

LFR Technology Development

Jun Liao

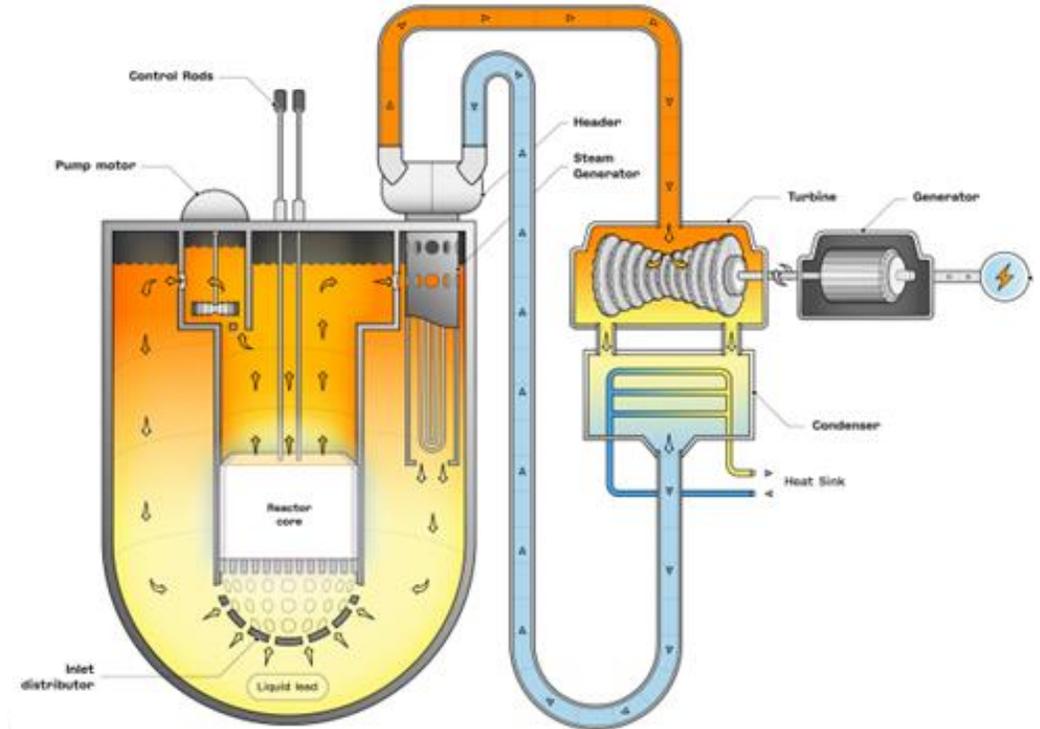
Westinghouse Electric Company, Cranberry Township, PA, USA

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Outlines

- LFR Technology: Benefits and Issues, Development Activities in USA
- LFR Safety: Safety System, Testing and Analysis
- LFR Safety: Regulations, and safety principles, licensing interaction



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Part II: LFR Safety: Safety System, Testing and Analysis

Safety Objective of LFR

Excellent Safety Features of LFR

- Enhanced passive safety systems
- IAEA passive safety category B for key systems
- Intrinsic safety of lead coolant:
 - High boiling point
 - Atmospheric pressure operation
 - Lack of exothermic reactions
 - Radionuclide retention capability
 - Excellent gamma shielding and neutronic properties



Implement Safety Guidance in Design

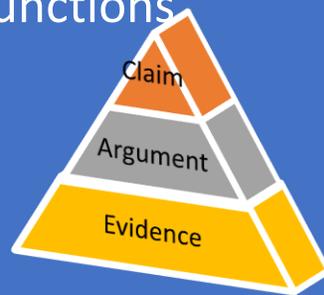
- US NRC Regulations
 - 10 CFR 50, 52, 53, LMP
- UK ONR SAPs and TAGs
- IAEA Safety Requirements



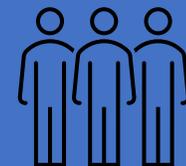
Superior Safety Performance

Develop a Strong Safety Case

- Ensure fundamental safety functions
- Implement defense in depth
- Robust safety case structure
- Engagement with regulators



Organization's Safety Culture

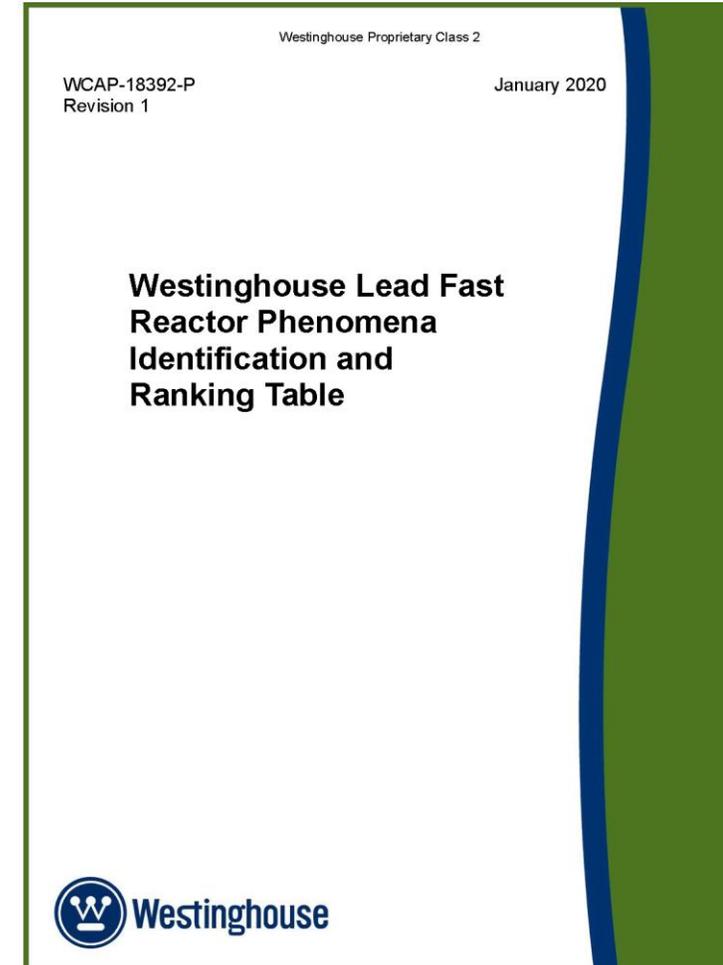
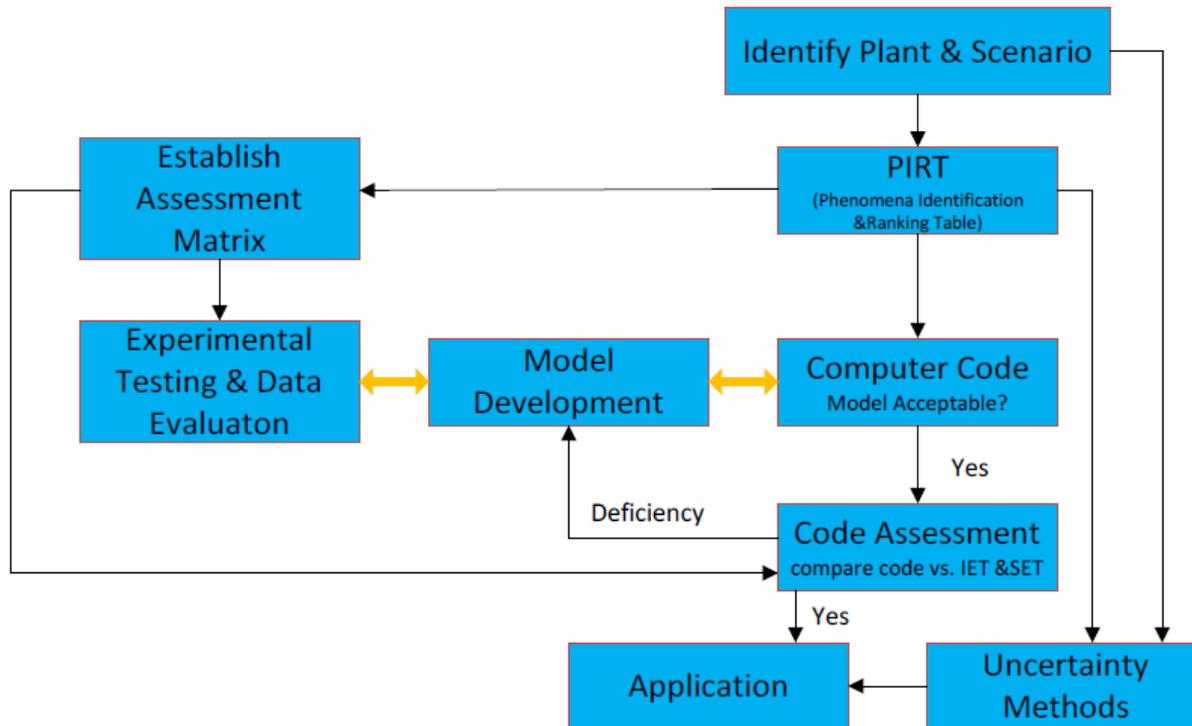


Principal Safety Functions

- Three principal safety functions:
 - Reactivity Control
 - Heat Removal from Core
 - Contain the radioactive material
- How are these objectives accomplished in various reactors?
 - Pressurized water reactor
 - Boiling water reactor
 - Candu heavy water reactor
 - LFR
 - ...

Testing and Safety Analysis Development

- US NRC developed EMDAP and PIRT to guide safety analysis development
- Evaluation Model Development and Assessment Process (EMDAP)
- Phenomena Identification and Ranking Table (PIRT) defines requirements for modeling and analysis tool and necessity of experimental data.

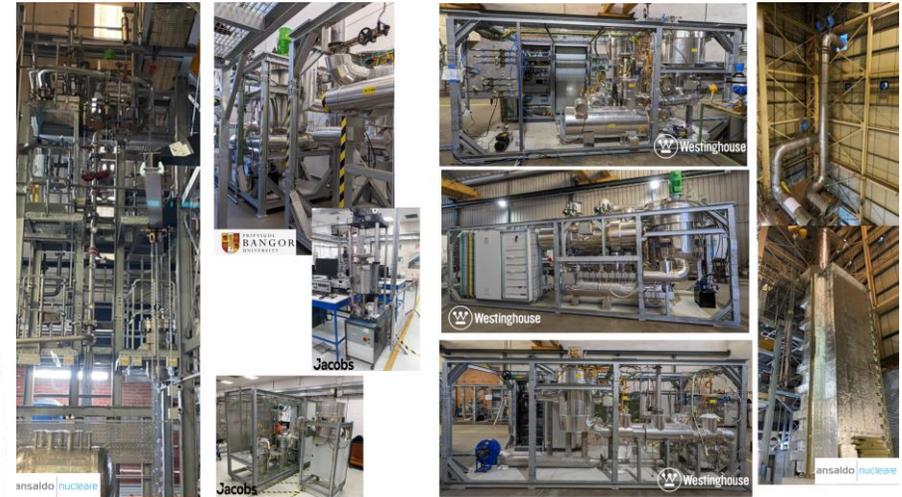


**Westinghouse LFR Safety PIRT guides
prioritization of testing**

Westinghouse Testing Program on LFR technology

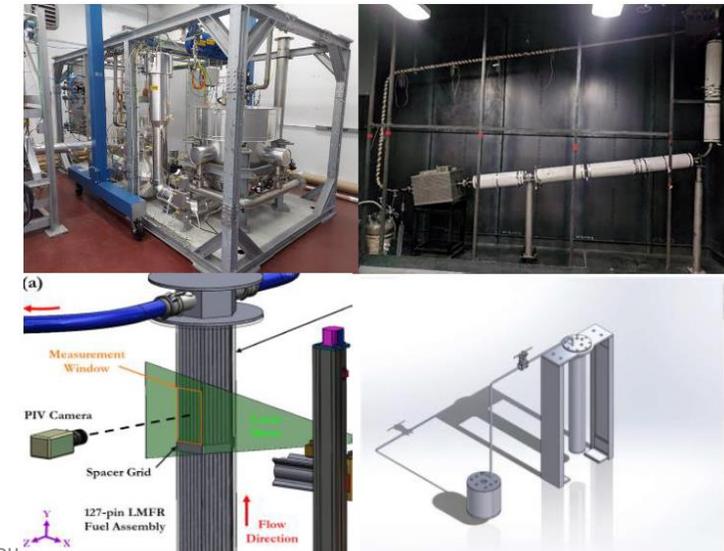
Testing program in the UK:

- Eight state-of-the-art LFR test facilities built between 2021-2023 in collaboration with nine international partners
 - Material testing (corrosion, erosion, mechanical properties)
 - Passive Heat Removal System testing
 - Component testing (rod bundle, primary heat exchanger)
 - Phenomena testing (freezing, heat exchanger failure)
- Co-funded by the UK Government



Testing program in the US:

- **Westinghouse (Churchill, PA):** testing of mechanical properties of materials when immersed in liquid Pb
- **University of Pittsburgh:** multi-purpose lead test rig (CORRERE)
- **Virginia Tech:** Fission Product Retention Facility (RELIEF) for measuring radionuclides retention capability of lead
- **Univ. of New Mexico:** Materials corrosion tests with high temperature lead in a loop configuration (LOBO loop)
- **Texas A&M Univ.:** Rod bundle testing in isothermal conditions for CFD validation



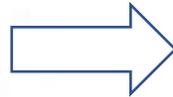
Corrosion/Erosion Test Facilities: Stagnant Rig, BULLET, MELECOR

Three corrosion/erosion facilities to comprehensively assess materials' corrosion/erosion performance. All oxygen-controlled. Complementary in operating conditions relative to Pb velocity and temperature

Static Pb corrosion rig



Step up in flow



BULLET: high-velocity
corrosion/erosion test loop



Step up in
temperature



MELECOR: high-temperature
corrosion test loop



Phenomena Testing Facilities

LEFREEZ: LEad FREEZing test facility

LEWIN: LEad-to-Water INteraction facility (a.k.a. heat exchanger failure test facility)



LEFREEZ

- Mission: assess effect of Pb freezing on immersed and surrounding structures
- “Untrapped” and “trapped” Pb test sections
- V&V of CFD modeling of freezing front propagation
- Designed to also host Under-Lead Viewing technology tests



LEWIN

- Mission: assess mechanical and fluid dynamics effects resulting from microchannel HX failure in liquid Pb
- HX failure reproduced through pre-set rupture of capillary tubes representative of failed microchannel
- Injection of 330 bar supercritical water in liquid Pb
- V&V of computer codes

Passive Heat Removal Facility (PHRF)

➤ The PHRF is to demonstrate the LFR's Passive Heat Removal System (PHRS)

- PHRS: annular pool of water surrounding the guard vessel, transitioning to indefinite air-cooling once water is depleted
- PHRF is 1:1 scale in height (23 m tall), 1:10 scaled azimuthally
- Prototypical in materials and thicknesses
- 0.5 MW power, operating temperature up to 700°C.

Largest power and highest temperature facility among similar ones

➤ Focuses on T/H phenomena:

- Transition from water cooling to air cooling in PHRS
- Natural convection heat transfer of air in PHRS
- PHRS air stack performance

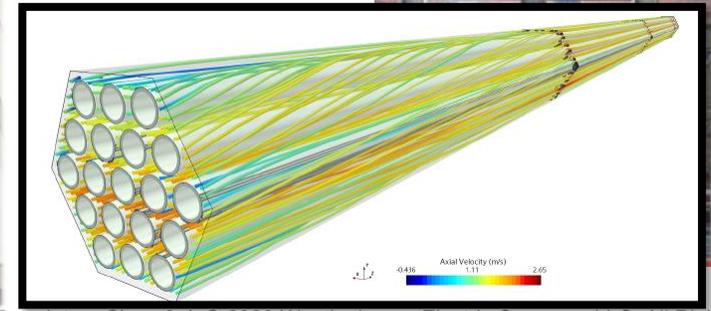
➤ 13 tests performed including forced convection, natural convection, and water-to-air transition

➤ Test results confirmed performance of the LFR's Passive Heat Removal System



Versatile Loop Facility (VLF)

- Large-size lead loop for component testing
 - Near-term mission: testing of rod bundle and primary heat exchanger
 - Facility's power: 0.5 MW
 - Lead Inventory: 3500 kg
 - Primary side: Liquid lead
 - Secondary side: Supercritical water
 - Footprint (W x D x H): ~ 5 x 5 x 10 m
- Primary heat exchanger mockup
- Fuel rod bundle mockup
- Facility installation is yet to completion.



Scaling Analysis for Experimental Facilities

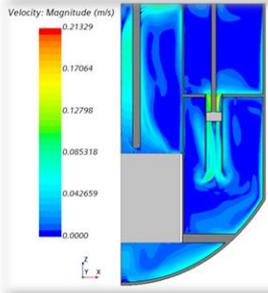
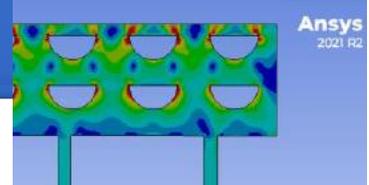
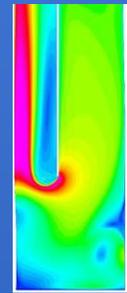
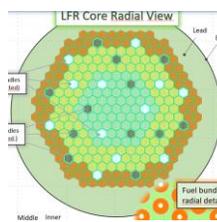
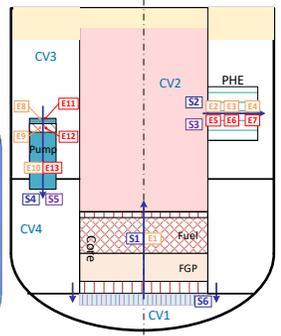
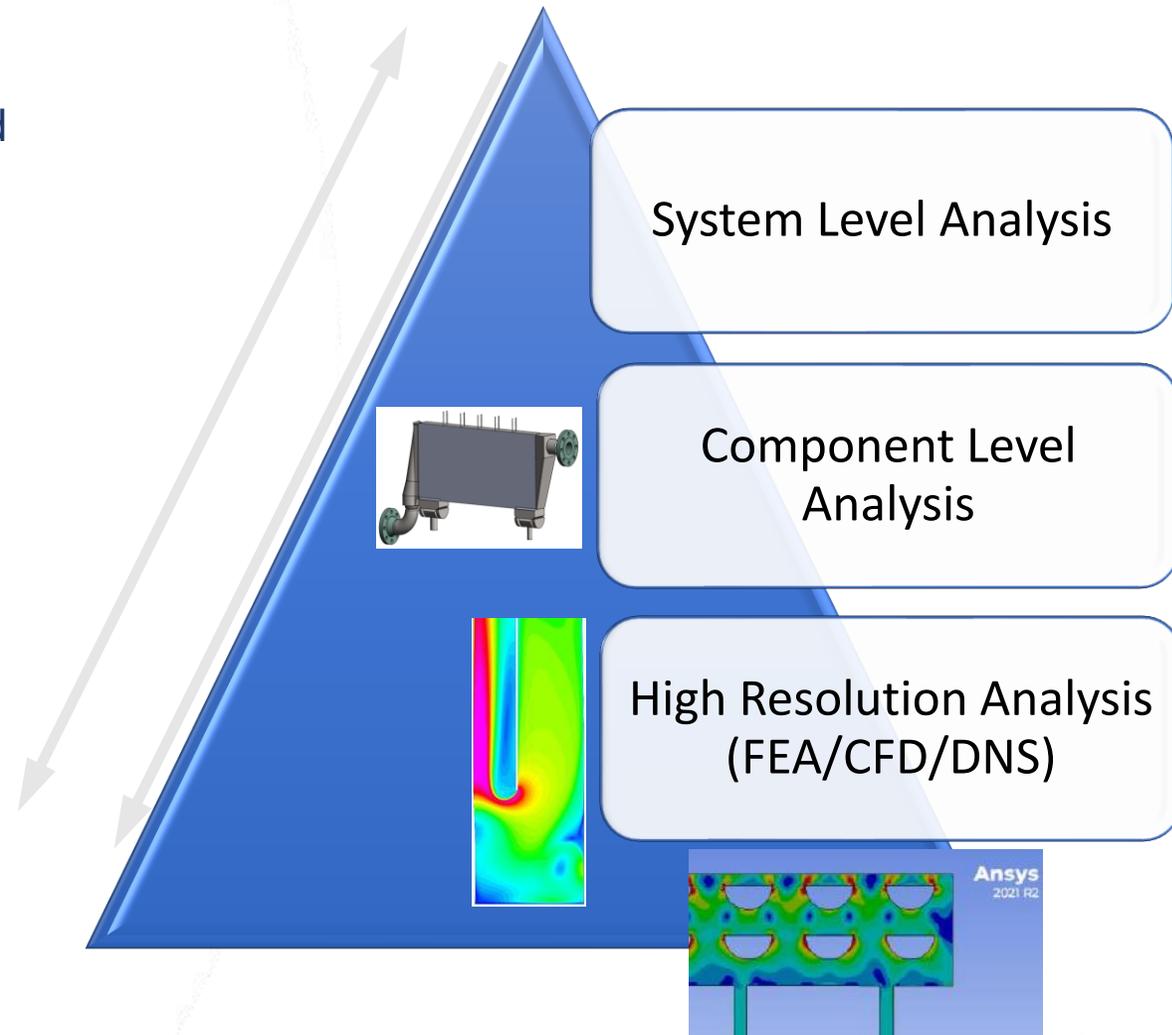
- Scaling methods for Nuclear Engineering
 - Π Theory (1914)
 - Linear Scaling Method (1965)
 - Power-Volume Scaling Method (1979)
 - **Hierarchical Two-Tiered Scaling Method (1998)**
 - Fractional Scaling Method (2005)
 - Dynamical System Scaling Method (2015)

- **Hierarchical Two-Tiered Scaling Method** is favorable for scaling analysis
 - Well-established scaling method and widely used in industry.
 - Extensive licensed experience with regulators.

Stage 1 SYSTEM DECOMPOSITION	Stage 2 SCALE IDENTIFICATION	Stage 3 TOP-DOWN SYSTEM SCALING ANALYSIS	Stage 4 BOTTOM-UP PROCESS SCALING ANALYSIS
PROVIDE: System hierarchy IDENTIFY: Characteristic: Concentrations Geometries Processes	PROVIDE HIERARCHY FOR: Volumetric concentrations Area concentrations Process time scales	PROVIDE: Conservation equations DERIVE: Scaling groups and characteristic time ratios ESTABLISH: Scaling hierarchy IDENTIFY: Important processes to be addressed in bottom-up process scaling analyses	PERFORM: Detailed scaling analysis for important processes DERIVE AND VALIDATE: Scaling groups

Hierarchy of Modeling and Analysis

- Multiphysics: thermal-hydraulics, reactor physics, structure, source term, etc.
- Phenomena involve a wide range of length and time scales.
- Multiscale/multi-resolution simulation hierarchy is needed.
- The LFR demands a collection of software
 - Rapid development
 - Low maintenance cost
 - Versatile applications
 - High resolutions
 - Coupled simulations
- These analyses support the reactor design, testing and licensing.



