INTERNATIONAL BENCHMARK FOR COUPLED CODES AND UNCERTAINTY ANALYSIS IN MODELLING: SWITCHING-OFF OF ONE OF THE FOUR OPERATING MAIN CIRCULATION PUMPS AT NOMINAL REACTOR POWER AT NPP KALININ UNIT 3

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ABSTRACT

The paper briefly describes the Specification of an international NEA/OECD benchmark based on measured plant data.

During the commissioning tests for nominal power at NPP Kalinin Unit #3 a lot of measurements of neutron and thermo-hydraulic parameters have been carried out in the reactor pressure vessel, primary and the secondary circuits. One of the measured data sets for the transient ‘Switching-off of one Main Circulation Pump (MCP) at nominal power’ has been chosen to be applied for validation of coupled thermal-hydraulic and neutron-kinetic system codes and additionally for performing of uncertainty analyses as a part of the NEA/OECD Uncertainty Analysis in Modeling (UAM) Benchmark.

The benchmark is opened for all countries and institutions. The experimental data and the final specification with the cross section libraries will be provided to the participants from NEA/OECD only after official declaration of real participation in the benchmark and delivery of the simulated results of the transient for comparison.

INTRODUCTION

Over the past years considerable efforts and progress have been made in various countries and organizations in incorporating full three-dimensional (3D) models of the reactor core into system transient codes. The coupled thermal-hydraulic (TH) and neutron kinetics (NK) code systems allow performing a “best-estimate” calculation of interactions between the core behavior and plant dynamics. Several benchmarks have been developed to verify and validate the capability of the coupled codes to analyze complex transients with coupled core-plant interactions for different types of reactors [1].

The Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD) has recently completed the VVER-1000 Coolant transient benchmark (V1000CT-1) and (V1000CT-2) for evaluating coupled TH system NK codes by simulating transients at the Bulgarian NPP Kozloduy Unit #6. The available real plant experimental data made the benchmark problem very valuable.

This paper is a further continuation of the above activities and it defines a coupled code benchmark problem for further validation of thermal-hydraulics system codes for application to Russian-designed VVER-1000 reactors based on actual plant data from the Russian NPP Kalinin Unit #3 [2]. The selected transient 'Switching-off of one Main Circulation Pump (MCP)' is performed at a nominal power and leads to an asymmetric core conditions with broad ranges of the parameter changes. The experimental data is very well documented. It is being measured with a quite high frequency and the measurements errors are known for almost all parameters. This fact allows to apply the studied transient not only for the validation purposes but also for uncertainty analysis as a part of the NEA/OECD Uncertainty Analysis in Modelling (UAM) Benchmark [3].

BACKGROUND, SCOPE AND GOALS

Under the guidance of the NEA/OECD a lot of benchmarks have been performed concerning the application of coupled 3D TH/NK codes. Some of them have utilized code-to-code comparisons, other have compared code predictions with real measured data.

Most transients in a VVER reactor can be properly analyzed with a system thermal-hydraulics code like ATHLET, using a simplified neutron kinetics models (point kinetics) of the core. A few specific transients require more advanced modeling for neutron kinetics for a proper description. A coupled thermal-hydraulics 3D neutron kinetics code is the most appropriate tool for such tasks.

The proposed benchmark problem is being analyzed with the coupled system code ATHLET-BIPR-VVER [4, 5] and the results compared with the measurements. A lot of very interesting challenging additional problems have to be solved in order to perform correctly the comparisons. This experience entered directly into the preparation of the specification.

The reference problem chosen for simulation is MCP #1 switching off at nominal power when the other three main coolant pumps are in operation, which is a real transient of an operating VVER-1000 power plant. This event is characterized by rapid rearrangement of the coolant flow through the reactor pressure vessel resulting into a spatially dependent coolant temperature change. This leads to insertion of spatially distributed positive reactivity due to the modeled feedback mechanisms and a non-symmetric power distribution. Simulation of the transient requires evaluation of core response from a multi-dimensional perspective (coupled 3D neutronics/core thermal-hydraulics) supplemented by a one-dimensional (1D) simulation of the remainder of the reactor coolant system. The purpose of this benchmark is four-fold:

- To verify the capability of system codes to analyze complex transients with coupled core-plant interactions and complicated fluid mixing phenomena.
- To fully test the 3D neutronics/thermal-hydraulic coupling.

