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Safety Analysis for the Institute of the Nuclear Physics Critical Assembly with LEU fuel

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ABSTRACT

Institute of Nuclear Physics (INP) in Alatau, Republic of Kazakhstan, operates the WWR-K research reactor, a Russian-designed and supplied reactor currently operating with HEU (36% U^{235} enrichment) fuel of Russian origin. Analysts from INP, in collaboration with the Nuclear Threat Initiative (NTI), Argonne National Laboratory (ANL), and Russian organizations, have selected an assembly design for the conversion of the reactor to LEU fuel. The qualification of the fuel assembly is proceeding by irradiating three lead test assemblies (LTA) in the WWR-K reactor to 60% average burnup.

WWR-K reactor complex includes a companion Critical Assembly (CA), having the same fuel assemblies as WWR-K reactor but operating at zero power (up to 100 W). The CA is used for physics experiments. In anticipation of successful LTA irradiation testing, INP is preparing to convert the CA to LEU fuel and DOE/NNSA is supporting this conversion.

As a necessary step in preparation for conversion, INP in collaboration with ANL has performed a safety analysis for CA with LEU fuel assemblies, including analysis of anticipated core configurations, reaching the first criticality and the final critical configuration, steady state, internal and external initiators accident analysis. The results of safety analysis are reported in the present paper.

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Introduction

In a period from 2003 to 2006, the Kazakhstan Institute of Nuclear Physics, which operates the WWR-K research reactor and Critical Assembly, carried out, under the US financial support (Nuclear Threat Initiative) search of relevant candidates for roles of new LEU fuel assembly composition and design [1-2]. Currently, VVR-C-type fuel assemblies with UO_2 –Al meat, 36% enriched in uranium-235 are used in the INP Kazakhstan Critical

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Assembly. Due to a number of technological reasons, the fuel composition on a base of uranium dioxide dispersed in aluminum matrix, having the uranium mass density 2.8 g/cm^3, enriched in U-235 to 19.7% was chosen. The thin-walled (1.6mm) eight-tube fuel assembly (FA) was recognized as the optimum design for the WWR-K reactor and Crittical Assembly. The nominal mass of uranium-235 in the FA is ~245 g; the area of heat-transfer surface reaches 1.34 m^2. As a necessary step in preparation for conversion, INP in collaboration with ANL has performed a safety analysis for CA with LEU fuel assemblies. In addition to general description of the CA safety-significant systems, CS geographical location and weather conditions, the report will include analysis of anticipated core configurations, the first-critical state and the final critical configuration, as well as core steady-states and accident analysis for internal and external initiators.

The critical facility function, type and specifics

The critical assembly (CA) is capable, in some cases (like, e.g., in case of conversion to new fuel composition) to reproduce the core of the water-water research WWR-K reactor. The critical assembly will be used for studies on substantiation of the WWR-K safe operation. Various reactor techniques will be assimilated; calculation codes will be verified, safety of experiments at the WWR-K reactor will be substantiated.

Specific features of the critical facility are as follows: design of the support plate makes it possible to compose critical assemblies with asymmetric configuration of the core, as well as to arrange experimental tubes of a diameter within the range from 63 to 320 mm.

The second feature of critical facility is a design of channels for the I&C driven elements, where small-size servodrives are used as executive mechanisms, which are fixed directly to a channel, making it possible, if necessary, to install the shim/safety channels (EE KO/EE AZ) practically, in any cell of the core.

Main parameters

1. The permitted thermal power of the critical assembly, which is dependent of biological shield, comprises 100 W.
2. Desalted water plays a role of moderator. Desalted water or beryllium serves as side reflector. Water also plays the roles of the upper and lower side reflectors.
3. The critical assembly involves six EE KO (shim elements), three EE AZ (protection rods) and a single EE AR (automatic rod). All control rods are equipped by independent drives of a mechanism responsible for control rod movements.
4. Values of the neutron flux density and relevant emergency protection by this parameter are controlled over three independent channels.
5. Control and emergency protection using the rate of increase in the neutron flux is also executed by three independent channels.
6. The moderator temperature is determined by the temperature of the premise (box) where critical assembly is located.
7. The thermal neutron flux density in the core central channels at the maximum power 100 W may reach $3.6 \cdot 10^9$ cm^{-2}s^{-1}.

The expected core maps

The following CA core maps, which are to be studied experimentally, have been considered via calculations:

- the core load which gives K_{eff} close to 0.98; this core is used for first assessment of efficiencies of EE KO and EE AZ;
- critical load, giving K_{eff} close to 1.0;
- the core initial operational core with water side reflector; reflector is formed by displacers.

