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International Agreement Report

Development of a Coupled TRACE/PARCS Model for KKL and Benchmark Against the Turbine Trip Test

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ABSTRACT

The Nuclear Power Plant of Leibstadt (KKL) is a participating member of the Code Applications and Maintenance Program (CAMP) of the U.S. Nuclear Regulatory Commission (USNRC) to validate the TRACE code for BWR/6 transient analysis. The application of TRACE for the safety assessment of BWRs requires a throughout verification and validation using experimental data from tests but also plant data for the modelling. The purpose of this work is the review of the KKL TRACE/PARCS model, the benchmark of the model against plant data recorded during a turbine trip test and an investigation of the core lumping effect on the turbine trip test.

A coupled TRACE/PARCS model has been developed to analyze fast transients in KKL. The first benchmark against a turbine trip test has shown differences between the test data and the results predicted by TRACE/PARCS such as the total core power and the dome pressure. This is mainly due to unstable steady-state conditions during the initialization process and modelling issues. The improvements introduced in this work to the TRACE model are including but not limited to the geometry of the reactor internals, the redesign of the main steam lines and the implementation of a rudimentary control system. Furthermore, the PARCS input model has been updated with the turbine trip test corresponding cross sections. The new designed coupled TRACE/PARCS model was eventually benchmarked against the same turbine trip plant data.

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A coupled TRACE/PARCS model has been developed to analyze fast transients in KKL. The first benchmark against a turbine trip test has shown differences between the test data and the results predicted by TRACE/PARCS such as the total core power and the dome pressure. This is mainly due to unstable steady-state conditions during the initialization process and modelling issues. The improvements introduced in this work to the TRACE model are including but not limited to the geometry of the reactor internals, the redesign of the main steam lines and the implementation of a rudimentary control system. Furthermore, the PARCS input model has been updated with the turbine trip test corresponding cross sections. The new designed coupled TRACE/PARCS model was eventually benchmarked against the same turbine trip plant data.

The comparison between the results of the TRACE/PARCS simulation and the recorded data of the turbine trip test has shown a very good agreement. The TRACE/PARCS model of KKL could reproduce the turbine trip sequence with a very good accuracy. Especially the power oscillations, indicating instable conditions in the core, could be eliminated. The only big discrepancies left result from the valve modeling in TRACE. The non-linear behavior of the closing and opening could not be reproduced so far and would need more investigation in the valve design.

Furthermore, the effect of the core lumping and mapping has been studied on the same turbine trip test. The same boundary conditions were used and revealed only small differences which mainly result from the different core pressure drop. As long as spatial radial effects are not of major importance, the lumped core can be used for a first estimate of the output.

In general, the new coupled TRACE/PARCS model performs very well. There are though, as the outlook states, further investigations needed in the design of the recirculation lines, including the jet pumps, the water level measurement and the valve design.

ACKNOWLEDGMENTS

The authors acknowledge the support for the PARCS input data from the Universitat Politécnica de València, namely P. Hidalga, R. Miro, T. Barrachina, P. Hidalga and G. Verdú. Special thanks go also to F. Giust from Axpo Power AG who helped preparing the CASMO input deck for the specific cycle of the turbine trip for the further process into the PARCS NEMTAB files.

ABBREVIATIONS

BPV	bypass valve
BWR	boiling water reactor
CPU	central processing unit
CSS	constrained steady-state
ECCS	emergency core cooling system
FCV	flow control valve
HPCS	high-pressure core spray
KKL	nuclear power plant Leibstadt (in German: Kernkraftwerk Leibstadt)
LOCA	loss of coolant accident
MSL	main steam line
MSIV	main steam isolation valve
NK	neutron kinetics
NPP	nuclear power plant
PARCS	Purdue Advanced Reactor Core Simulator
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RLS	reload licensing
RPV	reactor pressure vessel
SNAP	Symbolic Nuclear Analysis Package
SRI	selected rod insertion
SRV	safety relief valve
TCV	turbine control valve
TH	thermal-hydraulic
TT	turbine trip
TRACE	TRAC/RELAP Advanced Computational Engine

1 INTRODUCTION

System codes are widely used in nowadays nuclear industry for the evaluation of occurrences and incidents. The transient analysis with simulation tools is a very important subject as real experiments are very expensive and mostly not possible to perform without damaging the infrastructure and the integrity of the power plant.

The TRAC/RELAP Advanced Computational Engine (TRACE) is a modernized thermal-hydraulics system code designed to merge the capabilities of the legacy codes TRAC-P, TRAC-B and RELAP. The code is used to analyze different system transients and accident scenarios in pressurized (PWR) and boiling water reactors (BWR). TRACE has the capability to model thermal-hydraulic phenomena in the one- and three-dimensional space. The Purdue Advanced Reactor Core Simulator (PARCS) on the other hand is a computer code that solves the time-dependent two-group neutron diffusion equation in three-dimensional Cartesian geometry. The code is more commonly used in the analysis of reactivity-initiated accidents in light water reactors where spatial effects play an important role. Both system codes may be run in stand-alone or in coupled mode depending on the type of transient that is being investigated.

In coupled mode, they allow the user to design a model of a nuclear reactor and take all thermal-hydraulic and neutronic effects into account. The setup of such a model for the safety analysis is a highly time consuming process that is driven by an iterative procedure between modelling and validation. Each modification that is applied on an existing coupled model needs verification with multiple experimental data sets if possible. The result of these efforts is a multi-disciplinary model that is able to predict design and beyond-design transients. The analysis and study of those is an important part of the reload licensing. Despite the time and efforts that a numerically robust model needs, different accident scenarios have been studied with the coupled system code TRACE/PARCS and been published already all over the world. Depending on the type of accident, either the thermal-hydraulic or the neutronics part is investigated more thoroughly.

TRACE has for example been used to analyze the spurious opening of the valves of the automatic depressurization system (ADS) in a BWR/6 [1]. The safety relief valves which are responsible for the ADS, accidently opened during normal operation in a BWR/6. The recorded plant data is compared with a TRAC-BF1 and a TRACE simulation. Both models were able to reproduce the plant data of the incident. For certain main characteristic parameters like the dome pressure, the TRACE code was even better performing than the TRAC-BF1 code. In fact, «the TRACE code can satisfactorily predict the system behavior during the ADS transient in a BWR/6» [1].

Beside pure thermal-hydraulic scenarios, TRACE has also been coupled to study reactivity induced occurrences. In [2], a control rod withdrawal transient is simulated with TRACE/PARCS with a simplified model of the Cofrentes nuclear power plant core, also BWR/6. The model is validated with a TRAC-BF1/VALKIN and SIMULATE model of the same core geometry. As the results point out, this fast transient analysis is strongly dependent on the numerical solver and the time step size. However, the TRACE/PACRS model managed to show the same steady-state behavior as the other system codes.

These examples give a small overview of the capabilities of the system codes TRACE and PARCS. Since U.S. NRC declared TRACE to be their flagship thermal-hydraulic code, the Leibstadt Nuclear Power Plant (KKL) started with the design of a TRACE/PARCS model and the investigation of different design basis transients. The resulting models of this development are used to review and reproduce the outcome of the transient analysis of other institutes. The long term aim is the ability to conduct own sensitivity studies on different transients to estimate most conservative and limiting state conditions of the reactor design.

2 PLANT DESCRIPTION

KKL is a GE design BWR/6 with a Mark III containment. The plant has been in commercial operation since 1984. The initial power rating of KKL was 3012 MW_{th} which was increased in several power uprates and modernization steps to a nuclear thermal power of 3600 MW_{th}. The reactor coolant system (RCS) is arranged by two recirculation loops, each of them containing one recirculation pump, feeding in total 20 internal jet pumps inside the RPV. The reactor core itself is composed of 648 fuel elements and 146 control rods. The water level in the downcomer is measured with two different systems, the narrow (NR) and the wide range (WR) instrumentation. After the steam separator and steam dryer stage, four main steam lines lead the way to the high pressure turbine and the three low pressure turbines. At the exit of the low pressure turbine, the steam is condensed to its initial state. The heat sink is provided by the nearby river water which absorbs the heat in a passive cooling tower.

3 TRACE/PARCS INPUT MODEL DESCRIPTION

The development of a thermal-hydraulic model for KKL started with the design of a TRACG model for more sophisticated transient analysis. The model was later on converted into a TRAC-BF1 input model [3] to make the comparison of the results with other facilities easier since the TRAC-BF1 license was publicly available through the code application and maintenance program (CAMP) agreement. The TRAC-BF1 input model, on contrary to the TRACG input, could also be used for the publication of benchmarks against plant data of existing occurrences. With the release of TRACE V5.0 ([8 - 9]), the thermal-hydraulic model was once again adapted into the new input syntax. The two models were both benchmarked against KKL plant data of the event “Inadvertent Opening of ADS Valves” [4]. The results lead to the conclusion to focus the further development on the TRACE input deck rather than the TRAC-BF1 model, also because U.S. NRC declared TRACE as their future flagship in terms of thermal-hydraulic system codes.

The original nodalization and geometry of the RPV is depicted in Figure 1. The geometry was not changed, neither during the first conversion step from TRACG into TRAC-BF1 nor from TRAC-BF1 into the new TRACE model. The scheme shows 15 axial levels and four radial nodes, corresponding to three inner shroud rings and one downcomer.

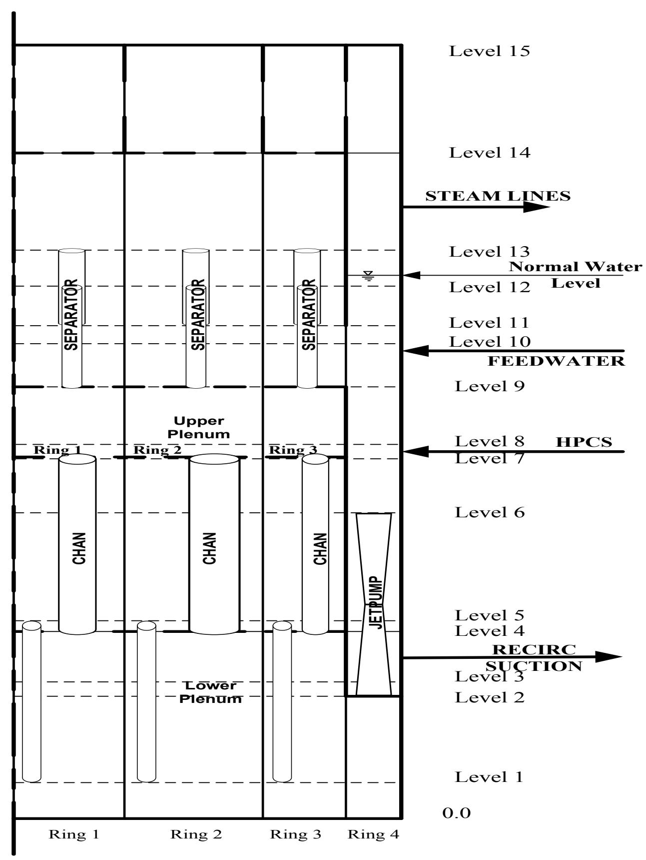

Figure 1 Original TRAC-BF1 nodalization scheme

Each of the inner core rings holds one lumped PIPE component for the guide tubes, one lumped CHAN component for the fuel elements and one lumped SEPD component for the steam separators. The connections for the two recirculation lines, the HPCS, the feedwater line and the steam lines are also shown at the proper axial level. The design of the piping from and to the RPV is illustrated with a SNAP [7] screenshot in Figure 2. The feedwater pump is represented by a FILL component as are the RCIC and HPCS safety systems. The main steam lines are split into two different pipes. The cross section ratio of this two steam lines is three to one. The division into two different sized cross section areas gave certain modularity for the transient simulations. With this configuration, the failure of one, three or all four main steam pipes or valves could be investigated.

