Innovative model of annular fuel design for lead-cooled fast reactors

Rowinski, Marcin Karol; White, Timothy John; Zhao, Jiyun

2015

Rowinski, M. K., White, T. J., & Zhao, J. (2015). Innovative model of annular fuel design for lead-cooled fast reactors. *Progress in Nuclear Energy*, 83, 270-282.

<https://hdl.handle.net/10356/98878>

<https://doi.org/10.1016/j.pnucene.2015.04.002>

© 2015 Elsevier Ltd. This is the author created version of a work that has been peer reviewed and accepted for publication by *Progress in Nuclear Energy*, Elsevier Ltd. It incorporates referee's comments but changes resulting from the publishing process, such as copyediting, structural formatting, may not be reflected in this document. The published version is available at: [<http://dx.doi.org/10.1016/j.pnucene.2015.04.002>].

Downloaded on 26 Apr 2023 10:26:31 SGT

Innovative Model of Annular Fuel Design for Lead-Cooled Fast Reactors

Marcin Karol Rowinski^{1,2}, Timothy John White¹, Jiyun Zhao^{3,*}

¹ Energy Research Institute at NTU, Interdisciplinary Graduate School, Nanyang Technological University, Singapore 639798

² School of Electrical and Electronic Engineering, Nanyang Technological University, Singapore 639798

³ Department of Mechanical and Biomedical Engineering, City University of Hong Kong, Kowloon, Hong Kong

* Corresponding author: Tel.: +(852)-3442-9395, E-mail address: jiyuzhao@cityu.edu.hk

Abstract

The results of an investigation of the analytical solutions of the default – solid fuel and innovative – annular fuel approach is summarized in this paper. Innovative annular fuel concept is described in details and shows that it is a promising technology in terms of safety aspects. Liquid metal cooled reactors are considered to burn used nuclear fuel from conventional nuclear power plants or processed weapon grade plutonium and therefore, higher safety standards must be assured.

During the design process seven fuel assemblies with annular fuel elements were created. The investigation was conducted in case of both commonly used fuel lattices i.e. square and hexagonal, moreover effect of spacer grids was taken into considerations. Results show that the annular fuel is superior in case of maximum fuel temperature, which is up to 757°C lower than in default base design – with use of spacer grids. The most promising designs are hexagonal lattice with 91 fuel elements and square lattice 18x18, where the maximum temperatures are 822°C and 732°C, while pressure drop of 185kPa and 128kPa, respectively.

The square lattice proved better performance according to our evaluation, it is possible to obtain very similar or smaller pressure drop than default – solid fuel – design. Hence, use of the same or even smaller coolant pumps is possible in case of annular fuel elements. Moreover, the innovative fuel also allows to overpower reactor up to 110% of nominal power, while securing all safety limits. Therefore, it is promising concept and further investigation would be interesting.

1. Introduction

The constant growth in energy demand in our life requires, despite building new power units, improvement in efficiency of current systems. Therefore, the production of the same amount of energy will require less resources, thus it will be friendlier to our environment. The simplest way to increase overall efficiency of a power station is to improve performance of a heat source, in case of nuclear power plant (NPP) it requires improvement of the fuel elements.

Nowadays, nuclear industry uses solid fuel elements due to years of experience and proven reliability. Nevertheless, it is not the most efficient solution in the case of heat exchange between heat source and coolant. Hence, in 2001 M. S. Kazimi [1] proposed annular fuel rods concept for water reactors and the investigation has shown better performance than the default solid fuel pin design. The feasibility study was conducted by the United States of America in case of water and sodium cooled reactors applications [2-4], while South Korea has conducted evaluation for water and gas-cooled reactors [5-10]. It was found that the temperature distribution is much lower in annular fuel element, while pressure drop along the channel can be decreased if the design of fuel assembly is optimized. There is no study related to application of annular fuel in lead cooled reactors.

The main difference between solid and annular fuel is to allow coolant to flow not only in a gap between fuel rods, but also through the element itself, thus the fuel is internally and externally cooled. Hence, the heat exchange area is much bigger and it leads to increase fuel

performance. It allows to use fuel with higher power density thus, the amount of heavy metal can be decreased while the overall power of the core will be the same or higher. At the same time, the maximum temperature in the fuel will be decreased significantly, providing higher safety margins in case of emergency or accidental conditions. On the other hand, increase of cladding material may lead to slightly higher enrichment of the fuel due to additional parasitic absorption.

All over the world researchers pursue Generation IV reactors, to this category following kinds of reactors can be assigned:

- Supercritical water reactors [11-14].
- Liquid metal fast reactors that can be further divided into:
 - Sodium cooled fast reactors [15-17].
 - Lead cooled fast reactors [18-21].
- Gas cooled reactors [6, 22, 23].

From the aforementioned types of reactors, the lead cooled fast reactor was chosen for further analysis due to coolant properties. The coolant is characterized by very high boiling point¹ at normal pressure conditions, lack of reactivity with water and/or air. Moreover, in case of leakage it would solidify, while different cooling medium could e.g. starts fire – sodium reacts violently with water and air. Together with relatively low melting temperature², low moderation ability and capability of forming compounds with iodine and cesium – convenient for coolant purpose – in case of radioactive leakage from fuel elements to the coolant, the fission product will stay inside the reactor vessel.

The fuel proposed for Generation IV reactors are mostly either mixed plutonium-uranium nitride or mixed plutonium-uranium oxide fuel, in order to flatten neutron flux distribution in

¹ Boiling point: a) lead – 1737°C, b) LBE – 1670°C

² Melting point: a) lead – 327°C, b) LBE – 125°C

core different plutonium enrichment zones are introduced. Two fuel lattice designs are investigated i.e. open square and closed hexagonal [19, 24, 25].

2. Methods

The process of calculations is divided into five parts from specification of fuel concept and reactor parameters through the design process to analysis of thermal parameters and pressure drop.

2.1. The fuel concept

The ordinary nuclear fuel has form of solid fuel pins. It is surrounded, usually, by cladding made of zircalloy which has very good neutronic properties, in between gas gap of inert gas is placed. The coolant flows around the fuel element in order to remove heat generated in the fuel. This kind of fuel rods can be found in most of commercial NPPs.

