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FUEL ASSEMBLY OUTLET TEMPERATURE PROFILE INFLUENC ON CORE BY-PASS FLOW AND POWER DISTRIBUTION DETERMINATION IN VVER - 440 REACTORS

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ABSTRACT

The incore instrumentation of the VVER–440 reactors consists of the thermocouple sys (TC) and the system of self powered detectors (SPD). The TCs are positioned about 50 above the fuel bundle upper flow-mixing grid. The usual assumption is that, the coola well mixed in the TC location, i.e. the temperature is constant through the flow cross-sec area. The present evaluations by using the FLUENT 5.5.14 code reveal that, this assump is not fulfilled. There exists a temperature profile that depends on fuel assembly geometry on inner power profile of the fuel assembly. The paper presents the estimation of this e and its influence on the core power distribution and the core by-pass flow determinat Comparison with measurements in Mochovce NPP will also be a part of this presentation.

1. INTRODUCTION

In the past twenty years of nuclear power utilization in Slovak Republic several cha in fuel assembly design have been introduced to improve the fuel performance. The cha concerned the fixed fuel assemblies mainly.

The first change with serious influence on power distribution determination in rea core based on incore thermocouple measurements was the change of the top grid and addi of the upper mixing grid. This change was introduced in about 1985 and led to n interpretation of the incore measurements (reactor operation on reduced power). The prob was published in former VMK Symposium in 1986 [1]. The root cause of discrepancy was change of the top grid hydraulic design, explained by P. Siltanen [2] in the case of m cores.

The second design change, reduction of the shroud wall thickness from 2,1 mn 1,5 mm (also decreasing the outer shroud diameter from 144 mm to 143 mm) for the f fuel assemblies, was introduced in about 1990. This modification was recognized by the slight change of the cycle lengths.

The next design change was the replacement of Fe spacer grids by Zr ones, started from 12th cycle of Bohunice NPP Unit 3 in 1995. This design change was handled correctly from the neutronics and thermal-hydraulics point of view by implementing of special core surveillance programme.

Starting from the 2nd cycle of Mochovce NPP Unit 1 in 1999 the profiled 3,82 % enriched fuel assemblies are implemented. The fixed fuel assemblies, in addition to enrichment profile, differ also in shroud diameter from previous ones. The shroud outer diameter was increased from 143 mm to 145 mm, keeping the wall thickness 1,5 mm. Even in this case the changes in neutronics were correctly handled. But concerning the thermocouple readings some problems had appeared. They will be explained in the next part of this paper.

In the near future there is an intention to introduce the new fuel assemblies with optimized fuel rod lattice pitch changed from present 12,2 mm to 12,3 mm and utilization of burnable absorber rods within the fuel assemblies. Not accounting these changes, even in interpretation of the incore monitoring system, can lead to an incorrect interpretation of the reactor core power distribution.

The paper also contains the short description of the code package that was used for evaluation of the fuel design changes influence on the core monitoring system.

At present there is a shortage of detailed design drawings of the fuel assemblies and part of the thermal-hydraulic models are not yet validated because of lack of experimental data, so the results presented in the paper are only informative. There is also an intention to draw the attention of the AER participants to devote more interest for solving the problem mentioned above.

2. CORE DESIGN OF MOCHOVCE NPP UNITS

The core composition of Mochovce NPP Unit 1 and 2 for the initial cores, as the fuel assembly enrichment concerned, is identically. The only difference between the core loadings in Unit 2 in comparison with Unit 1 is that all fuel assemblies are furnished with Zr grids. The core composition of the first cores of both units is shown in Fig. 1 and Table 1.

