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Assessment of the nuclear power plant behaviour during a small break LOCA combined with a SGTR assuming a station black out

P. Vryashkova [a], A. Stefanova [a,*], P. Groudev [a], Victor Hugo Sanchez-Espinoza [b]

[a] Institute for Nuclear Research and Nuclear Energy (INRNE) - Bulgarian Academy of Science, Tzarigradsko shaussee 72, Sofia 1784, Bulgaria
[b] Karlsruher Institut für Technologie, Kaiserstrasse 12, Karlsruhe 76131, Germany

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ABSTRACT

This paper presents the comparison of the results achieved by performing calculations using integral computer codes RELAP5/mod3.3 and TRACEv5.05. The scenario selected for this purpose is a multiple failure transient based on a “Small break loss of coolant accident (SB LOCA) combined with a steam generator tube rupture (SGTR) simultaneously with a Station blackout (SBO)”. In the investigated scenario is assumed that the break with equivalent diameter of 30 mm is located on the main coolant loop #1 between the main coolant pump and the inlet of the reactor vessel. In addition, a double ended guillotine tube rupture in SG #4 with the equivalent diameter of 13 mm is simulated simultaneously with a SBO.

The main objective of this work is focused on evaluation of VVER1000 plant response during multiple failure events and the ability of system thermal hydraulic computer codes to simulate the key phenomena and processes taking place during the evolution of the accidental scenario.

The computer codes used to perform the study are based on different approach for the modeling of the reactor pressure vessel (RPV) and core physics. The RELAP5 code is based on 1D thermal hydraulics modelling of RPV and point kinetics for reactor physics modelling, while in the TRACE code used 3D thermal hydraulics of RPV modelling and 3D neutron kinetics model for reactor physics modelling. The calculation is performed for the end of cycle (EOC) eight, when the reactor is operated at nominal power.

The investigations were performed in the frame of the EU H2020 CAMIVVER project. The comparison of the calculated results has shown that the codes predict the VVER1000 plant behavior under SB LOCA with SGTR simultaneously with a SBO in a good manner.

1. Introduction

A large number of VVER-type reactors are in operation in Russia and other East European countries e.g. Bulgaria, Hungary, Finland, etc. In the last decade VVER-1200 reactors were built and are being build e.g. in Bangladesh, Egypt, Hungary, Turkey, etc. which is characterized by passive safety system to cope with accidental conditions (Asmolov et al., 2017). There is an increased interest in the EU to increase the capabilities for the safety assessment of VVER-type reactors using different deterministic system thermal hydraulic codes as RELAP5, TRACE, CATHARE, ATHLET, etc. Consequently, the EU H2020 project CAMIVVER (Denis Verrier et al., 2021) was launched in 2020 with the goal to improve, validate and apply different safety analysis tools including system thermal hydraulics codes, CFD, and core physics tools for the analysis of selected transients such as SBLOCA, LBLOCA, Mains Steam Line Break (MSLB), where a code-to-code comparison played a key-role. In this framework, one goal of CAMIVVER project, was to analyse selected DBAs of VVER-1000 reactors using different computer codes. In

Abbreviations: BAS, Bulgarian Academy of Sciences; BDBA, Beyond design base accident; BRU-K, Steam Dumping Device to the Condenser; BRU-A, Steam Dumping Device to the Atmosphere; BZOK, Fast Acting Isolation Valve; CHV, Check Valve; DBA, Design base accident; EBIS, Emergency Boron Injection System; EECCS, Emergency Core cooling systems; EFW, Emergency Feed Water System; EOC, End of Cycle; HFP, Hot Full Power; HPIS, High Pressure Injection System; HHPIS, High-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; MCP, Main Coolant Pump; MIV, Main Isolating Valve; MSH, Main Steam Header; NC, Natural Circulation; NPP, Nuclear Power Plant; PRZ, Pressurizer; PRISE, Primary to Secondary; RPV, Reactor Pressure Vessel; SBO, Station Blackout; SCRAM, Reactor Protection Signal; SG, Steam Generator; SGTR, Steam Generator Tube Rupture; SV, Safety Valve; VVER, Water-Water Energy Reactor.

* Corresponding author at: INRNE-BAS, Sofia, Bulgaria.
E-mail address: antoanet@inrne.bas.bg (A. Stefanova).

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(Vryashkova et al., 2023), the comparative analysis of a MSLB-accident in the VVER-1000 Kozloduy plant was presented and discussed. In this paper, the analysis of a SBLOCA using the TRACE and RELAP5 codes with Point Kinetics. The initiating event of the Small Break Loss of Coolant Accident (SB LOCA) (Iegan et al., 2016), is a small break of 30 mm equivalent diameter located in the cold leg of the main coolant loop #1 between the main coolant pump (MCP) and the reactor vessel inlet. Additionally, a double ended guillotine steam generator #4 tube rupture (SGTR) is assumed. The SGTR is located at the end of the tube just before the cold collector at the elevation of 1.8–2.0 m. In this way, the small leakage from primary to secondary side (PRISE) is assumed in the scenario from the beginning of the transient.

Simultaneously with both initiating events, a station blackout (SBO) is assumed, which on the one hand will simplify the transient, but also will lead to more severe conditions. Involving a total loss of internal and external electrical power will allow observing the plant's response to the loss of primary coolant without any injection from make-up or emergency core cooling systems (HPIS (TQx13), LPIS (TQx2), and HHPIS (TQx4)) as well as the transition from the forced to the natural circulation (NC). The selected combination of the events allows to investigate important phenomena during the accident progression. Based on the selected initiating events, the accident progression requested special attention in preparation of models for flow leakages, models for regulation of secondary side pressure (BRU-As), etc. The main phenomena, critical (IAEA, 2000) for understanding the progression of loss of coolant accident combined of two additional events (PRISE and SBO) are: additional loss of coolant inventory to the damaged steam generator, loss of heat removal capability, core uncovery, core heat up, and long-term cooling. Another challenge is the simulation of the transition from forced to natural circulation, which can occur in different accidental sequences.

Including the SGTR event in the combination with a SB LOCA (Redondo-Valero et al., 2024; Queral et al., 2018) and SBO initiating events, request special treatments from the Unit operators as, SGTR will cause transfer of radioactive primary coolant to the secondary side and will request isolating the damaged SG by steam and feed water. The goal is to achieve maximum reduction of leaks from the primary circuit to the secondary side of the damaged SG. This could be performed by faster reducing of primary pressure close to the secondary pressure, but still slightly above it. Isolation of a damaged steam generator by steam line is necessary to protect the secondary side from radioactive contamination. Therefore, it is necessary to know that earlier isolation of a damaged SG by closing a fast-acting isolation valve (BZOK) should not be carried out so quickly due to a possible increase in the secondary side pressure of the damaged SG leading to the opening of the BRU-A of this SG and thus enabling a direct radioactive release into the environment. With other words, there is a need for the optimization of the isolation of the damaged SG.

The focus of this papers is the discussion of a comparative analysis of the SBLOCA sequence using different models developed for two system thermal hydraulic codes, namely RELAP5 and TRACE. This academic transient was selected in the CAMIVVER-project knowing that there is no plant data of VVER 1000 (MEDCoupling Developer's Guide, xxxx; Groudev et al., 2001) for a truly code validation. The intention is to compare the capability of two system thermal hydraulic codes in analysing such a complex accidental sequence, which combines a design basis accident (DBA) event such as a Small Break Loss of Coolant Accident (SBLOCA) (Groudev and Georgieva, 2010; Chatterjee et al., 2010) and primary to secondary side leakage (PRISE) caused by the steam generator tube rupture and the beyond design basis accident (BDBA) (Pavlova et al., 2008) such as the Station Blackout (SBO) (Pavlova et al., 2007).

To reduce the sources for deviations when performing a code-to-code comparison, in addition to the sequence of events, all assumptions for activation/deactivation of various safety systems are listed. The code-to-code comparison is also applied to the prediction of the steady-state conditions just before the SBLOCA started.

In this paper, after the introduction, the Chapter 2 is devoted to the detailed description of the transient scenario including all initial and boundary conditions. Then, the Chapter 3 describes the respective computer codes, RELAP5 and TRACE, followed by the details of the developed plant models for each code that includes all primary, secondary and safety systems of the VVER-1000 reactor. In Chapter 4, code-to-code comparisons of both the steady-state and transient results are discussed in detail. Chapter 5 presents the main conclusions of the investigations.

