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**Đorđe Petrović :: Advanced Nuclear Systems :: Paul Scherrer Institute**

# Coupled Neutronic/Thermal-Hydraulic Simulation of Unprotected Loss of Flow Test at Fast Flux Test Facility

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# Outline

- Introduction
- Benchmark Specifications
- Simulation Tools Applied
- Application to Fast Flux Test Facility
- Results
- Summary and Conclusions

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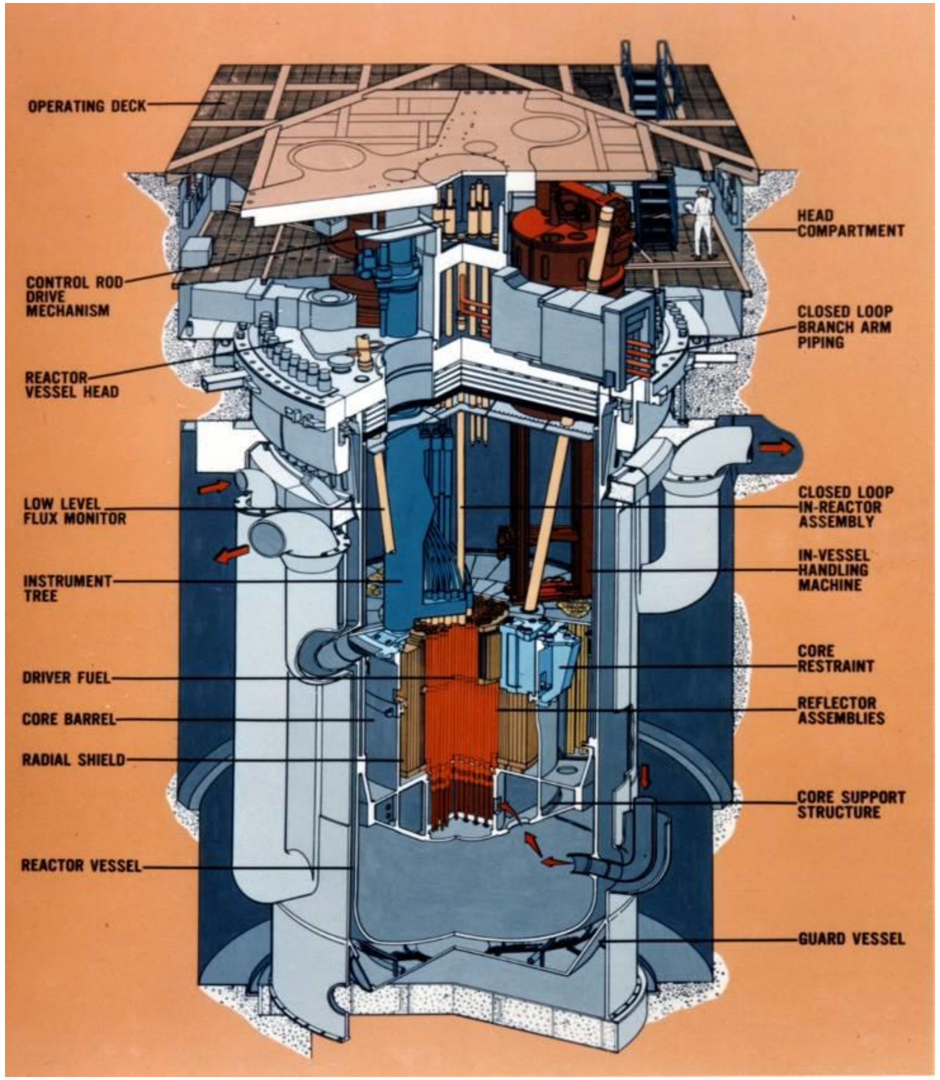
# Introduction: Fast Flux Test Facility

- Fast Flux Test Facility (FFTF)
  - Research reactor operated by U.S. Department of Energy
  - Thermal power: 400 MW
  - Coolant: Sodium
  - Fuel: Mixed-OXide, $UO_2$ - $PuO_2$

- Passive Safety Testing program
  - 13 unprotected **Loss of Flow WithOut SCRAM** (LOFWOS) tests
  - Confirmation of safety margins of Sodium-cooled Fast Reactor (SFR) design
  - Provision of data for **computer code validation**
  - Demonstration of inherent and passive safety benefits of **Gas Expansion Modules** (GEM)

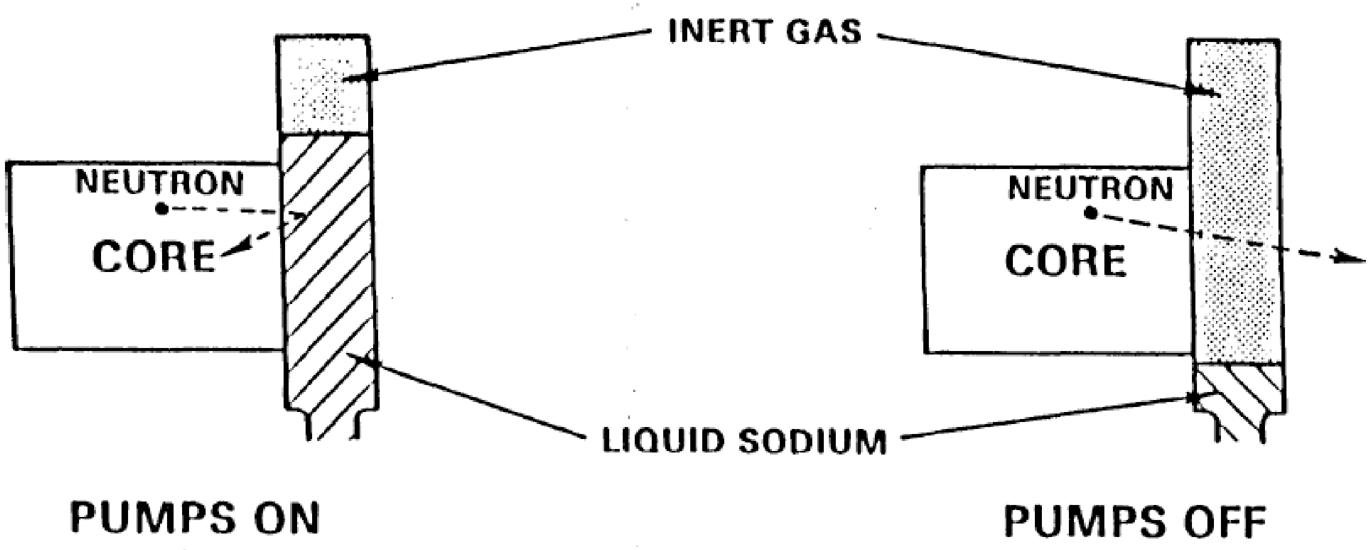


# Introduction: Loss of Flow Without SCRAM Test # 13 Benchmark

- Proposed by Argonne National Laboratory and Pacific Northwest National Laboratory
- IAEA Coordinated Research Project ‘Benchmark Analysis of FFTF Loss of Flow Without Scram Test’

- LOFWOS Test #13
  - Power: 49.8% of nominal power
  - Flow rate: 100% of nominal flow rate

- Special attention devoted to GEMs
  - Greatest fraction of **passive reactivity control system**
  - Principle of operation



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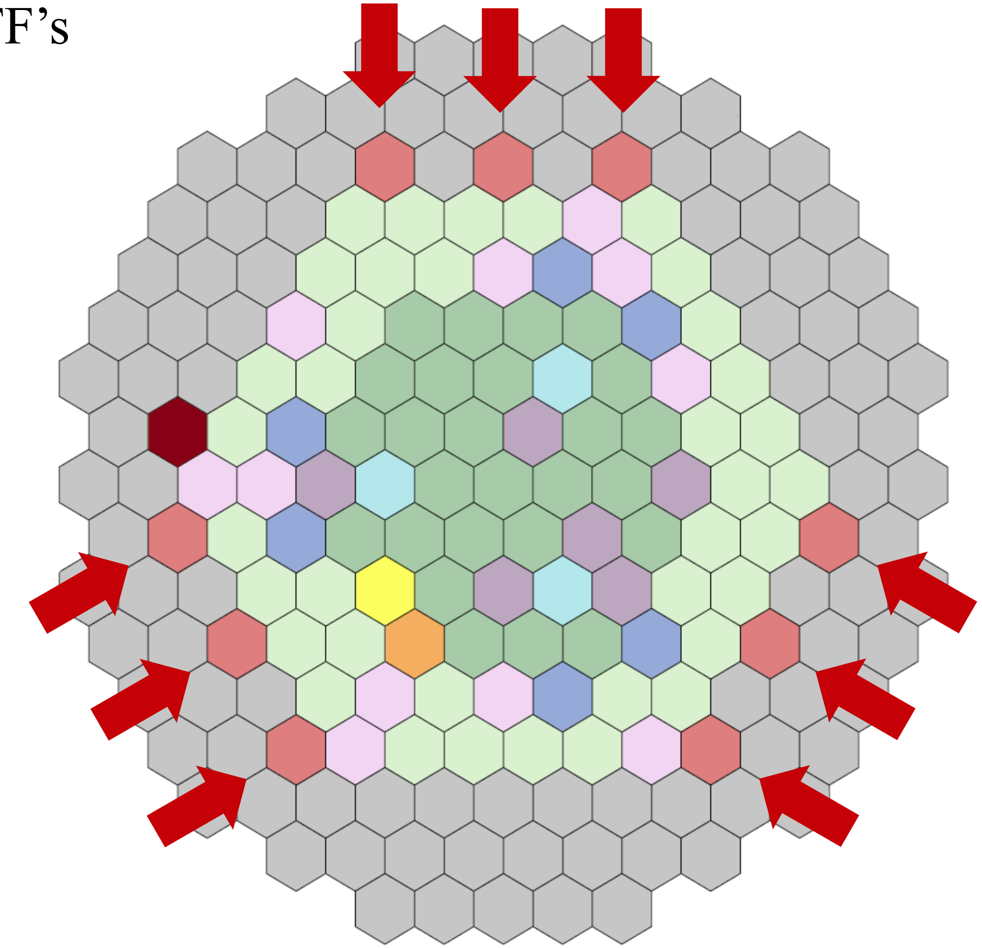
# Benchmark Specifications: Cycle 8C Core Configuration

- LOFWOS Test #13 performed during cycle 8C of FFTF's operation
- Assembly types present in the core during the corresponding cycle:
  - Driver Fuel Assembly
  - In-Core Shim Assembly
  - Reflector Assembly
  - Control Rod
  - Safety Rod
  - Materials Open Test Assembly
  - Fracture Mechanics Assembly
  - **Gas Expansion Module**



# Benchmark Specifications: Gas Expansion Modules

- Nominal conditions: pressure of the Sodium **compresses** the **gas** to a level above the top of the active fuel column

- Loss of flow transient: the pressure exerted on the gas by the Sodium decreases, allowing the **gas** to **expand**

