Fuel design for the experimental ADS MYRRHA

H. Aït Abderrahim, V. Sobolev*, E Malambu

Belgian nuclear research Centre SCK•CEN
Boeretang 200, B-2400 Mol, Belgium

ABSTRACT

Since 1997, the Belgian nuclear research Centre SCK•CEN, in partnership with IBA S.A and many European research laboratories, has being designed a multipurpose experimental Accelerator Driven System (ADS) called MYRRHA. This ADS is aiming to serve as a basis for the European experimental facility to study feasibility of MA transmutation in fast spectrum. In a first stage, the project mainly focuses on demonstration of the ADS concept, safety research on sub-critical systems and nuclear waste transmutation studies. In a later stage, it will also be applied for research of structural materials, nuclear fuel, liquid metal technology. Subsequently, it will be used as fast spectrum irradiation facility and as radioisotope production facility. One of the most important parts of the MYRRHA project is the design of sub-critical core, fuel assembly and fuel rod. The main results obtained in this part of the project are presented in this article. The first estimates concerning the replacement of MOX by fuel based on the low-enriched uranium are given at the end.

1. Introduction

The European Technical Working Group) on ADS concluded in April 2001 in its report "A European Roadmap for developing Accelerator Driven Systems (ADS) for Waste Incineration" [1] that the partitioning and transmutation (P&T) in association with the ADS and in combination with the geological disposal can lead to an acceptable solution for the nuclear waste management problem. It concluded that a strong support in this field from the European Commission and from the national programmes is needed to develop and build an experimental demo ADS in Europe. Since 1997 the Belgian nuclear research Centre SCK•CEN in partnership with IBA S.A. and many European research laboratories, have being designed a multipurpose ADS for R&D applications called MYRRHA (Multipurpose hYbrid Research Reactor for High-tech Applications) [2,3]. The associated R&D support programme is in progress. Conceptually, MYRRHA ADS consists of a proton accelerator delivering a 350 MeV proton beam with a current of about 5 mA to a liquid lead-bismuth eutectic (LBE) spallation target, and of the subcritical fast core fuelled with highly enriched MOX that is also cooled by LBE. One of the most important parts of the MYRRHA project is the design of the sub-critical core, fuel assembly and fuel rod. In this paper, the main results related to this topic are presented.

* Corresponding author: tel.: +32 14 33 22 67; fax: +32 14 32 15 29; e-mail : *vsobolev@sckcen.be*

2. MYRRHA ADS

The design of the reactor part of the MYRRHA ADS is based on a pool type configuration with a standing vessel (Figure 1) [2,3]. The pool design has been favoured because it avoids the penetration from beneath of the spallation target circuit into the main vessel and thus enhances the safety of the design. It allows also having an internal interim storage easing the fuel handling. The natural circulation for the residual heat removal in case of loss of flow (LOF) and loss-of-heat-sink (LOHS) is demonstrated to be feasible, particularly with the large thermal inertia that is also an argument in favour of this design. With an emergency cooling system based on natural circulation both on primary and secondary sides, a long-term core coolability can be ensured even in situation of complete power loss.