2. Investigated accident scenario

The selected accident is a SB LOCA with the equivalent diameter 30 mm (ID = 30 mm) on the main cold coolant loop #1 between the MCP and the RV inlet, along with a total station blackout at 0.0 s when the core is at End-of-cycle (EOC) conditions at full power conditions. In addition, it is assumed that a double ended break of one pipeline of the SG#4 with the equivalent diameter of 13 mm (ID = 13 mm) occurs simultaneously. The SG-pipeline break is located near the cold collector in the upper part of the tubes bundle (elevation 1.8–2.0 m from the bottom of SG). Moreover, an SBO is assumed to occur.

2.1. Initial and boundary conditions

The main thermal hydraulic parameters of the VVER-1000 plant at full power operation conditions are given in Table 1.

In addition, the following assumptions are considered for the analysis:

- Isolating of damaged SG by closing BZOK after the primary pressure is below the pressure in the faulted SG should be performed by operator, but this analysis is without any operator actions.
- Steam dump to atmosphere: all 4 Steam Dumping Device to the Atmosphere (BRU-As) are assumed to work to the end of calculations (they are powered by Accumulator battery)
- Leakages from MCP seals are not considered. The loss of coolant through the MCP seals will lead to a reduction in primary coolant mass and removing of reactor core residual heat. It is more conservative from the point of view of core cooling.

The reactor physics data of the core for the description of the core behavior during the accident considering the interaction of the neutronics and thermal hydraulics with the Point Kinetics model are provided in the Tables 2-4.

The initial HP core average axial relative power distribution is presented in Table 5.

Table 1
Main thermal hydraulic parameters of the VVER-1000 plant (Stefanova et al., 2020).

Parameter	Plant Design
Reactor thermal power, MW	3000
Primary pressure, MPa	15.7
Pressurizer Level, m	8.77
Average coolant temperature at reactor inlet, K	560.15
Average coolant temperature at reactor outlet, K	592.05
Mass flow rate through one loop, kg/s	4400.0
Boron concentration in the primary circuit, ppm.	53
Pressure in SG, MPa	6.27
Pressure in the main steam header (MSH), MPa	6.08
Steam mass flow rate through SG steam line, kg/s	408
SG Water Levels, m	2.40
Liquid mass in the SG secondary side, t	48.0

Table 2
Decay constants and fractions of delayed neutrons at EOC (Stefanova et al., 2020).

Group	Decay constant, s^{-1}	Relative fraction of delayed neutrons, β_i %	Normalized delayed neutrons
1	0.0125	0.01593	0.027000915
2	0.0305	0.12508	0.212007187
3	0.111	0.11092	0.188006373
4	0.305	0.22715	0.385013051
5	1.13	0.0826	0.140004746
6	3.0	0.0283	0.0479677

Table 3
Point kinetics parameters for Unit 6 of Kozloduy NPP, End of Cycle 8 (Stefanova et al., 2020).

Parameter	Data from KNPP (this data is used for current calculation)
HFP Moderator Temp. Coefficient (%/K)	$-54.866.10^{-3}$ (Moderator temperature coefficient including density temperature coefficient)
HFP Doppler Temp. Coefficient (%/K)	$-1.692.10^{-3}$
HFP neutron generation time (μs)	27.7 (2.77E-05, s)
βeff (%)	0.58998

Table 4
SCRAM reactivity vs. time (Scenario).

Time after beginning of SCRAM (s)	Scram worth (% dk/k)
0.0	0.0
4.0	−6.5

2.2. Sequences of events and assumptions

The accident scenario is characterized by the following sequence of events and the assumptions listed hereafter:

a) Initiation events: SB LOCA and simultaneous SGTR
b) Switching off all four MCPs due to assumed SBO.
c) Actuation of the Reactor SCRAM after 0.4 + 1.2 s due to "Three of Four MCPs switched off". As consequence of this signal, all control rods drop in 2–4 s to the bottom of the reactor core.
d) The Main Isolating Valve (MIV) closes in 1 s due to electrical protection actuation (condenser vacuum loss) and in this way the turbine is isolated.
e) The Steam Dumping Device to the Condenser (BRU-Ks) are not available due to loss of condenser vacuum as there is a total station blackout (SBO).
f) The Make-up system stops 2 s after the SBO and the draining line (Let down system) is closed.
g) The Feed Water Pumps switch off after 5 s due to SBO.
h) The Pressurizer Heaters switch off.
i) Opening of BRU-As at 7.25 MPa and support the secondary pressure at 6.67 MPa. If the secondary pressure is reduced below 6.27 MPa the BRU-As will be closed. The SG safety valves (SVs) will open in case of BRU-As failure.
j) Total dry out of the PRZ is expected due to the loss of coolant through the breaks.
k) Loss of natural circulation is expected after the loss of the SGs heat transfer effectiveness.
l) Heat-up of the reactor core is expected to occur.
m) Beginning of hot leg dry out and start of core uncover is expected to occur.
n) Termination of calculation after reaching of core exit temperature above 1473 K.

3. Description of used computer codes and of the developed plant models

3.1. Short description of the system thermal hydraulic code RELAP5

The RELAP5/MOD3.3 computer code (RELAP5/MOD3.3 CODE MANUAL, 2016; Nuclear Safety Analysis Division, xxxx) has been used to simulate the transients for VVER-1000/V320 NPP model. The code models specific simulations of transients in LWR systems, e.g., the coupled behaviour of the reactor coolant system and the core for loss-of-accidents and, operational transients such as anticipated transient without scram, loss of offsite power, loss of feedwater, station blackout, turbine trip, and loss of flow.

RELAP5 is a highly generic code in addition to calculating the behaviour of a reactor coolant system during a transient, it can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and nonnuclear systems involving mixtures of steam, water, non-condensable, and solute. Control system and secondary system components are included to permit modelling of plant controls, turbines, condensers, and secondary feedwater systems.

3.2. Description of the VVER-1000 model developed for RELAP5

The model was developed at INRNE for analyses of operational occurrences, abnormal events, and design basis scenarios. In the VVER-1000 input model, the primary system has been modelled using four coolant loops, each one including one MCP and a horizontal SG. The thermal–hydraulic model configuration provides a detailed representation of the primary, secondary and safety systems. The more detailed description of RELAP5 VVER1000 model is given in (Grudev and Pavlova, 2004). The Steam Generator RELAP5 model and SGTR location is present in Fig. 1.

3.2.1. Modelling of a small break at primary circuit

The SBLOCA with the equivalent diameter of 30 mm (ID = 30 mm) is simulated on the main cold coolant loop #1 between the main coolant pump #1 (MCP) and the reactor vessel (RV) inlet as it is shown in Fig. 2 below. The break for SBLOCA is modelled as a "Single-Junction" RELAP5-component. It is connecting to the component "tmdpvol" # 253 with a crossflow junction. The choking model is used and the Henry-Fauske critical flow model is active. The nonhomogeneous flow model is applied.

The full abrupt area change model is applied and the code calculates forward and reverse Kloss terms.

The momentum flux options s = 0 is used, which means that it uses momentum flux in both directions: to volume and from volume. As a Henry-Fauske critical flow model is active the discharge coefficient is entered by default as a 1.0 and 0.14.

Table 5
Relative axial core power distribution.

