Simulation of the OECD NEA Kalinin-3 benchmark (phase 2) by the
ATHLET-KIKO3D code system

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1. INTRODUCTION

As a logical continuation of the former benchmark activity of AER devoted to the VVER reactor physics [1-3], 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) [4].

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 the last task, analyzing a real plant transient, is good to verify the prediction capability of best estimate coupled codes. These code systems are characterized by direct coupling between the one dimensional thermal hydraulics plant models with three dimensional neutron kinetics models. Among others ATHLET-QUABOX/CUBBOX [5] ATHLET-DYN3D [6] TRAC-PARCS [7] CATHARE-CRONOS-FLICA4 [8] and the Hungarian ATHLET-KIKO3D [9] belong to this group.

The new, KALININ-3 transient „Switching-off of one Main Circulation Pump (MCP)” starts from the nominal power at a MOC burnup condition. During the transient the coolant flow is rearranged in the reactor pressure vessel and a spatially dependent coolant temperature change can be observed, meanwhile the limiting controller starts to decrease the reactor power by insertion of control group number 10 and 9.

At the beginning of the transient an asymmetric core condition can be observed which eventually leads to symmetric core behavior at lower power level. Broad ranges of the parameter changes can be observed. The experiment is very well documented, their uncertainties are known and the measurements were performed with a quite high frequency [4]. These facts allow applying the studied transient not only for validation purposes but also for uncertainty analysis as a part of the NEA/OECD LWR Uncertainty Analysis. The benchmark consists of four separate exercises as follows:

1. **Exercise**: 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 capability of thermal hydraulics. In our former paper, the ATHLET model using point reactor kinetics was used for the solution of this exercise. The results were compared to documented experimental data. Rather good agreement was observed [10].

2. **Exercise**: The spatial reactor kinetics simulation with given time dependent core boundary conditions, such as insertion of control groups inlet temperature and flow as a time dependent function are used. The appropriate data was distributed by GRS.

3. **Exercise**: full simulation of the reactor with coupled best estimate code using 3D reactor physics model (analysis of the transient in its entirety).

4. **Exercise**: The different types of uncertainties of the input parameters will be investigated.
In the paper the solution of the second exercise is outlined using the coupled ATHLET/KIKO3D code complex and the prescribed time dependent- boundary conditions (coolant inlet of assemblies and control rod positions as a function of time). New ATHLET input model and some code development concerning the KIKO3D code will be presented. The results are compared to some other solutions and the documented experimental data.

2. THE CORE LOADING AND THE TRANSIENT

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:

1. 48 FA with $U_{235}$ enrichment of 1.3 % (all pins has the same enrichment);
2. 42 FA with $U_{235}$ enrichment of 2.2 % (all pins has the same enrichment);
3. 37 FA with average $U_{235}$-enrichment of 2.98 % (303 pins with 3 % and 9 gadolinium fuel pins with 2.4 %-enrichment);
4. 24 radial profiled FA with average $U_{235}$-enrichment of 3.9 % (243 pins with 4 %, 60 pins with 3.6 % and 9 gadolinium pins with 3.3 %-enrichment);
5. 12 radial profiled FA with average $U_{235}$-enrichment of 3.9 % (240 fuel pins with 4 %, 66 fuel rods with 3.6 % and 6 gadolinium fuel rods with 3.3 %-enrichment).

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 $U_{235}$-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. 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.

Main scenario sequences recovered from the measured data histories can be systematized so [4]:

- Manually switching-off MCP number 1 at t=0s.
- The signal ‘one pump out of operation’ is generated at 1.41 s.
- 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 % $P_{nom}$. 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 % $P_{nom}$.
In connection of Control Bank, the further information is important:

- At the beginning of the transient the control group 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 CR 9 started by a delay of 7.29 s \[11\]

- According to the measurement system established at the NPP, the positions of Control Bank 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 of 352 cm. It corresponds to the 100% insertion of control rod.

The Exercise 2 focuses on the 3D core behavior and -for the easier comparison- the boundary conditions for the core are provided by the benchmark team.

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 successfully used for more than 12 years to examine security issues of VVER, to perform best-estimate analysis covering core thermal hydraulics, reactor physics and plant dynamics [9], [10].

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. That is why the detailed core model used in the first exercise was converted to a simplified one. It consists of 163 separate thermo-hydraulic channels with the upper plenum (see in Figure 3). 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 rather high during the transient progression. 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.

