Abstract
In the modern world, demand for electricity is constantly rising and concerns about climate changes and air pollution are major factors for extra interest in new nuclear power plants. Nuclear energy is very reliable, efficient electricity source without greenhouse gas emissions. Solution for those enormous demands can be provided with the next generation of nuclear power plants (Generation IV) which are based on improved safety, sustainability and efficiency. The crucial requirement is to make these reactors economically competitive. One of these potential candidates are fast reactors cooled by molten lead. To get additional knowledge and experience on this type of reactors a Belgium research reactor MYRRHA is in design project stage aiming to demonstrate the feasibility of the accelerator driven system and the lead cooled fast reactor concepts. It is listed as one of 50 projects which aim to increase important role of Europe in high-tech research.
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1. Introduction

First self-sustaining nuclear chain reaction was achieved in research reactor Chicago Pile-1 during the world war II (1942) and at the beginning of the second half of the 20th century first commercial exploitation of civil nuclear power appeared. Since then nuclear reactor technology is being constantly under development and divided on broad range of generations shown in figure 1, each presenting a major technical advance compared to previous generation either in terms of performance, cost or safety.

Generation I included first commercial power plants of different designs from gas-cooled and graphite moderated to first prototype water cooled and moderated. Generation II [1] includes standard light water pressurised (LWR) and boiling water (BWR) reactors which are operating today under the name thermal reactors. Generation III designs are being currently under construction and are an upgrade of current LWR technology with improved performance and evolutionary design. Improved safety mechanisms greatly extend operating lifetime and are more favourable in case of an extreme events (loss of coolant accident or core damage). A typical example is an AP1000 and the European Pressurised reactor (EPR).

Generation IV design presents the so called advanced fast reactors that could be operating commercially from 2030-2040. This technology is not new and such reactors have existed for decades though they have never been in commercial use (with exception of Russia reactors BN-600 and BN-800 in operation at Beloyarsk Nuclear Power Station). Their advantage is that they can extract more than 50 times more energy from the same amount of uranium than the standard reactors due to the breeding process of fertile U-238 into Pu-239 which is one of the main fissile materials. This particular characteristic is very favourable in a scenario of dwindling uranium reserves. However, fast reactors have also some specific technical challenges related to safety, non-proliferation and economy..

Currently, six types of Generation IV systems are being investigated for further research and development (4 of them are fast reactors):
Lead-cooled Fast Reactor (LFR), Sodium-cooled Fast Reactor (SFR), Gas-cooled Fast Reactor (GFR), Molten Salt Reactor (MSR), Supercritical Water-cooled Reactor (SCWR), and Very High Temperature Reactor (VHTR) [2, 3]

2. Thermal vs Fast reactors

All operating nuclear reactors are based on principal mechanism of nuclear fission, which may occur when a heavy fissile atomic nucleus absorbs a neutron and splits into two or more fast moving lighter nuclei (fission products). At the same time, free fast neutrons (and gamma photons) are produced as well which are later absorbed by other fissile nucleus and cause further fission events. This repeating process is so the called nuclear chain reaction.

But not every neutron can cause fission reaction with equal probability. Probability for fission reaction depends on the neutron fission cross section which is strongly depended on neutron kinetic energy for different heavy nuclei. Figure 2 presents a neutron capture and fission cross section for heavy nuclide U-235 and U-238 for different neutron kinetic energy. It is clear, that fission in U-235 will be more favourable for neutrons with kinetic energy bellow 1eV (fission cross section is more than 1000 times larger for thermal neutrons), but the ration between fission and capture will be much greater for neutrons with kinetic energy over 1 MeV. On the other hand, only neutrons with kinetic energy over 1 MeV can cause fission reaction in U-238 [2, 5].
Most nuclear reactors operating today are known as thermal reactors. Their principle of operation is based on fission neutrons, which are born with typical energies of around 1 MeV and must be slowed down to thermal energies (below 1eV) to sustain the chain fission reaction. Neutrons produced in fission reaction are slowed down through elastic scattering in a moderator (typical moderator in LWR is water). Equation bellow presents common fission reaction in thermal reactors

