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Comparison of MSLB transient results using the 3D coupled code TRACEv5p05/PARCS and the system thermal hydraulic code RELAP5

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A R T I C L E I N F O

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Point kinetics
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A B S T R A C T

This paper presents a comparative analysis of the Main Steam Line Break (MSLB) in a VVER-1000 reactor simulated with RELAP5 using Point Kinetics and the coupled code TRACE5-P05/PARCS using 3D kinetics. In the MSLB-scenario, it is assumed that the main steam line break of 580 mm inner diameter is located between the steam generator (SG) and the steam isolation valve (SIV), outside the containment. In a MSLB, a non-symmetric overcooling of the primary coolant takes place leading to a positive reactivity insertion. Hence, the main safety concern is to assess if the core may become critical despite SCRAM and it there is a considerable power increase (return-to-power). This paper will discuss the capabilities of different computational approaches to simulate the VVER-1000 plant behaviour during a MSLB; one approach based on 1D thermal hydraulics and Point Kinetics while the other one based on 3D thermal hydraulics of the reactor pressure vessel (RPV) and 1D thermal hydraulics for the remaining plant components based on a 3D neutron kinetics model. The analyses are performed for Beginning of Cycle (BOC) conditions i.e., with a fresh core loading when the plant is operated at nominal power. The neutron kinetic parameters for the RELAP5 Point Kinetics model were generated PARCS for the BOC assuming a boron concentration of 1630 ppm. The respective 2 energy group homogenized cross section libraries in PMAXS-format were generated by KIT using the SERPENT2 code.

The investigations were performed in the frame of CAMIVVER-project, which focus was the assessment and development of reliable neutron physical and system thermal hydraulic models for safety evaluations of VVER-1000 reactors. The comparative analysis for the MSLB has shown that both applied codes are able to qualitatively predicts the plant behaviour under MSLB-conditions in similar manner. Differences are caused by the different approach to represent the core and RPV followed by RELAP5 and TRACE5.05/PARCS as expected.

1. Introduction

A large number of VVER-1000 and VVER-1200 are under operation and construction in EU and worldwide. They are characterized by horizontal steam generators and hexagonal fuel assemblies compared to the U-tube ones and square fuel assemblies of Western-type PWRs. An important objective of the EU Horizon 2020 project CAMIVVER (Verrier et al., 2019) was the development of reliable both neutron physical core models and system thermal hydraulic models of VVER-1000 reactors for the analysis of transients, where distortions of the core behaviour are expected to occur, such as may occur in a main steam line break (MSLB). It is design basis accident appropriate for evaluation of the simulation

Abbreviations: BAS, Bulgarian Academy of Sciences; BOC, Beginning of Cycle; BOL, Beginning of Life; BRU-K, Steam Dumping Device to the Condenser; BRU-A, Steam Dumping Device to the Atmosphere; BZOK, Fast Acting Isolation Valve; CHV, Check Valve; CVCS, Control Volume and Chemical System; DTC, Doppler Temperature Coefficient; EBIS, Emergency Boron Injection System; EECCS, Emergency Core cooling systems; EFW, Emergency Feed Water System; FA, Fuel Assembly; HA, Hydro accumulator; HFP, Hot Full Power; HPIS, High-Pressure Injection System; ID, Inside Diameter; INRNE, Institute for Nuclear Research and Nuclear Energy; KIT, Karlsruhe Institute of Technology; LPIS, Low Pressure Injection System; MCPs, Main Coolant Pumps; MSIV, Main Steam Isolating Valve; MSH, Main Steam Header; MSLB, Main Steam Line Break; MTC, Moderator Temperature Coefficient; NPP, Nuclear Power Plant; PRZ, Pressurizer; RPV, Reactor Pressure Vessel; SCRAM, Reactor Protection Signal; SG, Steam Generator; SIV, Steam Isolating Valve (Russian abbreviation BZOK #4); SV, Safety Valve; TPEN, Triangular Polynomial Expansion Nodal; VVER, Water-Water Energy Reactor; XS, Cross Sections.

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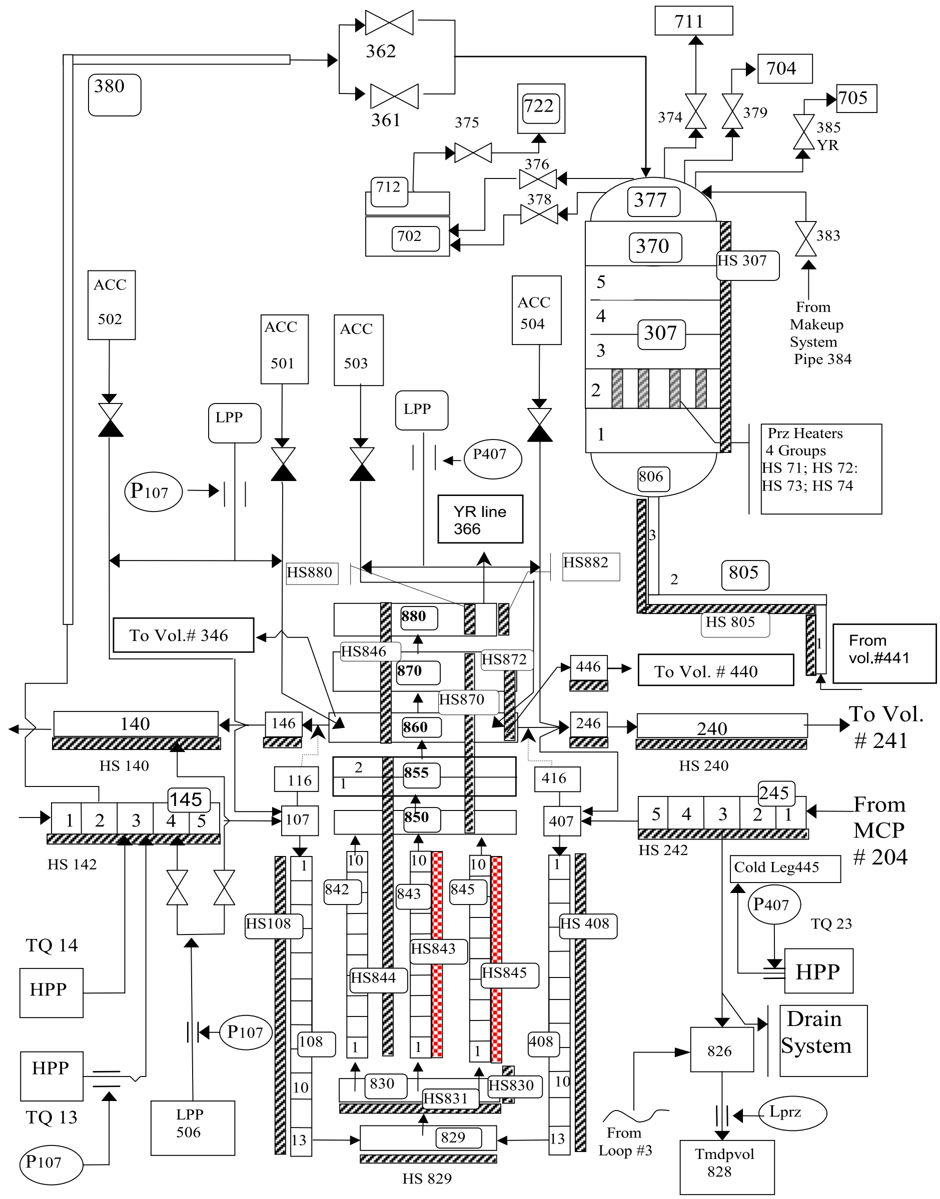

Fig. 1. Kozloduy VVER-1000 Reactor and Pressurizer RELAP5 Model.

capability of coupled thermal-hydraulics and neutronics codes. The scenario involves strong reactivity feedbacks from the coolant and fuel temperature. The MSLB-transient scenario defined in CAMIVVER is intended to assess the prediction capability of TRACE5.05/PARCS (USNRC, 2013) and RELAP5. Opposite to the OECD/NEA VVER-1000 MSLB benchmark (Kolev et al., 2006), where the MSLB is analysed for a burnt core loading (Cycle 6), the MSLB transient analysis here is performed for BOC-conditions (fresh core) with specific assumptions aimed to simplified the analysis. For the MSLB-analysis, integral plant models for TRACE5.05/PARCS and RELAP5 of the VVER-1000 Kozloduy Unit 6 were developed. In addition, homogenized and condensed cross sections for the BOC-core has been generated with Serpent2 for PARCS. Finally, the neutron kinetic parameters needed by the Point Kinetics model of RELAP5 code (NRC, 2010) such as reactivity coefficients, effective fraction of delay neutrons, prompt neutron lifetime, and the shut-down reactivity were generated wit static PARCS simulations. A consistent approach is provided for the comparison of the 3D core with point kinetics analysis. The plant data used for the development of the integral plant models were taken from (Stefanova et al., 2021).

