IAEA
International Atomic Energy Agency

Joint ICTP-IAEA Workshop on the Physics and Technology of Innovative Nuclear Energy Systems
20 – 24 August 2018, Trieste, Italy

Thermal Hydraulics of
Innovative Nuclear Energy Systems

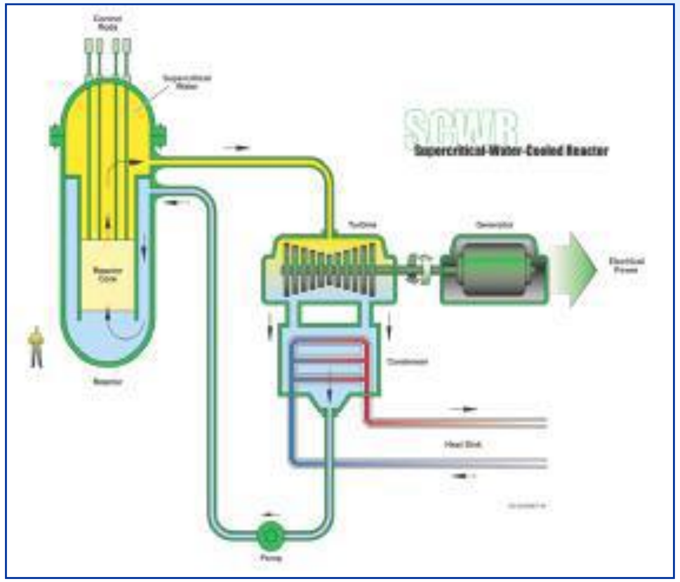
Vladimir Kriventsev
Fast Reactor Technology Development Team
Nuclear Power Technology Development Section
Department of Nuclear Energy

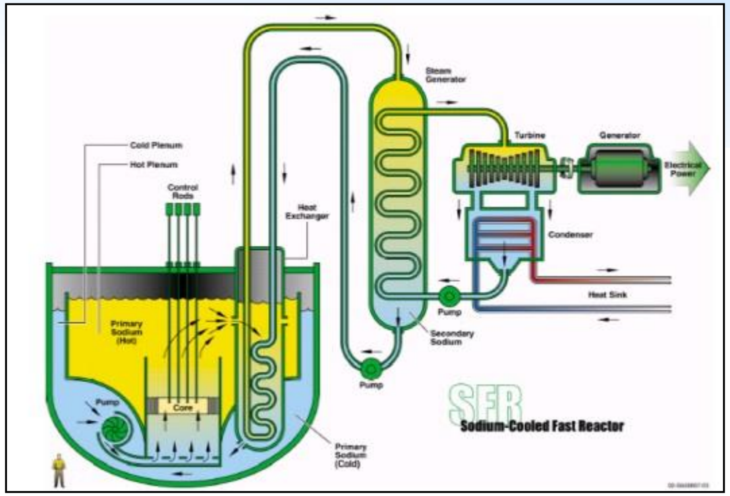
Outline

- 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

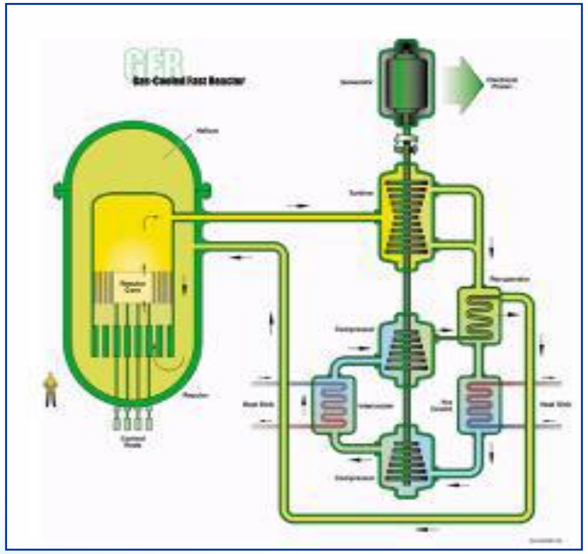
General Reactor Classification

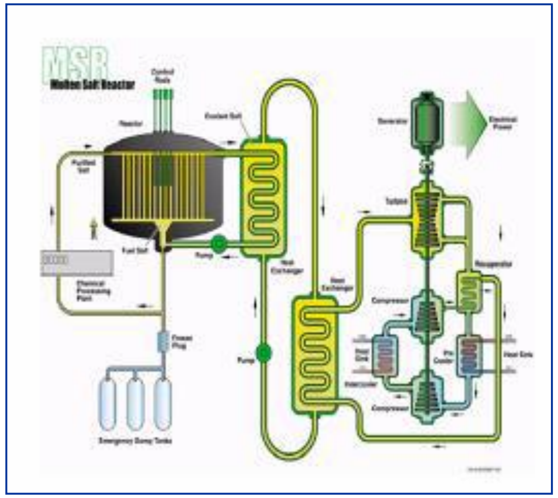
- Moderator
  - Water / Heavy Water
  - Graphite
  - None (fast neutron systems)
- Coolant
  - Water/Heavy Water
  - Liquid Metal
    - Sodium / Lead / Lead-Bismuth Eutectic (LBE)
  - Gas
    - Air / CO_2 / Helium
  - Molten Salt
- Fuel
  - UO2
  - MOX (UO_2 + PuO_2)
  - Metallic
  - Molten Salt
- Purpose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



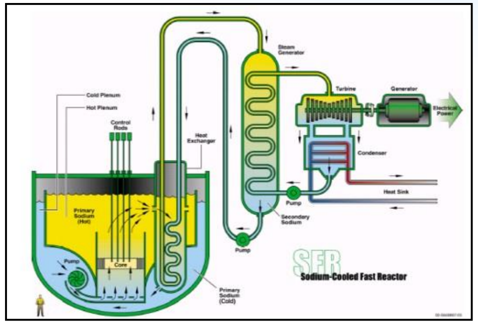


GEN-IV Reactors (GIF)

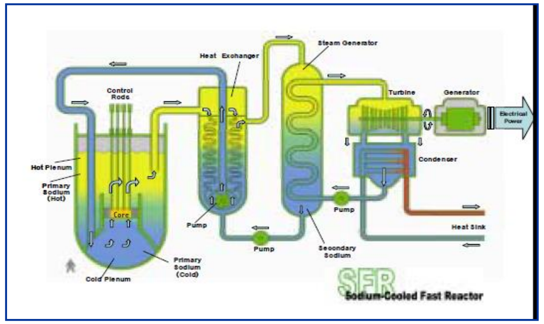




Six Generation IV Reactor systems

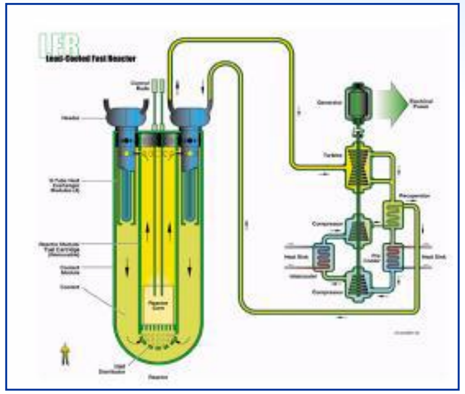


Pool type

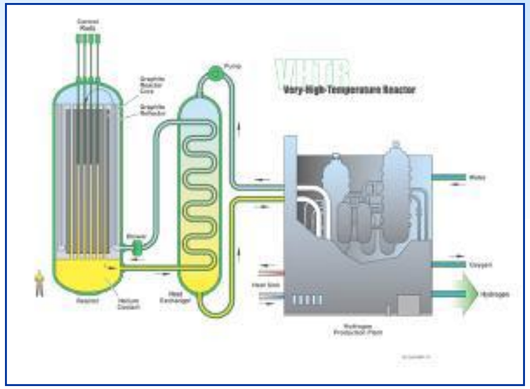


Loop type

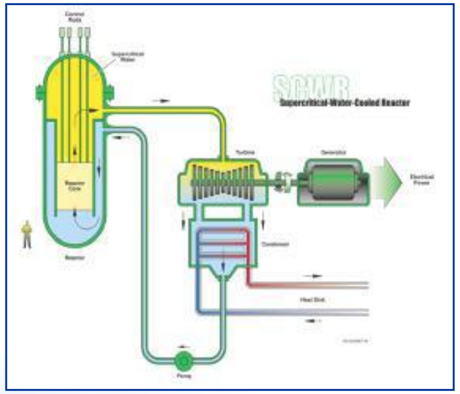
Sodium-cooled Fast Reactor (SFR)



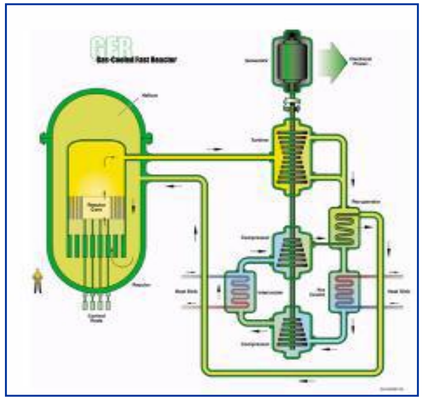
Lead-cooled Fast Reactor (LFR)



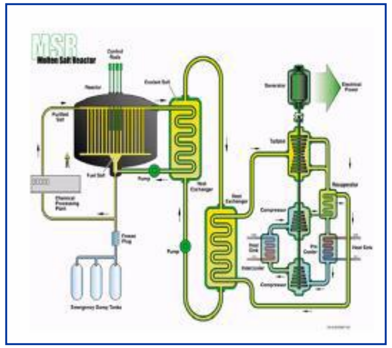
Very-High-Temperature Reactor (VHTR)



Supercritical-Water-cooled Reactor (SCWR)



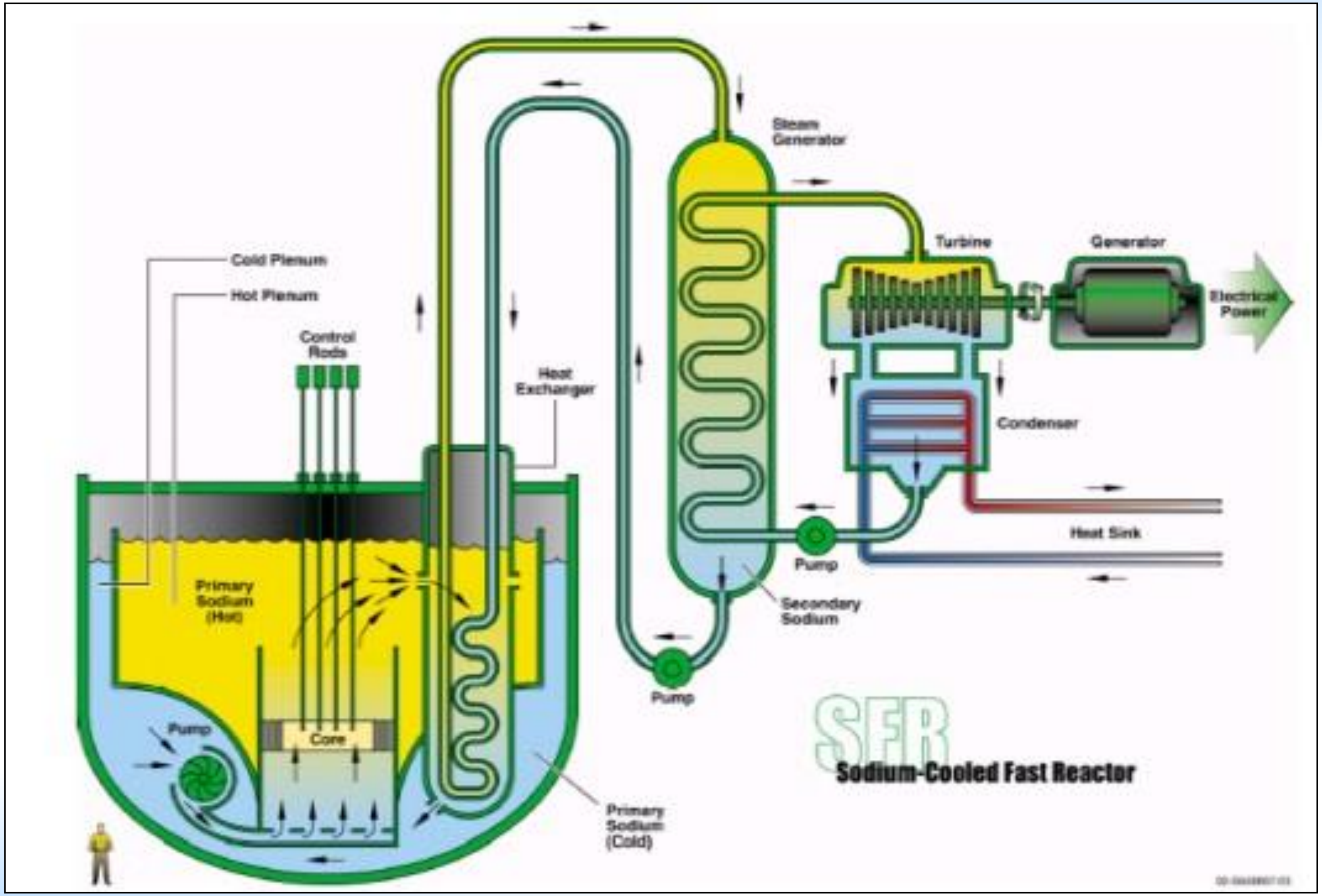
Gas-cooled Fast Reactor (GFR)



Molten Salt Reactor (MSR)

GIF website: www.gen-4.org

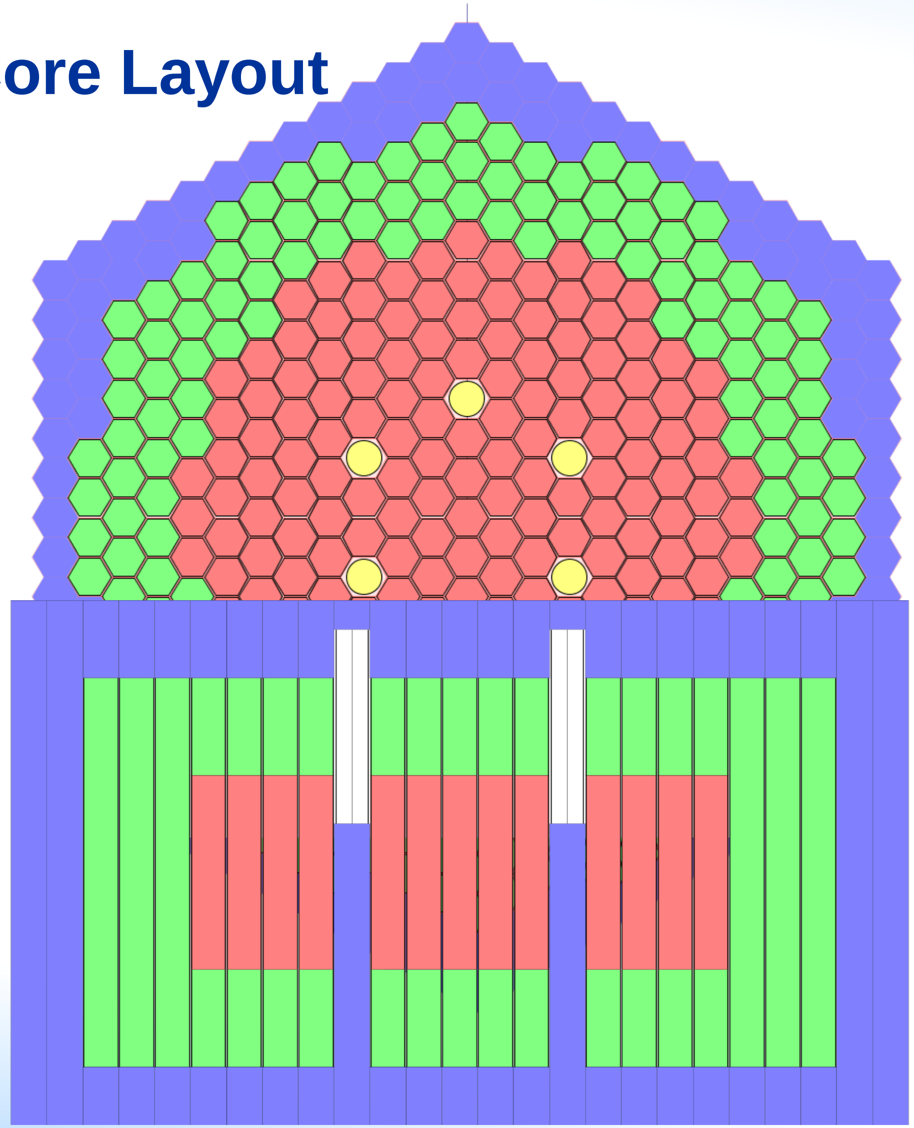
Sodium Cooled Fast Reactor (SFR)



