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OECD/NEA MAIN STEAM LINE BREAK PWR BENCHMARK SIMULATION BY TRACE/S3K COUPLED CODE

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Abstract

A coupling between the TRACE system thermal-hydraulics code and the SIMULATE-3K (S3K) three-dimensional reactor kinetics code has been developed in a collaboration between the Paul Scherrer Institut (PSI) and Studsvik. In order to verify the coupling scheme and the coupled code capabilities with regards to plant transients, the OECD/NEA Main Steam Line Break PWR benchmark was simulated with the coupled TRACE/S3K code. The core/plant system data were taken from the benchmark specifications, while the nuclear data were generated with the Studsvik's lattice code CASMO-4 and the core analysis code SIMULATE-3. The TRACE/S3K results were compared with the published results obtained by the 17 participants of the benchmark. The comparison shows that the TRACE/S3K code reproduces satisfactory the main transient parameters, namely, the power and reactivity history, steam generator inventory, and pressure response.

Introduction

The TRACE/S3K thermal-hydraulics and three-dimensional reactor kinetics coupled code was developed to consolidate the performance of two principal reactor analysis transient codes used at PSI by the STARS (Safety research related to Transient Analysis of Reactors in Switzerland) project [1] for supporting the existing Swiss Nuclear Power Plants (NPPs). The first code is the two group nodal kinetics code SIMULATE3K (S3K) [2] and the second one is the best-estimate thermal-hydraulic system code TRACE [3]. The motivation at the basis of the development of the TRACE/S3K coupled code is the continuing enhancement of the capability to perform best-estimate simulation of Light Water Reactors (LWRs) transients, where strong coupling between the core neutronics and the plant thermal-hydraulics and/or asymmetrical power generation take place, e.g. during Anticipated Transient Without Scram (ATWS).

The S3K and TRACE codes are intensively verified and validated by the developers and the user communities. However, additional validation and verification is needed for the coupled TRACE/S3K code. An independent verification matrix was compiled at PSI in order to assess the coupled code capability and accuracy. It includes PWR and BWR benchmarks as well as comparison with the available Swiss NPPs data. Previously, the OECD/NEA Peach Bottom 2 Turbine Trip Test 2 (BWR/4 NPP) has been successfully simulated by means of the

TRACE/S3K code [4], and a good agreement between the TRACE/S3K results and the experimental data has been demonstrated.

In the present work, the coupled TRACE/S3K code is validated against the OECD/NEA Main Steam Line Break (MSLB) PWR benchmark [5]. This benchmark was selected as part of the TRACE/S3K validation database because it is a dynamically complex event for which neutron kinetics in the core is strongly coupled with the thermal-hydraulics of the reactor primary and secondary systems. The OECD/NEA MSLB PWR benchmark is based on real plant design and operational data for the TMI-1 NPP. The purpose of this benchmark was three-fold: to verify the capability of system codes for analyzing complex transients with coupled core-plant interactions; to test the 3D neutronics/thermal-hydraulics coupling; and to evaluate discrepancies among the predictions of coupled codes in best-estimate transient simulations. The benchmark has been analyzed by 17 participants from 14 different organizations/companies. It should be clearly stated that experimental data is not available for a PWR MSLB transient scenario, so that the traditional code-to-data comparison methods are not applicable, in contrast to the previously mentioned Peach Bottom Turbine Trip Test benchmark. In addition, several benchmark participants have submitted results obtained with multiple versions of the same code. Consequently, not all of the sets of results submitted by the benchmark participants are completely independent of each other, and simple averaging techniques may not provide an accurate statistical representation of the data. To resolve these issues, the reference solution for all parameters is based upon a statistical mean value of all submitted values, corrected to account for the inter-dependence of some results. The statistical methods used for code-to-code comparisons in the OECD/NRC PWR MSLB benchmark are described in Ref. [6].

A brief description of the TRACE/S3K coupled code is given in section 2. The benchmark description and corresponding model nodalization are discussed in section 3, while section 4 provides the best-estimate TRACE/S3K results for the base case and the comparison with the mean benchmark data.

1. Description of TRACE/S3K coupled code

The S3K code is the principal 3-D kinetics solver within the STARS project at PSI and in that context, an assessment of the code for a wide range of core transients has been launched (e.g. [7], [8]). In parallel, the NRC-sponsored best-estimate thermal-hydraulics code TRACE was adopted to replace in a consolidated manner, the formerly used tools (e.g. TRAC-BF1, RELAP and RETRAN) for system transient analyses of the Swiss reactors. To support this migration, substantial assessment/validation efforts have been undertaken and continue to be carried out (e.g. [9], [10]). On the basis of the maturity achieved through these assessments, a coupling between the two codes was considered as an evident next step. In this context, although the TRACE code integrates the advanced neutronics PARCS solver, S3K was selected as the 3-D kinetics solver to take advantage of the CASMO-4(5)/SIMULATE-3 based methodology employed at PSI for the core modeling and steady-state analyses of the Swiss reactors [11].