Bottom					Top				
0.731	1.006	1.044	1.040	1.039	1.046	1.061	1.086	1.099	0.848

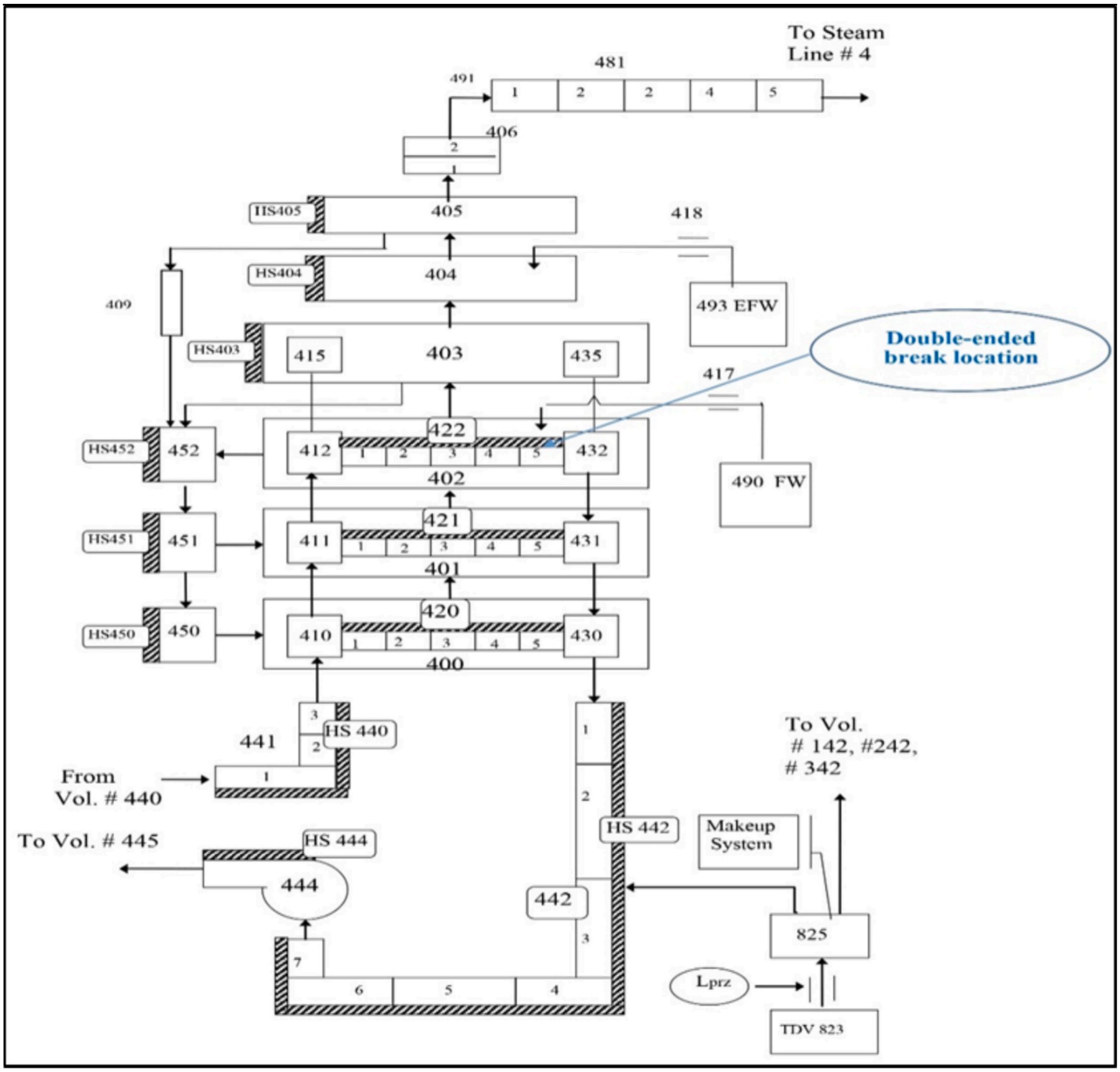

Fig. 1. Kozloduy steam generator #4 RELAP5 Model.

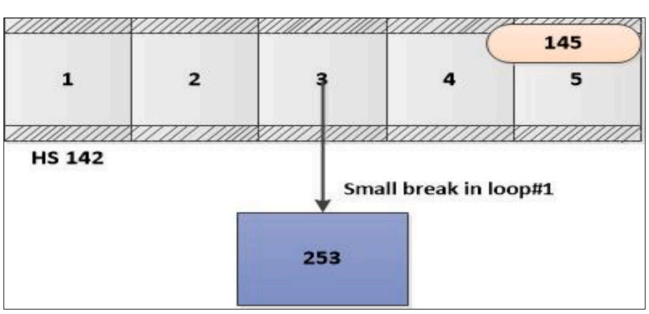

Fig. 2. Modelling of the break on cold loop #1.

3.2.2. SG tube break modelling

A single pipe # 423 with the same number of 5 sub – volumes is added to the heat structure of tubes that simulate heat transfer from primary to secondary sides in SG #4. The new single tube is located at the third level of the tubing structures in parallel to the tubes bundle of pipes # 422. To the pipe is added the heat structure for properly simulating heat transfer and flow rate.

The double ended break of the new single pipeline is simulated with two junctions: the first break is simulated on the end of the single pipe #423 from sub volume # 5 to the secondary side volume # 402 with a "Single-Junction" component; the second break is simulated from the cold collector #432 to the secondary side volume #402 as a "junction" as a part of a "BRANCH" component. The full abrupt area change model is applied for both breaks and the code calculates forward and reverse Kloss terms. The double ended break of the single pipeline in SG#4 with the equivalent diameters of 13 mm (ID = 13 mm) is activated simultaneously with the other initiating events.

3.2.3. Modelling of the BRU-As

The BRU-As are used in the simulation of the transient. They are modelled on each one steam lines between the SG and corresponding BZOK. The BRU-As begin to open after reaching their set opening point of 7.256 MPa and open fully in 15 s, and when the pressure drops below 6.277 MPa they closed in 15 s.

In the model of RELAP5, the BRU-A is modelled with two different types of valves combined in one common block connected with a pipe. The first one valve is a motor valve, which function is to open and close the BRU-A when it reaches the set points for opening and closing pressure in the corresponding steam lines. The second valve in the model is a servo valve and it supports the pressure of corresponding steam line at 6.668 MPa. When the BRU-A opens, steam starts to release to the atmosphere. The atmosphere is modelled with a "tmdpvol" RELAP5 component.

The all 4 BRU-As are considered operational during the whole transient.

3.3. Brief description of the system thermal hydraulic code TRACE

The system thermal–hydraulic code TRACE (U. NRC, TRACE V5., 2016) is a best-estimate system code of the U.S. NRC for the analysis of Light Water Reactor (LWR) and more recently extended for liquid metal cooled fast reactors. TRACE solves the static or time-dependent system of six conservation equations of a two-fluid mixture in 1D and 3D (Cartesian and Cylindrical coordinates) 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 (NRC, 2017).

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 conditions. In this approach, mechanical and thermal non-equilibrium situations are considered. Various models for components of an NPP e.g., pumps, valves, pipes, heat structures, as well as dedicated models for trips and control systems are also implemented in TRACE.

Two numerical methods, a semi-implicit method, and the SETS method are implemented in TRACE to solve any kind of slow and fast transients. 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]. Typically, system codes such as TRACE are coupled 3D (Zhang, 2019; Sanchez-Espinoza and Bottcher, 2006), and (Grudev and Pavlova, 2004; Ivanov et al., 2006) nodal diffusion solvers for the enhanced simulation of non-symmetrical transients in NPPs.

3.4. Description of the VVER-1000 model developed for TRACE

For the analysis of SBLOCA coincident with the SGTR, an integral plant model of the VVER-1000/V320 plant (Kozloduy, Unit 6) is developed, based on a previous VVER-1000/V320 RELAP5 model. The plant consists of four identical loops, each one consisting of primary and secondary side components. As it can be observed in Fig. 3, the beginning of the hot leg pipe and the end of the cold leg pipe are connected to the 3D-VESSEL component of TRACE, representing the reactor pressure vessel (RPV). Each loop is a network of 1D thermal hydraulic components of TRACE (PIPE, VALVE) to represent hot leg, the inlet plenum, horizontal tubes, the outlet plenum and the cold legs including the main coolant pumps. The flow conditions of the steam Generator on the primary side are modelled representing the flow through the tubes modelled, where the around 11,000 tubes lying horizontally are grouped in three groups at different elevations represented by a PIPE-component. There hot and cold collectors are also represented by different PIPE-components. On the secondary side, the large water volumes are also represented by three PIPE-components at different levels. The tubes itself are represented by HTSTR-components by cylindrical tubes, where the inner wall is connected with the PIPEs representing the primary coolant flow and the outer wall is connected with the secondary side volumes (PIPES). In this way, the heated-up primary coolant is cooled down along the SG-tubes and the heat is used to heat-up the secondary circuit until evaporation. At the upper part of the SG, a separator is considered and represented by SEPARATR-component of TRACE. There, the steam is separated from the entrained droplets and the liquid fraction flows back to the downcomer part of the SG.