In the benchmark specification the characteristics of the fuel assemblies for the KALININ-3 core are extensively described. Concerning the original sixty degree turning 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 divided 10 layers. There are reflector nodes in radial direction in all elevation and below and above all the assembly. There are 283 unrodded compositions. Take into account the two level structure of the control rod: one contains Dysprosium-Titanium and the other mainly B\(_4\)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 coefficients macroscopic cross section for scattering, absorption and fission as a function of moderator temperature and density and fuel temperature were calculated by HELIOS 1.9 code for each composition. The kappas, the group inverse neutron velocities, decay constants and betas are also provided for each composition.

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 total cross sections are obtained from the look-up tables using a linear interpolation scheme. To predict the actual cross section for
the bottom reflector a fuel temperature equal to the inlet coolant temperature and a coolant
density equal to the inlet coolant density are used. The same procedure used for the reflector
on the top and for radial reflector nodes.

4. THE STADY-STATE CALCULATIONS

Our modified calculation tool was verified by comparison our results to other one. The
benchmark team recommended the hot zero power 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/m3
- Boron concentration Cb=660 ppm or 3.6 g/kgH20
- Control rod group #10 is 82.95 % (302 cm according to the measurement) inserted
  from below
- the PSU/REL library was used

Calculations were done for the symmetrical case, where the newly loaded assembly was not
taken into account, and the asymmetrical case, too. In the second column of Table 1 it is sign
as “sym” or “asym”. The results presented here are preliminary result from the working
materials of the 4th Workshop of OECD Kalinin-3 Benchmark, held in Karlsruhe, Germany
in May [16-18].

<table>
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<tr>
<th>Code References</th>
<th>Core</th>
<th>Keff [-]</th>
<th>SCRM [%]</th>
<th>CR No 9 [pcm]</th>
<th>CR No 10 [pcm]</th>
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</table>

From Table 1 one can see that rather similar results were given by the different codes in hot
zero power cases. In that state the symmetrical case slightly differs from the asymmetrical
one. 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. It is slightly smaller for the asymmetrical core, as the enrichment
of the fresh fuel is smaller and the burnup level is not too high of the core.
Comparison of the radial power distributions calculated by different codes for the
symmetrical case can be seen on Figure 4. The agreement is acceptable. Figure 5 shows, how
much perturbation caused in the power distribution the insertion of a fresh low enriched fuel
assembly into the core.

The control group worth as a function of insertion can be seen on figure 6 (group 10) and 7
(group 9). Good agreement can be seen between the PARCS and KIKO3D calculation. It is
worth to see, that the control rod behavior practically does not depend from the core symmetry.
One dimensional axial power distribution for different state \((H_{10} = 302 / 352 \text{ cm})\) can be seen on Figure 9 and 10. The agreement is similar the above mentioned cases.

The initial state of the transient was investigated, too. In the Hot Power State only the asymmetric core loading was calculated. The efficiency of control rod group number 9 and 10 can be seen on Figure 11 and 12. The calculations made by KIKO3D and PARS show good agreement. The integral worth of rod groups is summarized in Figure 13.

5. SIMULATION OF THE TRANSIENT

The initial steady state calculation can be characterized by the following parameters:

- The thermal power: \(2962 \text{ Mw}\)
- Average fuel temperature: \(609.1 \text{ C}\)
- Average moderator temperature: \(305.4 \text{ C}\)
- Average moderator density: \(714.2 \text{ kg/m}^3\)
- Boron concentration \(C_b=660 \text{ ppm}\)
- Height of control group number 10 is 302 cm, according to the measurement
- the PSU/REL library was used
- \(K_{\text{eff}}\) for steady state case: \(0.99736\)

The radial power distribution at the beginning of the transient and the one dimension axial power distribution in some different time points can be seen on Figure 15 and 14. It has to be mention that all participants using the PSU/REL library slightly underestimate the steady state eigenvalue.

The time dependent boundary conditions are shown on Figure 16 and 17. Instead of presenting the all the time dependent flow data, nine of them signed by gray hexagon on Figure 15 were chosen to represent the transient.