1.1 Neutron Kinetics Code S3K

The S3K code is a best-estimate reactor analysis tool that employs advanced core neutronics coupled with detailed thermal-hydraulic channel models. Faithful modeling of assembly-by-assembly neutronics effects, including assembly pin power reconstruction, allows the application of S3K to a wide class of LWR core transients. In S3K the transient three-dimensional, two-group neutron diffusion equations are solved, including a six group model for delayed neutron precursors. S3K tracks dynamically nodal concentrations of fission products and accounts for the neutron sources due to spontaneous fissions, alpha-n interactions from actinides decay, and gamma-n interactions from long-term fission product decay.

The basic spatial integration model of S3K is formed via transverse integration of the 3-D equations separately over each spatial direction. This procedure creates an equivalent set of three one-dimensional equations coupled via a transverse leakage term. The flux distribution is expanded in terms of fourth-order polynomials (or analytical functions) in each direction and thus the spatial gradient of the flux can be analytically represented by a third-order function. This procedure yields the spatial difference equations for the two-group neutron flux. These difference equations also include assembly discontinuity factors (ADFs), which take into account the fact that adjacent assemblies may contain significant material heterogeneities. ADFs are generated by the lattice physics code CASMO as part of the core design process and are stored in the basic two-group data library. The frequency transform method is used for the time integration of the transient diffusion equations. This method separates the flux into two components, one with a pure exponential time dependence, and the other with primarily spatial (and weak temporal) dependence. The time integration of the diffusion equations is performed using backward differences. The knowledge of the intra-nodal flux and power distributions within each node can be used to compute the pin-by-pin power for every axial level of every fuel pin in the core. More details are given in Ref. [12].

1.2 System Code TRACE

TRACEv5.0 [3] is the latest in a series of advanced, best-estimate reactor system codes developed by the U.S. Nuclear Regulatory Commission (with the involvement of Los Alamos National Laboratory, Integrated Systems Laboratory (ISL), The Pennsylvania State University (PSU) and Purdue University) for analyzing the transient and stationary neutronic/thermal-hydraulic behaviour of Light Water Reactors (LWRs). The code is a result of a consolidation of the capabilities of previous USNRC supported codes, such as TRAC-PF1, TRAC-BF1, RELAP-5 and RAMONA. The most important models of TRACE include multidimensional non-equilibrium two-phase flow, generalized heat transfer, reflood, level tracking and reactor kinetics.

The set of coupled partial differential equations, together with the necessary closure relationships, is solved in a staggered (momentum solved at cell edges) finite difference mesh. Heat transfer is treated semi-implicitly, while the hydrodynamic equations (1, 2 and 3 Dimensional) make use of a multi-step time differencing scheme (SETS) that allows the material Courant limit to be violated, thus resulting in large time step sizes for slow transients, and fast running capabilities. The finite-difference equations for hydrodynamic phenomena form a system of coupled, nonlinear equations that are solved by the Newton-Raphson iteration method. The resulting linearized equations are solved by direct matrix inversion. A full two-fluid (6-equations) model is used to evaluate the steam-liquid flow, with an additional mass balance equation to describe a non-condensable gas field, and an additional transport equation to track dissolved solute in the liquid field.

1.3 TRACE/S3K Coupled Code

Typically, the TRACE PWR core thermal-hydraulics (TH) nodalization does not include a heat structure for each fuel assembly, but only 3 to 50 effective heat structures. These heat structures comprise from a few up to 100 fuel assemblies each. Since for the coupling between S3K and TRACE, the latter code must provide fuel assembly based TH parameters to S3K in order to take into account for the thermal feedback effects during a transient, a core mapping between the TRACE and the S3K core nodalization is needed. The core mapping can be freely selected by the user. However, there are two limitations imposed to the mapping: a) the axial nodalization in the active part of the core region must be identical in the TRACE and S3K input decks, b) each of the thermal-hydraulic volumes and heat structures that define a zone in the active fuel region must be unique and not reused in the other active fuel zones. The last limitation restricts a full-core mapping capability if a TRACE VESSEL component is used to represent a PWR RPV.

The linkage between TRACE and S3K is a direct, explicit coupling of the two codes on a synchronous time-step basis. The coupling provides a method of executing the S3K three-dimensional neutronics using the plant boundary conditions calculated by the TRACE thermal hydraulics code. The S3K calculated total core power and core power distributions “drive” the TRACE system model core.

The thermal-hydraulic conditions in the core and plenum regions calculated by TRACE are passed to the S3K model, which performs a calculation for the detailed core power. This is then passed back to the TRACE model, and is used for the next time step. There are three different coupling options that are available for the linkage between TRACE and S3K, “plenum”, “flat“, and “nodal“coupling. Each of these options is described below.

The “plenum” coupling option utilizes the S3K thermal-hydraulics calculation module for the core section. The inlet flow and enthalpy to the core and the exit pressure in the upper plenum is provided by TRACE. S3K will use this data to perform its own thermal-hydraulic calculations in the core region. Each fuel assembly in the core is modeled separately and a common bypass channel is used for all bypass flows. More details concerning the S3K thermal-hydraulics model are given in Ref. [2]. These thermal-hydraulic results are only used to provide feedback values on a nodal basis for the cross section evaluation. The resultant power distribution is then collapsed back to the coarse core nodalization used by the TRACE model. This option is under implementation at the moment.

