McSAFER

High-Performance Advanced Methods and Experimental Investigations for the Safety Evaluation of Generic Small Modular Reactors

Research and Innovation Actions

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– Deliverable –

D4.2: Analysis of NUSCALE plant with 1D system code and intercomparing between codes

Summary

This deliverable presents the work done within Task 4.1 dedicated to the performance of the classic application of the 1D thermal hydraulic codes to analyse the RPV behaviour under a postulated transient scenario for NuScale plant. The report contains a brief description of the NuScale reactor. Based on public data, the different 1D models of the RPV and the core are developed for the TRACE system code. For transient analysis the boron dilution scenario has been selected. Finally, the results of the calculations by the different individual models are compared. The exercise showed the users and code capabilities to model the specified boron dilution transient under typical SMR plant characteristics as NuScale reactor design although with the limitations of the 1D approximations. These results can therefore constitute the basis for the development of a more realistic three-dimensional multi-scale approach to be applied in the upcoming tasks.

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High-Performance Advanced Methods and Experimental Investigations for the Safety Evaluation of Generic Small Modular Reactors.

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Author(s)	**Oriol Parera Villacampa (JRC), Luca Ammirabile (JRC), Cesar Queral (UPM), Jorge Sanchez Torrijos (UPM), Kevin Fernandez Cosials (UPM), Victor Jammot (TBL)**
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Name	Oriol Parera Villacampa (JRC) Luca Ammirabile (JRC)	Marek Bencik (UJV)	Victor Sanchez (KIT)
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Signatures			Dr. V. Sánchez

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1. Introduction

The H2020 McSAFER project [1] has the main goal of advancing the safety research for SMR by combining experimental investigations with numerical simulations. The focus of McSAFER is on the development, improvement, validation, and application of numerical simulation tools (traditional, advanced low-order and high-fidelity) validated with experimental data generated in European facilities (COSMOS-H, MOTEL, HWAT) that are relevant for the majority of SMR-designs.

In McSAFER project, Work Package 4 - Multiscale Reactor Pressure Vessel (RPV) Analysis Methodologies for SMR - assesses the simulation of the three-dimensional thermal hydraulic phenomena inside the reactor pressure vessel of the integrated SMR-concepts by using multiscale thermal hydraulic tools in combination with traditional one-dimensional system thermal-hydraulic codes. These multiscale methods allow better understanding and description of the thermal hydraulic phenomena inside the RPV by increasing the spatial resolution of the computational domains. In this respect, the same problem is analysed by both approaches using 1D and 3D (coarse mesh) system thermal-hydraulic codes and multi-scale coupled codes, allowing as to compare them. Two SMR designs, NuScale and SMART, are selected as a representative case for thermal hydraulic analyses of RPV behaviour.

Within WP4, Task 4.1 is dedicated to the performance of the classic application of the 1D thermal hydraulic codes to analyse the RPV behaviour under a postulated transient scenario for SMART and NuScale plant respectively.

This report deals with the work performed on the NuScale plant under the postulated boron dilution scenario as a most suitable transient to test the above-mentioned multi-physic code systems [2].

The main text of the present report comprises:

- A brief description of the NuScale plant design and the boron dilution transient;
- The outline of the NuScale models developed by the interested partners for TRACE code;
- The analyses of results addressing the steady-state calculations and the constant power and point-kinetics transients.

2. NuScale Design

The NuScale Power Module ™ (NPM) is an integral concept of Small Modular Pressurized Water Reactor which relies on natural circulation to establish the primary coolant flow. Therefore, the Reactor Coolant System (RCS) and the pressurizer (PZR) are integrated within the cylindrical RPV and no pumps are needed in the primary side, see Figure 2-1.

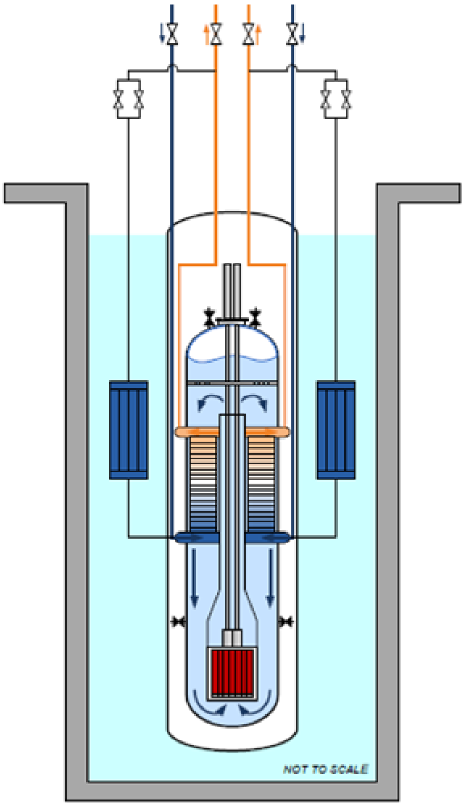

Figure 2-1: General arrangement of an NPM

The RCS is composed by a small-sized core (37 PWR fuel assemblies with 2 meters of active height) with a thermal power of 160 MW, a hot leg which is divided into lower, transition and upper riser regions, the primary side of the steam generators (SGs) (two independent and intertwined helically coiled SGs located surrounding the upper riser with 2 trains of 345 tubes each) and, the cold leg which is formed by the downcomer and the lower plenum.

Regarding the flow path within the RPV, the primary coolant is heated within the core flowing upwards through the riser region. When the flow reaches the top of the riser, it is radially redirected into the annular region between the riser and the RPV and, then it must go through the SGs tube bundles where the heat is transferred to the fluid in the secondary side. As a result, the primary coolant becomes denser and flows downwards by gravity reaching the downcomer and, eventually, the lower plenum region. Then, the primary coolant returns to the core region and primary loop is closed, see Figure 2-1.

The RPV is allocated within a cylindrical Containment Vessel (CNV), which is partially immersed within the reactor pool. The CNV is made in steel providing an enhanced protection against overpressure transients and its internal pressure is maintained at a vacuum during normal operation in order to minimize the heat losses, see [3].

Additionally, it is worth noticing that each NPM has an independent steam supply system (including the turbine), a dedicated Chemical and Volume Control System (CVCS), and two passive safety systems (Emergency Core Cooling System (ECCS) and Decay Heat Removal System (DHRS)).

The CVCS is responsible for controlling the boron concentration in the RCS, the RCS water inventory, the PZR spray for controlling RCS pressure and the extraction of the non-condensable gases accumulated in the PZR steam bubble during normal operation, see [9].

Regarding the safety systems, the ECCS has been designed to deal with accident scenarios in which the normal core cooling by means of the SGs is not achievable (such as LOCA accidents) or low temperature overpressure transients. It is formed by three Reactor Vent Valves (RVVs) and the two Reactor Recirculation Valves (RRVs) mounted at the RPV top head and around 1.8 m above the top of the core, respectively, see [7]. It relies on the CNV for collecting the steam discharged by the RVVs, condense it and, re-inject the condensed within the RPV by means of the RRVs.

On the other hand, the DHRS is devoted to remove the decay heat generated within the core when the reactor is tripped (under non-LOCA conditions and when the normal secondary cooling system is unavailable), see [6]. It consists of two trains of heat exchangers submerged in the reactor pool and connected to each steam (prior to the Main Steam Isolation Valves (MSIVs)) and feedwater lines, respectively, see Figure 2-1. When the NPM isolation is produced, a natural circulation flow rate can be established between the SG and the DHRS heat exchanger driven by the differences in height of the SGs inlet and the bottom of the DHRS heat exchangers.

3. Boron Dilution Transient

To compensate the excess reactivity over the fuel cycle in light water reactors without saturated boiling in the primary circuit under nominal conditions, boric acid is added to the reactor coolant. The coolant circulation during normal operation ensures that this boric acid is homogeneously distributed in the primary circuit.

The NuScale reactor is equipped by the chemical volume and control system (CVCS) which is responsible for controlling the boron concentration in the RCS. An operator error or a malfunction in CVCS system could result in an unintended decrease in boron concentration, which in turn increases the reactivity of the core and decreases the shutdown margin.

Within McSAFER project the boron dilution scenario is evaluated for Mode 1 in two variants:

1. Boron dilution scenario with constant power. Reactor power remains constant throughout the transient (no feedback effects are considered). The aim of the analysis is to determine the time to reach the critical boron concentration (1388.47 ppm). The main purpose of this scenario is to test input models and numerical techniques available in computer codes.
2. Boron dilution scenario with point kinetics where feedback effects are considered.

Each variant has the same initial and boundary conditions except CVCS makeup temperature and feed water mass flow assumption.

3.1. Initial and boundary condition

The following tables summarize the initial and boundary conditions used to calculate the boron dilution event. The parameters are taken from the Design Standard Application (DCA) report of NuScale [3], [4], [5], [6], [7], [8], [9], [10] and [11]. To minimize boron mixing, a minimum RCS mass flow rate is conservatively assumed.

Table 3-1: The primary and secondary system initial conditions

Parameter	**Value**	**Comment**
Primary Side		
Thermal power (MW)	160.0	Mode 1, nom
RCS pressurizer pressure (bar)	127.55	Nom
RCS mass flow rate (kg/s)	535.24	Min
Core mass flow rate (kg/s)	496.17	Min
Core inlet temperature (K)	531.48	Nom
PRZ level (%)	60.0	Nom
Initial boron concentration (ppm)	1600	Max Mode 1, ≥50 percent power
Secondary Side		

SG outlet pressure (bar)	34.47	Nom
FW temperature (K)	421.87	Nom
Total SG mass flow rate (kg/s)	67.07	Nom
CVCS		
Makeup flow rate (kg/s)	3.15	= letdown flow rate
Makeup temperature (K)	278.0	Point kinetics
	531.47	Constant power

Table 3-2: Core initial conditions

Parameter	**Value**	**Comment**
Time in cycle	BOC	Max. boron concentration
MTC (pcm/K)	0.0	Most positive
DTC (pcm/K)	-2.52	Most positive
Boron coefficient (pcm/ppm)	-10	Max, Mode 1
ß-eff	0.0059	BOC
Prompt lifetime (s)	18.35×10^{-6}	BOC
Control rods worth (pcm)	8154	BOC
Axial power profile	BOC	See Figure 3-1
Peak linear power (kW/m)	16.4	
Decay heat	ANS94	For PK delayed neutron groups see Table 3-3

Table 3-3: Delayed neutron groups

Neutron fraction	**Decay cte (s^{-1})**
1.54E-04	3.8700
7.57E-04	1.4000
2.40E-03	0.3110
1.11E-03	0.1150

