ATHLET/KIKO3D results of the OECD/NEA benchmark for coupled codes on KALININ-3 NPP measured data

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ABSTRACT

The paper presents an important part of the results of the OECD/NEA benchmark transient ‘Switching off one main circulation pump at nominal power’ analyzed by the coupled system code ATHLET-KIKO3D. The benchmark includes a set of input data for the NPP Kalinin-3 (VVER-1000) and consists of three exercises such as point kinetics plant simulation, boundary condition problem for coupled 3-D neutronics/core  T-H response evaluation and a best-estimate coupled code plant transient modeling. Some observations and comparisons with measured data for integral reactor parameters are discussed. Special attention is paid on the modelling and comparisons performed for the control rod movement and the reactor power history. Of primary interest are the comparisons done for the local in-core parameters –SPND signals at 7 layers of some fuel assemblies. An important step is done for the future performing of uncertainty analysis in the frame of the OECD/NEA activities.

1 INTRODUCTION

As a logical continuation of the former benchmark activity of AER devoted to the VVER reactor physics [1-2], now there is a rather large interest of the Kalinin-3 benchmark recently completed and published by the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD) [3]. The goals of these exercises were the evaluation of the prediction capability of advanced code systems by means of a code to code comparison. Meanwhile analyzing a real plant transient is good to verify the prediction capability of best estimate codes. These code systems are characterized by direct coupling between the one dimensional thermal-hydraulics plant models with three dimensional neutron-kinetics codes.

The selected transient „Switching-off of one Main Circulation Pump (MCP)” starts from the nominal power and leads to asymmetric core conditions. In Phase 1, the system thermal-hydraulic characteristics and modeling of the plant regulation were investigated without using the complicated 3D reactor physics model in order to clarify the ATHLET thermal hydraulics capability. In order to decrease the uncertainties coming from modelling of the primary and secondary NPP sides a boundary condition problem is defined on the base of
the measured data, in the second exercise. Full simulation of the reactor with coupled best estimate code using 3D reactor physics model (analysis of the transient in its entirety) completes the preparation to start the investigation of the different types of uncertainties of the input parameters.

The results and the discussions presented in this paper are related to the inter-comparison of the simulation accuracy of the standalone and coupled code system ATHLET, ATHLET-KIKO3D [4] within the activities of the Kalinin-3 transient benchmark. It is also a step forward in revealing some sources of model uncertainties and the need of some new interpretation of the available measured data. Additionally, the need of complimentary studies is determined. Some preliminary results, model developments and peculiarities by simulation the transient have been reported in details in [5-7]. The aim of this work is to compare mainly the available in-core measured thermal-hydraulic and neutron-physical core local (point) data with the simulated one in a case of a thermal-hydraulic asymmetry. These comparisons are of high importance by the validation of the coupled system code ATHLET-KIKO3D. They prove the influence and the importance of the interaction of local neutron-physical and thermal-hydraulic feedback parameters, which is in fact the main advantage of the „best estimate” 3D coupled code prediction capabilities compared to the point kinetics models.

In the paper the solution of the three exercises are outlined using the stand alone ATHLET and the coupled ATHLET/KIKO3D code complex. The appropriate thermo hydraulic and core input models of the transient are presented in the paper. The results calculated by different codes are compared to the documented experimental data.

2 DESCRIPTION OF THE BENCHMARK SCENARIO

The transient was measured during the first cycle of the Unit 3, NPP Kalinin. The original fuel loading of the core consists of five types of assemblies. A sixty degree symmetrical loading was developed from these assemblies, but at the 96th effective day a defected assembly with coordinates 07-32 had to be replaced by a “fresh” standard FA with U235-enrichment of 1.6%. The spacers and the leading tubes of these FAs were made of stainless steel. The fuel loading map in the reactor core of Unit 3 NPP Kalinin after the replacement of the defected assembly by a standard one is demonstrated in Figure 1.

