The KIT Computational and Experimental Research Platform for Nuclear Reactor Safety

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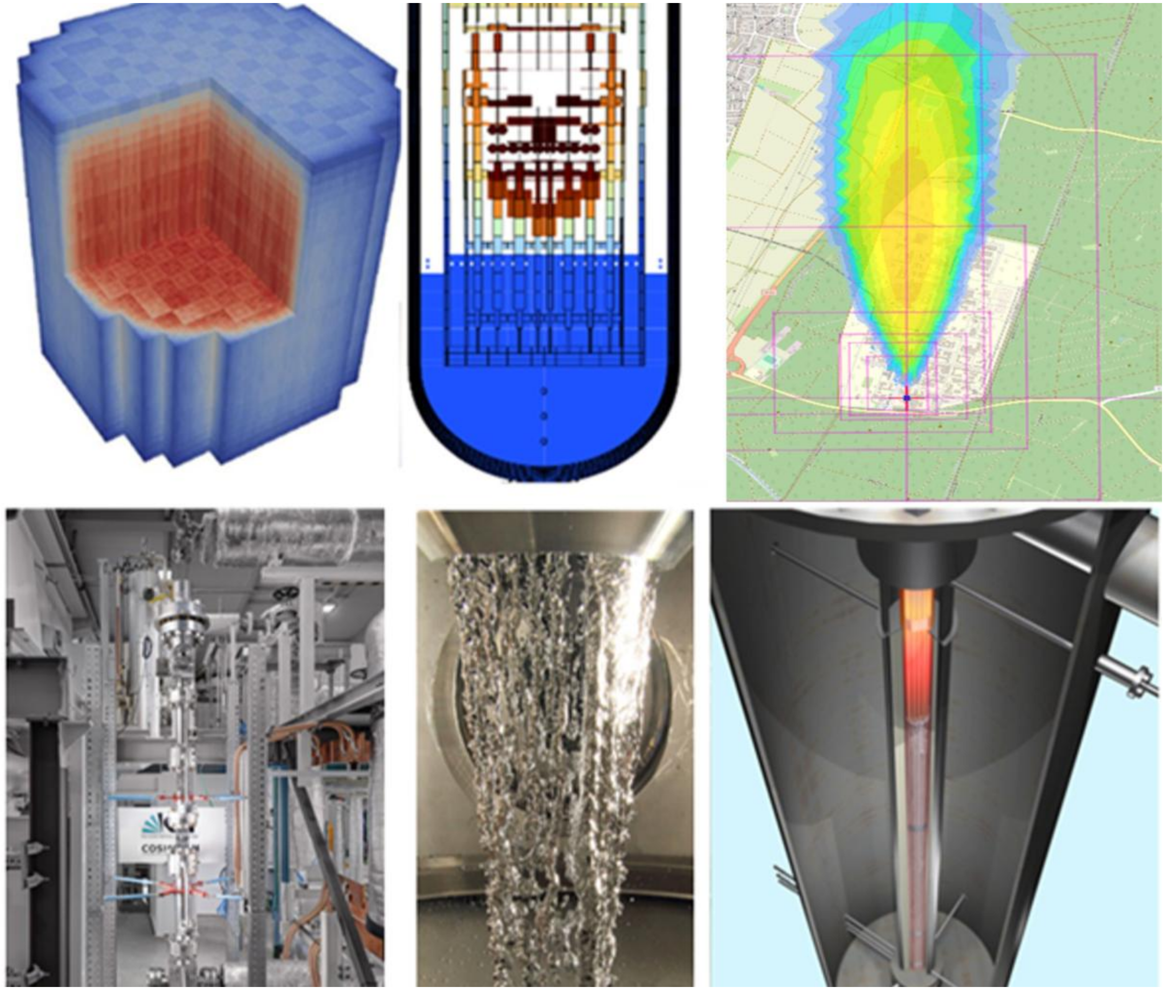
Reactor Physics and Dynamic Group (RPD)



The Helmholtz Association

Germany's Largest Research Organization

- 43,000 employees
- Annual budget of ≈ 5 billion euros
- 18 Helmholtz Centers

KIT participation in 11 programs of 4 Helmholtz Research Fields with a funding volume of 338 million euros per year (as of 2024)

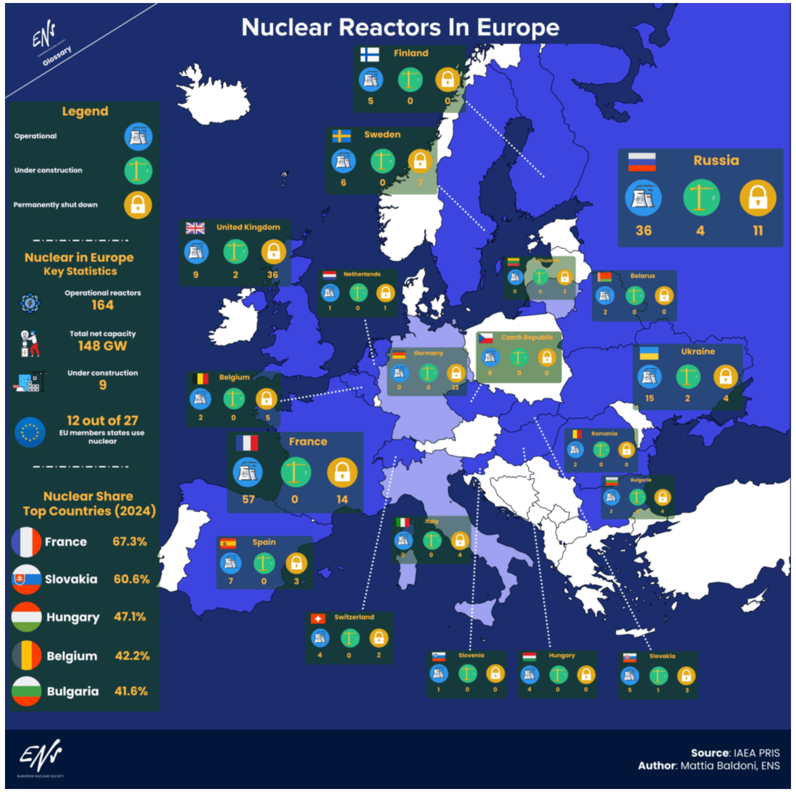
Helmholtz Research Fields

Energy	**Earth & Environment**	Health
Aeronautics, Space and Transport	**Information**	**Matter**

This makes KIT and Helmholtz unique partners with common goals.

Motivation

- Framework: Helmholtz NUSAFE long-term Research Program
- Many and different NPPs under operation and going to be deployed around Germany and worldwide
- Mission:
- **Keeping and improving expertise and knowledge** to support the **assessment of the hypothetical risks** that may arise from the operation of current operating and innovative designs NPPs, e.g., LWRs, SMRs, AMRs, Gen.IV



https://www.euronuclear.org/glossary/nuclear-power-plants-in-europe/

Strengthen the **Helmholtz Simulation Platform** by combining large-scale test facilities and advanced simulations for **nuclear reactor safety, Environmental Impact Assessment, and Emergency and Preparedeness Response** highly relevant to operating NPPs and innovative reactors systems in the EU and worldwide

Computational and Experimental Platform for Reactor Safety
Strategy

Providing reference experimental data for codes‘ validation to the international community (e.g., CFD, subchannel, system, and integral tools)

Past and new experimental campaigns, improvement of the existing large-scale facilities

Developing advanced calculation methods and methodologies **to provide reference solutions** for safety analyses in nominal and accident conditions

Development and improvement of codes and methods on

- **Multiphysics and multiscale** (core and plant analyses)
- **Severe accidents**
- **Uncertainty and Sensitivity**
- **EIA and EP&R**
- **Codes‘ validation using KIT and external experiments**
- **AI and ML**
- **Strategic international cooperations** (US NRC, ASNR, JAEA/CEA, VTT, JRC KA, CNL, ….)

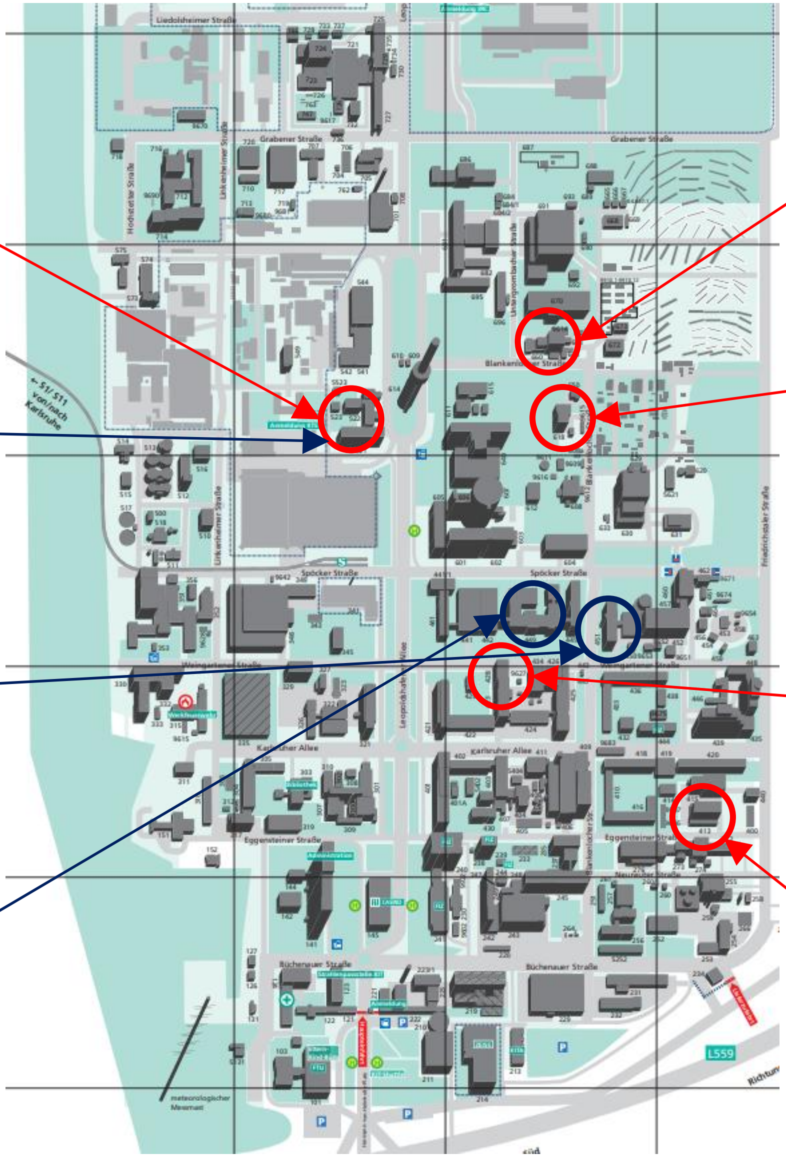
The Current KIT (Campus North) Wonderland

KASOLA
TH Sodium

Simulation groups
Reactor Safety

Simulation group
JRODOS

High-performance computing centre



HELOKA-US
TH MSR

QUENCH
TH+Materials

COSMOS-L and -H
TH+Materials

KALLA
TH+Materials

Water-cooled Nuclear Reactors' Safety

Large-Scale Facilities and Calculation Platform

The COSMOS and QUENCH Large-Scale Facilities

COSMOS

- Experiments on two-phase flow with heat transfer
- **Flow boiling phenomena including critical heat flux for classical and ATF materials**

QUENCH

- Investigation of hydrogen source term during early in-vessel phase and effect of reflood
- **Current focus on ATF cladding materials**

- **QUENCH+COSMOS allow covering the entire range from normal operation to DBA and BDBA**

COSMOS-L Low Pressure Water loop

(Courtesy of S. Gabriel)

Thermal-hydraulics Investigations

- Low pressure water loop with **0,3 MPa and 160°C**
- Thermal power 150 kW – 300kW, including 70 kW for the test-section

- More the 1000 CHF-events on different heater geometries
  - single tube
  - bundle of 5 tubes
  - flat heater for IVR-ERVC experiments

- **Transfer to COSMOS-H**
  - **New optical methods for image processing**
  - **Pre-experiments**

https://www.ites.kit.edu/english/128.php

COSMOS-H High Pressure Water/Steam Cycle

(Courtesy of S. Gabriel)

Thermal-hydraulics Investigations

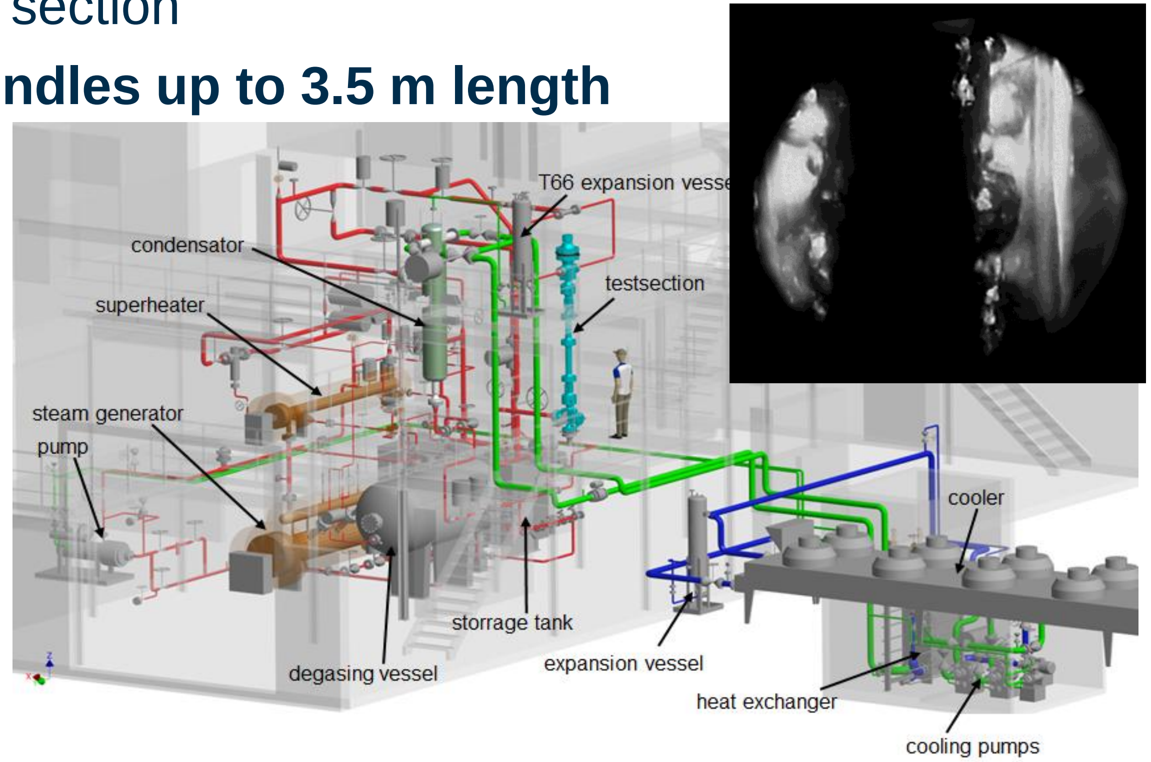
Facility features

- **New TH facility (commissioned in 2025)**
- **Water/Steam loop** @17 MPa, 370°C
- Thermal power 1.8 MW including 600 kW for the test section
- **Test section for heated single tubes and small bundles up to 3.5 m length**
- 1:1 conditions to most of the LWR Concepts

Experiments in BWR, PWR, and SMR conditions.

Research focus

- **Flow boiling phenomena** including **CHF**
- **Optical measurement** technology under **high pressure conditions**
- **Validation data** for CFD, system TH and subchannel codes, e.g., local void fraction, nucleation sites, bubble diameter, bubble velocity

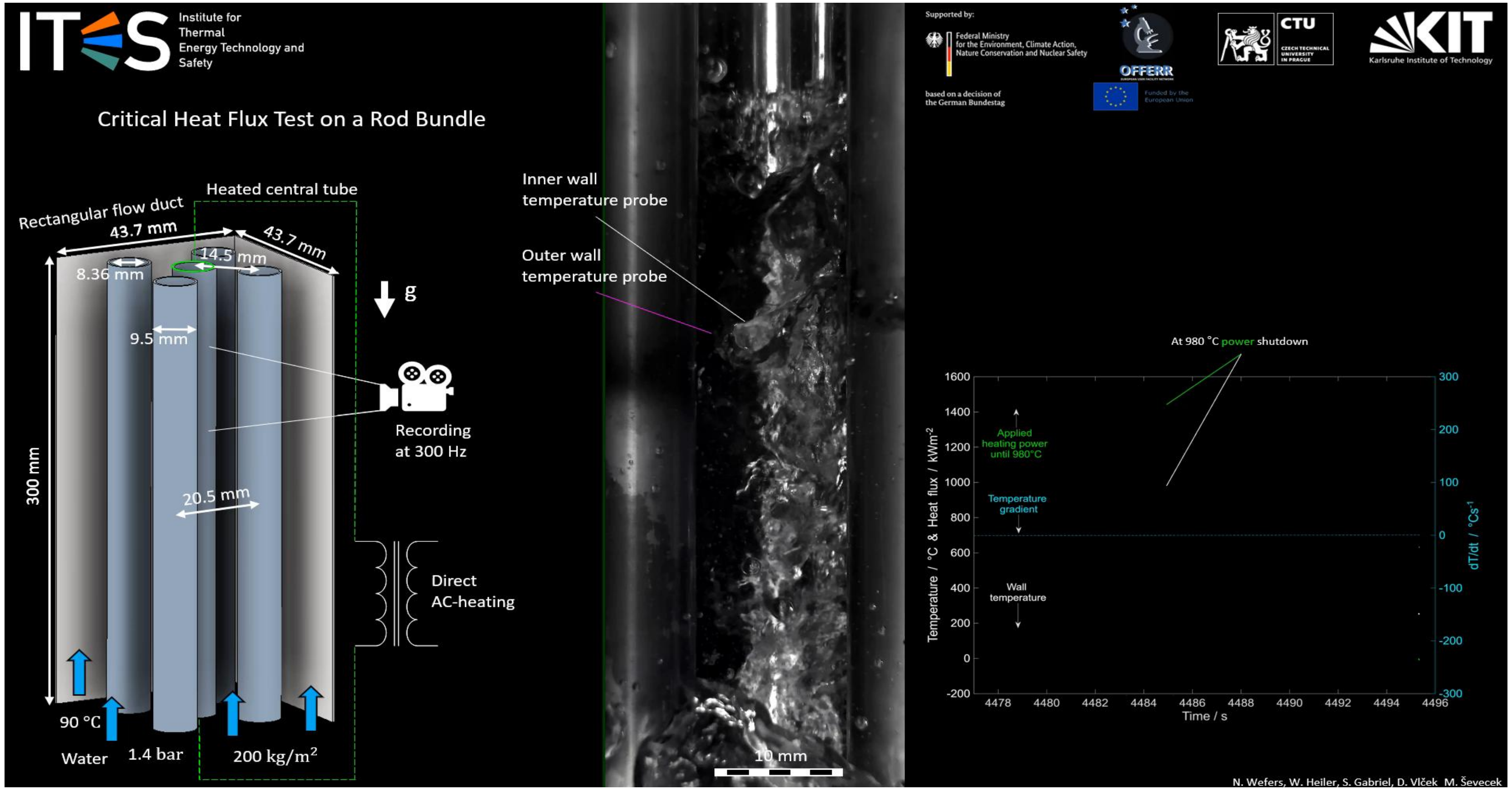


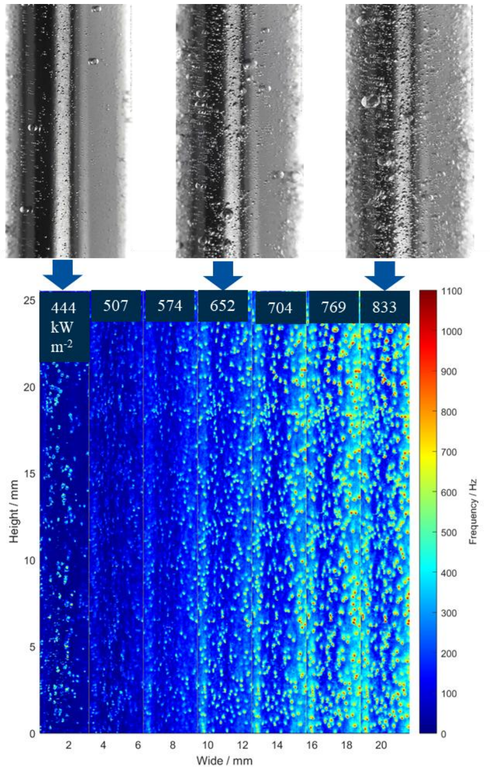
S. Gabriel, et al., 2022, https://data.europa.eu/doi/10.2777/662189

