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Preliminary Assessment of the Safety Performance of Westinghouse LFR

Technical Meeting on State-of-the-art Thermal Hydraulics of Fast Reactors

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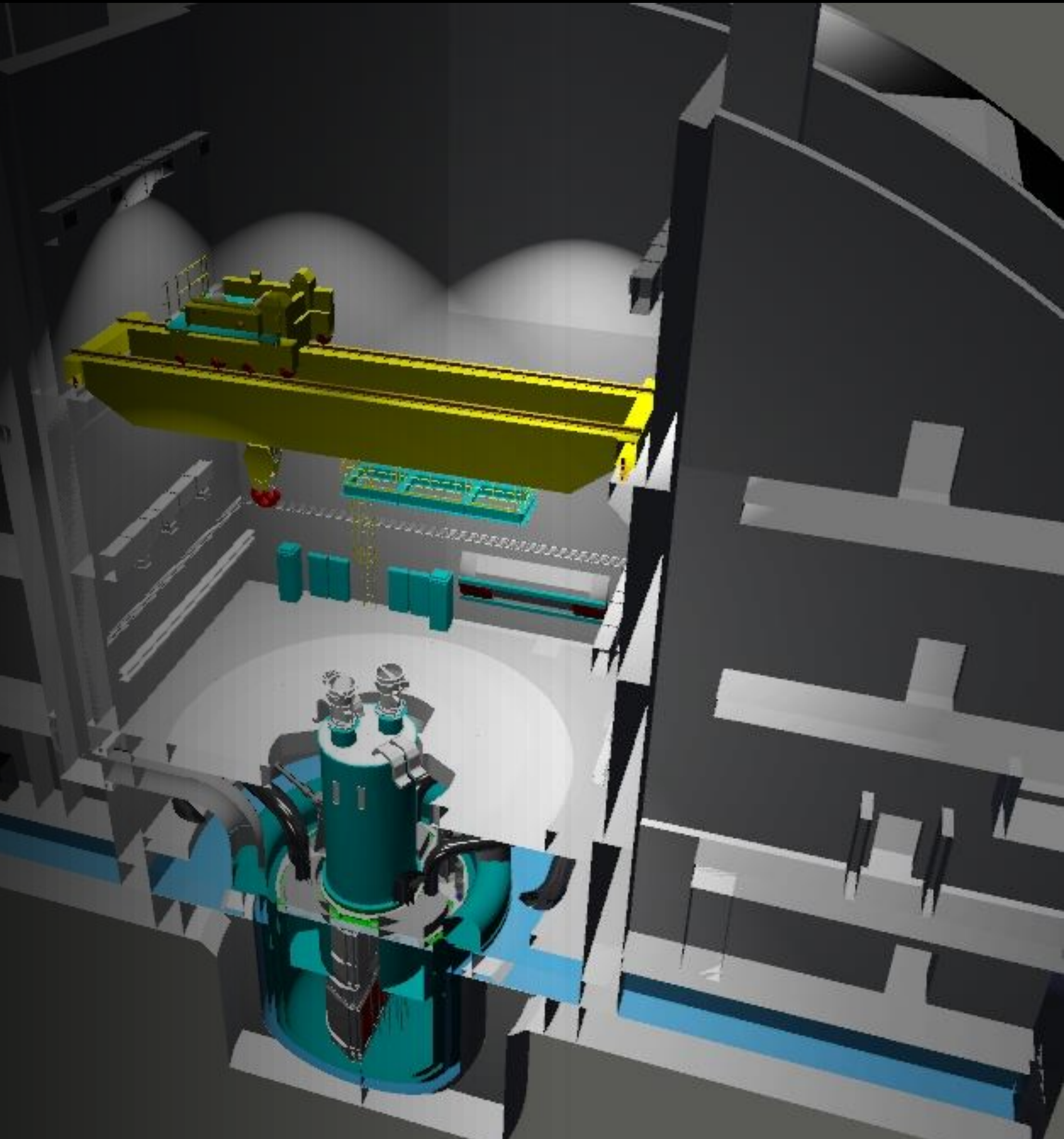
Sung Jin Lee (Fauske & Associates, USA)

Jun Liao, Dan L. Wise, Richard F. Wright, Paolo Ferroni (Westinghouse Electric Company, USA)



Outline

- Overview of the Westinghouse Lead Fast Reactor (LFR)
- Modeling Approach to Safety Performance Assessment
  - SAS4A/SASSYS-1 Model of the Reactor Vessel
  - GOTHIC Model of the Passive Heat Removal System
- Results of Preliminary Assessment
  - Protected Station Blackout (SBO)
  - Protected Transient Overpower (TOP)
- Summary

Introduction to the Westinghouse Lead-cooled Fast Reactor

Mission and Development Status

The Westinghouse LFR is a forward-thinking concept designed to:

- Achieve a step-change in economic competitiveness
- Achieve versatility in applications, beyond electricity
- Accommodate transition to closed fuel cycle, if/when needed

Developed leveraging Westinghouse's demonstrated experience in commercializing nuclear power plants globally

Strengthened by international collaborations selected to best complement capabilities

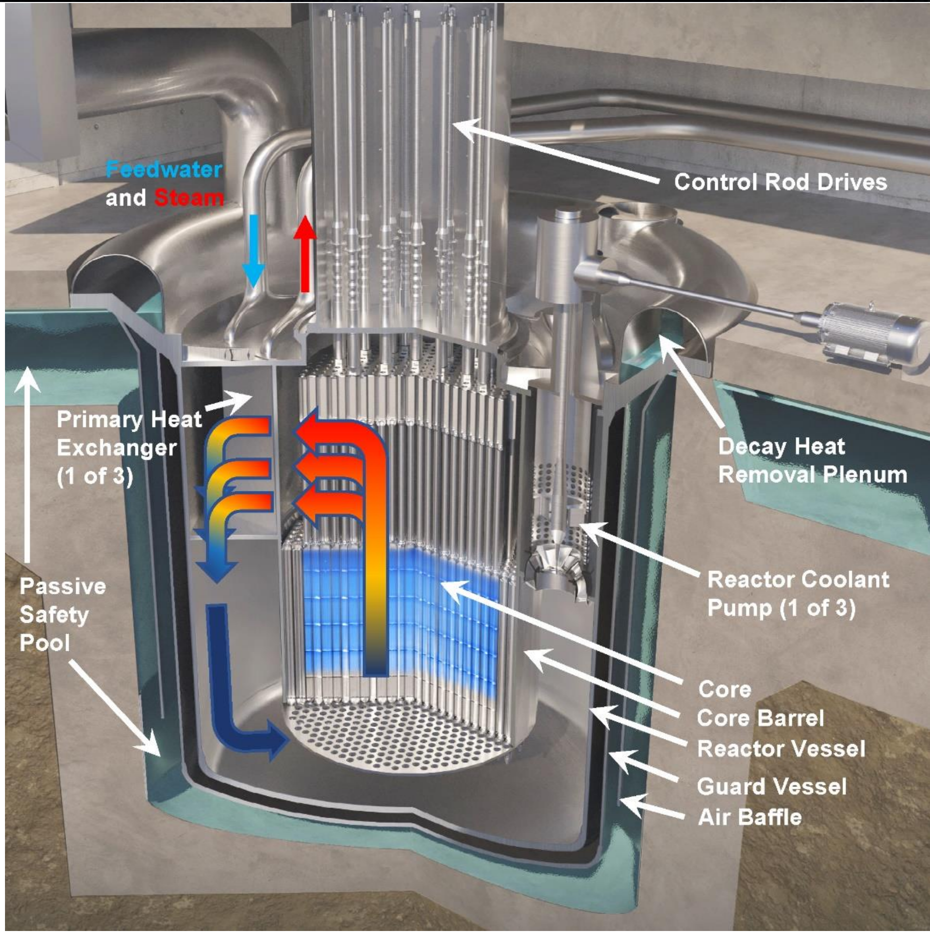
Development status:

- Near completion of conceptual design
- Demonstration of key systems, components and materials starting in 2022
- Pre-licensing engagement ongoing with UK Regulators

Westinghouse LFR's Key Characteristics

- Pool-type, passively safe, modular construction lead-cooled fast reactor

Reactor power	950 MWt (~450 MWe Net)
Efficiency	~47%
Primary coolant	Liquid lead
Secondary coolant	Supercritical water
Neutron spectrum	Fast
Configuration	Independent unit for single or two-unit site
Fuel	Oxide or Advanced fuel (future)
Operating pressure, MPa	0.1 (primary) / ~34 (secondary)
Lead coolant min/max temperature, ˚C	390 / 650



- Enhanced passive safety – shutdown, decay heat removal
- Fuel cycle flexibility typical of fast reactors
- Innovations to improve economics and enhance market versatility:
  - High-performance materials for heavy-duty components → improved economics through higher efficiency
  - Hybrid micro-channel-type heat exchangers → compact vessel and simplified safety
  - Atmosphere as the ultimate heat sink → enhanced siting opportunities with no need for vicinity of water bodies
  - Thermal energy storage → flexible electricity without changing core power

