ANALYSIS OF THE BOILING WATER REACTOR TURBINE TRIP BENCHMARK WITH THE CODES DYN3D AND ATHLET/DYN3D

Ulrich Grundmann, Soeren Kliem, and Ulrich Rohde

1. Introduction

The three-dimensional core model DYN3D was developed for steady-state and transient analysis of thermal reactors of western type with square fuel assemblies and Russian VVER type with hexagonal fuel assemblies [1]. It was coupled with the thermal-hydraulic system code ATHLET [2] of the German Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) for best-estimate analyses of the reactor systems. DYN3D consists of three-dimensional neutron kinetic models coupled with one-dimensional thermal hydraulics in parallel core channels. The ATHLET code has its own neutron models which consists in point kinetics or one-dimensional kinetics. The codes DYN3D and ATHLET can be used coupled or as stand alone. In accomplishing the coupling of ATHLET and DYN3D two basically different ways, the so-called internal and external couplings were used [3].

For the calculation of the benchmark, the codes have been coupled using a modified version of the external coupling, the so-called ‘parallel coupling’. Contrary to the pure external coupling, where the core is cut off from the system model and the DYN3D-core model is directly coupled to the system model through two interfaces in the lower and upper plenum, in the new coupling both core models are running in parallel. ATHLET containing its own core model calculates the behaviour of the whole plant and provides pressure boundary conditions at the core inlet and outlet and the core inlet coolant enthalpy. Using these boundary conditions, the whole core behaviour is calculated by DYN3D and the total core power is transferred to the ATHLET-core. This parallel coupling shows a more stable performance at low time step sizes necessary for a proper description of the feedback in the calculation of the current boiling water reactor benchmark problem.

The OECD/NRC (Organisation for Economic Co-operation and Development/Nuclear Regulatory Commission) Boiling Water Reactor (BWR) Turbine Trip (TT) Benchmark based on the turbine trip test 2 (TT2) in the American Peach Bottom 2 reactor [4] was analysed to validate the code DYN3D and the coupled system DYN3D/ATHLET for BWR’s. Moreover, groups from several countries around the world participate in the benchmark for the validation of the code systems.

The transient was initiated by the closure of the turbine stop valve. The resulting pressure wave propagates to the core. It is attenuated by opening the bypass valve. When the wave reaches the core the void in the core is reduced, which results in an increase of the reactivity and power. The power peak is limited by the Doppler effect and the reactor scram. The scram was initiated at t = 0.63 s. The control rod motion starts with a delay of 0.12 s at t = 0.75 s. It has a significant influence on the power after the power peak. The transient was investigated in the time interval from t = 0 to 5 s.

The benchmark consists of three different exercises:

- Exercise 1 - Simulation of the transient by means of an advanced thermal-hydraulic system code using a fixed power curve based on the experimental data.
- Exercise 2 - Calculation of the core response on given time-dependent thermal-hydraulic boundary conditions.
- Exercise 3 - Best-estimate coupled 3D neutron kinetic core/thermal-hydraulic system calculation.

The modelling of the Peach Bottom reactor is described in Sec. 2. Concerning the modelling of the BWR core several simplifications and their influence on the results are investigated for the Exercise 2 with the core model DYN3D. DYN3D allows calculations with and without assembly discontinuity factors (ADF) to study their influence on this transient. The calculations with DYN3D are performed with 764 coolant channels (1 channel per fuel assembly). Depending on the used codes several participants of the benchmark performed calculations with only 33 thermal-hydraulic channels. The influence of the number of coolant channels was studied by different calculations with DYN3D. The phase slip model of MOLOCHNIKOV [5] is the standard model of DYN3D for void fraction calculation. A comparison was made with the ZUBER-FINDLAY model [6]. The results of the different modifications are compared in Sec. 3 with the results of the standard model based on 764 coolant channels, the consideration of the ADF, and the phase slip model of MOLOCHNIKOV.

The Exercise 3 was calculated with the parallel coupling ATHLET/DYN3D. The core model in these calculations is the standard DYN3D model applied for the calculations for Exercise 2. The thermal-hydraulic model of ATHLET was used as applied for the calculations of Exercise 1. Key results are compared with the measurements in Sec. 4.