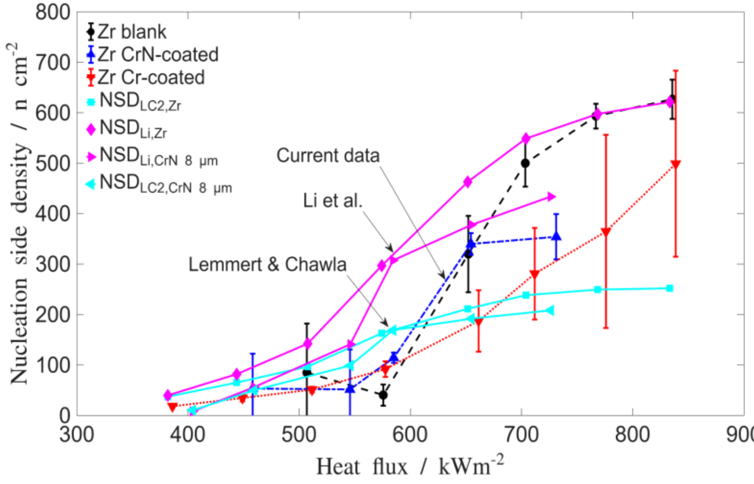
COSMOS-H and COSMOS-L

ATF-related activities at COSMOS Lab

- Investigating the heat transfer of Cr-coated Zr and FeCrAl ATF materials under normal and DBA conditions
- **Together QUENCH and COSMOS are used to cover the entire range from normal operation to DBA and BDBA**







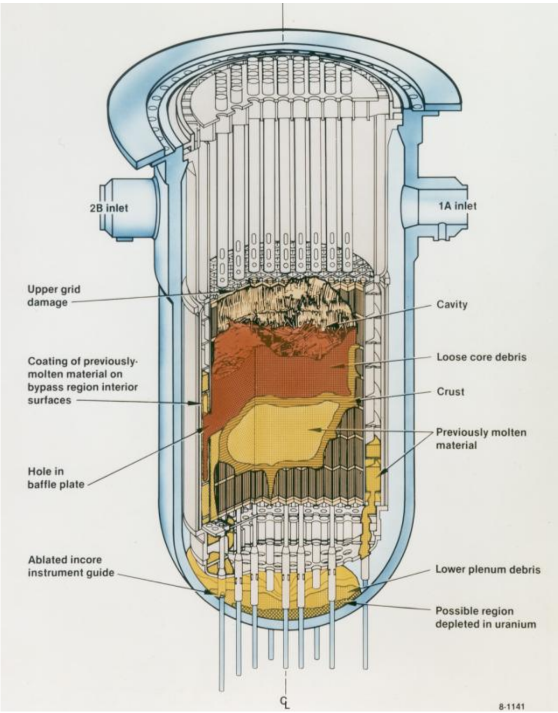
(courtesy of S. Gabriel)

QUENCH Facility: Historical Overview

(courtesy of J. Stuckert)

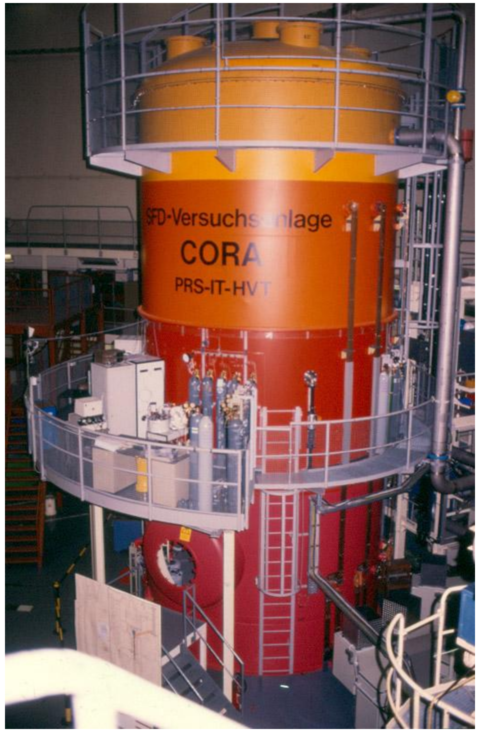
TMI-2-Accident

28 March 1979: 50% reactor core fragmented or melted, H_2 generation



CORA (1986 - 1993, 19 Tests)

Investigation of melt formation and -relocation



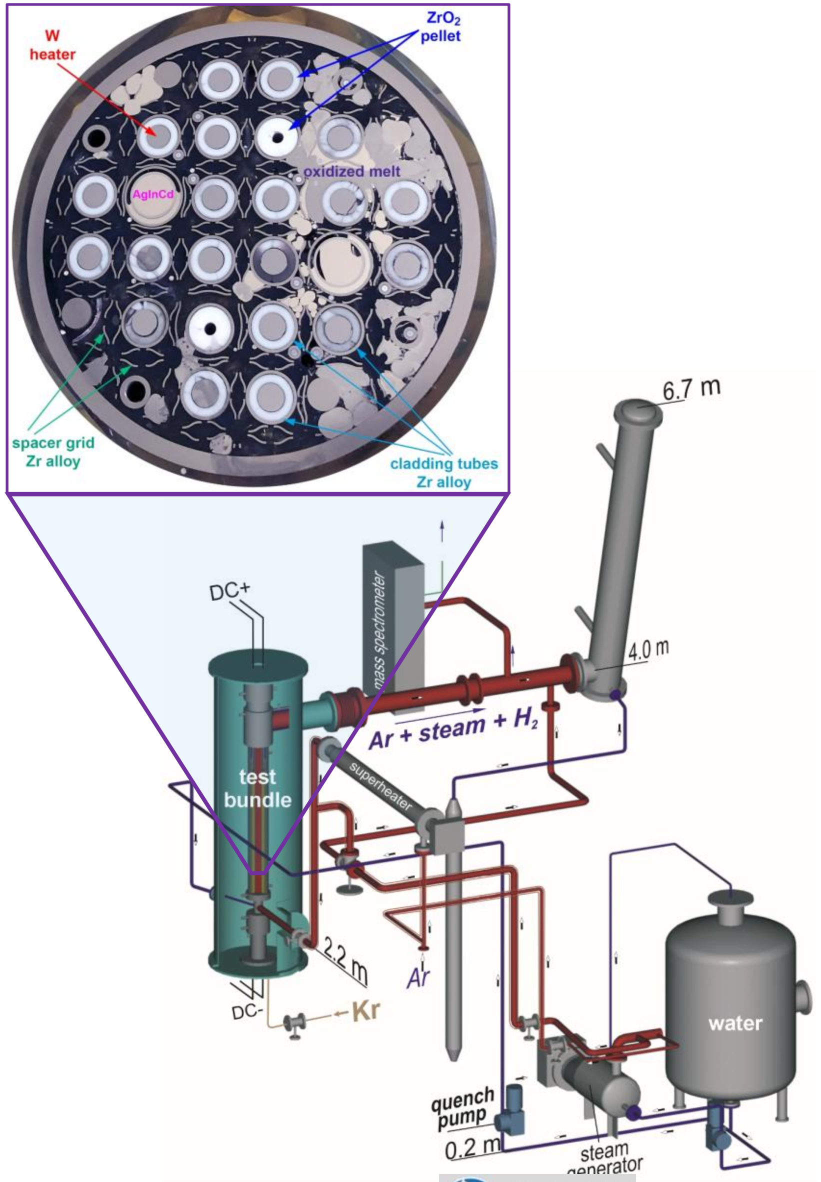
QUENCH (1997 → now) (21 Tests +7 LOCA)

Reflooding of overheated bundles, Material behaviour, Hydrogen generation

G. Schanz et al., . Information on the evolution of severe L WR fuel element damage obtained in the CORA program, JNM 188 (1992) 131-145
J. Stuckert et al., Experimental program QUENCH at KIT on core degradation during reflooding under LOCA conditions and in the early phase of a severe accident, in IAEA-TECDOC-CD-1775, 2013
T. Haste et al., A comparison of core degradation phenomena in the CORA, QUENCH, Phébus SFD and Phébus FP experiments, NED 283(2015) 8–20

QUENCH/LICAS high Temperature Facility for Bundle Tests
Innovative Materials Investigations

- Length of the test bundle: 2.2 m
- Number of tested rods: 21…31
- Electrical power: maximum 160 kW, variable in steps of 20 W
- Internal pressure in fuel rod simulators: adjustable up to 15 MPa



Unique worldwide large-test facility for investigating the materials' behaviour in
- **Design basis accidents** (T ≤ 1200 °C)
- **Beyond design basis accidents** (1200 °C < T < 2300 °C)
- **Dry storage conditions** (T < 450 °C for many months)

J. Stuckert et al., 2020, https://doi.org/10.1016/j.jnucmat.2020.152143

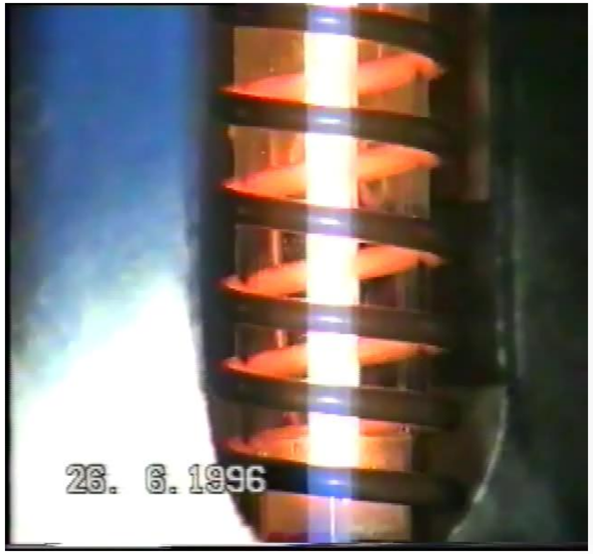
The QUENCH Experimental set-up

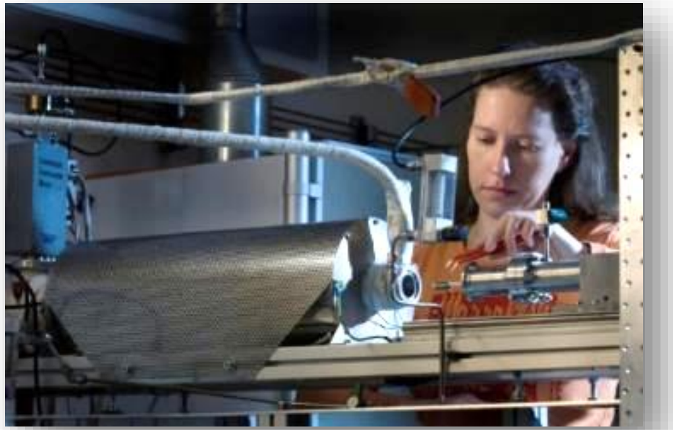
(courtesy of J. Stuckert, M. Steinbrück)

From Separate Effect Tests to Reactor Application

Bundle tests

Separate-effects tests





- 10-15 cm cladding tube segments
- Inductive heating up to 2000°C
- Water quench

- DTA-TG systems
- Max. temp. 1600°C

- Tube furnaces
- Max. temp. 1600°C
- Sample airlock

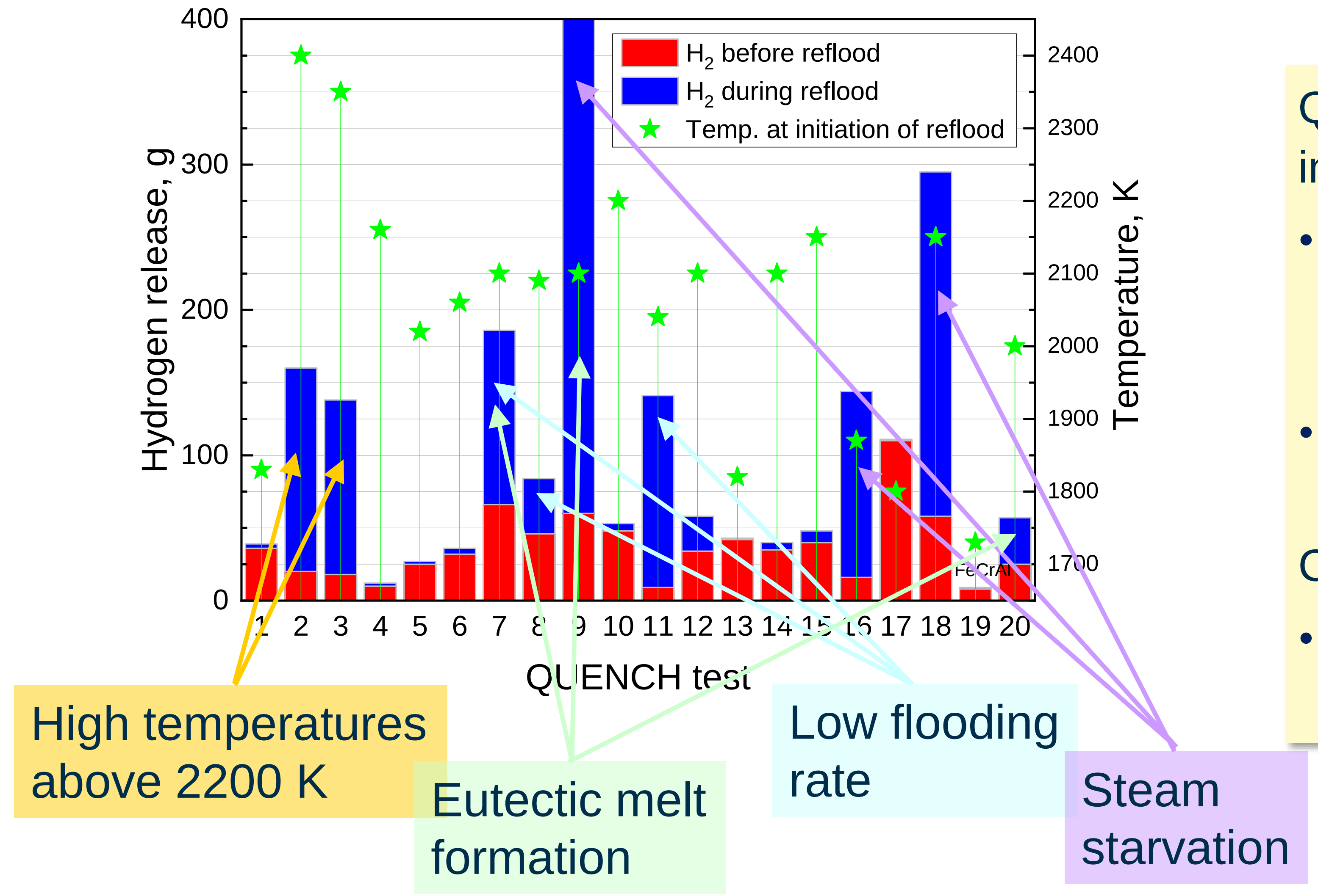
All systems with water steam supply and mass spectrometer coupling

J. Stuckert et al., 2020, https://doi.org/10.1016/j.jnucmat.2020.152143

IAM Institute for Applied Materials KIT

Summary of the SA QUENCH bundle tests

(courtesy of J. Stuckert, M. Steinbrück)



QUENCH bundle tests can be divided into those:

- with successful flooding resulting in immediate cooling and low hydrogen release
- with temporary temperature escalation and high H_2 production

Conditions for successful reflooding:

- No melt formation and flooding rate > 1 g/s.rod

W. Hering, 2007, https://doi.org/10.1016/j.nucengdes.2007.04.017
M. Steinbrück, 2010, https://doi.org/10.1016/j.nucengdes.2010.03.021

ATF cladding research at QUENCH

- HT behavior of all promising cladding materials
  - Cr-coated Zry, FeCrAl, SiC-based composites
- Separate-effects tests and bundle experiments
- Model development, code validation and code application
- Embedded in international cooperation programs
- and further international collaboration, e.g., with CEA, ASNR (France), Westinghouse (USA) and SNU (South Korea).

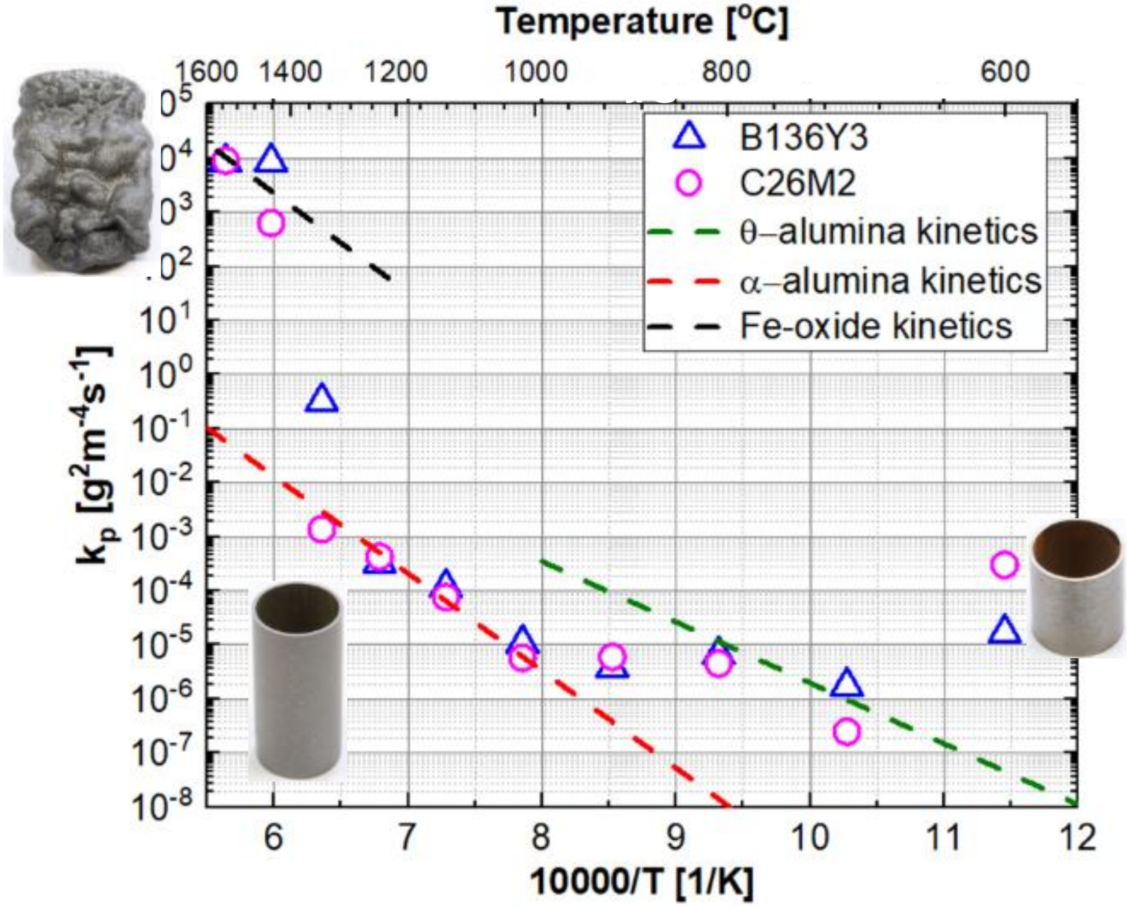
(courtesy of J. Stuckert, M. Steinbrück)

Source: https://de.wikipedia.org/wiki/Datei:Eugene_F._Kranz_at_his_console_at_the_NASA_Mission_Control_Center.jpg

QUENCH SETs on ATF

(courtesy of J. Stuckert, M. Steinbrück)

Oxidation kinetics of nuclear grade FeCrAl (600-1500 °C)



Steam oxidation kinetics of nuclear-grade FeCrAl alloys with kinetics of θ- and α-alumina and Fe-oxide

C. Kim, et al., 2021, https://publikationen.bibliothek.kit.edu/1000141509

- 700°C < T < 900°C:
  Transient oxide kinetics

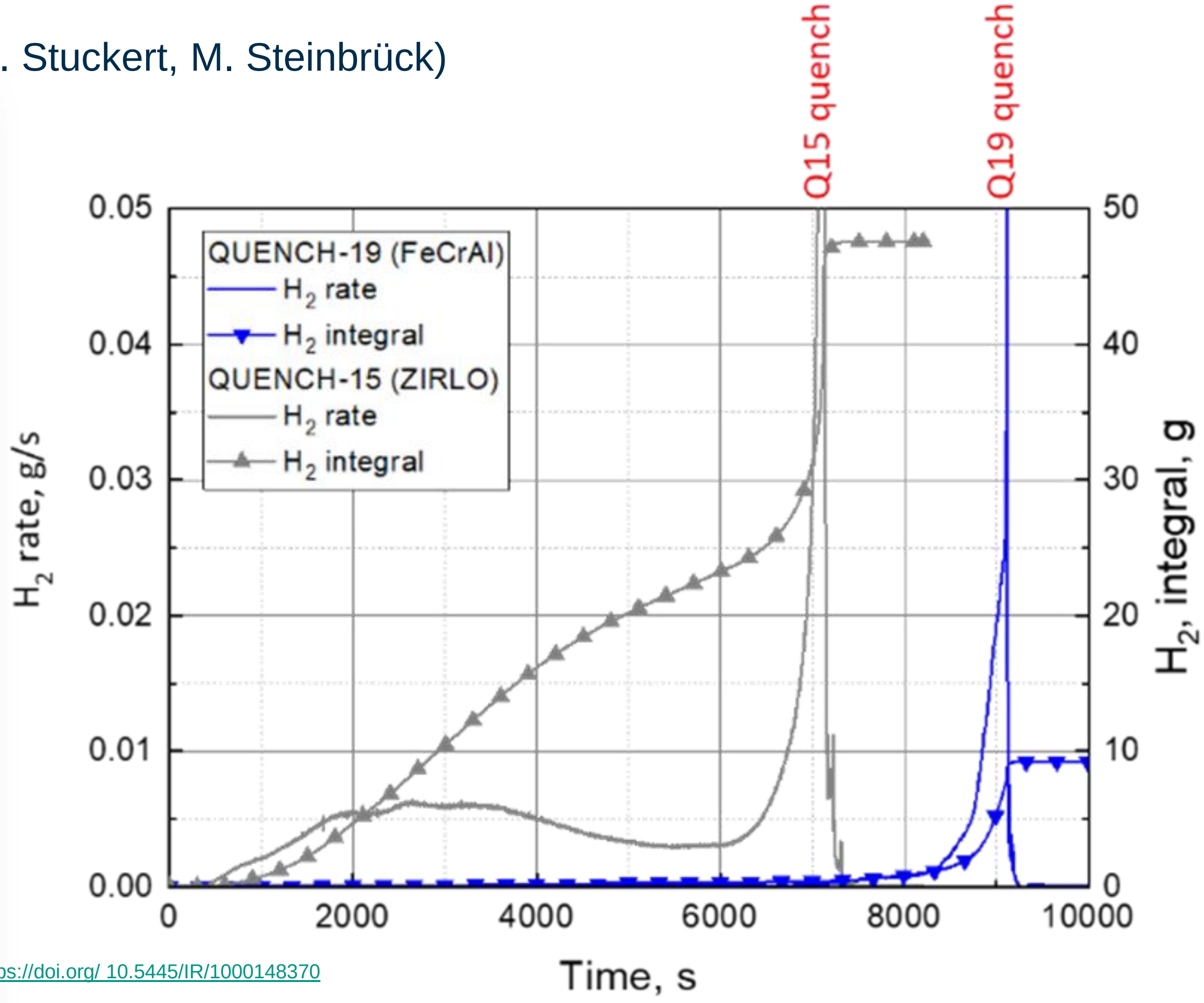
- 1000°C < T < 1350°C:
  α-alumina kinetics (2-3 orders of magnitude lower than of Zry oxidation)