1.25E-03	0.0317
2.25E-04	0.0127

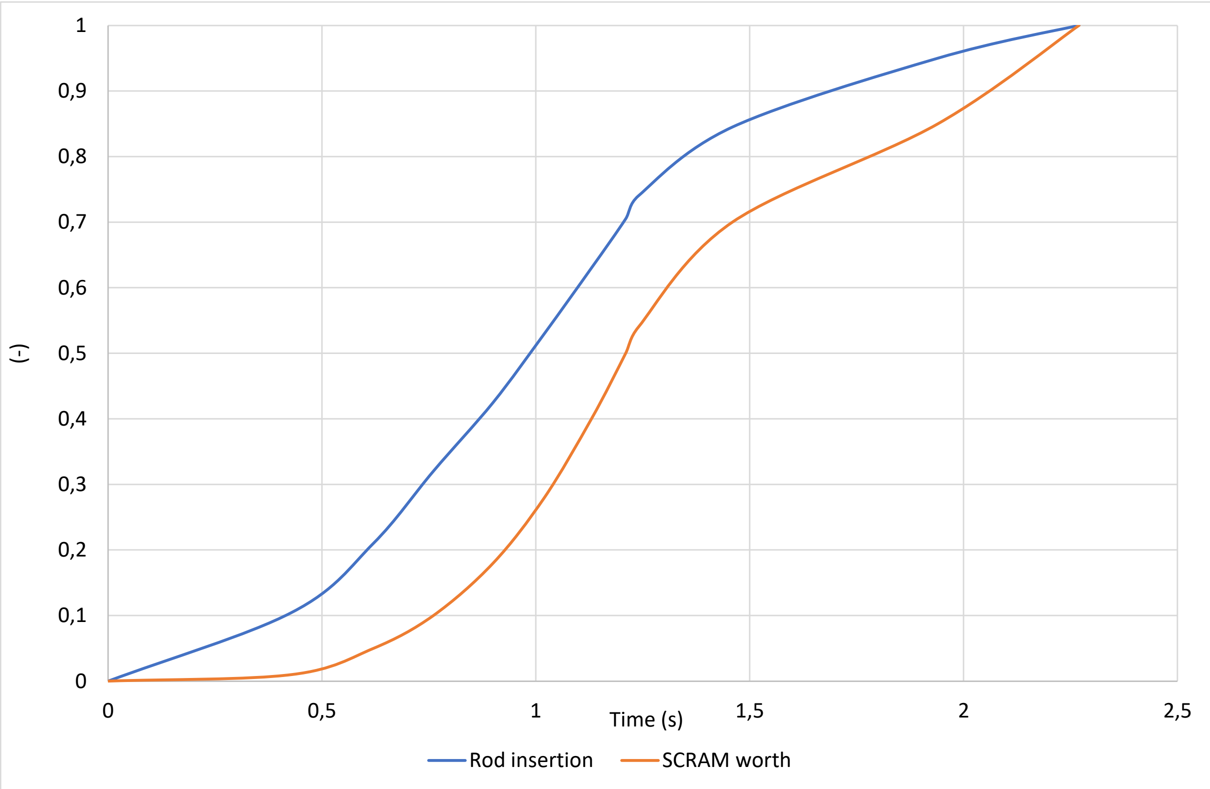

Figure 3-1 Normalized control rod insertion and SCRAM worth

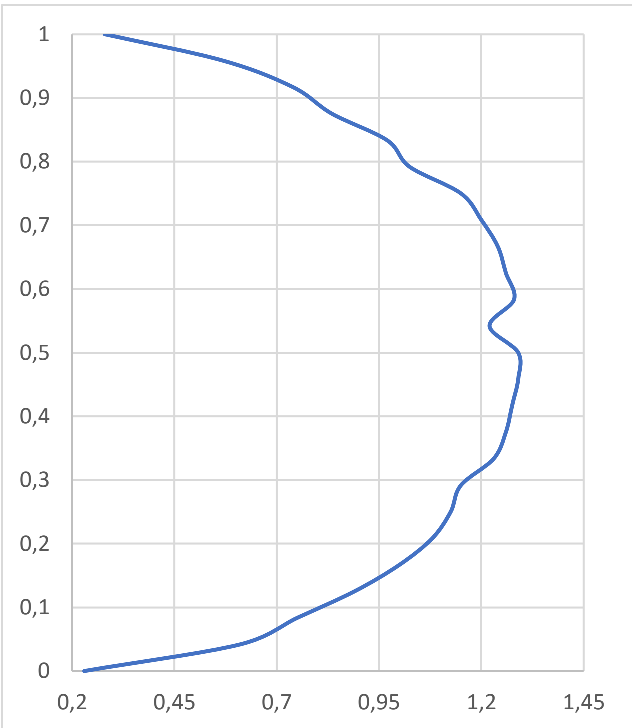

Figure 3-2 Axial peaking factor (BOC)

		0.895	0.911	0.888		
	1.033	1.137	0.957	1.135	1.033	
0.888	1.135	1.054	0.967	1.054	1.136	0.895
0.911	0.957	0.967	1.091	0.967	0.957	0.911
0.895	1.136	1.054	0.967	1.054	1.135	0.888
	1.033	1.135	0.957	1.137	1.033	
		0.888	0.911	0.895		

Figure 3-3 Fuel assembly power peaking factor (BOC)

Table 3-4: Reactor trip function considered in analyses

Signal	**Set point**	**Delay**
High Power Range Linear Power	120 %	2 s
High Pressurizer Pressure	13.79 MPa	2 s
High Narrow Range RCS Hot Temperature	594.26 K	8 s
High Pressurizer Level	80 %	3 s

Table 3-5: ESFAS Function considered in analyses

ESF Function	**Signal**	**Set point**	**Delay**
DHRS	High Pressurizer Pressure	13.79 MPa	2 s
	High Main Steam Pressure	5.516 MPa	2 s
	High Narrow Range RCS Hot Temperature	594.26 K	8 s
Secondary System Isolation	High Pressurizer Pressure	13.79 MPa	2 s
	High Narrow Range RCS Hot Temperature	594.26 K	8 s
	High Main Steam Pressure	5.516 MPa	2 s
	Low Low Pressurizer Pressure	11.032 MPa	2 s

	Low Low Pressurizer Level	20 %	3 s
	Low Main Steam Superheat	0 K	8 s
CVCS Isolation	SCRAM signal	-	7 s
Pressurizer Heaters Trip	Low Pressurizer Level	35 %	3 s
	High Pressurizer Pressure	13.79 MPa	2 s
	High Narrow Range RCS Hot Temperature	594.26 K	8 s
	High Main Steam Pressure	5.516 MPa	2 s

Both JRC and UPM explicitly modelled the Decay heat removal system (DHRS) in their analysis, while TRACTEBEL (TBL) defined ad-hoc boundary conditions for the secondary system after SCRAM.

4. Codes

4.1. TRACE code

The TRAC/RELAP Advanced Computational Engine (TRACE) code [12][12] is a system code developed by the U.S. Nuclear Regulatory Commission (NRC) designed for analysing transient and steady-state neutronic-thermal-hydraulic behaviour in light water reactors. It is the consolidation and combination of the capabilities of the legacy system codes TRAC-P, TRAC-B, RELAP and RAMONA into one computational tool.

The code has been designed to perform best-estimate analyses of loss-of-coolant accidents, operational transients and other accident scenarios for light water reactors. The addition of other fluid properties (sodium, lead, lead-bismuth, molten salts, helium) also allows analyses of innovative reactor systems. The capability exists to model thermal-hydraulic phenomena in both one-dimensional (1D) and three-dimensional (3D) space. It includes models that use multidimensional two-phase flow, non-equilibrium thermo-dynamics, generalized heat transfer, reflood, level tracking and reactors kinetics. The partial differential equations that describe the two-phase flow and heat transfer are solved using finite volume numerical methods and the heat transfer equations are evaluated using a semi-implicit time-differencing technique.

The code is based in a component approach to model the reactor systems. Each physical piece of equipment can be represented as some type of component, and each component can be further nodalized into some number of physical volumes (cells) over which the fluid, conduction, and kinetics equations are evaluated. The TRACE components can include pipes, pressurizers, pumps, separators, turbines, heaters, containments, valves, heat structures (for fuel or surface-convection), boundary conditions (for desired flow or pressure) and vessels.

The non-homogeneous and non-equilibrium modelling approach for two-phase flow includes a full two-fluid (six equations) hydrodynamic model to evaluate gas-liquid flow. A stratified-flow regime has been added to the 1D hydrodynamics; a seventh field equation (mass balance) describes a non-condensable gas field; and an eighth field equation tracks dissolved solute in the liquid field that can plated out on surfaces when solubility in the liquid is exceeded. The basic modelling approach for transient two-phase flow uses flow-regime dependent correlations for the interfacial heat, momentum and energy transfer processes. Nonetheless, the code does not evaluate the stress/strain effect of temperature gradient in structures, nor the effect of fuel-rod-gap closure due to thermal expansion or material swelling.

The code’s computer execution time is highly problem dependent and is a function of the total number of mesh cells, the maximum allowable timestep size, and the rate of change of the neutronic and thermal-hydraulic phenomena being evaluated. The stability-enhancing two-step (SETS) numerics in hydraulic components allows the material Courant limit to be exceeded. This allows very large time steps to be used in slow transients. This, in turn, can lead to significant speedups in simulations (one or two orders of magnitude) of slow-developing accidents and operational transients.

5. Models

For the purposes of the project, a database was created, which is the starting source for the preparation of individual models. The geometrical data and other modelling assumptions are taken mainly from the DCA report of NuScale [3], [4], [5], [6], [7], [8], [9], [10] and [11]; the expert judgement was applied in determining suitable values for missing data.

5.1. JRC model

The JRC NuScale model (Figure 5-1) is focused on developing a detailed primary system and a secondary system with two independent lines limited to the needs to cope with the DHRS system. Therefore, the secondary system has boundary conditions to simulate the feed water, the turbine and the pool.

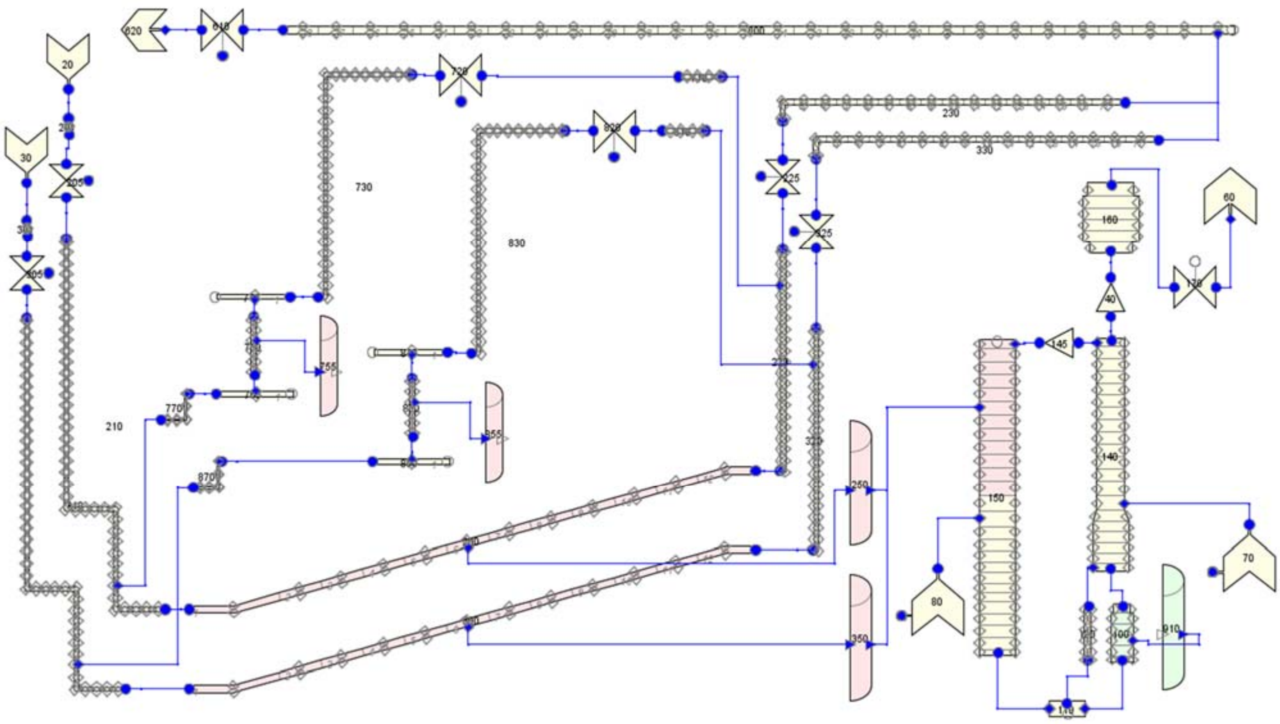

Figure 5-1 JRC NuScale nodalization for TRACE v5.0

The primary system is modelled using the following components:

- 6 PIPES for the core, bypass, riser, downcomer, lower plenum and pressurizer.
- 1 POWER component for the fuel rods.
- 1 HEAT STRUCTURE for core cladding.
- 2 SINGLE JUNCTIONS to connect the riser with the downcomer and the pressurizer.
- 1 VALVE component to model the safety valve of the pressurizer.
- 2 BOUNDARY CONDITIONS to model the CVCS (make-up and letdown).

