Comparison of Simple Design of Sodium and Lead Cooled Fast Reactor Cores

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Abstract

The purpose of this report was to present the results of a numerical simulation of thermal hydraulics processes in the liquid metal cooled fast reactor core, combined with the simple neutron population computing for infinite pin cell lattice. Two types of the coolant have been studied: liquid sodium and the liquid lead, with all requirements regarded to safety conditions. Temperature distributions along the cooling channel and distributions in radial direction have been prepared and in the next step the criticality calculations using MCNP Monte Carlo code for MOX fuel have been conducted.

I. INTRODUCTION

The aim of this paper was to assess and compare heat transfer in fuel element and cooling channel of the fast neutron reactor in steady state condition, cooled by liquid sodium and lead with specified assumptions and inlet conditions respectively. Both metal coolants can be used for fast reactor cores due to their high cooling capabilities, high thermal conductivity and high heat transfer coefficients. Additionally, they are convenient in terms of neutronic economy due to small cross sections for parasitic capture as well as very low moderation of neutrons. Those properties are crucial in fast reactor design. Sodium has lower than lead melting temperature and better thermo–physical properties, but is much more chemically active and has lower boiling temperature, whereas lead is chemically inert but highly corrosive for steel. The report also shows the methodology used to achieve the goal, presenting also results and conclusions.

The paper has been developed at the Information Platform TEWI.

Information Platform TEWI

TEWI stands for T - technology, E - education, W - knowledge, I - innovation. It is a information platform which aims to integrate Polish scientific community by creating an interregional platform to share the knowledge. The platform provides means to conduct the newest research and development, to create innovative products and to cooperate with the industry. In addition, the platform shall contribute to raising the professional qualification, promotion of the latest technological solutions and creating value added products through teamwork (Lipinski and Swirski (2012)).

All this will create conditions in which the influence of the platform will cause the increase of the competitiveness of economy and interregional cooperation will increase the competitiveness and knowledge in the region as well as in the whole country.

Plutonium resources

One of the problems of the contemporary nuclear power plants is that in thermal reactors only a small amount of fuel is utilized. Demand for electricity generated by the nuclear power plants is forecasted to increase, so there is a need to take into consideration the enhancement the utilization of the fuel. The solution of the issue are Fast Breeder Reactors (FBR) which enable a possibility of increased fuel usage. The Generation IV International Forum (GIF) is a cooperative international endeavor organized to carry out the research regarding the next generation nuclear energy systems. GIF identified and selected six nuclear energy systems for further development.

Three of the proposed generation IV reactor types are FBRs:

- Gas- Cooled Fast Reactor (GFR)
- Sodium- Cooled Fast Reactor (SFR)
- Lead- Cooled Fast Reactor (LFR)

The two most important fissile isotopes are uranium-235 and plutonium-239. Uranium-235 is found in nature (0.7% of natural uranium). Plutonium-239 is created artificially when uranium-238 (99.3% of natural uranium) absorbs a neutron and then the resulting nucleus undergoes two beta minus decays. Thermal reactors used in contemporary power plants utilize uranium very inefficiently. According to the International Atomic Energy Agency, demand for nuclear power will increase by 25% in the low projection and by 100% in the high projection (IAEA, 2012a). Total identified resources of uranium are sufficient for over 100 years of supply based on current requirements (IAEA, 2012b). So the question is what should be done to increase fuel usage. FBRs seem to be the answer. The technology of fast reactors has the potential to multiply by a factor of 50 to 100 the energy output from a given amount of uranium (with a full use of U-238, which is converted into Pu-239 by irradiating with fast neutrons). FBRs are able to provide energy for the next thousand of years

Facility	Country	Ther.power [MW]	El.output [MWe]	Operation	Coolant
Clementine	USA	0.025	-	1946-1952	mercury
BR-2	USSR	0.1	-	1956-1958	mercury
BM-40A*	USSR	155	-	1969-1990	LBE
BN-350	Kazakhstan	1000	90	1972-1999	sodium
Phenix	France	345	142	1973-2010	sodium
BN-600	Russia	1470	600	1980-pres.	sodium
Superphenix	France	3000	1242	1985-1998	sodium
Monju	Japan	714	280	1994-1995	sodium

*used in the Alfa-class submarines

Table 1: List of the most important fast reactors built to date (source: World Nuclear Association Reactor Database, http://world-nuclear.org; IAEA Power Reactor Information System http://www.iaea.org/pris).

with the already known uranium sources (SNETP, 2010). In the face of limited resources of uranium and increasing demand for nuclear power, there is a need for new nuclear systems.