In 2001 M. S. Kazimi [1] proposed use of annular fuel, which is characterized by superior cooling ability, thus further increase of power density, while retaining safety margins is possible. The main concept of the innovative design is to provide cooling internally and externally, this feature greatly increases the heat exchange area. Therefore, the same coolant interface temperature will be correlated with much lower temperature distribution inside the fuel element than in the default solid pin design. The comparison of solid and annular fuel design can be found in figure 1.

The model for solid fuel was made for both hexagonal and square lattice in order to have a reference point for further analysis. The main limitations regarding the use of lead as a coolant are the maximum fluid velocity due to erosion of cladding material and maximum cladding surface temperature (MCST) due to corrosiveness of lead in high temperature. The geometry of a default fuel design was created for hexagonal and square lattice (fig. 2). Based on the dimensions of the fuel assembly, reactor properties and coolant to fuel volume ratio the

innovative fuel assemblies were created (fig. 3). The gas gap and cladding thicknesses are the same in every design, thus annular fuel must have lower amount of fuel and higher amount of cladding material in the fuel assembly. Nevertheless, in order to achieve as close as possible reactor performance the coolant to fuel ratio shall be the same for every design, thus new parameter δ , coolant-to-fuel volume ratio, was introduced and it is equal to 1.676 in case of square and 1.842 in case of hexagonal lattice. Moreover, another parameter β , which is the ratio between default and particular fuel volumes, hence it is always lower than unity.

The lattice design scheme is based on the following dimensions such as:

- outer element radius – $R_{c,oo}$,
- inner element radius – $R_{c,ii}$,
- pitch between fuel rods – p ,
- length of the lattice side (square) – a , or radius of circumscribed circle of the hexagon – R – constant parameter.

The scheme of dimensions the fuel elements is presented in figure 4.

2.2. Reactor parameters

The ELSY reactor design was chosen for investigation as a SMR that uses pure lead as a coolant. It is in development under Sixth EURATOM Framework Programme by numerous European institutes including research organizations and universities like: Ente Nazionale per l'Energia Atomica (ENEA), Nuclear Research and Consultancy Group (NRG), Centre national de la recherche scientifique (CNRS) or universities like Royal Institute of Technology (KTH), Sweden and AGH University of Science and Technology, Poland [26]. Table 1 presents the most important parameters of the reactor. Since, the prediction suggests superior cooling ability, the calculations were also conducted in case of 10% over power, shown in brackets.

Table 1 Main ELSY reactor's parameter [19]

Parameter	Value	
Power [MW_{th}]	1500 (1650)	
Inlet temperature [$^{\circ}\text{C}$]	400	
Outlet temperature [$^{\circ}\text{C}$]	480	
Maximum allowed clad surface temperature [$^{\circ}\text{C}$]	550	
Maximum fluid velocity [m s^{-1}]	2	
Core's active length [m]	1.2	
Total length of fuel element [m]	2.5	
Type of lattice	Square	Hexagonal
Number of fuel assemblies	162	427
Number of elements per assemble	441	169
Element pitch [mm]	13.9	15.5

The main difference between solid and annular fuel is the inner channel inside the fuel element. Hence increase of cladding material inside the reactor core, it may lead to higher parasitic absorption and influence the performance of the fuel cycle. Therefore, next step will be an evaluation of fuel cycle length with respect to fuel enrichment. Memmott et al. [4] conducted an evaluation of SFR founding that use of annular fuel requires slightly more enriched fuel and decreases conversion ratio inside the core, thus the reactor has greater burning ability than solid fuel useful in burning reactors.

2.2.1. Material properties

The coolant properties i.e. isobaric specific heat, dynamic viscosity, thermal conductivity and density of coolant are estimated at mean fluid temperature of 440°C according to the relations from [27].

The thermal conductivity of fuel (UO₂) is estimated to be 2.9 W m⁻¹ K⁻¹ according to [28], while the thermal conductivity of cladding material is taken from ASME standards [29]. The property was taken for austenitic steel AISI 316 L, which is the candidate for cladding material for LMCR and can be found in “group J”, thus the thermal conductivity at 525°C which is the closest to mean cladding temperature is 22.6 W m⁻¹ K⁻¹.

The thermal conductivity of gas gap is 0.9 W m⁻¹ K⁻¹ and it was calculated according to relation found in M. S. Kazimi report [2]:

$$k_g = h_g R_g \ln \left(\frac{R_{c,oi}}{R_{f,o}} \right) \quad (1)$$

where gas conductance h is assumed to be equal to 6,000 W m⁻² K⁻¹, which is typical value for ordinary fuel element.

2.2.2. Material limitations

During the years, lead-bismuth eutectic (LBE) was used as a coolant in Russian submarines fleet with no major accidents, thus the concept had proven its reliability [30-33]. However, the main disadvantages of lead and LBE are: corrosiveness in high temperatures which limits maximum cladding surface temperature (MCST) to 550°C, especially in case of LBE, since bismuth boosts the corrosive properties of the coolant. It was found that the austenitic and martensitic steels are able to withstand temperatures up to 550°C in lead environment if sufficient oxygen control is provided [34]. Müller et al. [35] conducted tests with use of austenitic stainless steels such as AISI 316L, 1.4970 and MANET founding that all of them are able to withstand temperatures up to 550°C for a period of 4000 hours. Only surface coated with Al alloy layer is able to withstand 600°C in lead medium. Hence, the conservative estimation will limit MCST to 550°C.

Literature suggests that no considerable cladding strain due to mechanical interaction between fuel and cladding will occur during steady-state operation. It is due to high plasticity of oxide

fuel under irradiation conditions, especially by irradiation-induced creep, while carbide based fuels are more rigid, thus its plasticity is much lower [36]. Another difficulties are non-transparency of the medium and its erosive properties, thus the velocity of the coolant must not exceed 2 m s^{-1} . In case of LBE it is required to mention much higher production of radioactive polonium-210 (^{210}Po) and limited resources of bismuth in nature, thus pure lead is more reasonable option. According to review done by Zinkle et al. [37] the high energetic neutron flux will cause accelerated degradation of the material, mostly due to void swelling and phase stability during irradiation. Therefore, comprehensive research of suitable cladding alloy is still ongoing.

