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Safety analysis for the WWR-K research reactor converted to LEU fuel

S.N. Koltochnik*, A.A. Shaimerdenov

The Institute of Nuclear Physics, Almaty, Kazakhstan

E-mail: svetlana.koltochnik@gmail.com

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Recently in the WWR-K water-water research reactor the former HEU fuel, enriched to 36% in Uranium-235, was changed to LEU one, enriched to 19.7%, with substantial change of the core configuration. In view of reactor conversion, a new Safety Analysis Report (SAR) was developed for the WWR-K reactor. Substantiation of reactor safe operation under both normal operation and emergency conditions was done under thermal-hydraulic approach. In the analysis developed prior to physical start up it was assumed that a main circulation pump (MCP) provides the coolant flow rate in the core equal to 350 m^3/h (a certified value for the pump CB-321). However, in course of the reactor physical start up it was found that it is only 250 m^3/h. The reason was a decision of the reactor staff to reduce the primary pump power consumption, prolonging its life time.Therefor the thermal hydraulic analysis was revised, and the SAR was renewed. Safety analysis implies also consideration of some potential initiating events capable to develop into an accident. So, several typical initiating events are subject to thermal-hydraulic analysis to substantiate observance of nuclear and radiation safety in emergency situations. It is shown that, owing to proper operation of safety systems, the initiating events under consideration don't result into accidents, if two primary pumps provide not more than 585 m^3/h.

Keywords: research reactor, LEU fuel, safety analysis report, thermal-hydraulic analysis, fuel assembly, ONBR, coolant flow rate, steady states, transients.

Introduction

Prior to physical startup with new fuel, the Safety Analysis Report of the WWR-K research reactor converted to LEU fuel (SAR) was done. A significant part of the SAR is a summary of thermal-hydraulic analysis of the core steady states and of some initiating events capable to result in accidents against "hot" spots and the coolant flow rate in the core [1-2]. The coolant is ordinary water. Such hydraulic characteristics of the core elements as coolant passage areas, hydraulic diameters, the wet and heated perimeters, as well as “hot” spots, determined with application of the MCNP5 code [3] for the initial core configuration with fresh fuel, being independent of the primary flow core, are given in our previous publications (see [1] and references therein). Such hydraulic parameters of the core as the coolant flow rates and velocities in the core elements (fuel assemblies, displacers, experimental channels) were re-calculated.

This paper covers results of thermal-hydraulic analysis of the new core stationary states with application of the code PLTEMP v.4.1 [4] and analysis of the transients capable to result in accident - with the code PARET v.7.5 [5], as well as estimated safe limits of the reactor power as functions of the coolant flow rate [2], which were performed on a base of the neutron-physical calculations of the core in order to find “hot” spots subject to the analysis.

Statement of the work to be done is as follows:

• Hydraulic calculation of the core elements (coolant passage area, hydraulic diameters, hydraulic resistances, the coolant flow rates and velocities in core elements) – in Excel sheets.

• Neutron-physical calculation (with the MCNP5/MCNP6 code), including:

- maximum generated power density over the core;
- axial distribution of the power density in the “hot” channel;
- kinetic parameters of the core;
- efficiencies of the control rods.

• Thermal calculation (peak coolant/clad temperatures, min nucleate boiling ratio (ONBR) – with the PLTEMP code.

• Combined heat-reactivity-flow rate calculations (peak coolant/clad temperatures, min ONBR, temporal variations in the coolant flow rate, the peak coolant/clad temperature, reactivity) – with the PARET code.

Description of LEU FAs

On a base of metalloceramic composition UO_2-Al with uranium density 2.8 g/cm^3, enriched to 19.7% in uranium-235, two types of the VVR-KN FAs are developed: eight-tube (FA-1) and five-tube (FA-2) [1-2]. The net contents of uranium-235 in FA-1 and FA-2: $\approx$250 g and $\approx$200 g respectively. The eight-tube FA represents a structure composed of seven concentric tubular fuel elements (FE) of hexagonal cross section, the inner cylindrical FE, head and tail. Inside of an 8-tube FE, structural tube $\oslash$8.8 mm is located. The five-tube FA, used for installation of the CPS CR channels, represents a structure composed of five FEs of hexagonal cross-sections.

A FE represents three-layer tube 1.6 mm thick, composed of inner and outer clad, 0.45 mm each, fuel meat 0.7 mm thick and end plugs. A gap between FEs – for coolant flow – comprises 2 mm. Outer sides of all FEs have stiffening ribs. The FA all structural elements are made of SAV-1 alloy.

The VVR-KN FA-1 cross section is shown in Figure 1. FA-2 is like FA-1 but without three inner FEs and structural tube.

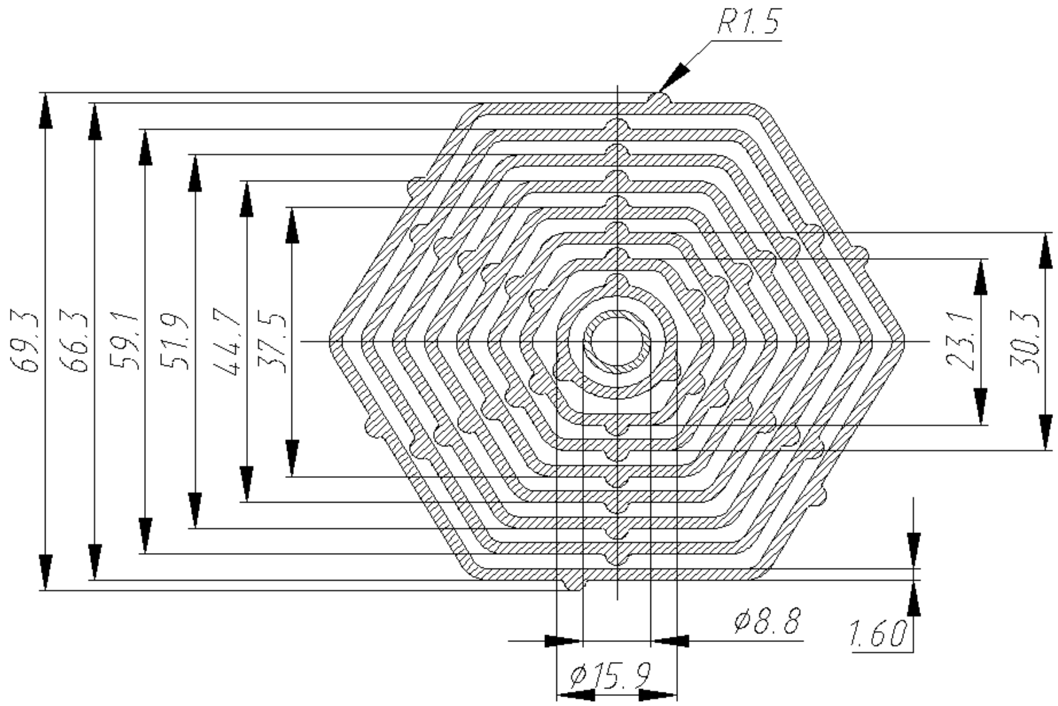

Figure 1. Cross sections of the VVR-KN eight-tube FA.

