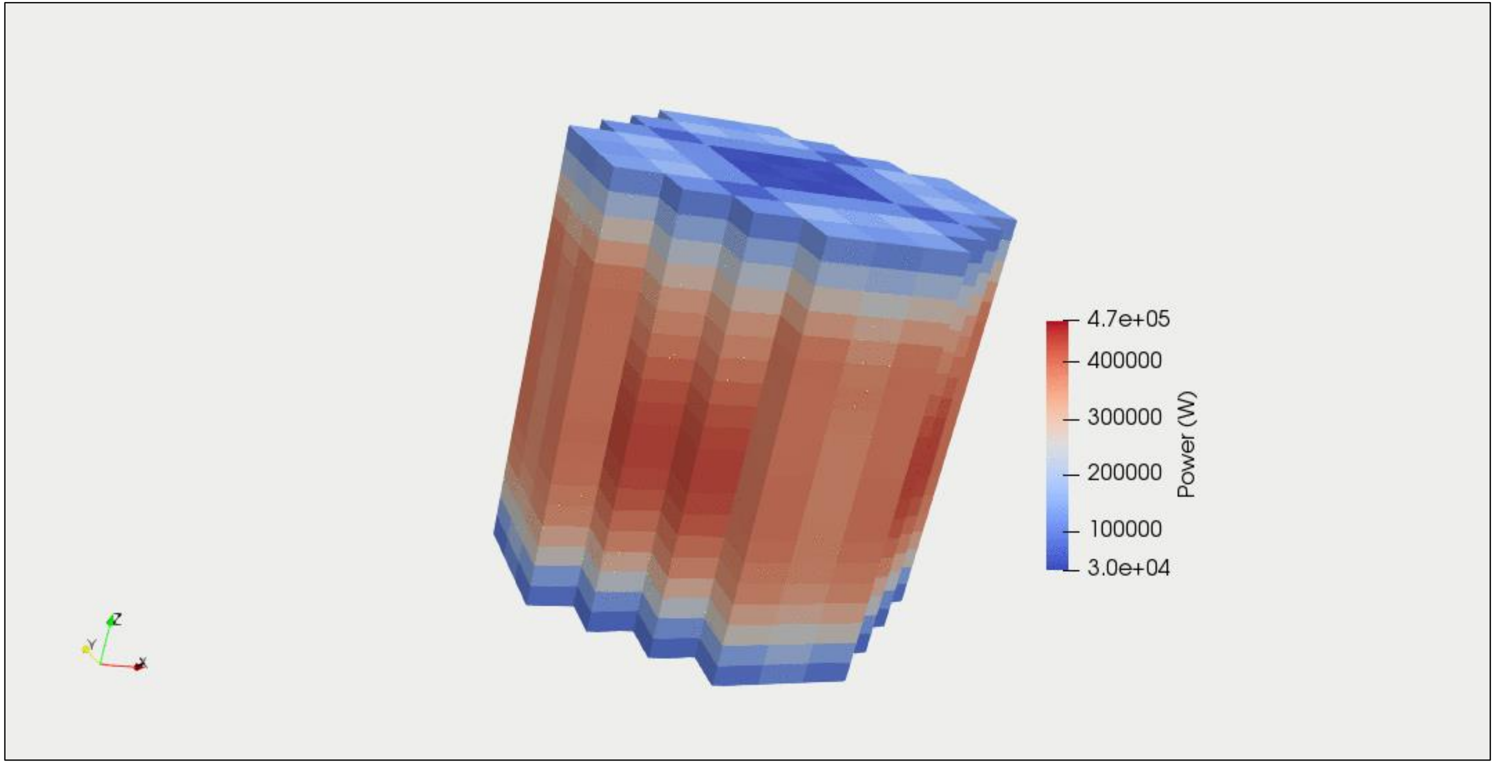

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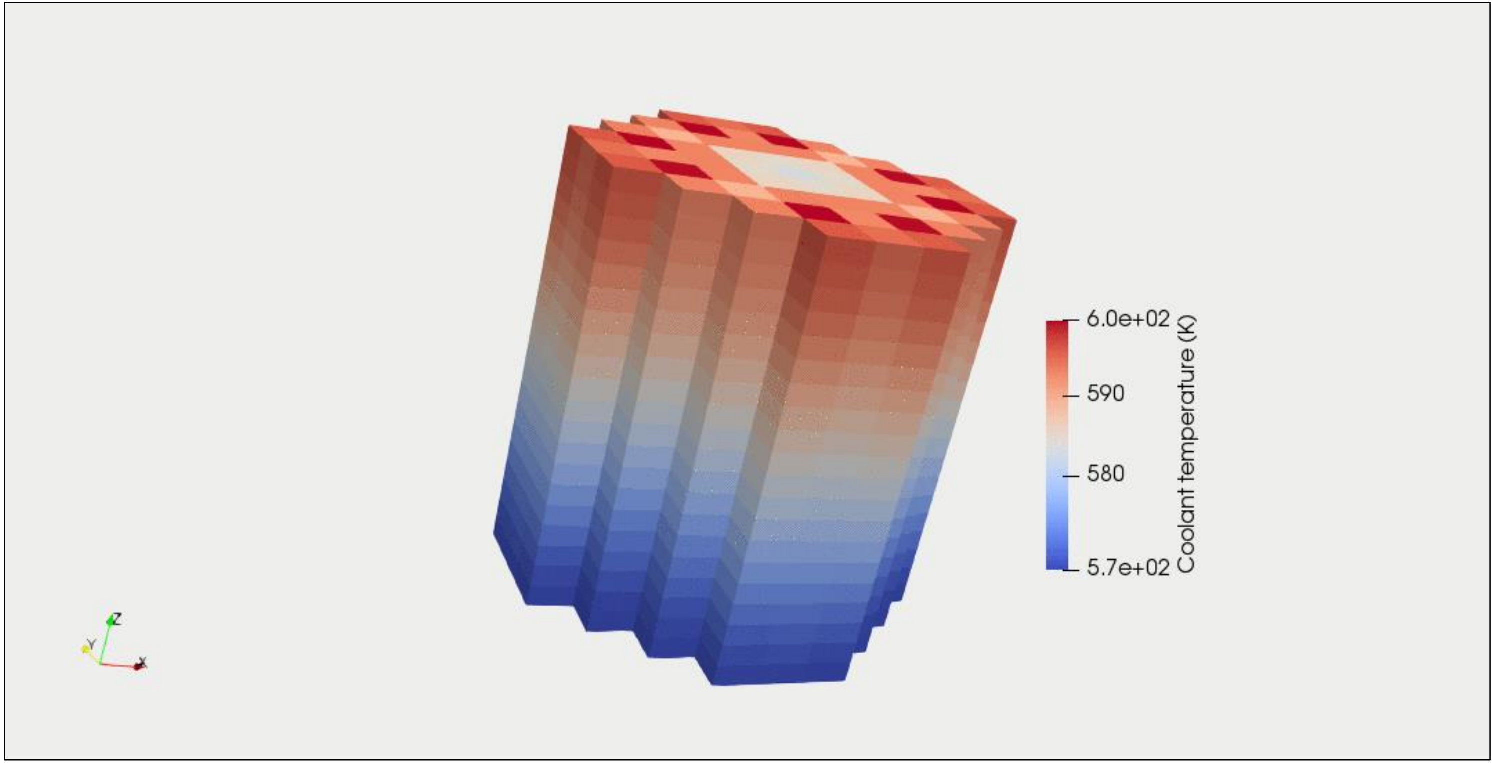

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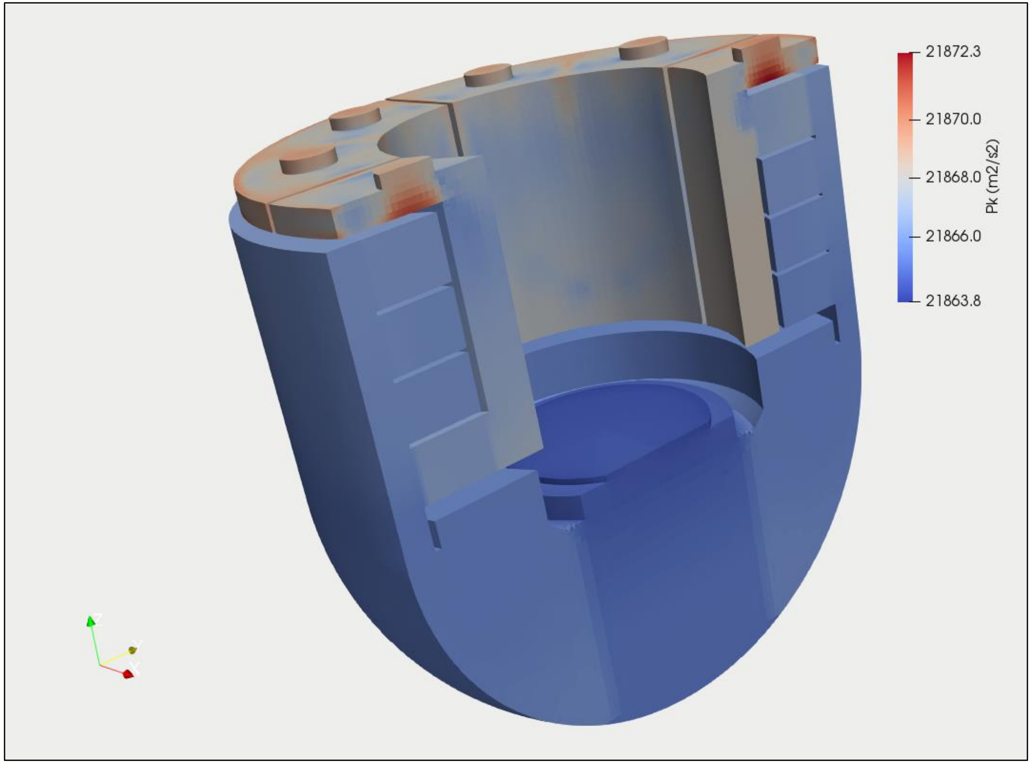
PARCS-TRACE-OpenFOAM steady-state simulation



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Fuel Cycle Design Optimization

Dr. Héctor Lestani
CNEA - Argentina

22.03.2022

McSAFER Training Course on
"SMR neutronics and thermal hydraulics"

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Instituto Balseiro
Bariloche

Presentation contents

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Motivation for this analysis to be done

- SMRs have different design enrichments, what's the reason behind that?
- If you wanted to use ATFs in SMRs, which have higher absorption materials, how can you compensate for that? Rising up Uranium enrichment?
- Can Neutronics and Thermal Hydraulics criteria answer these questions?

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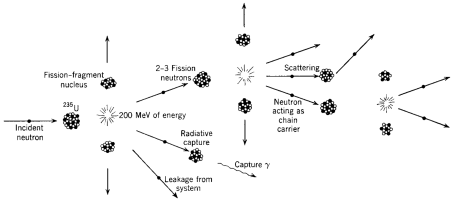
What do we know from reactor neutronics?

☛Fission Reactor: chain reaction needed.

☛Multiplication Factor: calculated quantity.
Indicates if reaction grows ($K > 1$), decreases ($K < 1$) or stays steady ($K = 1$).



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What do we know from reactor neutronics?

McSAFER

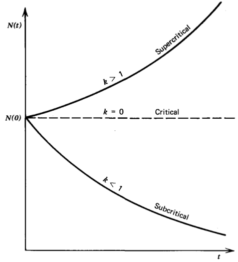
$$K = \frac{\int_{Vol}\int_0^\infty \nu(E')\Sigma_f(\vec{r},E')\phi(\vec{r},E',t)\mathrm{d}E'\mathrm{d}^3r}{\int_{Vol}\int_0^\infty \Sigma_t(\vec{r},E)\phi(\vec{r},E',t)\mathrm{d}E'\mathrm{d}^3r + \int_0^\infty \nabla^2\phi(\vec{r},E,t)\mathrm{d}E'}$$

- ☛ Fission Reactor: chain reaction needed.
- ☛ Multiplication Factor: calculated quantity.

Indicates if reaction grows ($K > 1$), decreases ($K < 1$) or stays steady ($K = 1$).

- ☛ Rod Radious: the optimum value as a function of ϵ can be calculated.

$$\begin{aligned}\frac{1}{v(E)}\frac{\partial\varphi(\vec{r},\widehat{\Omega},E,t)}{\partial t} &= S(\vec{r},\widehat{\Omega},E,t) + \frac{\chi(E)}{4\pi}\int_{4\pi}\int_0^\infty \nu(E')\Sigma_f(\vec{r},E')\varphi(\vec{r},\widehat{\Omega}',E',t)\mathrm{d}E'\mathrm{d}\Omega' \\ &+ \int_{4\pi}\int_0^\infty \varphi(\vec{r},\widehat{\Omega}',E',t)\Sigma_s(\vec{r},\widehat{\Omega}',E'\to\widehat{\Omega},E)\mathrm{d}E'\mathrm{d}\widehat{\Omega}' \\ &- \Sigma_t(\vec{r},E)\varphi(\vec{r},\widehat{\Omega},E,t) - \widehat{\Omega}\nabla\varphi(\vec{r},\widehat{\Omega},E,t)\end{aligned}$$



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CNEA

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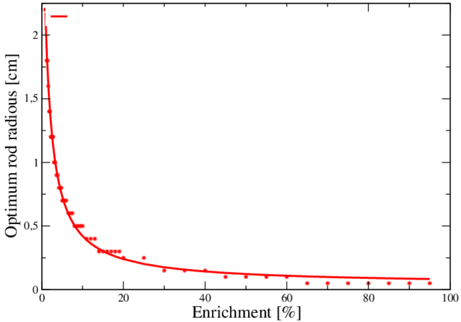
$$\frac{1}{v(E)}$$

- **Fission Reactor:** chain reaction needed.
- **Multiplication Factor:** calculated quantity. Indicates if reaction grows ($K > 1$), decreases ($K < 1$) or stays steady ($K = 1$).
- **Rod Radious:** the optimum value as a function of ϵ can be calculated.
- **Vm to Vf Ratio:** the optimum value as a function of ϵ can be calculated.



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What do we know from reactor neutronics?

$K = \cdots$

$\frac{1}{v(E)} \cdots$

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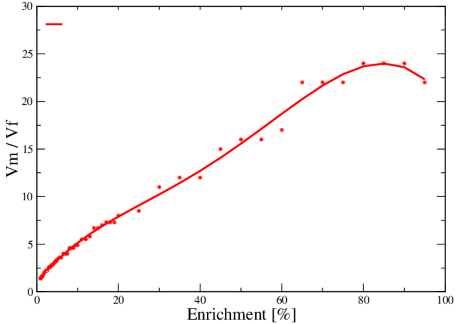
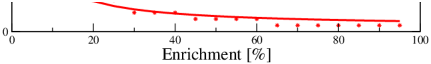
- Critical mass: the minimum value as a function of ϵ can be calculated.





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What do we know from reacto

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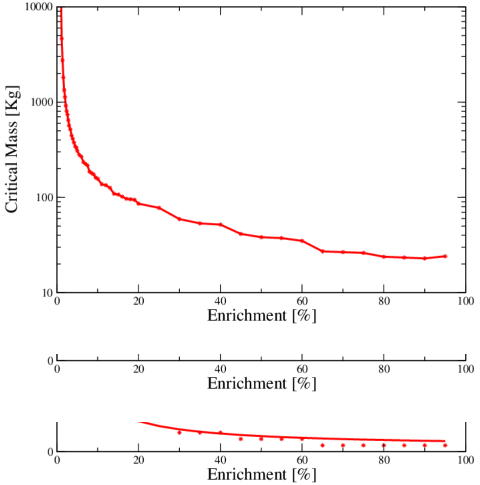
☛Critical mass: the minimum value as a function of ϵ can be calculated.

☛Burnup: the maximum value as a function of ϵ can be calculated.



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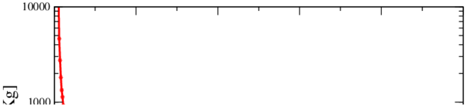
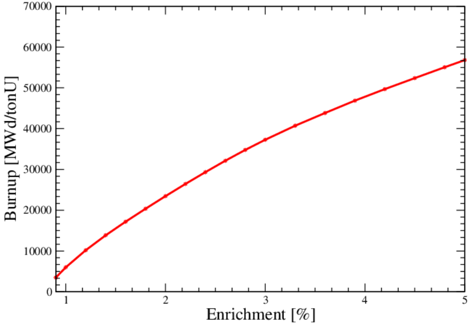
- Burnup: the maximum value as a function of ϵ can be calculated.
- "So, everything seems already settled..."





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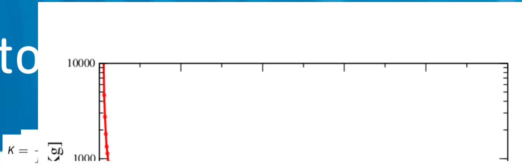

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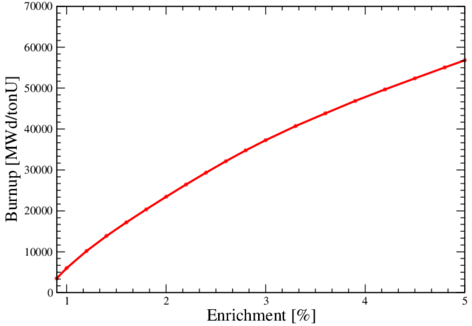
- "So, everything seems already settled..."

$\frac{1}{v(E)}$



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¿What else needs to be solved?

¿What's the open problem yet?

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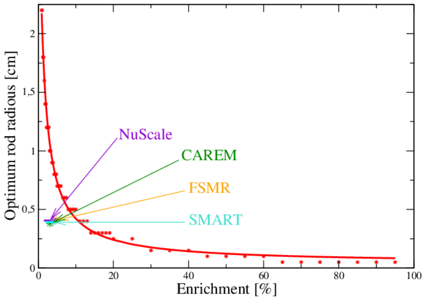
¿What else needs to be solved?

☛Rod Radious: its value is not only bound to neutronic optimization, due to its high impact also in mechanics, thermal hydraulics, fuel behavior, etc. Neutronics can't explain its value.

☛Pitch (Vm to Vf): optimizable, but high impact on safety, TH, fuel mechanics, etc. Neutronics alone can't explain its value.



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¿What else needs to be solved?

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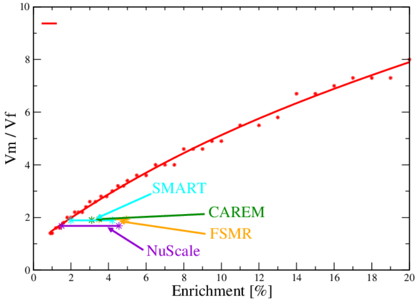
☛Pitch (Vm to Vf): optimizable, but high impact on safety, TH, fuel mechanics, etc. Neutronics alone can't explain its value.

☛Enrichment: it increases reactivity, lowers critical mass, and rises burnup. Monotonical dependencies. Presented this way, not optimizable, other aspects need to be considered.



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CNEA

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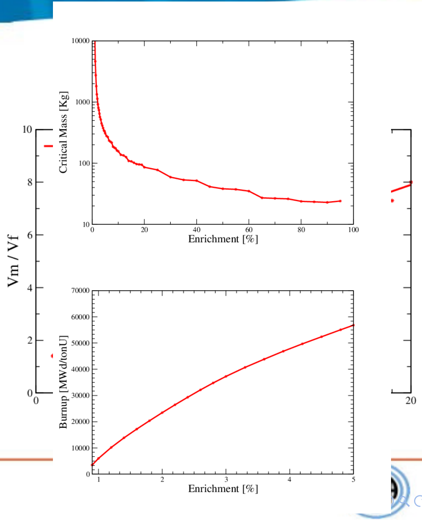

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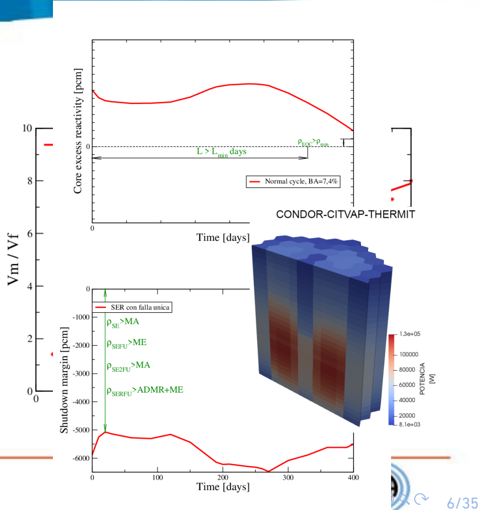

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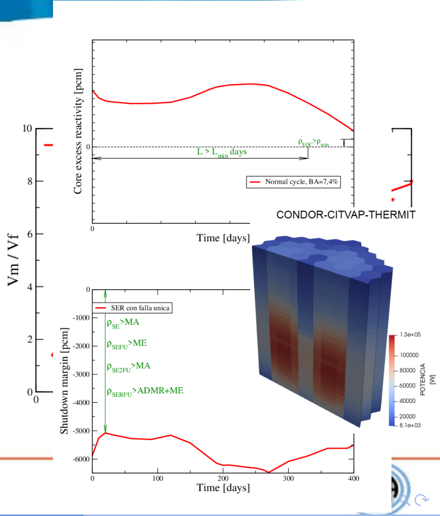

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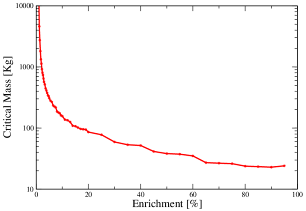
Optimization Criterion

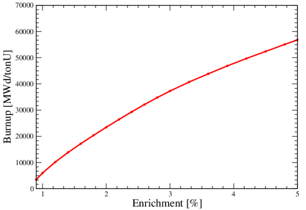
☛Enrichment rises burnup

In the four SMRs we are analyzing, with advanced design stages, the thermal power, fuel element mechanics, TH, and core mass are fixed. At constant mass, enrichment rises burnup, monotonically.
Burnup is a measure of how much energy we can extract from our fuel. It's hence desired to rise it.

☛¿Up to what level should we rise it?

Just like a screen that we'd like to buy, the bigger the better, but... how much does it cost to increase its size?
In our case, **how much does it cost to increase the enrichment?**





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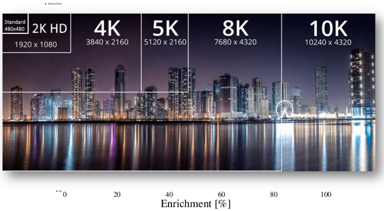

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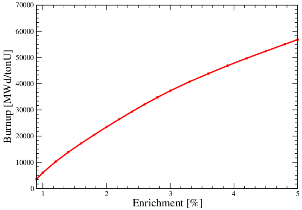

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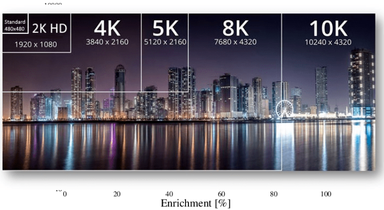

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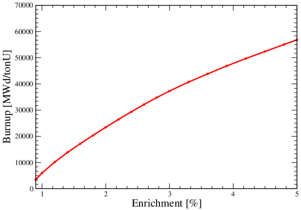

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Fuel cycle cost calculations

¿How do we model fuel cycle costs?

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Fuel cycle cost calculations(*)

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☛Refuelling cost

Calculated as the ratio between the cost of the fuels to be replaced and the energy produced during that period of time:

$$C_R = \frac{P_{th} * 365 * LF * 1000}{\eta * B_X}\left(\frac{\$}{KgU}\right)\frac{1}{P_{th} * 365 * 24 * 1000 * LF} = \left(\frac{\$}{KgU}\right)\frac{1}{\eta * B_X * 24}$$

The cost per unit energy of the replaced fuels is measured in $[\$/KWh]$

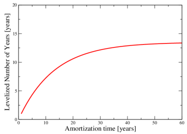
☛1st. core amortization cost

Calculated as the ratio between the cost of the first core loading, and the energy generated during the amortization time, the whole plant life:

$$C_{A1N} = \frac{\left(\frac{\$}{KgU}\right) * InvU * (1+d)^{t_{core}}}{\eta * P_{th} * 365 * 24 * LF * LNY} = \frac{\left(\frac{\$}{KgU}\right) * (1+d)^{t_{core}}}{\eta * \rho_{pow} * 365 * 24 * LF * LNY}$$

with LNY being the Levelized Number of Years of operation,

$$LNY = 1 + \frac{1}{(1+d)^1} + \frac{1}{(1+d)^2} + \cdots + \frac{1}{(1+d)^{(t_{amort})}} = \frac{[1 - (\frac{1}{1+td})^{t_{amort}}]}{[1 - \frac{1}{1+td}]}$$



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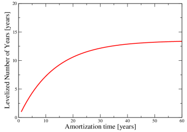

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Fuel cycle cost calculations(*)

☛Fuel cost per mass unit

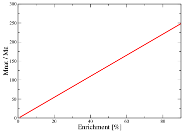
It is the cost of manufacturing the fuel, divided by the Uranium mass. It is calculated by adding up the costs of the steps involved in fuel fabrication: Mineral (YC) buying U_3O_8, conversion to UF_6, Enrichment services, conversion to UO_2 y fabrication.

