MODELLING AND CALCULATIONS OF CRITICAL CORE CONFIGURATIONS OF THE ANNA ZERO-POWER REACTOR IN KENO-VI/SCALE

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ABSTRACT

Between 1961-1972 four critical assemblies were constructed at the Institute of Nuclear Research (now National Centre for Nuclear Research - NCBJ) in Swierk, near Warsaw known under the names: ANNA, MARYLA, AGATA and PANNA, and one exponential assembly HELENA. They were used for experiments in the three domains of nuclear reactor physics: static, kinetics and noise measurements. The International Handbook of Evaluated Criticality Safety Benchmark Experiments, prepared by participants of International Criticality Safety Benchmark Evaluation Project, contains 567 evaluations representing 4874 critical, near-critical, or subcritical configurations. Only one Polish zero-power reactor – AGATA, with its critical experiment, was benchmarked and documented in the Handbook 17 years ago. The current reactor physics team of NCBJ decided to continue this work which was divided into few steps: collecting all necessary data, modeling in a Monte Carlo code and calculations performance. All of the Polish zero-power reactors does not exist anymore, they were in operation till the mid-80s. These facilities were the main tools for the development of fundamental reactor physics in Poland and furnished some input data necessary in design and construction of a high flux research reactor and a power reactor.

The article presents the study of the ANNA critical assembly. ANNA was a graphite-light water moderated assembly with enriched fuel, designed primarily as a mock-up of the high-flux reactor. ANNA consisted of a core, top and bottom reflectors, all surrounded by the radial graphite reflector. The whole system was 240 cm high and its horizontal cross section, octagonal in form, has an equivalent radius of 137.5 cm. The core was composed of 21% U-235 enriched fuel elements immersed in vertical coolant channels passing through a graphite matrix. The coolant channels and the graphite matrix extend throughout the top and bottom reflectors. The first critical experiment was carried out in June, 1963. This paper contains detailed description of the core and the critical experiments (core configurations and critical masses), model prepared in KENO-VI (SCALE) code and results of calculations.

1. Introduction

Between 1961-1972 four critical assemblies, also called zero-power reactors, were constructed at the Institute of Nuclear Research (INR) at Swierk (near Warsaw) known under the names: ANNA, MARYLA, AGATA and PANNA. These facilities were the main tools for the development of fundamental reactor physics in Poland in the 60's and 70's and furnished some input data necessary for design and construction of a high flux and power reactor. They were used for experiments in the three domains of nuclear reactor physics: statics, kinetics and noise measurements. These experiments may be described as operation of a reactor with: steady state of neutron flux, changes in neutron flux due to external disturbance in composition of a reactor material and fluctuation of neutron flux due to stochastic character of a chain reaction in a range of plus/minus one percent. In addition, in 1963 a subcritical assembly HELENA was constructed for verification of some measurement methods in reactor physics and as a neutron source for checking of neutron detectors constructed at the INR.

The International Handbook of Evaluated Criticality Safety Benchmark Experiments, prepared by participants of International Criticality Safety Benchmark Evaluation Project, contains only one Polish zero-power reactor – AGATA, which was benchmarked by dr. K. Andrzejewski and dr. T. Kulikowska in 2000. Now this work is continued to document remaining Polish zero-power reactors and critical experiments carried out in the Institute.

This paper is focused only on the ANNA critical assembly and describes in details its structure, critical experiments, modeling and k-eff calculations using KENO-VI/SCALE Monte Carlo code with uncertainty and sensitivity study.

2. Polish zero-power reactors

At the Institute of Nuclear Research in Swierk four critical assemblies and an exponential one were put into operation:

- ANNA, first critical in June, 1963, was a graphite-light water moderated assembly with enriched fuel, designed primarily as a mock-up of the high flux reactor (more details are described in chapter 3),
- MARYLA, first criticality on 29th December, 1963, was a pool-type facility for investigation of the light water systems. Both the construction and the control system were flexible to such an extent that they could be easily adapted to various types and geometries of the critical water assemblies.
- HELENA, put into operation in November, 1963, was a natural uranium-graphite exponential assembly,
- P-ANNA, put into operation in March, 1972, was the changed ANNA assembly with the fast central zone filled with triangular lattice of fuel rods with natural uranium without moderator,
- AGATA, built in 1973-74 in connection with construction of the 30 MW multipurpose high-flux reactor MARIA. The AGATA assembly was used to check the main reactivity characteristics of the reactor lattice and instrumentation designed and constructed at the Institute.

