Codes And Methods Improvements

for VVER comprehensive safety assessment

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D7.3 Results of transient-1 LOCA benchmark

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Summary

This Deliverable D7.3 titled “Results of transient-1 LOCA benchmark.” is a part of CAMIVVER Project, WP7, Task 7.3 “Transient-1 benchmark: “SB LOCA + SG line break” transient.

The aim of the benchmark is to demonstrate the capabilities of the models developed in Task 7.3 to properly simulate a small break loss of coolant (SB LOCA) simultaneously with a guillotine double ended break of a single pipe in one of the SGs in case of total loss of power. The task is performed as a benchmark by comparing the results obtained by simulating the selected multiple failures. The performing of such a task will increase the capabilities of the used codes in simulation of the accident analyses. The selected combination of the events will allow to investigate important phenomena and processes during the accident progression.

The important plant parameters, calculated by different codes are used in organizing the benchmark and it will help in developing and updating models for RELAP5, TRACE and CATHARE3 computer codes. In this way, the objective of Task 7.3 is to check the capabilities of the integral computer codes TRACE and CATHARE3 to produce consistent results compared to RELAP5, which is widely used for safety assessment of VVER reactors in preparation of the SAR, PSA L1, EOPs, as well as resolving other important issues connected with safety improvements.

Task 7.3 will also allow the evaluation of improvements related to the 3D T/H and 3D NC models for further using of CATHARE3 and TRACE with further inclusion of 3D vessel modelling.

Task 7.3 participants are:

- INRNE with RELAP5 Mod 3.3 code
- FRAMATOME with CATHARE3 code
- KIT with TRACE code
- ENERGORISK with RELAP5 Mod 3.2 code

Approval

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Table of contents

List of Tables

List of Figures

Abbreviations

AASSL	Actuation of Automatic Step by Step Load
ACRR	Annular Core Research Reactor
AC/DC	Alternating current / Direct current
AFW	Auxiliary Feed Water
AKHP	Russian Abbreviation for “Automatic Control for Neutron Flux”
ARM	Reactor Power Controller
BB - 440 kV	High Voltage breaker
BDBA	Beyond Design Basis Accident
DBA	Design basis accident
DG	Diesel Generator
BOL	Beginning of Life
BOC	Beginning of cycle
BRU - A	Steam Dump to Atmosphere
BRU - K	Steam Dump to Condenser
BRU - SN	Steam Dump Facility for House Load
BST	Block of shielding tubes
BZOK	Fast Acting Cut-off Valve
CAMP	Code Applications and Maintenance Program
CSF	Critical safety functions
ECCS	Emergency Core Cooling Systems
EOL	End of Life
EOC	End of fuel cycle
EOPs	Emergency Operating Procedures
FW	Feed Water
HA	Hydroaccumulator
HE	Hydrodynamic Element
HPP	High Pressure Pump
ICAP	International Code Assessment and Applications Program
LOFT	Loss-of-fluid Test
MCL	Main Circulation Loop
MCP	Main Coolant Pump
MCR	Main Control Room
MIV	Main Isolating Valve
MSH	Main Steam Header
MSIV	Main Steam Isolation Valve
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRU	National Research Universal

PBF	Power Burst Facility
PRZ	Pressurizer
PWR	Pressurizer Water Reactor
ROM	Reactor Power Limitation Controller
RV	Relief Valve
SAMG	Severe Accident Management Guidelines
SB LOCA	Small Break Loss of Coolant Accident
SBO	Total Station Blackout
SG	Steam Generator
SHS	Submersible Perforated Sheet
SCRAM	Emergency shutdown of the reactor (Safety control rod assembly moving)
SGTR	Steam Generator Tube Rupture
SVs	Safety Valves
TQ12, 22, 32	Low pressure injection pumps
TQx2	Low pressure system injection
TQx3	High pressure system injection
TQx4	High-High pressure system injection (piston type)
TQ40S	Valve for connection of HRS to primary side
TQ40S08, 09	HRS safety valves
UVC	Control and Computing System
VVER	Water-Water Cooled Reactor
WP-1	Warning Protection #1

1 Introduction

The report has been prepared in the frame of the CAMIVVER Grant agreement, NUMBER 945081 [1]. This Deliverable D7.3 titled “Results of transient-1 LOCA benchmark” is a part of the CAMIVVER Project, WP7, Task 7.3 “Transient-1 benchmark: “SB LOCA + SG line break” transient.

The initiated event of the “SB LOCA + SG line break” benchmark is a Small Break Loss of Coolant Accident (SB LOCA) with 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 guillotine double ended steam generator tube line break in the SG#4 is assumed, 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 (PRICE) is induced in the scenario from the beginning of the transient. Simultaneously with both initiating events, a station blackout (SBO) is assumed, which will on the one hand simplify the transient and also will lead to more severe conditions. Involving a total loss of internal and external electrical power will allow to observe 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)) and transition from the forced to the natural circulation.

Involving a PRICE event in the scenario with a SB LOCA allows to investigate the reversing of the flow rate from the secondary side with not borated water to primary side. The choice of such a size of primary break was based on reducing the primary pressure to the pressure of the secondary side but not below that of the activation of hydro accumulators (HAs) to avoid possible borating of primary circuit.

The results obtained by the participants are used to prepare code-to-code comparisons with subsequent analyses. This benchmark shows that all codes are able to simulate adequately the selected transient.

The report describes RELAP5/TRACE/CATHARE3 benchmark ‘’SB LOCA+SG line break’’ scenario for the VVER-1000, performed by four teams. The reference nuclear power plant (NPP) is Kozloduy NPP Unit 6, equipped with a VVER-1000 / V320 reactor type. All geometric data and plant specific equipment characteristics are presented in the Deliverable D3.2 [2]. In the transient, plant parameters for the end of fuel cycle #8 are used. The initial and boundary conditions are presented in this document. The selected combination of events allows important phenomena and processes during the accident progression to be investigated.

Task 7.3 participants are:

- INRNE with RELAP5 Mod 3.3 code

- FRAMATOME with CATHARE3 code

- ENERGORISK with RELAP5 Mod 3.2 code

- KIT with TRACE V5P5.1150 code

2 Description of the “SB LOCA + SG line break’’ Benchmark

The benchmark examines the capability to simulate the transition from forced to natural circulation, the dryout of pressurizer, the dryout of hot and cold legs, the dryout of SGs, the loss of natural circulation, the reactor core heat up, as well as the integral effects important for safety assessment, through a code-to-code comparison.

Based on the selected initiating events and initial and boundary conditions, heat removing from primary reactor core and primary circuit by the secondary side by activating BRU-As (steam dump to atmosphere) as long as the SGs are available is investigated. Simultaneously with a secondary side heat removing, the generated reactor core heat is removed by the break until voiding the break. After losing the capability of removing the reactor core residual heat, the consecutive dry out of hot and cold loops are simulated. Further, it is simulated a heat up of the reactor core and the calculation continues to 1200 – 1500 K, where the design based integral codes are capable of correctly predicting plant behaviour.

The chosen set of the initiating events allows one to simulate various important phenomena and processes and check the code's capabilities through code-to-code comparisons.

The calculated important plant parameters by different codes are used in organizing a benchmark and it will help in developing and updating models for RELAP5, TRACE and CATHARE3 computer codes. The objectives of the Task 7.3 are to check the capabilities of the integral computer codes TRACE and CATHARE3 to produce consistent results compared to RELAP5, which is widely used for safety assessment of VVER reactors in preparation of the SAR, PSA L1, EOPs, as well as, resolving other important issues connected with safety improvements.

Task 7.3 will also allow evaluating improvements related to the 3D T/H and 3D NC models for further using of CATHARE3 and TRACE with further including 3D vessel modelling.

The results from the Benchmark analysis were compared code-to-code. The report provides the results of the comparative assessment of four independent analyses of the Kozloduy NPP, Unit 6,''SB LOCA+SG line break '' transient.

2.1 Main phenomena during the benchmark exercise

The main phenomena observed during the execution of a transient benchmark:

- ✓ loss of core cooling due to loss of primary coolant, loss of MCPs and loss of heat sink due to SBO;
- ✓ loss of primary inventory;
- ✓ depressurization of primary circuit
- ✓ dryout of Steam Generators (SGs);
- ✓ dryout of pressurizer;
- ✓ reactor core heat removing by secondary side through work of BRU-As;
- ✓ transition from forced to natural circulation;
- ✓ loss of natural circulation, due to voiding of hot legs;
- ✓ heat up of the reactor core.

Combination of two small break leakages: one from a primary cold leg and the other from primary to secondary side and involving SBO make this analysis complicated from one side and interesting from the other side due to the possible challenging of many critical safety functions (CSF) as:

- ✓ Loss of Core Cooling
- ✓ Loss of Primary Inventory
- ✓ Loss of Heat Sink

Selecting a SB LOCA initiating event with SBO could lead to the loss of primary coolant without reducing the primary pressure to the set point of activation of passive safety systems HAs, which will create a condition for core heat up with further damaging at significantly high pressure at above 60 bars.

Usually during SB LOCA, it is observed a plateau when primary and secondary pressures are coupled. The performing of transient with involvement of natural circulation will allow checking of computer models for their capabilities to simulate reactor core heat removing through the SGs with activation of BRU-As. Completely dryout of the SGs demonstrates the loss of heat sink through the secondary side.

The initiating of loss of coolant allows comparison of simulation of differential and integral break flow rates from primary side, as well as from primary to secondary side.

The other important parameters that will be compared are the reduction of the pressurizer water level, as well as SGs water levels. Keeping in mind that in the scenario, a break of pipe is considered in one of SGs (SG#4), the comparison will examine two different behaviours of SGs water level reductions.

The simulation of the secondary side pressure behaviour will be compared, too. The activation of BRU-As will allow to investigate the capabilities of the developed computer models for this safety system to properly control and support secondary side pressure.

If the SB LOCA occurs, the reduction of primary pressure below the secondary side is possible to happen without further reduction of primary pressure below the set point of activation of HAs. If it happens, it will lead to a reversing of flow rate from the damaged SG to the primary circuit and will challenge the reactor subcriticality.

The developed transient scenario based on the selected initiating events combines two design basic accidents, such as small break loss of coolant accident and primary to secondary leakage with a beyond design basis accident (BDBA) such as total station blackout, where all the onsite and offsite alternating current (AC) electric power failed.

Table 2.1.1 List of important parameters for comparison code-to-code

No	**Name of parameter**	**Dimension**
	Primary side and secondary side	
1.	Total reactor power	MW
2.	Primary pressure (at core exit)	MPa
3.	Secondary pressure at MSH	MPa
4.	Core exit cladding temperature	K
5.	Core exit coolant (gas) temperature,	K
6.	Small break (SB LOCA) flow rate	kg/s
7.	Integral SB LOCA flow rate from primary circuit	kg
8.	Primary to secondary side flow rates	kg/s
9.	Integral break flow rate from primary to secondary side	kg
10.	Pressurizer water level	m
11.	BRU-As flow rate	kg/s

No	Name of parameter	Dimension
12.	Integral BRU-As flow rate	kg
13.	Temperature cold leg#1	K
14.	Temperature cold leg#2	K
15.	Temperature cold leg#3	K
16.	Temperature cold leg#4	K
17.	Temperature hot leg#1	K
18.	Temperature hot leg#2	K
19.	Temperature hot leg#3	K
20.	Temperature hot leg#4	K
21.	SG #1 water level	m
22.	SG #2 water level	m
23.	SG #3 water level	m
24.	SG#4 water level (damaged SG)	m
25.	SG #1 water mass	kg
26.	SG #2 water mass	kg
27.	SG #3 water mass	kg
28.	SG #4 water mass	kg
29.	Coolant flow rate in hot leg#1	kg/s
30.	Coolant flow rate in hot leg#2	kg/s
31.	Coolant flow rate in hot leg#3	kg/s
32.	Coolant flow rate in hot leg#4	kg/s
33.	Coolant flow rate in cold leg#1	kg/s
34.	Coolant flow rate in cold leg#2	kg/s
35.	Coolant flow rate in cold leg#3	kg/s
36.	Coolant flow rate in cold leg#4	kg/s
37.	Heat transfer of SG #1	MW
38.	Heat transfer of SG #2	MW
39.	Heat transfer of SG #3	MW
40.	Heat transfer of SG #4	MW

2.2 Methodology of comparison

This report contains the comparison of the results obtained by the integral analyses of four different models using three integral computer codes, where the RELAP5 is presented by two versions. The analysis discusses the computer codes ability to reproduce key phenomena and processes in the simulated transient.

Keeping in mind that for selected initiating events, there is no plant data for VVER – 1000, all calculations have been performed by representing teams without reference plant data for the investigated scenario and because of that the obtained results should be considered as a blind exercise. For reducing the uncertainty during the comparison, the scenario has been widely discussed and all possible assumptions for activation/deactivation of different systems have been included in the list of events in the scenario. In addition to that, before performing the analyses, the initial and boundary conditions have been discussed and accepted. They are described in the following subsection 2.3.

To address the challenges in the developed scenario, four teams developed and updated models for the computer codes RELAP5, CATHARE and TRACE.

The teams have selected appropriate models that correctly reproduced the expected processes and phenomena during the development of the accident based on their experience (assumptions) and code specific recommendations.

The performed benchmark is based on a simulation of a combination of DBA events such as Small Break Loss of Coolant Accident (SB LOCA) and primary to secondary side leakage (PRICE) and BDBA as Station Blackout (SBO). 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.

Exploring the codes' capability to simulate the plant's response in a scenario involving multiple failures is very important. Code-to-code comparison when accident progression data are missing is a valuable strategy to increase the accuracy of simulation of phenomena and processes from the codes. Such a comparison is appropriate for analysing and better understanding accident progression, as well as a good tool for increasing codes fidelities.

The benchmark carefully examines the potential to simulate critical phases such as transition from forced to natural circulation, coolant dry-out in both hot and cold legs, steam generators (SGs) dry-out, loss of natural circulation, heat up of the reactor core. Special attention is paid to the comparison of differential and integral break flow rates results as an important part of the accident's progression. This overall assessment is done by carefully comparing different codes, ensuring that their performance is consistent with respect to these key aspects of the analysis.

The selection of the code models for critical flow, which is used by four teams, has an important influence on the accident progressions during the transient and it is also discussed in the report.

It has been discussed a list of important parameters to be selected in order to better understand accident progressions. A list of important parameters for code-to-code comparison is presented in subsection 2.1.

The following list of integral computer codes have been chosen by the four teams:

- INRNE with RELAP5 Mod 3.3 code; FRAMATOME (FRA) with CATHARE3 code; KIT with TRACE V5P5.1150 code; ENERGORISK (ER) with RELAP5 Mod 3.2 code.

2.3 Initial and boundary conditions

The selected initiating event 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. Simultaneously is initiated a double ended break of one pipeline in SG#4 with the equivalent diameter of 13 mm (ID=13 mm). The SG pipe line break is assumed near the cold collector in the upper part of the tubes bundle (elevation 1.8-2.0 meters from the bottom of SG).

Plant design data is taken from D3.2 – "The CAMIVVER Definition report with specification for NPP with VVER 1000 reactor with respect to selected transients" [2].

The important reactor parameters for the initial state analysis are presented below:

- ✓ The reactor state is at the end of 8-th cycle (EOC).
- ✓ The reactor power is 100%.
- ✓ The initial SGs water mass is 48 000 kg.
- ✓ Boron concentration is 53 ppm which corresponds to the end fuel cycle.
- ✓ 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 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 loosing 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 nominal (100%) reactor power system parameters (operational data) are represented in Table 2.3.1.

Table 2.3.1 Initial plant design conditions

Parameter	Plant Design
Reactor thermal power, MW	3000.0
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
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	408.0
SG Water Levels, m	2.40
Liquid mass in the SG secondary side, t	48.0

The reactor physics parameters for point kinetics analysis are presented in the Table 2.3.2, Table 2.3.3 and Table 2.3.4.

Table 2.3.2 Decay constants and fractions of delayed neutrons at EOC

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 2.3.3 Point kinetics parameters for Unit 6 of Kozloduy NPP, End of Cycle 8

Parameter	**Data from KNPP (this data is used for current calculation)**
HFP MTC (%/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 2.3.4 Scram reactivity vs. time (Scenarios)

Time after beginning of scram (s)	**Scram worth (% dk/k)** Scenario
0.0	0.0
4.0	-6.5

The initial HP core average axial relative power distribution is presented in Table 2.3.5.

Table 2.3.5 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

The logics of work for BRU-As is presented in Table 2.3.6.

Table 2.3.6 Steam dump to atmosphere (BRU-A) logics

Description	Value	Dimensions
Time to fully open/close BRU-A	15	s
BRU-A opens if:	P_{sg}>74 (7.256)	kgf/cm^2 / (MPa)
Pressure controller switches on and keeps the pressure 68 kgf/cm^2 if:	P_{sg} =68 (6.668)	kgf/cm^2 / (MPa)
BRU-A closes if:	P_{sg}<=64 (6.276)	kgf/cm^2 / (MPa)
Flow rate through a fully open BRU-A, P=68 kgf/cm^2 (6.668 MPa)	250	kg/s

2.4 Transient scenario

The transient scenario is as follows:

1. Initiation events: SB LOCA (ID 30) on main cold coolant loop #1 between MCP and RV inlet along with a total station blackout at 0.0 s. Simultaneously is initiated a double ended break of one pipeline in SG#4 (ID=13 mm). It is assumed the break is near the cold collector in the upper part of the tubes bundle (elevation 1.8-2.0 meters from the bottom of SG)
2. Switching off all four MCPs due to SBO.
3. Actuation of the Reactor SCRAM after 0,4+1,2 s due to “Three of Four MCPs switched off” and after this signal all control rods drop in 2-4 s to the bottom of the reactor core.
4. 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.
5. The BRU-Ks are not available due to loss of condenser vacuum as there is a total station blackout (SBO).
6. The Make-up system stops 2 s after the blackout and the draining line (Let down system) is closed.
7. The Feed Water Pumps switch off after 5 s due to SBO.
8. The Pressurizer Heaters switch off.
9. Opening of BRU-As at 74 kgf/cm2 (7.256 MPa) and beginning to support secondary pressure at 68 kgf/cm^2 (6.668 MPa). If the secondary pressure is reduced below 64 kgf/cm^2 (6.276MPa) the BRU-As will be closed. The SG SVs will open in case of BRU-As failure.
10. Total dryout of PRZ is expected due to the loss of coolant.
11. Loss of natural circulation is expected after the loss of the SGs heat transfer effectiveness.
12. Reactor core heat up is expected.
13. Beginning of hot leg dryout is expected as well as beginning of core uncovery.
14. Termination of calculation after reaching of core exit temperature above 1200 ºC (1470 K).

3 MODELS DESCRIPTION

3.1 General Code Features

Table 3.1.1 Information of the codes used and contact persons

Institution	Code	Contact person (s)
INRNE	**RELAP5/ mod 3.3**	**VRYASHKOVA Petya, GROUDEV Pavlin**
FRAMATOME	**CATHARE3**	**BERNARD Olivier, MAS Alexandre**
KIT	**TRACE-V5P5.1150**	**D. SANCHEZ-ESPINOZA Victor Hugo (INR)**
LLC ENERGORISK	**RELAP5/ mod 3.2**	**HASHYMOV Artur, ONYSHCHUK Yurii**

3.1.1 INRNE

RELAP5/MOD3.3 computer code has been used to simulate the transients for VVER-1000/V320 NPP model [3]. 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, noncondensable, and solute. Control system and secondary system components are included to permit modelling of plant controls, turbines, condensers, and secondary feedwater systems.

The model was developed at INRNE-BAS for analyses of operational occurrences, abnormal events, and design basis scenarios. The actual four-loop system was modelled by four single loops for primary and secondary sides. The model provides a significant analytical capability for the specialists working in the field of NPP safety.

3.1.2 FRAMATOME

The code used by Framatome is CATHARE (Code for Analysis of Thermal-Hydraulics during an Accident of Reactor and Safety Evaluation) and is developed to perform best-estimate calculations of pressurized water reactor accidents: PWR loss of coolant (large or small break, primary and secondary circuit).

CATHARE includes several independent modules that take into account any two-phase flow behaviour:

- Mechanical non-equilibrium:
  - vertical: co- or counter-current flow, flooding counter-current flow limitation (CCFL), etc.
  - horizontal: stratified flow, critical or not critical flow co- or counter-current flow, etc.
- Thermal non-equilibrium: critical flow, cold water injection, super-heated steam, reflooding, etc.
- All flow regimes and all heat transfer regimes.

In order to take into account these phenomena the CATHARE code is based on a two-fluid and six equations model with a unique set of constitutive laws. Various modules offer space discretization adapted to volumes (0D), pipes (1D) or vessels (3D) ready to assemble for any reactor description.

3.1.3 KIT

The system thermal-hydraulic code TRACE 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.

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 [4]. 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 nodal diffusion solvers for the enhanced simulation of non-symmetrical transients in NPPs. At KIT, multi-scale coupling approaches for TRACE with CFD and subchannel codes are being developed [5] based on the ICoCo-Method [6].

3.1.4 LLC ENERGORISK

The RELAP5 code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. The code models the coupled behaviour of the reactor coolant system and the core for loss-of-coolant accidents and operational transients such as anticipated transient without scram, loss of offsite power, loss of feedwater, and loss of flow. A generic modelling approach is used that permits simulating a variety of thermal hydraulic systems. Control system and secondary system components are included to permit modelling of plant controls, turbines, condensers, and secondary feedwater systems.

The MOD3 version of RELAP5 has been developed jointly by the NRC and a consortium consisting of several countries and domestic organizations that were members of the International Code Assessment and Applications Program (ICAP) and its successor organization, Code Applications and Maintenance Program (CAMP) [7].

The RELAP5/MOD3 code is based on a nonhomogeneous and nonequilibrium model for the two- phase system that is solved by a fast, partially implicit numerical scheme to permit economical calculation of system transients. The objective of the RELAP5 development effort from the outset was to produce a code that included important first-order effects necessary for accurate prediction of system transients but that was sufficiently simple and cost effective so that parametric or sensitivity studies were possible.

The code includes many generic component models from which general systems can be simulated. The component models include pumps, valves, pipes, heat 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 noncondensable gas transport.

The system mathematical models are coupled into an efficient code structure. The code includes extensive input checking capability to help the user discover input errors and inconsistencies. Also included are free-format input, restart, renodalization, and variable output edit features. These user conveniences were developed in recognition that generally the major cost associated with the use of a system transient code is in the engineering labor and time involved in accumulating system data and developing system models, while the computer cost associated with generation of the final result is usually small.

RELAP5 represents the aggregate accumulation of experience in modelling reactor core behaviour during accidents, two-phase flow processes, and LWR systems. The code development has benefitted from extensive application and comparison to experimental data in the LOFT, PBF, Semiscale, ACRR, NRU, and other experimental programs [7].

3.2 Modelling of the NPP primary and secondary side

3.2.1 INRNE model description

In the RELAP5 model for VVER-1000/V320 NPP reactor vessel; core region represented by three channels; pressurizer system including heaters, spray and relief valves; safety system including high-pressure pumps, four accumulators and low-pressure injection pumps are included. In the model, it is also presented the make up /drain system including connection (control) with pressurizer. The secondary side is developed and is presented by eight SG safety valves, four BRU-A valves, BRU-K valves, steam pipe lines (including main steam header) and turbine including regulating valve in front of the turbine. The horizontal steam generator (SG) has been modelled. The model of natural circulation in horizontal SG has been presented in this RELAP5 VVER-1000 model. A separator model and the perforated sheet have been modelled in SG model, too. Main cooling pump (MCP) has been developed using homologous curves from real pumps.

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 reactor vessel model includes a downcomer, lower plenum, and outlet plenum.

The INRNE team uses in RELAP5/mod 3.3 VVER 1000 model, the ANS79-1 Standard data for calculating of fission product decay power.

The pressurizer (PRZ) system includes heaters, spray, and pressurizer relief capability. The safety system representation includes accumulators, high and low pressure injection systems, and the reactor scram system. The model of the make up and blow down systems includes the associated control systems. RELAP5 heat structures are used to represent 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 coolant to the water of the secondary side is modelled using heat structure components.

The Reactor and the Pressurizer RELAP5 Model is shown schematically in Figure 3.2.1.1. The Steam Lines RELAP5 Model is shown schematically in Figure 3.2.1.2.

Secondary side is modelled with four loops which included in total: four horizontal steam generator (SG); four steam pipe lines connected to common main steam header; eight SG safety valves; four BRU-A valves; four BRU-K valves; turbine including regulating valve in front of the turbine. The separator model and the perforated sheet have been modelled in the SG models, too. The Kozloduy Steam Generator RELAP5 model is present in Figure 3.2.1.3.

