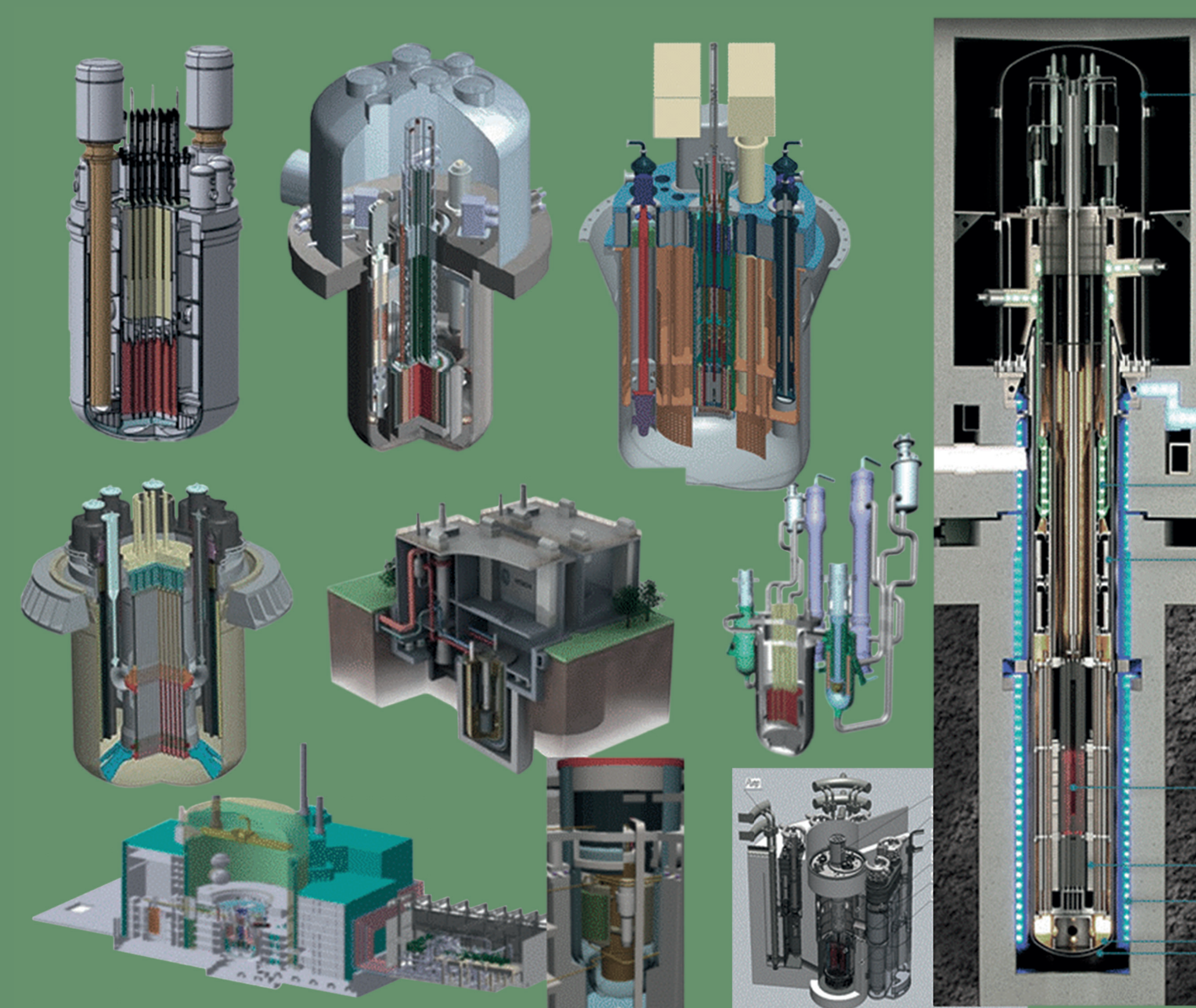
STATUS OF INNOVATIVE FAST REACTOR DESIGNS AND CONCEPTS

A Supplement to the IAEA Advanced Reactors Information System (ARIS)

http://aris.iaea.org



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International Atomic Energy Agency

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PREFACE

The important role of fast reactors and related fuel cycles for the long term sustainability of nuclear power has long been recognized. By the achievable breeding ratio and the multi-recycling of the fissile materials obtained from the spent fuel, fast reactors allow fully utilizing the energy potentiality of the natural resources (Uranium and Thorium), thus guarantying energy supply for thousands years and drastically enhancing the sustainability of nuclear power, especially in terms of resource preservation and management of high level and long-lived nuclear wastes.

