Fundamental Approaches to High-power Fast Reactor Core Development

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Abstract

The article examines approaches to developing a core for a high-power lead-cooled fast reactor in compliance with the formulated safety requirements and technical and economic specifications. A particular focus is made on achieving inherent safety properties, such as: a negative void effect, negative reactor power and coolant temperature coefficients, uniform coolant heating in the reactor core, as well as minimal operating reactivity margin and overshoot of reactivity during the core lifetime, which eliminates the potential risk of prompt neutron power excursion.

Constraints of reactor core characteristics in terms of acceptable power level, heating temperatures and lead coolant flow rate are considered in relation to the development level of available structural materials and lead coolant technology.

A comparative analysis of the impact of various factors (fuel rod geometry, spacing of triangular fuel rod lattice, fuel density) on characteristics of BR 1200 lead-cooled reactor core with a thermal power of 2,800 MW (void effect, reactivity margin during core lifetime, critical core loading) is conducted and trends of these characteristics are plotted depending on the fuel weight fraction.

Solutions aimed at reducing the irregularity of coolant heating and radial power peaking factor in a high-power reactor are evaluated.

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Key words: fast reactor, core, lead coolant, nuclear safety

1. Introduction

The «Proryv» project implies developing a comprehensive nuclear technology for a future innovative large scale nuclear power system based on fast reactors and a closed nuclear fuel cycle. A next generation large scale nuclear power industry like this should be founded on developing reactors with higher safety features and better economic performance than current mature nuclear or alternative energy technologies.

As part of project «Proryv», a 1200 MW(e) BR-1200 lead-cooled fast reactor is being developed. The BR-1200 reactor core allows multirecycling of nuclear fuel with the breeding ratio approximately equaling 1 and a small reactivity margin while needing only depleted uranium feedstock for refueling after reprocessing. The fuel composition therefore is defined by the requirement to facilitate a breeding ratio ~1 excluding a breeding blanket and the corresponding proliferation-resistant fuel reprocessing technology with no Pu extraction. The fuel composition is not a parameter that can be significantly affected by alternating the reactor core configuration [1].

Special emphasis must be placed on achieving inherent safety characteristics, such as a negative void effect, negative reactor power and coolant temperature coefficients, uniform coolant heating, as well as minimal operating reactivity margin and overshoot of reactivity during the core lifetime, which eliminates the potential risk of prompt neutron power excursion.

In order to fulfil the aforementioned requirements high-density nuclear fuel with high thermal conductivity characteristics is needed. In this way, mixed (uranium-plutonium-minor actinide) nitride fuel is being developed for the BR-1200 reactor.

2. Primary objectives for developing the BR-1200 reactor core concept

In order to enhance reactor core safety and performance characteristics the following objectives must be met, in accordance with the BR-1200 reactor core design requirements, defined in the requirements specifications document:

- reactivity change between reactor reloading ≈ β_{eff};
- average nuclear fuel burnup (12 % h.a.);
- achieving minimum fuel loading;
- minimization of the void effect.

Usually high-power fast lead-cooled reactors with uranium-plutonium fuel have a positive void reactivity effect. Taking into account the possibility that lead density in the reactor core might be altered due to gas or steam exposure, minimizing the void effect remains pertinent and requires implementing design solutions that affect the reactor core configuration [2].

As a result, a series of calculations for the BR-1200 lead-cooled fast reactor using uranium-plutonium nitride fuel were performed in relation to the neutronic and thermohydraulic characteristics of the core made up of hexagonal fuel assemblies with the fuel elements having identical fuel rod length and arranged in a triangular lattice.

According to the specified requirements, the reactor core's thermal power should equal to 2800 MW, average coolant heating – 120 °C. The reactor core coolant velocity is restricted to a maximum value of 2.5 m/s. Fuel element geometry (fuel pellet diameter, cladding thickness), triangular lattice pitch and fuel density were variable parameters. Different reactor core configurations were studied with various methods of forming power flux – including modifying the height of the fuel pellet column.