Modeling and Analysis Tools

➤ System Level Analysis Tools

- SAS4A/SASSYS-1
 - Liquid metal reactor safety analysis tool by ANL
- GOTHIC
 - General purpose T/H analysis tool by EPRI/NAI
- FATE
 - Mechanistic source term analysis tool by FAI
- RELAP5 MOD3.3
 - General purpose T/H analysis tool by NRC and INL
- SAM
 - Corrosion analysis tool by ANL

➤ Component Level Analysis Tools

- ARC/PyARC
 - Core design and analysis tool by ANL
- ANTEO+
 - Subchannel analysis tool by ENEA
- TRANSURANUS
 - Fuel performance tool by ITU
- SAS4A
 - Fuel performance tool by ANL
- Serpent
 - Radiation analysis tool

➤ High Resolution Analysis

- Finite element analysis
 - ANSYS Mechanical
 - ANSYS Workbench
- Computational fluid dynamics
 - CFX - ANSYS
 - STAR-CCM+ - Siemens

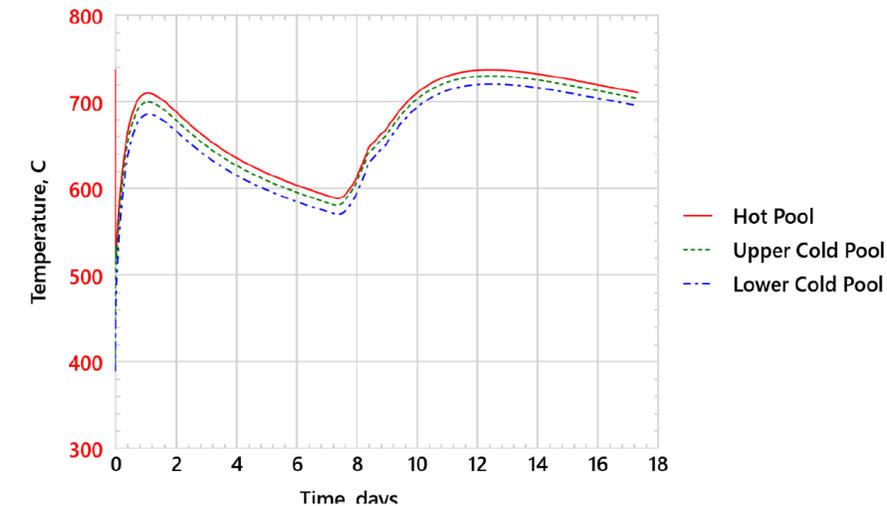
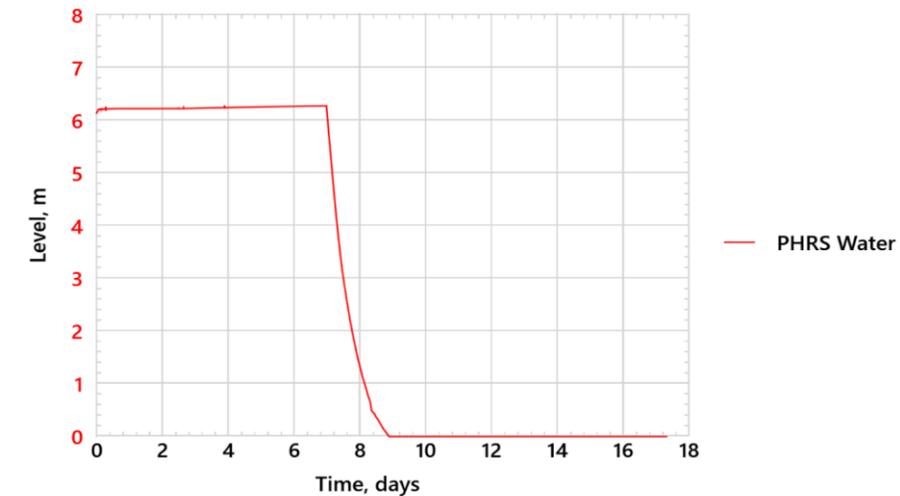
Focus of M&S so far has been primarily on application of reliable/ practical design tools, suited for this phase of LFR design development

Advanced M&S is of interest for LFR to:

- Validate or extend current analysis capabilities
- Demonstrate margins
- Fill M&S gaps in the current tools

Deterministic Safety Analysis

- Coupled SAS4A/SASSYS-1-GOTHIC simulation
- Accident analysis of Station Blackout (SBO) and Transient Overpower (TOP) were demonstrated with SAS-GOTHIC simulations⁽¹⁾
- Enhancing computer code applicability to LFR⁽²⁾
 - Verification and validation of SAS4A/SASSYS-1 and GOTHIC
 - Leverage existing SET and IETs and W-LFR test facilities
 - CIRCE-HERO, NACIE-UP, PHRF, VLF
- Support reactor design iterations
- Provide foundation for the LMP risk-informed safety analysis (next page)



1. Lee S.J., et al., 2022. Preliminary Assessment of the Safety Performance of Westinghouse LFR. IAEA Technical Meeting on State-of-the-Art Thermal Hydraulics of Fast Reactors, Brasimone, Italy.

2. Liao J., et al., 2022. Applicability Enhancement of SAS4A/SASSYS-1 Computer Code to Lead Fast Reactor Systems. IAEA Technical Meeting on State-of-the-Art Thermal Hydraulics of Fast Reactors, Brasimone,

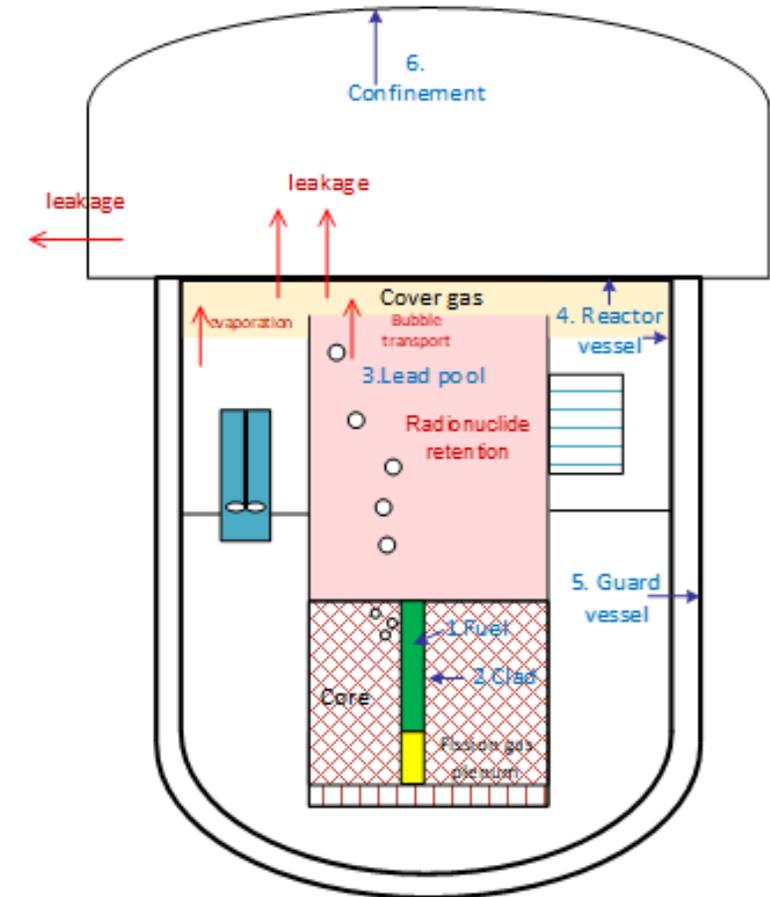
LFR LMP Development – Probabilistic Risk Assessment

- Probabilistic Risk Assessment (PRA) is an important tool for systematically and exhaustively evaluating and reducing the risk associated with a facility through the exploration of potential events and their consequences
- PRA techniques are used to define Licensing Basis Events (LBE) and their frequencies:
 - Initiating Event analysis
 - Event Tree analysis
 - Success and failure of systems that provide critical safety functions following initiation
 - Reactivity control
 - Heat removal from core
 - Radioactive material confinement
 - Fault Trees analysis
 - Quantify the probability of system failures
 - LBE Sequences defined by the paths on the event trees
- LBEs are analyzed for the identified accident sequences

Unlike traditional PRA, event sequences are not necessarily core damage scenarios

LFR LMP Development – Mechanistic Source Term

- Mechanistic Source Term (MST): magnitude, timing and composition (chemistry) of radioactive releases to the environment during accident scenarios
- Evaluation of initial radionuclide inventories and plant locations
 - Fission products in fuel pellets
 - Fission products in fuel-clad gap and plenum
 - Activation products in lead
 - Radionuclides in the cover gas
- Evaluation of attenuation factors provided by each barrier in functional containment of LFR
 - Fuel pellets, cladding, molten lead, reactor vessel, guard vessel, reactor building
- Evaluation of challenges to barriers integrity and timing of failures and releases
- Quantification of Mechanistic Source Term



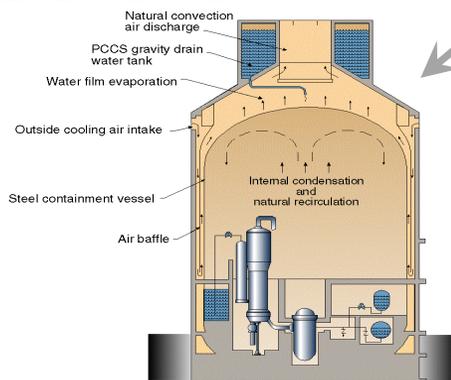
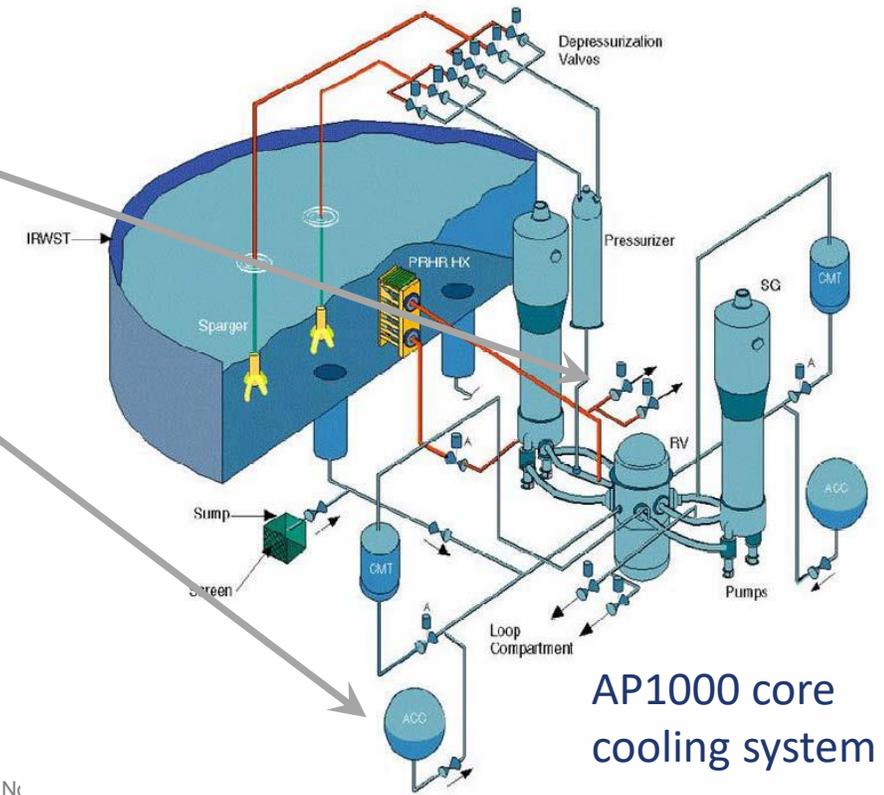
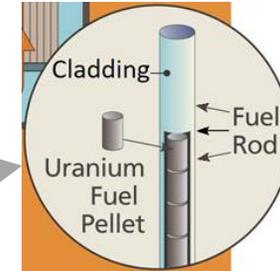
Model and Quantification of MST is performed by coupled SAS4A/SASSYS-1-GOTHIC-FATE simulations

Passive Heat Removal System

Passive Safety Category

The International Atomic Energy Agency (IAEA) classifies the degree of "passive safety" of systems/components according to what these systems/components do not use.

