2017 International Nuclear Atlantic Conference - INAC 2017
Belo Horizonte, MG, Brazil, October 22-27, 2017
ASSOCIAÇÃO BRASILEIRA DE ENERGIA NUCLEAR – ABEN

Agência Brasileira do ISBN
ISBN 978-85-99141-07-6
9 788599 141076

THERMAL HYDRAULIC CORE SIMULATION OF THE MYRRHA REACTOR IN STEADY STATE OPERATION

Gustavo H. N. Ferandes, Mário C. Ramos, Athos M. S. S. Carvalho, Carlos E. V. Cabrera, Antonella L. Costa, Claubia Pereira

Departamento de Engenharia Nuclear - Universidade Federal de Minas Gerais
Av. Antônio Carlos, 6627. Campus Pampulha
31.270-901, Belo Horizonte, MG

Instituto Nacional de Ciência e Tecnologia de Reatores Nucleares Inovadores/CNPq

ghnfernandes@gmail.com, marc5663@gmail.com, athos1495@yahoo.com.br, carlosvelcab@hotmail.com, antonella@nuclear.ufmg.br, claubia@nuclear.ufmg.br

ABSTRACT

MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) is a prototype nuclear subcritical reactor driven by a particle accelerator. As a special property, the reactor maintains the nuclear fission chain reaction by means of an external neutron source provided by a particle accelerator. The main aim of this work is to study two types of coolants, LBE (Lead-Bismuth Eutectic) and Na (Sodium) that are two strong candidates to be used in ADS systems as well as in *Generation IV (GEN-IV)* reactors. Firstly, it was developed a thermal hydraulic model of the MYRRHA core using the RELAP5-3D, considering LBE as coolant (original project). After this, the LBE was substituted by Na coolant to investigate the reactor behavior in such case. Results have demonstrated the high heat transfer capacity of the LBE coolant in this type of system.

1. INTRODUCTION

The Accelerator Driven Systems - ADS is a subcritical reactor which the main proposal is to transmute and regenerate nuclear burned fuel from traditional nuclear plants, generally LWR – Light Water Reactors plants, whose destination would be the final deposits. Basically, an ADS can be considered as an integrated system of three devices, being: a high energy particle accelerator, a target material capable of producing neutrons through *spallation* reactions, and the core of a reactor operating in subcritical mode.

ADS systems have had a resurgence in the last decades as it is possible to verify in the literature [1, 2, 3, 4]. The ADS is currently under development, directly or indirectly, by many countries around the world such as Belgium, Italy, the United States of America, Russia, Japan, France, Sweden, China, among others [5].

Probably the most important concept has been developed in the Belgian Nuclear Research Centre -SCK•CEN, it is the MYRRHA (Multi-purpose hYbrid Research Reactor for Hightech Applications). It is being developed since 1998 and it is expected to be completely operational until 2023. MYRRHA will be able to work either in subcritical (ADS) or in critical mode, and its fundamental concept involves fuel transmutations. MYRRHA started with the proposal to substitute the Belgian Material Testing Reactor (MTR) BR2 that has been working since 1962 [6]. When completely operational (around 2023) it is expected that

it will be able to support fuel developments, specially the transmutations of high-level nuclear waste.

Abderrahim et al. (2012) [6] describes the main MYRRHA proposes as: "*a) to demonstrate ADS feasibility; b) high level nuclear waste burned up fuel transmutation; c) fuel developments; d) materials developments for Gen IV reactors; e) medicinal radioisotopes production*"; etc. Since the start of the project, initiated under ADONIS name (1995-1997), MYRRHA original design and concept has already changed several times and finally a full design concept seems to be accepted. It is the MYRRHA-FASTEF, under the *Seventh Framework Programme of the European Commission – FP7* [7]. Its conception provides the creation of a flexible reactor.

MYRRHA is being projected to use Lead-Bismuth Eutectic (LBE) as primary coolant, a Heavy Liquid Metal (HML) [8], under investigation since the beginning of 1950s. The LBE has good physical and chemical characteristics for being used as coolant in breeder reactors, which the main aims are: high boiling point, relatively low melting point and thermal stability. However, LBE also demonstrates significant corrosion problems associated. Other liquid metals to be used as coolant have been investigated; the Sodium has also good aspects as: low melting point, thermal and radiation stability, good heat transfer coefficient, among others.

The main aim of this work is to present the thermal hydraulic behavior of the MYRRHA core simulated using the RELAP5-3D code. Available data from the core design, fuel characteristics and two types of coolants, LBE and Na, were used in the simulations.

2. MYRRHA-FASTEF – MAIN CHARACTERISTICS

MYRRHA is a pool type reactor projected to use LBE as coolant in the primary heat exchange system. It will have a linear proton beam accelerator of 600 MeV and 4 mA proton in critical mode coupled into the core. The spallation target will be LBE in solid state. In Fig. 1, the main components of this reactor are shown. In Fig. 2 are presented the main fuel pin and full assembly aspects.