The nodalization of the primary is structured in a way to obtain an average node height of around half meter. Therefore, the core is subdivided in six axial nodes, four of them dedicated to its active part. The whole fuel assemblies are modelled by a single power component appropriately distributed in a 4-node heat structure to simulate the axial power asymmetry. This power component is set up with the agreed point-kinetics parameters in order to give proper reactivity-feedback and with the desired control rod insertion timing.

The secondary system is modelled with two independent lines, each of them merging the two steam generator bundles into one pipe component. The helicoidal tubes bundle are linked to the upper downcomer region by a heat structure properly adjusted to achieve the desired steady-state heat transfer conditions. The DHRS system is modelled in detail with a fixed pool temperature on the outer side of the condenser tubes as boundary condition.

No heat losses to the containment are considered.

For the purposes of the boron dilution transient, only three signals have been considered (high power range, high pressurizer pressure and high main steam pressure) for the reactor trip, secondary isolation and DHRS actuation.

5.2. Tractebel model

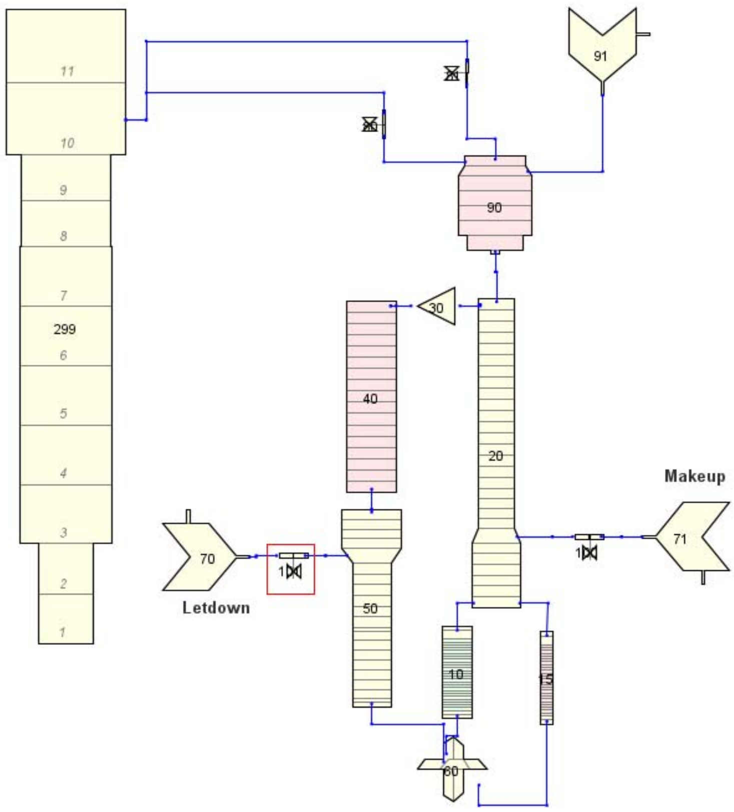

Figure 5-2 TBL - NuScale primary system model

The primary side model is depicted Figure 5-2. The core is represented by a 1D pipe, with an active core of 25 axial level. 1 power component is modelling the core power and connected to 1 heat structure modelling the fuel rods. CVCS makeup and letdown are modelled thanks to FILL components, and can be isolated via trip valves. The pressurizer is modelled by a pipe component, and the pipe wall option is used to simulate a proper thermal inertia. The two helicoidal Steam Generators (SG) are modelled on the primary side by one tube bank cross flow pipe, and on the secondary side by two curved pipes, see Figure 5-3.

On the secondary side, the SG trains have been merged and only one curve pipe is modelled by SG. A model of the Decay Heat Removal System (DHRS) has been developed and is included in Figure 5-3, but it is not used for the present boron dilution transients. In practice, it means that only the logic of the actuation of DHRS is used, but the transient is stopped before actual use of it. Agreed boundary conditions are imposed in the FILL and BREAK components representing the feedwater inlet and turbine, for the purpose of the transient simulation. These boundary conditions are to be found in Table 3-1.

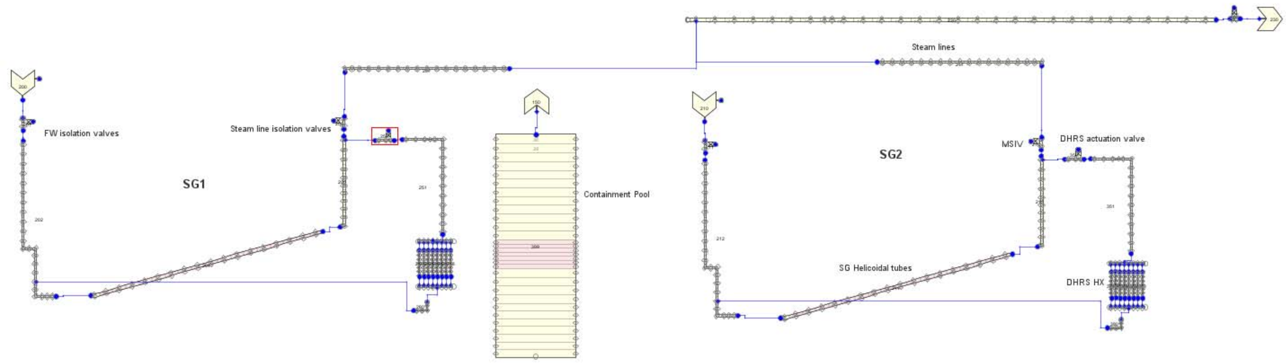

Figure 5-3 TBL - NuScale secondary side model

Heat losses to the containment are simulated via one heat structure modelling a constant heat transfer coefficient with a fixed overall pool sink temperature. These heat losses are taken at the SGs level and have been tuned at nominal power conditions.

All reactor trips and ESFAS signals potentially impacting the boron dilution transient are credited for the analysis.

5.3. UPM model

As its contribution to the task 4.1 of the WP4, UPM research group has developed a TRACE model of a NuScale Power Module using 1D components. The expected scope of the model for this task is limited to the RPV. Therefore, the hydraulic conditions in the secondary side, beyond the steam and feedwater lines up to the MSIVs and FWIVs, respectively, are considered in the model as boundary conditions, see Figure 5-4. Those BCs has been computed using a full-plant TRACE model also developed within the UPM research group see Figure 5-5 to Figure 5-7.

It is remarkable that the NuScale core region has been modelled using a PIPE component with 22 axial levels (20 axial levels are devoted to the active core height, the others are used for the top and bottom nozzles regions, respectively). Additionally, the Core Upper and Lower Plates and the core bypass have been also included in the model by independent PIPE components.

Given the special geometry of SGs in the primary side, a 'Tube Bank Crossflow' PIPE is selected in order to apply the most accurate correlations available in TRACE to model this singular region. The secondary side of the SG tubes is modelled by means of four PIPEs simulating each tubes bundle of the SGs. The 'Curved Pipe' option of the PIPE component in TRACE has been used to consider the helical shape of the tubes. The heat transfer in the SGs region is simulated with HTSTR components considering a reduction margin of 10% in the heat transfer area due to the tube plugging, see **Error! Reference source not found.**.

It must be also noted that the PZR is modelled by means of the dedicated PIPE type available in TRACE allowing the simulation of the PZR heaters. In that sense, the PZR heaters trip setpoints are deployed due to the actuation of the 'High PZR Pressure' signal or by low PZR level.

The CVCS makeup and letdown flow rate are considered by means of FILL components controlling the RCS boron concentration in the RCS during normal operation conditions. The CVCS isolation is achieved by the closure of the CVCS isolation valves with the actuation of any reactor scram signal.

All SCRAM signals described in the DCA report are included in the model which is very relevant because the transient BCs in the secondary side are activated when certain SCRAM signals are triggered. The reactivity feedbacks are included in the point kinetics model.

Finally, it is important to keep in mind that boron dilution is one of the sequences to be analysed. Therefore, the boron concentration evolution across the RCS should be computed. To do so, a high-order numerical technique is implemented in the model in order to avoid numerical diffusion, the option selected are the Van Leer method modified with flux limiters for solving the spatial differences and the semi-implicit method for the time integration.

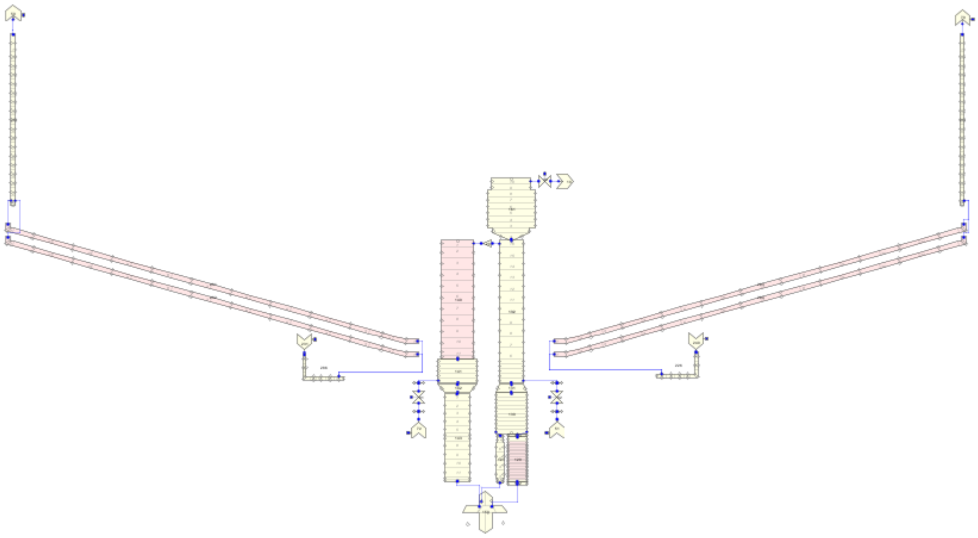

Figure 5-4 Nodalization scheme of the 1D TRACE model (UPM model)

