# ANALYSIS OF THE VVER-1000 COOLANT TRANSIENT BENCHMARK PHASE 1 WITH RELAP5/PARCS

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## ABSTRACT

As part of the reactor dynamics activities of FZK/IRS, the qualification of best-estimate coupled code systems for reactor safety evaluations is a key step toward improving their prediction capability and acceptability. The VVER-1000 Coolant Transient Benchmark Phase 1 represents an excellent opportunity to validate the simulation capability of the coupled code system RELAP5/PARCS regarding both the thermal hydraulic plant response (RELAP5) using measured data obtained during commissioning tests at the Kozloduy nuclear power plant unit 6 and the neutron kinetics models of PARCS for hexagonal geometries. Selected results of performed investigations will be presented and discussed in this paper. The overall trends of most plant parameters are in a reasonable agreement with the experimental data. Nevertheless multidimensional thermal hydraulic models are needed for a more realistic description of the coolant mixing phenomena within the reactor pressure vessel.

## KEYWORDS

VVER-1000 reactors, coupled codes, thermal hydraulics, 3D neutron kinetics, PARCS, RELAP5.

## 1. INTRODUCTION

FZK is involved in the overall qualification of computational tools for the safety evaluation of nuclear power plants of different design to improve their prediction capability and acceptability. In this framework, the code RELAP5/PANBOX was qualified within the OECD/NEA PWR MSLB-benchmark (Sanchez et al., 2000). As continuation of this work, partly as a contribution to the international Code Assessment and Maintenance Program (CAMP) of the US NRC, the coupled code system RELAP5/PARCS is being validated. The PARCS-capabilities for quadratic fuel assembly geometry has been qualified in the frame of both the PWR TMI-1 Main Steam Line Break (MSLB) (Kozlowski, T. et al., 2001) and the BWR Peach Bottom Turbine Trip (PBTT) (Bousbia-Salah, et al., 2004) benchmarks. Especially the new PARCS-capability to solve neutron kinetics for hexagonal geometries is of interest (Joo et al., 2002). These aspects are important not only for VVER-type LWR but also for innovative reactor concepts. The international OECD/NEA VVER-1000 Coolant Transient Benchmark Phase 1(Ivanov *et al.*, 2002) is an excellent opportunity to validate the overall simulation capability of RELAP5/PARCS regarding both the thermal hydraulic plant response (RELAP5) using measured plant data and the neutron physics (PARCS). The Phase 1 of this benchmark is devoted to the analysis of switching on one main coolant pump while the other three pumps are in operation. It covers following exercises: a) Exercise-1: investigation of the integral plant response using a best-estimate thermal hydraulic system code with a point kinetics model b) Exercise-2: analysis of the core response for given initial and transient thermal hydraulic boundary conditions using a coupled code system with 3D-neutron kinetics model and c) Exercise-3: investigation of the integral plant response using a best-estimate coupled code system with 3D-neutron kinetics. For the analysis of these exercises, the following steps are approached:

- Development of an integral model of the Kozloduy nuclear power plant (NPP) including all major systems for RELAP5,
- Development of three-dimensional core model for the coupled code system RELAP5/PARCS, and
- Integration of the above developed models in one integral model to investigate the plant response.

In this paper, the main modeling and nodalization issues will be presented and its relevance for the different exercises will be outlined. Finally selected results of Phase 1 will be given and discussed.

## 2. MAIN COOLANT PUMP (MCP) SWITCH-ON TEST DESCIPTION

The MCP switch-on test was performed during the decommissioning phase of the NPP Kozloduy, Unit 6, which is a four loop pressurized water reactor (PWR) of Russian-design equipped with one horizontal steam generator per loop (Ivanov et al., 2002). At the beginning of the experiment, the reactor power was 824 $MW_{th}$ and the MCP-3 was switched off while the other three pumps are working almost at nominal conditions. Therefore part of the coolant flow injected into the downcomer by the three pumps is flowing back through piping of the affected loop-3. This results in a considerable mixing of cold and hot coolant in the upper plenum. Hence total core mass flow rate reduces from around 18000 at nominal conditions to around 13600 kg/s. This reversed flow leads to multidimensional flow conditions in the down comer of the reactor and in the upper plenum of the core. The main parameters characterizing the primary circuit before the test are presented in **Table 4**. The test starts by switching on the MCP-3. During the first few seconds the flow in the third loop reverses leading to an increase of the total mass flow rate through the core. The increased coolant inventory leads to a decrease of the coolant temperature and to an increase of the coolant density. Hence the core power undergoes initially a rapid increase stabilizing later on at a power level higher than the initial power. Due to the coolant mixing in the downcomer, the mass flow rate of loop-1, -2 and -4 slightly decreases while the mass flow rate of loop-3 greatly increases until around 13 s. Afterwards the mass flow rate of all loops are similar. These modifications of the coolant stream influence the heat transfer across the steam generators leading e.g. to an increase of the water level of steam generator 1, 2 and 4 and to non-symmetrical core cooling. Concluding, the test is characterized by tight interactions between the core and the system thermal hydraulics. During the progression of the experiment that lasted for 130 s, most important primary and secondary thermal hydraulic parameters of the plant were measured. These data is summarized in **Table 4**, where also the error band of the measurement devices is given.

## 3 CALCULATIONAL MODELS AND NODALIZATIONS

To predict the plant response during the MCP switching-on test with the simulations tools both the primary and secondary plant systems as well as selected safety and control systems of the plant need to be represented in the numerical model (input deck). In the benchmark specifications (Ivanov *et al.*, 2002), all data needed to develop the respective models for PARCS and RELAP5 are given. Based on this information, an integral thermal hydraulic plant model including a point kinetics approach for Exercise-1, a multidimensional neutron kinetics and channel thermal hydraulic core model for Exercise-2, and an integral plant model with a multidimensional core model for Exercise-3 were elaborated. Details of these models will be given hereafter.

