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D7.2 - Results of Kozloduy-6 MCP start-up transient benchmark

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Summary

The objectives of WP7 are to improve the thermal-hydraulics modelling of VVER plant, especially challenge robustness and validation of CATHARE3 in the context of VVER reactors. Task 7.2 is dedicated to simulation of Kozloduy-6 Main Coolant Pump start-up transient. OECD/NEA VVER-1000 Coolant Transient Benchmark (V1000CT) Phase 1 Exercise 1 (see [2]) has been performed by each participant. This transient has been experimented during Kozloduy-6 start-up tests: one coolant pump has been started – the three others already operating, resulting in asymmetrical feeding of the vessel. The main objective of Task 7.2 is to check the validity of models within Task 7.1 [3], regarding the experimental data and code-to-code comparison.

Task 7.2 participants are:

KIT with TRACE code

INRNE with RELAP5 Mod 3.3 code

FRAMATOME with CATHARE3 code

ENERGORISK with RELAP5 Mod 3.2 code.

This document presents the results of Kozloduy-6 MCP start-up transient benchmark. The results show consistency with the test data and validity of models developed within Task 7.1 [3].

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

Approval

Version	First Author	WP leader	Project Coordinator
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Abbreviations

BOC	Beginning of Cycle
BRU-A	Steam Dumping Device to the Atmosphere
BRU-K	Steam Dumping Device to the Condenser
DTC	Doppler Temperature Coefficient
EHTC	Electro Hydraulic Turbine Controller
EFPD	Effective Full Power Days
HP	Hot Power
MCP	Main Coolant Pump
MSC	Main Steam Collector
MSH	Main Steam Header
MTC	Moderator Temperature Coefficient
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
OECD	Organization for Economic Co-operation and Development
PRZ	Pressurizer
RPC	Reactor Power Controller
RCS	Reactor Coolant System
SG	Steam Generator
TG	Turbo Generator
VVER	Water-Water Cooled Reactor

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1 DESCRIPTION OF VVER 1000 KOZLODUY – 6 "MCP START-UP" TRANSIENT BENCHMARK

1.1 Introduction

This Deliverable 7.2 on "Results of Kozloduy-6 MCP start-up transient benchmark" is a part of CAMIVVER, WP7, Task 7.2: "Simulation of Kozloduy-6 Main Coolant Pump start-up transient" in accordance with the CAMIVVER Grant agreement, NUMBER 945081 [1].

WP7 objectives are to improve thermal-hydraulics modelling of VVER plant, especially challenge robustness and validation of CATHARE3 in the context of VVER reactors. Task 7.2 is dedicated to simulation of Kozloduy-6 Main Coolant Pump start-up transient. OECD/NEA VVER-1000 Coolant Transient Benchmark (V1000CT) Phase 1 Exercise 1 [2] has been performed by each participant. This transient has been experimented during Kozloduy-6 start-up tests: one coolant pump has been started – the three others already operating, resulting in asymmetrical feeding of the vessel. The main objective of Task 7.2 is to check the validity of models developed within Task 7.1 [3], regarding experimental data and code-to-code comparison.

Task 7.2 participants are:

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

Objectives of the MCP start-up test analysis:

The purposes of this analysis are to test the primary- and secondary system model responses. The plant specification provides all the necessary point-kinetics data. Using this test, the partners can verify their input decks and they can eliminate all the problems coming from the user modeling, which later could be helpful for the best estimate comparisons.

The investigation of nuclear power reactor parameters behaviour in case of switching on one MCP when the other three MCPs are in operation on Unit 6, Kozloduy NPP [2], [4] called a "MCP start-up test", is an experiment that was conducted by Bulgarian and Russian engineers as a test during the plant commissioning phase at the Kozloduy Nuclear Power Plant - Unit #6 [2], [5], [6]. It was part of the start-up tests. The test was done according to the "Program for investigating of nuclear power reactor parameters behaviour in case of switching on MCPs". The purpose of the experiment was the complete testing of reliability of all power plant equipment, testing the reliability of the main regulators (RPC, EHTC and SG water level regulator) and defining a jump of the neutron reactor power in case of switching on one MCP when the other three MCPs are in operation.

The detailed description of the Kozloduy Unit 6 is presented in the Deliverable 3.2 [7]. The initial and boundary conditions are presented in this section.

1.1.1 Initial Steady State Conditions

Description of the MCP Switching on Test [2], [7], [8]:

Switching on one MCP when the other three MCPs are in operation is an experiment that was conducted by the Bulgarian and Russian engineers as a test during the plant commissioning phase at Kozloduy NPP, Unit 6. It was a part of the start-up tests, so that the main parameters behaviour to be investigated.

Initially the reactor has been operating at 50% of the nominal power for 20 Equivalent Full Power Days (EFPD) and after that the power was increased up to 75% for 10.7 EFPD.

Before the experiment reactor power level was reduced from 75% (2250 MW) to approximately 21% by consecutive switching off MCP#2 and MCP#3. A few hours before the experiment MCP#2 were switched on, and the power was stabilized at approximately 30%, following the Technical specification requirements.

According to the Technical specification for safety operation of the Units 5 and 6, switching on one MCP in operation is performed when the reactor power is 30 % of the nominal.

The event is characterized by rapid increase in the flow through the core resulting in a coolant temperature decrease, which leads to insertion of positive reactivity due to the modelled feedback mechanisms.

There is an axial neutronics asymmetry in the core during the transient. At the beginning of the test there is also a radial thermal-hydraulic asymmetry due to colder water coming from loop #3 into the reactor core. The test was made as it is important for the safety of the NPP with VVER-1000, model 320.

Initial conditions:

The initial conditions of Unit 6 before the test onset are:

- The reactor is at the beginning of cycle (BOC) with average core exposure of 30.7 EFPD and boron concentration 5.95 g/kg H_2O.
- Reactor power is 29.45% of the nominal power level. The reactor power increase to 29.8% after switch on MCP#3;
- MCPs #1, #2 and #4 are working under stable conditions and MCP #3 is out of operation.
- The inlet temperature in the reactor core is about 555 K.
- The Electro-Hydraulic Regulative System supports the pressure in MSH, at level 61.9 ±0.5 kgf/cm^2. All regulators are in automatic regime.

- The temperature differences between the hot and cold legs for the loops with working MCPs vary between 8.3 - 11.5 K, while the same temperature difference for loop #3 with MCP out of operation is negative due to the reverse flow (- 3.6 K).
- The total mass flow through the core is about 13 611 kg/s with average flow of 5 000 kg/s through each of the working loops and negative (reverse) flow of -1 544 kg/s in loop #3.
- There is a core axial non-symmetry as it could be seen from the initial value of the axial core power distribution.

The main measured plant data parameters of initial conditions on Unit 6 are given in the “CAMIVVER Definition report with specification for NPP with VVER 1000 reactor with respect to selected transients” – CAMIVVER Deliverable 3.2 [7] (see also Table 3.1.1 in Section #3 of this document).

Main physics data for MCP start up test:

The reactor physics parameters important for point kinetics analysis are presented in MCP Specification report [2], [7], [8] and summarized in the Table 1.1.1.1

Table 1.1.1.1 Summary of reactor physics parameters

Parameter	Value
HP MTC, pcm/K	-3.1
HP DTC, pcm/K	-1.661
HP delayed neutron fraction (β_{eff})	0.7268E-02
HP prompt neutron lifetime	0.267E-04
Control rod group #10 worth, %dk/k	0.91
Ejected rod worth, %dk/k	0.15
Tripped rod worth, %dk/k	7.85
Control rod group #5 worth, %dk/k	0.2

Table 1.1.1.2 Decay constant and fractions of delayed neutrons

Group	Decay constant (s^{-1})	Relative fraction of delayed neutrons in %
1	0.0125	0.0209
2	0.0305	0.1493
3	0.111	0.1368
4	0.305	0.2866
5	1.13	0.0984
6	3.0	0.0348

The total fraction of delayed neutrons: 0.7268%. The initial HP core average axial relative power distribution is presented in Table 1.1.1.3. [2], [7].

Table 1.1.1.3 Initial HP core average axial relative power distribution for 1st campaign

Bottom

0.82	0.86	0.89	0.91	0.92	0.93	1.02	1.32	1.21	1.13

Top

1.1.2 Transient scenario

The transient test scenario is as follows:

1. At reactor power 29.45% Nnom MCP#3 is switched on;
2. The flow rate in loop #3 reverses back to normal at 13th sec. after MCP #3 is switching on. The timing is consistent with reactivity increase, as observed through the reactor power setpoints;
3. The reactor power increases after MCP #3 is switching on due to the positive reactivity insertion in the reactor core caused by overcooling of coolant in the main coolant loop #3;
4. The Pressurizer water level decreases from 7.44 m to 7.28 m. There is a connection between RCS temperature and PRZ water level;
5. The water level in SG #3 decreases with 9 cm;
6. EHTC is supporting the pressure in MSH at level 61.9 $\pm$ 0.5 kgf/cm^2 (6.0+0.05 MPa) when the TG power is 164.0 $\pm$ 10 MW.

During the transient as a result of switching on MCP #3 there is an increased mass flow through the core. The core cooling is improved and re-distributed while the thermal core power level increases slightly during the transient (the total power increases from 29.45% to 29.8 % of nominal level).

At the end of the transient the temperature difference between hot and cold legs of loops #1, #2 and #4 slightly decreases:

- for loop #1 - from 11.5 K to 8.8 K;
- for loop #2 - from 8.3 K to 8.4 K:
- for loop #4 - from 10.9 K to 8.9 K;
- for loop #3 - from -3.6 K to 8.2 K

The most noticeable change is in the temperature difference for loop#3. This results in a dynamically changing spatial distribution of reactivity feedback during the transient and subsequently in a dynamically changing spatial power distribution. This type of effect is suitable for 3D kinetics code analyses. The present analysis demonstrates an increase of total reactor power and a change in total reactivity.

MCP start-up test plant data during the transient are presented in the MCP Specification report [2], [7]. These data have been distributed among the participants in this benchmark in Excel tables as well.

2 MODELS DESCRIPTION

2.1 General Code Features

Table 2.1.1 Information of the codes used and contact persons

Institution	Code	Contact person (s)
KIT	**TRACE V50P5**	**ETCHETO Javier**
INRNE	**RELAP5 mod 3.3**	**STEFANOVA Antoaneta**
FRAMATOME	**CATHARE 3 v2.1**	**BERNARD Olivier**
LLC ENERGORISK	**RELAP5 mod 3.2**	**HASHYMOV Artur**

2.1.1 Brief model description of KIT

The used computer code by KIT is TRACE (TRAC/RELAP Advanced Computational Engine).

TRACE is a thermal hydraulics code designed to consolidate and extend the capabilities of NRC's 3 legacy safety codes - TRAC-P, TRAC-B and RELAP. It is able to analyze large/small break LOCAs and system transients in both pressurized- and boiling-water reactors (PWRs and BWRs). The capability exists to model thermal hydraulic phenomena in both one-dimensional (1-D) and three-dimensional (3-D) space. The 3-D modelling feature was used in this work to simulate the reactor pressure vessel.

2.1.2 Brief code description of INRNE

The INRNE have used thermal-hydraulic RELAP5 computer code, which is a light water reactor transient analysis code [9] and it 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 non-condensable 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.

The RELAP5 hydrodynamic model is a one-dimensional, transient, two-fluid model for flow of a two-phase steam-water mixture that can contain non-condensable components in the steam phase and/or a soluble component in the water phase [3], [9].

2.1.3 Brief code description of FRAMATOME

The CATHARE (Code for Analysis of Thermal-Hydraulics during an Accident of Reactor and Safety Evaluation) code used from FRA 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 equation 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.

2.1.4 Brief code description used of LLC ENERGORISK

The thermohydraulic code of the improved evaluation RELAP5/MOD3.2 used from ER was developed at USA, at the Idaho National Technical Laboratory (INEL) and it is designed to simulate the following emergency and transient modes of nuclear power plants with water-cooled reactors:

a) loss of coolant, loss of feed water, rupture of steam lines, rupture of steam generator tubes;

b) transient modes not related to circuit depressurization.

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 scope of this code is limited to calculations of emergency modes without core destruction.

Code RELAP5/MOD3.2 is based on a one-dimensional, two-fluid model of a steam-water mixture. The model considers the phases of steam and water. The concept of a two-fluid model implies that for each phase, as for a separate liquid, the continuity equation, the momentum conservation equation, and the energy conservation equation are written. Each phase has its own velocity and temperature, that is, in general, steam and liquid are not in mechanical and thermal equilibrium with each other.

In addition to the basic two-fluid model, RELAP also allows you to describe the behaviour of non-condensing gases in steam, while it is assumed that the non-condensing gas is in thermal and mechanical equilibrium with steam, that is, the temperature and velocity of the non-condensing gas are equal to the temperature and velocity of steam. The hydrodynamic model of REL allows describing the transport of boron by the liquid phase, while it is assumed that boron is in mechanical equilibrium with the liquid phase.

2.2 Description of VVER 1000 computer models

2.2.1 KIT model description

The Kozloduy 6 plant model for the primary and secondary circuits used by KIT has been thoroughly described in Deliverable 7.1 (D7.1) **Error! Reference source not found.** and has not b een modified for this task.

The reactor pressure vessel is modelled using a TRACE 3-D vessel component with a cylindrical discretization (axial levels, rings and angular sectors). The main zones of the vessel, i.e. core, bypass, upper plenum, downcomer and lower plenum, are modelled explicitly. The exact dimensions of the nodalization scheme are specified in D7.1. The reactor core is modelled using point kinetics, with the kinetic parameters defined in the benchmark specifications.

Aside from the reactor pressure vessel, the plant model consists of the primary cooling circuit (cold and hot legs, reactor coolant pumps, pressurizer), the secondary circuit (feed-water and steam lines) and the steam generators linking both systems. The model follows the benchmark specifications, and a complete description can be found in D7.1 [3] as well.

2.2.2 INRNE model description

For the purpose of MCP switching on investigation it has been used RELAP5/MOD3.3 computer code [9]. The Baseline input deck for VVER-1000/V320 Kozloduy Nuclear Power Plant Unit 6 is developed by the Institute for Nuclear Research and Nuclear Energy - Bulgarian Academy of Sciences (INRNE-BAS) [10]. The model was developed for analyses of operational occurrences, abnormal events, and design basis scenarios. The model provides a significant analytical capability for the specialists working in the field of the NPP safety. Data and information for the modelling of these systems and components were obtained from the Kozloduy documentation and from the power plant staff.

The model was defined to include all major systems and equipment of the Kozloduy NPP, namely: core, reactor vessel, Main Coolant Pumps (MCP), Steam Generator (SG), Steam Generator steam line system and main steam header (MSH), emergency protection system, pressure control system of the primary circuit, makeup system, safety injection system, steam dumping devices (BRU-K, BRU-A, SG and Pressurizer safety valves), and main feed water system.

In the RELAP5 model of the VVER-1000, the primary system has been modelled using four coolant loops representing the four reactor loops. The RELAP5 model configuration provides a detailed representation of the primary, secondary, and safety systems. The following model includes: Reactor vessel including a downcomer, lower plenum, and outlet plenum; Core region represented by a hot and average heated flow paths and a core bypass channel; Pressurizer (PRZ) system with heaters, spray, and pressurizer relief capability; Safety system representation including the accumulators-, high- and low-pressure injection systems, and the reactor scram system; Make up and Blowdown system including the associated control systems.

In the RELAP5 VVER 1000 model, the secondary system has been modelled using four steam lines and four steam generators. The upper steam volume of each steam generator is modelled as a steam separator.

