Safety of fast reactors: phenomenology and modeling aspects

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Outline

- Horizon-2020 ESFR-SMART project
- Safety approach developed by GIF for all GIF systems
- Selected safety-related properties of SFR and LFR
- Characterisation of initiating events
- Design basis conditions example: Superphenix start-up (DBC-1)
- Design basis conditions example: protected loss of flow at Phenix (DBC-4)
- Design extension conditions
- LMFR calculational analysis: few examples of codes
- Summary: few examples of knowledge gaps in SFR safety analysis

ESFR-SMART

European Sodium Fast Reactor Safety Measures Assessment and Research Tools

proposal submitted to H2020 framework program; budget ~10 MEUR (5 MEUR from EU)

Coordinator-- Dr. K. Mikityuk (PSI)

PSI /CH	CEA /FR	CIEMAT /SP	CHALMERS /SW	EDF /FR	ENEA /IT	
Framatome/FR	GRS /DE	HZDR /DE	IPUL /LV	IRSN /FR	JRC /EU	KIT /DE
LEMTA /FR	LGI /BE	NNL /UK	UCAM /UK	UPM /SP	WOOD /UK	

Use legacy experiments

-- SFR operational data (SPX1/CEA)
-- sodium boiling (KNS-37/KIT)
-- molten fuel ejection (CABRI/IRSN)
-- molten pool behaviour (SCARABEE/IRSN)
-- aerosols in containment (FAUST/KIT)
-- aerosols in containment (NALA/KIT)
-- aerosols from sodium fire (FANAL/CEA)

Calibrate and validate codes

Conduct new experiments

-- MOX fuel properties (CEA → ITU)
-- forced-to-natural convection (KASOLA/KIT)
-- sodium boiling (SOLTEC/KIT)
-- chugging boiling (CHUG/PSI)
-- corium jet/catcher (JOLO/LEMTA)
-- corium/catcher (LIVE/KIT)
-- corium jet/concrete (MOCKA/KIT)

Assess new safety measures for ESFR

-- low void effect core design
-- corium discharge tubes
-- passive decay heat removal
-- large-inertia and passive pumps
-- improved natural circulation, *etc*

Develop new instrumentations

-- eddy current flowmeter (HZDR)

Establish new networks

-- students mobility grants to work at EU Na facilities
-- workshops and summer school

Demonstrate new reactor concept features

-- iso-breeder (produces fissile fuel for own needs)
-- safer than LWRs (no core meltdown in Fukushima-like accident)
-- safer than SFRs (low void effect)

Safety approach developed by Generation IV International Forum (GIF) for all GIF systems

Specific safety goals for Generation IV systems

Generation IV safety philosophy

- Safety improvement
- Defence in depth
- Safety functions
- Modeling and simulation

Specific safety goals for Gen IV systems

Generation IV nuclear energy systems

1. will excel in safety and reliability
2. will have a very low likelihood and degree of reactor core damage
3. will eliminate the need for offsite emergency response

The basic safety functions should be fulfilled in every reactor state:

- control of reactivity
- removal of heat from the fuel
- confinement of radioactive materials

Generation IV safety philosophy

- Opportunities exist to further **improve safety** of Generation IV systems, which is already very good or excellent for Generation II and III systems

- The principle of "**defence in depth**" must be preserved in the design of Generation IV systems

- The Generation IV design process should be driven by a **"risk-informed" approach**, i.e. using both deterministic and probabilistic methods

- For Generation IV systems, in addition to prototyping and demonstration, **modelling and simulation** should play a large role in the design and the assessment.

Safety improvement

Safety level achieved for Generation II and III systems is very good or excellent and can be kept as a **reference** for future reactors.

Further safety improvement for Generation IV systems are possible through

- **progress** in knowledge and technologies and
- application of a consistent safety philosophy **early** in the design process.

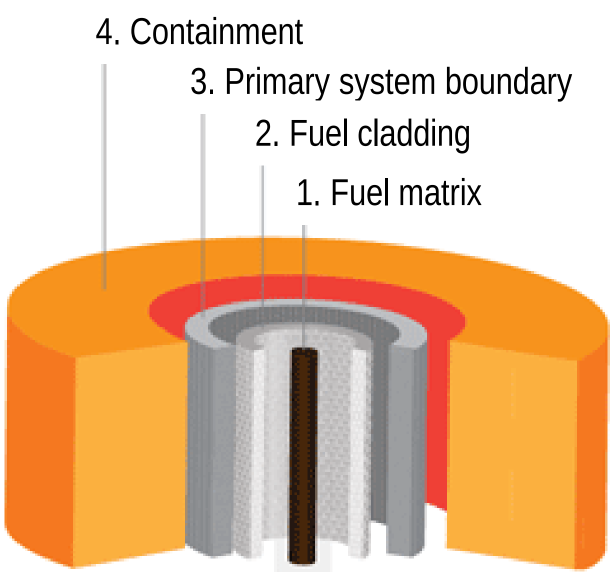
In Generation IV systems safety will be "**built-in**" to the fundamental design rather than "**added on**" to the system architecture.

Defence in depth: physical barriers

The central feature of defence in depth is the idea of multiple barriers and levels of protection between radioactive materials and the environment.

https://www.secureworldexpo.com/ditch-castle-moat

Barriers preventing radiation leak into the environment



https://www.seas.sk/safety-barriers

Defence in depth: levels of protection

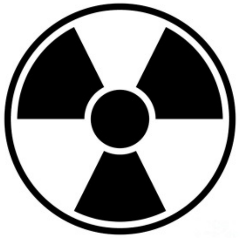
Mitigate off-site releases: on-site and off-site emergency plans to mitigate radioactivity release ⑤

Control severe accidents: measures to preserve the containment ④

Prevent severe accidents: safety systems for fulfilling three basic safety functions ③

Control abnormal operations: measures to detect failures and preserve two physical barriers ②

Prevent abnormal operation: conservative design, quality assurance, inspection activities, safety culture ①



Defence in depth: map of reactor states

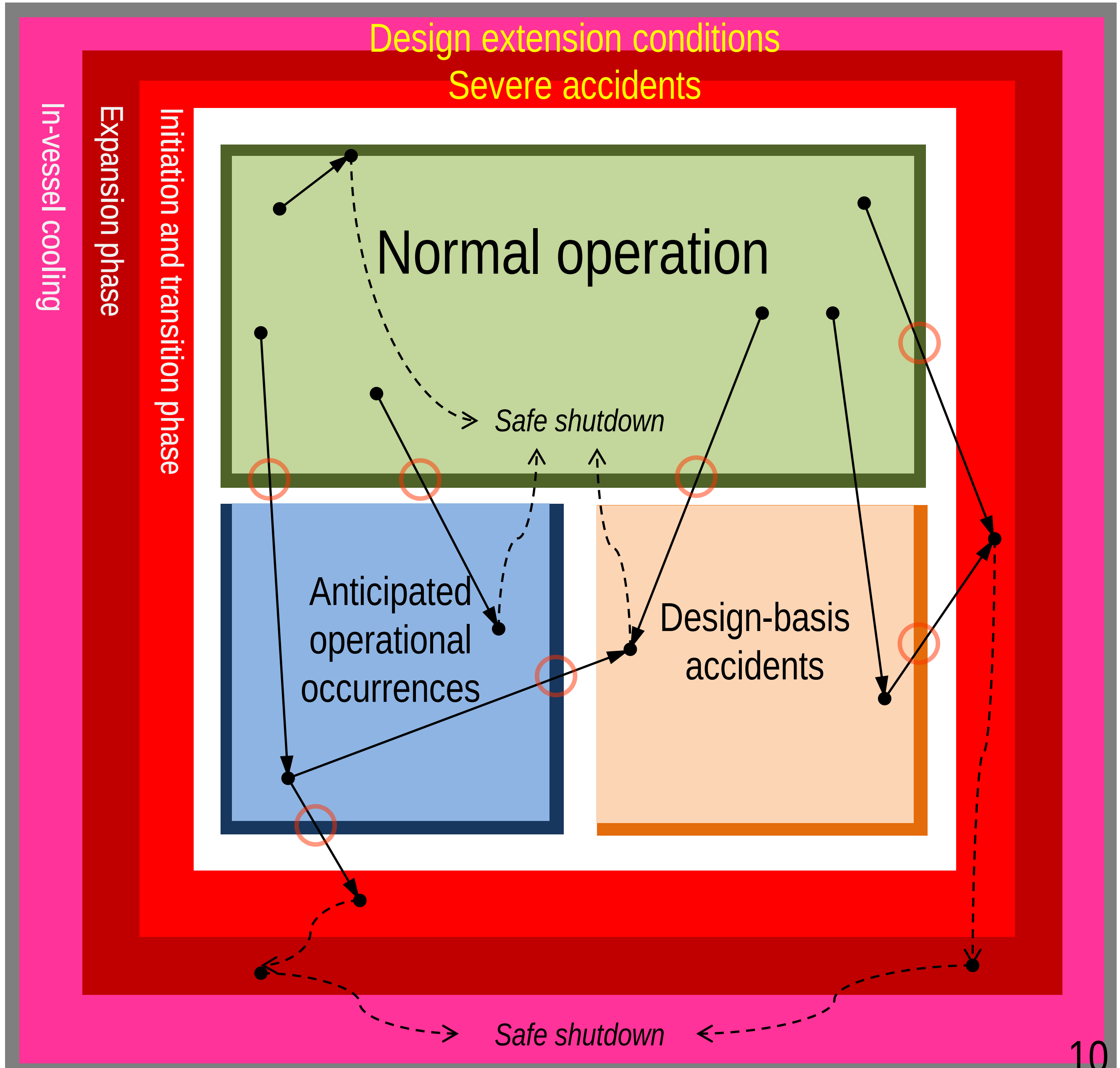
Defence in depth levels:

1 2 3 4

Initiating event

Safety measure

Failures



Safety functions for Gen IV systems

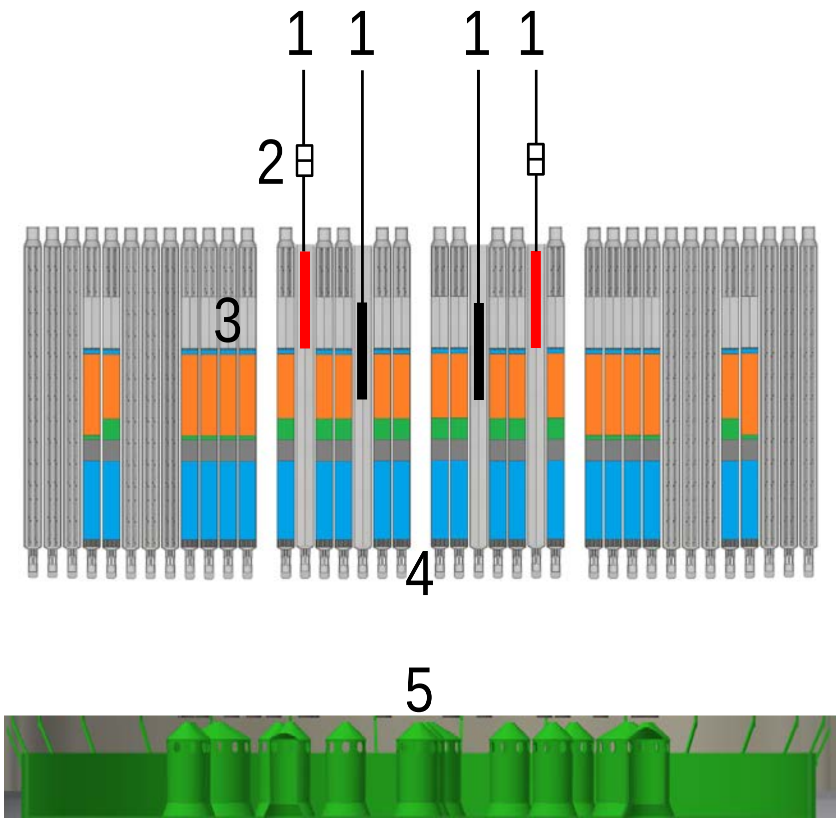
The basic safety functions should be fulfilled in every reactor state:

- control of reactivity
- removal of heat from the fuel
- confinement of radioactive materials

Control of reactivity: example of ESFR

In case of accident:

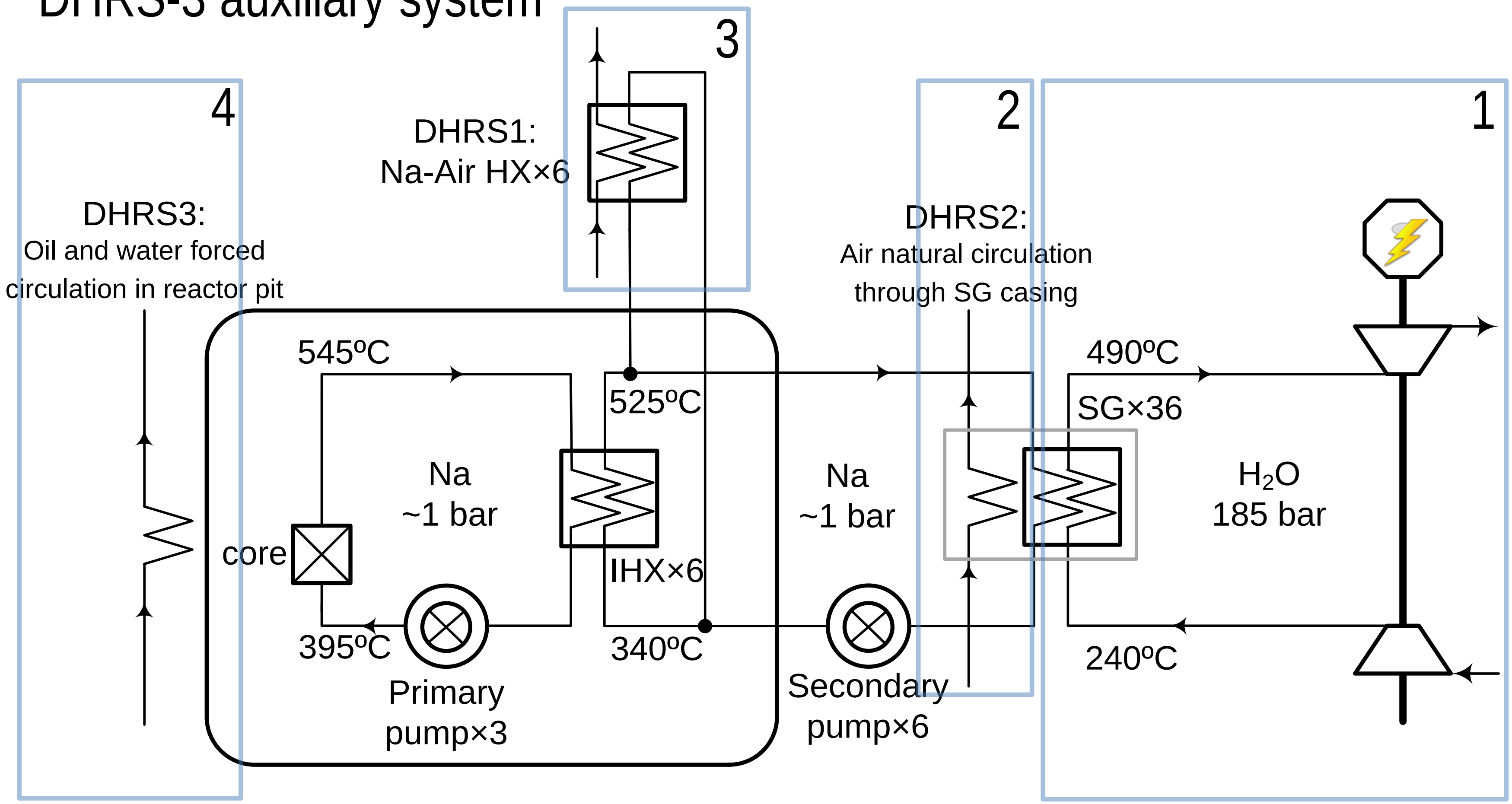
1. Scram activation by one of the signals or by operator
2. Curie-point devices on safety control rod drivelines for passive scram at temperature increase
3. Sodium plenum to avoid power run-away
4. Corium discharge tubes to avoid re-criticality in the core
5. Core catcher designed to guarantee subcriticality of corium



Removal of heat from the fuel: example of ESFR

In case of reactor shutdown:

1. Decay heat removal (DHR) through nominal cooling path
2. If feedwater is lost, DHRS-2 removes DH from SG surfaces
3. If secondary system is lost, DHRS-1
4. DHRS-3 auxiliary system



Confinement of radioactive materials: example of ESFR

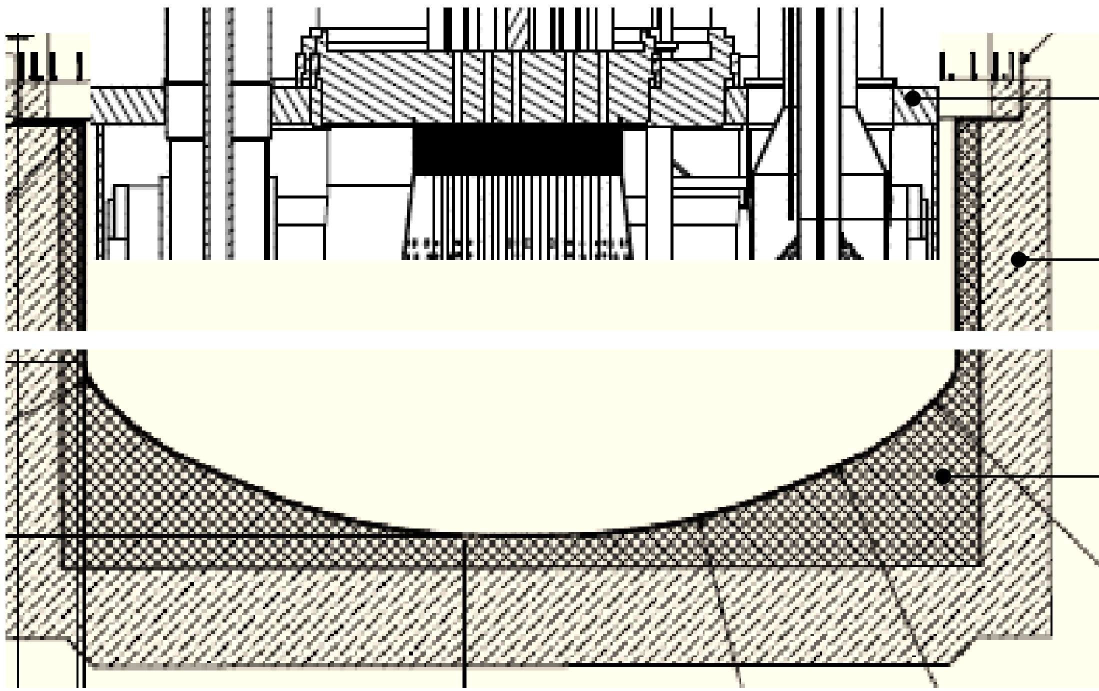
Optimization of confinement measures and economy

1. Massive steel reactor roof of 80 cm thick

— no water inside (no cooling and no sodium deposit in lower part)
— good radiation protection; no reactor dome
— minimisation of penetrations (components are welded on perimeter)
— solid rotating plugs with eutectic freezing seal

2. Reactor pit

— no safety vessel but metallic liner on isolation surface



Modeling and simulation

Prototyping and demonstration should be complemented by modelling and simulation while designing and assessing Generation IV systems.

- Prototyping and demonstration systems are expensive and contribute to the long lead time associated with the development of new technologies.

- Modelling and simulation can provide more thoroughly evaluations of a candidate design therefore reducing uncertainties and improving safety.