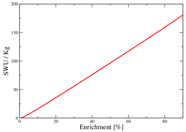
$$\left(\frac{\$}{KgU}\right) = \$_{U_3O_8}\left(\frac{M_{nat}}{M_\epsilon}\right)(1+d/100)^{T_{U_3O_8}} + \$_{UF_6}\left(\frac{M_{nat}}{M_\epsilon}\right)(1+d/100)^{T_{UF_6}} + \$_\epsilon(1+d/100)^{T_\epsilon} + \$_{UO_2}(1+d/100)^{T_{UO_2}} + \$_{fab}(1+d/100)^{T_{fab}}$$

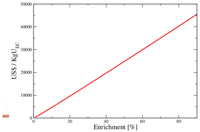
With $\$_\epsilon$ the cost of the enrichment services: the price of the *SWU* multiplied by the *SWU* needed for every Kg of Uranium: $\$_\epsilon = \$_{SWU} * \left(\frac{SWU}{KgU}\right)$, being:

$$\left(\frac{SWU}{KgU}\right) = M_\epsilon V(\epsilon) + M_{\epsilon_{waste}} V(\epsilon_{waste}) - M_{\epsilon_{feed}} V(\epsilon_{feed})$$

with $V(x) = (2x-1)ln\left(\frac{x}{1-x}\right)$ and $\left(\frac{M_{nat}}{M_\epsilon}\right) = \frac{\epsilon - \epsilon_W}{\epsilon_f - \varepsilon_W}$.







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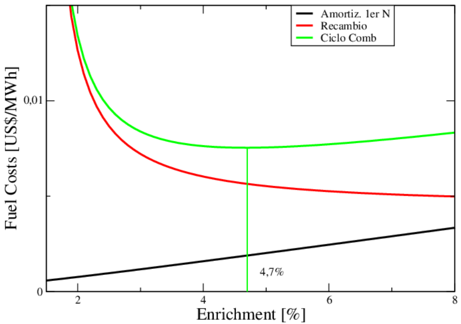
Fuel cycle cost calculations(*)

Enrichment optimization to minimize unit cost

Application of these equations to an **academic excercise using public data** corresponding to the **EPR**. The following curve is obtained.

Core data	Thermal Power [MW]	4590	Number of FEs	241	Power Density [MW/tonU]	35,53
	Electrical Power [MW]	1600	Burnup [MWd/tonU]	45.000	Load Factor	0,9
	Enrichment [%]	4,9	t amortization [years]	30	Discount Rate [%]	8
	Thermal Eficiency	0,33				
Fuel costs	Fabrication [US$/KgU]	250	t_{fabric} [years]	0,25	Conversion $UF_6 \rightarrow UO_2$ [US$/KgU]	6
	$t_{conversi}$	0,5	ϵ_{waste} [years]	0,35	Cost SWU [US$/UTS]	130
	$t_{enriquecimiento}$ [years]	1	Conversion $YC \rightarrow UF_6$ [US$/KgU]	6	$t_{convers}$	1,5
	YC [US$/KgU]	50	t_{YC} [years]	2	t_{core} [years]	1,5



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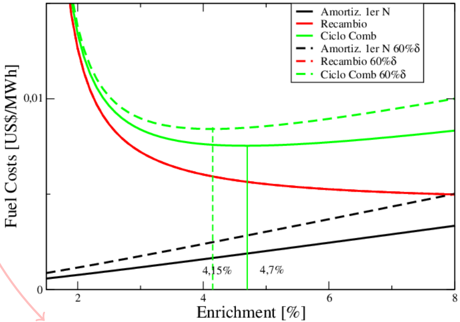

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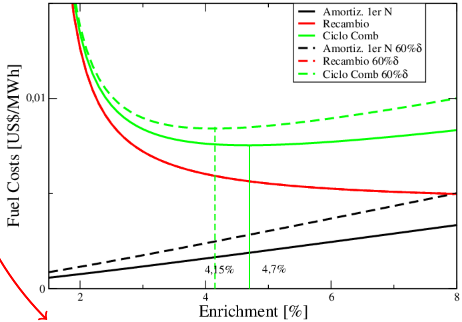
Application of these equations to an **academic excercise using public data** corresponding to the **EPR**. The following curve is obtained.

Core data	Thermal Power [MW]	4590	Number of FEs	241	Power Density [MW/tonU]	35,53
	Electrical Power [MW]	1600	Burnup [MWd/tonU]	45.000	Load Factor	0,9
	Enrichment [%]	4,9	t amortization [years]	30	Discount Rate [%]	8
	Thermal Eficiency	0,33				
Fuel costs	Fabrication [US$/KgU]	250	t_{fabric} [years]	0,25	Conversion $UF_6 \rightarrow UO_2$ [US$/KgU]	6
	$t_{conversi}$	0,5	ϵ_{waste} [years]	0,35	Cost SWU [US$/UTS]	130
	$t_{enriquecimiento}$ [years]	1	Conversion $YC \rightarrow UF_6$ [US$/KgU]	6	$t_{convers}$	1,5
	YC [US$/KgU]	50	t_{YC} [years]	2	t_{core} [years]	1,5

Neutronics...



$$B[MWd/tonU] = -18{,}000 + 12{,}857 * \epsilon[\%]$$

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Neutronic optimization of the fuel cycle

¿How is the fuel cycle neutronically optimized?

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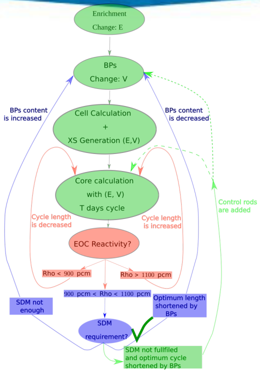
Fuel cycle neutronic optimization



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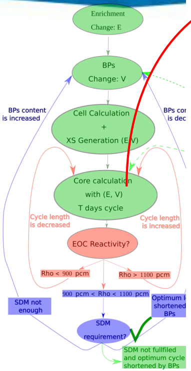
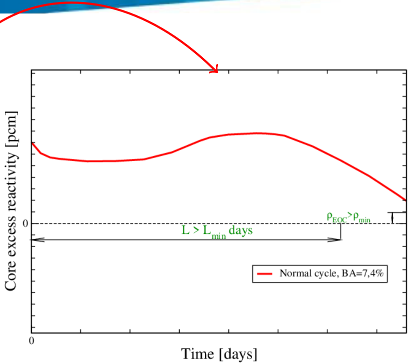
Fuel cycle neutronic optimization





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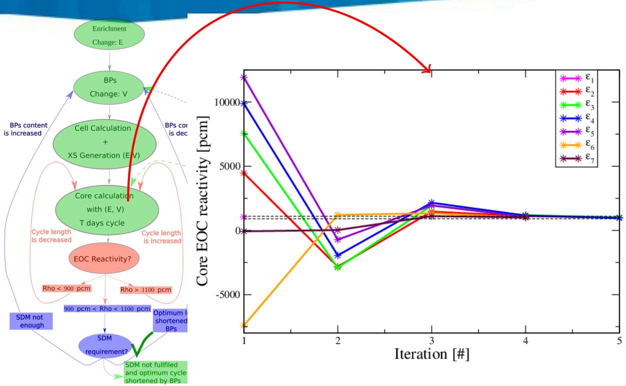
Fuel cycle neutronic optimization



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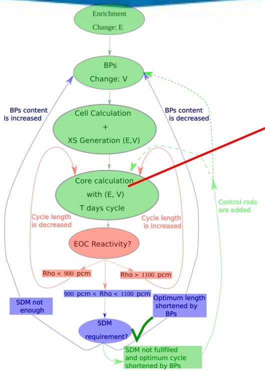
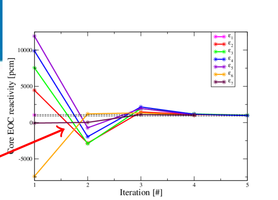
Fuel cycle neutronic optimization





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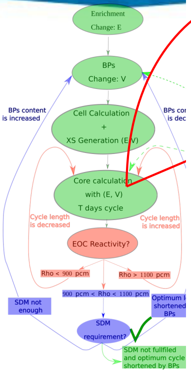
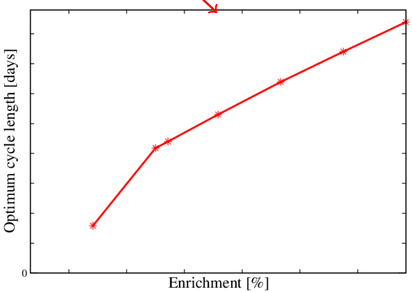
Fuel cycle neutronic optimization





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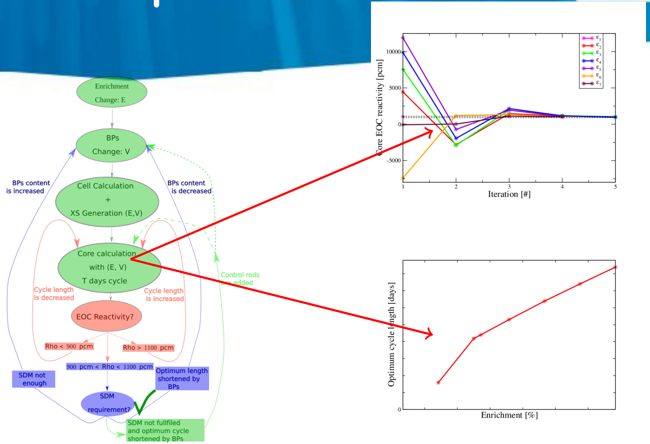
Fuel cycle neutronic optimization



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Fuel cycle neutronic optimization

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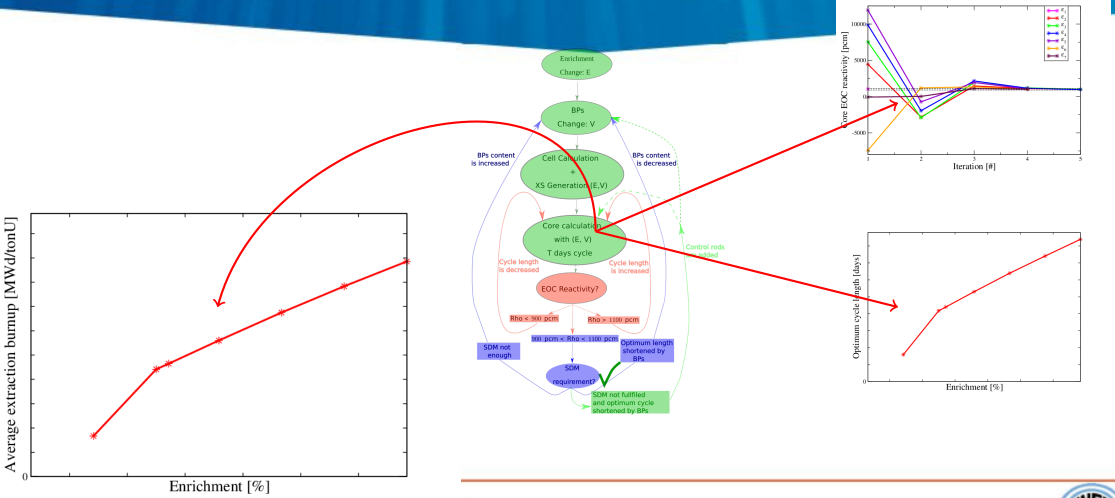

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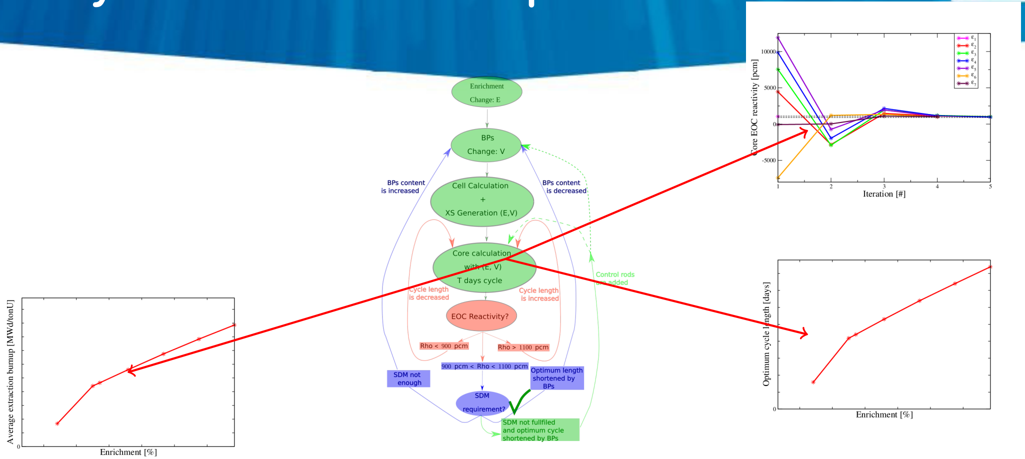
Fuel cycle neutronic optimization



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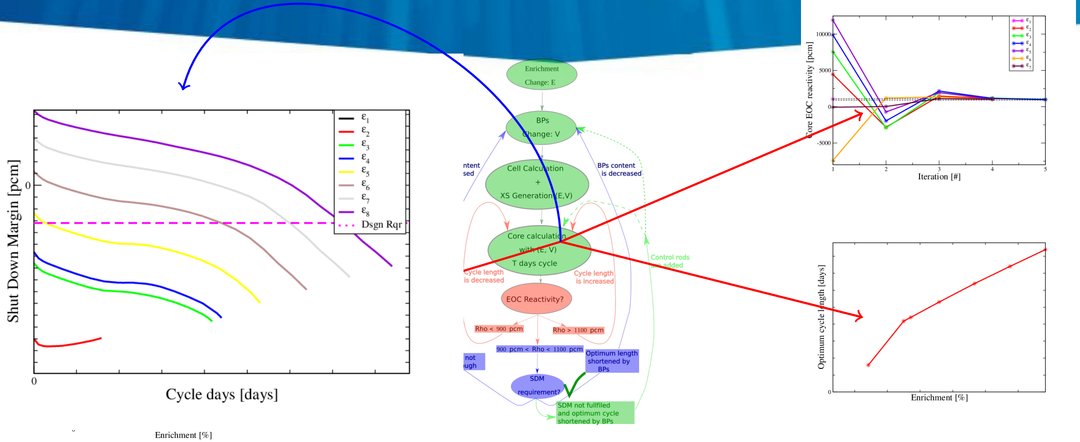
Fuel cycle neutronic optimization



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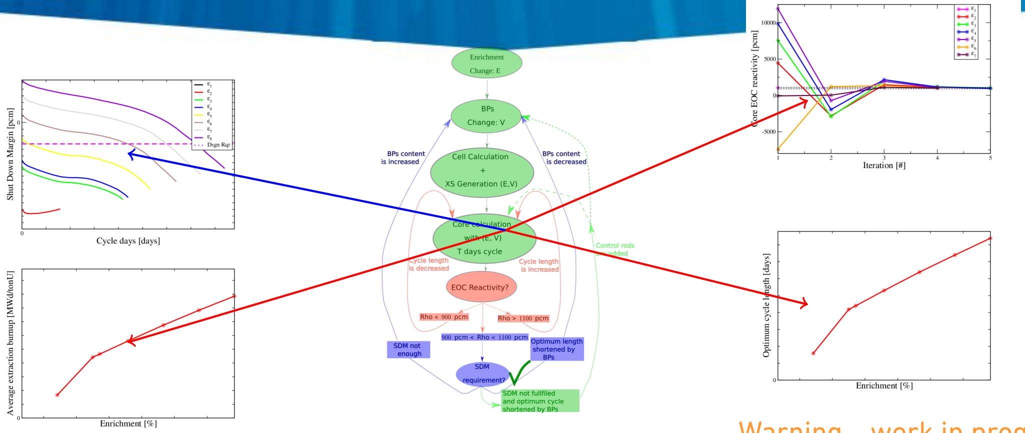

Warning... work in progress

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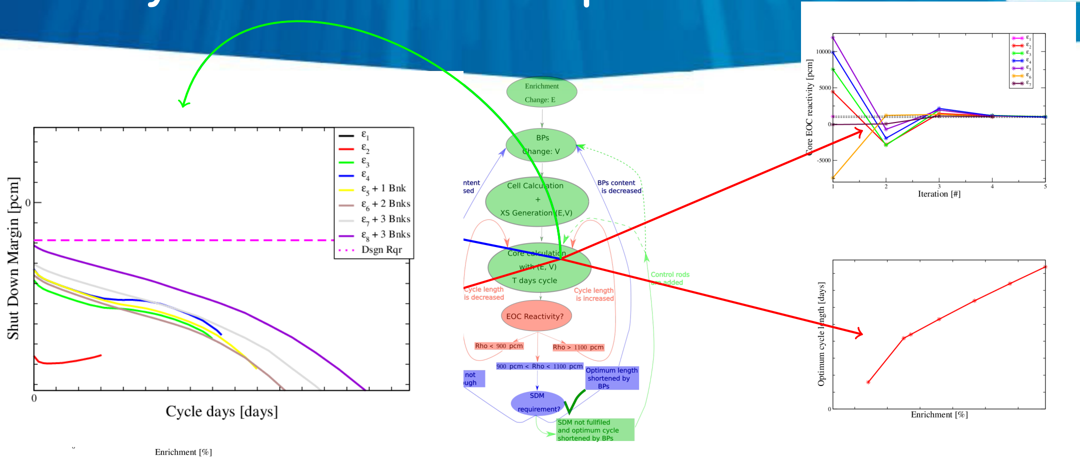
Fuel cycle neutronic optimization



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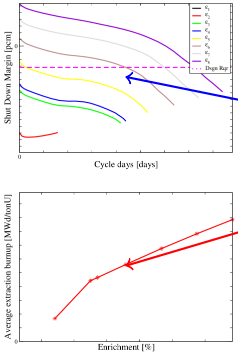
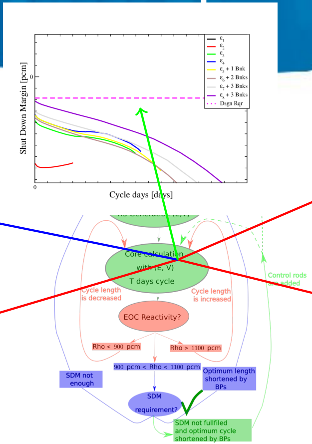
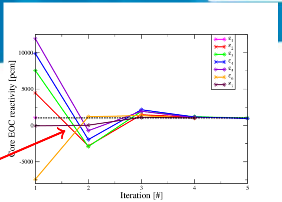
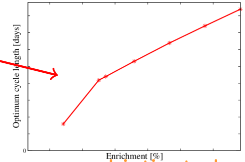
Fuel cycle neutronic optimization









A more sophisticated approach is also possible, and requires more computing time

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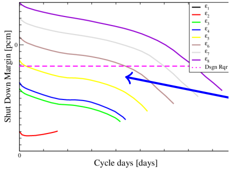
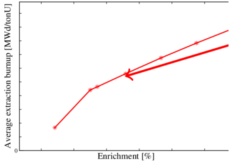
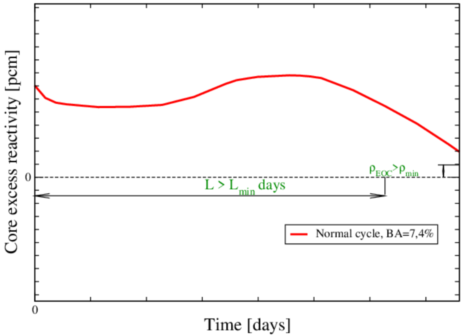
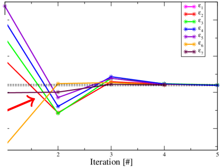
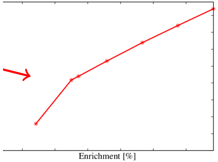
Fuel cycle neutronic optimization











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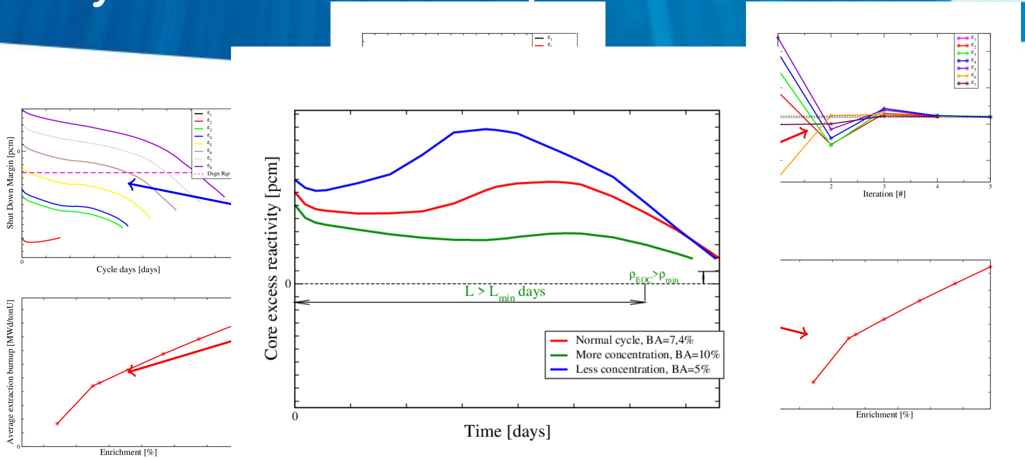
Fuel cycle neutronic optimization



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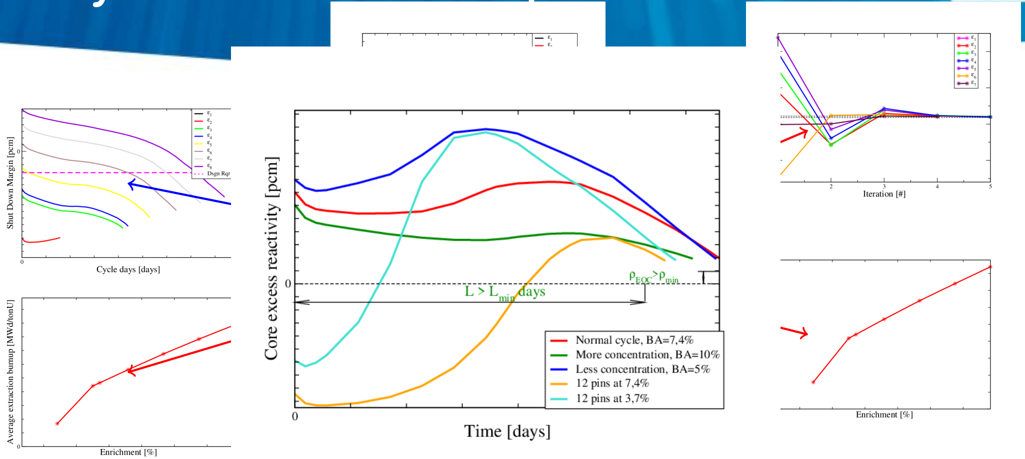
Fuel cycle neutronic optimization



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Preliminary Results

¿How is CAREM core economy influenced by fuel cycle changes?