### 3.1 Integral plant nodalization

The integral plant model developed for the Exercise-1 and -3 is shown in **Figure 1**, where only two of the four loops are exhibited (Metz, 2003). In this model most relevant primary, secondary, and safety systems of the Kozloduy plant are included. For the Exercise-1 a point kinetics model was implemented. The ***Core*** (volumes 845 and 843) is represented by two parallel volumes, one representing the core average channel and the other one the core bypass. The ***downcomer*** (volumes 108,208,308,408) is represented in four equal parts, each one connected to one loop so that the complex flow conditions prevailing during the pre-test phase and during the first 13 s of the transient are simulated appropriately. The ***primary circuit*** consisting of the piping system (loop-1: volumes 146, 140,141, 142, 144, 145), the pumps (volumes 144, 244, 344, 444), and the steam generator tubes (SG-1 volumes: 120, 121, 122) is fully incorporated in the model. In addition, the ***pressurizer (PZR)*** with the 4 groups of heaters is also included in the model. Furthermore the ***make-up and drainage system*** are included in the model. The full data of the main coolant pumps that are of Russian-type is taken from the specifications. Each ***steam generator (SG)*** consists of 11000 tubes that are horizontally arranged between the hot and cold collector tubes. They are vertically grouped in three units associated two the primary and secondary volumes.

The ***secondary side of the steam generators*** is large and characterized by a complex 3-dimensional flow inside this big volume. The back flow in the SG-downcomer (SG-1 volumes: 150, 151, 152, 109) is represented in the model, too **Figure 1**. The ***feedwater system*** (SG-1: volume 190) is simply modeled by a volume providing a constant coolant mass flow with the predefined coolant temperature. No ***emergency feed water system*** is considered in the model since these systems are not expected to be activated during the transient. The ***steam lines*** (loop-1 volumes: 181, etc.) are in detail modeled including the valves, common header, turbine stop valves and the associated safety steam valves and the steam dump valve groups. The core fuel pins, the steam generator tubes, the PZR-heaters, as well as the walls of all relevant primary and secondary systems (RPV, cold and hot legs, steam generator shell) are considered in the model as ***heat structure components*** with its respective heat transfer area, heater diameter, material data, and heat source when available. They are connected to the corresponding fluid volumes via convective boundary conditions. In the ***point kinetics model*** the given neutron physical data characterizing the VVER-1000 fuel like prompt neutron lifetime, effective fraction of delayed neutrons, decay constants of delayed neutrons, axial power profile, moderator and Doppler reactivity coefficients are implemented. The Doppler feedback is calculated using the following Doppler temperature ($T_{Doppler}$) instead of the volume averaged fuel temperature in all exercises:

$$T_{Doppler} = 0.7 \cdot T_{fuel}^{surface} + 0.3 \cdot T_{fuel}^{center} \; .$$

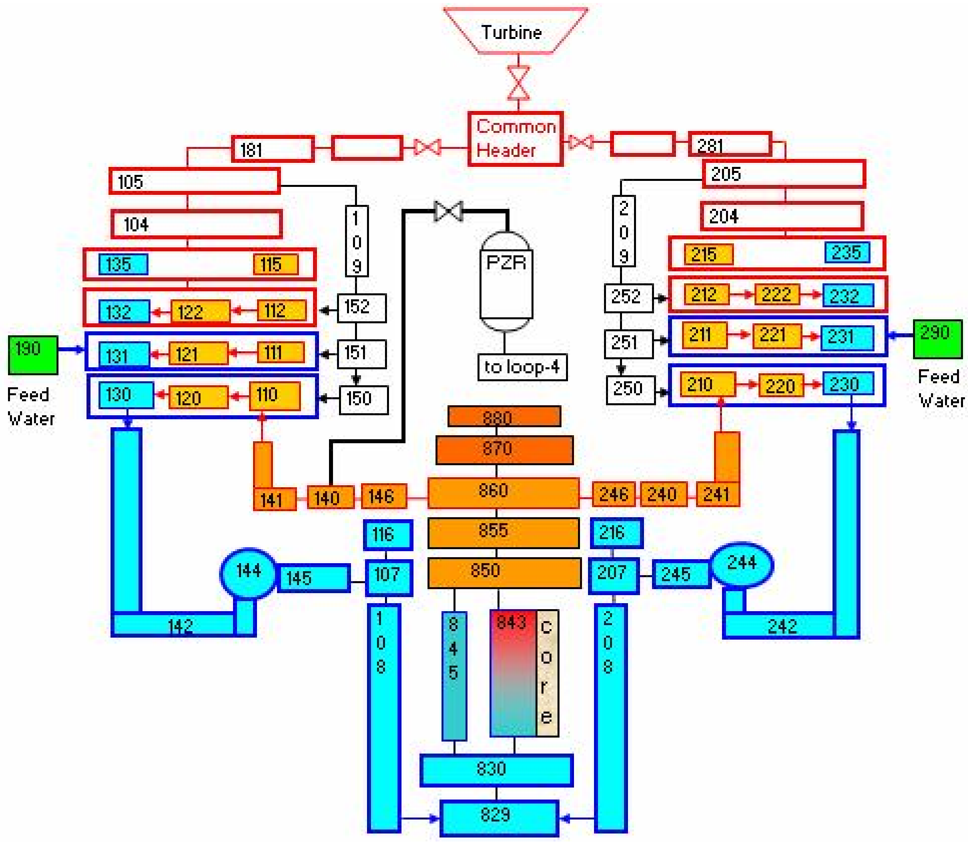

**Figure 1** Nodalization of the Kozloduy plant (reactor pressure vessel with two loops)

## 3.2 Multidimensional core modeling

For the Exercise-2 and -3 a multidimensional core model is needed for both the thermal hydraulics as well as the neutronics representation of the core for the coupled code system RELAP5/PARCS. The multidimensional core model is developed for the Exercise-2, where only the core behavior is evaluated for given boundary and initial conditions at the core inlet and outlet. Later on this model is fully merged with the integral plant model so that the transient can be analyzed with the coupled code system RELAP5/PARCS.

