Low-power lead-cooled fast reactor loaded with MOX-fuel

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Abstract. Fast reactor for the purpose of implementation of research, education of undergraduate and doctoral students in handling innovative fast reactors and training specialists for atomic research centers and nuclear power plants (BRUTs) was considered. Hard neutron spectrum achieved in the fast reactor with compact core and lead coolant. Possibility of prompt neutron runaway of the reactor is excluded due to the low reactivity margin which is less than the effective fraction of delayed neutrons. The possibility of using MOX fuel in the BRUTs reactor was examined. The effect of \( K_{\text{eff}} \) growth connected with replacement of natural lead coolant to \( ^{208}\text{Pb} \) coolant was evaluated. The calculations and reactor core model were performed using the Serpent Monte Carlo code.

1. Introduction
Existing training simulators are not fully recreating nuclear power plant conditions, so the creation of training reactor will provide an opportunity to gain valuable practical skills. In addition to education, such reactors can be used for scientific and industrial purposes. At present, training reactors already exist in the world, or they are developed. For example, there are SLOWPOKE reactors in Canada [1] and training reactor VR-1 in Czech Republic [2].

Of particular interest are lead-cooled fast reactor projects for training purposes (ELECTRA, BRUTs) [3]. The main features of the BRUTs are: low power – 0,5 MW, the small sized core with a hard neutron spectrum, the low reactivity margin which is less than the effective fraction of delayed neutrons [4].

Purpose of the investigation is to obtain the characteristics of the BRUTs core loaded with MOX fuel and cooled with different lead coolant composition.

2. BRUTs model
2.1. General design layout and reactor core
Pool-type configuration ensuring arrangement within the same reactor vessel of the reactor core with reflectors, as well as upper shielding plug, heat exchangers, coolant treatment equipment for maintaining thermodynamic activity of oxygen dissolved in the coolant, thermal engineering sensors, in-vessel shielding and buffer plenum above the molten lead free surface was suggested for the examined design of BRUTs reactor.

BRUTs reactor vessel represents rugged steel cylindrical containment with elliptical bottom and lid equipped with bores for installation and fastening the retractable in-vessel devices.

Reactor vessel is mounted inside concrete shaft with gap between the vessel and shaft walls designed for arrangement of natural air circulation circuit for removing residual heat through the vessel in passive mode and releasing this heat in atmosphere. Separating barrel dividing cold and hot
coolant flows and ensuring installation of the retractable core support barrel with reflectors and support plate and designed as part of the arrangement of the primary coolant circulation loop is available inside the vessel.

It is expected that circulation of coolant in the primary cooling loop will be achieved by natural convection. Selection of natural circulation for the primary cooling loop allows avoiding the problem of development and substantiation of operability of circulation pump for pumping heavy liquid metal coolant. Calculation estimations demonstrate that for achieving natural circulation of coolant in the primary loop it is sufficient to have riser with height of \( \sim 3 \) m above the reactor core [4].

Reactor core consists of seven fuel assemblies, with 252 fuel pins each. Fuel pins are arranged in hexagonal lattice. Technological channels are arranged in the center of each of the fuel assemblies for placement of (rods) of the reactor control and protection system (SUZ) (emergency shutdown system (RO AZ), reactivity compensation system (RO KR) and automatic reactor control system (RO AR)).

The core model used in the investigation has the parameters listed below in Table 1. Cross-section of the BRUTs reactor is shown in Figure 1.

| Table 1. Main technical characteristics of the BRUTs reactor core. |
|---------------------------------------------------------------|
| Characteristics, measurement unit | Value |
| Fuel pin diameter along the smooth part, mm | 12.7 |
| Fuel pellet diameter, mm | 11.5 |
| The gas gap width (helium), mm | 0.05 |
| Fuel pin lattice spacing, mm | 14 |
| Number of fuel pins in the fuel assembly, items | 252 |
| Reactor core diameter, mm | 618 |
| Reactor core height, mm | 780 |
| Fuel pin cladding material | Steel EP 823 |
| Technological channels outer/inner diameters, mm | 48/44 |
| Fuel composition | PuO\(_2\) + UO\(_2\) |
| Fuel mass density, g/cm\(^3\) | 9.2 |
| Fuel volume in the core, % | 61 |
| Coolant volume in the core, % | 25.5 |

2.2. Lead coolant
The primary coolant is lead coolant circulated by natural convection. Depending on the type of lead coolant, it may have a different isotopic composition. Natural lead (Pb-nat) includes 4 isotopes: \(^{204}\)Pb (1.5%), \(^{206}\)Pb (23.6%), \(^{207}\)Pb (22.6%), \(^{208}\)Pb (52.3%). Natural lead mass density is 10.686 g/cm\(^3\) (at 600°C), atomic weight – 207.2.

