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VALIDATION AND APPLICATION OF TRACE FOR TRANSIENT ANALYSIS OF A GERMAN BWR PLANT

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Abstract

The Karlsruhe Institute of Technology (KIT) is participating on CAMP to validate TRACE for LWR transient analysis. The validation of safety-relevant heat transfer models of TRACE, data BFBT bundle tests are available for institutions taking part on international OECD/NEA Benchmarks. In the first step, the BWR-relevant models of TRACE were validated using experimental data. Then, an integral plant model of a German BWR of Type-72 was developed using the multidimensional VESSEL-component of TRACE to simulate a plant events such as the TUSA. In this paper, the validation work, the plant modeling and selected results will be presented and discussed.

1. Introduction

The use of validated numerical simulation tools for the analysis of the plant response under off normal conditions is mandatory. In the framework of the Code Application and Maintenance Program (CAMP) of the US NRC the TRACE code is being validated for LWR safety investigations [1, 2]. The Karlsruher Institute of Technology (KIT) is participating on CAMP and performing validation work for LWR and innovative reactor applications. For the validation of safety-relevant TRACE heat transfer models, data from different bundle tests such as the NUPEC BFBT and PSBT test are available from the international OECD/NEA Benchmarks [3]. Very important data for the validation of models related to single and two-phase flow pressure drop, void fraction, burnout and DNB are accessible to benchmark participants. For the validation of void models of the CHAN-component in TRACE, data from several void fraction steady state tests performed at the BFBT (BWR Full-Size Fine-Mesh Bundle Test) facility were simulated by TRACE [4]. In addition, 69 pressure drop tests and 151 critical power steady state tests were investigated. In addition, plant data from BWR plant events are used by KIT for the overall TRACE validation.

The assessment of the pressure drop, void fraction and critical power is essential for BWR since any change in the thermal hydraulic conditions of the core will impact the neutron moderation and population within the core. Moreover any pressure change in the core will lead to a change of the void fraction distribution and by this means the core power will increase or decrease. The validity of the thermal hydraulic models of a safety analysis tool like TRACE needs to be checked against experimental data gained in single effect, bundle or integral tests.

In this paper the details of the BFBT tests and of the experiments performed for the validation of BWR-relevant models will be initially presented. Then, the TRACE modelling of the BFBT test will be described highlighting the parameter ranges and types of measured data available. A discussion of the comparison of TRACE predictions with BFBT test data will then follow. The TUSA event occurred in a German BWR plant will be briefly discussed followed by a discussion of the plant model developed for TRACE to simulate the TUSA event. Finally a comparison of the TRACE calculations with selected measured data of the plant is given and the main results are discussed.

2. Validation of TRACE for BWR Applications

2.1 Short description of BFBT-Tests

The BFBT (BWR Full-Size Fine Mesh Bundle Test) facility from NUPEC (Nuclear Power Engineering Cooperation) in Japan has been used for the measurement of void fraction and critical power for typical BWR reactor conditions [3]. Experiments covering a wide range of BWR-operating conditions (max. pressure of 10.3 MPa, max. liquid temperature of 588.15 K, max. power of 12 MW and a max. flow rate of 20.83kg/s) can be performed. In the test section of the facility, representative fuel assembly with different fuel rod arrangements and water rods can be assembled. The fuel rods are electrically heated rods (simulator) with the full-length of a BWR fuel assembly. The test section is shown in Figure 1and consists of a pressure vessel, electrodes and a flow channel which simulates a BWR fuel bundle. The fuel rod simulator consists of a heater (Nichrome) of 3.65 mm outer diameter, an insulator (Boron nitride) of 4.85 mm and the cladding (Inconel 600) of 6.15 mm outer diameter. The heated length is 3.708 m height.

The NUPEC BFBT tests were focused on the investigation of pressure drop for single and two phase flow situations, void fraction (steady state and transient) as well as critical power (steady state and transient) for different BWR assembly arrangements, radial and pin power distributions and bundle axial power profiles.