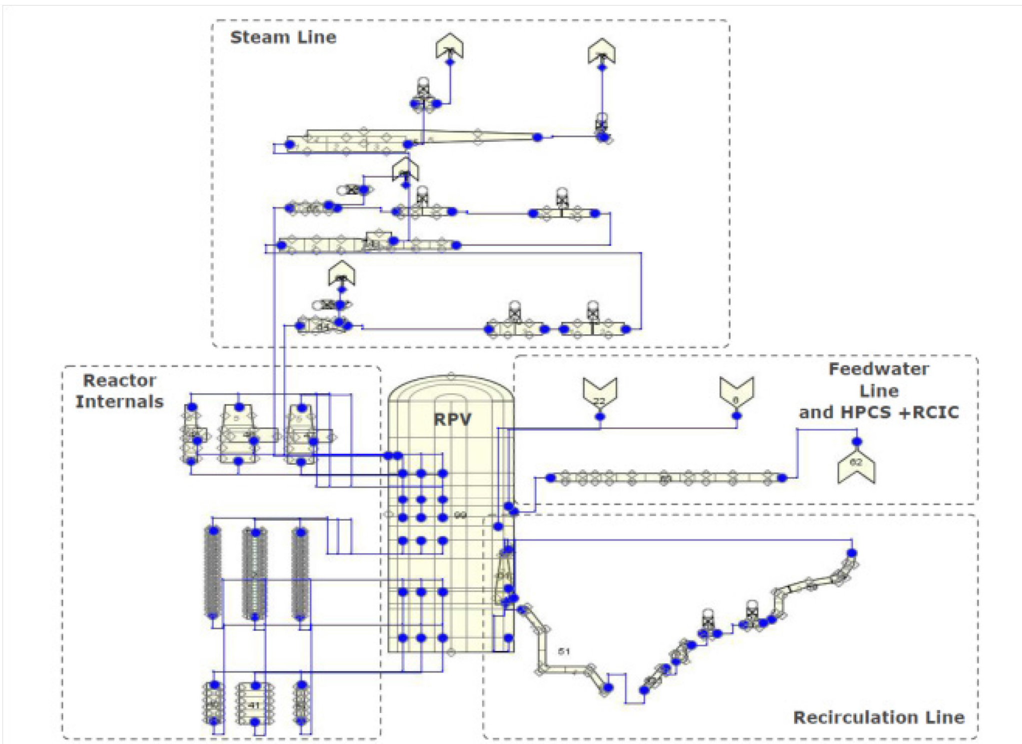

Figure 2 Initial KKL TRACE model converted from TRAC-BF1 – SNAP [7] hydraulics component view

In a further step, this KKL TRACE model was coupled for the first time to a generic core configuration to test the neutronic feedback during a turbine trip test [5]. The conclusion stated a lot of space for improvements in both, the thermal-hydraulic and the neutronic input deck.

3.1 <u>Geometry Review of the TRACE Model</u>

The revision of the input deck started with the TRACE part of the input. Plant documentation was accumulated to get all the updated information of the reactor vessel, the reactor core and internals. These data was condensed and inserted in the KKL TRACE input model. The only components that remained unchanged at the end were the two recirculation pumps. In Figure 3, the evolution of the free volume inside the vessel is displayed as it was also presented in [6]. From the fourth axial level onwards, the TRACE model is no longer over predicting the free volume. Especially for LOCA analysis, the correct free volume of the RPV will be essential.

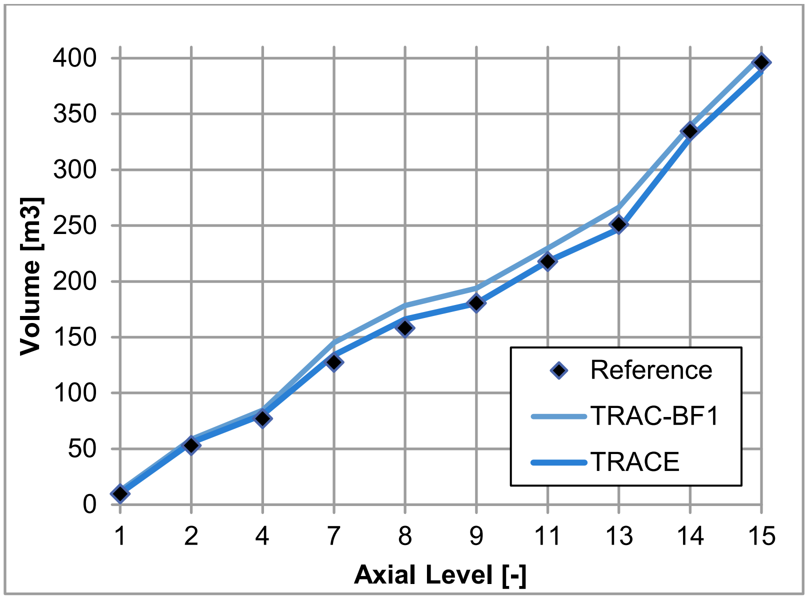

Figure 3 Free volume inside the RPV [6]

The geometries of the reactor internals underwent also some modifications. As displayed in Figure 4, except for the jet pumps, all other values could be improved. Especially the error in the core bypass was reduced from over 50% to less than 10%. The correct size of the bypass is of high importance for the heat losses from the fuel elements and the neutronic coupling, e.g. the temperature correction of the moderator. The redesigned jet pumps lost in volumetric but gained in geometric accuracy of the contraction ratio. The guide tubes are still overrated due to the different lengths of the tubes. The tubes are mounted at the bottom of the reactor pressure vessel, thus their length depends on their radial position. This was not taken into account due to the small influence of the guide tubes in the whole system dynamics.

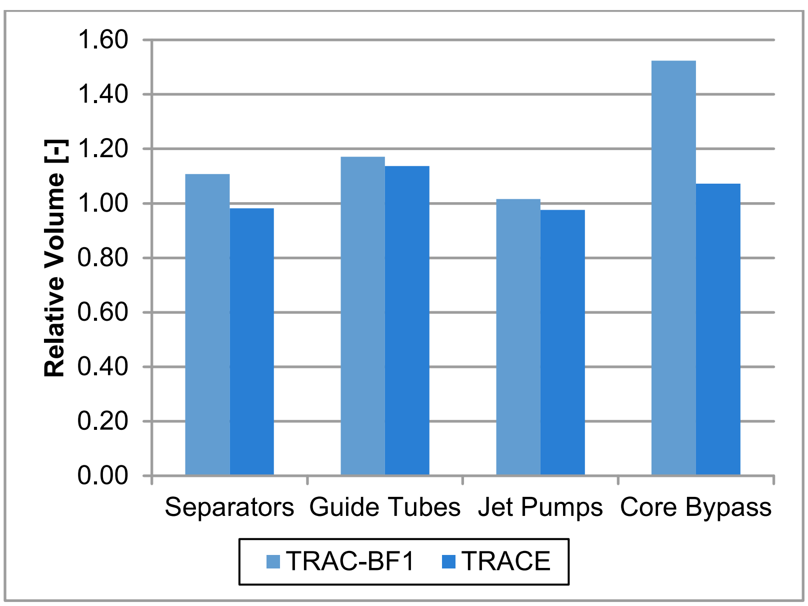

Figure 4 Volume inside the reactor internals

Despite the RPV and the reactor internals, one of the biggest modifications was done at the main steam lines. The lumped steam line models were replaced by four appropriate full length sized steam lines according to the plant documentation. Also the amount of valves was adjusted on each main steam line to the proper number, e.g. the S/R valves or MSIVs. In Figure 5, a SNAP screenshot is depicted of the main steam lines, including the position of the S/R valves, the bypass connection and the four turbine control valves.

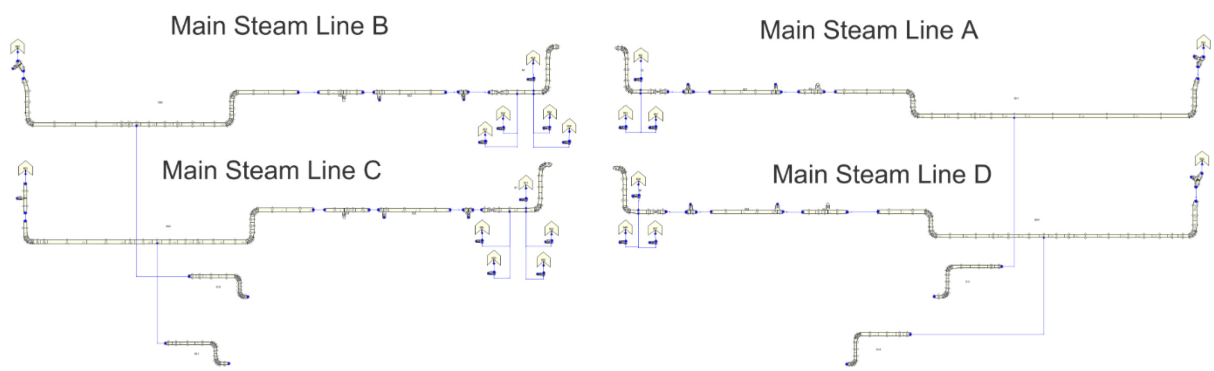

Figure 5 Main steam line geometry – SNAP hydraulics component view

The bypass lines lead to a manifold, the main steam header, as illustrated in Figure 6. This steam header supplies all the auxiliary steam demands of the power plant out of which two are of big importance for the normal operation. There is the connection to the reheater, which is running during normal operation, and the four bypass lines. These lines are used to reject the heat directly to the condenser and thereby bypassing the turbines. These lines were all introduced for the first time to the TRACE model of KKL.

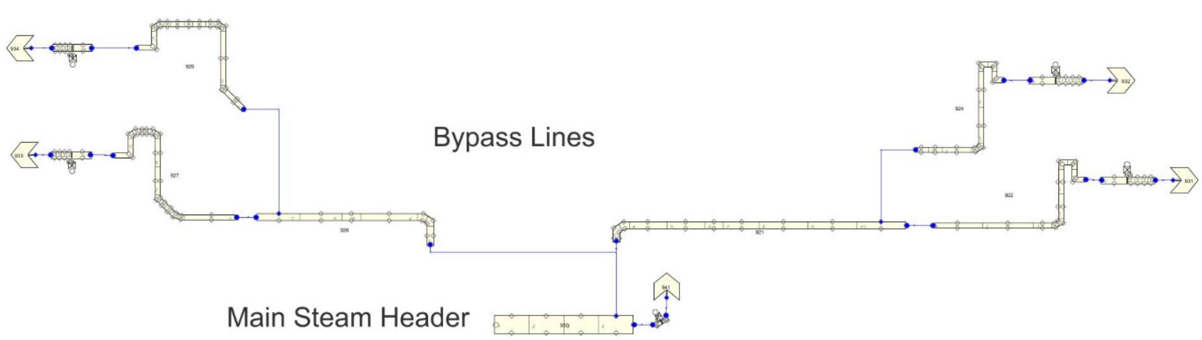

Figure 6 Bypass line geometry – SNAP hydraulics component view

The adjustments on the geometry of the input model were not enough for a converging steady-state calculation. The design of the controls of the plant model needed some additional attention.

3.2 <u>Control Design</u>

The plant balance was eventually achieved using two main control systems to keep the dome pressure and the water level stable in the TRACE calculation. The feedwater controller turned out to be one of the crucial controllers for the stability at the water-steam phase boundary. The pressure in the dome was contributing a lot on the flow stability inside the main steam lines.

