Nuclear Mathematical and Computational Sciences: A Century in Review, A Century Anew
Gatlinburg, Tennessee, April 6-11, 2003, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2003)

POLCA-T CODE VALIDATION AGAINST PEACH BOTTOM 2 END OF CYCLE 2 LOW-FLOW STABILITY TESTS

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ABSTRACT

The paper presents the results of the coupled 3D neutron kinetics / thermal-hydraulics code POLCA-T analysis of Peach Bottom 2 End of Cycle 2 Low-Flow Stability Tests PT1, PT2, PT3, and PT4. Assumptions and required corrections of the available measured data pointed out the need for a sensitivity study. The Peach Bottom 2 model developed for the OECD/NRC BWR Turbine Trip Benchmark is further modified and used in the present study. An overview of the Peach Bottom 2 system input decks and 3D core input model for End of Cycle 2 conditions is given. Code options, controllers settings and boundary conditions for stability analysis differ in some extent from ones used in turbine trip calculations. Obtained steady state results and core average axial power distributions are compared with measured data and P1 Edit data. Paper presents the results of reference calculations of stability parameters and their comparison with measured data and with previous ASEA ATOM results for all four tests. Performed sensitivity study considers the effects on stability parameters of the core inlet temperature, steam dome pressure, carry under (void fraction in bulk water), and gas gap heat transfer coefficient.

Key Words: BWR, stability, coupled codes, validation

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 following 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 unit 2 (PB2) end of cycle 2 (EOC2) turbine trip transients and low-flow stability tests [4, 5]. The participation in the OECD/NRC BWR Turbine Trip (TT) Benchmark [6] is a part of our efforts in this direction [7].

Westinghouse has extensively used PB2 EOC 2 turbine trip experiments for validation of BISON and RIGEL codes. PB2 low-flow stability tests have been also analysed by Atom Stability Analysis Model for BWR Applications (SAMBA) in 1988 [8]. This paper presents the results of the POLCA-T analysis of PB2 EOC 2 Low-Flow Stability Tests PT1, PT2, PT3 and PT4 [4, 5].

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2. PEACH BOTTOM 2 EOC 2 LOW-FLOW STABILITY TESTS: ASSUMPTIONS AND CORRECTIONS OF MEASURED DATA

Table I summarises the tests conditions, i.e. reactor power, steam dome pressure, mass flow, core inlet temperature obtained from references [4, 5, and 8]. Reference [4] reports the measured data obtained from PB2 EOC 2 low-flow stability tests PT1, PT2, PT3 and PT4. Appendix F of [5] reevaluates and corrects the results for EOC 2 after the additional low-flow stability tests performed during cycle 3. Moreover, the data reported in [4] need to be corrected and processed in order to obtain the required input data for POLCA-T calculations.

Table I. Tests conditions

Test =>	**PT1**		**PT2**		**PT3**		**PT4**	
Parameter, Units	[4, 5]	[8]	[4, 5]	[8]	[4, 5]	[8]	[4, 5]	[8]
Core thermal power, MWt	1995,00	1995	1702,00	1702	1948,00	1948	1434,00	1434
Initial power level, % of rated	60,6		51,7		59,2		43,5	
Reactor flow, kg/s	6627,60	6628	5418,00	5418	4901,40	4901	4901,40	4901
Corrected Reactor flow, kg/s	6753,60	6753	5657,40	5657	5216,40	5216	5203,80	5204
Total reactor flow, % of rated	51,3		42,0		38,0		38,0	
Reactor (Steam Dome) pressure, Pa	6798233	6878000	6743075	6812000	6832707	6909000	6791339	6866000
Corrected Steam Dome pressure, Pa	6860286	6860000	6814781	6812000	6904413	6881000	6863044	6867000
P1 Edit pressure, Pa	6860286		6812023		6880970		6867181	
Core pressure, Pa	6894760		6839602		6929234		6887865	
Corrected Core exit pressure, Pa	7063682		7008689		7098052		7056808	
Core inlet enthalpy, kJ/kg	1183,409		1174,340		1182,247		1179,456	
Corrected Core inlet enthalpy, kJ/kg	1184,610		1187,780		1184,610		1183,83	
Core inlet subcooling, kJ/kg	76,360	76,500	93,092	83,160	89,761	102,68	80,449	80,77
Corrected Core inlet subcooling, kJ/kg	75,160	75,107	79,648	79,477	77,442	76,714	76,075	75,749
Core inlet temperature, ºC	269,87		268,08		278,68		269,09	
Corrected Core inlet temperature, ºC	270,11		270,73		270,09		269,96	
Feed water mass flow, kg/s	963,90	963,90	808,92	808,90	941,22	941,20	671,58	671,20
Decay Ratio	0,259	0,170	0,303	0,206	0,331	0,386	0,271	0,263
Closed-Loop Resonant Frequency, Hz	0,449	0,382	0,449	0,347	0,430	0,365	0,391	0,324

2.1. Reactor Pressure

The core pressure given in [4] represents the pressure at the core exit. Previous Atom calculations [8] used the reactor pressure given in P1 edits as steam dome pressure. In this study the given pressure [4] was corrected with steam separators pressure drop to obtain the steam dome pressure. Table I shows that the differences in the corrected steam dome pressure are, however, negligible.

2.2. Core Flow

Reference [4] reports core flow taken from P1 edit of the plant process computer. These measurements were based on the plant instrumentation calibration during start-up to accurately indicate core flow near rated conditions. The Report [5] corrects core flow values to include an adjustment for dependence of jet pumps' loss coefficient on the ratio of suction to drive flow.