The differential and absolute pressures were measured using diaphragm transducers located at different axial locations as shown in Figure 2. Two different systems were used in BFBT to measured the averaged void fraction at three axial bundle elevation (X-ray densitometers) and to measure the detailed void distributions at radial plane located at the upper bundle part (X-ray CT scanner). The critical power was measured using thermocouples distributed at radial planes located at four axial positions in the upper part of the test section, where burn out is expected to occur. The critical power was measured by slowly increasing the bundle power while monitoring the individual heater thermocouple signals. The critical power defined by the benchmark team is reached when the peak simulator surface temperature became 14 K higher than the steady state temperature level. The inlet flow rate was measured using turbine flow meter. In the heater rods, the surface temperature was monitored at positions just upstream of the spacers by the 0.5 mm diameter chrome-alumel thermocouples, which were located in the heater rod cladding. In Table 1 the estimated accuracy of the measured parameters is given. There were three types of void fraction measurements: the sub-channel-averaged void fraction (averaged over more than 400 pixel elements), the local void fraction measured on a 0.3 mm × 0.3 mm square pixel element, and the cross–sectional averaged void fraction (averaged over more than 10^5 pixel elements). The accuracy of these void fraction measurements depends on the photon statistics of the X-ray source, the detector non-linearity, and the accuracy of the known fluid condition (temperature and pressure) measurements