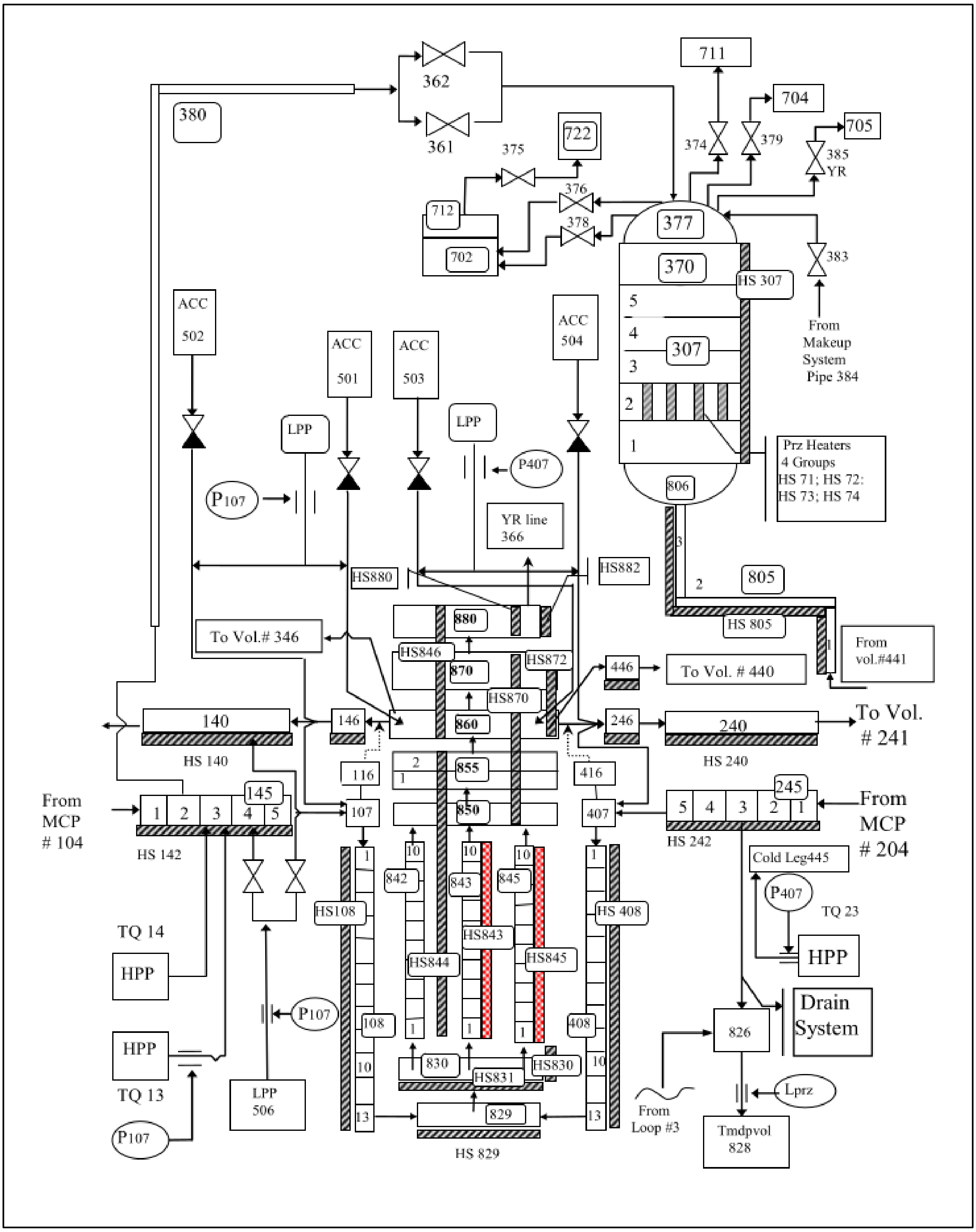

Figure 3.2.1.1 Kozloduy VVER-1000 Reactor and Pressurizer RELAP5 Model

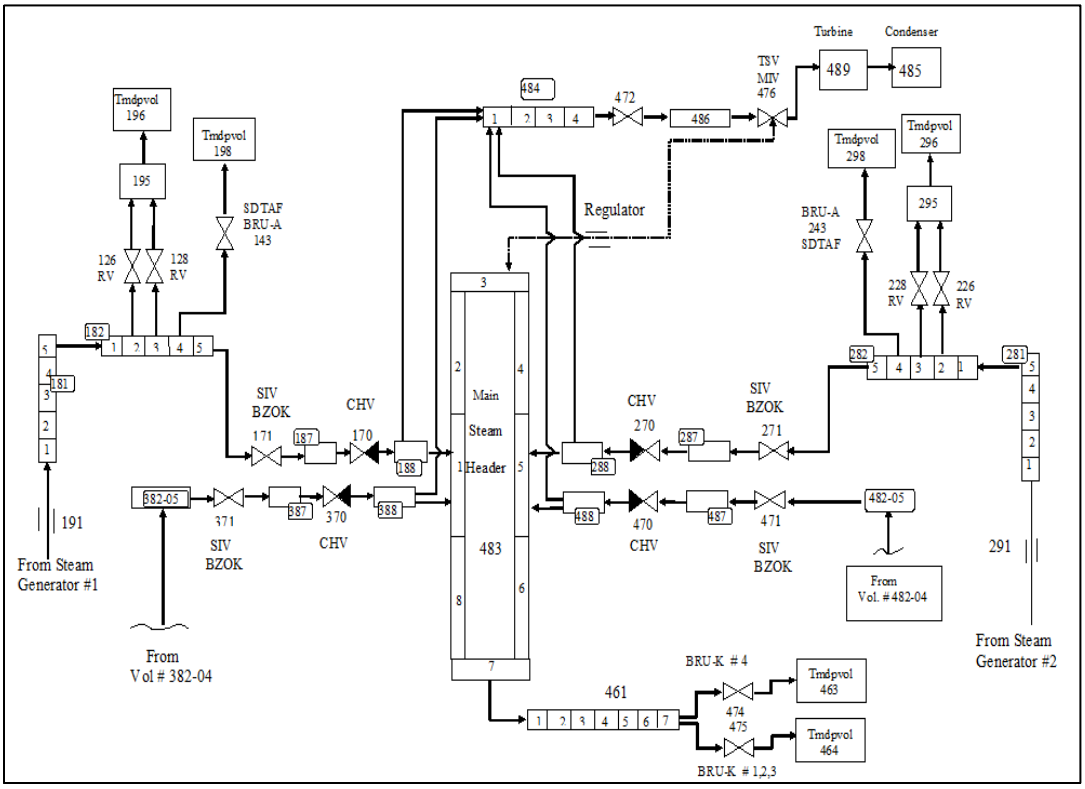

Figure 3.2.1.2 Steam Lines RELAP5 Model

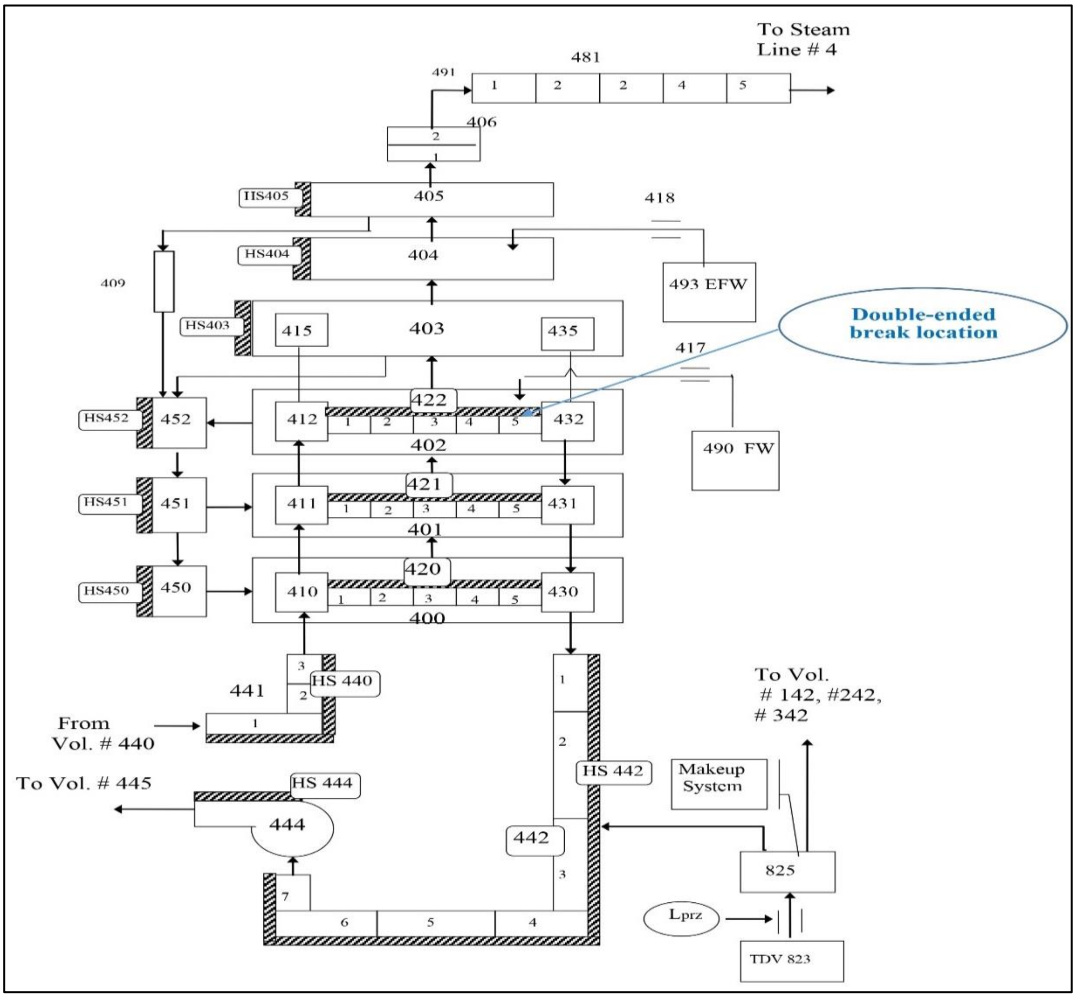

Figure 3.2.1.3 Kozloduy Steam Generator #4 RELAP5 Model

3.2.1.1 Modelling of a small break at primary circuit

The break for SBLOCA is modelled as a “Single-Junction” component.

The SB LOCA 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 Figure 3.2.1.1.1 below. 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.

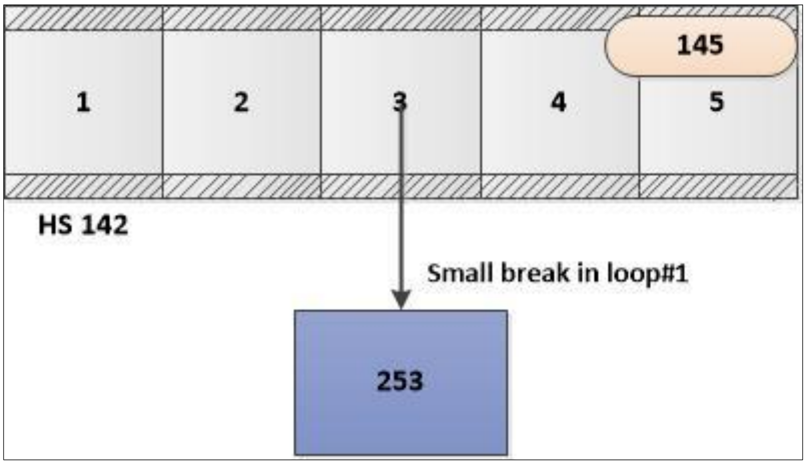

Figure 3.2.1.1.1 Break on cold loop n#1

3.2.1.2 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 the 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

3.2.1.3 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 start to open after reaching their set point for opening at 7.256 MPa and fully open for 15 s, and when the pressure drops below 6.277 MPa are closed for 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” component.

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

3.2.2 FRAMATOME model description

This section describes the modelling assumptions and nodalization for the development of a CATHARE3 model for VVER 1000, Unit 6 KNPP. The model was defined to include all major systems of the Kozloduy NPP.

In the CATHARE3 model of the VVER -1000, the primary system has been modelled using four coolant loops representing the four reactor loops. THE CATHARE3 model configuration provides a detailed representation of the primary, secondary, and safety systems. In the CATHARE3 VVER-1000 model, the secondary system has been modelled using four steam lines and four steam generators.

The nodalization is shown on the following figure, presenting only the loop #1. The safety systems are not presented. A more detailed presentation of the model can be found in the deliverable 7.1.

Besides, several modifications are made to model the transient. These data were taken from the Deliverables 3.2 and 3.3 respectively in references [2] and [8].

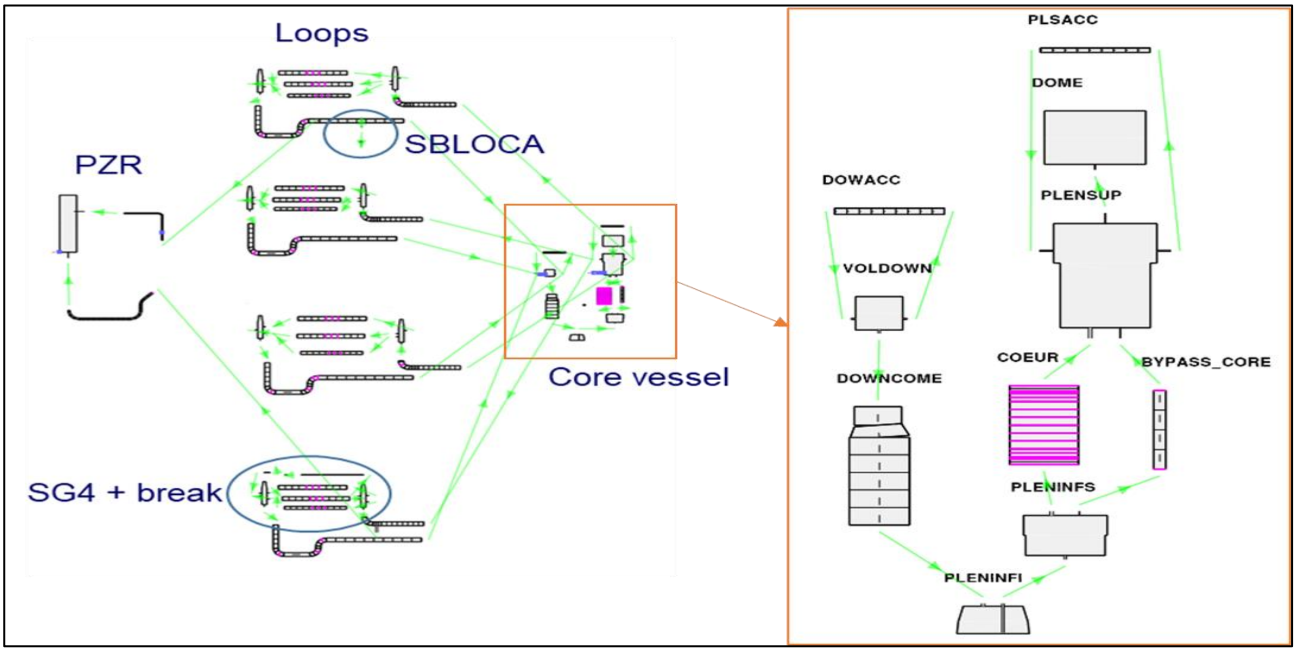

Figure 3.2.2.1 Primary circuit model in CATHARE3

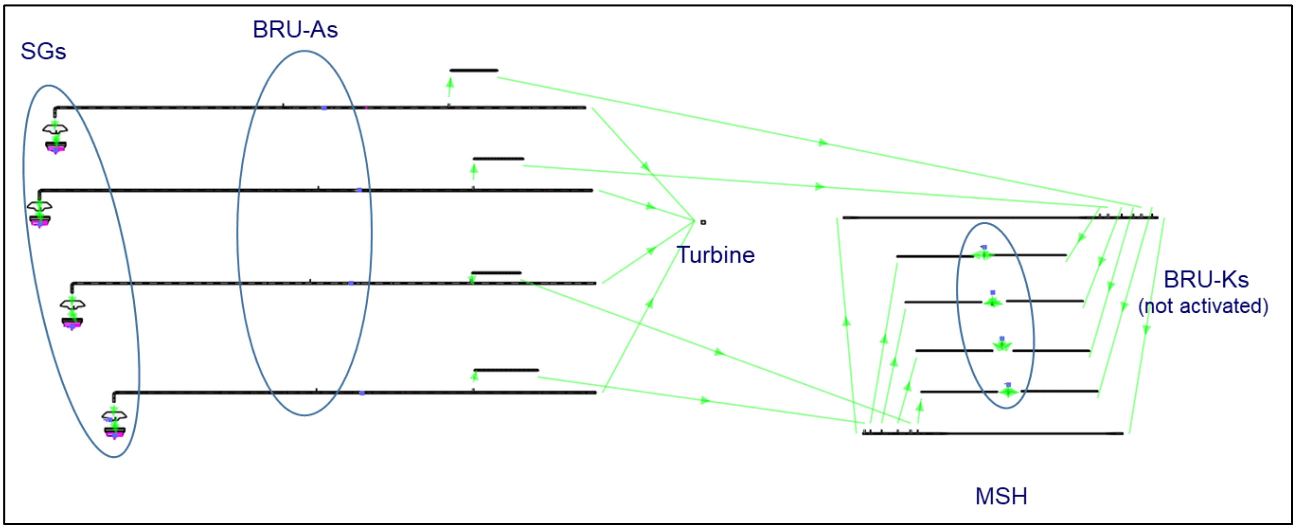

Figure 3.2.2.2 Secondary circuit model in CATHARE3

3.2.2.1 Small break on cold leg #1

The break is modelled thanks to a tee with one branch perpendicular to the axis of the cold leg. The extremity of the tee is set as a "blind" condition in the first phase of the calculation to reach the initial state of the transient, so no mass flow is possible out of the cold leg. Then, at the start of the transient, this boundary condition is changed into a pressure condition of 0.1 MPa. After that, fluid can leave the primary circuit through this boundary condition, potentially reaching critical velocity when steam is released.

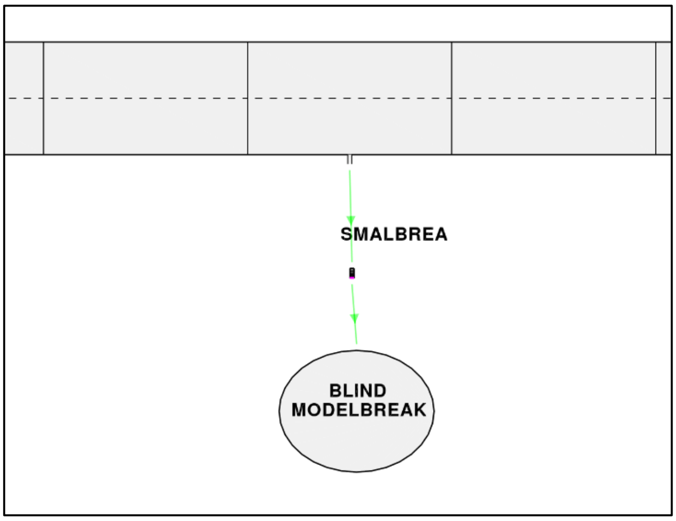

Figure 3.2.2.1.1 Break on cold loop #1

3.2.2.2 SG tube break

A tube in the SG #4 is broken at the start of the transient. To do so, it had to be isolated from the upper bundle of tubes modelled in the previous tasks. As a reminder, the tube in the SGs were grouped in 3 bundles (upper, middle and lower) to simplify their modelling. Here, the tube is opened at the start of the transient and water is then able to circulate between the primary circuit (at both ends of the broken tube) and the secondary circuit (the secondary side of the SG #4). As a consequence, should the pressure in the primary circuit become lower than that of the secondary circuit, water or steam from the secondary circuit would be able to flow into the primary circuit.

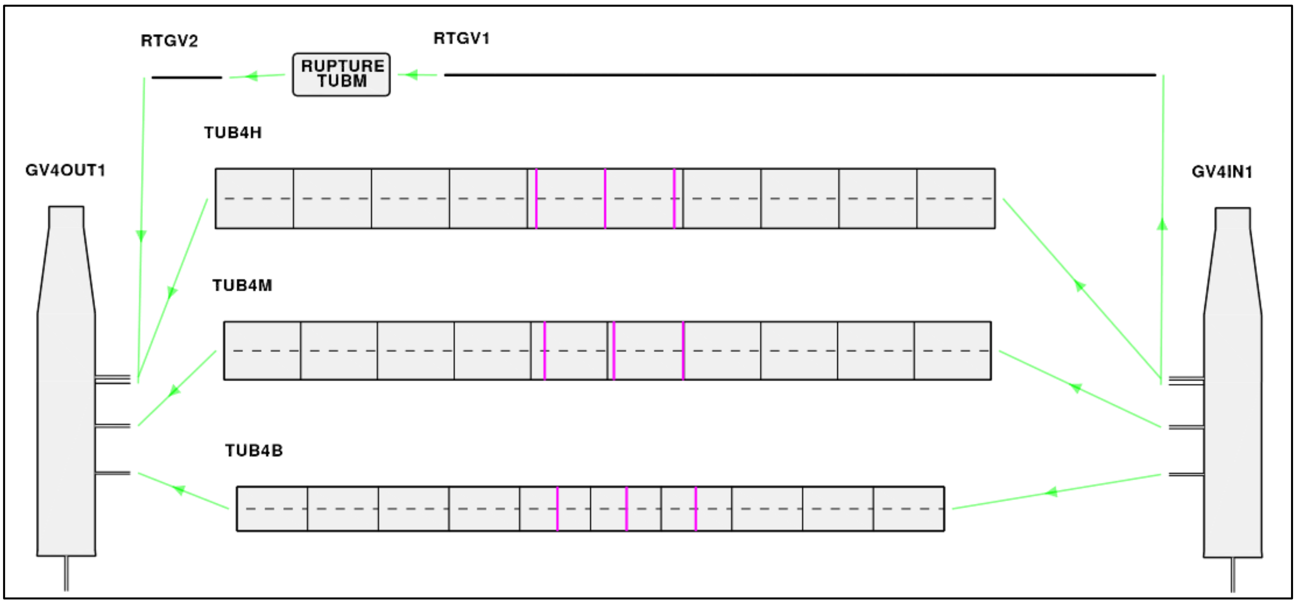

Figure 3.2.2.2.1: SG #4 tube break

3.2.2.3 Station blackout

The station black-out incurred few modifications of the model, mainly the switching off of several systems such as:

- ✓ The main coolant pumps (MCP) right from start of the transient, which leads to the SCRAM shortly after;
- ✓ The heaters in the pressurizer;
- ✓ The feed water pumps of the SG.

3.2.2.4 Addition of the BRU-As

The BRU-A are also added in the model, one on each steam line. They have a capacity of 900 t/h under a pressure of 66.7 bar (6.67 MPa). They start to open in 15 s when the pressure reaches 72.6 bar (7.26 MPa), and they start to close when the pressure drops below 62.8 bar (6.28 MPA) in 15 s too. The boundary condition at the extremity of the BRU-As is set as a pressure outlet at 1 bar (0.1 MPa).

However, it should be noted that before the transient, while reaching the initial state, the BRU-As cannot open whatever happens regarding the secondary pressure. When the transient starts, this condition changes and BRU-A are free to open to release steam if the pressure threshold is met. Besides, they are also set in the model in a way that allows to consider a critical flow of steam. The 4 BRU-A are considered operational during the whole transient.

3.2.2.5 Addition of the safety injection

3.2.2.5.1 Hydro Accumulators (HA)

Two accumulators are added in the CATHARE model. They have a volume of 60 m^3 and they activate if the pressure in the primary circuit goes below 59 bar (5.9 MPa). However, the accumulators are not expected to activate in this transient.

3.2.2.5.2 High-pressure safety injection

The high-pressure safety injection is not activated in this transient due to the SBO.

3.2.2.6 Check valves on steam lines

Check valves were added to the steam lines to prevent a reverse flow of steam from a SG to another. This way the steam can only circulate from the SG towards the BRU-As or the main steam header (MSH).

3.2.2.7 SCRAM

The reactor trip happens 1.6 s after stopping the MCPs at the beginning of the transient, reducing the core power. The rods drop in about 3 s and bring an anti-reactivity of 6.41%.

3.2.2.8 Residual power

Due to the modelling of the residual power in CATHARE 3, it is difficult to set it precisely without dedicated data. Thus, the residual power calculated by the INRNE was used as input data for CATHARE 3 as a table.

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3.2.3 KIT model description

For the analysis of the Small Break LOCA (SBLOCA) coincident with the Steam Generator Tube Rupture (SGTR), where it is assumed that one tube located in the upper group of tubes is broken, an integral plant model of the VVER-1000/V320 plant (Kozloduy Unit 6) is developed, based on a previous VVER-1000/V320 RELAP5 model [9].

The four loops are represented separately with 1D thermal hydraulic components of TRACE (PIPE, VALVE) consisting of the hot legs, the steam generator, the cold legs, and the main coolant pump. 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. 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. 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-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 conditions.

The flow conditions of the steam Generator on the primary side is modelled representing the flow through the tubes modelled, where the around 11000 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.

The Reactor Pressure Vessel (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 50856 fuel rods contained in the 163 hexagonal fuel assemblies. One six 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 in Figure 3.2.3.1.

A power component is also needed to represented 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.

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.

Finally, a hug number of signal and Trip variables, and 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.

The geometrical and thermal hydraulic data as well as neutron kinetic information of the core were taken from the CAMIVVER-deliverables describing the transient scenario [10] and the NPP-details [11].

In the first step, a reference plant model is developed to describe the steady state plant conditions at full power just before the transient takes place, see Figure 3.2.3.2.

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 [12], [13] for the full power conditions, the simulation of the transient was done.

For this purpose, the model was extended with the include

- The SBLOCA initiation at the primary loop-1,
- The small break of the SG-tube on the steam generator loop-4
- The SCRAM at time 0 s,
- The insertion of the control rods (POWER component),
- The isolation of the turbine by closing of the turbine stop valve,
- The witch-off of the main coolant pumps since station black out is assumed, and
- The switch-off of the feedwater lines since station black out is assumed.

