# SYSTEM FOR NEUTRONIC & THERMAL-HYDRAULIC DIAGNOSTICS OF CHANNEL-TYPE REACTOR PARAMETERS

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## ABSTRACT

This paper presents several advances in mathematical models and computer codes intended for operational support of RBMK reactors. The primary goal of the development is to check the correctness of local changes in the flux and power distributions with time. Mathematical algorithms for diagnostics of a variety of safety related parameters were developed. A specialised diagnostic and monitoring system based on the confluent analysis of measured data and results of neutronic and thermal-hydraulic calculations of a reactor core was designed. A multi level algorithm for diagnostics is triggered if certain disagreement between parameters related to calculated and measured data is observed.

## I. INTRODUCTION.

The main objective of the research is to create a tool that can find errors occurred in tracking of RBMK core history. A variety of factors cause distortions in the recorded evolution of flux and power distributions with time. Major factors are

a) high sensitivity of the neutron flux modelling to variations in the input data;

b) high uncertainties in reactor parameters;

c) errors in information associated with on-power channel refuelling;

d) distortions in control rod position data.

### A. High sensitivity of the neutron flux modelling to variations in the input data.

For quite some time, it has been assumed that a high accuracy of neutron flux simulation can be achieved by employing more powerful computers and developing precise techniques for solution of the diffusion equation. Despite a true-two-group diffusion model, which has been incorporated in the off-line NPP computer to calculate what can be referred fundamental model[1], it turned out that actual flux distributions in a real RBMK reactor core, as RBMK depends on deterministic factors as well as quasi-deterministic and stochastic factors. Furthermore, RBMK appeared to be remarkably sensitive to small variations in core parameters due to high neutronic decoupling of the core[2] and a positive void coefficient that affects both reactor kinetic properties and the spatial flux distribution.

### B. High uncertainties in reactor parameters.

Although an uncertainty in neutronic parameters is an inevitable factor in any reactor operation, the RBMK core is encountering a particularly significant impact of such. Suffice it to say that RBMK does not have a system for tracking 3D irradiation distributions with time. That factor negatively affects the accuracy of axial power and flux calculations, leads to an incorrect estimation of absorbing properties of control rods, and, therefore, contributes to the distortion of the global flux shape, which is calculated off-line by a diffusion-theory code.

Due to that and some other factors, the accuracy of flux mapping can be significantly diminished. In order to maintain high accuracy, an on-line program that performs the flux mapping, does not use the straightforward results of diffusion calculations, but utilises the more appropriate fundamental mode. This mode is also calculated by a diffusion code, which 'adjusts' depths of control rods so that global flux distribution is more close to a 'time-average' flux or to that in reality. Consequently, a significant reduction of standard deviation in the differences between computed and measured detector fluxes was achieved. Although this approach is displaying excellent performance, it also creates extra difficulties for diagnostic tools when someone tries to find and to localise an error that was inserted, for example, into control rod position data used in the calculations.

C. Errors in information associated with on-power channel refuelling.

Like CANDU, RBMK reactors are refuelled on-power -- there are between one and three refuellings a day. This is a technically complicated process involving an operation of a 100 ton "monster' -- a fuelling machine. Unlike CANDU, where double-ended fuelling is typically employed, the RBMK machine first removes the burnt-up fuel and then inserts the new fuel into the fuel channel. Thus, significant deformations of the axial neutron flux occur (Figure 1, Figure 2). At a certain moment of refuelling, the maximum fuel assembly line power may significantly -- up to 25% -- exceed the final value (Figure 3)[3],

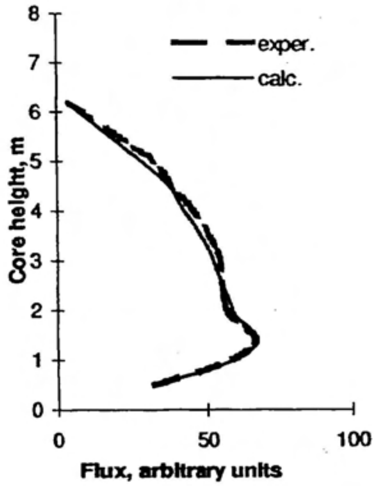


Figure 1: Normalised axial distribution, FA 100% loaded.