- To evaluate discrepancies between predictions of the coupled codes in best-estimate transient simulations with measured data.
- To perform uncertainty and sensitivity analyses having at disposal not only the measured values but also their accuracy respectively the measurement errors.

DEFINITION OF FOUR BENCHMARK EXERCISES

This benchmark employs many of the characteristics of the preceding NEA/OECD VVER-1000 Coolant Transient Benchmark (V1000CT-1) [6]. The current paper which summarizes the benchmark specification is also based on it and on the experimental data description [2] officially delivered from the Russian institutions to the OECD/NEA.

The benchmark includes a set of input data for the NPP Kalinin-3 and consists of four exercises.

Exercise 1 – Point kinetics plant simulation
The purpose of this exercise is to assess the primary and secondary system model responses. Provided are compatible point kinetics model inputs, which preserve the axial and radial power distribution, and CR #10 and #9 reactivities obtained from a 3D code neutronics model and a complete system description.

Exercise 2 – Coupled 3-D neutronics/core T-H response evaluation
The purpose of this exercise is to model the core and the vessel only. Inlet and outlet core transient boundary conditions are provided by the benchmark team on the basis of calculations performed with ATHLET-BIPR-VVER coupled code system; alternatively, the participants can apply the measured plant data. HFP state (**Exercise #2a**) of the core is required for comparison.

Exercise 3 – Best-estimate coupled code plant transient modeling
This exercise combines elements of the first two exercises in this benchmark and is an analysis of the transient in its entirety. For participants that have already taken part in the Kozloduy-6 NEA/OECD Benchmark [6], it is suggested to start directly with this exercise. As a first step for these participants, it is recommended to perform steady state core calculations at HZP state (**Exercise #3a**), HFP (**Exercise #3b**) and deliver results for comparisons. **Exercise #3a** and **Exercise #3b** will ensure a check out for the correct application of the cross section libraries, the core loading and the core design geometry.

Exercise 4 – Performing of uncertainty analysis for the purpose of Phase-III (System Phase) of the OECD Benchmark for Uncertainty Analysis in Best –Estimate Modelling (UAM) for Design, Operation and Safety Analysis of LWRs [3].

The aim and the specification of this exercise will be described in a separate volume which will depict the state of the art of the results and requirements gained after performing of UAM Exercises I and II.

NEUTRONICS CORE DATA

General

The geometrical and thermal-hydraulic data provided for Kozloduy-6 Benchmark in [6] completely define the Kalinin-3 benchmark exercise concerning the equipment geometry, piping, valves interlocks etc. and the needed modeling information for the NPP thermal-hydraulics. This is die to the fact that the NPP Kalinin-3 and NPP Kozloduy-6 have the same design. A Kozloduy NPP Unit 6 RELAP5 thermal-hydraulic skeleton input deck in the specification quoted above can be used in case that the participants have no experience with the V1000-CT benchmark; all other participants who have already participated in the NEA/OECD Benchmark [6] can apply directly the same thermal-hydraulic model of NPP Kozluduy-6 to simulate the Kalinin-3 NPP transient. Only the core design and loading are different and will therefore be described in more detail in this chapter.

Core and fuel assembly geometry

The core and fuel assembly geometry are the same as in the Kozloduy-6 specification [6]. There are differences in the core loading pattern and the radial location of the different control rod groups.

Radially, the core is divided into hexagonal cells (see Fig.1) with a pitch 23.6 cm, each corresponding to one fuel assembly (FA), plus a radial reflector of the same size. There are a total of 211 assemblies, 163 FA and 48 reflector assemblies. Axially, the reactor core is divided into 10 layers with a height (starting from the bottom) of 35.5 cm, adding up to a total active core height of 355 cm. Both upper and lower axial reflectors have a thickness of 35.5 cm. The axial nodalisation scheme accounts for material changes in the fuel design and for the exposure and moderator temperature (spectral history) variations. Zero flux boundary conditions are specified on the outer reflector surface for both radial and axial reflectors. The choice of the computational mesh is up to the participant, and should be chosen according to the numerical capabilities of the code.

