Safety Related Investigations of the VVER-1000 Reactor Type by the Coupled Code System TRACE/PARCS*

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Abstract
This study was performed at the Institute of Reactor Safety at the Forschungszentrum Karlsruhe. It is embedded in the ongoing investigations of the international code assessment and maintenance program (CAMP) for qualification and validation of system codes like TRACE(1) and PARCS(2). The chosen reactor type used to validate these two codes was the Russian designed VVER-1000 because the OECD/NEA VVER-1000 Coolant Transient Benchmark Phase 2(3) includes detailed information of the Bulgarian nuclear power plant (NPP) Kozloduy unit 6. The post-test investigations of a coolant mixing experiment have shown that the predicted parameters (coolant temperature, pressure drop, etc.) are in good agreement with the measured data. The coolant mixing pattern, especially in the downcomer, has been also reproduced quiet well by TRACE. The coupled code system TRACE/PARCS which was applied on a postulated main steam line break (MSLB) provided good results compared to reference values and the ones of other participants of the benchmark. The results show that the developed three-dimensional nodalization of the reactor pressure vessel (RPV) is appropriate to describe the coolant mixing phenomena in the downcomer and the lower plenum of a VVER-1000 reactor. This phenomenon is a key issue for investigations of MSLB transient where the thermal hydraulics and the core neutronics are strongly linked. The simulation of the RPV and core behavior for postulated transients using the validated 3D TRACE RPV model, taking into account boundary conditions at vessel in- and outlet, indicates that the results are physically sound and in good agreement to other participant’s results.

Key words: Best-Estimate, Coupled-Codes, PARCS, TRACE, VVER-1000

1. Introduction

The development of computer programs related to nuclear energy and safety is one of the main R & D works in nuclear engineering. The current development of nuclear safety related programs is focused on both the enhancements of existing codes (e.g. two-phase flow, multi-fluid models, 3D modeling, heat-transfer-correlations, friction-factor, etc.) and the validation of these codes by data obtained during experiments. One approach for improving results is to couple different codes. In this case the modeling of different physical processes is combined to describe the complete event (thermo-hydraulics + 3D neutronics).
The assessment of the coupled thermo-hydraulics code (TRACE) with the 3D neutronics code (PARCS) is the central point of this paper. The two codes TRACE and PARCS are the reference safety analytic tools of the U.S. Nuclear Regulatory Commission (U.S. NRC) and are still under development.

This paper is divided into 6 main parts. Chapter two gives a brief overview of the VVER-1000 Coolant Transient Benchmark followed by a short description of the used codes TRACE and PARCS given in chapter three. The fourth chapter is devoted to the development of a 3D-TRACE-Model of the RPV. Chapter five deals with the analysis of coolant mixing experiment carried out during the commissioning phase of Kozloduy NPP. The sixth part is dedicated to the investigation of a postulated MSLB by using the validated 3D Model. The last chapter gives a summary of the obtained results and an outlook.

2. VVER-1000 Coolant Transient Benchmark Phase 2

One of the current benchmarks for coupled code investigations is the OECD/NEA VVER-1000 Coolant Transient Benchmark. This benchmark investigates the Bulgarian nuclear power plant Kozloduy Unit 6 shown in Fig. 1.

![Primary side of the VVER-1000](image)

**Fig. 1** Primary side of the VVER-1000

The main technical characteristics of the VVER-1000 reactor are summarized in Table 1.