The “flat” coupling option does not utilize the S3K thermal-hydraulics calculation module. Each fuel assembly in a TRACE channel receives the same fuel temperature, coolant density, and boron concentration at a given axial plane from the TRACE calculation. This option is very robust, but it will approximate the accurate radial power distribution (especially for the hot assemblies or controlled assemblies) unless a large number (>100) of TRACE heat structures are modeled.

The “nodal” coupling option is a variation of the “flat” option. Once again, the S3K thermal-hydraulics calculation module is bypassed. However, an estimate of the true three-dimensional density and fuel temperature distributions is made utilizing the current nodal powers. The fuel temperature is estimated from the coarse value calculated by TRACE using a weight factor, evaluated as the ratio of the nodal power to the average power in the channel in that plane. The coolant density for a given fuel assembly is calculated using a simple enthalpy rise calculation and the same weight factor described for the fuel temperature calculation. The density calculation also includes a normalization step that preserves the mass of liquid for each TRACE channel.

2. MSLB description and modelling

The OECD/NEA MSLB international benchmark has been selected as one of the PWR cases for the validation of the coupled TRACE/S3K code. In the benchmark it is supposed that a rupture of one steam line upstream of the main steam isolation valves occurs. The reference design employed in the benchmark was based on the reactor geometry and operational data of the Three Mile Island Unit 1 NPP (TMI-1). The schematic TRACE thermal-hydraulic model for TMI-1 plant is shown in Figure 1. This model is based on the TMI-1 input deck received from the NRC [13].

The transient starts with a doubled-ended break of one main steam line at the tie with the cross-connection line. The 24 inch (60.96 cm) main steam line and 8 inch (20.32 cm) cross-connect rupture results in the highest break flow assumption and maximizes the Reactor Coolant System (RCS) cool-down. The break flow is simulated using the default TRACE choking model. The failure in the open position of the feedwater regulating valve to the broken steam generator (SG) is considered as the worst single failure. This failure in the open position causes feedwater flow from the intact SG to cross over to the broken SG across the common header and maximizes the feedwater flow to the broken SG. The feedwater flow is eventually terminated by closure of the feedwater block valve, which is conservatively assumed to close 30 seconds after the break occurs.

Subsequent to the break initiation, and following the reactor scram, the turbine stop valves in steam lines are assumed to close, isolating the intact SG. The 8 inch cross-connect between the two steam lines of the broken SG remains open.

The nuclear data were generated with the Studsvik's lattice code CASMO-4 and the core analysis code SIMULATE-3. The S3K core model relied on Studsvik's codes to generate the cross section data, and the history data. All data pertaining to the core (cross-sections, assembly dimensions, pin enrichments, etc.) were taken from the specification. The control rods were positioned as specified in the benchmark specification, with the control rod in location N-12 assumed to be stuck in a full out position for the entire duration of the transient. The average fuel pellet nodewise temperature is used as the Doppler temperature in each kinetic node in contrast to the specification where the Doppler temperature is given by the interpolation between the fuel temperature at the fuel rod center and the fuel rod surface. The latter method is used in some kinetics codes, e.g. PARCS [14].