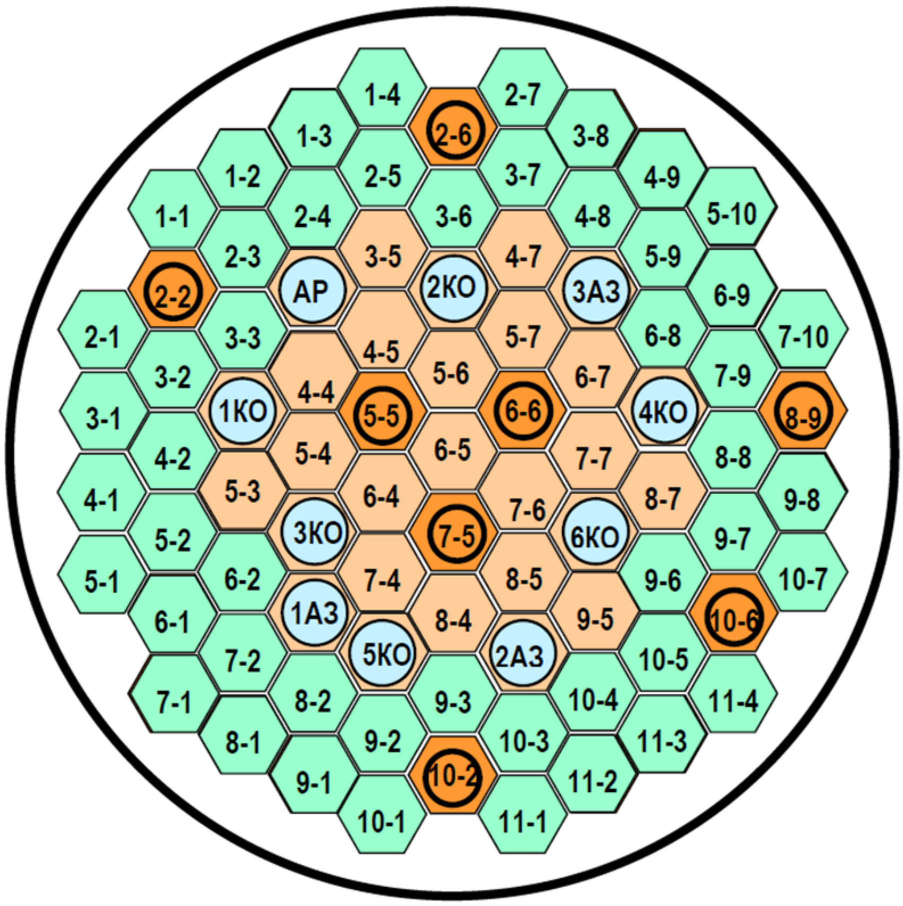

Figure 2. Initial work configuration of the core.

Configuration of the core with fresh fuel and water side reflector (start of the first operation cycle) under study is shown on Figure 2, where FA-1 are coloured with light-yellow; FA-2 are marked with blue inside; displacers, which form water side reflector, are marked with green; irradiation channels are equipped with bold circles. The core height (the FEs active part) is 610 mm. With the initial work load, critical state (k_{eff} =1) is reached with the control rods [automate regulator (AR) and six shim elements (KO)] inserted to 415 mm. Absorbing material of shim elements is boron carbide; automate regulator is made of stainless steel.

Recalculated hydraulic characteristics

The coolant flow rates and velocities in the core elements and pressure differential across the core are calculated in inter-connected Excel sheets, by one per every element and one sheet – for the entire core. Hydraulic characteristics (pressure differential as function of the coolant flow rate) of the FA-1 and FA-2 were determined earlier experimentally [1]. With the assumptions that the pressure differential is the same in all elements of the core and the coolant flow rate in the core is sum of the flow rates in its elements we obtain the hydraulic characteristics of the core as a whole along with the coolant flow rates and velocities in the core elements (see Table 1).

Table 1.
The flow rate in the core and core elements.

Core and core elements	Amount	Flow rate, m^3/h			
		Previous estimates		Last estimates	
		2 MCP	3 MCP	2 MCP	3 MCP
Core	1	700	1000	585	850
FA-1	17	17.9	25.87	15.08	21.98
FA-2	10	15.6	22.20	12.88	19.00
Irradiation channel	8	4.4	6.20	3.66	5.25
Displacer	51	4.1	5.77	3.41	4.89
Δp, MPa		0.0194	0.0387	0.0135	0.0277

Basing on the data from Table 1, pressure differential (Δ p, MPa) across the core versus the coolant flow rate (q, m^3/h) can be approximated by the power function: $\Delta p = 6.22\ 10^{-8}\ q^{1.931}$.

Neutron-Physical Calculations

Calculations of the generated power and power density in fuel as applied to the WWR-K RR initial work load (see Figure 2) were performed with the code MCNP5 v.1.51 [3] equipped with the cross-section library ENDF-VII.

Calculated values of the power (P), generated in the core cells filled with fuel, along with its FA-averaged power densities (Q) are given in Table 2.

Table 2.
Distribution of FA power and power density over FAs in the initial work load.

Cell #	Q, kW/cm^3	P, kW	Cell #	Q, kW/cm^3	P, kW
6-5	851	387.8	5-3	412	187.8
7-6	708	322.9	8-7	394	179.7
6-4	707	322.2	9-3	394	179.5
5-6	697	317.8	3-2 (AP)	505	185.6
7-4	545	248.7	4-3 (1KO)	382	140.4
5-4	518	236.0	4-6 (2KO)	454	166.8
5-7	536	244.3	6-3 (3KO)	414	152.2
6-7	542	247.3	7-8 (4KO)	389	142.9
4-5	564	257.2	8-3 (5KO)	376	138.1
8-4	524	239.0	8-6 (6KO)	458	168.4
8-5	597	272.2	7-3 (1AZ)	516	189.5
7-7	518	236.1	9-4 (2AZ)	567	208.1
4-7	424	193.5	5-8 (3AZ)	540	198.4
4-4	521	237.6	TOTAL:		6000

So, the hottest FA (387.8 kW) with the FA-average power density 851 W/cm^3 (against the core-average one, equal to 526 W/cm^3) is located in the core centre, in cell 6-5. Then its eight FEs are analyzed.

Distributions of the power (P) and the power density (Q) over FEs of the hottest FA are shown in Table 3. One may see that the FA hottest FE, with the power density 996 W/cm^3, is the outer one (FE-1).

Table 3.
Distributions of the power and the power density over FEs of the hottest FA.

FE No.	FE-1	FE-2	FE-3	FE-4	FE-5	FE-6	FE-7	FE-8
V, cm^3	94.02	83.48	72.94	62.40	51.86	41.32	30.78	19.1
P, kW	93.64	75.11	60.98	49.49	39.65	30.99	23.09	14.85
Q, W/cm^3	996	900	836	793	765	750	750	774

Distribution of the power and the power density over the hottest FE six rectangular sections (box) and six rounded corners (circ) are given in Table 4.

Table 4.
Distributions of the power and the power density.

FE element	box1	box2	box3	box4	box5	box6	circ1	circ2	circ3
V, cm^3	12.17	12.17	12.17	12.17	12.17	12.17	2.57	2.57	2.57
P, kW	15.30	10.21	15.21	10.27	10.27	15.46	2.85	2.86	2.62
Q, W/cm^3	1257	839	1249	844	843	1270	1113	1115	1023

Among six flats of this FE, the one faced to “wet” channel in cell 5-5 was found to be the hottest.The calculated height distributions of power density over 11 segments of the hottest flat (*box 6*) of the outer FE are shown in Figure 3. The segment *s7*, located in the outer flat centre (see Figure 4) and passing through cell 600, with power density nearly 1500 W/cm^3 is the hottest one.

