IAEA
International Atomic Energy Agency

3rd Joint ICTP-IAEA Workshop on
Physics and Technology of
Innovative Nuclear Energy Systems
12 – 16 December 2022, Trieste, Italy

Thermal Hydraulics of *Innovative Nuclear Energy Systems*

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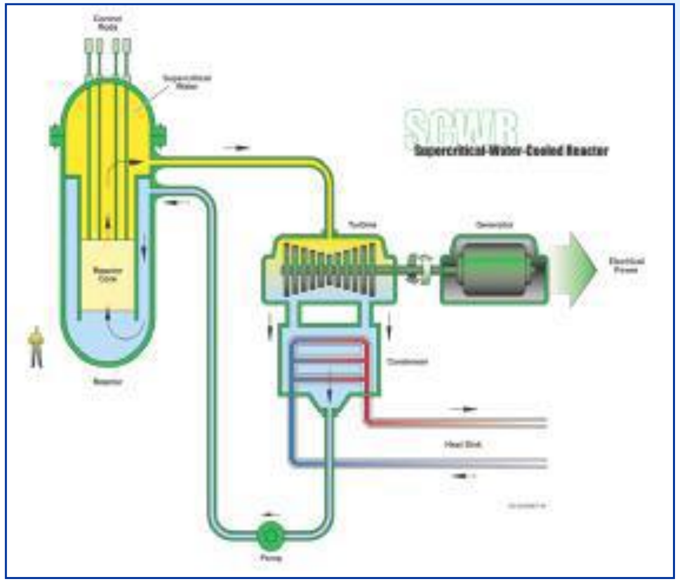
Nuclear Power Technology Development Section
Department of Nuclear Energy

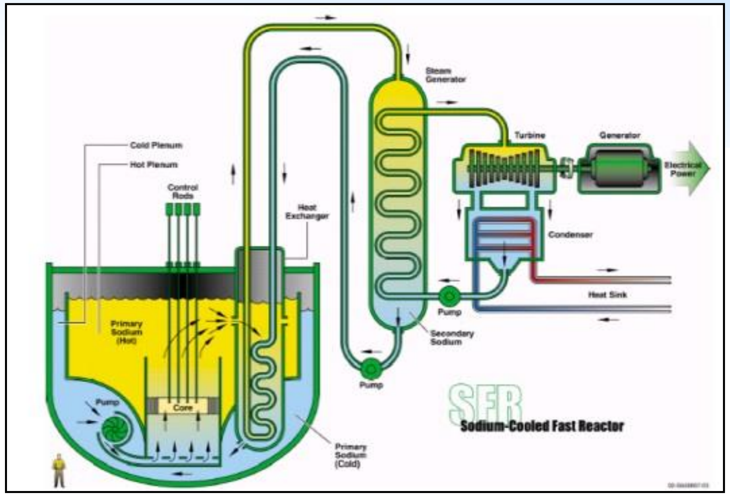
Outline

- Reactor Classification and Innovative Fast Neutron Systems
- Main Reactor Components
  - Reactor Core
  - Fuel Rod Bundle (Subassembly)
  - Fuel Rod (Pin)
- Comparison of Coolant Physical Properties
- TH Calculations on Design Temperature Limits
- Simulation of Real S/A under Irradiation
- Transient Analysis

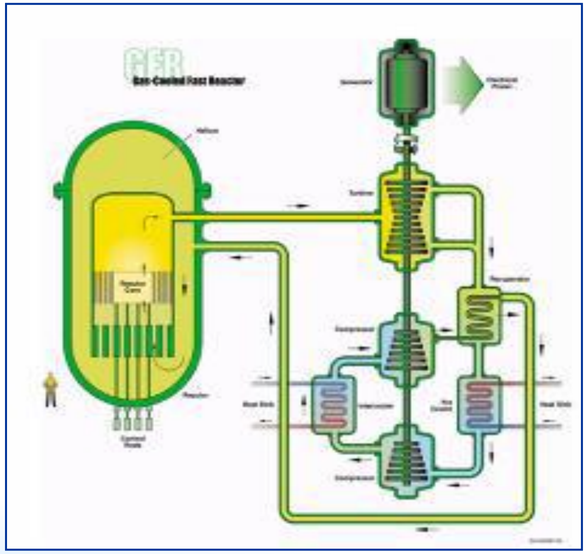
General Reactor Classification

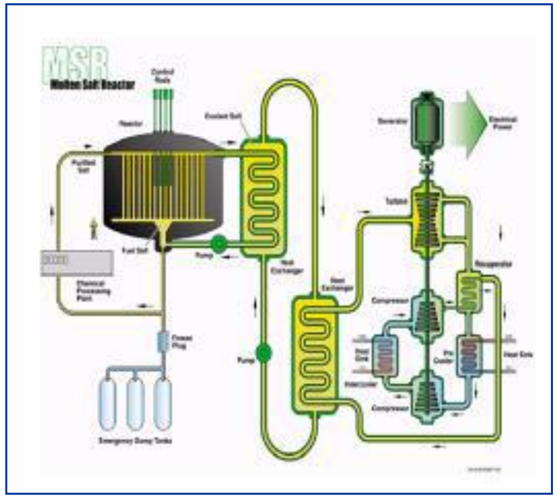
- Moderator
  - Water / Heavy Water
  - Graphite
  - None (fast neutron systems)
- Coolant
  - Water/Heavy Water
  - Liquid Metal
    - Sodium / Lead / Lead-Bismuth Eutectic (LBE)
  - Gas
    - Air / CO_2 / Helium
  - Molten Salt
- Fuel
  - UO2
  - MOX (UO_2 + PuO_2)
  - Metallic
  - Molten Salt
- Purpose
  - Electricity/Non-Electric Application
- Power
  - Low/Middle/High



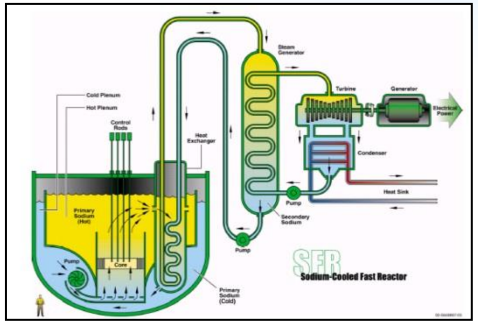


GEN-IV Reactors (GIF)

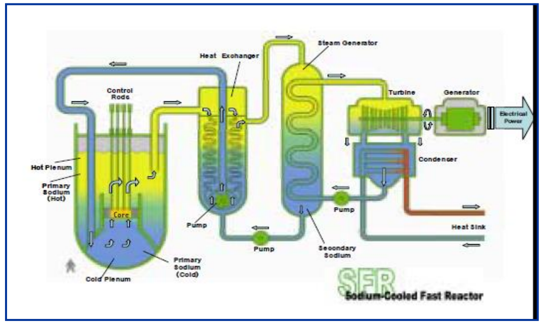




Six Generation IV Reactor systems

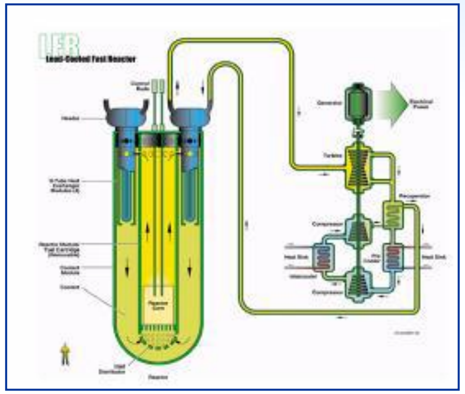


Pool type

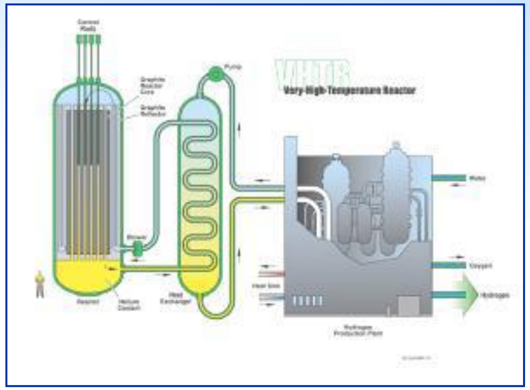


Loop type

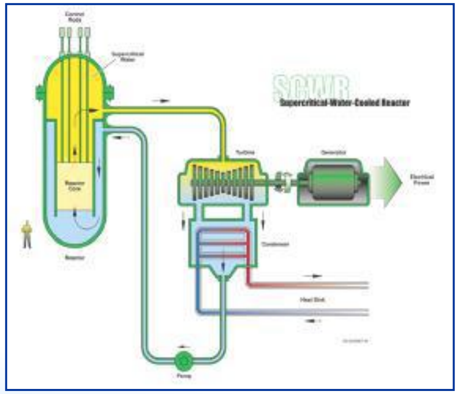
Sodium-cooled Fast Reactor (SFR)



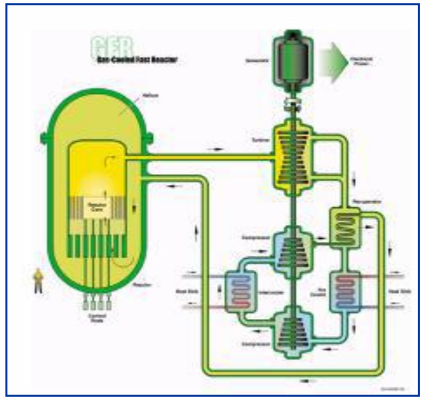
Lead-cooled Fast Reactor (LFR)



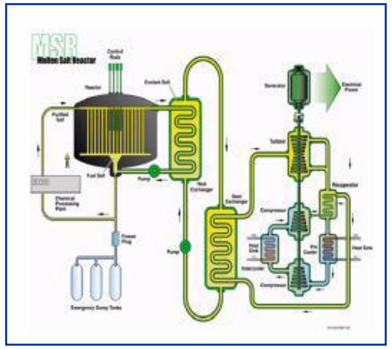
Very-High-Temperature Reactor (VHTR)



Supercritical-Water-cooled Reactor (SCWR)



Gas-cooled Fast Reactor (GFR)

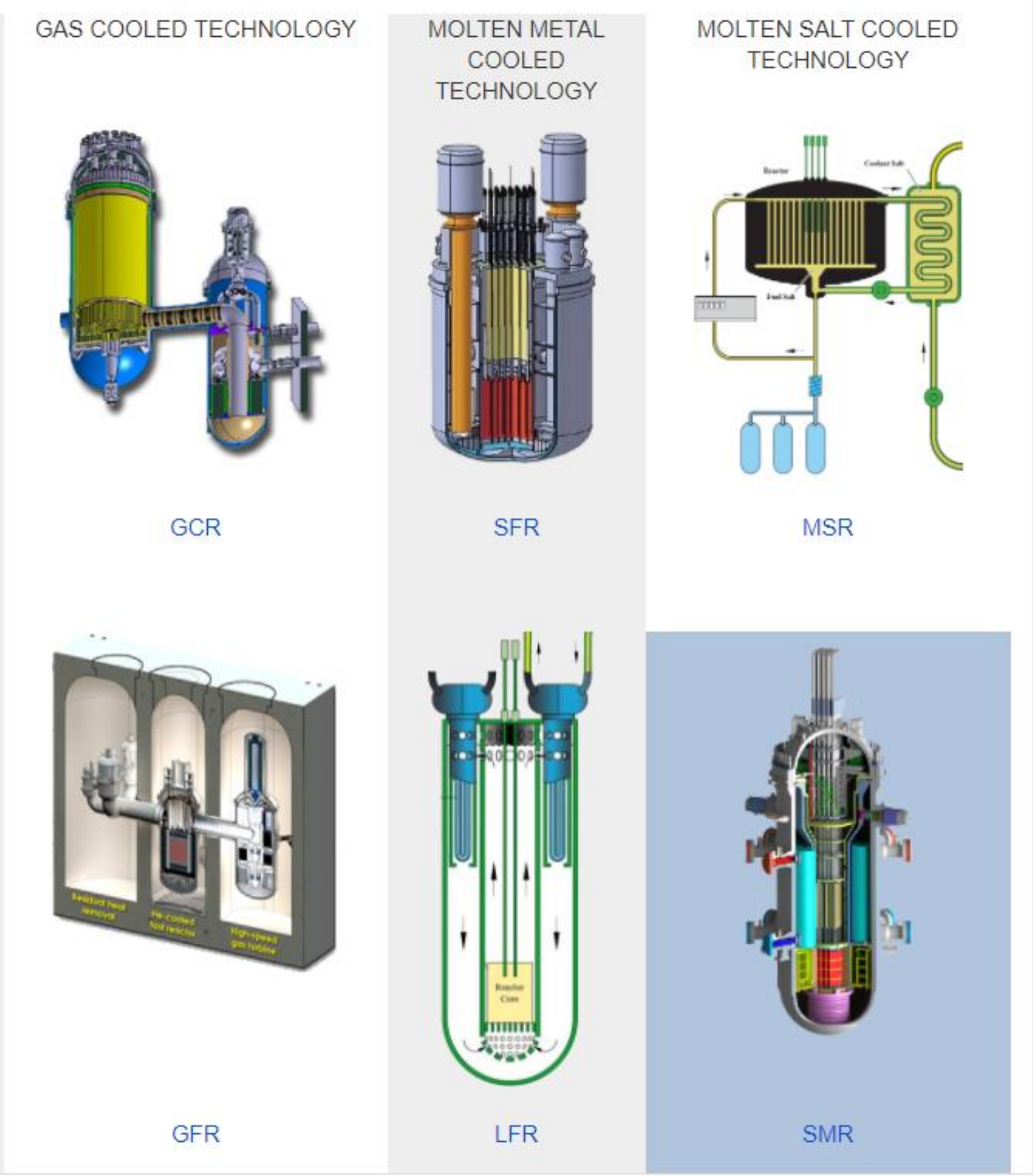


Molten Salt Reactor (MSR)

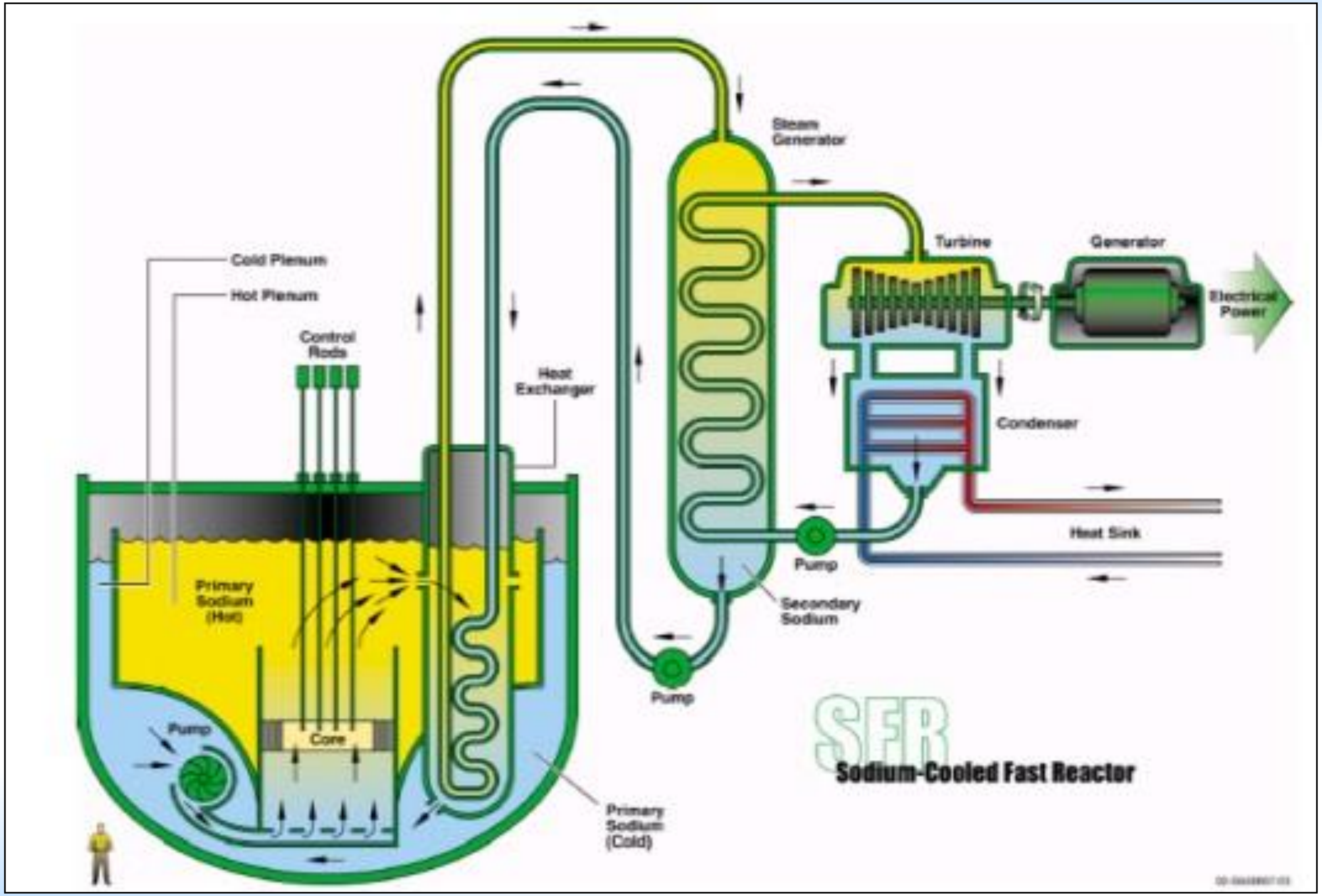
GIF website: www.gen-4.org

IAEA and GIF Terminology

- **Early Prototypes and Demonstration Plants Gen I**
- **Current Fleet Gen II-III**
- **Advanced Nuclear Reactors**
  - *Evolutionary designs* Gen III and III+
  - *Innovative designs* Gen IV
  - SMRs can be either *evolutionary* or *innovative*
- **ARIS:** IAEA Advanced Reactors Information System: https://aris.iaea.org/



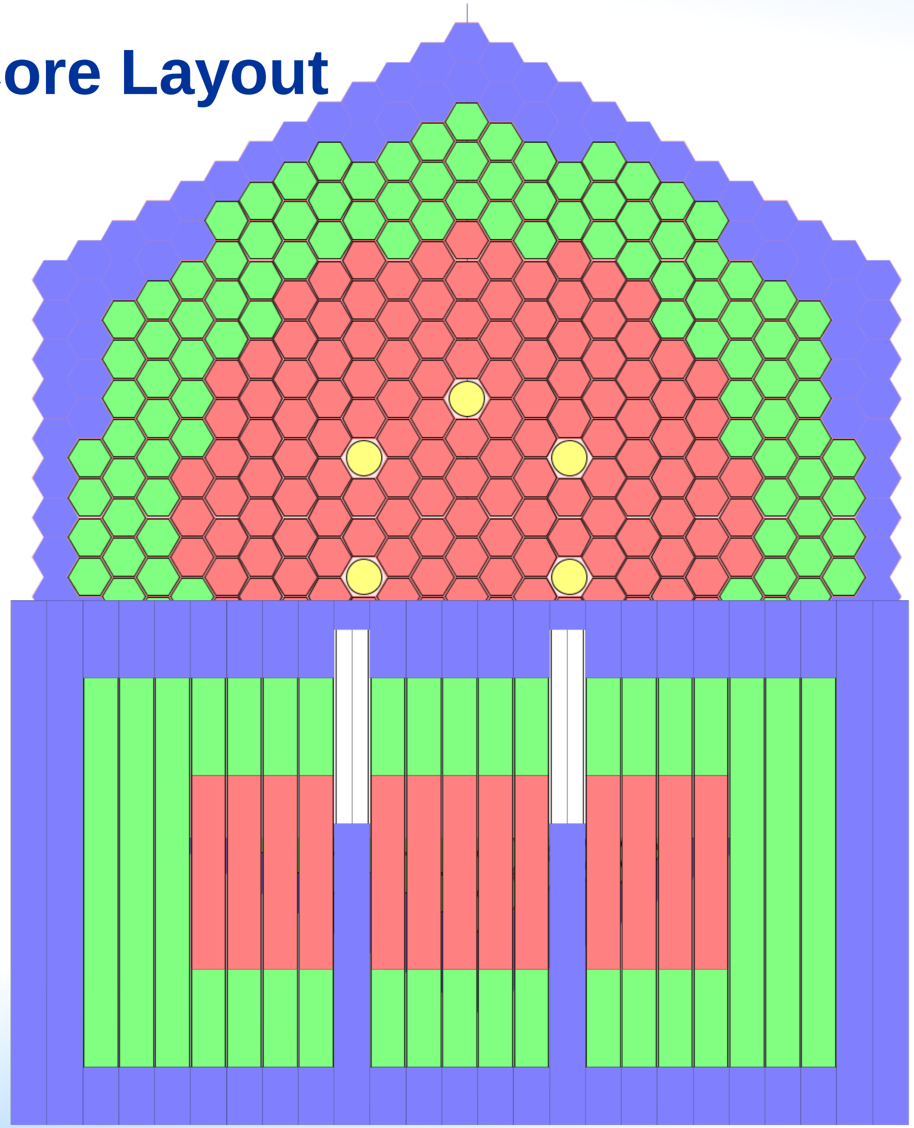
Sodium Cooled Fast Reactor (SFR)



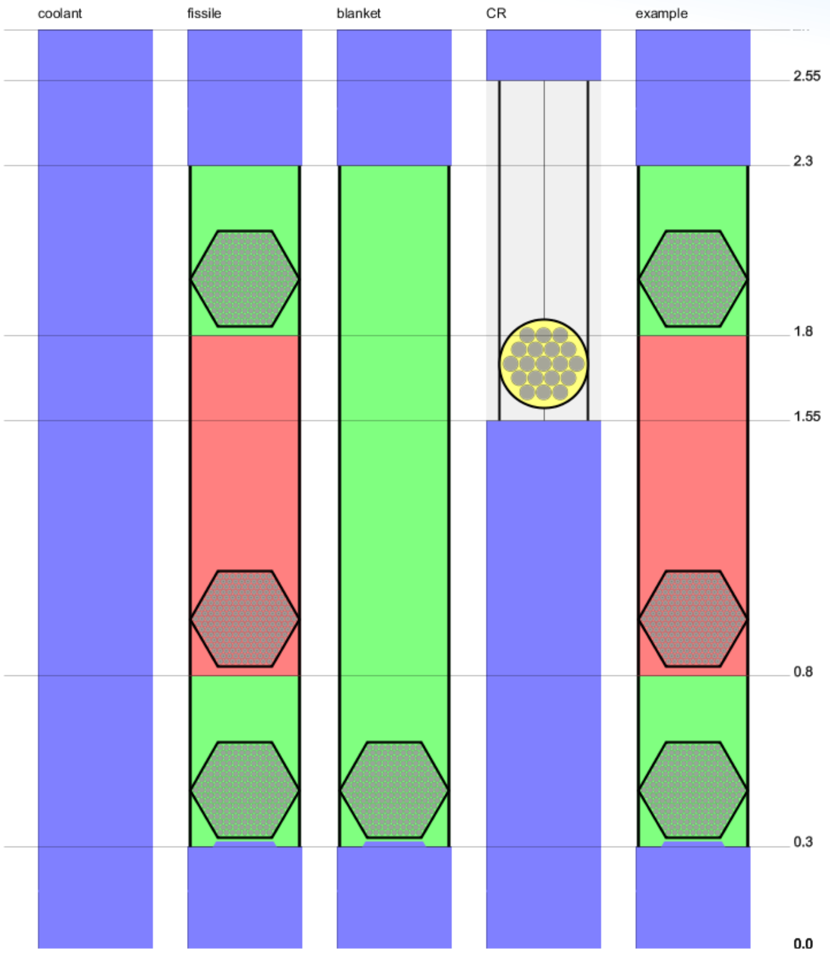
GIF website: www.gen-4.org,

Reactor Core

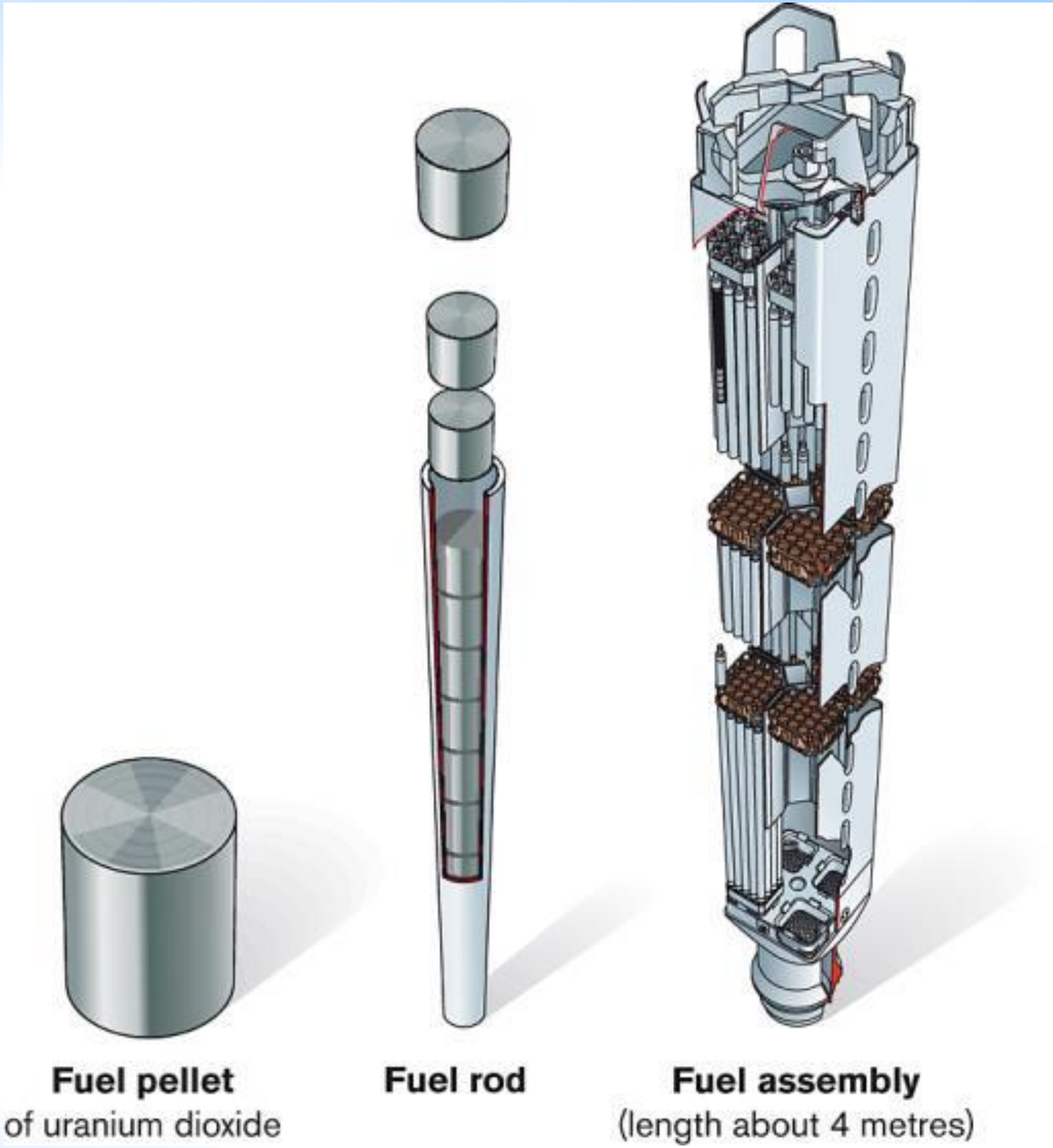
Core Layout



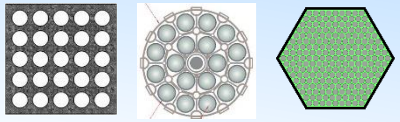
Sub-Assemblies (S/A)



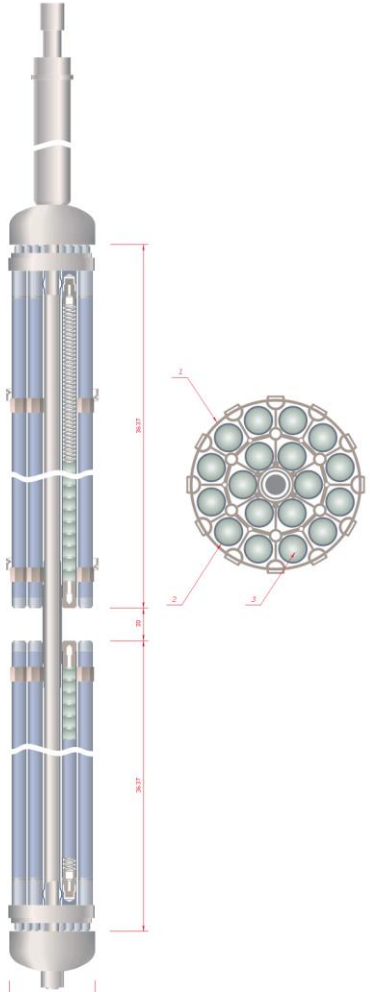
LWR Fuel Assembly (Rod Bundle)