2.3. Analysis of thermal parameters

In order to conduct analysis, it is necessary to specify known reactor parameters, calculate or estimate unknown factors and relevant equations are needed to be derived. The analytical analysis is based on thermodynamic models and following assumptions were made:

- cylindrical coordinate system,
- one dimension i.e. only $\frac{d}{dr}$,
- steady state i.e. $\frac{\partial}{\partial t} = 0$ and $q''' = \text{const}$,

2.3.1. Flow characteristics

The calculations of flow characteristics show how heat is removed from fuel elements, which is our heat source. In this case the most important factors are the velocity of the medium and the cross-section area together with hydraulic diameter.

Calculation of Reynolds number is as follow:

$$Re = \frac{\rho v D_H}{\mu} \quad (2)$$

Where ρ – density [kg m^{-3}], ν – velocity [m s^{-1}], μ – dynamic viscosity [Pa s] of coolant and the D_H is the hydraulic diameter [m] and can be described as follows:

$$D_H = \frac{4A}{W} \quad (3)$$

where:

A – cross section of flow area [m^2],

W – wetted perimeter of the cross section [m].

Let s be the ratio between fuel pitch and rod diameter. Therefore, for square and hexagonal lattice we can write following relations, respectively:

$$\begin{cases} D_H = 2R_{c,oo} \left[\frac{4}{\pi} (s)^2 - 1 \right] \\ D_H = 2R_{c,oo} \left[\frac{2\sqrt{3}}{\pi} (s)^2 - 1 \right] \end{cases} \quad (4)$$

The Nusslet number, it shows how heat transfer behave in the fluid flow in rod bundle of square lattice, relation for liquid heavy metal was found in [38]:

$$Nu = 7.55(s) - 14(s)^{-5} + 0.007Pe^{0.64+0.246(s)} \quad (5)$$

while for hexagonal lattice is used relation from [39]:

$$Nu = 0.047(1 - e^{-3.8(s)})(Pe^{0.77} + 250) \quad (6)$$

and [27] for inner lead flow:

$$Nu = 5 + 0.025Pe^{0.8} \quad (7)$$

where Peclet number (Pe) is defined as advective transport rate to diffusive transport rate:

$$Pe = \frac{\nu D_H}{\alpha} = \frac{\nu D_H \rho_{pb} c_{pPb}}{k_{pb}} \quad (8)$$

2.3.2. Temperature distribution

The temperature distribution analysis in a fuel element was conducted in order to see if all temperatures satisfy safety limits. It is dependent on power generation inside the fuel material and flow of coolant.

The base for thermal analysis of temperature distribution in fuel element is Fourier's law i.e.:

$$\vec{q}'' = -k\nabla T \quad (9)$$

where:

q'' – heat flux [W m^{-2}],

k – constant thermal conductivity [$\text{W m}^{-1} \text{K}^{-1}$],

∇T – temperature gradient [K m^{-1}].

In case of 1-D steady state solution in cylindrical coordinate (temperature depends on r – cylinder radius) system the eq. (9) will take form of:

$$T(r) = -\frac{q'''r^2}{4k} + C_1 \ln r + C_2 \quad (10)$$

For solid fuel element the following boundary conditions are given:

$$\left. \frac{dT}{dr} \right|_{r=0} = 0 \quad \text{and} \quad T|_{r=R_{f,o}} = T_{f,o} \quad (11)$$

where:

$T_{f,o}$ – fuel outer surface temperature [K],

R_o – outer radius [m].

After applying boundary conditions (11) to eq. (10) 1-D steady state temperature distribution in fuel is:

$$T_f(r) = T_{f,o} + \frac{q'''R_{f,o}^2}{4k_f} \left(1 - \frac{r^2}{R_{f,o}^2} \right) \quad (12)$$

For annular fuel element the boundary conditions are taking form of:

$$\left. \frac{dT}{dr} \right|_{r=R_m} = 0 \quad \text{and} \quad T|_{r=R_{f,o}} = T_{f,o} \quad (13)$$

Thus for 1-D steady state temperature distribution in annular fuel will be [40]:

$$T_f(r) = T_{f,o} + \frac{q''' R_{f,o}^2}{4k_f} \left(1 - \frac{r^2}{R_{f,o}^2} \right) + \frac{q''' R_m^2}{2k_f} \ln \frac{r}{R_{f,o}} \quad (14)$$

where:

R_m – radius of the maximum fuel temperature (mean fuel radius, since the fuel rod is designed to have the same temperature in internal and external channel) [m].

For gas gap and cladding temperature distribution there is no heat generation source and thus, in both cases the solution will take the following form:

$$\frac{1}{r} \frac{d}{dr} \left(rk \frac{dT}{dr} \right) = 0 \quad (15)$$

Therefore, solid fuel temperature distribution in gas gap and cladding is respectively:

$$\begin{aligned} T_g(r) &= T_{c,oi} + \frac{q'}{2k_g} \ln \frac{r}{R_{f,o}} \\ T_c(r) &= T_{c,oo} + \frac{q'}{2k_c} \ln \frac{r}{R_{c,oi}} \end{aligned} \quad (16)$$

In case of temperature at the interface between cladding and coolant the Newton's law of cooling is required:

$$q'' = h\Delta T \quad (17)$$

Therefore, for the “hot” rod the interface temperature will be:

$$T_{c,oo} = T_{pb} + \frac{q'}{2\pi r_{c,oo} h} \quad (18)$$

In case of annular fuel rod it is obliged to introduce new parameter γ_o and γ_i , which are outer and inner heat exchange surface area to the total heat exchange surface area ratio, respectively.

