Conceptual Neutronics Design for a High-FluxMulti-purpose Research Reactor

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Abstract: The paper presents calculation results of conceptual design for a 10-MWt high-flux multi-purpose research reactor of a Research Centre for Nuclear Energy Science and Technology (RCNEST) of Viet Nam. The Russian low-enriched uranium VVR-KN fuel type of 19.75% ^{235}U was selected for this design. The main characteristics of the designed reactor core were investigated to confirm about its safety operation and utilization capability. The established each core configuration in 6 cycles was considered under safety conditions in criticality and shutdown margin evaluation, etc. The safety parameters as well as kinetics parameters will be used for the thermal hydraulics and safety analysis of each core configuration. After 6 operating cycles with different power levels and core configurations, the equilibrium core configuration was determined. The neutronics computer codes of MCNP6.1 and REBUS-MCNP6.1 linkage system were applied for the design including fuel burn-up calculation. The detailed calculation on neutron flux distribution at vertical irradiation positions for typical applications such as neutron activation analysis (NAA), radioisotope production (RI), neutron transmutation doping (NTD), etc. was carried out and the evaluation of neutron flux at horizontal neutron beam ports for material science studies and basic researches on nuclear physics was also given in this paper.

Keywords: *Research reactor, conceptual design, VVR-KN fuel, MCNP6.1 code, REBUS-MCNP6.1 system code*

1. INTRODUCTION

The 500-kWt Dalat Nuclear Research Reactor (DNRR) is an unique reactor in Vietnam at present, however, with its low power thatdoesn't meet demands of its utilization serving for socio-economic in medicine, industry, as well as for advanced researches in nuclear physics and material science [1]. The conceptual design for the new research reactor is a necessary preparation step for its construction to adapt the safety requirements and utilization characteristics of recent advanced research reactor projects in the world [2, 3, 4, 5]. As the safety is an important issue so design calculation should also follow the research reactor safety requirements of IAEA safety guidelines [6, 7]. MTR fuel type and heavy water reflector were used in the design of the reactor cores of [2, 3, 4]. The design of the reactor core configuration of [5] was also used MTR fuel type and heavy water reflector but without horizontal beam ports. Besides, in the framework of the collaboration between Vietnam Atomic Energy Institute and Korean Atomic Research Institute, the conceptual nuclear design for two models of multi-purpose research reactors were also performed using rod-type and MTR fuels, respectively [8].

In this work, preliminary analyses to support the design of the new research reactor using VVR-KN fuel, which has been used in the WWR-K research reactor in Kazakhstan [9], were performed using neutronics computer codes as MCNP6.1 [10] and REBUS-MCNP6.1 [11, 12]. The operation power for fresh core is about 6 MWt and the final working core will be achieved with about 27 fuel assemblies (FAs) with 8 tubes (FA-1) and 10 FAs with 5 tubes (FA-2) with beryllium blocks setting around the core in order to create a reflector. At this working core, the operation power of the reactor was expected to 10 MWt and as normally the fuel cycle was from 25 to 30 days. For conceptual design of the reactor core, safety requirements and utilization ability need to be completely evaluated. This report mainly shows the safety of the designed reactor core and physics characteristics.

The total fuel cycle of the designed reactor core consists of 6 cycles. Detailed neutronics calculation was conducted for each cycle at start-up phase. From cycle 1 to cycle 3, the operation power of the reactor was about 6 MWt. At cycle 4, the power was put up to 8MWt and then from cycle 5 to 6, the operation power was put into 10 MWt. At the last cycle number 6, the characteristics of the reactor core in neutronics and thermal hydraulics were emphasized.

In neutronics calculation, all physical parameters of each cycle were estimated such as control rod worth, reactivity feedback coefficient, integral control rod worth, kinetics parameters, power peaking factor. Burn-up of each cycle was calculated by using REBUS-MCNP6.1 linkage code and beryllium poisoning was also taken into account. Especially, the neutron flux distribution of each irradiation positions and horizontal beam tubes were evaluated to confirm about application ability of the designed research reactor.

PLTEMP/ANL code [13] was also applied for evaluation of thermal hydraulics parameters in steady state of each cycle to confirm that the safety limit of fuel should not be violated as recommendation from vendor's fuel catalog. The obtained parameters of thermal hydraulics calculation are maximum temperature of fuel cladding and coolant temperature, minimum onset nucleate boiling ratio (ONBR), heat flux as well as flow rate of coolant. The PARET/ANL[14] and RELAP5MOD3.3 codes [15] were also applied for transient and safety analysis of each core configuration.

2. REACTOR CORE DESCRIPTION

2.1. General

The reactor core loaded with the VVR-KN fuel was analyzed and the reactor core structure was designed to maximize application ability of the designed research reactor [16, 17]. The main components of the reactor consist of reactor core with 7 cm in diameter neutron trap at core center, 11 vertical channels for RI or NAA, 4 vertical holes of 30 cm in diameter for NTD, a reserved position for cold neutron source in the near future, and 4 horizontal beam tubes of 7.7 cm in radius for material science studies and basic researches. To create a good neutron field on the reflector, beside beryllium material, graphite blocks were added to the side for all vertical channels. To control the fission chain of the reactor, 9 control rods (CR) were used and divided into three groups: (1) 2 safety rods named AZ1and AZ2, which are always hung up while reactor operating and they have safety function; (2) 6 shim rods named KC1 to KC6, which are used for reactor power control and (3) 1 regulating rod named AR. In design calculation, the flexible arrangement of these CRs was available. The total length of absorption part of all CRs is about 64 cm that is enough to cover whole the reactor core. The thickness of the reflector was about 45 cm with 60 cm height. The beryllium rods were put around the reactor core in order to create an extra reflector and it is very easy for setting up additional irradiation channels by removing beryllium rods.

As the burn-up of beryllium reflector blocks increases during reactor operation so 8-tube fresh FAs are inserted into the core to compensate for the reactivity loss. From the beginning with fresh core, 17 FAs with 8 tubes and 9 FAs with 5 tubes and CRs were loaded to set up a first cycle. As a problem of safety related to thermal hydraulics such as temperature of fuel cladding, ONBR value, so the cycles 1, 2, 3 and 4 were calculated at power level of 6 MWt and in cycles 5 and 6 the power was set up to 10 MWt.

Heat removal from the reactor core is carried out by forced convection of light water with the downward direction through the core.

The purposes for design calculation were to find out the "equilibrium" core with optimization of loaded fuel number and other requirements of technological parameters such as flow rate, operation power and operation limit conditions under abnormal or transient situations. In this work, the calculation results mainly focused on reactor core characteristics but not on the reactor technological systems.