The core map which corresponds to K_{eff} equal to 0.9835 (neutron multiplication factor of standard source is 58) is given in Fig. 1. There are 10 FA-2, under I&C EE, and 10 FA-1. Analysis of this load is mandatory and is fulfilled in compliance with requirements stated in regulatory documents.

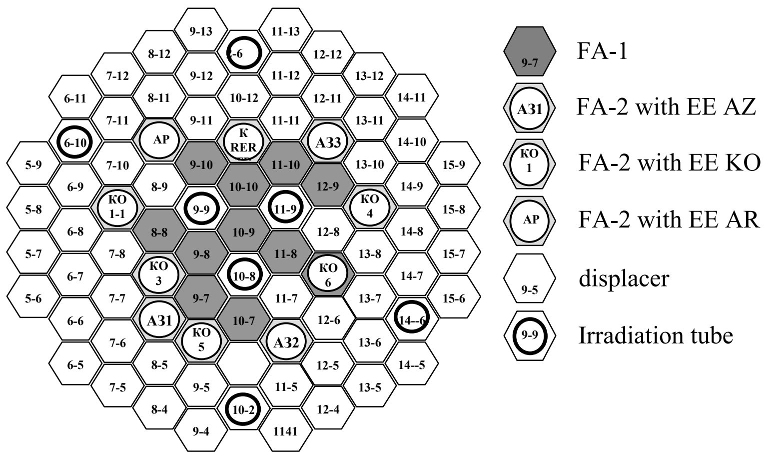


Fig. 1. The core map with 10 FA-1 and 10 FA-2

Figure 2 corresponds to critical state with 10 FA-2 and 11 FA-1. Figure 3 presents the cartogram with 10 FA-2 and 16 FA-1 and water side reflector, which corresponds to operational load of the core.

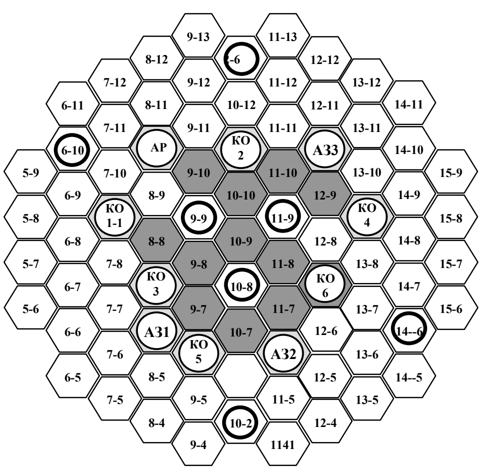


Fig. 2. The critical load core map

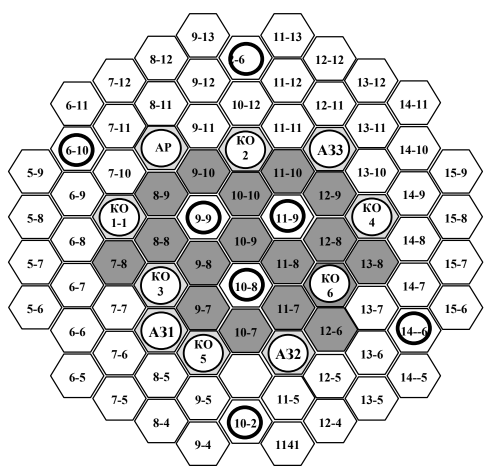


Fig. 3. The operational load core map

Neutron-physical characteristics of the core

Figure 1 presents the core map with K_{eff} =0.9835. The FA loading sequence prior to coming to this value of K_{eff} is given in Table 1.

Table 1 – Values of K_{eff} in course of the CA core critical mass building-up

Loaded FA	Amount of FAs	K_{eff}
8–6 (1AZ)	3 FA-2	0.3651(4)
11–6 (2AZ)		
12–10 (3AZ)		
10–11 (2KO)	6 FA-2	0.4864(4)
8–7 (3KO)		
12–7 (6KO)		
13–9 (4KO)	9 FA-2	0.6025(4)
7–9 (1KO)		
9–6 (5KO)		
8–10 (AR)	10 FA-2	0. 61354)
10–9, 11–8	2 FA-1	0.7328(4)
9–8, 10–10	4 FA-1	0.8213(4)
9–7, 8–8	6 FA-1	0.8919(4)
11–10, 12–9	8 FA-1	0.9345(5)
9–10, 10–7	10 FA-1	**0.9835(5)**
11–7	11 FA-1	**1.0012(5)**

In compliance with the RK nuclear safety regulatory requirements, the first evaluation of the I&C CR efficiencies is to be determined when K_{eff} ~ 0.98. Relevant calculated values of efficiencies for the I&C CRs are presented in Table 2.