2. Modelling of the Peach Bottom 2 reactor

The BWR Peach Bottom 2 reactor core consists of 764 fuel assemblies [4], each of them is modelled by one thermal-hydraulic channel in the standard case. The core was divided into 24 axial layers. Each of them has a height of 15.24 cm. The assemblies and their water gap have a width of 15.24 x 15.24 cm which determines the radial size of the nodes. The density of the bypass coolant flowing between the fuel assemblies (fuel assembly bypass) is different to the two-phase flow density inside the channels. About 1.7% of the generated power is released in the bypass. Performing the cell calculations with the CASMO code [4] the coolant density was assumed at saturation temperature in the bypass. Due to the specification, a density correction has to be taken into account in the nodal two-group cross section calculation, if the density in the bypass deviates from the saturation value. As a sufficient approximation all fuel assembly bypasses are lumped to one bypass channel in DYN3D. The DYN3D calculations are based on the given total mass flow rate through the core. Applying the thermal-hydraulic model FLOCAL of the DYN3D code the flows through the individual channels are calculated using

Table 1: Key parameters of the initial state of the turbine trip test 2

thermal power	2030 MW
pressure (core outlet)	6.798 MPa
inlet temperature	274 °C
total core mass flow	10445 kg/s
inchannel mass flow	9603 kg/s
Core average void fraction	30.4 %

the resistance coefficients of the channels according to the specification [4]. In the case of given total mass flow rate, the individual flows through the channels are determined by the condition of equal pressure drop over the core. The decay heat is calculated by the model implemented in DYN3D assuming an infinite operation at the power level of the initial state of the TT2. It is based on the German standard [7]. In DYN3D, each coolant channel is described by one average fuel rod. The power of a coolant channel is obtained by averaging the nodal powers of the fuel assemblies belonging to it. The main parameters of the initial state of TT2 are given in Table 1. The transient calculations for the Exercise 2 were performed with given transient thermal-hydraulic boundary conditions, i.e. pressure, at the core exit, the total mass flow rate and the core inlet temperature.

For the calculations of Exercise 3, the ATHLET-input deck developed and used by the Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) for Exercise 1 has been used. This input deck consists of the lower plenum, the core region with one single thermal-hydraulic core channel and one bypass channel, the upper plenum, stand pipes, and the separator. Fig. 1 shows a node scheme of the Peach Bottom 2 plant developed for the code RETRAN [4]. The downcomer section is splitted into several parts, to include the jet pumps and the diffusors. Two symmetric recirculation loops, one steam line up to the turbine stop valves and the turbine bypass line are modelled For the Exercise 3 calculations, the above described standard core model with 764 thermal-hydraulic channels was applied in DYN3D. The core boundary conditions were transferred from the ATHLET code to DYN3D. Instead of the total mass flow rate (Exercise 2) the pressure drop over the core calculated by the ATHLET-model was used as boundary condition.