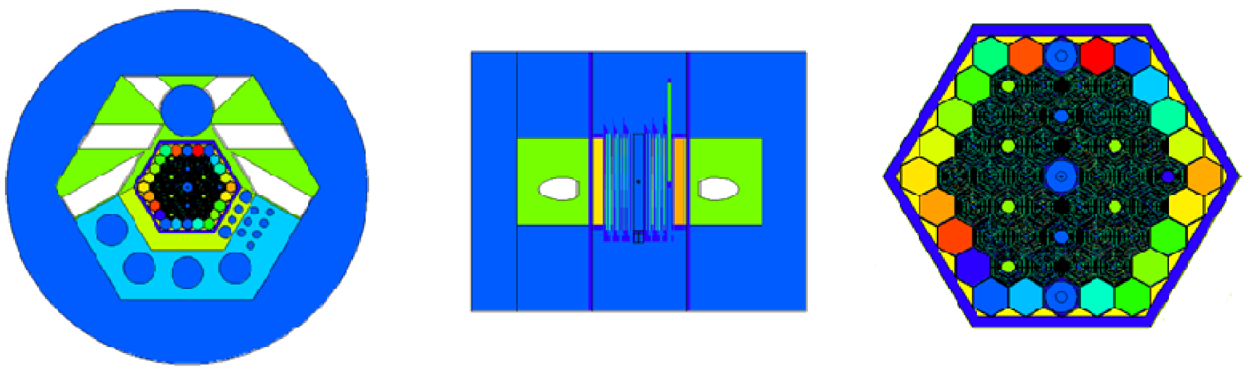

Fig. 1. Calculation model for new research reactor by MCNP code

2.2. VVR-KN Fuel

There are two types of LEUVVR-KN FAs named FA-1 and FA-2. FA-1 has 7 concentric tubular fuel elements (FE) of hexagonal cross section and an 8-th central cylindrical FE. There is a cylindrical structural tube interior to the 8-th FE. FA-2 has the same outermost 5 concentric tubular FE as in FA-1; interior to the FE is a cylindrical guide tube for CR. For safety and shim rods, B_4C is used asneutron absorption material with density of 1.69 g/cm^3 while regulating rod has stainless steel material for getting low worth with density of 7.8 g/cm^3. Dimensions of FAs are shown in **Fig. 2**. Corner rounding is 6.9 mm radius for outside of outermost FE, decreasing by 0.4 mm for each tube moving inward; inner corner rounding is 1.6 mm less than outer corner rounding for each FE. The ribs are actually trapezoid shape rather than the half circle implied by dimension "R1.5" in the figure.

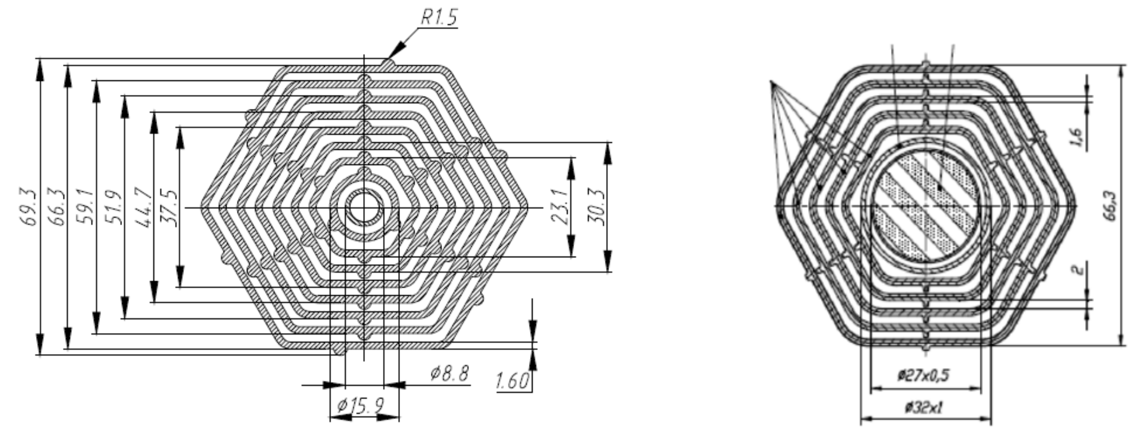

Fig. 2. LEU VVR-KN fuel assemblies with 8 and 5 tubes

The FEs areof 1.6 mm thick, consisting of 0.7 mm of fuel meat and 0.45 mm of cladding on each side. The fuel meat is UO_2-Al, enriched to 19.75% in U-235. The U-235 masses are 248.2 g in FA-1 and 197.6 g in FA-2; this yields a mean fuel density of about 2.8 g/cm^3 of uranium. Cladding and other structural items are made of the aluminum-alloy SAV-1. Ribs of 1.5 mm height provide stiffening of FE and help maintain 2 mm water gap between adjacent FE. The design of fuel meat is 0.6 m in length with a standard deviation of 0.002 m. In the analyses presented in this paper, the nominal dimensions and masses of the fuel were used.

2.3. Reactor core

The core loading for each cycle with number of FA-1, FA-2 and beryllium rods is shown in fully inserted while all safety rods are out and regulating rod is at center line of the reactor core. The number of CRs is constant for all cycles and can flexibly be re-arranged inside reactor core. The last two cycles were calculated to operate at power level of 10 MWt. The total number of FAs in the last core is 36 in which 27 of FA-1 and 9 of FA- 2. The reactor power for cycles 1, 2 and 3 is 6 MWt, cycle 4 is 8 MWt and cycles 5, 6 are 10MWt (see in **Table 1**). All the core loadings should have reactivity less than 1%Δk/k when all KCs full in, AZ1 full out, AZ2 full in and AR at center line.

Table 1. Number of FA-1, FA-2 fuels and beryllium rods in each cycle

Core	**17 FA-1 9 FA-2**	**17 FA-1 9 FA-2 9 Be rods**	**19 FA-1 9 FA-2 13 Be rods**	**23 FA-1 9 FA-2 13 Be rods**	**27 FA-1 9 FA-2 10 Be rods**	**27 FA-1 9 FA-2 22 Be rods**
Cycle	**1**	**2**	**3**	**4**	**5**	**6**

