Pure Lead as coolant of GEN IV Fast Reactors:
specific issues and comparisons with other coolants

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Seminar on «Fast Reactor Coolants» organized by the French Academy of Science
Paris, 19-20 February 2013

Hystorical introduction on Lead and HLM

The idea of developing fast spectrum reactors with molten lead (or lead alloy) as a coolant, was firstly pursued in the Soviet Union. Research and design of the use of lead-bismuth alloy as the coolant for nuclear reactors was initiated by Academician A. I. Leipunsky at the Institute of Physics and Power Engineering (IPPE) in Obninsk and was actively pursued (1950s through the 1980s) for the specialized role of submarine propulsion.

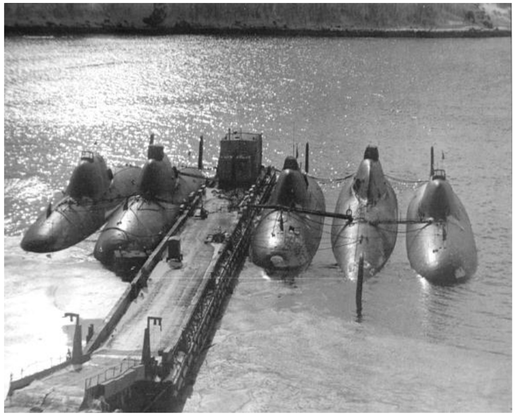
A total of 15 Lead Bismuth Eutectic (LBE) reactors were built:

- 2 submarine prototypes with 2 reactors each;
- 7 "Alpha Class" Submarines (155 MWe);
- 3 land system reactors;
- one replacement reactor for submarines.

At the start of this century, a spallation neutron source (MEGAPIE) based on LBE as coolant and target material was accomplished and operated, in Switzerland, by several Western Nuclear Institutions.

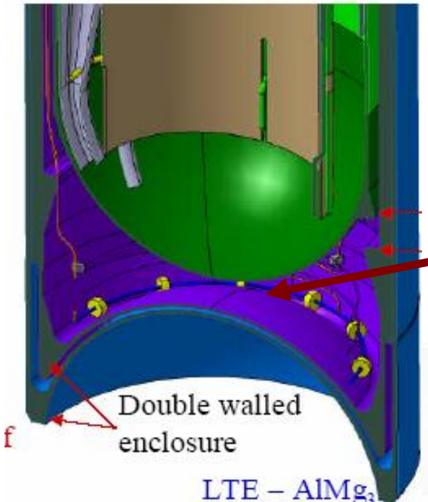
Successively, R&D activities conducted by European Industries and Research Institutes in the frame of the EURATOM 6th and 7th E.U. framework programs evidenced the advantages of pure lead against LBE as reactor coolant

Prototype nuclear submarine Project 645 (1963)



Nuclear submarine-705 serial (1976-1996)

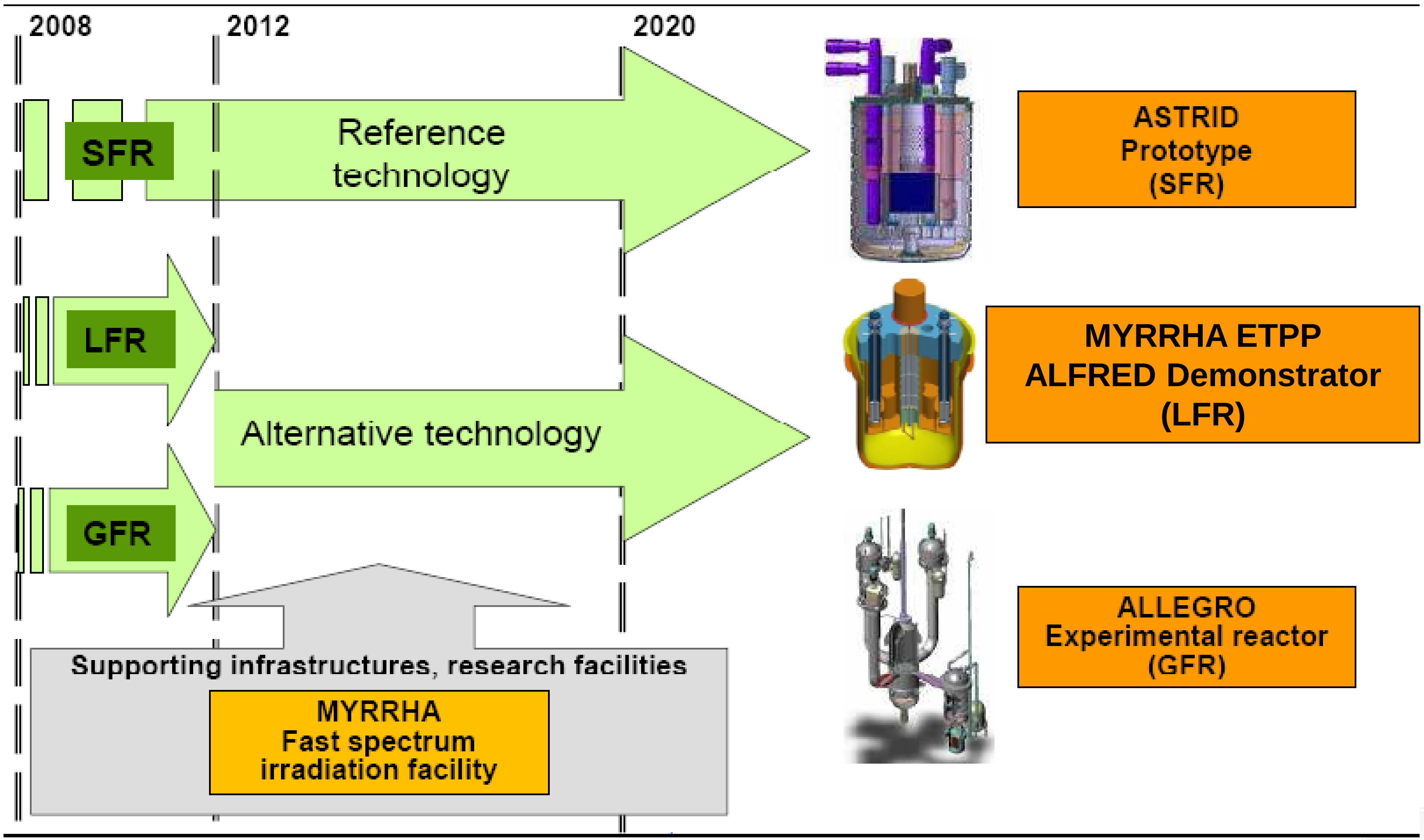
The acquired experience in the Soviet Union amounts to 80 reactor years.



window (T91): Irradiation effect

CNRS - SUBATECH

European strategy for Fast Neutron systems



ESNII Concept Paper, October 2010

Technical aspects of Lead features

Sodium and Lead properties at a glance

	Sodium	Lead
Natural Abundance	High : 2.8% of terrestrial rock	High: highly recyclable
Melting point	97,8°	327°
Boiling point	881,5°	1749°
Exothermic reactions	Yes : water, air, oxygen.	No
Stored potential energy 500° C	Very high: 10 GJ/m3	Very low: 1.09 GJ/m3
Thermal sink capability	Moderate: 0.5 GJ/m3	High: 2.5 GJ/m3
Density (@ 400° C)	Low: 856 Kg/m3	High: 10508 Kg/m3
Thermal conductivity	Very high: 72 W/(m*C)	High: 17 W /(m*C)
Kinematic Viscosity (ν)	Low: 3.3 E-7 m2/s	Low: 2.14 E-7 m2/s
Capability of natural convection: Grashof number	Moderate: Gr= 1.93 E13 (d=1 m and ΔT=1 K)	High: Gr= 2.47 E14 (d=1 m and ΔT=1 K)
Radio-activation	limited activation ^{22}Na: 2.6 years, ^{24}Na: 15 hours, no alfa emitters	^{204}Pb (only 1.4%) is alfa emitter Activated Po from Bi impurities
Neutron moderating power ($\xi\Sigma s$)	Low: 0.0176 barns	Very Low:0.00284 barns
Wetting capability (ISI)	Very good	Good (above 400° C)
Compatibility with steels	High	Low (above 450 ° C)

Stored potential energy

Toshinsky ICAPP 2011

The total potential energy (chemical, thermal and mechanical) that is stored per cubic meter of coolant is a **major indicator to assess the risk to spread the radionuclides out of the reactor in case of accident**.

Coolant	Water	Sodium	Lead, Lead-bismuth
Parameters	P = 16 MPa T = 300 ºC	T = 500 ºC	T = 500 ºC
Maximal potential energy, GJ/m^3, *including:*	~ 21,9	~ 10	~ 1,09
Thermal energy / *including* compression potential energy	~ 0,90 / ~ 0,15	~ 0,6 / None	~ 1,09 / None
Potential chemical energy of interaction	With zirconium ~ 11,4	With water 5,1 With air 9,3	None
Potential chemical energy of interaction of released hydrogen with air	~ 9,6	~ 4,3	None

Grashof number

The Grashoff number is an indicator of the capability of the fluid to circulate under the regime of thermal (or natural) convection.

It is the ratio between buoyancy force and friction force

$$Gr = (g\,\rho\, d^3\,(\Delta\rho/\rho))/(\mu/\rho)^2 = (g\, d^3\,\beta\,\Delta T)/\nu^2$$

g	acceleration of gravity
ρ	Density
d	characteristic dimension
μ	dynamic viscosity
ν	cinematic viscosity
β	thermal expansion coefficient

When d =1m and DT = 1 K, then :

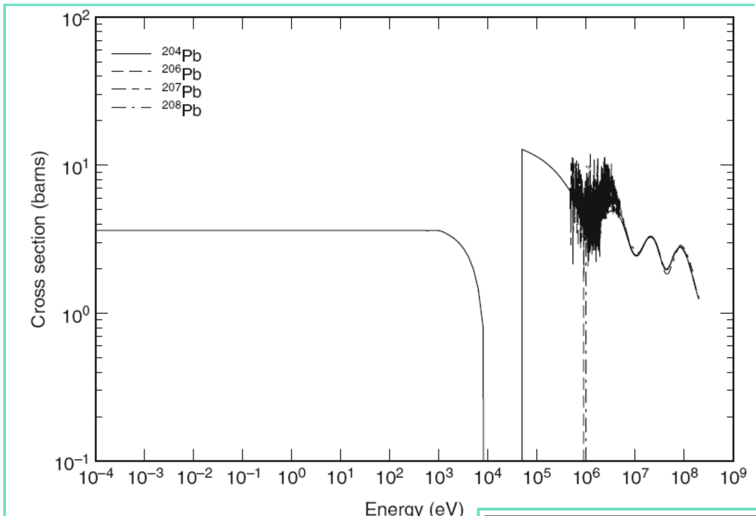
Grashof for sodium is **1.93 E13** and for lead is **2.47 E14**

Very low neutron moderating power

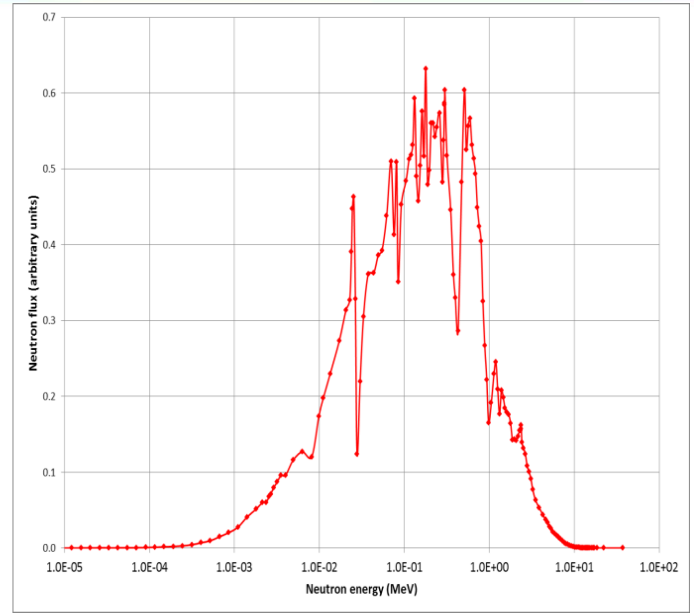
Average Lethargy change per elastic collision (ξ) is low due to the high atomic mass number of Pb. A= 207.2

$$\xi = 1 - \frac{(A-1)^2}{2A} \ln\left(\frac{A+1}{A-1}\right)$$

$$\xi = \left\langle \ln\left(\frac{E'}{E}\right) \right\rangle$$



Elastic scattering cross section (Σs) for Lead



Typical neutron spectrum in Fast Reactor core

The **limited neutron moderating power** by Lead allows a hard energy spectrum in the core. As a consequence **the distance among fuel pins increases with respect to sodium**.