All other actions such as opening or closure of the different valves on the SG-secondary side were considered already in the base model for the stationary plant simulation [14].

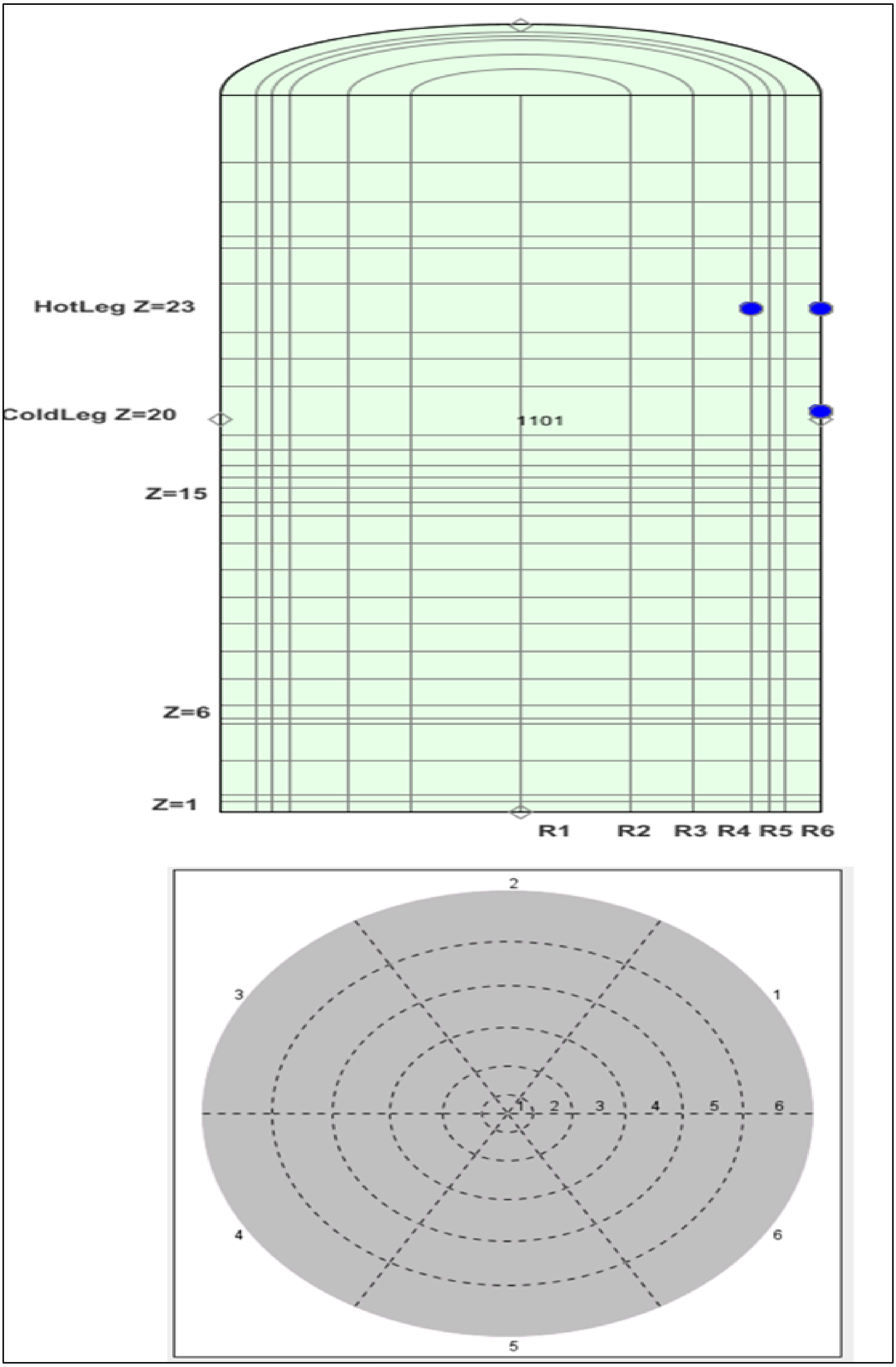


Figure 3.2.3.1 Nodalization of the Reactor Pressure Vessel (R, Z, Theta)

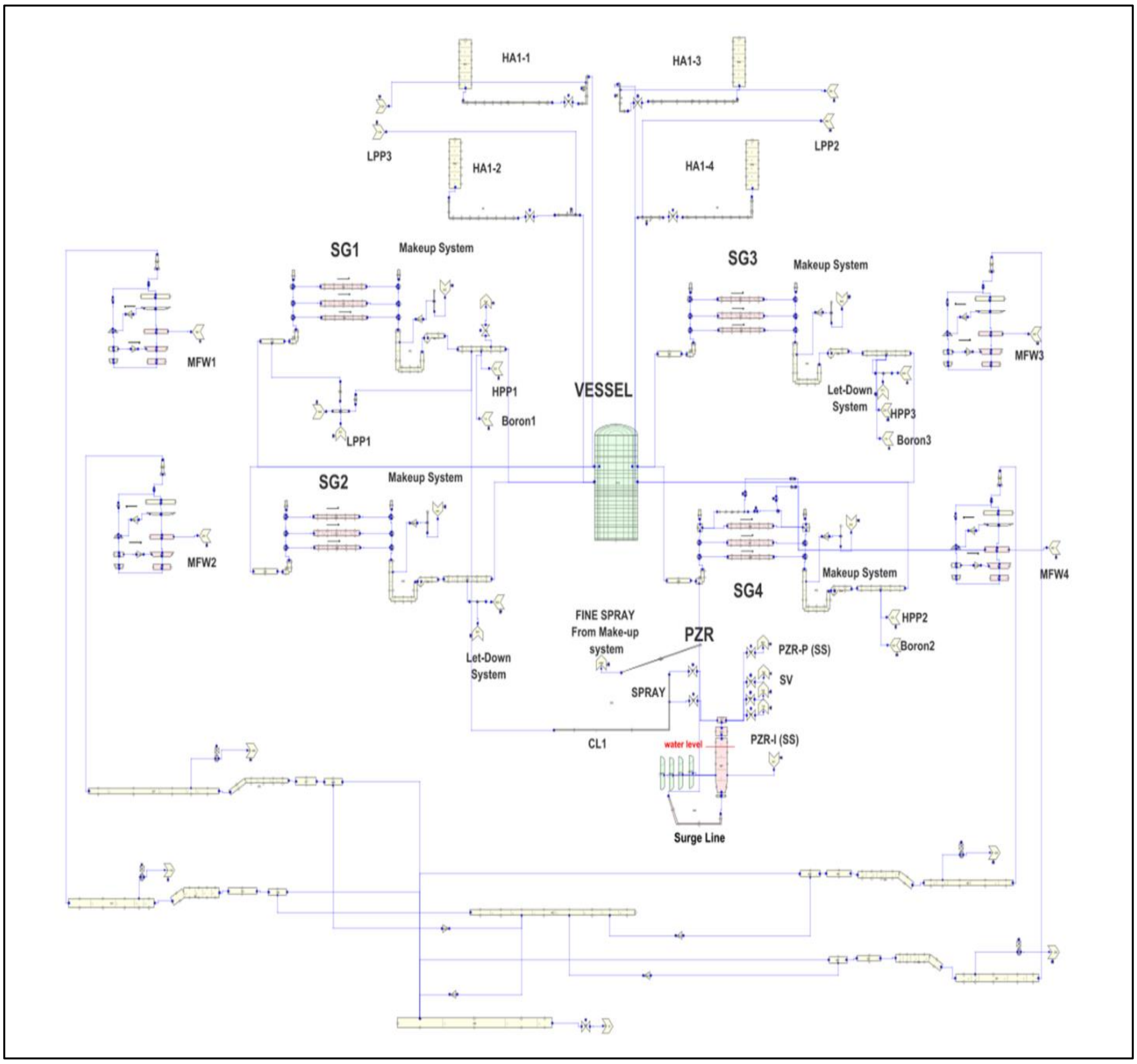


Figure 3.2.3.2 Integral model of the Kozloduy Nuclear Power Plant (KNPP) developed for TRACE (primary/secondary circuits, safety systems)

3.2.4 LLC ENERGORISK model description

The developed model of the reactor VVER-1000/320 is a 4-sector one with cross-links to simulate flows between sectors (Figure 3.2.4.1). This layout allows simulating the independent movement of the coolant within one loop. The area of the inlet and outlet pipes is divided into 8 equal parts, simulating annular gaps between the shaft and the reactor vessel. This allows you to properly separate flows during partial MCP operations [14].

The disturbance introduced by the ECCS branch pipes in the lowering section on the connections 67-1 and 69-1 has a turning effect on the flow of the coolant down the lowering section and causes the coolant to mix with the neighboring sector counterclockwise. The user can turn off the additional resistance of the ECCS pipes and get the so-called «symmetric» model, where the loop coolant enters the corresponding core sector almost unmixed. By default, the model is left «asymmetric», which is important for modelling asymmetric processes in the active zone.

The core is divided into 4 sectors, preserving the symmetry of the connected loops. No radial separation is provided. There are 3 channels allocated in each sector – medium fuel element, medium fuel element in hot fuel element and hot fuel element in hot fuel element. Bypasses are simply represented as common to all sectors.

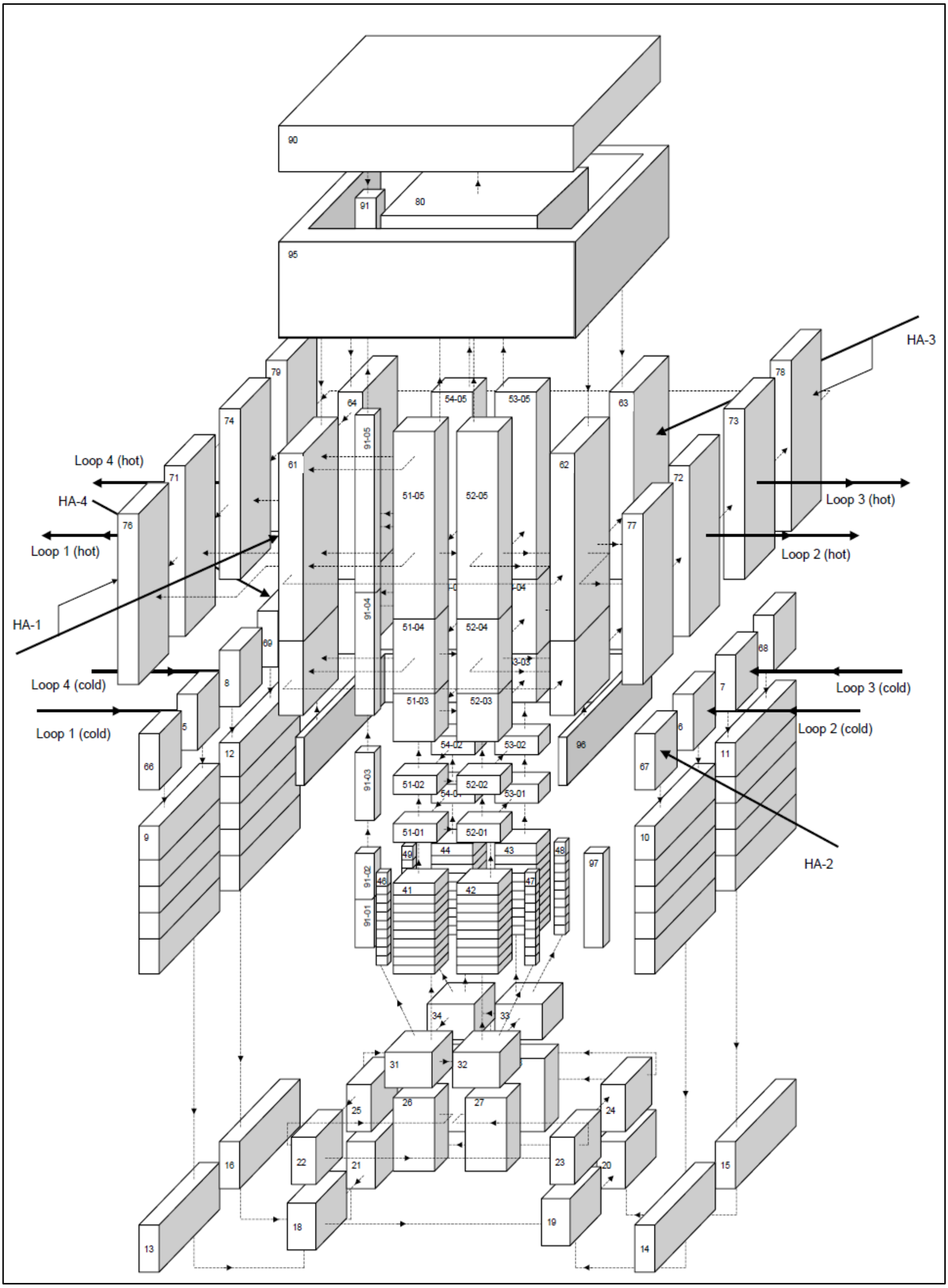


Figure 3.2.4.1 Nodalization diagram of the reactor

3.2.4.1 Primary circuit

The MCL model describes 4 loops. Each loop contains a «cold» and «hot» thread. In Figure 3.2.4.1.1 Nodalization scheme of the MCL loops is presented [14].

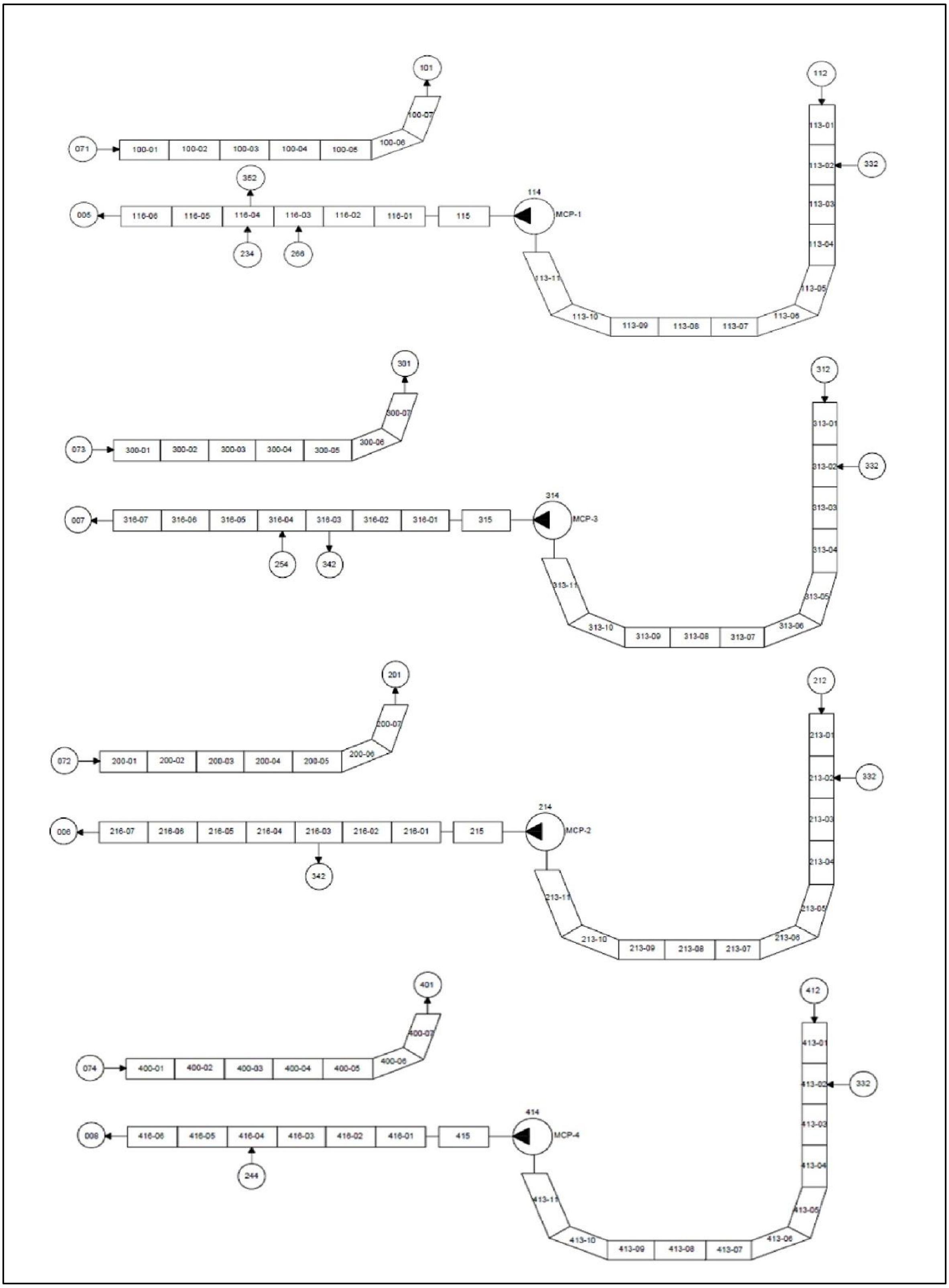


Figure 3.2.4.1.1 Nodalization scheme of the MCL loops

The nodalization scheme of the «hot» and «cold» SG reservoirs is shown in Figure 3.2.4.1.2. The first SG circuit consists of «hot» and «cold» collectors and pipes. HE 101 – «hot» SG collector, HE 112 – «cold» SG collector, HE 102,103,104,105,106,107,108,109,110,111-describe the SG tube.

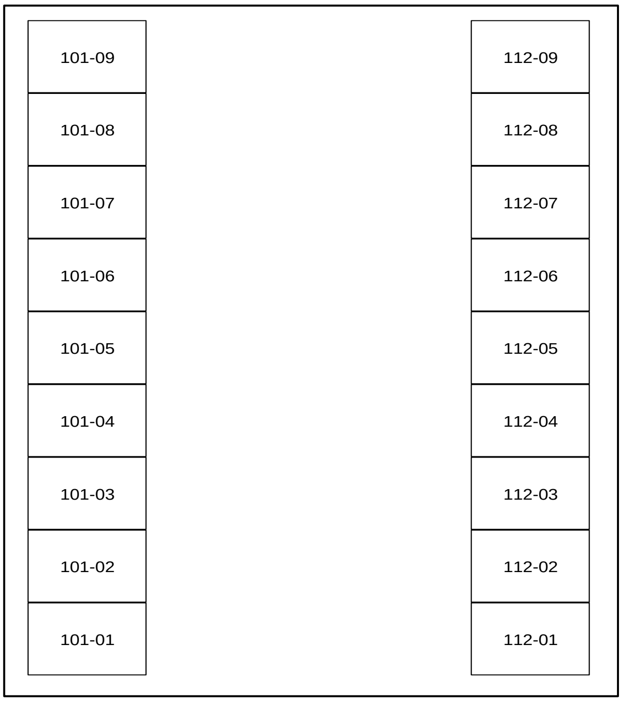
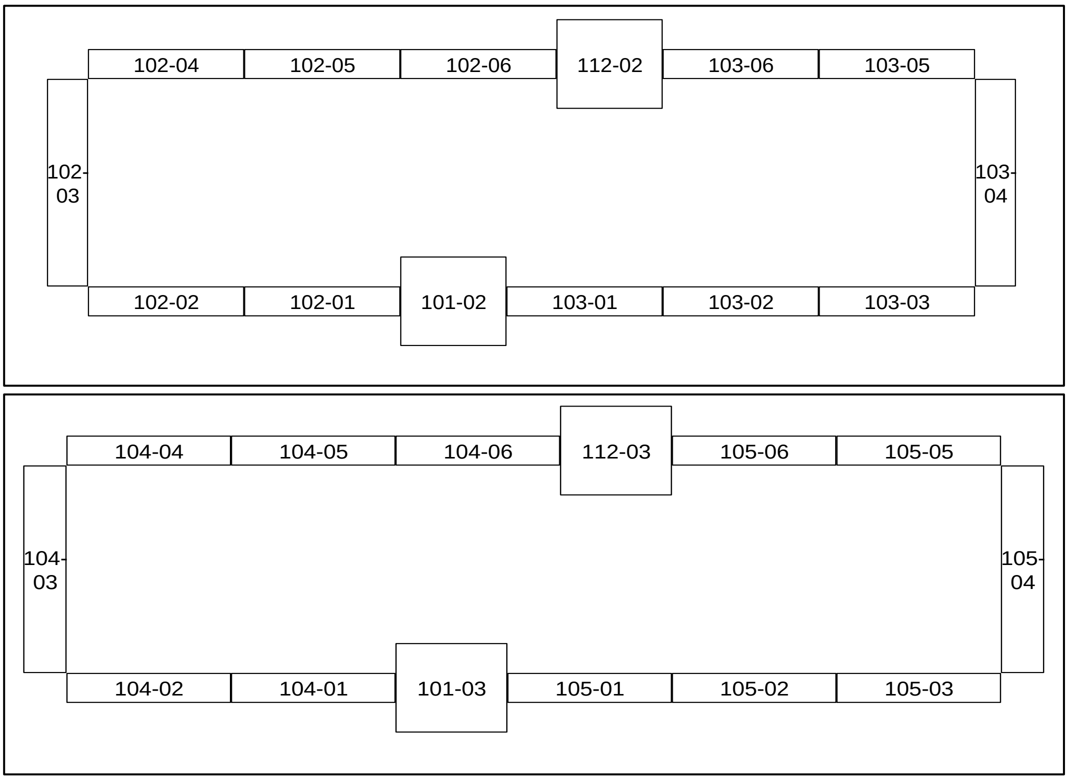


Figure 3.2.4.1.2 Nodalization scheme of «hot» and «cold» SG reservoirs

Hydrodynamic element of the tube bundle is located in the first five layers of the second circuit of the computational model. The nodalization diagram of the tube bundle for layers 1 - 5 is shown in Figure 3.2.4.1.3 Nodalization diagram of the tube bundle for layers 1 – 5.



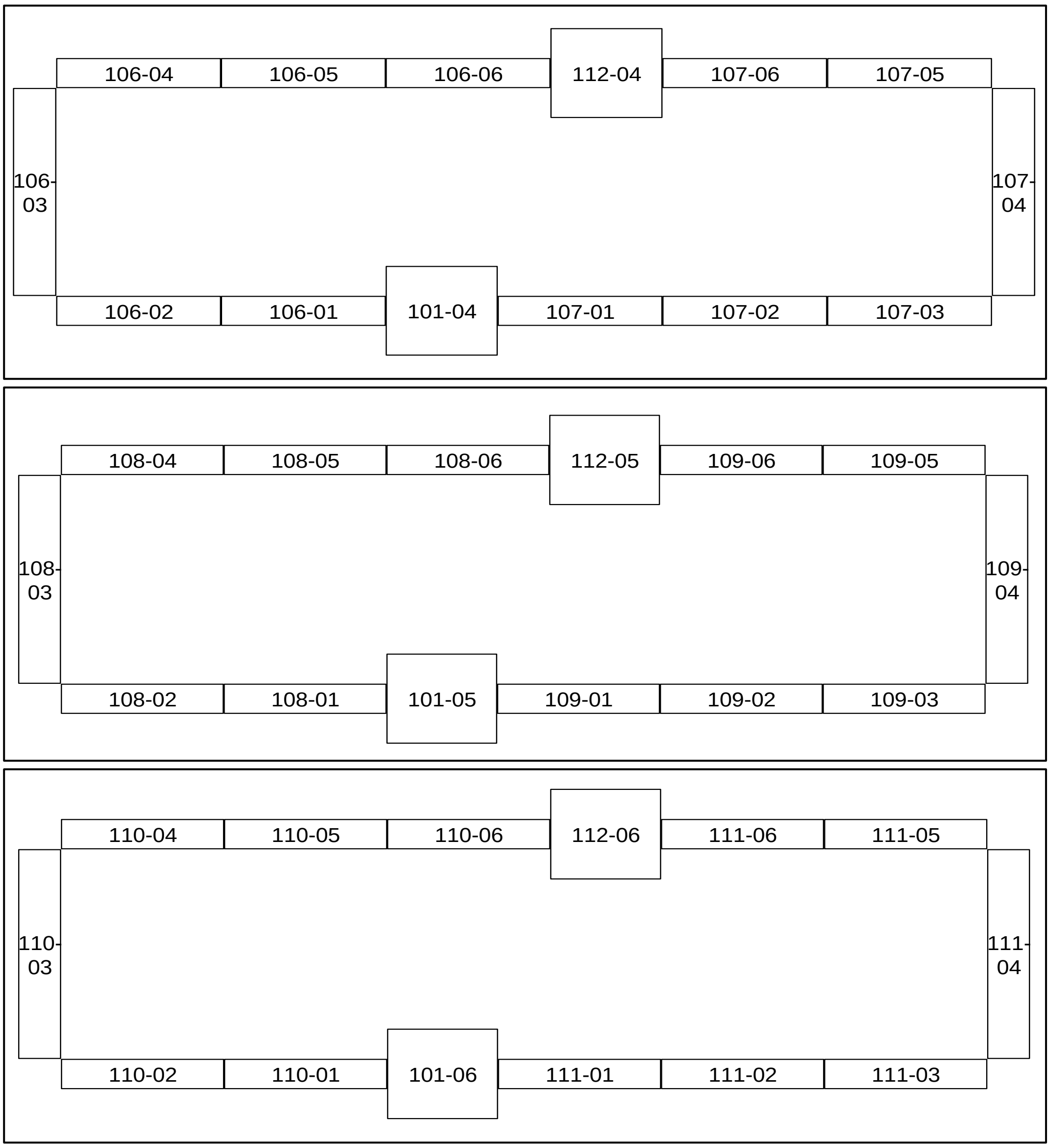


Figure 3.2.4.1.3 Nodalization diagram of the tube bundle for layers 1 – 5

3.2.4.2 Secondary circuit

Figure 3.2.4.2.1 shows the nodalization scheme of the second circuit of the PGV-1000M steam generator [14]. All 4 steam generators (SG) in the model are similar.