2.3 Core Inlet Temperature

The reported core inlet enthalpy (table 3-7 of [4]) was calculated from the heat balance using unadjusted mass flow. When later the mass flow was corrected [5] the core inlet enthalpy remained unchanged. Therefore, authors of [8] calculated the correct inlet subcooling from the heat balance with adjusted mass flow. The present study utilises the same approach to obtain the core inlet temperature.

Core inlet enthalpy given in [4] for PT3 test is rather high and requires feed water temperature as high as 250-260 ºC. Such high temperature of feed water at approximately the same power level, as in PT1 test seems to be unrealistic. The authors of [8] assumed a "printing error" for core inlet enthalpy of PT3 test reported in [4]. They have used the same feed water enthalpy and carry under fraction as in PT1. Using the same core inlet enthalpy as in PT1 we obtained an inlet subcooling value very close to one used in [8].

2.4 Feed Water Mass Flow and Temperature

Feed water mass flow is given in P1 edits [4]. The feed water temperature was not given in references [4, 5]. In the present calculations feed water temperature values for each test were set to get the core inlet temperature close to the corrected one.

2.5 Decay Ratio and Frequency

The measured decay ratio and closed loop frequency were revised, compare to original evaluation [4], using refined weighting procedure in the curve fit program [5].

3. POLCA-T MODEL FOR PEACH BOTTOM 2

POLCA-T system input deck for PB2 and input 3D core model for PB2 EOC2 conditions were developed for the participation in the OECD/NRC BWR Turbine Trip Benchmark [7]. POLCA-T input 3D core model, called also hereafter POLCA7 input data, consists of core physics and thermal-hydraulics data, core model options, control rods positions, fuel, and cross sections data. Systems input deck contains description of the system design, materials' geometry, materials' properties, core mapping, tests conditions, initial steady-state solution, system thermal-hydraulic data and control and balance of plant data.

3.1. Core Model (Polca7 Input Data)

The 3D core model comprises of 764 fuel channels (1 per assembly) and 122 channels for radial reflector. Each channel contains 24 axial fuel nodes. For each test reactor power and mass flow, steam dome pressure and core inlet temperature were given in POLCA7 input files. Control rods' positions correspond to the ones referred in P1 edit [4].

3.2. Systems' Model

The present study follows the established methodology for stability analysis. Main steam lines (MSL) and steam bypass lines models were removed from the system input deck for PB2 TT [7].

Boundary conditions with constant MSL pressure replaced them. The MSL pressure for each test was set to a value, that gave the corrected steam dome pressure (see Table I). Reactor pressure vessel (RPV) and recirculation loop describe the PB2 plant. The RPV model consists of 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, steam dryers and steam dome. The recirculation loop comprises of suction and discharge coolant legs, as well as main circulation pump. Separate calculations verified jet pump and recirculation loop models, together with the flow controller. Thus, we assured that the jet pump works along the operational curve.

3.3. Controllers Setings and Code Options

The simplified balance of plant consists of control rods speed and position controller, jet pump drive flow controller, feed water controller and RPV water level controller. We realised that the stability analysis had much higher requirements to the controllers and initial steady-state solution than the transients' analysis did. Both the controllers and initial steady-state solution should assure a steady state solution which has to be as close as possible to the state prior to the tests. Moreover, when the transient calculations start without any disturbances (so called zero transient calculations) the obtained transient solutions should be as close as possible to the steady state solution. In order to stabilise the zero transient calculations we introduced fine level measurements and modified the existing RPV level controller input signal and settings.

3.4. Boundary Conditions

The assumed boundary conditions include constant steam line pressure and constant feed water temperature. Water level controller governs the feed water flow in the stability calculations. The initial feed water's flow specified in the input data had to match as close as possible the expected steam flow when the settings of the level controller were modified to higher time constant.

4. STEADY STATE RESULTS AND COMPARISON WITH MEASURED DATA

4.1. Steady-State Calculations

The main parameters obtained in steady-state calculations and after zero transient calculations ware compared to make it sure that the stable conditions exist before the reactor is disturbed. Control rods' movement introduced the reactor disturbance. The list of compared parameters covers reactor power and mass flow, steam dome pressure, core inlet temperature, steam flow, RPV level, power peaking factors and core average axial power distribution. Table II presents the reactor main parameters prior to its disturbance for all four tests obtained in POLCA-T reference calculations and their comparison with available measured data. Calculated conditions before the reactor disturbance are very close to the measured prior to the tests. Only the feed water's mass flow differs from the measured values. The reason for this is that in the steady state calculations the feed water's mass flow was set to match the steam mass flow. This approach was established already in the stability methodology based on RAMONA code. This results in a little bit higher feed water mass flow in range of 2.1% for PT1 to 9.2% for PT3.

4.2. Core Average Axial Power Distributions

The core average axial power distributions obtained by POLCA-T code in the steady state calculations showed reasonable agreement with P1 edit for all four tests (see Figures 1 through 4). The same discrepancy as one seen before in TT2 calculations [7] is observed in all calculated cases. The power is underestimated in the lower part of the core and over estimated in the middle and upper parts. Differences in calculated and P1 edit axial power peaking factor and distributions observed in present results might be explained by the fact that Xenon number density were not taken into account. Moreover, the fuel burn-up and void history are given implicitly in the cross section data. However, the deviations are of the same magnitude as noticed before in core follow calculations at low power levels.

