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Review Article

Heavy liquid metal cooled fast reactors: peculiarities and development status of the major projects

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Abstract

Fast reactors with heavy liquid metal coolant (lead or eutectic bismuth-lead alloy) are one of the most advanced technologies capable to address the accumulated world nuclear energy issues. This innovative power technology is being developed in Russia, the USA, China and the European Union. Russia is the leader since it has focused on this topic for a number of decades. First concrete started to be poured in June 2021 to form the foundation of the Russian BREST-OD-300 lead cooled reactor scheduled to be started up in 2026. Attention is also given to the development status of the Chinese CLEAR reactor series. A large scope of R&D has been undertaken, and large-scale nonnuclear experimental facilities are under construction. International Euro-US consortiums for the development of the ALFRED, PLFR and MYRRHA reactors do not expect any unsolvable technical issues either and are currently formulating requirements to the test facilities and candidate materials and technologies required for further activities.

Keywords

fast reactor, heavy liquid metal coolant, closed nuclear fuel cycle, Russia, USA, China, European Union

Introduction

In the 1970s, the world's expert community viewed the prospects for the evolution of nuclear power as clearly optimistic expecting this industry to evolve further intensively to the extent that nuclear power would account for 30% of the world's energy generation by the 2020s (Adamov 2020). These expectations however failed to materialize.

As of the beginning of 2020, as shown by the IAEA data (IAEA 2022a), there were 444 nuclear power units in operation across the world with an installed (net) electric capacity of about 392 GW, and 54 nuclear units more (57.4 GW) in the process of construction. According to the 2016 data, nuclear power accounts for 5% of the world's energy generation as its share in electricity generation amounts to 10%. In Russia, atomic energy accounts for 20% of the total electricity generation.

Global energy consumption has grown continuously despite the recession of 2008–2009 and a great deal of uncertainty in the outlook for economic development, largely at the expense of developing countries, and this trend is expected to continue for at least decades to come.

Modern global nuclear power is based on uranium fueled thermal reactors (TR) in an open nuclear fuel cycle (ONFC). The technological framework for LWRs, which make the basis for the world's nuclear power, is sufficient to predict (in a span of up to 2050) the scale of the NPP construction, whereas its potential in addressing long-term energy issues is limited due to the engineering safety level failing to meet the key requirement to large-scale nuclear power, that is excluding accidents that would require evacuation of the population (Three Mile Island 1979, Chernobyl 1986, Fukushima 2011).

Requirements to new generation IV reactors were developed by the nuclear technology developing countries as part of Generation IV, a major international forum established in the early 21st century (Energy Information Administration 2003). The six technologies selected for the joint development include four different fast reactor and CNFC technologies.

Requirements to innovative nuclear power systems to satisfy the sustainability principles were also formulated as part of another major IAEA's international project, INPRO (IAEA 2022b). The INPRO investigations have confirmed as well the importance of developing the fast reactor and CNFC technologies, specifically for countries that possess an extensive NPP fleet or plan large-scale development of nuclear power.

One of the advanced technologies capable to address the outstanding issues of the world's nuclear power is heavy liquid metal cooled (HLMC) fast reactors. This paper provides an overview of the key projects in this field.

Abbreviations

ADS	(Accelerator Driven System) – ADS (a subcritical reactor with an external neutron accelerating source)
CBR	core breeding ratio
CLEAR	(China Lead-based Reactor) – Chinese fast reactor with lead (LBE or lead) based coolant
CNFC	closed nuclear fuel cycle
CPS	Control and Protection System
DBE	design based earthquake
ECS	energy conversion system
EU	European Union
FA	fuel assembly
FP	fission products
HALEU	(High-Assay Low Enriched Uranium) UO_2 fuel – uranium oxide fuel with 5 to 20% ^{235}U enrichment
HLMC	heavy metal liquid coolant
ICUF	installed capacity usage factor
LBE	lead-bismuth eutectics
LWR	Light Water Reactor
MOX SNF	MOX-fuel SNF
NDHRS	normal decay heat removal system
NPP	nuclear power plant
PDHRS	passive decay heat removal system
PFBS	passive feedback system
PLFR	Westinghouse: Prototype Lead Cooled Fast Reactor
RW	radioactive waste
SFA	spent FA
SG	steam generator
SNF	spent nuclear fuel
SNPP	small nuclear power plant
SNUP SNF	spent mixed nitride uranium-plutonium fuel
UDBE	ultimate design based earthquake
WLFR	Westinghouse: Lead Fast Reactor (PLFR)

BREST-OD-300 pilot and demonstration reactor facility

A power unit with the BREST-OD-300 reactor facility with dense nitride fuel and high-boiling lead coolant is developed as part of the *Proryv* project for the purpose of shaping, implementing and demonstrating innovative naturally safe nuclear power technologies based on fast reactors and their closed nuclear fuel cycle. The BREST-OD-300 reactor facility is considered as a prototype of future commercial BREST-type fast reactors for large-scale nuclear power that is capable to assume responsibility for most of the electricity generation growth and to solve the energy supply problem for the sustained development of humankind for many years to come.

The power unit is expected to be operated as part of the Pilot and Demonstration Energy Complex (PDEC) with an onsite closed nuclear fuel cycle. The second major goal is to test the complete process cycle. The key technical characteristics of the power unit and the reactor facility are given below (Adamov 2020).

Power unit characteristics

The power reactor, the reactor facility and most of their components are innovative and do not have comparable counterparts. The selection of the key technical characteristics and designs for the BREST-OD-300 facility, including the power level of 700 MW(th), a specific lead coolant circulation pattern, the normal and emergency cooldown system, etc., was defined not only by the need to demonstrate the natural safety properties of this reactor technology, but also given the requirements for the continuity of designs in reactor facilities of a greater power to be developed in the future.

Key characteristics of the power unit and the BREST-OD-300 reactor facility

Rated thermal/electric power, MW	700/300
ICUF	0.8
Refueling interval, eff. days	300
Core inlet/outlet coolant temperature, °C	420/535
Vapor temperature/pressure (SG outlet), °C/MPa	505/17

Efficiency, %	43.5
Primary loop coolant inventory, m^3	900
DBE/UDBE seismic resistance, earthquake magnitude	7/8
Service life, years	30
Number of plant personnel, persons	316

The BREST-OD-300 reactor facility is the key innovative solution as part of the power unit. Its layout is shown in Fig. 1. Specific to the reactor unit is its pool-type design with an integrally arranged lead loop containing the core with reflectors and CPS rods, steam generators (SG), pumps, refueling system components, a lead cleanup and oxygen control system, and other auxiliary components accommodated in the steel-clad central cavity and four peripheral cavities (the number of the cavities is equal to the number of the lead coolant circulation loops) in the metal-concrete vessel with a cooling system (Figs 2, 3). The temperature of the vessel outer surface at the interface with the civil structures is maintained within its permissible limits by the natural circulation air cooling system.

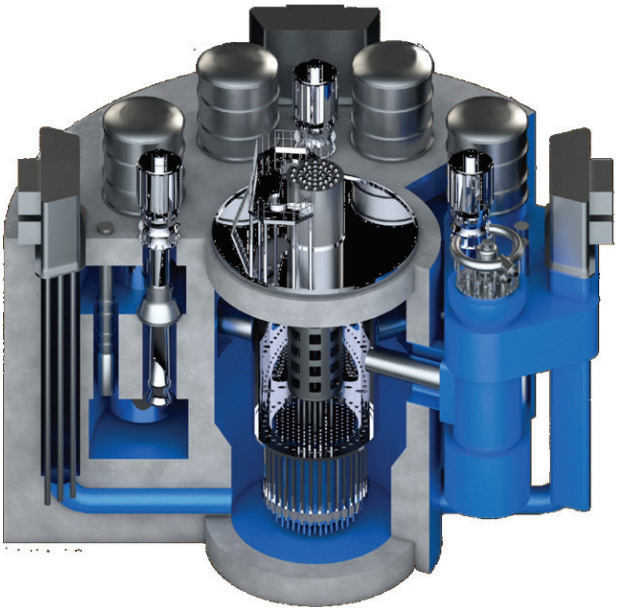

Figure 1. BREST-OD-300 reactor facility (Adamov 2020).

