OECD/NRC BWR TURBINE TRIP BENCHMARK: SIMULATION BY POLCA-T CODE

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ABSTRACT

Westinghouse transient code POLCA-T brings together the system thermal-hydraulics plant models and the 3D neutron kinetics core model. At present the code is under validation with emphasises on BWR stability and pressure increase transients' analyses. Code validation plan includes the calculations of Peach Bottom end of cycle 2 turbine trip transients and low-flow stability tests. The participation in the OECD/NRC BWR Turbine Trip Benchmark is a part of our efforts in this direction. The paper describes the objectives for Peach Bottom 2 (PB2) turbine trip analyses and benchmark participation. Brief overview of developed system input deck for PB2 and input 3D core model for PB2 EOC2 conditions is given.

The paper presents the results of benchmark exercise 1: system model without neutronics. Performed sensitivity studies cover the maximal time step, turbine stop valve position and mass flow, feed water temperature, and steam bypass mass flow. Obtained results including steam dome, core exit, lower and upper plenum, main steam line and turbine inlet pressures showed good agreement with measured data and previous RETRAN calculations.

Results of exercise 2: 3D core neutronic and thermal-hydraulic model with boundary conditions are also presented. Sensitivity studies include the coolant throttling on core periphery, core inlet temperature, cladding properties, and direct heating to core coolant and bypass (constant or density dependent). Calculated axial power distribution at the state prior to the transient agrees well with measured. Calculated during the transient power strongly depends on the direct heating, and also agree well with measured.

Thus the PB2 plant system thermal-hydraulic and core 3D models are validated separately. Next step is coupled calculations and validation of entire plant model. These activities will be performed in the frame of phase 3 of the benchmark.

1. INTRODUCTION

Westinghouse transient code POLCA-T brings together the system thermal-hydraulics plant models and the 3D neutron kinetics core model. The system thermal-hydraulics is based on RIGEL code [1]. The core model comprises the Westinghouse BWR and PWR 3D core simulator POLCA7 [2, 3]. POLCA-T also includes the BISON modules: SAFIR for balance of plant and PARA for steam lines simulation. At present the code is under validation with emphasises on BWR stability and pressure increase transients' analyses. Code validation plan includes the calculations of Peach Bottom end of cycle 2 turbine trip transients and low-flow-stability tests. The participation in the OECD/NRC BWR Turbine Trip (TT) Benchmark [4] is a part of our efforts in this direction [5].

2. OBJECTIVES FOR PEACH BOTTOM 2 TURBINE TRIP ANALYSES AND BENCHMARK PARTICIPATION

Our objectives for PB2 turbine trip analyses and benchmark participation are as follow:

1. Westinghouse had extensively used Peach Bottom 2 (PB2) end of cycle 2 (EOC2) TT1, TT2, and TT3 experiments for validation of BISON and RIGEL codes.
2. Qualify POLCA-T code for pressure increase transients (the PB2 EOC2 Turbine Trip experiments have been already included in code's Verification and Validation plan).
3. Validate 3D transient power distributions prediction for neutron kinetics with thermal-hydraulics feedback.
4. Validation of code's models for: heat transfer (fuel, gas gap, cladding, and to coolant); drift flux and void fraction (core average and local 3D distributions); two phase pressure drop; direct heating to the coolant in fuel assemblies and bypass channels; void and Doppler effects in 3D kinetics; control rod model and scram; jet pump model; turbine stop valves, steam bypass valves, and steam safety and relieve valves performance.
5. Validation of BWR entire plant system model coupled with 3D core neutron kinetics and thermal-hydraulics model.
6. Further development of 3D transient analysis methodology.

3. DEVELOPMENT OF PEACH BOTTOM 2 PLANT MODEL

System input deck for PB2 and input 3D core model for PB2 EOC2 conditions are developed [5 - 8]. The PB2 plant is described by reactor pressure vessel (RPV), recirculation loop, main steam lines (MSL), and steam bypass lines models. The RPV model contents down comer with feed water inlet and jet pump, lower plenum with control rods guide tubes, core with bypass channel, upper plenum, standpipes, steam separators and dryers, and steam dome (see figure 1). The recirculation loop comprises of suction and discharge coolant legs, and main circulation pump. The main steam lines includes steam lines, safety and relieve valves, turbine stop valves (TSV) and steam head. The steam bypass system model covers the bypass chest, valves, lines and orifice, and steam condenser.

The 3D core model comprises of 764 fuel channels (one per each fuel assembly) and 122 channels for radial reflector. Each channel is divided in 26 axial nodes (24 fuel nodes and 2 nodes for bottom and top reflectors).