\[ n + ^{235}_{92}U \rightarrow ^{236}_{92}U^* (\tau = 10^{-16} \text{ s}) \rightarrow \frac{4}{2}X + ^{236-A}_{92-Z}Y + kn \]  \hspace{1cm} (1)

where U-235 is major nuclear fuel, U-236* the unstable compound nucleus, which temporarily contains all of the charge and mass involved in the reaction and only exists for on the order of \(10^{-16} \text{s}\) (undetectable), X and Y fissile products and k value present number of released neutrons per fission.

As a name already implies, fast reactors got their name after energetic, fast neutrons which sustain chain reaction in reactor fuel. As there is no need for slowing them down, reactor core is more compact with absence of moderator. In addition, alternative materials, which are more efficient coolant, can be used because they do not have to serve as a moderator.

Due to the operation in a more energetic neutron spectrum, fast reactors can transform more efficiently the U-238 into the fissile material Pu-239 through capturing fast neutrons. Equation bellow presents capture of a fast neutron in a fertile nuclide U-238, which through two \(\beta\) decays transforms into Pu-239 with Np-239 as an intermediate step in the reaction

\[ n + ^{238}_{92}U \rightarrow ^{239}_{92}U \beta^{-}(23.5 \text{ m}) \rightarrow ^{239}_{92}Np \beta^{-}(2.35 \text{ d}) \rightarrow ^{239}_{94}Pu \]  \hspace{1cm} (2)

The ultimate goal of fast spectrum reactor is to breed fuel. The degree of conversion that occurs in a reactor is denoted by the general term conversion ratio (CR) which is defined as

\[ CR = \frac{\text{Fissile Material Produced}}{\text{Fissile Material Destroyed}} = \frac{\int_0^{T_f} dt \int_{\Omega_c} RR_{EP}^F (\vec{r}, t) \, d\Omega}{\int_0^{T_f} dt \int_{\Omega_c} RR_{EP}^D (\vec{r}, t) \, d\Omega} \]  \hspace{1cm} (3)
where $RR^p (\vec{r}, t)$ and $RR^d (\vec{r}, t)$ are the capture and absorption reaction rates, respectively. In general, $CR(\vec{r}, t)$ is also a property that is variable in space and time and can be averaged over reactor core volume $V_c$ and between periodic refueling $T_f$. If the conversion ratio is greater than unity ($CR > 1$), a reactor is called a “breeder”, otherwise ($CR < 1$) is called a “converter”. All present commercial thermal reactors (LWR) are converters, i.e., they produce around 40 - 70% of new fuel from U-238.

Another important value is the average number of new neutrons generated for each neutron absorbed in the fuel (averaged over the neutron flux spectrum). The parameter $\bar{\eta}$ is defined as

$$\bar{\eta} = \frac{\bar{\nu}_f \sigma_f}{\sigma_a} = \frac{\bar{\nu}_f}{1 + \sigma_c / \sigma_f} = \frac{\bar{\nu}_f}{1 + \bar{\alpha}}$$

where $\bar{\nu}_f$ is number of neutrons per fission, $\sigma_f$, $\sigma_c$ and $\sigma_a$ cross section for fission, capture and absorption, respectively and $\bar{\alpha}$ represent capture-to-fission ratio. The maximum possible value of conversion ratio is $CR_{max} = \bar{\eta} - 1$. The parameters $\bar{\nu}_f$ and $\bar{\alpha}$ are measured quantities. For each of the primary fissile isotopes, $\bar{\nu}_f$ is fairly constant ($\bar{\nu}_{U^{233}} = 2.50$, $\bar{\nu}_{U^{235}} = 2.44$ and $\nu_{Pu^{239}} = 2.88$) for neutron energies up to about 1 MeV and slowly rises at higher energies. On the other side $\bar{\alpha}$ varies considerably with energy and between isotopes: it rises sharply in the intermediate energy range between 1 eV and 10 keV and then drops again at high energies for Pu-239 and U-235. Behaviour of $\nu$ and $\alpha$ leads to variations of $\eta$ with energy shown in Figure 3.