3.2.1 Feedwater Controller

The estimation and control of the water level is one of the most challenging tasks inside a BWR model. The level was determined with the hydrostatic pressure equation (1), using the parameters in Figure 7 for the calculation.

$$\Delta p = \rho_g g(h_2 - h_l) + \rho_l g(h_l - h_1) \tag{1}$$

Beside the pressure difference over the phase boundary, the water and the vapor density (ρ_g and ρ_l) are needed. Two PIPE components were therefore attached to the RPV at two different heights. One pipe is set above, the other one below the regular water level.

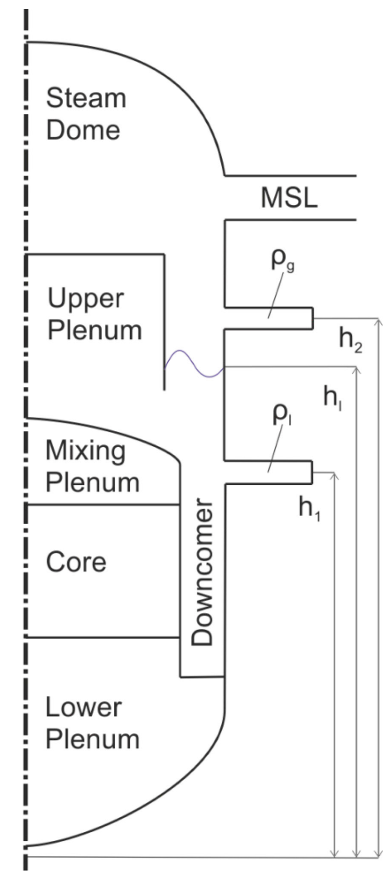

Figure 7 Level measurement instrumentation

The downcomer level was further on used as an input for the TRACE level controller (ICBN = 202, [9]) along with the feedwater flow, the main steam flow and the desired water level. The output of this controller is the new feedwater flow rate which serves as an input for the feedwater FILL component.

3.2.2 Implementation of a Pressure CSS

The first steady-state calculations showed a very sensitive behavior of the steam flow to power oscillations. This sensitivity was a result of the fixed turbine pressure at the end of the main steam lines. Steam flow waves were propagating through the steam lines back and forward, thus heavily influencing the dome pressure. The implementation of a pressure controller at the turbine control valve was able to dampen these wave oscillations and to help keeping the dome pressure at a fixed value.

The pressure regulators for the turbine control valves were introduced with the constrained steady-state (CSS) option. The CSS calculation runs the model with the given initial conditions until all significant parameters reach the desired and initially set convergence criterion, e.g. 1E-4. It also adds a stated number of controllers, such as pressure or velocity controllers for valve components which are only active during the steady-state calculation. Starting a transient simulation leads to the deactivation of all CSS controllers. The other possibility would be the implementation of the built-in pressure controller (ICBN = 20 [9]). Unfortunately, several attemps did not perform as described in [9], which probably comes from a programming error.

3.3 PARCS Neutronics Data

The results from [5] were obtained with a generic core configuration. The turbine trip test though was performed during cycle 18. In order to reproduce the core behavior properly, an update of the cross section data was necessary. The cross section input was generated by Axpo Power AG in CASMO-4. The PARCS input deck was created by the Institute for Industrial, Radiophysical and Environmental Safety (ISIRYM) of the Universitat Politècnica de València (UPV) with the SIMTAB methodology [11]. The CASMO files were used in SIMULATE3 to determine the neutron kinetics. In a second step, the SIMULATE3 [12] output was transformed into PACRS NEMTAB files.

The comparison of the k_{eff} between the SIMULATE3 and PARCS in Table 1 and Figure 8 and the axial power profile showed small deviations between the two codes. The root mean square error for the total axial power is 1.03%.

Table 1 k_{eff} comparison between SIMULATE3 and PARCS

System Code	k_{eff} [-]	Difference [pcm]
SIMULATE3	1.00519	-
PARCS v3.1	1.00251	267.7

The comparison of the radial power profile of the two different codes showed a good agreement for a wide range of the core. However, the relative error at the periphery and the rodded section increased to above 5%. This effect is still under investigation at the Universitat Politècnica de València. Nonetheless, the cross section data were good enough for the use in the KKL TRACE/PARCS model development.

As mentioned in the plant description, the reactor core has a total number of 146 control rods. The assignment of the control rods are depicted in Figure 9 in a SNAP scheme. This control rod pattern was used during the turbine trip test in cycle 18.

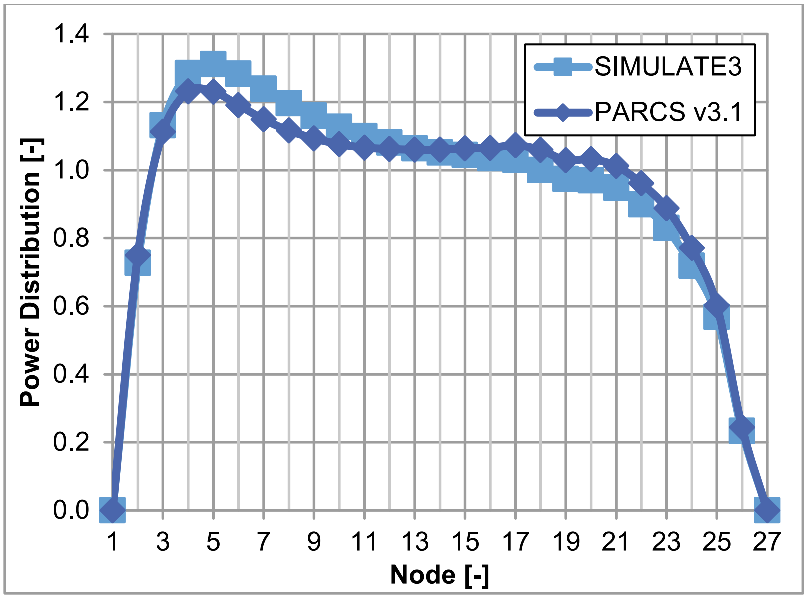

Figure 8 Axial power distribution in SIMULATE3 and PARCS

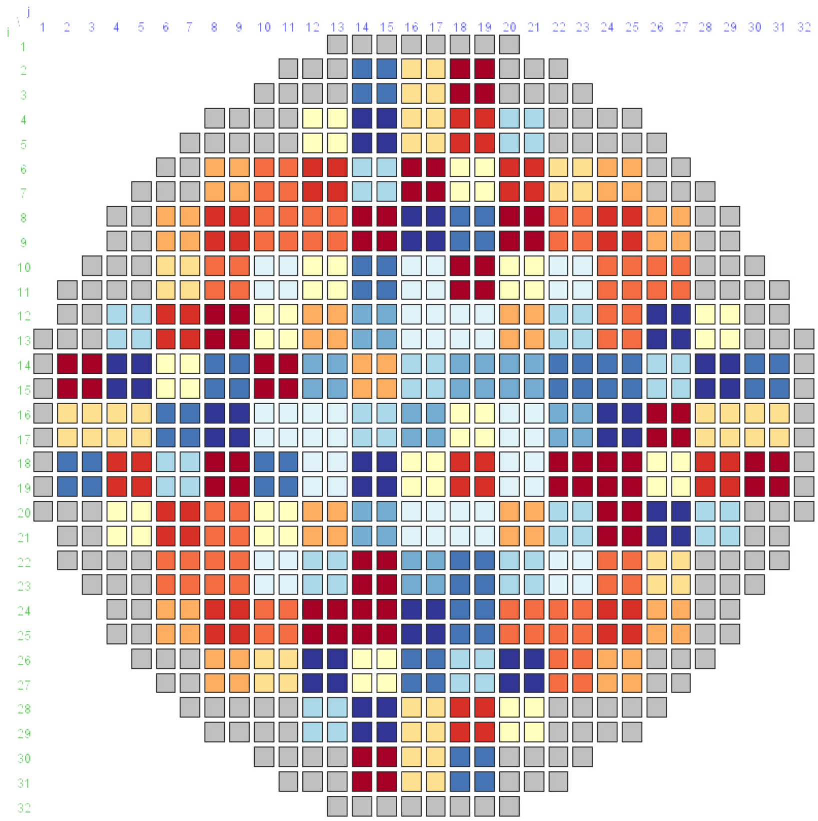

Figure 9 Control rod pattern in SNAP

3.4 Final TRACE/PARCS Model

With the modifications on the TRACE geometry and the update on the PARCS cross section input deck done, the KKL TRACE/PARCS model was ready for the benchmark against the turbine trip test. Figure 10 shows the new scheme for the KKL TRACE model in SNAP. The four radial rings of the RPV were collapsed to one inner core and one downcomer segment. As a consequence, also the separators and the guide tubes were condensed to one component. The lumping showed no significant influence on the system dynamics since no radial effects were of interest for the fast transient analysis. The biggest improvements were as described before the four proper main steam lines.

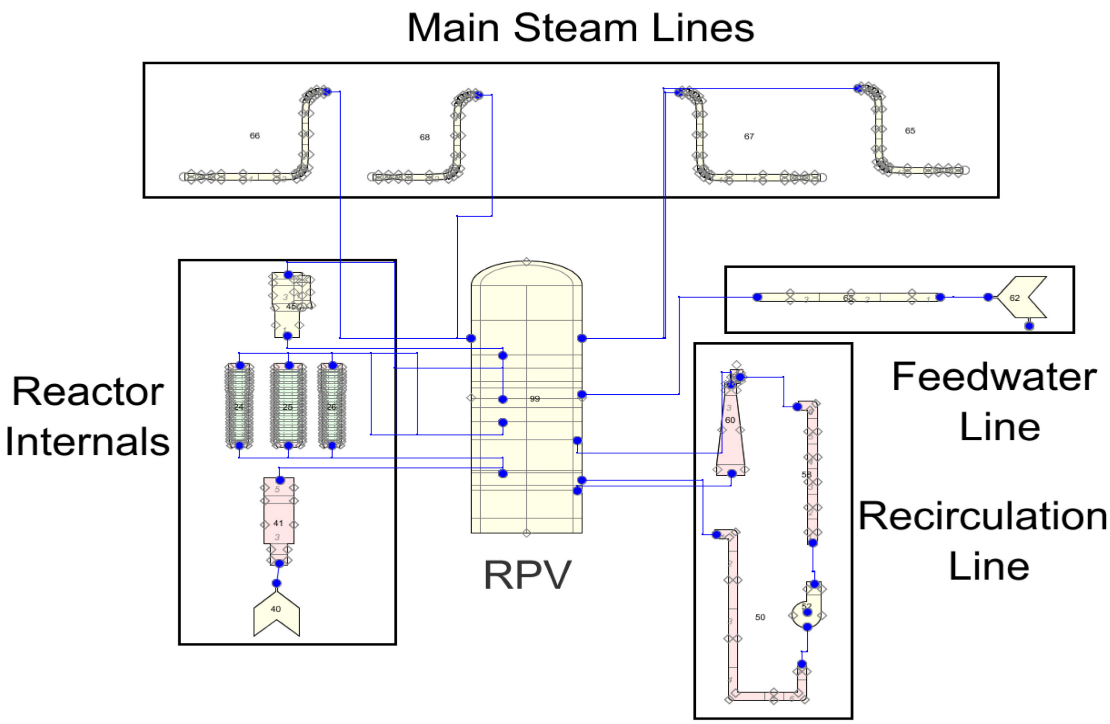

Figure 10 KKL TRACE model with modified nodalization scheme – SNAP hydraulics component view

The reactor core is divided in three parts. One outer peripheral ring, an inner hot core and the intermediate zone are holding 648 fuel elements. A second model was designed with 648 CHAN components where each CHAN represented one single fuel element. Thus, the effect of a lumped core during a fast transient could be investigated.