Table 1 presents an overview of the contents of Peach Bottom 2 Plant Model.

Table 1. Peach Bottom 2 Plant Model

Model Parts	System	Core	Total
Systems	52	765	847
Volumes	1473	19895	21368
Flow paths (junctions)	1580	20487	22067
Heat structures	2563	38200	40763
Nodes	6898	97334	104232

The balance of plant is simplified to control rods speed and position controller, scram controller, jet pump drive flow controller, feed water controller, RPV water level controller, and turbine pressure controller. The assumed boundary conditions include feed water flow and steam bypass valve position versus time.

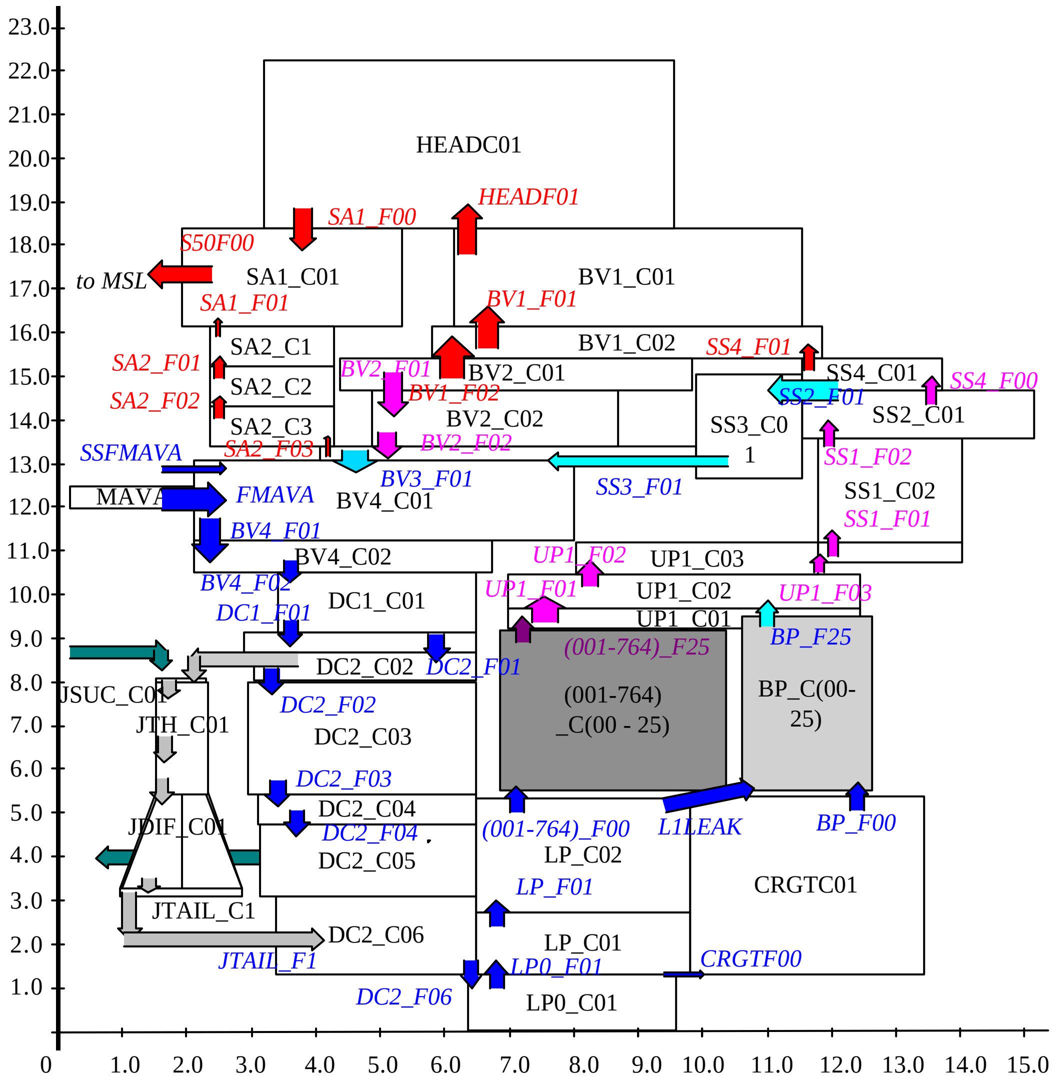


Figure 1. PB2 RPV nodalization in POLCA-T code.