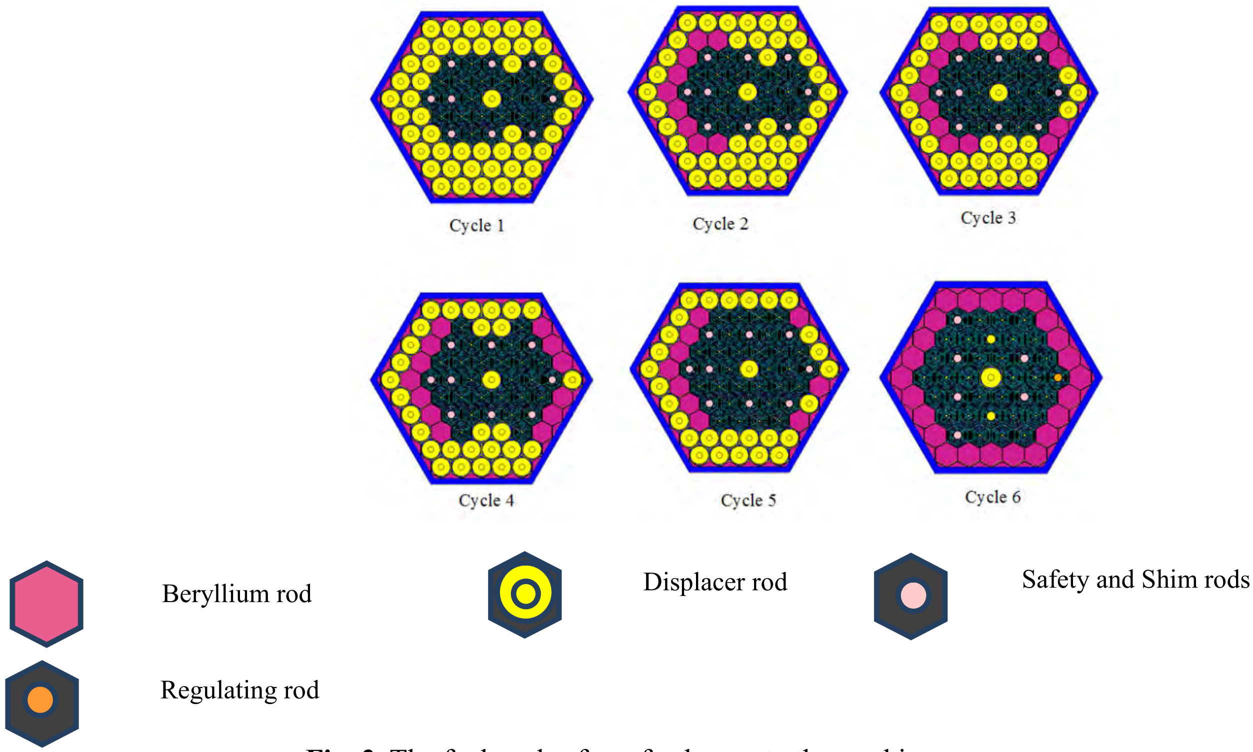

Fig. 3. The fuel cycles from fresh core to the working cores

Operation time and burn-up of 6 cycles is described in the **Table 2.** To assure about the nuclear safety, some parameters such as shutdown margin, excess reactivity at BOC, etc. were calculated.

Table 2. Core cycles and burn-up in operation time with reactivity

Core	**17+9+0 Be**	**17+9+9 Be**	**19+9+13 Be**	**23+9+13 Be**	**27+9+10 Be**	**27+9+24 Be**
Cycle	**1**	**2**	**3**	**4**	**5**	**6**
Power [MW]	6	6	6	8	10	10
Operation time [days]	28	110	82	67	41	86
Max burn-up FA-1[%]	4.271	20.303	30.487	37.762	45.565	56.738
Max burn-up FA-2[%]	4.428	20.561	30.567	40.170	45.675	56.064
Excess reactivity BOC [$]	8.149	11.233	9.722	10.433	9.271	13.753
Keff and reactivity[$] after 7-day cooling	1.04328 (5.437)	1.04233 (5.393)	1.04218 (5.396)	1.04205 (5.597)	1.04153 (5.632)	1.04036 (5.558)

The reactivity of Xenon poisoning of all cycles is about 4 to 4.5$ and average reactivity for

1 MWd burn-up is about 0.009 cent. The reactivity for experiments should be in range from 1.5$ to 2.7$. The excess reactivity of all cycles are about from 8.0$ to 13.7$ depending on loading patterns, that is enough for operation at least 25 to 30 days at power level of 10 MWt. Total operation days of the designed reactor core and 7 days of cooling in each cycle with excess reactivity changing are described in **Fig. 4**.

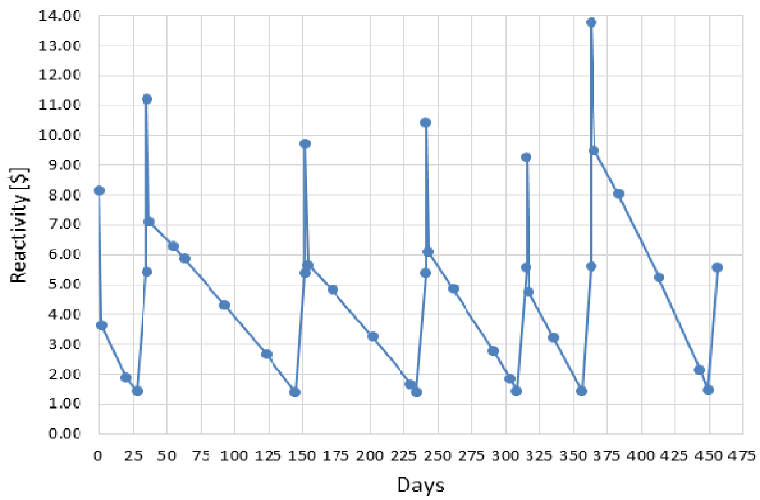

Fig. 4. Changing of excess reactivity following operation time and 7-day cooling in each operation cycle

3. CALCULATION RESULTS AND DISCUSSION

3.1.Neutronics parameters

In order to carry out steady state calculation, transients/accidents safety analysis, many neutronics parameters need to be prepared. The MCNP code and REBUS-MCNP linkage were used for this purpose.

The delayed neutron fraction β(i) and decay constant [λ(i)] for 6 groups plus effective delayed neutron fraction (β_eff) and prompt neutron generation time (Λ) are shown in **Table 3.**

Table 3. Kinetic parameters of 6 cycles

Core	**17+9+0 Be**	**17+9+9 Be**	**19+9+13 Be**	**23+9+13 Be**	**27+9+10 Be**	**27+9+24 Be**
Cycle	**1**	**2**	**3**	**4**	**5**	**6**
Delayed neutron fraction						
β(1)	0.00024	0.00024	0.00026	0.00024	0.00022	0.00021
β(2)	0.00123	0.00132	0.00126	0.00125	0.00128	0.00114
β(3)	0.00132	0.00126	0.00126	0.00116	0.00117	0.00109
β(4)	0.00341	0.0034	0.00336	0.00323	0.00326	0.00325
β(5)	0.00109	0.00095	0.001	0.00099	0.00084	0.00095
β(6)	0.00034	0.00036	0.00035	0.00034	0.00031	0.00033
β_eff	**0.00763**	**0.00753**	**0.00750**	**0.00721**	**0.00708**	**0.00698**
Decay constant						
λ (1) [1/s]	0.01249	0.01249	0.01249	0.01249	0.01249	0.01249
λ (2) [1/s]	0.03181	0.0318	0.03177	0.03175	0.03174	0.03172

λ (3) [1/s]	0.10947	0.10946	0.10945	0.10944	0.10945	0.10944
λ (4) [1/s]	0.31741	0.31741	0.31744	0.31744	0.31746	0.31745
λ (5) [1/s]	1.35292	1.35291	1.35253	1.35191	1.35089	1.34988
λ (6)[1/s]	8.66685	8.66877	8.67346	8.66992	8.66416	8.65643
Prompt neutron life time						
Λ[μs]	**43.04386**	**45.92251**	**47.58324**	**50.7326**	**47.52067**	**64.43571**

There are three types of CRs.2 safety rods (AZ1 and AZ2) are fully withdrawn from the core during reactor operation, they fall into the core due to gravity in response to a scram signal to terminate the nuclear chain reaction. 6 shim rods (KC1 through KC6) are partially withdrawn from the core during normal operation and are adjusted during operation to maintain criticality, these rods also fall into the core due to gravity in response to a scram signal. 1 automatic rod (AR) is partially withdrawn from the core during normal operation and its drive motor is attached to a logic circuit used to maintain (or make programmed adjustments to) power, it does participate in scram (but this small additional worth is ignored in the transient calculations). The reactivity worth of CRs is depicted in the **Table 4**.