In Fig. 3, the make-up system, the low and high pressure injection systems are also shown. There is also the location of the break in the primary loop 1 and of the break in the steam generator of loop-4 (third group of tubes). The wall of the steam generator and also the tubes are represented by the HTSTR-component of TRACE.

3.4.1. Break modelling of the cold leg and of the steam generator tube:

The modelling of the small break on the primary circuit loop1 and of the SG-tube is very important for the proper prediction of the outflow through the break and the evolution of the pressure on the primary/secondry systems. It is represented by a VALVE-component which is connected with the cold leg 1 and a BREAK-component that represent a mass sink. The similar approach is also followed for the break on the steam generator tube. For better comparisons with the simulations of the RELAP5-model, a RELAP5 motor valve type was selected. The choke flow model, the proper hydraulic diameter of the valve and edge firction

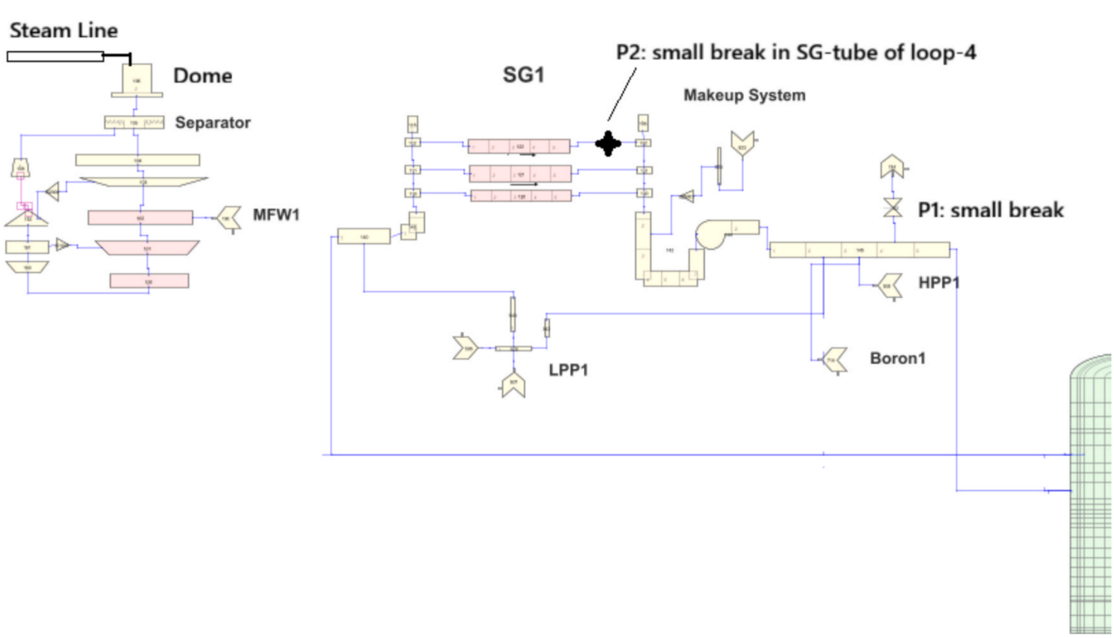

Fig. 3. TRACE Model of the loop 1 (primary/secondary circuits, safety systems) and location of the small breaks.

factors are provided and the internal additive valve flow loos model was activated. The connection between the cold leg node 4 is considered as crossflow with 90 degree angle where the offtake model is also activated. Similar to the RELAP5-model, a pipe and a HEATSTR are added to the steam generator to represent the fluid and the tube which is going to have a break while the original fluid volume and number of HEATSTR representing the group 3 is reduced by the fluid volume of one tube and by one tube. In this case, the Valve representing the break of one SG-tube is connected not to a BREAK-component but to the secondary side volume (#402). It means that after the break occurs, the primary circuit and the secondary circuit are connected to each other, Fig. 4.

3.4.2. 3D model of the RPV

Opossite to RELAP5, the RPV of the VVER-1000 plant is represented by a three-dimensional component: the VESSEL-component. It allows to discretize the RPV in three directions: axial, radial and azimuthal. In this case, the RPV was subdivided in 28 axial nodes, 6 azimuthal sectors and 6 radial sectors. The core is represented by three rings, the next ones represent the bypass, core barrel and downcomer. The VESSEL-component allows to consider the location of the lower and the upper grid plate as well as the upper head. The core itself is represented by 18 HTSTR-components to represent the 50,856 fuel rods contained in the 163 hexagonal fuel assemblies, Fig. 5. One sixth of the fuel rods are represented in each radial and azimuthal sector. The number of fuel rods represented in each HTSTR-component is described by the RDX-parameter (in this study RDX amounts 1924, 2808 and 3744). Each HTSTR must be axially and radially discretized so that the heat conduction equation can be solved for this discretization considering the heat source in the pellet material. A power component is also needed to represent the reactor power using different options e.g., as a table, or using the point kinetics model of TRACE and considering the provided reactivity coefficients. Also, the reactor SCRAM and the external reactivity to shut-down the reactor can be included in the POWER-component.

The geometrical and thermal hydraulic data as well as neutron kinetic information of the core were taken from the CAMIVVER-deliverables describing (Stefanova et al., 2020; Denis Stefanova et al., 2021) the transient scenario and the NPP-details.

3.4.3. Modelling of steam lines and feedwater lines including the BRU-As:

Fig. 6 shows the integral plant model including the RPV, Pressurizer, primary ans secondary loops (1, 2, 3 and 4) of the VVER-1000 plant. 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, where the secondary pressure as boundary condition is fixed.

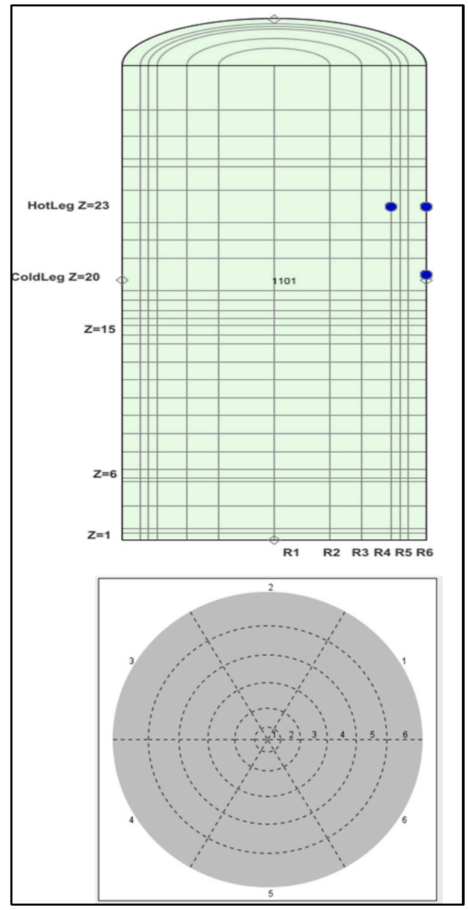

Fig. 5. Nodalization of the reactor pressure vessel (R, Z, Theta).

These valve types are considered in the steam lines: one steam dump valve to the atmosphere (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-SN) 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

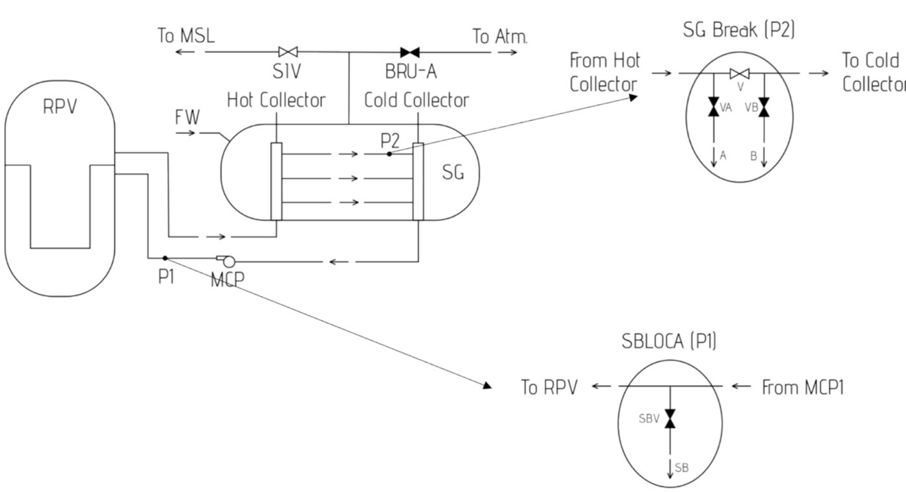

Fig. 4. Schematic representation of small break models in the cold leg of loop 1 (P1) and in the SG-Tube group 3 (highest elevation, P2) of the loop-4.