2.1 MARYLA

MARYLA was a poll type reactor with power of 250 W. The reactor achieved the first criticality on 29 December, 1963. Its basic configuration consisted of 20 fuel elements surrounded by the graphite reflector. MARYLA was operated in two configurations. The MARYLA-1 version was designated for testing the possibility of the EWA reactor operation with variable power and its power increase by adding the graphite reflector. The EWA research reactor was operated at the Institute of Nuclear Research in Swierk between 1958 and 1995. It was one of WWR-S type reactors produced in the former Soviet Union.

Study of the WWR-S type reactors power increase possibility by applying new type of fuel – WWR-SM – was performed using the critical assembly MARYLA-2. It was placed in the mid-depth of the same pool as the MARYLA-1 assembly. The first critical experiment was carried out on 12 February, 1967. 181 different configuration of the core were investigated The results of these experiments were used to increase power of the WWR-S type reactors up to 10 MW. The MARYLA assembly was dismantled in the mid 70's.

2.2 HELENA

HELENA was a subcritical assembly built to investigate the parameters of natural uranium-graphite reactor lattices. Its fuel consisted of Al-canned uranium slugs 300 mm long and 25 mm thick which form rod-type fuel elements of the required length. Due to the flexibility of the assembly constructed of small graphite blocks, some fifteen various lattices could be investigated. Some effects connected with air gaps, core anisotropy and influence of assembly dimensions on accuracy of buckling determination were studied. The assembly was also used for educational purposes.

ANNA, MARYLA and HELENA were used for investigations in reactor physics, both for our national program and for the cooperative program of the NPY-Project – a cooperative research program in reactor physics between Norway, Poland, Yugoslavia and IAEA.

2.3 P-ANNA

In 1971, the core configuration of the ANNA critical assembly was completely changed. The new assembly was called P-ANNA, where the inner part of the core was a zone of natural uranium and was known as the fast zone. This zone consisted of three tones of metallic natural uranium. The outer part of P-ANNA filled with enriched uranium was the thermal zone. Several experimental quantities such as critical mass, radial distributions of fission densities, reaction rates for activation detectors (threshold, resonance and thermal detectors), spectral index for fission detectors and axial distribution of reaction rate and resulting value of axial buckling were investigated. The experimental results were compared with critical and spectral calculations performed with numerical codes MIXER and EWA-I developed at the INR.

3. ANNA zero power reactor [2,3,4,5]

3.1 General information

The ANNA critical assembly (Fig. 1) was built in 1963, originally as a model of the Second Polish Research Reactor designed for material testing and basic investigations with high neutron flux. The reactor was to correspond to the Soviet high-flux reactor of the RFT type (Moscow). Then, in the next years the concept of the reactor structure changed. The ANNA critical assembly remained the subject of the reactor physics experiments. In the course of work certain experimental evidence was accumulated on several configurations of the assembly. It was reasonable to carry out a unified analysis of the assembly and its physical properties, and to give interpretation of the available data.

Unfortunately till today not all documentation – technical drawings, internal reports, descriptions of the experiments, etc., survived. That is why the detailed geometry, material composition and especially its uncertainties are unknown. Many problems with critical core configurations were met during its modeling. In available literature there is no all of the information how these experiments were carried out in details.

3.2 Description of the facility

ANNA consisted of a core, top and bottom heterogeneous reflectors, all surrounded by a radial graphite reflector. The whole system was 240 cm high, and its horizontal cross section, octagonal in form, had an equivalent radius of 137.5 cm. The core was composed of 21% U-235 enriched fuel elements immersed in vertical coolant channels running through a graphite matrix. The coolant channels and the graphite matrix extend throughout the top and bottom reflectors.

Both core and reflector along their x and y axes were provided for measuring purposes with a system of special vertical channels in which various flux-mapping detectors could be placed. The core of the assembly was surrounded by vertical channels in which control system detectors were located. One of the fuel elements was equipped with a driving mechanism which introduced the element into the core from below during the critical approach or the start-up of the assembly.

This movable fuel element as well as four safety rods and one control rod were included into the control and safety system. Rapid removal of water from all fuel channels was also possible. The control and safety system was based on two pulse channels, three linear and two logarithmic current channels.

The assembly and a part of the water filling system (water dump tank and header) were surrounded by 60 cm thick heavy concrete biological shield.

The vertical cross section of the ANNA assembly is presented in Fig. 2.