Selected safety-related properties of Liquid-Metal Fast Reactors (SFR and LFR): commonalities and differences

Safety related coolant properties

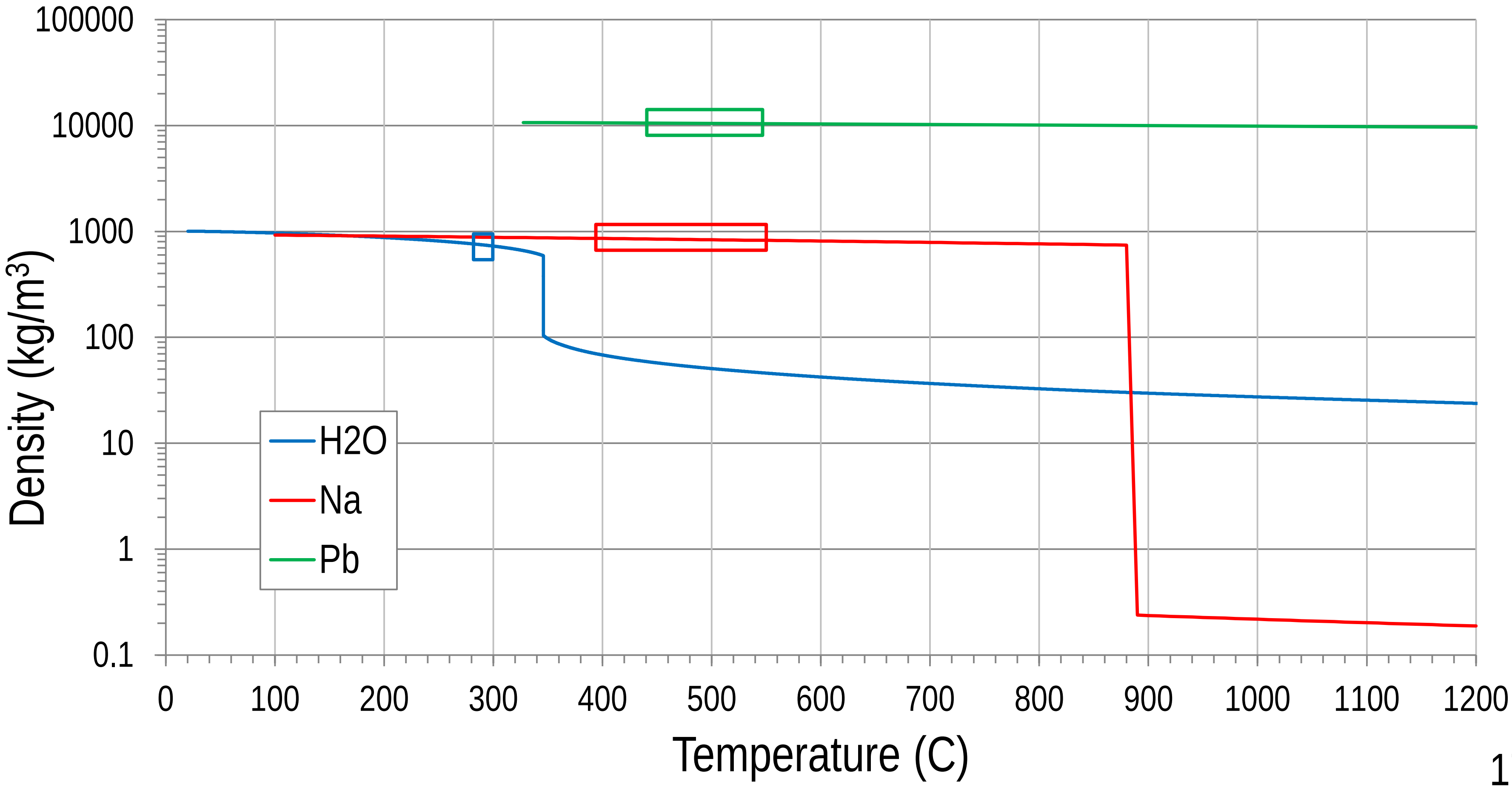
- Density
- Chemical activity
- Boiling
- Freezing

Main reactivity effects

- Doppler effect
- Coolant density / void effect
- Thermal expansion effects: fuel, clad, diagrid, strongback, vessel control rod drivelines

Sodium, lead and water density

- Margins to boiling for Na and Pb >> for H2O → increased importance of thermal expansion of core and reactor structures for reactivity
- Margins to freezing for Pb << for Na and H2O → special safety measures for LFR needed



Density

Lead has a very high density:

- powerful and reliable pumps are necessary;
- high requirements should be met to seismic stability of the facility;
- reactor vessel and support structures should have high strength;
- special measures should be envisaged to eliminate flowing up of fuel assemblies caused by high coolant buoyant force;
- probability of secondary critical mass formation after core degradation is low because coolant density is close to or higher than fuel density and coolant flow can distribute fuel fragments over primary circuit;
- probability of vapor or gas entrainment in the core is low due to high coolant buoyant force.

Chemical activity

Sodium exothermically reacts with air and water:

- danger of fire and explosion;
- danger of loss of coolant caused by coolant burning out;
- complication of reactor design to avoid contact with air and water: *e.g.* intermediate circuit, double-wall steam generator.

Lead slowly reacts with structural materials by dissolving and eroding them:

- special measures are required to protect claddings: *e.g.* on-line control of oxygen concentration in a narrow range to maintain protection oxide films on cladding surfaces and simultaneously to prevent precipitation of solid oxides in cold regions of primary system;

- dissolving of core structural materials in the coolant and their removal from the core can result in positive reactivity effect;
- coolant temperatures and velocities are limited by erosion.

Boiling

Sodium and lead have high boiling points:

- high temperatures (high efficiency) provided at low primary pressure, enhancing reactor safety and reliability, simplifying reactor design and facilitating fuel rod operation;
- high margin to boiling makes changes of the geometry due to thermal expansion very important for the core reactivity.

Lead boiling point is higher than stainless steel melting point

- danger of positive reactivity insertion due to melting or dissolving and removal of structural materials from the core.

Sodium boiling

- danger of cladding overheating by dryout;

- danger of positive reactivity insertion due to sodium boiling;

- danger of pressure shocks (mechanical and reactivity impact) caused by sodium boiling (collapse of bubbles).

Freezing

Lead has a (relatively) low margin to freezing (~130ºC)

- danger of coolant freezing during startup, repair and maintenance, shutdown, transients, requiring special measures, *e.g.* electrical heaters;
- difficulties of ISI at relatively high temperature (above freezing point);
- loss of coolant after circuit break is limited because of fast coolant freezing and possible self-healing of the break;
- rapid freezing of coolant eliminates deep penetration of radioactive coolant in the environment after accident with primary circuit break.

Doppler reactivity effect

Doppler effect is an apparent broadening of the normally narrow resonance peaks due to thermal motion of nuclei → reduction of the self-shielding effect.

Driven by the fuel temperature, an "instant" negative reactivity feedback of high importance for safety.

When fuel temperature rises, U-238 resonances broadened due to increased thermal agitation of nuclei. As a result, U-238 resonance capture rate ↑ and reactivity ↓

In fast spectrum: $d\rho_{Dop} = K_D \frac{dT_f}{T_f} \rightarrow \Delta\rho_{Dop} = K_D ln \frac{T_f}{T_{f0}}$

where K_D – Doppler constant

In thermal spectrum: $d\rho_{Dop} = \alpha_D dT_f \rightarrow \Delta\rho_{Dop} = \alpha_D (T_f - T_{f0})$

where α_D – Doppler coefficient

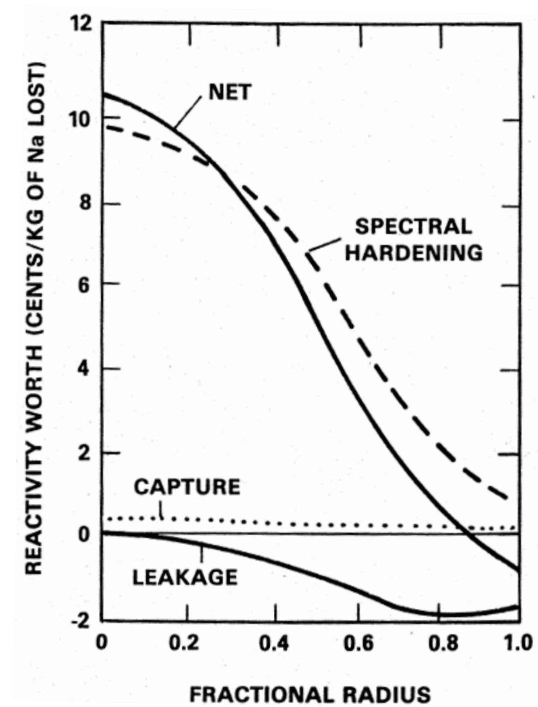
Change of stainless steel cladding temperature also contributes to Doppler effect!

Coolant density / void reactivity effects

Coolant density effect in fast reactors is a quick reactivity feedback that is a big challenge for safety (not present in thermal reactor)

Coolant density effect (reduction of density) or void effect (complete removal of coolant) leads to

- decrease of capture by coolant (small)
- increase of leakage
- hardening of spectrum → increase of production rate in high energy region

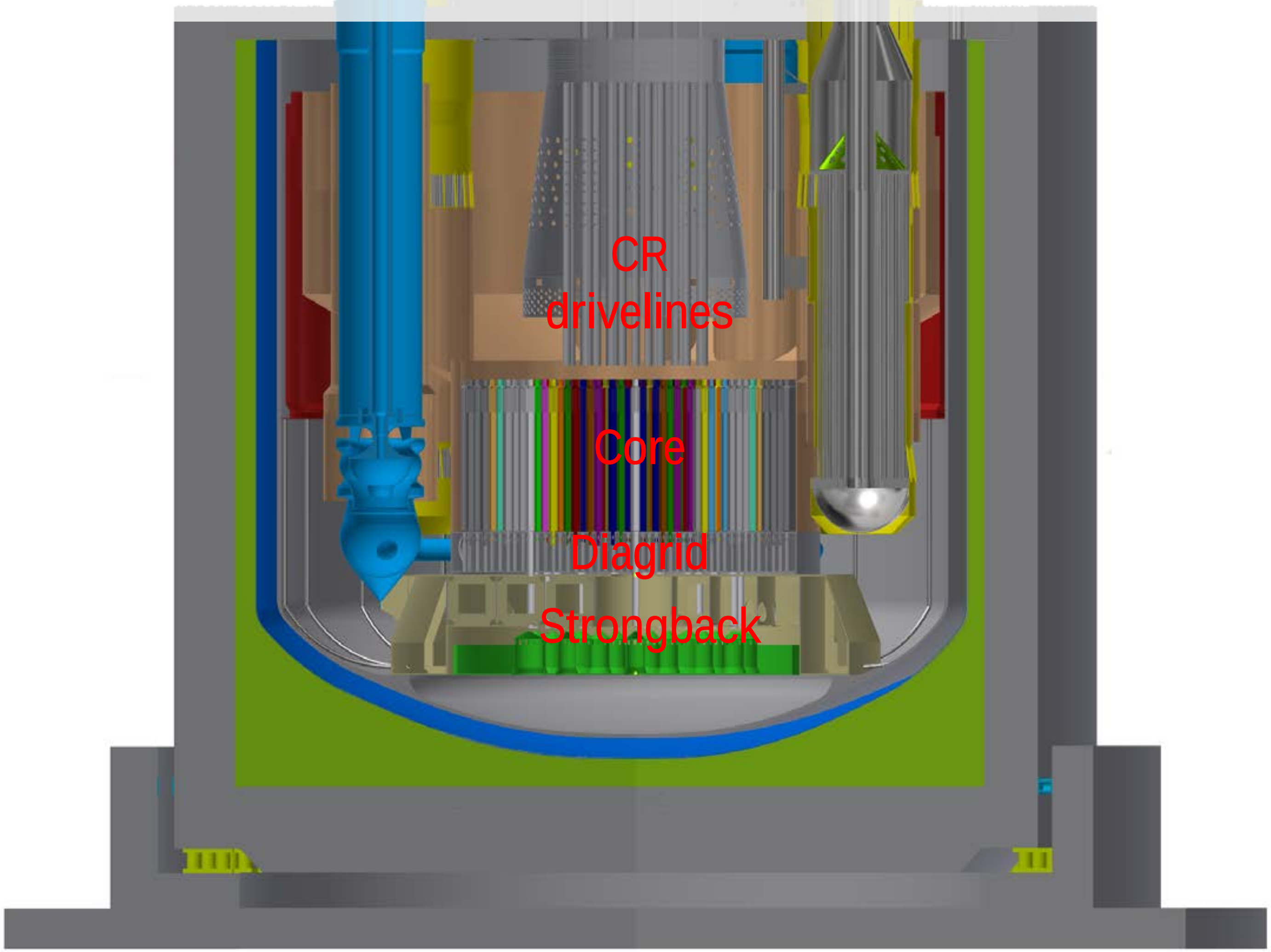


The effect is strongly 3D:

- positive in the core center (spectral hardening dominating)
- negative at core periphery & reflector (leakage dominating)

Thermal expansion reactivity effects

Differential expansion of fuel, cladding (wrapper), sodium and absorber driven by thermal expansion of different core and reactor components driven by different temperatures...



Fuel thermal expansion reactivity effect

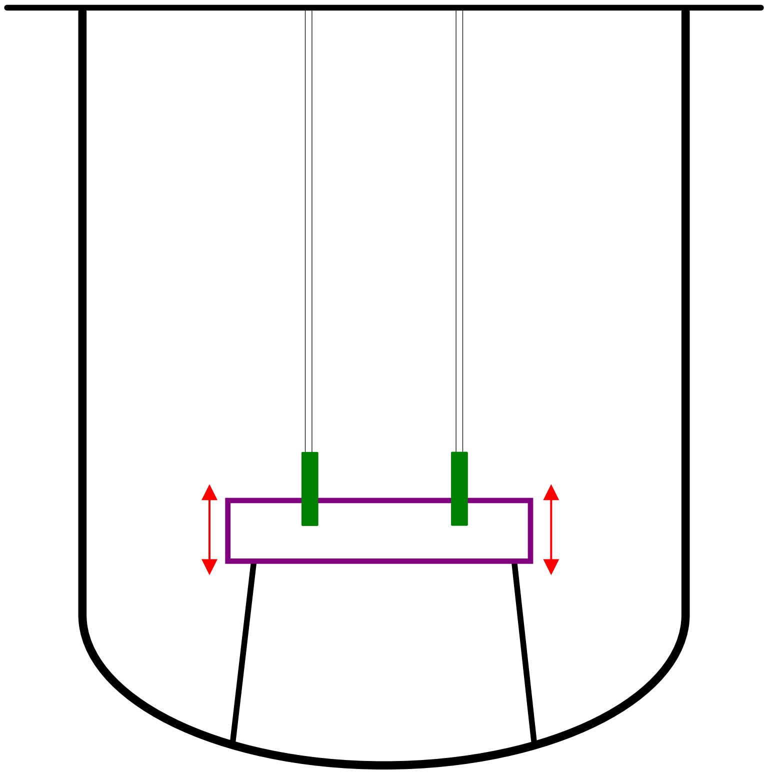
Heating up and axial thermal expansion of fuel

- more parasitic absorption by cladding
- more scattering by sodium
- slight insertion of control rods

Axial core expansion is driven by

- expansion of fuel when fuel-clad gap is open
- expansion of fuel-clad system when fuel-clad gap is closed (comprehensive thermal-mechanical analysis needed)

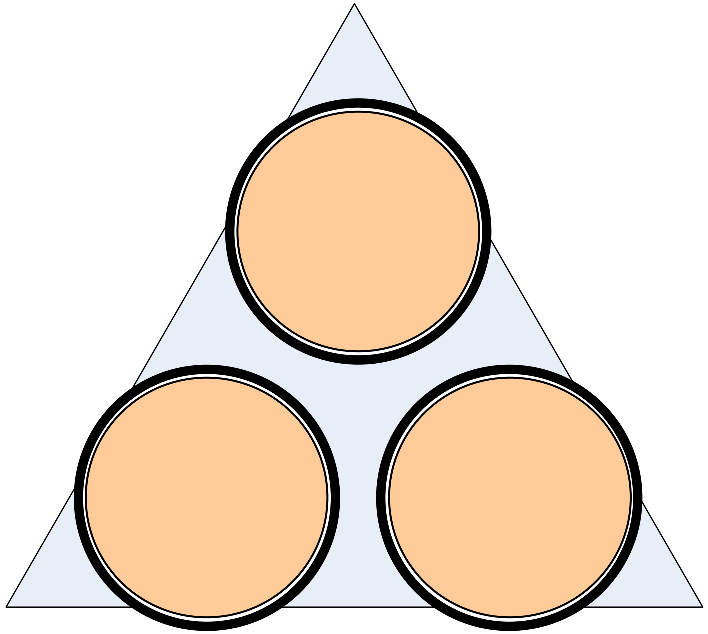
“Typical” values ~ –0.6 pcm/C

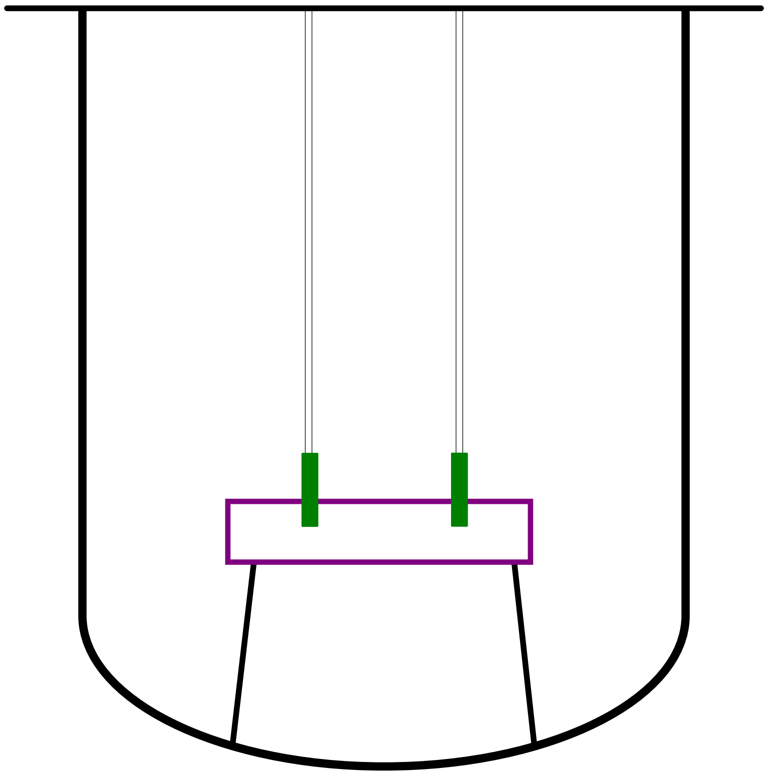


Clad thermal expansion reactivity effect

Heating up and axial + radial thermal expansion of cladding (and wrapper):

- axial: less parasitic absorption by cladding
- radial: less scattering by sodium (pushed out)



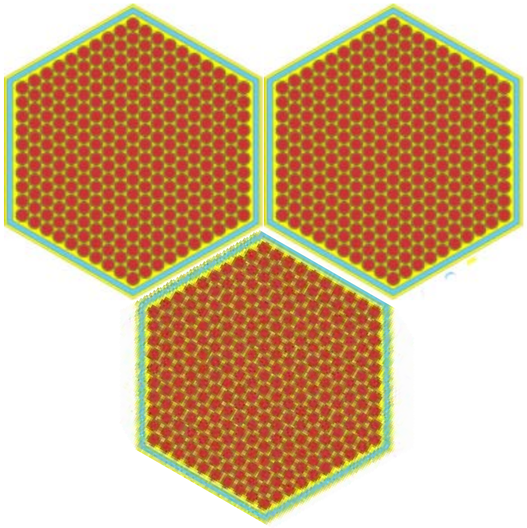


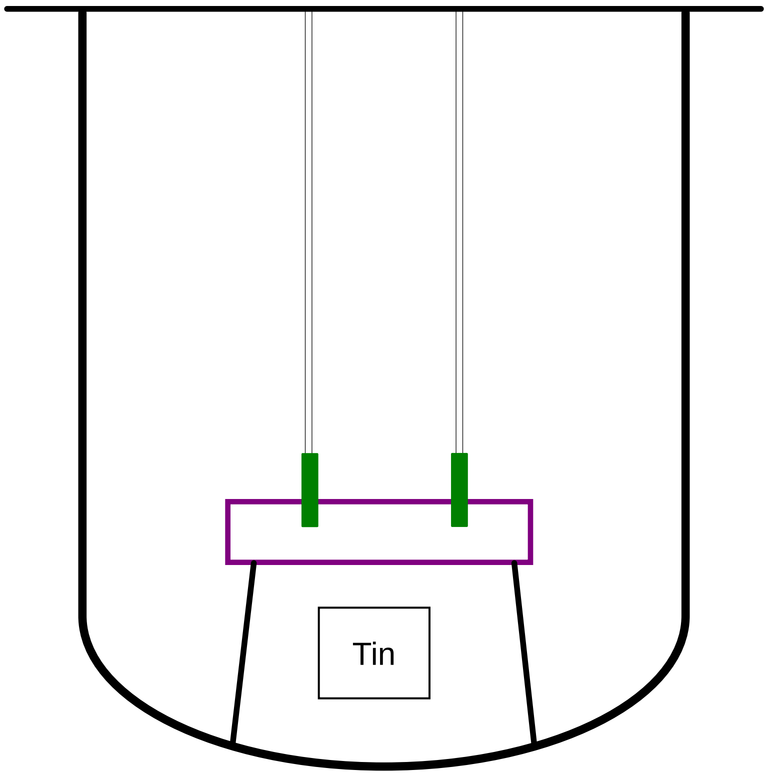
Diagrid thermal expansion reactivity effect

Heating up and radial expansion of the massive support structure:

- higher axial leakage
- more scattering by sodium

“Typical” values ~ –1 pcm/C





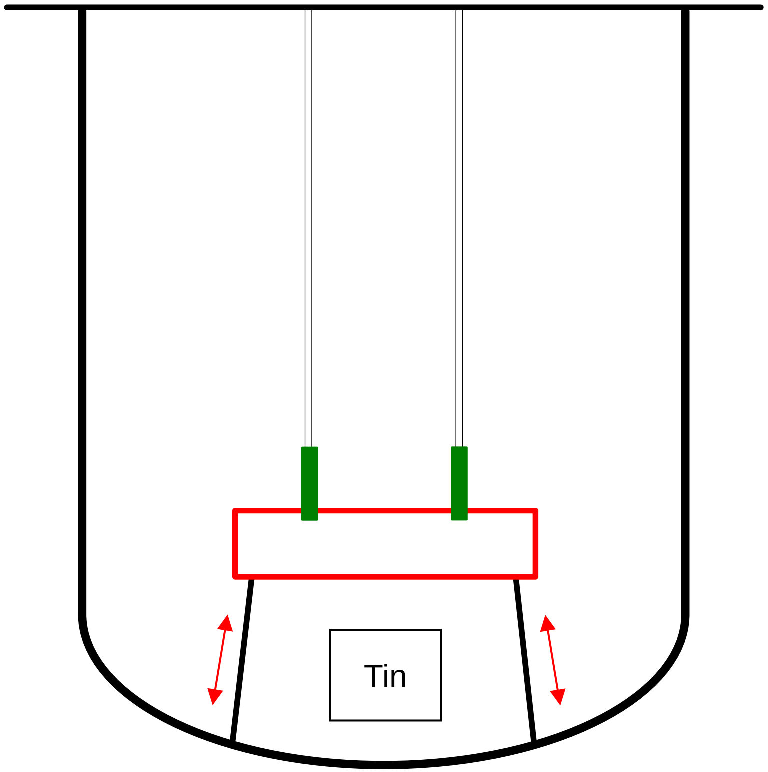
Strongback thermal expansion effect

Heating up and axial expansion of the massive support structure:

- slight insertion of control rods

Delayed with respect to diagrid expansion

“Typical” values ~ –2 pcm/C



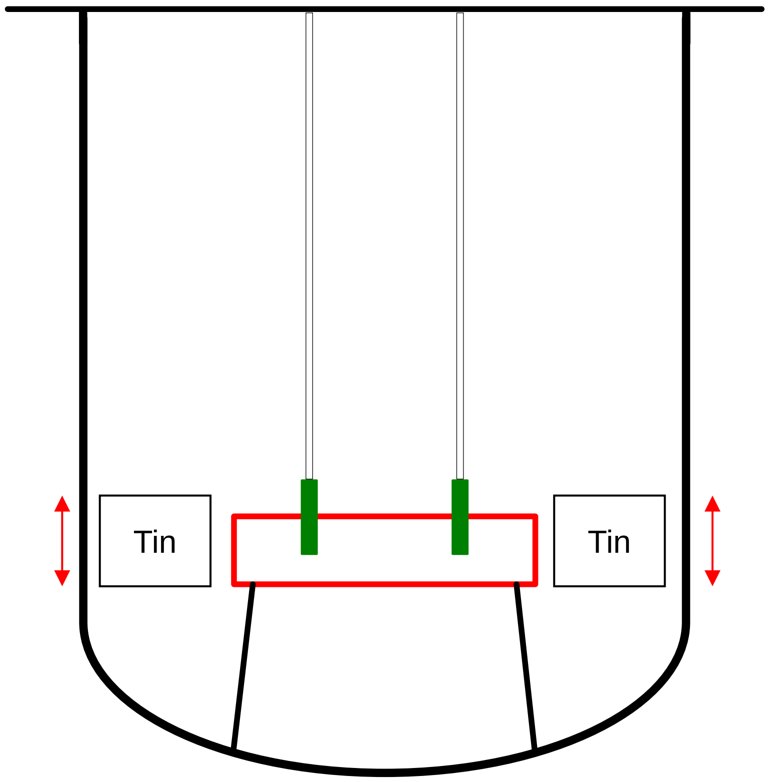
Vessel thermal expansion reactivity effect

Heating up and axial expansion of the massive vessel:

- slight withdrawal of control rods

Delayed with respect to diagrid and strongback expansions

“Typical” values ~ +4 pcm/C



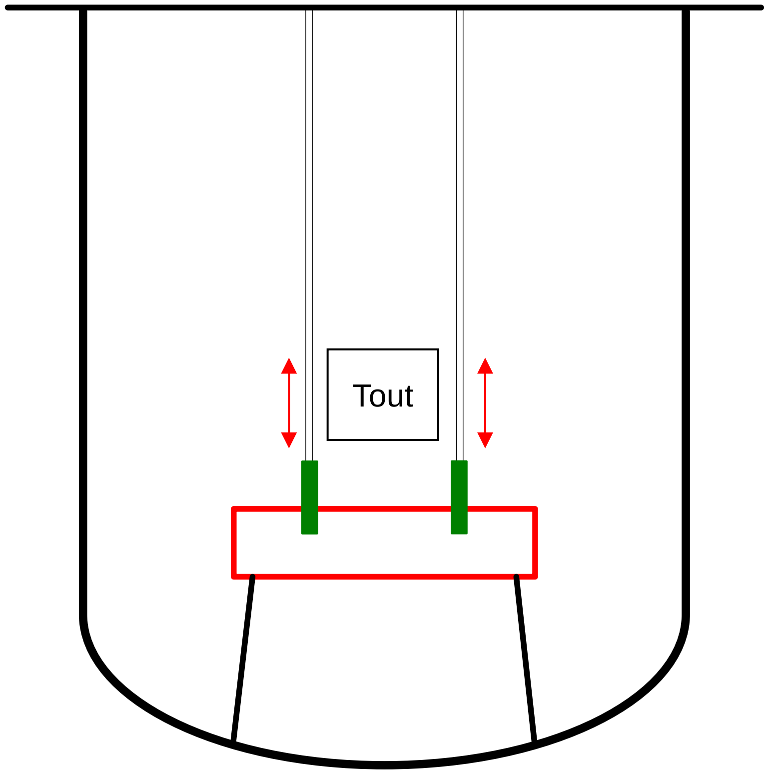
Control rod driveline expansion effect

Heating up and axial expansion of the control rod drivelines

- slight insertion of control rods

Delayed with respect to Doppler and coolant expansions

“Typical” values ~ –1 pcm/C

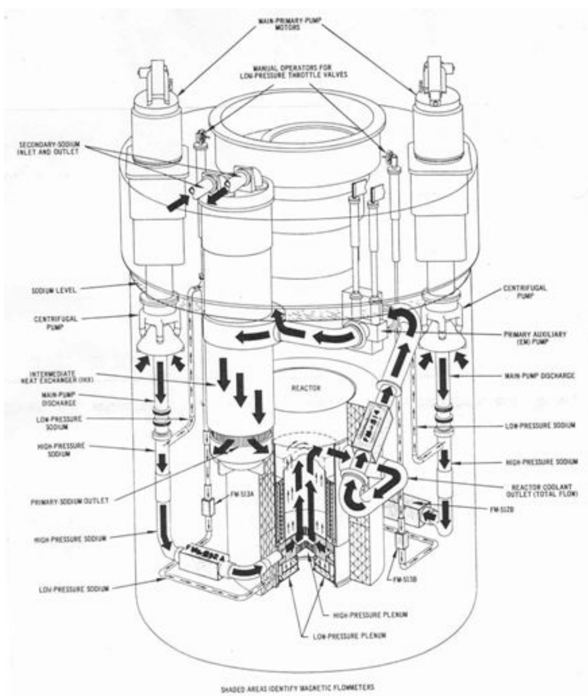


Thermal expansion reactivity effects

Differential expansion of fuel, cladding (wrapper), sodium and absorber during the transient before the boiling onset is an important reactivity feedback.

Understanding and using this feedback, the designer has an opportunity to create a self-protected system with the safety “**built-in**” to the design.

Demonstrated by SHRT-45R Unprotected Loss of Flow experiment at the **EBR-II** reactor



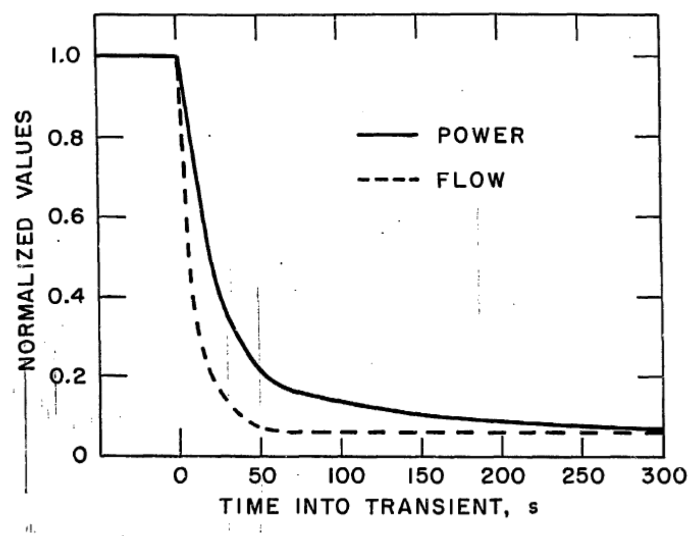


Fig. 2. Reactor Power and Flow Response during Transient

Characterisation of initiating events

Design basis conditions

DBC1: normal operating conditions, e.g.

- power operation,
- normal transients (start-up, shutdown, load following…),
- commissioning, ...

DBC2: incidents or Anticipated Operational Occurrences (10^{-2}), e.g.

- Reactivity insertion as a runaway of grouped control rods,
- Coastdown of all secondary pumps,
- Loss of feedwater on all SGs, ...

DBC3: accidents (10^{-4} – 10^{-2}), e.g.

- Coastdown of all primary pumps,
- Loss of Offsite power, ...

DBC4: accidents (... – 10^{-4})

- LIPOSO failure

Design extension conditions

1: Events for which the objective is to check that they do not deteriorate into a whole core accident

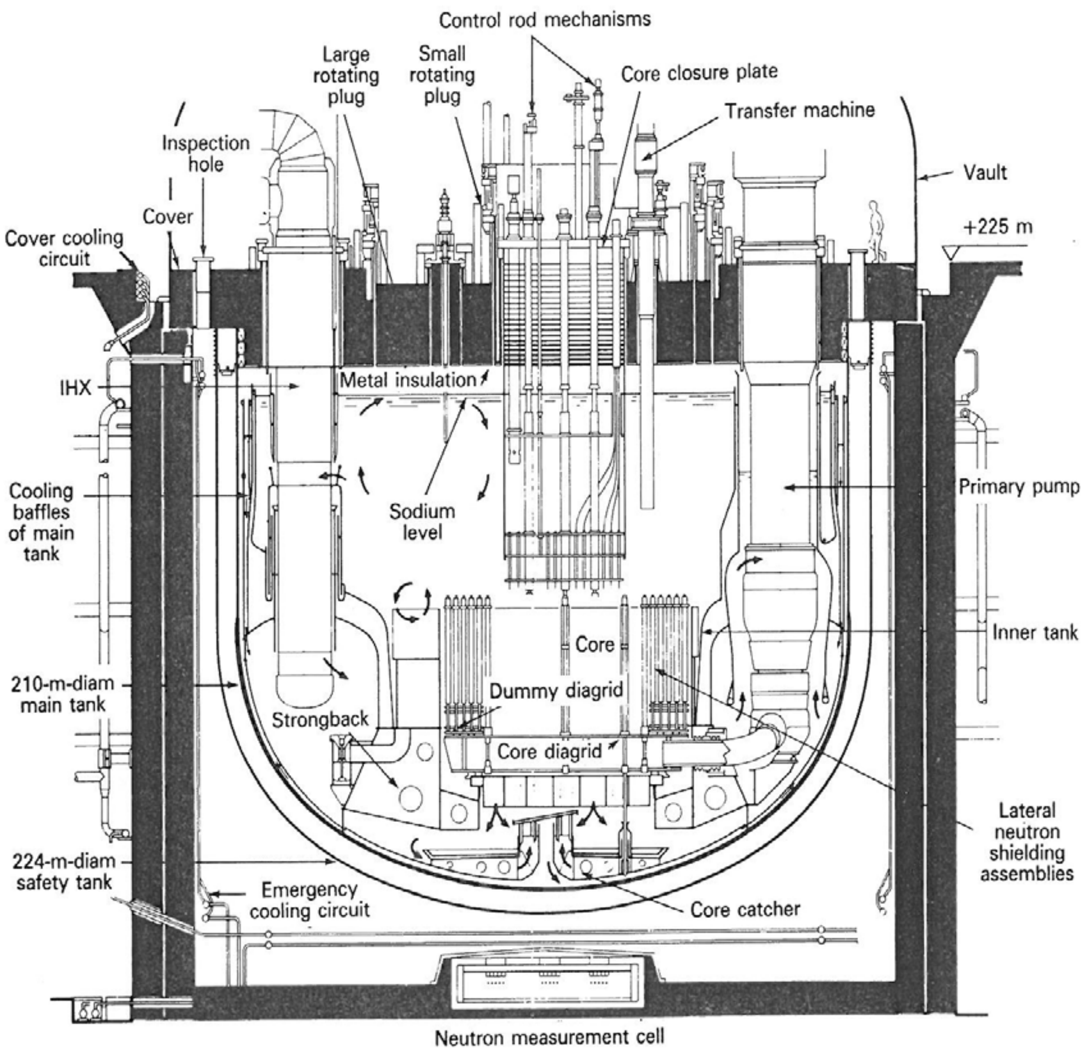
— complex sequences,
— limiting event, ...

2: Situations corresponding to whole core accidents, which are not practically eliminated

3: Practically eliminated situations

Design basis conditions example: Superphenix start-up (DBC-1)

Superphenix: large-power SFR



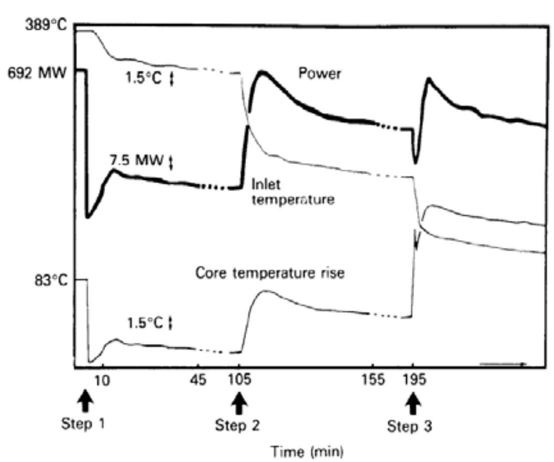
NSE 106 (1990)

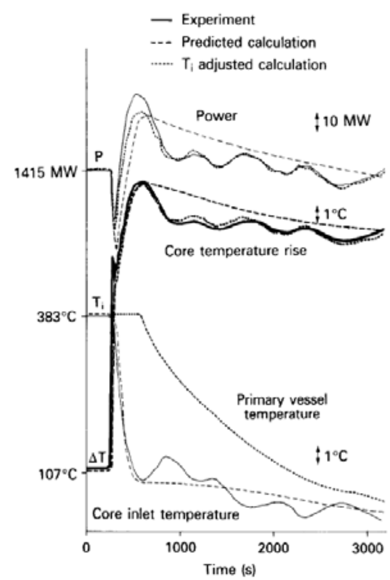
Power, MWt	3000
Flowrate, t/s	16.4
Inlet/outlet temp, C	395 / 545
Vessel height × diameter, m	22 × 21
Primary sodium mass, t	3200
Fissile core diameter, m	3.71
Fissile core height, m	1.0
Fissile core volume, m^3	10.82
Fuel pellet ID, mm	2.0
Fuel pin OD, mm	8.5
Number of fissile/fertile SAs	364/233
Fissile/fertile	MOX/UO2
Number of pins per SA	271

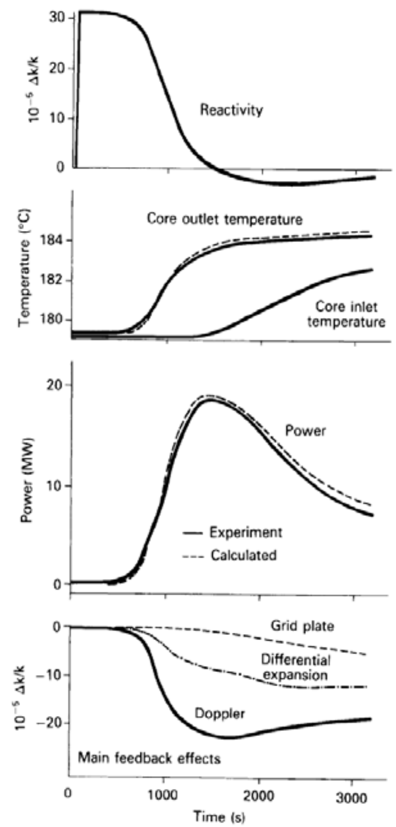
SPX start-up tests

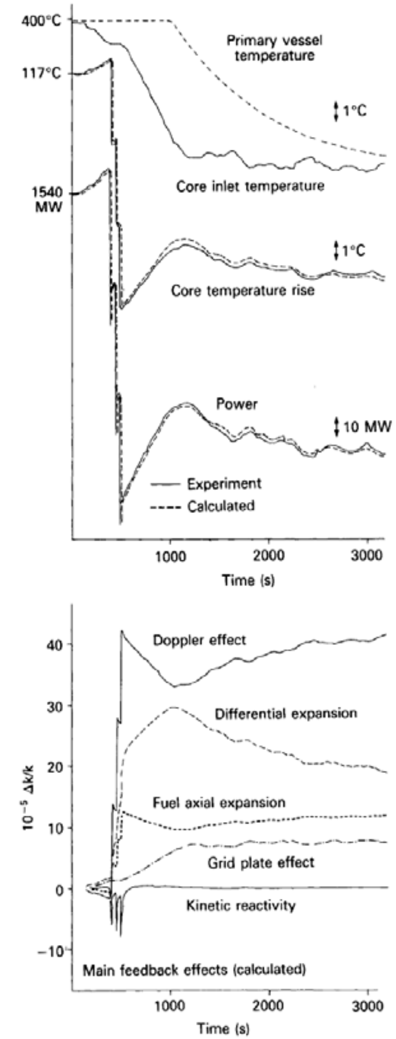
The measurements for a number of transients performed at SPX during the start up phase was published in NSE 106 (1990)

The task was to create a computer model of the SPX core and to find a unique set of reactivity coefficients providing the best fit for the data.









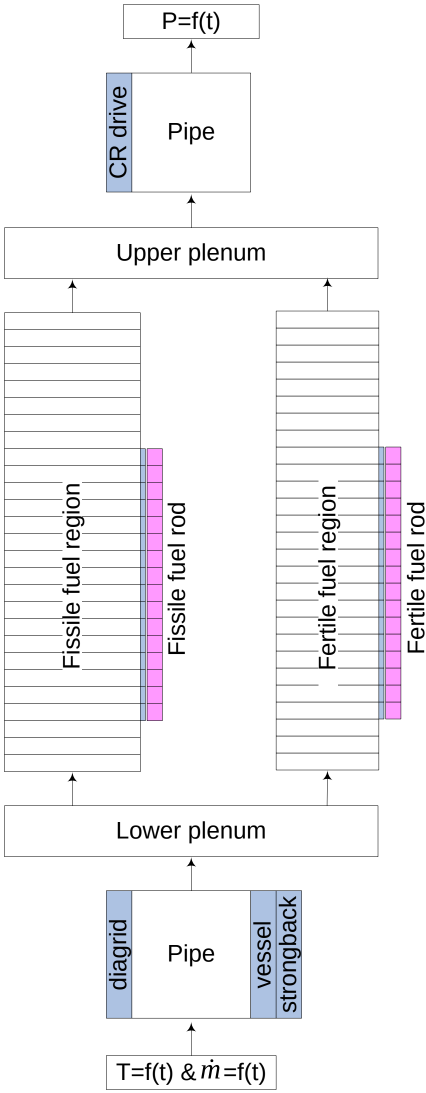
SPX start-up tests: computer model

Thermal-hydraulics model of the SPX core

- 2 channels and 2 heat structures (fuel rods) representing 364 SAs of the fissile core and 233 SAs of the fertile blanket, respectively.
- Fuel modeling: gas gap conductance is evaluated with account for fuel relocation (dependent on the power level) and fuel thermal expansion.
- Heat structures for vessel, strongback, diagrid and CR drive (thermal inertias – important).
- Boundary conditions: core inlet temperature and flow rate, outlet pressure.