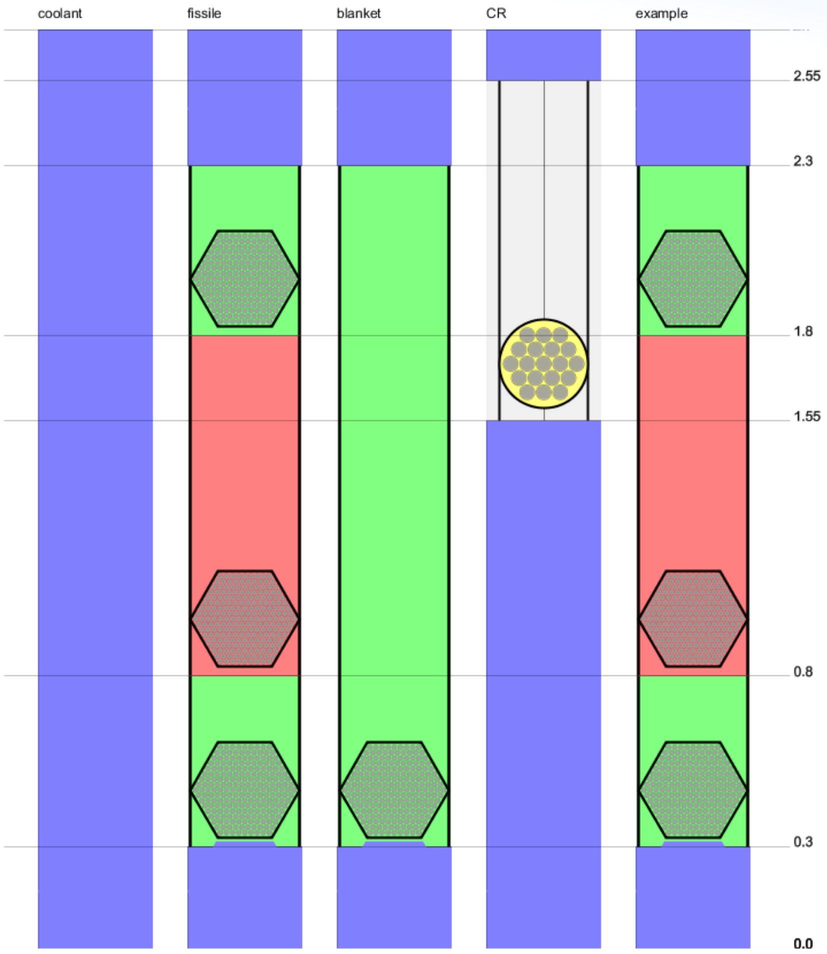
GIF website: www.gen-4.org,

Reactor Core

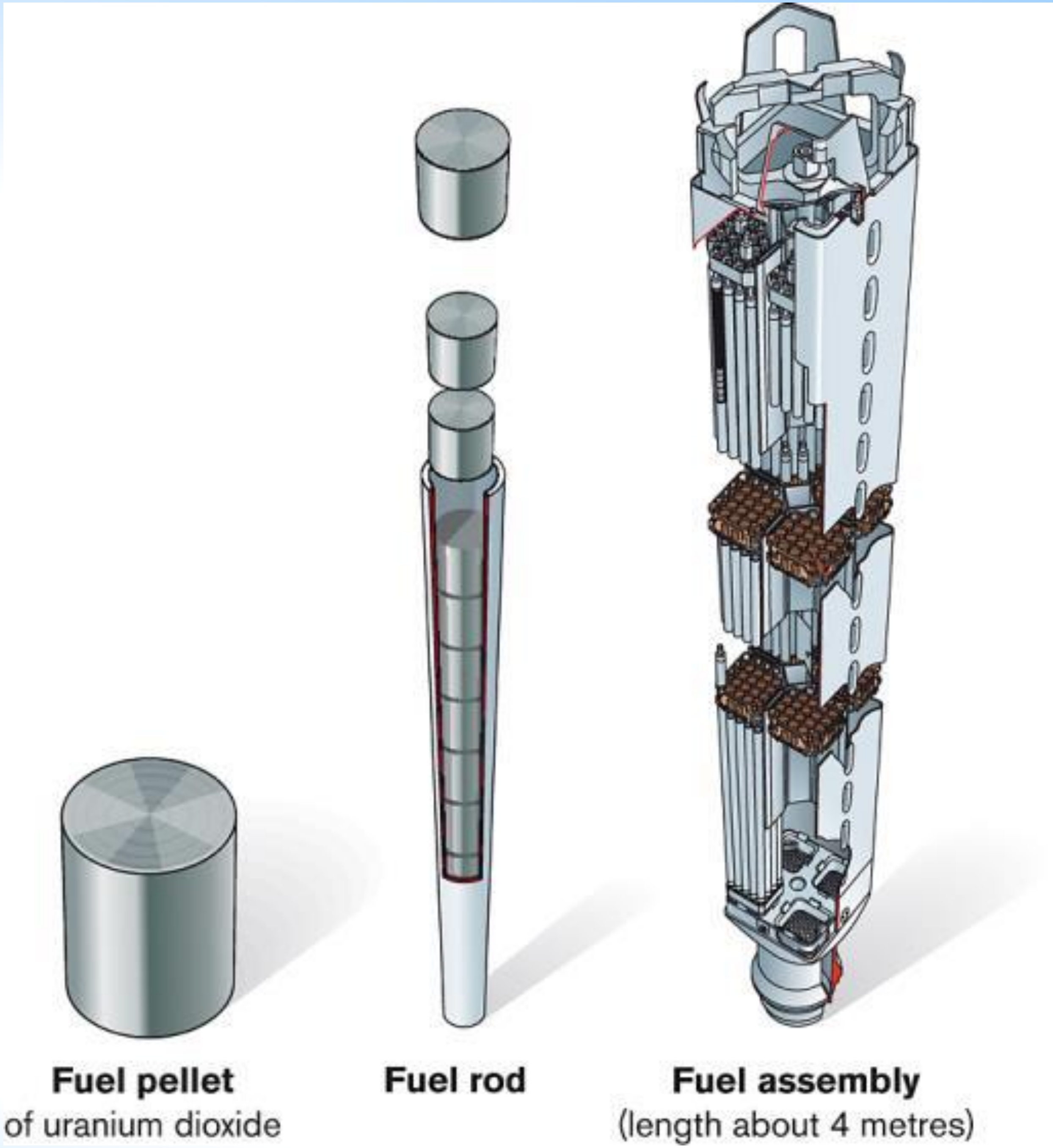
Core Layout



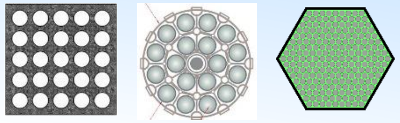
Sub-Assemblies (S/A)

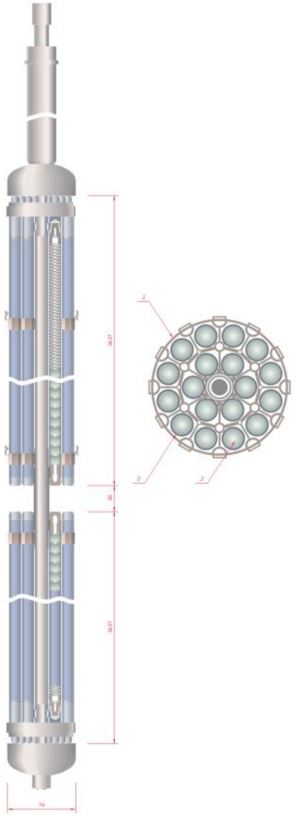


LWR Fuel Assembly (Rod Bundle)