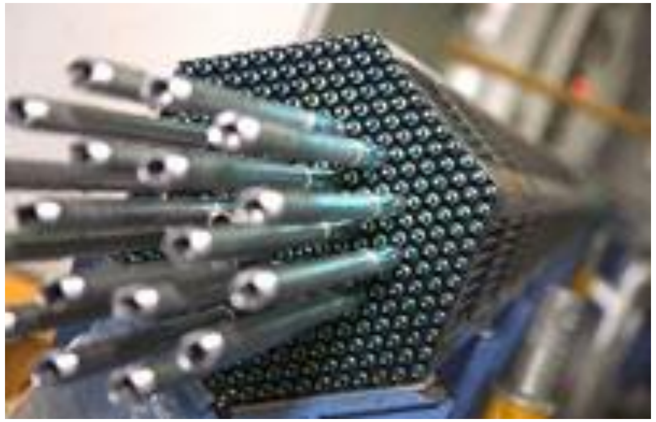
WCR Rod Bundles



PWR

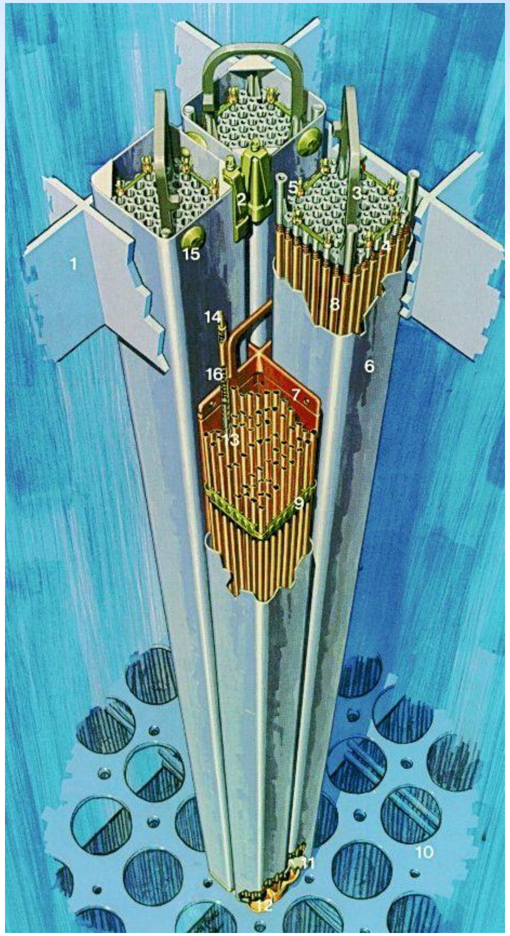


RBMK



VVER

CANDU

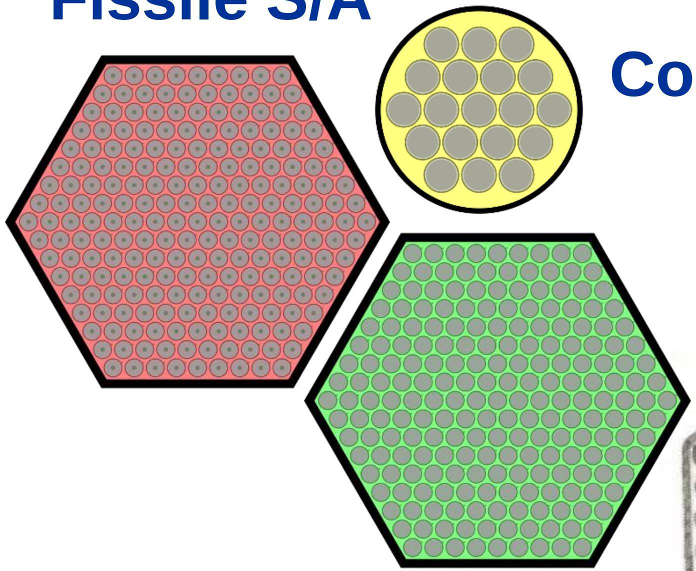


BWR

Sub-Assembly Types

Fissile S/A

Control Rod S/A

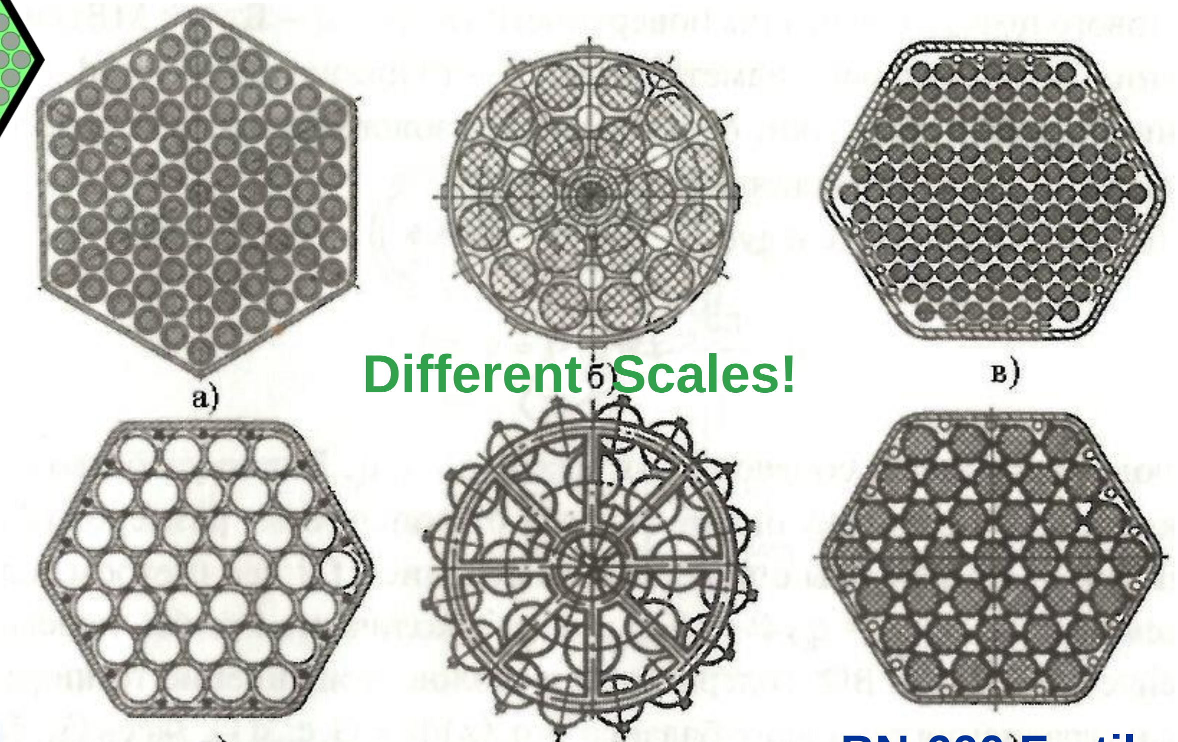


Fertile (Blanket) S/A

VVER-1000

RBMK-1000

BN-600 Fissile



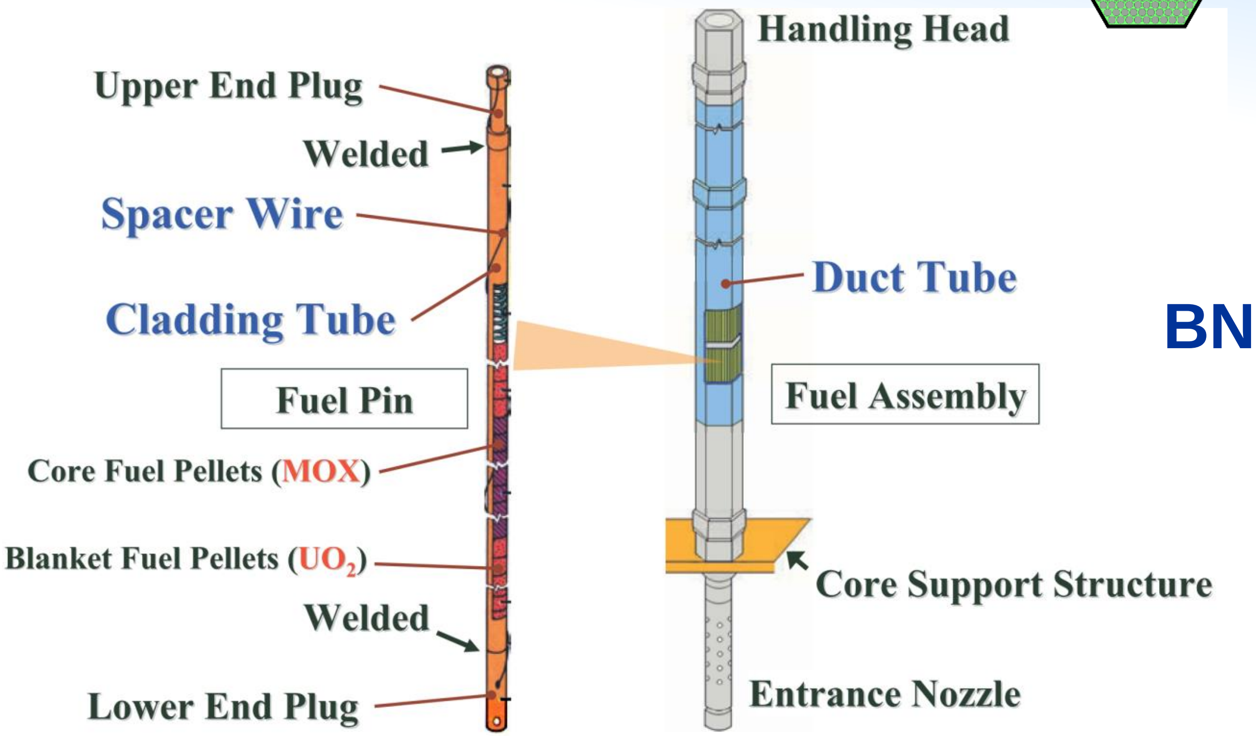
Different Scales!

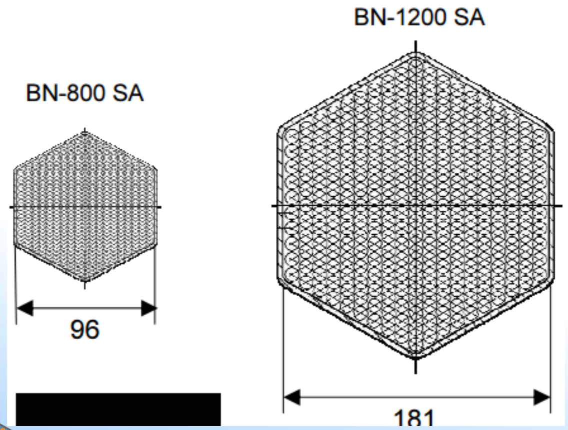
BOR-60

CANDU

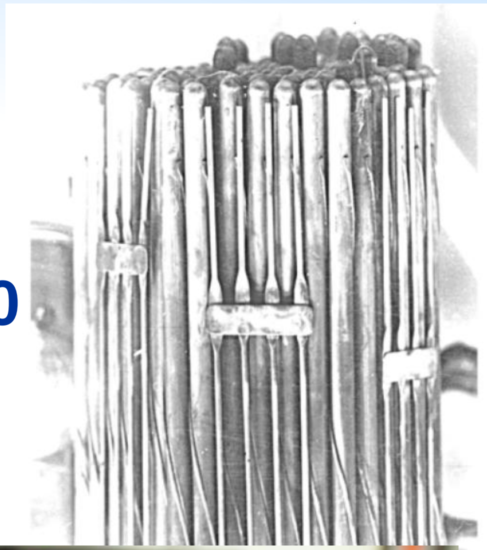
BN-600 Fertile

SFR Fuel Assemblies



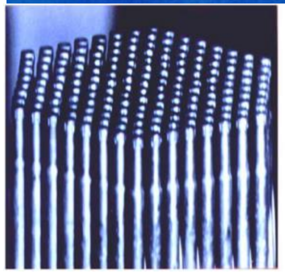


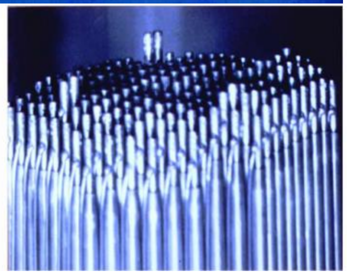
BN-600



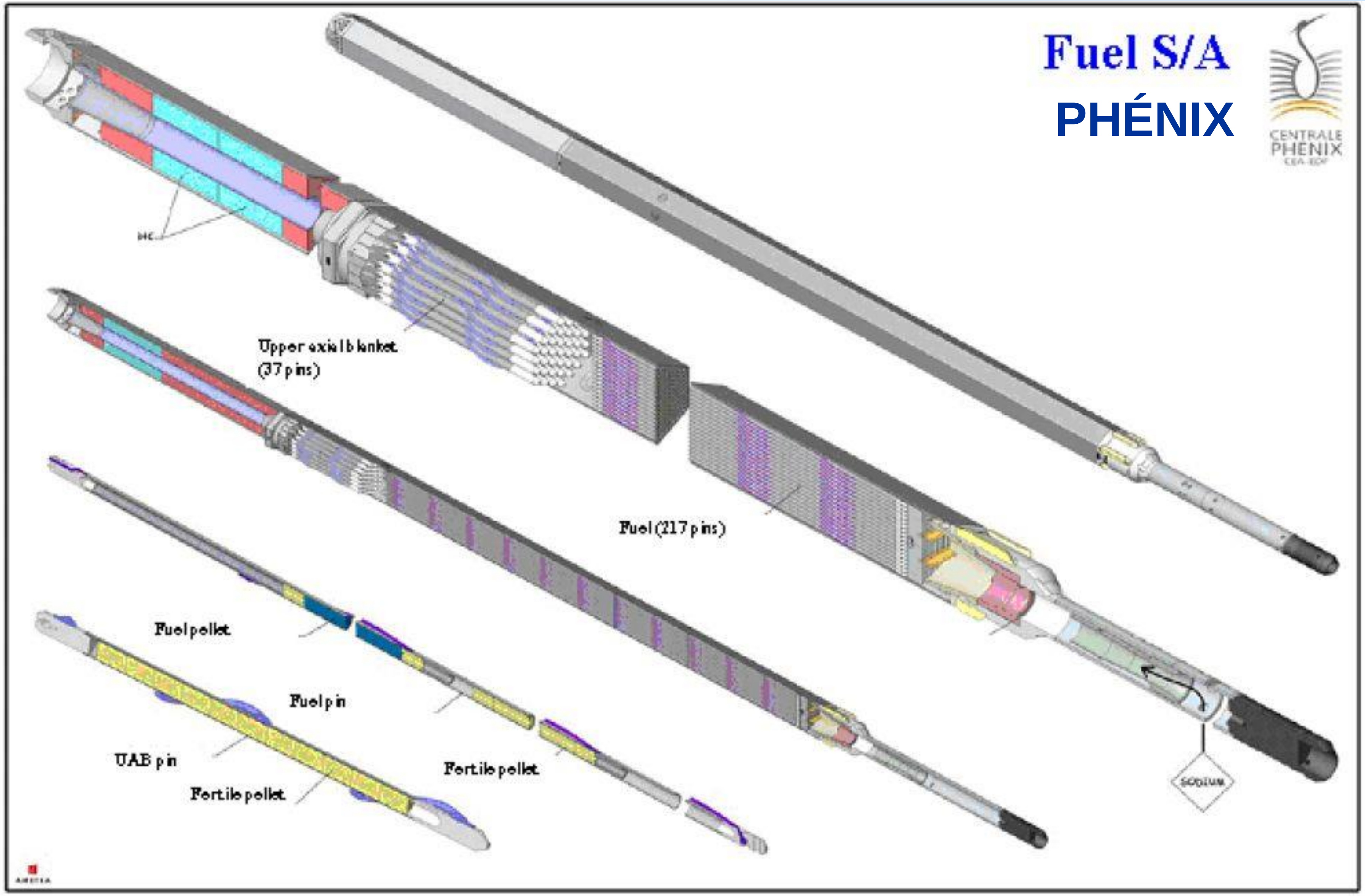
BOR-60

NATRIUM
(S/A mockup)



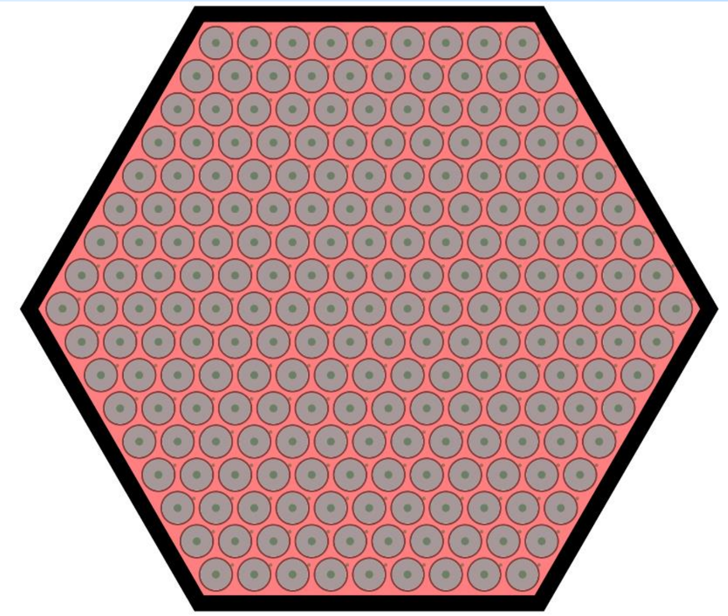


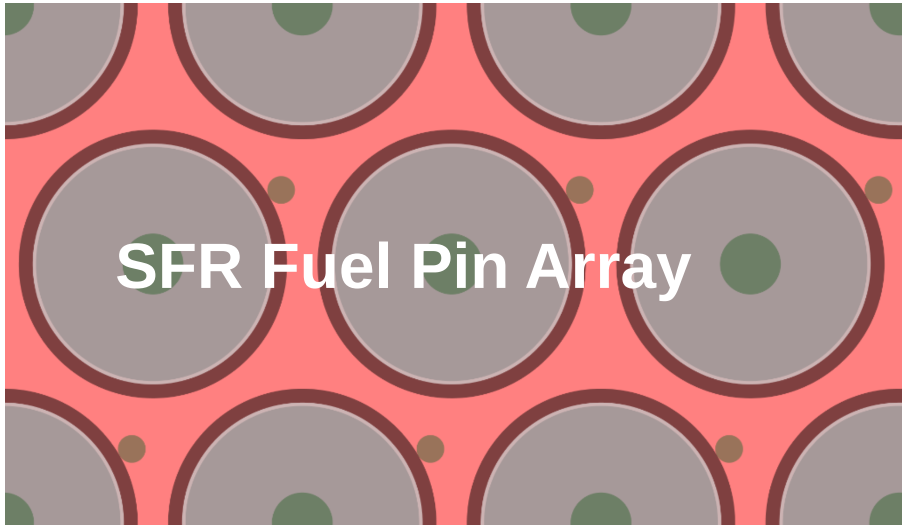
Phenix SFR Fuel Sub-Assembly



Fuel S/A: Pin Arrangement

	PWR/BWR	LMFNS
Fuel Pin/Rod OD, mm	9 - 14	6 - 9
Cladding Wall, mm	0.6 - 1	~0.5
Fuel Pellet Diameter, mm	7 - 10	5 - 7
Pitch-to-Diameter Ratio	1.4 - 1.6	1.1 - 1.3
Fuel Fraction	15 - 30 %	40 - 50 %
Coolant Fraction	50 - 70 %	35 - 50 %





Large Fuel Fraction:

- Triangular Array (in HexCan)
- Smaller P/D Ratio (in SFRs)
  - Cannot use grid spacers in SFRs
  - >> wire wrap
  - Both wire and grid spacers in LFRs

Sodium Properties: *several advantages*

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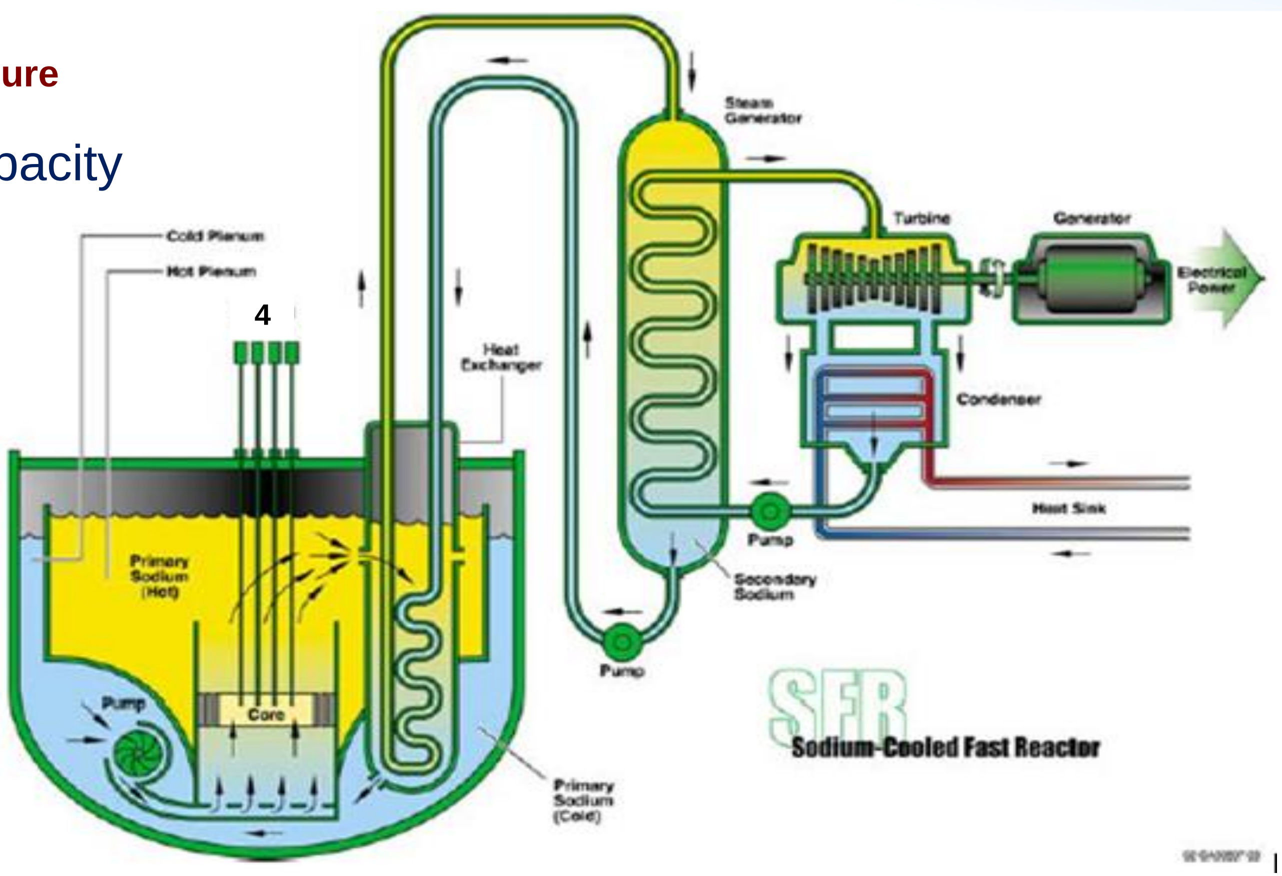
- Low melting point (97.8° C at 1 bar)
- Large temperature range of the liquid phase (97.8° C – 881.5° C at 1 bar)
- Low saturation vapor pressure
- Low density and viscosity

Primary system at atmospheric pressure

- Very high thermal conductivity and good heat capacity
- Excellent electrical conductivity
- Low activation and no alpha emitters
- No specific toxicity
- Cheap and largely available
- Perfectly compatible with steels
- Very limited amount of particles in sodium
- Low oxygen and hydrogen solubility
- Very good wetting



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Sodium Properties: *three main disadvantages*

- **Important: Violent reaction with water**
  - ✓ possible deleterious effects in Steam Generator Units (SGU), in case of pipe rupture
  - ✓ Na-H_2O interaction must be avoided or mitigated by design
    - Selection of a modular SGU
  - ✓ Na-H_2O interaction must be detected,
    - Thanks to the production of hydrogen
    - Risk of hydrogen explosion has to be mitigated

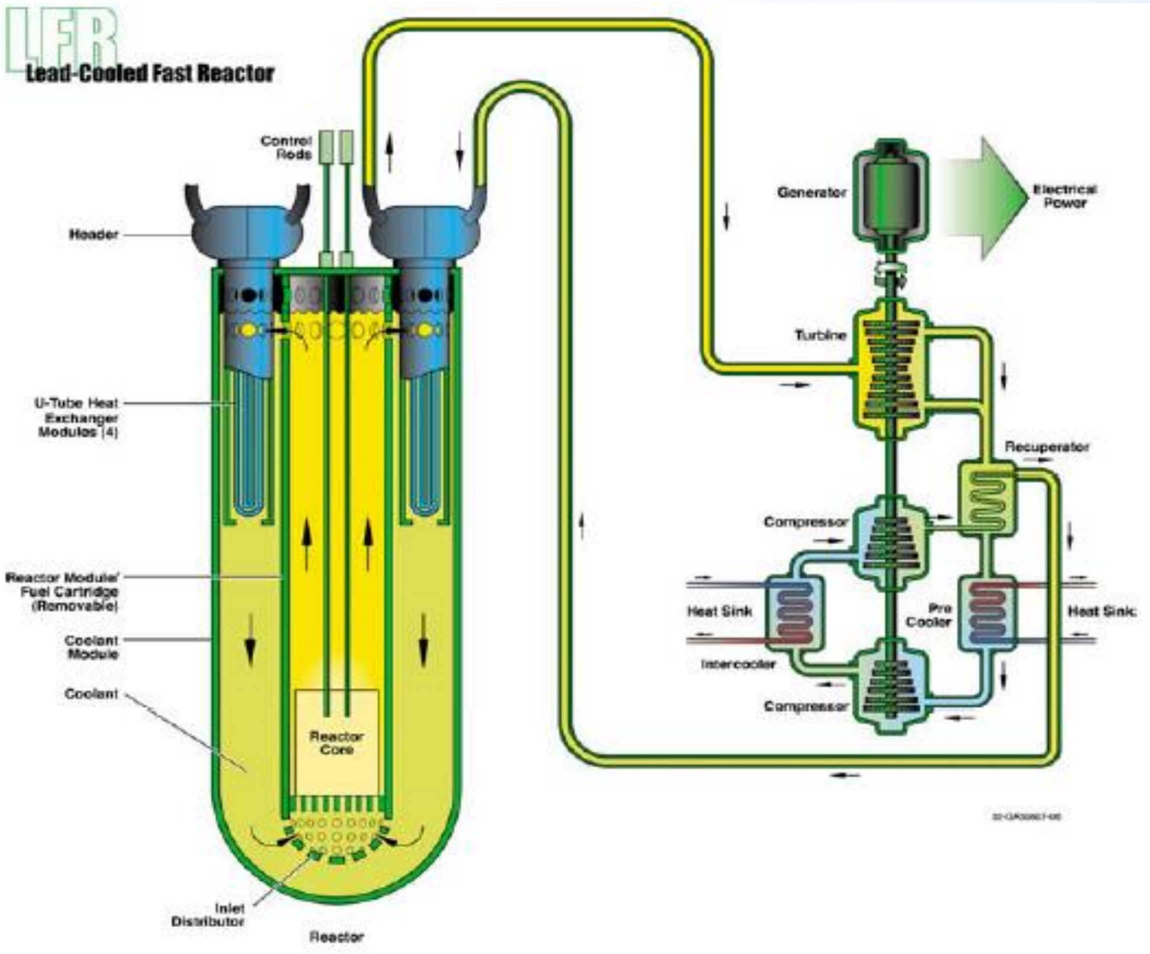
- **Important: Chemical reactivity with air**
  - ✓ Can induce Na fire
  - ✓ Need inert zones and confinement
  - ✓ Need early detection
- **Opacity**
  - ✓ Need specific equipments for under-sodium viewing and measurements

