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Progress in Conversion Analysis of IRT MEPhI Reactor

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ABSTRACT

IRT MEPhI reactor at the National Research Nuclear University MEPhI is 2.5 MW pool type research reactor. Feasibility studies for LEU conversion of the reactor were completed in 2011. LEU tube-type U-9%Mo fuel assembly IRT-3M was selected for the conversion analysis. Safety analysis is being performed. The report presents preliminary results of reactivity induced transients calculation. Problems of LEU core calculation validation and benchmarking are considered.

1. Introduction

NNSA (USA) and Rosatom (Russia) have agreed to study the feasibility of converting six research reactors in Russia to LEU fuel. One of these is the IRT MEPhI research reactor at the National Research Nuclear University MEPhI.
The feasibility studies were completed in November 2011. The comparison of performance of the reactor and its experiments using LEU fuel assembly (FA) with current reactor performance with HEU FA was carried out [1]. For feasibility studies of the IRT MEPhI reactor conversion tube-type FA IRT-3M with U9%Mo-Al fuel (enrichment of 19.7%) was chosen as a LEU fuel [2]. The initial step performed analysis sufficient to determine that conversion from HEU to LEU fuel is technically feasible. The next step is to perform safety analysis. The studies are being carried out in cooperation with Argonne National Laboratory (ANL) and at its funding support within the RERTR program under the contract No. 0J-30402 between ANL and MEPhI.

2. LEU core calculation benchmarking

Preparation for a conversion implies validation of LEU core calculation models. Validation of the codes for existing HEU cores calculation is based on the comparison with the reactor experimental data, with results of calculation of benchmark problems by other codes and on the comparison with calculation results for the reactor obtained by precision codes (if a diffusion code is being qualified). For LEU core with U9%Mo-Al fuel reactor experimental data are absent and the main way for codes validation is the comparison of calculation results. It should be noticed that model validation for existing HEU core is a necessary part of LEU core model

validation because the reflector and other non-fuel elements of the reactor are usually the same for HEU and LEU cores. The strategy of conversion feasibility studies within the framework of RERTR program ensures LEU core models validation based on calculations comparison: the studies are performed simultaneously and independently by reactor operators and ANL research team. Additional effort in this process is presented in this report.
The test problems for IRT-type reactor with U-Mo LEU fuel are proposed. One of them which illustrates the change of excess reactivity and other parameters of IRT type reactor core during the first burnup cycle is described further. Real geometry of fuel assembly is considered. Core configuration is simplified: experimental channels, end details of fuel assemblies, grid plate, etc are not considered. To reduce the number of materials for burnup calculation, one quarter of real core with symmetry boundary conditions is calculated.

2.1 Fuel assembly geometry

6-tube FA consists of 6 co-axial fuel tubes with control rod channel in the center. The dimensions of 6-tube FA are shown in Table 1. Outer and inner dimensions of fuel tube (S1, S2), outer and inner radii of rounded corners (R and r) are presented. The first tube is the outer tube.

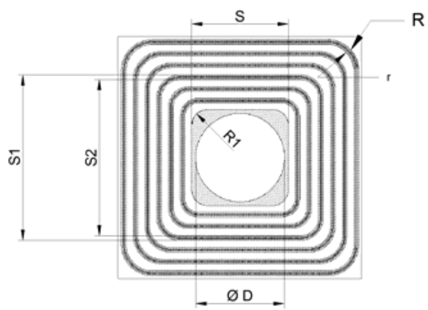


Table 1. FA dimensions, cm

Tube #	S1	S2	R	r
1	6.94	6.66	0.92	0.78
2	6.25	5.97	0.84	0.7
3	5.56	5.28	0.76	0.62
4	4.87	4.59	0.68	0.54
5	4.18	3.9	0.6	0.46
6	3.49	3.21	0.52	0.38

Figure 1 FA geometry

2.2 Core configuration

The core consists of 42 cells (6x7 positions) for FA and beryllium blocks. The core with 12 FA is considered. Lattice pitch - 7.15 cm. There is water between FA (beryllium blocks) and in the cells without FA or beryllium blocks. Core height - 58 cm. Water reflector thickness is 3x7.15 cm in X-Y direction and 29 cm in axial direction. There is vacuum boundary condition (BC) at external border. 1/4 of described system is considered in test problems: 1/2 in horizontal plane and 1/2 in axial direction. There are reflection boundary conditions at symmetry axes.

