OECD/NEA Multi-Physics Benchmarks for Traditional and Novel Coupled Codes

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Introduction

- The NPP predictive modelling capabilities have evolved from the so-called traditional coupled code calculations to first-principle high-fidelity multi-physics simulations.
- The multi-physics interactions in a NPP are manifested in both global-length-scale behavior and lower-length-scale behavior.
- Especially important are the multi-physics interactions in reactor core.
- OECD/NEA has developed and is further developing in international cooperation appropriate benchmarks for multi-physics simulation capabilities.
Multi-physics simulation tools can be classified in two groups (categories)

- “Traditional” (T) tools, which include mostly neutronics/thermal-hydraulics coupling into reactor core on assembly/channel basis. The coupling of reactor core to the system and coupling of the system to containment also belongs to traditional (T) multi-physics simulations;

- “Novel” (N) or state-of-the-art tools, which include high-fidelity coupling on pin/sub-pin (pin-resolved)/sub-channel level of several physics phenomena in reactor core such as neutronics (reactor physics), thermal-hydraulics, fuel performance, structural mechanics, chemistry, etc.
Outline

- Introduction
- OECD/NEA Benchmarks for Validation of Traditional Coupled Codes
- OECD/NEA Benchmarks for Validation of Novel Multi-Physics Tools
- Conclusions
OECD/NEA Reactor Stability and LWR Transient Benchmarks for Traditional Multi-Physics Tools

- **BFBT**: NUPEC BWR Full-size Fine-mesh Bundle Tests Benchmark
- **PSBT**: NUPEC PWR Sub-Channel Bundle Tests Benchmark
- **V1000CT**: VVER-1000 Coolant Transient Benchmarks – Kozloduy-6
- **KALININ3**: VVER-100 Coolant Transient Benchmarks – Kalinin-3
- **BWRTT**: Boiling Water Reactor Turbine Trip - Peach Bottom-2
- **BWRSB**: Stability Benchmark from BWR FORSMARK 1 and 2, time series / frequency analysis
- **OSKARSHAMN-2**: BWR Stability - feed-water transient benchmark
- **LWR UAM**: multi-physics multi-scale propagation of uncertainties

Critical Issues in Nuclear Reactor Technology (**CRISSUE-S**) with references to experimental and other databases

**Code to code comparison – Verification activities**

- Rod Ejection (PWR)
- Uncontrolled Withdrawal of Control Rods (PWR)
- MSLB - Main Steam-line Breaks (PWR) TMI (plant data)
- Cold water injection and core pressurisation (BWR)
- **PWR MOX/UO2**: Core Transient Benchmark
- **PBMR-400**: PBMR Coupled Neutronics/Thermal Hydraulics Transient Benchmark
- Benchmarks on Physics of Plutonium Recycling
  - Benchmarks on reactor based weapons-grade Pu disposition
Traditional coupling

Appropriate multi-physics core / system benchmarks have been developed in international co-operation led by NEA/OECD:

- OECD/NRC PWR Main Steam Line (MSLB) benchmark
- OECD/NRC BWR Turbine Trip (TT) benchmark
- OECD/DOE/CEA VVER-1000 Coolant Transient (V1000CT) benchmark
- OECD Kalinin-3 Coupled Neutronics Thermal-Hydraulic Transient benchmark
- OECD/NRC Oskarshamn 2 BWR Stability benchmark

These benchmarks provide a validation basis for the current generation of coupled best-estimate codes.

Temperature distribution at the end of the transient process in Kalinin-3 benchmark
These are complex benchmarks, which use different Phases, Exercises and Scenarios to validate different models and the coupling between them.

The developed systematic validation approach includes comparisons on different modelling levels – point kinetics and 3D kinetics; neutronics with and without thermal-hydraulic feedback; and core boundary conditions models and core-plant coupling.

The available measured data is utilized in combination with detail code-to-code comparisons.

The cross-section libraries are generated by the benchmark team, which removes the uncertainties introduced by using different cross-section generation and modelling procedures.

This approach allowed to develop a more in-depth knowledge of the capabilities of the current generation best estimate coupled thermal-hydraulic system codes.

Professional community of experts has been established, which advanced the state-of-the-art in the area of coupling research.
Simulated Main Steam Line Break (MSLB) for TMI-1:

- Break occurs in one steam line upstream of the cross-connect
- Control rod with highest worth is assumed stuck out

Event is featured with significant space-time effects in the core due to the asymmetric cooling;

Conservative assumptions utilized to maximize RCS cool-down;

Major concern: possible return-to-power and criticality.
OECD/NRC PWR MSLB Benchmark

- Study on the impact of different neutronics and thermal-hydraulics models as well as the coupling between them;
- Detail of spatial mesh overlays – important for local safety predictions;
- Modeling issues – mixing modeling, moderator/coolant density correlations, and spatial decay heat distribution.