0	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
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22	*	0	0	26	25	25	25	25	25	25	25	25	25	25	25	25	25	25	25	25	25	25	25	25	25	25	25	25	26	0	0	*
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32	*	*	*	*	*	*	*	*	*	*	*	*	0	0	0	0	0	0	0	0	*	*	*	*	*	*	*	*	*	*	*	*

Figure 11 Core Mapping between TRACE and PARCS

4 BENCHMARK AGAINST TURBINE TRIP PLANT DATA

The new model needed as a first step to be validated against plant data. The turbine trip test is a fast transient during which all reactivity removal mechanisms are acting, i.e. selected control rod insertion and core flow reduction. The transient is also acting in the high pressure region where all turbine control and bypass valves need to be activated to stabilize the reactor pressure. The turbine trip test during cycle 18 was therefore chosen due to the diverse reactor protection measures to validate the modified KKL TRACE/PARCS model.

4.1 Steady-State Calculations

In order to start the transient analysis, the TRACE/PARCS model needs first to run two steady-state calculations. In the first calculation, the TRACE model runs with no coupling to PARCS to find the steady-state conditions of the initialized model. This independent steady-state calculation speeds up the process since there is no neutronic feedback during this unstable initial state. The converged TRACE input is coupled in a second step to the PARCS input for another steady-state calculation. The stable outcome of this run holds the initial conditions prior to the transient. As mentioned before, two different models were designed for the TT test benchmark.

The quality of the lumped model design can be seen in Table 2. The deviations of the different characteristic parameters between the plant data and the TRACE/PARCS calculation are less than 1%. The water level, feedwater flow rate, the dome pressure and the steam flow rate are of particular interest. These four values are controlled by the level controller and the CSS respectively while the other values are basically fixed by the initial conditions like the recirculation flow rate and the core power or a result of the steady-state calculation, i.e. the core flow rate.

Table 2 Initial conditions of the lumped core model before the turbine trip test

	Plant Data	TRACE/PARCS	Relative Error [%]
Core Power [MW]	3557.63	3540.03	-0.495
Core Flow Rate [kg/s]	10170.88	10239.54	0.675
Dome Pressure [bar]	72.96	72.92	-0.055
Recirculation Flow Rate [kg/s]	3370.85	3391.46	-0.611
Steam Flow Rate [kg/s]	1955.02	1955.57	0.028
Feedwater Flow Rate [kg/s]	1952.99	1951.85	-0.058
Relative Water Level [-]	1.00	0.999	-0.007

The results of the full core model look slightly different. Once more, the only change in this model is the replacement of the three lumped channels with 648 channel components. Most of the values in Table 3 are still in the same order of magnitude as before except for the core flow rate. The relative error for the recirculation flow rate did also slightly increase. The core flow rate is determined by the two jet pump models that represent all 20 internal jet pumps. The sum of these flow rates corresponds to the total core flow.

Table 3 Initial conditions of the full core model before the turbine trip test

	Plant Data	TRACE/PARCS	Relative Error [%]
Core Power [MW]	3557.63	3540.34	-0.486
Core Flow Rate [kg/s]	10170.88	10450.13	2.746
Dome Pressure [bar]	72.96	73.07	0.151
Recirculation Flow Rate [kg/s]	3370.85	3400.54	0.881
Steam Flow Rate [kg/s]	1955.02	1957.06	0.104
Feedwater Flow Rate [kg/s]	1952.99	1953.51	0.027
Relative Water Level [-]	1.00	0.999	-0.007

According to the results in Table 2 and Table 3, there is a clear difference between the two core flow rates which cannot be explained only by the changed recirculation flow rates. As mentioned, the core flow rate is estimated by the jet pump flow rates. The jet pumps are on the one hand driven by the forced flow from the recirculation flow rate but also by the pressure drop alongside the downcomer. Since all initial conditions are set exactly the same for both models, the effect has to come from the different modeling of the core. In Table 4, the pressure drop between the mixing plenum and the lower plenum and the absolute pressure in the upper plenum is shown.

Table 4 Pressure drop inside the RPV

	Lumped Core	Full Core	Difference [bar]
Lower Plenum – Mixing Plenum Pressure Drop [bar]	1.38	1.25	0.13
Upper Plenum Pressure [bar]	72.92	73.07	0.15

The full core model has a clear influence in the pressure distribution in the core region. The core flow rate is established for each channel individually while the lumped core distributes the coolant among three channels. The pressure is hence also affected in the upper plenum and as a consequence in the downcomer. This pressure difference will influence the flow conditions in the jet pumps. The full core model has therefore already shown an effect on the results of the transient analysis.

4.2 Sequence of Event

As mentioned before, the event is followed by multiple system measures. In Table 5, the four events are chronologically summarized and compared with the TRACE/PARCS trip initiation. The turbine control valves (TCVs) start the fast closing 200 ms after the initiation of the transient. As a consequence, the bypass valves (BPVs) open with a delay of 40 ms to reduce the pressure in the RPV. The rod insertion is initiated another 60 ms later to compensate the added reactivity due to the pressure induced void collapse in the core. The second measure is activated 10 ms after the rod insertion which is called recirculation runback. The flow control valves of the recirculation line are closed to a predefined position to reduce the core flow and thus also reduce the reactivity in the core.

Table 5 Sequence of event for the turbine trip test

Event	Initiation after Trip	TRACE/PARCS
Turbine Control Valve Closing	0.20	0.20
Bypass Valve Opening	0.24	0.24
Selected Rod Insertion	0.30	0.30
Recirculation Runback	0.31	0.20

The delay of the recirculation runback is the only quantity that does not match with the test protocol which will be discussed in the results section. The closing sequence of the flow control valve illustrated in Figure 12 starts as described in Table 5 100 ms earlier than the reference and reaches the final state accordingly faster. The offset of the valve opening ratio at the beginning and the end comes from the modeling of the valve. The flow areas of all valves are designed in a way to agree with the properties like pressure or mass flow. The offset has only little effect on the behavior of the system as long as the closing times are matching.

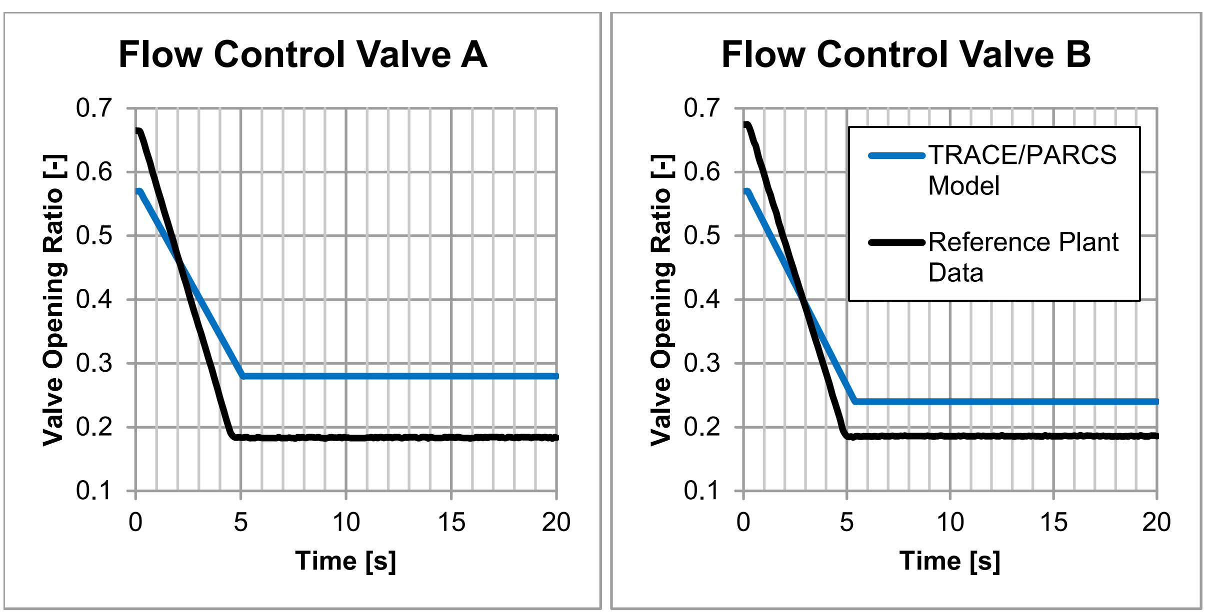

Figure 12 Closing sequence of the flow control valves

The opening ratio of the TCVs is as described earlier estimated by the CSS controller. The design of the valves was analyzed more thoroughly which made it possible to meet the opening ratio with the plant data during steady-state. As a result, the flow area did not correspond to the real geometrical TCV area but from a modeling point of view, the closing sequence could be represented very well except for the final state. The valve signal in the closed state always has an offset by the design of the measurement technique. Even for the TCV A which seems to be stuck open, the TCV indicator in the control room showed a closed position. Besides the big offset of the TCV A, there is also a small deviation of the TCV D at 0.3 seconds with an insignificant effect on the system response.

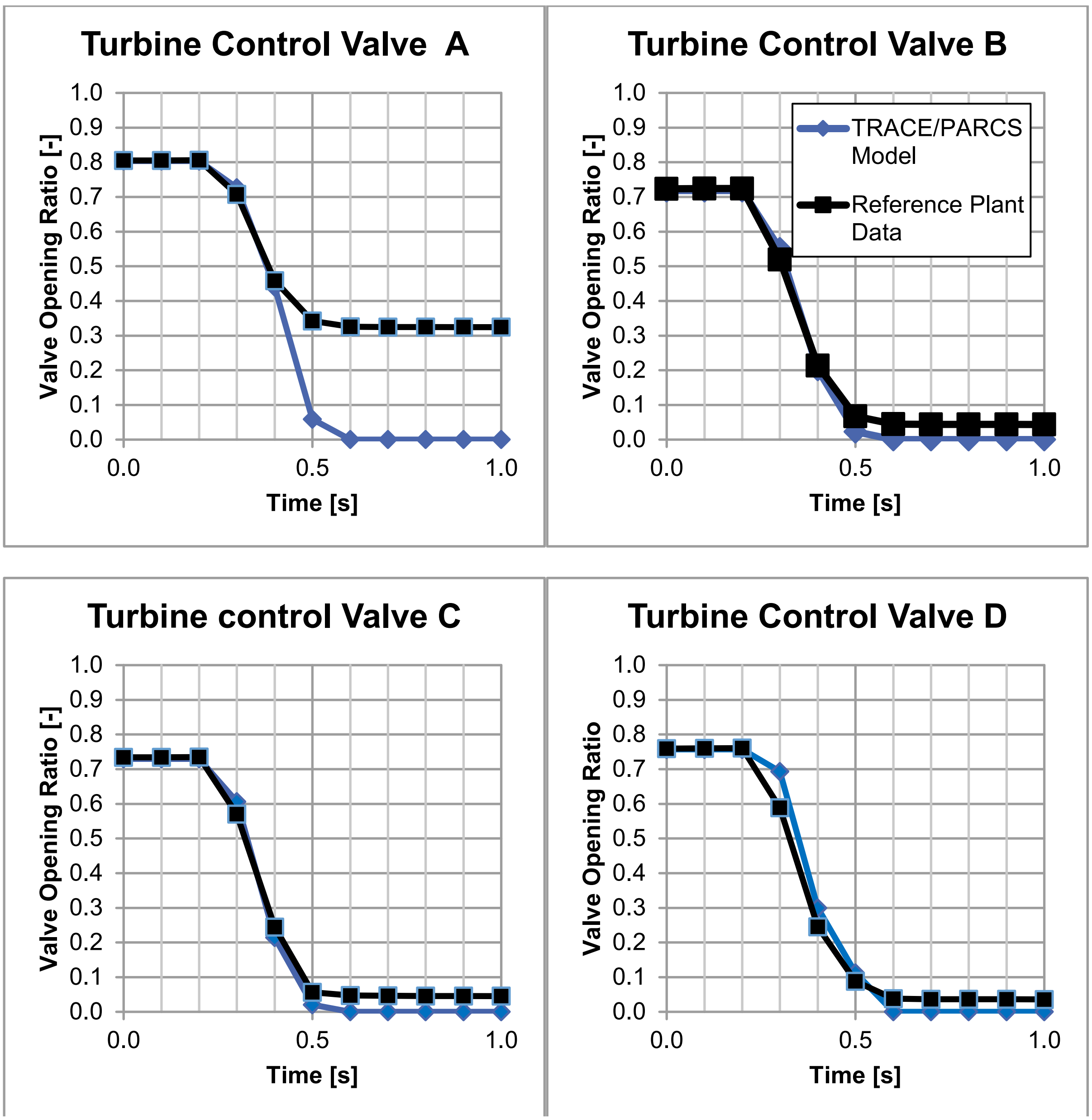

Figure 13 Closing sequence of the turbine control valves

The first action after the closing of the TCVs is the opening of the bypass valves. In the real plant, the bypass is regulated by a pressure controller. The attempt to model a pressure controller with the built-in version in TRACE unfortunately failed. The controller does not respond at all which probably comes from a programming error. Therefore, the signal was set manually with the help of the reference data. After several iterations, a stable configuration was found. In Figure 14, the valve opening ratio of the BPVs is pictured. The sequence maxima and minima were adjusted to be synchronous with the plant data. The value of the valve opening ratio though varied due to the same reason as the flow control valve opening.

