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International Agreement Report

Qualification of the Three-Dimensional Thermal Hydraulic Model of TRACE using Plant Data

Prepared by:

V. Sánchez-Espinoza
Forschungszentrum Karlsruhe GmbH
Hermann-von-Helmholtz-Platz 1
76344 Eggenstein-Leopoldshafen
Germany

A. Calvo, NRC Project Manager

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ABSTRACT

In this report the investigations performed to validate the 3D thermal hydraulic model of TRACE using data gained in the nuclear power plant Kozloduy Unit 6 regarding the coolant mixing within the reactor pressure vessel will be presented. These data were distributed to the scientific community in the frame of the VVER-1000 Coolant Transient Benchmark Phase 2. The measured data was recorded during the non-symmetrical core heat-up test caused by the closure of the isolation valve of the steam line of the loop-1. Since plant data for code validation is rather scarce, this coolant mixing data is very much appropriate for the qualification of the 3D thermal hydraulic models of the TRACE code.

A detailed multidimensional model for the RPV of the VVER-1000 was elaborated using the 3D VESSEL component of TRACE. The complete model consisted of more than 1000 3D thermal hydraulic cells. Using this model a post test analysis of the heat-up test was performed with the TRACE version V4160 in a Linux cluster.

The obtained results for the initial and final state are in very good agreement with the plant data. TRACE needed not more than six minutes for the simulation of the whole test duration of 1800 sec. It was demonstrated that the chosen 3D-nodalization of the RPV is adequate for the description of the coolant mixing phenomena in a VVER-1000 reactor.

FOREWORD

This validation report describes the investigations performed at Karlsruhe Institute of Technology (KIT) to validate the three dimensional thermal hydraulic models of the codes system TRACE by using coolant mixing data measured at the VVER-1000 nuclear power plant Kozloduy Unit 1 during the commissioning phase. This contribution is performed in the frame of the US NRC CAMP Program which is a very important international effort to increase the validation basis of safety analysis codes in the nuclear field.

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EXECUTIVE SUMMARY

In this report, the validation of the three-dimensional thermal hydraulic model of TRACE (3D VESSEL component) using data obtained at the nuclear power plant Kozloduy regarding the coolant mixing phenomena is presented. This work has been performed as part of the FZK-contribution to the international CAMP (Code Application and Maintenance Program) of the US NRC.

The coolant mixing data has been distributed to the scientific community in the frame of the OECD VVER-1000 Coolant Transient Benchmark Phase 2. The experiment was initiated by the closure of the steam isolation valve of the steam line of the loop-1 which caused a non-symmetrical heatup of the primary coolant entering the downcomer.

To catch the main mixing process taking place in the downcomer and upper plenum a full 3d model of the whole reactor pressure vessel including the downcomer, lower plenum, core and upper plenum was developed for TRACE. The main challenges developing this model were the complex constructive peculiarities of the lower and upper plenum of the VVER-1000 reactor that determine the flow paths for the coolant and hence influence the mixing process. The detailed multidimensional model for the RPV of the VVER-1000 consists of more than 1000 3D cells allowing for mass-, momentum- and energy exchange in axial, radial and azimutal direction. The solid structures within the RPV were also taken into account in the model.

The post test calculation of the heat-up test was performed with the TRACE version V4160 in a Linux cluster. The obtained results for the initial and final state are in very good agreement with the plant data. Also the few trends of the coolant temperature in the hot legs during the test time (1800 s) could be reproduced by TRACE.

The comparison of the measured coolant temperature at the fuel assembly outlet with the one predicted by TRACE is close to each other. Hence it can be stated that TRACE is able to describe the single phase coolant mixing process within the RPV of a VVER-1000 reactor in an acceptable manner. This demonstrates that the chosen 3D-nodalization of the RPV is adequate for the description of the coolant mixing phenomena.

For this single phase problem TRACE needed not more than six minutes CPU-time for the simulation of the test duration of 1800 sec.

This validated model will be used for the investigations of different transients such as main steam line break, etc. with coupled neutronic/thermal hydraulic system codes like TRACE/PARCS.

NOMENCLATURE

TRACE	TRAC/RELAP Advanced Computational Engine
PARCS	Purdue advanced reactor core simulator
VVER	Water-Water energy reactor
RPV	Reactor Pressure Vessel
CATHARE	French thermal hydraulic system code of CEA
ATHLET	Analysis of thermal hydraulics of breaks and transients
CFD	Computational fluid dynamics
MCP	Main coolant pump
BOC	Beginning of cycle
PZR	Pressurizer
PWR	Pressurized water reactor
KNPP	Kozloduy nuclear power plant
RA	Reflector assembly
FA	Fuel assembly
NEA	Nuclear energy agency
OECD	Organization for economic cooperation and development
US NRC	U. S. Nuclear Regulatory Commission

1 Introduction

The Institute of Reactor Safety is involved in the qualification of best-estimate coupled code systems for reactor safety evaluations since it is a key step toward improving their prediction capability and acceptability. In the frame of the VVER-1000 Coolant Transient Benchmark Phase1 the coupled code RELAP5/PARCS has been extensively assessed. The Phase 2 of this benchmark is focused on both multidimensional thermal hydraulics phenomena within the reactor pressure vessel (RPV) such as coolant mixing and core physics [Kolev04]. Hence it is an excellent opportunity to qualify the prediction capability of the new 3D thermal hydraulic model of TRACE (VESSEL component) taking into account plant data obtained in the Kozloduy nuclear power plant (KNPP) unit 6. The "heat-up test" performed at the Kozloduy plant by closing the steam isolation valve of the loop 1 when the plant was operated at low thermal power (281 MWth) is mainly focused on the coolant mixing phenomena within the RPV.

The main reason for these validation activities is the increasing need in the nuclear community for the use of multidimensional thermal hydraulics models to describe more accurately expected plant conditions during off-set plant situations within the primary circuit and also within the reactor pressure vessel (RPV). Typical transient where a multidimensional approach is required are e.g. pressurized thermal shocks (PTS), deboration transients, main steam line breaks, anticipated transient without scram, etc. Parallel to the increasing application of computational fluid dynamics codes like FLUENT, CFX, STAR-CD, etc. other thermal hydraulic system codes like RELAP-3D, CATHARE-3D, ATHLET-FLUBOX, TRAC-P, TRAC-B, TRACE, etc. includes three-dimensional (3D) based on "coarse mesh finite volume" approach at least for the multidimensional treatment of the RPV including the core. Knowing that the CFD codes are currently very CPU-time consuming for large problems, it is worth to evaluate the prediction capability of the 3D coarse mesh models implemented in system codes like TRACE [TraceMa07].

The heat-up test is characterized by the heat-up of the primary coolant of loop-1 caused by the closure of the Main Steam Isolation Valve (SIV-4) of the secondary loop 1 of the KNPP due to the degraded heat removal over the affected steam generator. Under such conditions, the hotter fluid of the loop-1 get mixed in the downcomer with the one of the neighbouring loops. Since the arrangement of the loops of the VVER-1000 reactor is not symmetrical, the resulting mixing pattern is complex. In this report the peculiarities of the VVER-1000 reactor and the facing model challenging are given first. Then the developed 3D model using the VESSEL component of TRACE is presented. The main results obtained with TRACE are compared to the test data and discussed in detail. Finally conclusions are drawn and the further investigations are outlined.