The first fuel loading of the reactor core in Unit 3, NPP Kalinin consists of AFA developed by OKBM (Experimental Engineering Bureau) in Nizhni Novgorod, with Uranium-Gadolinium fuel and without burnable absorbers. At an exposure of 96 effective full power days the core loading had five types of AFA:

- 48 FA with U_{235}-enrichment of 1.3 %;
- 42 FA with U_{235}-enrichment of 2.2 %;
- 37 FA with average U_{235}-enrichment of 2.98 % (303 fuel rods with 3 %-enrichment, 9 gadolinium fuel rods with 2.4 %-enrichment);
- 24 radially profiled FA with average U_{235}-enrichment of 3.9 % (243 fuel rods with 4 %-enrichment, 60 fuel rods with 3.6 %-enrichment, 9 gadolinium fuel rods with 3.3 %-enrichment);
- 12 radially profiled FA with average U_{235}-enrichment of 3.9 % (240 fuel rods with 4 %-enrichment, 66 fuel rods with 3.6 %-enrichment, 6 gadolinium fuel rods with 3.3 %-enrichment).

The fuel loading map in the reactor core of Unit 3 NPP Kalinin is shown in Fig.1. The scheme gives also information of the layout of control rods (CR) and their assignment to different groups.

The core layout in Fig.2 shows the accepted division of FA locations in the reactor core into 6 sectors with a 60^0-symmetry together with layout of CR and their assignment to groups as well as the locations of thermocouples (at FA-outlets) and self powered neutron detectors (SPND).

Neutronics modeling and cross-section library

The specification of the assembly types with their unrodded and rodded compositions will be delivered on CD-ROM together with the corresponding sets of cross sections. A complete set of macroscopic, two-group neutron diffusion cross sections and kinetic parameters defined for each assembly (composition) is provided in a NEMTAB-like format used for the NEA/OECD-CEA benchmark V1000-CT1 [6]. Two types of tables are available – one for the uncontrolled state (nemtab) and another for the controlled state for rodded nodes (nemtabr). For the assemblies which are controlled (CR are moving) during the transient two types of cross section data for controlled state are given (for dysprosium absorber and for boron absorber). Four reflector compositions are defined: upper reflector, bottom reflector and two radial reflectors. The approximation of nuclear data within the proposed table format will be done in three dimensional parameter space. Variables are the fuel temperature, the coolant (moderator) density and the moderator temperature; four support points are assumed for each variable. The burn-up distribution is accounted for in the composition numbers of the axial layers for each assembly type. Due to the fact that during the transient the boron concentration is assumed to be constant, the core macroscopic cross sections are derived for C_b=660 ppm [3.6 g/kg$H_2$0]. Xenon concentration is assumed to be in equilibrium state and is taken into account during the cross section generation process. Time constants and the local fractions of effective delayed neutrons are also provided. The assembly discontinuity factors (ADF), the group inverse neutron velocities and the delayed neutron parameters are also provided for each composition. For the first energy group two diffusion coefficients are provided for the radial and axial direction.

All data in the cross-section library have been obtained using the TVS-M cross section generation code. Each composition is assigned an individual cross section set containing separate tables for the diffusion coefficients and cross-sections, with each point in the table representing a possible core state. The expected range of the transient is covered by the selection of an adequate range for the independent variables as follows:

T_{fuel}: 540.0 K – 1700.0 K

ρ_{moder}: 660.0 [kg/m^3] – 790.0 [kg/m^3]

T_{mod}: 540.0 K – 600.0 K

A linear interpolation scheme is used to obtain the appropriate total cross sections from the tabulated ones based on the reactor conditions being modeled. Table 1 shows the macroscopic cross section table structure for an exemplary cross-section set. All cross-section data, along with a program for linear interpolation will be supplied in electronic form.

THERMAL-HYDRAULIC DATA

Component specifications for the full thermal-hydraulic system model

The design of NPP Kalinin-3 is identical to the NPP Kozloduy-6 design. This fact allows using all available data in [6] for the component description needed for modeling the thermal-hydraulic system. That means that the tables and the description of main equipment (reactor vessel, reactor coolant system, steam generator, feed water system etc.) can be used as described in the specification [6] of the NPP Kozloduy-6.

Definition of the core thermal-hydraulic boundary conditions model

By defining an inlet condition at the core bottom and outlet condition at the full Kalinin NPP Unit 3 thermal-hydraulic model can be converted to a core TH boundary condition (BC) model.

The BCs have been calculated using the ATHLET-BIPR-VVER [4, 5] best-estimate core-plant system code. Core inlet radial distributions will be provided in all 163 assemblies which have been modeled as separate thermal-hydraulic channels. On the base of this mapping scheme there will be given all needed parameters with their time dependent histories (0-300 s) like mass flow rate, inlet and/or outlet pressure, inlet coolant temperature, and positions of the control rod groups.

