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

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– Deliverable –

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

Summary

This deliverable contains a description of the SMART reactor. Starting from the SMART reactor description, two models are developed for the TRACE system code. In the first model, a 1D approach is used for most components including the reactor core. In the second model, the 3D flow-solution features are used to model the RPV and the core. Steady-state and transient calculations are performed. For the transient calculation, the ATWS transient has been chosen. The ATWS transient is a loss of normal feedwater without an automatic scram of the reactor. Finally, the results of the simulated ATWS transient using the 1D and 3D system models are compared.

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

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1. Introduction

The H2020 McSAFER project [1] has the main goal of advancing the safety research for Small Modular Reactors (SMRs) 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 the McSAFER project, Work Package 4 (WP4) - Multiscale RPV Analysis Methodologies for SMR, assesses the simulation of the three-dimensional thermal-hydraulic phenomena inside the Reactor Pressure Vessel (RPV) of the integrated SMR concepts by using multiscale thermal-hydraulic tools in combination with traditional one-dimensional system thermal-hydraulic codes. These multiscale methods provide a 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 multiscale coupled codes and the results are compared to assess the methodologies. Two SMR designs, NuSCALE and SMART, are selected as representative cases for thermal-hydraulic analyses of RPV behaviour.

Within WP4, Task 4.1 is dedicated to the application of traditional 1D thermal-hydraulic codes to analyse the RPV behaviour under a postulated transient scenario for the SMART and NUSCALE plants. The same scenarios are analysed in Task 4.2, this time with a more advanced approach consisting of system thermal-hydraulics with three-dimensional analysis of the flow in the RPV.

This report deals with the work performed on the SMART reactor under the postulated ATWS scenario as a most suitable transient to test the above-mentioned code systems [7]. The approaches used in tasks 4.1 and 4.2 are compared to assess both methodologies and the potential advantages of using a 3D modelling approach for the RPV.

The main text of the present report comprises:

- A brief description of the SMART plant design and the studied ATWS transient;
- The outline of the SMART models developed by the interested partners for the TRACE code;
- The analysis of results addressing the steady-state and transient calculations.

2. SMART reactor description

The System-integrated Modular Advanced Reactor (SMART) is an SMR design developed by the Korea Atomic Energy Research Institute (KAERI) [2], [3]. The reactor is a PWR-type SMR that produces 330 MW of thermal power to generate 100 MW of electricity.

The primary coolant system is integrated, i.e. all primary loop components are contained inside the Reactor Pressure Vessel (RPV), as shown in figure 1. The power is extracted from the RPV by eight modular-type, once-through helicoidal steam generators, which in nominal operating conditions produce 30 °C superheated steam in the secondary side. The forced-convection flow through the RPV and the core is maintained by four canned pumps. The in-vessel pressurizer is designed to control the system pressure at a nearly constant level over the entire design basis events [4]. The core is composed of 57 PWR-like fuel assemblies. A flow-mixing header assembly is located at the steam generator outlet to create a uniform coolant mass flow distribution at the core inlet. The reactor operates at about 15 MPa and a core inlet and outlet temperature of respectively 296 °C and 323 °C [5].

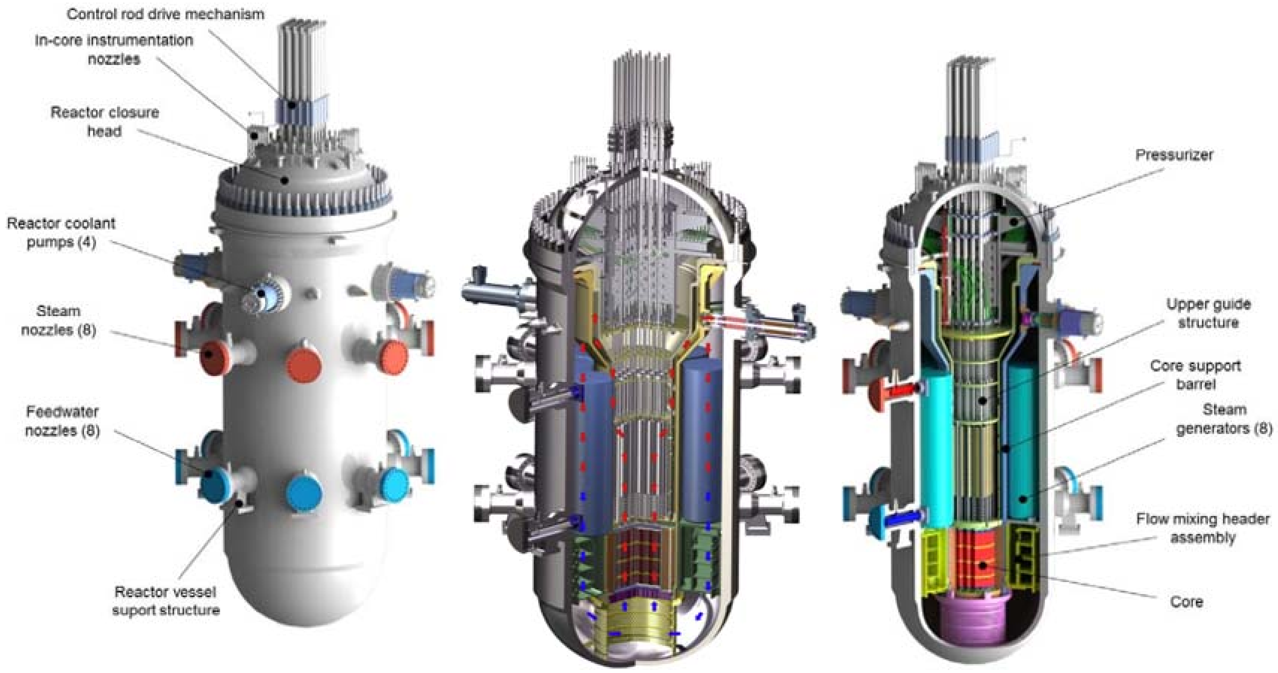

Figure 1. SMART reactor pressure vessel and integrated primary loop [4], [6].