LFR Fuel and Fuel Cycle

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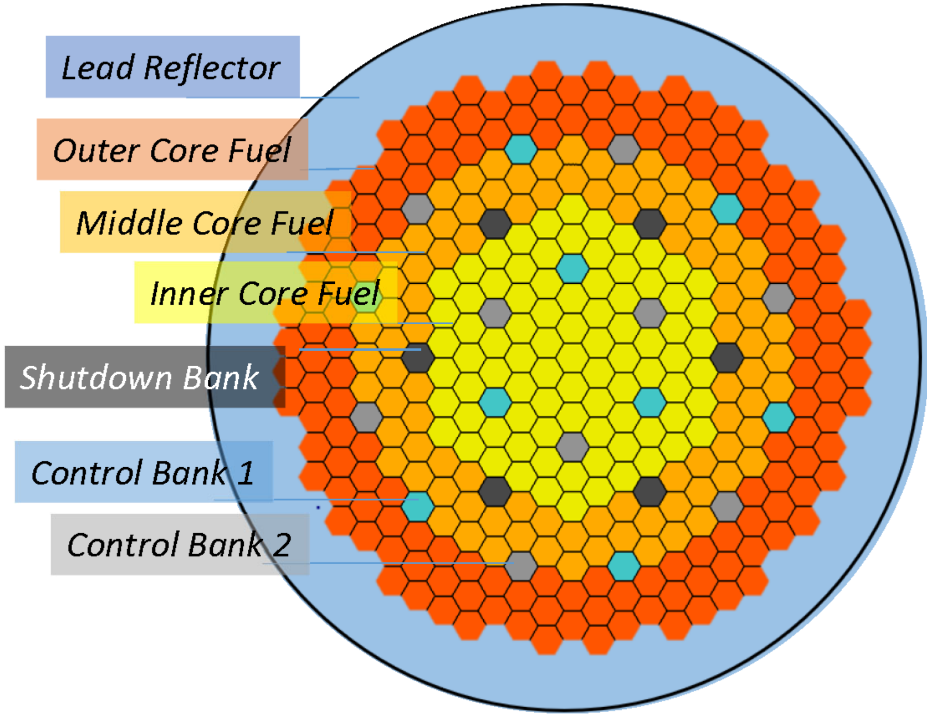
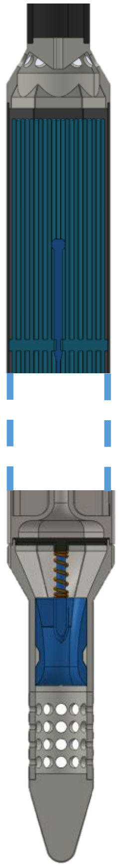
Fuel materials

- UO_2 with 15-15Ti-type austenitic steel cladding for LFR start-up core
- UN with advanced cladding (steel or SiC) for future performance enhancements
- MOX for Pu recycle if/when pursued
- Advanced fuel options are backfittable, no change to internals or control system will be required for incorporation
- Synergy with Westinghouse ATF and High Burnup/High Energy fuel development program

Fuel cycle and refueling

- Reference fuel cycle: open
  - Flexibility to transition to semi-open or closed cycle if pursued by national policies
- Long cycle length:
  - ~8 years (UO_2); ~15 years (MOX/UN)
  - Single-batch core designs
- Refueling scheme: direct-to-cask with no assembly shuffling. No spent fuel pool





Emergency Passive Heat Removal System (PHRS)

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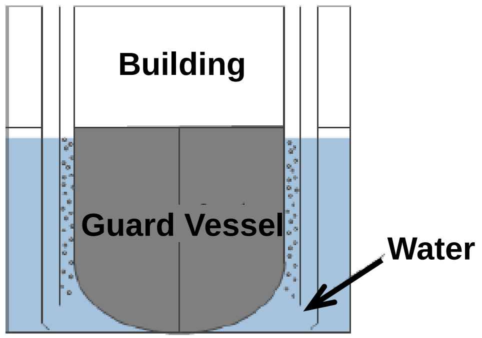
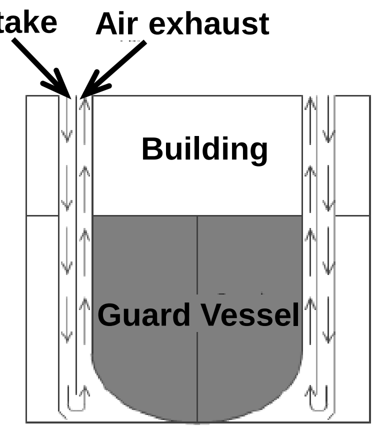
- LFR PHRS design requirements:
  - IAEA passive safety category B: No moving parts
  - Capable to remove decay heat of 950MWt LFR core
  - Capable of extended long term heat removal
- Leverage knowledge of AP1000® passive containment cooling system and SFR reactor vessel auxiliary cooling system

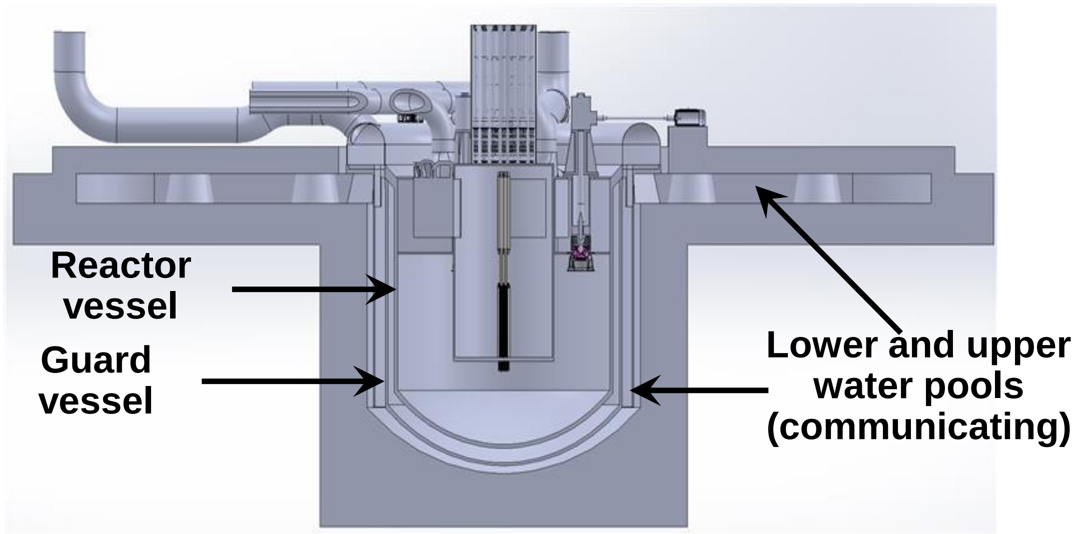


Water-cooling mode (notional)



Air-cooling mode (notional)

- The LFR PHRS design features
  - Pool of water surrounds Guard Vessel
  - Water-cooling during DBA (7 days)
  - Transition to (indefinite) air-cooling in extended long-term cooling
  - Fully passive, no I&C support, no need for actuation
  - System always on
  - Performance primarily driven by radiation heat transfer between RV and GV, which is very low during normal operation but kicks in during transients



Westinghouse LFR: A Global Program

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Westinghouse Sweden

Westinghouse US

Westinghouse Mangiarotti (Italy)

Westinghouse UK

Global collaborations

Argonne National Laboratory
ENEA Italian National Agency for New Technologies, Energy and Sustainable Economic Development
Fauske & Associates, LLC
Ansaldo Nucleare Ansaldo Energia Group
Westinghouse
Department for Business, Energy & Industrial Strategy
Los Alamos National Laboratory EST. 1943
Oak Ridge National Laboratory
National Nuclear Laboratory
Prifysgol Bangor University
PennState
BYU
VPE Vacuum Process Engineering
INL Idaho National Laboratory
Nuclear AMRC
Jacobs
Frazer-Nash Consultancy
Echogen power systems
University of Pittsburgh
Wisconsin University of Wisconsin-Madison
Manchester 1824

Modeling Approach to Safety Performance Assessment

Modeling Approach to Safety Performance Assessment

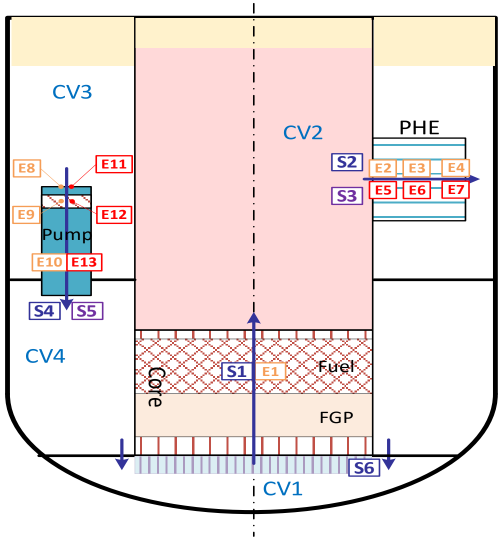
- The SAS4A/SASSYS-1 system code is coupled to the GOTHIC containment code to simulate the response of the Westinghouse LFR to operational transient and accidents.
- SAS4A/SASSYS-1 models the core, in-vessel thermal hydraulics, reactor components, and reactor vessel wall.
- GOTHIC models the passive heat removal system and guard vessel.
- Both codes run simultaneously, exchanging data via inter-process communication data files.
- At each time step, SAS4A/SASSYS-1 passes reactor vessel wall temperatures to GOTHIC and GOTHIC returns heat transfer rates on the vessel wall.