Fig.1: Core loading with the assembly which unmakes the 60 degree symmetry of the loading
Main scenario sequences recovered from the measured data histories can be systematized so [3]:

- Manually switching-off MCP number 1 at t=0 s,
- The signal ‘one pump out of operation’ is generated at 1.41 s, reactor limiting controller starts to decrease the power to a level of 67.2 %
- The following sequence of actuations for reactor limiting controller and automatic reactor power controller is recorded:
  - At t=1.41 s the reactor limiting controller starts to decrease the reactor power. Control rod bank (CRB) #10 starts to move downwards. When the CRB #10 reaches 50 % insertion depth (at about 60 s) the CRB #9 also starts to enter the active core according to the control rod movement algorithm.
  - Protection system level #1 of the automatic reactor power controller switches from option ‘T’ (keeping the secondary loops’ parameter constant) to option ‘H’ (keeping neutron power constant)
  - Control rod controller decouples from automatic reactor power controller.
- At t=71 s the reactor power load-off procedure is finished and power reaches a level of 67.2 % $P_{\text{nom}}$. At this moment the position of the CRB #10 is at 43.4 % and remains there till the end of the transient. CRB #9 is inserted into the core and reaches at 71 s the position of 93.1 % and stays there till 180 s. After that, it returns back to 100 %. The automatic reactor power controller is again switched on to the control rod controller with option ‘H’ and it starts to keep the power level in the range of 66.2 -67.3 % $P_{\text{nom}}$.

In connection of CRBs, the further information is important:

- At the beginning of the transient the CRB #9 is fully withdrawn from the core, its position is 362 cm (104% by the measurements), as the upper control rods’ end switches are located at 14 cm above the active core. It means that the real insertion of the CRB #9 started by a delay of 7.29 s [7]. See Fig. 4.
- According to the measurement system established at the NPP, the positions of CRB are given with respect to the position of the lower end switches. They are located 17.25 cm higher than the bottom of the reactor core. The length of the reactor core is 355 cm and the distance between the lower and the upper end switches is 352 cm. It corresponds to the 100% insertion of CRB.

3 THE ATHLET/KIKO3D COUPLED CODE SYSTEM

The coupling of ATHLET with the spatial kinetics code KIKO3D extends its application to a wide range of VVER plant transients. This coupled code complex has been routinely used to study safety issues of VVER-440, to perform best-estimate analysis covering core thermal hydraulics, reactor physics and plant dynamics [8-9]. Though the VVER-1000 differs in many details form the cases investigated earlier, it is not brought about changes in the program, but new types of input data set had to be generated.

The aim of this work is to compare mainly the available in-core measured thermal-hydraulic and neutron-physical core local (point) data with the coupled code results in case of a thermal-hydraulic asymmetry. These comparisons are of high importance for the validation of the coupled system code as they show the correctness of the applied methodology.

3.1 ATHLET input

The ATHLET nodalization of the Kalinin-3 NPP VVER-1000 reactor primary with the pressure vessel, cold and hot legs and the steam generator is developed. The four loops are
modeled separately. The vessel is divided azimuthally into 6 sectors for the appropriate modeling of the asymmetry induced by the switching of the pump in loop No. 1.

Between the nodes connections in the down comer, lower and upper plenum with their real geometry data are applied for the further investigation of the mixing and its influence on the temperatures at the measured positions. The pipes for the heat transfer in the steam generator are modeled in seven horizontal bundles.

In the first exercise when the ATHLET reactor point model was used, seven average fuel assemblies were applied corresponding to the six symmetrical sectors and a central one. Concerning the core, the lower and upper plenum, there are additional regions in the center.

In the second exercise, the inlet condition of core bottom and some outlet condition were prescribed on the basis of a detailed calculation made by ATHLET best estimate code using the measured flow data. That is why the detailed plant model used in the first exercise was converted to a simplified one. It consists of 163 separate thermo-hydraulic channels with the upper plenum. Each channel represents the flow condition of a given assembly wherever located in the core. No coolant mixing between these channels are considered as the mass flow rate is rather high during the transient progression. In the aspect of the core also this detailed core model was used in the third exercise, too. The other part of ATHLET input corresponds to that one, used in the stand alone calculation.

In all ATHLET input cases, the assemblies are divided in ten axial nodes of the same height. The fuel rod gap conductance as well as the material properties was taken from the final specification.

3.2 KIKO3D input

The data of the power load (necessary to calculate the fuel burnup) for Unit 3 NPP Kalinin from the beginning of the first fuel cycle up to the day when the experiment with the switching off of one MCP took place, was provided separately. In the benchmark specification the characteristics of the fuel assemblies for the KALININ-3 core are extensively described. Concerning the original sixty degree rotational symmetry of the core, it consists of 29 fuel assembly types, each one with unique axial material composition. Axially the fuel parts of the core are divided in 10 layers. There are reflector nodes in radial direction in all elevation and below and above all of the assemblies. There are 283 unrodded compositions. Taking into account the two level structure of the control rod: one contains Dysprosium-Titanium and the other mainly B\textsubscript{4}C absorber, there are further 2 times 110 rodded compositions. For the asymmetric sector a detached data set was developed, too. A complete set of diffusion type cross sections as a function of moderator temperature, density and fuel temperature were calculated by the HELIOS 1.9 code for each composition at GRS.