Table 4. Control rod worth [$]

Core	**17+9+0 Be**	**17+9+9 Be**	**19+9+13 Be**	**23+9+13 Be**	**27+9+10 Be**	**27+9+24 Be**
Cycle	**1**	**2**	**3**	**4**	**5**	**6**
AZ1	3.352	2.596	2.737	2.664	3.068	2.938
AZ2	3.369	2.561	2.689	2.601	2.962	3.011
All AZ	5.297	5.613	5.720	5.704	6.962	6.632
AR	0.496	0.382	0.475	0.673	0.439	0.566
KC1	1.691	2.232	1.702	1.464	1.714	1.656
KC2	2.070	1.338	2.143	2.887	2.104	2.860
KC3	1.049	1.593	1.137	0.881	0.918	3.592
KC4	1.689	2.199	1.657	1.512	1.738	2.946
KC5	2.058	1.336	2.085	2.765	2.230	3.579
KC6	3.383	3.354	2.537	2.090	2.112	1.859
Shutdown margin, Keff	0.97340	0.97579	0.95823	0.97793	0.97695	0.97722
Criticality condition, Keff	0.99450	0.9950	0.97839	0.99662	0.99437	0.99773

Note: + Shutdown margin is defined as all KCs full in, AZ1 full out, AZ2 full in and AR at center line
+ Criticality condition: k-eff < 1.0 when all KCs full in, 2 AZs full out and AR at center line.

In safety analysis, the response time of the reactor control system was assumed of about 0.3 s while the drop time of 2 AZ rods fully into the reactor core is about 0.6 s. For withdrawal of a shim rod, the velocity of moving is about 0.4 cm/s. In all six core configurations, KC rod with the highest worth was calculated.

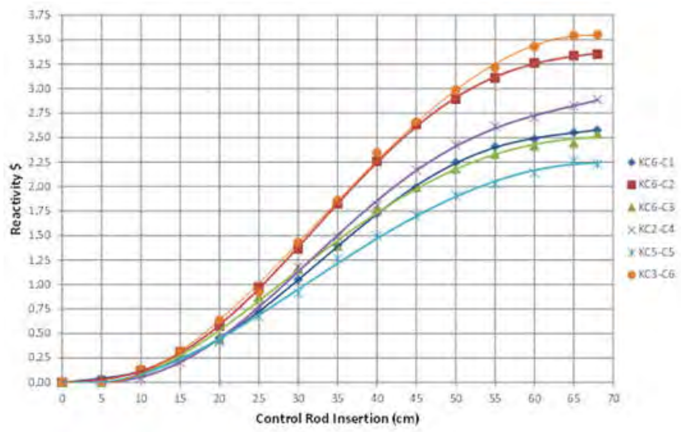

Fig. 5. Highest control rod worth as function of insertion for all cycles

The integral of highest worth KC at each cycle was calculated with 5 cm moving up each step and the results are shown in **Table 5**.

Table 5. Worth [$] versus withdrawal [cm] for maximum worth shim rod

Core	**17+9+0 Be**	**17+9+9 Be**	**19+9+13 Be**	**23+9+13 Be**	**27+9+10 Be**	**27+9+24 Be**
Cycle	**1**	**2**	**3**	**4**	**5**	**6**
Max. shim rod	KC6	KC6	KC6	KC2	KC5	KC3
Withdrawal (cm)						
0	0	0	0	0	0	0
5	0.0316	0.0187	0.0013	0.0180	0.0198	0.0099
10	0.1237	0.1185	0.0936	0.0416	0.1048	0.1314
15	0.2314	0.3124	0.2992	0.2009	0.2461	0.3175
20	0.4529	0.5758	0.4496	0.4274	0.4111	0.6336
25	0.7166	0.9710	0.8751	0.7863	0.6727	0.9232
30	1.0506	1.3624	1.1559	1.1788	0.9109	1.4295
35	1.3970	1.8216	1.3884	1.4864	1.2611	1.8550
40	1.7237	2.2583	1.7688	1.8115	1.4959	2.3451
45	1.9900	2.6320	1.9952	2.1700	1.6981	2.6576
50	2.2463	2.8926	2.1807	2.4262	1.9080	2.9850
55	2.4077	3.1078	2.3334	2.6148	2.0265	3.2137
60	2.4838	3.2601	2.4173	2.7029	2.1393	3.4295
65	2.5535	3.3387	2.4444	2.8242	2.2629	3.5438
68	2.5776	3.3539	2.5373	2.8867	2.2300	3.5519

Scram reactivity is as function of CR insertion in each cycle with two cases: A (AZ2+KC+AR) and B (AZ2+AR+KC-KC6) for cycles 1 to 3. CR insertion of only 10 to 15 cm is required to insert more than 1 $ of reactivity, thus leading to stop the nuclear chain reaction in all transients.

Table 6. Scram reactivity inserted [$] as a function of position of CRs

CYCLE1			CYCLE2			CYCLE3		
Pos. [cm]	Case A	Case B	Pos. [cm]	Case A	Case B	Pos. [cm]	Case A	Case B
0	0.000	0.000	0.00	0.000	0.000	0.00	0.000	0.000
7.5	-0.773	-0.573	6.50	-0.119	-0.084	8.50	-1.312	-0.543
15	-1.167	-0.958	13.00	-0.469	-0.266	17.00	-1.889	-0.947
24.5	-1.692	-1.477	20.00	-0.750	-0.562	25.50	-2.452	-1.255
34	-2.311	-2.068	27.00	-1.268	-1.042	34.00	-3.001	-1.606
51	-3.328	-3.127	34.00	-1.680	-1.478	44.00	-3.655	-2.137
68	-3.585	-3.368	42.50	-2.249	-2.061	54.00	-4.149	-2.721
			51.00	-2.734	-2.484	68.00	-4.338	-3.206
			59.50	-2.931	-2.751			
			68.00	-2.970	-2.761			

CYCLE4			CYCLE5			CYCLE6		
Pos. [cm]	Case A	Case B	Pos. [cm]	Case A	Case B	Pos. [cm]	Case A	Case B
0.00	0.000	0.000	0.00	0.000	0.000	0.00	0.000	0.000
6.30	-0.677	-0.593	7.00	-0.858	-0.711	4.00	-0.366	-0.308
12.60	-1.085	-0.991	14.00	-1.293	-1.059	8.00	-0.551	-0.455
19.60	-1.488	-1.294	24.00	-1.883	-1.580	14.00	-0.844	-0.732
26.60	-1.875	-1.640	34.00	-2.625	-2.255	24.00	-1.418	-1.334
34.00	-2.436	-2.163	44.00	-3.381	-2.885	34.00	-2.169	-2.068
44.00	-3.014	-2.737	54.00	-3.947	-3.323	44.00	-2.811	-2.768
54.00	-3.481	-3.214	68.00	-4.144	-3.526	54.00	-3.357	-3.229
68.00	-3.688	-3.331				68.00	-3.538	-3.479

The reactivity feedback coefficients associated with the change of coolant and fuel temperature and coolantdensity as well are shown in **Table 7**. All the reactivity coefficients are negative for the cycle in eachof the different core configurations. It is noted that for all cores, the lateral reflector temperature (either water or beryllium) was considered to be equal to room temperature.