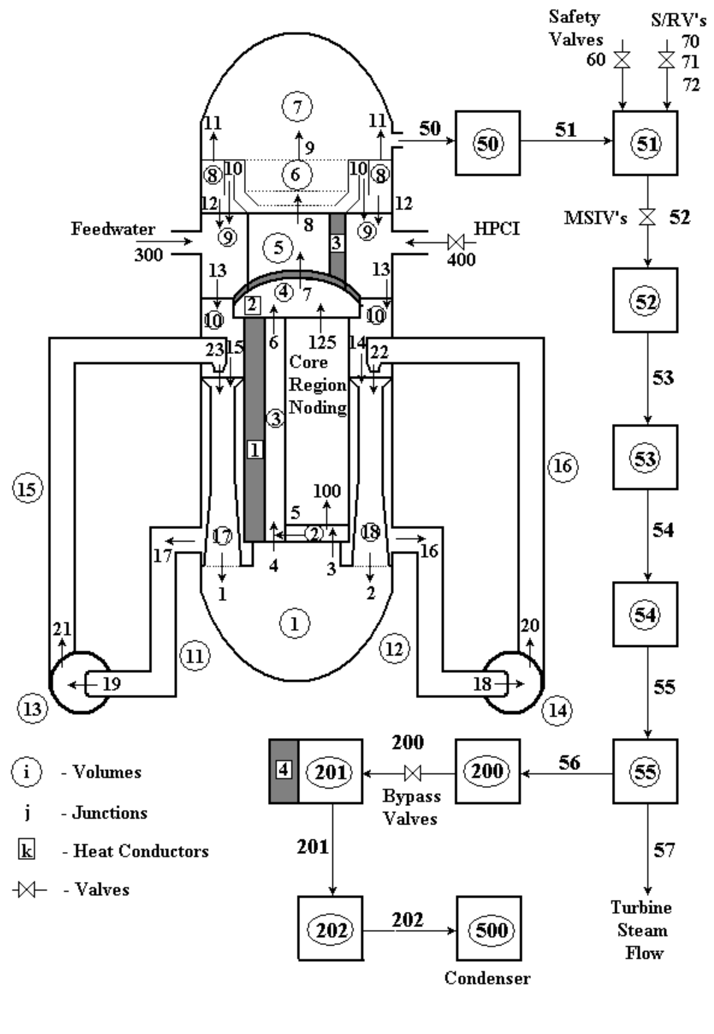

Fig. 1: RETRAN node scheme of the Peach Bottom 2 plant [4]

3. Steady-state and transient results for Exercise 2 using DYN3D

A hot zero power (HZP) state was defined with equal thermal-hydraulic parameters in all nodes for a first comparison of the codes. It was calculated with and without the ADF to investigate their influence. Table 2 gives the results for the eigenvalue k_{eff}, the 3D nodal power

Table 2: Influence of the ADF on k_{eff} and power peak factors at HZP and the initial state of the turbine test 2.

	HZP Without ADF	HZP With ADF	HZP Diff. (%)	TT2 Without ADF	TT2 With ADF	TT2 Diff. (%)
k_{eff}	0.99133	0.99654	0.53	1.00270	1.00410	0.14
F_Q	5.056	5.364	6.1	2.260	2.235	1.0
F_{xy}	1.884	1.998	6.1	1.448	1.448	-
F_z	2.692	2.698	0.2	1.494	1.459	2.4
Max. ass. 75	2.257	2.137	5.6	1.207	1.102	9.5
Max. ass. 367	3.584	3.293	8.8	1.766	1.716	2.9

peak factor F_Q, the assembly power peak factor F_{xy}, the axial power peak F_z of the radially averaged distribution and of the assemblies 75 (peripheral) and 367 (central). It is demonstrated that the ADF's influence the eigenvalue and the radial distribution (F_Q, F_{xy}). Therefore, the maximum values of the axial distribution in single fuel assemblies (for example assembly 75 and 367) show larger differences, while the influence on the averaged axial distribution is small. The power of the initial state of the TT2 is 2030 MW_{th}. Table 2 proves that the influence of the ADF on k_{eff}, F_Q and F_{xy} is smaller at that power level than in the HZP case. However, considering the single assemblies 75 and 367, the influence of the ADF is not negligible.

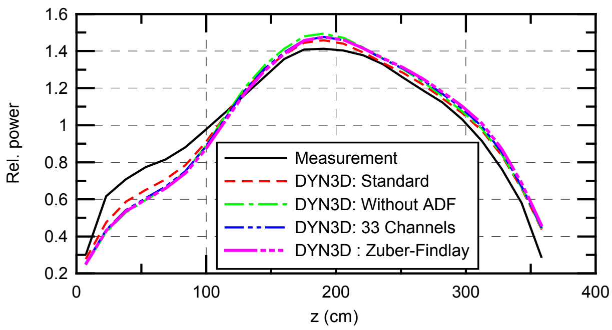

Fig. 2 : Averaged axial power distribution.

The averaged axial power distributions of the TT2 initial state are compared with the measured distribution in Fig. 2 for all different DYN3D calculations described in the introduction. The Figure shows that the differences of the calculations are rather small. The deviations from the measurement are in the range of the results of the other participants taking part in the benchmark. Nevertheless, the standard calculation is closer to the measurement than the other cases. The boiling model of ZUBER-FINDLAY yields only small changes in comparison to the standard model. The changes are caused by the different void fractions obtained with the models. The standard model of DYN3D provides 29.2 % average core void and the ZUBER-