4. RESULTS OF EXERCISE 1: SYSTEM MODEL WITHOUT NEUTRONICS

In order to validate the PB2 system model thermal-hydraulic calculations without core neutronics were performed. As in the coupled 3D neutronic / thermal-hydraulic calculations the core description was taken from POLCA7 input data. Thus the possible errors in input data are avoided and the core description is identical in all benchmark exercises 1, 2 and 3. Power versus timetable was given in this exercise 1 instead of performing the neutronic calculations.

Obtained results for steam dome and core exit pressures show good agreement with measured data (see figures 2 and 3). The pressures at lower and upper plenum also agree well with previous RETRAN calculations [6], distributed in the frame of the present benchmark as boundary conditions for exercise 2. Good agreement was observed also between obtained by POLCA-T results and measured data for RPV level, MSL and turbine inlet pressures.

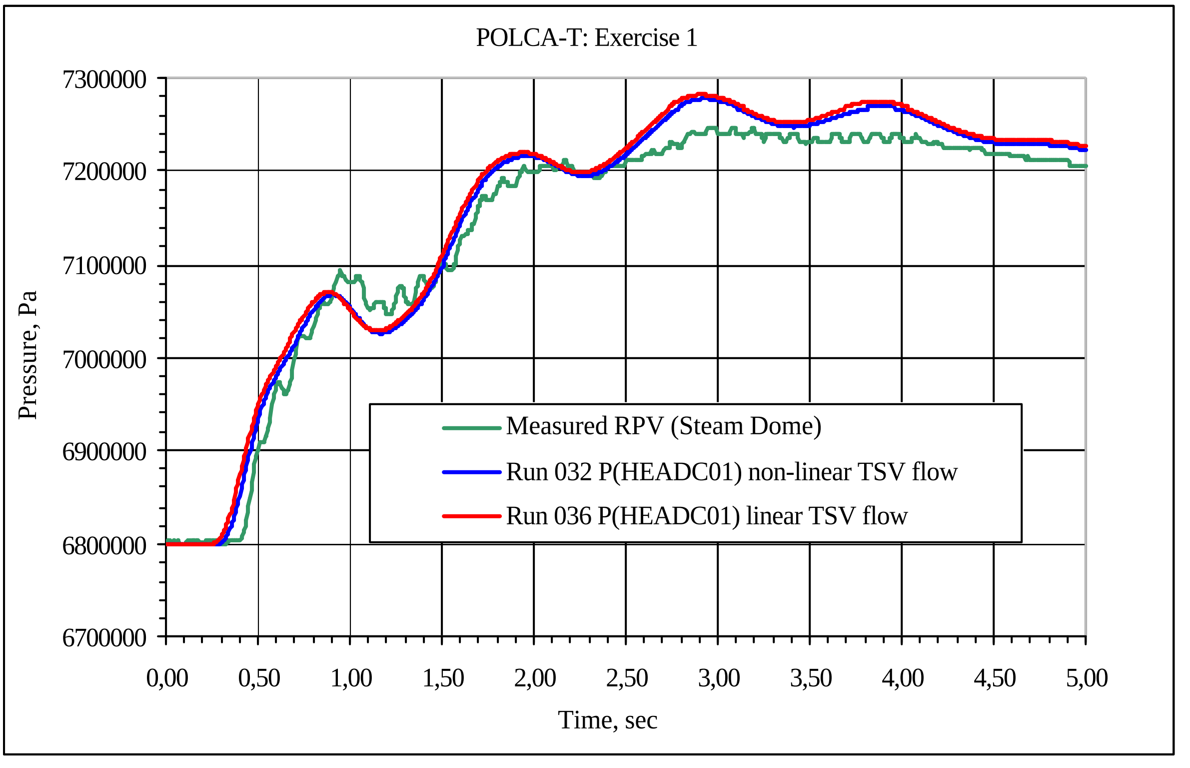


Figure 2. Steam Dome Pressure: effect of TSV flow linearization.