In the Chapter 2, a brief description of the selected computer codes is provided. Then, Chapter 3 deals with the details of the developed models of the plant for TRACE5.05/PARCS and RELAP5. Chapter 4 describes the MSLB-scenario including the initial and boundary. Finally, Chapter 5 is focus on the detailed discussion of the comparative analysis of the simulation's results obtained with the two codes. Conclusions are given in Chapter 6.

2. Brief description of used computer codes

2.1. Short description of RELAP5

The RELAP5 code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents developed by Idaho National Laboratory (INEL) for the US NRC (Nuclear Safety Analysis Division, 2001). It solves a system of six conservation equations for mass, momentum and energy for one dimensional two-phase flow. A large number of heat transfer modes for horizontal and vertical flow regimes are implemented for the wall/fluid heat transfer and also for the interface between the liquid and vapor phases. The component models include pumps, valves, and pipes, heat

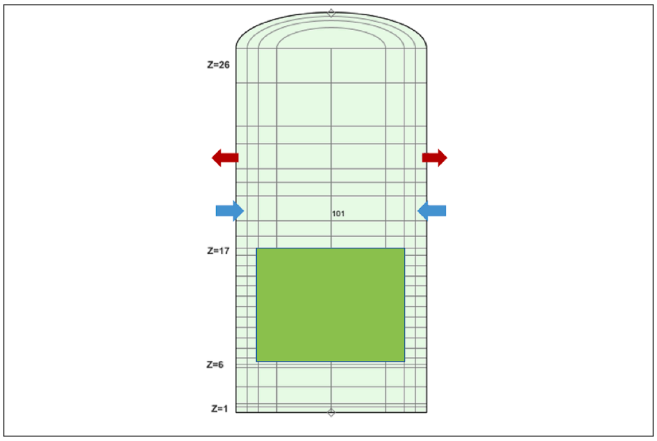

Fig. 2. 3D Reactor pressure vessel.

releasing or absorbing structures, reactor point kinetics, electric heaters, jet pumps, turbines, separators, accumulators, and control system components. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, boron tracking, and no condensable gas transport. In addition, it includes mathematical models for control systems, trips, signals, etc. allowing the representation of human actions and the ones of the reactor control and protection system to start or stop the operation of some safety systems. It is being extensively validated and develop in the frame of the Code Applications and Maintenance Program (CAMP) (RELAP5, 2016) of the US NRC. The RELAP5 is based on a nonhomogeneous and non-equilibrium model for the two- phase system that is solved by a fast, partially implicit numerical scheme to permit economical calculation of system transients. The code development has benefitted from extensive application and validation using quite large number of experimental data e.g., from LOFT, PBF, SEMISCALE, ACRR, NRU, and other experimental programs.

2.2. Short description of TRACE5.05/PARCS

TRACE5.05 /PARCS is a coupled code developed for the US NRC as a reference tool for the analysis of design basis accidents (DBA) of LWR of Gen-II, and III including SMRs. Hereafter a short description of each tool will be provided.

- **The system thermal hydraulic code TRACE**

TRACE is a best-estimate system code of the U.S. NRC for the analysis of Light Water Reactor (LWR); recently extended for liquid metal cooled fast reactors. It solves the static or time-dependent system of six conservation equations of a two-fluid mixture in 1D and 3D (Cartesian and Cylindrical coordinates for the reactor pressure vessel including the core) computational domain using the finite volume and donor-cell approach. Additional equations are formulated to describe the transport of boron in the liquid phase and of non-condensable gases in the gas phase. Due to its versatility, not only NPPs but also different experimental test sections or loops can be simulated with TRACE. A complete set of constitutive equations are formulated to close the balance equations describing the interphase and wall-to-fluid mass and heat transfer in all flow regimes of the boiling curve (i.e., pre- and post-CHF) for both horizontal and vertical flow regimes. In this approach, mechanical and thermal non-equilibrium situations are considered. The diverse component models e.g., pumps, valves, pipes, heat structures, as well as dedicated models for trips and control systems implemented in TRACE allow the representation of nuclear power plants of different design (PWR, BWR, SMRs). Two numerical methods, a semi-implicit method, and the SETS method are implemented in TRACE to solve any kind of slow and fast transients (U. NRC, 2016). Dedicated models describe specific physical phenomena such as thermal stratification, point kinetics, critical flow, etc. TRACE is recently equipped with an Exterior Communication Interface (ECI) for the coupling with any kind of solvers [ECI]. It is coupled with the 3D nodal diffusion solvers PARCS and recently it is coupled to subchannel and CFD (Zhang, 2019) based on the ICoCo-Method (MEDCoupling Developer's Guide, 2019).

- **The nodal diffusion neutronic code PARCS**

The PARCS (Purdue Advanced Reactor Core Simulator) neutronic code is a widely used reactor physics code developed by Purdue University for simulating the behaviour of nuclear reactor cores. It is a three-dimensional (3D) reactor core simulator, which solves the steady-state and time-dependent, multi-group neutron diffusion and SP3 transport equations in orthogonal and non-orthogonal geometries. PARCS is coupled directly to the thermal–hydraulic system code TRACE providing the temperature and flow field information to PARCS during the transient calculations via the few-group homogenized cross sections at fuel assembly level. A separate code module, GENPMAXS, is used to process the cross sections generated by a lattice physics code into the PMAXS format that can be read by PARCS (Downar et al., 2018).

3. The integral plan models of the VVER-1000 Kozloduy plant

3.1. RELAP5 VVER-1000 model

The Baseline input deck for VVER-1000/V320 Kozloduy Nuclear Power Plant, Unit 6 was developed by the INRNE-BAS (Pavlin and Pavlova, 2004). The VVER-1000 RELAP5 model was developed for analysis of operational occurrences, abnormal events, and design basis scenarios. The model provides a significant analytical capability for the specialists working in the field of the NPP safety. Data and information for the modeling of all main systems and components were obtained from the Kozloduy documentation and from the power plant staff. The reactor and the pressurizer model nodalization are shown schematically in Fig. 1.

The Kozloduy VVER-1000 RELAP5 model was defined to include all major systems of Kozloduy NPP, Unit 6, namely: reactor core, reactor vessel, Main Coolant Pumps (MCPs), SGs, steam lines and Main Steam Header (MSH), emergency protection system, pressure control system of the primary circuit, makeup system, safety injection system, steam dumping devices (BRU-K, BRU-A, SG and pressurizer safety valves), and main feedwater system (Pavlin and Pavlova, 2004).

In the RELAP5 model of VVER-1000, the primary system has been modeled using four coolant loops each one including one MCP and horizontal SG. The RELAP5 model configuration provides a detailed representation of the primary, secondary, and safety systems. A hot and average heated flow paths and a core bypass channel represent the reactor core region. The reactor vessel model includes a downcomer, lower plenum, and outlet plenum. The Pressurizer (PRZ) system includes heaters, spray and pressurizer relief valves. The safety system representation includes accumulators, high- and low-pressure injection systems, and reactor SCRAM system. The model of the make-up and blowdown systems includes associated control systems.

RELAP5 heat structure components are used to represent fuel rods, vessel structural internals (core barrel, core baffle, lower and upper plates, protective tube block and etc.) and the reactor vessel. Heat transfer from the primary to the secondary side is modeled by heat structure components. The Henry-Fauske critical flow model is accepted for modeling of break flow (Pavlin and Pavlova, 2004).