- T ≥ 1400°C:
  Catastrophic oxidation with kinetics similar to Fe oxidation

- **Correlations evaluated and provided for SA codes**

QUENCH-19 bundle test with FeCrAl cladding
Worldwide first bundle test with ATF cladding

(courtesy of J. Stuckert, M. Steinbrück)



post-test appearance

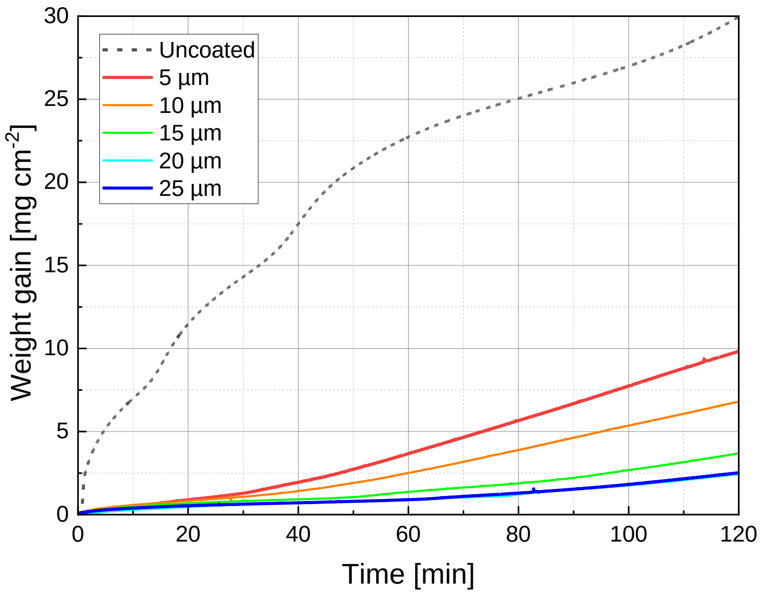
J. Stuckert, et al., 2022, https://doi.org/ 10.5445/IR/1000148370

QUENCH SETs on ATF

(courtesy of J. Stuckert, M. Steinbrück)

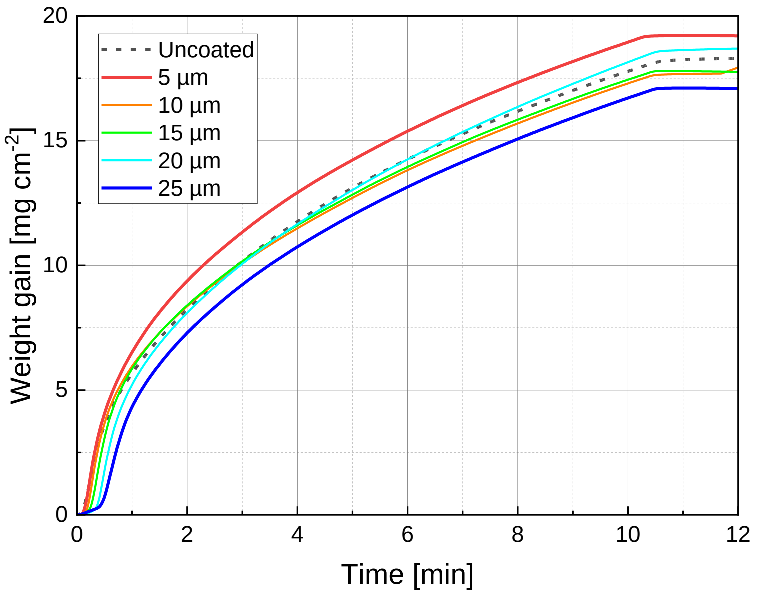
HT oxidation of Cr-coated Zry with varying thickness

1000°C



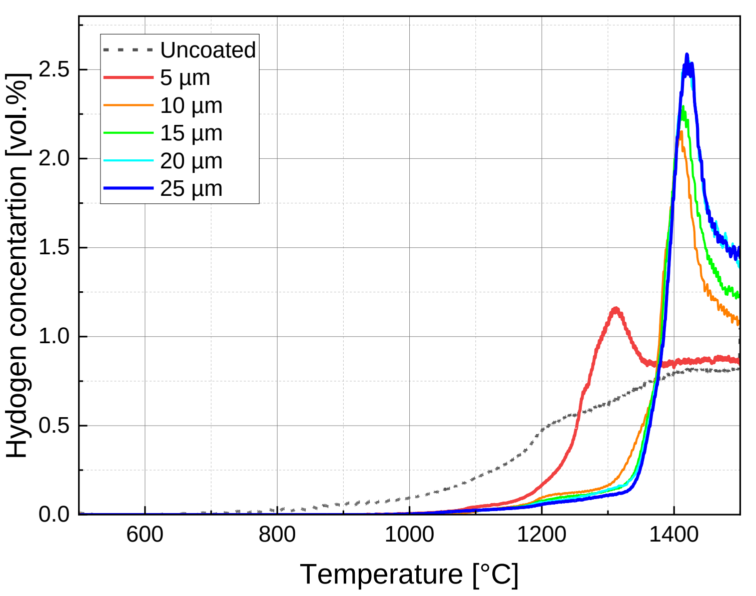
Protective effect of Cr coating below Zr-Cr eutectic temperature (1330°C)

1400°C



No protective effect of Cr coating above Zr-Cr eutectic temperature

500-1500°C



Loss of protective effect of Cr coating at Zr-Cr eutectic temperature

OECD-NEA Joint Undertaking QUENCH-ATF

- KIT as Operating Agent
- Three QUENCH bundle experiments with Cr-coated Zr alloy cladding tubes
  1. CS Cr/ZIRLO (Westinghouse) with slightly beyond DBA LOCA scenario (T_{max}=1300°C) – completed *July 2022*
  2. CS Cr/ZIRLO (Westinghouse) with BDBA scenario (T_{max}=1600°C) – completed *July 2024*
  3. PVD Cr/ZIRLO (WH/NNL) – scenario: modified QUENCH-11 with boil-off phase – *under preparation*
- Benchmark exercises for each test (coordinated by GRS)
- Extensive post-test analyses (KIT, ASNR, CRIEPI) and supporting SETs
- **Phase-2 of the project under preparation**

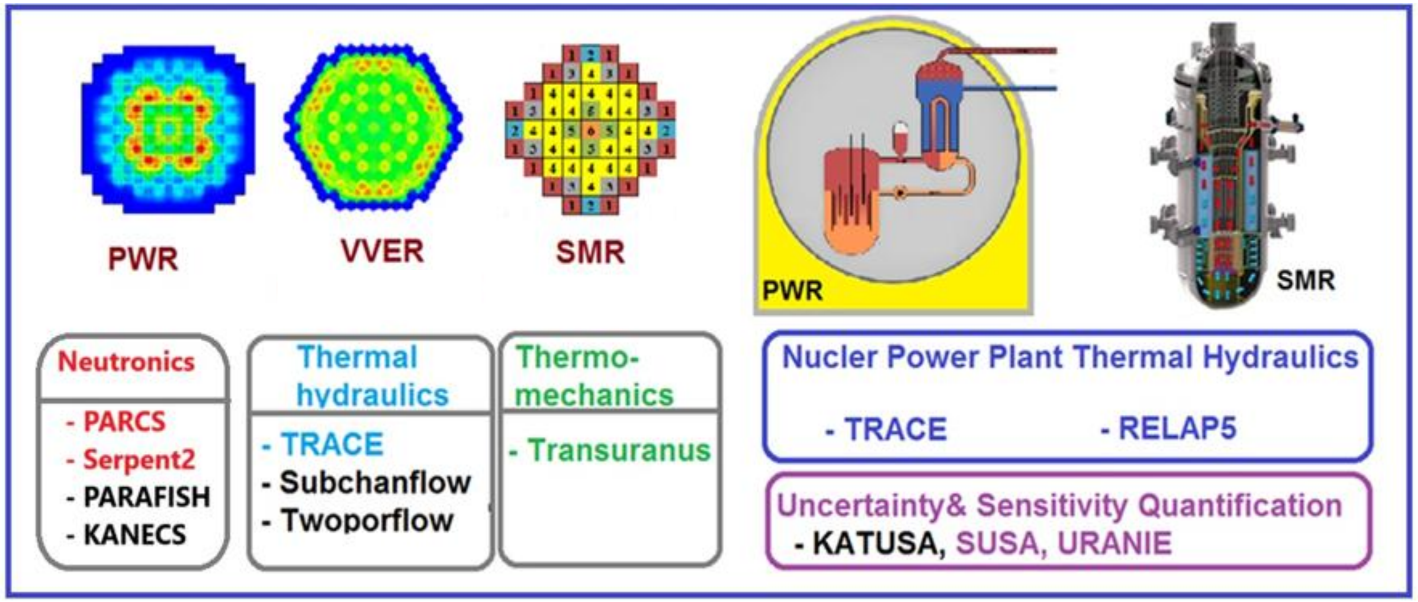
QUENCH-ATF 1 bundle

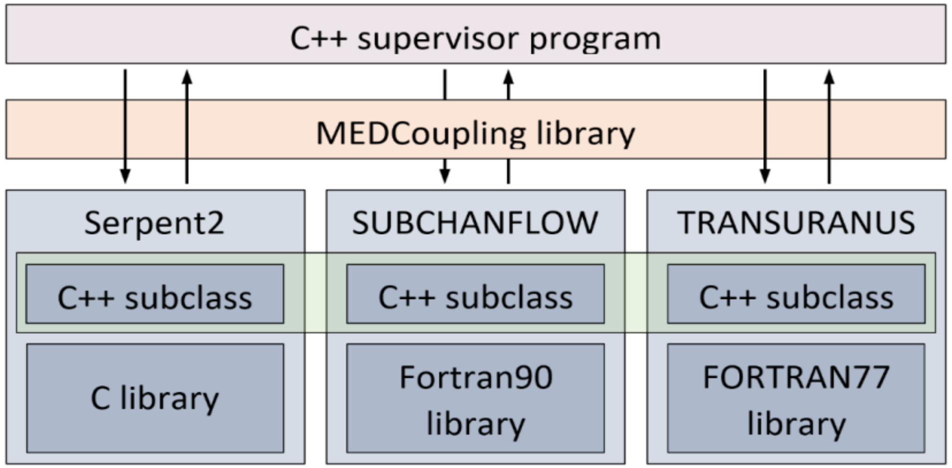
KIT Multi-Physics and -Scale Methods for Safety Assessment

Mission and Strategy

- Improve **core analyses** by developing **multi-physics coupled codes** combining neutronics, fuel thermo-mechanics, and thermal hydraulics
- Enhance **plant analyses** by developing **multi-scale coupling** of thermal hydraulic codes of different spatial resolution

Reference solutions for pin-by-pin/sub-channel core analyses.





KIT developed codes: PARAFISH, KANECS, SUBCHANFLOW, TWOPORFLOW, KATUSA

V.H. Sanchez-Espinoza, et al., 2023, https://doi.org/10.1016/j.nucengdes.2023.112573

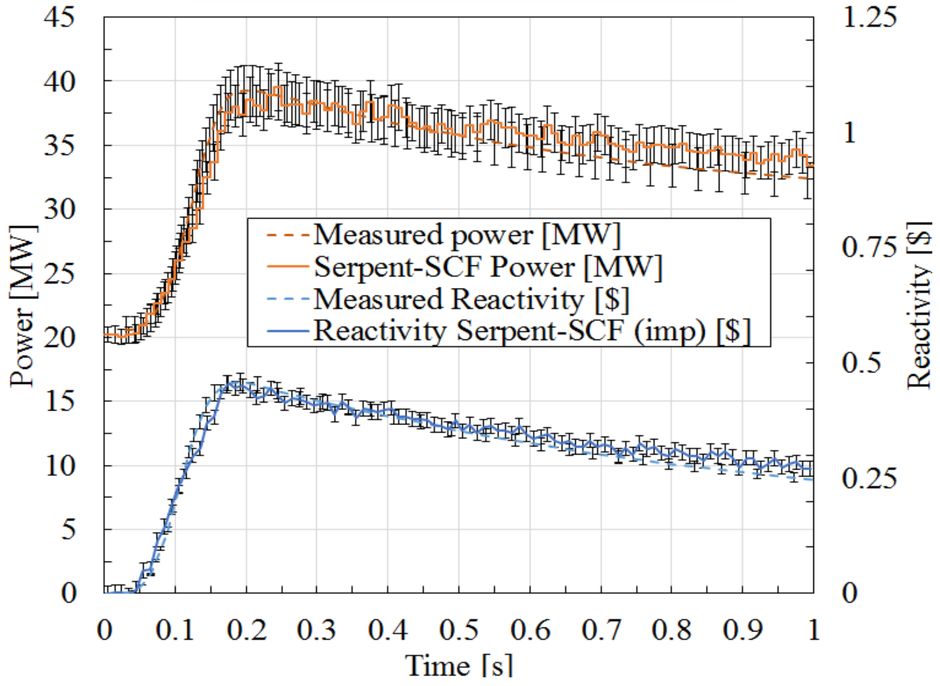
Highlight: High-fidelity Monte Carlo-based Tools

Validation of SERPENT2/SUBCHANFLOW

Validation against reference experiments with different fuel types (rods, plates).

- SPERT-IV Rod Ejection Accident Tests 84

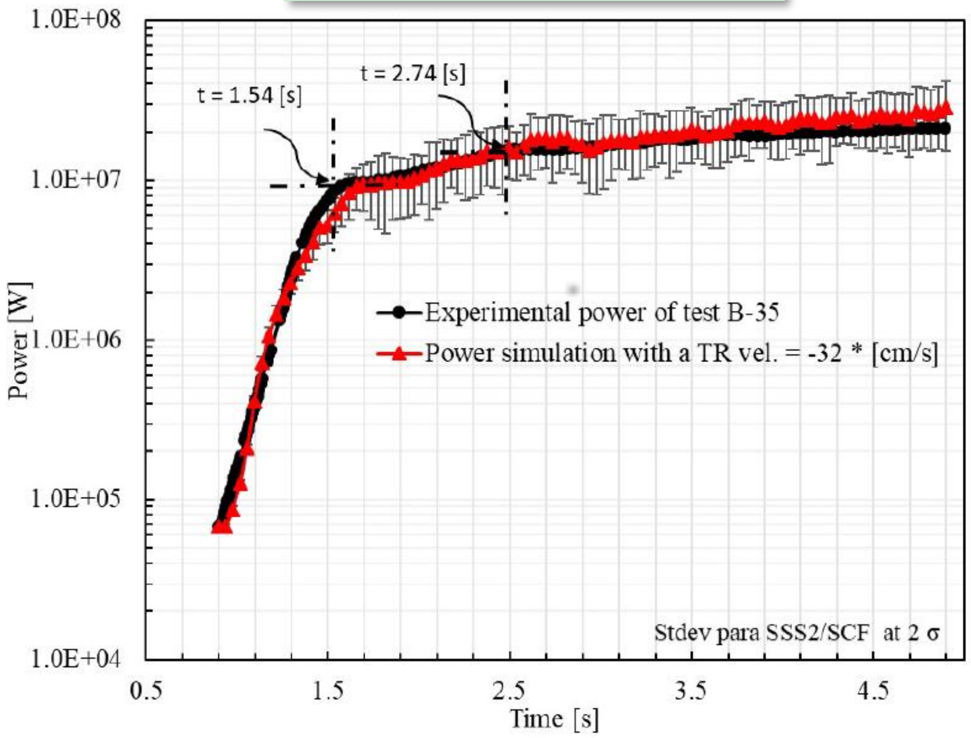
Cylindrical fuel rods



D. Ferraro, et al., 2020, https://doi.org/10.1016/j.anucene.2020.107387

- SPERT-IV D12-25 Rod Ejection Accident Tests B-35

Fuel plates



J. C. Almachi Nacimba, 2024, https://publikationen.bibliothek.kit.edu/1000169070

Reactor Physics and Dynamic

Highlight: High-fidelity Monte Carlo-based Tools

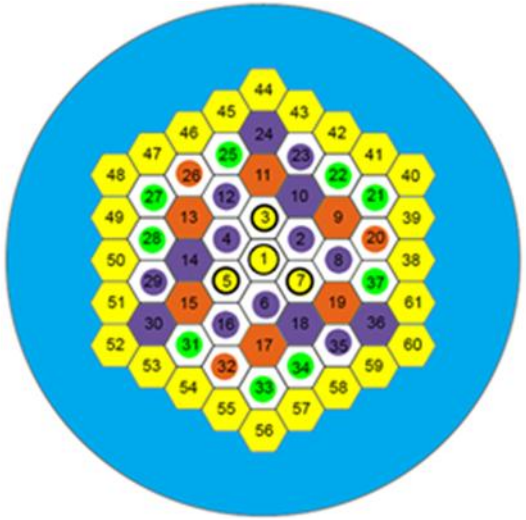
Transient analyses of a SMR-core

First-of-its-kind analysis of a long overcooling transient (50 s).

