ANALYSIS OF COOLANT MIXING IN VVER-440 FUEL ASSEMBLIES WITH THE CODE CFX

Summary of the Ph.D. thesis

SÁNDOR TÓTH

Supervisor:
DR. ATTILA ASZÓDI

Budapest University of Technology and Economics
Institute of Nuclear Techniques
2010

Preliminaries

In case of normal operation of nuclear reactors, leakage of the radioisotopes from fuel rods is not allowed therefore integrity of the claddings has to be sustained. Safety limits are prescribed for various thermal quantities in order to ensure the integrity of the fuel elements. The limited quantities have to be monitored during the cycle and operation has to be done without violations of the safety limits. Adequate planning of the fuel load patterns and adequate operation of the reactors are needed to satisfy the safety restrictions.

Knowledge of fuel assembly thermal hydraulics is important in terms of safe operation. In the fuel assemblies of the pressurized water reactors, volumetric boiling of the coolant is not allowed therefore outlet average temperatures of the subchannels are limited. In the industrial practice, outlet temperatures of subchannels are determined with subchannel codes. Research of the coolant mixing in the rod bundles is necessary in order to develop and verify these codes. In the case of VVER-440/213 reactors, the core outlet temperature field is measured above 210 fuel assemblies. Using these measured data as input, computational models are applied to determine the actual values of the limited parameters during operation. For this reason, the interpretation of the detected data is an important question, for which knowledge of the coolant mixing in the assembly heads is necessary.

Turbulent coolant mixing in the rod bundles has been studied mainly experimentally before the '90s. Measurements have shown that the structure of rod bundle flows is different from the structure of pipe flows [Trupp and Azad, 1975; Vonka, 1988] because of the higher anisotropy of the turbulence and presence of secondary flows. According to Vonka, the average magnitude of the secondary flows in P/D=1.3 (P: pitch, D: diameter) rod bundles is 0.1% of the bulk velocity.

Temperature field inside VVER-440 fuel assemblies has been investigated in detail by the Kurchatov Institute first [Kobzar and Oleksyuk, 2006]. The measurements were performed close to operational conditions. Temperature distributions have been measured at the end of the heated part of the rods and at the level of the in-core thermocouple. The measurements have revealed non-uniformity of the temperature field at the elevation of the in-core thermocouple.

Recently, rapid increase of computers' performance and intensive development of general purpose CFD (Computational Fluid Dynamics) codes allow to use them in the research area of the fuel assembly thermal hydraulics. The first CFD model of the VVER-440 active rod bundles has been developed by Finnish researchers using the FLUENT [Gango, 1997] and FINFLO codes [Rautaheimo et al., 1999]. Based on their results, the spacer grids have significant effect on the axial velocity distribution and pressure drop. A 240 mm long 60 degree segment model of the fuel assemblies has been developed using the CFX code at Budapest University of Technology and Economics [Aszódi and Légrádi, 2002]. Conclusions reached by the authors were similar to the ones by the Finish researchers. Common characteristics of the calculations were that the predicted temperature distributions have not been validated because of the lack of appropriate measurement data. Isothermal fully developed flow in a bare subchannel of the VVER-440 rod bundles has been studied with the Lattice Boltzmann numerical method [Mayer and Házi, 2006]. Large eddy simulations for Reynolds number of 21 000 have shown the fluctuations of the axial and peripheral velocity components and the secondary flows.

Investigation of the coolant mixing in the assembly heads is in focus of research in countries that are operated VVER-440 reactors. One of the first CFD models of the assembly heads has been developed at the Budapest University of Technology with the CFX code [Légrádi and Aszódi, 2003]. According to their results, significant differences (even about 4 °C) could be between the in-core thermocouple signal and the average coolant

temperature at the thermocouple level. Magnitudes of the deviations could not be confirmed by operational data. Temperature distributions in the assembly heads have been studied by Slovak VUJE Research Institute using the FLUENT code [Petényi et al., 2003]. Similar conclusions were drawn by the Slovak researchers as the Hungarian experts. The problem has been investigated by Finnish researchers with the FLUENT code [Toppila et al., 2004]. Based on their results, the coolant is not perfectly mixed at the level of the in-core thermocouple and the subchannel groups have different weight in the in-core thermocouple signal. Based on testing with operational data, the weight factors overestimated the effect of the imperfect coolant mixing therefore they have not been introduced in any nuclear power plants. It has to be mentioned that none of the head models has been validated acceptably because appropriate measurement data were not available.