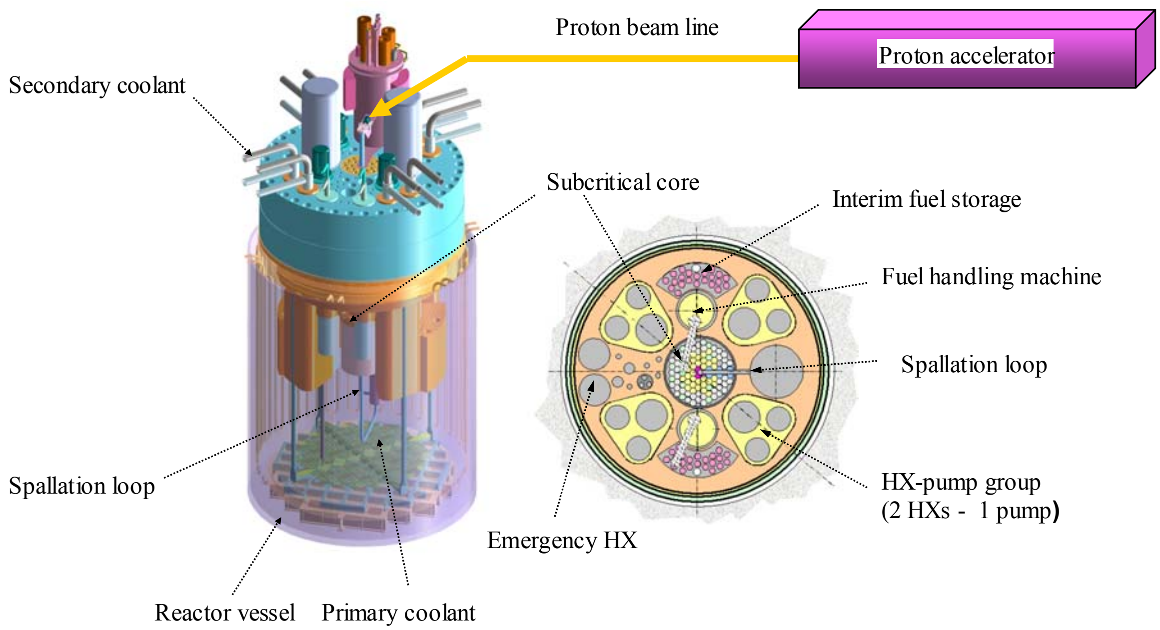

Figure 1. Schematic view of ADS MYRRHA.

The core pool contains a fast-spectrum sub-critical core loaded with typical fast reactor hexagonal assemblies cooled by LBE. The three central hexagons are left free for housing the spallation module. The spallation target circuit is fully immersed in the reactor pool and thermally interlinked with the core, but its liquid metal is physically separated from the core coolant. The core structure will be mounted on a central support column coming from the lid and being stabilised by the diaphragm separating the cold and hot parts of the LBE coolant. Since access from the top is very restricted and components introduced into the pool will be buoyant due to the high density of the LBE, the loading and unloading of fuel assemblies is foreseen to be carried out by force feed-back controlled robots in remote handling from underneath. The pool will also contain the liquid metal primary pumps, the heat exchangers and the two fuel handling robots based on the well known rotating plug technology of fast reactors.

The present MYRRHA ADS concept is driven by the flexibility and the versatility and aims at the following applications:

- ADS concept demonstration.
- Safety studies for ADS.
- Minor actinide transmutation studies.
- Long-life fission products transmutation studies.
- Medical radioisotopes production.
- Material research.
- Fuel research.

In the MYRRHA predesign, the project team has favoured as much as possible the mature or less demanding technologies (in terms of the needed R&D).

3. Design parameters of the core

3.1. Choice of fuel

Taking into account a large experience in the production of plutonium-uranium mixed oxide fuel (MOX) existing in Europe, this type of fuel was selected as the main candidate already at the beginning of the MYRRHA project. Moreover, MOX has better neutronic properties than uranium dioxide in a fast neutron spectrum [4,5].

Aiming at a compact core with high power density (high neutron flux) and based on the information cited above, MOX with a maximum enrichment of 30 wt. % RG Pu has been preliminary chosen as the driver fuel in the pre-design of the MYRRHA sub-critical core. However, this choice should still be checked against the non-proliferation requirements imposed to new test reactors by the RERTR (Reduced Enrichment of fuel for Research Testing Reactors) program launched by DOE US in 1996 and supported, in general, by the EU, Russian Federation and IAEA. This program aims at enhancement of the non-proliferation of fission materials and security in the nuclear research and testing reactors that use highly enriched uranium (HEU - usually containing 90-96 wt. % of isotope ^{235}U). It requires the conversion of these reactors to low enriched uranium (LEU - containing less than 20 wt. % of ^{235}U) [6]. A preliminary analysis of this problem, regarding the fuel alternatives for MYRRHA, is performed and other types of fuel are under study aiming at finding a backup solution and long term perspectives (U-Mo, Pu-U-Zr, Pu-U-Mo, …).

3.2. Cladding material

The choice of the fuel cladding material is of critical importance both from economic and safety viewpoints. The cladding material should exhibit:

a) reproducible fabrication, workability and weldability;
b) compatibility with LBE coolant;
c) mechanical resistance: sufficient strength with limited degradation of ductility and fracture toughness under irradiation in LBE;
d) dimensional stability: resistance to irradiation-induced swelling and/or creep;
e) heat resistance: limited decrease of strength and toughness with temperature.

