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POLCA-T Simulation of OECD/NRC BWR Turbine Trip Benchmark Exercise 3 Best Estimate Scenario TT2 Test and Four Extreme Scenarios

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Westinghouse transient code POLCA-T brings together the system thermal-hydraulics plant models and the 3D neutron kinetics core model. Code validation plan includes the calculations of Peach Bottom end of cycle 2 turbine trip transients and low-flow stability tests. The paper describes the objectives, method, and results of analyses performed in the final phase of OECD/NRC Peach Bottom 2 Boiling Water Reactor Turbine Trip Benchmark. Brief overview of the code features, the method of simulation, the developed 3D core model and system input deck for Peach Bottom 2 are given.

The paper presents the results of benchmark exercise 3 best estimate scenario: coupled 3D core neutron kinetics with system thermal-hydraulics analyses. Performed sensitivity studies cover the SCRAM initiation, carry-under, and decay power. Obtained results including total power, steam dome, core exit, lower and upper plenum, main steam line and turbine inlet pressures showed good agreement with measured plant data. Thus the POLCA-T code capabilities for correct simulation of turbine trip transients were proved.

The performed calculations and obtained results for extreme cases demonstrate the POLCA-T code wide range capabilities to simulate transients when scram, steam bypass, and safety and relief valves are not activated. The code is able to handle such transients even when the reactor power and pressure reach values higher than 600 % of rated power, and 10.8 MPa.

KEYWORDS: BWR, Turbine Trip, Benchmark, coupled code, 3D neutron kinetics, core and system thermal-hydraulics

1. Introduction

The complexity of phenomena, complicated technological processes, and composite equipment at nuclear power plant require the application of appropriate theoretical model of the entire reactor system. The need for comprehensive analysis states for the modeling of overall system dynamics. System's model should cover the processes of heating and cooling in the "non-nuclear" equipment, of neutronics and kinetics phenomena in the core, and of cooling of the fuel.

Moreover, the thermal-hydraulics and kinetics phenomena should be considered in their interactions in different events, such as oscillation events, operational events and loss of coolant accidents situations.

The application of such a system provides a means to forecast safety margins both in normal operation and during the postulated transients and accidents. This approach decreases the conservative unjustified assumptions, and overall uncertainty of the reactor safety analyses.

Best estimate reactor analysis codes incorporate a full 3D model of a reactor core into a system transient code. The coupled code provides a means to simulate interactions between reactor core behavior and plant dynamics.

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Boiling water reactor (BWR) turbine trip (TT) is a pressure increase transient that represents a demanding application of coupled 3D neutron kinetics and thermal-hydraulics codes. The objective of OECD/NRC BWR TT Benchmark [1] Exercise 3 Best Estimate (BE) Scenario is to perform comparison of coupled codes' results with available from Peach Bottom 2 (PB2) turbine trip test 2 (TT2) measured data. Challenging four extreme scenarios of Exercise 3 are aimed to test the coupled codes at extreme conditions and to provide data for code-to-code comparison.

Westinghouse participation in the OECD/NRC BWR TT Benchmark is a part of our efforts in the development and validation of coupled POLCA-T code [2, 3]. Emphasize is on BWR stability and transients' analyses.

Results of POLCA-T simulation of benchmarks' exercises 1 and 2 have been presented in [4]. Present paper describes the objectives, method, and results of the POLCA-T simulation of benchmarks' exercise 3 BE scenario and extreme scenarios.

2. POLCA-T Code, PB2 Plant Model and TT Benchmark Simulation

2.1 POLCA-T Code

The POLCA-T is a coupled computer code with 3D neutron-kinetics and thermal-hydraulics models. The code is able to perform steady state and transient analysis of BWR. Westinghouse experiences in developing models and tools for analysis of BWR reactors (POLCA7 [5], BISON, GOBLIN and RIGEL codes) is integrated in the code and the best features of those codes are adopted in POLCA-T.

The code utilizes new advanced methods and models in neutron kinetics, thermal-hydraulics and numerics. The main features of the code can be summarized as follow:

- Full 3D model of the reactor core.
- Advanced thermal-hydraulic model with thermal non-equilibrium description of the flowing steam-water mixture and its coupling to the heat structures.
- Boron transport and non-condensable gas transport and solution/dissolution.
- Use of the same thermal-hydraulics model for core and plant systems.
- 2D Fuel rod heat transfer model, fuel performance STAV7 code gas gap model and fission gas generation, and complete range of heat transfer regimes. Model of all plant heat structures.
- Dry-out and DNB correlations (using pin power distributions model).
- Full geometrical flexibility of the code. Code is able to analyze different power plants and test facilities.
- Balance of plant, control and safety systems: The RPV, external pump loops, steam system and feed-water system, ECC systems and steam relief system are modeled to the desired detail (SAFIR models are also used for this purpose).
- Stable numerical method. Low dependence on the size of the time step since the implicit numerical integration is close to second order by means of θ-weighting.

Because of the above mentioned features POLCA-T code makes possible the comprehensive approach to plant analysis with full consistency in steady state and transient results, and also between predicted core and system parameters and their behavior in very wide range of phenomena and processes.

2.2 POLCA-T PB2 Plant Model

Developed POLCA-T system input deck, input 3D core model, and balance of plant model (BOP) for PB2 end of cycle 2 (EOC2) conditions are described in [4]. In this paper the brief overview of the POLCA-T PB2 plant model is given.

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 of down comer with feed water inlet and jet pumps, lower plenum with control rods guide tubes, core with bypass channels, upper plenum, standpipes, steam separators and dryers, and steam dome.