MYRRHA is being projected to reach a power around 80 MW in sub-critical mode and 100 MW in critical one. To achieve this, some vacancies (37 positions) inside the main core were placed to possible changing mode of operation. It will give the reactor a large flexibility making possible many types of experiments which is the main proposal of this reactor [9]. In the Fig. 3, a typical core design of MYRRHA is presented.

Table 1 describes some of the main design parameters for the MYRRHA-FASTEF. The primary coolant system will operate with LBE in a velocity inside core of ~ 2.0 m/s. The total coolant mass flow rate is expected to be around 9,440 kg/s to remove heat of until 110 MW_{th} of generated power by the core and other components. As it does not pretend to generate electrical energy, the heat is dissipated in two more heat exchange systems, the secondary will use saturated water/steam as coolant and the tertiary will use atmosphere air as coolant. MYRRHA will use as fuel a Mixed Oxide Fuel – MOX, in a non-conventional proportion of PuO_2 (varying from 30% to 35wt%) from spent fuel [10]. In Table 2 are presented some core aspects.

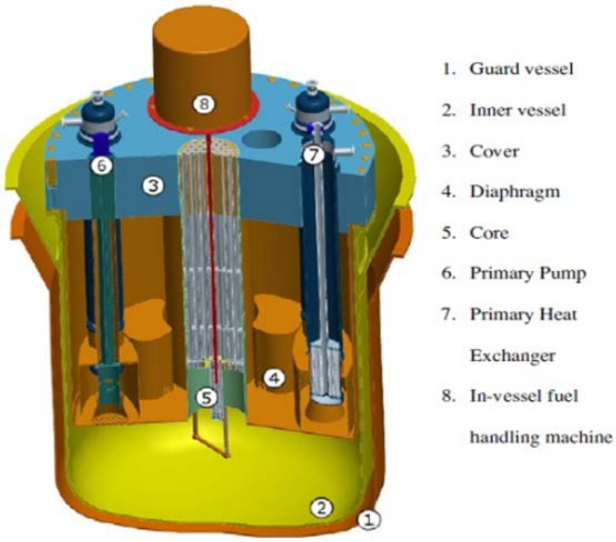

Figure 1: MYRRHA-FASTEF main components [6].

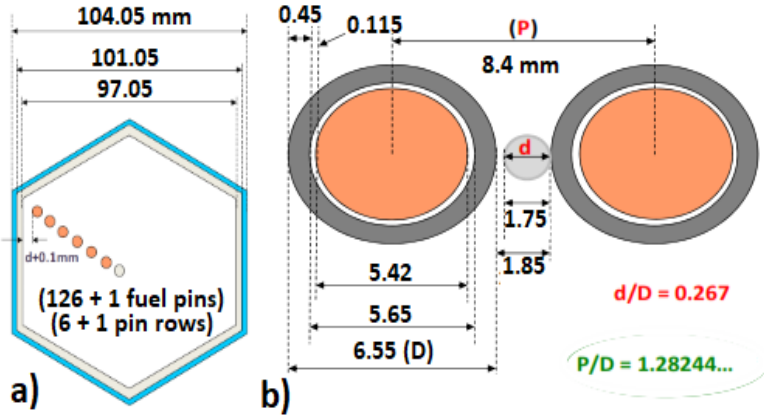

Figure 2: MYRRHA: a) fuel assembly radial view; b) fuel pin geometry [8].

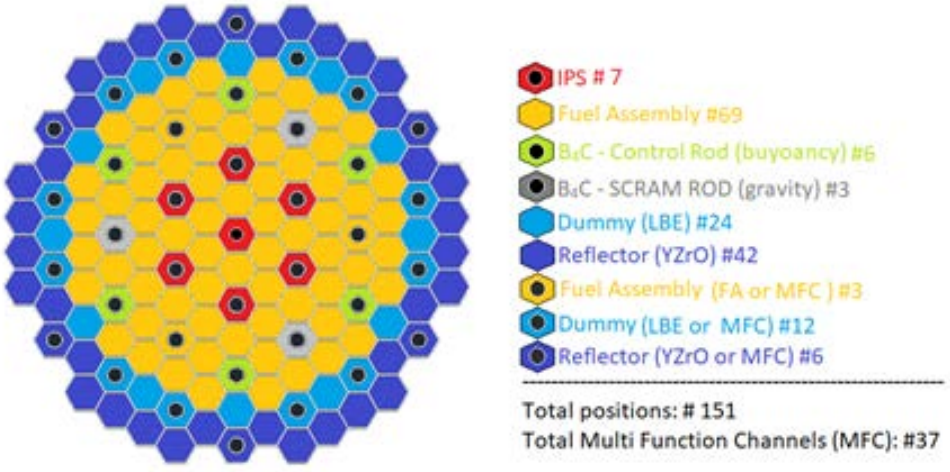

Figure 3: MYRRHA core design for critical mode operation (100 MW_{th}).