Lead/LBE Properties: *several advantages*

- Low absorption and elastic scattering cross-sections (neutrons just diffuse in lead)
- Effective gamma-rays shielding
- High retention of fission products
- High boiling point (1749/1670 °C at 1 bar)
- Very low vapor pressure
- High thermal capacity

Primary system at atmospheric pressure (applies to the two items above)

- Good heat transfer properties
- Chemically inert, in particular with water and air (allows elimination of intermediate circuit)
- No hydrogen formation
- Cheap and largely available



Lead/LBE Properties: *three main disadvantages*

- **Material compatibility: erosion, corrosion**
  - ✓ *Low coolant velocity*
  - ✓ *Limit in cladding Tmax*
  - ✓ *Hydrogen and oxygen control*
  - ✓ *New steels*
  - ✓ *Coatings*

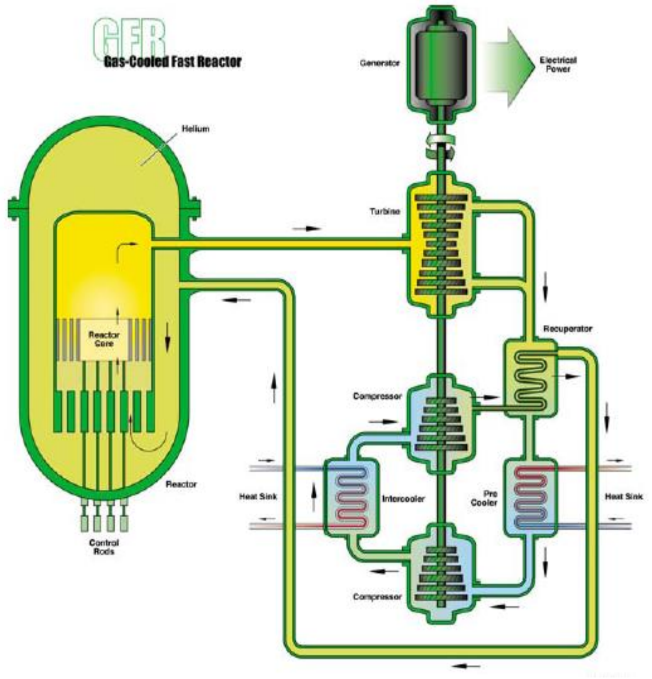
- **High density** (*also an advantage due to reduced risk of re-criticality in case of core melting*)

- **Opacity**
  - ✓ *Need specific equipment for under-lead viewing and measurements*

Very limited **operational experience** (Alpha-class submarines)

Gas (He) Properties: *advantages*

- Completely transparent to neutron (very hard neutron spectrum)
- Low reactivity insertion due to voiding of the coolant
- Chemically inert
- Single phase behavior
- Optical transparency
- Electrically non-conducting
- Possibility to adopt direct gas turbine cycle
- Very high temperature applications



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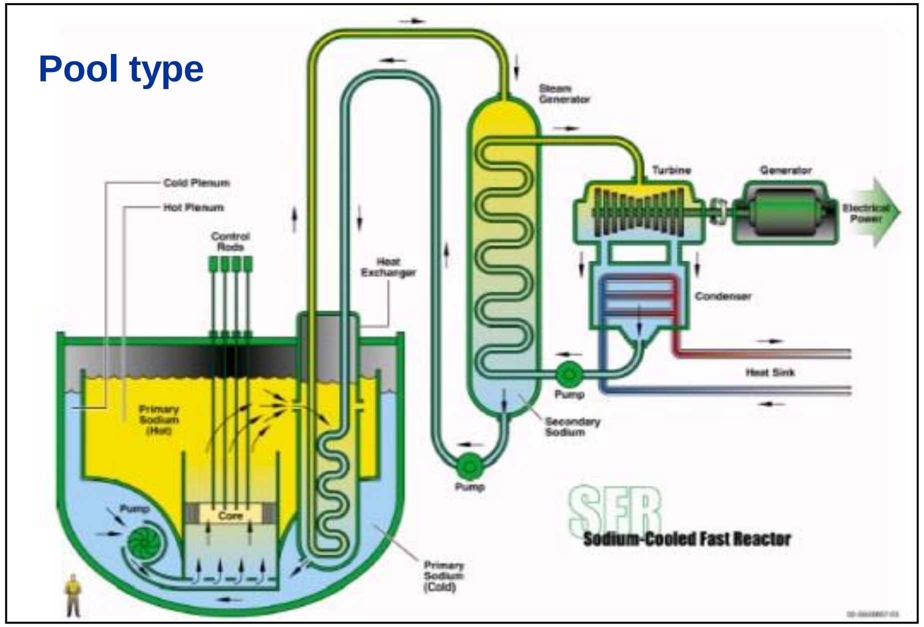
Gas (He) Properties: *four main disadvantages*

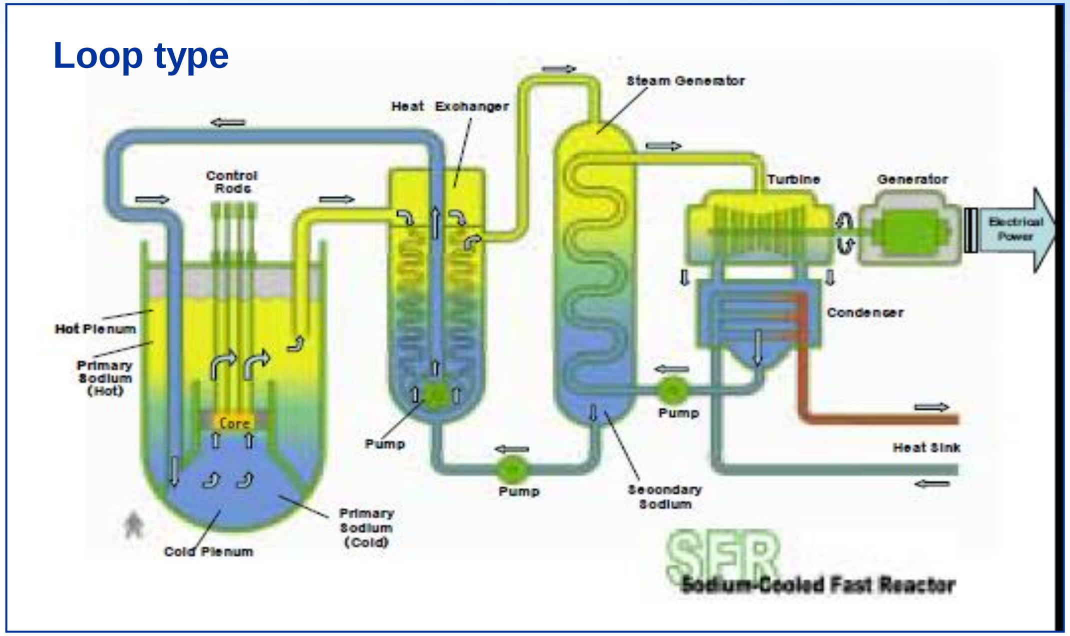
- **Low density creating requirement for pressurization**
  - *Likelihood and severity of a LOCA*
- **Inability to adopt a pool configuration**
  - *Core remains uncovered in case of breached primary circuit*
- **Non-condensable**
  - *Pressure loading the containment building in case of LOCA*
- **Low-thermal inertia**
  - *The reactor core heat up rapidly if forced cooling is lost*

No operational experience

Sodium Cooled Fast Reactor (SFR)

GIF website: www.gen-4.org



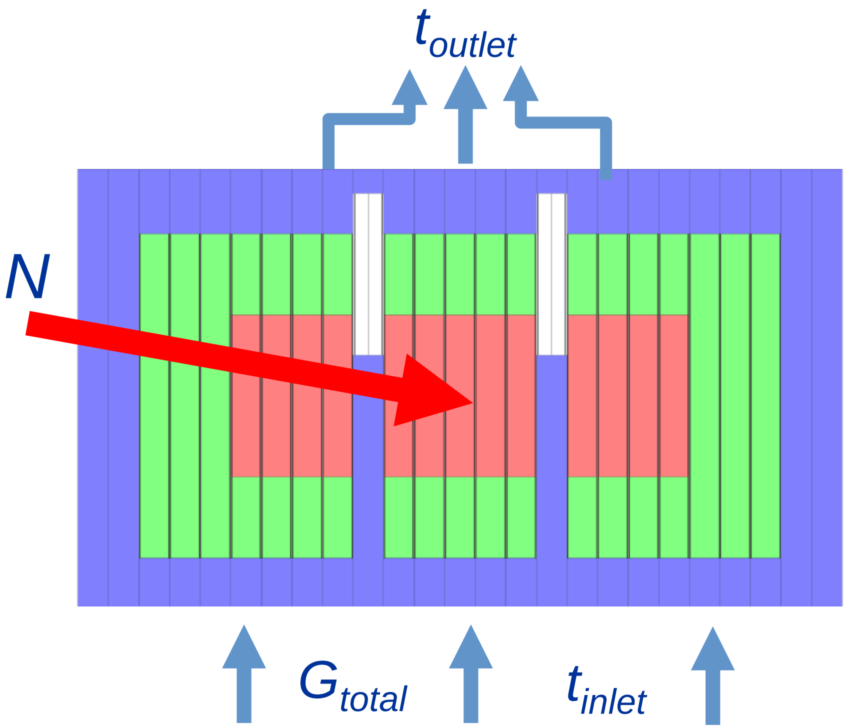


- ✓ Relatively low melting point; relative high boiling point: 97.8˚ .. 881.5˚ C at 1 bar
- ✓ Low density and viscosity
- ✓ Very high thermal conductivity and good heat capacity
- ✓ Excellent electrical conductivity
- ✓ Low activation and no alpha emitters
- ✓ Cheap and largely available
- ✓ Perfectly compatible with steels

- — Aggressive chemical **reaction with water**
- — Reaction with **air**: self-ignited **sodium fires**
- — Void reactivity effect
- — Not transparent: Need special equipment for control and inspections

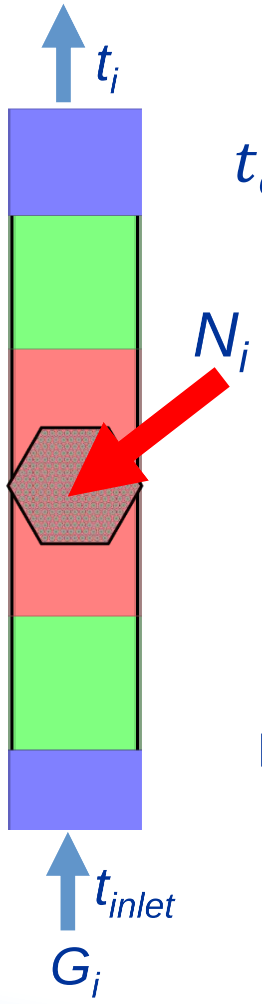
Reactor Core Power Balance

Core: $N = GC_p(t_{outlet} - t_{inlet})$



G_{total}	Total Flowrate through the Core, kg/s
t_{inlet}	Core Inlet Temperature, C
t_{outlet}	Bulk Outlet Core Temperature, C
N	Reactor Thermal Power, W

S/A: $N_i = G_i C_p(t_i - t_{inlet})$



$$t_{outlet} = \frac{\sum G_i t_i}{G}$$

$$N = \sum N_i$$

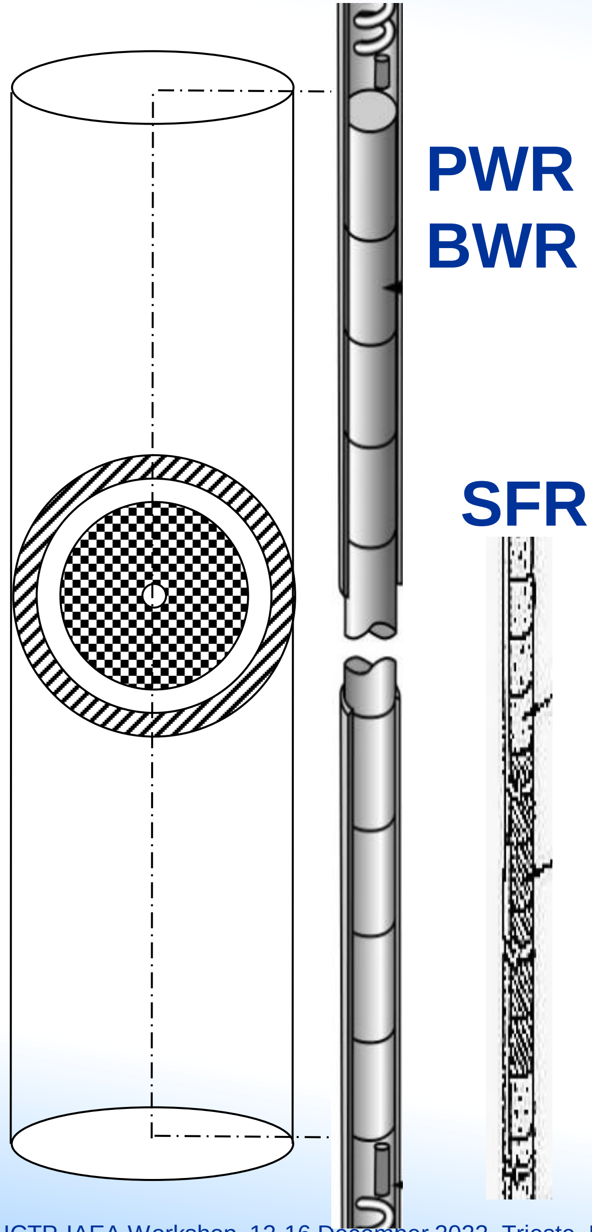
$$G = \sum G_i$$

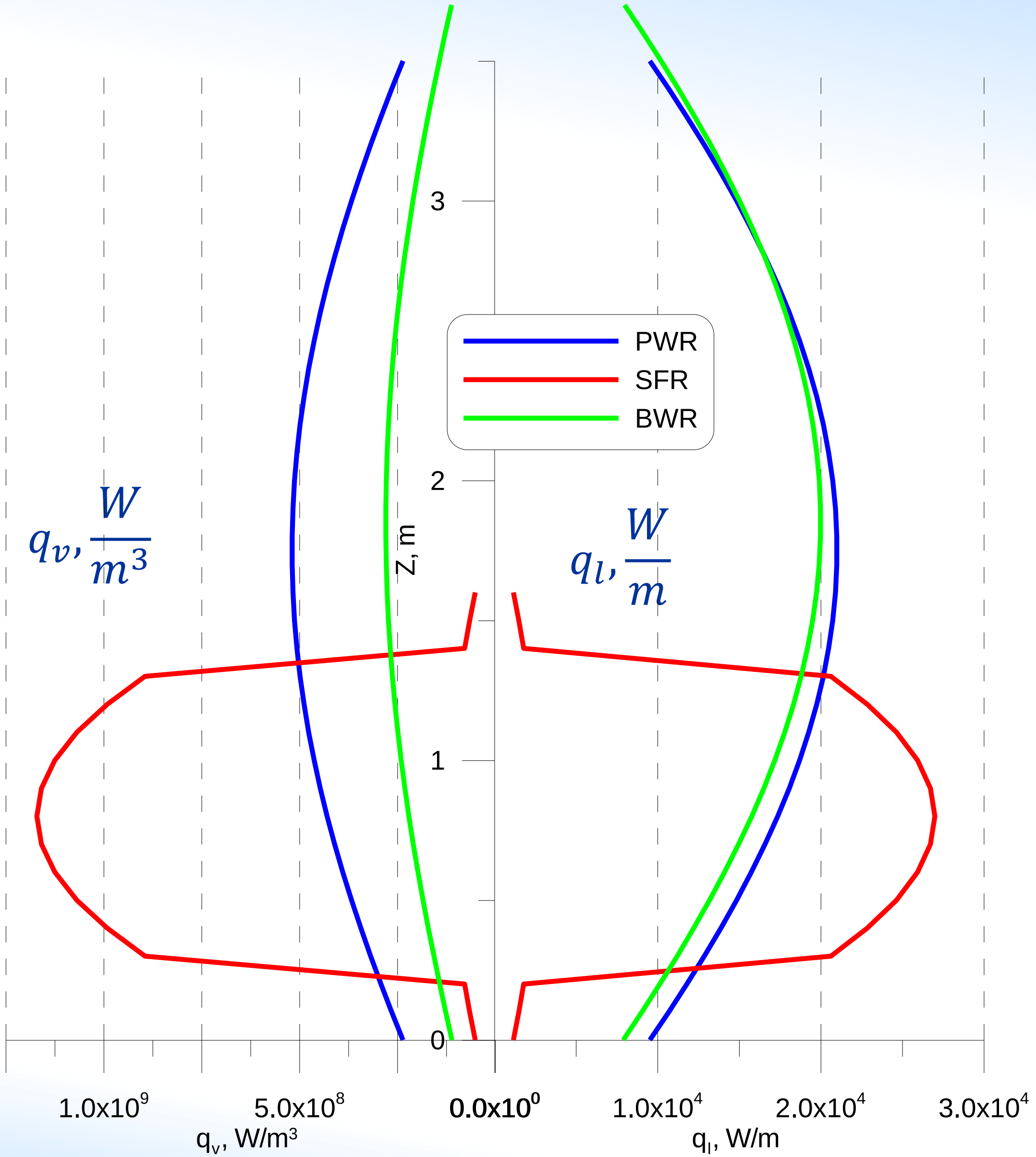
Power Density

$$q_l = \frac{dN}{dz}$$

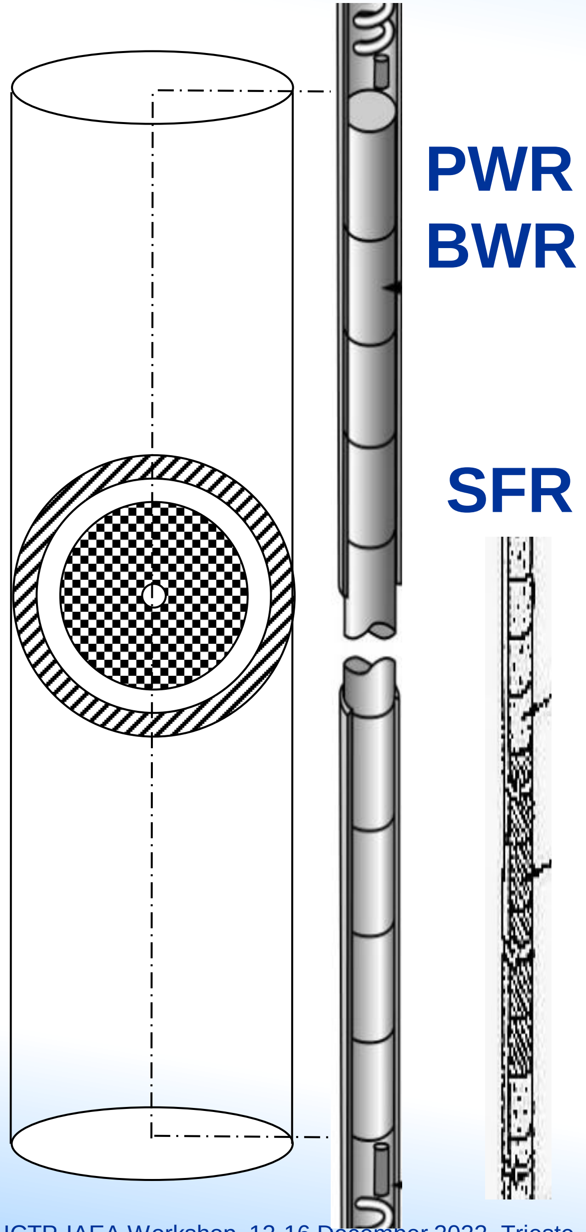
$$q_v = \frac{dN}{dV}$$

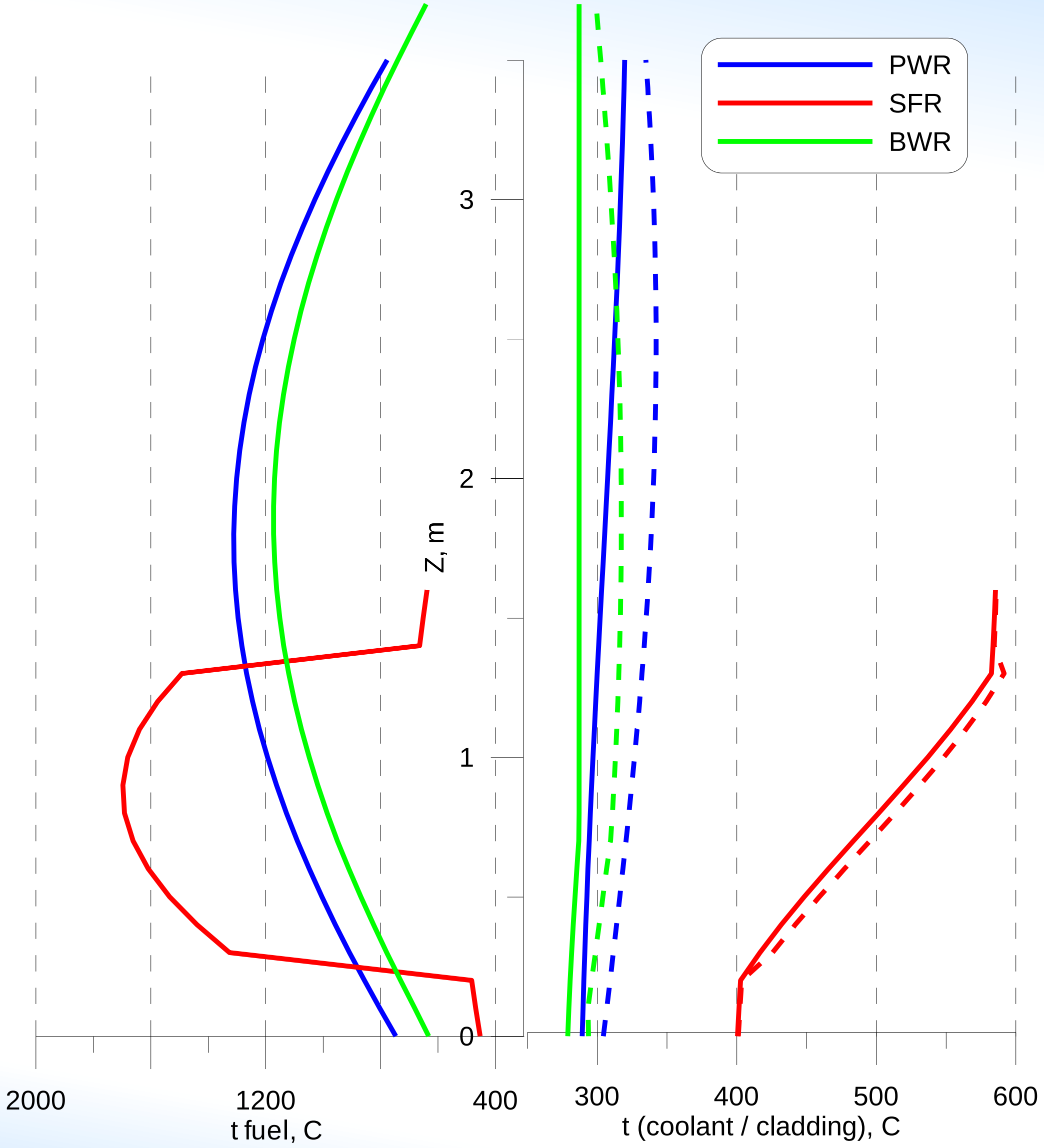
Fuel Pin: Power Density



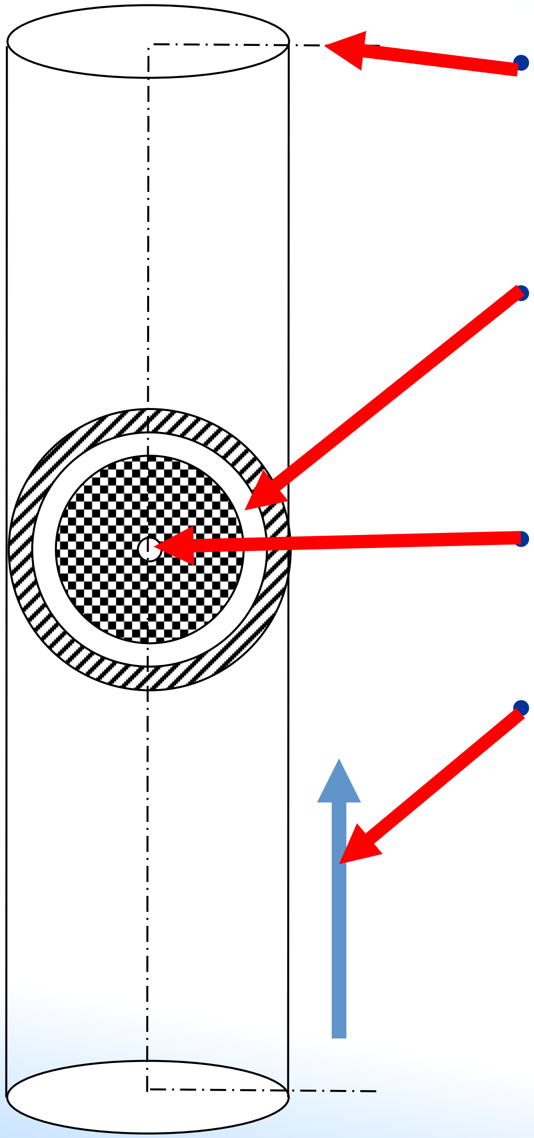


Fuel Pin: Temperature Profiles



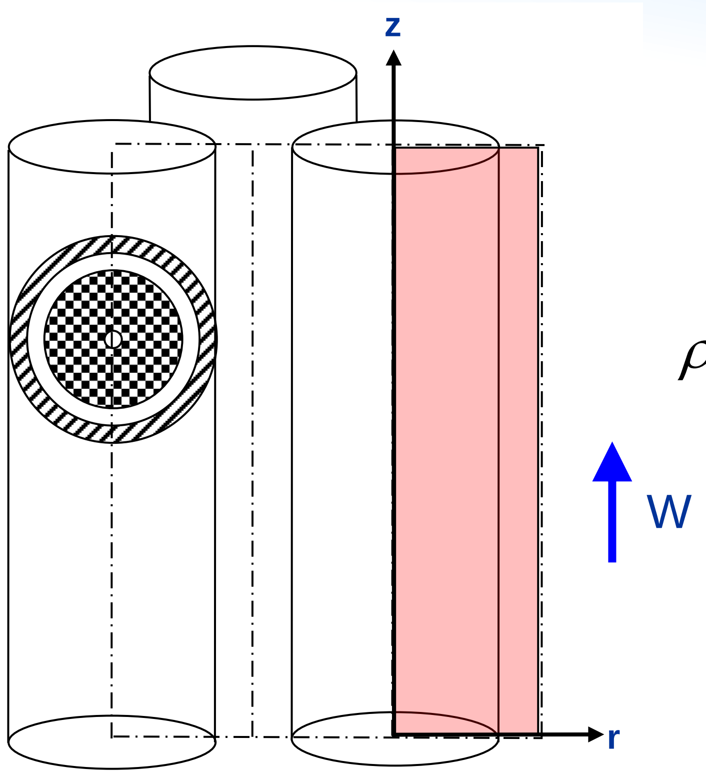


TH Limiting Parameters



- Maximal Coolant Temperature
  - Below Boiling Point (at least)
    - (ex: BWR, SCWR)
- Maximal Cladding Temperature
  - Zr: < 350 °C (< 1000°C under accident conditions)
  - SS: < 700 °C (< 1000°C under accident conditions)
- Maximal Fuel Temperature
  - Below Melting Point (< 2700C)
- Maximal Coolant Velocity
  - To prevent erosion and vibration problems
  - To minimize pressure drop in the core (pump power)
  - H_2O, Na: < 10 m/s
  - Lead, LBE: < 5 m/s

Governing Equations



Hear Conduction in Clad, Gap, and Fuel Pellet

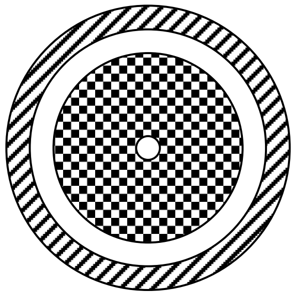
$$\rho c_p \frac{\partial t}{\partial \tau} = \frac{1}{r}\frac{\partial}{\partial r}\left(\lambda(t) r \frac{\partial t}{\partial r}\right) + \frac{\partial}{\partial z}\left(\lambda(t)\frac{\partial t}{\partial z}\right) + q_v$$