Table II. Reactor main parameters prior to its disturbance

Test =>	PT1		PT2		PT3		PT4	
Parameter, Units	Measured	POLCA-T	Measured	POLCA-T	Measured	POLCA-T	Measured	POLCA-T
Core thermal power, MWt	1995,00	1994,91	1702,00	1701,95	1948,00	1947,99	1434,00	1434,01
Steam Dome pressure, Pa	6860286	6860340	6814781	6814780	6904413	6904700	6863044	6863170
Total reactor flow, kg/s	6753,60	6749,70	5657,40	5655,10	5216,40	5216,61	5203,80	5204,74
Core inlet temperature, ºC	270,11	270,20	270,73	271,01	270,09	270,48	269,96	270,80
Core inlet enthalpy, kJ/kg	1184,610	1185,120	1187,780	1189,250	1184,610	1186,530	1183,830	1188,170
Core inlet subcooling, kJ/kg	75,160	74,649	79,648	78,182	77,442	75,522	76,075	71,735
Feed Water Mass Flow, kg/s	963,900	983,994	808,920	867,283	941,220	1028,107	671,580	699,800
Feed Water temperature, ºC		175,00		190,00		205,00		170,00
Steam Mass Flow, kg/s		983,978		867,235		1028,090		699,770
Water Level, m		14,8256		14,8141		13,5743		15,1703
K_{eff}		1,01092		1,01060		1,00672		1,00593
Power peaking factor		2,122		2,092		2,061		2,006
Axial power peaking factor	1,309	1,423	1,301	1,396	1,356	1,435	1,193	1,243

5. RESULTS OF REFERENCE CASES

Hereafter, we will call "results of reference calculations" or "reference results" the results obtained by POLCA-T in present study using most realistic in our opinion assumptions. Thus, we would like to distinguish them from results obtained in sensitivity studies and previous investigations. Table III presents the results of reference calculations and their comparison with measured data for all four tests. The decay ratio (DR) and frequency are calculated by Westinghouse software DRAMAT using method of autocorrelation functions. Table III presents along with stability parameters their deviations and errors. For all four tests, the deviations are less than 0,05 of calculated DRs, and less than 0,1 of calculated frequencies. Thus, the results show good agreement with measured data both for DR and frequency.

Table IV shows the comparison of POLCA-T reference results with previous Atom results obtained by SAMBA code [8]. The DRs obtained in present study are much closer to the measured data, while the frequencies in both studies show roughly the same deviations from ones observed in the tests.

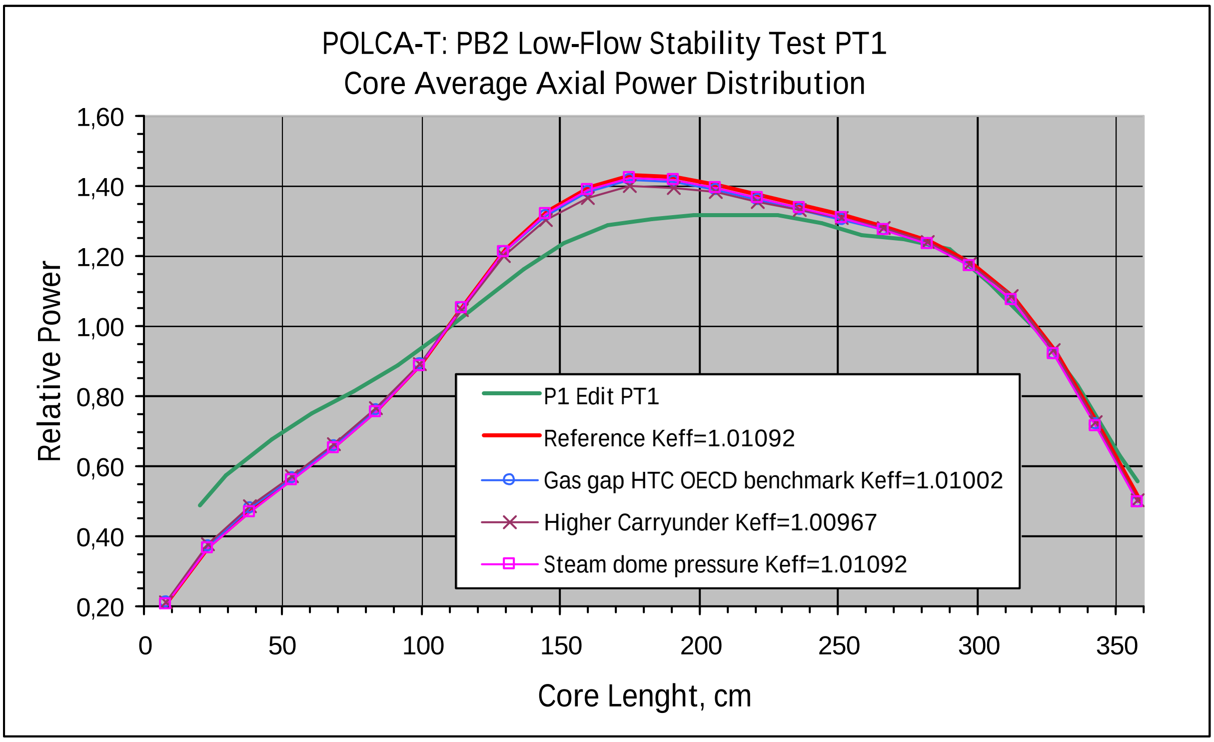

Figure 1. Comparison of POLCA-T calculated core average axial power distributions with P1 edit PT1 test.