Total power time evolution comparison
OECD/NRC PWR MSLB Benchmark

Two issues have impacted the final results of this benchmark:

- Choice of thermal-hydraulic model is very important for local parameters predictions during the transient (especially in the vicinity of the stuck rod);
- Different Decay Heat models have led to pronounced deviations in the transient snapshot axial power distributions after the scram.

Normalized power at the time of maximum return to power

Peak fuel temperature time evolution
Since the PWR MSLB benchmark indicated that further development of the mixing computation models in the multi-physics coupled codes is necessary, a coolant mixing experiment, Main Coolant Pump start-up and MSLB transients are selected for simulation in the OECD/DOE/CEA V1000CT benchmark.

Temperature profiles at core exit and side wall, K
OECD/NRC BWR TT Benchmark

- The benchmark was established to challenge the thermal-hydraulic/neutron kinetics codes against the Peach-Bottom-2 (PB2) Turbine Trip (TT) 2 test data;
- 5 scenarios: 1 base test/experiment case and 4 hypothetical (extreme) scenarios/cases.

OECD/NRC BWR TT benchmark

Key elements of Best Estimate Case and Extreme Scenarios
OECD/NRC BWR TT Benchmark

For initial steady state results the deviations in predicting core average axial void fraction distribution are also reflected in the comparison of the axial power profile predictions due to the void feedback mechanism.

Steady state core average axial void fraction (mean and std. deviation)

Steady state core average normalized axial power distribution (measured vs. mean and standard deviations)
OECD/NRC BWR TT Benchmark

For the best estimate transient scenario the modeling of the pressure wave propagation into the vessel (dome pressure time history) is critical for correct prediction of power time history.
Objectives:

- Validation of coupled codes on measured data and performing uncertainty analysis (studying the propagation of uncertainties at different stages of simulation) – interactions with UAM activities;
- Challenge code to the limits of their capability and test variability of the non-linear modeling between codes (Extreme scenarios).

Transient event:

- Work at the switchyard lead to Load Rejection which was interpreted differently by the turbine (not ON) and the reactor (OFF).

Data source:

- Oskarshamn-2 Nuclear Power Plant (Sweden);
- February 25, 1999 feed-water transient + 5 linear stability measurements before & after the event.
Kalinin-3 (K3) Coupled Code Benchmark

Objectives:

- Validation of coupled codes on measured data and performing uncertainty analysis (studying the propagation of uncertainties at different stages of simulation).

Transient event:

- Switch-off of one Main Circulation Pump (MCP) at nominal power, while the other three pumps remain in operation.

Data source:

- Russian VVER-1000, Kalinin Unit #3;
- High-quality measured data available (all main integral parameters in the primary and secondary loops, local coolant temperature, local flux, controller & regulator positions, measurement accuracy specified, etc.).
NEA/OECD Benchmarks for Novel Multi-Physics Tools

- These benchmarks provide a framework to address the current trends in the development of LWR multi-physics and multi-scale modeling and simulation.

- The benchmarks include the following common features:
  - Utilization of high-quality experimental data;
  - Refined local scale modeling in addition to global predictions;
  - More detailed comparisons and analysis;
  - Including uncertainty and sensitivity analysis of modeling predictions.

- These benchmarks include:
  - CASL Watts Bar Unit 1 Cycle 1 benchmark;
  - Rostov-2 VVER-1000 benchmark;
  - C5G7-TD benchmark.
OECD/NEA Multi-Physics Benchmark Based on the CASL Watts Bar Unit 1 Cycle 1 Benchmarks

• CASL has developed a set of progressive benchmark/test problems that have been used to guide the development and validation of the VERA core simulator

• Within the Core Physics Progression Problems the HZP Physics test comparisons are referred to in the progression problems as Problem 5, and the Cycle 1 depletion is Problem 9.

• These two problems have been evaluated for the potential to become a OECD/NEA EGMPEBV multi-physics benchmark

VERA is a Core Simulator capable of analyzing physical reactor and modeling challenge problems
The problems are based on Watts Bar 1 (WB1) geometry and non-proprietary information. The actual geometry is proprietary.

- Each of the problems is based on actual fuel and core geometries used in the Watts Bar Nuclear 1 (WBN1) initial core loading.
- The fuel is a Westinghouse 17x17 design utilizing discrete Pyrex burnable poisons and hybrid Al/C/B4C rod cluster control assemblies (RCCAs).
- The general dimensions and material content of this fuel which are applicable to each progression problem are given.
- The specifications are obtained from publicly available sources for WBN1 or similar power plant designs.
- All input is provided at cold conditions.
Reactor Core Loading Configuration

Core Fuel and Poison Loading Pattern (Quarter Symmetry).
Core RCCA Bank Positions (in quarter symmetry).