This implies:

- **Large flow-rate at low velocity**
- **Low pressure drop**
- **Low core outlet temperature**
- **Effective natural convection**
- **Power density of 110 MW/m3 (comparable with PWR and lower than SFR)**

Average lethargy (logarithmic energy loss) change per elastic collision and moderating power for some typical coolants/moderators

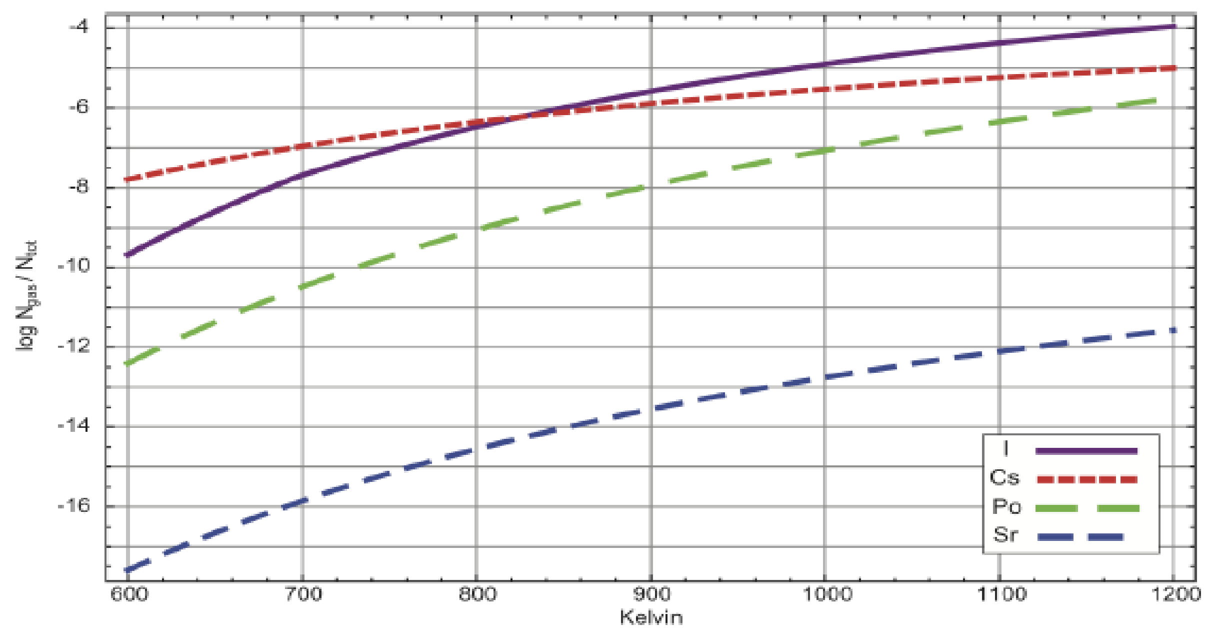
	ξ	$\xi\Sigma_s$
H_2O	0.920	1.425
D_2O	0.509	0.177
Helium	0.425	9.0e–6
Graphite	0.158	0.083
Sodium	0.0825	0.0176
Lead	0.00963	0.00284

Capability of HLM to retain fission gases

Fraction of volatilization of lead-soluble components.

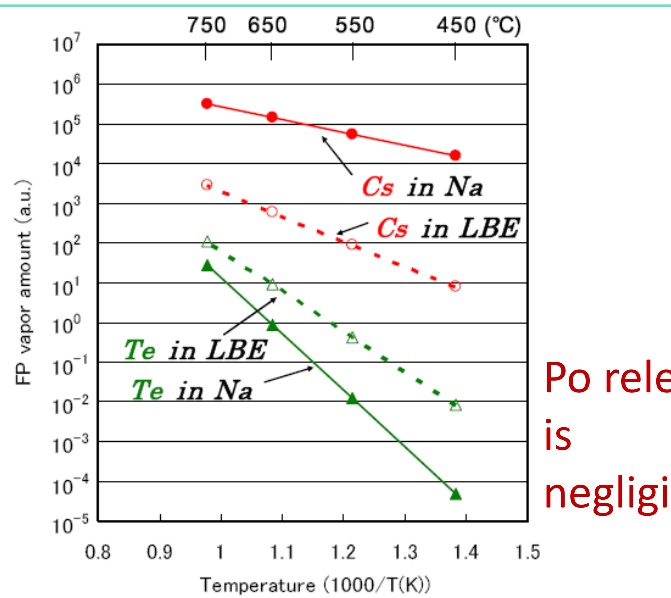
Tritium and noble gases are released to 100% .
I, Cs, Po and Sr are retained in Pb

Comparison between sodium and LBE cases for release of Cs and Te



Contribution of selected elements to radioactivity in 80 m³ volume.

Element	Specific radioactivity over element	Volatilised fraction 480 °C	Volatilised fraction 800 °C	Volatilised radioactivity, 480 °C	Volatilised radioactivity, 800 °C
Kr	$9.55 \cdot 10^{11}$ Bq/g	All Kr	All Kr	$8.13 \cdot 10^{13}$ Bq	$8.13 \cdot 10^{13}$ Bq
Xe	$2.90 \cdot 10^{13}$ Bq/g	All Xe	All Xe	$5.11 \cdot 10^{16}$ Bq	$5.11 \cdot 10^{16}$ Bq
I	$2.82 \cdot 10^{14}$ Bq/g	$9.51 \cdot 10^{-8}$	$2.99 \cdot 10^{-5}$	$3.41 \cdot 10^{9}$ Bq	$1.07 \cdot 10^{12}$ Bq
Cs	$2.06 \cdot 10^{12}$ Bq/g	$2.41 \cdot 10^{-7}$	$4.91 \cdot 10^{-6}$	$7.75 \cdot 10^{8}$ Bq	$1.58 \cdot 10^{10}$ Bq
Sr	$3.87 \cdot 10^{13}$ Bq/g	$7.53 \cdot 10^{-16}$	$5.34 \cdot 10^{-13}$	5 Bq	$3.71 \cdot 10^{3}$ Bq
Isotope					
^{3}H	$3.57 \cdot 10^{14}$ Bq/g †)	All ^{3}H	All ^{3}H	$3.23 \cdot 10^{13}$ Bq	$3.23 \cdot 10^{13}$ Bq
^{210}Po	$1.66 \cdot 10^{14}$ Bq/g †)	$2.09 \cdot 10^{-10}$	$2.96 \cdot 10^{-7}$	982 Bq	$1.39 \cdot 10^{6}$ Bq
Sum				$5.12 \cdot 10^{16}$ Bq	$5.12 \cdot 10^{16}$ Bq



Comparison of fission product vapor amount between Na and LBE Pool

Po release is negligible

Reactor design features

Lead drawbacks

Technological development and design provisions are necessary to overcome or mitigate the impact of the drawbacks.

Lead drawbacks	High Density	Opacity	High Melting Point	Corrosion of structural materials

Difficulties For:	Mechanical Design	Refueling	Material technology

With classical design solutions, refueling is particularly difficult because:

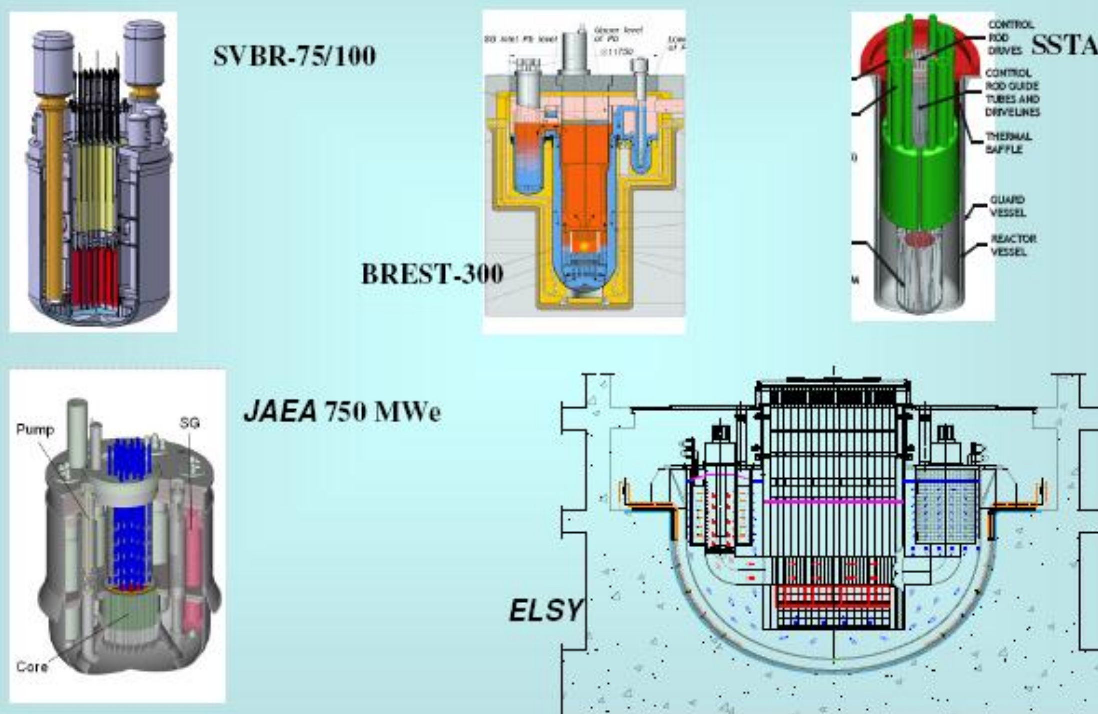
- **fuel elements must be fixed to the support grid to compensate buoyancy;**
- **opacity prevents vision;**
- **refueling must be performed at about 400 °C because of the high melting point;**
- **it is difficult to maintain the passive oxide layer on the sliding parts of the steels.**

Guidelines for system design

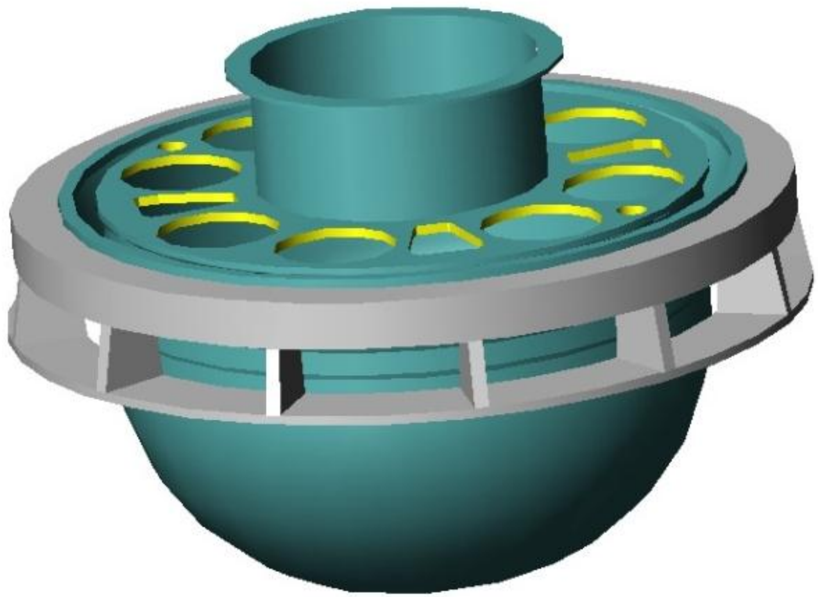
1. **The density of Lead is twelve times that of sodium. This property affects structural integrity: a particular concern for earthquake conditions.**
2. **Speed of Lead must be limited to about 2m/s to reduce erosion of the structural material**
3. **The absence of reaction with SG water makes useless the intermediate heat exchanger**

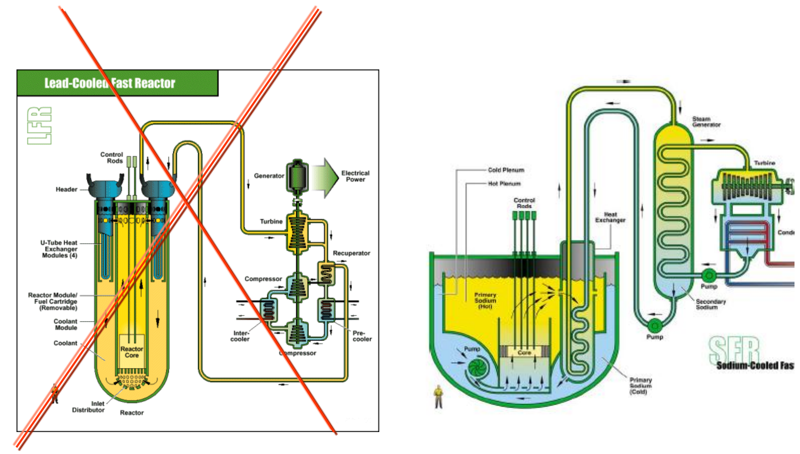
- **As a result, the LFR plant size is not more than medium scale even with adoption of 3D seismic isolation.**
- **Furthermore, a loop-type cooling system cannot be selected because of the of the difficulty in the design of piping supports.**
- **The pool type is without intermediate heat exchanger**
- **Economic appeal of the design can be only achieved by compact design having acceptable efficiency**

All LFR designs are characterized by the absence of the intermediate loop and by a pool configuration.



Compact design 1





Because of the lead high density a LFR must have a shorter vessel than a SFR!

The result is a reactor with a compact short-height vessel (~ 9 m) resistant to seismic loads and no intermediate loops.