Detailed description of RELAP5/MOD3.3 VVER-1000 model, including all major systems and equipment of the Kozloduy NPP is presented in the Deliverable D7.1 [3].

2.2.3 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. Just the loop 1 is presented. The safety systems are not presented.

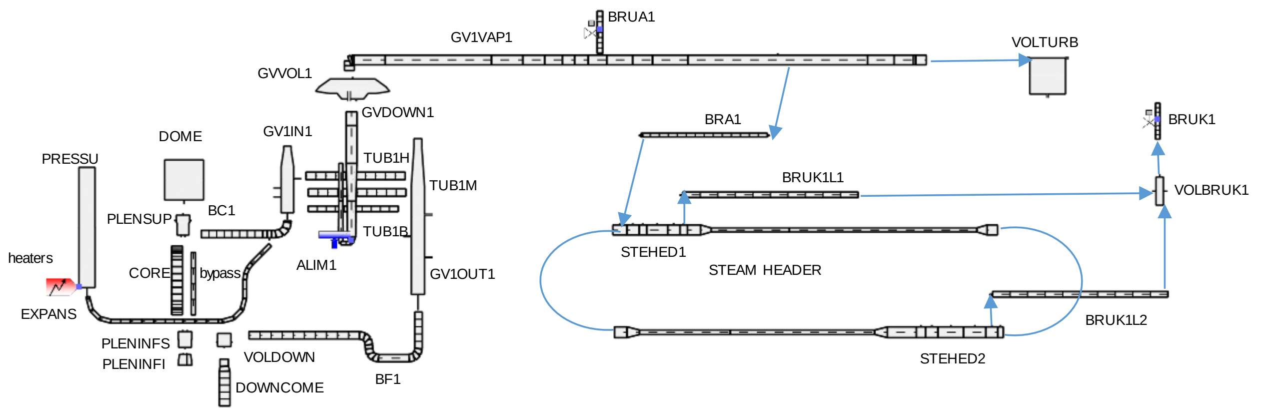

Figure 2.2.3.1 Kozloduy CATHARE3 model

For the computation, the primary and secondary circuits are modelled with several **elements**: pipes, capacities and boundary conditions, the names of which are as follows:

Primary Circuit:

BC_1C BC_2C BC_3C BC_4C GV1IN1 GV2IN1 GV3IN1 GV4IN1 TUB1H TUB2H TUB3H TUB4H TUB1M TUB2M TUB3M TUB4M TUB1B TUB2B TUB3B TUB4B GV1OUT1 GV2OUT1 GV3OUT1 GV4OUT1 BF1 BF2 BF3 BF4 VOLDOWN DOWACC DOWNCOME PLENINFI PLENINFS BYPASS_CORE COEUR PLENSUP DOME PLSACC EXPANS PRESSU (PZR)

Secondary Circuit:

GVDOWN1 GVDOWN2 GVDOWN3 GVDOWN4 GVVOL1 GVVOL2 GVVOL3 GVVOL4 GV1VAP1 GV2VAP1 GV3VAP1 GV4VAP1 BRUA1 BRUA2 BRUA3 BRUA4 BRA1 BRA2 BRA3 BRA4 STEHED1 STEHED2 BRUK1L1 BRUK1L2 VOLBRUK1 BRUK1 BRUK2L1 BRUK2L2 VOLBRUK2 BRUK2 BRUK3L1 BRUK3L2 VOLBRUK3 BRUK3 BRUK4L1 BRUK4L2 VOLBRUK4 BRUK4 VOLTURB

For a detailed descriptiion of the different elements, refer to the Deliverable D7.1 [3].

2.2.4 LLC ENERGORISK model description

The model of the VVER-1000 reactor plant was developed using the calculation code RELAP mod 3.2. The model includes a reactor, first and second circuit systems, as well as auxiliary systems. The main systems of the first and second circuits are also modelled in the model: main circulation pipelines, MCP, steam generators, pressure compensation system, systems for feeding and maintaining the level in PG and others. For systems whose operation has a direct impact on the results of the simulation analysis, a detailed modelling of auxiliary systems is performed at the functional level.

The developed model of the reactor is a 4-sector one with cross-links to simulate flows between sectors (Figure 2.2.4.1). 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.

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. Further on, this reactor model will be called «asymmetric» in the text. 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».

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.

A more detailed description of the model is given in the Deliverable D7.1 [3]

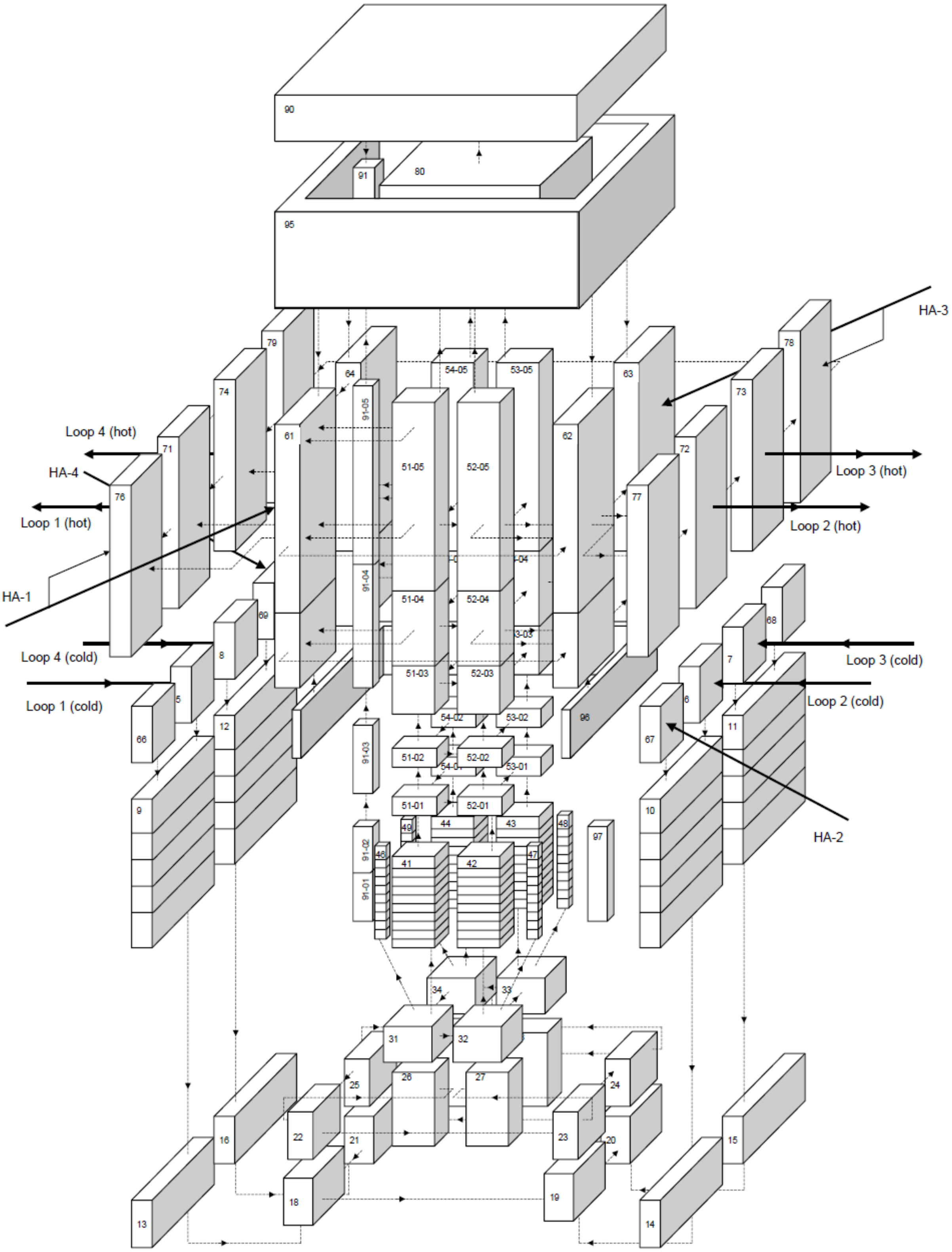

Figure 2.2.4.1 Nodalization diagram of the reactor

3 RESULTS AND DISCUSSION

This section presents the comparative analyses of the Kozloduy-6 MCP start-up transient benchmark.

Expected results of Kozloduy-6 MCP start-up transient benchmark are as follows:

• Modelling of MCP start up test;

• The calculated results to be compared with the test data, prepared in the framework of Task 3.2 [7] and uncertainty differences in the results of modelling on different models to be analyzed.

Output requested by the participants

• All data should be prepared in SI System (kg, m, s);

• For time histories, data should be at 1.0 second;

• The requested data should be provided in EXCEL format by each of participants in the benchmark (the EXCEL table with the requested data have been distributed to the participants).

The initial steady state results and the transient results are compared.

3.1 Initial state

The main measured plant data parameters of initial steady state and calculation results are presented in Table 3.1.1.

Table 3.1.1 Comparison of main measured plant parameters of initial steady state conditions and calculation results during MCP start up test for KNPP unit 6 at 883.5 MWt

Parameters	Value	Exp. data	Uncertainty	KIT	INRNE	FRA	ER
Core power, at 29,45%	MW	**883,5**	±60	883,5	883,5	883,5	883,5
Primary side pressure,	MPa	**15,6**	±0.3	15,6	15,6	15,6	15,59
RCS first cold leg temperature,	K	**555,55**	±2.0 K	555	555,08	554,64	555,7
RCS second cold leg temperature	K	**554,55**	±2.0 K	553,9	554,56	554,65	556,39
RCS third cold leg temperature	K	**554,35**	±2.0 K	554,3	554,84	554,91	556,46
RCS fourth cold leg temperature	K	**555,25**	±2.0 K	554,8	555,06	554,63	557,35
RCS first hot leg temperature	K	**567,05**	±2.0 K	567,2	566,2	565,59	569,77
RCS second hot leg temperature	K	**562,85**	±2.0 K	562,5	564,72	565,59	565,34
RCS third hot leg temperature	K	**550,75**	±2.0 K	551	550,2	549,44	551,95
RCS fourth hot leg temperature	K	**566,15**	±2.0 K	566,2	566,23	565,59	568,76
Core flow rate	kg/s	**13611**	±800.0	13061	13382	13588	13226
First loop flow rate	kg/s	**5031**	±200.0	5046,6	4975,54	5053	4951
Second loop flow rate	kg/s	**5069**	±200.0	5094,3	4992,37	5053	4977
Third loop flow rate	kg/s	**-1544**	±200.0	-1753,7	-1574,49	-1571	-1617

Fourth loop flow rate	kg/s	**5075**	±200.0	5073,5	4980,3	5053	4935
Pressurizer level	m	**7,44**	±0.15	7,44	7,43	7,44	7,51
Water level in SG1	m	**2,3**	±0.075	2,3	2,28	2,30	2,31
Water level in SG2	m	**2,41**	±0.075	2,41	2,40	2,40	2,35
Water level in SG3	m	**2,49**	±0.075	2,49	2,45	2,50	2,48
Water level in SG4	m	**2,43**	±0.075	2,43	2,43	2,40	2,35
Secondary side pressure	MPa	**5,937**	±0.02 MPa	6,098	6,16	5,962	6,22
Pressure difference in the reactor	MPa		±0.02 MPa	0,135	0,26	0,24	0,292
Pressure difference in MCP1	MPa	**0,49**	±0.02 MPa	0,563	0,49	0,51	0,501
Pressure difference in MCP2	MPa	**0,47**	±0.02 MPa	0,56	0,49	0,51	0,500
Pressure difference in MCP3	MPa	**0,18**	±0.02 MPa	0,151	0,19	0,18	0,177
Pressure difference in MCP4	MPa	**0,50**	±0.02 MPa	0,56	0,49	0,51	0,507

The table above presents the main plant initial parameters in comparison with the stabilized parameters by the participants. The comparison of steady state results shows very good agreement of stabilized initial parameters compared to the plant data. The comparison of hot and cold leg temperatures in each coolant loops are in good agreement, except hot leg temperature in loop #2. It is observed some overestimation of initial hot temperature in second loop in both organizations: FRA and ER. Also, it is observed significantly big difference at initial value of PRZ level in ER result.

The initial MCP pressure drop plant records indicates different initial values in each loop, while in the initial stabilized parameters by most of the participants are identical in Loop #1, #2 and #4, except Loop #3. The KIT stabilized initial value of MCP pressure drop is slightly out of the range of the uncertainty in all loops.

The comparisons of all other parameters are in their ranges of the uncertainties, like: primary side pressure, flow rate in loops, flow rate through the core and etc.

The comparison of all parameters vs plant data shows that reached steady states conditions by all the participants provides stable inputs for run of investigated transient “Switching on one MCP when the other three MCPs are in operation”.

3.2 Transient

In this section are presented comparison results of main selected parameters in comparison with available plant data.

The table of event time calculations during the transient was requested by the participants. The results are presented in Table 3.2.1

Table 3.2.1 Event time calculations

Event description	Experiment, s	KIT	INRNE	FRA	ER
MCP #3 switched on	0,0	0,0	0,0	0,0	0,0
Minimum pressure above the core	12	18	18	18	15
MCP #3 pressure drop stabilizes	12	16	14	17	16
Reactor pressure drop stabilizes	16	15	15	16	16
Primary side pressure stabilizes	app. 25	-	-	-	31
Pressurizer water level stabilizes	app. 60	-	53	-	60
Hot leg #3 temperature stabilizes	60	10	-	-	24
Transient end	129	129	129	129	129

Graphical comparison of calculated results and test data have been performed.

3.2.1 Total core power

The comparison of reactor core power is presented on Figure 3.2.1.1. There is no plant data record for the reactor power during the transient. As a whole, the comparison of calculated core power by the participants shows similar trends. The predicted reactor power by all participants has the same initial values, after the transient initiation. The core power starts to increase and reaches different total value from all of the participants. Both organizations: INRNE and FRA predicted almost the same power increase, the max. core power predicted by INRNE is 939 MW (31.3% Nnom), while FRA value is 941 MW (31.4% Nnom). The other two organizations predicted significantly higher rate of core power increase. The observed max. core power by KIT at 128 s is 975 MW (32.5% Nnom) and a little bit lower power predicted by ER 964 MW (32.1% Nnom). The observed higher core power at the end of test could be explained with coolant temperature decrease, which leads to a positive reactivity. Also, it reflects, to core power increase, which begins to decrease after increase in the parameters of the reactor plant due to negative power feedbacks.

Based on the test record, the core power increases from 29.45% to 29.8 % of nominal level.

Generally, the comparison shows that both organizations: INRNE and FRA are in the range of the measurement uncertainty of +/-60MW, while the other two organizations exceed slightly the range of the uncertainty. The observed differences could be explained with using of different codes and models by the participants.