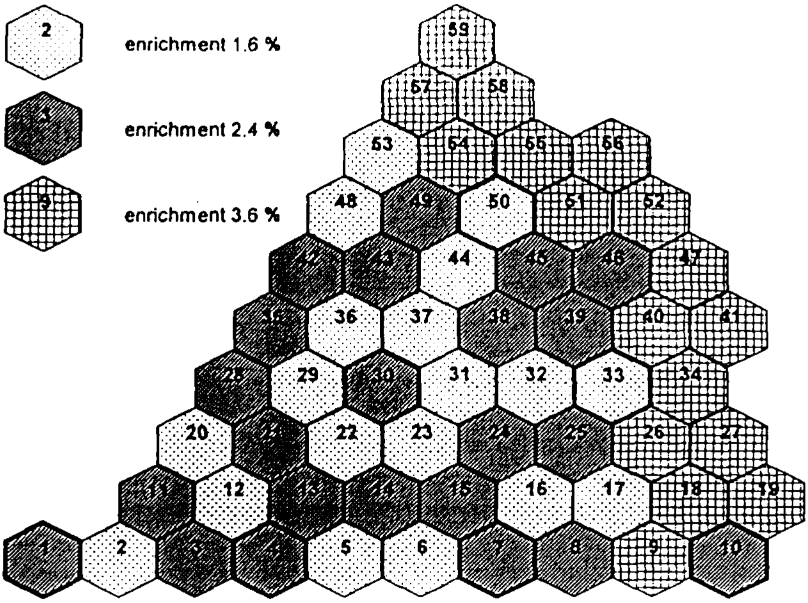


Fig. 1 The core composition of Mochovce NPP Unit 1 and 2 for the initial cores

Enrichment	Grid	Wall thickness [mm]	Unit 1		Unit 2	
			Fuel assemblies	Control assemblies	Fuel assemblies	Contr assemb
1,6 %	Fe	1,5	4	-	-	-
		2,1	98	12	-	-
	Zr	1,5	-	-	102	12
2,4 %	Fe	1,5	34	-	-	-
		2,1	74	25	-	-
	Zr	1,5	-	-	108	25
3,6 %	Fe	1,5	102	-	-	-
		2,1	-	-	-	-
	Zr	1,5	-	-	102	-

Table 1 First loadings of both Mochovce NPP units active core

Starting with second cycle of both units low-leakage fuel management strategy implemented. In the same time the first profiled 3,82 % enriched fuel assemblies v introduced into loadings. In all cycles, except of 4th cycle, the number of profiled assemblies was the same in active core of both units. Difference between hydra characteristics of both units depends on number of non-profiled fuel assemblies with Fe ;

and wall thickness 2,1 mm. The 5[th] cycle is the first loading of Unit 1 without fuel assemblies with wall thickness 2,1 mm. The composition of the present loadings is shown in Table 2.

Enrichment	Grid	Wall thickness [mm]	Unit 1, Cycle 5		Unit 2, Cycle 4	
			Fuel assemblies	Control assemblies	Fuel assemblies	Control assemblies
1,6 %	Fe	1,5	-	1	-	-
	Zr	1,5	-	-	-	-
2,4 %	Fe	1,5	3	-	-	-
	Zr	1,5	2	-	1	7
3,6 %	Fe	1,5	21	-	-	-
	Zr	1,5	4	-	101	-
3,82 %	Zr	1,5	282	36	210	30

Table 2 Present loadings of both Mochovce NPP units active core

3. COMPARISON OF THE RESULTS OF THE CORE FLOW BY-PASS MEASUREMENTS BASED ON THE HYDRAULIC AND THERMAL-HYDRAULIC METHODS

At the beginning of each cycle the core by-pass flow (i.e. flow rate outside of the fuel assemblies) is measured. The core by-pass flow is determined by two methods:

- "Hydraulic by-pass flow" - as a part of hydraulic measurements in the course of zero power reload start-up tests, from the core pressure gradient of flow area and coefficients of hydraulic resistance [3],
- "Thermo-hydraulic by-pass flow" - based on enthalpy ratio measured on the core and on the reactor, it is expressed by:

$$\Theta_{TH} = \left(1 - \frac{\Delta i_R}{\Delta i_{AZ}}\right).100 \quad [\%] \qquad (1)$$

where is:

Θ_{TH} - the by-pass flow ratio
Δi_R - the measured enthalpy rise on the reactor
Δi_{AZ} - the measured enthalpy rise on the core

The enthalpy rise on the reactor is determined from the loop temperatures and pressure. The Δi_{AZ} is determined from the reactor inlet and outlet temperatures and pressure [3].

In the Table 3, there are presented the results of by-pass measurements by both mentioned methods on Mochovce NPP during start-up tests [4], [5]. There are present the by-pass values from the safety analysis report for Mochovce NPP [6]. The agreen values measured by hydraulic methods with values from safety analysis report is satisfactory. The values measured by enthalpy balance differ from hydraulic ones, and t not seen the decreasing trend by the increasing number of profiled fuel assemblies as the case of hydraulic measurements. At the beginning of 5-th cycle of Unit 1 almost the core is fuelled with profiled fuel, (there is only 30 non-profiled fuel assemblies in active The by-pass flow is expected 5,9 %, but the value of the Θ_{TH} is nearly the same as beginning of the first loading.