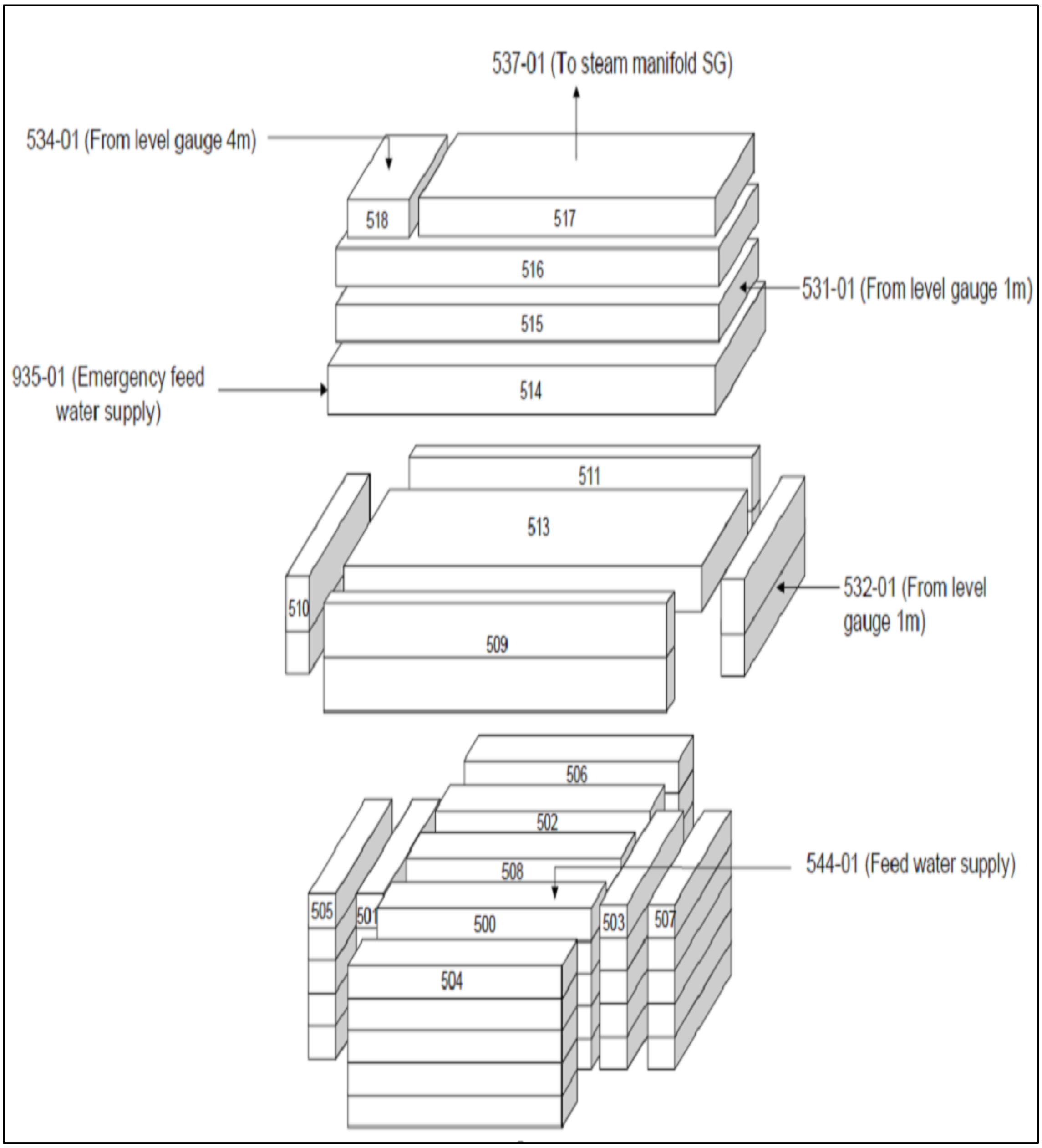


Figure 3.2.4.2.1 Nodalization diagram of the second circuit of the PGV – 1000M model

The calculated model of the PGV-1000M is made in the 3D approximation of the RELAP5/Mod3.2 code. The three-dimensional approximation was chosen to correctly distribute the heat load over the volumes of the second SG circuit.

HE 500,502 – side packages that describe the volume of the second circuit enclosed in the pipe bundle, HE 501,503-end packages that describe the volume of the second circuit enclosed in the pipe bundle. HE 504, 506-side bypasses, which up to the 4th element describe the volume of the second circuit between the external pipe bundle and the SG body and between the external and main pipe bundles. The 5th element of these HE includes the volume of the second SG circuit between the outer and main pipe bundles, as well as between the outer pipe bundle and the SHS SG rim. HE 505, 507-end bypasses. Distribution of the second circuit volume by The height distribution in HE 505, 507 is similar to that in HE 504, 506. HE 508 is the central bypass, which describes the volume in the center of the SG surrounded by a tube. HE 509,510,511,512 – volume of the second circuit of the SG between the edge of the SHS SG and the body of the SG. HE 513 – volume of the second circuit of the SG between the SHS SG and the upper row of the tube. HE 514,515,516,517,518- describe the vapor space of SG. The steam collector is represented by HE 537,538. HE of the first circuit are connected to HE of the second circuit by thermal structures.

Steam generated in the steam generators of the reactor plant is transported through steam lines to the high-pressure cylinder of the turbine with a total steam flow rate of 6154.2 t/h.

This section provides some basic assumptions and assumptions for modelling the entire steam pipeline system. When modelling, we will use some assumptions and simplifications.

The change in the volume of the steam line due to its finite diameter at the bends will be ignored. We assume that the volume of the element is its length along the centerline of the steam pipeline multiplied by the cross-sectional area (we consider this to be true for any steam pipelines, including those leading to steam-throwing valves). That is, the lengths of bends, if they are not given in the data, will be calculated along the centerline of the pipeline, taking into account its diameter and radius of the bend. MSV are not modelled separately. Their functions are performed by the turbine stop and control valves. The hydraulic resistance of the MSV is taken into account in the corresponding connection of the steam line. The roughness of steam pipes is assumed to be 10^{-4}m for seamless steel pipes.

Heat losses from the surface of steam pipes are not taken into account (thermal structures are not modelled). The turbine is modelled by the boundary condition as a relation to a constant pressure volume.

The nodalization diagram of the steam pipeline system is shown in Figure 3.2.4.2.2

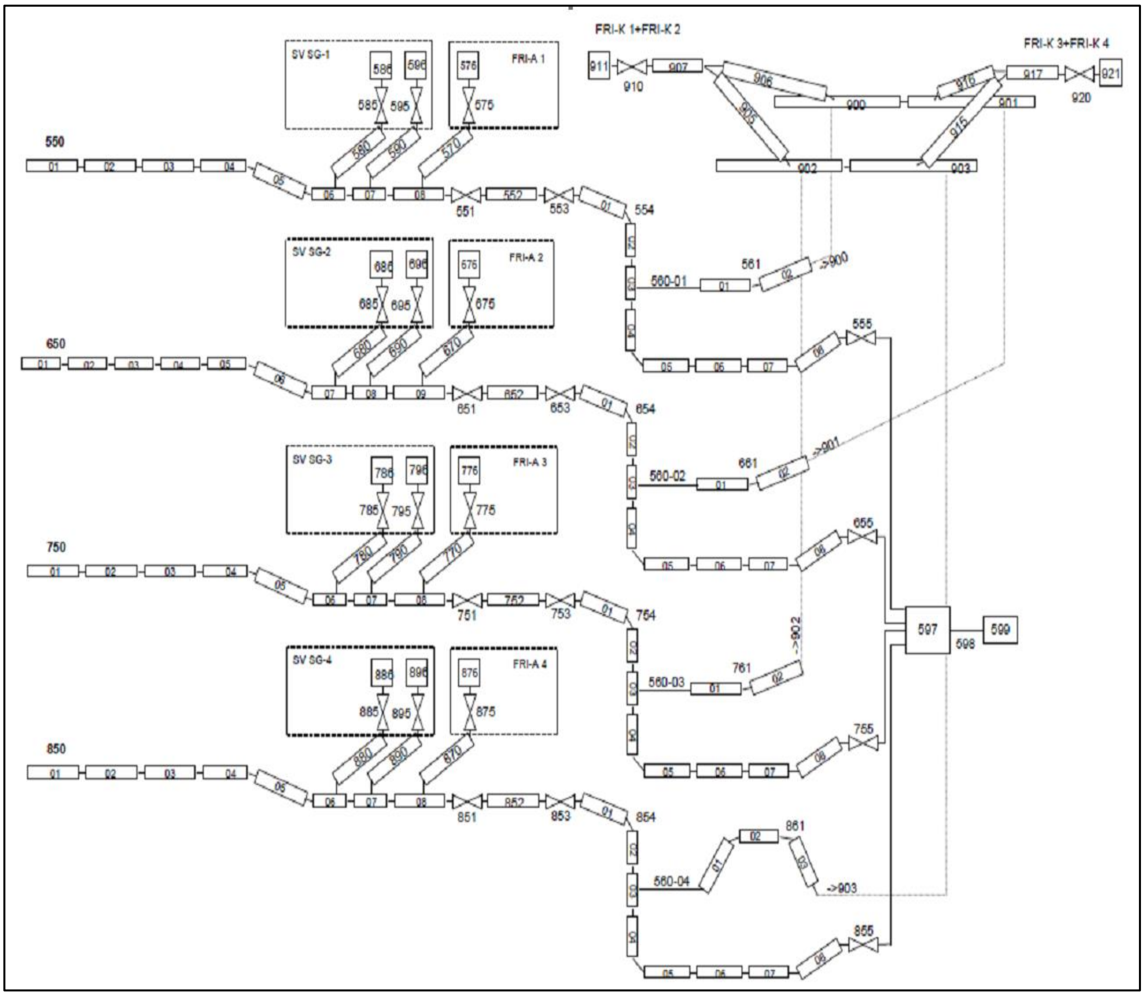


Figure 3.2.4.2.2 Nodalization scheme of the steam pipelines

3.2.4.3 SB LOCA model

- the leak is modelled on main cold coolant loop #1 between MCP and RV inlet (element 116-06) with internal diameter 30 mm using a valve (Figure 3.2.4.3.1).
- the homogeneous flow model is used;
- the formation of the leak occurs in one time step of the calculation (0.01 s);
- the pressure in the volume simulating the containment is equal to 1.013×10^5 Pa and conservatively remains constant throughout the transition process;
- the parameters of the environment in the containment based on a conservative approach are determined as superheated steam with a temperature of 150°C.

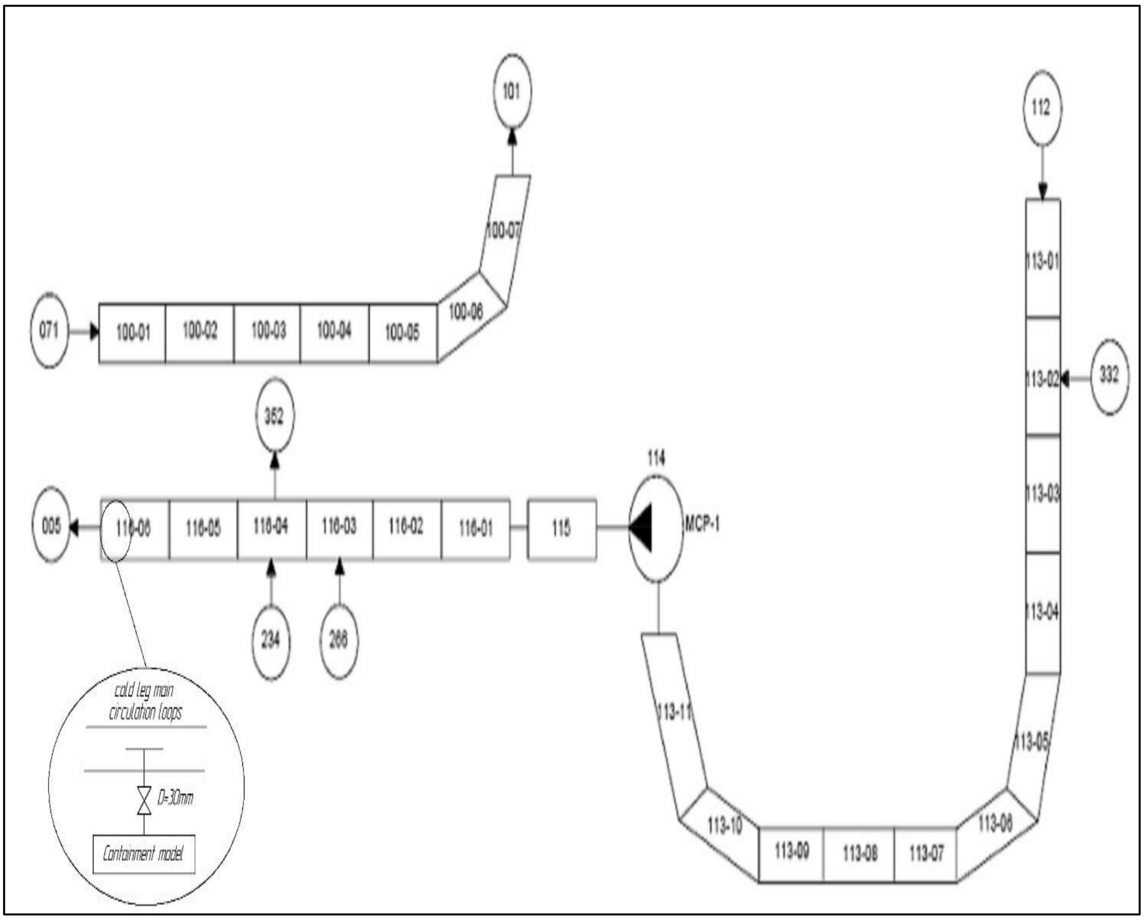


Figure 3.2.4.3.1 Nodalization scheme of the SB LOCA model

3.2.4.4 SG line break model

- the leak is modelled as double ended break of one pipeline with internal diameter 13 mm in SG#4 using three valves (Figure 3.2.4.4.1).
- the leak is modelled from element 410-06 to 802-05 (from side of pipe) and 412-06 to 802-05 (from side of collector);
- the homogeneous flow model is used;
- the formation of the leak occurs in one time step of the calculation (0.01 s);

- the standard procedure for renodalization of the tube layer of the emergency SG was applied for such an accident: element 410 represents all tubes of this layer, with the exception of the broken one, and new element corresponding to one "emergency" tube was added. Thus, in this problem, the original element 410 is presented as the sum of parallel element, with preservation of the total cross-section and volume.
- thermal structures have also changed accordingly. Thus, the specified changes do not affect the thermal-hydraulic parameters of the model in the absence of leakage, but allow to correctly model the leakage itself.

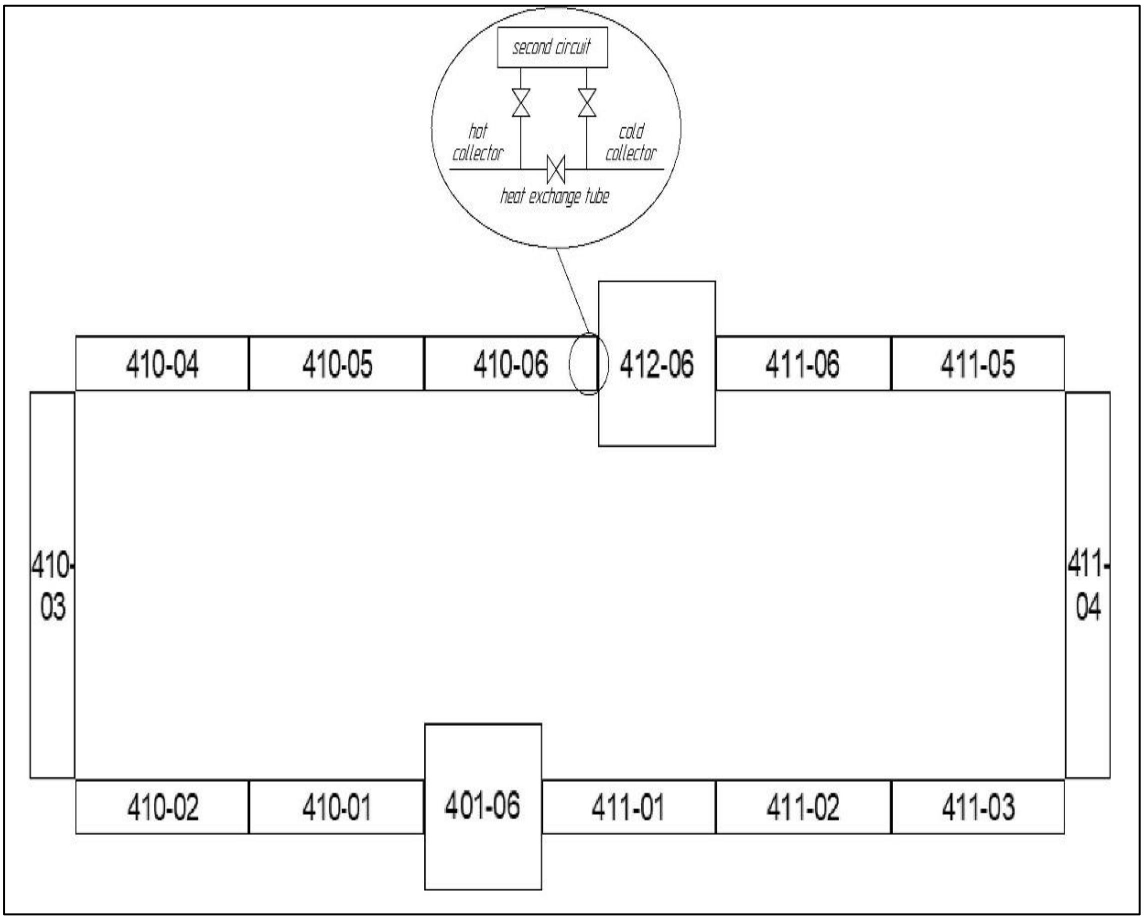


Figure 3.2.4.4.1 Nodalization scheme of the SG line break model

4 RESULTS AND DISCUSSION

4.1 Comparisons of a steady state results between the codes

The steady state plant simulation was performed by all participants 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 4.1.1.

Table 4.1.1Comparison of references plant parameters and the steady state calculation results during the “SB LOCA + SG line break” transient for KNPP unit 6

Parameter	Plant Design	INRNE RELAP5/ mod3.3	FRA CATHARE3	KIT TRACE-V5P5	ER RELAP5/ mod 3.2
Reactor thermal power, MW	3000.0	3000.0	3000.0	3000.0	3002.3
Primary pressure, MPa	15.7	15.7	15.8	15.55	15.7
Pressurizer Level, m	8.77	8.76	8.77	8.71	8.77
Average coolant temperature at reactor inlet, K	560.15	560.2	562.7	560.8	563.9
Average coolant temperature at reactor outlet, K	592.05	591.0	592.9	591.38	593.9
Mass flow rate through one loop, kg/s	4400.0	4395.9	4370.0	4403.16	4397.8
Pressure in SG, MPa	6.27	6.17	6.26	6.08	6.27
Pressure in the main steam header (MSH), MPa	6.08	6.03	6.02	5.62	6.08
Steam mass flow rate through SG steam line, kg/s	408.0	408.03	394.8	408.23	409.3
SG Water Levels, m	2.40	2.38	2.34	2.51	2.40
Liquid mass in the SG secondary side, t	48.0	48.0	48.2	48.2	47.9

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

This good agreement between references plant data and predictions 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 comparisons

This section presents the results of the analysis of the “SB LOCA + SG line break” under the initial and boundary conditions described in subsection 2.3. In the frame of the ‘’SB LOCA+SG line break’’ transient benchmark, the results of the participants were collected and compared.

In Figure 4.2.1 Figure 4.2.40 the code-to-code comparison of all parameters have been compared including the sequence of the main events. The calculated chronological sequence of events is presented in the Table 4.2.1.

As the initial and boundary conditions used by all teams are very close or almost identical where it is possible, the important parameters that could contribute to the deviations during the accident progression are break flow rates (due to usage of different assumptions and models), calculations of residual powers (as all teams have selected their own models for calculation of residual powers), heat transfer from primary to secondary and work of BRU-As to support secondary pressure, as well as flow rates of main coolant loops of primary circuit during the natural circulations. The deviations can also be based on the use of different codes and developed models with different nodalization (some models have been more detailed or have some simplifications and etc.).

Table 4.2.1 Chronological sequence of events calculated for the “SB LOCA + SG line break” transient

	INRNE	**Framatome**	**KIT**	**LLC Energorisk**
Event description	Time, s	Time, s	Time, s	Time, s
Time for total dryout of Pressurizer	430	380	393	440
Time for first opening of BRU-A in loop#4	8	2	175	8
Time for first opening of BRU-A in loop#1, #2, #3	8	2	175	8
Time for loss of natural circulation	2850	2580	3540	2980
Beginning of boiling in hot legs	630	590	500	420
Time for hot leg dryout	3598	3165	3540	3510
Time for beginning of core uncovery	4220	3640	-	5160
Time for reaching 650 °C (923 K) – time for leaving EOPs and entrance in SAMG	7980	7455	7779	6200
Total dryout of reactor core	8200	7325	-	-
End of calculation at 1200 °C (1470 K)	9400	7660	9390	6510

In Figure 4.2.1 is shown the total reactor power. At the beginning of the transient, the MCP are stopped, and the core power drops quickly due to the SCRAM. The main part of the residual heat is removed from the reactor core and primary side through the secondary side by a natural circulation until all SGs are effective and there is enough coolant in a primary circuit. Additionally, part of the residual heat is removed through the leakage of coolant mass to the containment due to the SB LOCA.

It can be seen from Figure 4.2.1 that the trends of the calculated total reactor powers for all calculations are very close. The total reactor core decreases for the first 30 seconds from 3000 MW as follows for the individual teams: for INRNE to 160 MW, FRA to 160 MW, KIT to 154 MW and for ER to 174 MW. It can be observed that in the first 100 s, the residual heat for three teams: INRNE, FRA and KIT have the similar values (99 MW, 98 MW and 96 MW), while for ER team is slightly higher value of 118 MW.

After the first 1000 s, the residual heat calculated by the three teams INRNE, FRA, and KIT has almost identical values (55 MW, 56 MW, and 54 MW), while the ER team calculated higher value for residual heat, which is 70 MW. The bigger calculated residual power contributes in faster reducing of water level in steam generators (SGs), it is presented in Figure 4.2.21 to Figure 4.2.24.

The very similar behaviour for residual heat can be seen at around 5000 s for the same three teams, INRNE, FRA, and KIT (33 MW, 34 MW, 34 MW), while for the ER team it continues to have higher values for the residual heat and at 5000 s the value is 43 MW. This comparison explains earlier dryout of SGs in ER team calculation as it is shown in Figure 4.2.25 to Figure 4.2.28.

The higher residual heat could lead to faster dry out not only of the SGs, but also to faster loss of natural circulation and causing this way earlier heat up of reactor core as it could be seen in the comparison of corresponding parameters presented below.

At the end of the calculated transients, it is observed the following values for the residual heats: for INRNE team is 27 MW at 9400 s; for FRA team is 29 MW at 7660 s; KIT team is 28 MW at 9390 s and ER team is 40 MW at 6510 s. It should be kept in mind that the calculated values are for different times corresponding to the end of calculations done by each team.

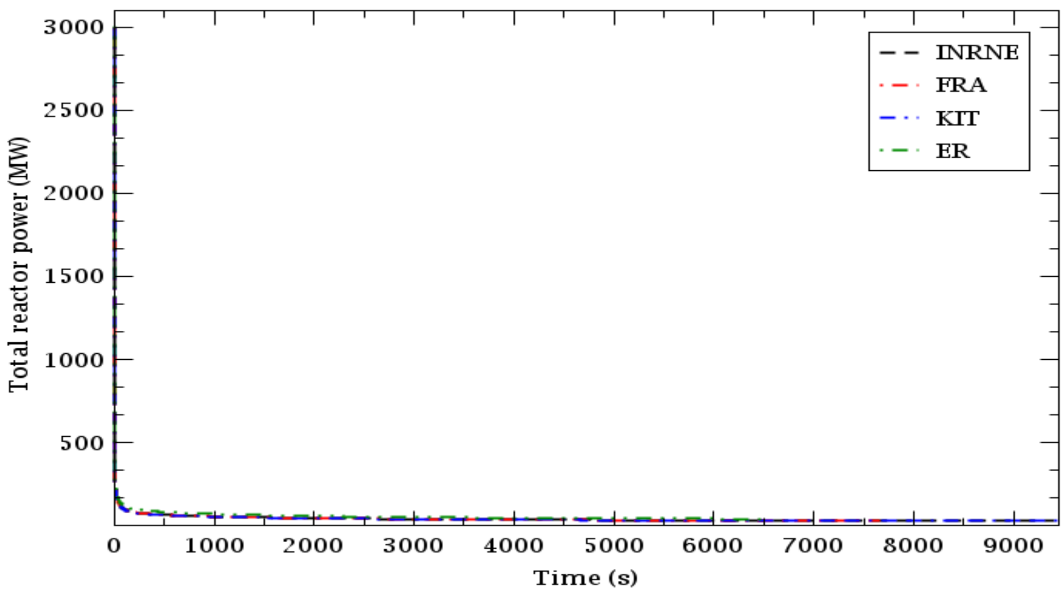


Figure 4.2.1 Total reactor power

The Figure 4.2.2 presents the primary pressure calculated by the participants. The primary pressure drops due to the loss of the coolant through the all breaks (from primary circuit to the containment and from primary to secondary side) in the beginning of transient.