Table 7. Temperature and density feedback coefficients

Core	**17+9+0 Be**	**17+9+9 Be**	**19+9+13 Be**	**23+9+13 Be**	**27+9+10 Be**	**27+9+24 Be**
Cycle	**1**	**2**	**3**	**4**	**5**	**6**
Coolant Temp [$/K]						
294<T<350	-1.14184E-02	-1.29489E-02	-1.32460E-02	-1.38437E-02	-1.41585E-02	-1.66078E-02
350<T<400	-1.24056E-02	-1.41191E-02	-1.4268E-02	-1.52266E-02	-1.44767E-02	-1.76637E-02
294<T<400	-1.18841E-02	-1.35009E-02	-1.36407E-02	-1.44960E-02	-1.43086E-02	-1.71058E-02
Coolant Density [$/%]						
0 – 5%	-4.86969E-01	-4.61866E-01	-4.55067E-01	-4.36385E-01	-4.54551E-01	-3.72373E-01

Fuel Temp [$/K]						
294<T<600	-3.01378E-03	-3.01573E-03	-3.21777E-03	-3.39960E-03	-3.56441E-03	-3.53462E-03

3.2. Reactor core power

The reactor core power, the power of the hottest fuel assembly and the power peaking factors (the results of multiplying local power peaking factor, radial and axial power peaking factors) are shown in **Table 8** for a total core power of 6 MWt for cycles 1 to 3, 8 MWt for cycle 4 and 10 MWt for cycles 5 and 6. In general, total power peaking factor defines as result of multiplication of local peaking factor inside FA, relative radial power of FA in all the reactor core configurations and axial power of the FA. Peak FA power occurs in core at position 6-5 in all of these cores; peak FA power is 0.409 MWt in cycle 1 and decreases in later cycles. Calculations for power peaking factor of all cycles were performed by MCNP code at critical status of BOC each cycle.

Table 8. Power in each fuel assembly

Core	**17+9+0 Be**	**17+9+9 Be**	**19+9+13 Be**	**23+9+13 Be**	**27+9+10 Be**	**27+9+24 Be**
Cycle	**1**	**2**	**3**	**4**	**5**	**6**
Power [MW]	6	6	6	8	10	10
Max. Power [MW]	0.4780	0.4587	0.4360	0.5099	0.5353	0.5035
Local power peaking	1.7432	1.7520	1.7979	1.7150	1.7287	1.6751
Max. Radial	1.4685	1.4020	1.3717	1.4271	1.338	1.346
Max. Axial	1.2775	1.2679	1.2376	1.2189	1.2051	1.2561
Total power peaking factor	**3.2703**	**3.1143**	**3.0521**	**2.9832**	**2.7874**	**2.8321**

Because the number of FA-1 in loading scheme for cycles 3 to 5 were increased, the total power in absolute value and total power peaking factor were decreased. So condition for operation as well as for transients will be satisfied in safety. Power distribution in axial direction of all cycles is depicted very detail in **Table 9**.

Table 9. Power distribution in axial direction of hottest channel

Core	**17+9+0 Be**	**17+9+9 Be**	**19+9+13 Be**	**23+9+13 Be**	**27+9+10 Be**	**27+9+24 Be**
Cycle	**1**	**2**	**3**	**4**	**5**	**6**
Position (cm) from top to bottom						
1	0.6340	0.6490	0.6668	0.7120	0.7335	0.6589
3	0.5671	0.5808	0.5826	0.6230	0.6459	0.6006
5	0.6214	0.6366	0.6340	0.6743	0.6946	0.6603
7	0.7035	0.7154	0.7107	0.7521	0.7712	0.7433

9	0.7845	0.7970	0.7889	0.8347	0.8535	0.8252
11	0.8613	0.8774	0.8697	0.9186	0.9416	0.9050
13	0.9343	0.9399	0.9046	0.9250	0.9144	0.9658
15	1.0036	1.0063	0.9717	0.9937	0.9855	1.0305
17	1.0561	1.0644	1.0274	1.0476	1.0356	1.0842
19	1.1135	1.1161	1.0778	1.0937	1.0764	1.1304
21	1.1582	1.1607	1.1218	1.1350	1.1151	1.1705
23	1.1972	1.1988	1.1628	1.1742	1.1526	1.2031
25	1.2303	1.2250	1.1747	1.1667	1.1291	1.2241
27	1.2590	1.2471	1.2010	1.1903	1.1553	1.2430
29	1.2734	1.2607	1.2192	1.2054	1.1698	1.2526
31	1.2734	**1.2679**	1.2291	1.2122	1.1775	**1.2561**
33	**1.2775**	1.2662	1.2324	1.2118	1.1766	1.2521
35	1.2686	1.2572	1.2274	1.1990	1.1664	1.2395
37	1.2550	1.2431	**1.2376**	**1.2189**	**1.2051**	1.2285
39	1.2241	1.2163	1.2169	1.1931	1.1794	1.1991
41	1.1922	1.1828	1.1913	1.1630	1.1534	1.1638
43	1.1515	1.1400	1.1586	1.1265	1.1194	1.1250
45	1.0970	1.0892	1.1139	1.0795	1.0772	1.0763
47	1.0371	1.0291	1.0556	1.0164	1.0145	1.0155
49	0.9662	0.9708	1.0322	1.0215	1.0583	0.9582
51	0.8908	0.8932	0.9532	0.9376	0.9729	0.8808
53	0.8134	0.8101	0.8707	0.8536	0.8902	0.7982
55	0.7256	0.7285	0.7877	0.7727	0.8089	0.7153
57	0.6716	0.6731	0.7337	0.7221	0.7596	0.6569
59	0.7586	0.7573	0.8458	0.8259	0.8668	0.7371