3. Reactor core neutronic characteristics research findings

In the result of the aforementioned calculations principal neutronic characteristics were obtained, dependencies (or trends) of these characteristics from the fuel mass fraction in the core were established. Fuel mass fraction in the reactor core can be defined as $\varepsilon_m = \varepsilon_v \cdot \rho_\kappa/\rho_T$, where ε_v is the fuel volume fraction in the core, ρ_κ is the fuel density for a particular variant, ρ_T is a fixed density value (11.8 g/cm^3).

The results are shown as dependencies of reactor core characteristics from fuel mass fraction in Figures 1-5. Marker definitions for different variants and the variable parameters used in these variants are given in Table I. Figure one shows the relationship between reactor critical load and mass fraction. Figure 1 illustrates that critical loading reaches a shallow minimum in the 0.3-0.4 mass fraction range. Figure 2 illustrates the relationship between reactor core diameter and fuel volume fraction. Reactor core height dependency from fuel mass fraction is also shown (Figure 3).

TABLE I – Maker description for each variant in Fig. 1-5

Marker	Reactor core variable parameters
◆	Change in triangular lattice pitch from 11 mm to 15.4 mm. Forming power flux along the radius of the reactor core by the two diameters of fuel element in the two zones of the reactor. Fuel density 11.8 g/cm^3.
■	Change in fuel density from 12.2 g/cm^3 to 13.0 g/cm^3. Forming power flux along the radius of the reactor core by the two diameters of fuel element in the two zones of the reactor
●	Change in triangular lattice pitch from 13 mm to 16.4 mm without forming power flux along the radius of the core. Fuel density 12.25 g/cm^3.
●	Fuel element diameter 6.9 mm. Change in triangular lattice pitch from 7.9 mm to 10.6 mm forming power flux along the radius of the core. Fuel density 11.8 g/cm^3.
○	Test variant