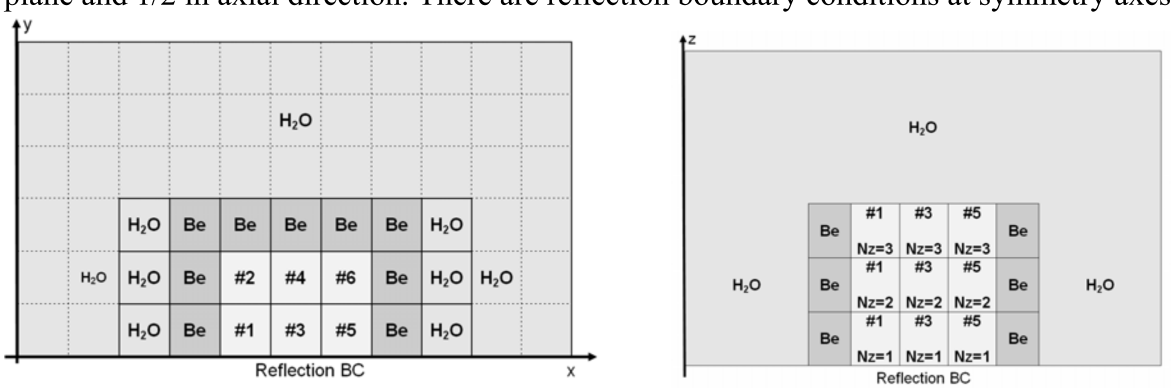


Figure 2 Core diagram (#1÷#6 – FA numbers, Nz=1÷3- axial layers numbers)

Top and bottom reflector is water. FA end details, grid plate and control rods (CR) in top and bottom reflector are not considered. Only 6-tube FA are considered in the test problem.

General description of the design parameters is as follows:

Reactor design description

Reactor type…	Pool-type IRT
Power, MW…	2.5
Number of fuel assemblies …	12-16
Grid plate, positions …	6x7
Reflector…	Be, water
Moderator, coolant …	Water
Water temperature, °C …	20
Fuel temperature, °C …	20

Fuel assembly design description

Type…	IRT-3M
Lattice pitch, cm …	7.15
Enrichment, % …	19.7
U density in fuel meat, g/cc …	5.4
Fuel meat:	
material …	U-Mo (9 wt. % Mo)
thickness, cm …	0.05
Clad:	
material …	Al
thickness, cm …	0.045
^{235}U per fuel assembly, g:	
6-tube …	355.1
Fuel meat length, cm …	58

Atom densities for FA cell, atom/ (barn cm):

Fuel meat:	
^{235}U …	2.726E-03
^{238}U …	1.097E-02
Mo …	3.352E-03
Al …	3.935E-02
Clad:	
Al …	0.0602

Control rod channel

Outer dimension S, cm …	2.82
Rounded corner radius R1, cm…	0.45
Inner channel diameter ØD, cm…	2.6
Material…	Al.

CR has no displacer (follower). There is water in the inner channel (2.6 cm diameter).

Atom densities for CR channel, atom/(barn cm):

Al ……	0.0602

Control rod

Control rod consists of absorber (boron carbide cylinder) and stainless steel tube (clad).

Absorber radius, cm …	1.06
Clad outer radius, cm …	1.15

Atom densities for CR, atom/barn cm:

Absorber:	Clad:
^{10}B …0.0146	Fe …0.0558
^{11}B …0.0644	Cr …0.0134
C …0.01975	Ni …0.0136

Reflector

Dimensions of Be block 6.9 x 6.9 x 58 cm (without facets).