The nodalization of the reactor pressure vessel (RPV) is illustrated in [4]. The recirculation loop comprises of suction and discharge coolant legs, and main circulation pumps. The main steam lines includes steam lines, four groups safety and relief valves (SRV), turbine stop valves (TSV) and steam head. The steam bypass system model covers the bypass chest, bypass valves (DRV), 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).

The BOP is simplified to control rods speed and position controller, scram controller, jet pumps drive flow and main circulation pumps controllers, feed water controller, RPV water level measurement and controller, SRV controllers and turbine pressure controller. The assumed boundary conditions include feed water flow and steam bypass valve position versus time.

The four groups SRV model and their control is important for the analyses of the exercise 3 four extreme scenarios. The nodalization of the MSL with SRV is illustrated in figure 1. Figure 2 presents the SRV controllers developed with Westinghouse SAFIR code and implemented in POLCA-T model. SRV set-points and capacities are described in Table 1.

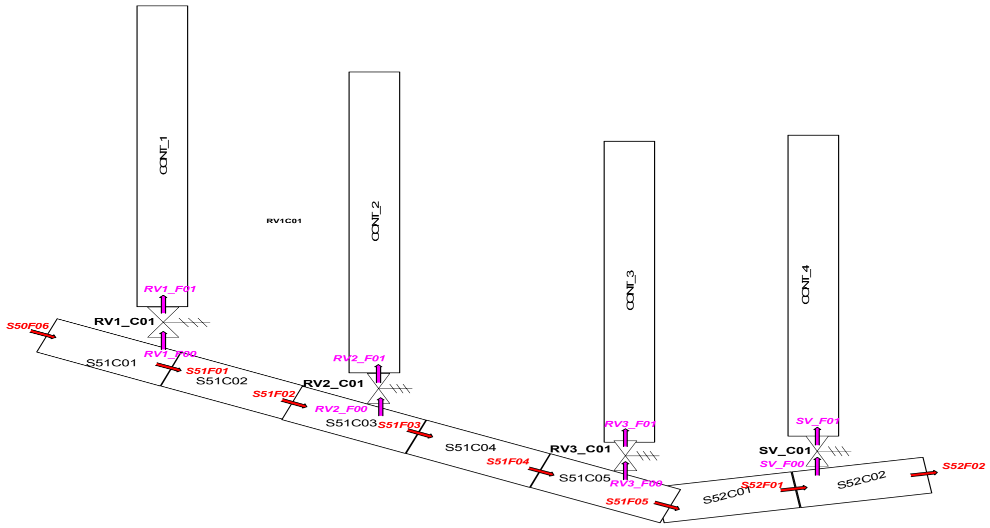


Fig.1 POLCA-T code MSL with SRV nodalization.

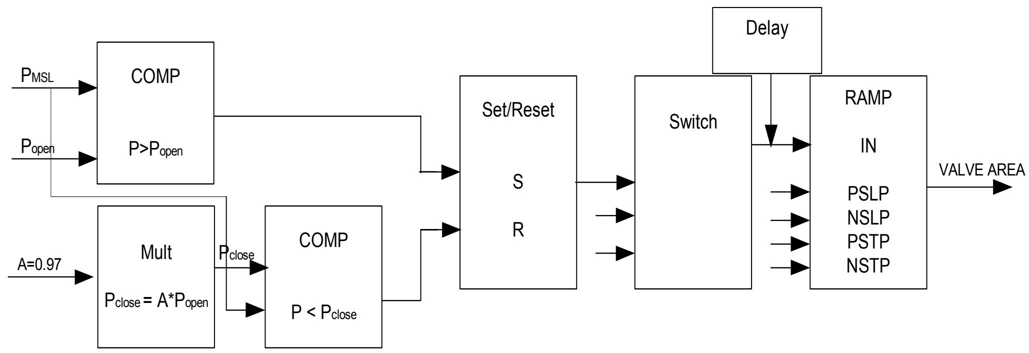


Fig.2 POLCA-T code SRV Controller - Open/Close Logic (SAFIR) for each SRV group.

2.3 POLCA-T Simulation of TT Benchmark

POLCA-T utilizes fully the 3D methodology used by coupled codes. The PB2 plant model and POLCA-T input data were qualified using separate models validation and coupled 3D neutron kinetics and thermal-hydraulic analyses.

Table 1 SRV set-points and capacity [1]

	Valves' Group	*Number of valves*	*Set Pressure, MPa*	*Capacity kg/s each*	*Total Capacity, kg/s*	*Open delay time, msec*	*Open stroke time, msec*
Relief valves	RV1	4	7.720	103.19	412.76	400	150
	RV2	4	7.789	104.20	416.80	400	150
	RV3	3	7.858	105.08	315.24	400	150
	Total	11 (5)			1144.80		
Safety valves	SV	2	8.582	118.40	236.80	0	300
	Total	13			1381.60		

Notes: Close pressure is equal to 97% of the set pressure; capacity is given at 103% set pressure; close delay time and close stroke time are set to 0 sec.

The PB2 plant system thermal-hydraulic and core 3D models were validated separately in the frame of OECD/NRC BWR TT Benchmark Exercises 1 and 2 [4]. System thermal-hydraulic model was validated in Exercise 1 calculations: system analysis with total power given as boundary condition [4]. The obtained results and POLCA-T system model response to TSV closure were compared with measured plant data [6].