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>VVER-1000</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power, MW</td>
<td>3000</td>
</tr>
<tr>
<td>Electric power, MW</td>
<td>1000</td>
</tr>
<tr>
<td>Pressure above the core, MPa</td>
<td>15.65</td>
</tr>
<tr>
<td>Temperature in the Pressurizer, K</td>
<td>620.0 ± 1</td>
</tr>
<tr>
<td>Temperature at the core inlet, K</td>
<td>560.0 ± 2</td>
</tr>
<tr>
<td>Temperature at the core outlet, K</td>
<td>593 ± 3.5</td>
</tr>
<tr>
<td>Mass flow, kg/s</td>
<td>17610 ± 400</td>
</tr>
<tr>
<td>Characteristic</td>
<td>VVER-1000</td>
</tr>
<tr>
<td>-------------------------------------------</td>
<td>-----------</td>
</tr>
<tr>
<td>Pressure at SG inlet, MPa</td>
<td>15.64</td>
</tr>
<tr>
<td>Temperature at SG inlet, K</td>
<td>591 ± 1</td>
</tr>
<tr>
<td>Temperature at SG outlet, K</td>
<td>560 ± 2</td>
</tr>
<tr>
<td>Av. coolant temperature, K</td>
<td>576.15</td>
</tr>
<tr>
<td>RPV height (vessel + cover), m</td>
<td>12.433</td>
</tr>
<tr>
<td>Height of the active zone, m</td>
<td>3.55</td>
</tr>
<tr>
<td>Outside diameter, m</td>
<td>4.535</td>
</tr>
<tr>
<td>Inside diameter, m</td>
<td>4.136</td>
</tr>
<tr>
<td>Wall thickness (bottom), mm</td>
<td>237</td>
</tr>
<tr>
<td>Wall thickness (cylinder), mm</td>
<td>192</td>
</tr>
<tr>
<td>Coolant loop diameter, mm</td>
<td>850</td>
</tr>
</tbody>
</table>

The benchmark consists of two phases and each phase has several exercises. Phase 1 is already finished (5). Phase 2 is led by the French Commissariat à l’Energie Atomique and is three-folded: (1) Test of flow mixing models (CFD, coarse-mesh and mixing matrix) against measured data and in code-to-code comparison. (2) Test of the coupling of 3D neutronics and vessel thermal-hydraulics. (3) Evaluation of discrepancies between predictions of coupled codes in best-estimate transient simulations (3).

3. Brief Description of TRACE and PARCS

The challenges mentioned in chapter two are asking for tools which are able to handle multidimensional phenomena by using an appropriate 3D model of the RPV. TRACE and PARCS are especially suitable for the work on the benchmark because TRACE has 3D thermo-hydraulics capabilities and PARCS offers the opportunity to use hexagonal fuel assembly (FA) nodes which are typical for Russian designed reactors like the VVER-1000 and innovative reactor concepts.

The code TRACE (TRAC/RELAP Advanced Computational Engine) combines capabilities of the RAMONA, RELAP5, TRAC-B and TRAC-P codes and was developed by the Los Alamos National Laboratory (LANL), the Information Systems Laboratory (ISL), and the Penn State University (PSU) for use best-estimate analysis of light water reactors and Generation IV systems. To meet these challenges TRACE uses many new features like multidimensional flow modeling and 2D heat conduction. TRACE is able to use different coolant types like H₂O, D₂O, He, Na and PbBi as well. The partial differential equations that describe two-phase flow and heat transfer are solved in TRACE using finite difference numerical methods (1).

The neutron kinetics code PARCS (Purdue Advanced Reactor Core Simulator) is being developed by Purdue University. It solves the two group/multigroup 3D neutron diffusion equation for both rectangular and hexagonal nodes. The implemented hexagonal multigroup nodal diffusion kernel is based on the Triangular Polynomial Expansion Method (TPEN). PARCS has also the following features: eigenvalue calculation, transient calculation, xenon transient, decay heat and pin power calculation (3). Both codes are written in FORTRAN90 and have a modular structure.
4. Development of the 3D-TRACE-Model

The best way to develop a TRACE model of the RPV is to use the VESSEL element since this element allows a 3D description of the flow within the RPV. A detailed description of the 3D model is given in\(^7\). An example of the RPV nodalization shows Fig. 2. This nodalization of the RPV requires the definition of a certain number of 3D cells. The nodalization of the space directions is based upon the flow conditions in the RPV, because areas with a complex flow path require a finer subdivision. Each cell requires parameters to specify the thermo-hydraulic conditions (e.g. fraction of flow area, hydraulic diameter, etc.) in the cell. The nodalization of the RPV is a compromise between the CPU time and the resolution of the RPV so that the underlying physical phenomena are well described.

Fig. 2  Example for a 3D nodalization of a RPV

The following subchapters describe the nodalization in all three space directions.