Table 1: MYRRHA - main reactor parameters [8]

Core external diameter	1,450 mm
Core height / FA length	2,000 mm
Fueled length	600 mm
Vessel diameter (ID)	7,600 mm
Vessel cover thickness	2,000 mm
Vessel length	1,1000 mm
Nominal Core Power	100 MW_{th}
Maximum core power	110 MW_{th}
Core inlet temperature	543 K
Core average outlet temperature	683 K
Coolant velocity in core	2 m/s
Coolant pressure drop	0.25 MPa
Secondary coolant	Saturated water/steam
Tertiary coolant	Air
Cold shutdown state temperature	200 °C
Reactor vessel materials	AISI 316L
Reactor cover material	AISI 316L and concrete

Table 2: Main core parameters [9]

Fuel Type	MOX, 30wt%PuO_2 from spent fuel
Fuel assembly	
Number of fuel pins	127
Cladding material	T91
Spacer type	Wire spacer in 15-15 Ti
Core	
Number of positions	151
Diameter	Around 1,500 mm
Layout	Centralized around central position

3. HEAVY METAL LIQUID - HML

In Table 3 are shown some physical-chemical characteristics for two main coolants candidates to be used in ADS.

In Table 4, the HMLs are compared in according with: (a) low vapor pressure at operating temperatures; (b) high boiling point; (c) low melting point; (d) excellent heat-transfer properties; (e) radiation stability; (f) thermal stability; (g) low thermal neutron capture cross-section; (h) short term induced radioactivity; (i) compatibility with reactor and heat transfer loop materials; (j) low cost; (k) low pumping power; (l) non-toxic and free from operating hazards; (m) non-reactive with air and water [10].

Table 3: Thermophysical properties of Sodium and LBE coolants [11]

Physical Properties	LBE	Sodium (Na)
Density (for 300°C)	10,300 kg/m^3	880 kg/m^3
Melting point	123.5 °C	97.8 °C
Boiling point	892°C	1,670 °C
Thermal Conductivity (for ~300°C)	11 W/mK	76 W/mK
Heat capacity (for ~300°C)	1.50 MJ/m^{3}K	1.14 MJ/m^{3}K

Table 4: Selection criteria and assessment of two liquid metals [10]

HML	Meets criterion	Partially meets criterion	Does not meet criterion
Sodium	(c), (d), (e), (f), (h), (k), (l)	(a), (g), (i)	(b), (j), (m)
LBE	(a), (b), (d), (e), (f), (g), (m)	(i)	(h), (j), (k), (l)

4. MYRRHA CORE - RELAP5-3D MODEL

The RELAP5-3D code was developed by Idaho National Laboratory - INL. This code includes several functions that allow analysis of safety, design, operation, modification, etc. These analyses are calculated based on three pillars: balance equations, transfer and heat conduction equations and neutron equations [12].

The nodalization (Fig. 4) was performed considering five Thermal Hydraulic Channels (THCs) divided axially in 24 volumes; 5 Heat Structures (HS) divided into 24 axial active volumes parts. Table 5 presents the main parameters used to perform the model in the RELAP5 code. In the Table 5, the acronyms are: IPS (In-Pile Section), CR (Control Rod), SR (Safety Rod).

Figure 5 shows how the core was divided in relation to the flow area corresponding to each channel of the Fig. 4. Five 5 rings in the core were considered to perform the modeling. The coolant flow area of each ring corresponds to the flow area of the THCs shown in Figure 4.

As MYRRHA system is still in development, many data are not available in the literature. Because of this, it not could be possible to take all information from the same source. The data used to perform the model are shown in Table 5.

In the Figure 6 is shown the modeled heat struture used to simulate the reactor heat source. Point kinetics model was used in the calculations.

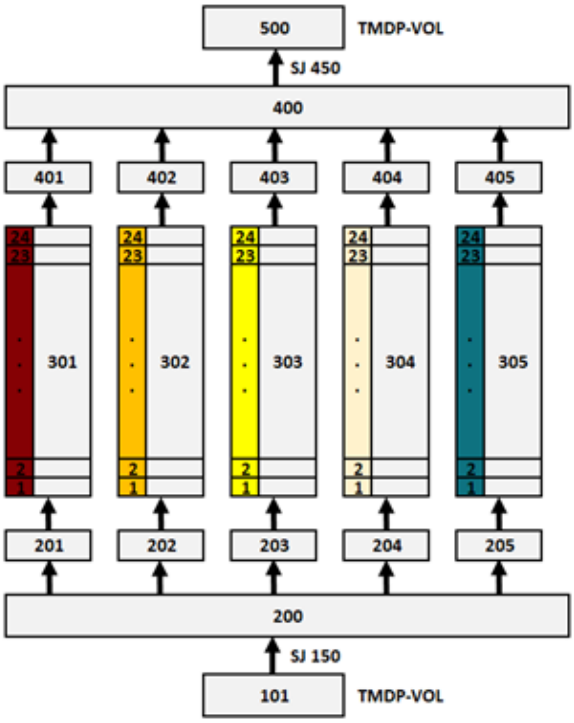

Figure 4: RELAP5-3D model used for the core reactor simulation.