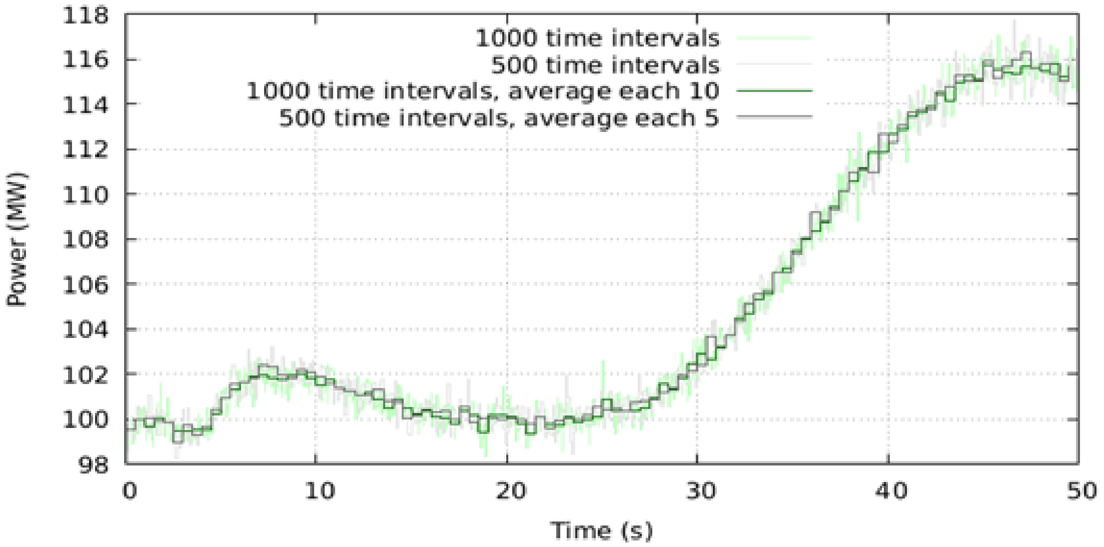
- SERPENT2/SUBCHANFLOW (developed by KIT and VTT)
- **3D** static, depletion, and **transient analyses** of cores at pin/subchannel level for the **prediction of local safety parameters**

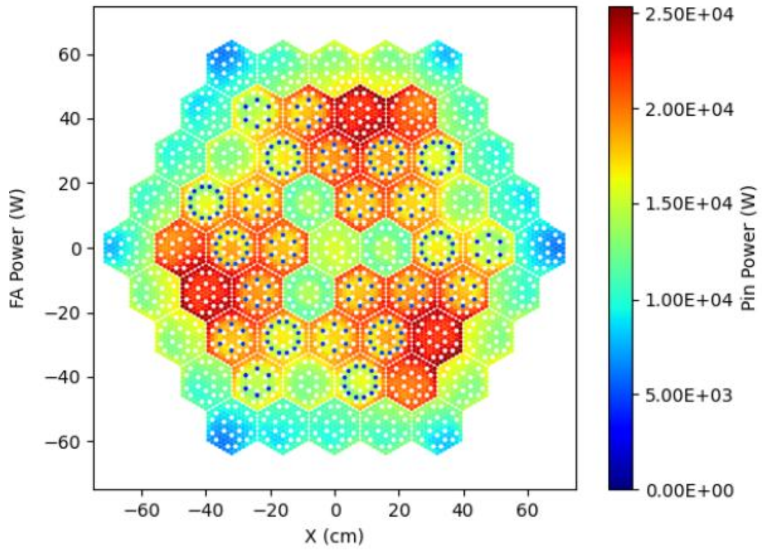
Access to the KIT High Performance Computer Karlsruhe (HoreKA)

CAREM SMR-Like core



Cold Water Transient Analysis





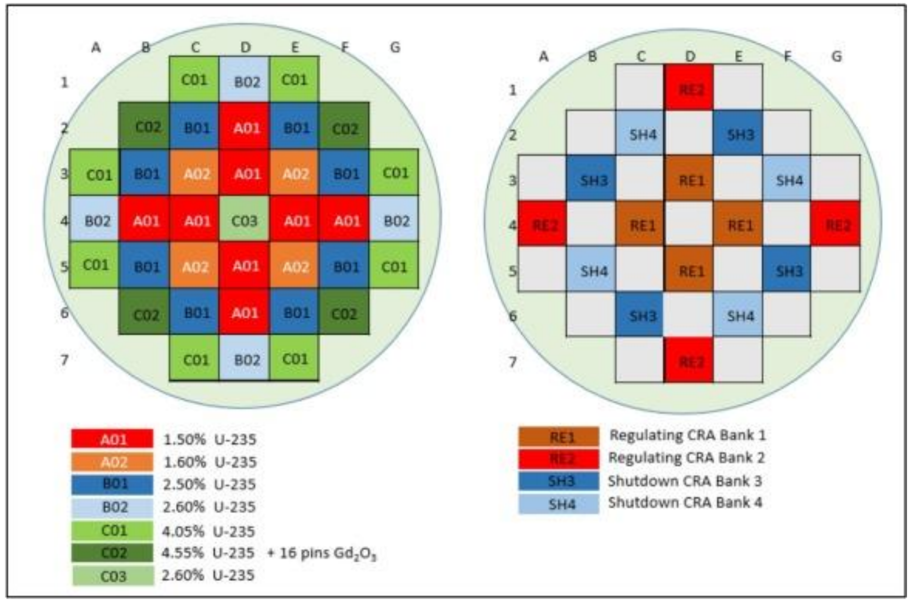
V. H. Sanchez Espinoza, at al., 2022, Int. Conf. on Topical Issues in Nuclear Installation Safety, Vienna, Austria, 18-21 October

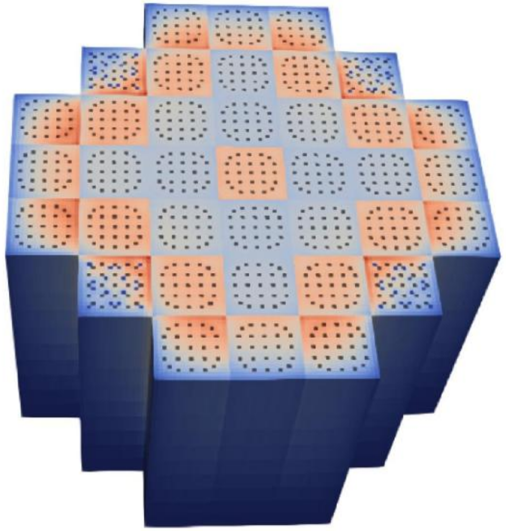
Reactor Physics and Dynamic

Highlight: High-fidelity Monte Carlo-based Tools

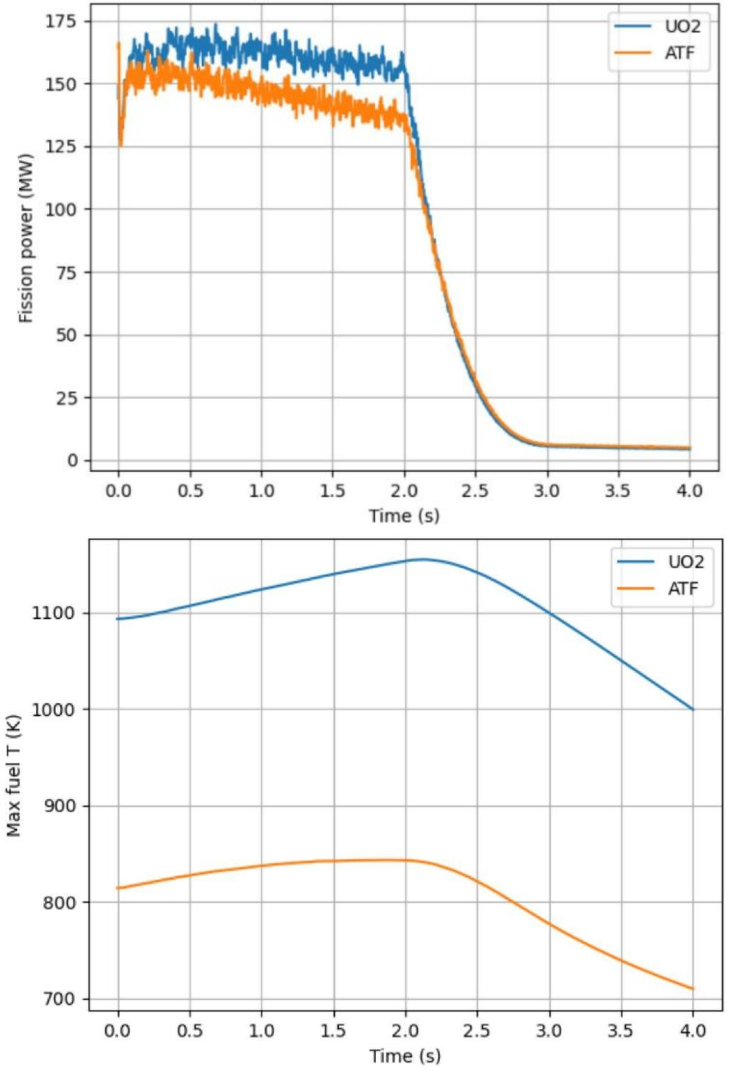
High Fidelity – NuScale REA with N, TH, TM

- SERPENT/SUBCHANFLOW/TRANSURANUS
- UO_2 and U_3Si_2 (FeCrAl cladding)





REA1: UO_2 - ATF



Z. Soti, 2025, https://doi.org/10.1016/j.nucengdes.2025.114183

Highlight: Multi-scale Analysis of Plant Transients
3D Analysis of Steam Line Break of a SMR

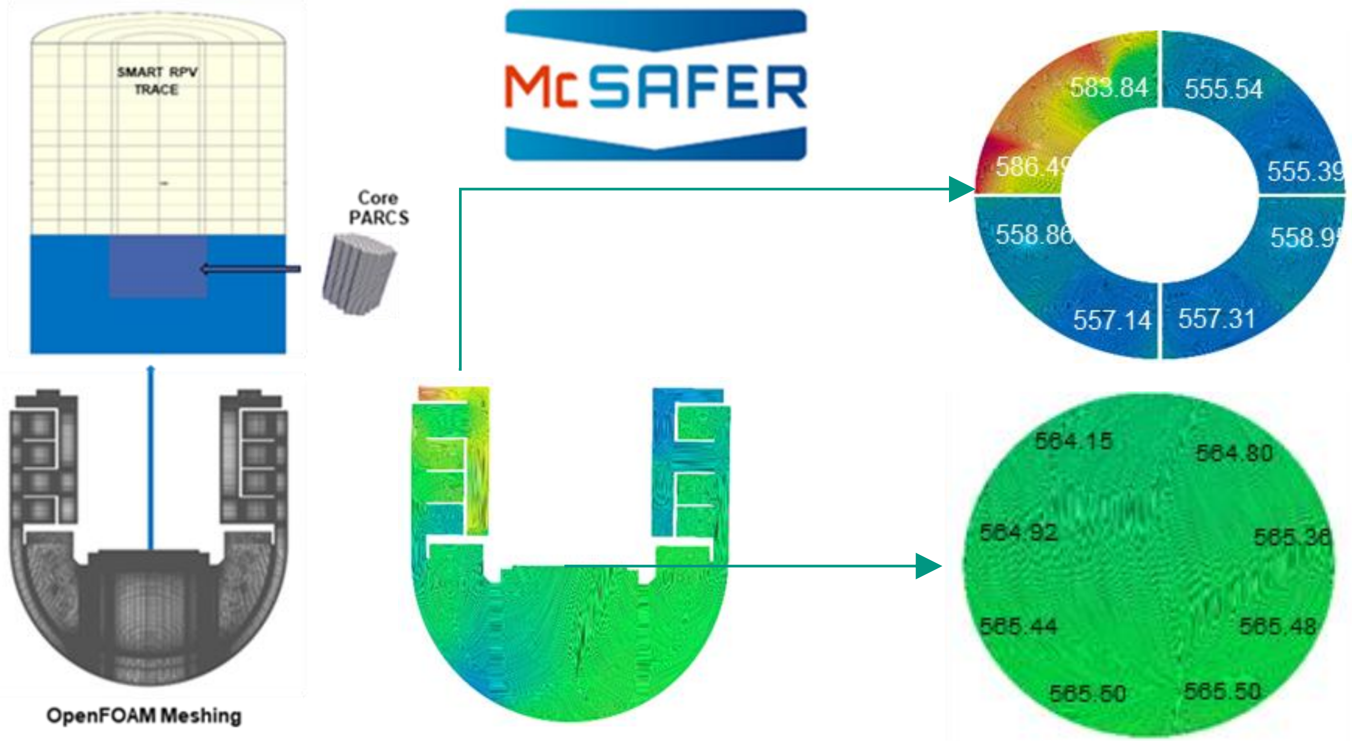
- Coupling Neutronics/Thermal hydraulics/CFD codes
- SMR SMART Plant Analysis - Steam Line Break on one loop

Moving beyond the 1D/2D approach.

TRACE

PARCS

OpenFOAM



OpenFOAM Meshing

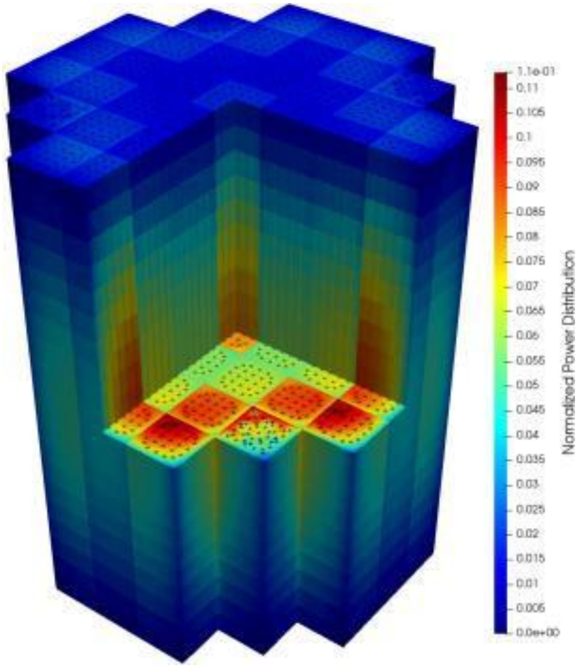
3D modelling necessary to simulate reactor transients induced by asymmetrical initiators.

Reliable prediction of local temperatures and velocity distribution in the downcomer and in the lower plenum.

Reactor Physics and Dynamic

KIT Multi-Physics and -Scale Methods for Safety Assessment
KArlsruhe NEutronics Core Simulator (KANECS)

- Cartesian, Hexagonal, Unstructured Meshes
- Simplified spherical harmonic (SPN) method {1,3,5,7}
- Multi-group approximation
- Continuous Galerkin (CG) Finite-element
- Steady-state + Time-dependent calculations



- NUSCALE: KANECS vs. Serpent2 benchmark

Discretization	Code	Keff (ARO)	pcm	Max. ΔP	Avg. ΔP
Pin	Serpent2	1.03046	-	-	-
Pin	KANECS (SP3)	1.03084	38	2.86%	1.08%
Fuel Assembly	KANECS (SP3)	1.03170	124	3.07%	1.35%

J. Duran, et al., 2025, https://doi.org/10.3389/fenrg.2025.1498331

KIT Strategy for Severe Accident Analyses
Mission and Strategy

Realistic estimation of the source term and **dispersion of fission products** (radiological risk) into the environment **in large and small nuclear reactors**

Use of **state-of-the-art codes** at each simulation stage.

Going beyond: Development, extension, and validation of integral codes by using the KIT experiments.

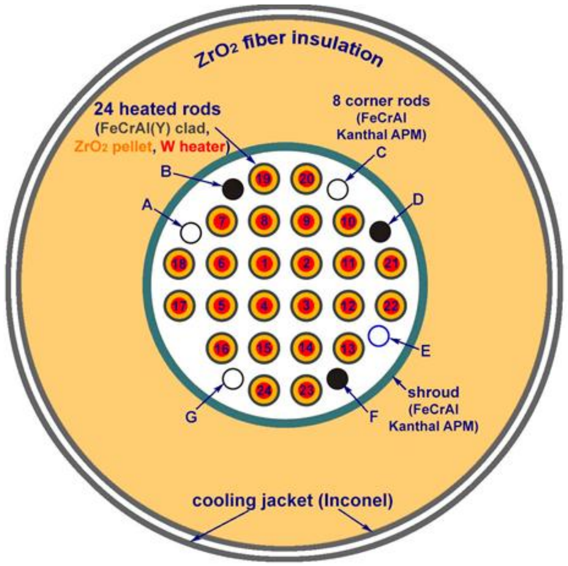
Risk estimation requires the evaluation of...

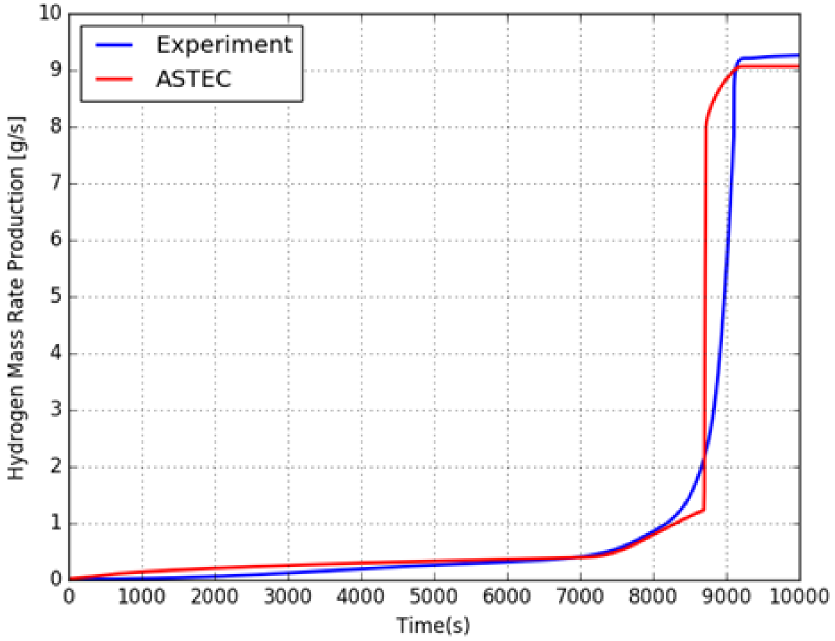
- **Fuel inventory and decay heat**
  - Neutronic codes (CASMO, SCALE)
- **Severe accident progression**
  - Integral codes (ASTEC, MELCOR)
- **Uncertainty quantification of the source term**
  - KATUSA tool (KIT development)
- **Fission product dispersion in the environment**
  - JRODOS (KIT development)
- **…while, improving the integral codes‘ performance**
  - AI/ML algorithms (under development in ASSAS)

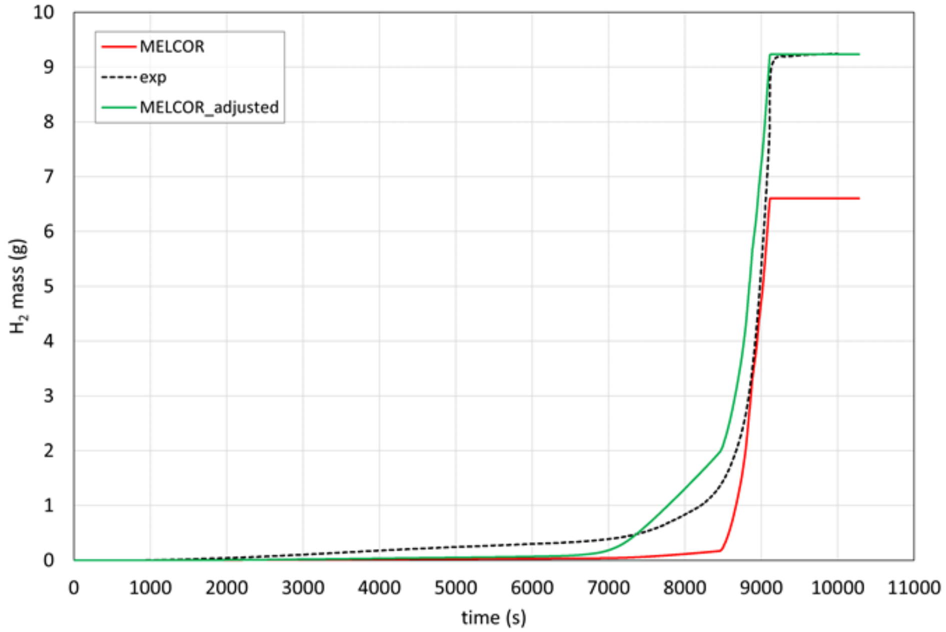
V.H. Sanchez-Espinoza, et al., 2023, https://doi.org/10.1016/j.nucengdes.2023.112573

Validation of Severe Accident Codes
ASTEC and MELCOR validation against QUENCH-19

- **QUENCH Experimental data employed**
  - **Separate-effects tests** for the development of physical models
    - Provision of correlations for cladding oxidation
  - **Bundle-tests** for the **validation of integral codes** in representative conditions







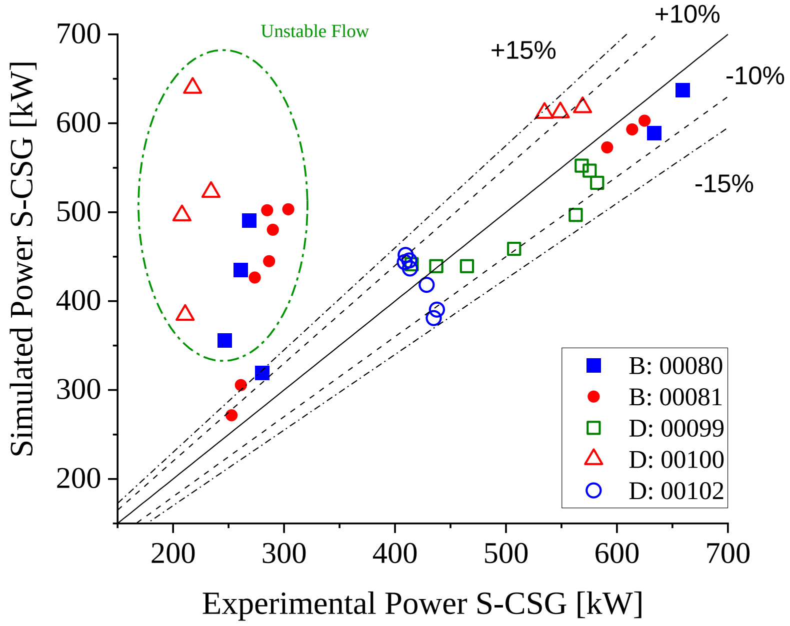
Z. I. Jimenez Balbuena, ERMSAR2026
M. E. Cazado, 2026, International Webinars on Nuclear Energy
F. Rosi,, et al., 2026, ERMSAR2026
F. Gabrielli, et al., 2023, https://publikationen.bibliothek.kit.edu/1000165938

(courtesy of M. E. Cazado)

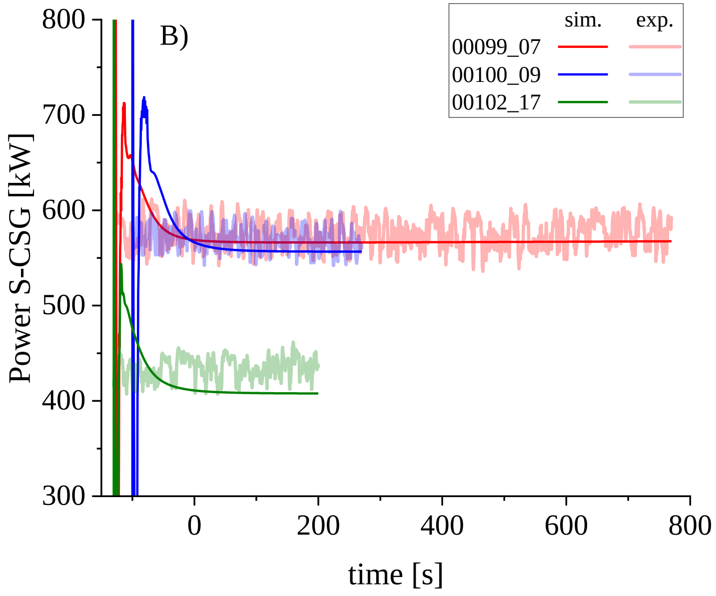
Validation of Severe Accident Codes
External experimental data and integral codes' validation

