4.7 PRIMARY CIRCUIT AND REACTOR CORE T-H CHARACTERISTICS DETERMINATION OF VVER 440 REACTORS

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ABSTRACT

The VVER-440 nuclear fuel vendor permanently improves the assortment of produced nuclear fuel assemblies for achieving better fuel cycle economy and reactor operation safety. During unit refuelling there also could be made some other changes in hydraulic parameters of primary circuit (change of impeller wheels, hydraulic resistance coefficient changes of internal parts of primary circuit, etc.). Therefore it is necessary to determine real coolant flow rate through the reactor during units start-up after their refuelling, and also to have the skilled methodology and computing code for analyzing factors, which affecting the inaccuracy of coolant flow redistribution determination through reactor on flows through separate parts of reactor core in any case of parallel operation of different assembly types. Computing code TH-VCR and CORFLO are used for reactor core characteristics determination for one type of fuel and control assemblies and also in case of parallel operation of different assembly types. The code TH-VCR is able to calculate coolant flow rate for different combinations of three different fuel assembly channel types and three different control assembly channel types. The CORFLO code deals the area of the reactor core which consists of 312 fuel assemblies and 37 control assemblies. Regarding the rotational 60^{o} symmetry of reactor core only 1/6 of reactor core with 59 fuel assemblies is taken into account. Computing code CORFLO is verified and validated at this time. Paper presents some results from measurements of coolant flow rate through reactors during start-up after unit refuelling and short description of computing code TH-VCR and CORFLO with some calculated results.

1 THE AIM OF THE PRIMARY CIRCUIT HYDRAULIC MEASUREMENT

Flow through the reactor is determined by hydraulic measurements on the primary circuits and by power balance measurements on the secondary circuits. This flow is in reactor core redistributed into 312 fuel assemblies and 37 control assemblies. This flow redistribution is determined by different computing codes. Reactor core is analysed as collection of parallel channels. Fundamental prerequisite which were taken into account in calculation of flow distribution in the core is constant (same) pressure difference in all reactor core channels. Pressure difference in all channels equals to pressure difference in reactor core.

The aim of the primary circuit hydraulic measurement after unit refuelling is:

- to evaluate all differences that could cause changes in hydraulic parameters of primary circuit (new type of fuel assemblies, change of impeller wheels, etc.),
- to determine coolant flow through reactor at 6 main circulation pumps (MCP) in operation - compare it with coolant flow in previous fuel cycle,
- to determine by calculation coolant flow through reactor internals,
- to determine by calculation coolant flow through fuel parts of fuel assemblies (FA) and control assemblies (CA) - in dependence on their position in reactor core and geometric properties,
- to determine hydraulic by-pass and compare it with thermo-hydraulic by-pass.

2 REALISATION OF THE PRIMARY CIRCUIT HYDRAULIC TESTS AFTER REFUELLING

Coolant circulation flow through reactor is performed by 6 MCP types 317, which are installed in cold circulation loops. Because real flows through the particular MCP delivered by design MCP characteristics (delivered by suppler of MCP) are untrustworthy, were during first start-up determined for particular MCP new-real Q-H characteristics.

Consequently these characteristics are possible to specify on basic of hydraulic characteristics measurement of primary circuit and also measurement of secondary circuit heat balances for each fuel cycle.

Realisation of reactor coolant flow measurements on primary circuit are performed in three phases:

a) during physics start up:
- steady state
- reactor power 10^{-3} - 10^{-1} N_{nom}
- 6 MCP in operation
- average coolant temperature in primary circuit 260 ^{o}C
- nominal coolant pressure in primary circuit

b) during physics start-up after reactor shut-down:
- complete 6 MCP in operation
- incomplete operation of MCP - 5, 4, 3 and 0 MCP

c) during power start -up:
- steady state
- 6 MCP in operation
- reactor power 95 ÷100 % N_{nom}

3 BRIEF DESCRIPTION OF THE COMPUTING METHODOLOGY

3.1 Determination of coolant flow through the reactor

Coolant flow through i-th Primary Circulation Loop (PCL) corrected to nominal frequency is during physics and power commissioning after refueling determined from corrected Q-H characteristics of i-th MCP, which were determined during hot non-active tests during first start-up. Corrected Q-H characteristics are determined as second-degree multi-nominal. Mean values of measured parameters during steady state are used in the algorithm. Coolant flow is determined for actual grid frequency during the measurement:

$$Q_{HCS_i} = \frac{f_{siete}}{50} * \sum_{k=0}^{2} A_i(k) * \left(\Delta p_{HC\check{C}_i} * \left(\frac{50}{f_{siete}} \right)^2 * \frac{\rho_{270}}{\rho_{vst}} \right)^k$$

where:

Q_{HCSi}	[m^3/hour] is coolant flow through i-th PCL,
$\Delta p_{HC\check{C}\,i}$	[kPa] is measured pressure drop at i-th RCP,
$A_i(k)$	are multi-nominal coefficients of i-th RCP,
f_{siete}	[Hz] is measured grid frequency,
ρ_{270}	[kg/m^3] is coolant density for reference temperature 270 °C,
ρ_{vst}	[kg/m^3] is coolant density for reactor inlet temperature.