Originally the KIKO3D reactor kinetic code calculation is based on response matrix library which had to be modified for the above depicted table given for the benchmark. Based on the actual reactor conditions the appropriate cross sections are obtained from the look-up tables using a linear interpolation scheme. To evaluate the actual cross section for the bottom reflector a coolant temperature equal to the inlet coolant temperature and a coolant density equal to the inlet coolant density are used. The same procedure was used for the reflector on the top and for radial reflector nodes.

Our modified calculation tool was verified by comparison of our results to other ones. The benchmark team recommended the hot zero power (HZP) state for that purpose. Steady state calculations were performed using the following parameters:

- 0.1% of nominal power
- Moderator and fuel temperature: 552.15 K
- Moderator density: 767.1 kg/m\textsuperscript{3}
• Boron concentration $C_{b}=660$ ppm or 3.6 g/kgH2O
• CRB #10 is 302 cm according to the measurement

Calculations were done for the asymmetrically loaded core at HZP and hot full power case (HFP), too. The results presented here are preliminary results from the working-materials of the 4th Workshop of OECD Kalinin-3 Benchmark, held in Karlsruhe, Germany in May 2013.

Table 1: Comparison of the eigenvalue of different calculations table

<table>
<thead>
<tr>
<th>Code [reference]</th>
<th>HFP $K_{eff}$[-]</th>
<th>HZP $K_{eff}$[-]</th>
</tr>
</thead>
<tbody>
<tr>
<td>PARCS [10]</td>
<td>0.99601</td>
<td>1.01098</td>
</tr>
<tr>
<td>DYN3D [11]</td>
<td>0.99700</td>
<td>1.01131</td>
</tr>
<tr>
<td>ATHLET KIKO3D</td>
<td>0.99290</td>
<td>1.00770</td>
</tr>
</tbody>
</table>

From Table 1, one can see that rather similar results were given by the different codes in hot/zero power cases. The frozen xenon concentration corresponds to the full power case and the boron concentration which is the critical value of the full power case give explanation for the supercritical eigenvalue.

4 RESULTS, COMPARISONS AND DISCUSSIONS

All the three exercises were solved by our code system. Due to the point reactor model in the first exercises only some limited results could be compared to the measurements. The following circumstances render mode difficult these tasks:

• In the figures below, the point model solution is denoted by “c_phase1”, and the others, the coupled calculations by “c_phase2” and “c_phase3”.
• In order to compare the calculated temperatures with the measured data the inertia of the heat transfer process of the measuring temperature sensors has to be also taken into our modelling. Such model is not reported in the benchmark specification however its necessity is known [3].
• Even the positions of the Self-Powered Neutron Detectors (SPND) were reported, further information is needed about the quality and accuracy of the measured SPND data. Generally, for the accurate modeling of SPND signal information is needed from the SPND burnup, induced cable current effects, SPND current-power transformation procedure, power used for normalization, etc.

4.1 ATHLET with point kinetic

In case of using point kinetics, the reactivity coefficients and the control rod reactivity were applied according to the benchmark specification. The axial positions of working groups #9 and #10 are controlled according to the plant logics, which was simulated by the ATHLET GCSM module. In this case, the most important phenomenon is the pump run-out which can be characterized by the pump pressure drops and the primary loop flow rates. On the basis of the measured and calculated pump pressure drops and flow rates, the rated hydraulic torque was set to 30000 Nm in order to obtain a relatively good agreement of the mentioned parameters, as it is shown in Figure 2.

4.2 ATHLET KIKO3D Coupled calculations

The whole plant transient was investigated in two steps by the ATHLET-KIKO3D coupled code system. In the first one, only the core was simulated as a boundary condition problem in order to decrease the sources of uncertainties eliminating the influence of the NPP

Proceedings of the International Conference Nuclear Energy for New Europe, Portorož, Slovenia, September 8-11, 2014
secondary side modelling then the full system was modelled. In this exercise the inlet condition of core bottom and some outlet condition were prescribed on the basis of a detailed calculation made by ATHLET-BIPR-VVER best estimate code. Compared and analyzed are mainly the local in-core parameters – assembly outlet coolant temperatures at 93 measured points and SPND powers at 7 layers of 64 fuel assemblies, in that case. The initial steady state calculation can be characterized by the following parameters:

- The thermal power: 2962 MW
- Boron concentration $C_b=660$ ppm
- Height of CRB #10 is 302 cm, according to the measurement

For the initial steady state the $K_{ef}$ was 0.9923 in both input set. It has to be mentioned that all participants using the PSU/REL library slightly underestimate the steady state eigenvalue.