Table 5: MYRRHA core parameters used for RELAP5 modeling

Reactor Power	100 MW_{th}
Main core parameters	
Total length [18]	2,000 mm
Active length$^{(considered)}$ [18]	600 mm
Number of pins $^{(per\ fuel\ assembly)}$ [18]	127
Fuel assembly area [18]	9,375.94 mm^2
Total pins area [18]	4,279.33 mm^2
Total coolant area $^{(fuel\ assembly,\ LBE\ Dummies)}$ [18]	4,410.38 mm^2
Total coolant area $^{(IPS,\ CR,\ SR)}$ [18]	1,285.26 mm^2
Total coolant area $^{(Outer\ dummies)}$ [18]	2,934.39 mm^2
Coolant flow rate $^{(fuel\ assembly,\ LBE\ Dummies)}$ [16]	75.15 kg/s
Coolant flow rate $^{(IPS,\ CR,\ SR)}$ [16]	21.90 kg/s
Coolant flow rate $^{(Outer\ dummies)}$ [16]	50.00 kg/s
Inlet coolant pressure	1.5 MPa
Pellet MOX radius [18]	2.710 mm
GAP radius [18]	2.820 mm
Clad radius [18]	3.275 mm
Inlet temperature for all THCs [7, 18]	273°C
Coolant mass flow THC 1	1,055.10 kg/s
Coolant mass flow THC 2	1,352.70 kg/s
Coolant mass flow THC 3	1,324,35 kg/s
Coolant mass flow THC 4	1,803.60 kg/s
Coolant mass flow THC 5	3,903.60 kg/s

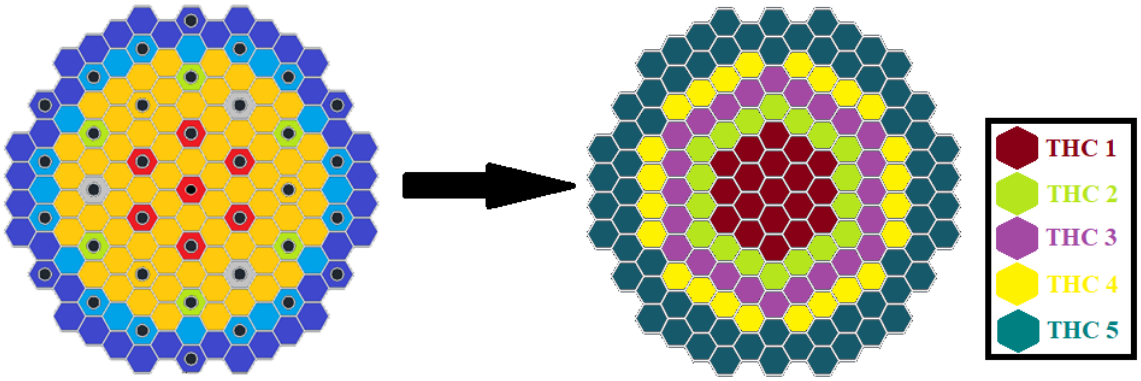

Figure 5: Division of the MYRRHA core to model the 5 THCs.

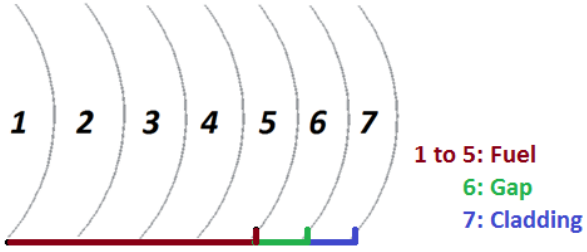

Figure 6: Heat structure intervals.

The fuel used in MYRRHA will be a MOX type with a non-conventional percentage of PuO_2 (around 30%). The thermal conductivity of the MOX type fuels could be calculated using the expressions (1), (2) and (3) [13].

$$k_{MOX} = K_{UO_2}\left(1 - a.P_{PuR}^{\frac{2}{3}}.\left(1 - \frac{1}{1+\frac{1}{a}.P_{PuR}^{\frac{1}{3}}.\left(\frac{K_{UO_2}}{K_{PuR}}-1\right)}\right)\right); \quad \textbf{(1)}$$

where:

k_{MOX} = thermal conductivity of heterogenous MOX fuel;
k_{UO2} = thermal conductivity of UO_2 and PuO_2 in solid solution;
k_{PuR} = thermal conductivity of Pu rich particles;
P_{PuR} = volumetric fraction of Pu rich particles;
a = ansiotropy factor; [a = 1, isotropic pore distribution].

$$k_{UO_2} = \frac{1}{0.1149+4.015x10^{-3}xBU+(2.475x10^{-4}-6.982x10^{-7}xBU)xTC} + 0.0132xe^{0.00188xTC}; \quad \textbf{(2)}$$

$$k_{PuR} = \left(\frac{1}{1.528\sqrt{x+0.00931}-0.1055+0.44xBU+2.2885x10^{-4}xTK} + 76.38x10^{-12}xTK^3\right)xf(p); \quad \textbf{(3)}$$

where:
f(p) = correction factor for porosity; [f(p)=1, no porosity];

BU = burnup in MWd/kgHM [BU=0, initial fuel state];
x = stoichiometry [x = 1];
TK= temperature in K;
TC= temperature in °C.