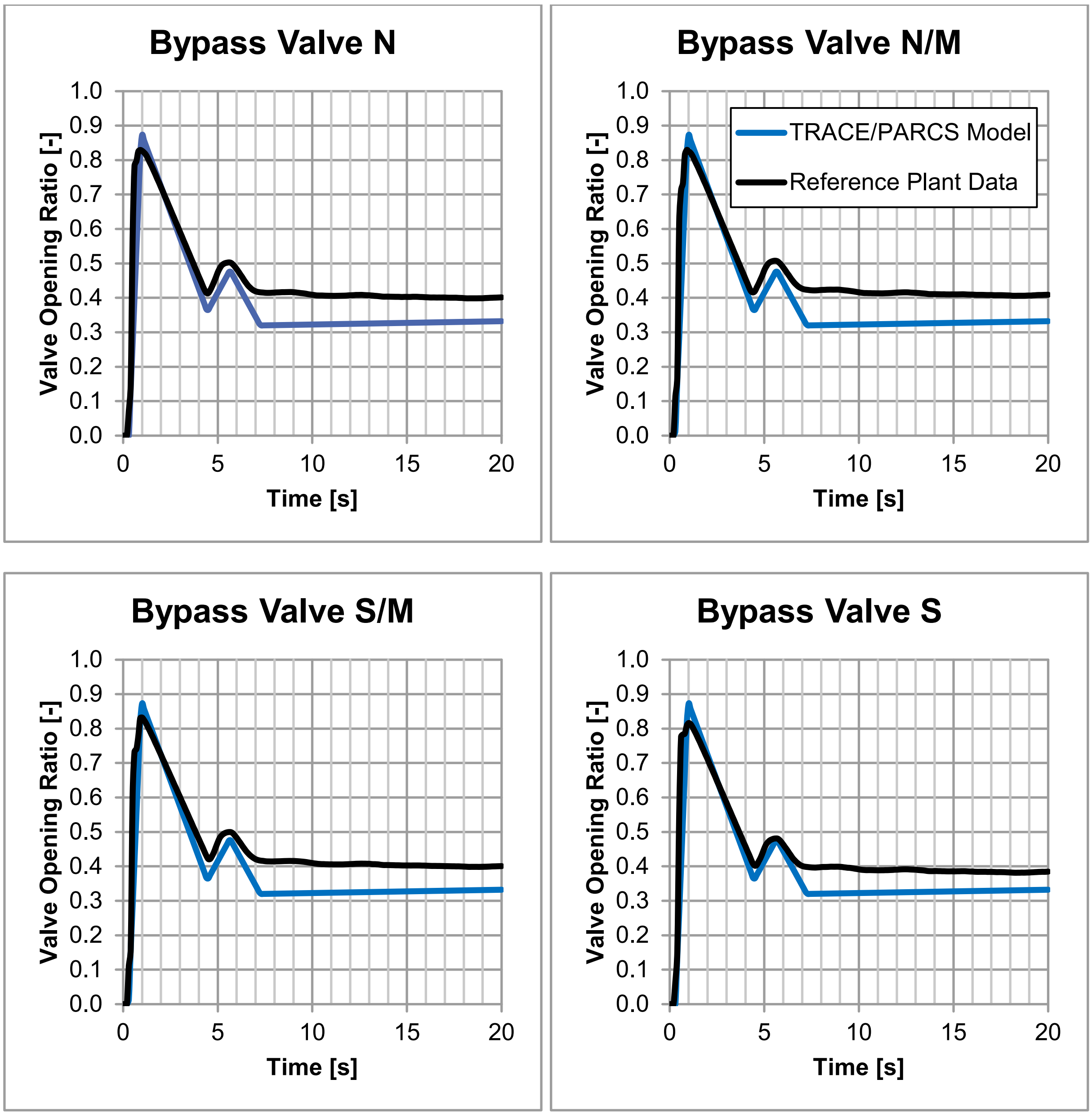

Figure 14 Opening of the bypass valves

In general, the model responded as expected with the given initiation sequence. The selected rod insertion was modeled in the PARCS part of the input. The different positions of the control rods were set in the PARCS input deck as it was documented in the test report.

5 RESULTS

The turbine trip test scenario was performed on the two different TRACE/PARCS inputs – the lumped core and the full core model. The problem time was set to 20 seconds with a maximal time step size of 10 milliseconds. The results were compared with the plant data of the turbine trip. Five characteristic parameters were chosen for comparison: core power, dome pressure, core flow, main steam flow and the water level. The final conditions of the transient after the 20 seconds are also summed up to conclude the results of the calculation.

5.1 Lumped Core Model

The results of the lumped core model shall be discussed first. The evolution of the core power is the most difficult value to control due to its strong coupling to many other parameters like the dome pressure, core flow or coolant temperature. In Figure 15, the core power during the turbine trip test is depicted.

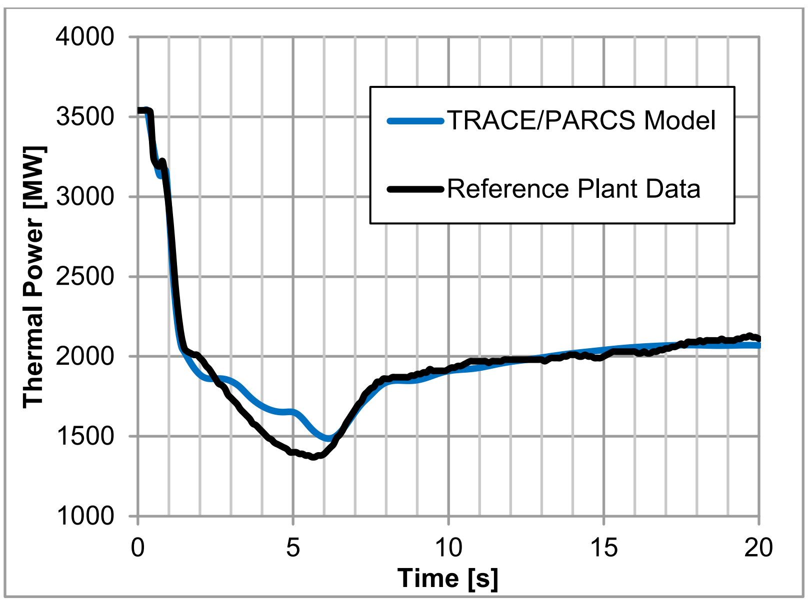

Figure 15 Core power behavior during the turbine trip test – lumped core model

As it can be seen, the deviation of the simulation results from the plant data is during most of the transient time very small. An exception is the time period between 1.5 and 6 seconds where TRACE/PARCS is clearly overestimating. The power excursion at 5 seconds can be explained with the evolution of the dome pressure, illustrated in Figure 16. At 5 seconds, the dome pressure experiences a peak caused by the pressure regulator of the bypass valves. This pressure collapses the void in the channels and induces a power response. In the plant data, this peak is not measured. The explanation can be found in the pressure distribution in the RPV. The dome pressure in the model is apparently propagating through the separators to the core while in the real plant a pressure wave is dampened by the separators. Therefore, some further improvements would be necessary for the reactor internals.

The estimated dome pressure by TRACE in Figure 16 is like the core power deviating only very little. As described before, the pressure was controlled manually with the bypass opening so it was very difficult to keep the dome pressure on a constant value. Besides this drawback, the calculation results were very good.

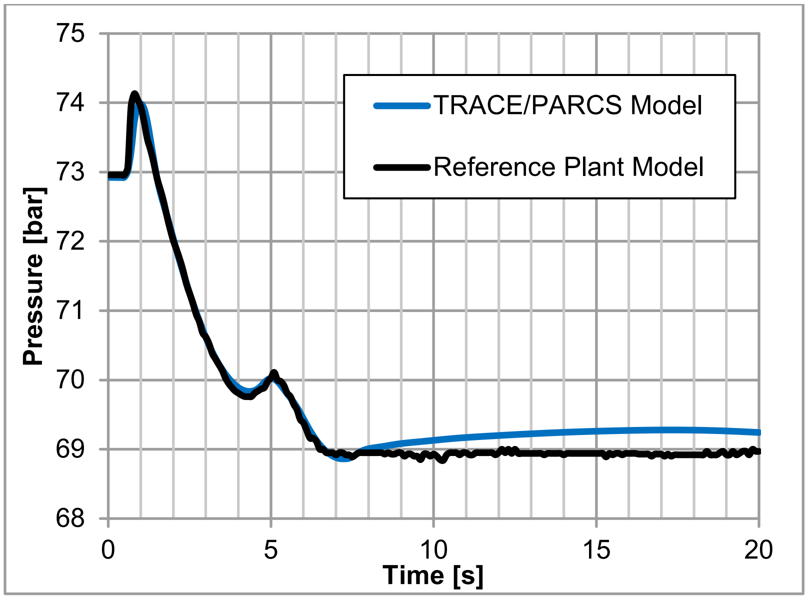

Figure 16 Dome pressure evolution during the turbine trip test – lumped core model

The values of the main steam flow in Figure 17 were the first to show a major discrepancy from the reference plant data. While the values for the maxima did match the reference one, the minima were always far underestimated. Also the timings of the peaks were always shifted by around 500 milliseconds. The design of the bypass valves can be one explanation for the different behavior. The modeling of the valve characteristic of the bypass valves is a difficult task. During normal operation, these valves are kept closed so there is little data available to benchmark the model against. This would explain the time shift in the response. The choked flow model could also have its contribution to the deviation. Both explanations would need further investigation to improve the behavior of the main steam flow.

