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Conceptual design of critical assembly using Low enriched uranium fuel and moderated light water

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Abstract: Critical assembly is a very important facility to serve for fundamental reactor physics research, application of neutron source, training and education. In nuclear engineering, critical assembly is a facility for carrying out measurement of reactor physics parameters, creating benchmark problem, validation of neutron physics calculation tool in computer codes and nuclear data. Basing on concept using commercial Nuclear Power Plant (NPP) fuels such as PWR (AP-1000) and VVER-1000 fuel rods with limited 2 meter in length and fully controlled by water level, the conceptual design of the critical assembly has been carried out in neutronic, thermal hydraulics and safety analysis. Ten benchmark critical core configurations of critical assembly are established and investigated to show safety during normal opeartion and accident conditions. Design calculation results show that NPP fuels are fully adequate for critical assembly operating under nominal power 100W and having average neutron flux about 3×10^9 neutron/cm^2.s.

Keywords: *Critical assembly, NPP fuels, neutronics, thermal hydraulics, safety analysis.*

I. INTRODUCTION

The development of nuclear program of any country is based on experimental portion conducted on research reactor or critical assembly, mock-up. The diversity of reactor parameters and their characteristics means that it involes verification and validation all physical data and calculation models so carrying out experiment on critical assembly is nessesity. Critical assembly is very flexible, adaptable, easy to access and to instrument while still reliable and safety. Neutron source of critical assembly can be used for neutron activation analysis, radioisotope productions, neutron radiography and other application to demonstrate in traing nign and education. A various topics related to experimetal reactor physics can be conducted on critical assembly for research and training.

Using low enriched uranium fuel from 1.6 to 5% U-235 of commercial NPPs [1, 2], 10 critical configurations with 2.0cm arrangement pitch of critical assembly have been investigated detail about neutronics, thermal hydraulics, safety analysis and gamma shielding. The advantage of these types of fuel is that they are be using in many commercial NPPs in the world and proved safety and integrity during operation under high power, temperature and pressure conditions.

The critical assembly has been designed with open tank in pool type and controlled completely by water level. The demineralized light water is chosen for moderation, cooling and biological shielding functions of critical assembly. The tank of critical assembly has cylindrical geometry with 1m radius and 2m height. Two grid plates are located at bottom and top of the reactor tank with radius 40cm. They have functions to fix fuel rods and neutron source driven rod in the center as well as aluminum tube containing neutron detector out side the core. Neutron source tube made by aluminum

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tighten air is used for loading and unloading neutron source for critical assembly.

Ten benchmarks configuration cores using different fuel enrichments are determined and detailed considered in neutronics, thermal hydraulics and safety aspect. For neutronics analysis, all characteristics parameters such as critical water level, neutron flux and power distribution, safety control rods reactivity, feedback parameters, kinetics parameters have been considered. Thermal hydraulics analysis is evaluated in case operating critical assembly under 100W power at steady state to evaluate fuel pellet and fuel cladding temperature. Safety analysis has been carried out in case reactivity insertion accident when water level increases over critical level or loading neutron source in to the core at critical status of critical assembly. Shielding calculation of critical assembly has also evaluated whether having concrete shielding around critical assembly tank or not.

Calculation results show that the designed critical assembly is safe in opreration and accident conditions together with effective utilizations and it is essential tool in fundamental research as well as training in nuclear engineering.

II. STRUCTURE AND APPLICATION OF CRITICAL ASSEMBLY

A. Structure of critical assembly

The main structure, system and component of designed critical assembly are not complicated when comparing with research reactor. Reactor tank of critical assembly contains fuel rods and water together with other control and instrument equipment. The water level in reactor tank is controlled by main and auxiliary pumps. Neutron source of critical assembly is loaded and unloaded to the core through tighten aluminum tube by hydraulics pump. Water tank supply is directly connected to the reactor tank by a pipe with 20 cm diameter. **Fig.1** depicts main structure of designed critical assembly.

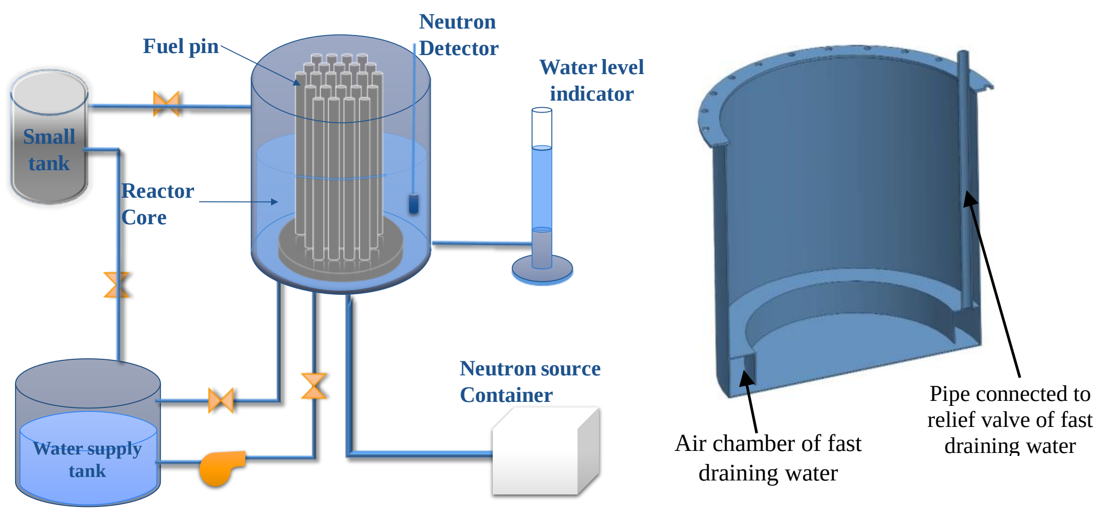


Fig. 1. Main structure of designed critical assembly using low enrichment fuel from 1.6 to 5% U-235 and reactor tank

Fuels of critical assembly are fuel rods in fuel assembly of commercial NPPs and have 1.5 to 2m in height with 20 cm header and 10 cm tail made by aluminum. Chosen neutron source of the critical assembly should be Am-Be because this source has long life time more than 400 years. After loading for start-up critical assembly, the neutron source is kept in lead and paraffin shielding container. Three neutron detection channels using wide range and independence are served for neutron flux detection and monitor of control system. To supply water for critical assembly, one pump will be operated to take water from supply water tank to the tank of critical assembly. Flow rate of water from supply tank to reactor tank is around 100 to 150 liters/minute and flow rate is controlled by a valve. This pump and valve can be considered as shim rod of critical assembly. A lower power pump is used for supplying water to reactor tank from supply tank, flow rate of this pump is about few liters/minute. This system can be considered as regulating rod of critical assembly. The control system of critical assembly has buttons to control open or close of two pumps and valves just like control rods.

The safety system of critical assembly includes safety control rod in plate shape made by Cadmium with dimensions 0.5cm×10cm×200cm, fast and slow draining water systems. In fast draining system, water level of reactor tank will be immediately reduced 10 cm by opening the relief valve to make balance of air pressure between water in the tank and air inside chamber (**Fig. 1**) then water will be completely occupied the air. In slow draining water system, water in reactor tank will be withdrawed by opening release valve then water will be passed through big pipe with 20cm diameter to water tank supply.

Other auxiliary systems like dosimeter, ventilation, electricity, water purification and supplying, mechanical and experiment supporting are included in the whole system of critical assembly.

B. Application of the critical assembly

Critical assembly can be considered as a low power research reactor [3, 4] so it can be served for basic research in reactor physics, training and education or application related to nuclear engineering. Neutron source of critical assembly can be used for some applications such as radioisotope productions, neutron activation analysis and neutron radiography. Testing new equipment for neutron detection or fabricated fuel pellet can be carried out on critical assembly. Related to nuclear engineering training and education, critical assembly is a tool to measure reactor physics parameters characteristics, setting new critical configurations with different enrichment or pitch arrangement of fuel rod and experimental results can be used for validation of model and calculation method of computer code.