Volume coolant flow through the reactor is defined as a sum of coolant flows through the individual PCL:

$$Q_R = \sum_{i=1}^{6} Q_{HCSi}$$

Volume coolant flow through PCL and reactor [m^3/hour] is converted into mass flow [kg/s] for actual coolant temperature at reactor inlet:

$$G_R = Q_R * \frac{\rho_{vst}}{3600} \qquad G_{HCS_i} = Q_{HCS_i} * \frac{\rho_{vst}}{3600}$$

Thermal power N_{tep_i} of individual PCL is defined:

$$N_{tep_i} = G_{HCS_i} * (h_{v\acute{y}st_i} - h_{vst_i})$$

where:

G_{HCSi}	[kg/s] is coolant flow through i-th PCL,
$h_{v\acute{y}st_i}$	[kJ/kg] is coolant enthalpy at reactor outlet for i-th PCL,
h_{vst_i}	[kJ/kg] is coolant enthalpy at reactor inlet for i-th PCL.

3.2 Methodology of determination of coolant flow through the reactor internals

Coolant flows through reactor internals are determined on the basis of:

- known hydraulic resistances of reactor internals including their uncertainties,
- known power distribution in all FA and CA,

- geometric properties of FA and CA,
- position of FA and CA in the reactor core,
- calculated coolant flow through the reactor and their uncertainties,
- knowledge of coolant transfer in reactor core.

Computing scheme of reactor internal parts for VVER-440 is on the Figure 1. After entering the coolant into the reactor through input nozzles coolant passes through the down comer, then through the perforation of the core barrel bottom the flow through RC is divided into 312 FA channels and into 37 CA channels, removing generated heat in fuel pins and it led away through outlet nozzles.

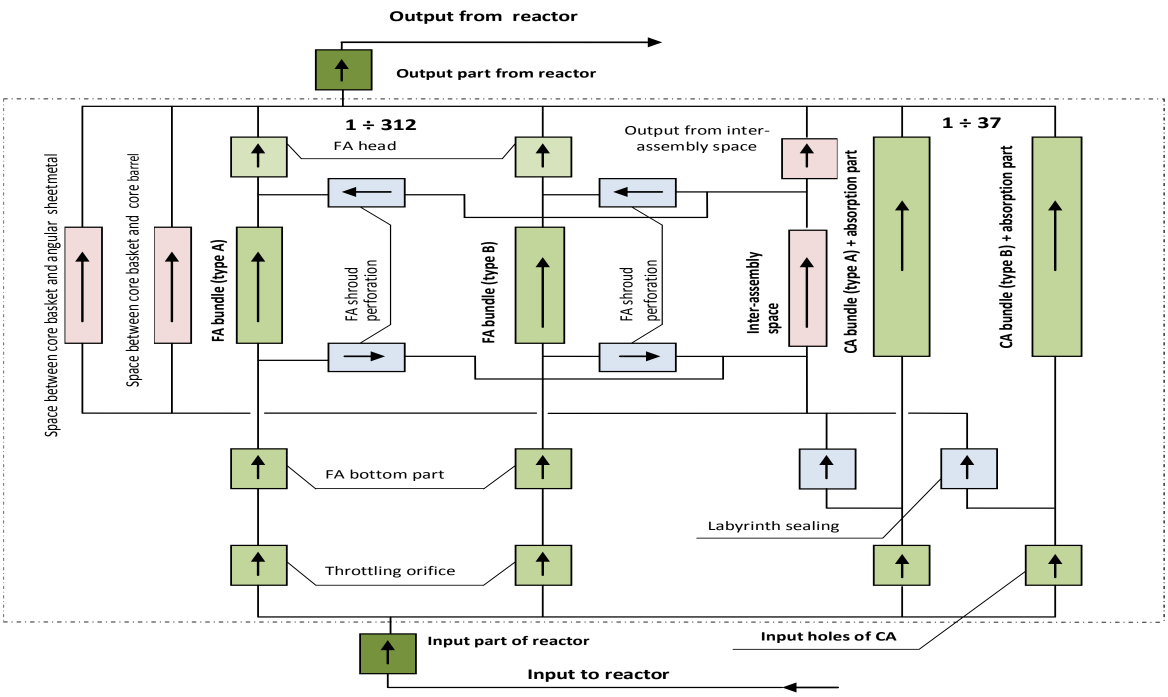
In the FA channel coolant flows enter through the throttling orifice in the upper plate of the core barrel bottom) and afterwards it is divided into a flow through the bundle of fuel pins and a bundle by-pass through the lower perforation of the shroud into the inter-assembly space and through the upper perforation back into the channel. There it mixes with heated flow rising from fuel bundles. Then joined flow rises into the upper plenum.

In CA channel coolant passes firstly through input holes in protection tubes and then is divided into a flow through the fuel bundle of the CA assembly and an assembly by-pass through the labyrinth. After passing the labyrinth coolant flows through the inter-assembly space, there an RC by-pass is separated by the angular sheet metal and basket core barrel (RC shell) and finally it rises to the upper plenum. The main flow in the CA channel passes fuel bundle, then the absorption assembly part and afterwards through holes in protection tubes it rises also into the upper plenum.

A part of coolant from FA, having passed through the holes in the lower shroud perforation, and also from the CA channel, passing through the labyrinth sealing, flows into the inter-assembly space and it does not take part in heat removal from the bundle of fuel pins.

Coolant being under the total pressure difference in the active core is divided between FA and CA fuel parts according to their hydraulic resistances.

Figure 1 Computing scheme of reactor internal parts for reactor VVER-440



Pressure differences in the core and hydraulic resistance

Pressure differences in the core on internal reactor parts in stationary modes are calculated on the basis of design hydraulic resistances, gained on the basis of experiments, results of commissioning and reference manuals.

