Sensitivity and Uncertainty Information Incorporated Loss of Flow Accident Analysis for Research Reactors

Tao Liu and Zeyun Wu

Department of Mechanical and Nuclear Engineering
Virginia Commonwealth University, Richmond, Virginia

VCU College of Engineering

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Outline

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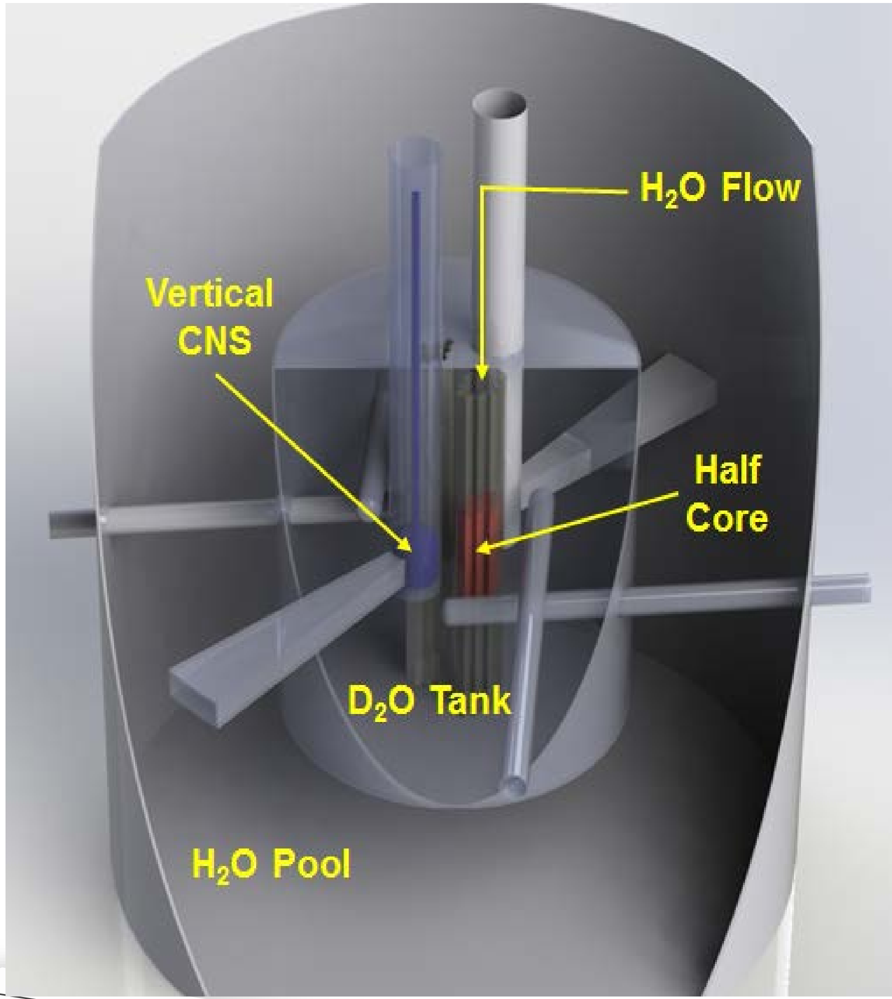
- Computational Models
- Protected Loss of Flow Accident
- Sensitivity and Uncertainty Analysis
- Results and Discussion
- Conclusions