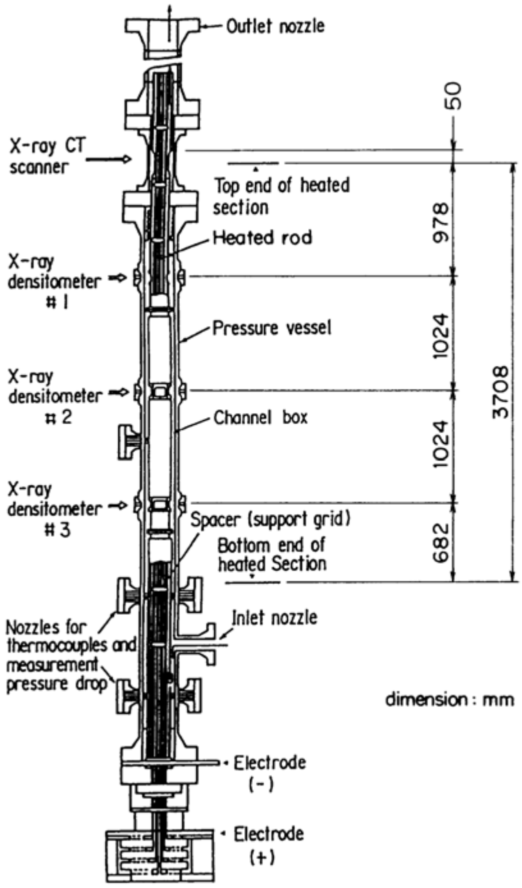

Figure 1 Vertical cut of the NUPEC BFBT test facility

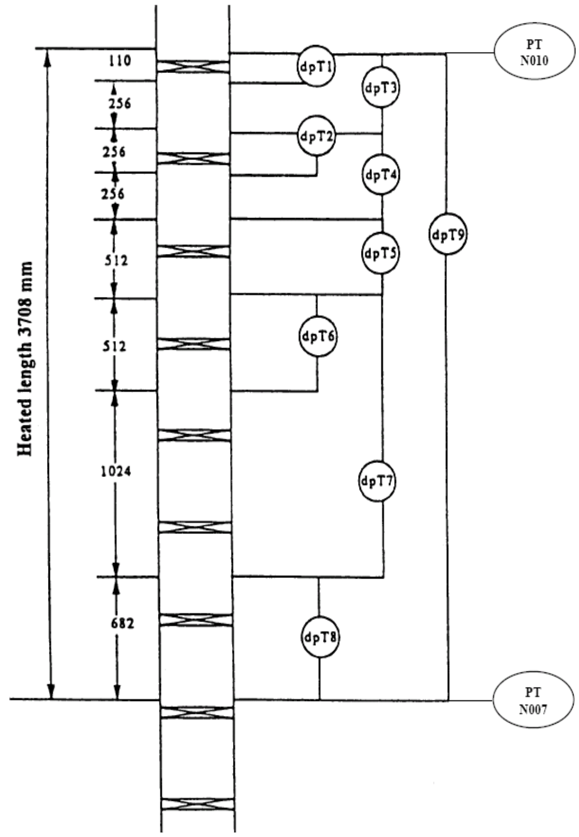

Figure 2 Locations of the pressure drop measurements

Table 1 Estimated accuracy of the measured parameters in BFBT test

	Quantity	Accuracy
Generals	Pressure	1%
	Flow	1%
	Power	1.5%
	Inlet fluid temperature	1.5 Celsius
X-ray CT scanner	Sub-channel void fraction measurements	3%
	Cross-sectional void fraction measurements	2%
	Spatial resolution	0.3 mm x 0.3 mm
	Scanning time	15 seconds
X-ray densitometer	Sampling time	Max. 60 seconds

A detailed description of the test series for the pressure drop, void fraction and critical power measurement can be found in the BFBT benchmark description [3].

2.2 TRACE Modeling for post-test simulation of BFBT test

For the simulation of the large number of tests devoted to the measurements of different quantities using different thermal hydraulic parameters, power profiles and fuel assemblies geometries (no water rods, one water rod, two water rods, different number of simulator rods) TRACE models were developed for each test series characterized by the same geometrical arrangement. Then with the help of Perl or Python scripts the large number of tests was simulated by changing the initial and boundary conditions of each test automatically. Hereafter a TRACE model for the simulation of the experiment number 1071-53 will be described as representative for all other tests to avoid repetition.

The TRACE modelling is focused on the BFBT test section only i.e. the heated zone (heater and water rods) and the lower and the upper plenum, where the boundary conditions of each test are defined. For the modelling of bundle part, the BWR-specific component (CHAN) is used. The bundle conditions at the inlet and outlet are represented in TRACE by the FILL (inlet mass flow, inlet temperature) and the BREAK (outlet pressure) component. The CHAN components allows a very detailed representation of each simulator, water rods and channel box taking into account the power of each simulator. In Figure 3 the CHAN model of the 8x8-2 BWR bundle is exhibited, where each different colour of the simulator indicates a different radial power. The two gray rods are the water rods. In Figure 4 the TRACE representation of the whole test section is shown indicating the axial nodalization (24 nodes) as well as the boundary conditions at the inlet and outlet: FILL (Number 100) and BREAK (Number 300). The seven spacer grids are modelled by an additive pressure drop at the particular positions. Each simulator rod is subdivided in 22 radial mesh points to catch the radial temperature distribution. The bundle power in all heaters (simulator rods) is defined in the POWER-component of TRACE.

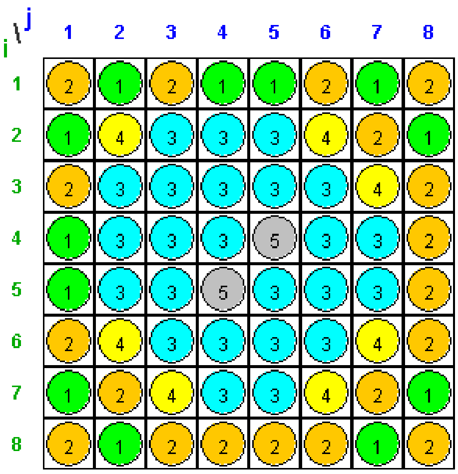

Figure 3 CHAN representation of the BFBT Bundle **Figure 4** TRACE representation of the heated test section

2.3 Selected results of the TRACE simulations

A detailed description of the assessment of all pressure drop, void fraction and critical power BFBT tests investigated with TRACE can be found in [5]. Hereafter only selected results demonstrating the validation of the BWR-models will be presented.

2.3.1 Pressure drop tests

In Figure 5 and Figure 6 a comparison of TRACE predictions with the experimental data for both single and two phase flow pressure drop measurements is shown. It can be seen that TRACE tends to under predict the single phase pressure drop in the whole pressure range but the deviations are within the 15 % of margin error. On the contrary for the two phase flow experiments, TRACE predictions are closer to experimental values except for few cases. Here almost all predictions are within the 10 % margin of error along the whole bundle elevation.