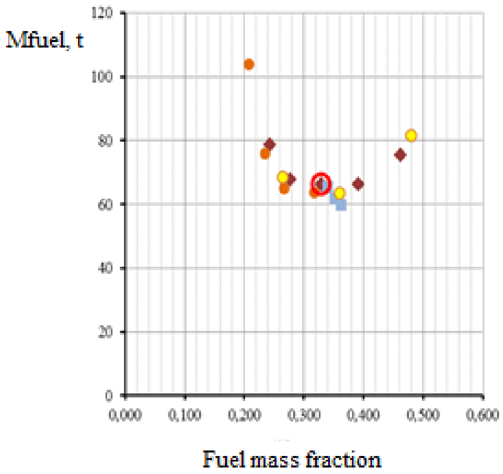

Fig.1 – Critical load dependency from fuel mass fraction

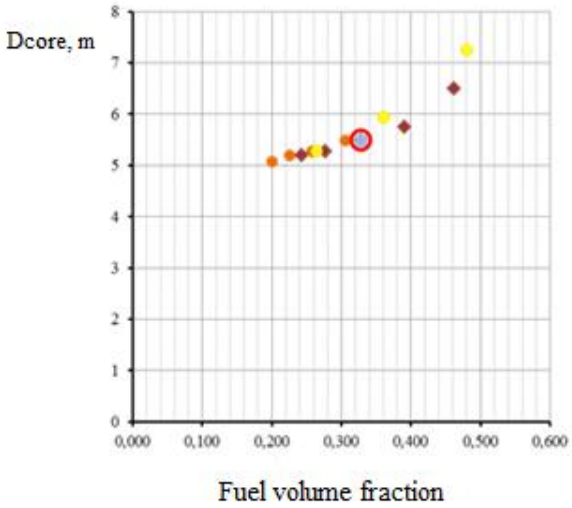

Fig.2 – Core diameter dependency from fuel volume fraction

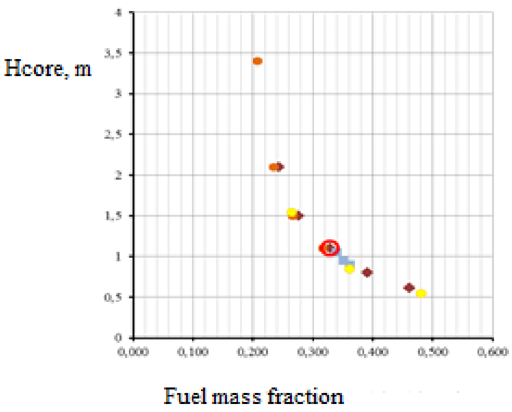

Fig.3 – Core height dependency from fuel mass fraction

Void effect dependency (in this case density-wise – lowering lead density by 5%) from fuel mass fraction is given in Figure 4. Void effect dependency from core height is also noteworthy (Figure 4), although the relationship between core height and fuel fraction is understandable and this is just another graphical representation of void effect dependency from interrelated parameters. Figure 5 illustrates that height decrease of the reactor core results in a zero or negative void effect.

By analyzing the graphical representation of the calculation results one is able to conclude that the characteristics of the core, made up of hexagonal fuel assemblies with fuel elements arranged in a triangular lattice with fuel rods identical in height, behave as composite functions of fuel mass fraction. Also noteworthy, within the accepted margin of error in this analysis, there is no difference what combinations of variable parameters resulted in specific mass fraction values.

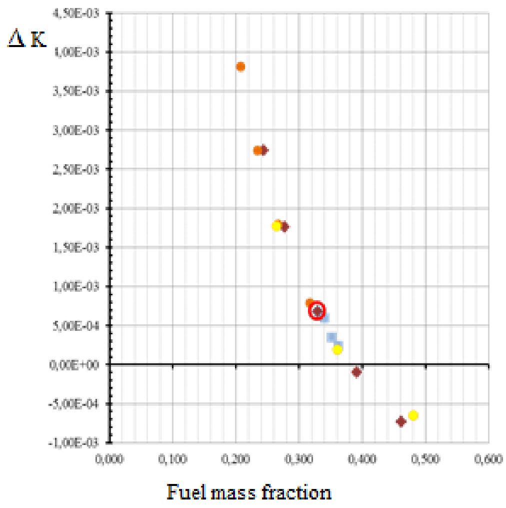

Fig.4 – Void effect dependency (in this case density-wise – lowering lead density by 5%) from fuel mass fraction

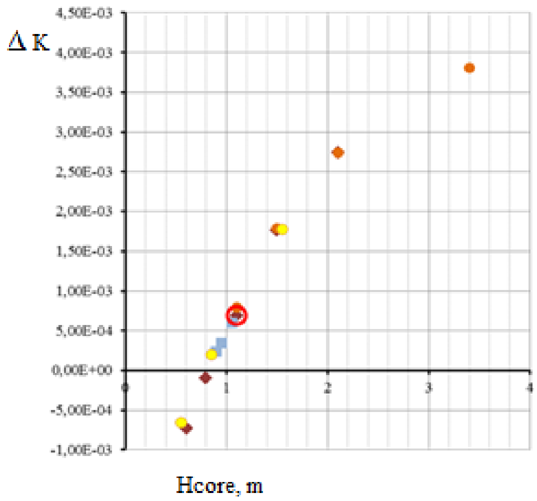

Fig.5 – Void effect dependency (in this case density-wise – lowering lead density by 5%) from height core

A comparative analysis concluded that the following is true for feasible fuel mass fractions ($\mathcal{E}_M$) in the 0.2÷0.5 range:

– critical fuel loading where $\mathcal{E}_M$ = 0.3÷0.4 reaches a shallow minimum ≈ 62÷64 t and rapidly increases for other $\mathcal{E}_M$ values;

– reactor core height decreases from 3.5 m to 0.5 m when $\mathcal{E}_M$ increases from 0.2 to 0.5, whereas the diameter increases from 5.2 m to 7.2 m. For minimal fuel loadings the diameter was within the 5.4÷6.1 m range and the height equaled to 1.25÷0.75 m.