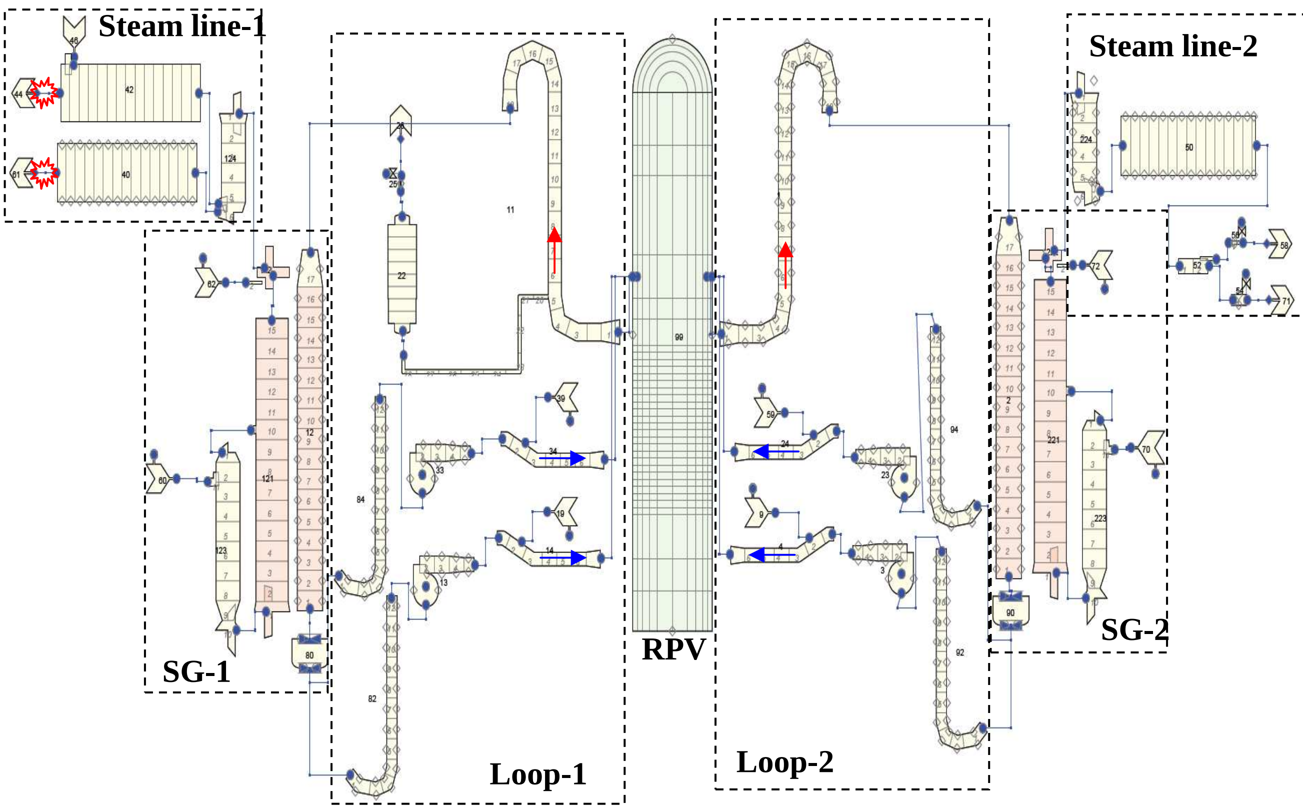

Figure 1 Schematic TRACE model for MSLB.

The reference reactor core includes 241 assemblies: 177 fuel assemblies and 64 reflector assemblies. Axially, the reactor core is divided into 24 layers. The fuel assemblies in the TRACE deck are mapped using 18 sectors of the vessel component, accompanied with 18 heat structure components. The corresponding radial and azimuthal nodalization of TRACE vessel component is shown in Figure 2 while the S3K fuel assembly mapping is shown in Figure 3.

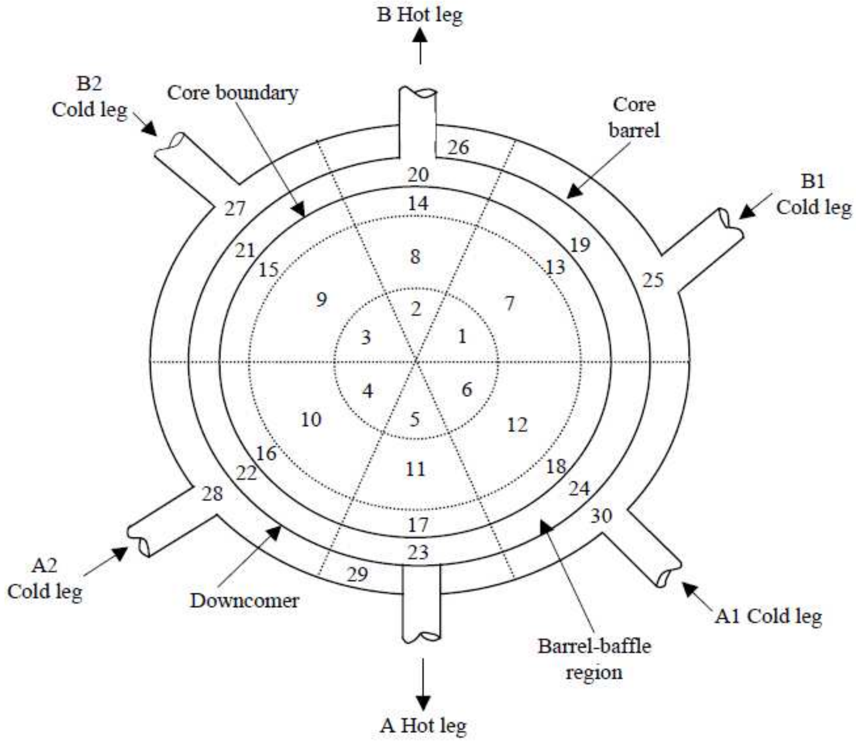

Figure 2 TRACE vessel radial and azimuthal nodalization.

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0	0	0	0	0	0	0	0	0	0	17	17	17	17	17	17	17	17	17	17	0	0	0	0	0	0	0	0	0	0

Figure 3 S3K fuel assembly mapping to TRACE vessel component.

All four RCS pumps are assumed to operate during the event, since maximizing the primary to secondary heat transfer will cause maximum RCS cool-down. No credit is taken for the operation of the pressurizer heaters. This conservative assumption enhances the RCS depressurization.

The reactor trip is modeled to occur when the neutron power reaches 114% of 2772 MW, or when the primary system pressure reaches 13.41 MPa. A delay of 0.4 seconds is used for the high neutron flux trip; while the low RCS pressure delay is modeled as 0.5 seconds. These values represent the delay from the time the trip condition is reached to the time the control rods are free to fall, and bound the actual delays for TMI-1.

The high pressure injection (HPI) starts with a 25 second delay after the primary system pressure drops below 11.34 MPa. The HPI system is expected to activate because of the large overcooling which occurs during this transient. No credit is taken for negative reactivity insertion deriving from boron addition. No other emergency core coolant system (ECCS) action is expected to occur during the transient.

