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

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• 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, $\text{UO}_2 - \text{PuO}_2$

- 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)
• 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
• Introduction

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• 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
• 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

Benchmark Specifications: Gas Expansion Modules

<table>
<thead>
<tr>
<th>Pressure</th>
<th>Sodium Level</th>
<th>Radial Leakage</th>
<th>Reactivity</th>
</tr>
</thead>
</table>

![Diagram showing the flow of gas and sodium in a core with pumps on and off.]
Benchmark Specifications: Experimental Data

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

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

Calculated evolution of power-to-flow ratio

Evolution of the coolant outlet temperature as measured [1]

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

[Graph showing the evolution of reactivity over time, with a sharp decrease followed by a small increase]

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

[Graph showing the breakdown of reactivity into various components over time, with different colors representing different factors]

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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
Outline

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• Summary and Conclusions
• 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
• **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
• **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>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Effective Multiplication Factor</td>
<td>1.00574 ± 0.00003</td>
</tr>
<tr>
<td>Reactivity</td>
<td>571 ± 3 pcm</td>
</tr>
<tr>
<td>Delay Neutron Fraction</td>
<td>(3.209 ± 0.001) ∙10^{-3}</td>
</tr>
<tr>
<td>Prompt Neutron Lifetime</td>
<td>(5.524 ± 0.001) ∙10^{-7} s</td>
</tr>
<tr>
<td><strong>Reactivity Feedback Coefficients</strong></td>
<td></td>
</tr>
<tr>
<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><strong>Reactivity Worths</strong></td>
<td></td>
</tr>
<tr>
<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><strong>Incremental Reactivity Worths</strong></td>
<td></td>
</tr>
<tr>
<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>
</tbody>
</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
Comparison of the **coolant outlet temperature** evolution to the experimental data

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

Results: Axial Power Profiles

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

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

\[
\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} n(t) - \lambda_i C_i(t)
\]
- Representativeness of Gas Expansion Modules in modeling the **purposeful voiding** of fast reactor cores

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Gas Expansion Modules</th>
<th>Inner Sodium Plenum</th>
</tr>
</thead>
<tbody>
<tr>
<td>Leakage</td>
<td>Radial</td>
<td>Axial</td>
</tr>
<tr>
<td>Total Reactivity Worth</td>
<td>475 ± 7 pcm</td>
<td>1386 ± 21 pcm</td>
</tr>
<tr>
<td>Outward Facing Surface</td>
<td>1.814 m²</td>
<td>5.920 m²</td>
</tr>
<tr>
<td>Surface Reactivity Worth</td>
<td>262 ± 4 pcm/m²</td>
<td>234 ± 4 pcm/m²</td>
</tr>
</tbody>
</table>
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

• 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