The primary pressure for all teams in the first 600 s in the beginning of the transient follows a very similar trend. The minor fluctuations of primary pressure have been seen between 600 s and 2000 s and have similar behaviour for all teams.

The fluctuations in KIT model are in 1000 s and 1450 s, it could be explained with the numerical instability.

The increasing of pressure at different time is observed in all calculations. The increased pressure is caused by loosing of capabilities of SGs to remove the heat from primary circuits, as well, as the break flow rates. At this time, they are not sufficient to reduce the pressure since breaks are voided at corresponding models and break flow rates are reduced significantly.

It can be observed the increase of the primary pressure after 2500 s for both teams (FRA and ER), while for KIT team is around 4000 s and for INRNE team after 7000 s.

The pressure rises in the primary circuit because of the hot legs completely dry up. It causes loss of natural circulation, with further loss of heat removing by steam generators (SGs).

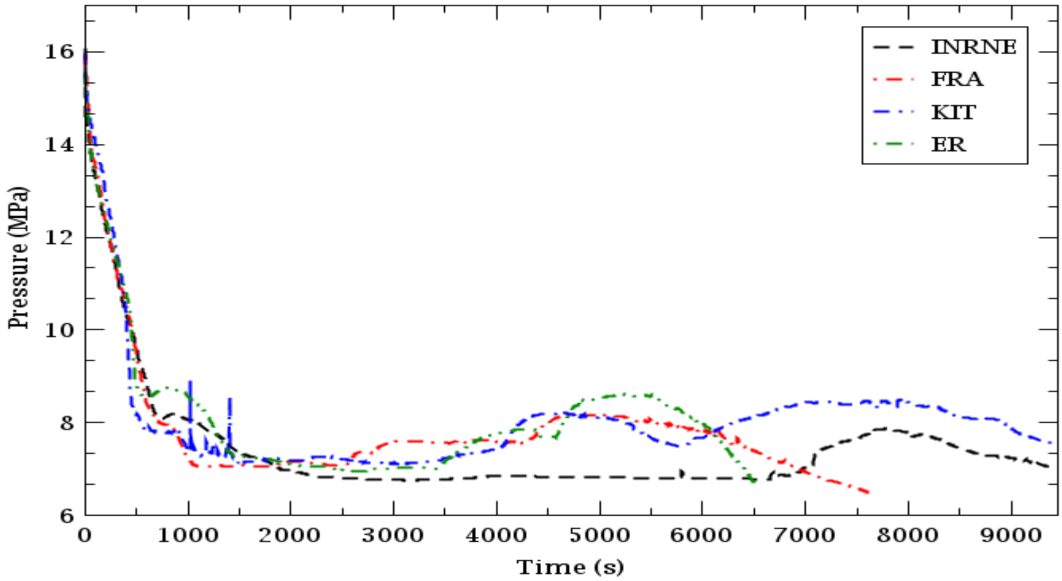


Figure 4.2.2 Primary pressure (at core exit)

The secondary pressure at MSH is shown in Figure 4.2.3. In the first seconds, the primary temperature and pressure drop, while the secondary side pressure shows a sharp increase.

After closing of MSIV in front of the turbine, the secondary pressure starts to increase which leads to the set point of BRU-As opening. BRU-As open and start to reduce secondary pressure to the controlled level of pressure at 6.67 MPa until to the end of transient.

The calculated secondary pressure shows very similar trends for INRNE, KIT and ER teams, but calculated by the FRA team secondary pressure is higher.

After opening the BRU-As in FRA team calculation, the secondary pressure starts to reduce significantly slow to the controlled level and has overestimated value at the end of the transient compared to the other three teams. The different behaviour of secondary pressure for FRA team could be explained with the characteristics of their BRU-As model. In the model of FRA team, the pressure is correctly regulated at 68 bar by the BRU-As as I expected in the SGs. However, after the check valve in the steam line (after the BRU-As going to the steam header) the pressure is much higher due to is assumed that a check valves isolated fully MSH, while in the other models is assumed minor leakages.

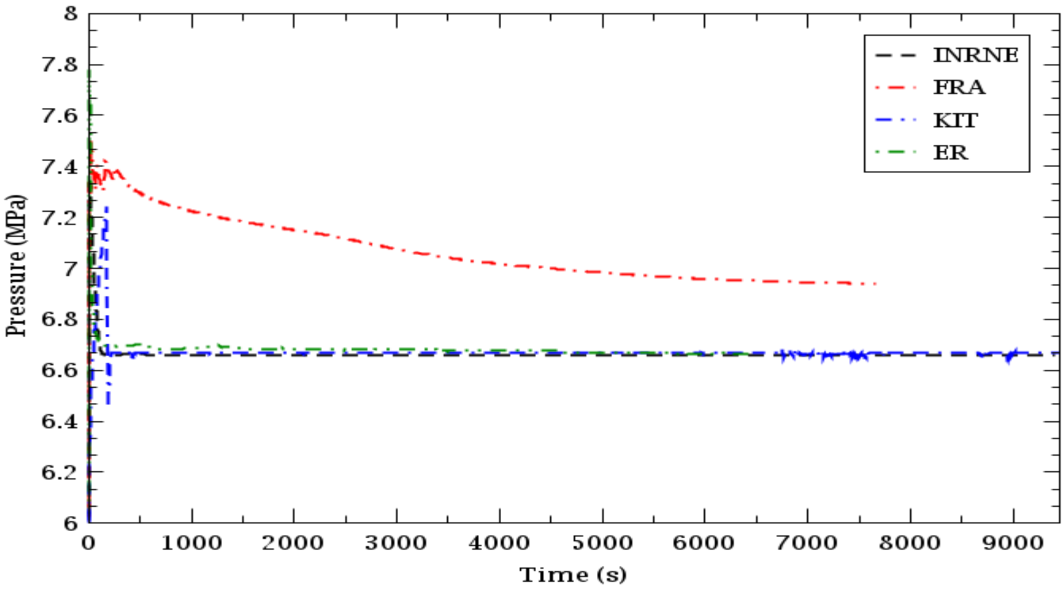


Figure 4.2.3 Secondary pressure at MSH

In the Figure 4.2.4 is shown the core exit cladding temperature. The core exit cladding temperature decreases initially due to the reactor scram.

The core exit cladding temperature starts to increase after 5000 s for both teams (FRA and ER), while for INRNE and KIT teams around 7000 s due to loss of cooling and beginning of core uncovery. As it is seen, the temperature increases from 600 K to more than 1400 K for all calculations at different times.

The core exit cladding temperature has the similar behaviour for FRA and ER teams. For both teams (INRNE and KIT) the core exit cladding temperature follows a similar trend with delay of 2000 sec. Such a delay could be explained with differences in calculated total reactor residual powers, as well as with differences in the integral break flow rates.

The earlier increasing of core exit cladding temperature for ER team corresponds to earlier losing of steam generators (SGs) due to higher residual power (see Figure 4.2.1 and Figure 4.2.21 to Figure 4.2.24).

The behaviour of core exit cladding temperature of FRA team could be explained with the earlier losing the effectiveness of SGs and higher integral break flow rate compared to the others which can be seen on corresponding Figure 4.2.7 and Figure 4.2.21 to Figure 4.2.24. It could also be explained by the level of water decreasing in the upper plenum. Until ~4600 s there is still a little bit of water coming down from the upper plenum to the core (see the void fraction in the upper core between ~3500 s and ~4600 s) but after this time, this flow stops and since the level in the core has already significantly decreased the cladding heats very quickly.

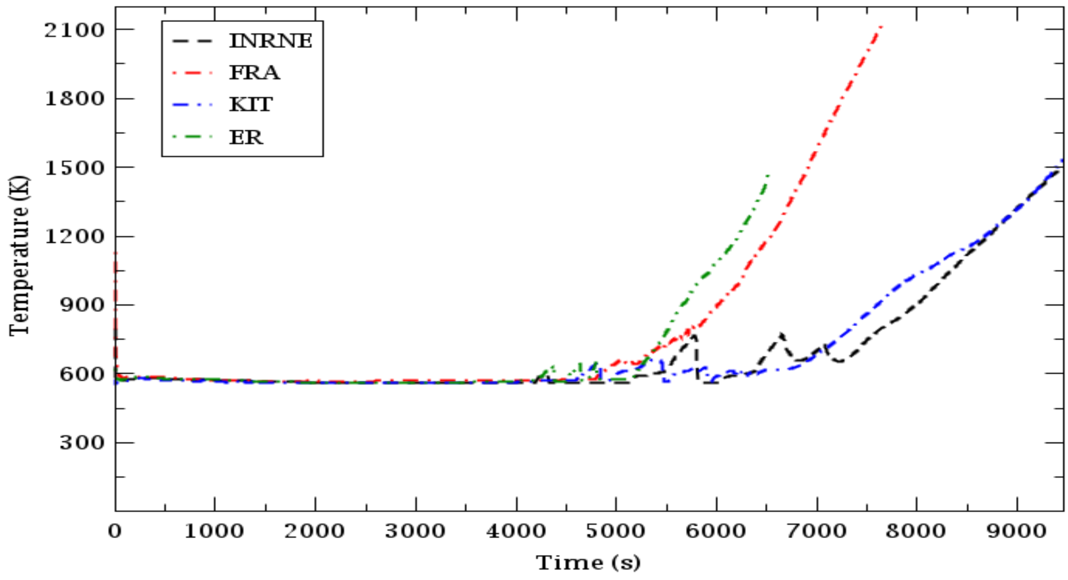


Figure 4.2.4 Core exit cladding temperature

The Figure 4.2.5 shows the core exit coolant (gas) temperature. For all teams core exit temperature sharply increases in different time and reaches around 1500 K.

The core exit coolant (gas) temperature starts to increase for three teams (INRNE, FRA and KIT) after 7000 s, while for ER team is at around 5100 s. The behaviour of ER team core exit coolant (gas) temperature seems reasonable corresponding to the behaviour of fuel cladding temperature and has the same explanation.

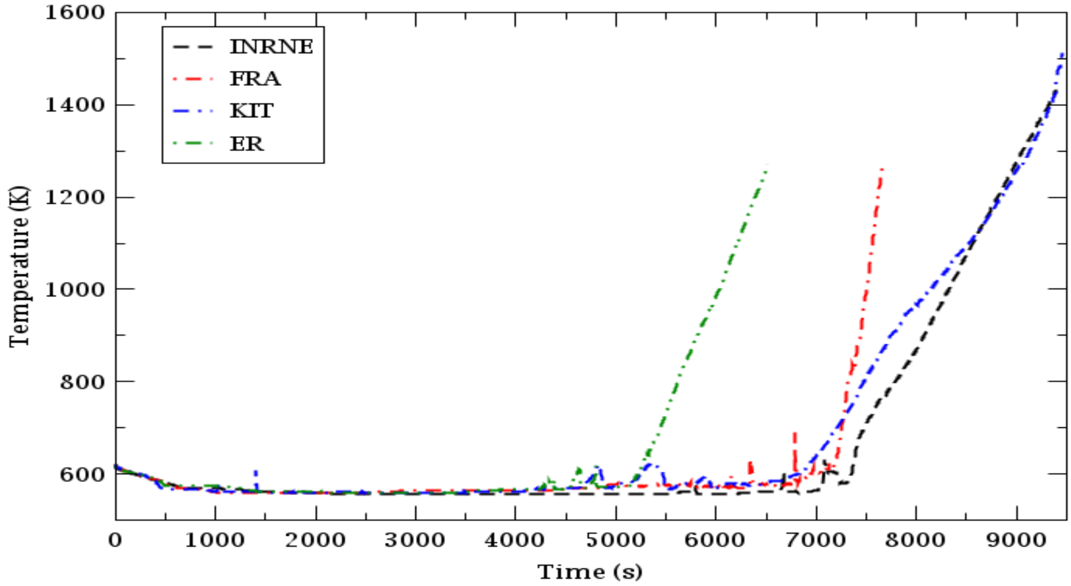


Figure 4.2.5 Core exit coolant (gas) temperature

The SB LOCA flow rates as predicted by the codes are shown in Figure 4.2.6. The loss of coolant from primary side reduces capability for removing of residual heat from the reactor core by steam generators (SGs) from one side. Also, the residual heat is removed by losing the primary coolant.

In the beginning of transient, the behaviour of flow rates is very similar during the first 1000 s for all calculations. In the beginning of the transient the maximum break flow rates varied with +-10 kg/s.

The SB LOCA flow rates show similar trends for both teams (INRNE and FRA) and a rapid decrease of flow rate is observed due to voiding of the break at around 4500 s. For the KIT team there is a rapid decrease in the flow rate due to voiding of break in the primary circuit at around 5500 s.

For the ER team, the flow rate between 1500 s and 3000 s shows fluctuations due to an increase in primary pressure, after that a rapid decrease in flow rate is observed until around 4500 s. It is seen on the graph that the flow rate is increasing again, and the voiding of the break is about 4800 s.

The differences in behaviour of break flow rates could be explained by using the different models for break simulations, as well as using of homogenous model by ER team, compared to the nonhomogeneous models used by others participants.

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 the obtained deviations in prediction of deferential break flow rates. From the other side, the trends of integral break flow rate have the same shape and very similar behaviour by some teams.

Because 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 for all calculations.

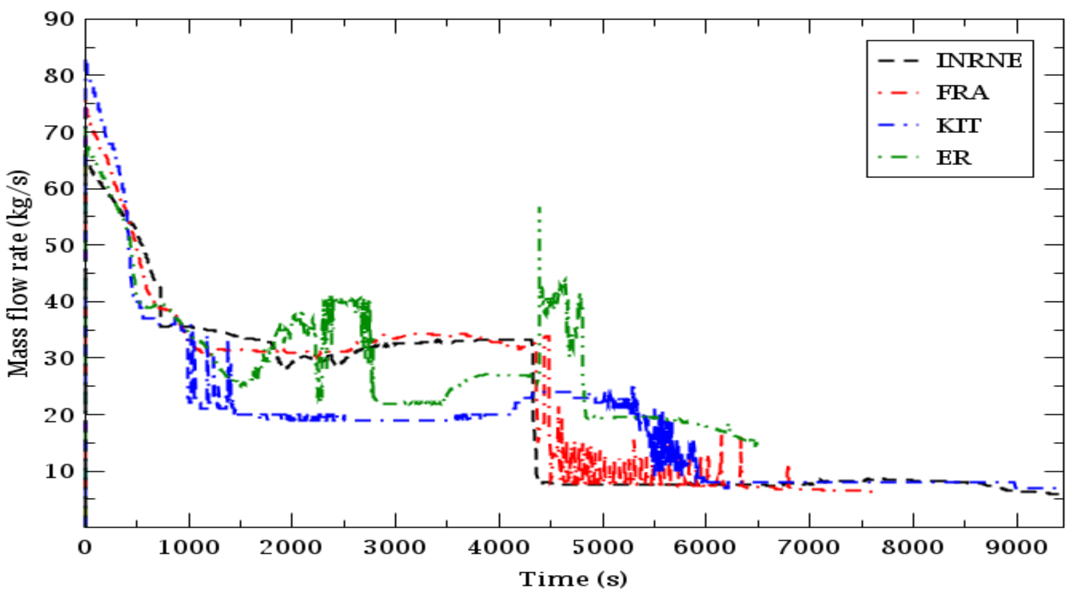


Figure 4.2.6 SB LOCA flow rate

In the Figure 4.2.7 is shown the integral SB flow rate from primary circuit. The total mass of the coolant through the primary break has a similar shape for all the calculations.

The total loss of coolant through the primary break for all the teams (INRNE, FRA, KIT and ER) is very close for first 1500 s and it is around 67 t.

The all calculations for the mass flow of small break opening are between 65 kg/s and 80 kg/s, after that decrease to about 30 kg/s approximately at 1500 s.

However, the total loss of coolant through the primary break varied by +-15 t at the end of calculations. For example, the INRNE team lost 196 t of coolant through the primary break at 9400 s, the FRA team lost 187 t at 7660 s, the KIT team lost 181 t at 9390 s, and the ER team lost 193 t at 6510 s.

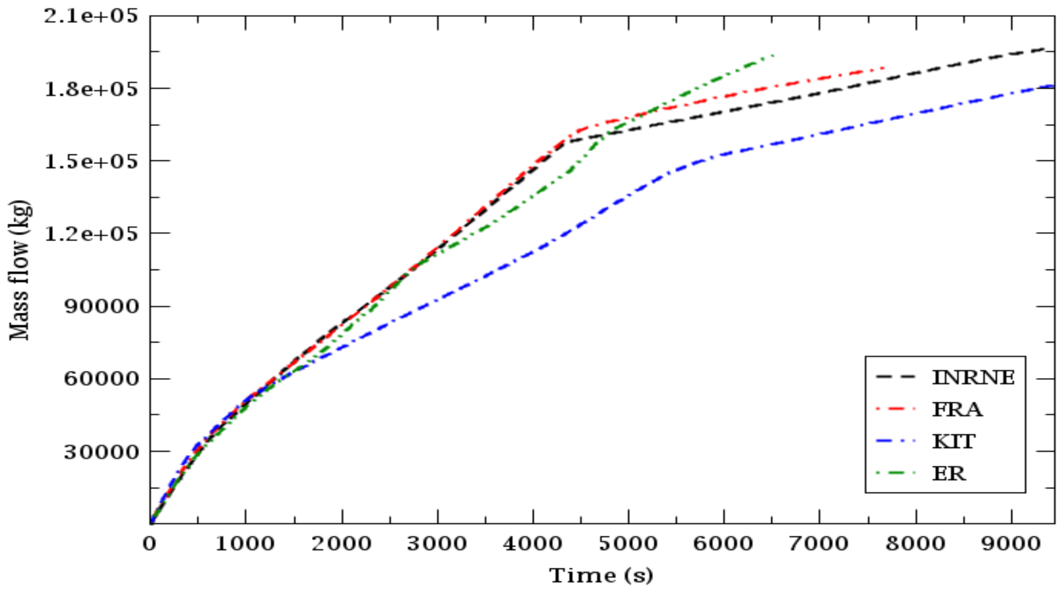


Figure 4.2.7 Integral SB LOCA flow rate from primary circuit

The Figure 4.2.8 shows the primary to secondary side flow rates. In the steam generator #4, the broken tube causes the primary circuit to flow in the secondary with a flow rate of approximately 12.5 kg/s in total when the tube breaks for all teams.

The primary to secondary side flow rates have similar trends for three teams (INRNE, KIT and ER). It can be observed that in the first 500 s, the primary to secondary side flow rates for all teams have a similar behaviour.

The FRA team mass flow rate drops quickly after the first 1000 s of the transient because the pressure in the primary circuit reaches that of the secondary circuit and the tubes at the level of the break start to dry up in the primary side of the SG#4.

Further, the primary to secondary side flow rates at the end of the transient are seen for the INRNE team to be 0.57 kg/s at 9400 s, for the FRA team the value is -0.025 kg/s at 7200 s due to higher secondary pressure, for the KIT team the value is 0.98 kg/s at 9455 s, and for the ER team the value is 0.19 kg/s at 6510 s.

Because of the primary side behaviour, the variances for all the calculations have very different values. The increasing of break flow rate values closely follows the increasing of pressure in primary sides for different teams.

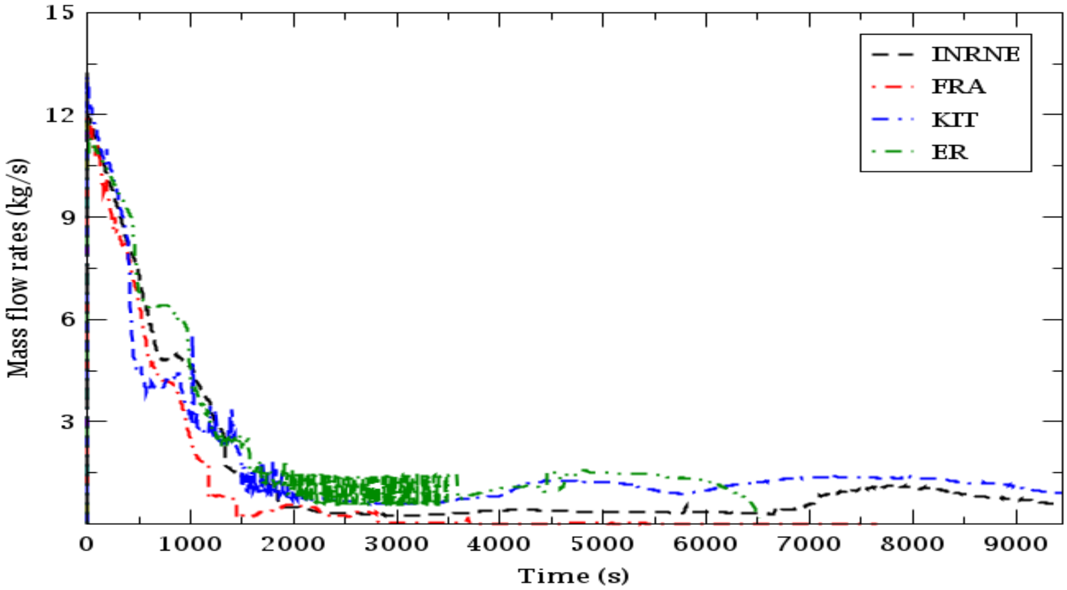


Figure 4.2.8 Primary to secondary side flow rates (in total)

The integral (cumulative) break flow rates from primary to secondary sides are shown in Figure 4.2.9. A similar trend can be seen in the integral break flow rates from primary to secondary side for first 1000 s for all teams. The both teams INRNE and KIT have a similar shape during the whole transient.

It is the same for both teams FRA and ER for first 4000 s, but after the 4000 s, for the ER team starts to increase, while for the FRA team remains stable until the end of the transient. The explanation is that in the FRA team, calculation of the secondary side pressure is close to primary and the tubes at the level of break start to dry up in the primary side of the SG#4. All this leads to reducing the break from primary to secondary and even reversing the flow.

The integral break flow rate from primary to secondary side reaches the total mass of coolant for individual teams as follow: for the INRNE team is 13.3 t, for the FRA team is 7.6 t, for the KIT team is 16.4 and for the ER team is 10.4 t at the end of transient.

For the total mass of coolant from primary to secondary side the deviation is + - 8 t.

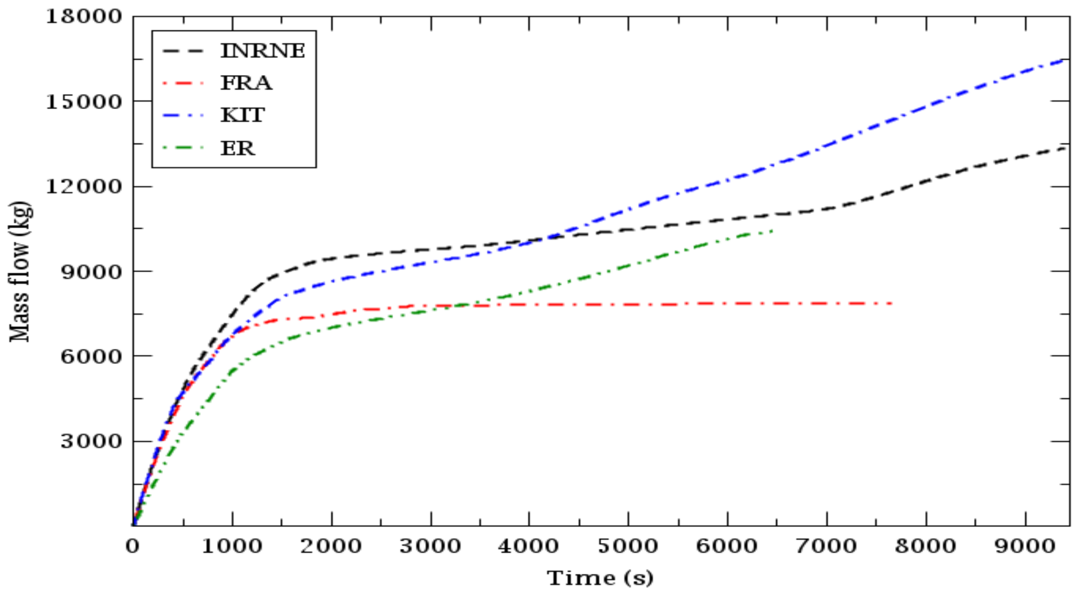


Figure 4.2.9 Integral break flow rate from primary to secondary side

The loss of coolant inventory and the primary pressure reduce lead to the reduction of the water level in the Pressurizer, which is shown in Figure 4.2.10.

It is observed a significant decrease in the water level in the pressurizer almost to the bottom because of the loss of coolant for all teams within the initial 400 s. The calculations of all teams demonstrate similar behaviour for the first 400 s.