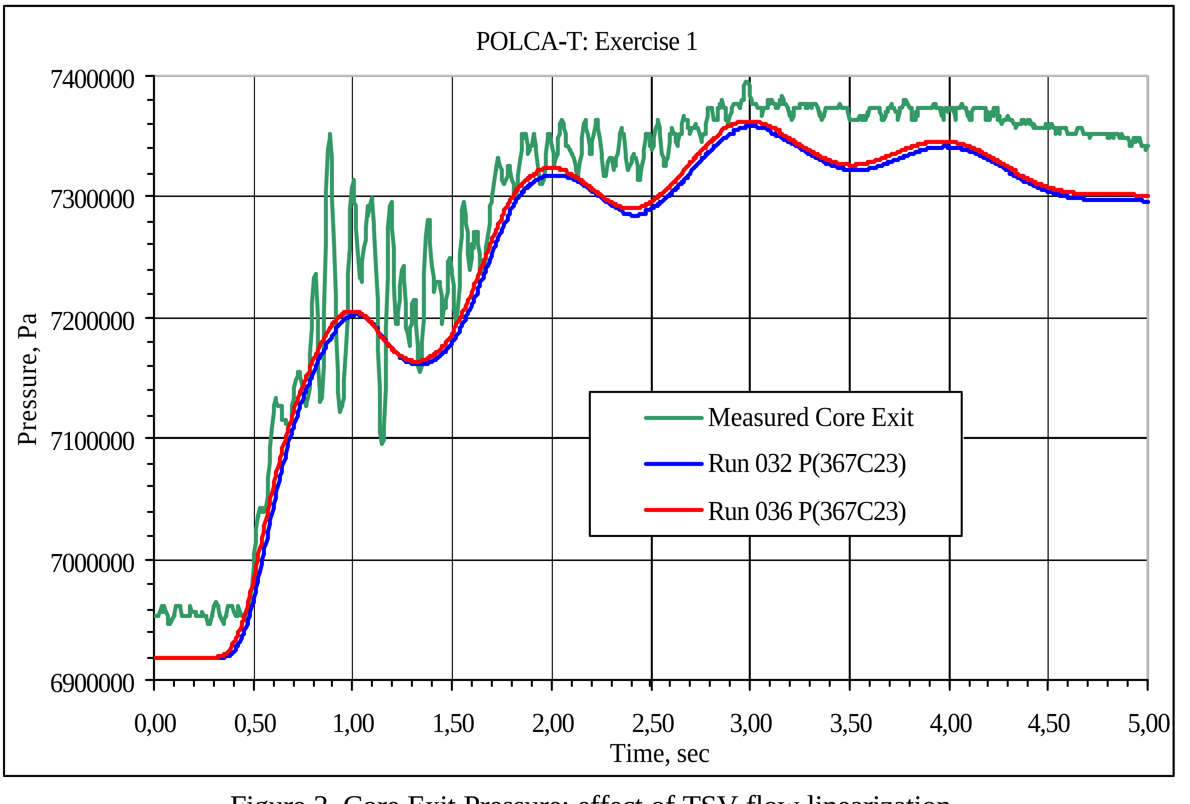


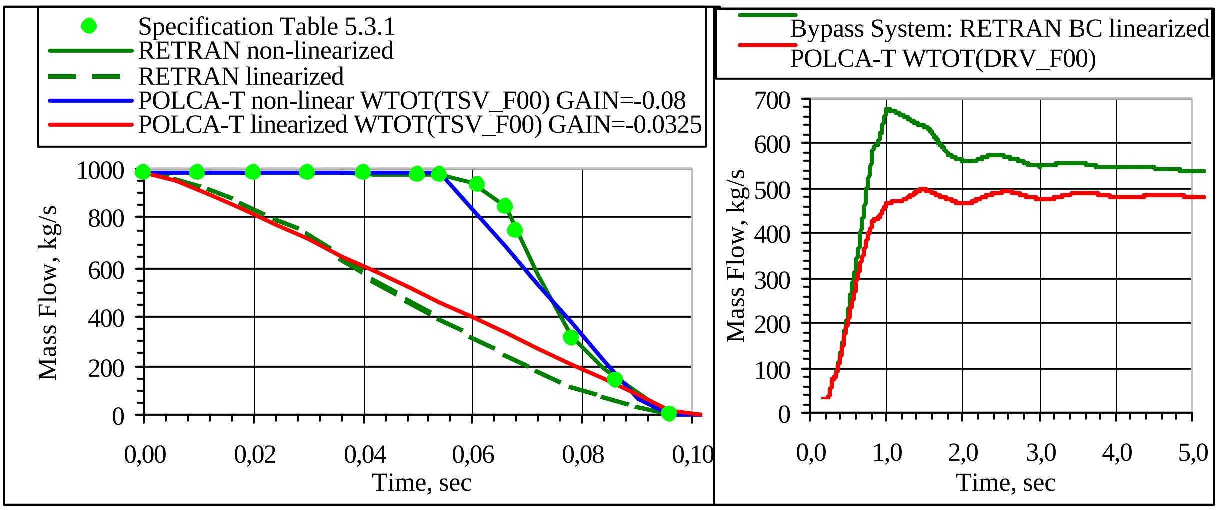
Figure 3. Core Exit Pressure: effect of TSV flow linearization.