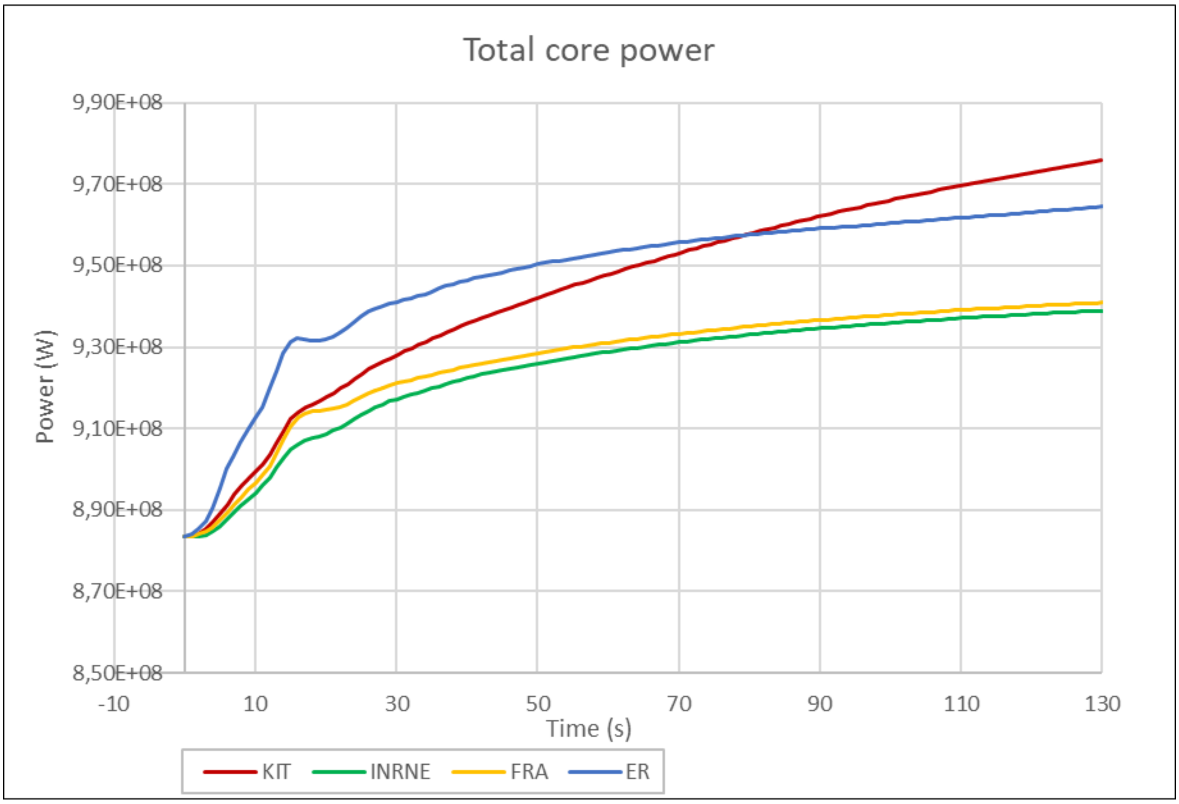

Figure 3.2.1.1 Core power during transient

3.2.2 Reactivity

The comparison of participant’s reactivity is presented on Figure 3.2.2.1. There is no available plant data for the reactivity during the transient.

The comparison of the reactivity results demonstrates that all participants predicted almost the same trend. There are observed differences mainly in the max. reactivity predicted by the participants, which was observed in all calculations at approximately 15 s. The differences of calculated reactivity could be explained with Doppler and moderator temperature feedback input. Also, the reactivity reflects from total core power change during the transient and core average fuel temperature and core average moderator temperature time histories. The observed differences arise also from differences in fuel rod component modelling and gas-gap conductance model.

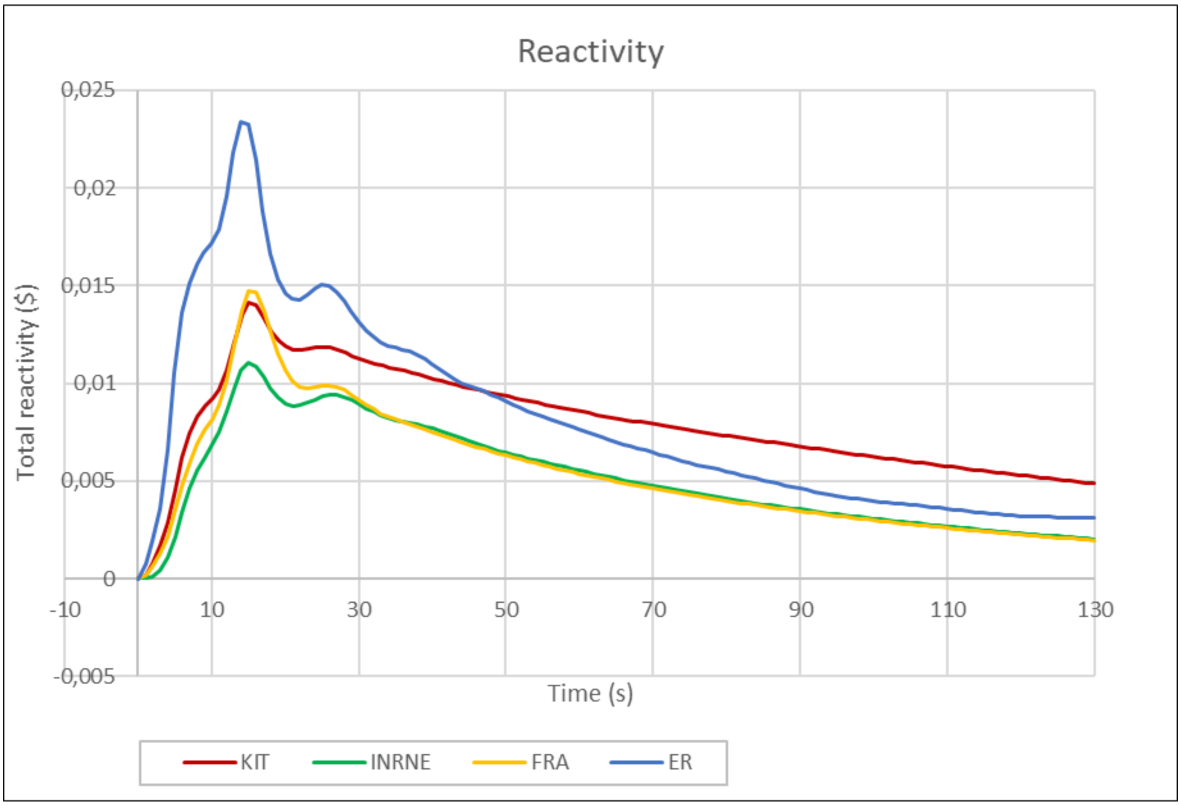

Figure 3.2.2.1 Reactivity during transient

3.2.3 Cold leg temperatures

The comparison of temperature evolution of cold leg temperatures and plant data is presented from Figure 3.2.3.1 to 3.2.3.4. The comparison of cold leg temperatures shows that FRA predicted significantly close results vs experiment, while the other three showed poor agreement between the calculated values and the plant data for some of the legs. The largest differences of predicted temperatures by the participants in loop #3 are observed at the beginning of the test at the interval between 5 to 15 seconds, when it is observed a sharp drop of the coolant temperature (approximately 4 to 5 K), after switch on MCP #3, which is not observed in the plant data. This effect could be explained that with switching on MCP #3, when the other pumps are on, the direction of the flow in this loop reverses. The temperature changes due to changes in the flow rate through the loops and due to changes in the distribution of the energy release field in the core and redistribution of fuel temperature and coolant parameters (density + temperature)

After the pump starts, the flow which once passed through the SG is forced back and goes again through the SG, leading to further temperature decrease. All codes predict the exact behaviour. The missing of drop effect in cold temperature in plant data record could be explained with measurements work, which did not register this effect due to time delay in the system.

Generally, the comparison of predicted coolant temperature at loops #1, #2 and #4 is within the uncertainty range of the experimental data. The comparison of calculated results vs plant data for loop #3, shows that when the main coolant pump is switched on, the results are slightly outside from the uncertainty range at the beginning of the transient, but at the end they became within the error band at the final conditions.