The Pressurizer is totally dry out for INRNE team in 430 s, for FRA team in 380 s, for KIT team in 393 s and for ER team in 440 s (see the Table 4.2.1).

The water level in the pressurizer temporarily rises a little because of the primary pressure increasing which causes the condensation.

The increasing of water level in pressurizers corresponds to increasing of primary pressure for each of the teams. The biggest increasing of water level is observed in KIT team calculation, while the smallest return of water level is predicted by INRNE team.

Additionally, the back of water level in the pressurizer could be explained by liquid water stuck between the hot leg (in the surge line) and the bottom of the pressurizer.

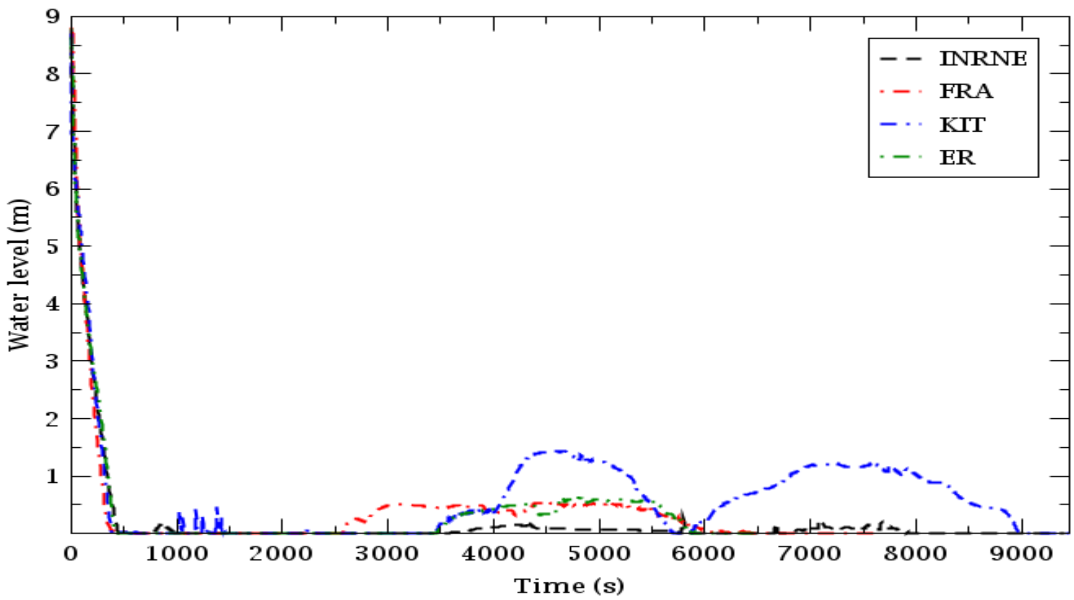


Figure 4.2.10 Pressurizer water level

In the Figure 4.2.11 is shown the comparison of differential BRU-As flow rate. The pressure in the secondary circuit increases rapidly at the very start of the transient after closing the MSIV and leads to open the BRU-As after reaching their set point. BRU-As open and start to reduce secondary pressure to the controlled level of pressure at 6.67 MPa until the end of transient.

The time to reach the setpoint to open the BRU-As is as follows: INRNE team at 8 s, FRA team at 2 s, KIT team at 175 s and ER team at 8 s. It is observed that both teams INRNE and ER have the same value, while FRA team open very fast BRU-As and KIT team have a significant delay for BRU-As opening.

The delay of BRU-As opening in KIT team could be explained with an observed condensation of water in SGs after reactor SCRAM, which causes from one side increasing of water level, and from the other side delay in secondary side pressure.

The initial flow rate of FRA team is close to the expected in the beginning at the opening pressure of BRU-As.

The observed differences could be explained with using of different codes and models by the participants.

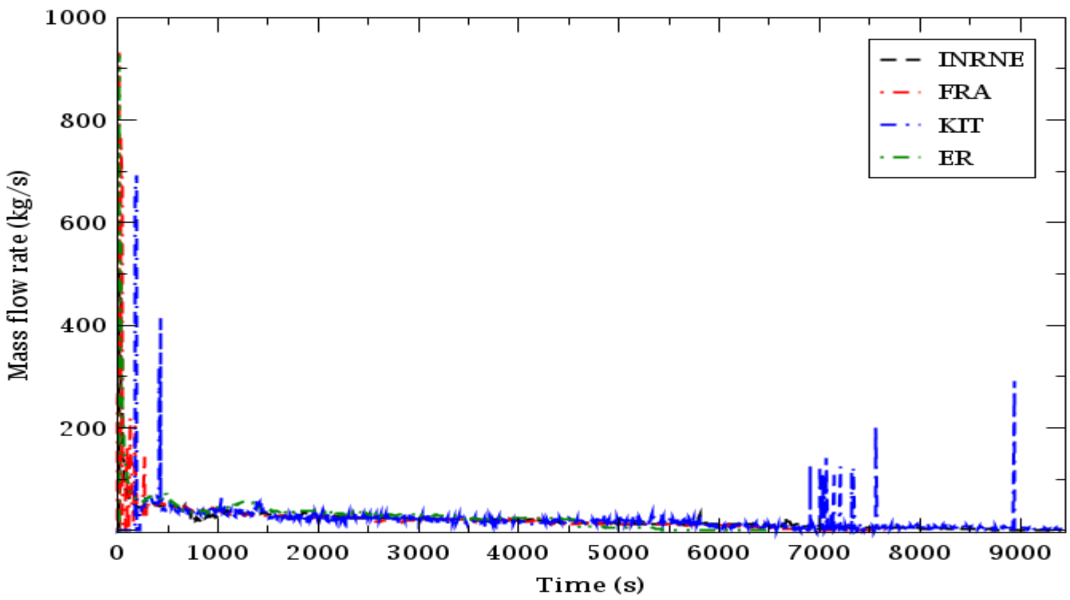


Figure 4.2.11 BRU-As flow rate

The comparison of the Integral BRU-As flow rate is presented in Figure 4.2.12.

Overall, the comparison of calculated by the participants integral flow rate of all 4 BRU-As shows similar trends.

Both teams INRNE and FRA have almost identical behaviour until 6200 s, after which slightly increasing of integral flow rate of BRU-As is observed for INRNE team, while for FRA team it remains stable to end of the transient.

The integral flow rate of BRU-As for ER team is overestimated compare to other teams.

For KIT team the integral flow rate of BRU-As seems to be slightly underestimated compared to both teams INRNE and FRA.

The integral flow rate of all 4 BRU-As reaches for individual teams as follow: INRNE team is 188 t, FRA team is 170 t, KIT team is 190 and ER team is 199 t at the end of calculations. For the integral flow rate of all 4 BRU-As the deviation is + - 29 t.

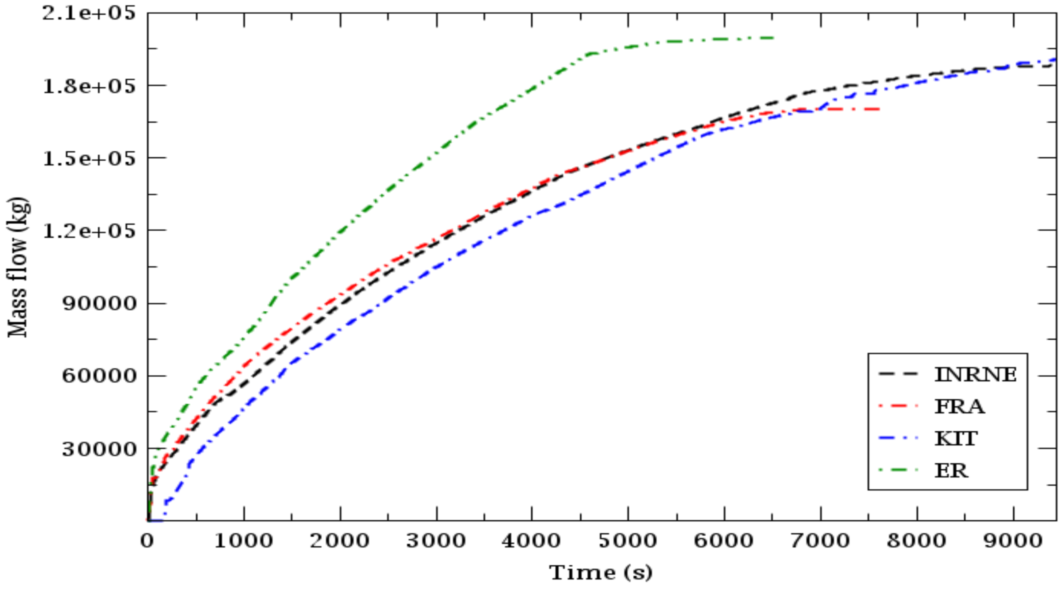


Figure 4.2.12 Integral BRU-As flow rate

Comparisons of cold leg coolant temperatures are presented from Figure 4.2.13 to Figure 4.2.16.

Coolant temperature trends in the cold legs for all computer codes are almost similar for the first 1000 s.

The cold leg coolant temperature starts to increase and reaches different values for all participants. The comparison of all calculations shows that for all teams it varied with +/- 15 K.

The rise in the temperature depends on heat transfer from primary to secondary sides and corresponds to increasing of the primary pressure. As it is explained above, the increasing of primary pressure comes when heat transfer to the secondary side is disturbed for all teams.

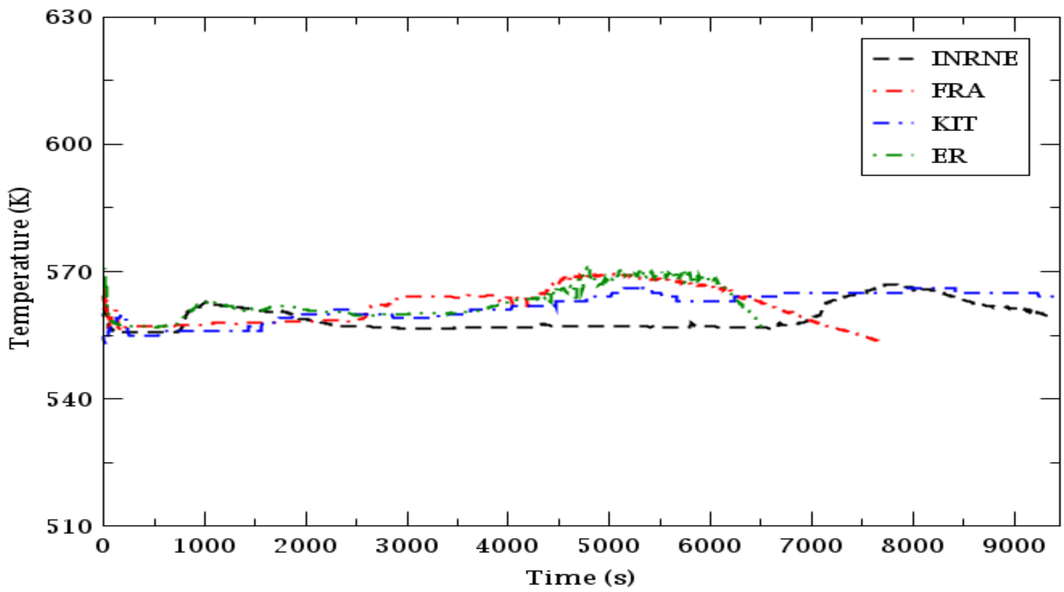


Figure 4.2.13 Temperature in cold leg#1

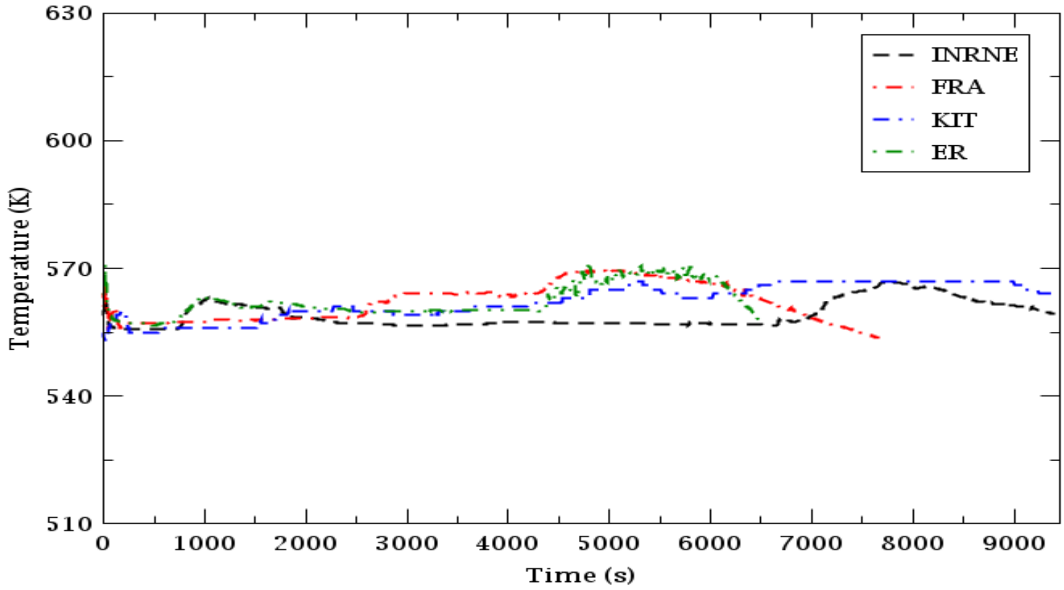


Figure 4.2.14 Temperature in cold leg#2

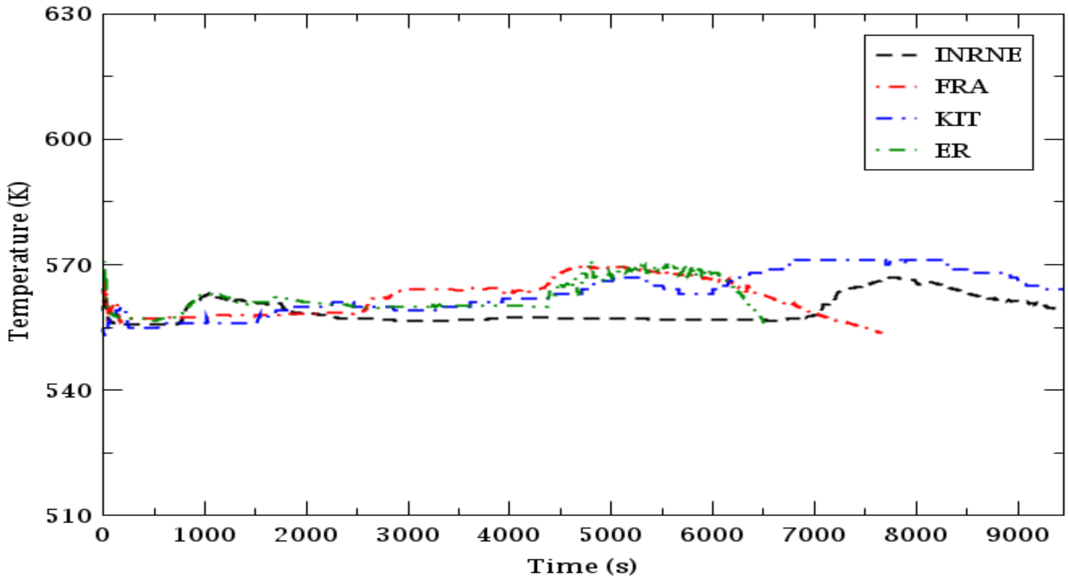


Figure 4.2.15 Temperature in cold leg#3

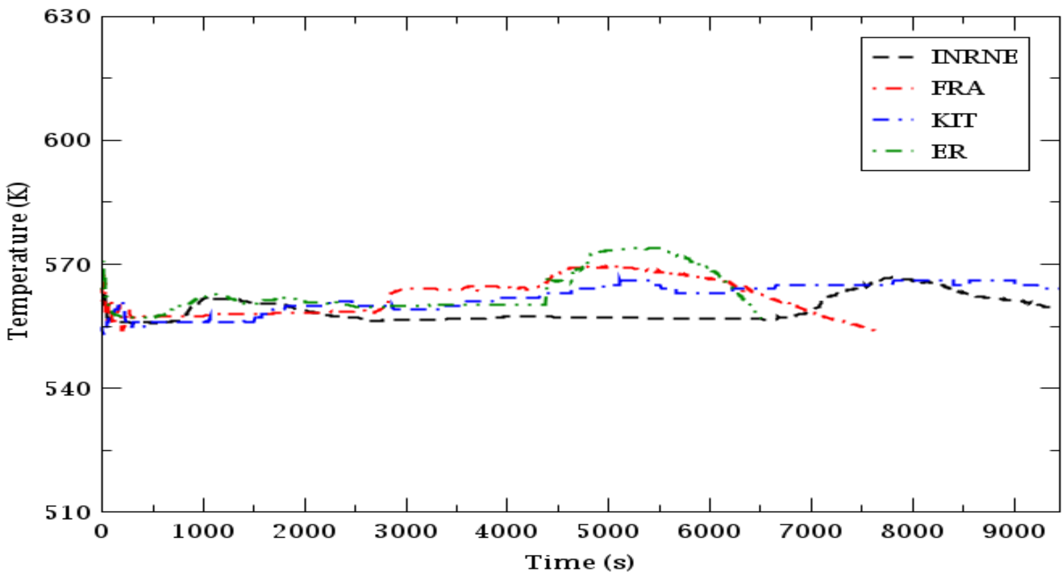


Figure 4.2.16 Temperature in cold leg#4

The results of hot leg coolant temperatures are presented from Figure 4.2.17 to Figure 4.2.20. Comparison of the temperatures in the hot legs for all computer codes are almost similar for the first 2500 s.

After that an increase of the temperature is observed, which follows the behaviour of the primary side pressure due to inefficient heat removing by steam generators (SGs).

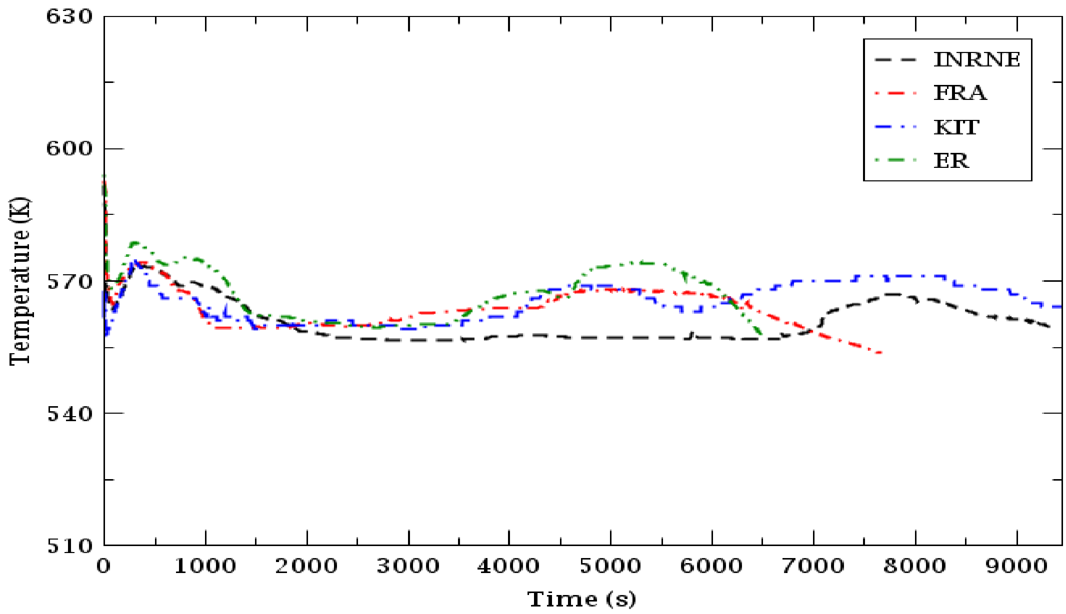


Figure 4.2.17 Temperature in hot leg#1

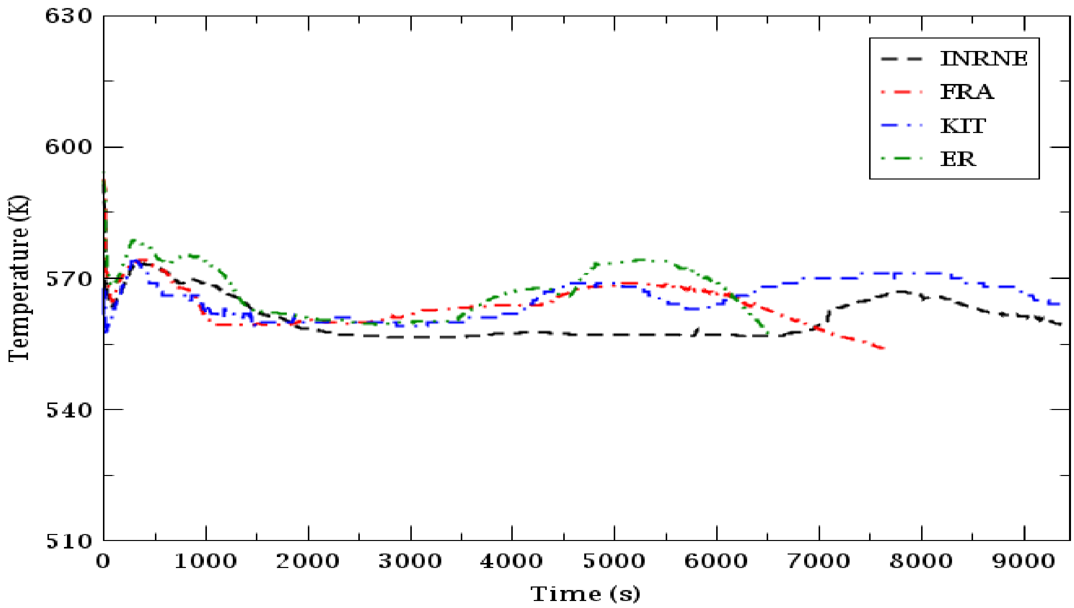


Figure 4.2.18 Temperature in hot leg#2

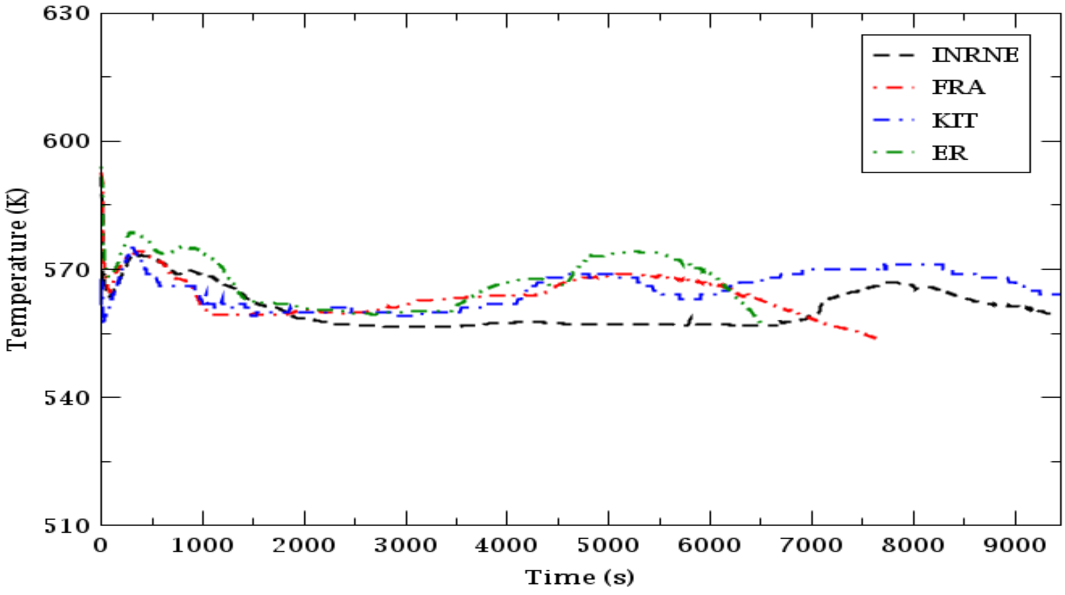


Figure 4.2.19 Temperature in hot leg#3

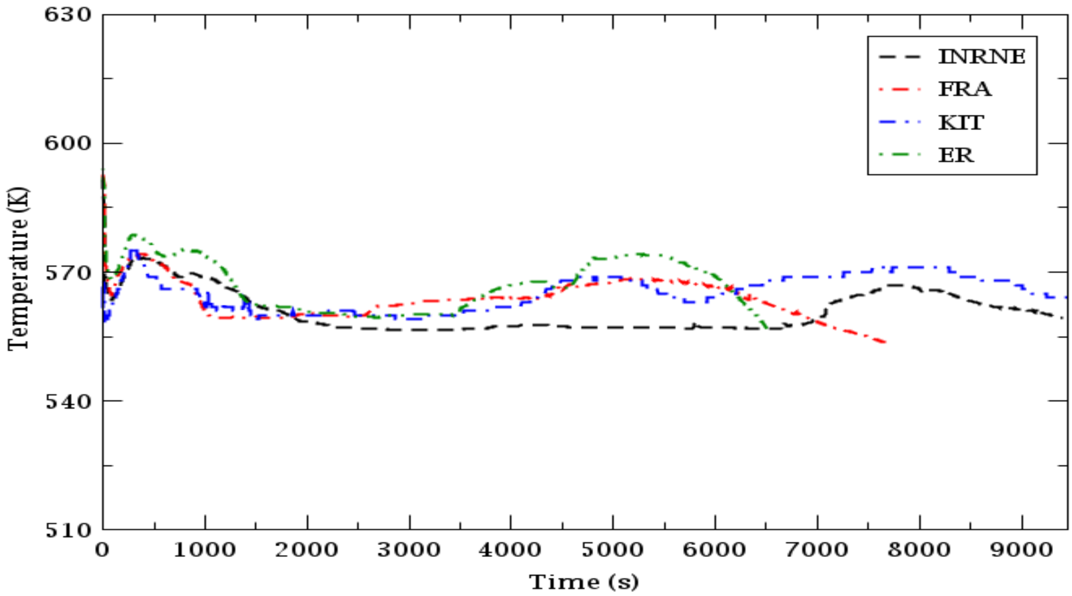


Figure 4.2.20 Temperature in hot leg#4

In the Figure 4.2.21 to Figure 4.2.24 Steam generators water levels for all teams are presented. Due to the loss of feed waters (as SBO), the steam generator water levels are reduced significantly. Steam generators (SGs) did not dry out completely for three of the teams (FRA, INRNE and KIT) because of the loos of primary coolant.