Three dimensional core neutron kinetics and thermal-hydraulic model was validated using benchmark's exercise 2 [4]. Exercise 2 consists of steady-state hot zero power (HZP) calculations with fixed state parameters (i.e. without thermal-hydraulic feedback); steady-state hot full power (HFP) calculations with thermal-hydraulic feedback, and core transient calculations. In all three analyses core inlet flow and temperature, and outlet pressure were given as boundary conditions. The correctness of generated cross section data and their link to and use by POLCA-T code were confirmed by HZP calculations. HFP calculations with thermal-hydraulic feedback validated the entire core model. Obtained by POLCA-T results of steady-state calculation (HZP and HFP), as well as the initial state prior to the transient were compared with results of Westinghouse core simulator POLCA7 [5]. Among the observed good agreements were the core node-wise power distributions. The transient analyses confirmed the correctness of generated neutron kinetics data as well as the transient response of the entire core model.

Hereafter the method and results of the simulation of benchmark's exercise 3 BE and extreme scenarios are present.

Coupled calculations enable the validation of entire plant model. In present work the exercise 3 BE scenario was used for this purpose. Performed sensitivity studies include the effect of scram initialization, carry-under (void fraction in bulk water) and decay power.

The analyses of extreme scenarios require the model of safety and relief valves (SRV) and their control. These cases give the possibility to test the coupled code behavior and capabilities under quite extreme condition. Moreover, they give good opportunity to investigate the feedback mechanism in coupled core 3D neutron kinetics and system thermal-hydraulics analyses.

Four extreme scenarios proposed and analyzed in the frame of the benchmark exercise 3 are as follows:

- case A - TT without steam bypass system opening;
- case B - TT without scram;
- case C- combined cases A and B, i.e. without steam bypass system opening and without scram; and
- case D - case C without activation of the safety and relief valves (SRV).

The effect of decay power was investigated in performed sensitivity studies for all four extreme scenarios.

3. POLCA-T Results of Exercise 3

3.1 Best Estimate Scenario Results: Comparison with PB2 TT2 Test

After the stand-alone validation of PB2 plant system thermal-hydraulic and core 3D models, the core model was implemented in the system model. In order to validate the entire plant model the coupled calculations were performed in the frame of benchmark's exercise 3 BE scenario.

The predicted major plant process parameters, the total power, the dome pressure, and the turbine inlet pressure, were compared with the measured data in this work as shown in Figures 3 and 4.

Table 2 presents the sequence of events. In general predicted by POLCA-T system response agree well with one observed in TT2 test.

Obtained results including total power (see figures 3), steam dome (see figure 4a), turbine inlet (see figure 4b), and MSL (see figure 8a) pressures showed good agreement with measured data. The POLCA-T code slightly over predicts the peak power with about 10%. Use of decay power model reduces the over prediction to less than 7%. The predicted timing of the maximum power is by about 30 milliseconds later than the measured. It is very difficult to identify influential factors for this amount of subtle discrepancy. It has to be noted that among them is probably the assumed in the benchmark specification the constant (in time, temperatures and fuel types) heat conductivity through the gas gap between the fuel pellets and the cladding. The effect of investigated in the sensitivity study parameters on the timing of the power peak is quite contradictory. While the lower carry-under decreases the discrepancy of predicted timing of the power peak the decay power increases it.

Core exit, lower and upper plenum pressures, as well as the RPV level agree well with TT2 data [4].

Figure 10 shows the total reactivity history during this transient. Calculated by POLCA-T reactivity behavior agrees well with TRAC-M/PARCS results [7] and is close to average behavior obtained by others benchmark participants.

Obtained by POLCA-T core average axial power and void fraction distributions were compared with results of Westinghouse core simulator POLCA7 [5]. Calculated core average axial power distribution at the state prior to the transient agreed well with the plant's P1 Edit data. The results showed also good agreement with POLCA7 results for core axial and radial power distributions. Transient core 3D power distributions were also investigated. Figure 12a illustrates the change of axial power distribution in fuel assembly (FA) number 75 (52-17) during the transient. Significant redistribution of the power is observed at the end of calculations (at 5 seconds) due to the SCRAM.

Thus performed calculations and obtained results for PB2 TT2 proved the POLCA-T code capabilities for correct simulation of pressurizing transients with very fast power increase and the PB2 model was qualified for further analysis of extreme scenarios.

3.2 Best Estimate Scenario Sensitivity Study

Performed during the coupled calculations, sensitivity studies include the effect of SCRAM initialization, carry-under (void fraction in the bulk water), and decay power. The results are present in figures 4 and 5. The cases that have been run, the investigated parameters and their values or set-points are as follows:

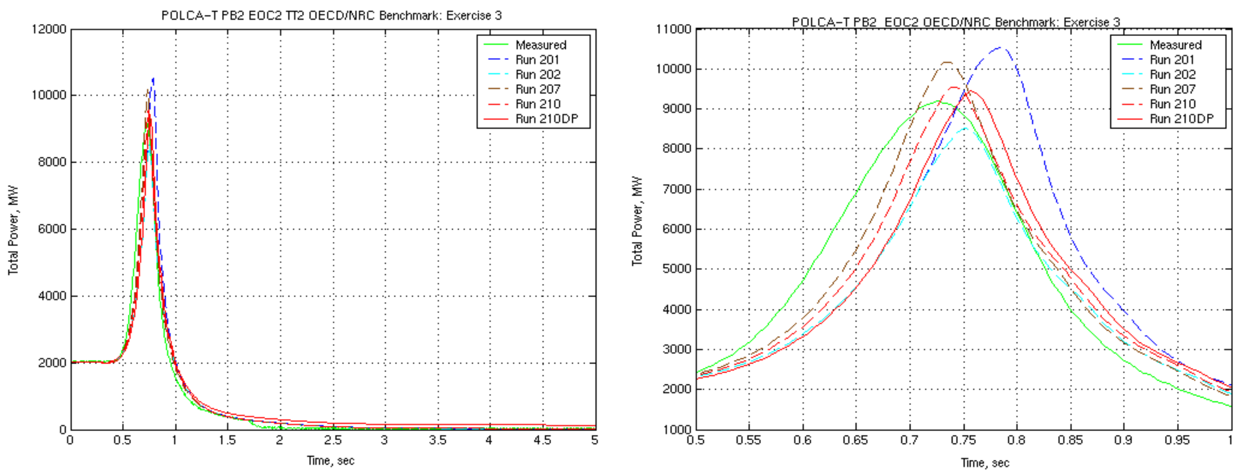
- Run 201: SCRAM initialization at time 0.75 seconds
- Run 202: SCRAM initialization at 95% rated power
- Run 207: Steam separator (SS) improved model (according to the recommendation of [7]), lower carry-under
- Run 210: New code version release, decay power not used in this run
- Run 210DP: Decay power model used