<table>
<thead>
<tr>
<th>Core Fuel and Poison Loading Pattern (Quarter Symmetry)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core RCCA Bank Positions (in quarter symmetry)</td>
</tr>
</tbody>
</table>

- **Core Fuel and Poison Loading Pattern (Quarter Symmetry):**
  - Enrichment: Number of Pyrex Rods
  - A: 2.1
  - B: 2.6
  - C: 2.1
  - D: 2.6
  - E: 2.1
  - F: 2.6
  - G: 2.1
  - H: 2.6
  - I: 2.1
  - J: 2.6
  - K: 2.1
  - L: 2.6
  - M: 2.1
  - N: 2.6
  - O: 2.1
  - P: 2.6
  - Q: 2.1

- **Core RCCA Bank Positions (in quarter symmetry):**
  - H: D
  - G: A
  - F: D
  - E: C
  - D: A
  - C: D
  - B: C
  - A: B

The Consortium for Advanced Simulation of LWRs
A DOE Energy Innovation Hub
Ten VERA Core Physics Benchmark Progression Problems

• #1 2D HZP BOC Pin Cell
• #2 2D HZP BOC Lattice
• #3 3D HZP BOC Assembly
• #4 3D HZP BOC 3x3 Assembly CRD Worth
• #5 Physical Reactor Zero Power Physics Tests (ZPPT)
• #6 3D HFP BOC Assembly
• #7 3D HFP BOC Physical Reactor w/ Xenon
• #8 Physical Reactor Startup Flux Maps
• #9 Physical Reactor Depletion
• #10 Physical Reactor Refueling
Problem #5: Physical Reactor ZPPT

Description
Prediction of startup Zero Power Physics Tests (ZPPT) results for Cycle 1 of a Westinghouse 4-loop 17x17 physical reactor. The tests include measurement of critical boron concentration, isothermal temperature coefficient, control bank worths, and soluble boron worth. The tests are performed at zero power no xenon conditions without T/H or fuel temperature feedback.

Required Capabilities
• Definition of core baffle/shroud for radial reflector
• Full or quarter core geometry with rotational symmetry about core axes (i.e. cut assemblies on axes and at the core center)
• Definition of control rod banks and ability to position based on notch position
• Ability to simply specify multiple assembly types, with multiple lumped burnable poison (LBP) arrangements (WABA’s or PYREX), in a specified core layout
• Mesh generation for radial reflector (may affect in-core meshing)
• Automatic domain and energy decomposition for efficiency in massive parallelization
• Output of normalized 2D and 3D assembly power distributions and core average axial power distribution

Purpose
• Benchmark VERA to measured reactor data
• Address radial core boundaries (baffle/shroud)
• Address partial assemblies on core axes
• Demonstrate problem size and required resources
• Code comparisons to measurements (validation)
• Code-to-Code comparisons for more detailed distributions with MCNP and KENO
Problem #5: Physical Reactor Zero Power Physics Tests

To predict the eigenvalue and core reactivity coefficients without thermal-hydraulic feedback or depletion. Provides the opportunity to compare to measured startup data from WBN1.