Point-kinetics model

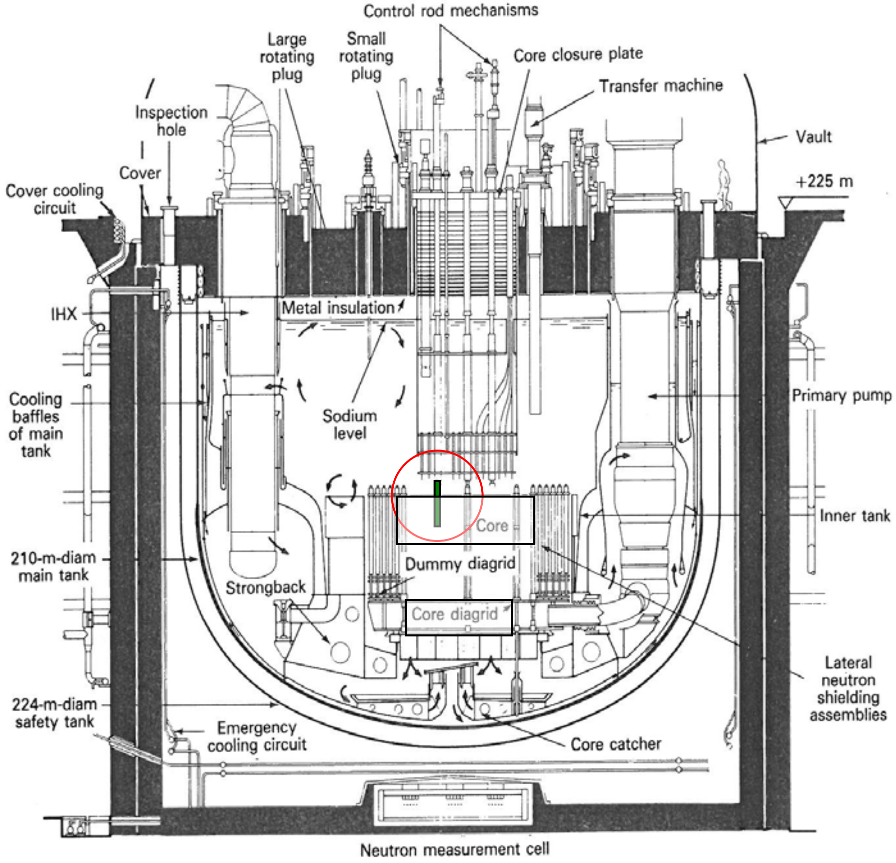
- The same set of reactivity coefficients was used in all calculations
- Reactivity decomposition: Total reactivity = Doppler + FuelExp + CoolantExp + DiagridExp + StrongbackExp + VesselExp + CRdriveExp

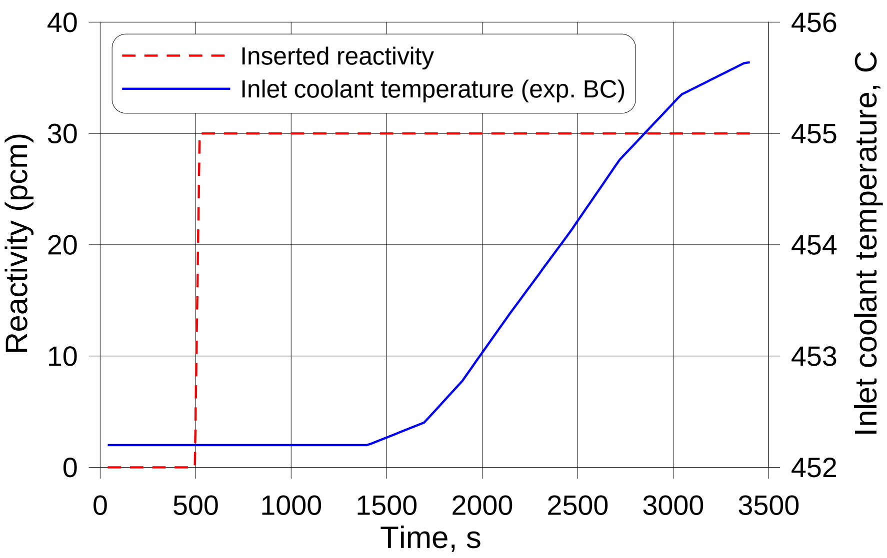


SPX start-up tests: positive reactivity step

Start: iso-thermal (452ºC), zero power (85 kWth), 20% flowrate (3.2 t/s)

Perturbation: control rod withdrawal (+30 pcm) ⟶ reactor start-up. In ~1000s inlet coolant temperature starts to grow

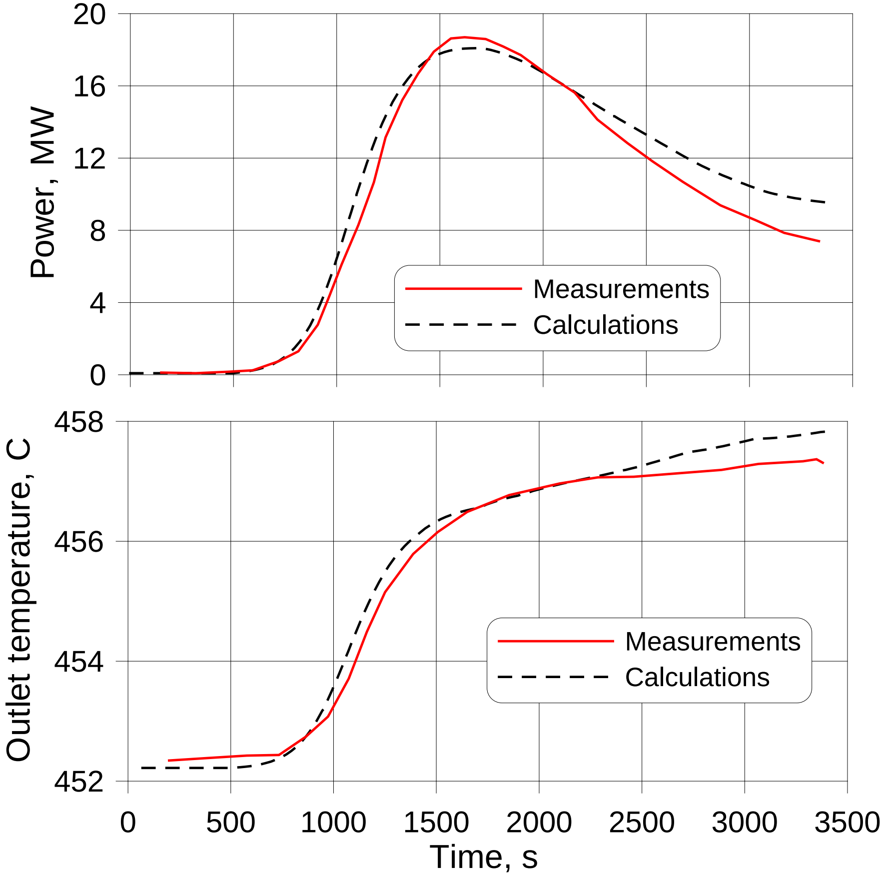


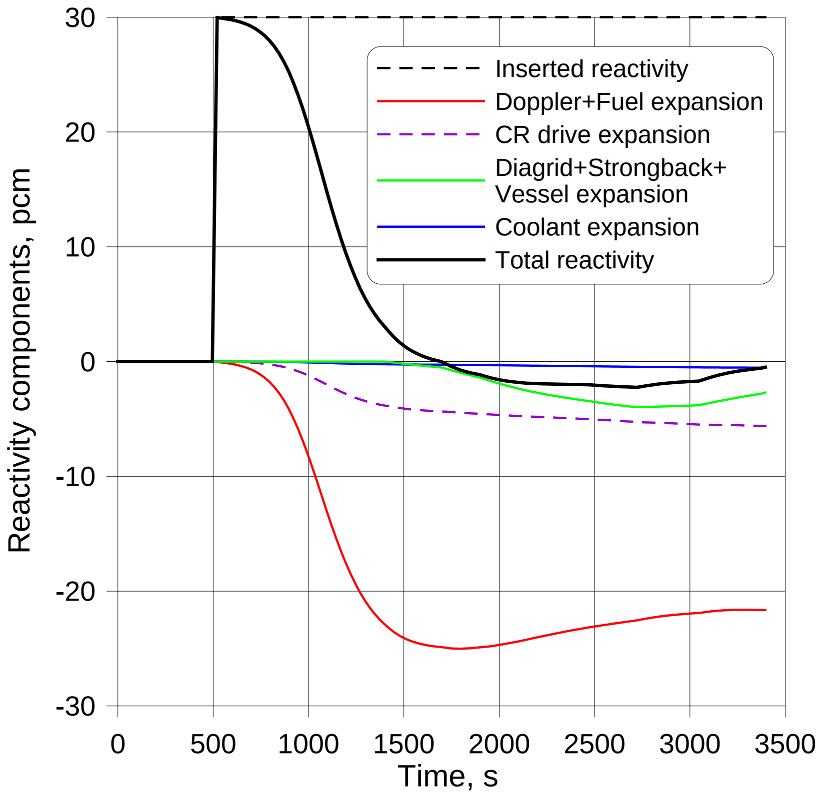


SPX start-up tests: positive reactivity step

Results: withdrawal of absorber → power growth. Doppler (immediately) and expansion effects (with some delay) tend to bring the power back (reduce)

Validation: esp. fuel thermal response, CR driveline expansion & vessel expansion

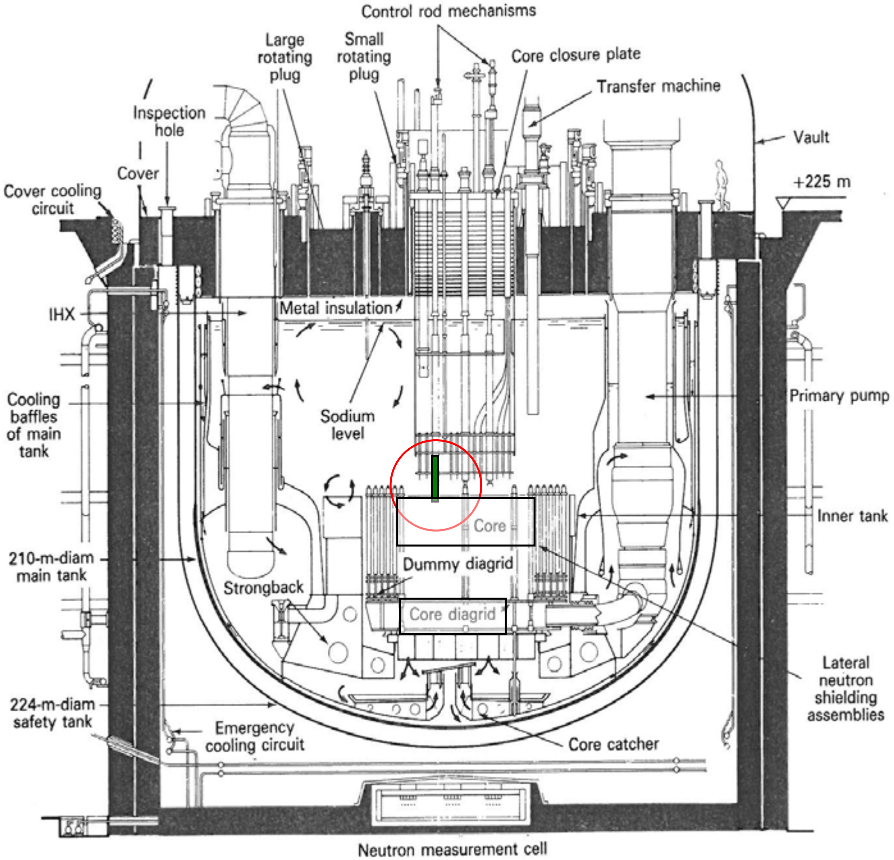


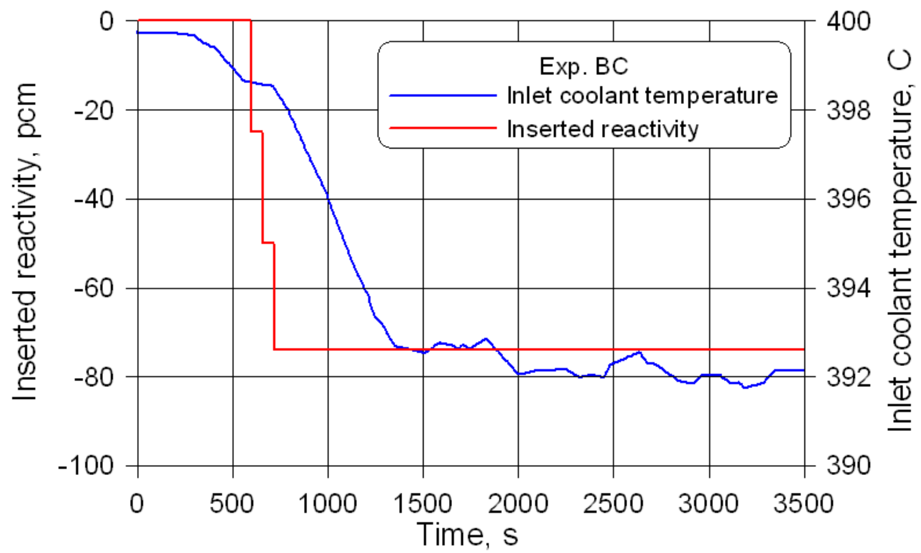


SPX start-up tests: negative reactivity step

Start: 51% power (1540 MWth) 63% flowrate (10.4 t/s)

Perturbation: inlet coolant temperature reduction and
control rod insertion in three steps (–25 pcm × 3)

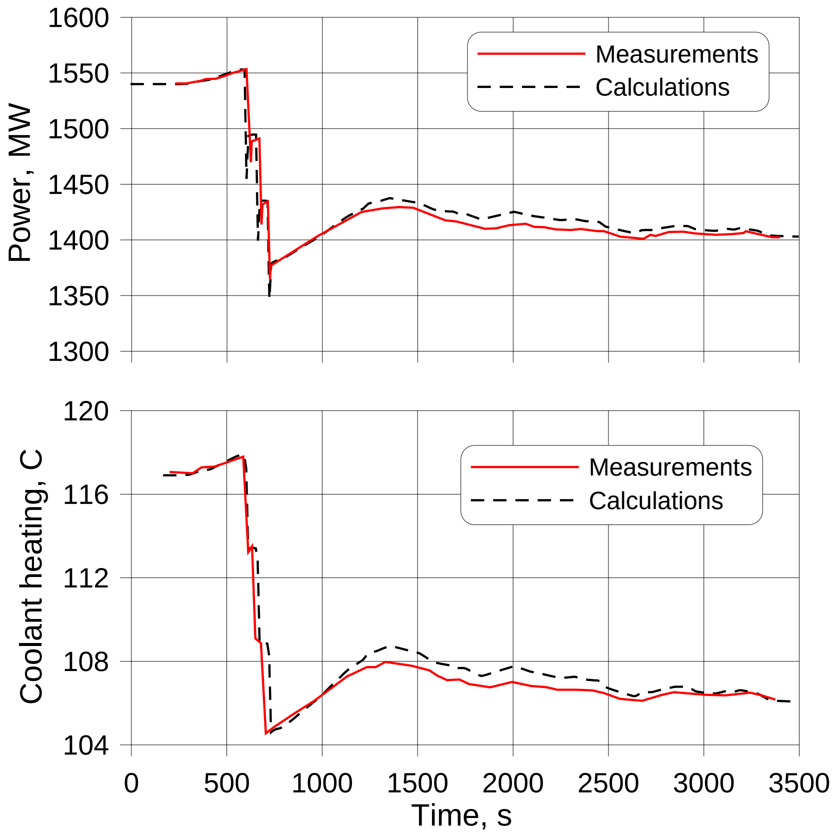


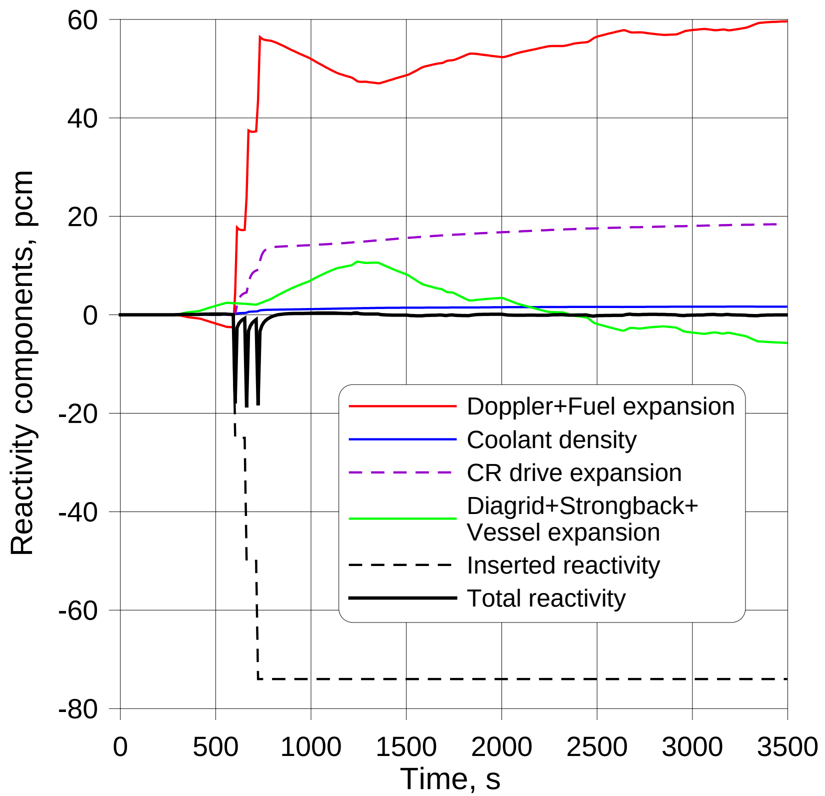


SPX start-up tests: negative reactivity step

Results: insertion of absorber → power drop. Doppler (immediately) and expansion effects (with some delay) tend to bring the power back (increase)

Validation: esp. fuel thermal response, CR driveline expansion & diagrid/strongback/vessel expansions

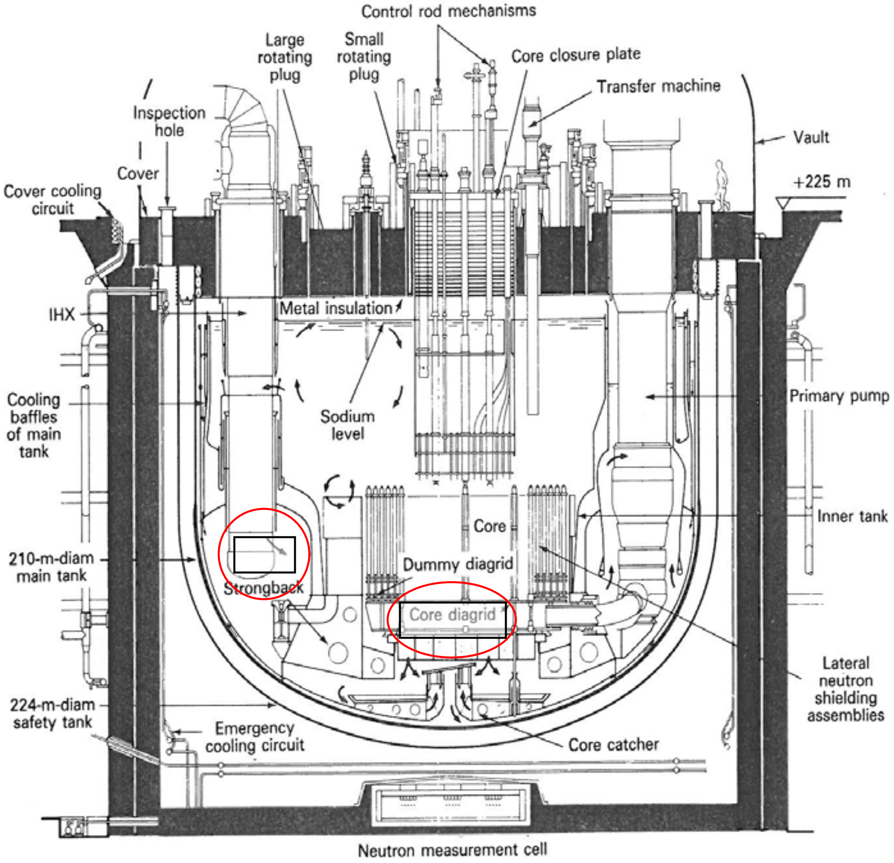


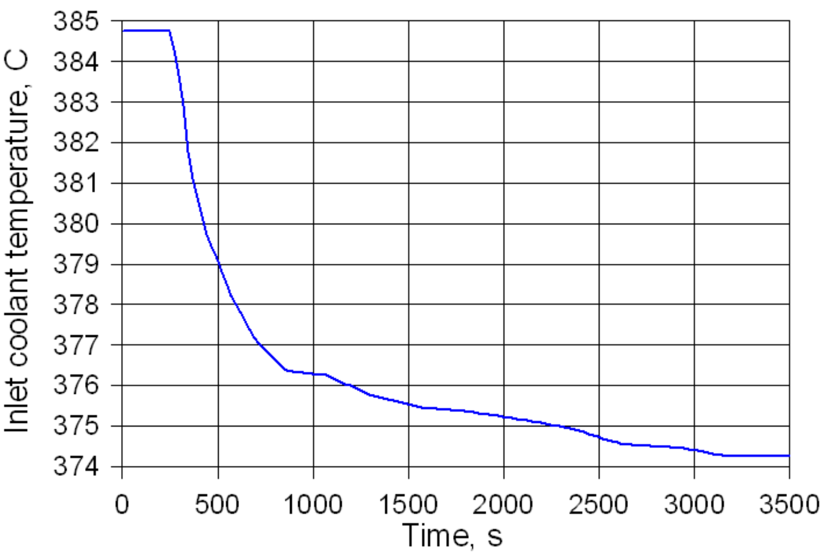


SPX start-up tests: overcooling

Start: 21% power (633 MWth) 40% flowrate (6.5 t/s)

Perturbation: inlet coolant temperature reduction

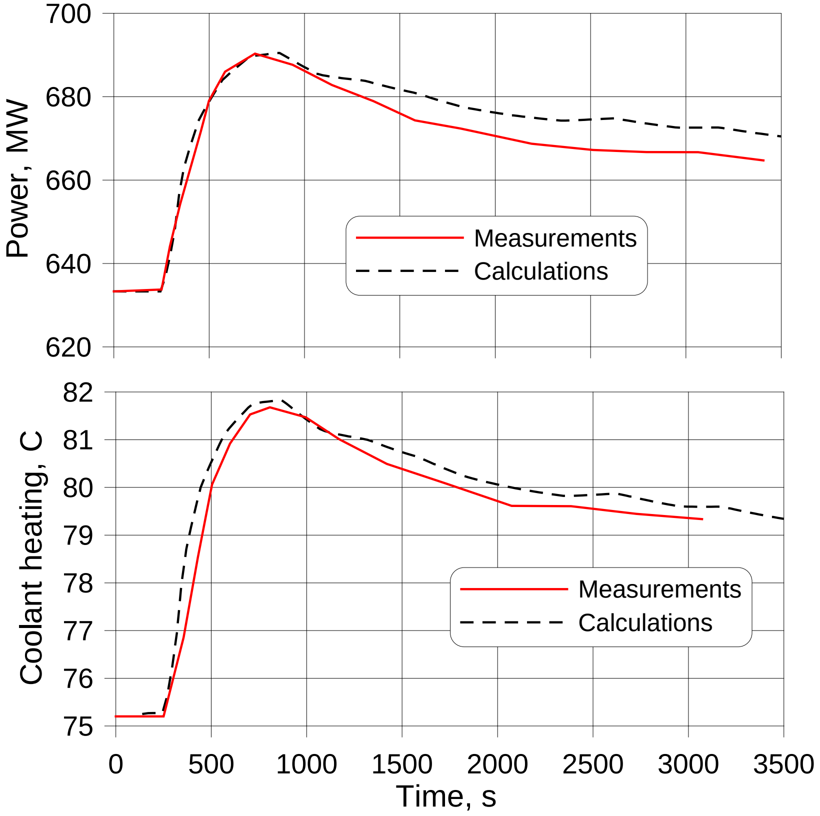


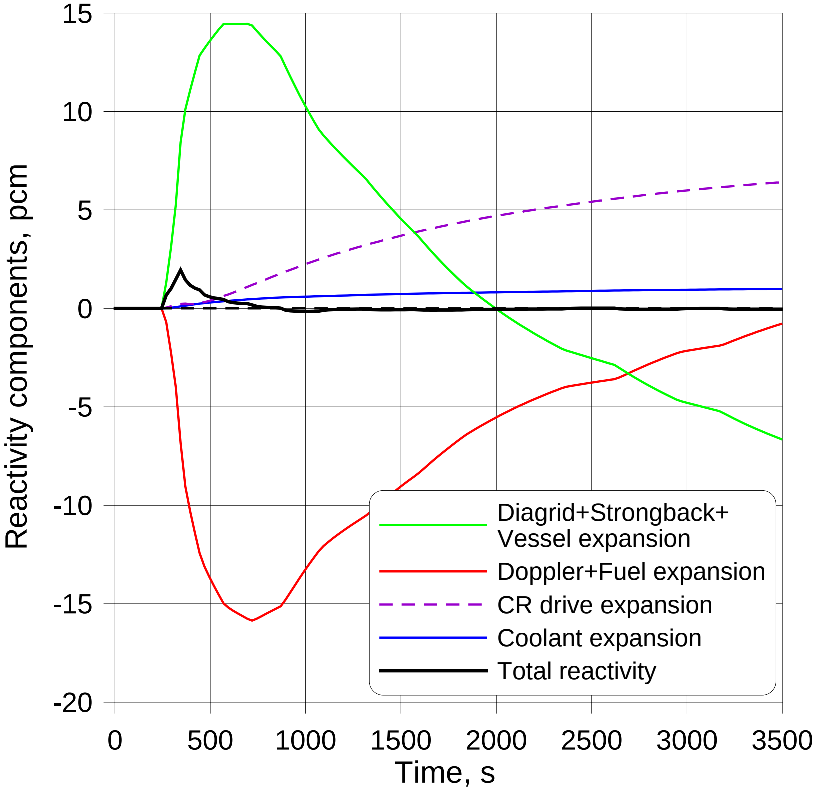


SPX start-up tests: overcooling

Results: positive reactivity due to cooling down at core resulted in a slight core compaction and downward shift is compensated by fuel heats up. With some delay both vessel (ρ↓) and CR driveline (ρ↑) axially contract.