Such operational conditions are very challenging for a material to be selected from the existing materials for nuclear applications. Ferritic-martensitic steels (FMS) appear to be promising candidates both for the fuel cladding and the structures [7]. At this stage, FMS T91 has been chosen as the main candidate for the fuel cladding in MYRRHA. This choice was based on the fact that T91 shows a lower swelling rate and embrittlement under irradiation at T > 350 °C, and higher resistance to dissolution in the oxygen-free LBE, compared to austenitic stainless steels (ASS). Moreover, this group of FMS is considered as a candidate for a GEN-IV reactor cooled by liquid lead and for fusion reactors. Important research programs have already started for development and studies of these FMS.

However, taking into account that all FMS suffer stronger from irradiation embrittlement below 350 °C and show higher corrosion rate in presence of oxygen than ASS, the well-known steels AISI 316 L and modified 316Ti were selected as a backup solution. These steels have already demonstrated their good performances in LMFBR where they were used as the fuel cladding material. The most serious limitation on their use is associated with the irradiation induced swelling and with the helium induced embrittlement at high irradiation doses (at low temperatures severe embrittlement can appear at the doses > 50 dpa). The modified 316Ti steel shows a better radiation resistance and its swelling at operation temperatures can be tolerated up to the radiation damage doses of 120 dpa [8] .

The available experimental data on LBE technology and corrosion resistance of different steels in contact with LBE indicate that their long-term operation (> 10 000 h) is possible only at temperatures lower than 560 °C [9]. Therefore, the allowable maximum local temperature of 450 °C has been chosen for the LBE coolant normal operation. (It could be higher during a limited period in DBC transients). The minimum coolant temperature has a natural limit of ~ 125 °C due to the LBE melting. In order to have a technological margin the minimum LBE operation temperature of 200 °C has been chosen at this stage of the pre-design. However, one should keep in mind that embrittlement problems related to joint effects of LBE and neutron damage on the cladding materials can force us to increase this temperature up to 250-350 °C. The LBE velocity limit of 2 m/s has been fixed because of a possible erosion of structures by heavy metal coolant during a long-term operation at higher velocities.

3.3. Fuel rod design

The standard MOX fuel almost mandatory requires choosing a traditional geometry of a fuel pin: a rod with a cylindrical cladding loaded with cylindrical fuel pellets. Given the maximum heat power deposited in unit volume of pellet (called "fuel power density") and the pellet thermal conductivity, the maximum allowed pellet diameter is limited by the fuel melting temperature. The higher fission rate, the thinner pellets should be used [10].

The upper limit of the power density in the fuel can approximately be estimated using the maximum aimed neutron flux and the neutron cross-sections of the considered MOX in the typical neutron spectrum of LBE-cooled ADS. The first estimates show that the fuel power density of ~ 1.5 kW/cm^3 should be envisaged in the MYRRHA core to obtain the fast neutron flux of ~ $1 \cdot 10^{15}$ n cm^{-2}s^{-1} at the beginning of the fuel life (*BOL*).

The melting temperature of MOX decreases with the PuO_2 content and with the fuel burnup [10]. The "fresh" MOX with 30%Pu in HM has the liquidus temperature of 2730 °C. At aimed burnup of 100 MWd/kg-iHM, this temperature will drop down to 2410 °C. The maximum design temperature of the fuel has to be below these limits. In order to keep the fuel enough away from a completely plastic state, the maximum operation temperature is usually chosen lower than about $0.9 \cdot T_{melt}\ (K)$.

A degradation of the MOX thermal conductivity with burnup is another factor limiting the allowed power density. In the expected operation temperature range of 500-2000 °C, the average thermal conductivity of the fresh MOX estimated with the Baron-Hervé correlation [11] is 2.25 W/m K, and it decreases to about 1.5 W/m K at burnup of 100 MWd/kg iHM.

After simplified calculations based on the above mentioned values, the fuel pellet diameter of 5.40 mm has been chosen (which is very close to that of LMFR PHENIX). A height of the fuel pellet is usually selected between one and two diameters [4, 10]. Shorter pellets are easier for fabrication and provide a bit more free space for fission products (due to a greater number of chamfers, dishes and interfaces); longer pellets allow to reduce the total number of pellets in the fuel column, that makes a bit cheaper the fuel pin production in an industrial scale. A value of h_{pellet} = 6.0 mm (close to the pellet diameter) has been chosen for MYRRHA taking into account its experimental character.