In the last phase the same ATHLET input was used for the plant as in the point model calculation excluding the core.

![Mass flow rate in the loops](image)

**Fig. 2:** Measured and calculated mass flow rate in the four loops

### 4.3 Comparison of the global parameters

Figure 3 shows the predicted integral power evolution during the transient compared to the measured one, once on the basis of the ex-core Fission Ionization Chambers (FIC) and another one on the basis of the in-core self-powered neutron detectors (SPND). At the beginning the power is reduced by the insertion of the control groups 10 and 9 (Fig. 4) under the algorithm for fulfilling the safety requirements of the NPP operation with one switched off MCP. After reaching the minimum power level at about 70 s the power starts to increase within about 2-3% after which it stabilizes at a level of 68.7%. This effect of power increase is due to the colder coolant flow that enters the core and so introduces a positive density reactivity effect. The analysis of the curves in Fig. 3 shows that: After the 70-th sec of the transient there is a difference of the measured power evolution by the two systems - ex-core and in-core. The SPND based power (restored from neutron flux sensors located in 64 assemblies in 7 layers) ‘catches’ the power increase but based on FIC power does not ‘notice’ it. The SPND power differs in absolute value in comparison with the FIC, the reason can be the normalization procedure of the SPND readings. The ATHLET-KIKO3D coupled calculations fit to the measurement and depicts correctly the slight power return, too.
**Fig. 3:** Comparison of calculated total reactor power evolution with measured by the FIC and SPND systems.

**Fig. 4:** Comparison of predicted CRB #9 & #10 insertion histories with the measured data.

**Fig. 5:** Comparison of predicted and measured hot legs’ coolant temperatures.
As another example of comparison of global parameters are presented in Figs. 5-6. The time histories of the predicted coolant temperatures fit the measurements but the lack of the inertia model for the measurements can be seen.

4.4 Comparison of local parameters

In case of a coupled calculation the comparisons of the measured data of the thermocouples at the core exit and the data of SPND detectors with the results of the simulation are particularly interesting.
For all of them measured data are available with a time resolution of 1 s. Concerning the measured and calculated outlet temperature a rather large discrepancy can be seen on Fig. 7. Its reason was discussed in [3-5] a very detailed manner. As in our calculations there are no sub channel model for guide tubes their outlet temperatures are not known and the coolant mixing cannot be simulated which is necessary for the appropriate simulation of the measurement.

The SPND sensors recorded the power decrease and the rod banks’ insertion. Practically a small number of SPND were out of order during the experiment and have been excluded from our data processing and any comparisons. Examples of comparisons predicted with the ATHLET-KIKO3D system code local powers with the SPND measurements are shown in the following figures. Figures 8-9 show the comparison of the axial power distributions before and after the transient and the time when the CR insertions were significant (T=45 s) for assemblies 08-25 and 12-25.

Fig. 8: Comparison of axial power distributions for assembly 08-25 at T= 0.0 / 45.0 and 300s

Fig. 9: Comparison of axial power distributions for assembly 12-25 at T= 0.0 / 45.0 and 300s
The first observations with regard to Figures 7-9 are the good correspondence between the two types of coupled calculations. Figure 8-9 also indicate that the ATHLET-KIKO3D code can predict the local power value quite well. To quantify the accuracy an additional statistical analysis has been performed on the results of the third phase.

The main results of the performed statistical analysis are summarized in Tables 2 and in Fig. 10-12. The comparisons are made between the measured SPND data and those predicted by the system code ATHLET-KIKO3D without any corrections.