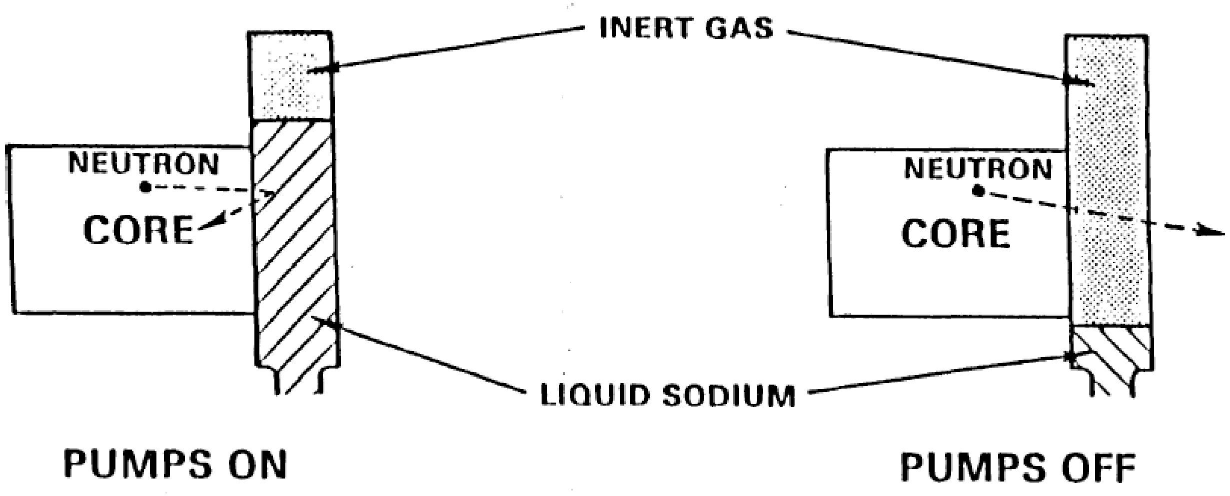
- Low flow rates: the Sodium-gas interface level within each GEM would be below the bottom of the active fuel column. The displaced Sodium at the periphery of the core leads to the **increased radial neutron leakage** and the corresponding **decrease** of the **core reactivity**

Pressure ↘

Sodium Level ↘

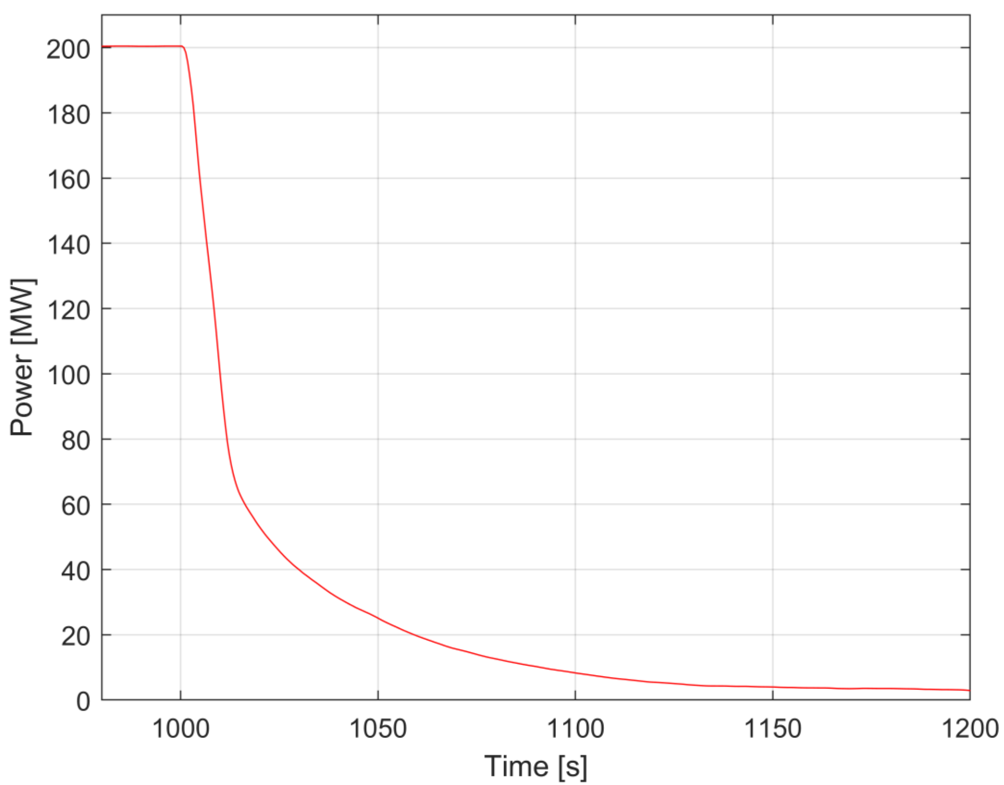
Radial Leakage ↗

Reactivity ↘

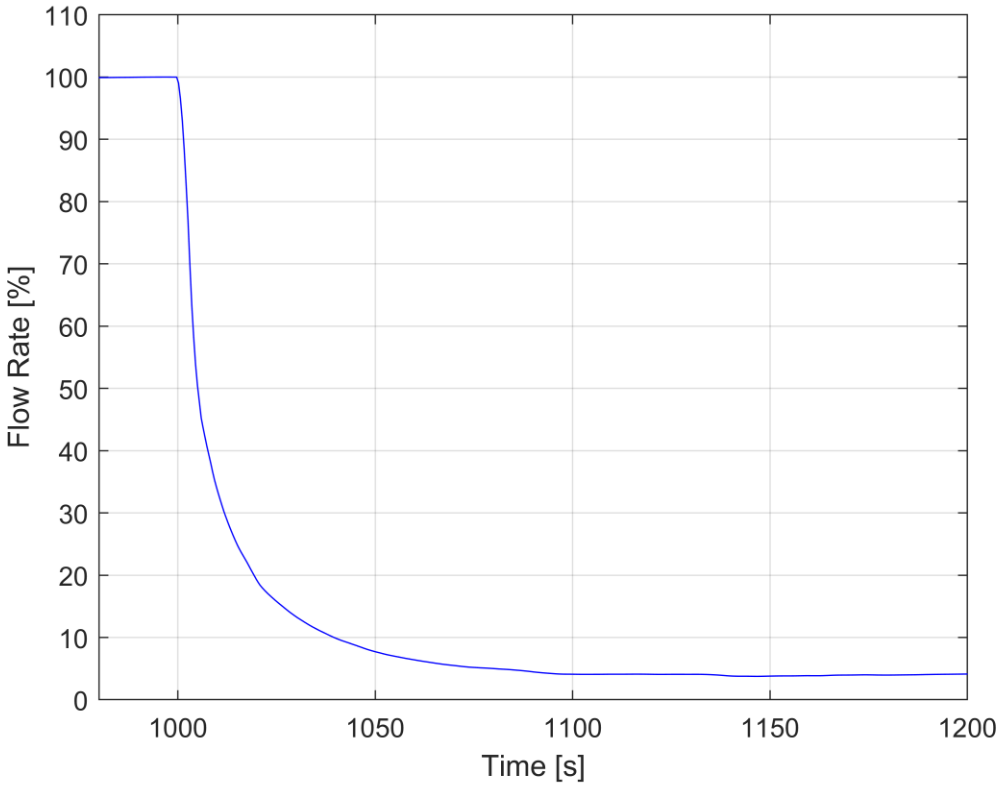


# Benchmark Specifications: Experimental Data

Evolution of the core **power** as **measured** [1]




Evolution of the core **flow rate** as **measured** [1]