The cladding diameter and thickness are directly related to the pellet diameter. A conservative choice can be done assuming that the intrinsic thermal stresses in the clad itself and the stresses induced by the mechanical interaction between the pellet and the cladding don't exceed the elasticity limit [4,5]. In order to escape the pellet-cladding mechanical interaction, (at least at *BOL*), a gap is foreseen between them. If the gap is large enough, it can compensate both thermal and radiation induced pellet swelling. However, a larger gap will result in a higher fuel temperature. Optimisation of this parameter is an important item, which requires a modelling with thermomechanical fuel performance codes. In fuel rods of the well-known reactors, where oxide fuel pellets is helium-bounded with the stainless-steel clad, the gaps with the design (cold) thickness of *0.075-0.15 mm* are used, depending on pellets diameter [4, 10, 11]. The gap thickness of *0.075 mm* has been chosen as the first guess in the preliminary design of the MYRRHA fuel rod.

Thermal stresses will be built over the cladding because of temperature differences. Moreover, the coolant pressure will act from the outside of the cladding and the filling/fission gas pressure from the inside. The cladding must keep sufficient strength and ductility, in order to resist these stresses. Often, a less conservative requirement that the cladding plasticity must allow the plastic deformation not more than 0.4-1 % is used in pre-design stage for ASS claddings [4, 11]. The inner and outer cladding corrosion and the irradiation induced degradation of its mechanical properties must also be accounted for the needed cladding thickness. The maximum corrosion loss can be assumed to be of 10 % of the thickness during operation life of the fuel pin. So, the cladding with a thickness of $\delta_{clad} = 0.5$ mm can be chosen for the fuel pins of MYRRHA. It will keep pressure of 20 MPa even after the loss of 50 μm in the thickness.

The active fuel length of *600 mm* was selected for the MYRRHA subcritical core in the earlier neutronic calculations [12].

Yttria stabilised zirconia (YSZ) ceramics has been chosen as material for the reflector segments, accounting for its rather good neutronic and satisfactory thermal insulating properties. The length of two reflector segments (upper and below the fuel stock) has been chosen on the basis of analogy with the existing LMFR fuel, where 200-400 mm breeder segment are usually placed in the axial reflector zone. In experimental fuel rods without breeding material, the shorter reflector segments (50-100 mm) made of steel are often used. [10, 13]. The reflector segments of 100 mm were proposed for MYRRHA fuel, keeping in mind flexibility for possible modifications. These segments should still be optimised (may be shorter) after a more detailed neutronic analysis.

A gas plenum within the fuel rod has to withstand the total pressure of the filling gas and the fission gas products released from the fuel. Usually, the filling gas initial pressure of 1.1 to 10 bar is used in LMFR. The value of 5 bar was fixed after the preliminary modelling of the fuel rod behaviour with FEMAXI-V.1m code. The gaseous fission products (mainly Xe and Kr) is generated within MOX fuel of the theoretical density with the rate of about $\eta_{FG} = 11.5\ mol/m^3$ per *MWd/kg-iHM* [5]. Assuming that fuel in the hottest rod can reach peak burnup of 100 MWd/kg-iHM and that all gas can be released in the plenum and cooled down to 250 °C by the input flow of LBE, one can obtain the needed gas plenum length of 210 mm, provided the maximum allowed pressure is 20 MPa (the cladding diameter and thickness were fixed above). Under accidental conditions accompanied by a rapid increase of the fuel temperature (especially in the case of the fuel partial melting), the gas fission products can arrive into the plenum with a higher temperature. In this case the gas pressure at first moment will be higher. More sophisticated analysis of this situation has to be performed later. In order to be on a safe side, the plenum volume was increased by factor 1.5. Moreover, a space was added for a spring in the upper part and for a support element in the lower part. So the total length of the two-chamber gas plenum (a sum of the upper and lower parts) was fixed to be 360 mm.

A schematic view of the MYRRHA fuel rod with elements presented in a simplified manner is illustrated by Figure 2.