4.1 Axial nodalization

The first step in the nodalization is subdividing of the height in an appropriate number of axial levels. There are different options to accomplish this. One possibility is to choose a constant level height (e.g. 1m or 20 cm). An alternative method is to divide the axial levels according to the underlying physical process and the constructive peculiarities. This means, for example, that the lower core support plate represents one level. Other levels are dedicated to the lower plenum support columns. The advantage of this kind of axial nodalization is the simplified calculation of the thermo-hydraulic parameters required in TRACE because the cell geometry of the RPV and its internals are constant. The lower plenum of the VVER-1000 differs strongly from lower plenums of western nuclear power plants because the whole lower plenum is loaded with support columns, each support column consists of a lower massive part and a perforated upper part, and the columns are detached between the perforated core barrel bottom and the lower core support plate. This requires special efforts for the modeling of the lower plenum, but simplifications are still needed in the VVER-1000 model. One simplification is the planate of the elliptic RPV bottom and the elliptic core barrel bottom respectively. These approaches lead to 30 axial layers as shown in Fig. 3.
The first layer represents the gap between the RPV bottom and the bottom of the core barrel. The next level is the core barrel bottom itself. Layers number 3 and 4 are dedicated to the support columns of the lower plenum. The next two levels represent the lower core support plate and the unheated part of the core. The core itself is divided into 10 layers followed by the layers of the upper unheated part of the core and two layers for the FA heads. The upper core support plate builds the next level. The next six levels represent the layers between the upper core support plate and the plate at the top of the control rod guide tubes. This plate is also represented as one level. The last three levels represent the head of the RPV.

4.2 Azimuthal nodalization

The division in azimuthal direction accounts for the arrangement of the main coolant loops with respect to the core. As seen in Fig. 1 the coolant inlet nozzle is arranged below the outlet nozzles and the loops are not symmetrically arranged. The embedded angle between the two loops is 55° on the right side and also for the left side in Fig. 1. Hence, the RPV is divided into 6 parts instead of 4 or 8 parts by symmetrical arrangement of the coolant nozzles, and the division of the core becomes easier because this leads to a symmetrical zoning of the core with its 163 hexagonal FA’s (see Fig. 4).

4.3 Radial Nodalization

The third direction to subdivide is the radial one. This follows the same principles like the axial direction. Here, the recommended principle is dividing the RPV into rings with constant geometrical properties. This leads to six rings. The outer ring is dedicated to the downcomer followed by the ring for the core barrel and the reflector area. The other three rings cover the active zone of the core (see Fig. 4).

Hence, the 3D model of the vessel consists of 1080 cells forming by 30 axial levels, six rings, and six azimuthal sectors.
5. Post-test calculation of a coolant mixing experiment

5.1 Coolant mixing experiment

This part of the paper deals with the validation of the developed 3D model. The chosen scenario serves for the checking of the correct nodalization. The experiment is a pure thermal-hydraulic problem and disregards the influence of the feedback coefficients. A coolant mixing experiment carried out during the commissioning phase of Kozloduy #6 in 1991 serves as the reference case for the qualification of the TRACE model. The experiment was performed at a thermal power of 281 MW and was initiated by closing steam isolation valve 1 (SIV-1) and isolating the steam generator 1 (SG-1). This produced a temperature rise of 13.6 K and a mass flow rate decrease of 3.6 % in loop #1. The heat up of the coolant in the core is approximately 3 K. 

Due to the coolant mixing in the vessel, the temperatures of the unaffected loops increase slightly. After 90 s the temperature of cold leg #1 exceeds the temperature of the corresponding hot leg. Within 20 min the temperature differences are stabilized to within 0.7 degrees. The final state of the experiment was reached at 30-35 min after the initiation of the transient. The boundary conditions for the coolant inlet temperatures, the mass flow rates and the system pressure have been provided by the benchmark specification and were used in the simulation. Fig. 5 shows the given RPV inlet temperatures during the experiment.
5.2 Selected results.

The investigation of the coolant mixing consists of two calculations. The first calculation is dedicated to the steady state and the second to the transient case. A detailed description of this investigation is given in (7). The obtained results for the steady state are in good agreement with the measured data provided by the benchmark specification (see Table 2, initial state). Thus, the 3D model is able to describe the coolant mixing within the RPV.

After reaching the steady state, a second calculation was performed. As an example Fig. 6 shows the obtained temperature distributions during the transient in the hot legs.