Validation: thermal response of fuel and reactor structures at core inlet followed by CR driveline contraction

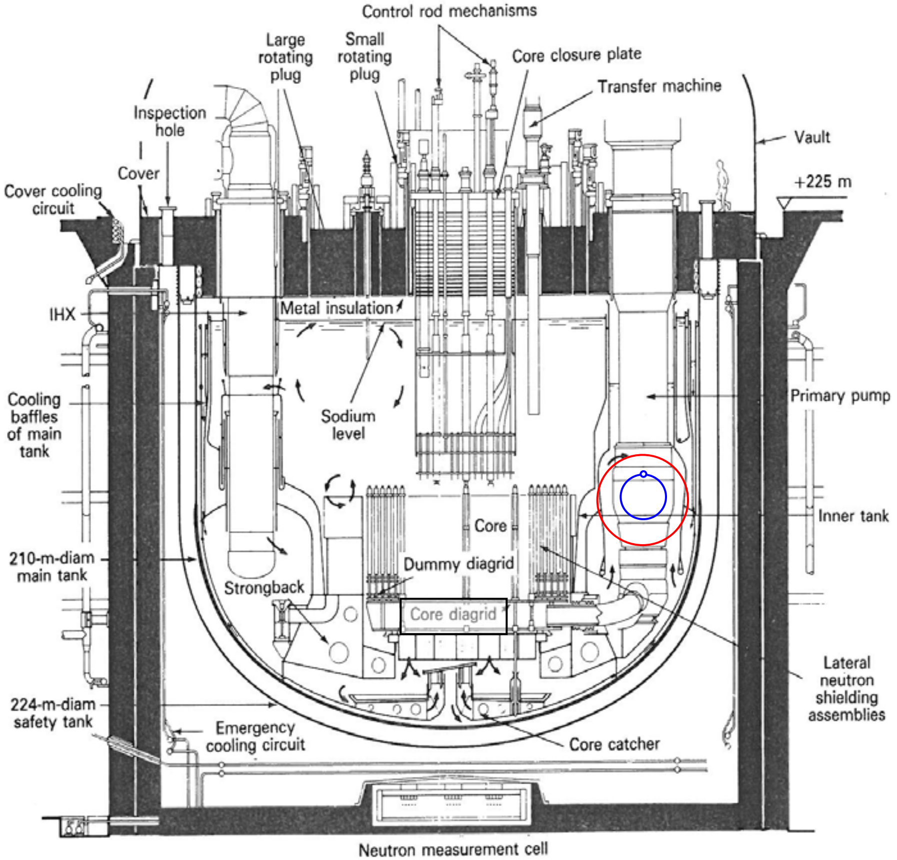


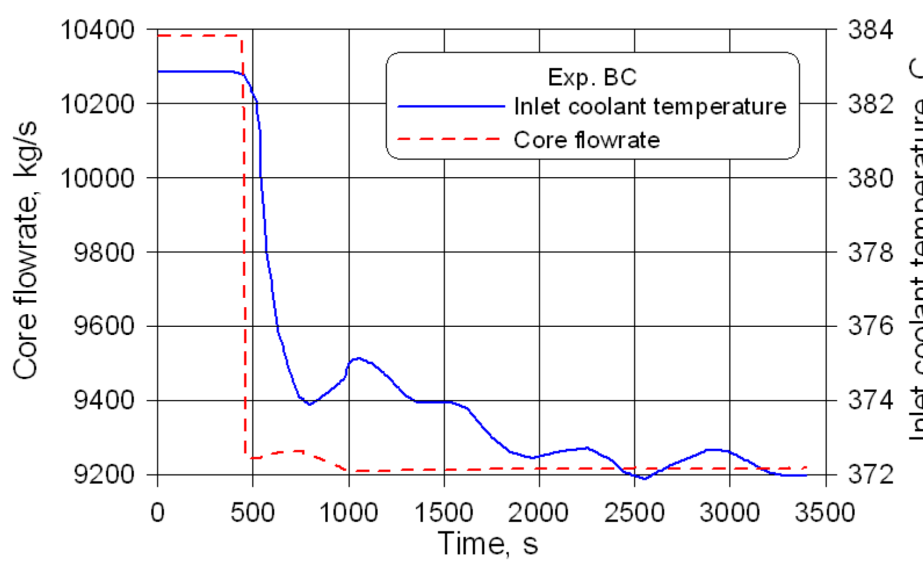


SPX start-up tests: loss of flow

Start: 47% power (1415 MWth) 63% flowrate (10.4 t/s)

Perturbation: ~12% decrease of the primary flowrate, followed by decrease of inlet coolant temperature

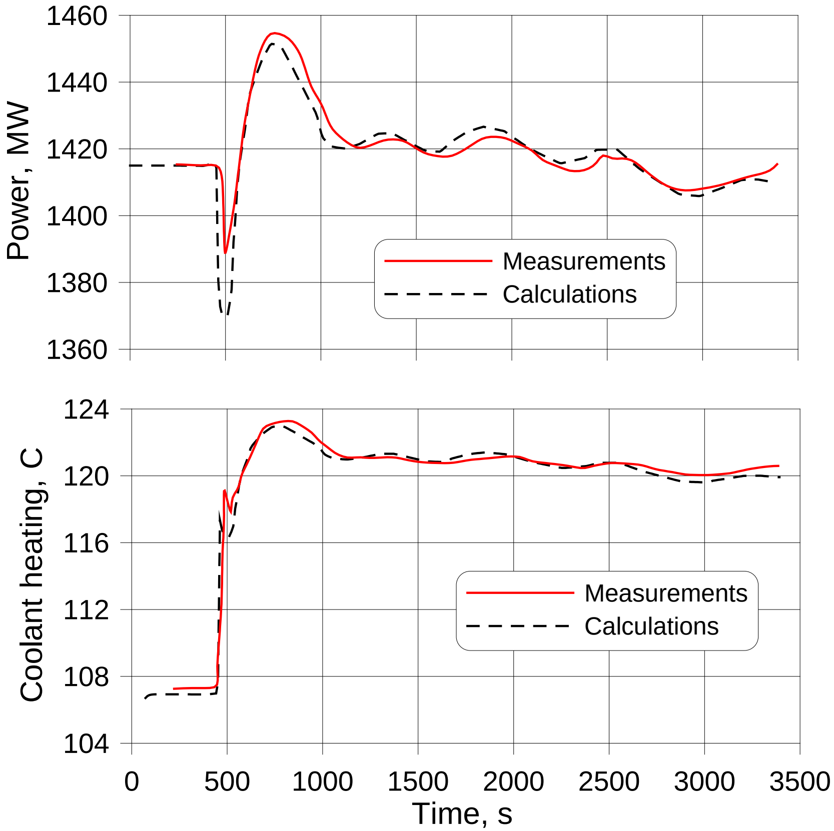


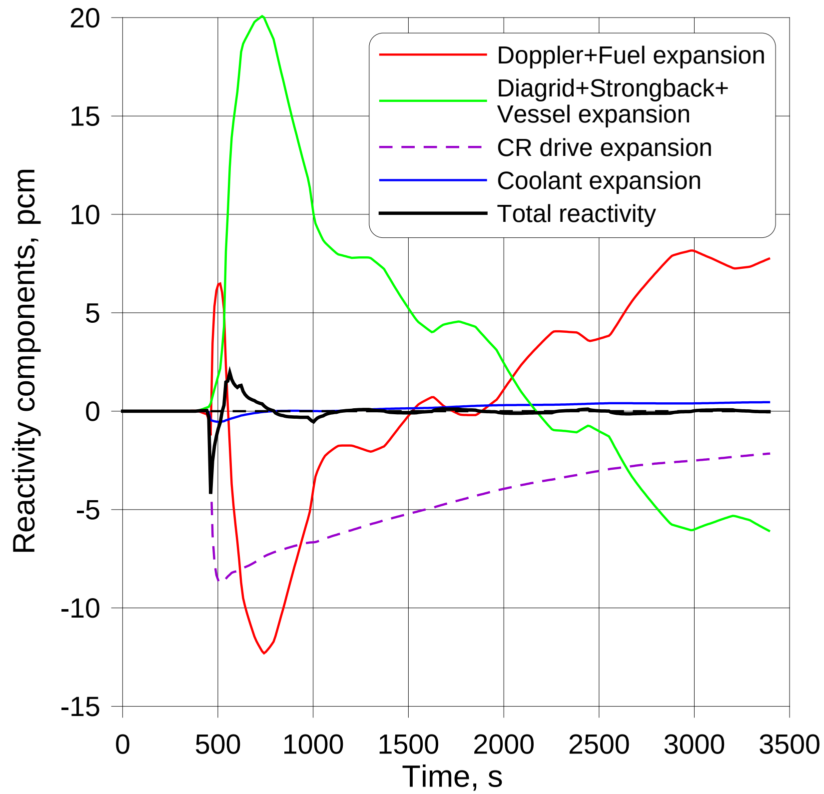


SPX start-up tests: loss of flow

Results: first response is outlet coolant temperature increase and CR drive expansion (ρ↓) smoothened by Doppler (ρ↑). Increase of IHX efficiency leads to cooling down of reactor structures at core inlet (ρ↑). Finally with the delay vessel axially contracts and slightly shifts the core up (ρ↓).

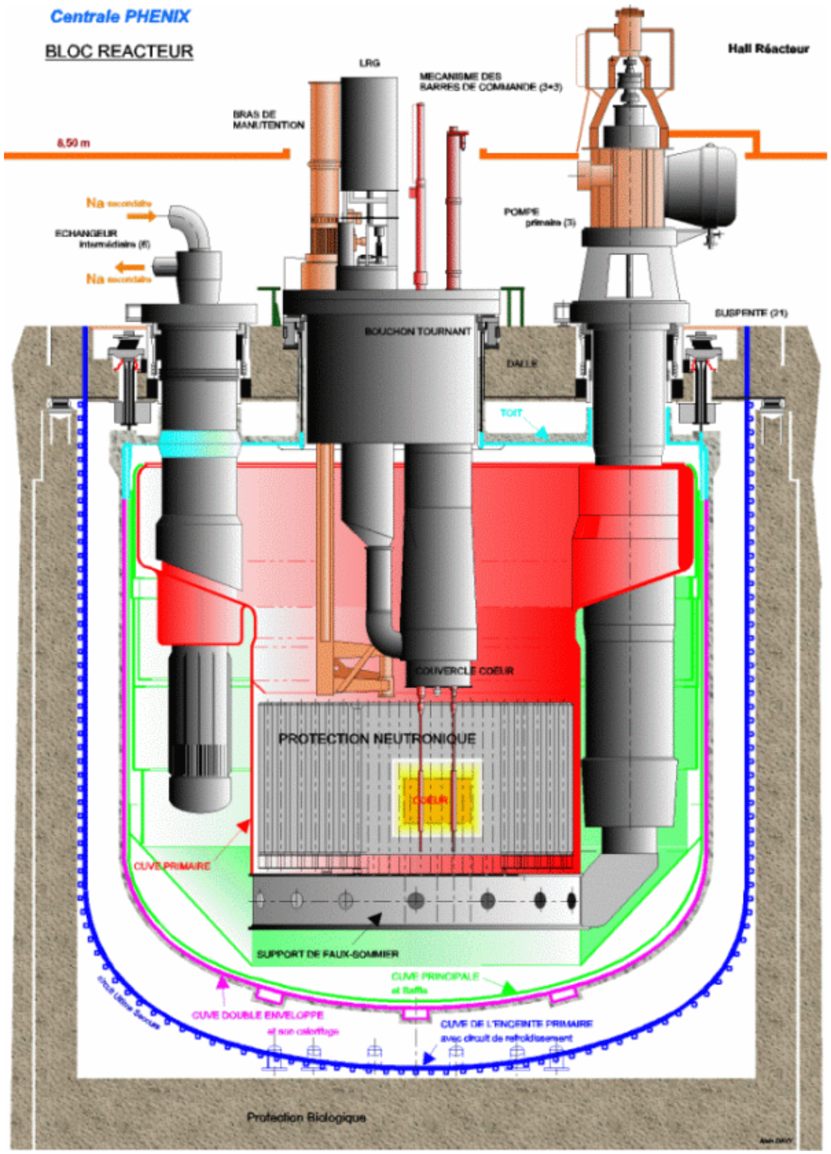
Validation: esp. dynamic thermal response of CR driveline and vessel





Design basis conditions example:
protected loss of flow at Phenix (DBC-4)

Phenix: medium-power SFR



CEA courtesy

Power, MWt	560
Flowrate, t/s	2.8
Inlet/outlet temp, C	400 / 560
Vessel height × diameter, m	10 × 12
Primary sodium mass, t	800
Fissile core diameter, m	1.50
Fissile core height, m	0.85
Fissile core volume, m^3	1.51
Fuel pellet ID, mm	0.0
Fuel pin OD, mm	6.6
Number of fissile/fertile SAs	106 / 86
Fissile/fertile	MOX/UO2
Number of pins per SA	217

PX End of Life Tests

Large test program was carried out in 2009 by CEA, EDF and AREVA before final shutdown:

Thermal-hydraulic tests:

- Asymmetrical transient
- Natural convection

Core physics tests:

- Decay heat
- Control rod offsetting
- Subassembly reactivity worth
- Control rod worth
- Sodium void

Fuel test:

- Partial fuel melting

Negative reactivity transient investigations:

- Carrier/Blankets interactions
- Core flowering

Proceedings of ICAPP 2011, Nice, France, May 2-5, 2011, Paper 11298

The Phenix Final Tests

A. VASILE, B. FONTAINE. M. VANIER, P. GAUTHE, V. PASCAL, G. PRULHIERE, P. JAECKI, L. MARTIN

CEA Cadarache, BP 1, 13108 St Paul Lez Durance – France

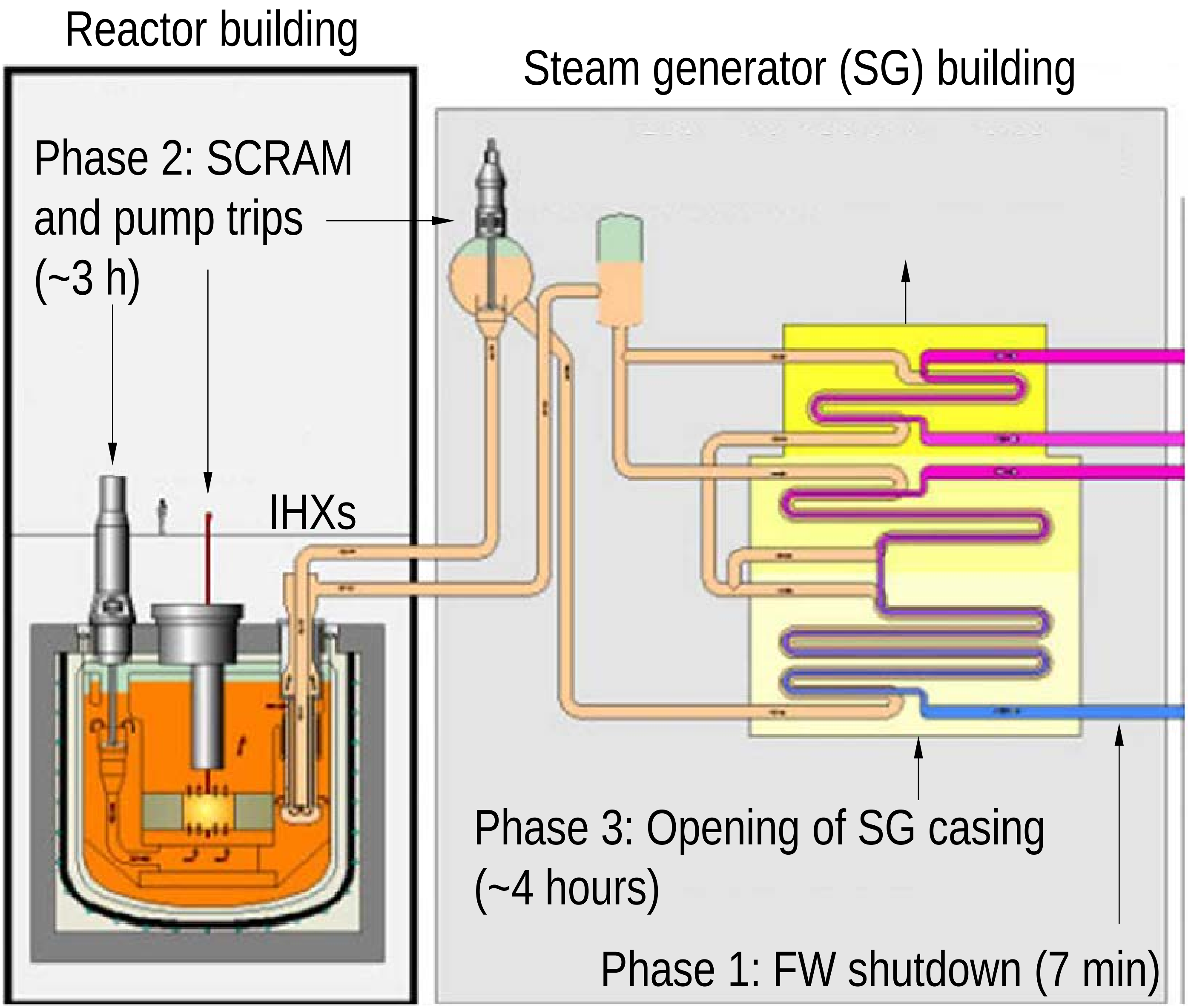
D. TENCHINE
CEA Grenoble - France

D. ROCHWERGER, P. BARRET
Centrale Phenix – CEA Marcoule - France

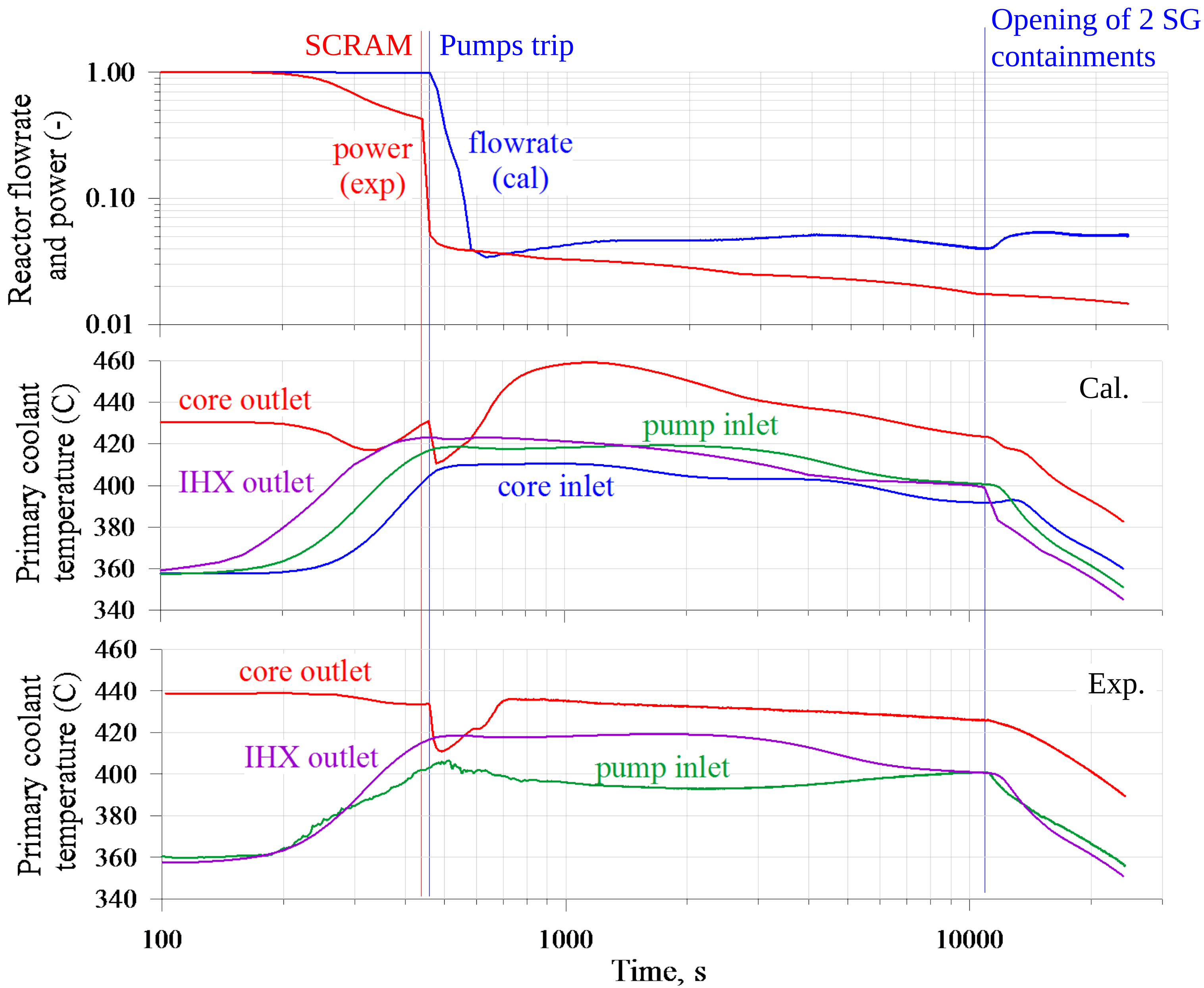
D. VERWAERDE
EDF, 1 Av. du Gral. De Gaulle 92141 Clamart - France