- ELSMOR experimental facility: testing and validating the passive decay heat removal systems of Small Modular Reactors (SMRs)
- Compact Steam Generator (CSG) to simulate a reactor's primary circuit and a natural circulation loop to remove decay heat into a water pool

S-CSG Power Exchange: ASTEC vs. Experiment



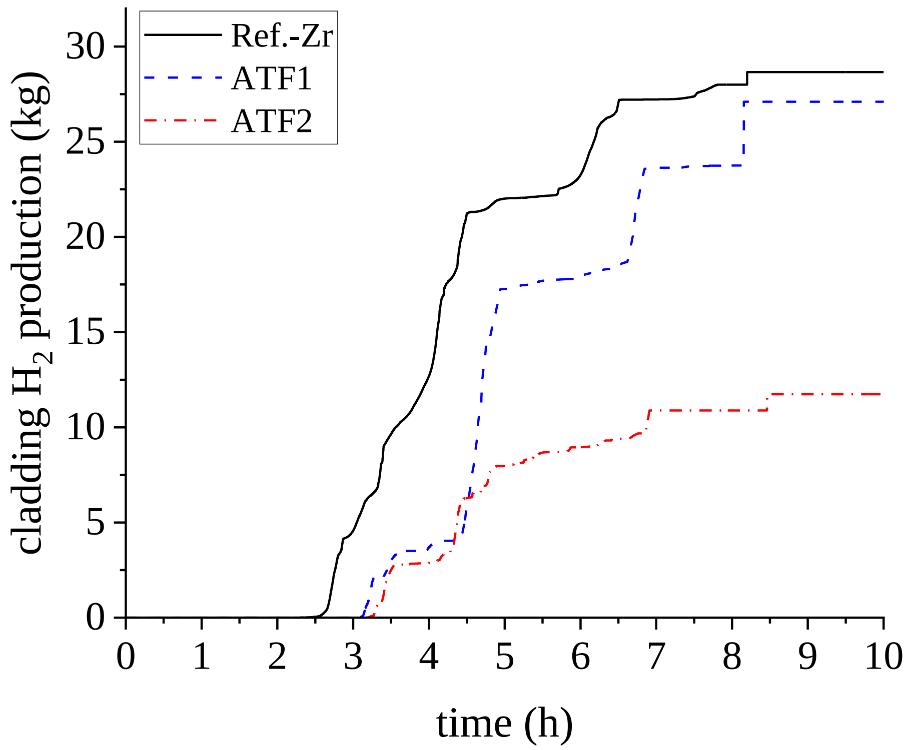
S-CSG Power Exchange: ASTEC vs. Exp.



Severe Accident in SMRs with ASTEC
Highlight: Employment of ATFs in a generic SMR

(courtesy of M. E. Cazado)

- QUENCH experimental results employed
- Scenario: 100% break on the CVCS makeup line + Loss of AC + failure of 2 RRVs (ASTEC)



- **ATF-1**
  - Oxidation correlations only
- **ATF-2**
  - Oxidation correlation + physical properties

- **'Reliable' results up to the clad break**
  - Modelling after this event under assessment

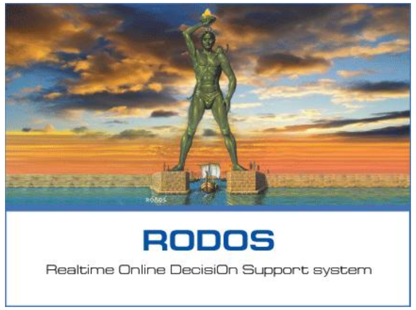
M. E. Cazado, 2025, Impact of Advanced Technology Fuel Cladding Materials on the Progression of Severe Accidents in a Generic Natural-Circulation iPWR, NURETH-21

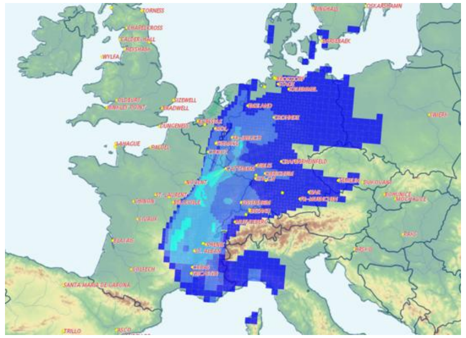
Reactor Physics and Dynamic

Calculation Platform for SA Analyses

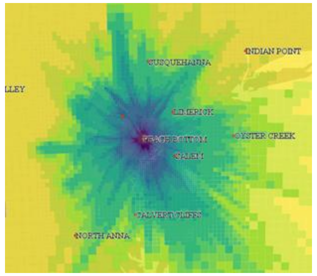
CASMO

KATUSA



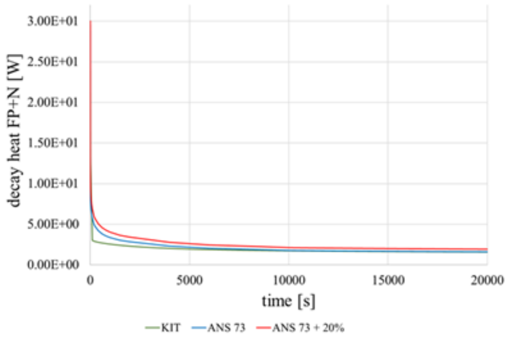


Statistical Analysis

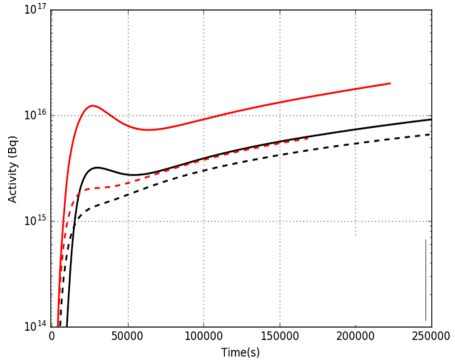


Risk Map

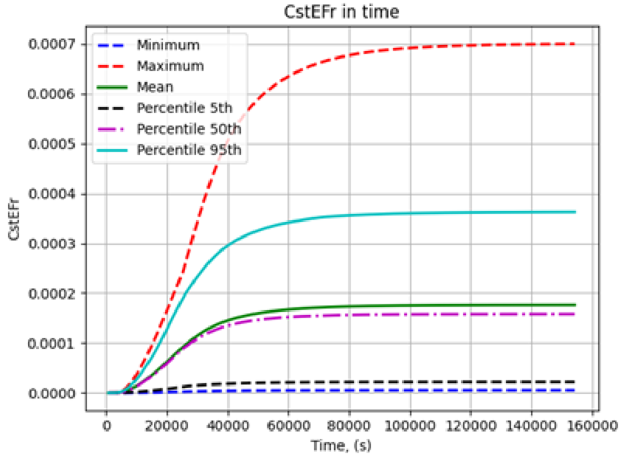
- Fuel inventory
- Decay heat



- Accident progression
- Source term



- Uncertainty and sensitivity analyses of the source term



- Environmental Impact Assessment
- Emergency Response and Preparedeness

(courtesy of O. Murat)

O. Murat, et al., 2025, https://doi.org/10.1016/j.anucene.2025.111277
O. Murat, et al., 2023, https://doi.org/10.1016/j.nucengdes.2023.112227
A. Mercan, et al., 2022, https://doi.org/10.1016/j.nucengdes.2022.112078

Reactor Physics and Dynamic

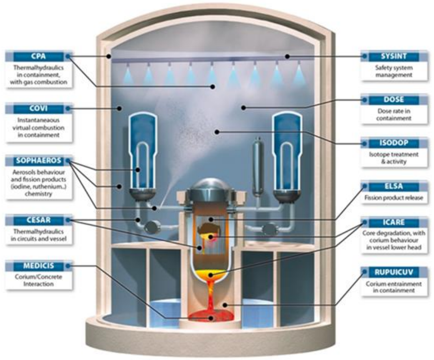
KIT Computational Chain for Radiological Impact Prediction

ASTEC and JRODOS coupling

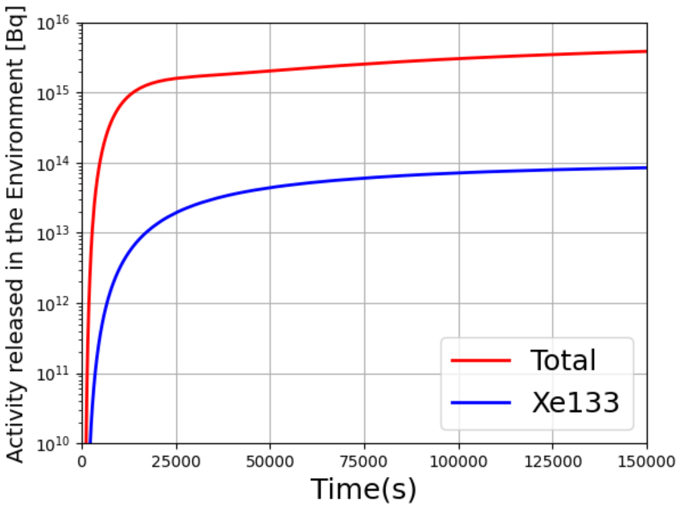
- **Unique once-through approach** for the estimation of the radiological impact of hypothetical severe accidents

First-of-its-kind: Complete calculation chain from fuel inventory to the dispersion of the fission products in the environment **by using state-of-the-art codes.**

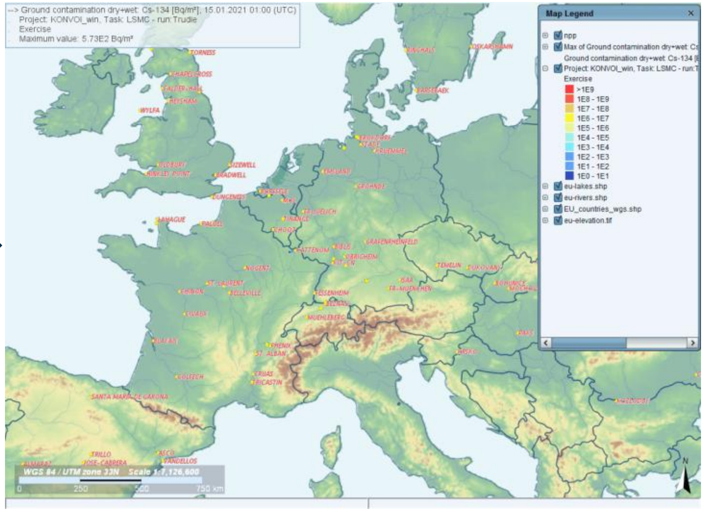
ASTEC



Source Term
Isotope-wise activity release



Cloud arrival time [hours]



O. Murat, et al., 2023, https://doi.org/10.1016/j.nucengdes.2023.112227
A. Mercan, et al., 2022, https://doi.org/10.1016/j.nucengdes.2022.112078

Reactor Physics and Dynamic

JRODOS: Java Real-time On-line Decision Support System

KIT developed tool for radiological impact prediction

(courtesy of W. Raskob, S. S. Ottenburger)

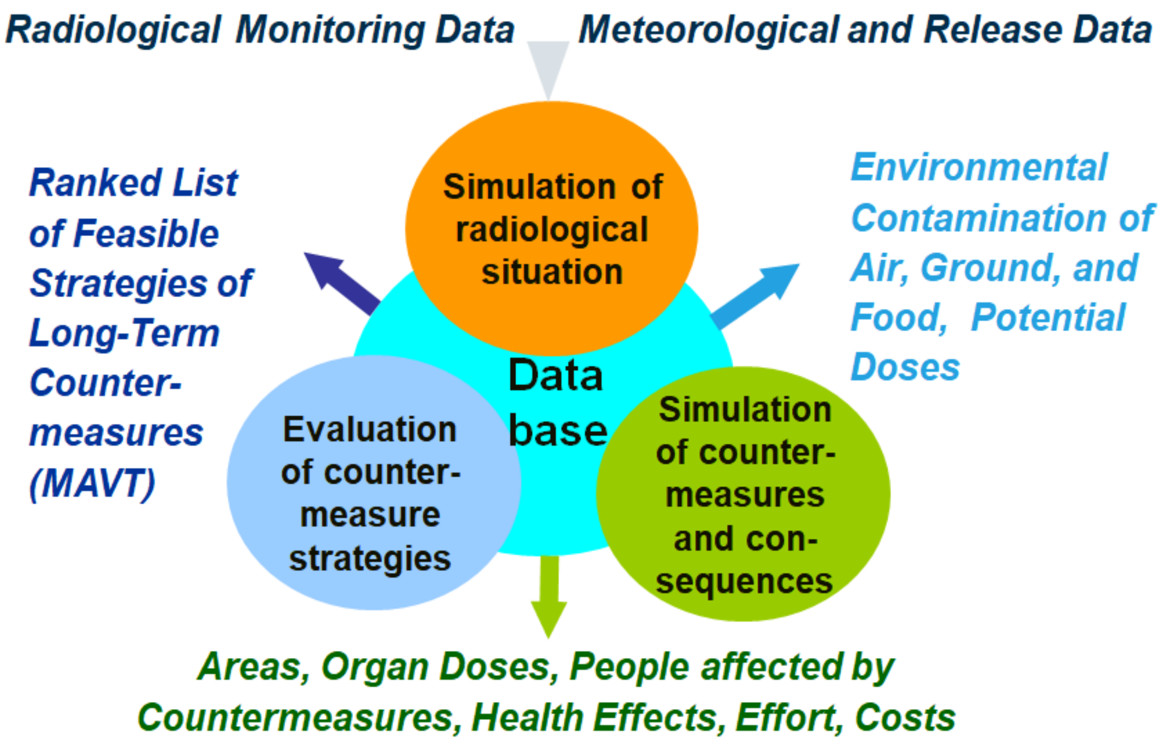
- Simulation of **radionuclide release and environmental impact on fall out zone in early and later phase of an accident**

Enhancing emergency planning capabilities for decision-maker on **local/national/international scales** for all **relevant emergency actions and countermeasures**

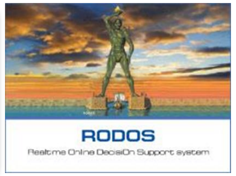
One or several users in about 50 countries worldwide (governmental authorities, local communities, research centers, universities, NPP operators, etc.).

W. Raskob, 2012, https://doi.org/10.1051/radiopro/20116865s

RODOS simulation procedure



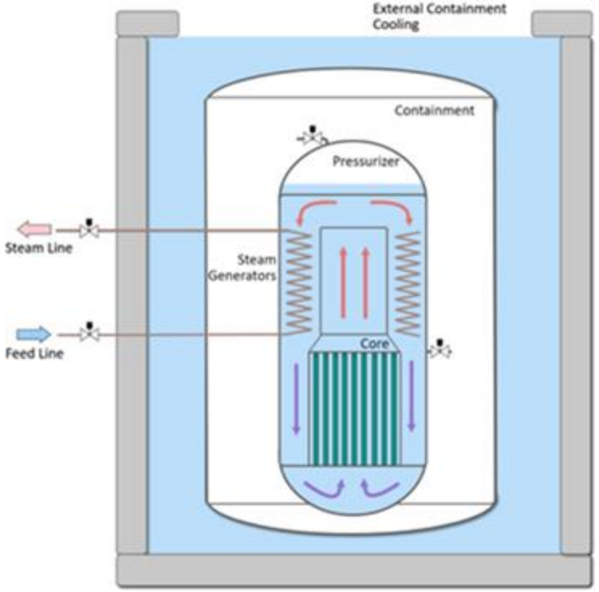
Presently simulating every hour scenarios for four NPPs in Ukraine.

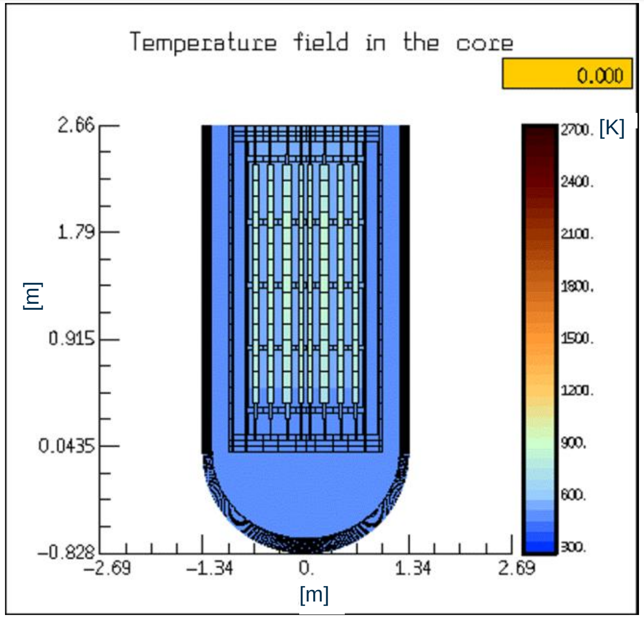


Severe Accidents Analysis & AI Methods for Reactor Safety
Highlight: SMR Analysis & EU ASSAS project

- KIT as Working Package leader

Estimation of the radiological risk after postulated severe accidents in a water cooled SMR.





F. Gabrielli, et al., 2024, https://publikationen.bibliothek.kit.edu/1000174165/v2

- Building a SA simulator based on the ASTEC code

- Large simulation campaign is ongoing at KIT (about 1,000 scenarios up to now)
- Strong collaboration with the KIT Scientific Computing Center a key for success!

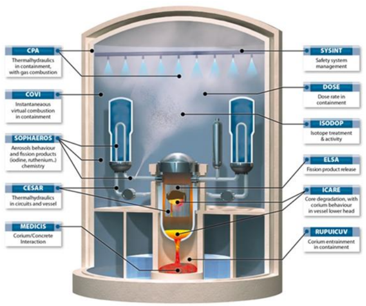
- ASSAS KIT Hub assessed to host the ASTEC database for training AI
- Use of High Performance Computer Karlsruhe (HoreKA) crucial!

KIT Radiological Source Term Prediction
Uncertainty analysis of the source term

- Statistical analysis of the radiological risk

MUSA

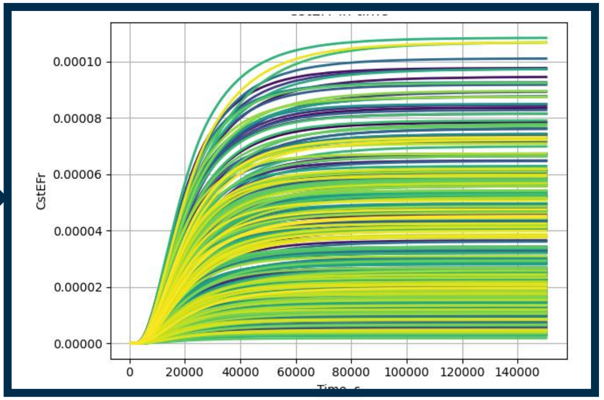
ASTEC

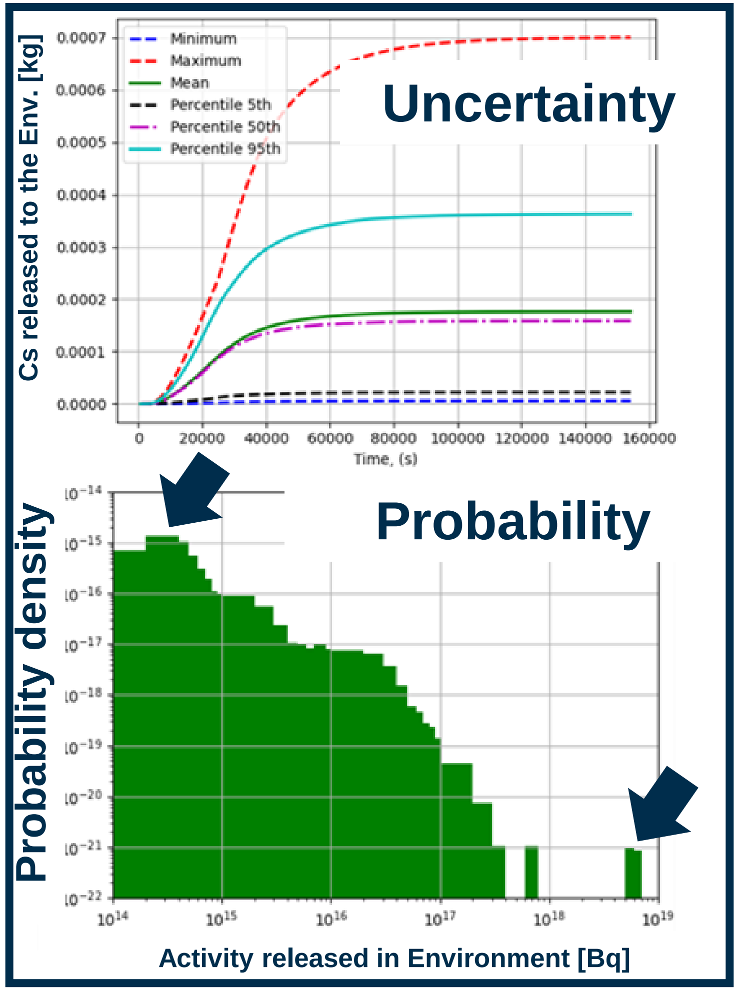


KATUSA

MC sampling of uncertainties, operator actions, accident initiators

Source Term Database





- **Key information for Emergency & Preparedness Response**
  - **Uncertainty** of the **timing** and the **amount** of the release
  - **Identification of the most severe** accident scenario based on the **probability** of the **radiological risk**.