Table 2 – Efficiencies of the I&C CRs for core map 10 FA-1+10 FA-2 (K_{eff} = 0.9835)

CPS CR	Efficiencies, % Δk/k	CPS CR	Efficiencies, % Δk/k
1AZ	1.26	3KO	2.36
2AZ	1.04	4KO	0.72
3AZ	0.88	5KO	1.34
ΣAZ*	3.27	6KO	1.35
1KO	1.12	ΣKO	8.64
2KO	1.75	ΣKO*	9.76

*all rods are inserted to the core simultaneously

Figure 2 shows the CA critical load core map. The critical state occurs after loading of FA in position 11-7 as shown in Table 1.
The operational core load consists of 16 FA-1 and 10 FA-2; the core excess reactivity is equal to 6.2%Δk/k. Efficiencies of the I&C control rods are given in Table 3.

Table 3 – Efficiencies of I&C EEs for operational core load (k_{eff}≈1)

EE	Efficiency, %Δk/k	
	MCU[3]	MCNP5 [4]
1AZ	0.81	0.88
2AZ	1.02	1.13
3AZ	0.86	0.79
All AZ*	2.93	2.97
AP**	0.31	0.21
1KO	1.00	1.06
2KO	1.65	1.63
3KO	2.08	2.14
4KO	1.01	1.15
5KO	1.10	1.21
6KO	2.33	2.33

*all AZ are inserted simultaneously, ** in calculations with MCU and MCNP different grades of steel were used as candidate materials for AR

Analysis of the data presented in Tables 2 and 3 shows that interference of EE KO is positive, i.e., the difference between the efficiencies of simultaneously inserted EE KO and sum of individual efficiencies of EE KOs is positive, making the core safer.

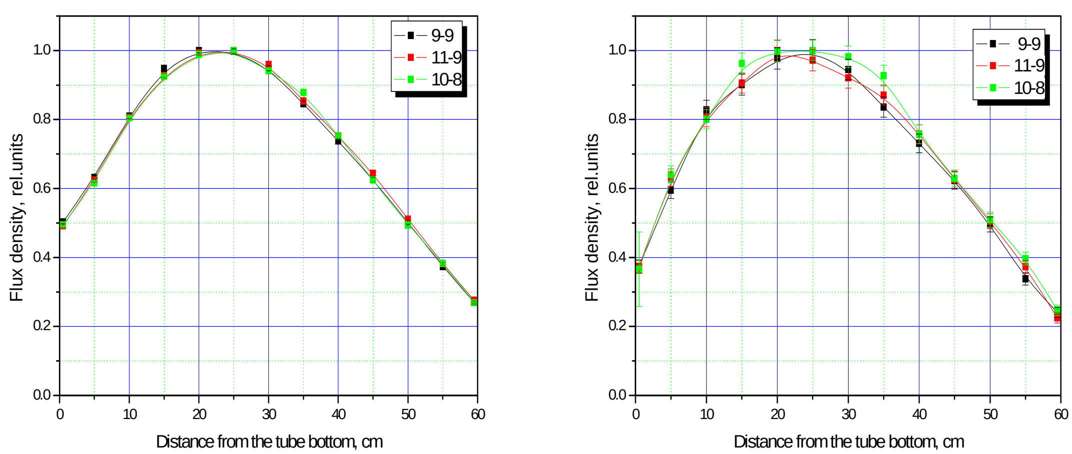
Table 3 shows that the I&C EEs fully meet regulatory requirements. For example, the net efficiency of EE AZ without the most effective one comprises 2.2 $ß_{eff}$. The efficiency of the EE AR is less than 0.7 $ß_{eff}$. The calculated value of the neutron life time is $0.47 \cdot 10^{-4}$ s; the effective fraction of delayed neutrons $ß_{eff} = 0.757\%$.

For the operational load core map with water side reflector, the thermal/fast neutron flux densities in irradiation tubes were calculated. Three irradiation tubes will be located in central cells 9-9, 11-9 and 10-8. Table 5 presents values of the thermal (En<0.4 eV) and fast (En>1.15 MeV) neutron flux densities at the critical assembly nominal power 100 W.