4.1 SENSITIVITY STUDIES

Performed sensitivity studies pointed out that in case of so fast power increase the maximal time step should be lower than 0.01 second. There were not observed significant differences between the results obtained by time steps limited to 0.006 and 0.002 seconds.

It was found to be important to investigate also the effects on the results of some input data for plant nodalization, distributions and assumed boundary conditions. More sensitivity studies are performed for this purpose. Among the investigated parameters are the TSV position and steam mass flow: non-linear and linearized, steam bypass mass flow, and direct heating to core coolant and bypass: as specified constant or density dependent. Feed water temperature, jet pump parameters, steam separator inertia terms, and void fraction in bulk water (carry-under) are expected also to have strong influence on the obtained results.

The linearization of the TSV position and mass flow (see figure 4a) does not affect the obtained results when the neutronic calculation are not performed (see figures 2, 3, 5 and 6). Our preliminary results with coupled 3D neutronic / thermal-hydraulic calculations showed significant effect of TSV linearization [7, 8]. Feed water temperature does not change significant the core inlet temperature and has no effect on obtained results (see figures 5 and 6). Steam bypass mass flow (see figure 4b) has a strong effect on calculated RPV vessel pressure (see figures 5 and 6).



a) TSV flow linearization b) steam bypass flow

Figure 4. Sensitivity studies.

5. RESULTS OF EXERCISE 2: 3D CORE NEUTRONIC AND THERMAL-HYDRAULIC MODEL WITH BOUNDARY CONDITIONS

Polca7 and POLCA-T codes were modified in order to implemented a special option that links to Pennsylvania State University generated XS data library [4]. Codes' modifications and XS data link are tested by performed hot zero power calculations at 1% of rated power (32.93 MW) and fixed thermal-hydraulic conditions everywhere in the core (fuel temperature and coolant density) [5].

Core inlet coolant mass flow and temperature, and outlet pressure are specified boundary conditions for this case (exercise 2 with only 3D core model).

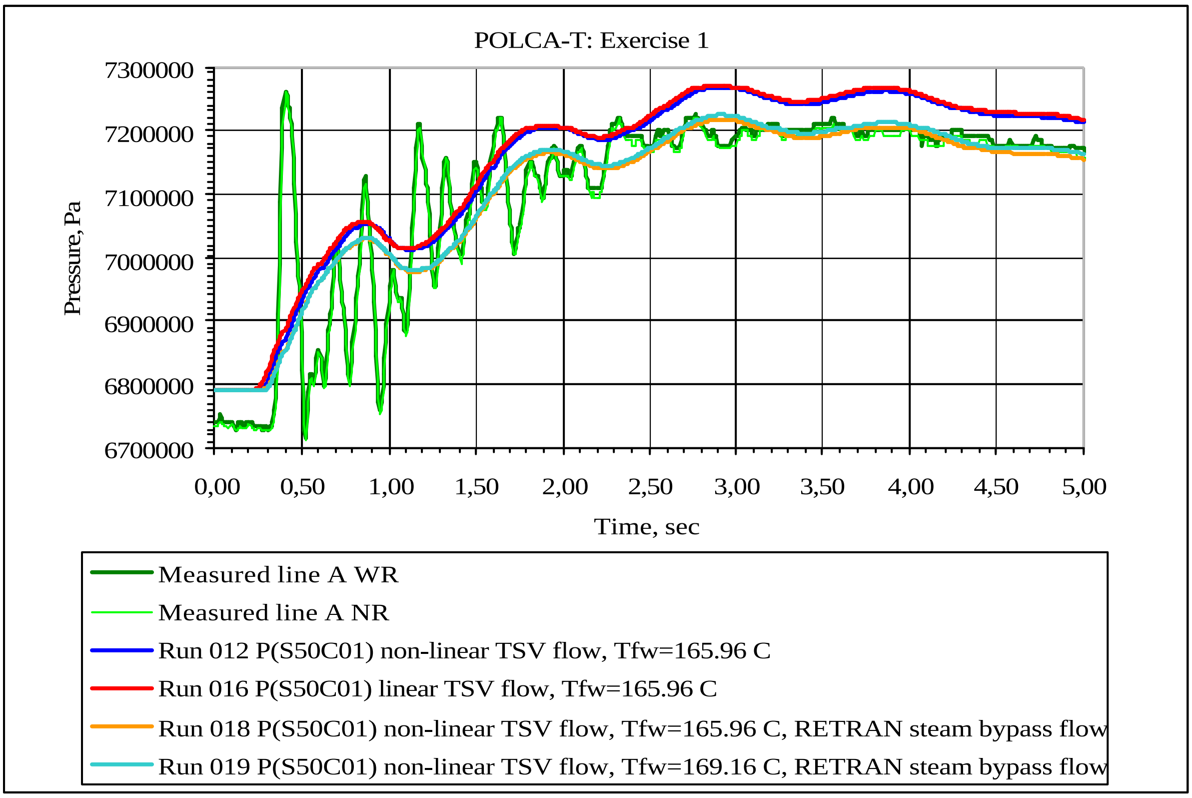


Figure 5. Main Steam Line Pressure: effect of TSV flow linearization, steam bypass flow, and feed water temperature.