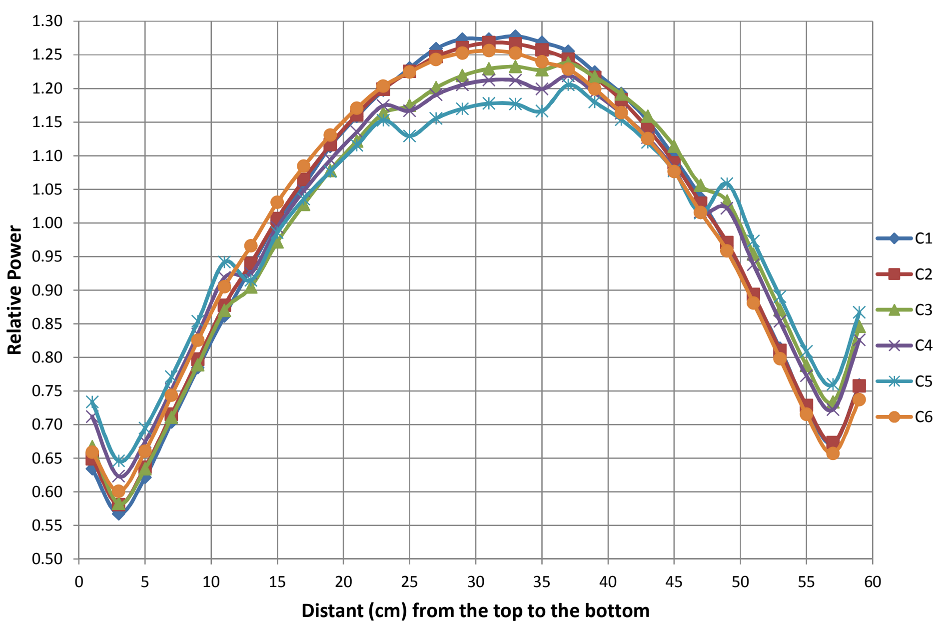

Fig. 6. Relative power distribution in axial direction of 6 cycles

3.3. Neutron flux at irradiation positions and horizontal beam tubes

To confirm about utilization ability of the designed reactor, neutron flux in 3 groups: thermal (E_{th} < 0.625 eV), epi-thermal (0.625 eV< E_{epi} < 0.821 MeV) and fast (0.821 MeV < E_f < 10 MeV) at irradiation positions and nose of horizontal beam tubes were investigated at BOC of cycles 5 and 6. Detail of neutron flux at each position is depicted in the **Table 10** and **Fig. 7**. On these cycles, the reactor is operated at nominal power of about 10 MWt.

Table 10. Neutron flux at irradiation positions [n.cm^{-2}.s^{-1}]

CYCLE	5			6		
Neutron trap						
	Thermal	**Epi-thermal**	**Fast**	**Thermal**	**Epi-thermal**	**Fast**
Average	2.24E+14	1.67E+14	6.72E+13	2.02E+14	1.48E+14	5.89E+13
Maximum	2.82E+14	2.08E+14	8.29E+13	2.54E+14	1.82E+14	7.20E+13
Cold neutron source						
Average	9.62E+12	1.28E+12	4.04E+11	1.23E+13	1.59E+12	4.02E+11
Maximim	1.22E+13	1.64E+12	5.12E+11	1.57E+13	2.07E+12	5.12E+11
SiD-1						
Average	2.97E+12	2.02E+11	4.88E+10	1.02E+13	6.99E+11	1.29E+11
Maximum	3.75E+12	2.68E+11	6.19E+10	1.30E+13	9.33E+11	1.65E+11
SiD-2						
Average	2.22E+12	1.25E+11	3.51E+10	5.65E+12	2.97E+11	6.29E+10
Maximum	2.80E+12	1.63E+11	4.45E+10	7.16E+12	3.94E+11	8.04E+10
SiD-3						
Average	1.62E+12	8.16E+10	1.93E+10	6.16E+12	3.04E+11	5.24E+10
Maximum	2.04E+12	1.07E+11	2.44E+10	7.82E+12	4.06E+11	6.70E+10
SiD-4						
Average	2.65E+12	1.52E+11	3.39E+10	1.01E+13	6.18E+11	1.03E+11
Maximum	3.34E+12	2.00E+11	4.27E+10	1.29E+13	8.31E+11	1.32E+11
B1						
Average	4.90E+13	9.44E+12	2.22E+12	5.73E+13	1.02E+13	1.81E+12
Maximum	6.14E+13	1.21E+13	2.81E+12	7.15E+13	1.31E+13	2.29E+12
B2						
Average	2.64E+13	4.13E+12	1.11E+12	5.94E+13	1.12E+13	2.02E+12
Maximum	3.34E+13	5.31E+12	1.40E+12	7.60E+13	1.46E+13	2.58E+12
S1						
Average	4.99E+13	1.07E+13	2.35E+12	6.85E+13	1.52E+13	2.58E+12
Maximum	6.27E+13	1.38E+13	2.99E+12	8.70E+13	1.97E+13	3.30E+12
S2						
Average	4.16E+13	8.55E+12	1.90E+12	6.97E+13	1.58E+13	2.68E+12
Maximum	5.25E+13	1.10E+13	2.40E+12	8.90E+13	2.06E+13	3.42E+12
S3						
Average	2.15E+13	2.17E+12	5.17E+11	2.76E+13	2.35E+12	4.87E+11
Maximum	2.71E+13	2.86E+12	6.56E+11	3.53E+13	3.15E+12	6.20E+11
S4						
Average	1.90E+13	1.98E+12	4.80E+11	2.85E+13	2.66E+12	5.56E+11
Maximum	2.40E+13	2.58E+12	6.15E+11	3.62E+13	3.51E+12	7.17E+11
S5						

Average	1.43E+13	1.40E+12	3.57E+11	2.64E+13	2.40E+12	5.09E+11
Maximum	1.80E+13	1.81E+12	4.55E+11	3.33E+13	3.16E+12	6.57E+11
S6						
Average	1.34E+13	8.78E+11	1.66E+11	1.55E+13	7.99E+11	1.40E+11
Maximum	1.75E+13	1.28E+12	2.14E+11	2.08E+13	1.27E+12	1.85E+11
S7						
Average	1.06E+13	6.62E+11	1.61E+11	1.38E+13	7.09E+11	1.60E+11
Maximum	1.34E+13	8.92E+11	2.09E+11	1.76E+13	9.68E+11	2.08E+11
S8						
Average	8.27E+12	5.24E+11	1.37E+11	1.28E+13	6.91E+11	1.64E+11
Maximum	1.03E+13	6.91E+11	1.77E+11	1.62E+13	9.15E+11	2.10E+11
S9						
Average	6.10E+12	3.62E+11	9.72E+10	1.19E+13	6.43E+11	1.46E+11
Maximum	7.62E+12	4.82E+11	1.26E+11	1.50E+13	8.58E+11	1.89E+11
Beam tubes (average neutron fluxes inside beam tubes)						
HBT1-1	1.27E+13	9.53E+13	1.54E+13	3.52E+13	3.65E+14	3.67E+13
HBT1-2	1.30E+13	9.58E+13	1.61E+13	3.50E+13	3.31E+14	3.66E+13
HBT2-1	3.90E+13	3.19E+14	5.11E+13	4.76E+13	3.95E+14	4.39E+13
HBT2-2	4.00E+13	3.29E+14	4.98E+13	5.02E+13	4.29E+14	4.69E+13
HBT3-1	2.07E+12	1.77E+12	4.95E+11	6.16E+12	4.98E+12	8.56E+11
HBT3-2	2.04E+12	1.76E+12	4.81E+11	6.10E+12	4.53E+12	8.11E+11

The neutron fluxes at irradiation positions of the designed reactor completely meet requirements for utilizations and applications for RI, NAA, NTD, as well as for basic research and material science study using neutron beam tubes.

Fig. 8 and **Fig. 9** show the thermal and fast neutron flux in average distribution of cycles 5 and 6 at beginning of cycle in case all CRs withdrawn completely.