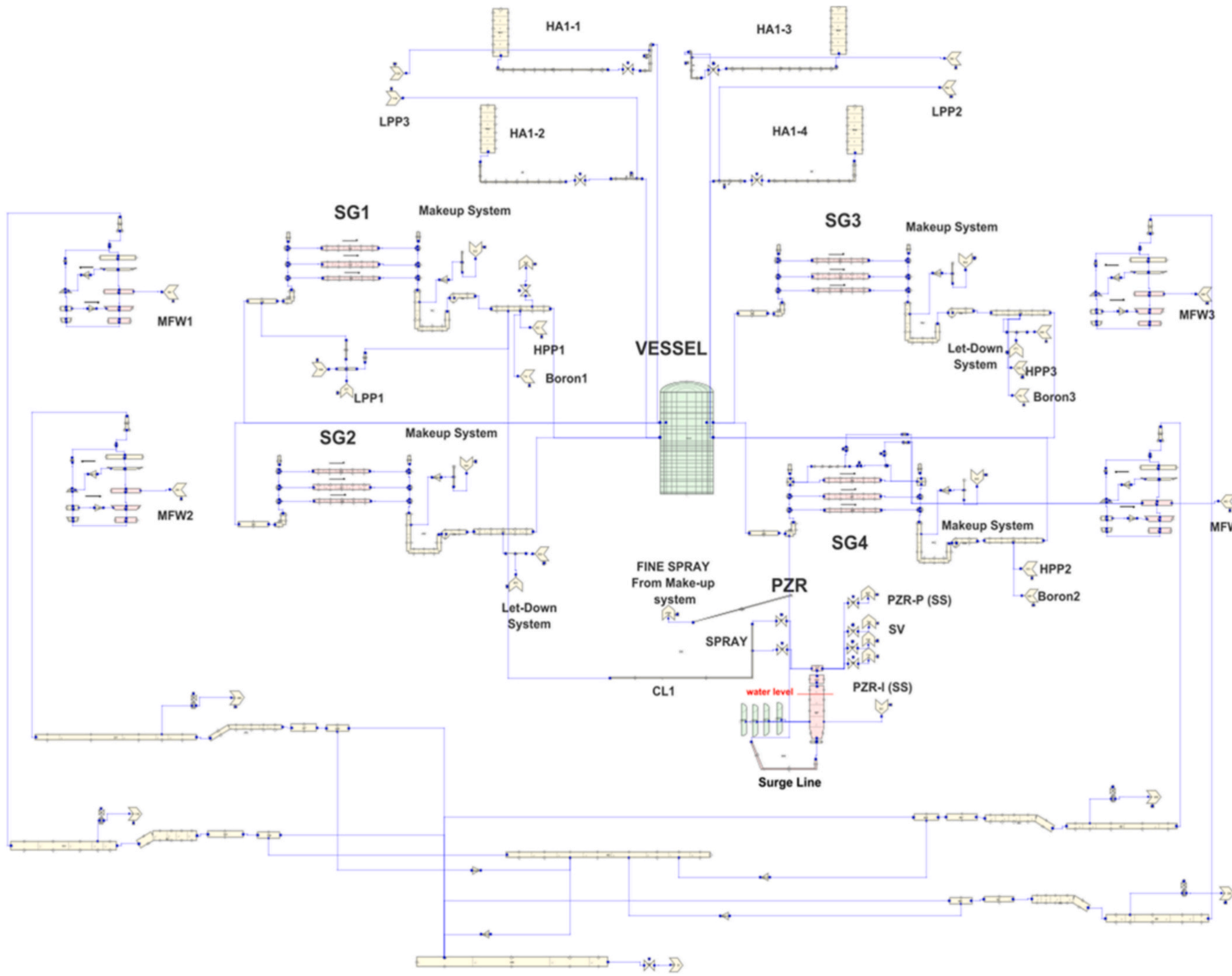

Fig. 6. Integral model of the Kozloduy Nuclear Power Plant (KNPP) developed for TRACE (primary/secondary circuits, safety systems).

conditions. As in RELAP5-model, only the BRU-As are activated during the transient. They connect the steam lines between the SG and BZOK-valve. The BRU-A-valves opens when their set point (7.256 MPa) is reached and in 15 s they are fully open. They are closed within 15 s if the pressure goes below 6.277 MPa.

3.4.4. *Safety systems and Pressurizer modeling:*

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 system (EFW) are included in the basic model. Control blocks are included in the model to facilitate the initiation of actions of the reactor protection and control system (RPCS) during the accident progression. In addition, the Pressurizer is also modelled by different 1D-volumes, HTSTR-component to represent the heater together with the POWER-component. It is connected to the cold leg #1 with the surge line and to the hot leg 4 with the spray lines.

Once the integral plant model to describe the stationary plant conditions is ready, the first step is to check if the model predicts the reference plant parameterts as good as possible, that are valid just before the transient takes place. A comparison of the stationary plant conditions with the one of the reference plant data was done and after acceptable values were predicted by TRACE (RELAP5/MOD3.3 CODE MANUAL, 2016) for the full power conditions.

4. Comparison of main results from analyses

First, the code predictions for the steady-state full power conditions just before the accident starts are compared to the reference values to assure that the codes are able to predict these important initial conditions of the plant. Then, the evolution of key thermal hydraulic parameters predicted by the two codes for the accidental phase are compared with each other.

4.1. *Comparisons of steady state between the codes*

The steady state plant parameters stabilization was performed, and the results were compared with the reference plant conditions to assure that the integral models are appropriate for the analysis of the transient plant conditions. The references plant data parameters and the steady state calculation results are presented in the Table 6.

It can be seen that the deviations of most parameters are very small for both calculations. Some of the parameters are slightly under predicted and others are slightly over predicted by participants.

This good agreement between references plant data and predictions

Table 6
Comparison of references plant parameters and the steady state calculation results during the "SB LOCA + SG line break" transient for KNPP unit 6.

Parameters	Plant Reference data	INRNE RELAP5/mod3.3	KIT TRACE
Reactor thermal power, MW	3000	3000	3000
Primary side pressure, MPa	15.7	15.7	15.55
Pressurizer water level, m	8.77	8.76	8.71
Average coolant temperature at reactor inlet, K	560.15	560.2	560.8
Average coolant temperature at reactor outlet, K	592.05	591.0	591.38
Mass flow rate through one loop, kg/s	4400.0	4395.9	4403.16
Pressure in SG, MPa	6.27	6.17	6.08
Pressure in the main steam header (MSH), MPa	6.08	6.03	5.62
Steam mass flow rate through SG steam line, kg/s	408	408.03	408.23
SG Water Levels, m	2.40	2.38	2.51
Liquid mass in the SG secondary side, t	48	48	48.2

for the steady state plant conditions before the transient demonstrate that the developed integral plant models are appropriate for subsequent analyses of the plant transients.

4.2. Code-to-code comparison of the accident progression

4.2.1. Code-to-code comparison of sequence of events

The calculated chronological sequence of events is presented in the Table 7. As the initial and boundary conditions used by both teams are very close or almost identical where it is possible.

4.2.2. Code-to-code comparison of selected parameters during the accident progression

In this subchapter, selected parameters predicted by the two codes for the same accidental sequence are compared to each other. Fig. 7 shows the comparison of the total reactor power predicted by RELAP5 (INRNE) and TRACE (KIT). There can be observed that the total power drastically decreases after SCRAM activation and later on it slowly decreases with the time. The agreement between the two predictions is very good.