3.2. TRACE5.05 VVER-1000 model

The integral plant model of the TRACE5.05 VVER-1000/V320 plant (Kozloduy Unit 6) is developed, based on a previous VVER-1000/V320

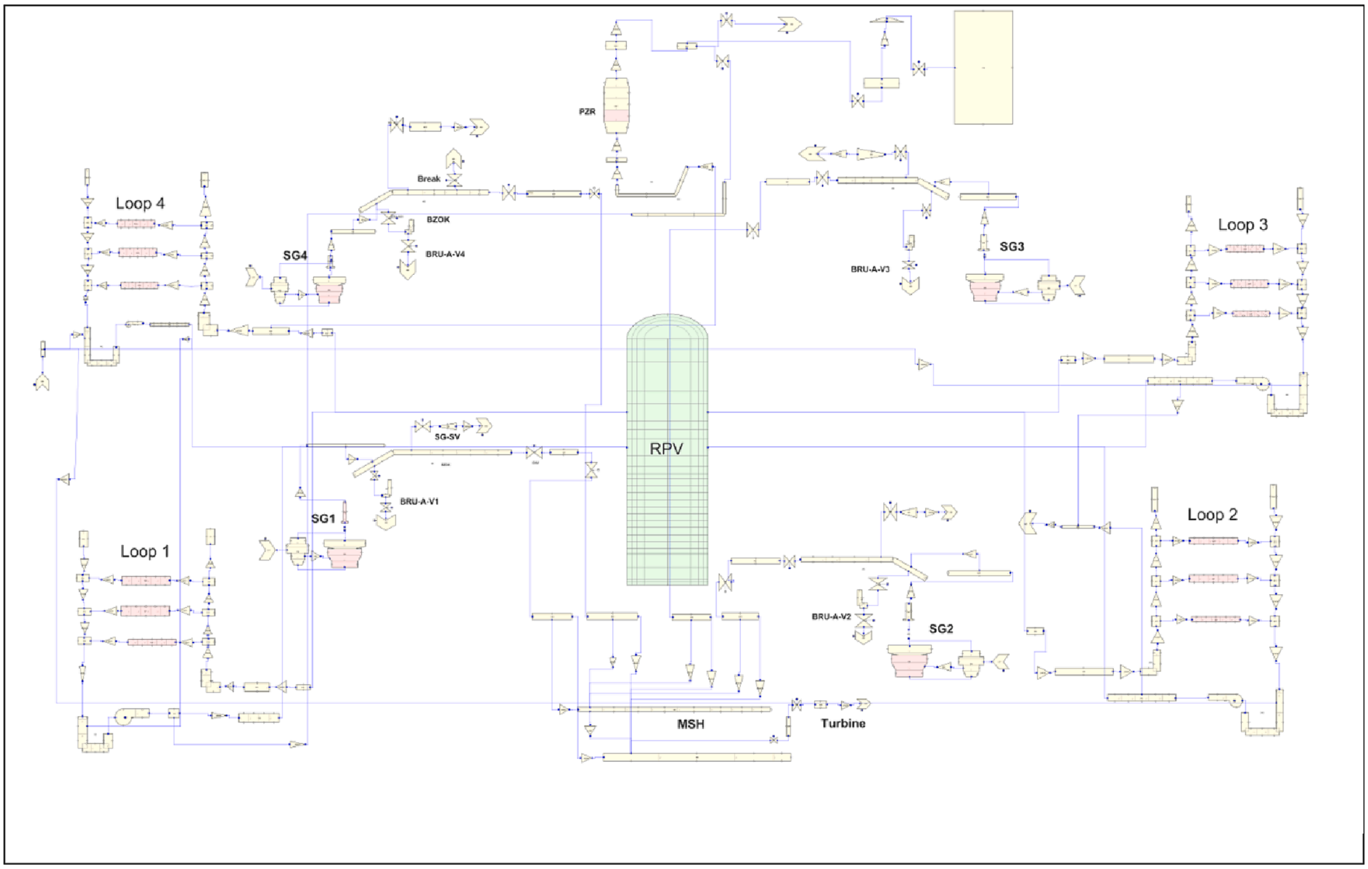

Fig. 3. Primary and Secondary side nodalisation scheme of VVER-1000 TRACE5.05.

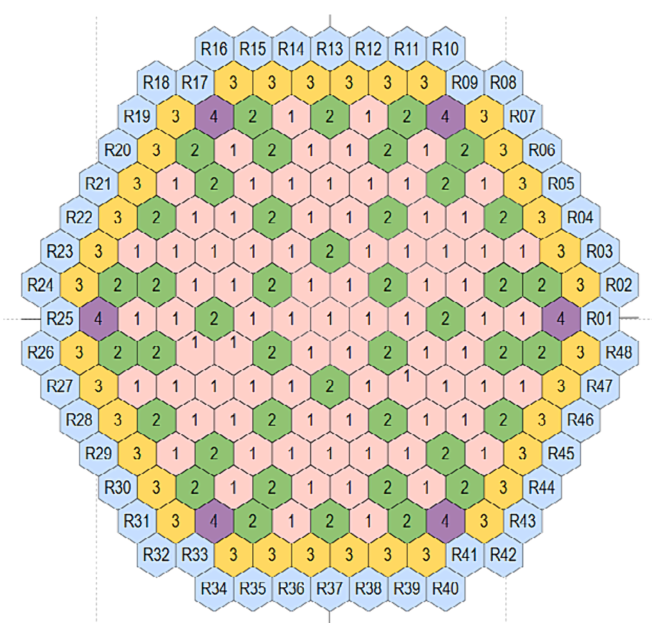

Fig. 4. Core model developed in PARCS.

RELAP5 model (Dinkov and Popov, 2005). The four loops are represented separately with 1D thermal hydraulic components of TRACE5.05 (PIPE, VALVE) (Fig. 3) consisting of the hot legs, the steam generator, the cold legs, and the main coolant pump. In addition, the Pressurizer also is modelled by different 1D-volumes, HTSTR-component to represent the heaters together with the POWER-component. The pressurizer is connected to the cold leg 1 with the surge line and to the hot leg 4 with the spray lines. On the secondary side, each loop consists of the steam lines, the different valves to control the pressure in the secondary side, the common header, the turbine stop valve and the turbine represented by a BREAK-component. Following valves are considered in the steam lines: one steam dump valve to the containment (BRU-A), two safety valves (SV), one main isolation valve (BZOK), and a check valve (CHV). The steam header is connected with the condenser via the steam dump valves (BRU-K), with the atmosphere by the steam dump valves (BRU-A) and with the turbine by the main steam isolation valve (MSIV). The feedwater lines are represented in a very simplified manner by a short PIPE-component and by the FILL-component, where the mass flow rate and temperature of the feedwater are defined as boundary conditions. The Reactor Pressure Vessel (RPV) of the VVER-1000 plant is represented by a three-dimensional VESSEL-component. It allows to discretize the RPV in three directions: axial, radial and azimuthal. The core is represented by three rings, the next ones represent the bypass, core barrel and downcomer. In the investigation, the core neutronics was described by the PARCS diffusion solver. Hence, the core representation of the core coolant channels and the fuel assemblies in TRACE5.05 is done using the 3D Cylindrical VESSEL (Ring 1 to 3, axial nodes: 6 to 17, azimuthal: 1 to 6) as indicated in Fig. 2. For the coupling of TRACE5.05 thermal hydraulics with the PARCS core model, in order to exchange feedbacks during the simulation, the core in TRACE5.05 needs also to consider a node below and above the core, which will provide the coolant temperature for the radial reflector cross-sections.

All additional and needed safety systems e.g., the Emergency Boron Injection System (EBIS), the Control Volume and Chemical System (CVCS) that consists of the Make-up and the Let-down system, the Emergency Core cooling systems (EECCS) including the passive accumulators (HA) the high-pressure injection system (HPIS) and the low-pressure injection system (LPIS), as well as the emergency feed water

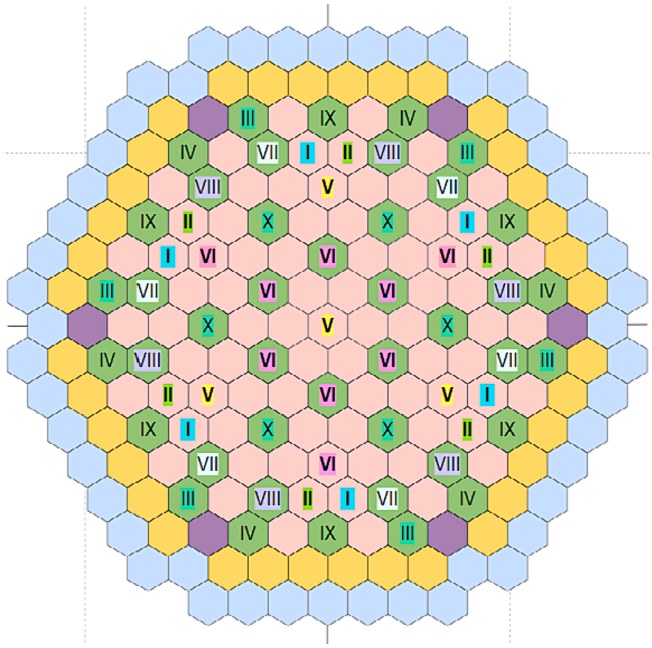

Fig. 5. Control rod banks layout.

Table 1
Variation points considered for the macroscopic cross section generation.

Parameter	HFP, nominal condition	Variations points for XS generation
Coolant density (g/cm^3)	0.725	0.1, 0.2, 0.3, 0.4, 0.5, 0.6, 0.8, 0.9
Boron concentration (ppm)	1200	1, 2000
Fuel temperature (K)	900	600, 1400
Coolant temperature (K)	574	470, 620

Table 2
Decay constants and fractions of delayed neutrons at BOL at 1st cycle.

