Hindawi Publishing Corporation
Science and Technology of Nuclear Installations
Volume 2012, Article ID 247482, 9 pages
doi:10.1155/2012/247482

Research Article

Validation and Application of the Thermal Hydraulic System Code TRACE for Analysis of BWR Transients

V. H. Sánchez,[1] M. Thieme,[2] and W. Tietsch[3]

[1] *Institute for Neutron Physics and Reactor Technology (INR), Karlsruhe Institute of Technology (KIT), 76344 Eggenstein-Leopoldshafen, Germany*
[2] *TÜV SÜD Industrie Service GmbH, Westendstraße 199, 80686 Munich, Germany*
[3] *Westinghouse Electric Germany GmbH, 68140 Mannheim, Germany*

Correspondence should be addressed to V. H. Sánchez, victor.sanchez@kit.edu

Received 13 April 2012; Accepted 7 August 2012

Academic Editor: Boštjan Končar

The Karlsruhe Institute of Technology (KIT) is participating on (Code Applications and Maintenance Program) CAMP of the US Nuclear Regulatory Commission (NRC) to validate TRACE code for LWR transient analysis. The application of TRACE for the safety assessment of BWR requires a throughout verification and validation using experimental data from separate effect and integral tests but also using plant data. The validation process is normally focused on safety-relevant phenomena for example, pressure drop, void fraction, heat transfer, and critical power models. The purpose of this paper is to validate selected BWR-relevant TRACE-models using both data of bundle tests such as the (Boiling Water Reactor Full-Size Fine-Mesh Bundle Test) BFBT and plant data recorded during a turbine trip event (TUSA) occurred in a Type-72 German BWR plant. For the validation, TRACE models of the BFBT bundle and of the BWR plant were developed. The performed investigations have shown that the TRACE code is appropriate to describe main BWR-safety-relevant phenomena (pressure drop, void fraction, and critical power) with acceptable accuracy. The comparison of the predicted global BWR plant parameters for the TUSA event with the measured plant data indicates that the code predictions are following the main trends of the measured parameters such as dome pressure and reactor power.

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 Applications and Maintenance Program (CAMP) of the US NRC, the TRACE code is being validated for LWR safety investigations [1]. An extensive validation of coupled neutron kinetics/thermal hydraulic codes is taking place worldwide in the frame of national and international benchmarks related to both pressurized and boiling water reactors (PWR and BWRs) using plant data such as the Peach Bottom turbine trip test [2, 3], the Oskarshamn-2 Instability event [4] and the Ringhals Loss of Feedwater Case [5]. The Karlsruhe Institute of Technology (KIT) is participating on CAMP program and performing validation work for light water reactors (LWRs) [6] and Generation 4 reactors [7]. For the validation of safety-relevant TRACE heat transfer models, data from different bundle tests such as the NUPEC BFBT and PSBT (PWR Subchannel and Bundle Test) tests are available from the international OECD/NEA Benchmarks [8, 9]. 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 facility were simulated by TRACE [10].

In addition, 69 pressure drop tests and 151 critical power steady-state tests were investigated [11]. Furthermore, plant data from BWR plant events are used by KIT for the overall TRACE validation [12, 13].

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

In this paper, details of BFBT tests and of the experiments used for the validation of the TRACE will be presented. The TRACE modelling of the BFBT test will be described indicating the parameter ranges and types of measured data available. A discussion of the comparison of TRACE predictions with BFBT test data will follow. The turbine trip event occurred in a German BWR plant will be briefly discussed followed by a description of the plant model developed for TRACE to simulate the turbine trip event. Finally, a comparison of 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 facility from NUPEC in Japan has been used for measuring the void fraction and critical power for typical BWR reactor conditions [8]. 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 20.83 kg/s) can be performed. In the test section of the facility, representative fuel assemblies with different fuel rod arrangements and water rods can be arranged. The fuel rods are electrically heated rods (simulator) with the full length of a BWR fuel assembly. The test section is shown in Figure 1 and 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 measure the averaged void fraction at three axial bundle elevations (X-ray densitometers) and to measure the detailed void distributions at radial plane located at the upper bundle part (X-ray CT scanner). The rapid temperature escalation when critical power conditions are achieved was measured using thermocouples distributed at radial planes located at four axial positions in the upper part of the test section, where burnout is expected to occur. The critical power was measured by slowly increasing the bundle power while monitoring the individual heater thermocouple signals.