Several side circuits and safety systems are connected to the reactor pressure vessel but are not discussed in detail in the present deliverable since only the RPV itself is modelled within WP4 of the McSAFER project. One of the main safety systems is the pressurizer safety valve on the RPV to protect the RPV against an overpressure. Another important safety system is the Passive Residual Heat Removal System (PRHRS) which is connected to the RPV via the secondary circuit to keep the reactor cooled during an abnormal shutdown. When the PRHRS is actuated, the steam on the secondary side is directed towards a large pool in which the steam is condensed to water. The water circulates back to the steam generators inlet via natural circulation. The PRHRS can operate for 72 hours without any corrective action by the operators [5], [7].

3. ATWS transient

This work deals with the analysis of an Anticipated Transient Without Scram (ATWS) for the SMART reactor, which is the accidental scenario selected for WP4. The models considered deal only with the RPV, which is the scope of WP4, and hence are closed with boundary conditions on the secondary side of the steam generators.

3.1. ATWS transient selection

An ATWS accident is classified as a Beyond Design Basis Event (BDBE). These are characterized by an anticipated operational occurrence followed by the failure of the first shutdown system. For SMART, two typical ATWS initiating events are [8]:

- **Decrease in the heat removal by the secondary system**: a loss of feedwater, a loss of load or a loss of condenser vacuum result in an increase in the primary system coolant temperature, as well as in an increase in the pressurizer level and therefore in its pressure. In the case of a loss of load or a loss of condenser vacuum, the pressure of the secondary system also increases due to a closure of the turbine stop valve. The reactor shuts down due to the negative moderator temperature feedback, and the long-term cooling is established by the PRHRS, which is activated normally.
- **Loss of offsite power**: the reactor coolant pumps as well as the feed water pumps, shut down after offsite power is lost, reducing sharply the core mass flow rate and leading to a Total Loss of Coolant Flow Accident (TLCFA). The rapid increase in the coolant temperature due to the loss of flow leads to a quick shutdown of the reactor due to the moderator feedback. The PRHRS ensures the long-term cooling of the reactor.

Within WP4 the decrease in the heat removal by the secondary system is selected to be the studied ATWS transient. The transient initiator is a total loss of feedwater modelled by the sudden closure of all feedwater isolation valves. The automatic reactor scram is not considered in the transient analysis, but the reactor shuts down due to the negative reactivity coefficient. The feedback effects are modelled by using a point kinetic model of the core. In the analysis, it is assumed that the offsite power remains available. Consequently, the forced circulation of coolant through the RPV is maintained.

3.2. Initial and boundary conditions ATWS

The initial conditions are summarized in the tables below [9]. No secondary-side parameters are listed, since only the RPV is modelled within the WP4.

The model used in WP4 needs to be closed with boundary conditions for the RPV for the selected ATWS transient. These are obtained from a calculation in which the whole plant was simulated, i.e. the RPV and the secondary side. The secondary side consisted of the steam lines and the passive residual heat removal system. In this calculation the ATWS transient is simulated and the PRHRS is actuated 5.6 s after the sudden closure of the feedwater isolation valves allowing for a short time delay of the signals and opening of the PRHRS valves. From this calculation the inner surface steam generator temperature corresponding to the secondary side of the steam generators is taken and implemented in the model used for the WP4 ATWS analyses.

Parameter	Value
Thermal power (MW)	330.0
RCS pressurizer pressure (MPa)	15.0
RCS mass flow rate (kg/s)	2090
Core mass flow rate (kg/s)	45.98
Core inlet temperature (K)	568.55
Core outlet temperature (K)	596.15

Table 1. Primary-system initial conditions.

Parameter	Value
DTC (pcm/K)	-2.0
MTC (pcm/K)	-76.0
β-eff	0.00687
Power profile	Axial

Table 2. Main core parameters.

4. Computer codes description

The thermal-hydraulic code used by both Tractebel and KIT for this task is the TRACE plant analysis tool, which is described next.

4.1. TRACE code

The TRAC/RELAP Advanced Computational Engine (TRACE) code [10] 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) numeric 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 description

The TRACE models used by Tractebel and KIT are described in the next two sections. While the two models are very similar, the former consists of a 1D flow description while the latter uses a fully 3D scheme.