Group	Decay constant	Relative fraction of delayed neutrons, βi %	Normalized delayed neutrons
1	0.0125	0.0209	0.028756192
2	0.0305	0.1493	0.205421024
3	0.111	0.1368	0.188222345
4	0.305	0.2866	0.394331315
5	1.13	0.0984	0.135388002
6	3.0	0.0348	0.047881123

Total effective fraction of delayed neutrons, β eff = 0.7268E^{-02}.

Table 3
Reactor physics parameters for Unit #6 of Kozloduy NPP at the beginning of 1st cycle.

Parameter	Value
HP Moderator temperature coefficient (MTC), pcm/K	−3.1
HP Doppler temperature coefficient (DTC), pcm/K	−1.661
HP delayed neutron fraction (β_{eff})	0.7268E-02
HP prompt neutron lifetime, [s]	0.267E-04
Control rod group #10 worth, %dK/K	0.91
Tripped rod worth, % dK/K	7.85
Control rod group #5 worth, % dK/K	0.2

system (EFW) are included in the basic model.

3.3. PARCS 3D VVER-1000 core model

The Kozloduy core consists of 163 hexagonal fuel assemblies of four types (Sánchez-Espinoza and Böttcher, 2006). In this study a fresh core loading is considered. A 3D core model was developed in PARCS v3.3.1 including radial and axial reflectors (Fig. 4). FA type 1 (FA1) and 2 (FA2) with enrichments of 2 % and 3 %, respectively, fill almost 75 % of the center part of the core. FA type 3 (FA3) of enrichment of 3 %wt U235 is distributed in the peripheral part of the core. Finally, FA type 4 (FA4) of combined enrichments of 3 % and 3.3 %, complete the peripheral corners of the core, as illustrated in Fig. 5 (Calgaro and Huaccho, 2023). Additionally, the core is surrounded by 48 reflectors with the same hexagonal FA shape. The distinction between each reflector is considered to evaluate the influence of surface discontinuity factors. For the reactivity control of the core, ten control rod banks are distributed in the core, as illustrated in Fig. 5. Control rod banks are only associated with FAs types 1 and 2. Axially the active core is 355 cm long and divided into 30 axial slices, and extra widths of 23.6 cm each are considered for the bottom and top axial reflectors.

The Triangular Polynomial Expansion Nodal (TPEN) method, available in PARCS for hexagonal geometry, is used to solve the neutron diffusion equation with 2 energy group condensed and homogenized macroscopic XS. The TPEN kernel solves two transverse-integrated neutron diffusion equations for a single hexagonal node. One is the radial equation defined for a hexagon and the other is the axial equation defined for the z-direction. The radial problem is solved by splitting the hexagon into six triangles and the employing a polynomial expansion of the flux within each triangle. The axial problem is then solved using the nodal expansion method (NEM) (Downar et al., 2012). Homogenized XS in a 2-energy group structure is generated with SERPENT and then processed with GenPMAXS to obtain the XS in PMAXS-format as required in PARCS. Additionally, a full core model is developed in SERPENT to verify the consistency of the XS generation process (Sanchez et al., 2023). There, the maximum difference of the core reactivity between Serpent2 and PARCS using the XS-generated with Serpent2 using ADF was around 200 pcm (underprediction).

For the MSLB transient analysis, XS depending of thermal hydraulic parameters were generated to cover all the MSLB transient conditions. The following table shows the variation points of thermal hydraulic parameters considered in Serpent2 for the branch calculations. In total 94 branches are considered (considering permutations between the variation points), Table 1.

4. Investigated MSLB scenario

4.1. Initial plant conditions

The transient is initiated by a main steam line break with ID 580, which is occurred between the steam generator (SG) and the steam isolation valve (SIV), outside of the containment. It is assumed the reactor SCRAM simultaneously with a break opening (Spasov et al., 2017). The accepted assumption is not realistic, but it has been taken between the partners in the benchmark, to simplify the simulated scenario and to avoid the possible uncertainties. The reactor state is at the beginning of 1-st cycle, beginning of life (BOL). The reactor power is at 100 %. The initial SGs water mass is accepted by the partners to be 48 000 kg. The main physics data for investigated MSLB scenario are presented below. In the Table 2 and Table 3 are given the main data for point kinetics modelling.

4.2. Main assumptions and investigated scenario

MSLB transient description, summarizing the main assumptions: The selected scenario is completed for 600 sec. The reason for choosing the

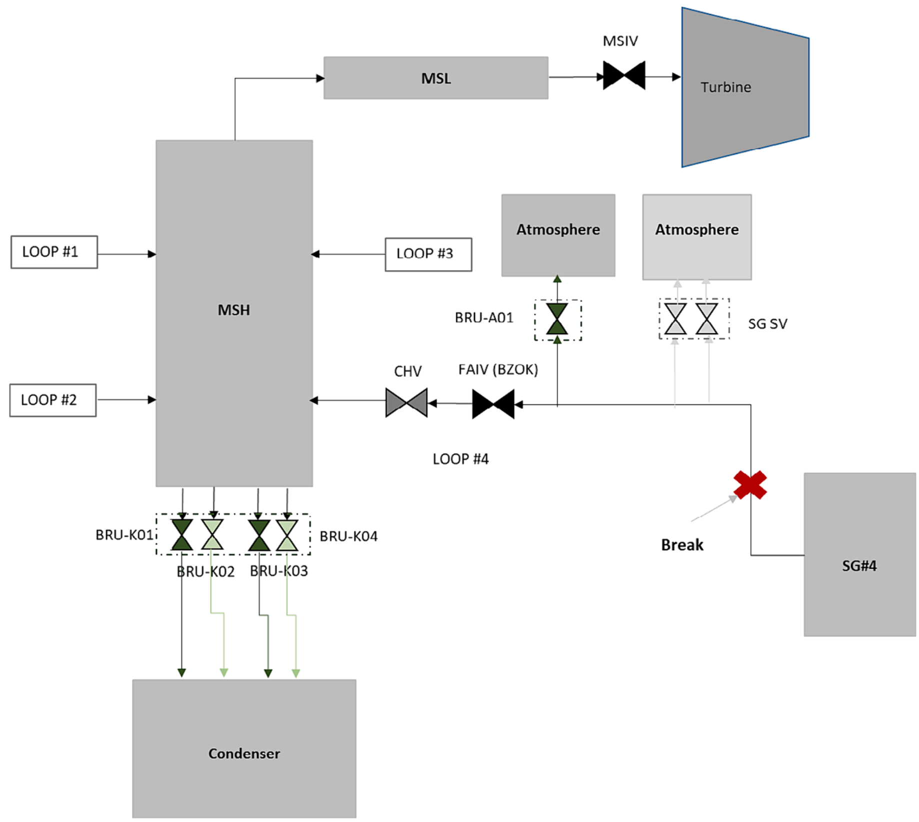

Fig. 6. Break location with simplified steam system.

Table 4
Core kinetic parameters.

Parameter	Value
Total beta effective (pcm)	705
Neutron generation time (μs)	25.7
Group of precursors	8

Table 5
Delayed neutron fraction and decay constant data.

Group	Beta fraction	Lambda (1/s)
1	2.085E-04	1.2467E-02
2	1.023E-03	2.8292E-02
3	5.940E-04	4.2524E-02
4	1.334E-03	1.3304E-01
5	2.264E-03	2.9247E-01
6	7.558E-04	6.6649E-01
7	6.261E-04	1.6348E + 00
8	2.474E-04	3.5546E + 00

Table 6
Reactivity coefficients for nominal and critical boron concentration.

State, boron	\$/ΔCoolant density \$/kgm^3	\$/ΔBoron in coolant \$/ppm	\$/ΔFuel temperature \$/K	\$/ΔCoolant temperature \$/K
HFP, 1200 ppm	2.23E-03	−1.53E-02	−3.37E-03	−1.03E-03
HFP, 1630 ppm	−4.54E-03	−1.43E-02	−3.36E-03	−3.40E-03

Table 7
Comparison of the sequence of events predicted by RELAP5 and TRACE5.05/PARCS.