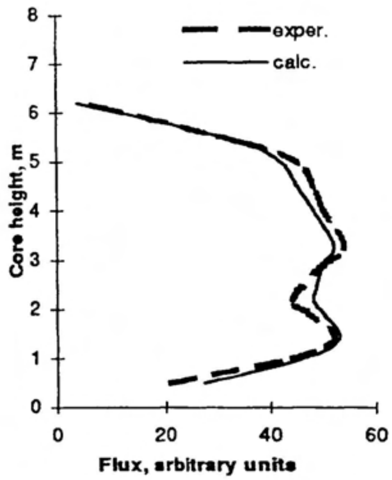


Figure 2: Normalised axial distribution, FA withdrawn on 3 m.

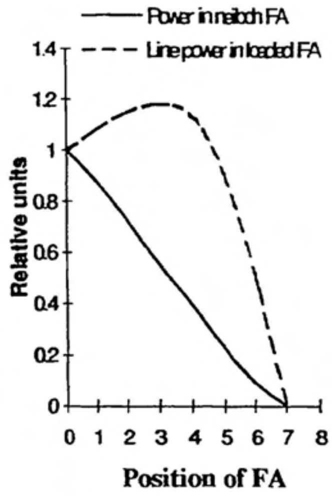


Figure 3: Safety parameters versus the depth of FA unloading.

which usually just complies with the safety margin. Even small movements of control rods or a fuel assembly may lead to a dramatic increase of linear power, whereas the online flux monitoring system does not generate any signals to worry about, because the system is essentially two dimensional. These and some other facts make personnel very concerned with safety and reliability aspects of operation during a refuelling. However, the burden of registration and recording of parameters also rests on the personnel who can – and actually did -- insert errors in the safety important information, which is used in further reactor operation. Even though errors of that kind appear very rarely, fidelity in information associated with on-power channel refuelling is not 100%.

D. Distortions in control rod position data.

Control rod movements can also contribute to the distortion of the flux and power distributions due to such causes as self-moving of a control rod, non-identified failure of a control rod driver, and other causes resulting in the erroneous control rod position information. Those errors can remain hidden for a long time.

However, such kind of errors can be identified by a diagnostic algorithm based on confluent analysis of measured data and results of neutronic & thermal-hydraulic calculations of the reactor core. Multi level algorithm for diagnostics is triggered if certain disagreement between parameters related to calculated and measured data is observed.

## II. NEUTRONIC DIAGNOSTICS

Development of mathematical algorithms for diagnostics of a variety of safety related parameters required the following task to be solved:

a. to create a calculational tool so that the three-dimensional core flux distribution can be calculated with a diffusion-theory code and then the flux distribution will be reconstructed using a statistical interpolation so as to produce a best fit to the detector readings,

b. to create an algorithm to check the correctness of some important neutronic parameters of the core -- control rod positions and fuel assembly burn-up -- using accurate three-dimensional flux distributions,

c. to develop an algorithm that allows us to find inconsistencies between corrected neutronic parameters and thermal-hydraulic parameters.

### A. Calculational tool for the three-dimensional core flux calculation and reconstruction.

Assuming that we know the power density in a number of cells containing a detector, the one-group neutron field local structure (LS) technique was developed so that it can be used to obtain the power density distribution, which a quantity which, in fact, has a "one-group representation". Additionally, an algorithm was developed that takes into account in-core detector readings by correcting the power density LS, which was obtained from the neutronic calculations. A method was also developed to deal with three-dimensional geometry by using a set of the two-dimensional equations[4]. Finally, the KRATER computer code was developed that is a program for three-dimensional power distribution monitoring. Its flux reconstruction is based on measured data (in-core detector readings) and results of neutronic calculations produced by the code itself. The first version of the code was implemented into trial operation at the Smolensk NPP in 1984, and at the Chernobyl NPP. At that time NPPs were equipped with low-power ES-1022 type computers. Hence, there were strong limitations on random access memory and program execution time. Besause of that, an efficient algorithm had to be developed, which transforms a 3-D problem to a set of 2-D auxiliary problems.