### 3.2.1 Neutron kinetic core model

The transient neutron kinetic response of the core is simulated by the core simulator PARCS version 2.5 which is coupled to RELAP5 via PVM (Barber, et al., *1998*). PARCS solves time-dependent neutron diffusion equation for up to six energy groups using the triangular polynomial expansion (TPEN) method (Joo et al., 2002*).* In the model, 8 hexagonal fuel assembly (FA) rings and a reflector assembly ring are considered, where each assembly represents a numerical radial node. Here a total of 28 fuel assembly types and one reflector assembly are considered. Axially, all assemblies are subdivided in 12 equal nodes, one for the lower and upper reflector and 10 for the fuel zone. In **Figure 2**, the numbering of the FAs with and without control rods (Roman numbers) within the VVER-1000 core is given. A total of 283 material compositions (unrodded) are considered for this problem. Each one is characterized by unique material properties like enrichment, density as well as burn-up, absorber rod history and spectral history. These data were prepared and generated by the benchmark team for this benchmark and it was made available to the participants in two cross-section libraries (look-up tables). These look-up tables include information for two energy groups such as diffusion coefficients, scattering, absorption, and fission macroscopic cross sections, assembly discontinuity factors, etc. Additional information about the delayed neutron fractions, decay constants and neutron velocity are provided in the tables. PARCS has several models to read in different formats of cross-section libraries. An additional cross-section library containing 260 material compositions (rodded) was also delivered to account for absorber rods movement. The look-up tables are functions of fuel temperature and coolant density. A suitable parameter range of these variables was selected to cover the expected parameter changes for the steady state and during the transient progression. PARCS uses a multidimensional interpolation scheme for the online update of the cross-sections during the transient phase in dependence of the actual parameters.

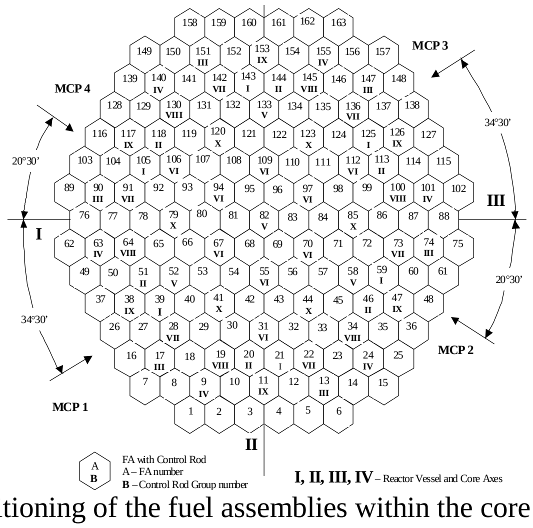


**Figure 2** Radial positioning of the fuel assemblies within the core

### 3.2.2 Thermal hydraulic core model

The thermal hydraulic core model for the coupled calculations is based on the nodalization and core channels/FA-mapping proposed in the specification, **Figure 3**. According to this, the whole core is in radial direction divided in 19 thermal hydraulic channels, 18 for the core region and 1 for the radial reflector. All fuel assemblies with the same number are associated to one thermal hydraulic channel. An additional channel is considered (channel 19) to represent the flow area of the 48 reflector assemblies (RA). In axial direction, the parallel channels are subdivided in 12 nodes, the bottom and top nodes for the axial reflector and the remaining 10 nodes for the active core, **Figure 4**. The additional fluid volumes at core inlet and outlet are represented because they are needed to define the initial and boundary conditions for Exercise-2. At the core inlet (volume 601 up to 619), the mass flow rate as well as the coolant temperature is given. The system pressure is defined in at the core outlet (volume 800). In the RELAP5-model, 18 ***heat structures components*** representing the 18 groups of FA are modeled that are linked to the 18 core channels by convective boundary conditions. These heat structures have the same axial nodalization like the corresponding fluid channels. In radial direction each heat structure is subdivided in 7 zones, 4 in the fuel, one gap and two in the cladding material.

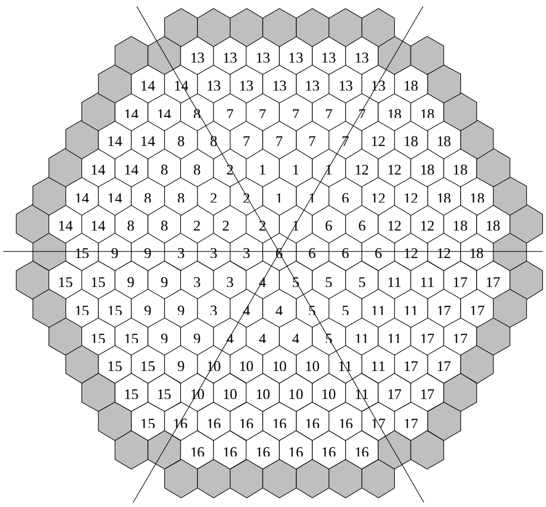


**Figure 3** Radial grouping of the fuel assemblies for the mapping with the neutronic nodes i.e. 18 parallel channels in the core and 1 channel for the bypass flow (reflector assemblies)

The multidimensional neutron kinetics and thermal hydraulics core model was entirely incorporated in the integral plant model to perform the coupled calculations for Exercise-3. The volumes 702, 800 and 601 up to 619 are not part of the integral model.