Next part shortly describes the methodology of T-H parameters determination in reactor core.

Note 1: Used notation:

ξ - hydraulic resistance coefficient [-]

ρ - coolant density [kg/m^3]

(index 1. at coolant inlet temperature to RC)
(index 2.. at coolant output temperature from assembly)
(index s. at average coolant temperature)

Note 2: All coefficients ξ refer to the same flow rate and to the same coolant density corresponding to reference temperature**.**

Present scheme shows detailed pattern of internal reactor parts resistances. The following formulas are used:

Reactor (reference temperature t_s-average temperature)

$$\xi_R = \xi_R^{VST} * \frac{\rho_S}{\rho_1} + \xi_{AZ} + \xi_R^{VYST} * \frac{\rho_S}{\rho_2}$$

Input part of reactor (reference temperature t_1 - inlet temperature) ξ_R^{VST}

Output part of reactor (reference temperature t_2 - outlet temperature) ξ_R^{VYST}

Reactor core (reference temperature t_s - average temperature)

$$\xi_{AZ} = \left(\frac{349}{\left(\sum_{i=1}^{312} \frac{1}{\sqrt{\xi_{PKNi}}} + \sum_{j=1}^{37} \frac{1}{\sqrt{\xi_{HRKNj}}} \right)} \right)^2$$

Fuel assembly channel (reference temperature t_{Si} - average temperature)

$$\xi_{PKNi} = \xi_{Ci} * \frac{\rho_{Si}}{\rho_1} + \xi_{PKi}$$

Throttling orifice (reference temperature t_1 - inlet temperature) ξ_{Ci}

Fuel assembly (reference temperature t_{Si} - average temperature):

$$\xi_{PKi} = \xi_{VSTi} * \frac{\rho_{Si}}{\rho_1} + \frac{1}{\left(\frac{1}{\sqrt{\xi_{PKi}^{SV}}} + \frac{1}{\sqrt{\xi_{PKi}^{P+MP} * \frac{\rho_{Si}}{\rho_1}}} \right)^2} + \xi_{VYSTi} * \frac{\rho_{Si}}{\rho_{2i}}$$

Inlet part of fuel assembly (reference temperature t_1 - inlet temperature) ξ_{VSTi}
Outlet part of fuel assembly (reference temperature t_{2i} - outlet temperature) ξ_{VYSTi}
Perforation of FA shroud and inter-assembly space (reference temperature t_1) ξ_{PKi}^{P+MP}
Fuel bundle (reference temperature t_{Si} - average temperature) ξ_{PKi}^{SV}

Control assembly channel (reference temperature t_{Sj} - average temperature)

$$\xi_{HRKNj} = \xi_{HRKj}^{VST} * \frac{\rho_{sj}}{\rho_1} + \frac{1}{\left(\frac{1}{\sqrt{\xi_{HRKj}^{L+M}}} * \frac{\rho_{Sj}}{\rho_1} + \frac{1}{\sqrt{\xi_{HRKj}^{PAL+A}}} \right)^2}$$

Input part of CA channel (reference temperature t_1) ξ_{HRKj}^{VST}
Labyrinth and inter-assemblies space (reference temperature t_1) ξ_{HRKj}^{L+M}

Fuel and absorption extension of CA (reference temperature t_{Sj})

$$\xi_{HRKj}^{PAL+A} = \xi_{HRKj}^{PAL} + \xi_{HRKj}^{A} * \frac{\rho_{Sj}}{\rho_{2j}}$$

Fuel part of CA reference temperature t_S) ξ_{HRKj}^{PAL}
Absorption extension of CA (reference temperature t_2) ξ_{HRKj}^{A}

Note: Coefficients ξ_{PKi}^{SV} (fuel part of FA) a ξ_{HRKj}^{PAL} (fuel part of CA), their friction parts depend on coolant flow and are determined on the basis of Reynolds number.
ξ_{PKi}^{SV} as a complement of ξ_{VSTi} and ξ_{VYSTi} on ξ_{PKi},
ξ_{HRKj}^{PAL} as a complement of ξ_{HRKj}^{A} on ξ_{HRKj}.

Fundamental prerequisite, which were taken into account in calculations of flow distribution in the core, is constant (same) pressure difference in all reactor core channels. Pressure difference in channel equals to pressure difference in core and is given by formula:

$$\Delta p_{AZ} = \xi_{PK,HRK}^{KAN} * \frac{\left[G_{PK,HRK}^{KAN} \right]^2 * 10^{-3}}{2 * F_K^2 * \rho_S} \qquad \text{[kPa]}$$

where $G_{PK,HRK}^{KAN}$ [kg/s] - average coolant flow through FA or CA channel
$F_K = 0{,}009238\ m^2$ - reference flow area of FA
ρ_S [kg/m^3] - average specific weight
$\xi_{PK,HRK}^{KAN}$ - hydraulic resistance coefficient (HRC) FA or CA channel

For individual channels in reactor core (FA and CA) are determined resistance coefficients $\xi_{PK,HRK}^{KAN}$ (with taken into account by pass and various specific weights - given by various powers of channels). Providing the same pressure differences Δp_{AZ} are determined coolant flow through the individual channels $G_{PK,HRK}^{KAN}$:

$$G_{PK,HRK}^{KAN} = \sqrt{\frac{\Delta p_{AZ} * 2 * \rho_s}{\xi_{PK,HRK}^{KAN} * 10^{-6}}} * F_K$$

Coolant flow through the fuel bundle of assembly is determined with taken into account the influence of by-pass through the perforation and inter-assembly space (or through the labyrinths and inter-CA space).