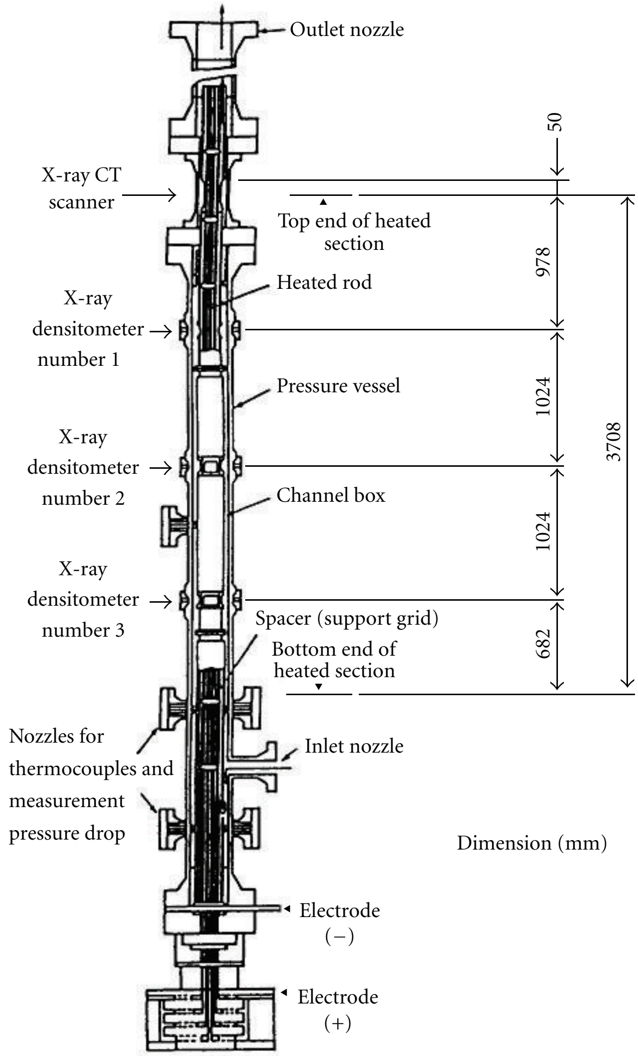


Figure 1: Vertical cut of the NUPEC BFBT test facility.

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. 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 nonlinearity, and the accuracy of the known fluid condition (temperature and pressure) measurements.

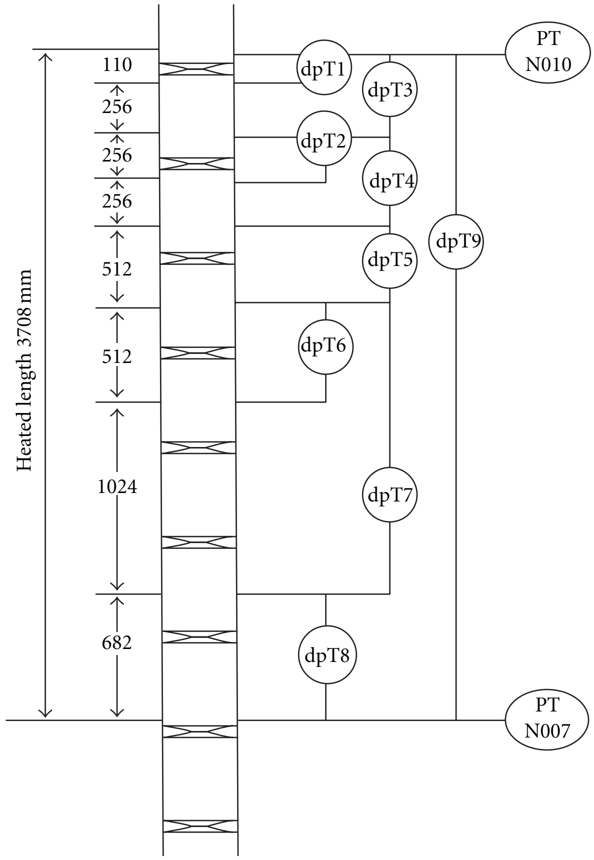


Figure 2: Locations of the pressure drop measurements.

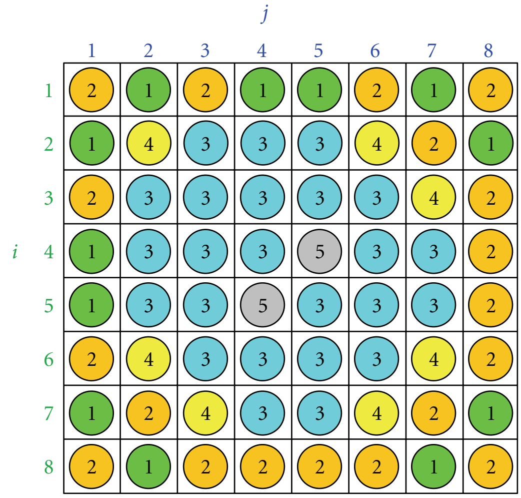


Figure 3: CHAN representation of the BFBT bundle.

Table 1: Estimated accuracy of the measured parameters in BFBT test [8].

	Pressure	1%
	Mass flow rate	1%
	Power	1.5%
	Fluid temperature at bundle inlet	1.5°C
X-ray CT scanner	Subchannel void fraction measurements	3%
X-ray CT scanner	Cross-sectional void fraction measurement	2%
X-ray CT scanner	Spatial resolution	0.3 mm × 0.3 mm
X-ray CT scanner	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 [8].