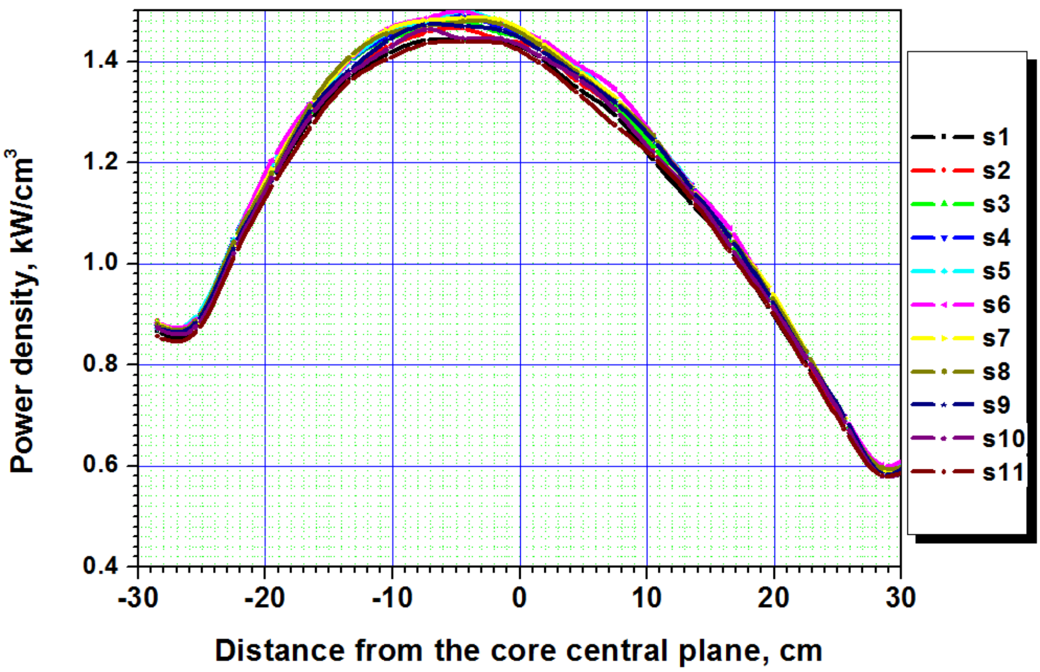

Figure 3. Height distributions of the power density in 11 segments of the hottest FA outer FE.

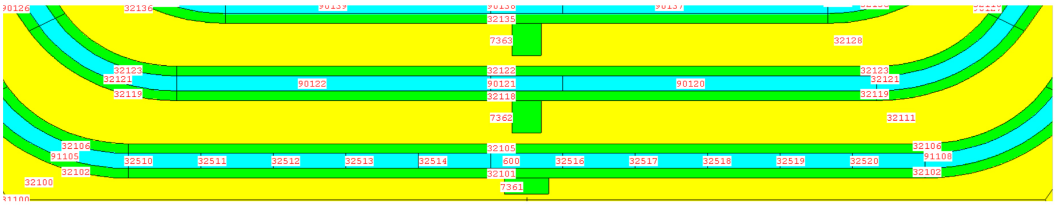

Figure 4. Segment s7 is cell 600 on screen shot of the MCNP Visual Editor image.

The calculated values of control rod efficiencies (needed for calculations with PARET relatively the critical state are given in Table 5.

Table 5.
Control rod efficiencies.

CR	AR	1KO	2KO	3KO	4KO	5KO	6KO	1AZ	2AZ	3AZ	1AZ&3AZ
Worth, % Δk/k	0.27	1.12	1.90	2.16	1.09	1.36	1.97	0.89	1.09	0.96	1.92

Kinetic parameters (needed for calculations with PARET) were obtained with the MCNP6 code, equipped with the cross-section library ENDF-VIII. A value of the prompt neutron lifetime is $4.6 \cdot 10^{-5}$ s; the effective fraction of delayed neutrons β_{eff}, is 0.0076 ± 0.0001.

Reactivity feedbacks:

- on coolant temperature (293-350 K): -0.009%(Δ k/k)/K);

- on fuel temperature (293-600 K): -0.002%(Δ k/k)/K);

- void feedback (over 5-% range of coolant density): -0.32%(Δ k/k)/%.

Steady-State Thermal-Hydraulic Calculations

Thermal-hydraulic analysis of the core stationary states is performed with application of the code PLTEMP v.4.2 [5]. Input is axial distribution of the specific generated power in "hot" channel, the flow rates in gaps between fuel elements, the pressure differential across the core as function of the coolant flow rate for every element of the core and such FA-1 hydraulic parameters as hydraulic diameter, wet perimeter, the coolant flow passage area, velocity [1], as well as specific heat conductivity of clad and fuel.

Thermal-hydraulic calculation has shown that for normal reactor operation the clad temperature and the onset of ONBR don't exceed relevant top thresholds stated by the VVR-KN FA developer (98 °C and 1.3 respectively), provided the coolant flow rate in primary circuit comprises not less than 585 m^3/h in the core at the coolant inlet temperature 45 °C (see Figure 5 and Table 6, where the "hot" FE segment height × width × thickness: is $61 \times 0.55 \times 0.16$ cm. The "hot spot" is a portion of "hot" segment 3 cm high, located by 145 mm lower that the core centre).

Table 6.
Main thermal-physical parameters of the hottest FA.

Parameter	Value	
The reactor thermal power, kW	6000	
the power generated in the hottest FA, kW	388	
the power generated in the "hot" segment, kW	14.7	
the core-averaged power density, W/cm^3	526	
the FA-averaged power density, W/cm^3	850	
the "hot" segment averaged power density, W/cm^3	1129	
the peak power density, W/cm^3	1500	
the core-averaged heat flux, W/cm^2	17.5	
the FA-averaged heat flux, W/cm^2	27.7	
the "hot" segment averaged heat flux, W/cm^2	37.6	
the "hot spot" heat flux	50.1	
the flow rate (m^3/h): in primary circuit /in FA-1	585/15.1	850/22
the coolant inlet temperature, °C	45	
the coolant peak temperature, °C	71.0	63.0
the clad peak temperature, °C	91.4	79.4
the ONB temperature, °C	114.6	112.5
the ONBR min value	1.48	1.95

Analysis of transients

For emergency conditions, it must be shown that reactor safety systems are capable to bring reactor into a safe state in any case. Analysis of the transients capable to result in accident has been performed with the code PARET v.7.5. Several typical initiating events (such as loss of external power supply, failure of primary