Goals

Research of coolant mixing in fuel assemblies is important because of the reasons mentioned above. Actuality of the present research is given by the near future introduction of the fuel assemblies with burnable poison at Paks Nuclear Power Plant. There are differences in the geometry and power distribution between the "new" and the "actual" assemblies therefore the internal thermal hydraulic processes differ as well.

My research goals were the development of validated CFD models for various parts of VVER-440 fuel assemblies and the investigation of three-dimensional turbulent coolant mixing with these models. I intended to study the effects of the mixing grid and spacer grid on the coolant flow and the relation between the in-core thermocouple signal and the outlet average temperature in the case of various fuel assemblies. Moreover, I intended to determine weight factors of rod bundle regions for the in-core thermocouple.

CFD codes have been applied only since a decade in the research of fuel assembly thermal hydraulics, therefore there are many open questions in this field. I looked for the answer for some of these questions as well.

With my research results, I intended to contribute to the deeper understanding of VVER-440 fuel assembly thermal hydraulics; to the improvement of subchannel codes and core monitoring system of Paks Nuclear Power Plant. Moreover, I intended to give guidelines in some questions connected to the CFD modeling of fuel assemblies.

New scientific results

1st thesis

I have proved that the Reynolds-Averaged Navier-Stokes Simulation (RANS) is suitable for calculating the Reynolds stresses and secondary flows accurately in bare rod bundles with pitch to diameter ratio of 1.35. I have specified guidelines for proper selection of the spatial resolution and turbulence model. [1][2]

I have developed a CFD model for a bare subchannel of VVER-440 fuel assemblies (*P/D* of 1.35) with the ANSYS CFX code. I have investigated isothermal, fully developed turbulent flow of Reynolds number 60 000. I have proved with validating my results to Trupp and Azad measurement data (1975) that RANS simulation is suitable for calculating the Reynolds stresses and secondary flows accurately in bare rod bundles of *P/D*=1.35. Based on detailed sensitivity studies and validation, I have specified the following guidelines for the mesh and turbulence model in order to calculate the Reynolds stresses and secondary flows accurately:

The inner edge lengths of the cells need to be smaller than (*P-D*)/18 and $y^+ \approx 20\text{-}40$ or $y^+ \approx 1$ criterion needs to be satisfied for the near wall mesh by using automatic near wall treatment.

Among the Reynolds stress models, the BSL Reynolds stress model is suitable to calculate accurately the turbulent stresses and secondary flows in the investigated rod bundles.

2nd thesis

I have shown that the turbulent mixing between the subchannels of VVER-440 fuel assemblies is more intensive than the convective mixing between them. The spacer grids of the assemblies intensify the turbulent mixing between the subchannels up to ~50 mm downstream and they cause significant convective mixing between them up to ~70 mm downstream. In the remaining region between two spacer girds, mixing processes present in bare rod bundles dominate so there is no significant convective mixing between the subchannels. [1][2]

I have developed CFD models of a section of VVER-440 rod bundles with and without spacer grid in order to investigate the effects of the spacer on the coolant mixing. In the model development, I have applied the guidelines determined through the subchannel analysis. With the models, I have investigated turbulent flows of near operational Reynolds number (230 000). According to my results, the turbulent mixing between subchannels of the rod bundle with spacer grid is more intensive than the convective mixing between them (effective velocities of the mixings are 130–240 mm/s and 0–50 mm/s respectively). The spacer grid intensifies the turbulent mixing between neighbouring subchannels up to ~50 mm downstream and causes significant convective mixing (effective velocity of the mixing > 5 mm/s) between them up to ~70 mm downstream. In the remaining region between two spacer grids, the convective mixing between the subchannels decreases (3–4 mm/s) and turbulent mixing processes typical for bare rod bundles (130–145 mm/s) dominate. In accordance with others' results [Gango, 1997; Rautaheimo et al., 1999], I have concluded that the spacer grid has significant effect on the axial velocity profile and the outlet average temperatures of the subchannels.

