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SAFETY ANALYSIS FOR LTA IRRADIATION TEST AT THE WWR-K RESEARCH REACTOR [1]

F.M. Arinkin, L.V. Chekushina, P.V. Chakrov, Sh.Kh. Gizatulin,
S.N. Koltochnik and A.A. Shaimerdenov
The Institute of Nuclear Physics, 1 Ibragimov str., Almaty, 050032 – Kazakhstan

ABSTRACT

Final results of safety analysis performed for the forthcoming irradiation test of three LTAs in the existing core of the WWR-K research reactor are reported. Preliminary results of analysis partially were reported at previous RERTR meetings. However, only recently a final design of the fuel assembly with low-enriched uranium fuel was identified, and hydraulic test of imitators of new assemblies was performed. Due to this, both neutron-physical and thermal-hydraulic calculations were revised with renewed information taken into account. Thermal hydraulic analyses were performed with the US computer codes PLTEMP and PARET with the data obtained from the hydraulic test taken into account. As a result, safety ranges for the primary coolant flow rate and the inlet coolant temperature were chosen.

1. Introduction

This autumn we expect start of irradiation test of three lead test assemblies (LTA) in the existing core of the WWR-K reactor. LTA designer has changed slightly the LTA design –shape of stiffening ribs has changed as well as radii of corner rounding. So, safety analysis of LTA irradiation test is to be revised. In addition, recently in Russian institute VNIINM hydraulic test of LTA imitators has been performed. Test results also are taken into account in revised safety analysis. This report focuses, mainly, on thermal-hydraulic analysis of core steady-states with code PLTEMP v.3.5 and transient analyses with code PARET.

[1] The study was done under financial and technical support from the Argonne National Laboratory

2. Steady-State Analysis of the LTA Irradiation Core

2.1. LTA parameters

The lead test assembly (LTA) is composed of eight coaxial tubular fuel elements (seven have hexagonal cross-section, and one –circular one), head, shank and structural tube. A fuel element is a three-layer tube which incorporates inner and outer clad, fuel meat and end corks. Thickness of clad/meat/water gap is 0.45/0.7/2.0mm. Length of fuelled section is 600 mm. Cross-sections of fuel elements (FE) are shown in Fig.1. FE characteristics are presented in Table 1.

