5.8 DEFINITION OF THE 7-TH DYNAMIC AER BENCHMARK – VVER-440 PRESSURE VESSEL COOLANT MIXING BY RE-CONNECTION OF AN ISOLATED LOOP

(EDITION 1)

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ABSTRACT

The 7th dynamic benchmark is a continuation of the efforts to validate systematically codes for the estimation of the transient behavior of VVER type nuclear power plants. This benchmark is a continuation of the work in the 6th dynamic benchmark. It is proposed to be simulated the transient - re-connection of an isolated circulating loop with low temperature or low boron concentration in a VVER-440 plant. It is supposed to expand the benchmark to other cases when a different number of loops are in operation leading to different symmetric and asymmetric core boundary conditions. The purposes of the proposed benchmark are:
1) Best-estimate simulations of an transient with a coolant flow mixing in the Reactor Pressure Vessel (RPV) of VVER-440 plant by re-connection of one coolant loop to the several ones on operation,
2) Performing of code-to-code comparisons.
The core is at the end of its first cycle with a power of 1196.25 MWt. The basic additional difference of the 7-th benchmark is in the detailed description of the downcomer and bottom part of the reactor vessel that allow describing the effects of coolant mixing in the RPV without any additional conservative assumptions. The burn-up and the power distributions at this reactor state have to be calculated by the participants. The thermohydraulic conditions of the core in the beginning of the transient are specified. Participants self-generated best estimate nuclear data is to be used. The main geometrical parameters of the plant and the characteristics of the control and safety systems are also specified. Use generated input data decks developed for a VVER-440 plant and for the applied codes should be used. The behaviour of the plant should be studied applying coupled system codes, which combine a three-dimensional neutron kinetics description of the core with a pseudo or real 3D thermohydraulics system code.

1. INTRODUCTION

During the last years a series of six 3D hexagonal dynamic benchmarks have been defined in the framework of the Working Group D „VVER Safety Analysis“ of the AER. The first three benchmarks were concerned with the asymmetric control rod ejection at low power in a VVER- 440 core. The issue of the fourth benchmark was a boron dilution accident in the same core. The complexity of the benchmark tasks was increased step by step. The first benchmark considered only neutron kinetics without feedback, the second one included the effect of the Doppler feedback, while the third and fourth benchmarks included a full thermohydraulics model of the core. Nuclear cross section data was specified in the first two benchmarks, while use of own generated data was requested in third and fourth benchmarks. The fifth benchmark was the first one for coupled thermohydraulics system/3D hexagonal neutron kinetics core models. In this benchmark, the thermohydraulics boundary conditions for the core were not longer defined as in the first four benchmarks but hat to be calculation by the participants. In the sixth benchmark with the help of the coupled three-dimensional neutron kinetics / thermohydraulics system code was modelled in more detail the processes in the primary and secondary sides of a NPP: double ended break of one main steam line (asymmetric leak) in a NPP with VVER-440. Additional new features in the sixth benchmark are: asymmetric operation of the feed water system, effects of coolant mixing in the reactor vessel, defining of a fixed isothermal re-criticality temperature for normalising the nuclear data. The results of these benchmarks have been published in the AER Symposium proceedings [1 – 6].

The seventh benchmark is a continuation of the efforts for testing the performance and validation of coupled neutron/thermal-hydraulic system codes. The initiating event is the opening of the main isolation valve (MIV) and so re-connecting the loop with its main circulation pump (MCP) on operation. One control rod in the sector of highest overcooling is assumed stuck at its fully withdrawn position after SCRAM. A special attention should be paid on modelling of the downcomer and bottom plenum of the reactor vessel in order to simulate correctly the coolant mixing at these parts of the RPV

In section 2 the main geometrical parameters and setpoints of the relevant for the transient control and safety systems are specified. Further the necessary nuclear and thermohydraulics input data for the initial state and the scenario of the transient are presented. The results and format requested for the comparison are described in section 3.

