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Our Experiences with the Benchmark "Rostov-2" - Neutron Physical Modeling of the Initial Reactor State -

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ABSTRACT

The OECD/NEA benchmark „Reactivity compensation of boron dilution by stepwise insertion of control rod cluster into the VVER-1000 core" is based on measurements of neutron physical and thermal-hydraulic behavior during a corresponding transient in unit 2 of the Rostov nuclear power plant. The measurement data were provided for simulations in the course of the benchmark exercises.

As the first exercise, the neutron physical modeling of the initial reactor state has to be prepared. In particular, this requires a generation of all necessary cross sections of the fuel assemblies and reflectors. This paper summarizes our preliminary results. In detail, we present two different approaches of cross section generation. One approach is based on TRITON/NEWT from SCALE 6.2.4, the other one on CASMO5 from CMS5-VVER. In the next step, we present our preliminary results of the reactor depletion calculations with SIMULATE5 from CMS5-VVER and compare them with the provided measured data.

1 INTRODUCTION

TÜV NORD EnSys (TNE) is a technical support organization that can provide a wide range of reactor safety analyses. In this respect, TNE has many years of experience in providing services to support safety or regulatory authorities, plant operators and research institutions in ensuring, maintaining and improving the safety of nuclear facilities. For example, we have tested new core design methodologies, core loading strategies, and core transient analyses taking into account the signals from reactor core monitoring and protection systems. Reactor core simulations are fundamental for the safe operation of nuclear power plants. Currently, we use different simulation tools. Here, we perform reactor simulations using the internationally proven software packages SCALE 6.2.4 [1] distributed by Oak Ridge National Laboratory (ORNL) and the STUDSVIK SCANDPOWER core management system CMS5-VVER [2] with the lattice transport code CASMO5 [3, 4] and the nodal reactor code SIMULATE5 [5].

All codes provide solvers for the hexagonal geometry required for reactor core calculations of Vodo-Vodyanoi Energeticheskiy Reactor (VVER).

For the first benchmark exercise, we have applied two computational paths for the cross section generation. In both cases, we used a two-step procedure, which consists of an assembly transport calculation with lattice physics code, and reactor core calculation with nodal diffusion code. For the several branches of lattice calculations for fuel assemblies and reflectors we used the deterministic transport codes TRITON/NEWT [6] and CASMO5, respectively, to generate cross section libraries for the 3D nodal diffusion code SIMULATE5. In the second step, the reactor core behavior is then analyzed by SIMULATE5.

This paper is structured as follows. At the beginning, a short description of the benchmark [7] is given followed by the lattice physics and cross section generation as well as a verification of the results at the assembly level. Afterwards, we compare our results to the measured data [7] and to the results provided by the Kurchatov Institute (KI) [8, 9]. Finally, conclusions are drawn and further plans are presented.

2 SHORT DESCRIPTION OF ROSTOV-2 BENCHMARK

The Rostov-2 benchmark was recently developed by the OECD/NEA to enable the validation of reactor physics code systems especially novel high-fidelity multi-physics codes. A detailed description of the benchmark and the exercises is given in the specification [7].

2.1 Reactor core data

The Rostov-2 reactor is a VVER-1000 type reactor with 3000 MW nominal thermal power. The reactor has an active core height of 368 cm and uses five different TBC-2M fuel assembly types [7].

The reactor core is radially separated into hexagonal cells with a pitch of 23.6 cm, each corresponding to one fuel assembly (FA), plus a radial reflector. In total, the core consists of 163 FA, whereby each FA consists of 312 fuel rods [7]. Figure 1 presents the radial layouts of five FA types loaded in the reactor.

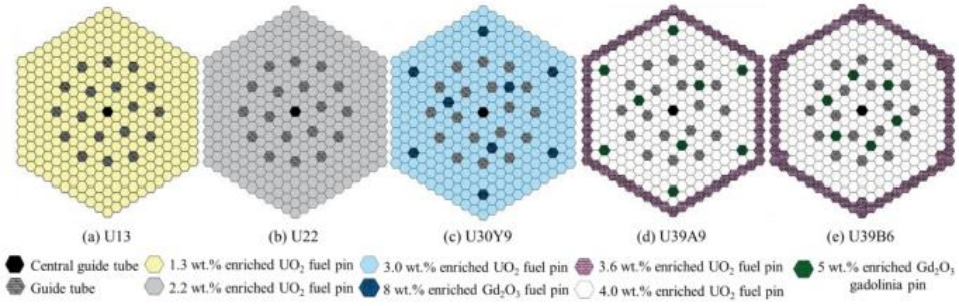

Figure 1: Radial layouts of FA types [10]

Table 1 presents the detailed information of the FA types [7].