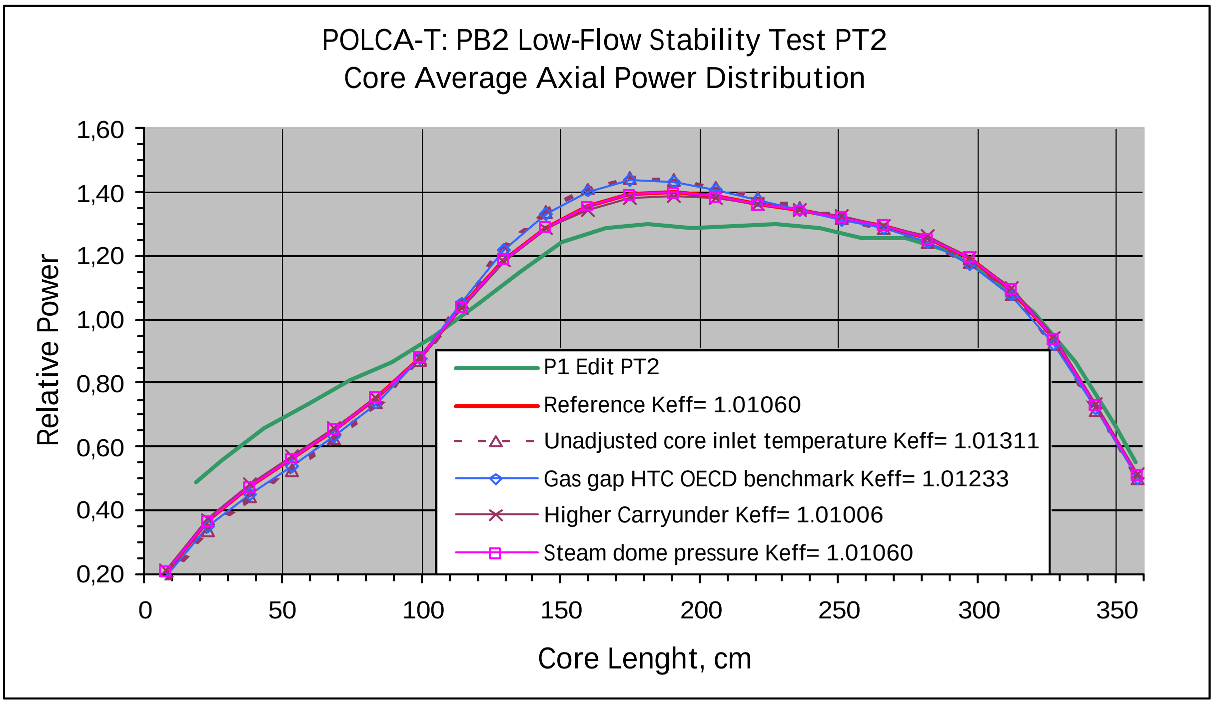

Figure 2. Comparison of POLCA-T calculated core average axial power distributions with P1 edit PT2 test.

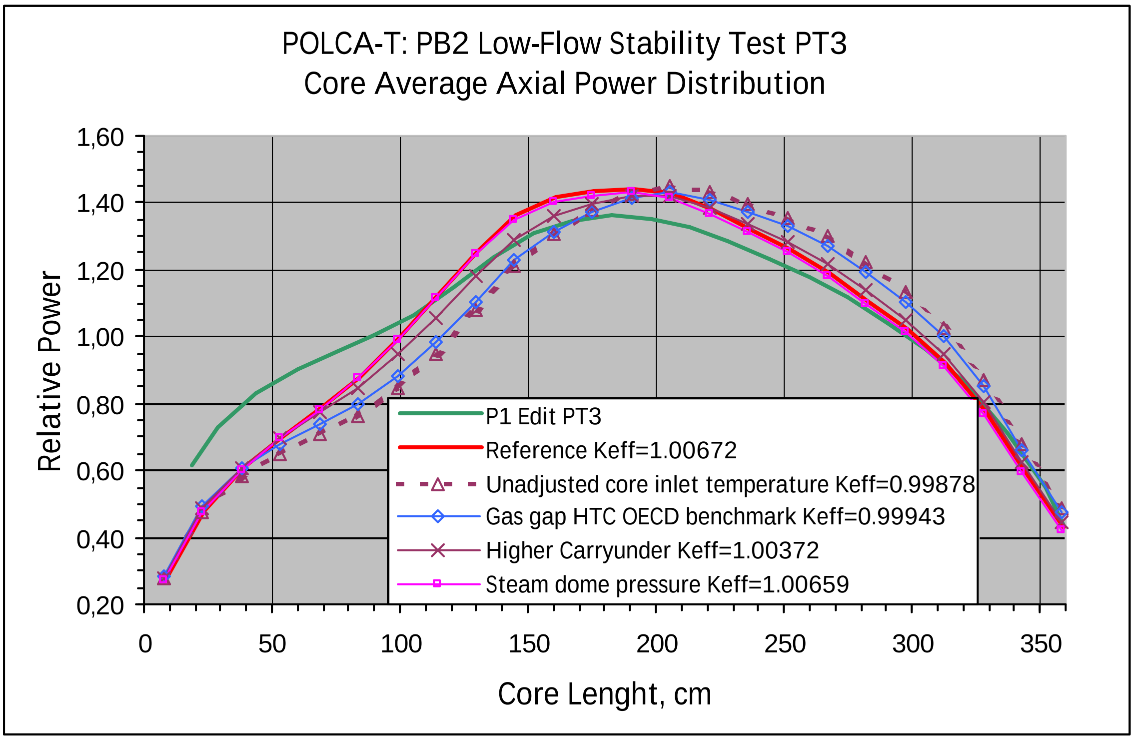

Figure 3. Comparison of POLCA-T calculated core average axial power distributions with P1 edit PT3 test.