2 Peculiarities of the VVER-1000 RPV

The Koloduy NPP is a Russian design VVER-1000 reactor of type W320 with a thermal power of 3000 MW_{th} and located in Bulgaria. The plant consists of four loops, each one with a horizontal steam generator (SG) and a main coolant pump (MCP) [Kolev04]. Details of the primary loops arrangement are shown in **Figure 1**. It can be seen there that the loops are not symmetrically arranged. The horizontal steam generator is characterized by a large water inventory on the secondary side compared to western-designed vertical steam generators. The reactor pressure vessel design differs also from that of western PWR, especially due to the constructive peculiarities in the lower and upper plenum that strongly impacts the flow patterns during normal and accidental situations. In **Figure 2** a vertical cut through the reactor pressure vessel (RPV) is given. The lower plenum consists of an elliptical cone with many perforations that result in a narrowing gap in direction of the central RPV-point. The diameter of the inlet and outlet nozzles amounts 850 mm while the inlets of the safety injection amount 280 mm. In addition 163 support columns are present in the lower plenum. The lower part is a full slab while the upper part is a tube with perforated walls (perforations with different size). Hence the flow coming from the downcomer has to pass through very complex flow paths to enter into the core. In the upper plenum, two concentric cylinders with perforations are present, where the lower part of the outer cylinder is conic.

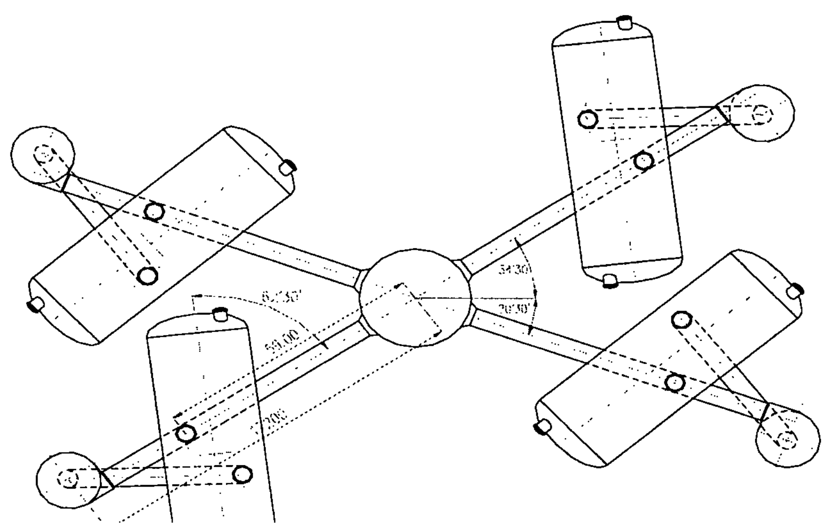


Figure 1 Horizontal arrangement of the primary loops of the VVER-1000 plant

The core consists of 163 fuel assemblies (FA) and 48 reflector assemblies (RA), **Figure 3**, each one with 312 fuel pins and one water rod. The main data about the FA and fuel rod design are given in **Table 1**. The fuel pins are arranged in a triangle within the FA, where the central position is occupied by an instrumentation rod. In addition the fuel pins have a central hole of around 1.4 mm diameter while the western-type pins are a full slab.

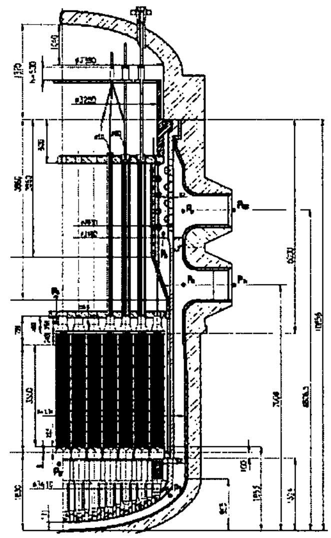


Figure 2 Vertical cut through the reactor pressure vessel of the VVER-1000 plant

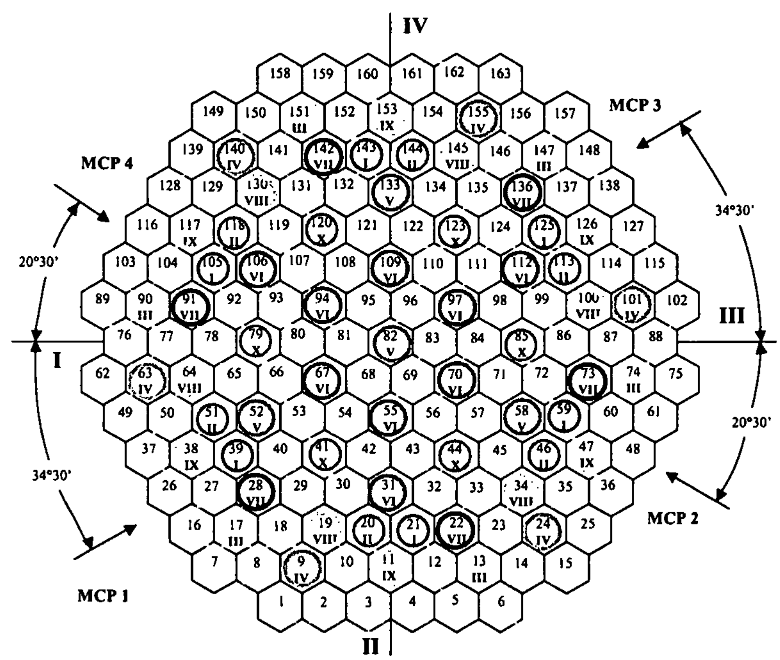


Figure 3 Core configuration with the position of the different control rod groups

Table 1: Dimensions of the fuel rod and fuel assembly of the VVER-1000

Parameter	Value
Pellet diameter, mm	7.56
Central void diameter, mm	1.4
Clad diameter (outside), mm	9.1
Clad wall thickness, mm	0.69
Fuel rod total length, mm	3837
Fuel rod active length (cold state), mm	3530
Fuel rod active length (hot state), mm	3550
Fuel rod pitch, mm	12.75
Fuel rod grid	Triangular
Number of guide tubes	18
Guide tube diameter (outside), mm	12.6
Guide tube diameter (inside), mm	11.0
Number of fuel pins	312
Number of water rods/assembly	1
Water rod diameter (outside), mm	11.2
Water rod diameter (inside), mm	9.6
FA wrench size, mm	234
FA pitch, mm	236

The peculiar constructive design of the VVER-1000 RPV internals represents a real challenge for the development of a 3D RPV model including the constructive details. The most challenging aspects are summarised here:

(1) Lower plenum: Radial core barrel elliptical bottom, support columns (solid hallow with perforation of different size)
(2) Core design: Fuel assemblies and fuel pin arrangement
(3) Upper grid plate with upper perforated fuel assembly head
(4) Upper plenum: inner perforated cylinder (lower part: conic) and outer perforated cylinder

In Figure 4 the complex flow path along the lower plenum is depicted. There the coolant has to pass first through the 163 holes of the core barrel elliptical bottom. Then it flows upwards along the support columns and enters through the perforated support columns upper part. Finally it flows through the lower core support plate into the fuel assembly.