5.1. Tractebel model

The RPV model used for the Tractebel calculations is presented in Figure 2. The vessel is represented by a 3D vessel component but the flow paths in the azimuthal direction are blocked to stay as close as possible to the 1D system code as requested by Task 4.1 description. A cylindrical vessel is used with 21 axial, 5 radial and 8 azimuthal sectors. These 8 azimuthal sectors are required since the steam generators are symmetrically divided around the core riser. All components are connected to this RPV component. One of these components is the core which is represented by a 1D pipe. To represent the core power a heat structure is created for the core and a power component is assigned to this heat structure. Since there is a small core bypass, this is also modelled by a 1D pipe component to capture the correct coolant flow rate through the heated core. Note that the core and core bypass pipes are connected to the RPV via 8 single junctions. This ensures that the coolant is distributed evenly through the 8 azimuthal sectors.

The coolant flows upwards through the core and core riser (inside the RPV). On the top of the RPV 4 reactor coolant pumps are modelled by using a pump component ensuring the correct total flow through the RPV. The reactor coolant pumps force the flow through the 8 steam generators which are modelled on the primary side by the tube bank crossflow pipe type. Each steam generator is assigned to his azimuthal sector. Each steam generator also includes a heat structure for the heat transfer to the secondary side. Since in WP4 no secondary side is modelled, boundary conditions are implemented (see Section 3.2) on the outer surface representing the secondary side. From the steam generators the flow is directed downwards and back at the core inlet.

At the top of the RPV the pressurizer is modelled via a pipe component which contains water in the lowest axial layers and steam in the top axial layers. On the top of the pressurizer a safety valve is modelled via the valve component to protect the RPV against an overpressure.

No reactor scram signals are modelled in the WP4 as the considered scenario (ATWS) assumes a failure of the first shutdown system.

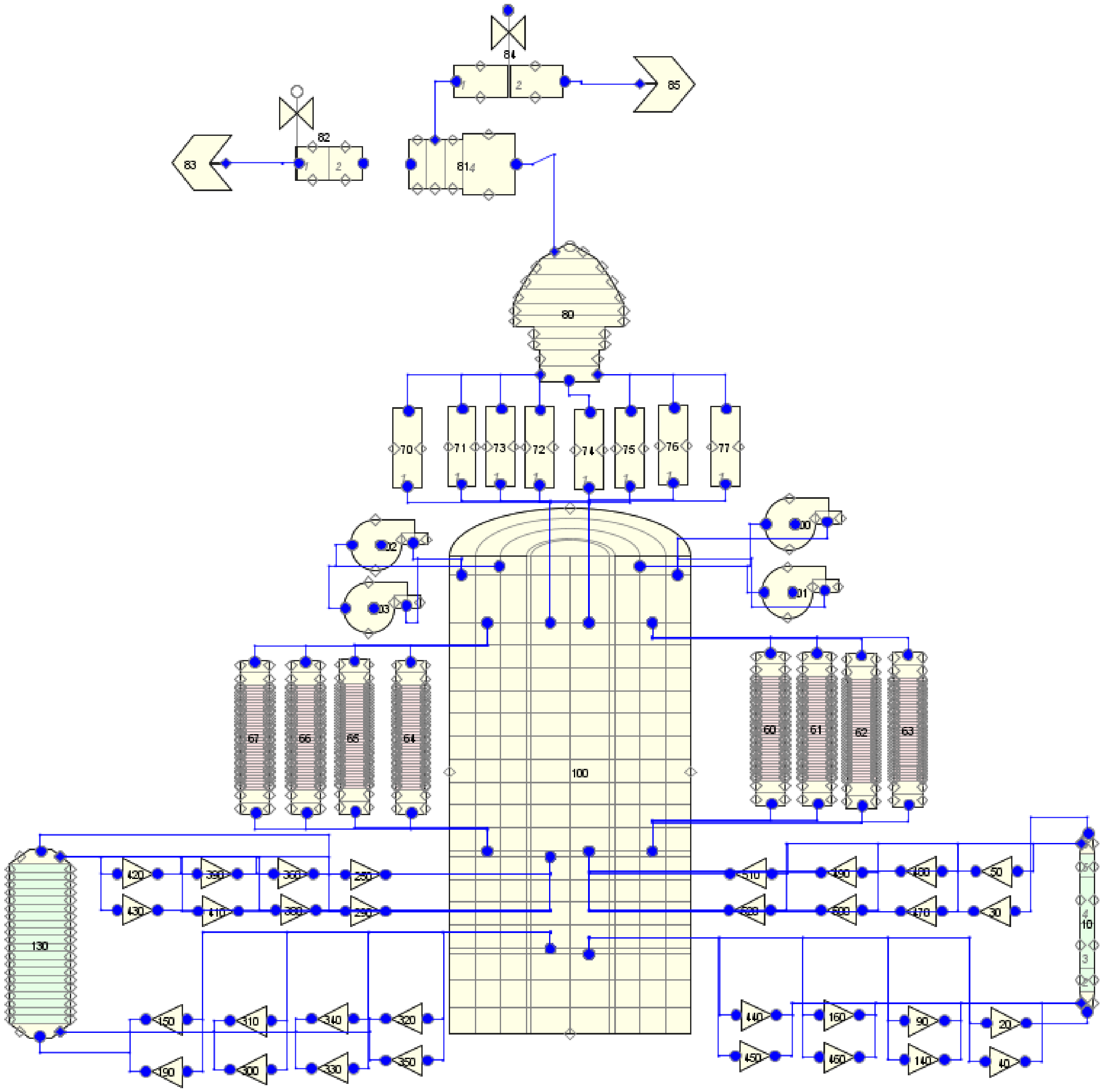

Figure 2. Tractebel RPV SMART model.