This effect also influences the relation between the relative power distribution k_{qi} and ΔT_i values (used in some reactors for fuel assemblies power limitation).

Note:

The values of thermo-hydraulic by-pass flow are presented in Table 3 in depende fuel loading. As the values of thermo-hydraulic by-pass flow are derived from thermoc readings they depend on actual work conditions of thermocouple system too, mai temperature of thermocouples cool end.

Unit	Cycle	Thermo-hydraulic by-pass flow [%]	Hydraulic by-pass flow [%]	Core by-pass flow [6] [%]
1	1	10,4	9,4	8,3
	2	10,7	9,2	8,0
	3	11,0	9,0	7,5
	4	11,0	6,5	6,9
	5	10,1	5,4	5,9
	stationary cycle	-	-	5,1
2	1	9,3	8,7	9,3
	2	8,9	8,4	8,7
	3	9,5	7,5	8,0
	4	9,9	6,9	7,0
	stationary cycle	-	-	5,1

Table 3 The values of by-pass flow measured by both methods at the beginning of cycle of Mochovce NPP Unit 1 & 2, comparison with values in safety ana report [6]

4. FUEL ASSEMBLY MODELING AND RESULTS OF THE CALCULATIONS

For evaluation of the fuel assembly outlet temperature distribution BIPR-7 [7], PERMAK [8], CALOPEA [9] and FLUENT 5.5.14 [10] codes were used.

4.1 The fuel assembly model

The fuel assemblies of the VVER-440 type reactors consist of 126 fuel pins in hexagonal symmetry. The active length of the fuel pins is 242 cm, the outer diameter is 9,1 mm. The fuel assembly consists of 10 spacer grids and central tube with outer diameter 10,3 mm for incore neutron detectors. For the flow, temperature and velocity distribution modeling in the fuel assembly up to the upper end of the fuel column the CALOPEA 3D code was used. The fuel assembly in x-y plane is represented by 259 subchannels (Fig. 2). In axial direction 31 mesh points are used.

The upper part of the fuel assembly including the fixation of the fuel assembly to the unit of the shielding tubes was modeled by FLUENT code. The simplified geometry of the fuel assembly with the thermocouple is on Fig 3. More detailed animated view of this detail is on Fig. 4. The proper meshing of the complicated geometry of the fuel assembly head including the upper fixation was the greatest problem. Due to the complicated geometry the combined structured and unstructured meshing was used. The upper part of the fuel assembly approximately by 500 000 control volumes is represented. Two types of basic meshes were created for fuel assemblies with inner diameter 142 mm (profiled) and 140 mm (non-profiled).

