ANALYSIS OF THE BOILING WATER REACTOR TURBINE TRIP BENCHMARK WITH THE CODE DYN3D

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ABSTRACT

Considering the phase 2 of the OECD/NRC BWR Turbine Trip Benchmark several analyses were performed with the help of the DYN3D code. Thermal-hydraulic boundary conditions of the core are given for this part of the benchmark. Concerning the modelling of the BWR in the DYN3D code several simplifications and their influence on the results were investigated. The standard calculations with DYN3D were performed with 764 coolant channels (1 channel per fuel assembly). It is shown that the power peak obtained for the given boundary conditions is close to the measured value. For numerical stability reasons, preliminary calculations were carried out neglecting the instationary mass balance equation. This approximation provided a stronger reduction of the core void that results in a higher power peak. The impact of the assembly discontinuity factors (ADF) was studied. It is shown that the influence on core-averaged values of the steady state and the transient is small. Considering local parameters the influence is not negligible. Several participants of the benchmark perform calculations with 33 thermal-hydraulic channels. The influence of the number of coolant channels has also a small effect on the core averaged values, but local parameters as axial power distribution in single fuel assemblies are affected. The phase slip model of MOLOCHNIKOV is the standard model of DYN3D for void fraction calculation. The ZUBER-FINDLAY model shows only small deviations from the standard case for both global and local values, however not in the direction of the measurement. Using the thermal-hydraulic boundary conditions the best agreement with the experiment was obtained for the standard model.

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 of the German Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) [2] 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 as well as stand alone. There are two options of coupling, the external and the internal coupling. The core is completely described by DYN3D in the external coupling whereas the neutron kinetics of ATHLET is replaced by the 3D kinetics of DYN3D in the internal coupling. Most of the problems were analysed with the option of external coupling.

The OECD main steam line break (MSLB) benchmark [3] was analysed for validation of the code system for pressurized water reactor systems. Similar benchmarks for VVER reactors were analysed in the frame of the Atomic Energy Research (AER) cooperation of the research institutions which are concerned with VVER reactors. The OECD/NRC Boiling Water Reactor (BWR) Turbine Trip (TT) Benchmark based on the turbine trip test 2 (TT2) in the reactor Peach Bottom 2 [4] is analysed for validation of the codes and the coupled systems for BWR's. The phase 2 of the benchmark consists in the calculation of the core response for given thermal-hydraulic boundary conditions. This part of the benchmark is used for the validation of the DYN3D code for the calculations with the coupled code system.

The transient was initiated by the closure of the turbine stop valve. The pressure wave, which is moved to the core is attenuated by the opening of 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.

Concerning the modelling of the BWR several simplifications and their influence on the results of the considered transient are investigated. If the axial variation of the mass flow rates in correspondence with the instationary mass balance equation is taken into account, the calculations with very small time steps can lead to numerical stability problems of the thermal hydraulics of DYN3D. Therefore preliminary calculations were carried out assuming axially constant mass flow rates. It is shown that this approximation leads to a stronger reduction of the core void during the pressure increase. The consideration of assembly discontinuity factors (ADF) is possible not in all three-dimensional codes. DYN3D allows calculations with and without the ADF to study their influence on this transient. The calculations with DYN3D are performed with 764 coolant channels (1 channel per fuel assembly). Several participants of the benchmark perform calculations with 33 thermal-hydraulic channels, which corresponds to the thermal-hydraulic map used in the TRAC-BF1/NEM model [4]. The influence of the number of coolant channels is studied also in this paper. 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 with the results of the standard calculation based on 764 coolant channels, the consideration of the ADF, the solution of the instationary mass balance equation and the phase slip model of MOLOCNIKOV.

2. MODELLING OF THE PEACH BOTTOM 2 CORE IN DYN3D

The BWR Peach Bottom 2 consists of 764 fuel assemblies [4], each of them is modelled by one thermal-hydraulic channel in the standard case. For sensitivity studies, calculations with 33 coolant channels were performed. The core was divided into 24 axial layers. Each of them has a length 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 coolant flowing between the fuel assemblies (fuel assembly bypass) has a density, which is different to the two-phase flow density inside the channels. About 1.7% of the generated power is released in the bypass. Saturation density was assumed in the bypass by performing the cell calculations with the

CASMO code [4]. Due to the specification, a density correction has to be taken into account in the nodal two-group cross section calculation. 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 to core. The flows through the individual channels are calculated by using the resistance coefficients of the channels in the thermal-hydraulic model FLOCAL of the DYN3D code. Equal pressure drop over all channels is requested, which is determined from the condition of given total flow rate. The decay heat is calculated by the model implemented in DYN3D by assuming an infinite operation at the power level of the initial state of the TT2. It is based on the German standard [7]. The fuel assemblies which are lumped to the coolant channels of the 33 channel model can be seen in Fig. 1. In DYN3D, each coolant channel is described by one fuel rod. The power of a coolant channel is obtained by averaging the nodal powers of its assemblies.