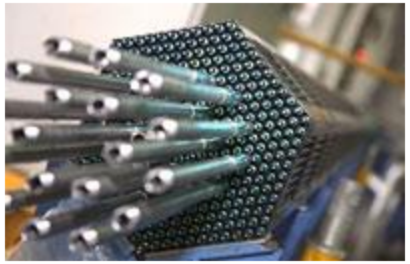
Rod Bundles





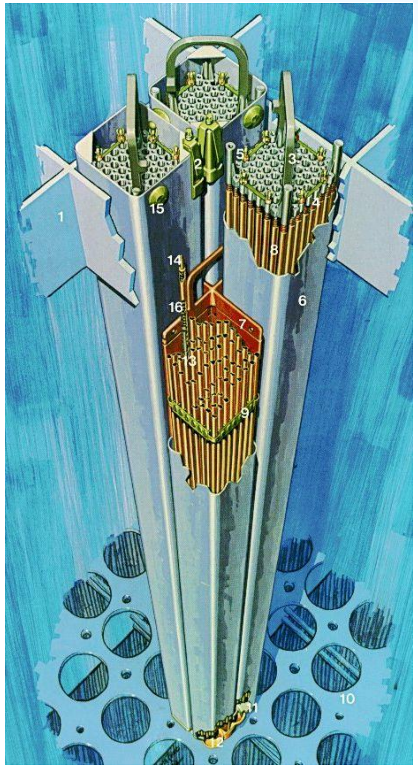
PWR

CANDU



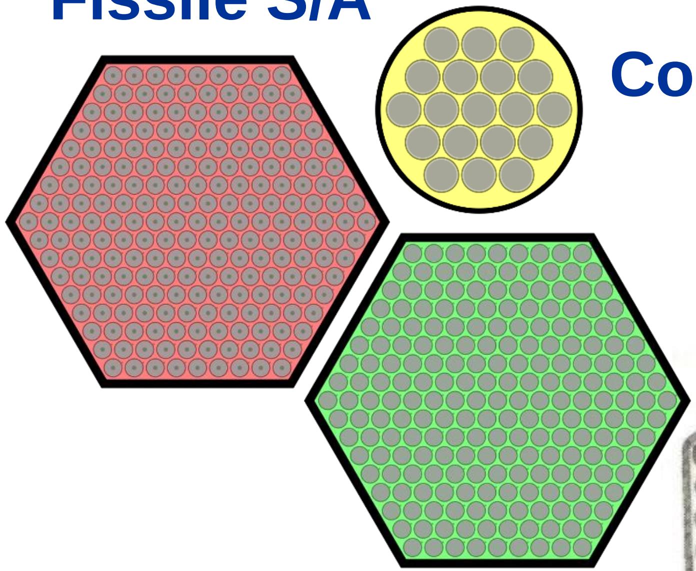
VVER

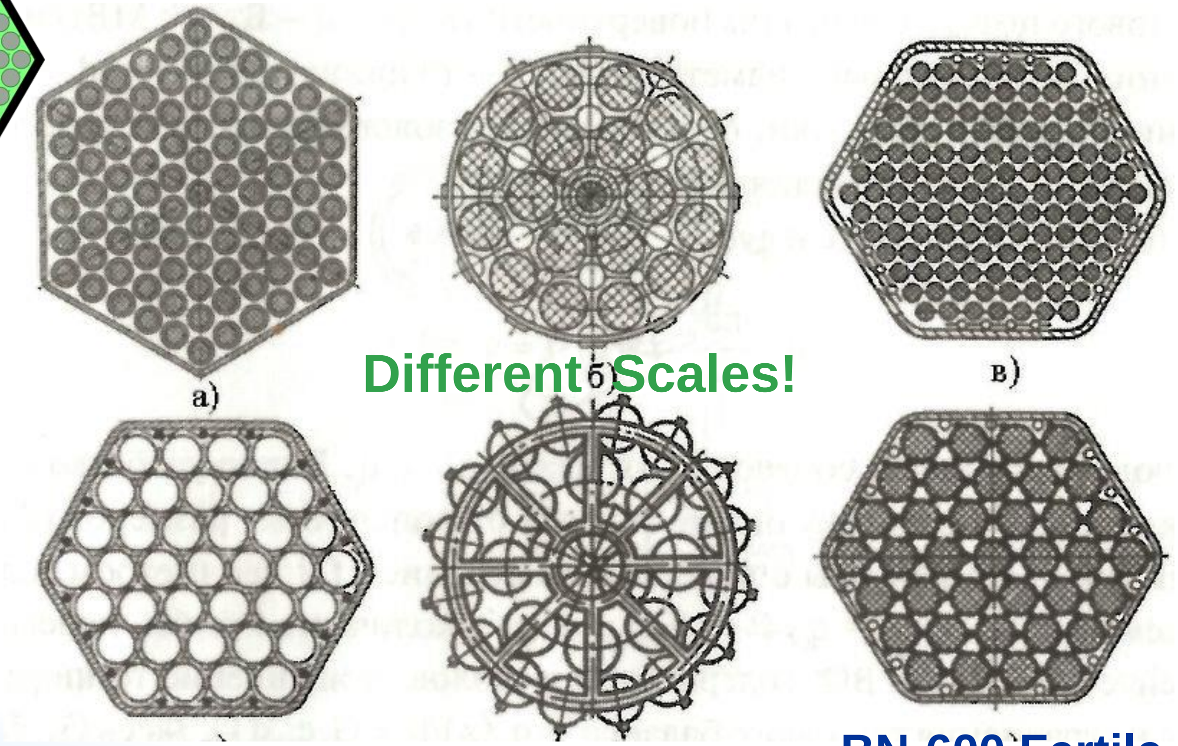
RBMK



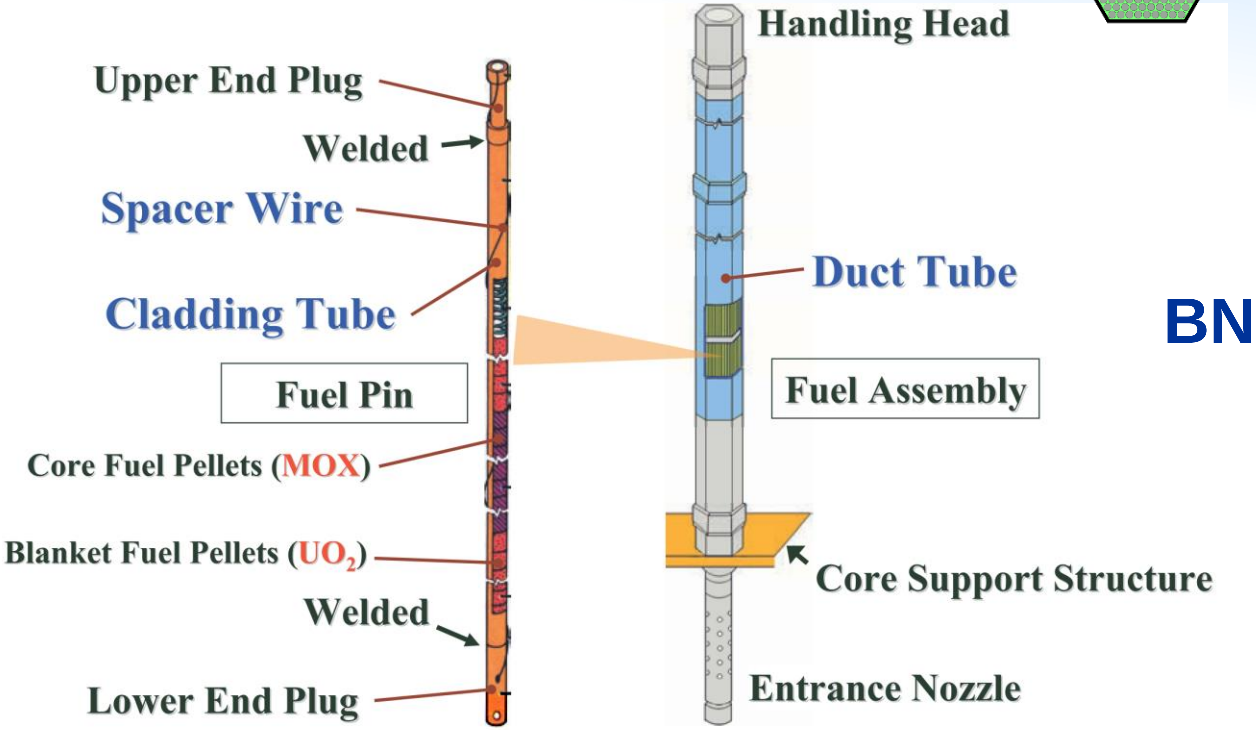
BWR

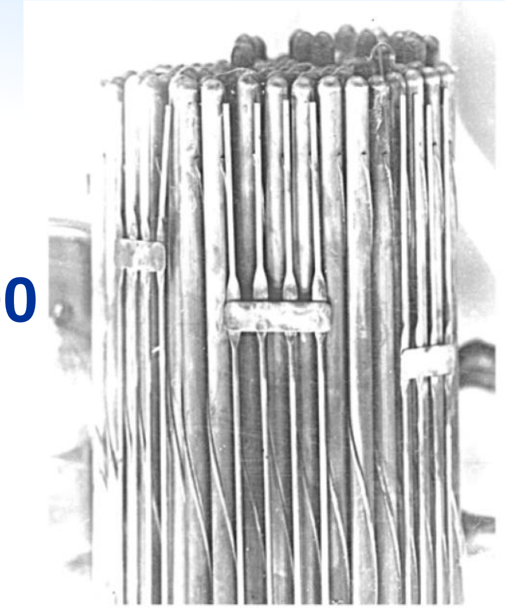
Sub-Assembly Types





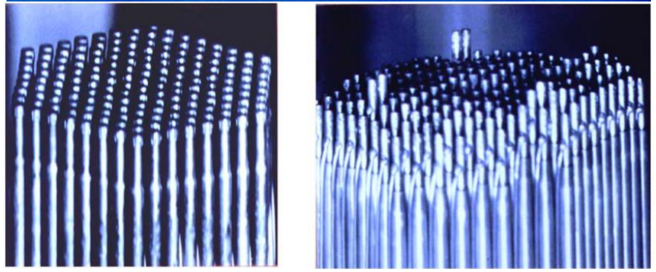
SFR Fuel Assemblies

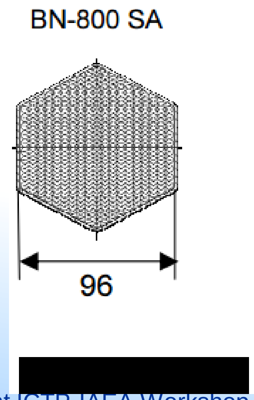


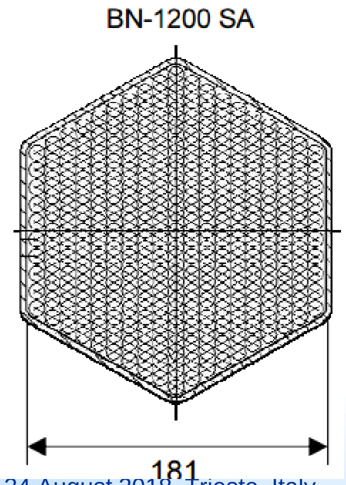


BN-600

FFTF

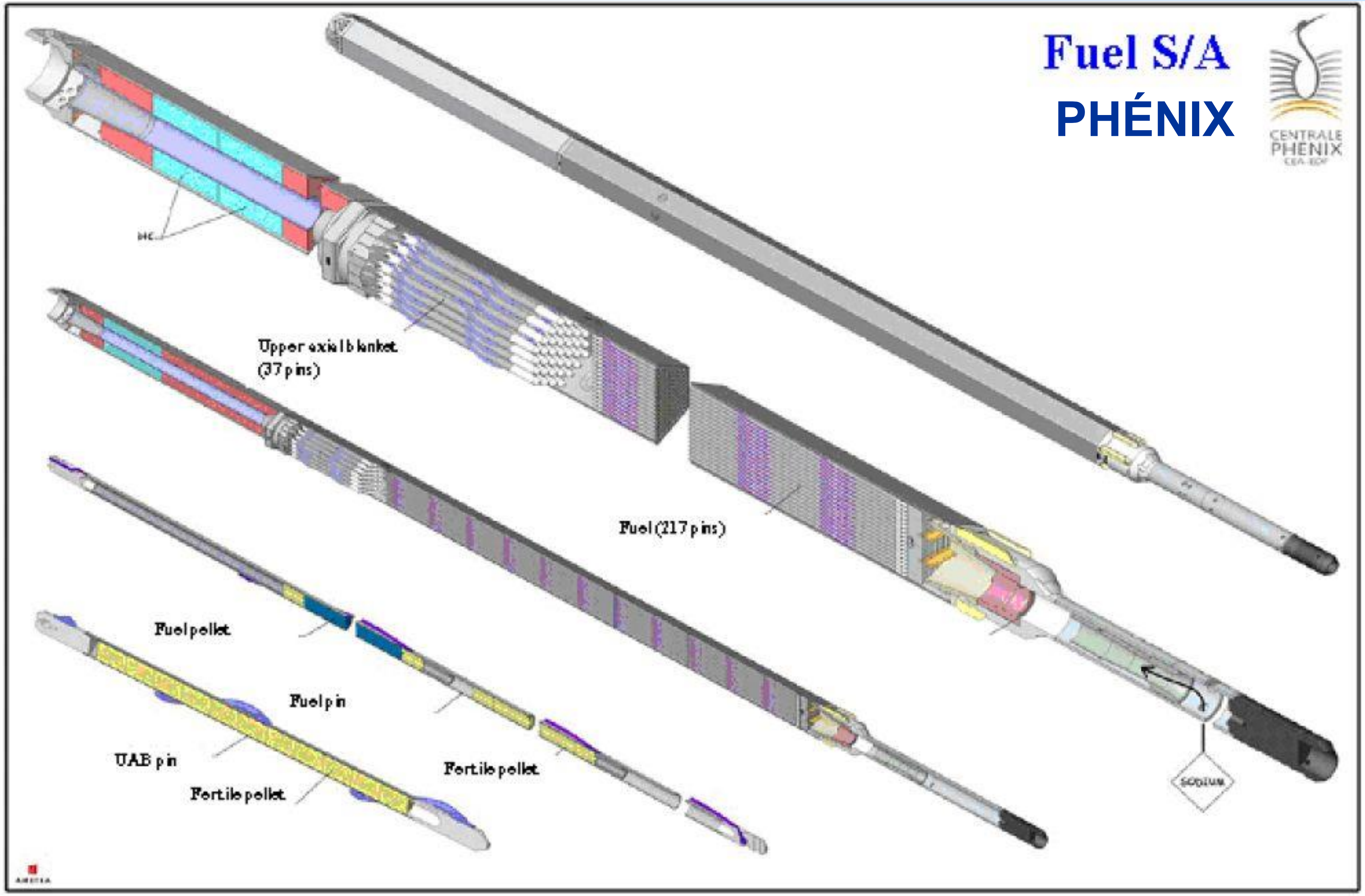






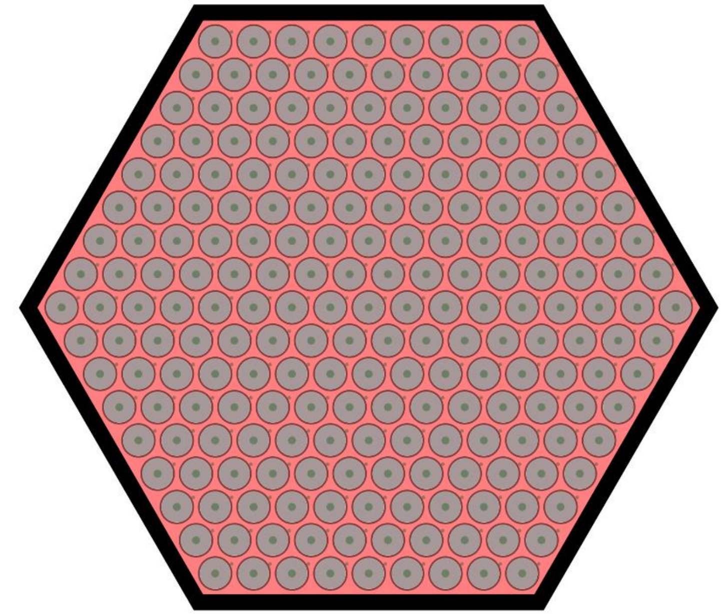
BOR-60

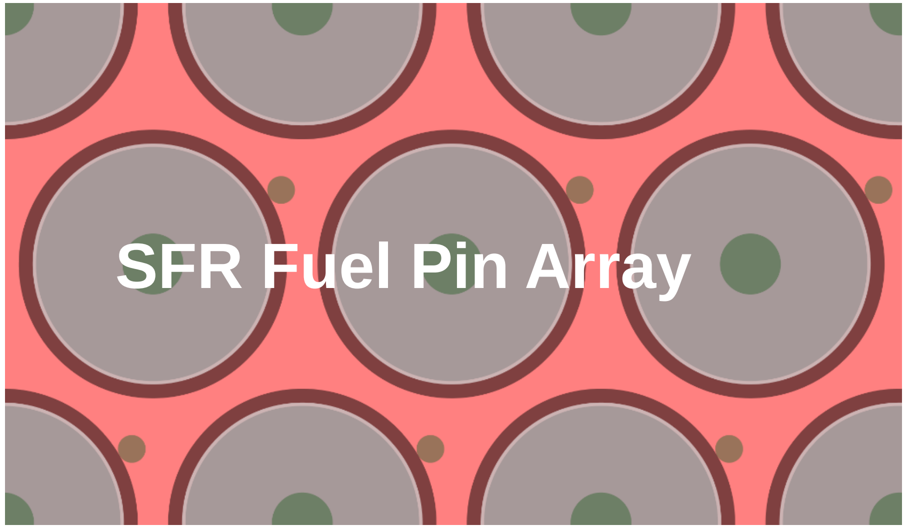
Phenix SFR Fuel Sub-Assembly



Fuel S/A: Pin Arrangement

	PWR/BWR	LMFNS
Fuel Pin/Rod OD, mm	9 - 14	6 - 7
Cladding Wall, mm	0.6 - 1	~0.5
Fuel Pellet Diameter, mm	7 - 10	5 - 6
Pitch-to-Diameter Ratio	1.4 - 1.6	1.1 - 1.2
Fuel Fraction	15 - 30 %	40 - 50 %
Coolant Fraction	50 - 70 %	35 - 50 %





Large Fuel Fraction:

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

Fast Reactor Coolants: Neutronic Considerations

- Neutrons interact with the atoms of the coolant
- The strength of the overall effect is governed by the probability of a particular interaction (absorption or scattering) and the number density of the coolant atoms
- Absorption removes neutrons from the system
- Scattering causes the neutrons to “bleed” energy thus slowing them down (moderation)
- Both of these mechanisms add negative reactivity
- If the coolant is removed (lost or “voided”), the loss of negative reactivity is equivalent to an insertion of positive reactivity:

Void Reactivity effect

FR Coolants: *key physical properties* (1/3)

- Melting temperature: impact on the reactor's cold shutdown temperature for fuel handling
- Boiling point and liquid phase temperature range
- Thermal characteristics: Cp, λ, Prandtl number
- Thermal stability: decomposition close to high temperature, safety margin
- Density: impact on power pumping required, internal dynamic pressures, seismic behavior
- Interaction with structural materials: Dissolution (solubility of metal elements), corrosion, embrittlement and potential mass transfer
- Chemical reactivity with surrounding fluids (air, water, organic products, etc) and impact on operating safety

FR Coolants: *key physical properties* (2/3)

- Interaction with primary coolant when used as different intermediate coolant: corrosion, contamination.
- Interaction with ECS coolants (water, SC CO2, etc) when used as different intermediate coolant: corrosion, contamination
- Transparency/opacity: special in-service inspection methods
- Vapor pressure: impact on aerosols production and deposition
- Ability to “block” the Tritium produced in the primary system (Tritium is the only radioactive contaminant capable to cross metal walls)
- Capability to be purified and meet quality standards

FR Coolants: *key physical properties* (3/3)

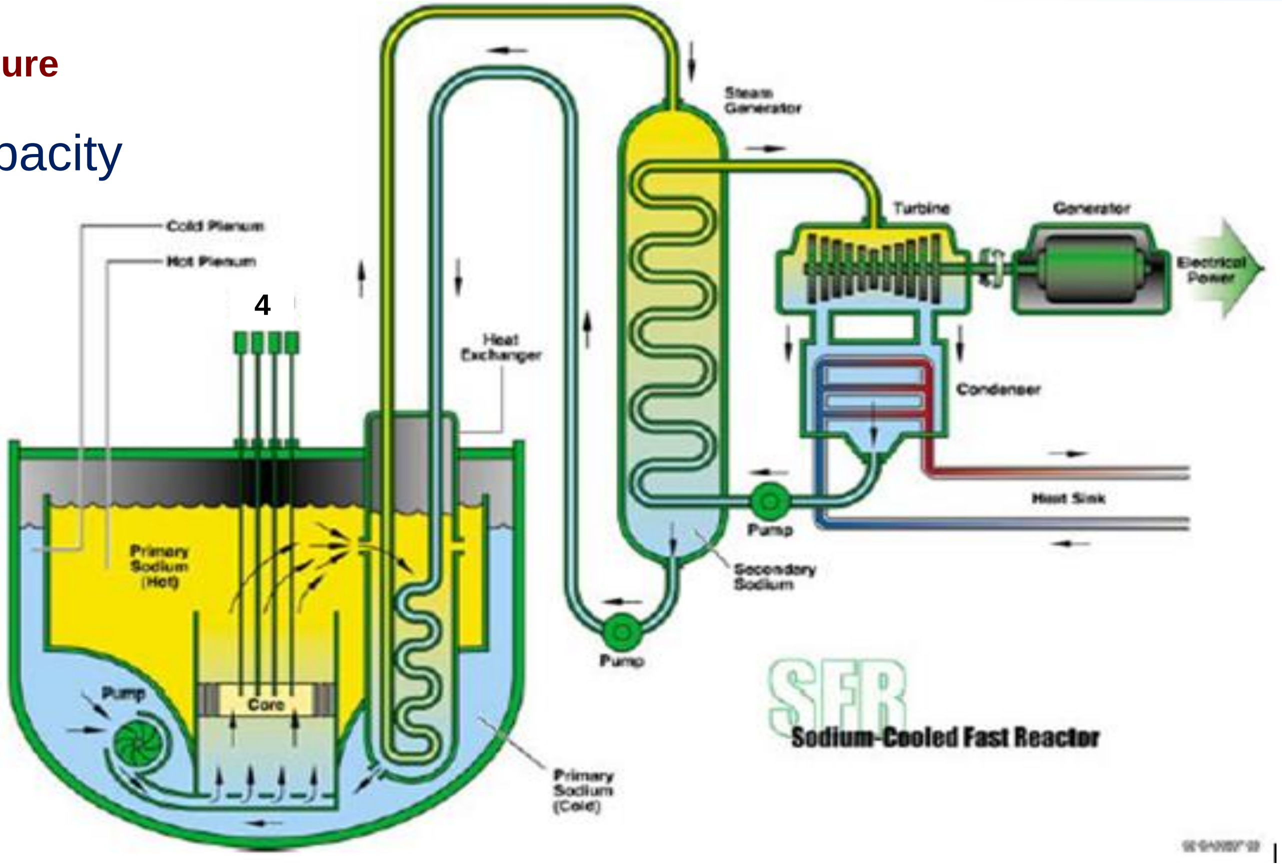
- Potential structures wetting: impact on fluid-material interactions, instrumentation, quality of ultra-sound transmission, maintenance
- Toxicity: need to confine the coolant during handling and repair
- Possibility of processing during dismantling, including specific systems like cold trapping
- Production of wastes and their processing during operation and dismantling
- Availability in nature
- Cost

Sodium Properties: *several advantages*

- 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



17

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

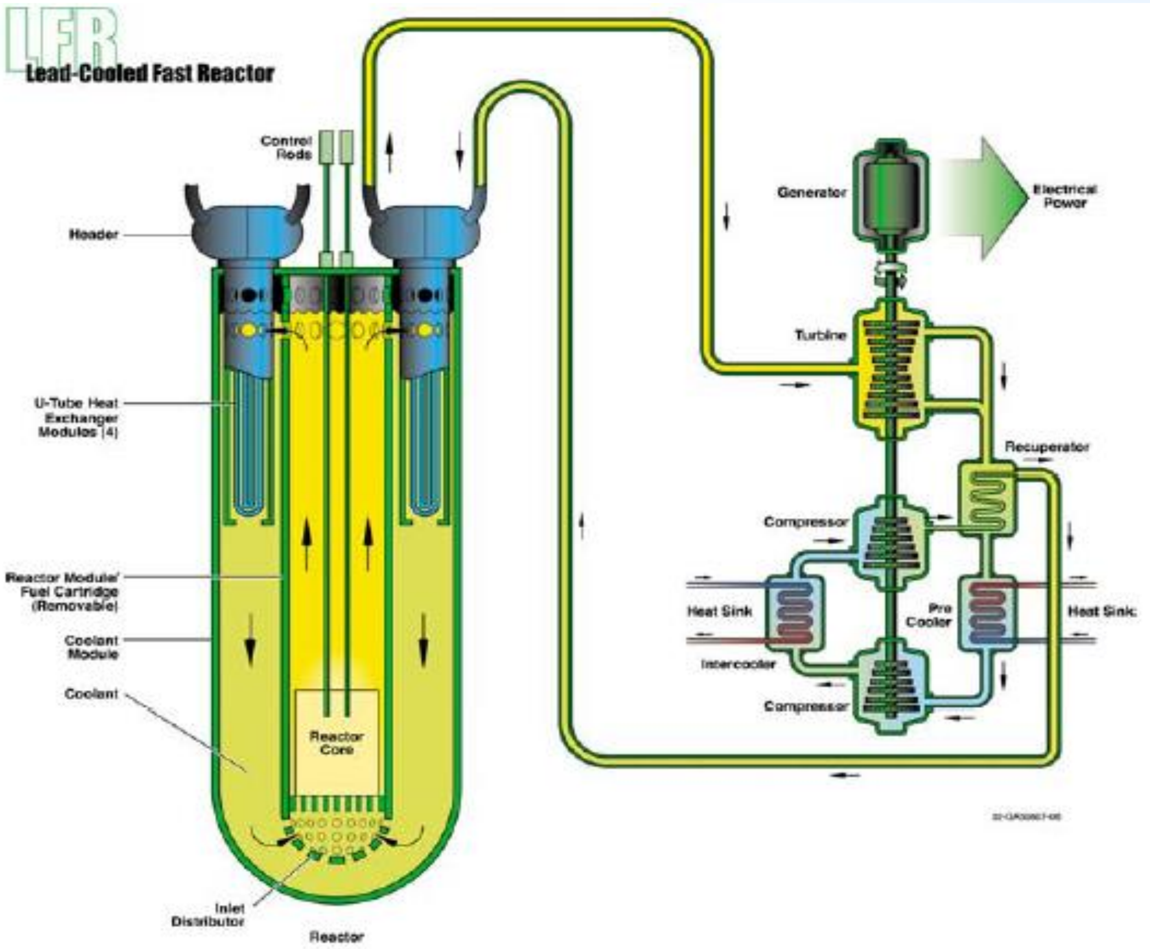
- **Important: Chemical reactivity with air**
  - Can induce Na fire
  - Need inert zones and confinement
  - Need early detection
- **Opacity**
  - Need specific equipments for under-sodium viewing and measurements

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

- 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*

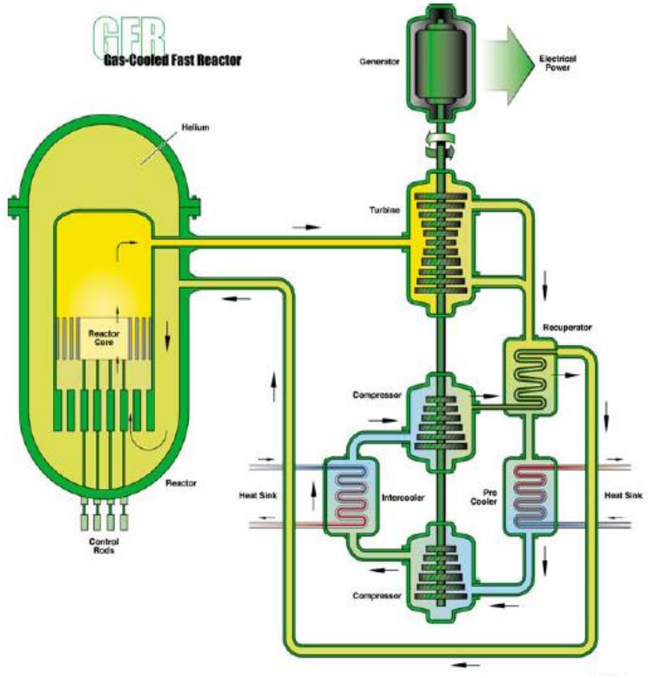
- **High density** (*also an advantage due to reduced risk of re-criticality in case of core melting*)

- **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



21

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

Coolant Thermal-Physical Properties

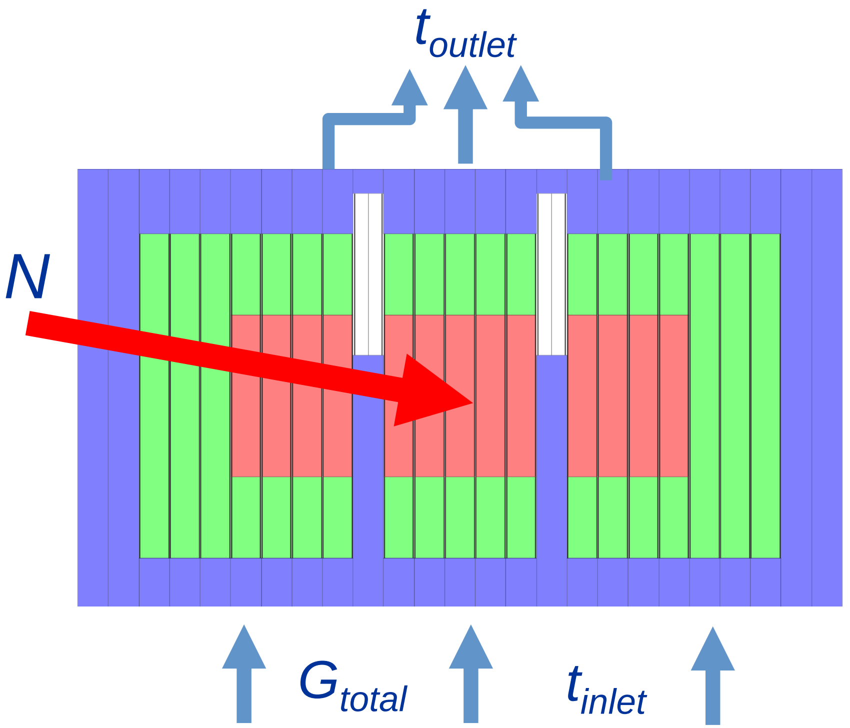
		H_2O	Na	Pb	LBE	He
Atomic Weight		18	23	207	208	4
Melting Point	°C	0	97.8	327.4	123.5	
Boiling Point	°C	100/ 287	892	1737	1670	-267
Density	kg/m3	1000	832	10460	10080	0.178
Vol. Heat Capacity	MJ/m3/K	4.18	1.05	1.53	1.47	0.0009
Specific Heat Capacity	J/kg/K	4180 5682	1264	147	146	5200
Thermal Conductivity	W/m/K	0.6	70	18	15	0.152 0.238
Kinematic Viscosity	$m^2/s \times 10^6$	1 0.12	0.28	0.11	0.13	0.15 0.71
cold 20 °C hot water 300 °C hot LM/He 500 °C						

FastRTeam

Please Don’t Sleep!!!