3^{td} thesis

I have shown with a validated CFD model of the active part of the VVER-440 rod bundles that the point-like measurement data of the Kurchatov Institute can not be directly used to verify the outlet temperature field of a subchannel code. I have applied this CFD model to verify the maximum subchannel outlet temperature calculated with the COBRA code. [3][4][5][6]

I have developed a three-dimensional CFD model of VVER-440 rod bundles in order to calculate the outlet temperature field. By comparing my results to measurement data of the Kurchatov Institute performed on a full scale test facility, I have proved that the CFD model is able to predict acceptably the outlet temperature field of the rod bundles. With the analysis, I called the attention that the measurement data can not be directly employed to verify the outlet temperature field of a subchannel code because there are significant differences (0.5–2 °C) between the outlet average temperatures of the subchannels and the temperatures in the measuring points at the center of the subchannels. For this reason, the direct comparison could be misleading.

I have applied the rod bundle model with the suitable approximations to verify the maximum subchannel outlet temperature calculated with the COBRA code. The differences between temperatures calculated with two methods are not higher than 0.4 °C.

4th thesis

Based on detailed sensitivity studies and comparison to measurement data, I have specified guidelines concerning the inlet boundary conditions and difference scheme of the convective terms for calculating the coolant mixing in VVER-440 fuel assembly heads with CFD codes. [4][5][6][7]

In order to investigate the coolant mixing in VVER-440 fuel assembly heads, I have developed CFD model. I have validated the model with the measurement results of Kurchatov Institute. Based on detailed sensitivity studies and comparison to measurement data, I have specified guidelines concerning the inlet boundary conditions and difference scheme of convective terms for calculating the coolant mixing in VVER-440 fuel assembly heads with CFD codes.

According to my investigations, the assembly head model is less sensitive to the quality of the inlet boundary conditions. Therefore they can be determined with a subchannel code and there is no need for using a CFD code, which demands massive hardware resources and computational costs.

Second order accuracy scheme has to be used in the assembly head calculations for differencing the convective terms. First order upwind scheme overestimates the coolant mixing in the assembly heads due to strong numerical diffusion therefore it provides non-conservative results. First order scheme is not able to predict the unsteady character of the flow because of the before mentioned reason.

5th thesis

5th thesis

With numerical calculations, I have shown that the coolant mixing is rather intensive in the head parts of various VVER-440 fuel assemblies but the distance between the ends of the fuel pins and the thermocouple (~360 mm) is too short to obtain perfect mixing. The difference between the thermocouple signal and the outlet average temperature of a fuel assembly depends on the assembly type and its pin power distribution. For this reason, the thermocouple correction has to depend on the pin power distribution. Deviations calculated with the CFD models are in good agreement with the deviations experienced in the nuclear power plant. [8][9]

I have developed CFD models for the head parts of the 3.82% enrichment profiled fuel assemblies with 12.2 mm and 12.3 mm pitch and the head parts of the 4.2% enrichment fuel assemblies with burnable poison. I have studied the coolant mixing in these assemblies taking different pin power distributions into account. Based on my results, coolant mixing is rather intensive in the fuel assembly heads – as the temperature fluctuations refer to it – but the ~360 mm distance between the ends of the fuel pins and the thermocouple is too short to achieve perfect mixing. The difference between the thermocouple signal and the outlet average temperature of a fuel assembly depends on the assembly type and its pin power distribution. In the case of fuel assemblies with burnable poison, the deviation strongly depends on the burn-up as well. For these reasons, the thermocouple correction has to depend on the pin power distribution. Deviations calculated with the CFD models are in good agreement with the deviations experienced in the nuclear power plant.

6th thesis

I have determined the weight factors of VVER-440 rod bundles' annular regions and the central tube for the in-core thermocouple. Using the weight factors and the outlet enthalpies of the subchannels and central tube, the in-core thermocouple signal can be predicted with a linear relation. [8][9]

I have investigated in details the coolant mixing in the head parts of VVER-440 fuel assemblies using numerical tracers. Based on the distribution of the numerical tracers, I have determined the weight factors of five rod bundle regions and the central tube for the in-core thermocouple. The analyses showed that the coolant flow from the five inner subchannel rings and flow from the central tube influence significantly the thermocouple signal. Using the weight factors and the outlet enthalpies of the subchannels and central tube, the in-core thermocouple signal can be predicted with a linear relation. The pin power dependent correction of the measured thermocouple signal can be determined from the predicted thermocouple signal and the outlet average temperature calculated from the heat balance. Using the correction, the accuracy of the core monitoring can be improved.