Coolant heat up at fuel bundles is calculated from enthalpy increase:

$\Delta h_i = \frac{N_i}{G_{PK,HRKi}^{PC}}$,where: N_i is power and $G_{PK,HRKi}^{PC}$ is coolant flow through fuel bundle of FA_i

3.3 Basic characteristics of fuel parts of fuel and control assemblies

Fuel assemblies are fixed in bottom plate of core basket in triangular grid with a pitch of 147 mm.

Basic characteristics of different types of FA and fuel parts of CA are listed in Table 1.
All the values in the table are average.

Table 1 Basic characteristics of FA and fuel parts of CA

Name	**Dimen-sion**	**Type A** (Fe SG)	**Type B** (Zr SG)	**Type C** (Profiled assembl.)	**Type D** (Gd-II 9,07 mm)	**Type E** (Gd-II 9,1 mm)
Pin pitch	[mm]	12,2			12,3	
Outer diameter of fuel pin	[mm]	9,1			9,07	9,1
Spacing grid pitch	[mm]	250		240	250	
Spacing grid height	[mm]	10			10 and 20	
Number of spacing grid - height 10 mm	[pcs]	10			7	
Number of spacing grid - height 20 mm	[pcs]	0			3	
Fuel assemblies						
Outer key dimension / Shroud thickness	[mm]	143,0 / 1,5		145,0 / 1,5		
Flow area	[m^2]	$88{,}40\cdot10^{-4}$		$91{,}84\cdot10^{-4}$	$92{,}38\cdot10^{-4}$	$91{,}84\cdot10^{-4}$
Hydraulic diameter of fuel pin	[mm]	8,60		8,90	8,99	8,90
Fuel length in cold state	[mm]	2420			2480	
Fuel parts of control assemblies						
Outer key dimension / Shroud thickness	[mm]	144,2 / 2,1		145,0 / 1,5		

Name	Dimension	Type A (Fe SG)	Type B (Zr SG)	Type C (Profiled assembl.)	Type D (Gd-II 9,07 mm)	Type E (Gd-II 9,1 mm)
Flow area	[m^2]	$87{,}00 \cdot 10^{-4}$		$91{,}84 \cdot 10^{-4}$	$92{,}38 \cdot 10^{-4}$	$92{,}38 \cdot 10^{-4}$
Hydraulic diameter of fuel pin	[mm]	8,40		8,90	8,99	8,90
Fuel length in cold state	[mm]	2320			2360	

Coolant what enters the reactor from cooling loops, flows through the down-comer (a downward part between the core barrel and reactor vessel), and proceeds to the barrel bottom (lower plenum) through holes in the elliptical core barrel bottom part. In the elliptical core barrel bottom part coolant divides itself between FA and CA fuel parts. Coolant for FA cooling passes through the throttling orifice, placed in the upper plate of the core barrel bottom, and further it passes through the FA lower nozzle, part of coolant leaves into the inter-assembly space through lower perforations of the FA shroud, but main part passes through the bundle of fuel pins. In the FA head, the main stream, which passed through the bundle of fuel pins, mixes with coolant coming from the inter-assembly space through upper perforations of the FA shroud, and it proceeds into the volume of protective-tube bank.

Coolant coming for cooling the bundle of fuel pins of the CA fuel part passes through input holes of CA protective tubes, then a part of coolant proceeds through the labyrinth sealing of the basket plate into the inter-assembly space and the main coolant stream passes through the bundle of fuel pins of the CA fuel part.

A negligible part of coolant passes from the inter-assembly space into spaces between the angular sheetmetal and the basket and between the basket and the core barrel and its basic part passes between FA shrouds and CA fuel parts. Coolant passes from the inter-assembly space into the volume of the guard-tube bank through the upper perforation of FA as well as through the perforation of guard tubes, before that it intermixes with coolant, which passed through the bundle of fuel pins of CA fuel parts. From the volume of the protective-tube bank (upper plenum) all coolant having passed through the core barrel perforation, proceeds into cooling loops.

The values of HRC of the coolant in-vessel route in the reactor, FA, and CA fuel parts, gained on the basis of FA and CA fuel parts dummy tests and during the trial operation of NPP VVER 440 units, list of parameters used at present are introduced in Table 2.