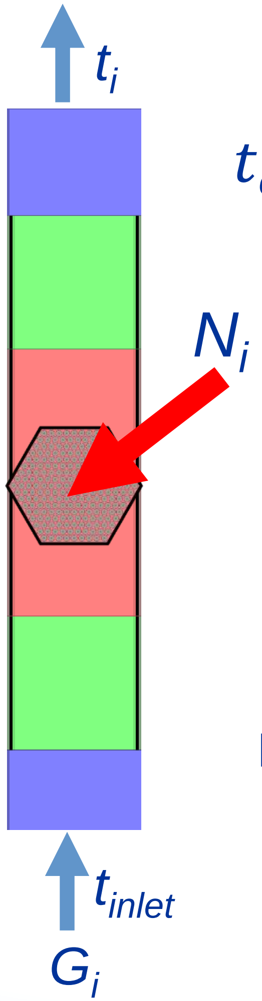
Reactor Core Power Balance

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



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

$$N = \sum N_i$$

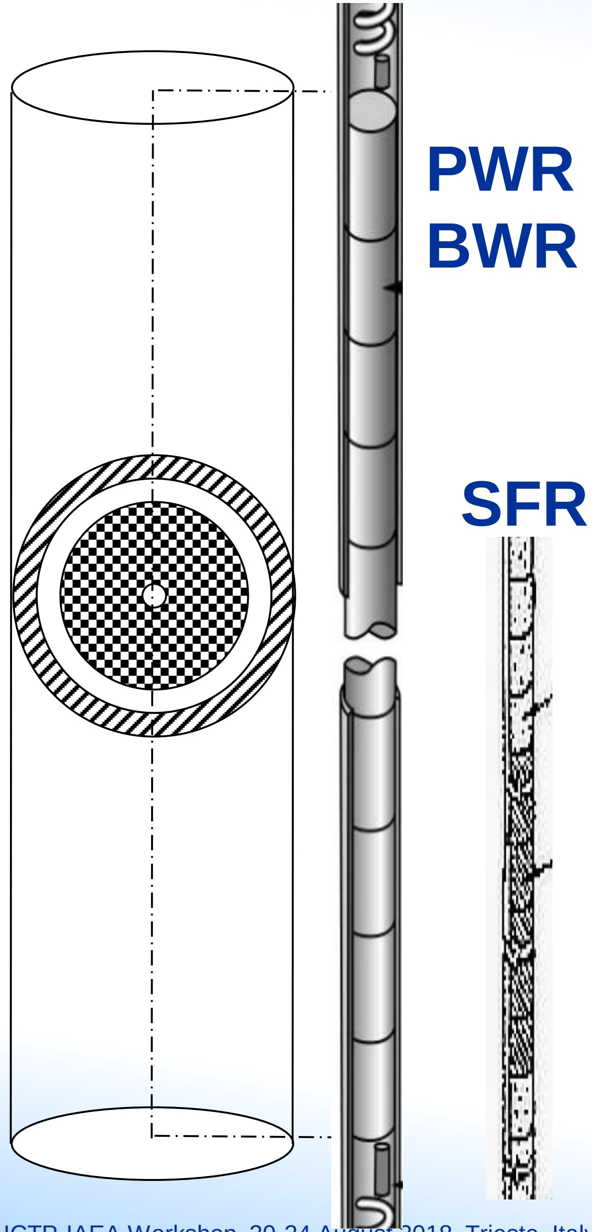
$$G = \sum G_i$$

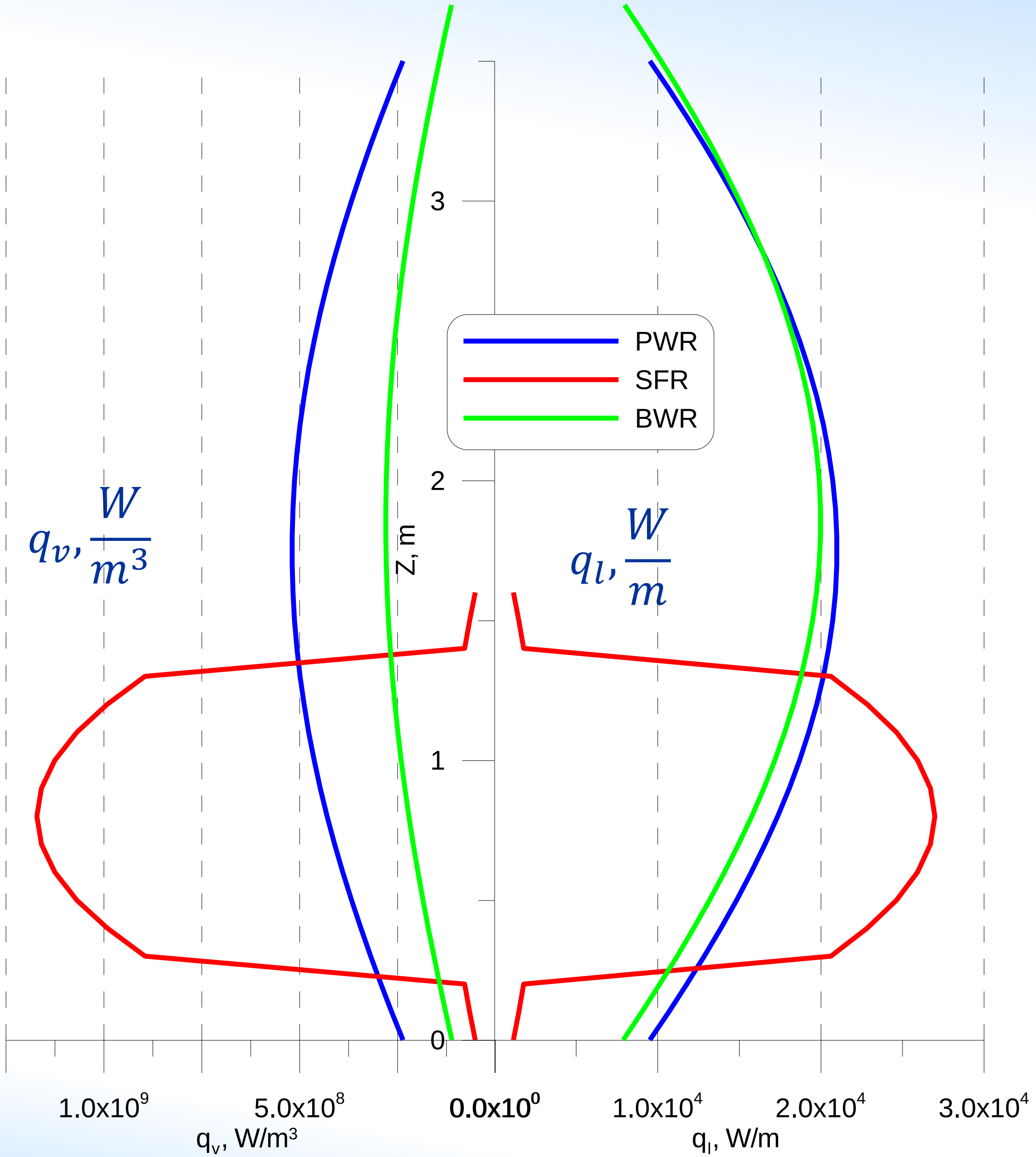
Power Density

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

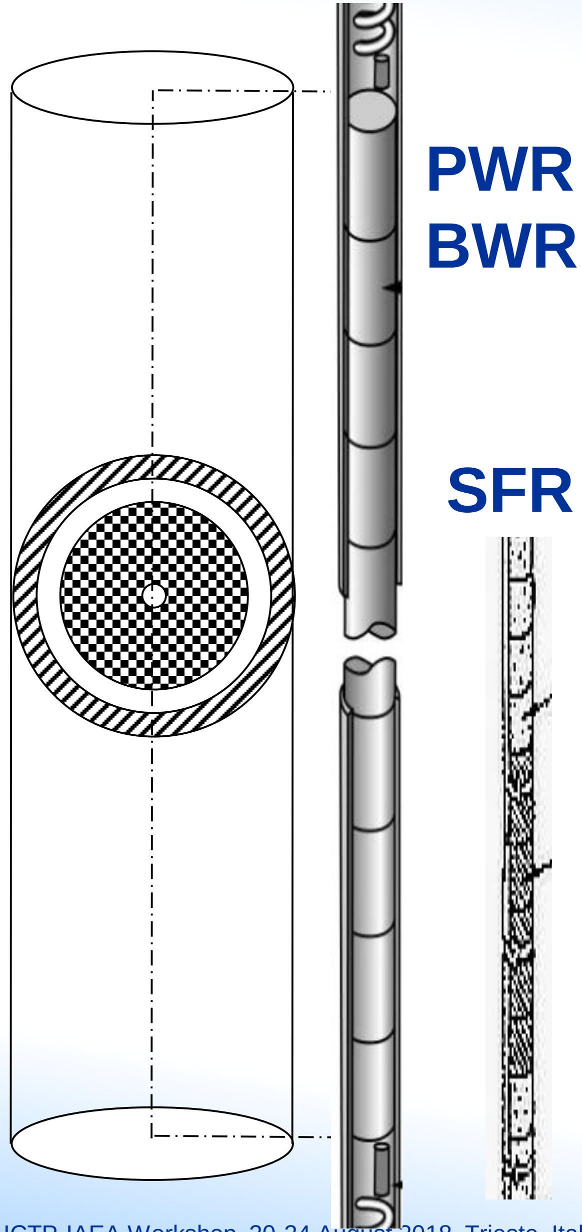
$$q_v = \frac{dN}{dV}$$

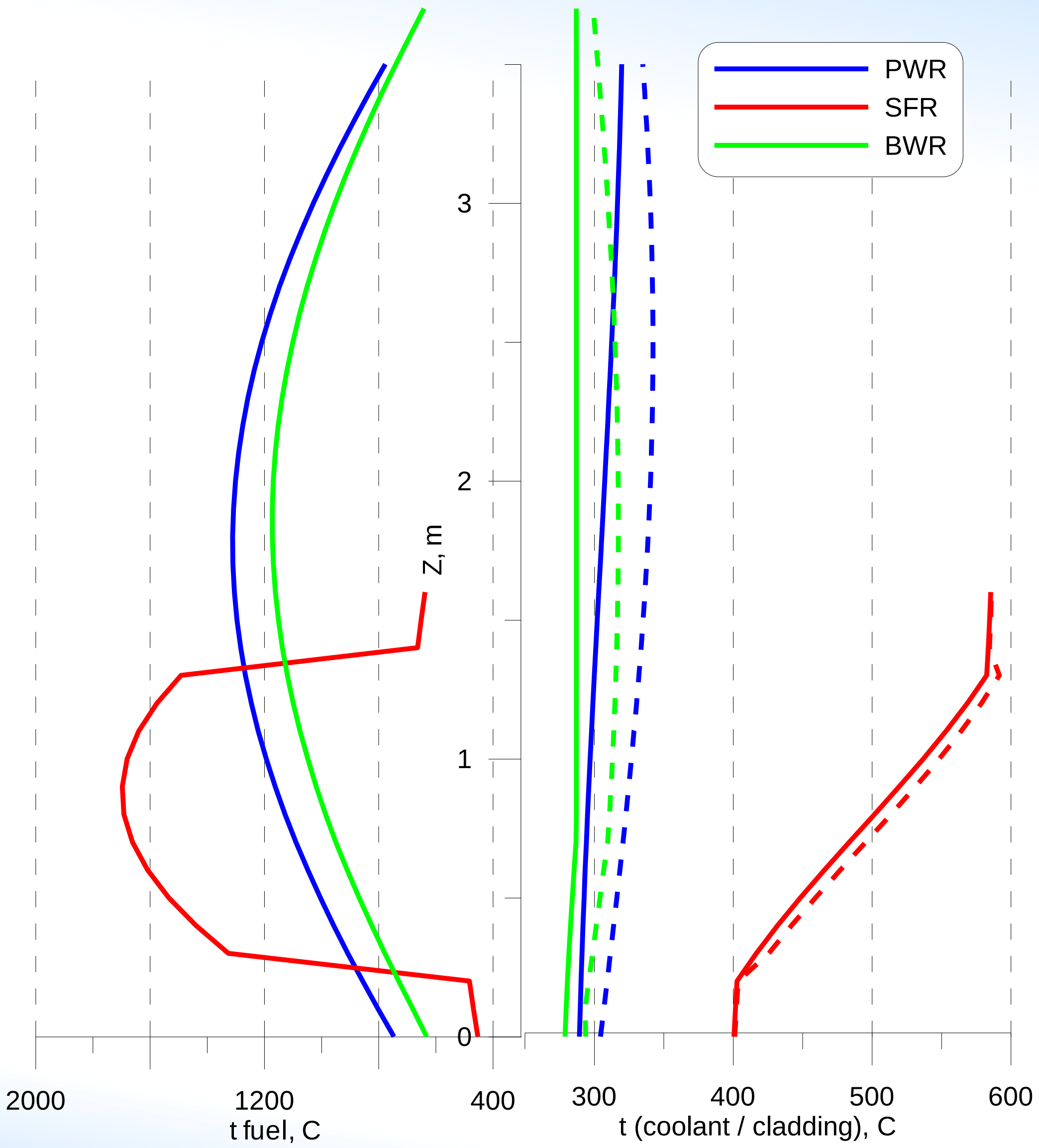
Fuel Pin: Power Density



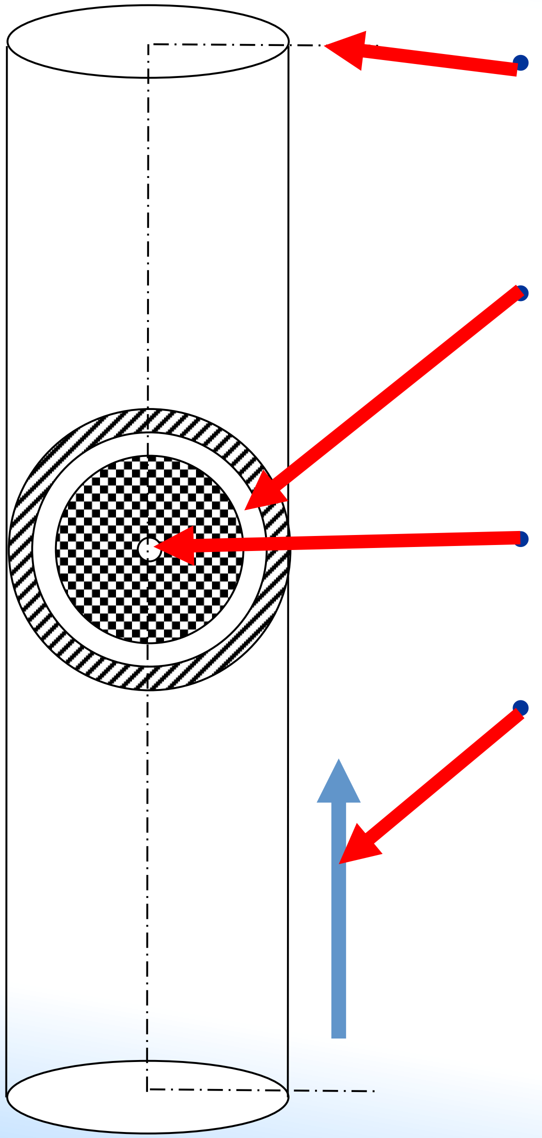


Fuel Pin: Temperature Profiles



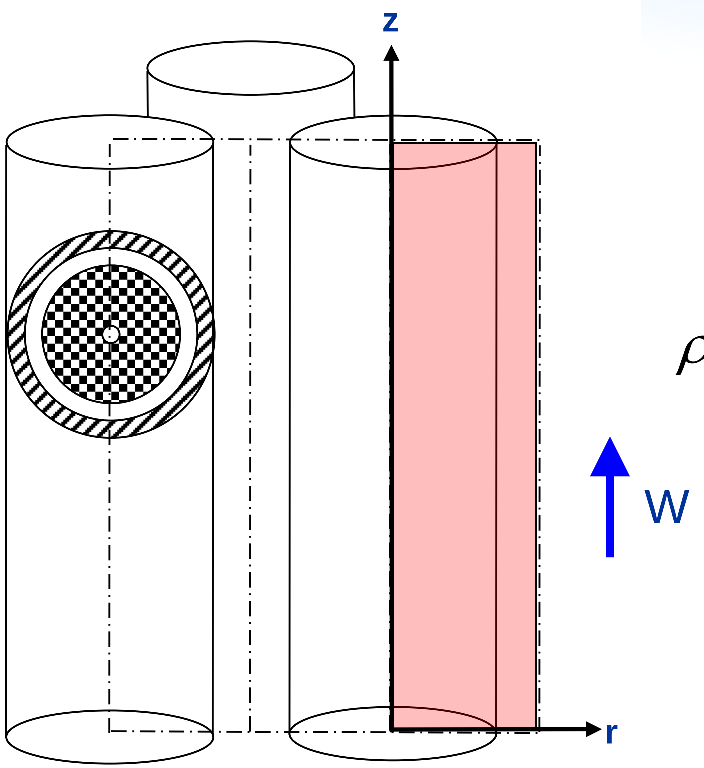


TH Limiting Parameters



- 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

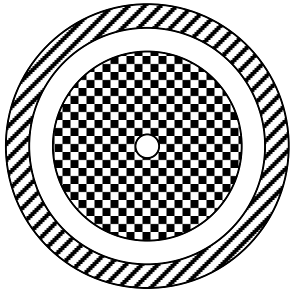
$$\rho c_p \frac{\partial t}{\partial \tau} = \frac{1}{r}\frac{\partial}{\partial r}\left(\lambda(t) r \frac{\partial t}{\partial r}\right) + \frac{\partial}{\partial z}\left(\lambda(t)\frac{\partial t}{\partial z}\right) + q_v$$

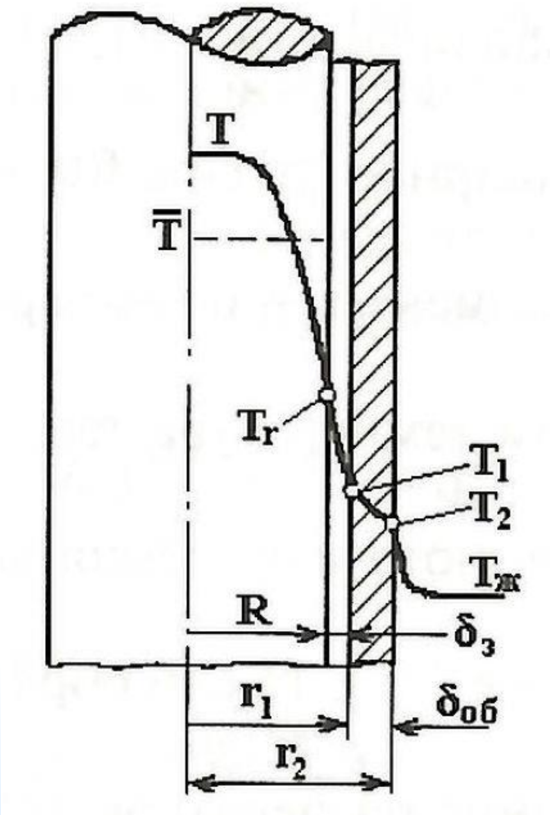
(3D Effects are neglected)

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)

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

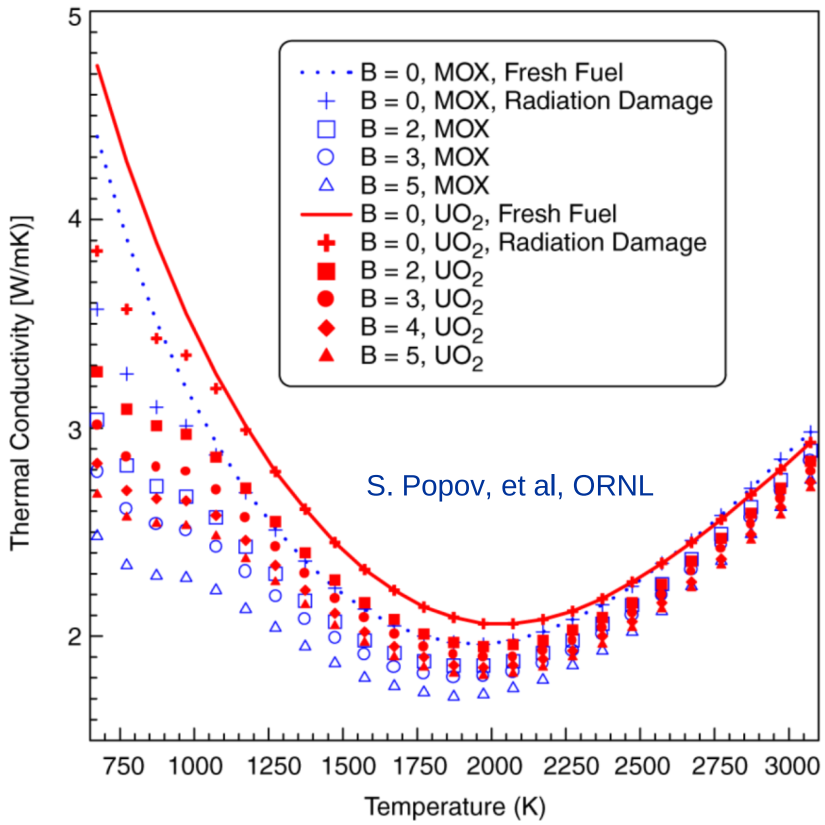
$$t_{coolant}(z) = t_{inlet} \int_{-h/2}^{z} c_p G_i q_l(z) dz$$

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

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

But…

Non-Linear Effects



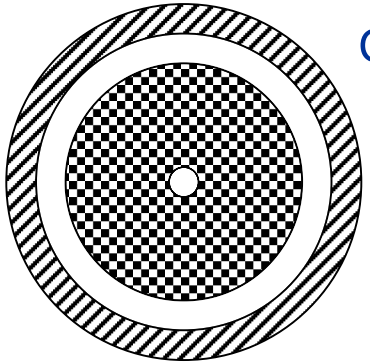
$$\rho c_p \frac{\partial t}{\partial \tau} = \frac{1}{r}\frac{\partial}{\partial r}\left(\lambda(t) r \frac{\partial t}{\partial r}\right) + \frac{\partial}{\partial z}\left(\lambda(t)\frac{\partial t}{\partial z}\right) + q_v$$

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!