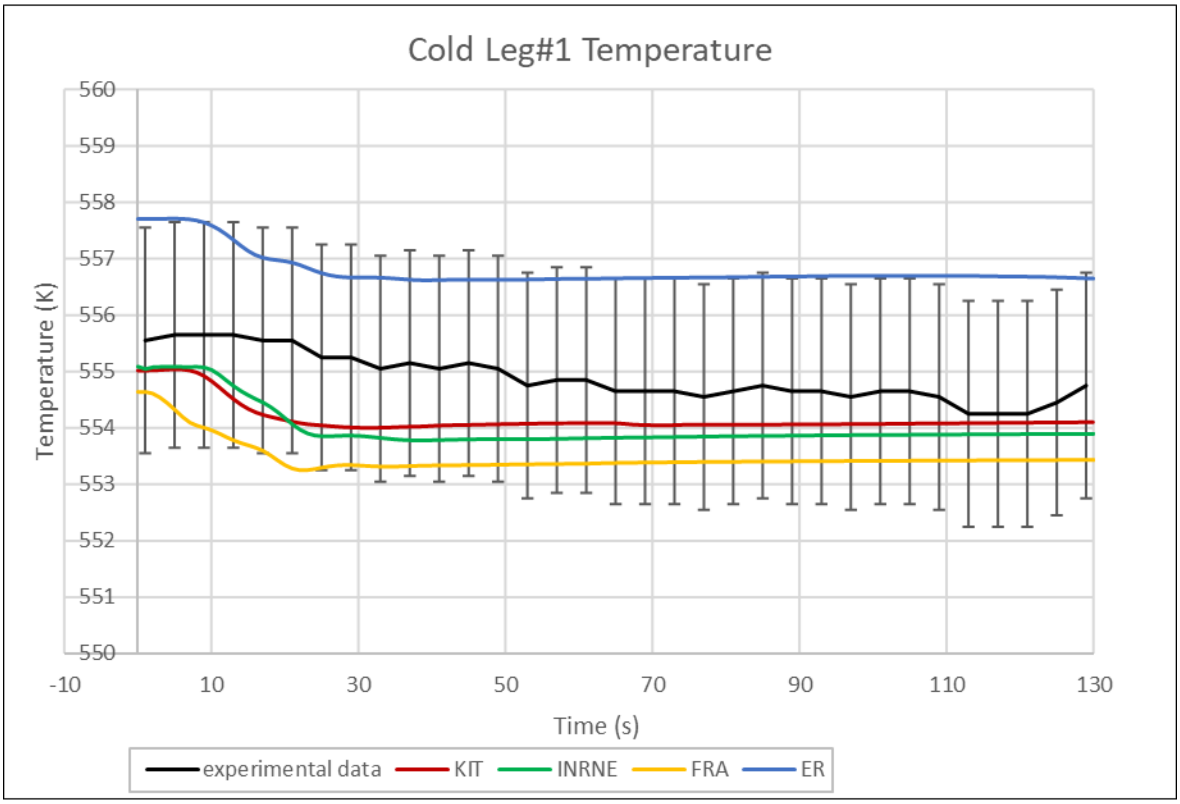

Figure 3.2.3.1 Cold leg #1 temperature

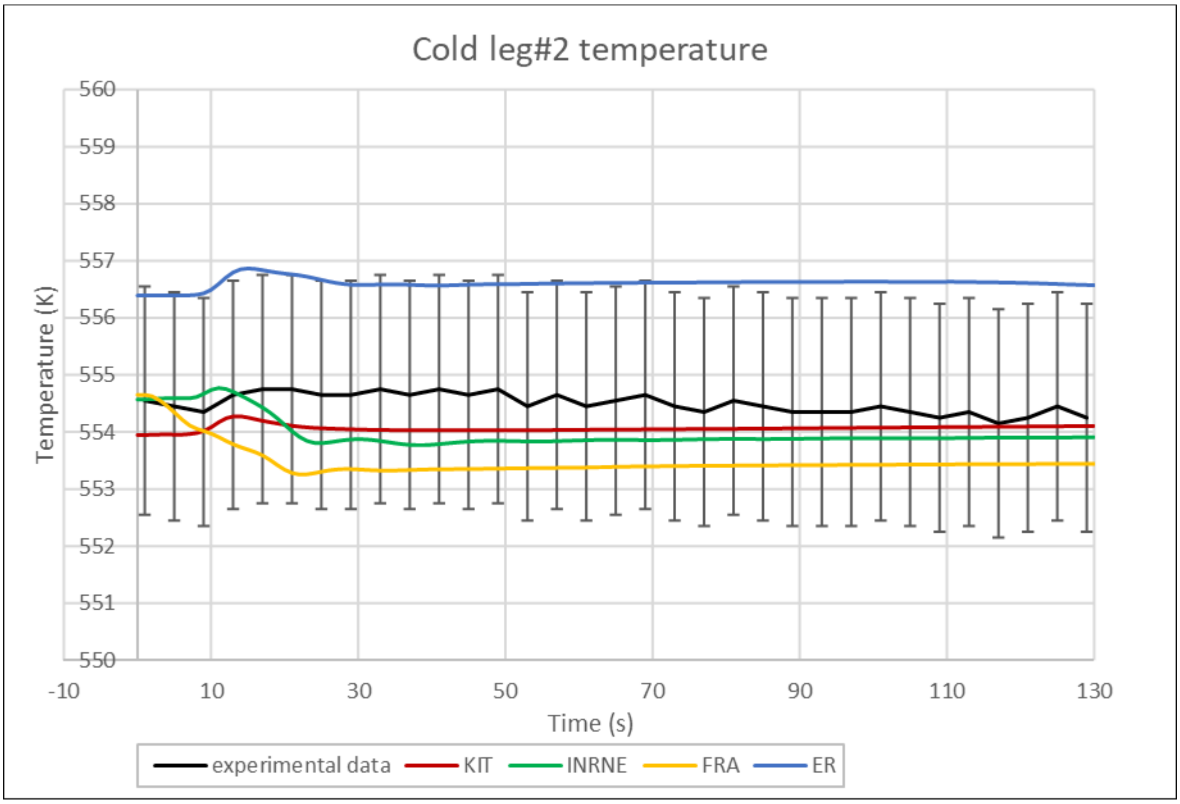

Figure 3.2.3.2 Cold leg #2 temperature

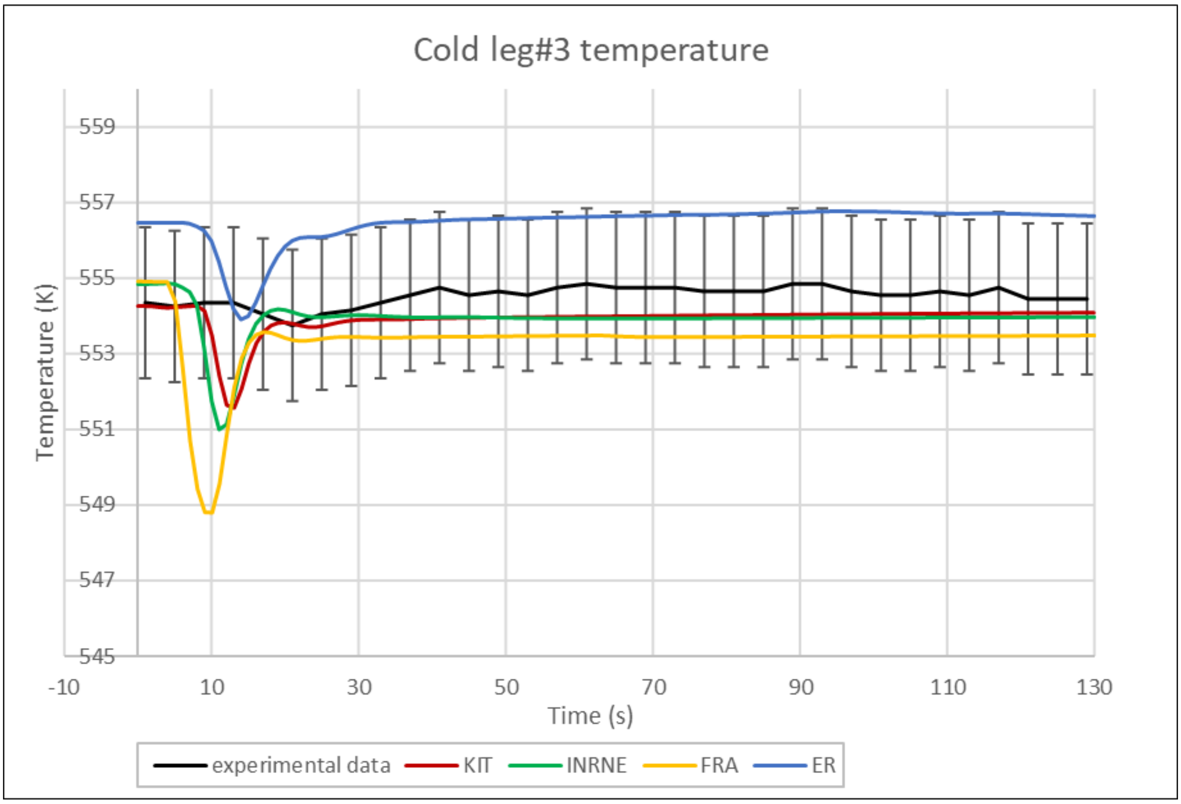

Figure 3.2.3.3 Cold leg #3 temperature

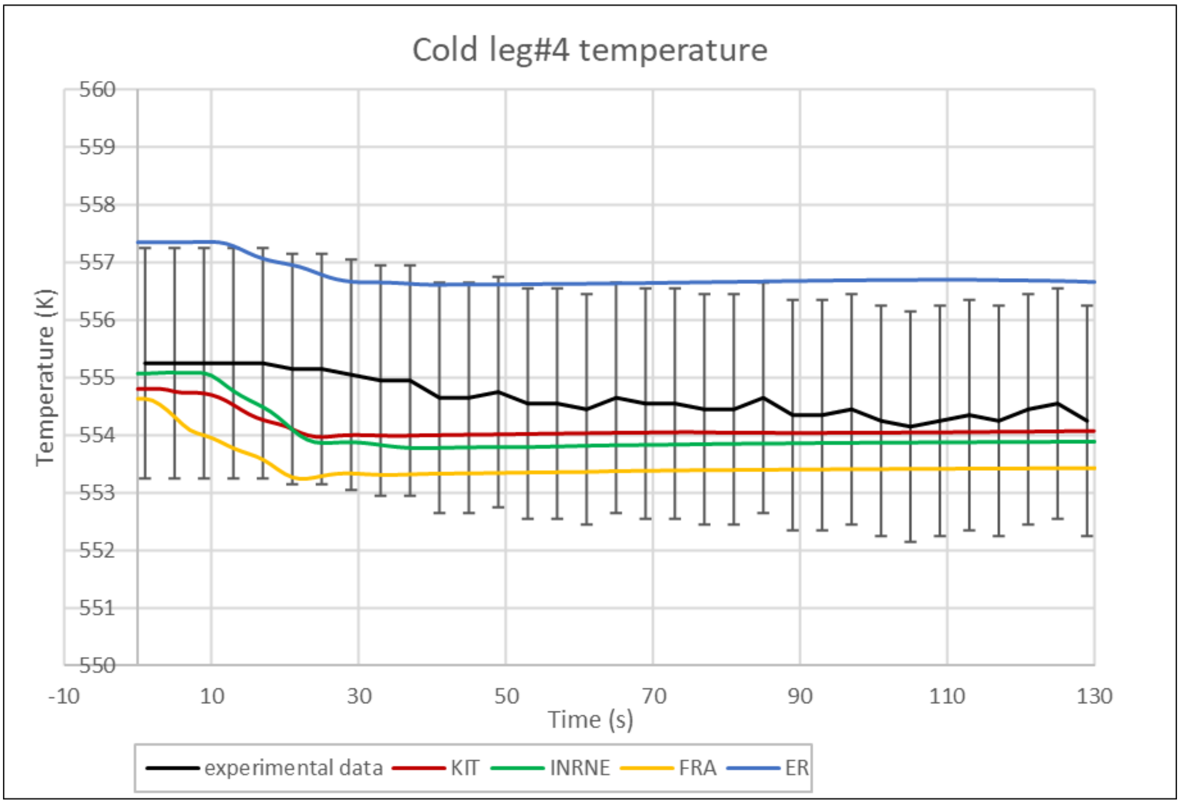

Figure 3.2.3.4 Cold leg #4 temperature

3.2.4 Hot leg temperatures

Figures 3.2.4.1 – 3.2.4.4 show the temperature evolution for the hot legs for the four cooling loops. As with the cold leg coolant temperatures, the results of hot legs temperatures in loops #1, #2 and #4 are within the uncertainty range of the experimental results for most of organizations: KIT and FRA and INRNE, while the ER are out of the uncertainty range for loops #1, #2, #3 and #4. The comparison of hot legs temperatures initially showed poor agreement between the codes' predictions and measured data. The observed largest difference is in loop #3 at the interval from 0 to 20 s: the most codes predicted a peak in the hot temperature in loop#3, while this peak is not observed in the plant data. This observed hot leg temperature peak could be explained with the differences between the loops: before the transient starts, the temperature of the coolant in hot leg #2 is lower than the temperatures in hot legs #1 and #4 due to the coolant with lower temperature coming from loop #3. The affected loop is loop#2, because it is closest to loop #3. Due to reverse back flow from loop #3, the flow from loop #1 and #4 with higher coolant temperature enter loop #2. The predicted hot temperatures in loop #3, where the MCP is switched on at the beginning are slightly off vs the experiment, while in the end of transient are within the range of the uncertainty. It could be explained with later system response of the measured temperature.

As it is seen later into the transient, the most participants' results of coolant hot leg temperatures in loops #1, #2 and #4 stabilize. Once again, the explanation for this significant difference is the time delay of the measurement system. The CATHARE3 model doesn't take into account this dissymmetric effect so that calculated temperatures of loops #1, #2 and #4 are the same. Also, the participants did not model the time delay of the thermo-couples measurements.