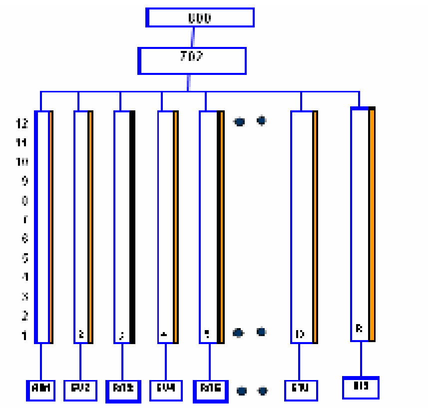

**Figure 4** Thermal hydraulic core nodalization for Exercise-2 and -3

## 3. SELECTED RESULTS OF THE CALCULATIONS

Since the main goal of this benchmark is to assess the prediction capability of coupled code systems with 3D-neutorn kinetics models, most of the results are mainly related to the Exercise-2 and -3. Exercise-1 is necessary to develop a well balanced integral plant model to satisfactorily predict the Kozloduy plant behavior during the MCP-switch-on test. On the contrary, Exercise-2 is aimed to demonstrate the operability of the coupled code system such as reading correctly the cross-sections (look-up tables), the appropriateness of the interpolation scheme for the cross-section update, the convergence of both the neutronics and the thermal hydraulic model in a coupled calculation, etc. The numerical simulation of the MCP switch-on test with RELAP5/PARCS-V2.5 was performed on a LINUX platform with PVM-environment including the following steps: a) Run RELAP5-stand alone with the null transient option for some 200 s until stable thermal hydraulic plant conditions are reached b) Run the RELAP5/PARCS coupled system with the steady state option until the eigenvalue calculation of PARCS converged c) Then, run the coupled code with the transient option restarting from the latter steady-state condition for both codes until 130 s (duration of the test).

### 3.1 Steady state results of Exercise-2

Several calculations were performed with RELAP5/PARCS-V2.5 for both the hot zero power (HZP) and hot full power (HFP) conditions of the Kozloduy core using the multidimensional model described above. The initial and boundary conditions as well as the control rod positions used to predict the different reactivity worth of the HZP-state are taken from the specifications *(*Ivanov et al., 2000*)*. HZP-conditions mean a nominal power of 0.1 % of the total power and fixed feedback thermal hydraulics conditions i.e. the moderator density in the core is 767.1 kg/m³ and the fuel temperature amounts 552.15 K. In order to assess the developed models the effective multiplication factor ($k_{eff}$) for different core states defined by different positions of the absorber rod groups were predicted. In **Table 1** neutron physical parameters of the HZP-state predicted by RELAP5/PARCS are compared with some data from the specification, where predictions and reference values are in a reasonably good agreement. The reactivity worth for different HZP-core states calculated by the coupled code are compared to the values given in the specifications. Both are close to each other.

| Calculated parameters | RELAP5/PARCS | Reference Data |
|---|---|---|
| $K_{eff}$ | 0.999669 | |
| Radial power peaking factor | 1.4034 | |
| Axial power peaking factor | 1.514 | |
| Axial offset | -0.1726 | |
| Ejected rod worth %dk/k | 0.078 | 0.09 |
| Control rod group 10 worth, %dk/k | -0.69 | - 0.61 |
| Tripped rod worth, %dk/k | -7.24 | - 7.02 |

**Table 1:** HZP results obtained with RELAP5/PARCS compared to reference values

The hot full power (HFP) core state is characterized by beginning of cycle (BOC) fuel conditions with an average exposure of 30.7 effective full power days (EFPD) and a thermal power of 824 MW. For this core state both steady state and transient calculations were performed with the coupled code system using the initial and transient boundary conditions given in the specifications (Ivanov, et al., 2002). The position of the absorber rod group attained for the HFP-state in comparison to the HZP-state is indicated in **Table 2**. The main neutron physical parameters predicted for the stationary conditions of the HFP by RELAP5/PARCS are summarized in **Table 3**.

| Core state | G1-4 | G5 | G6-8 | G9 | G10 | G10 EjRod |
|---|---|---|---|---|---|---|
| Hot Zero Power (HZP) | 100 | 100 | 100 | 64 | 0 | 0 |
| Hot Full Power (HFP) | 100 | 100 | 100 | 100 | 36 | 36 |

**Table 2:** Position of the control rod groups for the HFP states (100 means all rods out, 0 means all rods in)

| Calculated Parameters | RELAP5/PARCS |
|---|---|
| $K_{eff}$ | 1.000425 |
| Radial power peaking factor | 1.3471 |
| Axial power peaking factor | 1.408 |
| Axial offset | -0.1734 |

**Table 3:** HFP results obtained with RELAP5/PARCS for the steady state conditions

The RELAP5/PARCS predicted a non-symmetrical axial power distribution for the steady state conditions expressed by an axial offset of 17.34 %. In **Figure 5** the predicted core averaged axial power peaking is compared to the one given in the specification that was calculated by the benchmark team (PSU). It can be seen, that both curves show the same trends with slightly deviations mainly around 1 m and 2 m height.