2. INPUT DATA

The most part of the required input data is derived from the definition of a benchmark six [6] The reference plant for this benchmark is a VVER-440/213 NPP. The definition is based on the assumption that all possible participants of the benchmark have input data decks for the VVER-440 which has been developed according to the needs of their own thermohydraulics system/neutron kinetic core models. Therefore information about all details necessary for the generation of a new input data deck is not provided. The main geometrical characteristics are specified to adjust the existing input data decks.

a: Geometry of the primary circuit

Tab. 1: Hot Leg Geometry

Elevation (z) [m]	Length (x) [m]	Diameter (d) [m]
0.0	0.0	0.496
0.0	1.2	0.496
-1.40	5.28	0.496
-1.40	13.52	0.496
-0.45	14.47	0.496
-0.25	14.67	0.800
2.38	17.30	0.800
2.73	17.65	0.550
3.53	18.45	0.496

Tab.: 2 Cold Leg Geometry

Elevation (z) [m]	Length (x) [m]	Diameter (d) [m]
3.53	0.0	0.496
2.73	0.80	0.550
2.38	1.15	0.800
-0.48	4.01	0.800
-0.60	4.13	0.496
-2.92	6.45	0.496
-2.92	15.76	0.496
-1.15	17.53	0.496
-1.15	18.43	0.496
-1.40	18.68	0.496
-1.40	26.28	0.496

Tab. 3: RPV Elevations

Elevation z [m]	
-9.86	lowest RPV elevation
-6.02	beginning of the unheated core part
-5.38	lower fuel boundary
-2.94	upper fuel boundary
-2.46	end of the unheated core part
3.56	highest RPV elevation

Tab. 4: RPV Volumes

Object	Volume [m^3]
Downcomer	18.8
Lower plenum	23.1
Core region	12.7
Upper plenum including vessel head	40.9

The hot leg nozzle is set to the elevation 0.0m. All elevations provided in the Tables 1-3 and 5, 6 are related to this reference point. For checking the geometry of the primary circuit, the most relevant data of the hot leg together with the steam generator (SG) inlet collector are presented in Tab. 1 and of the SG outlet collector together with the cold leg in Tab. 2. It should be known, that the geometry data presented in Tab. 1 and 2 should be considered only as a key data for tuning the participants data sets. Details of the geometry are not provided. The 5536 U-tubes have an inner diameter of 13.2 mm, an outer diameter of 16.0 mm and an averaged length of 9.02 m. They are located on the collectors' elevation from z=0.14 m to z=1.96 m. The RPV elevations are shown in Tab. 3. Table 4 contains the water volumes of the main parts of the RPV. The pressurizer is connected to one loop (No. 5 acc.in Fig. 2) by two surge lines of a diameter of 0.21 m and a length of 25.37 m, each. The connecting point of the surge lines to the hot leg is at x=7.37 m (acc. to Tab. 1). The surge lines can be modelled by one two fold line. The lowest elevation of the pressurizer is -1.15 m and the highest is 8.85 m. The diameter is 2.40 m, the whole volume 44.0 m^3. The volume control system is connected to the cold legs of loops 2 and 6 (acc. to Fig. 2) at x= 8.45m (acc. to Tab. 2). The main isolation valves are located on the hot leg at x=8.34 m (acc. to Tab. 1) and on the cold leg at x=21.94 (acc. to Tab. 2). It is considered, that MIV opens for 36 sec according to a linear law.

b: Geometry of the secondary circuit

This section completely coincides with corresponding section in [6].
The SG has an inner diameter of 3.21 m (the corresponding elevations are z=-0.025 m and z=3.185 m). It is connected on the top to the main steam line (MSL). The connecting lines between SG and MSL can be omitted. The elevations and lengths of the MSL are shown in Tab. 5. The steam line isolation valves are located at 41.33 m away to the outlet of the steam generators. The main steam header (MSH) is a pipe with a diameter of 0.425 m and a length of 83.40 m (Tab. 6). It is directly connected to the MSL (without any small connecting pipes) at x=51.25 m (acc. to Tab. 5). The MSH isolation valve is not modeled.