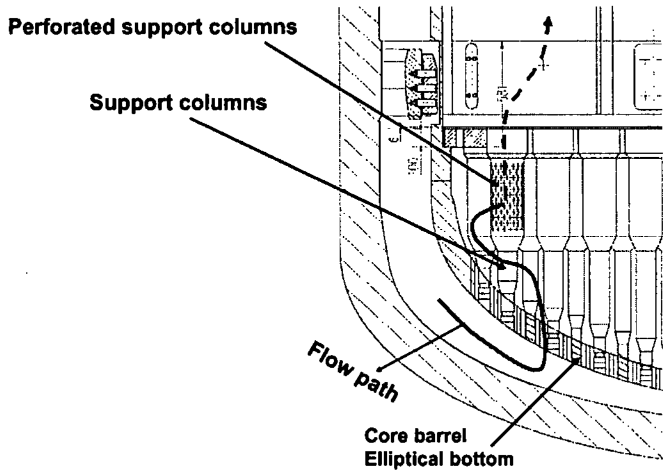


Figure 4 Constructive details of the lower plenum and complex flow path

A further challenge is the fuel pin arrangement in the core regarding the azimuthal and radial nodalization. Assumptions and engineering judgement hast to be made to estimate the main thermal hydraulic parameters at the faces, Figure 5. Especially the prediction of the following input deck parameters is crucial: (1) cell face fraction through for fluid flow (2) hydraulic diameter and (3) additive friction loss coefficients.

Furthermore the peculiarities of the upper end of the fuel assembly and the upper grid plate, see Figure 6, result in complicated flow paths through the perforated cone part of the fuel assembly head and finally through the upper core support plate. Finally the upper plenum is characterized by two concentric cylinders with a lot of perforations, through which the coolant has to pass, Figure 7. The presence of the guide tubes makes the flow more complex. The coolant leaving the core flows through the perforated part of the inner ring and then through the perforated core barrel.

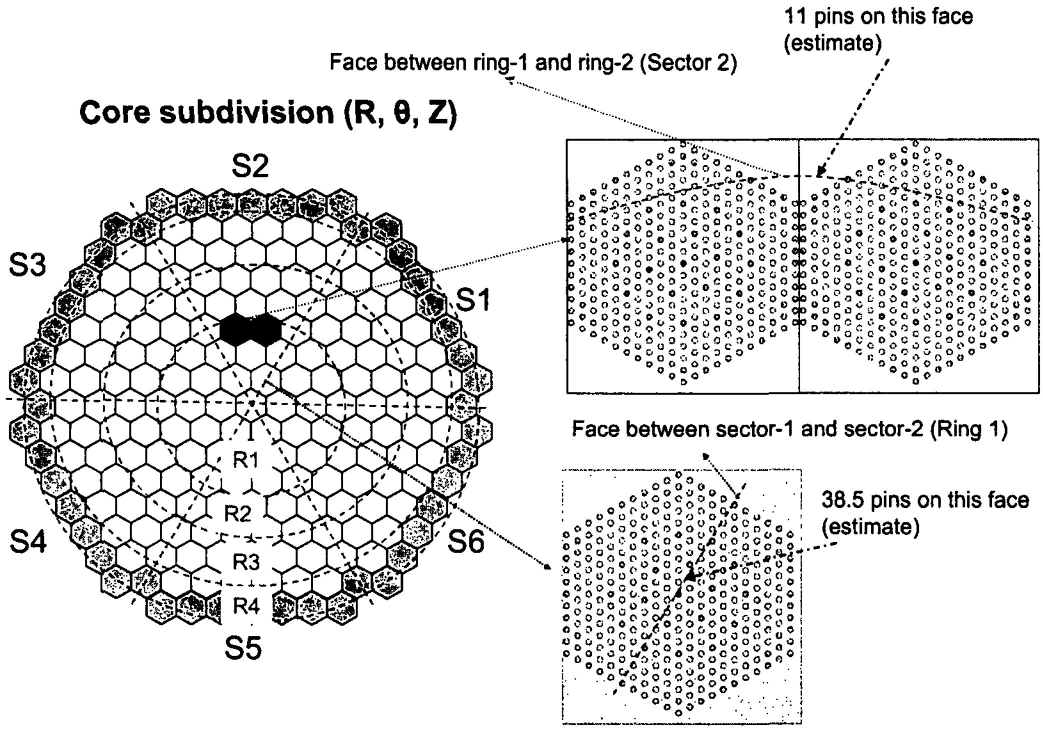


Figure 5 Pin arrangement regarding the nodalisation lines in radial and azimuthal direction

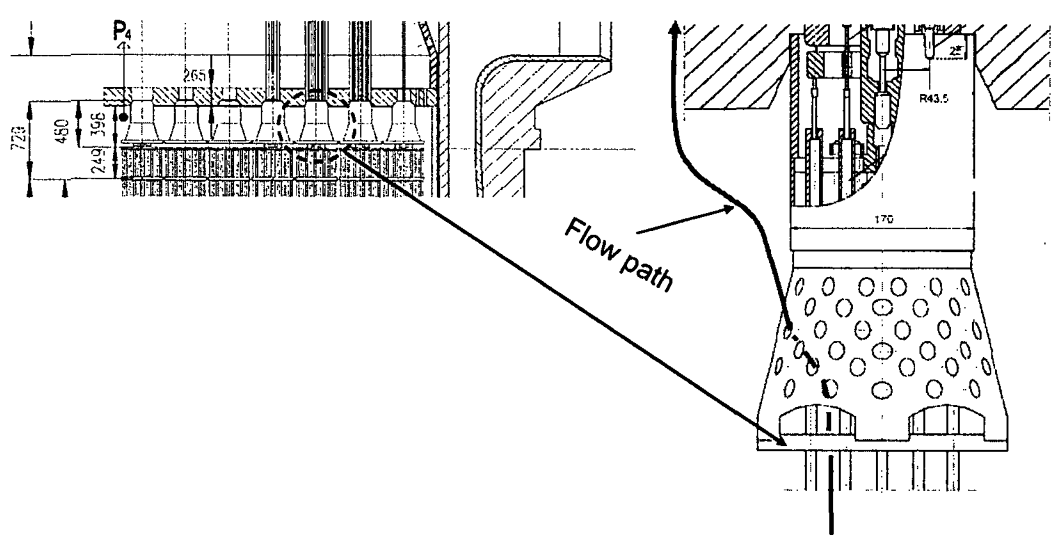


Figure 6 Flow conditions at the fuel assembly head and upper grid plate

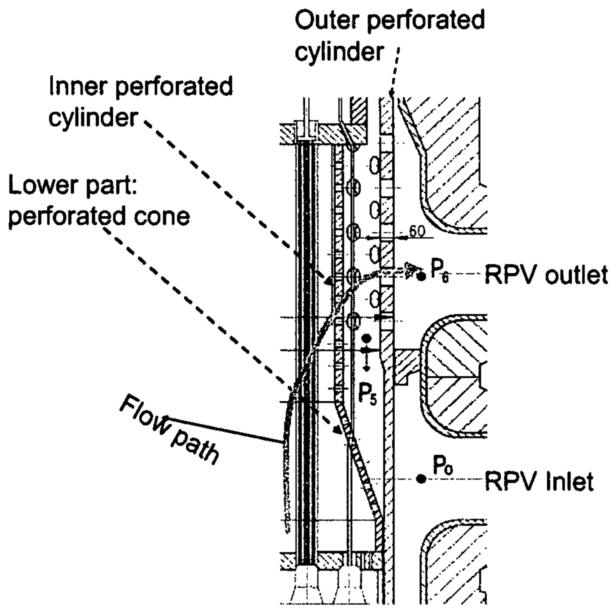


Figure 7 Vertical arrangement of the VVER-1000 primary components

3 Test description

3.1 Pre-test phase

Before the test, the nuclear power plant Kozloduy was operated at around 9.36 % of the nominal power i.e. 281 MWth with all main coolant pumps running. The main operational parameters are summarized in **Table 2** where also the measurement accuracy is summarized. On the secondary side all steam generators were available. The core was loaded with fresh fuel i.e. at beginning of cycle conditions (BOC) with a core averaged exposure of 0.4 effective full power days (EFPD) and a boron concentration of 7.2 g/kg (the value of 7.5 g/kg that corresponds to a zero moderator temperature coefficient). Hence the feedbacks between the core neutronic and the thermal hydraulic conditions are rather negligible. The position of the control rod groups were as follows: group #9 and #10: fully inserted; groups #1-#7: fully withdrawn and the regulating rod group #8 was about 84% withdrawn from the bottom of the core. The coolant temperature at core inlet was 20 K lower that the one at nominal conditions. Finally the steam generator levels were as high as the ones at nominal conditions. The main steam header pressure amounts 5.07 MPa, about 1.0MPa lower than the nominal value.