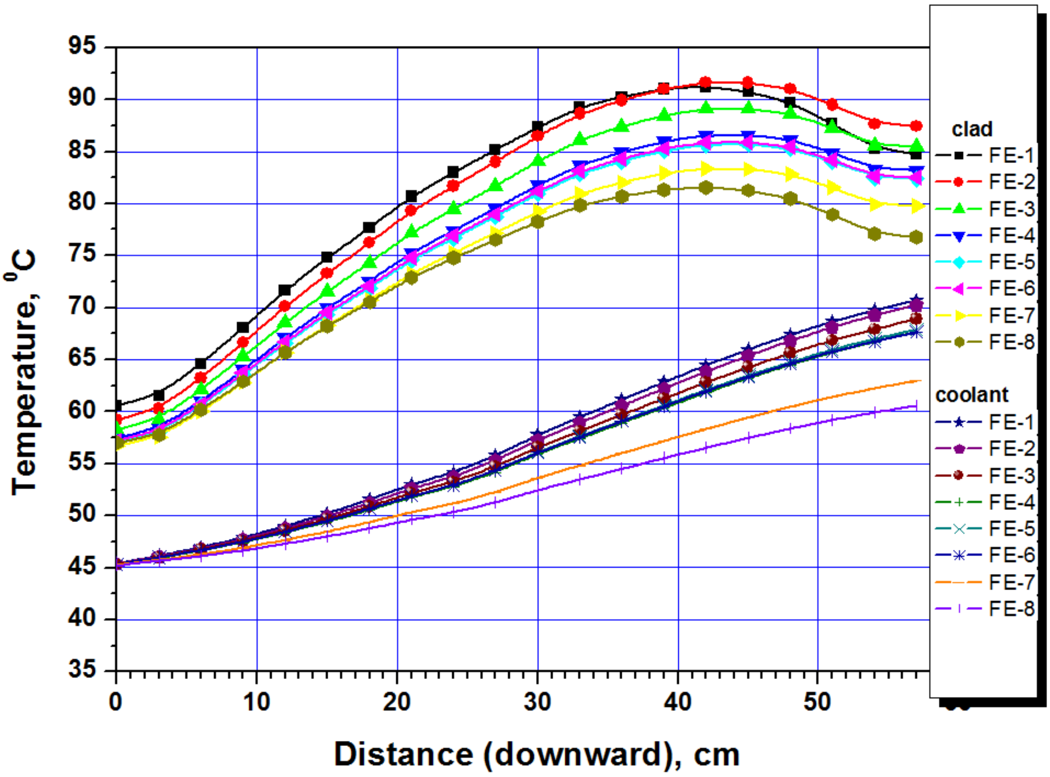

Figure 5. Height distributions of the coolant and clad temperatures over the hot sector.

pump) were subject to thermal-hydraulic analysis to substantiate observance of nuclear and radiation safety in emergency situations.

Figures 6-8 illustrate outcomes of the analysis for spontaneous withdrawal of the most effective compensation rod 3KO, when the most effective protection rod (2AZ) (see Table 5) is stuck. Scram happens after 6.3 second as reaction to 20-% increase in power (from 6 to 7.2 MW). Five out of six compensation rods and two out of three protection rods are inserted to the core for less than 1 second. However, 3KO still moves upwards with the rate 4 mm/s (the maximum rate value guaranteed by manufacturer), introducing positive reactivity. For 3KO, initially inserted to the core to 415 mm, it implies that it will be fully withdrawn for $\approx$104 s, however, reactors stays in deep subcritical state. The calculated peak values of the clad / coolant temperatures: 105.1 $^\circ$C / 77.9 $^\circ$C; the ONB temperature 114 $^\circ$C; ONBR >1.3. Thus, the emergency situation doesn't lead to occurrence of an accident.

Figures 9-12 illustrate outcomes of the analysis for failure of one of two primary pumps.

Signal on the 20-% reduction of the flow rate is generated in 0.4 second, reactor is shutdown automatically by emergency protection system for $\approx$1 second. So, this initial event doesn't lead to occurrence of an accident either.

Figures 13-16 illustrate outcomes of the analysis of loss of external power. Auxiliary emergency pump of the emergency cooling system (see Figure 17) provides extra 45 m^3/h after 1 minute.

Maximum coolant temperature does not exceed 74 $^\circ$C, being much lower than start of the water nucleate boiling, whereas maximum temperature of the «hot»

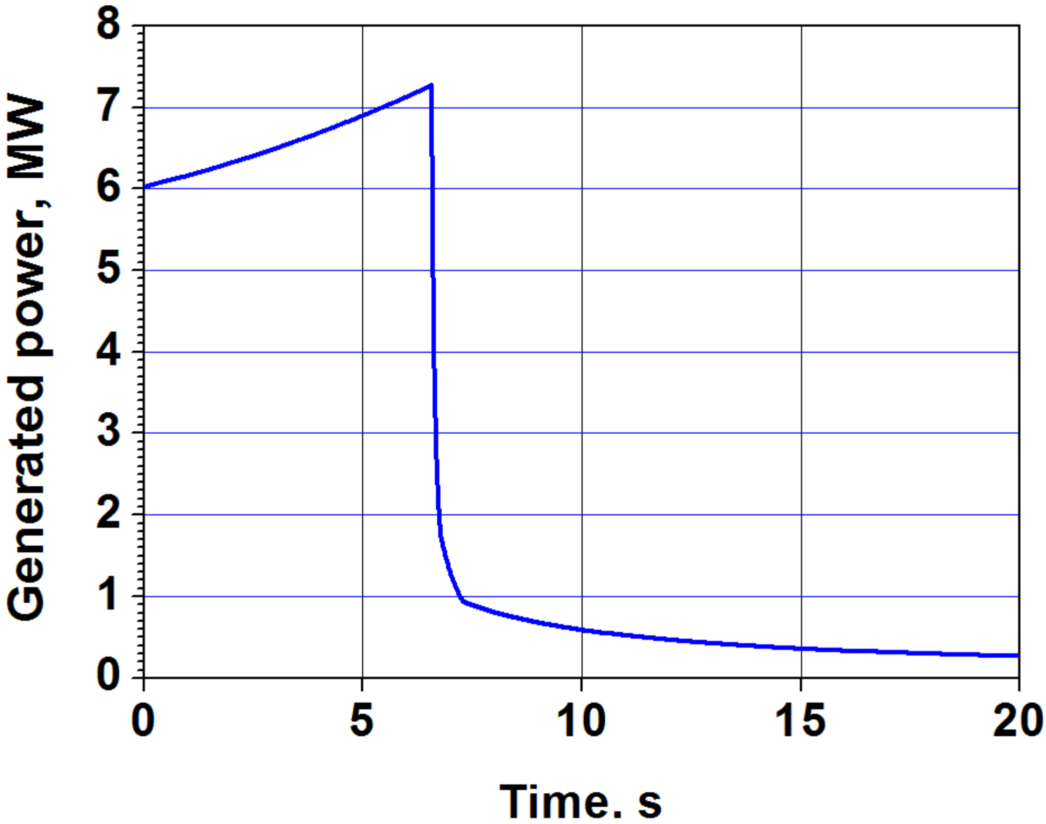

Figure 6. Spontaneous withdrawal: Power vs. time.