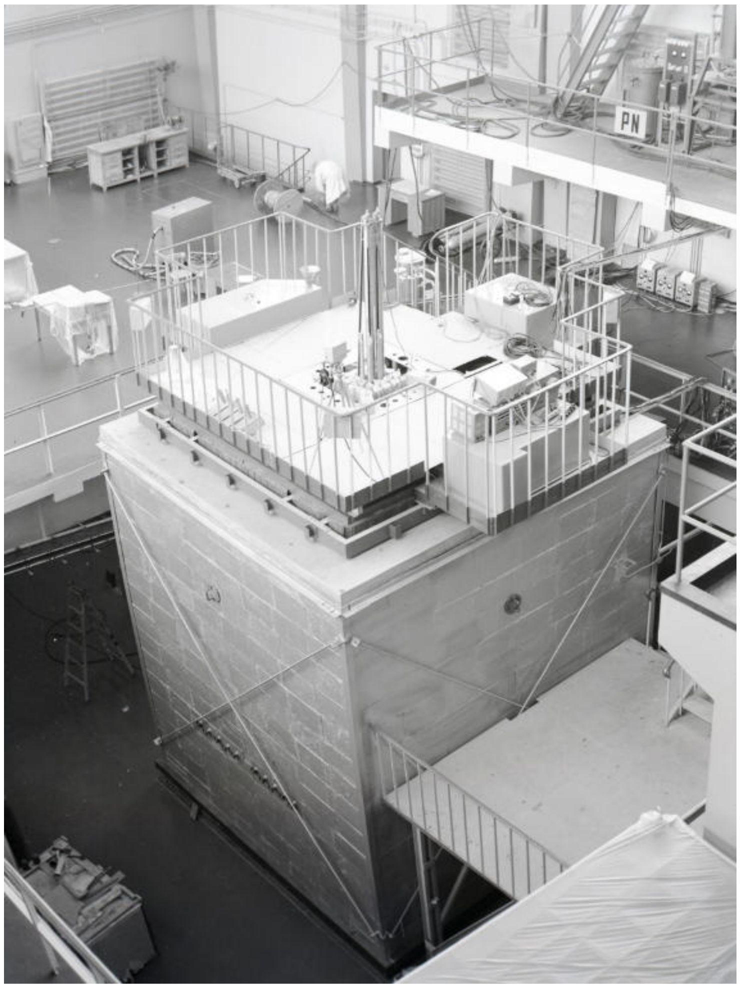

Fig. 1 ANNA critical assembly at the Institute of Nuclear Research at Swierk

<u>3.3 Critical experiments</u>

Many critical configurations of the ANNA reactor core were performed during its operation. But this paper contains the study of 11 of them, described in the article [2] and [3].

Four of the configurations, denoted by P01-P04, and called here symmetric systems, had identical top and bottom reflectors. Their core heights amounted to 100 cm corresponding to the full fuel element active length. These systems differed between themselves in respect of the lattice pitch – P01, P03: 14 cm, P02, P04: 19.8 cm, and of whether the fuel elements contained water – in the case of P01, P02, or air – P03, P04. In the case of P03 system, with channels containing air, the criticality could not be attained with 30 fuel elements available, therefore coolant channels of 2.8 fuel elements had to be filled with water. In the top reflector four safety rods (diameter 3.4 cm, B_4C – boron density 1.9 g/cm^3) were inserted down to 5 cm above the top reflector-core interface.

Next seven critical configurations (P05-P11) of 14 cm lattice pitch and with water in coolant channels were mounted without top reflector. In the sequel they were called asymmetric systems. They were distinguished by various numbers of fuel elements and by correspondingly different active core heights. The active core heights were determined by the depths of fuel elements immersion into the critical system below its upper surface. In the paper [3] we can find also the information that the change of height of the critical assembly could be achieved only by changing the water level. Consequently, it would be necessary to find a region near the edge of the graphite which is adequate for measuring the differential reactivity. In this region, on the one hand, increase of the water level in the technological channels should be equivalent approximately to the increase of the total height of the reactor and, on the other hand, the effect of the dry uranium-graphite layer located above the critical level of the water should be negligibly small. In practice, all fuel elements were suspended in the core at a height such that the critical level of the water in the technological channel was always lower than the upper surface of the graphite stack by approximately 10 mm. This reserve was essential for measuring the differential reactivity and simultaneously, as follows from special investigations, it had no very strong effect on the measured results.

All of the configurations are shown on the Fig. 3. The main structural features of these systems are given in Table 1.