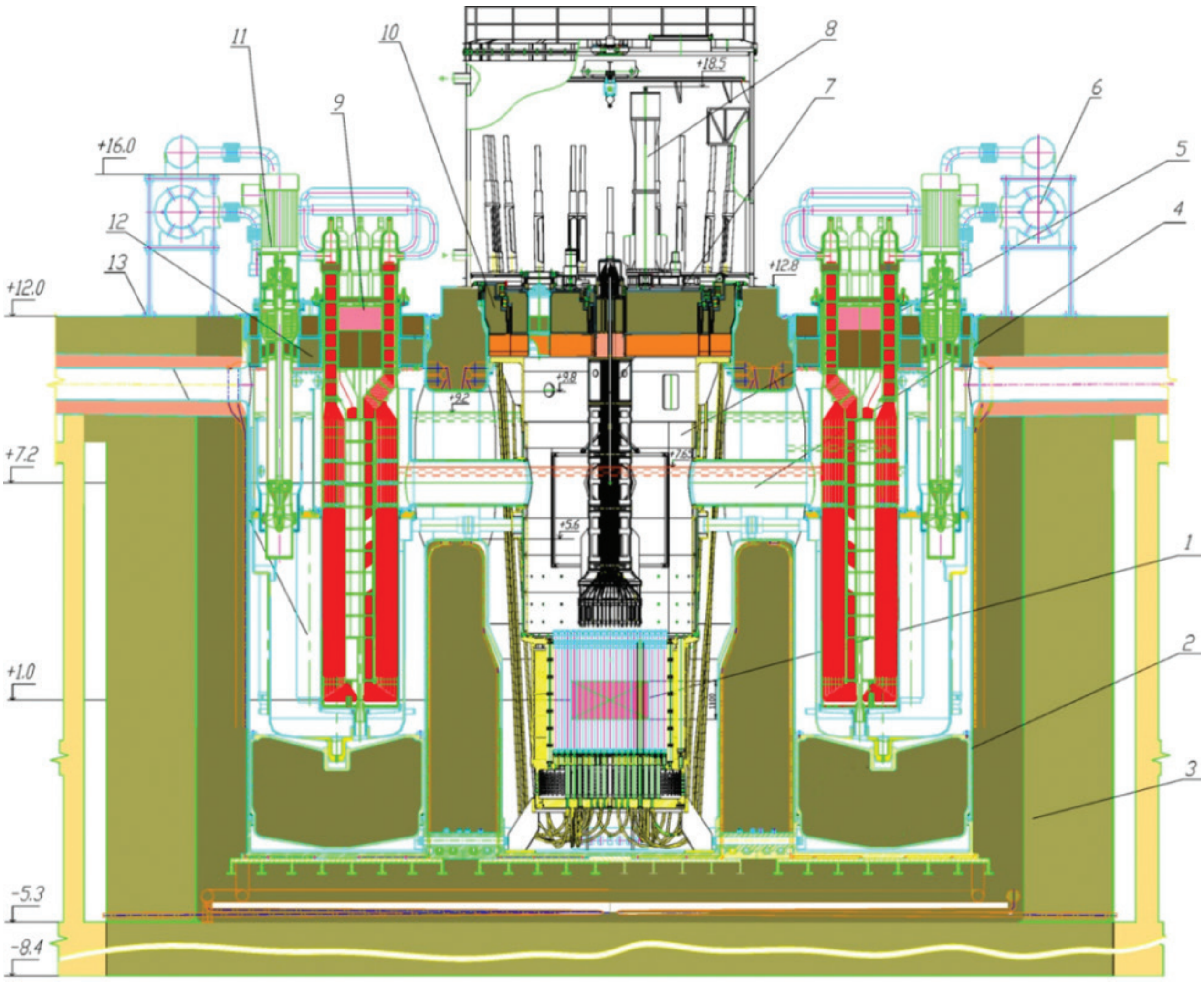

Figure 2. Longitudinal cut of the BREST-OD-300 reactor facility (Adamov 2020): 1 – core; 2 – block of vessels; 3 – reactor pit; 4 – header pipeline; 5 – core basket; 6 – cooldown system; 7 – instrumentation string; 8 – in-pile refueling machine; 9 – steam generator; 10 – upper plate; 11 – reactor coolant pump; 12 – SG-MCP block; 13 – filter.

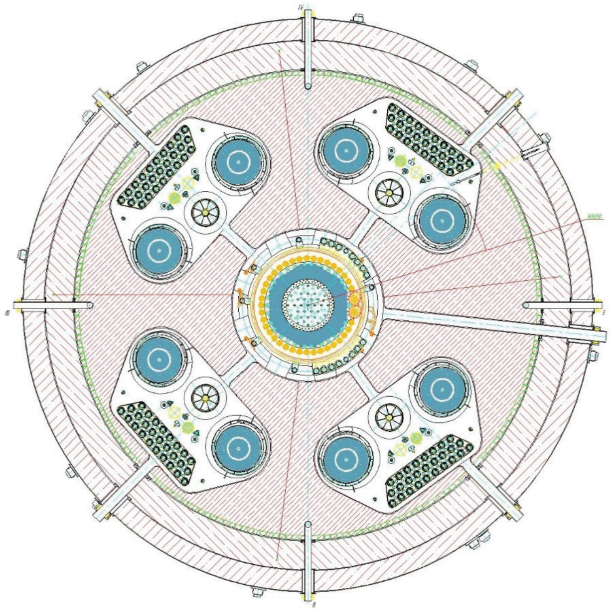

Figure 3. Cross-sectional cut of the BREST-OD-300 reactor facility (Adamov 2020).

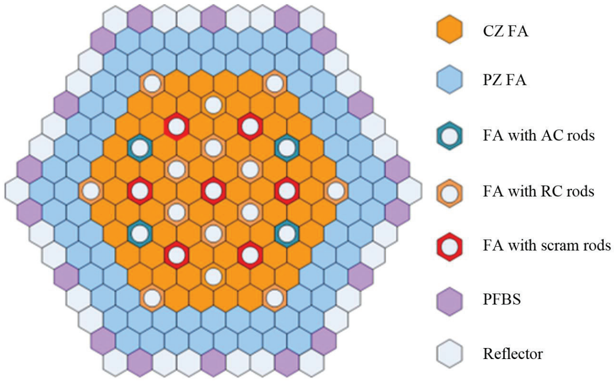

Figure 4. BREST-OD-300 core (Adamov 2020).

Forced circulation of lead through the core is ensured by the difference created by the pumps in the "cold" and "hot" coolant levels. The lead cooled in the SG is supplied by the pumps to the upper (pressure) level and further, through the loop's annular downcomer portion in the central cavity, to the core inlet where it is heated as it flows upwards through the core. The lead further enters the SG and flows down, through the tube space, while giving heat to the secondary water and steam, into the suction chambers from where it goes up to the upper free level. As it leaves the pump, the lead coolant contacts the gas circuit for separation of the lead-captured gas and steam (in the event of the SG pipe leak).

The secondary loop feedwater heated up by live steam in a mixing high-pressure heater to 340 °C is supplied to each SG at a pressure of 18.5 MPa. This is achieved by preventing the lead coolant temperature drop to below the coolant melting point (327 °C) in the reactor startup modes and in emergencies. Specific to the secondary loop is that, unlike the existing earlier NPP designs, it does not assume the safety function with respect to the emergency core heat removal.

BREST-OD-300 core made up of 169 jacket-free hexagonal FAs is designed in the form of two radial zones: a central zone (CZ) and a peripheral zone (PZ) (Fig. 4). Grids are used for fuel rod spacing and retention in the FAs. The fuel composition and the rod number and pitch are the same for all FAs. The power and maximum fuel/coolant heat-up temperature flattening is achieved by the radial fuel design and by the lead flow ensured by using fuel rods of a smaller diameter in the CZ FAs and fuel rods of a larger diameter in the PZ FAs. The stability of the distributions flattened over the fuel life is achieved by using fuel with one and the same composition in all FAs provided fissionable nuclides are bred in full in the core (CBR ~ 1).

Key characteristics of the BREST-OD-300 core

Number of FAs in core	169
Maximum power reactivity margin, β_{eff}	0.54
Temperature/power effect, %	–0.57
Average heat density, MW/m^3	175
Maximum linear load on fuel rod, W/cm	410
Maximum fuel burn-up, % h.a.	6*/9
SNUP fuel starting load weight, t	20.8
Weight of plutonium in starting load, t	2.67
Fuel weight during refueling with maximum burn-up of 6%/9% h.a., t	7.2 / 4.8
Breeding ratio (BR and CBR)	1.05

* For initial operation stage

Along with fuel rods, some of the CZ FAs include a control and protection system (CPS) rod. The combination of the CPS rods form two independent reactor shutdown systems one of which, made up of scram rods, is an emergency protection system, and the other, made up of shim and automatic reactivity control (AC) rods, forms the second system. The CPS rod drives are contained in the upper rotary plug and the rods as such, when in a withdrawn position, are beneath the core. For refueling, the drives are disengaged from the rods which float up into the core and are held there due to Archimedes buoyant force providing so a deeply subcritical reactor state.

The use of jacket-free FAs, as compared with jacketed FAs, ensures a higher safety level. For a jacket-free FA, heat is removed, in the event of the coolant flow blockage, by the coolant overflow from the adjoining FAs. Calculations have shown that the postulated stoppage of the coolant flow at the inlet of seven jacket-free FAs in the central part of the core does not lead to a fuel surface temperature growth in excess of the acceptability criterion of 800 °C.

The core is surrounded by rows of changeable lead reflector and steel shielding blocks, and there is no blanket in the design. Each reflector block is designed as a hexagonal tight steel shroud filled with lead coolant with a small circulation rate. Some of the lead reflector blocks are designed as devices similar to a gas bell, the lead column level in which "traces" the coolant pressure (flow rate) and affects the neutron escape. The channels with lead columns, which are components of the passive feedback system (PFBS), tie the reactor power (reactivity) to the coolant flow rate (discharge) through the core, make it possible to reduce the power reactivity margin on the control rods, and form an important safety factor as they introduce a negative reactivity when the forced flow rate is reduced or stopped.

The absence of the uranium blanket traditional for fast reactors and the use of lead reflector blocks instead of it excludes generation of weapon-grade plutonium (a technological measure for strengthening the nuclear nonproliferation regime), contributes to flattening the neutron field spatial distribution, and excludes the positive reactivity insertion as the lead level in the loop decreases in the event of the coolant leakage or evacuation.

ALFRED reactor (Advanced Lead-cooled Fast Reactor European Demonstrator)

The ALFRED reactor was initially designed with an increased safety margin (Frignani 2019b) and its engineering design is being modified extensively to simplify the reactor design, increase the stability of the reactor and make it scalable. FALCON, an international consortium consisting of Ansaldo Nucleare, ENEA and RATEN-ICN, intends to convert the ALFRED reactor by 2035–2040 to the prototype of a competitive commercial lead-cooled fast reactor for a small nuclear power plant (SNPP). A phased plan has been proposed for the demonstration evolution program of the ALFRED reactor the key parameters of which are given in Table 1.