As the ER team is using higher residual heat power, the steam generators (SGs) are dried out before losing significant amounts of water in the primary circuit and their earlier dry out is due to the need of removing significantly more power from the primary circuit.

The observed increase of water level in KIT model in the beginning of the transient could be explained with the observed condensation in the secondary side.

In 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 for all teams follow similar trends until 2500 s. Because 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 as follows: INRNE team at 2850 s, FRA team at 2580 s, KIT team at 3540 s and ER team at 2980 s (see Table 4.2.1).

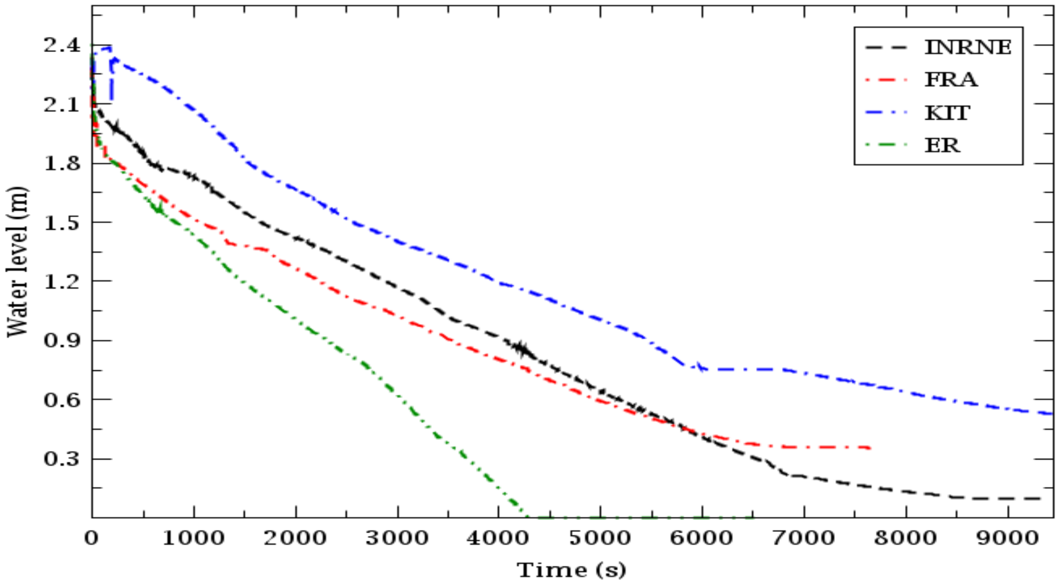


Figure 4.2.21 SG#1 water level

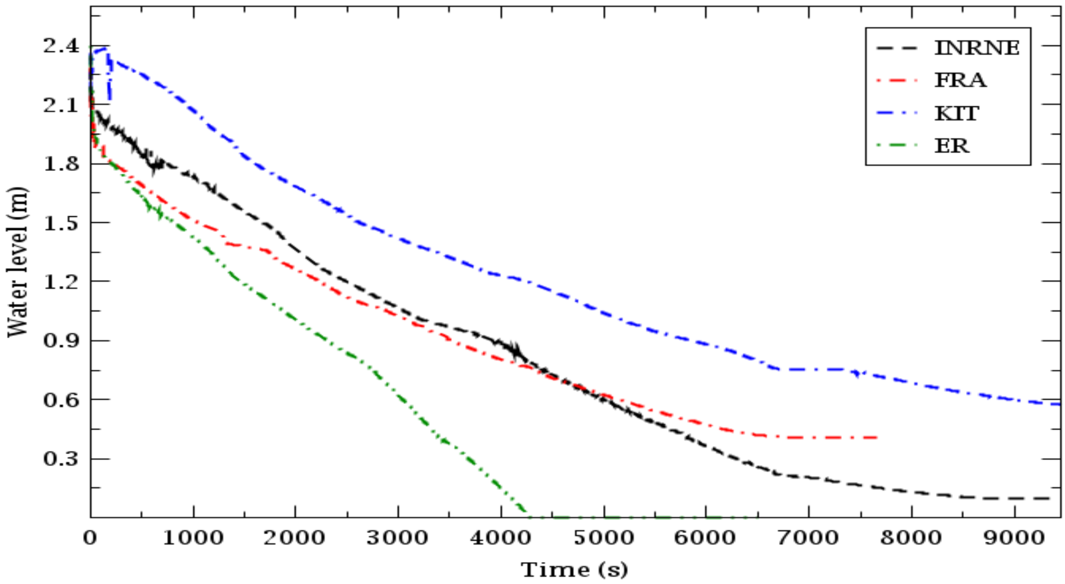


Figure 4.2.22 SG#2 water level

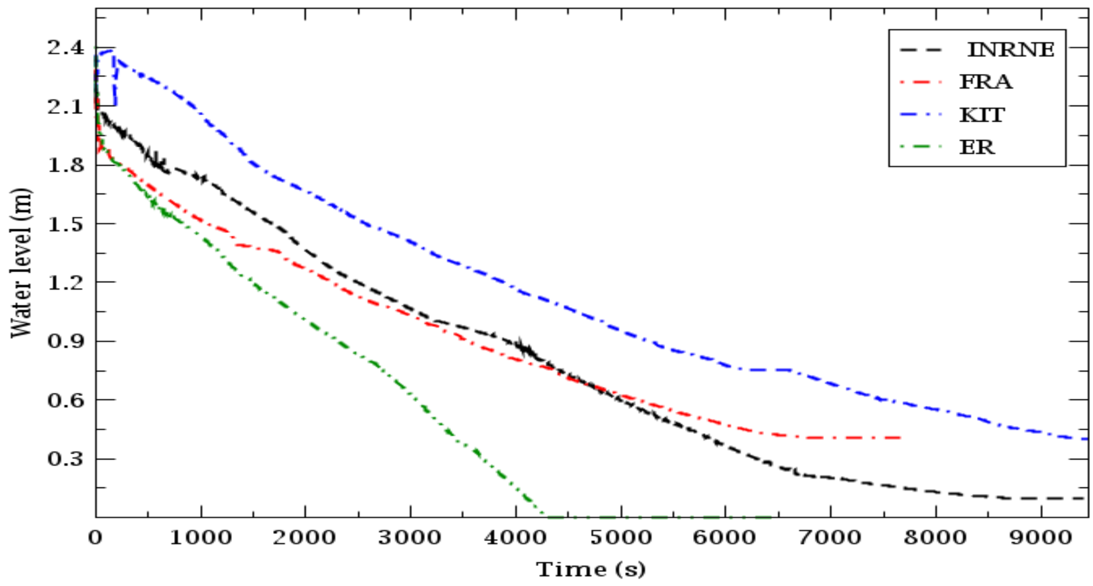


Figure 4.2.23 SG#3 water level

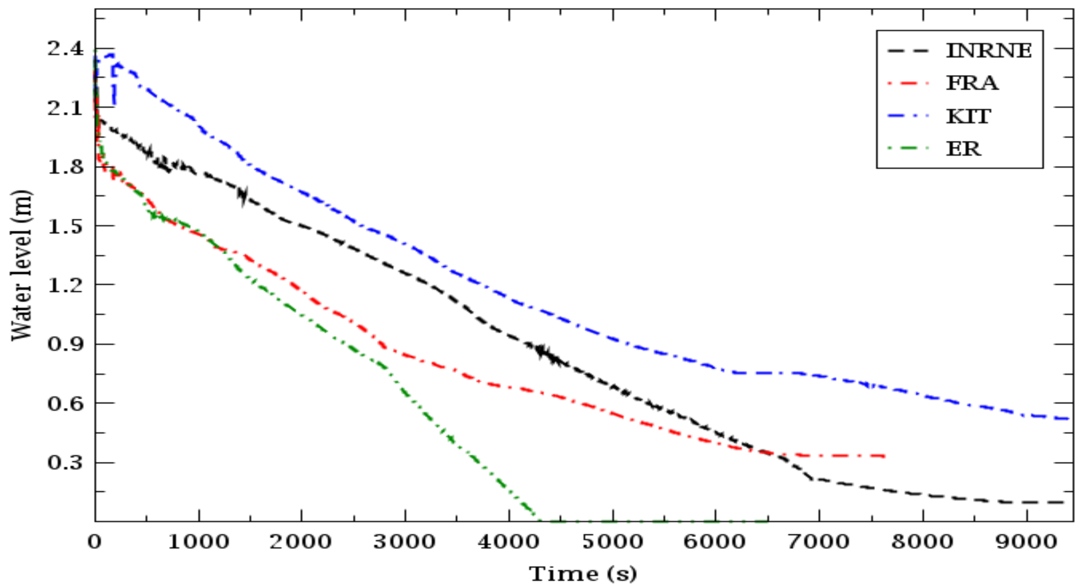


Figure 4.2.24 SG#4 water level

In the Figure 4.2.25 to Figure 4.2.28 are presented Steam generators water masses for all calculations. Residual heat is removed from the reactor core and primary side by the natural circulation as long as SGs are effective. The loss of natural circulation is due to the loss of primary coolant from one side and reducing the water mass of steam generators (SGs).

Due to the loss of feed waters (as SBO), the steam generator water mass is reduced significantly. Steam generators water mass for all teams has similar shape until 4000 s.

For both INRNE and FRA teams, it has almost identical behaviour until 6500 s, after which it is observed slightly increasing of SGs water mass for FRA team and it remains stable to end of the transient.

The SGs water mass for KIT team is overestimated compared to other teams, while for ER team is seen that the SGs water mass is underestimated and SGs dry out.

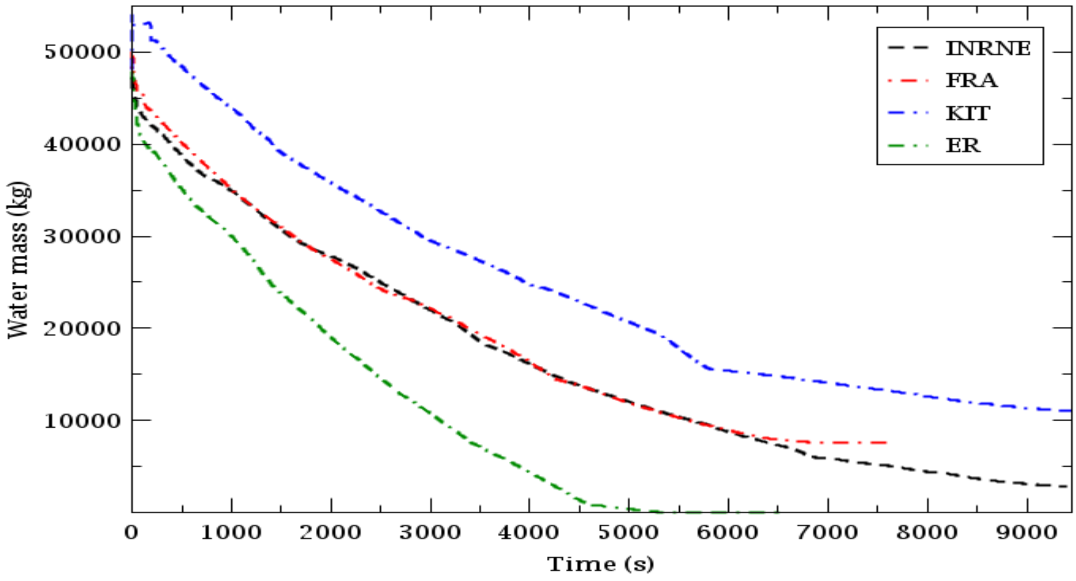


Figure 4.2.25 SG#1 water mass

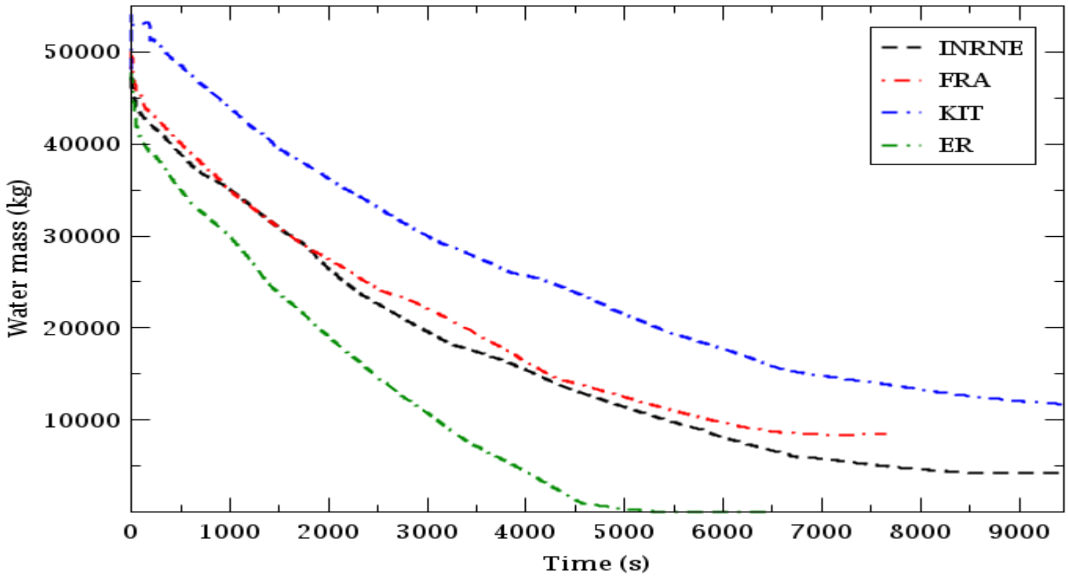


Figure 4.2.26 SG#2 water mass

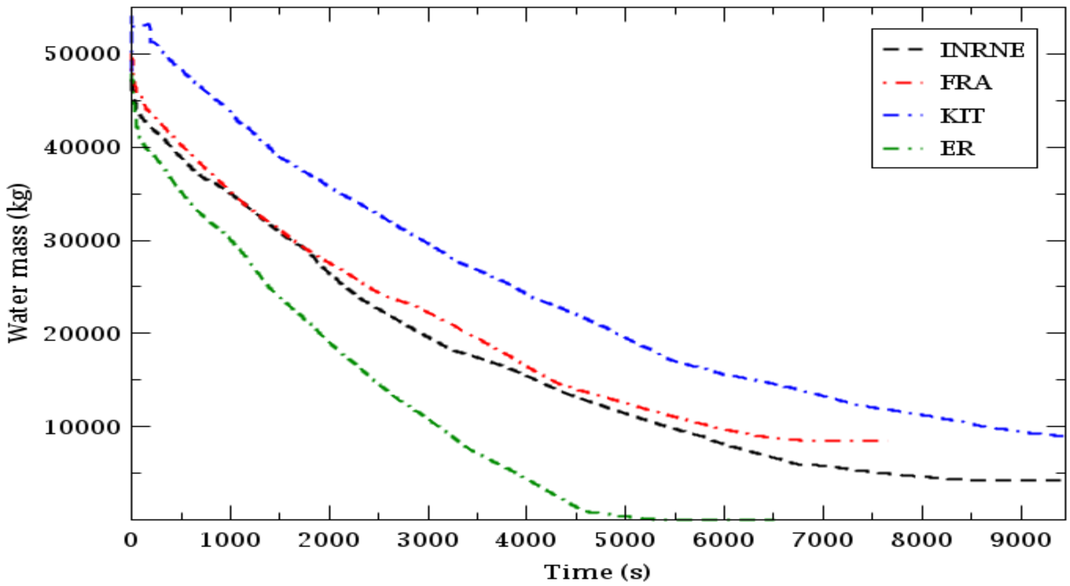


Figure 4.2.27 SG#3 water mass

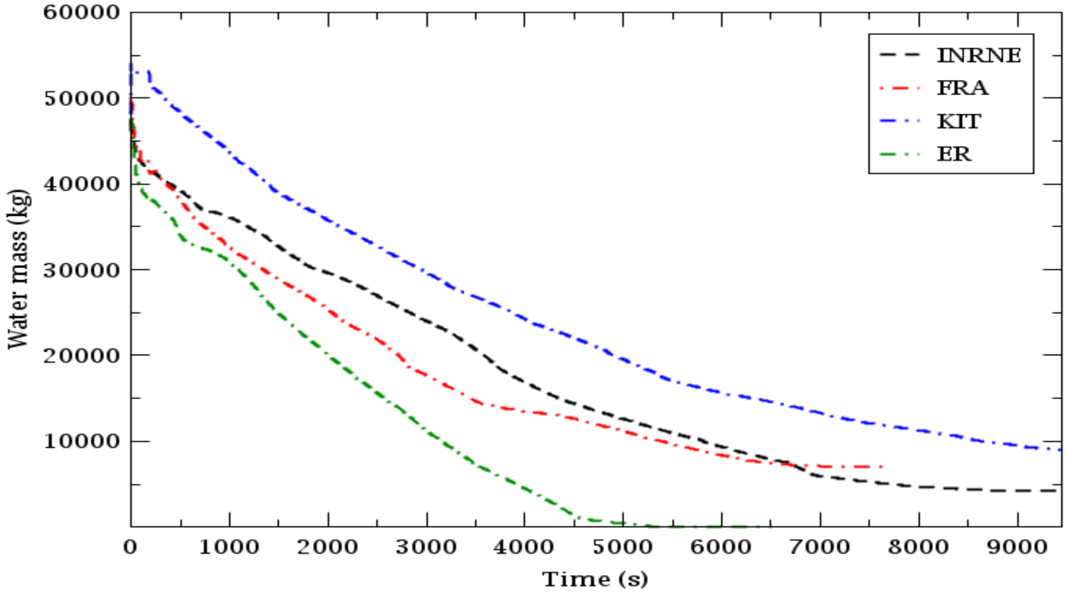


Figure 4.2.28 SG#4 water mass

Comparison of Coolant flow rate in hot legs for all calculations is shown in the Figure 4.2.29 to Figure 4.2.32. As it is shown in these figures, there is also a good agreement between all the codes. The behaviour of the coolant flow rates in hot legs follows similar trends.

In hot loops, the coolant started to boil for INRNE team at 630 s, FRA team at 590 s, KIT team 500 s and ER team at 420 s. The fluctuations in KIT model at 1000 s and 1450 s could be explained with the numerical instability.

The core uncovery is observed between 3500 s and 5200 s for all teams. The hot legs completely dry out and this leads to loss of natural circulation, also it deteriorates even further the cooling of the primary circuit.

The hot legs dryout occurs for all calculations as follows: for INRNE team is 3598 s, for FRA team 3165 s, for KIT team is 3540 s and for ER team is 3510 s.

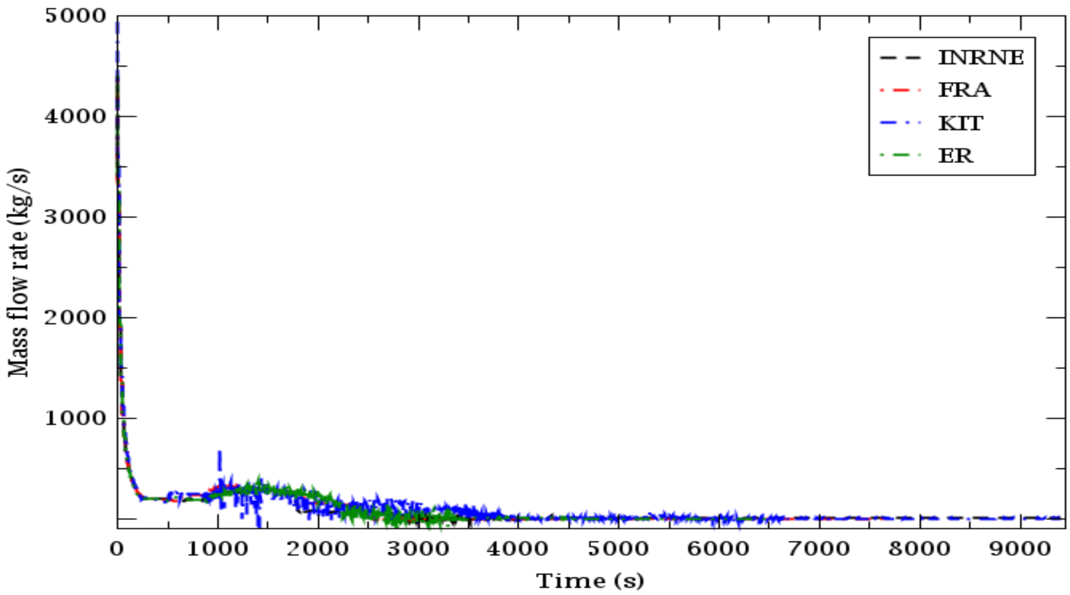


Figure 4.2.29 Coolant flow rate in hot leg#1

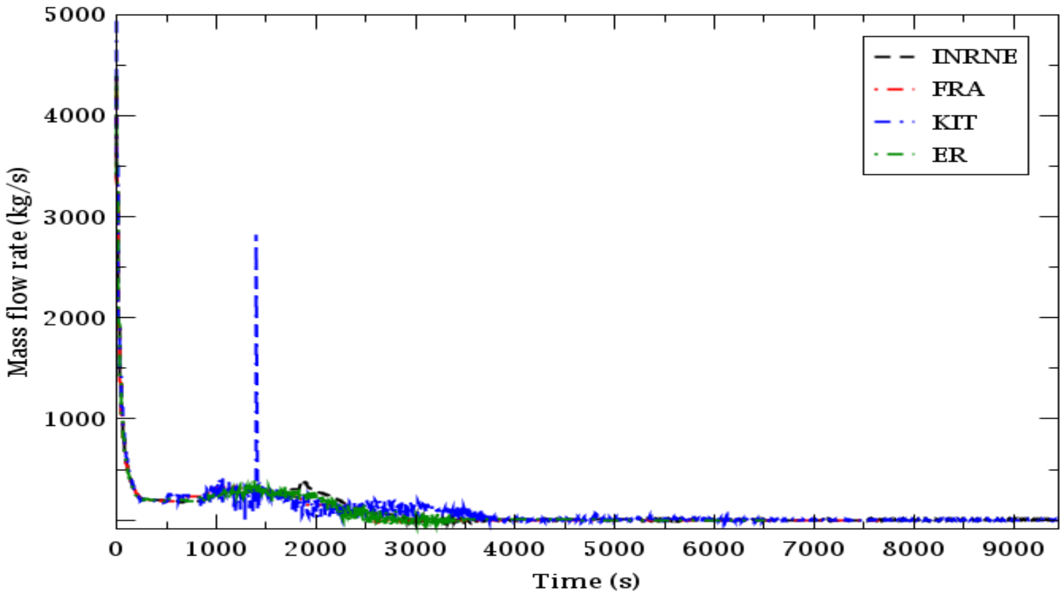


Figure 4.2.30 Coolant flow rate in hot leg#2

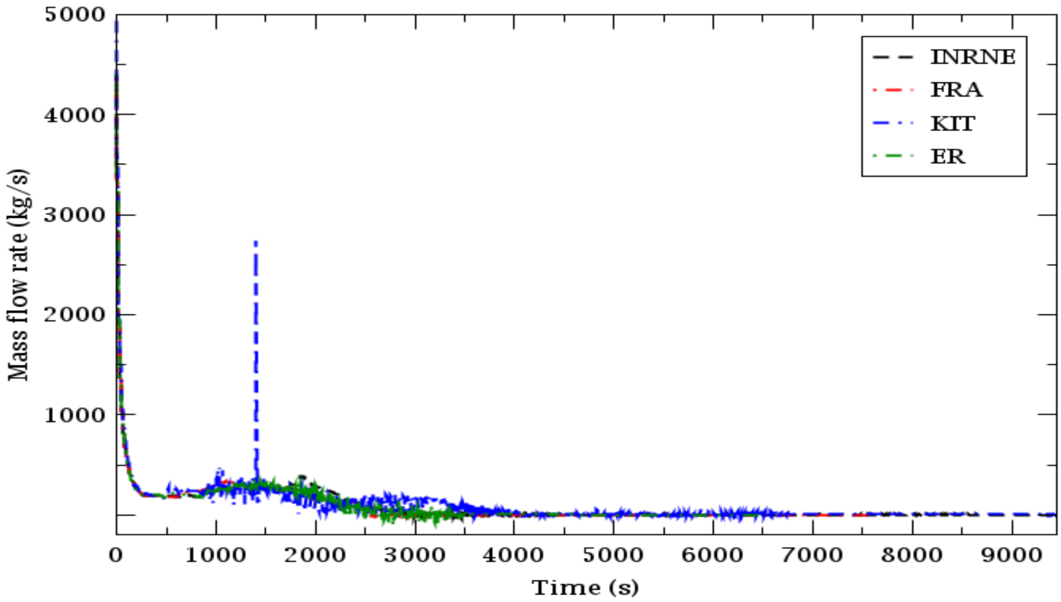


Figure 4.2.31 Coolant flow rate in hot leg#3

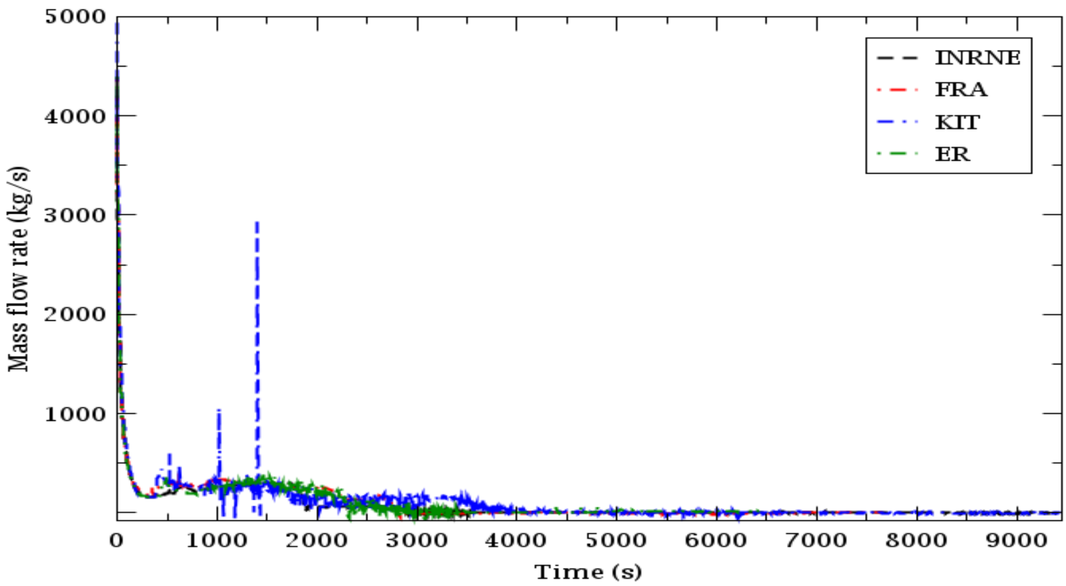


Figure 4.2.32 Coolant flow rate in hot leg#4

Comparison of Coolant flow rate in cold legs for all calculations is shown in the Figure 4.2.33 to Figure 4.2.36.