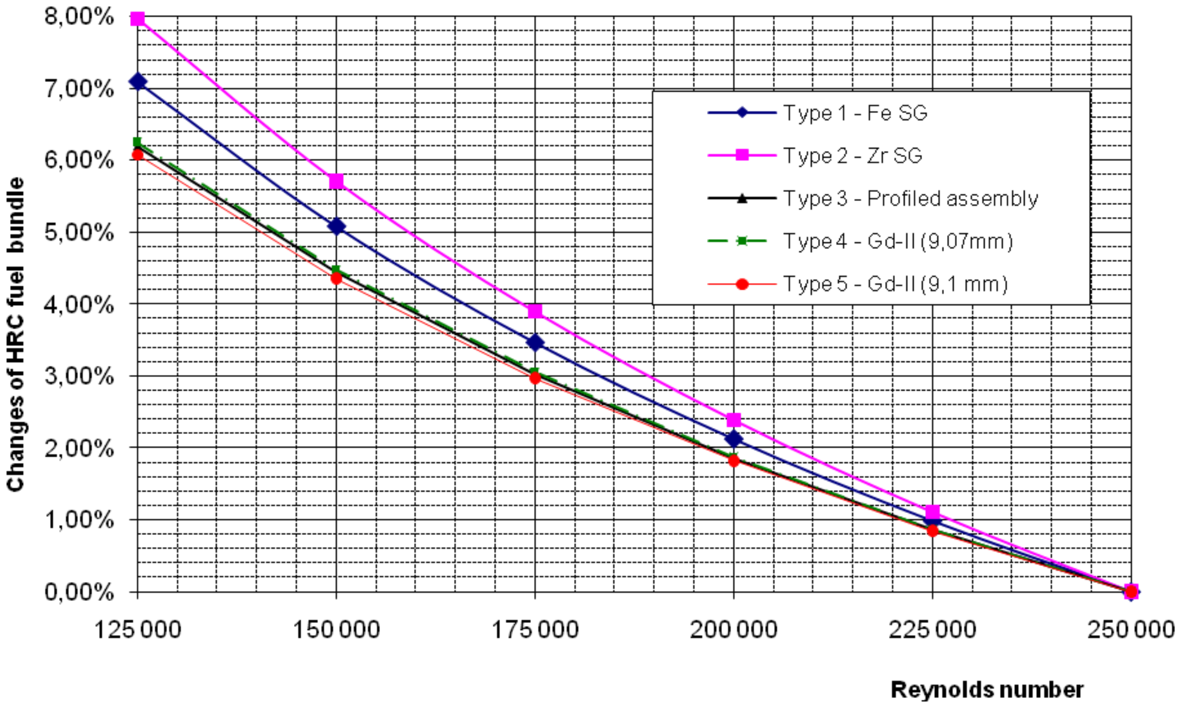
Table 2 Example of list parameters of hydraulic resistance coefficients (HRC) of reactor internal parts that are delivered by fuel assemblies vendor which are used by evaluation codes

Num.	Name of parameter
1	HRC of FA together with throttling orifice for type assemblies with outer key dimension 145,0 mm (pin diameter 9,07 mm) for reference flow area in m^2
2	HRC of FA for type assemblies with outer key dimension 145,0 mm (pin diameter 9,07 mm) for reference flow area in m^2

Num.	Name of parameter
3	HRC of throttling orifice for reference flow area in m^2
4	HRC of FA base for type assemblies with outer key dimension 145,0 mm (pin diameter 9,07 mm) for reference flow area in m^2;
5	HRC of FA head for type assemblies with outer key dimension 145,0 mm (pin diameter 9,07 mm) for reference flow area in m^2
6	HRC of CA channel inlet for reference flow area in m^2
7	HRC of labyrinth sealing of CA channel for reference flow area in m^2
8	HRC of perforation FA shroud (for reference flow area, $F=7{,}63\cdot10^{-4}$ m^2)
9	HRC of space between core basket and formed tinplate (for reference flow area, $F=0{,}4398$ m^2)
10	HRC of space between core basket and core barrel (for reference flow area $F=0{,}1458$ m^2)
11	HRC output from between assemblies space for reference flow area in m^2
12	HRC of CA type for reference flow area in m^2

Values of HRC internal reactor parts are in literature standard presented for Reynolds number 250 000. Fuel bundle HRC dependence of different assembly construction and Reynolds number is presented on the Picture 2. Presented range is valid approximately for coolant flow rate through reactor core of reactor type VVER-440 from 3 to 6 MCP in operation.

Figure 2 Fuel bundle HRC dependence on assembly construction and Reynolds number of VVER-440 reactors



Accuracy of calculation of T-H characteristics of reactor core is given by inaccuracy of power distribution in reactor core calculation (from BIPR-7) and uncertainty of coolant flow determination (and its distribution in the core)

Uncertainty of coolant flow through the fuel bundles of fuel assemblies determination depends on values of HRC of individual reactor internal parts. Coefficients of hydraulic resistance are taken with tolerances. Uncertainties of coolant flow through the reactor internals result from these tolerances. Values of HRC internal reactor parts are in literature presented for Reynolds number 250 000.

The coolant heat-up is considered only in fuel part of fuel bundles. The heat-up of by-passes by convection through assembly shroud is not considered – heat balance of channels is not influenced by that and otherwise heating would make an influence only by changing the medium density at the pressure loss evaluation, but that is respected by normalising of hydraulic resistance coefficients to a certain temperature.

Calculation accuracy also depends on accuracy of formulas used for thermo-physical properties of coolant. First of all they are specific mass and specific enthalpy (accuracy of used multi-nominals is circa 0,03 %).

4 EVALUATION PROGRAMS (CODES) OF COOLANT FLOW RATE THROUGH REACTOR INTERNALS

Computing codes developed in VUJE used for stationary analyses of T-H characteristics of the reactor core are “TH-VCR” and “CORFLO”.

4.1 Description of TH-VČR code

Computing code TH-VČR is used for stationary computing analyses T-H characteristics of reactor core. This code was developed in VUJE. Program TH-VČR is used for detailed solving of coolant flow distribution in reactor. Scheme of flow distribution can be seen on the Figure 1. Program can calculate flow rate through FA and CA channels, dummy elements, and flow outside FA and CA and outside core.

Program is able to calculate coolant flows through internal parts of reactor core for different combination of three different fuel assembly channels and three different control assembly channels.

The coolant heating is considered only in fuel part of fuel bundles. The heating of by-passes by convection through assembly walls is not considered – heat balance of channels is not influenced by that and otherwise heating would make an influence only by changing the medium density at the pressure loss evaluation, but that is respected by normalising of hydraulic resistance coefficients to a certain temperature.