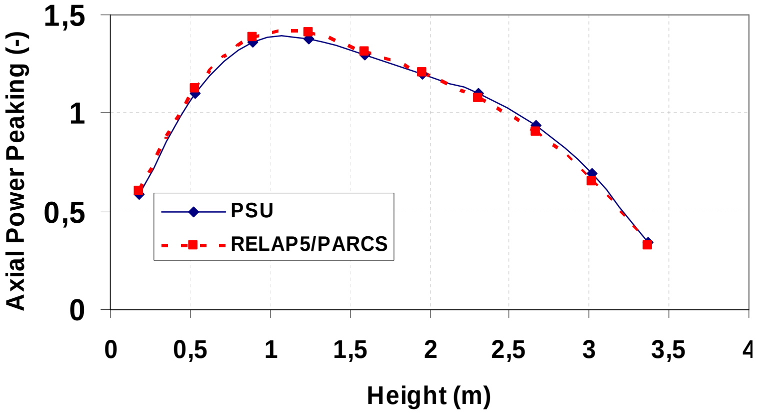


**Figure 5** Comparison of the predicted axial power profile for the HFP steady state

### 3.2 Steady state results of Exercise-1 and-3

The appropriate simulation of the complex flow conditions in the primary circuit and especially in the downcomer and the upper plenum, schematically shown in **Figure 6** and **Figure 7**, is important for the successful simulation of the steady state plant conditions at the time before the MCP switch-on test with both the stand-alone RELAP5 and the coupled code system RELAP5/PACRS. Hence the downcomer was divided in four volumes that are connected to each other by cross-flow junctions. To assess the flow conditions in the upper plenum, especially the distribution of the reverse flow of about 1500 kg/s entering into the outer ring between the RPV-wall and the inner perforated shell, an isolated CFX-model was developed for the upper plenum (Boettcher, 2004).

In **Figure 8** the predicted flow redistribution in the upper plenum, especially in the outer ring is illustrated. It can be seen that a considerable part of the cold back flow of loop-3 is flowing sideward in the outer ring to the outlet orifice of the loop-2 since loop-3 and loop-2 are located next to each other. A minor part of the coolant of loop-3 is also sideward redirected into the outlet orifice of loop-4. It has to be noted that the rest of the cold flow of the loop-3 is entering the inner volume of the upper plenum through the perforations of the inner shell. Based on these results, the fluid volumes representing the outer ring of the upper plenum were connected by cross-flow junction in the RELAP5-model so that the reverse flow of loop-3 can go to the outlet orifices of loop-2 and -4. The mixing of a part of the flow of loop-3 in the upper plenum before leaving is also allowed. With this model extensions the prediction of the stationary plant conditions were improved, **Table 4**. The plant data taken from the specifications is compared with the predictions using the RELAP5-stand alone code and the coupled code RELAP5/PARCS. It can be seen that the deviation of most parameters is very small for both calculations. Some of the parameters are slightly under-predicted and others are slightly over-predicted by both calculations.

As part of the qualification of the initial steady state attention was paid also to the primary circuit power balance, i.e. the difference between the core power plus Main Coolant Pump (MCP) power and the thermal power transferred over all four steam generators, which amounts around 12 MW. The good agreement between data and predictions for the stationary plant conditions before the test demonstrates that the developed integral plant model is appropriate for the subsequent study of the plant response.