Table 1: Specification of FA types

FA type	Number of Fuel Pins/Enrichment (wt.% of ^{235}U)	Number of Gadolinia Pins (wt.% of Gd_2O_3/ ^{235}U)	Number of FA in the Core
U13	312/1.3	--	48
U22	312/2.2	--	42
U30Y9	303/3.0	9 (8.0/2.4)	37
U39A9	243/4.0, 60/3.6	9 (5.0/3.3)	24
U39B6	240/4.0, 66/3.6	6 (5.0/3.3)	12

The benchmark experiment is part of the nuclear commissioning tests of Rostov-2. At the beginning of the experiment, the reactor is in the Xe equilibrium state by a nominal thermal power of 69.37 % (2081.0 MW). The average cycle burnup is 36.37 effective full power day (efpd) and the boron concentration is 4.6 g/kg H_2O. The control rods are grouped into 10 control rod clusters (CRC).

When the reactor is operated at rated power, all control rods are at the top position (above the lower limit switch), except the CRC No. 10 (CRC10). Usually, the CRC10 position is 80-90 % (above the lower limit switch). The CRC10 is used to compensate small changes of reactivity due to variations of coolant temperature, boron concentration, etc. At the beginning of the transient, CRC10 is in 84.4 % withdrawn (wd) position, while the CRC 1-9 are in all controls rods out position. The evolution of the reactor power and the CRC10 positions for the first 36.37 efpd are shown in Figure 2.

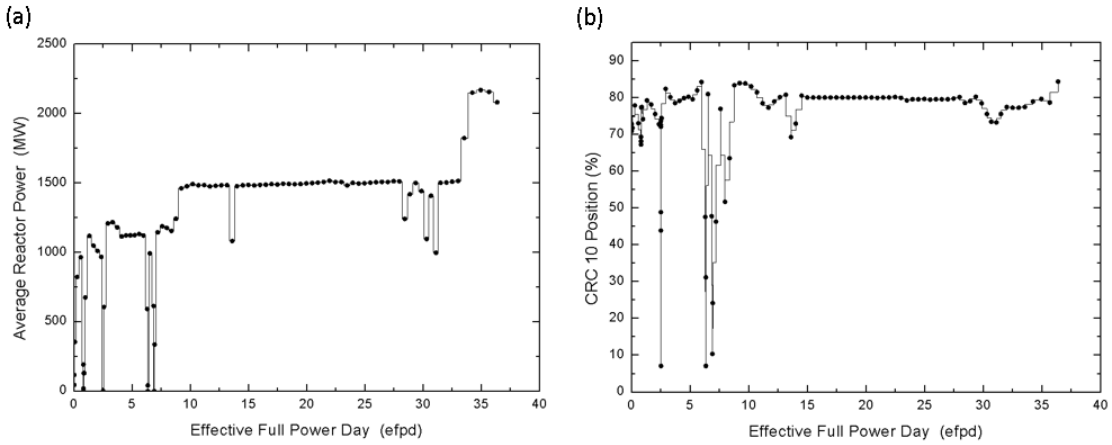

Figure 2: Evolution of (a) the reactor power and (b) the CRC10 positions for the first 36.37 efpd

The transient started with a boron dilution that leads to a decrease of boron concentration in the primary circuit coolant and to a subsequent increase of the core power. To keep the power level inside control limits the CRC10 is being inserted with small steps. At the end of the experiment, the CRC10 is at 32 % wd position. The changes of the position of the CRC10 during the transient lead to redistribution of the power distribution towards core periphery in radial plane and bottom part of the core in axial direction. The simulation of the transient is not a subject of the current presentation.

3 LATTICE PHYSICS CALCULATIONS AND FEW-GROUP DATA GENERATION FOR SIMULATE5

In this work, we use the few-group nodal diffusion code SIMULATE5 for reactor core simulation. The solution of the 3D neutron diffusion equation for reactor core requires information on geometry, material composition, thermal-hydraulic conditions and boundary conditions of the computational domain. Therefore, the whole core is discretized in nodes that contain the information on few-group macroscopic cross sections of the different reaction types, e.g. absorption, capture and scattering cross section. These cross sections are based on multi-group lattice physics calculations performed with deterministic 2D transport codes TRITON/NEWT and CASMO5, respectively. In order to cover all possible core conditions numerous sets of cross sections for each fuel assembly and reflector type have to be generated as a function of depletion, e.g. fuel temperature, moderator temperature, moderator density, boron concentration, rodded/unrodded compositions (so-called branch calculations).