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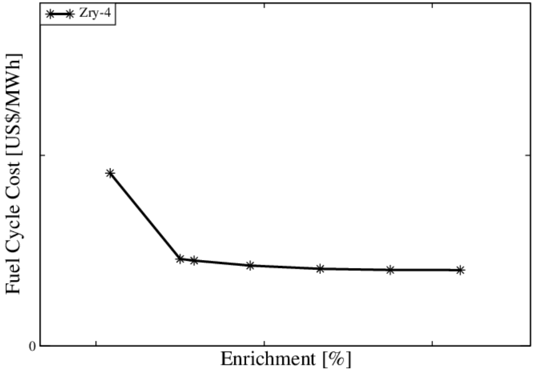
Preliminary Results



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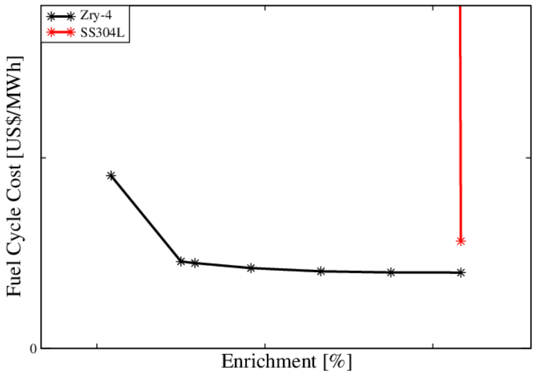
Preliminary Results



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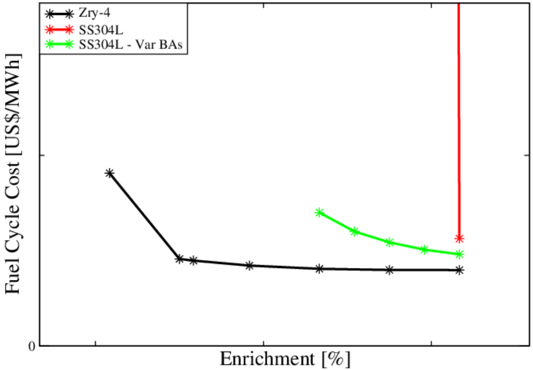
Preliminary Results



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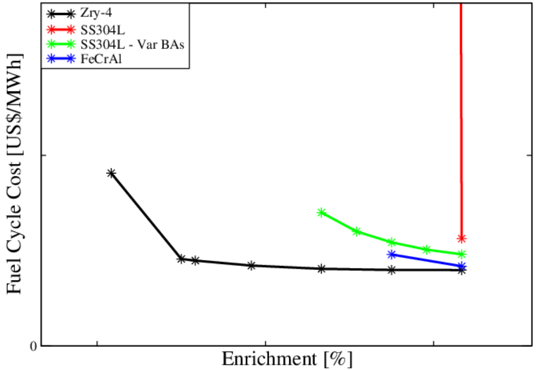

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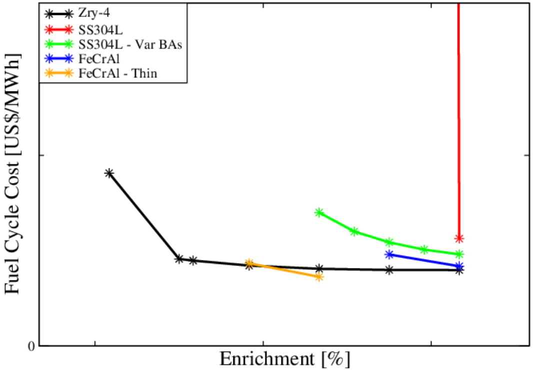
Preliminary Results



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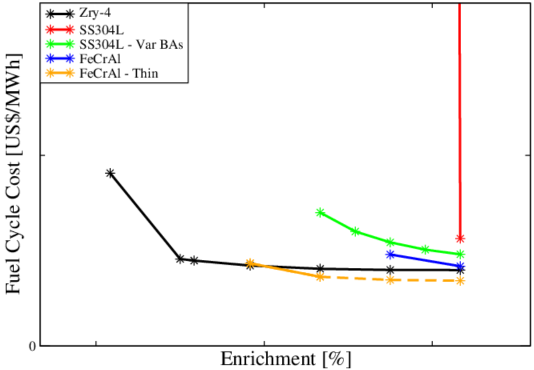

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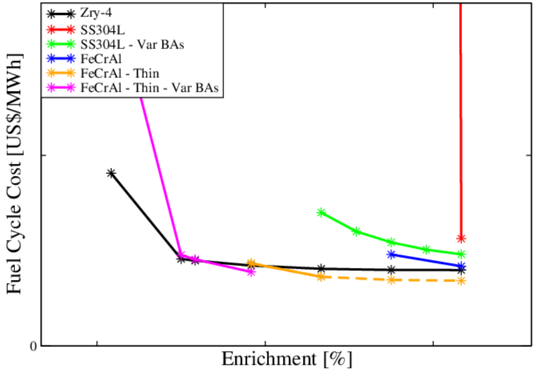
Preliminary Results



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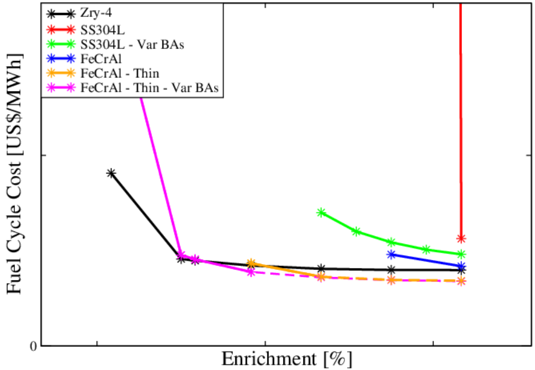
Preliminary Results



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Conclusions

¿What could be concluded so far?

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Conclusions

- A fuel cycle analysis implies the study of design requirements, something funny, not only interesting
- The proposed optimization method complied with the objective, but further enhancements could be done analyzing the distribution of BPs
- For Zry fuels, an increase in uranium enrichment seems economically feasible
- The SS fuels analyzed were not able to achieve more economical fuel cycles than that for Zry at today's prices
- The FeCrAl fuels analyzed are very competitive due to lower absorptions than SS.
- The SS and FeCrAl results could be further refined with a smarter use of BPs
- Power distribution still needs some improvements for configurations with higher enrichment
- There is still a long way ahead with promising fuel cycle possibilities

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Thanks
for your attention...

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Kraken multi-physics framework for SMR analyses

Introduction to multi-physics code coupling in the Kraken framework

Ville Valtavirta, VTT Technical Research Centre of Finland Ltd

23.3.2022

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Outline

- Kraken on one slide

- Motivation for Kraken development.
- Overarching development goals.
- Coupling scheme in Kraken.
- Development history and future.

- Licensing and distribution.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Kraken on one slide

VTT is replacing its legacy reactor analysis toolchains (HEXTRAN, TRAB-3D) with a new set, **Kraken**, building largely on VTT's own modern solvers.

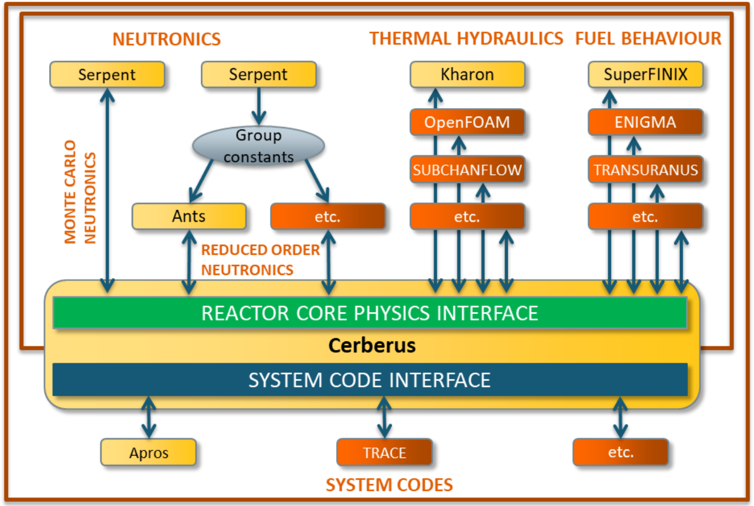
Kraken will provide VTT with the **tools** required for future safety analyses and the **expertise** to use those tools in a proper manner.

Kraken is designed both for **independent determinist safety analyses** and **evaluation of new reactor concepts**.

Basic capabilities for **steady state**, **fuel cycle** and **transient** analyses implemented during 2019-2021.

Validation effort ongoing with focus on demonstrating capabilities required for deterministic safety analyses.

A non-commercial user license is being drafted with international distribution planned through OECD/NEA data bank and RSICC later in 2022.



A schematic representation of the plans for the completed Kraken framework. Finnish solver modules developed at VTT are shown in yellow, while potential state-of-the-art third party solvers to be coupled are shown in orange.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Background and development goals

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Motivation for Kraken development

Historically the independent deterministic safety analyses for Finnish reactors have been conducted by Finnish organizations using Finnish reactor analysis tools (HEXBU, HEXTRAN, TRAB, SMABRE, Apros...).

Many of the tools previously in use at VTT were originally developed in the 80's and 90's educating a whole generation of experts into the field of reactor analysis.

The aging of both tools and expertise leads to challenges.

New reactor types (e.g. small modular reactors) are expected to enter the market.

More recently, Serpent development and user community have given some valuable lessons on:

- Active code development providing source code knowledge and motivation for young scientists.
- International user community tackling wider research topics and providing valuable feedback and contributions.
- Active participation in international research projects

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Motivation for Kraken development

Furthermore, much recent work in Serpent development on:

- Generating group constants for nodal solvers. For which nodal code? Ants development started in 2017.
- Coupled multi-physics calculations. With which solvers? Couplings made e.g. in McSAFE project.

Aging legacy codes and experts + new applications (SMRs, HTGRs, …) + good experience from Serpent development:

Start a project to build Kraken, VTT‘s reactor analysis framework for future research work and safety analyses.

Provide a non-commercial license for the framework early on so that interested scientists around the world can get involved.

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Development goals for Kraken

Kraken aims to be

Capable: Can evaluate fulfilment of design bases according to Finnish YVL-guides and NUREG-0800.

Usable: Offers a reasonable user interface while automating the routine parts of analyses.

Modular: Allows cross-verification of single physics solvers even in coupled transients.

Alive: Maintains source-code level expertise of the different parts of the framework.

Excellent: Uses state-of-the-art and beyond-the-state-of-the-art approaches whenever possible.

Kraken is intened both as a

Safety analysis tool, able to conduct Finnish deterministic safety analyses in the future, and

A **research and design tool** that can be applied to reactor core related research problems including the design of new reactor concepts.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Code coupling approach in Kraken

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Couplings for coupled calculations

Codes can be coupled in many ways:

- Memory level coupling:

Codes compiled together or linked as libraries. Executed as a single process.

- MPI or socket based coupling:

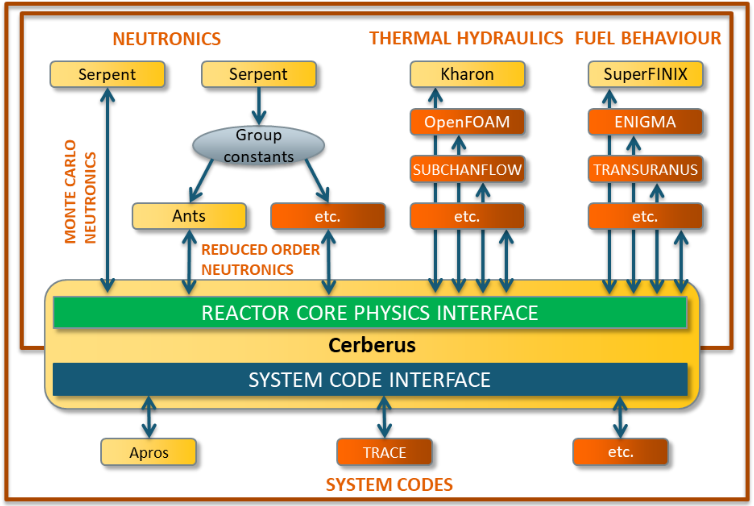
Data exchanged between separate processes using common protocols.

- Input/file based coupling:

External driver updates code inputs based on the outputs of others.

Kraken is intended to bring together solvers from various sources:

- Those developed at VTT specifically for Kraken.
- Those already developed at VTT, not specifically for Kraken.
- Externally developed codes, for which source code may be available.
- Externally developed codes, for which source code is not available.



A schematic representation of the plans for the completed Kraken framework. Finnish solver modules developed at VTT are shown in yellow, while potential state-of-the-art third party solvers to be coupled are shown in orange.

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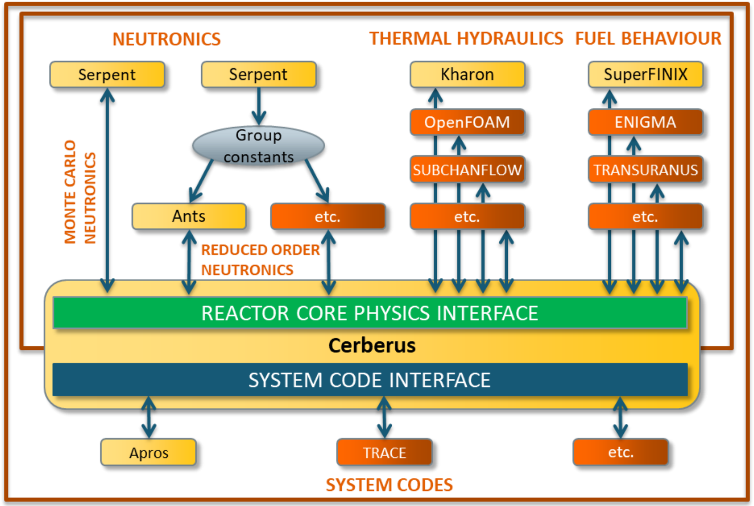
Couplings for coupled calculations

Code coupling in Kraken:

- A central multi-physics driver Cerberus.
    - Each solver only needs to communicate with Cerberus.
- Data transfer through sockets.
    - Native Kraken solvers support socket communication automatically.
    - Others utilize wrapper programs. SCFWrap, TUWrap, TRACEWrap.
- Code agnostic and modular coupling approach.
    - Cerberus does not know which solver is which.
    - All solvers look similar through Cerberus.
    - Can exchange solver modules to a higher or lower fidelity easily without changes to other solvers or simulation model as a whole.



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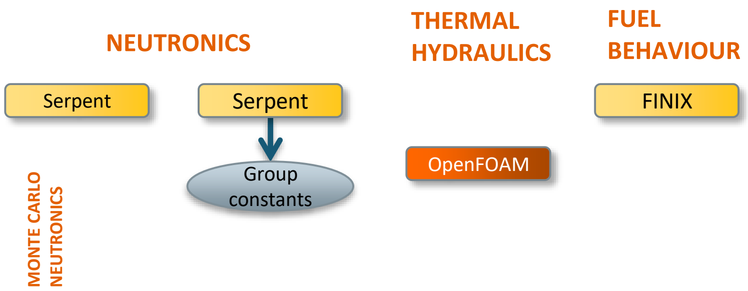
Development history and future plans

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Development history and future plans

Pre 2017:

- Serpent developed at VTT since 2004.
- Serpent 2.1.0 in 2012.
- Serpent 2 development focused on:
  - Group constant generation for reduced order solvers.
  - Coupled multi-physics calculations.
- FINIX fuel behaviour solver developed at VTT since 2012.
- VTT participates in OpenFOAM development.



Previous computational framework in use at VTT for deterministic safety analyses.

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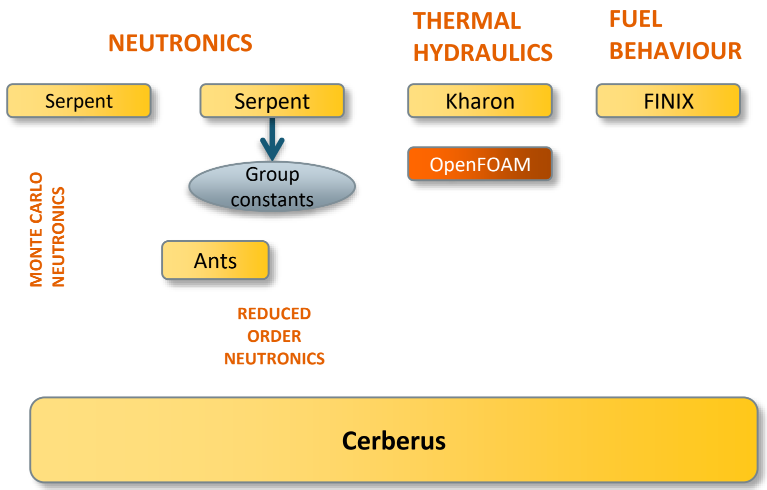
Development history and future plans

2017:

The idea of the renewal of VTT‘s reactor analysis framework and expertise proposed and accepted.

- Development of Ants nodal neutronics program starts.
- Development of the simple two phase closed channel porous medium TH solver Kharon starts.
- First commit in the Cerberus multi-physics driver repository.



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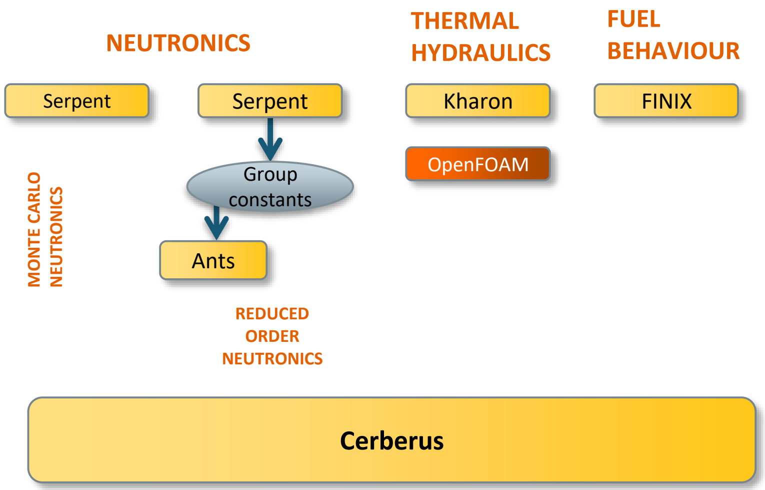
Development history and future plans

2018:

- Development work related to Ants.
- Kharon development.
- Drafting plans and project proposals related to large scale Kraken development.



Sahlberg, V., Rintala, A.
“Development and first results of a new rectangular nodal diffusion solver of Ants”
In Proc. PHYSOR 2018, Cancun, Mexico, April 22-26, 2018

Rintala, A., Sahlberg, V.
"Extension of Nodal Diffusion Solver of Ants to Hexagonal Geometry"
In Proc. 28th Symposium of AER on VVER Reactor Physics and Reactor Safety, October 8 – 12, 2018, Olomouc, Czech Republic.

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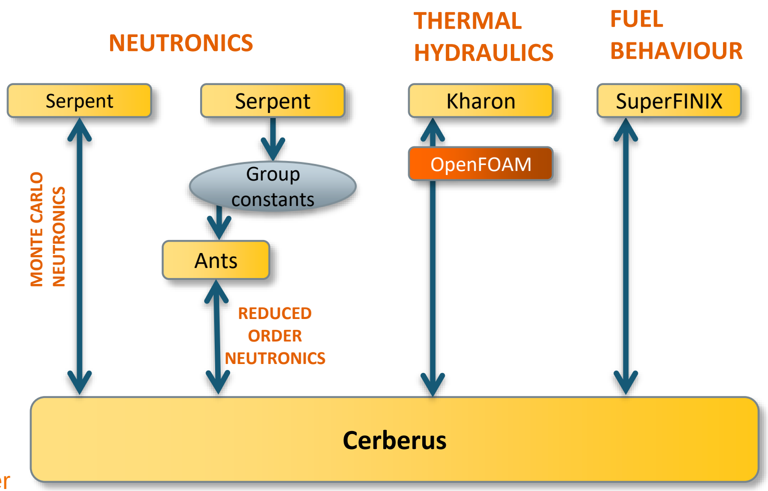
Development history and future plans

2019:

LONKERO 2019-2022 project starts.

- SuperFINIX core level fuel behaviour solver created.
- Socket communication syntax established for Cerberus.
- Couplings to several solver modules.
- Coupled steady state calculations (reactivity coefficients etc.)



Valtavirta, V., Peltonen, J., Lauranto, U., Leppänen, J. "SuperFINIX – A Flexible-Fidelity Core Level Fuel Behavior Solver for Multi-Physics Applications" NENE 2019, September 9-12, 2019, Portoroz, Slovenia

Rintala, A., Sahlberg, V. "Pin Power Reconstruction Method for Rectangular Geometry in Nodal Neutronics Program Ants" 28th International Conference Nuclear Energy for New Europe, September 9-12, 2019, Portoroz, Slovenia

Rintala, A., Sahlberg, V. "Extension of Nodal Diffusion Solver of Ants to Hexagonal Geometry" Kerntechnik 84 (2019)

Valtavirta, V., Hovi, V., Loukusa, H., Rintala, A., Sahlberg, V., Tuominen, R., Leppänen, J., "Kraken – an upcoming Finnish reactor analysis framework" M&C 2019, August 25-29, 2019, Portland, OR, USA.

Lauranto, U., Valtavirta, V., Rintala, A., Leppänen, J. Evaluating the fulfilment of control rod related nuclear design bases for an SMR core using the Kraken computational framework, symposium article at Nuclear Science and Technology Symposium – SYP2019, Helsinki, Finland, 30-31 October 2019

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

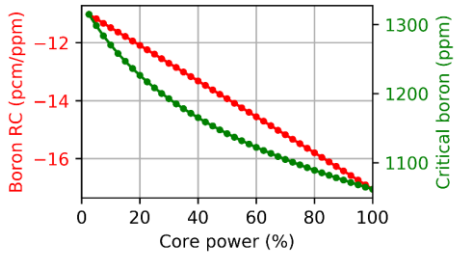
Some work conducted in 2019

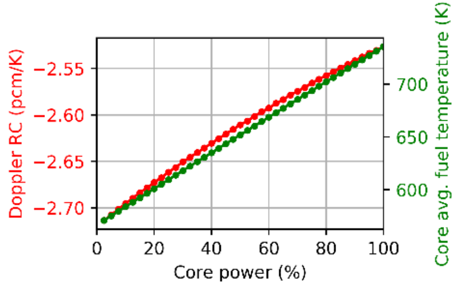
- Establishing the basic solver modules.
- Establishing a coupling between the solvers.

- Demonstrating the **capability** and **modularity** at steady state:

	CZP			HZP			HFP		
	Ants	**Serpent**	**A–S**	**Ants**	**Serpent**	**A–S**	**Ants**	**Serpent**	**A–S**
RB1	861	874	–13	1974	2012	–38	2084	2221	–137
RB2	2094	2092	+2	2218	2161	–57	2290	2285	+5
SB3	2592	2597	–5	3547	3559	–12	3612	3697	–85
SB4	2592	2596	–4	3547	3560	–13	3612	3703	–91

Using high-fidelity solver to verify reduced order solver performance also in coupled calculations: Control rod group worths in an SMR core evaluated by Ants and Serpent based coupled calculation sequences in cold-zero-power (CZP), hot-zero-power (HZP) and hot-full-power (HFP) conditions.





Evaluating licensing relevant data:
Two reactivity coefficients calculated with Ants-Kharon-SuperFINIX for the SMR core at various power levels:
Top: Boron reactivity coefficient (red) and critical boron (green).
Bottom: Doppler reactivity coefficient (red) and core average fuel temperature (green).

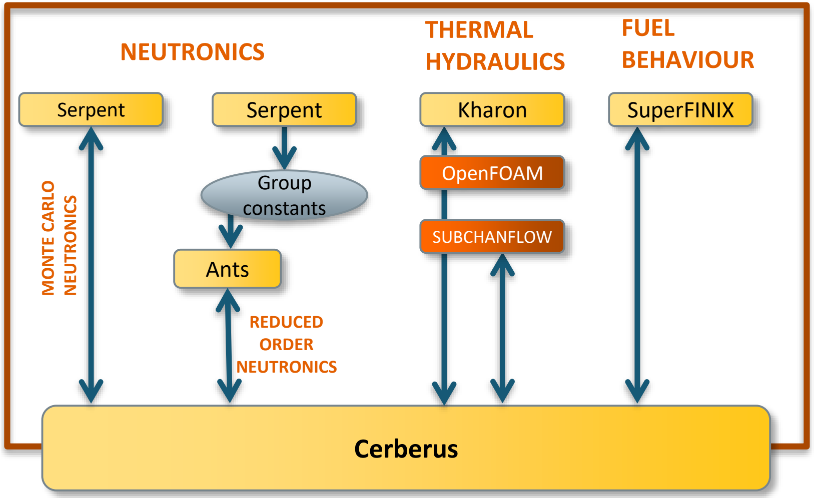
This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Development history and future plans

2020:

Moving from steady state to operating cycle analyses.

- Burnup capabilities implemented.
- A separate ReactorSimulator Python module automates fuel cycle analyses.
- Coupling SUBCHANFLOW through SCFWrap.



Unna Lauranto
Developing a generic Python based group constant library generator module for Serpent
Special Assignment Report, PHYS-E0441, Aalto University, School of Science, Department of Applied Physics, 2020.