Tab. 5: Main Steam Line (MSL)

Elevation (z) [m]	Length (x) [m]	Diameter (d) [m]
3.185	0.00	0.425
3.185	9.80	0.425
5.385	12.00	0.425
5.385	26.80	0.425
11.485	32.90	0.425
11.485	66.33	0.425
2.785	75.03	0.425
2.785	83.13	0.425

Tab. 6: Main Steam Header (MSH)

Elevation (z) [m]	Length (x) [m]	Diameter (d) [m]
11.485	0.00	0.425
12.985	1.50	0.425
12.985	81.90	0.425
11.485	83.40	0.425

c: Reactor core geometry and material parameters

This section completely coincides with corresponding section in [6].
The core loading pattern with three different fuel enrichments is applied (Fig. 1). As follows from Tab. 3 the active core length is 2.44 m. The unheated parts below and above the active core have the same hydraulic diameter and free flow cross section like the core. The main fuel parameters are given in Tab. 7. All types of fuel assemblies have the same heat transfer characteristics. 97.5 % of the total power are released uniformly in fuel pellet, the other 2.5 % directly in the coolant of the respective fuel assembly due to γ-radiation. The gas-gap heat transfer coefficient is to be kept constant during the whole transient: 3000 W/ (m^2*K). Radial thermal conductivities and thermal capacities of fuel pellet and cladding are described with best available data of each participant. Axial transfer of heat is neglected in fuel pellet and cladding.

Each participant should use own best estimate nuclear cross section data and other neutronic related information. The decay heat has to be taken into account.

Tab. 7: Main Fuel Parameters

Fuel assembly pitch	14.7cm
Number of heated pins per assembly	126
Fuel pellet inner diameter	0.14cm
Fuel pellet outer diameter	0.76cm
Cladding inner diameter	0.78cm
Cladding outer diameter	0.91cm
Free flow cross section per fuel assembly	89.0cm^2
Equivalent hydraulic diameter	0.86cm
Fuel density	10.4 g/cm^3
Cladding density	6.25 g/cm^3

d: Heat structure modelling

This section completely coincides with corresponding section in [6].
The following components have to be included in the heat structure modelling: the RPV, the primary coolant pipes, the heat exchanger tubes and the pressurizer with the surge lines.

e: Mixing in the reactor vessel

Each participant should use own models for the description of the coolant mixing in the lower and the upper reactor plenum according to the possibilities of the available codes. Both upper and lower plenum mixing are applied to temperature and boron acid concentration. Allocation of any sectors in the core connected to a certain loop should not be fixed. The aim of the RPV modeling should be to allocate as much as possible calculation volumes at the description of downcomer, upper plenum and lower plenum and so to describe coolant cross flow between these calculation volumes (for example by cross-connections in ATHLET).
Comparing the results of different coupled codes will allow each participant to check up its own developed model of coolant mixing.

f: Characteristics of considered control and safety systems

In this section are specified the characteristics and the set points of the control and safety systems that should be considered by the modeling. Only systems and signals which are mentioned explicitly have to be taken into account. All others should be neglected. The pressurizer has four groups of heaters. The power, the activation pressure, and the deactivation pressure are shown in Tab. 8. It is recommended to model the reaching of full heater power after switching-on by a low pass filter with a time constant of 5s. When the collapsed level in the pressurizer measured from the bottom drops below 2.56 m, the heaters are automatically switched-off, also through the same low pass filter.

Tab. 8: Pressurizer Heater Groups

	Power [kW]	Activation Pressure [MPa], measured in the PRZ	Deactivation Pressure [MPa], measured in the PRZ
Group 1	180	12.0	12.1
Group 2	180	11.9	12.0
Group 3	540	11.8	11.9
Group 4	540	11.5	11.8

The volume control system is activated in the following manner: The 1st pump of this system is connected to loop No.2. The pump is started when the pressurizer collapsed level drops over 10 cm from the nominal level (L_{nom} = 5.97 m). If the 1-st MCP is working for a period of about 40 s (i.e. the difference between actual and nominal level is continued), then the 2-nd pump is started, which is connected to the loop No.6. The mass flow rate is 1.7 kg/s per pump. The supplied water has the same boron concentration like the reactor coolant and enters the reactor coolant system with a temperature of 260°C. By reaching the nominal level the volume control system is switched-off.

The feed water system has to be modeled in the following manner:
The feed water and steam flow in the initial state should be adjusted to the full power conditions for a SG in operation. The controlling value is fixed on a collapsed level (2.015±

0.10) m.
In SG №1 before transient also the controlling value is fixed on a collapsed level (2.015± 0.10). (During same time in the loop №1 U-tubes as the part of the primary circuit is the coolant with temperature 100°C between closed MIVs.)
After a scram, the feed water temperature decreases linearly from 220°C to 160°C within 50s and then remains constant.