Table 1. Key parameters of the ALFRED reactor at different demonstration program stages

	Stage 0 (startup)	Stage 1 (low temperature)	Stage 2 (medium temperature)	Stage 3 (high temperature)
Core inlet temperature (°C)	390	390	400	400
Core outlet temperature (°C)	390	430	480	520
Thermal power (MW)	≈0	100	200	300

The core outlet coolant temperature will gradually rise with each subsequent stage changing from the level reached at the research facility to the level required for commercial plants. In parallel with the ALFRED demonstration program, which will make it possible to accumulate the operating experience and to examine safety issues, an R&D program will be undertaken to support the demonstration program with improved process and design solutions.

The lead coolant circulation loop of the ALFRED reactor (configuration, key zones, lead flow direction) is shown in Fig. 5.

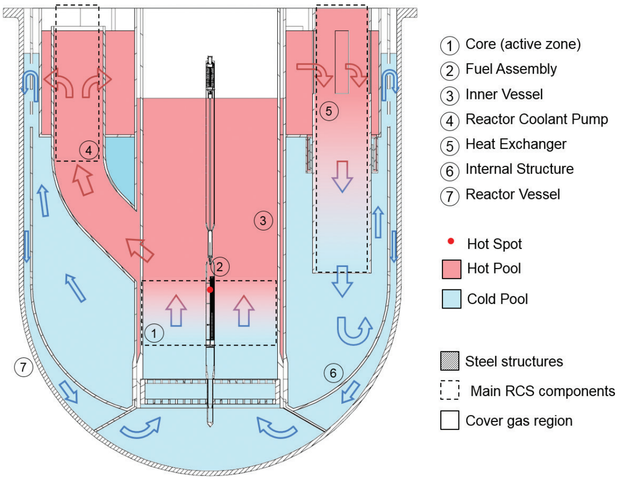

Figure 5. ALFRED lead coolant circulation loop (configuration, key zones, lead flow direction) (Frignani 2019b).

Stage 1 is to investigate two major factors of the lead-cooled fast neutron technology:

- compatibility of lead with structural materials;
- control of the lead coolant's physical and chemical properties.

The ALFRED reactor is expected to have the concentration of dissolved oxygen in liquid lead maintained in a range of 10^{-6} to 10^{-8} wt. % and to use nuclear grade structural materials (specifically, austenitic steels of type 316 or 15-Ti), which have shown themselves to be compatible with lead in the selected temperature range. However, no validated technology has been developed so far in the EU countries for pool systems to control the concentration of dissolved oxygen locally, so ALFRED will operate at the initial stage with an approximately uniform concentration of dissolved oxygen at the level of 10^{-7} wt. % plus or minus one order of magnitude.

Further stages will require special qualification programs to deploy the process solution in the reactor and various strategies to be used in the reactor as such or, in parallel, in individual experimental facilities. Additionally, other lead cooled fast reactor technology issues will be addressed successively to:

- minimize the primary coolant flow rate (except the easy-to-replace circulation pump impeller region) and areas with abrupt momentum changes (e.g., heavy turbulence or flow collision areas);
- deploy a self-controlled passive decay heat removal system, thanks to which the estimated time to the lead coolant freezing has decreased considerably as compared with the earlier declared 72-hour interval;
- use more compact RCS components (the entire nuclear island shall be based on high-damping rubber supports) proven earlier in conditions of full-scale dynamic oscillations;
- ensure that all components to be inspected in service would be retrievable and replaceable to enable

inspection and repair in the absence of molten lead and to expand, additionally, the ALFRED demonstration capabilities.

The temperature conditions for the normal operation of structural materials in the hot pool (the core outlet header and the heat exchanger inlet headers) are 430, 480 and 520 °C for the reactor thermal power of 100, 200 and 300 MW respectively (Frignani 2019b). In emergency conditions, the maximum temperature of structural materials in the same area with the same reactor thermal power values is 480, 590 and 680 °C.

It should be noted that the maximum temperature of structural materials is reached not in the hot pool but at the so-called core hot spot with its non-uniform power density, that is, on the outside wall of a fuel rod in the hottest channel. Depending on the reactor thermal power (100, 200 and 300 MW), the maximum temperature of structural materials in the hot spot was 450, 535 and 600 °C for normal operation and 520, 650 and 800 °C for emergencies.

A strategy has been developed, given the temperature and radiation conditions, for the approach to different RCS components of the ALFRED reactor. A logic has been proposed to categorize protective measures with identification of respective risks. A table has been developed for categorizing preliminarily protective measures for the ALFRED reactor RCS components, which suggests that the risk is moderately high for the fuel assemblies (FA) and fuel rods, and the risk level is somewhat lower, that is moderate, for the inner vessel, the core basket, the diagonal lattice, the internal structure, and the heat exchangers (steam generators and the decay heat removal system, the shaft and the impeller). The lowest risk level is for the reactor vessel.

The most heated and irradiated component is the fuel cladding. For this reason, it is planned that FAs will be replaced every five years, and the FAs to be used at a further stage are preliminarily tested at the current stage by being placed in the core center.

The phased ALFRED evolution demonstration program shown in Table 1 envisages (Grasso et al. 2019), specifically at stage 3 (full power operation), greatly different conditions of operation than expected for the initial plant concept. There are not enough irradiation facilities in the EU for in-service in-pile qualification testing of the improvements proposed as applied to the changed conditions of operation. The loss of time hampers the entry of the lead cooled fast reactor technology into the commercial market. This circumstance has triggered the ALFRED core redesign for qualification tests to be undertaken for the proposed improvements at the current stage, and for the successfully tested improvements to be implemented at further stages.

For the ALFRED core redesign, the criterion the developers were guided by was the suitability of the resultant design at all three stages of the reactor evolution demonstration program, specifically at the final stage when the plant operates at full power.

The desire to keep high safety margins and the maintainability inherent in the ALFRED demonstration facility has led to the need to reduce the lead coolant mass flow rate. The developers have achieved this not only by reducing the lead velocity; they have simultaneously reduced the FA fuel grid spacing and increased the fuel rod active length to improve the neutronic performance. Besides, it was decided to install a measuring dummy rod at the center of each FA for the in-pile monitoring of operating parameters. The key changes in the core parameters are shown in Table 2.

Table 2. Key changes in the core parameters

	Previous configuration (LEADER project)	New configuration (FALCON project)
Lead mass flow rate, kg/s	≈25 694	≈17 174
Lead velocity, m/s	≈1.368	≈1.278
Fuel rod grid spacing in FA, mm	13.86	13.60
Fuel rod active length, cm	60	81

Calculations show that the core pressure drop has decreased from 1.08 to 0.78 bar, and the neutronic performance has improved thanks to an increase both in the volume fraction of fuel in the FAs (from 30.3 to 31.6%) and in the core height/diameter ratio (from 0.489 to 0.752). Actually, a longer fuel rod (see Table 2) makes it possible to reduce the number of fuel rods required to achieve the reactor rated power with the same linear load and thus reduce the required number of FAs. This leads, accordingly, to a smaller radius of the core, which, in this case, fits the inner vessel with a diameter of 2.9 m.

The FA redesigns have led to the reactor core configuration (see Fig. 6) comprising 134 FAs, 12 control rods, 4 scram system rods, and 1 dedicated position for in-pile irradiation experiments. These assemblies are surrounded by 102 dummy FAs (displacers), which form two concentric rings: the inner ring reflects the escape neutrons back into the core, and the outer ring provides radiation shielding for the inner vessel.

Calculations using the ERANOS 2.2N code (a deterministic code, European Reactor Analysis Optimized System (ERANOS)) have shown that the core configuration meets the requirements with respect to the reactor

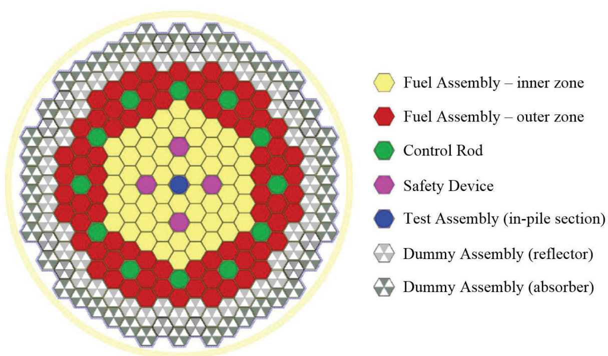

Figure 6. Cross-section of the new ALFERED core configuration (Frignani 2019b).

criticality in the course of the reactor operation, reducing the inequality between the fuel rods in terms of power density distribution and achieving the adequate counter-reactivity efficiency of the power control system and the scram system.

It should be noted that the control rods are inserted into the core from below as in the initial ALFRED design. They can be inserted passively, that is under the action of buoyancy force (as the electromagnetic latch is opened) when these rods play the role of the primary scram system.

The reactor scram system differs greatly from that adopted for the initial ALFRED design, which is currently in the process of patenting. It is not based on the absorber rod insertion into the core and starts to operate both in response to the respective control signal and on its own.

The efficiency of the control rod systems and the new scram system has been confirmed by ERANOS calculations, the results of which show the ALFRED safety margins to be rather high.

Recent years have seen many major redesigns of the ALFRED reactor aiming to accelerate the commercialization of the heavy liquid metal cooled fast reactor. A systemized description of the ALFRED reactor's improved lead coolant circulation loop is presented in Frignanti et al. 2019a, Alemberti et al. 2020.

As noted by Alemberti et al. 2020, the entire lead coolant circulation loop is accommodated in the steel vessel thanks to which the core and the pipelines connected to it remain submerged in lead in the event of the reactor vessel failure. The reactor vessel with a height of 10 m and an internal diameter of 8.3 m is a vertical cylinder with a semispherical bottom. The wall thickness is 50 mm and steel 316 LN (or L) has been selected as the material. The reactor vessel is designed to operate for the entire reactor life, that is, for 40 years as a minimum. Currently under consideration are the manufacturing and transportation issues of the reactor vessel and the guard vessel, which are the largest components.