Critical assembly is a subject for carrying out experiment related to reactor physics, reactor kinetics, nuclear reactions and data, some applications in neutron activation analysis, radioisotope production for short live isotopes and non-destructive testing. In basis research related to reactor physics, critical assembly can be used for: studying neutron transport in different media like water, graphite, heavy water; measuring of neutron flux and spectrum by using foil or multi foils activation; determining reactor parameters using pulsed neutron source. Kinetics parameters can be determined with different methods using critical assembly. Studying about fuel loading patterns, criticality calculation and correcting calculation models or libraries are application of critical assembly. Environmental samples, geological materials and others can be analyzed by neutron activation analysis using neutron flux about 10^{12} n/cm^2.s of critical assembly. Neutron, gamma shielding research with different materials, testing new neutron detection equipment and new fabricated fuel are an object in utilization of critical assembly.

In university or training center, students and nuclear engineering staffs are mainly objects for training in reactor engineering. For student, some subjects will be paid attention such as reactor theory, reactor kinetics and dynamics, radiation measurement, neutron flux and spectrum determination, critical status critical assembly with different loading configurations. Effect of reactivity of different materials inside reactor core or fuel loading patterns, reactor kinetics and shielding experiments are important for training of operating staffs and researchers. In case studying about neutron reaction or neutrino particle, precise detection particle equipment should be used and set up.

Critical assembly can be used as an objective simulation for developing calculation programs and it is a very good facility for trainee to consolidate knowledge about reactor physics, reactor dynamics. To support for trainee when carrying out experiments on critical assembly, computer codes for reactor neutronic, thermal-hydraulics calculation need to be trained.

III. NEUTRONICS, THERMAL HYDRAULICS AND SAFETY ANALYSIS OF CRITICAL ASSEMBLY

To investigate detail physics characteristics in design calculation of critical assembly, many computer codes are applied. In neutronics analysis, mainly MCNP5 [5] and SRAC [6] codes are used for neutronics parameters investigation including neutron flux and power distribution, critical water level determination, control rod worths, reactivity feedback temperature coefficients, kinetics parameters, burn-up analysis. Couple codes WIMS-ANL [7] and REBUS [8] are also applied in critical calculation with determined water level. For shielding calculation, photon spectrum in 18 energy groups of each core configuration was estimated by ORIGEN [9] code and gamma dose rate evaluation was performed by MCNP code. In thermal hydraulics, temperature of fuel and cladding are estimated by the analytical method and RELAP5/MOD3.2 [10] code. For safety analysis of critical assembly, reactivity insertion accident (RIA) is analyzed with different positive reactivity insertion by RELAP5 code. This means that all parameters of critical assembly with different loading enriched fuels PWR (AP-1000) and VVER-1000 are carefully considered to confirm about safety and utilization capability of critical assembly.

A. Neutronics design calculation

Fuel characteristics

Two fuel types to be used in critical assembly are investigated by MCNP and SRAC code. The calculation model of fuel and basic parameters of these fuel types are presented in **Fig.2** and **Table I**.

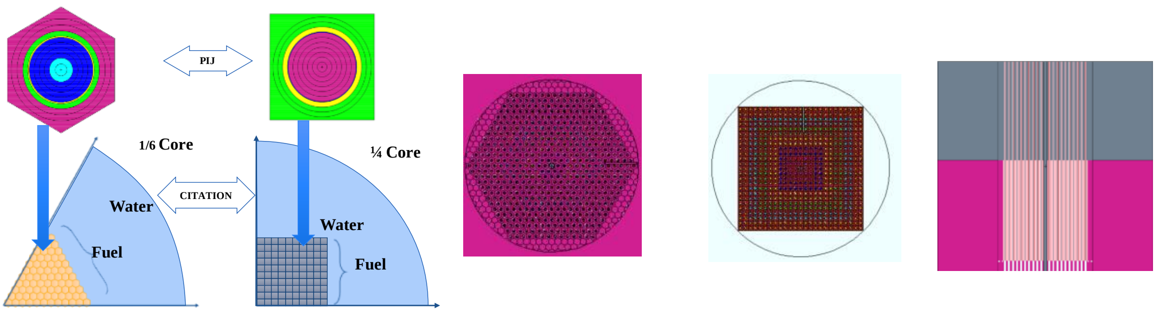


Fig.2. Fuel and whole core calculation models of VVER and PWR cores in the SRAC and the MCNP codes

Table I. Characteristics parameters of VVER and PWR fuels

Parameters	PWR fuel pin	VVER fuel pin
Density (g/cm^3)	~10.40	~10.40
Enrichment U-235 (%)	1.6 – 2.1 – 2.6 – 3.2 – 4.2 – 4.8	1.6 – 2.4 – 3.3 – 3.7 – 4.1 – 4.5
Calculation temperature (^{o}C)	20	20
Pitch (cm)	1.26	1.275
Dimension (cm) Fuel radius Helium gap Cladding	 0.46955 0.00955 0.0673	 0.14 - 0.38 0.006 0.069
Cladding material	Zr-4	Zr-4 – 1%Nb

The main difference of two fuel types is geometry, thickness of fuel, cladding material and fuel masses. Enrichment of these fuels can be considered the same and having values from 1.6% to approximate 5% U-235. The changing of infinite multiplication factors of two fuel types has been investigated to determine effectiveness of these fuels. **Table II** and **Fig. 3** show the calculation results of MCNP and SRAC codes with different calculation libraries.

Table II. Infinite multiplication factors of both fuel types depending on enrichment.

VVER fuel rod		**Enrichment**					
Library	**Code**	**1.60%**	**2.40%**	**3.30%**	**3.70%**	**4.10%**	**4.50%**
ENDF/B VII	SRAC	1.14457	1.27661	1.36263	1.38963	1.41212	1.43116
	MCNP	1.14705	1.27902	1.36557	1.39288	1.41558	1.43461
JDL33	SRAC	1.11681	1.25199	1.34090	1.36894	1.39237	1.41222
	MCNP	1.12312	1.25884	1.34837	1.37697	1.40080	1.42110
JEF31	SRAC	1.11949	1.25430	1.34289	1.37082	1.39414	1.41391
	MCNP	1.12449	1.25958	1.34853	1.37657	1.40032	1.42009
PWR fuel rod		**Enrichment**					
Library	**Code**	**1.60%**	**2.10%**	**2.60%**	**3.20%**	**4.20%**	**4.80%**
ENDF/B VII	SRAC	1.15824	1.22808	1.27613	1.31731	1.36372	1.38377
	MCNP	1.15574	1.22542	1.27345	1.31470	1.36125	1.38140
JDL33	SRAC	1.15200	1.22210	1.27037	1.31175	1.35840	1.37854
	MCNP	1.15585	1.22588	1.27432	1.31544	1.36279	1.38326
JEF31	SRAC	1.15465	1.22448	1.27253	1.31371	1.36011	1.38015
	MCNP	1.15449	1.22422	1.27230	1.31348	1.36003	1.38005

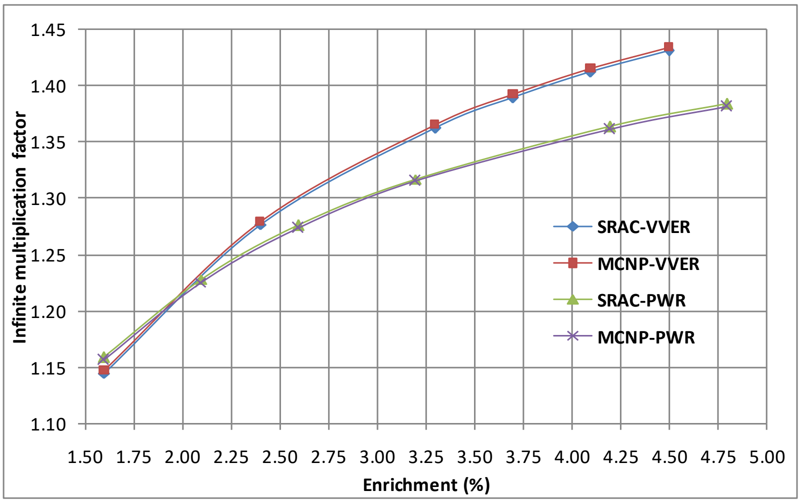


Fig.3. Infinite multiplication factor depending on enrichment of fuel VVER or PWR types

The standard deviation in critical calculation of MCNP5 and SRAC codes is about ±0.0015% and ±0.001% respectively. The difference of infinite multiplication factors from results of both codes is about 0.2%. The infinite multiplication factors of fuel rods are also considered in case changing of V_f/V_m ratio (volume of fuel per volume of moderator ratio). It means the different arrangement pitches of fuel rods inside the core. Arrangement pitches of fuel increasing are corresponded with decreasing of V_f/V_m ratio. Calculation results of infinite multiplication factors are presented in **Fig. 4**