The comparison of calculated coolant loops flow rate in cold legs by the participants demonstrates similar trend. The loss of natural circulation is observed between 2500 s and 3540 s for different codes. The SGs are still removing primary heat by very small flow rate and observed fluctuations.

It can be seen very good agreements in flow rate of loop#1 predicted by different computer codes. Fluctuations are observed in KIT results compared to the other participants because of the numerical instability.

The observed differences in the participant's results could be explained by the use of different models and computer codes.

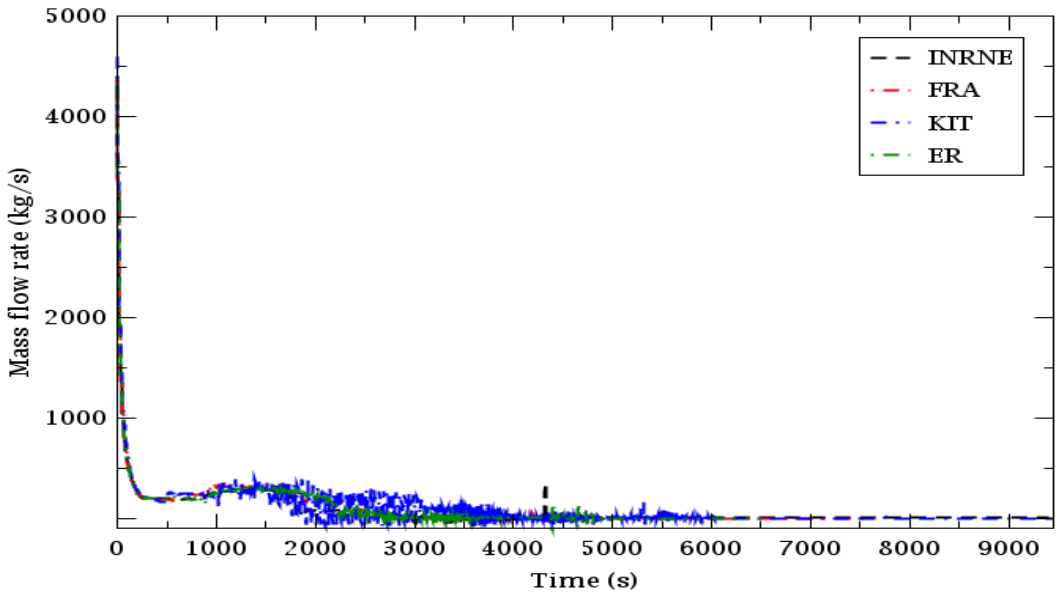


Figure 4.2.33 Coolant flow rate in cold leg#1

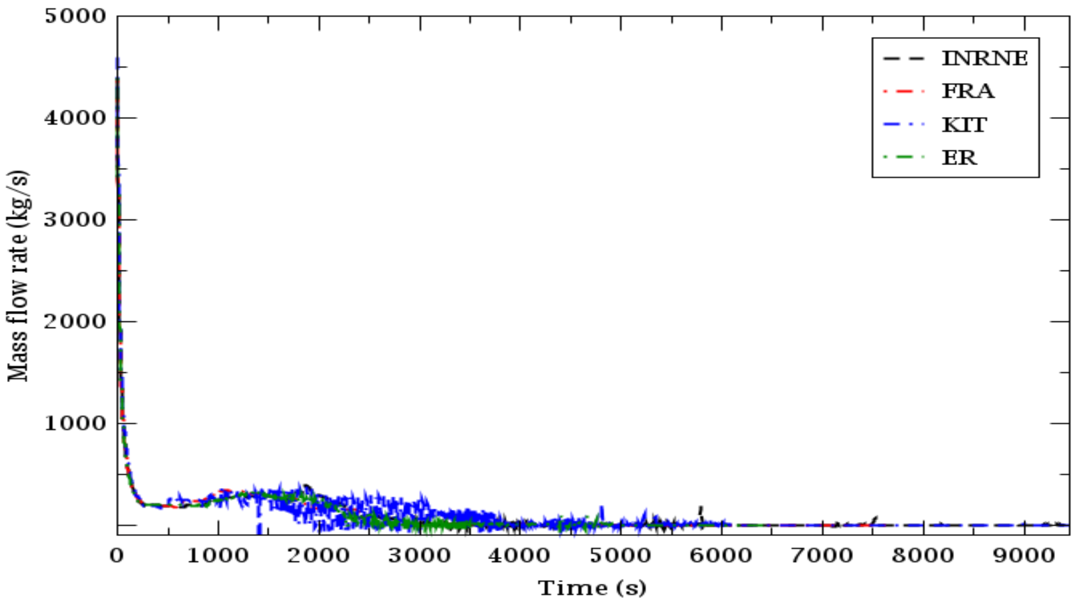


Figure 4.2.34 Coolant flow rate in cold leg#2

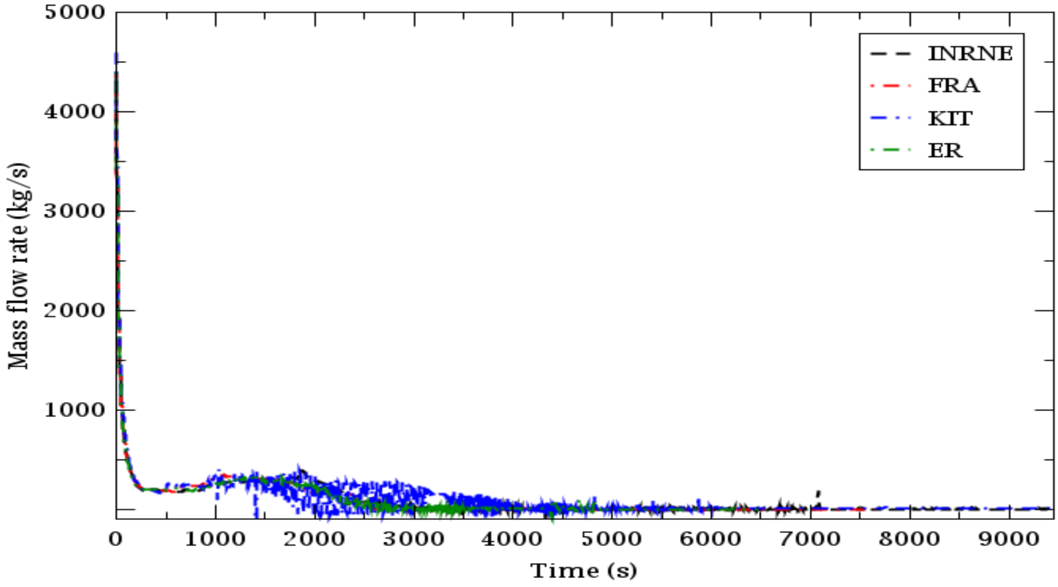


Figure 4.2.35 Coolant flow rate in cold leg#3

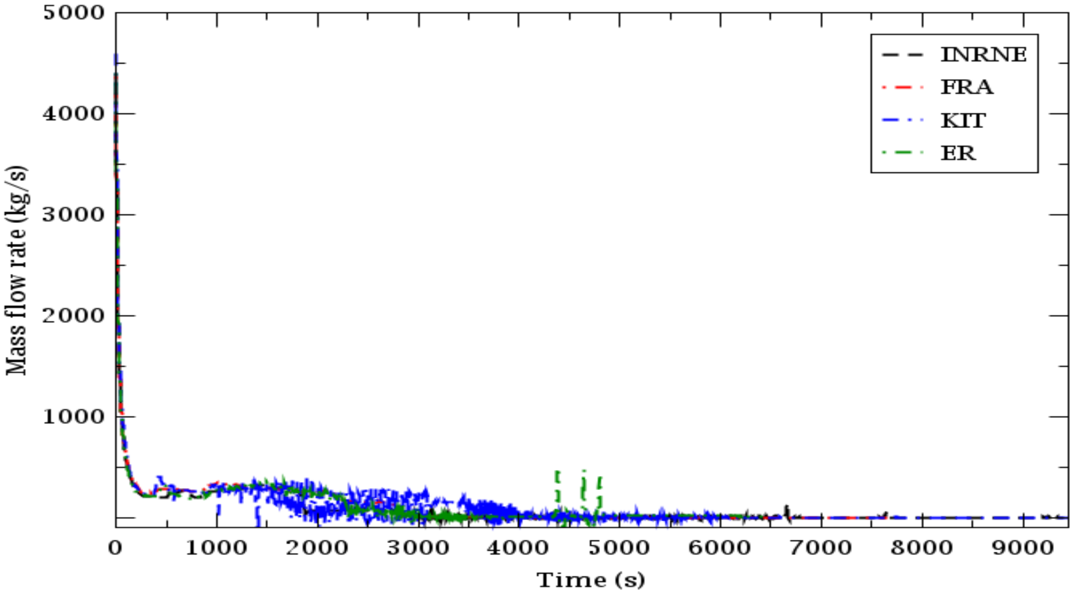


Figure 4.2.36 Coolant flow rate in cold leg#4

Comparison of Heat transfer for 4 SGs for all calculations is shown in the Figure 4.2.37 to Figure 4.2.40.

The comparison of calculated heat transfer in all steam generators shows similar trend. The initial HT in SG#4 is more than 750 MW for all participants.

The calculated heat transfer between primary and secondary side for INRNE team is observed to increase at around 20 s, after that it starts to decrease until 80 s and remains stable to the end of transient.

The trend of the calculated heat transfer between primary and secondary side seems to decrease smoothly during the whole transient for ER team.

It can be observed that the behaviour of the calculated heat transfer between primary and secondary side for FRA team follows the trend of INRNE team.

It is seen that calculated heat transfer between primary and secondary side for FRA team compared to others is higher and it is noticed fluctuations in the first 200 s of the calculation.

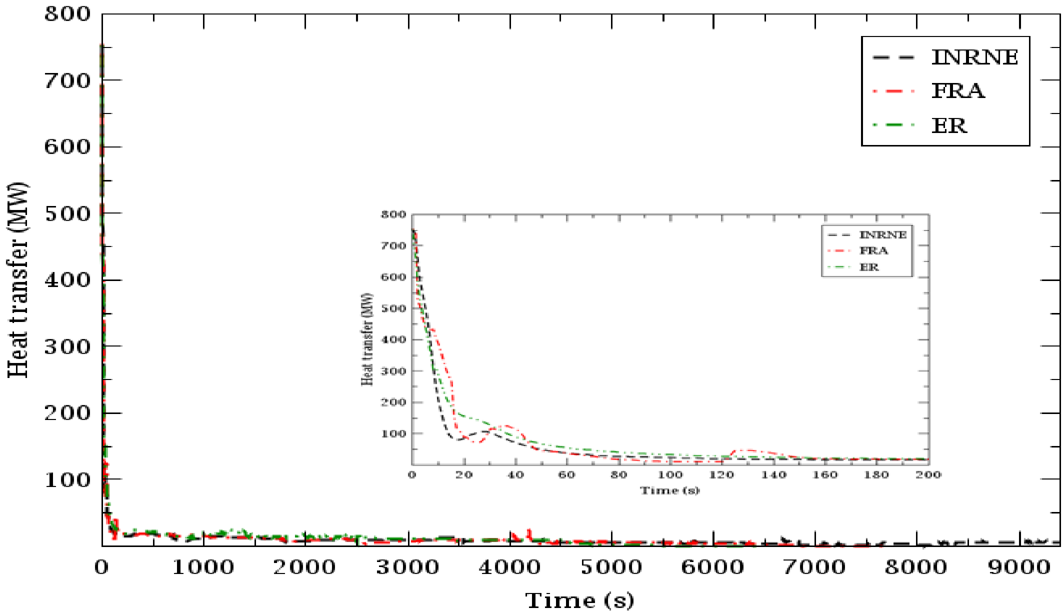


Figure 4.2.37 Heat transfer SG#1

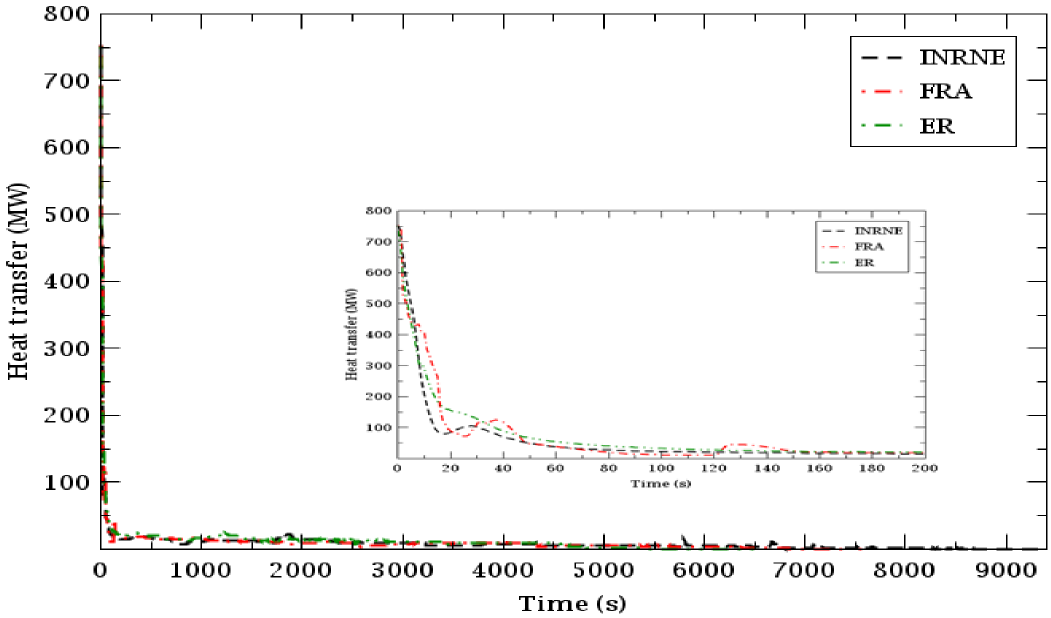


Figure 4.2.38 Heat transfer SG#2

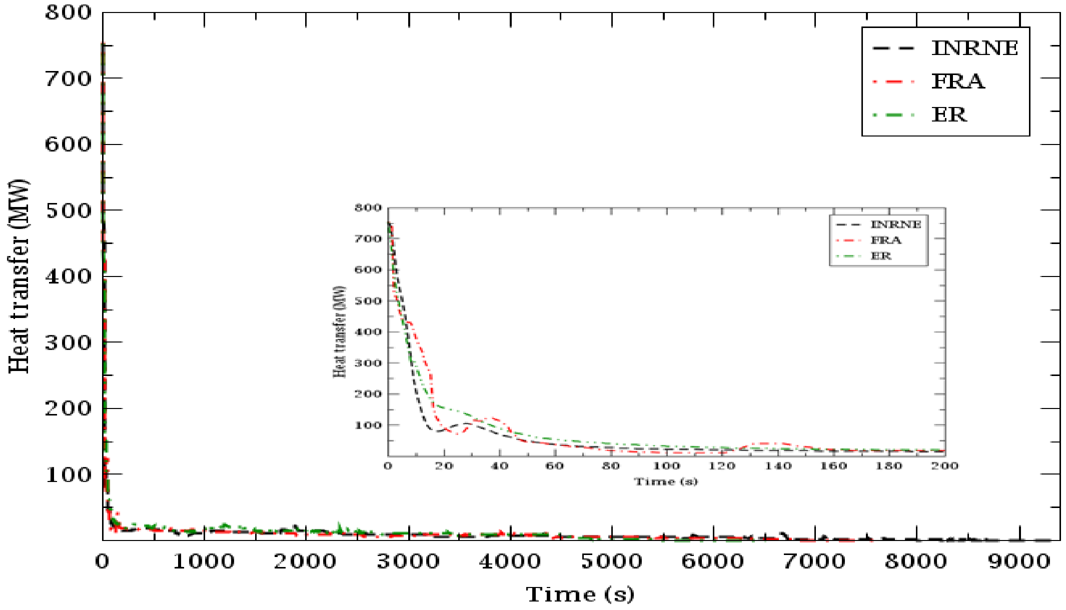


Figure 4.2.39 Heat transfer SG#3

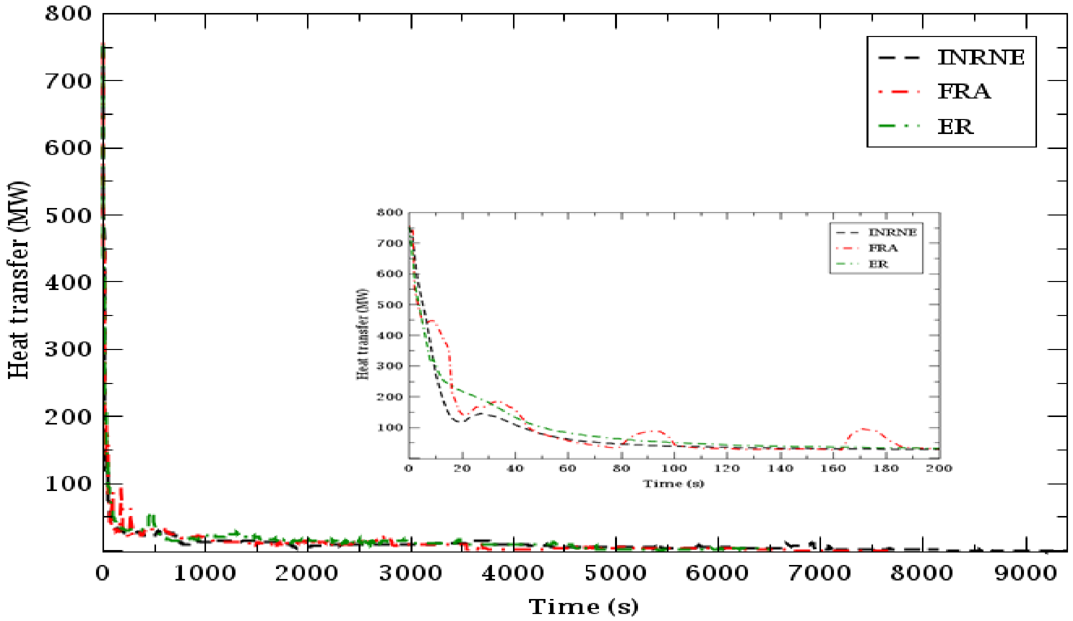


Figure 4.2.40 Heat transfer SG#4

At it is seen the heat transfer from primary to secondary side for the first 10 s is very close between participants. After the 10 s there a bigger heat transfer is observed for FRA team compared to the other participants. In Figure 4.2.40 it is observed after the first 50 s to the 200 s very similar results as in the other Figure 4.2.37 to Figure 4.2.39, but there are two very small increases in this period of time in heat transfer for FRA team at around 90 and 170 s. These increases in the heat transfer are synchronized with the opening of the BRU-As, caused by fluctuations of the secondary pressure.

Generally, the heat transfers from primary to secondary side is very close for all participants and cannot be considered as contributor to the deviations observed in some other parameters.

5 General conclusions, challenges and limitations observed during the performance of the Benchmark

By comparing the accident progressions predicted by all participants it could be concluded that all parameters' trends are very similar as well as the simulated process of transition from forced to natural circulation, dryout of pressurizers, integral break flow rates etc. From the other side there are some deviations that are observed and described and explained.

This section is intended to describe the aspects that might make it difficult to carry out the simulations and to interpret the code-to-code comparison. 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 present benchmark has challenged the existing system analysis codes in the case of an "SB LOCA + SG line break simultaneously with a SBO" scenario and has proven that there remains physical processes that could be further studied and they could be used for further model improvement. As calculating of residual power by partners causes faster dryout of SGs in some of calculations compared to the others. Very fast or very slow increasing of secondary pressure is also an example that could be included in further investigations. Some deviations in the predictions of deferential break flow rates could be also subjects for future investigations. The explanations presented in the report just explained possible reasons for such deviations. To avoid some observed deviations, not only the use of common initial and boundary conditions could be included, but also some common assumptions in modelling as, for example, in the modelling of the breaks.

The main phenomena expected during the transient were observed, 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 clearly be identified.

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.

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

6 REFERENCES

[1] "Grant agreement, NUMBER 945081 – CAMIVVER".

[2] "CAMIVVER Deliverable 3.2 “D3.2 - The CAMIVVER Definition report with specification for NPP with VVER 1000 reactor with respect to selected transients”, Work Package 3," 2021.

[3] "RELAP5/MOD3.3 CODE MANUAL, VOLUME I: CODE STRUCTURE,SYSTEM MODELS, AND SOLUTION METHODS, NUREG/CR-5535/Rev P5-Vol I, Prepared by Information Systems Laboratories, Inc. Rockville, Maryland, Idaho Falls, Idaho," June 2016.

[4] "U. NRC, TRACE V5.1051 Theory Manual, Washington, DC: Division of Safety Analysis Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission," 2016.

[5] K. Zhang, "The multiscale thermal-hydraulic simulation for nuclear reactors: A classification of the coupling approaches and a review of the coupled codes," *International Journal of Energy Research,* 2019.

[6] "MEDCoupling Developer’s Guide," SALOME-platform, [Online],Available: http://docs.salome-platform.org/latest/dev/MEDCoupling/developer/index.html. [Accessed 04 02 2019]..

[7] "Nuclear Safety Analysis Division 2001, RELAP5/Mod3.3 code manual Volume I: Code Structure, System Models, and Solution Methods., vol. 1.".

[8] "Deliverable 3.3 – Definition reports of SB LOCA + SG tubing break benchmark".

[9] V. Sanchez-Espinoza and M. Bottcher, "Investigations of the VVER-1000 coolant transient benchmark phase with the coupled system code RELAP5/PARCS," *Progress in Nuclear Energy,* vol. 48, 2006.

[10] A. Stefanova, N. Zaharieva, P. Vryashkova and P. Groudev,, "“The CAMIVVER Definition report with the specifications of rht NPP with VVER_1000 reactor with respect to selected transients,”," CAMIVVER, , Paris, 2020.

[11] KNPP, "Data Base for VVER1000: Safety Analysis Capability Improvement of KNPP (SACI of KNPP) in the field of Thermal Hydraulic Analysis," INRNE, Sofia, 2015.

[12] "NRC, "TRACE V5.840, User’s Manual: Input Specification," 2017".

[13] S. Iegan, A. Mazur, Y. Vorobyov, O. Zhabin and S. Yanovskiy,, ""TRACE/RELAP5 Comparative Calculations For Double-Ended LBLOCA and SBO",," NUREG/IA-0475, , November 2016..

[14] "Project of in-depth safety analysis of power unit № 5 of Zaporizhia NPP. Data base on the YPPU of unit № 5 of the ZNPP. Final report. 10044DL12R. ZAES. 2000.".