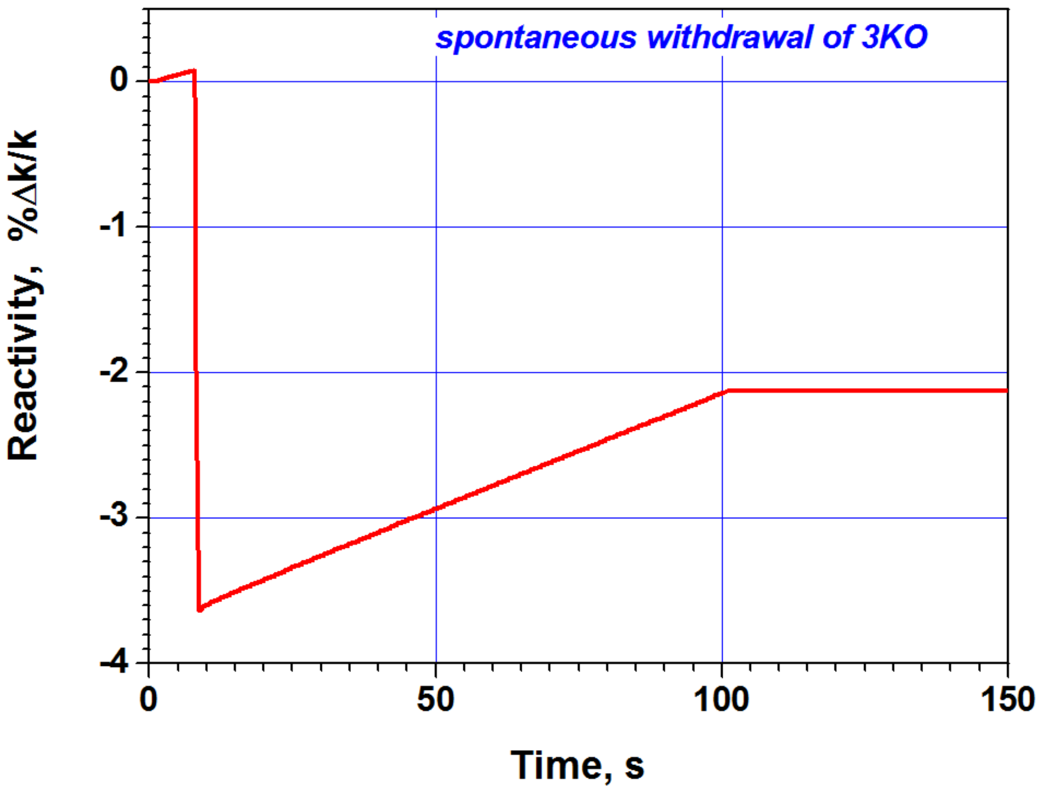

Figure 7. Spontaneous withdrawal: Reactivity vs. time.

section of the hottest fuel element doesn't exceed 97.1 ° C, being much lower than the SAV melting point. Thus, with one operating standby pump, which provides 45

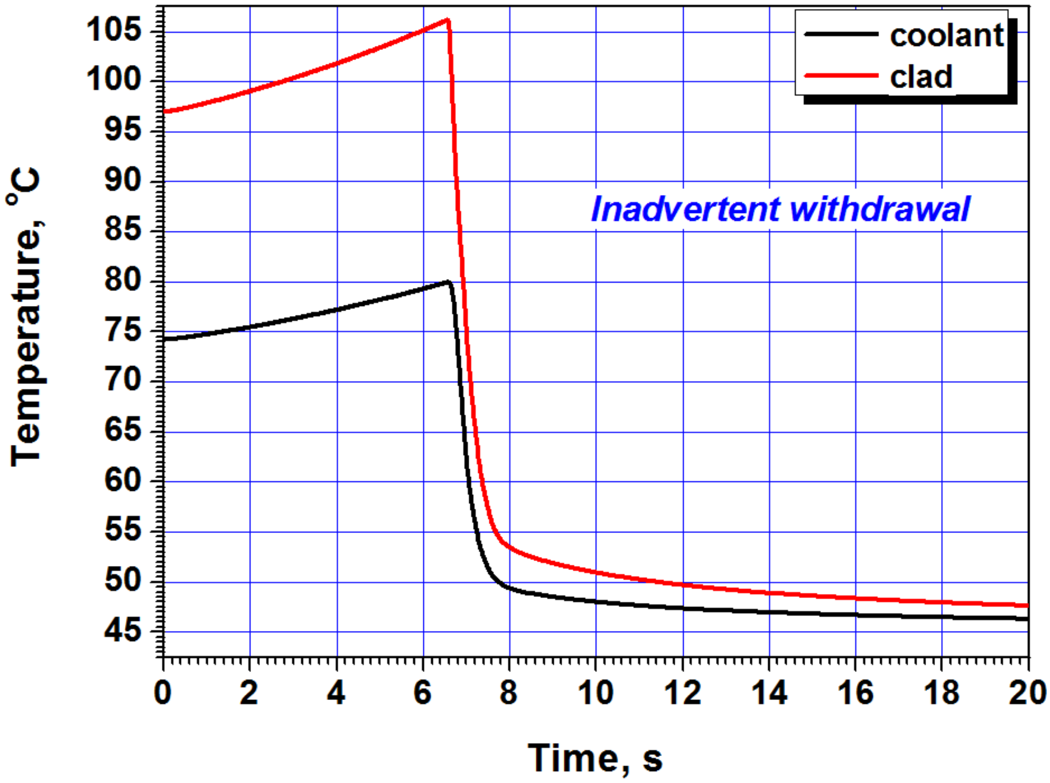

Figure 8. Spontaneous withdrawal: temperature vs. time.

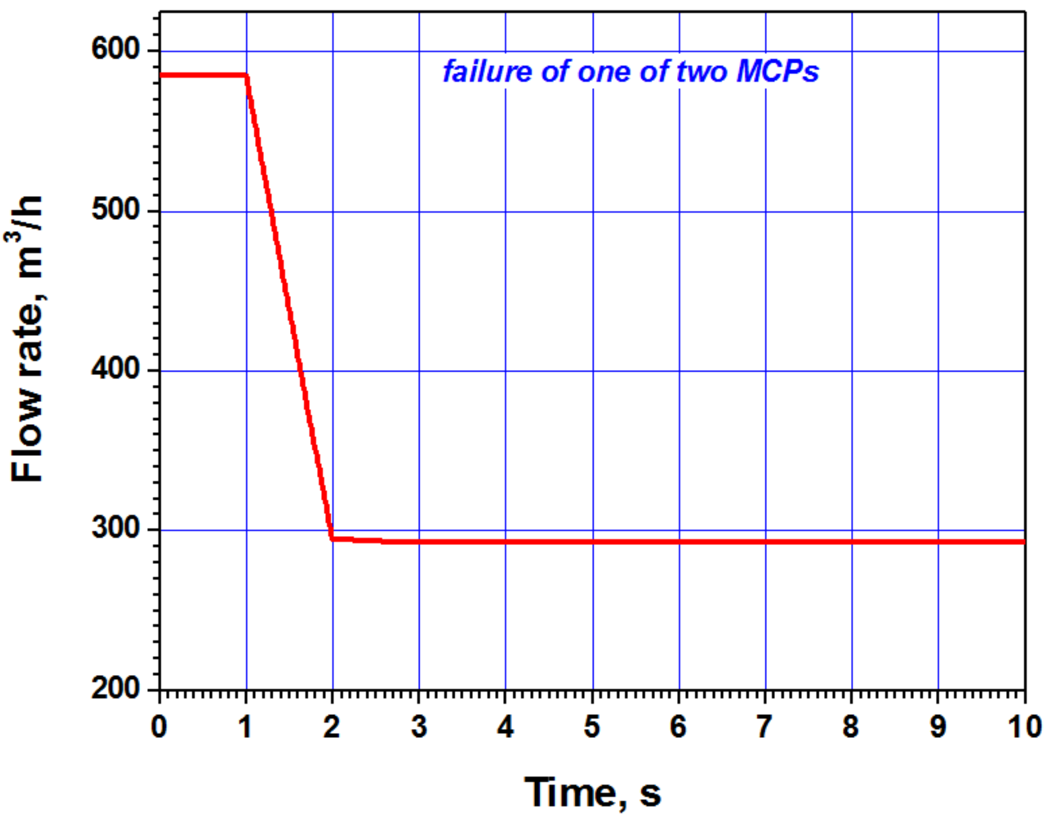

Figure 9. Variation in the coolant flow rate for 10 s.

m^3/h of the emergency cooling system, the initial event doesn't lead to occurrence of an accident.