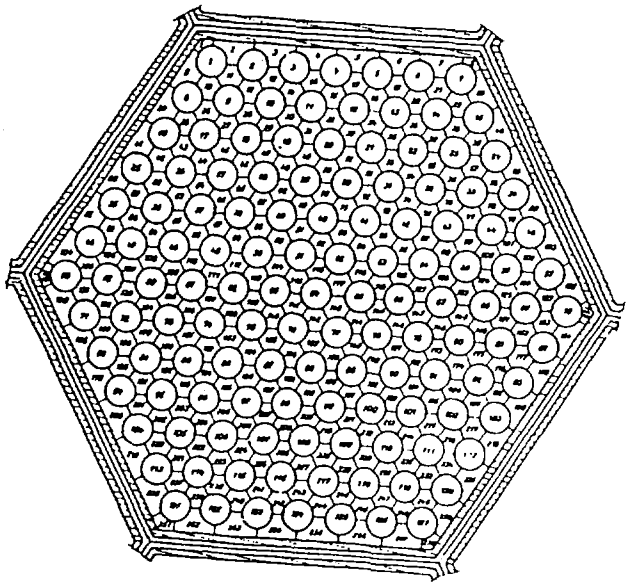


Fig. 2 Thermal-hydraulic subchannels of fuel assembly

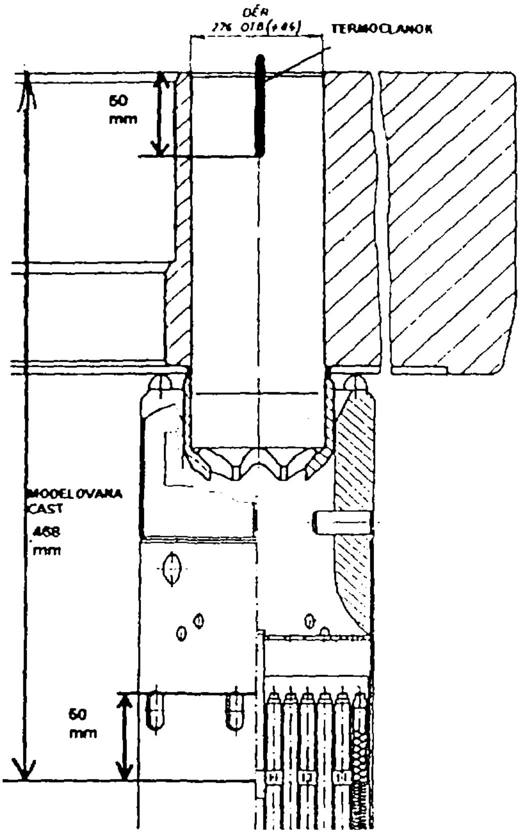


Fig. 3 The area considered in calculations

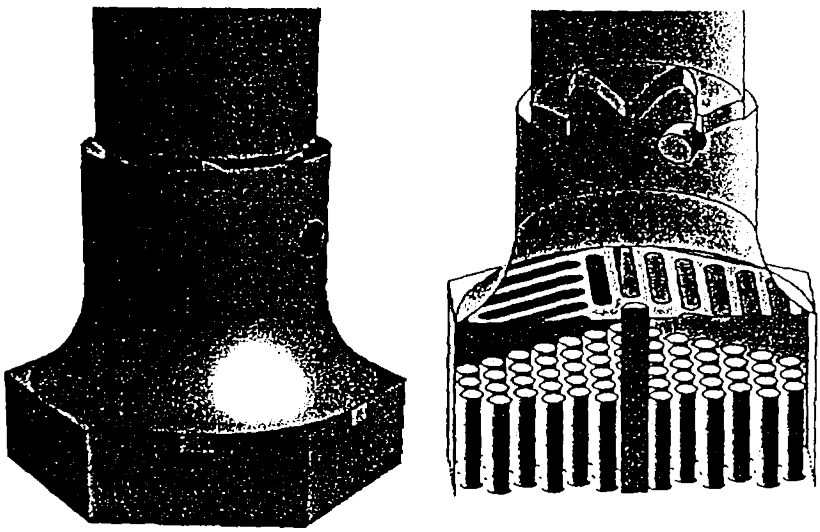


Fig. 4 The detailed view on the upper part of fuel assembly

4.2 Initial and boundary conditions for thermal-hydraulic analyses

The goal of these calculations was the determination of coolant temperature profile in the position of the thermocouple. For studying of this phenomenon the following power profiles, flow conditions and assembly inner diameters were selected:

Variants A, B and C

These cases represent the fuel assembly with inner diameter 142 mm with different power profiles. The power profile in case A represents an idealistic power distribution of non-profiled fuel with idealistic power distribution. By the same condition is calculated the power distribution for the 3,82 % enriched fuel assembly - case B. The power distribution for the case C was selected in such a way that it represents the power distribution for the assemblies from the core periphery. The assembly power for the cases A and B was selected as a maximal allowed power i.e. 5,32 MW, assembly power in case C is 1,43 MW (approximately as the power of peripheral fuel assemblies). The idealistic power distribution for cases A and B was calculated by BIPR-7 and PERMAK codes in a way that the interested fresh fuel assembly is placed into the core central part in the 3rd cycle of Mochovce NPP Unit 2. The main characteristics needed for calculations of these variants are in Tables 4 and 5.

Variants K09, K11, K24 and K26

These variants have realistic power distributions and power corresponding to the core positions – groups of symmetry No. 09, 11, 24 and 26 on the beginning of the 3rd cycle of Mochovce NPP Unit 2. The main characteristics of these variants are in Table 6.

Variants A1 and B1

These variants have the same power conditions as variants A and B, but they differ in fuel assembly inner diameter that is 140 mm (like the non-profiled fuel) and in flow rate that is 108 m^3/h. This flow rate corresponds to the flow rates of non-profiled and profiled fuel in Mochovce NPP Unit 2. The main characteristics of these variants are in Tables 4 and 5.