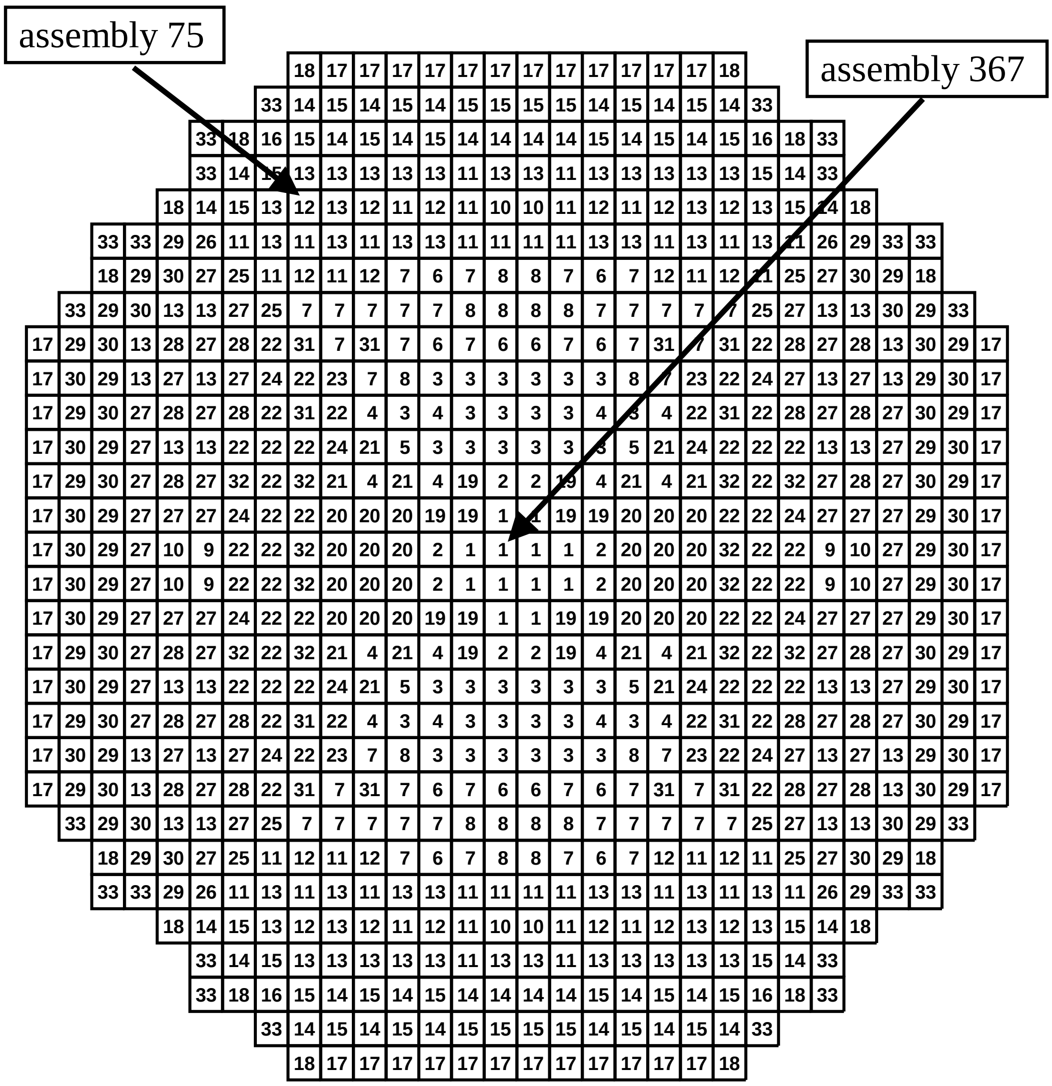

Fig. 1: Core map with numbers of the 33 thermal-hydraulic channels and the positions of the assemblies 75 and 367.