The reactor protection (scram) is activated after reaching a reactor power level of 110% of the nominal value NN (nominal power is 1375 MW). The time delay of starting the following control rod drop is 0.5 s. The velocity of control rod insertion is 25.5 cm/s. Until the scram initiation, both turbines work at the power level with the constant mass flow rate of the initial state. The reactor scram signal with the delay of 10 s initiates the turning-off of turbines by closing the turbine isolation valves within 0.5 s.

g: Initial conditions

Burn-up

Because the benchmark calculation will be performed for the end of the first fuel cycle (EOC) conditions, a burn-up calculation for the first loading of the VVER-440 core is required. This calculation should be made at a power level of 1375 MW until the critical boron concentration reaches the value of zero. During the burn-up calculation all control rod groups are fully withdrawn. Their position will not change. The thermohydraulics conditions during the burn-up calculation correspond to the conditions of the initial state before the transient. The burn-up distribution obtained at the end of this calculation should be used in the transient calculation.

Initial neutronic conditions

At the beginning of the transient the reactor is at full power level (1196.25 MW). All control rod groups are fully withdrawn with the exception of control rod group K6 which is at the position 175 cm from the bottom of core. This location is different from the rod position during the burn-up calculation. The locations of the control rod groups are presented in Fig.3. Xe and Sm concentrations are assumed to be in equilibrium (state with partly inserted K6 rods). No boron acid is in the coolant. The multiplication cross section $\nu\Sigma_f$ are divided by the initial k_{eff} to obtain a critical state at the beginning of the transient.

Initial thermohydraulic conditions

The following thermohydraulic input data are specified:

Primary circuit

Upper plenum pressure:	12.26 MPa
Core inlet temperature:	267.4 ^{o}C
Loop mass flow rate (working loops):	1470 kg/s
Core bypass mass flow rate:	3%
Pressurizer collapsed level (measured from the bottom):	5.97 m

Secondary circuit

Pressure at SG outlet (working loops SG):	4.63MPa
Feed water temperature:	220°C
Feed water mass flow rate to a single SG:	124.5kg/s
SG collapsed level (measured from the bottom):	2.015m

h: Scenario of the transient calculation

It is necessary to achieve such an initial condition that before the beginning of the transient at MIV of the isolated loop a coolant with temperature 100°C is reached. The initiating event is a re-connection of a loop №1 with coolant temperature in it of 100°C. The scenario of connection is: First opens the MIV of the hot leg, after that the MCP is started. After the MCP reached full flow the MIV in the cold leg starts to open. A water slug with a lowered temperature enters the core that results in increase of reactor neutron power.
The activation of the reactor scram is caused by the corresponding power level signal with the indicated time delay. The stuck rod belonging to group K4 (position 293) is located in the sector of highest overcooling (see Fig. 3) A fully withdrawn position is assumed. As a result of the scram, the turbines are turned-off by closing the turbine isolation valves. It is postulated, that all main coolant pumps (MCP) remain in operation. It is recommended to perform the calculation until at least 200 s after the MIV opening.

3. RESULTS FOR COMPARISON

a: Requested Results

1. " Single key parameters "

- Total core power [MW]:
  - Ÿ at the beginning of the transient t=0.0s
  - Ÿ at the power maximum

- Total prompt fission power [MW]:
  - Ÿ at the beginning of the transient t=0.0s
  - Ÿ at the power maximum

- 3D power peak factor F_Q (with information about positions according to Fig.1 and core layer):
  - Ÿ at the beginning of the transient t=0.0s
  - Ÿ at the power maximum

2. Spatial nuclear power distribution (" Power distribution")

The normalized two-dimensional assembly-wise power distributions and the axial distribution generated by radial averaging of the 3D normalized power distribution (for each of ten equal core layers) are to be given. The nodes with absorber material belongs to the core volume in any normalization procedure. This normalization has to be applied also for the determination of the power peaking factors F_Q. The values of the assembly powers are to be provided according to the numbering used in Fig. 1 (row-wise from left to right and from bottom to top).