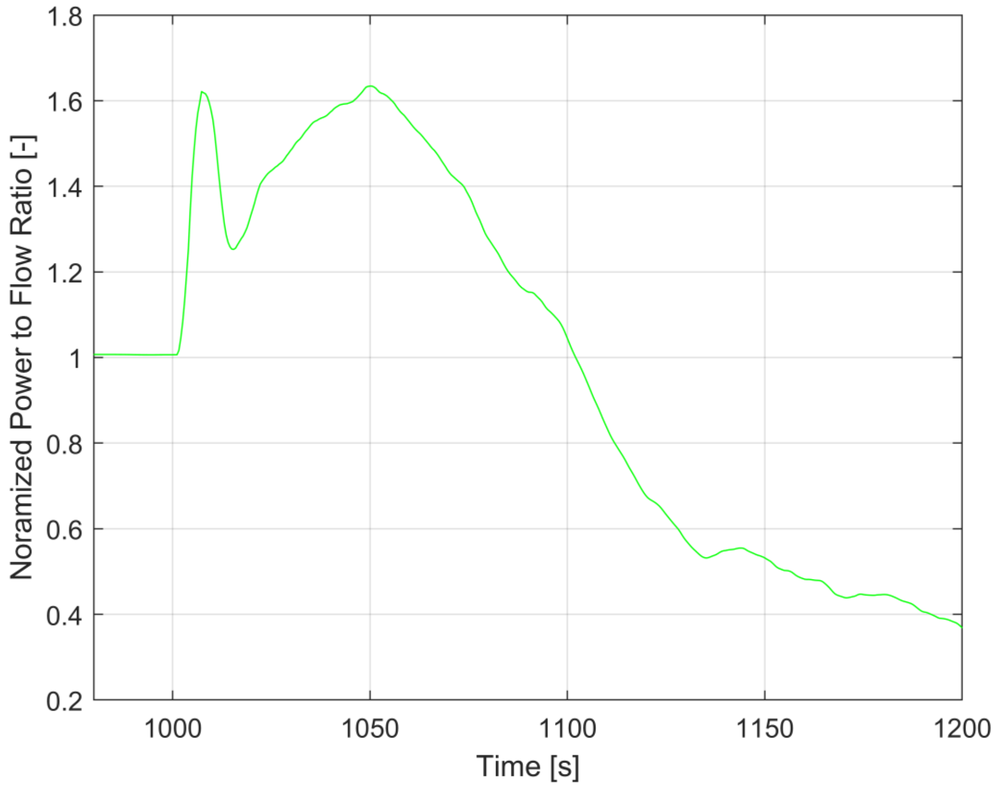




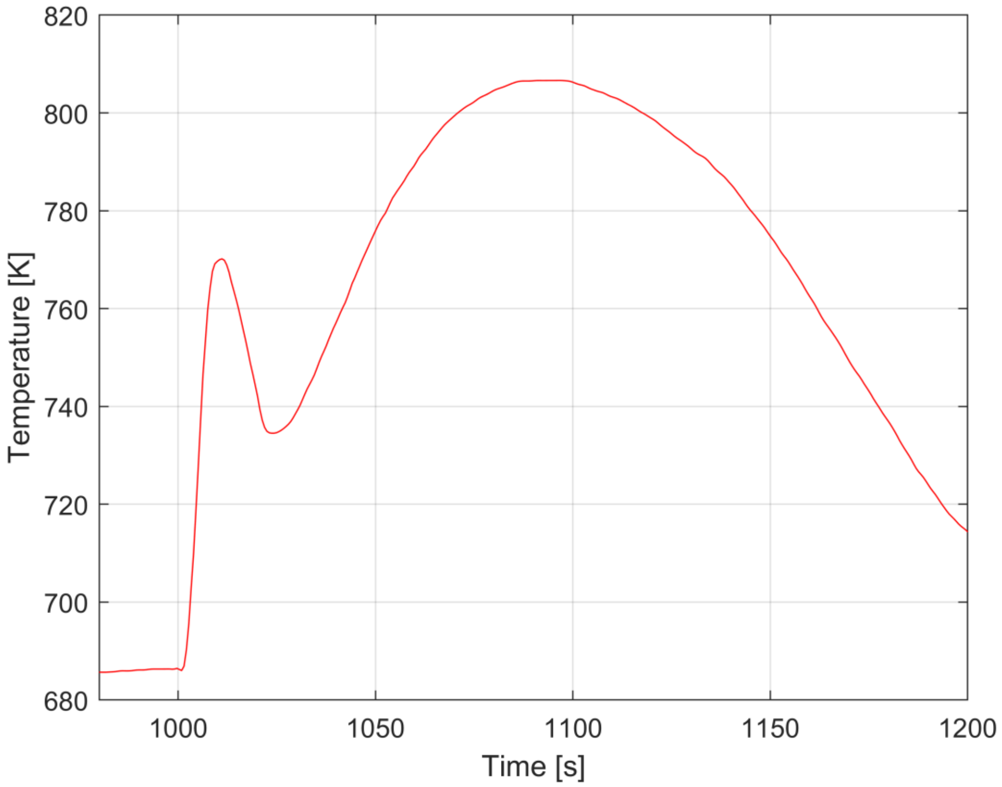
[1] Lucoff, D. M., September 1987, 'Passive Safety Testing at the Fast Flux Test Facility', WHC-SA-0046-FP, Westinghouse Hanford Company, Washington, United States.

# Benchmark Specifications: Experimental Data

**Calculated** evolution of **power-to-flow** ratio



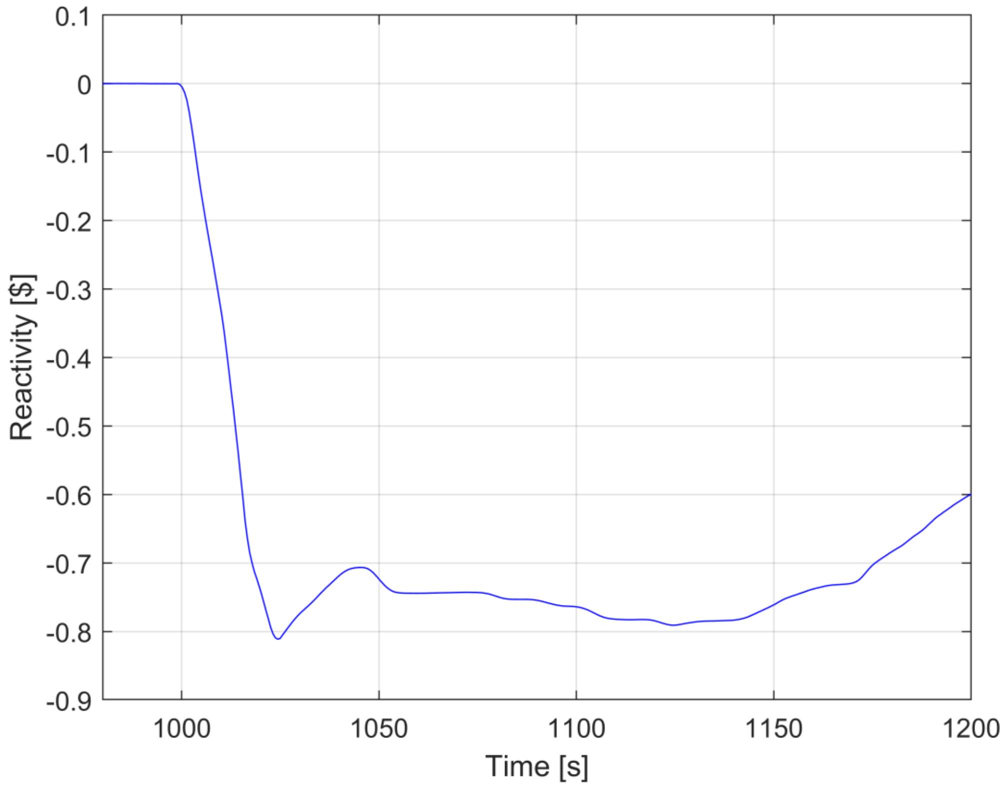

Evolution of the **coolant** outlet **temperature** as **measured** [1]



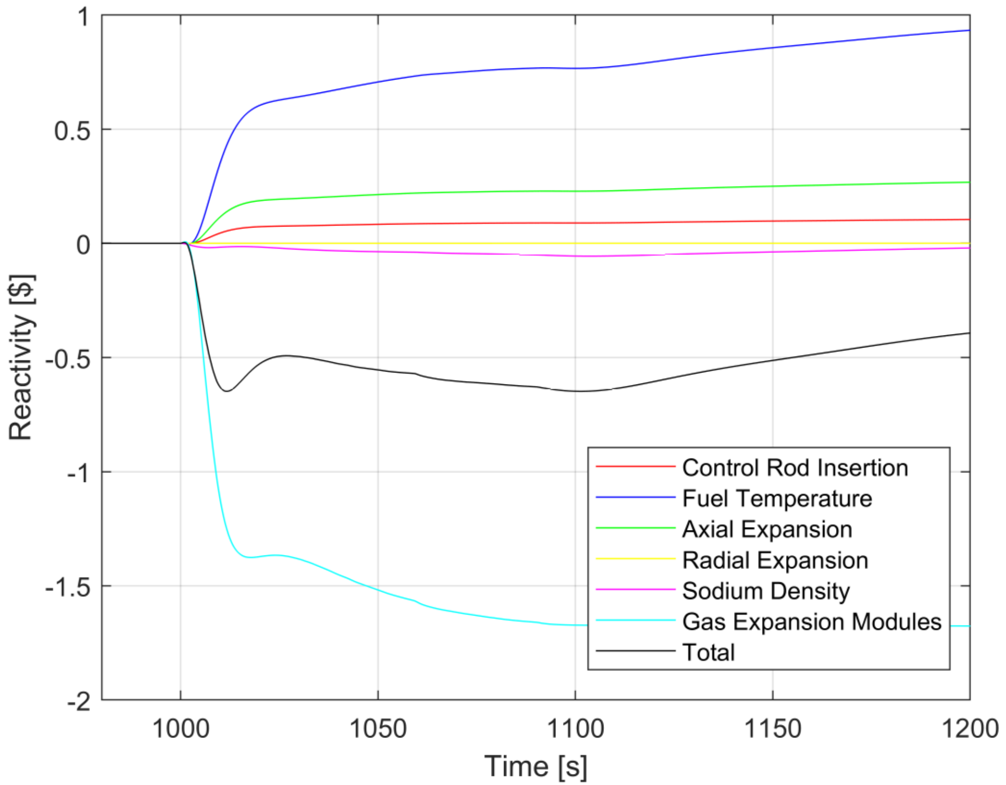

[1] Lucoff, D. M., September 1987, 'Passive Safety Testing at the Fast Flux Test Facility', WHC-SA-0046-FP, Westinghouse Hanford Company, Washington, United States.

# Benchmark Specifications: Experimental Data

Evolution of the core **reactivity** as **measured** [1]




**Breakdown** of the core **reactivity** evolution as **simulated** by Point Reactor Kinetics model




[1] Lucoff, D. M., September 1987, 'Passive Safety Testing at the Fast Flux Test Facility', WHC-SA-0046-FP, Westinghouse Hanford Company, Washington, United States.

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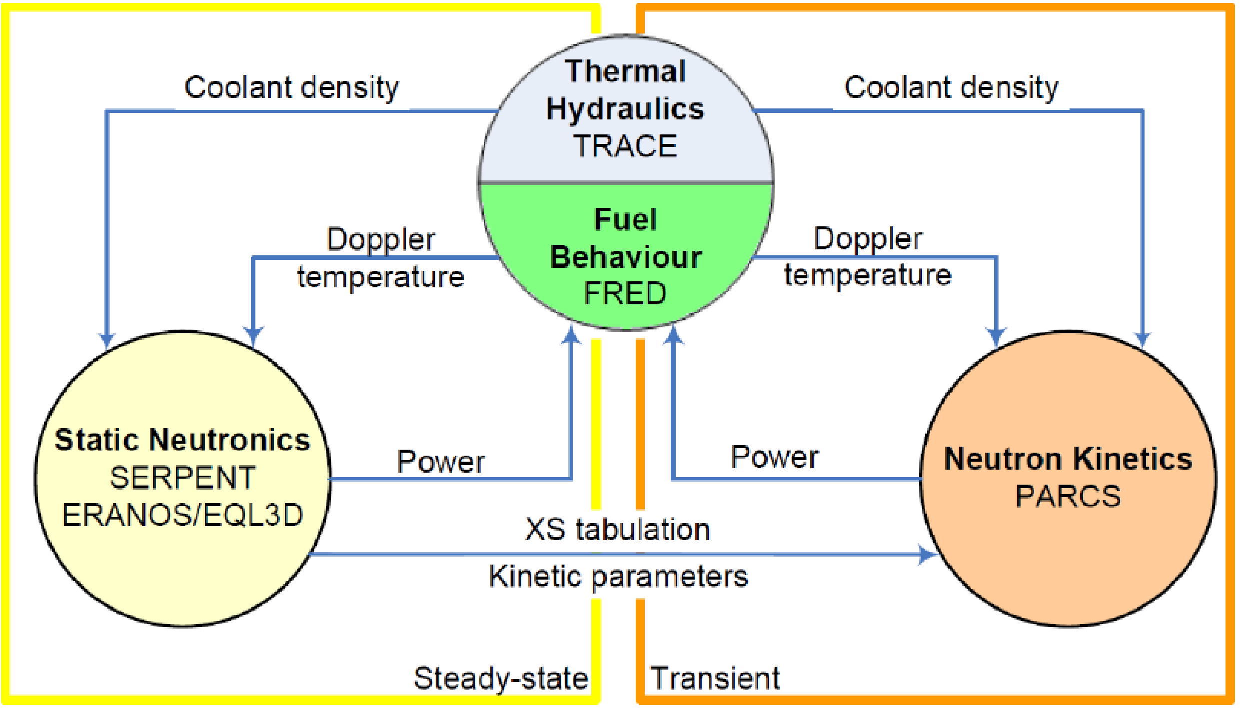
# Simulation Tools Applied: FAST Code System