2.2. TRACE Model for Post Test Simulation of BFBT Test. TRACE models have been developed for simulating a large number of tests series characterized by the same geometrical arrangement devoted to the measurements of different quantities using and different thermal hydraulic parameters, power profiles, and fuel assembly geometries (no water rods, one water rod, two water rods, and different number of simulator rods). Input decks for the large number of tests were automatically generated using Python scripts taking into account the specific initial and boundary conditions of each test. 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 [10] is focused on the BFBT test section only, that is, the heated zone (heater and water rods) and both lower and upper plenums, where the boundary conditions of each test are defined. For modelling of the bundle part, the BWR-specific CHAN component is used. The bundle conditions at the inlet and outlet are represented in TRACE by the FILL (inlet mass flow and inlet temperature) and the BREAK (outlet pressure) components. The CHAN components allow 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 $8 \times 8 - 2$ BWR bundle is shown, 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. In Figure 3, there are four rod groups (1 to 4) characterized by a different relative radial power, and the two rods number 5 represent the water rods.

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 [10]. Hereafter only selected results demonstrating the validation of the BWR models will be presented.

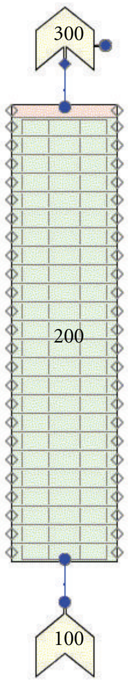


FIGURE 4: TRACE representation of the heated test section.

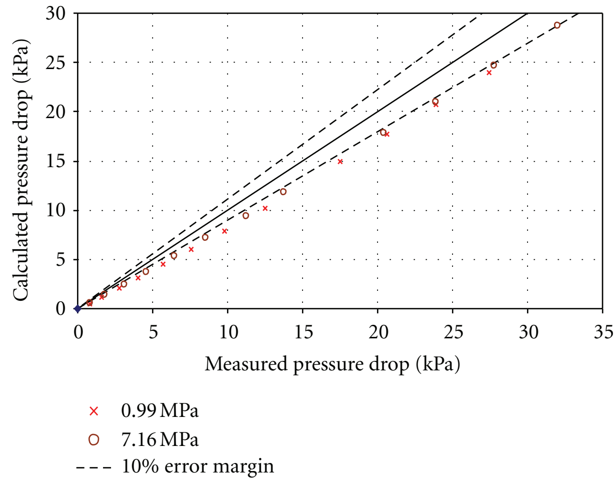


FIGURE 5: Comparison of predicted and measured single-phase pressure drop in BFBT tests.

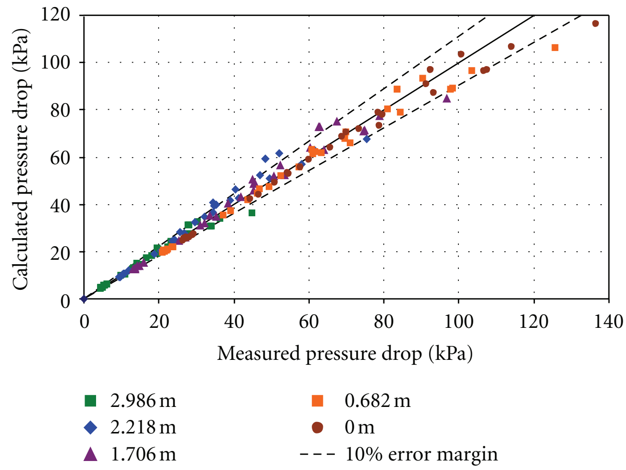


FIGURE 6: Comparison of predicted and measured two-phase pressure drop in BFBT tests.

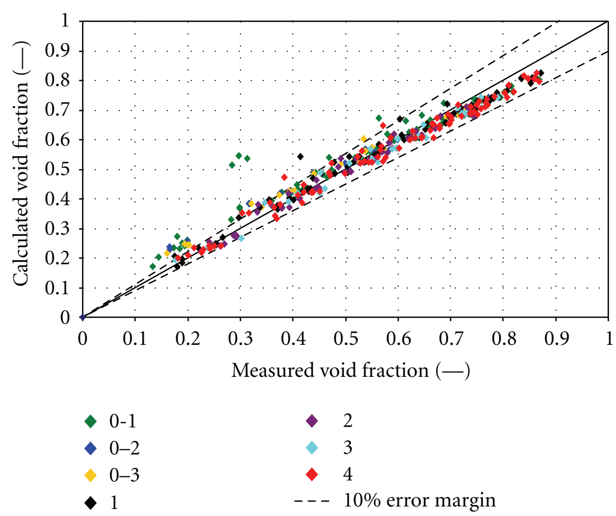


FIGURE 7: Comparison of predicted and measured void fraction results at 3.758 m.