The power distributions are to be given at following times:

- t=0.0s, initial state
- time of power maximum (before SCRAM)

3. "Table of events" (switching-off/on, closing and opening of the different systems ...)

Time (s)	**Event**
0.0	Begin opening MIV on cold leg
	...
x.x	SCRAM value reached
	...
x.x	End of calculation

4. “Time functions 1”

- Total nuclear power of the core [MW] (FPOW)
- Total prompt fission power of the core [MW] (PFPOW)
- Total power transferred to coolant [MW] (THPOW)
- Reactivity [pcm] (REAC)
- Upper plenum pressure, measured at the hot leg inlet elevation [MPa] (PUP)
- Lower plenum pressure, measured at the assemblies inlet elevation [MPa] (PBT)
- Pressure, measured at the cold leg outlet elevation [MPa] (PCLO)
- Pressurizer collapsed level measured from the pressurizer bottom [m] (CLPRZ)
- Maximum fuel pellet centerline temperature [^{o}C] (TFMAX)
- Averaged fuel temperature in the core [^{o}C] (TFAVE)

(The fuel temperature should be averaged over all nodes of the active core including the absorber parts of inserted control assemblies. In the absorber parts, the fuel temperature is assumed to be equal to the coolant temperature.)

5. “Time functions 2”

- Cold leg outlet coolant mass flow rate of loop 1 [kg/s] (MFCL1)
- Cold leg outlet coolant mass flow rate of loop 2 [kg/s] (MFCL2)
- Cold leg outlet coolant mass flow rate of loop 3 [kg/s] (MFCL3)
- Cold leg outlet coolant mass flow rate of loop 4 [kg/s] (MFCL4)
- Cold leg outlet coolant mass flow rate of loop 5 [kg/s] (MFCL5)
- Cold leg outlet coolant mass flow rate of loop 6 [kg/s] (MFCL6)
- Cold leg outlet coolant temperature of loop 1 [^{o}C] (TCL1)
- Cold leg outlet coolant temperature of loop 2 [^{o}C] (TCL2)
- Cold leg outlet coolant temperature of loop 3 [^{o}C] (TCL3)
- Cold leg outlet coolant temperature of loop 4 [^{o}C] (TCL4)
- Cold leg outlet coolant temperature of loop 5 [^{o}C] (TCL5)
- Cold leg outlet coolant temperature of loop 6 [^{o}C] (TCL6)
- Hot leg inlet coolant temperature of loop 1 [^{o}C] (THL1)
- Hot leg inlet coolant temperature of loop 2 [^{o}C] (THL2)
- Hot leg inlet coolant temperature of loop 3 [^{o}C] (THL3)
- Hot leg inlet coolant temperature of loop 4 [^{o}C] (THL4)
- Hot leg inlet coolant temperature of loop 5 [^{o}C] (THL5)
- Hot leg inlet coolant temperature of loop 6 [^{o}C] (THL6)

6. “Time functions 3”

- Secondary side collapsed level in SG-1 [m] (CLSG1)
- Secondary side collapsed level in SG-2 [m] (CLSG2)
- Secondary side collapsed level in SG-3 [m] (CLSG3)
- Secondary side collapsed level in SG-4 [m] (CLSG4)

- Secondary side collapsed level in SG-5 [m] (CLSG5)
- Secondary side collapsed level in SG-6 [m] (CLSG6)
- Steam pressure at SG-1 outlet [MPa] (PSG1)
- Steam pressure at SG-2 outlet [MPa] (PSG2)
- Steam pressure at SG-3 outlet [MPa] (PSG3)
- Steam pressure at SG-4 outlet [MPa] (PSG4)
- Steam pressure at SG-5 outlet [MPa] (PSG5)
- Steam pressure at SG-6 outlet [MPa] (PSG6)
- Steam mass flow at SG-1 outlet [kg/s] (GSG1)
- Steam mass flow at SG-2 outlet [kg/s] (GSG2)
- Steam mass flow at SG-3 outlet [kg/s] (GSG3)
- Steam mass flow at SG-4 outlet [kg/s] (GSG4)
- Steam mass flow at SG-5 outlet [kg/s] (GSG5)
- Steam mass flow at SG-6 outlet [kg/s] (GSG6)
- Total power transferred to secondary side in SG-1 [MW] (POWSG1)
- Total power transferred to secondary side in SG-2 [MW] (POWSG2)
- Total power transferred to secondary side in SG-3 [MW] (POWSG3)
- Total power transferred to secondary side in SG-4 [MW] (POWSG4)
- Total power transferred to secondary side in SG-5 [MW] (POWSG5)
- Total power transferred to secondary side in SG-6 [MW] (POWSG6)

7. “Time functions 4”

- Fuel assembly№1 inlet coolant temperature [^{o}C] (TFA1)
- Fuel assembly№2 inlet coolant temperature [^{o}C] (TFA2)

...