Thus, can be described as follows:

$$\begin{aligned}\gamma_o &= \frac{2\pi R_{f,o} L}{2\pi (R_{f,i} + R_{f,o}) L} = \frac{R_{f,o}}{R_{f,i} + R_{f,o}} \\ \gamma_i &= \frac{2\pi R_{f,i} L}{2\pi (R_{f,i} + R_{f,o}) L} = \frac{R_{f,i}}{R_{f,i} + R_{f,o}}\end{aligned}\quad (19)$$

Therefore, eq. (16) for annular fuel is described as:

$$\begin{aligned}T_{g,oo}(r) &= T_{C,oi} + \gamma_o \frac{q'}{2k_g} \ln \frac{r}{R_{f,o}} \text{ and } T_{g,io}(r) = T_{C,io} + \gamma_i \frac{q'}{2k_g} \ln \frac{r}{R_{g,i}} \\ T_{c,oo}(r) &= T_{C,oo} + \gamma_o \frac{q'}{2k_c} \ln \frac{r}{R_{c,o}} \text{ and } T_{c,io}(r) = T_{C,ii} + \gamma_i \frac{q'}{2k_c} \ln \frac{r}{R_{c,i}}\end{aligned}\quad (20)$$

The temperature of the interfaces for annular fuel will be similar as for eq. 18:

$$T_{c,oo} = T_{pb} + \gamma_o \frac{q'}{2\pi r_{c,oo} h} \quad \text{and} \quad T_{c,ii} = T_{pb} + \gamma_i \frac{q'}{2\pi r_{c,ii} h} \quad (21)$$

The predicted temperature distribution in solid fuel is shown in figure 5 and the prediction of temperature distribution in annular fuel is shown in figure 6. The maximum temperature in fuel element is estimated to be at the mean radius (R_m) of the fuel.

2.4. Pressure drop

The calculations of pressure drop are based on results of experimental investigation done by South Korean team [7]. They have conducted experiments on how different kind of spacer grids affects pressure drop in solid and annular fuel rod assemblies with use of water as medium. Their investigation considered two kinds of spacer grids: plain and twist-vane in case of annular fuel and three kinds: plain, split-vane and hybrid vane spacer grids in case of solid fuel rods for square lattice. Our evaluation will be conducted for plain spacer grid in both annular and solid fuel approaches.

The pressure drop in the inner channel can be determined according to the following relation:

$$\Delta P_{Ch,in} = f \frac{L_{Tot} \rho v^2}{2D_H} \quad (22)$$

where L_{Tot} is the total length of the fuel rod and friction coefficient f for inner flow is taken from McAdams correlation and depends on Reynolds number i.e. the inertial to viscous forces in the fluid flow [7]:

$$f = 0.184 \text{Re}^{-0.2} \quad (23)$$

for the fuel lattice the friction coefficient is:

$$\begin{cases} f = 0.177 \text{Re}^{-0.224}; & \text{for } s = 1.08 \\ f = 0.214 \text{Re}^{-0.215}; & \text{for } s = 1.35 \end{cases} \quad (24)$$

The expressions are for annular fuel and for solid fuel, respectively. The values were derived from experimental results using data regression process.

It is oblige to take into consideration the influence of spacer grids in case of pressure drop calculations. Therefore, the pressure loss coefficient at the spacer grid K_{SG} is necessary to be introduced. In article mentioned before the following relations were found:

$$\begin{cases} K_{SG} = 3.682 \text{Re}^{-0.172}; & \text{for } s = 1.08 \\ K_{SG} = 3.02 \text{Re}^{-0.0746}; & \text{for } s = 1.35 \end{cases} \quad (25)$$

Therefore, the pressure drop in the outer channels will take form of:

$$\Delta P_{Ch,out} = f \frac{L_{Tot} \rho v_{out}^2}{2D_H} + n K_{SG} \rho v_{out}^2 \quad (26)$$

where, n is number of spacer grids. Our investigation will be performed for $n = 7$ as the total length of the fuel element (L_{Tot}) is 2.5 m and the spacers are placed every 0.4 m.

In case of annular fuel rod bundle it is oblige to make sure that pressure in inner and outer channels are the same and the mass flow rates are balanced. Thus, following relations are needed:

$$\begin{cases} \Delta P_{Ch,in} - \Delta P_{Ch,out} = f \frac{L_{Tot} \rho v_{in}^2}{2D_H} - \left(f \frac{L_{Tot}}{2D_H} + n K_{SG} \right) \rho v_{out}^2 = 0 \\ N_{el} \dot{m}_{in} + \dot{m}_{out} = \dot{m} \end{cases} \quad (27)$$

where N_{el} and $\dot{m}$ are number of elements in fuel assembly and total mass flow rate respectively.

2.5. The design process

The main challenge with the design process is to make sure that pressure drop in inner and outer channels will be balanced in all of the models. Since the size of fuel assembly has to be the same in order to secure possibility for ease replacement of default elements, the only parameter that can vary is the size of the inner and outer cladding diameters together with pitch between fuel elements. The number of fuel elements will be smaller in annular fuel due to the fact that the place for inner channel is required, thus external radius must be greater.

All relations mentioned before have to take part in the design process, thus optimizing function must had been written. The change of the number of elements per assemble creates the dimensions for the fuel element.

3. Results and Discussion

The results were obtained for both cases i.e. with and without spacer grids inside the fuel assembly in order to better understand the phenomena occurring in annular fuel. In case of temperature distribution the calculations were performed for the “hot” rod in the lattice due to the fact that the temperature distribution in this rod is the highest, hence the chance of failure is the biggest. Moreover, the calculations were conducted in case of normal and maximum possible overpower conditions, which was estimated to be 10%. The core radial and axial peaking factor were taken from [24] i.e. radial peaking factor 1.14 and axial peaking factor 1.24 where neutronic calculations were performed for default solid fuel and square fuel lattice. Since the fuel assembly dimensions and coolant to fuel volume ratio are the same, the peaking factors are assumed to be valid for all of the designs including solid and annular fuel.

The pressure drop calculations were performed for whole fuel assemblies.

3.1. Assembly without spacers

In case of rod bundles without spacer grids the dimensions of models that were created can be found in table 2 and 3 for the square and hexagonal lattices, respectively. It can be noticed that the pitch to element diameter (s) is similar for annular fuel, for square lattice it is around 1.075, it is very similar to the value in Lee et al. [8] experiment which is 1.08, hence the correlations for pressure drop are valid. For hexagonal it is around 1.227.