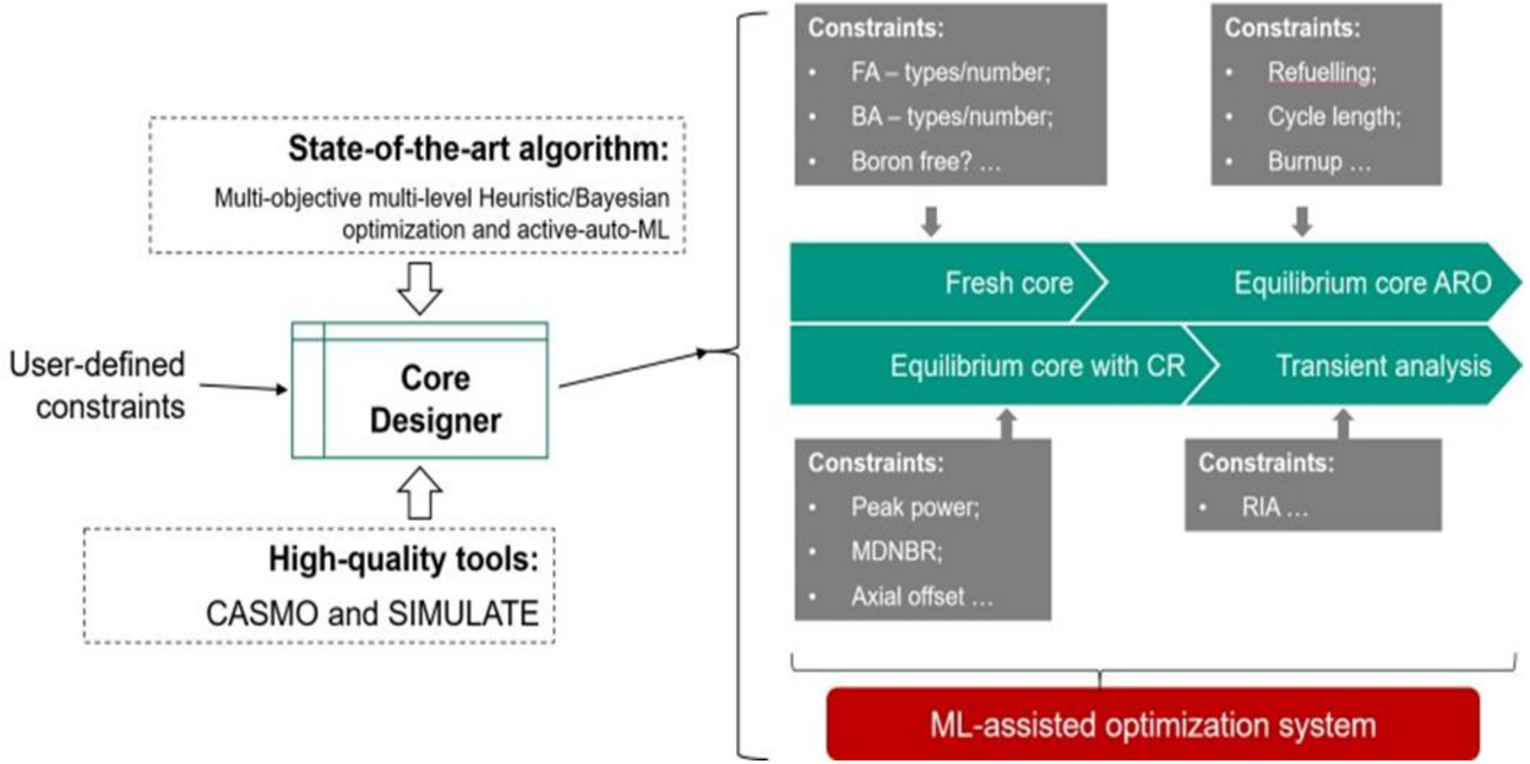
A. Stakhanova, et al., 2023, https://doi.org/10.1016/j.anucene.2023.109964
A. Mercan, et al., 2022, https://doi.org/10.1016/j.nucengdes.2022.112078

Multiphysics: AI/ML methods

(courtesy of K. Zhang)

- **Design Optimization of SMR-Cores**
  - Core economy and safety enhancement
  - Potential for next-generation core optimal design system

- **Methods for reactor safety**
  - Prediction of safety margin
  - Develop core monitoring system

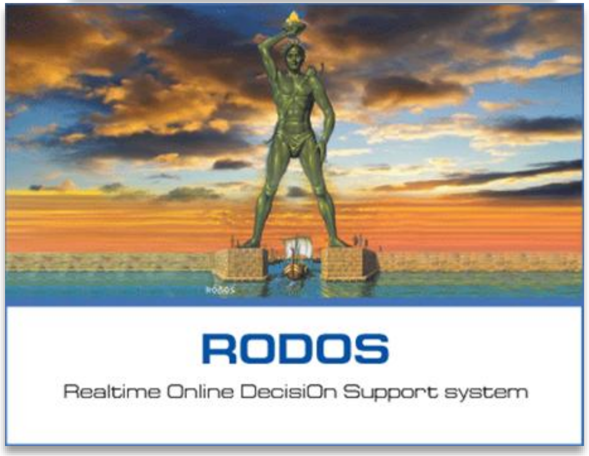


(courtesy of W. Raskob, S. S. Ottenburger)

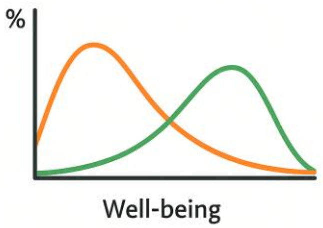
JRODOS: AI & Simulation for optimized measures

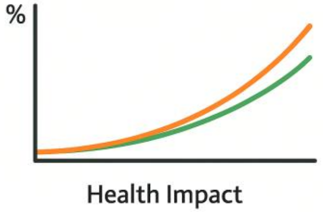
Towards AI-powered Decision Support

- **KIT ITES-RESIS** is building a **first prototype for AI-based decision support** for optimized preparedness measures by summer 2026, coupling **JRODOS** with **FRAMESS** to enable faster, uncertainty-aware assessments and more effective protective-action planning



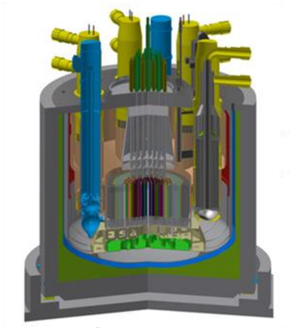
Assessment

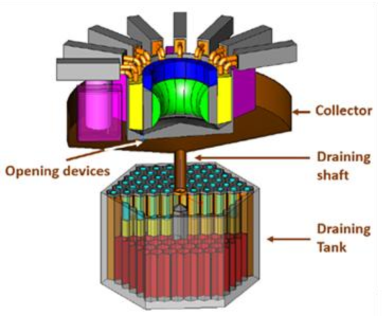




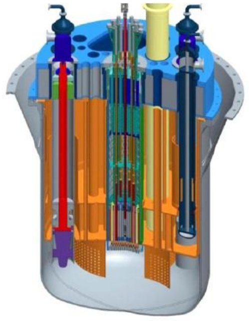
Advanced Nuclear Reactors’ Safety

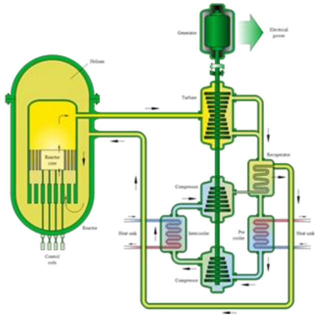
Large-Scale Facilities and Calculation Platform


SFR



MSR


MYRRHA


GFR

Liquid Metal Thermal Hydraulics and Materials Research
Mission and Strategy

(courtesy of K. Litfin, A. Weisenburger)

We provide essential benchmark and reference data for **CFD codes, system codes and licensing tools** used **for liquid metal cooled nuclear systems** to the **interna-tional scientific community.**

Comprehensive investigations of **liquid metal ...**

- **thermal hydraulics** in fuel bundles …
  - heat transfer
  - pressure losses
- **… with increasing compexity and fidelity …**
  - blockages, deformations
  - inter wrapper flow
- **… and including effects on structural material**
  - corrosion
  - degradation

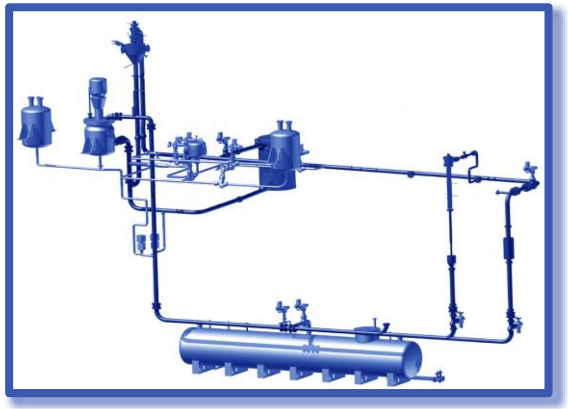
Liquid Metal Thermal Hydraulics and Materials Research

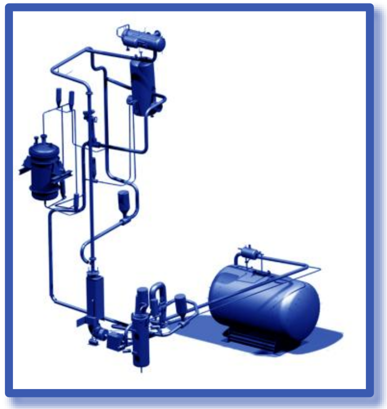
Strategic Foundations: Infrastructure and Scientific Program

(courtesy of K. Litfin, A. Weisenburger)

Loop facilities

- THEADES
- THESYS
- KASOLA

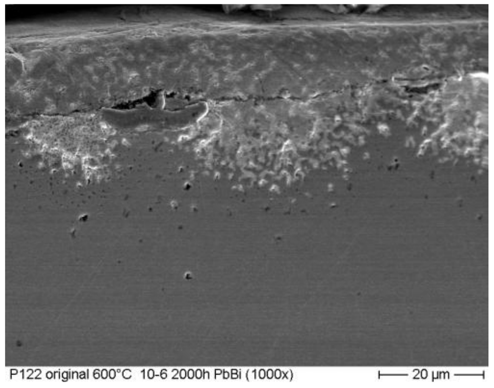




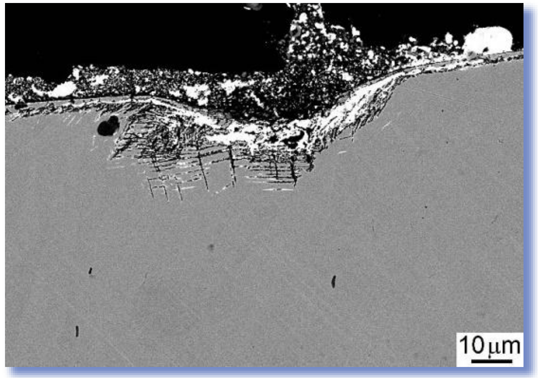
Worldwide uniqe facilities, leading contributor to the international liquid metal technology community.

Material test facilities

- Corrosion, erosion, degradation of mechanical properties
- Thermal stability and high temperature strength



P122 – 600°C
2000h

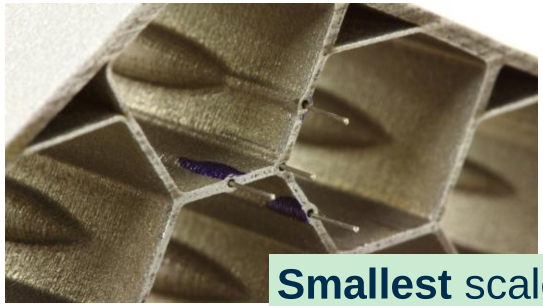


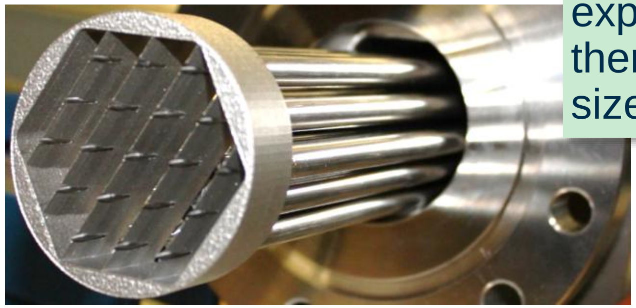
316l – 400°C
4000h

Example established Infrastructure Thermal Hydraulics
THEADES loop at Karlsruhe Liquid Metal Lab KALLA

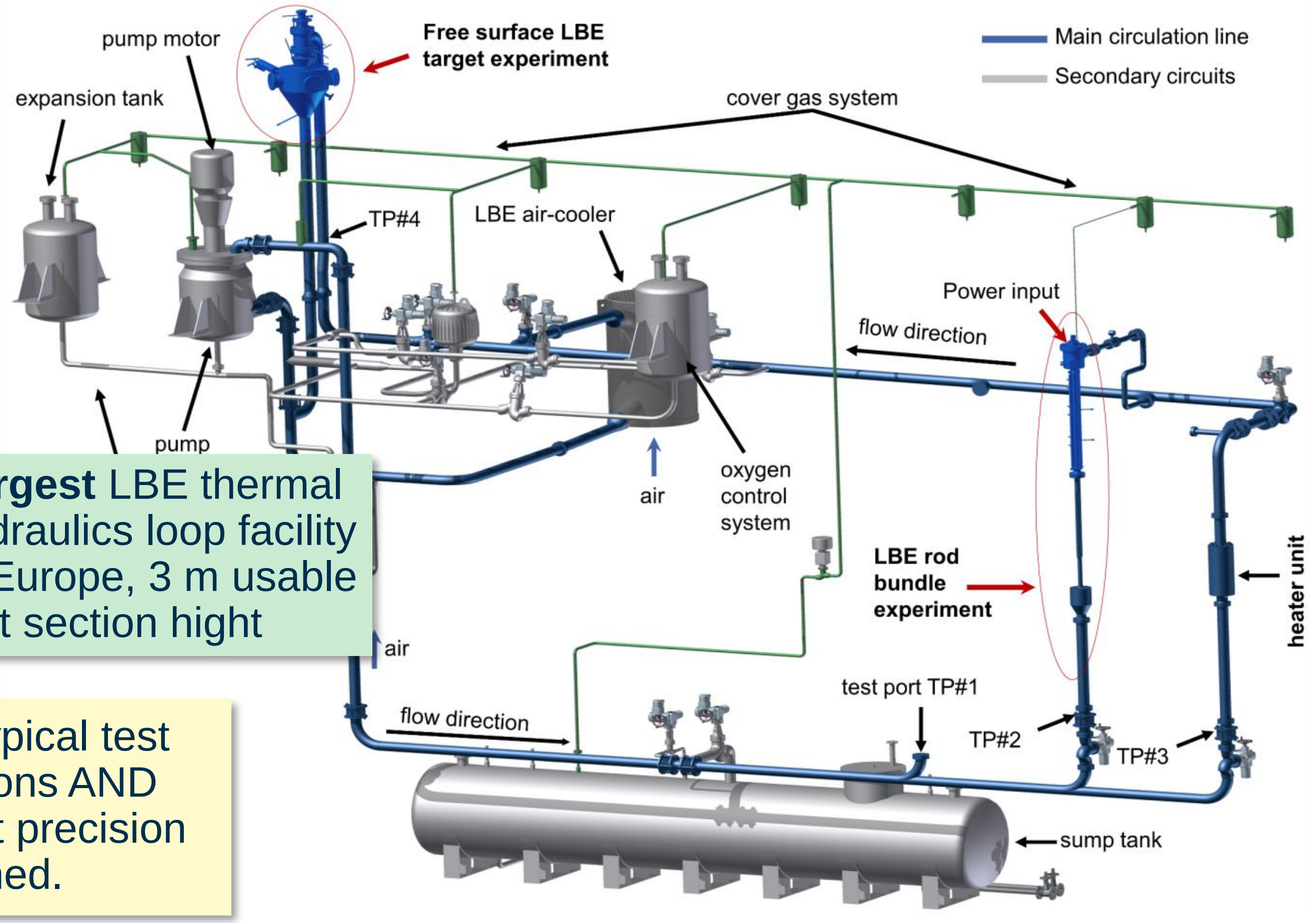
(courtesy of K. Litfin, A. Weisenburger)

- Inventory: 44 to Lead Bismuth Eutectic (LBE)
- Temperature: 180°C - 450°C
- Flow rate: 47 m³/h (136 kg/s) max.
- Heating power: 500 kW max.





Smallest scale in experimental setup: thermocouple tip size 0.25 mm



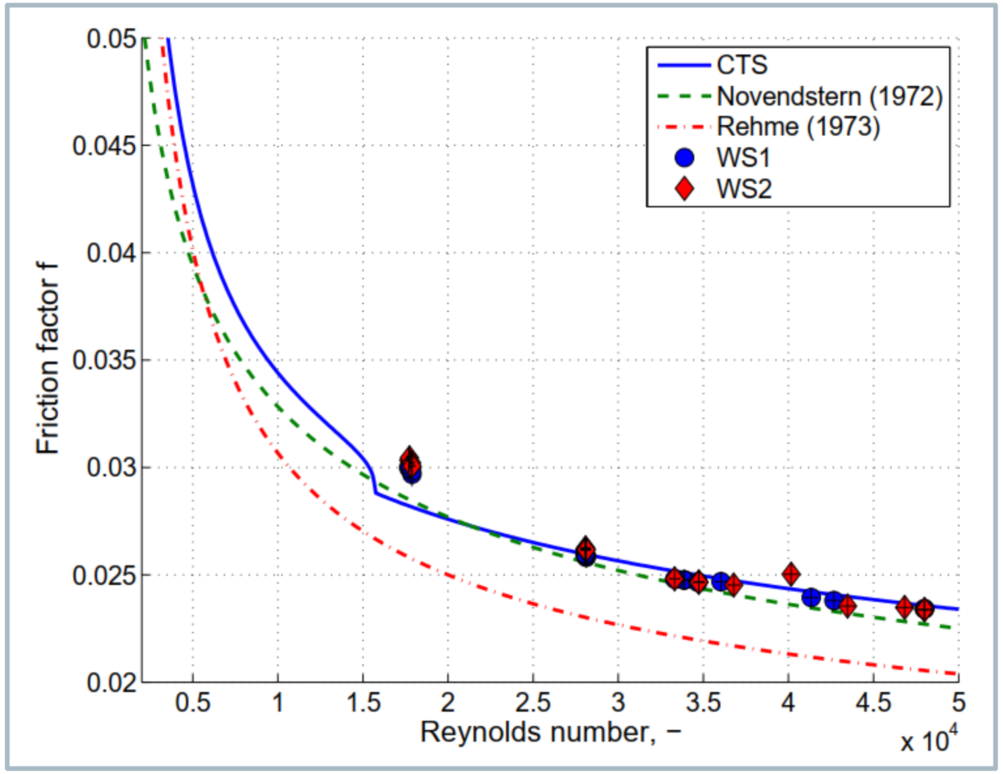
Largest LBE thermal hydraulics loop facility in Europe, 3 m usable test section hight

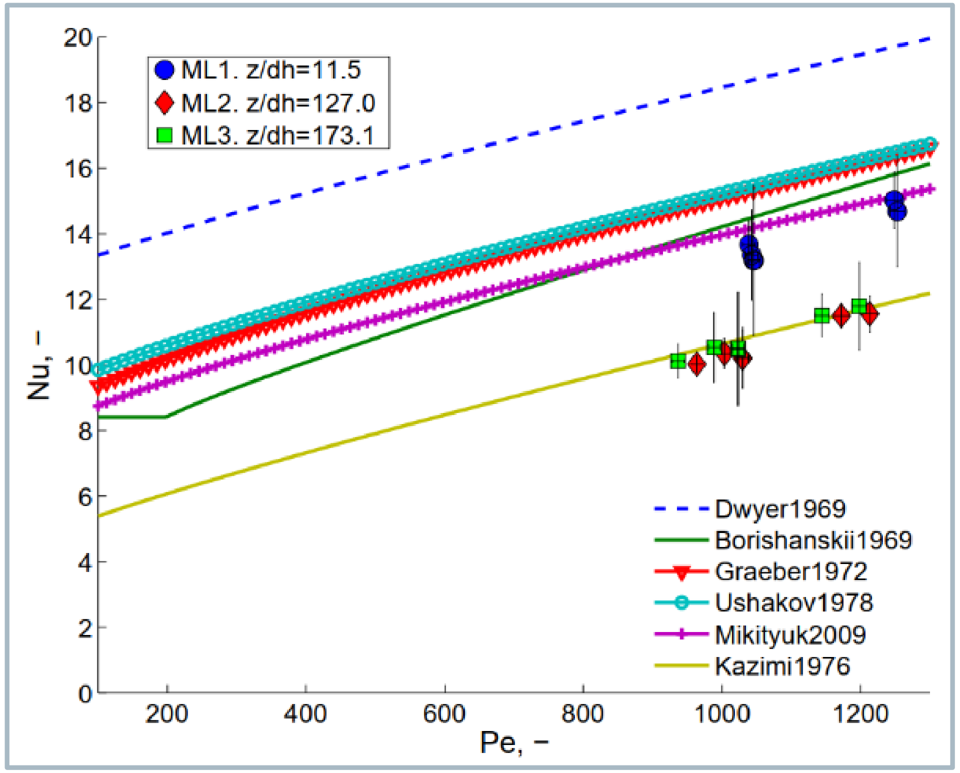
Prototypical test conditions AND highest precision combined.