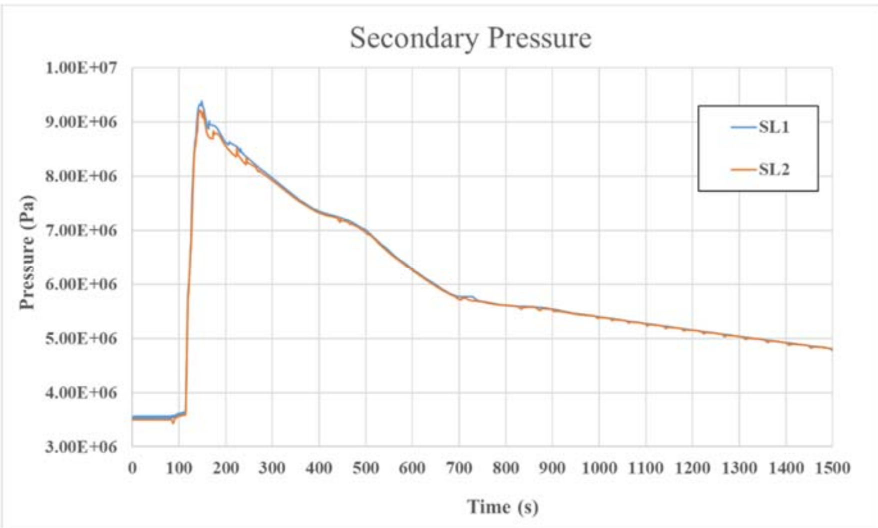

Figure 5-5 Secondary side hydraulic BCs: SG Outlet Pressure

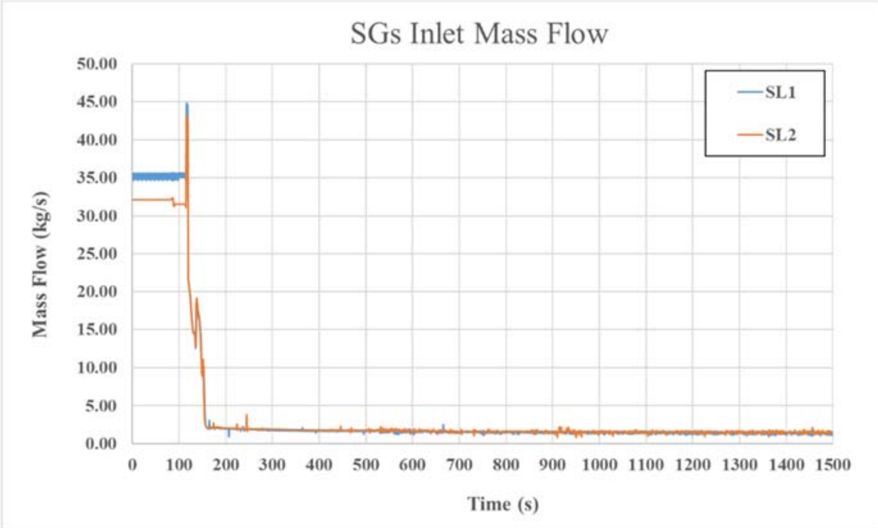

Figure 5-6 Secondary side hydraulic BCs: SG Inlet Mass flow

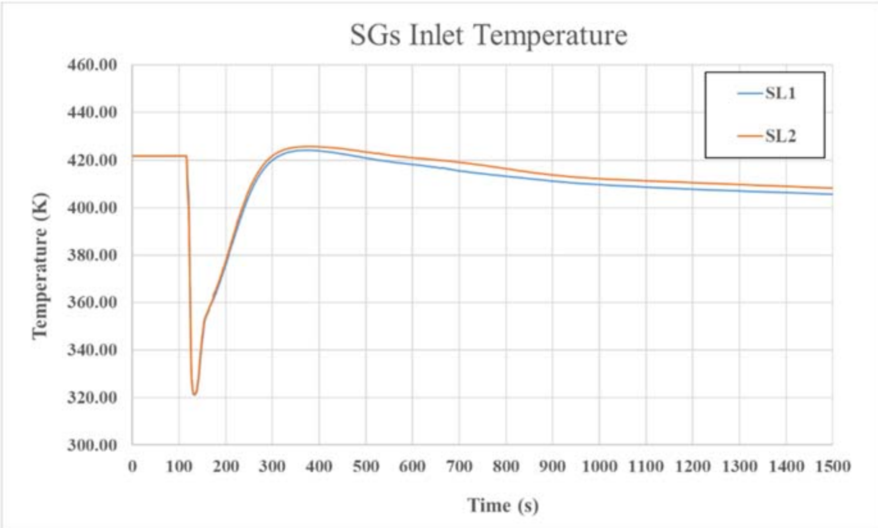

Figure 5-7 Secondary side hydraulic BCs: SG Inlet Temperature

6. Analysis of Results

6.1. Steady-state Calculations

The Table 6-1 shows a comparison of the main steady state thermalhydraulic plant parameters. Calculated parameters are also compared (if possible) with reference values taken from DCA report. It can be stated that the calculations are in good agreement with the DCA, the differences found are minimal.

Table 6-1: Steady state parameters

Parameter	DCA	JRC (error %)	UPM (error %)	TBL (error %)
Primary pressure [bar]	127.55	127.59 (0.03)	127.70 (0.18)	127.64 (0.07)
Core Power [MW]	160	160 (0.0)	160 (0.0)	160 (0.0)
Core inlet T [K]	531.48	531.46 (0.0)	533.04 (0.29)	532.00 (0.09)
Core outlet T [K]	-	591.55	592.66	591.99
RCS mass flow rate [kg/s]	535.24	535.27 (0.0)	535.10 (0.03)	535.13 (0.02)
Core mass flow rate [kg/s]	496.17	495.86 (0.06)	497.02 (0.17)	496.03 (0.01)
PRZ level [%]	60	60.02 (0.03)	60.00 (0.0)	60.39 (0.65)
Boron concentration [ppm]	1600	1600.04 (0.0)	1600.00 (0.0)	1600.00 (0.0)
Primary inventory [kg]	-	47382.90	46751.37	47533
FW mass flow rate [kg/s]	67.07	67.08 (0.01)	67.10 (0.04)	67.07 (0.0)
SG secondary inlet T [K]	421.87	421.89 (0.0)	421.93 (0.01)	421.96 (0.02)
SG secondary outlet T [K]	580.04	585.59 (0.96)	567.64 (2.14)	567.26 (2.20)
Secondary pressure [bar]	34.47	34.87 (1.16)	34.88 (1.19)	34.47 (0.0)

6.2. Transient Calculations: Constant Power

The purpose of these calculations is testing the numeric techniques available in computer codes for boron transport simulation. For this reason, the effect of feedback is not considered, and the reactor power remains constant throughout the calculation.

The boron dilution can be described by slug flow or dilution front (wave front) mixing model where unborated water injected into the RCS is assumed to mix with a slug of borated water at the injection point. The diluted slug is assumed to move through the RCS (i.e. through the riser, steam generators, downcomer, and finally though the reactor core). The change in core boron concentration with time depends on the location of the diluted slug.

Based on the parameters listed in the DCA report (see [11]), UPM calculated the time to loss the SDM for the slug flow model to be 1879 s and boron concentration at this instant equal 1388.47 ppm.

A comparison of the calculated values of the time to loss the SDM with DCA value is shown in Table 6-2.

Obviously, all three calculations show very similar behaviour – see Figure 6-1. The boron dilution rate is in good agreement with the DCA report.

Table 6-2: Results of constant power scenario

Parameter	DCA	JRC	UPM	TBL
Time to loss of SDMs [s]	1879	1823	1835	1864.6

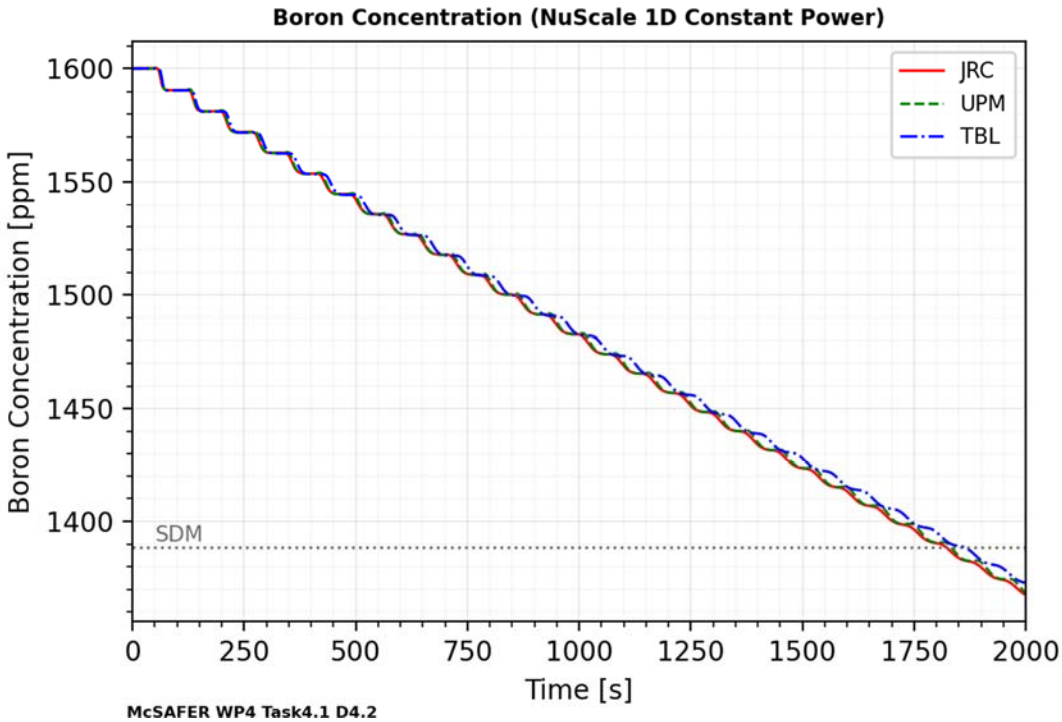

Figure 6-1 Boron Concentration for the Constant Power transient

6.3. Transient Calculations: Point Kinetics Power

The transient calculations using point kinetics models were performed using the specifications provided in Section 3.1.

Error! Reference source not found. shows the chronology of events in the respective transient calculations. In general terms, JRC and UPM results show similar evolution with about 20 seconds delay in the SCRAM signal actuated by the High Pressurizer Pressure signal, while TBL results show a later SCRAM signal actuated instead by the High Core Power signal. It is important to note that the goal of this work was not to perform a benchmark exercise but rather to assess the code capabilities to model the specified boron dilution transient under typical SMR plant characteristics as NuScale reactor design.

A comprehensive comparison among the calculated time trends is given below.

Table 6-3 Results of Point Kinetics chronology

Event	JRC	UPM	TBL
Boron Dilution begins [s]	0	0	0
SCRAM set point [s]	121	97	135.4
SCRAM signal	High Pressurizer Pressure	High Pressurizer Pressure	High Core Power
CRs start to insert [s]	123	99	137.4
MSIVs and FWIVs fully closed [s]	128	104	208.4
ESFAS set point [s]	129	105	195.4
ESFAS signal	High Main Steam Pressure	High Main Steam Pressure	Low Steam Superheat
CVCS fully isolated [s]	130	106	144.4
DHRS valves fully opened [s]	153	129	NA

As shown in Figure 6-2, it takes about 50 seconds after the initiating event before the diluted slug reaches the core causing a drop in the boron concentration. This leads to positive reactivity insertion (Figure 6-3) that is only partially compensated by the Doppler effect (Figure 6-4) and results in a total positive reactivity feedback (Figure 6-5). The Figure 6-6 shows a close comparison between the total reactivity and the boron concentration where it can be seen that the effect of an early SCRAM signal (consequently ending the dilution) by UPM avoids reaching the second diluted slug, whereas for JRC and TBL this is later enough to cause a second drop on the boron concentration.