- Fast-spectrum Advanced Systems for power production and resource managemenT (FAST) code system
  - A general tool for the analysis of core **statics** and **dynamic behavior** of advanced fast spectrum reactor concepts
  - Assembled from **already existing codes**, which, where necessary, have been **modified** to simulate the features of the fast reactors

- Constituents
  - **Serpent 2** Monte Carlo code
  - **PARCS** reactor kinetics code
  - **TRACE** thermal-hydraulics code
  - **FRED** thermal-mechanics code



# Simulation Tools Applied: Constituents and Modifications

**Serpent 2: A Continuous-energy Monte Carlo Reactor Physics Burnup Calculation Code**

- Nuclear data library: **ENDF/B-VII.0**

**Purdue Advanced Reactor Core Simulator**

- Modification of macroscopic cross-section data calculation
  - Logarithmic dependence of core reactivity on fuel temperature
  - Radial expansion
  - Axial expansion
  - Relative insertion of control rods

**TRAC/RELAP Advanced Computational Engine**

- **Rehme** correlation: calculation of friction factor to account for the presence of the wire wraps around the fuel bundles
- **Mikityuk** correlation: calculation of the heat transfer to the liquid metal coolants

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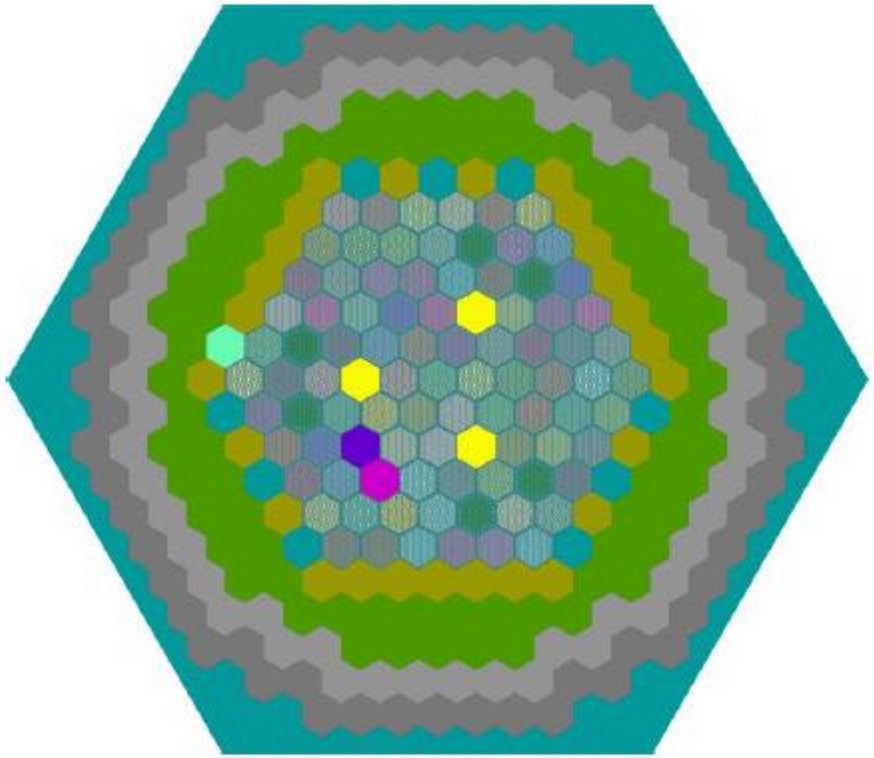
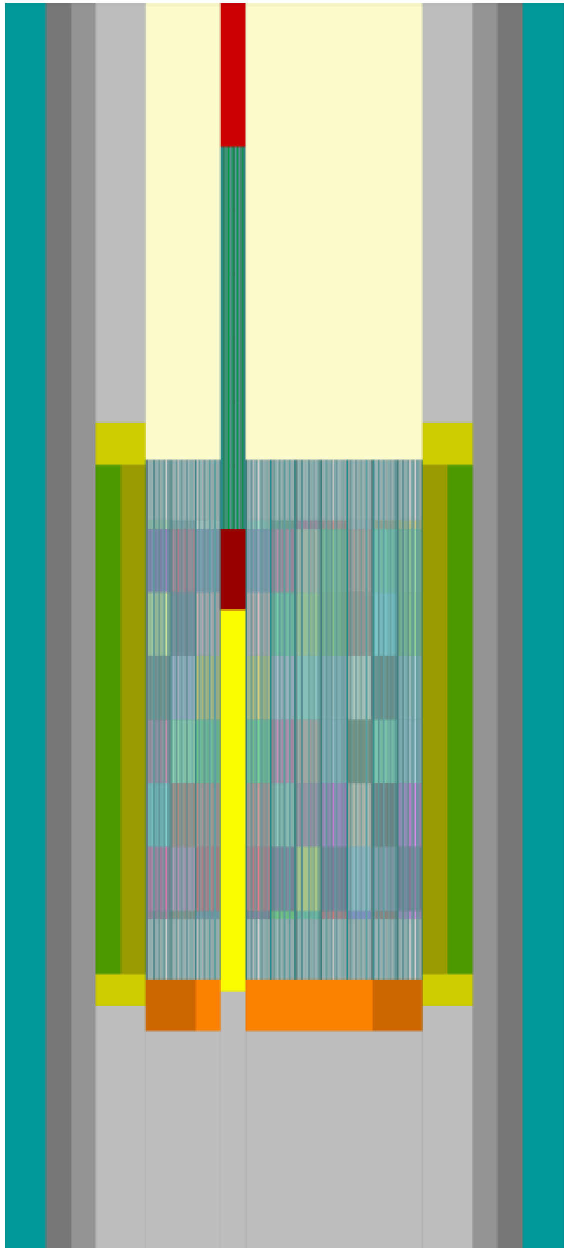
# Application to Fast Flux Test Facility: Static Neutronics Model

- Core **geometry** and **material composition** provided in the benchmark specifications
- Statistical uncertainty on the **order** of **1 pcm**
- Reactivity feedback effects
  - Axial expansion
  - Radial expansion
  - Fuel Doppler constant
  - Fuel density
  - Structure density
  - Sodium density
  - Control rods
  - Safety rods
  - Gas Expansion Modules

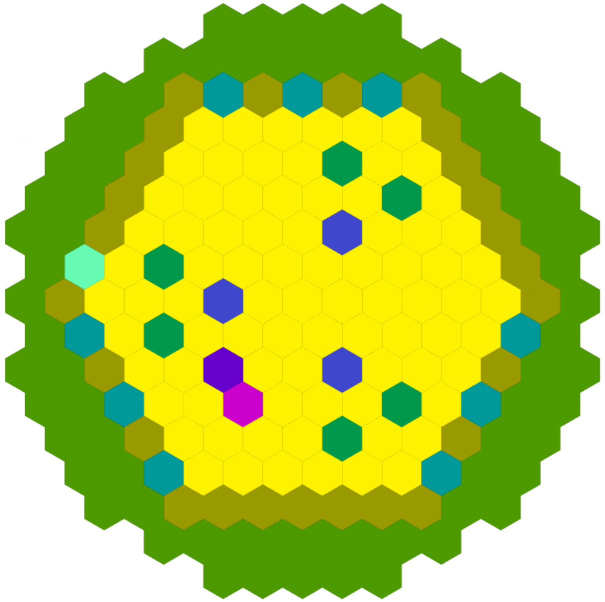


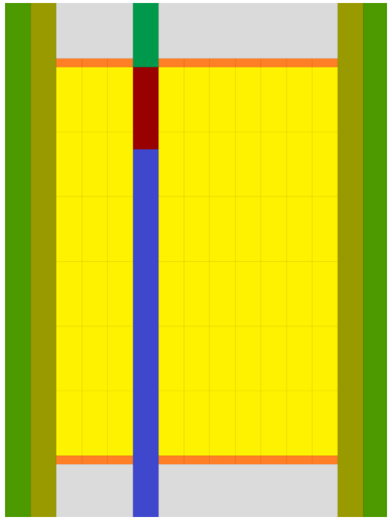


# Application to Fast Flux Test Facility: Neutron Kinetics Model

- **Reduction** of **size** to simplify convergence and coupling
- **Reproduction** of **power distribution** obtained by **static neutronics model**

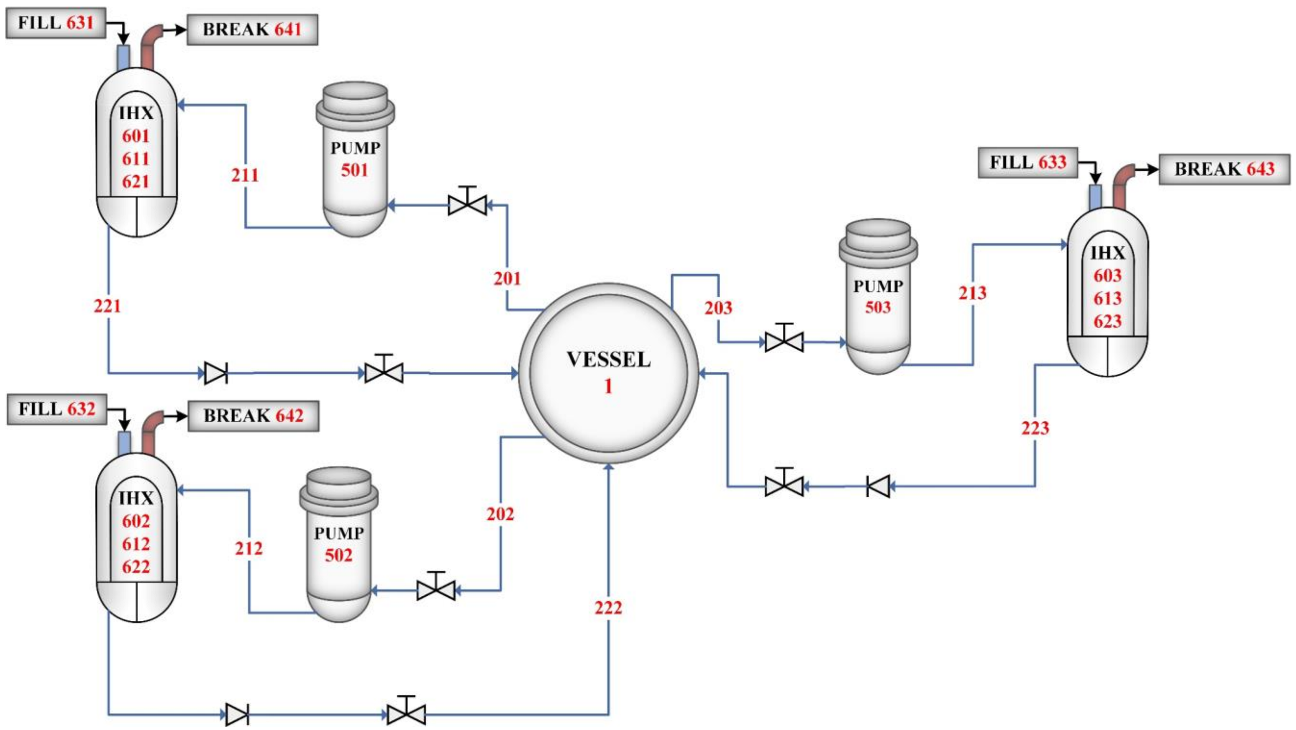
- Number of neutron energy groups: 24
- Number of delay neutron groups: 8