A successful large-scale deployment of fast reactors is reasonably achievable only if the research and technology developments create the conditions to accomplish the full potential of the fast neutron systems and related closed fuel cycles, and if the criteria of economic competitiveness, stringent safety requirements, sustainable development, and public acceptability are clearly satisfied.

In this context, it is of paramount importance the understanding of different design options and related safety characteristics, based on past knowledge and experience, as well as on new scientific and technological research efforts. Since 1960 significant fast reactor development and deployment programmes have been pursued worldwide, bringing the knowledge on fast reactor and associated fuel cycle technologies to a high level of maturity. Furthermore, nowadays a new generation of fast reactors is being developed at national and international level in order to comply with higher standards of safety, sustainability, economics, physical protection and proliferation resistance. Member States, both those considering their first nuclear power plant and those with an existing nuclear power programme, are interested in information about innovative fast reactor designs and concepts, as well as new development trends.

The International Atomic Energy Agency (IAEA), which has been accompanying and supporting the development of fast reactors and related fuel cycles for almost fifty years, plays a prominent role, representing for the interested Member States a major international forum for scientific and technical cooperation in this field. Furthermore the IAEA regularly publishes technical reports which provide comprehensive and detailed information on the different aspects of the fast neutron system science and technology.

The objective of this booklet is to provide Member States with an overview of the status of innovative fast reactor designs and concepts that are currently under development. The report is intended as a supplementary booklet to the IAEA's Advanced Reactor Information System (ARIS), which can be accessed at http://aris.iaea.org. It is also to be regarded as a complementary publication to the recently released *Status of Fast Reactor Research and Technology Development* (IAEA-TECDOC-1691).

The IAEA officer responsible for this publication was S. Monti of the Division of Nuclear Power.

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INTRODUCTION

Background

The International Atomic Energy Agency (IAEA) devotes a number of its initiatives in support of the development and deployment of innovative fast reactors (FR) and the related closed fuel cycles, recognizing their importance in ensuring the long term sustainability of nuclear power. The objective of this booklet is to provide Member States, both those considering the initiation of nuclear power programmes and those already having practical experience in nuclear power, including the ones with an active FR programme, with a brief introduction to the IAEA's Advanced Reactor Information System (ARIS) by presenting a balanced and objective overview of the status of innovative FRs. The designs and concepts presented in the booklet were mostly derived from: a) the recently published *Status of Fast Reactor Research and Technology Development* (IAEA-TECDOC-1691), b) the proceedings (in preparation) of the *International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios* (FR13), held in Paris on March 4-7 2013, c) presentations/consultations during the last meetings of the Technical Working Group on Fast Reactors (TWG-FR), and d) several interactions with the developers of the different FR concepts indicated by the respective Member States. The scope of the publication covers briefly the history of fast reactors, from the first experimental reactors and prototypes up to nuclear power plants that are currently under construction, for leading to present on-going initiatives on innovative fast reactor designs and concepts, followed by short but informative design descriptions.

Fast reactors have the potential to enhance sustainability indices and correspond to cross-cutting prerequisites for large scale introduction of nuclear energy in the world's energy mix, including economics, safety, natural resource utilization, reduction of waste, non-proliferation and public acceptance [1]. Fast neutron reactors differs from thermal reactors in the possibility to achieve the breeding ratio, meaning that during the operation they convert fertile material (Uranium-238) into fissile material (Plutonium-239) at a rate faster than they consume the fissile material by fission reactions. In the typical fast reactor fuel cycle, after irradiation, the reprocessing of the spent fuel allows extracting this fissile material which can be reused to produce fresh fuel. In this way, fast reactors allow fully exploiting the intrinsic energy potential also of the fertile Uranium, which represents the 99.3% of the natural resource, and thus to extract sixty-to-seventy times more energy from uranium than thermal reactors do. Furthermore, the physical characteristics of the fast neutron spectrum offer the possibility to burn more efficiently the long life transuranic wastes, reducing the amount of high level wastes, the heat load to the geological disposal as well as the required isolation time for from tens of thousands of years to hundreds of years for fission products only. In short, by improving the utilization of natural resources and reducing the amount and radiotoxicity of radioactive wastes, fast reactors offer significant benefits in making nuclear energy production more sustainable.

Fast reactors have been under development for many decades: primarily conceived as breeders, in recent years their development has also focused on burning high level waste. The closed fuel cycle that has been demonstrated and the experimentally confirmed effective breeding ratio for these reactors encourages future deployment.