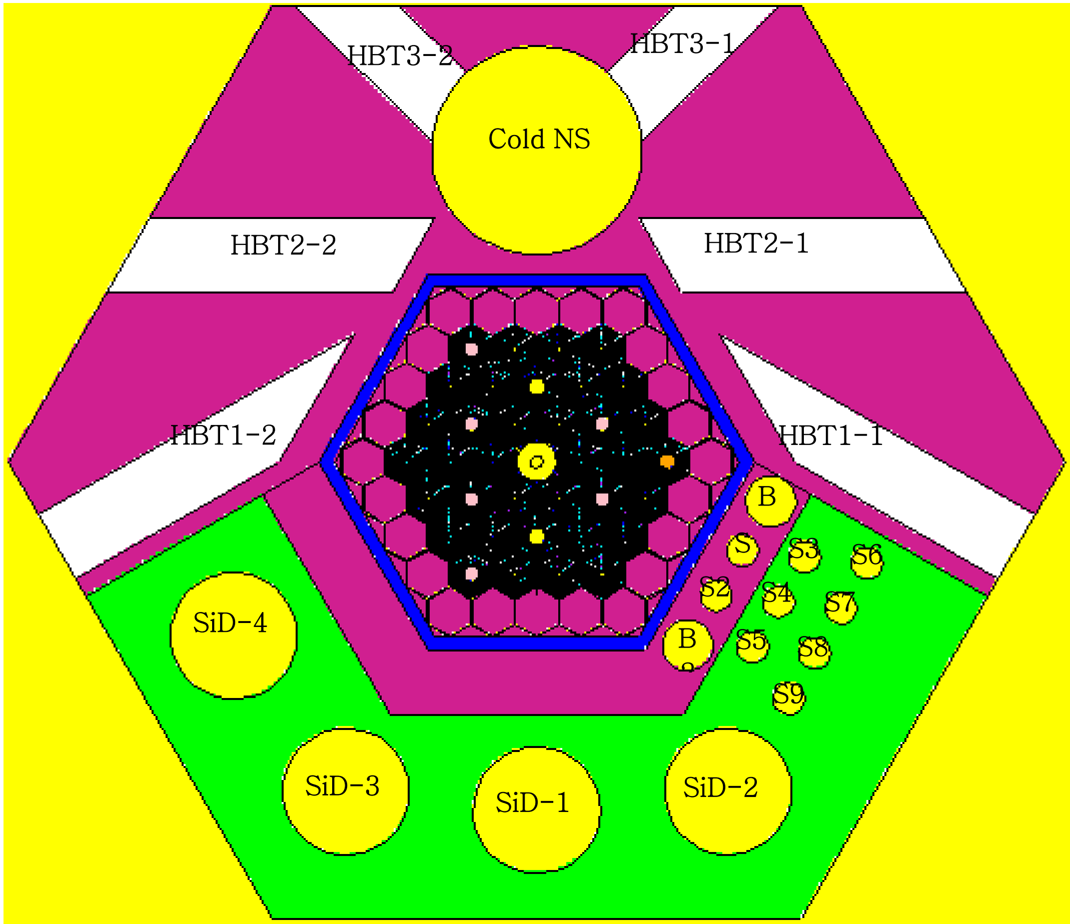

Fig. 7. Cross section of the reactor core configuration and reflector at cycle 6

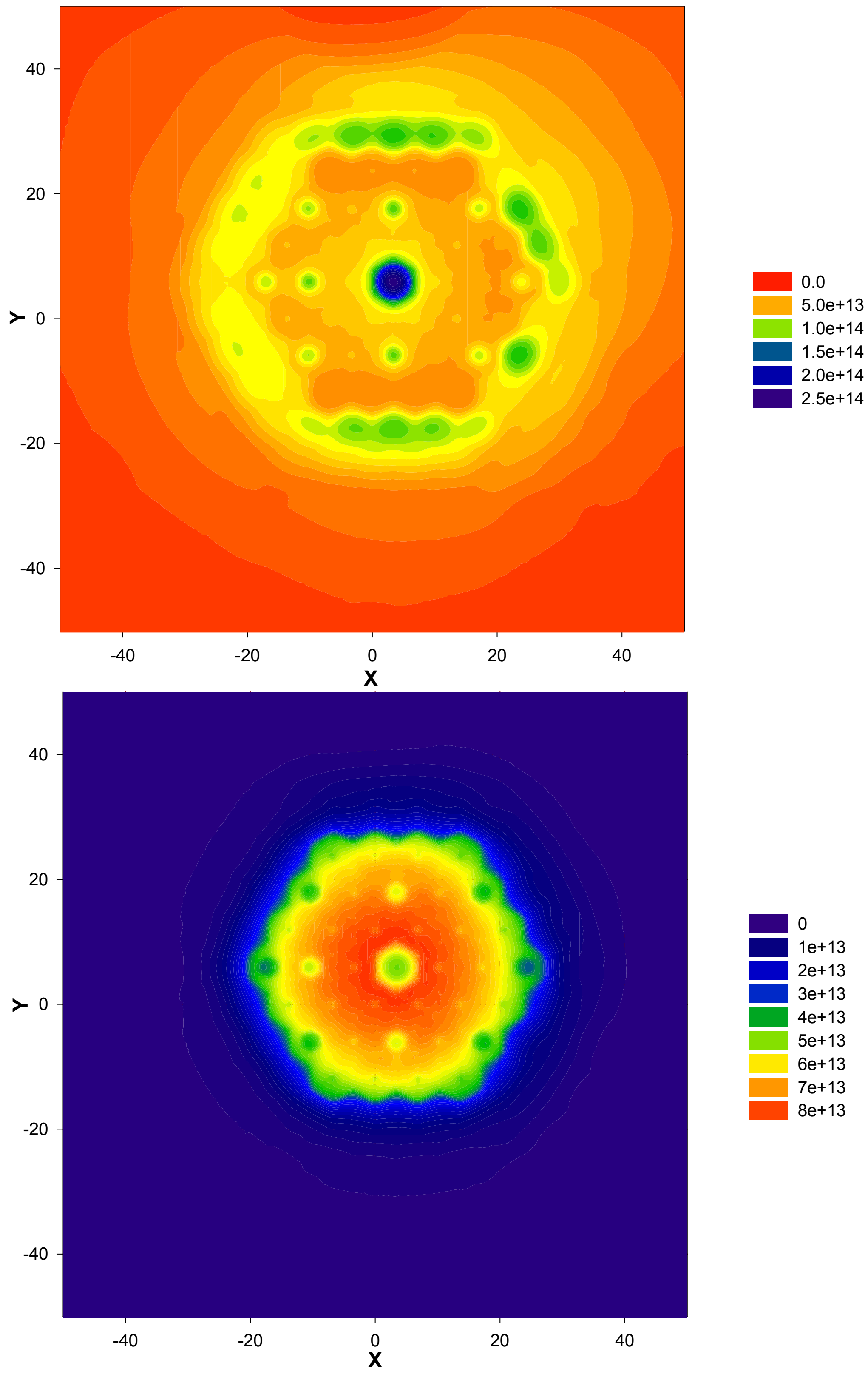

Fig. 8. The neutron fluxes in average: thermal and fast of beginning of cycle 5 in case all control rods out of the core