Table 2 Input parameters for square lattice calculations

	$R_{c,oo}$ [cm]	$R_{c,oi}$ [cm]	$R_{f,o}$ [cm]	$R_{f,i}$ [cm]	$R_{c,io}$ [cm]	$R_{c,ii}$ [cm]	s [cm]
Solid_21	5.25	4.65	4.5	N/A	N/A	N/A	1.323
Annular_19	7.177	6.577	6.427	4.393	4.243	3.643	1.070
Annular_18	7.560	6.960	6.810	4.638	4.488	3.888	1.073
Annular_17	7.989	7.389	7.239	4.913	4.763	4.163	1.075
Annular_16	8.471	7.871	7.721	5.223	5.073	4.473	1.077
Annular_15	9.017	8.417	8.267	5.573	5.423	4.823	1.079

Table 3 Input parameters for triangular lattice calculations

	$R_{c,oo}$ [cm]	$R_{c,oi}$ [cm]	$R_{f,o}$ [cm]	$R_{f,i}$ [cm]	$R_{c,io}$ [cm]	$R_{c,ii}$ [cm]	s [cm]
Solid 169	5.3	4.7	4.55	N/A	N/A	N/A	1.462
Annular 127	7.311	6.711	6.561	4.218	4.068	3.468	1.222
Annular 91	8.562	7.962	7.812	4.970	4.820	4.220	1.231

The figure 7 plots the parameters that affects neutronics and thermalhydraulics performance for both square (left) and hexagonal (right) lattice. The annular assembly has lower amount of fuel than solid one due to the fact that the amount of cladding material is inversely proportional. Hence, the normalized amount of fuel to the default design is always lower than unity and it is decreasing with number of rods, while cladding material is above unity and increases. From nautronic point of view, greater number of fuel elements leads to higher parasitic absorption in claddings. Nevertheless, the cooling surface area is much greater i.e. in the worst case the cooling surface area is 2.4 times bigger, which is great advantage over solid fuel. It is shown that the cooling surface area is up to 3.08 and 3.57 times greater for square and hexagonal lattice, respectively.

In table 4 it is possible to see how the maximum power density and linear power density changes for different models in square lattices. As we know, the reactor power is constant but number of fuel elements is different in each design, all of the values are for the default power output of 1500MW_{th}. The ratio of maximum linear power density per fuel element is up to 96% higher for annular approach and is decreasing with number of fuel elements. Nevertheless, the power density ratio in the fuel behaves proportional to number of power rods. The highest value in comparison to the solid fuel is only 12% higher.

Table 4 Maximum power density and linear power density per fuel element in square lattice

	Total number of fuel elements	δ	β	Ratio between particular and default power density	Ratio between particular and default linear power density
Solid_21	71442	1.676	100%	100%	100%
Annular_19	58482	1.676	88.98%	112.39%	122.16%
Annular_18	52488	1.676	90.21%	110.85%	136.11%
Annular_17	46818	1.676	91.45%	109.35%	152.60%
Annular_16	41472	1.676	92.70%	107.88%	172.27%
Annular_15	36450	1.676	93.95%	106.44%	196.0%

In case of hexagonal lattice, the table 5 shows ratio of maximum power density per annular fuel element is up to 9% higher than for solid fuel approach but in case of linear power density ratio the value is up to 185% the value of default fuel. The behaviour of those parameters are similar to the behaviour in case of square lattice.

Table 5 Maximum power density and linear power density per fuel element in hexagonal lattice

	Total number of fuel elements	δ	β	Ratio between particular and default power density	Ratio between particular and default linear power density
Solid 169	72163	1.842	100%	100%	100%
Annular 127	54229	1.842	91.68%	109.07%	133.07%
Annular 91	38857	1.842	94.49%	105.83%	185.71%

3.1.1. Temperature distribution

The base design is characterized by the highest temperature distribution inside fuel element. Figure 8 shows the comparison between the standard – solid fuel – element and the innovative – annular – design for square lattice.

The annular fuel shows superior advantage in case of temperature distribution, the maximum temperature is over 700°C lower than in base fuel design. Worth to mention is the fact that the temperature is decreasing with increase of fuel elements even though temperature of the interface between cladding and coolant is constant i.e. 513°C. Maximum temperature of annular fuel can be found in lattice 15x15 - 786°C. The smallest is in the lattice 19x19 - 707°C, while the centreline temperature of solid element is up to 1473°C, which is over double the value. Figure 9 shows the temperature distribution in hexagonal fuel lattice. Again, the base design, solid element, is characterized by highest temperature.

The innovative fuel designs show about 700°C lower maximum temperature. Again, the coolant temperature is the same in every case but this time the distribution has smaller spread for annular fuel. The temperature decreases with number of fuel rods in assemble. Maximum temperature of annular fuel is in lattice 91 - 788°C, the smallest is in the lattice 127 - 725°C, while the temperature of solid element is up to 1461°C. Worth to mention is fact that the temperatures are slightly higher than in square lattices in case of annular approach, due to compactness of the geometry.

The overpower (110%) calculations shows that temperature distributions (dashed lines) are only slightly higher than in 100% power level case, nevertheless, higher rise in power output exceeds the limit for coolant velocity in square lattice. Hence, to keep the same temperature at the coolant to cladding interface and ensure the velocity limit, further power uprates over 10% are not recommended. Unfortunately, hexagonal lattice does not permit for any overpowering, if it would, the temperature distributions would behave similar to the square case (dashed lines).

3.1.2. Pressure drop

Pressure drop is another important property, which should be investigated due to the fact that required pump power depends on it. Figure 10 and 11 presents the pressure drop in different lattices.

The pressure drop in square and hexagonal lattices increases with number of fuel rods, due to smaller outer flow area, thus smaller hydraulic diameter. For both inner and outer flow, the value is higher than for the base design in this case. Interesting is fact that the pressure drop is lower in case of base hexagonal lattice while for annular elements the pressure drop seems to be lower in square lattice.

3.2. Assembly with spacers

In case of fuel assemblies with spacer grids calculations were performed in the same manner as for the assemblies without spacers. The number of elements per fuel assembly is the same as in previous case. Nevertheless, the element dimensions were adjusted in order to obtain balance between inner and outer pressure loss in the elements since the pressure drop will be higher due to presents of spacer grids.

In tables 6 and 7 it is shown that the dimensions of the fuel rods are slightly smaller than in case without spacers. Therefore, the results are going to differ from the previous case.