# Application to Fast Flux Test Facility: Thermal-Hydraulics Model

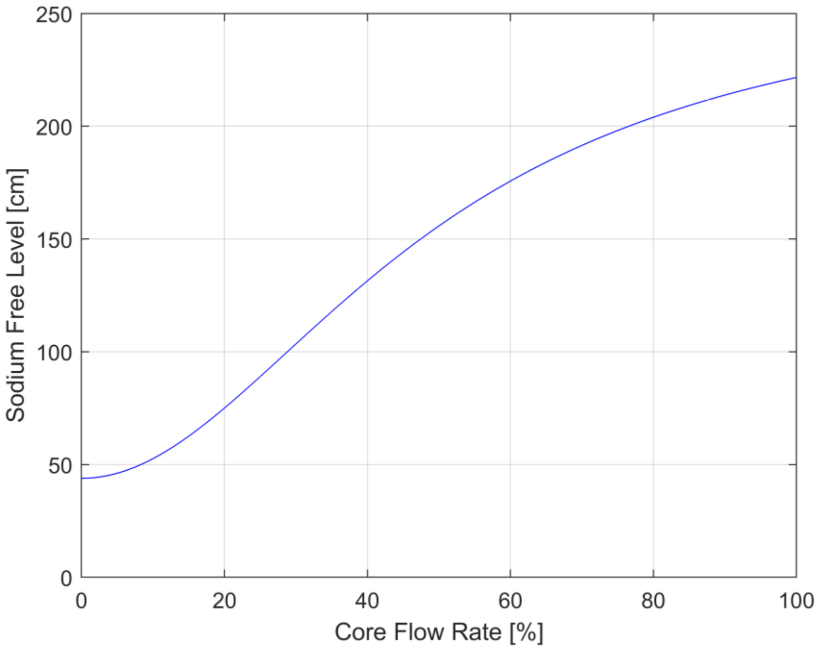
- **Point Reactor Kinetics** approach employed in modeling the core **neutronics**
- **Boundary conditions** provided in the benchmark specifications
  - Pump speeds of the primary pumps
  - Secondary loop flow rates
  - Sodium outlet temperature for 12 dump heat exchanger modules

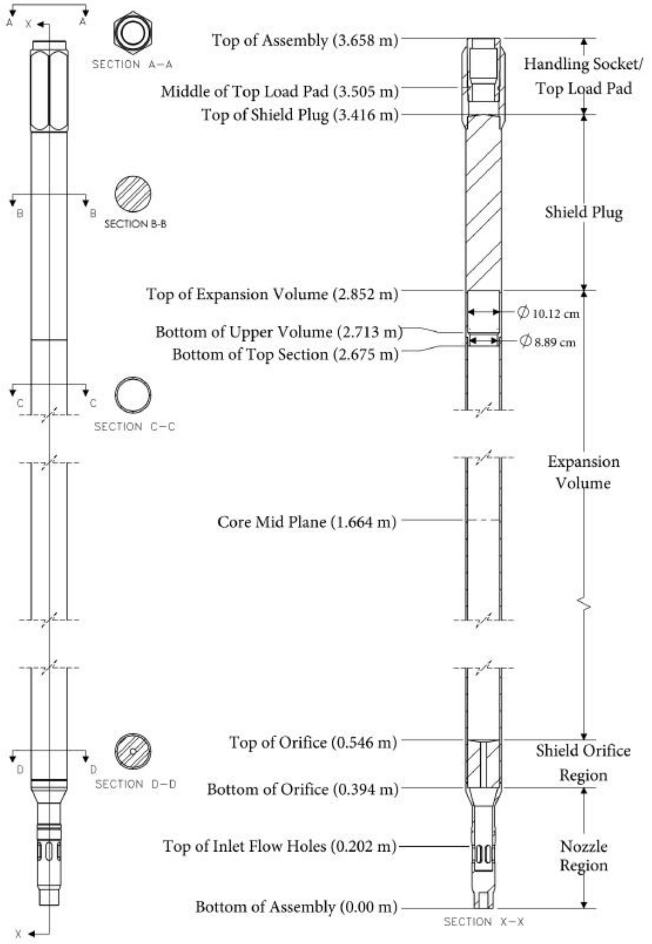


# Application to Fast Flux Test Facility: Coupled Model

- **Coupling scheme** based on the exchange of data
  - Power profiles
  - Temperature and density fields
  - Flow rates

- Special attention devoted to the model of **Gas Expansion Modules**



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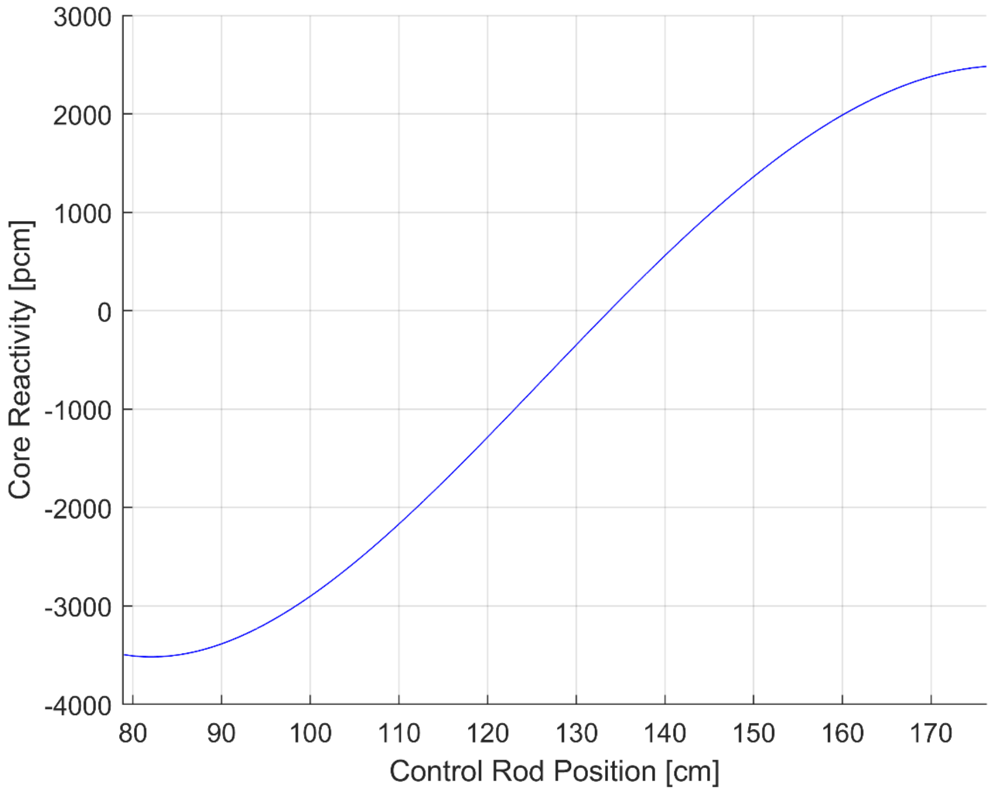
# Results: Static Neutronics Simulation

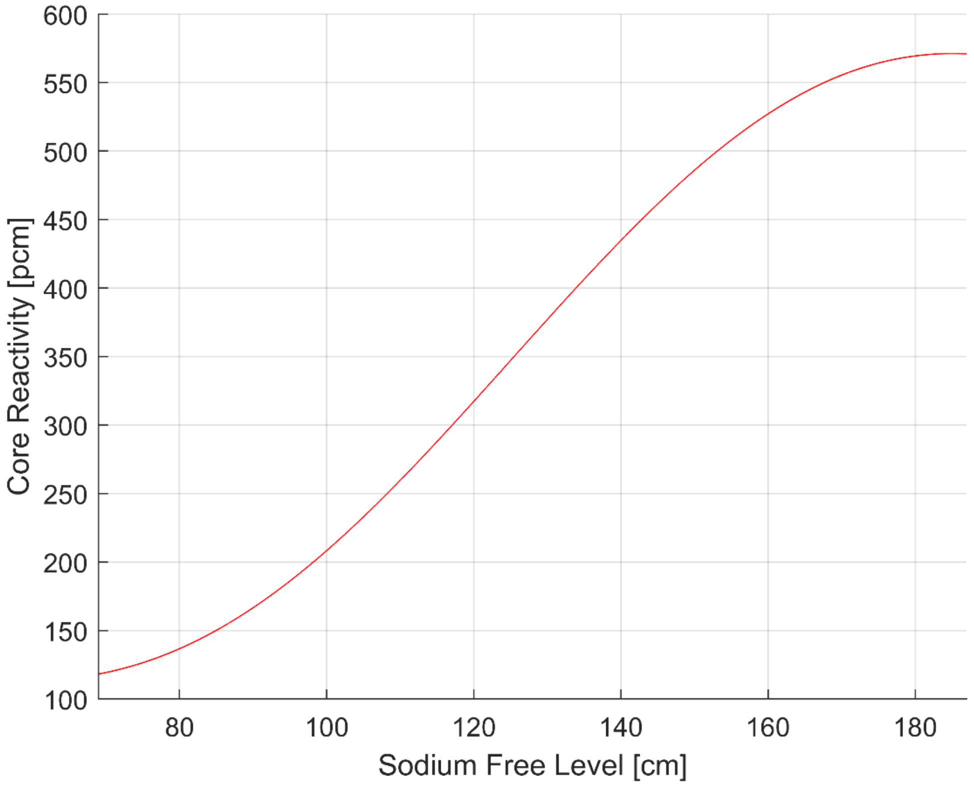
<table>
<tr><th colspan="2">Parameter</th><th>Value</th></tr>
<tr><td colspan="2">Effective Multiplication Factor</td><td>1.00574 ± 0.00003</td></tr>
<tr><td colspan="2">Reactivity</td><td>571 ± 3 pcm</td></tr>
<tr><td colspan="2">Delay Neutron Fraction</td><td>$(3.209 \pm 0.001) \cdot 10^{-3}$</td></tr>
<tr><td colspan="2">Prompt Neutron Lifetime</td><td>$(5.524 \pm 0.001) \cdot 10^{-7}$ s</td></tr>
<tr><td rowspan="6">Reactivity Feedback Coefficients</td><td>Axial Expansion</td><td>-0.221 ± 0.007 pcm/K</td></tr>
<tr><td>Radial Expansion</td><td>-1.522 ± 0.012 pcm/K</td></tr>
<tr><td>Fuel Doppler Constant</td><td>-658 ±10 pcm</td></tr>
<tr><td>Fuel Density</td><td>-1.363 ± 0.020 pcm/K</td></tr>
<tr><td>Structure Density</td><td>-0.039 ± 0.009 pcm/K</td></tr>
<tr><td>Sodium Density</td><td>-0.274 ± 0.023 pcm/K</td></tr>
<tr><td rowspan="3">Reactivity Worths</td><td>Safety Rods</td><td>-5809 ± 10 pcm</td></tr>
<tr><td>Control Rods</td><td>-6014 ± 10 pcm</td></tr>
<tr><td>Gas Expansion Modules</td><td>-475 ± 7 pcm</td></tr>
<tr><td rowspan="2">Incremental Reactivity Worths</td><td>Control Rods</td><td>-8.95 ± 0.65 pcm/mm</td></tr>
<tr><td>Gas Expansion Modules</td><td>-0.49 ± 0.01 pcm/mm</td></tr>
</table>