Variant B1o

This variant corresponds with variant B1. The only difference is that there is modeled also the perforation in fuel assembly shroud. The perforation is modeled by 6 holes (9,5 x 13,4 mm) instead of 12 holes with Φ 9 mm. The holes are situated just bellow the upper mixing grid. The by-pass flow of 1,1 kg/sec. is accounted which corresponds to 5 % of flow rate of the fuel assembly inlet flow. The temperature of by-pass flow was calculated by CALOPEA code, it was 276,9 °C. The gross by-pass flow in the reactor is approximately 7÷8 % for cores with non-profiled fuel, from which 5 ÷ 6 % returns by perforations and about 2 % flows out (guide tubes of the control assemblies, core barrel, etc.).

CALOPEA Code	All variants (except of A1, B1 a B1o)	Varia ar
Absolute inlet pressure [MPa]	12,26	
Inlet temperature [°C]	266,74	2
Flow rate in fuel assembly [m^3/h] / [kg/sec]	110,0 / 23,91	108,
Fuel assembly diameter (shroud diameter x wall thickness) [mm]	145 x 1,5	14
FLUENT Code		
Inlet hydraulics diameter [mm]	10,0	
Outlet hydraulics diameter [mm]	90,0	
Turbulence intensity on the inlet and outlet of FA [%]	10,0	
Coolant temperature in perforation [°C] **(variant B1o only)**	-	:
Integral flow rate through perforation [m^3/h] / [kg/sec] **(variant B1o only)**	-	5,0
Turbulence model	standard k-ε	star
Numerical scheme of calculation	segregate	se
Linearization	implied	in

Table 4 Initial and boundary conditions

	Var. A	Var. B	Var. C	Var. A1	Var. B1
Fuel	non-profiled	profiled	non-profiled	non-profiled	profiled
Average enrichment U^{235} [%]	3,60	3,82	2,40	3,60	3,82
FA position in active core	centre GS No.1**	centre GS No.1	periphery GS No.27	centre GS No.1	centre GS No.
Thermal flux from FA [$kW.m^{-2}$]	590,92	590,92	158,26	590,92	590,92
FA thermal power [MW]	5,321	5,321	1,425	5,321	5,321
Average coolant temperature in FA [°C] *	307,8	307,8	278,5	308,5	308,5

* - average coolant temperature in outlet from fuel part of FA

** - GS = group of symmetry

Table 5 Variants specification

	Var. K09	Var. K11	Var. K24	Var. K26
Fuel	profiled	profiled	profiled	profiled
Average enrichment U^{235} [%]	3,82	3,82	3,82	3,82
FA position in active core	GS No.9**	GS No.11	GS No.24	GS No.26
Thermal flux from FA [$kW.m^{-2}$]	577,77	568,31	571,85	516,37
FA thermal power [MW]	5,205	5,111	5,144	4,670
Average coolant temperature in FA [°C] *	307,06	306,47	306,68	303,13

* - average coolant temperature in outlet from fuel part of FA

** - GS = group of symmetry

Table 6 Variants specification

4.3 Results of calculations

Study of the influence of by-pass flow on the measured outlet temperature

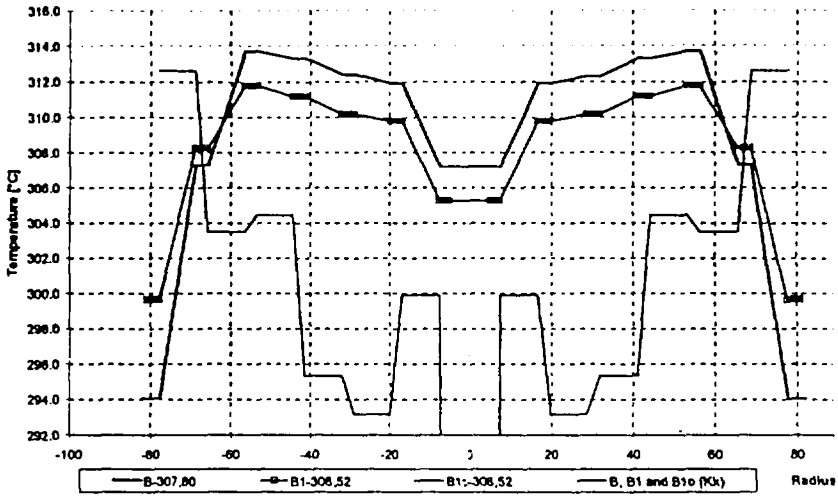
This effect was evaluated on fuel assemblies of inner diameter 140 mm with flat power distribution (profiled fuel). The results of the calculation are in Table 7 (variant B1 and B1o) and on Fig. 5, 6 and 7. From the Table 7 and Fig. 5 it can be seen that the influence of by-pass flow on measured temperature is small (approx. 0,13 °C). For the fuel assemblies of 142 mm inner diameter it is even less due to decreased by-pass flow (about 3 % is returning through the perforation). Due to this fact for the study of power profile influence and fuel assembly shroud diameter change the by-pass flow was neglected.