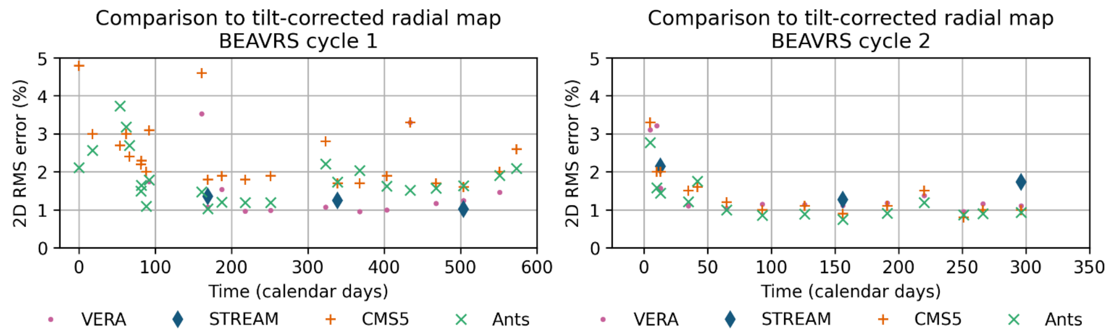
Valtavirta, V., Lauranto, U., Hovi, V., Peltonen, J., Rintala, A., Tuominen, R., Leppänen, J.
“High fidelity and reduced order solutions to an SMR-level progression problem with the Kraken computational framework”
PHYSOR 2020, March 29-April 2, 2020, Cambridge, UK.

Valtavirta, V., Rintala, A., Lauranto, U.
Validating the Serpent-Ants calculation chain using BEAVRS fresh core HZP data
29th International Conference Nuclear Energy for New Europe, September 7-10, 2020, Portoroz, Slovenia

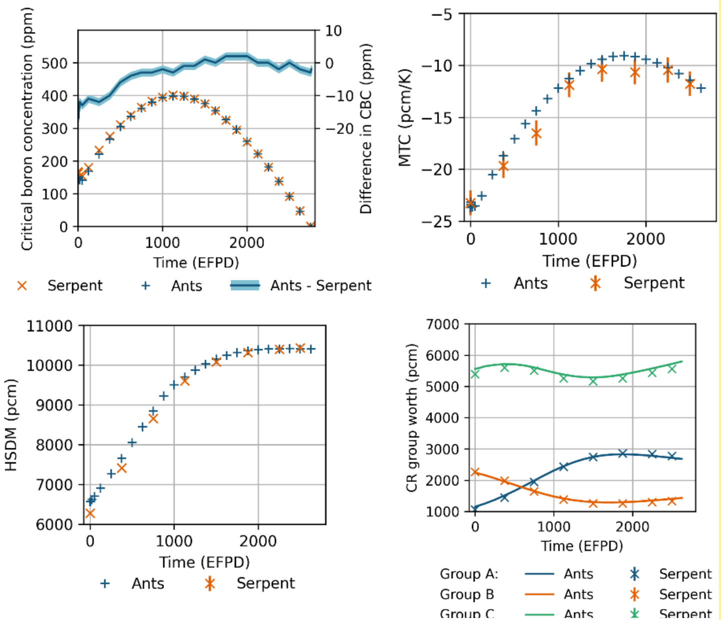
This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Some work conducted in 2020

- Extending the solvers and coupled solution to operating cycle analyses.
- Constructing a reactor core simulator for operating cycle analyses.
- Demonstrating the **capability**, **usability** and **modularity** of the core simulator in operating cycle analyses.
- Beginning the validation of the framework for operating cycle analyses.



Validation: 2D RMS errors when comparing calculated results to measured detector maps during the two operating cycles of the BEAVRS benchmark. Various industry and scientific leaders and Kraken (Ants).



Automatic evaluation of licensing relevant data during the simulation of an SMR operating cycle. Verification by switching one physics from reduced order solver (Ants) to a high-fidelity one (Serpent), while Kharon and SuperFINIX models are kept constant

Top left: Boron letdown curve.

Top right: Moderator temperature reactivity coefficient.

Bottom left: Instantaneous hot shutdown margin.

Bottom right: Control rod group worths.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Development history and future plans

2021:

From operating cycle analyses to transient calculations.

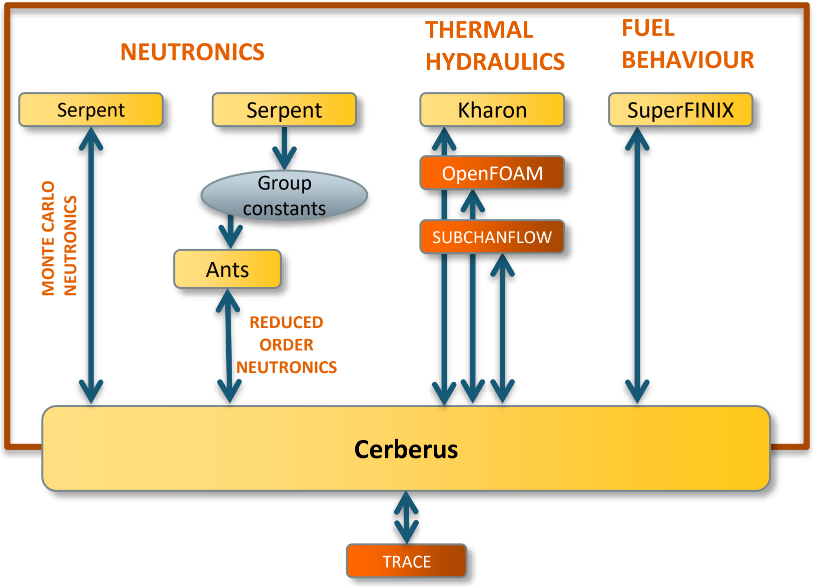
- Coupling to TRACE via TRACEWrap and ECI.
- Initial coupling to OpenFOAM.
- Begin drafting a non-commercial license.

Valtavirta, V., Tuominen, R.
A simple reactor core simulator based on VTT's Cerberus Python package
M&C 2021, April 11-15, 2021, Raleigh, NC

Hirvensalo, M., Rintala, A., Sahlberg, V.
Triangular geometry model for Ants nodal neutronics solver
M&C 2021, April 11-15, 2021, Raleigh, NC

Markus Hirvensalo
Runtime optimization of SuperFINIX multi-rod fuel performance program
M.Sc. Thesis, Department of Applied Physics, School of Science, Aalto University.

Rintala, A., Valtavirta, V., Leppänen, J.
Microscopic cross section calculation methodology in the Serpent 2 Monte Carlo code
Annals of Nuclear Energy 164 (2021)

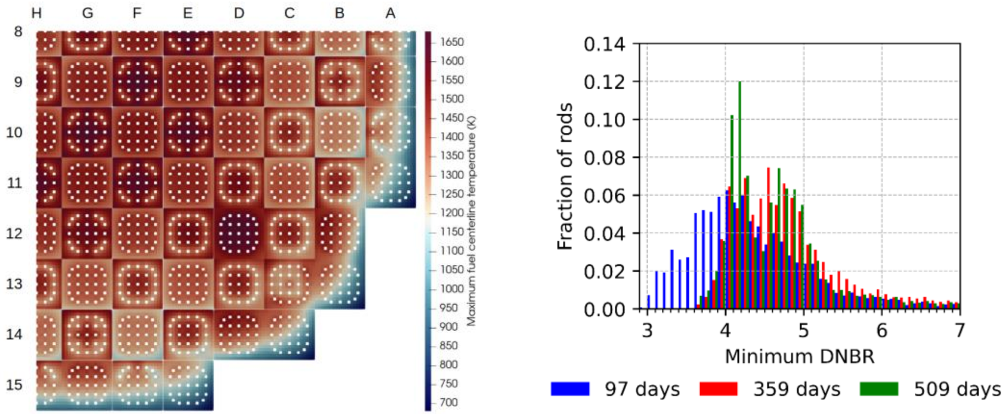


Valtavirta, V., Rintala, A., Lauranto, U. “Validating the Serpent-Ants Calculation Chain Using BEAVRS Fresh Core HZP Data”
Journal of Nuclear Engineering and Radiation Science (2021),

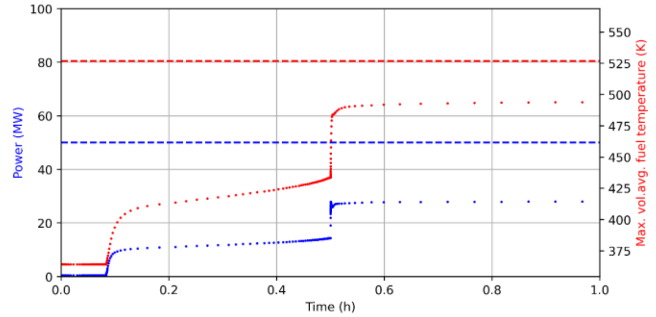
This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

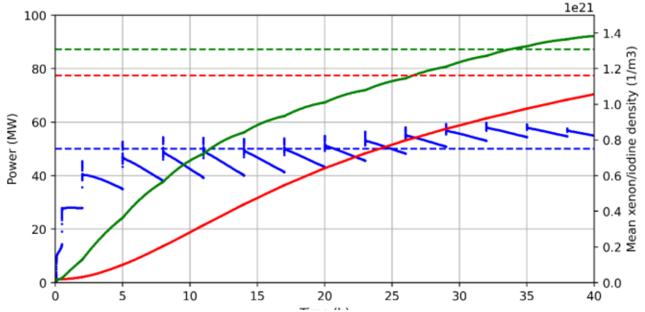
Some work conducted in 2021

- Extending the solvers and coupled solution to **time dependent simulations**.
- Verification of Ants neutron kinetics and dynamics.
- Verification of the Serpent-Ants chain in hexagonal lattice neutronics.
- Advanced methods for **in-line thermal margin evaluation**.



Rod resolved operating cycle analyses: PWR operating cycle modelled with Ants (pin power reconstruction) - SUBCHANFLOW (subchannel resolved) - SuperFINIX (rod resolved): **Left:** Maximum fuel centreline temperatures at 97 days. **Right:** Rod minimum DNBR distributions at 97, 359 and 509 days.





Starting an SMR from hot zero power to full power over several days. Modelled with Ants-SUBCHANFLOW.
Top row: Reactor power and maximum volume averaged fuel temperature during the first hour of the startup
Bottom: Reactor power and concentrations of ^{135}I and ^{135}Xe during the first 40 hours of the startup process. Dashed lines indicate hot full power steady state values.

Development history and future plans

2022 (ongoing):

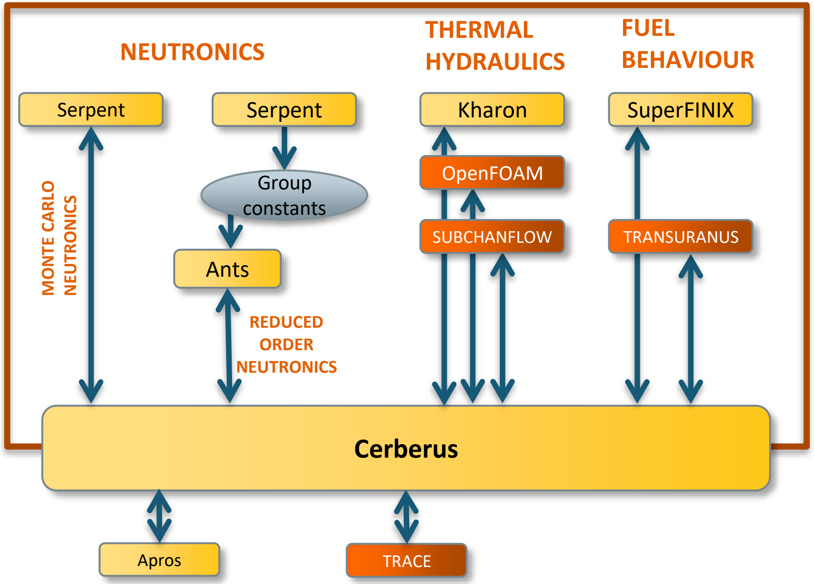
Various applications in McSAFER. Validation in LONKERO.

- Coupling to TRANSURANUS via TUWrap, coupling to Apros.
- Ants axial rehomogenization.
- First one-day Kraken training (McSAFER training course).
- First international Kraken workshop on ANS conference (PHYSOR2022)

Valtavirta, V., Lauranto, U., Rintala, A.
Evaluating the X2 initial core zero power physics tests with Serpent–Ants
PHYSOR 2022, May 15-20, 2022, Pittsburgh, PA

Leppänen, J., Valtavirta, V., Rintala, A., Hovi, V., Tuominen, R., Peltonen, J., Hirvensalo, M., Dorval, E., Lauranto, U., Komu, R.
Current Status and On-Going Development of VTT's Kraken Core Physics Computational Framework, Energies 15 (2022)

Unna Lauranto
Verification of Ants time-dependent nodal neutronics model
M.Sc. Thesis, Department of Applied Physics, School of Science, Aalto University.



Valtavirta, V., Rintala, A., Lauranto, U. "Pin power reconstruction for hexagonal geometry in nodal neutronics program Ants"
Submitted to Annals of Nuclear Energy

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Current status of Kraken

Capabilities for steady state, operating cycle and transient analyses.

Modular structure with several options for different solver modules.

Validation work for safety analyses ongoing (a large future topic).

Widely used in the core design of the Finnish district heating reactor concept LDR-50.

Applied in McSAFER to REA and MSLB analyses.

- REA:
    - Ants – SUBCHANFLOW.
    - Serpent – SUBCHANFLOW.
    - Serpent – SUBCHANFLOW – TRANSURANUS.
- MSLB:
    - Ants – TRACE.
    - Ants – TRACE – OpenFOAM.

J. Leppänen *et al.* “A Finnish district heating reactor: Background and general overview”. Proceedings of ICONE-28, August 4-6, 2021, Virtual Conference, USA.

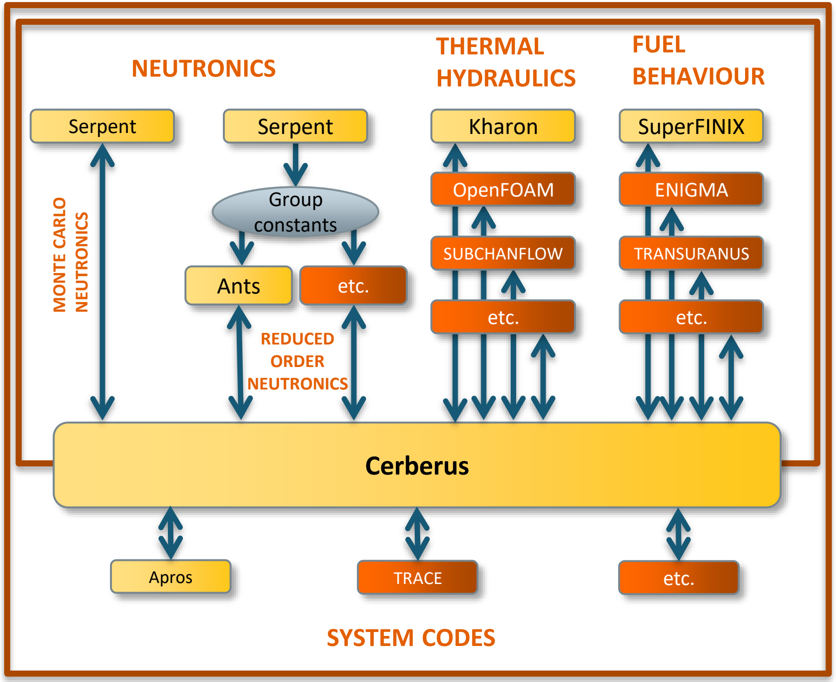
J. Leppänen *et al.* “A Finnish district heating reactor: Neutronics design and fuel cycle simulations”. Proceedings of ICONE-28, August 4-6, 2021, Virtual Conference, USA.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Development history and future plans

Future:

- Distribution for non-commercial use via OECD/NEA Data Bank and RSICC.
- Development of Ants nodal neutronics program continues:
  - Adjoint flux solver and related capabilities.
  - Improved group constant models.
  - From diffusion to transport?
- Validation for safety analyses:
  - International benchmarks.
  - Finnish NPP models.
- Improved capabilities for reactor design (LDR-50 development).
- Secondary analyses: Final disposal, radiation shielding, dosimetry, safeguards etc.



This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Licensing and distribution

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Licensing and distribution

Ongoing work for drafting a non-commercial license, establishing export control practices and starting distribution via OECD/NEA Data Bank and RSICC.

Practices similar to Serpent 2.2.0.

Initially covers:

- The **Ants** nodal neutronics code.
- The **FINIX** fuel performance code.
- The **SuperFINIX** core level fuel behaviour solver.
- The **Kharon** thermal hydraulics code.
- The **libFluid** fluid properties library.
- The **Cerberus** multi-physics driver package.
- The **KrakenTools** package of accessory modules.

Further modules most likely added on yearly basis (requires modifications to licenses and export control documents).

At this point, changes may still happen in communication syntax etc.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

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This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Kraken multi-physics framework for SMR analyses

Introduction to the physics solvers

Ville Valtavirta, VTT Technical Research Centre of Finland Ltd

23.3.2022

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Outline

- Neutronics solvers
    - Serpent
    - Ants
- Thermal hydraulics solvers
    - Kharon
    - OpenFOAM
    - SUBCHANFLOW
- Fuel behavior solvers
    - SuperFINIX
    - TRANSURANUS
- System codes:
    - Apros
    - TRACE

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Neutronics

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Serpent Monte Carlo code

Continuous energy Monte Carlo multi-purpose particle transport code developed at VTT since 2004.

Approximately 1000 users in over 250 organizations in 44 countries.

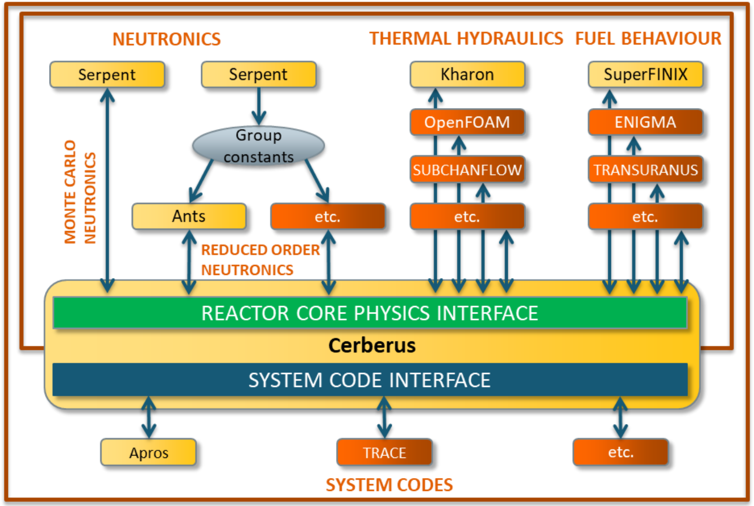
Initially designed for group constant generation.

Coupled multi-physics calculations a major development direction.

Flexible geometry. Neutron and photon transport.

Steady state, burnup and transient.

- Serpent is the one and only tool for group constant generation in the Kraken framework.
- The aim is to leverage the advanced capabilities of Serpent in the two step calculation chain.
- Direct Serpent reference solutions used to identify weaknesses in the two step calculation chain.



J. Leppänen *et al.* "The Serpent Monte Carlo code: Status, development and applications in 2013". Annals of Nuclear Energy 82 (2015), pp. 142–150.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Ants nodal neutronics program

Multi-group nodal neutronics code developed at VTT since 2017.

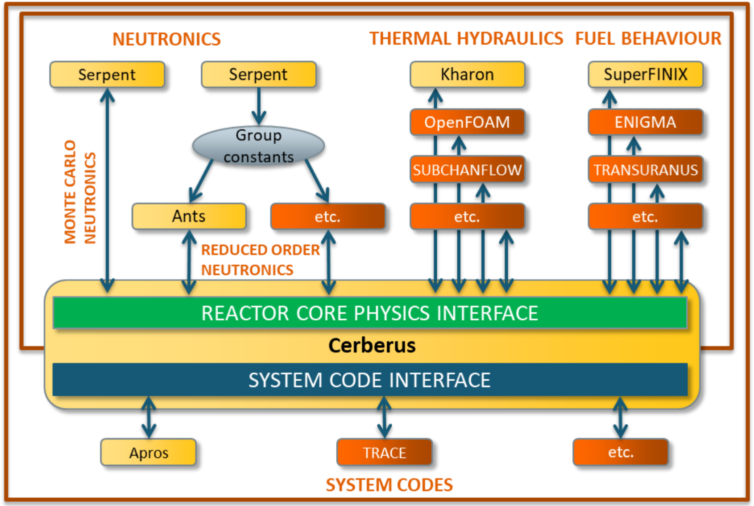
Currently uses nodal diffusion.

Combines AFEN and FENM approaches for flux solution.

Rectangular, hexagonal and triangular nodal models.

Steady state, burnup and transient.

Ants serves as the reduced order neutronics solver in Kraken providing solutions to stationary, depletion and transient neutronics problems in a reasonable time.



V. Sahlberg and A. Rintala. “Development and first results of a new rectangular nodal diffusion solver of Ants”. Proc. PHYSOR 2018. Cancun, Mexico, Apr. 2018

A. Rintala and V. Sahlberg. “Extension of nodal diffusion solver of Ants to hexagonal geometry”. Kerntechnik 84 (2019), pp. 252–261.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Thermal hydraulics

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Kharon thermal hydraulics solver

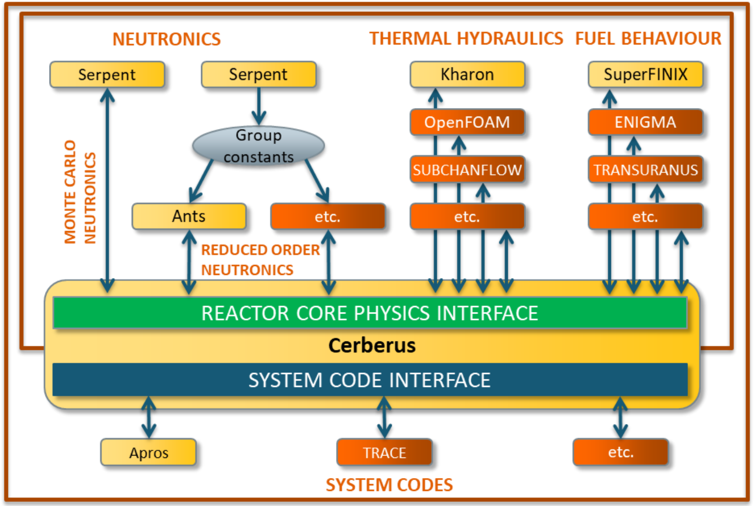
A simple core level thermal hydraulics solver developed at VTT:

- Two phase.
- Time-independent.
- Closed channel.
- Porous medium.

Models flow based on channel inlet and outlet boundary conditions and basic geometry.

Also models heat transfer from fuel rod cladding to coolant providing boundary condition for fuel behaviour codes.

Utilized in stationary and fuel cycle simulations. Transient simulations need another tool.



This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

OpenFOAM computational fluid dynamics code

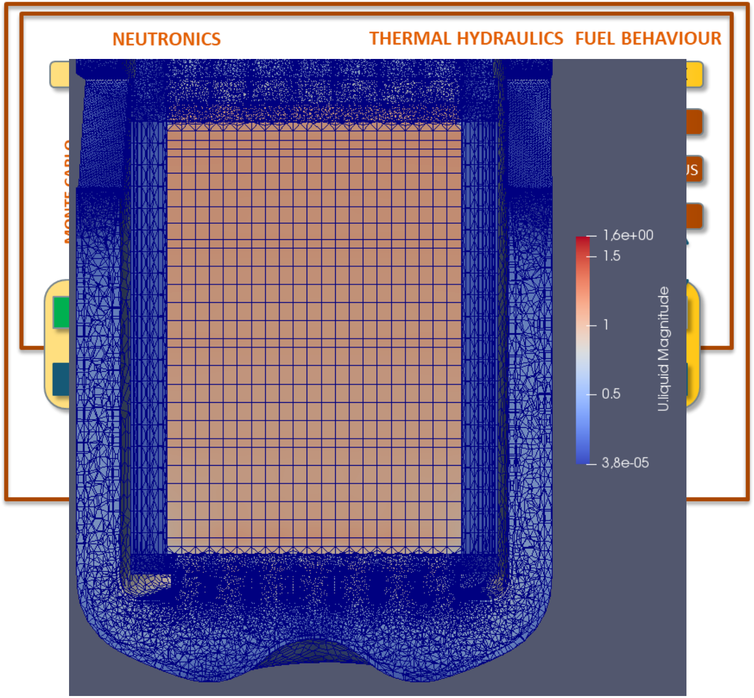
Open source software for CFD modelling and other purposes: https://openfoam.org/

VTT is a contributor in the project.

Past history in coupling Serpent and OpenFOAM in many ways.

Mostly using the multiphase porous medium solver in Kraken applications.

- Stationary and transient coarse mesh solutions inside the reactor core.
- Mixing, natural circulation etc. inside the reactor pressure vessel.