Code „TH-VCR” was used for determination of coolant flow through reactor internals during first commissioning of NPP Dukovany, Jaslovské Bohunice and Mochovce. Program is still innovated and is stagnantly used for determination of coolant flow through the internal reactor parts of 1st and 2nd unit Mochovce NPP and 3rd and 4th unit Bohunice NPP during periodical measurement of hydraulic characteristics of I.O. after refuelling.

Main aim of calculation is to determine T-H characteristics of reactor core (fuel assembly power, coolant temperature at fuel bundle outlet, coolant flow through fuel bundle of fuel assemblies for various types of FA loaded into core and hydraulic by-pass of fuel). Program calculates parameters of FA with average and maximum power.

Accuracy of calculation of T-H characteristics of reactor core is given by inaccuracy of power distribution in core calculation (from BIPR-7) and uncertainty of coolant flow determination (and its distribution in the core).

A lot of measurements of hydraulic characteristics of reactor and the whole I.O. and II.O in year's 1984÷1987 during consecutive commissioning of individual reactors type VVER-440 were performed. These measurements were after evaluation the basis of specification of hydraulic characteristics of reactor core and the whole I.O. and the basis of correction of MCP characteristics.

4.2 Description of CORFLO code

The CORFLO code calculates steady thermo-hydraulic characteristics of VVER-440 RC within 60° symmetry. The CORFLO code was configured based on the knowledge of resistance characteristics of reactor internal parts which were determined within measurements and evaluations of hydraulic characteristics at preoperational complex testing, commissioning and operation of Czechoslovak NPPs with VVER 440 reactors of the type V230 and V213 in years 1978÷2006. The resistance scheme of the VVER 440 reactor is presented in Fig. 1. The CORFLO code deals with the area of the active core which consists of 312 FA and 37 CA. Regarding the rotational 60° symmetry of RC only 1/6 of RC with 59 fuel assemblies is calculated, numbering of assemblies is according to BIPR code. Eventual asymmetry of coolant temperatures at RC inlet, which can occur in cases when output temperatures from SG of single loops are different and perfect intermixing is not in the lower plenum, is not considered in this code for now. The coolant flow distribution in single RC fuel channels is calculated while their resistance characteristics are the most important, but also the influence of a different power of single fuel assemblies (by means of different specific masses of coolant) and of their arrangement in RC are taken into account, which is essentially lower. Coolant flow rates through fuel bundles are decreased against input coolant flow rates into fuel channels by coolant by-passes through the shroud perforation of FA and by coolant by-passes through labyrinths of CA into the inter-assembly space. Based on the knowledge of resistance characteristics and the power of single assemblies, corresponding flow rates and coolant temperatures at the output of assembly fuel bundles are iteratively determined.

A main objective of a performed calculation is except for gaining RC thermo-hydraulics characteristics (including the map of relative and absolute powers, output temperatures from fuel bundles and flow rates through bundles) also to determine the most loaded assemblies of the active core. Then their parameters (power, flow rate) enter the CALOPEA code, which performs a subchannel analysis of these assemblies.

The basic assumption of the CORFLO code, from which it follows at the calculation of the flow rate distribution in the reactor core, is an assumption of an equal pressure loss on all reactor core channels (equal to the pressure loss on RC). For single RC channels (FA and CA) HRC are determined with considering by-passes and partially different specific masses (given by different powers of single channels) and then under the assumption of reaching equal pressure loss on the RC particular flow rates are determined iteratively. Input HRC values are given for Reynolds number = $2{,}5 \cdot 10^5$. The code distinguishes friction and local losses of assemblies therefore consequently it calculates HRC of these assemblies for relevant flow rate.

Flow rates through assembly fuel bundles are determined then with considering the influence of the by-pass through the perforation and inter-assembly space (labyrinth and inter-assembly space in case of CA, respectively). Heating on fuel bundles are calculated from enthalpy increase.

The validation of CORFLO code which is closely connected with its inputs a neutron-physical calculation of RC. In order to validate the calculated power distribution in single assemblies of RC a comparison was done with parameters measured at EMO12 units. The records of those parameters are performed systematically in the beginning of particular cycles. The calculations confirmed a quite good compliance of CORFLO code results with measured coolant heating (those were averaged for symmetric assemblies from single sextant of RC) on assembly outlet.

5 ACCEPTANCE CRITERIA OF THE HYDRAULIC TEST

- Coolant flow through the reactor at nominal reactor power and grid frequency 50 Hz is 43 000 ± 2000 m^3/hour.
- Coolant flows through the fuel parts of assemblies at nominal reactor power are within the safety criteria limits

  100 ÷ 130 m^3/hour.
- Pressure difference at MCP at nominal reactor power is 450 ± 25 kPa
- Pressure difference at the reactor at nominal reactor power is 255 ÷314 kPa