An initial step of the KRATER verification was carried out in the course of code implementation into trial operation. The verification was performed by using data from the SCALA regular monitoring system and the results of axial neutron flux measurements. These measurements were carried out using a copper wire activation at the Smolensk and Chernobyl NPPs. An estimated error of the channel-by-channel axial power peak factor (Kz) was about 3.7%.

After KRATER was tested at the RBMK-1000 reactors, a new version of the code was developed for the Ignalina NPP RBMK-1500 reactors. This program was designed for the much more powerful ES-1045 computer and had an advanced service shell.

Afterwards, the KRATER code was installed at the Kursk and Leningrad NPPs. Operation capability was provided for any ES-type computers (from 1020 to 1066). The calculation algorithm and user interface were improved and a lot of calculations for axial field scanning experiments were done.

A new step in KRATER code development was made when personal computers began to spread everywhere including NPPs. It appeared to be possible to design a PC version of the code because of algorithm flexibility and efficiency that were initially put into the code. The PC version of the code could be run even on IBM PC AT-286/287.

At the same time, the first KRATER user interface was created. It manages initial data input and displays calculation results using graphic capabilities of the PC monitor. This shell was developed according to NPP staff recommendations and users found it very convenient for operation. Another direction of the KRATER code modernisation was the development of new additional modules. They were designed to solve the following problems:

- calculation of the operational reactivity margin of control rods and axial distribution of reactivity margin on the base of the 3D neutron field distribution;
- calculation of the departure factor from critical power and critical linear power taking into account axial power density distribution;
- channel-by-channel calculation of the graphite temperature distribution;
- calculational detection of defective radial detectors.

Nowadays the KRATER code finds a new level of application at the Chernobyl NPP. According to Order of Ukrainian Nuclear Regulation Board (Ukraine Gosatomnadzor, Resolution N 18 of June 8, 1993) a use of the KRATER for computation of the axial channel peaking factor was included into operation regulation. The code has been adopted as the official source of 3-D power distribution , used to compute a safety margin coefficient of acceptable linear power in the reactivity control rod channels (Kzl) (the so called Kramerov's criterion). In particular, the Chernobyl NPP operation rules have the following paragraph: "At reactor power level higher than 70% of nominal, the Reactor Operator on Duty must run the KRATER code in order to compute the 3D power distribution as often as once per hour and must pass a printout of the channel linear power to the Chief Reactor Operator. The Chief Reactor Operator must run the code and analyse the printout himself in case when Kzl<1.00, Kzl>1.24, and after elimination of these deviations."

This operation condition is a result of verification researches which were carried out at the Chernobyl NPP. One set of calculations on IBM PC 386/387 DX 40 takes 3-4 min only. It allows the personnel to monitor very carefully the maximum channel linear power, which is of significant importance due to its influence on the maximum channel graphite temperature. The former affects the magnitude of the (positive) void-power effect[5]. Due to this effect, the value of the maximum channel linear power is one of the major safety related parameters[6].

### B. An algorithm to check correctness of control rod positions and fuel assembly burn-up

On a basis of the KRATER methodology and code, a diagnostic algorithm and module were developed for calculational detection of non-authentic control rod depths, that is breaking control rod detection.

There are 1884 fuel channels (FC) in the RBMK core, and 130 in-core radial flux detectors are uniformly distributed in the core. Their signals are measurable parameters, those can also be calculated using a neutronic code. As was mentioned above, there is a disagreement between the results of neutronic and thermal-hydraulics calculations and measured data even under ideal conditions of operation. Major 'natural' contributors to that disagreement are variations of nuclide composition, physical, and chemical properties of fuel and uncertainties in control rod absorption. 'Artificial' contributors are errors in the burn-up and in control rod positions (i.e. discrepancy between a real position and what the instrumentation indicates or what is used as input data for off-line neutronic calculations).