5.2. KIT model

The RPV model used for the KIT calculations is presented in Figure 3. The vessel is represented by the same 3D vessel component as in the Tractebel model, but here a full 3D flow calculation is used, i.e. the azimuthal direction is simulated. The core is modelled with a 3D cartesian mesh with the nodal-level geometry, which consists of 57 fuel assemblies. The core bypass consists of 8 1D pipes, one for each azimuthal sector. The rest of the primary-side components, i.e. the pressurizer, the primary coolant pumps and the steam generators, are the same as for the Tractebel model.

In this model, the flow through the downcomer, the flow-mixing header assemblies, the lower plenum, the core, the upper plenum and the riser are modelled with a fully 3D approach, the core on a cartesian mesh and the rest on the 3D vessel component. A traditional 1D scheme is used for the pressurizer, the pumps and the steam generator, as well as for the core bypass.

The secondary-side boundary conditions are implemented in the same way as in the Tractebel model.

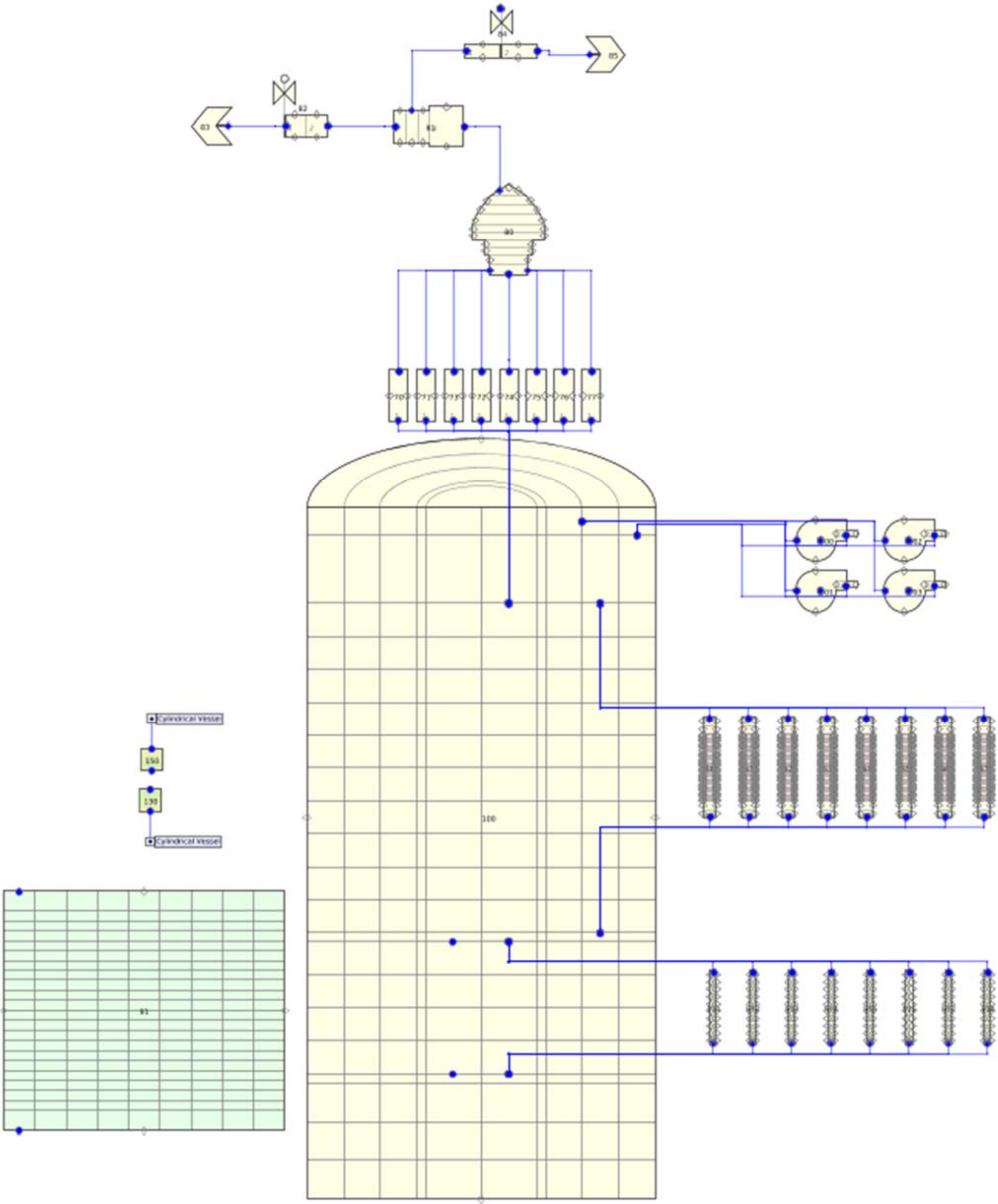

Figure 3. KIT RPV SMART model.

6. Analysis of results

The next two sections present the steady-state and transient results obtained with the Tractebel and KIT models.

6.1. Steady-state calculation

Table 3 shows a comparison of the main thermal-hydraulic plant parameters computed by the TRACE code considering the models in sections 5.1 and 5.2, as well as results published in the literature. The differences between them are judged acceptable. The only large deviation is related to the pump impeller speed, but since the correct total RPV flow rate is obtained and the pumps are kept running throughout the studied transient, this was not investigated in detail. Note also that no secondary parameters are mentioned as only the RPV is modelled for WP4.