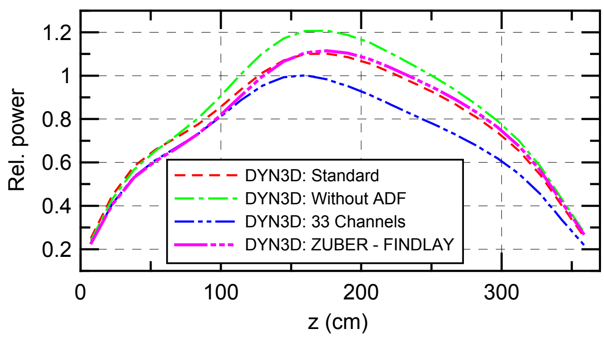

Fig. 3: Axial power distribution in the assembly 75.

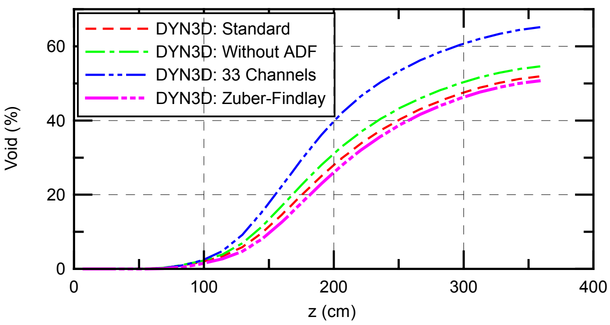

Fig. 4: Void distribution in the assembly 75.

FINDLAY 28.1%, which results in a higher neutron flux in the upper core region when using the ZUBER-FINDLAY model. The results for the peripheral fuel assembly 75 show larger deviations (see Fig. 3). The effect of the ADF depends on the control rod positions. If an assembly is rodded, the deviation of the ADF from unity is larger than for unrodded assemblies. In case of the initial state of TT2, the assembly 75 is rodded. Therefore the consideration of the ADF has a considerable effect on the power generation in this assembly. The higher power in assembly 75 of the calculation without the ADF leads to higher void in the coolant channel (see. Fig. 4). The lumping of the thermal-hydraulic channels of the fuel assemblies to 33 channels also leads to larger differences of power generation in single elements. The model with 33 channels produces a higher void (Fig. 4), because the fuel assembly 75 is linked together with fuel assemblies of higher power. The application of the ZUBER-FINDLAY slip model instead of the standard model provides only small differences in the power and the void distribution, which are in the same order of magnitude as for the radial averaged power distribution (Fig. 2).

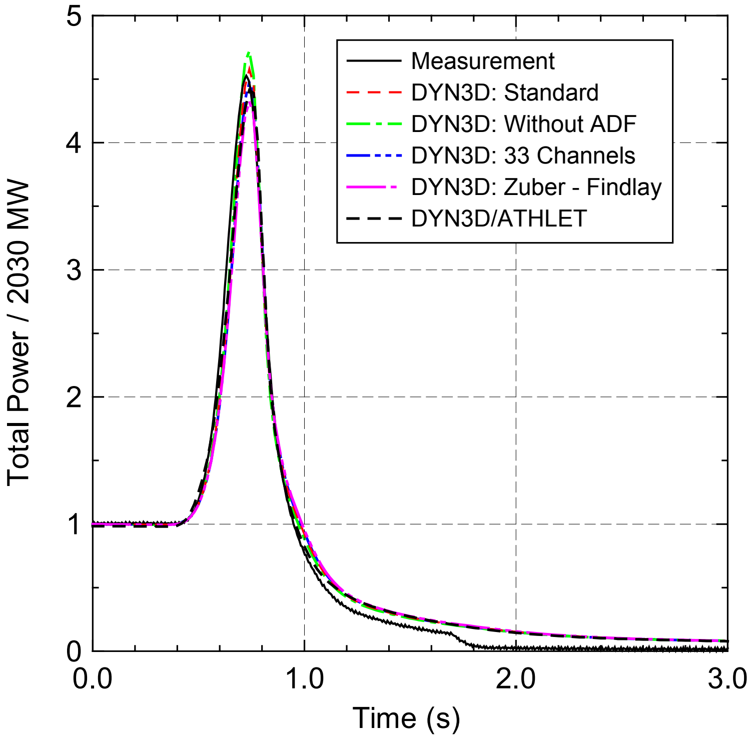

Fig. 6: Total power versus time of the different DYN3D calculations and the DYN3D/ATHLET calculation compared with measurement

In the first phase of the transient up to t = 0.65 s, the core void decreases mainly as a result of the pressure increase. The reactivity change is determined by the change of the coolant density in the core as long as the fuel temperature is unchanged and the shut down is not activated. Fig. 5 shows the power versus time. The reactivity increase by the reduced core void leads to a power excursion. All calculations including the standard simulation provide a relative power peak close to the measured value of 4.5.