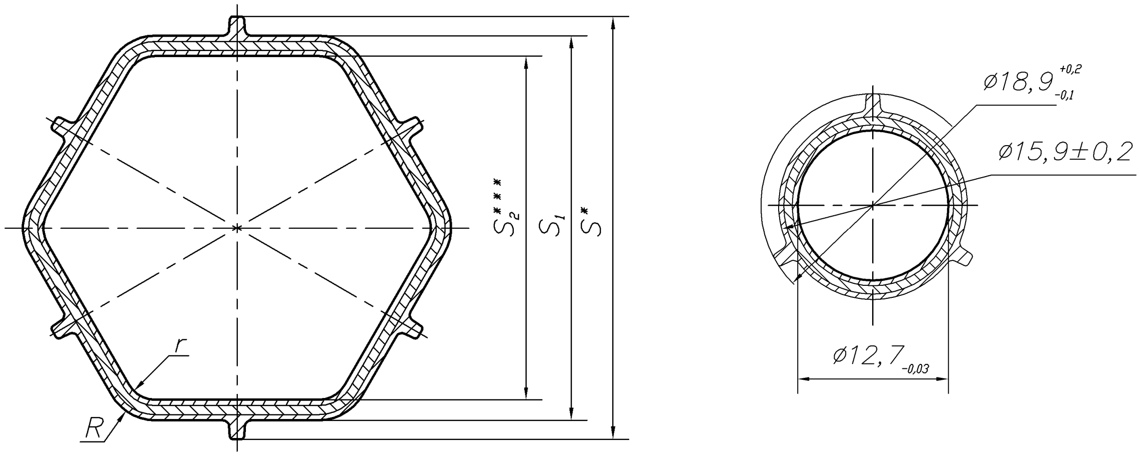

Fig. 1. Cross-sectional view of the LTA fuel elements

Table 1 – LTA parameters

FE #	Dimensions, mm						U-235 mass, g
	S*	S_1	S_2***	L	R	r	
1	$67.8^{+0.10}_{-0.20}$	66.3±0.20	$63.10_{-0.03}$	$718_{-0.5}$	6.9	5.3^{+1}	51.9±2.6
2	$62.1^{+0.10}_{-0.20}$	59.1±0.20	$55.90_{-0.03}$	$650_{-0.5}$	6.5	4.9^{+1}	46.2±2.3
3	$54.9^{+0.10}_{-0.20}$	51.9±0.20	$48.70_{-0.03}$	$650_{-0.5}$	6.1	4.5^{+1}	40.4±2.0
4	$47.7^{+0.10}_{-0.20}$	44.7±0.20	$41.50_{-0.03}$	$650_{-0.5}$	5.7	4.1^{+1}	34.6±1.7
5	$40.5^{+0.10}_{-0.20}$	37.5±0.20	$34.30_{-0.03}$	$650_{-0.5}$	5.3	3.7^{+1}	28.8±1.4
6	$33.3^{+0.10}_{-0.20}$	30.3±0.20	$27.10_{-0.03}$	$650_{-0.5}$	4.9	3.3^{+1}	23.0±1.2
7	$26.1^{+0.10}_{-0.20}$	23.1±0.20	$19.90_{-0.03}$	$650_{-0.5}$	4.5	2.9^{+1}	17.3±0.9
8	18,9±0,02	15.9±0.02	12.7±0.02	$650_{-0.5}$			10.4±0.5

The expected LTA irradiation test core will be composed of 38 regular WWR-C-type assemblies, three LTA, placed inside special irradiation device from beryllium in the core central part, and side beryllium reflector, compose of 28 blocks of beryllium (see Fig. 2 below).

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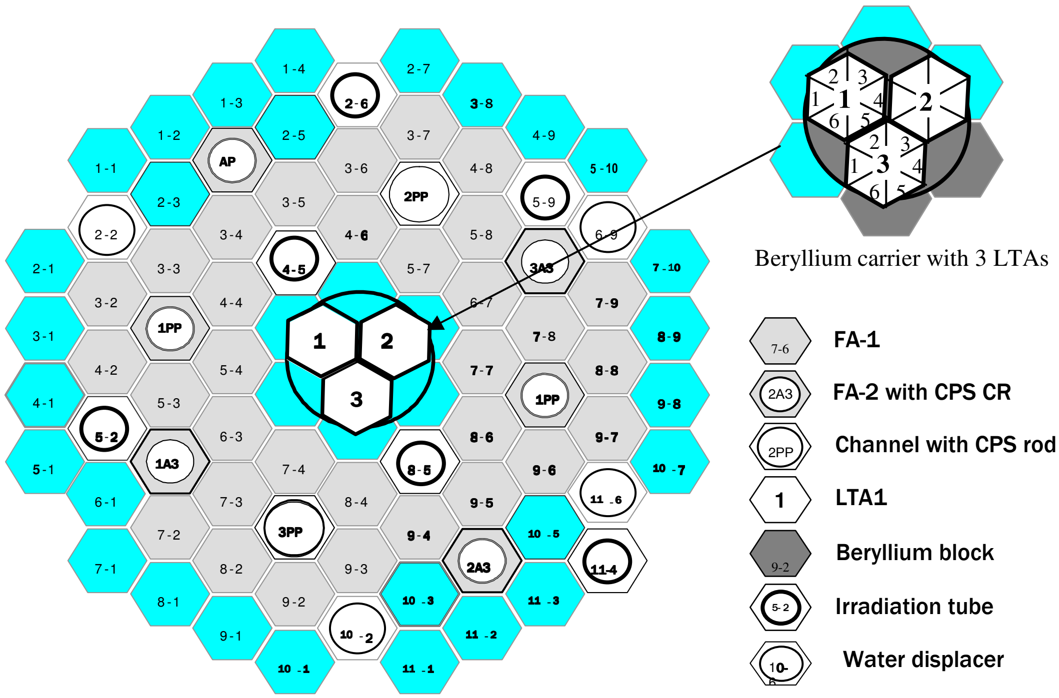