MOX heat capacity calculation is a function of temperature, UO_2 and PuO_2 fractions, oxygen and metal atoms ratio O/M and fuel burnup. It is given by the expression, valid for a range between 300 ~ 1,400 K [14]:

$$FC_p = \frac{K_1.\theta^2.e^{\left(\frac{\theta}{T}\right)}}{T^2.\left(e^{\left(\frac{\theta}{T}\right)}-1\right)^2} + K_2.T + \frac{Y.K_3.E_D}{2.R.T^2}.e^{\left(\frac{-E_D}{R.T}\right)}; \quad \textbf{(4)}$$

where:
FC_p = Specific heat capacity [J/kg.K];
T = temperature [K];
Y = oxygen to metal ratio;
R = universal gas constant = 8.3143 [J/mol.K];
Θ = the Einstein temperature [K], typical for each type of fuel;
E_D = energy per amount of material [J/mol].

Since that O/M and burnup effects over heat capacity are very small, they can be neglected [15]. Others constants are given in Table 6 [14].

Table 6: Constants for UO_2 and PuO_2 heat capacity expression (4) [14]

Constant	UO_2	PuO_2	Units
K_1	296.70	347.40	J/kgK
K_2	$2.43x10^{-2}$	$3.95x10^{-4}$	J/kgK^2
K_3	8.75×10^7	3.86×10^7	J/kg
Θ	535.29	571.00	K
E_D	1.58×10^5	1.97×10^5	J/mol

MOX specific heat capacity can be found by an average considering the PuO_2 weight proportion. Then, heat capacity [J/m^3K] can be calculated by a simple multiplication of Eq. (4) x (5) or (6), depending of temperature range. MOX density expression is given below [15]:

$$\rho_s(T) = \rho_s(273)x\left(\left((9{,}973x10^{-1}) + (9{,}802x10^{-6})x(T - 2{,}705x10^{-10})xT^2 + 4{,}391x10^{-13} * T^3\right)^{-3}\right); \quad (5)^*$$

* Valid from 273 to 973 K;

$$\rho_s(T) = \rho_s(273)x\left(\left((9{,}967x10^{-1}) + (1{,}179x10^{-5})x(T - 2{,}429x10^{-9})xT^2 + 1{,}219x10^{-12} * T^3\right)^{-3}\right); \quad (6)^{**}$$

** Valid up to 973 K;

where:

$\rho_{s(UO2)}(273) = 10{,}970\ kg/m^3$;
$\rho_{s(PuO2)}(273) = 11{,}600\ kg/m^3$.

Density for MOX (U1-yPuy) depends of mole fraction of PuO_2 and it is calculated by the linear expression (7) [15]:

$$\rho_{s(UO2)}(273) = 10{,}970 + 490y. \quad (7)$$

In Table 7, it is possible to verify the calculated thermal proprieties for MOX (30%w, PuO_2) that was used in the RELAP5 input data.

Table 7: Thermal properties for MOX (30%w, PuO2)

Temperature [K]	Thermal conductivity [W/mK]	Heat capacity [J/m^{3}K]
273	8.5854	2.59E+06
350	6.7141	2.73E+06
500	5.3287	2.92E+06
650	4.4228	3.23E+06
800	3.7858	3.37E+06
950	3.3155	3.44E+06
1,100	2.9567	3.49E+06
1,250	2.6771	3.54E+06
1,400	2.4571	3.57E+06
1,550	2.2844	3.60E+06
1,700	2.1515	3.63E+06
1,850	2.0544	3.65E+06
2,000	1.9913	3.68E+06
2,150	1.9630	3.71E+06
2,300	1.9722	3.74E+06
2,450	2.0238	3.76E+06
2,600	2.1252	3.79E+06
2,750	2.2874	3.82E+06
2,900	2.5255	3.85E+06
3,050	2.8609	3.89E+06
3,200	3.3226	3.92E+06

The reactor core simulations using the RELAP5 code were performed varying the coolants, LBE or Sodium, whose thermal characteristics were listed in Table 3. The results are being presented in section 5.

Agência Brasileira do ISBN
ISBN 978-85-99141-07-6
9 788599 141076

5. STEADY STATE MYRRHA SIMULATION RESULTS

In Table 8 is presented a comparison for results obtained for temperatures and pressures for both simulations using LBE and Na coolants.