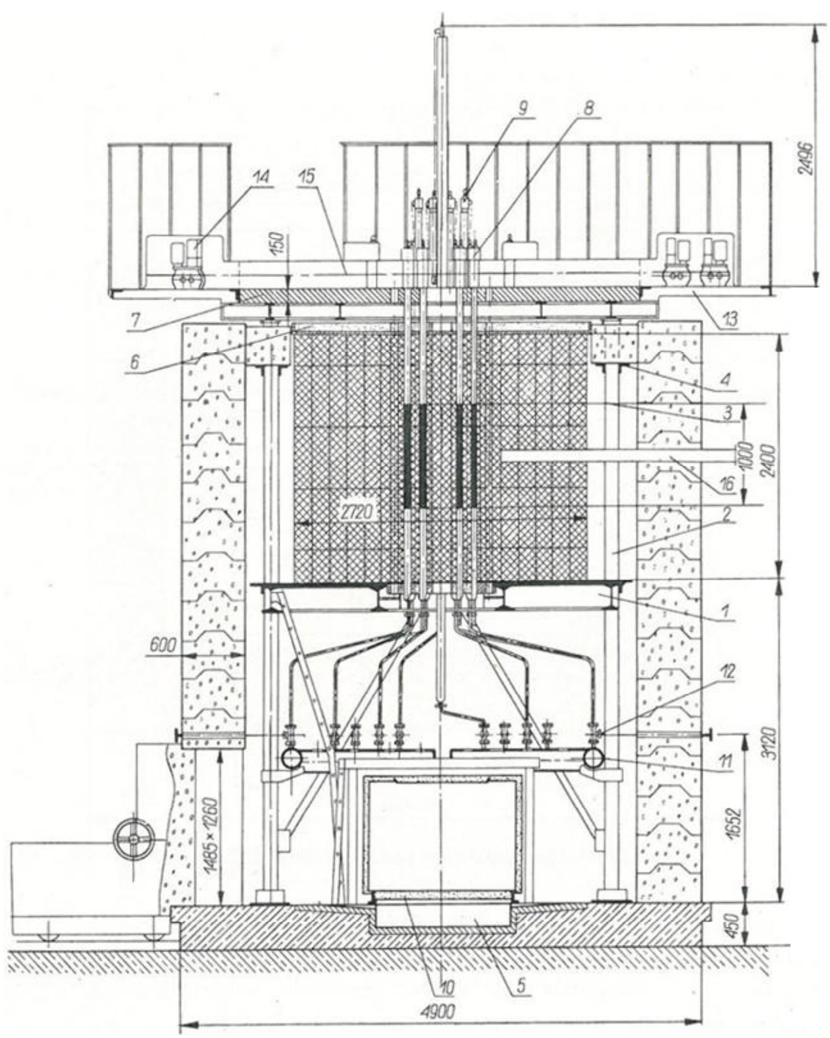

Fig. 2. Vertical cross section of the ANNA assembly [3]
1-4 -framework, 5 - cavity, 6 - boron-paraffin shield, 7 - cast iron plate, 8 - fuel channel, 9 - safety rod mechanism, 10 - water dump tank, 11 - header, 12 - valve, 13 - upper platform, 14 - safety rod motor and gearbox, 15 - steel cable, 16 - pulsed neutron source channel

Table 1 Structural features of the ANNA critical systems

No.	Lattice pitch [cm]	Channel filled wih	h_c [cm]	h_1 [cm]	h_2 [cm]	Crit. mass [g U-235]	No. of fuel elements	Core area** [cm^2]
P01	14.0	water	100	70	70	2825	11.3	2237
P02	19.8	water	100	70	70	3375	13.5	5292
P03	14.0	air	100	70	70	7450	27.0/2.8*	5900
P04	19.8	air	100	70	70	6000	24.0	9409
P05	14.0	water	96.5		143.5	3135	13.0	2548
P06	14.0	water	83.0		157.0	3320	16.0	3136
P07	14.0	water	77.0		163.0	3465	18.0	3528
P08	14.0	water	71.5		168.5	3575	20.0	3920
P09	14.0	water	69.5		170.5	3649	21.0	4116
P10	14.0	water	64.5		175.5	4031	25.0	4900
P11	14.0	water	60.5		179.5	4386	29.0	5684

* 27.0 air channels, 2.8 water channels,
** Core area = area of the horizontal cross section of the core,
h_c – core height,
h_1 – top reflector height,
h_2 – bottom reflector height.