Atom densities for beryllium block, atom/(barn cm):

Be …	0.01235

Water

Atom densities for pool water and water in FA channels, atom/barn cm:

H …	0.066854
O …	0.033427

2.3 Test problem specifications

The test problem illustrates the change of excess reactivity of IRT type reactor core during the first burnup cycle. At the first time step Xe-free core with fresh FA is considered. For fresh core the composition of all fuel zones is the same. Two CR are fully inserted: in FA# 3 and in symmetrical to it FA. The reactor operation at full power is calculated. Integral energy generation of the core during the first cycle is 20000 MWh (333.3 full power days). Reactor power is 2.5 MW. As one quarter of the core is considered, the power of 625 kW is intended in calculation. It is assumed that in horizontal plane all fuel tubes of one FA have the same burnup (they burn as one material). There are 3 burnup layers in axial direction. The numeration of axial segments is presented in Figure 2. CR position does not change during the burnup process. ^{10}B does not burn. Beryllium poisoning is not considered.

2.4 Results of calculation

The calculated results of three codes are presented: REBMC code (the linkage between REBUS-PC and MCNP codes) [3], MCU-PTR code [4] and MCU5-REA code. REBMC calculation was performed by ANL research team. It is performed with very good statistics and is considered as a reference variant. MCU calculations were performed at MEPhI. MCU-PTR code is the version of MCU5 package for the calculation of pool and tank types research reactors. MCU5-REA code is the version of MCU5 package for the calculation of the reactors of all types. The main difference between MCU-PTR and MCU5-REA is in cross sections for beryllium. Table 2 presents the results of neutron multiplication factor calculations.

Table 2. Neutron multiplication factor vs. burnup during the first cycle

Full power days	MCREB		MCU-PTR		MCU5-REA	
	K_{ef}	ρ, %ΔK/K	K_{ef}	ρ, %ΔK/K	K_{ef}	ρ, %ΔK/K
0*	1.08209	7.59	1.0819	7.57	1.0851	7.84
1	1.05429	5.15	1.05418	5.14	1.0578	5.46
2	1.04969	4.73	1.04936	4.70	1.0517	4.92
3	1.04854	4.63	1.04863	4.64	1.0519	4.93
33.3	1.03523	3.40	1.03519	3.40	1.0389	3.74
99.99	1.01898	1.86	1.01956	1.92		
166.7	1.00347	0.35	1.00411	0.41	1.0063	0.63
250	0.98317	-1.71	0.983704	-1.66		
333.3	0.96148	-4.01	0.96263	-3.88	0.9648	-3.65
333.3*	0.99062	-0.95			0.9939	-0.61

*) - without Xe.

The standard deviation in K_{ef} is as follows: REBMC - ±0.00008, MCU-PTR - ±0.0002, MCU5-REA - ±0.0007. MCU5-REA results have a bias ~0.3 %ΔK/K from the other codes results. The difference between MCU-PTR results and REBMC results is -0.02%ΔK/K at the first time step and 0.12%ΔK/K at the end of the cycle.
Tables 3,4,5 present concentrations of ^{235}U, ^{238}Pu, ^{239}Pu calculated by REBMC and MCU-PTR.

Table 3. ^{235}U atom densities, 1/(barn cm)