Table 2: Main parameters of the four loops before the test

Parameter	Initial State	Accuracy
Thermal power, MW	281	± 60
Pressure above core, MPa	15,593	± 0,3
Pressure drop over RPV, MPa	0,418	± 0,043
Coolant temperature at core inlet #1, K	541,75	± 1,5
Coolant temperature at core inlet #2, K	541,85	± 1,5
Coolant temperature at core inlet #3, K	541,75	± 1,5
Coolant temperature at core inlet #4, K	541,75	± 1,5
Coolant temperature at core outlet #1, K	545	± 2,0
Coolant temperature at core outlet #2, K	545	± 2,0
Coolant temperature at core outlet #3, K	544,9	± 2,0
Coolant temperature at core outlet #4, K	545	± 2,0
Mass flow rate of loop #1, kg/s	4737	± 110
Mass flow rate of loop #2, kg/s	4718	± 110
Mass flow rate of loop #3, kg/s	4682	± 110
Mass flow rate of loop #1, kg/s	4834	± 110

3.2 Test phase

The test was performed in 1991 at the Kozloduy NPP. It was initiated by the isolation of the steam generator of loop-1 due to the closure of the main steam isolation valve [Kolev04] and isolation of the steam generator from feed water. As a consequence, the primary coolant temperature of loop-1 started to increase up to about 14 °C compared to the coolant temperature of the other loops. Under such conditions a coolant mixing occurred first of all in the downcomer region. The resulting mixing pattern propagates through the lower plenum, core and upper plenum. Since the power was relatively low, the feedbacks between thermal hydraulics and core neutronics are negligible according to the recorded data. Due to the mixing, the temperature of the unaffected loops, especially of the loop closer to the loop-1 (loop-2) increased too. The test lasted for 1800 s. At that time the power increased only up to 286 MW. Different data was recorded at the Kozloduy plant during the test. The coolant temperature at the cold/hot legs was measured with one thermal resistor at the level of pipe axis and two thermocouples in the lower part of the flow section. At some fuel assembly positions the coolant temperature at the core outlet was measured too. Measured fuel outlet temperatures and experimental determined mixing coefficients from cold legs to fuel assembly outlets were also measured for the qualification of CFD-codes. From this data the fuel assembly inlet temperatures was derived assuming that the relative temperature rise distribution does not change during the transient [Kolev04].

In Figure 8 the recorded data of the four hot legs is given for the whole test i.e. 1800 sec. There it can be seen that the coolant temperature of the affected loop-1 starts to increase very rapidly at around 130 sec. due to the deteriorated heat transfer over the steam generator-1. From 500 s onward the increase rate becomes smaller stabilizing at a value below 556 K. Due to the coolant mixing in the downcomer the temperature of the loop-2 experienced a higher temperature than the one of the loop-4 indicating that the mixing pattern is not in clock-wise direction. Note that the position of the loops is not symmetrical, see Figure 3. The core parameters at the end of the test (at 1800 sec.) are given in **Table 3**.

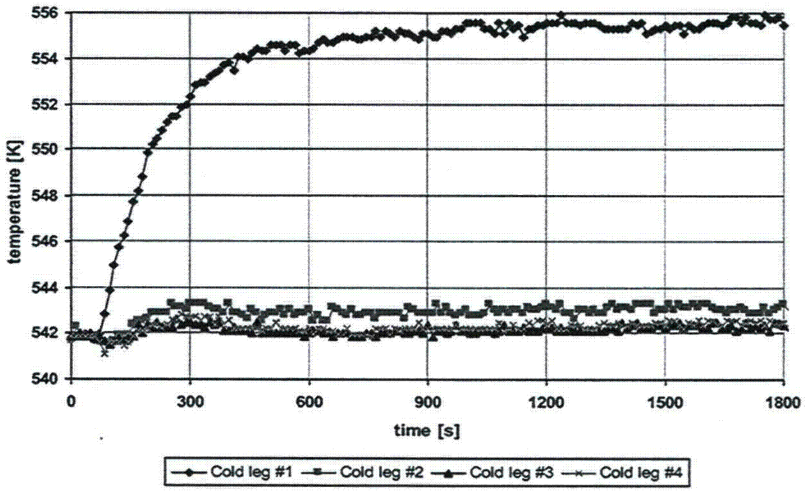


Figure 8 Measured evolution of the hot legs during the test at the KNPP

Table 3: Main parameters of the NPP at the end of the test (1800 sec)

Parameter	Final State	Accuracy
Thermal power, MW	286	± 60
Pressure above core, MPa	15,593	± 0,3
Pressure drop over RPV, MPa	0,417	± 0,043
Coolant temperature at core inlet #1, K	555,35	± 1,5
Coolant temperature at core inlet #2, K	543,05	± 1,5
Coolant temperature at core inlet #3, K	542,15	± 1,5
Coolant temperature at core inlet #4, K	542,35	± 1,5
Coolant temperature at core outlet #1, K	554,85	± 2,0
Coolant temperature at core outlet #2, K	548,55	± 2,0
Coolant temperature at core outlet #3, K	545,75	± 2,0
Coolant temperature at core outlet #4, K	546,45	± 2,0
Mass flow rate of loop #1, kg/s	4566	± 110
Mass flow rate of loop #2, kg/s	4676	± 110
Mass flow rate of loop #3, kg/s	4669	± 110
Mass flow rate of loop #1, kg/s	4816	± 110

4 Short description of the used code TRACE

The system code TRACE (TRAC/RELAP Advanced Computational Engine) is being developed by Los Alamos National Laboratory (LANL) and the Pennsylvania State University as a reference tool for the US NRC [TraceMa07]. It should unify the simulation capabilities of well know code system like TRAC-P and –B, RAMONA and RELAP5. TRACE is characterized by a modern modular conception linked to a powerful pre-and post-processing software –called SNAP (Symbolic Nuclear Plant Analyzer). The fluid dynamics model consists of a set of two-fluid models in one and three-dimensions coupled to a one and two-dimensional heat conduction model for structures (with/without heat source) e.g. fuel pins, pipe walls, etc. The heat transfer package includes not only vertical but also a horizontal flow regime for all relevant flow regimes that can be expected during the normal and accidental conditions of nuclear reactors. TRACE is especially developed to investigate any kind of operational events, transients and design basis accidents of both Boiling water (BWR), Pressurized water reactors (PWR) and innovative reactor systems. Hence not only water but also liquid metals and gases are included as working fluids. In addition, it contains not only a point kinetics model based on the Kaganove-approach but also a powerful three dimensional core reactor simulator called PARCS. The coupled system TRACE/PARCS is a powerful tool appropriated for the simulation of such transients where a strong power distortion within the core exist and where the thermal hydraulic core behaviour is strongly related to the core neutronics like in the case of MSLB, ATWS, boron dilution, etc.