For the generation of the few-group data, it is necessary to collapse and homogenize these multi-group data under consideration of the corresponding multi-group flux distribution. Subsequently, all these few-group data are transformed in a large parameterized library with an appropriate format for the core simulator SIMULATE5.

For the generation of the cross section library for SIMULATE5, we use two different lattice physics codes, TRITON/NEWT and CASMO5. The main methodical differences between both codes concern amongst others resonance self-shielding and transport methods. For the detailed description of these codes, we refer to the following references [1, 3, 4, 6].

Furthermore, there are differences in the chosen multi-group structure (TRITON/NEWT: 252 groups, CASMO5: 586 groups) of the lattice physics calculations. Both codes use the most recent ENDF/B-VII nuclear data.

4 RESULTS

4.1 Lattice physics calculations

Representative for the numerous branch calculations, we show in Figure 3 a comparison of the burnup-dependent neutron multiplication factors of the five fuel assembly types under the following conditions: T_{fuel} = 900 K, T_{clad} = 616.6 K; T_{coo} = 578 K, c_{boron} = 600 ppm.

As can be seen in Figure 3, both approaches show a due to the differences in the methodologies expected reactivity behavior for the five FA. Here, the largest contribution corresponds to the typical bias of 200 - 500 pcm [1] due to the different self-shielding methodology of both codes.

These results verify that both approaches are appropriate for the neutron physical description of the FA and reflectors and provide the necessary quality for the reactor simulation (see section 4.2).