The Fig. 8 presents the primary pressure evolution during the accident progression as calculated by both codes. After the initiation of the transient the primary pressure drops, due to the loss of the coolant through all the breaks (from primary circuit to the containment and from primary to secondary side).

As it is seen the predicted primary pressure in the first 600 s from the beginning of the transient follows a very similar trend. The main fluctuations between the predicted results by both codes are observed

Table 7
Chronological sequence of events calculated for the "SB LOCA + SG line break" transient.

Event description	INRNE Time, s	KIT Time, s
Time for total dryout of Pressurizer	430	393
Time for first opening of BRU-A in loop#4	8	175
Time for first opening of BRU-A in loop#1, #2, #3	8	175
Time for loss of natural circulation	3550	3540
Beginning of boiling in hot legs	630	500
Time for hot leg dryout	3598	3540
Time for beginning of core uncovery	4220	4901
Time for reaching 923 K – time for leaving EOPs and entrance in SAMG	7980	7779
Total dryout of reactor core	8200	8882
End of calculation at 1470 K	9400	9390

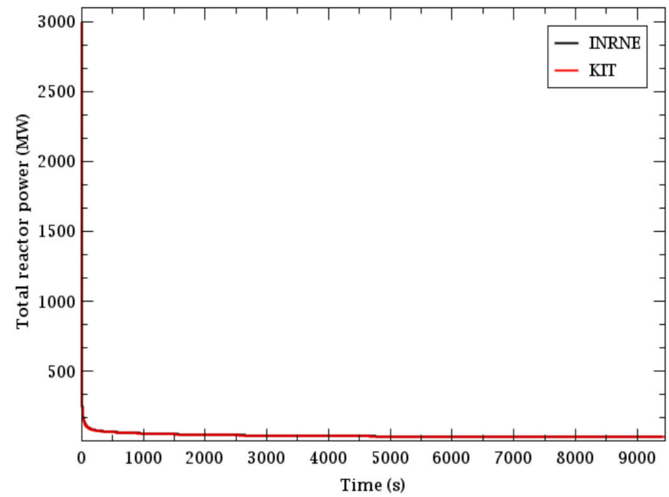

Fig. 7. Total reactor power.

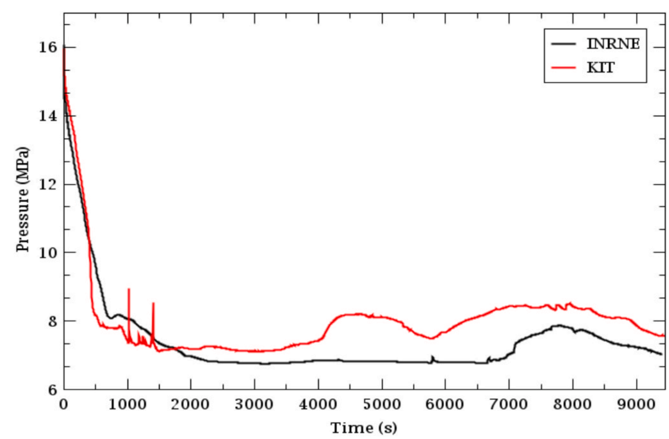

Fig. 8. Primary pressure (at core exit).

between 1000 s and 1450 s, it could be explained with the large changes of the coolant (liquid and vapor) mass leaving the small break, which is not seldom in such kind of simulations. The reliability of TRACE results during the entire transient process is assessed not only with respect to one parameter, but with respect to all the most important parameters describing the main ongoing phenomena. Overall, the global trends of the calculated results are qualitatively similar. Of course, the differences of the break modeling (Small break and the one in the SG-tubes) in RELAP5 and TRACE have an impact on the pressure reduction and loss of coolant (liquid and vapor) through the small break. In addition, water inventory at the beginning of the transient predicted by TRACE is slightly higher than the one of RELAP5 and this fact impacts also the heat transfer from the primary to the secondary side.

The pressure increase at different time during the transient process in both calculations is observed after 4000 s, due to the loss the ability of SGs to remove heat from primary circuits, as well, as due to the flow of coolant through the break. At this time, they are not sufficient to reduce the pressure due to the voiding in the brake at the corresponding models and break flow rates are reduced significantly.

The primary pressure increases after 4000 s for the KIT team, while in RELAP5 it is observed after 7000 s. The pressure increases in the primary circuit as the hot legs completely dry out. This leads to a loss of natural circulation, with further heat loss from the steam generators (SGs).

The comparison of secondary side pressure at MSH is shown in Fig. 9. As can be seen, in the first seconds of event, the primary temperature

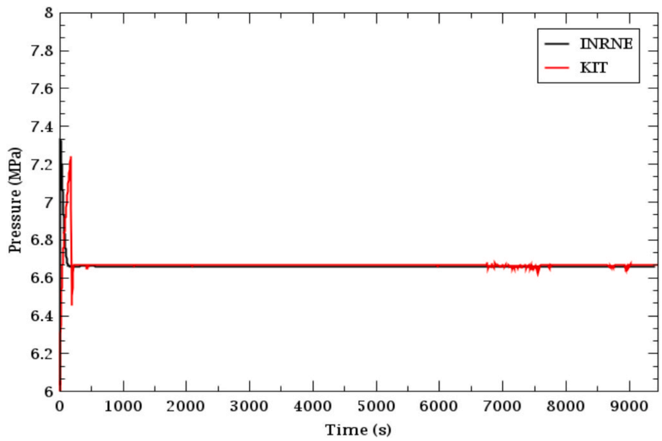

Fig. 9. Secondary pressure at MSH.

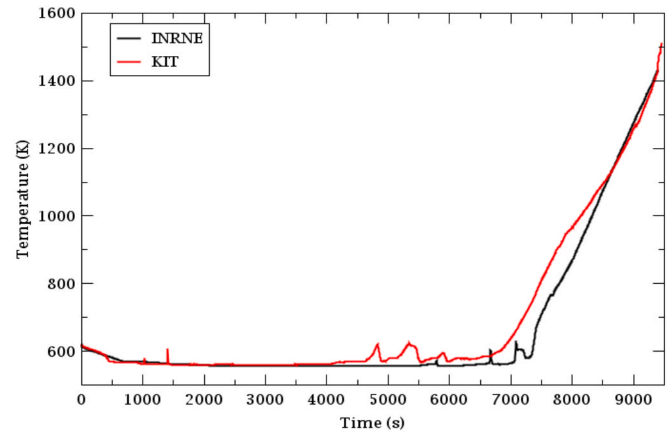

Fig. 11. Core exit coolant (gas) temperature.

and pressure drop, while the secondary side pressure shows a sharp increase. After the MSIV is closed upstream of the turbine, the secondary pressure starts to increase, which leads to the set point for opening of BRU-As. BRU-As open at 7.25 MPa and start to reduce the secondary pressure to the controlled pressure level of 6.67 MPa by the end of transient.

The Fig. 10 shows the comparison of core exit cladding temperature. As it is seen the behaviour of the core exit temperature is very close in both calculations. The temperature starts to increase sharply, after 7000 s and reaches to app. 1500 K in both calculations. The cladding temperature begins to increase after SGs dry out.

The Fig. 11 shows the core exit coolant (gas) temperature. The core exit coolant temperature sharply increases at 7000 s and reaches around 1500 K in both calculations. The coolant temperature starts to increase after SGs dry out.

The SB LOCA flow rates predicted by both codes are shown in Fig. 12. The loss of coolant on the primary side reduces the possibility of removing residual heat from the reactor core by the steam generators (SGs) on one side. Also, residual heat is removed by loss of primary coolant.

At the beginning of the transient, the flow rates sre very similar in the first 1000 s in both calculations. The SB LOCA flow rates show similar trends for both teams, with a rapid decrease in flow rate due to the voiding at the break in the primary circuit at around 4500 s in RELAP5, while in TRACE the flow rate decrease is after 5500 s.