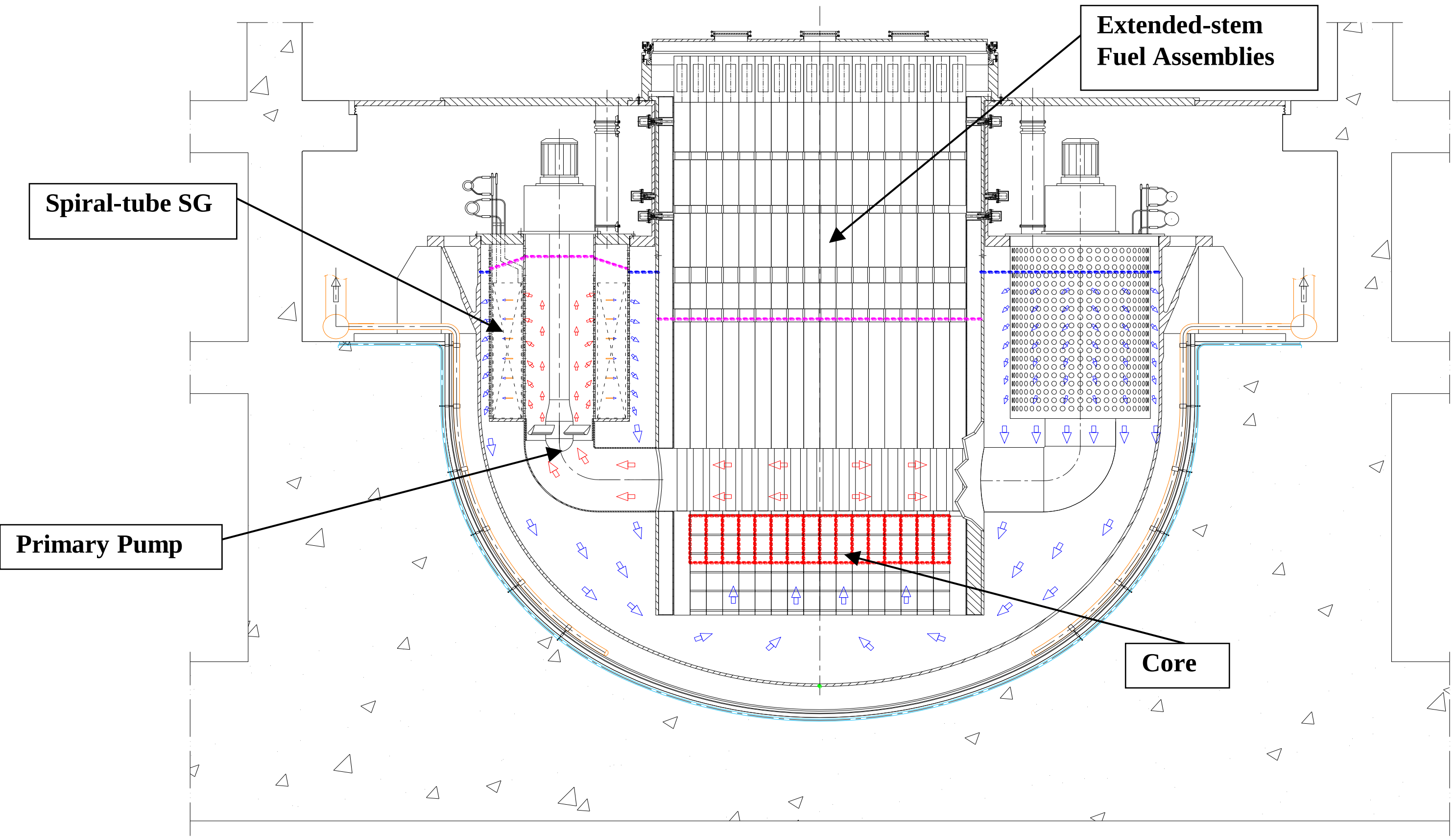
Preliminary mechanical analyses confirm the feasibility of a compact 600MWe reactor.

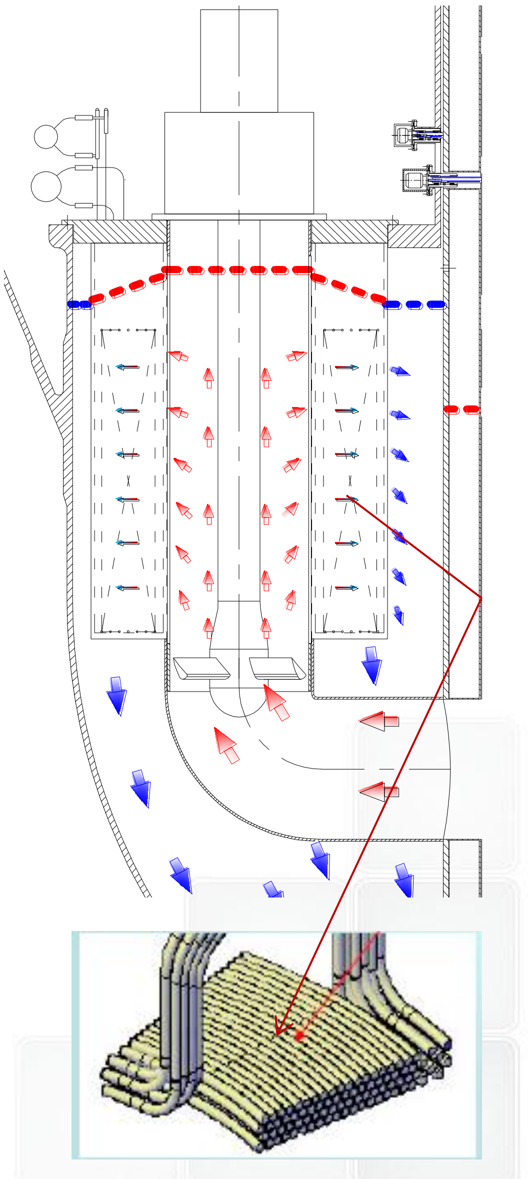
ELSY	m^3/MWe	~1,5
SFR, pool-type	m^3/MWe	~ 2

Compact design 2

The compact design is achieved by 2 innovations

1. Mechanical extension, out of Lead, of the fuel assemblies and subsequent elimination of:
   - «inside vessel» refueling machine
   - «inside vessel» fuel storage
2. Integration of Steam Generator and Pump based on the following:
   - Plane spiral SG
   - Pump in hot coolant



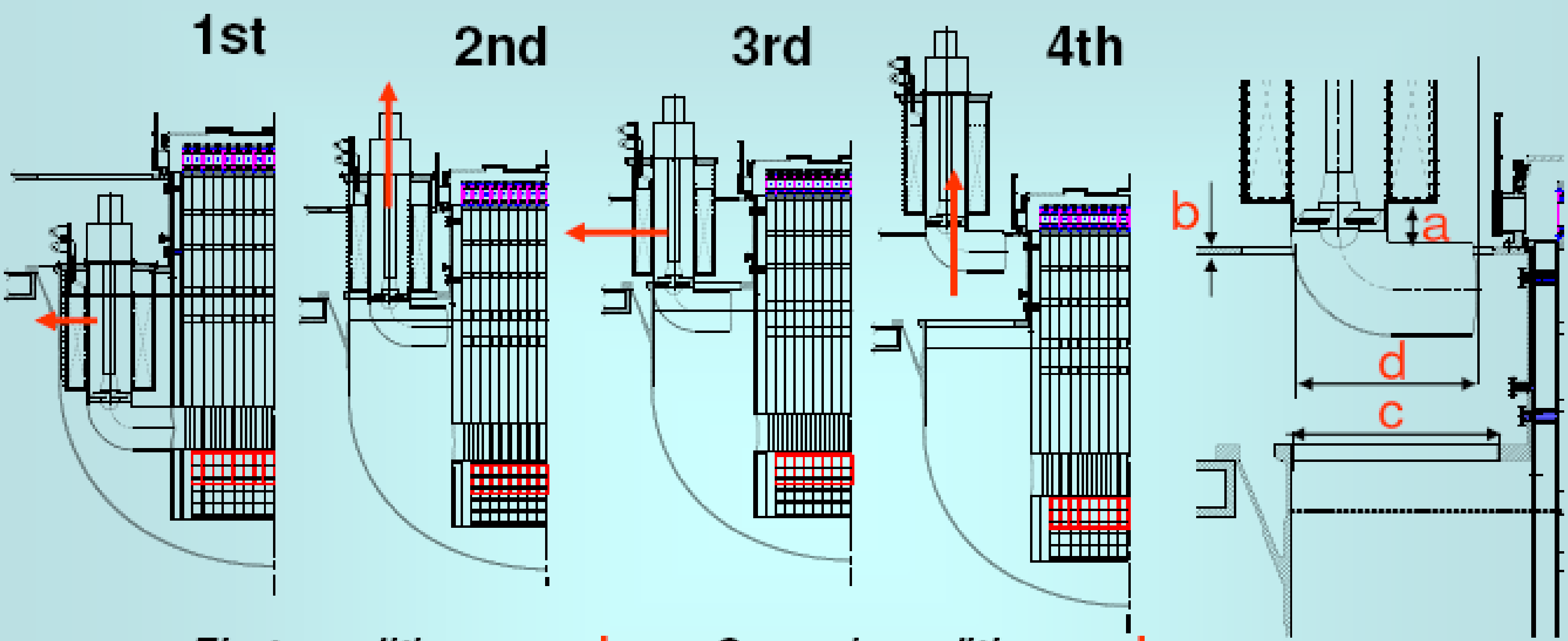


Removable internals for ISI&R

Drawback : Lead opacity and high melting point make difficult the repair operation in lead

Proposed solution : Design replaceable components

The Pump- Steam Generator Assembly for a LFR with removable internals



First condition: a>b, *Second condition:* c>d

All in-vessel components are removable, including the Cylindrical Inner Vessel and the SG that can be disengaged from the Inner Vessel by a combination of radial and vertical displacements (investment protection).

Refueling and ISI of Fuel Assemblies

<u>Drawback</u> : Lead opacity and high melting point make difficult the refuelling operation and FA ISI in lead

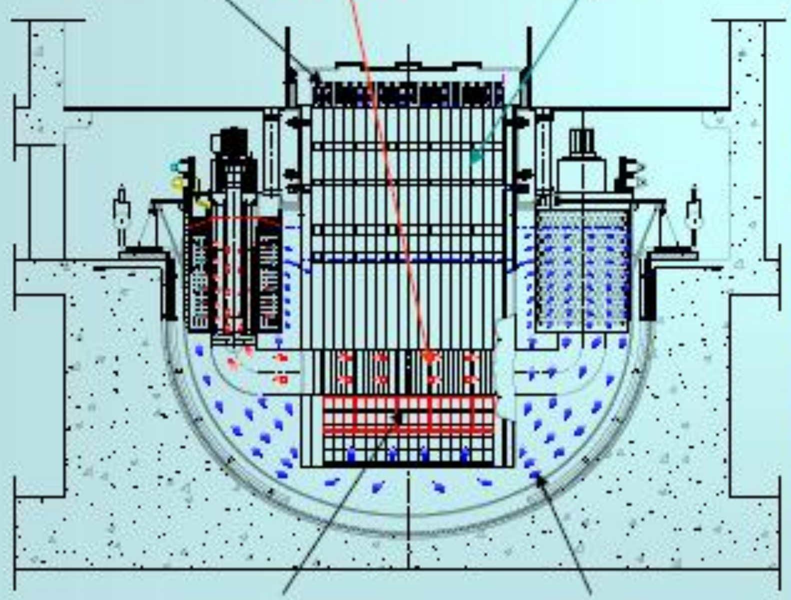
<u>Proposed solution</u> : Design refueling machine operating in gas

FA are sustained by buoyancy, kept in vertical position by structures and inspected

The Fuel Assemblies are extended above the lead free level

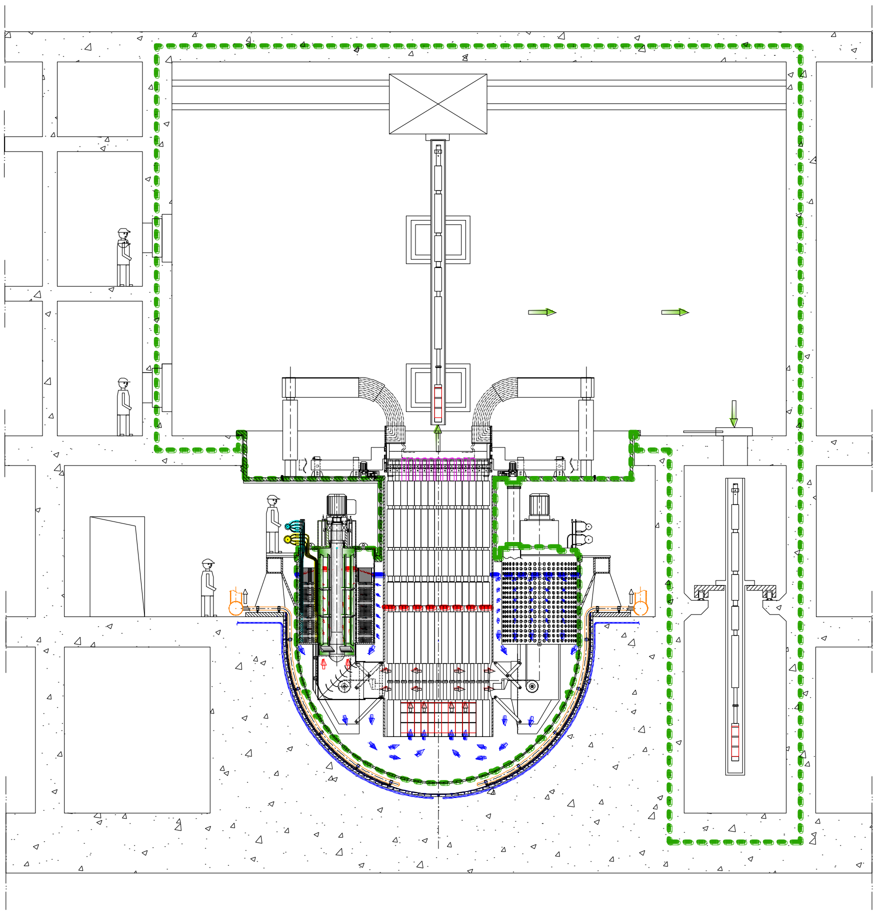
- ✓ The Refueling machine operates in gas.