In order to find an error in any given set of control rod positions, the diagnostic technique analyses discrepancies between two different calculations of the distribution of channel power. The first distribution ('restored') is calculated by

KRATER using the input data concerned and corresponding detector readings. The second distribution ('predicted') is calculated using older -- e.g. accurate – neutronic input data and then corrected to account for recent detector readings. Correction coefficients, i.e. differences between calculated and measured values, are determined only for channels, those are considered as authentic. It turned out, that 'old' correction coefficients are valid for power prediction if the burn up is negligible during the prediction period. In addition, consequent -- from day to day -- application of similar diagnostic techniques makes it possible to find systematic errors in detector signals used and to exclude them.

If there are no errors in the 'new' input data, the distribution predicted in such a way should be close to the distribution restored with using 'new' data. The restored and predicted power distribution disagreement is not more than 5% in absence of errors in control rod positions or in fuel assembly burnup. If an error occurred, for example, in some control rod position indicator, a local perturbation of the power distribution appears in the rod involved. According to this technique, the magnitude of discrepancy between the predicted and measured values may be used for determination of the authentic position of a control rod or fuel assembly burn up. Application of this the technique to a real NPP requires that the discrepancy existing between the predicted and measured values be decomposed into a random component (background) and a deterministic component originated from disagreement between the real core and its misrepresentation in the reactor computer system.

The following assumptions are a basis of this diagnostics method:

- There are at least two files of input data available and one of these files can be considered as correct or a corrected one, while the other one may contain errors to be found. This means that errors occurred after the first file was recorded.
- If the control rod position seems to have an error (an undefined failure of a control rod drive), the probability of appearance of similar error in the nearest control rod is low;
- No more new errors of any type appear in the neighbourhood of the erroneous control rod after the first file was recorded.
- The discrepancy between the restored $\vec{w}^r$ and the predicted $\vec{w}^p$ power distributions in channels containing a detector can be written as:

$$\vec{\varsigma} = \vec{x} + \lambda\,\vec{y}, \tag{1}$$

$$\vec{\varsigma} = F\,(\vec{w}^r - \vec{w}^p)\ , \tag{2}$$

$$(F)_{ij} = \begin{cases} f_1, & i=j \\ 0, & i \neq j \end{cases},\ (i,j = 1,2,\ldots,N_d),$$

$N_d$ - the number of detectors in the core;

$\vec{x}$ - the discrepancy between the restored and predicted power distributions without errors;

$\vec{y}$ - the perturbation form-function;

$f_1$ - some value, derived from measured data;

$\lambda = f(\delta h)$ - the perturbation magnitude when an error in the control rod position equal to $\delta h$.

The perturbation form-function was derived from calculations using the diffusion-theory part of the computer code KRATER. The different stencils (the disposition of control rods and detectors) were used: 6-detector stencil; 8-detector stencil and 12-detector stencil.

Suppose there is an error in position of a control rod. Define the disagreement between correct and erroneous calculated power distributions as follows:

$$\delta w_j^i = 1 - w_j^i / w_j^0$$

where: $w_j^0$ - calculated power distribution without error;

$w_j^i$ - calculated power distribution with error in control rod #1 position;

Figure 4 shows the maximum disagreement between the correct and erroneous power distributions depending on the error in the rod position $\delta h^i$ for different stencils. This dependence was calculated by the diffusion-theory part of computer code KRATER.