The main parameters of the initial state of TT2 are given in table I. The transient calculations for the phase 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. The transient was investigated in the time interval from t = 0 to 5 s. 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.

Table I: 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 %

3. STEADY-STATE RESULTS

A hot zero power (HZP) state with equal thermal-hydraulic parameters in all nodes was defined for a first comparison of the codes. It was calculated with and without the ADF to investigate their influence. The ADF have an influence on the eigenvalue k_{eff} and the assembly power distribution. Table II shows the results for the eigenvalue k_{eff}, the 3D nodal power 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 and 367. The position of the assemblies which were chosen in [4] can be seen in Fig. 1. It is demonstrated that the ADF's have an impact on 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.

Table II: 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

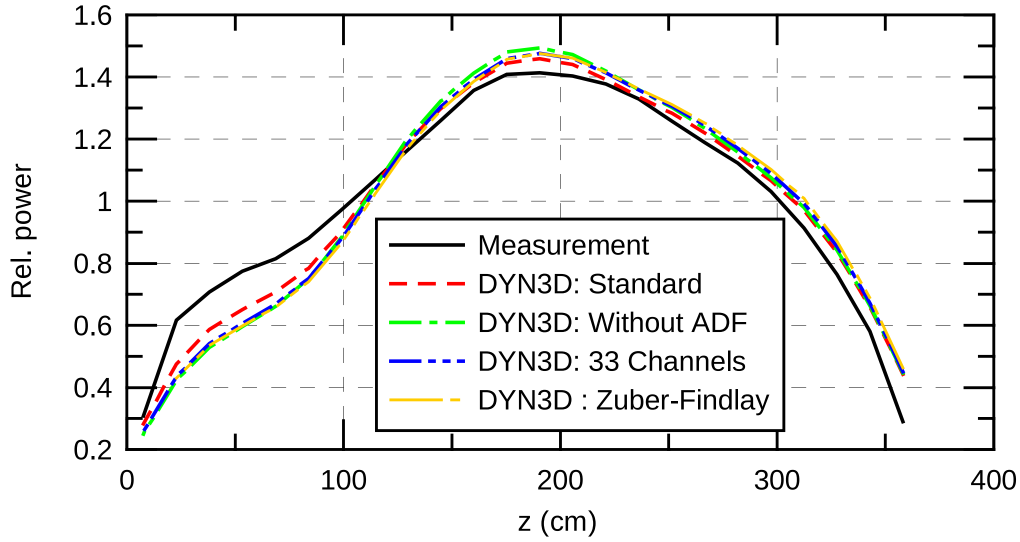

Fig. 2 : Averaged axial power distribution

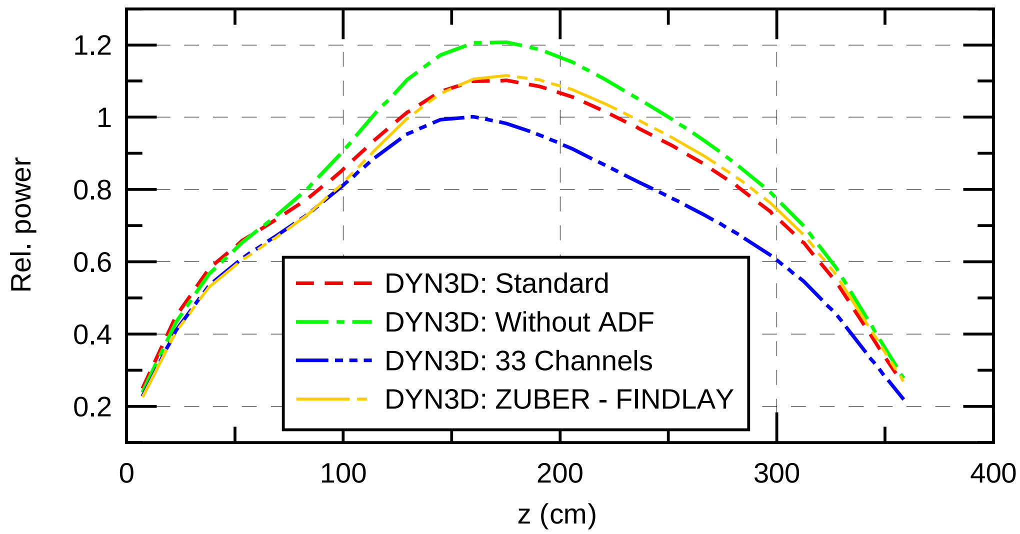

Fig. 3: Power distribution in the assembly 75.