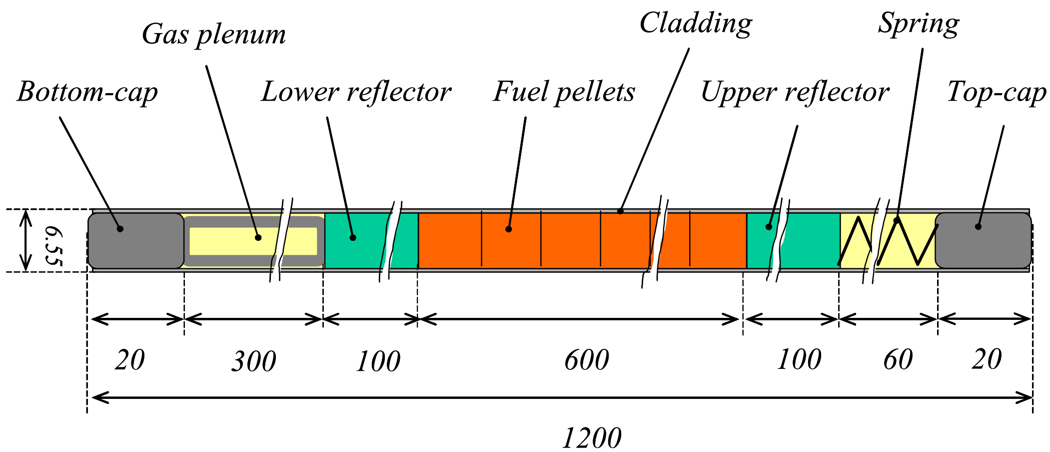

Figure 2. Axial schematics of the MYRRHA fuel rod.

3.4. Fuel assembly

At the first step of the pre-design of a fuel bundle, a type of the fuel rod grape and a distance between neighbour rods (pitch) should be chosen basing on the admissible values of the fuel volume fraction in the active zone, on the heat removal capacity of the coolant and on the acceptable coolant pressure drop on the core. Based on joint neutronic and thermohydraulic estimations, a dense triangular type fuel cell with the pitch of *8.55 mm* (distance between the centres of the neighbour rods) was fixed. The number of the fuel rods in a hexagonal bundle was chosen to be 91, in order to assure the needed flexibility in the core management and to limit the radial differences of temperature and neutron flux. The shroud wall thickness of 1.75 mm was fixed based on the previous thermo-mechanical estimations [14]. A typical LMFR assembly was used as a prototype for the axial assembly design with few adaptations to the LBE coolant [10,13].

The axial and radial schematic of the hexagonal fuel assembly of ADS MYRRHA are presented in Figure 3 below.

