Safety Demonstration of *new*cleo's LFR-AS-30

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Contents

Introduction

What is a ‘Safety Demonstration’?

- IAEA definition (draft Safety Standard DS537):
  “‘***Safety demonstration***’ *refers to a* ***comprehensive process*** *to* ***validate*** *and* ***substantiate*** *the* ***safety claims*** *made during the* ***facility’s design****, which considers* ***findings*** *of a safety assessment* ***supported*** *by a* ***statement*** *of* ***confidence*** *in these findings (see* ***safety case***†*), and* ***implies acceptance*** *of* ***interested parties*** *(e.g. designer, operating organisation,* ***regulatory body****).”*

- A nuclear reactor’s **operating license** relies on a **Safety Analysis Report** — a **comprehensive** and “**living**” **document** that **provides assurance** to a **regulatory body** that the **plant** can **operate without** posing **unacceptable risk** to **public health** and **safety**

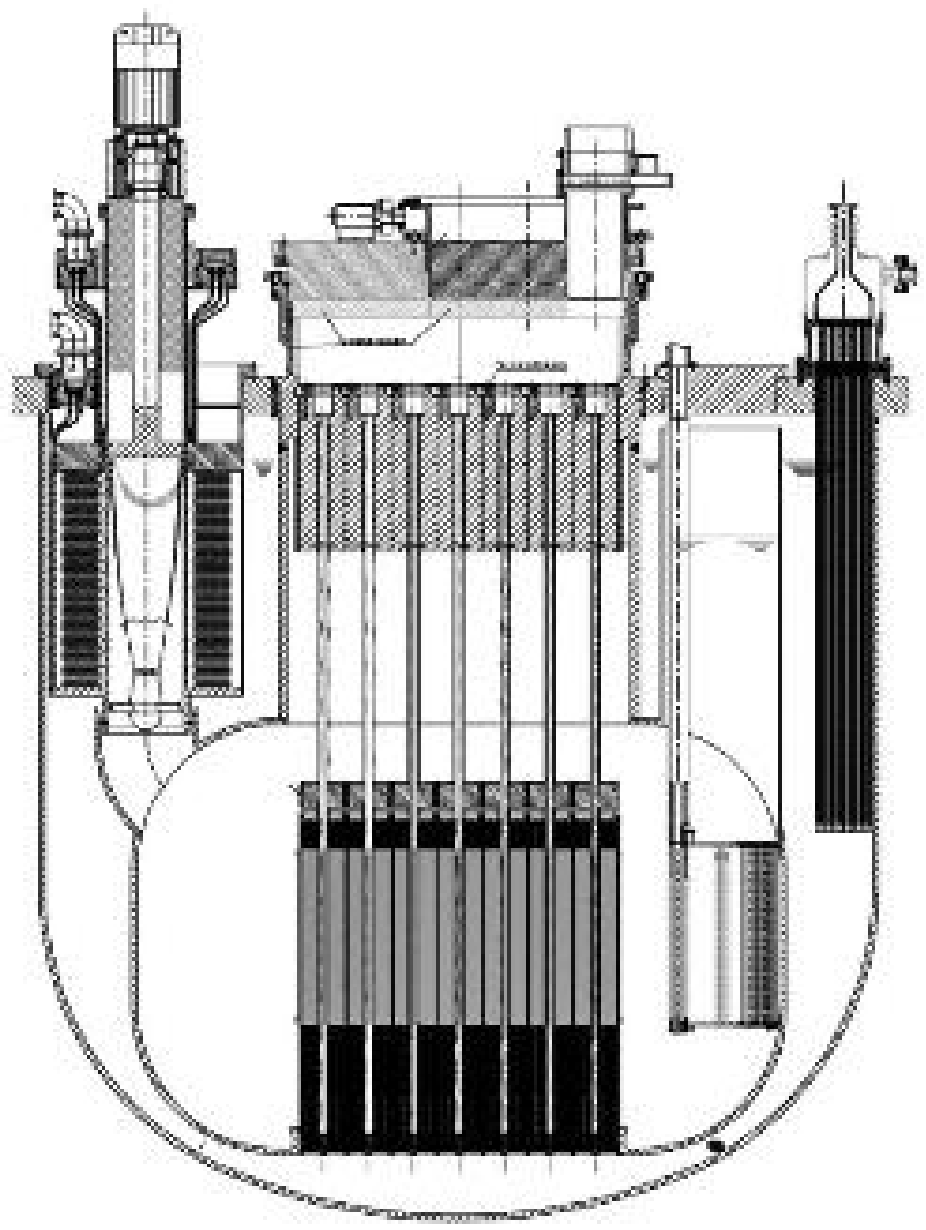
- The **Safety Analysis Report** is built on the ‘**Claim–Argument–Evidence**’ concept (→ **safety case**†)

- The **Safety Demonstration** provides the **scientific evidence** for the **safety case**

† ***Safety case****: the* ***collection of arguments and evidence*** *to demonstrate the safety of a facility or activity and normally includes the findings of a safety assessment and a statement of confidence in these findings.*

*new*cleo's LFR-AS-30 Safety Architecture

LFR-AS-30: Amphora Shaped, 30MWe



- *new*cleo's first LFR, demonstrating feasibility, operability, maintainability and more
- Single-unit plant
- Reactor **conceptual design** completed in March 2023, basic design in progress
- **First phase** of **technical meetings** with **ASN** and **IRSN completed** in **June 2024**, **LFR-AS-30 design and related safety options submitted to ASNR for review and assessment in December 2025**

	1st phase Low temp. and power	**2nd phase** High temp. and full power
Power	**60MWth**	**90MWth**
Core coolant temperature	**inlet 370°C, outlet 440°C**	**inlet 420°C, outlet 530°C**
Steam at turbine inlet	**400°C, 150 bar**	**500°C, 150 bar**
Core coolant	**Pure lead**	
Layout	**Pool-type**	
Circulation	**Forced: 3 pumps**	
Spectrum	**Fast**	
Fuel form	**Extended-stem fuel assembly**	
Fuel	**MOX**	
Secondary side fluid	**Water**	
Steam generators	**3 spiral-tube SG**	
Design life	**60 years**	

Fundamental Safety Functions (IAEA SSR 2/1 Rev. 1)

1. Control of Reactivity

- ✓ Prevent unintended or uncontrolled chain reactions
- ✓ Ensure shutdown systems can reliably stop fission
- ✓ Maintain subcriticality under all operational conditions

2. Removal of Heat from the Core

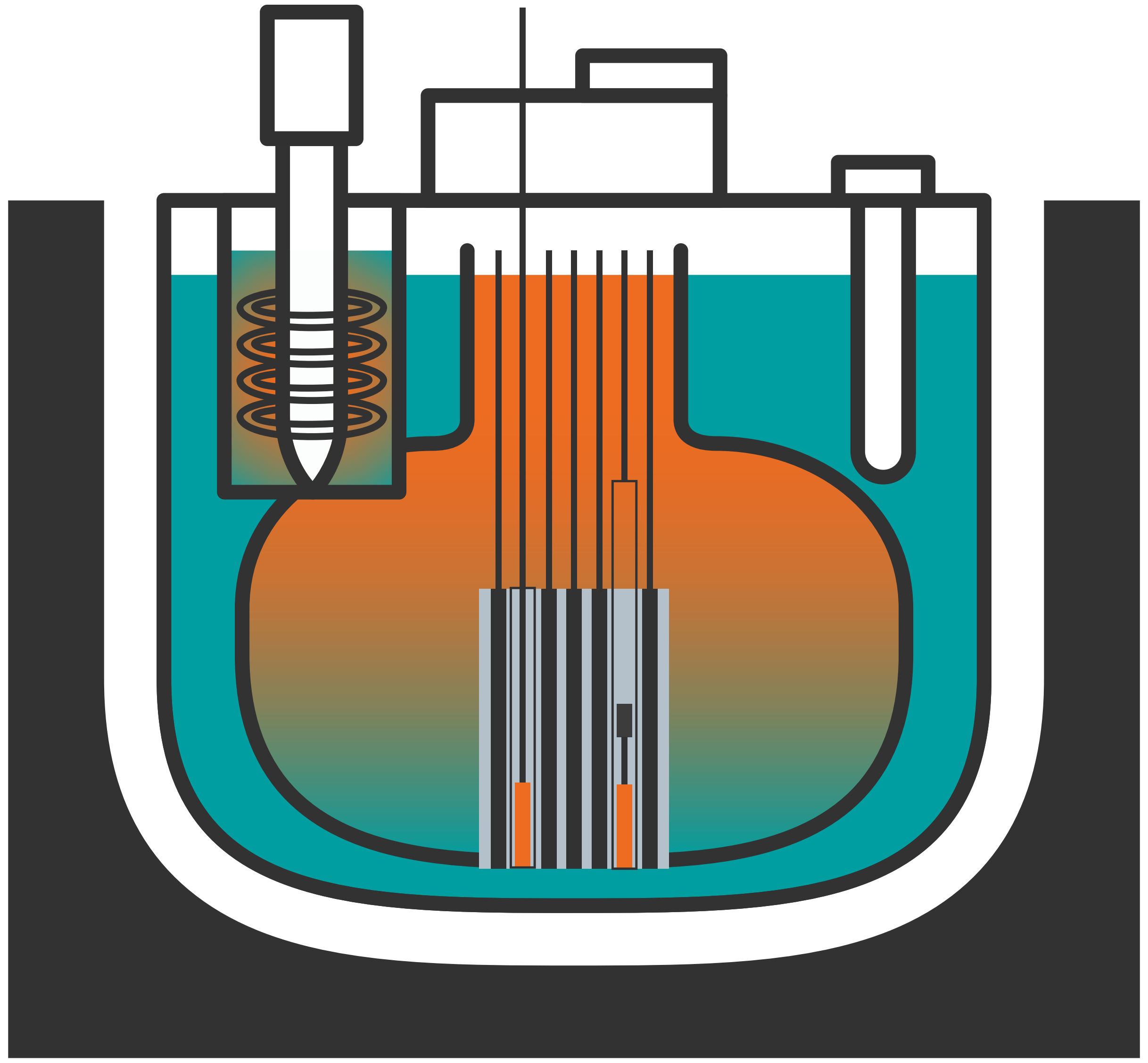
- ✓ Transfer residual heat from the reactor core to prevent overheating.
- ✓ Includes active and passive cooling systems.
- ✓ Critical for avoiding fuel damage and maintaining structural integrity.