Two ways to initiate the SCRAM were considered: first one with SCRAM initiated at 0.75 seconds, as it was specified in the benchmark [1]; second one with SCRAM initiated when the power reaches 95%

of rated, as it was set-up during the PB2 turbine trip tests [6]. It was concluded that the SCRAM initialization has strong effect on the power peak value (see figure 3). Initialization by power set point reduces the peak value in direction to measured value. The timing of the power peak is also shifted closer to that observed in the TT2 test. In case of initialization by power set point, the steam dome (see figure 4a), core exit, and turbine inlet pressures (see figure 4b) are reduced after 2.5 seconds of the transient. Thus the power set point improves the agreement with the measured power and in contrary increases the deviation from measured pressures. RPV level was not affected by the change of SCRAM initialization. In the results present hereafter only SCRAM initiated at 95% power set-point is used.

In the very first coupled calculation it was found that the steam separator (SS) model gives unrealistically high void fraction in bulk water. This pointed out the need to improve the steam separation model. The solution has been found using moment of inertia in the junctions in upper plenum, SS inlet, liquid and gas paths. The conclusions of sensitivity study performed by TRAC-M/PARCS code in the frame of the PB2 TT benchmark [7] were taken in to account. The improved SS model gave more realistic value of carry-under. This increases peak power value which however stays below the one observed in the calculations with higher carry-under and SCRAM initiation set by time (see figure 3b). The time of the peak is shifted further close to that observed in the TT2 test. Steam dome (see figure 4a) and core exit pressures are increased after 0.5 - 1 second of the transient and showed good agreement with measured data. The effect of carry-under on turbine inlet pressure (see figure 4b) is observed after 1 second of the transient and it became significant after about 2.5 seconds of the transient. Turbine inlet pressure in this run agreed well with measured data. Predicted RPV level showed also better agreement with measured data when the improved SS model was used.

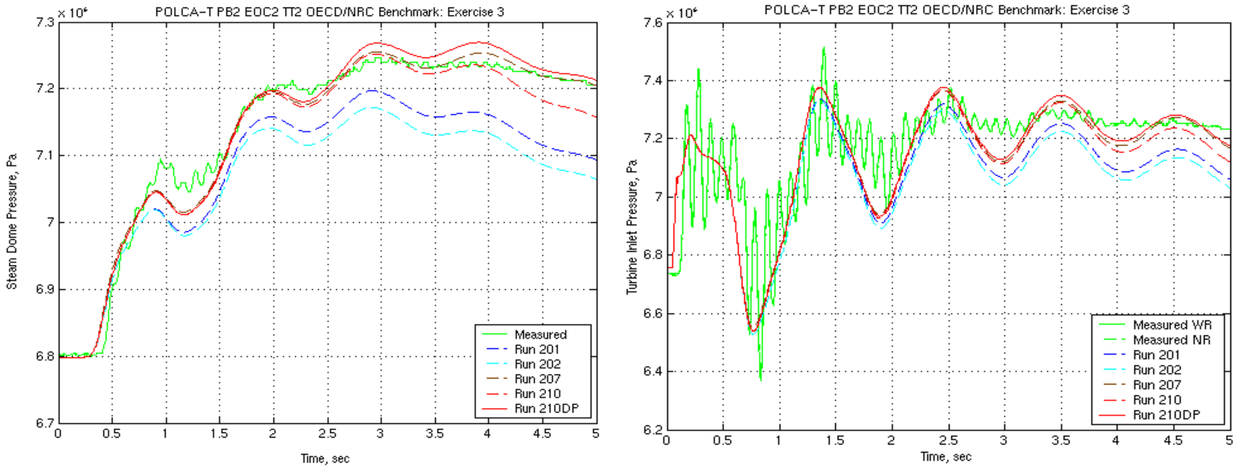
It was found important to investigate the effect of use of decay power model. Suffix DP in the runs' identification shows the use of decay power model. Use of the model reduces the code's over prediction of the peak power and increases the discrepancy of predicted timing of the power peak (see figure 3b). Decay power increases the steam dome and turbine inlet pressures after about 2 – 2.5 seconds of the transient (see figure 4). Thus the use of decay power model improves the agreement of predicted time histories with measured data. The effect of decay power will be discussed hereafter also in connection to the results of the simulation of extreme scenarios.



a) total power

b) total power zoom 0.5 – 1.0 second

Fig.3 PB2 TT Benchmark Exercise 3 BE scenario total power.



a) steam dome pressure b) turbine inlet pressure

Fig.4 PB2 TT Benchmark Exercise 3 BE scenario steam dome and turbine inlet pressures.