Background and Objective

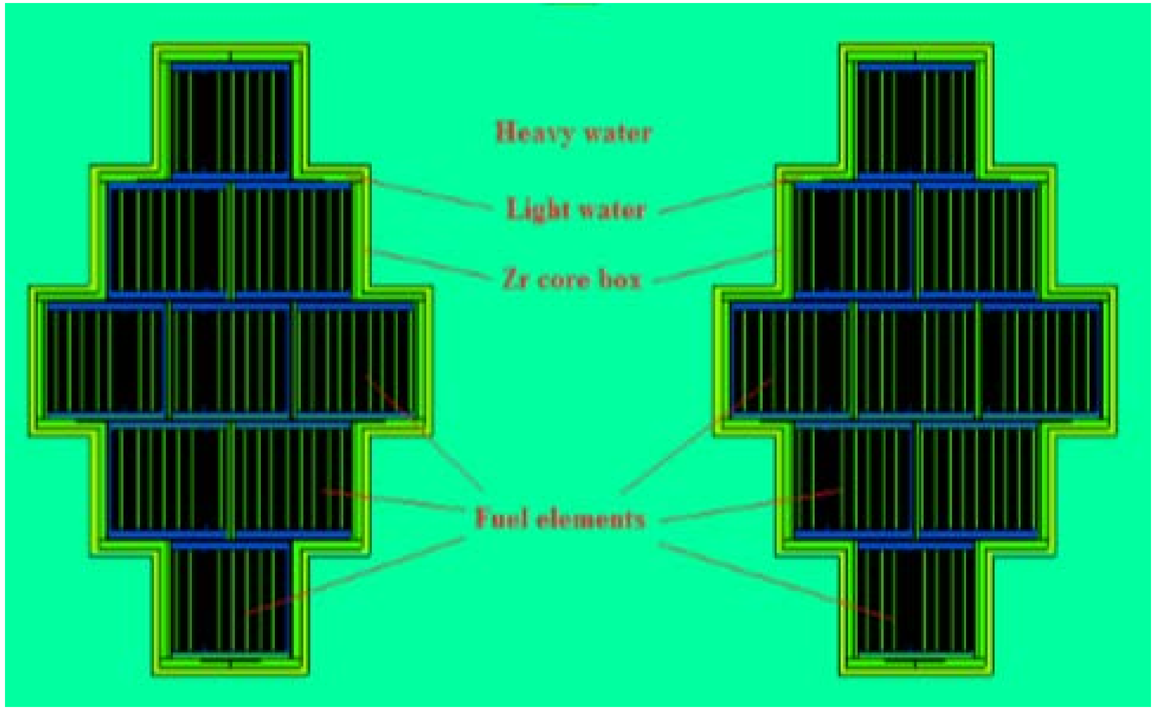
- Nowadays research reactors are widely used in the world as important research or production facilities.
- The safety analysis for research reactors is paramount important as that for commercial power reactors.
- To better assess the transient modeling capability and understand the discrepancies observed in the simulations, sensitivity analysis and uncertainty quantification were needed in the safety analysis to provide best-estimated predictions.

The NIST Conceptual Research Reactor Design



- Tank-in-pool type research reactor
- A heavy water tank immersed in a light water pool
- Beam-type research reactor as an advance neutron source facility
- 20 MW thermal power
- 30-day operating cycle

Horizontal Split Core Design



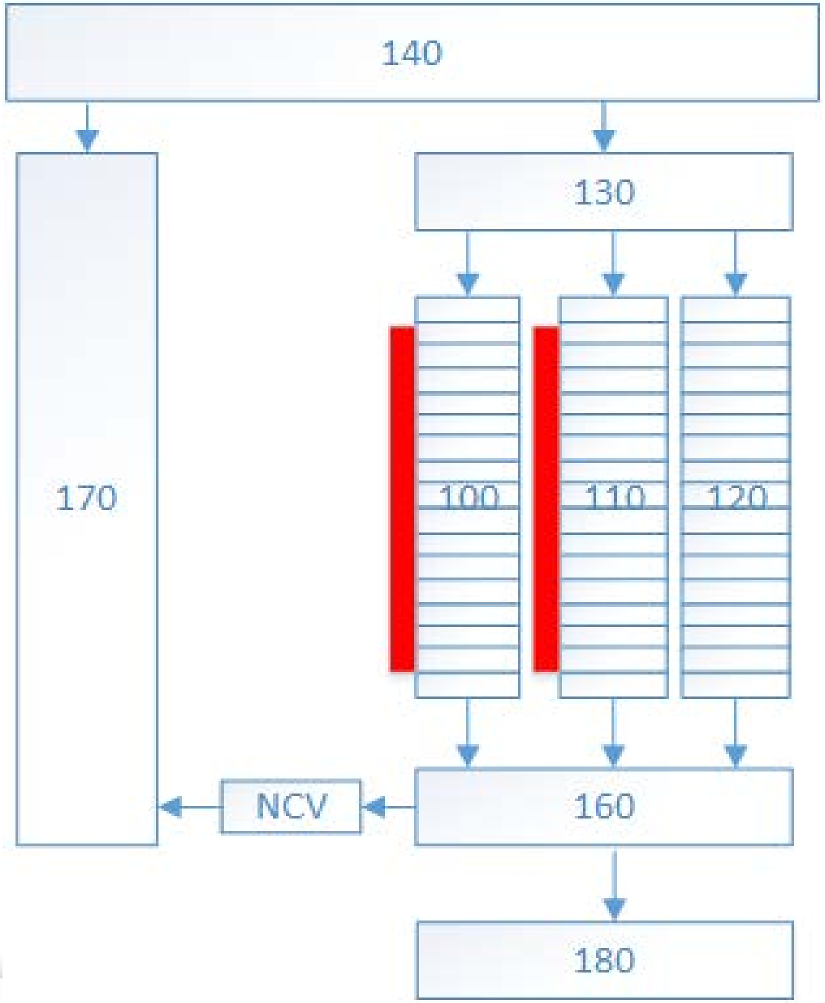
- 18 fuel element distributed to two splitted half cores
- Fueld with low enriched uranium (LEU) – U_3Si_2-Al
- Cooled by forced downward circulation
- Moderated by heavy water