I have studied the sensitivity of the weight factors on the fuel assembly types (3.82% enrichment profiled assemblies with 12.2 mm and 12.3 mm pitch, 4.2% enrichment assembly with burnable poison) and on the pin power distributions. The analyses showed that the weight factors of various assemblies and weight factors of assemblies with different pin power distributions do not differ significantly in terms of the prediction of the thermocouple signal. The weight factors can be used for other assembly types and for assemblies with different pin power distributions within the bound of the investigation.

With assembly head calculations, I have pointed out that the orientation of the mixing grid influences significantly the thermocouple signal in the cases of fuel assemblies with declined pin power profiles. This causes an uncertainty in the in-core temperature measurement, which has not been taken into account until now. In the investigated case, the orientation causes 2.5% uncertainty in calculation of the assembly power. [9]

Based on my observations of fresh VVER-440 fuel assemblies, the orientation of the mixing grid is not consistent and all four orientations can be found randomly. For this reasons, I have studied the effects of the mixing grid orientation on the coolant mixing and the thermocouple signal in the case of a fuel assembly with declined pin power profile. The investigations showed that the orientation of the mixing grid has significant effect on the cross-sectional temperature distribution at the level of the thermocouple. Consequently, the grid orientation influences the thermocouple signal namely it influences the monitoring of the fuel assembly power during the cycle. In the investigated case, the orientation causes 0.8 °C (2.5%) uncertainty (the average heat up of the coolant is 32 °C) in the in-core temperature measurement. This means 2.5% uncertainty in the determination of assembly power. With fixing the orientation of the mixing gird, this uncertainty could be avoided.

Utilization of the results

Based on statistical investigations of the registered data of Paks Nuclear Power Plant Unit 4, the dependence of the deviations between the measured and calculated thermocouple signals on the fuel assembly type can be compensated with the weight factors. In the case of the "actual" profiled fuel assemblies with 12.3 mm pitch, the average of the deviations between measured and calculated signals decreases from 0.8 °C to -0.25 °C (from 2.4% to -0.7% of the average heat up). This is due to the use of the new model based on the weight factors instead of the old model based on the radiation heat up. Implementation of the weight factors in the VERONA core monitoring system of Paks Nuclear Power Plant is in progress.

Publications related to my Ph.D. thesis

[1] S. Tóth, A. Aszódi: CFD analysis of flow field in a triangular rod bundle, Nuclear Engineering and Design, 240, 352–363, 2010, ISSN 0029-5493, Elsevier

[2] S. Tóth, A. Aszódi: CFD analysis of flow field in a triangular rod bundle, Proc. 12th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, NURETH-12-175, Pittsburgh, PA, USA, 30 September–4 October 2007, ISBN 0-89448-058-8

[3] Tóth S., Aszódi A.: VVER-440 kazetta pálcakötegére vonatkozó CFD számítások eredményei, Magyar Energetika, 2007/5, 75–79, ISSN 1216-8599

[4] S. Tóth, A. Aszódi: CFD study on coolant mixing in VVER-440 fuel rod bundles and fuel assembly heads, Nuclear Engineering and Design, doi:10.1016/j.nucengdes.2009.11.022, ISSN 0029-5493, Elsevier

[5] S. Tóth, A. Aszódi: CFD study on coolant mixing in VVER-440 fuel assembly head, Proc. International Congress on Advances in Nuclear Power Plants, ICAPP '08-8278, Anaheim, CA, USA, 8–12 June 2008, ISBN 0-89448-061-8

[6] S. Tóth, A. Aszódi: CFD study on coolant mixing in VVER-440 fuel rod bundle and assembly head, Proc. Experiments and CFD code Applications to Nuclear Reactor Safety, MIX-05, OECD/NEA & IAEA Workshop, Grenoble, France, 10–12 September 2008

[7] Tóth S., Aszódi A.: VVER-440 kazettafej modell előzetes validációs számításai, Nukleon, I. évf., 3, 18, 2008, ISSN 1789-9613

[8] S. Tóth, A. Aszódi: Investigation of coolant mixing in head parts of VVER-440 fuel assemblies with burnable poison, Kerntechnik, 74, 265–272, 2009, ISSN 0932-3902, Carl Hanser Verlag

[9] S. Tóth, A. Aszódi: Detailed analysis of coolant mixing in VVER-440 fuel assembly heads, Proc. 18th Symposium of AER on VVER Reactor Physics and Reactor Safety, 413–425, Eger, Hungary, 6–10 October 2008, ISBN 978-963-372-637-2