Events	Time (s)	TRACE5.05/ PARCS code	RELAP5 code
SLB with ID = 580 mm at line #4 between the SG#4 and BZOK#4, sec	0.0	0.0	0.0
SCRAM activation, sec	0.0	0.0	0.0
Make up and Let-down systems switch off, sec	2.0	2.0	2.0
Turbine stop valves (MSIV) close, sec	10.0	5.0	10.0
CHV closing in loop#4, sec	0.1	0.1	0.1
BZOK #4 closing, sec	2.9	0.1	0.1
BRU-Ks opening, sec	–	9.0	46.0
SGs water initial mass at initial time, t	48.0	50.0	48.0
SG#1 water mass at 600 sec, kg	–	74 472.0	63 921.0
SG#4 water mass at 600 sec, kg		1 612.0	629.0
Minimum water level at Pressurizer (PRZ), sec	–	323.0	187.0
Integral break flow rate at 600 sec, kg	–	68197.0	70550.0
Re-criticality achieved, sec	no	no	no
Return-to-power peak, s / value, MW	no	no	no
Transient end, sec	600.0	600.0	600.0

studied scenario only for the first 600 s is because after 300 - 400 s the affected SG #4 is almost empty and its operation is not effective. The rapid and large cooling of the damaged primary side is observed in the first 100-200 sec. The uncontrolled cooling of the primary side continues until the affected SG#4 completely dry out. After that, the transition process stabilizes the plant parameters by the work of the other circuits and BRU-Ks.

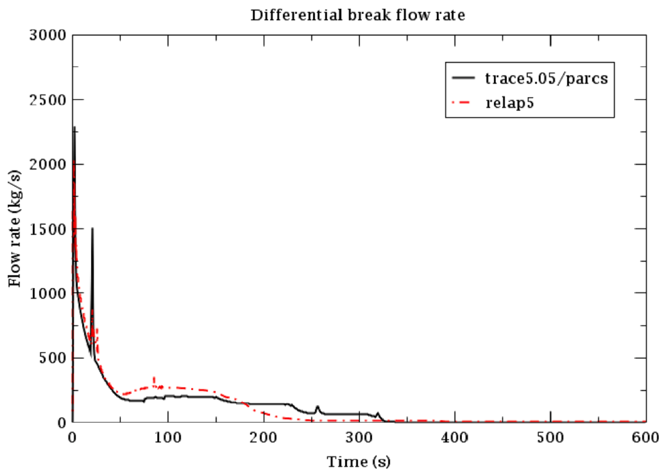

Fig. 7. Comparison of differential break flow rate.

- Break: A large break of the steam line of loop-4 is assumed to occur at time 0 sec. The break is located between the steam generator (SG) and the steam isolation valve (SIV) (see Fig. 6). The steam line pipe diameter is 580 mm,
- The break happens when the plant is operated at nominal plant conditions,
- Core is loaded with fresh fuel at begin of cycle (BOC) conditions,
- SCRAM is assumed just after the break initiation for simplifications (normally SCRAM is caused either by low secondary side pressure, low primary circuit pressure or high thermal power);
  - o Most reactive peripheral control assemblies remain stuck out in case of SCRAM
  - o Location of stuck rod is close to the core sector with the highest overcooling,
  - o The time for the full control rod insertion is 4 sec
- MCP #4 coast down for app. 55 s.
- Feed water valve on damaged secondary loop #4 fails and remains open for some period of time (additional FW into SG#4); The boundary condition tables for intact and broken loops is given in (Kolev et al., 2006; Stefanova et al., 2021).
- Turbine stop valves (MSIV) closes for 10 s after the reactor SCRAM. Time for fully open/close MSIV is around 0.2 s;
- Turbine bypass to condenser (BRU-Ks) starts to open and switches to MSH pressure control mode after closing MSIV. The Opening of all 4 BRU-Ks at $P_{MSH} > 6.67$ MPa and supporting P_{MSH} is 6.2807 MPa. BRU-Ks closes at 5.79 MPa and will be re-open again only if $P_{MSH} >$ 6.67 MPa;
- BZOK #4 (FAIV) is closing after reaching the signals. The signals applied in the scenario is: Signal for close SIV-4 (BZOK) is $P_2 < 4.9$ MPa and $T_{S1} - T_{S2} > 75$ °C;
- Make up and Let-down systems are used only during the steady state. During the transient the systems are not used for reducing the uncertainty- suggested to be isolated for 2 sec;
- PRZ heaters are switched on after primary depressurization for some period until primary pressure is back after dry-out of SG#4 and due to decay power. In our case they are switched off after PRZ water level became below 4.2 m which happen in first 50 s.

4.3. Reactivity coefficients

The used kinetic parameters at point kinetics RELAP5 VVER-1000 model for BOC of 1st cycle (Table 4 and Table 5), were extracted by KIT using PARCS code with considering HFP state at critical boron concentration 1630 ppm.

Reactivity coefficients are provided for system code that use point kinetic models. The level of boron concentration level has a significant impact on the values of reactivity coefficients, particularly the one associated with the coolant density. At critical boron concentration (1630 ppm) a negative value was obtained for this reactivity coefficient, while a positive value was obtained for the reference boron concentration (1200 ppm), as presented in Table 6.

5. Comparison and discussion of calculated results

This section presents the comparative results between system thermal hydraulic system code RELAP5 with point kinetics approach and the 3D kinetics reactor core with coupled codes TRACE5.05/PARCS. In the table below is given the comparison of sequence of main events predicted by RELAP5 and TRACE5.05/PARCS codes (Table 7).

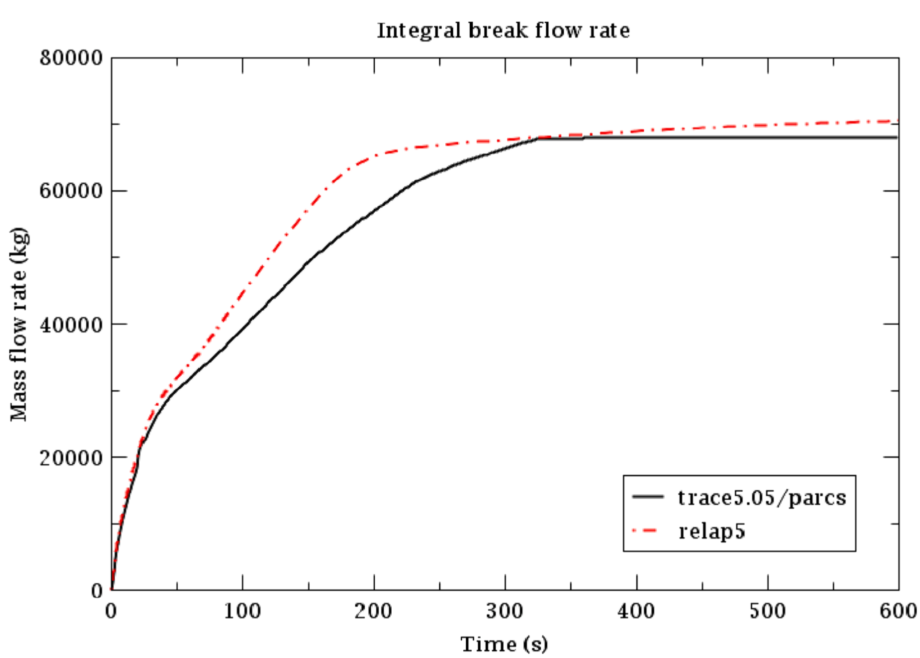

Fig. 8. Comparison of integrated break flow rate.

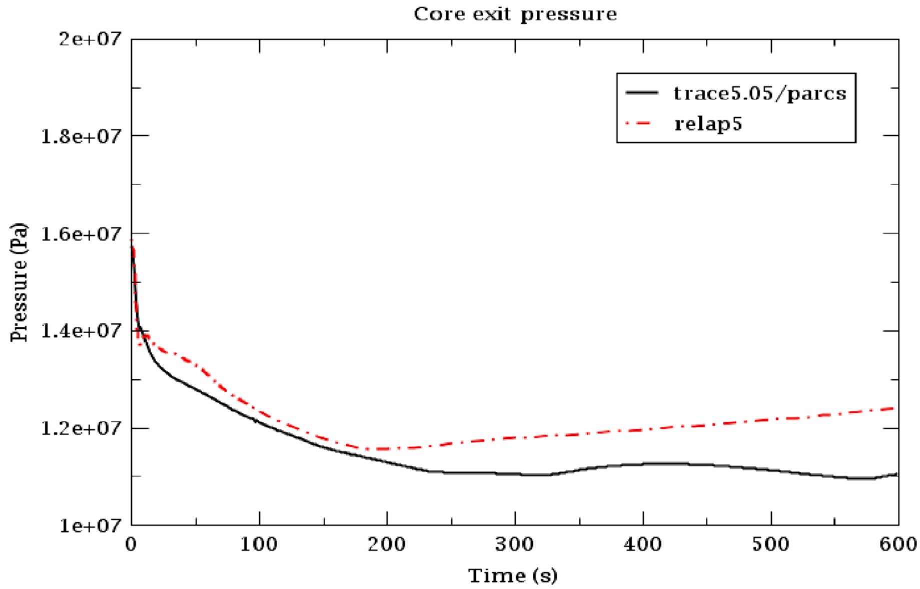

Fig. 9. Comparison of core exit pressure.