2.3.1. Pressure Drop Tests. In Figures 5 and 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 underpredict the single-phase pressure drop in the whole pressure range, but the deviations are within the 10% of margin error. On the contrary for the two-phase flow experiments, TRACE predictions are closer to experimental values except for few cases. Almost all predictions are within the 10% margin of error along the whole bundle elevation.

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 Figures 7 and 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 subcooling was investigated [10]. It confirmed that TRACE predictions are worse for low-quality and mass flow conditions [10] since these conditions are not completely in the validation range of correlations.

2.3.3. Critical Power Tests. In case of the BFBT critical power tests, TRACE tends to overpredict the measured data. Nevertheless the root square mean error (RSME) is below 0.82 MW [11]. The comparison of the predicted (C) and

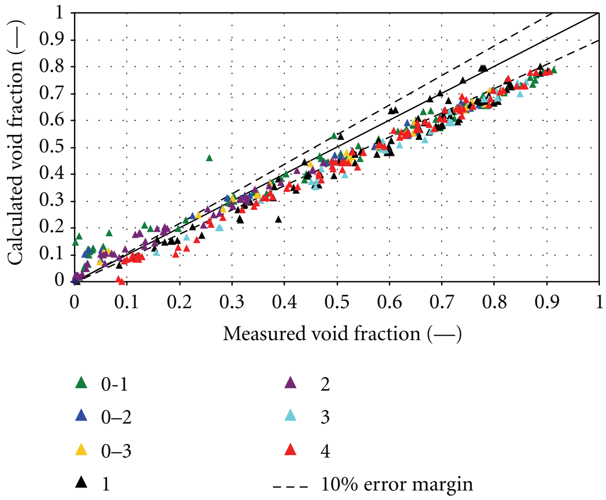


Figure 8: Comparison of predicted and measured void fraction results at 2.730 m.

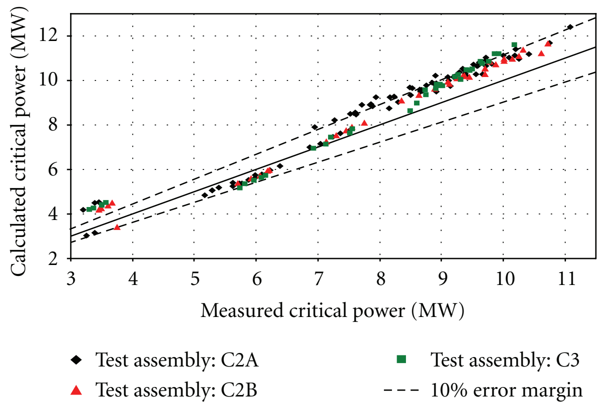


Figure 9: Comparison of calculated with measured critical power for different bundle arrangements.

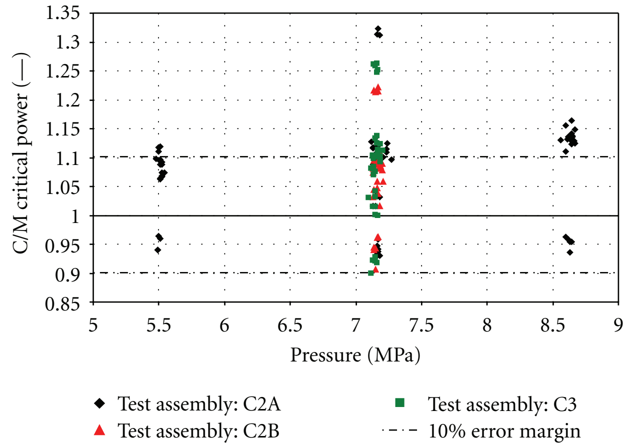


Figure 10: C/M ratio as function of the bundle pressure.

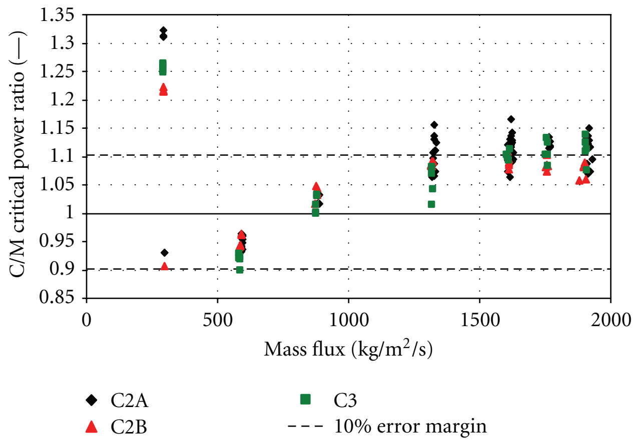


Figure 11: C/M ratio as function of bundle mass flux.