Table 1 Structure and key of the macroscopic cross-section table

****
*   Nemtab and nemtabr   – Cross-Section Table Input
*   number of support points of:
* Fuel temperature   Rho   Moderator temperature
4   4   4
*
*   Tf   - Doppler (fuel) temperature, (K)
*   ρm - moderator density, (kg/m^3)
* T_m – moderator temperature, (K)
*

Tf_1 Tf_2 Tf_3 Tf_4
Tm_1 Tm_2 Tm_3 Tm_4
ρm_1 ρm_2 ρm_3 ρm_4

*
*********** X-Section Set
*
* Group No. 1
****
*************Radial Diffusion Coefficient Table
*

$D1_1 (Tf_1, Tm1, \rho m_1)$ $D1_2(Tf_2, Tm1,\rho m_1)$ $D1_3(Tf_3, Tm1,\rho m_1)$ $D1_4(Tf_4, Tm_1, \rho m_1)$ $D1_5(Tf_1, Tm_2, \rho m_1)$
$D1_6 (Tf_2, Tm_2, \rho m_1)$ $D1_7 (Tf_3,Tm_2,\rho m_1)$ $D1_8 (Tf_4, Tm_2,\rho m_1)$ …………………………………………
……………………………………………………………….. $D1_{63} (Tf_3, Tm_4, \rho m_4)$ $D1_{64} (Tf_4, Tm_4, \rho m_4)$

*
************* Absorption X-Section Table
************* Scattering from Group 1 to 2 X-Section Table
************* Axial Diffusion Coefficient Table
************* Nu-Fission X-Section Table
************* Kappa X-Section Table
****
* Group No. 2
************* Diffusion Coefficient Table
************* Absorption X-Section Table
************* Nu-Fission X-Section Table
************* Kappa X-Section Table
****
*Additional parameters
************* ADF in radial direction Table
************* Inverse Neutron Velocities (2 values)
*Delayed neutron parameters
************* Betta (6 values)
************* Lambda (6 values)

NEUTRONIC/THERMAL-HYDRAULIC COUPLING

The transient calculations for Exercise #3 must be performed with coupled system codes which should take into account the following effects:

- Fluid mixing in the downcomer, upper and bottom plenum. The measurements showed that the flow through the active core is more or less laminar and no flow mixing is observed in it.
- In order to predict correctly the measured coolant temperatures at the 96 assembly outlet positions, it is necessary to model the mixing of the bypass fluid flow through the control rod guide tubes with the main assembly flow. If not possible, mixing coefficients pre-calculated with ATHLET-BIPR-VVER [7, 8] can be supplied to the participants.
- The delay (inertia) terms of the measurements (mainly coolant temperature) should be modeled in order to allow a direct comparison with the measured thermocouples' readings [9].
- For the simulation of the SPND predictions, the real positions of the sensors should be taken into account; these will be given on CD-ROM in a special file. The quantities to be compared are the relative SPND currents (simulated nodal relative power)

In case that the participants have difficulties to model the secondary circuit response either the experimental data measured at the stem generators (SG) can be applied as boundary conditions or one may acquire the ATHLET-BIPR-VVER simulated data from the benchmark team.

Each participant should use his own coupling TH/NK scheme and methodology.

REACTOR CONTROL SYSTEM

Short description of the control system logic operating by this transient

The purpose of the experiment at Kalinin-3 which is selected for this benchmark is the complete testing of reliability of all power plant equipment, testing the reliability of the main controllers (Automatic Reactor Power Controller (ARC), Electro-Hydraulic Turbine Controller and the controller of the steam generator filling level) and to check the expected neutron reactor power change in case of switching off of one MCP.

The ARC is a part of the Unit Power Control System and operates in coordination with the reactor power limitation controller and the Turbine Electrohydraulic Controller (TEC). The controller stabilizes the reactor power and allows following the load requested from the turbine.

The ARC usually uses the control rod group #10 to operate. In this particular transient the control rod group #10 and group #9 are changing their position during the transient. The reactor power limiting controller (PLC) is used to limit the maximum thermal and neutron power to set points automatically chosen depending on the operational status of certain plant components. The reactor control and protection system (CPS) inserts the control rod group #10 and #9 with a nominal operational speed of 2 cm/sec. When CPS is in operation, ARC is automatically disconnected and AZ-1 signals are not used. Depending on the initiating event, the reactor power is lowered to and then kept at specified set points by CPS.