Characters of system	Category	Examples
No moving working fluid	A	Fuel clad/RV/containment
No moving mechanical part	B	PWR surge line condensation on heat sink
No signal inputs of 'intelligence'	C	Pressure relief valve PWR accumulator
No external power input or forces	D	Passive containment cooling system PWR emergency shutdown



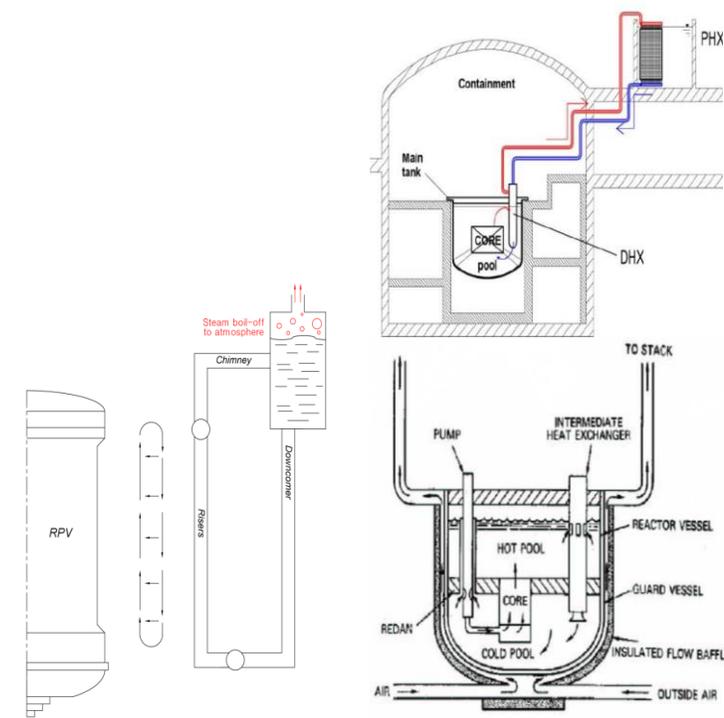
AP1000 containment cooling system

AP1000 core cooling system

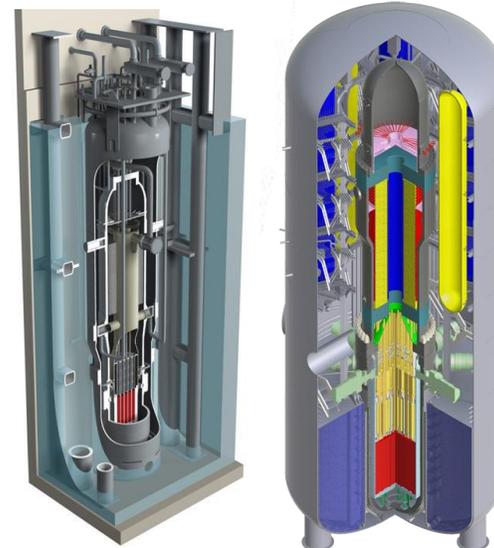
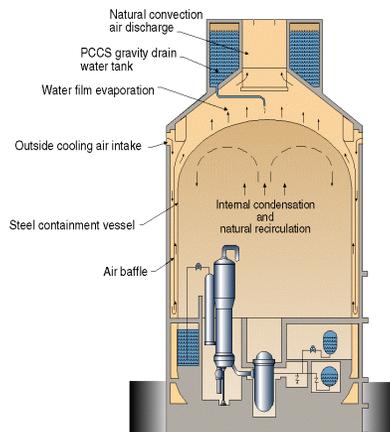
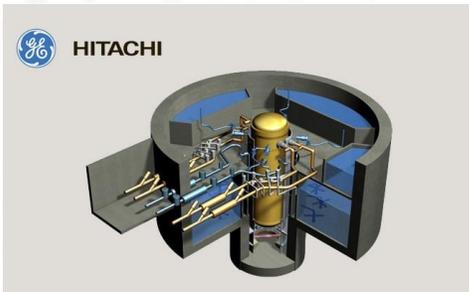
Passive Heat Removal in NPPs

- Limited passive safety in Generation II PWRs for large break LOCA.
 - Accumulator, ice condenser in containment
- Passive heat removal system is expanded to mitigate small break LOCA, non-LOCA and severe accident for Generation III+ LWR designs
 - AP600/AP1000 (1990s)
 - SBWR/ESBWR (1990s)
- 100% reliance on natural forces
 - Evaporation, condensation, gravity
- Passive safety design is widely accepted by industry, regulator, and public.

- Passive heat removal design is increasingly adopted by small modular reactors.
 - NuScale SMR
 - Holtec SMR
 - GE BWRX-300 (BWR)
 - AP300
- SMRs with passive heat removal performing primary safety function.



- Passive heat removal is widely adopted in Gen IV reactor designs
 - Reactor vessel auxiliary cooling system (Sodium Fast Reactor)
 - Reactor cavity cooling system (gas cooled reactor)
 - Isolation condenser decay heat removal system (lead fast reactor)
 - PHRS in LFR**
 - More



LFR Decay Heat Removal System

Normal decay heat removal: through primary HXs, turbine bypass, condenser.

Non safety grade

Emergency decay heat removal system requirements:

Safety-grade system

IAEA passive safety category B: *simple and reliable*

Capable to remove decay heat of high power LFR core

Capable of extended long term heat removal (indefinite)

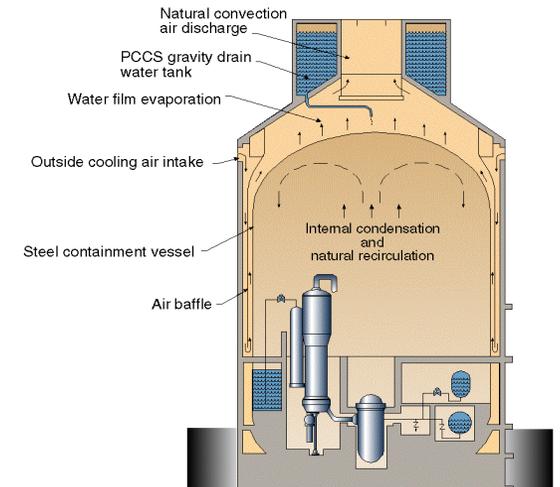
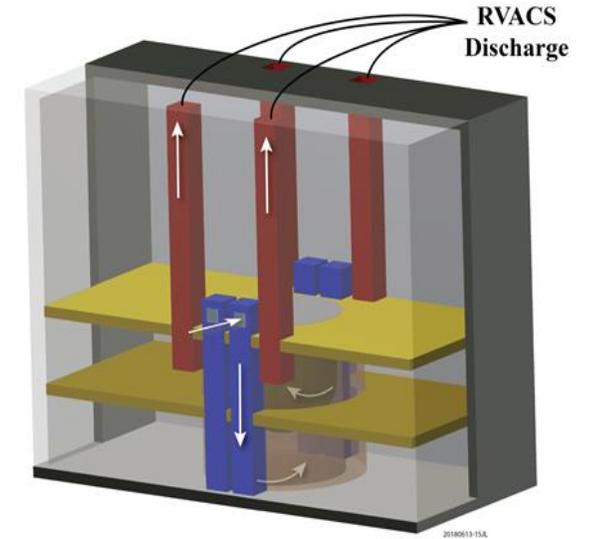
Leverage knowledge of previous nuclear power plant design

AP1000® passive containment cooling system

SFR reactor vessel auxiliary cooling system

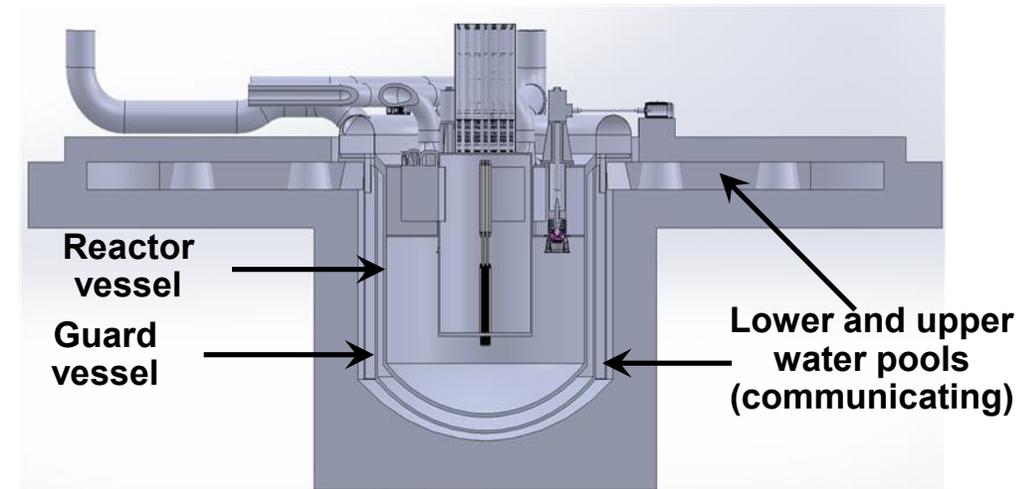
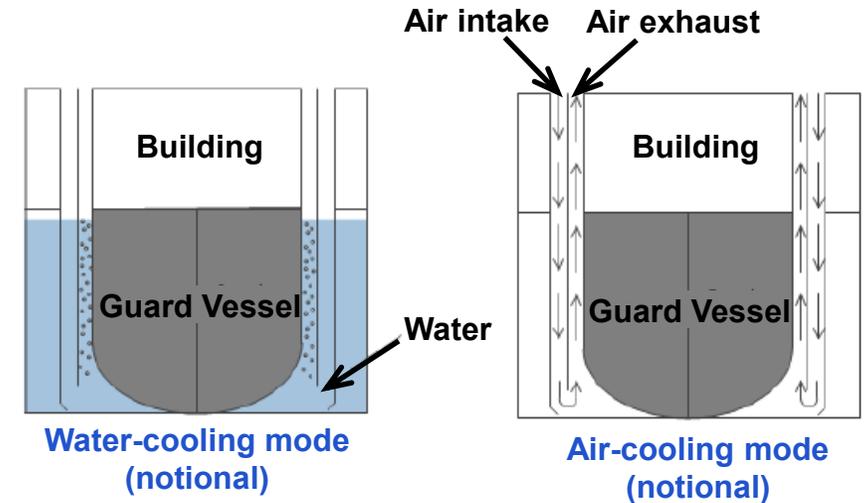


LFR Passive Heat Removal System (PHRS) design is innovative but based on previously proven technical basis.



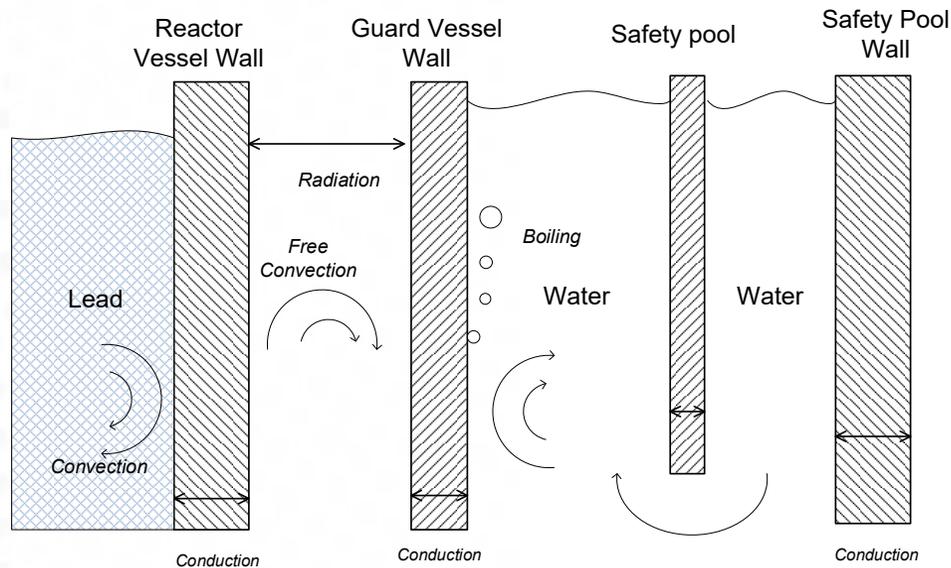
Passive Heat Removal System (PHRS) Design

- **LFR PHRS design rationale**
 - Water cooling is applied to address early cooling challenge due to high power reactor core.
 - Air cooling is used as ultimate heat sink as air supply is unlimited.
- **LFR PHRS design features**
 - Pool of water surrounds Guard Vessel.
 - Water pool is monitored and maintained during normal operation.
 - In accident, water cooling is provided for 7 days
 - Transition to (indefinite) air-cooling in extended long-term cooling
 - IAEA passive category B
 - No moving part
 - No I&C support, no need for actuation
 - No need for external power
 - System always on
 - Parasitic heat loss is addressed in the engineering design.