3.3 Results of Extreme Scenarios Simulation

In order to test the POLCA-T code behavior and capabilities under extreme condition, to investigate the feedback mechanism and to submit the results for benchmark's code-to-code comparison four extreme scenarios as described before in section 2.3 were analyzed.

The predicted major plant process parameters, the total power, the dome pressure, the MSL pressure, the SRV mass flow, and the total reactivity obtained in this work are shown from Figure 5 through Figure 10. The results of BE scenario as well as the measured data (when available) are present in the figures for comparison.

Table 2 presents the sequence of events, including the activation and the closure of SRV, when observed. SRV is activated in extreme cases B and C only. However, even in these cases the system pressure never reaches the set point of safety valves to open. Relief group 1 and 2 open in case B, and groups 1, 2 and 3 open in case C. Figure 9 illustrates the activation and work of the SRV in cases B and C. This figure shows the mass flow through each SRV group as well as the total mass flow through the valves.

Obtained results including total power (see Figures 5 and 6), steam dome (see Figure 7), core exit, and MSL pressures (see Figure 8), showed quite different behavior in each extreme case. The reactor power has only one peak value in cases when the scram is activated (BE and case A). In cases without SCRAM: B, C and D the power has several peaks, in general with decreasing magnitude. In case D the second and following peaks has approximately the same magnitude. For all cases, the first peak has the higher value of 287, 366, 417, 604 and 604% of rated power for BE, A, B, C, and D scenarios (see Table 2). The maximal steam dome pressure increases from BE scenario to extreme cases, from case A to B, etc. In case D the maximal steam dome pressure of 10.8 MPa is observed at the end of calculations (at 10 seconds). The core exit, lower and upper plenum, MSL (see Figure 8), and turbine inlet pressures show similar to the steam dome pressure behavior.

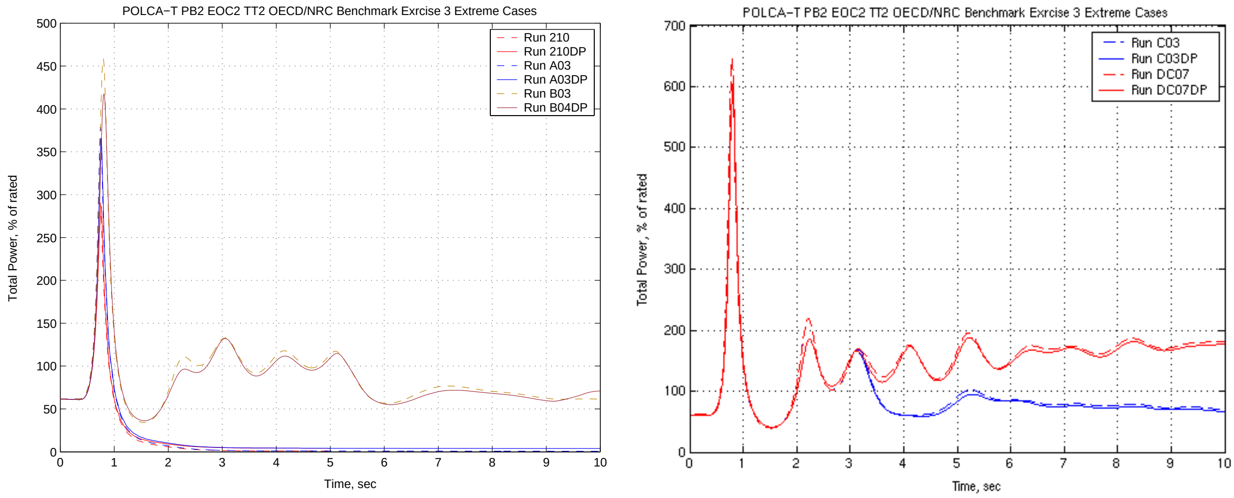
Figure 10 shows the total reactivity history during this transient. Calculated by POLCA-T reactivity behavior agrees well with TRAC-M/PARCS results [7] and is close to average one obtained by others benchmark participants.

The effect of decay power was investigated in all extreme cases and found to be strong on the pressure in cases A and B, and insignificant in cases B, C and D. The effect of decay power on the total power increases with the power level increase. While in the BE scenario the use of decay power model reduces the power peak with about 3% of the rated power in extreme case C and D the decrease is more than 40% of the rated power (see Table 2 and figures 5 and 6).