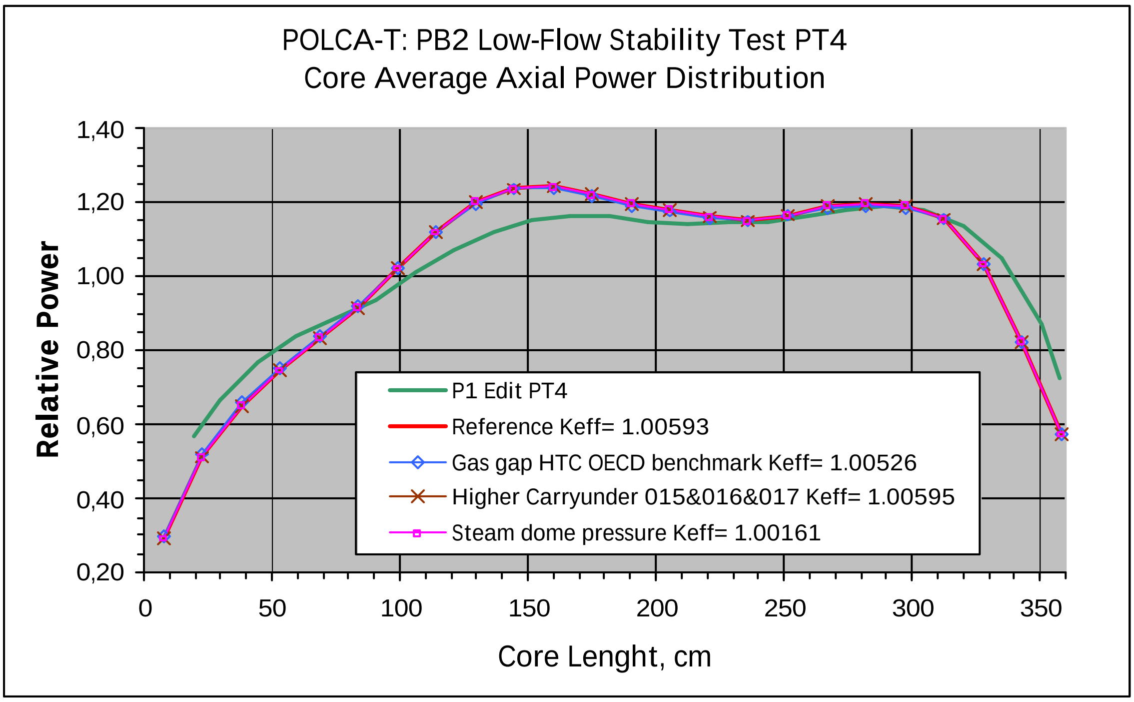

Figure 4. Comparison of POLCA-T calculated core average axial power distributions with P1 edit PT4 test.

Figures 5 and 6 illustrate the comparison of results obtained in present and previous [8] studies with measured DRs and frequencies in all four tests. Lines in figures 5 and 6 illustrate deviations from measured data equal to ± 0.05 and ± 0.10. Some of the results of performed by POLCA-T sensitivity studies, described in following sections, are also included in the figures.

Table III. Results of POLCA-T reference cases and comparison with measured data

Test	***DR***				***Frequency***			
	measured	*calculated*	*deviation*	*error, %*	*measured*	*calculated*	*deviation*	*error, %*
PT1	0,259	0,253	-0,006	-2,3	0,449	0,381	-0,068	-15,1
PT2	0,303	0,260	-0,043	-14,2	0,449	0,357	-0,092	-20,5
PT3	0,331	0,362	0,031	9,4	0,430	0,367	-0,063	-14,7
PT4	0,271	0,279	0,008	3,0	0,391	0,333	-0,058	-14,8
Average			-0,003	-1,0			-0,070	-16,3
RMS			0,027	8,7			0,071	16,5

where

deviation = calculated- measured;
error = (calculated - measured)/measured;
RMS = ((S error^2)/4)$^{1/2}$.

Table IV. Comparison of the reference results of present POLCA-T and previous SAMBA studies [8]

Test	***DR***					***Frequency***				
	Measured	*Present study: reference*		*Previous study*		*Measured*	*Present study: reference*		*Previous study*	
		calculated	*deviation*	*calculated*	*deviation*		*calculated*	*deviation*	*calculated*	*deviation*
PT1	0,259	0,253	-0,006	0,170	-0,089	0,449	0,381	-0,068	0,382	-0,067
PT2	0,303	0,260	-0,043	0,206	-0,097	0,449	0,357	-0,092	0,347	-0,102
PT3	0,331	0,362	0,031	0,386	0,055	0,430	0,367	-0,063	0,365	-0,065
PT4	0,271	0,279	0,008	0,263	-0,008	0,391	0,333	-0,058	0,324	-0,067
Average			-0,003		-0,035			-0,070		-0,075
RMS			0,027		0,071			0,071		0,077

6. RESULTS OF THE SENSITIVITY STUDY

In order to investigate the effects of assumptions and corrections of input data, discussed in section 2, we performed sensitivity studies. We found that is important to investigate the effects on the results of some other parameters. Hereafter, the effects on stability parameters of the core inlet temperature, steam dome pressure, carry under (void fraction in bulk water) and gas gap heat transfer coefficient will be considered.