Table 6 Input parameters for square lattice calculations with spacers

	$R_{c,oo}$ [cm]	$R_{c,oi}$ [cm]	$R_{f,o}$ [cm]	$R_{f,i}$ [cm]	$R_{c,io}$ [cm]	$R_{c,ii}$ [cm]	s [cm]
Solid_21	5.25	4.65	4.5	N/A	N/A	N/A	1.323
Annular_19	7.018	6.418	6.268	4.128	3.978	3.378	1.095
Annular_18	7.381	6.781	6.631	4.340	4.190	3.590	1.099
Annular_17	7.785	7.185	7.035	4.575	4.425	3.825	1.103
Annular_16	8.239	7.639	7.489	4.838	4.688	4.088	1.107
Annular_15	8.751	8.151	8.001	5.132	4.982	4.382	1.112

Table 7 Input parameters for triangular lattice calculations with spacer

	$R_{c,oo}$ [cm]	$R_{c,oi}$ [cm]	$R_{f,o}$ [cm]	$R_{f,i}$ [cm]	$R_{c,io}$ [cm]	$R_{c,ii}$ [cm]	s [cm]
Solid 169	5.3	4.7	4.55	N/A	N/A	N/A	1.462
Annular 127	7.076	6.476	6.326	3.796	3.646	3.046	1.262
Annular 91	8.232	7.632	7.482	4.377	4.227	3.627	1.280

From the figure 12 it is possible to notice that in case of fuel assembly with spacers grid the cooling surface is slightly lower than without spacers (fig. 7). The assembly with the highest number of fuel rods has the cooling area 2.31 times higher than default design, while the highest

value is 2.92 and 3.35 times greater for square and hexagonal lattice, respectively. The rest parameters are negligible different than in case without spacers.

In table 8 it is possible to see how the maximum power density and linear power density changes for different models in square lattices at 100% power output. The ratio of linear power density per fuel element is the same as in previous case i.e. 196% of the value of solid fuel element. Nevertheless, the power density ratio is slightly different. The highest value in comparison to the solid fuel is only 11% higher, thus slightly lower than in case without spacers.

Table 8 Maximum power density and linear power density per fuel element in square lattice

	Total number of fuel elements	δ	β	Ratio between particular and default power density	Ratio between particular and default linear power density
Solid_21	71442	1.676	100%	100%	100%
Annular_19	58482	1.676	89.94%	111.19%	122.16%
Annular_18	52488	1.676	91.18%	109.67%	136.11%
Annular_17	46818	1.676	92.43%	108.19%	152.60%
Annular_16	41472	1.676	93.69%	106.74%	172.27%
Annular_15	36450	1.676	94.95%	105.32%	196%

In case of hexagonal lattice (table 9) linear power density ratio, again the value is up to 185% value of the solid fuel, while maximum power density ratio per fuel element slightly smaller i.e. 7% higher than for solid fuel approach. It is due to smaller dimensions of the fuel and higher value of β , which defines the amount of fuel in the assembly.

Table 9 Maximum power density and linear power density per fuel element in hexagonal lattice

	Total number of fuel elements	δ	β	Ratio between particular and default power density	Ratio between particular and default linear power density
Solid 169	72163	1.842	100%	100%	100%
Annular 127	54229	1.842	92.94%	107.60%	133.07%
Annular 91	38857	1.842	95.76%	104.43%	185.71%

3.2.1. Temperature distribution

The slightly different size of fuel elements has impact on temperature distribution inside the fuel rods. It can be found that the temperatures are slightly higher in case with spacers, nevertheless, the rise in temperature distribution is marginal.

Figure 13 presents results obtained for square lattices. The maximum temperature is over 700°C lower than base fuel design. Maximum temperature of annular fuel can be found in lattice 15x15 - 806°C, the smallest is in the lattice 19x19 - 715°C, while the temperature of solid element is up to 1472°C.

Figure 14 shows the temperature distribution in hexagonal fuel lattice. Again, the base design – solid element is characterized by highest temperature. The innovative fuel designs show over 700°C lower temperature distribution. Maximum temperature of annular fuel is in lattice of 91 fuel elements – 823°C, the smallest is in the lattice of 127 – 747°C, while the temperature in case of solid element is up to 1461°C. Again, annular fuel in hexagonal lattices are characterized by higher temperatures than in case of square lattice.

Temperature distribution of 110% power core (dashed lines), like in case without spacers, shows that the temperatures are slightly higher than default core. The maximum temperature difference in annular fuel is about 20°C, while for solid fuel is near 100°C than in nominal power core. Again, the hexagonal lattice does not allow any overpowering the reactor but potential results are shown (fig. 14 – dashed lines).

3.2.2. Pressure drop

The correlations between pressures drop in default and annular approaches are different. Figure 15 and 16 shows the pressure drop in square and hexagonal lattice in the outer channels. Sum of pressure drop across the outer channel and influence of spacer grids must be equal to the pressure drop in inner channel i.e. ΔP_{total} is equal to ΔP_{inner} . The pressure drop in square and hexagonal lattice increases with number of fuel rods in assembly. In case with spacer grids

only one square lattice approach design has higher pressure drop than the default one, which is very desirable.

It can be seen that share of pressure drop related to grid spacers decreases significantly to become almost half of the default one.

Table 10 Pressure loss characteristics in square lattice

	Delta P_{Ch}	Delta P_{K_SG}	Delta P_{total}	f_{in}	f_{out}	K_{SG}	$\dot{m}_{in}$	$\dot{m}_{out}$	$\dot{m}_{total}$
Solid_21	47.215	84.840	132.055	N/A	0.0181	0.8974	N/A	1	128371
Annular_19	78.929	60.375	139.304	0.0147	0.0200	0.5460	0.325	0.675	128371
Annular_18	70.146	57.794	127.940	0.0146	0.0198	0.5409	0.327	0.673	128371
Annular_17	62.049	55.270	117.318	0.0144	0.0196	0.5356	0.329	0.671	128371
Annular_16	54.594	52.797	107.391	0.0142	0.0193	0.5299	0.332	0.668	128371
Annular_15	47.745	50.369	98.114	0.0140	0.0191	0.5240	0.334	0.666	128371

Pressure drop in hexagonal lattice is always higher for annular design as it is shown in the figure 16. Further decrease of fuel elements would decrease the pressure drop, but it would also increase temperature inside the fuel element, thus it is not convenient, since the temperatures are already higher than in case of square lattice. Hence, the hexagonal lattice possesses much less desirable properties and is less likely for further evaluation.