3. Confinement of Radioactive Materials

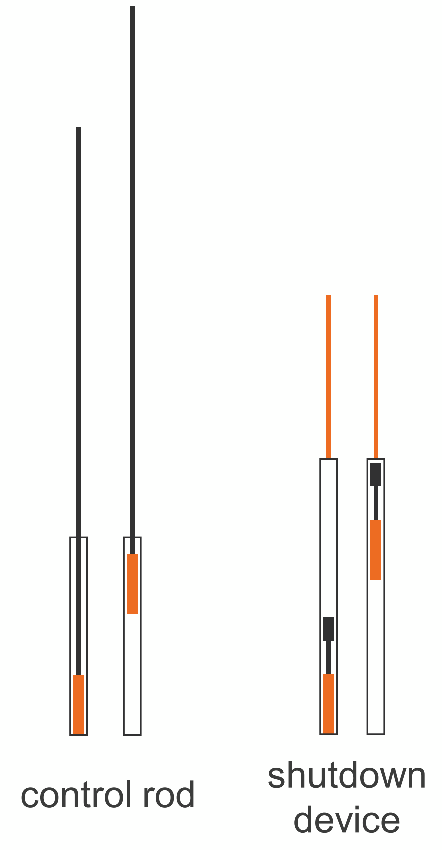
- ✓ Prevent release of radioactive substances to the environment
- ✓ Involves robust barriers (e.g., fuel cladding, reactor vessel, containment)
- ✓ Includes **radiation protection** measures for workers and the public

Reactivity control

- Two neutron absorber systems of diverse design, located in the core
  - **Control rods** – Release of power to drive mechanism leads to insertion into the active core by buoyancy
  - **Shutdown devices** – Release of pressure to piston leads to insertion into the active core by buoyancy. Passive fusible plug provides an additional I&C free means of operation
- Reactivity feedbacks ensure acceptable behaviour in unprotected transients
- Design of core and supports to ensure geometry maintained and compaction avoided
- Designed to protect against gas insertion events (SGTR)

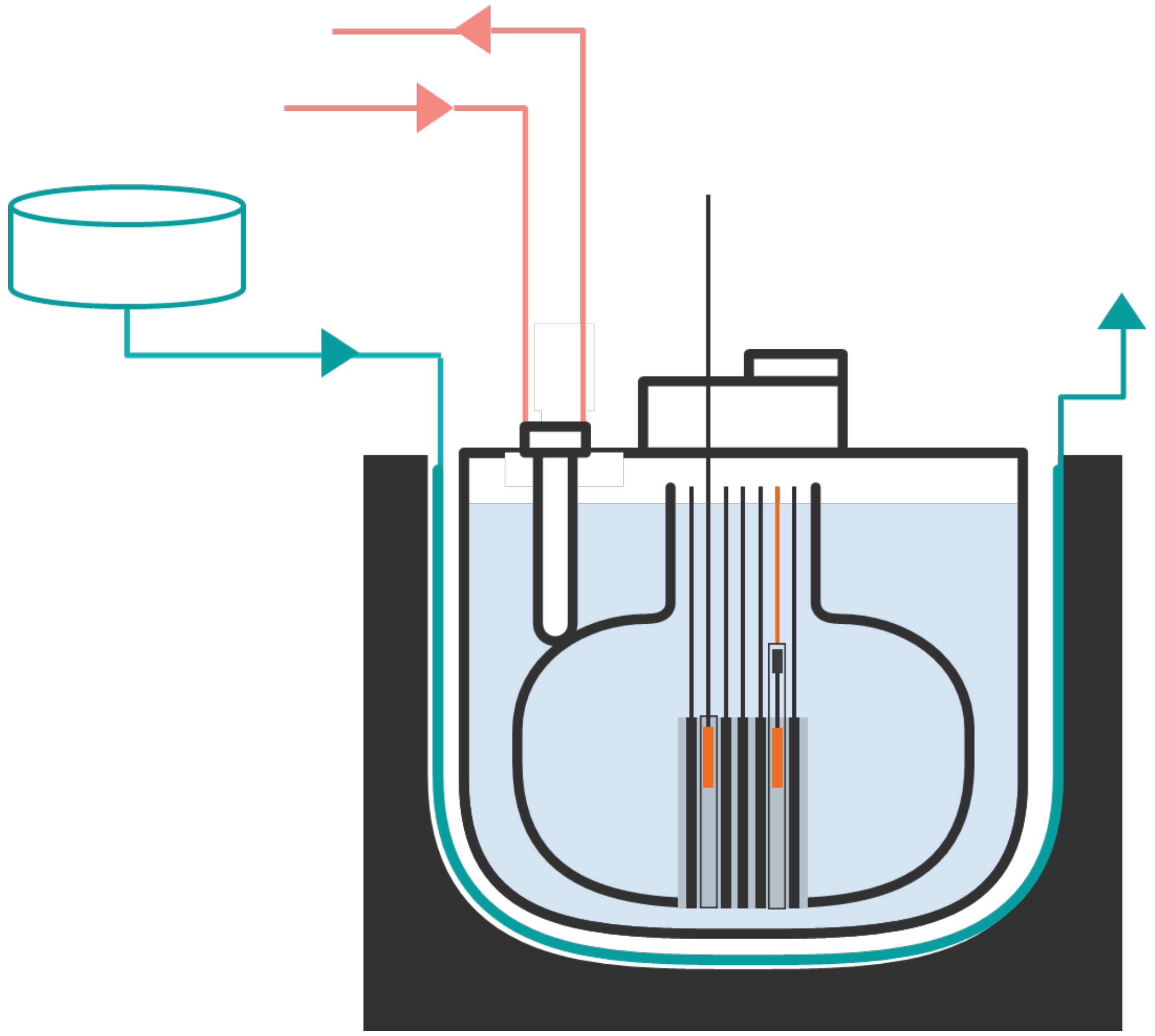


- *Able to shut the core down with either absorber system in all transients.*
- *Able to tolerate unprotected transients*
- *Practical elimination of prompt criticality*



Decay Heat Removal (DHR)

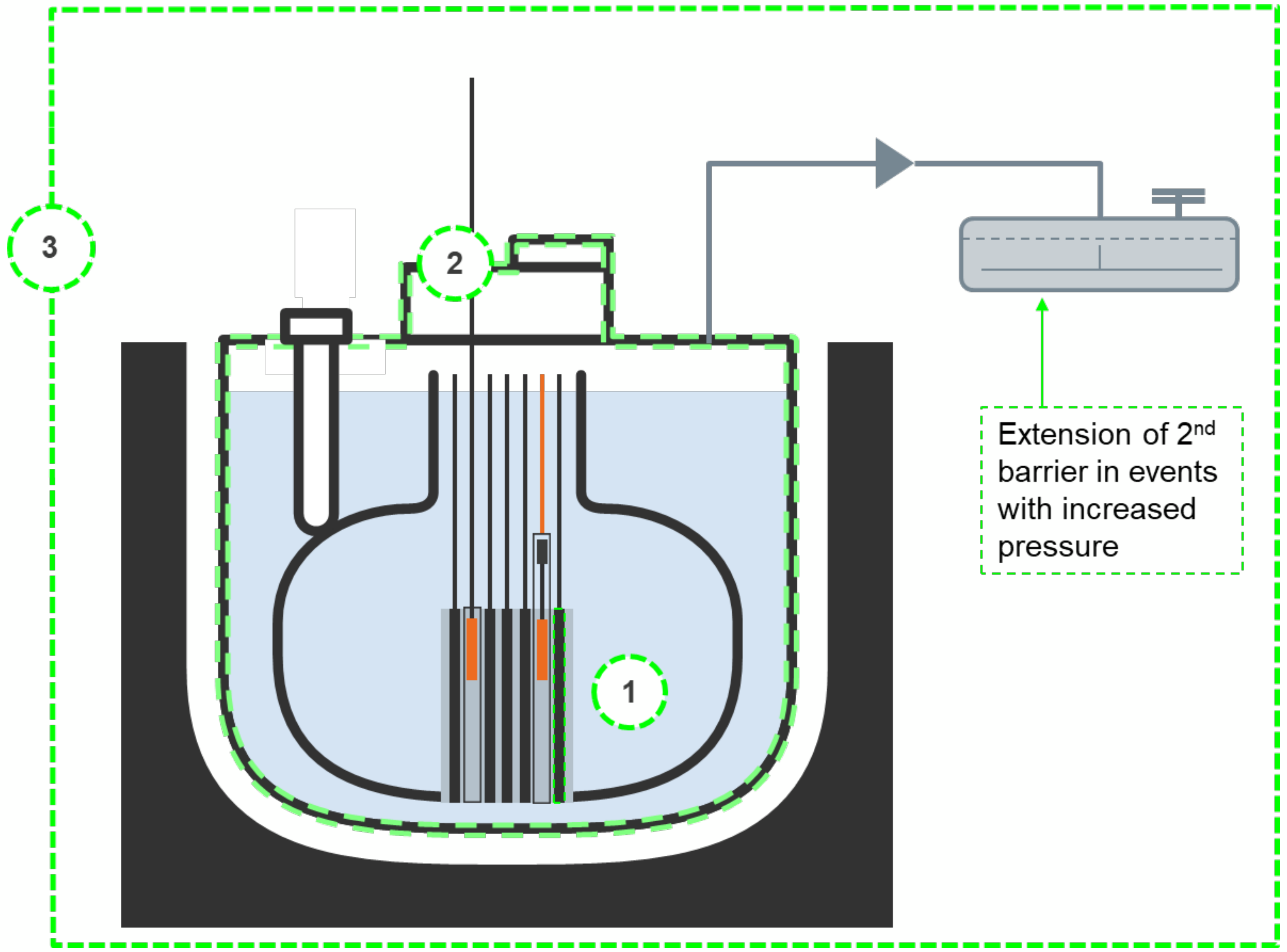
- Two highly reliable, diverse means of removing decay heat:
  - ✓ **In-vessel** DHR system used in **design basis conditions** – heat exchangers immersed in Pb pool (“dip coolers”)
  - ✓ **Out-of-vessel** system for **design extension conditions**
- **Simple**, **passive** design and strong **independence**
- Additional **risk-reduction** measures against **total prolonged loss** of **DHR function** and for **long-term operation** using mobile equipment if needed
- **Pb solidification prevention** in design basis conditions, freezing and remelting made **tolerable** in design extension conditions
- **Prevention** of **loss** of **Pb inventory** via reactor vessel and qualified reactor pit (ability to cope with vessel leaks)



- *Practical elimination of prolonged total loss of decay heat removal*
- *Ensuring acceptable Pb levels in all circumstances*
- *Low frequency of freezing and demonstration of tolerability of freezing*

Confinement

- Fuel clad (**1st barrier**) **integrity** ensured via DBC and DEC-A **acceptance criteria** and **operating limits** to prevent clad failure through various degradation mechanisms
- Primary circuit (**2nd barrier**) **integrity** via **leakage limits** and **design** for performance in accident conditions. Lack of high-energy LOCA equivalents is beneficial, SGTR challenge to be managed via overpressure protection system
- **3rd barrier design** being developed to ensure **acceptable offsite** radiological **releases** (demonstration of leakage rates)
- **Pb retention** of **fission products** is a **key advantage** of **LFR technology** that reduces the demand on 2nd and 3rd barriers in the event of 1st barrier failure. **R&D** ongoing to **demonstrate** benefit of **retention**