SAS4A/SASSYS-1 Overview

- SAS4A/SASSYS-1 is a safety analysis code for liquid metal fast reactors
  - Developed by Argonne National Laboratory in the 1970s', primarily for sodium fast reactors
  - Has been used as the safety analysis tool for sodium fast reactors extensively.
  - Expanded to lead based reactor since lead properties subsequently added in 1990s.

- Westinghouse has been using SAS4A/SASSYS-1 for LFR since 2017

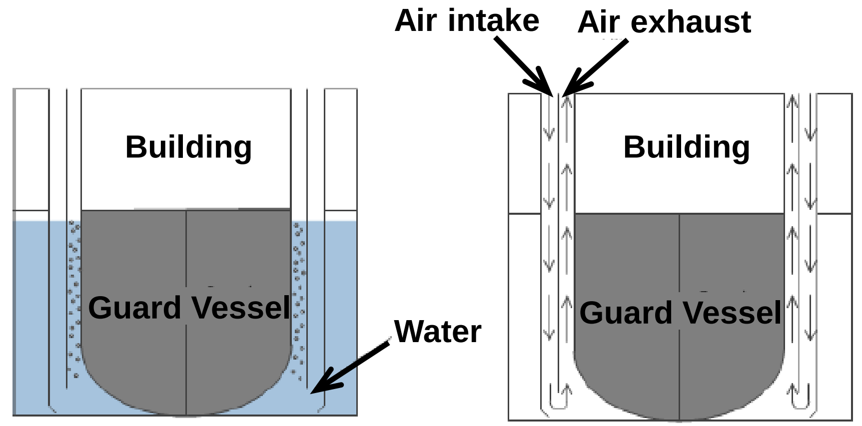
- SAS4A/SASSYS-1 plays an important role in the development of LFR.
  - Analysis/scoping tool for the design of W-LFR system
  - Support the development of testing plan and PIRT.
  - Safety analysis code (AOO, DBA, BDBA) for the licensing of LFR.



Presentation of "Applicability Enhancement of SAS4A/SASSYS-1 Computer Code to Lead Fast Reactor Systems" on Thursday gives more information.

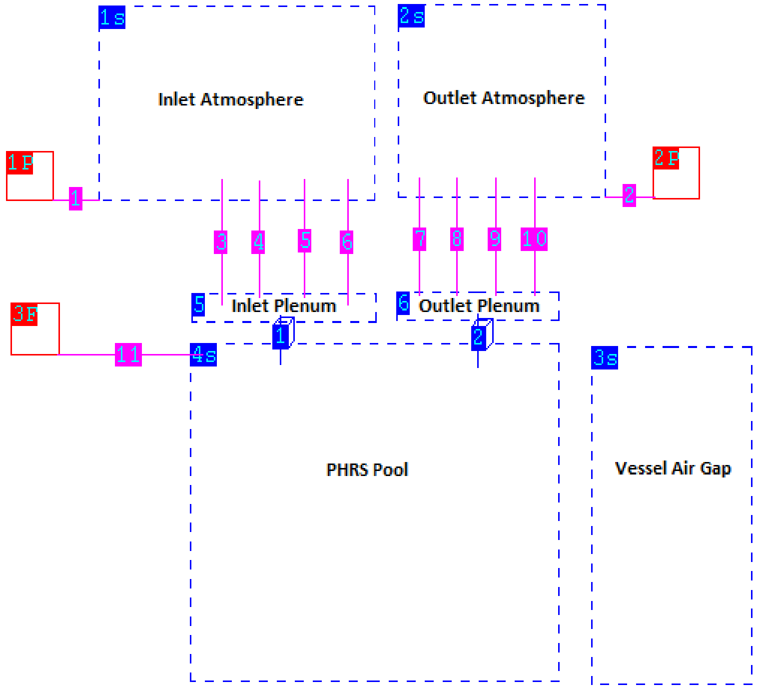
GOTHIC Overview

- GOTHIC is a high-pedigree system/containment analysis computer code.
- GOTHIC is a general-purpose thermal-hydraulics software package for design, licensing, safety and operating analysis of nuclear power plant containments, confinement buildings and system components.
- Capable to model two-phase flow and heat transfer.
- Westinghouse has extensive usage of GOTHIC for light water reactors.
- The GOTHIC code is selected to model LFR PHRS and it is coupled with the SAS4A/SASSYS-1 code to perform safety analysis of the LFR.



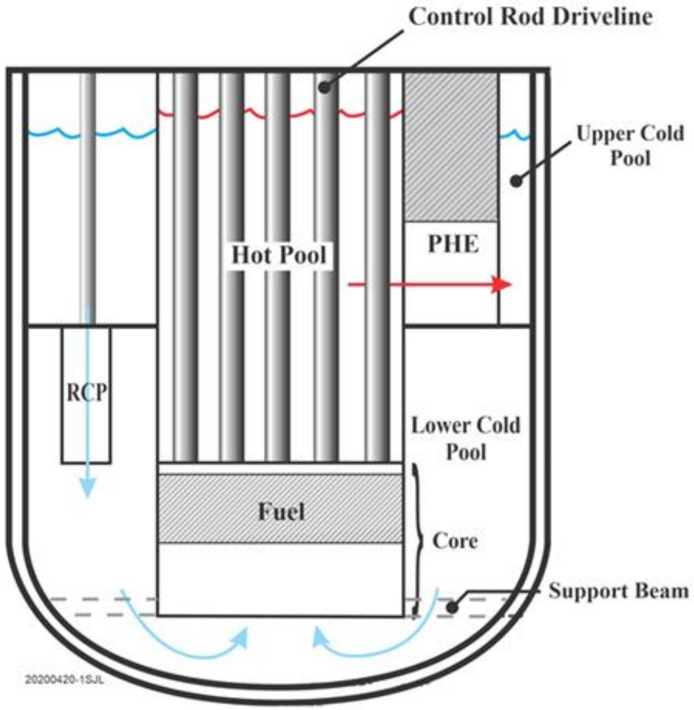
Water-cooling mode (notional)

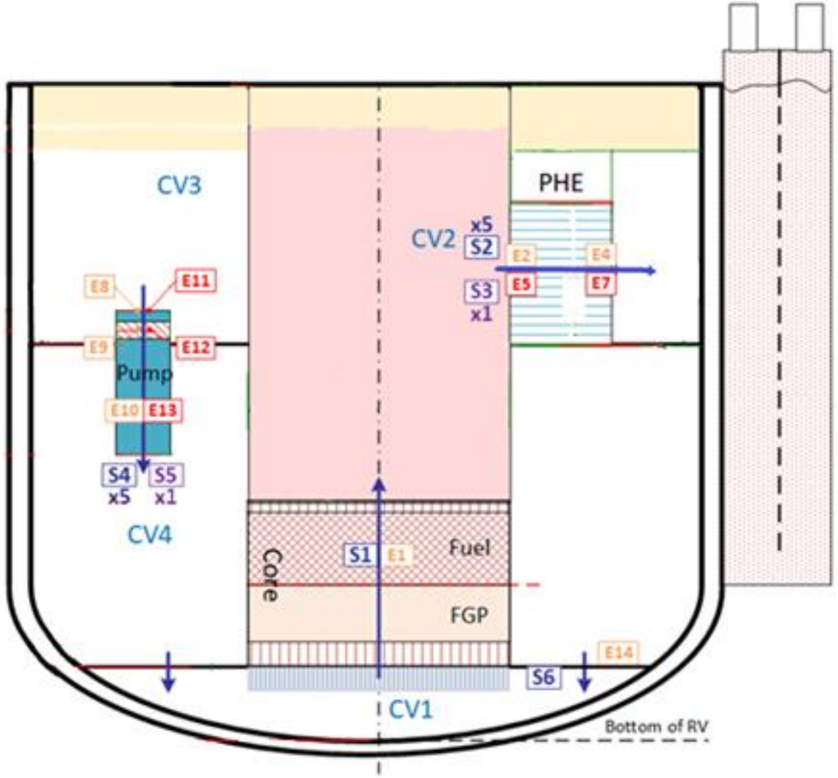
Air-cooling mode (notional)



SAS4A/SASSYS-1 Model of the Reactor Vessel

- Four compressible volumes are used to represent the primary coolant system: CV1 for the inlet plenum, CV2 for the hot pool, CV3 for the upper cold pool, and CV4 for the lower cold pool.
- Primary heat exchangers (PHEs) and reactor coolant pumps (RCPs) are modeled.