Fig. 2. WWR-K initial core configuration for LTA irradiation test

Currently, two primary-circuit circulation pumps, capable to provide the flow rate over the core up to ~690 m^3/h, are used in operation of the WWR-K reactor. The LTA power densities in the start of LTA test will be highest, and at coolant inlet temperature 45 °C (the highest value used in existing WWR-K SAR) the flow rate of two primary pumps will be insufficient to provide heat removal sufficient to assure the FE clad temperature lower than that given by fuel designer (98ºC for LTA and 95ºC for VVR-C). So, the flow rate values which correspond to operation of three pumps, capable to provide ~1000 m^3/h, are taken for coolant inlet temperature 45°C. Calculations are to show to the values of the flow rate as functions of inlet coolant temperature at which safe operational temperatures of the coolant and clad are assured as well as safe minimum values of the ONB ratio.

2.2. Hydraulic Calculations

Pressure drop and coolant flow rate in the LTA inter-fuel-element gaps 1 through 8 has been determined by experimental hydraulic characteristics which were obtained as a result of hydraulic test of the LTA imitator. The characteristics have been approximated by power functions of a form $\Delta P = C{\cdot}G^a$, where C and a are adjusted parameters. Coolant flow rate in the LTA outer gap was not modeled in hydraulic test; so it has been calculated approximately from the flow rate in the first gap under assumption of proportionality to the gap flow area ($G_0=G_1{\cdot}S_0/S_1$). Calculated distribution of the coolant velocity and the mass flow rate over gaps of LTA and regular FA for value of the pressure drop across the core 17000 Pa, which corresponds to the coolant low rate in the core 1000 m^3/h, is presented in Table 2.

Table 2 – Distribution of the coolant velocity and the mass flow rate in LTA and regular FA

	gap #	Flow area, m^2	Wet P, m	D_H m	v, m/s	G, kg/s
8-tubeLTA	1	0.0004805	0.47663	0.00403	2.24	1.07
	2	0.0004171	0.43254	0.00386	2.24	0.93
	3	0.0003669	0.38317	0.00383	2.09	0.76
	4	0.0003167	0.33381	0.00379	2.03	0.64
	5	0.0002665	0.28444	0.00375	2.09	0.55
	6	0.0002163	0.23507	0.00368	1.91	0.41
	7	0.0001661	0.18571	0.00358	1.87	0.31
	8	0.0001363	0.12544	0.00435	1.67	0.23
	9	0.0000659	0.06754	0.00390	3.23	0.21
5-tube VVR-C FA	1	0.0006942	0.45241	0.00614	2.57	1.77
	2	0.0005902	0.40010	0.00590	2.52	1.47
	3	0.0004905	0.32701	0.00600	2.54	1.24
	4	0.0003804	0.25357	0.00600	2.54	0.96
	5	0.0002759	0.17586	0.00628	2.60	0.71
	6	0.0001546	0.10304	0.00600	2.54	0.39

Hydraulic calculations of the core under assumption of equal pressure drop values for all elements of the core are summarized in Table 3. Flow rate in the core equals 1000 m^3/h. Relevant value of the pressure differential across the core is 17000 Pa.

Table 3 – Flow rate (m^3/h) through elements of the core

Core element	Qnty	Partial flow rate, m^3/h	Total flow rate, m^3/h
LTA	3	18.380	55.14
WWR-C Type-1 FA	32	21.615	691.68
WWR-C Type-2 FA	6	17.530	105.18
Block of beryllium	28	3.803	106.484
Irradiation channel	6	3.803	22.818
Water displacer	4	3.803	15.212
CPS channel	2	3.803	7.606
TOTAL:	**81**		**1004**

2.3. Thermal Calculations

The following results of neutron-physical calculations for reactor power 6 MW with MCU-REA code are used as initial data for thermal calculations:

- peak power values of the hottest LTA and regular FA: 370 and 192 kW respectively
- peak values of the power density: in LTA and FA: 1.51 and 1.31 kW/cc.