	Variant A	Variant B	Variant C	Variant A1	Variant B1	Variant B1o
Temperature in position of temperature measurement (thermocouple) [°C]	310,87	311,65	278,78	308,90	309,65	309,50
Average coolant temperature [°C] *	307,80	307,80	278,50	308,50	308,50	308,50 307,15**
Temperature difference [°C]	3,07	3,85	0,28	0,40	1,13	1,00/2,35
(δT_i / ΔT_i) [%]	7,5	9,4	2,4	1,0	2,8	2,4

* - average coolant temperature in outlet from fuel part of FA

** - average coolant temperature in the thermocouple position

Table 7 Coolant temperatures in the upper part of fuel assembly



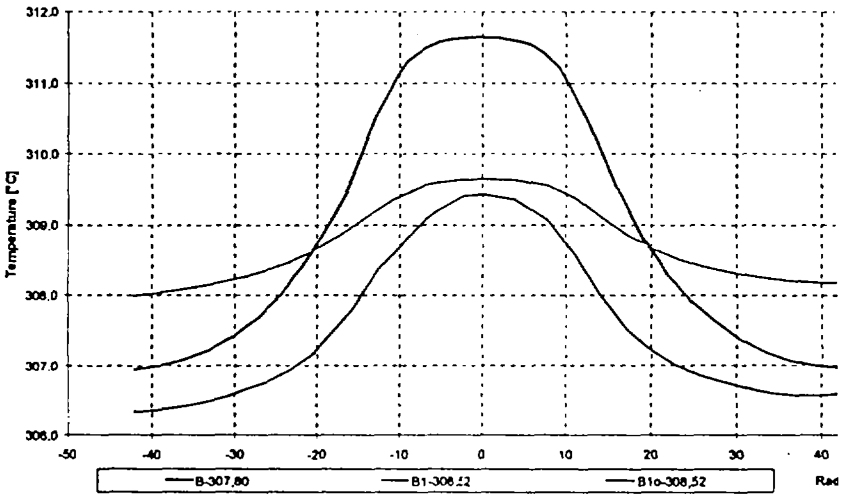


Fig. 5 Distribution of power and temperature profile at the end of fuel column and in thermocouple position – variants B, B1 and B1o

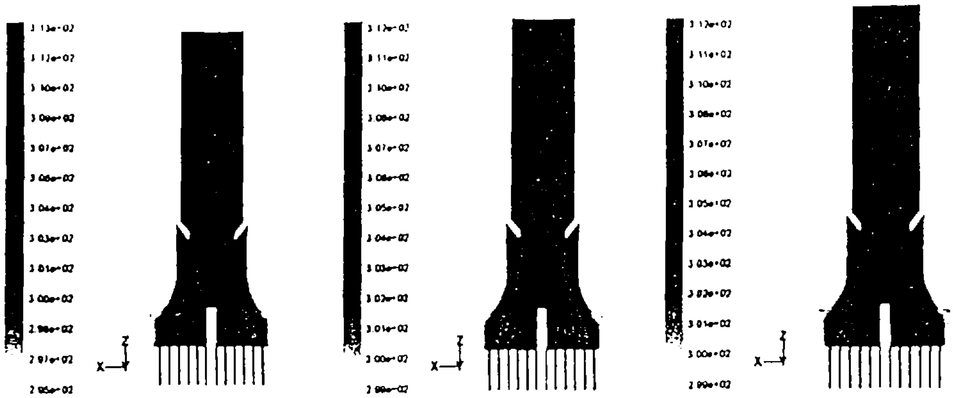


Fig. 6 Coolant temperature in vertical section of fuel assembly – variants B, B1 and B1o