Table 8: Simulations results for MYRRHA core using RELAP5

Parameters	Using LBE coolant	Using Sodium coolant
Core inlet temperature [°C]	273.0	273.0
THC 1 outlet temperature [°C]	393.0	304.0
THC 2 outlet temperature [°C]	385.0	301.0
THC 3 outlet temperature [°C]	358.0	292.0
THC 4 outlet temperature [°C]	349.0	289.0
THC 5 outlet temperature [°C]	275.0	274.0
Core average channel outlet temperature (upper plenum) [°C]	352.0	292.0
Maximum fuel temperature [°C]	813.0 Axial 13. Mesh 1, HS1	686.0 Axial 13. Mesh 1, HS1
Maximum gap temperature [°C]	481.0 Axial 16. Mesh 6, HS1	372.0 Axial 14. Mesh 6, HS1
Maximum cladding temperature [°C]	427.0 Axial 17. Mesh 7, HS1	314.0 Axial 17. Mesh 7, HS1
Core Inlet pressure [MPa]	1.5797	1.4863
Core Outlet pressure [MPa]	1.3703	1.4639

The hottest channel was the central channel, THC1 (see Figure 5). The average outlet temperature calculated in THC 1 was 393.0 °C for a thermal power of 100 MW_{th}. Castelliti (2013) using RELAP5-3D found a value of 400 °C for the outlet coolant channel temperature at 110 MW_{th}. Note that there are differences, just as number of channels, axial and radial volumes, etc., between the used models, but these values can be considered as reference data [18].

The project proposes an average core outlet coolant temperature of about 350 °C for a thermal power of 100 MW_{th} [18]. In this work, it was found 352°C using a total mass flow rate of 9,440 kg/s and a thermal power of 100 MW_{th}. Therefore, the calculation agrees very well with the reference data.

The total pressure drop found in this work was ~0,21 MPa and the value considered in project is ~0,25 MPa. Here the difference found can be easily associated with simplifications adopted for this work, just as no material roughness used, that influences the friction and the flow resistance.

About Sodium results, no comparison works were found. However, the core outlet average temperature was much lower (292.0 °C) than in the case of use of the LBE (358 °C), that is, ΔT of 79°C and 19 °C, for Na and LBE, respectively. The low outlet sSdium temperature needs be studied with more details, because it was not expected.

In the Fig. 7 are presented the coolant and heat structure temperatures behavior, for LBE (left side) and Na (right side).

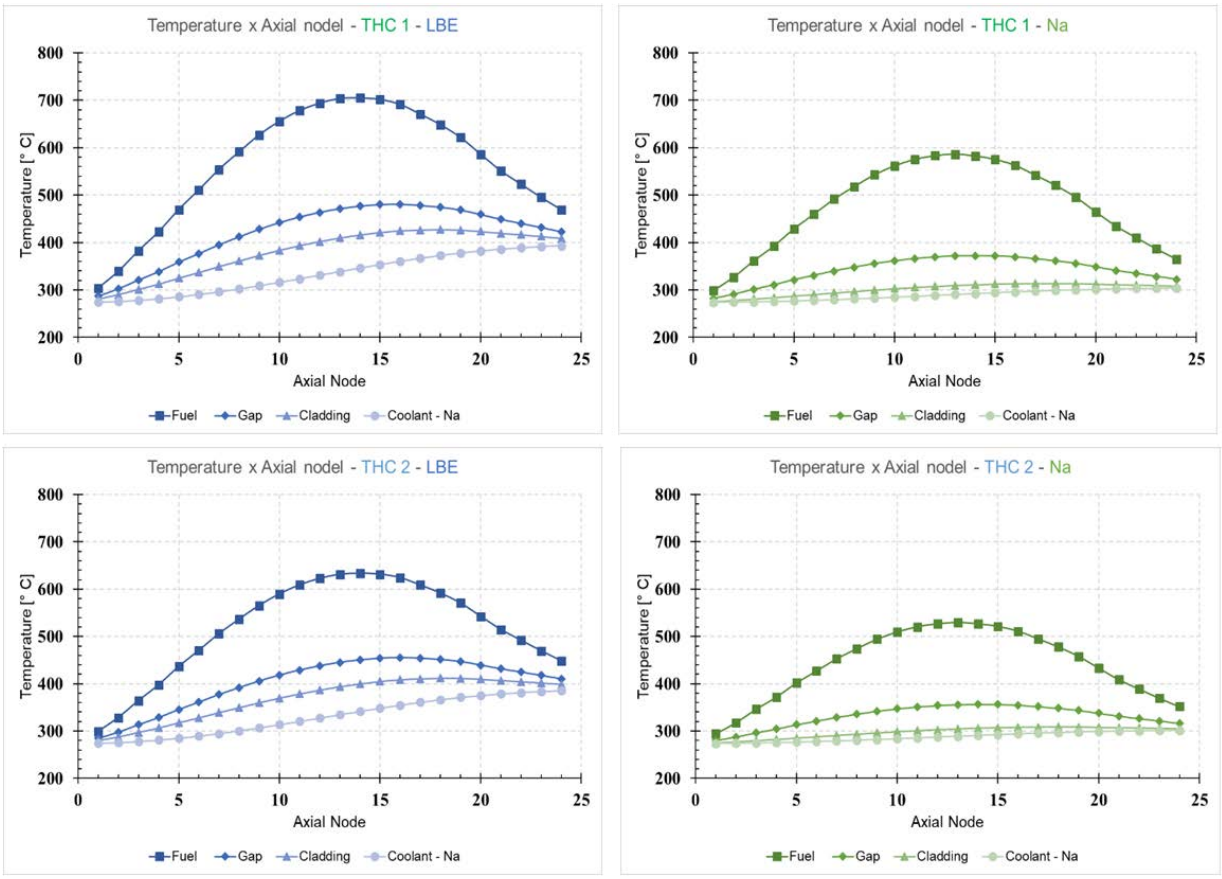

Figure 7: Temperature behavior along the HS and THC (1 and 2) for: LBE (left) and Na (right).