Besides, the following data are used as input:

- the core inlet coolant pressure and temperature :0.135 MPa and 45ºC respectively
- specific heat conductivity of clad/meat material in LTA: 0.17 / 0.10 kw/m-K, in regular FA: 0.17 / 0.08 kW/m-C
- thicknesses of the LTA/ FA FE meat and clad : 0.7 /0.6 and 0.45/ 0.85 mm respectively

– data of hydraulic calculations (flow areas of inter-FA gaps, hydraulic diameters, wetted and heated perimeters, coolant flow rates for FAs of two types

Thermal calculations with code PLTEMP v. 3.5 for a value of the coolant inlet temperature 45 °C and a value of the coolant flow rate in the core 1000 m^3/h have shown to maximum value of the LTA clad temperature, 88,8 °C, which is associated with outer wall of the second fuel element, and maximum value of the coolant temperature, 68.6 °C; it is in a gap between the first (outer) and second fuel elements of LTA. The minimum value of the ONB ratio is 1.6.

The lowest safe value of the coolant flow in the core (785 m^3/h) for coolant inlet temperature 45 °C has been found. With this flow rate, the LTA maximum clad temperature is 97.7 °C, the maximum coolant temperature in LTA is 75 °C, and the minimum ONBR is 1.31. Results of this calculation as pressure differential across the core versus the coolant flow rate in the core are shown in Fig. 3.

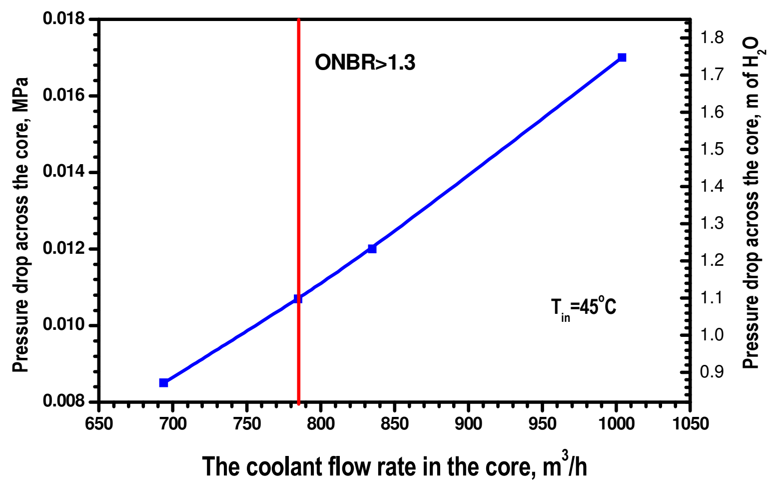

Fig. 3. Pressure drop across the core versus the coolant flow rate in the core

Region of safe values of the coolant flow rate is beyond red vertical line, which corresponds to 785 m^3/h.

Inaccuracies in determination of the reactor power and the power of the hottest LTA may increase the LTA power by ~20%. So, with the increased value of the LTA power up to 444 kW, the lowest safe value of the flow rate in the core is 956 m^3/h, provided inlet coolant temperature is 45 °C. Relevant maximum clad temperature is 97°C, maximum coolant temperature is 74 °C, the minimum ONB ratio by Bergles-Rohsenow – 1.31.

So, for the coolant inlet temperature in the core 45 °C, values of the clad temperatures in the hottest LTA and FA are lower than appropriate safe operational limits dictated by designer - 98 ºC for LTA and 95 ºC for VVR-C type FA, when three operating primary pumps provide the coolant flow rate not less than 956 m^3/h. Maximum calculated values of the coolant temperature in gaps of both hottest LTA and regular FA in all cases under consideration are lower than water boiling point.