<table>
<thead>
<tr>
<th>Case</th>
<th>Boron (ppm)</th>
<th>Temp (K)</th>
<th>Bank Position (steps withdrawn)*</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1285</td>
<td>565</td>
<td>- - - 167 - - - -</td>
<td>Initial</td>
</tr>
<tr>
<td>2</td>
<td>1291</td>
<td>-</td>
<td>- - - - - - - -</td>
<td>ARO</td>
</tr>
<tr>
<td>3</td>
<td>1170</td>
<td>↓</td>
<td>0 - - 97 - - - -</td>
<td>Bank A</td>
</tr>
<tr>
<td>4</td>
<td>↓ ↓</td>
<td>- 0</td>
<td>113 - - - - -</td>
<td>Bank B</td>
</tr>
<tr>
<td>5</td>
<td>↓ ↓</td>
<td>- - 0</td>
<td>119 - - - - -</td>
<td>Bank C</td>
</tr>
<tr>
<td>6</td>
<td>↓ ↓</td>
<td>- - - 18</td>
<td>- - - - - -</td>
<td>Bank D</td>
</tr>
<tr>
<td>7</td>
<td>↓ ↓</td>
<td>- - 69 0</td>
<td>- - - - - -</td>
<td>Bank SA</td>
</tr>
<tr>
<td>8</td>
<td>↓ ↓</td>
<td>- - 134 0</td>
<td>- - - - - -</td>
<td>Bank SB</td>
</tr>
<tr>
<td>9</td>
<td>↓ ↓</td>
<td>- - 71 0</td>
<td>- - - - - -</td>
<td>Bank SC</td>
</tr>
<tr>
<td>10</td>
<td>↓ ↓</td>
<td>- - 71 0</td>
<td>- - - - - -</td>
<td>Bank SD</td>
</tr>
<tr>
<td>11</td>
<td>↓ ↓</td>
<td>- - - - - - - -</td>
<td>ARO</td>
<td></td>
</tr>
<tr>
<td>12</td>
<td>↓ ↓</td>
<td>0 - - - - - - -</td>
<td>Bank A</td>
<td></td>
</tr>
<tr>
<td>13</td>
<td>↓ ↓</td>
<td>- 0 - - - - - -</td>
<td>Bank B</td>
<td></td>
</tr>
<tr>
<td>14</td>
<td>↓ ↓</td>
<td>- - 0 - - - - -</td>
<td>Bank C</td>
<td></td>
</tr>
<tr>
<td>15</td>
<td>↓ ↓</td>
<td>- - 0 - - - - -</td>
<td>Bank D</td>
<td></td>
</tr>
<tr>
<td>16</td>
<td>↓ ↓</td>
<td>- - 0 - - - - -</td>
<td>Bank SA</td>
<td></td>
</tr>
<tr>
<td>17</td>
<td>↓ ↓</td>
<td>- - - - - - - 0 - -</td>
<td>Bank SB</td>
<td></td>
</tr>
<tr>
<td>18</td>
<td>↓ ↓</td>
<td>- - - - - - - 0 - -</td>
<td>Bank SC</td>
<td></td>
</tr>
<tr>
<td>19</td>
<td>↓ ↓</td>
<td>- - - - - - - 0 - -</td>
<td>Bank SD</td>
<td></td>
</tr>
<tr>
<td>20</td>
<td>1291</td>
<td>560</td>
<td>- - - - - - - -</td>
<td>Low temp</td>
</tr>
<tr>
<td>21</td>
<td>↓ ↓</td>
<td>570</td>
<td>- - - - - - - -</td>
<td>High temp</td>
</tr>
<tr>
<td>22</td>
<td>1230</td>
<td>565</td>
<td>- - - - - - - -</td>
<td>D @ 0%</td>
</tr>
<tr>
<td>23</td>
<td>↓ ↓</td>
<td>- - - - - 23 - - - -</td>
<td>D @ 10%</td>
<td></td>
</tr>
<tr>
<td>24</td>
<td>↓ ↓</td>
<td>- - - - - 46 - - - -</td>
<td>D @ 20%</td>
<td></td>
</tr>
<tr>
<td>25</td>
<td>↓ ↓</td>
<td>- - - - - 69 - - - -</td>
<td>D @ 30%</td>
<td></td>
</tr>
<tr>
<td>26</td>
<td>↓ ↓</td>
<td>- - - - - 92 - - - -</td>
<td>D @ 40%</td>
<td></td>
</tr>
<tr>
<td>27</td>
<td>↓ ↓</td>
<td>- - - - - 115 - - - -</td>
<td>D @ 50%</td>
<td></td>
</tr>
<tr>
<td>28</td>
<td>↓ ↓</td>
<td>- - - - - 138 - - - -</td>
<td>D @ 60%</td>
<td></td>
</tr>
<tr>
<td>29</td>
<td>↓ ↓</td>
<td>- - - - - 161 - - - -</td>
<td>D @ 70%</td>
<td></td>
</tr>
<tr>
<td>30</td>
<td>↓ ↓</td>
<td>- - - - - 184 - - - -</td>
<td>D @ 80%</td>
<td></td>
</tr>
<tr>
<td>31</td>
<td>↓ ↓</td>
<td>- - - - - 207 - - - -</td>
<td>D @ 90%</td>
<td></td>
</tr>
<tr>
<td>32</td>
<td>↓ ↓</td>
<td>- - - - - - - - - -</td>
<td>D @ 100%</td>
<td></td>
</tr>
</tbody>
</table>

Problem 5 Cases for WBN1 ZPPT
Problem #5: Physical Reactor Zero Power Physics Tests