The observed differences in the behaviour of break flow rates could be explained by using the different models for break simulations. The use of different assumptions in the development of the break flow model, as well as the use of different critical flow models, further contribute to

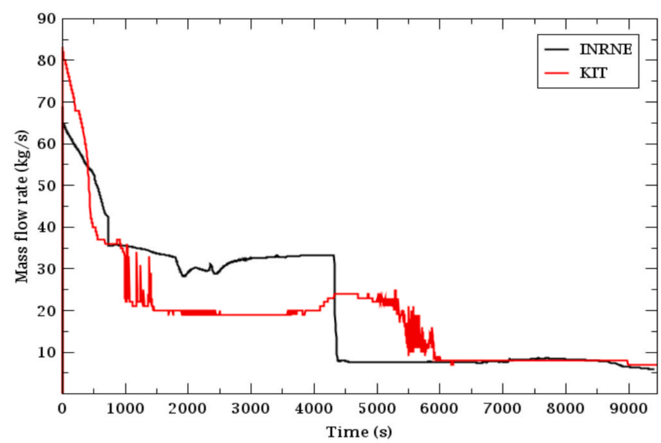

Fig. 12. SB LOCA flow rate.

the obtained deviations in prediction of deferential break flow rates. From the other side, the trends of the integral break flow rate have the same shape and very similar behaviour. In results of the break in the primary circuit, the hot legs and the upper part of the core start to dry up between 390 s and 650 s in both calculations.

The different options for modelling the break, can affect on the total liquid and vapor leaving the primary loop. These modeling issues impacts the pressure decrease on the primary side, but this is also influence by the heat transfer through the steam generators.

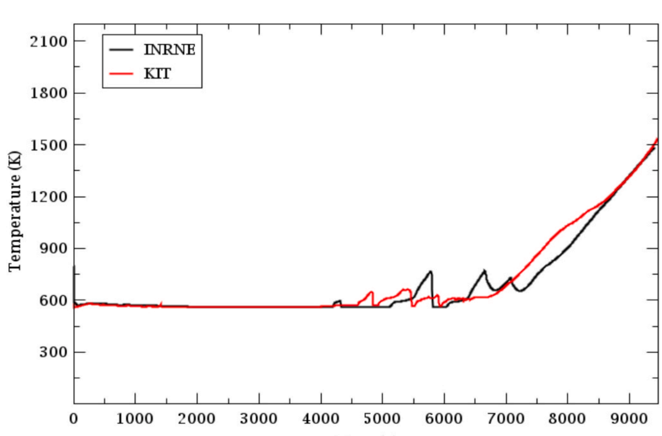

Fig. 10. Core exit cladding temperature.

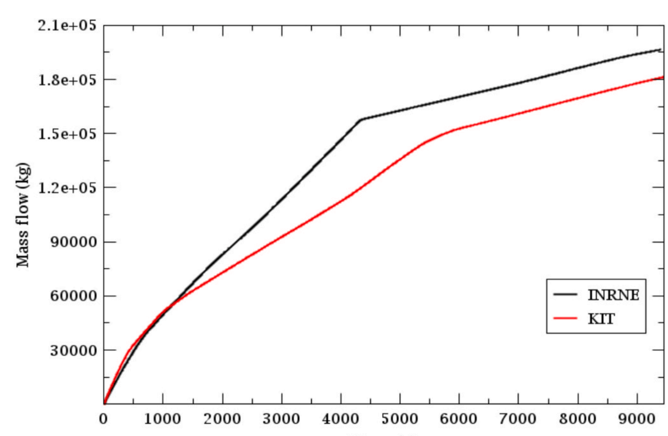

Fig. 13. Integral SB LOCA flow rate from primary circuit.

Fig. 13 presents a comparison of integral SB flow rate from primary circuit. The total mass of the coolant through the primary break has a similar shape for both calculations. The predicted loss of coolant through the break in the primary side in both calculations is very close during the first 1500 s and is about 67 t. Then the mass flow start to decrease in both calculations to approximately 30 kg/s at about 1500 s. After 1500 s until the end of transient it is observed small difference in the predicted mass flow from both codes.

Fig. 14 shows the primary to secondary side flow rates. In steam generator #4, the broken tube causes flow from primary to in the secondary circuit with a flow rate of approximately 12.5 kg/s total at the tube breaks in both cases. The mass flow rates from primary-to-secondary side have similar trends in both calculations. It can be seen that during the first 500 s, the flow rates from the primary to the secondary side have a similar behaviour. After 500 s, small deviations in the predicted values are observed until the end of the transient. RELAP5 predicts 0.57 kg/s at 9400 s at the end of transient, while TRACE predicts 0.98 kg/s at 9455 s.

The comparison of integral (cumulative) break flow rates from primary to secondary sides is given in Fig. 15. A similar trend can be seen in the predicted integral break flow rates of the primary to secondary side during the whole transient in both calculations. The predicted integral break flow rate from primary to secondary side reaches a total coolant mass is 13.3 t in RELAP5, while in TRACE is 16.4 t.

The loss of coolant inventory and the primary pressure decrease lead to decrease in the Pressurizer water level, which is shown in Fig. 16. A significant decrease in the pressurizer water level almost to the bottom is observed due to loss of coolant in both calculations during the first 400 s. The predicted results in both codes show similar behavior in the first 400 s. Complete dry out of the pressurizer occurs around 430 s in RELAP5, while in TRACE it occurs after 393 s. A significant increase in the pressurizer water level predicted by TRACE compared to RELAP5, between 3500 s and 9000 s, can be explained by return of water from the hot leg (in the surge line) and the bottom of the pressurizer. Also, the increase in the water level in pressurizers corresponds to an increase in primary pressure in both calculations.

The comparison of the Integral flow rate of BRU-As is presented in Fig. 17. Overall, the comparison of the calculated integral flow rate of all 4 BRU-As shows similar trends in both calculations. As can be seen, the integral flow rate through BRU-A predicted by both codes is almost identical during the whole transient. The flow rate of BRU-A starts to increase after reaching the set points for their opening. Overall, the predicted flow rate of BRU-A predicted by TRACE is slightly underestimated compared to RELAP5. The integral flow rate of all 4 BRU-As reaches 188 t in RELAP5 and 190 t in TRACE at the end of the calculations.

The comparison of predicted Steam generators#4 water levels by two codes is presented in Fig. 18. Nevertheless, that the initial mass of water in SGs in both models are almost equal, after the initiation of transient, in the first 100 s is observed increasing of mass of water in the TRACE model, while in RELAP5 model the SGs water mass is reduced, which reflect on the behavior of both models to the end of transient. The initial deviations in behavior of the two SG models could be explained by the use of different condensation models. In general, for whole transient, as it is seen from the results, the water level in SG#4 decreases

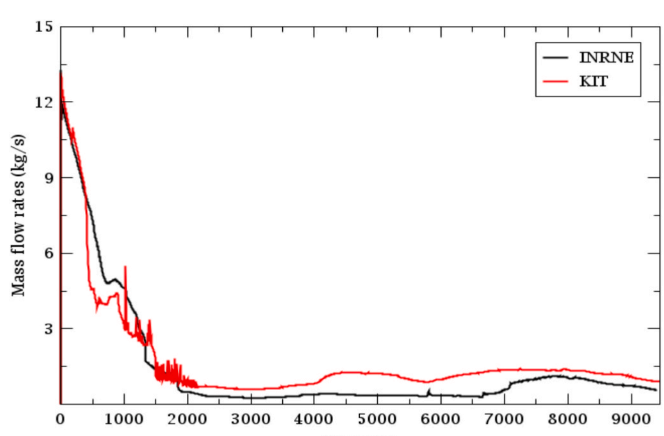

Fig. 14. Primary to secondary side flow rates (in total).

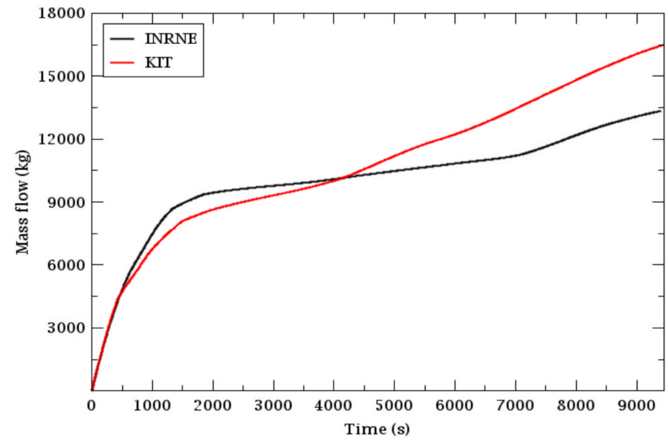

Fig. 15. Integral break flow rate from primary to secondary side.