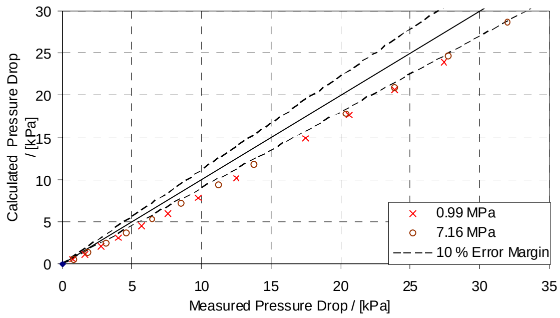

Figure 5 Comparison of predicted and measured single phase pressure drop in BFBT tests

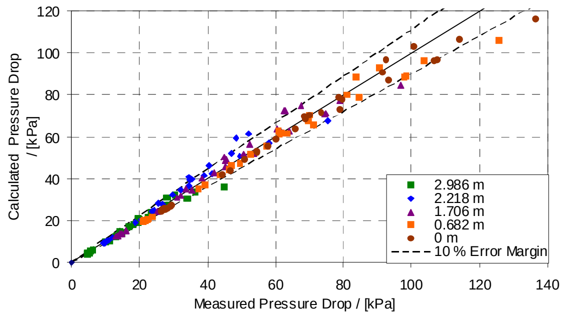

Figure 6 Comparison of predicted and measured two phase pressure drop in BFBT tests

2.3.2 Void fraction tests

Many void fractions tests were simulated with TRACE and the results have been compared to the measured data at four axial bundle elevations. In Figure 7 and Figure 8 predicted void fraction at the bundle outlet and at the upper bundle part is compared to the experimental data of the test series (0-1, 0-2, 0-3, 1, 2, 3, and 4). It can be observed that the majority of the predictions are within the 10 % error band, except for few tests. The deviations of the TRACE predictions compared to the data become larger for the lower bundle levels Figure 8. Based on sensitivity studies, the influence of the four input parameters such as outlet pressure, outlet quality, flow rate and inlet sub-cooling was investigated. It confirmed that the TRACE predictions are worse for low quality and mass flow conditions [5] since these conditions are not completely in the validation range of correlations.

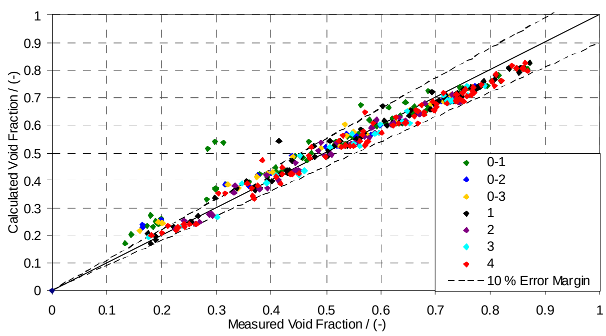

Figure 7 Comparison of predicted and measured void fraction results (3.758 m)

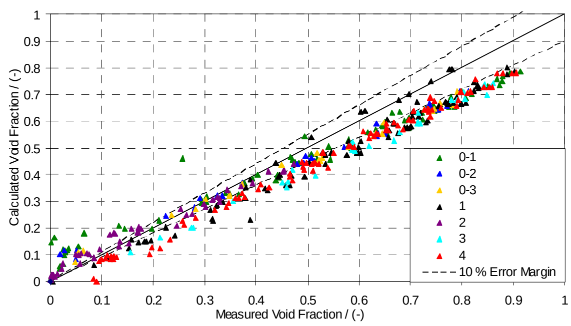

Figure 8 Comparison of predicted and measured void fraction results (2.730 m)