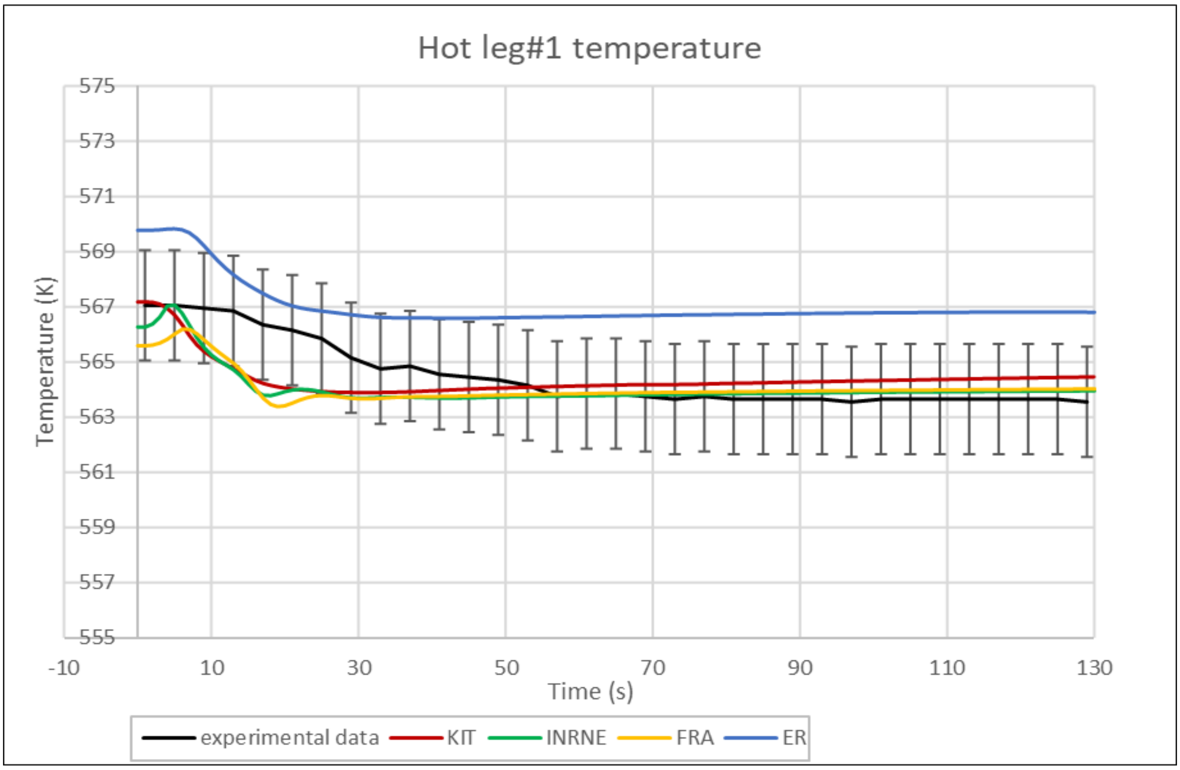

Figure 3.2.4.1 Hot leg #1 temperature

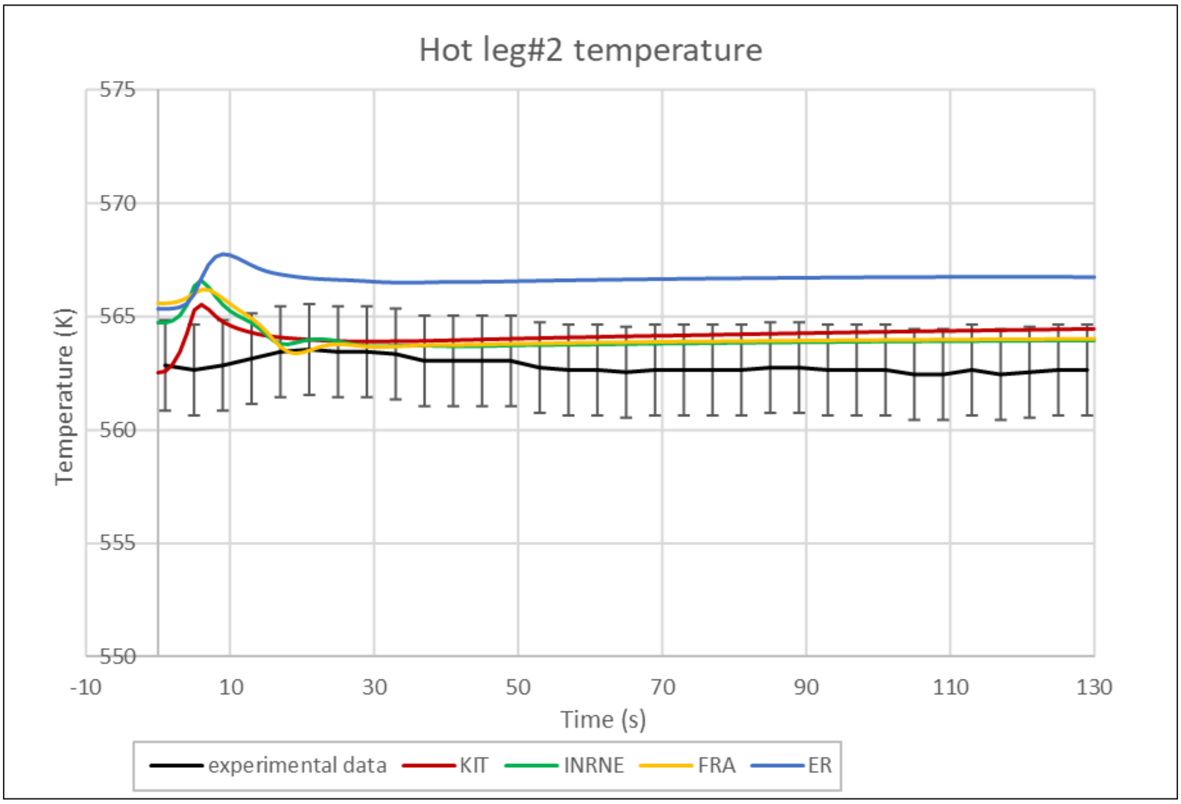

Figure 3.2.4.2 Hot leg #2 temperature

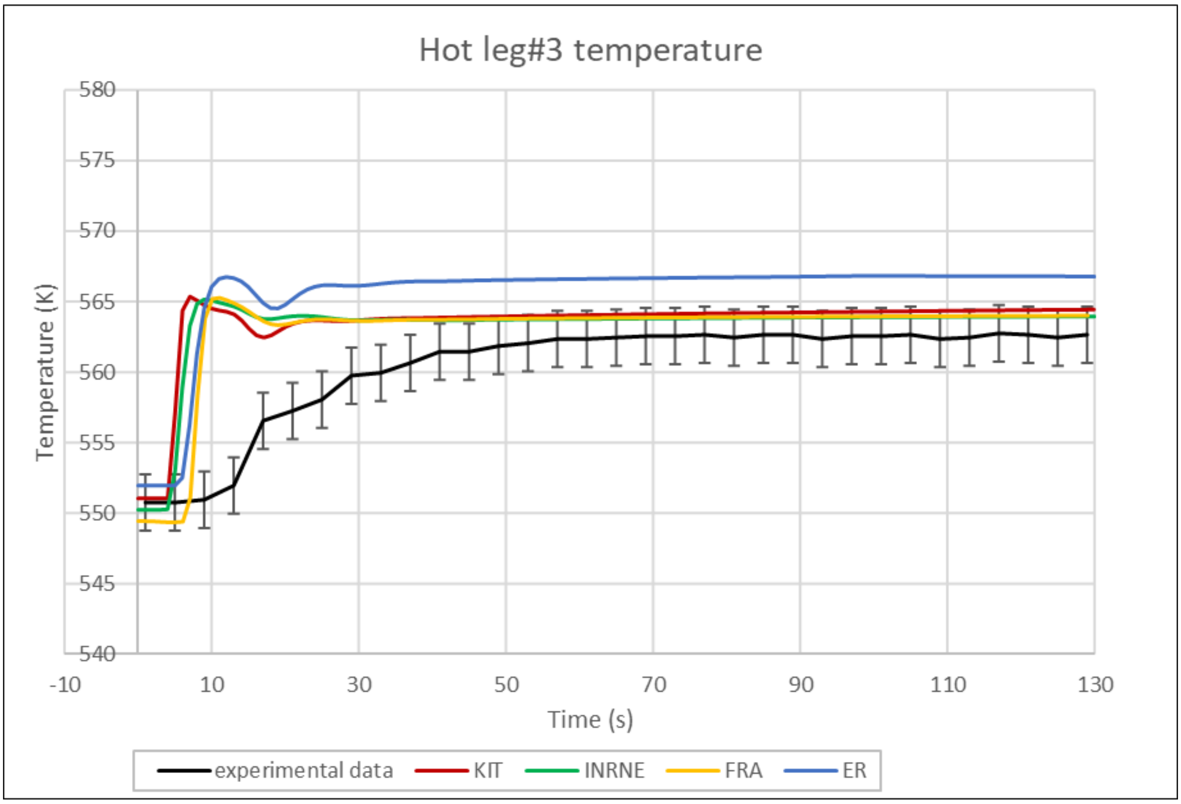

Figure 3.2.4.3 Hot leg #3 temperature

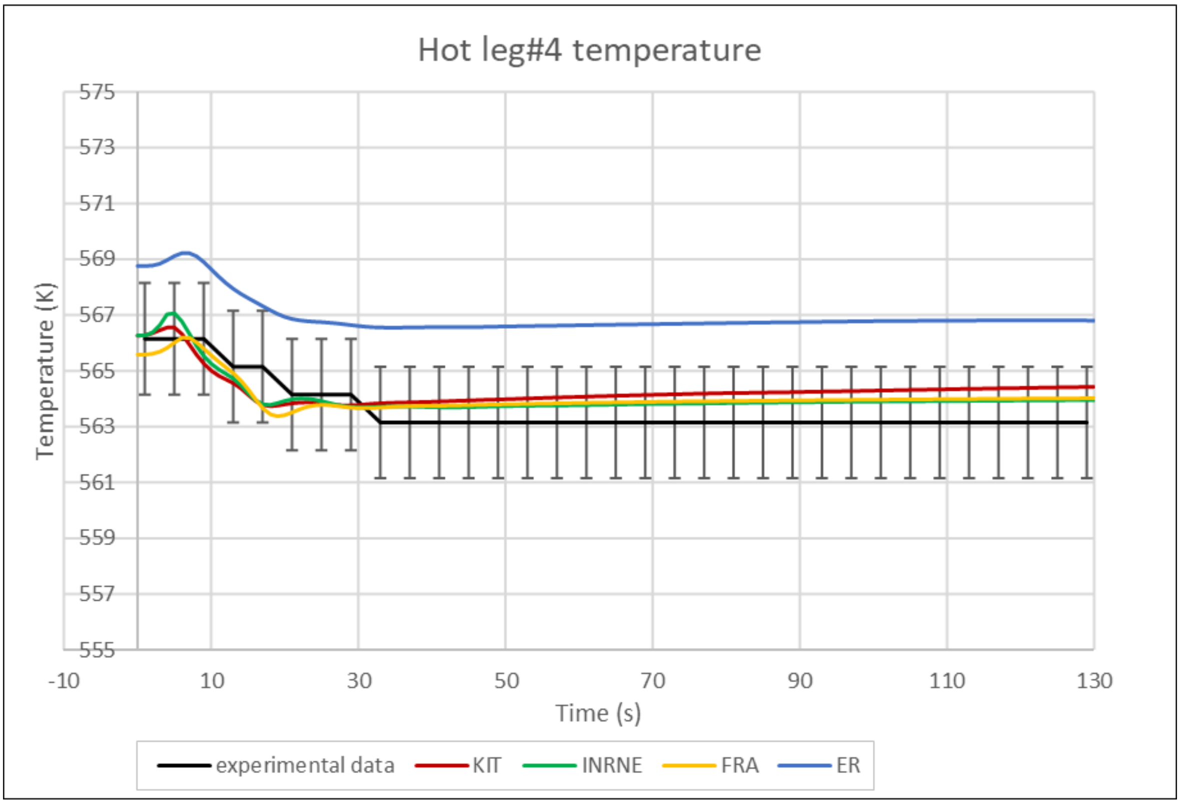

Figure 3.2.4.4 Hot leg #4 temperature

3.2.5 PZR water level

Figure 3.2.5.1 shows the comparison of the participants' results for the pressurizer liquid level during the transient. After starting the idle MCP, the coolant temperature of the first circuit begins to decrease due to an increase in the heat sink from the first circuit to the second. This leads to decrease in the pressure compensator level, which is stabilized by pressurizer heaters.

The comparison of this parameter shows one of the largest deviations in the trend vs plant data, among all of the compared parameters. At the beginning of the transient the comparison of calculated results by the participants is in significantly good agreements with measured data.

The trend of most predicted water level results by the participants are in the range of the uncertainty, except ER where at the end of test the pressurizer water level increases slowly and go a little bit out of the range of the uncertainty. Most of the predicted pressurizer levels by the participants are stable earlier in the transient compared to the measured value, and at the end of the transient the pressurizer level became higher than the experiment, but the results are still in the range of the uncertainty. The observed differences could be explained with using of different models and codes by the participant.

Some of the participants have regulation systems in their models, like: make-up and let-down systems, while the others do not, as in FRA model, where the Make-up/Let-down system is not modelled. Also, the secondary side pressure in FRA model is not regulated and SG levels are maintained constant with a very simple regulation (P action only). It has to be mention that the participants have modelled make-up and let-down systems using different controller's logic, which reflect to their work.

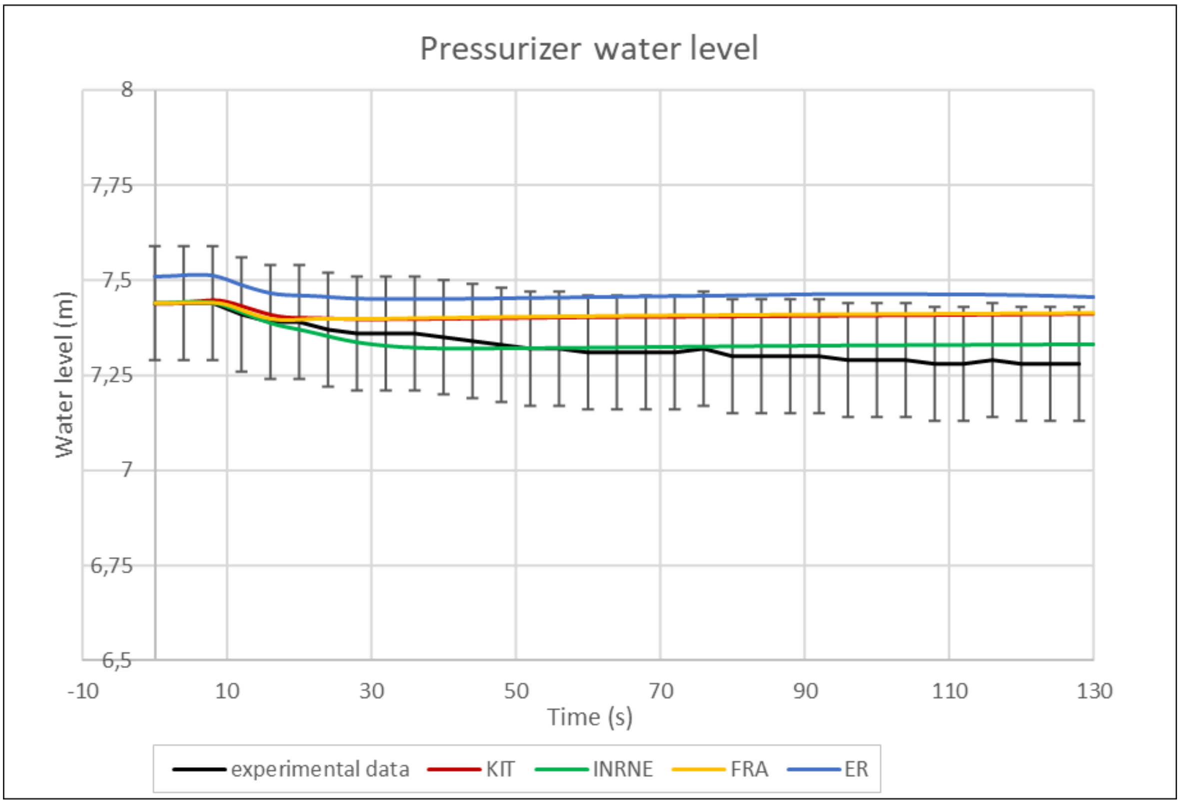

Figure 3.2.5.1 Pressurizer water level during transient.

3.2.6 Loop Flow rates

The comparison of flow rate in each loop is presented on Figures 3.2.6.1 through 3.2.6.4. There is no available plant data during the test.

The comparison of calculated loops flow rate by the participants demonstrates almost the same trend. It can be seen very good agreements in flow rate of loop#3 predicted by different computer codes. It is observed differences in loop #1, #2 and #4 in FRA results compared to the other participants. The observed differences in the participant's results could be explained with using of different pump characteristics.