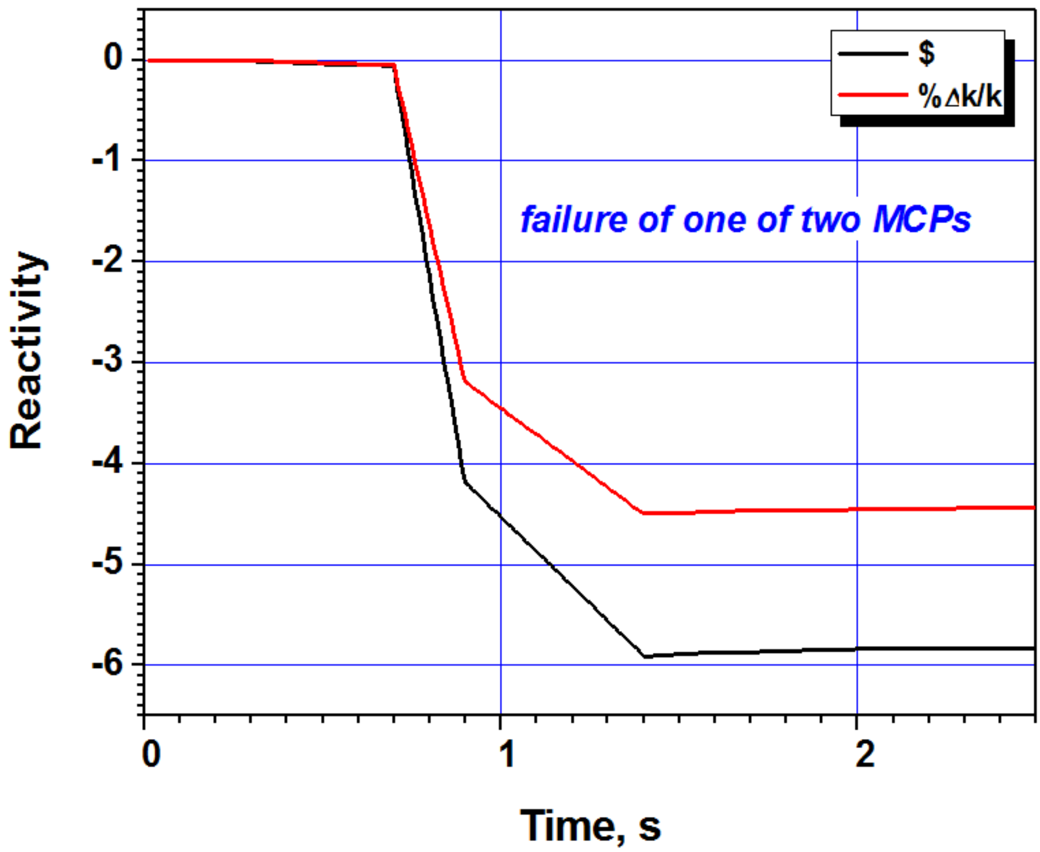

Figure 10. Variation in reactivity for 2.5 s.

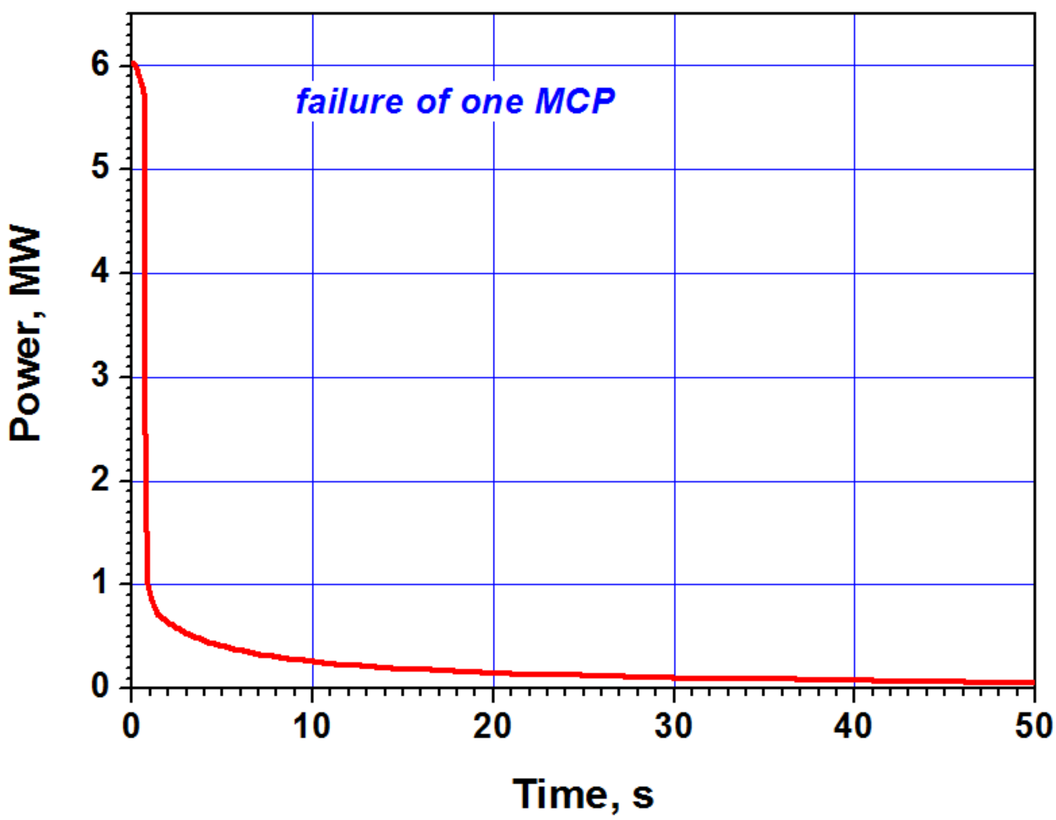

Figure 11. Variation in reactor power for 50 s.