Thus, LTA irradiation test in the WWR-K reactor core won't violate limits of the reactor safe operation provided the coolant flow rate in the core is not less than 986 m^3/h

3. Transient Analysis with PARET

3.1. Inadvertent withdrawal of the most effective CPS control rod

To determine the core thermal characteristics in case of spontaneous insertion of positive reactivity, scenario of inadvertent withdrawal of the most effective control rods (1PP) occurred as a result of CPS failure. So, reactor is at critical state. Excess reactivity is 6.1 %Δk/k. Reactor sub-criticality with inserted automate rod and all shim rods comprises 1.1 %Δk/k. An assumption is made that a couple of shim rods is inserted fully at critical state (worst case), whereas the rest shim rods are inserted by half. Inadvertent withdrawal of the most effective shim rod, 1PP, has occurred for 43 s. The rod efficiency is 2.5 %Δk/k; i.e., the rate of reactivity insertion is ~0.06 (%Δk/k)/s. The most efficient safety rod, 3AZ, having efficiency 0.91 %Δk/k, is assumed to be stuck, and is not included for scram. Following scram signal, safety rods 1AZ and 2AZ are dropped, and shim rods are inserted fully. Time needed to drop safety rods is 0.7 s. Time needed to insert shim elements is 10.5 s. Time of delay for signal to actuate emergency protection is 0.3 s. β_{eff} =0.738%; prompt neutron lifetime is $9.8 \cdot \times 10^{-5}$ s.

For the 21-d first cycle, excess reactivity varies from 6.0 to1.3% Δk/k and restores up to 4 %Δk/k for 6-day period of break, i.e., reactor poisoning by *Sm* and *Xe* comprises ~3 %Δk/k (see fig.4).

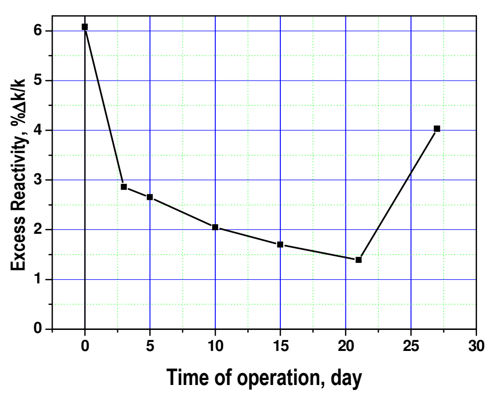

Fig. 4. Excess reactivity vs. time of reactor operation. First cycle

Reactivity feedback coefficients (%Δk/k/°C) for coolant temperature: 0.013±0.001, for coolant void: 0.194±0.003, for fuel temperature: 0.0010±0.0001; β_{eff} = 0.738 %Δk/k. Values of worthies of control rods are given in Table 5 below.

Table 4 - Worthies of control rods, %Δk/k

Σ 1PP	2PP	3PP	AP	1AZ	2AZ	3AZ	ΣPP+AP*	ΣAZ*
2.51	1.73	1.83	0.23	0.86	0.79	0.80	7.17	2.52

* All control rods are inserted to the core.

With all parameters taken into account, combined thermal-hydraulic-reactivity transient calculation has been performed with code PARET v.7.2 [4] for a 100-s time period beginning with start of 1PP withdrawal.

Model-based core consists of two parts – one includes three LTAs, another one – all WWR-C-type FAs in the core. For every part separate calculation is carried out. Each of the two parts is divided into two channels: “hot” channel, representing the FA hottest outer face (half of fuel layer, adjacent clad layer and half of adjacent coolant layer), and “average” channel, representing all FAs of the modeled part of the core with adjacent half-layers of coolant minus “hot” channel. Results of calculation are presented below.

Fig. 6 shows power variation in the core as a result of withdrawal of the “heaviest” shim rod 1PP and scram with protection rods 2AZ and 3AZ and other shim rods and automatic rod as a response to signal of power overrun by 20%. In 2.5 s reactor power increases by 23% for several seconds. Then, when 2AZ, 3AZ, 2PP and 3PP are inserted to the core, power decreases, reaching 0.55 MW. As withdrawal of the 1PP rods continues, reactor power increases to ~0.67 MW and then slowly and monotonically decreases.