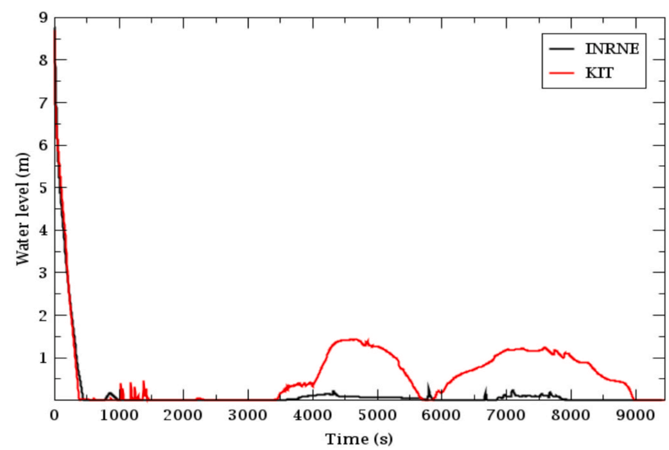

Fig. 16. Pressurizer water level.

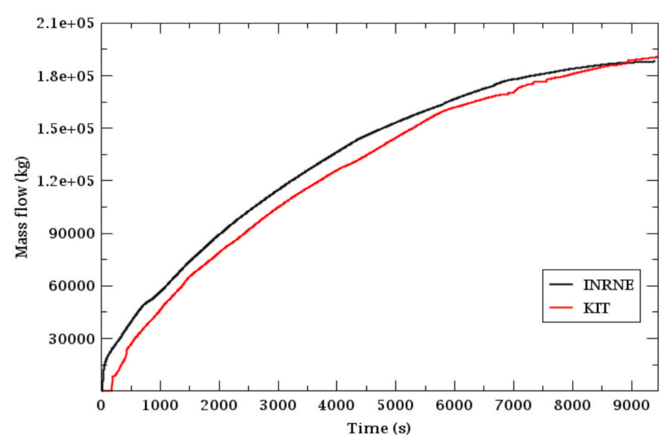

Fig. 17. Integral BRU-As flow rate.

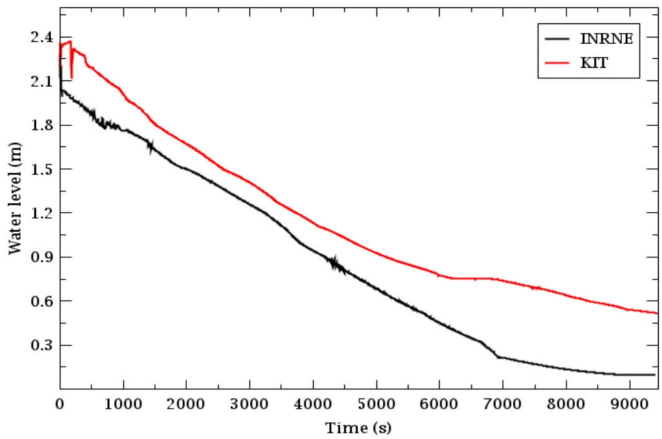

Fig. 18. SG#4 water level.

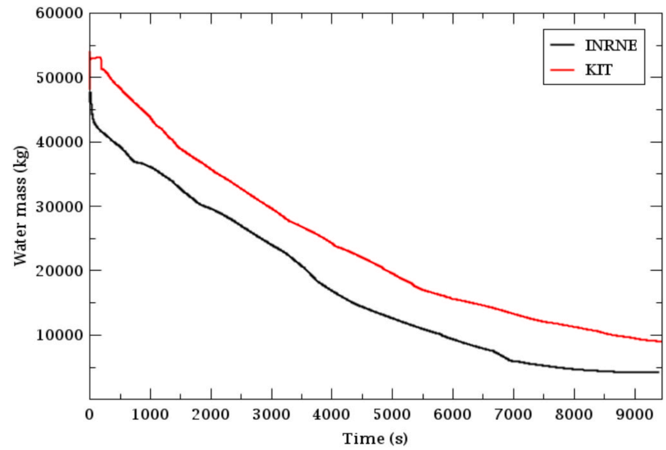

Fig. 19. SG#4 water mass.

significantly in both models, due to the loss of feed waters (as a result of SBO). The comparison shows that the Steam generators (SGs) have not completely dried out in both codes due to the loss of primary coolant. The observed increase in water level in TRACE results at the beginning of the transient could be explained with the observed condensation in the secondary side, as mentions above. At the beginning of the accident, the feed water switches off to all steam generators and decay heat is removed from the core by natural circulation. Steam generators water levels predicted by both codes has similar trends until 2500 s. Because physics of the steam release from the secondary circuit, the level in the steam generators (SGs) decreases quickly. The residual heat is removed from the reactor core and primary side by natural circulation as long as SGs are effective, as well as through the breaks. The loss of natural circulation occurs at 2850 s in RELAP5 and 3540 s in TRACE (Table 6).

The comparison water mass of SG#4 is presented in Fig. 19. The residual heat is removed from the reactor core and primary side by natural circulation, while the SGs are efficient. The loss of natural circulation is observed as a result of the loss of primary coolant and the reduction of the SGs water mass.

Due to the loss of feed waters (such as SBO), the water mass of the steam generator decreases significantly. The water mass of the steam generators has the same shape until the end of calculation in both calculations. As can be seen from the comparison, there is a significant underestimation of the SG#4 water mass and SG dry out predicted by TRACE code, compared to REALP5.

5. Conclusions

A complex accidental sequence for the VVER-1000 was analyzed by two system thermal hydraulic codes, RELAP5 and TRACE. The main difference between the plant models is the use of the VESSEL- component in TRACE which allows a 3D representation of the reactor pressure vessel compared to the 1D representation in RELAP5. By comparing the accident progressions predicted by both codes it can be concluded that trends of all parameters are very similar as well as the simulated process of transition from forced to natural circulation, dry out of pressurizers, integral break flow rates etc. From the other side there are some deviations that are observed and explained.

The developed scenario is based on three independent initiating events that happen simultaneously and have different contributions to processes development as well as on the parameters' behaviour.

The main phenomena expected during the transient were observed and investigated, such as the decrease of the level in the pressurizer and in the SGs; the opening of the BRU-As because of the high pressure in the secondary circuit; the loss of natural circulation because of the dry up of the hot legs; heat up of the reactor core, etc. The instants when the mass flows at the small break and the broken tube switch from water to steam can be identified clearly.

Two phases can be distinguished during this transient: before and after the dry-up of the hot legs. While there is still a natural circulation of liquid water in the hot legs (and the rest of the primary circuit), the pressure and temperature of the primary circuit remain rather stable after the initial events. However, once natural circulation is lost, the primary circuit fails to evacuate the residual power and the level in the core starts to decrease, which leads to a sharp increase in temperature at the very end of the transient.

The expected reversing of primary to secondary flow rate was not observed in any of the calculations as the break flow rate was too small for reducing the primary pressure compared to the primary pressure supported by residual heat. Observed deviations in returning of water level in Pressurizer during the accident progressions were explained, but still need further investigation.

To further improve the code predictions, additional efforts should be made to ensure that the steady-state predictions of the two codes are closer to each other, as deviations already present in the steady-state simulations will propagate through the transient phase. Furthermore, the modeling of break in RELAP5 and TRACE should be chosen in such a way that they are comparable to each other. Only in this way is it guaranteed that the break model will lead to similar pressure drops and leakages upon interruption. Furthermore, it should be ensured that the secondary side water levels of all steam generators already in the steady-state simulation are equal to the reference values, for example by using a control function.

Despite of all observed deviations, the comparison of calculated results from code-to-code comparison demonstrates a reasonable agreement between the obtained results. The gathered knowledge will contribute to further model improvements and better understanding the phenomena and process arising during the accident progressions.

Declaration of competing interest

The authors declare the following financial interests/personal relationships which may be considered as potential competing interests: Antoaneta Stefanova reports financial support was provided by Horizon 2020. If there are other authors, they declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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Data availability

The data that has been used is confidential.

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