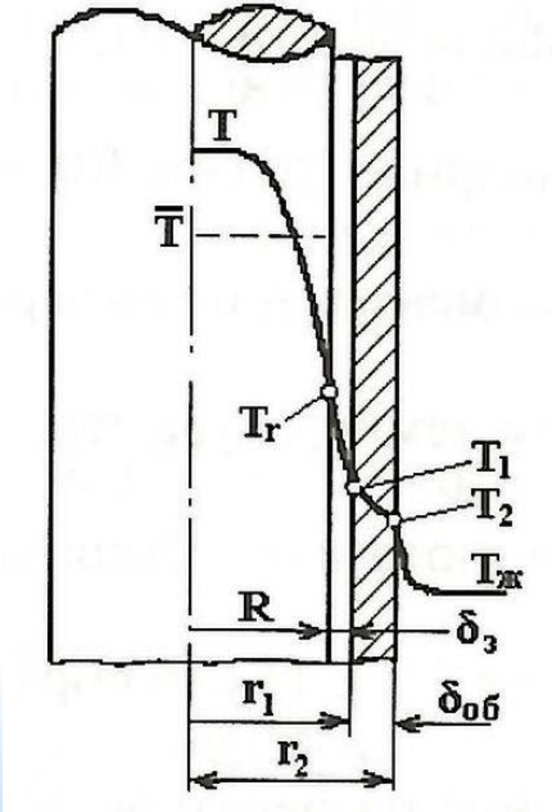
(3D Effects are neglected)

Energy Conservation in Coolant

$$\rho c_p \frac{\partial t}{\partial \tau} + \rho c_p W(r)\frac{\partial t}{\partial z} = \frac{1}{r}\frac{\partial}{\partial r}\left(\left(\lambda + \lambda^r_{turb}(r)\right) r \frac{\partial t}{\partial r}\right) + \frac{\partial}{\partial z}\left(\left(\lambda + \lambda^z_{turb}(r)\right)\frac{\partial t}{\partial z}\right)$$

Steady Temperature Profiles: Inside Pin





$$\rho c_p \frac{\partial t}{\partial \tau} = \frac{1}{r}\frac{\partial}{\partial r}\left(\lambda(t) r \frac{\partial t}{\partial r}\right) + \frac{\partial}{\partial z}\left(\lambda(t)\frac{\partial t}{\partial z}\right) + q_v$$

- No transient term
- Axial heat conduction can be neglected

Easy to Solve in 1D (Analytically)

$$t_{\max}(z) = t_{coolant}(z) + \Delta t_{colant} + \Delta t_{clad} + \Delta t_{gap} + \Delta t_{fuel}$$

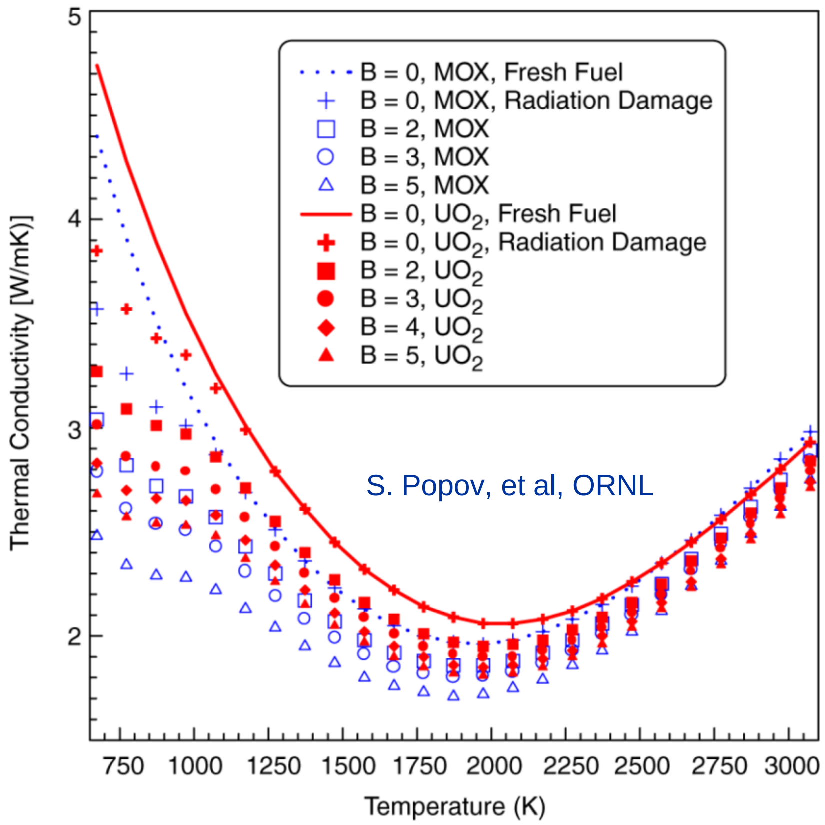
$$t_{coolant}(z) = t_{inlet} \int_{-h/2}^{z} c_p G_i q_l(z) dz$$

$$\Delta t_{coolant} = \frac{q_l(z)}{\alpha \pi d_{pin}} \qquad \Delta t_{clad} = \frac{q(z)\Delta_{clad}}{\lambda_{clad}}$$

$$\Delta t_{gap} = \frac{q(z)\Delta_{gap}}{\lambda_{gap}} \qquad \Delta t_{fuel} = \frac{q_v(z) d_{fuel}^2}{16\lambda_{fuel}}$$

But…

Steady-State but Non-Linear Effects



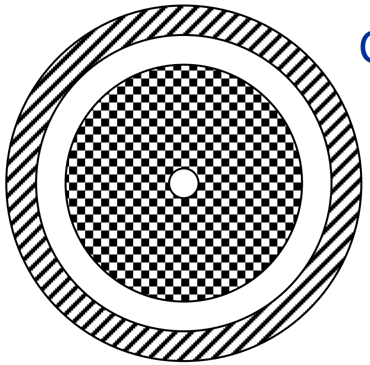
$$\rho c_p \frac{\partial t}{\partial \tau} = \frac{1}{r}\frac{\partial}{\partial r}\left(\lambda(t) r \frac{\partial t}{\partial r}\right) + \frac{\partial}{\partial z}\left(\lambda(t)\frac{\partial t}{\partial z}\right) + q_v$$

Fuel Conductivity depends on temperature

Cannot use simple relation!

$$\Delta t_{fuel} = \frac{q_v(z) d_{fuel}^2}{16\lambda_{fuel}}$$

Radiation Heat Transfer in the gap



Cannot use simple relation!

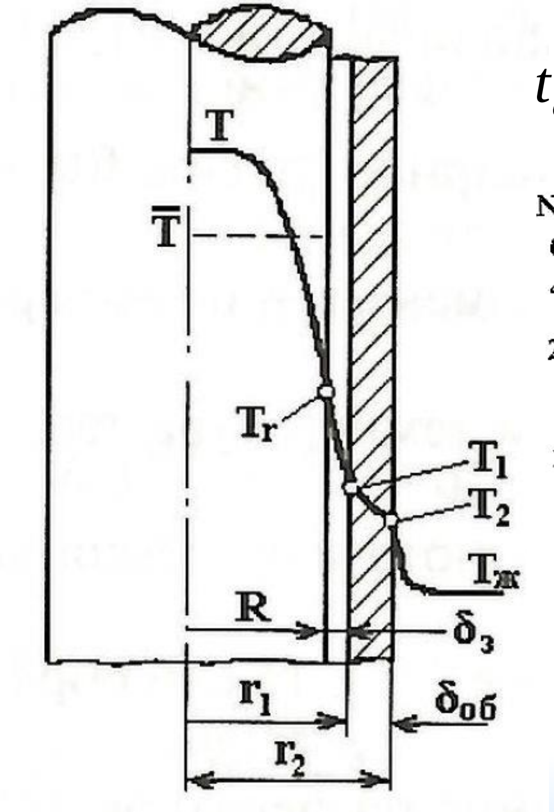
$$\Delta t_{gap} = \frac{q(z)\Delta_{gap}}{\lambda_{gap}}$$

$$q = \varepsilon_{eff}\sigma\left(T_{pellet(out)}^4 - T_{clad(in)}^4\right) + \lambda_{gap}\frac{T_{pellet(out)} - T_{clad(in)}}{\Delta_{gap}}$$

Coolant-Cladding Heat Transfer

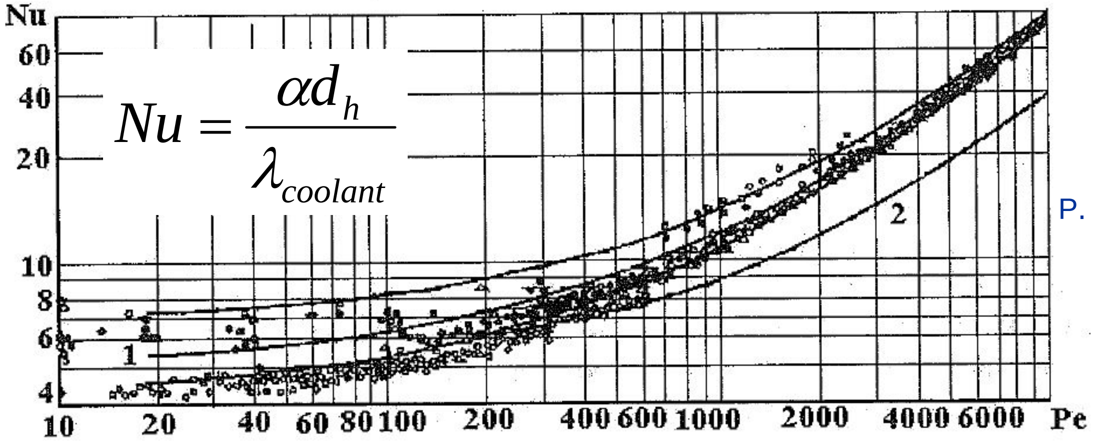
Energy Conservation in Coolant

$$\rho c_p \frac{\partial t}{\partial \tau} + \rho c_p W(r)\frac{\partial t}{\partial z} = \frac{1}{r}\frac{\partial}{\partial r}\left(\left(\lambda+\lambda^r_{turb}(r)\right) r \frac{\partial t}{\partial r}\right) + \frac{\partial}{\partial z}\left(\left(\lambda+\lambda^z_{turb}(r)\right)\frac{\partial t}{\partial z}\right)$$



$$t_{coolant}(z) = t_{inlet} \int_{-h/2}^{z} c_p G_i q_l(z)\,dz \qquad \Delta t_{coolant} = \frac{q_l(z)}{\alpha \pi d_{pin}}$$

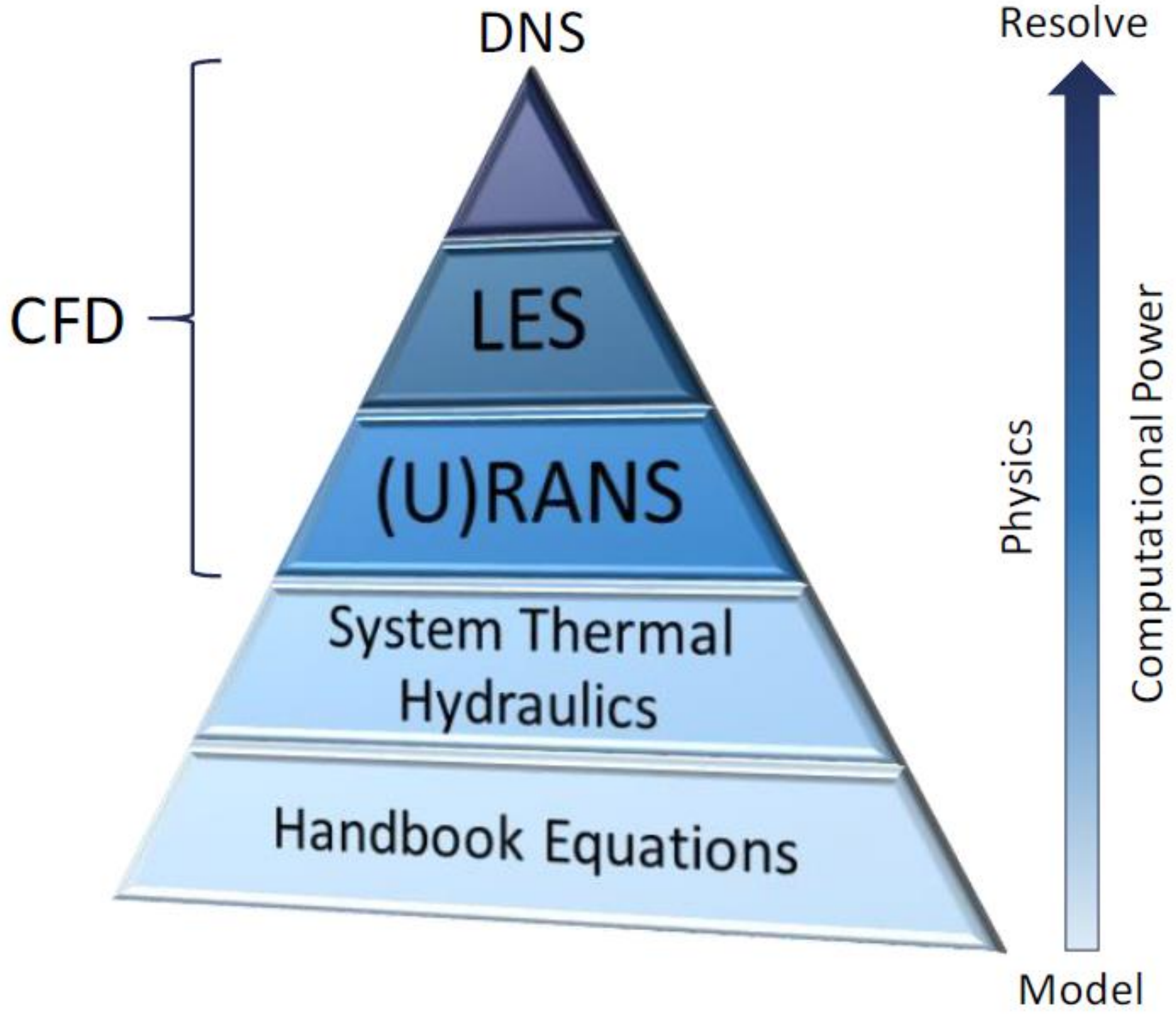
$$Nu = \frac{\alpha d_h}{\lambda_{coolant}}$$

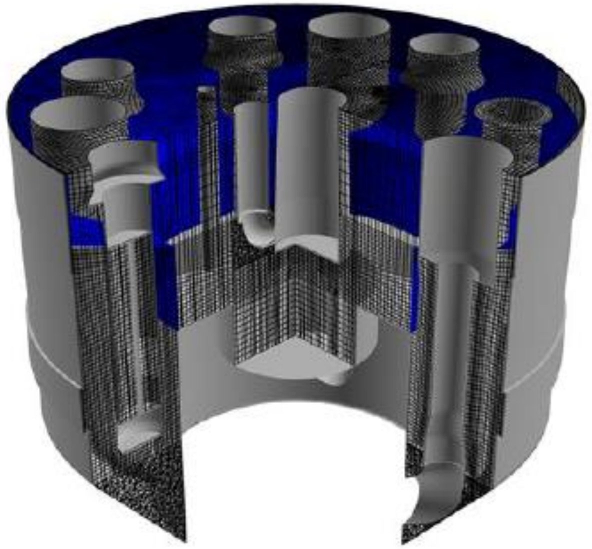


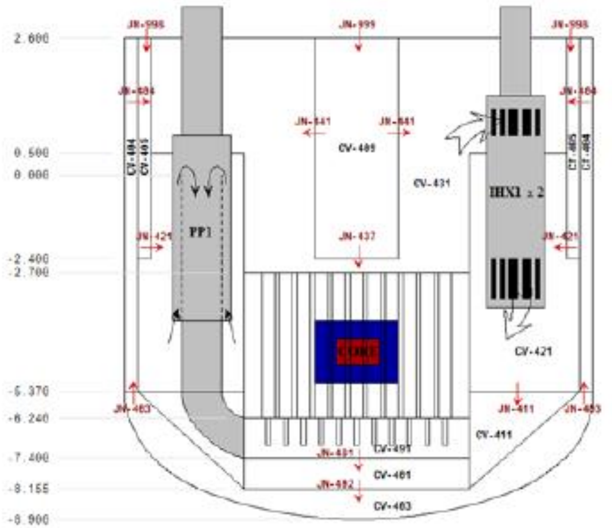
P. Kirillov, IPPE

$$Nu = 0.58\left(1.1\left(\frac{p}{d}\right)^2 - 1\right)^{0.55} Pe^{0.45} \quad (Pe = 400..4000;\ \Pr \le 0.04)$$

From simple experimental correlations to DNS







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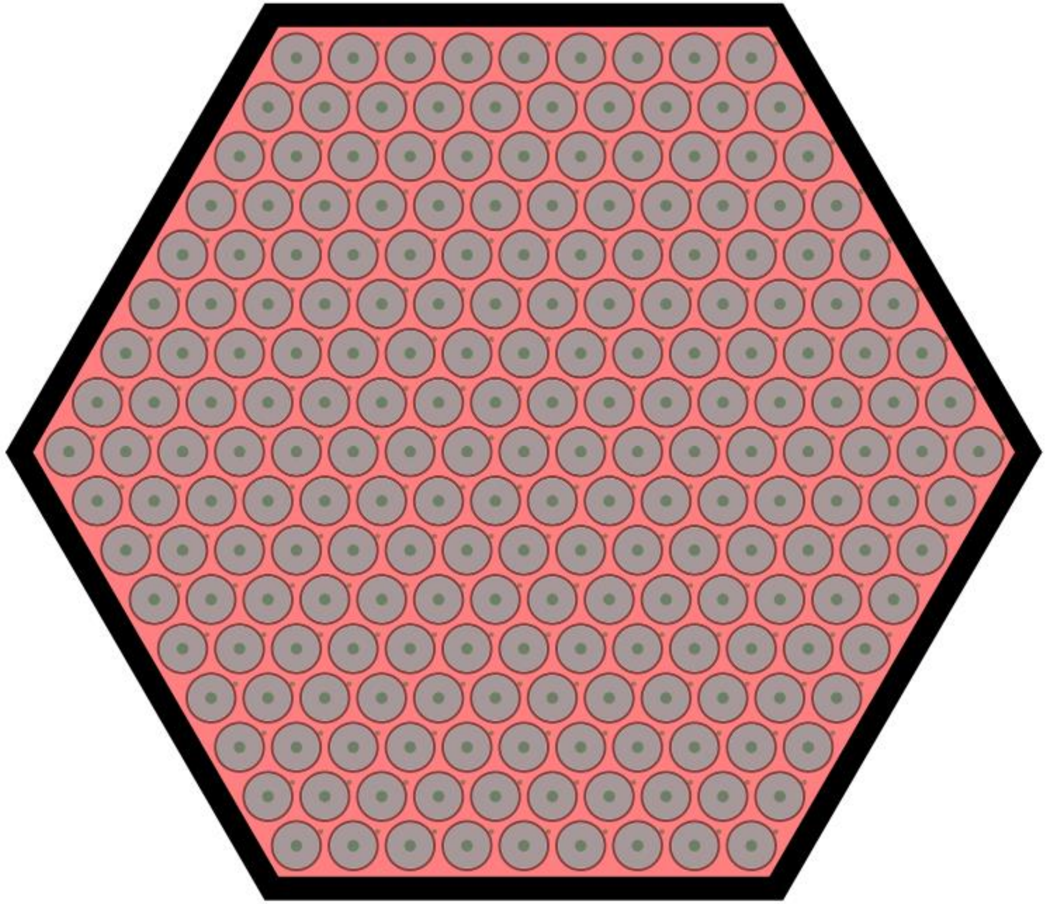
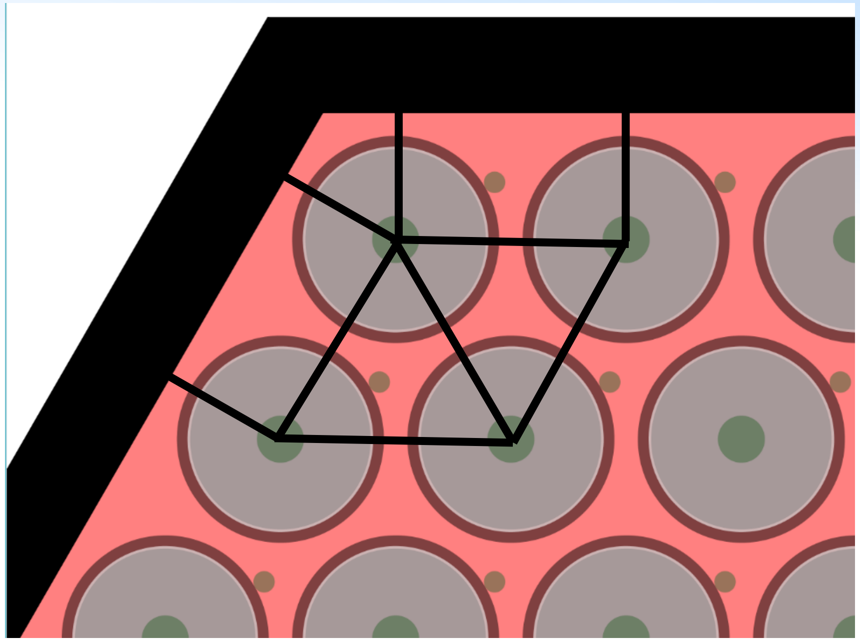
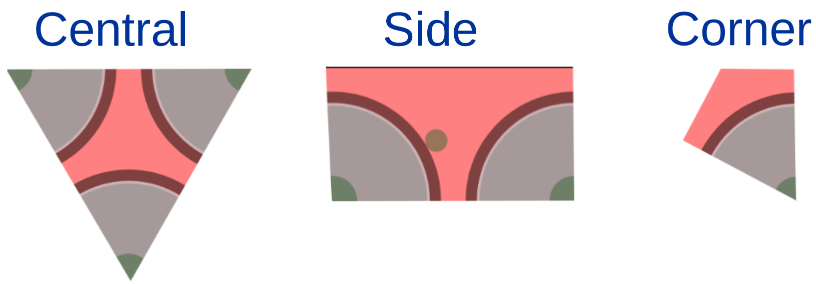
Temperature Distribution within S/A: **Subchannel Analysis**







Power-to-Flow Ratio in Subchannel

$$\frac{Power}{Flowrate} = \frac{Heat\ Flux \times \Pi}{Velocity \times Area}$$

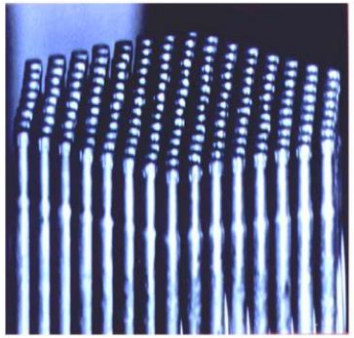
Power-to-Flow Ratio in Central Subchannel is

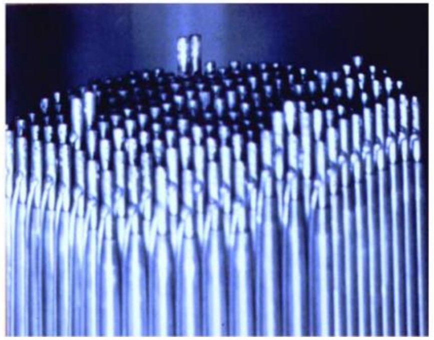
- 1.2 – 1.4 higher (if isolated)
- 1.1 – 1.15 in real S/A, thanks to mixing

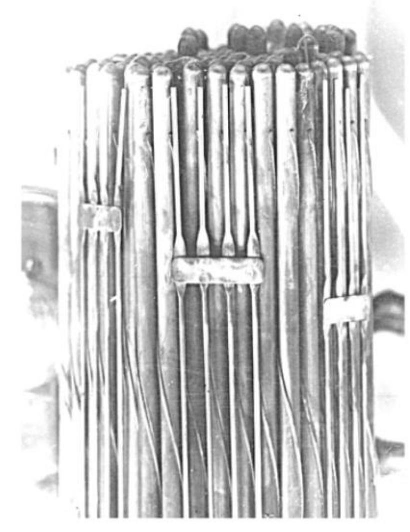
S/A Deformation Under Irradiation

IAEA

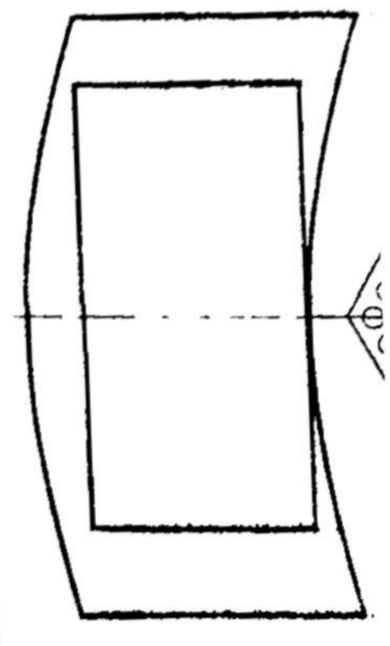
FFTF





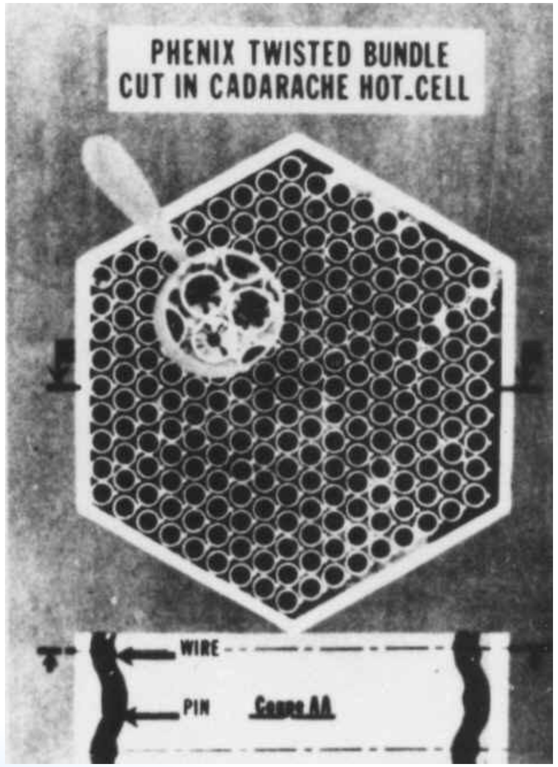


BN-600

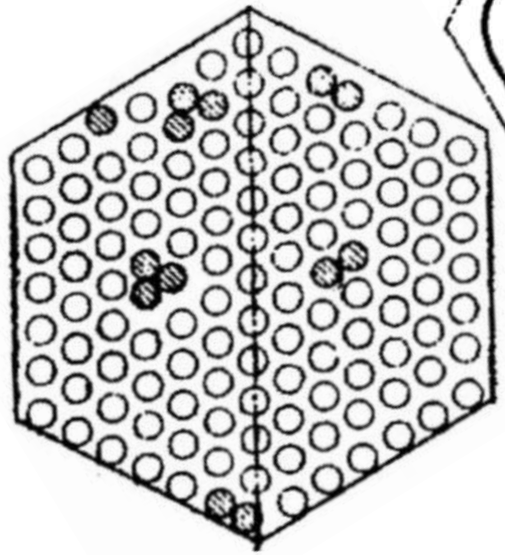


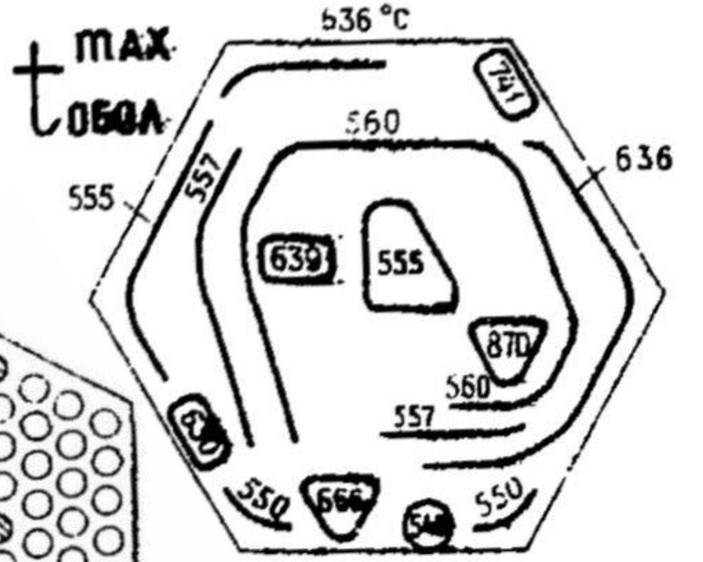
Axial Deformation

Phenix



Random Deformation Inside S/A

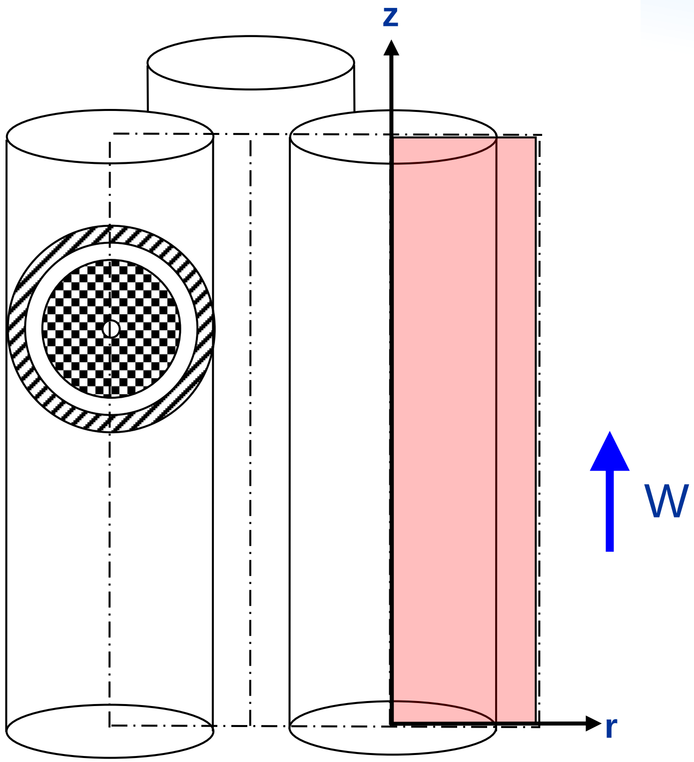




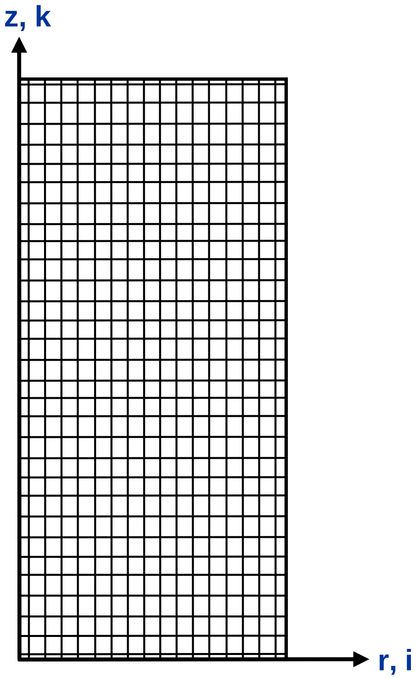
Max Clad Temperature, C

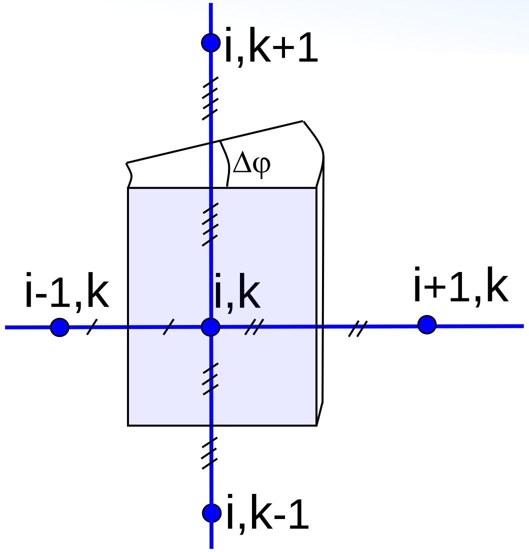
A.Sorokin, et al, IPPE

Numerical Simulation (CFD)