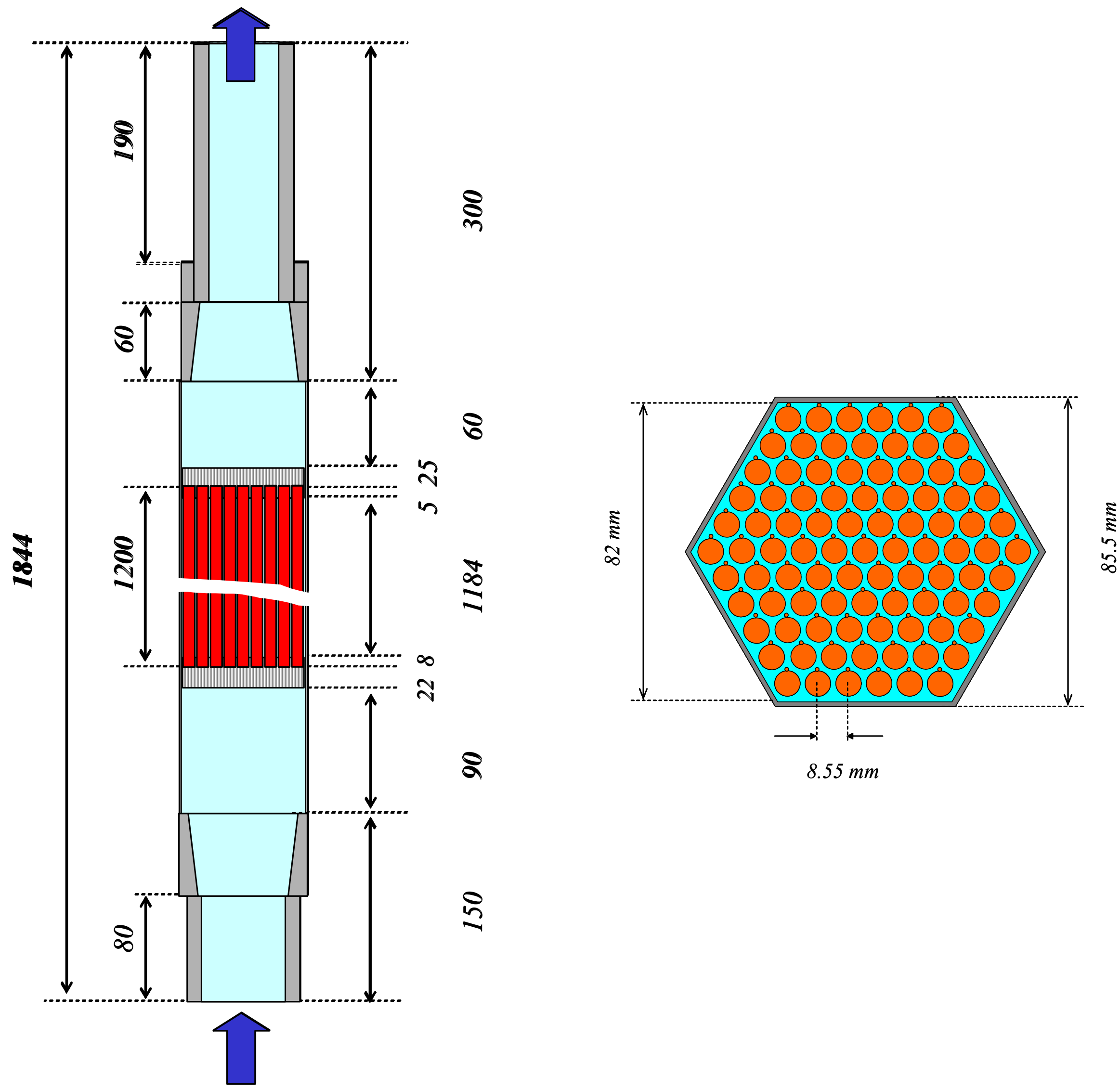

Figure 3. Axial and radial (at mid-plane) schematics of the MYRRHA fuel assembly.

3.5. Sub-critical core configuration

The neutronic configuration of the MYRRHA sub-critical core is similar to that of a classical LMFR. The core has 102 hexagonal positions with pitch ("centre-to-centre") of 87 mm (Figure 4). Three central positions are free and occupied by the spallation target. The target is surrounded by the active zone loaded with 45 (or more) fuel assemblies. In their turn, they are surrounded by the reflector zone. The maximum core radius is 1000 mm, the core height determined by the assembly length is 1844 mm and the active core height is 600 mm. A typical MYRRHA configuration with k_{eff} of 0.95 can be achieved by using 45 fuel assemblies with 30 % MOX. There are 19 positions accessible through the reactor lid capable to house experimental devices.