The internal structure installed in the reactor vessel is intended to shape the lead flow, to separate the hot pool of the lead circulation loop from the cold pool, and to accommodate the retrievable leak-tight components. It is also designed for operation throughout the reactor life (> 40 years) but is not a safety component. However, it contacts the circulating lead hot pool, so it requires protection against the lead corrosive action apart from protection through the dissolved oxygen concentration control. At the present time, it is proposed that a coating of Al_2O_3 be applied to that end.

The reactor core is in the inner vessel that can be withdrawn after all fuel assemblies (FAs) are withdrawn. The service life of the inner vessel is as short as 20 years due to radiation effects. Each hexagonal FA consists of 126 fuel rods. MOX fuel has been selected as the fuel, with the concentration of plutonium in the fuel at the core center (56 FAs) being 20.5 wt. % and the plutonium content in the fuel on the periphery (78 FAs) being 26.2%. There is a test portion at the core center for irradiation experiments. The core has a control system of 12 control rods (absorber rods of boron carbide) in the peripheral part of the core and 4 safety devices on the periphery. The safety devices represent a diversified and redundant reactor shutdown system with two alternative mutually exclusive methods for introducing high-enrichment boron carbide into the core by buoyant force. There is a passive actuation system in case of the control rod system failure or a safety device actuation signal.

Heat is transferred to the system for the thermal energy conversion to electric power via three symmetrical shell-and-tube heat exchangers (bayonet tubes) submerged in the circulating lead and partially performing the containment function. One of the key advantages offered by the use of double-wall tubes is that this excludes water drops or vapor bubbles being captured by the lead flow with their ingress into the core and the respective insertion of positive reactivity. Strictly speaking, it is not altogether excluded that both walls fail simultaneously though this is highly unlikely. Additional safeguards are provided by the deflector (baffler) of the internal structure that forces lead to flow up to the free cover gas level facilitating the transfer of water or water steam into the gas space.

It is also very important for the lead circulation loop pumps to select the right material and to take into account the conditions of operation. High velocities of the fluid and periodic variations of the fluid momentum require the pumps to be protected against both corrosive and erosive lead effects. It is important in the pump selection process to select an impeller with a low hydraulic resistance in the event of the pump seizure so that no pump hampers natural circulation during an accident.

The decay heat removal system shall meet the following criteria:

- the permissible maximum hot or cold pool lead temperature not exceeded;
- retention time: the maximum time for which the lead temperature in the given pool (cold, hot) can remain above the lower limit (temperatures during normal operation, occurrence of creep phenomena) and below the maximum temperature.

This system is designed to perform the following safety functions:

- ensuring adequate cooling of the lead circulation loop following a postulated initiating event;
- delaying the lead freezing by a passive method through the natural circulation of lead in the primary loop.

The ALFRED reactor design includes the standard and emergency decay heat removal systems of different designs.

When analyzing the circulation of lead in the initial design of the ALFRED reactor, the authors of (Frignanti et al. 2019a) identified two major thermal-hydraulic problems typical of pool-type fast reactors: (i) temperature stratification in the pool's upper part, and (ii) potential capture of steam in the event of the steam generator tube failure or leak. Besides, one more problem has been identified: (iii) lead freezing risk, specifically as applied to the passive removal of decay heat during an accident.

The following has been identified as the fundamental criteria and respective design features: (i) a compact and efficient pool configuration which excludes the need for the lead circulation outside the reactor vessel; (ii) an additional guard vessel around the reactor vessel which provides a decay heat removal channel in the event of the reactor vessel failure; (iii) the layout of the reactor core and internals which allows natural circulation of lead and an increase in the grace time available for the emergency response in the event of the lead flow stoppage; (iv) jacketed hexagonal FAs which extend above the free lead level thus facilitating the FA handling; (v) hollow MOX fuel pellets which make it possible to reduce the maximum fuel temperature and reach the target fuel burn-up; (vi) two redundant reactor shutdown systems based on different principles of action; (vii) once-through steam generators without an intermediate loop which improve the economic competitiveness; (viii) axial circulation pumps installed at the core inlet that make it possible to minimize the shaft length and simplify the lead flow configuration in the pool conditions; (ix) two diversified, redundant and fully passive decay heat removal systems based on water/steam as the cooling medium capable of providing at least 72 hours available for emergency response.

To solve the above issues of the lead temperature stratification and the steam capture with insertion of positive reactivity, it has been proposed that an internal structure be installed to separate the hot pool and the cold pool, as well as to shape the lead flow from the steam generators such that it goes up to the cover gas volume boundary thus facilitating the transfer of steam bubbles or water drops into this space. A negative aspect of the internal structure installation is the inevitable increase in the reactor vessel diameter; however this effect has turned out to be within the permissible limits.

Any liquid metal cooled reactor requires that coolant freezing be avoided so as to exclude complete or partial blockage of the coolant flow path. Freezing can be caused by any thermal unbalance source. There are two mutually exclusive requirements in the case of the ALFRED reactor: (i) to ensure efficient removal of decay heat so that no fuel, cladding, reactor vessel and structural material temperature limits are exceeded, and (ii) to prevent coolant freezing. Decay heat decreases exponentially, and the passive system for removing this heat cannot be monitored or controlled during the particular available time, so unconventional solutions are required to avoid thermal unbalance.

The standard decay heat removal system operates with the connection to the steam generators, and the backup system uses a gas cylinder thanks to which it has assumed the self-control property since gas reduces the heat transfer coefficient and, therefore, the amount of the heat removed. This also reduces the pressure and establishes the thermal balance at a new level.

It should be noted that FALCON is not confined to the conceptual studies of the ALFRED design. Thus, for example, information is provided in (Frignanti et al. 2019a) on the experimental facilities built or under construction for testing and justifying solutions for this design. Two scale test facilities will be built in Romania: ATHENA (Advanced Thermo-Hydraulics for Nuclear Application facility) and ELF (Electrical Long-running Facility).

ATHENA is a pool-type experimental facility with an electrically heated FA model with the thermal power of 2.21 MW and the vessel diameter of 3.2 m and the length of 10 m. This facility is designed for: (i) R&D activities involving lead technologies (e.g., control of oxygen concentration in a large volume); (ii) integral research concerned with normal operation of the reactor (pool thermal hydraulics, functionality and performance of the steam generator, functionality and performance of the primary coolant pump); (iii) integral research concerned with safety assessments (e.g., loss of electric power, margin to lead freezing for different accident progression scenarios); (iv) full-scale testing of individual reactor components (e.g., steam generator, primary coolant pump, decay heat removal system); (v) investigation of individual safety-related effects (e.g., steam generator tube rupture, partial FA clogging); (vi) obtaining an extensive experimental framework for validation and verification of codes used in the lead cooled reactor design and licensing.

The key component of the ELF pool facility is a vessel filled with liquid lead which accommodates a core model (10 MW), four steam generators of 2.5 MW each, and a diversified decay heat removal system consisting of two heat exchangers of 500 kW each, and two prototype vertical pumps with a rated capacity of 150 m^3/h each at a temperature of 480 °C. This facility simulates the main lead movement path in the primary loop of the ALFRED reactor and is designed for prolonged experiments concerned with the future reactor operation (including coolant corrosion and chemistry).

Apart from pool-type test facilities, a vertical loop facility, HELENA-2, is planned to be built in Romania specifically for thermal-hydraulic studies for the highest power FA (2.44 MW). It consists of two vertical tubes (riser and downcomer portions) connected via two horizontal tubes. The FA model is accommodated in the riser's lower part, and the heat exchanger is in the downcomer's upper part. The ELF experiments in the natural circulation mode include bypassing of the centrifugal circulation pump. The FA power and the fuel rod number are simulated in the 1:2 scale, however such distinctive parameters as linear power density, heat flux density, fuel element diameter, bundle pitch, and others are simulated in the 1:1 scale. The following will be studied at this facility: (i) fuel element wall temperature; (ii) lead temperature in the fuel bundle cell; (iii) heat transfer coefficient; (iv) maximum temperature hot spots and points; (v) general mixed lead circulation. Following certain modifications, the facility is expected to be used to investigate vibrations caused by the lead flow.

Apart from the issues involved in the design and experimental justification of the adopted technical solutions, the ALFRED developers give much attention to developing and improving the required codes. Estimates were presented in Castelluccio et al. 2021 concerning the target accuracy of the integral parameters important to the neutronic calculation of the ALFRED reactor core, and the inverse problem was solved then based on the respective requirements to the target accuracy of nuclear data which allows to achieve the above target accuracy of the integral parameters. A two-sided approach was used to estimate the target accuracy of the integral parameters: the current level of the integral parameter uncertainty was estimated driven by the current state of the nuclear data accuracy, and the maximum permissible error was found which would make it possible to prevent the core design from being overburdened with excessive safety margins.

Uncertainty was estimated for the following integral parameters of the ALFRED reactor core: effective neutron multiplication factor (k_{eff}), coolant density effects, fuel Doppler effect (temperature coefficient), physical (reactivity) weight of control rods, effective fraction of delayed neutrons, fuel expansion effects and core local power density maximum. A deterministic code ERANOS, was selected for the calculations. As a result, the current uncertainty values concerned with nuclear data were calculated for each of the integral parameters.

Then, based on considering the required corrective actions for the underestimated or overestimated value of any integral parameter, the requirement for its target accuracy was defined. Table 3 presents the current and target uncertainties for each of these parameters. It can be seen from the table that the target accuracy increase is actually required only for the effective neutron multiplication factor, k_{eff}. An optimized solution for the inverse problem will make it possible to determine the uncertainty level for each type of nuclear data required to satisfy the target accuracy requirements for the integral parameters.