The void effect reaches zero value and becomes negative when mass fraction exceeds 0.38 and when core height value is lower than 80 cm.

Further calculations regarding configuration, composition and geometry of the reactor core were compared with the aforementioned trends. This allowed assessing advantages or disadvantages of altering configurations and evaluating the next direction of the research.

As a result of performing these calculations a conclusion could be made that the required characteristics, including a negative void effect and a good levelling of energy release along the radius of the reactor core, could be obtained by increasing fuel column height from the central area to the peripheral area (with the minimum fuel column height in the central area). This kind of forming power flux allows the use fuel elements of the same diameter, with the only difference being fuel column height i.e. the quantity of fuel pellets in the fuel elements in different subzones.

4. Main results of neutronic calculations for the BR-1200 reactor core

As a result the BR-1200 reactor core was configured with thermal power equaling N=2800 MW, comprising 547 hexagonal fuel assemblies of various purpose, including fuel assemblies with reactor control and protection systems (CPS). For the purpose of leveling the neutron flux and power radial distribution the reactor core was divided into three subzones – central, intermediate and peripheral. The central subzone contains 72 fuel assemblies, 1 fuel assembly with an emergency protection system (EPS), 6 fuel assemblies with a reactivity compensation system (RCS) and 6 fuel assemblies with an automatic control system (ACS). In the intermediate subzone 168 fuel assemblies are installed, 18 fuel assemblies with EPS, 30 fuel assemblies with RCS and 6 passive devices insertion of negative reactivity (PDINR). In the peripheral subzone 222 fuel assemblies are installed (Figure 6). The reactor core is surrounded by three rows of side lead reflector units (Side lead reflector units not shown in Figure 6). All the fuel assemblies have a hexagonal steel casing width across fields 196 mm and with 2.5 mm wall thickness. The fuel assemblies in the triangular lattice nodes are arranged at a pitch of 199 mm. The fuel elements are arranged in a triangular lattice at a pitch of 13 mm.

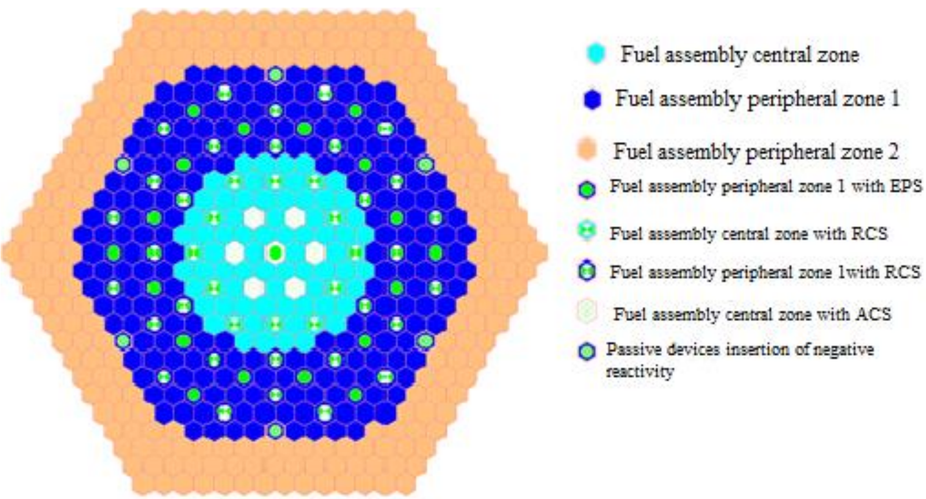

Fig.6 – BR-1200 reactor core composition (thermal power – 2800 MW)

Power flux forming is achieved by changing the fuel column height (Figure 7) while preserving fuel element diameter. This method of power density allows power flux forming uniform power and temperature distribution along the reactor core radius, without hydraulic profiling [3,4].