Coolants for FBRs- stages of development

The first fast reactor called Clementine was constructed in 1946 in the USA. The core was cooled by mercury. The reactor was decommissioned due to a breakdown in 1952- the uranium slugs swelled bursting the cladding and released plutonium into the mercury coolant. The next fast reactor cooled by mercury was BR-2 built in 1956 in the USSR. Similarly to the Clementine, the BR-2 was damaged- plutonium fuel was not stable under irradiation even at low temperatures and mercury leaked from pipe joints and corroded the steel cladding. Because of the mercury's strong corrosive effect, it is no longer used or considered as a coolant in reactors. Later, sodium was used as a coolant for FBRs except reactors used in the Alfa-class submarines cooled by LBE. A list of the most important fast reactors built to date is presented in Table 1.

The idea of building FBRs was forced in the early 1970s, when a significant development of the nuclear power was expected and there were concerns of a rapid exhaustion of the world's uranium sources. This resulted in increase of the uranium price from $6 to $40 on the spot market between 1973 and 1976(Bunn et al., 2005). The rate of development of nuclear power reduced in the 1980s and the interest in FBRs decreased, which was caused by the reduction of nuclear arms (the use of fuel from nuclear warheads) and the changes in Eastern Europe.

With the establishment of the GIF in 2001 and the appearance of the project to build generation IV reactors, the research on FBRs has been intensified. As a new type of coolants for the fast reactors were selected: helium, sodium, lead and LBE.

Coolant	Na	Pb	LBE	He
Melting Points [°C]	97.8	328	125	-
Boiling Point [°C]	883	1750	1670	-
Density [kg/m3]	790	10540	10200	3.8
Specific heat [J/kgK]	1260	160	146	5260
Th. Conductivity [W/mK]	62	16	11	0.29

Table 2: Thermophysical parameters of the coolants (approximate values at 700°C and 1 atm, except 7.5 MPa for helium) (Zhang et al., 2009; Fanning, 2007).

Thermophysical properties

Basic thermophysical parameters of sodium, lead and LBE such as melting and boiling point allow a safe operation of FBR at the standard atmosphere pressure- primary circuit's temperature is between 400 and 600°C. A disadvantage of liquid metals is the fact that the melting temperature is so high that the cooling system must be heated to prevent solidification of the metal. This problem does not occur in the case of helium, where the change of the physical state in the core is impossible. Its specific heat is high, but the density is very low. These properties affect high pressure in the cooling system (between 7.0 and 8.5 MPa) and the high velocity (~100 m/s), which can lead to significant vibrations of the fuel pins.

Lead-alloy coolant velocities are limited by erosion concerns of protective oxide layers to about 2.5–3.0 m/s. Typical sodium velocities are up to 8–10 m/s, hence lead has, in practice, a lower heat removal capacity (Tucek et al., 2006).

Low thermal conductivity of helium results in poor heat transfer even at high coolant velocity. Cladding surfaces can be roughened (it results in an increased pressure drop over the core, and a higher requirement on pumping power) to improve heat transfer (4x), but it is still 8-9x lower than for sodium (Fanning, 2007).

Material properties

The main problem in the case of sodium is its high chemical reactivity. It reacts very rapidly with the air. Its reaction with water is highly exothermic and can proceed explosively- it requires a very careful design of the water- sodium heat exchanger. Helium, lead and LBE are chemically inert to water and air, which improves safety during operation of the reactor.

One of the drawbacks of lead and LBE is their strong corrosive (LBE is even more corrosive because the addition of bismuth). To protect the pump and cladding materials against corrosion they are covered with oxide layers. These layers are sensitive to the coolant temperature above 590°C and very high flow rates- hence velocity is limited to 2.5-3.0 m/s (Tucek et al., 2006). The use of helium (directly in Brayton cycle) and sodium does not require any special measures to protect against corrosion in the core.

Each of the coolants is characterized by high radiation stability.