#FA	Nz	Full power days					
		33.3		166.7		333.3	
		REBMC	MCU-PTR	REBMC	MCU-PTR	REBMC	MCU-PTR
1	1	2.638E-03	2.638E-03	2.294E-03	2.296E-03	1.887E-03	1.890E-03
	2	2.652E-03	2.653E-03	2.362E-03	2.365E-03	2.009E-03	2.015E-03
	3	2.675E-03	2.676E-03	2.468E-03	2.474E-03	2.208E-03	2.220E-03
2	1	2.637E-03	2.638E-03	2.289E-03	2.292E-03	1.878E-03	1.883E-03
	2	2.652E-03	2.653E-03	2.358E-03	2.364E-03	2.002E-03	2.012E-03
	3	2.675E-03	2.676E-03	2.469E-03	2.476E-03	2.210E-03	2.225E-03
3	1	2.666E-03	2.666E-03	2.427E-03	2.429E-03	2.137E-03	2.139E-03
	2	2.676E-03	2.676E-03	2.475E-03	2.477E-03	2.225E-03	2.230E-03
	3	2.690E-03	2.690E-03	2.544E-03	2.547E-03	2.357E-03	2.364E-03
4	1	2.641E-03	2.641E-03	2.310E-03	2.311E-03	1.916E-03	1.917E-03
	2	2.655E-03	2.656E-03	2.376E-03	2.379E-03	2.036E-03	2.041E-03
	3	2.677E-03	2.678E-03	2.479E-03	2.485E-03	2.230E-03	2.241E-03

Table 4. ^{238}Pu atom densities, 1/(barn cm)

#FA	Nz	Full power days					
		33.3		166.7		333.3	
		REBMC	MCU-PTR	REBMC	MCU-PTR	REBMC	MCU-PTR
3	1	9.215E-11	1.130E-10	1.124E-08	1.286E-08	8.729E-08	9.644E-08
	2	5.351E-11	7.417E-11	6.680E-09	7.884E-09	5.325E-08	5.941E-08
	3	1.538E-11	2.878E-11	1.982E-09	2.673E-09	1.663E-08	1.966E-08

Table 5. ^{239}Pu atom densities, 1/(barn cm)

#FA	Nz	Full power days					
		33.3		166.7		333.3	
		REBMC	MCU-PTR	REBMC	MCU-PTR	REBMC	MCU-PTR
1	1	8.630E-06	7.900E-06	3.749E-05	3.783E-05	6.225E-05	6.371E-05
	2	7.227E-06	6.590E-06	3.237E-05	3.227E-05	5.577E-05	5.656E-05
	3	4.575E-06	4.138E-06	2.147E-05	2.119E-05	3.941E-05	3.942E-05
2	1	7.979E-06	7.350E-06	3.477E-05	3.517E-05	5.783E-05	5.944E-05
	2	6.677E-06	6.110E-06	2.992E-05	3.014E-05	5.170E-05	5.270E-05
	3	4.172E-06	3.779E-06	1.967E-05	1.952E-05	3.616E-05	3.631E-05
3	1	7.739E-06	7.044E-06	3.506E-05	3.479E-05	6.155E-05	6.189E-05
	2	6.474E-06	5.845E-06	2.996E-05	2.975E-05	5.417E-05	5.425E-05
	3	4.126E-06	3.715E-06	1.984E-05	1.939E-05	3.751E-05	3.707E-05
4	1	8.305E-06	7.618E-06	3.630E-05	3.663E-05	6.072E-05	6.207E-05
	2	6.954E-06	6.344E-06	3.127E-05	3.131E-05	5.428E-05	5.494E-05
	3	4.386E-06	3.972E-06	2.067E-05	2.047E-05	3.809E-05	3.823E-05

The difference in the concentrations at the end of the cycle is: ^{235}U – 0÷ 0.7%, ^{239}Pu – -1 ÷+2.8%, ^{238}Pu – +8 ÷+18%.
The presented results demonstrate a good agreement between the calculations by the chosen codes.