6 RESULTS

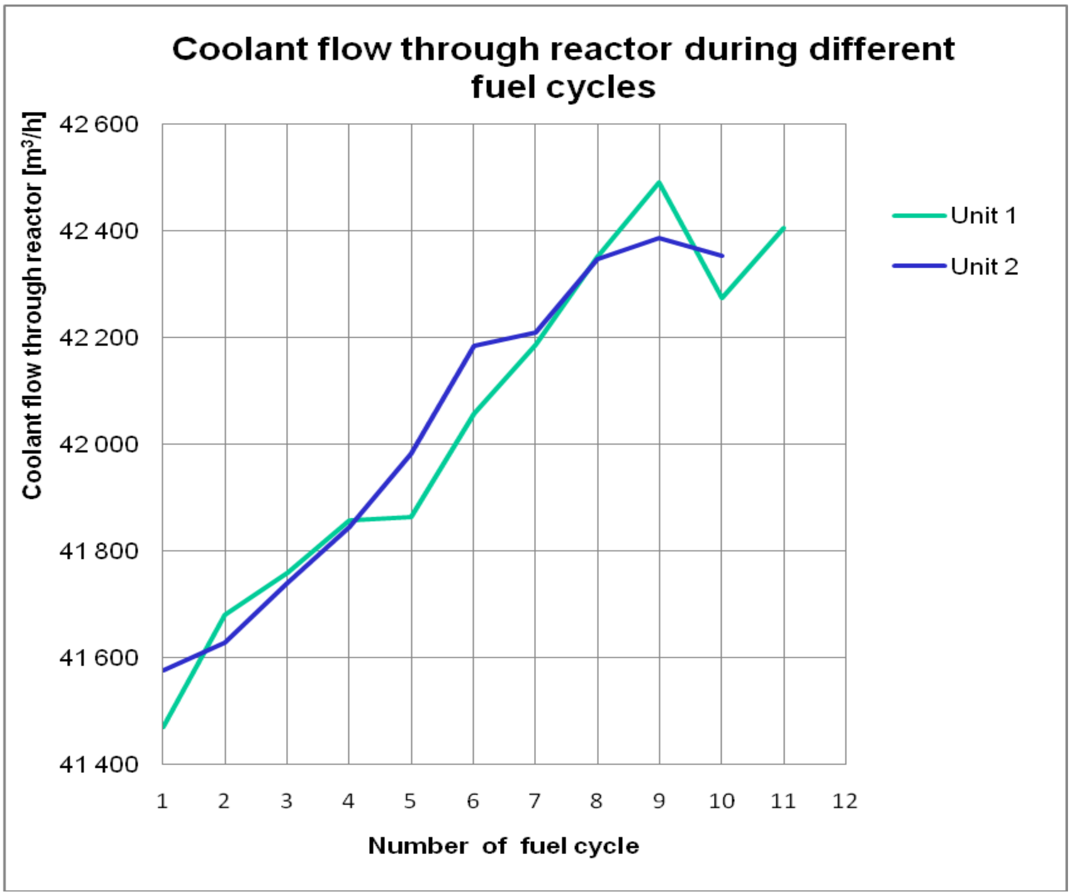
6.1 Coolant flow through reactor VVER 440

Coolant flows through reactors of the units VVER-440 operated in Slovakia are determined regularly by measurement at the beginning of fuel cycles after their refuelling. Changes of coolant flow through reactors could be subjected by different hydraulic resistance coefficients of new fuel assemblies charging in reactors and by changing of some impeller wheels of MCP. There are pointed some results from measurements at two Slovak units in Table 3. During all operation of these two units were not changes any MCP impeller wheels, all coolant flow through reactor are produced by changes of hydraulic resistance coefficients of internal parts of primary circuit. Values from the Table 3 and Picture 3 indicate that coolant flows through reactors were increased approximately 2 %, because new fuel and control assemblies (profiled and Gd-II) charged in the reactor core had lower hydraulic resistant coefficients than older type assemblies which were taken out from reactor.

Table 3 Coolant flow through reactor during different fuel cycles

Number of fuel cycle	Unit 1		Unit 2	
	value	change	value	change
	[m³/h]	[%]	[m³/h]	[%]
1	41 470	0	41 575	0
2	41679	0,50	41 628	0,13
3	41 760	0,70	41 740	0,40
4	41 858	0,94	41 845	0,65
5	41 863	0,95	41 983	0,98
6	42 058	1,42	42 184	1,46
7	42 187	1,73	42 209	1,52
8	42 351	2,12	42 346	1,85
9	42 491	2,46	42 387	1,95
10	42 273	1,94	42 353	1,87
11	42 406	2,26		

Picture 3 Coolant flow through reactor during different fuel cycles



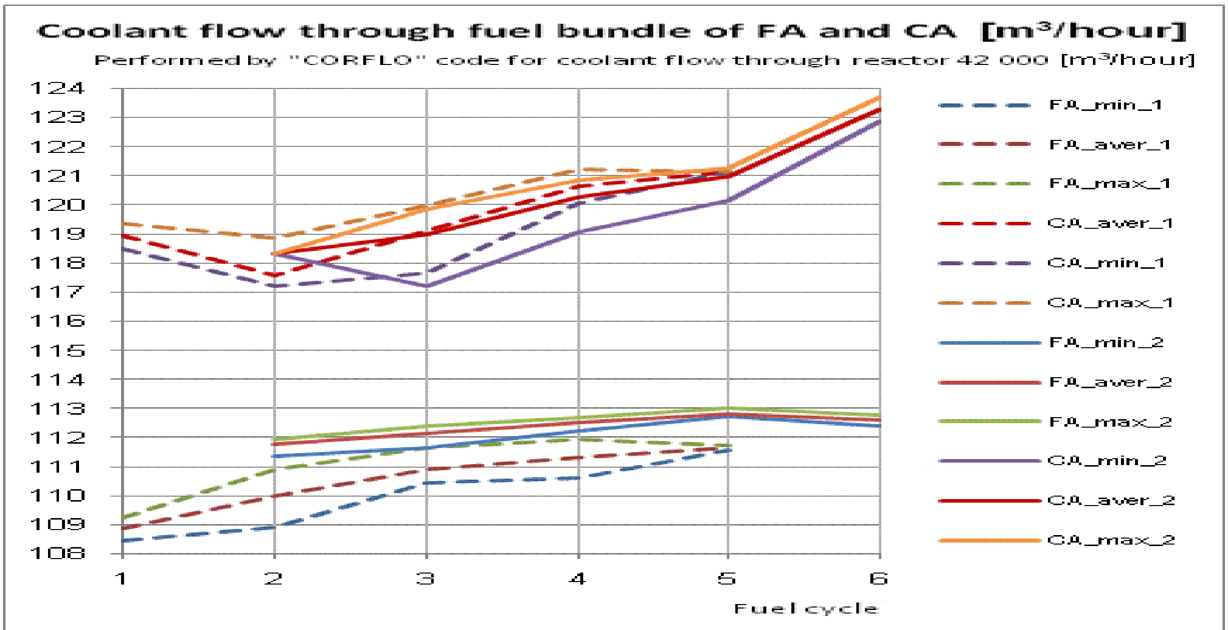
6.2 Coolant flow through the internal part of reactor core