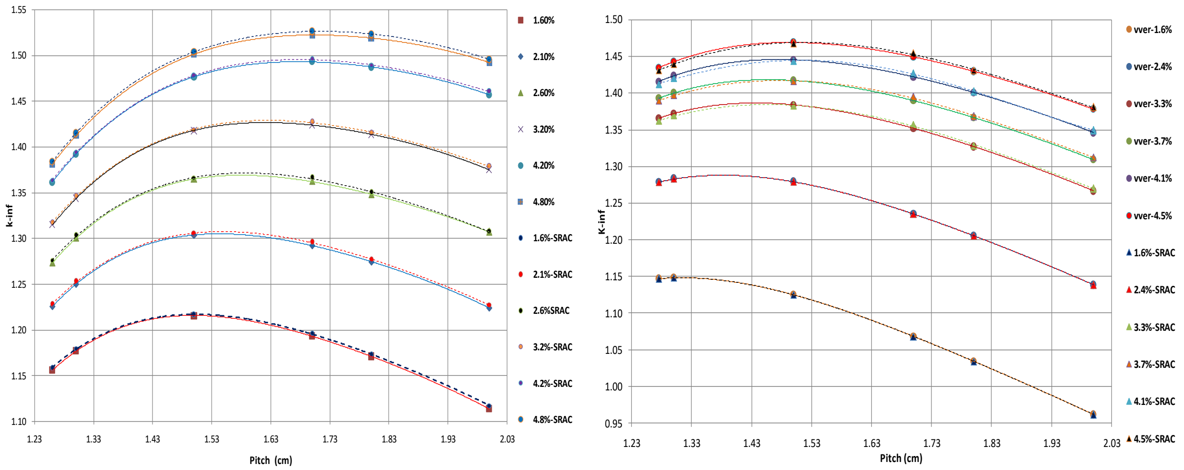


Fig.4. Calculation results of infinite multiplication factors with different V_f/V_m ratio of PWR and VVER fuel by MCNP and SRAC codes using ENDF/B7.0 library

Calculation results show that infinite multiplication factors are depended complicatedly on V_f/V_m ratio as polynomial curve. PWR fuels with high enrichment above 2.4% have multiplication factor increasing together with expanding of arrangement pitch. By contract, multiplication factor of VVER fuels decrease following the increase of arrangement pitch. Calculation results can be used for setting up critical configurations with different arrangement pitches for research purposes. Micro and macro cross sections of two fuel types have been calculated to prepare for whole core calculation by diffusion code in SRAC system. Number of neutron energy groups in whole core calculation is 7 or 2 groups. Burn up of fuel rod was calculated with U-235(%) depletion up to 9% because critical assembly will be operated at low power. MVP-Burn [12] and SRAC codes were applied for this investigation and calculation results are shown in **Fig. 5**.

Critical assembly is operated under nominal power about 100W and neutron flux about 3×10^9 n/cm^2.s so the fuels can be used for long time and had low burn-up of U-235. The calculation results between SRAC and MVP-Burn codes are good agreement, the discrepancy of results is caused by two codes using different libraries: ENDF/B7.0 and ENDF/B6.8 to PIJ (SRAC code) and MVP [11], MVP-Burn respectively.

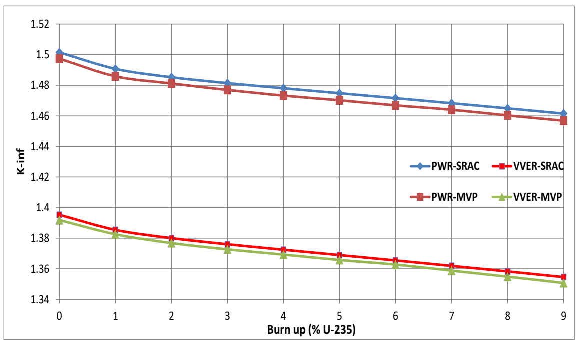


Fig.5. Infinite multiplication factor of fuel pin following burn-up (%) of U-235

Benchmark critical configurations

After investigation of fuel, many core configurations are established with different arrangement pitches and fuel enrichments from 1.6% to 5%. Ten representative critical configurations with 2.0 cm pitch have been established. In these core configurations, number of loading fuels in critical assembly and water levels is different.

Table III. Number of fuel rods and water level of ten benchmark critical configurations (Uranium mass is assumed with 2m height of fuel rod)

Fuel type (Core name)	Water level (cm)	Number of fuel rods	Uranium mass (kg)	U-235 mass (kg)	U-238 mass (kg)
PWR1	95.0	360	214.72	5.58	209.03
PWR2	70.0	288	126.57	4.05	122.46
PWR3	58.0	224	81.57	3.43	78.10
PWR4	100.0	168	105.47	5.06	100.36
PWR5	34.6	624	135.55	4.98	130.51
Fuel type (Core name)	**Water level (cm)**	**Number of fuel rods**	**Uranium mass (kg)**	**U-235 mass (kg)**	**U-238 mass (kg)**
VVR1	54.5	630	123.41	4.07	119.30
VVR2	48.0	546	94.20	3.49	90.68
VVR3	56.0	396	79.71	3.27	76.41
VVR4	116.0	270	62.11	2.79	59.29
VVR5	60.0	546	125.60	4.88	120.68

The enrichment of fuel in configurations from 1 to 4 of both fuel types has values from 2.4% and 2.1% for VVER and PWR fuels respectively. PWR5 and VVER5 cores are mixed core configurations. In two configurations, all of fuels with different enrichments were used in the core. To estimate effective multiplication factor of 10 benchmark configurations, SRAC, MCNP and WIMS-ANL, REBUS codes were used and calculation results are not much different among three codes. This means neutronics characteristics of critical assembly can be determined by these codes. In whole core calculation, 7 neutron energy groups were applied to REBUS and SRAC codes.

Table IV. Calculation results of effective multiplication factor of 10 benchmark critical configurations

Core	No. of fuel rod	Water level (cm)	MCNP	WIMS-ANL and REBUS	PIJ – CITATION (SRAC)
VVER1	630	54.5	0.99996	0.99978	1.0002
VVER2	546	48.0	1.00031	1.00021	1.00067
VVER3	396	56.0	1.00004	0.99979	1.00042
VVER4	270	116.0	1.00032	0.99995	1.00105
VVERM	546	60.0	1.00045	1.00023	1.00121
PWR1	360	95.0	1.00018	0.99923	1.00052
PWR2	288	70.0	1.00023	0.99975	1.00067
PWR3	224	58.0	0.99986	0.99871	1.00038
PWR4	168	100	1.00052	1.00002	1.00137
PWRM	624	34.6	1.00010	0.99998	1.00062

Ten benchmark configurations were investigated in neutronics characteristics, thermal hydraulics and safety analysis. Some parameters effect to the effective multiplication of critical configurations including enrichment, density of uranium, fuel meat radius, fuel cladding and arrangement pitch. The fluctuation of these parameters is about ±5%. Total effect of these factors to multiplication factors is about 0.2%$\Delta k/k$ to mix-core and 0.16%$\Delta k/k$ to homogeneous core (using 1 kind of fuel type only).

Neutron flux and power peaking factor

Neutron flux and power distribution in radial and axial direction of 10 critical configurations were calculated under nominal power. Neutron flux distribution of 10 critical configurations is described in **Table V**.