R. DUPRAZ
AREVA NP - 10 rue Juliette Recamier 69456 LYON Cedex - France

PX EOL natural convection test

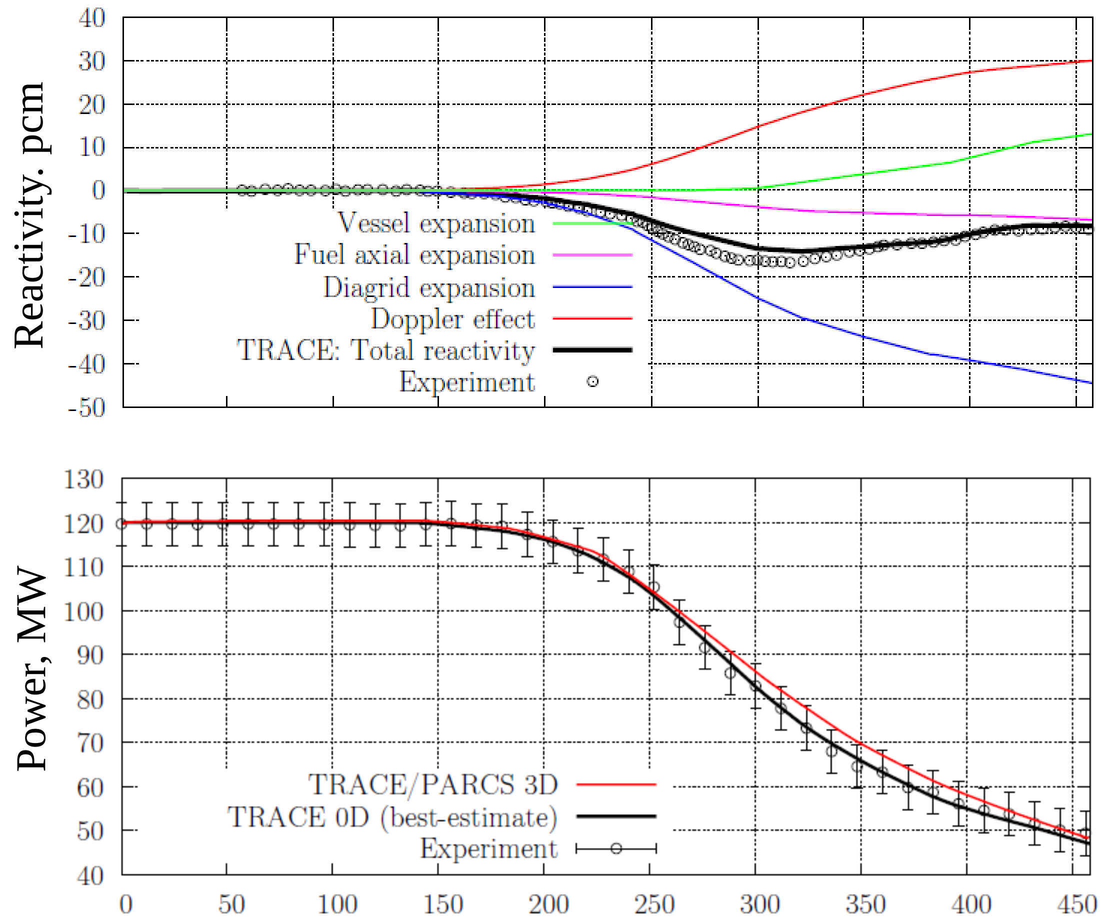


PX EOL natural convection test



PX EOL natural convection test

- First 7 min of the test is the Unprotected Loss of Heat Sink
- 0D (TRACE) and 3D (TRACE/PARCS) neutron kinetics solvers. Good agreement for reactivity (< 3 pcm error) and power
- Decomposition of reactivity: diagrid radial expansion is the most important feedback.
- Fissile and fertile fuel zones bring opposite feedbacks due to axial expansion.
- Sensitivity study: need to accurately predict *core inlet temperature, fuel-clad gas gap conductance and fuel expansion mechanisms*.

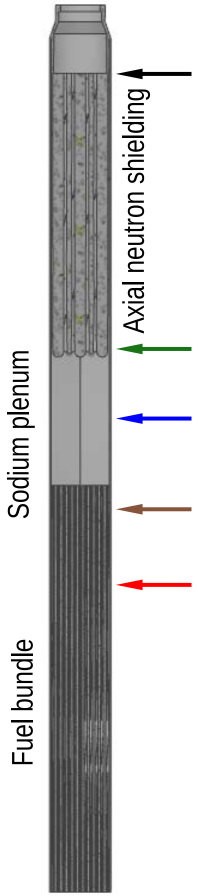


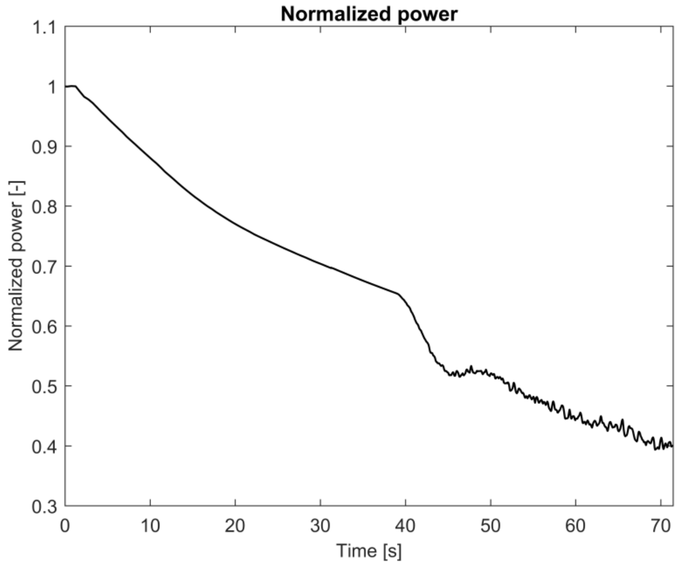
Design-extension conditions

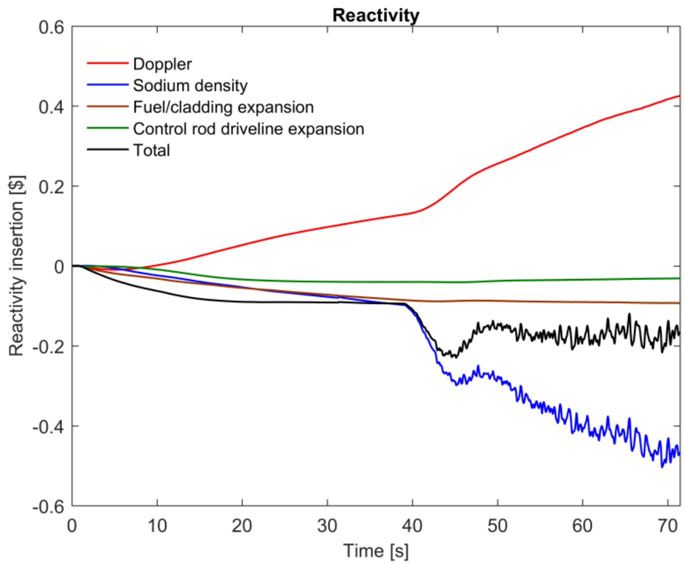
ULOF in low-void core (TRACE results)

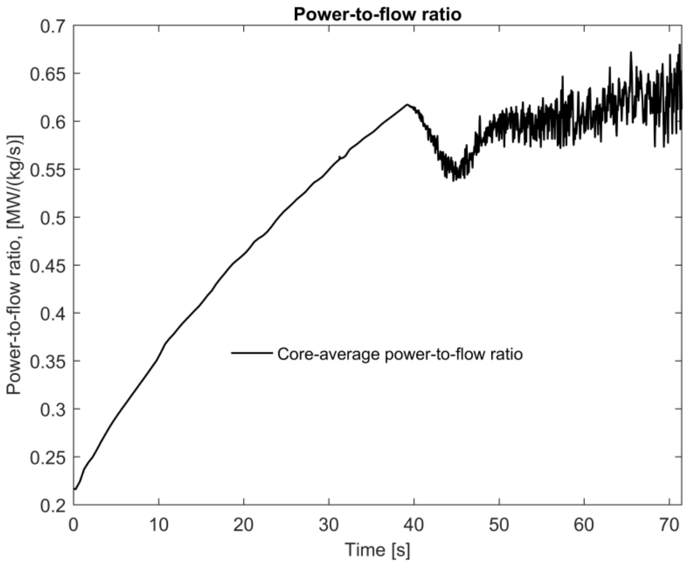
No power run-away

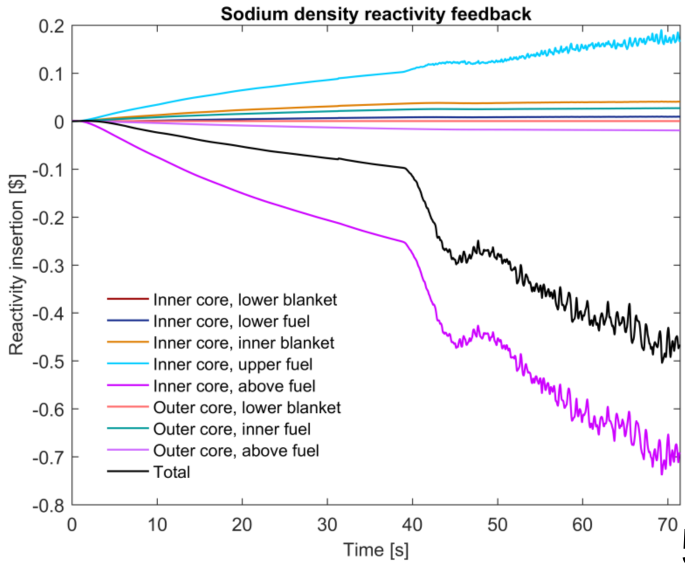
Main players: +Doppler+upper fissile IZ –plenum IZ



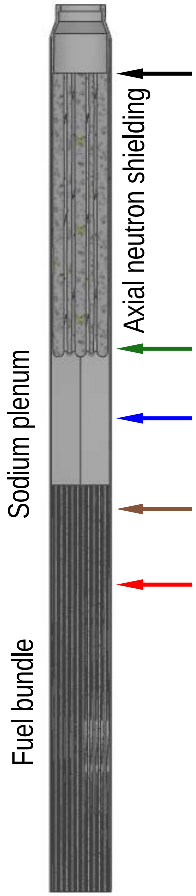




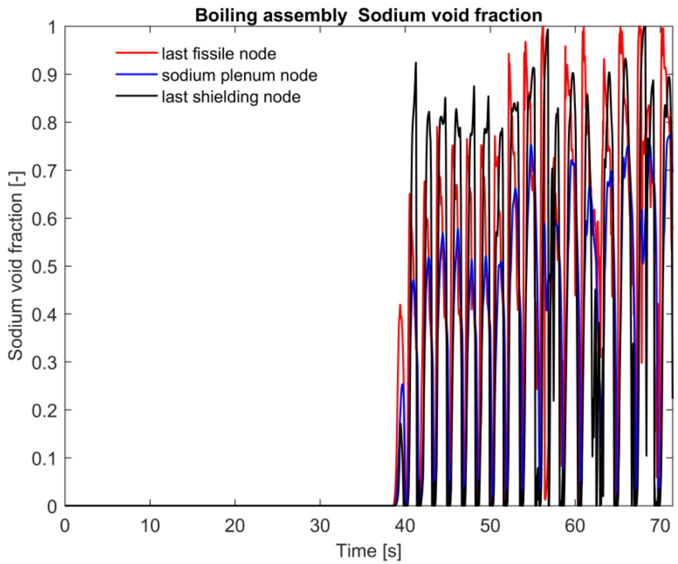


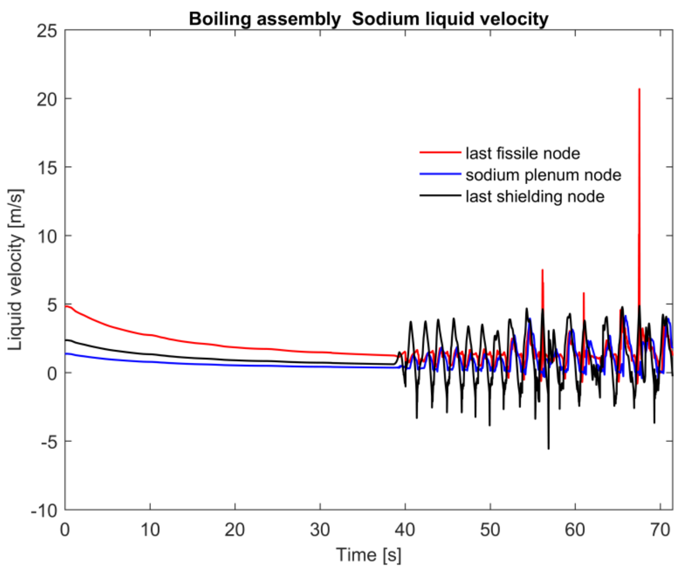


ULOF in low-void core (TRACE results)

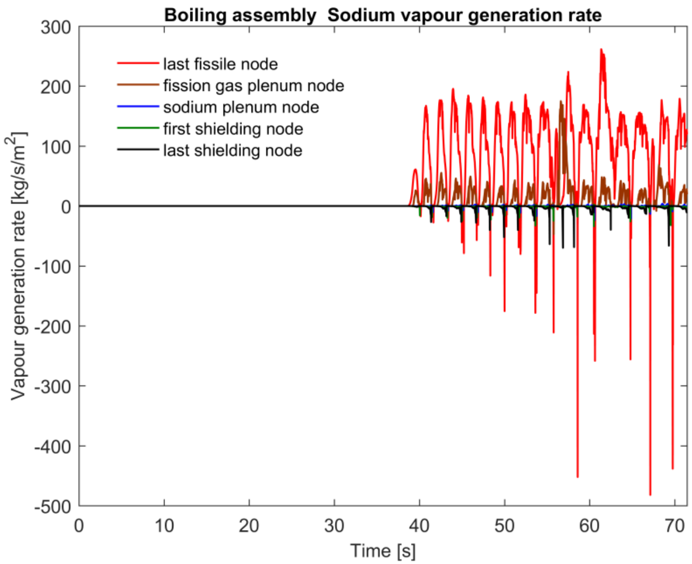


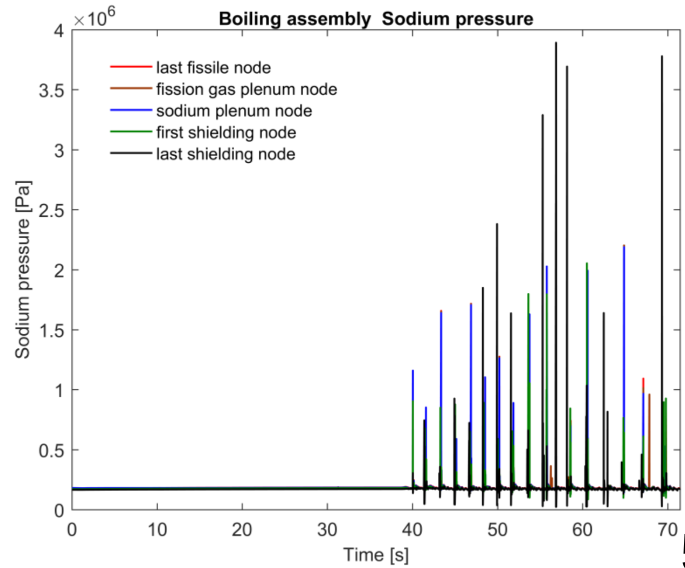
Cyclic boiling regime (chugging)



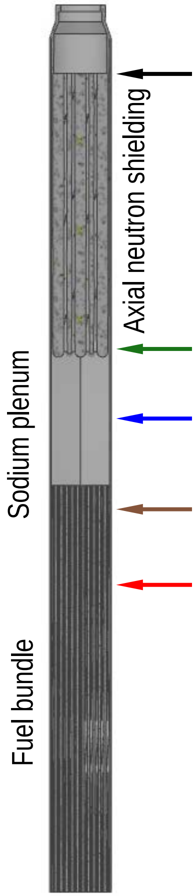


Condensation-induced pressure pulses

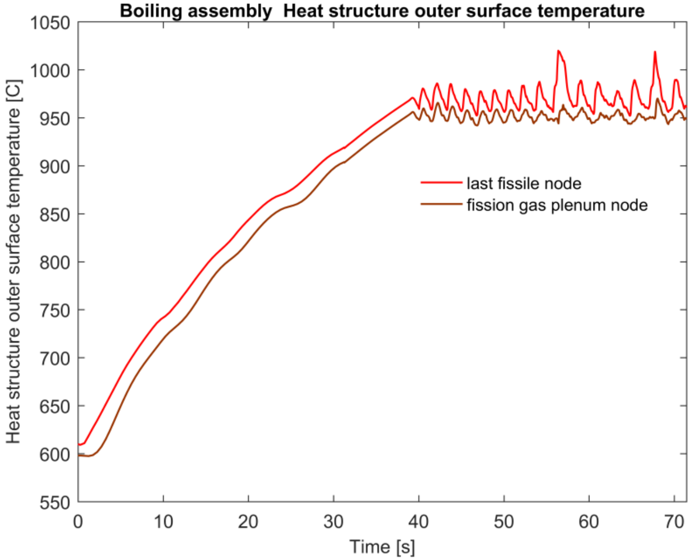




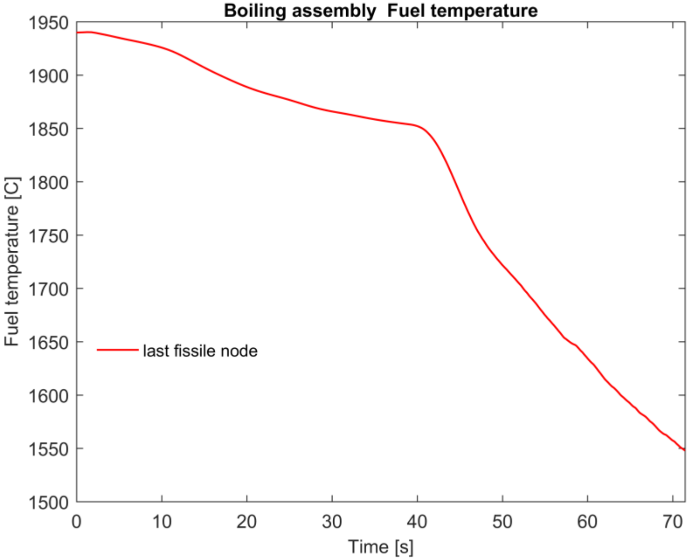
ULOF in low-void core (TRACE results)