Neutronic properties

Sodium, lead and LBE introduce a small amount of parasitic absorption. Fast neutrons in contact with the nuclei of these metals show a small slowdown. In addition, lead and LBE are neutron reflector- neutrons which leaked out from the core are directed back to it. Hence, we can also infer that the neutron economy of the lead-alloy-cooled systems would be better than for sodium-cooled counterparts having the same geometry. For example, lead-alloy-cooled, fuelled systems require smaller plutonium enrichments than sodium counterparts to reach criticality (Tucek, 2004).

Helium is transparent to neutrons. Its low density leads to negligible moderation, but also affects a higher neutron leakage fraction from the core.

Despite the small neutron capture during irradiation, both sodium and LBE form radioactive isotopes. Sodium creates Na-24 with a half-life of 15 hours. Bismuth-209 turns into bismuth-210, which decays to polonium-210. This element is very toxic, however, it rapidly forms a stable compound with lead - PbPo, which is well retained in the Pb-Bi coolant.

Safety and costs

A very important criterion when comparing properties of the coolants for FBRs is safety. The main issue in the case of sodium is that its reaction with water is highly exothermic. If it leaks out and contact with water, the reaction may lead to a fire. High reactivity of sodium with air and water causes the service within the reactor difficult to perform and time consuming. Moreover radioactive sodium-24, which is formed in the core during the irradiation, requires a construction of intermediate sodium loop. It is inserted in order to prevent a steam generator fire dispersing radioactive isotope- primary loop cools the core and exchange heat with the intermediate loop, which transfer the heat to the steam generator.

Liquid lead for FBRs is a proper protection against gamma rays and for this reason it is used as an additional protective barrier for spent nuclear fuel. Spent nuclear fuel storing prior to reprocessing is presumed to realize as follows. After the spent fuel sub-assembly has been extracted from the reactor, it is installed in a penal, in which lead has been previously heated in an electric furnace over its melting point. Then the penal is sealed and transported to the "dry" storage with natural convection air-cooling. At this, lead in the penal is solidifying gradually and forms an additional protection barrier (Zrodnikov et al., 2008).

The use of bismuth dopant in lead causes that the melting point of LBE is lower than of pure lead, but it also increases the possibility of radioactive contamination. In case of the coolant leakage through the reactor vessel or the steam generator, a significant release of polonium to the reactor room might occur, which could pose a serious radiological problem. PbPo compound evaporates directly or interacts with water and forms volatile alpha-emitting aerosol-both are extremely dangerous when inhaled because of emitting alpha particles (Buongiorno et al., 2003).

Helium is an inert gas, but as the coolant it also entails risks. In case of a depressurization event, the core's cooling system might be inefficient.

Direct costs of the coolants are as follows (USGS, 2012):

- sodium: \$3.5/kg (\$2 765 for m^3, density 790 kg/m^3)
- lead: \$2.5/kg (\$26 350 for m^3, density 10 540kg/m^3)
- bismuth: \$25.6/kg (\$245 760 for m^3, density 9 600kg/m^3)
- helium: \$5.77/m3 (\$5.77 for m^3, but under pressure of 7.5 MPa \$430 for m^3)

Taking into account only the direct costs, the cheapest coolant is helium and the most expensive is LBE (because of bismuth dopant). However, there are also indirect costs, which should be taken into consideration such as: pumping requirements, construction of the intermediate loop (in case of SFR) and safety-related systems.

A preliminary analysis of the costs of the first generation IV FBRs in Europe is presented below :

- Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID)- pre-industrial scale prototype fast reactor with an electrical power of the order of 600 MWe, which is planned to be enable commissioning by 2022; cost: 5 000 M€
- Advanced Lead Fast Reactor European Demonstrator (ALFRED)- cooled by LBE with a power of approximately 100 MWe that will allow connection to the grid, commissioning is planned in 2025; cost: 1 000 M€
- ALLEGRO- the world's first gas-cooled fast reactor, in the range of 70 to 100 MW, with construction in the 2020s; cost: 1 200 M€

In this paper only sodium and lead was considered in further studies.

II. METHODOLOGY

Initially, the geometry of the fuel cell was set up. Due to a selected hexagonal lattice a cell unit is triangular. The following dimensions: fuel rod active length, pin diameter, cladding thickness, gas gap thickness and pin to pitch ratio were assumed. Geometry is presented in Figure 1.