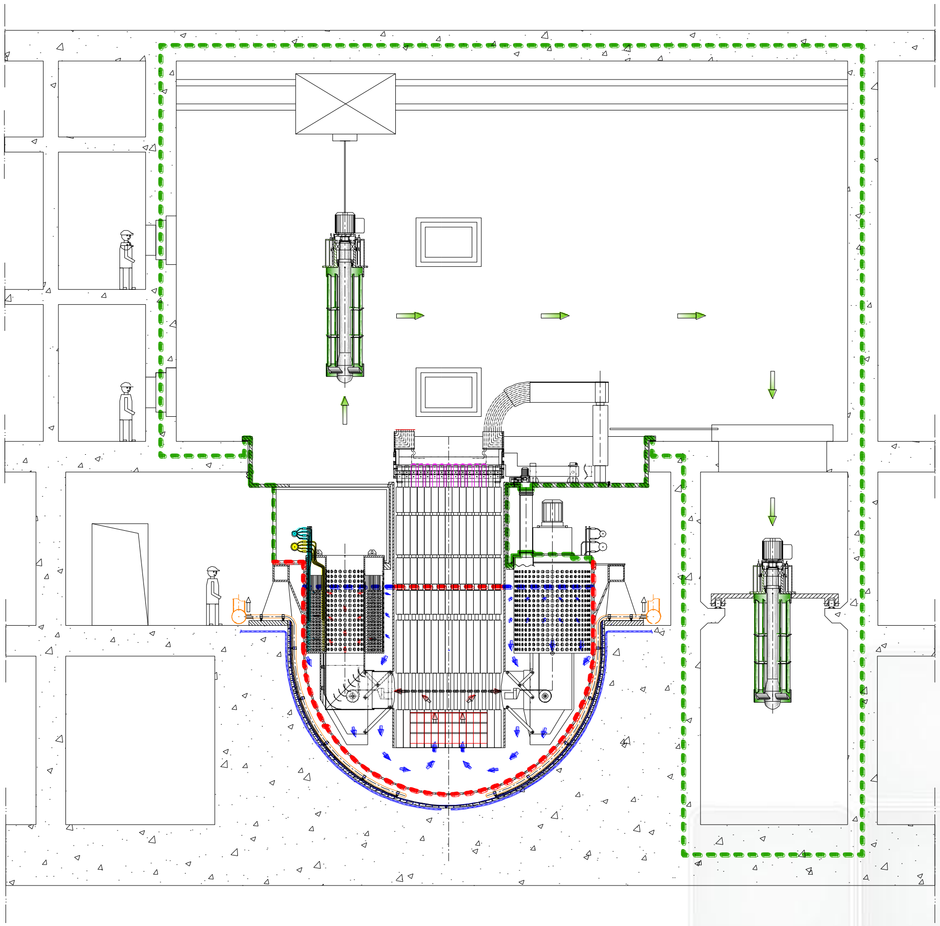
  Refuelling is difficult in sodium which is opaque, would be more difficult in lead which is opaque, has a high melting point and high density
- ✓ FA are sustained by buoyancy, kept in the vertical position by a support system in gas.
- ✓ Reactor structures which constitute the vertical support of the FA can be inspected from outside in gas space.



ISI&R Approach

AGENZIA NAZIONALE PER LE NUOVE TECNOLOGIE, L'ENERGIA E LO SVILUPPO ECONOMICO SOSTENIBILE

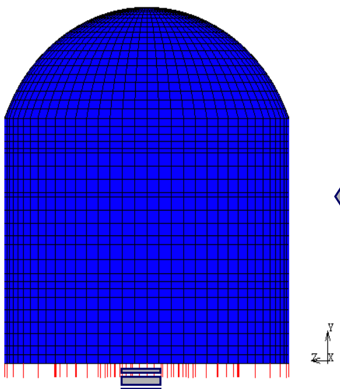


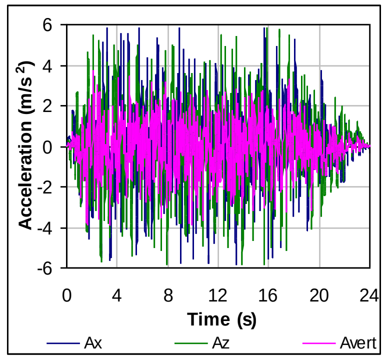


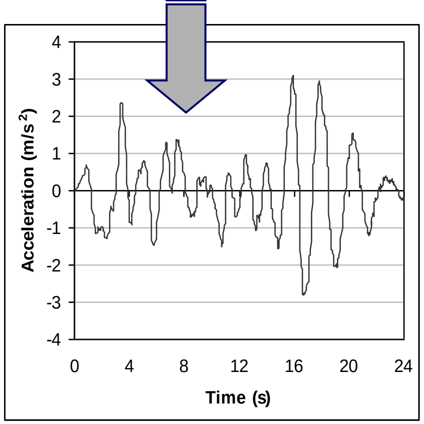
Seismic loads

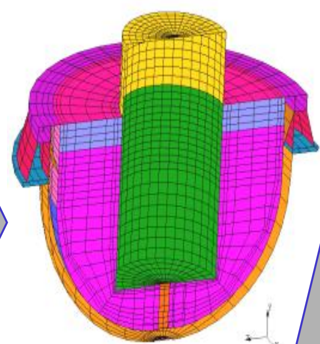
Seismic loads represent the limiting factor to the LFR size. 3D seismic isolators are foreseen in ELSY building fundations

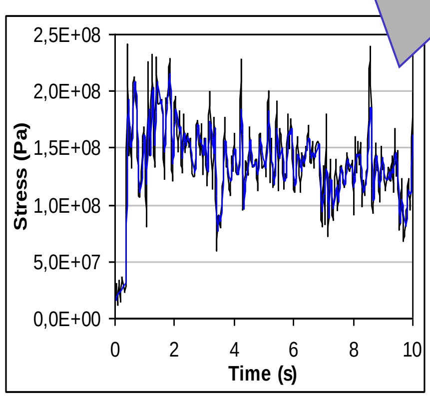
Inside the vessel liquid lead is present. During an earthquake loading, the liquid can start to slosh. The influence of sloshing onto the internals has been analysed. The stress in the steam generator housing at the junction with the vessel cover has been calculated. Based on a transient analysis the stress intensity is 120MPa with a safety factor of 0.76.

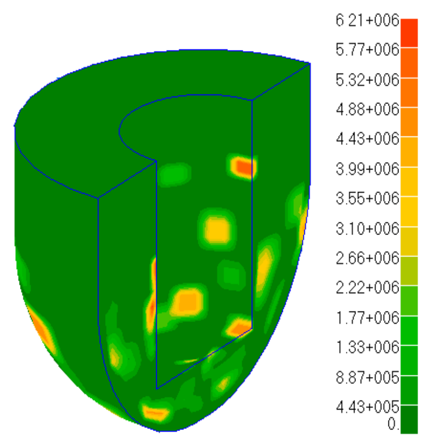












Max Von Mises stresses

Considering that for the large European interconnected electrical grid there is a limited interest towards small-size reactors, the 600 MWe size was chosen since the beginning of the project as the minimum size of interest in the European context, 1000 MWe remaining the final target for a reactor fleet of lead cooled systems

Guidelines for core design

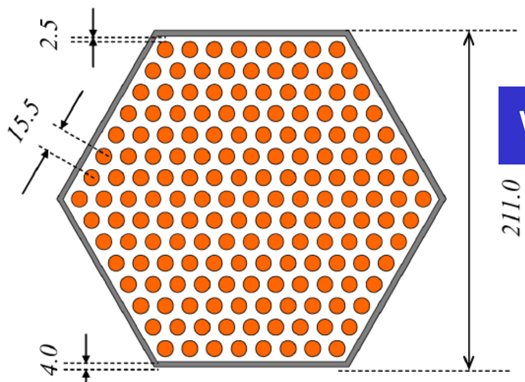
1. Lead is corrosive , therefore the outlet coolant temperature must be low (< 500° C)
2. Lead is erosive and has high density, therefore the coolant velocity must be low (< 2 m/s)
3. Lead has high melting point, therefore the inlet coolant temperature must be high enough (>327° C)
4. Lead has very low moderating power, therefore P/D ratio of fuel pins can be higher than in sodium
5. Lead has high boiling point (1749 C) , therefore the risk of voids in the core, due to lead boiling, is null.

Consequences

- Typical P/D ratio in Lead is 1.34 to 1.4 while in ASTRID is around 1.19
- The large coolant cross section, with respect to the solid cross section, allows a high flow rate even at low speed (1 m/s)
- The high flow rate, together with the high heat capacity, allows a low ΔT: Tinlet=400° , Toutlet=480° .

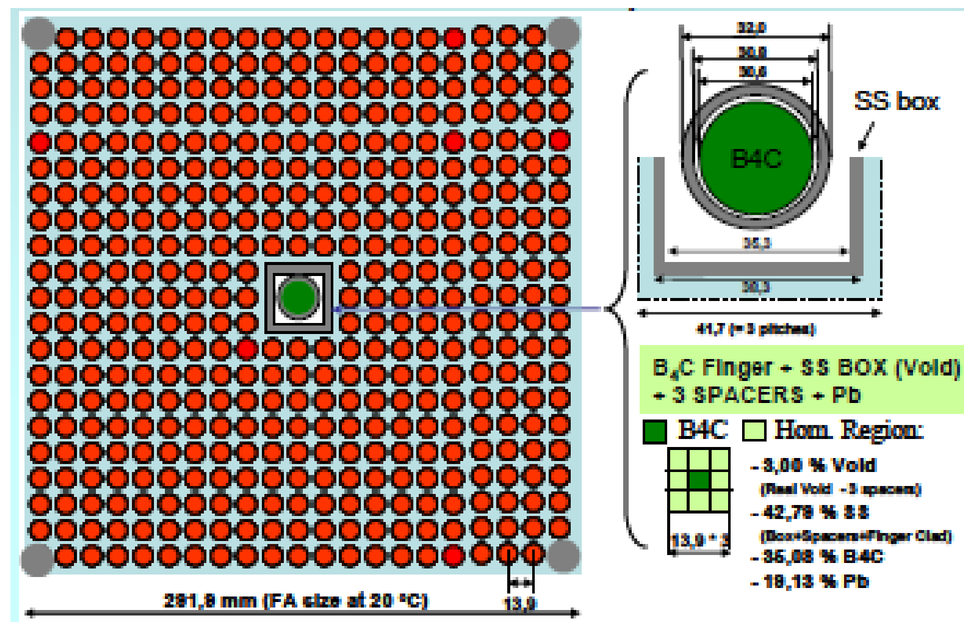
LFR core design

Two options for core configurations: Wrapped option and Wrapperless



Wrapped option

N° fuel pins / FA	169
Fuel column height (mm)	1200
N° Fuel Assemblies	162
Pellet diameter (mm)	9,1
Fuel rod outer diameter (mm)	10.6
Fuel rod pitch (mm)	15
Clad thickness (mm)	0,6
Thickness wrapper (mm)	4
Gap between FA's (mm)	5



Wrapperless option

N° fuel pins / FA	428
Fuel column height (mm)	900
N° Fuel Assemblies	162
Fuel rod outer diameter (mm)	10,5
Fuel rod pitch (at 20° C) (mm)	13,9
Clad thickness (mm)	0,6

The limited neutron capture by Lead allows a larger coolant fraction in the core.

This implies:

- **Large flow-rate at low velocity**
- **Low pressure drop**
- **Low core outlet temperature**
- **Effective natural convection**

Main parameters of ELSY design

	ELSY
Power, MWe	600
Thermal efficiency %	42
Primary coolant	Pure lead
Primary coolant circulation (at power)	Forced
Primary coolant circulation for DHR	Natural
Core inlet temperature (°C)	400
Core outlet temperature (°C)	480
Fuel	MOX with and without MA
Neutron spectrum	Fast
Fuel pin diameter, (mm)	10.5
Fuel cladding temperature (max) °C	~ 550
Active core dimensions Height/ equivalent diameter, (m)	0.9/4.32
Fuel column height. (mm)	900
N° Fuel Assemblies (FA)	162
FA geometry	Open (wrapperless)
FA pitch, (mm) .	294
N° fuel pins / FA	428
Fuel pin pitch at 20° C, (mm)	13,9 square
Enrichment, (%wt HM)	14.54/17.63/20.61 Pu, three radial zones
Power conversion system	Water-superheated steam at 18 MPa, 450° C
Primary/secondary heat transfer system	Eight SGs

Safety Features

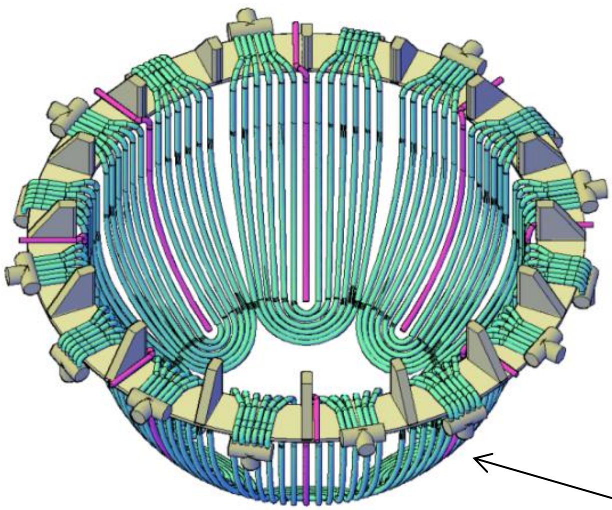
Main LFR safety features resulting from the Lead features

- Water and air are used as cooling fluids for the DHR loops of ELSY
- Enhanced natural circulation even without pumps
- Fuel dispersion in case of core meltdown
- Other safety issues

The Reactor Vessel Air-Cooling System (RVACS)

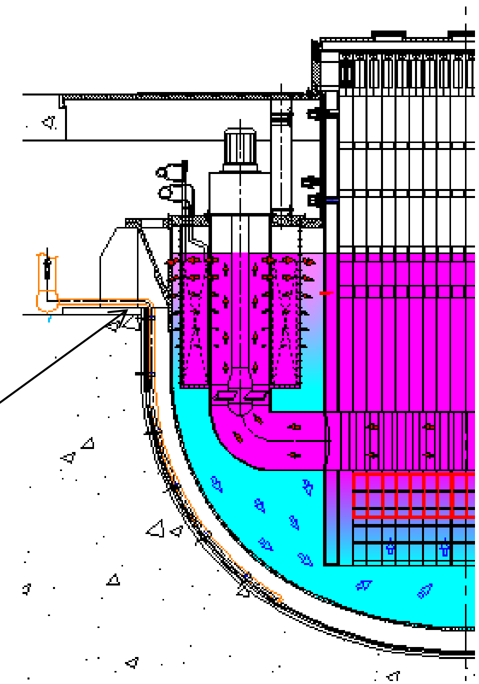
The RVACS is a decay heat remover that continuously dissipates to the external air the heat radiating from the reactor vessel.