Finite-Difference Approximation on Mesh





$$\rho c_p W(r)\frac{\partial t}{\partial z} = \frac{1}{r}\frac{\partial}{\partial r}\left(\left(\lambda + \lambda^{r}_{turb}(r)\right) r\frac{\partial t}{\partial r}\right) + \frac{\partial}{\partial z}\left(\left(\lambda + \lambda^{z}_{turb}(r)\right)\frac{\partial t}{\partial z}\right) + Q_v$$

$$a^1_{ik} t_{i,k} + a^2_{ik} t_{i-1,k} + a^3_{ik} t_{i+1,k} + a^4_{ik} t_{i,k-1} + a^5_{ik} t_{i,k+1} + a^6_{ik} = 0$$

TH Analysis: at Nominal Power

- Core Design Verification Calculations
  - For the given core design and power, to check if temperatures and velocities are below the limits
  - Input
    - Core Design, S/A and Pin Geometry
    - Max Pin or S/A Power (number of pins/SA) (from Reactor Power Distribution)
    - Axial Power Profile (or peaking factor)
    - Inlet Coolant Temperature
    - Coolant Velocity or Flowrate/SA
  - Output
    - Outlet Coolant Temperature
    - Maximal Cladding Temperature (or Distribution)
    - Maximal Fuel Temperature (or Distribution)

TH Analysis: Max Nominal Power

- Design Study Calculations
  - For the given core configuration, what can be a maximal pin/SA/core thermal power?
  - Input
    - Core Design, S/A and Pin Geometry
    - Inlet Coolant Temperature
    - Axial and Radial Power Profiles (or peaking factors)
  - Output
    - Max Pin or S/A Power; Total Reactor Power

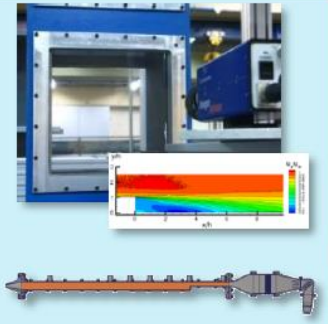
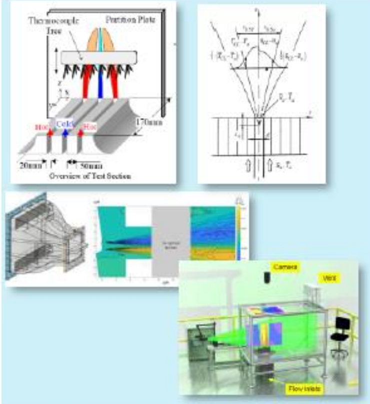
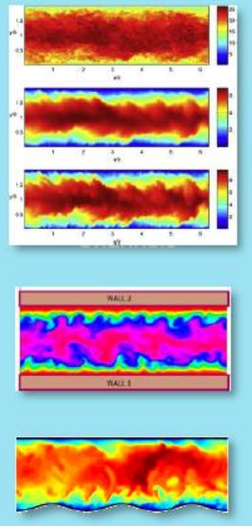
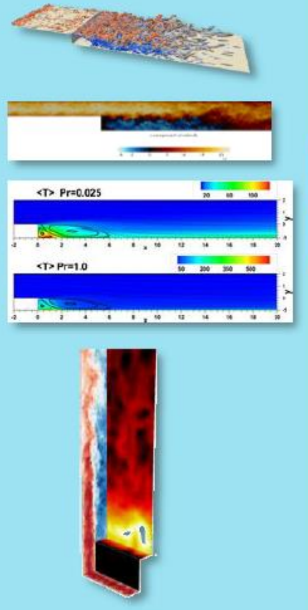
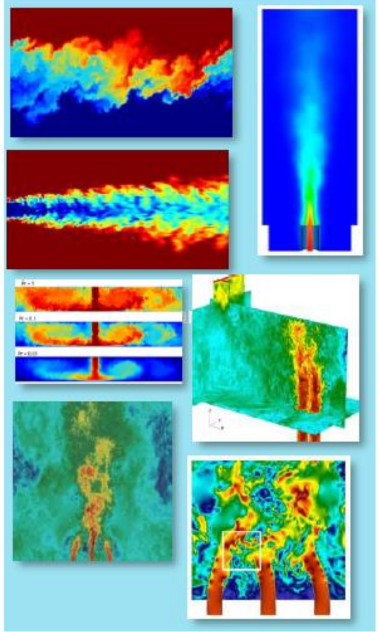
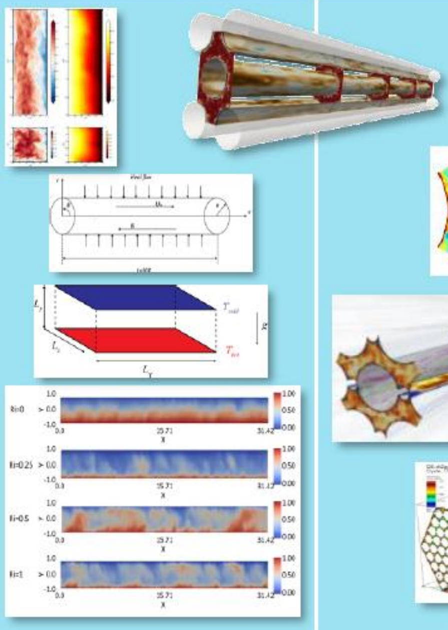
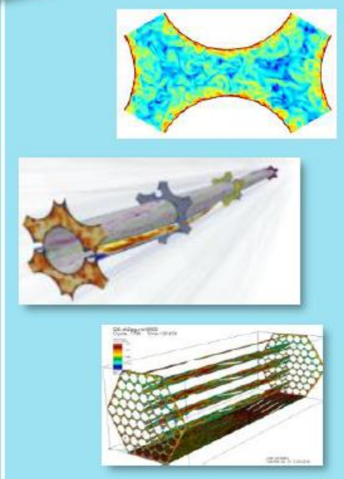
TH Analysis: Transients

Reactor Accidental Transient Scenarios

- DBC (Design Basis Condition) accidents
  - *Reactor shut-down normally (Protected)*
  - Drop/Release of Single Control Rod
  - Loss of one or all primary pumps
- DEC (Design Extension Conditions) accidents
  - *Severe Accidents, May Result in Core Melting*
  - **ULOF** (Unprotected Loss of Flow)
    - For LMFNS, ULOF is considered as most serious accident
  - **UTOP** (Unprotected Trip of Power)
    - Drop/Release of Control Rod Bank
  - Core Flow Blockage (incl. **TIB** – Total Instantaneous Blockage)
    - May results in core melting/damage. Simulations should reject/confirm the possibility of propagation
  - Loss of Heat Sink (**LOHS**)
- **Required**: Coupling TH/Neutronics/Mass Transfer/EOS

CFD Analysis: Basic Flows

IAEA

	Channels	Flow Separation	Jets	Mixed Convection	Rod Bundle
Experiment					
High Fidelity Reference Simulation (LES/DNS)					

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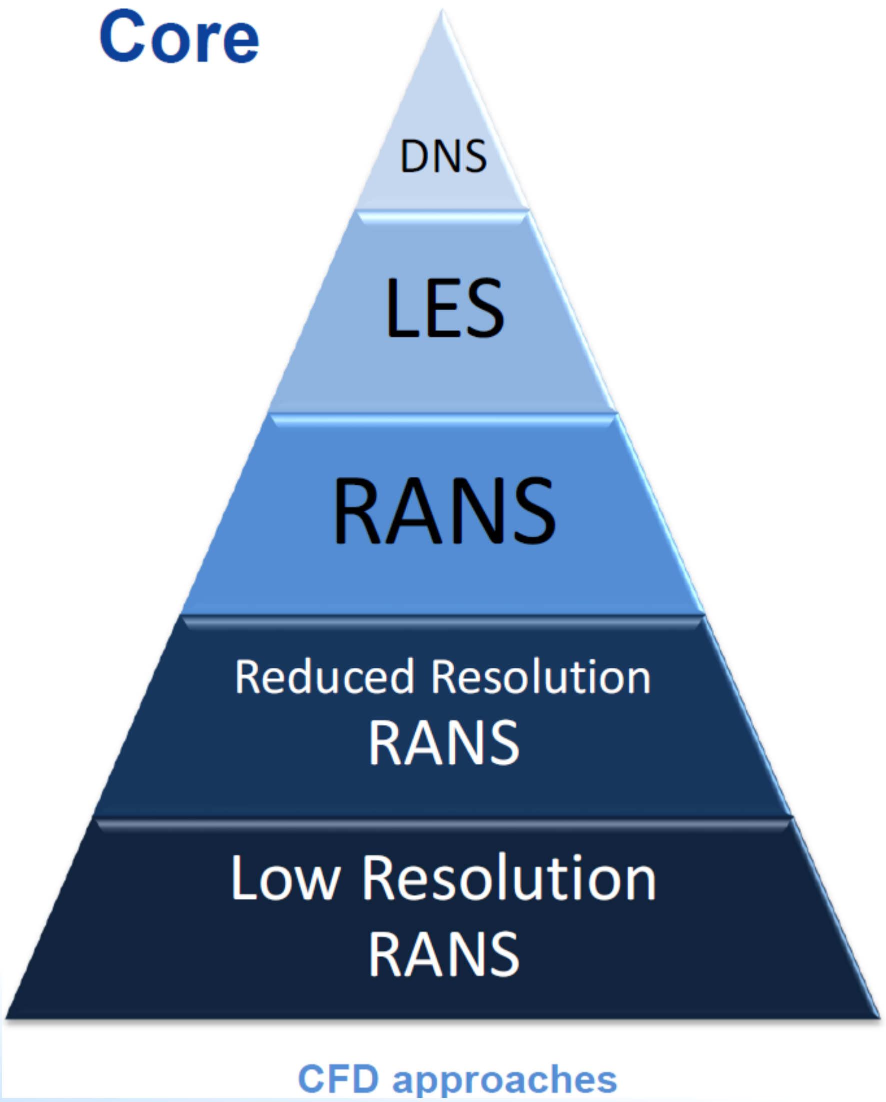
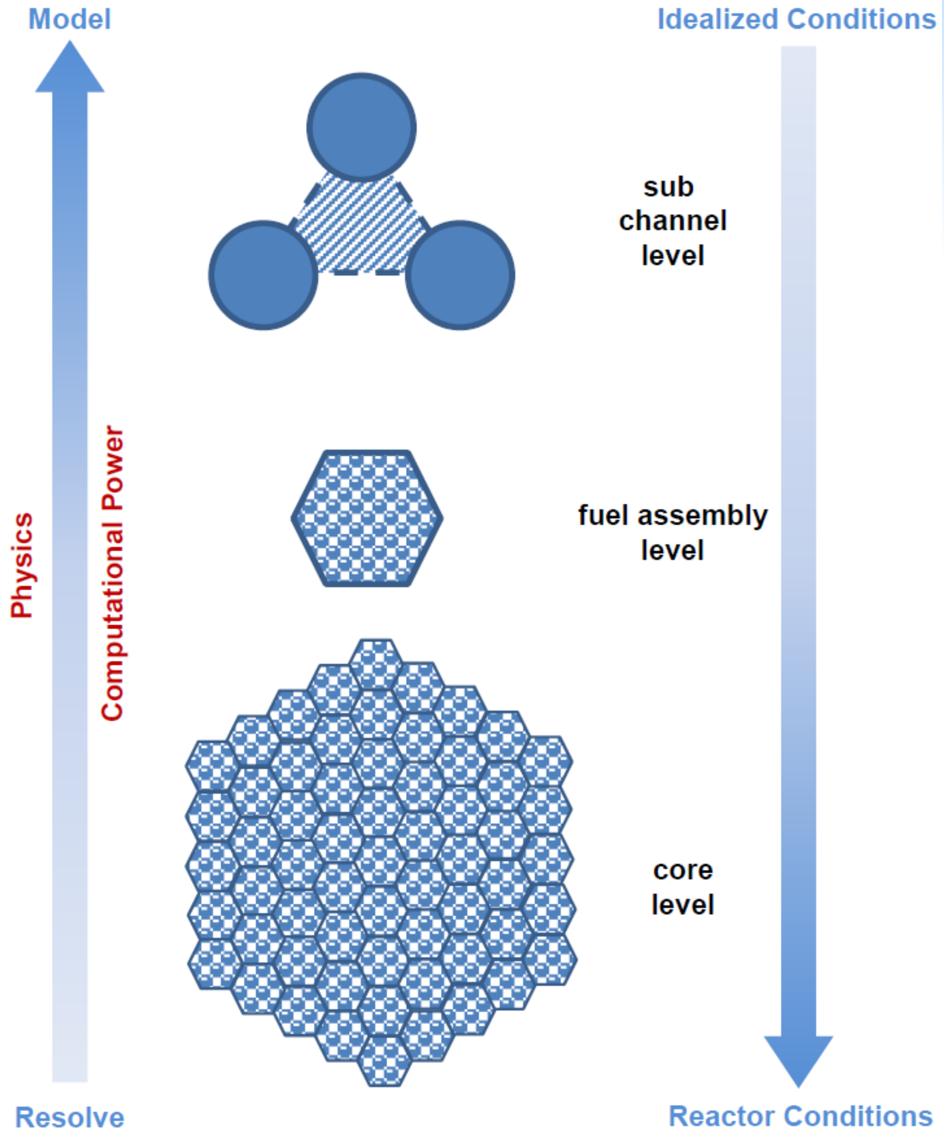
CFD Analysis: Reactor Core





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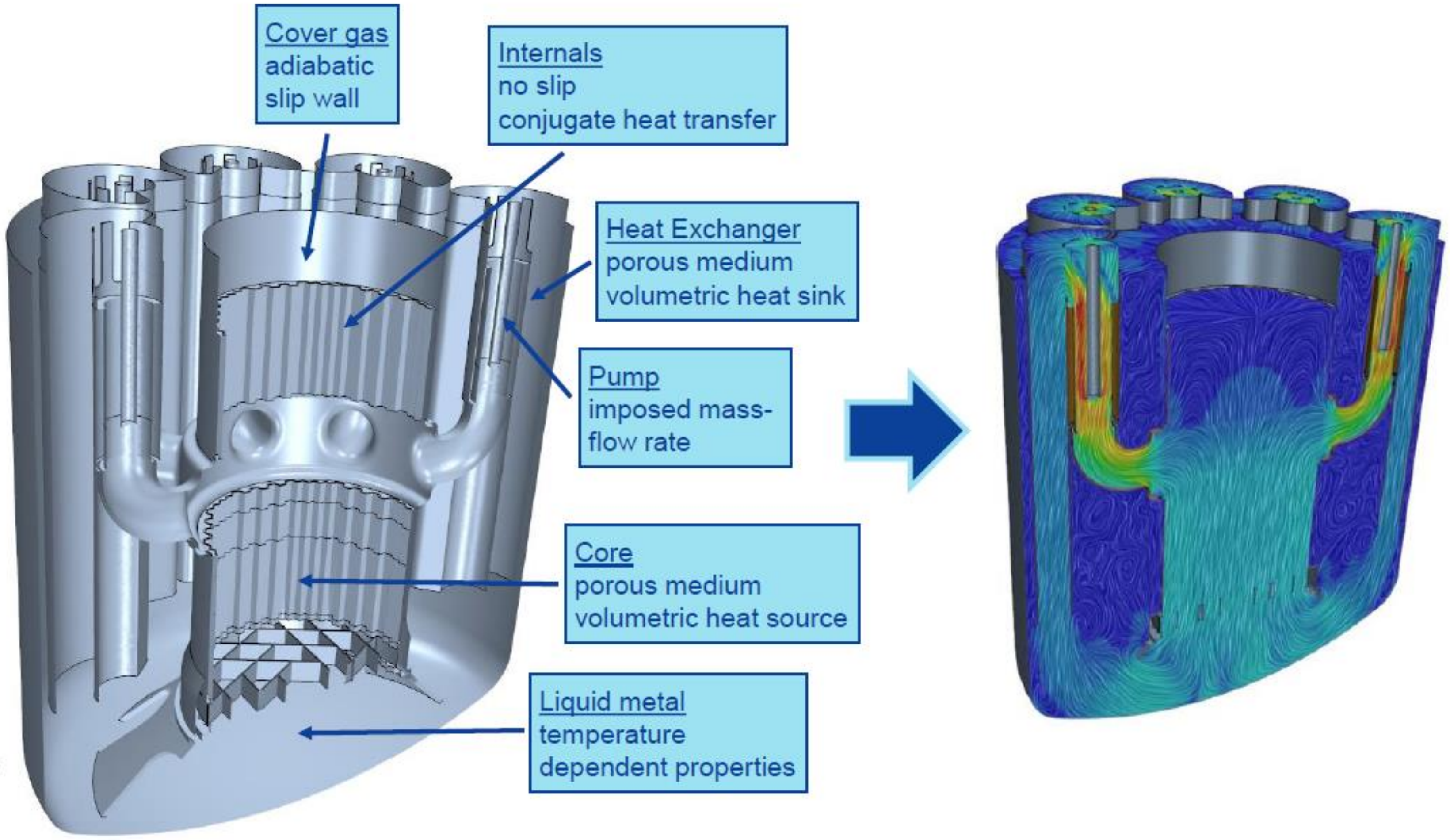
CFD Analysis: LMFR Vessel and Pool

- Modelling strategy for various components:
  - Cover gas
  - Internals
  - Heat exchangers
  - Pumps
  - Core
  - Liquid metal
- Experimental validation:
  - CIRCE
  - ESCAPE
- Application
  - SEALER designs
  - ALFRED



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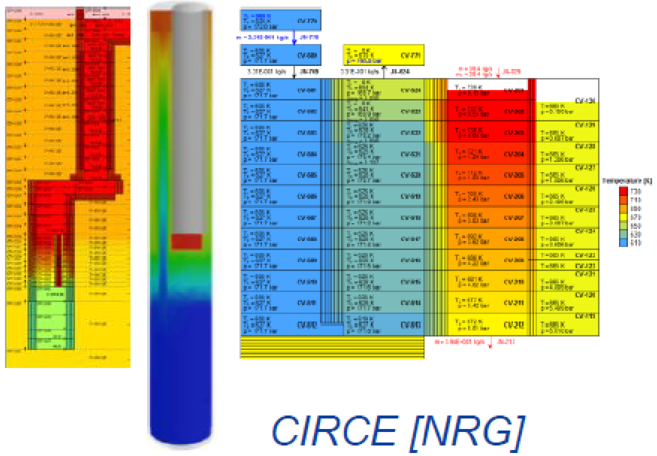
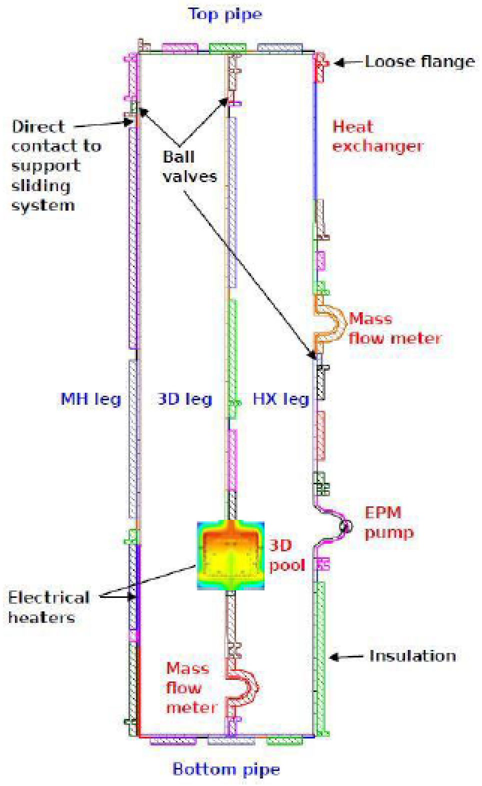
System Thermal Hydraulics

- System thermal hydraulics codes are nice tools but… they lack details which may be essential (e.g. 3-dimensional effects)
- Details can be modelled in CFD
- CFD is a nice tool but…computationally very expensive

- Still there is a need of a tool which can combine sufficient details and reasonable computational costs
  - 3-D modules in system codes
  - Multi-scale modelling through coupled CFD – System Thermal Hydraulics codes



CIRCE [NRG]



TALL-3D [KTH]

NRG

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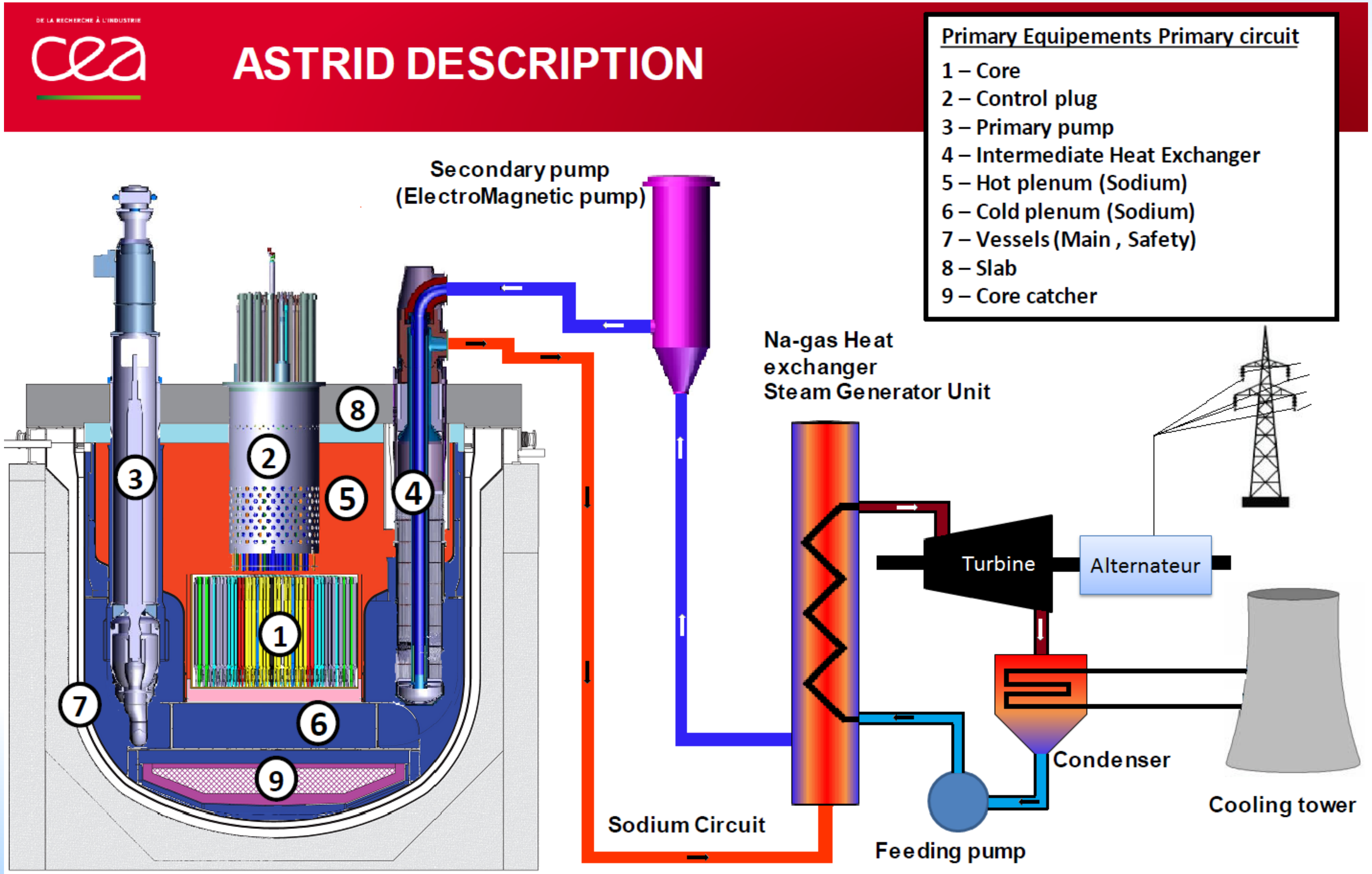
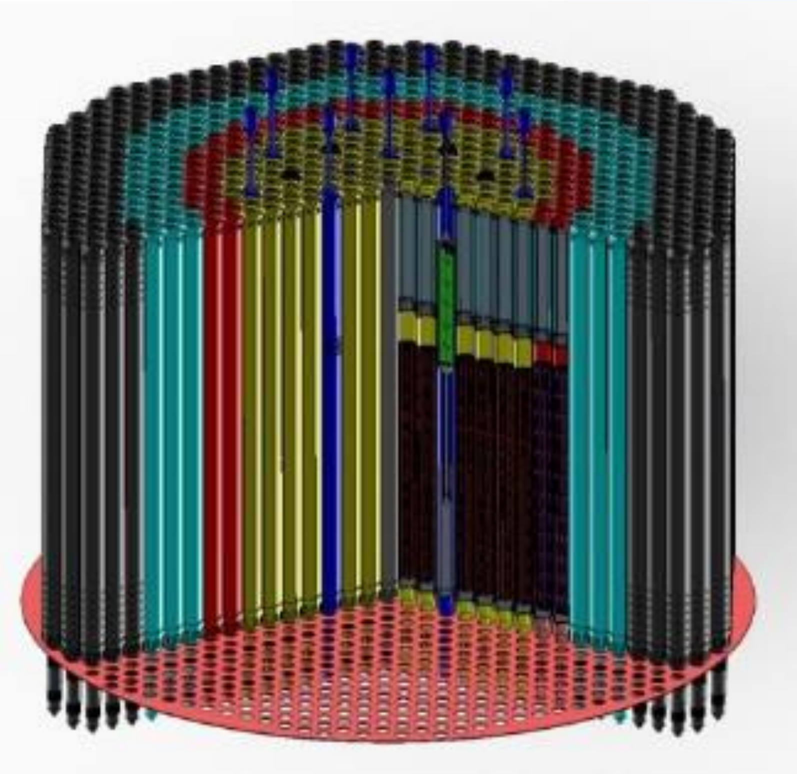
ASTRID: Advanced Sodium Technological Reactor for Industrial Demonstration