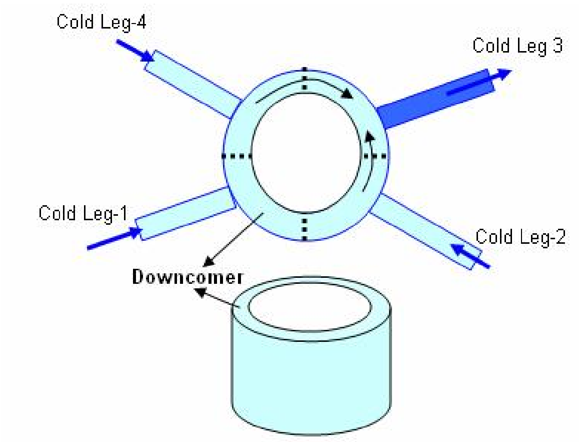


**Figure 6** Flow conditions in the downcomer

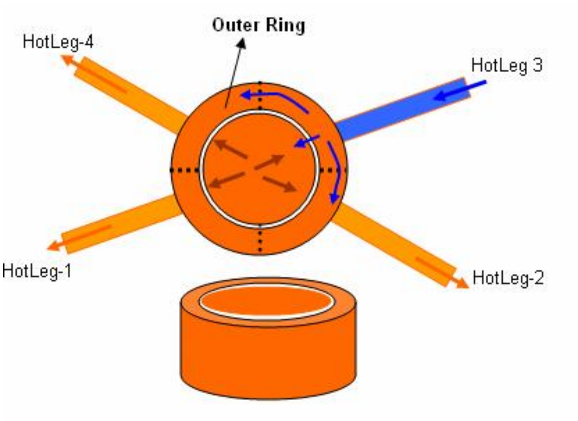


**Figure 7** Flow conditions in the upper plenum

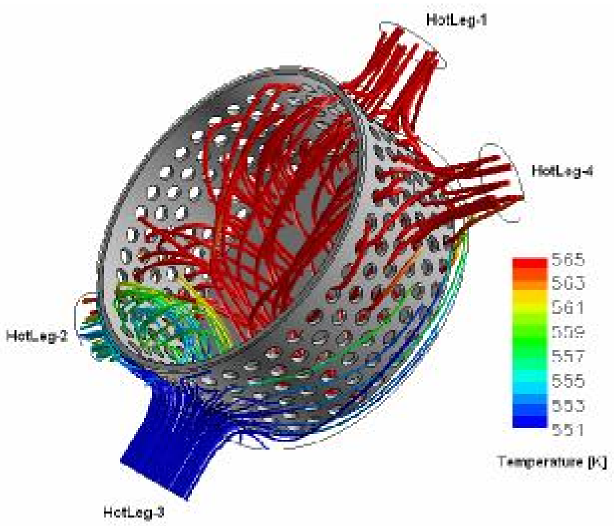


**Figure 8** Complex coolant mixing in the upper plenum of the Kozloduy NPP predicted by CFX-5

| Parameter | Benchmark Data | | | RELAP5 | Deviation | R5/PARCS |
|---|---|---|---|---|---|---|
| | Unit | Data | Accuracy | Exercise-1 | % | Exercise-3 |
| Thermal core power | MW | 824 | ± 60 | 824 | 0 | 824 |
| RCS mass flow rate | Kg/s | 13611 | ± 800 | 13577 | -0.25 | 13612 |
| Primary side pressure | MPa | 15.60 | ± 0.3 | 15.62 | 0.13 | 15.62 |
| Sec. side pressure | MPa | 5.94 | ± 0.2 | 6.105 | 2.83 | 6.106 |
| Cold leg temp. loop-1 | °K | 555.55 | ± 2 | 555.43 | -0.02 | 555.44 |
| Cold leg temp. loop-2 | °K | 554.55 | ± 2 | 554.61 | 0.01 | 554.62 |
| Cold leg temp. loop-3 | °K | 554.35 | ± 2 | 554.94 | 0.11 | 554.95 |
| Cold leg temp. loop-4 | °K | 555.25 | ± 2 | 555.16 | -0.02 | 555.17 |
| Hot leg temp. loop-1 | °K | 567.05 | ± 2 | 566.18 | -0.15 | 566.19 |
| Hot leg temp. loop-2 | °K | 562.85 | ± 2 | 563.71 | 0.15 | 563.72 |
| Hot leg temp. loop-3 | °K | 550.75 | ± 2 | 550.65 | -0.02 | 550.66 |
| Hot leg temp. loop-4 | °K | 566.15 | ± 2 | 565.43 | -0.13 | 565.43 |
| Mass flow rate loop-1 | Kg/s | 5031 | ± 200 | 5021 | -0.20 | 5029 |
| Mass flow rate loop-2 | Kg/s | 5069 | ± 200 | 5036 | -0.65 | 5043 |
| Mass flow rate loop-3 | Kg/s | -1544 | ± 200 | -1503 | -2.66 | -1491 |
| Mass flow rate loop-4 | Kg/s | 5075 | ± 200 | 5034 | -0.81 | 5041 |
| PZR water level | m | 7.44 | ± 0.15 | 7.44 | 0.00 | 7.44 |
| Water level in SG-1 | m | 2.30 | ± 0.075 | 2.305 | 0.22 | 2.304 |
| Water level in SG-2 | m | 2.41 | ± 0.075 | 2.409 | -0.04 | 2.409 |
| Water level in SG-3 | m | 2.49 | ± 0.075 | 2.439 | -2.05 | 2.439 |
| Water level in SG-4 | m | 2.43 | ± 0.075 | 2.458 | 1.15 | 2.457 |
| DP over core | MPa | 0.225 | ± 0.2 | 0.2570 | 14.22 | 0.255 |
| DP over MCP-1 | MPa | 0.492 | ± 0.2 | 0.4845 | -1.52 | 0.4825 |
| DP over MCP-2 | MPa | 0.469 | ± 0.2 | 0.4818 | 2.73 | 0.4798 |
| DP over MCP-3 | MPa | 0.179 | ± 0.2 | 0.1811 | 1.17 | 0.1787 |
| DP over MCP-4 | MPa | 0.500 | ± 0.2 | 0.4824 | -3.52 | 0.4804 |

**Table 4:** Comparison of the predicted main parameters of the Kozloduy plant with the data given in the benchmark specifications

## 3.3 Transient results

Selected results obtained with the RELA5/PARCS are presented and compared to results of Exercise-1 and to the plant data.

**Global plant response (Exercise-1 versus Exercse-3)**

The transient is initiated by the switch-on of the MCP-3. As a consequence the mass flow rate of the loop-3 starts to re-invert, **Figure 9**, leading to a continuously increase of the coolant mass flow, **Figure 10**. After about 15 s, all loops reached similar mass flow rates which remain almost unchanged during the transient. As a result, the core averaged coolant temperature decreases some degrees for the first 15 s, **Figure 11**.

The fuel temperature undergoes the same trend during the first 15 s like the coolant temperature. Consequently the total reactor power increases, **Figure 12**, rapidly until around 15 s. Later on this trend continues slowly until the end of the transient. The power increase predicted with the point kinetics model (Ex-1) is higher (5.5 % of nominal power) than the one predicted with the 3D-neutron kinetics model (Ex-3, 3.7 % of nominal power) at the end of the transient. The reason is the use of an axial power peaking, Doppler and moderator reactivity coefficients estimated for the core conditions at the beginning of the test during the whole transient. This leads to an overestimation of the reactivity inserted into the core. On the contrary, PARCS solves a 3D-problem with local estimation of the feedbacks by means of using cross-section sets depending on local thermal hydraulic parameters that represents a more realistic description of the underlying asymmetrical core behavior. The predictions of Exercise-1 may improve if a more detailed and sophisticated point kinetics model is used for this problem. But this was not the aim in the frame of this analysis.