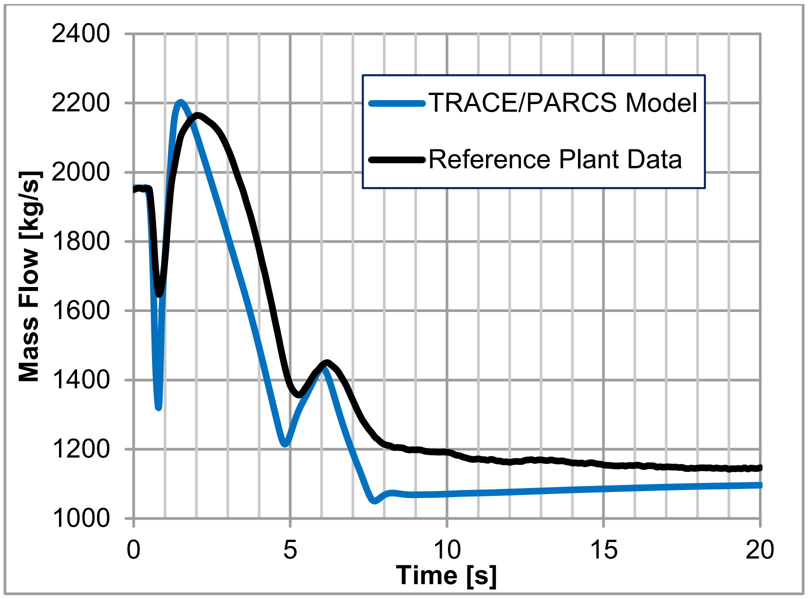

Figure 17 Main steam flow rate during the turbine trip test – lumped core model

The evolution of the core flow is shown in Figure 18. As mentioned in the section above, the recirculation runback is initiated 100ms earlier. This anticipation allows the TRACE/PARCS model to meet the linear part of the core flow drop between 1 and 4 seconds where most of the reactivity is decreased. The input was also here not able to predict the non-linear behavior of the real valves. Beside these side effects, the core flow follows the plant data with a very good agreement.

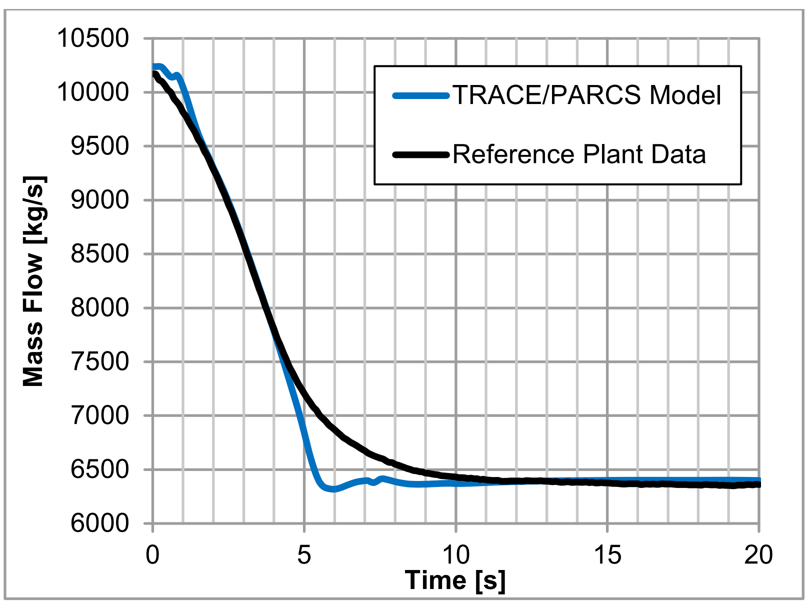

Figure 18 Core flow rate during the turbine trip test – lumped core model

The recirculation flow in Figure 19 on the other hand did show a good agreement. The discrepancy of the two recirculation lines is by far the biggest of all the presented parameters. As mentioned in the section above, the jet pump characteristic was found to be the reason. The flow control valve was adjusted in a way to meet the conditions for the core flow rate. As a consequence, the results of the recirculation flow rate suffered.

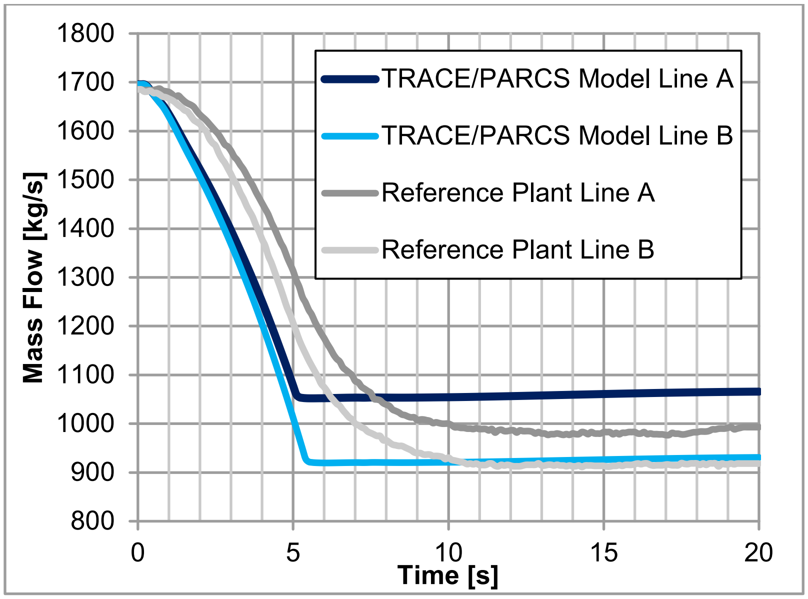

Figure 19 Recirculation flow rate during the turbine trip test – lumped core model

The jet pump flow ratio is pictured in Figure 20. The two different behaviors of the input model and the plant data drift clearly apart. The plant data shows first a decrease, followed by a recovery to an even higher value than the initial one. The jet pump flow ratio for the TRACE/PARCS model starts immediately to increase but stabilizes at about the same time as the reference plant data on a lower value. This big discrepancy originates probably from the pressure distribution in the RPV. Compared to the real plant, as it could be seen in the core power in Figure 15, the pressure drop from the dome to the lower region is too small and the internals respond too sensitive on dome pressure changes. Thus, the flow in the jet pump is accelerated by the higher pressure in the downcomer while the plant data indicates this response after 4 seconds. Concluding these results, the recirculation lines and the jet pumps will need some modeling work in the near future.

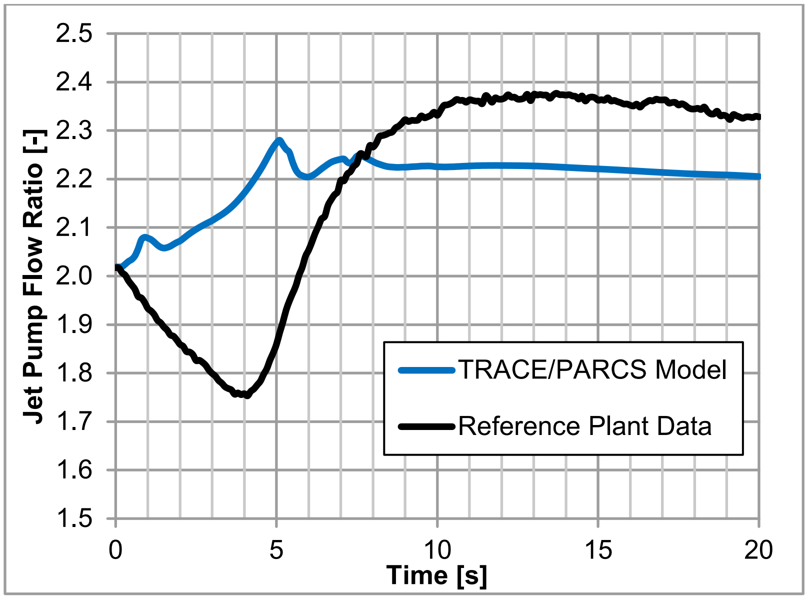

Figure 20 Jet pump flow ratio performance during the turbine trip test – lumped core model

The new level water level instrumentation was also benchmarked against the plant data. In Figure 21, the comparison of the calculation and the reference is depicted. The level measurement reacts very sensitive on the pressure difference which explains the noisy behavior of the estimated water level. The preassigned method in (1) to determine the water level is thus only working for the steady-state calculation where the pressure difference is also converging towards a stable value. During a transient, where the internal pressure is rapidly changing, a different method has to be found to get a reliable water level during transient calculations.

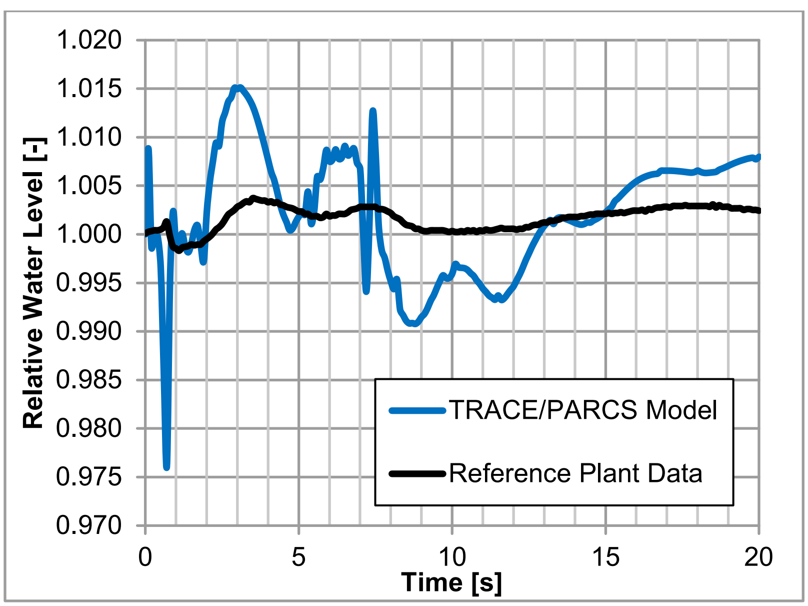

Figure 21 Water level determination during the turbine trip test – lumped core model

The final values of the transient were again summarized and can be found in Table 6. As it can be seen, the relative errors increased for some parameters, i.e. core power, recirculation and steam flow rate. The core power is a very difficult parameter to control since it is influenced by almost all other values. Therefore, a relative error of within 2 % is a good result.

Table 6 Final conditions after the turbine trip test for the lumped core model

	Plant Data	TRACE/PARCS	Relative Error [%]
Core Power [MW]	2110.00	2069.19	-1.934
Core Flow [kg/s]	6360.62	6399.39	0.610
Dome Pressure [bar]	68.97	69.24	0.391
Recirculation Flow Rate [kg/s]	1911.15	1996.54	4.468
Steam Flow Rate [kg/s]	1146.94	1096.12	-4.431
Feedwater Flow Rate [kg/s]	1102.78	1103.61	0.075
Relative Water Level [-]	1.002	1.008	0.599

As mentioned above, the recirculation and steam flow rate suffer both of compromises that had to be made. The recirculation flow rate had to be increased to keep the core flow at the power plant level. Again, the recirculation line needs another revision to widen the working range of the jet pumps. The steam flow rate has been already discussed at Figure 17.

In principal, the lumped core model performed very well, especially comparing with the previous state of the model in [4]. There is still some space for improvements, e.g. the recirculation lines, the pressure controller for the bypass valves, the water level instrumentation or the pressure distribution in the RPV. The TRACE/PARCS model is though able to reach coupled steady-state conditions with a good convergence criterion in order to run some transient calculations.

5.2 <u>Full Core Model</u>

The full core model has been setup according to the initial conditions listed in Table 3. The model was running the same transient with the same trips, closing and opening times and control rod insertion. The aim of this calculation was to investigate the influence of a thermal-hydraulic lumped core.

The first illustration in Figure 22 shows the core power evolution of the full core model. The differences between the core power in Figure 15 and Figure 22 appear to be very small. The selected control rod insertion is the only measure that shows a better response at 1.5 seconds than in the lumped core case. The fuel elements around the control rods are in this case more sensitive to inserted control rods than the lumped fuel element where the effect is averaged over an assembly of fuel elements.