<table>
<thead>
<tr>
<th>Test Result</th>
<th>Measured</th>
<th>CE KENO-VI</th>
<th>Difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial Criticality†</td>
<td>1.00000†</td>
<td>0.999899 ± 0.000010</td>
<td>-10 ± 1 pcm</td>
</tr>
<tr>
<td>Bank A Worth (pcm)</td>
<td>843</td>
<td>898 ± 2</td>
<td>6.4% ± 0.2%</td>
</tr>
<tr>
<td>Bank B Worth</td>
<td>879</td>
<td>875 ± 2</td>
<td>-0.5% ± 0.2%</td>
</tr>
<tr>
<td>Bank C Worth</td>
<td>951</td>
<td>984 ± 2</td>
<td>3.5% ± 0.2%</td>
</tr>
<tr>
<td>Bank D Worth</td>
<td>1342</td>
<td>1386 ± 2</td>
<td>3.3% ± 0.1%</td>
</tr>
<tr>
<td>Bank SA Worth</td>
<td>435</td>
<td>447 ± 2</td>
<td>2.6% ± 0.4%</td>
</tr>
<tr>
<td>Bank SB Worth</td>
<td>1056</td>
<td>1066 ± 2</td>
<td>1.0% ± 0.2%</td>
</tr>
<tr>
<td>Bank SC Worth</td>
<td>480</td>
<td>499 ± 2</td>
<td>3.9% ± 0.4%</td>
</tr>
<tr>
<td>Bank SD Worth</td>
<td>480</td>
<td>499 ± 2</td>
<td>4.0% ± 0.4%</td>
</tr>
<tr>
<td>Total Bank Worths</td>
<td>6467</td>
<td>6654 ± 4</td>
<td>2.9% ± 0.1%</td>
</tr>
<tr>
<td>DBW (pcm/ppm)</td>
<td>-10.77</td>
<td>-10.21 ± 0.02</td>
<td>0.56</td>
</tr>
<tr>
<td>ITC (pcm/F)</td>
<td>-2.17</td>
<td>-3.18 ± 0.04</td>
<td>-1.01</td>
</tr>
</tbody>
</table>

![Figure P5-2: Problem 5 KENO-VI Geometry](image)

Figure P5-2: Problem 5 KENO-VI Geometry
# Problem #9: Physical Reactor Depletion

## Description
Full simulation of Cycle 1 of a Westinghouse 4-loop 17x17 physical reactor. Operating power history provided. Measurements include critical boron concentrations, axial offset, and axial flux map data.

## Purpose
- Benchmark VERA to measured reactor data
- Demonstrate fuel depletion and isotopic decay
- Address depletion time step sizes
- Gain experience in handling data from full cycle
- Code comparisons to measurements (V&V)
- Code-to-code comparisons for more detailed distributions

## Required Capabilities
- Input definition of core follow core power, flows, CRD positions, and exposures
- Perform accurate depletion (by power) at pin level for user input of up to monthly time-steps (sub-step methodology, time step averaging, etc.) with T/H and fuel temperature feedback
- Accommodate mid-cycle outages (if necessary) with isotopic decay
- Output of critical boron concentration and limiting powers ($F_{\Delta h}$ and $F_q$) vs. cycle exposure
- Output of assembly level 3D exposures
- Output of assembly average and core average exposures
- Output of peak assembly and pin exposures (and locations)
- Output of fuel pin average 3D exposures for selected assemblies
- Ability to save (and archive) all fuel rod isotopics and exposures for restart capability
Problem #9: Physical Reactor Depletion

Slow time-dependence (cycle depletion) of the reactor at operating conditions in pseudo-steady state - time scale to the length of a typical 18-month fuel cycle

<table>
<thead>
<tr>
<th>Input</th>
<th>Value</th>
<th>Section</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rated Power (100%)</td>
<td>3411 MW</td>
<td>3</td>
</tr>
<tr>
<td>Rated Coolant Mass Flow (100%)</td>
<td>131.7 Mlbs/hr</td>
<td>3</td>
</tr>
<tr>
<td>Reactor Pressure</td>
<td>2250 psia</td>
<td>3</td>
</tr>
<tr>
<td>Cycle Length</td>
<td>441.0 EFPDs</td>
<td>3</td>
</tr>
<tr>
<td>EOC Exposure</td>
<td>16.939 GWd/MT</td>
<td>3</td>
</tr>
<tr>
<td>RCCA Overlap (steps withdrawn)</td>
<td>128</td>
<td>3</td>
</tr>
</tbody>
</table>
### Problem #9: Physical Reactor Depletion

<table>
<thead>
<tr>
<th>Case</th>
<th>EFPD</th>
<th>Cycle Exposure (GWd/MT)</th>
<th>Power (%)</th>
<th>Inlet Temp. (F)</th>
<th>Bank D Position (steps)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.0</td>
<td>0.000</td>
<td>0.0</td>
<td>557.0</td>
<td>186</td>
</tr>
<tr>
<td>2*</td>
<td>9.0</td>
<td>0.346</td>
<td>65.7</td>
<td>557.6</td>
<td>192</td>
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<tr>
<td>3*</td>
<td>32.0</td>
<td>1.229</td>
<td>99.7</td>
<td>558.1</td>
<td>219</td>
</tr>
<tr>
<td>4</td>
<td>50.0</td>
<td>1.920</td>
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<td>558.2</td>
<td>218</td>
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<td>5</td>
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<td>100.0</td>
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<td>6</td>
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<td>99.7</td>
<td>558.7</td>
<td>215</td>
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<tr>
<td>7</td>
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<td>99.7</td>
<td>558.6</td>
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<td>99.8</td>
<td>558.8</td>
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<td>9</td>
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<td>4.644</td>
<td>99.8</td>
<td>558.4</td>
<td>220</td>
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<td>10</td>
<td>133.8</td>
<td>5.139</td>
<td>99.5</td>
<td>557.9</td>
<td>219</td>
</tr>
<tr>
<td>11</td>
<td>148.4</td>
<td>5.700</td>
<td>98.0</td>
<td>558.0</td>
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</tr>
<tr>
<td>12</td>
<td>163.3</td>
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<td>95.1</td>
<td>557.9</td>
<td>216</td>
</tr>
<tr>
<td>13</td>
<td>182.2</td>
<td>6.998</td>
<td>94.8</td>
<td>557.9</td>
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<td>10</td>
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</table>
Problem #9: Physical Reactor Depletion