Coolant flows through internal parts of reactor core were calculated by both computing codes TH-VCR and CORFLO. Maximal differences were recognized during substitution profiled or Gd assemblies (outer key dimension 145 mm) onto older assembly types (outer key dimension 143 mm). Results from calculation for different fuel cycles performed by code CORFO for coolant flow through reactor 42 000 m^3/h, inlet temperature 268 °C a grid frequency 50 Hz are presented in Table 4 and Picture 4.

Table 4 Coolant flow through the internals part of reactor core for reactor coolant flow 42 000 [m^3/h] during different fuel cycles

Name	Variant	V_1	V_2	V_3	V_4	V_5	V_6
Number of fuel assemblies	Type 1	312	246	174	102	24	0
	Type 2	0	66	138	210	288	312
Number of control assemblies	Type 1	37	25	19	7	1	0
	Type 2	0	12	18	30	36	37
FA average fuel bundle flow [m^3/h]	Type 1	108,9	110,0	110,9	111,3	111,7	-
	Type 2	-	111,8	112,1	112,5	112,8	112,6
CA average fuel bundle flow [m^3/h]	Type 1	118,9	117,6	119,1	120,6	121,2	-
	Type 2	-	118,3	119,0	120,3	121,0	123,3
FA flow difference [m^3/h]	Type 1	0,78	1,99	1,21	1,32	0,15	-
	Type 2	-	0,58	0,76	0,46	0,32	0,35
CA flow difference [m^3/h]	Type 1	0,87	1,64	2,32	1,16	0	-
	Type 2	-	0	2,64	1,76	1,12	0,83
Total flow difference [m^3/h]	FA	0,78	2,99	1,97	2,06	1,46	0,35
	CA	0,87	1,64	2,76	2,13	1,12	0,83
Fuel bundle core flow by-pass	[%]	8,64	7,64	6,71	6,12	5,6	5,47

Picture 4 Coolant flow through fuel bundle of FA and CA during different fuel cycles



6.3 Uncertainties of coolant flow rate through reactor and fuel bundle of FA and CA

Uncertainties of coolant flow rate through reactor and through fuel bundle of fuel and control assemblies were determined from uncertainties of measured parameters and uncertainties of HRC of internal parts of reactor core. Their values are presented in Table 5.

Table 5 Uncertainties of coolant flow rate through reactor and fuel bundles of fuel and control assemblies

Name of parameter	Uncertainty
Coolant flow rate through reactor VVER 440	± 1,0 %
Coolant flow rate through fuel bundle of fuel assemblies	± 1,7 %
Coolant flow rate through fuel bundle of control assemblies	± 2,4 %

7 CONCLUSION

1. Realizations of the hydraulic measurements on the primary circuit during units start-up after their refuelling confirm smaller changes of coolant flow rate through the reactors in dependence of hydraulic resistance coefficient changes of internal parts, changes of impeller wheels, etc. In spite of this, it is necessary to determine real coolant flow rate through the reactor during units' start-up after their refuelling.

2. As the VVER-440 nuclear fuel vendor permanently improves the assortment of produced nuclear fuel assemblies for achieving better fuel cycle economy and reactor operation safety, it is necessary to have the skilled methodology and computing code for analyzing factors, which affecting the inaccuracy of coolant flow redistribution determination through reactor on flows through separate parts of reactor core in any case of parallel operation of different assembly types.

REFERENCES

[1] Hermanský J., Závodský M. et al.: Coolant flow rate determination trough individual assemblies of reactor core VVER-440, VUJE, Trnava, august 2007

[2] Hermanský J. et al.: PRESAR Chapter 6.1.4 Thermal and hydraulic Project for NPP Mochovce, unit 3 and 4, 2/2008

[3] Klučárová K., Hermanský J. et al.: Actual stage of solution in temperature in homogeneity coolant flow rate on the outlet of fuel assemblies and in determination of relative coolant flow trough the assemblies in case of parallel operation of different fuel assembly types, VUJE Trnava, V01-32645/2008

[4] Klučárová K., Hermanský J. et al.: Computing modelling of temperature in homogeneity on the outlet of fuel assemblies and creation of methodology for determination of relative coolant flow through the assemblies, VUJE Trnava, V01-26674/2009, November 2009