Modeling Codes Used in this Work

- **PARET**
  - Developed by Argonne National Laboratory (ANL) for plate-type research reactor safety analyses.
  - Consists of a one-dimensional T/H model and a point-kinetics model
  - Modular channel analysis code: unable to model complete cooling loops in the reactor

- **Relap5-3D**
  - Developed by Idaho National Laboratory (INL) for for the analysis of transients and accidents in water-cooled nuclear power plants.
  - Multidimensional thermal hydraulics and neutron kinetic modeling capabilities.
  - Able to model complete cooling loops in the reactor.

Computational Models for the Reactor Core



Boundary Conditions

- Time-dependent control volumes and junctions

Hydrodynamic channels

- Hot, average and bypass channel
- Divided into 17 control volumes
- Reactor pool

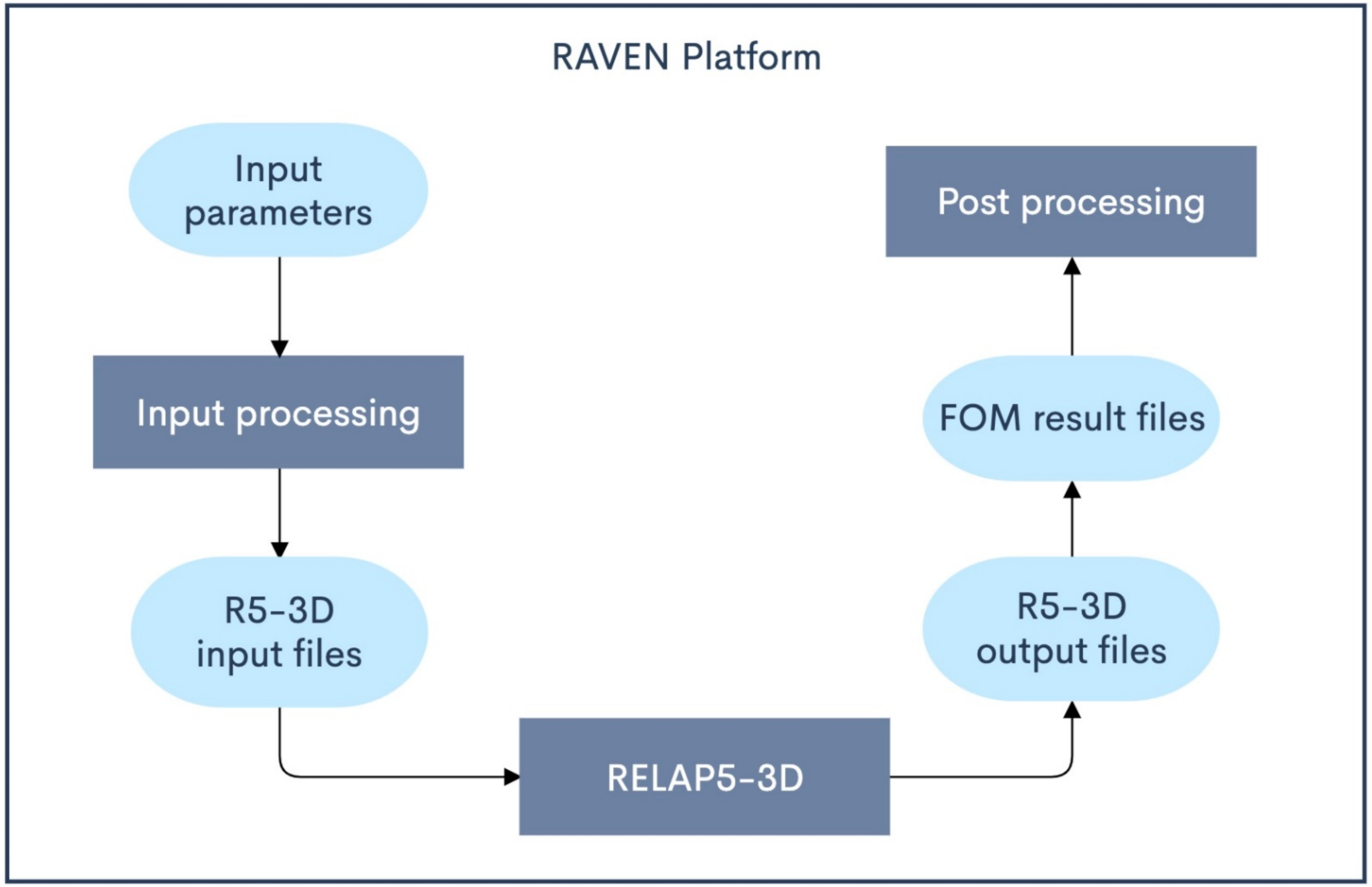
Upper and bottom plenum

- Branch

Fuel element

- Heat structures

Uncertainty Quantification Procedure



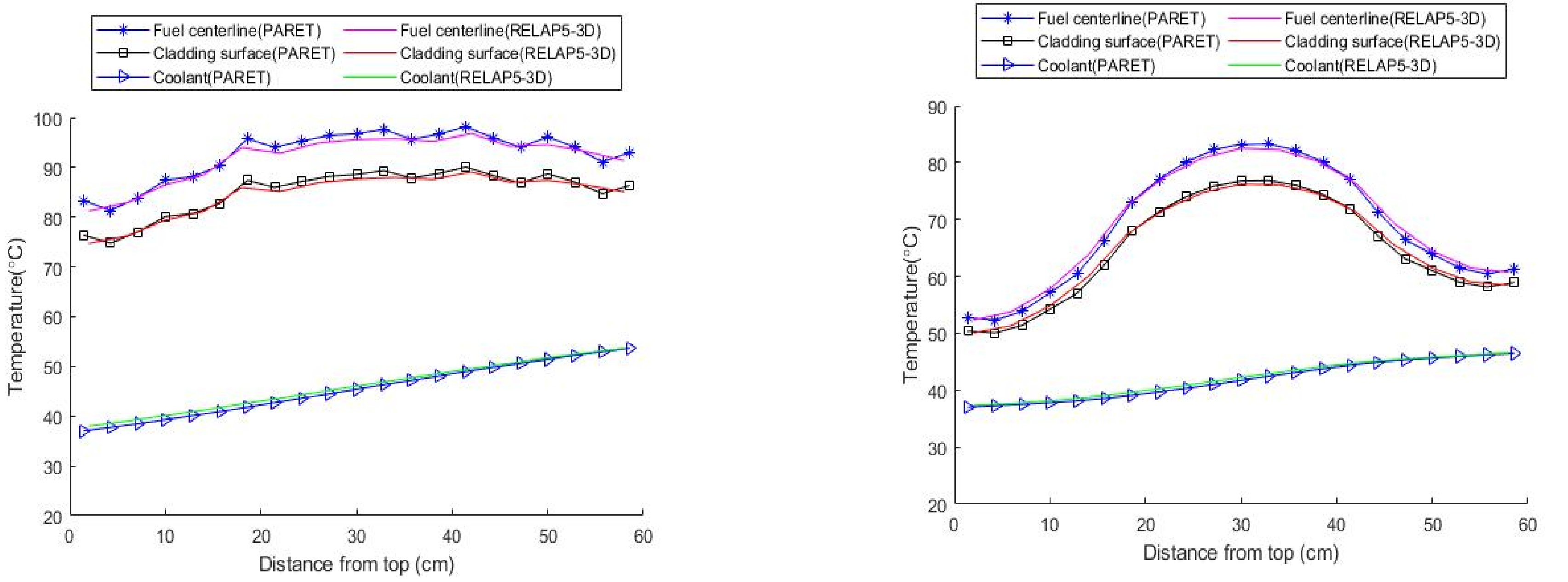
- **RAVEN**: Risk Analysis Virtual Environment
- **Uncertainty quantification** were carried out with RELAP5-3D coupled to the data analysis code RAVEN