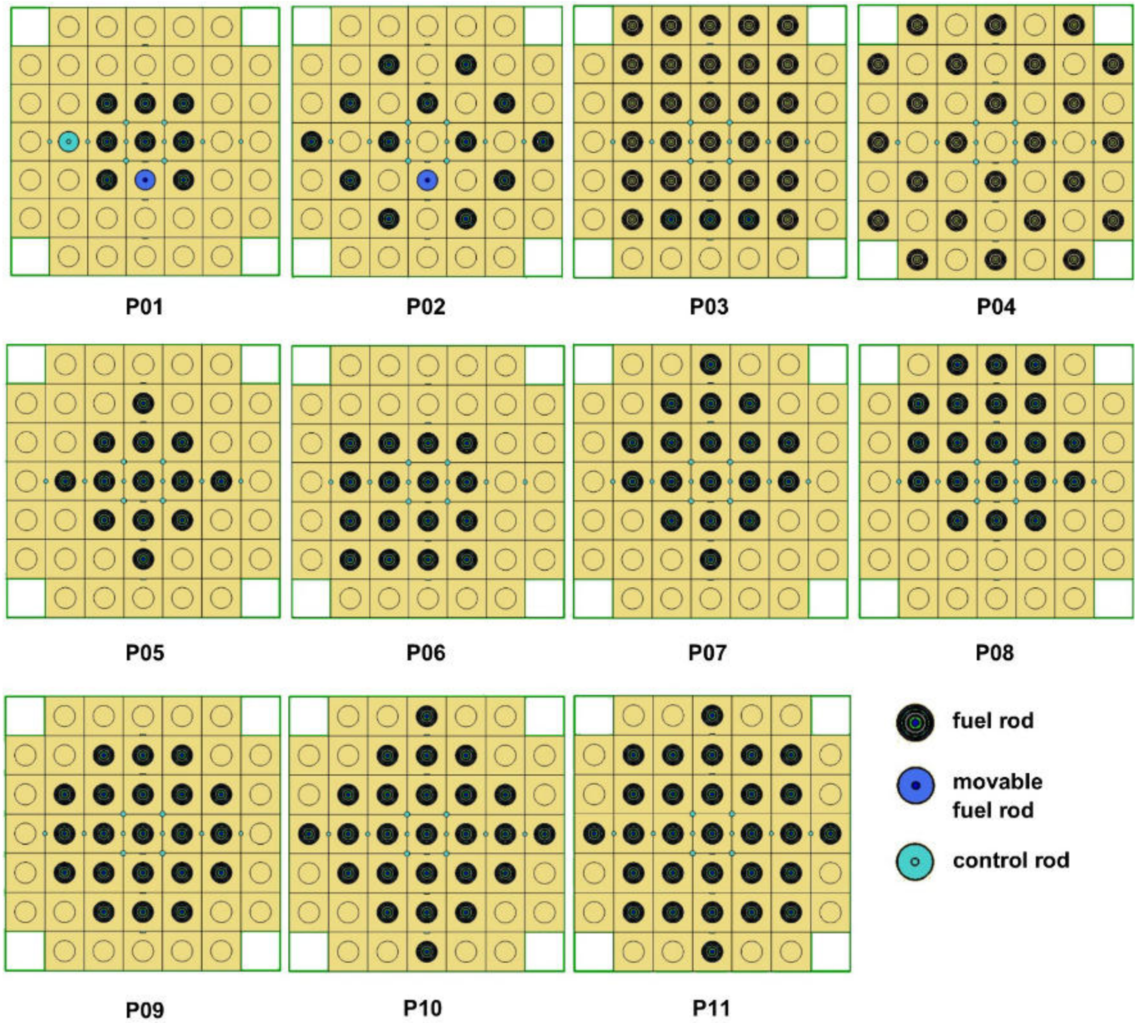

Fig 3. Critical configurations of ANNA zero power reactor

3.4 GEOMETRY AND MATERIAL COMPOSITION

Data collected below come from:

- Reviewed articles: [2], [3], [4], [5],
- Internal reports of INR: [7], [8], [9], [10], [11], [12], [13],
- A few technical drawings found in the Institute,
- Accounts given by the staff engaged in the experiments: prof. E. Józefowicz and dr. J. Kubowski.