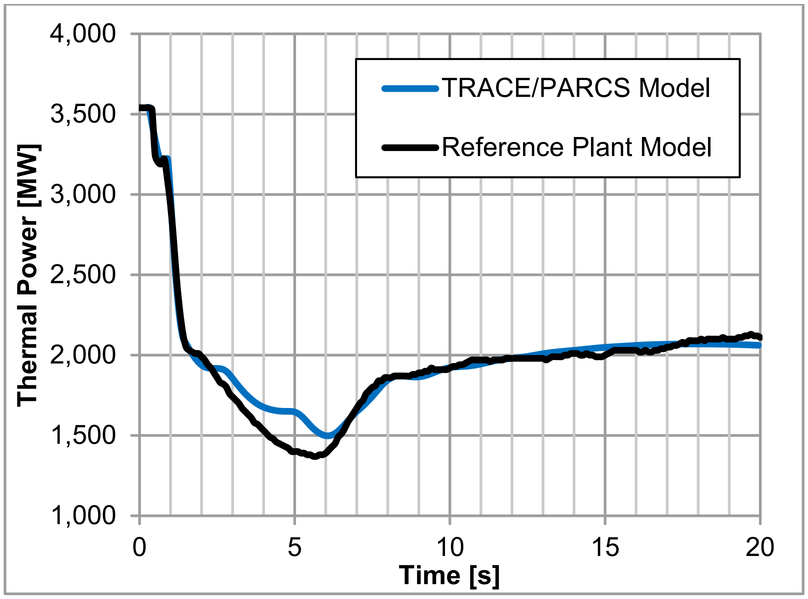

Figure 22 Core power behavior during the turbine trip test – full core model

The pressure in the dome, pictured in Figure 23, is following the plant data at beginning with a high accuracy. The discrepancy between the simulation and the reference data starts to increase after 3 seconds of problem time. The pressure remains from there on overestimated for the rest of the transient. The explanation for this behavior has already been given in the previous section. Since the core pressure drop is smaller in the full core model than in the lumped core, it is not surprising to see an influence also in the behavior of the dome pressure.

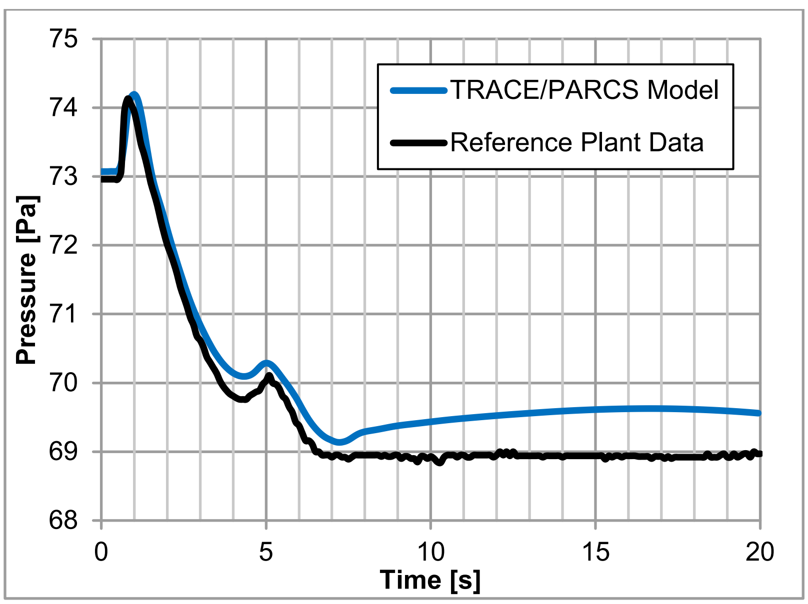

Figure 23 Dome pressure evolution during the turbine trip test – full core model

The main steam flow rate in Figure 24 looks very similar to the flow rate in Figure 17. Also a closer look does not reveal any significant difference. The root-mean-square deviation between the results of the lumped core model and the full core model is less than 1%. It is no coincidence to find basically the same results concerning the steam flow rate. The mass flow rate is influenced by the activation of the TCVs and BPVs. The steam boiling in the core is compared to the acting pressure waves in the main steam lines relatively slow so the mass flow will be determined rather by the valve action than the power excursion. The opening respectively closing sequence of the valves did not change for the two TRACE/PARCS model so the difference between the two is expected to be small.

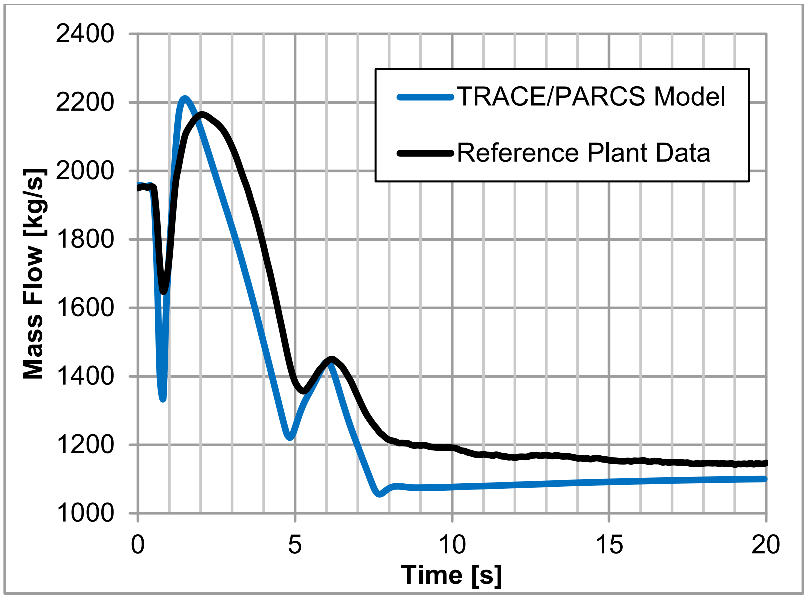

Figure 24 Main steam flow rate during the turbine trip test – full core model

The comparison between the recirculation flow rate of the full core model in Figure 25 and the lumped core model in Figure 19 shows even fewer differences than the main steam flow rate. The root-mean-square error of the two flow rates is actually less than 0.2 %.The similarity is again no surprise, since the recirculation pumps and the FCV closing sequences are exactly the same for both models, which is eventually the driving force of the recirculation flow rate.

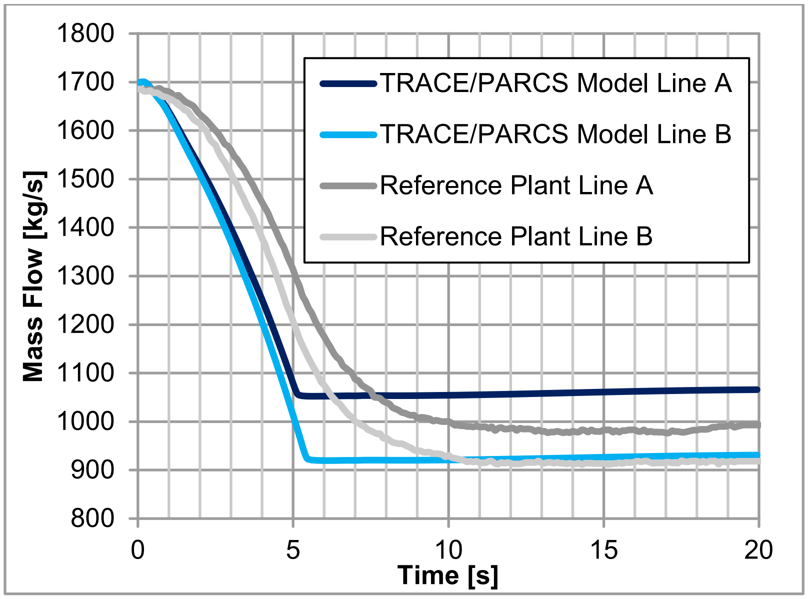

Figure 25 Recirculation flow rate during the turbine trip test – full core model

The core flow rate on the other hand is not only influenced by the forced flow of the recirculation lines but also by the pressure difference between the downcomer and the lower plenum. As mentioned before, the core pressure drop of the full core model is slightly smaller than in the lumped core model. Therefore, the dome pressure was also higher which has a direct impact on the inlet pressure of the jet pump. In Figure 26, the total core flow rate is illustrated for the full core model. The flow rate is overestimated for most of the time except for the missing non-linearity at around 5 seconds.

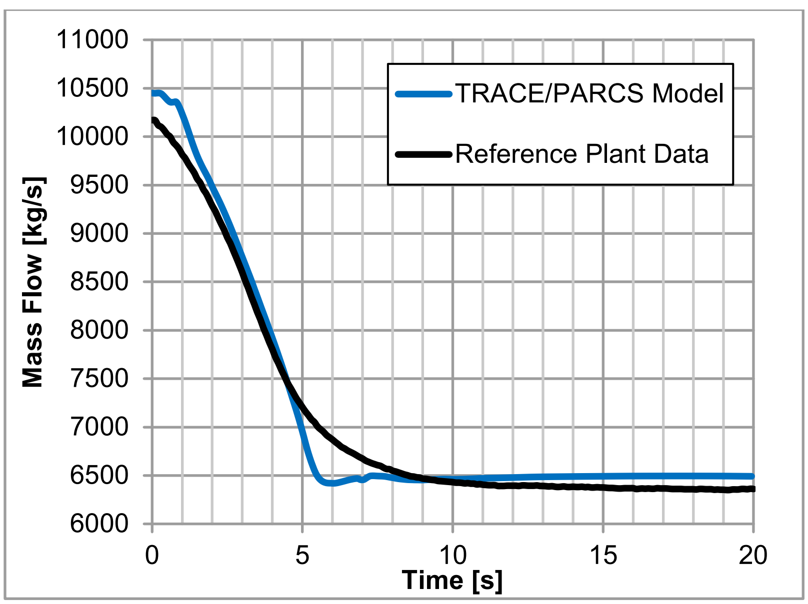

Figure 26 Core flow rate during the turbine trip test – full core model

The effect of the pressure on the jet pump is also visible in the jet pump flow ratio in Figure 27. Comparing the results with Figure 20 shows the same behavior during the transient but shifted on the ordinate. The jet pump flow ratio is in this case around 2 % higher.

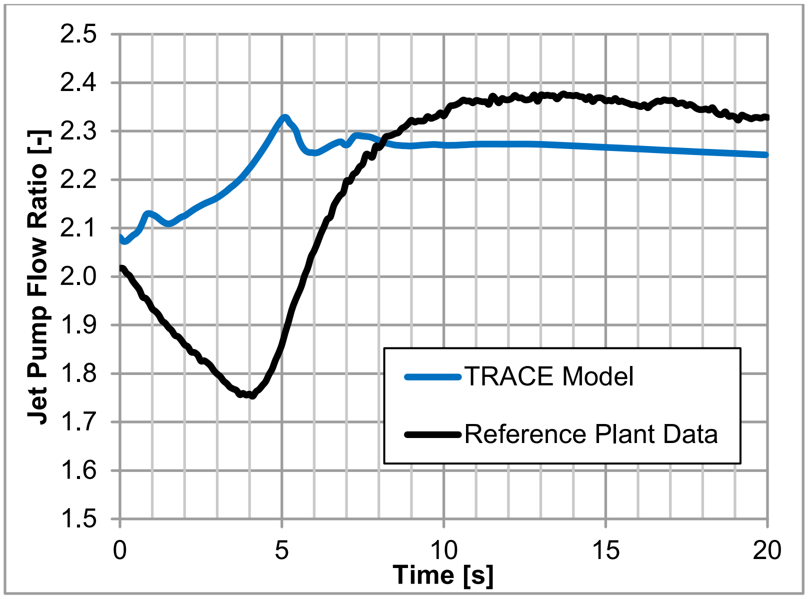

Figure 27 Jet pump flow ratio performance during the turbine trip test – full core model

In Figure 28, the evolution of the water level is depicted. Once more, the deviations to the results of the lumped core model seem to be very small. The water level is as described above estimated via the hydraulic pressure difference at two different points in the downcomer. A higher pressure in the dome will increase the pressure everywhere, from the downcomer to the upper plenum. Thus, the difference will remain approximately constant and the water level determination will not be strongly affected.