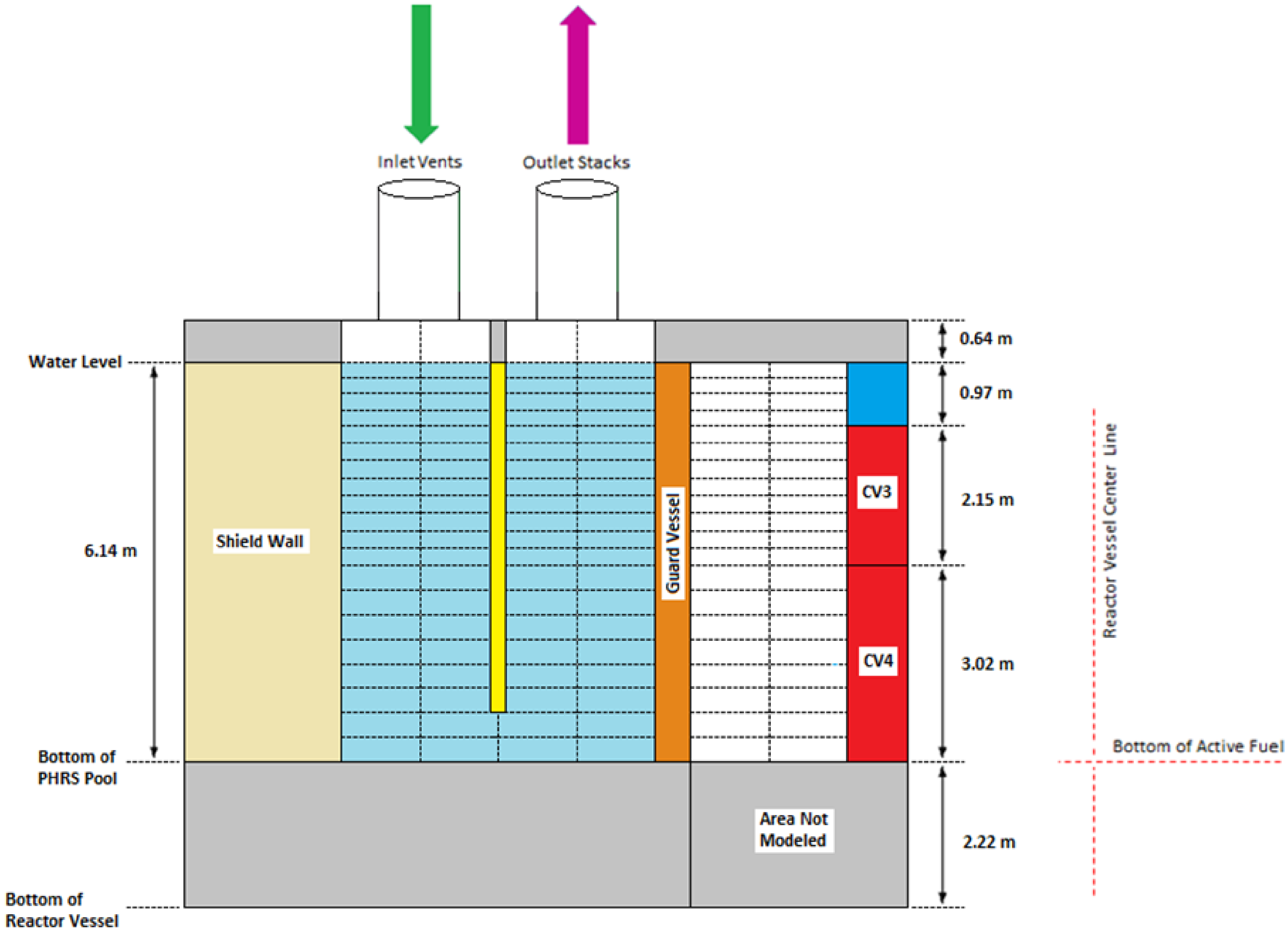




SAS4A/SASSYS-1 Model of the Reactor Vessel

GOTHIC Model of the Passive Heat Removal System (PHRS)

- The PHRS is modelled using six control volumes: inlet atmosphere, outlet atmosphere, PHRS pool, RV-GV (Reactor Vessel-Guard Vessel) air gap, inlet plenum, and outlet plenums.
- The PHRS pool is subdivided into twenty vertical nodes and four radial nodes to represent local conditions such as pressure, temperature, void fraction, heat transfer rate, and flow rate.
- The inlet and outlet atmosphere control volumes are each subdivided into two vertical nodes to represent static heads in inlet ducts and outlet stacks.



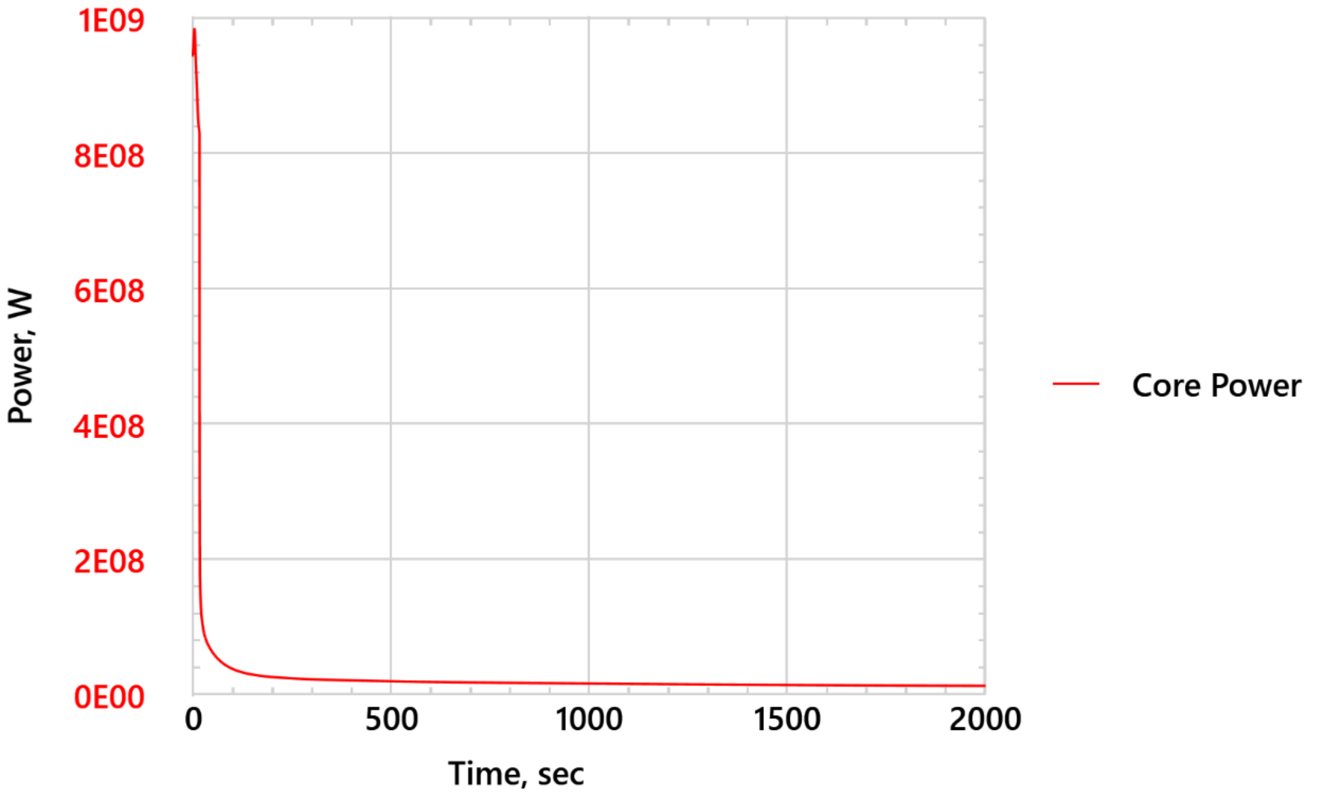
Results of Preliminary Assessment

Results – Station Blackout (SBO)

- In station blackout, a loss of offsite power event is postulated.
- The accident is assumed to occur at the end-of-cycle (EOC) in the reactor operation. EOC is selected for the highest fission gas pressure in the fuel rod. The fluence to clad and the decay heat in fuel are also highest at EOC.
- Heat removal by primary heat exchangers (PHE) is assumed to stop instantaneously at time zero.
- The reactor coolant pump (RCP) torque is reduced to zero in one second for all pumps. Subsequently, the pumps coast down following the homologous curve.
- The normal reactor shutdown system is assumed to fail – the passive shutdown system is actuated by high hot pool temperature. The transient is Protected station blackout.
- The normal decay heat removal system is assumed unavailable.