\[Figure 3. Neutrons produced per absorption vs. energy for fissile isotopes reproduced from [2].\]

In order to breed, yield factor has to be at least $\eta \geq 2 + L$, due to absorption of one neutron in a fissile isotope, in order to continue the chain reaction, and losses by parasitic absorption or by leakage from the reactor (L). For simplification, absorption by any material other than fissile or fertile is parasitic.

3. Lead fast reactor

One of the 6 concepts for Generation IV reactors are lead fast reactors (LFR). Due to the excellent heat transfer in heavy liquid metal coolant LFR concepts offer substantial potential in terms of safety (lead inherent properties and effective forced convective heat removal),
proliferation resistance (breeding factor is slightly larger than 1 compared to sodium reactors where breeding factor is almost 1.3), improved resource utilization (extract more than 50 times more energy than thermal reactors from the same mass of uranium), longer core life, effective burning of minor actinides (concentration of the long lived minor actinides is less than 0.1 % of all the fission products) and hopefully the resulting economic competitiveness [7, 8, 9, 10].

The current experience base for Heavy Liquid Metal coolants (HLMCs) includes 80 reactor years operating experience in the Soviet Union and subsequently the Russian Federation with Lead-Bismuth Eutectic (LBE) cooled reactors for strictly military purposes. Two 73 MWth and eight 155 MWth submarine reactors were operated with 40 reactor years operation underway together with 70 and 155 MWth land prototypes. Detailed design information about LBE-cooled submarine reactors has never been provided. For example, drawing of the designs have never appeared in the open literature. In 1998 (and later on conferences held in Obninsk in 2003 and 2008) Russia declassified a lot of research information derived about the LBE-cooled reactors, LFR concepts and HLMC technology. Since then US interest in using Pb or PBE for small reactors has increased subsequently. Distinctive features of lead as a coolant are presented below [2, 11].

3.1 Advantages

The term lead-cooled fast reactor (LFR) usually applies to a fast reactor utilizing either of two heavy liquid metal coolant materials. The first is lead (Pb) itself which has excellent cooling properties and a melting temperature of 327.45°C and an atmospheric boiling temperature of 1743°C. The second material is lead-bismuth eutectic alloy (LBE), which is composed of 45 at. % Pb and 55 at. % Bi with even better characteristic with a melting temperature 124.5°C and boiling point at 1671°C. Both densities at 480°C are similar, about 10000 kg/m³. The high HLMC density is a benefit in some respect, including improved safety (limits void growth and downward penetration due to the buoyancy).

The very high boiling temperatures of the HLMCs allow for potential operations at system temperatures that are not limited by coolant boiling but by the integrity of the steel cladding and structures. The operating pressure is very low thus enabling significant advantages of a low pressure compact liquid metal system. With proper design of the guard vessel even in the event of a failure of the reactor vessel, system would not suffer a loss of primary coolant.

Heavy liquid metal coolants have several key inherent properties (do not react vigorously with water, steam, or air) in contrast to the alkali-based coolants (NaK). A design of such reactors can potentially reduce the plant cost and improve safety, thus designer does not need to incorporate features similar to those needed to accommodate sodium-water reactions.

3.2 Disadvantages

The high HLMC density compared with the density of sodium or water require greater structural thicknesses and results in greater cost for structural steel. This is significant factor in limiting the size of a LFR due to the need to accommodate seismic events. Also fuel assemblies containing oxide fuel have to be strained due to buoyancy.

HLMC dissolves Fe, Cr, and Ni from unprotected steels at rates that increase with temperature. Dissolved elements need to be removed (filters) from the coolant to avoid the
potential for plugging of narrow passages. To increase corrosive resistance new ferritic/martensitic steel cladding is under investigation.