2.3.3 Critical power tests

In case of the BFBT critical power tests, TRACE tends to over predict the measured data. But the root square mean error (RSME) is below 0.82 MW. The comparison of the predicted (C) and measured (M) critical power is given in Figure 9. There, three regions can be distinguished: (1) for critical power below 4 MW most of the tests are over predicted (2) for critical power between 5 and 6.5 MW TRACE under predicts the data but the calculated values are inside the 10 % error band (3) for critical powers above 7 MW TRACE over predicts the measured data but a large number of predictions are within the 10 % error band. It has to be noted that around the pressure of 7.2 MPa the predicted critical powers are different since the power profile of the assemblies C2A and C2B (cosine shaped) are different from that of C3 (skewed peak shape), see Figure 10. In addition, C2A and C2B have different radial power profiles. TRACE over predicts the measured critical power in the pressure range of 5.5 MPa to 8.6 MPa. Finally Figure 11 indicates that for low mass flux conditions (< 500 kg/m²s) the over prediction of TRACE is between 20 and 35 % while for larger mass fluxes the predictions are within the 10 % error margin [6].

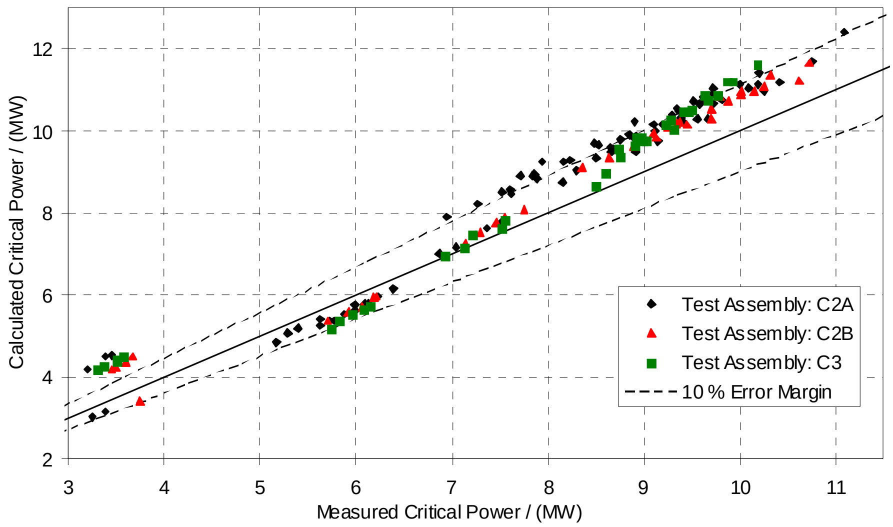

Figure 9 Comparison of calculated with measured critical power for different bundle arrangements

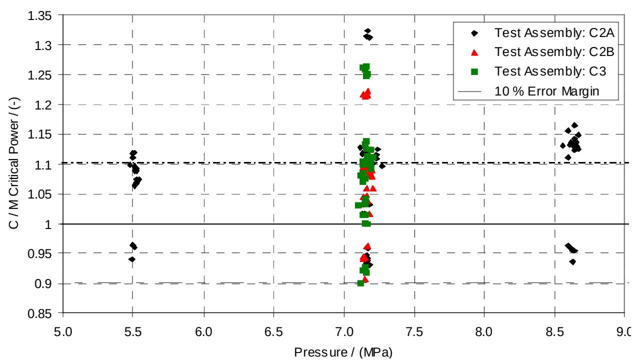

Figure 10 C/M ratio as function of the bundle pressure

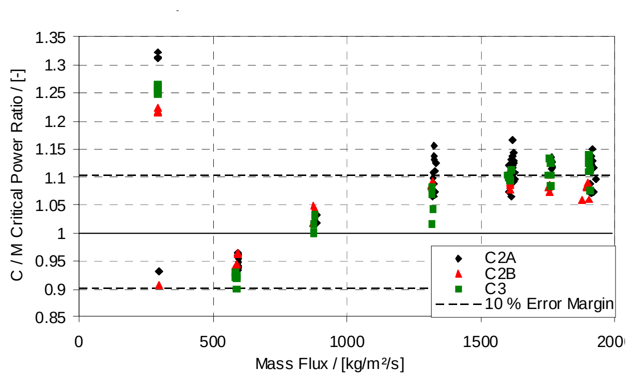

Figure 11 C/M ratio as function of bundle mass flux

3. Application of TRACE for the Simulation of a TUSA Event

For the application of TRACE to simulate BWR transients it is important to demonstrate that TRACE is working properly by the post-test analysis of real plant events. To do so, a turbine trip (TUSA) event in the German BWR type-72 plant that happened in 1998 (cycle 13) was selected for the analysis with the code TRACE using point kinetics model. First of all, the TUSA event will be described together with the initial and boundary conditions; followed by the description of the developed integral plant model.