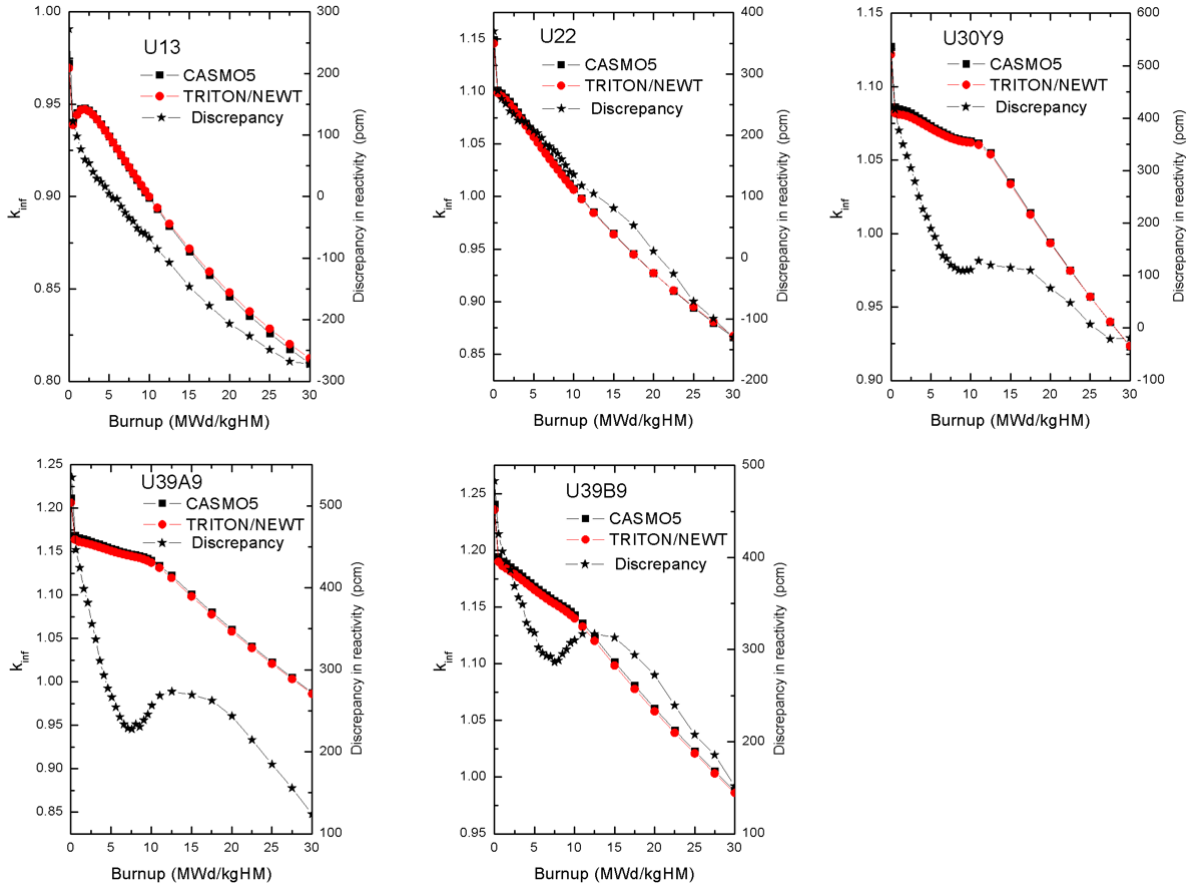

Figure 3: Reactivity behavior of the five FA types: Infinite neutron multiplication factor (k_{inf}) and discrepancy in reactivity as a function of burnup

In this context, we would like to note that the generation of input files with TRITON/NEWT is time-consuming and prone to error and further scripts are needed in order to achieve desired results (e.g. few group constants for the reactor calculations). In contrast, CASMO5 enables a user-friendly generation of input files and comprises scripts for the evaluation of results. All in one, CASMO5 offers a fast and reliable solution. Due to the limited paper size, we present in the following only results using CASMO5/SIMULATE5.

4.2 Simulation of initial reactor state by CASMO5/SIMULATE5

Based on the neutron library generated by CASMO5, the initial state (prior to the transient) is simulated with SIMULATE5 (S5) according to the reactor behavior presented in Figure 2.

- *Comparison with BIPR8 calculation results*

Table 2 shows the calculation results achieved by the code BIPR8 and given by the benchmark specification (Spec) [7] as well as our predicted calculation results for Xe equilibrium core condition (TNE).

Table 2: Comparison of the calculated values (BIPR8 vs. S5) of selected core parameters

				Boron		Kq		Kv		AO	
	T_{in}	CRC10	PN	Spec	TNE	Spec	TNE	Spec	TNE	Spec	TNE
efpd	˚C	%	MWth	g/kg	g/kg					%	%
0	280	80	0	6,97	6,88	-	1,413	-	2,177	-	-4,1
10	282,8	80	1200	5,1	5,24	1,27	1,258	1,74	1,756	-8,30	-6,58
10,05	283,5	80	1500	4,93	5,08	1,26	1,244	1,72	1,739	-9,29	-7,35
15	283,5	80	1500	4,88	5,05	1,25	1,24	1,71	1,732	-9,25	-7,09
17,5	283,5	80	1500	4,85	5,04	1,25	1,237	1,70	1,727	-9,17	-7,02
20	283,5	80	1500	4,82	5,04	1,24	1,234	1,69	1,721	-9,09	-6,94
20,05	285,2	90	2250	4,51	4,79	1,21	1,199	1,63	1,660	-8,33	-6,36
30	285,2	90	2250	4,43	4,71	1,20	1,205	1,60	1,652	-7,91	-5,86

In general, fair agreement between both simulations (BIPR8 / S5) is obtained in terms of critical boron concentration and a good agreement for radial peaking factor (Kq) and for axial peaking factor (Kv). Higher discrepancies are observed in terms of axial offset (AO).

- *Comparison with measured data*

Several FA power values based on flux measurements with self-powered neutron detectors (SPND) as well as computational results (e.g. radial FA power, axial power shape) from KI [8] were used to assess the quality of our simulation of the initial state (36.37 efpd, critical cboron = 805 ppm). Figure 4 shows a comparison of relative FA power values between the experimental values (EXP) and the computational values from KI as well as ours (TNE). Our calculated critical boron concentration for the initial state is 831 ppm. The examined FA positions show a good agreement for the radial power of the initial state. The maximum discrepancy of our results to the measured values remains smaller than or equal to 2.3 % (the standard deviation is 1.13), whereby the maximum discrepancy of the KI results to the measured values is below 3.4 % [8]. It should be noted that no conspicuous, systematic tendency can be observed here.