The fuel element diameter is 10.4 mm, shell thickness is 0.5 mm, the gap between the fuel and the shell is 0.1 mm. Fuel pellet diameter is 9.2 mm. Fuel column height in the central subzone is 58 cm, in the intermediate – 68 cm, in the peripheral – 83 cm. Spacing is achieved by means of a 2.6x0.5 mm strap with a rectangular cross-section twisted on itself and winded around the shell.

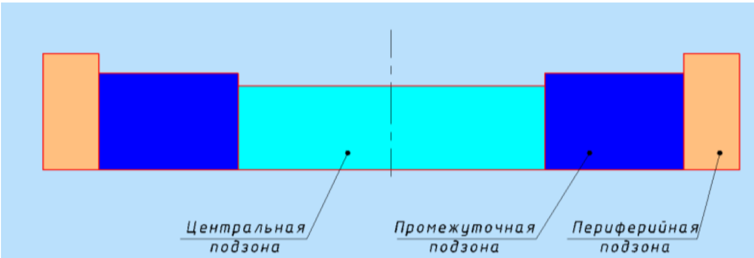

Fig. 7 – Three zone profiling power flux of reactor core using height of fuel rods

In the result neutronic characteristics were obtained for the BR-1200 reactor core with fuel elements, containing a gaseous (helium) cavity and a liquid-metal sublayer (TABLE II). For the liquid-metal sublayers moistening (sodium) and not moistening (lead) liquid-metals were considered.

Reactor core variants with a helium cavity and a sodium sublayer have identical configuration and insignificantly differ from one another in terms of neutronic characteristics. All temperature reactivity coefficients in both variants are negative, except one, which is related to temperature induced change of lead density only in the reactor core. The Doppler-effect of these zones in comparison with the zone with a helium sublayer decreases by a factor of 2.2, same as the power effect. The difference in reactivity effects is triggered by a decrease in fuel temperature due to liquid-metal sublayer application.

TABLE II – Main characteristics of the core

Characteristics	Helium gap	Sodium gap
Fuel (U-Pu-MA)N, Fuel density, г/см^3 (Pu+MA) content,%	11,8 14,1	
Fuel element diameter, mm Central zone Peripheral zone	10,4x0,5 10,4x0,5	
Fuel assembly quantity Central zone/Peripheral zone$_1$/Peripheral zone$_2$	547 85/240/222	
Mass fraction of fuel, steel, lead: $\varepsilon_{fuel}/\varepsilon_{steel}/\varepsilon_{lead}$	0.430/0.160/0.401	
H_{fuel} Central zone/Peripheral zone$_1$/Peripheral zone$_2$, cm	58/68/83	
Fuel element pitch, mm	13	

Fuel assembly width across fields, mm	196	
Fuel assembly pitch, mm	199	
Fuel element quantity: In fuel assembly In EPS In RCS and ACS	 217 90 156	
Reactor control and protection system materials	Reactor core, RCS – B_4C (80 % ^{10}B); EPS – Er_2O_3	
Average power of fuel assembly, MW	5.12	
Maximum power of fuel assembly Central zone/Peripheral zone$_1$/Peripheral zone$_2$, MW	5.02/5.90/6.50	5.01/5.91/6.51
Irregularity coefficient along the radius, K_r	1.27	
Power release irregularity coefficient (height) $K_{Z1}/K_{Z2}/K_{Z3}/$	1.13/1.17/1.20	
Fuel load mass, tHM	64.5 (60.9)	
average nuclear fuel burnup 1500 eff.days, MWd/kg (% h.a.)	72 (7.02)	
Maximum (local) fuel burnup through fuel cycle, MWd/kg (% h.a.)	107 (10.65)	
Maximum reactivity overshoot for steady-state operation conditions for time period between refueling, β_{eff}	0.2	
Reactor core breeding ratio (fuel cycle average)	1.06	
Full power effect (693K-1300K) Doppler effect B_{eff} Prompt neutron lifetime, τ	-5.21E-03 -4.00E-03 3.80E-03 3.95E-07	-2.48E-03 -1.81E-03 3.80E-03 3.95E-07
ΔK, for changing lead density in the entire reactor by 5%	-2.95E-04	-2.85E-04

The void reactivity effects for the entire reactor in zones with a helium as well as a sodium sublayer are negative and are practically the same, since the change from helium to sodium does not affect fuel column height and mass fraction.