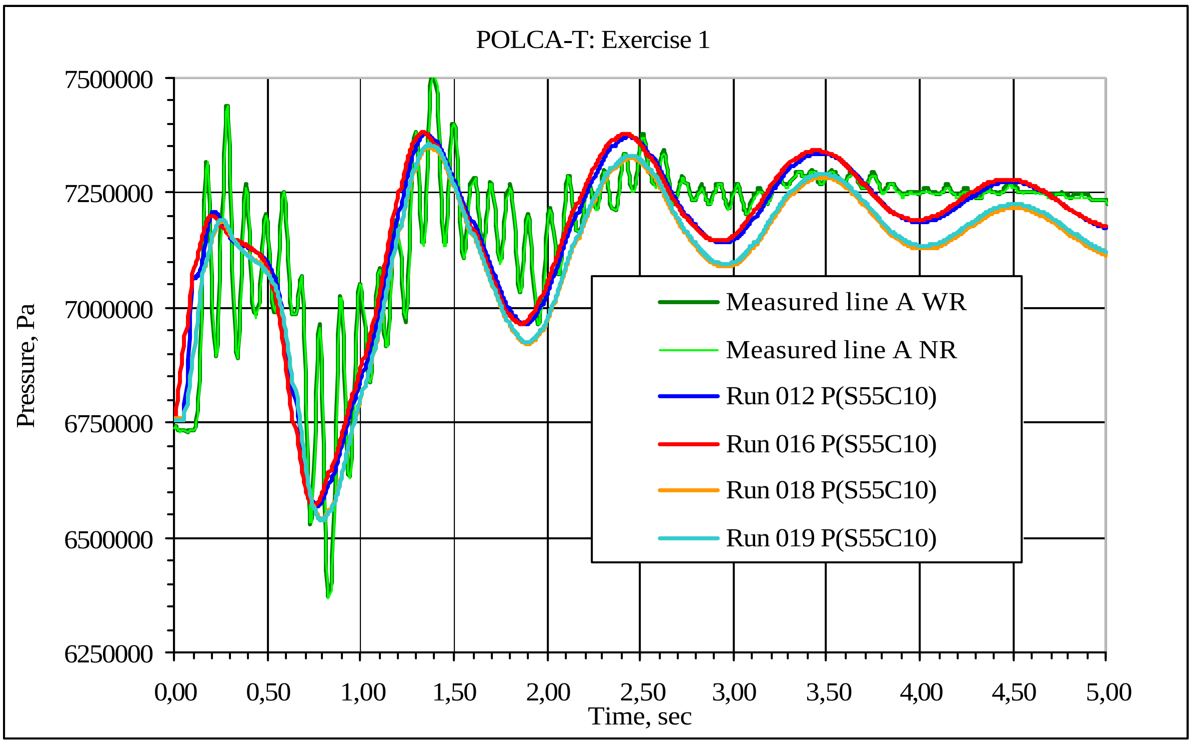
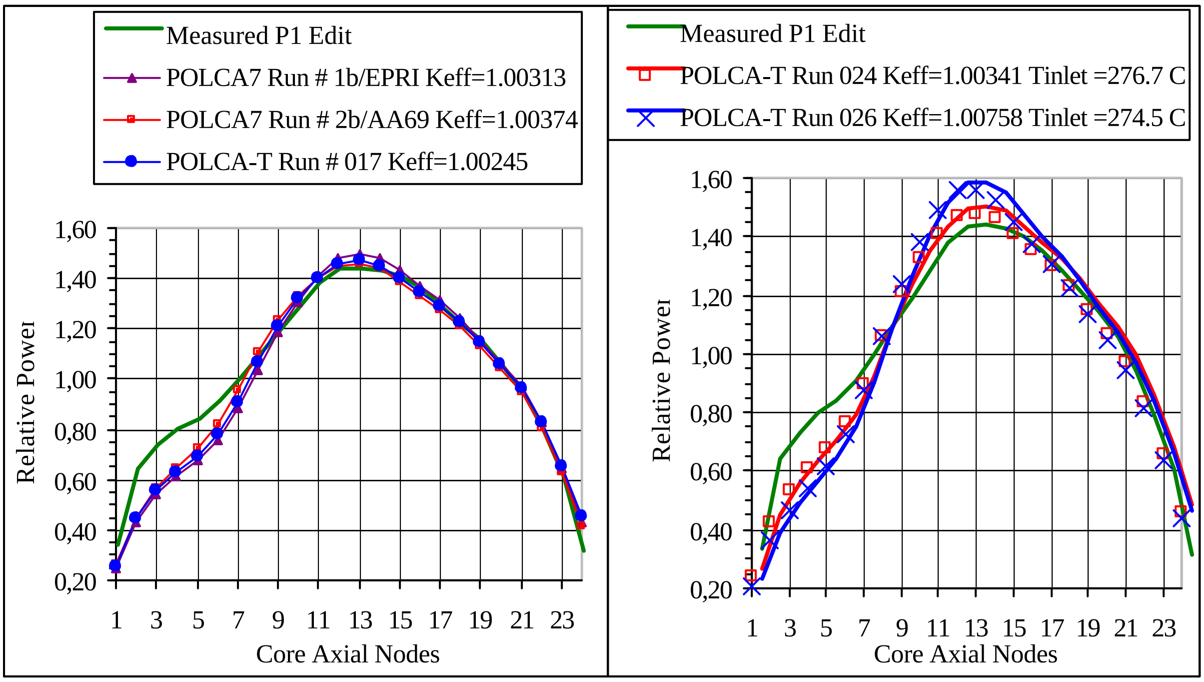