Core: The fuel elements were placed in graphite blocks and form a square lattice of 14.0 cm pitch – for systems P01, P03, P05-P11, or of 19.8 cm pitch – for systems P02, P04. The cylindrical fuel element consisted of three coaxial fuel tubes canned in aluminium, of a central supporting aluminium tube and of an outer aluminium tube. The inner and outer radii of the tubes are the following:

	IR (cm)	OR (cm)
1. Supporting Al tube	0.700	0.800
2. First fuel tube:		
Al cladding	1.300	1.385
fuel	1.385	1.615
Al cladding	1.615	1.700
3. Second fuel tube:		
Al cladding	2.200	2.285
fuel	2.285	2.515

Al cladding	2.515	2.600
4. Third fuel tube:		
Al cladding	2.900	2.985
fuel	2.985	3.215
Al cladding	3.215	3.300
5. Outer Al tube	3.590	3.740
6. Channel in the graphite block 3.750		

<u>Graphite:</u> blocks of 1.70 g/cm^3 density.
<u>Fuel:</u> 38.5 weight % $U0_2$ enriched to 21% of U-235, the rest: Al.

Material composition:

Fuel	U-235	0.0006336
	U-238	0.002352
	O	0.005972
	Al	0.04764
Cladding	Al	0.06034
Moderator		
in the coolant channel	H	0.06674
(in the case of P01, P02, P05-P11)	O	0.03337
In the graphite block	C	0.08533

<u>The radial reflector:</u>
The radial reflector was composed of graphite blocks of 1.70 g/cm^3 density. Atomic density (unit: 10^{+24} cm^{-3}): C 0.08533

4. DESCRIPTION OF THE MODEL

The calculations were performed using KENO-VI/SCALE-6.2.1 code. **S**tandardized **C**omputer **A**nalyses for **L**icensing **E**valuation – SCALE computer system, developed at Oak Ridge National Laboratory, is widely used for criticality safety analyses. KENO-VI is a 3-D generalized geometry Monte Carlo code that allows for versatile modelling of complex geometries and has a capability to perform calculations in the multigroup approach and in the continuous energy mode. A horizontal and vertical cross section of the benchmark model is shown in Fig. 4 and Fig 5.

The model includes:

- core, i.e. fuel elements without upper and lower ends, graphite blocks, borated-paraffin layer and iron plate and air gap between them, experimental channels,
- inserted control rods,
- graphite lateral reflector,
- channels with safety rods.

There was no information or the information was unreliable related to some geometrical and material data. That is why the following simplifications and assumptions were made to the benchmark model:

- heights of paraffin, air and iron layer, it was assumed:
    - height of paraffin layer – 10 cm,
    - height of air gap – 15.5 cm,
    - height of iron plate – 15 cm.
- material composition of borated paraffin – the content of boron was unknown, but the composition was assumed from [AK] as:

  h-1 6.5282e-02

c	2.5842e-02
o-16	1.1530e-02
b-10	7.6098e-04
b-11	3.0823e-03

- whether all channels in the graphite blocks in the core were filled with plugs – according to the stuff engaged in the experiments, some cases could have empty channels in the graphite blocks. In the model it was assumed that all channels in the graphite blocks had plugs,
- position of the safety rods in the cases P01-P04, assumption in this model – 5 cm above the reactor core,
- impurities of graphite were unknown. Because the AGATA critical assembly studied in the paper [6] had the graphite blocks, which was manufactured in the same company in USSR, that is why its composition of impurities of graphite was assumed in this work:

c	0.08523511
b-10	1.88446e-09
b-11	7.58519e-09
cd	1.821466e-09

- it was assumed that inner and outer supporting Al tube had length 280.5 cm, so from the bottom of the graphite blocks to the top of iron plate,
- the concrete blocks surrounded reactor, support structure and core support plate were not modeled.