Since the primary-to-secondary heat transfer is the driving force for the RCS cool-down and depressurisation, the SG inventory is maximised to provide the largest cool-down capacity. An initial fluid inventory of 57 320 lbm (26 000 kg) was assumed. One can obtain the desired mass by either decreasing the aspirator flow until the downcomer quality is just saturated, or adjusting the initial void fraction in the bundle region of the SG. In addition to the initial inventory, the mass of the feedwater present between the feedwater isolation valve and the downcomer of the broken steam generator, which was calculated to be 16103 kg, is modelled and contributes to the overcooling and depressurisation of the RCS. For the purposes of this benchmark the additional feedwater is modelled as an extended boundary condition of feedwater rate vs. time shown in Figure 4. Additional details are available from the benchmark final specification [5].

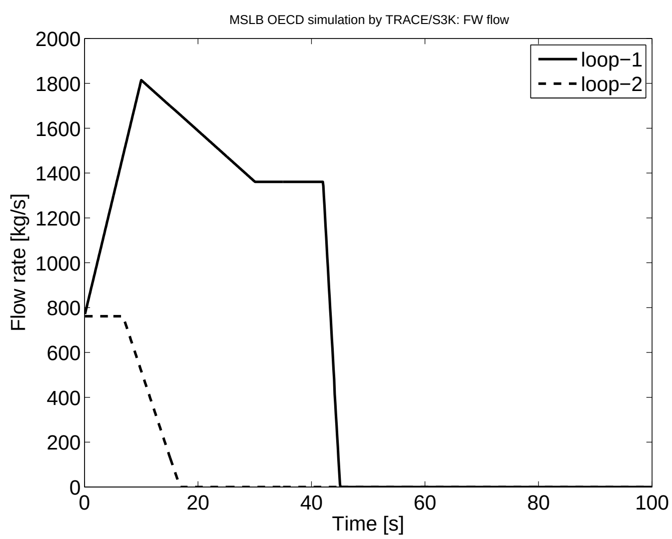

Figure 4   Steam generator feedwater flow rate.

3. Results

One of the most important parameters that might affect the MSLB transient is the secondary side fluid inventory of the SGs. In order to comply with the benchmark specification, this value was fitted by slightly modifying the SG secondary side geometry (volumes of downcomer and boiler). The resulting initial steady-state conditions are given in Table 1, together with the benchmark specifications. The initial distribution of the average axial core power profile is plotted in Figure 5. The black line is the mean curve from all the benchmark participants [15]. The TRACE/S3K predicts a flatter power profile, however within of the overall spreading of participants' results. The TRACE/S3K steady-state parameters are close to the specifications. The transient simulation was carried out employing the "flat" coupling option.

Table 1 Initial conditions.

Parameter	TRACE/ S3K	MSLB data	Δ, %
Power, MW	2772	2772	0.00
Core flow, kg/s	16167	16052	0.71
Bypass flow, kg/s	1480	1550	-4.74
Lower plenum pressure, MPa	15.37	15.36	0.04
Upper plenum pressure, MPa	15.17	15.17	-0.16
Hot leg temperature, K	590.6	591.4	-0.13
Cold leg temperature, K	563.3	563.8	-0.09
Pressurizer level, m	5.61	5.59	0.32
Feedwater/steam flow per SG, kg/s	761.6	761.6	0.00
Feedwater temp., K	510.9	510.9	0.00
SG exit pressure, MPa	6.41	6.41	0.00
SG initial mass, kg	25988	26000	-0.05
keff	1.0037768	1.0039	0.00

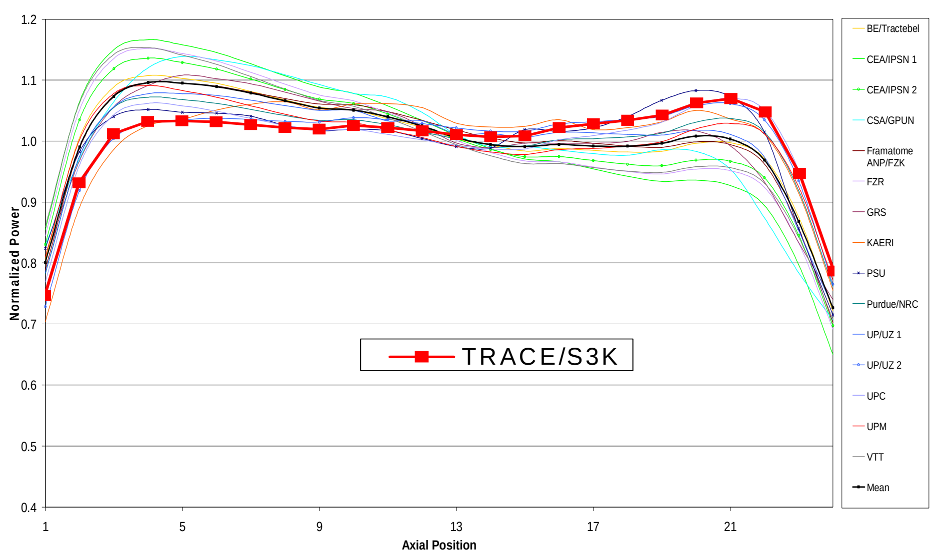

Figure 5 Initial average core power profile.