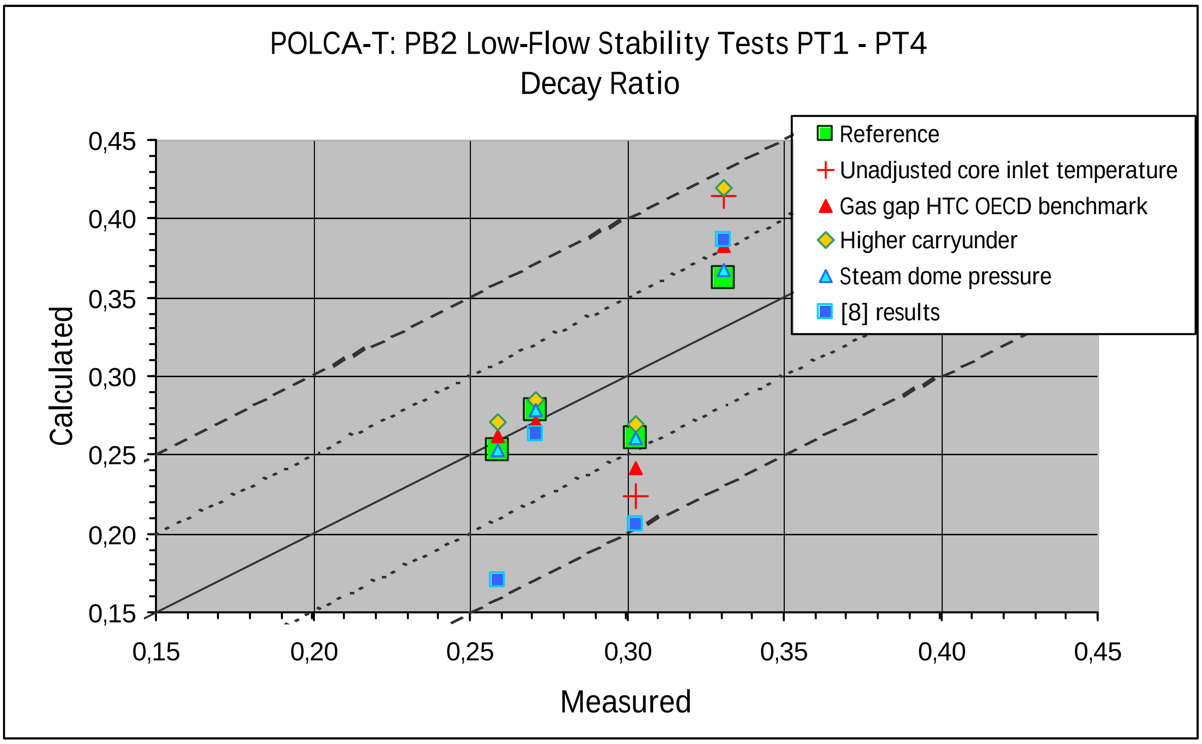

Figure 5. Comparison of POLCA-T and previous [8] results with measured in PT1 - PT4 decay ratios.