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 underpredicts 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) is 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/m^{2}s) the overprediction of TRACE is between 20 and 35% while for larger mass fluxes the predictions are within the 10% error margin [11].

3. Application of TRACE for the Simulation of a Turbine Trip Event

For the application of TRACE to simulate BWR transients it is important to demonstrate that TRACE simulates properly real plant events. A turbine trip event in the type-72 German BWR plant that happened in 1998 (cycle 13) was selected for the analysis with the code TRACE using point kinetics model. First of all, the turbine trip 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 Turbine Trip Event. The turbine trip event was initiated by the erroneous activation of the condenser controller when the pressure was above 0.3 bar [14]. In reality, the condenser pressure was not higher than 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 the 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 hinder a pressure increase in the steam line. It has to be noted that after the turbine stop valve (TSV) started to 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 and feed-water lines are connected to the reactor pressure vessel. In the core 784 fuel assemblies of uranium oxide (UO_2) and mixed oxide fuel (MOX) were loaded in 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 (SRVs) and the TSV and TBV. Finally, the BREAK and FILL components were used to define, for example, 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 [12]. There was shown that a 3D model and a 2D model predict the same results for the TUSA event which represents a global perturbation of the core behaviour. But for the investigation of nonsymmetrical transients like a rod drop accident, a 3D thermal hydraulic model is mandatory to catch the local perturbation of, for example, the reactor power distribution.

Since the dynamic response of the pumps does not play an important role during the TUSA, they were represented by a simplified 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%) [12]. 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 Figures 13 and 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 leads to a pressure spike shown in Figure 15.

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.

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 was 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 underpredict the single phase pressure drop while the calculation of the two phase pressure drop agrees well with the measured data. A comparison of