Table 3. Current and target accuracy of the ALFRED core integral parameters

Integral parameter	Uncertainty (%)	
	Current	Target
Effective multiplication factor	0.768	0.433
Control rod weight	1.08	2.5
Local power density maximum	0.264	1.33
Effective fraction of delayed neutrons	0.840	(>)
Doppler effect	2.94	(>)
Coolant density effect	14.7	(>)
Longitudinal (linear) fuel expansion	0.753	(>)

There were three different groups of the nuclear data target accuracy requirements formed by assigning weight factors to cross-sections of various reactions (Group A, Group B and Group C). It has been determined for each of these what reactions of which specific isotopes and in which energy intervals require the greatest increase in the measurement (determination) accuracy.

The fact that the target accuracy increase is required only for the effective neutron multiplication factor, k_{eff}, is explained, first, by the high accuracy of the latest nuclear data library, and, second, by peculiarities of the ALFRED facility designed to operate, due to its technological mission, with relatively high safety margins.

The solution of the inverse problem has shown, the accuracy of cross-sections for which specific nuclear reactions is most relevant for increasing the accuracy of the effective neutron multiplication factor, k_{eff}. It is interesting to note that most of the contribution could come from decreasing the uncertainty of measuring the ^{239}Pu fission reaction cross-section to the values below 1% in an energy range of 2 keV to 4 MeV; however this appears to be highly difficult to achieve (it is, rather, unachievable) in an experiment. Much greater efforts need to be applied to reduce the ^{239}Pu capture reaction cross-section uncertainty that is rather high at the present time amounting to 10 to 20%, so it is comparatively easy to reduce. It would be also perfectly good to get more accurate information on the inelastic scatter reaction cross-section for ^{207}Pb.

Occasionally, when experiments to define the ALFRED nuclear data more accurately seem to be impossible to undertake, it will be quite acceptable to use differential and integral experiments.

PLFR reactor (WLFR, Westinghouse Lead Fast Reactor)

The purpose of this project initiated by Westinghouse Electric as part of the Generation IV International Forum is to build a competitive scalable modular passive lead cooled medium reactor meeting high safety and stability standards (Ferroni 2019).

Westinghouse pursues a phased approach, according to which a prototype reactor, PLFR (Prototype Lead Cooled Fast Reactor), with an electric power of 300 MW will be built in the near future, to be used for several years to demonstrate and improve the energy technology based on HLMC fast reactors. Later, the first commercial unit with an electric power of 465 MW is expected to be built based on the demonstration stage, which will include the key solutions for the PLFR reactor but will differ from it in terms of using more advanced structural materials and fuel (Lee 2019).

At the present time, the design is at the conceptual development stage. Westinghouse collaborates with leading US and foreign organizations in the field of nuclear power plant development: ENEA and Ansaldo Nucleare in Italy, Nuclear Advanced Manufacturing Research Center, National Nuclear Laboratory, University of Manchester and University of Cambridge in Great Britain, and Fauske&Associates, Argonne National Laboratory, Oak Ridge National Laboratory, Los Alamos National Laboratory, University of New Mexico and Brigham Young University in the USA.

The major difference of the Westinghouse approach is that it aims at the following innovations (Ferroni 2019):

- *Materials capable to operate in liquid lead at temperatures of up to 650 °C.* These materials are tested in parallel with the PLFR development. Westinghouse aims to obtain the power unit efficiency values close to 50%.
- *Compact hybrid microchannel primary heat exchangers (PHX).* A compact PHX allows reducing the reactor vessel dimensions and weight.
- *Improved energy conversion system (ECS) with supercritical carbon dioxide (sCO$_2$).* The selection of such system allows reducing the turbomachinery equipment dimensions, to provide a compact layout, to increase the power unit efficiency, and to optimize the use of air as the ultimate sink for the removed heat.
- *Thermal energy accumulation system.* This system makes it possible to ensure a variable load with a nearly constant thermal power of the reactor.
- *Innovative fuel (uranium nitride) for commercial power units.* Such fuel improves the safety of units and makes them more cost-effective.

The lead temperature in the circulation loop is as follows: 420 ˚C at the core inlet, up to 530 ˚C at the outlet in the PLFR prototype, and 655 ˚C in a commercial NPP, and the pressure is close to the atmospheric pressure. The reactor vessel serves to retain the lead and is used to remove (dissipate) decay heat in the event of a failure of the normal decay heat removal system that uses the PHX. It should be noted that the reactor vessel contacts lead only in the low temperature region, thanks to which the corrosion and erosion processes slow down, this expected to extend the service life.

An innovative difference of the Westinghouse design from other lead cooled fast reactors is the use of hybrid microchannel primary heat exchangers (PHX) that transfer heat from the primary loop to the secondary loop. A schematic view of such heat exchangers used earlier in nonnuclear industries is shown in Fig. 7.

HAELU fuel (uranium dioxide with the ^{235}U enrichment from 5 to 20%) (Ferroni 2019) will be initially loaded into the PLFR prototype reactor, but the use of MOX fuel is not excluded given the experience of using it in fast reactors. Such approach, thanks to using well-studied materials and technologies, is expected to accelerate the licensing process.

Nitride uranium fuel (UN) has been selected for being loaded into the commercial reactor. This fuel will raise the unit's electric power from 300 to 465 MW with the same reactor vessel dimensions, and both safety and cost effectiveness of the plant will be improved.

Stainless steel of the 15–15Ti grade with a coating of aluminum oxide (Al_2O_3), for which there is an extensive database, has been selected for the midterm as the fuel cladding material. For a longer term, Westinghouse

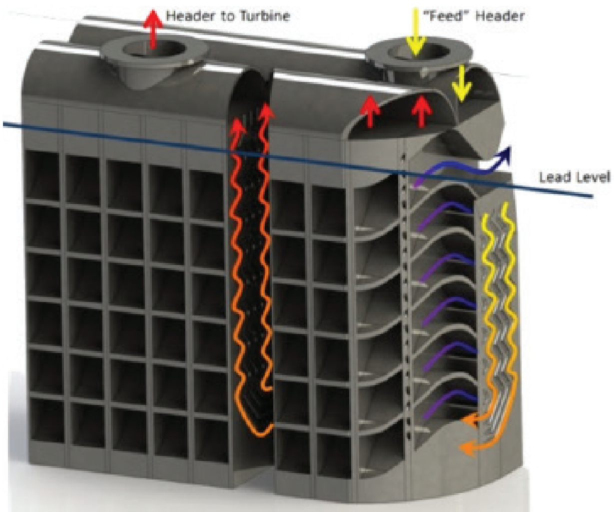

Figure 7. Schematic view of a compact hybrid microchannel heat exchanger (Ferroni 2019).

considers using other materials, such as austenitic steels with the formation of an aluminum oxide film, carbide-silicon composites, and molybdenum and niobium alloys. At the present time, the ENEA laboratories carry out studies to investigate corrosion of the candidate materials in lead at temperatures of interest for commercial power units.

The PLFR has 4 independent reactor shutdown systems. Two of these are based on the control rod insertion into the core and can be used both to control the reactor power and to shut down the reactor, and the two others are based on another principle of action which makes it possible to shut down the reactor when no control rods can be inserted due to the core deformation. The latter two systems are designed exclusively for the reactor shutdown.

The key role in ensuring the reactor safety is played by the passive decay heat removal system (PDHRS) which cools down the reactor in the event the normal decay heat removal system (NDHRS) fails. The PDHRS function is implemented thanks to the following processes:

- heat conductivity through the reactor vessel wall;
- heat transfer by radiation and convection from the reactor vessel wall to the guard vessel wall;
- heat conductivity through the guard vessel wall;
- heat transfer by natural convection and boiling to a larger water volume outside the reactor vessel;
- transition to the heat removal due to natural convection of air circulating outside the reactor vessel after water boils out.

The PDHRS operates continuously (even during normal operation of the power unit or a prolonged outage). This leads to heat losses but since the reactor vessel temperature is just 370 to 420 °C, the loss is not great (≈2 MW). It is practically inappreciable for the unit efficiency and do not threaten with the lead freezing since lead will freeze not earlier than after ~20 days even in a hypothetical case of a very prolonged outage and the auxiliary heat-up system failure.

The use of an ECS with supercritical carbon dioxide allows making the unit more compactly laid out and increasing the unit's efficiency as compared with the option using a conventional steam-water ECS. Unlike technologies, which are traditional for NPPs, Westinghouse proposes that an air-cooled condenser (ACC) be used in the ECS with supercritical carbon dioxide.

At the present time, Westinghouse develops thermal energy accumulators to level the grid loads, with the thermal energy accumulation system integrated directly with the unit turbine and the generator for a greater cost effectiveness. Such approach enables continuous full-power operation of the reactor, and the load change can be traced by increasing or reducing the working medium (sCO_2) mass flow rate through the turbine.

A concept of such thermal energy accumulators considered by Westinghouse represents a modular thermal energy accumulator in the form of a steel shell filled with the heat-transfer liquid with concrete plates stowed in it. The use of thermal energy accumulators offers a simple and cost-effective solution to the problems of grids that include NPPs operating steadily in the base load mode, and alternative electricity sources the power of which depends on the time of the day or the wind speed.

As shown in Table 4, Westinghouse plans to start building a prototype PLFR reactor around 2030.

Table 4. Key stages of the Westinghouse program to build the lead cooled reactor technology

Commencing date/Stage content	
2015	Prior consideration of innovative reactor technologies, selection of a lead cooled fast reactor for further development
2017	Completion of preconceptual design, adoption of a new pool-type design
~2030	Commencement of the full-scale prototype reactor construction (PLFR) and its further operation for technology demonstration
~2035	Transition from the PLFR to a commercial unit reactor, start of the reactor operation

Westinghouse, Argonne National Laboratory (ANL) and Fauske & Associates, LLC (FAI) are jointly developing the software to calculate the radionuclide transport and escape into the environment during different design-basis accidents at liquid metal cooled fast reactors (Lee 2019). The codes, SAS4A/SYSSYS-1 (ANL developed) and FATE (FAI developed), are used as the basis. The former simulates transients and fuel damage and the latter simulates the radionuclide transport into the primary coolant, into the gas cavity and into the containment and further escape into the atmosphere beyond the NPP site. The basis for the code integration was a good agreement of the simulation results for the lead cooled fast reactor thermal hydraulics.