Parameter	Literature [9]	TBL (error %)	KIT (error %)
Primary pressure (MPa)	15.0	15.0 (0.0)	15.0 (0.0)
Core Power (MW)	330.0	330.0 (0.0)	330.0 (0.0)
Core inlet T (K)	568.85	570.25 (0.2)	568.22 (0.1)
Core outlet T (K)	596.15	597.85 (0.3)	595.69 (0.1)
Total RPV flow (kg/s)	2090.0	2090.0 (0.0)	2088.3 (0.1)
Pump impeller speed (rad/s)	179.0	106.8 (40.3)	129.0 (27.9)
Core pressure drop (kPa)	Between 5-45	24.3	27.9

Table 3. Steady-state results.

6.2. Transient calculation

In the transient calculation the boundary conditions retrieved from the full-scale model are implemented to close the RPV models used here (see Section 3.2). The ATWS transient is initiated at t = 100 s. The main results obtained with both models are shown in Figures 4 to 11.

The ATWS results in the closure of all feedwater isolation valves and the failure of the first shutdown system. Before the closure of the feedwater isolation valves, the core power is stable at 330 MW. Once the isolation valves close, the core power decreases rapidly due to the total negative feedback effects. The sudden closure of the feedwater isolation valves starts the PRHRS but a 5.6 s delay is introduced considering the signal delay and opening of the valves. The heat removal by the PRHRS is insufficient at the initial stage and the core average temperature increases, resulting in an increase in the PRZ level and pressure. The total RPV pressure increases and can only be reduced by the opening of the PRZ safety valve. Once the safety valve opens the RPV pressure reduces and due to the lower core power, the capacity of the PRHRS becomes sufficient. Consequently, the core average temperature reduces and no return to criticality is observed in the first 5 minutes. This allows sufficient timing for the operators to manually trip the reactor.

Regarding the comparison between the Tractebel and KIT models, no significant differences can be observed during the duration of the transient. The differences in fuel and coolant temperatures are of the order of a few degrees and lay in the typical uncertainty range arising from modelling decisions. The power evolution is quite similar for both models and the primary pressure is essentially the same. This is obviously to be expected, since both TRACE models are quite similar and the accidental scenario has no asymmetry that could lead to large differences between the 1D and 3D flow models.

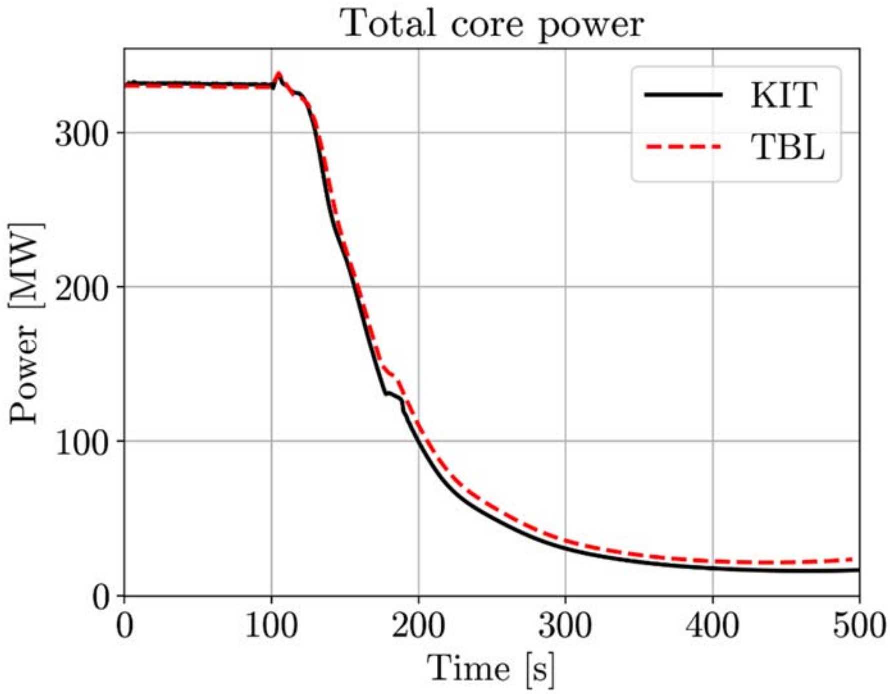

Figure 4. Total core power.