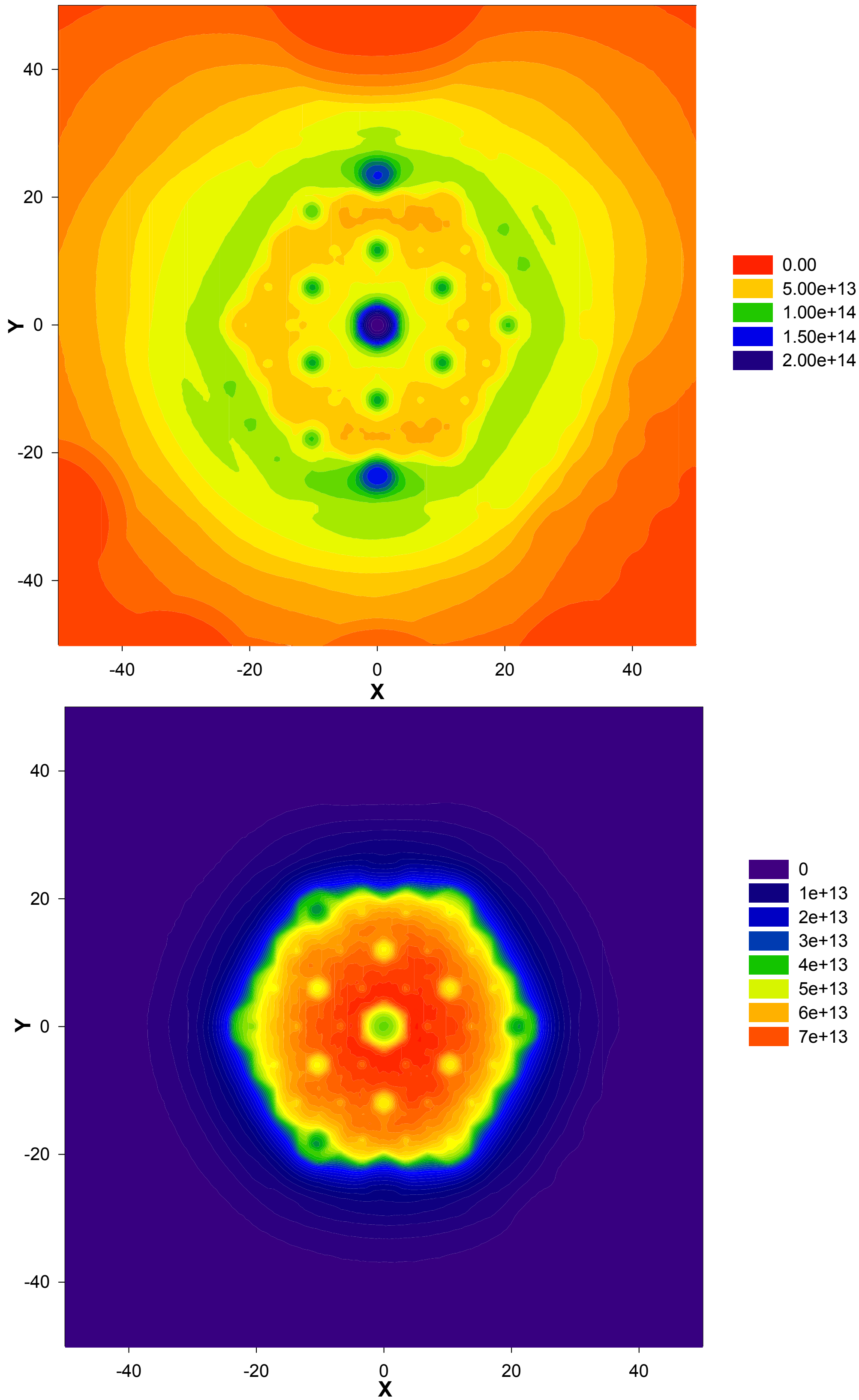

Fig. 9. The neutron fluxes in average: thermal and fast of beginning of cycle 6 in case all control rods out of the core

4. Conclusion

The neutronics calculation of the designed research reactor including burn-up was implemented to find out the "equilibrium core" after 6 cycles of operation with different core configurations and reactor power levels. Each operation cycle was confirmed about nuclear safety as critical conditions and thermal hydraulics safety limit incase of steady state operation. The safety of the designed reactor was also assured by having negative temperature feedback coefficients of coolant and fuel as well as void coefficients.

For utilization and application, the detailed neutron flux distribution at irradiation positions were calculated and it was quite good not only for traditional utilizations as RI productions, NAA, but also for advanced ones as NTD, material science study; setting up a cold neutron source for the near future as well as a neutron trap for fuel and material irradiation test-loop.

For detailed calculation in thermal hydraulics and safety analysis, the parameters such as kinetics, power peaking factors, maximum control rod worth, etc. were estimated by neutronics codes. The results of thermal hydraulics and safety analysis of the designed reactor cores will be presented in another paper.

ACKNOWLEGEMENTS

This work was performed with the financial support of Ministry of Science and Technology of Vietnam under the national research project with code ĐTĐL.CN-50/15.

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TÍNH TOÁN THIẾT KẾ KHÁI NIỆM VỀ NƠTRON CHO LÒ PHẢN ỨNG NGHIÊN CỨU ĐA MỤC TIÊU THÔNG LƯỢNG CAO

Tóm tắt: Bài báo trình bày các kết quả tính toán về thiết kế khái niệm cho một lò phản ứng nghiên cứu công suất khoảng 10 MWt với thông lượng nơtron cao, đa mục tiêu cho Trung tâm nghiên cứu khoa học công nghệ hạt nhân của Việt Nam. Nhiên liệu độ giàu thấp loại VVR-KN do Liên bang Nga chế tạo được lựa chọn cho thiết kế lò phản ứng. Các đặc trưng chính của vùng hoạt lò phản ứng được xem xét để đảm bảo các yêu cầu về an toàn và khả năng sử dụng của lò phản ứng. Cấu hình vùng hoạt của lò phản ứng thiết lập trong 6 chu trình làm việc được xem xét với điều kiện an toàn khi đánh giá về giới hạn an toàn và điều kiện tới hạn. Các thông số về an toàn cũng như các thông số động học được sử dụng để phân tích thủy nhiệt và an toàn cho mỗi cấu hình vùng hoạt. Sau 6 chu trình vận hành với các cấu hình vùng hoạt và các mức công suất khác nhau, cấu hình vùng hoạt cân bằng được thiết lập. Chương trình tính toán thiết kế neutron MCNP6.1 và hệ chương trình liên kết REBUS-MCNP6.1 được sử dụng cho thiết kế cũng như tính toán cháy cho lò phản ứng. Các tính toán chi tiết về phân bố thông lượng neutron tại các vị trí chiếu xạ trong vùng hoạt và vùng phản xạ để sử dụng cho các mục đích sản xuất đồng vị phóng xạ, phân tích kích hoạt, pha tạp vật liệu bằng neutron, cũng như sử dụng các kênh dẫn dòng neutron nằm ngang để phục vụ cho các nghiên cứu về khoa học vật liệu, vật lý hạt nhân cũng được thực hiện và trình bày.

Các từ khóa: *Conceptual design, VVR-KN fuel type, MCNP code, REBUS-MCNP system code*