Measured results (critical boron concentration as function of burnup, and 3-D flux maps at 13 points in cycle depletion)
Summary

The TVA Watts Bar Cycle 1 data will be used for developing a OECD/NEA Multi-Physics Benchmark for novel high-fidelity multi-physics tools

• The benchmark will be in 3 exercises:

  **Exercise 1** – Validation of stand-alone 3-D neutronics model at HZP conditions

  **Exercise 2** – Validation of multi-physics steady state model for HFP conditions (neutronics + thermal-hydraulics + fuel)

  **Exercise 3** – Validation of multi-physics cycle depletion model for Watts Bar Cycle 1 (neutronics + thermal-hydraulics + fuel with depletion)

• Some examples of high-fidelity calculated results are shown.

---

*The Consortium for Advanced Simulation of LWRs (CASL)*

A DOE Energy Innovation Hub
OECD/NEA Rostov-2 Multi-Physics Benchmark

- A large number of tests with a multitude of well-documented neutron-physics and thermal-hydraulics measurements data have been performed at Rostov - Unit 2 NPP.

- Integral (plant) data and local measured data (core) were collected during the test, which will be used for the validation of both traditional and novel multi-physics codes.

- The measurement and recording of parameters was performed by the standard means available at NPP and by a special system of experimental control (SEC).

- As a result, the specification of the “Phase 1” for the first OECD/NEA EGMPEBV multi-physics benchmark is being developed, based on Rostov-2 NPP experimental, operation and measured data.

- The reactor type is VVER-1000 with new modern fuel assemblies TBC-2M which enable an 18-month fuel cycle length.
Introduction

- The benchmark team selected a test (transient), which will allow validation of novel multi-physics codes developed last years in the frame of different national and international projects.

- The difference in comparison with all previous NEA/OECD Benchmarks for coupled code validation is the implementation of high fidelity multi-physics simulation codes that could predict pin-by-pin power distributions and flow mixing in the primary loop, in the reactor pressure vessel including its active core part.

- The reference benchmark problem chosen for simulation and comparison with the measured data is based on a test characterized with the following scenario –

  “Reactivity compensation with diluted boron by stepwise insertion of control rod cluster into the VVER core”
Coolant temperature in hot leg #2 and movement of CR group #10

1. Thermocouple #1 readings
2. Thermocouple #2 readings
3. Thermoresistor readings
4. Insertion depth of the CR#10
Available measured data to compare with

1. Integral values
   - Integral power (several values measured with different methods)
   - Hot and cold leg temperatures (at several azimuthal positions at several locations)
   - Hot and cold leg mass flows
   - Pressurizer data (pressure, heaters control, surge line control)
   - Steam generators’ data (water levels, feed water temperature and mass flows)
   - Make up data (mass flow)
   - $C_{\text{boron}}$ at the measuring point in the downcomer

2. Local values
   - SPND readings at 7 levels in $nnn$ assemblies
   - $T_{\text{coolant}}$ at core exit (at $nn$ assemblies’ heads)
Benchmark Specifications

Benchmark on reactivity compensation with diluted boron by stepwise insertion of control rod cluster into the VVER-1000 core

Specifications and Support Data
Version 1.1

Maria Avramova¹, Kostadin Ivanov², Kirill Velkov³, Ihor Pasicnyk⁴, Sergii Nikonov⁵, Margarita Denisova⁶, Aleksei Denisenko⁷, Pavel Gordienko⁸, Boris Shnurskij⁹

¹NCSU - USA, ²GRS - Germany, ³GRS scientific guest, ⁴VNIIAES – Russia, ⁵KI - Russia

February 2017
OECD Nuclear Energy Agency
### Benchmark Specifications

Table 2.7 – Fundamental error of measurement/calculations of the main technological parameters [8]

<table>
<thead>
<tr>
<th>Name of the measured or calculated parameter</th>
<th>Assumed fundamental error (confidence factor 0.95)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 Temperature: TC, °C, not more than RT, °C, not more than</td>
<td>±1.0  ±0.5</td>
</tr>
<tr>
<td>2 DCD currents, %, not more than</td>
<td>±0.05</td>
</tr>
<tr>
<td>3 Peaking factor on the core volume (FA level), %, not more than</td>
<td>±5.0</td>
</tr>
<tr>
<td>4 Weighted-average reactor thermal power, %, not more than</td>
<td>±2.0</td>
</tr>
</tbody>
</table>

**Note**
1. Confirmation of the metrological characteristics of the parameter 1 must be carried out in an isothermal condition of the reactor.
2. Confirmation of the metrological characteristics of the parameters 3 and 4 must be carried out at the nominal power level of the reactor in the steady state.
3. The values of the assumed fundamental error are listed at the full functionality of the hardware.
4. Design error limits of the channels for the parameter 2 are listed at % from the top limit of the measurement.