Control rod group #10 and #9 are changing their position during the selected transient. Analysis of the initial 3D relative power distribution showed that this insertion introduces an axial neutronical asymmetry in the core. At the beginning of the transient there is also a thermal-hydraulic asymmetry stemming from the asymmetric coolant change introduced in ¼ of the core when MCP #1 is switched off. This causes a spatial asymmetry in the reactivity feedback, which is then propagated throughout the transient.

TRANSIENT DESCRIPTION

Initial steady-state conditions (HFP- Exercise #3b)

The reactor is at the middle of cycle (MOC) with an average core exposure of 130.6 EFPD and boron concentration of 3.6 [g/kgH2O]. The definition of the initial steady state is given in Table 2 and is derived from the measurements.

Table 2 The main reactor parameters at the beginning and at the end of the transient

Parameters	Values	
	Initial state	Final state
Date	02.10.2005	02.10.2005
Time, h:min:s	20:30:00	20:34:42
$T_{eff.}$ eff. days	128.50	128.50
N_{core}, MW	2907	1946
N_{PC}, MW	2918	1926
N_{SC}, MW	2877	1938
N_{DCS}, MW	2887	1948
N_{NFC}, MW	2965	1996
N_{el}, MW	986	625
H_{1-8}, cm (%)	352 (100)	352 (100)
H_{10}, cm (%)	292 (82.95)	160 (45.45)
C_B, g/kg$H_2$0	3.60	3.60
Tk_i, ^{0}C	288.14; 287.81; 287.69; 287.50	284.96; 287.40; 287.83; 284.80
$\Delta T_{loop\ i}$, ^{0}C	29.23; 28.87; 28.74; 29.26	-7.98; 23.86; 25.40; 17.88
T_{inlet}, ^{0}C	287.79	286.68
ΔT_{loop}, ^{0}C	29.03	22.38
P_{PC}, MPa	15.52	15.46
ΔP_r, MPa	0.38	0.21
$\Delta P_{MCP\ i}$, MPa	0.569; 0.564; 0.565; 0.562	0.153; 0.460; 0.448; 0.431
G_{loopi}, m3/h	22292; 22223; 21784; 21772	-7198; 24668; 24280; 24725
G_r, m3/h	88073	66475
L_{PRZ}, cm	860	780
L_{SGi}, cm	222; 220; 220; 222	229; 215; 216; 221
G_{i-SG}, t/h	1445; 1367; 1364; 1360	143; 1241; 1283; 937
T_{SG-i}, ^{0}C	215.70; 215.50; 216.40; 214.50	208.90; 201.70; 205.50; 201.10
P_{SGi}, MPa	6.27; 6.30; 6.25; 6.24	6.02; 6.27; 6.23; 6.16
P_{MSH}, MPa	6.02	5.99
δW_{DCS}, %	-6.06	-23.40

δW_{core}, % $K_{q\ max}$/FA $\Delta T_{c\ max}$, ^{0}C/FA $K_{v\ max}$/FA/layer	-2.15 1.27/08-25 28.71/08-25 1.50/10-31/2	-20.70 1.29/12-21 27.69/08-25 1.81/10-31/2

An additional HZP state is defined for initialization of the 3D core neutronics model for Exercise #2 – Exercise #2a and for Exercise #3 – Exercise#3a (only for those participants that do not need to perform Exercise #1 and #2 because of availability of a system model for NPP-Kozloduy -6).

The HZP (Exercise #2a or Exercise #3a) conditions are defined as follows:
The power level is 0.1% of the nominal power; the fuel and moderator temperature are 552.15 K and the moderator density is 767.1 [kg/m^3]. Only control rod group #10 is 82.95 % inserted from below. Boron concentration is 3.6 [g/kgH_20].

Transient scenario

The transient scenario (recovered from the measured data histories) is listed shortly below.