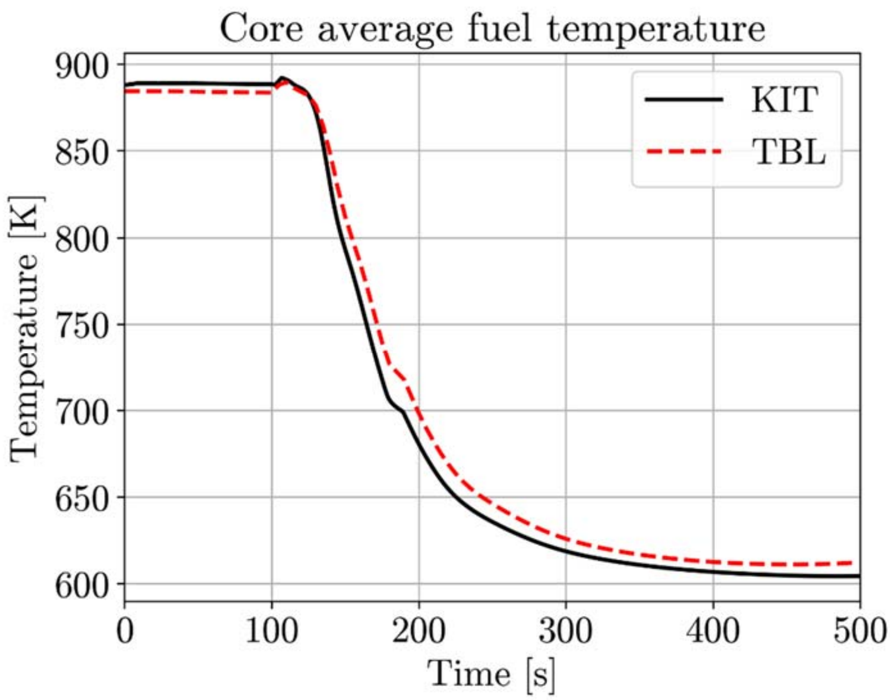

Figure 5. Core average fuel temperature.

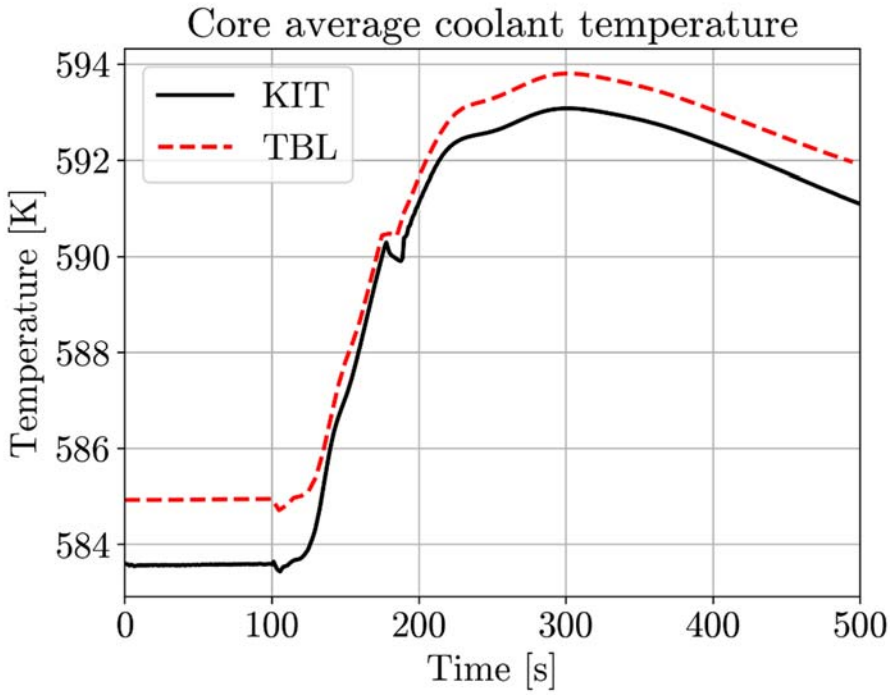

Figure 6. Core average coolant temperature.

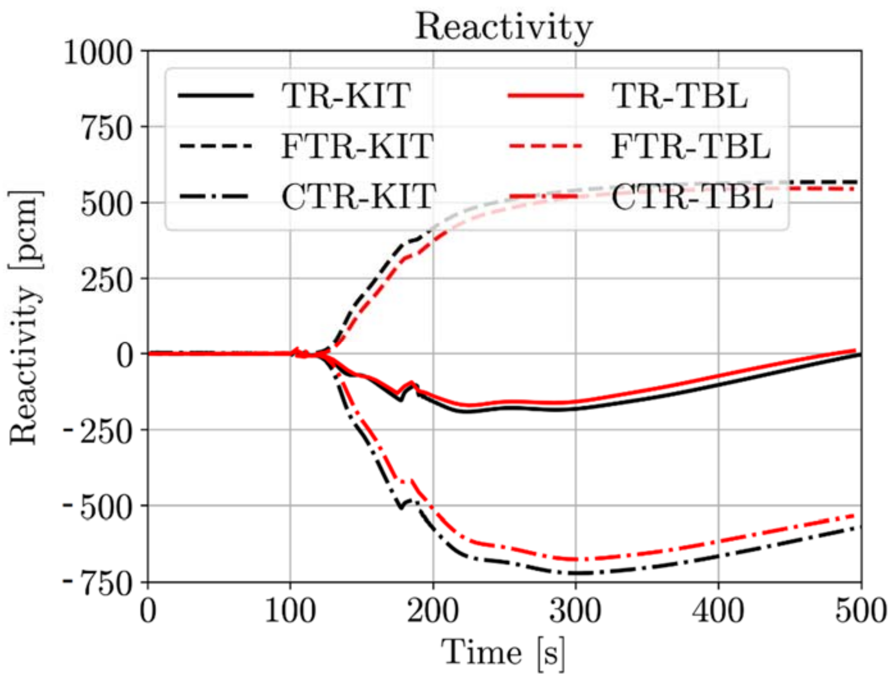

Figure 7. Total (TR), fuel-temperature (FTR) and coolant-temperature (CTR) reactivity.