3. Safety analysis

Conversion from one fuel type to another requires a re-evaluation of the safety analysis. Changes to the material properties, reactivity worth of CR, shutdown margin and power density need to be taken into account. Postulated initiating events which trigger accident scenarios are as follows:

1. Uncontrolled withdrawal of control rods from critical state at nominal power level.
2. Uncontrolled withdrawal of control rods during reactor startup.
3. Jamming of automatic control rod during shim rods withdrawing.
4. Core loading accident (FA loading to critical reactor with maximum allowed rate).
5. Reactivity insertion due to ejection/loading of a movable experimental facility.
6. Loss of the offsite electricity supply.
7. One of two primary pumps shutdown.
8. Blockage of the cross section of the primary circuit pipe.
9. Rupture of the primary coolant boundary leading to a loss of flow (small leakage of the primary circuit pipe).
10. Rupture of the primary coolant boundary leading to a loss of flow (big leakage of the primary circuit pipe).
11. Split of pool covering, thermal column covering.
12. Failure of beam tubes.

Two reactivity insertion accidents (RIA) are considered further (#1 and #4). Design accidents are analyzed (accidents with realization of safety systems function). Transient calculations were performed using PARET 7.5 code [5]. In this analysis, the hottest fuel tube was modeled, along with the rest of the core modeled as the average channel; i.e. two channels were used in the PARET model. Calculations were performed for the operational HEU core and proposed operational LEU core (both cores of 16 fuel assemblies, core average burnup is ~25%).

3.1 Reactivity insertion accidents

For RIA #1 (uncontrolled withdrawal of control rods from critical state at nominal power level) the boundary conditions are as follows: the initial reactor power is 2.5 MW; over-power trip point is 3 MW; two coolant primary circuit pumps operate in initial state and continue to operate following scram (core pressure drop is $9\cdot10^3$ Pa, inlet coolant mass velocity is 1500 kg/(m^{2}s)); coolant flow direction is up-down; coolant inlet temperature is 45°C .
For RIA #4 (core loading accident) the boundary conditions are as follows: the initial reactor power is 5 kW; over-power trip point is 250 kW; period trip point is 10 s; forced coolant circulation does not work; natural convection is assumed; coolant inlet temperature is 45°C.
Delay time between crossing of overpower trip point and start of control rod motion for scram is 0.2 s. Delay time for the period trip is 3 s for IRT MEPhI C&I system.
The AZ safety control rods travel from fully withdrawn to fully inserted in 0.8 s; the motion of the rods is assumed to insert the total available negative reactivity linearly in time. Shutdown reactivity worth is -13.4$ for HEU core and -12.1$ for LEU core.
In course of reactor operation the rates of shim rods (KC-1,2,3) motion are adjusted in such way that positive reactivity insertion rate is less than or equal 0.07$/s. It is obtained by setting the individual rate for KC-1,2,3 in mm/s. The rate in mm/s is defined according with linear part of S-curves. That is, if control rods worth significantly changes their motion rate in mm/s would be corrected. So positive reactivity insertion rate in $/s is approximately the same for all CR and in transient calculations the difference in the worth of KC rods for different cores is ignored. Positive reactivity insertion rate is assumed to be 0.07$/s for KC rods (for all HEU and LEU cores). The value 0.07$/s is a maximum permissible value of positive reactivity insertion rate according to nuclear safety regulations. It should be noticed that the worth of CR groups is large (KC-3 ~7$, AR~2$) and scram will operate earlier their full withdrawing. And it is not useful to take into account the difference in CR worth of ~20% for different cores.
So, in transient calculations for HEU and LEU cores the difference will be in material properties, fuel meat and clad thicknesses, kinetic parameters and shutdown reactivity worth (AZ worth).

3.1.1 Uncontrolled withdrawal of control rods from critical state at nominal power level

The transient is assumed to be initiated by the upwards movement of shim rods from their critical position to the fully withdrawn position. Possible causes of the accident are: the failure in KC servo-motor blocks, failure of “MIRAZH” block of automatic control section of C&I system.