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SUBCHANFLOW thermal hydraulics solver

Subchannel level TH-solver developed by KIT.

https://www.inr.kit.edu/english/1008.php

Long history in coupling Serpent and SUBCHANFLOW using various approaches.

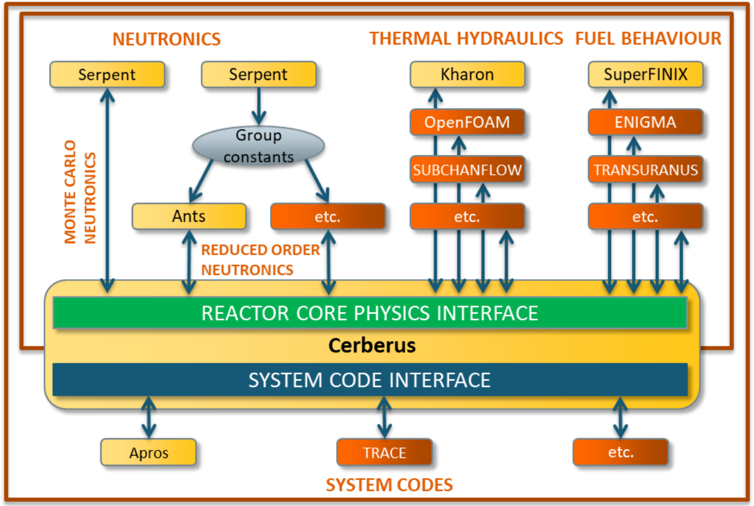
Most recently in the McSAFE project:

- ICoCo coupling.
- Master-slave coupling: sss-scf-tu and sss-scf.

Kraken coupling utilizes the pre-existing C API and a Kraken-specific wrapper layer (SCFWrap) to handle communications to/from Cerberus.

Applied in stationary, depletion and transient analyses.

Python preprocessor created in McSAFE utilized in generation of calculation mesh and interpolations.



This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Fuel behaviour

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

FINIX fuel behaviour module

The FINIX fuel behaviour module has been developed at VTT since 2012.

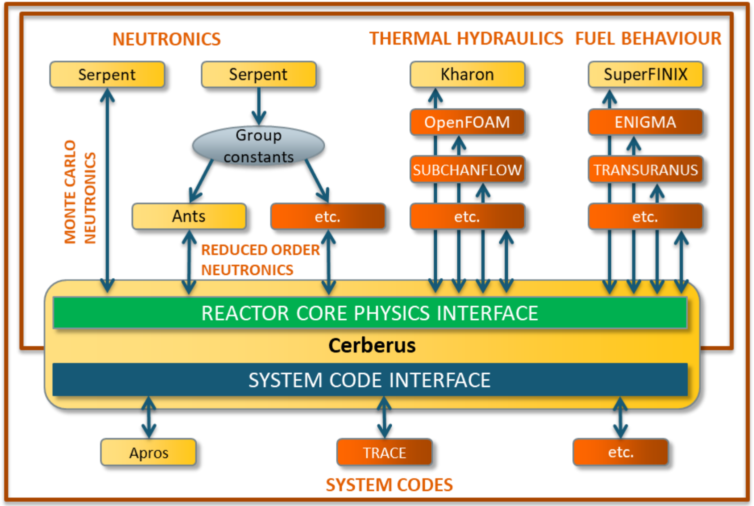
FINIX is a traditional 1.5 dimensional single rod fuel performance code.

Originally developed as a simple fuel behaviour solver module that could be coupled to reactor analysis codes at the source code level.

Developed for LWR applications.

Verified against FRAPTRAN and FRAPCON in RIA and steady state scenarios and compared against experimental Halden reactor data.

In the Kraken framework, FINIX is used through SuperFINIX, the core level fuel behaviour solver.



T. Ikonen *et al.* “Module for thermomechanical modeling of LWR fuel in multiphysics simulations”. Annals of Nuclear Energy 84 (2015), pp. 111–121.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

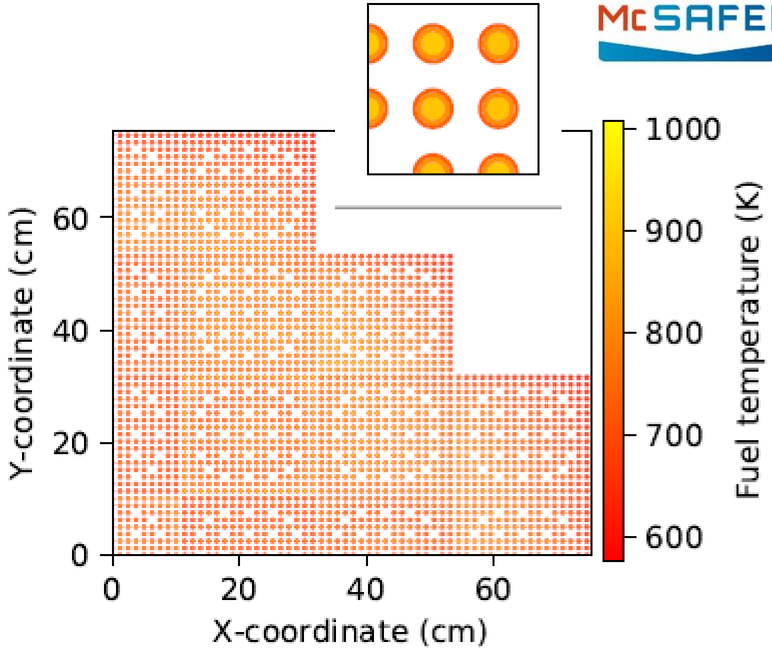
SuperFINIX core level fuel behaviour solver

The SuperFINIX core level fuel behaviour solver was written in 2019.

FINIX models a single fuel rod. LWR cores contain hundreds of fuel assemblies, tens of thousands of fuel rods.

Flexible fidelity for field input and output:

- Nodal codes, such as Ants require one fuel temperature value per node.
- Monte Carlo codes, such as Serpent can utilize individual rod radial distributions for fuel temperatures.
- Conversely power distribution may be evaluated at assembly, quarter assembly, rod or sub-rod level.
- SuperFINIX accepts input fields and provides output fields at multiple levels of discretization for the same model.



This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

TRANSURANUS fuel performance code

European fuel performance code developed by the JRC.

https://data.jrc.ec.europa.eu/collection/transuranus

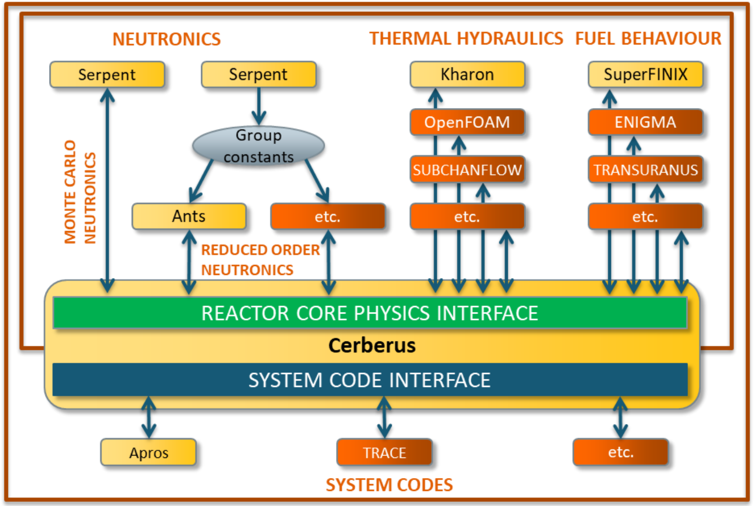
Coupled with Serpent in the McSAFE project:

- ICoCo coupling.
- Master-slave coupling: sss-scf-tu and sss-tu.

Single rod solver, but Kraken coupling utilizes pre-existing C and C++ layers and a Kraken-specific wrapper layer (TUWrap) to handle communications to/from Cerberus.

To be applied in stationary, depletion and transient analyses.

Python preprocessor created in McSAFE utilized in generation of calculation mesh and interpolations.



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System codes

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Apros system code

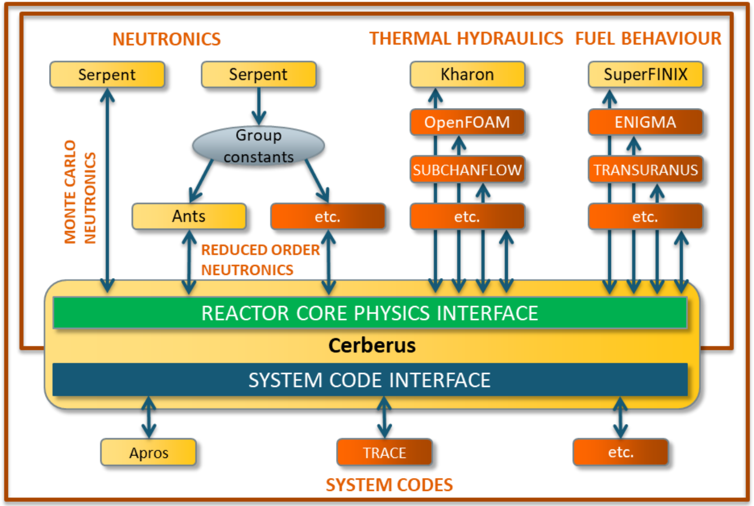
A system code / process simulator developed at VTT and Fortum for a long time.

https://www.apros.fi/

Used in the safety analyses of Finnish NPPs.

Also used in the development of VTT's LDR-50 district heating reactor concept.

Coupling work with Kraken ongoing.



E. Silvennoinen et al., "The APROS software for process simulation and model development", Technical Research Centre of Finland, Research reports 618 (1989).

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

TRACE system code

TRAC/RELAP Advanced Computational Engine (TRACE)

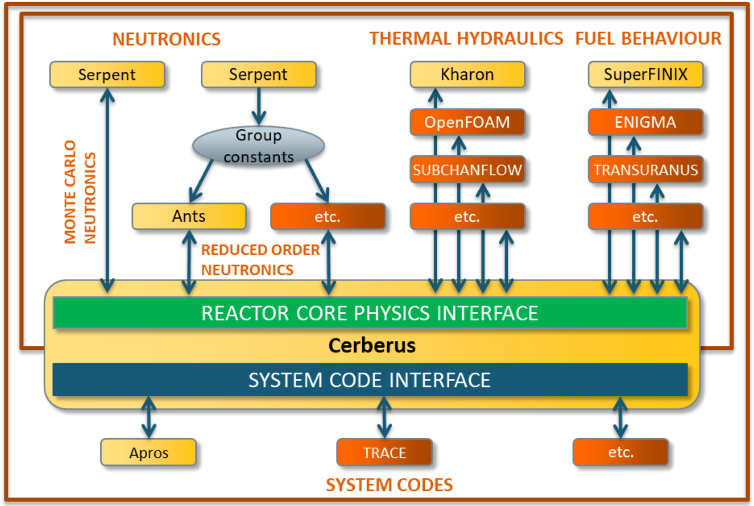
A system code developed by US NRC for LWR transient analyses.

Being adopted in Finland for independent deterministic safety analyses.

Finland participates in US NRC's Code Applications and Maintenance Program (CAMP).

Coupled to Kraken using a separate wrapper TRACEWrap, which communicates with TRACE using the Exterior Communications Interface (ECI).

Used as an independent verification tool for Apros analyses.



R. Tuominen, R. Komu and V. Valtavirta "Coupling TRACE with Nodal Neutronics Code Ants Using the Exterior Communications Interface and VTT's Multiphysics Driver Cerberus". PHYSOR 2022

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

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Kraken multi-physics framework for SMR analyses

Python level introduction to coupled calculations with Kraken using Cerberus

Ville Valtavirta, VTT Technical Research Centre of Finland Ltd

23.3.2022

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Outline

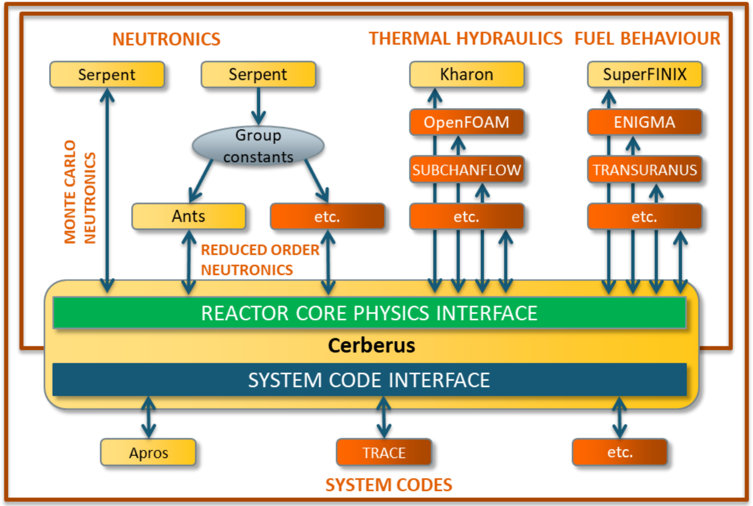
- Cerberus Python package
  - Idea
  - Capabilities
- Using Cerberus to set up Kraken calculations.
  - Boron iteration.
  - Control rod iteration.
  - Transient calculations.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Cerberus Python package

- Code agnostic multi-physics driver of the Kraken framework.
- Provides high-level API for solvers, fields and variables on Python side.
- Python makes building coupled calculation schemes simple and fun.

- Cerberus aims to hide most of the boring and technical stuff from the user.
- Strikes a balance between simplicity and flexibility.
- Aimed for expert users, who can package common calculation sequences into further Python packages/modules for non-expert users.



A schematic representation of the plans for the completed Kraken framework. Finnish solver modules developed at VTT are shown in yellow, while potential state-of-the-art third party solvers to be coupled are shown in orange.

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Cerberus Python package

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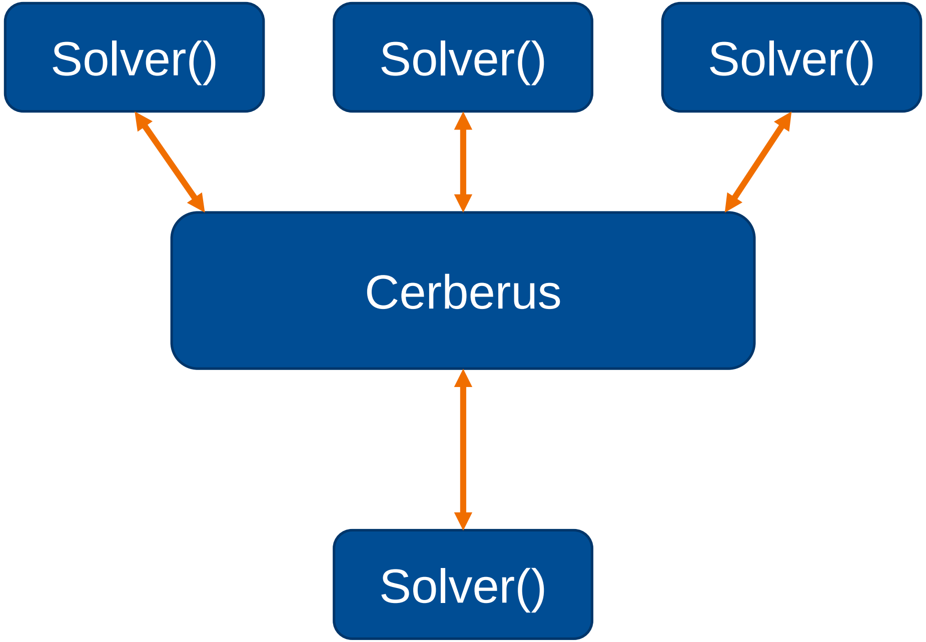


This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

The Solver class of Cerberus

All solvers participating in the calculation are based on the Solver class.

- Solver.initialize()
- Solver.get_transferrable()
- Solver.solve()
- Solver.set_current_time()
- Solver.suggest_next_time()
- Solver.move_to_time()
- Solver.write_restart()
- Solver.read_restart()
- Etc.

The user does not need to know what happens "under the hood" when calling one of these methods from Python.

All solvers provide the same functionality to Python even if actual implementation in the solver module may differ.

Cerberus does not know (or care) which solver handles neutronics and which thermal hydraulics.

The expert user, of course does care.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Transferrables in Cerberus

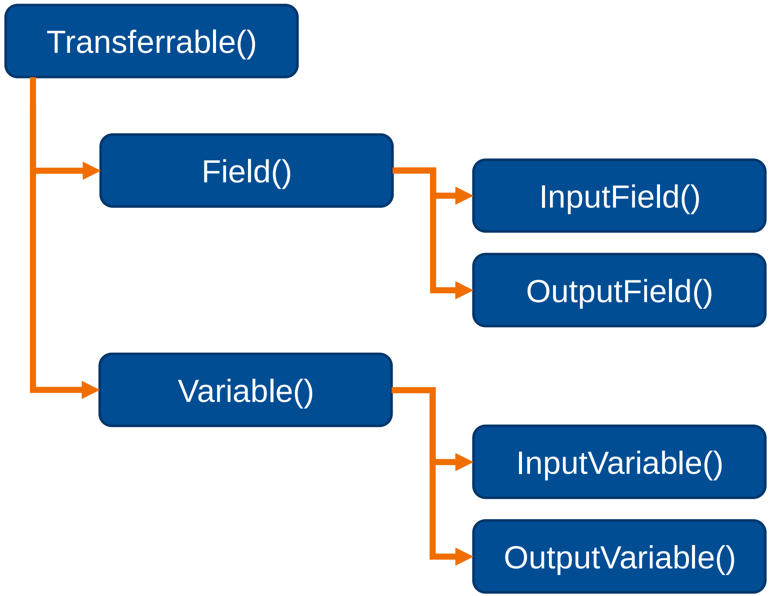
The transferrable superclass covers all input and output data needs of the solver that can be transferred between the solver and Cerberus.

- Transferrable.communicate(): Exchange data with solver.
- Transferrable.write_simple(): Write data to file.
- Transferrable.value_vec: Current values of data.
- Transferrable.get_conv_crit(): Evaluate convergence criterion between current and previous values.
- …

Field() and Variable() are sub-classes of Transferrable().

Field is a (physical) dataset with a spatial representation (mesh).

Variable is a more general set/piece of data.
Often single valued.



This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

On Field()s

Field data on Python side is in SI-units and uses a default (global) indexing for the mesh.

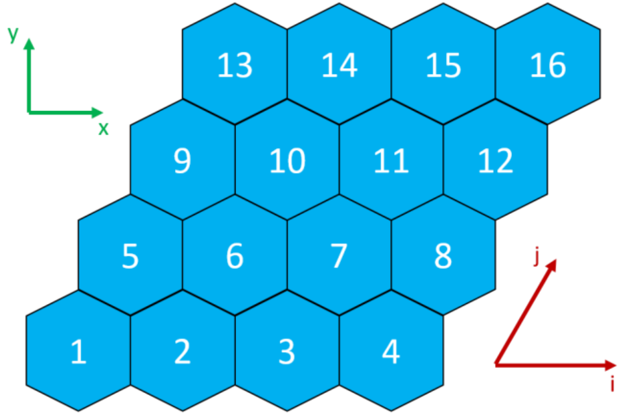
Conversions between solver (local) and Cerberus (global) units and indexings are handled automatically by Cerberus based on data Cerberus obtains from the solver.

Cerberus output can be written to files separately using the global and local indexings.

Meshes may be used for automatic generation of interpolations between fields in the future. At the moment, interpolations need to be pre-generated by the user.

Mesh information is also written in files and can be re-created in postprocessing using krakentools.kraken.Mesh class, which offers some useful functionalities for plotting etc.

Automated output to FoamFiles is partly supported with additional support added in the future.



Global indexing for mesh type 3: One axial layer of structured x-type 60 degree hexagonal mesh.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

On Field()s

Field data on Python side is in SI-units and uses a default (global) indexing for the mesh.

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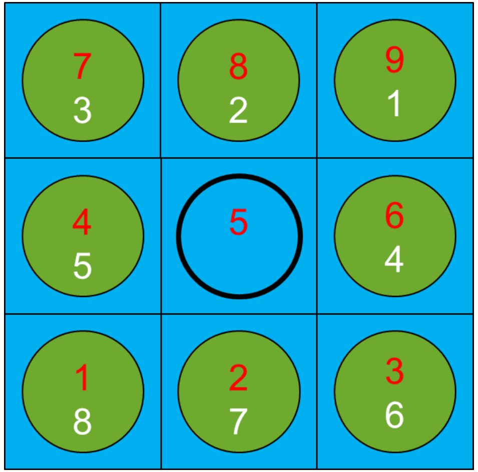


Cerberus will automatically handle conversions between solver indexing to global indexing to provide one uniform indexing scheme across all solvers and fields on the Python side.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

The Interpolator() class

Handles data transfer between two Field()s using a chosen interpolation scheme:

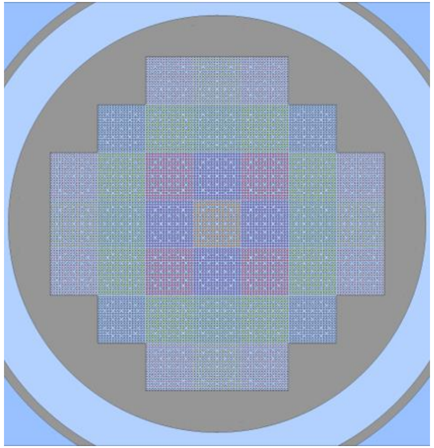
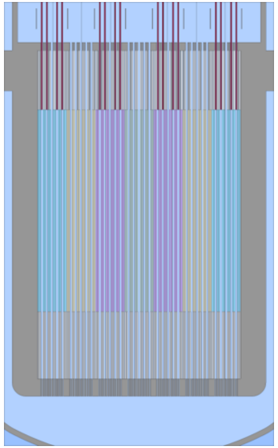
- One-to-one mapping.
- User supplied interpolation matrix:
  - $\bar{d} = \bar{\bar{A}}\bar{s}$

$$\begin{pmatrix} d_1 \\ \vdots \\ d_{N_d} \end{pmatrix} = \begin{pmatrix} a_{1\to 1} & \cdots & a_{N_s\to 1} \\ \vdots & \ddots & \vdots \\ a_{1\to N_d} & \cdots & a_{N_s\to N_d} \end{pmatrix} \begin{pmatrix} s_1 \\ \vdots \\ s_{N_s} \end{pmatrix}$$

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

A simple time-independent coupled sequence

1. Initialize solvers
2. Set initial conditions
3. Coupled iteration loop
    1. Solve first physics.
    2. Transfer results.
    3. Solve second physics.
    4. Transfer results.
    5. …
    6. Iterate if not converged.
4. Save final results

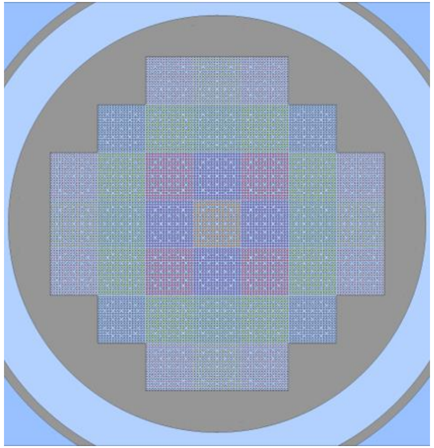



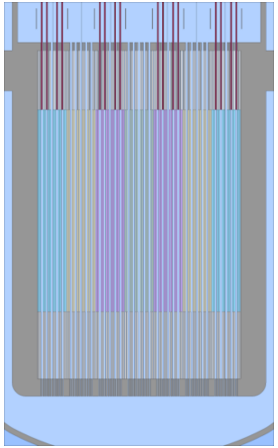
Converge the critical boron (CB), all rods out (ARO), hot full power (HFP) state of a 50 MW small modular reactor using Ants and SUBCHANFLOW.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

A simple critical control rod position search scheme

1. Initialize solvers
2. Set initial conditions
3. Control rod search loop
    1. Coupled iteration loop
        1. Solve first physics.
        2. Transfer results.
        3. Solve second physics.
        4. Transfer results.
        5. …
        6. Iterate if not converged.
    2. Check reactivity of reactor.
    3. Move control rods to next guess or break if critical.
4. Save final results