![Fig. 6  RPV outlet temperature distributions during the transient](image)

The temperature difference in the cold legs causes an asymmetric temperature profile at the RPV inlet. This yields to coolant mixing in the downcomer and in the lower plenum. Hot leg #1 has the same behavior as the corresponding cold leg. The clear temperature increase of hot leg #2 indicates the influence of the coolant mixing due to the asymmetric feed of the coolant. Hot legs #3 and #4 are not affected. Experimental investigations at the ROCOM facility have shown that the temperature perturbation is limited to the faulted loop (8)(9). The following two figures (Fig. 7 and Fig. 8) show a comparison of measured and calculated values of two selected hot leg temperatures during the transient. Both figures show small discrepancies between the recorded data and the predicted ones during the transient. Nevertheless Fig. 7 and Fig. 8 respectively show good agreements between the behavior of the measured data and the TRACE predictions.

The comparison for leg #2 would be of higher benefit but since the measured temperatures of leg #2 is not reliable due to a failure in the monitoring system a comparison between measured and calculated temperatures trends is not appropriate. Temperature loop’s data only for the initial and the final state (i.e. at 0 and 1800 sec.) are available for comparison (see Table 2).
An overview of some selected results at the final state of the test is given Table 2. This comparison confirms the conclusions made from the figures shown above. All results are within the measurement uncertainty. Thus, the developed 3D model can be used for the following investigations of a postulated MSLB.

Table 2 Comparison of the reference values and the TRACE predictions at initial state (t = 0 s) and final state (t = 1800s)

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Experimental values</th>
<th>TRACE V4.160</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>initial state</td>
<td>final state</td>
</tr>
<tr>
<td>Thermal power, MW</td>
<td>281</td>
<td>286</td>
</tr>
<tr>
<td>Pressure above core, MPa</td>
<td>15.593</td>
<td>15.593</td>
</tr>
<tr>
<td>Pressure drop (RPV), MPa</td>
<td>0.418</td>
<td>0.417</td>
</tr>
<tr>
<td>Coolant temp. (inlet #1), K</td>
<td>541.75</td>
<td>555.35</td>
</tr>
<tr>
<td>Coolant temp. (inlet #2), K</td>
<td>541.85</td>
<td>543.05</td>
</tr>
<tr>
<td>Coolant temp. (inlet #3), K</td>
<td>541.75</td>
<td>542.15</td>
</tr>
</tbody>
</table>
6 Calculation of a main steam line break transient

6.1 MSLB Scenario

The following investigation is about a postulated MSLB during normal operation. The break occurs outside the containment between the steam generator (SG) and the steam isolation valve (SIV) of loop #4. This leads to asymmetric core overcooling. The main purpose of this investigation is to analyze a possible return to power after reactor scram due to overcooling.

In the MSLB scenario, a reactor scram with a “stuck-rod” assumption immediately after the break initialization and the shut down of the main coolant pump (MCP) of the affected loop is assumed. Due to the MCP shut down, a reverse flow is established in the faulted loop in around 16 sec. A detailed description of the MSLB is given in (3). Fig. 9 shows the temperatures in the cold legs, and Fig. 10 shows the mass flow rates during the transient. These values will serve as boundary conditions for the TRACE/PARCS investigations of the RPV and core behavior.

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Experimental values</th>
<th>TRACE V4.160</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>initial state</td>
<td>final state</td>
</tr>
<tr>
<td>Coolant temp. (inlet #4), K</td>
<td>541.75</td>
<td>542.35</td>
</tr>
<tr>
<td>Coolant temp. (outlet #1), K</td>
<td>545.00</td>
<td>554.85</td>
</tr>
<tr>
<td>Coolant temp. (outlet #2), K</td>
<td>545.00</td>
<td>548.55</td>
</tr>
<tr>
<td>Coolant temp. (outlet #3), K</td>
<td>544.90</td>
<td>545.75</td>
</tr>
<tr>
<td>Coolant temp. (outlet #4), K</td>
<td>545.00</td>
<td>546.45</td>
</tr>
<tr>
<td>Mass flow rate (loop #1), kg/s</td>
<td>4737</td>
<td>4566</td>
</tr>
<tr>
<td>Mass flow rate (loop #2), kg/s</td>
<td>4718</td>
<td>4676</td>
</tr>
<tr>
<td>Mass flow rate (loop #3), kg/s</td>
<td>4682</td>
<td>4669</td>
</tr>
<tr>
<td>Mass flow rate (loop #4), kg/s</td>
<td>4834</td>
<td>4816</td>
</tr>
</tbody>
</table>