Table 2 Results of BE and extreme cases A, B, C, and D: Sequence of Events

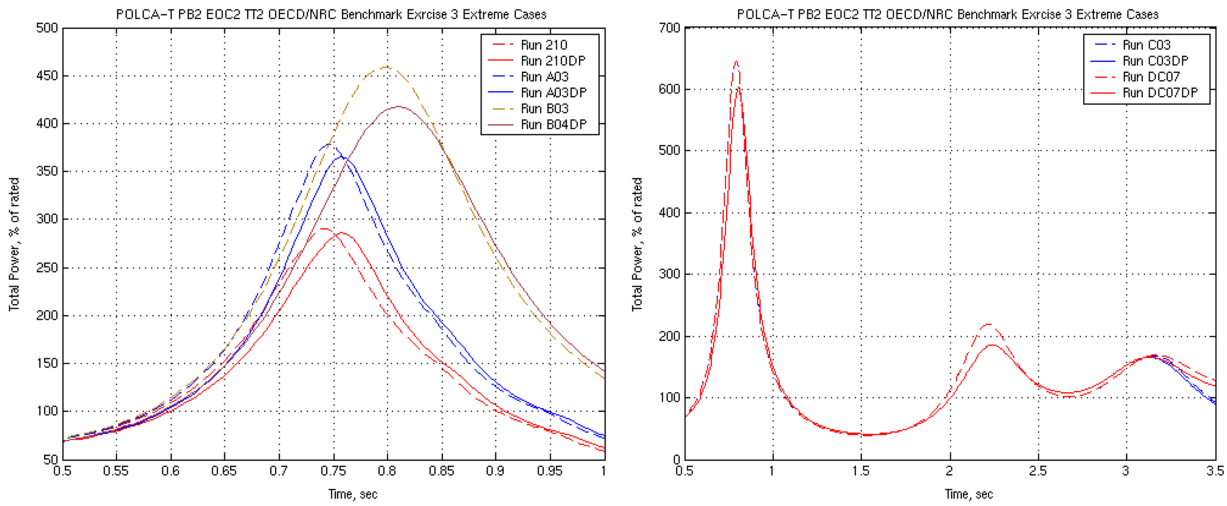
Event *Case*	*measured*	*BE*		*A*		*B*		*C*		*D*	
Run #	*[6]*	*210*	*210DP*	*A03*	*A03DP*	*B03*	*B04DP*	*C03*	*C03DP*	*DC07*	*DC07DP*
TSV begin to close [a]	0	0(54)	0(54)	0(54)	0(54)	0(54)	0(54)	0(54)	0(54)	0(54)	0(54)
TSV closed [a]	90 - 96	87	90	90	90	87	90	90	87	87	87
DRV Bypass begin opening [a]	60 - 78	66	66	-	-	66	66	-	-	-	-
DRV Bypass full open [a]	840 - 846	846	846	-	-	846	846	-	-	-	-
Time of scram initiation [a]	630	582	600	582	594	-	-	-	-	-	-
Time delay prior to rod motion [a]	120	120	120	120	120	-	-	-	-	-	-
Initiates CR insertion [a]	750	702	720	702	714	-	-	-	-	-	-
Turbine pressure initial response [a]	102 - 126	54	54	54	54	54	54	54	54	54	54
Steam line pressure initial response [a]	348 - 378	300	300	300	300	300	300	300	300	300	300
Dome pressure initial response [a]	432	342	342	342	342	342	342	342	342	342	342
Core exit pressure initial response [a]	486	444	438	444	438	438	438	438	438	438	438
Time of first Power peak [a]	726	744	756	744	756	798	810	792	804	792	804
Peak Power, % of rated	280.0	290.1	286.9	378.4	365.9	458.8	417,5	645,5	604.1	646.3	604.1
RV1 signal to open [b]	-	-	-	-	-	4.460	4.502	2.612	2.630	-	-
RV1 begin opening [b]	-	-	-	-	-	4.866	4.908	3.018	3.036	-	-
RV1 full open [b]	-	-	-	-	-	5.010	5.052	3.162	3.180	-	-
RV1 closed [b]	-	-	-	-	-	-	-	-	-	-	-
RV2 signal to open [b]	-	-	-	-	-	4.760	4.802	2.744	2.762	-	-
RV2 begin opening [b]	-	-	-	-	-	5.160	5.208	3.150	3.168	-	-
RV2 full open [b]	-	-	-	-	-	5.310	5.352	3.294	3.312	-	-
RV2 closed [b]	-	-	-	-	-	-	8.970	-	-	-	-
RV3 signal to open [b]	-	-	-	-	-	-	-	2.900	2.930	-	-
RV3 begin opening [b]	-	-	-	-	-	-	-	3.306	3.336	-	-
RV3 full open [b]	-	-	-	-	-	-	-	3.450	3.480	-	-
RV3 closed [b]	-	-	-	-	-	-	-	-	-	-	-

Note: a – in milliseconds; b - in seconds



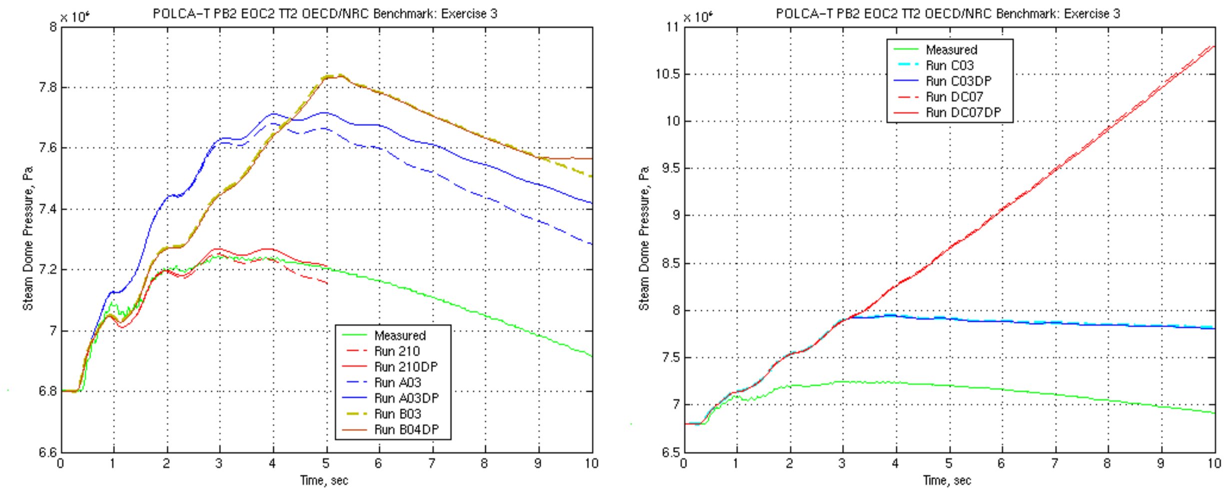
a) BE, extreme cases A and B b) extreme cases C and D

Fig.5 BE and Extreme scenarios total power.