The decay heat is rejected to the ultimate heat sink over the following path: fuel assemblies → primary coolant → main reactor vessel → RVACS → external atmosphere



Advantages:

- ➢ Passive initiation
- ➢ Passive operation
- ➢ Simplicity
- ➢ Reliability

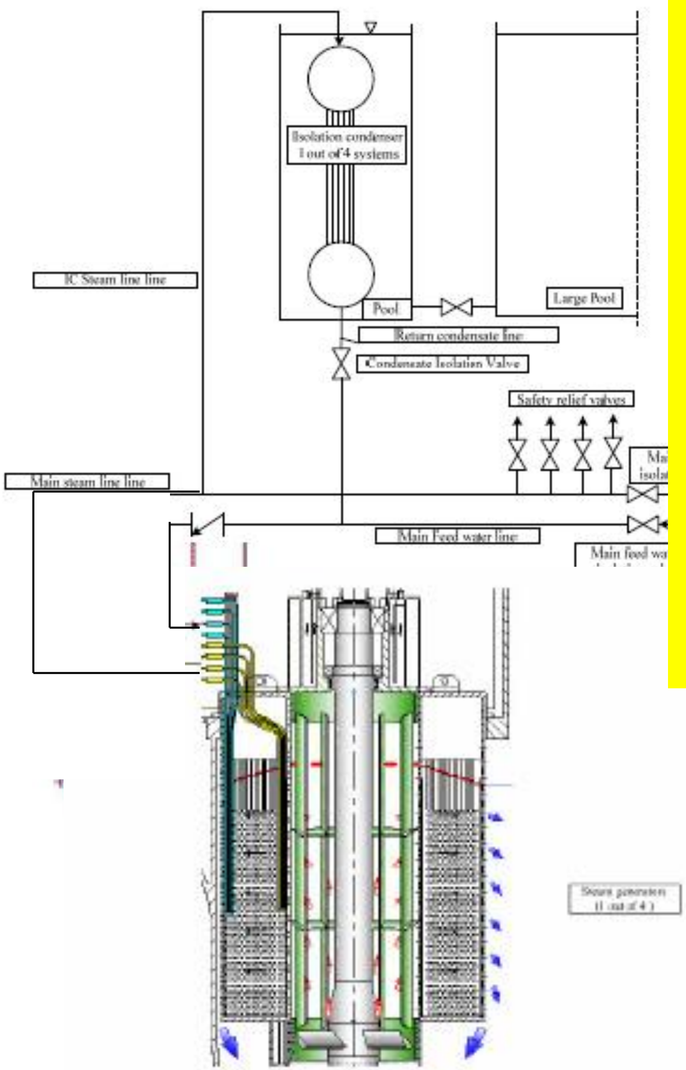


One reactor vessel air cooling loop (2 MW)

Drawback:

- ➢ Too low power for a large plant (2 MW at T-vessel=430° is effective only after 1 month since the shutdown)
- ➢Performances affected by atmospheric conditions (temperature, wind speed)
- ➢ Large diameter piping outside the Reactor Building

DHR2: Condensers on the main steam lines



- Four out of eight steam generators are connected with isolation condensers and discharge a total power of 30MWth. This is the decay power after 24 minutes after shutdown
- Each Isolation Condenser loop comprehends:
  - Heat exchanger (Isolation Condenser), constituted by a vertical tube bundle with an upper and lower header
  - Water pool, where the isolation condenser is immersed (the amount of water contained in the pool is sufficient to guarantee 3 days of operation)
  - Condensate isolation valve

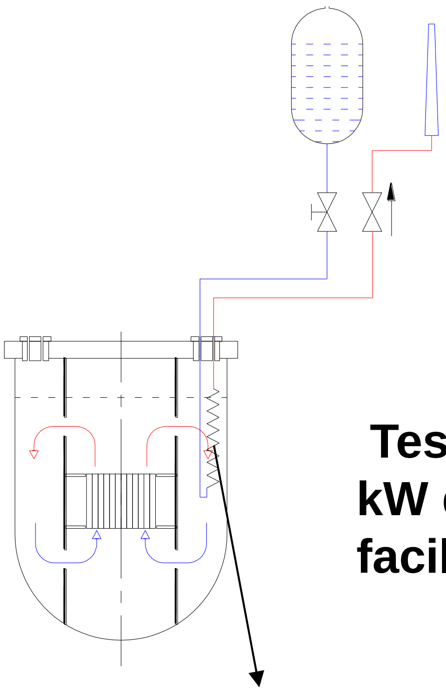
Advantages:

- ➢ **Passive operation**
- ➢ **No impact on the Reactor Block size**

Drawbacks:

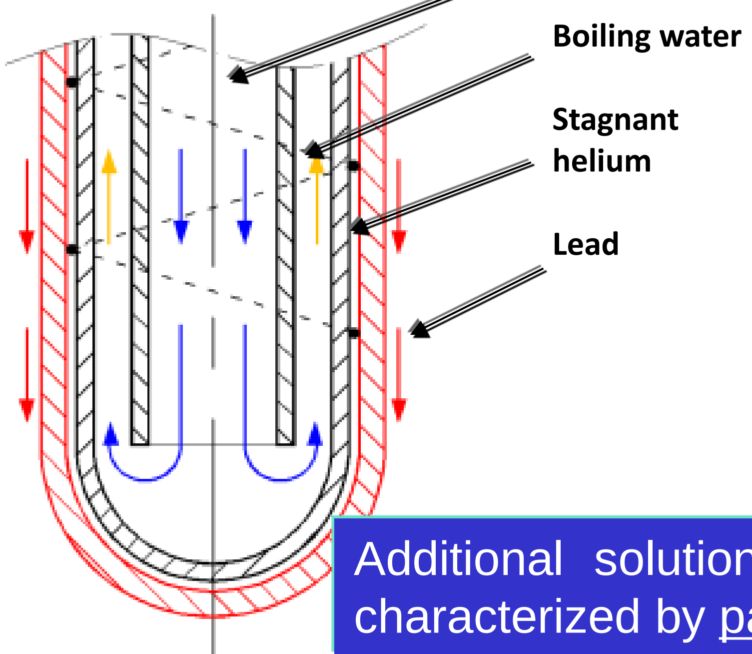
- ➢ **The secondary loops are complicate,**
- ➢ **Risk of flow instability inside the steam generator** **(high thermal load associated to high pressure)**
- ➢ **Need of fast discrimination of the failed SG in case of steam generator line break**

DHR1: Water Direct Reactor Cooling (W-DRC) system



The system is made of 4 loops which globally remove 30 MW.
Each W-DRC loop is mainly constituted by a cooling water storage tank, a water-lead Dip Cooler, interconnecting piping, and steam vent piping to discharge steam into the atmosphere.
Dip-Cooler with Double-wall, helium-bonded-outer-tube bayonet tubes with continuously monitored double barrier between primary system and outside.

Tests are ongoing on a 800 kW dip cooler at the ENEA facility of Brasimone).



Advantages:

- **Passive operation**
- **Simplicity**
- **Reliability**
- **Low cost**
- **High physical protection** **(only a small diameter vent piping outside the Reactor Building)**

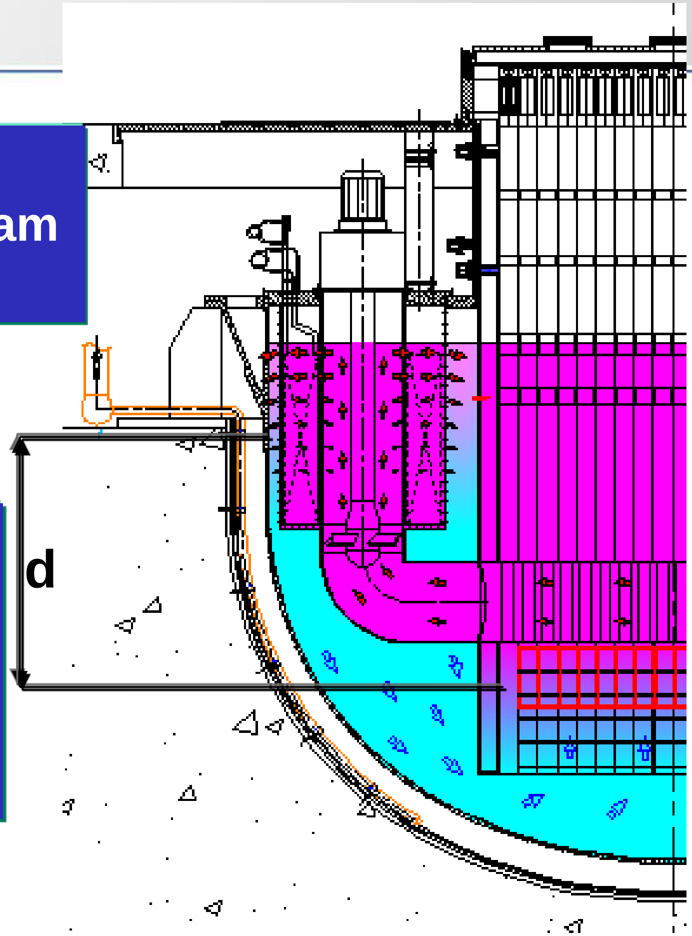
Drawbacks:

- **High water consumption if operated with low steam title**
- **Risk of flow instability if operated with high steam title.**
  (Not yet assessed if operation under unstable condition is mechanically acceptable.

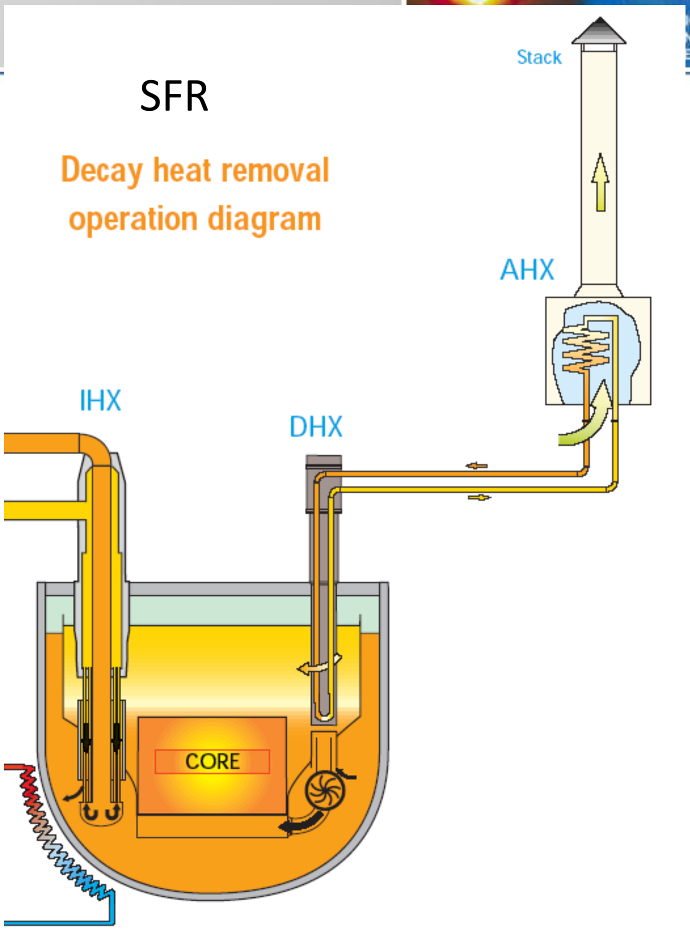
Additional solutions (not yet published), based on air or water as coolant characterized by passive initiation and passive operation. are under investigation

The spiral-tube SG for lead natural circulation

The case of LOF and loss of the water-steam loops



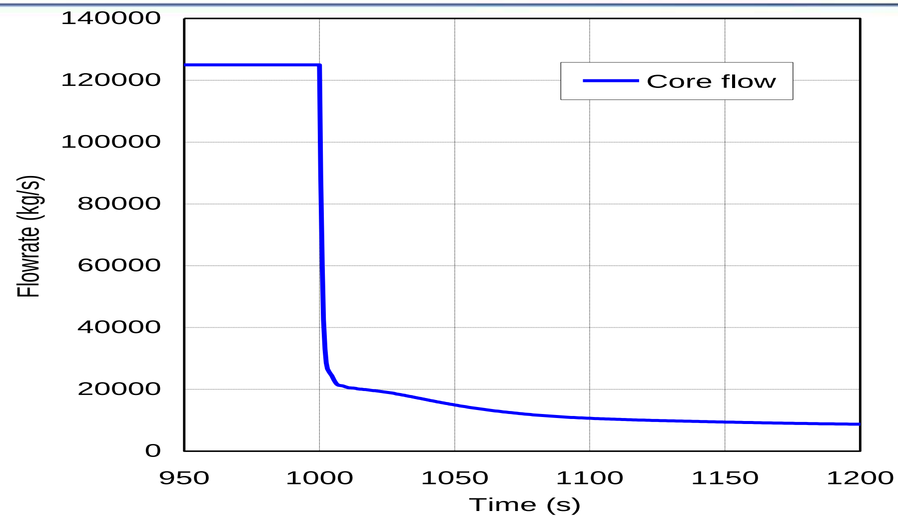
Buoyancy is created because of the large distance "d" between core mid plane and the level at outlet of SG (about the level of half height of the SG bundle).