Table 5 - The neutron flux densities in the CA irradiation tubes

Cell	Neutron flux density	
	Thermal (En < 0.4 eV)	Fast (En > 1.15 MeV
9-9	$(3.4 \pm 0.1)\ 10^9$	$(6.3 \pm 0.2)\ 10^8$
11-9	$(3.5 \pm 0.1)\ 10^9$	$(7.0 \pm 0.2)\ 10^8$
10-8	$(3.6 \pm 0.1)\ 10^9$	$(7.12 \pm 0.2)\ 10^8$

Figure 6 shows axial distribution of the thermal and fast neutron flux densities in central irradiation tubes. The distribution corresponds to the CA core critical state.



(a) (b)

Fig. 6. Axial distribution of the thermal (a) and fast (b) neutron flux density

The axial irregularity factor for thermal and fast neutrons is 1.41 and 1.46 respectively

Analysis of potential accidents at the critical facility

The following initiating events are analyzed:

1.1. Spontaneous insertion of positive reactivity. This initiator is assumed to be caused by the spontaneous withdrawal of the most effective EE KO. The efficiency of this control rod is equal to 2.33% (see Table 3), and it is assumed that the rod is withdrawn with a speed of 8 mm/s. The CA is at its maximum permitted power 100 W at the time of the event. (each control rod has its own drive.) Simultaneously, in course of actuation of emergency protection, it is conservatively assumed that the most effective safety rod (EE AZ) is stuck in its uppermost position

1.2. As a result of personnel's mistake, extra FA or block of beryllium is loaded to the core (to the closest to the core center cell 10-6) with maximum reactivity worth.

1.3. As a result of failures in the system of control for gate valve for emergency water discharge, it opens spontaneously after operation of CA at 100W for a shift (7 hours). On conservative grounds, it is assumed that the core dries at once.

The results of analysis of the initiating events described above are provided below:

2.1. Calculations performed with the PARET [5] code have shown that the power runaway in this case does not exceed 122 W (see Fig. 7 below). Emergency protection will actuate with 0.3-s delay, and the safety (AZ) will be fully inserted into the core from their withdrawn position in 0.7 s the control (KO) rods will drop into the core in less than 0.7 s since they are already partially inserted into the core bringing the CA power to decay heat level in few seconds, and in about 100 second the CA power will reduce to about 0. About 1000 J of energy will be released for this period of time. (When CA operates at 100W, $2.5 \cdot 10^6$ J is released for 7 hours). The temperatures of coolant, clad and fuel stay, practically, unchanged from their steady-state values.

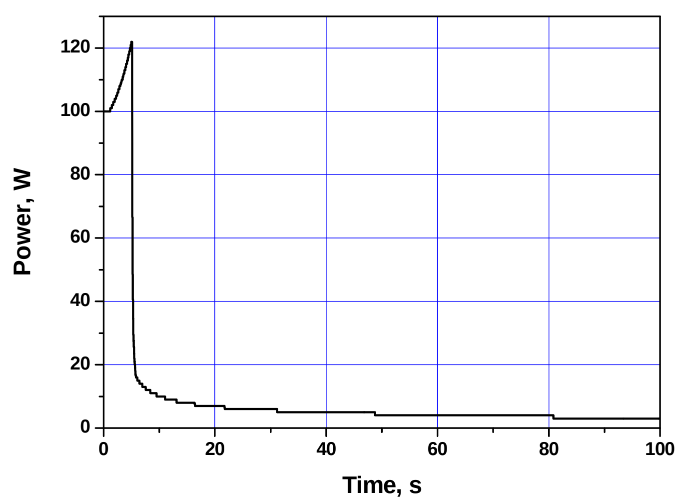


Fig. 7. Power vs. time in case of spontaneous withdrawal of the most effective shim element

2.2. As a result of erroneous loading of extra FA to cell 10-6, the core excess reactivity has increased from 6.2% to 6.9%, which can be reliably compensated by the EE KO. Thus, this initiating event does not result in occurrence of emergency situation (scram or trip).

2.3. By result of calculation with the MCNPX code, in case of core uncover the gamma radiation dose rate in front of the CA box outer protective door is equal to ~0.9 mR/h; so, there is no threat to the health of the operating personnel in the CA control room.

Conclusion

The INP Safety Analysis report has been developed for the CA LEU fuel core. Potential core maps have been analyzed with their relevant neutron–physical characteristics.

Three initiating events with unauthorized insertion of excess reactivity have been analyzed. Calculations have shown that the considered events do not present threat neither to health of the CA personnel in CA control room or to neighboring population outside the reactor building.

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