Table V. Average neutron flux (n/cm^2.s) of 10 critical configurations

Core	Thermal neutron flux (<0.625eV)	Epi-thermal neutron flux (0.625eV< and <0.82MeV)	Fast neutron flux (0.82MeV< and <20 MeV)	Total neutron flux
VVER1	5.29E+08	1.29E+09	4.85E+08	2.31E+09
VVER2	7.15E+08	1.55E+09	7.08E+08	3.27E+09
VVER3	6.78E+08	1.83E+09	7.15E+08	3.23E+09
VVER4	4.46E+08	1.25E+09	4.98E+08	2.19E+09
VVER5	5.14E+08	1.29E+09	4.89E+09	2.29E+09
PWR1	3.87E+08	1.11E+09	4.23E+08	1.92E+09
PWR2	5.10E+08	1.62E+09	6.42E+08	2.77E+09
PWR3	6.69E+08	2.44E+09	1.01E+09	4.13E+09
PWR4	4.66E+08	1.80E+09	7.63E+08	3.03E+09
PWR5	5.03E+08	1.64E+09	6.48E+08	2.79E+09

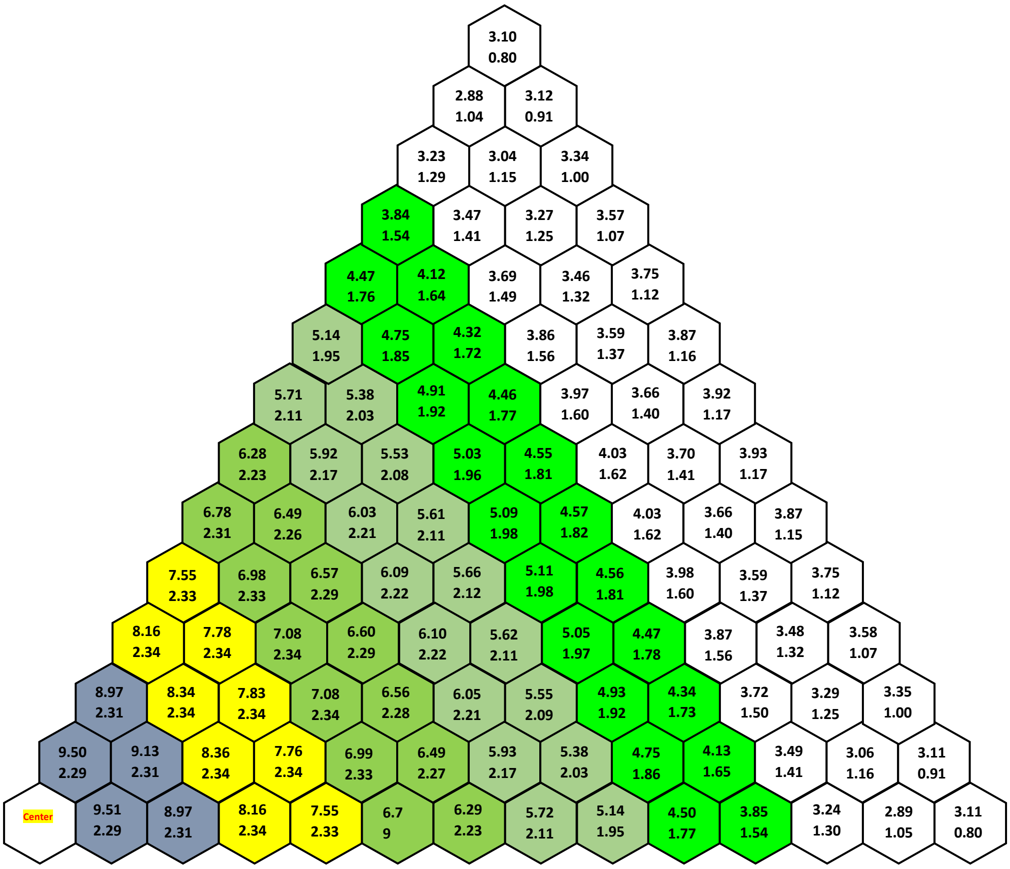
Average neutron flux of each critical configuration is nearly the same however neutron fluxs are depended on the water level and mass of uranium loading inside the core. The core with high water level and using low enriched fuel has low neutron flux and vise versa. Detailed neutron flux distribution at axial and radial direction on reflector using different materials such as beryllium, graphite, heavy water and light water has been calculated. Light water is chosen as reflector material of critical assembly.

Power peaking factor is an important parameter and will be used in safety analysis of facility so detailed calculation of power peak for hottest channel to be estimated. **Table VI** shows calculation results of total power peaking factor of all benchmark critical configurations.

Table VI. Power peaking factors of benchmark critical configurations

Core	Radial power peaking factor	Axial power peaking factor	Total power peaking factor	Average linear power (W/m)
VVER1	1.640	2.262	3.710	2.9125E-03
VVER2	1.601	1.889	3.024	3.8156E-03
VVER3	1.533	2.098	3.216	4.5094E-03
VVER4	1.440	2.115	3.046	3.1928E-03
VVER5	1.297	1.809	2.346	3.0525E-03
PWR1	1.572	2.262	3.556	2.92398E-03
PWR2	1.513	2.278	3.447	4.96032E-03
PWR3	1.488	1.990	2.961	7.69704E-03
PWR4	1.368	1.993	2.726	5.95238E-03
PWR5	1.202	1.567	1.884	4.63169E-03

Normally, the hottest fuel rod is located in center of the core but in mix core the hottest fuel rod will be moved to a different position in the core. In the mix core, the way to arrange fuel is carried out with low enriched in center and increasing enriched fuel to the edge of the core, the hottest fuel rod is not located in the center. Detailed neutron flux distribution of VVER5 and PWR5 configurations is presented in **Fig. 6.**



Center	9.12E+08 2.80E+09	8.66E+08 2.82E+09	8.03E+08 2.86E+09	7.63E+08 2.88E+09	7.04E+08 2.87E+09	6.60E+08 2.83E+09	5.92E+08 2.76E+09	5.35E+08 2.64E+09	4.54E+08 2.47E+09	3.95E+08 2.23E+09	3.46E+08 1.95E+09	3.90E+08 1.61E+09
9.11E+08 2.80E+09	8.91E+08 2.81E+09	8.61E+08 2.83E+09	8.01E+08 2.86E+09	7.61E+08 2.87E+09	7.02E+08 2.86E+09	6.58E+08 2.82E+09	5.90E+08 2.75E+09	5.36E+08 2.63E+09	4.53E+08 2.46E+09	3.95E+08 2.22E+09	3.45E+08 1.94E+09	3.86E+08 1.61E+09
8.66E+08 2.82E+09	8.60E+08 2.82E+09	8.41E+08 2.83E+09	7.89E+08 2.86E+09	7.54E+08 2.86E+09	6.97E+08 2.85E+09	6.52E+08 2.80E+09	5.85E+08 2.73E+09	5.29E+08 2.61E+09	4.48E+08 2.44E+09	3.89E+08 2.20E+09	3.42E+08 1.91E+09	3.83E+08 1.59E+09
8.01E+08 2.86E+09	7.99E+08 2.86E+09	7.91E+08 2.85E+09	7.70E+08 2.85E+09	7.45E+08 2.84E+09	6.89E+08 2.82E+09	6.45E+08 2.77E+09	5.77E+08 2.70E+09	5.21E+08 2.57E+09	4.42E+08 2.40E+09	3.82E+08 2.16E+09	3.37E+08 1.88E+09	3.76E+08 1.56E+09
7.63E+08 2.87E+09	7.60E+08 2.87E+09	7.55E+08 2.85E+09	7.45E+08 2.84E+09	7.26E+08 2.81E+09	6.76E+08 2.79E+09	6.32E+08 2.72E+09	5.68E+08 2.64E+09	5.11E+08 2.51E+09	4.31E+08 2.34E+09	3.73E+08 2.11E+09	3.26E+08 1.83E+09	3.64E+08 1.51E+09
7.04E+08 2.87E+09	7.02E+08 2.86E+09	6.96E+08 2.85E+09	6.90E+08 2.82E+09	6.74E+08 2.78E+09	6.49E+08 2.73E+09	6.12E+08 2.66E+09	5.51E+08 2.57E+09	4.95E+08 2.43E+09	4.17E+08 2.27E+09	3.61E+08 2.04E+09	3.15E+08 1.77E+09	3.50E+08 1.46E+09
6.59E+08 2.82E+09	6.57E+08 2.82E+09	6.51E+08 2.80E+09	6.43E+08 2.76E+09	6.32E+08 2.72E+09	6.12E+08 2.66E+09	5.82E+08 2.58E+09	5.26E+08 2.48E+09	4.72E+08 2.34E+09	3.99E+08 2.17E+09	3.45E+08 1.95E+09	3.01E+08 1.69E+09	3.33E+08 1.39E+09
5.93E+08 2.76E+09	5.93E+08 2.75E+09	5.88E+08 2.73E+09	5.78E+08 2.69E+09	5.68E+08 2.64E+09	5.50E+08 2.56E+09	5.26E+08 2.48E+09	4.92E+08 2.36E+09	4.47E+08 2.22E+09	3.76E+08 2.05E+09	3.25E+08 1.83E+09	2.84E+08 1.59E+09	3.14E+08 1.31E+09
5.35E+08 2.63E+09	5.34E+08 2.62E+09	5.29E+08 2.60E+09	5.21E+08 2.56E+09	5.10E+08 2.50E+09	4.92E+08 2.43E+09	4.73E+08 2.33E+09	4.46E+08 2.22E+09	4.06E+08 2.07E+09	3.46E+08 1.91E+09	3.00E+08 1.70E+09	2.61E+08 1.47E+09	2.90E+08 1.21E+09
4.54E+08 2.46E+09	4.52E+08 2.45E+09	4.47E+08 2.43E+09	4.39E+08 2.39E+09	4.30E+08 2.34E+09	4.15E+08 2.26E+09	3.98E+08 2.17E+09	3.76E+08 2.05E+09	3.46E+08 1.91E+09	3.09E+08 1.74E+09	2.72E+08 1.55E+09	2.37E+08 1.33E+09	2.61E+08 1.09E+09
3.94E+08 2.22E+09	3.94E+08 2.22E+09	3.89E+08 2.19E+09	3.80E+08 2.15E+09	3.72E+08 2.10E+09	3.59E+08 2.03E+09	3.42E+08 1.94E+09	3.23E+08 1.83E+09	3.00E+08 1.70E+09	2.70E+08 1.55E+09	2.39E+08 1.37E+09	2.09E+08 1.17E+09	2.32E+08 9.61E+08
3.46E+08 1.94E+09	3.46E+08 1.93E+09	3.40E+08 1.91E+09	3.33E+08 1.87E+09	3.26E+08 1.83E+09	3.14E+08 1.76E+09	3.00E+08 1.69E+09	2.83E+08 1.59E+09	2.61E+08 1.47E+09	2.37E+08 1.33E+09	2.10E+08 1.18E+09	1.88E+08 1.00E+09	2.07E+08 8.21E+08
3.90E+08 1.61E+09	3.85E+08 1.62E+09	3.82E+08 1.58E+09	3.77E+08 1.55E+09	3.64E+08 1.51E+09	3.52E+08 1.45E+09	3.34E+08 1.40E+09	3.13E+08 1.32E+09	2.88E+08 1.21E+09	2.60E+08 1.10E+09	2.33E+08 9.62E+08	2.08E+08 8.22E+08	1.85E+08 8.02E+08