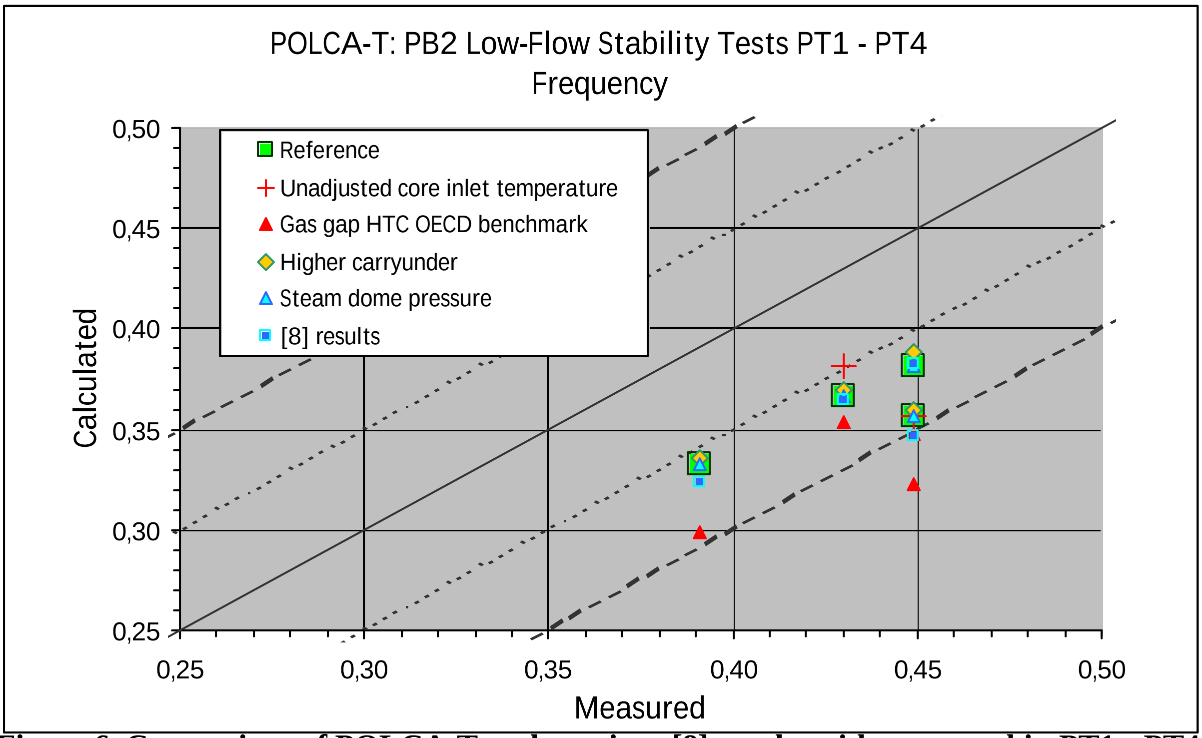

Figure 6. Comparison of POLCA-T and previous [8] results with measured in PT1 - PT4 frequencies.

6.1. Core Inlet Temperature

We investigated the effect of core inlet temperature for PT2 and PT3 tests only. Both tests have relatively high difference between reported (in [4]) and corrected (in present study) values (see Table I). In the other two tests, PT1 and PT4, the correction is less than 1ºC. We performed additional POLCA-T calculations without correction of the core inlet temperature for PT2 and PT3 tests. Table V presents the results and their comparison with POLCA-T reference cases.

Table V. Effect of core inlet temperature

Test	***DR***					***Frequency***				
	Measured	*Reference*	*uncorrected*	*deviation*	*error, %*	*Measured*	*Reference*	*uncorrected*	*deviation*	*error, %*
PT1	0,259	0,253	**0,253**	-0,006	-2,3	0,449	0,381	**0,381**	-0,068	-15,1
PT2	0,303	0,260	**0,224**	-0,079	-26,1	0,449	0,357	**0,357**	-0,092	-20,5
PT3	0,331	0,362	**0,415**	0,084	25,4	0,430	0,367	**0,381**	-0,049	-11,4
PT4	0,271	0,279	**0,279**	0,008	3,0	0,391	0,333	**0,333**	-0,058	-14,8
Average				0,002	0,0				-0,067	-15,5
RMS				0,058	18,3				0,069	15,8

Note: Deviations and errors are given for the results obtained with uncorrected inlet temperature.

Deviations of the DR obtained without correction of the core inlet temperature for PT2 and PT3 tests from the measured are more than double, compared to ones observed in the reference cases (see Tables V and III). The DR RMS of deviations and errors are also more than two times higher. The correction of core inlet temperature does not change significantly the calculated frequency. The average error and RMS of reference cases are with less than 1% higher than obtained without correction.

6.2. Steam Dome Pressure

In order to investigate the effect of steam dome pressure for all four tests we performed additional POLCA-T calculations with steam dome pressure used in previous Atom study [8] (see Table I). Table VI presents the results and their comparison with POLCA-T reference cases.

Table VI. Effect of steam dome pressure

Test	***DR***					***Frequency***				
	Measured	*Reference*	*pressure [8]*	*deviation*	*error, %*	*Measured*	*Reference*	*pressure [8]*	*deviation*	*error, %*
PT1	0,259	0,253	**0,253**	-0,006	-2,3	0,449	0,381	**0,381**	-0,068	-15,1
PT2	0,303	0,260	**0,260**	-0,043	-14,2	0,449	0,357	**0,357**	-0,092	-20,5
PT3	0,331	0,362	**0,367**	0,036	10,9	0,430	0,367	**0,367**	-0,063	-14,7
PT4	0,271	0,279	**0,279**	0,008	3,0	0,391	0,333	**0,333**	-0,058	-14,8
Average				-0,001	-0,7				-0,070	-16,3
RMS				0,028	9,1				0,071	16,5

Note: Deviations and errors are given for the results obtained with pressure used in [8].

DRs calculated with and without correction of steam dome pressure are identical in three of the tests (PT1, PT2 and PT4). The difference obtained for PT3 test is negligible (0,005). The frequencies are identical for all four tests.

6.3. Carry Under

We investigated the effect of carry under (void fraction in bulk water) for all four tests. The POLCA-T steam separator model gives low carry under in the reference calculations (less than 0,002 for all tests). We modified steam separator model in order to obtain carry under higher than observed in reference cases. In performed additional POLCA-T calculations carry under lies in range of 0,004 to 0,006 for all tests. Table VII presents the results and their comparison with POLCA-T reference cases.

Table VII. Effect of carry under

Test	***DR***					***Frequency***				
	Measured	*Reference*	*higher carry under*	*deviation*	*error, %*	*Measured*	*Reference*	*higher carry under*	*deviation*	*error, %*
PT1	0,259	0,253	**0,271**	0,012	4,6	0,449	0,381	**0,388**	-0,061	-13,6
PT2	0,303	0,260	**0,269**	-0,034	-11,2	0,449	0,357	**0,360**	-0,089	-19,8
PT3	0,331	0,362	**0,420**	0,089	26,9	0,430	0,367	**0,370**	-0,060	-14,0
PT4	0,271	0,279	**0,285**	0,014	5,2	0,391	0,333	**0,336**	-0,055	-14,1
Average				0,020	6,4				-0,066	-15,4
RMS				0,049	15,0				0,068	15,6

Note: Deviations and errors are given for the results obtained with higher carry under.