Further publications

Conference papers

1. S. Tóth, A. Aszódi, G. Légrádi: CFD analysis of coolant flow in VVER-440 fuel assemblies with the code ANSYS CFX 10.0, Proc. 14th International Conference on Nuclear Engineering, ICONE14-89497, Miami, FL, USA, 17–20 July 2006, ISBN 0-7918-3783-1

2. S. Tóth, A. Aszódi: Detailed analysis of coolant flow in VVER-440 fuel rod bundle, Proc. 16th Symposium of AER on VVER Reactor Physics and Reactor Safety, 465–481, Bratislava, Slovakia, 25–29 September 2006, ISBN 978-963-372-633-4

3. S. Tóth, A. Aszódi: Calculations of coolant flow in a VVER-440 fuel bundle with the code ANSYS CFX 10.0, Proc. Workshop on Modeling and Measurements of Two-Phase Flows and Heat Transfer in Nuclear Fuel Assemblies, Stockholm, Sweden, 10–11 October 2006

4. Tóth S., Aszódi A.: VVER-440 kazetta pálcakötegére vonatkozó CFD számítások eredményei, V. Nukleáris Technikai Szimpózium kiadványa, 405, Paks, 2006. november 30.– december 1., ISBN 978-963-420-916-4

5. S. Tóth, A. Aszódi: Analysis of mixing processes in VVER-440 rod bundle with RANS method, Proc. 17th Symposium of AER on VVER Reactor Physics and Reactor Safety, 513–528, Yalta, Crimea, Ukraine, 23–29 September 2007, ISBN 978-963-372-634-1

6. S. Tóth, A. Aszódi: Preliminary validation of VVER-440 fuel assembly head CFD model, Proc. 17th Symposium of AER on VVER Reactor Physics and Reactor Safety, 499–512, Yalta, Crimea, Ukraine, 23–29 September 2007, ISBN 978-963-372-634-1

7. S. Tóth, A. Aszódi: Determination of weight factors for VVER-440 fuel assemblies with burnable poison, 19th Symposium of AER on VVER Reactor Physics and Reactor Safety, 483–496, Varna, Bulgaria, 21–25 September 2009, ISBN 978-963-372-640-2

Conference presentations

1. Tóth S., Aszódi A.: Gőzfejlesztő tápvízszabályozó szelep 3D numerikus modellezése, IV. Nukleáris Technikai Szimpózium, Budapest, 2005. december 1–2.

2. S. Tóth, A. Aszódi: CFD analysis of coolant flow in VVER-440 fuel assemblies with the code ANSYS CFX 10.0, AER Working Group C and G Joint Meeting, Balatonfüred, Hungary, 22–23 May 2006

3. S. Tóth, A. Aszódi: CFD analysis of flow field in a triangular rod bundle, AER Working Group C and G Joint Meeting, Ráckeve, Hungary, 17–18 May 2007

4. Tóth S., Aszódi A.: VVER-440 kazettafej modell előzetes validációs számításai, VI. Nukleáris Technikai Szimpózium, Budapest, 2007. november 29–30.

5. S. Tóth, A. Aszódi: CFD study on coolant mixing in VVER-440 fuel assembly head, AER Working Group C and G Joint Meeting, Siófok, Hungary, 29–30 May 2008

6. S. Tóth, A. Aszódi: CFD study on coolant mixing in VVER-440 fuel assembly head, IAEA Regional Workshop on Application of CFD Codes in Nuclear Safety, Budapest, Hungary, 17–20 June 2008

7. Tóth S., Aszódi A.: VVER-440 kazetta fejrészében kialakuló hűtőközeg-keveredés vizsgálata, VII. Nukleáris Technikai Szimpózium, Budapest, 2008. december 4–5.

8. S. Tóth, A. Aszódi: Coolant mixing in head parts of Gd fuel assemblies, AER Working Group C and G Joint Meeting, Tengelic, Hungary, 11–12 June 2009

Reports

1. Tóth S., Aszódi A.: Gőzfejlesztő tápvízszabályozó szelep 3D numerikus modellezése, Kutatási jelentés, BME NTI-310/2005, Budapest, 2005. november 15.

2. Tóth S., Aszódi A.: Aktív pálcaköteg hatvanfokos szegmensének modell leírása, Kutatási jelentés, BME NTI-323/2006, Budapest, 2006. február 20.