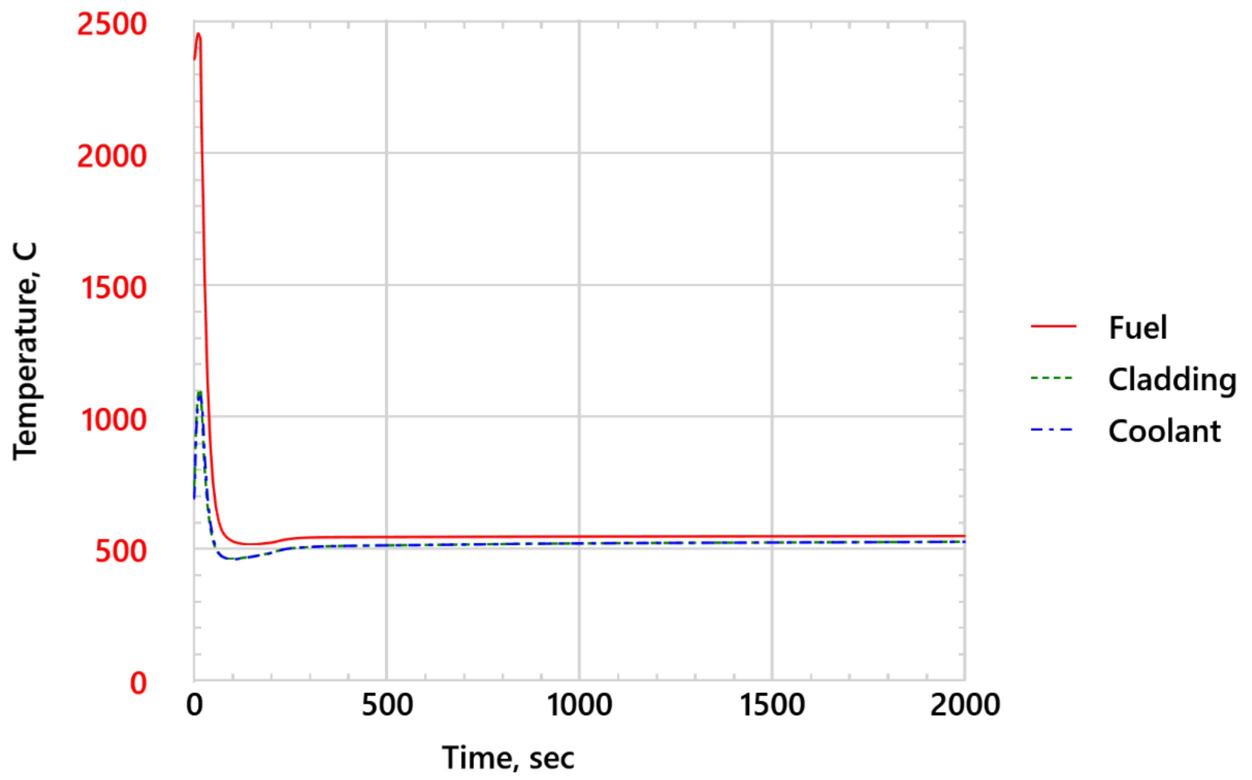
Results – Station Blackout (SBO)

- Core Power – The reactor is tripped after 20 seconds by the passive shutdown system triggered by high coolant temperature.



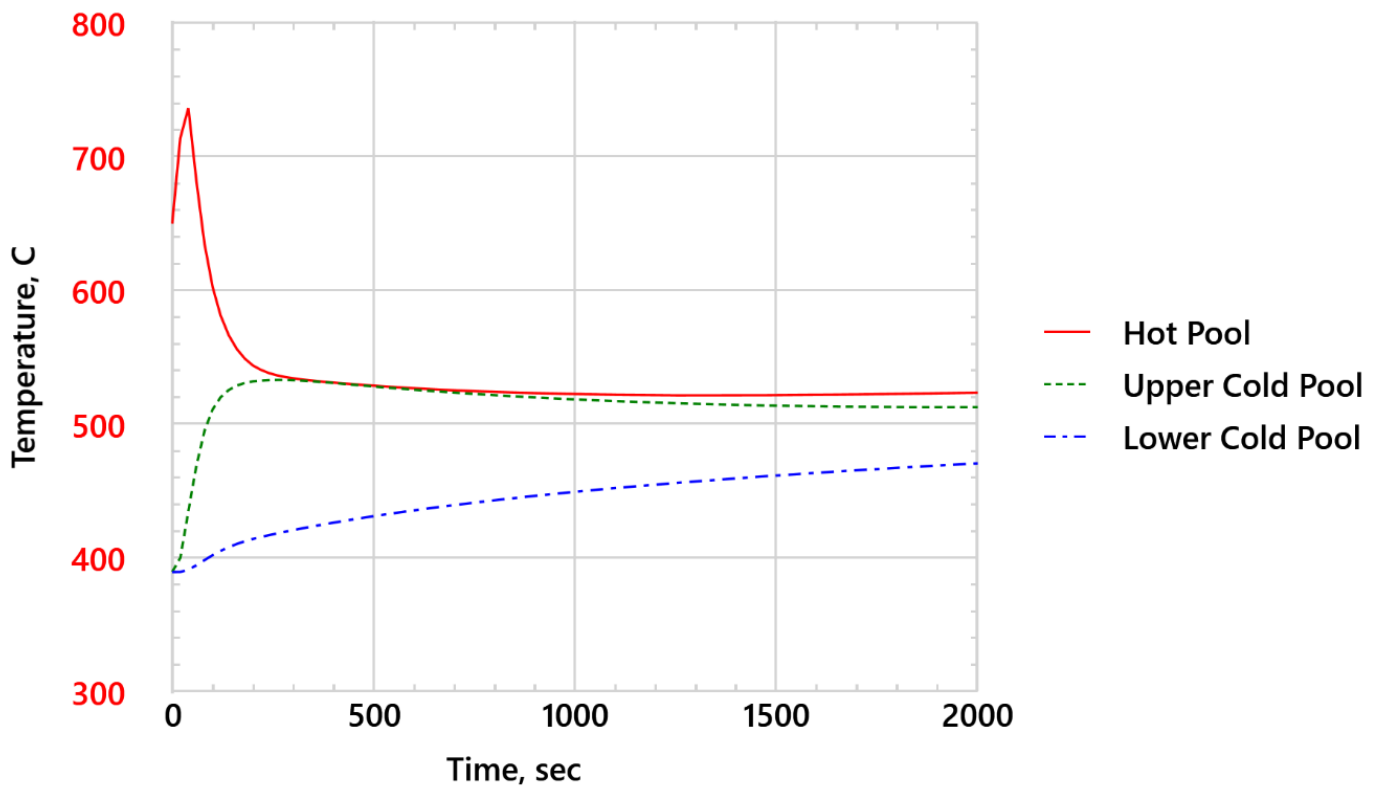
Results – Station Blackout (SBO)

- Fuel, Cladding, and Coolant Temperatures – Temperatures increase initially due to the fission power without heat removal by primary heat exchangers; fuel does not melt, and cladding does not fail.



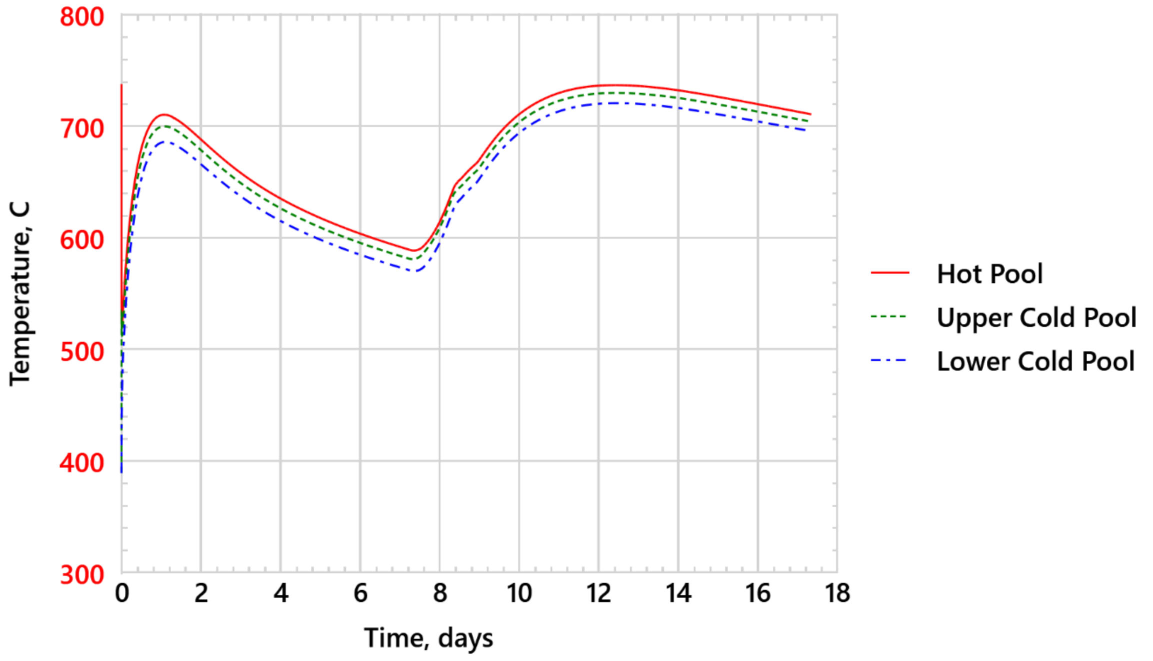
Results – Station Blackout (SBO)

- Short Term Pool Temperatures – following loss of primary heat exchangers, pool temperatures converge; the lower cold pool has significantly larger thermal mass