The temperature has a cosine profile following the relative power distribution. The maximum temperature found was 876.0 °C for axial node 13, HS 1, mesh 1 (fuel center), using LBE as coolant. In Sodium case, it was found 736.0 °C, for axial node 13, HS 1, mesh 1 (fuel center).

In the Figure 8 is shown the heat structure temperature behavior for three axial levels, for models using LBE (left) and Na (right) as coolants for the hottest THC 1, respectively, in the bottom (axial 3), middle (axial 13) and upper (axial 22) positions, taking into account the radial temperature distribution, starting from the fuel center until the cladding.

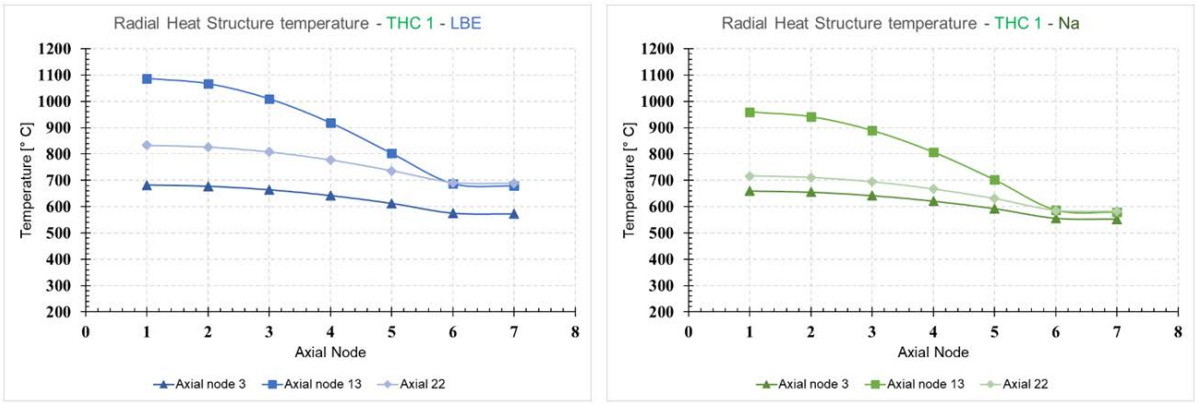

Figure 8: Radial temperature behavior for HS 1 for: LBE (left) and Na (right).

In the Fig. 9 is shown a thermo-graphic for temperature (°C) behavior in the hottest channel, HS 1, for both cases, LBE and Na as coolants. Again, the low temperature found along the radial HS in the case of use of Na as coolant needs further studies.

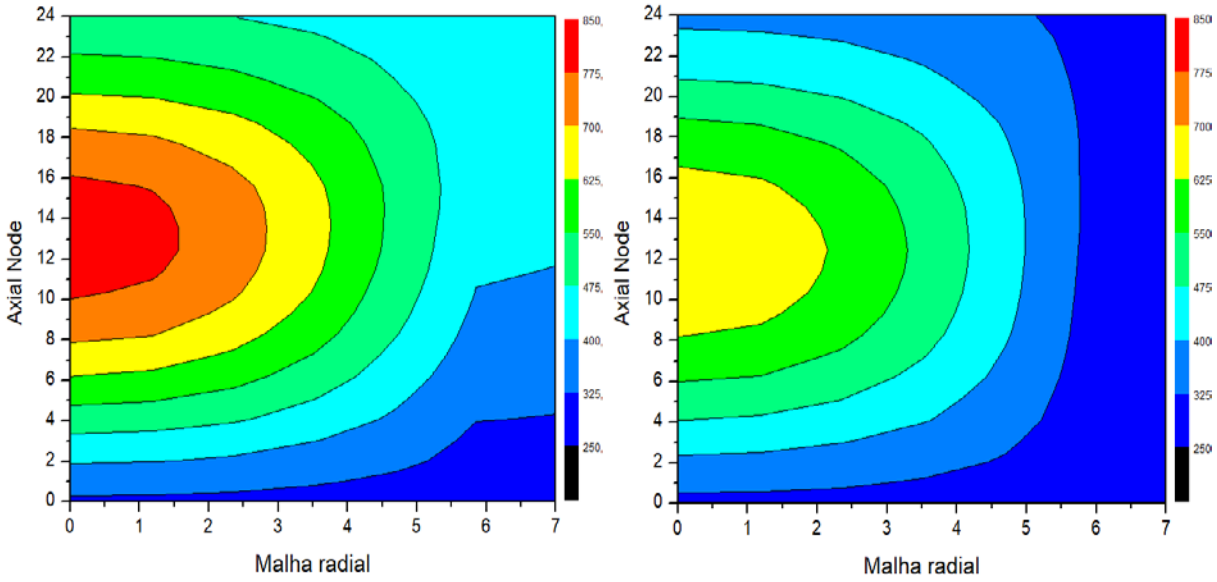

Figure 9: Temperature distribution in hottest channel (THC 1) for: LBE (left) and Na (right).