- Manually switching off of MCP #1 at t=0s.
- After the signal 'one pump out of operation' which is generated after 1.41 s, rector limiting controller starts to decrease the power to a level of 67.2 %.
- The following sequence of actuations for rector limiting controller and automatic reactor power controller is recorded:
  - At t=1.41 s the reactor limiting controller starts to decrease the reactor power. CR #10 starts to move downwards. When the CR #10 reaches 50 % insertion depth (at about 60 s) the CR #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 off 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 cancelled and power reaches a level of 67.2 % P_{nom}. At this moment the position of the CR #10 is at 43.4 % and remains there till the end of the transient. CR #9 is inserted into the core and reaches at 71 s the position of 93.1 % and stays there till 180 s, returning 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}.
  With the reactor limiting controller the reactor power was decreased from 98.9 % P_{nom} to 67.2 % P_{nom} within 71 s. The speed of reactor power decrease (load-off) within the reactor limiting controller operations is 26.8 % /min. The change of the coolant heat-up in the core decreases from 29 ^{o}C to 23.3 ^{o}C.
- Due to reactor limiting controller operation and switching off the automatic reactor power controller, the electronic controller of the turbine generator electro-hydraulic automatic controller starts the load-off operation of the turbine generator. At t=222 s the power of the turbine generator corresponds to the reactor power and stabilizes at 625.5 MW.
- The pressure in the main steam line changes from initial 6.01 MPa to a level of 5.86 - 6.02 MPa. At 300 s the pressure is stabilized at 6.02 MPa.
- Primary pressure changes from initial 15.52 MPa to 15.12 -15.56 MPa following the change of the mean primary coolant temperature.

- The temperature decrease of the affected loop #1 (within the time interval from 30 s to 140 s) leads to a decrease of the mean reactor coolant temperature and in turn leads to a decrease of the volume of the coolant in the primary loop. That affects (decreases) the pressurizer level which leads to a decrease of the primary loop pressure. At 94 s of the transient the pressure is stabilized at 15.13 MPa. Due to the pressurizer heaters operation starting from 140 s the pressure starts to increase and at 300 s it stabilizes at 15.47 MPa. As a result the pressurizer level changes from 858.5 cm at the beginning to 801.1 cm at the end of the transient.

Point kinetics model inputs (Exercise #1)

The point kinetics model is necessary only for Exercise #1 which should be performed in case that the participants have not taken part in the OECD/DOE/CEA V1000CT Benchmark [6] or have no consistent model for VVER-1000. Point kinetics model inputs, which preserve axial and radial core power distributions obtained with 3D neutronics model BIPR-VVER are given on the CD-ROM. The following parameters for the point kinetics model can be found:

- Control Rod Group #10 and #9 worth;
- Axial power distribution;
- Moderator temperature coefficient;
- Moderator density coefficient;
- Doppler temperature coefficient;
- Other kinetics parameters (delay neutron parameters, etc.).

Transient core calculations (Exercise #2)

Exercise #2 is a boundary condition problem with the aim of testing the correctness of participants' core loading, neutronic data and core thermal hydraulic without modelling the primary loops. The required thermal-hydraulic data will be recorded on the CD-ROM. It includes the time histories of the following parameters calculated with the coupled system code ATHLET-BIPR-VVER; alternatively they can be taken from the measurements directly:

- Core inlet assemblywise mass flow distribution;
- Core inlet assemblywise coolant temperature distribution;
- Core inlet/outlet pressure;
- Position of the CR groups #9 and #10.

Transient coupled calculations (Exercise #3)

Exercise #3 is the final goal of the Benchmark - to predict the NPP response in a 'best estimate' manner with a coupled code system.

The changes of the main parameters at the end of the transient are given in Table 1. MCP test plant data recorded with time interval of 1 s and the transient analysis are described in detail in the specification. The simulated results should be compared with the real measured data.

OUTPUT REQUESTED

The requested output is as closer as possible to the same format and data quantity like in the specification of the Kozloduy-6 Benchmark [6]. It includes local and global thermal-hydraulic and neutron-kinetic data which are in detail described in the specification.

The analysis results will be presented in a benchmark analysis report, which will be made available in both hard copy and electronic form.

Requirements for the participants' data delivery:

- Results should be presented in digital form.
- All data should be in SI units (kg, m, sec).
- For time histories, data should be at 1.0-second intervals.
- Graphical comparison of calculated results and test data should be performed.