Fig. 9  Cold leg temperature distribution during the transient
6.2 Pre-investigations with the coupled code TRACE/PARCS

Since the RPV and core behavior during a MSLB scenario will be analyzed with the coupled code TRACE/PARCS, a check of the mapping scheme and the reading of the cross sections are necessary. For this purpose, seven hot zero power scenarios have been studied. These scenarios have been performed at hot zero power (3 kW) with a constant fuel and moderator temperature of 552 K. These scenarios are distinguished by different control rod positions shown in Table 3.

Table 3 Definition of hot zero power states

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Control rod position</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Groups 1-10: ARO&lt;sup&gt;1&lt;/sup&gt;</td>
</tr>
<tr>
<td>2</td>
<td>Groups 1-5: ARO, Group 6: 81 % wd, Groups 7-10: ARI&lt;sup&gt;2&lt;/sup&gt;, and Rod #90: 100 % wd&lt;sup&gt;3&lt;/sup&gt;</td>
</tr>
<tr>
<td>3</td>
<td>Groups 1-10: ARI</td>
</tr>
<tr>
<td>4</td>
<td>Groups 1-10: ARI, #90: 100 % wd</td>
</tr>
<tr>
<td>5</td>
<td>Groups 1-10: ARI, #63: 100 % wd</td>
</tr>
<tr>
<td>6</td>
<td>Groups 1-10: ARI, #140: 100 % wd</td>
</tr>
<tr>
<td>7</td>
<td>Groups 1-10: ARI, #140 and #117: 100 % wd</td>
</tr>
</tbody>
</table>

The arrangement of the control rods corresponding to Table 3 shows Fig. 11.

The TRACE model is the validated 3D-RPV model used for the coolant mixing problem. A mapping file was created<sup>(10)</sup> taking into account the provided cross section libraries to link the thermal hydraulic nodes with the neutronic nodes, since no automatic mapping exists for hexagonal geometry. Eigenvalue calculations with TRACE/PARCS were performed for different control rod positions.

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<sup>1</sup> All Rods Out
<sup>2</sup> All Rods In
<sup>3</sup> wd = withdrawn
An overview over the $k_{eff}$ values calculated by PARCS and a comparison of these values with values calculated by other benchmark participants is given in Fig. 12. All results are close and, the discrepancies can be explained due to the use of different programs and different nodalizations (PSU and FZK used TRACE/PARCS, UPISA used RELAP5/PARCS)\textsuperscript{(11)(12)}.

6.3 Selected results of the MSLB calculation

As mentioned in chapter 5.2, a steady state calculation prior to the transient calculation has been performed here too. The predicted results are also in good agreement with the benchmark specification. Fig. 13 shows the calculated values for the temperatures in the hot legs during the transient, and Fig. 14 shows a comparison of the inlet and outlet temperatures (loop #1 and #4) together with the mass flow rate (loop #4). It can be seen that

\textsuperscript{4} PSU provided no results for scenario 7
the temperature decreases in all hot legs after the MSLB and the reactor scram.

![Fig. 13](image)

**Fig. 13** Hot leg temperature distribution during the transient

The biggest impact is shown in hot leg #4. In the first seconds after the MSLB, the temperature in hot leg #4 has the same trend as the other legs. By means of Fig. 14, the behavior of the reactor can be explained.