As consequence of the positive reactivity, the core power increases (Figure 6-7) as well as the primary system pressure (Figure 6-8). While the rate of power increase is similar in JRC and UPM simulations, TBL results show a lower rate, due to a lower gradient in the boron concentration drop (Figure 6-2), and the primary pressurisation is also lower. This explain why JRC and UPM calculations reach first the High Pressurizer Pressure SCRAM signal instead of TBL results where SCRAM signal is due to High Core Power (Table 6-3).

Following the SCRAM signal, the insertion of the control rods makes the power drop to decay heat rate and triggers the CVCS isolation, ending the boron dilution. After SCRAM, the induced thermal unbalance generates oscillations observed in the core inlet (Figure 6-9) and outlet (Figure 6-10) temperatures and in the riser (Figure 6-11), core (Figure 6-12) and core by-pass (Figure 6-13) mass flows with a tendency to converge asymptotically to new operational conditions.

For JRC and UPM the High Pressurizer Pressure signal leads also to the Secondary System isolation (closure of the MSIV and FWIV) but for TBL the Secondary System isolation is triggered later by the Low-Steam Superheat signal. Due to the closure of steam and feedwater lines, the pressure in the secondary systems rises up to 94 bars in JRC and UPM calculations as well as the outlet steam generator temperature (Figure 6-15 and Figure 6-17). Afterwards the activation of the DHRS systems leads to a gradual depressurisation of the secondary system. The DHRS activation can also be observed in Figure 6-16 where the secondary mass flow drops until reaches the new balance with the functioning of DHRS. Figure 6-18 show the evolution of heat transfer between primary and secondary following the secondary steam and feedwater closure and the opening of the

DHRS. For TBL calculations the dynamics of the secondary are different because the scram signal for high core power does not trigger the closure of the secondary MSIV and FWIV. After scram, the steam generators continue to work normally until the low-steam superheat signal leads to the closure of the secondary MSIV and FWIV. The higher heat transfer between primary and secondary after scram explains the drop in pressure (Figure 6-8) and in pressurizer level (Figure 6-14) observed in TBL calculations.

For JRC and UPM the DHRS actuation is triggered by the High Main Steam Pressure signal shortly after the Secondary System isolation and enables a satisfactory heat removal strategy through the secondary as it can be seen in Figure 6-16 and Figure 6-18. Nevertheless, for TBL the DHRS is not actuated during the transient and the residual heat is transferred as heat losses to the containment and to the primary, as it can be seen in the increases of pressure and pressurizer level after its secondary isolation.