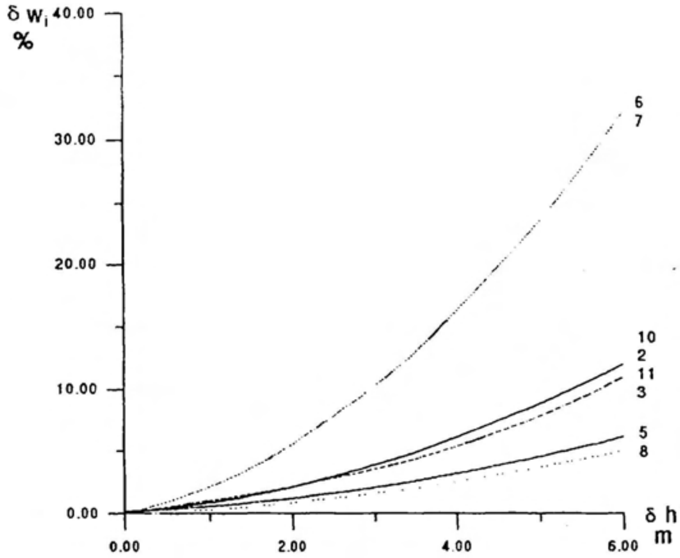


**Fig. 4: Maximum disagreement between correct and erroneous power distributions versus the value of error in the control rod position $\delta h^i$.**

The power distribution perturbation is a local effect, so it is reasonable to take into account only the signals of the detectors that are placed at a short distance from the "suspicious" control rod. Therefore, dimensions of vectors in formulae (1), (2) can be decreased from $N_d$ to $N_d^t$ - the number of detectors in a stencil, so:

$$(\vec{y})_j = \delta w_j, \ (j = 1, 2, \ldots, N_d^t).$$

where $\delta w_j$ is the discrepancy between the restored and predicted power distributions when the error in control rod position is 6 meters. In order to find the 'erroneous' control rod and the value of the error $\delta h$, let us consider the problem of defining the perturbation magnitude. We will search for the perturbation magnitude $\lambda$ using the folowing equation:

$$\min_\lambda \|\vec{\varsigma} - \lambda \vec{y}\|.$$

as a norm we will consider the quadratic A-norm:

$$\|\vec{x}\|_A = \sqrt{(\vec{x},\vec{x})_A} = \sqrt{(\vec{x}, A\,\vec{x})}$$

and $\lambda$ can be written as:

$$\lambda = \frac{(\vec{\varsigma}, \vec{y})_A}{(\vec{y}, \vec{y})_A} = \frac{(\vec{\varsigma}, \vec{p})}{(\vec{y}, \vec{p})}\ , \qquad (3)$$

where: $\vec{p} = A\,\vec{y}$.

Let us select À to satisfy the condition to minimise the variance $\lambda$ at the noise signal set:

$$\left\{\vec{x}_k,\ \ k = 1, 2, \ldots, N\right\}.$$

In the final analysis $(\vec{p})$ can be defined as:

$$(\vec{p})_j = y_j - \frac{1}{N_d^t}\sum_{j=1}^{N_d^t} y_j \qquad (4)$$

So, the diagnostics algorithm consists of the following steps:

1. Perform the diffusion-theory calculation in order to derive the perturbation form-function $\vec{y}$ for each control rod.
2. Calculate the weight coefficients $\vec{p}$ using Eq. (4).
3. Calculate the predicted power distribution using correct -- 'old' -- input data.
4. Calculate the restored power distribution using the set of correction coefficients.
5. Using Eq. (2), calculate discrepancies between restored and predicted power distributions $\vec{\varsigma}$ for each control rod for detectors belonging to its stencil.
6. Obtain the perturbation magnitude according to Eq. (3) for each control rod in the core.
7. Identify the control rods, for which the value of $\lambda$ is higher than the variance of $\lambda$ (control rods "suspected" in incorrect position indication).

The dependence of the power distribution perturbation magnitude $\lambda$ on the value of control rod position error $\delta h$ is presented in Figure 5. The magnitude calculations were carried out on the 8-detector stencil. A drop of the control rod from the heights of 1 m, 2 m, ..., 6 m to 0 m was modelled. In order to estimate a selectivity of the algorithm, the perturbation magnitude $\lambda$ in neighbouring rods was calculated depending on the value of the error and on the type of stencil, Figure 6. Study shows that the 8-detector stencil is optimal from point of view of the computational efficiency, selectivity, and sensitivity. However, control rods that are close to side reflector have less than eight detectors in their vicinity.