3.1 Description of the TUSA event

The TUSA event was initiated by the erroneous activation of the condenser alerter when the pressure was above 0.3 bar [7]. In reality, the condenser pressure was never above 0.145 bar. As a consequence the reactor power was reduced from the nominal value (3840 MWth) to about 35 % of nominal power by partial insertion of control rods and reduction of the rotational speed of eight main recirculation pumps (MRP) to almost minimal value (600 U/min). In addition one MRP was shut down due to unknown reasons. Furthermore four groups of the safety relief valves were manually opened for a short time to ameliorate pressure increase in the steam line. It has to be noted that after the turbine stop valve (TSV) started to get close the turbine bypass valve (TBV) started to open. But the diameter of the TBV (bypass line) is smaller than the one of the TSV (steam line). Consequently a pressure increase was propagated from the steam line to the reactor pressure vessel leading to a void collapsing and hence to a power increase. But since the mass flow rate through the core is considerably reduced due to the MRP speed reduction, more void is generated in the core leading to a power decrease.

3.2 Description of the integral BWR plant model

The reference plant is a German BWR of type-72 consisting of eight internal recirculation pumps (MRP). Four steam lines and feed-water lines are connected to the reactor pressure vessel. In the core 784 fuel assemblies of uranium oxide (UO_2) and mixex oxide fuel (MOX) were loaded in the cycle 13.
An integral plant model was developed for TRACE using the three-dimensional VESSEL component for the representation of the reactor pressure vessel (RPV), the CHAN component for the fuel assemblies, the SEPT component to model the separators and dryers in eight groups, the PUMP component to model the MRPs as well as various PIPE components for the representations of the steam and feedwater lines. The VALVE component was used to model the safety relief valves (SRV) and the TSV and TBV. Finally the BREAK and FILL components were used to define e.g. the turbine (pressure boundary conditions) and the feed water injection (mass flow rate and temperature). The POWER component using the point kinetics option was selected to describe the power change during the simulation of the transient.
The RPV was subdivided into 22 axial nodes taking into account the constructive peculiarities of the internals below and above the core as well as in two rings and one azimutal sector (2D model). In Figure 12 the representation of the integral BWR plant model is given. More details of this model are given in [8]. There it was shown that a 3D model and a 2D model predict comparable results for the TUSA event since no unsymmetrical behaviour can be expected in the core, where normally three dimensional phenomena may plays a role. This will not be the case, if a transient with asymmetrical core behaviour is evaluated with both 2D and 3D models. In this model the core was represented by the CHAN component. The separators and dryers were modelled in eight SEPT-component each. Since the

dynamic response of the pumps do not play an important role during the TUSA, they were represented by a simplified model.