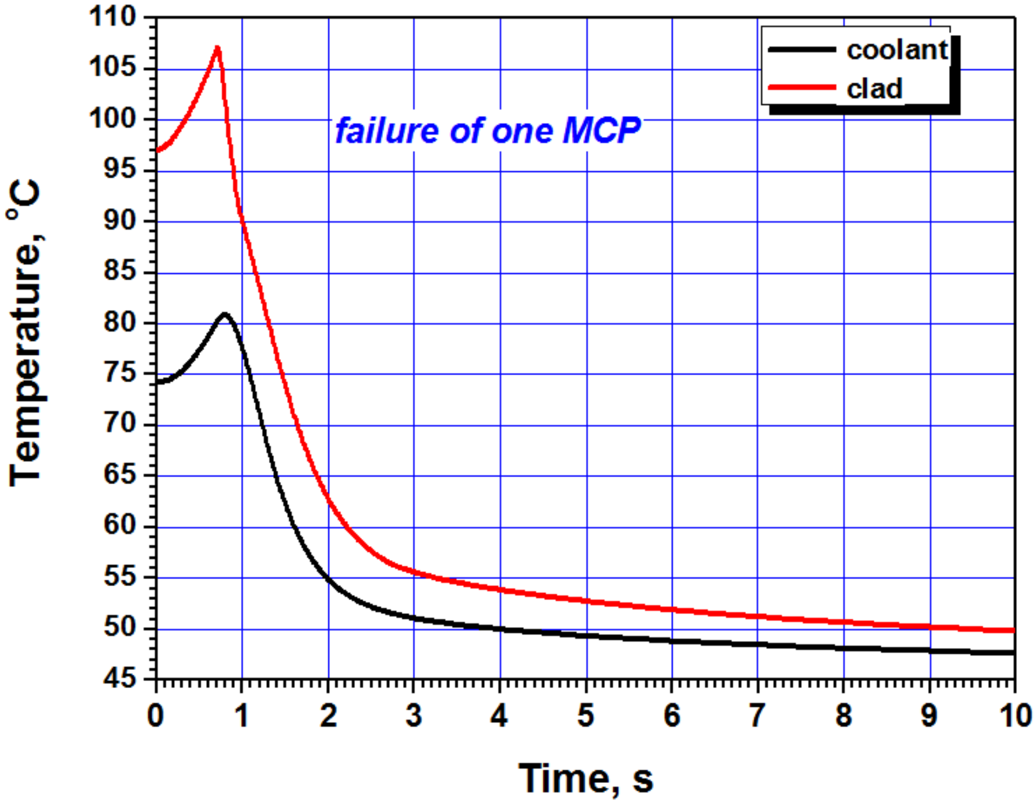

Figure 12. Variation in temperatures of coolant and clad for 10 s.

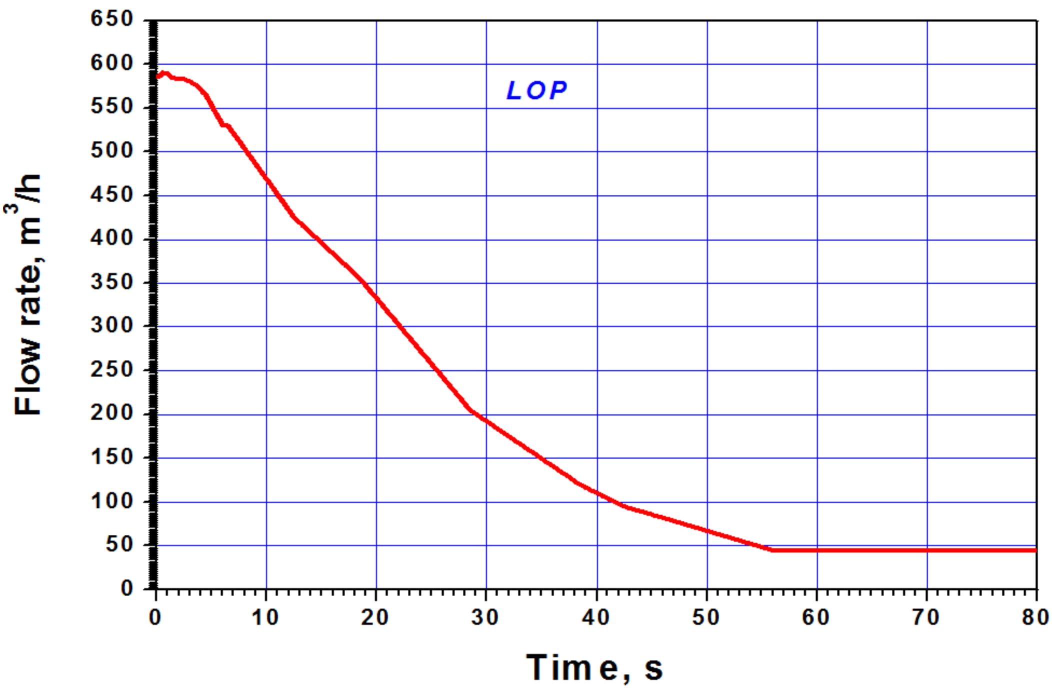

Figure 13. Flow rate coast-down.

It should be mentioned that the authors are engaged in the analysis like the presented here as applied to the WWR-K reactor conversion for many years [6-14], when various candidates for WWR-KN FA were under study, in view of proper choice for the WWR-K reactor core. ANL was a reviewer of the WWR-K RR Safety Analysis Report prepared to physical start up for the flow rate in the core 700

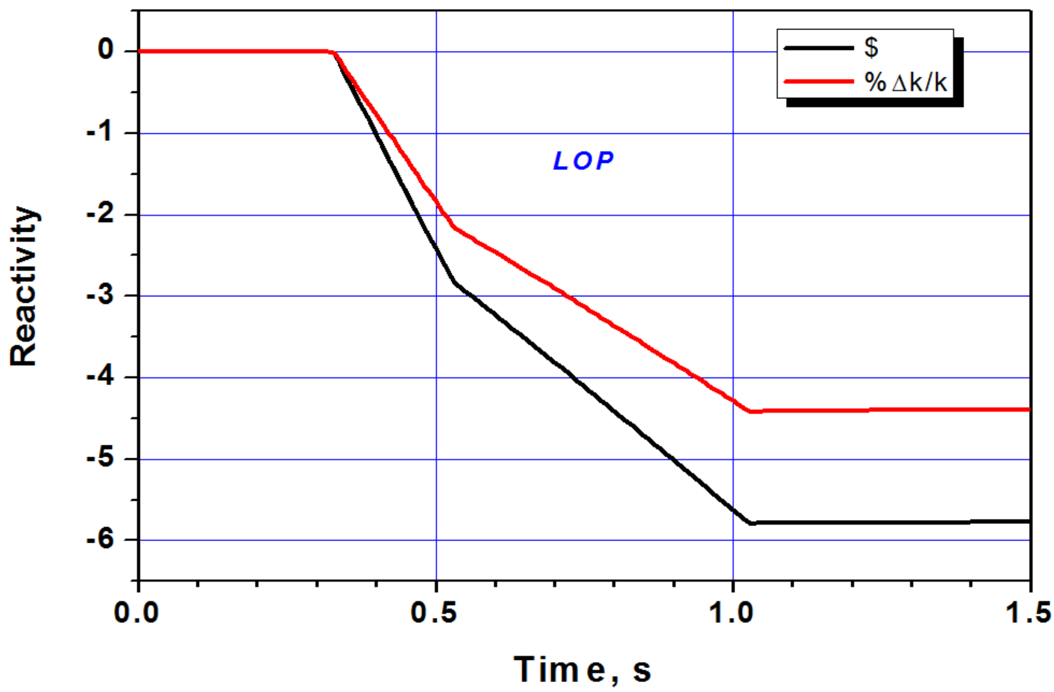

Figure 14. Reactivity vs. time for 1.5 s.