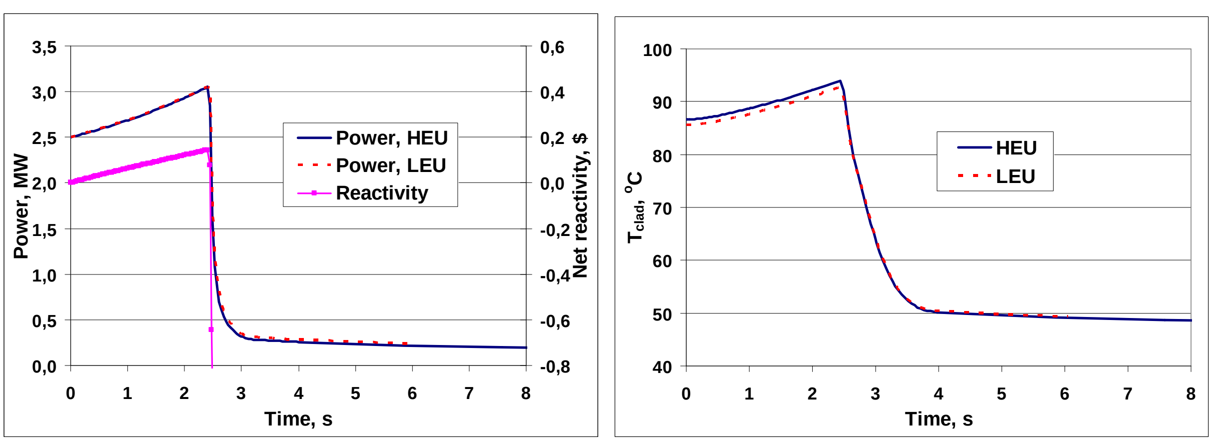


Figure 3 PARET results for shim rods withdrawal (power, max. clad temperature)

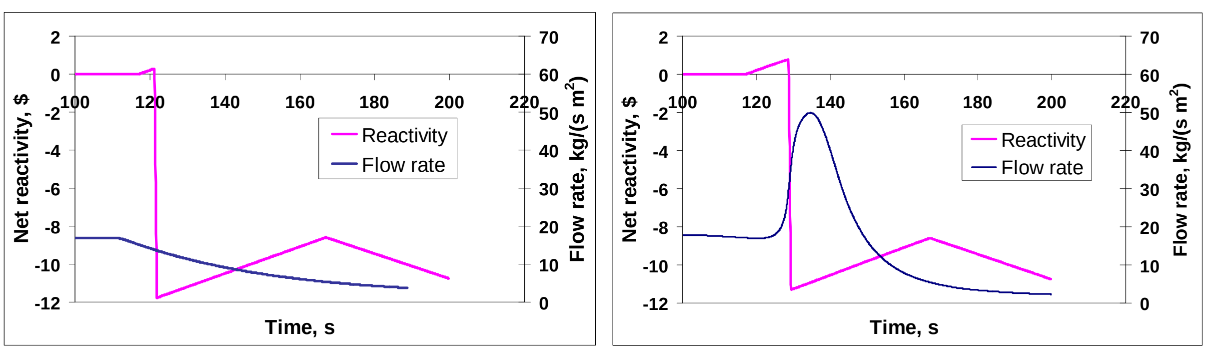
Figure 3 presents calculated results for HEU and LEU cores. The gradual reactivity insertion results in a gradual increase in power from 2.5 to 3 MW, which is the scram limit, over 2.4 s for HEU and LEU core. The reactor period becomes lower than 10 s at 1.89 s for HEU core and at 1.64 s for LEU core. Due to the delay time for the period trip (3 s) overpower scram occurs at first. The peak power is 3.05 MW. The peak cladding surface temperature is 93.8°C for HEU and LEU cores; this value is ~8°C higher than the steady-state values; the peak value is well below the limit at which fuel damage might occur.

3.1.2 Core loading accident

Adding of fresh FA during the core loading is done with scram rods fully withdrawn and AR, KC rods fully inserted. Before fresh FA adding excess reactivity of the core is estimated (using AR, KC S-curves) by reactor transition to critical state with power level of 5 kW. After that AR, KC rods are fully inserted. Accident can occur as a result of following staff errors: 1) after one of the excess reactivity estimations the reactor was not transited to subcritical state (AR, KC rods were not fully inserted); 2) chief of refueling works did not check the reactor transition to subcritical state and gave instruction to add fresh FA. According to this instruction operator began to add fresh FA so that positive reactivity insertion rate is less than 0.07 $/s.