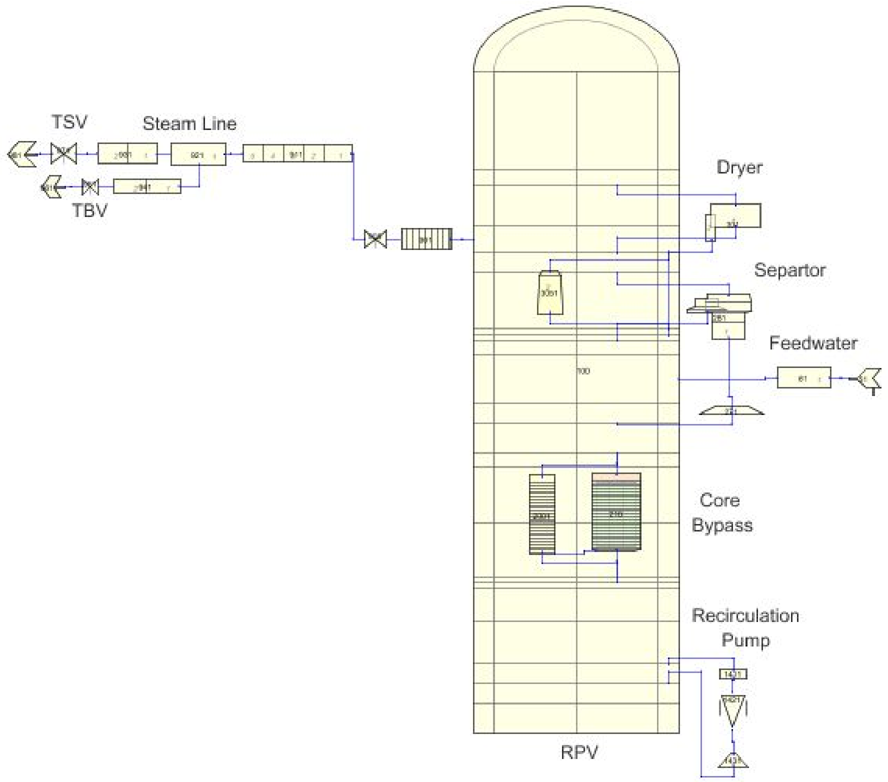

Figure 12 Integral BWR plant model of the BWR as represented by SNAP (2D model)

3.3 TRACE simulation of the TUSA event

Steady state simulation

Using the described TRACE model of the BWR plant, a steady-state simulation of the plant conditions just before the event occurred was simulated. It was shown that the predicted parameters were in a good agreement with the reference plant data. Most of the predictions were close to the reference values (deviations less then 5 %). Only the pressure drop predicted over the steam dryer showed the largest deviation (about 18 %), [8]. For example the water level within the RPV predicted by TRACE amounted 14.49 m compared to the reference value of 14.36 m.

Transient Simulation

Based on the good agreement obtained for the steady state BWR conditions, the integral model was extended to take into account the boundary conditions during the TUSA event such as the reduction of the recirculation velocity of the 8 MRPs, the opening and closure of the TSV and TBV after the initiation of the event. In addition the reduction of the MRP flow was also taken into account in the modelling of the TUSA event; see Figure 13 and Figure 14. Considering these boundary conditions the TUSA transient was simulated with TRACE using the point kinetics model.

The evaluation of the simulation has shown that the TUSA progression was dominated mainly by the two competing effects, namely the void reactivity and the Doppler coefficient, in the short term and by the behaviour of the recirculation pumps in the long term. The magnitude of the void effect was much larger than the one of the fuel temperature increase. As expected, after the TUSA a sharp void collapsing was predicted. This was caused by the pressure wave propagation from the steam line to the core, and it lead to a pressure spike shown in Figure 15.