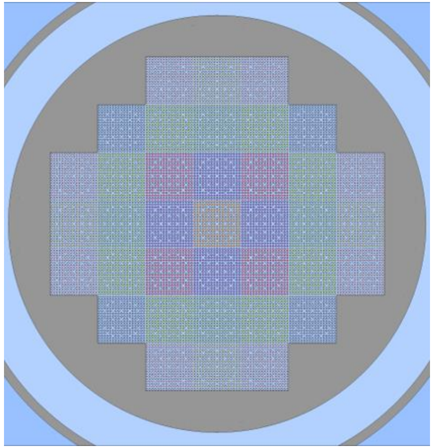


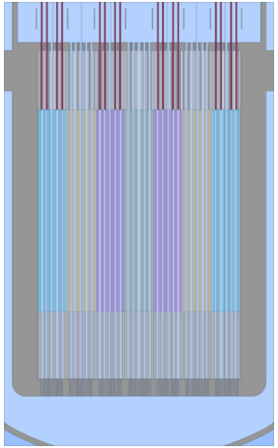
Using previously iterated boron concentration, converge critical control rod position at hot zero power (HZP) state of a 50 MW small modular reactor using Ants and SUBCHANFLOW.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

A simple time-dependent calculation scheme

1. Initialize solvers
2. Set initial conditions
3. Time integration loop
   1. Coupled iteration loop
      1. Solve first physics.
      2. Transfer results.
      3. Solve second physics.
      4. Transfer results.
      5. …
      6. Iterate if not converged.
   2. Process results from current time step.
   3. Move to next time step or end calculation.
4. Save final results



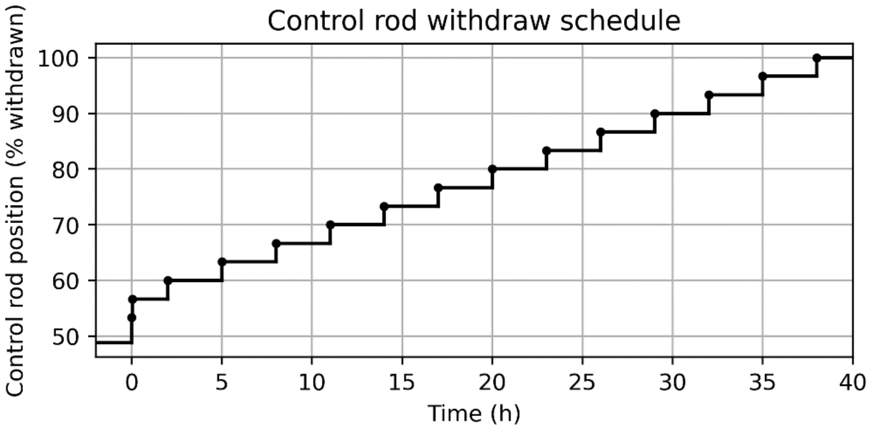


Using previously iterated boron concentration, and control rod position. Model a zero-transient starting from the HZP state of a 50 MW small modular reactor using Ants and SUBCHANFLOW.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

A more interesting time-dependent calculation scheme

1. Initialize solvers
2. Set initial conditions
3. Time integration loop
   1. Move control rods based on startup sequence.
   2. Coupled iteration loop
      1. Solve first physics.
      2. Transfer results.
      3. Solve second physics.
      4. Transfer results.
      5. …
      6. Iterate if not converged.
   3. Process results from current time step.
   4. Move to next time step or end calculation.
4. Save final results



Using previously iterated boron concentration, and control rod position. Model the startup (from HZP to HFP) of a 50 MW small modular reactor using Ants and SUBCHANFLOW.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Ville Valtavirta

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Kraken multi-physics framework for SMR analyses

NuScale input models

Ville Valtavirta, VTT Technical Research Centre of Finland Ltd

23.3.2022

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Outline

- Introduction to the rod ejection accident scenario.
- Serpent model.
  - Group constant generation.
  - Reference calculation model.
- Ants model.
- SUBCHANFLOW model.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Introduction to the REA scenario

Introduction to NuScale REA scenarios

- Core specifications detailed in McSAFER D3.1.
  - Exact loading is not public information.
  - Fresh fuel loading approximating equilibrium cycle.
  - Axial fuel blankets replaced with normal fuel.
  - Annular AIC part of control rods replaced with solid AIC.
- Rod ejection accident scenarios specified separately.
  - Initial control bank positions.
  - Initial reactor power.
  - Ejected control rod assembly.
  - Stuck control rod assembly.
  - Fuel rod properties for simplified modelling.

4.2.3.1. Loading pattern and control rod bank positions

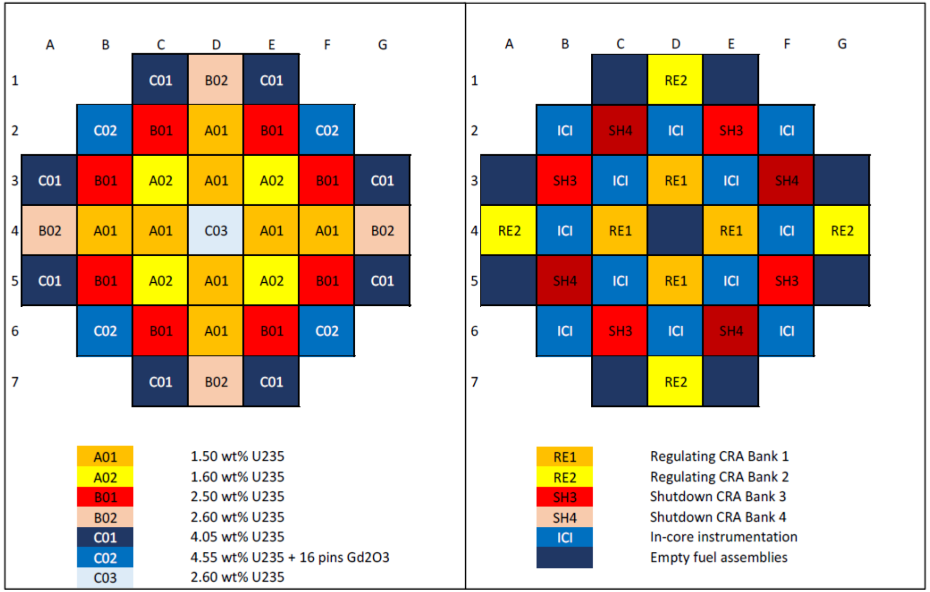

Figure 4.8 Core loading pattern and control rod assembly locations.

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Introduction to NuScale REA scenarios

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    - Initial reactor power.
    - Ejected control rod assembly.
    - Stuck control rod assembly.
    - Fuel rod properties for simplified modelling.

4.2.3.2. Radial reflector description

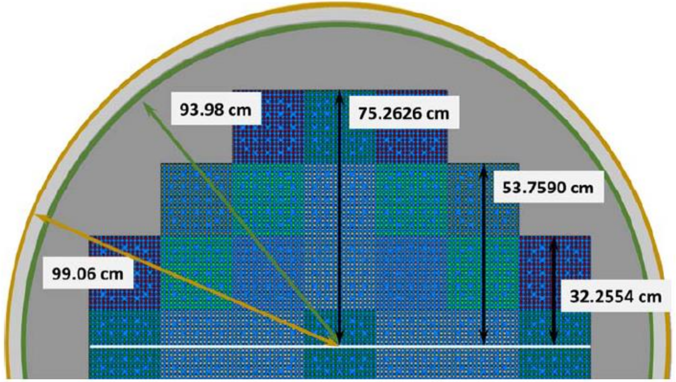

Figure 4.9 Geometrical information for radial reflector modelling. Radial reflector consists of the heavy reflector (dark grey) surrounding the active core and the cylindrical core barrel (light grey) surrounding the heavy reflector.

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Introduction to NuScale REA scenarios

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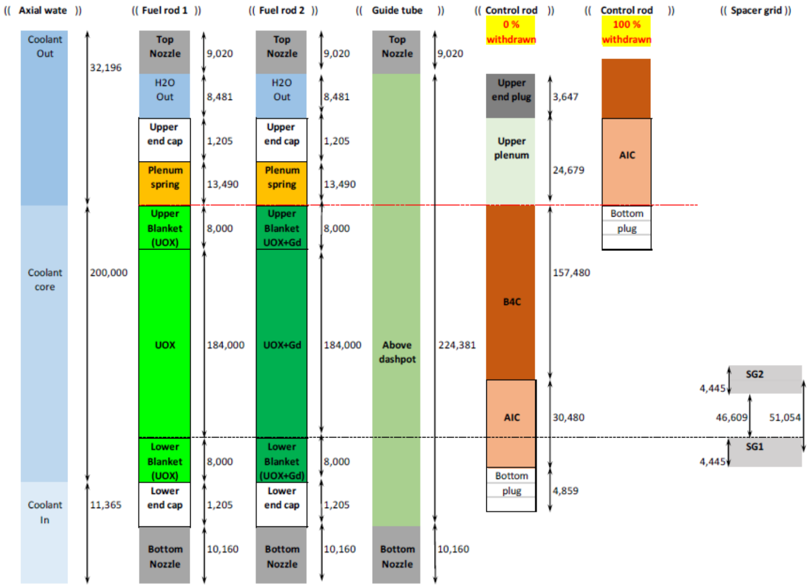
4.2.2.3. Axial structure



All units in cm			Fuel stack	200,000	Fuel stack	200,000						
			Fuel rod	215,900	Fuel rod	215,900	GT	224,381	CR active	187,960	CR active	0,000
Total		243,561	Total	243,561	Total	243,561	Total	243,561	Total	221,145	Total	0,000

Figure 4.7 Axial structure and axial alignment of the different components of the active core.

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Introduction to NuScale REA scenarios

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  - Fuel rod properties for simplified modelling.

4.2.3.1. Loading pattern and control rod bank positions

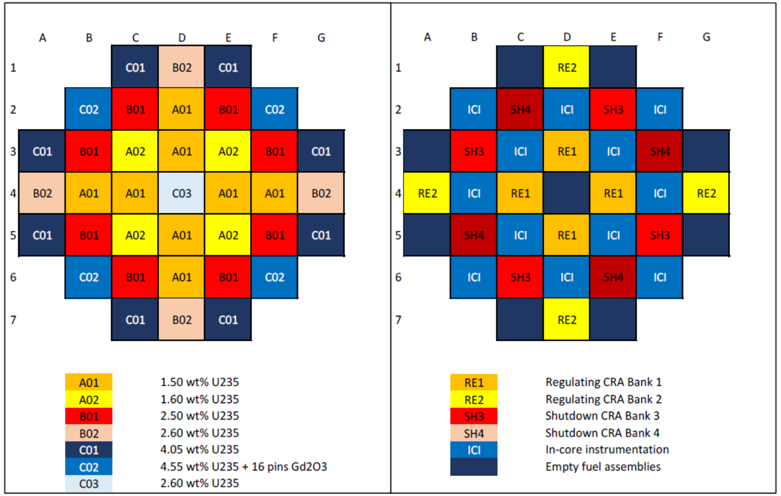

Figure 4.8 Core loading pattern and control rod assembly locations.

Table 16: Summary of the initial conditions for different REA scenarios

Power		Flow rate	T inlet	Operating pressure	Boron	CRA position			
						RE1	RE2	SH3	SH4
%	MWt	kg/s	°C	bar		% withdrawal			
75%	120	521.6	262	127.5	Critical	100	56	100	100

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Introduction to NuScale REA scenarios

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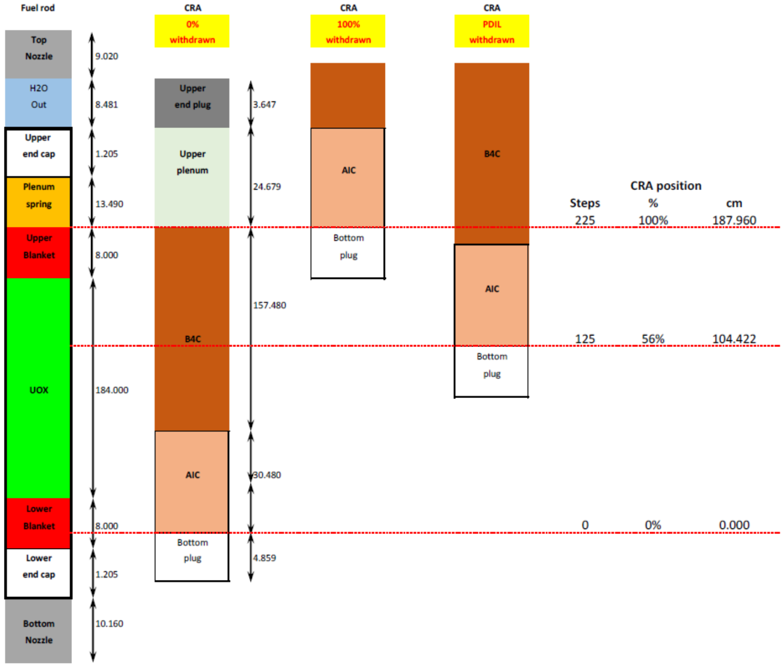

Figure 56. Clarification on CRA positioning.

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Introduction to NuScale REA scenarios

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  - Initial reactor power.
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5.1.5. Sequence of Events

- t≤0: the system is first made critical at the core state and power level specified for the transient scenario by adjusting CBC.
- 0.0≤ t ≤ 0.1 sec: CRA assembly is ejected from the core at a constant speed within 0.1 sec.
- 0.1≤ t ≤ 2.0 sec: Transient without any additional changes applied to the system.
- 2.0≤ t ≤ 3.0 sec: SCRAM at a constant speed within 1.0 sec. All CRAs except a) ejected rod; b) stuck rod (see Table 18)
- 3.0≤ t ≤ 4.0 sec: Transient without any additional changes applied to the system.

Table 18: Initial position of CRA banks, ejected and stuck CRAs

Initial position				Ejected CRA	Stuck CRA
% withdrawal					
RE1	RE2	SH3	SH4		
100	56	100	100	A4 (RE2)	B5 (SH4)

Table 17: Fixed thermal-physical properties of the fuel rod

Property	Units	Value
Fuel density	g/cm^3	10.5216
Fuel thermal conductivity	$W/(m \cdot K)$	3.5
Fuel specific heat	$J/(kg \cdot K)$	290
Clad density	g/cm^3	6.5500
Clad thermal conductivity	$W/(m \cdot K)$	13.0
Clad specific heat	$J/(kg \cdot K)$	300
Gap conductance	$W/(m^2 \cdot K)$	10000

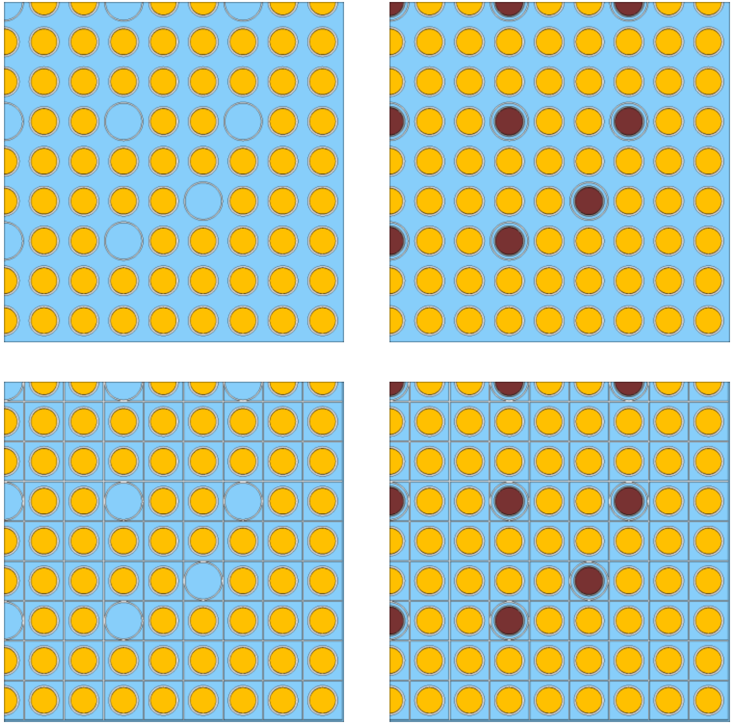
This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Group constant generation with Serpent

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Group constant generation with Serpent (HZDR)

- Fuel assembly group constants generated in ¼ assembly models.
- Group constants generated based on continuous energy Monte Carlo transport solution.
- Intermediate condesation of data to a 70-group structure.
- Final condensation to 4-group structure using:
  - Infinite spectrum **or**
  - Fundamental Mode (FM) leakage correction.
- Diffusion coefficients calculated using Cumulative Migration Method.
- Discontinuity factors calculated at assembly outer and inner boundaries.
- Serpent branch calculation capabilities utilized in running all thermal hydraulic state points with a single input.



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- Serpent branch calculation capabilities utilized in running all thermal hydraulic state points with a single input.

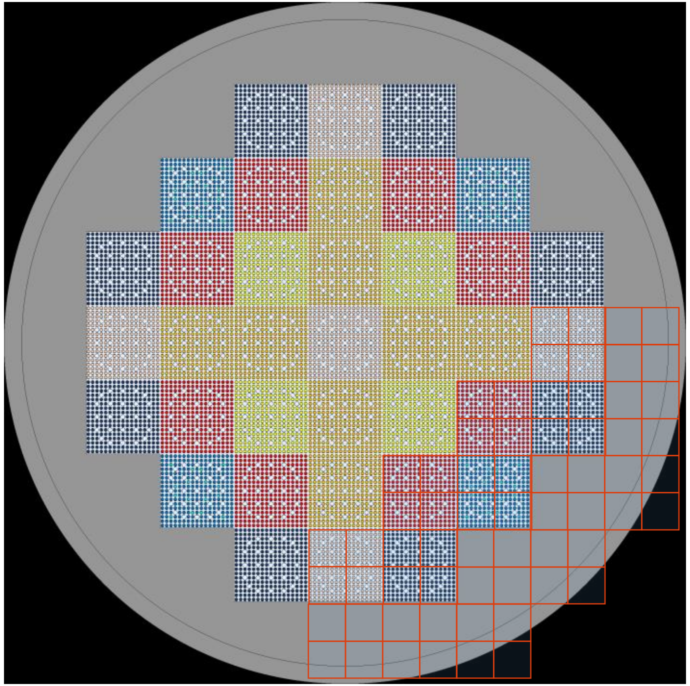
```
coef 1 0.0
6 ro09 ro08 ro07 ro06 ro05 ro03
3 cb0 cb1000 cb2000
3 tf500 tf900 tf1800
3 no_spa zirc inc
4 no_cr plug aic b4c
```

Using the coef card of Serpent based on branch definitions.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Group constant generation with Serpent (HZDR)

- Radial reflector was homogenized in a 2D full core calculation.
- Group constants generated based on continuous energy Monte Carlo transport solution.
- Intermediate condesation of data to a 70-group structure.
- Final condensation to a 4-group structure using infinite spectrum.
- Diffusion coefficients based on out-scatter approximation with transport correction for ^1H in H_2O.
- Discontinuity factors evaluated based on Serpent heterogeneous fluxes and Ants single node homogeneous flux solutions.



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Group constant generation with Serpent (HZDR)

1. The reflector side DF is first evaluated simply as the ratio of the heterogeneous surface flux from the Serpent full core solution and the homogeneous surface flux from a single node Ants calculation using group constants and boundary condition currents from the Serpent3D solution:

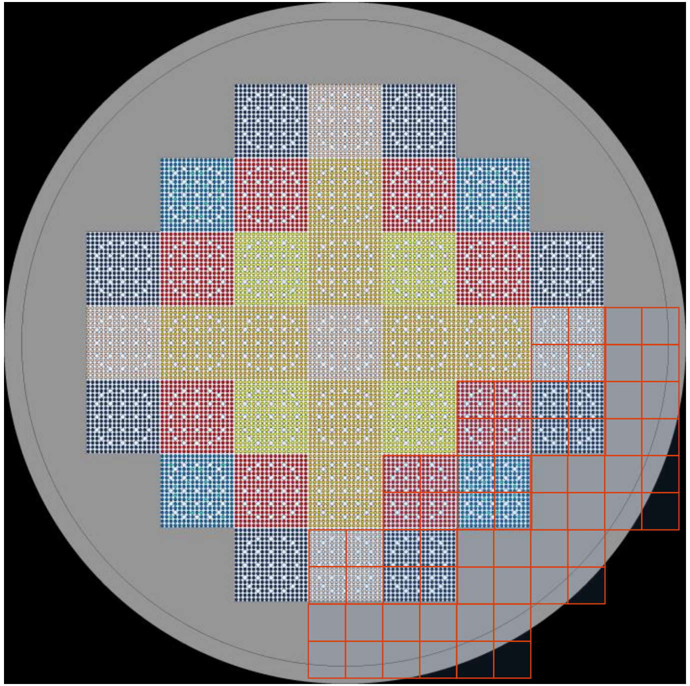
$$f_{\text{refl.}}^{\text{Ants}} = \frac{\phi_{\text{refl.}}^{\text{Serpent3D}}}{\boldsymbol{\Phi}_{\text{refl.}}^{\text{Ants}}}$$

2. The fuel side DF is similarly evaluated

$$f_{\text{fuel}}^{\text{Ants}} = \frac{\phi_{\text{fuel}}^{\text{Serpent3D}}}{\boldsymbol{\Phi}_{\text{fuel}}^{\text{Ants}}}$$

3. This DF is then corrected* by the ratio of the assembly discontinuity factor $f_{\text{fuel}}^{\text{ADF}}$ evaluated for the fuel assembly in the infinite lattice 2D Serpent calculation and $f_{\text{fuel}}^{\text{Ants}}$:

$$f_{\text{refl.}} = f_{\text{refl.}}^{\text{Ants}} \times \frac{f_{\text{fuel}}^{\text{ADF}}}{f_{\text{fuel}}^{\text{Ants}}}$$



* K. S. Smith. “Nodal diffusion methods and lattice physics data in LWR analyses: Understanding numerous subtle details”. *Progress in Nuclear Energy* 101 (2017), pp. 360–369

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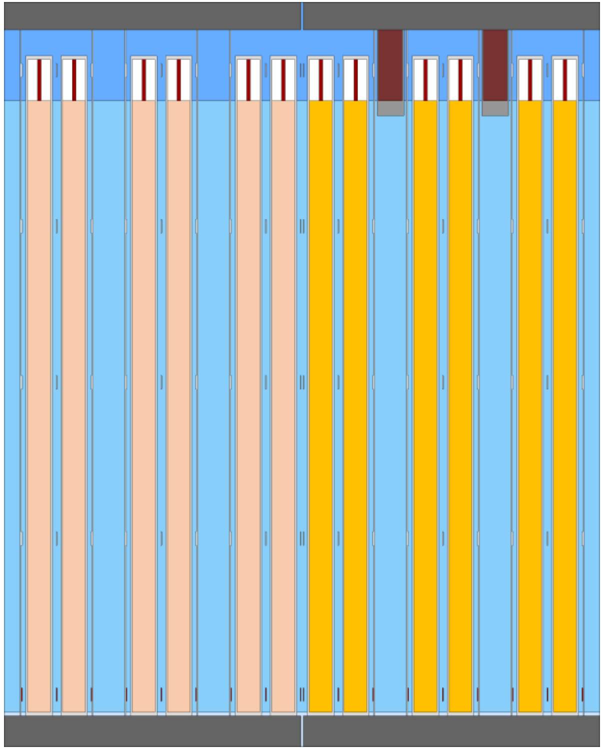


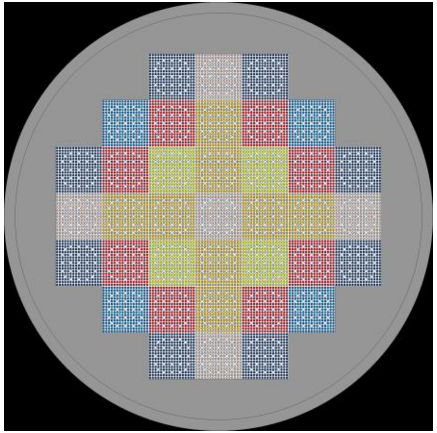
This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

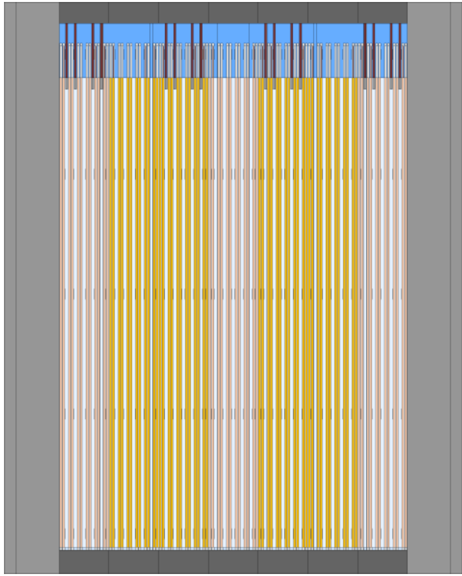
Serpent model for reference solution

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Serpent reference calculations

- A 3D full core Serpent model was used for verification of reduced order neutronics models during group constant generation and nodal model development.
- The same model will be used for high-fidelity coupled transient calculations in T3.4.
- Control rod definitions constructed to allow the movement of different CRAs using the transformation feature of Serpent.
  - Can be accessed from Python via Cerberus.