The power of the initial state of the TT2 is 2030 MW_{th}. Table II shows that the influence of the ADF on k_{eff}, F_Q and F_{xy} at that power level is smaller than in the HZP case. Considering the single assemblies 75 and 367, the influence of the ADF is not negligible.

The averaged axial power distributions of the different calculations are compared with the measured distribution in Fig. 2. The figure shows that the differences of the calculations in comparison to the measurement are rather small. Nevertheless the standard calculation is closer to the measurement than the other cases. In the results of the single fuel elements 75 and 367, the different options show larger deviations (see Fig. 3 and 4).

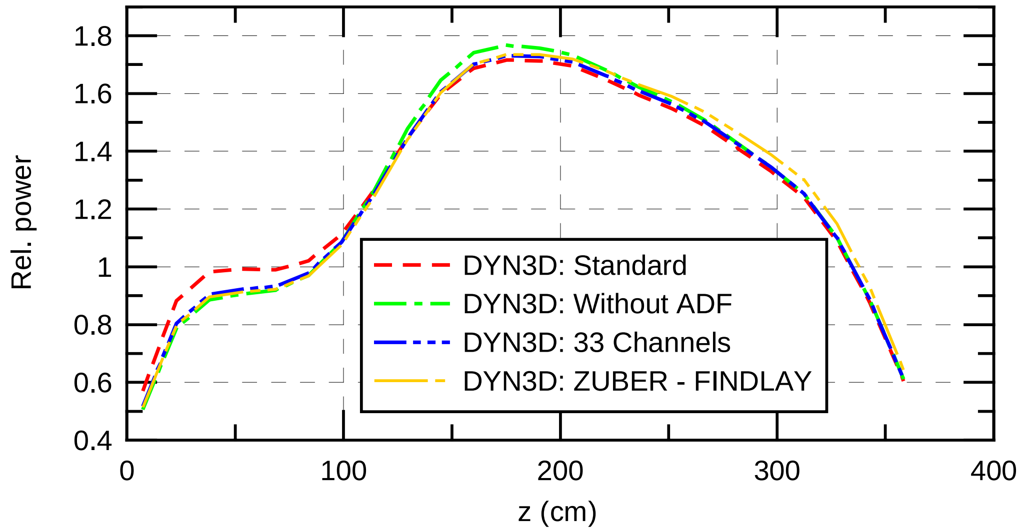

Fig. 4: Power distribution in the assembly 367.

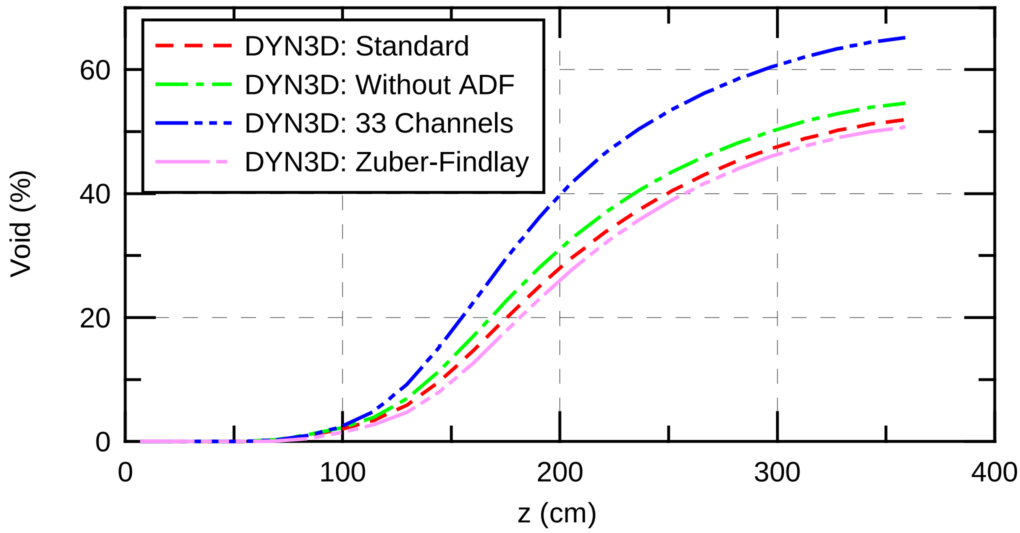

Fig. 5: Void distribution in fuel assembly 75.