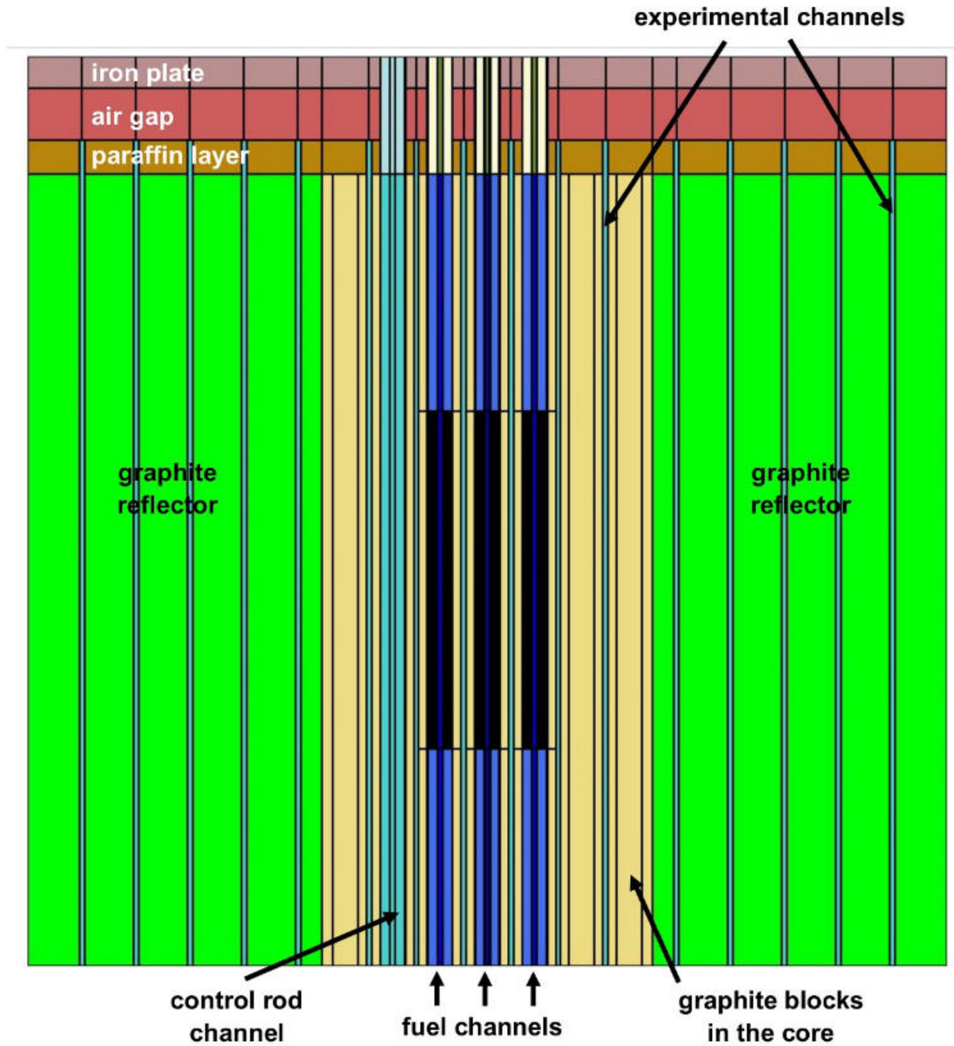

Fig. 4 Horizontal cross section of the KENO-VI/SCALE model of the ANNA critical assembly