Example Scientific Program - Thermal hydraulics
highlight: Wire wrapped 19 pin bundle in LBE

Most conservative heat transfer cor-relation recom-mended for the MYRRHA safety assessment

Friction coefficient and Nusselt Number – Test and provision of correlations





J. Pacio et al., Technical Report MAXSIMA Deliverable D3.4
J. Pacio et al., NED, https://doi.org/10.1016/j.nucengdes.2016.03.003
J. Pacio et al., NED, https://doi.org/10.1016/j.nucengdes.2018.01.034

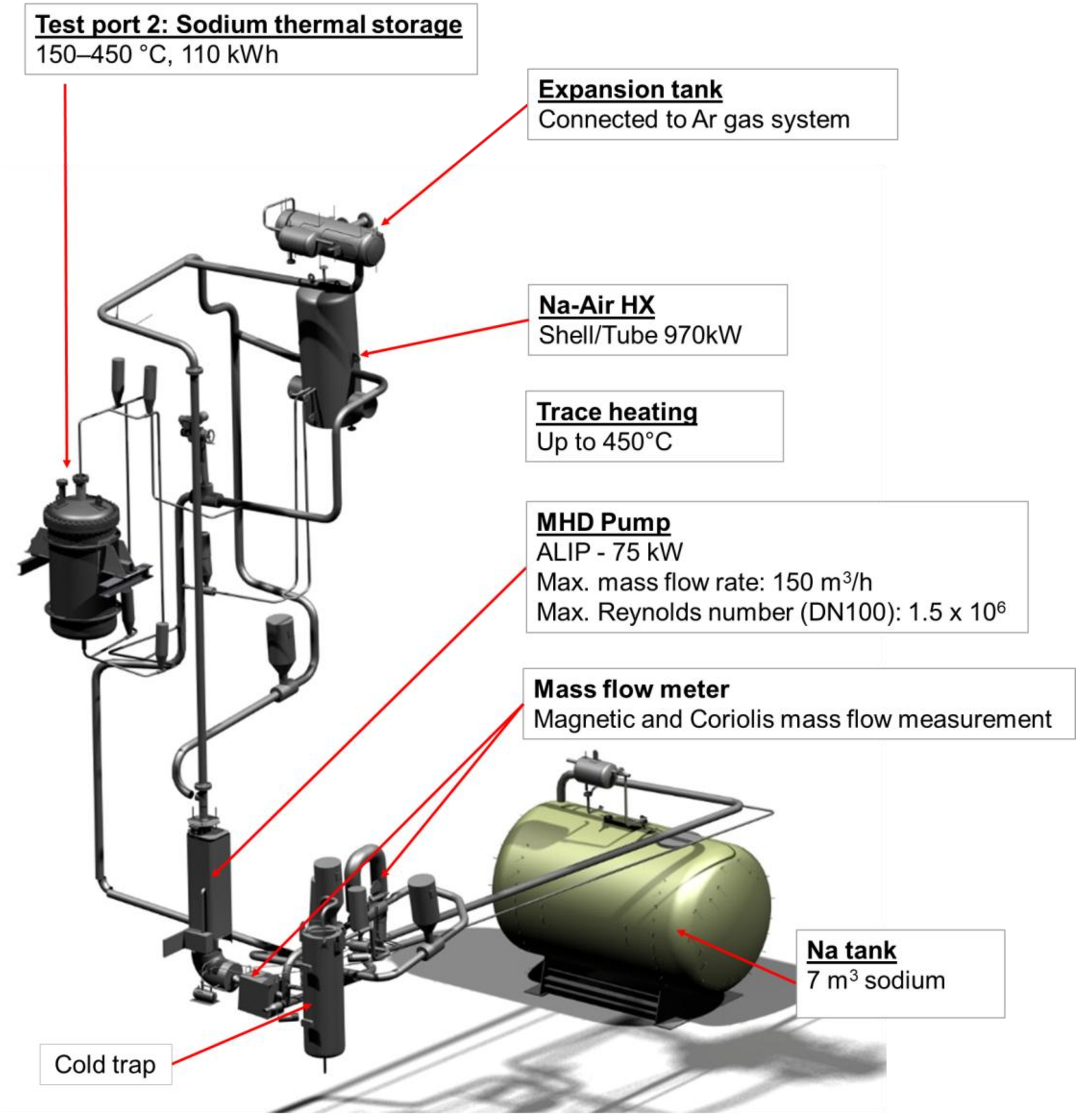
(courtesy of K. Litfin, A. Weisenburger)

KASOLA
Karlsruhe Sodium Lab

(courtesy of S. Ruck, S. Perez-Martin)

- Inventory: 7 m^3 Sodium
- Temperature: 150 - 450°C
- Flow rate: 150 m^3/h
- Flexible test space: 6 x 2 x 2 m^3
- Operational since: **July 2024**

- Versatile research platform: from **fundamental thermal hydraulic flow phenomena** through qualification of instrumentation up to **prototypical component testing** e.g. for
  - Advanced Modular Reactor (AMR)
  - Sodium Fast Reactor (SFR)

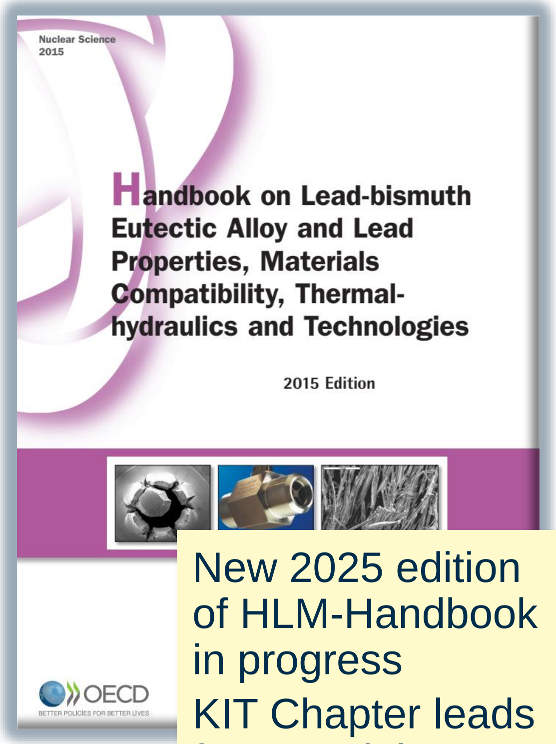


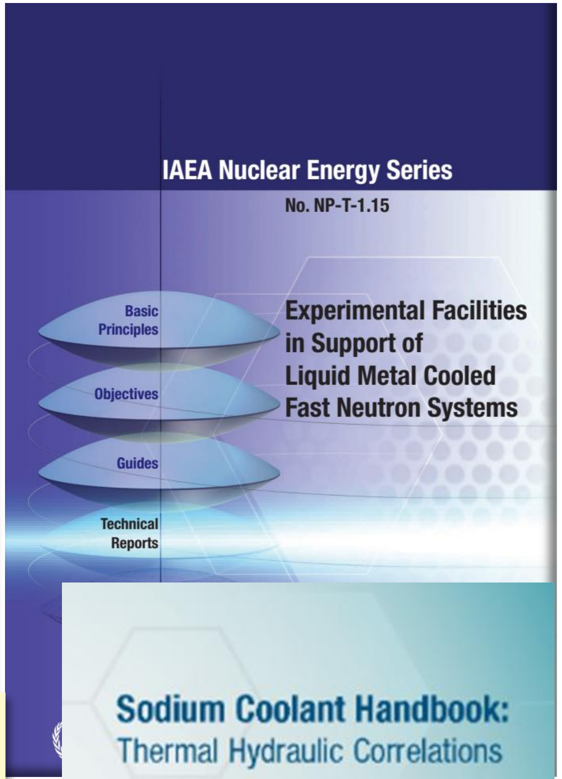
Isometric view of the KASOLA model

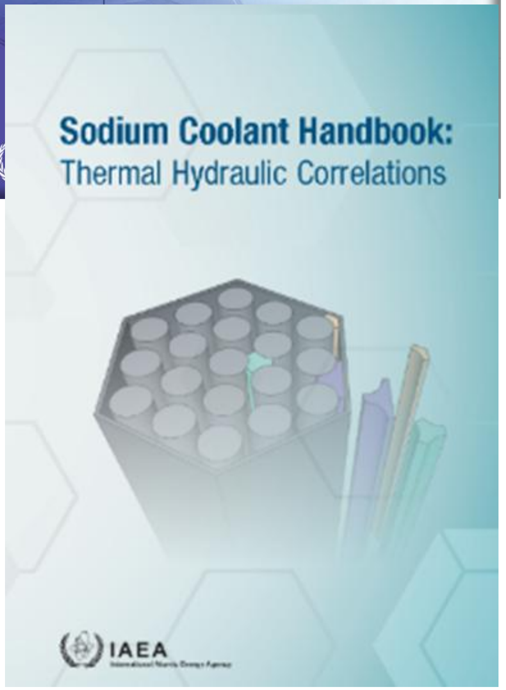
Scientific Program Thermal Hydraulics AND Materials

Outreach and international collaboration

- OECD-NEA Expert Group on Reactor Coolants / Components Technology (EGCoCoT)
- IAEA-LMFNS Facilities Database
- IAEA Coordinated Research Projects
- IAEA Network for Experiments and code validation sharing (NEXSHARE)
- IAEA CRCP/SOD/003 978-92-0-104824-0







New 2025 edition of HLM-Handbook in progress

KIT Chapter leads for materials, corrosion and instrumentation

MAXSIMA

Transparant

PASCAL

SEARCH

PATRICIA

OECD Publication: Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermal-hydraulics and Technologies – 2015 Edition
IAEA Publication: Experimental Facilities in Support of Liquid Metal Cooled Fast Neutron Systems
IAEA Publication: Sodium Coolant Handbook: Thermal Hydraulic Correlations, CRCP/SOD/003 978-92-0-104824-0

(courtesy of S. Ruck, S. Perez-Martin)

HELOKA-US
Molten Salt Loop

- **Commissioned in 2025**

- Thermal hydraulic research platform with **thermal energy storages** for the investigation of **molten salt flow phenomena and heat transfer under various operation conditions**

- Three coupled heat transfer loops with 260 kW thermal power
  - **Heat source**: Electrical heater or helium loop (300 - 520 °C, 80 bar)
  - **Energy storage system**: Molten salt loop (270 - 465 °C, 6 bar)
  - **Heat Sink**: Water Loop (160 - 220°C, 45 bar)

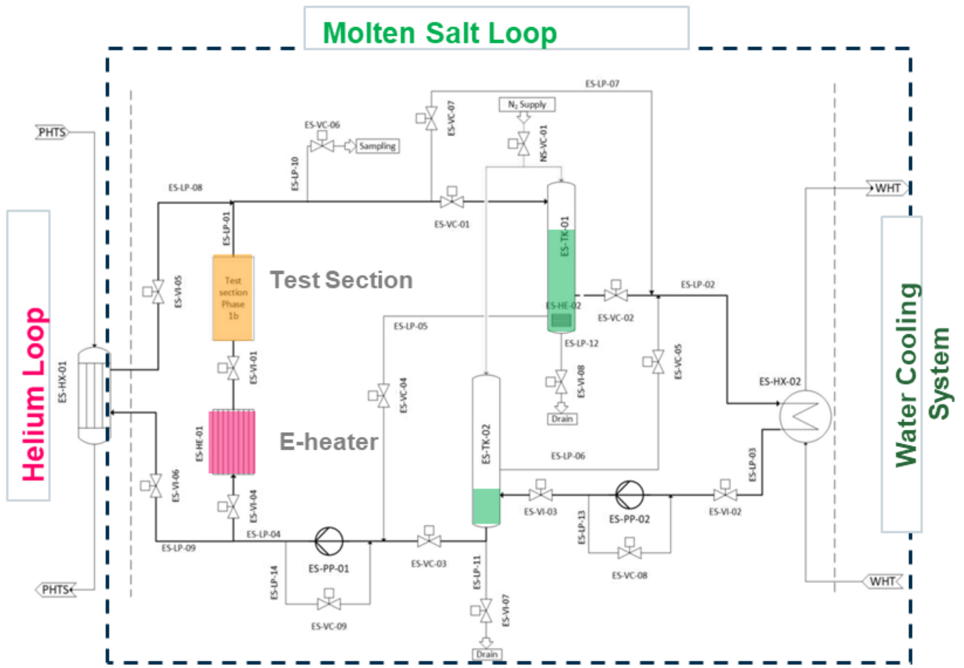
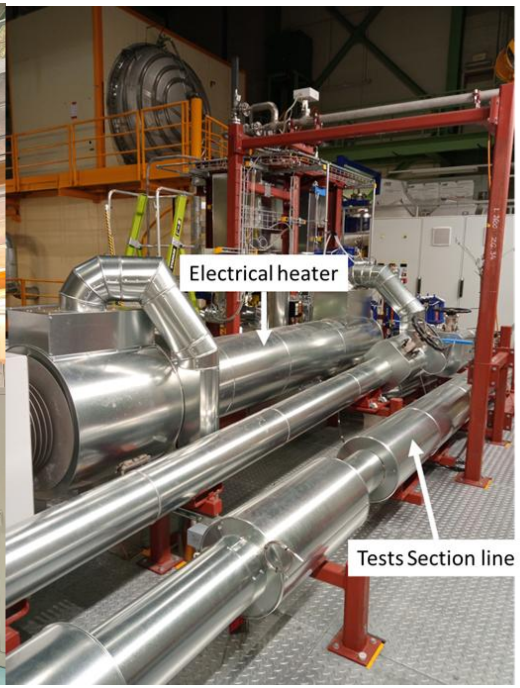


Figure: P&F Diagram HELOKA-US.

	High Power	Low Power
Thermal power (kW)	260	2.60
Duration (s)	7,200	600
MS IHX in / out temp. (°C)	270/465	270/299
MS SG in/out temp. (°C)	465/270	453/270
MS hot tank temp. (°C)	465	465
MS cold tank temp. (°C)	270	270
MS flow rate in charg. line (kg/s)	0.854	0.0581
MS velocity in charg. line (m/s)	0.452	0.031
MS flow rate in disch. line (kg/s)	0.789	0.727
MS velocity in disch. line (m/s)	0.417	0.385

HELOKA-US
Molten Salt Loop

- Fluid: **HITEC** eutectic mixture of potassium nitrate, sodium nitrite and sodium nitrate (KNO_3 (53%), $NaNO_2$ (40%) and $NaNO_3$ (7%)
- Dimensionless numbers in the relevant range



	LiF-ThF4-Fx	LiF-ThF4-235UF4-(TRU)F3	NaF-NaBF4/8%–92%	HITEC
Use	Fuel salt	Fuel salt	Intermed. salt	**Intermed. salt**
Melting Point (°C)	565	581	694	**142**
Cold temperature (°C)	650	675	600	**270**
Hot temperature (°C)	750	775	670	**465**
Average temperature (°C)	700	725	635	**367,5**
Density at Ave. Temp. (kg/m3)	4124.7	4286.1	1800.6	**1811**
Dyn. Viscosity at Ave. Temp. (Pa.s)	1.01E-02	9.59E-04	1.03E-03	**1.84E-03**
Th. Conduct. at Ave. Temp. (W/mK)	1.0097	1.7000	0.4448	**0.357**
Heat Capacity at Ave. Temp. (J/kgK)	1594.36	1010.00	1506.00	**1562**
Prandtl Number	16.0	0.6	3.5	**7.5 - 14.6**

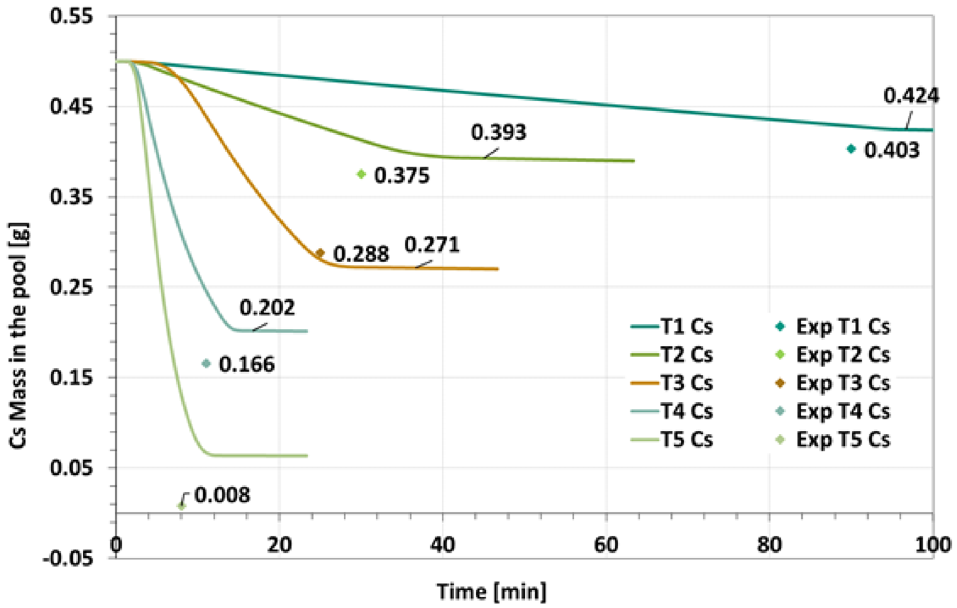
(courtesy of S. Ruck, S. Perez-Martin)

KfK Sodium Source Term Experiments

(courtesy of S. Ruck, S. Perez-Martin)

Validation of the CONTAIN-LMR code

- KfK programs: FAUNA (Na fires), NALA (secondary source term), FAUST (primary source term)
- NALA Program: Laboratory- and technical-scale experiments at different Na temperature and Cs content conducted in an inert-gas atmosphere (Ar or N2)
- Results of NALA tests for assessing Fission Products Retention Factors



NALA-I	T1 Exp.	T1 sim.	T2 Exp.	T2 sim.	T3 Exp.	T3 sim.	T4 Exp.	T4 sim.	T5 Exp.	T5 sim.
Na pool temperature (° C)	437	444	530	530	625	633	723	694	814	822
Cs RF^{-1}	30	27	18	40	11	12	6	5	4	15
Ratio Cs/Na vapor pressures	-	26	-	15	-	9	-	9	-	4

M. Garcia, L-E. Herranz, 2025, https://doi.org/10.1016/j.anucene.2024.111059
S. Perez-Martin, FR26

Calculation Platform for Advanced Systems

(courtesy of Perez-Martin)

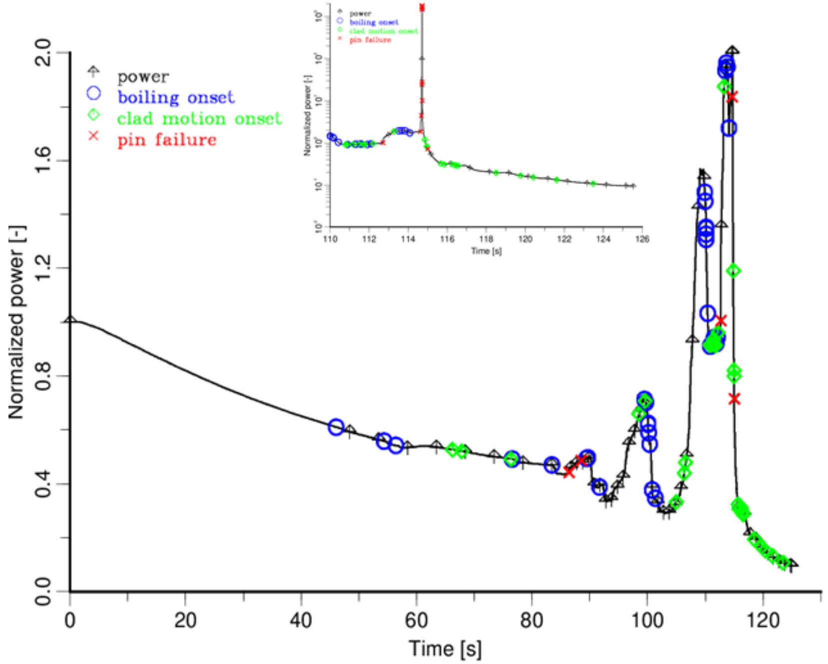
	SFR	MSR	LFR	GFR
Simulation Codes	SAS-SFR, SIM-SFR, SIMMER, CONTAIN-LMR	SIM-MS, SIMMER	SAS-LFR, SIM-LFR, SIMMER	SIM-GFR
Current Projects	EU ESFR-SIMPLE	EU SAMOFER	ANSELMUS, MYRRHA	TREASURE, IAEA S-allegro