In summary, fast reactors promise a large variety of advantages: at first place, and in addition to safe and economical electricity production, they can utilize uranium resources to the greatest extent possible due to the breeding ratio achievable in a fast spectrum, which allows exploiting the energy potential of the fertile nuclides contained in the natural uranium resources. Furthermore, radioactive waste is significantly reduced, also thanks to the fact that Minor Actinides (MA), which represent the long-lived component of high-level nuclear waste, are produced at a lower rate and are even consumed through fission. Finally, fast reactors deal successfully with the issue of 233U production with low 232U production in a Uranium-Thorium fuel cycle. These facts, in the long-term perspective, support the acceptability and worth of nuclear energy in terms of providing sustainable energy security as well as clean environment [2].

Major Fast Reactor Options

There are four major fast reactor options, and innovative designs of each one of them are presented in this Booklet: the sodium-cooled fast reactor (SFR), the heavy liquid metal-cooled (HLMC) fast reactor, the gas-cooled fast reactor (GFR), and the molten salt fast reactor (MSFR).

The sodium-cooled fast reactor (fig. 1) features a fast-spectrum, sodium coolant, and a closed fuel cycle for efficient management of actinides and conversion of fertile uranium. The SFR can be also designed for management of plutonium, minor actinides and, in general, high-level wastes. Important safety features of the system include a long thermal response time, a large margin to coolant boiling, a primary system that operates near atmospheric pressure, and an intermediate sodium system between the radioactive sodium in the primary system and the water and steam in the energy conversion system. With innovations to reduce capital cost, the SFR can serve markets for electricity and, in the future, non-electrical applications. The SFR's fast spectrum, together with fuel reprocessing and re-fabrication technology, also makes it possible to use available fissile and fertile materials (including depleted uranium) considerably more efficiently than thermal spectrum reactors with once-through fuel cycles [3].

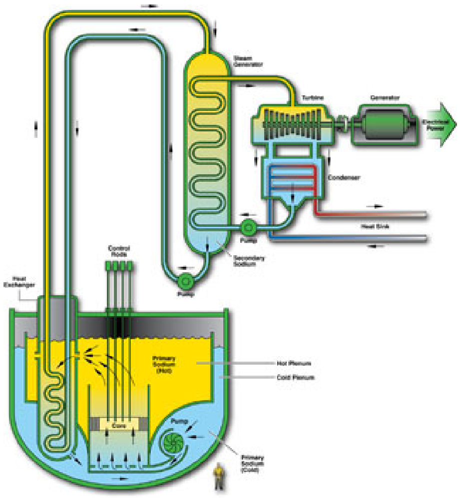


FIG. 1. Schematic representation of SFR

For heavy liquid metal-cooled fast reactors (fig. 2), there are two possible coolant materials: the first is pure Lead (Pb) and the second is lead-bismuth (Pb-Bi) eutectic (LBE) which is composed of 45.0 at% Pb and 55.0 at% Bi. The lead-cooled fast reactor system (LFR) features a fast-neutron-spectrum, lead or lead/bismuth eutectic liquid coolant, and a closed fuel cycle for efficient conversion of fertile uranium and management of actinides. Two LFR systems have been conceived: a more conventional medium-large size plant for grid-connected power stations and a small plant operating on a closed fuel cycle with a very long refueling interval (15 to 20 years) cassette core or replaceable reactor module, designed to meet market opportunities for electricity production on small grids, and for developing countries that may not wish to deploy indigenous fuel cycle infrastructure to support their nuclear energy systems. The LFR system is designed for distributed generation of electricity and other energy products, including hydrogen and potable water. The fuel is oxide, metal or nitride-based, containing fertile uranium and transuranics [4].

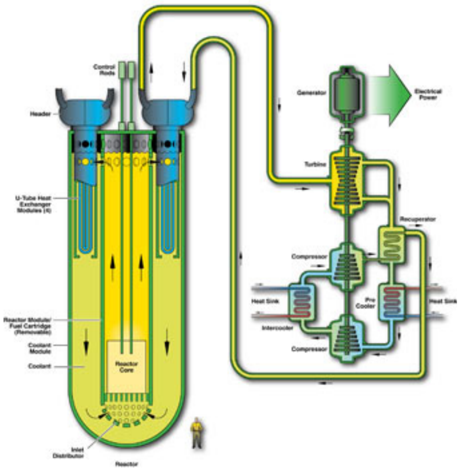


FIG. 2 Schematic representation of HLMC reactor

The gas-cooled fast reactor system (fig. 3) features a fast-neutron-spectrum, helium coolant, and a closed fuel cycle. The GFR uses a direct-cycle helium turbine for electricity generation, or can optionally use its process heat for production of hydrogen. Through the combination of a fast spectrum and full recycle of actinides, the GFR minimizes the production of long-lived radioactive waste. The GFR's fast spectrum, together with fuel reprocessing and re-fabrication technology, also makes it possible to use available fissile and fertile materials (including depleted uranium) much more efficiently than thermal spectrum gas reactors with once-through fuel cycles. Several fuel forms are candidates that hold the potential for operating at very high temperatures and ensure an excellent retention of fission products: composite ceramic fuel, advanced fuel particles, or ceramic-clad elements of actinide compounds. Core configurations may be based on pin- or plate-based assemblies or on prismatic blocks. The GFR reference has an integrated, on-site spent fuel treatment and refabrication plant [5].