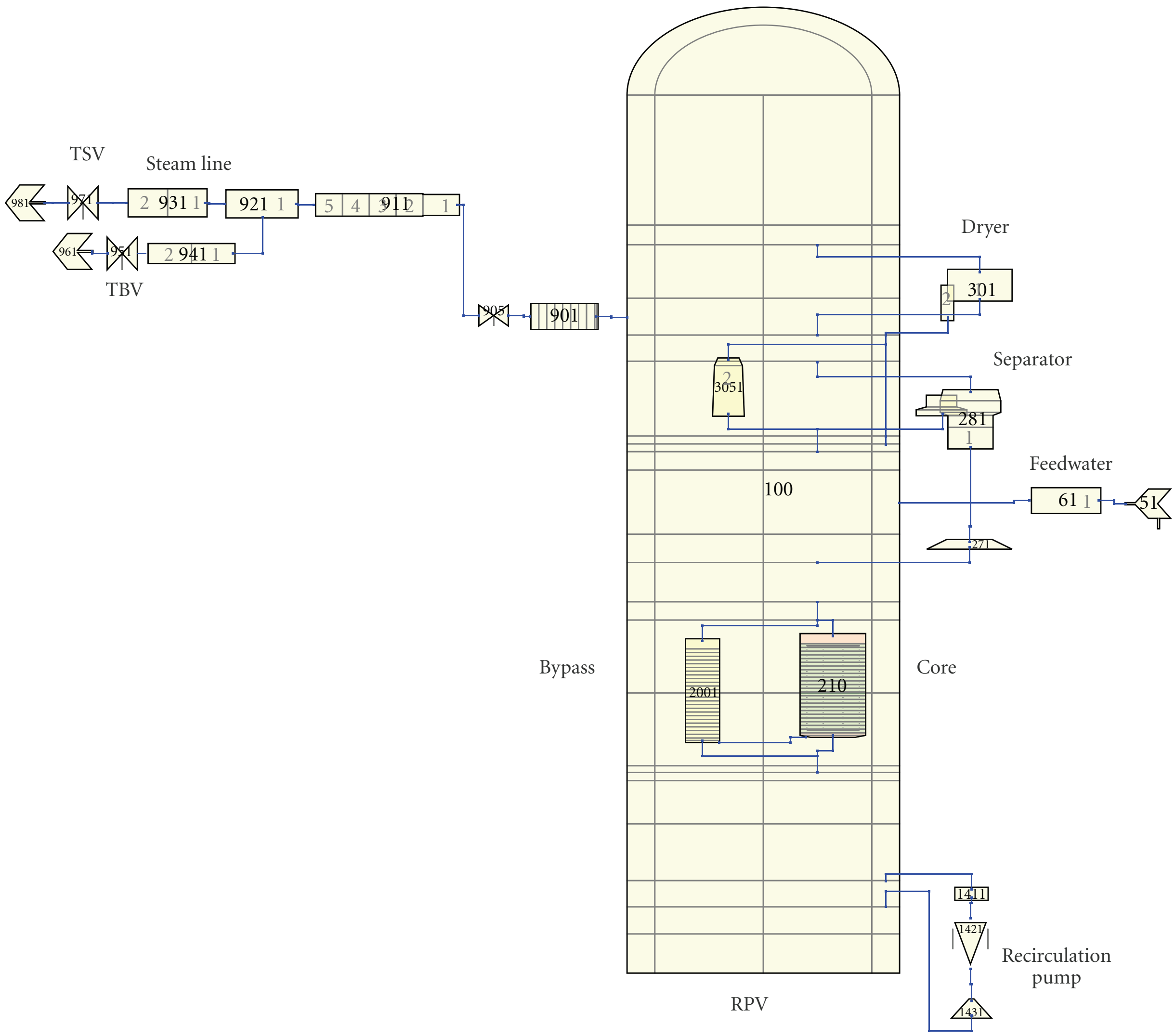


FIGURE 12: Integral BWR plant model of the BWR as represented by SNAP (2D model).

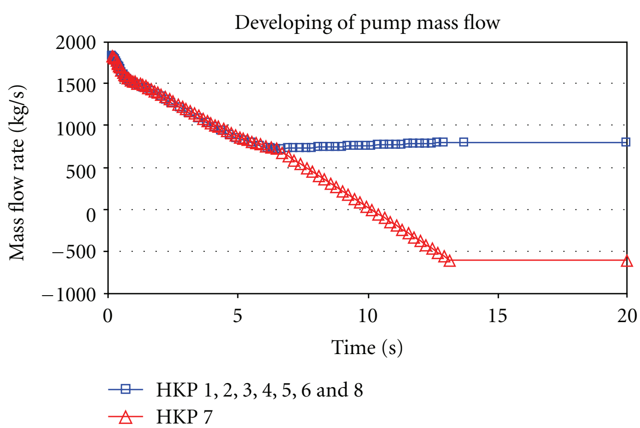


FIGURE 13: MRP mass flow rate during the transient.

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 posttest 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, for example, 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 behavior 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 is already underway [13] and will be published in a subsequent publication.

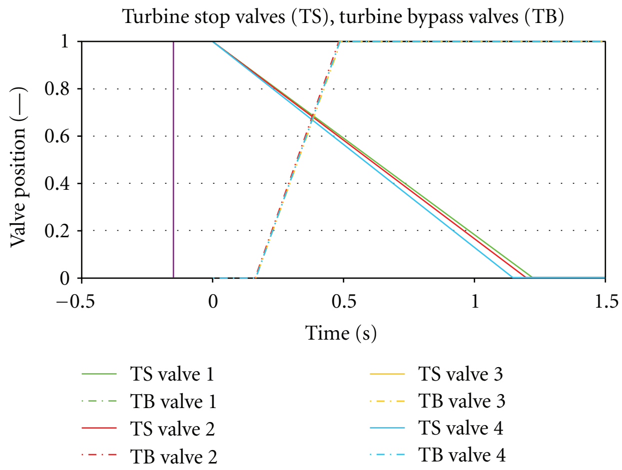


FIGURE 14: Dynamic of the closing and opening of TS and TB.

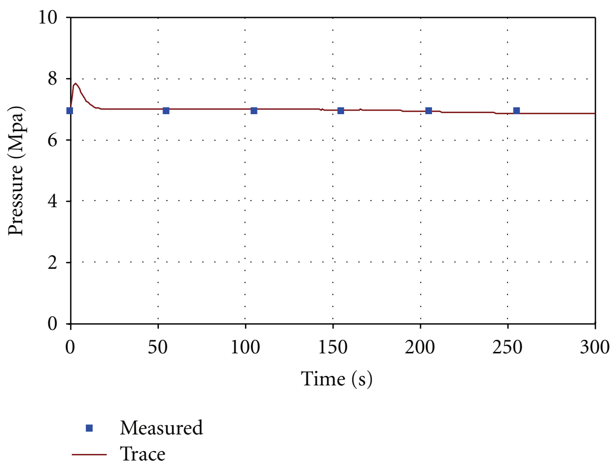


FIGURE 15: Comparison of predicted and measured dome pressure during transient.

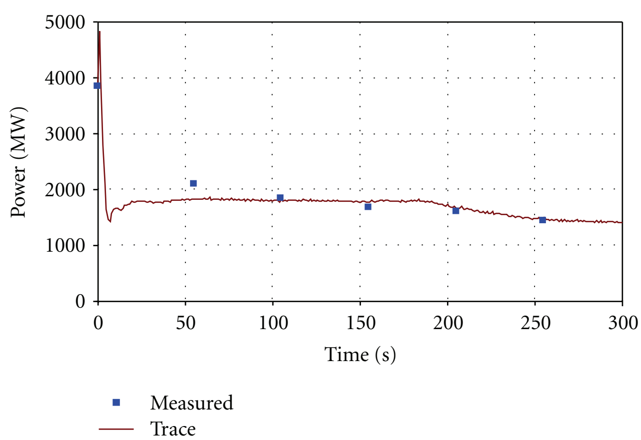


FIGURE 16: Comparison of predicted and measured reactor power during transient.