- *1st barrier integrity requirements for DBCs and DEC-A (no or limited failures depending on frequency)*
- *2nd barrier reinforced with aim of minimising release into 2nd barrier including in SGTR events*
- *3rd barrier design to ensure acceptable off-site consequences*

Safety Demonstration and R&D

R&D needs for the Safety Demonstration of LFR-AS-30

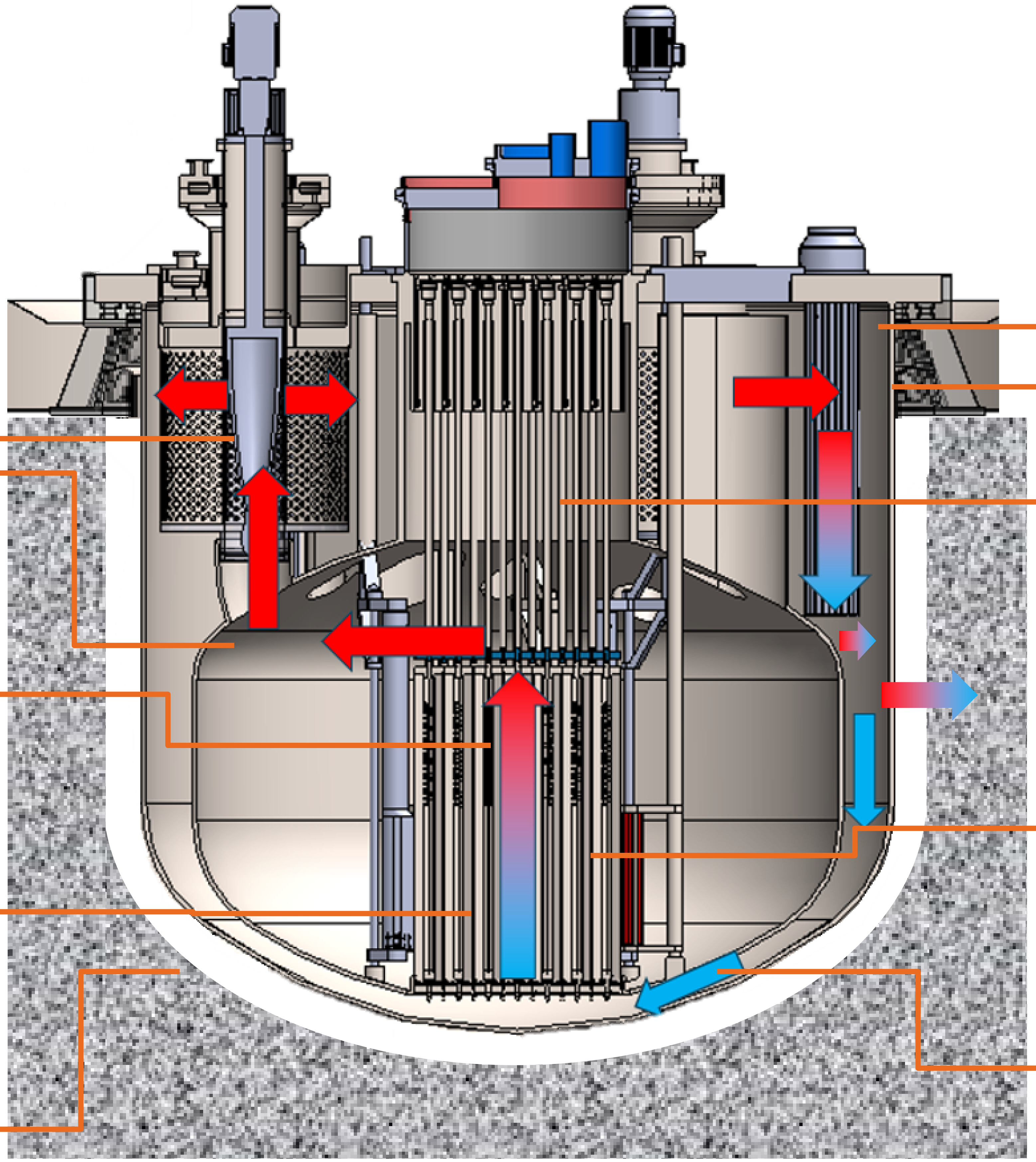
Steam Generator Tube Rupture

In-Pb retention of radionuclides

Fuel-coolant interaction

Reliability of reactivity control provisions

Core instrumentation and in-service inspection techniques



Reliability of DHR provisions

Fluid-structure interaction in seismic events

Feasibility of “self-supporting core”

Pb corrosion resistance of core materials

Demonstration of natural circulation performance

Reactivity Control

Safety Demonstration of Reactivity Control Provisions

| Core Support and Restraint Functions

- *new*cleo's LFR-AS-30 exploits **buoyancy** of **steel** in **Pb** to minimise need for traditional bottom support structures → "**self-supporting core**" concept

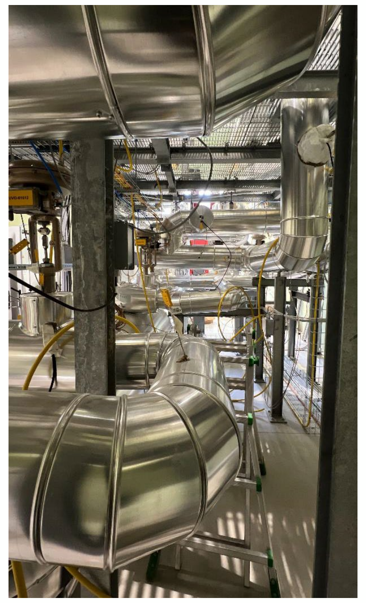
- **Vertical restraint** of fuel assemblies ensured through **mechanical interlocking** between top-mounted **ballasts** and upper part of **Amphora-Shaped Inner Vessel** (ASIV)

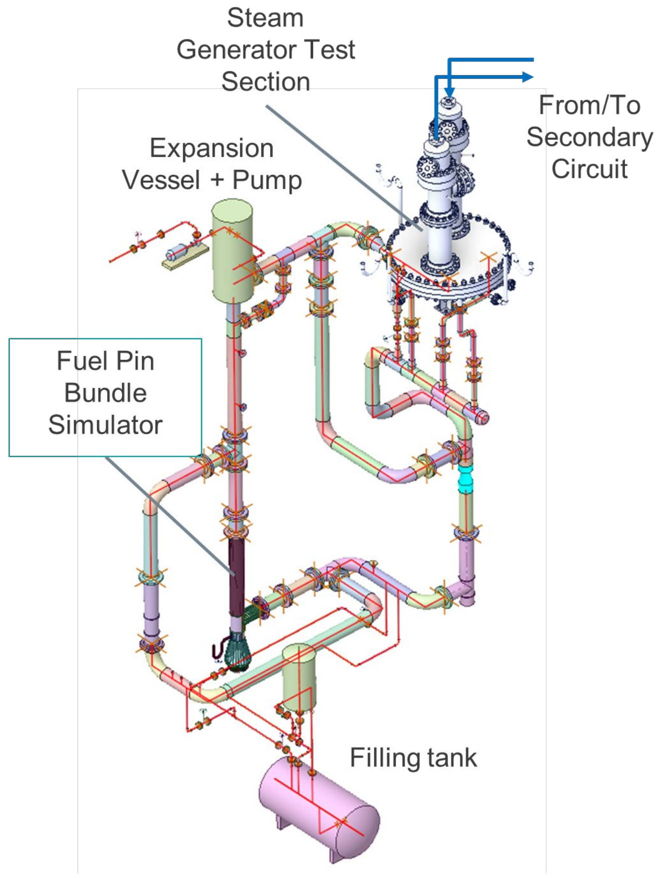
- Safety demonstration of self-supporting core concept relies on complex mechanical analyses and R&D:
  - ✓ **Static** simulations under normal operation and accident conditions (using RCC-MRx criteria)
  - ✓ Spatial characterization of **stress state** in all areas of assembly
  - ✓ **Seismic** response of assemblies and interlocking mechanism (notably **sloshing** risk)
  - ✓ **Buckling** risk assessment
  - ✓ **Experimental** campaigns to evaluate parameter **uncertainties** for numerical simulations:
    - Phenomenological investigations (separate and integral effects)
    - Validation of numerical models

Safety Demonstration of Reactivity Control Provisions

| Core Support and Restraint Functions – Experimental Facilities @ ENEA Brasimone, Italy

- **MAT LAB**
  → **Mechanical** characterization of core support materials (slow strain rate tests, creep-rupture, fretting)
- **CORE, CAPSULE, RACHEL LAB, CHEM LAB**
  → Pb-induced **corrosion** and Pb-steel interactions (erosion, embrittlement)
- **OTHELLO** (FIV test section)
  → **Flow-induced vibrations** in normal and accident Pb flow conditions using realistic fuel assembly geometry
- **MANUT-in-lead**
  → Realistic testing of top core support structure mechanical performance (notably interlocking mechanism)
  → Validation of robustness of ASIV/bulkhead under normal and accident conditions (e.g., deformations beyond neutron-induced creep effects)



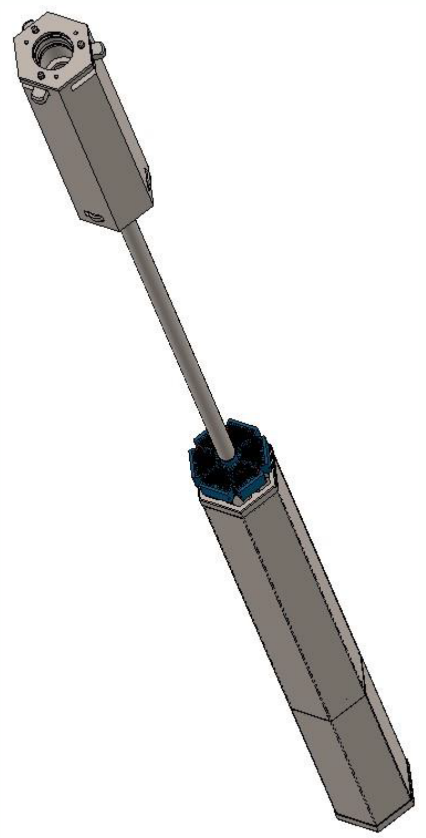


Safety Demonstration of Reactivity Control Provisions

| Reactivity Control Devices

CONTROL RODS

Reactivity control in normal operation + reactor trip, completed/ongoing analyses:



- **Neutronic efficiency**
  → Best Estimate shutdown margin evaluated @ BoL and EoL
  → Adequate margin to single failure criterion implemented by design
  → Assessment of long-term effectiveness against rod depletion
- **Mechanical integrity**
  → Design studies under normal and accident conditions in progress
- **Operational parameters**
  → **Rod insertion time** estimated analytically (~1 s) ➔ experimental validation in **MANUT-in-lead**
  → High buoyancy and short travel distance (active zone ~0.7 m)
  → Assessment of insertion performance in seismic event
  → Design margins to be revised as per results of experimental campaign