C. Latge at Joint IAEA-ICTP Workshop
August 2016, Trieste, Italy

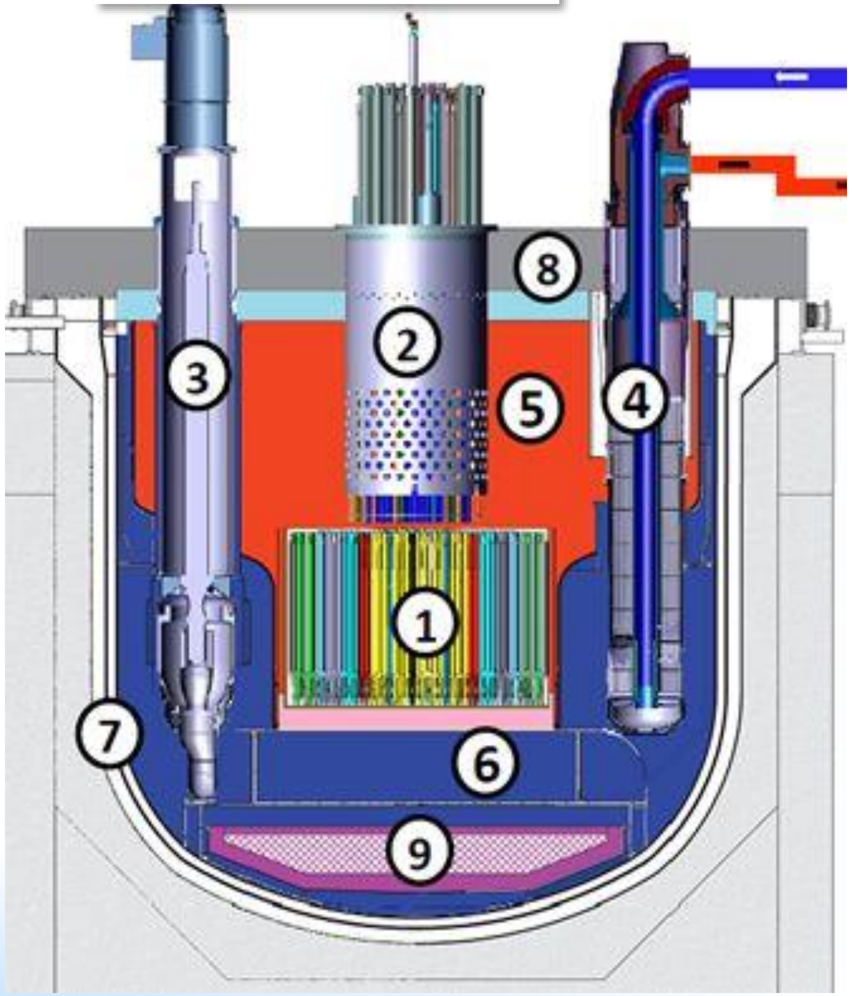


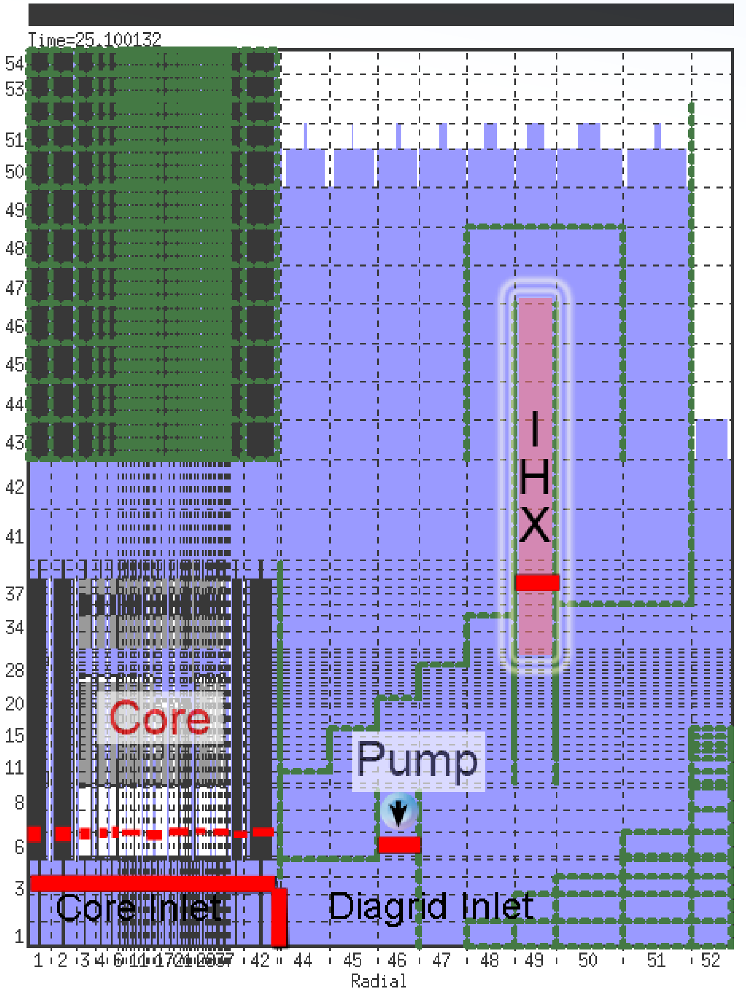
Innovative Core Design for Enhanced Safety Features



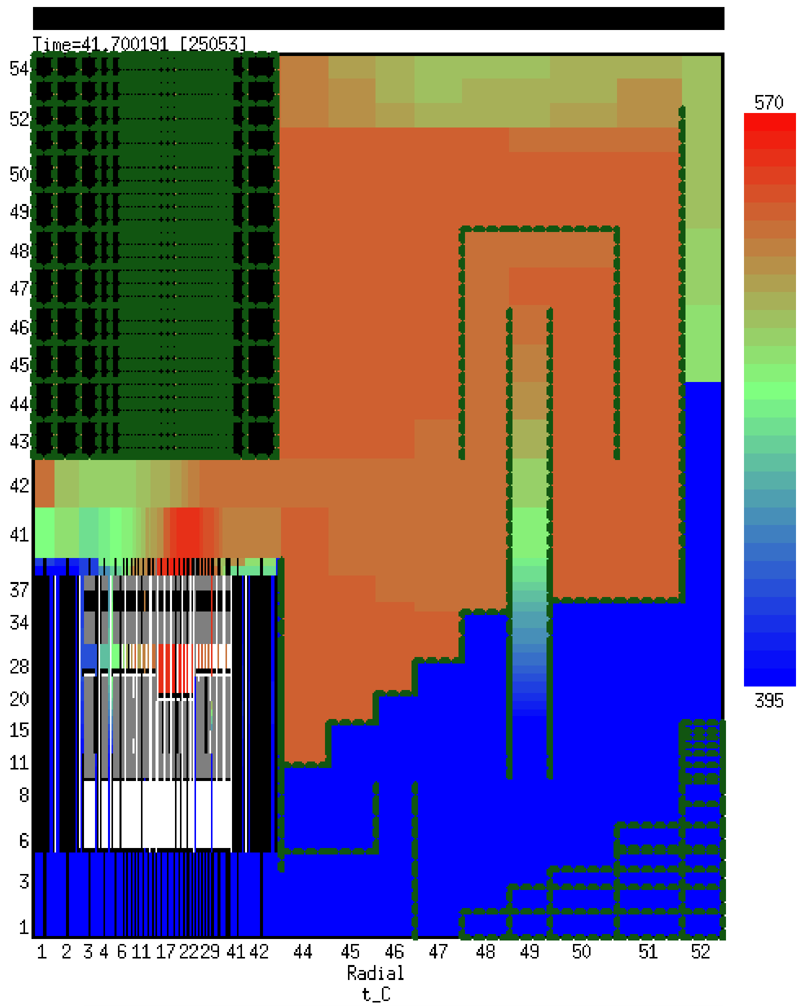
ASTRID: ULOF Simulations with SAS4A and SIMMER-III codes

One of considered ASTRID designs





SIMMER-III Model: Reactor Vessel in 2D (r-z)

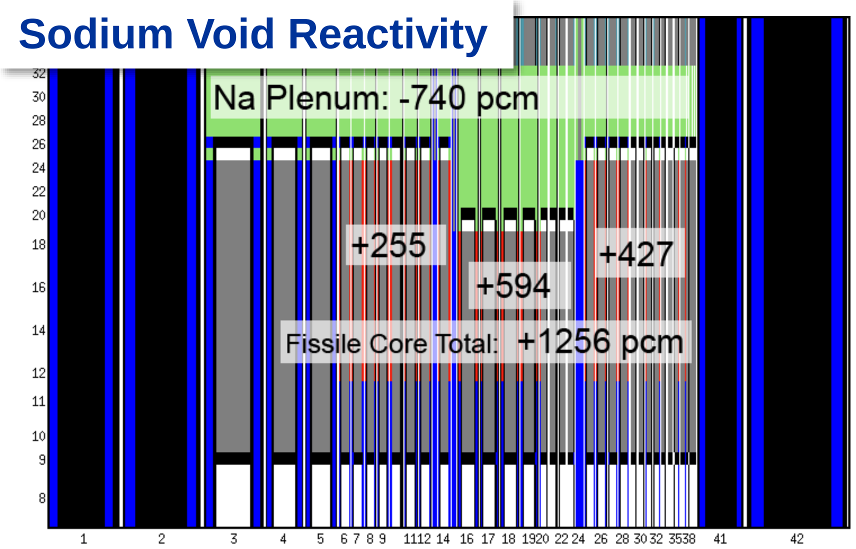


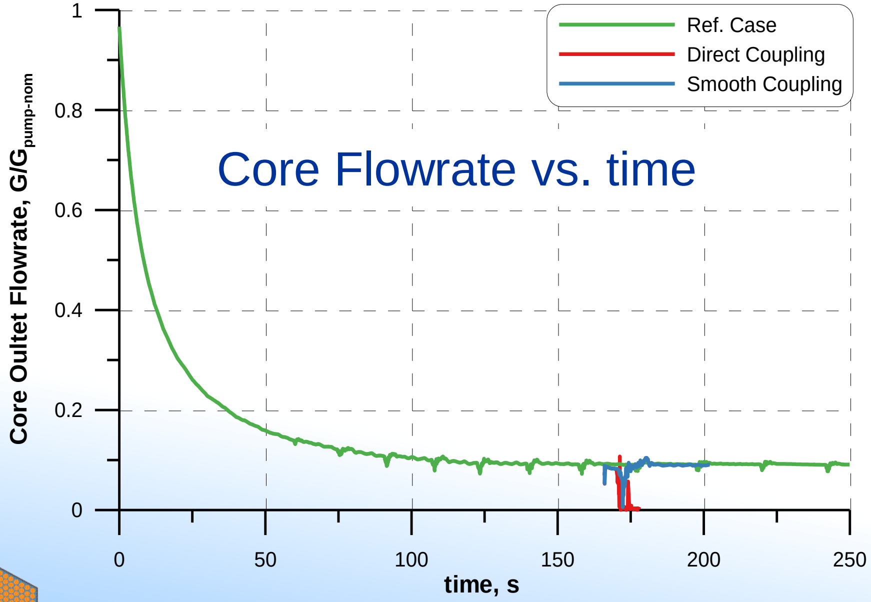
Na Temperature Distribution

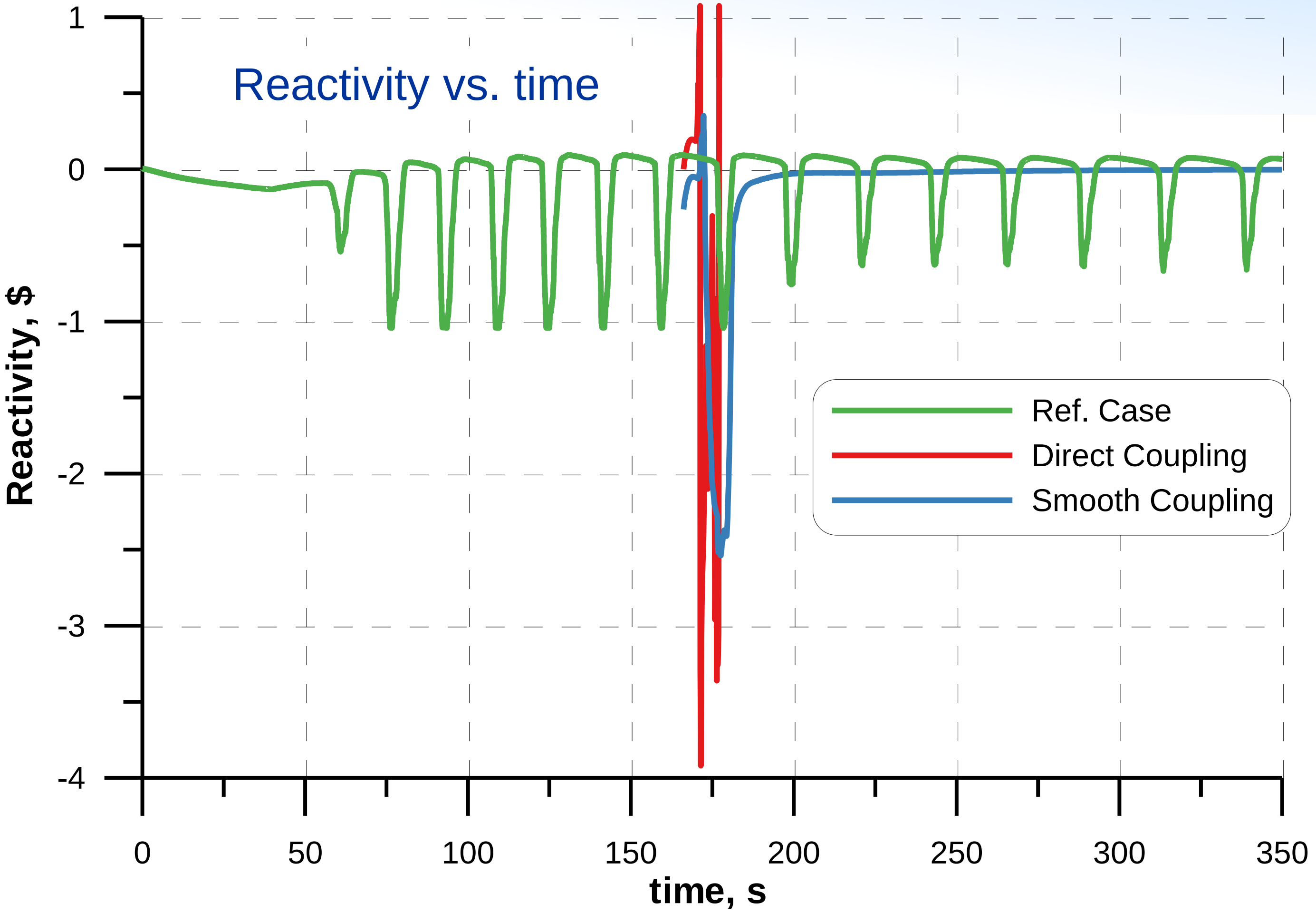
V. Kriventsev, 2015, KIT, Germany

ULOF Simulations with SAS4A and SIMMER-III Codes

Sodium Void Reactivity



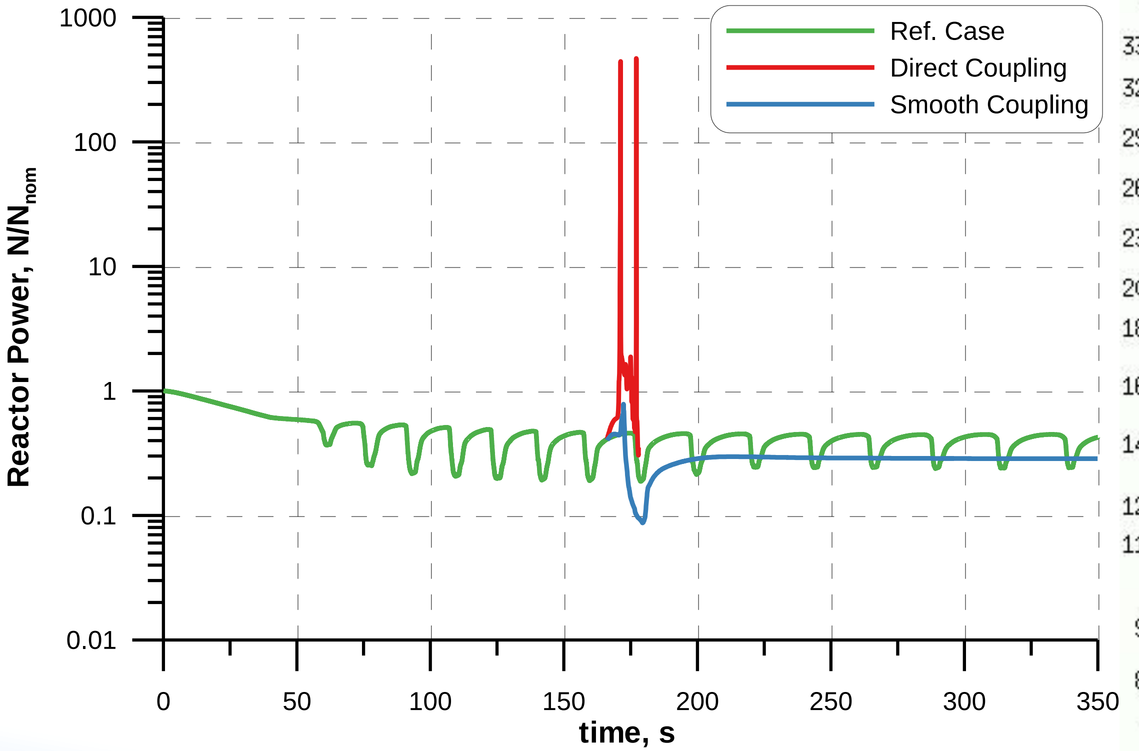




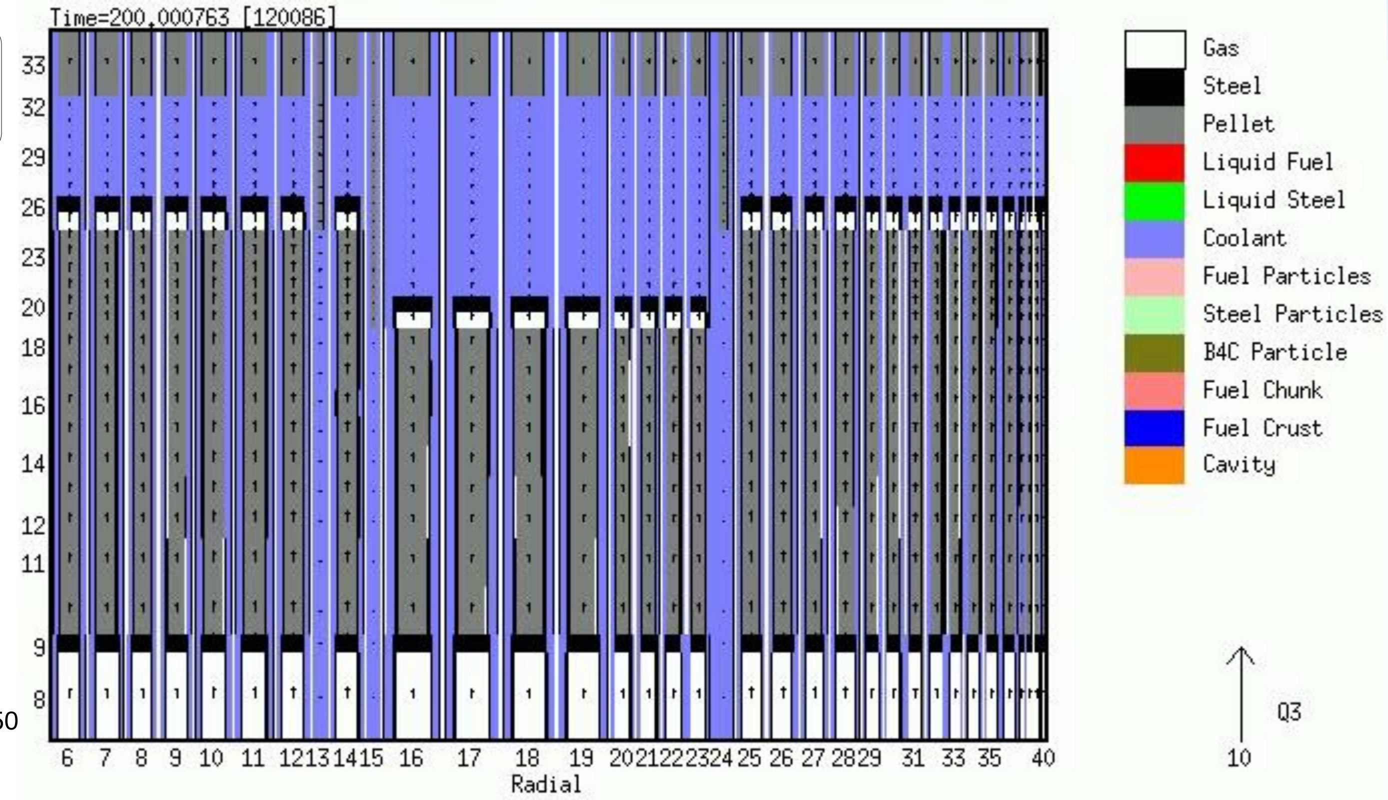
V. Kriventsev, 2015, KIT, Germany

ULOF Simulations with SIMMER-III Code

Reactor Power



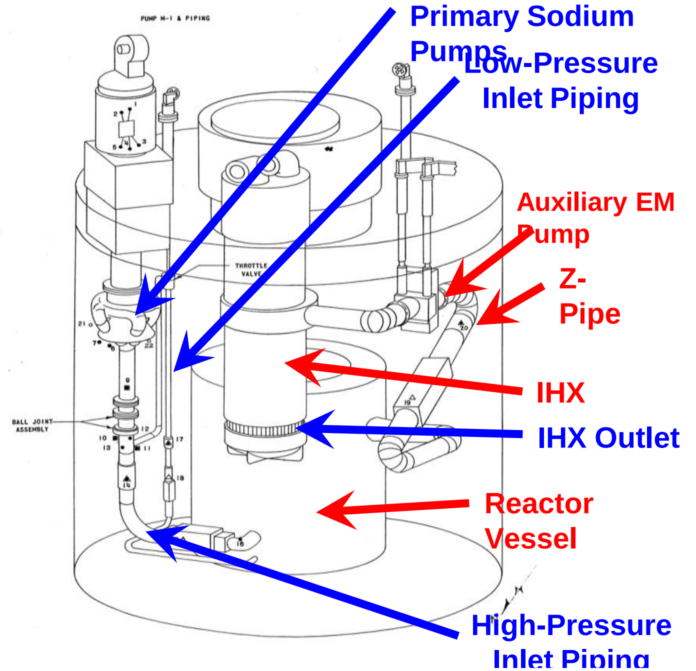
Material Distribution in Core vs. Time

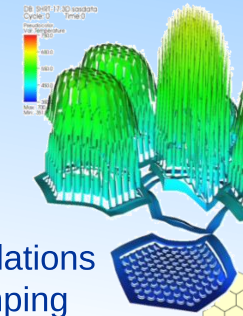


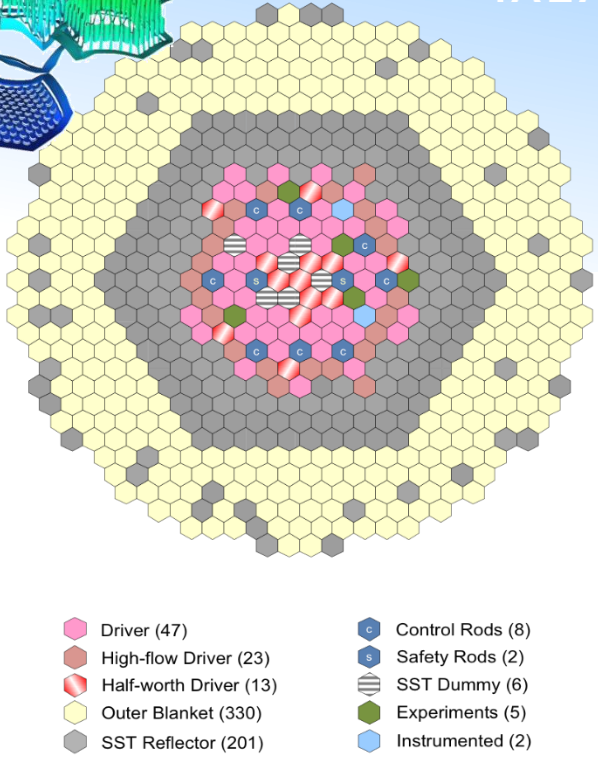
V. Kriventsev, 2015, KIT, Germany

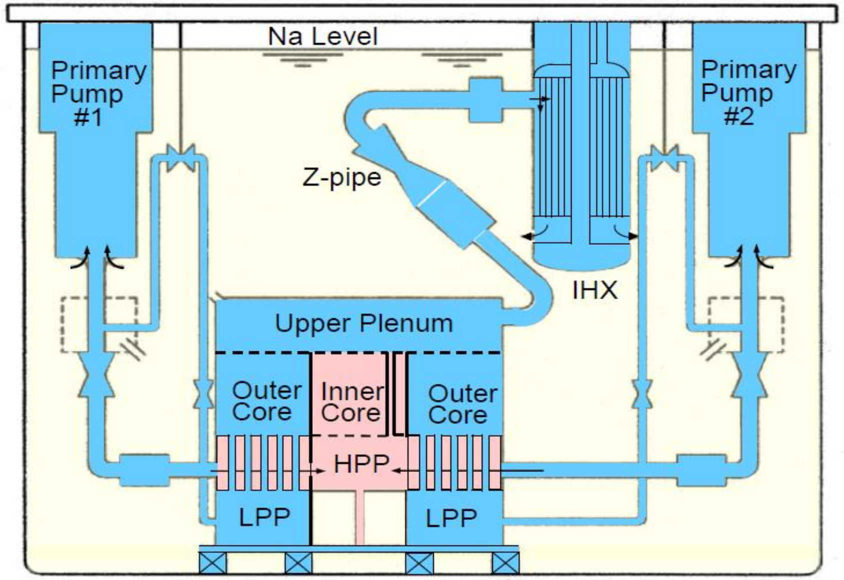
CRP on Benchmark Analysis of ***EBR-II*** Shutdown Heat Removal Test (*2012-2016*)

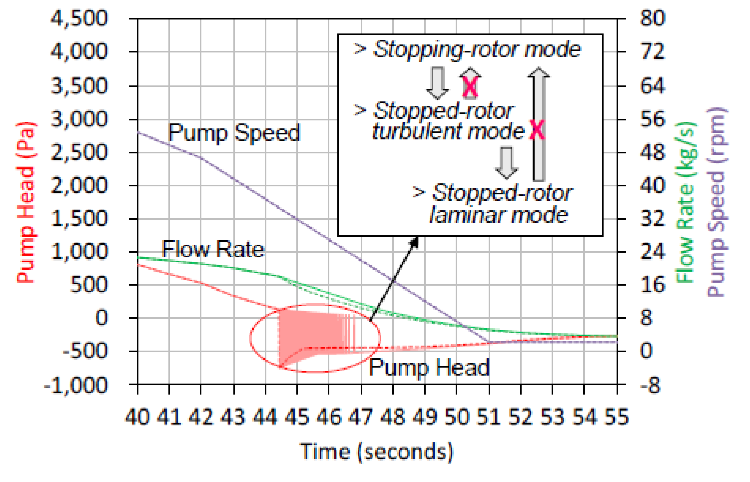
- Coupled Neutronics and Thermalhydraulic Transient Simulations
- SHRT-17 (Protected): Loss of normal and emergency pumping
- SHRT-45 (Unprotected): Loss of normal flow, scram disabled, station blackout