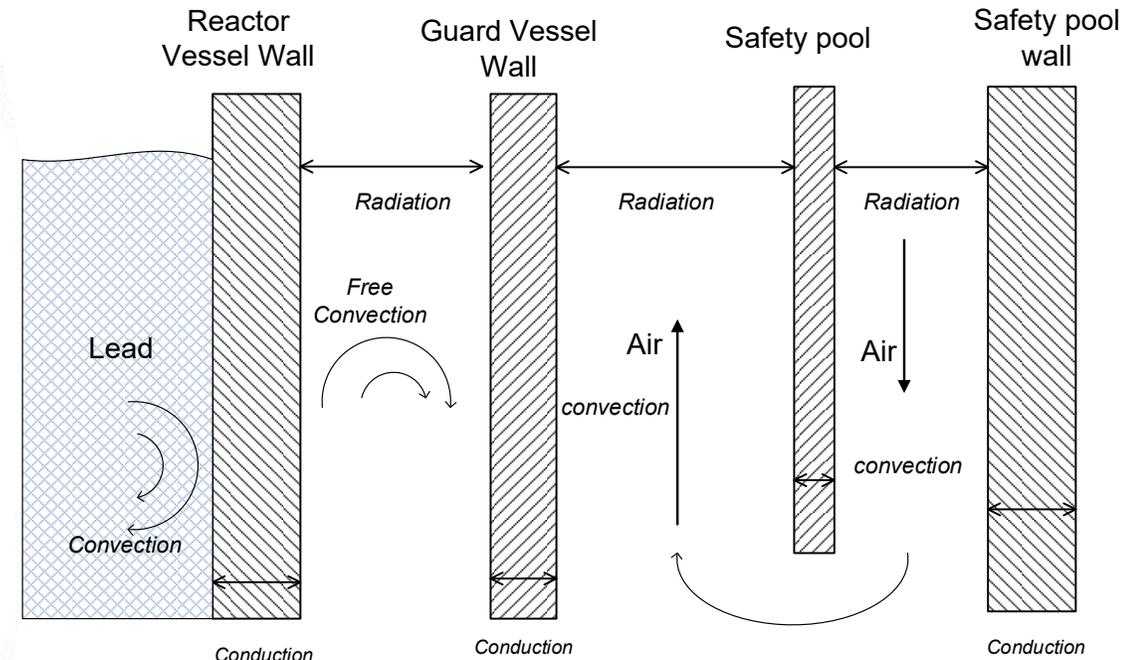


Passive Heat Removal System (PHRS) T/H

- Important T/H phenomena in PHRS are identified in Westinghouse LFR PIRT (phenomena identification and ranking table)
- Buoyancy-driven natural circulation
- Heat transfer phenomena
 - Convective heat transfer
 - Conduction heat transfer
 - Radiation heat transfer
 - Very low during normal operation but kicks in during transients when temperature increases.



Water Cooling



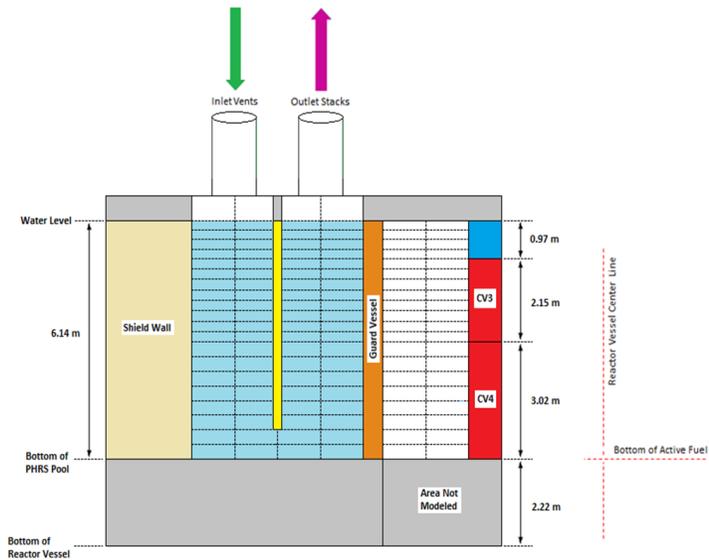
Air Cooling

Demonstration of PHRS is supported by analysis and testing.

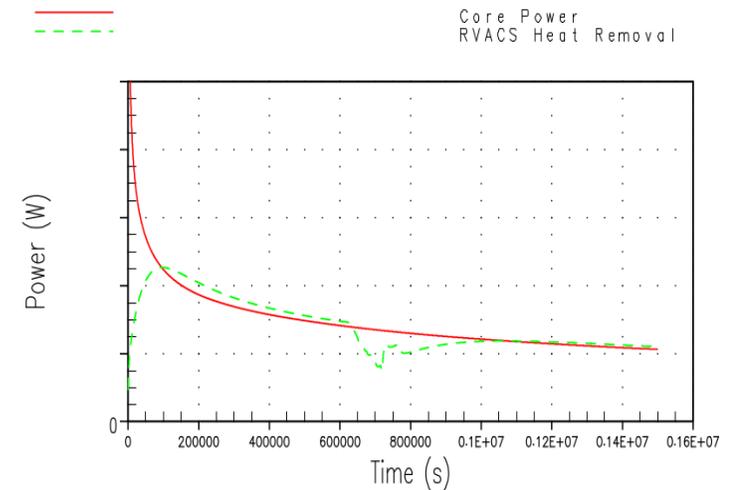
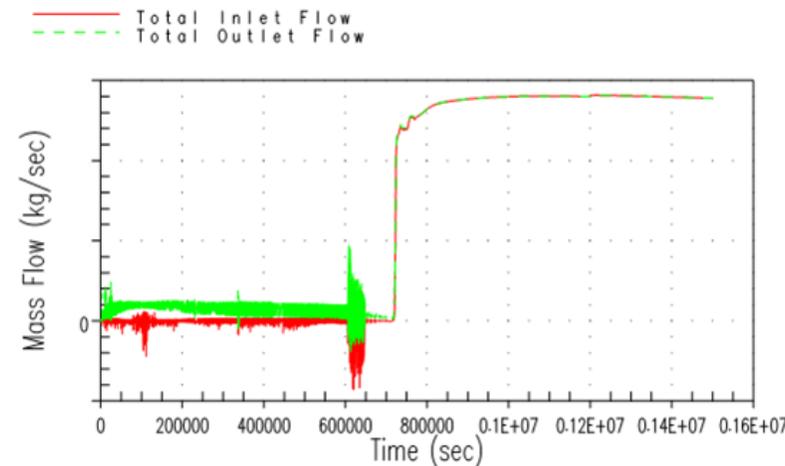
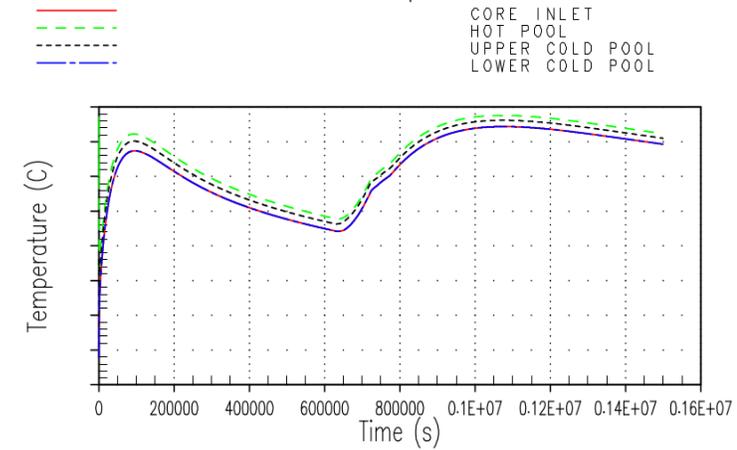
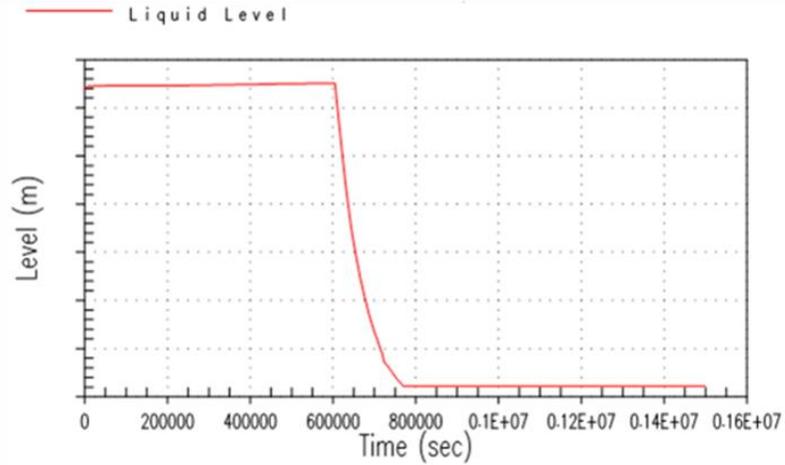
PHRS Performance Analysis

- PHRS is modeled by GOTHIC computer code
 - GOTHIC is a high pedigree system/containment analysis computer code.
- GOTHIC is coupled with SAS4A/SASSYS-1 to perform LFR safety analysis.