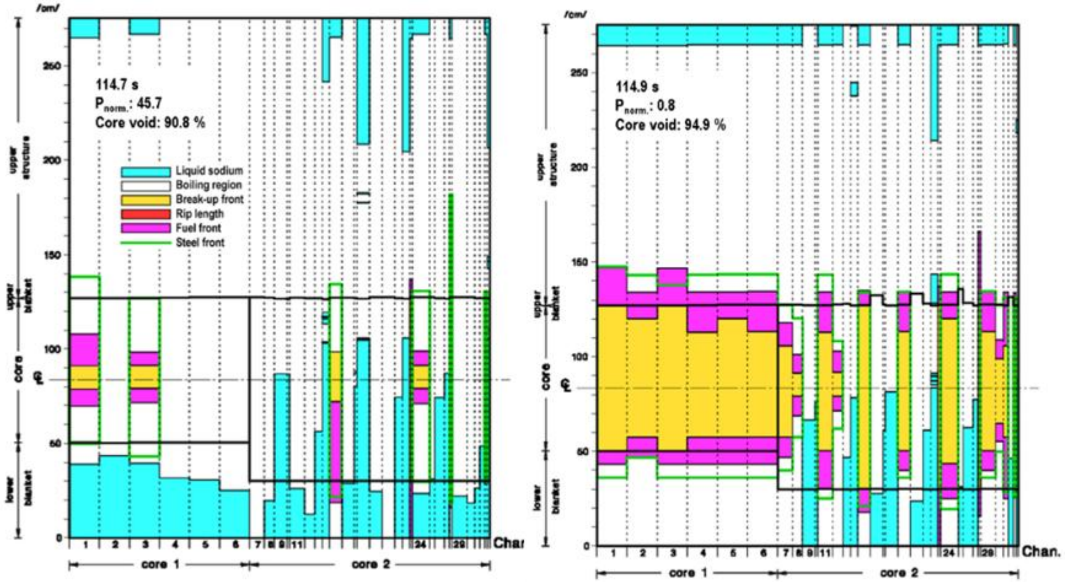
- SFR - Code development and application
- Initiation phase of accidental transients → SAS-SFR, SIM-SFR
- Transition and Expansion phase of accidental transients → SIMMER
- Source term evaluation in severe accidents → CONTAIN-LMR Code

Initiation Phase of Accidental Transients
SAS-SFR Code (KIT code developer)

ESFR-SMART
sodium fast reactor safety
(courtesy of Perez-Martin)

- SAS-SFR Code: Complete transient analysis including post-boiling and post-cladding failure phases up to hexcan integrity failure





~ 17 % min. flow rate to avoid pin failure
... Now being applied to SMR (360 MWth) SFR

- Currently: extension to metallic-fuels, validation against EBR-II and TREAT tests, and code-to-code benchmarking with SAS4A code

KIT Calculation Platform for Advanced Reactors

The SIMMER code

(courtesy of A. Rineiski)

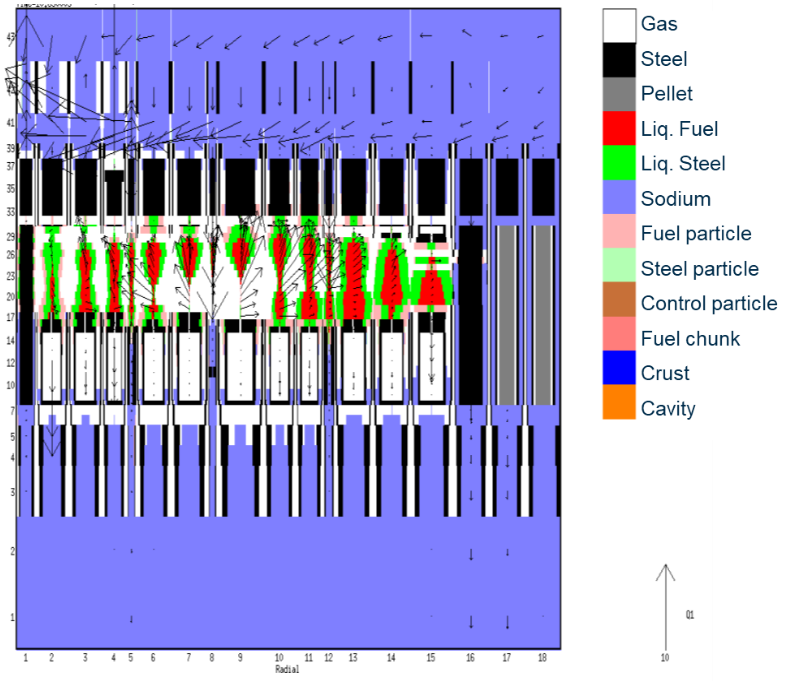
- **Safety assessment** of **innovative reactor systems (LFR, SFR, AMRs)** and **development of accident mitigation strategies**

- **KIT in key strategic role for 3 decades as co-developer** of the **SIMMER-code in cooperation with JAEA, CEA, and now with other partners**
  - **2D/3D fluid-dynamics** code coupled with a **structure model** and a **space-, time- and energy-dependent neutron transport model**

- **SIMMER as reference code** for Core Disruptive Accident (CDA) analyses in fast neutron spectrum reactors

- Originally developed for Sodium Fast Reactors

Material distribution in a SFR during a severe accident



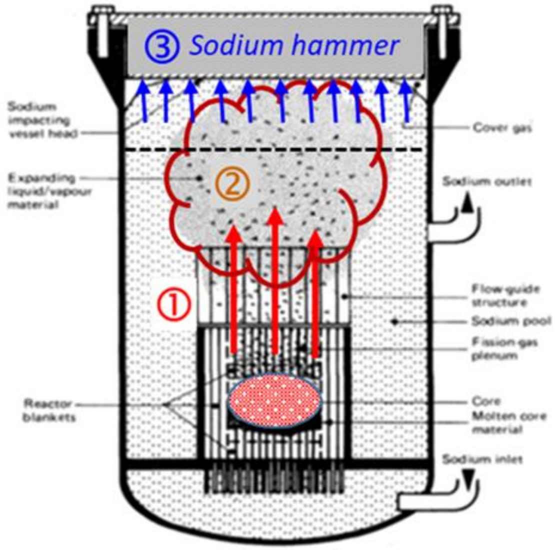
Extension of SIMMER to Lead- and Gas-cooled reactors and to Molten Salt reactors.

Transition and Expansion Phases in SFRs

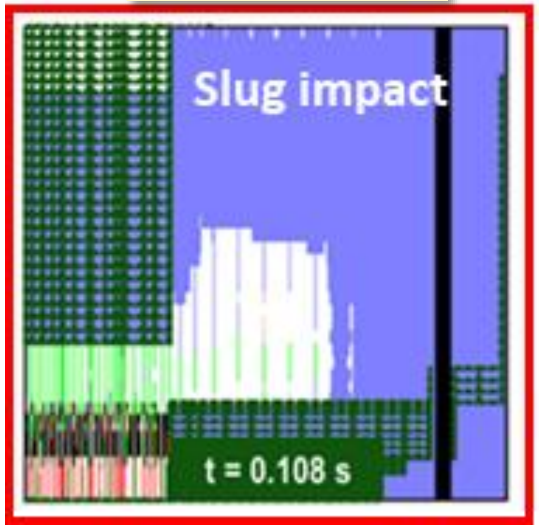
(courtesy of A. Rineiski)

Highlight: Reduction of the Mechanical Energy Release

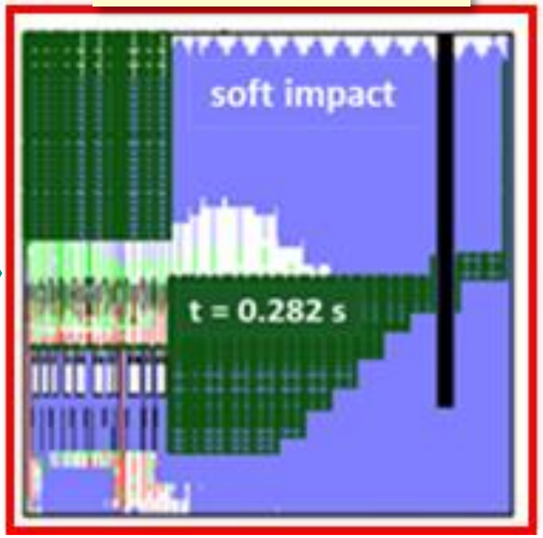
- Impact of **new safety measures to prevent sodium boiling** and **multiple re-criticalities**
- **From conventional (CP_ESFR) to ESFR-SMART design**
  - High sodium plenum above the core instead of reflector and low plenum
  - Corium transfer tubes
  - Passive safety devices



CP-ESFR
W=794 MJ



ESFR-SMART
W=396 MJ

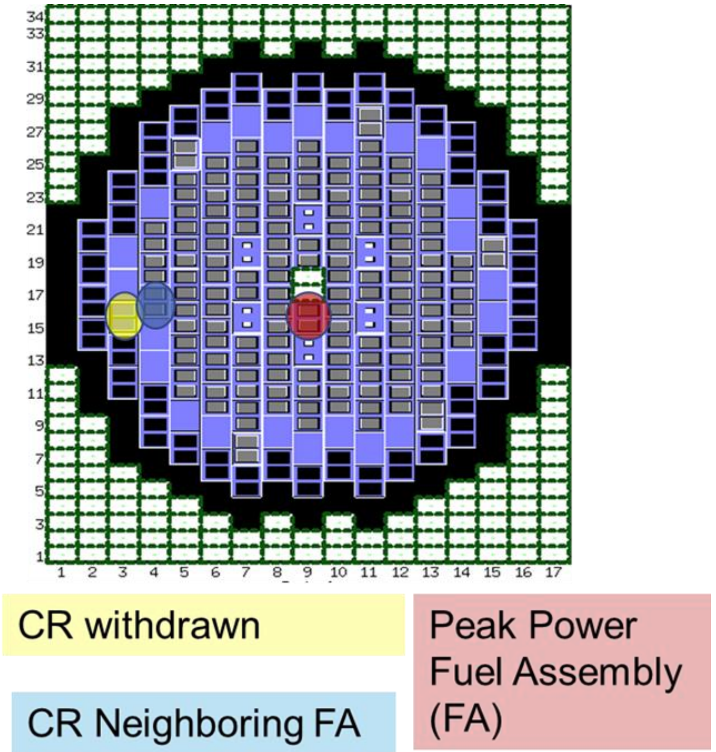


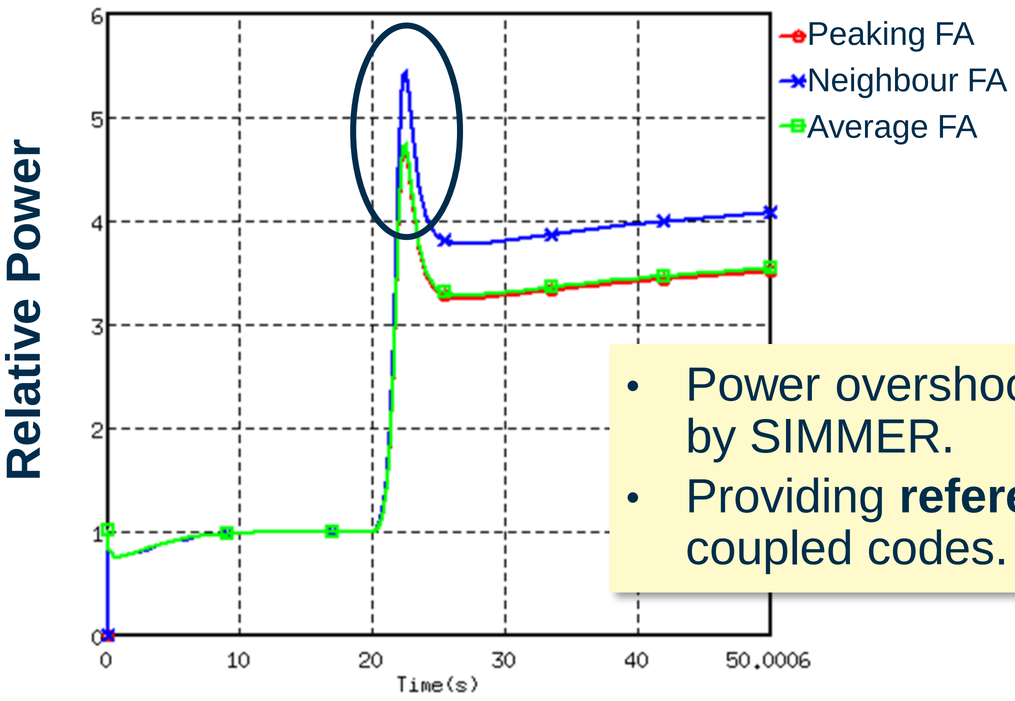
MYRRHA-Lead Fast Reactor

(courtesy of A. Rineiski)

Highlight: 3D SIMMER Capability for Transient Analyses

- Supporting the validation of codes applied for Lead Fast Reactor licensing, such as RELAP
- **3-D neutronics & fluid-dynamics coupled calculation** of a Control Rod withdraw transient





- Power overshooting (15%) computed by SIMMER.
- Providing **reference solutions** to **not** coupled codes.

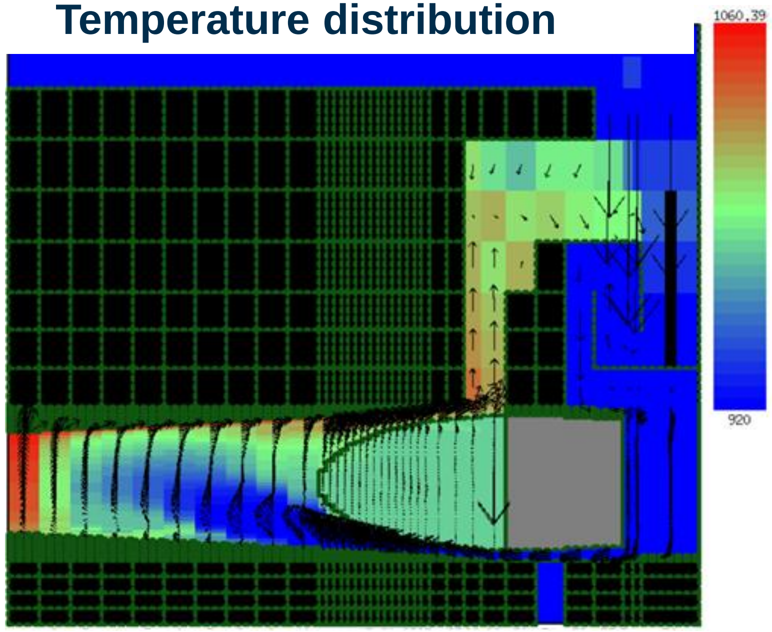
Safety Evaluations of Molten Salt Fast Reactors

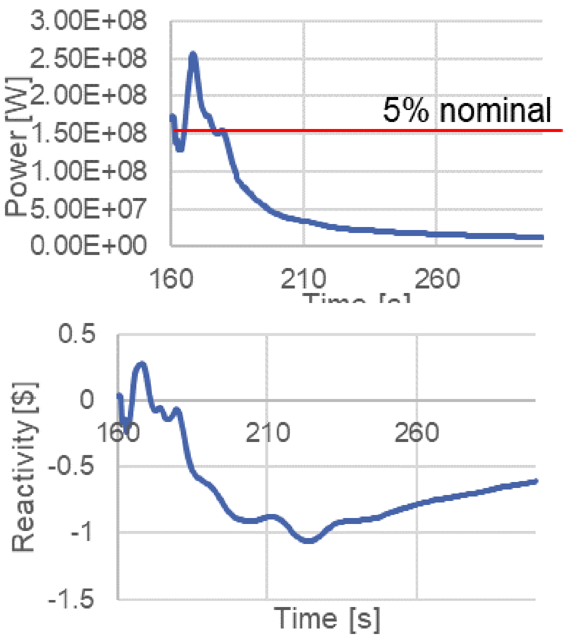
(courtesy of B. Kędzierska)

Highlight: 3D SIMMER Analyses

- SIMMER developments for MSR-specific physics
  - **New equations of state for liquid salts (fluorides, chlorides under development) with fuel isotopes**
  - **Delayed neutron precursor drift model**

Temperature distribution





Successful verification against CFD tools.

Demonstration of the safe shut down of the system following a loss of flow transient.

X.-N. Chen, et al., Simulation of gas injection into liquid with SIMMER, Technical Meeting on Compatibility Between Coolants and Materials for Fusion Facilities and Advanced Fission Reactors, IAEA Headquarters, Vienna, Austria, Nov. 2023
B. Kędzierska, et al., Modified Algorithm for Taking into Account Delayed Neutron Precursor Movement in the Improved Quasi-Static Scheme of the SIMMER-III Code, accepted to M&C 2025, Denver, CO, USA, Apr. 2025
B. Kędzierska, et al., Decay Heat Removal from the MSFR CORE through the Passive Safety System, in Proc. of the Int. Conference Nuclear Energy for New Europe, Portorož, Slovenia: Nuclear Society of Slovenia, September 2024

Education and Training

- **Internships/student, Master theses, Doctoral theses**
- Lectures on reactor physics, reactor dynamics and safety analysis methods at the **Mechanical Engineering Faculty of KIT**
- **Framatome** Nuclear Professional School (https://www.fps.kit.edu/)
- **KIT/CEA** Frederic Joliot/Otto Hahn Summer School on Nuclear Reactors (https://www.fjohss.eu/)
- **Training courses on the SIMMER code**
- **QUENCH Workshop, 30 Editions!,** https://quench.forschung.kit.edu/21.php
- Since 2022 support to the organization of '**European Review Meeting on Severe Accident Research' (ERMSAR)**, SNETP/NUGENIA TA2, SEAKNOT co-organized with IAEA and OECD/NEA (next edition in Madrid, organized by CIEMAT, https://ermsar2026.com/)

Summary

- **Preserving, extending, and sharing the knowledge regarding reactor safety, EIA, and EP&R for current and advanced nuclear NPPs**
- Large scale **Experimental Platform** and extend to new materials and safety-relevant phenomena of innovative reactors
  - **Providing worldwide recognized, independent, high quality reference data and know-how** on water cooled and liquid metal thermal hydraulics and material research for code development, safety assessments, etc.
- Extending the **Computational Platform** for innovative reactors, new materials, passive safety systems, vulnerable siting
  - **Providing beyond state-of-the-art calculation methods for safety analysis**, reference data for code validation, and high-fidelity reference solutions to the international research community
- **Keep active role in the international community**, e.g., EU research programs and strategic cooperation with leading international institutions

Acknowledgement

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- KALLA: K. Litfin, A. Weisenburger (https://www.ites.kit.edu/english/140.php)
- KASOLA, HELOKA-US: S. Ruck, S. Perez-Martin (https://www.inr.kit.edu/58.php)
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- JRODOS: S. S. Ottenburger, W. Raskob, D. Trybushnyi (https://www.ites.kit.edu/136.php)
- Computation Water-cooled NPPs: V. H. Sanchez-Espinoza, M. E. Cazado, A. Campos Munoz, J. Duran Gonzalez, G. Huaccho Zavala, Z. Jimenez Balbuena, I. Karaaslan, L. Mercatali, O. Murat, Y. Song, A. Stakhanova, S. Wang, K. Zhang, F. Kretzschmar (https://www.inr.kit.edu/342.php, https://www.multiphysics.kit.edu/)

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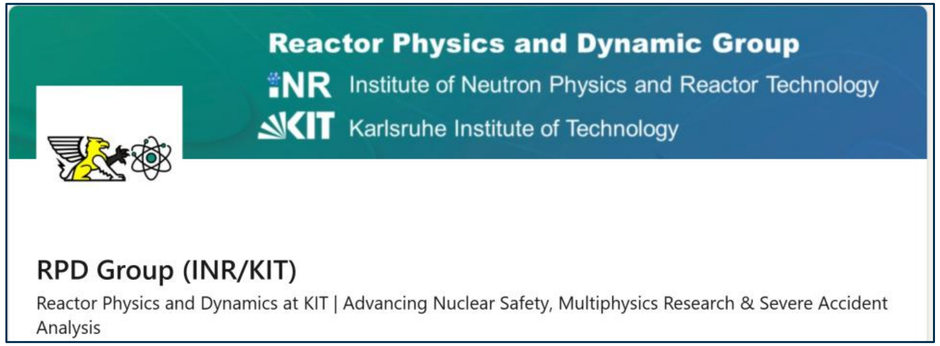
- KALLA https://www.ites.kit.edu/english/140.php
- KASOLA, HELOKA-US https://www.inr.kit.edu/58.php
- Computation Advanced NPPs https://www.inr.kit.edu/english/967.php
- JRODOS https://www.ites.kit.edu/136.php
- Computation Water-cooled NPPs – INR/RPD https://www.inr.kit.edu/342.php

https://www.multiphysics.kit.edu/

https://www.linkedin.com/company/rpd-group-inr-kit



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