Figure 6. Turbine Inlet Pressure: effect of TSV flow linearization, steam bypass flow, and feed water temperature.

The results obtained both by POLCA7 and POLCA-T codes in the steady-state calculations of the state prior to the transient showed good agreement with P1 Edit for axial power distribution (see figure 7a). In these calculations the influence of the fuel assembly mass flow distribution due to orifice loss coefficients (throttling), void fraction (use of AA69 or EPRI correlations in POLCA7), inlet temperature/subcooling, cladding heat capacity and thermal conductivity, and direct heat to the coolant in fuel assemblies and bypass have been investigated. Among mentioned parameters only the inlet temperature/subcooling has significant impact on the core axial power distribution (see figure 7b and Table 2).



a) comparison of Polca7 and POLCA-T results b) sensitivity of the Results to Inlet Temperature

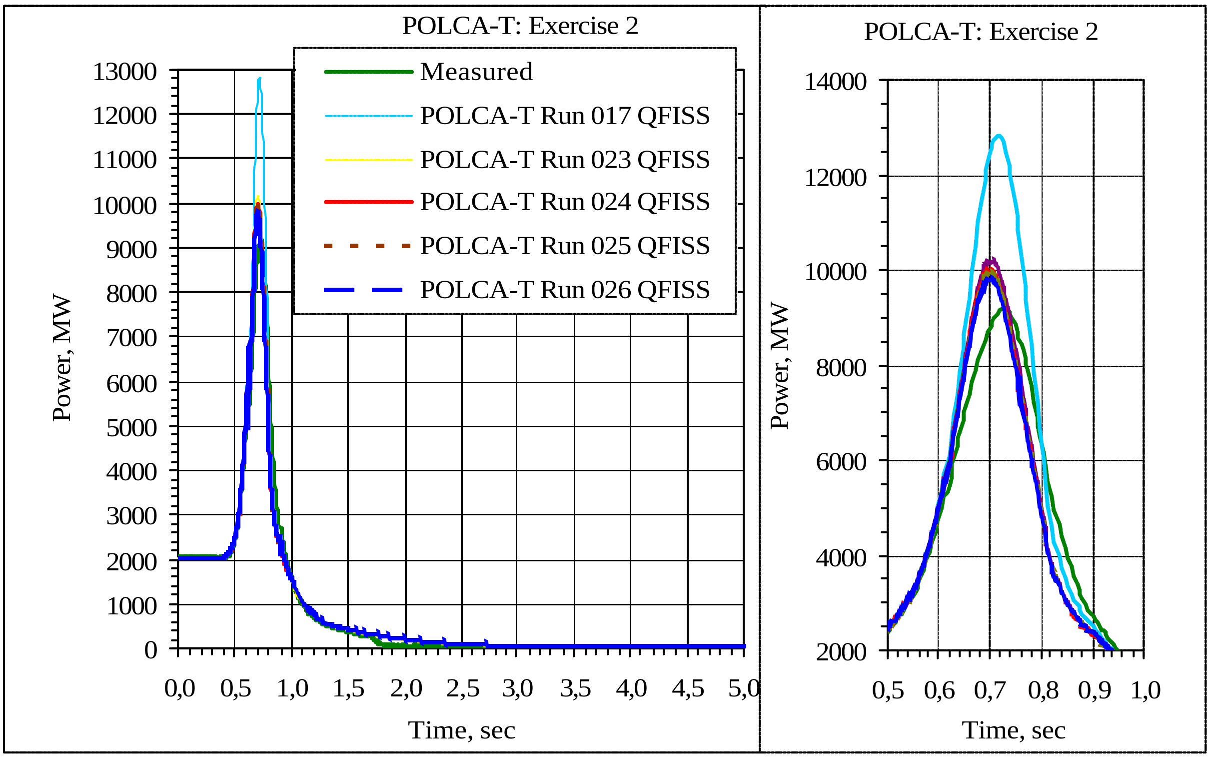
Figure 7. Core Average Axial Power Distribution.