4. Calculation of Exercise 3 with the coupled code DYN3D/ATHLET

Fig. 6 shows the calculated steam dome pressure and the pressure at the core exit compared with the measured value in the steam dome and the specified value at core exit which was provided from a TRAC-BF1/NEM calculation [4]. The DYN3D/ATHLET pressures are slightly lower in the initial state than the measured and the specified one. Nevertheless, the time, when the pressures starts to increase is well reproduced by the ATHLET plant model. The calculated absolute pressure increase during the first 0.8 s (the most important time interval in the calculation) is in full agreement with the measurement. This excellent description of the thermal-hydraulic aspects of the transient together with the appropriate DYN3D core model are the basis for the good agreement of the time course of the core power between calculation and measurement which can be seen in Fig. 5. The shape of power increase, the time of the maximum and the maximum value itself agree very well with the measurement. In the later phase of the transient (after t = 2 s) , the calculation slightly overpredicts the system pressure. But this is of no influence on the core power behaviour, because at that time, the core is already in a subcritical state. Fig. 7 shows the comparison of calculations with the measurements of the local power range monitors (LPRMs). The values A, B, C, and D are the radial averaged values of 42 micro fission chambers at four axial levels, which are located at 45.72, 137.16, 228.6 and 304.8 cm above the core bottom. The agreement between calculated and measured time behavior is similar to the power in Fig. 5.

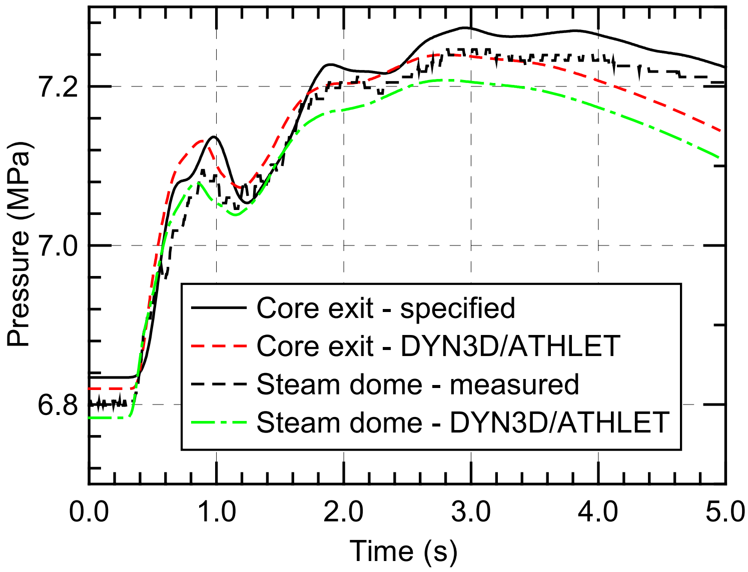

Fig. 6: Transient behaviour of the calculated pressure in the steam dome and at the core exit with measured or specified values.

5. Summary

The Exercise 2 of the OECD/NRC BWR TT Benchmark was analysed with the core model DYN3D using different model options. The results of the standard case that includes the consideration of 764 thermal-hydraulic channels (one channel/passembly), the ADF and the standard phase slip model of MOLOCHNIKOV show a good agreement with the measurements. The results without ADF, with 33 thermal-hydraulic channels or with the ZUBER-FINDLAY slip model yields only small differences, when core-averaged values are compared. However, the calculations without the ADF or with 33 thermal-hydraulic channels provide deviations to the standard case in single fuel assemblies. The available computer allows transient calculations with such detailed models as in the standard case which is in agreement with the models used for design and fuel cycle calculations. For the calculations of Exercise 3, the parallel coupling of DYN3D and ATHLET has been used. The results of the calculation were com-