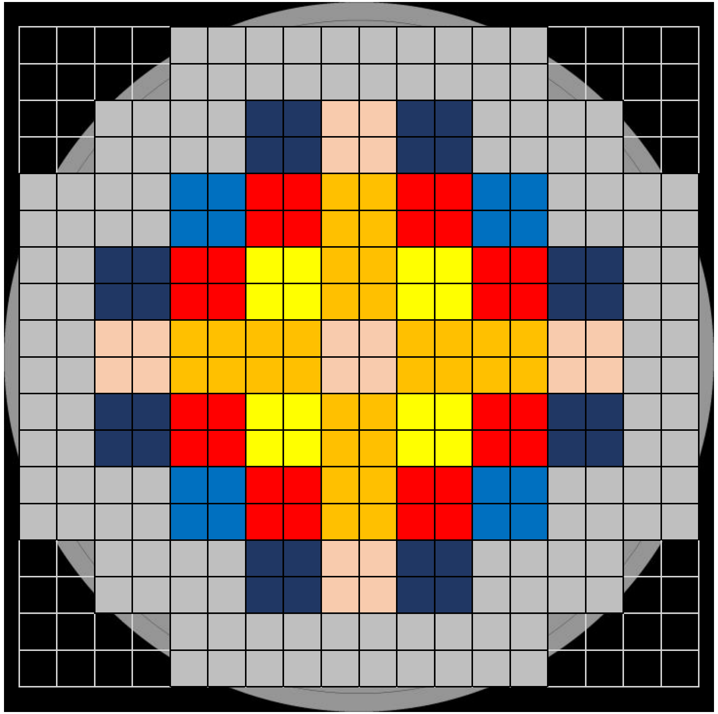
This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Ants nodal model

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Ants nodal neutronics model

- Uses 2x2 radial subnodalization.
- Base model uses 29 axial nodes.
    - Additional higher fidelity nodalizations used in transient analyses.
- 1/8 symmetry utilized in reflector definition.
    - KrakenTools automatically averages Serpent results over symmetric positions in reflector group constant generation calculations.
- Control rod banks and individual rods indicated for movement via Cerberus.
- Three part control rods:
    - Steel plug.
    - AIC part.
    - B4C part.



This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Group constant parametrization

FA GCs
With proper ADFs
And pin power
Form functions

Radial reflector
group constants
incl. DFs

Axial reflector
group constants
(no zDFs)

- Generic polynomial model implemented in Ants with a polynomial fit for momentary state parameters. (T_{fuel} , T_{cool} , ρ_{cool} , C_B).
- Control rod, spacer grid and instrumentation tube are treated as select variables with separate nominal values and polynomial coefficients tabulated for each possible combination.

$$\Sigma(\mathrm{T_f}, \rho_c, \rho_B) = \Sigma\left(\mathrm{T_f^0}, \rho_c^0, \rho_B^0\right) + c_1 \Delta\sqrt{\mathrm{T_f}} + c_2\left(\Delta\sqrt{\mathrm{T_f}}\right)^2 + c_3 \Delta\rho_c + c_4(\Delta\rho_c)^2 + c_5(\Delta\rho_c)^3 + c_6\Delta\rho_B + c_7(\Delta\rho_B)^2 + c_8(\Delta\rho_B)^3 + c_9\Delta\rho_c\rho_B$$

- History effects could be handled using a plutonium history approach (with microdepletion).

V. Valtavirta, A. Rintala. “Specifications for the generic polynomial group constant model of Ants”, Research report (public), VTT-R-00154-21, 2021.
Y. Bilodid. “Spectral history modelling in the reactor dynamics code DYN3D”, PhD thesis, Technical University of Dresden, 2014 (HZDR-051).

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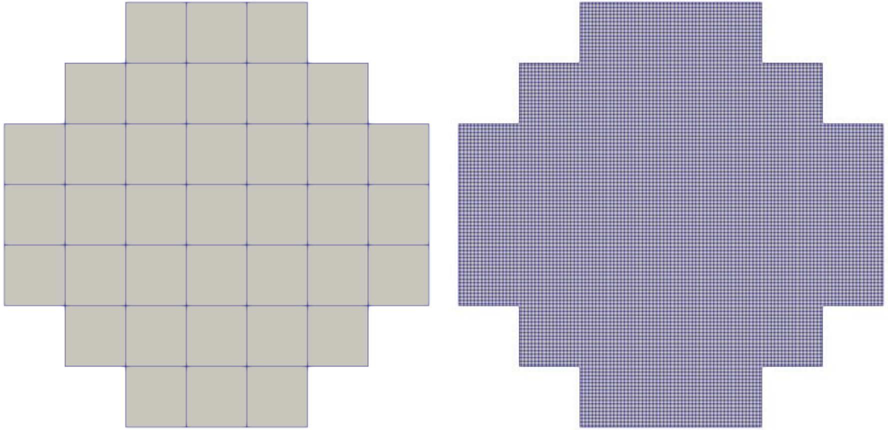
SUBCHANFLOW model

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

SUBCHANFLOW models

- Two different SUBCHANFLOW models were created:
  - Assembly level channels.
  - Fuel rod level channels (channel centered).
- The Python and MED based preprocessor developed in the McSAFE project was utilized in the construction of the model and interpolations between Ants and SUBCHANFLOW meshes.
- The constant fuel properties defined in the scenario specifications were used instead of the full internal fuel behavior solver of SUBCHANFLOW.
- $T_f^{\mathrm{Doppler}} = 0.7T_f^{\mathrm{surf}} + 0.3T_f^{\mathrm{center}}$



Illustrations of the assembly level (left) and fuel rod level (right) channel configurations used with SUBCHANFLOW.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

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This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.


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CURIOUS

EXPERIMENTAL INVESTIGATIONS OF SMRS WITH THE MOTEL TEST FACILITY

Design and first results

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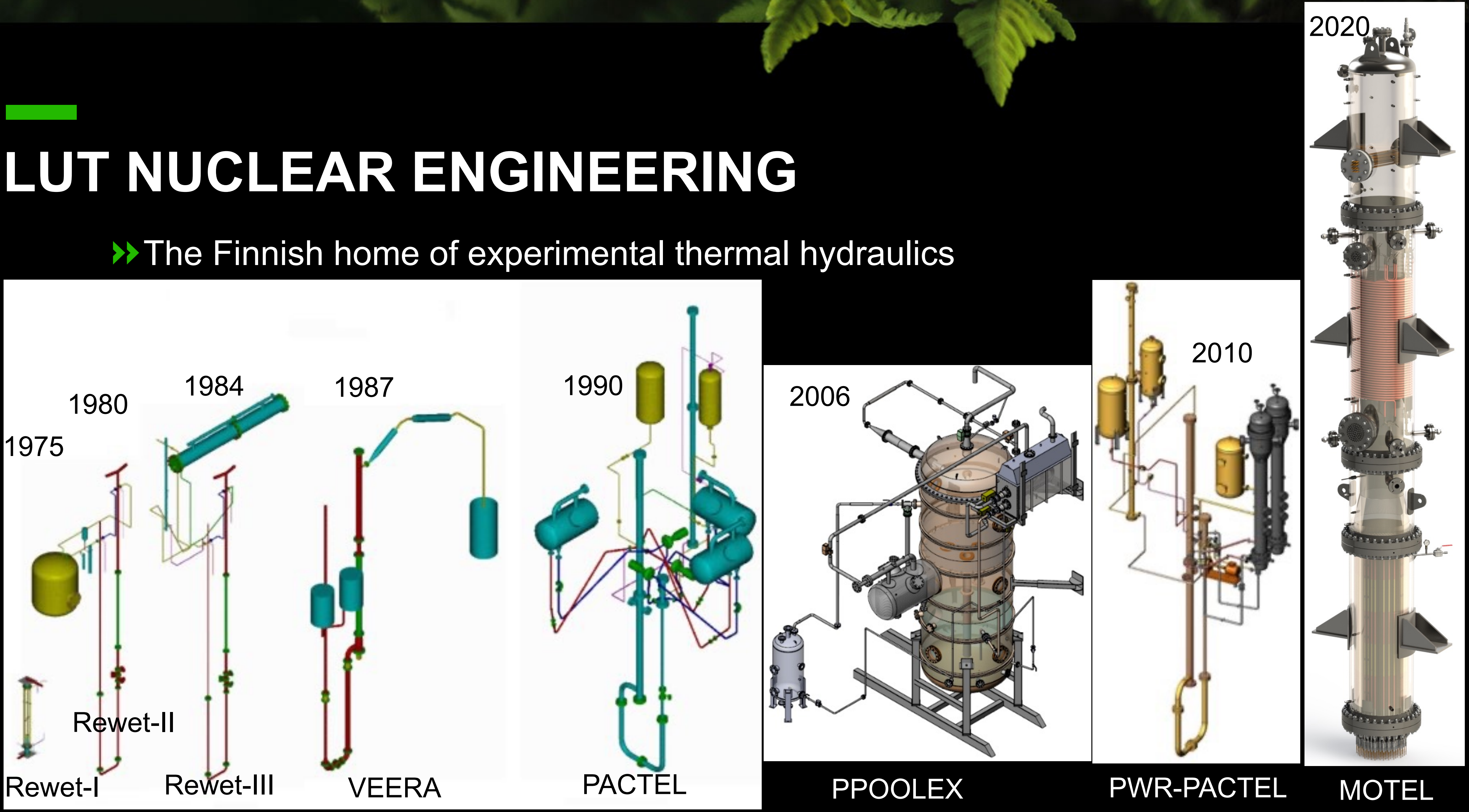
MODULAR TEST LOOP - OUTLINE

- LUT Nuclear Engineering laboratory
- Finnish interest in SMRs
- Basic architecture decisions
- Scaling considerations
- MOTEL design
  - Core
  - Process
  - Instruments
- Some interesting results
- Summary and outlook

LUT NUCLEAR ENGINEERING

›› The Finnish home of experimental thermal hydraulics



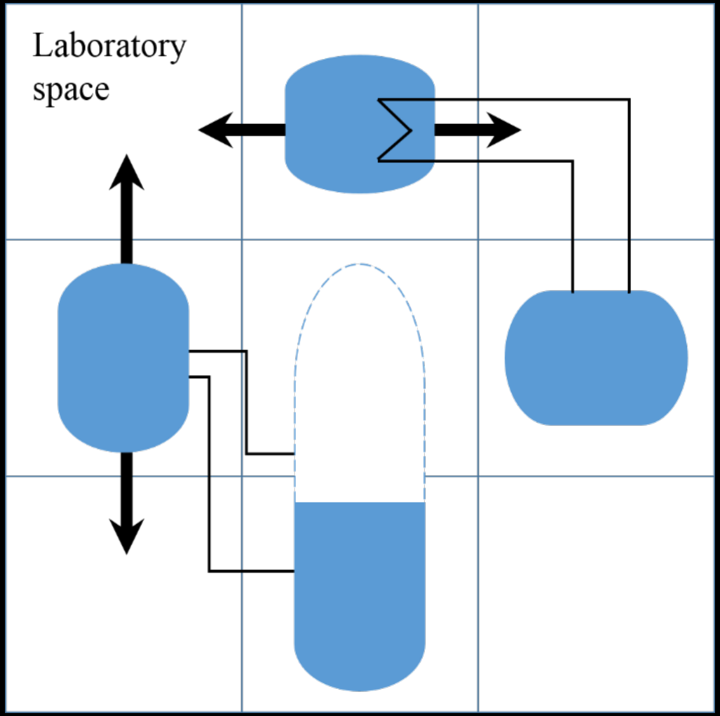
FINNISH INTEREST IN SMRS

- Nuclear power is well accepted in Finland
- Recent experience of large reactor construction – Olkiluoto 3 and Hanhikivi 1 – less encouraging
- Need to decarbonize heating, in addition to electricity → Cogeneration

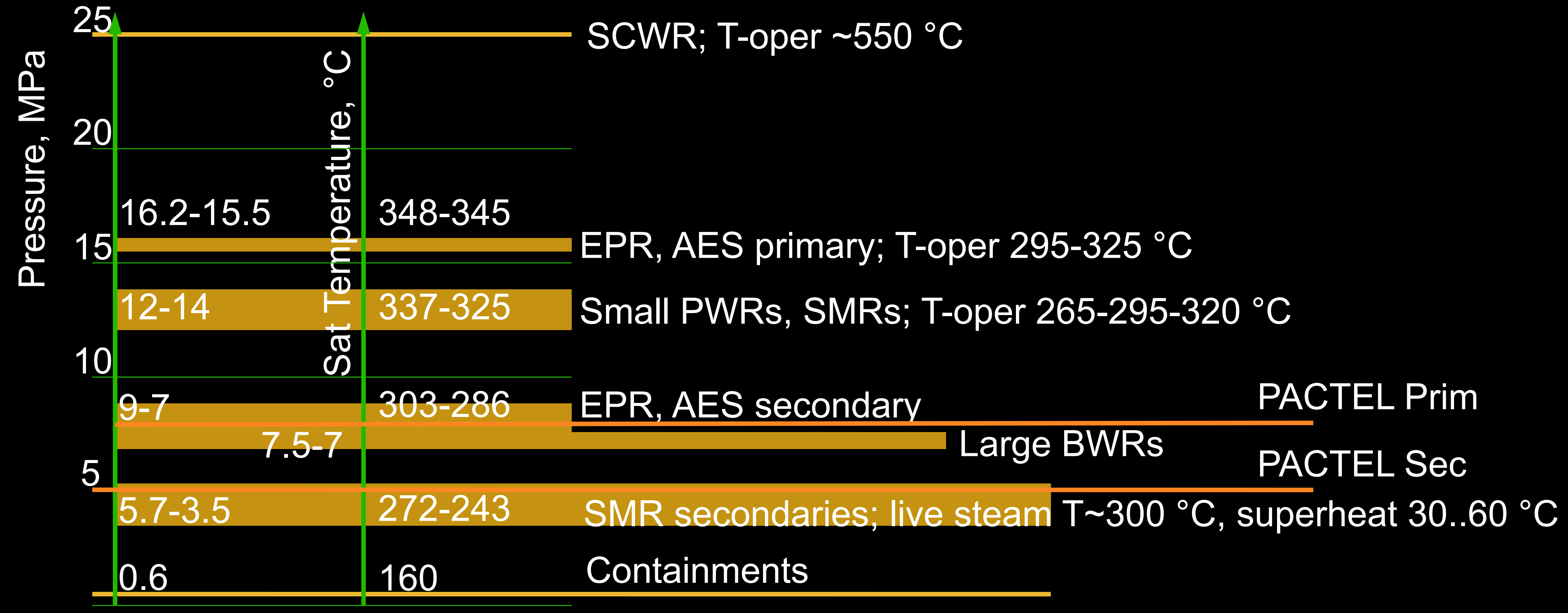
- SMRs promise to address all needs
- New nuclear sites are politically feasible, using cogenerating SMRs or dedicated heating reactors, connected to existing district heating networks
- First-hand experience of novel SMR design features is needed to support eventual deployment →new process features and new process components

MOTEL ARCHITECTURE DECISIONS

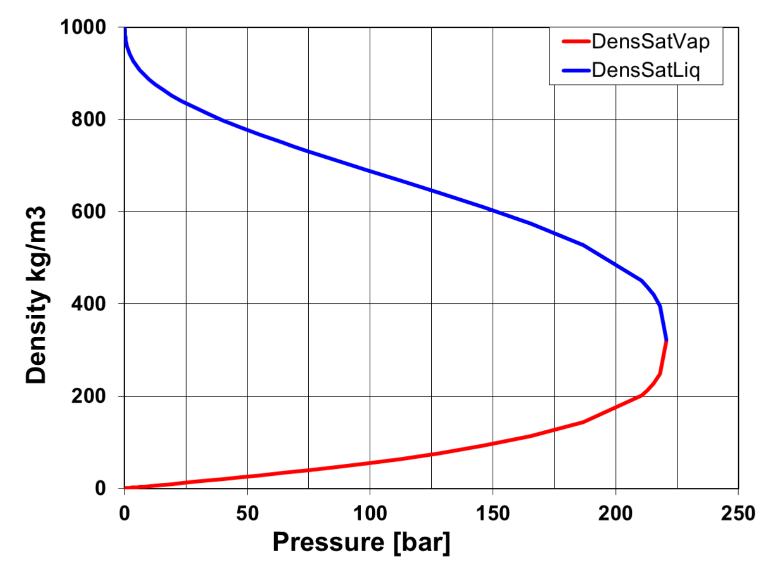
- Must
  - Be **adaptable**, to represent all different reactor systems in use in Finland (VVER, BWR, PWR, …)
  - Have L/D ratios enabling **somewhat realistic 2D/3D** geometric representation
- Must be cost-effective
  - Reduced pressure
  - Standard size pipes, valves, flanges, heaters only
- First iteration must **look forward** → Integral PWR configuration chosen

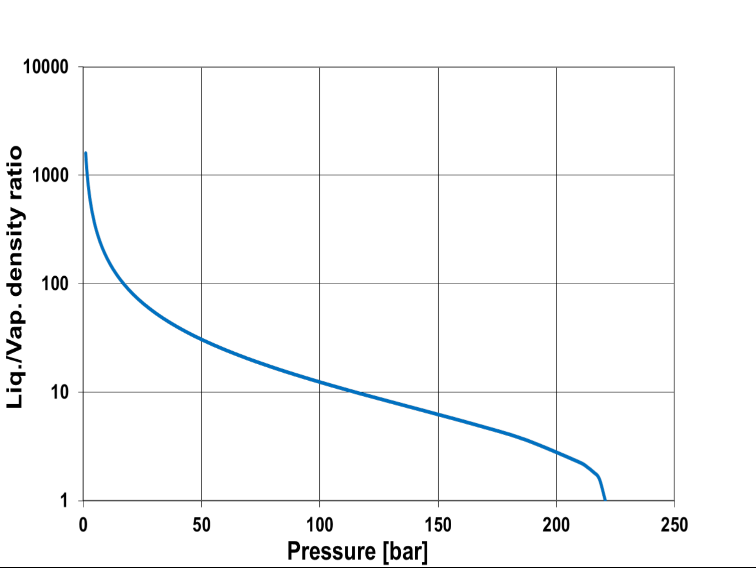


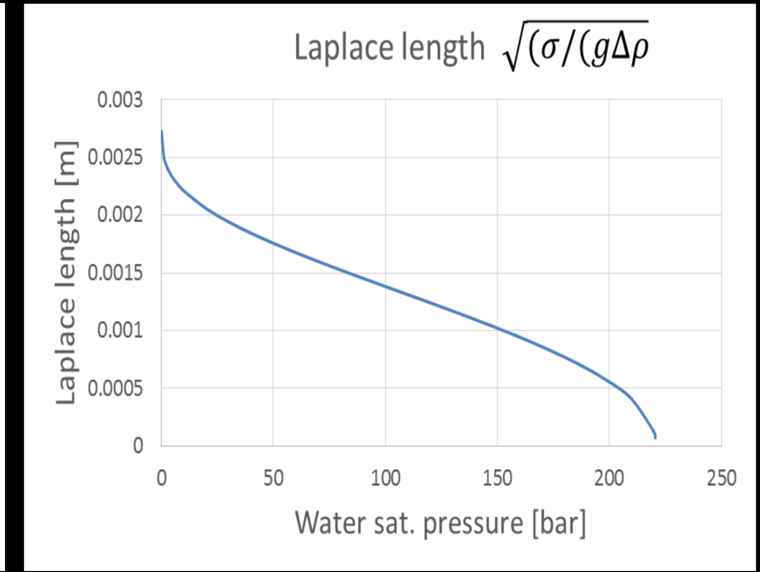
DESIGN PRESSURE AND TEMPERATURE



SOME KEY MATERIAL PROPERTIES



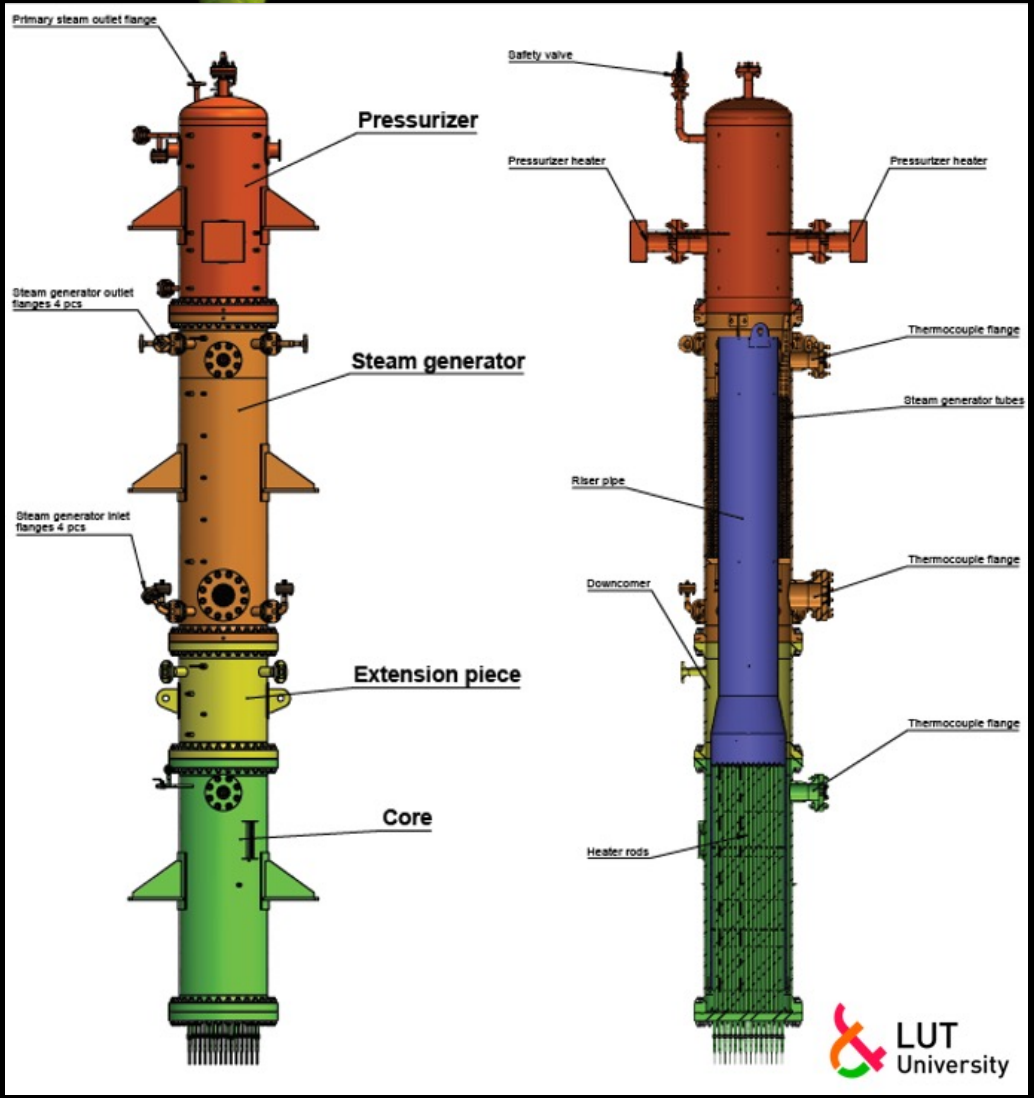




- Density ratio – sensitivity to voiding
- Laplace length – scales to bubble/droplet size

SCALING CONSIDERATIONS

- No specific reference plant – but resembles naturally circulating iPWR
- Generic geometry is good for
  - Phenomena characterization, as long as representative flow regimes reproduced throughout the transients
  - Code validation data generation
- Large D requirement forces reduced height, ~1:2
- Representative flow regimes are obtained by maintaining Froude and Reynolds numbers
  - Local similarity by representative diameters in SG tubes
  - Approximate Re similarity in core; unheated (dummy) rods for reasonable P/D