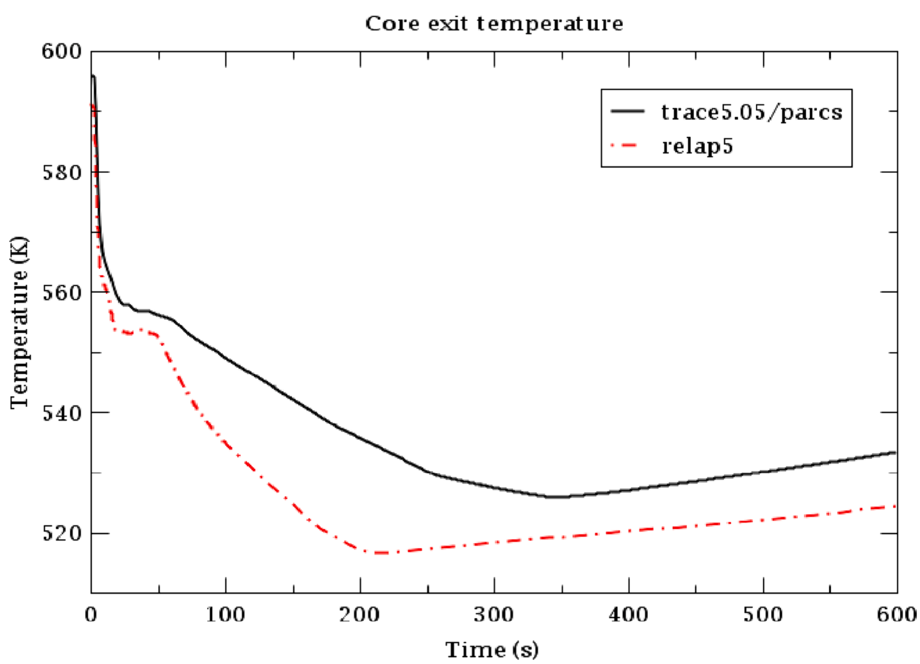

Fig. 10. Comparison of core exit temperature.

5.1. Differential break flow rate

The comparison of the differential break flow rate predicted by both codes is presented in Fig. 7. The comparison shows that both codes predicted very close results of the differential break flow rate. The coolant starts to decrease after the break initiation. As it is seen the both codes predicted flow rate through the break at around 2250 kg/s. The differential break flow rate reduces to 0 kg/s after 300 sec due to the pressure decrease in the affected SG#4. In the period between 50 s and 320 sec it is observed small fluctuation in the predicted results, which could be explained by the numerical problems in the computer codes.

5.2. Integrated break flow rate

A comparison of the integrated break flow rate is given in Fig. 8. As it can be seen, the both codes predict similar behavior. The integrated flow rate predicted by both codes starts to increase rapidly after 25 s, after 50 s the integrated flow rate of RELAP5 becomes higher relatively compared to TRACE5.05/PARCS and after 300 s the integrated flow rate becomes close in both codes.

5.3. Core exit pressure

The comparison of the core exit pressure is given in Fig. 9. Overall, the trend of predicted core exit pressure in both codes is close until 160 sec after that it became different until the end of transient. The observed difference in the predicted pressure results after 200 sec, could be explained by the work of BRU-Ks. The core exit pressure in both calculations begins to decrease rapidly after the initiation of the break in loop #4. The pressure drops rapidly due to the loss of coolant through the

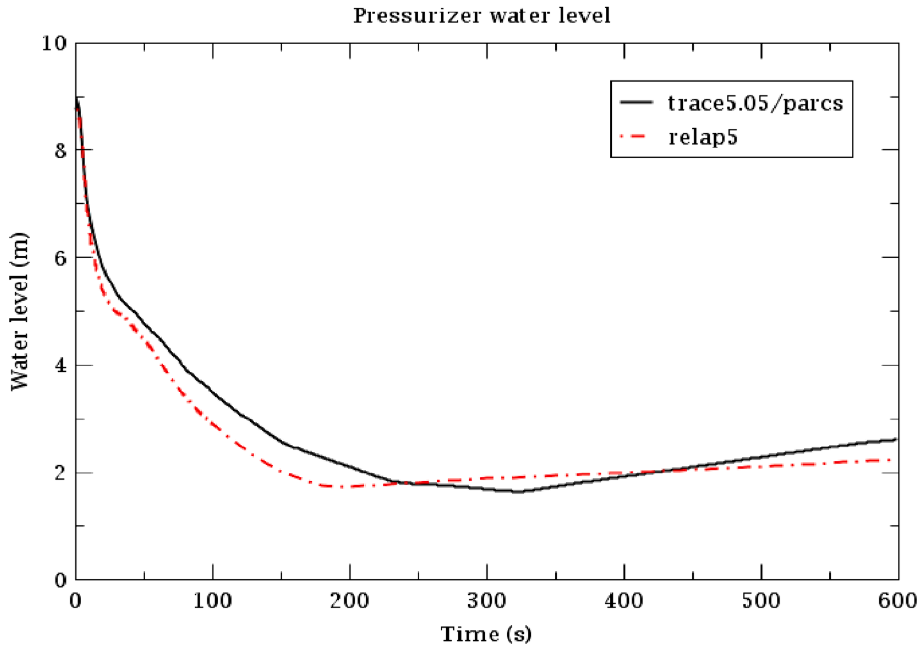

Fig. 11. Comparison of PRZ water level.

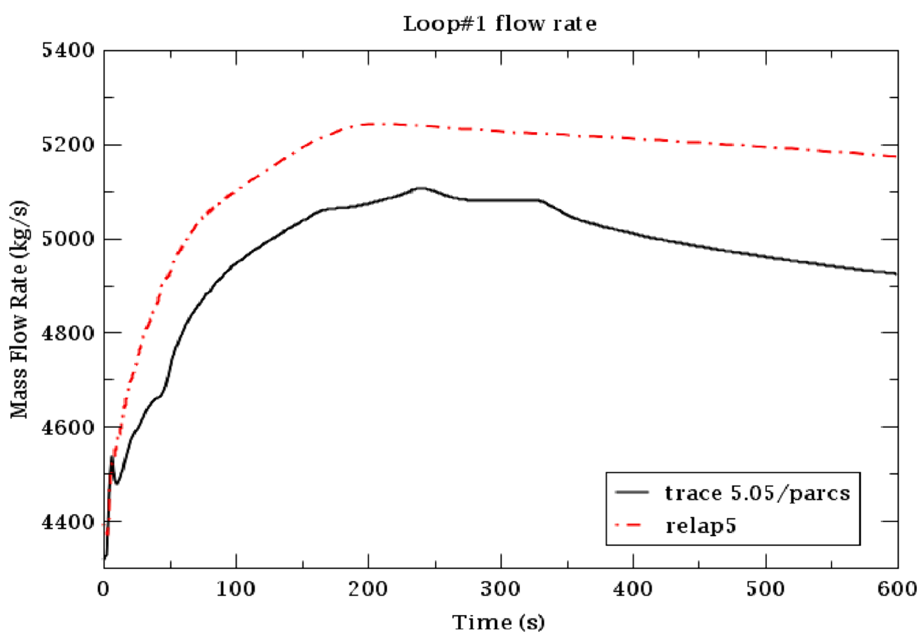

Fig. 12. Comparison of Loop#1 flow rate.

break and subcooling of the primary system by the steam flow through the break in faulted SG. As a result of it, the primary pressure starts to increase slowly after 180 sec in RELAP5 until the end of the transient, while in TRACE5.05/PARCS the pressure continues to decrease slowly after 200 sec.

5.4. Core exit temperature

The comparison of predicted core exit temperature is presented in Fig. 10. As it can be seen, after the MSLB event is observed in loop#4, the core exit temperature starts to decrease rapidly in both calculations, and then starts to increase slowly. The core temperature predicted by both codes has a similar trend, but the periods of decrease and increase in core temperature are different. The observed difference in TRACE5.05/PARCS vs RELAP5 is due to the earlier Turbine isolation, which is in 5 sec in TRACE5.05/PARCS, while in RELAP5 it close at 10 sec. The reason to simulate this way is to create conditions for opening BRU-K in TRACE5.05/PARCS to have the same conditions as in RELAP5 for BRU-K activation. After the isolation of the affected SG#4, the temperature of the reactor coolant began to rise slowly and reaches 530 K in RELAP5 and 535 K in TRACE5.05/PARCS. The smooth increase of the core temperature after 200 sec could be considered as a plant response during the stabilization process, after a large cooling down of the primary side.