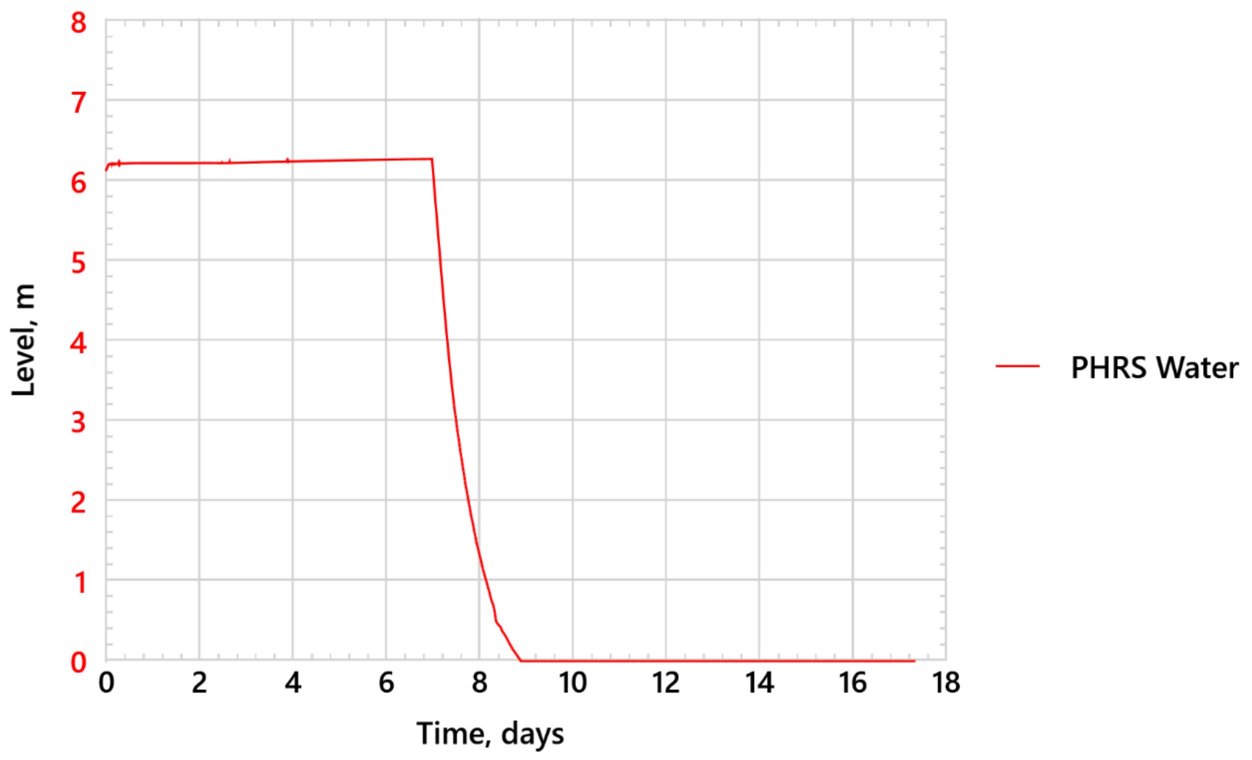
Results – Station Blackout (SBO)

- Long Term Coolant Temperatures – three temperature peaks, initial peak due to fission power, short-term peak due to the mismatch between decay heat and water cooling, and long-term peak due to the mismatch between decay heat and air cooling are predicted



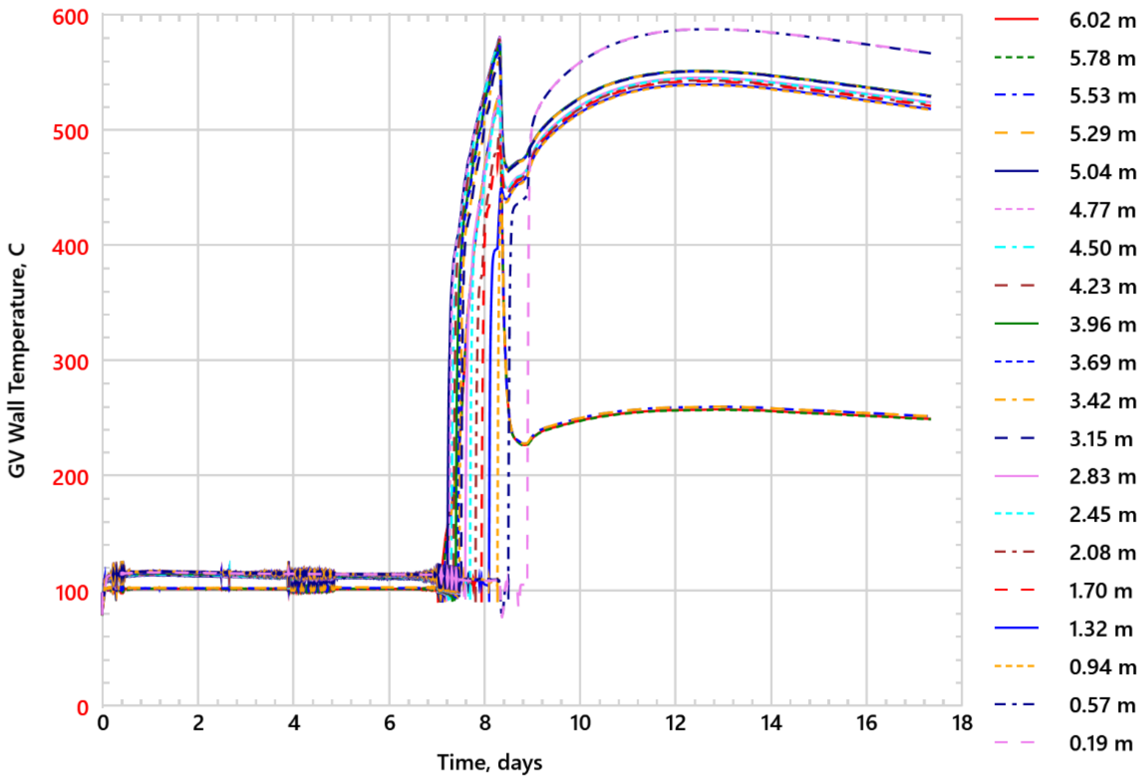
Results – Station Blackout (SBO)

- PHRS Water Level – enough water to remove decay heat for seven days without power is stored in the safety pool.



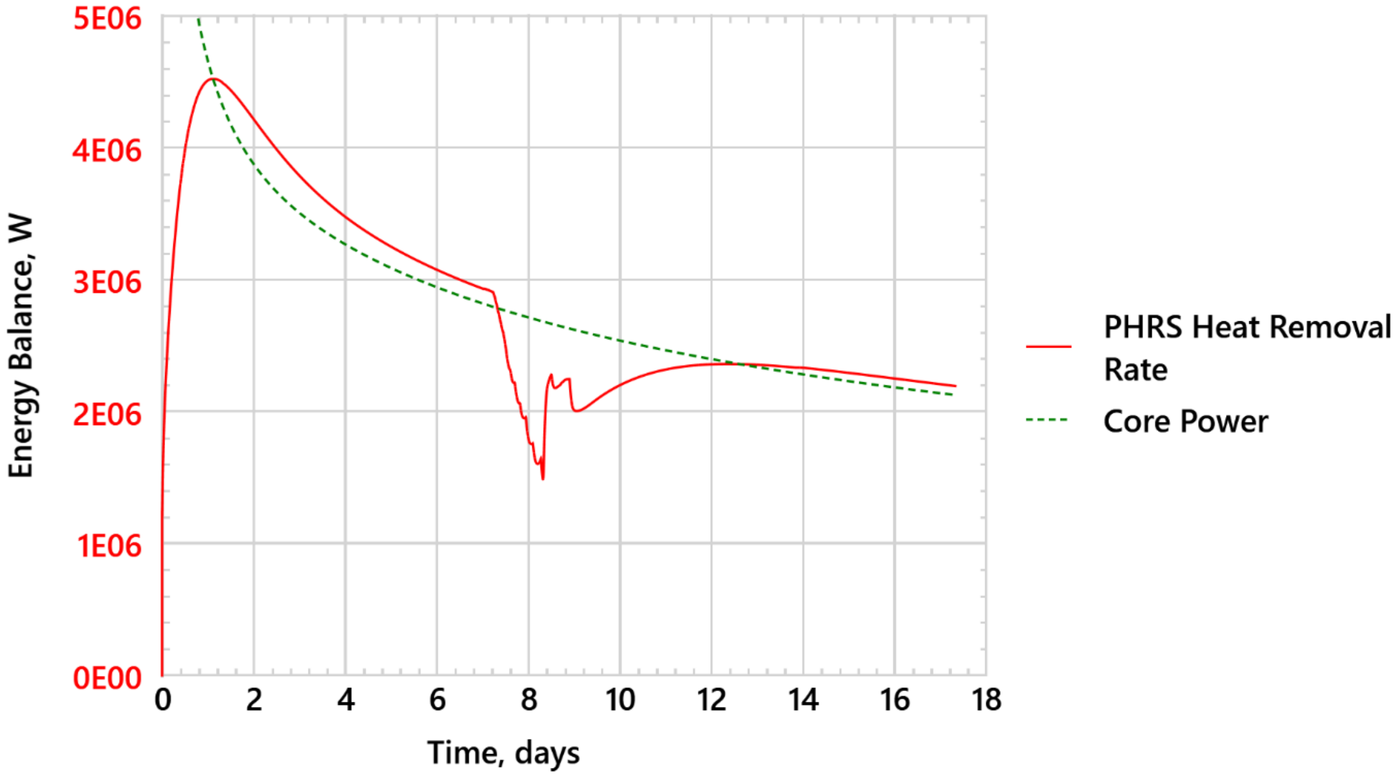
Results – Station Blackout (SBO)

- GV Wall Temperatures – GV wall temperatures rise rapidly starting from the top as the wall becomes uncovered.