The RRM (Radionuclide Release Module) module was built as part of these activities to simulate the escape of radionuclides from overheated fuel and their retention in the coolant. The RRM was validated based on experiments undertaken in the USA, France and Japan.

The capability of the SAS4A-FATE integral code to simulate the fuel heat-up and failure and the radionuclide escape into the fuel and their further transport in the lead coolant, in the gas cavity and in the containment was demonstrated based on an example of an accident with an uncontrolled power growth. Further studies will require experimental data to identify the composition and distribution of radionuclides in the lead coolant.

One of the mandatory tasks pursued by developers of any NPPs is to build an effective decay heat removal system (DHRS). A number of advanced DHRS concepts were analyzed by Westinghouse experts for the WLFR development, and the PDHRS concept with an air cooled reactor vessel has been opted for (Liao and Utley 2020). This concept is shown schematically in Fig. 8.

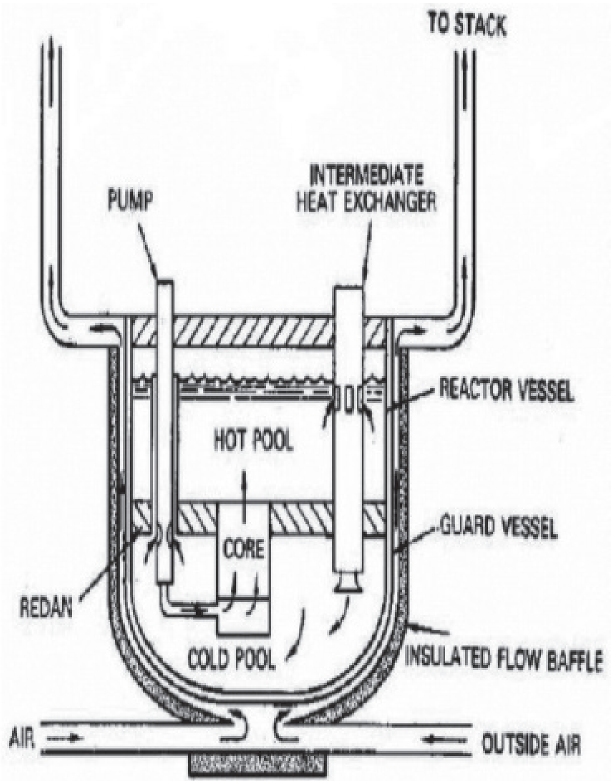

Figure 8. Schematic of the lead cooled pool-type reactor vessel air cooling concept (Liao and Utley 2020).

The selection of this concept is typical of liquid metal (sodium or lead) cooled reactors and some water cooled reactors (e.g., AP600^{TM} and AP1000). Its key advantages are as follows:

1. *Passivity* – does not require operator interference, there are no moving or active components (pumps or valves), and it is enough to open the venting holes during thermal expansion for the DHRS actuation.
2. *Self-controlled decay heat removal* – self-control is representative of air cooling, since one of its key mechanisms is heat transfer from the reactor vessel to the guard vessel by radiation, which depends heavily on the absolute wall temperature values for these vessels raised to the fourth power. In normal operating conditions, this mechanism leads only to

minor thermal energy losses. Besides, self-control makes it possible to delay the threat of the primary loop freezing if the primary coolant has a high melting temperature, e.g., as lead.
3. *Unlimited heat sink capacity* – unlike systems that require the addition of cooling agent after some time, air cooling in this case is based on the heat flow to the air environment which actually represents an unlimited heat sensing capacity.

Alongside, the drawbacks of this concept need to be noted:

1. *Limited heat removal capability* – cooling is effected through heat transfer by radiation (from the reactor vessel to the guard vessel) and by convective heat transfer from the latter to the air in the deflector, both mechanisms being characterized by a high thermal resistance. In addition, the contribution of heat transfer by radiation at a low temperature of the reactor vessel is small since it is determined by the fourth power of the absolute value of this temperature.
2. *Heavy dependence on local conditions* – air cooling is effected through the natural circulation of air through the deflector and the stack. The driving head is the difference in the air density values, which depends on the local onsite weather conditions including the following: bulk air temperature, side wind and reactor room temperature. Besides, an important role is played by hydraulic resistance in the deflector and in the ventilation path.
3. *Heat loss* – a heat loss occurs if the vessel cooling system is designed such that it is opened in the reactor normal operating mode. Thanks to a low temperature level in this mode, heat losses are limited but they cannot be viewed as negligibly small, and are expected to amount to ~ 0.36% of the reactor core thermal power.

To analyze the efficiency of the PDHRS concept under consideration, three codes were used in parallel: a simplified standalone Westinghouse-developed code, the WCOBRA/TRAC-TF2 circuit code, and the SAS4A/SYSSYS-1 circuit code. These codes differ in complexity, the validation level and, specifically, the convenience of undertaking “fast prototype” assessments. The standalone code suits best for the initial rapid assessment with respect to the influence of the geometrical parameters (including the reactor vessel size), the SAS4A/SYSSYS-1 code suits best for calculations at the most advanced and complex design stages, and the WCOBRA/TRAC-TF2 code, in this respect, is intermediate between them.

The results of calculations using the standalone code and the WCOBRA/TRAC-TF2 code satisfactorily agree with the experimental data obtained earlier based on the NSTF test facility at ANL. These codes are suitable for rapid assessments at the initial conceptual design stages preceding a more complex analysis using the SAS4A/SYSSYS-1 code. These three codes have been developed independently of each other, so the close agreement of the results from using these raises their level of confidence.

Westinghouse plans (in accordance with the PIRT-process results) to build a facility for experiments to justify the development of the PDHRS design with air cooling of the WLFR reactor vessel, and to go on, in parallel, with improving the air cooling model in the loop codes with further verification based on earlier experimental data and based on results of the experiments at the planned facility. This will make it possible to observe the qualification requirements and support the WLFR licensing.

The above PIRT (Phenomena Identification and Ranking Table) process was developed in its time for the PWR when analyzing the loss of coolant accident with a major pipe break. In Liao et al. 2021, the authors used this efficient method to develop an integrated safety analysis methodology, which encompasses the code development, the model development and the experimental verification. The PIRT process comprises the following processes:

- Identification of the characteristic quantity (CQ);
- Identification of all phenomena that may affect the CQ;
- Estimation of the phenomenon’s relative importance for the CQ, including justification;
- Estimation of the phenomenon’s relative exploration status (ES) , including justification;
- Grouping of phenomena in terms of importance and the ES (low, moderate, high);
- Documenting.

The following accidents have been selected by an international expert team for the PIRT process as applied to the WLFR analysis:

- Postulated design-basis accidents:
    - station blackout (SBO);
    - transient overpower (TOP);
    - failure of the inter-loop heat exchanger (PHX)
- Hypothetical beyond-design-basis accidents:
    - unprotected (reactor protection system failure) transient overpower (UTOP);
    - unprotected (reactor protection system failure) station blackout (USBO).

Initially, phenomena with a high importance level and a low (or moderate) ES level are chosen from the results of using the PIRT process for the WLFR safety analysis. To raise the ES level for important phenomena, a matrix of experiments (special, integral and laboratory types) is developed, which is an integral part of the preliminary WLFR development stage. The PIRT process has also confirmed the practicability of the further SAS4A/SYSSYS-1 code development for considering phenomena with a high level of importance and a low (or moderate) ES level.

The PIRT process has revealed the following group of phenomena with a high level of importance and a low (or moderate) ES level based on an example of the postulated design and hypothetical beyond-design-basis accidents selected above for the WLFR:

- Design-basis accidents:
  - reactivity response to the radial core expansion;
  - a transient with the PDHRS switchover from water cooling to air cooling;
  - heat exchange during free convection of the cooling air in the PDHRS;
  - efficiency of the passive reactor emergency protection system;
- Beyond-design-basis accidents:
  - chemistry of radionuclides in the lead coolant;
  - onset of the fuel melt movement inside the fuel element (if fuel has melted);
  - reactivity effects from the fuel melt movement inside the fuel element (if fuel has melted);
  - dispersion of the fuel pieces and re-criticality;
  - solid fuel interaction with lead coolant;
  - movement of fuel melt and/or fuel pieces;
  - interaction of melted fuel with the fuel cladding and the jacket wall (if fuel has melted).

MYRRHA ADS (Multi-purpose hYbrid Research Reactor for High-tech Applications) program

Belgium has been implementing a multipurpose hybrid research program, MYRRHA ADS, to develop a lead-bismuth eutectic cooled fast reactor (De Bruyn 2019). It was since the time of the program development in 1998 that the Belgian Nuclear Research Center (SCK-GEN) initiated a large volume of R&D to support the program for the purpose of justifying a range of solutions, including those involving the use of lead-bismuth eutectic (LBE) as the reactor coolant and the proton beam target. In 2008, the government of Belgium made a decision to fund the construction of the first ADS and its operation as from 2038. The construction of the infrastructure is planned to be started in 2026, and full-scale operation of the MYRRHA ADS is expected to be started in 2036 (De Bruyn 2019).

The rated thermal power of the MYRRHA reactor is 100 MW. It is brought out of the subcritical state by the proton beam of a linear accelerator (LINAC) with the proton energy of 600 MeV and the current intensity of 2.5 A. The reactor is capable to operate both in a subcritical state and in a critical state. In 2005, Belgium and SCK-GEN opened the MYRRHA ADS program for the EU member countries, as well as for the leading countries in the world nuclear community to get them involved in the program development and in the subsequent construction and operation of the MYRRHA ADS.