Station Blackout Transient

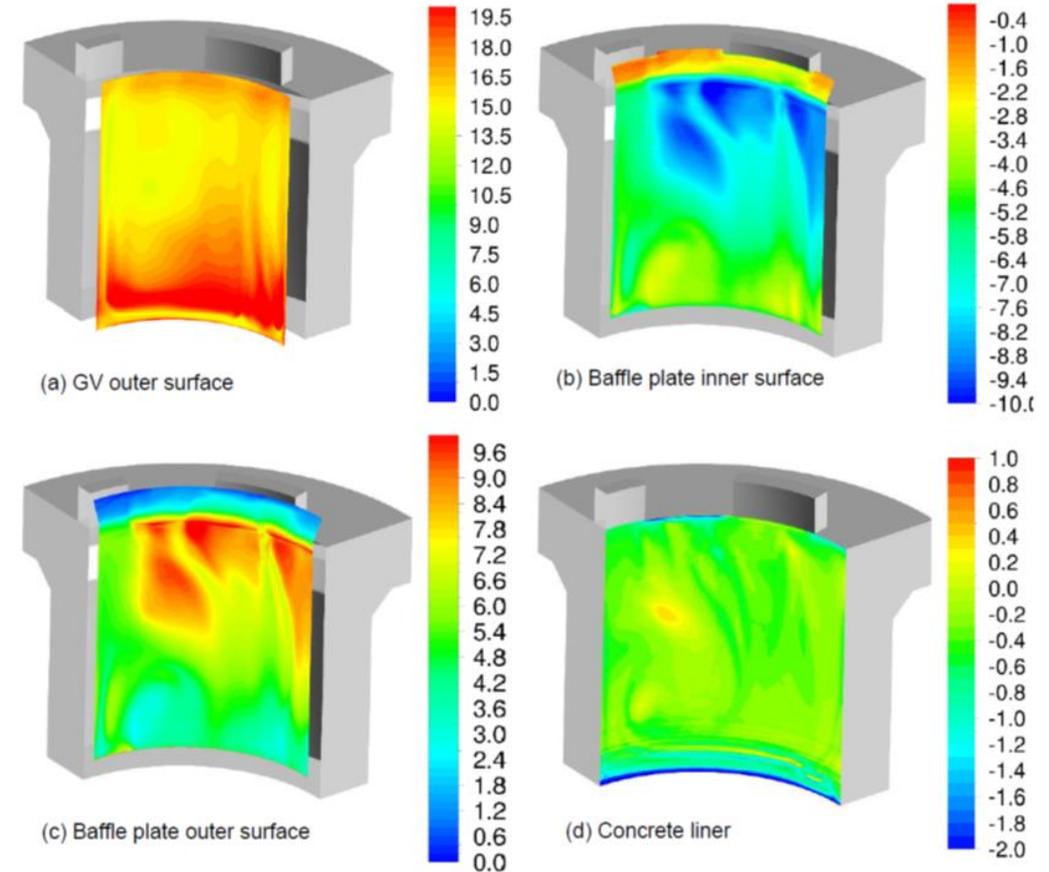
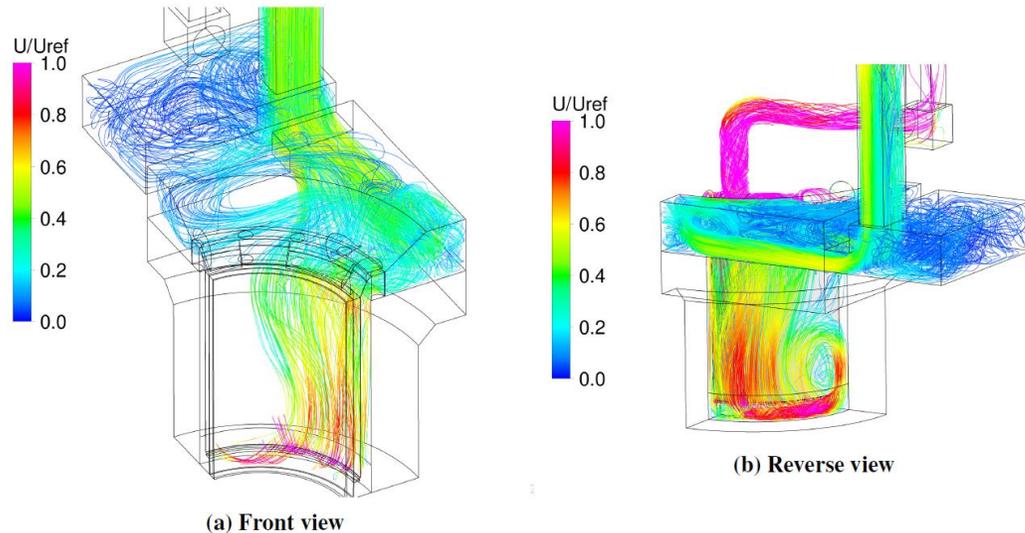


PHRS:GOTHIC



High Fidelity CFD Analysis of PHRS

- Performance of PHRS was studied by Frazer-Nash using ANSYS CFX.
- Study was focused on air cooling phase.
- ANSYS CFX simulation was benchmarked against simulation by GOTHIC.
- The study assesses and improves the PHRS design.



Demonstration of LFR's Passive Heat Removal System - PHRF

➤ Passive Heat Removal Facility for demonstrating the LFR's PHRS

- 1:1 scale in height (23 m tall) , 1:10 scaled azimuthally.
- Prototypical in materials and thicknesses
- 0.5 MW power, operating temperature up to 700°C.

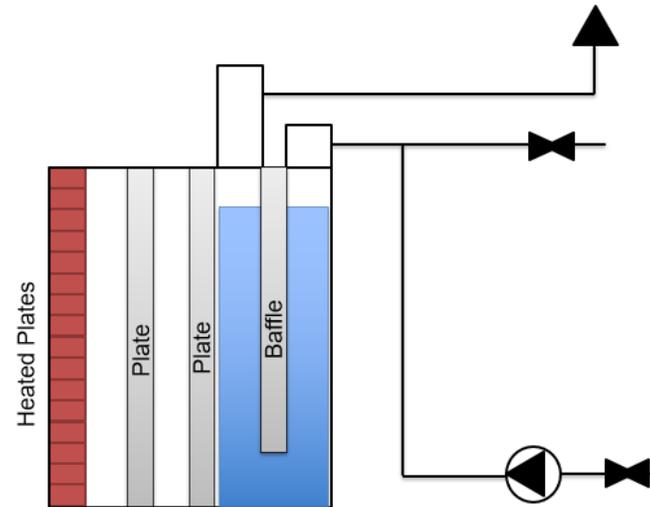
Largest power and highest temperature in similar type facilities

➤ Focuses on T/H phenomena:

- Transition from water cooling to air cooling in PHRS,
- Natural convection heat transfer of air in PHRS,
- PHRS air stack performance.

➤ 13 tests were performed including forced convection, natural convection, and transition at 400°C, 550°C, and 700°C.

➤ The first phase test results demonstrated the capability of PHRS of LFR.

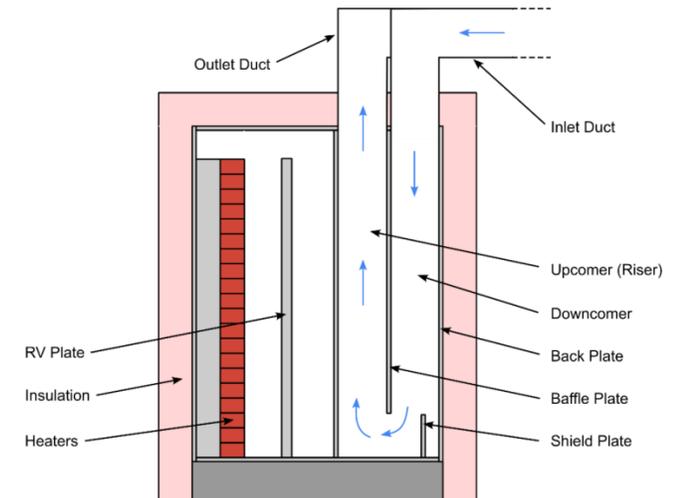
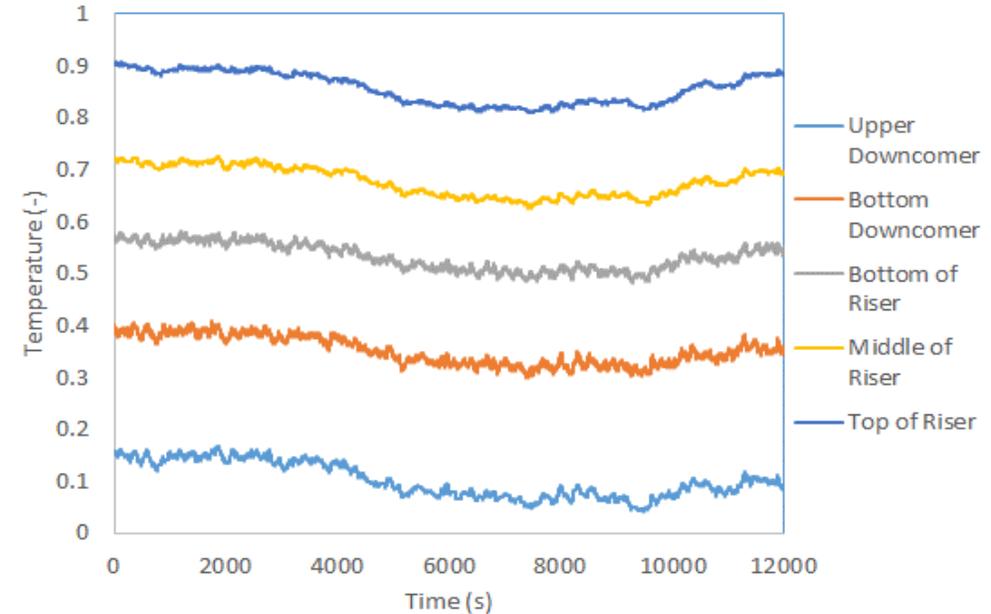


The PHRF Experiments

-Air Cooling Tests

- Once the PHRF reached steady state, the wall temperatures from the heating plate, reactor plate, to guard plate gradually decreased.
- The measured air temperature increased gradually from the inlet, the downcomer, the riser and the outlet.
- Some thermocouple temperatures in the cavity exceed the air outlet temperature, indicating the possible occurrence of circulation phenomena within the PHRF.

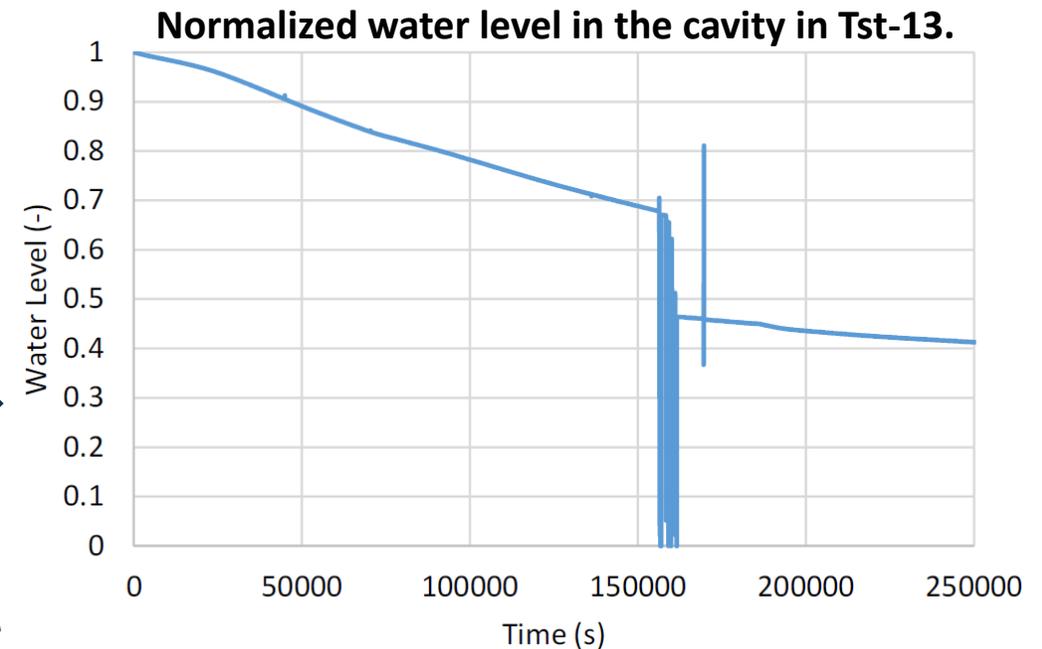
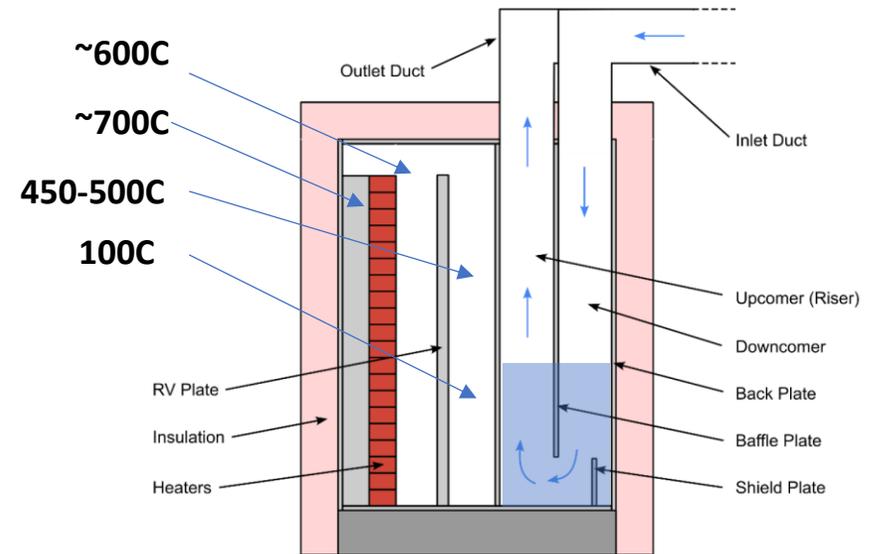
Normalized air temperature in the PHRF cavity in Tst-1