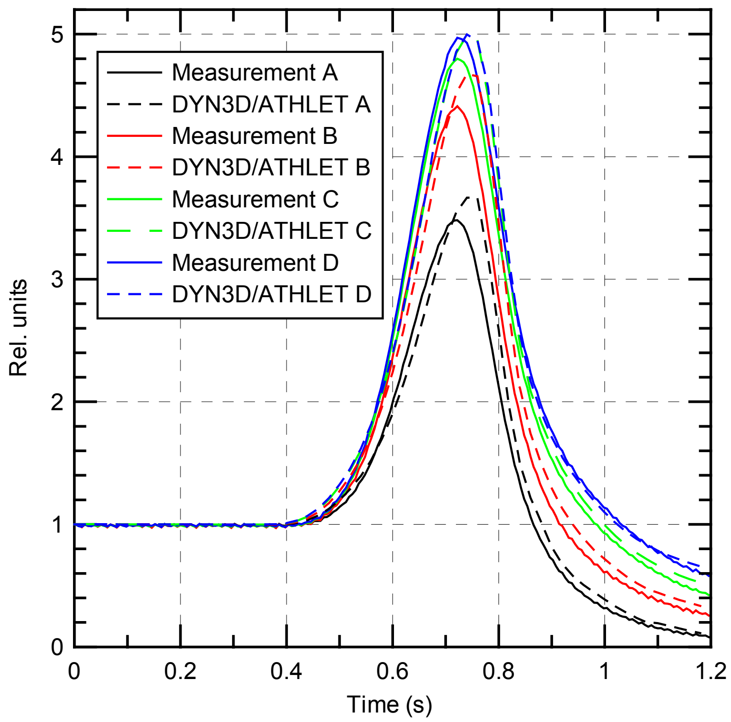

Fig. 7: Comparison of DYN3D results with the LPRM signals at the four axial levels A, B, C, D.

pared with measuring data. Only small deviations are found between the calculated and measured core power and the LPRM signals.

The calculations of Exercise 2 and 3 prove the applicability of the codes DYN3D and ATHLET/DYN3D to the simulation of such boiling water reactor transients with a strong feedback and are a substantial contribution to the validation of DYN3D and ATHLET/DYN3D.

References

[1] U. Grundmann, U. Rohde, and S. Mittag (2000), DYN3D – three-dimensional core model for steady-state and transient analysis of thermal reactors, Proc. of Int. Topl. Mtg. on Advances in Reactor Physics and Mathematics and Computation into the Next Millennium (PHYSOR2000), Pittsburgh (USA), May, 7- 11, 2000.

[2] V. Teschendorff, H. Austregesilo, and G. Lerchl (1996), Methodology, status and plans for development and assessment of the code ATHLET, Proc. OECD/CSNI Workshop on Transient Thermal-Hydraulic and Neutronic Codes Requirements, Annapolis, USA, Nov. 5-8, 1996, p. 112, NUREG/CP-0159, NEA/CSNI/R(97)4.

[3] S. Kliem, U. Grundmann, and U. Rohde (1997), The coupled code complex DYN3D/ATHLET - application to main steam line break analysis, Proc. Joint International Conference on Mathematical Methods and Supercomputing for Nuclear Applications, pp.1358-1366, ANS, La Grange Park, IL (1997).

[4] J. Solis, K.N. Ivanov, B. Sarikaya, A. M. Olson, and K.W. Hunt (2001), Boiling water reactor turbine trip (TT) benchmark", Volume 1: Final Specifications, NEA/NSC/DOC (2001) 1, Paris, June 2001.

[5] Yu. S. Molochnikov (1982), Generalisation of experimental data on volumetric void fraction for sub-cooled boiling of water (in Russian). Teploehnergetika No.7, p. 47.

[6] N. Zuber and J. A. Findlay (1965), Average volumetric concentration in two phase flow systems, Journal of Heat Transfer, 87c, p.453.

[7] DIN Norm 25463, Part 1 (1990), Berechnung der Nachzerfallsleistung der Kernbrennstoffe von Leichtwasserreaktoren.

Acknowledgements

The participation in the OECD/NRC BWR TT Benchmark was funded by the German Federal Ministry of Economics and Labour (BMWA).