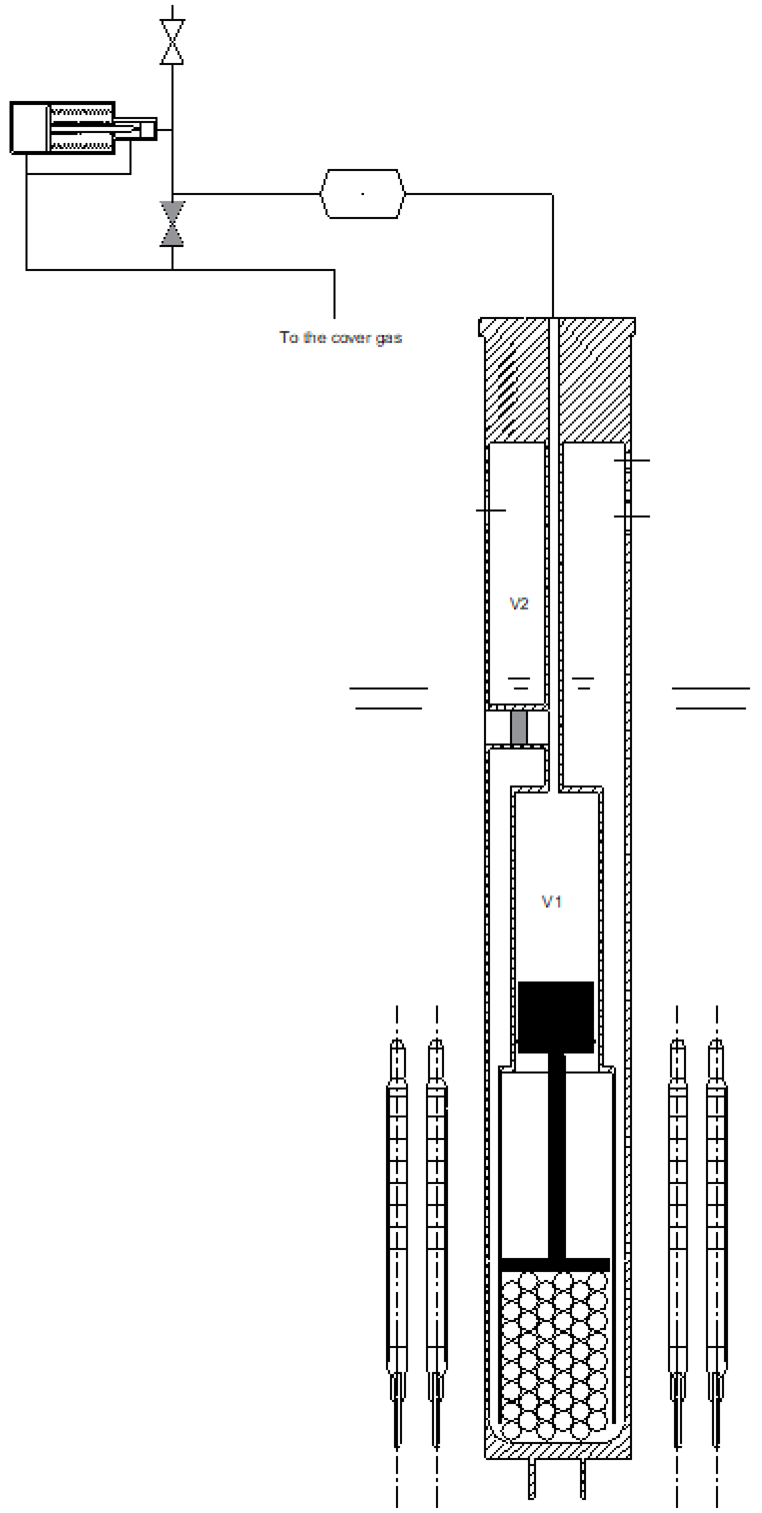
Safety Demonstration of Reactivity Control Provisions

| Reactivity Control Devices

DIVERSE SHUTDOWN DEVICES

Emergency shutdown, completed/ongoing analyses:

- **Positioning uncertainty assessment** relative to active core height
- **Full qualification program**
  → Insertion speed
  → Geometric positioning accuracy and stability under operational stress
- **Feasibility campaign ➔ MANUT-DRY** and **MANUT-in-lead**
  → Control rod mock-up insertion
  → Shutdown device actuation and positioning



Safety Demonstration of Reactivity Control Provisions

| Nuclear Data Improvement @ TAPIRO (ENEA Casaccia)

- Deployment of fleet of GEN-IV commercial reactors requires **reduction** of **nuclear data uncertainties** for **fast spectrum** materials cross-sections, notably **transuranic** elements
  - ➔ Reduce **overall uncertainty** on neutronic parameters (cf. k_{eff}) and **reactivity feedback** coefficients
- Novel evaluation of cross-sections starting from new and more accurate differential measurements for:
  - ✓ ^{206}Pb, ^{207}Pb
  - ✓ ^{238}U, ^{241}Pu, MA
- Needs identified by dedicated NEA expert group fit experimental capabilities of **TAPIRO fast neutron source reactor** in **ENEA Casaccia** research centre

Table C1 reproduces some of the needs identified by several authors as being relevant to the TAPIRO experimental capabilities. These needs cover mainly advanced reactor concepts in the fast range: breeders, burners, comprising either sodium, lead, lead/Bi or gas designs.

Table C1. Nuclear data needs relevant to TAPIRO experimental capabilities.

		Energy Range	Current Accuracy (%)	Target Accuracy (%)
U238	σ_{inel}	6.07 ÷ 0.498 MeV	10 ÷ 20	2 ÷ 3
	σ_{capt}	24.8 ÷ 2.04 keV	3 ÷ 9	1.5 ÷ 2
Pu241	σ_{fiss}	1.35MeV ÷ 454 eV	8 ÷ 20	2 ÷ 3 (SFR,GFR,LFR) 5 ÷ 8 (ABTR,EFR)
Pu239	σ_{capt}	498 ÷ 2.04 keV	7 ÷ 15	4 ÷ 7
Pu240	σ_{fiss}	1.35 ÷ 0.498 MeV	6	1.5 ÷ 2
	ν	1.35 ÷ 0.498 MeV	4	1 ÷ 3
Pu242	σ_{fiss}	2.23 ÷ 0.498 MeV	19 ÷ 21	3 ÷ 5
Pu238	σ_{fiss}	1.35 ÷ 0.183 MeV	17	3 ÷ 5
Am242m	σ_{fiss}	1.35MeV ÷ 67.4keV	17	3 ÷ 4
Am241	σ_{fiss}	6.07 ÷ 2.23 MeV	12	3
Cm244	σ_{fiss}	1.35 ÷ 0.498 MeV	50	5
Cm245	σ_{fiss}	183 ÷ 67.4 keV	47	7
Fe56	σ_{inel}	2.23 ÷ 0.498 MeV	16 ÷ 25	3 ÷ 6
Na23	σ_{inel}	1.35 ÷ 0.498 MeV	28	4 ÷ 10
Pb206	σ_{inel}	2.23 ÷ 1.35 MeV	14	3
Pb207	σ_{inel}	1.35 ÷ 0.498 MeV	11	3
Si28	σ_{inel}	6.07 ÷ 1.35 MeV	14 ÷ 50	3 ÷ 6
	σ_{capt}	19.6 ÷ 6.07 MeV	53	6

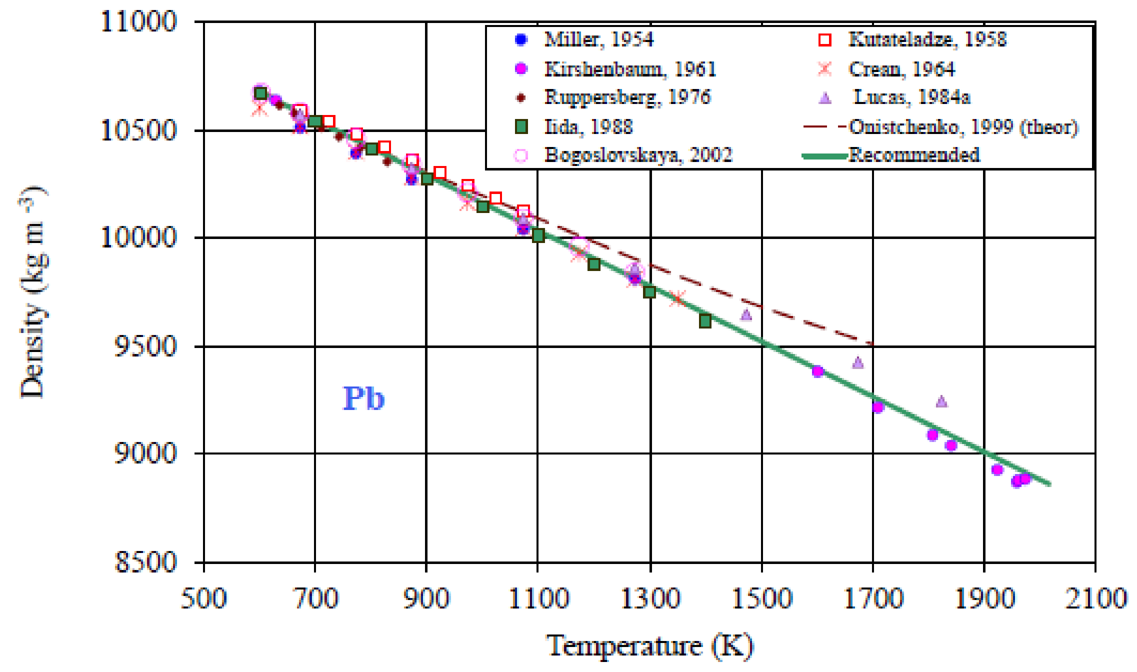
Cooling

Safety Demonstration of Residual Heat Removal Provisions

| Natural Circulation in Primary Coolant Circuit

- **Intrinsic** properties of **Pb** leveraged in design to **enhance natural circulation**
  - → Large density of Pb ⇒ very high **buoyancy** effect
  - → Low moderating power of Pb ⇒ low neutron absorption σ
    - ⇒ can increase hydraulic diameter of core channels
    - ⇒ core **head loss minimised**

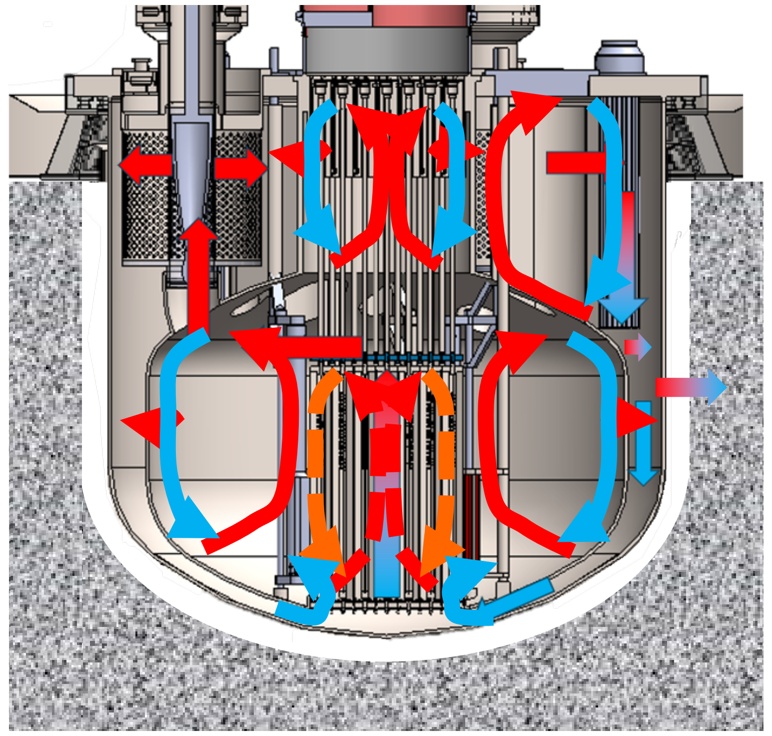
Figure 2.11.1(a): Density of molten lead as function of temperature at normal pressure



Source: Sobolev (2011).