Protected Loss of Flow Accident - Description

- The flow rate reduction caused by the pump coastdown is assumed to follow an exponential function exp $(-t/\tau)$, where τ is considered as the time constant of the flow rate decay. In this study, the time constant τ is set to be 1 s to mimic the fast PLOFA.
- During the LOF transients, the reactor SCRAM is tripped by a low coolant flow signal when the coolant flow reaches 85% of its nominal operation value.
- The safety control rods react to the trip signal with a time delay of 0.2 s. This short delay is considered to account for the reaction time needed by mechanical and electronic circuit operations.
- All reactivity feedback effects and period trip are neglected in the analyses

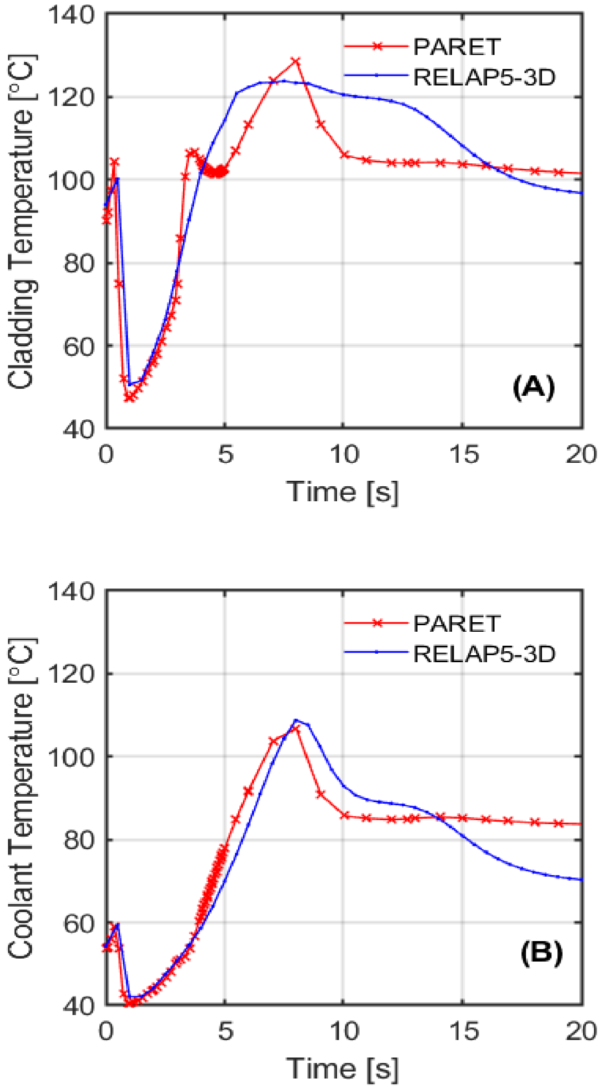
Steady-State Conditions

The steady-state results are compared against PARET results to verify the correctness of the modeling procedure and outcome.



Temperatures of hot (left) and average channel(right) in the steady-state

PLOFA Transient Results



Properties	R5-3D	PARET	Deviation
1st PCT[1] [°C]	100.25	104.57	4.13%
1st PCT time [s]	0.50	0.40	25.00%
1st PCoT[2] [°C]	59.47	59.72	0.42%
1st PCoT time [s]	0.50	0.40	25.00%
2nd PCT [°C]	123.81	128.67	3.78%
2nd PCT time [s]	7.5	8.00	6.25%
2nd PCoT [°C]	108.77	106.76	1.88%
2nd PCoT time [s]	8.00	8.00	0.00%