The module PARCS [Joo02] includes 3D neutronic solvers for both square and hexagonal fuel assemblies for static and time dependent solutions in diffusion or transport approximation using multigroup cross sections. To take into account for feedbacks mechanisms different schemes for the online-update of cross-sections during the transient calculations are present.

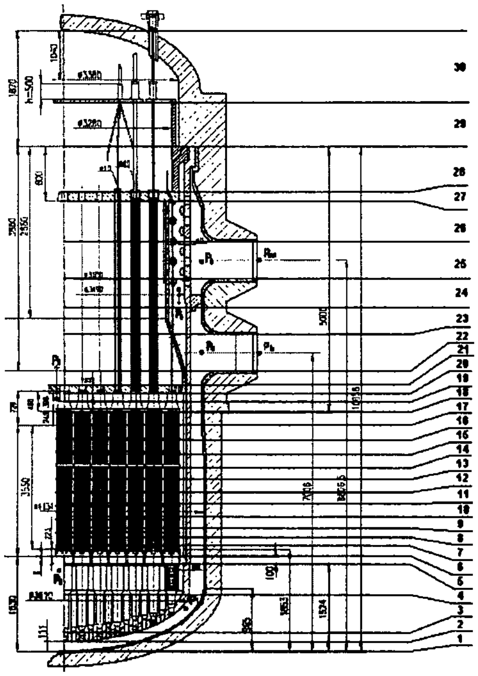
5 Development of a 3D model for the RPV

For the foreseen investigations, detailed models of the RPV including the core were developed for TRACE including e.g. the downcomer, lower plenum, core, core outlet, upper plenum and RPV-inlet and outlet pipes. A detailed description of this model can be read in [Jaeger06]. The 3D VESSEL component of TRACE was used for the representation of the RPV. According to this, the whole RPV is subdivided in 30 axial, six radial and six azimuthal direction, Figure 9 and Figure 10. The size of the respective nodes depends of the existing flow conditions along the main flow paths within the RPV determined by the constructive peculiarities of the RPV-internals. From the 30 axial elevations of the RPV 10 axial nodes belong to the core region while two to the lower and upper axial reflector. The azimuthal sectors (S1 to S6, Figure 10) were defined so that the cold legs are connected to sector 4 (cold leg-1), sector 6 (cold leg-2) sector 1 (cold leg 3) and sector 3 (cold leg -3).

In radial direction also 6 rings are considered, three of them in the core region, Figure 10. For each of the 3D-volume elements the main thermal hydraulic parameters for each direction such as hydraulic diameter, flow area, heated diameter and form loss coefficients, etc. are derived from the detailed plant data. To catch the non-symmetrical coolant mixing expected to occur mainly in the downcomer and lower/upper plenum a rather fine nodalisation of the RPV in azimuthal and radial direction is needed. One has to keep in mind that the finer nodalization the higher the CPU. A reasonable compromise between accuracy and CPU-cost is here mandatory. In Figure 11 the radial and azimuthal nodalization of the core is shown. In developing the 3D model using the VESSEL component the following aspects had to be kept in mind:

- Make use of geometrical symmetry (R, θ, Z),
- Select the size of cells (radial, axial, angular) as small as necessary (based on underlying physics), and
- Consider the details of flow paths as much as necessary.

Otherwise the 3D model may become unnecessary complex.



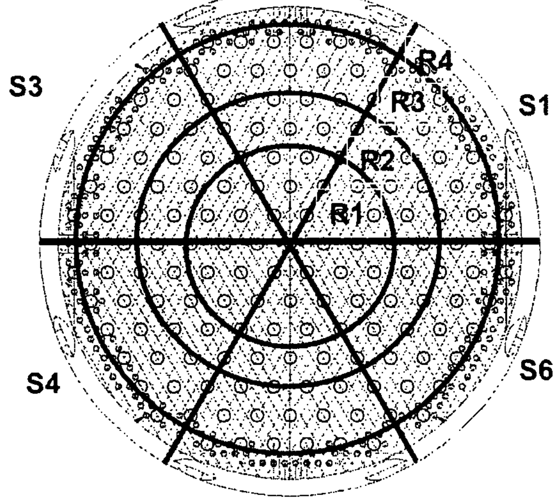
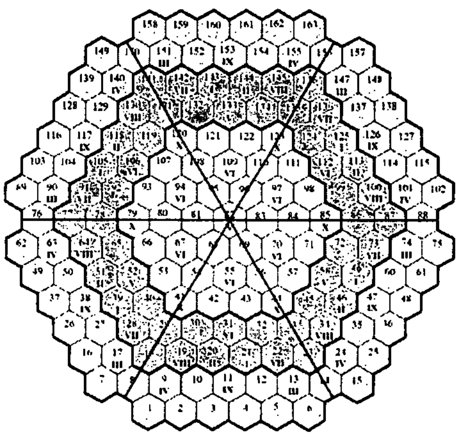


Figure 9 TRACE axial nodalization of the RPV

Figure 10 TRACE radial and azimutal subdivision of the core



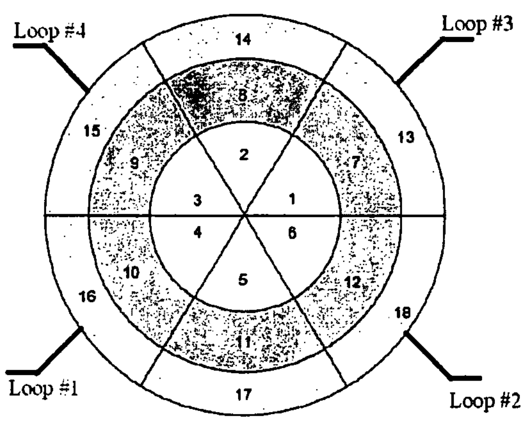


Figure 11 TRACE Nodalisation of the core and relative position of the cold/hot legs

The complete TRACE model as represented by SNAP (Pre- and Postprocessor) is depicted in **Figure 12**. Part of the hot and cold legs as well as the mass source and sinks are represented with pipes and FILL and BREAK components. They are necessary to define the initials and boundary conditions of the problem being investigated. These boundary conditions are coolant temperature of the loops, system pressure and loops mass flow rate.

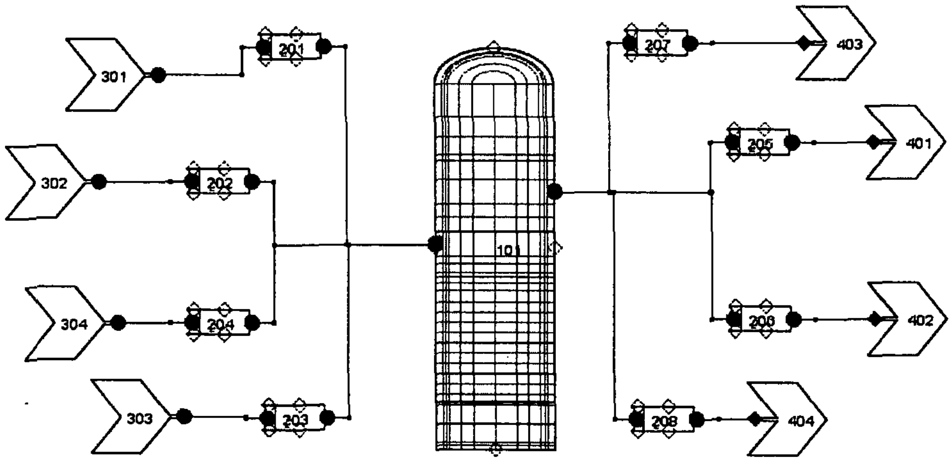


Figure 12 TRACE 3D Nodalisation of the RPV with cold/hot legs

6 Simulation of the heat-up experiment

The TRACE post-test calculations of the coolant mixing experiment have been performed in two steps. First of all a steady state calculation was carried out to predict the plant conditions just before the test. Secondly a transient run was made for the 1800 sec. test duration in order to determine the final state of the plant. The time dependent given boundary conditions e.g. loops flow rate, coolant temperature of the cold legs and the system pressure were defined in the Benchmark Specifications.