# Results: Static Neutronics Simulation

S-curve characteristic of **control rods**

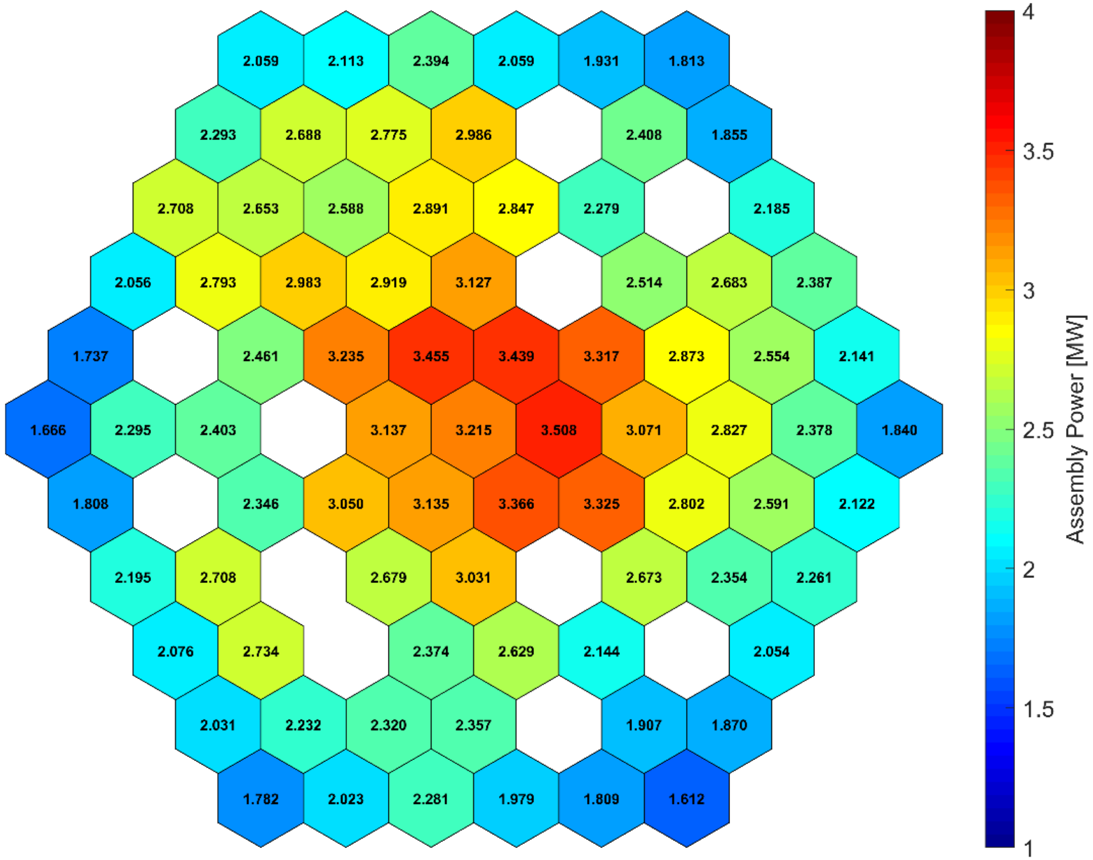



S-curve characteristic of **Gas Expansion Modules**

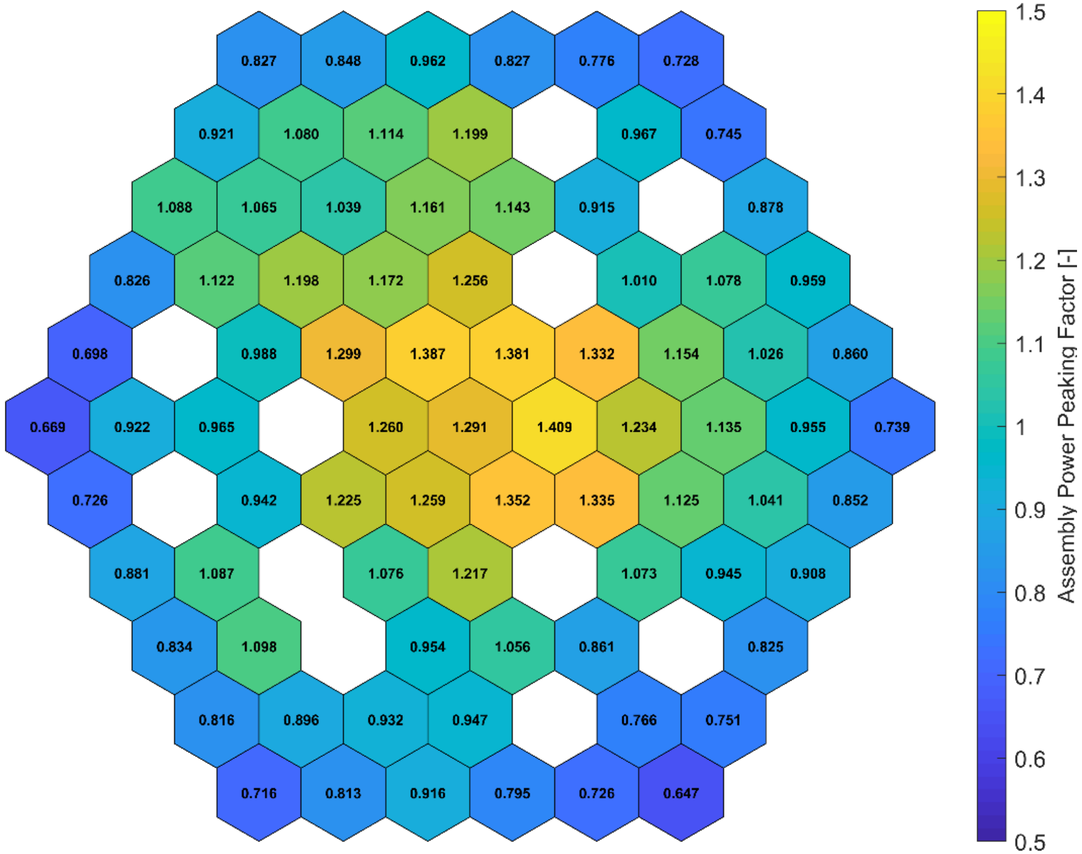


# Results: Static Neutronics Simulation

## Assembly **power** distribution

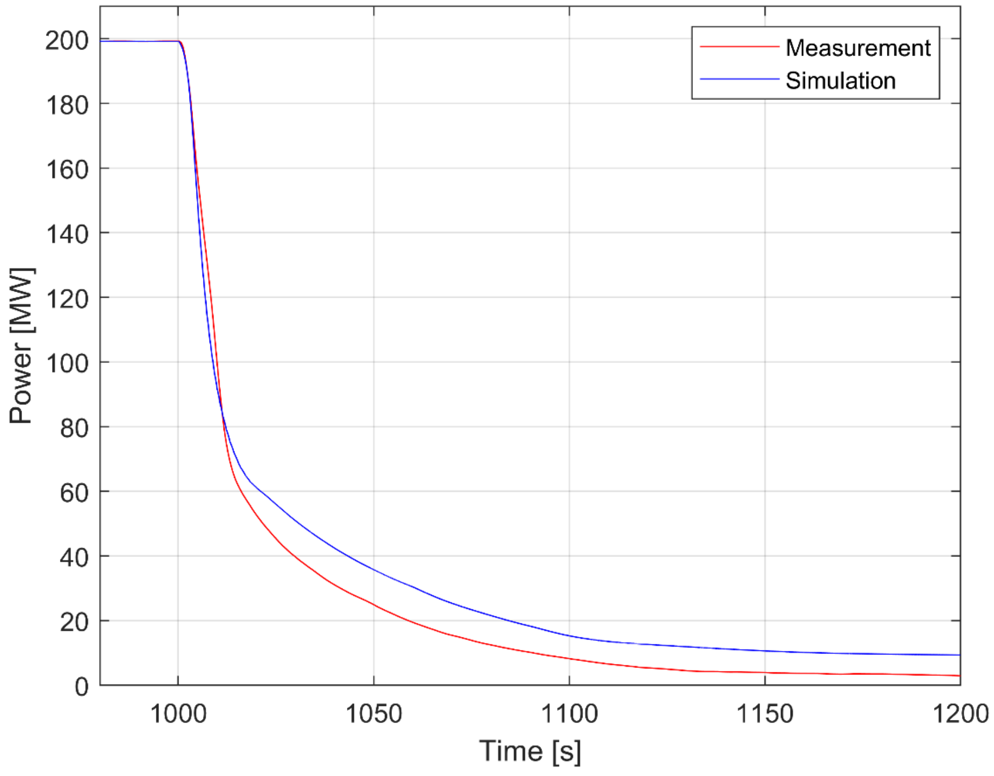



## Map of assembly **power peaking factors**

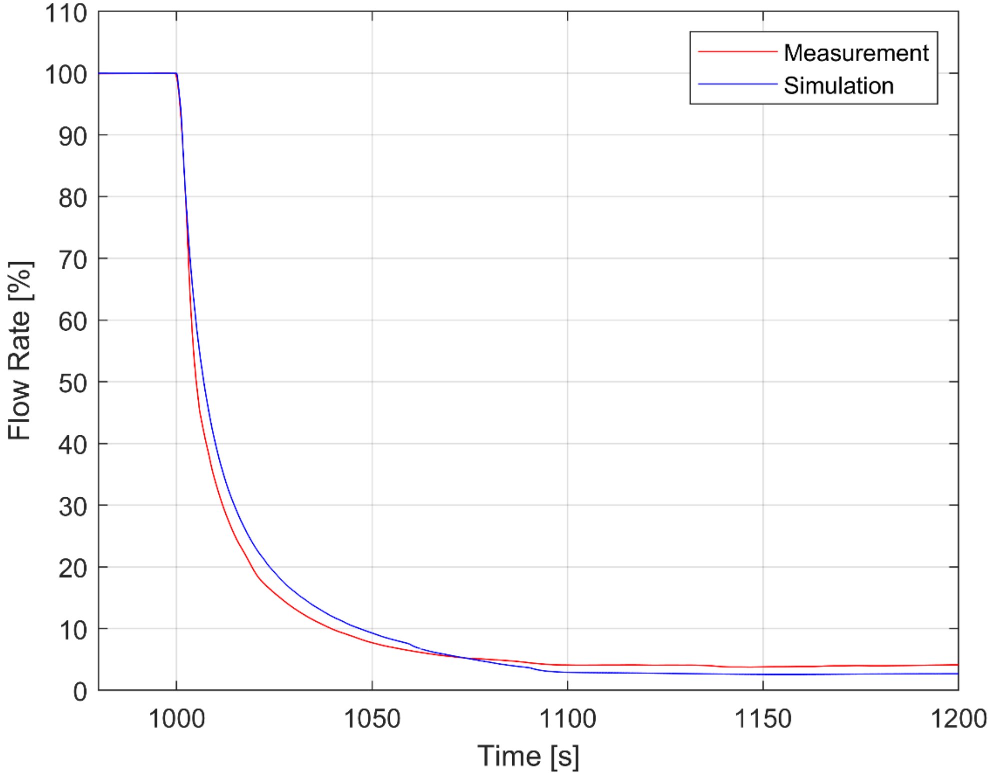


# Results: Coupled Simulation and Comparison to Experimental Data

Comparison of the core **power** evolution to the experimental data

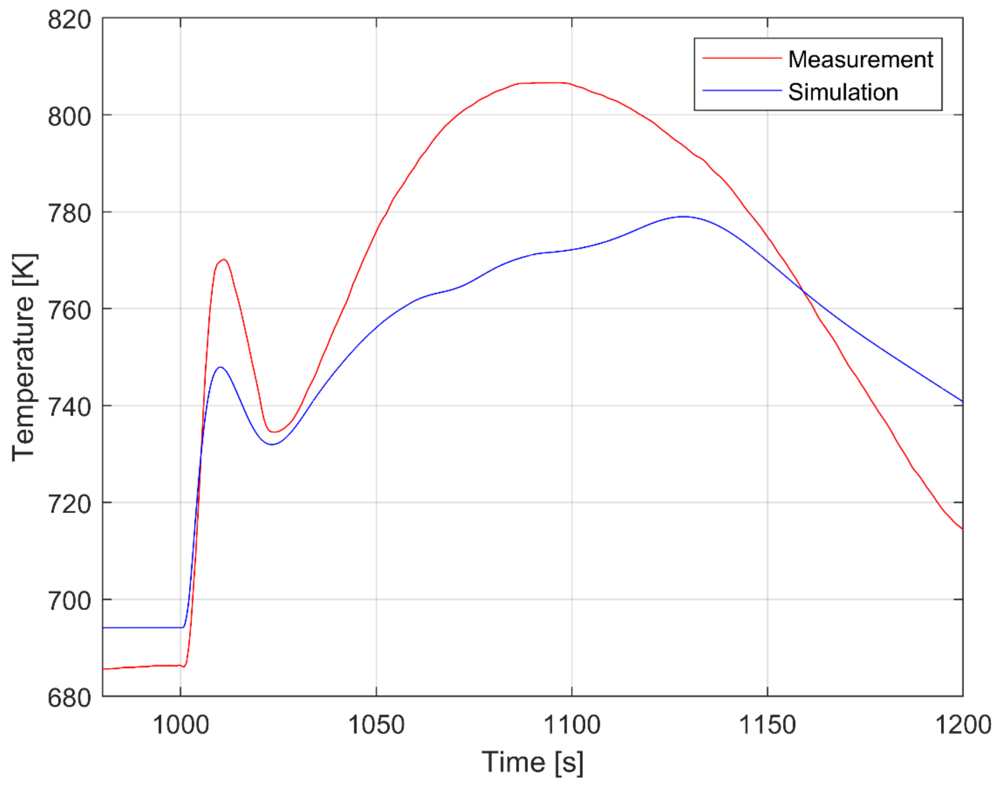



Comparison of the core **flow rate** evolution to the experimental data

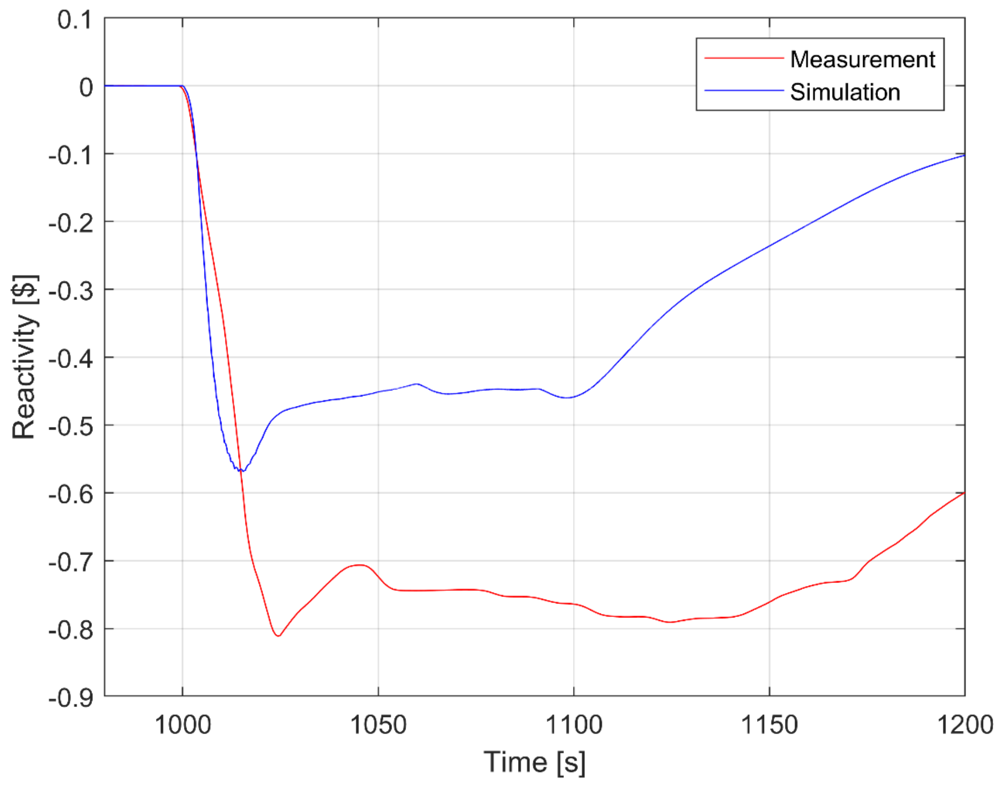


# Results: Coupled Simulation and Comparison to Experimental Data

Comparison of the **coolant** outlet **temperature** evolution to the experimental data

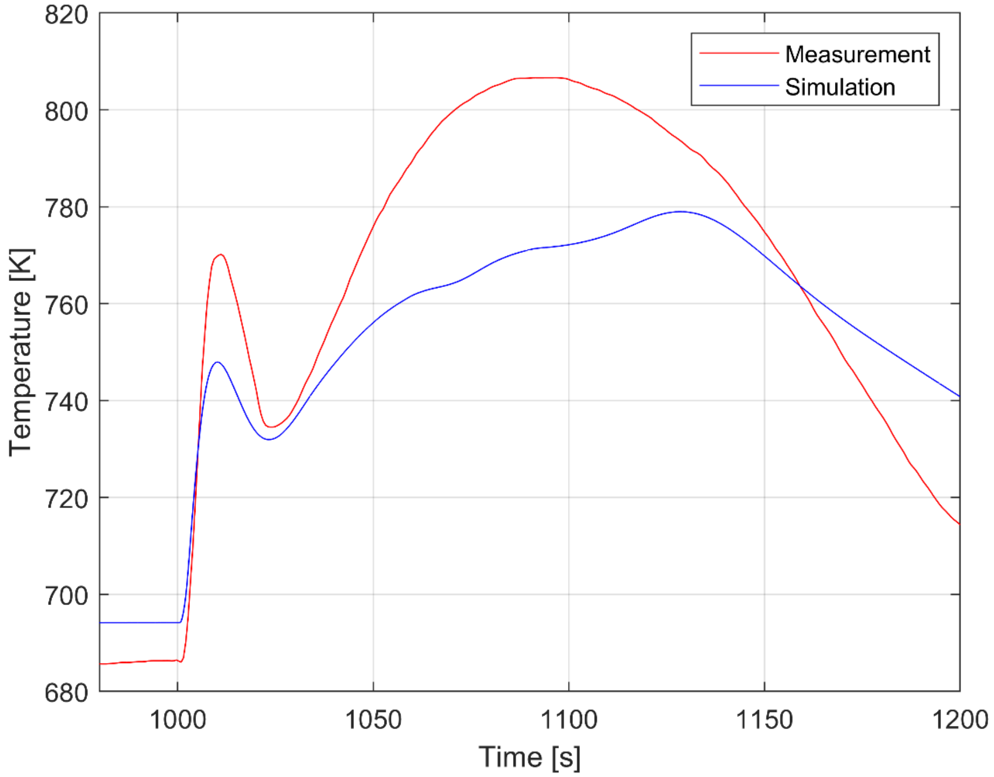