The measured and calculated time dependent power can be seen on Figure 18. One can see that the final power state slightly underestimated. In Figure 19 a measured and calculated pressure is presented. The reason of the difference between the two curves is the fact, that the position of the simulation was not perfect. Figure 20 -22 show the average parameters of the core, during the transient. These figures well characterize the slight perturbation of the core and are good for further inter-comparisons. The power and temperature outlet for the chosen assemblies can be seen on Figure 23-24. One can see the effect of the temperature perturbation one side and the power decrease due to control group insertion on the other side. Some of the assemblies (number 26, 36, 37 and 52) are the vicinity of the MCP 1. Two assemblies number 52 112 belong to the control group 10. The smallest outlet temperature belongs to the fresh assembly (number 97).

Concerning the measured and calculated outlet temperature a rather large discrepancy can be seen on Figure 25. Its reason was discussed in [11-13] a very detailed manner. That is why the \(K_q\) were evaluated from the calculated temperature rises. The measured and evaluated \(K_q\) as a function of time show rather good agreement (see Figure 26 and 27).
6. CONCLUSIONS

The second phase of the KALININ-3 benchmark was analyzed with the ATHLET-KIKO3D code complex. Appropriate ATHLET nodalization was developed for this task. In case of spatial reactor kinetic code, the PSU cross section library was implemented into the KIKO3D code. The code to code comparison revealed that the results prepared by the ATHLET-KIKO3D results are in good agreement with most of the trends predicted by similar codes. The reactivity values of different control rod groups and power distribution at HZP and HFP were used to prove that the given cross section library works properly.

Some comparison of the result obtained for the KALININ 3 benchmark transient with the appropriate measurements clearly shown the efficiency of the coupled code.

REFERENCES


[16] Yann Perin, OECD KALININ-3 BENCHMARK: NEUTRONIC CALCULATIONS OF EXERCISES 3A AND 3B WITH PARCS

[17] Jan Hádek, R. Meca, Progress in Solution of Kalinin-3 Benchmark Exercise 1 and 2

[18] Emiliya Georgieva, K. Ivanov, Status of PSU/REL Benchmark Cross Section Library
Figure 1  Final Core loading with the assembly (signed by letter N) which unmake the 60 degree symmetry of the loading

Figure 2  Arrangement of the control rod groups within the core
Figure 3  ATHLET nodalisation for exercise 2

Figure 4  Radial power distribution at HZP state for the symmetrical core calculated by ATHLET-KIKO3D (bellow) with comparison to PARCS results (above)
Figure 5  Assembly wise normalised power distribution at HZP calculated by ATHLET-KIKO3D. The fresh assembly and its 60 degree symmetrical counterparts are signed by grey HZP Control Rods Worth (CR09).

Figure 6  Control rod worth calculated by ATHLET-KIKO3D and PARCS for group 9 at HZP state.
Figure 7  Control rod worth calculated by ATHLET-KIKO3D and PARCS for group 10 at HZP state

Figure 8  Integral control rod worth calculated by ATHLET-KIKO3D for all groups at HZP case
Figure 9  Axial power distribution calculated by ATHLET-KIKO3D, PARCS and some coupled codes for HZP case $H_{10}=302$ cm

Figure 10  Axial power distribution calculated by ATHLET-KIKO3D, ATHLET-DYN3D for HZP case $H_{10}=352$ cm
Figure 11 - Control rod worth calculated by ATHLET-KIKO3D and PARCS for group 10 at HFP state

Figure 12 - Control rod worth calculated by ATHLET-KIKO3D and PARCS for group 10 at HFP state
Figure 13  Integral worth of Control rod groups calculated by ATHLET-KIKO3D at HFP state

Figure 14  One dimension axial power distribution during the transient
Fig. 15  Assembly wise normalised power distribution at HFP calculated by ATHLET-KIKO3D. The fresh assembly and those which were chosen to present the transient signed by grey color.
Figure 16  Position of the control rod group during the transient

Figure 17  Inlet temperatures of the chosen assemblies as a function of time
Figure 18 Comparison of the measured and calculated core power

Figure 19 Comparison of the measured and calculated pressure above the core
Figure 20 The core average density during the transient

Figure 21 The core average fuel temperature during the transient
Figure 22 The core average coolant temperature during the transient

![Average coolant temperature in the core](image1)

Figure 23 The calculated relative power of the chosen during the transient

![Assemblies relative power](image2)
Figure 24 The calculated outlet temperature of the chosen during the transient

Figure 25 Comparison of the measured and calculated temperature outlet for a given assembly
Figure 26 Comparison of the measured and calculated Kq for one of the chosen assembly during the transient.

Figure 27 Comparison of the measured and calculated Kq for one of the chosen assembly during the transient.