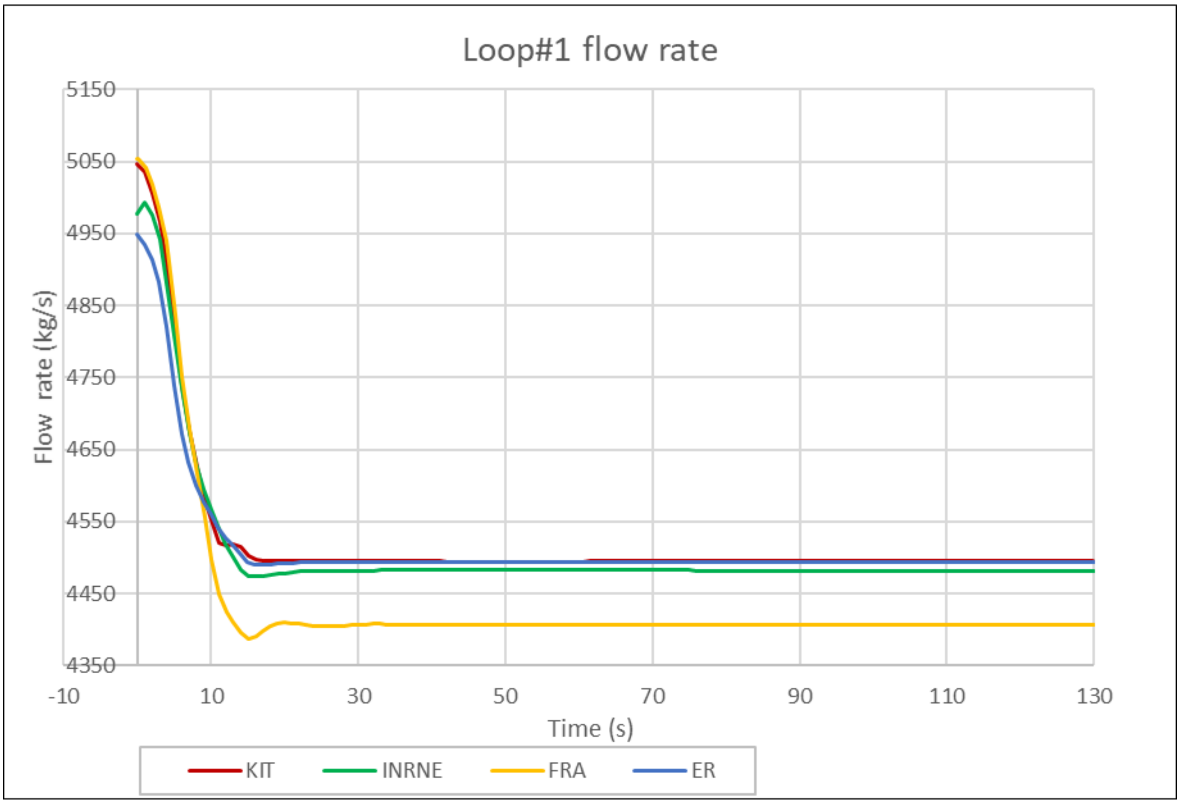

Figure 3.2.6.1 Loop #1 flow rate during the transient

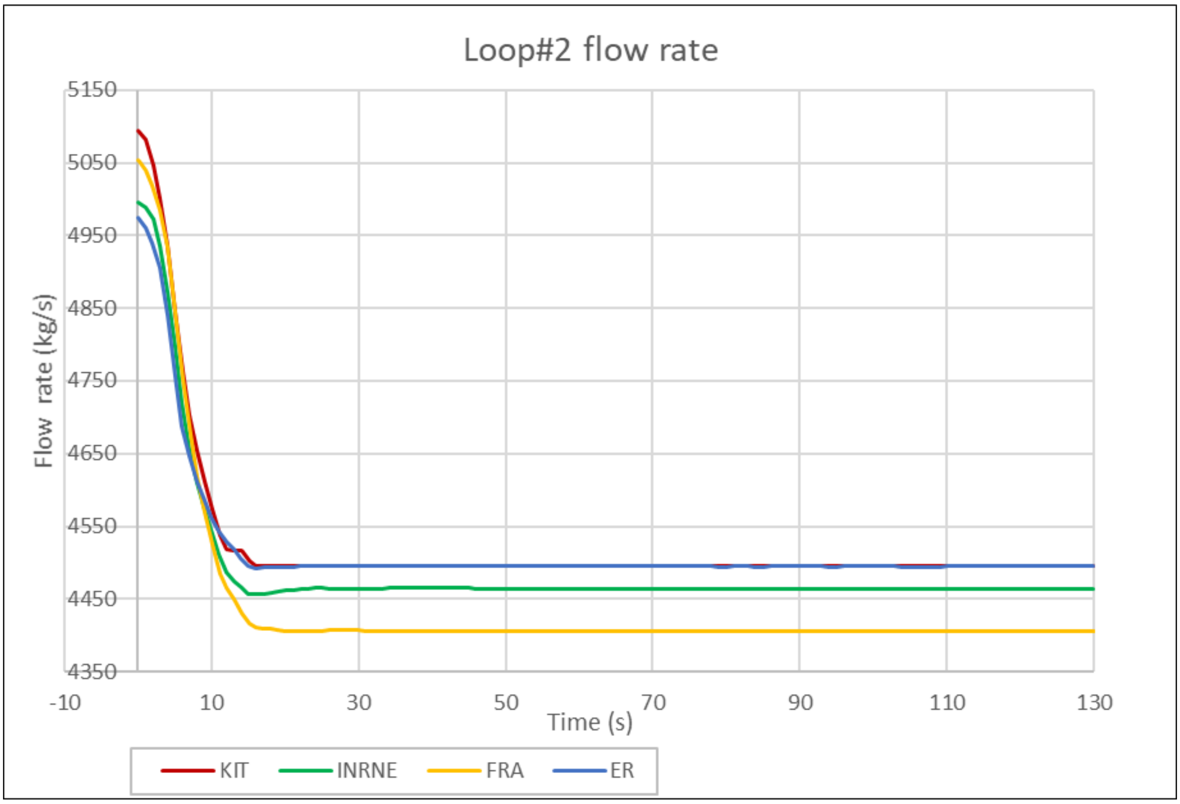

Figure 3.2.6.2 Loop #2 flow rate during the transient

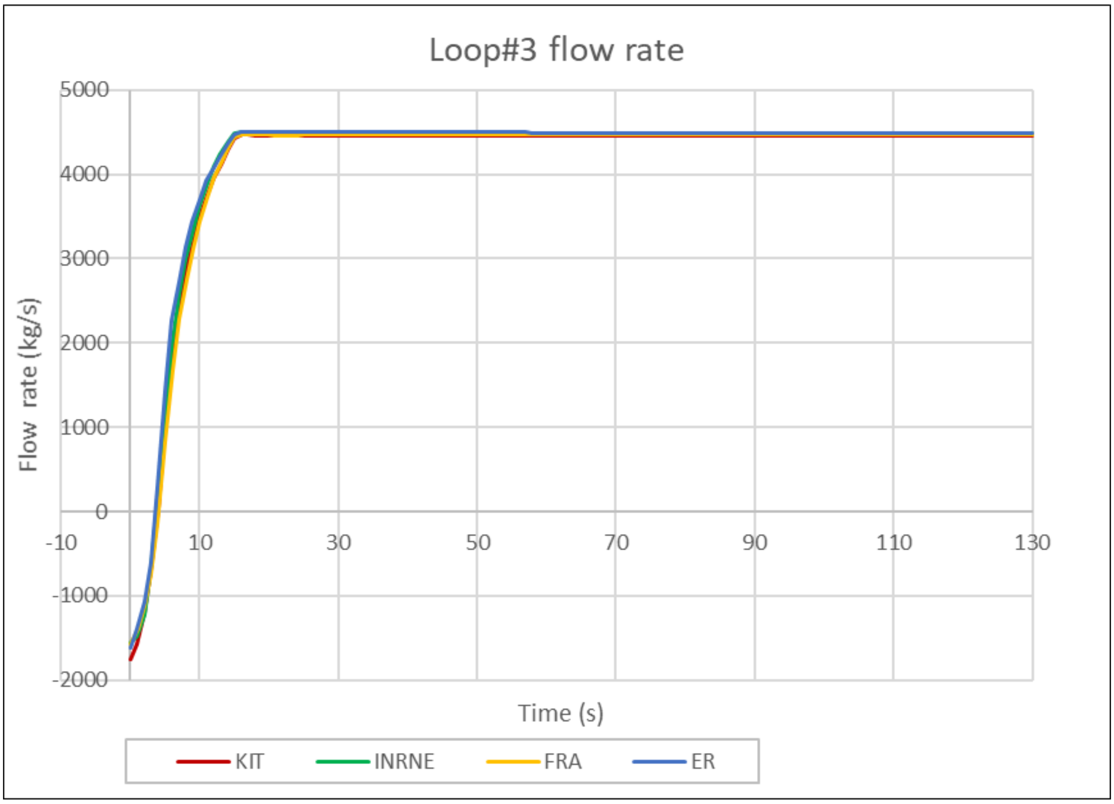

Figure 3.2.6.3 Loop #3 flow rate during the transient

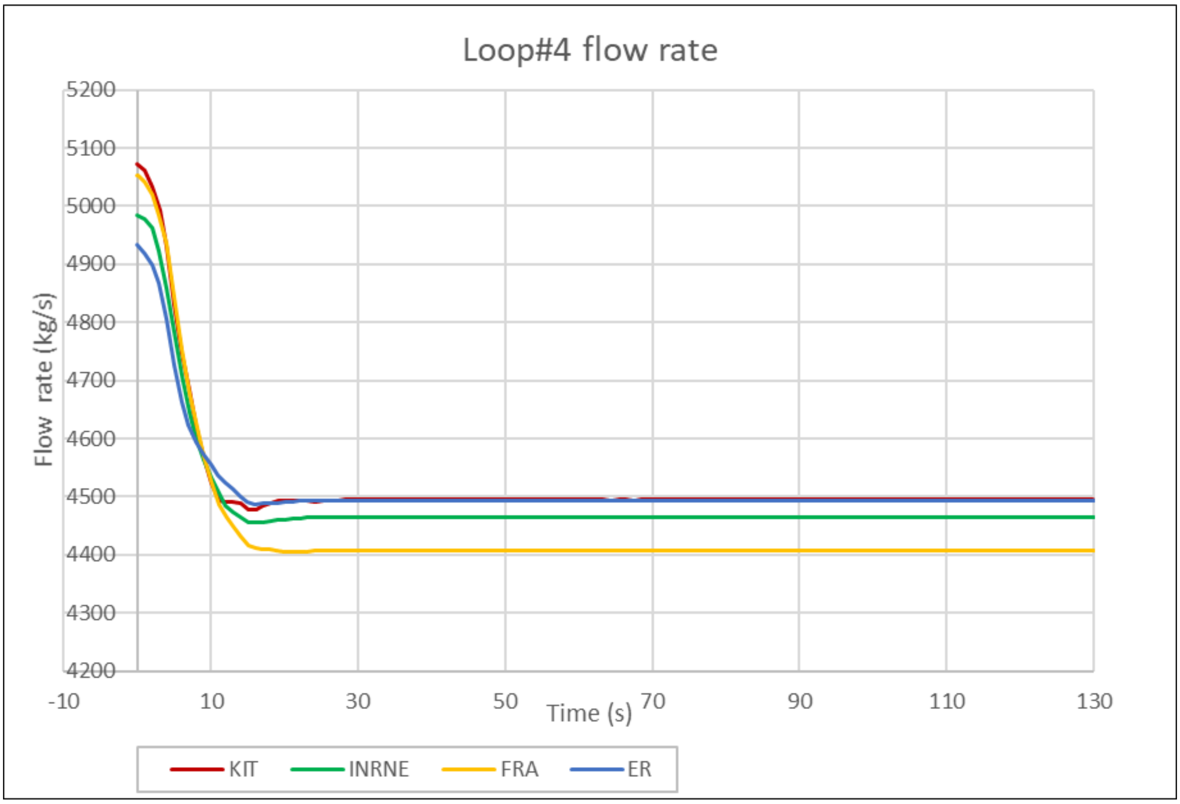

Figure 3.2.6.4 Loop #4 flow rate during the transient

3.2.7 Pressure above the core

The comparison of pressure above the core calculated by the participants vs plant data is presented on Figure 3.2.7.1.

The comparison shows that both of participants predicted primary pressure in significantly good agreement with plant data, while in the other participants' results some deviations are observed during the test. As it has been explained, the primary pressure strongly follows the PRZ water level behaviour. The observed discrepancy could be explained with different models used by the different participants. As it is mentioned, some of the participants have modelled regulation system like: make-up and let-down, while other have not. Also, using different models for PRz heaters play very important role in regulation of primary pressure.

Generally, the comparison shows that all participants predicted the primary pressure in the range of the uncertainty.

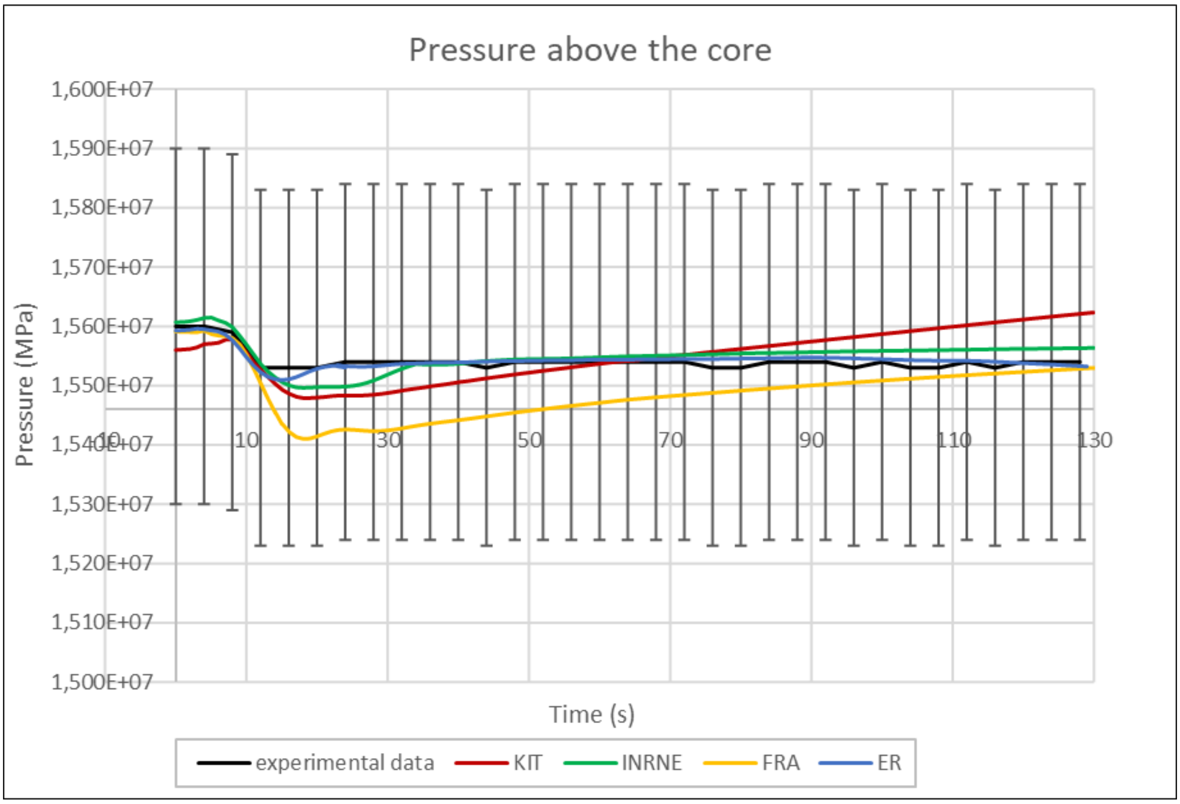

Figure 3.2.7.1 Pressure above the core

3.2.8 Core pressure drop

The comparison of core pressure drop vs plant data is presented on Figure 3.2.8.1. The comparison of core pressure drop predicted by the participants vs plant data show that most of the participants predicted almost the same pressure drop as the plant data, except ER result. The initial value of the reactor pressure drop predicted by the participants is higher than the experiment in first

20 sec of the transient, only the pressure drop predicted by ER starts with significantly big deviation of 2,5 MPa higher.

It can be concluded that most of the participants predicted their results in the range of the uncertainty, only one partner is outside the uncertainty range.

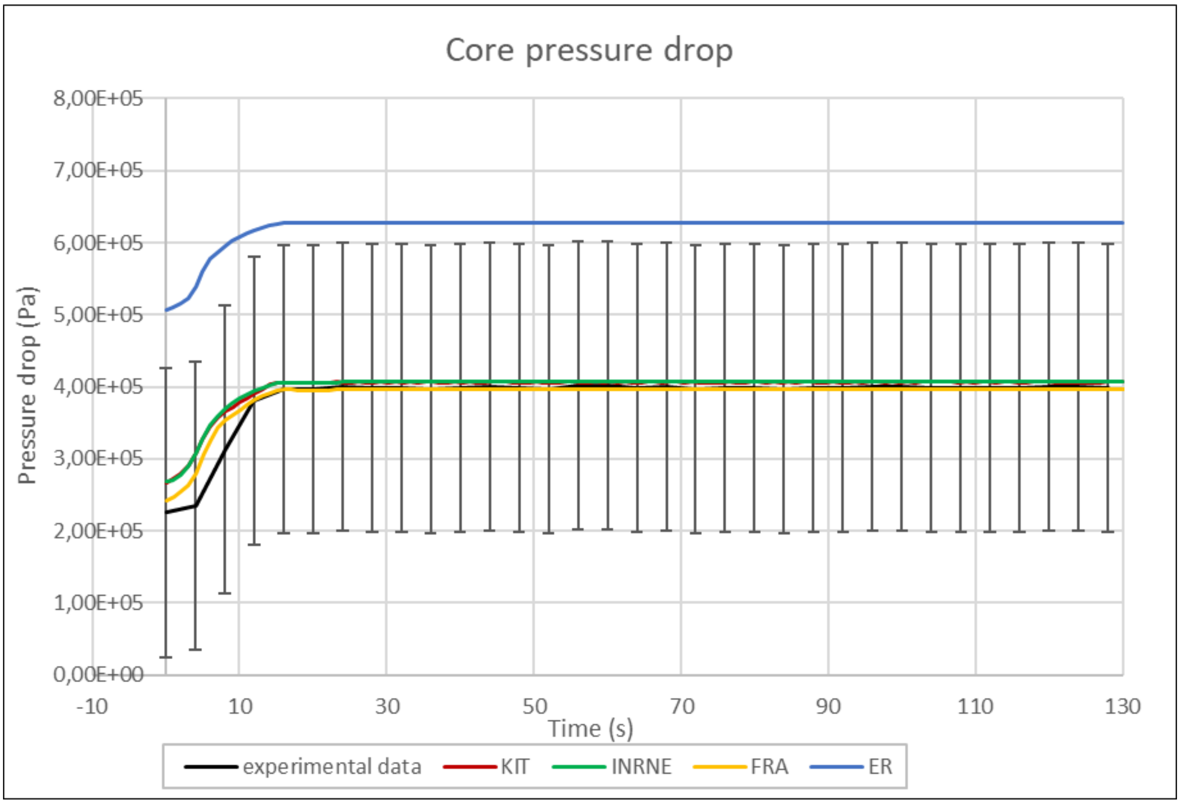

Figure 3.2.8.1 Core pressure drop

3.2.9 Main coolant pump pressure drop for the four loops

Figures 3.2.9.1 - 3.2.9.4 show the pressure drop for the main coolant pumps (MCPs) for the four cooling loops vs plant data.

The comparison of MCP pressure drop predicted by the participants vs test data demonstrate significant differences in the results. Both of the participants: FRA and INRNE predicted significantly good results for MCP pressure drop in all loops compared to the plant data, the predicted results are in the range of the uncertainty. In the predicted results by KIT, it is observed small overestimation compared to the test data.

The predicted by KIT results are overestimated but close to the range of the uncertainty for loop #1, #2, #3, and #4. The results predicted by ER are very different, in loop #1, the pressure drop is predicted with very big deviation, while for loop #2, #3 and #4 are overestimated, but very close to the uncertainty range.