No permanent dryout – no clad melting



No fuel melting

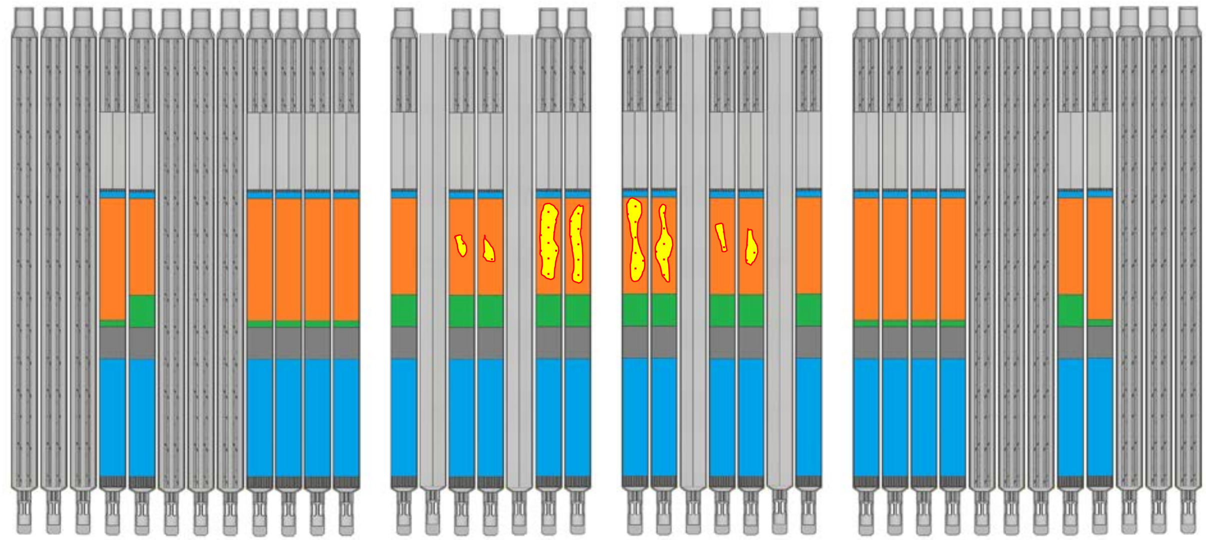


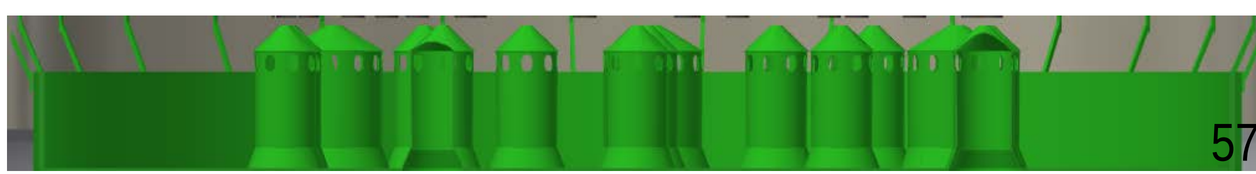
In frame of ESFR-SMART project we launched a CHUG experiment to better understand the chugging boiling conditions injecting saturated steam into the subcooled water

Severe accident phenomenology

- **Initiation phase**
- Transition phase
- Expansion phase
- In-vessel cooling

— From fuel (or clad) melting onset till hexcan wall melting
— Localized inside fuel subassembly
— Energetics depends on competition between negative (Doppler and fuel relocation) and positive (sodium void, cladding relocation) effects
— Code used: SAS4A/ SAS-SFR validated using CABRI and TREAT data

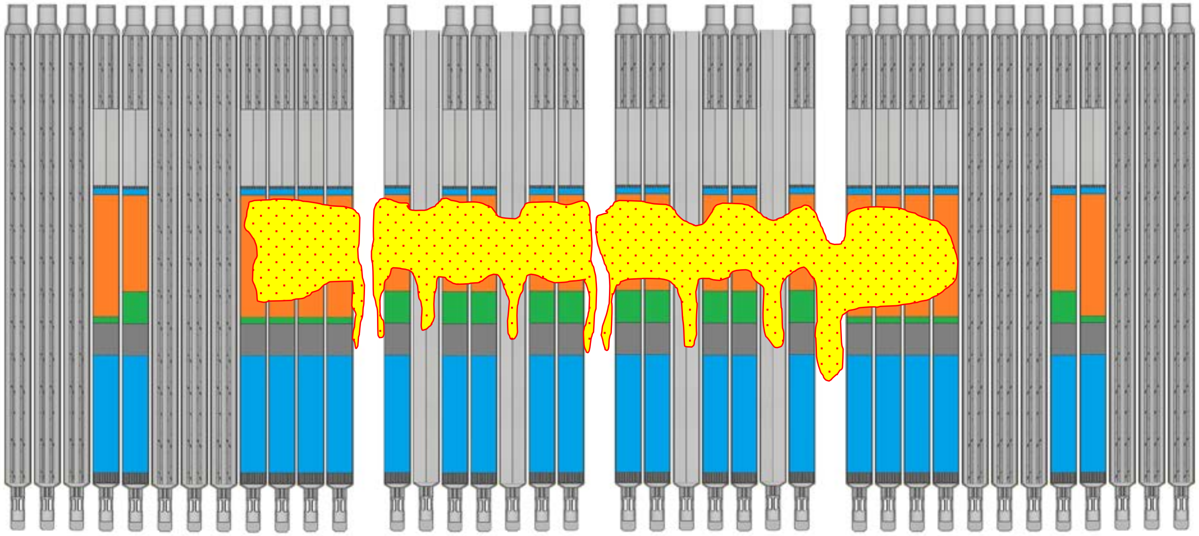


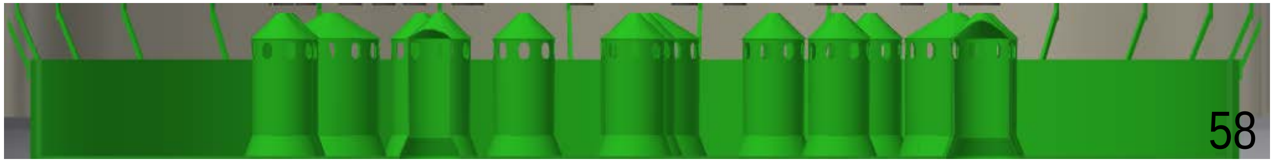


Severe accident phenomenology

- Initiation phase
- **Transition phase**
- Expansion phase
- In-vessel cooling

— From hexcan wall melting till re-criticality
— Radial propagation of melting front
— Energetics depends on materials stratification in the molten pool
— Code used: SIMMER validated using CABRI and SCARABEE data





Severe accident phenomenology

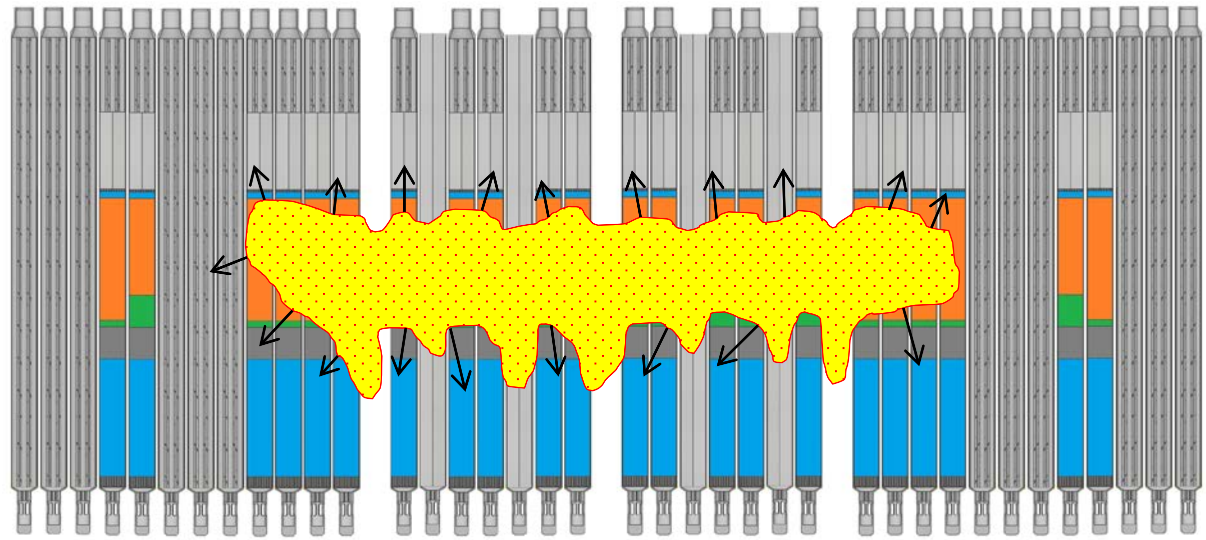
- Initiation phase
- Transition phase
- **Expansion phase**
- In-vessel cooling

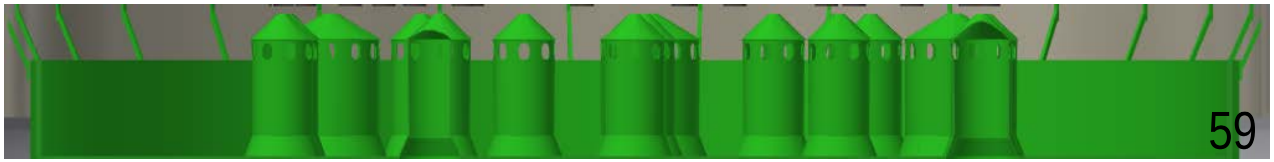
— From re-criticality till corium relocation from the core

— Nuclear energy release with energetics depending on re-criticality level

— Mechanical impact on vessel or reactor pit in case of vessel failure

— Code used: SIMMER validated using CABRI and SCARABEE data

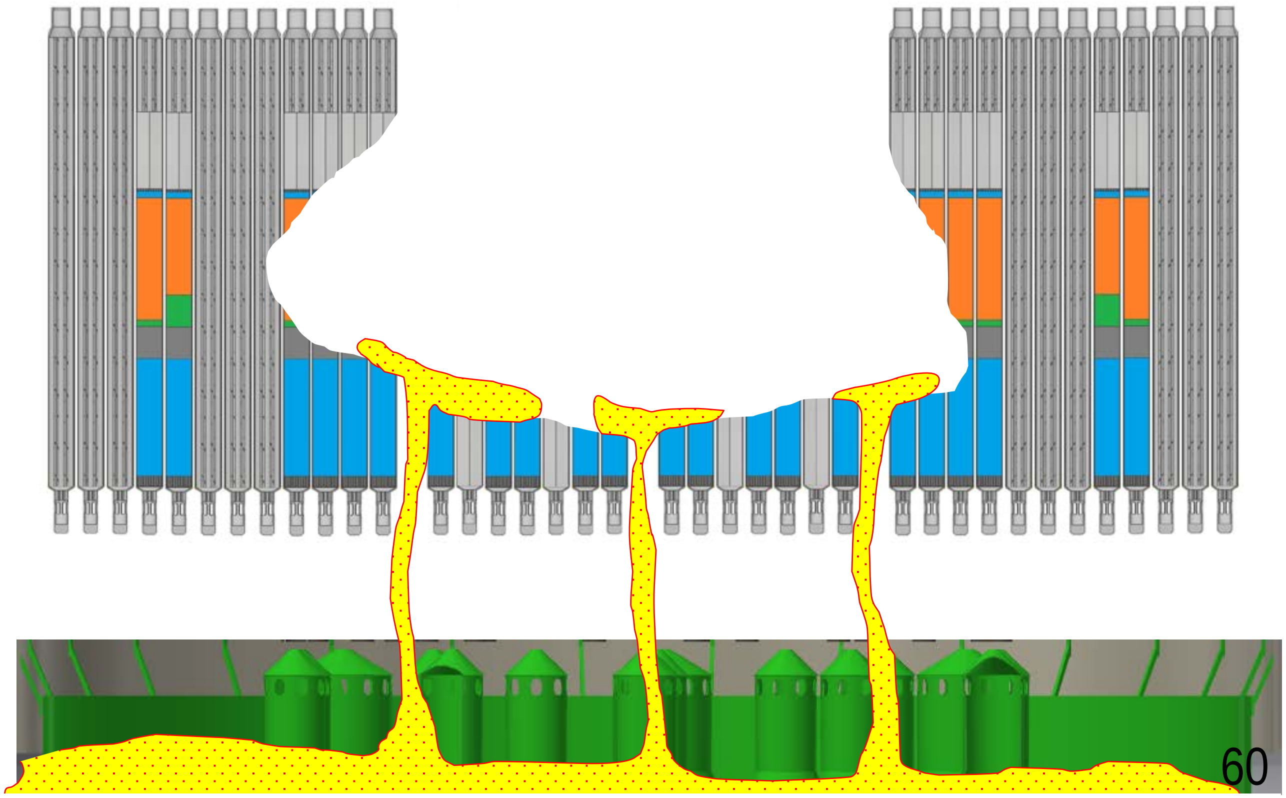




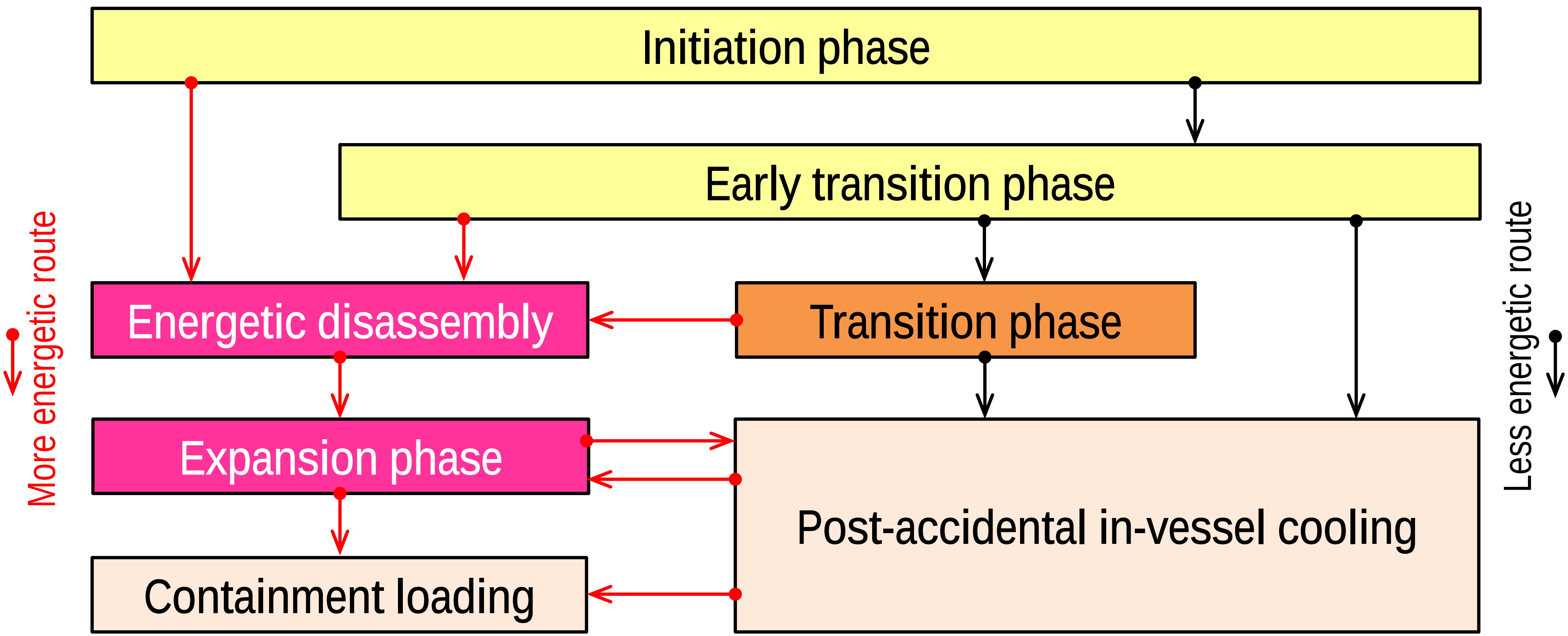
Severe accident phenomenology

- Initiation phase
- Transition phase
- Expansion phase
- In-vessel cooling

— Stabilization
— Decay heat release
— Codes used: CFD



Severe accident routes classification



- *Initiation phase*: fuel melting onset; 1D fuel flow phenomena; damage limited to pin/SA
- *Transition phase*: fuel pool formation; 2D fuel flow phenomena; recriticalities; high thermal energies
- *Energetic disassembly phase*: large-scale core break-up
- *Expansion phase*: mechanical energy release; challenge of vessel?
- *Containment loading phase*: fuel/concrete interaction; challenge of containment?
- *Post-accidental in-vessel cooling phase*: stabilization / aggravation?

From: Marco Marchetti, PhD thesis “Neutronics Methods for Transient and Safety Analysis of Fast Reactors”, KIT Scientific Report 7728, 2016

LMFR safety analysis: few examples of codes

Overview of codes used in Europe for LMFR modeling

Examples of codes used for LMFR DBA and DEC

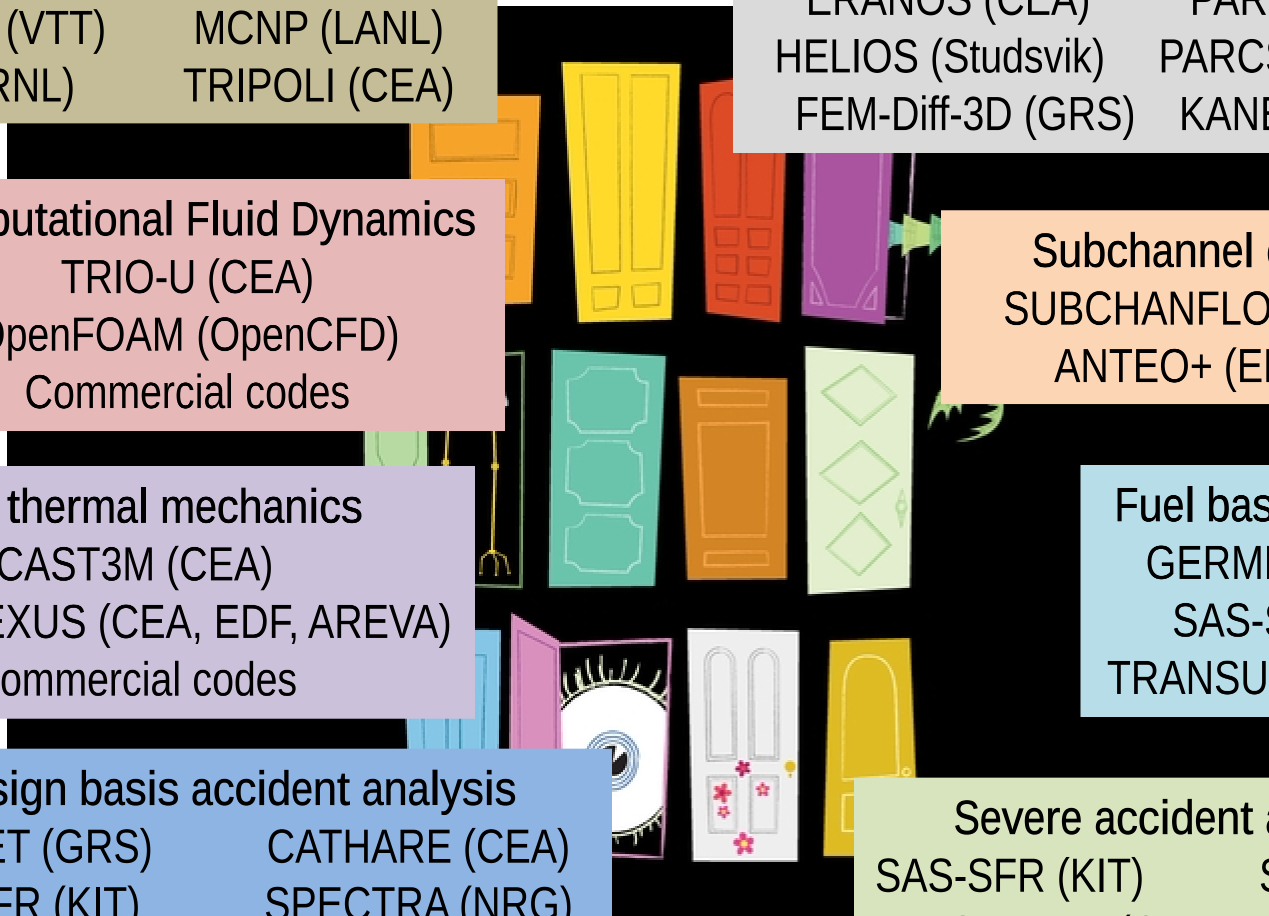
- TRACE
  - Sodium boiling features
  - TRACE 6-eq. model for 2Φ sodium
  - Sodium boiling test: KNS sodium loop

- SIMMER
  - General structure
  - Nodalization
  - Heat and mass transfer
  - Neutronics

Codes used in Europe for LMFR modeling

Monte Carlo neutronics
SERPENT (VTT) MCNP (LANL)
SCALE (ORNL) TRIPOLI (CEA)

Deterministic neutronics
ERANOS (CEA) PARIS (CEA)
HELIOS (Studsvik) PARCS (Purdue)
FEM-Diff-3D (GRS) KANEXT (KIT)



Computational Fluid Dynamics
TRIO-U (CEA)
OpenFOAM (OpenCFD)
Commercial codes

Subchannel codes
SUBCHANFLOW (KIT)
ANTEO+ (ENEA)

Core thermal mechanics
CAST3M (CEA)
EUROPLEXUS (CEA, EDF, AREVA)
Commercial codes

Fuel base irradiation
GERMINAL (CEA)
SAS-SFR (KIT)
TRANSURANUS (ITU)

Design basis accident analysis
ATHLET (GRS) CATHARE (CEA)
SIM-SFR (KIT) SPECTRA (NRG)
RELAP5 (ENEA) TRACE (PSI)
GeN-Foam (PSI, EPFL)

Severe accident analysis
SAS-SFR (KIT) SAS4A (ANL)
SIMMER (CEA, KIT, JAEA)
ASTEC-Na (IRSN)

Examples of codes used for LMFR DBA

TRACE – example of a legacy (system) code

- US NRC code developed for LWR DBA analysis
- Includes other coolants, in particular single-phase sodium and lead
- Modified at PSI for two-phase sodium flow using existing coding (6-equations) and modifying the closure relations when needed
- Validated using available test data

Sodium boiling features

Due to the thermal-physical properties of sodium (high k_l and low ρ_g / ρ_l) the boiling is characterised by:

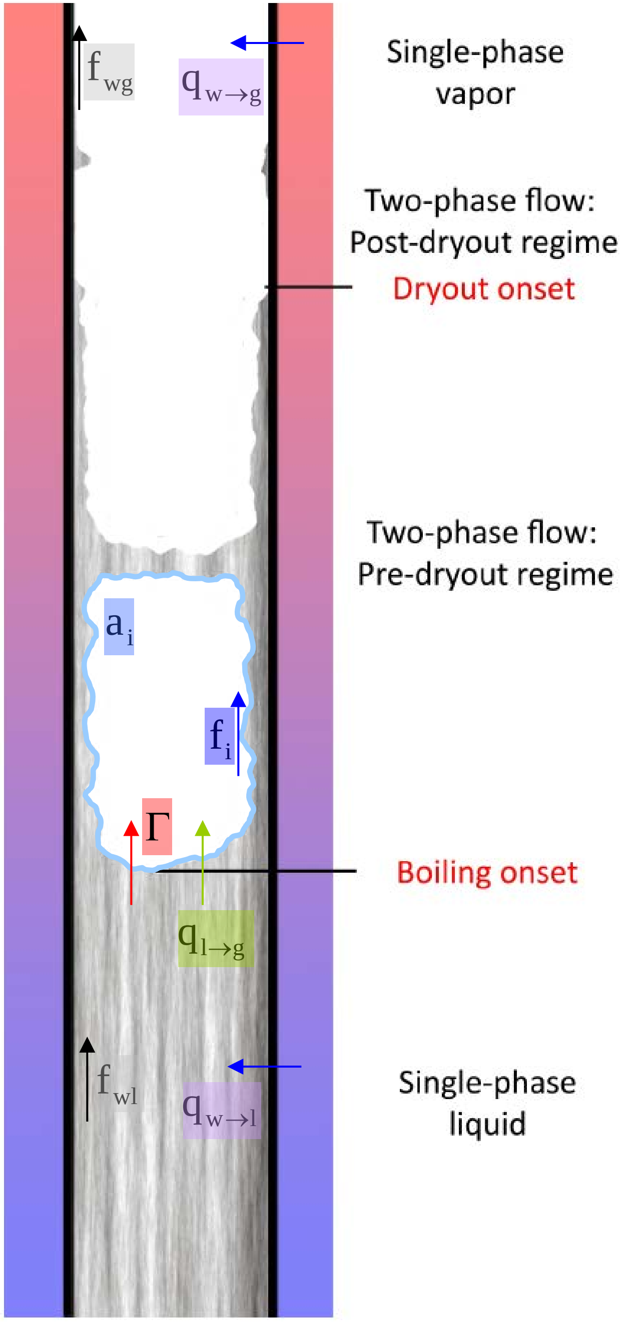
-- quick formation of big bubbles

-- domination of annular flow regime

-- dryout mechanism of crisis

Closure relations:

- a_i interfacial area density (m^2/m^3)
- Γ intefacial mass transfer rate (kg/m^{3}s)
- f_i intefacial drag coefficient
- $q_{l\to g}$ interfacial heat transfer (W/m^3)
- f_{wl} wall-to-liquid drag coefficient
- $q_{w\to l}$ wall-to-liquid heat transfer (W/m^3)
- f_{wg} wall-to-gas drag coefficient
- $q_{w\to g}$ wall-to-gas heat transfer (W/m^3)



TRACE 6-eq. model for 2Φ sodium

Liquid- and gas-phase mass conservation

$$\frac{\partial[(1-\alpha)\rho_l]}{\partial t}+\nabla[(1-\alpha)\rho_l V_l]=-\Gamma$$

$$\frac{\partial(\alpha\rho_g)}{\partial t}+\nabla(\alpha\rho_g V_g)=\Gamma$$

Liquid- and gas-phase momentum conservation

$$\frac{\partial[(1-\alpha)\rho_l v_l]}{\partial t}+\frac{\partial[(1-\alpha)\rho_l v_l^2]}{\partial z}=-(1-\alpha)\frac{\partial P}{\partial z}+a_i f_i\frac{\rho_l v_r|v_r|}{2}+\Gamma v_r-\frac{f_{wl}}{D_h}\frac{\rho_l v_l|v_l|}{2}+(1-\alpha)\rho_l\vec{g}\cdot\vec{i}$$

$$\frac{\partial[\alpha\rho_g v_g]}{\partial t}+\frac{\partial[\alpha\rho_g v_g^2]}{\partial z}=-\alpha\frac{\partial P}{\partial z}-a_i f_i\frac{\rho_g v_r|v_r|}{2}-\Gamma v_r-\frac{f_{wg}}{D_h}\frac{\rho_g v_g|v_g|}{2}+\alpha\rho_g\vec{g}\cdot\vec{i}$$

Liquid- and gas-phase energy conservation

$$\frac{\partial[(1-\alpha)(\rho_l e_l+\rho_l v_l^2/2)]}{\partial t}+\frac{\partial}{\partial z}[(1-\alpha)(\rho_l e_l+\rho_l v_l^2/2+P)v_l]=-q_{l\to g}+q_{w\to l}-\Gamma h_l$$

$$\frac{\partial[\alpha(\rho_g e_g+\rho_g v_g^2/2)]}{\partial t}+\frac{\partial}{\partial z}[\alpha(\rho_g e_g+\rho_g v_g^2/2+P)v_g]=q_{l\to g}+q_{w\to g}+\Gamma h_v$$

Closure relations:

Interfacial:

- Γ mass transfer rate
- a_i area
- f_i drag coefficient
- $q_{l\to g}$ heat transfer

Wall-to-liquid:

- f_{wl} drag coefficient
- $q_{w\to l}$ heat transfer

Gas-to-liquid:

- f_{wg} drag coefficient
- $q_{w\to g}$ heat transfer

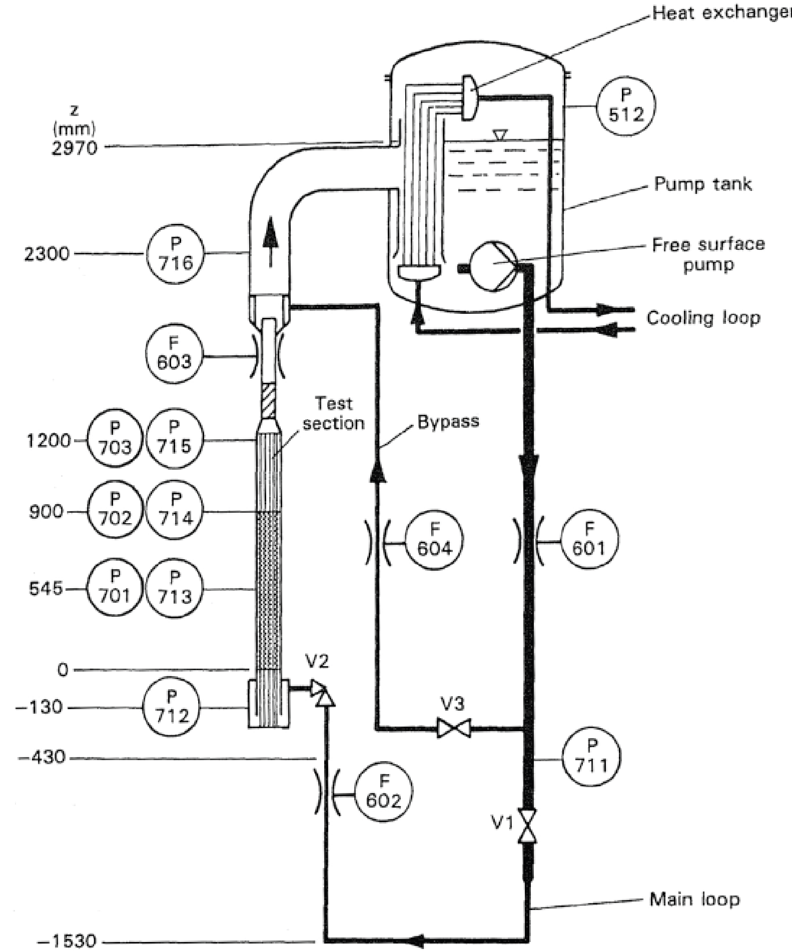
Sodium boiling test: KNS sodium loop

Test section

- 37-pin bundle with e-heaters
- Representative for the SFR SA (SNR-300)
- Power distribution : cosine

Pump

- Free-surface pump tank with heat exchanger
- Pump coastdown characteristics representative for SNR-300 ($t_{1/2}$ ~ 2 s)



Sodium boiling test: KNS sodium loop

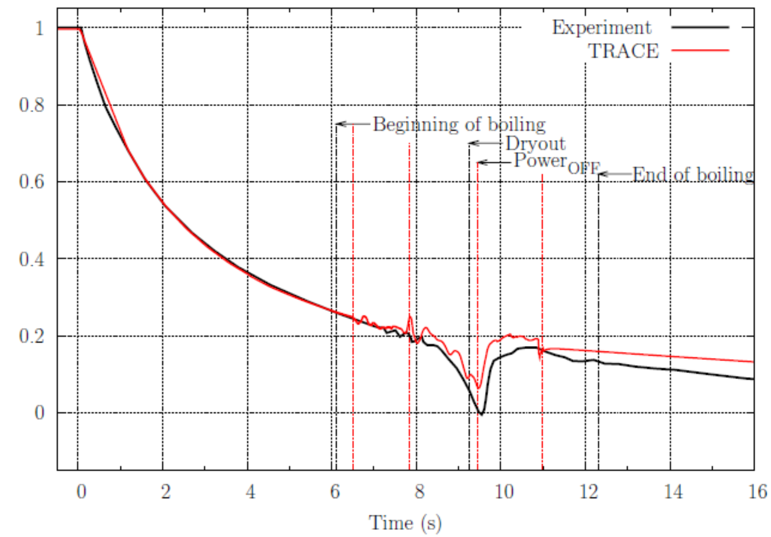
Boiling phenomenology:

- Boiling at the top of the heated section in the bundle center
- First, boiling propagates radially across the central part of the bundle
- Then, it reaches the test section wall (**blocks the flow area**) and then expands axially

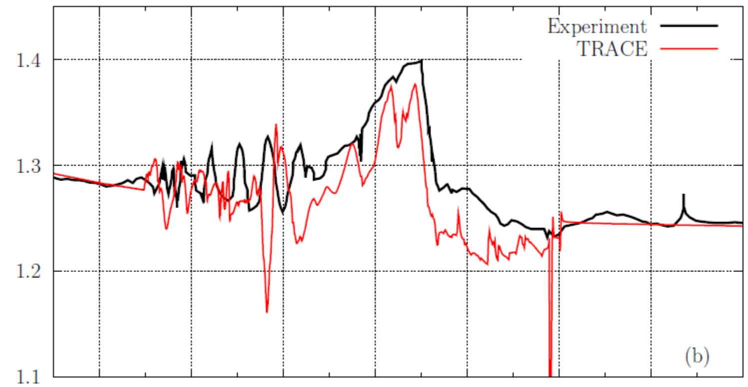
2D TRACE model

- Boiling onset well predicted
- Good prediction of radial expansion during first stage
- Pressure increases when the boiling front reaches the test-section wall

Flowrate (rel.units)

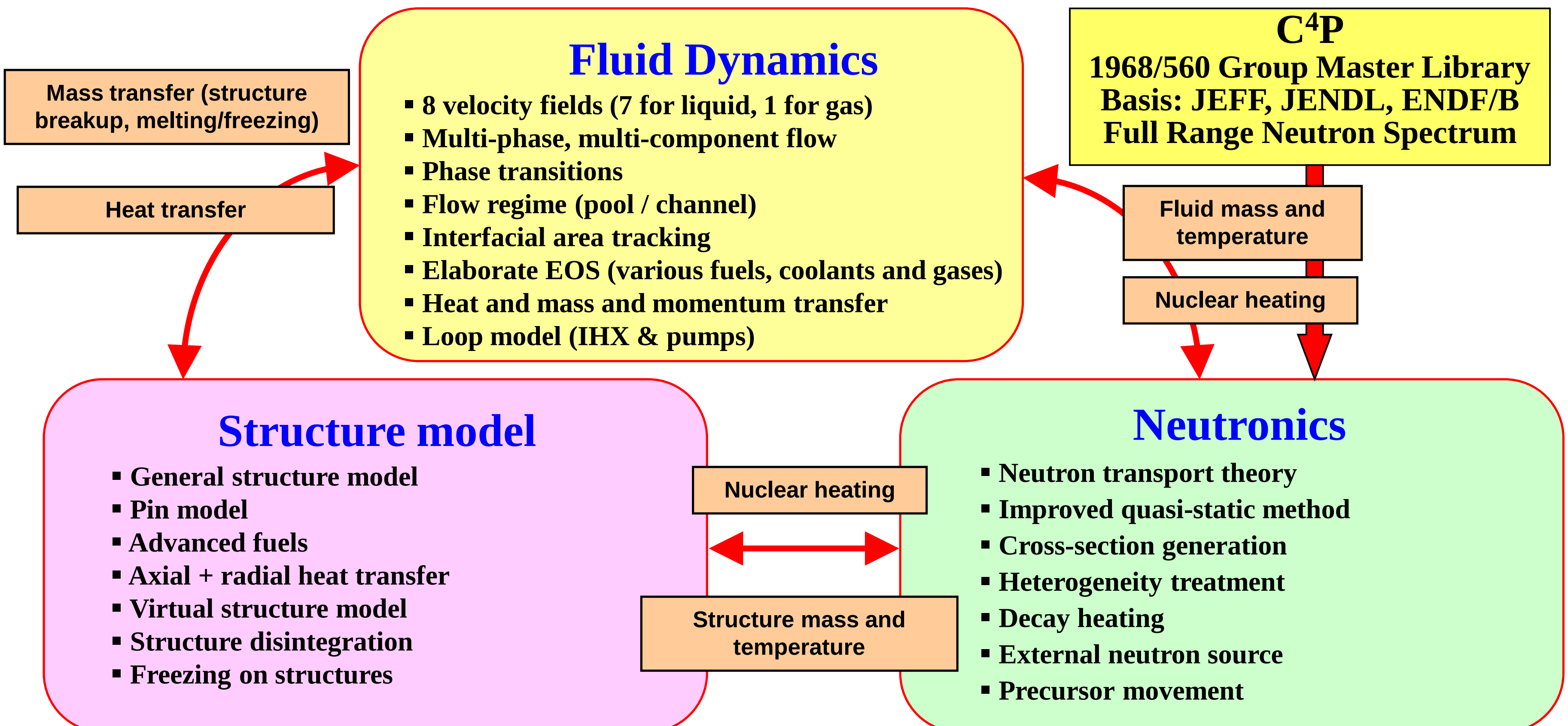


Pressure (bar)



SIMMER: general structure

SIMMER-III and SIMMER-IV codes are 2D and 3D, multi-velocity-field, multi-phase, multi-component, Eulerian, fluid-dynamics code system coupled with a structure model for fuel-pins, hexcans and general structures, and with a space-, angle-, time- and energy-dependent neutron transport model.



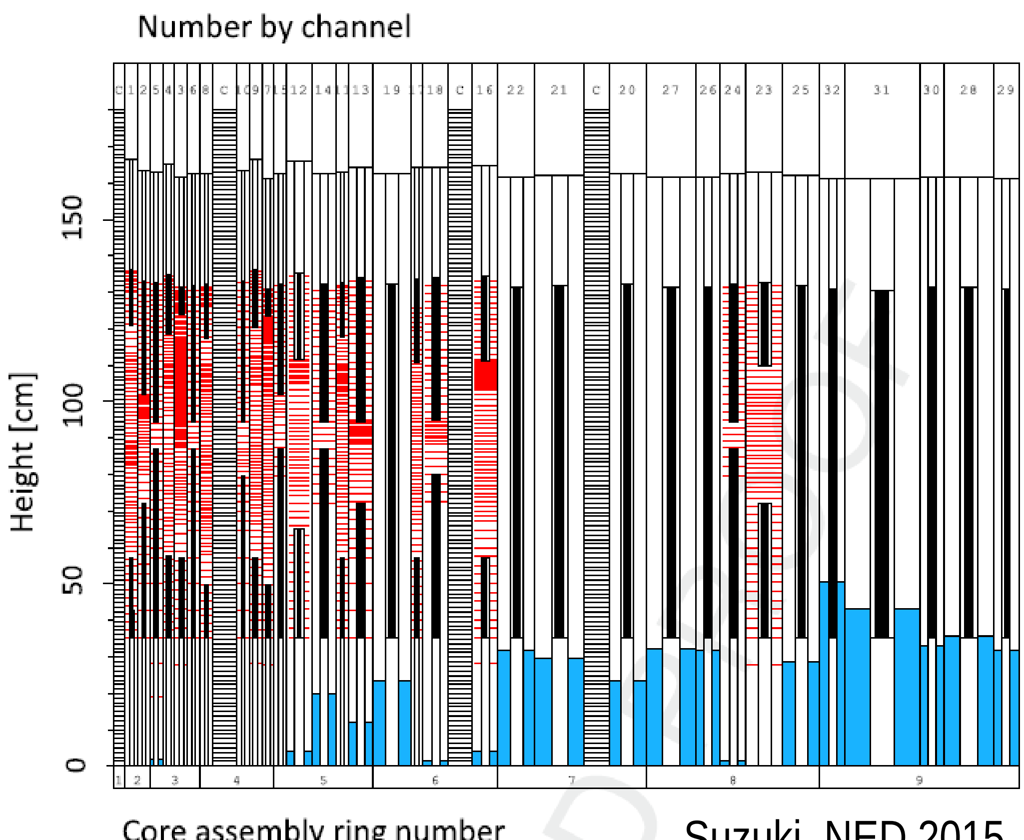
SIMMER: developed by...

SIMMER code family developed by Japan Atomic Energy Agency (JAEA) in cooperation with KIT, CEA and other partners mainly for simulation of hypothetical core disruptive accidents (HCDA) in SFR but also extended for ADS, LFR, GFR and MSR.

SIMMER: nodalization

For SFR pool-type reactor:

- thermal-hydraulic domain can include the whole primary system subdivided in R, Z (2D) or X, Y, Z (3D) directions;
- neutronics domain is a smaller sub-region, including reactor core and its surroundings, with a finer spatial mesh.



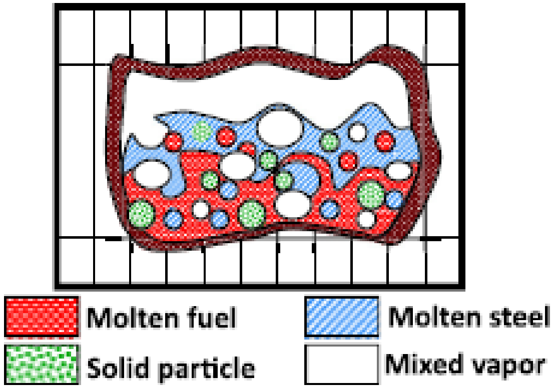
Suzuki, NED 2015

SIMMER: heat and mass transfer

SIMMER simulates five SFR core materials: fuel, steel, coolant, control, and fission gas at different physical states (solid, liquid, gas)

Constitutive models describe intra-cell transfer of mass, momentum and energy at the fluid interfaces including model for convection of interfacial areas based on Ishii's approach.

For each of multiple flow regimes heat and mass transfer includes source terms, momentum exchange functions for each flow regime, inter-cell heat transfer due to conduction, melting and freezing, structure break-up, vaporization and condensation.



SIMMER neutronics

Self-shielded macroscopic cross-sections are calculated from XS library containing infinitely diluted cross sections and Bondarenko (self-shielding) f-factors.

Neutron transport calculations performed by SN neutron transport models, based on the DANTSYS and PARTISN codes.

SIMMER has been extensively validated and verified (V&V) for many applications

Summary:
few examples of knowledge gaps in SFR safety analysis
(*personal opinion*)

1) Thermal mechanics of SFR core and reactor structures

Challenges are:

– 3D modeling of thermal expansion of reactor structures and the impact on reactivity;

– 3D modeling of core mechanics in normal operation;

– 3D modeling of core mechanics in transient conditions including very fast ones (e.g. driven by the pressure pulse caused by sodium vapour bubble collapse);

– ...

2) SFR thermal hydraulics and heat transfer

Challenges are:

- 3D modeling of transition from forced convection to natural convection for the pool-type SFR including fluid dynamics and thermal stratification (Phenix EOL NC test benchmark as an indicator of the modeling uncertainties);
- 3D modeling of Heat transfer and fluid dynamics in Decay Heat Removal systems (reactor pit, cooling of the Steam Generator surfaces, etc.)
- Dynamic modeling of thermal electromagnetic pumps;
- ...

3) Coupled neutronic/thermal-hydraulic analysis of sodium boiling in low-void SFR core

Sodium boiling in ULOF in conventional (e.g. SPX) SFR cores resulted in a positive reactivity insertion, power excursion, melting and relocation of the core materials.

For the newly developed low-void SFR cores (e.g. ASTRID or ESFR) the period between sodium boiling onset and fuel or cladding melting onset becomes longer and the phenomenology of this period is less known:

- there are experimental evidences that under certain conditions the sodium boiling may stabilize (no permanent dryout, no power excursion);
- some calculations show that during the sodium boiling under certain conditions high-amplitude condensation-induced pressure waves occur;
- the impact of these pressure waves on the core geometry and reactivity is to be studied.

4) Severe accident analysis

Modeling of melting, boiling, condensation, solidification as well as relocation and interaction of several core materials are challenging from viewpoint of

- Heat transfer
- Mass transfer
- Neutron transport

There is a need in development and validation of new codes (to complement the legacy SAS and SIMMER families)

Thank you for your attention

Questions?

GIF Webinars (ad):

https://www.gen-4.org/gif/jcms/c_9260/public