Table 11 Pressure loss characteristics in hexagonal lattice

	Delta P_{Ch}	Delta P_{K_SG}	Delta P_{total}	f_{in}	f_{out}	K_{SG}	$\dot{m}_{in}$	$\dot{m}_{out}$	$\dot{m}_{total}$
Solid 169	31.973	65.942	97.915	N/A	0.0176	0.8987	N/A	1	128371
Annular 127	105.827	122.522	228.350	0.0123	0.0196	0.4762	0.301	0.699	128371
Annular 91	75.684	109.671	185.354	0.0119	0.0189	0.4620	0.306	0.694	128371

Such behaviour can be explained with the fact that the pressure drop in inner channels has to be in balanced with the pressure drop in the outer channels, thus the sum of pressure drop related to the friction factor and spacer grids. Therefore, in order to obtain higher pressure drop in the inner channel the hydraulic diameter must be smaller than in case without spacers, due to the fact that pressure drop is inversely proportional to the inner radius. Smaller number of fuel elements in assembly would lead to increase of the inner rod hydraulic diameter. Unfortunately, the temperature distribution would increase with this process.

The main contributor to the pressure drop in case of annular flow is length of the fuel rod, thus friction acts on much bigger area due to greater size of fuel elements and decreases rapidly with number of fuel rods. While, the effect of grid spacers is smaller thus, decreases much slower.

Table 10 and 11 shows the characteristics of the pressure loss in both square and hexagonal lattices. The total mass flux in the core is constant while the share between inner and outer channels varies. The mass flux in the outer channels is proportional with number of fuel elements, thus negatively correlated with inner channels. On the other hand, all friction coefficients i.e. in inner channel, outer channel and at the spacer grid always are correlated with number of elements.

Characteristic of the pressure drop is similar to the one reported by M. S. Kazimi [2], thus suffices to validate the model.

4. Conclusions

The calculation results present analytical solutions of the default – solid and innovative – annular fuel approach. Temperature distribution in the “hot” fuel rod including cladding, gas gap and fuel together with pressure drop across fuel channels was obtained. Seven new designs were prepared and estimated during the calculation process. For a given operating conditions and assumptions, it was found that maximum temperature decreases with number of annular elements. The annular fuel is superior in case of maximum fuel temperature, which is up to 757°C lower than in base design – with use of spacer grids.

Unfortunately, it was found that pressure drop is proportional with number of fuel elements, thus the optimal design has not yet been found and must be a trade-off between temperature distribution and pressure drop. Further literature review and optimization is needed in order to create the most convenient design. Fortunately, the pressure drop, in case of spacer grids, is usually smaller than the default design. At the moment the most interesting designs are

hexagonal lattice with 91 fuel elements and square lattice 18x18 the maximum temperature are 822°C and 732°C, while pressure drop of 185kPa and 128kPa, respectively. In case of hexagonal lattice the pressure drop is higher than in default design while square lattice is more promising due to the similar or lower pressure drop i.e. 132kPa and the use of the same or smaller pump would be possible.

The preliminary results shows the annular fuel concept has a high potential to increase both performance and safety. It is due to superior cooling ability and in case of square lattice, the pressure drop can be decreased if needed. Moreover square lattice allows to uprate of power output up to 110% of nominal power, while ensuring safety requirements i.e. temperature and coolant velocity criteria. Therefore, it is worth further development and evaluation. Hexagonal lattice on the other hand is characterized by higher coolant velocity and limits power uprates.

References:

1. Kazimi, M.S., et al., *High performance Fuel Design for Next generation PWRs*. 2001: NERI proposal.
2. Kazimi, M.S., *High Performance Fuel Design for Next Generation PWRs: Final Report*. 2006, Cambridge, CA, USA.
3. Jiyun Zhao, H.C.N., Mujid S. Kazimi, *Mechanical Analysis of High Power Internally Cooled Annular Fuel*. Nuclear Technology 2004. **146**(2): p. 164-180.
4. Memmott, M., J. Buongiorno, and P. Hejzlar, *An Evaluation of the Annular Fuel and Bottle Shaped Fuel Concepts for Sodium Fast Reactors*. Nuclear Technology, 2011. **173**(2): p. 162-175.
5. Han, K.H. and S.H. Chang, *Development of a thermal-hydraulic analysis code for annular fuel assemblies*. Nuclear Engineering and Design, 2003. **226**(3): p. 267-275.
6. Han, K.-H., et al., *Development of a thermal hydraulic analysis code for gas-cooled reactors with annular fuels*. Nuclear Engineering and Design, 2006. **236**(2): p. 164-178.
7. Lee, C.Y., C.H. Shin, and W.K. In, *Pressure drop in dual-cooled annular and cylindrical solid fuel assemblies for pressurized water reactor*. Nuclear Engineering and Design, 2012. **250**(0): p. 287-293.
8. Lee, C.Y., et al., *An experimental investigation on turbulent flow mixing in a simulated 3x3 dual-cooled annular fuel bundle using particle image velocimetry*. Nuclear Engineering and Design, 2013. **260**(0): p. 134-144.
9. Shin, C.-H., et al., *Thermal hydraulic performance assessment of dual-cooled annular nuclear fuel for OPR-1000*. Nuclear Engineering and Design, 2012. **243**(0): p. 291-300.