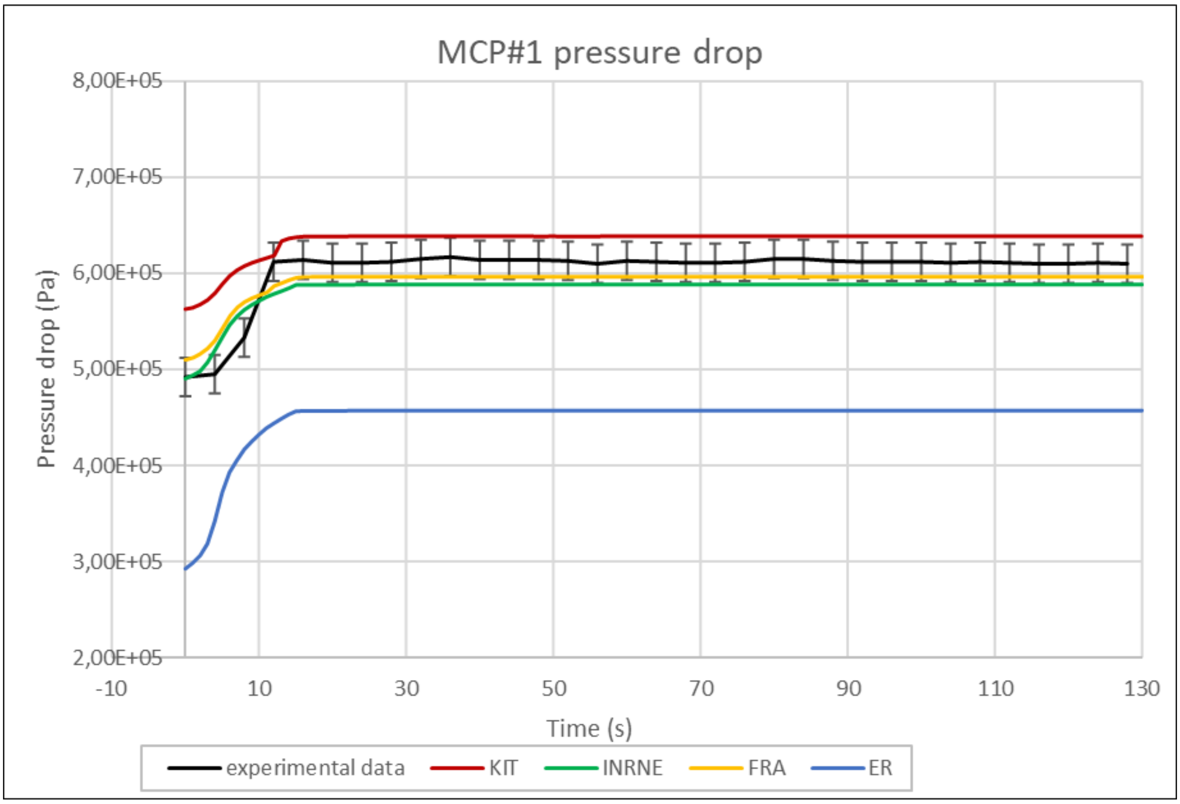

Figure 3.2.9.1 MCP #1 pressure drop

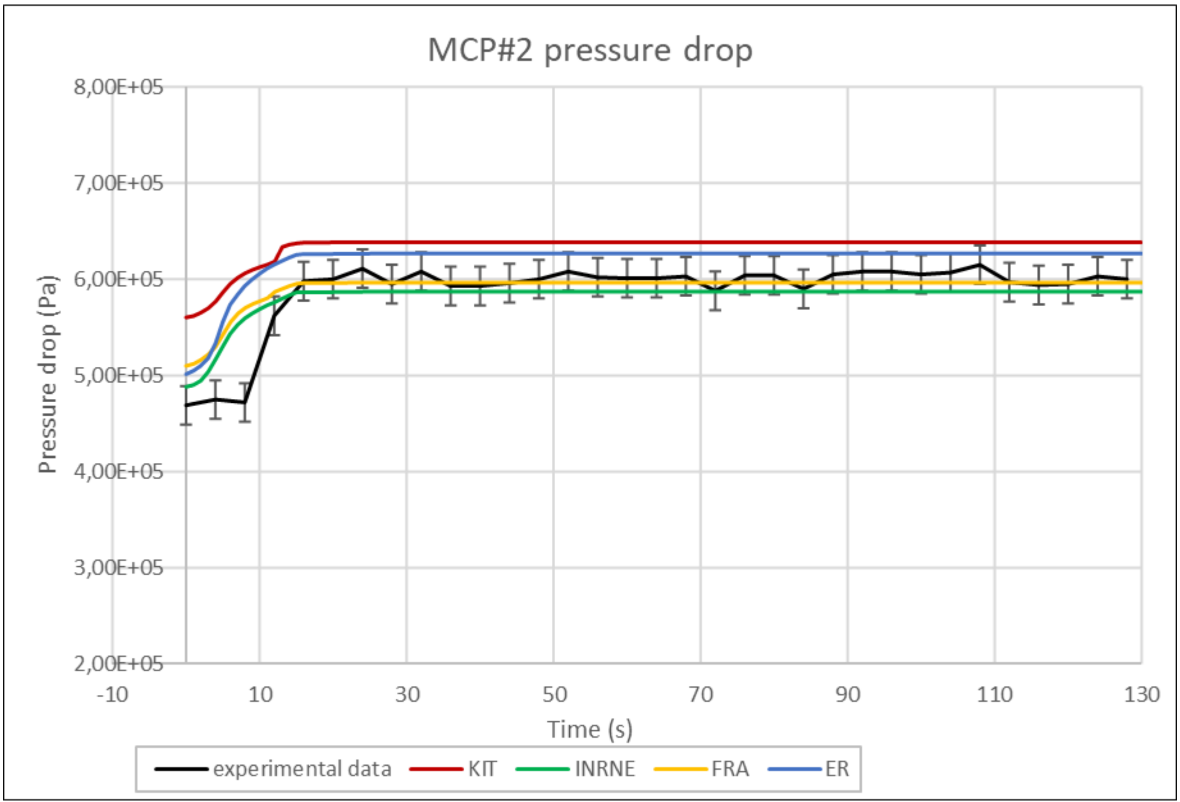

Figure 3.2.9.2 MCP #2 pressure drop

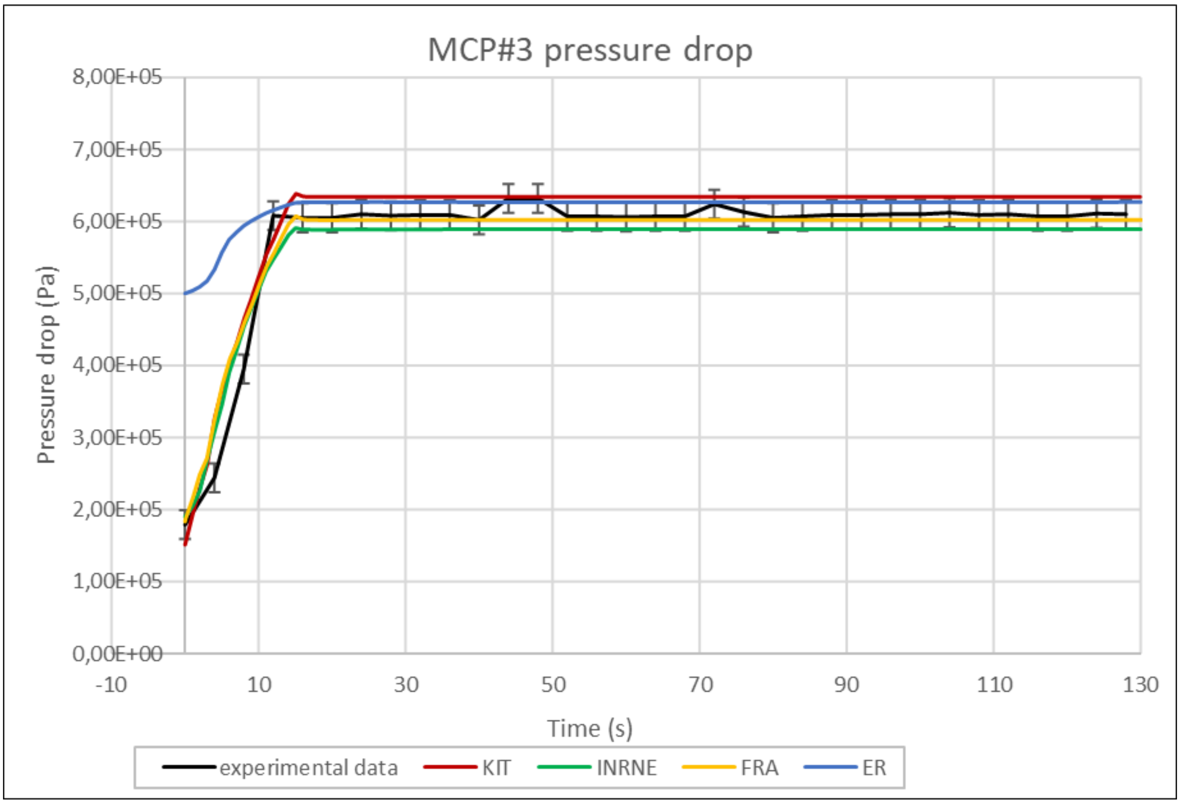

Figure 3.2.9.3 MCP #3 pressure drop

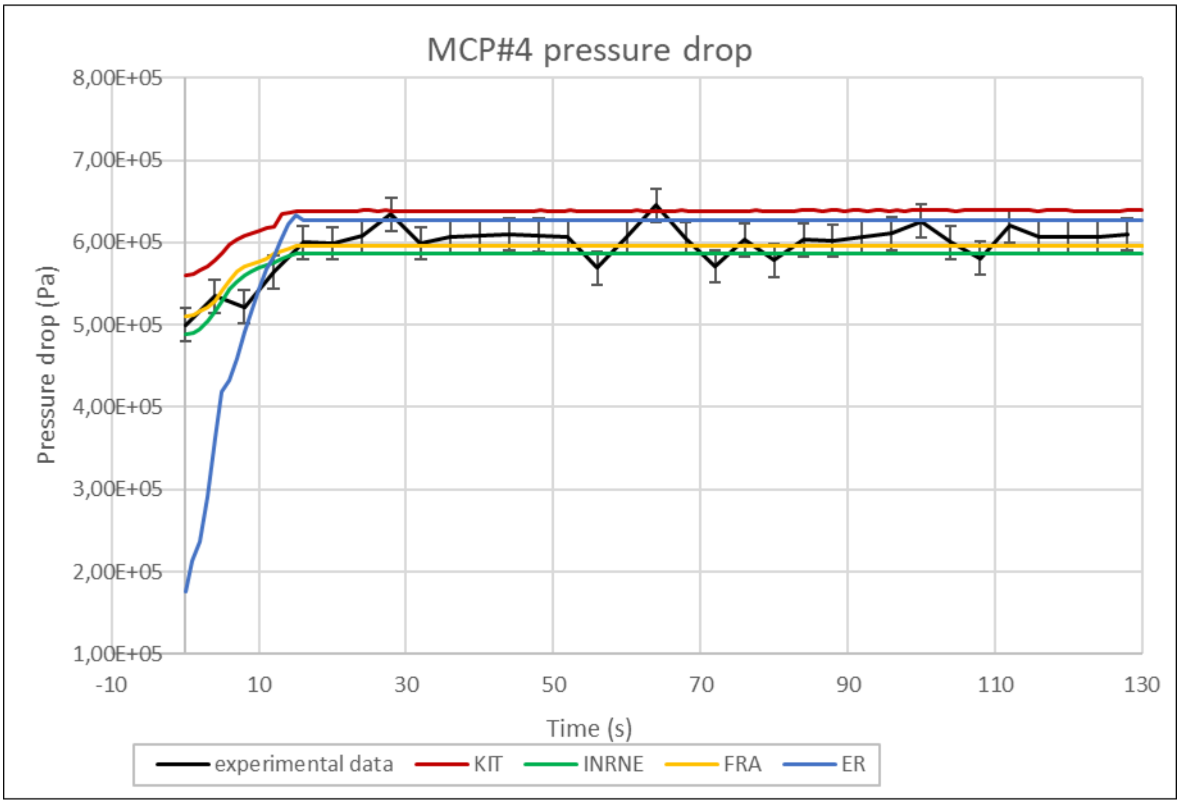

Figure 3.2.9.4 MCP #4 pressure drop

3.2.10 Coolant heat-up temperature (Delta Temperature change) in the four loops

The coolant heat-up temperatures in all four loops are presented on Figures 3.2.10.1 to Figure 3.2.10.4. The comparison of heat-up in loops vs plant data show the temperature difference between hot and cold temperatures in different loops. The comparison of temperature differences in loop #1 predicted by the participants, are in close agreement in comparison with test data. It is observed a little bit different behavior of calculated results after 30 sec from the beginning of transient.

The plant data temperature difference in loop# 1 decrease slightly, while all participants temperature difference in loop#1 slightly increases. As a whole all the predicted results in loop #1 are in the range of the uncertainty. The comparison of temperatures in loop#2 and #3 shows that all the participant's results are overestimated vs plant data, the temperature difference behavior in loop #3 closely follow the observed pick in cold and hot legs temperature in loop#3. As a whole all the predicted results in loop #2 and #3 are out or close to the upper side in the range of the uncertainty. In the comparison of all predicted temperature difference in loop #4, it is observed big deviation with 2 K higher temperature in comparison with plant data. The observed discrepancy between the calculations vs plant data could be explained with the time delay of the thermo-couples measurements.

All the predicted results of temperature differences in loop #4 are close to the upper side of the range of the uncertainty.

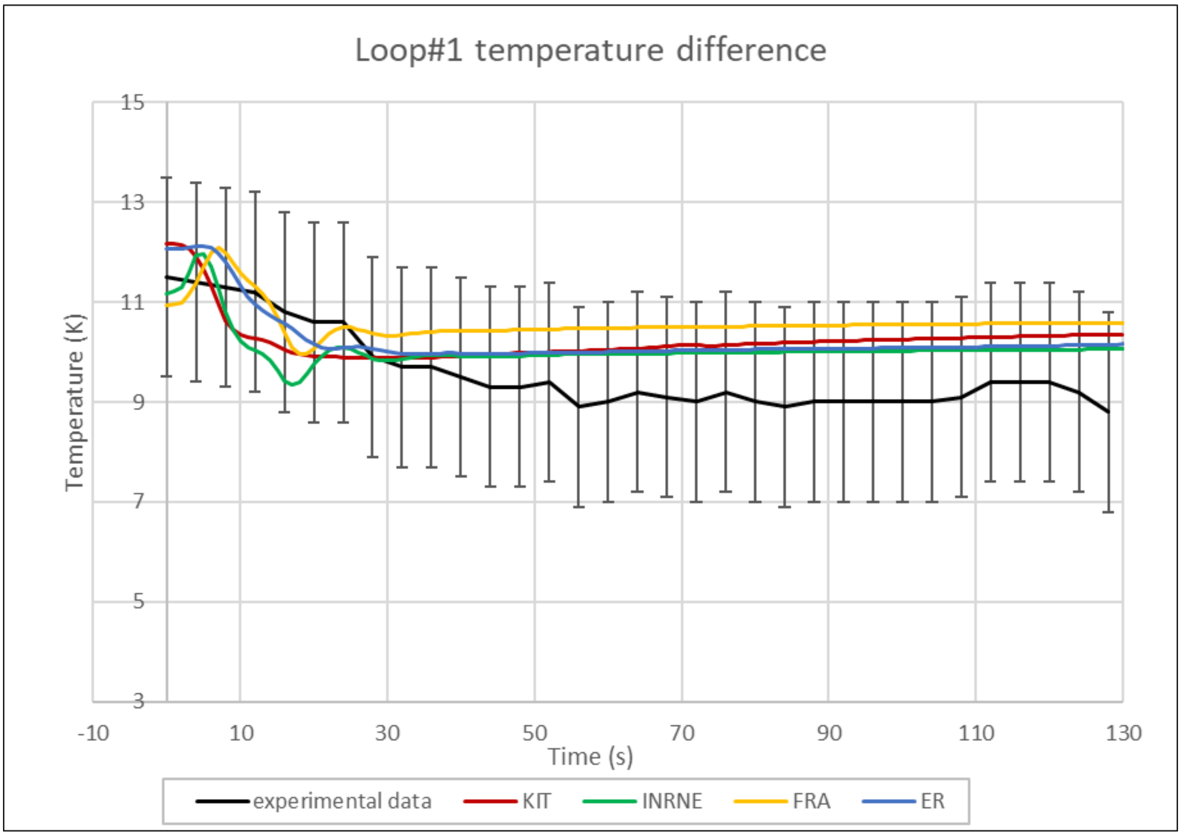

Figure 3.2.10.1 Heat-up in loop #1 change during the transient.