- The measurement uncertainties are provided
- The initial and boundary conditions are characterized and associated uncertainty is quantified
Data-base of measured data and cross-section library

CRC #10 positions, N- SPND measurements, T – TC measurements
Benchmark Phase and Exercises

1. Phase 1 - Assembly wise analysis
   a. Exercise #1 – T-H plant simulation using power tables
   b. Exercise #2 – Coupled 3-D neutronics/core T-H response evaluation
      - Exercise #2a - HZP state
      - Exercise #2b – 75% HP state
   c. Exercise #3 - Best-estimate coupled code plant transient modeling

2. Phase 2 - Full core pin-by-pin analysis
   a. Exercise #1 – Boundary condition steady-state problem
   b. Exercise #2 – Boundary condition transient calculations
   c. Exercise #3 – Best-estimate coupled pin-by-pin transient calculations

REMARK:
Instead of full core: at least hot channel pin-by-pin or 7 assemblies around CRC 10 (#112) pin-by-pin
Location of the selected 7 assemblies where p-b-p calculations are required
OECD/NEA deterministic time-dependent neutron transport benchmark (C5G7-TD)
Motivation

- Currently available benchmark problems
  - Simplified diffusion benchmarks: computational domain consists of several homogeneous regions characterized by few-group diffusion macroscopic cross sections
  - Heterogeneous benchmarks but material specification was provided with respect to the isotopic concentrations and feedback description was sometimes defined – additional uncertainties

- A new benchmark problem is required to satisfy the demand for verifying numerical methods aiming at homogenization-free time-dependent transport calculations
- OECD/NEA benchmark for time-dependent neutron transport calculations without spatial homogenization: C5G7-TD
Overview

- Approved by OECD/NEA Nuclear Science Committee (NSC) Working Party on Scientific Issues in Reactor Systems (WPRS) in the meeting in February 2015
- First benchmark specification released in March 2016 and being continuously updated (currently in version 1.7)
- First workshop C5G7-TD-1 was held in PSI in May 2016
- Results being collected from participants
- Second workshop C5G7-TD-2 was held in Erlangen in May 2017

Benchmark team
- National Research Centre “Kurchatov Institute” (KI)
  - Victor F. Boyarinov and Peter A. Fomichenko
- North Carolina State University (NCSU)
  - Jason Hou and Kostadin Ivanov
- Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH
  - Alexander Aures, Winfried Zwermann and Kiril Velkov

For more information, please contact Jason Hou (jason.hou@ncsu.edu)
C5G7-TD benchmark: schedule

Phase I: Kinetics Phase
- Verification of methods/codes for heterogeneous time-dependent neutron transport calculations without feedback
- Currently in progress: collecting results and performing comparative analysis

Phase II: Dynamics Phase
- Verification of methods/codes for heterogeneous time-dependent neutron transport calculations with feedback
- Currently in progress: draft specification released

Phase III: High-fidelity Phase
- Uncertainty propagation in high-fidelity multi-physics calculations
- Future: starting from 2018
C5G7-TD benchmark: background

- Based on well-known steady-state Benchmark on Deterministic Transport Calculations Without Spatial Homogenization (C5G7 benchmark)
  - Originally a 2D benchmark
  - Extended version is a 3D model and it contains control rod characteristics
- Miniature LWR core with quarter-core symmetry and with given seven-group cross-sections
- Core consists of 8 MOX and 8 UO$_2$ assemblies surrounded by moderator
- No uncertainties in initial data
  - 7-group cross sections and kinetics data prepared by benchmark teams
  - All possible differences in results will be caused only by methodical errors and differences of applied codes
- Accurate model specification of the C5G7 benchmark is available
C5G7-TD geometry: 2-D configuration

Cross sections prepared for 2 zones
C5G7-TD geometry: 3-D configuration

Fission chambers and control rods present in axial reflector region should be modelled.
Transient problems

- 2D transients
  - Exercise 0 (TD0): control rod (CR)
  - Exercise 1 (TD1): control rod (CR)
  - Exercise 2 (TD2): CR
  - Exercise 3 (TD3): moderator density

- 3D transients
  - Exercise 4 (TD4): CR
  - Exercise 5 (TD5): moderator density

- Initial state: critical
  - Divide neutron production $N_u$ by initial core $k_{eff}$
2-D cases: TD0, TD1, TD2