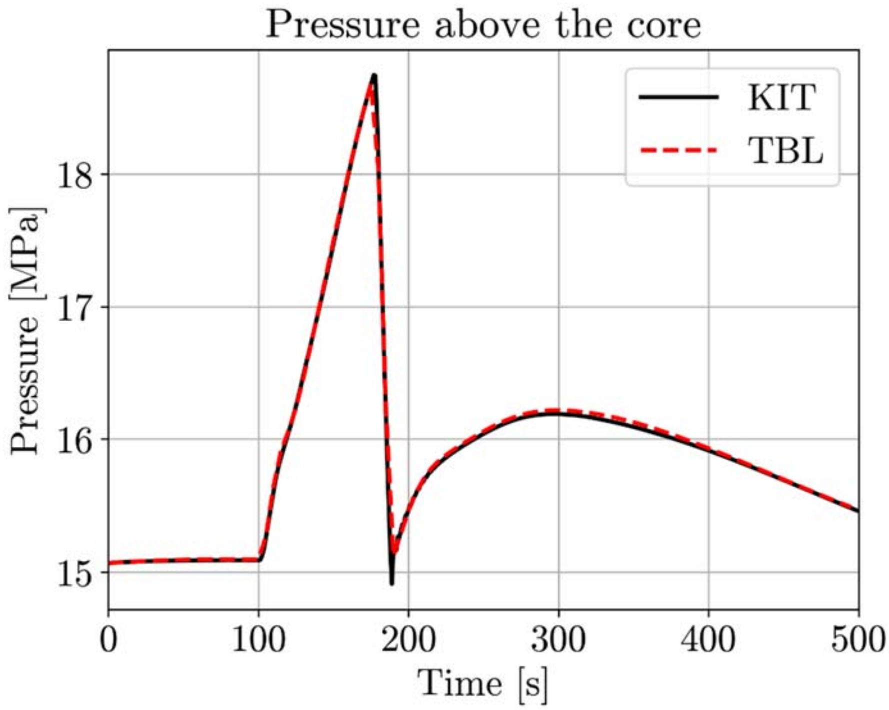

Figure 8. Pressure above the core.

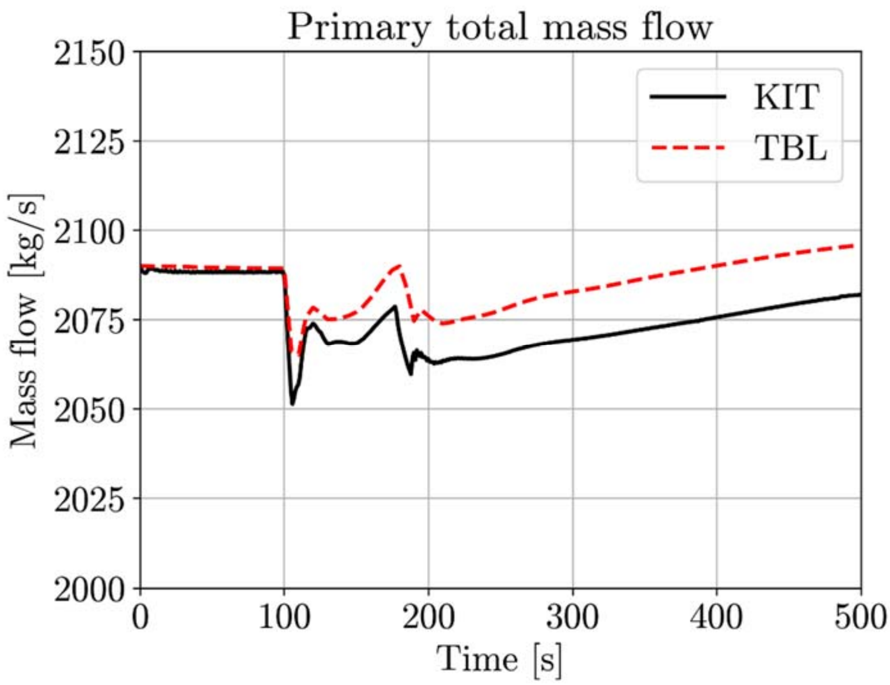

Figure 9. Primary total mass flow.