Calculated DRs for runs with higher carry under are higher than obtained in the reference cases. The DR has also higher average deviation and RMS than ones obtained in the reference cases (see Tables VII and III). The differences in carry under do not influence significantly the calculated frequency. The error's RMS of reference cases is 1% higher with less than obtained with higher carry under.

6.4. Gas gap heat transfer coefficient

We investigated the effect of gas gap's heat transfer coefficient (HTC) for all four tests. POLCA-T reference calculations used a constant gas gap's HTC of 6568 W/(m^2K) from previous Atom study [8]. We performed additional POLCA-T calculations with constant gas gap HTC equal to 4542,56 W/(m^2K), specified for OECD BWR TT benchmark [6]. Table VIII presents the results and their comparison with POLCA-T reference cases.

DRs and frequencies with different constant gas gap's HTC are close to each other. Specified for OECD BWR TT benchmark HTC gives slightly higher deviations and errors compared to the value used in reference cases and [8]. The effect of gas gap HTC is stronger on calculated frequencies. The temperature and fuel burn-up dependent gas gap's HTC should give better estimation of the tests frequencies. Such a fuel model based on Westinghouse fuel thermal mechanics code STAV-7 is already implemented in POLCA-T.

Table VIII. Effect of gas gap's heat transfer coefficient

Test	***DR***					***Frequency***				
	Measured	*Reference*	*BWR TT gap*	*deviation*	*error, %*	*Measured*	*Reference*	*BWR TT gap*	*deviation*	*error, %*
PT1	0,259	0,253	**0,262**	0,003	1,2	0,449	0,381	**0,348**	-0,101	-22,5
PT2	0,303	0,260	**0,241**	-0,062	-20,5	0,449	0,357	**0,323**	-0,126	-28,1
PT3	0,331	0,362	**0,382**	0,051	15,4	0,430	0,367	**0,354**	-0,076	-17,7
PT4	0,271	0,279	**0,271**	0,000	0,0	0,391	0,333	**0,299**	-0,092	-23,5
Average				-0,002	-1,0				-0,099	-22,9
RMS				0,040	12,8				0,100	23,2

Note: Deviations and errors are given for the results obtained with BWR TT gas gap's HTC.

7. CONCLUSIONS

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

1. Decay ratios calculated by POLCA-T are in good agreement with measured data. This proves the POLCA-T capability to perform stability analysis at low-flow conditions.
2. The calculated frequency has reasonable values, however there is a systematic underestimation in POLCA-T calculations.
3. Obtained core average axial power distributions are in relatively good agreement with P1 edit data. The comparison gives, however the same discrepancy as the one observed in TT2 calculations.
4. Performed sensitivity studies pointed out that core inlet temperature, carry under (void fraction in bulk water) and gas gap's HTC effect the stability parameters. Among them, the carry under and gas gap's HTC have stronger effect on the results. Some further efforts to investigate the temperature and burn-up dependent gas gap's HTC on frequency would be required, to improve current modelling.

ACKNOWLEDGMENTS

The authors would like to express their gratitude to their colleagues: to Ulf Bredolt for his efforts in POLCA-T thermal-hydraulics development and readiness to discuss the results and to criticize the work; to Juan Casal, Per Jerfsten, and Jari Toivanen, for their support, helpful proposals and ideas.

REFERENCES

1. H. Wijkström, "ABB Atom's New Code for 3D Static and Transient Analysis", *Proceeding of the German Nuclear Society Workshop on "Thermal and Fluid Dynamics, Reactor Physics and Computing Methods"*, Rossendorf, Germany, January 31 - February 1, (2000).
2. S-Ö. Lindahl, E. Müller, "Status of ABB Atom's Core Simulator POLCA", *PHYSOR96*, Mito, Japan, September 16-20, (1996).
3. E. Müller, P. Forslund, Y. Parmar, "Validation of the ABB Atom Core Physics Methods", *ANS Topical Meeting on Advances in Nuclear Fuel Management*, Myrtle Beach, South Caroline, USA, March 23-26, (1997).
4. L. A. Carmichael and R. O. Niemi, Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2. EPRI NP-564, Topical Report, June (1978).
5. Low-Flow Stability Tests at Peach Bottom Atomic Power Station Unit 2 During Cycle 3. EPRI NP-972, Final Report, April (1981).
6. J. Solis, et al., Boiling Water Reactor Turbine Trip (TT) Benchmark Volume I: Final Specifications. NEA/NSC/DOC(2001) 1, June (2001).
7. D. Panayotov, OECD/NRC BWR Turbine Trip Benchmark: Simulation by POLCA-T Code, Invited paper in *PHYSOR2002 International Conference on the New Frontiers of Nuclear Technology: Reactor Physics, Safety and High-Performance Computing, Track H-2, Special Session: "Numerical and Computational Issues of Coupled 3-D Kinetics/Thermal-Hydraulic System Code Simulations: OECD/NRC BWR TT Benchmark"*, Paper 3C-02, Seoul, Korea, October 7 - 10, (2002).
8. Stability Analysis Model for BWR Applications (SAMBA). Volume 4: Code Qualification Analyses. EPRI NP-5620-CCML, March (1988).
9. N. H. Larsen, Core Design and Operating Data for Cycles 1 and 2 of Peach Bottom 2. EPRI NP-563, Topical Report, June (1978).