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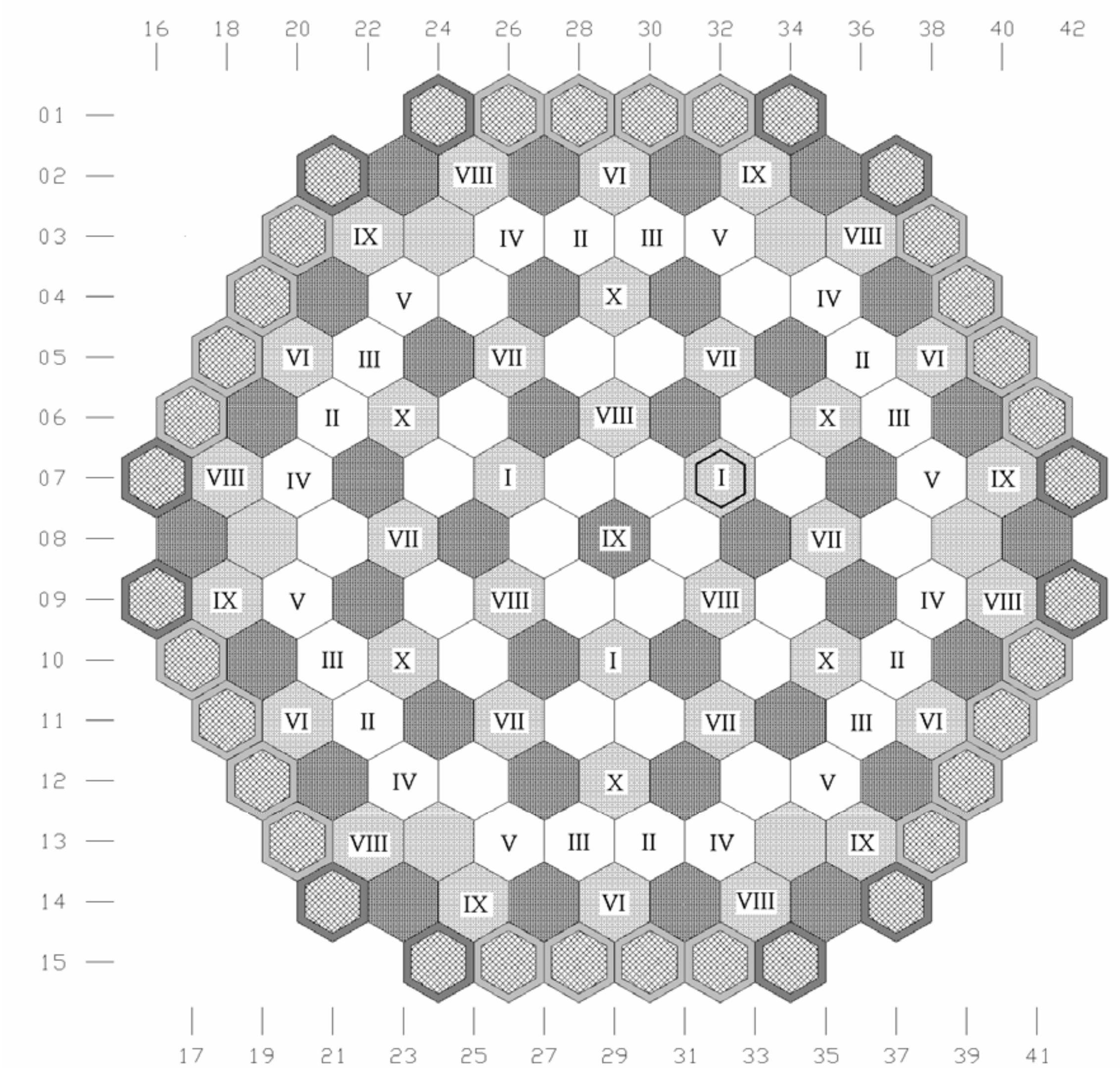

Enrichment of U-235 1.3 % - 48 FA

Enrichment of U-235 2.2 % - 42 FA

Average enrichment of U-235 2.98 % - 37 FA
(303 FA with enrichment 3 %, 9 gad-FA with enrichment 2.4 %

Average enrichment of U-235 3.9 % - 24 „profiled“ FA (243 FA with enrichment 4.0 %, 60 FA with enrichment 3.6 %, 9 gad-FA with enrichment 3.3 %)

Average enrichment of U-235 3.9 % - 12 „profiled“ FA (240 FA with enrichment 4.0 %, 66 FA with enrichment 3.6 %, 6 gad-FA with enrichment 3,3 %)

Enrichment of U-235 1,6 % (stainless steel)

Fig. 1: Core loading

Fig. 2: Division of the reactor core in 60 °-symmetry sector, layout of FA, CPS control rods and CPS CR group allocation, layout of thermal control sensors at FA outlets and assemblies with SPND