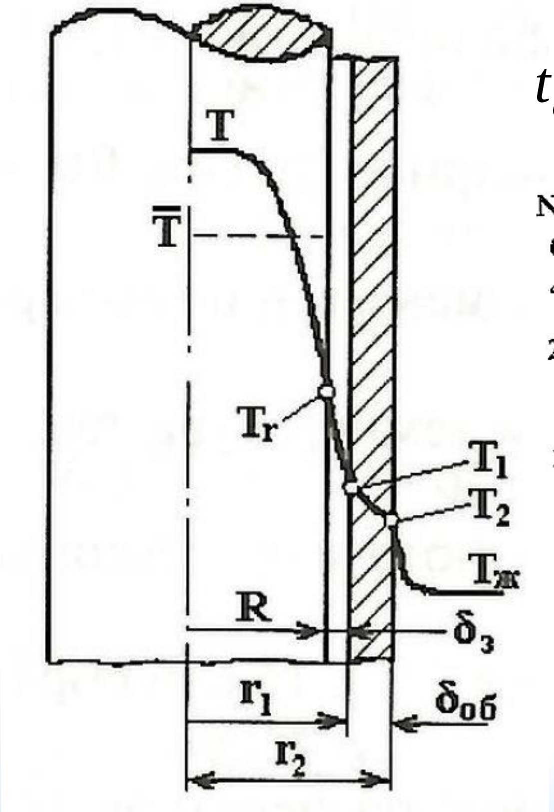
$$\Delta t_{gap} = \frac{q(z)\Delta_{gap}}{\lambda_{gap}}$$

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

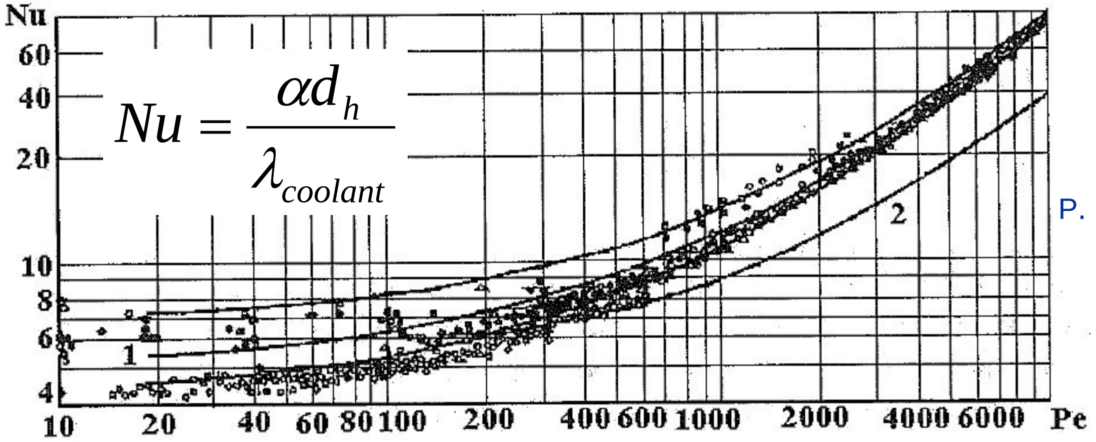
Coolant-Cladding Heat Transfer

Energy Conservation in Coolant

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



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

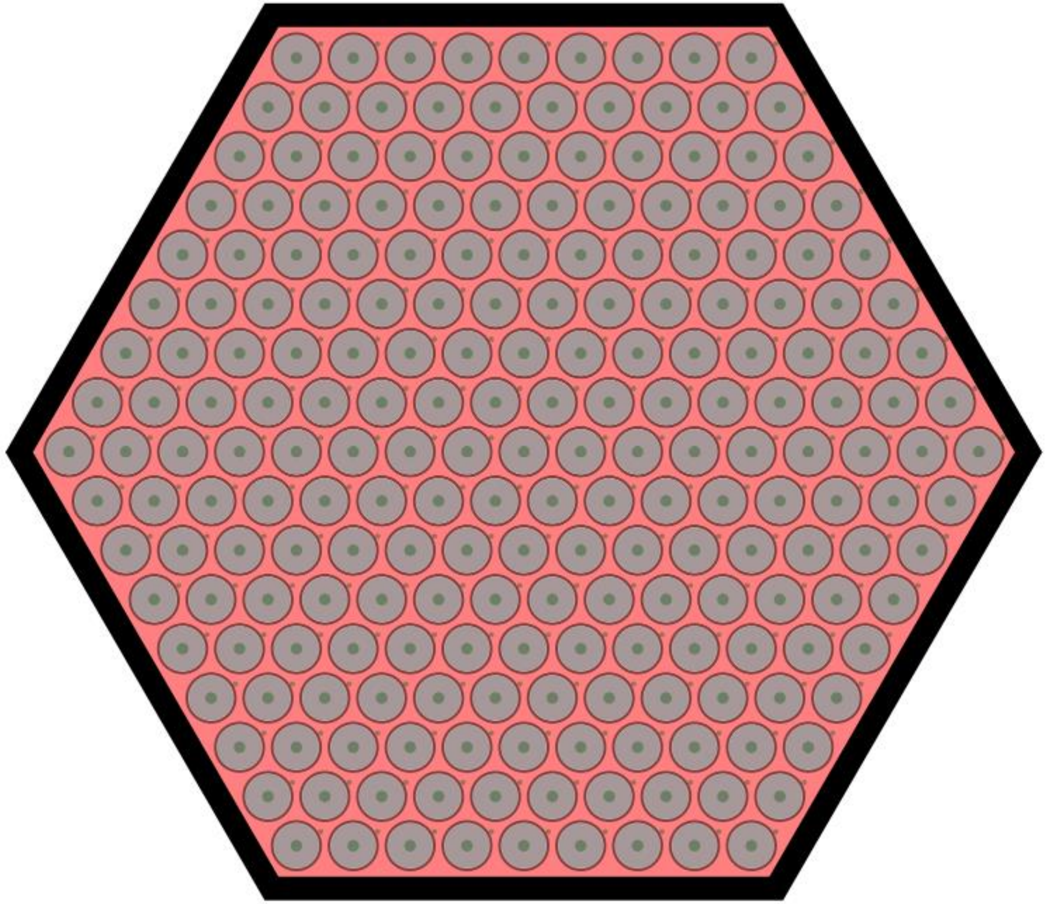


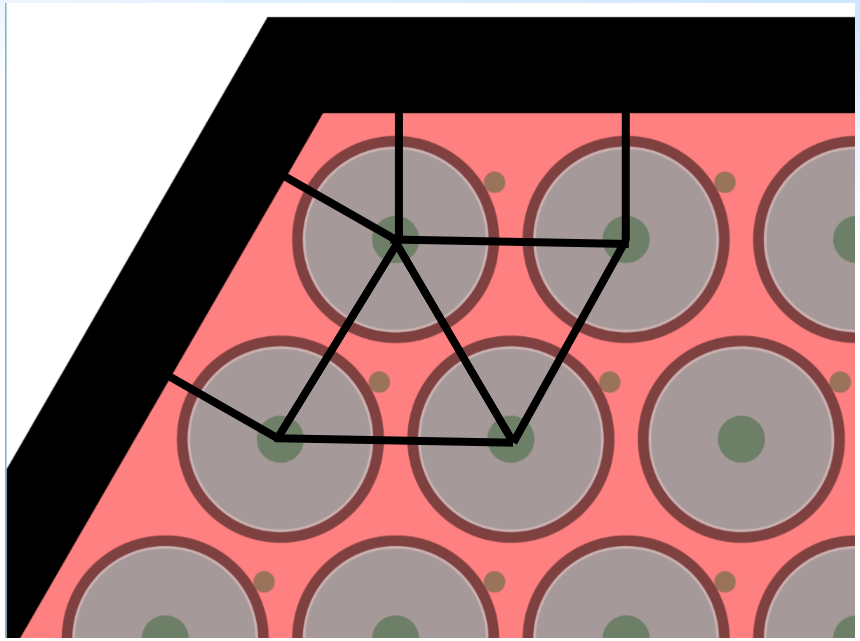
$$Nu = \frac{\alpha d_h}{\lambda_{coolant}}$$

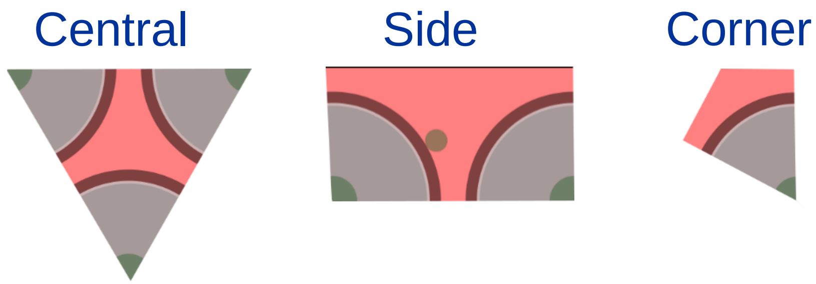
P. Kirillov, IPPE

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

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







Power-to-Flow Ratio in Subchannel

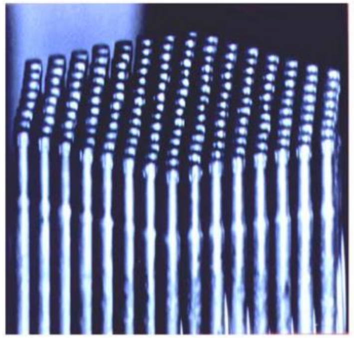
$$\frac{Power}{Flowrate} = \frac{Heat\ Flux \times \Pi}{Velocity \times Area}$$

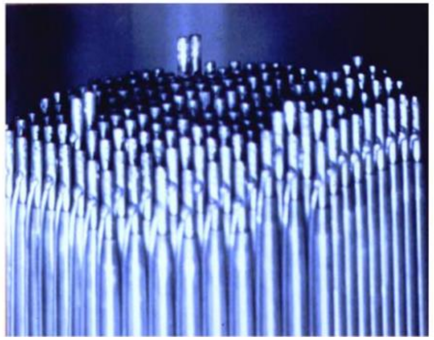
Power-to-Flow Ratio in Central Subchannel is