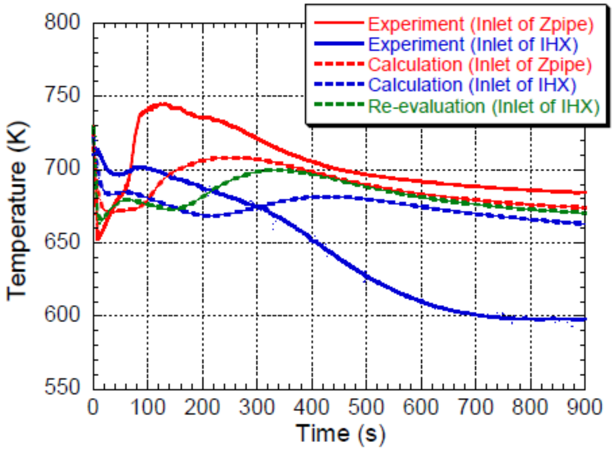


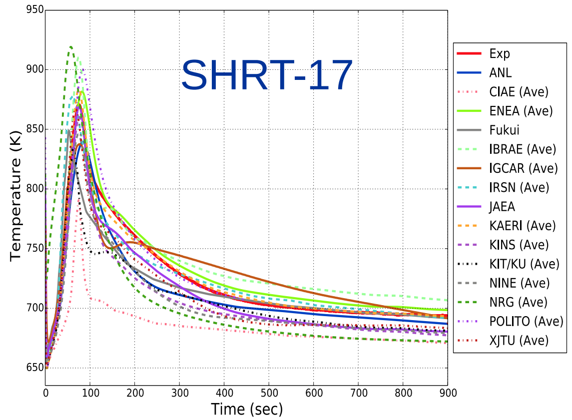












Benchmark Analysis of ***EBR-II*** Shutdown Heat Removal Test

- Coupled Neutronics and Thermalhydraulic Transient Simulations
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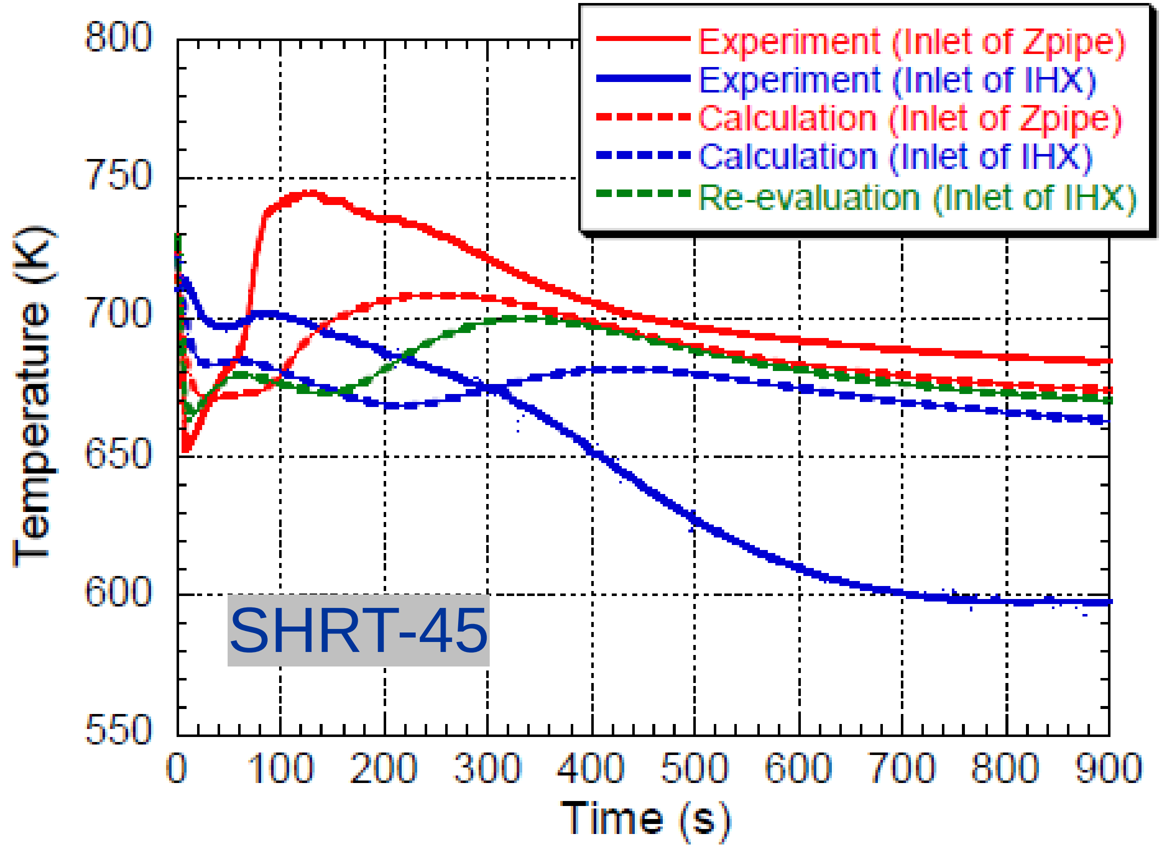
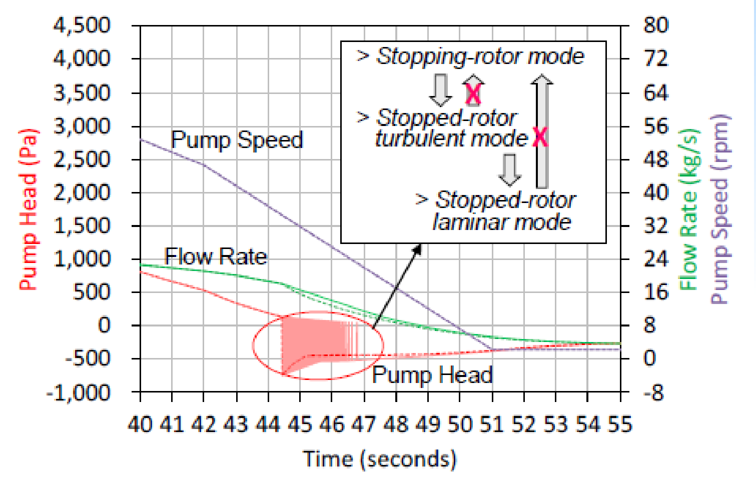
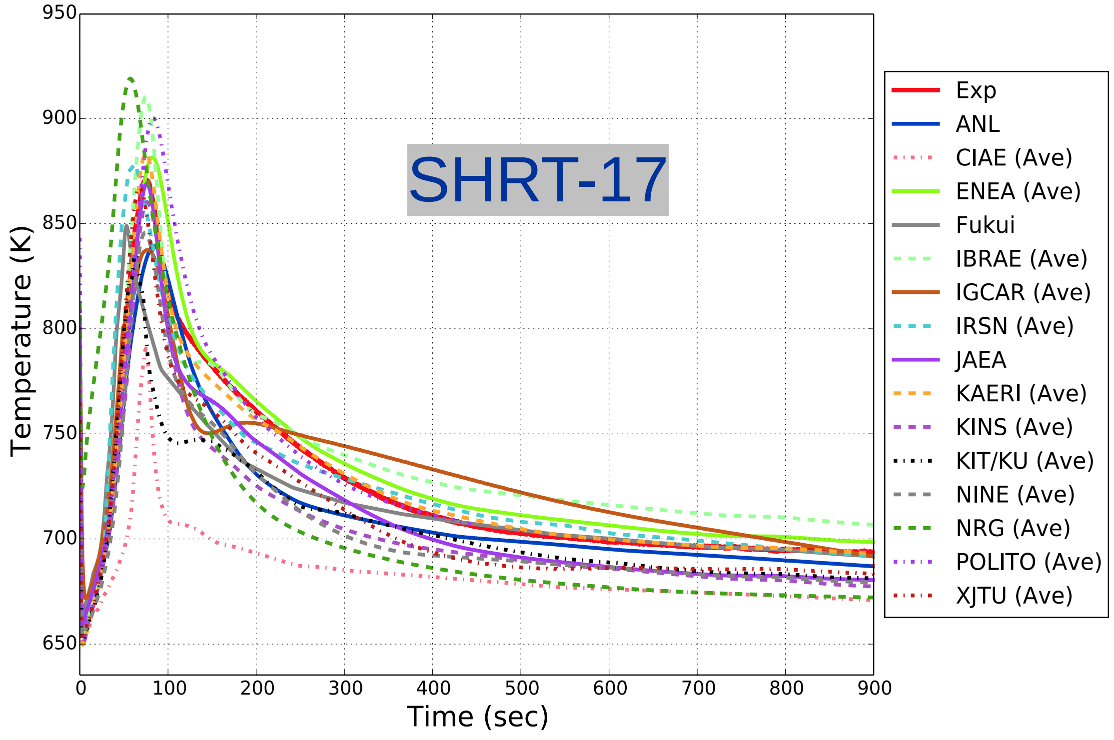
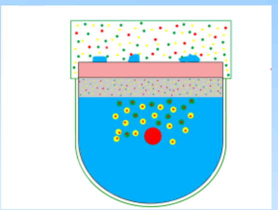


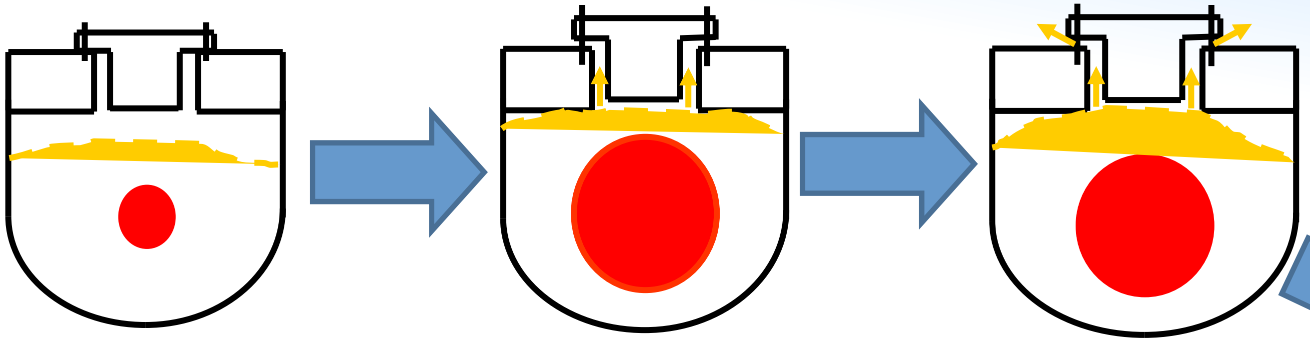




CRP on Radioactive Release from Prototype SFR under Severe Accident Conditions (2016- 2020)



CDA development and propagation in pool type SFR

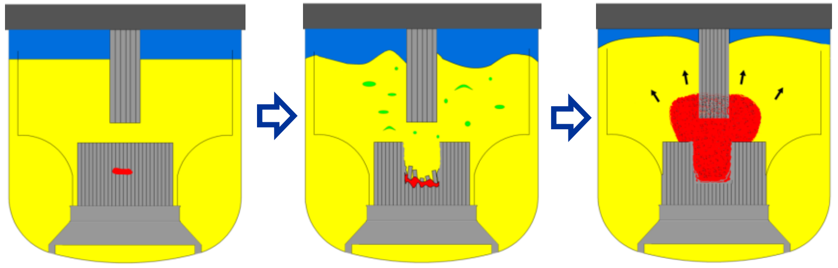


Initiation *(neutronics), and* ***Transition*** *(fuel relocation) Phases*

Core Melt/Bubble is formed

I. *Expansion Phase*

Core bubble expands in sub-cooled sodium



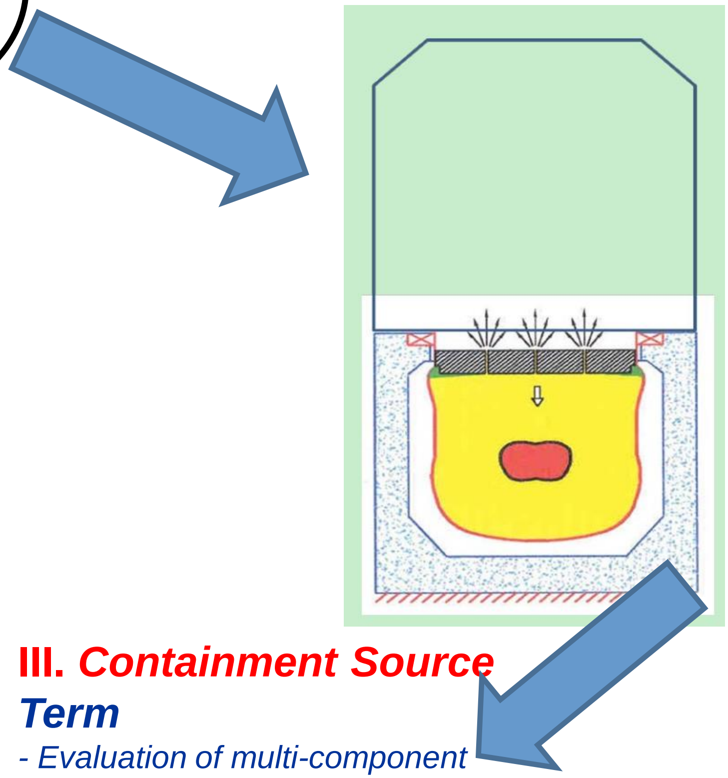
Incipient melting and early relocation

Extended relocation and core compaction

Rapid fuel vapor bubble expansion

II. *Quasi-static Phase*

Release of sodium to the Reactor Containment Building (RCB)



Reference design for the safety analysis:

500 MWe pool type PFBR

Very complicated multi-physics phenomenon
Can be a Standard Benchmark for Verification of Safety Analysis Codes and Models

III. *Containment Source Term*

- Evaluation of multi-component aerosol evolution is required

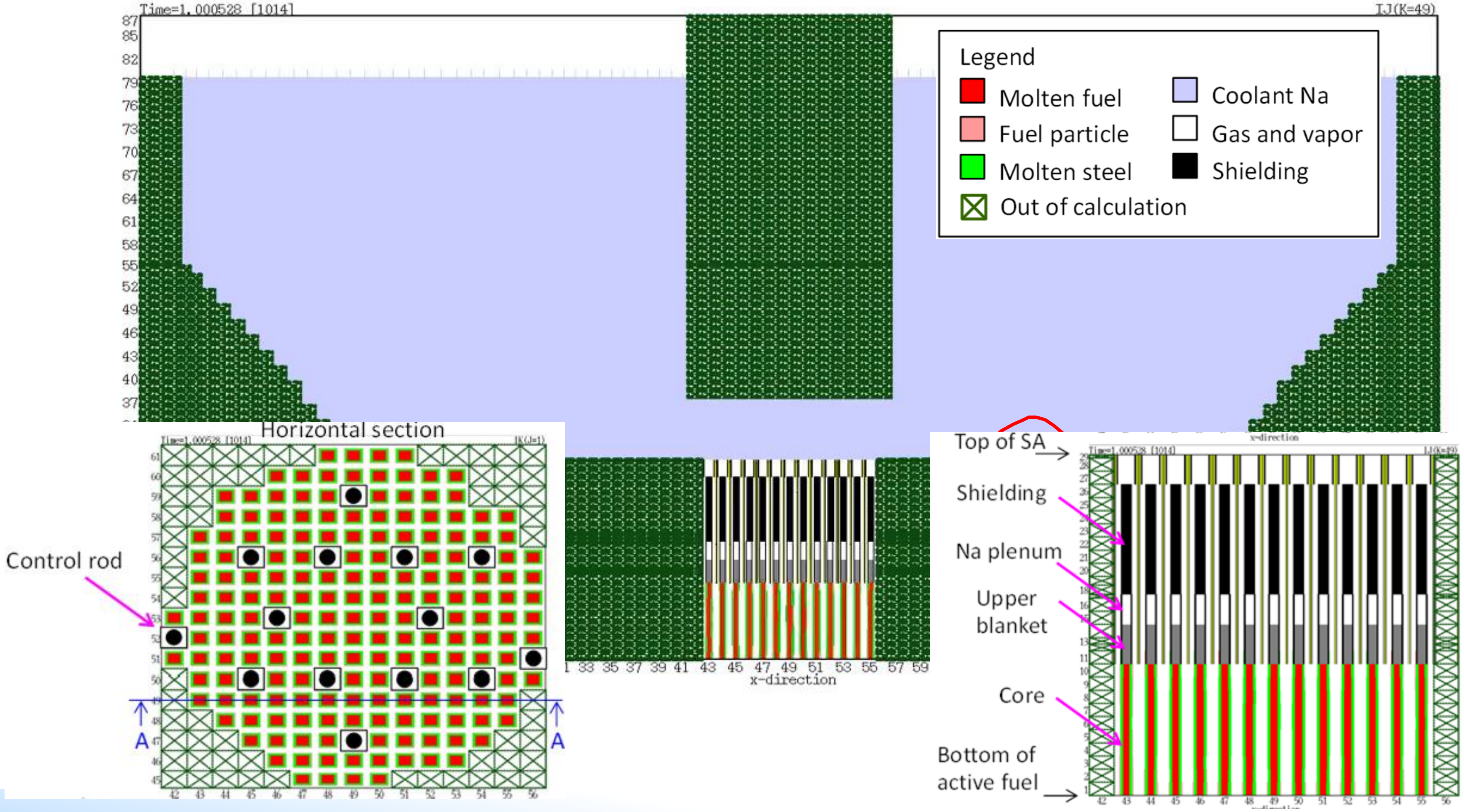
- Two typical sodium fire accidents:

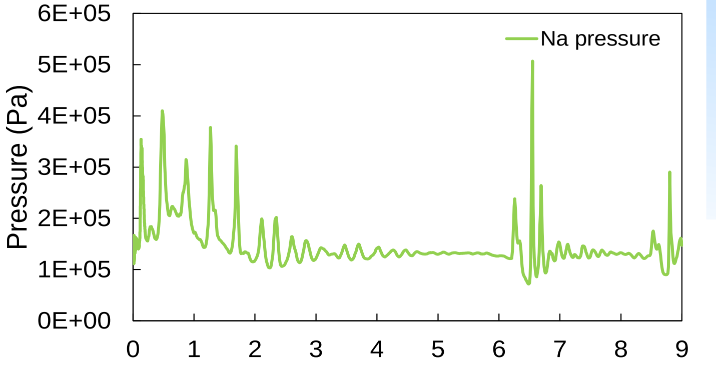
- *sodium pool fire accident*
- *sodium spray fire accident*

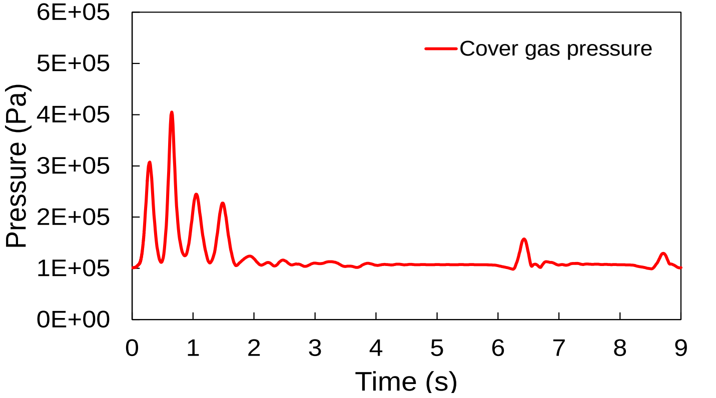
CRP on Radioactive Release from Prototype SFR under Severe Accident Conditions (2016- 2020): Expansion Phase

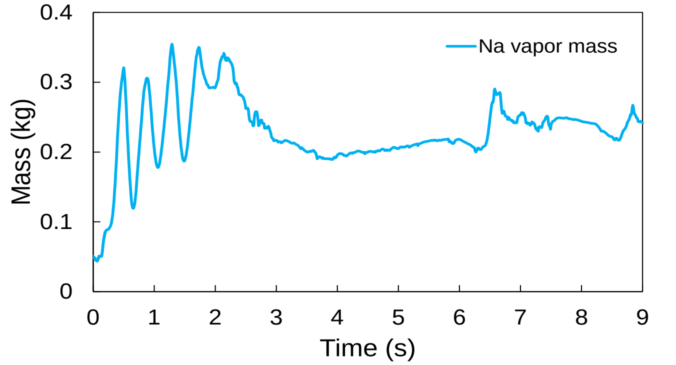
IAEA

Click to play SIMMER-IV Video
(provided by JAEA)







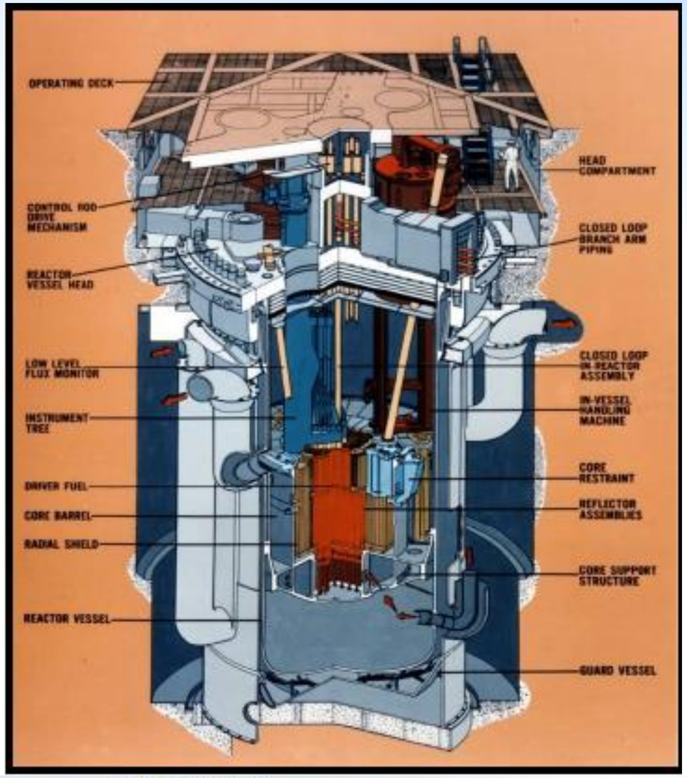


WP-1. *Sodium Bubble Expansion Phase*

IAEA CRP: **Benchmark Analysis of FFTF Loss of Flow Without Scram Test**

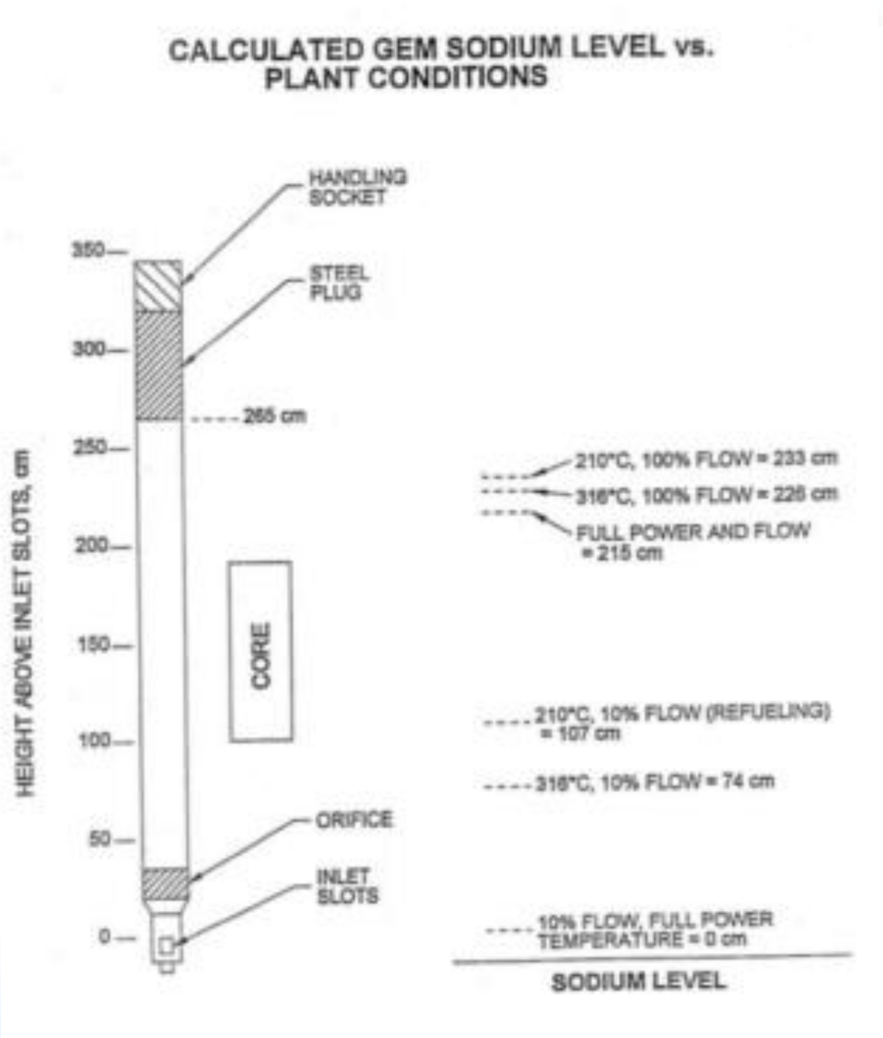
IAEA

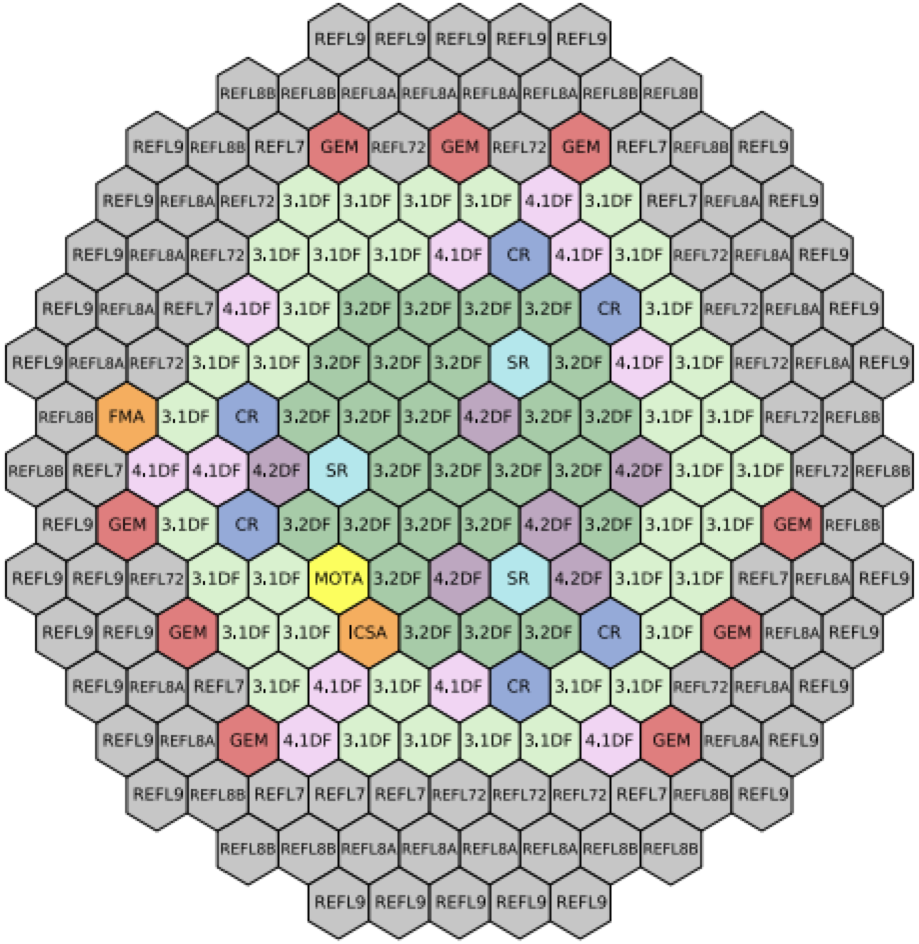
- FFTF Reactor:
  - 400 MW(th) sodium cooled fast test reactor
  - Mixed UO2-PuO2 (MOX) fuel
  - Loop type plant, axial and radial reflectors
  - Prototypic size
    - ~1m^3 core volume
    - ~91 cm high, ~120 cm diameter
  - Series of Passive Safety Tests
    - Demonstrated passive safety of SFRs
    - Demonstrated efficacy of negative reactivity insertion safety devises (GEMs)