[1]PCT = Peak cladding temperature
[2]PCoT = Peak coolant temperature

Sensitivity and Uncertainty Analysis

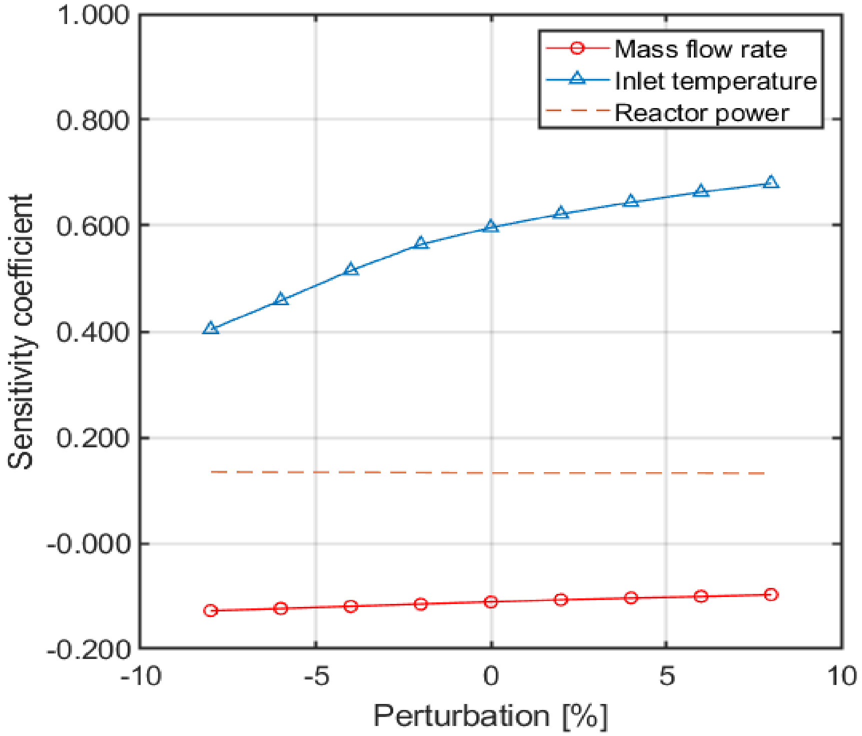
- **Figure of Merit (FOM):**
  - Peak cladding temperature (PCT) and Peak coolant temperature (PCoT)

- **Input Parameters of Interest:**

Uncertain parameter	Nominal value	Uncertainty range	Distribution
Inlet coolant Temp. [°C]	37	±10%	Normal
Inlet coolant mass flow rate [kg/s]	516.83	±10%	Normal
Reactor core power [MW]	20	±10%	Normal

Sensitivity Analysis Results and Discussion

- **Relative Sensitivities of Input Parameters at steady state** $\alpha = \frac{x_0}{R_0}\frac{\partial R}{\partial x} \simeq \frac{x_0}{R_0}\frac{R(x+h)-R(x-h)}{2h}$