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

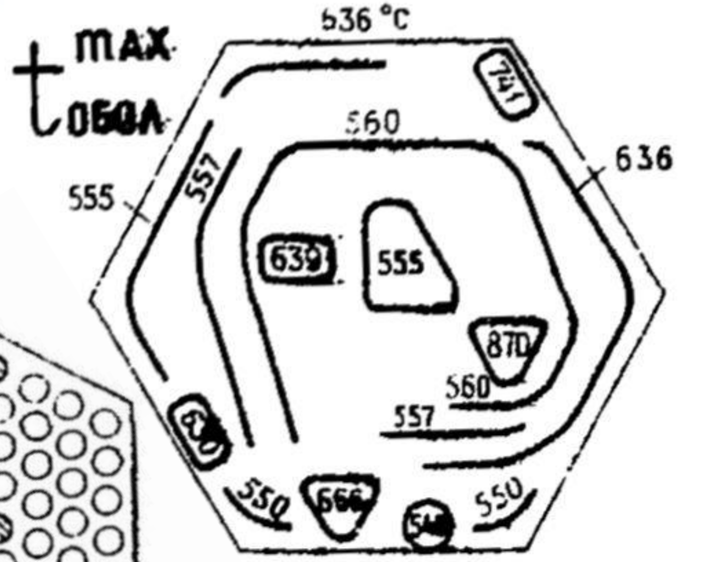
S/A Deformation Under Irradiation

FFTF





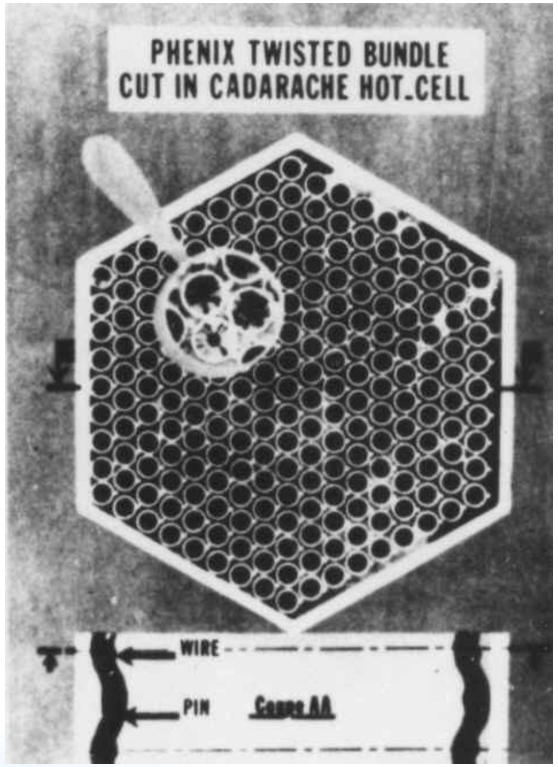
Random Deformation Inside S/A

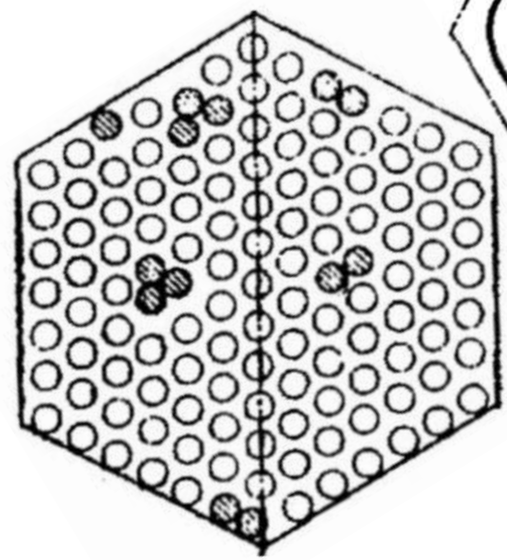


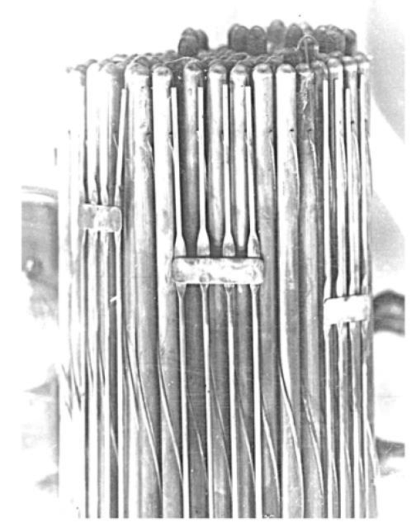
Max Clad Temperature, C

A.Sorokin, et al, IPPE

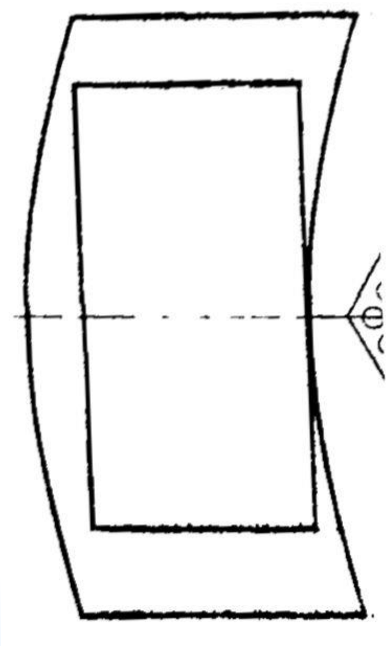
Phenix







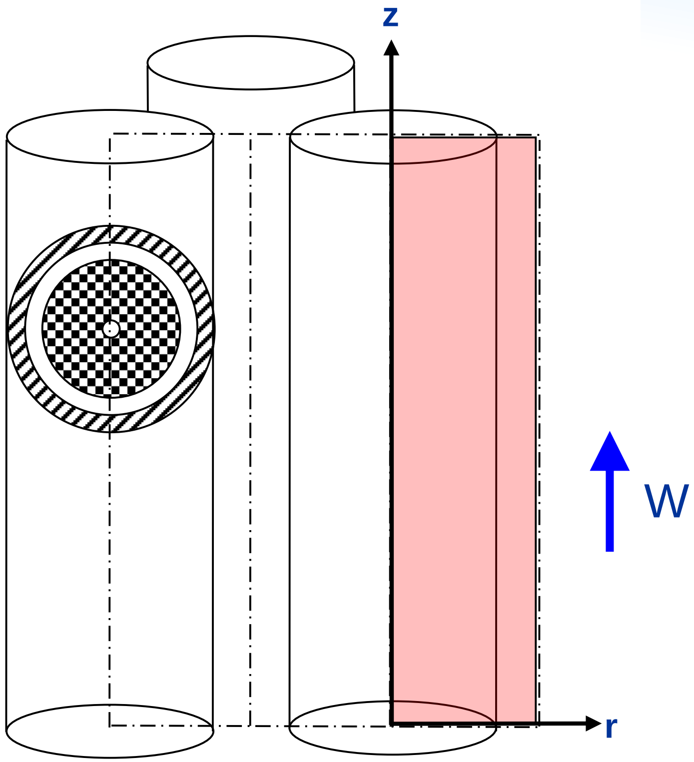
BN-600

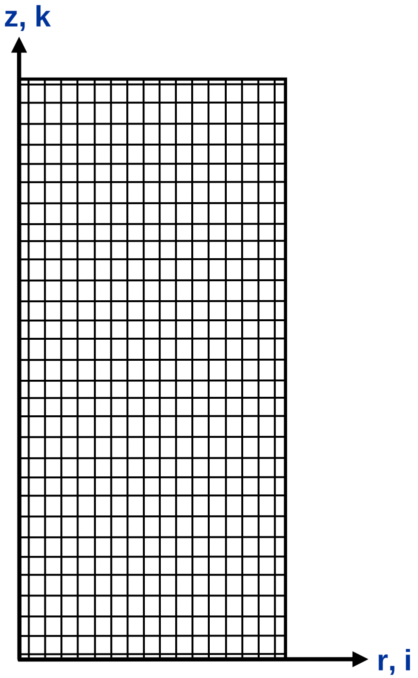


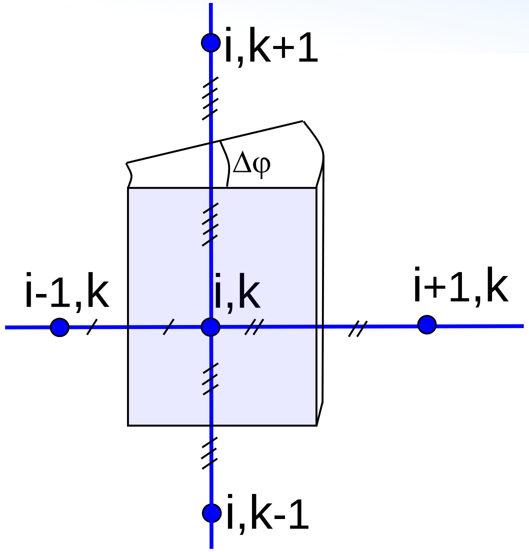
Axial Deformation

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

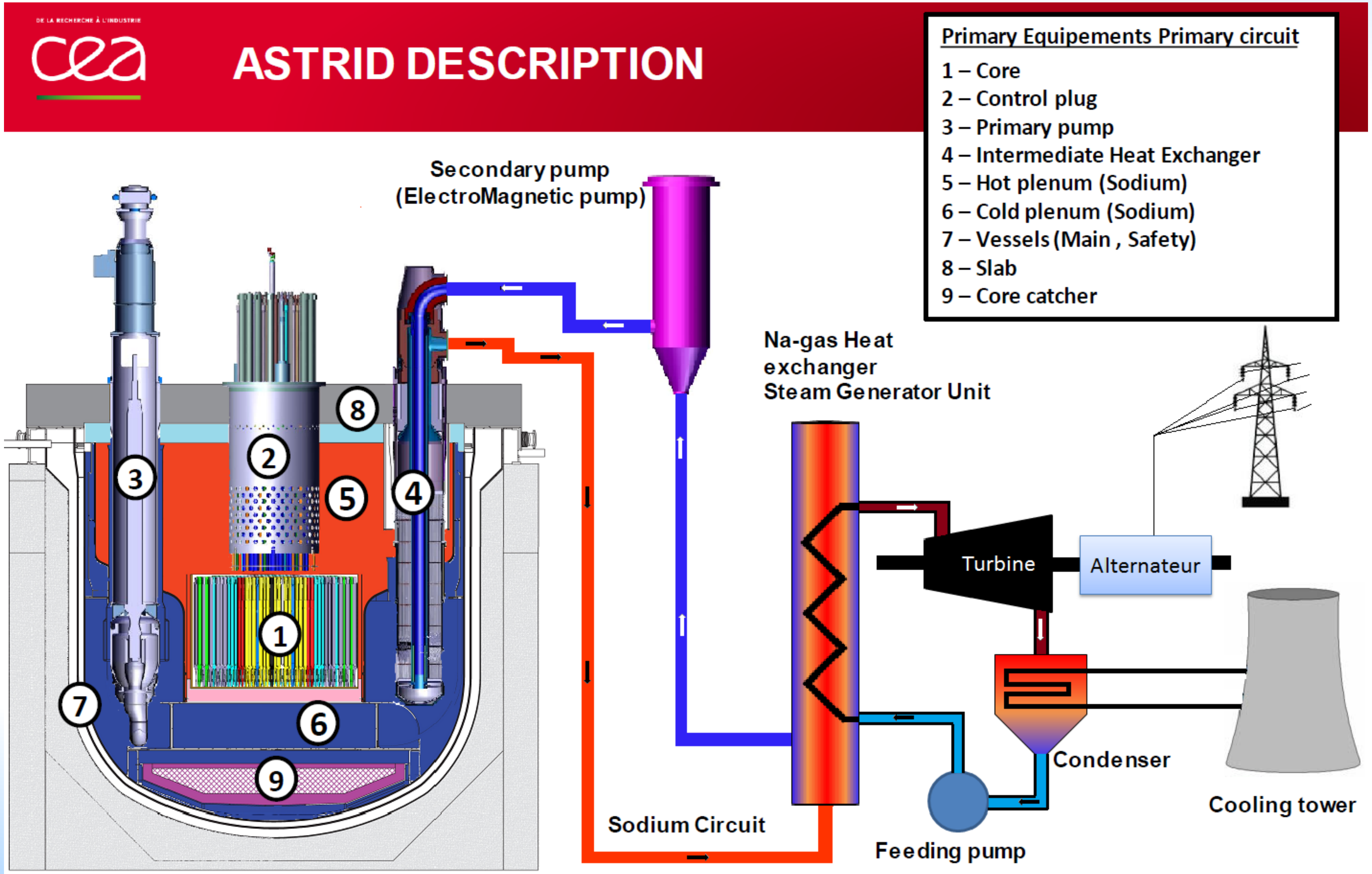
Reactor Accidental Transient Scenarios

- 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

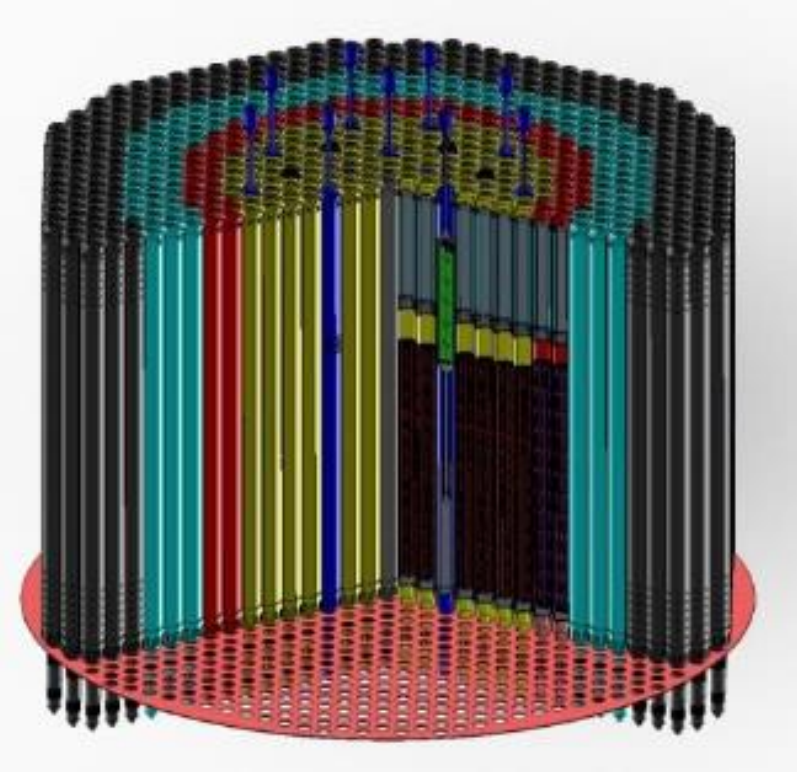
ASTRID: Advanced Sodium Technological Reactor for Industrial Demonstration

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

IAEA

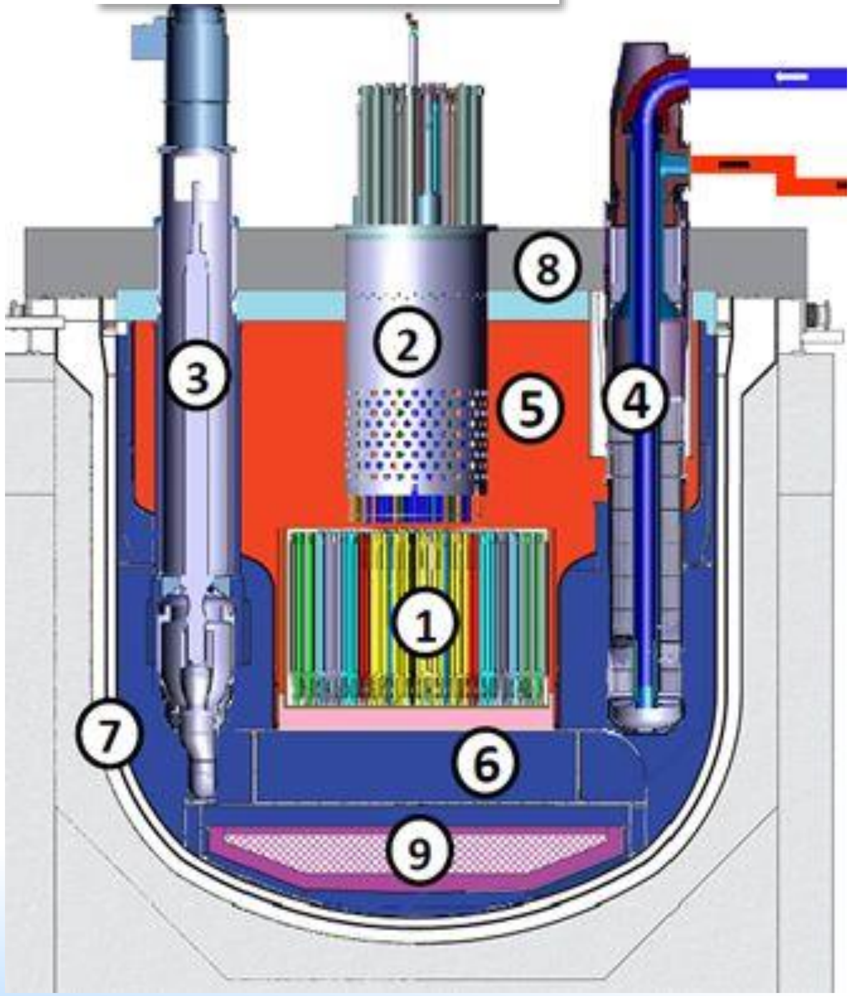


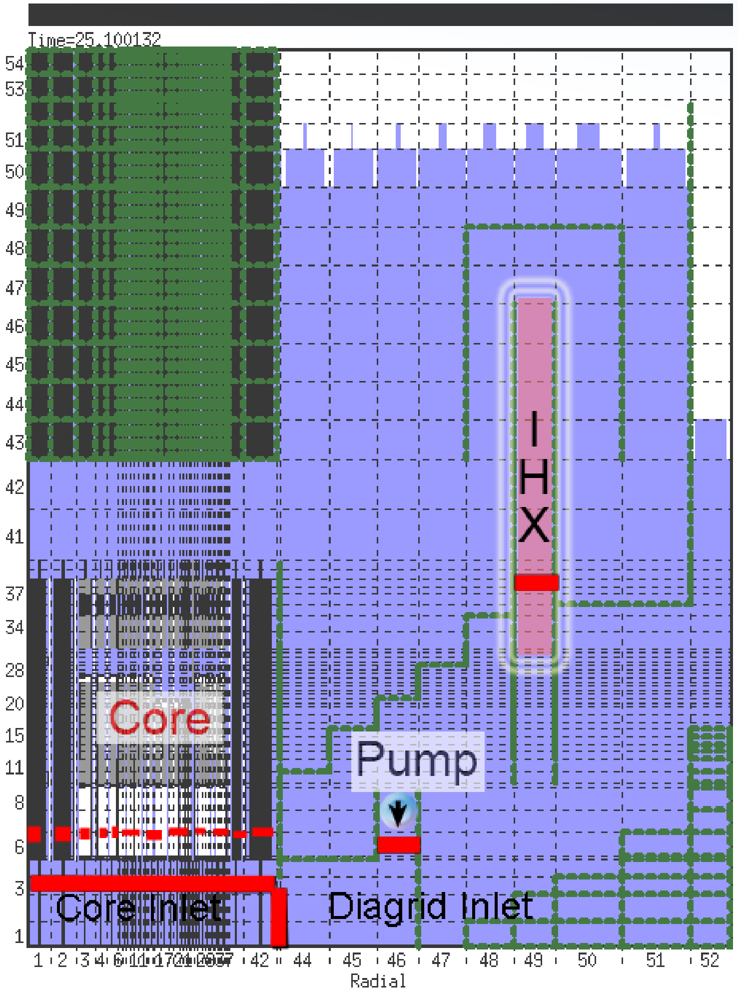
Innovative Core Design for Enhanced Safety Features



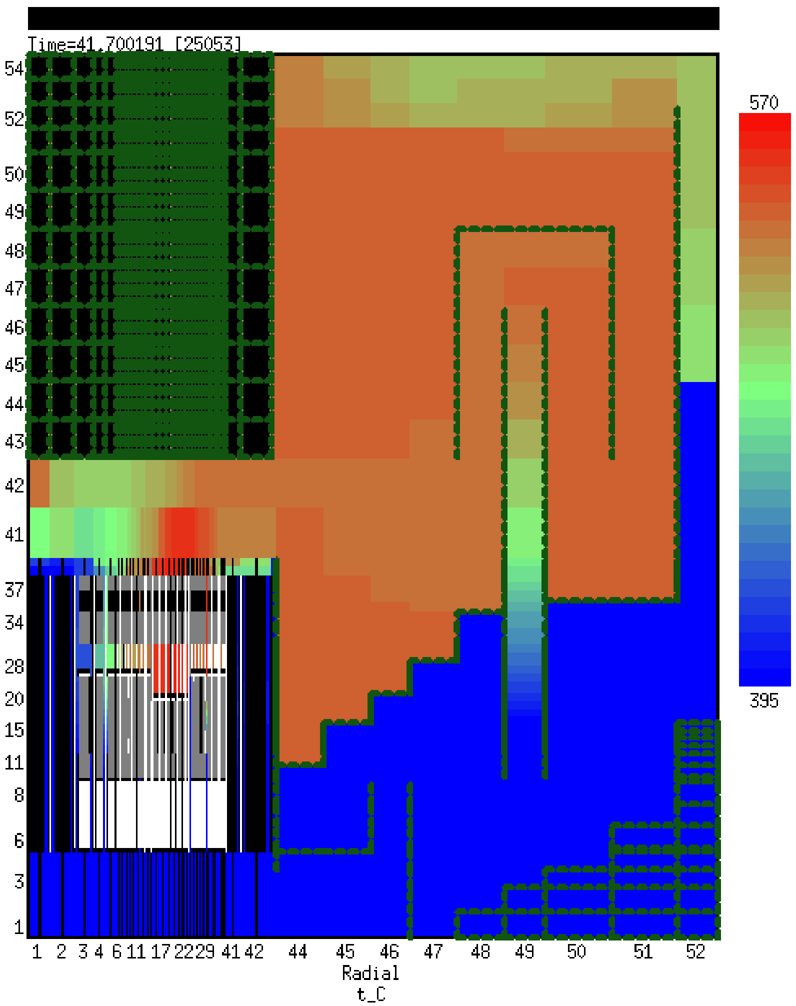
ASTRID: ULOF Simulations with SAS4A and SIMMER-III codes

One of considered ASTRID designs





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



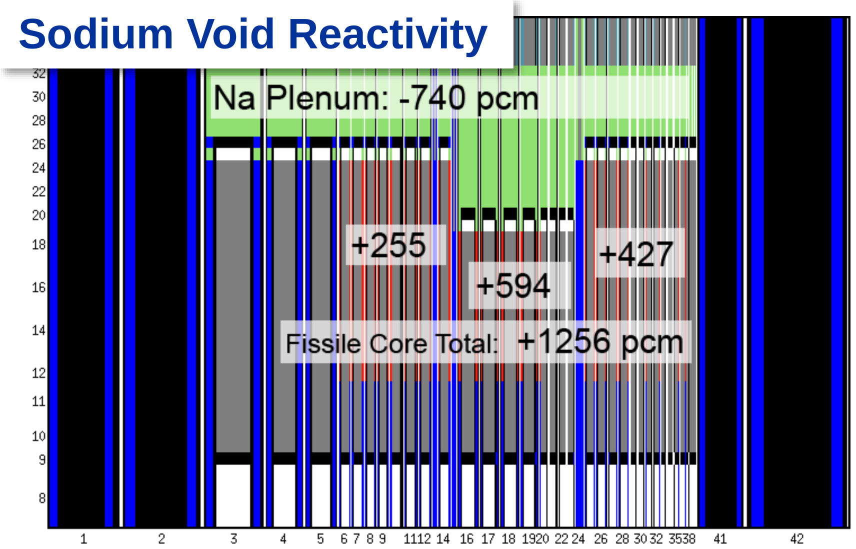
Na Temperature Distribution

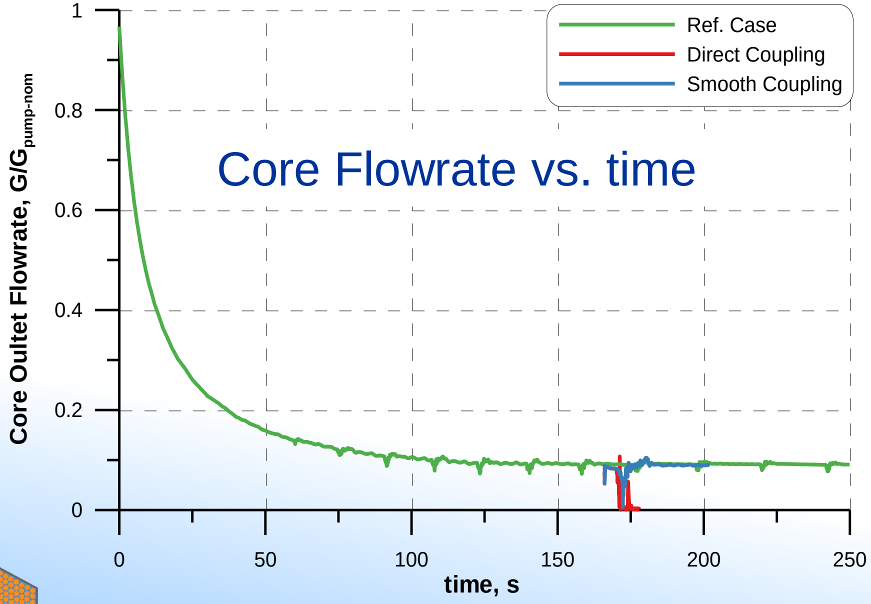
V. Kriventsev, 2015, KIT, Germany

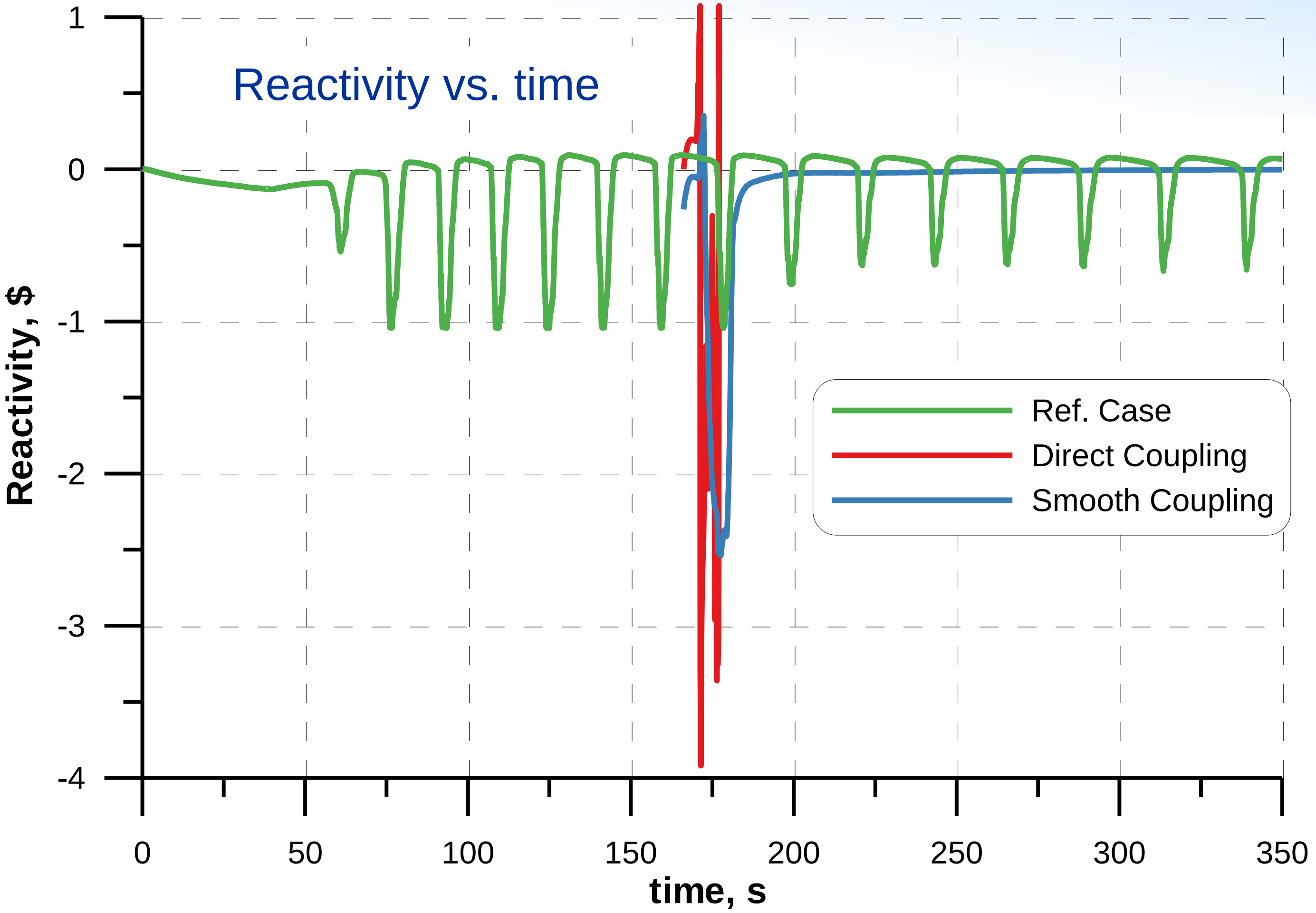
ULOF Simulations with SAS4A and SIMMER-III Codes

