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

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Outline

• Background and Objective
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• Sensitivity and Uncertainty Analysis
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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
- Fuelled with low enriched uranium (LEU) – U₃Si₂-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 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 $\tau$ is considered as the time constant of the flow rate decay. In this study, the time constant $\tau$ 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

<table>
<thead>
<tr>
<th>Properties</th>
<th>R5-3D</th>
<th>PARET</th>
<th>Deviation</th>
</tr>
</thead>
<tbody>
<tr>
<td>1st PCT${}^1$ [°C]</td>
<td>100.25</td>
<td>104.57</td>
<td>4.13%</td>
</tr>
<tr>
<td>1st PCT time [s]</td>
<td>0.50</td>
<td>0.40</td>
<td>25.00%</td>
</tr>
<tr>
<td>1st PCoT${}^2$ [°C]</td>
<td>59.47</td>
<td>59.72</td>
<td>0.42%</td>
</tr>
<tr>
<td>1st PCoT time [s]</td>
<td>0.50</td>
<td>0.40</td>
<td>25.00%</td>
</tr>
<tr>
<td>2nd PCT [°C]</td>
<td>123.81</td>
<td>128.67</td>
<td>3.78%</td>
</tr>
<tr>
<td>2nd PCT time [s]</td>
<td>7.5</td>
<td>8.00</td>
<td>6.25%</td>
</tr>
<tr>
<td>2nd PCoT [°C]</td>
<td>108.77</td>
<td>106.76</td>
<td>1.88%</td>
</tr>
<tr>
<td>2nd PCoT time [s]</td>
<td>8.00</td>
<td>8.00</td>
<td>0.00%</td>
</tr>
</tbody>
</table>

${}^1$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:**

<table>
<thead>
<tr>
<th>Uncertain parameter</th>
<th>Nominal value</th>
<th>Uncertainty range</th>
<th>Distribution</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inlet coolant Temp. [°C]</td>
<td>37</td>
<td>±10%</td>
<td>Normal</td>
</tr>
<tr>
<td>Inlet coolant mass flow rate [kg/s]</td>
<td>516.83</td>
<td>±10%</td>
<td>Normal</td>
</tr>
<tr>
<td>Reactor core power [MW]</td>
<td>20</td>
<td>±10%</td>
<td>Normal</td>
</tr>
</tbody>
</table>
Sensitivity Analysis Results and Discussion

- Relative Sensitivities of Input Parameters at steady state

\[ \alpha = \frac{x_0}{R_0} \frac{\partial R}{\partial x} \approx \frac{x_0}{R_0} \frac{R(x+h) - R(x-h)}{2h} \]

Fig. 9: Sensitivity coefficients of PCT (left) and PCoT (right)
Uncertainty Analysis Results at Steady State

<table>
<thead>
<tr>
<th>PCoT [ °C]</th>
<th>PCT [°C]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mean</td>
<td>54.25</td>
</tr>
<tr>
<td>Standard Dev.</td>
<td>14.88</td>
</tr>
<tr>
<td>95% Lower C.L.</td>
<td>53.32</td>
</tr>
<tr>
<td>95% Upper C.L.</td>
<td>55.18</td>
</tr>
<tr>
<td>Maximum</td>
<td>97.74</td>
</tr>
</tbody>
</table>

Peak Temperature Distribution Statistics

(A) Peak Coolant Temperature [ °C]

(B) Peak Cladding Temperature [ °C]
Uncertainty Analysis Results for PLOFA

(A) Cladding Temperature [°C]

(B) Coolant Temperature [°C]
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?