- **Primary Pumps <u>not</u> credited** in **accident scenarios** in *new*cleo LFR
  ➔ rely on **natural circulation**

- Safety classification of Primary Pumps and power supplies in line with **passivity** goal:
  - ✗ No classification for long-term operation
  - ✓ PP safety classification and power supplies may be needed to <u>initiate natural circulation</u> (but preference for mechanical inertia)



Safety Demonstration of Residual Heat Removal Provisions

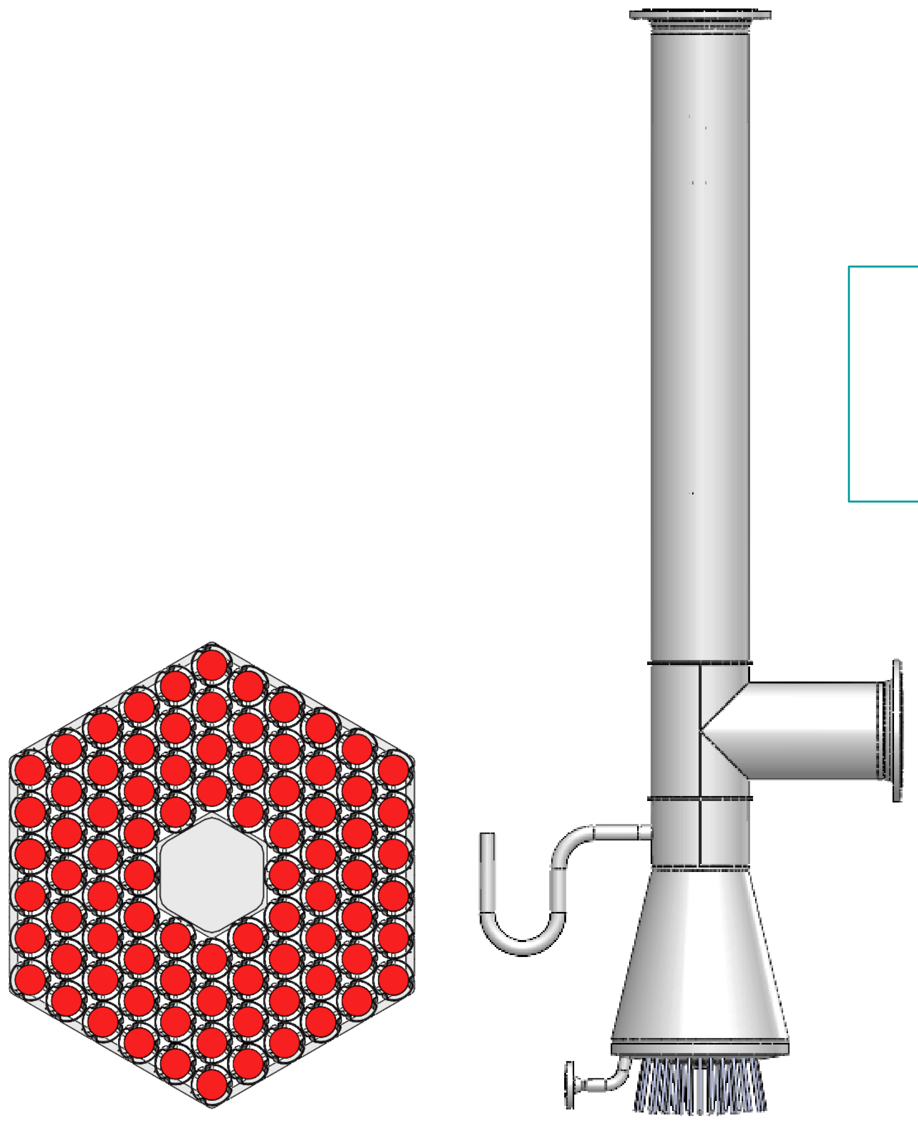
| OTHELLO (Oxygen controlled Thermal Hydraulic Experimental Lead Loop) @ ENEA Brasimone

Separate-effect test facility

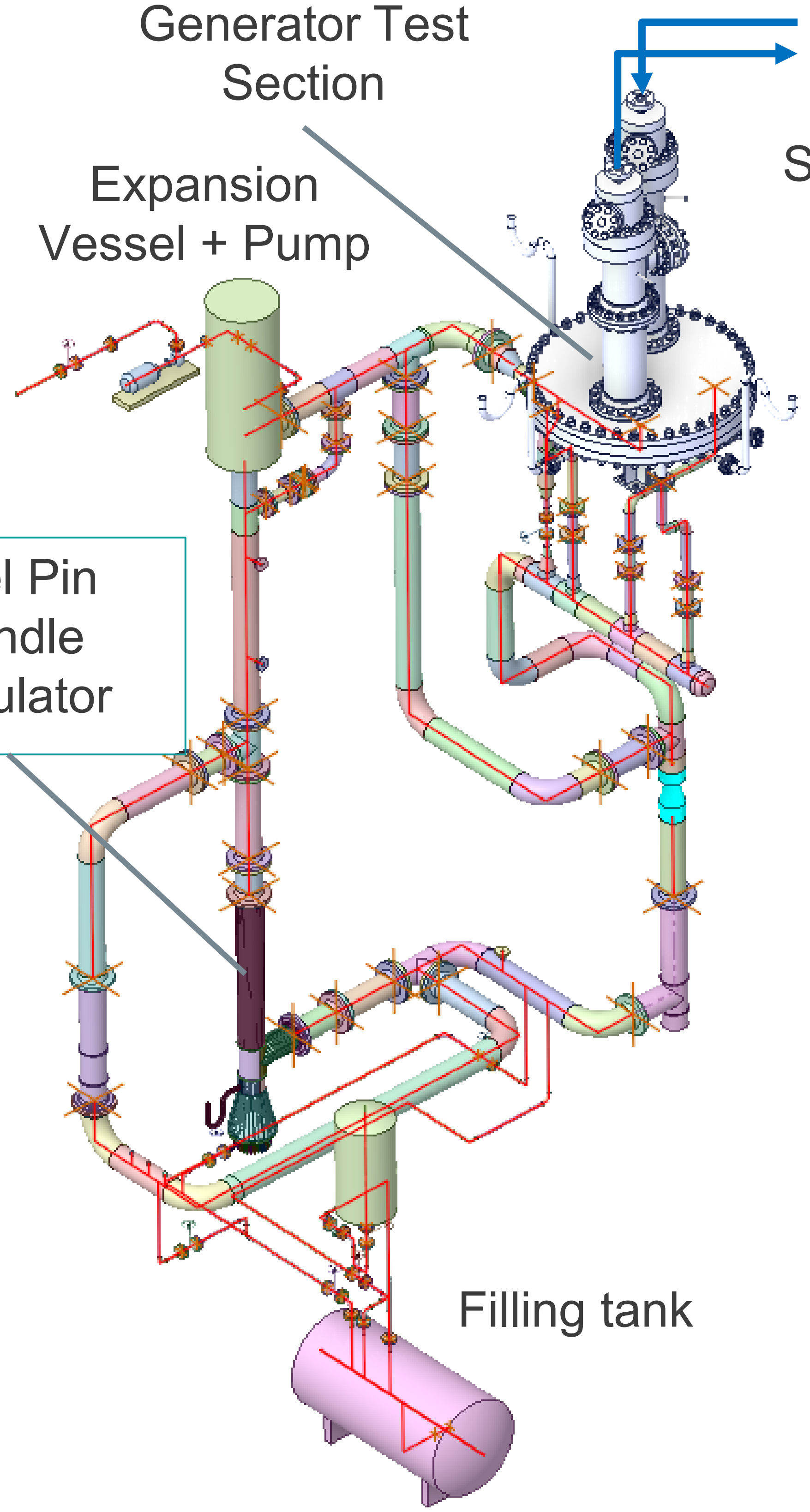
- ✓ Assess performance of key primary components
- ✓ Support **validation** of **thermal-hydraulic codes**
- ✓ Test instrumentation and chemistry control systems
- ✓ Investigate specific **thermal-hydraulic phenomena**

Key test sections

- **Electrically heated Fuel Pin Bundle Simulator**
  - ✓ Largely representative of LFR fuel assembly
  - ✓ Characterization of convective heat transfer over range of TH conditions and flow regimes
- **Unheated Fuel Pin Bundle Simulators**
  - ✓ Largely representative of LFR fuel assembly
  - ✓ Investigation of FA head losses, effect of pin spacers, fluid-induced vibrations in FA, etc.