3. Tóth S., Aszódi A.: Növelt rácsosztású kazettára vonatkozó számítások eredményei, Kutatási jelentés, BME NTI-332/2006, Budapest, 2006. április 30.

4. Tóth S., Aszódi A.: Tápvízszabályozó szelep 108%-os blokkteljesítmény melletti üzemére vonatkozó CFD számítások, Kutatási jelentés, BME NTI-373/2006, Budapest, 2006. október 15.

5. Tóth S., Aszódi A.: VVER-440 kazettafej-modell ellenőrzése szubcsatornakódok által számított adatok felhasználásával, Kutatási jelentés, BME NTI-395/2007, Budapest, 2007. augusztus 30.

6. Tóth S., Aszódi A.: VVER-440 pálcaköteg-modell ellenőrzése, Kutatási jelentés, BME NTI-403/2007, Budapest, 2007. november 5.

7. Tóth S., Aszódi A.: VVER-440 kazettafej modell ellenőrzése a CFX kód által számított adatok felhasználásával, Kutatási jelentés, BME NTI-407/2007, Budapest, 2007. november 30.

8. Tóth S., Aszódi A.: 12,3 mm rácsosztású kazetta in-core termoelemre vonatkozó súlyfaktorainak meghatározása, érzékenységvizsgálatok, Kutatási jelentés, BME NTI-445/2008, Budapest, 2008. október 1.

9. Tóth S., Aszódi A.: 12,2 mm rácsosztású kazetta in-core termoelemre vonatkozó súlyfaktorainak meghatározása, Kutatási jelentés, BME NTI-455/2008, Budapest, 2008. december 1.

10. Tóth S., Aszódi A.: Gd kazetta in-core termoelemre vonatkozó súlyfaktorainak meghatározása, Kutatási jelentés, BME NTI-473/2009, Budapest, 2009. május 15.

11. Tóth S., Aszódi A.: A COBRA kód kilépő hőmérsékleteloszlásának ellenőrzése a CFX kód által számított adatok felhasználásával, Kutatási jelentés, BME NTI-480/2009, Budapest, 2009. június 30.

References

- Trupp, A.C., Azad, R.S., 1975, The structure of turbulent flow in triangular array rod bundles, Nuclear Engineering and Design, 32, 47–84

- Vonka, V., 1988, Measurement of secondary flow vortices in a rod bundle, Nuclear Engineering and Design, 106, 191–207

- Kobzar, L.L., Oleksyuk, D.A., 2006, Experiments on simulation of coolant mixing in fuel assembly head and core exit channel of VVER-440 reactor, Proc. 16th Symposium of AER on VVER Reactor Physics and Reactor Safety, 95–117, Bratislava, Slovakia

- Gango, P., 1997, Application of CFD models for 3D analysis of single phase thermal-hydraulics in the VVER-440 fuel assembly, Proc. 7th Symposium of AER on VVER Reactor Physics and Reactor Safety, 967–988, Hörnitz, Germany

- Rautaheimo, P., Salminen, E., Siikonen, T., Hyvärinen, J., 1999, Turbulent mixing between VVER-440 fuel bundle subchannels: a CFD study, Proc. 9th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, San Francisco, CA, USA

- Aszódi A., Légrádi G., 2003, Detailed CFD analysis of coolant mixing in VVER-440 fuel assemblies with the code CFX-5.5, Proc. 10th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Seoul, Korea

- Mayer G., Páles J., Házi G., 2007, Large eddy simulation of subchannels using the lattice Boltzmann method, Annals of Nuclear Energy, 34, 140–149

- Légrádi G., Aszódi A., 2003, Detailed CFD analysis of coolant mixing in VVER-440 fuel assembly heads performed with the code CFX-5.5, Proc. 13th Symposium of AER on VVER Reactor Physics and Reactor Safety, 773–790, Dresden, German

- Toppila, T., Lestinen, V., Siltanen, P., 2004, CFD simulation of coolant mixing inside the fuel assembly top nozzle and core exit channel of a VVER-440 reactor, Proc. 14th Symposium of AER on VVER Reactor Physics and Reactor Safety, 567–579, Helsinki, Finland

- Petényi, V., Klučárová, K., Remiš, J., Chapčák, V., 2003, Fuel assembly outlet temperature profile influence on core by-pass flow and power distribution determination in VVER-440 reactors, Proc. 13th Symposium of AER on VVER Reactor Physics and Reactor Safety, 695–710, Dresden, German