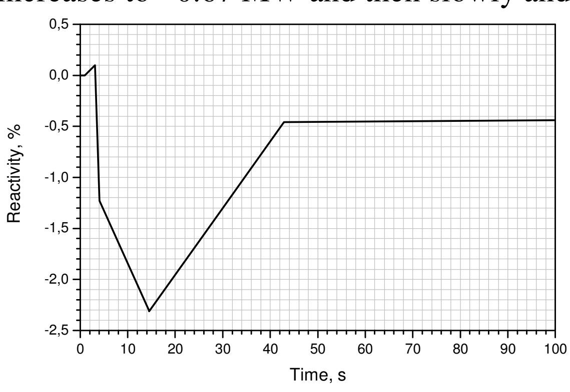

Fig.5. Reactivity versus time for 100-s period

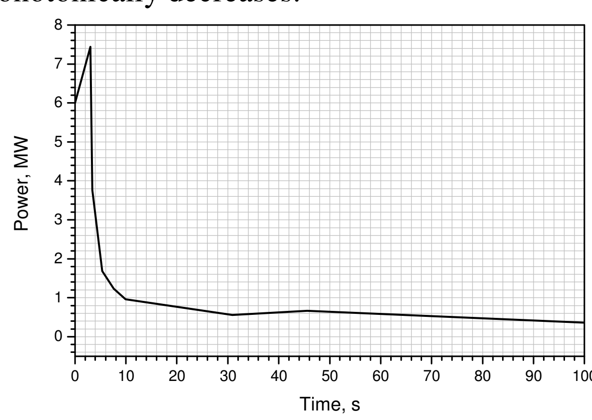

Fig.6. Power versus time

Fig.7 shows peak temperatures at outer-tube inner wall and at coolant in a gap between the first and second tubes for LTA (hottest channel) and the hottest regular FA as functions of time. Temperature maxima are reached in ~3 s after start of inadvertent withdrawal of 1PP, comprising ~99˚C for a tube in LTA and ~81˚C – in regular FA; ~78˚C for coolant in LTA and ~60 ˚C – in regular FA. The LTA clad peak temperature exceeds the relevant designer’s operational limit (98 C) by several fractions of a second (~0.5 s).

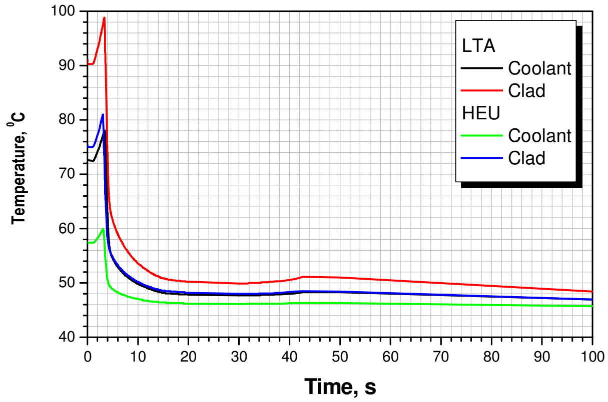

Fig.7 Peak temperatures of coolant and clad as functions of time

3.2. Failure of primary pumps

3.2.1. Failure of primary pumps as a result of loss of external electric power supply

For safety analysis of initiating cases like this one, pump “rundown” was found experimentally (residual flow rate of coolant) for a case of refusal of three primary pumps within the time range 0 to 50 second. Simultaneously, time of DG turning on was measured, found to be equal to 5 to 10 s. Also time needed to reach stationary circulation has been measured in case of turning on of the existing standby core cooling system, which incorporates two standby pumps of the ESN trade mark, providing the total coolant flow rate 20 m^3/h. The stationary mode is reached for 26 to 30 s. After scram (AZ drop for ~1 s) shim rods are inserted to the core for ~10 s. DG is turned on automatically for 5-10 s, and the standby core cooling system actuates.