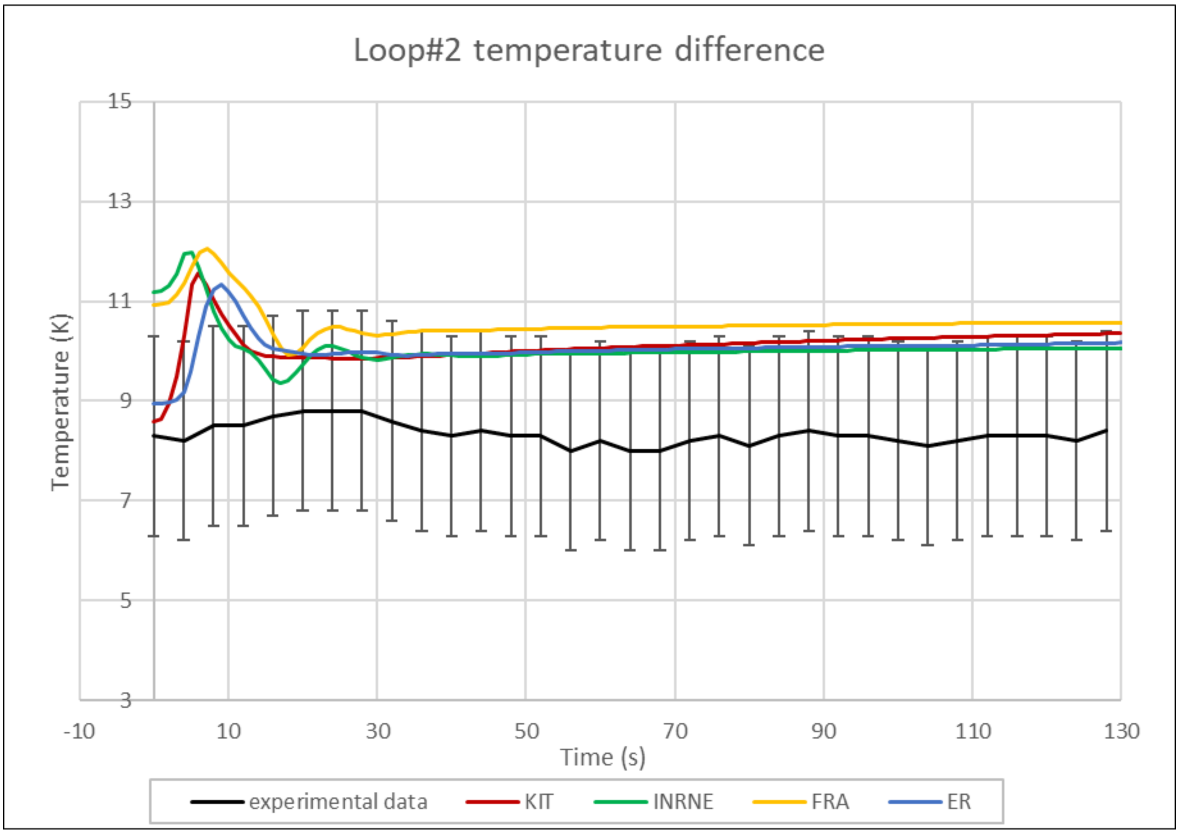

Figure 3.2.10.2 Heat-up in loop #2 change during the transient

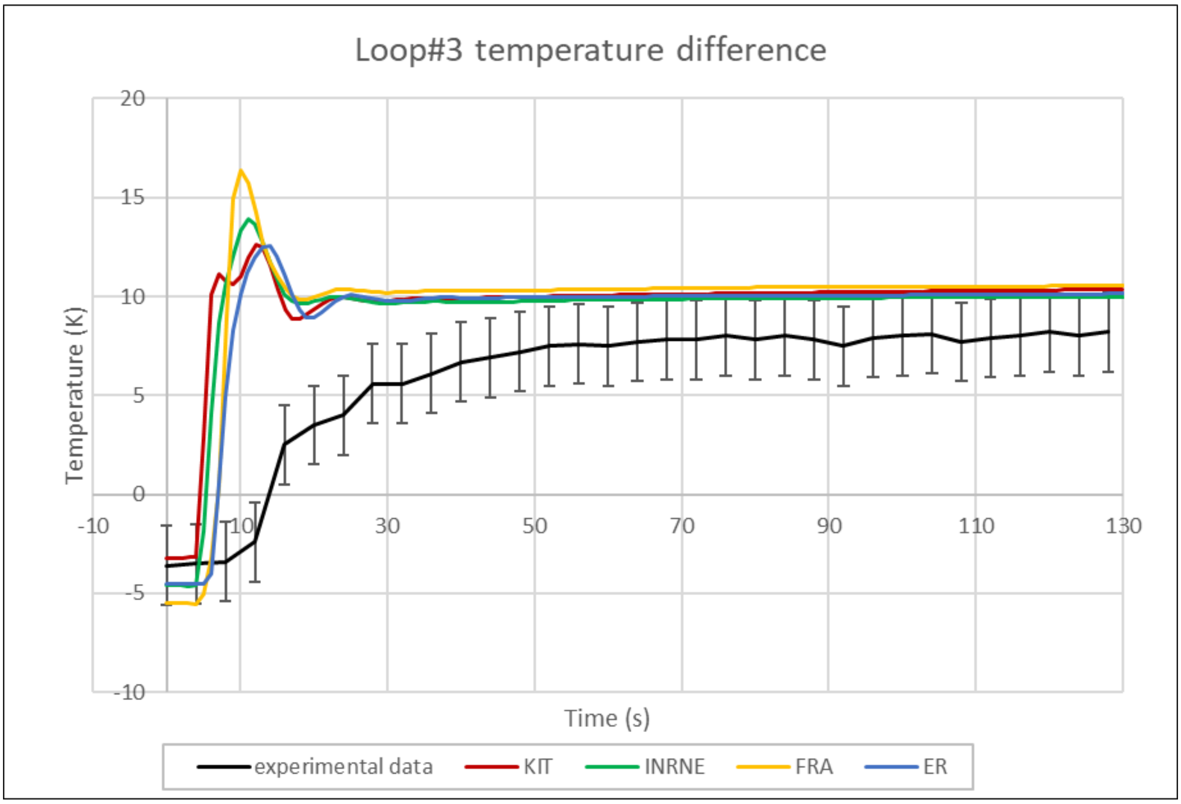

Figure 3.2.10.3 Heat-up in loop #3 change during the transient

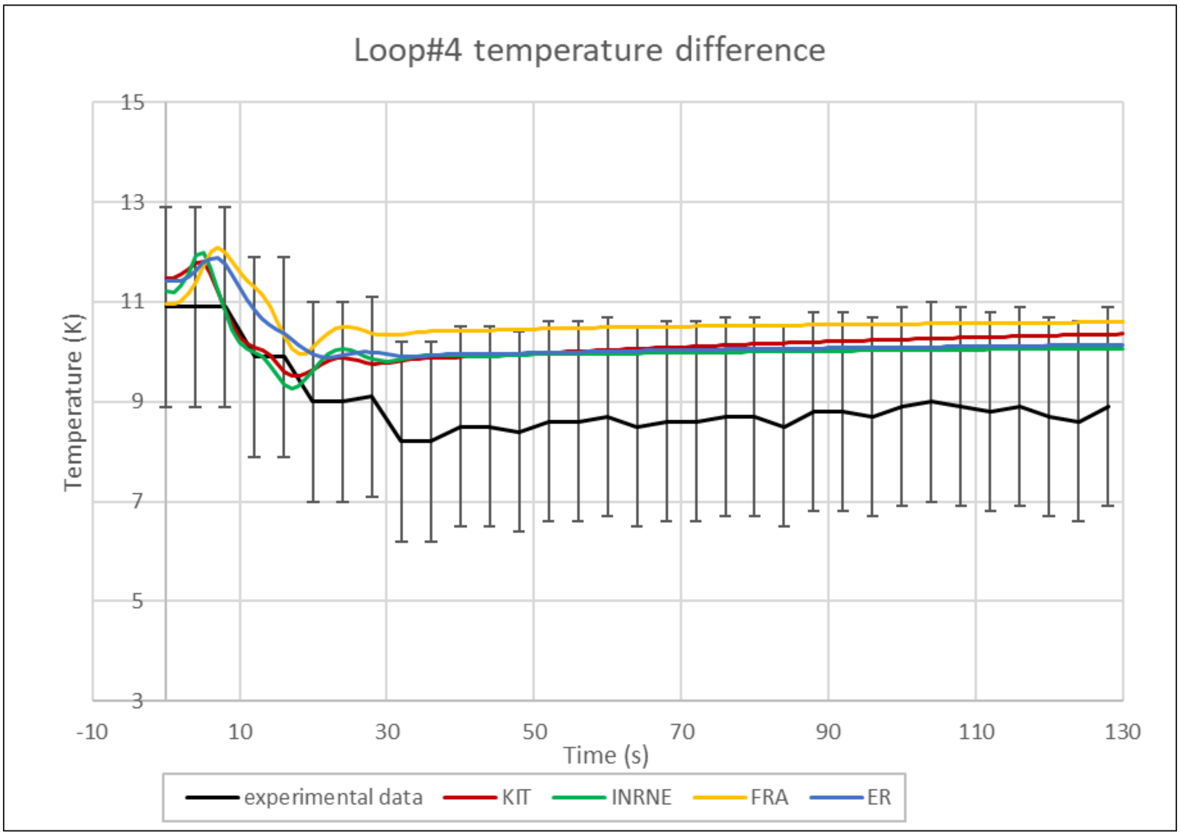

Figure 3.2.10.4 Heat-up in loop #4 change during the transient

3.2.11 Core average coolant temperature

The comparison of core average coolant temperature is given on Figure 3.2.11.1. There is no available core average coolant temperature plant data during the test.

As a whole, the comparison of participants average coolant temperature show that all of the participants predicted almost the same coolant temperature trend during the whole transient. As it is seen, when the coolant temperature decreases, the reactivity starts to increase.

There are differences in ER results, where the average coolant temperature is higher with 20 K than the other participants, which could be explained with higher level of core power in their results.

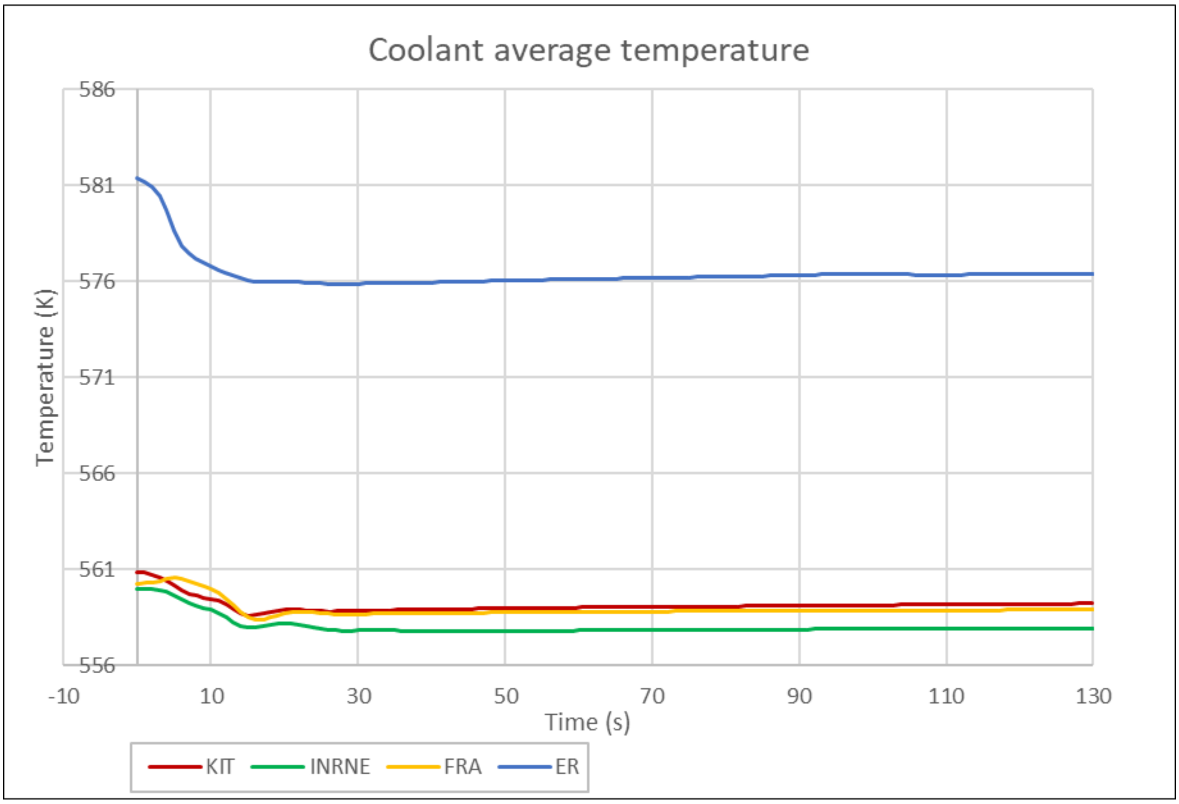

Figure 3.2.11.1 Core average coolant temperature during the transient

3.2.12 Core average fuel temperature

The comparison of fuel average temperature is presented on Figure 3.2.12.1. There is no available plant data.

The comparison of participant's average fuel temperature demonstrates different average fuel temperature predicted by most of participants. There is a big difference in ER result which is significantly higher with app 250K than the other participants, also KIT predicted lower temperatures compared to INRNE and FRA. The differences could be explained with the different fuel models applied in the codes.

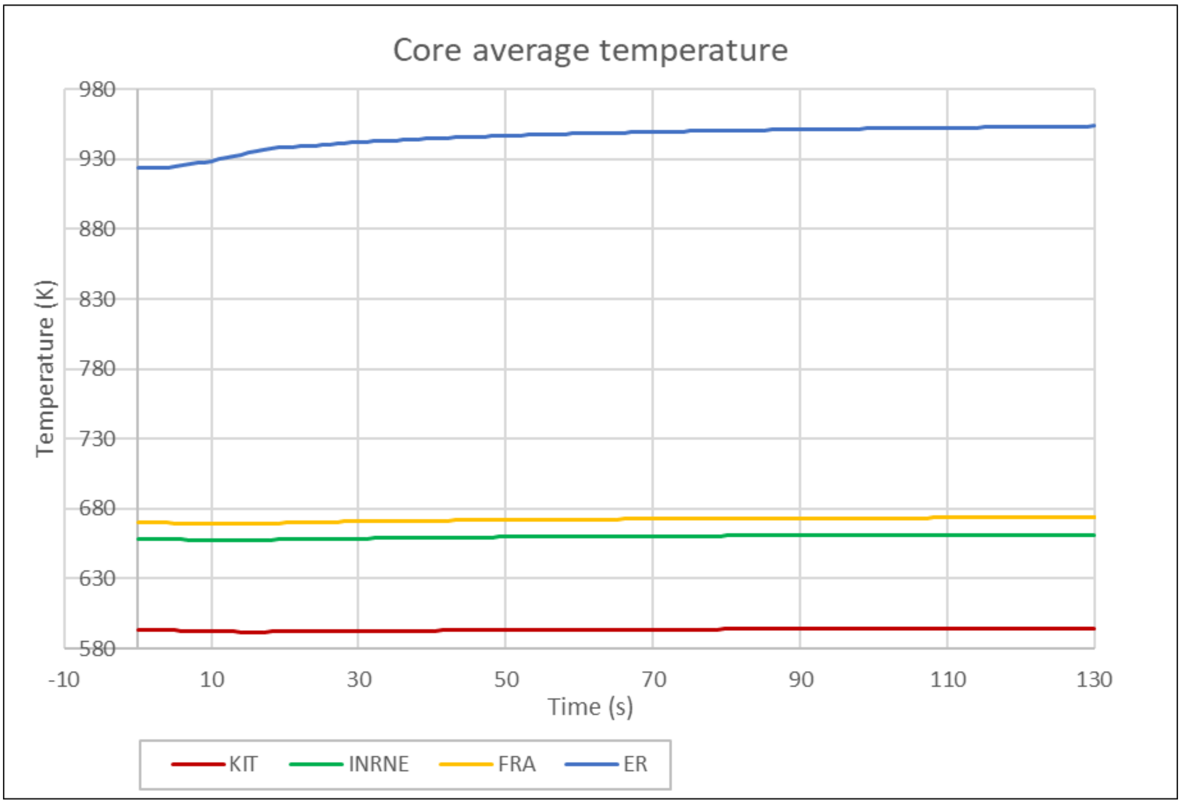

Figure 3.2.12.1 Core average fuel temperature during the transient

4 CONCLUSIONS

The benchmark was performed to improve the thermal-hydraulics modelling of VVER plant, especially challenge robustness and validation of CATHARE3 in the context of VVER reactors.

The main objective of this benchmark was to check the validity of models, regarding the experimental data and code-to-code comparison. The purpose of this benchmark was to test the primary and secondary system model responses using a point kinetics simulation. This exercise enables participants to initialize and verify their thermal-hydraulics models before focusing on the main objective of the benchmark: the testing of coupling methodologies in terms of numerics, temporal and spatial mesh overlays. The benchmark is based on a real plant transient wherein one main coolant pump switches on while the other three pumps are in operation.

The analyses of the results of benchmark show that the participants' results for each parameter are in significantly good agreement with some exceptions. The explanation of observed discrepancies in the participants' results is presented in the section #3 with description of results.

As a general conclusion this benchmark has been performed successfully from four organizations of four countries. The submitted results by the participants are used to prepare code-to-code and code-to-plant data comparisons with subsequent analyses. This benchmark shows that all codes are able to simulate adequately the selected transient. The calculation with CATHARE3 model must be completed with the required regulations and systems.

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