5.5. PZR water level

The comparison of the PZR water level is presented in Fig. 11. The comparison of PZR water level behavior predicted by both codes show

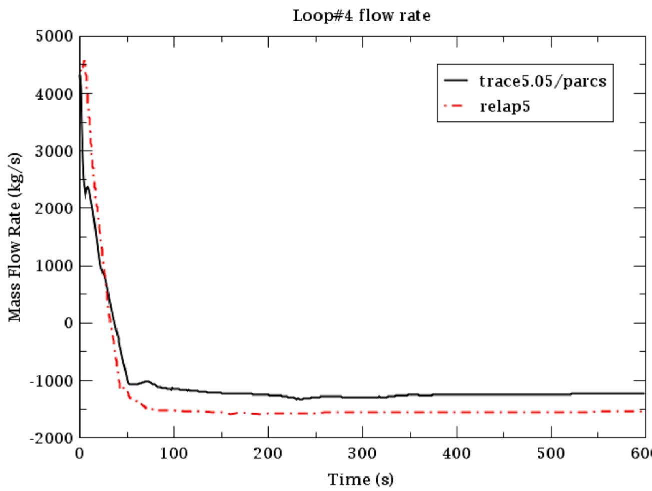

Fig. 13. Comparison of Loop#4 flow rate.

close agreement. The PZR water level starts to decrease after the break initiation. It drops to 1.8 m at app. 190 sec in RELAP5, while in TRACE5.05/PARCS the minimum water level of 1.6 m is reached at 323 sec. After that the PZR water level rises slowly.

5.6. Loop#1 flow rate

The comparison of Loop#1 flow rate is given in Fig. 12. After the break initiation the coolant flow rate increase significantly, as a results of coolant temperature and density decreasing. The behavior of other two loops #2 and 3 is identical. Overall, the comparison of Loop#1 flow rate predicted by both codes has a similar trend but different flow rate.

One of the reasons for this deviation can be explained by the different coolant density in both codes resulting from the small deviations in the cooling of primary circuits. A significantly higher mass flow rate is observed in the RELAP5 calculations compared to TRACE5.05/PARCS. The mass flow rate in loop #1 in both calculations starts to increase after the onset of the break and reaches 5190 kg/s in the RELAP5 calc. and approx. 4920 kg/s in TRACE5.05/PARCS calc., until to the end of the transient.

A comparison of the flow rate of loop #4 predicted by the two codes is presented in Fig. 13. The flow rates in the failed loop #4 began to decrease immediately after the break initiation and reduce the flow to (−1500) kg/s at 45 sec from the start of event. In this way it is observed revers flow, due to MCP #4 switching off, while the other 3 MCPs are in operation. The largest flow rate closes to coolant heat up in damaged SG#4 and cool down of primary side.

5.7. MSH pressure

A comparison of MSH pressure behavior predicted by both codes is presented in Fig. 14 and Fig. 15. After the break initiation the pressure in MSH decreases shortly in first 10 sec than start to increase rapidly, after the turbine isolation by MSIV and isolating the affected SG#4 from the MSH using the BZOK (SIV in loop #4). After closing the main steam isolating valve, the steam flow goes through the MSH to the BRU-Ks. The SGs are fed with deeply subcooled water coming from the deaerator. In fact, after the reactor SCRAM the temperature of feed water is reduced from 493 K to 437 K for a very short time. The investigated event causes rapid depressurization of the damaged secondary side and uncontrollable cooldown of the damaged loop primary side, which causes the insertion of colder water into the reactor vessel and reactor core resulting in increasing of the reactivity. The pressure increases until the BRU-K opening set point is reached. After reaching their set points, the BRU-Ks open and begin to regulate the pressure in the secondary side. They open when the pressure in the secondary side is 6.67 MPa, reducing the pressure and controlling it to 6.28 MPa. When the pressure drops below 5.79 MPa, they close. As it can be seen from the comparison, the BRU-Ks in TRACE5.05/PARCS open significantly earlier compare to RELAP5. They open at 9.0 s due to earlier turbine isolation in TRACE5.05/PARCS, while in RELAP5 they open at 46.0 s, where the turbine is isolated at 10.0 s.

5.8. SG#4 pressure

A comparison of the secondary side pressure of the affected SG#4 predicted by both codes is presented in Fig. 16. As the SG# 4 feed water

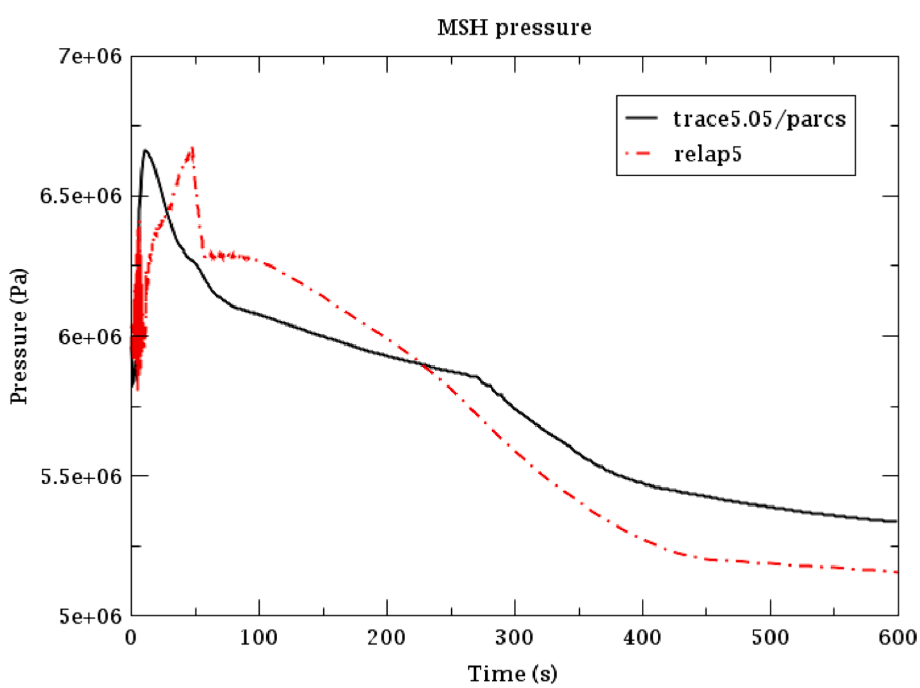

Fig. 14. Comparison of pressure in MSH.

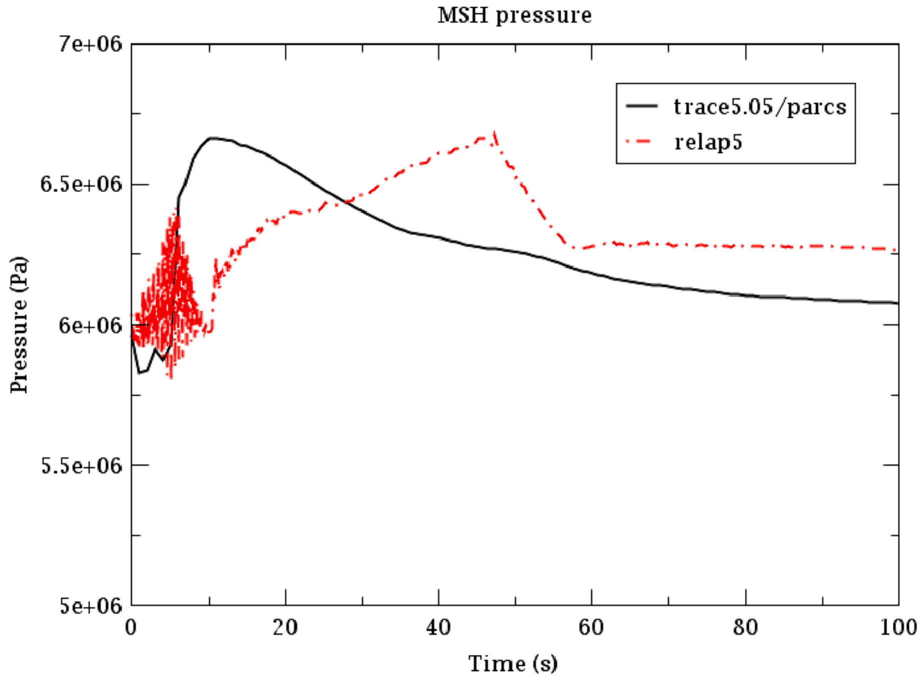

Fig. 15. Comparison of pressure in MSH for first 100 sec.