The sequence of the main events predicted by TRACE/S3K is given in Table 2. The transient starts with simultaneous opening of 24- and 8-inch breaks of main steam line 1 at the tie with the cross-connection line at 0.01 s. It results in a fast depressurization of the secondary side of the broken loop 1 and, consequently, in a rapid cooling of the primary coolant system. A large amount of fluid is lost through the breaks. The fluid outflow through the 24-inch break calculated by TRACE/S3K is shown in Figure 6, together with the results obtained by the 14 benchmark participants. The black line in the figure represents the mean result. There is a noticeable discrepancy between the TRACE/S3K break flow and the mean value at 40-50 s of the transient, where TRACE/S3K over-estimates the break flow in comparison to the benchmark participants. However, some participants predicted a similar peak flow rate, though at a different time instant. In order to simplify the data comparison, the S3K/TRACE results presented in Figure 7 to Figure 12 are compared only against the benchmark mean values. As it was mentioned previously, the mean values are based upon a statistical mean of all submitted values, corrected to account for the inter-dependence of some of the results [6].

Table 2. Main events sequence

Event	Time, s
Break opens	0.01
Overpower (114%)	6.37
Control rod insertion	6.77
Turbine isolation valve loop-2 closed	7.27
Loop 2 safety relief valve opens	7.49
Loop 2 safety relief valve closed	44.21
HPI starts	44.23
Max. power after SCRAM (8.42%)	71.40
End of transient	100

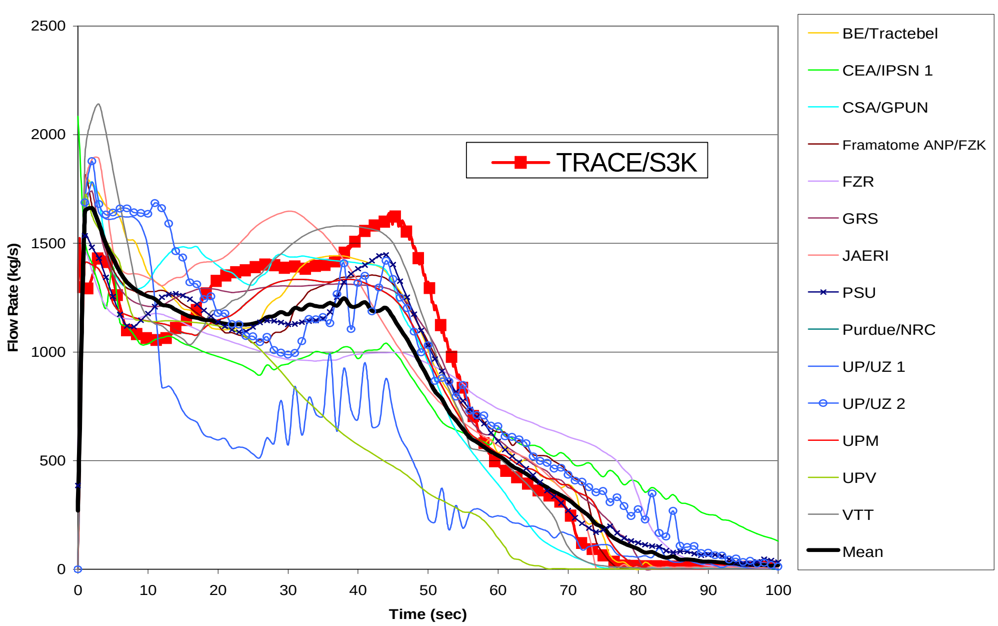

Figure 6 Flow rate for 24-inch break.

Despite the discrepancy in the break flow, the total SG fluid inventories are close to the mean values, as reported in Figure 7. Because of the intensive evaporation, the heat removal over SG-1 is very effective and much larger than that over the intact SG. The significant cooldown of the affected loop results in a considerable difference between the cold and hot leg temperatures of intact and broken loops (Figure 8 and Figure 9). After 60 s of transient the affected SG becomes almost empty. As a consequence, the coolant cold (and hot) leg temperature of the affected loop start increasing. The loop pressure follows the decrease of the average coolant temperature, as plotted in Figure 10.

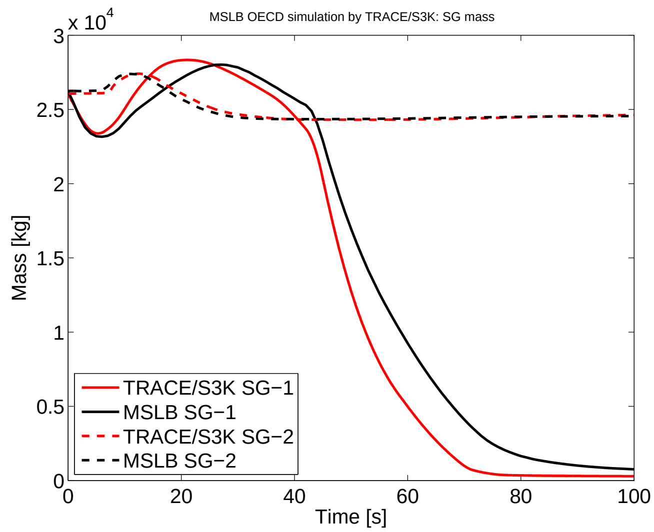

Figure 7   Steam generator mass.