As noted earlier HLMC has a high melting point and the solidification can present a serious threat and renders the reactor inoperable. An additional heating system must be providing to melt the coolant and maintain it in a molten state prior to start-up (also during refuelling, inspections and maintenance operations).

Another disadvantage of LBE coolant is neutron capture in Bi-209 (LBE) which directly leads to production of the isotope Po-210 \((^{209}\text{Bi} + n \rightarrow ^{210}\text{Po} + e^{-})\). The Po-210 is an alpha emitter with a half-life of 138 days. Besides that, elemental Po has a high melting and boiling temperatures it has also a high volatility. Contact with moisture can result in the formation of polonium hydride \((\text{PoH}_2)\) which are health hazards gaseous compounds and aerosols. Po-210 is the main reasons why Pb coolant is more favourable in some LFR designs (produce almost 2-4 orders of magnitude lower amount of Po-210 relative to LBE).

4. Multi-purpose hYbrid Research Reactor for High-tech Applications

To get additional knowledge and experience on fast reactors a Belgium research reactor MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications), also known as ETPP (European Technology Pilot Plant) [7, 12] is in design project which aim to demonstrate the feasibility of the accelerator driven system (ADS) and the lead cooled fast reactor concept with two possible configurations: sub-critical or critical. The project is managed by Belgian Nuclear Research Centre SCK•CEN [13] and it will be built based on the experience gained from the first successful demonstration project GUINEVERE [14]. It is listed as one of 50 projects which aim to increase important role of Europe in high-tech research.

Main MYRRHA characteristics, shown in figure 11, are a 2.1 MW proton beam (600 MeV – 3.5 mA), a spallation target and LBE coolant. A multiplying core with mixed oxide (MOX) fuel will provide around 65-100 MWth for subcritical mode and around 100 MWth for critical mode.

![Figure 4. Scheme of the MYRRHA research reactor reproduced from [12].](image)
4.1 Design

A compact core is needed to meet the requirements of the desired high flux levels (around $10^{15} \text{n/cm}^2\text{s}$) and high power density. Therefore, central hole which is intended for the spallation target should not be larger than 10 cm, where a window target design is proposed (with physical separation between the proton beam and the liquid target material).

Due to thermal inertia of the coolant a pool-type system is designed in which the components of the primary loop are inserted from the top (pumps, heat exchanger, fuel handling tools, experimental rigs, etc.). On the other hand, FA will be loaded from underneath due to flexibility for the experimental devices and safety reason as all other structures are already in place before the core loading.

The main components of the MYRRHA reactor are shown in Fig. 7: 1 Reactor vessel, 2 Guard vessel, 3 Cover, 4 Diaphragm, 5 Heat exchangers, 6 Primary pumps, 7 Fuel storage zone, 8 Spallation target and core, 9 Spallation loop, 10 Fuel manipulators.

The cooling systems are designed to effectively evacuate core power of 50-100 MWth, additional heat produced by the spallation target, pumps, decay heat in the in-vessel storage zones and additional heat due to the polonium decay. Lead-bismuth eutectic ($150 \text{ m}^3$) serves not only as a coolant for both the spallation target and core but also as a reflector/shielding for the fast neutrons and gamma rays. During operation (at full power) a coolant inlet and outlet temperatures are 270°C and 400°C, respectively (melting point at 123°C).

4.2 An Accelerator Driven System (ADS)

ADS is a neutron source created by coupling to a proton accelerator, a spallation source (target) and a sub-critical core. The main advantage of an ADS is its inherent safety due to its non-critical fission core. A non-critical core cannot sustain the fission chain reactions on its own. The accelerator, which is the driver of the ADS system, provides the high-energy protons that are used in the spallation target to create neutrons which in their turn feed the sub-critical core. The reactor is immediately turn down with the absence of proton beam.