6. CONCLUSIONS

This is a work in progress. Core simulations of MYRRHA reactor have been performed in this work using the RELAP5-3D code. Five THCs were considered to simulate the reactor core with LBE as coolant. Simulations of the core using Sodium as coolant were also performed, since Sodium is a possible coolant candidate to several future reactors. For the developed model using the LBE the results were in according with the reference data. However, when the coolant was changed by Sodium, all the temperatures were very low. Therefore, the model using the Sodium will be investigated with more details to find an

explanation for such behaviour. One possible sensitivity test will be change the Sodium flow rate decreasing its value to verify the behavior of heat to coolant.

ACKNOWLEDGMENTS

The authors are grateful to CAPES (*Coordenação de Aperfeiçoamento de Pessoal de Nível Superior)*, FAPEMIG (*Fundação de Amparo à Pesquisa do Estado de Minas Gerais*) and CNPq (*Conselho Nacional de Desenvolvimento Científico e Tecnológico*) for the support. Thanks also to Idaho National Laboratory for the RELAP5-3D license.

REFERENCES

1. Furukawa, K. et al., "The combined system of accelerator molten salt breeder (AMSB) and molten salt converter reactor (MSCR)", *Japan–US Seminar on Thorium Fuel Reactors* (1982).
2. Bowman, C. D. et al. "Nuclear energy generation and waste transmutation using an accelerator-driven intense thermal neutron source". *Nuclear Inst. and Methods in Physics Research*, A, **v. 320,** n. 1–2, p. 336–367 (1992).
3. Carminiati, F. et al. "An energy amplifier for cleaner and inexhautable nuclear energy production driven by a particle beam accelerator". *European organisation for nuclear research*, **v. 47,** n. November, p. 1–72 (1993).
4. Nifenecker, H. et al. "Basics of accelerator driven subcritical reactors". *Nuclear Instruments And Methods In Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment*, **v. 463,** n. 3, p.428-467 (2001).
5. GIF. Generation IV international forum. "GIF 015 Annual Report". *Oecd Nuclear Energy Agency*, 141 p (2015).
6. Abderrahim, H. A. et al., "MYRRHA – A multi-purpose fast spectrum research reactor". *Energy Conversion and Management*, [s.l.], **v. 63,** p.4-10, (2012).
7. Sarotto, M. et al. "The MYRRHA-FASTEF cores design for critical and sub-critical operational modes (EU FP7 Central Design Team project)". *Nuclear Engineering And Design*, **v. 265,** p.184-200, (2013).
8. "MULTI-PURPOSE hYbrid Research Reactor for High-tech Applications". *Unkown: Sck*, **34 slides**, (2017)
9. De Bruyn, D. et. al., "The fast-spectrum transmutation experimental facility FASTEF: Main design achievements (Part 1: Core and primary system) within the FP7-CDT collaborative project of the European Commission". *United States: American Nuclear Society – ANS*, (2012)
10. Zhang, J., "Lead-Bismuth Eutectic (LBE): A Coolant Candidate for Gen. IV Advanced Nuclear Reactor Concepts". *Advanced Engineering Materials*, **v. 16,** n. 4, p.349-356, (2013).
11. Barros, G. P., "Estudo inicial de um ADS para a produção de 233U utilizando o código MCNPX". 2009. 53 f. Dissertação. *Programa de Pós-graduação em Ciências e Técnicas Nucleares, Departamento de Engenharia Nuclear, Universidade Federal de Minas Gerais*, **53 f.** (2009).
12. The RELAP5-3D© Code Development Team, "RELAP5-3D© Code Manuals", INEEL-EXT-98-00834, *Idaho National Laboratory,* (2009).
13. Lee, B., Koo, Y., Sohn, D., "Modelling of MOX fuel's thermal conductivity considering its microstructural heterogeneity", *International Atomic Energy Agency (IAEA)*, IAEA-TECDOC—1233**,** (2001).

14. Hagrman, D. T. et al. “MATPR0-A Library of Materials Properties for Light-Water-Reactor Accident Analysis”. *Idaho Falls: Idaho National Engineering Laboratory*, **667p.,** (1995)
15. C., J. J. “Thermophysical Properties of MOX and UO2 Fuels Including the Effects of Irradiation”; *University of North Texas Libraries*, (1995).
16. Koloszar, L. et al. “MyrrhaFoam: A CFD model for the study of the thermal hydraulic behavior of MYRRHA”. *Nuclear Engineering And Design*, **v. 312,** p.256-265, (2017).
17. Heusdains, S., Arien, B., Abderrahim, H. A. "Simulation of the primary system of MYRRHA with the RELAP code", *Proceedings of the International Workshop on P&T and ADS Development*, (2003).
18. Castelliti, D. Comparison of MYRRHA RELAP5 mod 3.3 and RELAP5-3D models on steady state and PLOF transient; *Inl – Idaho Falls*, **34 slides**, color, (2013).