IAEA

Sodium Void Reactivity





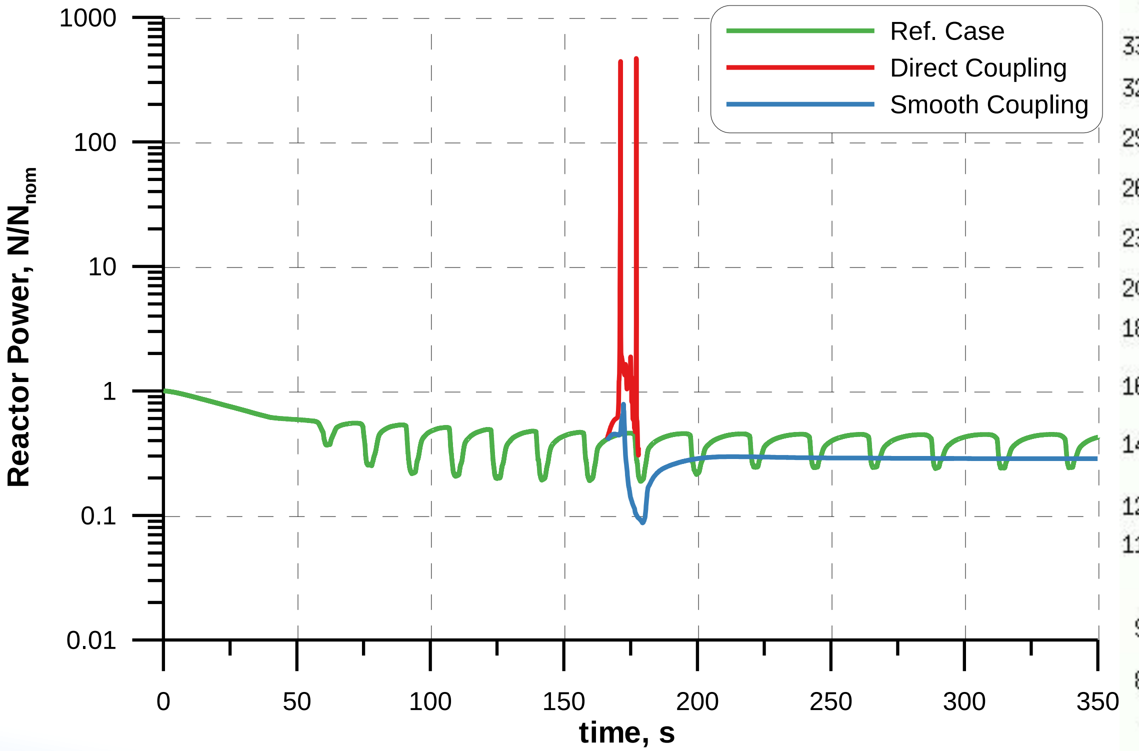


V. Kriventsev, 2015, KIT, Germany

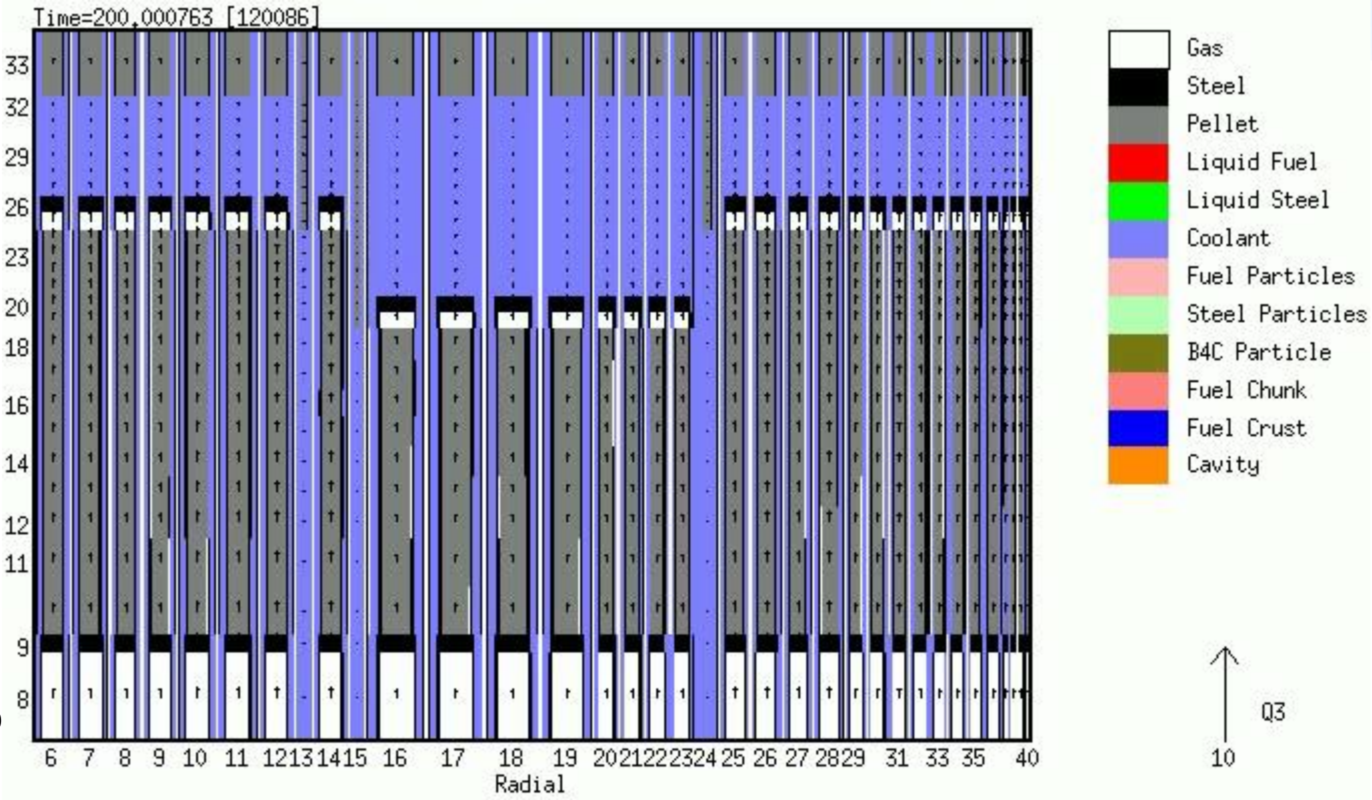
ULOF Simulations with SIMMER-III Code

IAEA

Reactor Power



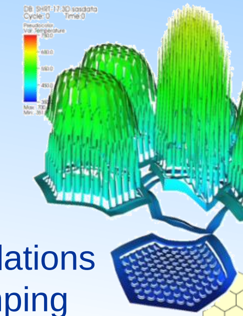
Material Distribution in Core vs. Time

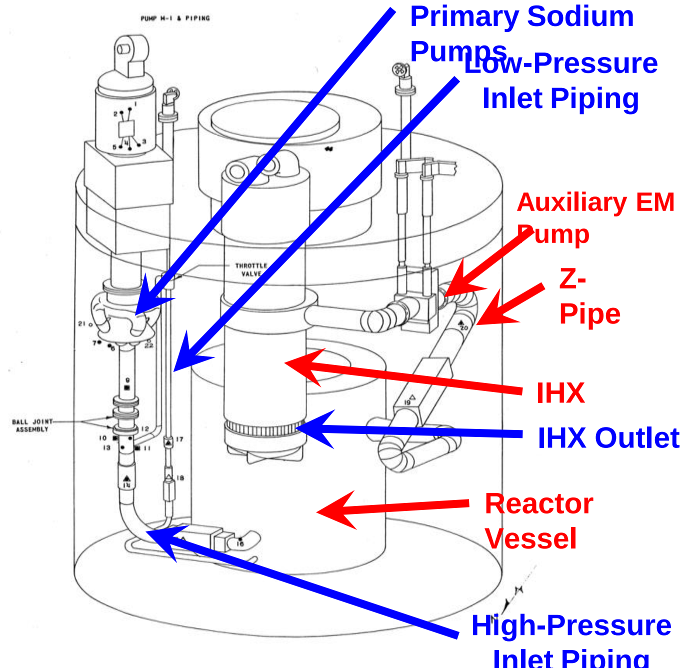


V. Kriventsev, 2015, KIT, Germany

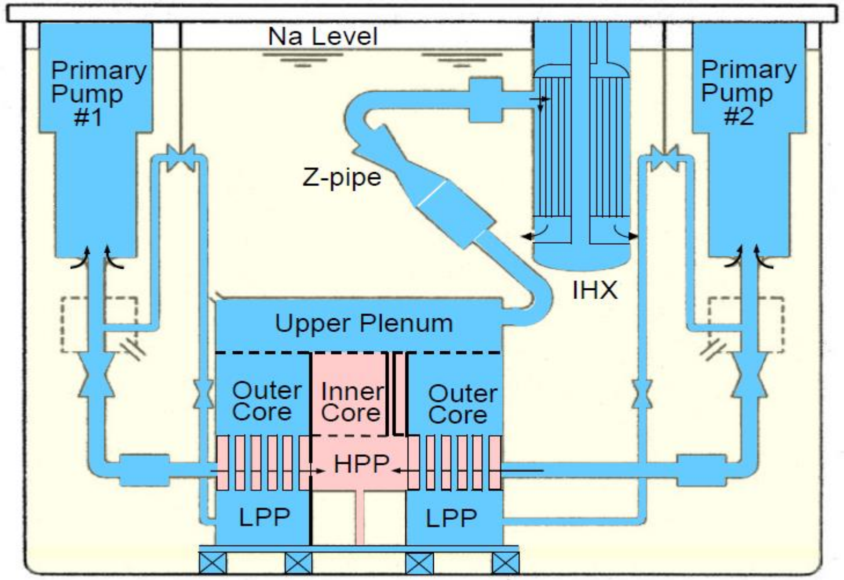
Kindly Wake Up Now !!!

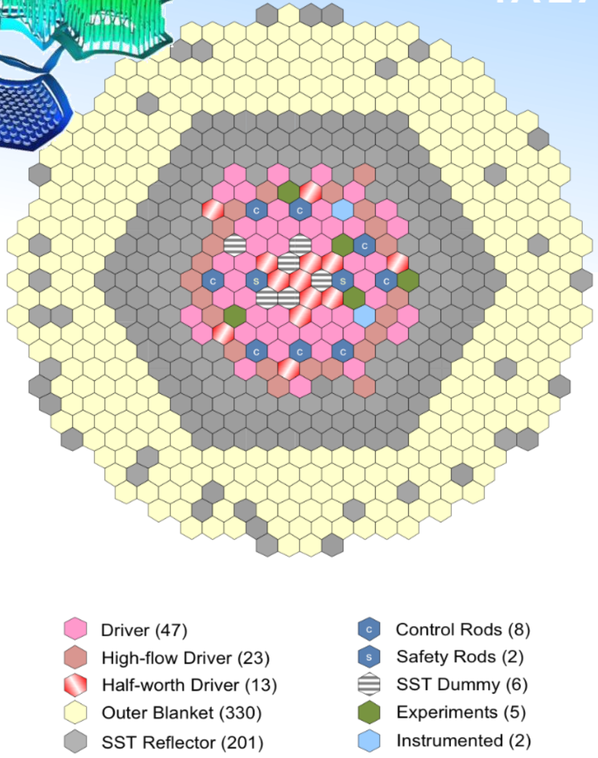
CRP on Benchmark Analysis of *EBR-II* Shutdown Heat Removal Test (*2012-2016*)

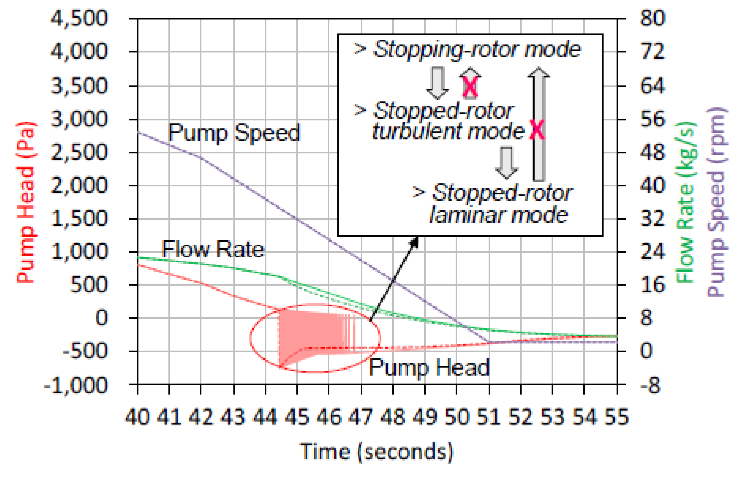


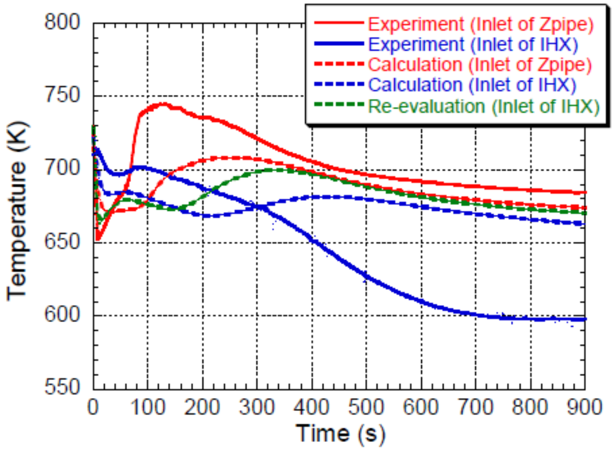


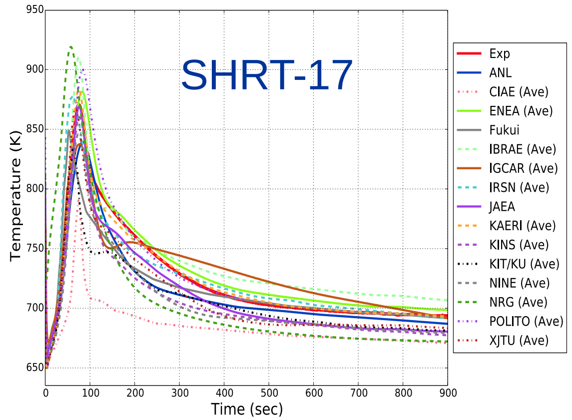
- Coupled Neutronics and Thermalhydraulic Transient Simulations
- SHRT-17 (Protected): Loss of normal and emergency pumping
- SHRT-45 (Unprotected): Loss of normal flow, scram disabled, station blackout