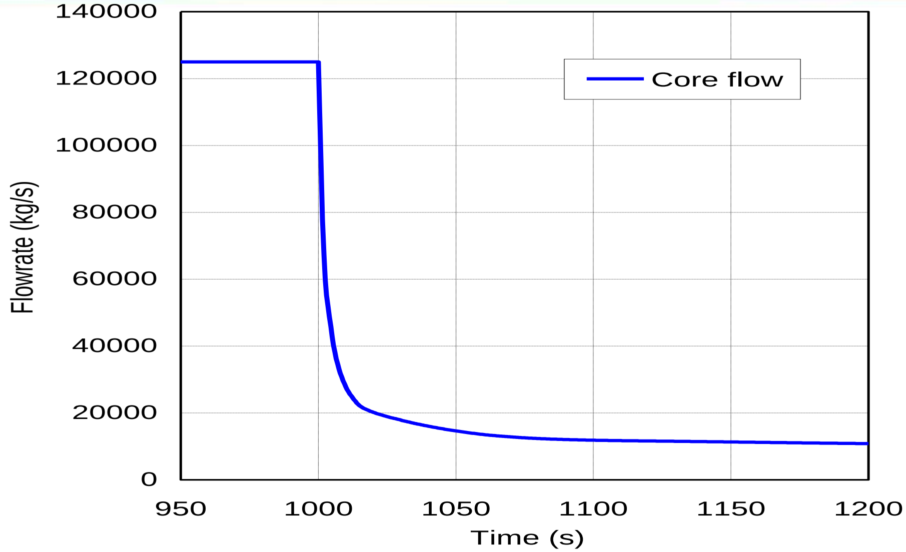


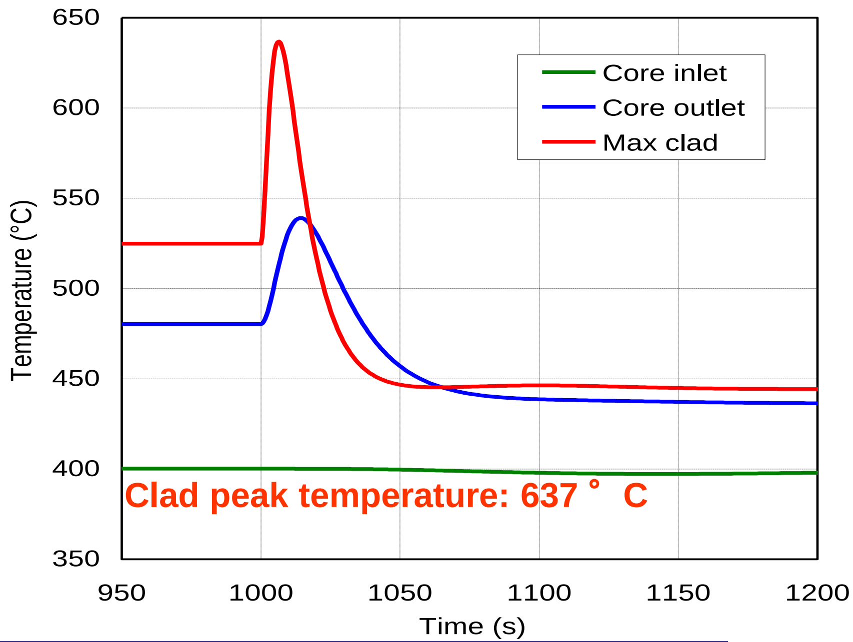
The spiral-tube SG outlet is positioned well above the core mid level.

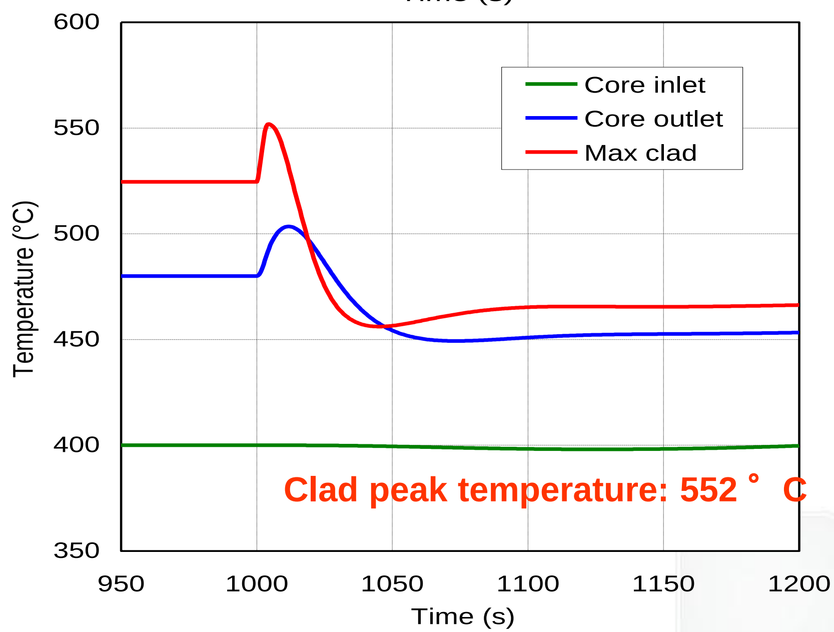
→ **lead natural circulation is possible in case of decay heat removal trough RVACS or dip coolers and water-steam loops unavailable even in a primary system with cylindrical inner vessel configuration.**

Protected Station Blackout (PLOF+PLOH)







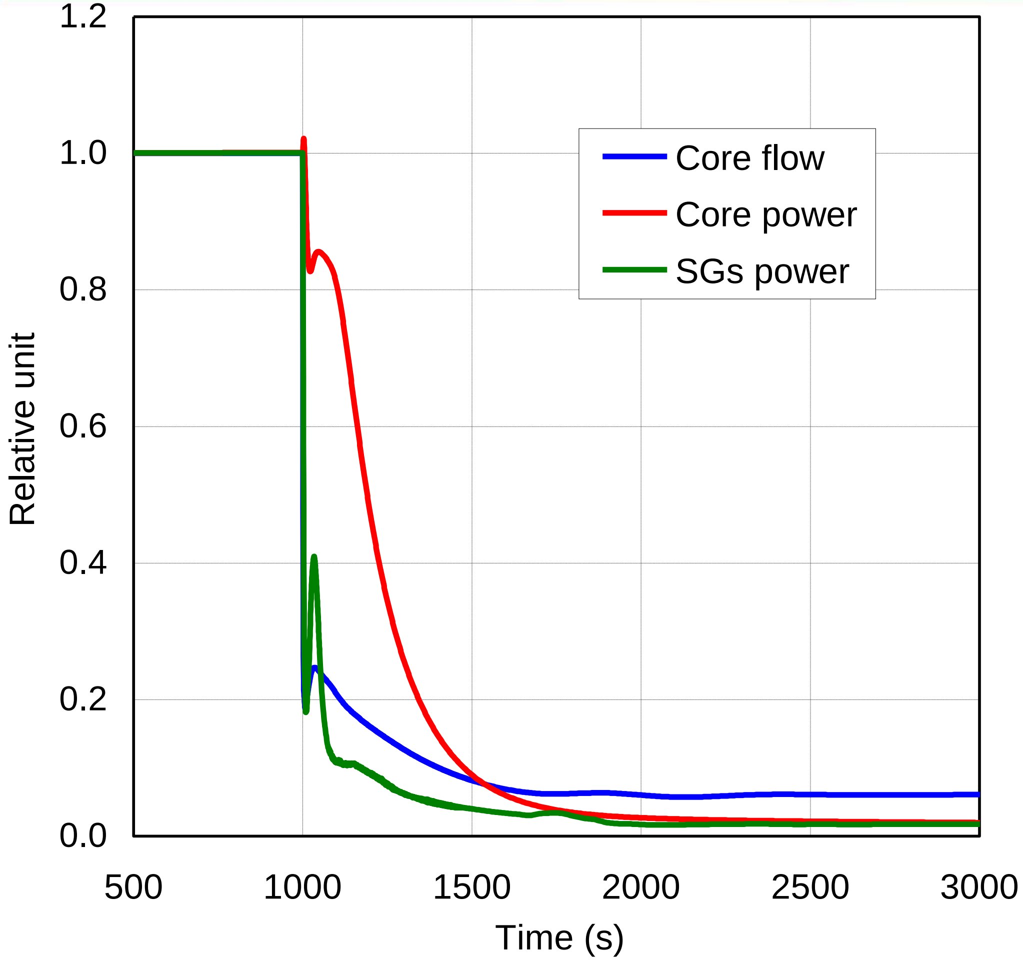


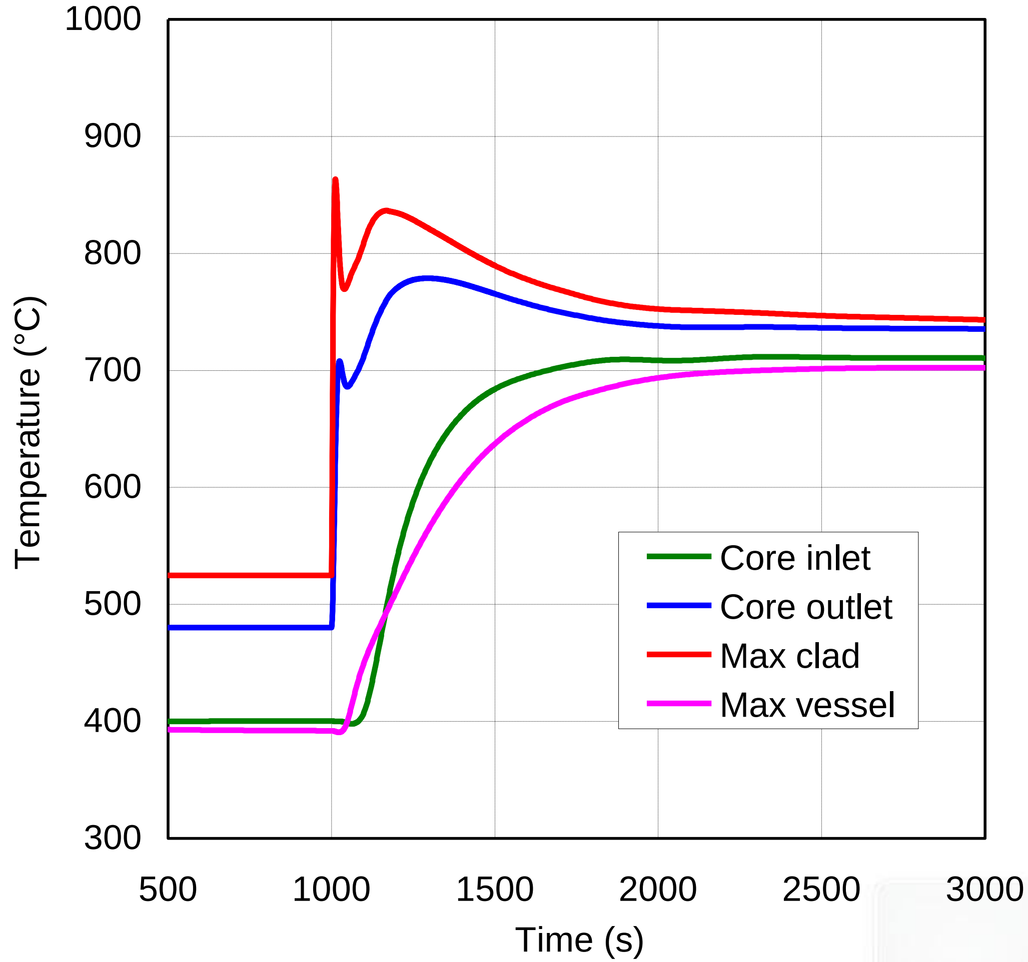
Pump inertia = 864 kg-m2.