The PHRF Experiments

- Water-Air Transition Tests (1 of 2)

- Purpose: assess the water-cooling capability of PHRF and the transition from water-cooling to air-cooling, and provide experimental data for the transition phenomenon.
- The initial water level in the transition test was 3 m, which was located at near 1/3 height of the cavity.
- In Tst-13, heating plate temperature at 700°C, and reactor plate maintained a relatively uniform temperature of ~600°C.
- On the guard plate, thermocouples located above the water level registered values around 450-500°C, whereas those below showed temperatures slightly above 100°C, indicating the occurrence of boiling within the cavity.
- The measured water level in the cavity gradually decreased due to water boiling. Once the water level dropped near the bottom of the baffle plate, the water seal at the bottom of the box was cleared near 160,000 seconds (~44 hours)

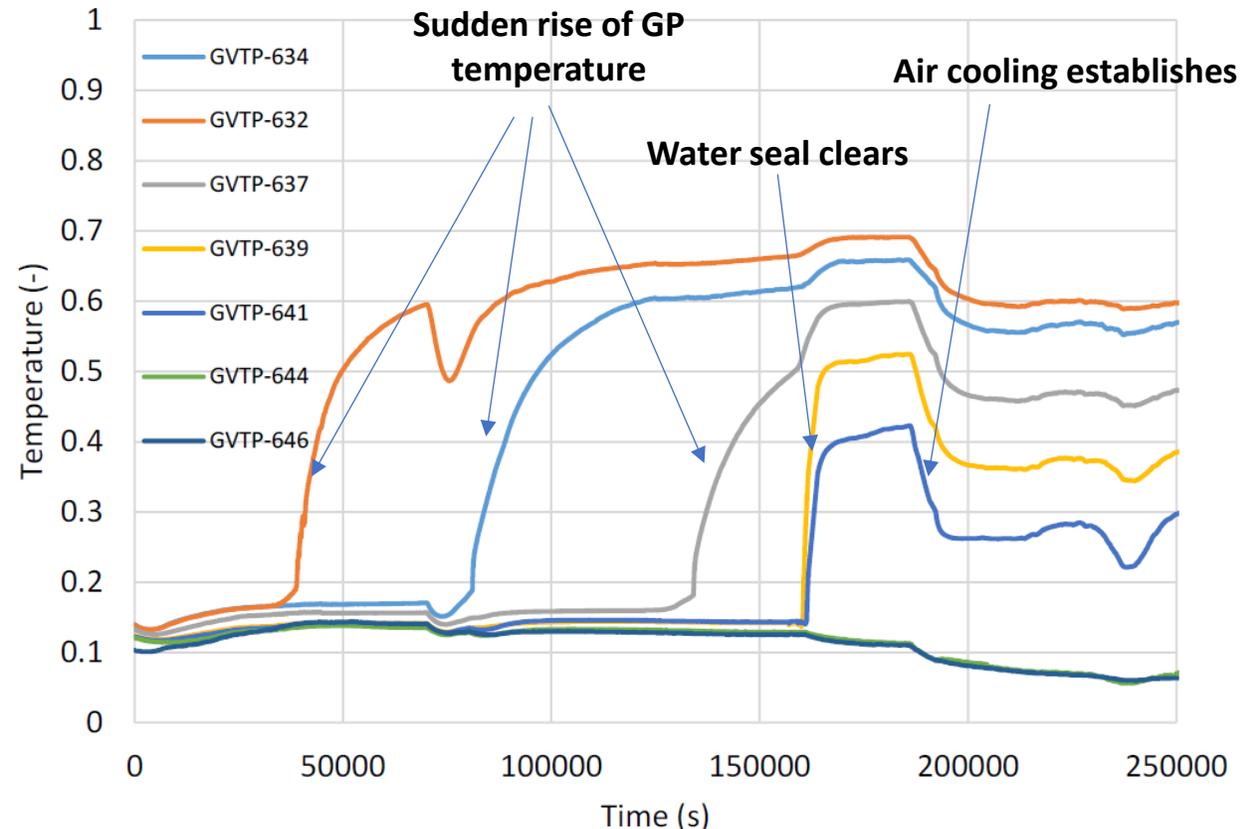


PHRF Experiments

- Water-Air Transition Tests (2 of 2)

- During the water boil off, when a thermocouple transitioned from water to air heat removal zone due to the decreasing water level, the sudden reduction in heat removal capability led to a quick rise of the guard plate temperature.
- Once the water level dropped below the bottom of the baffle plate, the water seal at the bottom of the box clears, and the air natural circulation establishes to provide the cooling to the guard plate.
- After ~ 180,000 s (50 hrs), the guard plate temperatures dropped to a steady-state with air cooling as the cooling mechanism. This indicated the successful transition from water- to air-cooling in the PHRF.
- During the transition period, the heat removal rate of the PHRF was relatively stable and the magnitude of heat removal rate was comparable to that during the air-cooling mode.

Normalized guard plate (GP) temperature on the cavity side (Tst-13)



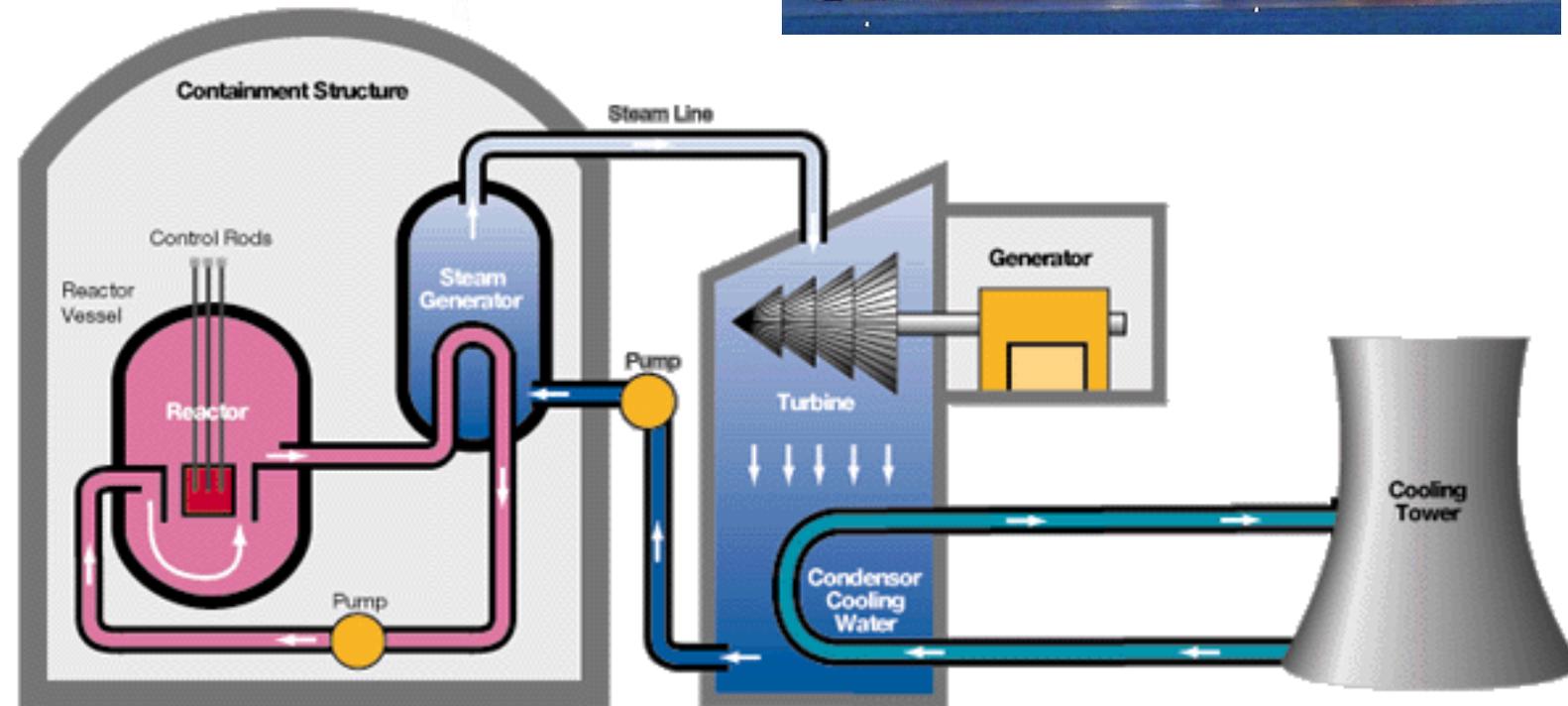
Functional Containment for LFR

What is Containment?

- Definition: a robust, airtight structure - usually made of reinforced concrete and steel- that encloses the reactor and associated systems to prevent the release of radioactive materials into the environment during normal operation and accident conditions.
- It is considered the final physical barrier in the defense-in-depth safety strategy, following the fuel cladding, reactor vessel.
- Safety functions (SSG-53)
 - (i) confinement of radioactive substances in operational states and in accident conditions;
 - (ii) protection of the reactor against natural external events and human induced events;
 - (iii) radiation shielding in operational states and in accident conditions.

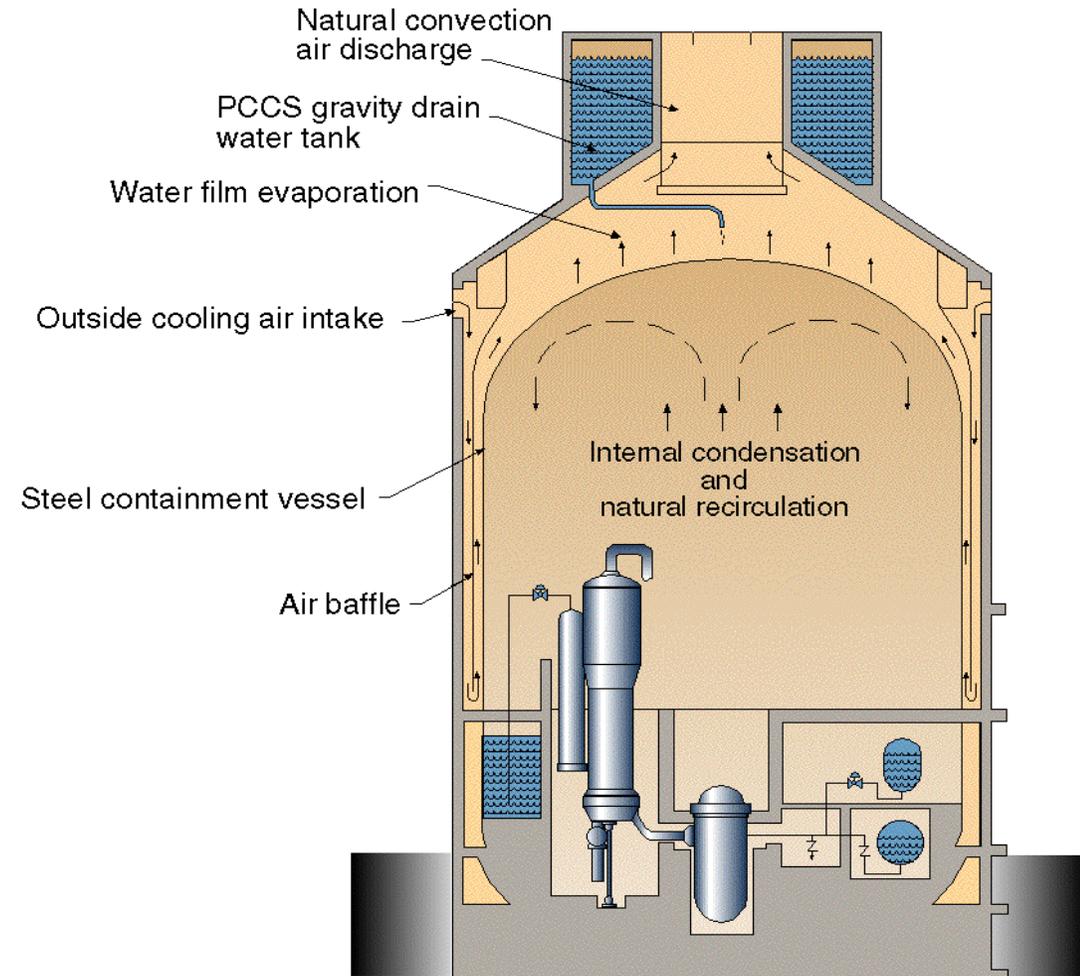
Pressurized Water Reactor Containment

- By Design
 - Dry containment
 - Ice condenser containment
 - Sub-atmospheric Containment
- By Structure
 - Reinforced or Prestressed Concrete Containment
 - Steel Containment