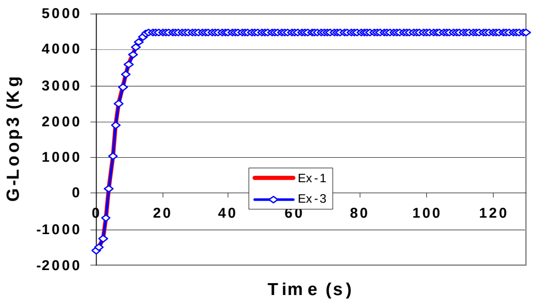


**Figure 9** Predicted reverse flow of the loop-3 during the test

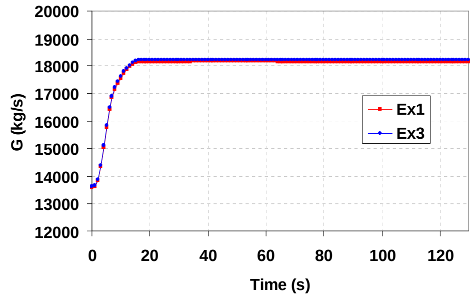


**Figure 10** Predicted Change of the total primary mass flow rate during the test

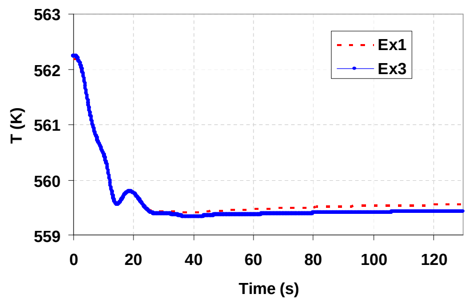


**Figure 11** Predicted core averaged coolant temperature during the test

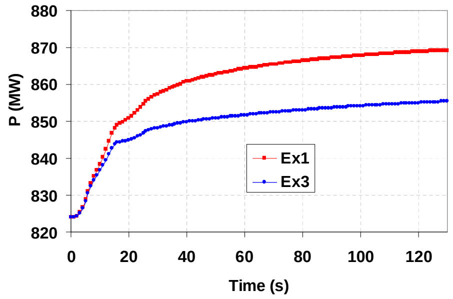


**Figure 12** Predicted power increase for Exercise-1(Ex-1) and Exercise-3 (Ex-3)

## Transient plant behavior: Predictions versus experimental data

During the MCP switch-on test, several parameters of the plant were measured and its error bands were estimated, see **Table 4**. To show the quality of the RELAP5/PARCS-predictions selected parameters are chosen and compared to available data. In **Figure 13** the pressure of the upper plenum predicted by the codes is compared to the measured values. The couple code is able to predict the initial time and decrease rate of the pressure during the first 10 s. Later on the contraction of the primary system coolant is over-predicted by the simulations, which results in a faster decrease of the PZR-level as can be observed in **Figure 14**. Around 55-80 s in the transient both predictions and measured PZR-level are very close to each other. The primary to secondary heat transfer is almost constant in the calculation while a slowly but steady cooldown of the primary system is observed in the measured data. The changes of coolant temperature of the loop-1 for both cold and hot legs predicted by the codes in comparison with the measurements data are exhibited in **Figure 15** and **Figure 16**.

It must be noted that the changes of the coolant temperature are moderate and smaller than the error band of the temperature measurement devices. The overall trend of the measured data can be reproduced by the calculation. The predictions tend to estimate a larger variation of the coolant temperature than one shown by the data. In **Figure 17** the measured pressure drop over the MCP of loop 3 is compared with the values predicted by the code. The agreement is quite good for the whole transient. On the secondary side, the measured variation of the water level in the steam generator 1 is also compared with the predicted one in **Figure 18**. Even though prediction and data start from around the same level, the code overestimates the heat transferred to the secondary side for the first 15 s. Then, the primary-to-secondary heat transfer is underestimated until the end of the transient.