4.3 The spallation target

The spallation target is a primary neutron source which provides neutrons that are multiplied by the surrounding sub-critical core with the multiplication factor $k_{eff} = 0.95$, which represents number of neutrons in one generation compared to previous generation. The primary neutrons are produced by the spallation reaction of heavy target nuclide (high Z number: LBE) which are bombarded by high-energy protons (600 MeV) generated by the accelerator.
In a first stage of the spallation reaction the incident particles (high energy protons) interact with the nucleons of the target and lead to the emission of very energetic secondary particles (neutrons, protons, pions, alpha-particles, etc.). During a second stage, highly excited nucleus will de-excite by releasing few MeV energy neutrons which can cause high energy fission. With this process, a large amount of spallation neutrons can be produced depend on the initial energy of the incident protons and on the atomic number of the target nuclei. For example, a lead target bombarded with 600 MeV protons can yield about 13 neutrons per incident proton, at 1 GeV one expects almost 25 neutrons per incident proton.

A LBE has been chosen as target material to obtain a high neutron gain and to allow effective forced convective heat removal as during the process almost 65 % of the total beam power (2.1 MW) is deposited as heat in the target.

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**Figure 6. Scheme of the spallation reaction and further fission reproduced from [12].**

### 4.4 Fuel and sub-critical core

The main candidate for fuel is a mixture of plutonium-uranium oxide (MOX, with enrichment of 30 to 35 % of plutonium) due to a large experience in the production and its better neutronic properties in the fast spectrum compared to uranium dioxide.

The fresh core is designed to have a lattice of 183 hexagonal channels of which 68 are loaded with fuel assemblies (FA). At the centre of the sub-critical core are 3 free spaces intended for the spallation target module. A ten-step reshuffling scheme has been studied for 90-day cycle where every fuel assembly undergoes the operation 8.3 times on average. It is predicted that after 3 months of operation $k_{eff}$ will drop from 0,95 to 0,9493. To compensate the power loss during the operational cycle of 90 days two options are available. First option includes gradually increasing the beam current during cycle, and the second option is using a burnable poison.

MYRRHA will also be favourable for different applications. It will try to demonstrate the physics and technology of an Accelerator Driven System (ADS) for transmuting long-lived radioactive waste in order for sustainable fission energy. By that, it will reduce the burden on the nuclear waste disposal.

Due to its fast spectrum and spallation target MYRRHA can also fulfil need for the development of materials and fuels which can only be performed in an irradiation facility where fully controlled and representative experimental conditions can be obtained. Furthermore, MYRRHA will aim at testing new technological developments for sensors and
instrumentation for innovative fission reactors, fusion reactors, space applications, production of radioisotopes for nuclear medicine and other fundamental research (reactor and fuel cycle technology).

MYRRHA will be operational at full power around 2025. Before that, construction of the facility and assembly of the components is foreseen in the period 2017-2021. After that, 3 years are foreseen for the full commissioning of the facility. The overall investment cost was estimated in 2009 at around 960 M€.

5. Conclusion

Generation IV reactors promise a new advanced technology to utilize extreme demand for electricity and reducing air pollution.

One of them are fast spectrum reactors (fast reactors) which are undoubtedly the most efficient systems for the effective utilization of uranium resources, due to possibility to produce electricity while recycling the uranium stored in used fuel. With this technology, the energy potential of uranium increases significantly (by a factor of 60) compared to LWRs. In addition, the radioactive wastes containing long lived minor actinides become practically insignificant (approximately less than 0.1 % of all the fission products). With these unique features, fast reactors can minimize the mining effort and also reduce storage space and the time required in a used fuel repository (by a factor of 1000) to reduce the waste radiotoxicity to the required level. Hence, fast reactor could offer a realistic path forward to achieve energy sustainability consistent with environmental stewardship. The main drawback appears from the economical point of view as we currently have not managed to construct a cheap and good fast reactor.

The MYRRHA project will fulfil the role as 1st European Technology Pilot Plant in the roadmap for the development of the lead fast reactors.

References

[7] ESNII, The European Sustainable Nuclear Industrial Initiative, 2010