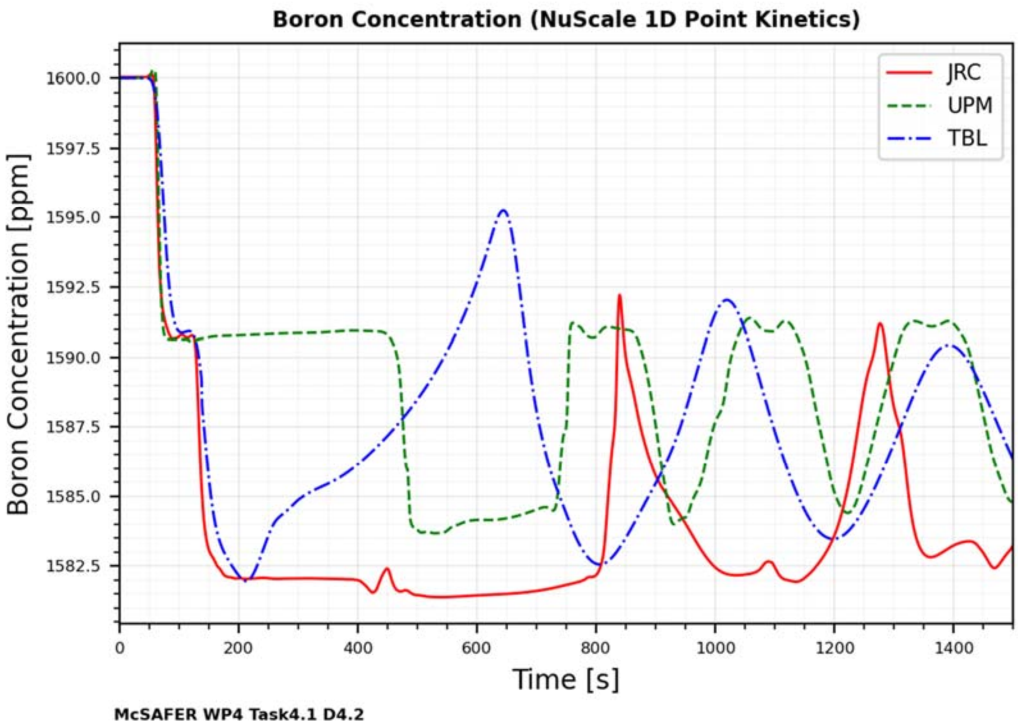

Figure 6-2 Average boron concentration in the core

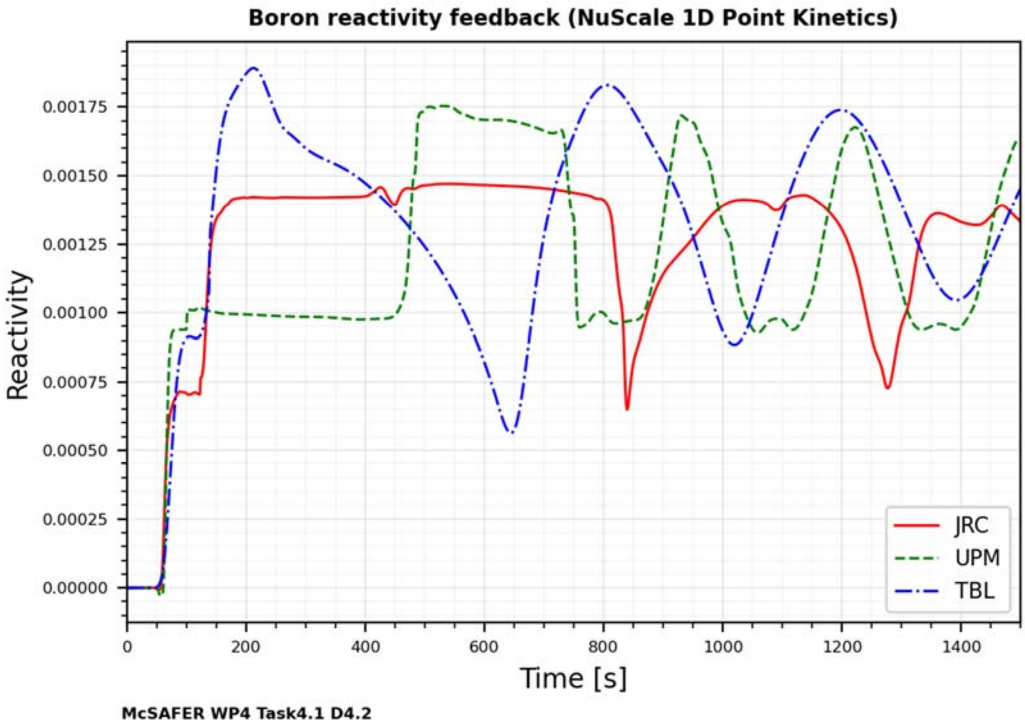

Figure 6-3 Boron reactivity feedback

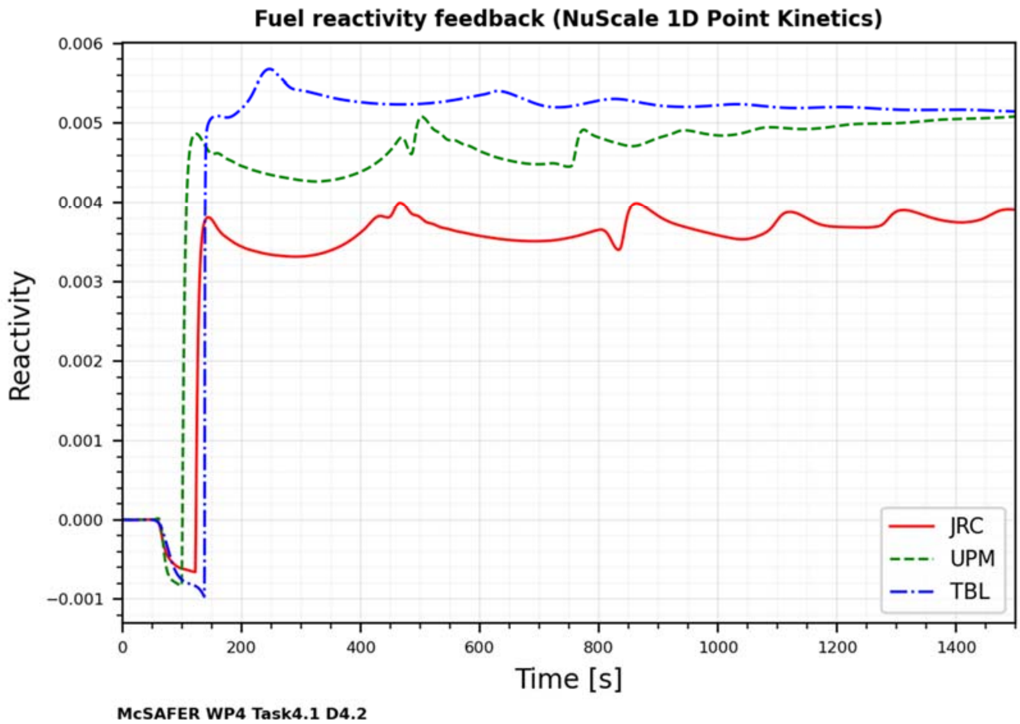

Figure 6-4 Fuel reactivity feedback

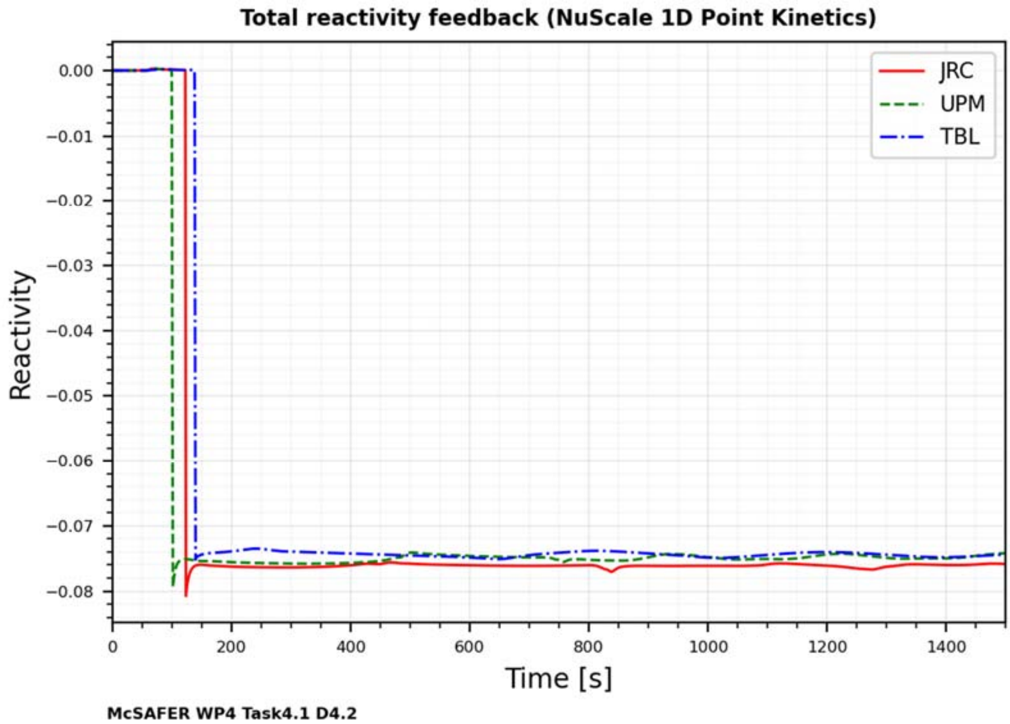

Figure 6-5 Total reactivity feedback

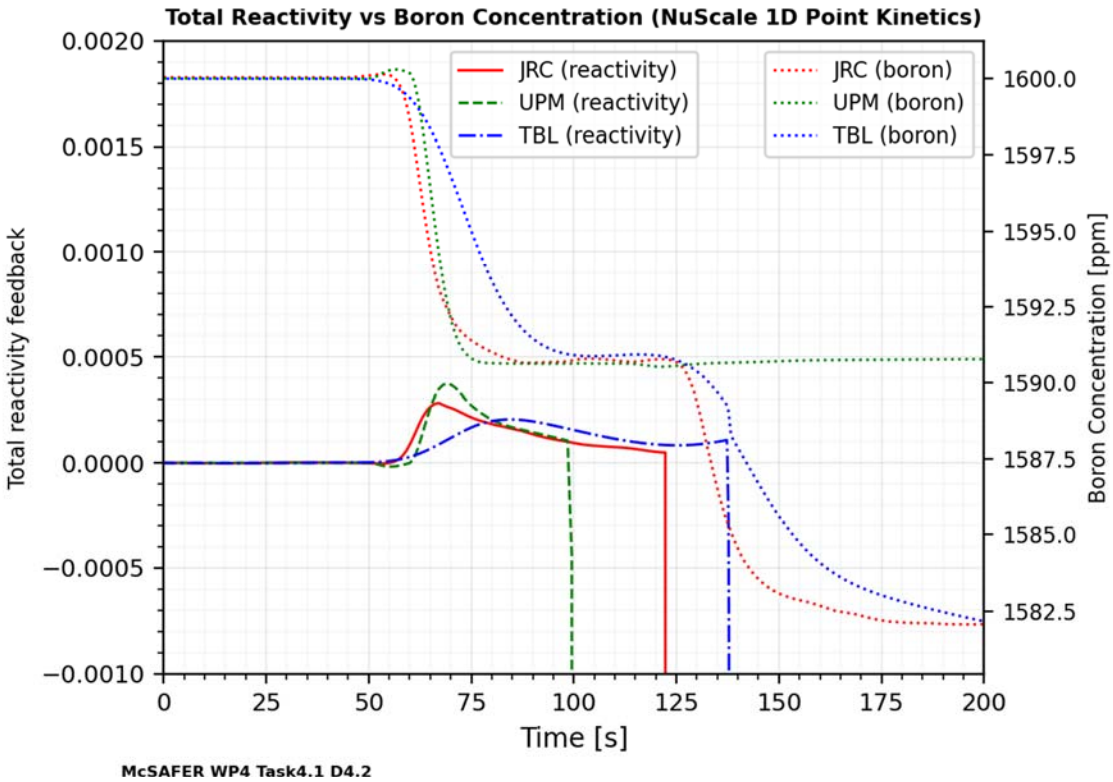

Figure 6-6 Total reactivity feedback vs Boron Concentration

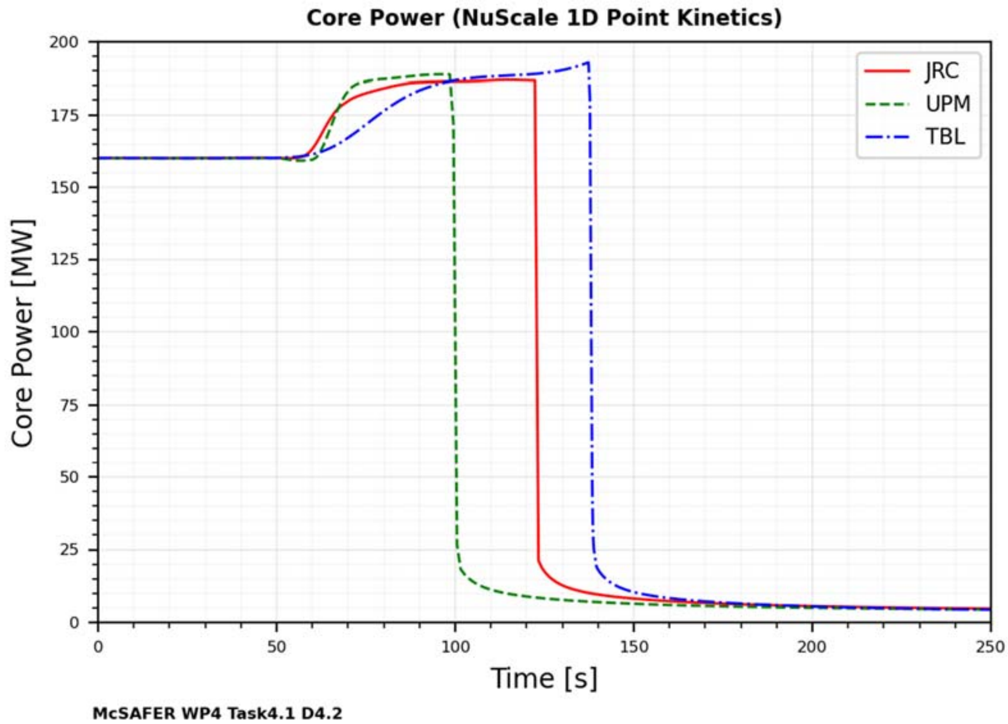

Figure 6-7 Core Power

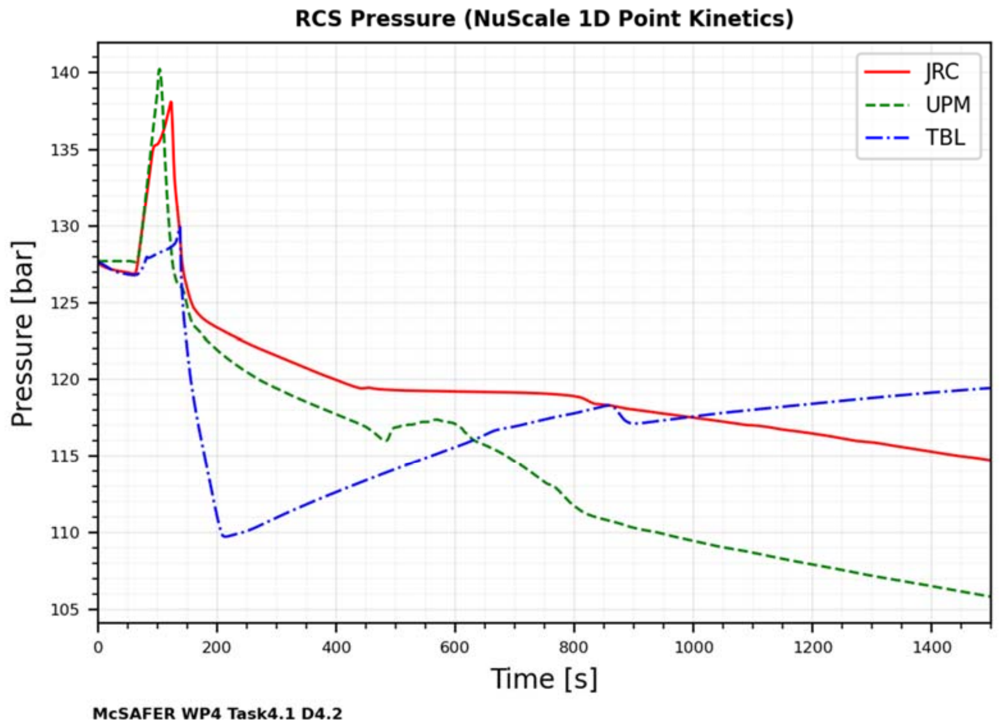

Figure 6-8 RCS Pressure

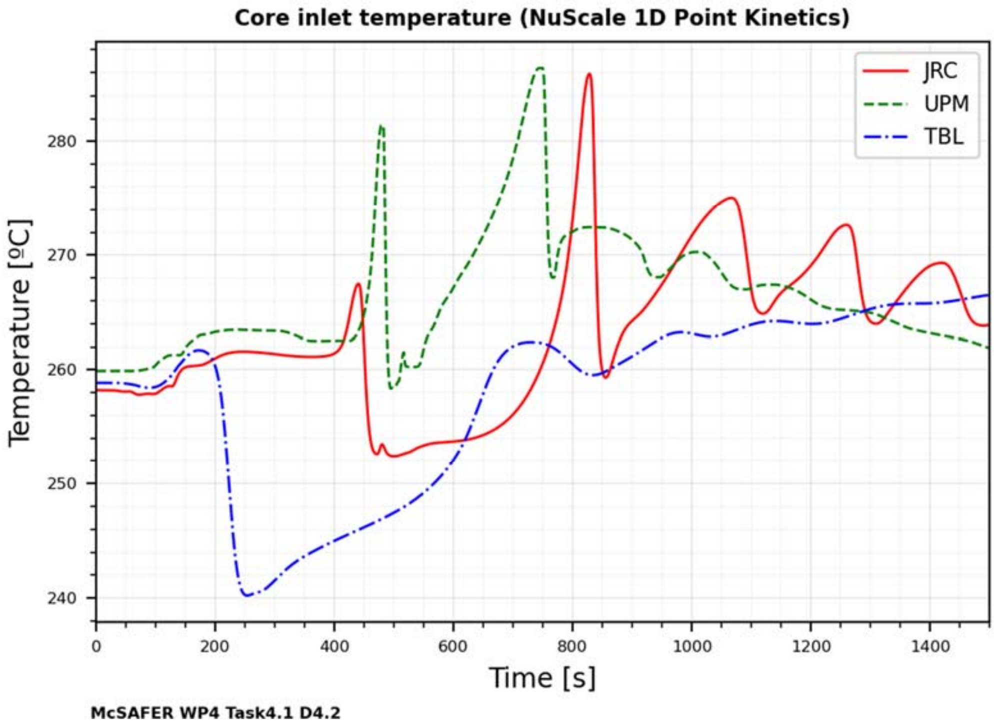

Figure 6-9 Core inlet temperature

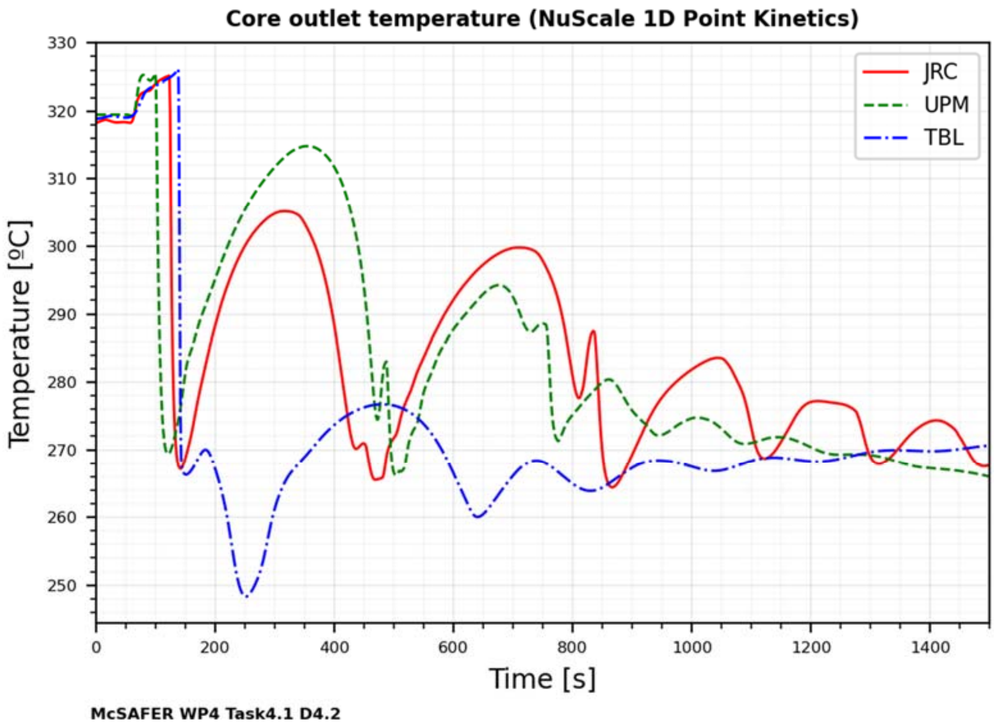

Figure 6-10 Core outlet temperature

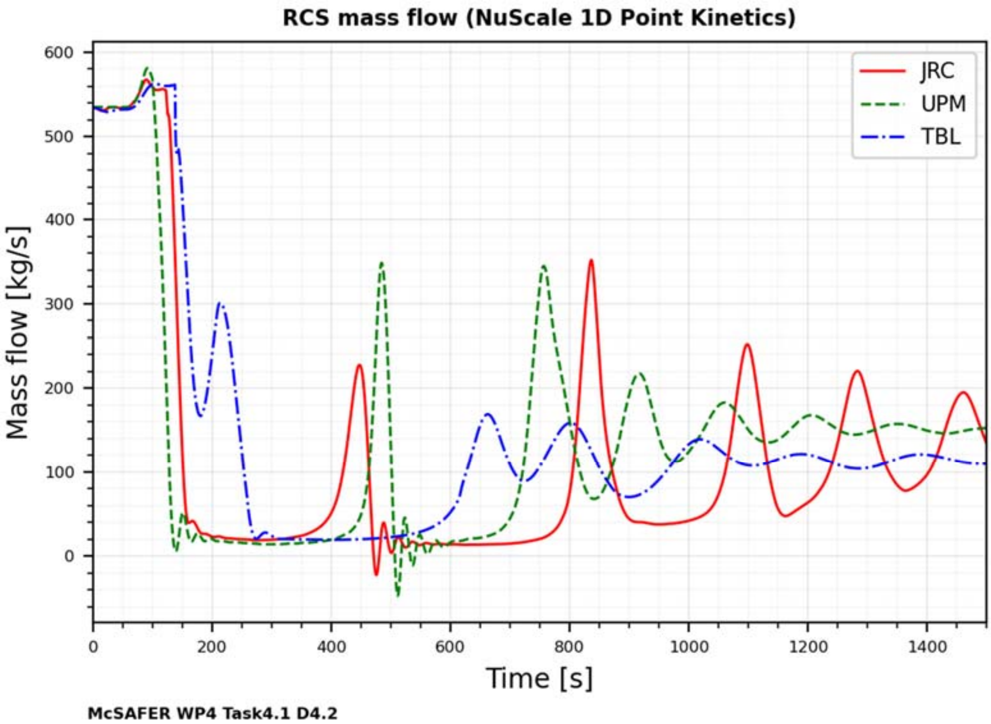

Figure 6-11 RCS mass flow

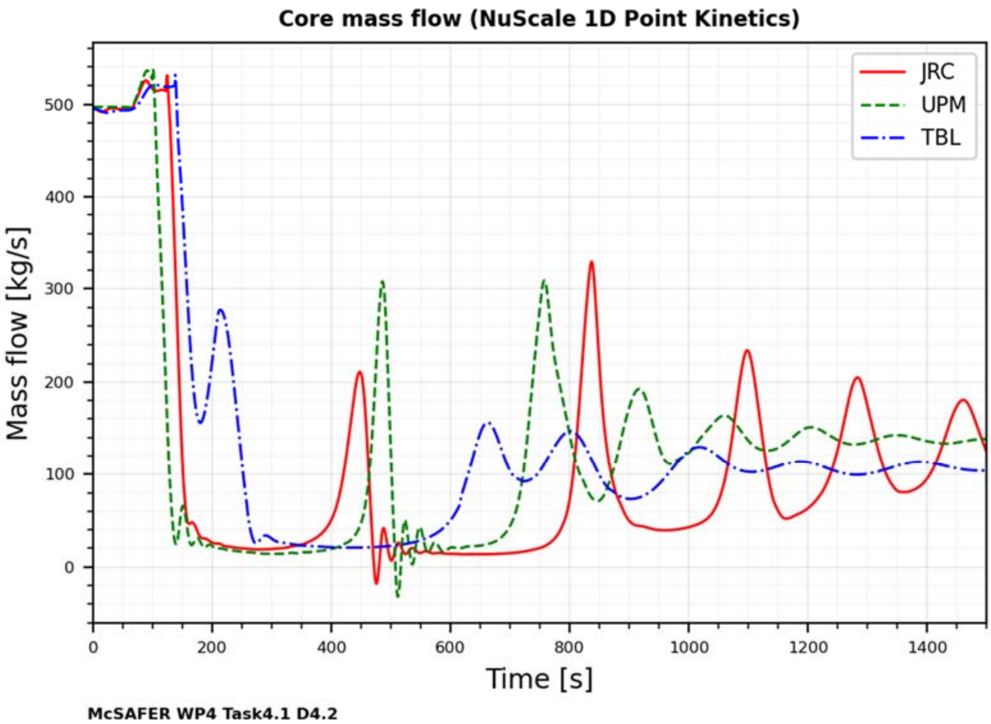

Figure 6-12 Core mass flow

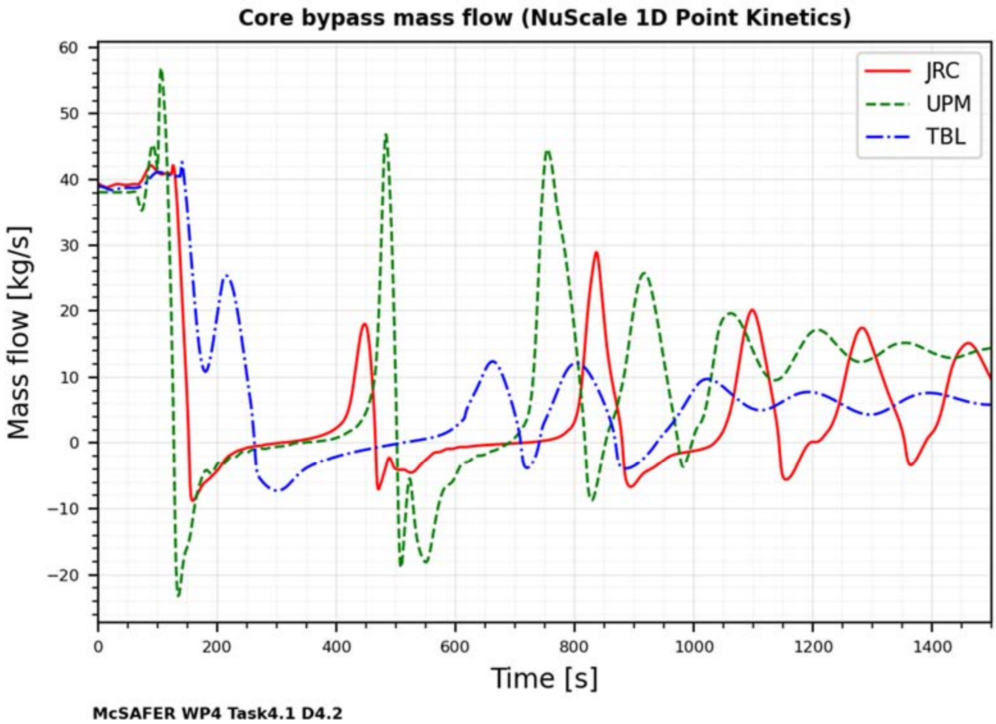

Figure 6-13 Core bypass mass flow

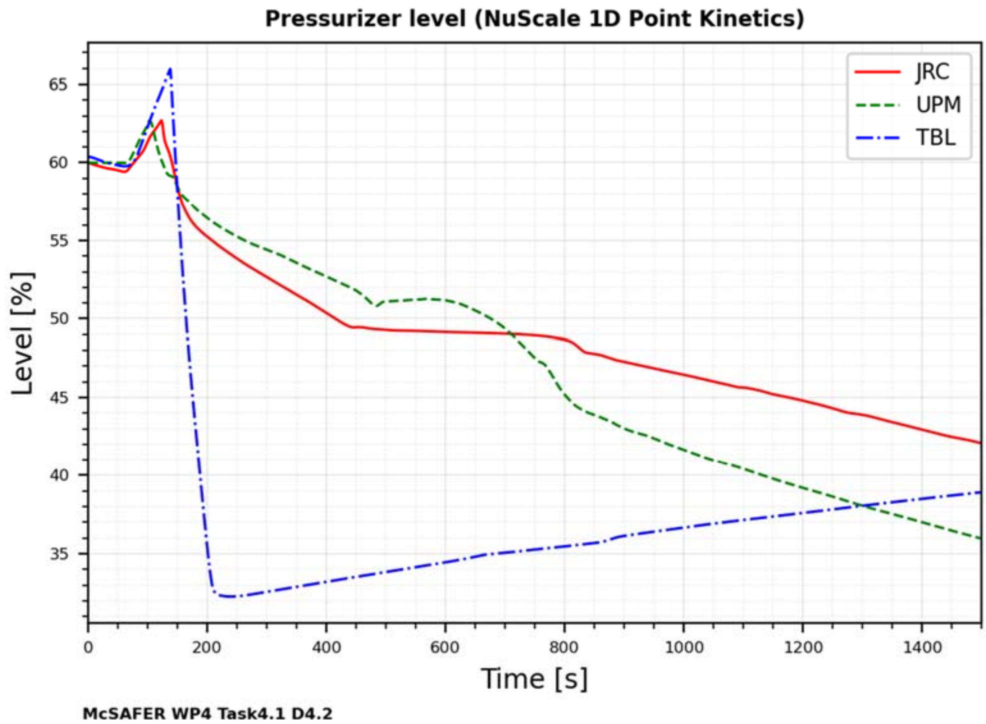

Figure 6-14 Pressurizer level

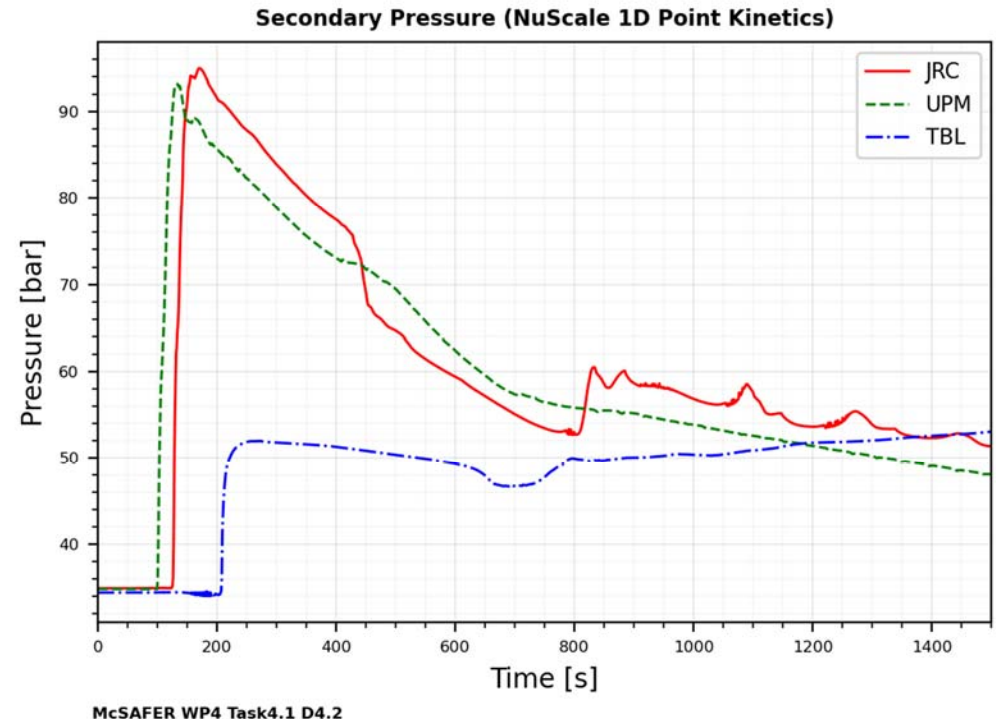

Figure 6-15 Secondary Pressure

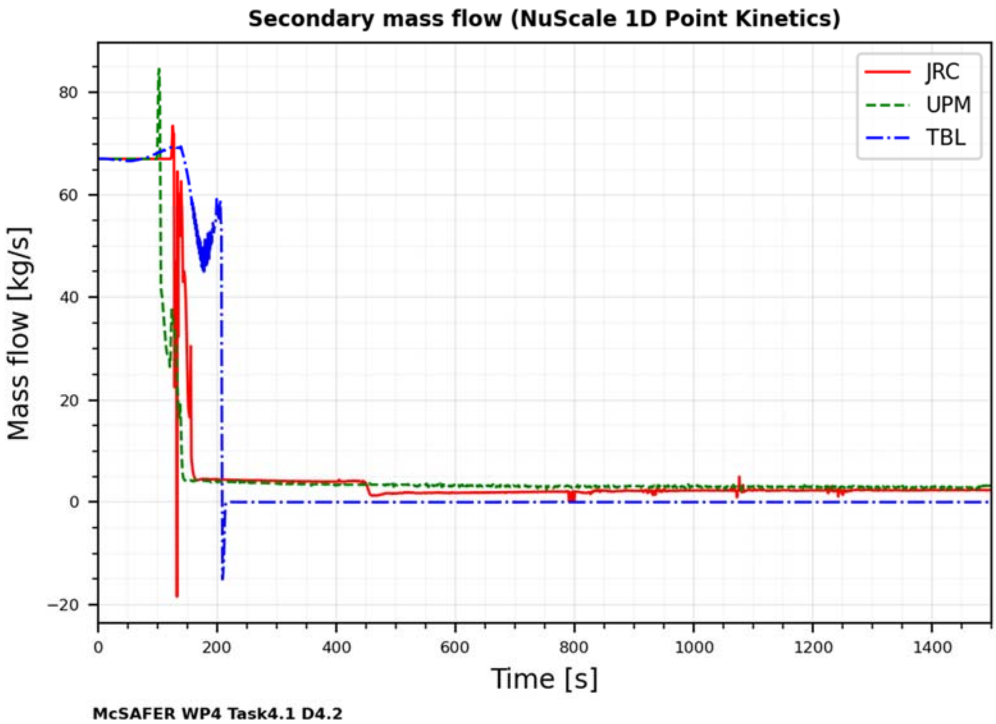

Figure 6-16 Secondary mass flow

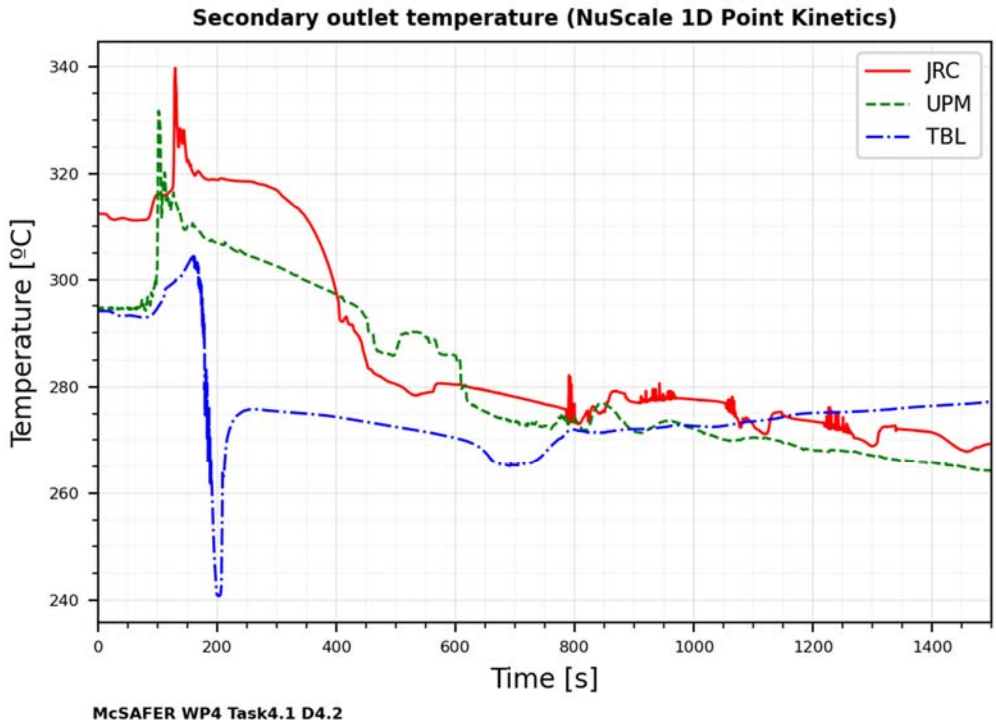

Figure 6-17 Secondary outlet temperature

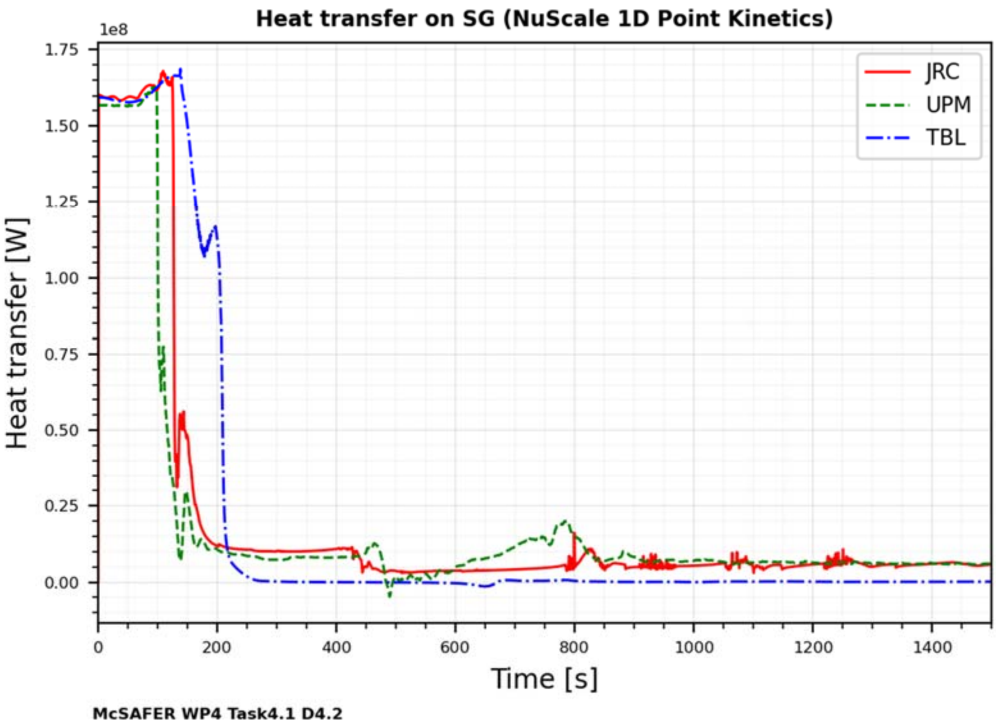

Figure 6-18 Heat transfer between the primary and secondary system

7. Conclusions