- CR movement at various speed in different assemblies.
  - TD0: Step change in CR position
  - TD1/2: ramp change in CR position
  - Can be realized by material composition change
  - Transient simulation for 0-10 s

<table>
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<tr>
<th>Test cases</th>
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<th>TD1</th>
<th>TD2</th>
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<td>Bank 1</td>
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<td>Bank 4</td>
<td>Bank 4</td>
</tr>
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<tr>
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<td>Bank 1-4 simultaneously</td>
<td>Bank 1-4 simultaneously</td>
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</tbody>
</table>

TD0 expression

\[
\Sigma_x(t) = \Sigma_x^{GT}, t = 0, t \geq 2s
\]
\[
\Sigma_x(t) = \Sigma_x^{GT} + 0.1 (\Sigma_x^R - \Sigma_x^{GT}), 0 < t \leq 1s
\]
\[
\Sigma_x(t) = \Sigma_x^{GT} + 0.05 (\Sigma_x^R - \Sigma_x^{GT}), 1s \leq t < 2s
\]
2-D cases: TD3

- Moderator density changes uniformly across the core
  - Perturbation is approximated by linearly decreasing moderator cross section in all energy groups during density decrease
  - Then perturbation continues by linearly increasing these cross sections to initial values during density increase
  - Speed of change varies for test problems
  - 4 test problems: transient simulation for 0-10 s
3-D cases: TD4 (1/2)

- Control rod movement
  - Initial state: CR fully withdrawn
  - Insertion/withdrawal of (combination of) control rods in different fuel assemblies
  - Rod bank moves at constant speed, which allows it to be fully inserted into the assembly from the fully withdrawn position within 6 s
  - Hypothetic value proposed only for the purpose of reducing the computational effort in the transient calculation
3-D cases: TD4 (2/2)

- 5 test problems: transient simulation for 0-15 s
3-D cases TD5: moderator density transients

- Initial state: nominal mod density across core
- Change of mod density at various speed in different fuel assemblies
- No axial variation of moderator density within assembly
- Unrodded state during transient
- Water density in both radial and axial reflector maintained in its nominal value throughout the transient
### Submitted results for Phase I

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* Diffusion and diffusion w/ Improved Quasistatic Solution (IQS)
TD1-1: CR ramp change transient

Test cases

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<td>Bank 1, 3 and 4 simultaneously</td>
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<tr>
<td>5</td>
<td>Bank 1-4 simultaneously</td>
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- $\Delta \rho \sim -0.7$
- All participants reported on this case
- Prediction on $\beta_{\text{eff}}$ is self-consistent
TD1-1: CR ramp change transient (cont.)

Test cases TD1

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<td>Bank 1, 3 and 4 simultaneously</td>
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<tr>
<td>5</td>
<td>Bank 1-4 simultaneously</td>
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Max diff. ~ 500 pcm
TD1-1: CR ramp change transient (cont.)

- Good agreement found in assembly power comparison
TD4: CR ramp change transient

- TD4-2
- TD4-4
Phase II exercise are focused on TH feedback including fuel temperature and moderator density

Fast feedback on fuel temperature:

- Initial temperature distribution is uniform at $T_0$
- XS for fuel prepared for multiple temperature points
  - Small temperature interval (5 K): lookup table + interpolation
  - Large temperature interval: polynomial
- Initial criticality achieved by adjusting neutron production $\nu$
- Initial critical flux distribution should be normalized such that core is at zero power
- Adiabatic model for rate of change in local (rod) fuel temperature $T_k$
  - $\frac{\partial T_k(t)}{\partial t} = \alpha f_k(t) - \beta_k$
  - $f_k(t) = \sum G \sum_{f}^{k,g}(t) \phi^{k,g}(t)$: local fission rate
  - $\alpha = 3.83 \times 10^{-11}$ K $\cdot$ cm$^3$: conversion factor
  - $\beta_k = \alpha \cdot f_k(t_0)$: time independent parameter

Transient scenario
- CR extraction
- Type a: abrupt extraction
- Type b: linear extraction
Conclusions

- This presentation summaries two sets of NEA/OECD multi-physics benchmarks

For Traditional Multi-Physics Tools
  - OECD/NRC PWR Main Steam Line (MSLB) benchmark
  - OECD/NRC BWR Turbine Trip (TT) benchmark
  - OECD/DOE/CEA VVER-1000 Coolant Transient (V1000CT) benchmark
  - OECD Kalinin-3 Coupled Neutronics Thermal-Hydraulic Transient benchmark
  - OECD/NRC Oskarshamn 2 BWR Stability benchmark

For Novel Multi-Physics Tools
  - CASL Watts Bar Unit 1 Cycle 1 benchmark
  - Rostov-2 VVER-1000 benchmark
  - C5G7-TD benchmark