- Fuel assembly№349 inlet coolant temperature [^{o}C] (TFA349)

b: Files, Format

Each type of the described output data should be preceded by the keyword given in the heading, and each power distribution additionally by the time for the distribution. The time functions should be presented with a time resolution of at least 1s. It is recommended, to use a finer output during the power peak.
The data arrays of all time functions should contain the time (in s) and the values of the requested quantities (in the given order) for successive time points. The first point t=0.0s corresponds to the cold leg MIV opening. Each data array with time functions should contain a heading line with the keyword ”TIME” in the first column and the abbreviations for the provided quantities given above in the other columns.

All output data except for “Time functions 4”should be given in one file: "OUTPUT1". The data on “Time functions 4”should be given in the file: "OUTPUT2".

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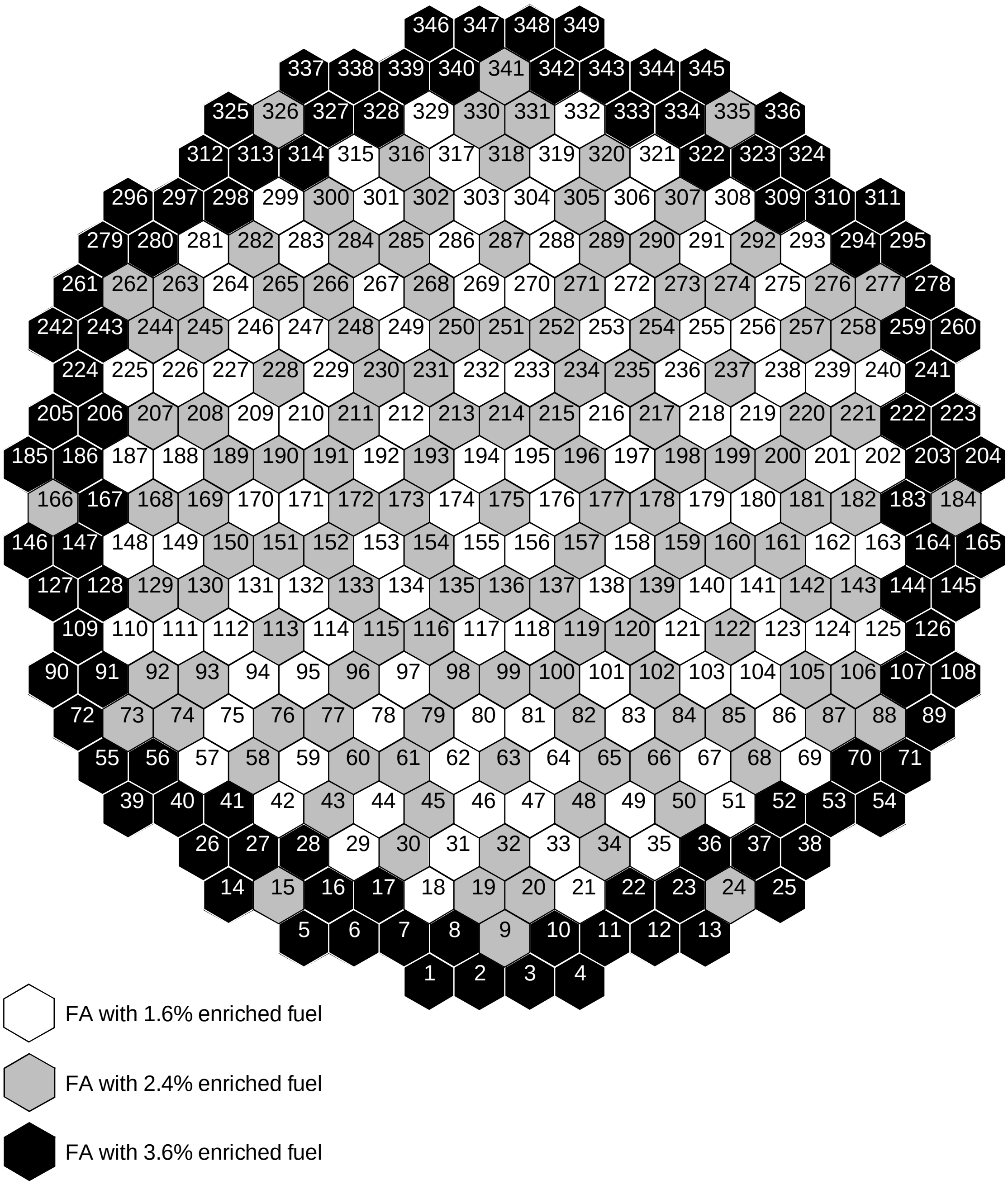