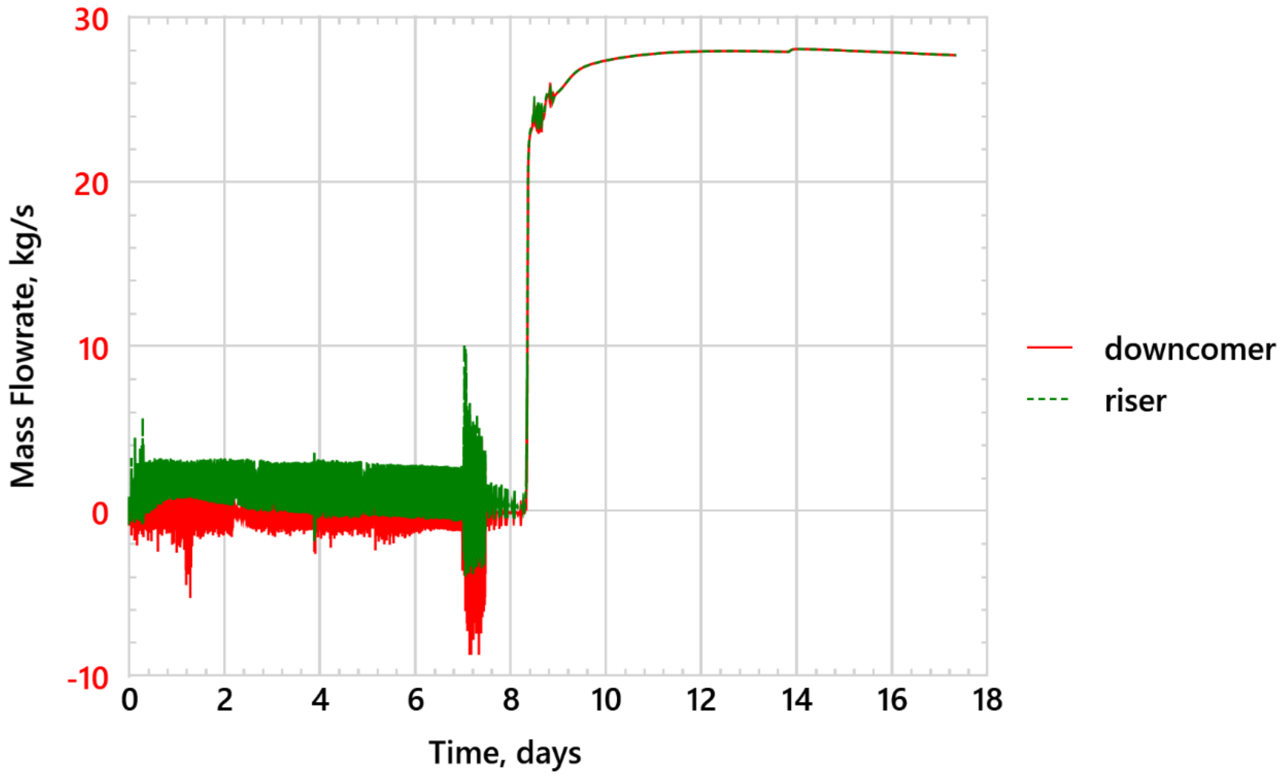
Results – Station Blackout (SBO)

- Energy Balance – After one day, the passive heat removal system (PHRS) can remove the decay heat constantly except during the water-to-air cooling transition.



Results – Station Blackout (SBO)

- PHRS Gas Flow Rate – the gas flow rate shows the steaming rate followed by the air circulation rate when the baffle is cleared after 8 days.

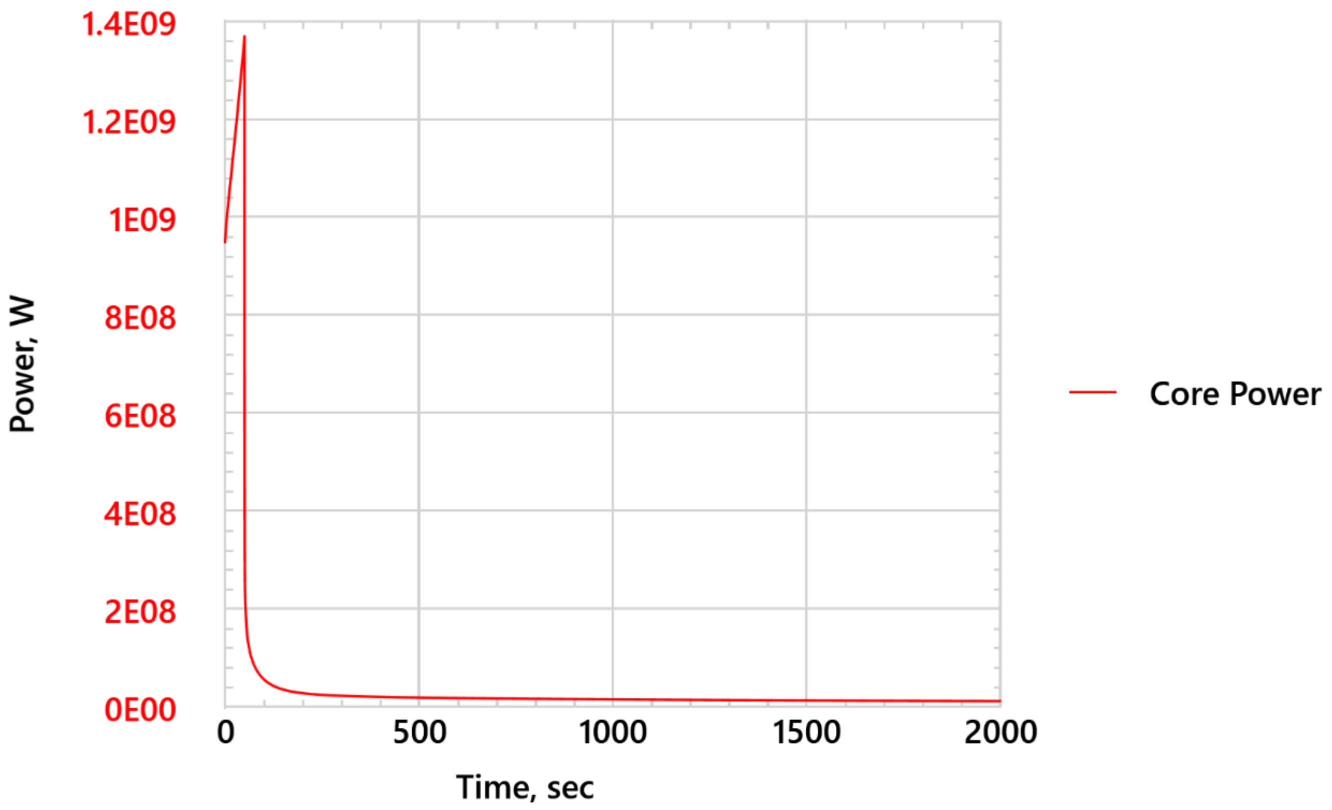


Results – Transient Overpower (TOP)

- A transient overpower accident is initiated by inadvertent withdrawal of most reactive control rods, inserting a positive reactivity of 74 cents in 60 seconds.
- All primary heat exchangers (PHEs) continue operating.
- All reactor coolant pumps continue operating.
- The accident is assumed to occur at the beginning-of-cycle (BOC) in the reactor operation. BOC is selected for the largest reactivity in the core at the beginning of cycle.
- In the protected transient overpower, the normal reactor shutdown system is assumed to fail – the passive shutdown system is actuated by high hot pool temperature.

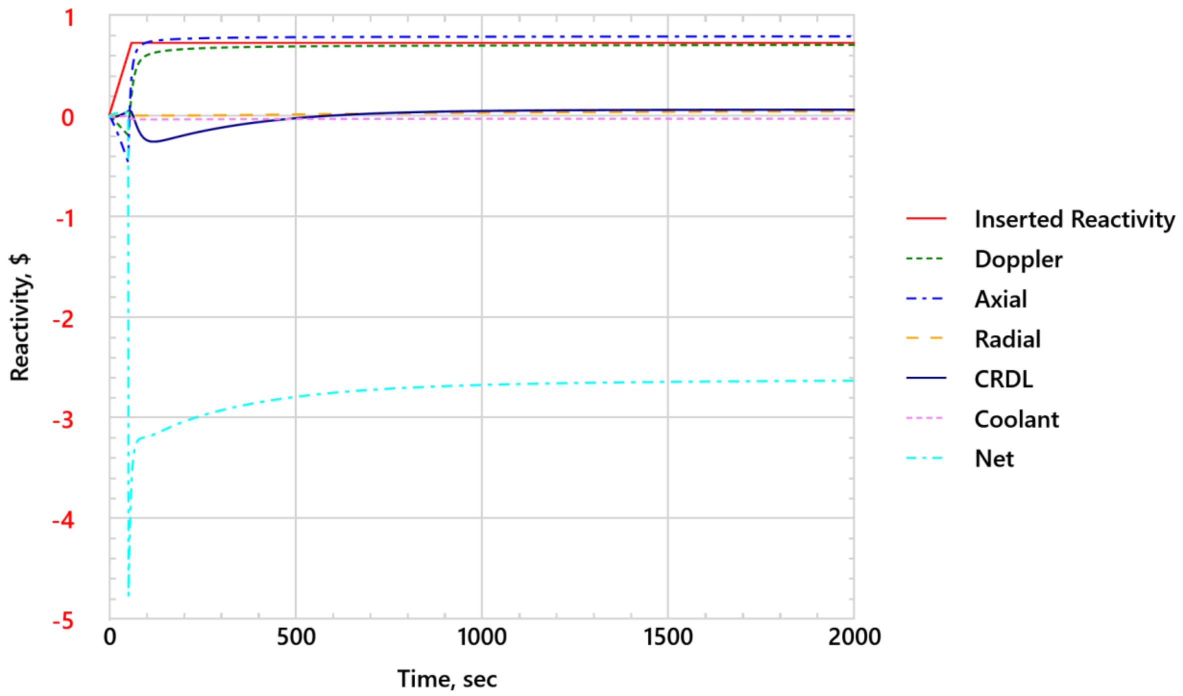
Results – Transient Overpower (TOP)

- Core Power – The core power increases to 140% of the normal power due to the positive reactivity insertion until the passive shutdown system is actuated when the hot pool temperature exceeds 700 ℃ after 50 seconds.