The next step was to set up coolant inlet parameters: temperature and velocity. In case of sodium as a coolant, velocity is limited to 9 m/s due to mechanical vibration of fuel rods, while the lead velocity cannot exceed 2 m/s

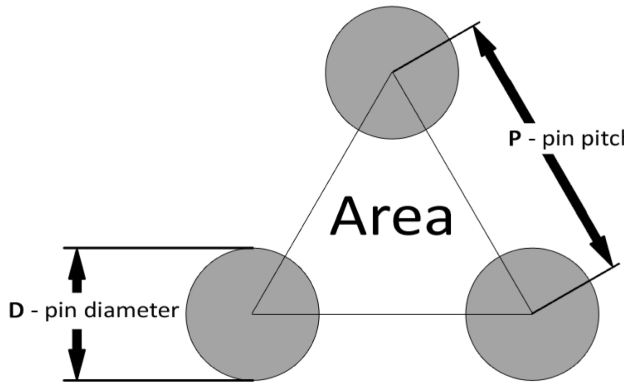

Figure 1: Geometry description

Cr	**Mo**	**Si**
8.35	1.03	0.45
V	**Mn**	**Fe**
0.202	0.47	remaining part

Table 3: Alloying elements of T91 steel, wt.%.(Williams et al., 1984)

due to erosion concerns (Jimenez et al., 2009). To have reasonable error margin, velocities of 8 m/s and 1,5 m/s for sodium and lead respectively were selected. Cause of the difference in coolant flow velocities, the hydraulic diameter for lead had to be greater than that for sodium due to increase of the flow area that leads to improvement of heat removal capabilities. Final P/D (pitch to diameter, as depicted in Figure 1) ratios corresponding to velocities were set at 1.2 and 1.6 for sodium and lead respectively.

Next parameter to set up, was the maximum of linear power density – 300 W/cm. Axial power profile was approximated by a cosine distribution, due to approximated neutron flux distribution (Kiełkiewicz, 1987). Extrapolation length was set to 35% of radius for sodium and 50% for lead. Coolant inlet temperature was 673 K, which is the minimal temperature to avoid solidification of lead. Same temperature was set for sodium. For the sodium coolant an austenitic steel 15-15Ti (Cheon et al., 2009) was selected. With respect to the corrosion produced by liquid lead, a ferritic-martenisitic steel T91 (Jimenez et al., 2009; Williams et al., 1984) was selected as a cladding material considering its ability to form Cr-oxide protective layer at cladding surface as well as better performance than austenitic steel. The composition of the austenitic steel is shown in Table 3 and the ferritic-martensitic in Table 4 . The assumptions and the design parameters are presented in Table 5.

Knowing all of the necessary parameters and additionally assumed the composition of the MOX fuel that consist of 20% of plutonium and 80% of depleted uranium, it is possible to perform thermal hydraulics calculations. Oxygen to

Cr	**Ni**	**Mo**
15	15	1.2
Si	**Mn**	**Ti**
0.6	1.5	0.4
C	**P**	**Fe**
0.1	0.03	remaining part

Table 4: Alloying elements of 15-15Ti steel, wt.%.(Cheon et al., 2009)

Coolant	**Sodium**	**Lead**
Rod active length [m]	1.0	
Pin diameter [cm]	0.61	
Cladding thickness [cm]	0.4	
Gas gap thickness [cm]	0.015	
Max. linear power density [W/m]	300	
Inlet coolant temperature [K]	673	
Coolant velocity [m/s]	8.0	1.5
P/D ratio	1.2	1.6

Table 5: Assumptions and design parameters.

metal ratio was assumed to be 1.98 and fuel density was set to 95% of theoretical density. Fuel was composed of recycled plutonium and depleted uranium from spent PWR fuel with initial enrichment of 4.5% and burnup of 45 MWd/kg after 15 years, working in the PWR reactor core (Mazgaj, 2010). The fuel density was: 10.559 g/cm3 . Fuel vectors are presented in Table 6. Critically calculations were accomplished only for BOL fuel state, which means that fuel was taken as fresh in all calculations.