It can be seen that the consideration of the ADF has a considerable effect on the power generation in the single fuel element 75. The lumping of the thermal-hydraulic channels of the fuel assemblies according to Fig. 1 leads also to larger deviations of power generation in single elements. The boiling model of ZUBER-FINDLAY gives only small deviations in comparison to the standard model. The deviations are caused by the different void obtained by the models. The standard model of DYN3D provides 29.2 % average core void and the ZUBER-FINDLAY 28.1%, which results in a higher neutron flux in the upper core region when using the ZUBER-FINDLAY model.

Fig. 5 shows the void in the fuel assembly 75. The model with 33 channels produces a higher void because the fuel assembly 75 is linked with fuel assemblies of higher power to the thermal-hydraulic channel 12 of Fig. 1. The normalized assembly powers p_i of the assembly 75 and the three assemblies at the symmetrical positions are $p_i \approx 0.78$. The values p_i of the other 16 assemblies of channel 12 are in the range $1.17 < p_i < 1.22$. It leads to a higher void of channel 12 in comparison to the void in channel 75 of the standard model. The higher void in channel 12 of the 33-channel model is responsible for the lower power in assembly 75. It can be assumed that the deviations can be minimised by an optimal choice of channels, however it requires an additional effort. Considering the calculation without the ADF, the higher power of the fuel assembly 75 is caused by the differences of the neutronic model. Therefore the higher power leads to higher void in the coolant channel of assembly 75. The ZUBER-FINDLAY boiling model shows only small differences of the power and the void distribution in the two assemblies 75 and 367 (Fig. 3 to Fig. 5). The void distributions in the assembly 367 are not shown here, because there are only small differences that are comparable with the axial power distributions in Fig. 4. Considering the 33-channel model the values of the normalized assembly power distribution of the assemblies belonging to the thermal-hydraulic channel 1 are close together, $1.22 < p_i < 1.24$.

4. TRANSIENT RESULTS

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 (see Fig. 6). 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. If the axial variation of the mass flow rates in correspondence with the instationary mass balance equation is taken into account in DYN3D, the calculations with very small thermal-hydraulic time steps can lead to numerical stability problems. For this reason, preliminary calculations were carried out by assuming axially constant mass flow rates. However with a sufficiently small thermal-hydraulic time step Δt_{TH} = 20 ms the calculation was stable for the option of axially dependent mass flow rates obtained from the solution of the mass balance equation. This Δt_{TH} together with the neutron kinetic time step Δt_{NK} = 2 ms was chosen for standard calculation. Considering the change of the coolant density in the core at the beginning of transient (Fig. 7), a stronger increase of the coolant density is observed in the calculation without the solution of the instationary mass balance equation.

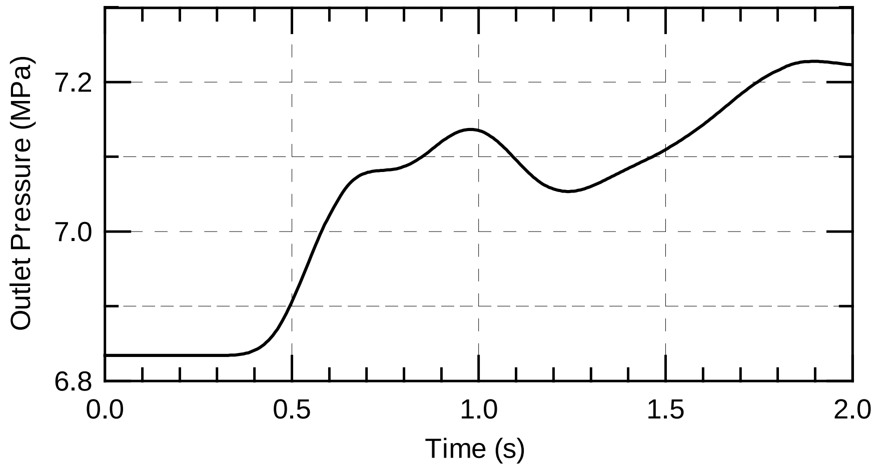

Fig. 6: Specified pressure at the core outlet.