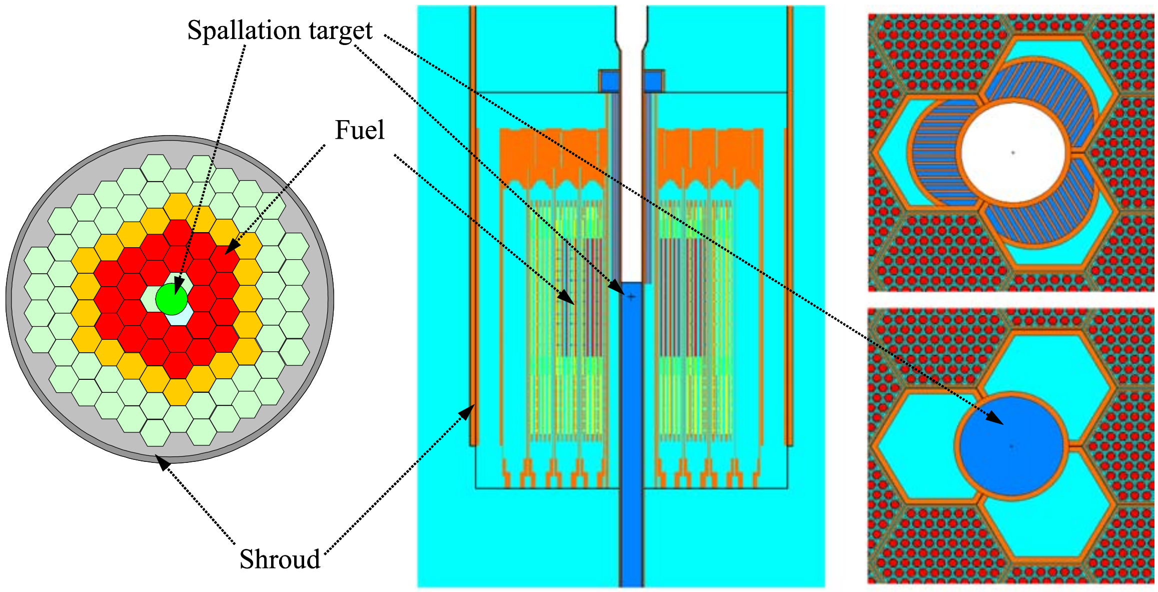

Figure 4: Typical configuration of the MYRRHA core.

The performances of an ADS in terms of flux and power levels are dictated by the spallation source strength, which is proportional to the proton beam current at a particular energy and the sub-criticality level of the core. The sub-criticality level of k_{eff} = 0.95 has been considered as an appropriate level for a first of kind medium-scale ADS. The reactivity worth of one fuel assembly is ranging between ~ 450 to 1600 pcm depending on the position.

Fixing the sub-criticality level and the desired neutron flux in the position of the irradiation location for minor actinide transmutation, determines the required strength of the neutron spallation source. In order to achieve the needed performances at a modest total power level of few tens of MW, one has to limit the central hole diameter. The maximum cylindric hole diameter of ~100 mm is available in the MYRRHA core.

The calculated performances in terms of the fast neutron flux, the hottest fuel rod linear power and the thermal powers of the core and of the target in MYRRHA at the beginning of fuel life are summarised in Table 1 below [12].

Table 1 MYRRHA facility performances.

System operation parameter	value	unit
Accelerator		
Energy of protons	350	MeV
Proton bean current	5.0	mA
Spallation target		
Neutron source intensity	$1.88 \cdot 10^{17}$	neutrons s^{-1}
Effective heating length	13.0	cm
Total thermal power	1.43	MW
Subcritical core (at start)		
Neutron multiplication factor (k_{eff})	0.9552	
Neutron source importance	1.127	
Maximum neutron flux total (at the hottest rod) fast (En > 1 MeV)	$4.1 \cdot 10^{15}$ $0.80 \cdot 10^{15}$	neutrons cm^{-2} s^{-1}
Fuel average specific power	101.0	W/g-HM
Total thermal power	51.8	MW
Radial power form-factor	1.25	
Primary coolant mass flow rate	2500	Kg s^{-1}
Coolant inlet temperature	200	°C
Coolant average outlet temperature	340	°C
Coolant pressure at core inlet	0.5 + hydrostatic	MPa
Pressure drop of the coolant on the core	0.18	MPa
Hottest fuel rod (at start)		
Peak linear heating rate	352	W cm^{-1}
Axial form-factor	1.29	
Cladding maximum temperature	~440	°C
Fuel maximum temperature	~1800	°C

The MYRRHA operation fuel cycle will be determined by the k_{eff} drop as a function of the core burn-up. The targeted operating regime is 3 months of operations and 1 month for core reshuffling, loading and maintenance. One cycle of operation of the fresh core will lead to the reactivity drop of about 1000 pcm. After every cycle, the core will be re-configured, and fresh fuel will be added in order to compensate this reactivity loss.

Studies of the possibilities to use fuel based on low-enriched uranium (LEU) (~ 20 % U-235) in the place of MOX, in order to satisfy the non-proliferation requirements of the RERTR program [6] imposed to new research reactors, were started. At the first stage, the existing design of the MYRRHA core was used for these studies,where MOX was replaced by LEU based fuels. The k_{eff} of these cores has been estimated. In these calculations, the we aimed to compensate the lower enrichment by a higher fuel density. The preliminary results are presented in Table 2 below.

Table 2. Effect of the replacement of MOX by LEU based fuel on the MYRRHA core size and k_{eff}

Fuel type	Material	Density g cm^{-3}	HM density g cm^{-3}	Enrichment wt.%	Number of fuel assemblies	k_{eff}
MOX	ceramic	10.55	9.30	30% RG Pu	45	0.955
UO_2	ceramic	10.40	9.16	20 % ^{235}U	45	0.709
UO_2	ceramic	10.40	9.16	20 % ^{235}U	72	0.801
UO_2	ceramic	10.40	9.16	20 % ^{235}U	99	~ 0.9
UN	ceramic	13.57	12.82	20 % ^{235}U	72	0.884

From Table 2 one can see that, in order to reach the aimed k_{eff} *~0.95* with LEU fuels, the initial amount of fuel in the core (hence core sizes) must be significantly increased. It will also affect the core neutronic performances and total power. Thus, the use of LEU based fuel can require serious modifications in the current MYRRHA ADS design.