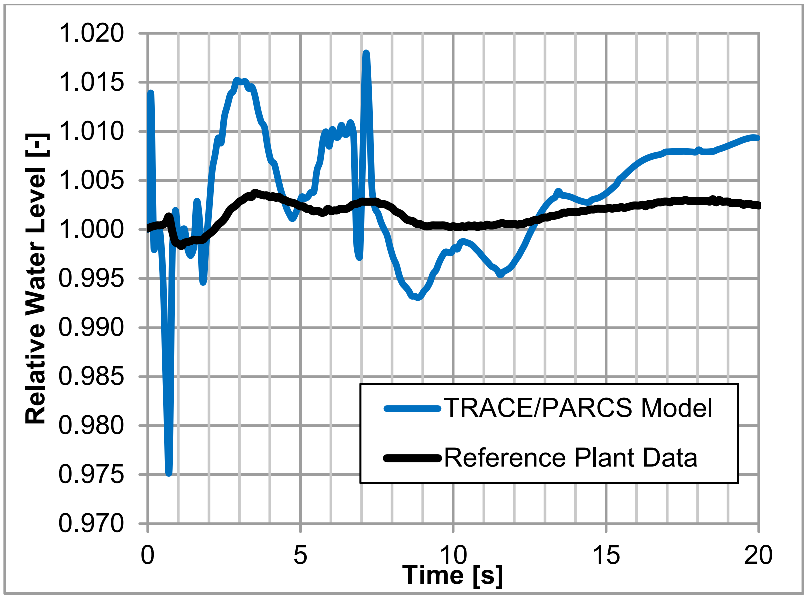

Figure 28 Water level determination during the turbine trip test – full core model

Last but not least, the final conditions after the turbine trip test are summarized in Table 7. Comparing the relative errors of the simulation with the lumped core model in Table 6, the only significant difference can be found for the core power, the dome pressure and the core flow rate. As also seen in the results, the core pressure drop affected only these three values.

Table 7 Final conditions after the turbine trip test for the full core model

	Plant Data	TRACE/PARCS	Relative Error [%]
Core Power [MW]	2110.00	2061.41	-2.303
Core Flow Rate [kg/s]	6360.62	6491.61	2.059
Dome Pressure [bar]	68.97	69.56	0.855
Recirculation Flow Rate [kg/s]	1911.15	1996.71	4.477
Steam Flow Rate [kg/s]	1146.94	1100.00	-4.093
Feedwater Flow Rate [kg/s]	1102.78	1106.60	0.346
Relative Water Level [-]	1.002	1.009	0.699

5.3 Difference Between Full Core and Lumped Core Model

The differences between the fully mapped core and the lumped three channel model appeared to be very small and insignificant. The lumped core seemed even to be slightly more conservative in terms of core power and pressure distribution inside the core. The question arises of whether the full core model is necessary for the transient analysis or if the lumped core would be sufficient. This question cannot be fully answered with only one transient calculation, but the results give a small hint. In Figure 29, the pressure difference between the lower and the mixing plenum is shown. The pressure drop for the lumped core stays above the pressure drop of the full core model.

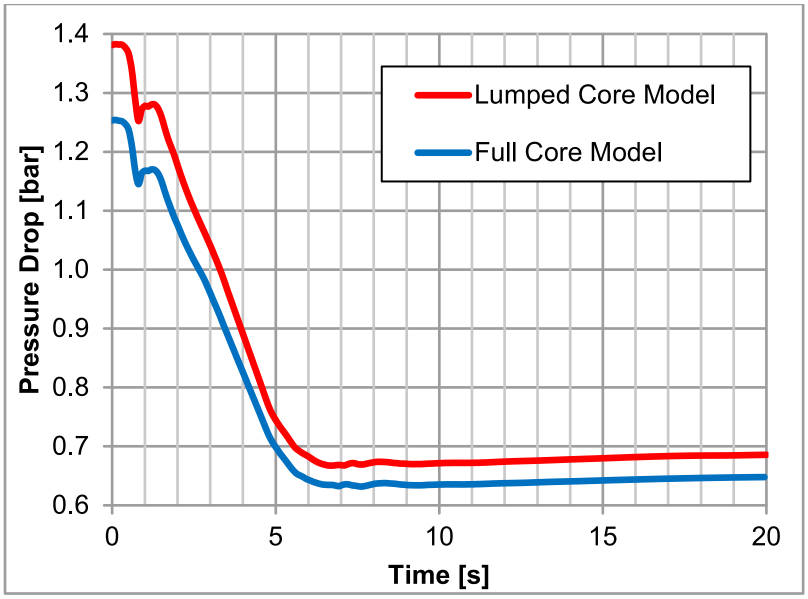

Figure 29 Pressure drop lower plenum – mixing plenum during the turbine trip test

Therefore, the pressure distribution will not change inside the RPV, relatively speaking. This is what the results of the full core model have shown as well. The only differences originated in the different pressure drop inside the core. Eventually, the deviations remained very small. So in terms of fast transients where the neutronic feedback has not too much time to respond, the lumped core model can enough for modeling the transient. If more accurate results are needed, e.g. hottest rod position, exact pressure distribution in a particular fuel element or if the margins are small, the full core should be represented in the TRACE/PARCS model.

6 RUN STATISTICS

The calculations on the lumped core model with the TRACE/PARCS computer code (TRACE V5.0 p3, PARCS 2.7) were performed on a work station which is built on an Intel Xeon 5450 CPU (based on the Intel Core microarchitecture @ 3.00 GHz), having 4 cores and 4 threads in total. The operating system is Windows 7 Service Pack 1 (x86_64) with 16 GB of RAM memory.

Table 8 Run statistics for the lumped core model

Characteristic Value	Stand-Alone Steady-State	Coupled Steady-State	Turbine Trip Transient
Problem Time [s]	176.786	78.170	20.002
CPU Time [s]	478.642	157.436	805.717
Number of Time Steps [-]	9505	810	2024
Maximum Time Step Size [s]	1e-2	1e-1	1e-2
CPU Time / Problem Time [-]	2.707	2.014	40.282
Number of Thermal-Hydraulic Cells [-]	81		
Number of Heat Structures [-]	225		

In Table 8, the run statistics for the transient turbine trip benchmark are listed. The total number of thermal-hydraulic cells was 81 and 225 heat structures. The transient calculations run due to the small time step size 40 times slower than the real time. Stand-Alone steady-state calculations lasted 176.786 seconds of problem and 478.642 seconds of CPU time, requiring 9505 time steps to reach the convergence criterion while the coupled steady-state needed 78.17 seconds of problem time, 157.436 seconds of real time and 810 steps.

The full core model was running on a work station with two Intel Xeon E5-2643 CPUs (based on the Sandy Bridge microarchitecture @ 3.30 GHz), having 8 cores and 16 threads in total. The operating system is Windows 7 Service Pack 1 (x86_64) with 32 GB of RAM memory.

The run statistics for the full core model are presented in Table 9. The problem times remained in the same order of magnitude while the CPU times increased significantly. The number of thermal-hydraulic cells went up to 17496 and the number of heat structures to 48600. Therefore, more computations were needed per time step which increased the CPU time to 20812.748 seconds for the stand-alone case, to 5866.339 seconds for the coupled steady-state calculation and 611.453 for the transient. The worst CPU / problem time ratio turned out to be during the transient calculation with 1 second of problem time being more than 10 minutes of CPU time. The number of time steps though stayed approximately on the same level for both models.

Table 9 Run statistics for the full core model

Characteristic Value	Stand-Alone Steady-State	Coupled Steady-State	Turbine Trip Transient
Problem Time [s]	199.886	126.805	20.009
CPU Time [s]	20812.748	5866.339	12234.565
Number of Time Steps [-]	11850	1310	2049
Maximum Time Step Size [s]	1e-2	1e-1	1e-2
CPU Time / Problem Time [-]	104.123	46.263	611.453
Number of Thermal-Hydraulic Cells [-]	17496		
Number of Heat Structures [-]	48600		

7 CONCLUSION

This work continued the improvement of a TRACE/PARCS model which has been started some time ago ([3 - 5]). The TRACE model has been reviewed with existing technical documentation and improved in terms of control systems and cell volume terms and a second model has been developed where the full core has been modeled with single fuel element channels. Both TRACE models have been coupled with the neutronics system code PARCS and validated with existing plant data resulting from a turbine trip test. For the analysis of further fast transients, the validation required to be done with another fast transient. The turbine trip is a very favorable transient, since all the measures to maintain the reactor in operation are taken, i.e. selected rod insertion, recirculation runback and the activation of the turbine bypass.

The results of the benchmarks against the plant data were very promising. The numerical instabilities were tremendously reduced and the characteristic parameters, i.e. dome pressure, core flow, core power and steam flow, were able to follow and to a certain degree also match the existing data. The initial and final conditions are consistent within 5 % accuracy. The discrepancy of the valve opening and closing behavior with the real valves could not be resolved yet and will need future work. The built-in pressure controller in TRACE for the bypass valves did not perform as described in the TRACE manual which made manual adjustments of the bypass opening necessary. Last but not least, the jet pump characteristic was not properly modeled and will require further research and modification.

The difference of the lumped core and the full core model revealed a different pressure drop inside the core. In the full core model, the core inlet flow is established elementwise in each fuel assembly and thus also the pressure drop. In the lumped core model, the core flow is distributed within three channels which lead to different flow conditions in the core and hence a different pressure drop.

In general, the TRACE/PARCS coupled models showed a very good performance. Not only that the results in the benchmark against the turbine trip data were met with a good accuracy, the results in the fast transient analysis could also be reasonably explained. First steps have been done in understanding the effect of lumping for fast transients and will be expanded in the future work. However, the nuclear power plant of Leibstadt is now one step further in their development of a TRACE/PARCS model that can already be used for the validation of existing transient analysis.

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11. ABSTRACT (200 words or less)

The Nuclear Power Plant of Leibstadt (KKL) is a participating member of the Code Applications and Maintenance Program (CAMP) of the U.S. Nuclear Regulatory Commission (USNRC) to validate the TRACE code for BWR/6 transient analysis. The application of TRACE for the safety assessment of BWRs requires a throughout verification and validation using experimental data from tests but also plant data for the modelling. The purpose of this work is the review of the KKL TRACE/PARCS model, the benchmark of the model against plant data recorded during a turbine trip test and an investigation of the core lumping effect on the turbine trip test.

A coupled TRACE/PARCS model has been developed to analyze fast transients in KKL. The first benchmark against a turbine trip test has shown differences between the test data and the results predicted by TRACE/PARCS such as the total core power and the dome pressure. This is mainly due to unstable steady-state conditions during the initialization process and modelling issues. The improvements introduced in this work to the TRACE model are including but not limited to the geometry of the reactor internals, the redesign of the main steam lines and the implementation of a rudimentary control system. Furthermore, the PARCS input model has been updated with the turbine trip test corresponding cross sections. The new designed coupled TRACE/PARCS model was eventually benchmarked against the same turbine trip plant data.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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Main Steam Isolation Valve (MSIV)
Reload Licensing (RLS)
Selected Rod Insertion (SRI)
GE Boiling Water Reactor (BWR/6)
Automatic Depressurization System (ADS)
Reactor Core Isolation Cooling (RCIC)
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