The relative deviations (RD) and standard deviations (SD) estimated with the formulas 1 and 2 are being calculated for the whole set of available measured points (64 assemblies with 7 layers for 300 s).

\[
RD = \frac{\sum_{k=1}^{n} (\frac{x_{k,\text{exp}} - x_{k,\text{calc}}}{x_{k,\text{exp}}})}{n} \quad [1]
\]

\[
SD = \sqrt{\frac{\sum_{k=1}^{n} (\frac{x_{k,\text{exp}} - x_{k,\text{calc}}}{x_{k,\text{exp}}})^2}{n}} \quad [2]
\]

**Table 2** Layerwise RD and SD for all SPND readings at all-time points

<table>
<thead>
<tr>
<th>LAYER</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
<th>TOTAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>DEVIATION,%</td>
<td>-3.39</td>
<td>-4.14</td>
<td>-2.28</td>
<td>-0.98</td>
<td>0.75</td>
<td>5.05</td>
<td>14.81</td>
<td>1.40</td>
</tr>
<tr>
<td>SD %</td>
<td>7.61</td>
<td>6.57</td>
<td>3.75</td>
<td>3.73</td>
<td>6.41</td>
<td>8.51</td>
<td>17.95</td>
<td>8.98</td>
</tr>
</tbody>
</table>

The achieved results for the comparison in all measured points for all transient time steps are summarized in Table 2. The total SD has a value of 9.0% and the maximum RD is in the range of -4.14% to +14.8%.

Figures 10-12 show the differences of measured and calculated detector readings during the whole transient for detectors located at different axial core layers (1 –bottom, 7-top) for two different assemblies. The simulation is particularly interesting for assembly 08-25 (Fig. 10) which is located near the CRB #10.

Fig. 12 shows the time history of the RD for all assemblies with SPND sensors. The maximum RD is in the range of -8% to +16%.

From the analysis of these curves it can be observed that before the transient initiation RD is within the range of -3% to +12% and after the moment of CR bank #10 insertion stop (t > 70 s) the RD stabilizes in the range of -5 to +15%. Almost for all assemblies a negative RD is observed for the first core layer of SPND locations (bottom part of the core) and a large positive RD is denoted in the last (upper) core layer (7) of SPNDs.

**Fig. 10**: Comparison of normalised power histories of assembly 08-25 for four axial SPND location layers (near CRB #10)
That shows that the calculated power distributions are overpredicted at the bottom part of the core and underpredicted it in the upper part. Similar results for other NPPs with VVER-1000 reactor are reported in [12].

Our results indicate that the implementation of some correction factor in the form of an axial shift of the SPND locations could be necessary. However it is not clear and still not yet proved whether it is a measurement or a simulation problem.

The explanation of the observed differences is not a simple task and is an object of our future detailed work connected with performing systematic uncertainty and sensitivity studies and analysis. According to our experience and first estimations the differences may be attributed to the following reasons and considerations:

1. Small number of axial nodes (10 axial layers) of the active core model, meanwhile the value for the power comparison at the axial points where the SPNDs are located is achieved through spline approximations.
2. The thermal-hydraulic parameters for the neutronic model are influenced by a large number of parameters and model descriptions of the whole core and also of
approximations made by modeling of separate local effects, as for example the flow mixing phenomena in the reactor pressure vessel.

3. Correct generation and homogenization of the nuclear cross section data and its uncertainties and correctly defining the fuel burnup in each assembly at the time moment when the measurements have been performed.

4. Correctly taking into account of the radial distribution of the coolant parameters (approximately 7% of the assembly cross section water area is the ‘cold’ water located in the control rod guide tubes).

5. How exactly can the position of each SPND in each assembly at each axial layer be defined.

6. More information is needed about the quality and accuracy of the measured SPND data.

5 CONCLUSIONS

First step in the verification and validation of the coupled ATHLET & KIKO3D code for VVER-1000 plant has been performed. All calculations done till now prove the successful functioning of coupled code system and demonstrate physical plausibility of the simulations. The coupled system code is intended to be used in the future analysis for VVER reactors. The reliable prediction of the global and local reactor parameters is important issue to achieve a high validation stage of the coupled neutron-physics/thermal-hydraulics system codes which nowadays are considered to be the state of art by performing of safety analysis.

Performed is analysis of the OECD/NEA benchmark transient switching off one MCP by nominal power. Compared and analysed are mainly the local in-core parameters – assembly outlet coolant temperatures at 93 measured points and SPND powers at 7 layers of 64 fuel assemblies. Revealed are some sources of inaccuracy and in that way an important step is done for future performing of uncertainty and sensitivity analysis in the frame of the OECD/NEA activities. The analysis of the simulations performed has proved that the coupled system code ATHLET/KIKO3D allow predicting of global and local parameters with a rather good accuracy. The applied methodology by performing coupled analysis proved its efficiency and accuracy and is an important step towards the overall validation of the coupled system code ATHLET-KIKO3D on real plant measured data.

REFERENCES