Fig. 6. Neutron flux (n/cm^2.s) distribution of VVER5 and PWR5 benchmark configurations
Upper values are thermal neutron and lower values are epi-thermal and fast neutron
(For VVER5 the upper values $\times 10^8$ and lower values $\times 10^9$)

Calculation results have relative error smaller than 0.5%. Because all the benchmark cores have simple geometry the symmetry of neutron flux of the cores through diagonal line is very clearly. The neutron flux distribution is reduced from center to the edge of the core but power peaking factor is not in the center because enriched fuel rods around the center have lower enrichment. In case same fuels enrichment, fuel rod having highest neutron flux and power peaking factor is located near center of the core.

Temperature reactivity coefficients and kinetics parameters

To confirm inherent safety of benchmark critical configurations, temperature reactivity feedback coefficients of moderator and fuel are evaluated by MCNP and SRAC codes. Detailed calculation results are shown in **Table VII** and **VIII**.

Table VII. Temperature reactivity feedback (%Δk/k/°C) of moderator

Core	Temp. 296°K – 350°K		Temp. 350°K – 400°K	
	MCNP	SRAC	MCNP	SRAC
PWR1	-7.498E-03	-7.717E-03	-1.278E-02	-1.302E-02
PWR2	-1.025E-02	-1.112E-02	-1.640E-02	-1.827E-02
PWR3	-1.314E-02	-1.446E-02	-1.931E-02	-2.339E-02
PWR4	-1.303E-02	-1.440E-02	-1.857E-02	-2.317E-02
PWR5	-9.608E-03	-1.039E-02	-1.519E-02	-1.721E-02
VVER1	-9.652E-03	-1.048E-02	-1.566E-02	-1.625E-02
VVER2	-1.279E-02	-1.303E-02	-1.915E-02	-2.015E-02
VVER3	-9.718E-03	-9.802E-03	-1.456E-02	-1.577E-02
VVER4	-9.695E-03	-9.736E-03	-1.435E-02	-1.515E-02
VVER5	-4.412E-03	-4.530E-03	-7.540E-03	-8.016E-03

Table VIII. Temperature reactivity feedback (%Δk/k/°C) of fuel

Core	Temp. 300-400°K		Temp. 400-500°K		Temp. 500-600°K	
	MCNP	SRAC	MCNP	SRAC	MCNP	SRAC
PWR1	-1.687E-03	-1.709E-03	-1.527E-03	-1.577E-03	-1.391E-03	-1.455E-03
PWR2	-1.537E-03	-1.613E-03	-1.385E-03	-1.494E-03	-1.298E-03	-1.370E-03
PWR3	-1.391E-03	-1.484E-03	-1.271E-03	-1.376E-03	-1.092E-03	-1.269E-03
PWR4	-1.358E-03	-1.412E-03	-1.133E-03	-1.299E-03	-9.744E-04	-1.220E-03
PWR5	-1.566E-03	-1.638E-03	-1.467E-03	-1.504E-03	-1.254E-03	-1.387E-03
VVER1	-1.397E-03	-1.451E-03	-1.307E-03	-1.359E-03	-9.802E-04	-1.231E-03
VVER2	-1.398E-03	-1.416E-03	-1.183E-03	-1.287E-03	-1.099E-03	-1.230E-03
VVER3	-1.280E-03	-1.382E-03	-1.298E-03	-1.303E-03	-1.092E-03	-1.192E-03
VVER4	-1.264E-03	-1.316E-03	-1.182E-03	-1.286E-03	-1.085E-03	-1.189E-03
VVER5	-1.360E-03	-1.431E-03	-1.218E-03	-1.328E-03	-1.125E-03	-1.223E-03

The average difference of calculation results between two codes is about 10%. The main reason is the treatment of neutron cross section of water and fuel with different temperature in both codes. All calculation results show that 10 benchmark configurations have negative temperature reactivity feedback coefficient, this mean safety of these cores is assured.

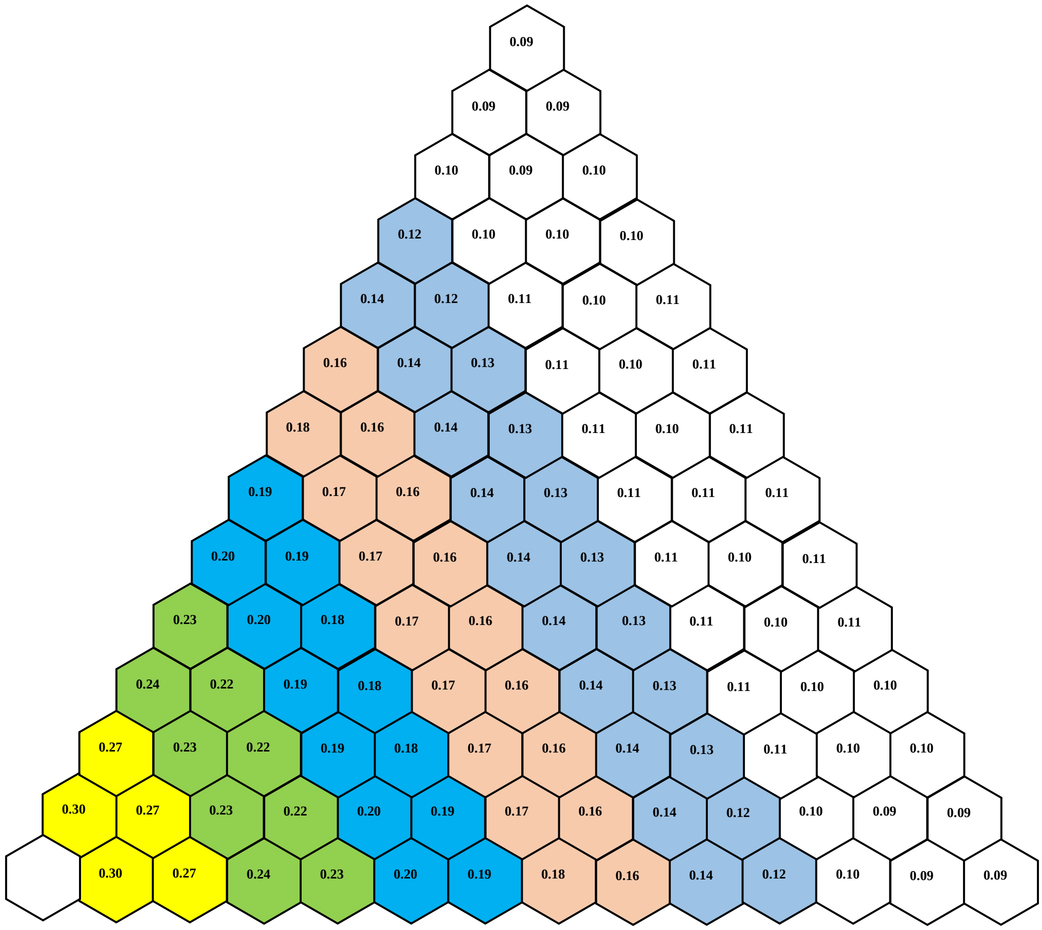
For kinetics parameters calculation, in SRAC code point reactor kinetics model is applied together with adjoint neutron flux calculation. In MCNP code, delayed neutron faction can be estimated by evaluating ratio of effective multiplication factor with and without delayed neutron using **totnu** option. To estimate prompt neutron life time by MCNP code, method 1/v insertion with boron-10 is applied. The method is better than point kinetics method because calculation model is treated in energy and space. Error of delayed neutron fraction and prompt neutron life time in MCNP code is about 5%. **Table IX** depicts calculation results of both codes with difference about 5% and 10% to delayed neutron fraction and prompt neutron life time respectively.