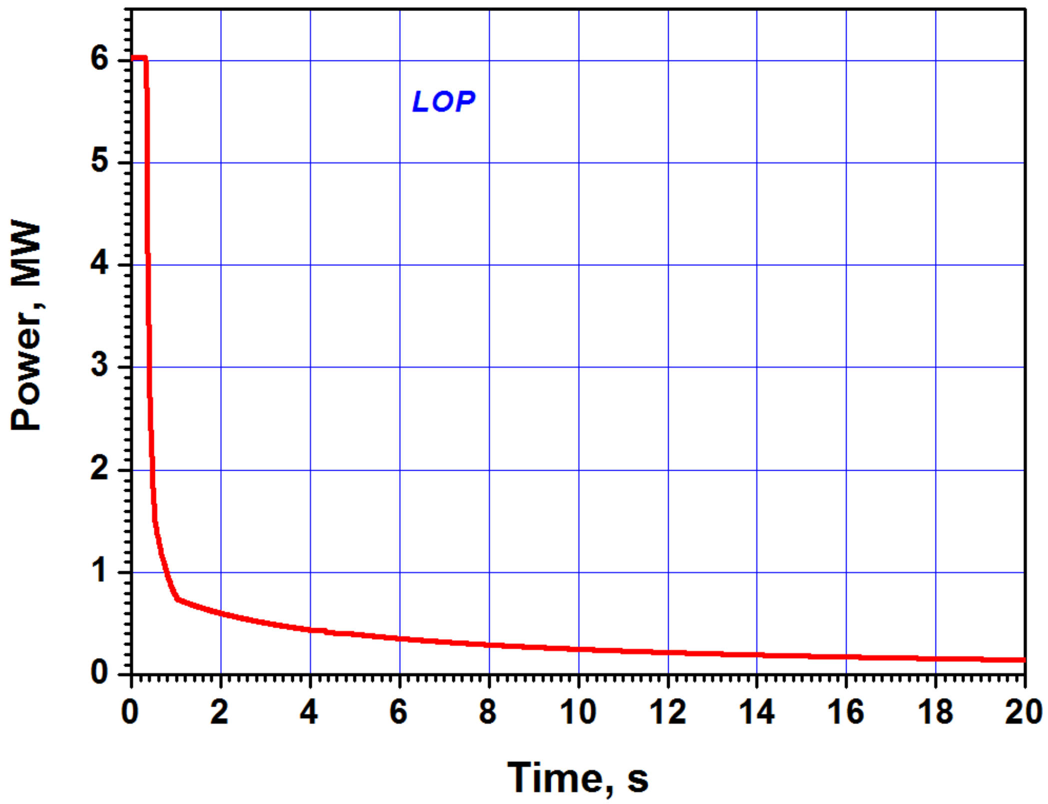

Figure 15. Power vs. time.

m^3/h; they performed some verifying thermal-hydraulic calculations with the code RELAP [7], obtaining the results very close to ours.

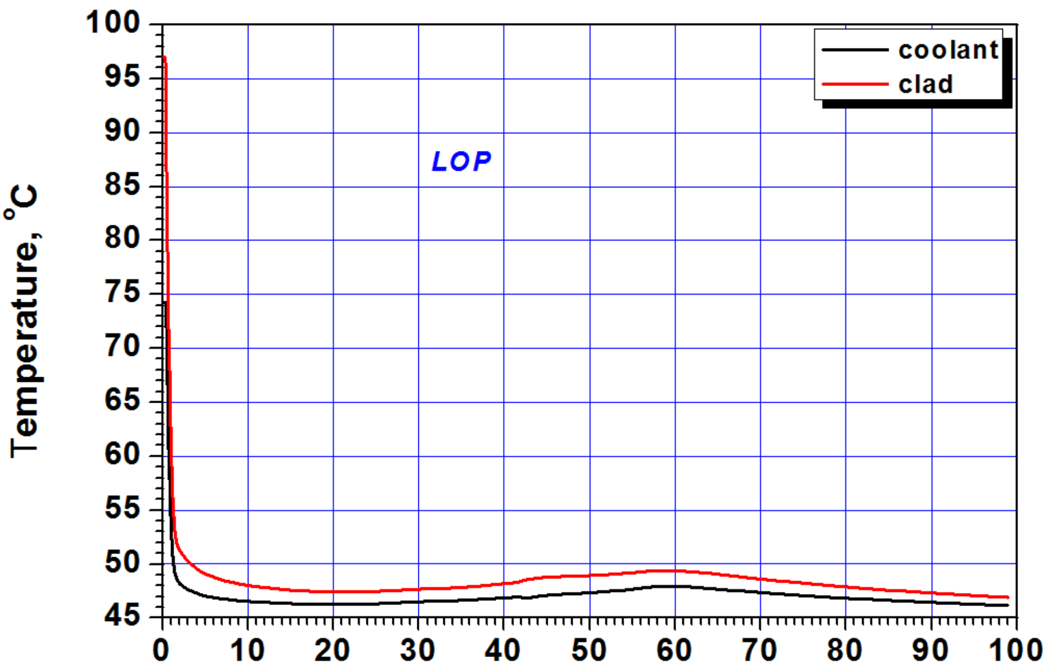

Figure 16. Temperature vs. time.

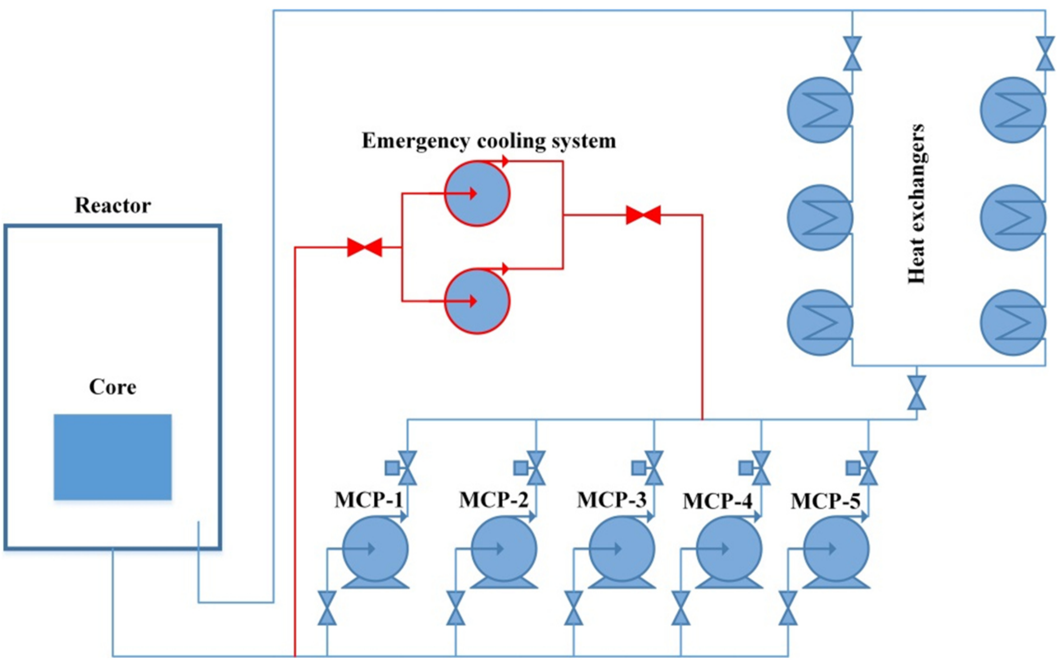

Figure 17. Primary circuit simplified schrmatic view.

Conclusion

Heat-hydraulic analysis of the steady state of the initial work load for the WWR-K RR core as well as analysis of potential accidents have proved that operation with LEU FAs will be safe at 6 MW, provided the coolant flow rate in the core comprises not less than 585 m ^{3}/h.

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