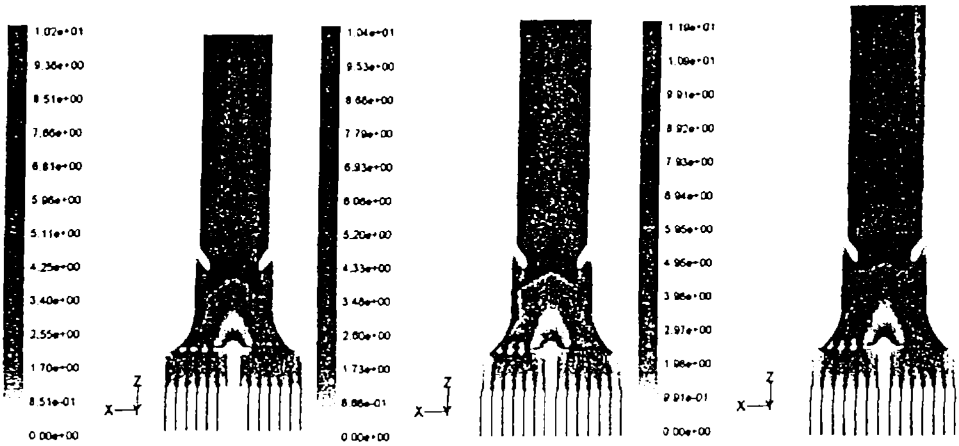


Fig. 7 Coolant speed in vertical section of fuel assembly – variants B, B1 and B1o

Power profile influence on measured outlet temperature

This effect was evaluated on the fuel assemblies with inner diameter 142 mm. The results of the calculation for variants A, B and C are in Table 7.

From Table 7 it can be seen the influence of power profile on measured temperature. For further understanding of the power profile influence there were calculated also the outlet temperatures on measured positions with realistic power distributions - Table 8 and Fig. 8. From the Table 7 and 8 it can be seen that in the core central part the ratio of

$$\frac{\delta T_i}{\Delta T_i} = \frac{T_i^{meas} - T_i}{T_i - T_{in}} \qquad (2)$$

where:

T_i^{meas} – temperature in the position of thermocouple

$\bar{T}_i$ – the mean outlet temperature

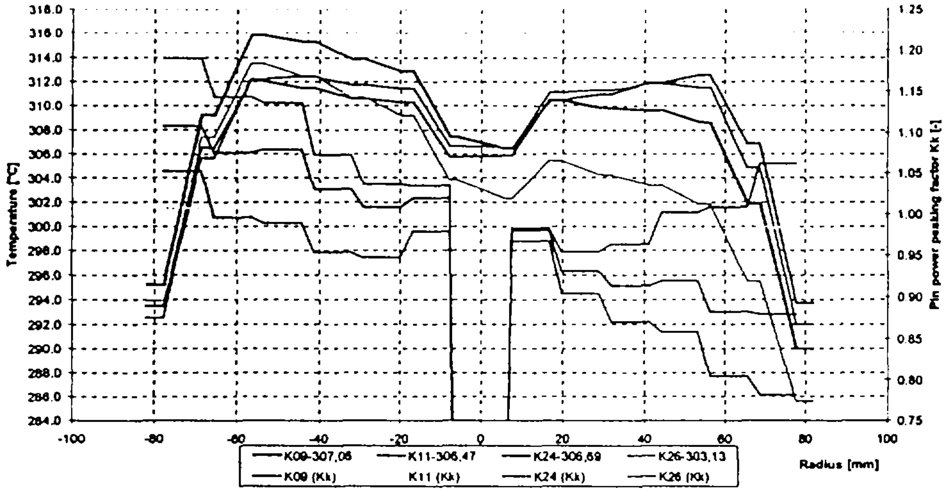
T_{in} – the inlet temperature

is about $\overline{\dfrac{\delta T_i}{\Delta T_i}} \approx (9{,}1 \pm 1{,}4)\%$ while in periphery it is nearly zero. From this it can be concluded that the fuel assembly power in central part is overestimated in compari peripheral ones. Also the by-pass flow due to this effect using the relation (1) i predicted in comparison to hydraulic measurements.