Comparison of the core **reactivity** evolution to the experimental data



# Results: Coupled Simulation and Comparison to Experimental Data

Comparison of the **coolant** outlet **temperature** evolution to the experimental data

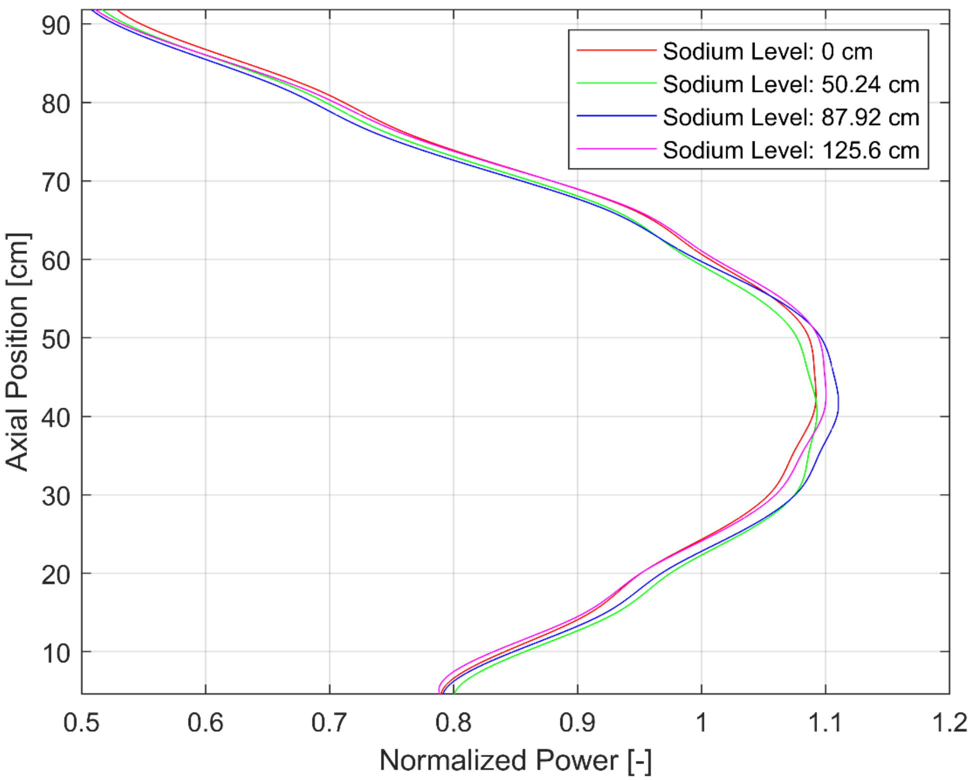



- According to [2], **discrepancy** can be explained
  - **Magnitude**: Simulated heat transfer by interassembly flow and the radial core heat transfer might be overestimated
  - **Time shift**: Heat transfer coefficient between primary and secondary loop and a possible error in thermal inertia of intermediate heat exchangers
  - **Reactivity**: simply ‘follows’ the state variables. In addition, absence of the neutronic model of Core Restraint System


[2] Wang, S., Mikityuk, K., Petrović, Đ., Zhang, D., Su, G., Qui, S., Tian, W., December 2021, ‘Validation of TRACE capability to simulate unprotected transients in Sodium Fast Reactor using FFTF LOFWST Test #13’, Annals of Nuclear Energy 164, 108600.