![Fig. 14](image)

**Fig. 14** Comparison of the inlet and outlet temperatures for loop #1 and #4 and the mass flow rate of loop #4

At $t = 0$ s the reactor operates at normal conditions. The core heats up about 30 K. Due to the MSLB and the occurrence of boil-off on the secondary side of SG #4, the heat transfer ratio rises. This leads to a cool down of loop #4 and to coolant mixing in the downcomer. The temperature of cold leg #1 shows a slight impact because of mixing ($0 \ s < t \leq 16 \ s$). A MCP trip in loop #4 is initialized to mitigate the overcooling of the core. The measured temperature in cold leg #4 is now nearly the same as the temperature in cold leg #1 due to coolant mixing ($t \approx 16 \ s$). The shut down of MCP #4 yields a reverse flow through the faulted loop. The coolant from the unaffected loops, mainly loop #1, feeds the inlet of loop #4. Thus, the coolant flows backward through SG #4. Due to the increasing of the reverse flow in loop #4, the temperature decreases quickly ($20 \ s < t \leq 45 \ s$). After approximately 50 s, the reverse flow rises slightly and the outlet temperature of loop #4 decreases slightly. This coolant enters the reactor at the upper plenum and mixes there with
coolant from the core due to the large temperature difference. This yields a decrease of the
temperature in the unaffected loops. After approximately 100 s, the feed water supply of the
secondary SG side is shut down\(^{(3)}\). This leads to a faster evaporation of the secondary SG
side due to the broken steam line and a temperature increase of hot leg #4. This also has an
influence on the unaffected loops again (120 s < t < 600 s).

The main reason for this investigation was assessing the possibility of returning to
power after overcooling. Fig. 15 shows the development of the reactivity and the thermal
power during the transient. It is visible that the reactivity increases in the first 200 s of the
transient despite that the reactor is scrammed. The temperatures in the hot legs decrease in
the same period. On the other hand, it can also be seen that the power decreases after the
scram during the whole transient.

![Reactivity and thermal power comparison](image)

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Fig. 15  Comparison of the reactivity and the thermal power during the transient

No return to power has been predicted for the given boundary conditions at vessel in-
and outlet, which were predicted by the benchmark team with a complete integral model of
the whole plant. It means that the improved heat transfer over the affected steam generator
(leg #4) and thereby the amount of primary coolant temperature reduction (of the RPV and
core) which implies a positive reactivity insertion (predicted by the coupled code) is not
simulated here. It is taken into account by the boundary conditions. This is also valid for the
reverse flow through the affected loop. This investigation shows clearly how the change of
the thermal hydraulic core conditions during the MSLB affects the 3D core neutronic
behavior and vice versa as predicted by the coupled system TRACE/PARCS.

Since the MSLB is a postulated transient no experimental data are available for
further validation of the core neutronics, like the reactivity coefficient prediction, and the
thermal behavior.

7. Summaries and Outlook

The objective of this paper was the qualification of a 3D TRACE model and the
application of TRACE/PARCS in the frame of the OECD/NEA VVER-1000 Coolant
Transient Benchmark Phase 2. The first task was the post-test calculation of a coolant
mixing experiment and the second was the investigation of a postulated MSLB. Due to
multidimensional thermal hydraulic phenomena such as coolant mixing and asymmetric
core cooling, a 3D model of the RPV was required. This model was developed and
described in chapter 4. The qualification of this model has been done by applying it to a
post-test calculation of a coolant mixing experiment. The presented results are very
satisfying because all results are close to the measured values and the predicted flow patterns are physically sound. The 3D TRACE model was applied to simulate a postulated MSLB transient of the same VVER-1000 reactor. The obtained results are also physically sound and in good agreement to results of other benchmark participants. Hence, the developed 3D TRACE model is appropriate for VVER-1000 coolant mixing phenomena, and the coupled system TRACE/PARCS is useable for investigations of transients where the thermo hydraulics and the neutronics are strongly linked.

Acknowledgements

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References

(4) Ivanov, B., and Ivanov, K., Comparative Analysis of Exercise 1 of V1000CT-1 Benchmark, 3rd Workshop on VVER-1000 Coolant Transient Benchmark (V1000-CT), Garching, Germany, (2005-4).
(5) Ivanov, B., and Ivanov, K., Comparative Analysis of Exercise 2 of V1000CT-1 Benchmark, 3rd Workshop on VVER-1000 Coolant Transient Benchmark (V1000-CT), Garching, Germany, (2005-4).
(10) Kozlowski, Th., private communication, 2006
(11) Ivanov, B. and Ivanov, K., TRACE/ PARCS Modeling and Results for Exercise 2 of V1000CT-2, 4th Workshop on VVER-1000 Coolant Transient Benchmark (V1000-CT), Pisa, Italy, (2006-4).