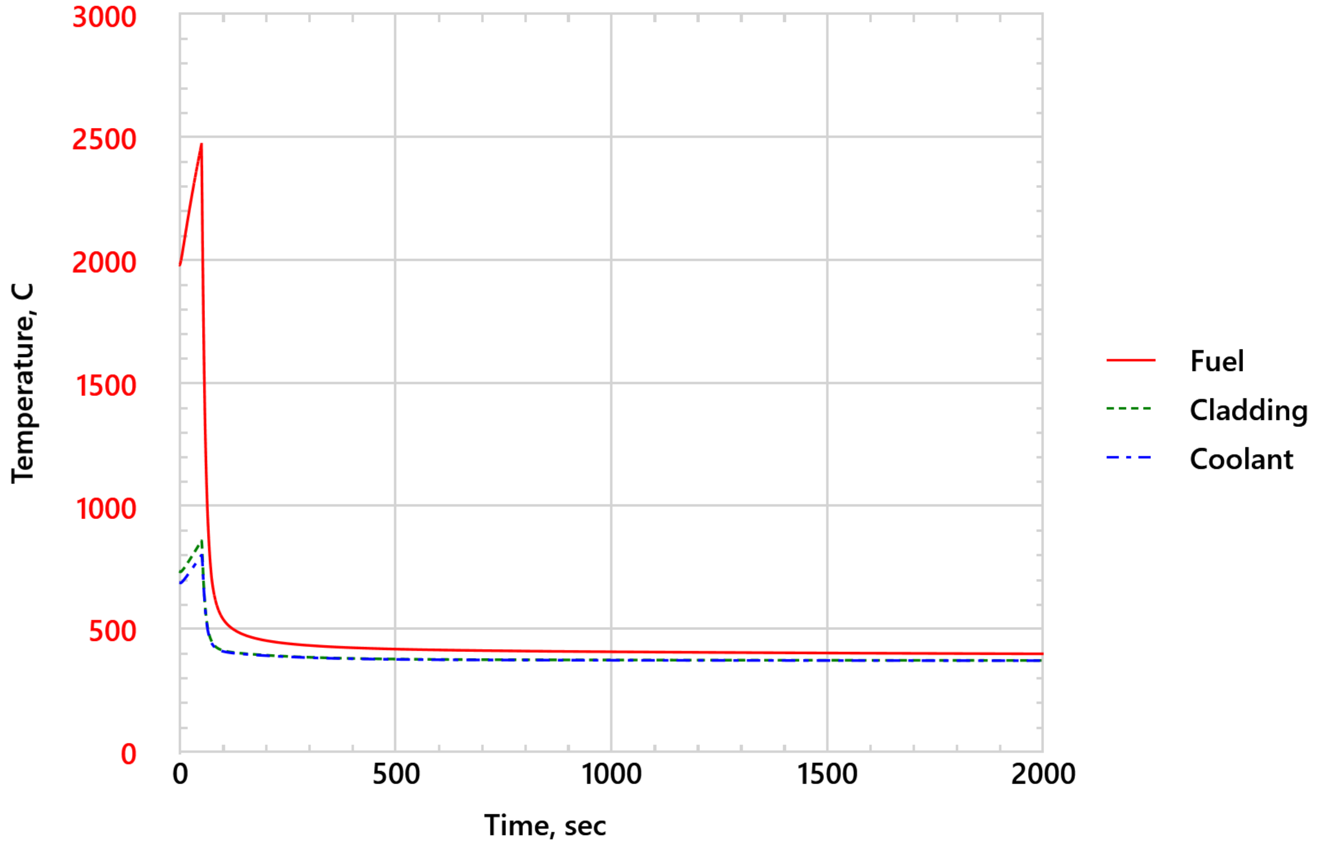
Results – Transient Overpower (TOP)

- Reactivity – the negative reactivity of passive shutdown system (not shown in the plot) overwhelms the inserted reactivity due to rod ejection; the reactor shuts down.



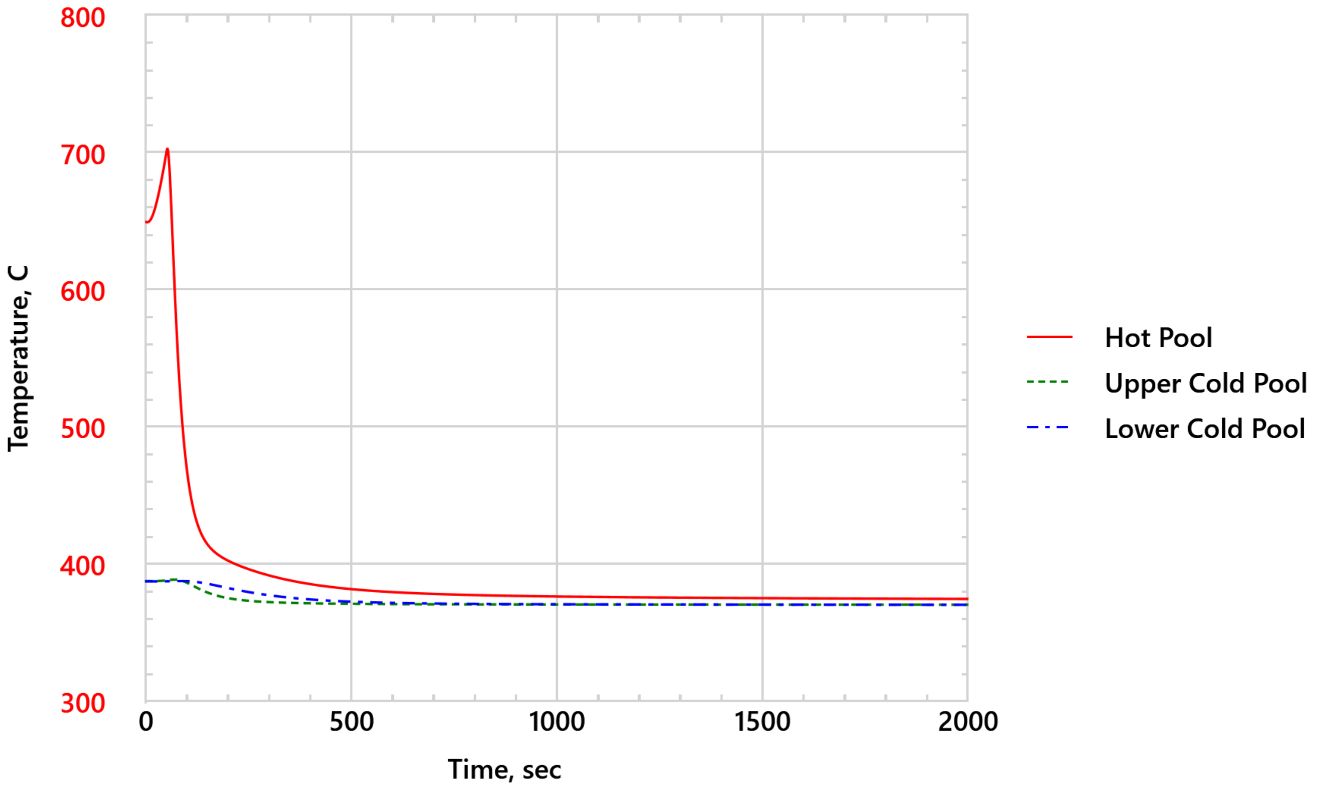
Results – Transient Overpower (TOP)

- Fuel, Cladding, and Coolant Temperatures – the fuel and cladding heat up initially due to the overpower. Then, they cool down rapidly after the reactor shuts down because of the primary heat exchangers and reactor coolant pumps. The fuel cladding does not reach the failure temperature.



Results – Transient Overpower (TOP)

- Coolant Temperatures – the coolant cools down and mixes due to operating PHEs and RCPs, approaching the primary heat exchanger secondary side temperature, 370 C.



Conclusions

- The responses of the Westinghouse LFR during SBO and TOP are assessed using the SAS4A/SASSYS-1 system code coupled to the GOTHIC code.
- For protected SBO, the performance of the passive heat removal system comprised of the safety (water) pool surrounding the GV, which transitions to air-cooling after the water in safety pool has boiled off, is evaluated.
- For protected TOP, temperature reactivity feedbacks in the core determine whether fuel melting and cladding failure will occur during the initial power excursion in the core.
- The analyses demonstrate capability of the coupled SAS4A/SASSYS-1 and GOTHIC codes to assess safety performance of Westinghouse LFR.