Table IX. Calculation results of kinetics parameters by MCNP and SRAC codes

Core	Delayed neutron fraction β_{eff} (%)		Prompt neutron life time (s)	
	MCNP	SRAC	MCNP	SRAC
PWR1	7.751E-01	7.678E-01	5.470E-05	5.603E-05
PWR2	8.016E-01	7.802E-01	4.988E-05	5.173E-05
PWR3	8.063E-01	7.942E-01	4.535E-05	4.705E-05
PWR4	8.165E-01	7.988E-01	4.453E-05	4.611E-05
PWR5	7.902E-01	7.780E-01	5.037E-05	5.211E-05
VVER1	7.728E-01	7.555E-01	5.754E-05	6.121E-05
VVER2	7.978E-01	7.640E-01	5.304E-05	5.810E-05
VVER3	8.009E-01	7.712E-01	5.281E-05	5.667E-05
VVER4	8.175E-01	7.742E-01	5.338E-05	5.600E-05
VVER5	7.737E-01	7.561E-01	5.865E-05	6.196E-05

Burn-up calculation for VVER5 and PWR5 cores is performed by MCDL [13] code (in-house development) with assuming that critical assembly is operated continuously 600 full power days and 100 days cooling. Burn-up percent of U-235 in each fuel rod only have small values under the operation time. **Fig.7** presents burn-up distribution of VVER5 and PWR5 benchmark cores.

The MCDL code is developed with more convenience for users in which the isotope Pr-141 is used to replace the lumped fission product because its cross section is nearly the same with the lumped fission product. Also, 21 actinide and 50 fission product isotopes are considered to suffice for depletion calculation. Reaction chains including reaction types used in the MCDL code have been taken from the SRAC system. The library of the MCDL code contains data of isotope label in the MCNP5 code, user's defined name of 71 isotopes, decay constants and fission yield products of 21 actinide isotopes to 50 fission product isotopes. In MCNP code calculation, ENDF/B.7.0 library is applied and each fuel rod is divided to 5 nodes in axial direction with the same volumes. The MCDL code can be run on Windows or Linux operating system under MPI environment for parallel calculation.



0.0011	**0.0011**	**0.0011**	**0.0011**	**0.0010**	**0.0010**	**0.0009**	**0.0009**	**0.0008**	**0.0007**	**0.0007**	**0.0006**	**0.0006**
0.0010	**0.0010**	**0.0010**	**0.0010**	**0.0010**	**0.0009**	**0.0009**	**0.0008**	**0.0008**	**0.0007**	**0.0006**	**0.0006**	**0.0006**
0.0012	**0.0012**	**0.0012**	**0.0012**	**0.0011**	**0.0011**	**0.0011**	**0.0010**	**0.0009**	**0.0008**	**0.0007**	**0.0006**	**0.0007**
0.0014	**0.0014**	**0.0014**	**0.0013**	**0.0013**	**0.0013**	**0.0012**	**0.0011**	**0.0011**	**0.0009**	**0.0008**	**0.0007**	**0.0007**
0.0017	**0.0017**	**0.0017**	**0.0017**	**0.0016**	**0.0016**	**0.0015**	**0.0014**	**0.0013**	**0.0011**	**0.0009**	**0.0008**	**0.0008**
0.0019	**0.0019**	**0.0018**	**0.0018**	**0.0018**	**0.0017**	**0.0017**	**0.0016**	**0.0014**	**0.0011**	**0.0010**	**0.0008**	**0.0009**
0.0021	**0.0021**	**0.0021**	**0.0021**	**0.0020**	**0.0020**	**0.0019**	**0.0017**	**0.0015**	**0.0012**	**0.0011**	**0.0009**	**0.0009**
0.0022	**0.0022**	**0.0022**	**0.0022**	**0.0022**	**0.0021**	**0.0020**	**0.0017**	**0.0016**	**0.0013**	**0.0011**	**0.0009**	**0.0010**
0.0025	**0.0025**	**0.0025**	**0.0024**	**0.0024**	**0.0022**	**0.0020**	**0.0018**	**0.0016**	**0.0013**	**0.0011**	**0.0010**	**0.0010**
0.0026	**0.0026**	**0.0025**	**0.0025**	**0.0024**	**0.0022**	**0.0021**	**0.0018**	**0.0017**	**0.0013**	**0.0012**	**0.0010**	**0.0011**
0.0029	**0.0028**	**0.0028**	**0.0025**	**0.0025**	**0.0022**	**0.0021**	**0.0018**	**0.0017**	**0.0014**	**0.0012**	**0.0010**	**0.0011**
0.0031	**0.0030**	**0.0028**	**0.0026**	**0.0025**	**0.0022**	**0.0021**	**0.0019**	**0.0017**	**0.0014**	**0.0012**	**0.0010**	**0.0011**
	0.0031	**0.0029**	**0.0026**	**0.0025**	**0.0022**	**0.0021**	**0.0019**	**0.0017**	**0.0014**	**0.0012**	**0.0010**	**0.0011**

Fig.7. Burn-up (%U-235) distribution of VVER5 and PWR5 benchmark cores (burn-up values in VVER5 core is multiplied with 10^{-2})

Burn-up distribution of both benchmark cores is also symmetry through diagonal line and low burn-up percent values of U-235. This is an advantage of the critical assembly and fuels can be used for long time and handled after cooling 2 or 3 days.

Shielding calculation

Shielding calculation for critical assembly is only paid attention about gamma rays during critical assembly operation. ORIGEN code is used for calculation of photon spectrum with 18 energy groups and the spectrum is put in input of MCNP code for gamma dose rate estimation. Gamma dose rate calculation is assured with error under 1% and ICPR-21 table to convert photon flux to gamma dose rate is applied. Calculation model is depicted in **Fig. 8** and shielding calculation results are presented in **Table X** with or without 50cm thickness of concrete around critical assembly. Variance reduce technique in MCNP for shielding calculation is chosen with weight window option that treats depending on space and energy.

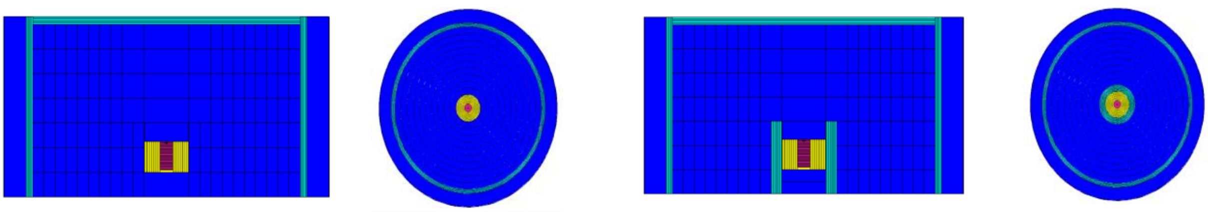


Fig. 8. Calculation model in MCNP code for gamma dose rate estimation

Concrete using in the problem has density ablout 2.30 g/cm^3 as normal compositions. Thickness of wall concrete around critical assembly hall is about 30cm. The shielding around critical assembly with different materials is also an object in research of critical assembly. Two representative benchmark cores VVER2 and PWR5 have highest photon flux are calculated detaily. If necessary, heavy concrete with high density (3 to 3.5 g/cc) can be used for gamma shielding around critical assembly building especially at the wall interface with control room. Because the top of critical assembly is not shielded by any heavy material, refeclect effect on ceiling is mainly contributed in gamma dose rate in calculation.