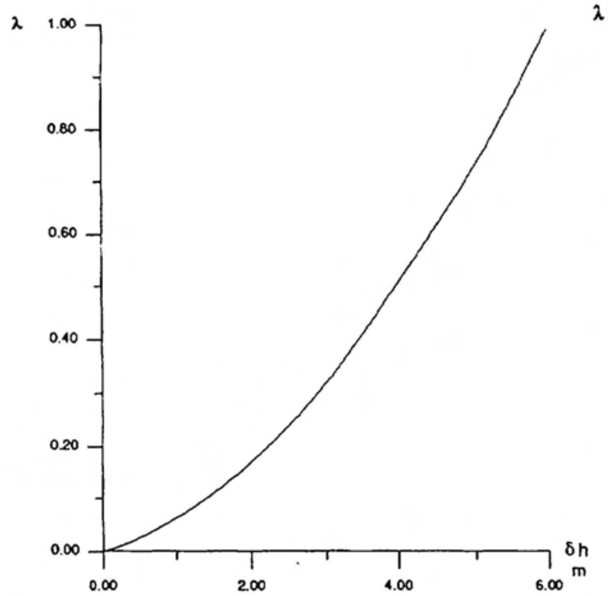


Fig. 5: The dependence of power distribution perturbation magnitude $\lambda$ on the value of control rod position error $\delta h$. ($\lambda$ is presented in per-unit, 1 corresponds to movement for a distance of 6 m)

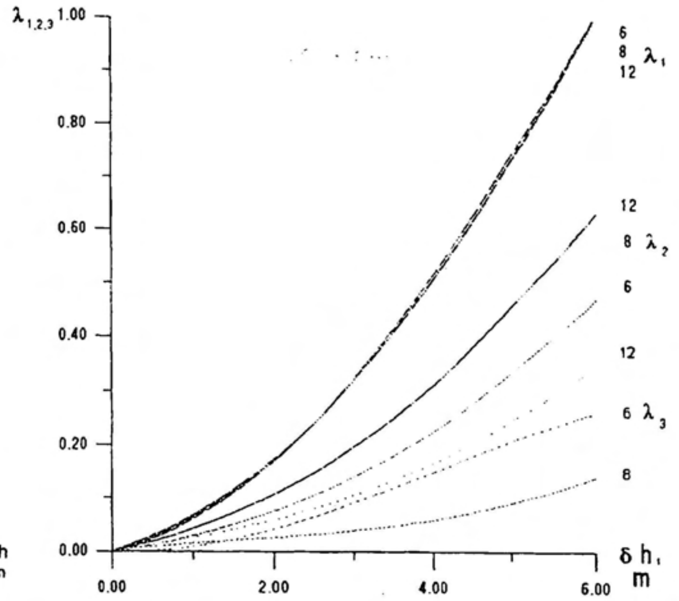


Fig. 6. The dependence of perturbation magnitude $\lambda$ for rods #2 and #3 on value of error in rod #1 position using stencils with 6, 8 and 12 detectors.

A response of the neutron and power distributions to a move of a control rod is sufficiently high to be captured by the nearest in-core detectors. Since the average distance between control rods is close to that between detectors, there are always enough 'nearest' detectors around a control rod and, therefore, even a small error in a control rod position can easily be found by the technique similar to one described above.

Detection of errors in channel burn up is a more complicated problem. First, we assume that response of the neutron and power distributions to that kind of error is sufficiently high. This assumption corresponds to the situation when reactor operating personnel records that a fresh fuel assembly was loaded in the core whereas, in reality, it was a spent fuel assembly, or vice versa. Although nearest detectors would capture the unexpected deformation of neutron flux, the grid of detectors is coarse as compared to that of fuel channels, therefore it is difficult to localise an erroneous fuel channel exactly. Now, assume that a small error was introduced in the burnup data. Experience shows that this is quite possible, and that it may significantly reduce the efficiency of the fuel assembly use. Moreover, it may lead to a cladding failure. However, that kind of problem cannot be resolved by using the diagnostic algorithm based solely on power distribution detectors.