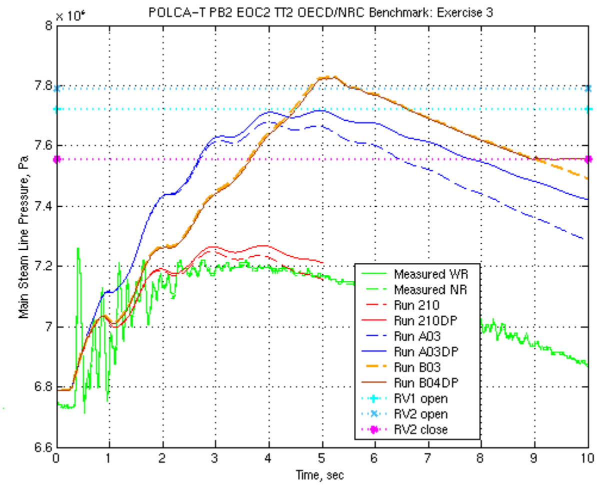
a) BE, extreme cases A and B b) extreme cases C and D

Fig.6 BE and Extreme scenarios total power zoom.

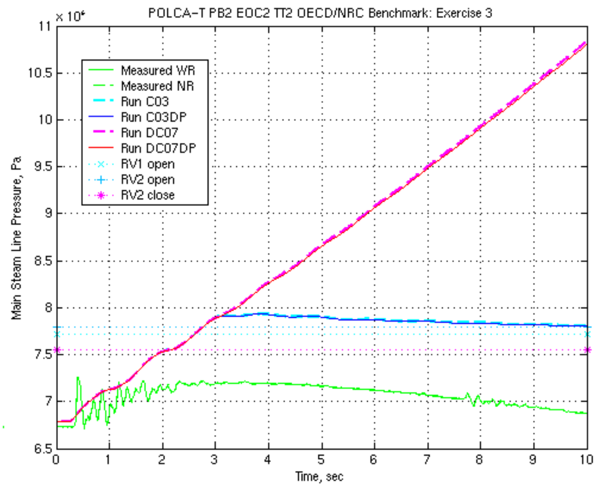


a) BE, extreme cases A and B b) extreme cases C and D

Fig.7 BE and Extreme scenarios steam dome pressure.

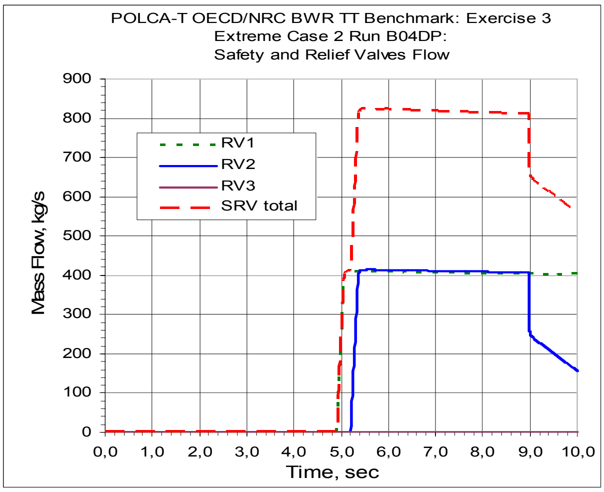


a) BE, extreme cases A and B

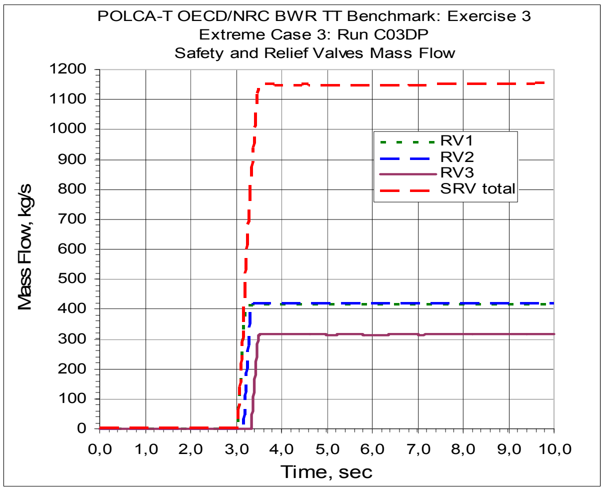


b) extreme cases C and D

Fig.8 BE and Extreme scenarios main steam line pressure.



a) extreme case B run B04DP



b) extreme case C run C03DP

Fig.9 Extreme scenarios SRV mass flow.