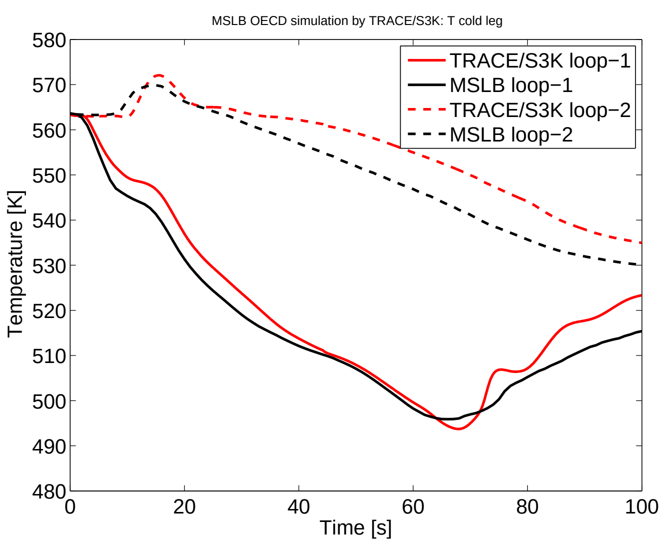

Figure 8   Cold leg temperature.

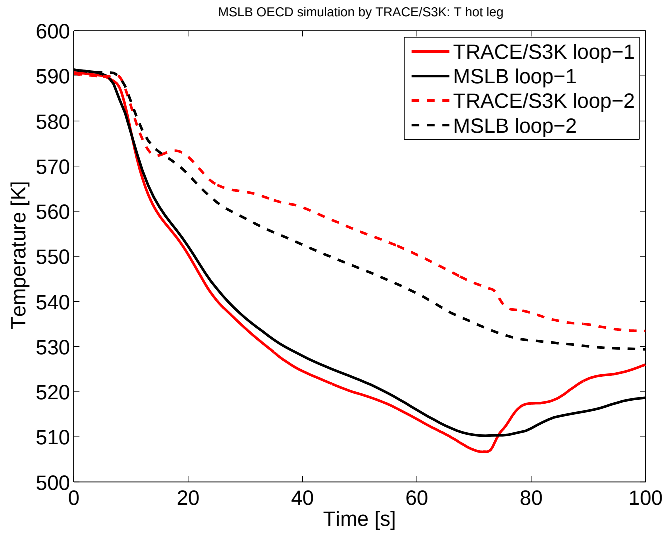

Figure 9 Hot leg temperature.

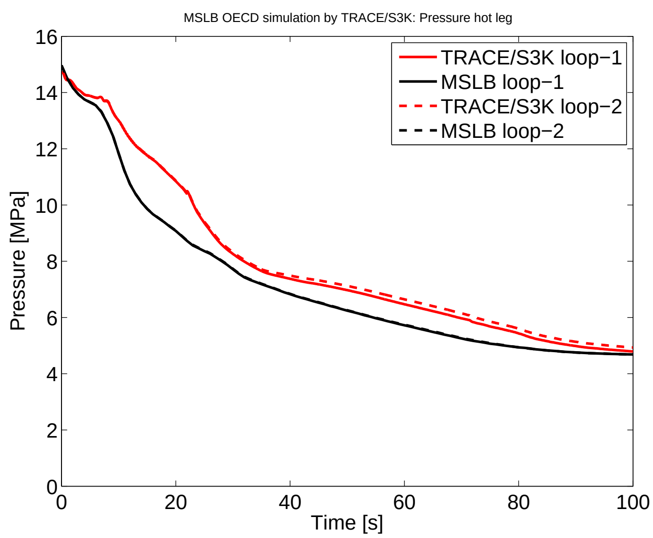

Figure 10 Hot leg pressure.

The reactor trip occurs at 6.77 s. The core peak power reaches 3238 MW (Figure 11). The mean benchmark peak power is 2993 MW, with the results from the benchmark participants ranging from a minimum of about 3059 to a maximum of about 3294 MW. The mean benchmark peak power is lower than the minimum predicted power peak because the peak power times are different among the benchmark participants. It also should be noted that the benchmark mean data are available with a 1 second increment only, so that the power peaks are likely to be hidden between two adjacent values. The total reactivity is very close to the benchmark mean reactivity, as shown in Figure 12. The peak reactivity after the scram is predicted to occur at approximately 70 s, similar as for the mean benchmark data.

The asymmetric core cooling yields an asymmetric increase of the reactor power. The part of the core that is close to the affected loop experiences a higher power increase, as illustrated in

Figure 13. Here the difference is presented between the 2D power distribution at the instant when the peak power is reached, and the initial power distribution. The maximal fuel assembly power reaches 165% of the initial average value.