Abbreviations

BFBT: Boiling Water Reactor Full-Size Fine-Mesh Bundle Test
PSBT: PWR Subchannel and Bundle Test
BWR: Boiling water reactor
CT: Computer tomography
CAMP: Code Application and Maintenance Program
LWR: Light water reactors
NUPEC: Nuclear Power Engineering Corporation
NRC: US Nuclear Regulatory Commission
MRP: Main recirculation pump
MOX: Mixedoxide fuel
RPV: Reactor pressure vessel
RSME: Root square mean error
TUSA: Turbine trip event
TBV: Turbine bypass valve
TSV: Turbine stop valve.

Acknowledgments

The authors greatly appreciate the technical advice from Mr. Holzer (Simpelkamp Nukleartechnik) and Mr. Scholz and Dr. Förster (Nuclear Power Plant Gundremmingen). The fruitful discussions with them helped us to clarify important plant specific issues for the model development. Finally, the authors thank the Program "*Nuclear Safety Research*" of KIT for the financial support of the Research Topic "*Multiphysics Methodologies for Reactor Dynamics and Safety.*"

References

[1] "TRACE V5. 0 assessment manual," Main Report, U. S. Nuclear Regulatory Commission, Washington, DC, USA, 2005.

[2] K. Ivanov, A. Olson, and E. Sartori, "OECD/NRC BWR turbine trip transient benchmark as a basis for comprehensive qualification and studying best estimate coupled codes," in *Proceedings of the International Conference on the New Frontiers of Nuclear Technology (PHYSOR '02)*, Seuol, Korea, October 2002.

[3] K. Nikitin, J. Judd, G. M. Grandi, A. Manera, and H. Ferroukhi, "Peach bottom 2 turbine trip 2 simulation by TRACE/S3K coupled code," in *Proceedings of the Advances in Reactor Physics to Power the Nuclear Renaissance Pittsburgh Conference (PHYSOR '10)*, pp. 2283–2293, CD-ROM, American Nuclear Society, Pittsburgh, Pa, USA, May 2010.

[4] T. Kozlowski, S. Roshan, T. Lefvert et al., "TRACE/PARCS validation for BWR stability based on OECD/NEA oskarshamn-2 benchmark," in *Proceedings of the 14th Annual Topical Meeting on Nuclear Reactor Thermalhydraulics*, Toronto, Canada, September 2011.

[5] J. Bánáti, M. Stålek, C. Demazière, and M. Holmgren, "Analysis of a loss of feedwater case at the ringhals-3 NPP using RELAP5/PARCS coupled codes," in *Proceedings of the 16th International Conference on Nuclear Engineering (ICONE '08)*, pp. 135–144, Orlando, Fla, USA, May 2008.

[6] W. Jäger and V. Sánchez, "Development of a 3D-VVER-1000-RPV-model for the investigations of a coolant mixing

experiment with the best-estimate code TRACE," in *Proceedings of the Annual Meeting on Nuclear Technology*, Karlsruhe, Germany, May 2007.

[7] W. Jäger, V. Sánchez, and R. Macián-Juan, "On the uncertainty and sensitivity analysis of experiments with supercritical water with TRACE and SUSA," in *Proceedings of the 18th International Conference on Nuclear Engineering (ICONE '10)*, Xi'an International Conference Center, May 2010.

[8] B. Neykov, F. Aydogan, L. Hochreiter, and K. Ivanov, *NUPEC BWR Full-Size Fine-Mesh Bundle Test (BFBT) Benchmark—Volume I: Specifications*, NEA/NSC/DOC(2005)5, Pennsylvania State University, 2005.

[9] A. Rubin, A. Schoedel, M. Avramova, H. Utsuno, S. Bajorek, and A. Velazquez-Lozada, *OECD/NRC Benchmark Based on NUPEC PWR Subchannel and Bundle Tests (PSBT), Volume I: Experimental Database and Final Problem Specifications*, US NRC OECD Nuclear Energy Agency, 2010.