Table X. Calculation results of gamma dose rate (μSv/h) of critical assembly

Without shielding concrete around critical assembly

VVER Height 1m-3m	**Distance from outside critical assembly (cm)**									
	50	**100**	**150**	**200**	**250**	**300**	**350**	**400**	**450**	**500**
operation	8.00E+03	3.62E+03	1.92E+03	1.15E+03	7.60E+02	5.40E+02	4.07E+02	3.20E+02	2.62E+02	2.22E+02
cooling 1 day	2.78E+01	1.23E+01	6.47E+00	3.85E+00	2.54E+00	1.81E+00	1.37E+00	1.09E+00	8.95E-01	7.64E-01

Outside concrete 30cm

VVER Radius 1m	**Height (cm)**						
	50	**100**	**150**	**200**	**250**	**300**	**Average**
Operation	5.34E+00	6.13E+00	6.75E+00	7.89E+00	1.01E+01	1.29E+01	**8.18E+00**
Cooling 1 day	5.85E-03	7.10E-03	8.47E-03	1.13E-02	1.64E-02	2.25E-02	**1.19E-02**

Shielding 50cm concrete around critical assembly

VVER Height 1m-3m	**Distance from outside critical assembly (cm)**								
	50	**100**	**150**	**200**	**250**	**300**	**350**	**400**	**450**
Operation	5.28E+01	4.33E+01	3.68E+01	3.20E+01	2.83E+01	2.53E+01	2.29E+01	2.09E+01	1.94E+01

Cooling 1 day	1.10E-01	1.17E-01	1.15E-01	1.09E-01	1.02E-01	9.49E-02	8.81E-02	8.20E-02	7.72E-02

Outside concrete 30cm

VVER Radius 1m	**Height (cm)**						
	50	**100**	**150**	**200**	**250**	**300**	**Average**
Operation	1.68E-01	1.95E-01	2.16E-01	2.34E-01	2.64E-01	2.93E-01	**2.28E-01**
Cooling 1 day	6.99E-05	8.74E-05	1.03E-04	1.22E-04	1.43E-04	1.70E-04	**1.16E-04**

Without shielding concrete around critical assembly

PWR Height 1m-3m	**Distance from outside critical assembly (cm)**									
	50	**100**	**150**	**200**	**250**	**300**	**350**	**400**	**450**	**500**
Operation	1.84E+04	8.66E+03	4.68E+03	2.81E+03	1.84E+03	1.28E+03	9.48E+02	7.33E+02	5.90E+02	4.91E+02
Cooling 1 day	6.82E+01	3.16E+01	1.70E+01	1.02E+01	6.66E+00	4.68E+00	3.49E+00	2.73E+00	2.22E+00	1.87E+00

Outside concrete 30cm

PWR Radius 1m	**Height (cm)**						
	50	**100**	**150**	**200**	**250**	**300**	**Average**
Operation	7.83E+00	9.14E+00	1.07E+01	1.50E+01	2.25E+01	3.15E+01	**1.61E+01**
Cooling 1 day	8.87E-03	1.12E-02	1.51E-02	2.48E-02	4.07E-02	5.93E-02	**2.67E-02**

Shielding 50cm concrete around critical assembly

PWR Height 1m-3m	**Distance from outside critical assembly (cm)**								
	50	**100**	**150**	**200**	**250**	**300**	**350**	**400**	**450**
Operation	1.18E+02	9.71E+01	8.23E+01	7.13E+01	6.28E+01	5.59E+01	5.02E+01	4.57E+01	4.22E+01
Cooling 1 day	2.60E-01	2.76E-01	2.69E-01	2.54E-01	2.37E-01	2.19E-01	2.03E-01	1.88E-01	1.77E-01

Outside concrete 30cm

PWR Radius 1m	**Height (cm)**						
	50	**100**	**150**	**200**	**250**	**300**	**Average**
Operation	3.01E-01	3.60E-01	4.11E-01	4.79E-01	5.84E-01	6.91E-01	**4.71E-01**
Cooling 1 day	1.43E-04	1.88E-04	2.25E-04	2.84E-04	3.43E-04	4.12E-04	**2.66E-04**

While operating at nominal power 100W, if no shielding around critical assembly is equipped gamma dose rate of critical assembly is very high inside critical assembly building with values about 220 to 490 μSv/h of VVER and PWR cores respectively. The gamma dose outside 30 cm concrete thickness critical assembly building is about 8 to 16 μSv/h to VVER and PWR cores respectively. It is very high gamma dose rate. If using 50cm concrete thickness for shielding around critical assembly tank, gamma dose rate is extremely reduced. After cooling 1 day, gamma dose rate at any position is low enough. It is better to use 50 cm concrete thickness for shielding around critical assembly tank so finance investment is economic and more effective for shielding than using 50 cm concrete thickness around critical assembly building.

B. Thermal hydraulics calculation and safety analysis

Thermal hydraulics calculation

Because of operating at low power, natural convection for cooling fuel inside critical assembly core is applied and enough for heat removal from fuel rods. Cooling system is not necessary for operation of critical assembly so it makes structure of critical assembly became simpler than research reactor. All benchmark critical configurations are calculated for fuel pellet and cladding temperature with assuming temperature of water inside the core tank is about 27^{o}C and this value is higher than normal operation. Analytical calculation together with using RELAP5/MOD3.2 code is applied to evaluate temperature of fuel and cladding at the hottest fuel rod in benchmark critical configurations at steady state. Calculation model in RELAP code of VVER and PWR benchmark critical cores is shown in **Fig. 9**.

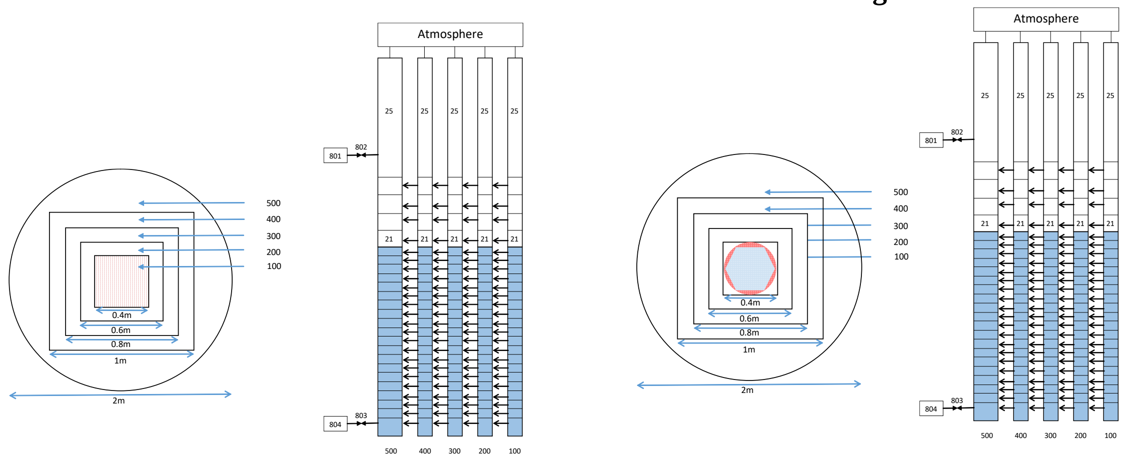


Fig. 9. Calculation model in RELAP code of VVER and PWR benchmark critical configurations

Calculation results of fuel temperature and cladding are presented in **Table XI**. Temperatures of fuel and cladding are not changed so much when comparing with water temperature inside critical assembly tank. This means critical assembly can be safely operated under nominal power around 100W.

Table XI. Calculation results of fuel pellet and cladding by the RELAP5 code

Core	**Temperature (ºC)**		**Core**	**Temperature (ºC)**	
	Fuel pellet	**Cladding**		**Fuel pellet**	**Cladding**
PWR1	27.0178	27.0172	**VVER1**	27.0174	27.0170
PWR2	27.0177	27.0171	**VVER2**	27.0173	27.0169
PWR3	27.0176	27.0170	**VVER3**	27.0173	27.0170
PWR4	27.0180	27.0174	**VVER4**	27.0179	27.0175
PWR5	27.0172	27.0167	**VVER5**	27.0174	27.0170

Safety analysis

Safety analysis of critical assembly has been carried out with technical specific parameters of benchmark critical configurations in **Table XII**.