Plutonium		**Uranium**	
Isotope	Atom. fraction (%)	Isotope	Atom.fraction (%)
^{238}Pu	2.3477	^{234}U	0.0031
^{239}Pu	57.0151	^{235}U	0.4091
^{240}Pu	26.9515	^{236}U	0.0101
^{241}Pu	6.0693	^{238}U	99.5777
^{242}Pu	7.6164		

Table 6: The fuel vectors of Pu and U used in calculations.Mazgaj (2010)

All of essential heat transfer correlations in steel were taken from Jimenez et al. (2009); Wallenius (2010). The temperature dependence of thermal conductivity of the MOX fuel is described in Waltar and Reynolds (1981) and the complete heat transfer model were taken from Kiełkiewicz (1987); Wallenius (2010); Pfrang and Struwe (2007); Todreas and Kazimi (1990). Three ways of heat transfer can appear in the open gap: radiation, conduction and convection. Due to very high working temperature of the fuel, radiation is the most important while the convection is negligible (Waltar and Reynolds, 1981). In this case, due to comparable pin and fuel rod diameters, and the small size of the gas gap,Sobolev et al. (2009) recommends heat transfer coefficient of the value of 6000 W/m^2K for the gas gap. All the necessary equations were coded in MATLAB environment and the iterative process was necessary to run to figure out the results. Fuel rod was divided axially into 500 elements. Fuel and coolant temperatures were calculated with steps 1K and 0.1K respectively. This part of code is called outer iteration. Inner iteration procedure with step 1K for temperature was used to find radial temperature distributions for every axial element with assumption of no axial heat transfer in fuel element. It was possible to calculate exact temperature distribution only in fuel and in cladding. The rest of the founded temperatures are for external or internal surface and central point of cooling channel. Temperature in the fuel is not strictly parabolic due to dependence of thermal conductivity of MOX with temperature. This problem was solved using proper iterative methods and numerical integration by the Newton – Cotes formulas. The results, containing fuel, cladding and coolant temperature profiles, were compared with maximum assumed allowable temperatures. Then, the geometry and material content were inputted into MCNP5 Monte Carlo code to achieve criticality for infinite pin lattice and to find adequate geometrical configuration to complete design process (Goorley, 2004). During MCNP calculations, 1.5 m layer of coolant above and under reactor core was added to take into account finite size of reactor in axial direction. It the next step, another MATLAB script was used to estimate effective criticality using simple relations for radial leakage with assumption of core radius equaling 1 m, for assumed radial extrapolation length equaling 25 cm for both sodium and lead cases and for diffusion lengths of 15 cm and 18 cm respectively. Effective criticalities were calculated with margin for loss of reactivity during fuel irradiation (Kiełkiewicz, 1987; Waltar and Reynolds, 1981).

III. RESULTS

Results of the calculations, briefly described in methodology, were summarized in Table 7 and in a set of charts showing certain parameters of axial profiles for heat transfer in liquid sodium and lead.

Power density distribution profiles presented in Figure 2 are both described by a cosine functions, but in case of lead as a coolant there is higher flux extrapolation length because it reflects more neutrons into core than sodium does (Wallenius, 2010; Tucek et al., 2006).

Coolant	**Sodium**	**Lead**
Coolant outlet temperature [K]	735	787
Max. outer cladding temperature [K]	743	824
Max. inner cladding temperature [K]	756	840
Max. fuel outer surface temperature [K]	1007	1085
Max. fuel temperature in the pellet center [K]	1933	2038
Critically for the infinite pin lattice	1.31729	1.22576
Criticality standard deviation	0.0005	0.00044
Effective criticality	1.21602	1.094509

Table 7: Results obtained from the numerical calculations.

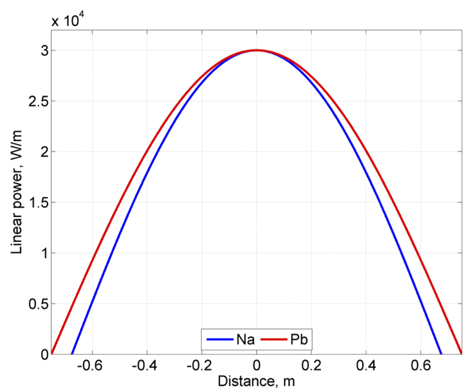

Figure 2: Approximate power density profiles.