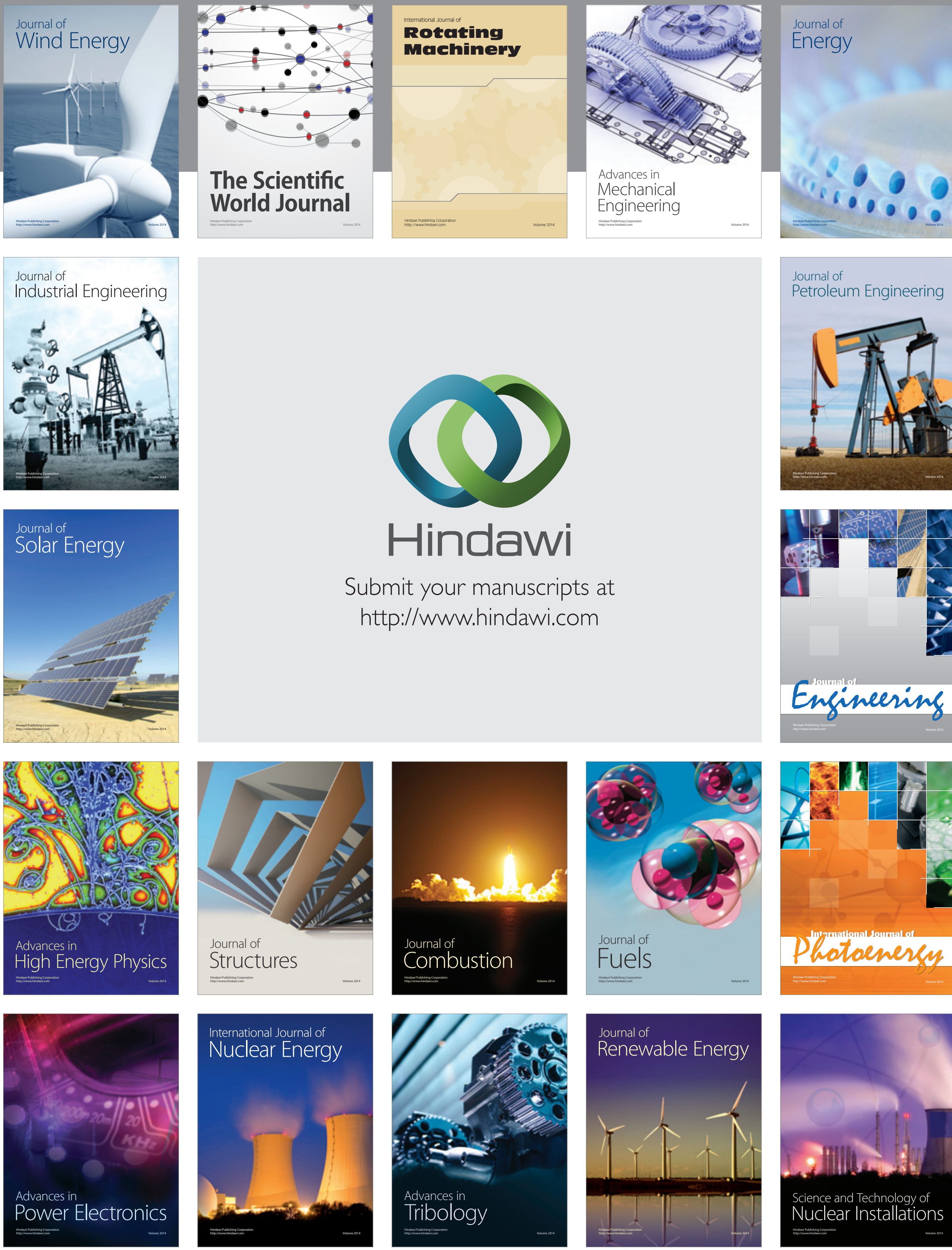
[10] M. Thieme, W. Tietsch, R. Macian, and V. Sanchez, "Validation of TRACE using the void fraction tests of the NUPEC BFBT facility," in *Proceedings of the International Congress on Advances in Nuclear Power Plants (ICAPP '09)*, Tokio, Japan, May 2009.

[11] M. Thieme, W. Tietsch, R. Macian, and V. Sanchez, "Post-test investigations of BFBT critical power tests with trace," in *Proceedings of the 17th International Conference on Nuclear Engineering (ICONE '09)*, pp. 503–513, Brussels, Belgium, July 2009.

[12] M. Thieme, "Qualifizierung des best-estimate programmsystems TRACE für die sicherheitsbewertung von störfallabläufen in siedewasserreaktoren," Internal Report, 2009.

[13] Ch. Hartmann, V. Sánchez, and W. Tietsch, "Analysis of a German BWR Core with TRACE/PARCS using different cross section sets," in *Proceedings of the 14th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH '11)*, Toronto, Canada, September 2011.

[14] Rusch, "TUSA Auslösung über hydraulischen Kondensatordruckwächter," Störungsbericht KRB II, BLOCK C. Aktenzeichen A-30/234. 28. 7., 1998.


Journal of Wind Energy
The Scientific World Journal
International Journal of Rotating Machinery
Advances in Mechanical Engineering
Journal of Energy
Journal of Industrial Engineering
Hindawi
Submit your manuscripts at
http://www.hindawi.com
Journal of Petroleum Engineering
Journal of Solar Energy
Journal of Engineering
Advances in High Energy Physics
Journal of Structures
Journal of Combustion
Journal of Fuels
International Journal of Photoenergy
Advances in Power Electronics
International Journal of Nuclear Energy
Advances in Tribology
Journal of Renewable Energy
Science and Technology of Nuclear Installations