The other simulations show only small deviations of the coolant density change against the standard calculation.

Fig. 8 shows the power versus time . The reactivity increase by the reduced core void leads to a power excursion. The power peak is very sensitive to the change of the coolant density. Therefore the calculation without the solution of the instationary mass balance equation leads to a power peak, which is 9 times the initial power value. It provides also the highest increase of the maximum fuel centerline temperature drawn in Fig. 9. The other calculations including the standard simulation provide a power peak close to the measured value of 4.5. The maximum fuel temperature of the calculation with 33 coolant channels is lower due to lumping the thermal-hydraulic channels. The deviations of the other simulations are rather small.

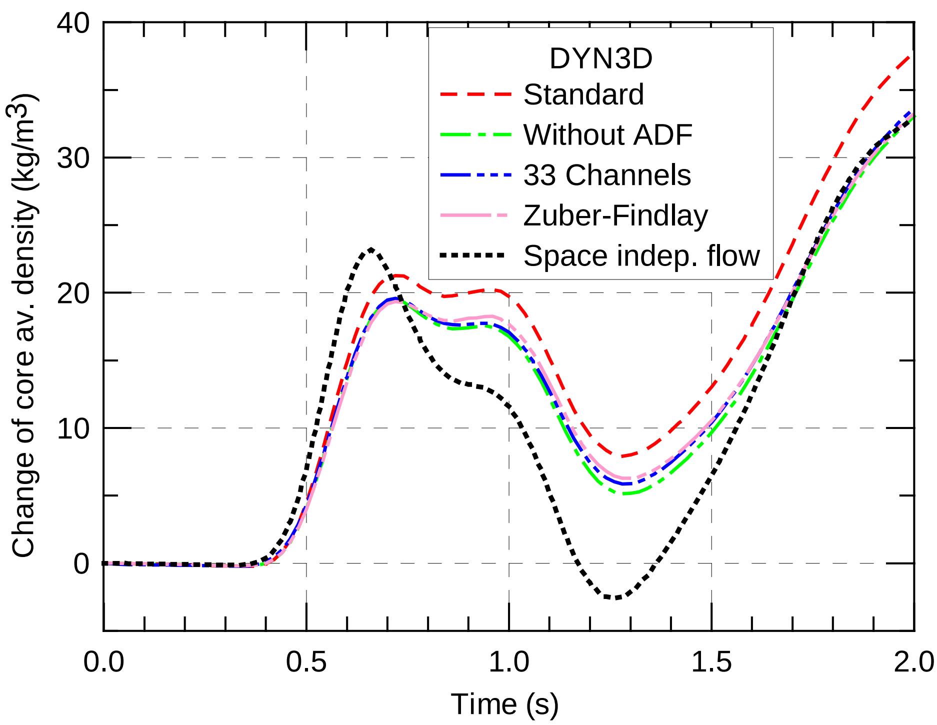

Fig. 7. Change of core averaged coolant density versus time of the DYN3D results.