The MYRRHA ADS is developed to:

a. Test and implement transmutation of long-lived and most toxic spent nuclear fuel (SNF) radionuclides to reduce their amount (by a factor of 100) and to reduce the half-life (from hundreds of thousands of years to several hundred years). Transmutation has a positive effect both on safety and on the SNF handling economy;
b. Produce medical radioisotopes;
c. Examine and test materials for existing and future nuclear reactors and thermonuclear facilities;
d. Build a multifunction proton accelerator for fundamental and applied research.

The following staged approach was adopted as a result of analyzing different MYRRHA ADS program implementation options:

stage 1 – the accelerator with a proton energy of 0 to 100 MeV;
stage 2 – the accelerator with a proton energy of 100 to 600 MeV;
stage 3 – the reactor.

This suggests that stage 3 can be implemented in parallel with or after stage 2.

Over the time since the program commencement, SCK-GEN has built and put into operation a number of experimental facilities to investigate LBE related issues:

- **HELIOS3** – in operation since 2013; designed to prepare the LBE melt for other facilities, investigate the LBE preparation methods and study the consequences of the steam or water ingress into LBE;
- **MEXICO** – in operation since 2014; a specially designed circulation loop with 7 tons of LBE melt intended to investigate the dissolved oxygen concentration control in LBE and LBE filtering;
- **CRAFT** – in operation since 2014; a medium-scale facility with 6 tons of LBE melt intended for experimental studies of corrosion in LBE with well-controlled flow parameters (flow rate, temperature and chemistry) at a temperature of 270 to 500 °C;
- **LIMETS** – a facility for various investigations with structural materials in LBE melt (fatigue, tensile strength, crack resistance), including using irradiated samples;
- **RHAPTER** – in operation since 2011; designed to study the behavior of mechanical (moving) components (bearings, gear wheels and power cables);
- **E-SCAPE** – in operation since 2017; designed to investigate the liquid metal thermal hydraulics in the pool reactor (the MYRRHA reactor vessel in a geometrical scale of 1/16);
- **COMPLOT** – in operation since 2014; designed to investigate the hydraulic and hydrodynamics of the MYRRHA reactor components and is of special

interest since its components (FAs, spallation model, reactivity control and scram rods) have been made in a scale of 1:1.

The following has already been studied at the COMPLOT test facility:

- FA pressure losses;
- control rod hydrodynamics;
- hydrodynamically excited FA vibrations;
- thermal hydraulics in the FA vicinity.

It should be noted that the MYRRHA reactor and the research facilities involved in its development, especially COMPLOT, may be of interest as well for the lead cooled fast reactor technology. In particular, this relates to the small modular fast reactor since its components are dimensionally similar to the MYRRHA reactor components.

The key issue involved in considering the MYRRHA reactor's thermal-hydraulic performance and assessing the reactor safety is concerned with a complex pattern of the coolant flow in the primary loop with open cold and hot spaces and pronounced 3D phenomena that may lead to accidents, such as LOFA (loss of flow accident). One-dimensional loop thermal-hydraulic codes (system thermal hydraulic codes), such as STH, normally used for the transient analysis and the nuclear unit licensing, may turn out inefficient for taking into account and displaying the above phenomena. At the same time, CFD codes have been increasingly used in nuclear power thanks to their capability to simulate perfectly well complex flows and thermal-hydraulic phenomena. These codes however require large volumes of computer memory and CPU time, which complicates their practical use for the integral simulation of large systems.

A coupled STH/CFD model of the MYRRHA reactor, combining the capabilities of the RELAP5-3D unidimensional circuit thermal-hydraulic code and the FLUENT 3D CFD code, has been developed to represent realistically these 3D effects in operation (Toti et al. 2018). The resultant methodology is based on decomposing the computational domain and an innovative implicit numerical scheme.

One of the key objectives pursued by that paper is to compare the transient (LOFA) analysis results using a standalone thermal-hydraulic code and coupled simulation. A model of the MYRRHA reactor primary loop has been developed to that end as part of this study. The lower and upper zones in the model are characterized by a pronounced 3D pattern of the coolant flow, so they are simulated by CFD codes. The rest of the primary loop (core, intermediate heat exchangers, circulation pumps) and the secondary and tertiary loops are simulated using a circuit thermal-hydraulic code.

After the circulation pumps are shut down and the scram rods are inserted into the core, natural circulation is established in the MYRRHA primary loop, and the reactor's secondary and tertiary loops continue to operate normally. Coupled simulation makes it possible to simulate more reliably this natural circulation expected to provide the decay heat removal. The results of the coupled STH/CFD simulation agree well with the standalone calculation using the RELAP5-3D unidimensional circuit thermal-hydraulic code. Disagreements have been recorded only for local temperatures in the complex mixing zones, e.g. in the event of the flow reversal at the intermediate heat exchanger inlet. It should be noted that it is proposed that the developed coupled simulation methodology be validated based on the results obtained at the experimental facility.

The MYRRHA project implementation program provides for a linear accelerator to be built at stage 1 for the proton beam generation to control the criticality of this ADS reactor. The proton beam quality factor and emittance are defined to a large extent by an injector consisting of a 4-rod high-frequency quadrupole (RFQ), two quarter-wave comb accelerating structures (QWR) and 16 normally conducting CH-type resonators (the so-called drift tubes). At stage 1, the MYRRHA injector will consist of an ion source, a high-frequency quadrupole (RFQ), two quarter-wave comb structures (QWR) and the first seven CH-resonators. Information on the status of the resonator-related activities is provided in (Kumpel et al. 2018).

As noted by the authors hereof, the CH-1 resonator was manufactured by NTG, a German company, and installed for further studies at the experimental room of IAP, also a German company, after a copper coating was applied at Galvano-T, Germany. The resonance frequency values measured prior to and after the copper coating application agree well with the calculation made with the STEP model for the resonator without the copper coating. The CH-2 resonator has already been manufactured by PINK GmbH, Germany, and is prepared for the copper coating application. A good agreement has also been obtained for this between the resonance frequency measurement and calculation but so far without the copper coating.

Very important for the high reliability of the CH resonators is to ensure that they are cooled effectively. In connection with this, a decision was made to develop a new cooling system with additional channels for the resonators and to manufacture resonators CH-3 through CH-7 with the same cooling system as for CH-1 and CH-2.

The investigation and testing of the CH-1 and CH-2 resonators will be completed during next year, and the tender procedure has been started for the fabrication of the next five resonators. It has also been noted that resonators CH-8 through CH-15 have already been thermally simulated.

Chinese CLEAR reactors

China develops a family of the CLEAR heavy liquid metal (lead-bismuth eutectic or lead) cooled fast reactors. The reactor family consists of the CLEAR-M, CLEAR-I and CLEAR-A reactor facilities for a variety of applications (Wu 2018a).

The CLEAR-M reactor is designed for a small modular nuclear power plant, and CLEAR-10 of 10 MW(el), the characteristics of which are given in Table 5, is considered as an example (Ali et al. 2020).

It is planned initially to build the CLEAR-M10a reactor of the thermal power 10 MW. Its detailed design has already been developed but it will use LBE with the core outlet temperature of 380 °C rather than lead as the primary coolant. Stainless steel of the 15–15Ti grade has been selected as the fuel cladding material as in the above Westinghouse designs.

Since 2011, the Chinese Academy of Sciences has been developing the ADS for RW and SNF transmutation. The CLEAR-I pool-type reactor (Fig. 9) is expected to be used as the ADS reactor component at the initial stage. LBE with a weight of 600 t is used as the primary coolant, and pressurized water circulates in the secondary loop. Noteworthy, the CLEAR-I is not designed for electricity generation.

The CLEAR-I fuel is UO_2 with an enrichment of 19.54%, and the core is designed to operate in two modes (critical and subcritical). Heat is transferred from the primary loop to the secondary loop via four heat exchangers submerged in a pool containing LBE coolant. Two circulation pumps are used for the LBE circulation.

The next ADS development stage is the system improvement such that it generates electricity in addition to the RW and SNF transmutation. The CLEAR-A subcritical reactor of the travelling wave type is planned to be developed to that end. The characteristics of this reactor are presented in Table 6, and the reactor schematic is shown in Fig. 10.

It should be noted that the travelling-wave reactor concept has been strongly criticized by many experts since it suggests disposal of RW and SNF with a high content of plutonium. As reported by media (*China Daily*), China National Nuclear Power Co Ltd (CNNP), a Chinese company, announced in the autumn of 2017 its plans to establish a subsidiary to develop the travelling-wave reactor technology.

Table 5. Key characteristics of the CLEAR-M reactor

Description	CLEAR-M
Electric power	10 MW(el)
Cycle efficiency	>40%
Primary coolant	lead
Core outlet coolant temperature	>500°C
Secondary coolant	supercritical CO_2
Fuel	UO_2/MOX
Dimensional requirements	transportability
Refueling interval	≥5~10 years

Table 6. Key characteristics of the CLEAR-A reactor

Description		CLEAR-A
Power		400 MW(th)/150 MW(el)
External neutron source	Accelerator	Proton cyclotron (900 MeV/10 mA)
	Target	Pb
Neutron generation intensity		~1.1×10^{18} (n/s)
k_{eff}		0.97~1
Primary coolant		Pb
Nuclear fuel		U-Zr
Design life		60 years

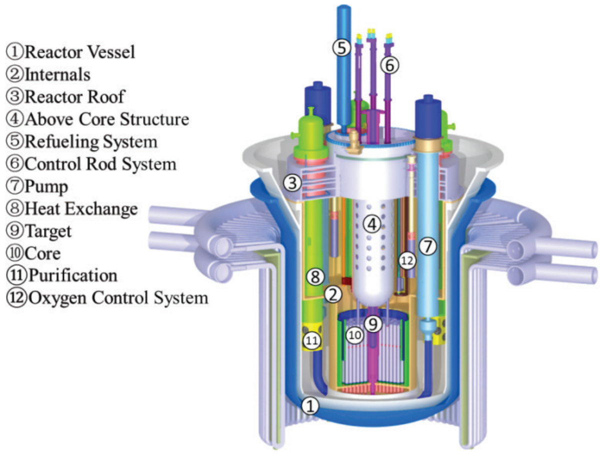

Figure 9. Overall view of the CLEAR-I reactor (Wu 2018a).