For conservative calculation, responses time of safety system of critical assembly are extended but in practice, these responses time have smaller than 2 or 3 seconds only.

Table XII. Benchmark critical configuration parameters used for safety analysis

Parameters	**Values**
Power, W	100
Coolant inlet temperature, ºC	27
Peaking factor at critical status	
- Axial peaking factor	1.889 – 2.287
- Radial peaking factor	1.202 – 1.640
Reactor kinetics	
- Prompt neutron life, s	4.453-5.865×10^{-5}
- Delayed neutron fraction (1$)	7.737-8.175×10^{-3}
Temperature reactivity coefficients	
- Moderator, %/K; (300-350ºK)	- 0.441-1.314×10^{-2}

- Fuel, %/ºC;	
(300-400ºK)	- 1.26-1.69×10^{-3}
(400-500ºK)	- 1.13-1.53×10^{-3}
(500-600ºK)	- 0.974-1.39×10^{-3}
Reactivity control	
- Shutdown worth, %	0.4
Reactor protection characteristics	
- Control rod drop, s	10
- Response time to drain water tank, s	
Fast relief valve	20
Reducing by gravity force	30

Control rod worths in safety analysis are assumed 0.4%Δk/k for conservative calculation but in fact, effective reactivity of control rods is higher than this value. The reactivity insertion into three cases calculation is approximated 1.0$, 2.0$ and 3.5$. In safety analysis, the safety system of critical assembly includes safety control rod, fast relief valve for reducing of 10cm water level and valve to drain water in critical assembly tank by connected with supply water tank. To describe action of changing water level and reactivity in RELAP code, a table of water level depending on reactivity is established and used in trip option. Detailed calculation results in three cases with different reactivity insertion are described in **Table XIII** and **Fig.10**. For safety analysis, reactivity insertion in the core only when water level is increased after opening water supply valve and reloading neutron source in the core.

Table XIII. Parameters of critical assembly with power 110W to 550W in representative VVER and PWR cores

Power from 112 to 151W

Parameter	VVER	PWR
Maximum power	151.8W	112.87W
Temperature of water in tank	300.150ºK or 27.000ºC	300.150ºK or 27.000ºC
Temperature of fuel pallet	300.260ºK or 27.110ºC	300.244ºK or 27.091ºC
Temperature of fuel gap	300.253ºK or 27.103ºC	300.240ºK or 27.09ºC
Temperature of fuel cladding	300.251ºK or 27.101ºC	300.238ºK or 27.088ºC

Power from 246 to 297W

Parameter	VVER	PWR
Maximum power	296.9W	246.7W
Temperature of water in tank	300.15ºK or 27.000ºC	300.150ºK or 27.000ºC
Temperature of fuel pallet	300.314ºK or 27.164ºC	300.277ºK or 27.127ºC
Temperature of fuel gap	300.310ºK or 27.160ºC	300.273ºK or 27.123ºC
Temperature of fuel cladding	300.298ºK or 27.148ºC	300.268ºK or 27.118ºC

Power from 538 to 540W

Parameter	VVER	PWR
Maximum power	540.2W	538.2W
Temperature of water in tank	300.15ºK or 27.000ºC	300.15ºK or 27.000ºC
Temperature of fuel pallet	300.417ºK or 27.267ºC	300.382ºK or 27.232ºC
Temperature of fuel gap	300.409ºK or 27.259ºC	300.374ºK or 27.224ºC
Temperature of fuel cladding	300.387ºK or 27.237ºC	300.362ºK or 27.212ºC

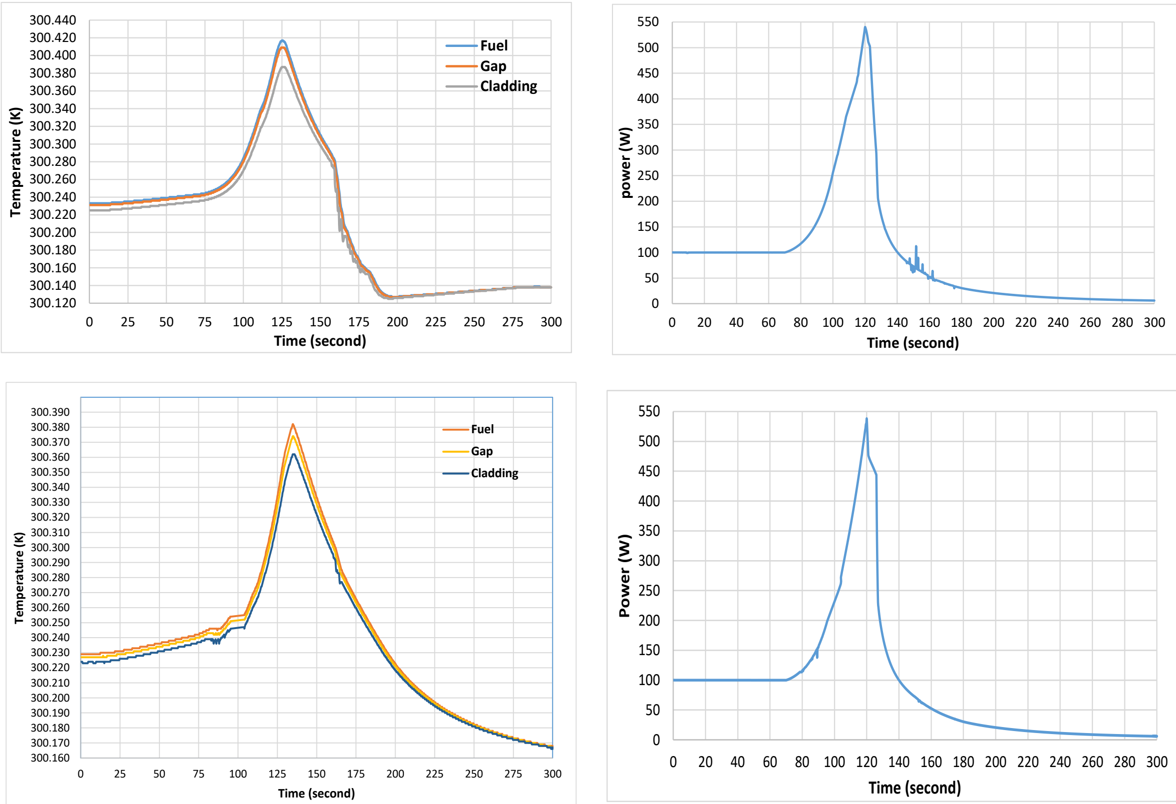


Fig. 10. Fuel temperature and power of critical assembly in case power increasing from 100 W to approximate 500 W of VVER core (upper) and PWR core (lower)

Calculation results show that critical assembly is safety facility even power can be reached to 5 times higher than nominal power but temperature of fuel and cladding is only slightly increased. In fact, the temperature of water inside reactor tank is not high as 27^{o}C so the temperature of fuel and cladding will be lower than calculation results.

IV. CONCLUSIONS AND REMARKS

Neutronics, thermal hydraulics and safety analysis for critical assembly using low enriched fuel of commercial NPPs have been carried out and based on calculation results, it is can concluded that:

- Critical assembly has very simple structure, system and components comparing with research reactor,

- Fuel of NPPs is completely used for critical assembly under low power and neutron flux,

- Temperature reactivity feedback coefficients of fuel and moderator have negative values to confirm about inherent safety of designed critical assembly,

- U-235 burn-up of critical assembly fuel is very low so fuel can be used very long time,

- Gamma shielding during operation of critical assembly has to be paid attention because gamma dose rate at normal operation of critical assembly is quite high. But after cooling 1 or 2 days the gamma dose rate decreases significantly,

- In thermal hydraulics and safety analysis aspect, temperatures of fuel pellet and cladding are slightly changed in steady state or transient with big positive value reactivity insertion.

The advantages of critical assembly are easy in siting, low cost investment and operating, low burn-up fuel, simple in operation and having good applications in basic research, training and education. However, critical assembly also has some disadvantages including low neutron flux to reduce sensitivity for neutron activation analysis, limited in radioisotope production, licensing requirement because of simple structure and only using one safety rod.

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