4. Conclusions

- The preliminary core design of the experimental ADS MYRRHA loaded with highly enriched MOX (30 wt.% RG Pu in HM) and cooled with LBE has been completed. Neutronic and thermomechanical modelling showed that its performances satisfy well the aimed requirements. The estimated safety margins are sufficient for about three years of the normal operation with cycles of three months followed by reshuffling periods of one month.

- A possibility of the replacement of MOX by the fuel based on LEU is under study. First estimates indicate that, even in a case of metallic LEU alloys, the core must be increased to obtain the desired neutronic performances. It will require serious modifications in the current MYRRHA design.

5. References

[1] OECD/NEA, "Accelerator-driven Systems (ADS) and Fast Reactors (FR) in Advanced Nuclear Fuel Cycles", ISBN 92-64-18482-1 (2002).

[2] H. Aït Abderrahim, P. Kupschus & MYRRHA-team, "MYRRHA - A Multipurpose Accelerator Driven System for Research & Development. March 2002 Pre-Design Report", Scientific Report of SCK·CEN R-3595 (2002).

[3] H. Aït Abderrahim, et al. "MYRRHA: A multipurpose accelerator driven system for research & development," *Nuclear Instruments & Methods in Physics Research,* **A 463** (2001) 487-494.

[4] B. R. T. Frost, "Nuclear Fuel Elements", Pergamon Press, Oxford (1982).

[5] K. Wirtz, "Lectures on Fast Reactors". Kernforschungszentrum und Universität Karlsruhe. Karlsruhe, Postfach 3640, 7500 Karlsruhe, Germany (1978).

[6] D. Huizenga, M. Clapper, D. Newton, "US Department of Energy research reactor initiatives in support of global nuclear non-proliferation, nuclear security and nuclear safety", Trans. 7th Int. Top. Meet. "Research Reactor Fuel Management" (RRFM-2003). March 9-12, 2003, Aix-en-Provence, France.

[7] R. L. Klueh, D. S. Gelles, S. Jitsukawa, A. Kimura, G. R. Odette, B. van der Schaaf, and M. Victoria, "Ferritic/martensitic steels – overview of recent results." J. Nucl. Mater. **307-311** (2002) 455-465.

[8] R. L. Klueh and D. R. Harries, "High-Chromium Ferritic and Martensitic steels for Nuclear Application", Library of Congress Cataloging in Publication data by American Society for Testing and Materials (ASTM) (2001).

[9] A.G. Bespalov, et al. "Irradiation Performance of Structural Materials in Reactors with Pb-Bi Coolant", in: Proc. "Heavy Liquid Metal Coolants in Nuclear Technology", v. 2, pp. 640-647. Obninsk, RF (1999).

[10] Y.S. Tang, R.D. Coffield, R.A. Markley, "Thermal Analysis of Liquid-Metal Fast Breeder Reactors". Library of Congress Cataloging in Publication Data. ANS (1978).

[11] "Le Combustible Nucleaire des Reacteurs à Eau Sous Pression et des Reacteurs à Neutrons Rapides". Ed. By H. Bailly, D. Ménissier, C. Prunier. Comissariat à l'Énergie Atomique. EYROLLES, Paris (1996).

[12] E. Malambu et al. , "Status of Neutronics Calculations for the Pre-Design Proposal of the MYRRHA ADS", In "*Proceedings of the International Workshop on P&T and ADS Development"*, Mol, Belgium, October 6-8, (2003).

[13] P.L. Kirillov, S. Yuriev Yu., V.P. Bobkov "Handbook: Thermohydraulic Calculations (Nuclear Reactors, Heat Exchangers, Steam Generators)". Moscow: Energoizdat, (1990) (in Russian).

[14] M. Lippens, Ch. Potten, "Preliminary Study of a Fuel Pin Design for the MYRRHA Reactor Core 1: Assembly Design". Contract Report of Belgonucleaire for SCK·CEN (2003).