6.1 Prediction of the initial plant state

In **Table 4** a comparison of the TRACE-predictions with the plant data is given for the initial plant state is exhibit. It can be seen that the agreement between data and prediction is quite good. At the initial state the coolant temperature at the core inlet/outlet is uniformly since all pumps and steam generators are in operation, **Figure 13** and **Figure 14**. This will change drastically during the heat-up test.

Table 4: Comparison of TRACE predictions with plant data for the initial state

Parameter	Initial State	Accuracy	TRACE	Deviation (%)
Thermal power, MW	281	± 60	281	0,0000
Pressure above core, MPa	15,593	± 0,3	15,592	0,0064
Pressure drop over RPV, MPa	0,418	± 0,043	0,404	3,3493
Coolant temperature at core inlet #1, K	541,75	± 1,5	541,78	-0,0055
Coolant temperature at core inlet #2, K	541,85	± 1,5	541,88	-0,0055
Coolant temperature at core inlet #3, K	541,75	± 1,5	541,78	-0,0055
Coolant temperature at core inlet #4, K	541,75	± 1,5	541,78	-0,0055
Coolant temperature at core outlet #1, K	545	± 2,0	544,63	0,0679
Coolant temperature at core outlet #2, K	545	± 2,0	544,7	0,0550
Coolant temperature at core outlet #3, K	544,9	± 2,0	544,61	0,0532
Coolant temperature at core outlet #4, K	545	± 2,0	544,62	0,0697
Mass flow rate of loop #1, kg/s	4737	± 110	4749	-0,2533
Mass flow rate of loop #2, kg/s	4718	± 110	4735	-0,3603
Mass flow rate of loop #3, kg/s	4682	± 110	4750	-1,4524
Mass flow rate of loop #1, kg/s	4834	± 110	4737	2,007

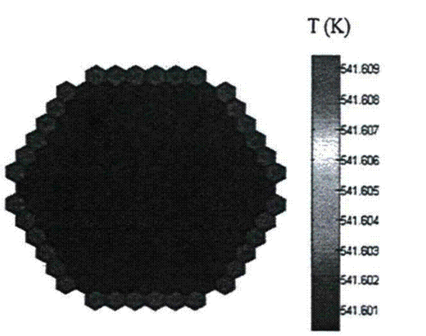


Figure 13 Predicted coolant temperature at core inlet

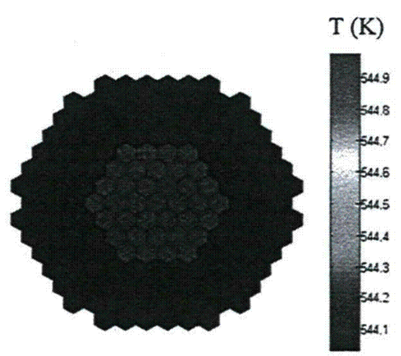


Figure 14 Predicted coolant temperature at core outlet

6.2 Predicted final plant state

The transient phase started with the isolation of the main steam isolation valve and lasted for 1800 sec. The final plant state predicted by TRACE is compared to the plant data in **Table 5.** There can be observed that the code predictions are close to the plant data. In addition to the hot/cold leg temperatures also the pressure drop is in good agreement with the data. Since during the test the hot leg temperature of the loop-1, see **Figure 15**, was continuously increasing while the one of the other loops were not, a considerable macroscopic coolant mixing took place in the downcomer. The predicted temperature in the six sectors of the downcomer are shown in **Figure 16**, where the increase of the temperature in sector two and three can be noted due to the mixing process. It is worth to mention that the mixing took place counter clock-wise direction.

Table 5: Comparison of TRACE predictions with plant data for the final state

Parameter	Final State	Accuracy	TRACE	Deviation (%)
Thermal power, MW	286	± 60	286	0
Pressure above core, MPa	15,593	± 0,3	15,591	0,013
Pressure drop over RPV, MPa	0,417	± 0,043	0,404	3,118
Coolant temperature at core inlet #1, K	555,35	± 1,5	555,39	-0,007
Coolant temperature at core inlet #2, K	543,05	± 1,5	543,08	-0,006
Coolant temperature at core inlet #3, K	542,15	± 1,5	542,18	-0,006
Coolant temperature at core inlet #4, K	542,35	± 1,5	542,38	-0,006
Coolant temperature at core outlet #1, K	554,85	± 2,0	555,14	-0,052
Coolant temperature at core outlet #2, K	548,55	± 2,0	548,66	-0,020
Coolant temperature at core outlet #3, K	545,75	± 2,0	545,44	0,057
Coolant temperature at core outlet #4, K	546,45	± 2,0	545,69	0,139
Mass flow rate of loop #1, kg/s	4566	± 110	4657	-1,993
Mass flow rate of loop #2, kg/s	4676	± 110	4693	-0,364
Mass flow rate of loop #3, kg/s	4669	± 110	4724	-1,178
Mass flow rate of loop #1, kg/s	4816	± 110	4724	1,910

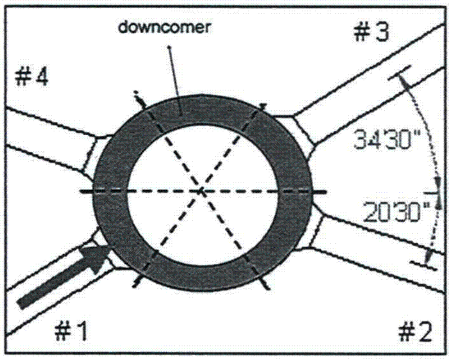


Figure 15 Location of the loops with respect to the downcomer

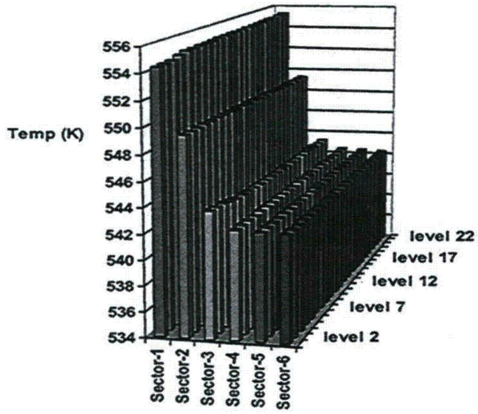


Figure 16 Predicted coolant temperature in the sectors of the downcomer (levels:2 to 22)

The predicted coolant temperature of each fuel assembly at the core outlet is compared in **Figure 17** and **Figure 18** for the initial and final state. Specially in **Figure 18** the mixing pattern within the core can be observed. The hotter fluid of the loop-1 get mixed with the one of the sector between the loop-1 and loop-2, i.e. in counter-clock-wise direction as observed in the tests.