Table 2. Sensitivity of the parameters of the state prior to the transient to some input data and models' options

Parameter	Value	K_{eff}	Axial peaking factor	Radial peaking factor	Axial offset, %	Core average void, %
Orifice loss	5.95*	1.00303	1.467	1.454	8.56	31.53
	7.50	1.00305	1.471	1.459	9.11	31.85
Void Correlation	EPRI	1.00313	1.473	1.499	11.97	31.99
	AA69	1.00374	1.467	1.459	8.86	30.99
Inlet Temperature	274.5	1.00758	1.451	1.562	12.26	28.76
	276.7	1.00341	1.464	1.480	11.40	31.52
Cladding Properties	*	1.00341	1.464	1.480	11.40	31.52
Thermal conductivity	-10%	1.00335	1.464	1.480	11.39	31.52
Direct Heating	2.00*	1.00245	1.462	1.469	10.69	31.96
	3.41**	1.00302	1.463	1.476	11.10	31.70
	4.11**	1.00341	1.464	1.480	11.40	31.52

Note: * - specified, ** - coolant density dependent.

The transient calculations performed by POLCA-T code showed also significant effect of the inlet temperature on the value of observed power peak. Much stronger effect on the power peak value however had the direct heating (see figure 8 and Table 3).



a) 5 seconds of the transient b) zoom 0.5 - 1.0 sec.

Figure 8. Fission Power.

Table 3. Sensitivity of the Transient Peak Power to some input data and models' options

Parameter	Value	Time of peak power, sec	Calculated peak power, % of rated	Calculated peak power, MW	Calculated/ measured peak power	Effect on peak power, %	Run #
Measured		0.726	279.1	9190	1.0000		
Inlet Temperature	274.5	0.702	297.6	9799	1.0663		026
	276.7	0.702	302.7	9967	1.0846	1.83	All others
Cladding Properties	*	0.702	302.7	9967	1.0846		All others
Thermal conductivity	-10%	0.702	301.7	9934	1.0810	-0.36	025
Direct Heating	2.00*	0.714	389.5	12826	1.3956		017
	3.41**	0.708	324.7	10692	1.1634	-23.22	023
	4.11**	0.702	302.7	9967	1.0663	-31.10	024

Note: * - specified, ** - coolant density dependent.

CONCLUSIONS

The results of performed calculations and sensitivity studies can be summarised and the conclusions can be drawn as follow:

1. The POLCA-T results are in good agreement with measured data: steam dome and core exit pressures, MSL and turbine inlet pressures, RPV water level, etc. The results for pressure at lower and upper plenum also showed good agreement with previous RETRAN calculations, distributed in the frame of the present benchmark as a boundary conditions for exercise 2.

2. In performed sensitivity studies there were not observed significant differences between the results obtained by time steps limited to 0.006 and 0.002 seconds.

3. Linearization of TSV mass flow does not have significant effect on the obtained results.

4. Feed water temperature does not change significant the core inlet temperature and has no effect on obtained results.

5. Steam bypass mass flow has a strong effect on calculated RPV vessel pressure.

6. The steady-state results for the state prior to the transient obtained with POLCA7 and POLCA-T codes are very close to each other. They are in good agreement with the available measured data.

7. The transient results obtained with POLCA-T code are in good agreement with the available measured data.

Thus the PB2 plant system thermal-hydraulic and core 3D models are validated separately. Next step is coupled calculations and validation of entire plant model. These activities will be performed in the frame of phase 3 of the benchmark.

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