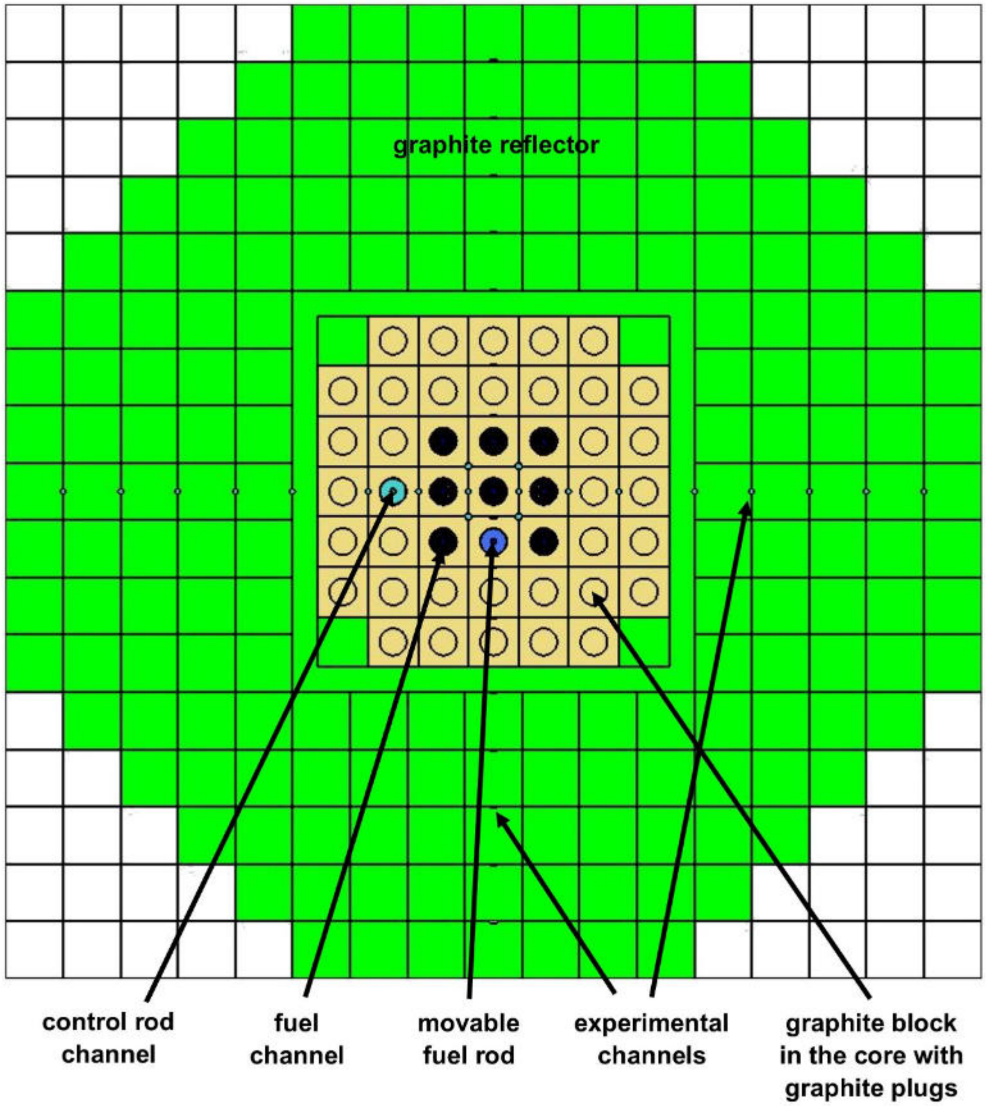

Fig. 5 Vertical cross section of the KENO-VI/SCALE model of the ANNA critical assembly

5. UNCERTAINTY AND SENSITIVITY STUDY

The total combined keff uncertainty – Δkeff and its components are given in Table 3. The composition of aluminum alloy used in the ANNA critical experiments are sown in Table 2.

Table 2. Composition of aluminum alloy

Source of Data	Polish Norms: PN-56/H-82160
Fe	≤0.30
Si	0.30
Fe +Si	≤0.45
Cu	0.015
Al	Up to 100%
Sum of impurities	0.50
Density [g/cm^3]	2.7 (PN/H-93669)

Due to the lack of specification sheet related to fuel rod, the needed data for uncertainty calculations was taken from the AGATA critical assembly [6] – fuel used in both reactor was manufactured by the same company in USSR.

Table 3. Total combined uncertainty

Parameter	Calculated variation	Calculated effect	Δkeff
Fuel density	-2.2%	-0.00518	0.016
	+2.2%	0.0161	
Uranium enrichment	-0.5%	-0.00692	0.0069
	+0.5%	0.00511	
Impurities of aluminum of clad of fuel	(Table 2)	-0.00032	0.00032
Fuel clad thickness	-0.2 mm	0.02426	0.027
	+0.2 mm	-0.0273	
Core height	-2 mm	-0.00068	0.00030
	+2 mm	0.0003	
Temperature	-5 °C	0.00033	0.00056
	+5 °C	-0.00056	
Total			**0.051**

6. RESULTS OF SAMPLE CALCULATIONS

Calculated keff values using geometry and material composition described in chapter 3 and 4 are given in Table 3. Calculated keff results for three cases are 3% higher than the benchmark-model value, and three cases are 1 to 1.5% high. Such a large deviation (higher than 3%) from value 1.0 relates to cases P01, P03 and P04. P01 was the first critical experiment and from the direct account given by dr. Kubowski, who was engaged in this project, we know that experiment's description was not complete – some channels in graphite blocks in reactor core could be filled with water. Unfortunately, the reason of high keff value for case P03 and P04, so cases for fuel channels filled with air, is unknown.

Table 3. Sample calculation results

Case Number	KENO-VI/SCALE (Continuous-energy ENDF/B-VII.1)
P01	0.93514 ± 0.00030
P02	1.01045 ± 0.00027
P03	1.05024 ± 0.00030
P04	1.07302 ± 0.00028
P05	1.01168 ± 0.00031
P06	1.01359 ± 0.00031
P07	1.00992 ± 0.00028
P08	1.00597 ± 0.00030
P09	1.00616 ± 0.00028
P10	1.00146 ± 0.00029
P11	1.00071 ± 0.00028

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