The cell cladding external temperature profile is presented in the Figure 3. As it can be seen the outlet temperature for lead coolant is noticeably higher than for sodium one (1085 K and 1007 K respectively). At this point it can be clearly proven, that the heat removal ability of sodium is higher due to its higher heat transfer abilities.

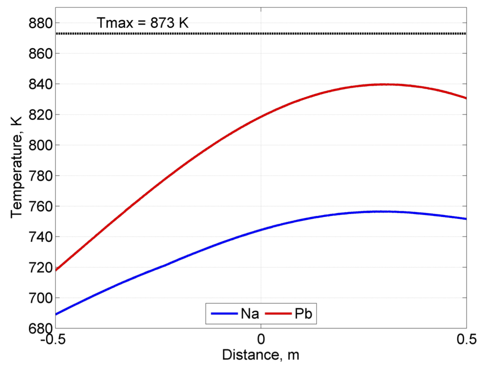

Figure 3: Cladding outer temperature profile.

Distribution of temperature in coolants as presented in Figure 4 supports this statement, and as it can be seen, the temperature of lead at the outlet is noticeably higher than in sodium (787 and 735 respectively) due to better heat transfer characteristics of the second one.

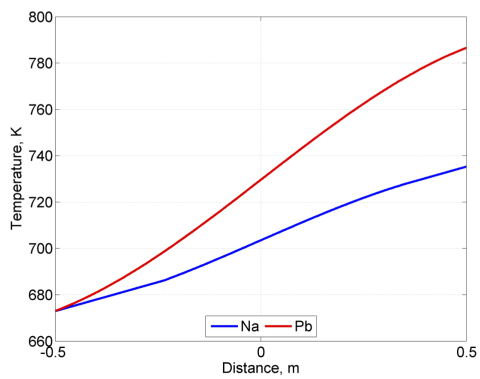

Figure 4: Axial coolant temperature distribution through the channel.

In Figure 5 the fuel temperature distribution in the fuel pellet is presented. As it can be seen, the fuel temperature for lead coolant is about 100 K higher and reaches maximum axial value of 2038 K, versus only 1933 K for sodium coolant respectively. The profile is slightly asymmetrical and has higher values for the half part that is closer to coolant outlet.

Figure 6 shows the radial temperature distribution in the fuel pin.

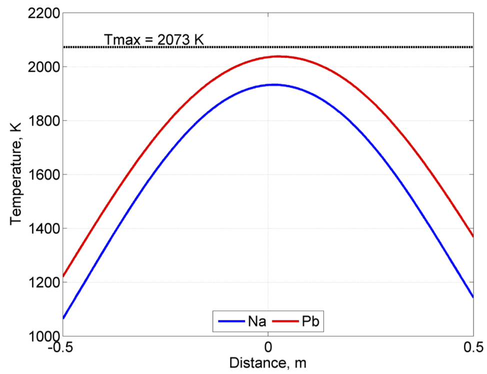

Figure 5: Fuel axial temperature profile.

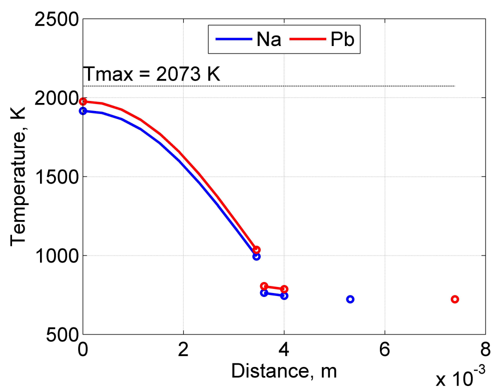

Figure 6: Fuel radial temperature profile in the hottest section of fuel pin.

IV. CONCLUSIONS

The results obtained seem to be compatible with the data presented in available literature. Although sodium has some disadvantages, such as high chemical reactivity and worse neutron reflection properties than lead, the thermal hydraulics analysis has confirmed that as far as regards thermo-physical properties, liquid sodium coolant is superior to lead. The main advantages are: better heat-removal capabilities as well as both the lower fuel and cladding temperatures. These features together with higher flow velocity of sodium lead to higher linear power available and lower pitch to diameter ratio required. Moreover the technological and operational experience gathered in the past half century is much higher for sodium than for lead applied as coolants respectively.

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