Conceptual model of OTHELLO FPBS



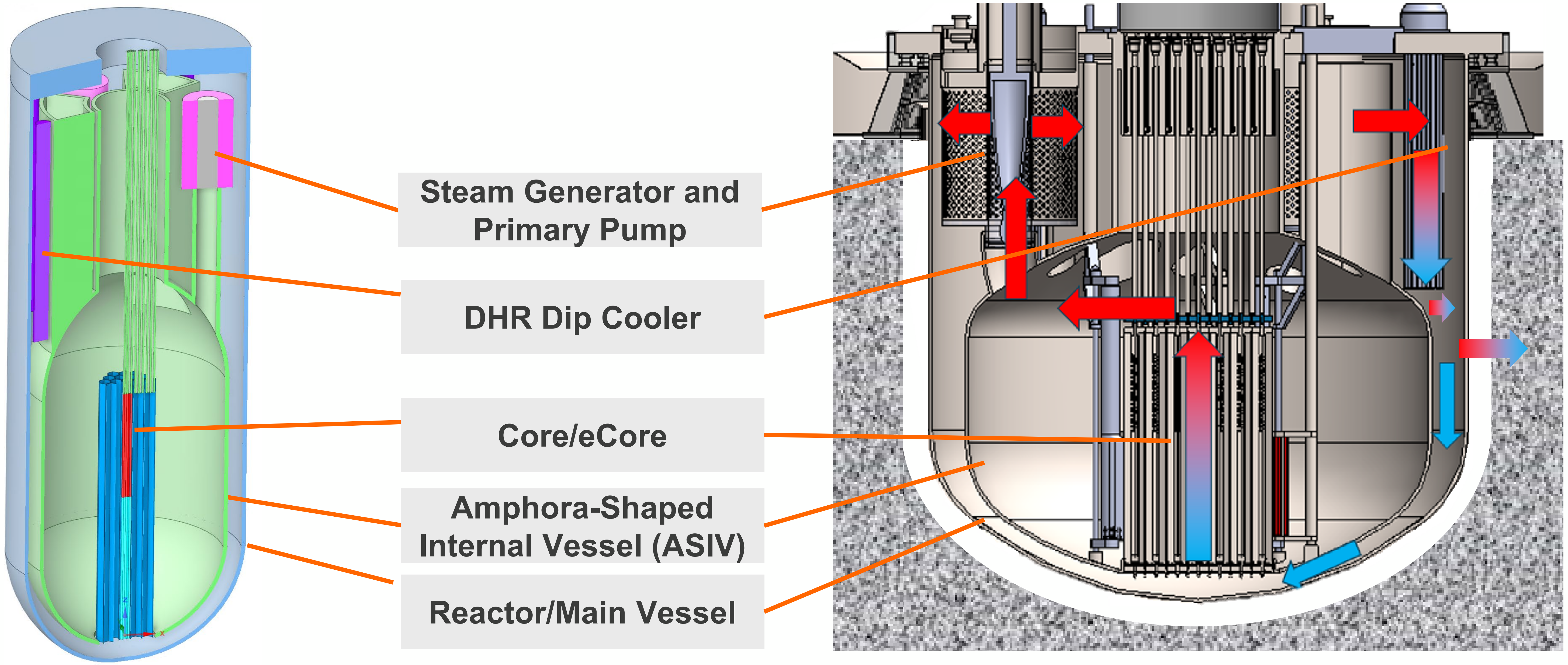
Safety Demonstration of Residual Heat Removal Provisions

| PRECURSOR @ ENEA Brasimone

Precursor is a molten-lead, pool-type, electrically heated, **integral-effect test facility** representative in size (1:9 power-to-volume), complexity and thermal hydraulic performances, of the LFR-AS-30 reactor

Precursor | **LFR-AS-30**

Power	**10MWth**
Core coolant temperature	**inlet 420°C, outlet 530°C**
Steam at turbine inlet	**500°C, 150 bar**
Core coolant	**Pure lead**
Layout	**Pool-type**
Circulation	**Forced: 3 pumps**
Secondary side fluid	**Water**
Steam generators	**3 spiral-tube SG**



Safety Demonstration of Residual Heat Removal Provisions

| PRECURSOR @ ENEA Brasimone

- Thermal-hydraulic performance in **nominal** (stationary) **operating conditions**
  - ✓ Steam (and electrical power) production in stationary operation at various power levels
  - ✓ Stationary operation of DHR systems
- Thermal-hydraulic performance in **operational transient conditions**
  - ✓ System start-up
  - ✓ System shutdown
- **Accident transient behaviour**, including 3D phenomena/imbalances
  - ✓ Partial loss of flow (loss of pumps)
  - ✓ Partial loss of DHR systems during shutdown transients
  - ✓ Transients initiated in the Balance of Plant (turbine trips, loss of feedwater etc.)
- Assessment of **induced vibrations** on relevant structures

Safety Demonstration of Residual Heat Removal Provisions

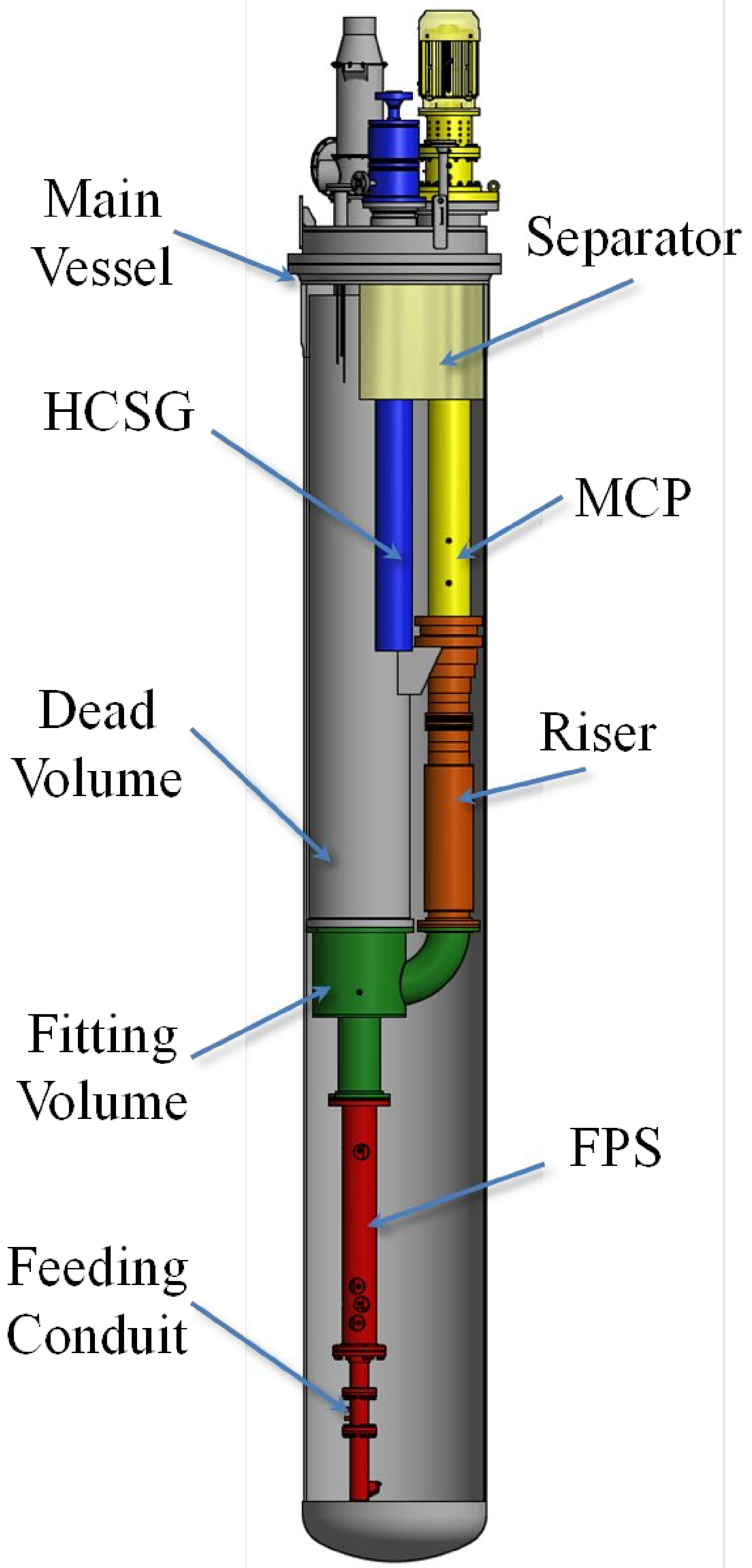
| Refurbishment of CIRCE-NEXTRA @ ENEA Brasimone

CIRCE: largest European HLM **pool-type facility**

- ✓ Main vessel 8.5m high and 1.2m in diameter
- ✓ Up to ~90 tons of molten LBE
- ✓ Includes argon cover gas and recirculation system, LBE heating and cooling systems, secondary loop to supply water up to ~180 bar and 335°C, LBE storage and transfer tanks, data acquisition system

CIRCE-NEXTRA is a stepwise refurbishment of CIRCE facility, devoted to:

- ✓ Long-duration **endurance tests of axial flow pump** operating with Heavy Liquid Metals (HLM)
- ✓ **Steam Generator Tube Rupture** (SGTR) investigation

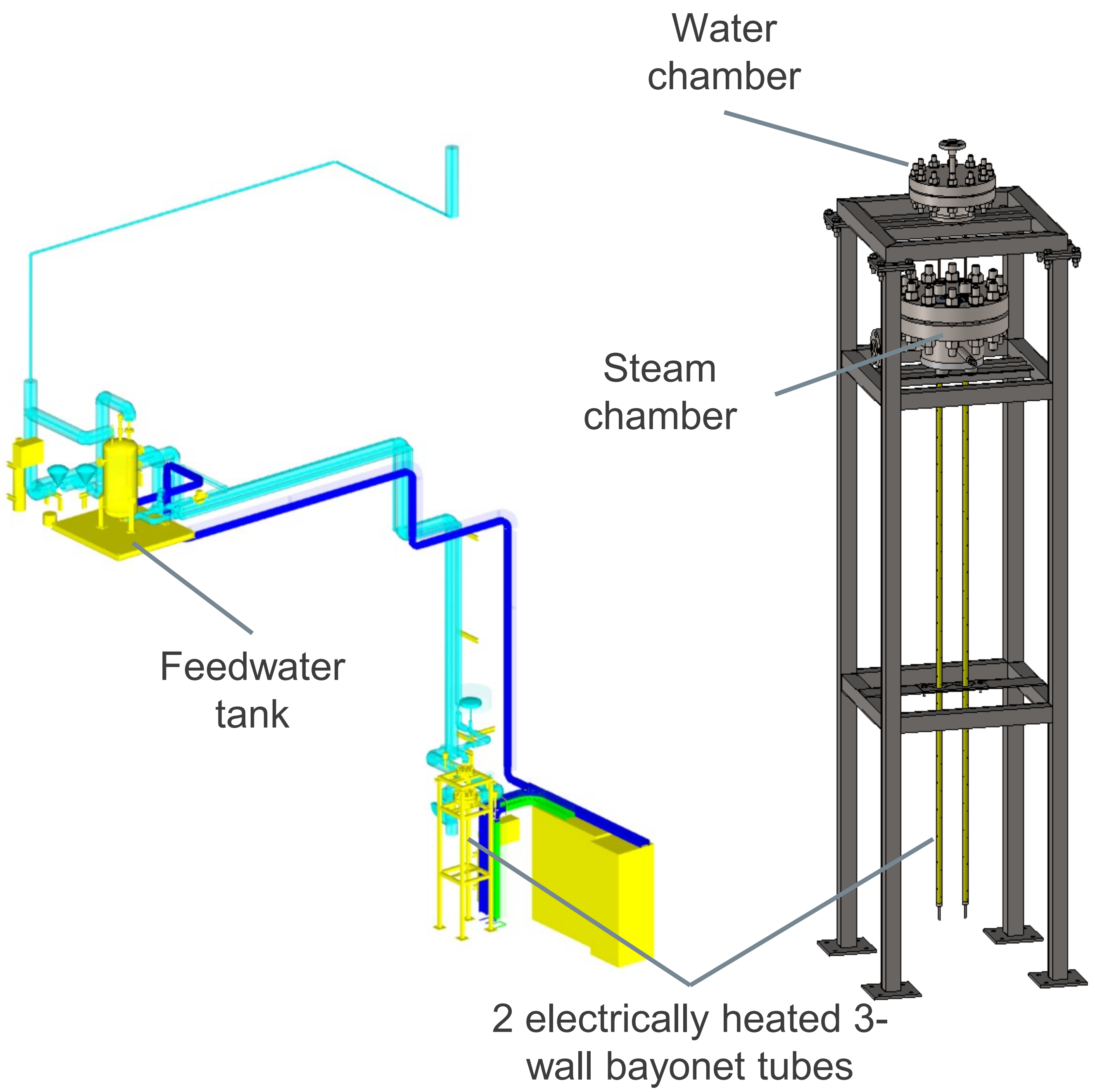


Safety Demonstration of Residual Heat Removal Provisions

| DCI (Dip Cooler Instability) Test Facility @ Polytechnic of Turin

DCI: experimental loop to investigate T/H behaviour of **DHR heat exchanger** (“**dip cooler**”), notably **start-up transient** and parallel channel **two-phase flow instabilities**