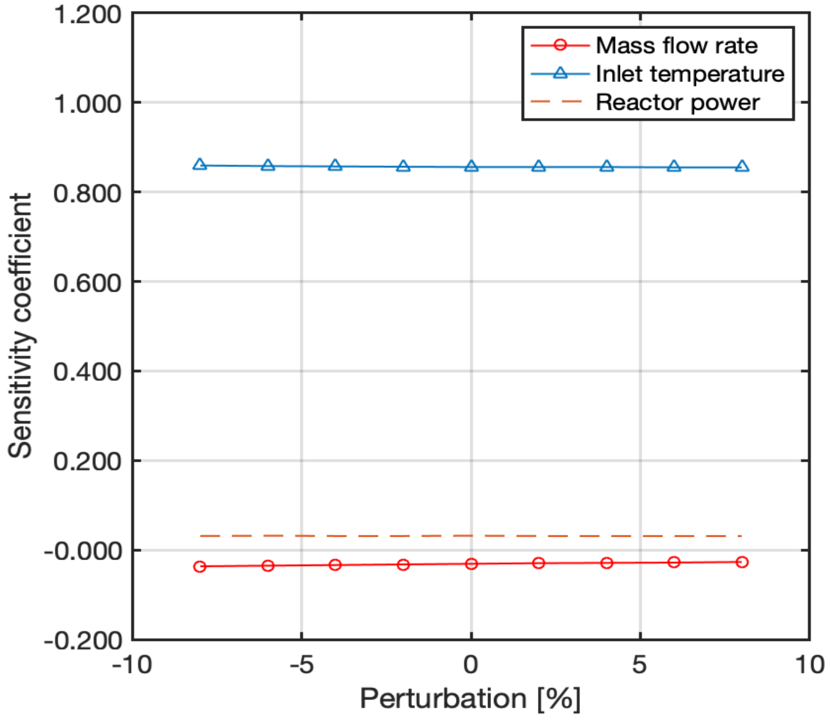


Fig. 9: Sensitivity coefficients of PCT (left) and PCoT (right)

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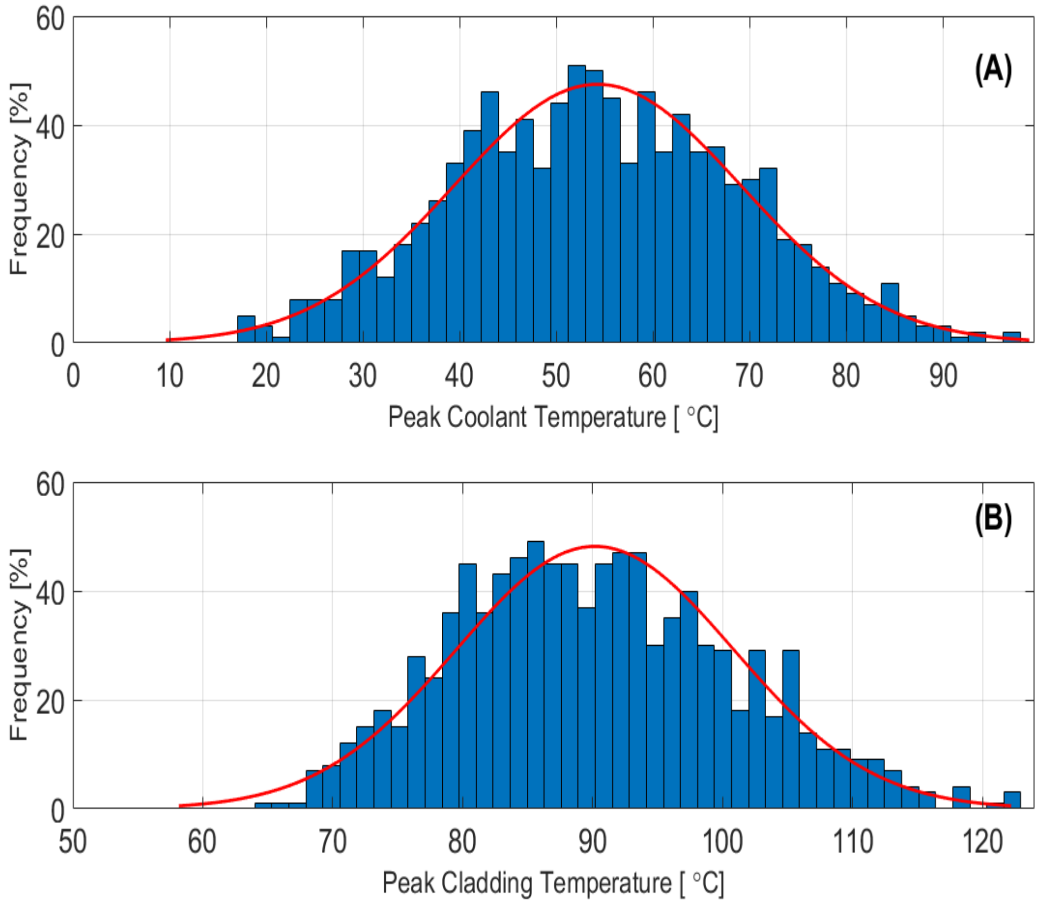
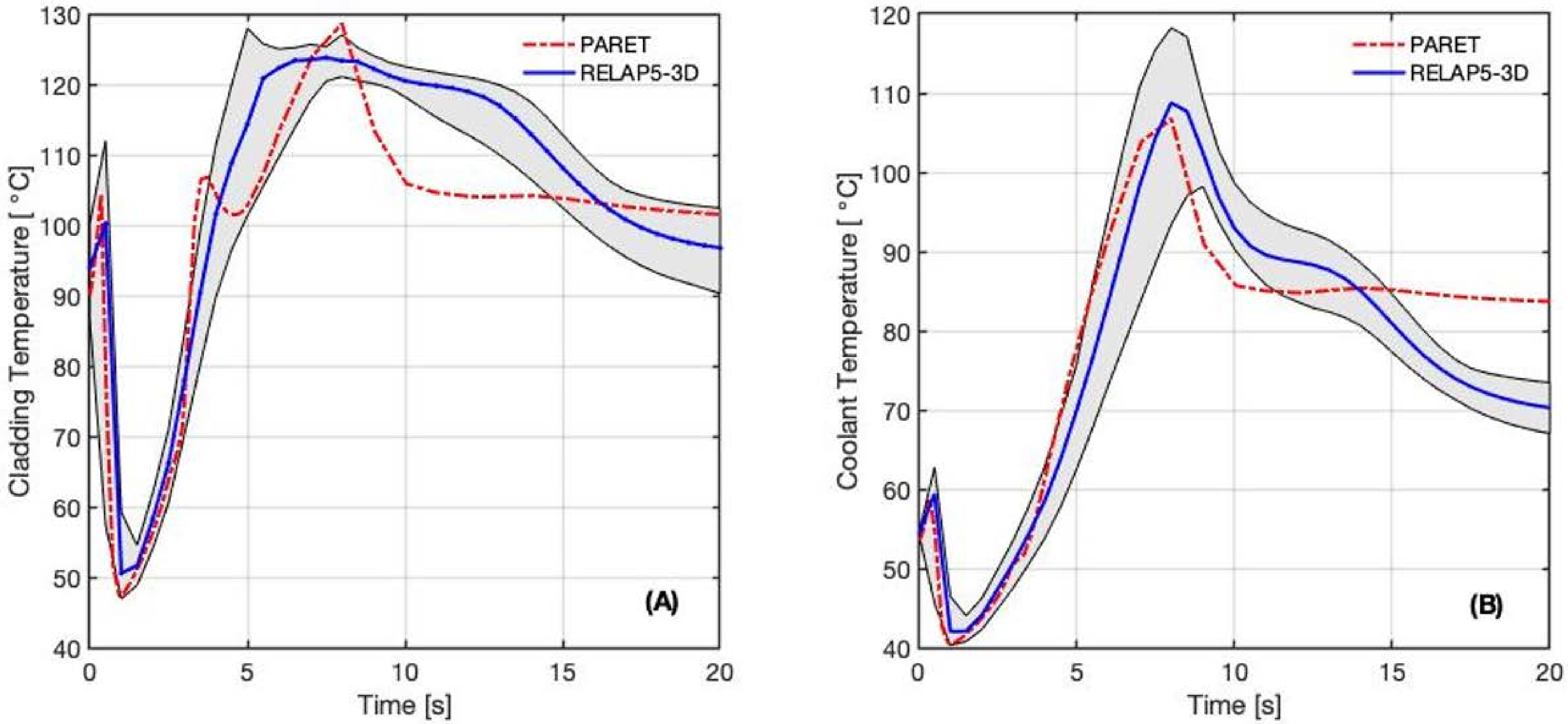
Uncertainty Analysis Results at Steady State



Peak Temperature Distribution Statistics

	PCoT [°C]	PCT [°C]
Mean	54.25	90.17
Standard Dev.	14.88	10.67
95% Lower C.L.	53.32	89.51
95% Upper C.L.	55.18	90.84
Maximum	97.74	122.83

Uncertainty Analysis Results for PLOFA



Conclusions

- This work presents a **sensitivity** and **uncertainty** incorporated reactor safety analysis for research reactors under the framework of **RELAP5-3D** and **RAVEN**.
- A design basis **protected LOF accident** is used as a representative transient accident for this work.
- The **relative sensitivities** obtained from the sensitivity analysis procedure reveals insights of different level influencing impacts of different input variables on the responses.
- The **uncertainty analysis** informs the deviations of the responses contributed by the errors of various input components.

Thank you!

Questions?

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