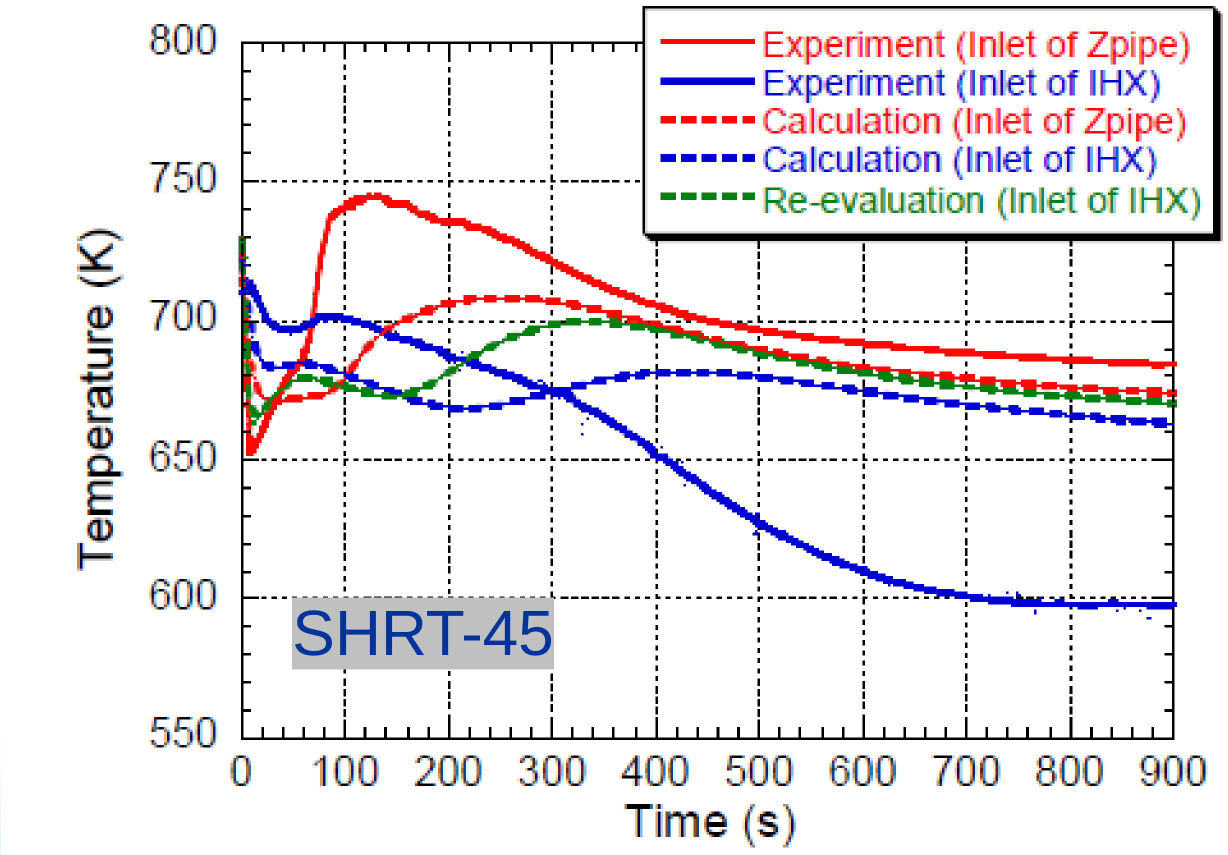


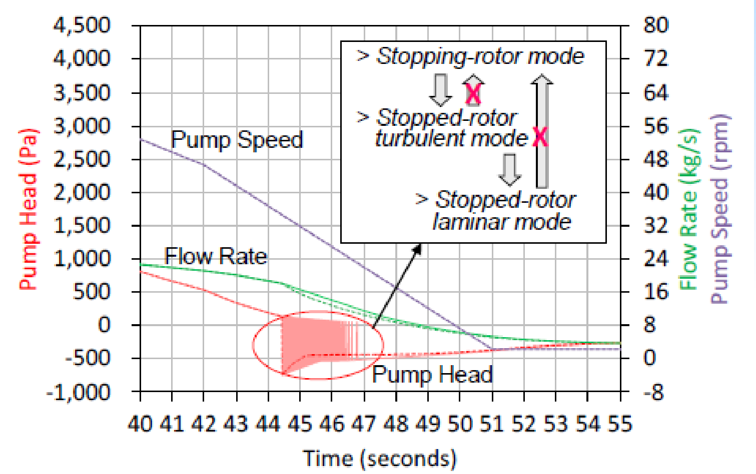


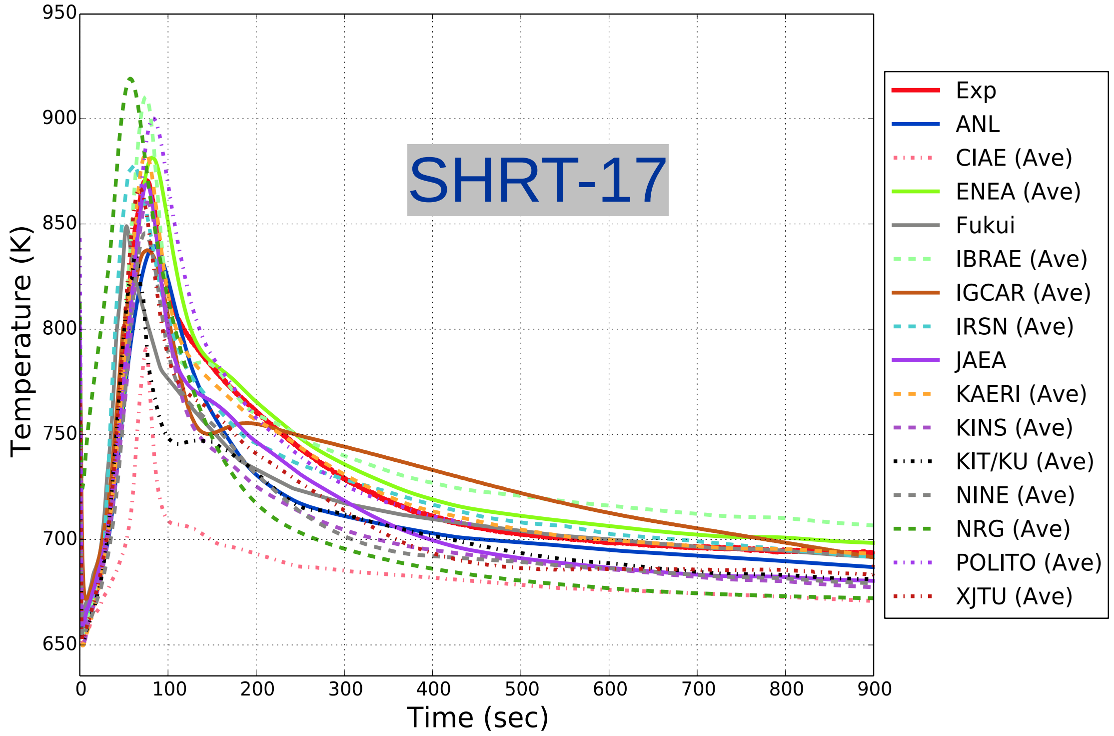


Benchmark Analysis of ***EBR-II*** Shutdown Heat Removal Test

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

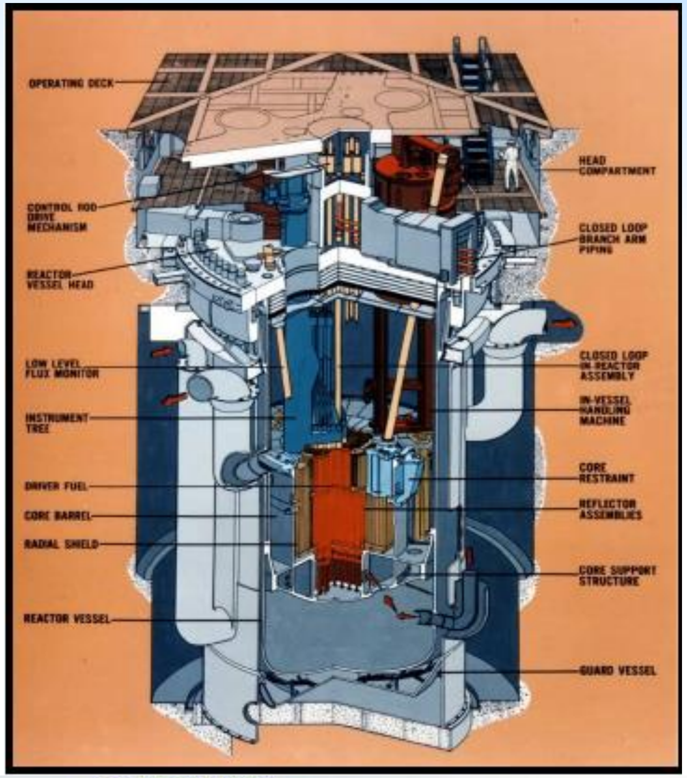






New IAEA CRP: **Benchmark Analysis of FFTF Loss of Flow Without Scram Test**

- 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)



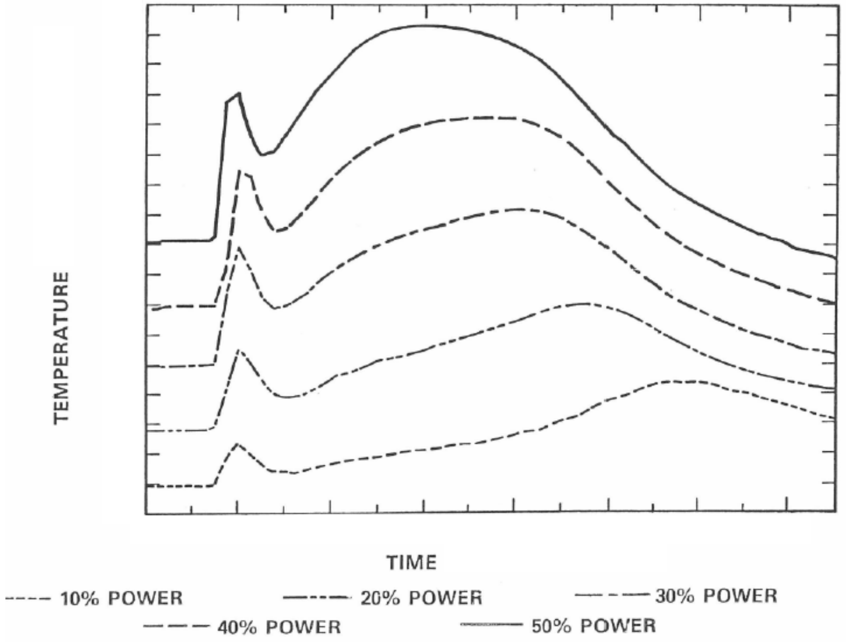
PNNL/ANL at Consultants' Meeting
November 2017, IAEA, Vienna

Benchmark Analysis of FFTF Loss of Flow Without Scram Test (ULOF)

ULOF to Natural Circulation Tests

Pacific Northwest NATIONAL LABORATORY
Proudly Operated by Battelle Since 1965

- With reactor at 50% power, main coolant pumps were turned off and normal control rod scram response was disabled
- GEMs and inherent core reactivity feedback mechanisms took the core subcritical with a modest peak coolant temperature transient
- Peak reached 85 °C above the pre-transient value, >400 °C below sodium boiling point



- Initial Flow Coast Down Causes First Sharp Peak
- Negative GEM Reactivity Feedback Causes Power to Drop
- Broad Peak Is Caused by Flow and Power Reaching Quasi Steady-State Values
- Subsequent Decline Is Caused by Reduction in Decay Heat

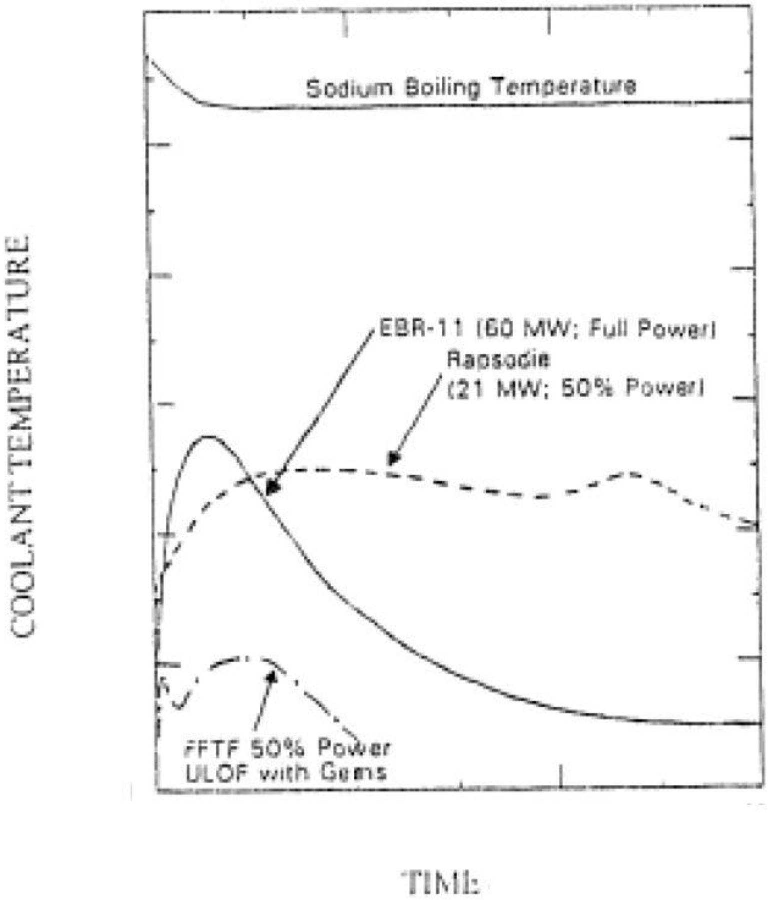
PNNL/ANL at Consultants' Meeting
November 2017, IAEA, Vienna

Benchmark Analysis of FFTF Loss of Flow Without Scram Test

Passive Safety Tests Demonstrated Advantages of LMRS for Surviving ULOF

Pacific Northwest
NATIONAL LABORATORY
Proudly Operated by Battelle Since 1965

- The result of loss of flow tests in Rapsodie, EBR-II, and FFTF was that the peak coolant temperatures were several hundred degrees below the sodium boiling point.
- While the driver fuel for the FFTF passive safety tests was oxide, the structural reactivity feedbacks are independent of fuel type.

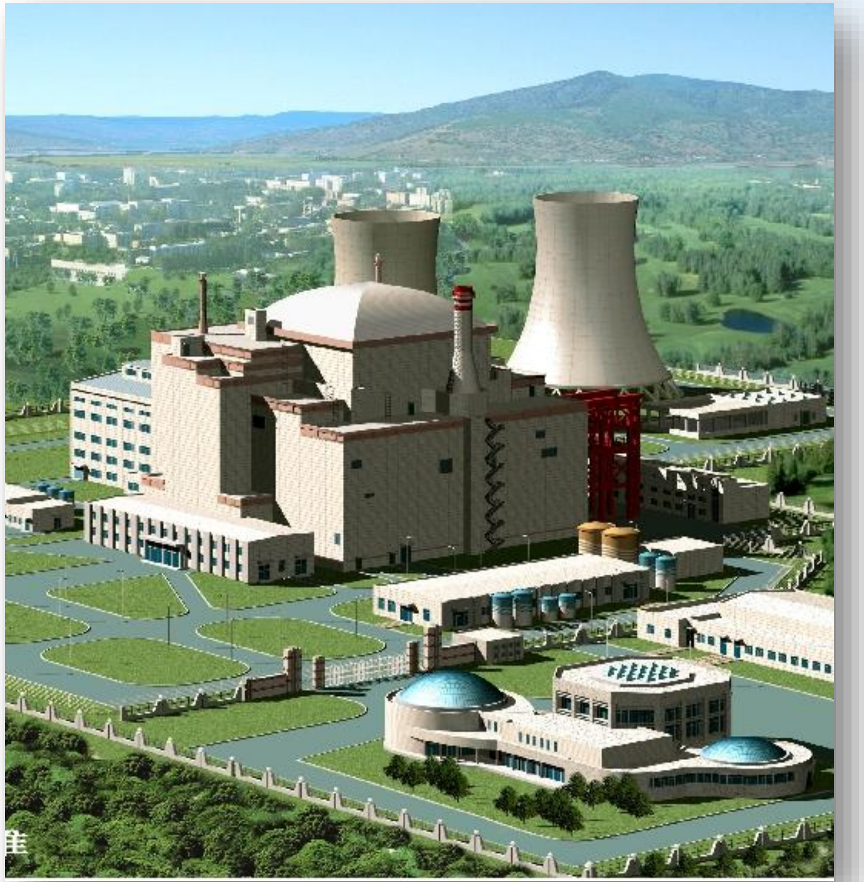


PNNL/ANL at Consultants' Meeting
November 2017, IAEA, Vienna

New CRP: **Neutronics Benchmark of CEFR Start-Up Tests**

中国原子能科学研究院
中核集团 CHINA INSTITUTE OF ATOMIC ENERGY

- China Experimental Fast Reactor
  - Sodium-cooled fast reactor with nominal power of 65MW(th), 20MW(e)
  - Reached the first criticality in 2010
  - Generated electricity at 40% full power and was connected firstly to the grid in July 2011
  - Generated electricity at 100% power in December 2015 and operated for more than 40 effective full power days



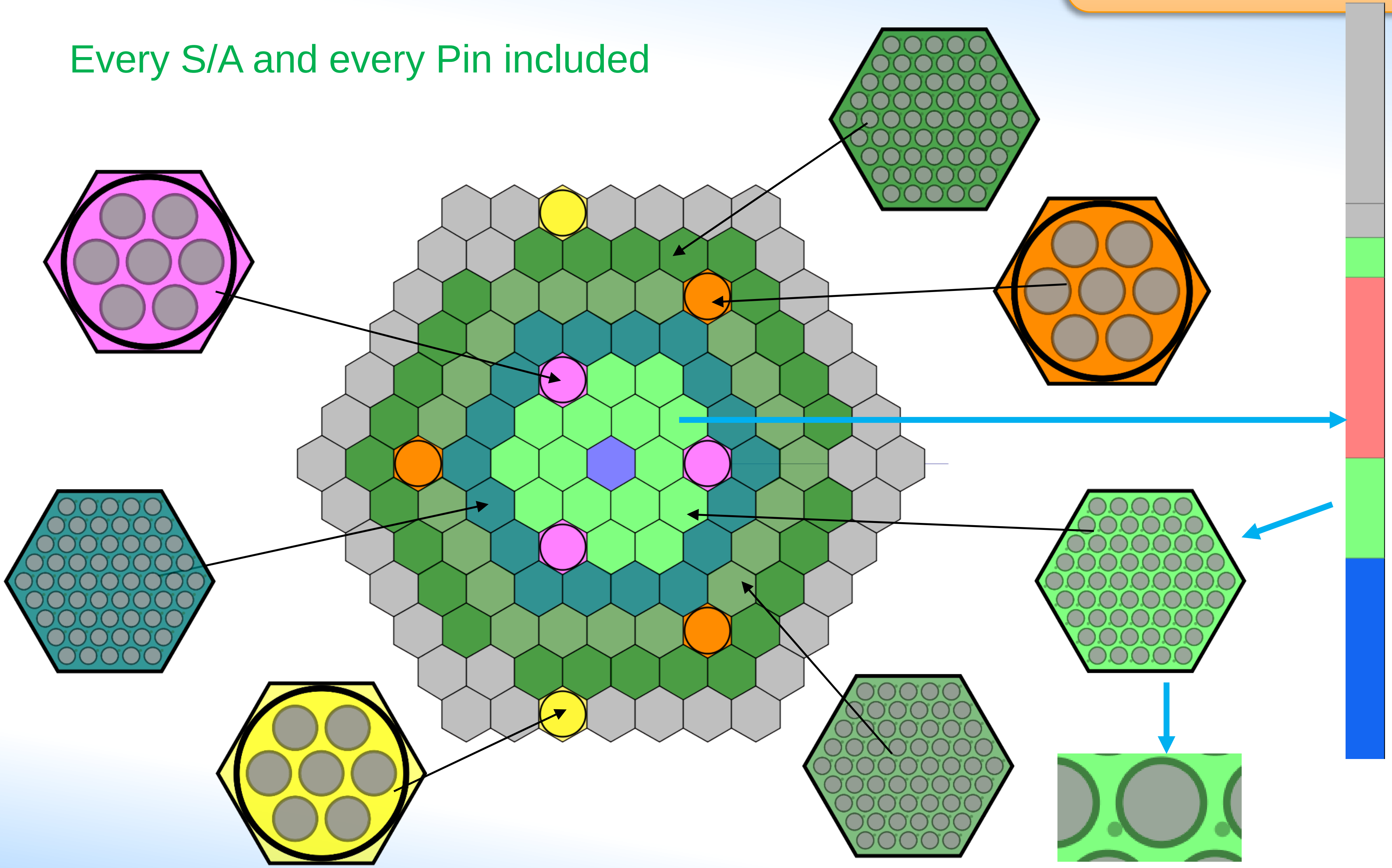
CIAE at Consultants' Meeting November 2017, IAEA, Vienna

Digital Nuclear Reactor: Core Map

CEFR Start-Up Benchmark

IAEA

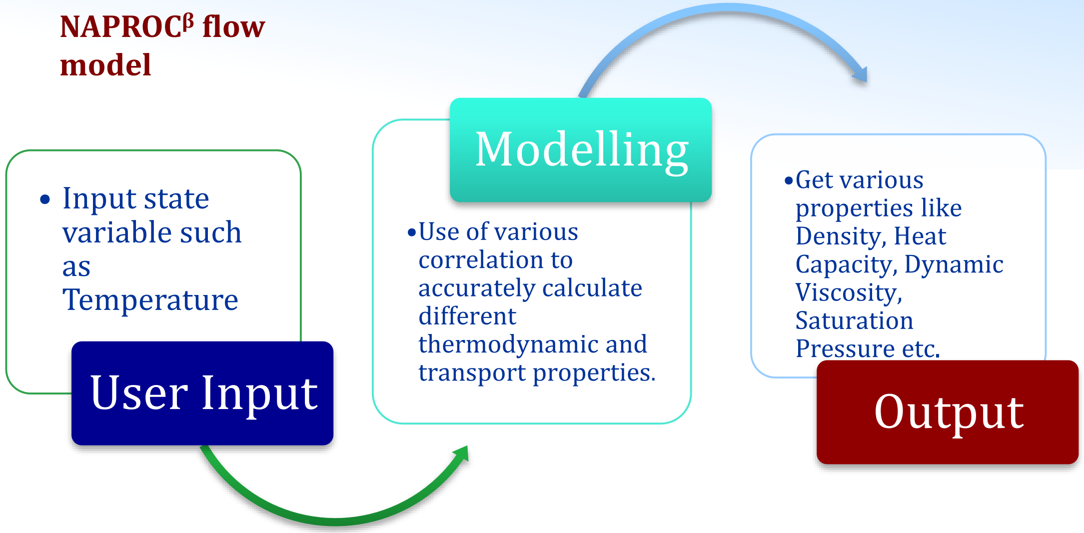
Every S/A and every Pin included



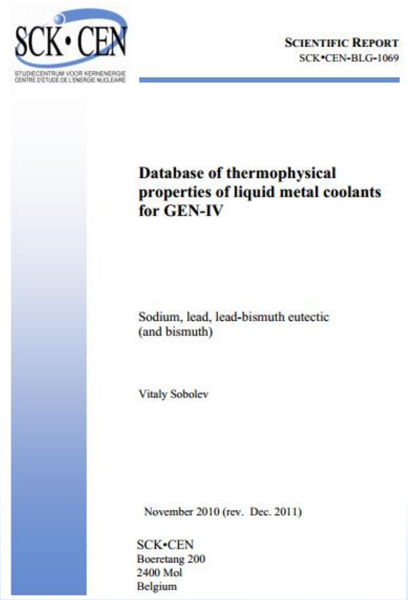
NAPROC $^{\beta}$: Sodium Properties Calculator

- 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

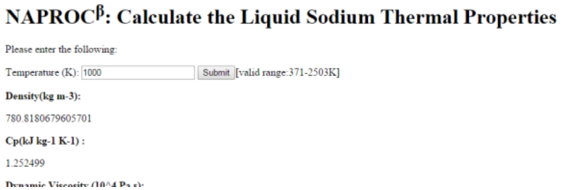
Sodium, lead, lead-bismuth eutectic (and bismuth)

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:

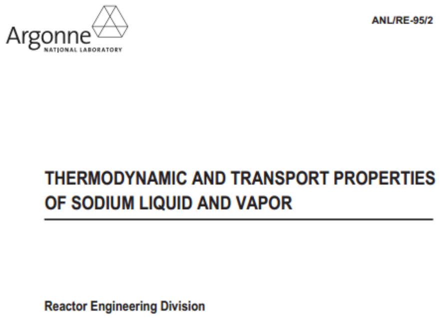
Temperature (K): 1000 Submit [valid range:371-2503K]

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

Thank You!