The Work Package 4 (Multiscale Reactor Pressure Vessel (RPV) Analysis Methodologies for SMR) of McSAFER project aims at assessing three-dimensional thermal hydraulic phenomena inside the reactor pressure vessel of the integrated SMR-concepts by using multiscale thermal hydraulic tools in combination with traditional one-dimensional system thermal-hydraulic codes.

Within WP4, Task 4.1 deals with the performance of the classic application of the 1D thermal hydraulic codes to analyse the RPV behaviour under a postulated transient scenario for SMART and NuScale plant respectively.

In this report, it was described the work performed on the NuScale plant under the postulated boron dilution scenario as a most suitable transient to test the above-mentioned multi-physic code systems.

For the purposes of the project, a database was created, based on the geometrical data and other modelling assumptions taken mainly from the DCA report of NuScale, while expert judgement was applied in determining suitable values for missing data.

The three project partners involved in the task, JRC, TRACTEBEL and UPM, developed respectively three NuScale-type models using the thermalhydraulic code TRACE.

The comparison of the main steady state thermalhydraulic plant parameters calculated by the three models with reference values taken from DCA report showed a general good agreement with minimal differences.

A first set of calculations was performed considering a constant thermal power without reactivity feedback to test the numeric techniques available in computer codes for boron transport simulation. All three calculations show very similar behaviour and the computed boron dilution rate is in good agreement with the DCA report.

The second set of transient calculations was performed using the point kinetics models based on the specifications provided in Section 3.1.

The chronology of events in the respective transient calculations showed some discrepancies that could be, nonetheless, explained by the differences in the models and the numerical scheme used to compute the boron transport.

In fact, it is important to stress that the objective of this work was not to perform a benchmark exercise but rather to assess the code capabilities.

In this respect, the exercise showed the users and code capabilities to model the specified boron dilution transient under typical SMR plant characteristics as NuScale reactor design although with the limitations of the 1D approximations.

These results can therefore constitute the basis for the development of a more realistic three-dimensional multi-scale approach to be applied in the upcoming tasks.

8. References

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