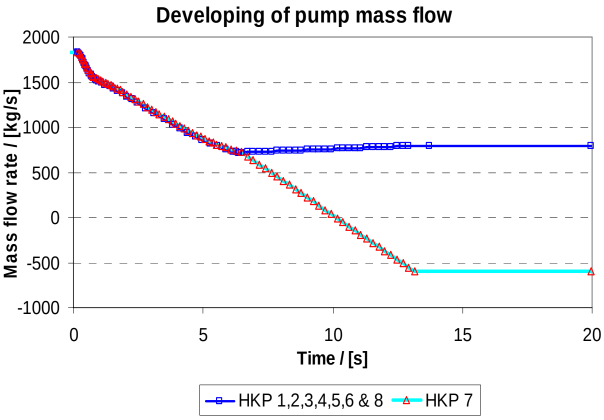

Figure 13 MRP mass flow rate during the transient

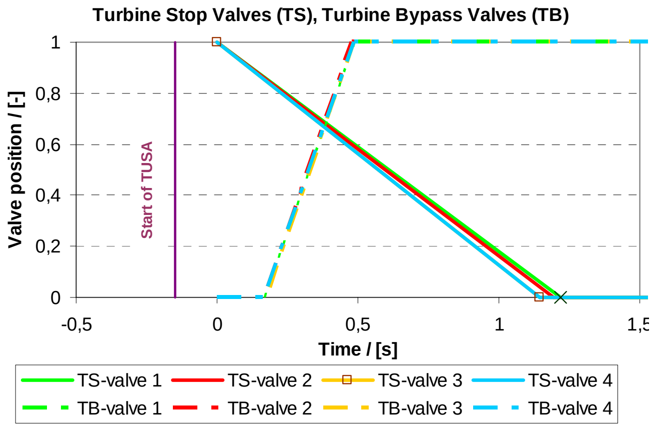

Figure 14 Dynamic of the closing and opening of TS and TB

As a consequence the reactor power increased rapidly due to the increased moderation of the neutrons in the core. Then, the power increase was stopped by the increased void generation in the core as a consequence of the reduction of the core mass flow rate, (see Figure 13) stabilizing after 50 s at a much lower power level (~ 40 %) than the nominal one, as shown in Figure 16. The reactor approached stationary conditions at around 300 s.

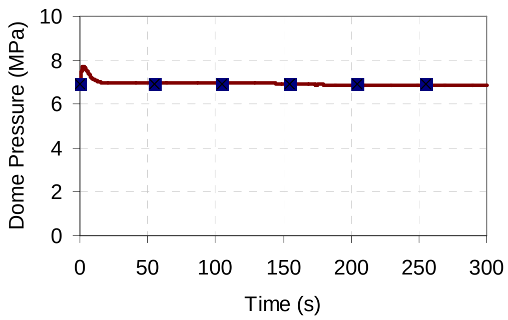

Figure 15 Comparison of predicted and measured dome pressure during transient

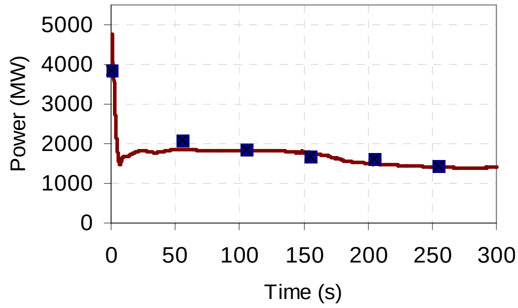

Figure 16 Comparison of predicted and measured reactor power during transient

Further sensitivity analysis was performed to find out the most important parameters influencing the progression of this TUSA event; specifically the reactor power, the dome pressure and the water level inside the RPV. To do so, the delay time for the opening (TSV) and closing (TBV) of the steam line valves were varied. It was found out that these parameters can influence the maximal water level as well as the pressure peak and power peak during TUSA. Also the uncertainty in the global reactivity coefficients will influence the response of these global parameters.

4. Summary and further work

The presented validation work using BFBT bundle data has shown that TRACE is appropriate to describe the main BWR phenomena. For the single and two phase pressure drop tests, TRACE tends to under predict the single phase pressure drop while the calculation of the two phase pressure drop agrees well with the measured data. A comparison of predicted void fractions for the different bundle arrangements with the data indicates that TRACE is able to predict reasonably well the void fraction at all axial measurement positions. The predictions are specifically for the bundle outlet close to the data.

The critical power tests were well-predicted with TRACE except for low mass fluxes conditions, where TRACE tends to over predict the critical heat flux. The post test analysis of the TUSA event with the 2D plant model using point kinetics demonstrated that the predictions are in good agreement with the recorded plant data.
Despite these encouraging results, further investigations are needed to improve the code's prediction capability e.g. for burnout under transient conditions. A detailed review of the models for the prediction of the critical power is still necessary. In addition, the qualification of the 3D RPV model of the BWR German plant needs to be performed using plant data for situations where the thermal behaviour of the core is asymmetrical. For such situations, the coupling of this 3D thermal hydraulic model with a 3D neutronic core model is needed. This work will be presented in a companion paper [9].

Acknowledgements

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