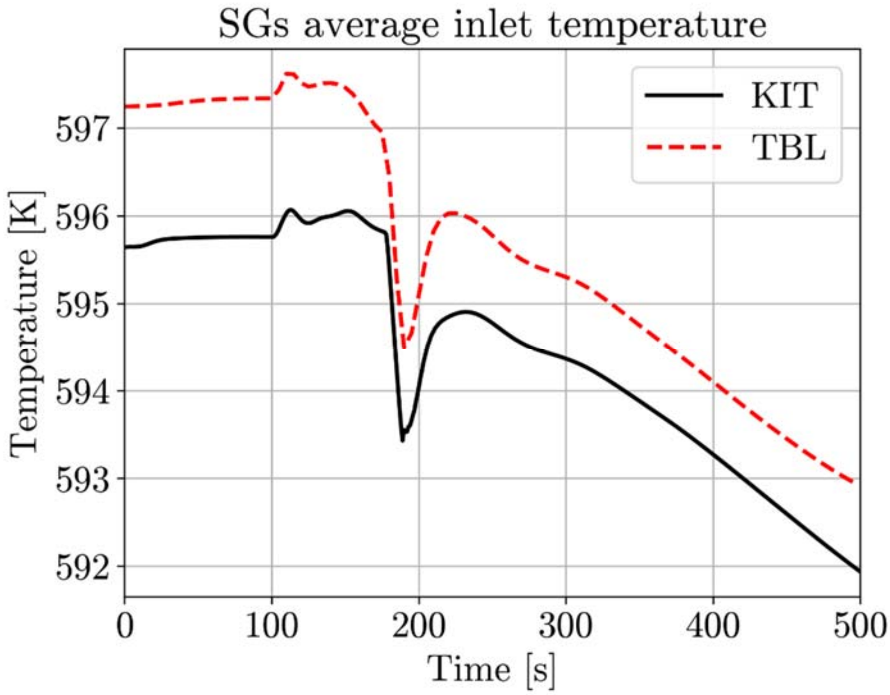

Figure 10. Steam-generator (SG) average inlet temperature.

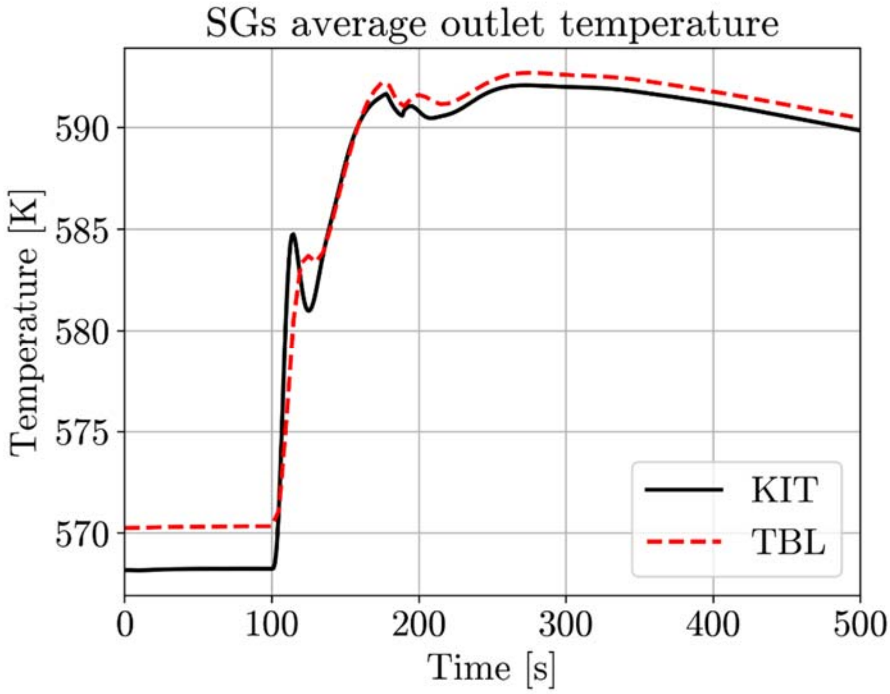

Figure 11. Steam-generator (SG) average outlet temperature.

7. Conclusions

ATWS calculations have been performed for the SMART reactor using the TRACE system code with 1D and 3D flow models for the RPV. The results obtained are physically consistent and agree with results published in the open literature. Furthermore, no significant differences have been observed between the two models used, mainly because the accidental scenario is symmetrical and therefore the potential improvements in the flow solution using the 3D approach do not have a substantial impact.

8. References

[1] EU Horizon 2020, grant agreement 945063. “McSAFER - High-Performance Advanced Methods and Experimental Investigations for the Safety Evaluation of Generic Small Modular Reactors”. url: http://cordis.europa.eu/project/id/945063.

[2] KAERI. "Status report 77 - System-Integrated Modular Advanced Reactor (SMART)”. 2011.

[3] J. Yoon, S.Y. Ryu, B.S. Choi, M. Lee and S.K. Zee. “Plant System Design Features of SMART”. GENES4/ANP2003, Kyoto, Japan, 15-19 September, 2003.

[4] K. Langdon. “NuScale Small Modular Reactor (SMR) Overview”. INPRO Dialogue Forum on Opportunities and Challenges in Small Modular Reactors, Ulsan, Republic of Korea, July 02-05, 2019.

[5] IAEA. “Advances in small modular reactor technology developments”. Supplement to IAEA Advanced Reactors Information System (ARIS), pages 35-38, 2018.

[6] Y.J. Chung, H.R. Kim, J.H. Chun, S.H. Kim and K.H. Bae. “Strength assessment of SMART design against anticipated transient without scram”. Progress in Nuclear Energy 85, pages 617-623, 2015.

[7] S. Kliem, M. Garcia, C. Queral, M. Bencik and V. Sanchez-Espinoza. “McSAFER - Milestone MS07: Selection of accident scenarios for WP4”.

[8] S.H. Yang, Y.J. Chung, H.C. Kim and S.Q. Zee. “Performance evaluation of an advanced integral reactor against an anticipated transient without scram”. Annals of Nuclear Energy 33, pages 655-663, 2006.

[9] V. Sanchez-Espinoza, V.M. Marin, Y. Alzaben, G. Jimenez and R. Stieglitz. “Integral Smart Plant Model Developed for the System Thermal Hydraulic Code Trace for Transient Analysis”. NUTHOS-12, Qingdao, China, October, 2018.

[10] TRACE V5.0 Theory Manual, User Manual and Assessment Manual. 2007.