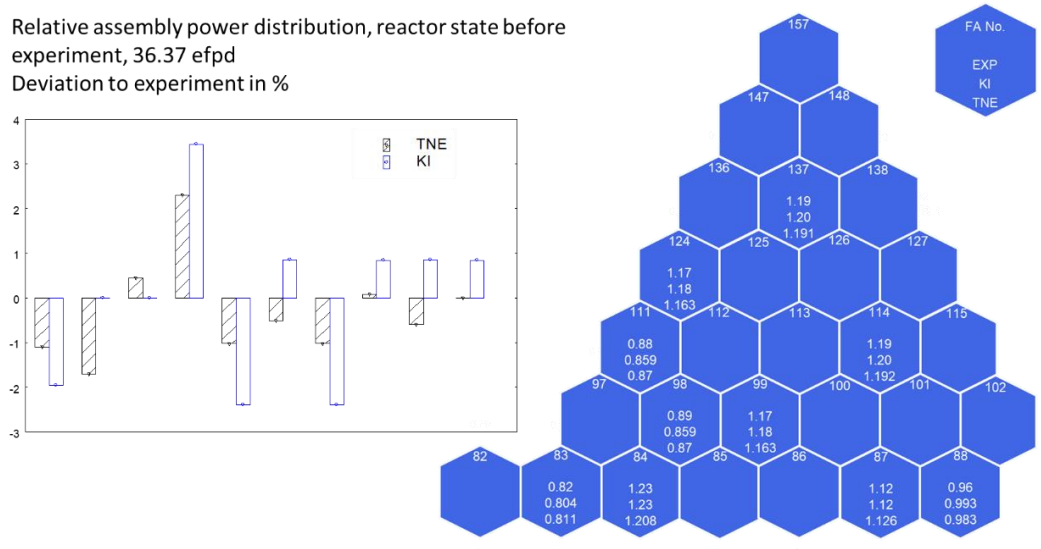

Figure 4: Relative FA power values of the initial reactor state (36.37 efpd), statistic of deviation of relative assembly power (EXP-TNE and EXP-KI)

The last quantity investigated is the axial power profile for the FA with the highest radial power peaking factor (FA 84) and for the FA with the lowest one (FA 111). Figure 5 shows the calculated and measured axial power profiles of these FA [8, 9].

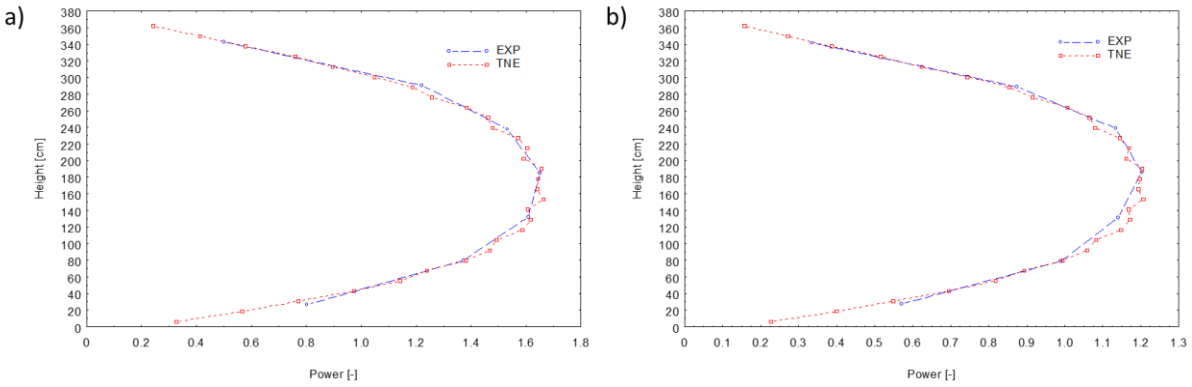

Figure 5: Axial power profiles of the FA with (a) the highest radial power peaking factor (FA 84) and (b) with the lowest one (FA 111). The measurement uncertainty is below 5 %.

A good agreement of our results (TNE) with the measured data (EXP) is achieved and no unusually conspicuous, systematic tendencies can be observed.

5 CONCLUSIONS

We used the OECD/NEA "Rostov-2" benchmark to validate our methodology and computational tools for VVER core analysis. Based on the specification of the benchmark, lattice physics calculations with TRITON/NEWT and CASMO5 as well as reactor depletion calculations with SIMULATE5 are successfully performed.

It is demonstrated that the developed CASMO5/SIMULATE5 model enables a precise reproduction of the initial hot power state in terms of boron concentration, radial assembly power and selected assembly axial power profile.

In the next step, we will reproduce the transient scenario at the assembly and at the pin level to complete the validation process of our computational tools for hexagonal lattices.

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