10. Tak, N.-I., et al., *Thermo-fluid investigation on a double-side-cooled annular fuel for the prismatic very high temperature gas-cooled reactor*. Nuclear Engineering and Design, 2008. **238**(10): p. 2821-2827.
11. Zhang, S., et al., *Numerical investigation on heat transfer of supercritical fluid in a vertical 7-rod bundle*. The Journal of Supercritical Fluids, 2014. **92**(0): p. 8-15.
12. Dobashi, K., Y. Oka, and S. Koshizuka, *Core and plant design of the power reactor cooled and moderated by supercritical light water with single tube water rods*. Annals of Nuclear Energy, 1997. **24**(16): p. 1281-1300.
13. Squarer, D., et al., *High performance light water reactor*. Nuclear Engineering and Design, 2003. **221**(1-3): p. 167-180.
14. Reiss, T., et al., *The Simplified Supercritical Water-Cooled Reactor (SSCWR), a new SCWR design*. Progress in Nuclear Energy, 2010. **52**(2): p. 177-189.
15. Ueda, N., et al., *Sodium cooled small fast long-life reactor "4S"*. Progress in Nuclear Energy, 2005. **47**(1-4): p. 222-230.
16. Ishii, K., H. Matsumiya, and N. Handa, *Activities for 4S USNRC licensing*. Progress in Nuclear Energy, 2011. **53**(7): p. 831-834.
17. Ichimiya, M., *The Status of Generation IV Sodium-Cooled Fast Reactor Technology Development and its Future Project*. Energy Procedia, 2011. **7**(0): p. 79-87.
18. Smith, C.F., et al., *SSTAR: The US lead-cooled fast reactor (LFR)*. Journal of Nuclear Materials, 2008. **376**(3): p. 255-259.
19. Alemberti, A., et al., *European lead fast reactor—ELSY*. Nuclear Engineering and Design, 2011. **241**(9): p. 3470-3480.
20. Aït Abderrahim, H., et al., *MYRRHA – A multi-purpose fast spectrum research reactor*. Energy Conversion and Management, 2012. **63**(0): p. 4-10.
21. Yican Wu, et al. *Conceptual Design of China Lead Alloy - based Zero Power Reactor CLEAR - 0*. in ICENES 2013. Madrid, Spain.
22. Zhang, Z., et al., *Design aspects of the Chinese modular high-temperature gas-cooled reactor HTR-PM*. Nuclear Engineering and Design, 2006. **236**(5-6): p. 485-490.
23. Kiryushin, A.I., et al., *Project of the GT-MHR high-temperature helium reactor with gas turbine*. Nuclear Engineering and Design, 1997. **173**(1-3): p. 119-129.
24. Bandini, G., P. Meloni, and M. Polidori, *Thermal-hydraulics analyses of ELSY lead fast reactor with open square core option*. Nuclear Engineering and Design, 2011. **241**(4): p. 1165-1171.
25. Chandra, L. and F. Roelofs, *Inter fuel-assembly thermal-hydraulics for the ELSY square open reactor core design*. Nuclear Engineering and Design, 2010. **240**(10): p. 3009-3019.
26. Bortot, S., et al., *Preliminary core characterization of a Generation IV lead fast reactor DEMO: Goals, design rationales and options*. Energy Conversion and Management, 2010. **51**(9): p. 1806-1812.
27. *LOW PRANDTL NUMBER THERMAL-HYDRAULICS*, in *Handbook of Lead-bismuth Eutectic Alloy and Lead Properties*, R. Stieglitz, Editor. 2007.
28. Tulenko, J.S., *Nuclear Reactor Materials and Fuels*, in *Nuclear Energy*. 2013, Springer: USA. p. 207.
29. ASME, *ASME BOILER AND PRESSURE VESSEL CODE AN INTERNATIONAL CODE PART D (METRIC)*. 2004, ASME: USA.
30. Alemberti, A., et al., *Overview of lead-cooled fast reactor activities*. Progress in Nuclear Energy, (0).
31. Cinotti, L., et al., *Lead-cooled system design and challenges in the frame of Generation IV International Forum*. Journal of Nuclear Materials, 2011. **415**(3): p. 245-253.

32. G.I. Toshinsky, O.G.K., K.G. Mel'nikov, *Nuclear power technologies at the stage of sustainable nuclear power development*. Progress in Nuclear Energy, 2011(53): p. 6.
33. Mitenkov, F.M. and V.I. Polunichev, *Small nuclear heat and power co-generation stations and water desalination complexes on the basis of marine reactor plants*. Nuclear Engineering and Design, 1997. **173**(1–3): p. 183-191.
34. Zhang, J. and N. Li, *Review of the studies on fundamental issues in LBE corrosion*. Journal of Nuclear Materials, 2008. **373**(1–3): p. 351-377.
35. Müller, G., et al., *Behavior of steels in flowing liquid PbBi eutectic alloy at 420–600 °C after 4000–7200 h*. Journal of Nuclear Materials, 2004. **335**(2): p. 163-168.
36. Banerjee, S. and T.R. Govindan Kutty, *10 - Nuclear Fuels*, in *Functional Materials*, S.B.K. Tyagi, Editor. 2012, Elsevier: London. p. 387-466.
37. Zinkle, S.J. and G.S. Was, *Materials challenges in nuclear energy*. Acta Materialia, 2013. **61**(3): p. 735-758.
38. Colombo, M., et al., *Transfer function modelling of the Lead-cooled Fast Reactor (LFR) dynamics*. Progress in Nuclear Energy, 2010. **52**(8): p. 715-729.
39. Mikityuk, K., *Heat transfer to liquid metal: Review of data and correlations for tube bundles*. Nuclear Engineering and Design, 2009. **239**(4): p. 680-687.
40. Sathiyasheela, T. and P. Mohanakrishnan, *Annular fuel pin heat transfer and lumped model correction*. Progress in Nuclear Energy, 2010. **52**(5): p. 487-490.

Figure captions

Fig. 1 Comparison of default solid fuel (left) and proposed annular fuel (right)

Fig. 2 Scheme of default assembly lattices a) square b) hexagonal [19]

Fig. 3 Schemes of proposed assembly lattices a) square b) hexagonal

Fig. 4 Dimensions of the fuel elements

Fig. 5 Predicted temperature distribution in solid fuel element

Fig. 6 Predicted temperature distribution in annular fuel

Fig. 7 Key parameters normalized to the default designs

Fig. 8 Temperature distribution in various square lattices

Fig. 9 Temperature distribution in various hexagonal lattices

Fig. 10 Pressure drop in square lattice

Fig. 11 Pressure drop in hexagonal lattice

Fig. 12 Key parameters normalized to the default design (with spacers)

Fig. 13 Temperature distribution in various square lattices (with spacers)

Fig. 14 Temperature distribution in various hexagonal lattices (with spacers)

Fig. 15 Pressure drop in square lattice in the outer channels

Fig. 16 Pressure drop in triangular lattice in the outer channels

Table captions

Table 1 Main ELSY reactor's parameter [19]

Table 2 Input parameters for square lattice calculations

Table 3 Input parameters for triangular lattice calculations

Table 4 Maximum power density and linear power density per fuel element in square lattice

Table 5 Maximum power density and linear power density per fuel element in hexagonal lattice

Table 6 Input parameters for square lattice calculations with spacers

Table 7 Input parameters for triangular lattice calculations with spacer

Table 8 Maximum power density and linear power density per fuel element in square lattice

Table 9 Maximum power density and linear power density per fuel element in hexagonal lattice

Table 10 Pressure loss characteristics in square lattice

Table 11 Pressure loss characteristics in hexagonal lattice