Fig. 1 Core loading map of the VVER-440 reactor

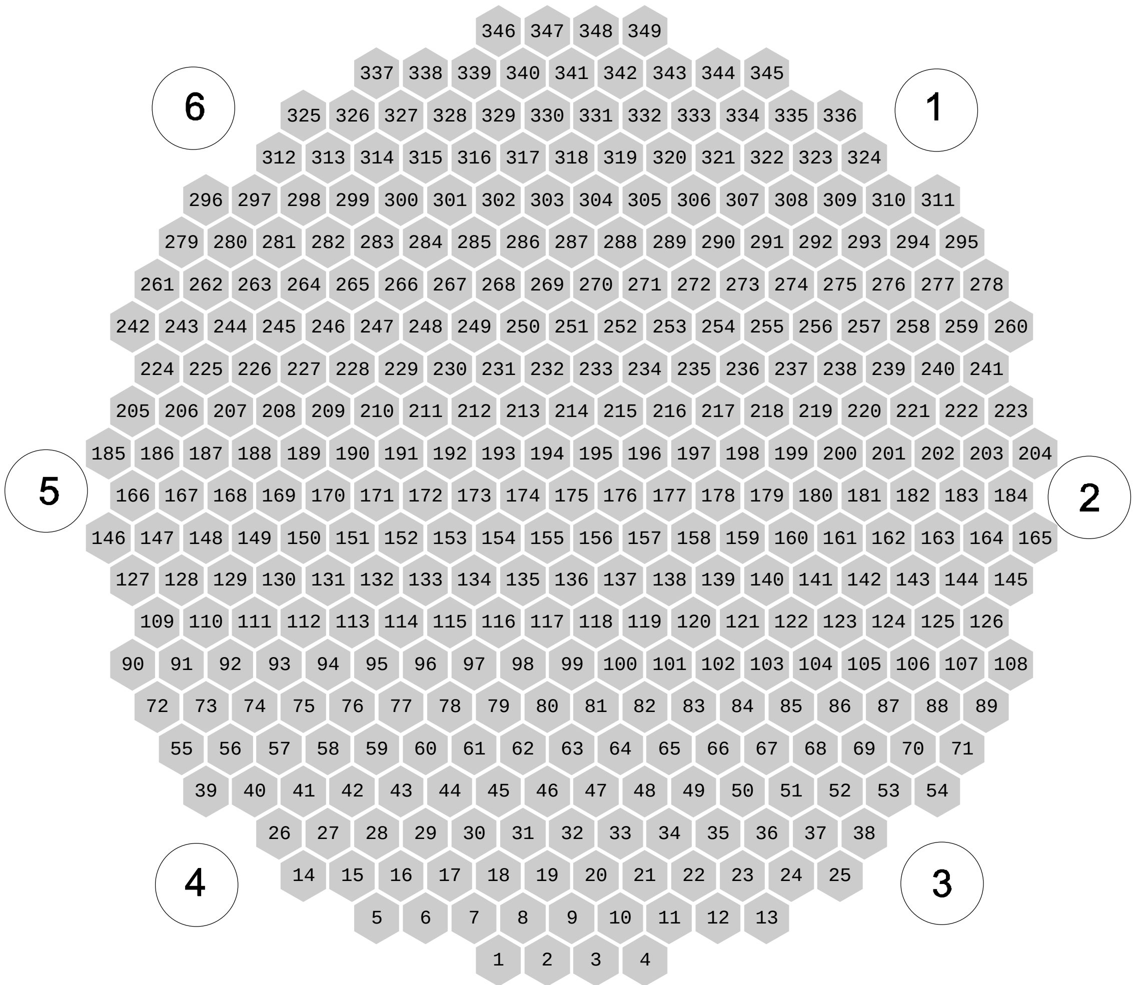

Fig. 2 Loops’ nozzles locations of VVER-440 RPV