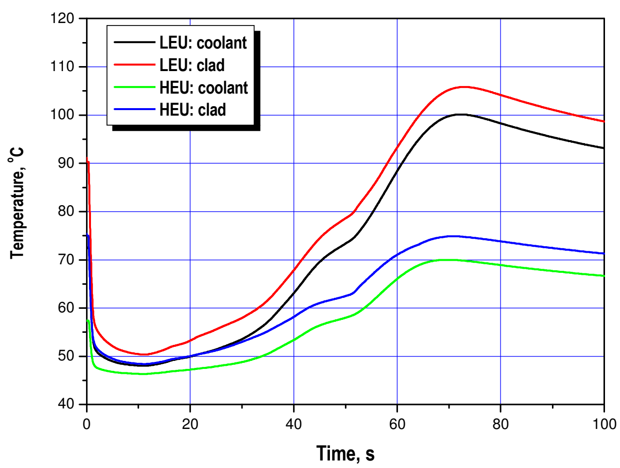

Fig. 8. Temperature of the LTA/regular FA coolant and clad versus time for a 100-s time interval

Appropriate transient thermal hydraulic calculation has been performed with the code PARET. Fig.8 shows the 100-s variation in temperatures of coolant in a gap between the first and second fuel element of LTA/regular FA and of the wall of outer fuel element and. In calculations rundown of the residual flow rate after shutdown of three primary pumps is taken into account. The calculated peak values of the clad and coolant for LTA are, respectively, 105.8 °C and 100.3 °C, and for regular FA - 75 °C and 70°C. (coolant inlet temperature is 45 C). It follows from all said above that this initiating event does not lead to accident.

3.2.2. Failure of one primary pump as a result of blade seizure

In case of failure of one primary pump, scram is actuated by emergency signal "*Reduction of coolant flow by 20%*", and operation of two primary pumps continues. Appropriate transient thermal hydraulic calculation for the coolant inlet temperature 45 °C has been performed with code PARET. Results obtained are presented in Fig.9.

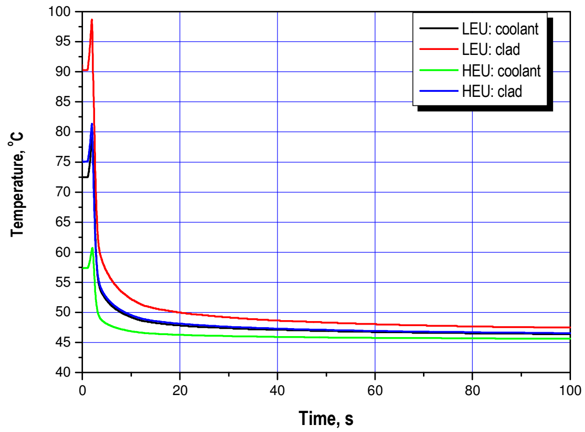

Fig.9. Temperature of the LTA/regular FA coolant and clad versus time for a 100-s time interval.

3.3 Full blockage of gaps between the LTA fuel elements

The postulated accident, related to such initiating event as blockage of coolant passage in the hottest LTA in the core is considered as BDA. It results in LTA melting and gaseous/volatile fission products (GFP) release to environment. An accident occurs in an end of the first operation cycle of the LTA irradiation test, when maximum level of the generated activity is expected.

Analyzed inventory of the activity per nuclide which is accumulated in LTA was obtained with the MCU-BUR. The calculated total activity generated in fuel to an end of the first operation cycle comprises $6.06 \cdot 10^{16}$ Bq.

By results of experimental studies, ~90% of accumulated fission products (FP) will stay in grains of uranium dioxide and ~ 10% in aluminum matrix [5]. We use also estimates of the radionuclide activity matrix-to water release fractions which are presented in Table 6 [5].

Table 6– Fractions of nuclide activity released from fuel to coolant

Xe, Kr	*I, Br*	*Cs, Rb*	*Te, Se, Sb, As*	*Sr, Ba*	*Ru, Rh, Pd, Mo, Tc*	*La, Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, Nb, Ge, Ag, Cd, In, Sn*
1	0.5	0.1	0.15	0.1	0.03	0.03

A value of the integrated activity released to reactor stack was calculated by the formula (1):

$$\boldsymbol{\mathcal{A}} = \boldsymbol{k} \times \Sigma_i\, \mathcal{A}_i^{w} \times c_i^{w} \times\{1\text{-}\exp[\lambda_i + c_i^{w} + c_i^{p}) \times t]\}/(\lambda_i + c_i^{w} + c_i^{p}), \qquad (1)$$

where $\mathcal{A}$ is the net activity released via reactor stack for time $\boldsymbol{t}$, $\boldsymbol{k}$ is fraction of the activity release from fuel meat to aluminum matrix (0.1), $\mathcal{A}_i^{w}$ is the activity of an i-th nuclide in water ($\mathcal{A}_i^{w}$ includes factors from table 6), c_i^{w} is the constant of activity retention in coolant for an i-th nuclide, λ_i is the decay constant; c_i^{p} is the purification factor for an i-th nuclide at the expense of coolant circulation through ion-exchange resins, t is the time elapsed from a moment of activity release from fuel.