### C. An algorithm to check consistence between corrected neutronic parameters and thermal-hydraulic parameters.

In order to solve the above-mentioned problems it was suggested to use additional measured information, such as coolant flow rates which are available for each fuel channel, contrary to the sparse grid of in-core neutron detectors. Completely independently from measurements, the set of coolant flow rates can be calculated by a neutronic & thermal-hydraulics model. Statistical comparative analysis of both sets creates a basis for diagnostics.

The neutronic & thermal-hydraulics code BARS/COTT [3], which computes channel-by-channel distributions of all essential physical parameters of an RBMK core, was chosen as a calculational tool for the diagnostic algorithm concerned. Being fully coupled, the BARS/COTT model is able to account for thermal feedback and offers greater accuracy of neutronic calculations. Such safety parameters as margin for burnout, critical power margin, maximal temperatures of fuel and moderator can also be obtained by this code. The code allows virtually any desired operational situation (including cold criticality and departure from nucleate boiling) to be modelled. The thermal-hydraulic model with BARS/COTT has been extensively validated against measurements at Kursk NPP.

An iterative procedure of coupled steady state calculations involves the following stages:

1. Diffusion-theory calculation of the channel power distribution by the BARS code.

2. Thermal-hydraulic calculation by the COTT code using the power distribution as input data. The code COTT provides a set of channel physical characteristics, such as the axial distribution of the coolant temperature, pressure, and density; fuel temperature distribution and others.

3. Based on the results of the thermal-hydraulic calculations, some of the initial data for neutronic calculations are recalculated. For example, macroscopic cross sections of the fuel channel are functions of temperature and coolant density distributions.

Subsequently, steps 1-3 are iterated until the required accuracy is achieved.

Because of the time consuming character of the iterative procedure and due to limited capacities of computers available at the NPP, some simplifications of the procedure were to be implemented without loosing a required accuracy. As a logical limit of the possible accuracy deterioration, a value of existing uncertainty of the fuel channel technological data was chosen.

Thus, taking all these factors into account, the following assumptions were used:

- An axial one dimension model is applied to the steam and water mixture, which moves from the lower channel edge to its upper edge. One dimension means that in every cross section of the fuel channel only average values of the parameters are used. These parameters depend on an axial co-ordinate only. This simplification becomes reasonable because of a negligible radial peaking factor, which is 1.04 or even less within the fuel channel cross section.

- The two-phase coolant flow is treated as a steam and water mixture at thermodynamic equilibrium. In order to take into account the difference between the water and vapour flow rates a so-called "slip"-coefficient is used.

There are also some other less important assumptions.

The main purpose of the thermal-hydraulics calculation for the diagnostics problem is to calculate the channel coolant flow rates and to compare them with the current indications of flow rate detectors. The equations and balance formulas used in the thermal-hydraulics model are presented earlier. Here, we only point out the essence of the diagnostic algorithm based on coolant flow indication. If an error was inserted in input data during the registration of refuelling, it results in a corresponding perturbation of the calculated power distribution, which affects the calculated flow rates in turn. Thus, there is set of measured coolant flow rates and the calculated one. A procedure of comparison and statistical filtration reveals regions of systematic disagreement between sets. The existence of such regions could be interpreted indication of possible mistakes of one type or another. That initiates additional calculations using the method of statistical filtration, similar to what was described above with respect to control rod position diagnostics.

## III. CONCLUSIONS.

Mathematical algorithms for diagnostics of variety safety related parameters were developed.

A specialised diagnostic and monitoring system based on the confluent analysis of measured data and results of neutronic & thermal-hydraulic calculations of a reactor core was designed to check the correctness of control rod positions and fuel assembly burn-up. Minimum errors that can be detected by the system are as follows: 1.0 meter in control rod position, 6.0 MWD/kgU in a fuel assembly burn up.

The developed algorithms can probably be used for operational support of other reactor types.

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