FastRTeam

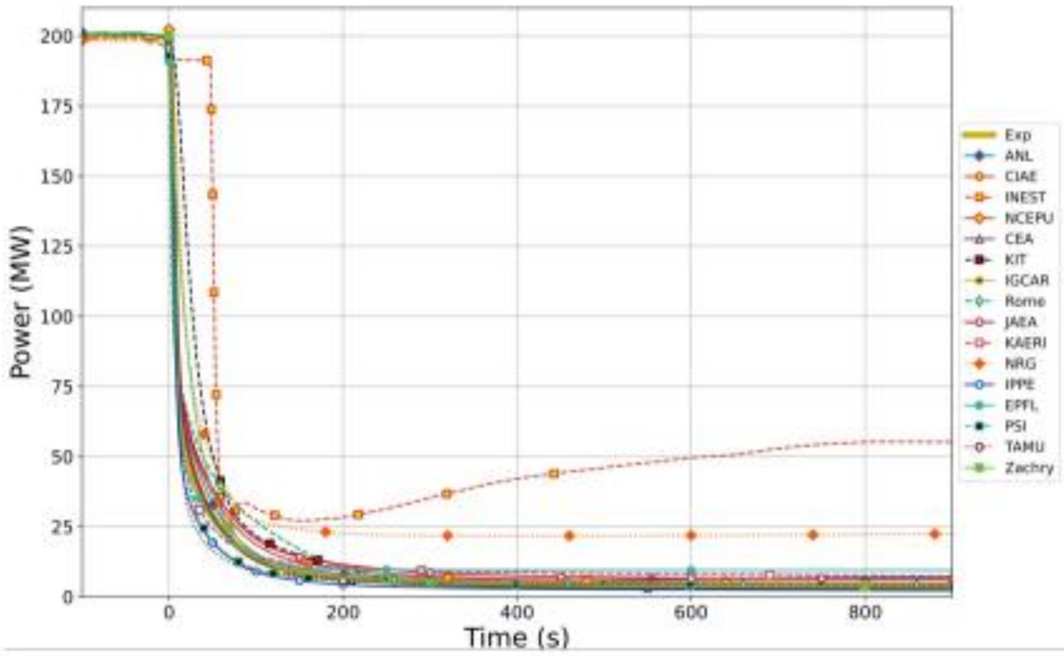
FFTF: Gas Expansion Module (GEM)



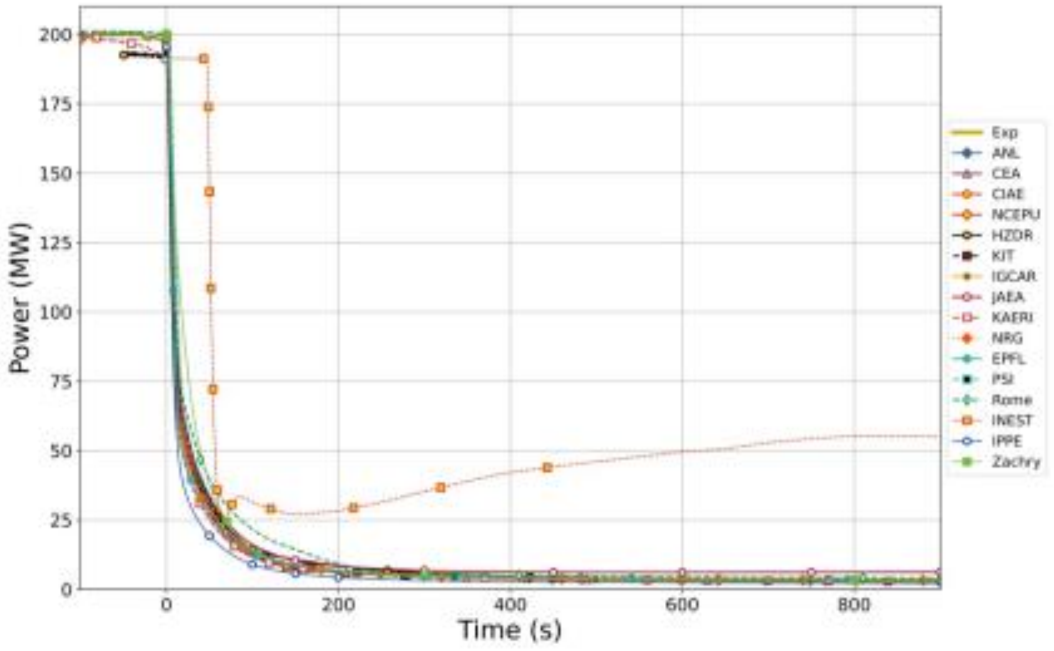


FFTF: Comparing Results: Reactor Power

TOTAL POWER

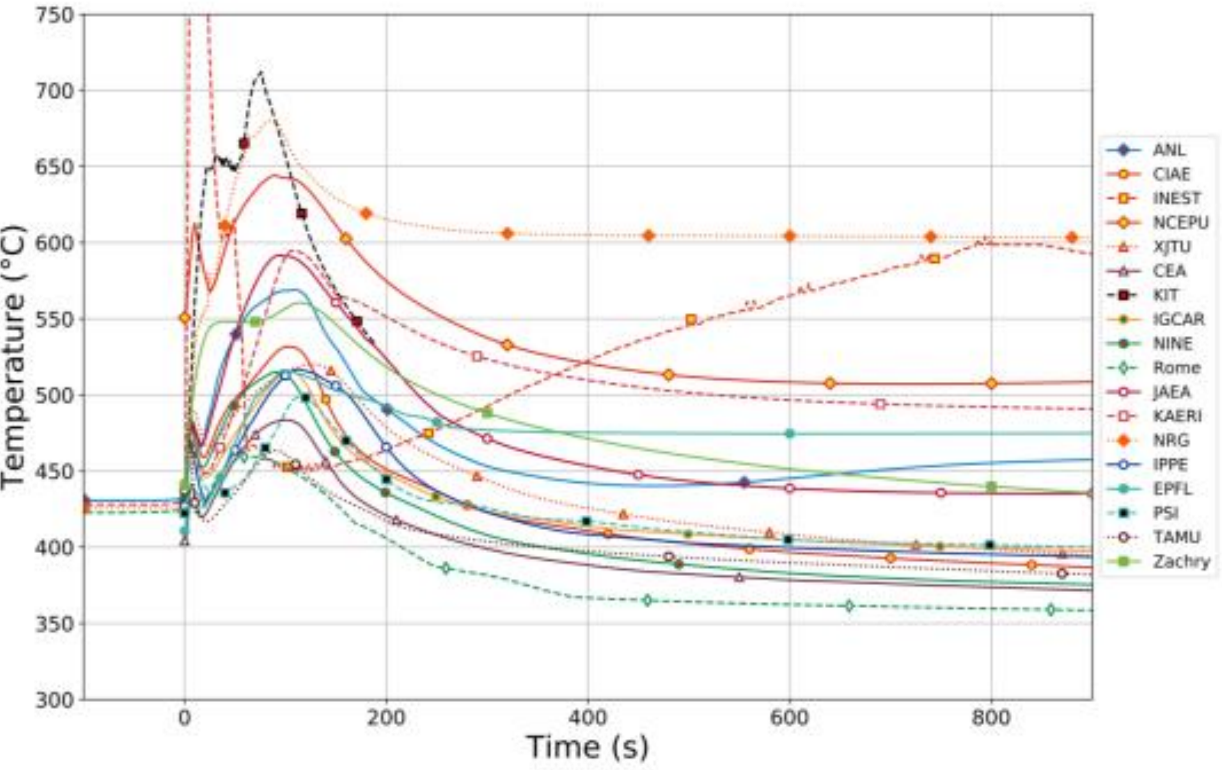


Blind Results

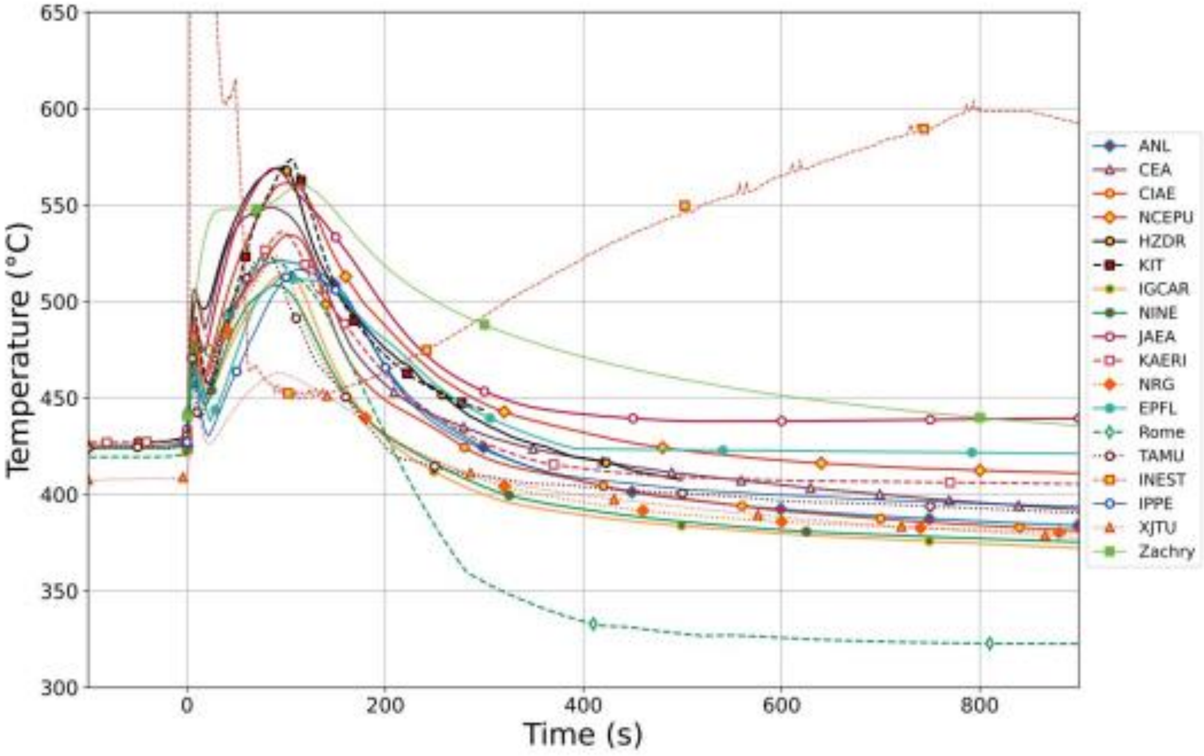


Final Results

FFTF: Comparing Results: Coolant Temperatures

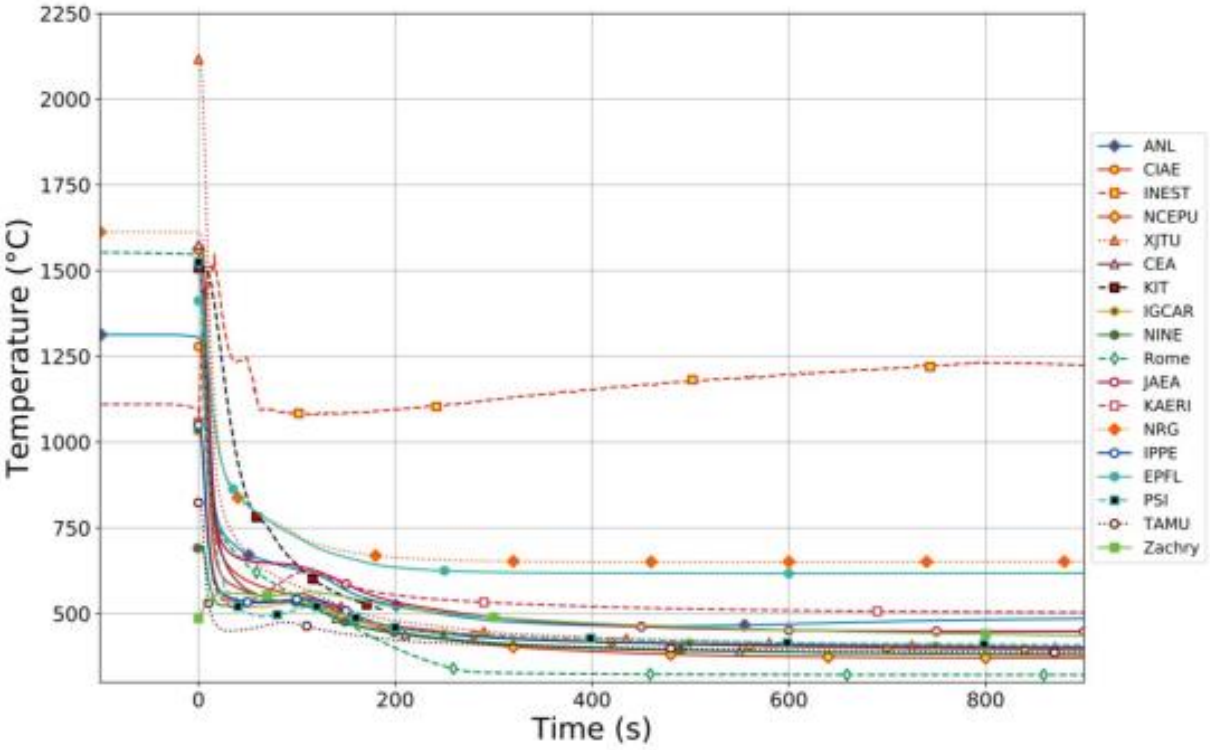


Blind Results

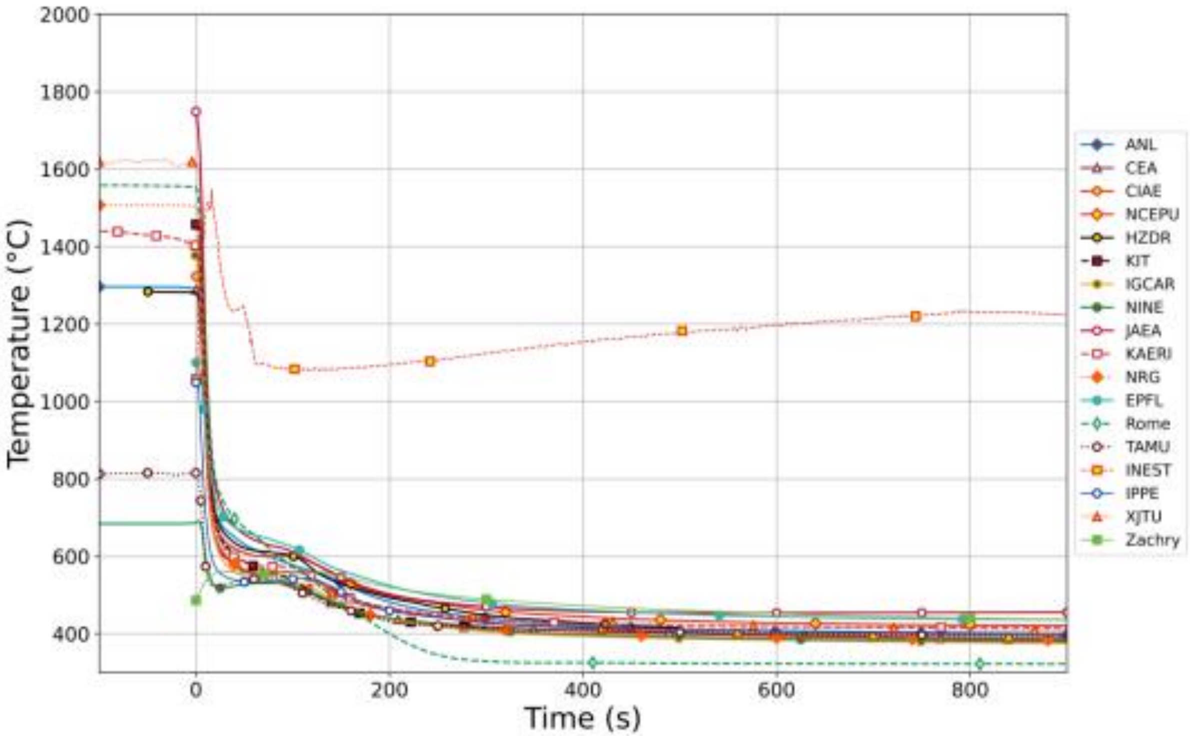


Final Results

FFTF: Comparing Results: Fuel Temperatures



Blind Results

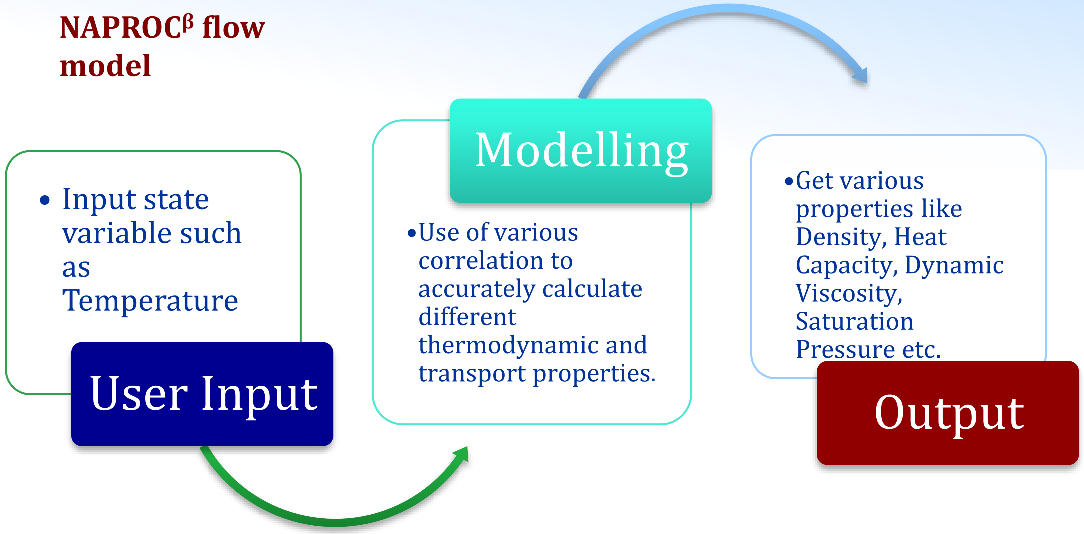


Final Results

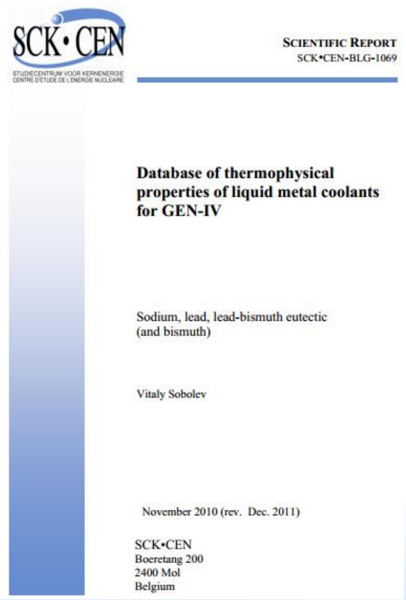
NAPROC $^{\beta}$: Sodium Properties Calculator

- Easy to use software to get the thermo-physical of liquid sodium.
- Input the required state variables and get all desired properties.
- Beta version under development.
- Modelling based on the use of various correlations.
- If possible, benchmarking against available database.

NAPROC$^{\beta}$ flow model



Used for software modeling


SCK•CEN
STUDIECENTRUM VOOR KERNENERGIE
CENTRE D'ETUDE DE L'ENERGIE NUCLEAIRE

SCIENTIFIC REPORT
SCK•CEN-BLG-1069

Database of thermophysical properties of liquid metal coolants for GEN-IV

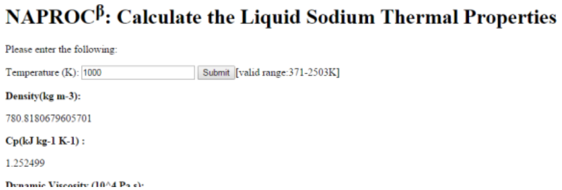
Sodium, lead, lead-bismuth eutectic (and bismuth)

Vitaly Sobolev

November 2010 (rev. Dec. 2011)

SCK•CEN
Boeretang 200
2400 Mol
Belgium

Current Development


NAPROC$^{\beta}$: Calculate the Liquid Sodium Thermal Properties

Please enter the following:

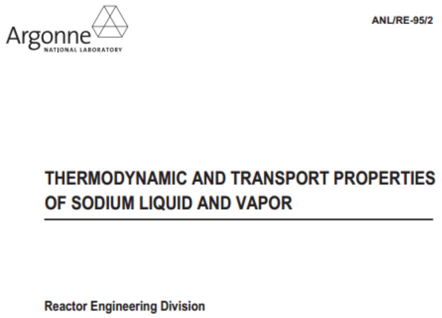
Temperature (K): 1000 Submit [valid range:371-2503K]

Density(kg m-3):

780.8180679605701

Cp(kJ kg-1 K-1) :

1.252499


Argonne NATIONAL LABORATORY

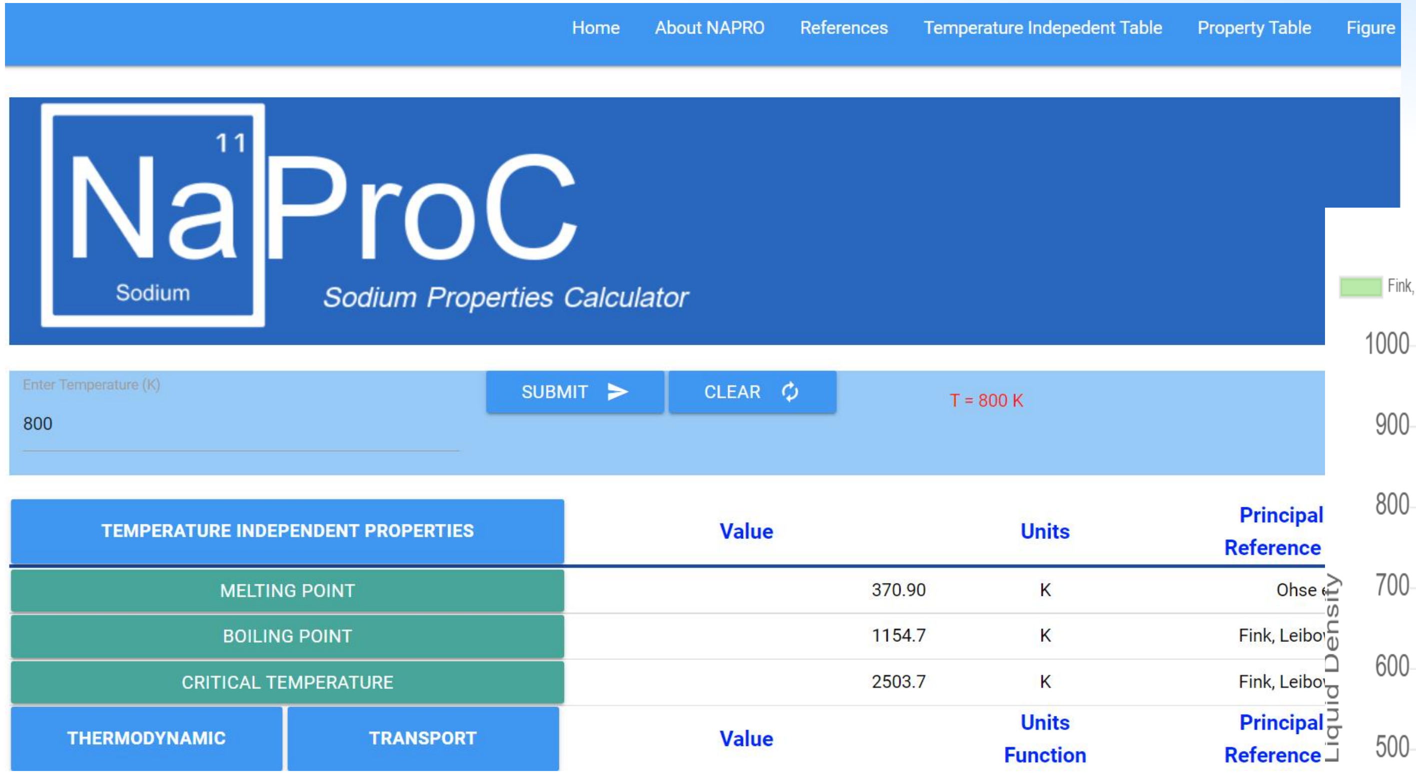
ANL/RE-95/2

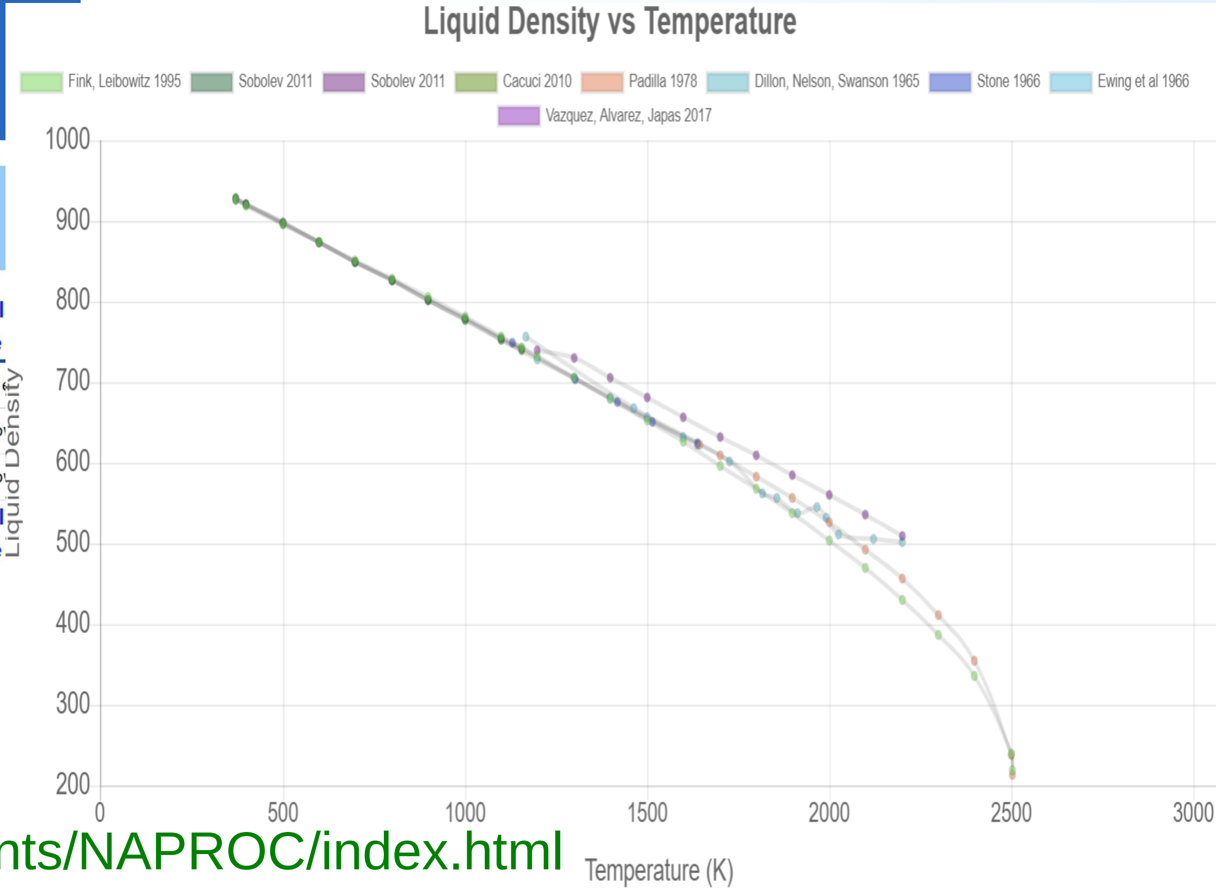
THERMODYNAMIC AND TRANSPORT PROPERTIES OF SODIUM LIQUID AND VAPOR

Reactor Engineering Division

Used for Benchmarking

NAPRO: Sodium Properties Calculator





https://nucleus.iaea.org/sites/fr/Shared%20Documents/NAPROC/index.html

Thank You!