- **Test section**:
  - ✓ Two full-length bayonet tubes HX, connected to a water header and a steam header
  - ✓ Fed by water tank @ several meter elevation
  - ✓ Equipped with electrical heaters to achieve uniform outer wall temperature
  - ✓ Water flows downward inside inner tube, boils in annular gap between inner and intermediate tube, and exits as superheated steam
  - ✓ Helium fills gap between intermediate and outer tube
- System designed to enable tests:
  - ✓ @ different **pressure**, **temperature** and **flow rate**
  - ✓ In **open** and **closed-loop** configuration

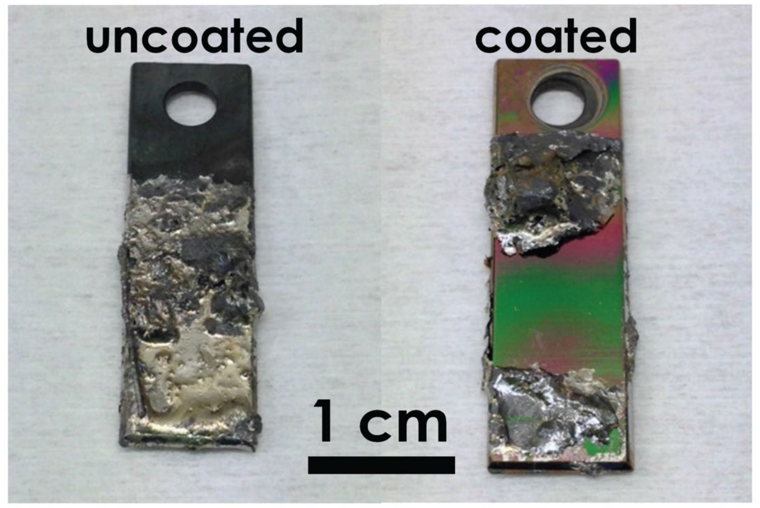


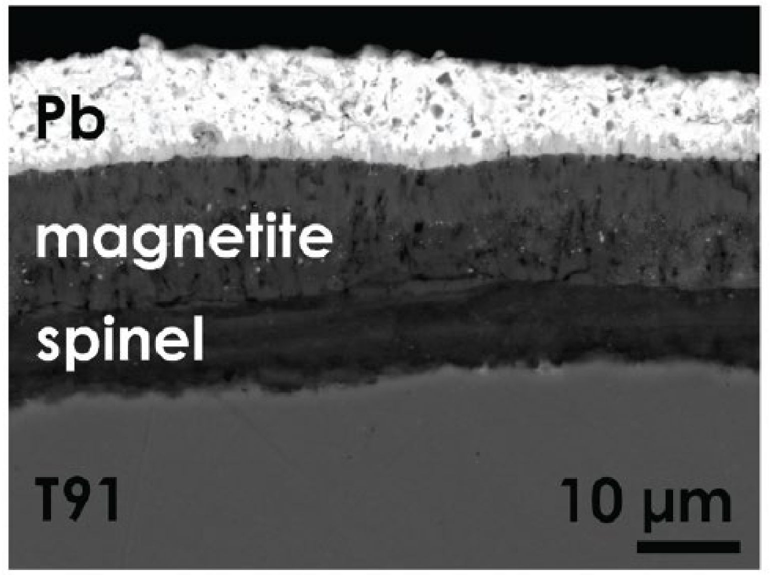
Confinement & Radiation Protection

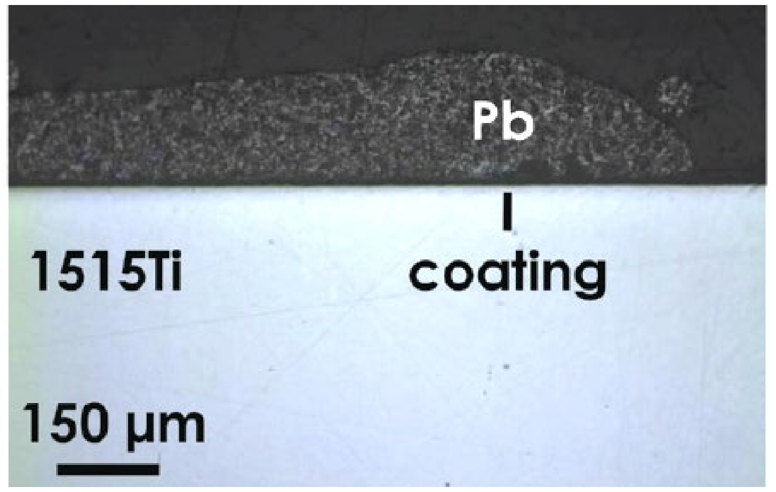
Safety Demonstration of Confinement Provisions

| Material Characterisation

- Corrosion test in flowing lead
- OCS testing in loop
- Component qualification
- Instrumentation Test