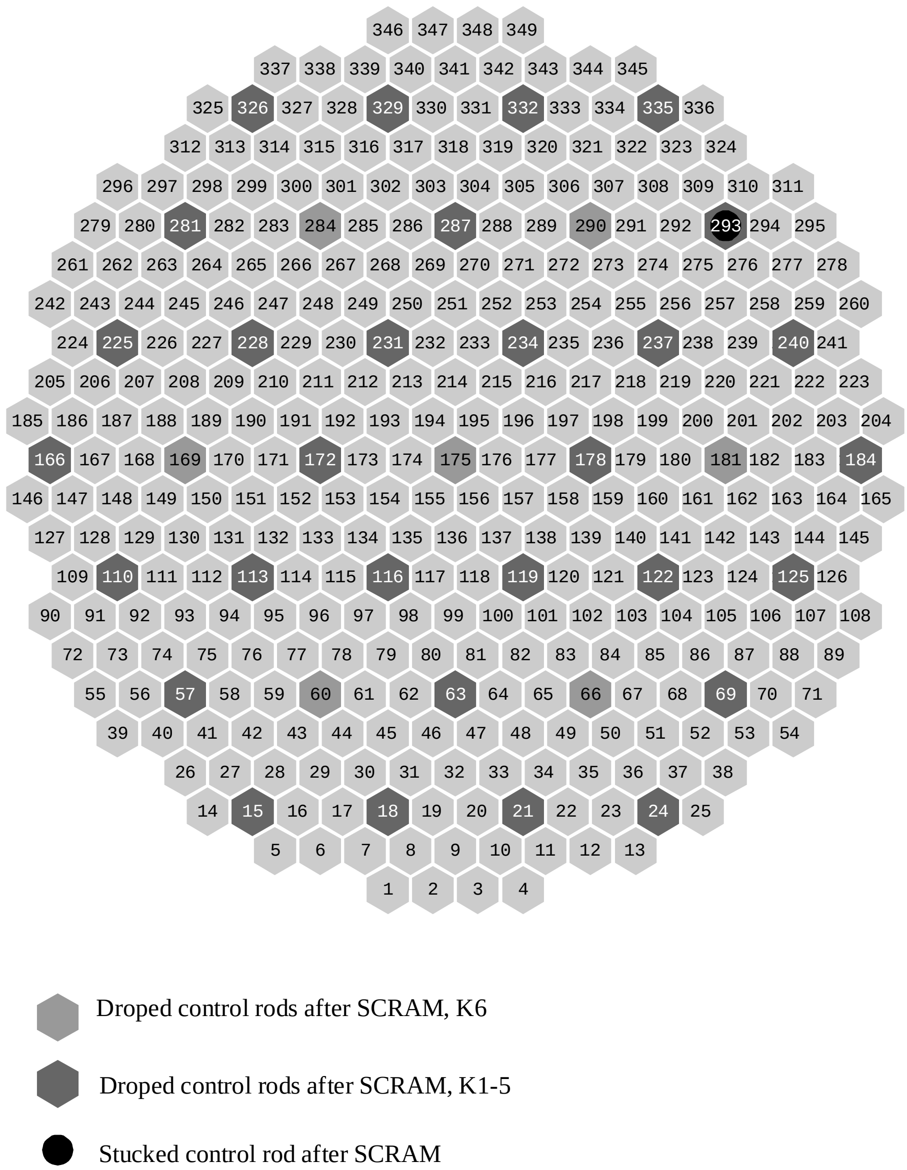

Fig. 3 Location of control rod groups in VVER-440 reactor core