REDUCED HEIGHT IS THE NORM THESE DAYS

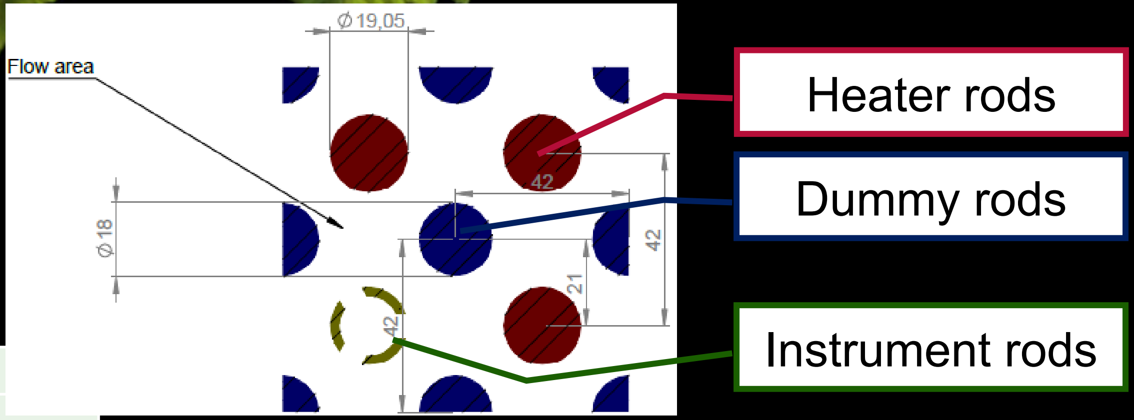
Facility	Owner	Ref. plant	Volume scale	Height scale	Time scale	Max pressure
APEX	Oregon State U	AP600	1/192	1/4	1/2	~2.5 MPa
PUMA	Purdue U	SBWR	1/400	1/4	1/2	~1 MPa
ATLAS	KAERI	APR1400	1/288	1/2	$1/\sqrt{2}$	18.7 MPa

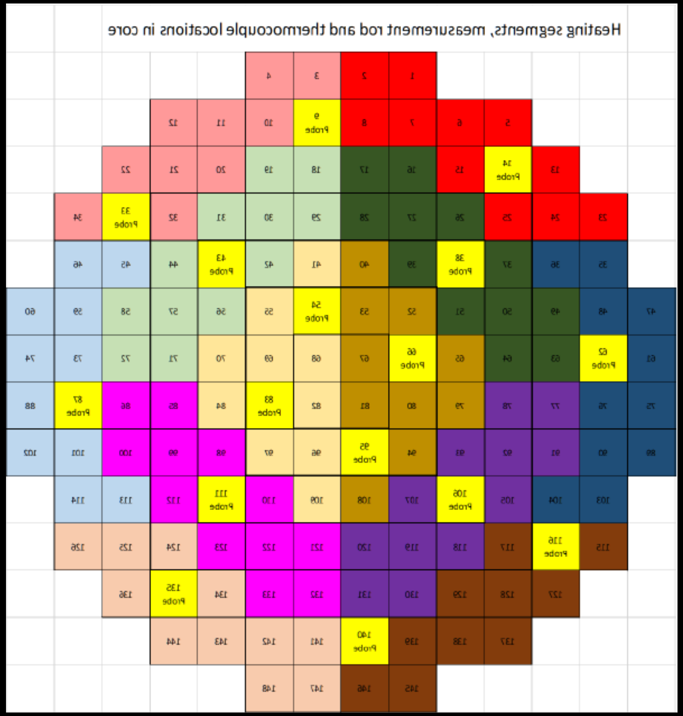
MOTEL OVERALL DESIGN

Characteristic	MOTEL test facility
Reference system	None – overall geometry similar to MASLWR and NuScale
Height scale (riser & downcomer pipelines, heat exchanger) (approximate)	1:2
Maximum pressure, reactor vessel	**4 MPa**
Maximum **heating power**	**990 kW**
Maximum temperature, reactor vessel	523 K (250 °C)
Height of the vessel	7.4 m
Height of core	1.83 m
Height of the helical coil steam generator	1.311 m
Elevation difference from core midplane to steam generator midplane	3.23 m
Core outer diameter	0.602 m
Downcomer annulus gap (riser / core)	98 mm / 36 mm
Main material of the components	Stainless steel
Insulation material / thickness	Mineral wool / 120 mm

MOTEL CORE DESIGN

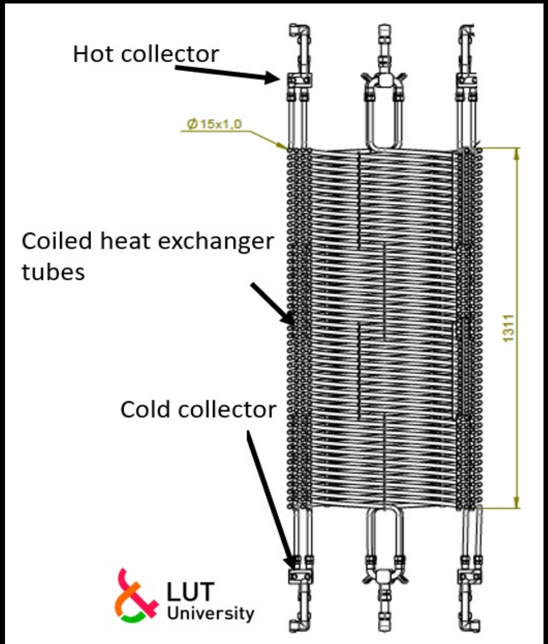
Characteristic	MOTEL test facility
Height of core	1.83 m
Core diameter	0.6 m
Electrically heated rods	132
Heated rod diameter	19.05 mm
Rod pitch	29.7 mm
Heated rod power profile	Chopped cosine, 5 steps
Rod power individual / total	7.5 / 990 kW
Independently controllable core heating segments	**12**
Dummy rods	145
Dummy rod diameter	18 mm
Instrument rods	16
Instrument rod diameter	18 mm
Largest achievable average heat flux	63 kW/m²
Largest achievable average heat flux, relative to initial NuScale design	17 %





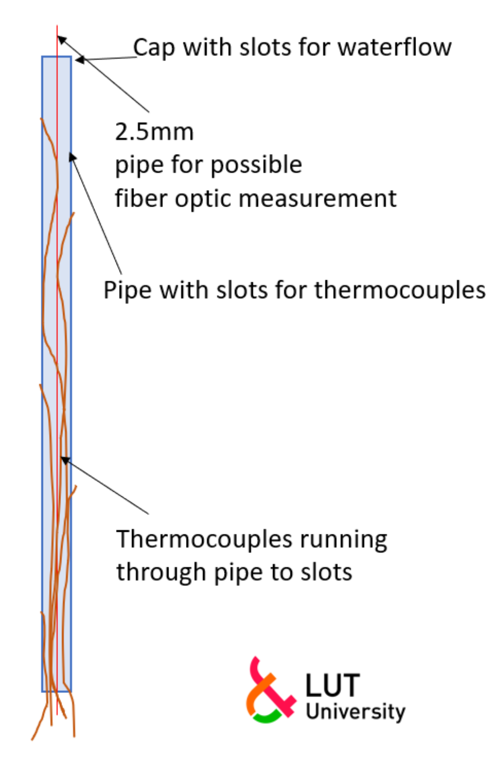
MOTEL HELICAL COIL STEAM GENERATOR

Characteristic	MOTEL test facility
Height of the helical coil steam generator	1.311 m
Total number of tubes / Number of layers	16 / 4
Tube outer diameter / wall thickness, mm	15 / 1
Tube lengths, m	20.0, 21.7, 23.4, 25.1
Coil diameters, mm	515, 560, 605, 650
Total heat transfer area	17 m^2
Area matched to core heat transfer area	
Tube material	Stainless steel

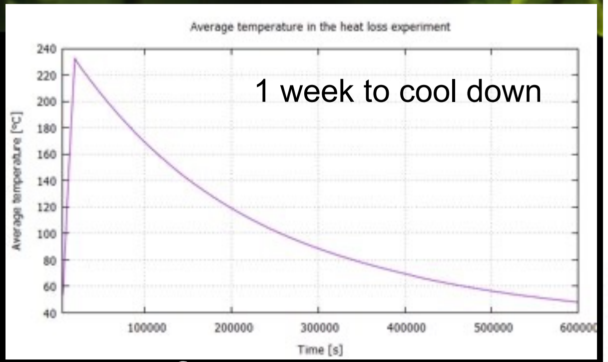


MOTEL INSTRUMENTS

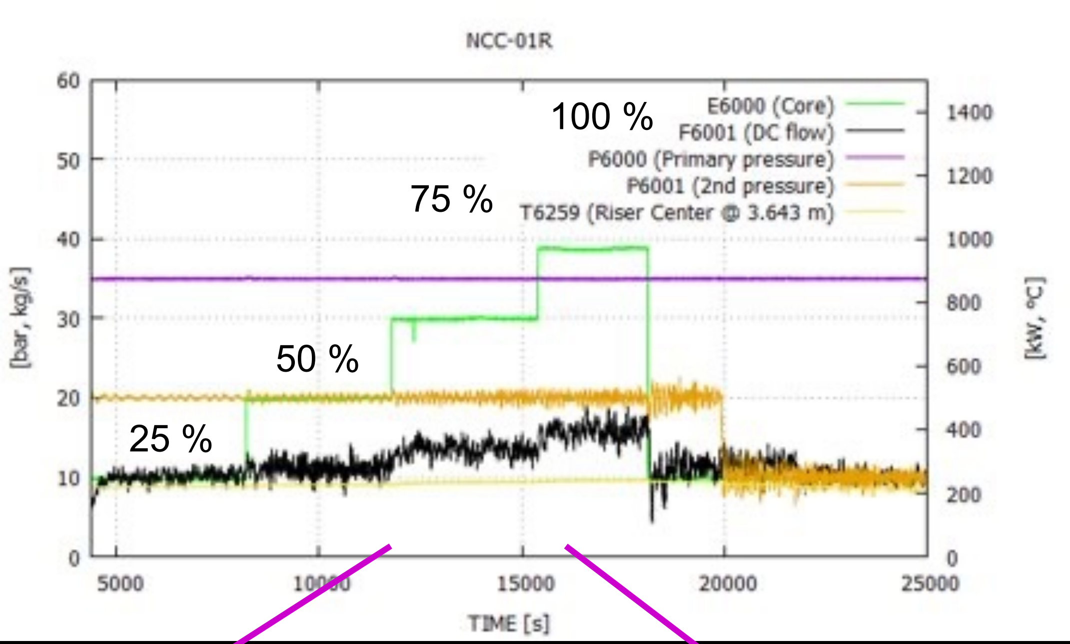
- 132 J-type thermocouples inside heater rods
- 80 core region fluid temperature measurements in instrument rods (5 each)
  - Capillary pipe for later installation of fiber optic sensors
- Inlet, outlet and 3 intermediate thermocouples on **each SG tube**
- 13 thermocouples in the downcomer, 20 in the riser, 5 at each of 4 elevations
- Ultrasonic flow meters for downcomer flow
- Pressure and pressure differentials
- Separate process control, process safety and data acquisition systems

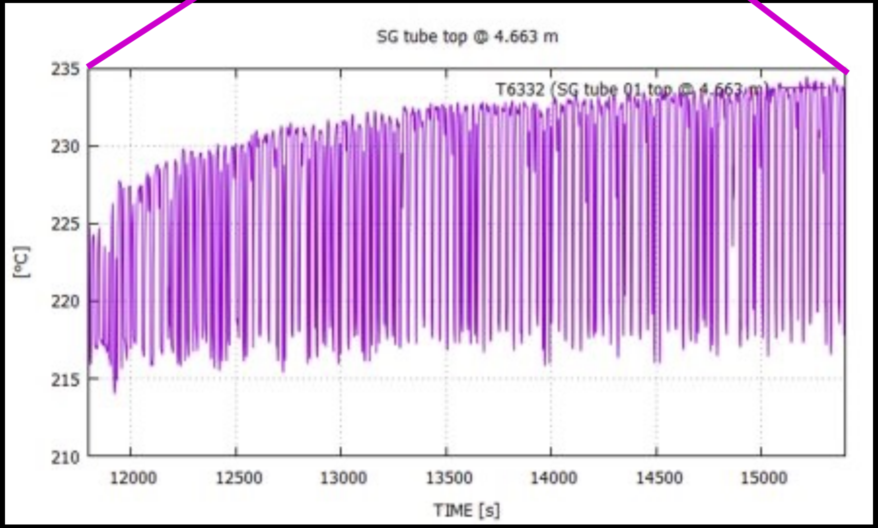


FIRST RESULTS



- Shakedown testing successfully verified performance and data acquisition
- Pressure drop and heat loss tests to facilitate code calculations – very low heat losses observed, as expected, thanks to large D
- Steady-state natural circulation tests at constant power levels 25%/50%/75%/100%
- Interesting noise observed in the helical coil steam generator: apparently density wave oscillations develop in SG tubes



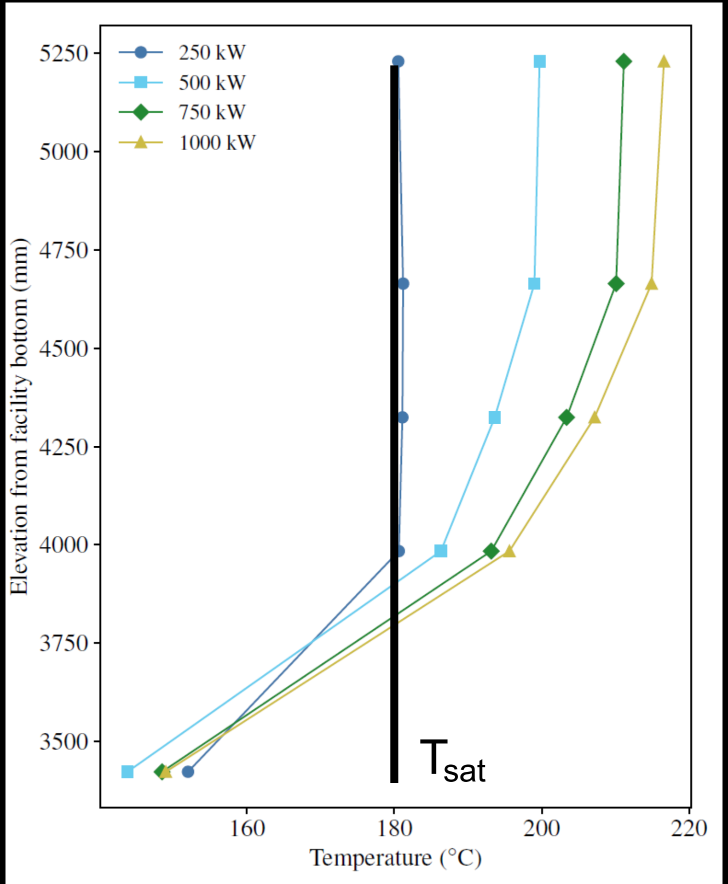


RESULTS FROM MCSAFER SG TESTING

- Power steps 250 / 500 / 750 / 1000 kW
- Primary pressure 3.5 MPa
- Secondary pressure 1 MPa
  → T_{sat} 180 °C
- Feedwater temperature ~20 °C

- The steam generator can superheat steam by 30+ °C already at full MOTEL power, about 16 % of the NuScale prototypic heat flux

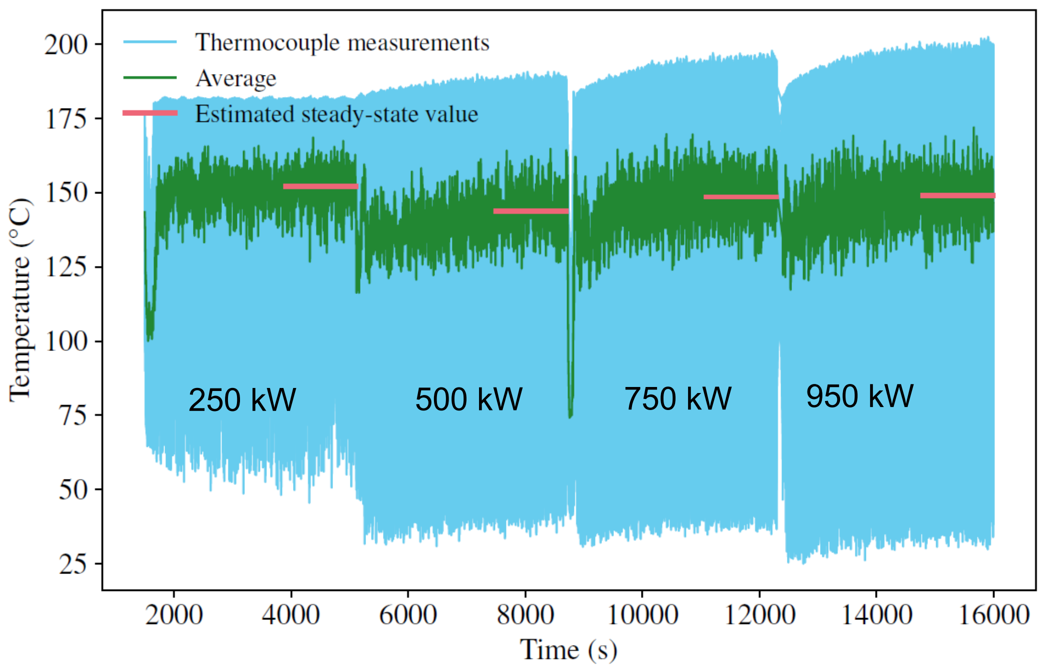
SG tube temperature profiles (smoothed)



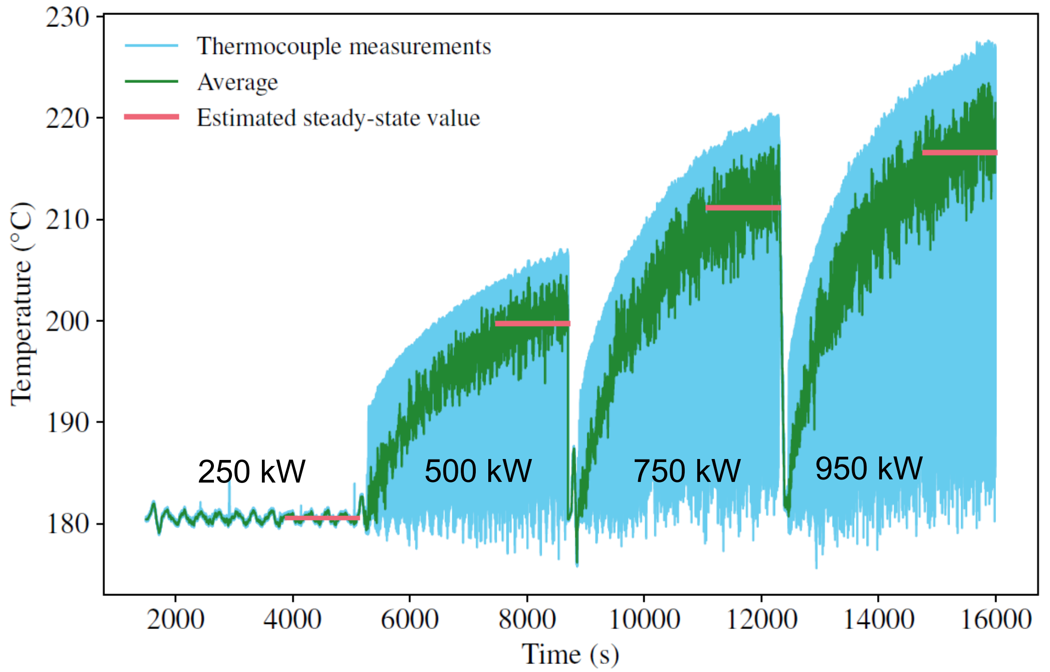
BOILING IS OSCILLATORY – DENSITY WAVES?

- Cold collector T oscillates between feedwater and saturation → momentary backflows
- Hot collector oscillates between saturation and 30 °C superheat → slug penetration

SG cold collectors



SG hot collectors



NATURAL CIRCULATION IN THE PRIMARY

Simple gravity driven loops in steady state satisfy pressure balance equation

$$(\rho_{CL} - \rho_{HL})gH = F \cdot {q_m}^n, \qquad n \cong 2$$

Where $n \cong 2$ for fully turbulent (roughness) flow and F contains all friction factors, local and distributed.

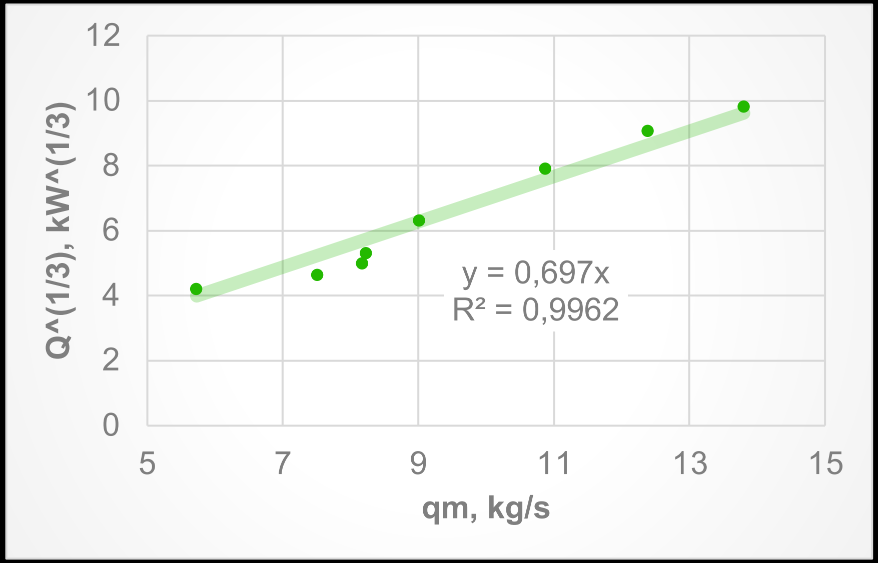
Using energy balance $Q = q_m(h_{HL} - h_{CL})$, and calculating enthalpy and density differences using loop ΔT ($\rho_{CL} - \rho_{HL} = \rho_{av}\beta\Delta T, h_{HL} - h_{CL} = c_p\Delta T$), one finds that

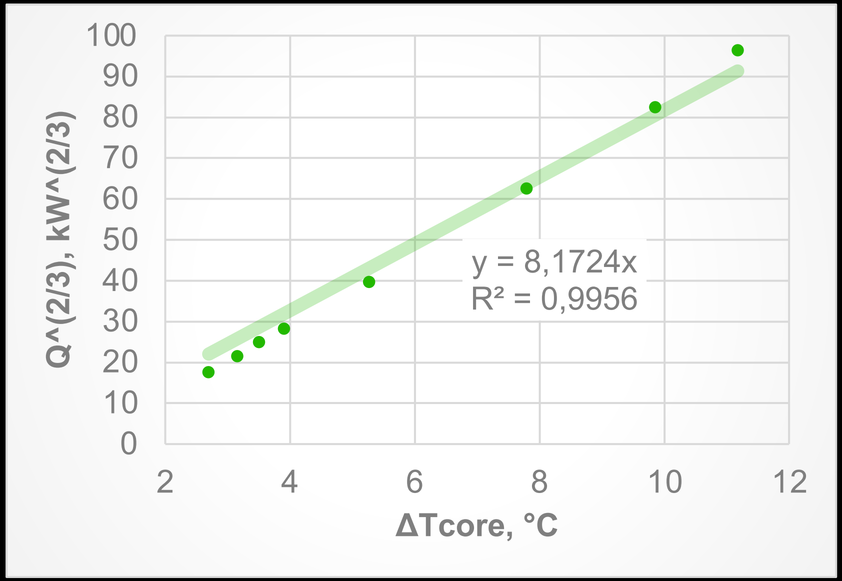
- Loop mass flow $q_m \sim \left(\frac{Q}{F}\right)^{\frac{1}{n+1}} = \left(\frac{Q}{F}\right)^{\frac{1}{3}}$ $for\ n = 2$
- Temperature rise $\Delta T \sim Q^{\frac{n}{n+1}} F^{\frac{1}{n+1}} = Q^{\frac{2}{3}} F^{\frac{1}{3}}$
- Power transferred $Q \sim \left(\frac{\Delta T^3}{F}\right)^{\frac{1}{2}}$

Friction factors F appear only under square or cube roots, therefore single-phase natural circulation is not at all sensitive to uncertainties in friction.

NATURAL CIRCULATION IN THE PRIMARY

Observed primary flow – heating power and core temperature rise – heating power relations in McSAFER tests





Dependences $q_m \sim Q^{\frac{1}{3}}$ and $\Delta T \sim Q^{\frac{2}{3}}$ seem to hold rather well.

SUMMARY AND OUTLOOK

- MOdular TEst Loop successfully built and commissioned
- Already yielding valuable insights into natural circulation cooled integral PWR inherent behaviours
  - Boiling oscillations in Helical coil steam generator
  - Robustness of natural circulation
- EU Horizon 2020 project McSAFER to produce further data on helical coil steam generator and non-uniform core power distribution – test program ongoing
- Ready for extension with additional systems and/or components, of relevance to SMRs, existing and planned Finnish reactors, or future district heating reactors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