\(^{208}\)Pb has the best neutronic properties among listed isotopes. Compared to other isotopes it has the smallest cross-section of the one-group neutron capture and the highest threshold for inelastic neutron scattering [5][6].
2.3. MOX-fuel
Current study deals with MOX-fuel with the following composition of civil (reactor-grade) plutonium: 
\[ ^{234}\text{U} - 0.03\%, \quad ^{235}\text{U} - 0.03\%, \quad ^{236}\text{U} - 0.04\%, \quad ^{237}\text{Np} - 0.1\%, \quad ^{238}\text{Pu} - 1.19\%, \quad ^{239}\text{Pu} - 63.05\%, \quad ^{240}\text{Pu} - 21.5\%, \quad ^{241}\text{Pu} - 4.07\%, \quad ^{242}\text{Pu} - 4.12\%, \quad ^{241}\text{Am} - 5.87\%. \]

Previous BRUTs reactor model is used for the study, which was created to calculate the BRUTs core cooled with \(^{208}\text{Pb}\) and loaded with UO\(_2\) fuel [7].

Possibility of prompt neutron runaway of the reactor can be excluded by the low reactivity margin which is less than the effective fraction of delayed neutrons. MOX-fuel has \(\beta_{\text{eff}}=0.0035\).

![Figure 1. General appearance of the BRUTs reactor: 1 – reactor core; 2 – heat exchanger; 3 – reactor control and protection system drives; 4 – reactor core barrel; 5 – reactor vessel](image_url)

3. Calculation results
The enrichment of MOX-fuel has been consistently chosen to meet the low reactivity requirement. At constant technical parameters of the core with the use of natural lead coolant (Pb-nat) the desired enrichment was obtained 22.8\%, and \(K_{\text{eff}} = 1.00310 \pm 0.00042\). MOX-fuel with \(^{239}\text{Pu} = 14.4\%\) and \(^{241}\text{Pu} = 0.93\%\) is acceptable for use in the BRUTs. All calculations in the study were carried out with Serpent 1.1.17, which is Monte Carlo reactor physics calculation code. It allows calculation of various characteristics of systems containing nuclear fissile material. Serpent calculation code is capable of full-core modeling of research reactors, SMR's, and other closely coupled systems [8].
Replacement of Pb-nat coolant to $^{208}$Pb with other constant core parameters gives increased effective multiplication factor equal to $K_{\text{eff}} = 1.03183 \pm 0.00088$, and $\Delta K_{\text{eff}} \approx 2.86\%$. This effect was examined in previous studies [7]. As compared with uranium oxide fuel MOX-fuel gives stronger effect (the effect for $\text{UO}_2$ was $\Delta K_{\text{eff}} \approx 1.6\%$).

As the percentage of the coolant in the reactor core is low, increase in $K_{\text{eff}}$ is insignificant. In more powerful reactors with greater fraction of coolant in the reactor core will provide greater effect [5]. The same considerations are acceptable for achievement of harder neutron spectrum with $^{208}$Pb [9].

**Conclusion**

The possibility of using MOX fuel in the BRUTs reactor was examined. Fuel enrichment of reactor-grade plutonium was obtained. Low reactivity margin which is less than the effective fraction of delayed neutrons excludes prompt neutron runaway. The effect of $K_{\text{eff}}$ growth connected with replacement of natural lead coolant to $^{208}$Pb coolant was evaluated – $K_{\text{eff}}$ growth on MOX-fuel is higher. Calculations confirm that the use of lead coolant enriched by $^{208}$Pb can reduce the initial enrichment of MOX fuel or to reduce the amount of the fuel loading [10].

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