Transitioning to a lead sublayer leads to gap width increase due to technical reasons, an insignificant increase of fuel load (~2%) and an increase of the void reactivity effect but within the negative range.

Effective profiling power flux along the reactor core radius by means of altering the fuel column height while preserving the diameter of the fuel element and keeping the same pitch and fuel composition allows uniform power distribution along the reactor core radius with the irregularity coefficient equaling $K_r \sim 1.27$. It is noteworthy that due to the homogeneity of the fuel composition and the breeding ratio being ~ 1, the uniform distribution undergoes almost no change throughout the fuel cycle.

Calculations showed that under steady-state refueling conditions when using nuclear fuel close to equilibrium composition, with a 1500 effective days fuel cycle length (interval between refueling – 500 effective days) the reactivity overshoot value equaled to 0.2 β_{eff}. The average fuel cycle breeding ratio in this case totaled 1.06. Fuel load mass totaled 64.5 t, average fuel burnup equaled 7.02% h.a. This fuel burnup value is consistent with the requirements originally established for the reactor core.

Average burnup for the steady-state refueling conditions of the aforementioned reactor core variant composition could reach 12% h.a. with fuel cycle length equaling 3000 effective days (interval between refueling – 600 effective days) and a 69 t fuel load. Having said so, the reactivity shift would not exceed ~$\beta_{eff.}$ It should be said that the limiting factor for increasing fuel burnup value is the restriction concerning the damaging dose on the fuel element shell material. However, in this analysis, this factor was not taken into account, due to the fact that this issue requires conducting s separate study and only the neutronic characteristics of this issue were examined.

Calculations concerning CPS efficiency for the reactor core composition, given in Figure 6 showed, that 19 EPS, 36 RCS and 6 ACS allow establishing two separate systems for reactor shutdown, and performance-wise comply with nuclear safety regulation requirements. Recently there has been growing interest in passive reactor safety systems without mechanical moving parts. These passive negative reactivity insertion devices (PDINR).are allocated in the reactor core being examined in this study and contains liquid cadmium as an absorber. The efficiency of the six PDINR equaled to 0,6 β_{eff}. Amplifying PDINR efficiency through increasing PDINR quantity was also studied. Thus, the PDINR quantity was increased to 12. Overall efficiency equaled 1,2 β_{eff}. An efficiency value of a passive negative reactivity insertion system of that scale allows the fuel, cladding and coolant temperature to stay within the maximum limits defined in the design requirements even in the event of beyond design-basis accidents related to both reactor shutdown systems failing.

5 Conclusion

A series of calculations have been conducted concerning neutronic and thermohydraulic reactor core characteristics for a fast lead-cooled nuclear reactor using nitride uranium fuel. As a result principal neutronic characteristics were obtained, trends (or dependencies) of these characteristics from the fuel mass fraction were established.

The study of neutronic and thermal characteristics for the BR-1200 reactor showed reactor core configuration designs variants which comply with void effect requirements, reactivity overshoot between refueling ~ $\beta_{eff,}$ fuel load minimization and average fuel burnup requirements. These design solutions are mainly related to increasing fuel ratio (while limiting maximum fuel element cladding temperature) and using a forming fuel load along the reactor core radius by means of modifying the fuel column and preserving one fuel element frame-size in the jacket fuel assembly.

Under steady-state refueling conditions, when using nuclear fuel close to equilibrium composition, average burnup values equaling to 12% h.a. could be achieved with fuel cycle length equaling to 3000 effective days (interval between refueling – 600 effective days). The reactivity shift related to increase in burnup would not exceed ~$\beta_{eff.}$

A variant for reactor core configuration was proposed, which could be considered as one of the possible variants for the future stages of reactor core design.

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