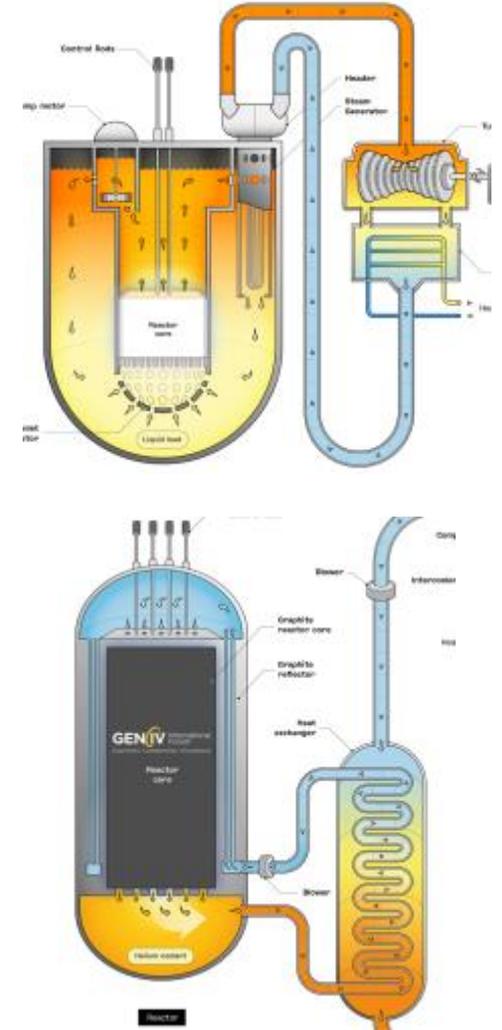
AP1000 PWR Containment

- AP1000 is a Generation III+ pressurized water reactor (PWR) with 1150 MWe output.
- Passive safety system
 - Relies on gravity, natural circulation, evaporation, condensation, and convection for safety functions, eliminating dependence on active components like pumps or diesel generators.
- Passive Containment Cooling System
 - 72 hr of water evaporation into air
 - Afterwards use available onsite or offsite water supplies
 - Air only cooling prevents failure
- Shield Building- protection of containment



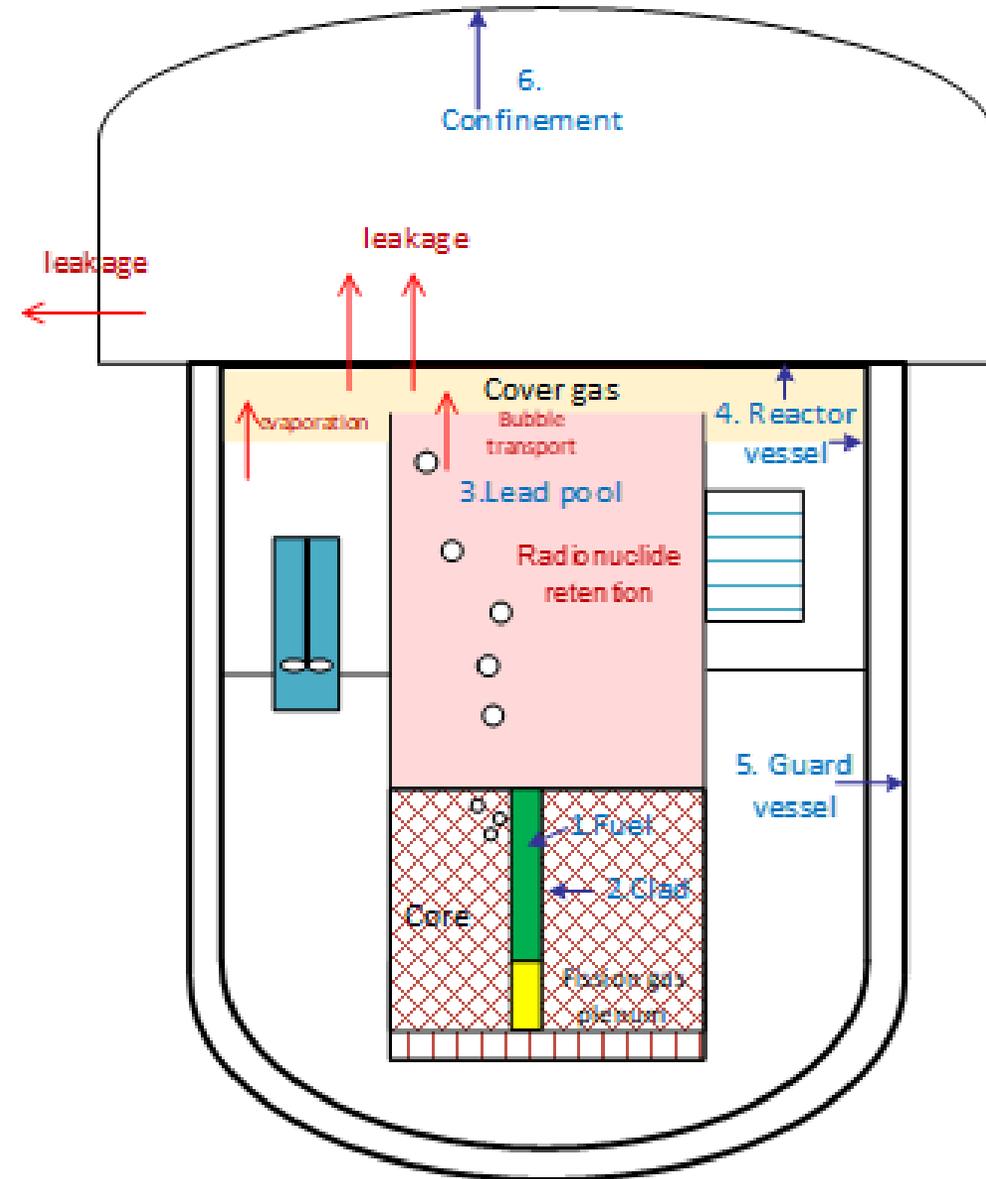
Containment Function for Non-Water Cooled Reactor

- Characteristics of advanced non-WCRs
 - Advanced fuel cycle
 - Safety enhancement
 - Economic competitiveness
 - Versatile applications
- Advanced non-WCRs do NOT present the high mass and energy release associated with the WCRs.
 - **Relax the need for a structurally robust containment**
- The non-WCR system imposes additional layers to confine radionuclides during operation and accidents.
 - Significantly reduce radionuclides release
 - Severe accident is largely reduced or eliminated.



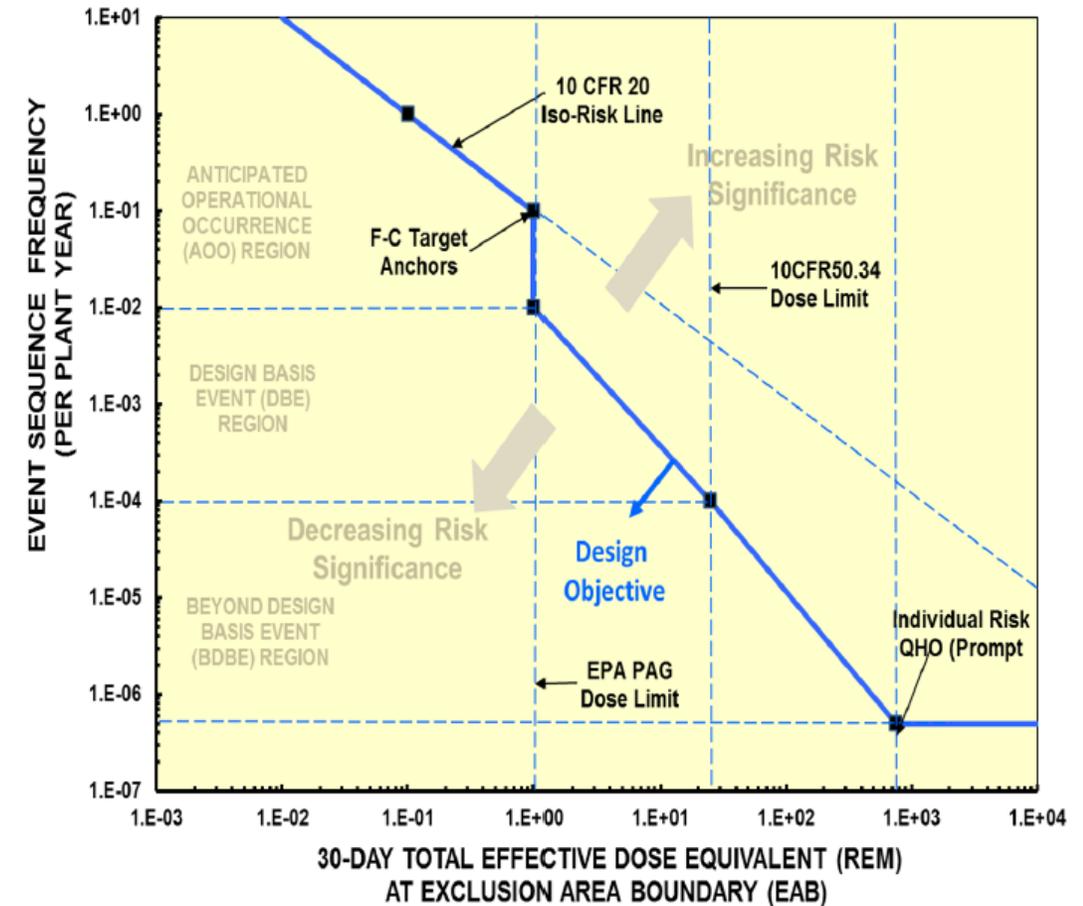
LFR Functional Containment

- Functional containment consists multiple layers of barriers and each barrier attenuates the amount of radionuclides
- Barrier in functional containment of LFR
 - Fuel pellets
 - Cladding
 - Liquid lead
 - Reactor vessel
 - Guard vessel
 - Reactor building
- Each barrier attenuates the amount of radionuclides presented to subsequent barriers thus reducing the magnitude of the release to the environment to an acceptable magnitude



Risk-Informed Safety Analysis for Functional Containment

- **Risk-informed performance-based (RIPB)** licensing framework for non-WCRs is piloted by the Licensing Modernization Project (LMP)
- LMP methodology is intended to
 - Select and evaluate Licensing Basis Events.
 - Classify Structures, Systems and Components (SSCs) based on their holistic and realistic contribution to risk
 - Determine Defense-in-Depth (DiD) adequacy
- **Mechanistic Source Term (MST)** refers to the magnitude, timing, and composition (chemistry) of radioactive releases to the environment during accident scenarios
- Evaluation of initial radionuclide inventories and locations in the plant
- Quantification of the Mechanistic Source Term



RELIEF (Radionuclide Retention in Liquid lead Experimental Facility)

- Developed **decontamination factor** (DF) model for fission product radionuclide retention in **lead pool** is unique for LFR
- Testing on fission product radionuclide retention of lead coolant is important to confirm the DF model
- Purpose: demonstrate the radionuclide retention capability of lead pool and provide experimental data to validate the DF model.
- The test facility is supported by Westinghouse and located at Virginia Tech (USA).
 - Commissioned in 2026 and retention testing is ongoing.