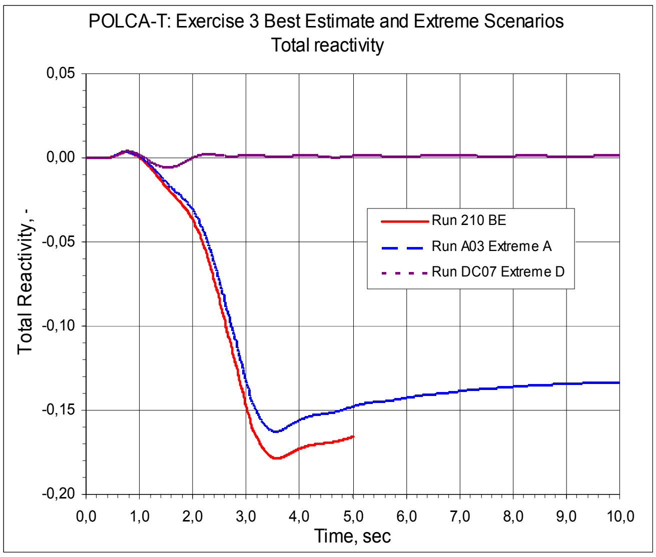
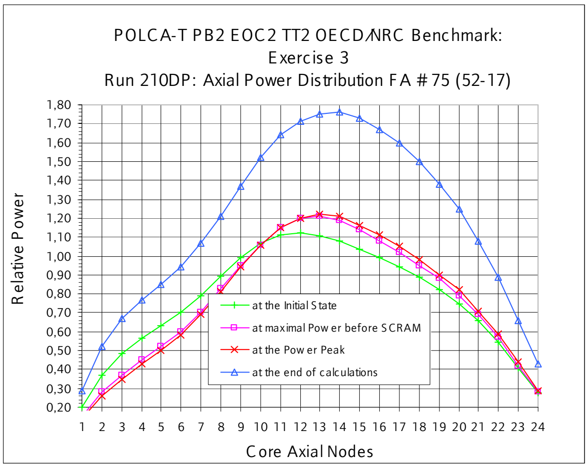


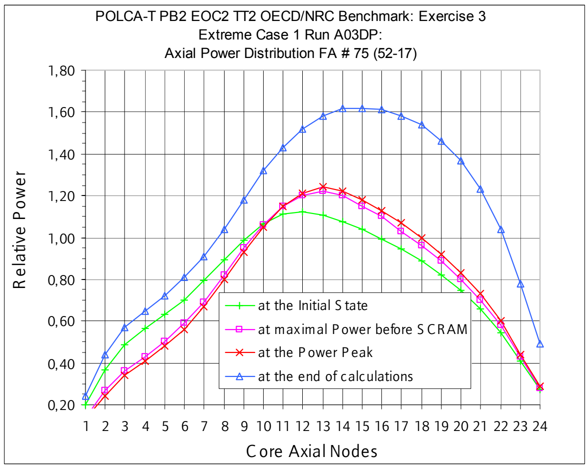
Fig.10 BE and Extreme scenarios total reactivity.

Transient core 3D power distributions are also investigated in the simulation of extreme scenarios. Figure 11 presents the redistribution of the axial power in FA number 75 (52-17) during the transient. Significant redistribution of the power is observed at the end of calculations in BE and A scenarios (at 5 and 10 seconds). In scenarios without SCRAM activation B, C and D the observed axial power profile is more stable.

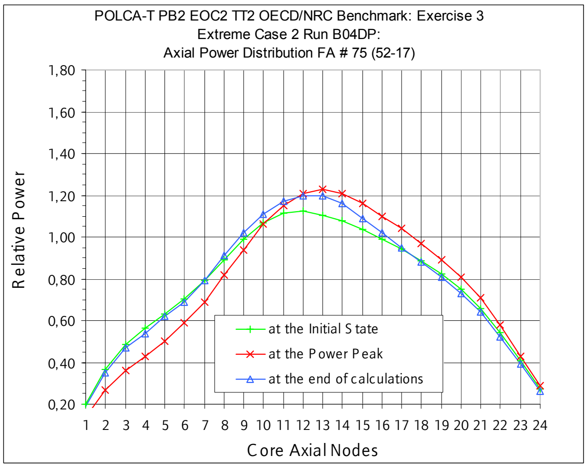
The obtained results for extreme scenarios demonstrate the POLCA-T code capability to analyze transients with pressure increase up to 10.8 MPa and power peaks higher than 600 % of rated power.



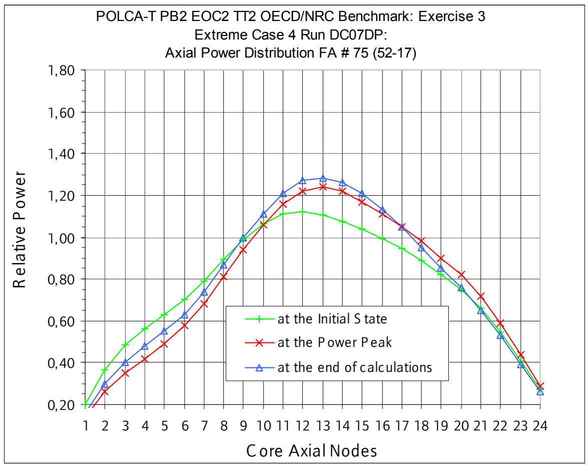
a) BE



b) extreme case A



c) extreme case B



d) extreme case D

Fig.11 BE and Extreme scenarios axial power distribution in FA # 75 (52-17).

4. Conclusion

The results of performed calculations and sensitivity studies can be summarized as follow:

1. Calculated for BE scenario time histories of main integral parameters: total power, steam dome, core exit, MSL, and turbine inlet pressures, and RPV level, are in good agreement with measured data.
2. Decay power plays important role in the predictions of power behavior, as it has strong effect on the peak power value. This effect is increasing with the increase of the peak value itself. Steam dome, core exit, and MSL pressures are affected by decay power modeling only in BE and case A scenarios. In other cases, the effect on the pressure is negligible.

3. 3D power distributions show significant redistribution of the power in BE and case A scenarios. In other three cases without SCRAM, the power distributions are more stable and similar for all cases.
4. The performed calculations and obtained results for extreme cases demonstrate the POLCA-T code wide range capabilities to simulate transients when scram, steam bypass, and SRV are not activated. The code is able to handle such transients even when the reactor power and pressure reach values higher than 600 % of rated power, and 10.8 MPa.

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