Also the following assumptions, based of experimental data [5], are used:

- on retention in water:
  - for gaseous FP the constant of release from water is 10^{-6} s^{-1};
  - for iodine the constant of release from water is $3{\cdot}10^{-10}$ s^{-1};
  - for non-gaseous FP the constant of release from water is $3{\cdot}10^{-10}$ s^{-1};
- on filter efficiencies:
  - for non-gaseous FP the filtering constant (ion-exchange filters) is 10^{-5} s^{-1};
  - efficiency of particulate/iodine filtering is not taken into account.

The calculated by formula (1) activity released for ~700 days, **$1.14{\cdot}10^{15}$ Bq,** will come to coolant immediately. Due to slow diffusion in water and retention in filters, $2.55{\cdot}10^{10}$ Bq release via stack for the first minute. The integrated release for about 70 days comprises, in total, $2.78{\cdot}10^{13}$ Bq. The release rate, averaged for nearly 70 days, comprises $2.78{\cdot}10^{13}$ / $(70{\cdot}24{\cdot}60)$ $=2.78{\cdot}10^{8}$ Bq/min., which is less than the release for the first minute. So, in view of conservatism, we shall use in our analysis the release rate for the first minute. i.e., $2.55{\cdot}10^{10}$ Bq/min.

t	t, s	A, Bq	ΔA/Δt, Bq/min
1 m	60	2.55E+10	2.55E+10
2 m	120	4.60E+10	2.30E+10
10 m	6.E+02	1.72E+11	1.72E+10
1.7h	6.E+03	9.96E+11	9.96E+09
16.7 h	6.E+04	5.33E+12	5.33E+09
6.9 d	6.E+05	2.22E+13	2.22E+09
69.4 d	6.E+06	2.78E+13	2.78E+08
694.4 d	6.E+07	2.78E+13	2.78E+07

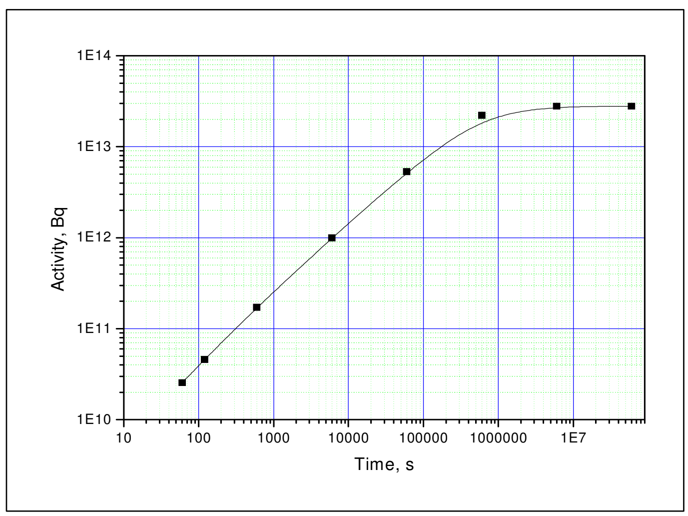

Fig.10. Temporal dependence of the integrated GFP activity which is released via stack within time range of about 70 days

Evaluation of accident consequences has been performed with the code RASCAL 2.2 [7]. It was found that calculated values of absorbed doses are much less than relevant regulatory limits.

4. Conclusion

Thermal-hydraulic analysis of the LTA test core has identified safe ranges of the coolant flow rate and temperatures. Analysis of transients has proved that the LTA test won't overrun limits of safe reactor operation.

5. References

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