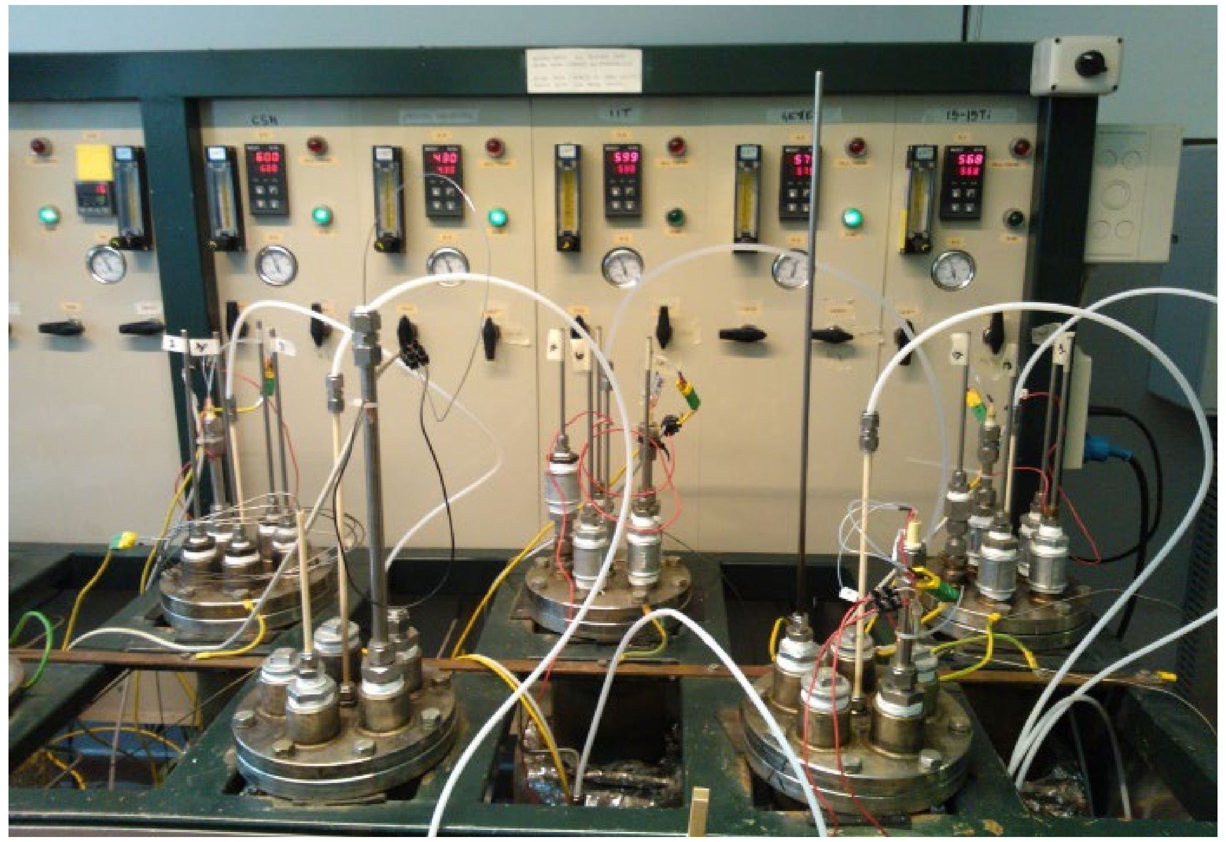




Safety Demonstration of Confinement Provisions

| Coolant Chemistry

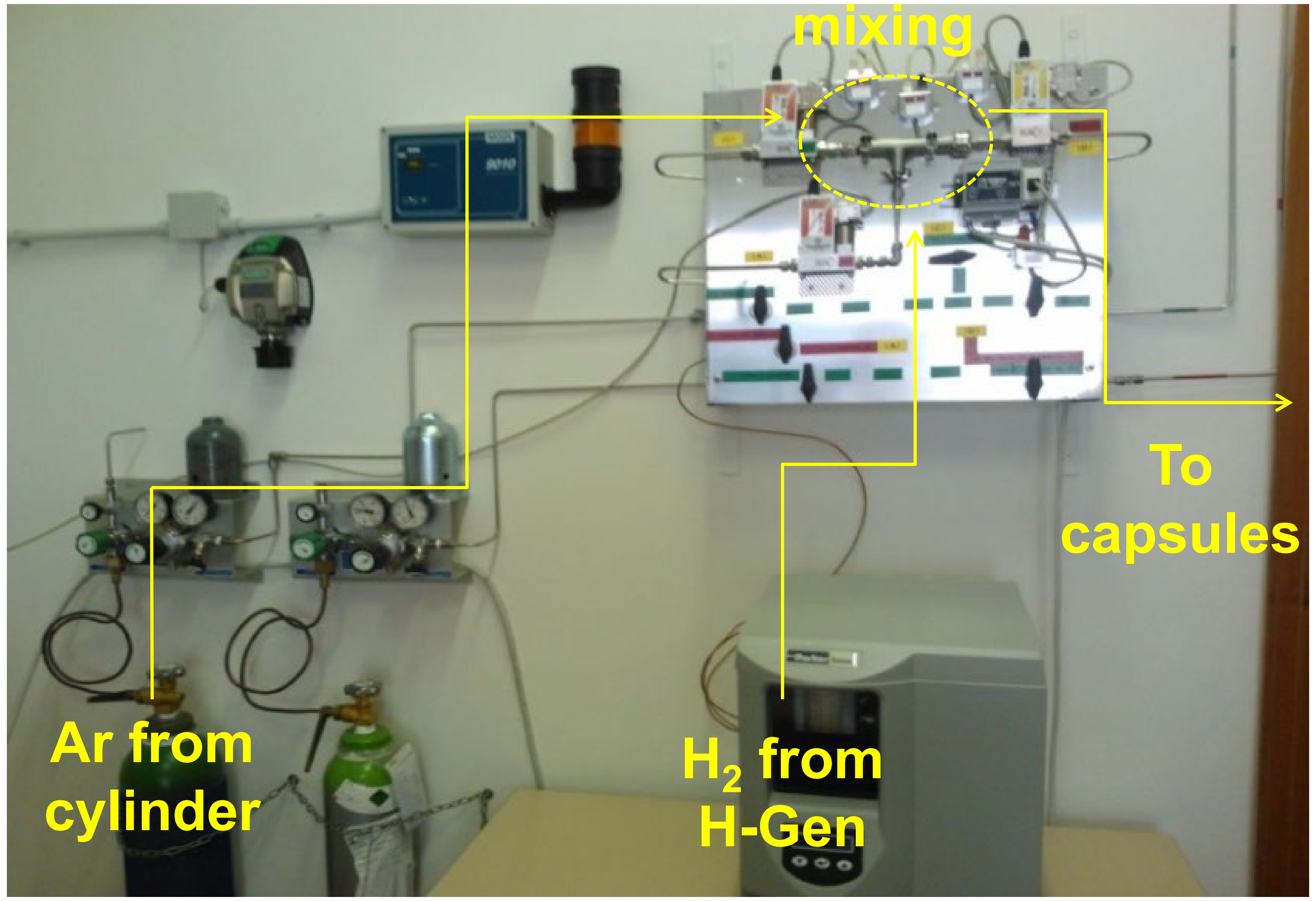
capsules for HLM chemistry (oxygen sensor testing, deoxygenation with gas) & corrosion tests of materials in Pb alloys



Stagnant Test up to 750°C

BID-ONE

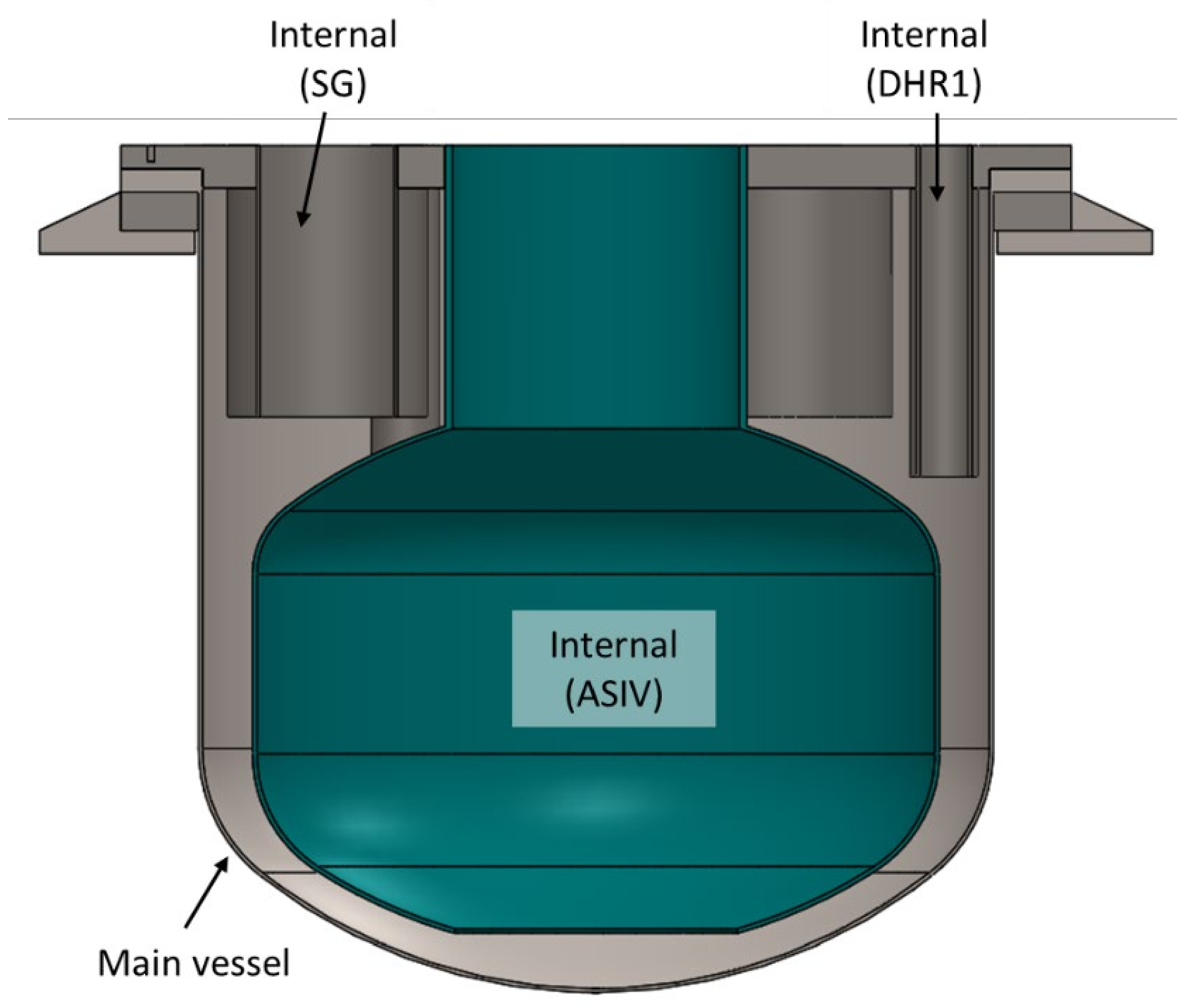
Gas control system (Ar-H_2 injection)

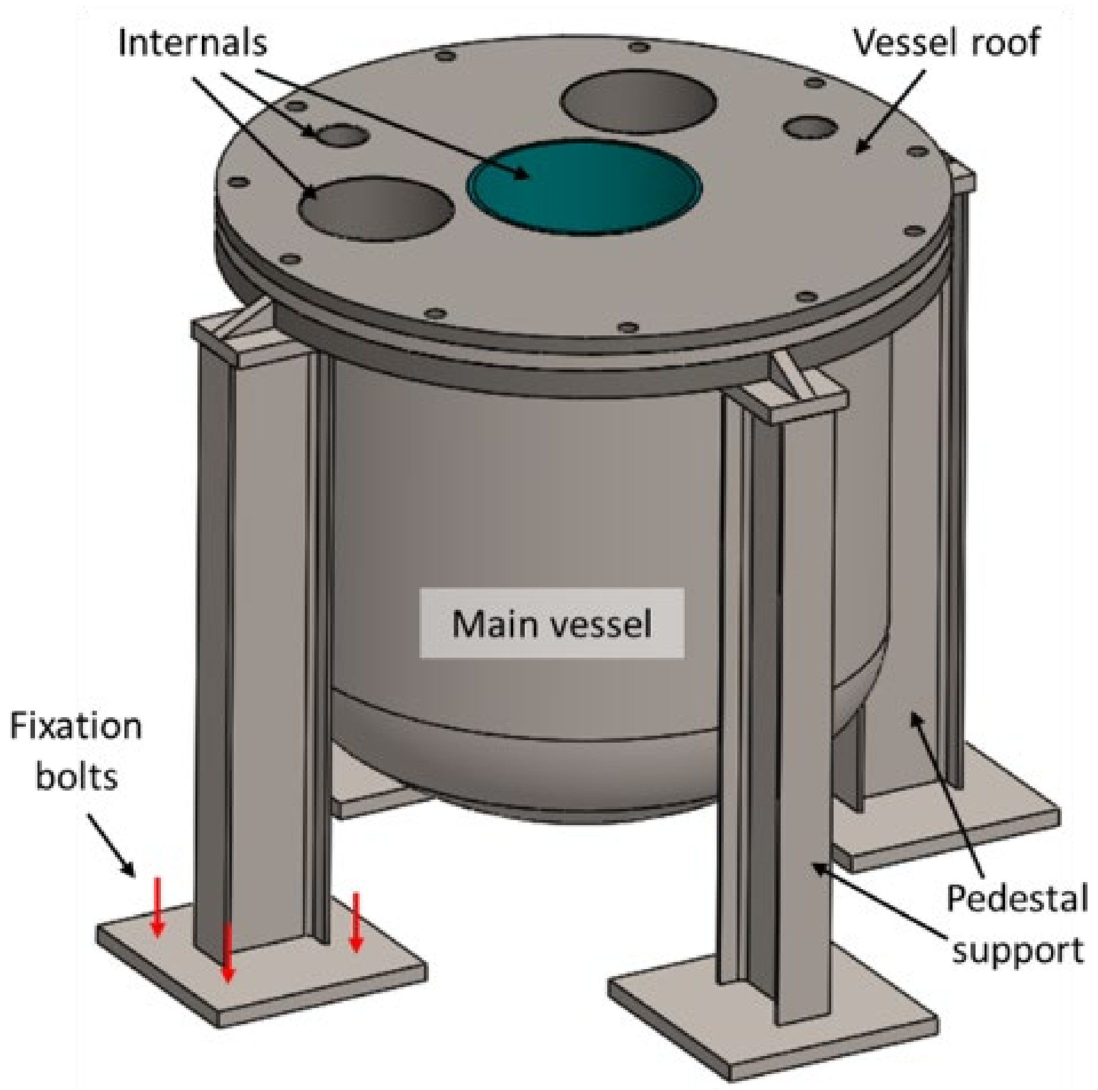


Safety Demonstration of Confinement Provisions

| EFESTO (Experimental FSI for Earthquake and Sloshing Test Observation) @ ENEA Casaccia

- Validation of calculation methods used for structural integrity justification
- Investigation of **Fluid-Structure Interaction** phenomena (e.g., sloshing impact on reactor vessel and internal structures)
- Provide a test bench for in-Pb instrumentation and in-service inspection sensors and strategies





Thank you