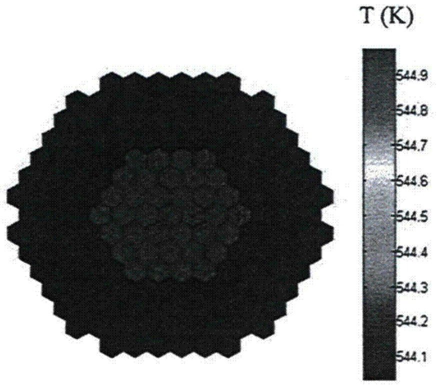


Figure 17 Predicted coolant temperature at the core outlet at initial state (0 sec)

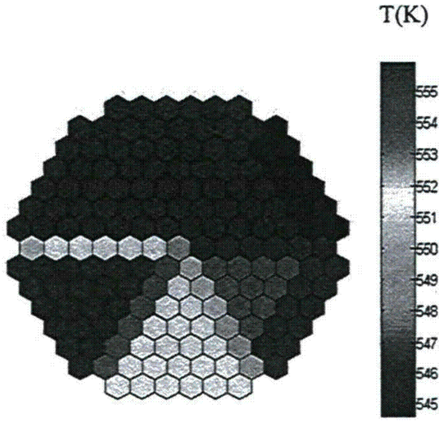


Figure 18 Predicted coolant temperature at core outlet at final state (1800 sec)

The predicted coolant temperature at the fuel assembly outlet is compared with the derived data for the initial state in **Figure 19**. The agreement is reasonable taking into account the measurement error and the large sectors of the core where only averaged values of the coolant temperature are predicted in each sector. To catch the local gradients of the coolant temperature that exist due to the radial burnup of the fuel assemblies within the core, a refinement of the radial and azimuthal discretitation were necessary. This kind of analysis can be done preferably with CFD or subchannel codes.

In **Figure 20** a comparison of the measured coolant temperature at the fuel assembly outlet with the predicted values by TRACE is given. It can be seen that the TRACE-predictions follows qualitatively the trend of the measured data. In some positions TRACE tends to over predict and in others to under predict the data. But the differences between data and predictions there are within the measurement error. These trends are comparable to the trends predicted by CFX-5 [Sanchez07].

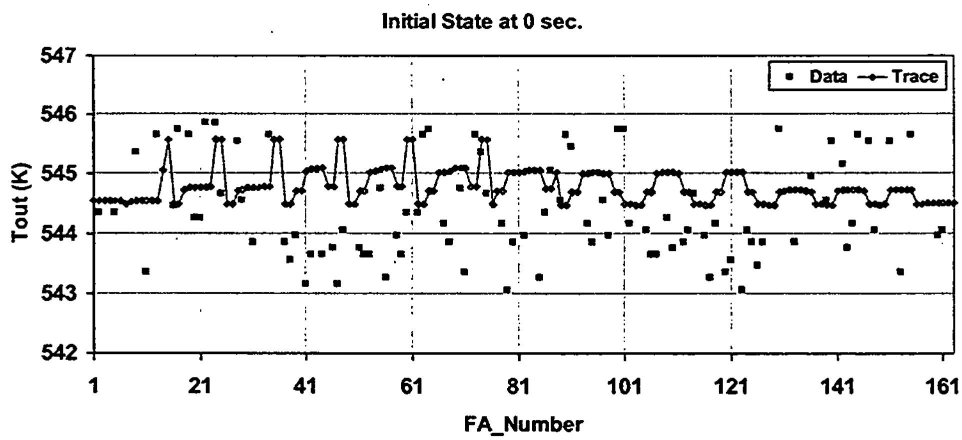


Figure 19 Comparison of the predicted coolant temperature at FA-outlet with data

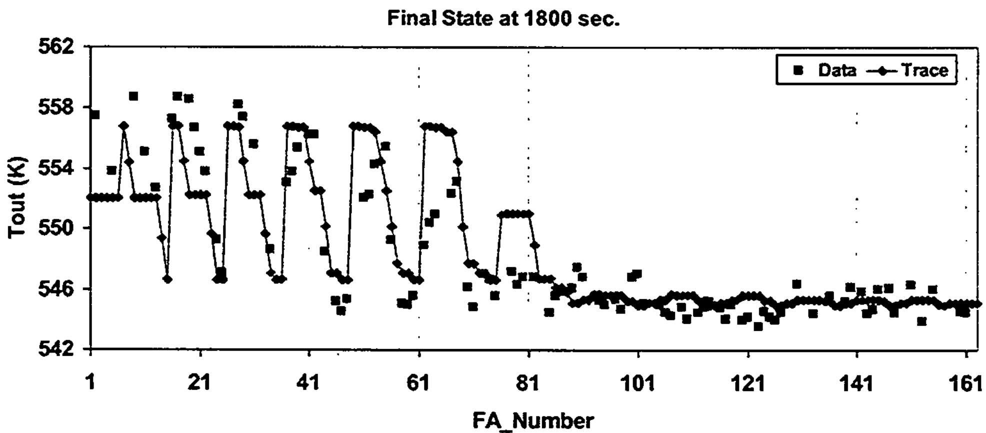


Figure 20 Comparison of the predicted coolant temperature at FA-outlet with data

6.3 Global parameters

In addition to the measured data for the initial and final state of the core and RPV conditions, there is also data recorded during the test duration. These data was derived from the thermocouples located at the hot legs and taking into account the different ways of power measurements applied at the nuclear power plant. According to the re-evaluation of the data by the plant and benchmark team, [Kolev06], the evolution of the temperature of the hot leg-2 hast to be corrected to a final value of 548.55 K as given in **Table 3**. This has to be kept in mind comparing the time evolution of the hot leg temperature of loop-2 hereafter.

The coolant temperature of the hot legs as derived from the data is compared with the predictions of the TRACE simulation in **Figure 21** to **Figure 24**. It can be seen that the predictions are in very good agreement with the experimental data. Apparently this is not the case for the loop-2. But -as mentioned earlier- re-evaluations of this experiment performed in [Kolev06] lead to the conclusion that the measured value for the loop-2 should be around 548 K instead of 545 K. Considering the measurement error for the coolant temperature are the predictions very reasonable.