Pump inertia = 3000 kg-m2 and t(1/2ω)=2,1 s

Flowrate in the primary circuit stabilizes at 6% of nominal value

ELSY-Unprotected Station Blackout (ULOF+ULOH)





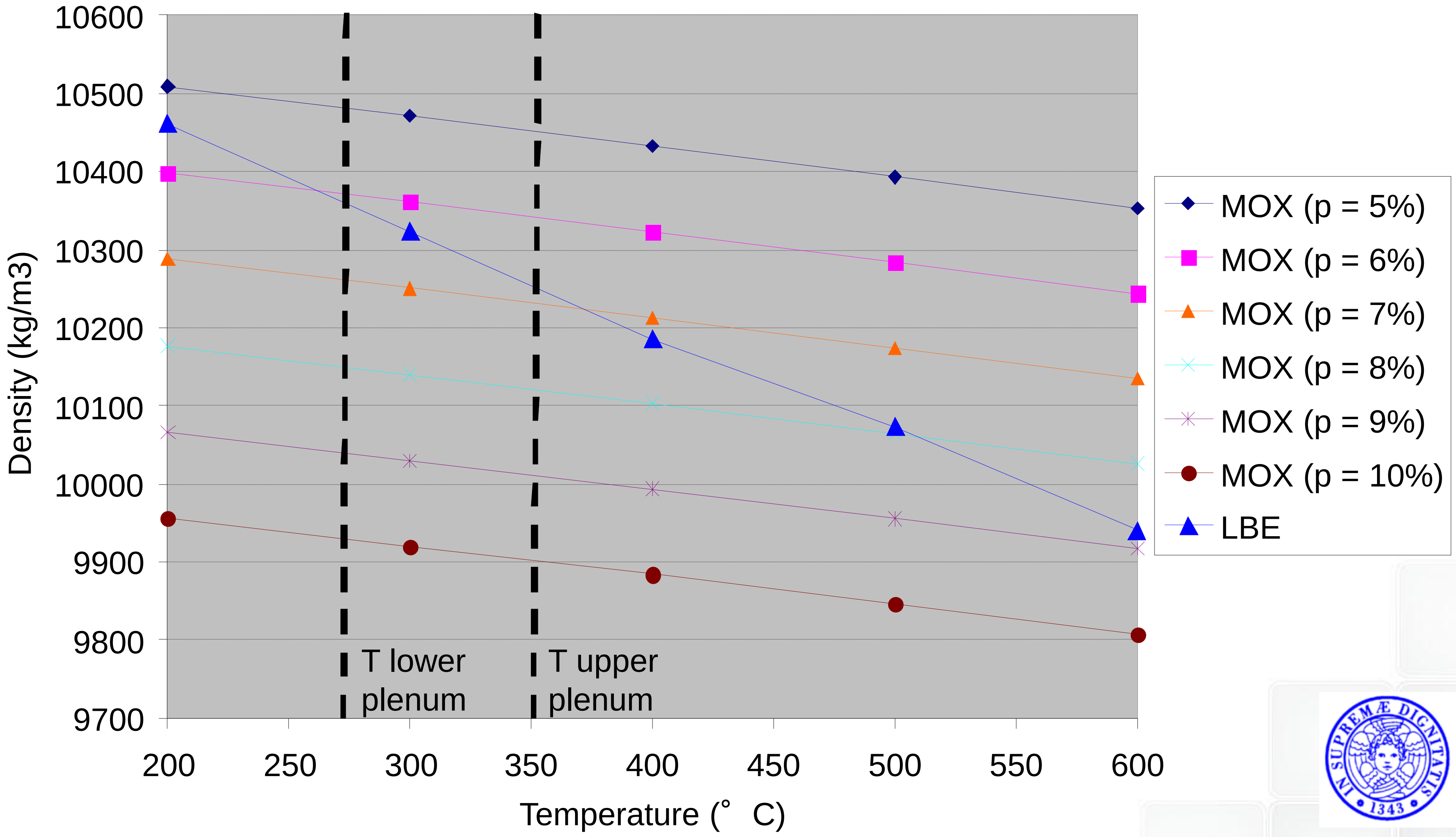
The continuous increase of core inlet temperature and relative negative feedback reduces the nuclear power to decay level at about t = 2000 s.

The maximum clad temperature slowly reduces down to 720 ° C at t = 6000 s.

The maximum vessel wall temperature of 702 ° C is reached around t = 3000 s.

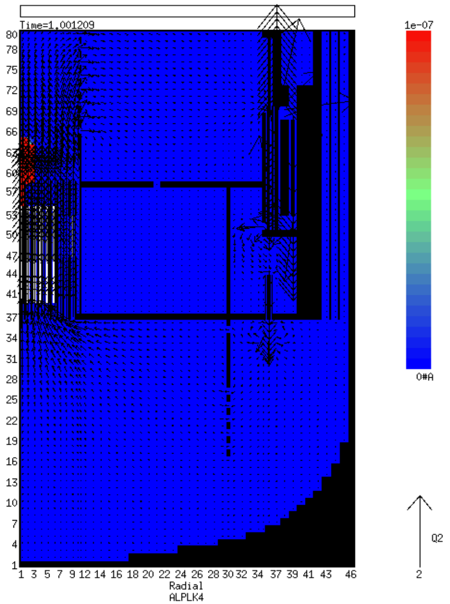
Fuel Dispersion: a safety effect of high density

MOX (30%Pu) and LBE densities

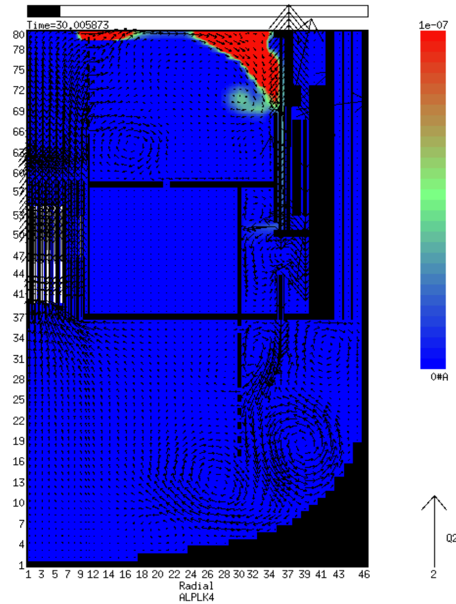


Fuel dispersion in primary system:
Fuel porosity = 10%

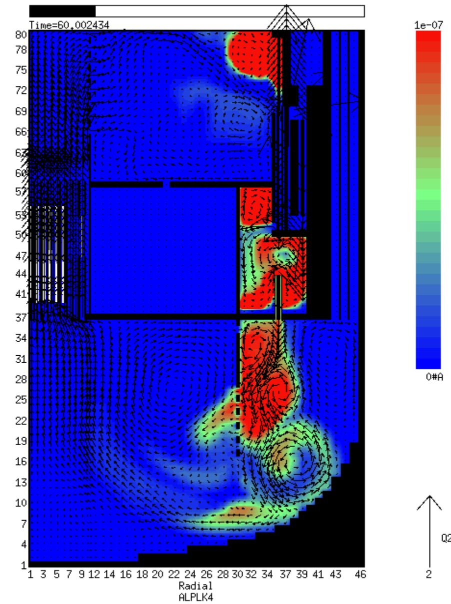
Time = 1 s



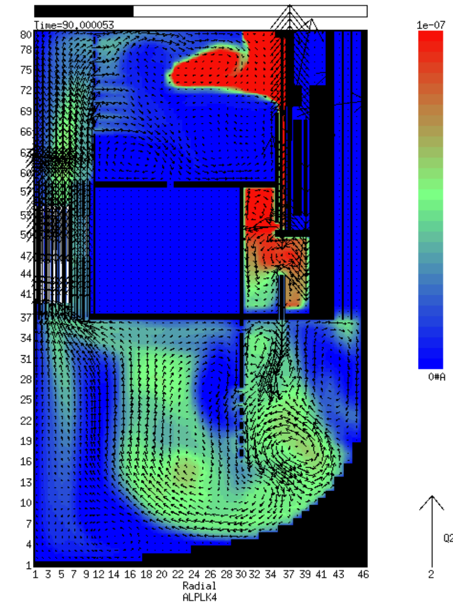
Time = 30 s



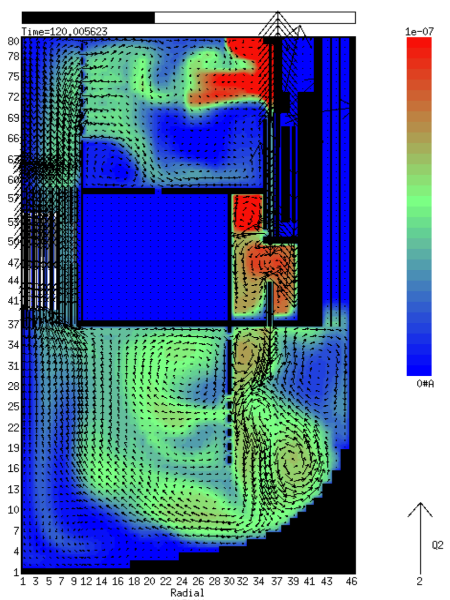
Time = 60 s



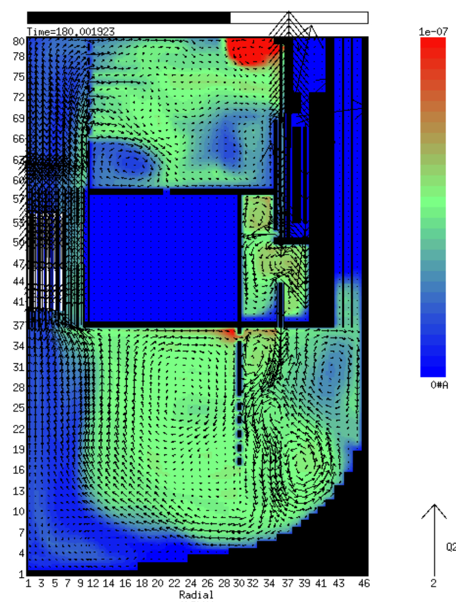
Time = 90 s



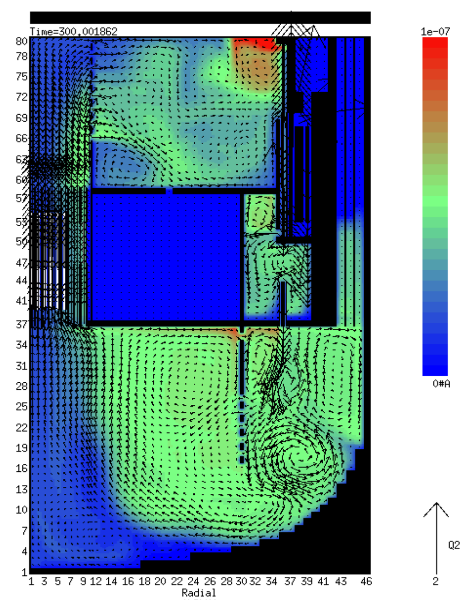
Time = 120 s



Time = 180 s



Time = 300 s



The fuel particles density is slightly lower than the LBE density → fuel particles tends to float towards upward or free surfaces in the long term

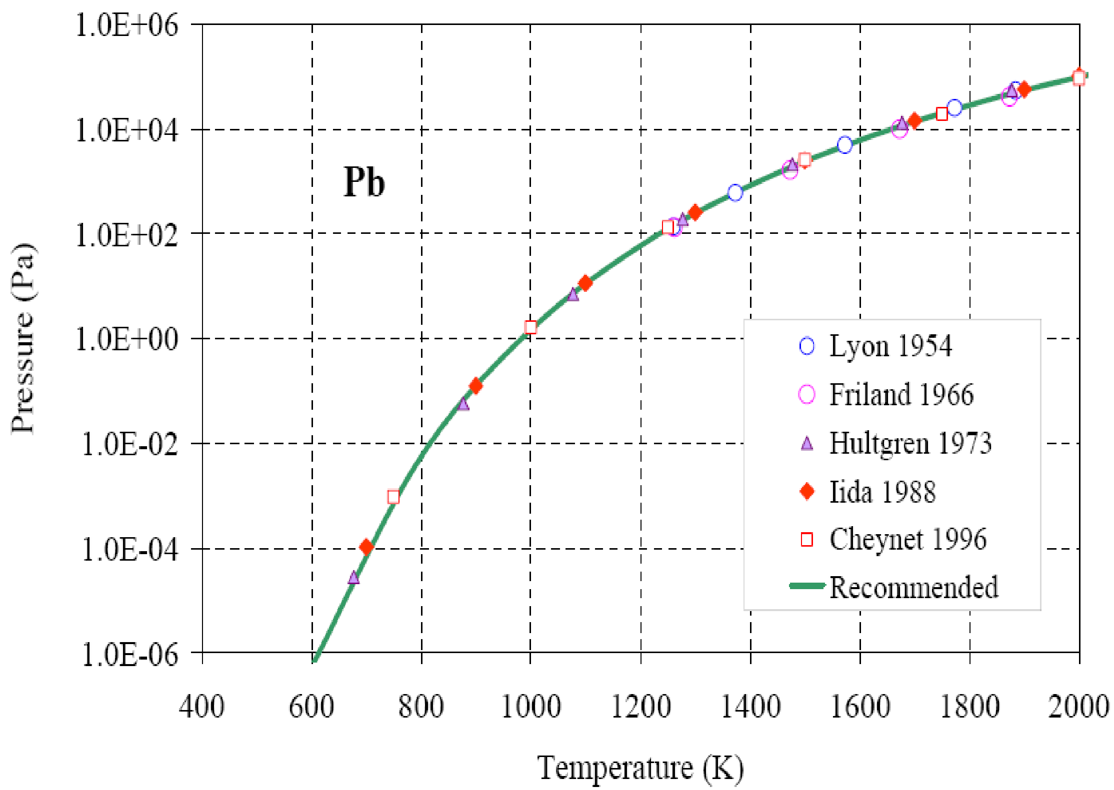
Accidental release of Pb vapour

Lead is highly toxic, nevertheless its low vapour pressure prevents the release of Pb vapours

Example

- The cover gas volume of ELSY reactor is 28 m^3
- At 500° (773K) the saturated vapour pressure is 0.002 Pa and the fraction of Lead in the cover gas is 2 mg
- The concentration is 67 $\mu g/m^3$
- This value is less than the maximum allowed concentration of lead in air in Italy: 150 $\mu g/m^3$

Saturated vapour pressure of molten lead versus temperature



Flow Blockage

The flow blockage accident in a Fuel Assembly (FA) of a nuclear reactor consists in **a partial or total occlusion of the flow passage area.**

Two different effects can be distinguished:

✓A *local effect*, if the blockage occurs inside, or near, the active part of the FA, due to the stagnation-recirculation/wake region downstream the blockage, with a local minimum heat transfer and a clad temperature peak;

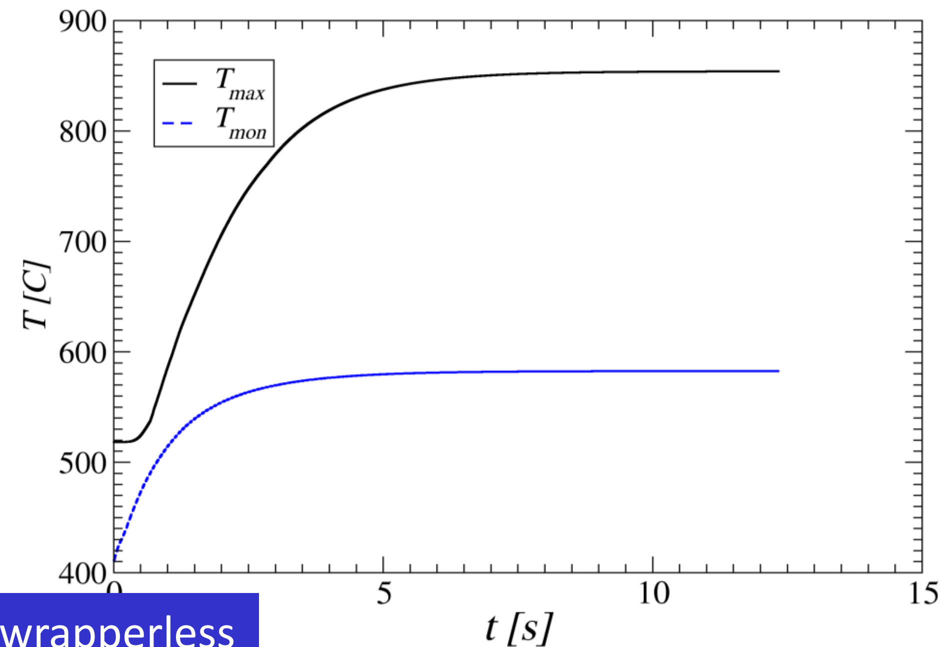
✓A *global effect* due to the lower mass flow rate in the blocked subchannels; this fact leads to an increase of the bulk fluid temperature with respect to the ‘unblocked’ regions and a consequent peak in the clad temperature at the end of the active region.

Flow Blockage

AGENZIA NAZIONALE PER LE NUOVE TECNOLOGIE, L'ENERGIA E LO SVILUPPO ECONOMICO SOSTENIBILE

Two types of FA are being considered for LFR:
- Open (wrapperless) FA and
- Closed (wrapped) FA

The simulated blockage regards 20% of the spacer grid

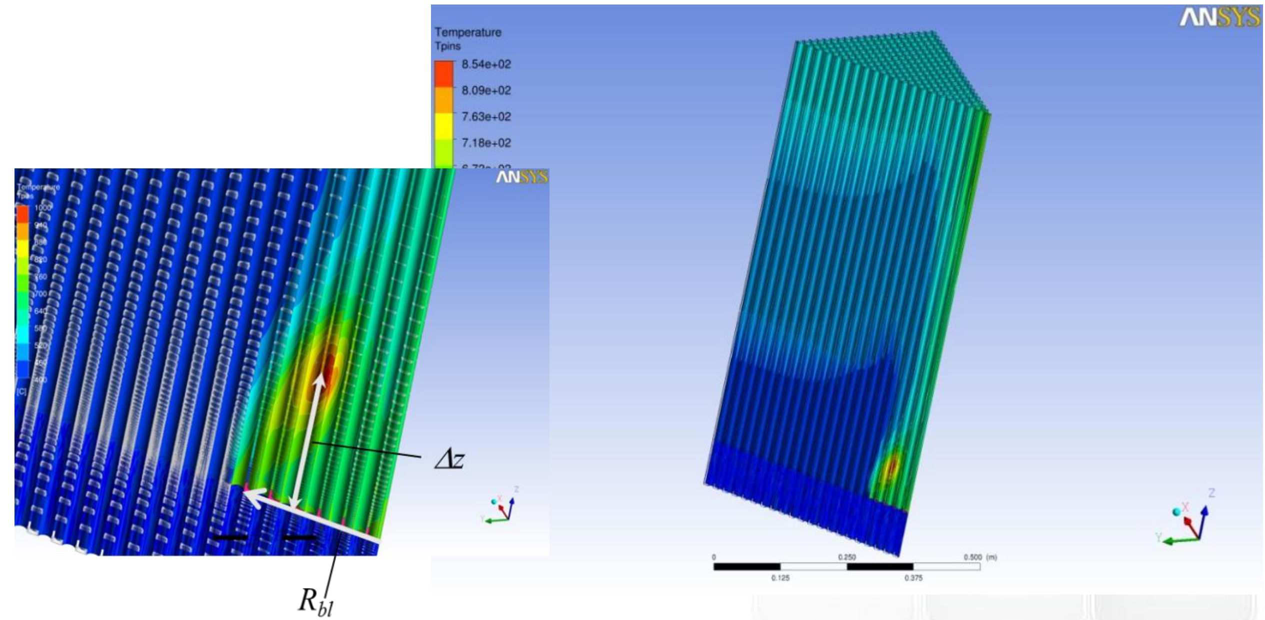


Case of wrapperless

Results of the analyses indicate that:

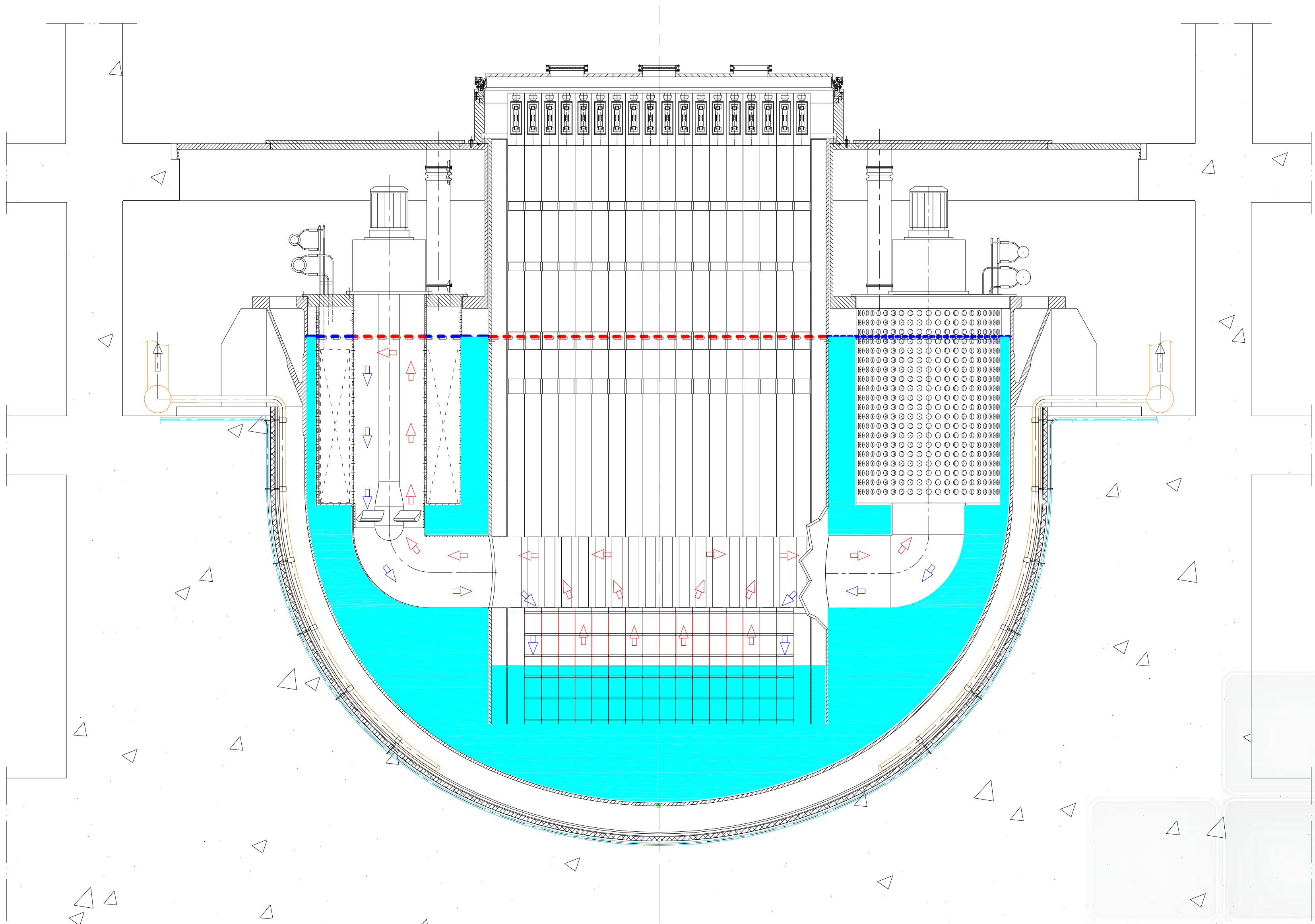
- *the main advantage of the open FA with respect to the closed FA* is the fact that in case of blockage the velocity far upstream and downstream the occlusion remains unchanged, and therefore the *mass flow rate across the FA is not altered*.

- On the opposite, in a *closed FA* a flow blockage *increases the hydraulic resistance* and leads to a lower overall mass flow rate.

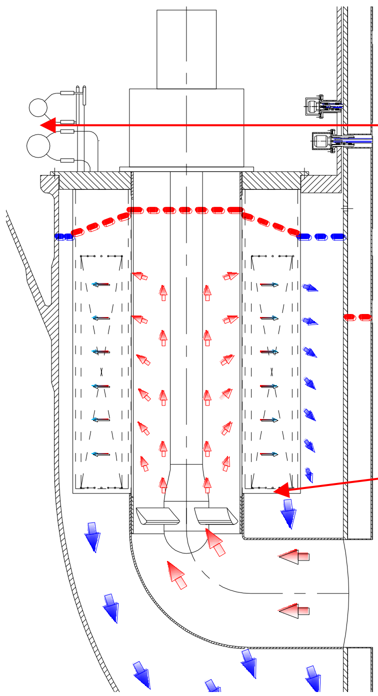


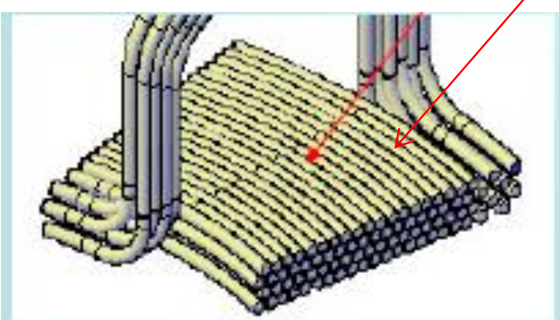
Lead freezing

It is expected that wrapperless fuel elements will allow core cooling even in case of frozen primary loop. (Cooling the FA by downward flow +cross flow toward the central FA with progressive heating up and formation of a central hot plume). Calculations are ongoing



The spiral-tube SG to mitigate the SGTR accident





a) **Feed water and steam collectors are installed outside the reactor vessel.**
No risk of catastrophic primary system pressurization

b) **The tube bundle of the spiral-tube SG is positioned up in the reactor vessel.**
In case of Steam Generator Tube Rupture steam is released near the lead free level reducing lead displacement.

c) **The tube bundle of the spiral-tubes SG is closed in the bottom part.**
In case of Steam Generator Tube Rupture no downward steam jet is possible and only small bubbles can be entrained inside the core.

d) **The tube bundle of the spiral-tube SG is constituted by long , high-pressure-loss tubing with a superheated or supercritical cycle.**
In case of Steam Generator Tube Rupture water-steam flow rate is limited.

e) **The tube bundle of the spiral-tube SG is constituted by few tubes with excess flow valves at water inlet and check valves at steam outlet.**
In case of Steam Generator Tube Rupture water-steam flow rate is promptly interrupted.

Conclusions:
The LFR in the post-Fukushima era

There is a reasonable expectation of demonstration of the capability of the LFR system to avoid scenarios leading to severe core damage.

- ✓ **At Fukushima, the common mode failure of diesel generators has prevented the core cooling function.**
  A LFR does not need diesel generators to cool the core, DHR is passively operated.

- ✓ **At Fukushima primary coolant has been lost.**
  Lead cannot be lost (lead can leak into the guard vessel, but the core always remains covered).

- ✓ **At Fukushima DC power has been lost as well as all control logics.**
  A LFR can survive unprotected transients.
  DHR can be passively initiated.
  Passive initiation should be also provided to face cyber attacks. (After the Stuxnet virus has infected the Bushehr nuclear power, Yukiya Amano, the director general of the International Atomic Energy Agency, has expressed concern about cyber attacks on nuclear facilities).

Conclusions:
The LFR in the post-Fukushima era

There is a reasonable expectation of demonstration of the capability of the LFR system to manage extreme events in degraded plant conditions.

✓ **At Fukushima interaction of steam with fuel cladding has produced hydrogen and associated explosions.**
In a LFR there in no hydrogen generation.

✓ **At Fukushima the plant thermal capacity has resulted in a slow progression of the accident allowing a timely evacuation of the people in the surrounding of the plant.**
A LFR has a high thermal capacity.

✓ **At Fukushima the main contamination is due to Cs.**
Lead retains Cs.

✓ **At Fukushima a more severe catastrophic scenario has been prevented by means of ultimate intervention with use of sea water.**
Ultimate interventions with the use of water is thinkable in a LFR.