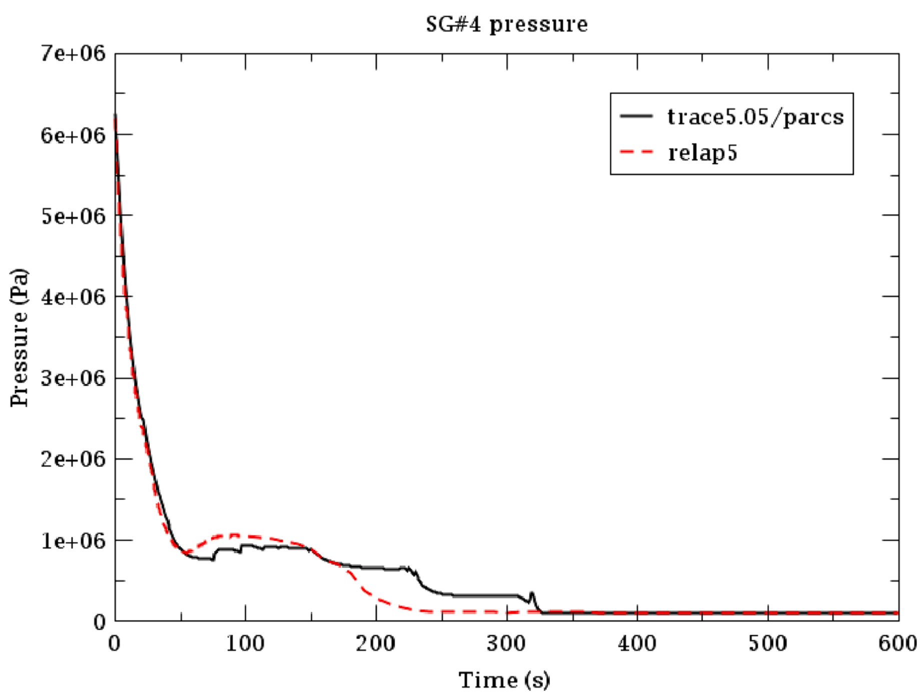

Fig. 16. Comparison of SG#4 pressure.

is almost isolated in 75 sec, the SG #4 dries out very fast and depressurizes completely for appoximately 300 s as it could be seen in Fig. 16. After that time the SG#4 no longer cooldown the primary side of the damaged loop and the reactor heat is removed through the other three SGs. A comparison of the calculated results predicted by the both codes show good agreement except the period between 50 and 320 s where small pressure deviations is observed in the predicted results. After the break initiation, the pressure decreases rapidly in both codes and continues to decrease slowly until the SG is emptied. The decay heat in the first 50 s is removed mainly from work of SG#4, after the depressurization of SG#4 at around 320 s, the decay heat is removed from the operation of BRU-K.

5.9. *Reactor power*

The Fig. 17 presents the total power reduction after the SCRAM as predicted by the two codes. As it is seen from the comparison the trend predicted by both codes is the same.

5.10. *Total reactivity*

The comparison of the total reactivity predicted by both codes shows

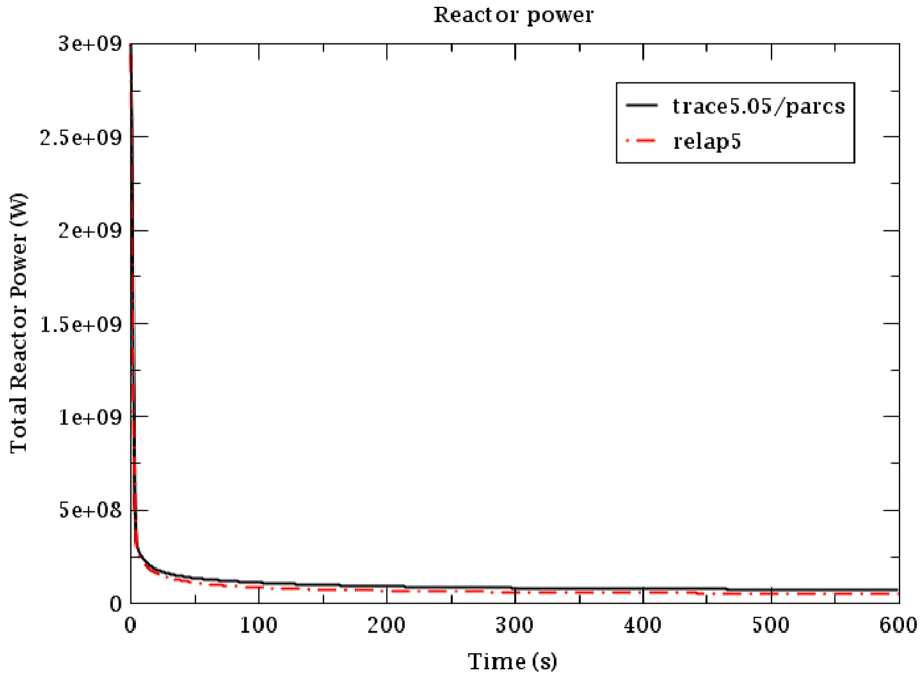

Fig. 17. Comparison of Reactor power.

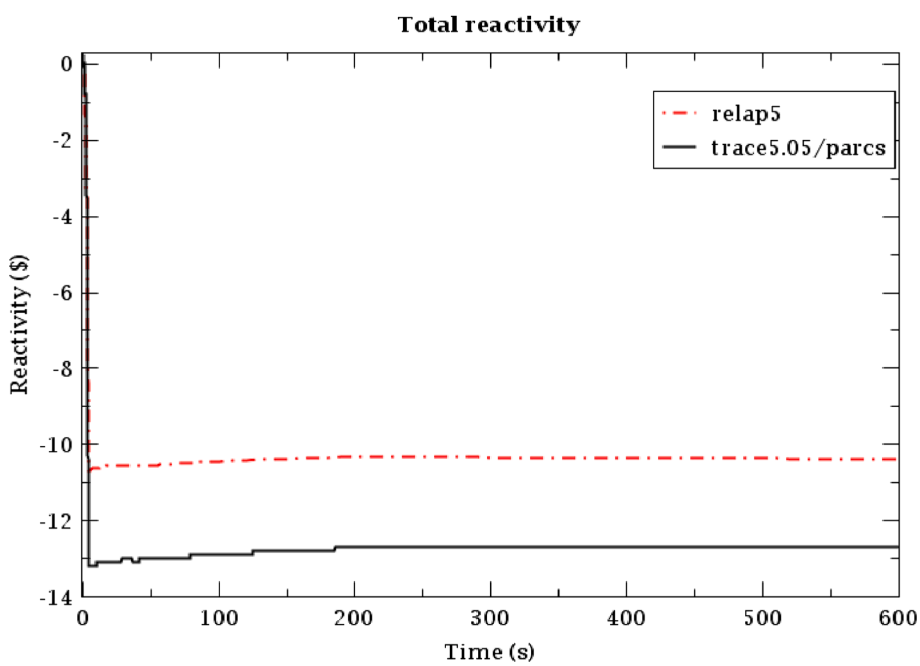

Fig. 18. Comparison of total reactivity.

similar results (Fig. 18). After the water inventory from the affected SG is blowdown through the break, the temperature in the primary system start to rise slowly, leading to decrease in the feedback reactivity. The total reactivity in both codes is negative, it is −12.7 $ in TRACE5.05/PARCS and at around −10.5 $ in RELAP5.

6. Conclusions

Based on the obtained simulation's results for the MSLB-scenario, it can be stated that both codes predict the overall behavior of the VVER-1000 plant quantitatively in good agreement for the majority of the compared parameters. There are some small deviations among them. Both simulation tools are able to catch the key-thermal hydraulic phenomena of the MSLB-transient. The deviations may be caused by the different thermal hydraulic model of the core and RPV: RELAP5 uses 1D while TRACE5.05/PARCS a 3D approach. In addition, the core neutronic analysis is performed with a Point Kinetics model in RELAP5 and with a 3D kinetics in TRACE5.05/PARCS. Here, the RELAP5 results tends to over predicts the inserted positive reactivity to the core compared to the 3D TRACE5.05/PARCS model: −10.5 $ compared to −12.7$. The deviation of the global thermal hydraulic parameters predicted by REALP5 and TRACE5.05/PARCS are a result of the different break modelling in both codes. It leads to different pressure reduction and break-outflow mass flow rates during the transient. Last but not least, the fast evaporation and critical flow through the broken steam line 4 are in reasonably good agreement. In general, the results predicted by both computer codes demonstrates that the plant behaviour during the MSLB-accident progression is safe i.e., far from re-criticality and no significant power increase after SCRAM.

CRediT authorship contribution statement

Antoaneta Stefanova: Writing – original draft, Investigation, Conceptualization. **Pavlin Groudev:** Writing – review & editing, Investigation, Conceptualization. **Victor Hugo Sanchez-Espinoza:** Writing – review & editing, Investigation. **Gianfranco Huaccho Zavala:** Investigation.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

The authors do not have permission to share data.

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