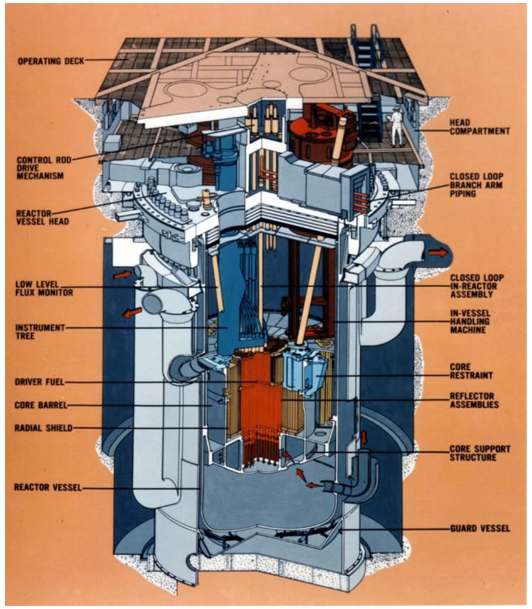
# Results: Axial Power Profiles

Evolution of the core **axial power profile** subsequent to the **activation** of **Gas Expansion Modules**




$$\frac{dn(t)}{dt} = \frac{\rho(t) - \beta}{\Lambda} n(t) + \sum_i \lambda_i C_i$$

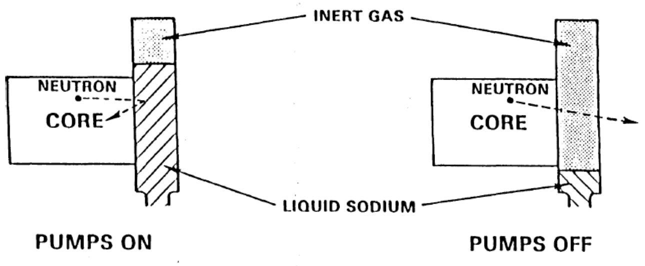
$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} n(t) - \lambda_i C_i(t)$$

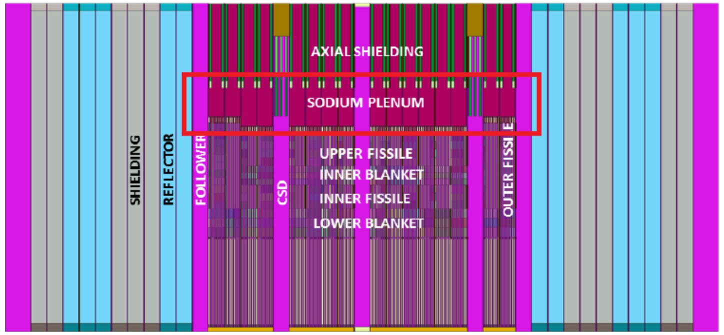


# Results: Representativeness of Gas Expansion Modules

- Representativeness of Gas Expansion Modules in modeling the **purposeful voiding** of fast reactor cores

| Parameter | Gas Expansion Modules | Inner Sodium Plenum |
|---|---|---|
| **Leakage** | Radial | Axial |
| **Total Reactivity Worth** | 475 ± 7 pcm | 1386 ± 21 pcm |
| **Outward Facing Surface** | 1.814 $m^2$ | 5.920 $m^2$ |
| **Surface Reactivity Worth** | 262 ± 4 $pcm/m^2$ | 234 ± 4 $pcm/m^2$ |

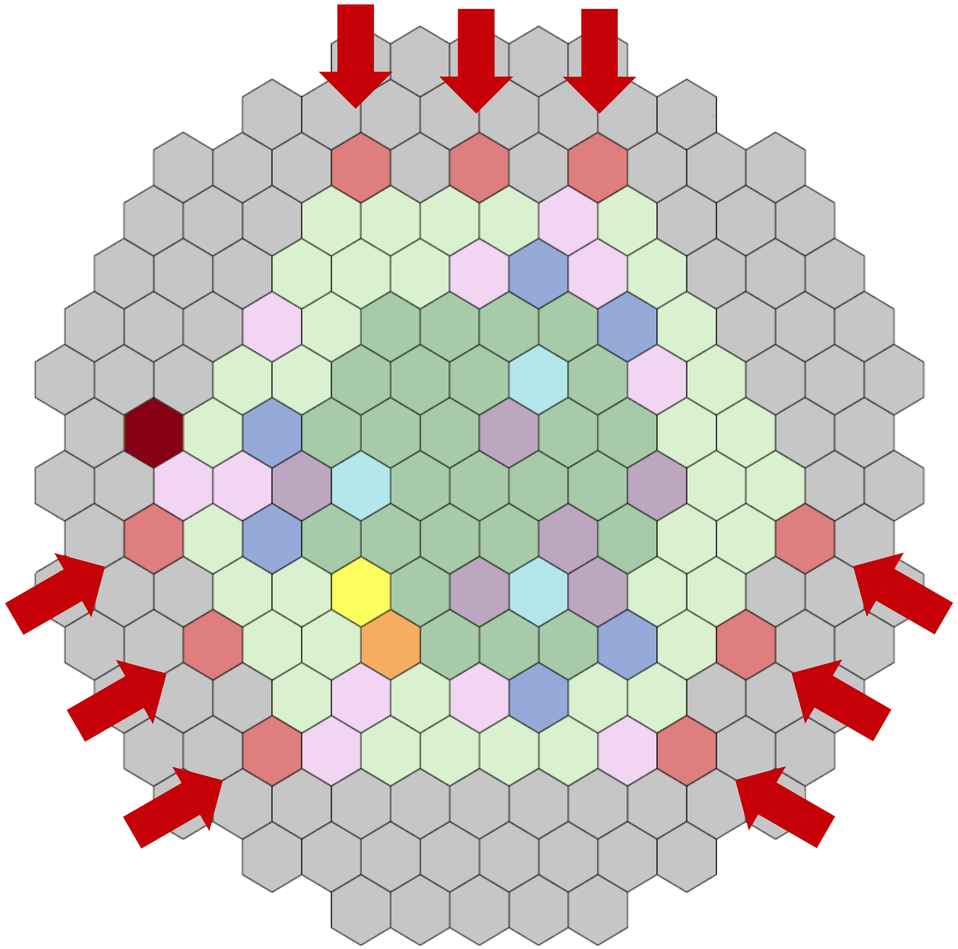



# Results: Importance of Gas Expansion Modules

**Coolant boiling** would occur 67 s
after the beginning of the LOFWOS Test #13

Gas Expansion Modules play
a crucial role in preserving the
**intrinsic** and **passive safety**
of the FFTF's core during
the cycle 8C



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# Summary and Conclusions

## Summary

- Static neutronics / Serpent 2 Monte Carlo simulation
- Neutron spatial kinetics / PARCS simulation
- Thermal-hydraulics / TRACE simulation
- Coupling

## Conclusions

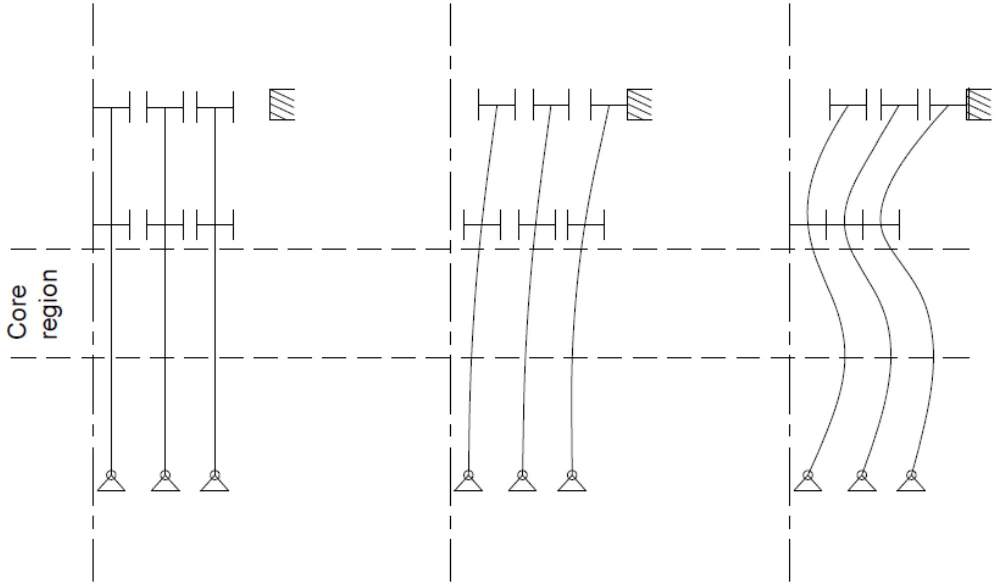
- Reactor core of an **SFR** features potential to achieve the **inherent** and **passive safety**
- Lessons learned during the operation of FFTF prove the capability of **Gas Expansion Modules** to **mitigate** the consequences of **Unprotected Loss of Flow** accident
- **Validation** of the **FAST code system** and the proof of its performance in modeling the transients relevant for safety assessment of the SFR design and fast reactor cores in general

# Future Work

- Correct model of **heat transfer phenomena** in TRACE
- Explicit model of **Gas Expansion Modules** in TRACE

- Model of **thermal-mechanics** and **fuel performance** aspects of the LOFWOS Test #13
  - Coupling to the fuel behavior code **FRED**

- Correction of the radial (and axial) expansion reactivity feedback effect in order to account for the presence of the **Core Restraint System**



# Wir schaffen Wissen - heute für morgen