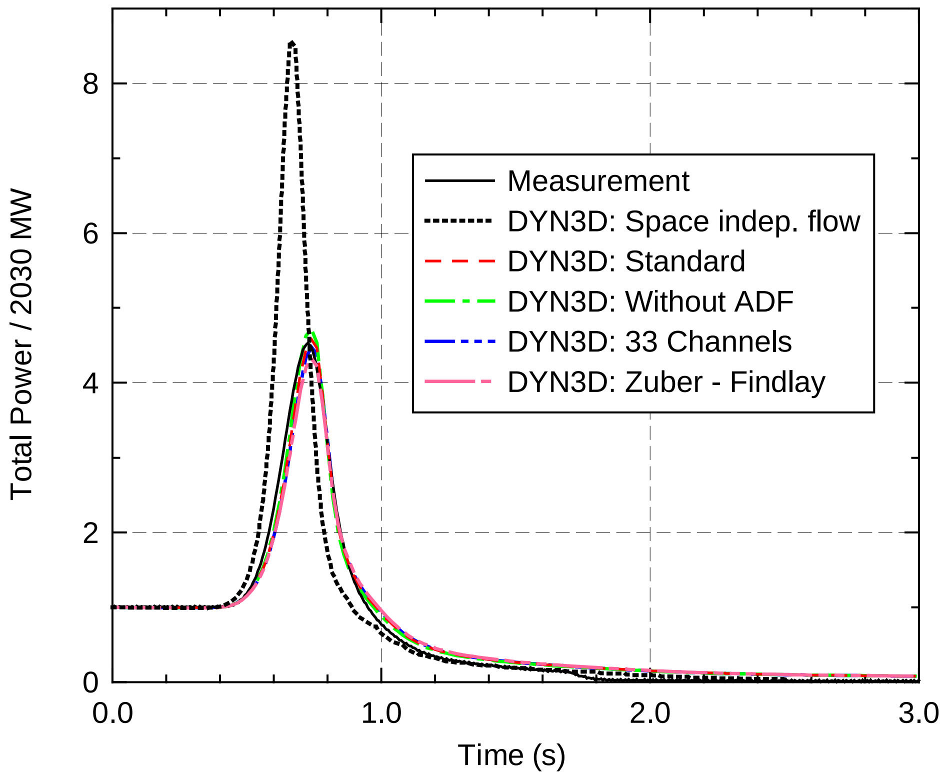

Fig. 8: Total power versus time of measurement and the different DYN3D calculations

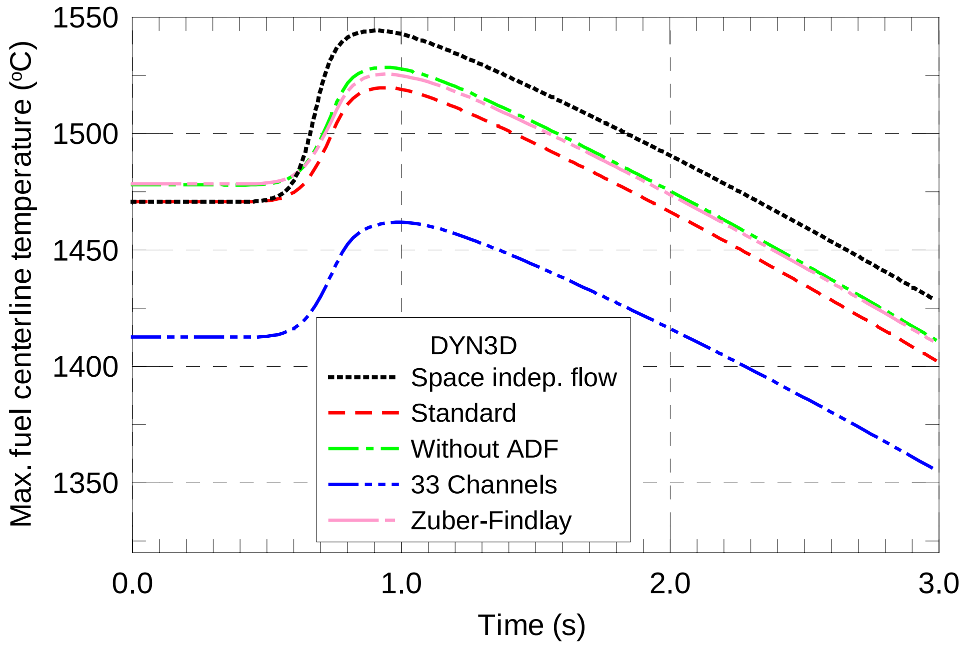

Fig. 9: The maximum fuel centerline temperature versus time of measurement and the different DYN3D calculations

CONCLUSIONS

The phase 2 of the OECD/NRC BWR TT Benchmark was analysed with the core model DYN3D by using different model options. The results of the standard case that includes the solution of the instationary mass balance equation, the consideration of one thermal-hydraulic channel per assembly, the ADF and the standard phase slip model of DYN3D show a good agreement with measured values. It was demonstrated that the consideration of the instationary mass balance equation is important for the transient behaviour. The influence of the ADF is small, if core-averaged values are compared. Differences in the results of single fuel assemblies were observed. If the core is described by only 33 thermal-hydraulic channels, it is observed too. Especially, larger deviations are observed, if extreme values as the maximum fuel temperature are considered. The ZUBER-FINDLAY slip model shows only small deviations to the standard model of MOLOCHNIKOV.

If the transient is calculated by the coupled code system, the core behaviour is strongly influenced by the interaction of the thermal hydraulics of the core with the rest of the system. Nevertheless the reactor core should be described as detailed as possible. This is important in the case that local and/or extreme values are considered.

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