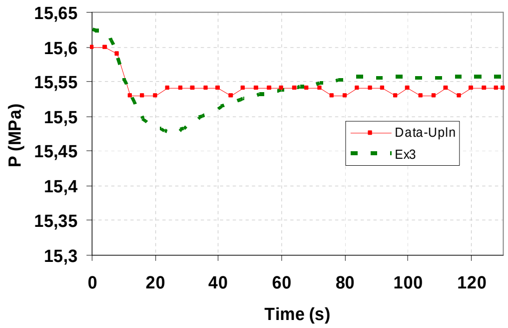


**Figure 13** Comparison of the pressure measured in the upper head with the calculated trend

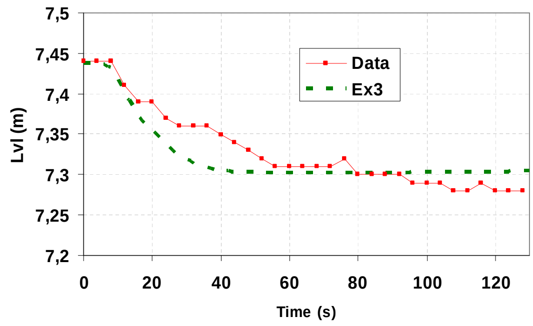


**Figure 14** Comparison of the predicted and measured PZR-water level

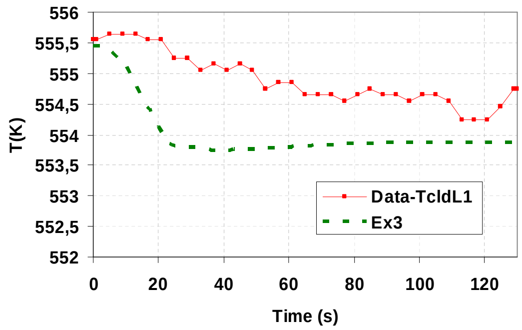


**Figure 15** Comparison of the predicted coolant temperature of the cold leg-1 with the data

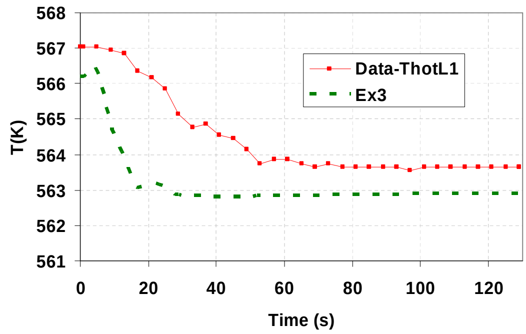


**Figure 16** Comparison of the predicted coolant temperature of the hot leg-1 with the data

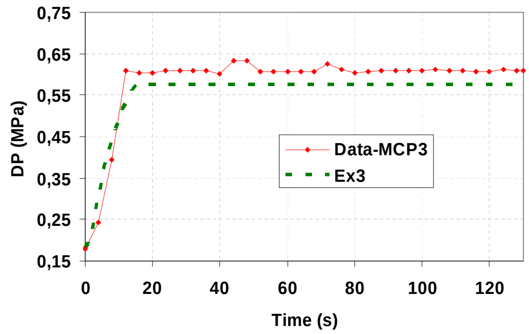


**Figure 17** Comparison of predicted pressure drop over MCP-3 with the experimental data

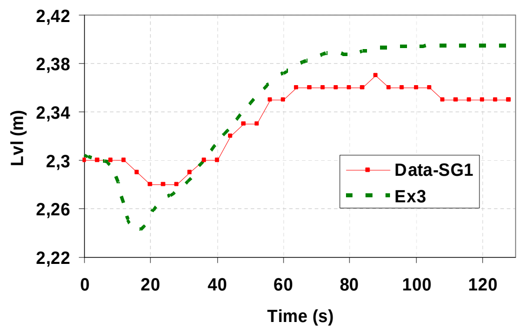


**Figure 18** Comparison of the measured water level of the steam generator (loop-1)secondary side with the prediction trend

## Multidimensional core behavior

The use of coupled codes with 3D-neutron kinetics models allows a more detailed analysis of the core response compared to the point kinetics. In **Figure 19** the core averaged axial power peaking predicted by PARCS for three time windows during the transient is shown. It can be observed that for the basic scenario there is only a very moderate variation of the power peaking. A similar trend was observed when the core averaged radial power profile was analyzed. There are minor changes of the local radial power profile at different time windows. Hence only the relative radial power profile at 15 s transient time is indicated in **Figure 20** with a radial peak value of 1.39.

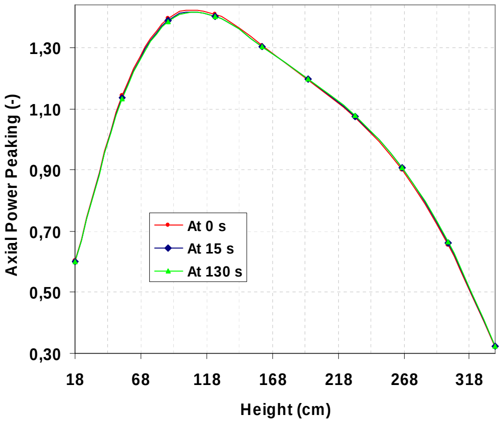


**Figure 19** Predicted core averaged axial power peaking for three different times during the transient progression

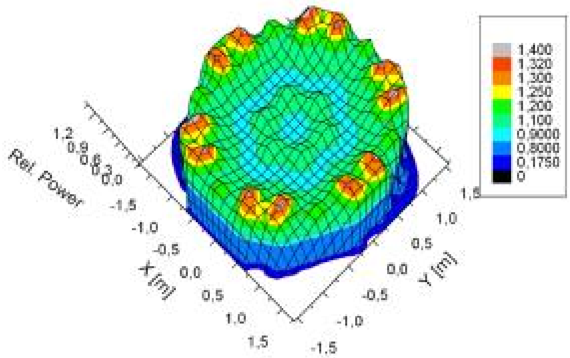


**Figure 20** Predicted core averaged relative radial power distribution for the basic scenario at 15 s transient time

## 4. CONCLUSIONS AND OUTLOOK

The Phase 1 of the V1000 Coolant Transient was analyzed with different computational tools using both point kinetics and 3D-kinetics models. It was demonstrated that the developed integral plant model as well as the multidimensional core model are appropriate to describe the main plant and core response what is reflected in the good agreement of the predicted plant initial conditions against the measured data. For the transient phase, most of the predicted parameters show trends that are qualitatively in good agreement with the available experimental data obtained during the test. The predicted reactivity worth for the different states of the cold zero power is close to the ones given in the specifications. Also the axial power peaking factor for the steady state hot full power estimated by RELAP5/PARCS agreed well with the reference curve given in the specifications. From the multidimensional results it is apparent that the non-symmetrical spatial power perturbation for the investigated test is rather moderate. The analysis of an additional scenario where the moderate non-symmetrical power distribution is reinforced by assuming the ejection of a control rod is additionally being investigated to check the codes' capabilities for such conditions. These investigations clearly illustrate the usability of coupled codes systems with 3D-neutorn kinetics models as promising simulation tools to predict local hot spots within the core. It can be stated that the time dependent neutron diffusion solution for hexagonal geometries of PARCS works quite well in connection with the thermal hydraulic part (RELAP5) and that the RELAP5/PARCS works very stable and fast enough under both Linux and Windows platforms. Finally the analysis of the MCP switch-on test, characterized by multidimensional flow within the RPV, shows the limits of one dimensional thermal hydraulic models like the ones implemented in RELAP5 to describe multidimensional flow problems such as coolant mixing. Consequently a more realistic description of such transients may only be possible using three-dimensional thermal hydraulics (CFD-like) models loosely coupled with the multidimensional neutron kinetics models. This kind of investigations is envisaged for the Phase 2 of this benchmark (V1000CT-2).

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