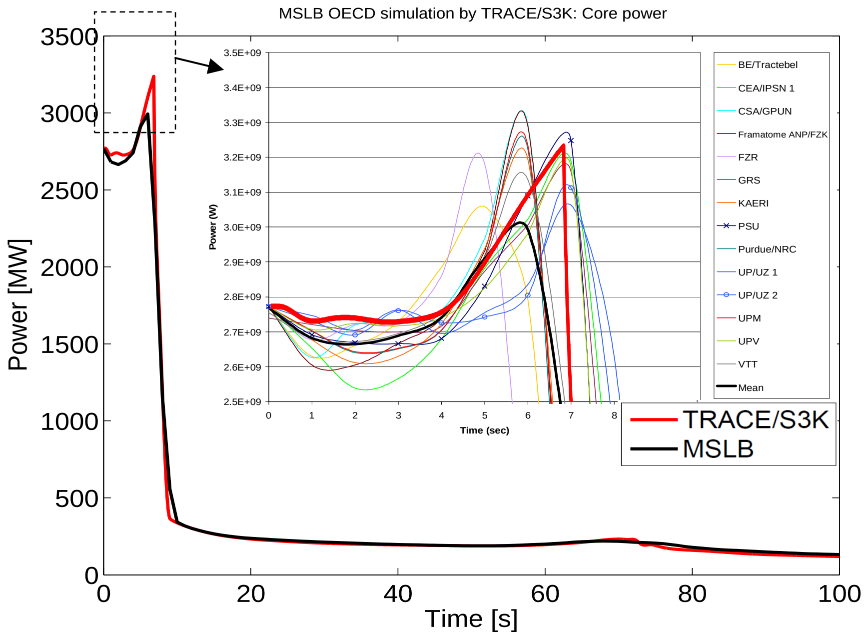

Figure 11   Core power.

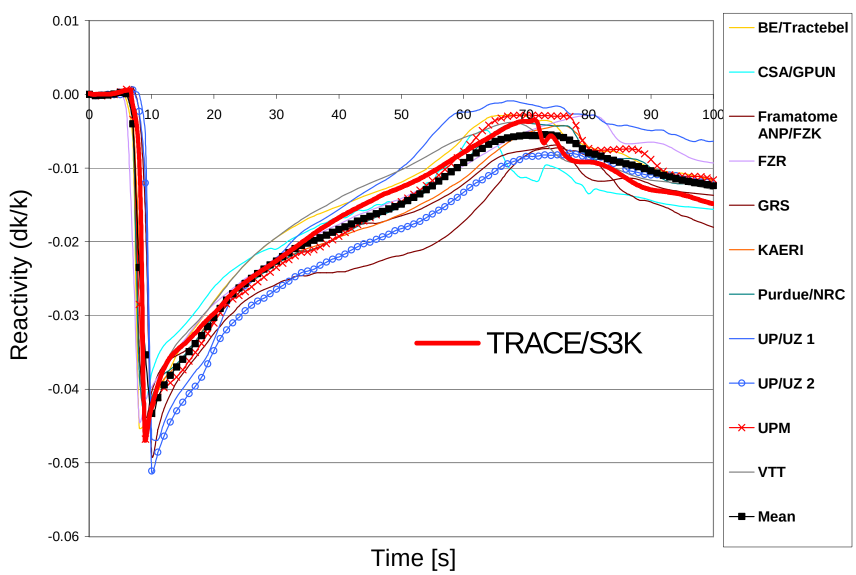

Figure 12   Total reactivity.

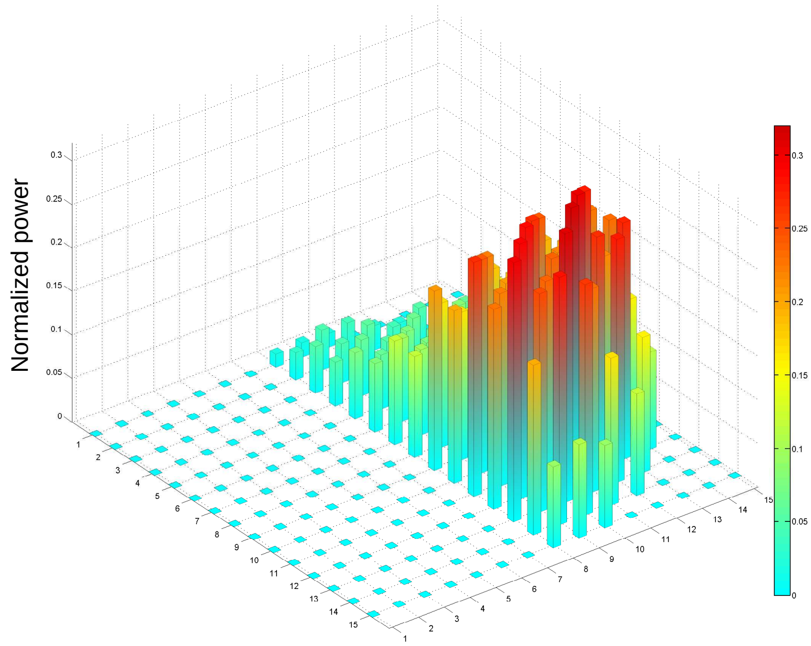

Figure 13 Normalized power increase at the instant when the peak power is reached (half of the core is shown).

4. Conclusion

The recently developed coupled code TRACE/S3K was verified by simulating the OECD/NEA MSLB PWR benchmark. The comparison with the benchmark data shows that the TRACE/S3K code is able to satisfactory reproduce the main transient parameters, namely, the power and reactivity history, change of the steam generator inventory, cold leg overcooling, and pressure response. In the near future, the TRACE/S3K code will be further validated against Swiss reactors plant data.

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