	Variant K09	Variant K11	Variant K24	Variant K26
Temperature in position of temperature measurement (thermocouple) [°C]	311,05	310,78	310,16	306,57
Average coolant temperature [°C] *	307,06	306,47	306,69	303,13
Temperature difference [°C]	3,99	4,31	3,47	3,44
$(\delta T_i / \Delta T_i)$ [%]	9,9	10,9	8,7	9,5

* - average coolant temperature in outlet from fuel part of FA

Table 8 Coolant temperatures in the upper part of fuel assembly



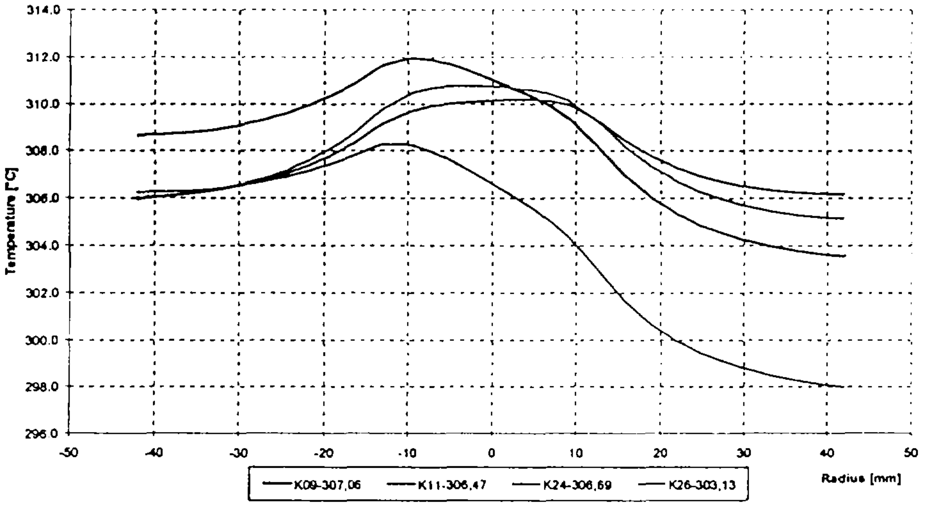


Fig. 8 Distribution of power and temperature profile at the end of fuel column and in the thermocouple position – variants K09, K11, K24 and K26

Influence of geometric changes on measured outlet temperature

For this study the fuel assemblies with ideal power distribution were selected – variants A, B, A1 and B1. The flow rate was selected as it is approximately in Mochovce NPP Unit 2, i.e. 110 m^3/h. and 108 m^3/h. The results of the calculations are presented in Table 7 and in

Fig. 5 ÷ 7. From this table it can be seen that the design changes significantly influe thermocouple readings and also the temperature ratio described by formula (2). Thi leads also to incorrect interpretation of core power distribution in mixed cores and ca power limitations as it was described in the introduction.

5. CODE DESCRIPTION

The results and description of the BIPR-7 and PERMAK codes are well knowr community of AER members, so the code description is omitted.

FLUENT 5.5.14

This code interprets liquid convection in 2D or 3D objects by using CFD ı (Computational Fluid Dynamics). TGRID or GAMBIT pre-processors genera computational mesh for CFD analysis. This analysis offers values of local flows pressures and temperatures from one elementary volume to other.

CALOPEA

CALOPEA Code calculates the profile of coolant temperatures and speed. The re the calculation is a file with 258 values of coolant temperatures and speed at the ou each fuel assembly subchannel. It calculates thermal-hydraulic parameters of fuel asse working in steady states in reactors VVER-440 and VVER-1000 by using numerical Seidel solving of continuity, energy and momentum equations.

6. CONCLUSIONS

On the basis of above mentioned calculations it can be concluded:

- The measured outlet temperatures by thermocouple system differ from the expected coolant temperatures from the fuel assemblies
- There is a difference between the mean assembly outlet temperatures and measured · in the core center and periphery
- For correct interpretation of the core power distribution based on thermocouple measurements the influence of flow temperature profile needed be accounted
- The difference between by-pass measured by hydraulic and thermal-hydraulic methc be explained by temperature profile of the coolant in the position of thermocouple
- The changes on the fuel assemblies (power profile, geometry, mixing grid, etc.) shou evaluated also in the relation to the incore measurement system

ABBREVIATIONS

FA - fuel assembly
GS - group of symmetry
k_k - fuel pin relative power
k_q - fuel assembly relative power
NPP - nuclear power plant
SPD - self powered detector
TC - thermocouple

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