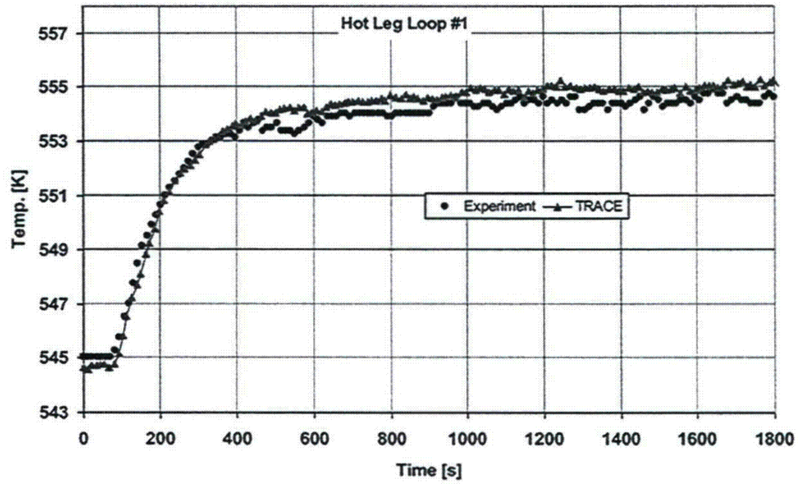


Figure 21 Comparison of predicted hot leg with data the for loop-1

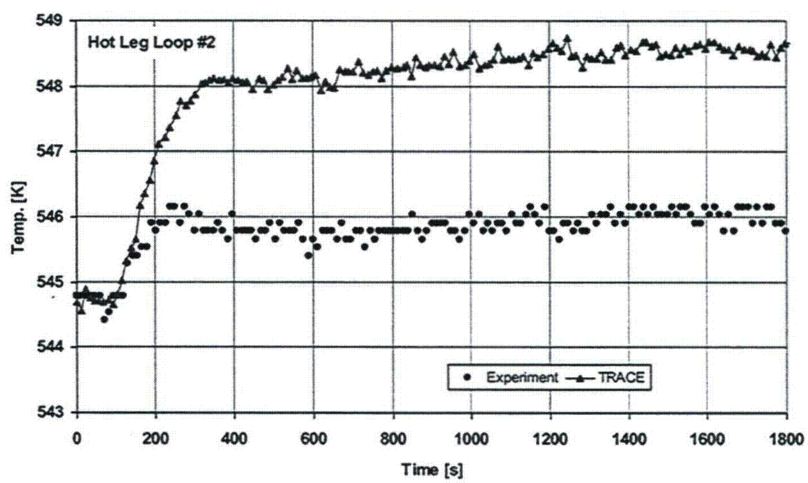


Figure 22 Comparison of predicted hot leg with data the for loop-2

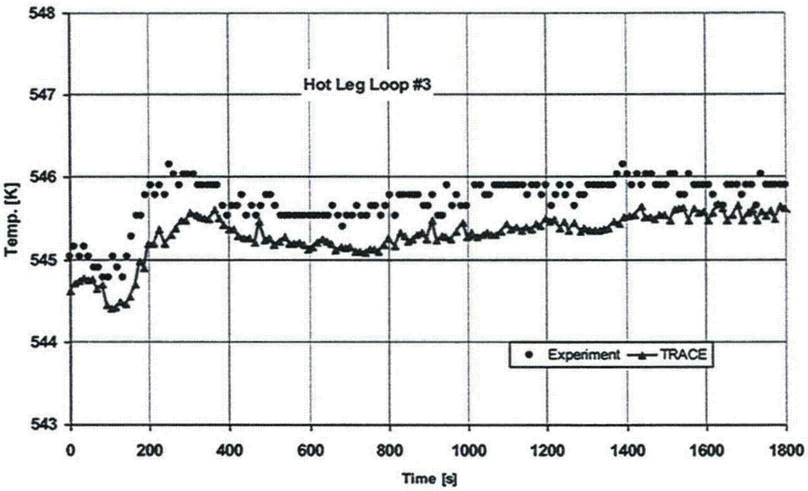


Figure 23 Comparison of predicted hot leg with data the for loop-3

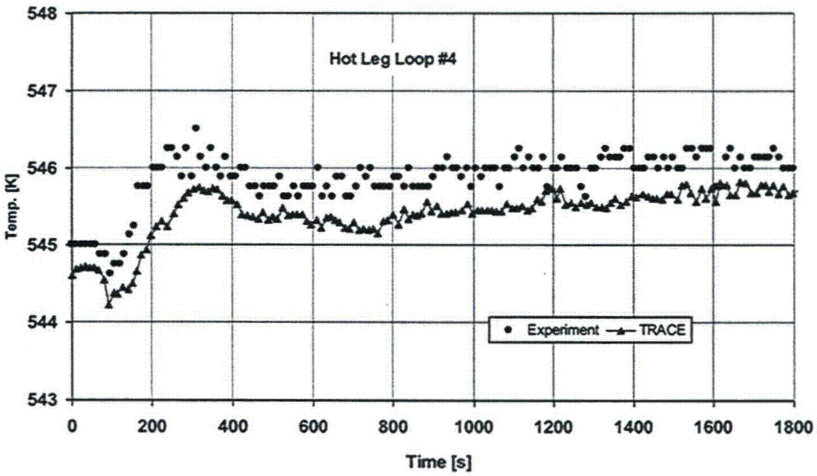


Figure 24 Comparison of predicted hot leg with data the for loop-4

7 Conclusions

In this report investigations performed to validate the 3D-thermal hydraulic model of TRACE based on plant data (Kozloduy nuclear power plant) are presented. The main issue was the coolant mixing within the RPV, especially in the downcomer, caused by the isolation of the steam generator of loop-1 while the others were working at nominal conditions. Details of the developed 3D thermal hydraulics model for TRACE using the VESSEL component were presented extensively. The constructive peculiarities of the RPV of importance for the elaboration of the 3D models are also outlined. The test conduction and initial and boundary conditions are also briefly presented.

From the comparison of the calculated parameters by TRACE with the available measurement data the following conclusions can be drawn:

- The initial plant conditions just before the test were predicted by TRACE in a very good agreement with the plant data,
- The final plant state predicted by TRACE is close to the plant data and the deviations are within the measurement error band,
- The evolution of important plant parameter predicted by TRACE follows nicely the measured trends indicating that the mixing within the RPV is well described by simulations (hot leg temperature of all loops),
- A detailed comparison of the calculated coolant temperature at the core outlet for each fuel assembly position with available data showed a good trends, and
- TRACE was also able to predict the counter-clock-wise rotation i.e. the mixing preferably in direction of loop-2 and -3 instead of loop-4.

In general it can be stated that the chosen nodalisation scheme (30 axial, six radial and six azimuthal nodes) seems to be appropriate to catch the underling physics of the coolant mixing process within the RPV of VVER-1000 reactors.

These results are very encouraging and underline the capabilities of the 3D VESSEL component of TRACE which is very flexible allowing simulations in from 1D, 2D and 3D geometry. Consequently the validated 3D model of the RPV of the VVER-1000 reactor can be used to investigate transients where the coolant mixing is a key issue such as boron dilution, main steam line break, etc.

For the simulation of plant transients with coupled thermal hydraulics/neutron kinetics codes a well validated 3D thermal hydraulic model is an important prerequisite.

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10. SUPPLEMENTARY NOTES
A. Calvo, NRC Project Manager

11. ABSTRACT *(200 words or less)*

In this report the investigations performed to validate the 3D thermal hydraulic model of TRACE using data gained in the nuclear power plant Kozloduy Unit 6 regarding the coolant mixing within the reactor pressure vessel will be presented. These data were distributed to the scientific community in the frame of the VVER-1000 Coolant Transient Benchmark Phase 2. The measured data was recorded during the non-symmetrical core heat-up test caused by the closure of the isolation valve of the steam line of the loop-1. Since plant data for code validation is rather scarce, this coolant mixing data is very much appropriate for the qualification of the 3D thermal hydraulic models of the TRACE code.

A detailed multidimensional model for the RPV of the VVER-1000 was elaborated using the 3D VESSEL component of TRACE. The complete model consisted of more than 1000 3D thermal hydraulic cells. Using this model a post test analysis of the heat-up test was performed with the TRACE version V4160 in a Linux cluster.

The obtained results for the initial and final state are in very good agreement with the plant data. TRACE needed not more than six minutes for the simulation of the whole test duration of 1800 sec. It was demonstrated that the chosen 3D-nodalization of the RPV is adequate for the description of the coolant mixing phenomena in a VVER-1000 reactor.

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