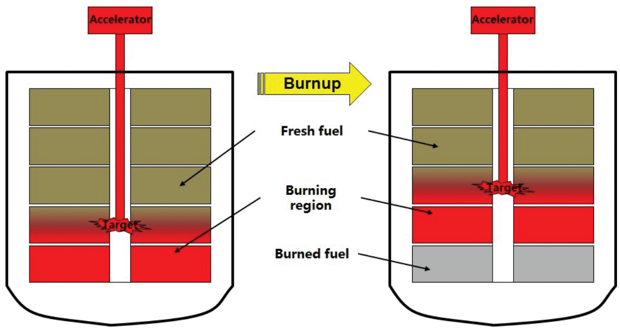

Figure 10. Schematic of the CLEAR-A (Wu 2018a).

By now, a great amount of R&D work has been undertaken on the key process solutions for the CLEAR-A reactor justification, and all key components have been manufactured and tested: the primary coolant pump, the heat exchanger, the CPS drive and the refueling system. Technologies have been developed for LBE melting and flow cleanup, for monitoring and control of dissolved oxygen concentration in LBE, and for cleaning LBE from ^{210}Po. A cold trap and a magnetic trap have been developed for the LBE flow cleaning of impurities. The developed oxygen detectors (Pt/air, Bi/Bi_2O_3, Cu/CuO) and the LBE dissolved oxygen concentration control system (gas-phase and solid-phase) have proved to be serviceable and provided for the stable state of the coolant for more than 6 000 hours. Apart from ordinary stainless steel, a new graphene-based composite was used as the filtering material to clean LBE from ^{210}Po.

As is known, in a transient with a decrease or loss of the coolant flow (LOFA, or loss of flow transient accident), a pool configuration of the reactor can cause potentially a temperature stratification of the coolant leading to fatigue effects in the reactor vessel and the components it accommodates. This phenomenon was investigated numerically in Ali et al. 2020 as applied to the world's largest (as of 2021) nonnuclear experimental facility, CLEAR-S, (Wu 2018b), the vessel of which contains an electrically heated model of the core and the CLEAR reactor prototype components submerged in the lead-bismuth eutectic melt.

FLUENT, an ANSYS code, was used for this study. The calculations used a standard k-ϵ turbulence model with a standard wall function that has turned out to be the most effective one, as compared to other models, with a fairly acceptable accuracy. A porous body model has been selected to determine the pressure loss in different CLEAR-S components (inter-loop heat exchanger, core model, circulation pump, decay heat removal system).

The obtained numerical model was verified by varying the time step and the iterations at each step. With a step of 0.01 sec and 40 iterations per step, the solution was no longer dependent on the time step size and the Courant criterion became less than unity.

Initially, steady-state solutions were obtained for 100, 50 and 25% of the rated coolant flow rate, which served as the initial conditions for the temperature stratification of the coolant. At t = 0, the circulation pump shuts down, natural circulation is switched over to, the core model power drops to 7% of the rated power (transition to decay heat), the inter-loop heat exchanger trips, and the decay heat removal is activated. Calculations were done only up to t = 1000 sec due to time and funding restrictions.

Key findings:

a. Temperature stratification takes place in the coolant's cold and hot pools during a LOFA accident.
b. With all initial LBE coolant flow rates (100, 50 and 25% of the rated value), there were 5–6 layers with different temperatures formed in the hot pool, which differ from the adjoining layers in temperature by 5 to 7 K, 11 to 12 K and 24 to 25 K respectively. In other words, the LBE temperature increased on both sides of each layer as the initial coolant flow rates decreased. With time, the stratification layer in the hot pool went up, and the upper layer width decreased until the layer disappeared in full after which a new layer formed at hot pool's lower end.
c. The stratification started at the decay heat removal system's outlet in the cold pool with the initial flow rate of 100% of the rated value. The stratification layer formed at the bottom of the cold pool and went up after some time, as the temperature drop over the layer was ~10 K. With the initial flow rate values of 50 and 25% the stratification started at the inter-loop heat exchanger. The temperature drop over the layer was 17 and 27 K depending on the flow rate. Therefore, the temperature drop over the stratification layer in the cold pool grows as the initial flow rate value decreases. With time, the stratification layer goes up until it disappears. The lowest layer stabilizes throughout the estimated time at different levels, depending on the initial coolant flow rate values.

A distinctive feature of the reactors included in the ADS is that there is a spallation model at the core center, which extremely complicates the refueling using an ordinary vertical grip. Thus, for instance, the spallation target in the CLEAR-1 reactor is attached to the end of the tube via which the proton flux comes from the accelerator and which cannot be removed for the refueling time. As compared to a vertical grip, a grip of different design is required, and it is supposed to meet a whole range of requirements including the following: resistance to corrosive and thermal effects of the LBE coolant, operation in buoyancy prevalence conditions, capability to transfer FAs with a dead matter of depleted uranium, remote use in a high-temperature opaque medium. These requirements are hard to comply with for the CLEAR-I reactor refueling system, including the FA grip.

The MYRRHA reactor design envisages refueling from below, where the proton-conducting tube does not pose a major obstacle for the L-shaped refueling grip. Such approach is however associated with a major increase of the reactor unit dimensions.

The FA grips in PWR reactors use spring actuation mechanisms and position sensor probes. These springs, likewise the sensors, are however vulnerable to the aggressive impact of the LBE coolant.

With regard for the above circumstances and the FA shapes, an in-vessel cantilever-type grip has been proposed for the CLEAR-1 reactor (see Fig. 11 for the schematic).

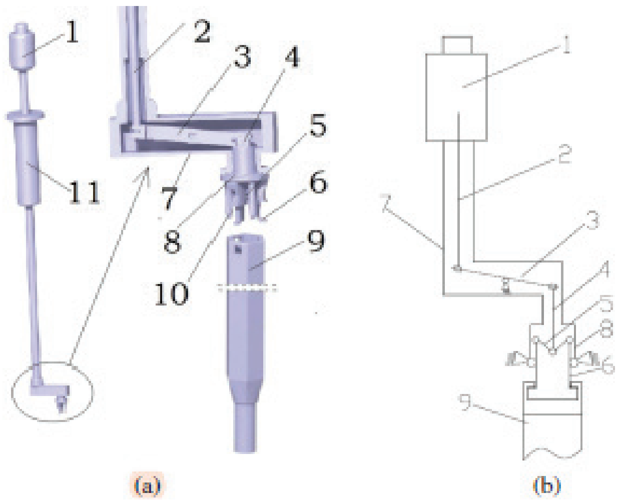

Figure 11. Cantilever-type grip for the CLEAR-1 reactor (Zeng et al. 2018) (a) Structural Model, and (b) Simple Structural Diagram]: 1 – motor; 2 – spindle; 3 – lever; 4 – connecting rod; 5 – tie rod; 6 – hitch; 7 – external bushing; 8 – retainer; 9 – FA; 10 – pin; 11 – guide cylinder.

A structural analysis and kinematic simulation have shown that the grip moves smoothly and flexibly in the working range of parameters. It retains reliably the FA head and ensures that the FA moves without jolting. The selection of the 316L-type stainless steel ensures high mechanical strength of the grip and its resistance to the temperature and corrosive effects of the LBE coolant. The obtained results can be useful in developing a refueling robot for liquid metal cooled reactors, specifically in the case of ADS.

Conclusion

Liquid metal cooled fast reactors have been identified by the Generation IV International Forum as one of the promising trends for the evolution of the world's nuclear power. During the past decade, however, the evolution of this energy technology was a top priority only in Russia for which purpose Rosatom State Corporation had accumulated the scientific and engineering potential in the *Proryv* project. A graphic evidence of this leadership has been the commencement of construction of the lead cooled BREST-OD-300 reactor (first concrete was poured for the reactor foundation on 8 June 2021).

The achievements of Rosatom in this field led to growing interest on the part of foreign counterparts as well, which has been clearly shown in recent years by the growth of the publication activities on the topic.

As one can judge from publications, international consortiums for the development of HLMC reactors (ALFRED in Romania, PLFR in the USA and MYRRHA in Belgium) do not expect any unsolvable technical issues and formulate requirements to experimental facilities and candidate materials and technologies needed for further activities.

Of special note is the development status of the Chinese CLEAR reactor family. Extensive development activities are under way encompassing electricity generation and transmutation of RW and SNF. A very large volume of R&D has been undertaken and large-scale nonnuclear experimental facilities are under construction to justify technical solutions adopted for the reactor designs.

The feasibility of a liquid metal cooled fast reactor was demonstrated several decades ago. The investigations undertaken since that time confirm that it is possible to build a large NPP. Certain issues (e.g., fuel cycle closure, reliability, maintenance) can be evidently investigated only at an operating experimental power unit. The possibility for switching to large-scale nuclear power with a closed fuel cycle based on a proven NPP is driven, primarily, by economic considerations, including the actual cost of fuel supplies for operating NPPs of all types. Bearing in mind that "politics is the concentrated expression of economy", the environmental considerations, and the issues of the public perception of nuclear power and even of nonproliferation have been addressed in one way or another depending on the cost of nuclear electricity as compared with alternative sources in given regions and countries. Therefore, provided there is confidence in resolving the scientific and technical problems of the fuel cycle closure using fast reactors, the focus in the future engineering activities shall be on a commercial NPP that can be economically advantageous as compared with alternative options.

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