The transient boundary conditions are imposed starting at 117 s. Over the period of ~100 s natural convection in PARET calculation completes its adjustment to equilibrium condition with coolant mass velocity of 17 kg/(m^{2}s) at the power of 5 kW. The reactivity addition is taken to be 3.5$ over 50 s (0.07$/s). The gradual reactivity insertion results in a gradual increase in power and in increase of natural convection intensity (Figure 4).

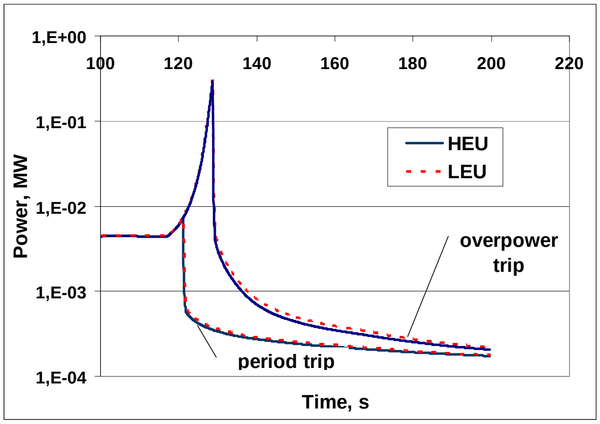
The calculated results for HEU and LEU cores are shown in Figure 5. The reactor period becomes lower than 10 s in 1 s after reactivity insertion start (for HEU and LEU cores). The period trip occurs at 121 s (after 4 s after reactivity insertion start). The peak power is 7.3 kW and 7.4 kW for HEU and LEU cores accordingly. The peak cladding surface temperature is 49°C for HEU and LEU cores; this value is not higher than the initial values.



a) b)

Figure 4 LEU core PARET results for core loading accident (net reactivity and coolant mass flow rate). a) Period scram; b) Overpower scram.

The calculated transient confirms the importance of period scram in some cases. Figures 4,5 also present the results of calculation of the transient without period scram. In this case overpower scram occurs at 129 s (after 12 s after reactivity insertion start). The peak power is 292 kW and 304 kW for HEU and LEU cores accordingly. The peak cladding surface temperature is 63.3°C and 62.9°C for HEU and LEU cores accordingly; these values are 14°C higher than their initial values.



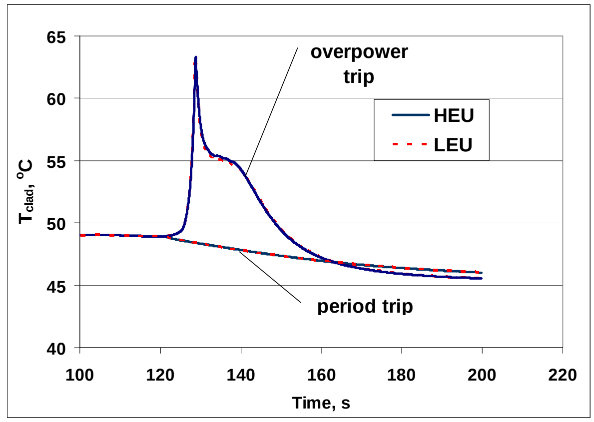


Figure 5 HEU and LEU core PARET results for core loading accident (power, clad temperature)

4. Conclusions

Preliminary results of safety analysis for HEU and LEU cores of IRT MEPhI reactor show that when designed accidents are considered and deviation of operational parameters from the nominal values is not large the difference in the results of transient calculations for HEU and LEU cores is insignificant. Future work will include the analysis of other RIA, loss of forced flow and loss of coolant designed accidents and the analysis of beyond designed accidents.

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