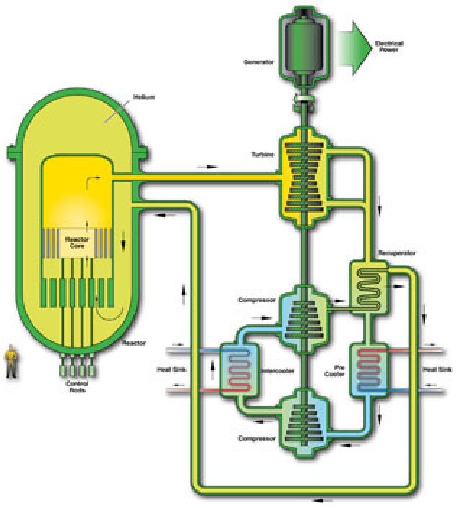


FIG. 3. Schematic representation of GFR

The Molten Salt Fast Reactor (fig. 4) combines the generic assets of fast neutron reactors (reduced neutron absorptions in the fission products, waste minimization) with those related to molten salt fluorides as fluid fuel and coolant (low pressure, high boiling temperature and optical transparency). As opposed to thermal molten salt reactors [6], the specificity of the MSFR is the absence of any solid moderator (usually graphite for thermal MSR systems), in the core. This concept has been recognized as a long term alternative to solid fuelled fast neutron systems with a unique potential, such as large negative temperature and void coefficients, lower fissile inventory, no initial criticality reserve, simplified fuel cycle, and wastes reduction. In the MSFR, the liquid fuel processing is performed during the reactor operation. The on-site salt management of the MSFR combines a salt control unit, an online gaseous extraction system in the core and an offline lanthanide extraction component by pyrochemistry where a small stream of the molten salt is set aside to be processed for fission product removal and then returned to the reactor. This is fundamentally different from a solid fuel reactor where separate facilities produce the solid fuel and process the spent nuclear fuel. Because of this, the MSFR can operate with widely varying fuel compositions. Thanks to this fuel composition flexibility, the MSFR concept may use as its initial fissile load, ^{233}U or enriched natural uranium and/or also the transuranic (TRU) elements currently produced by PWRs in the world [7].

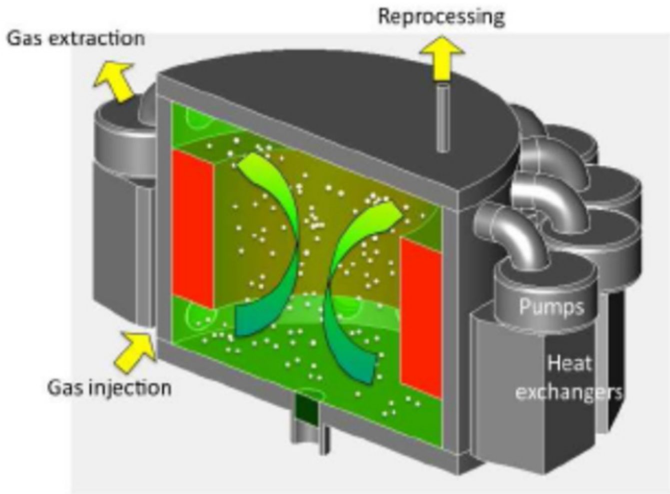


FIG. 4. Schematic representation of MSFR

Experience acquired with demonstration and prototype reactors operation

Fast reactors, mainly SFR technology, have already a long history with about 350 reactor-years of operation acquired [2]. Their overall performance has been notable, with important achievements such as the demonstration of the breeding and of the feasibility of the fast reactor fuel cycle, thermal efficiencies reaching values of 43-45%, which is the highest in the nuclear practice, and accumulation of indispensable experience in the decommissioning of several of these reactors.

Fast breeder technology has been developed since the 1960s with experimental, prototype, demonstration and commercial size reactors operating in a number of countries, including China, France, Germany, India, Japan, the Russian Federation, the United Kingdom and the United States. Past SFR realizations are summarized in Table 1, Table 2 and Table 3. A significant number of experimental fast reactors have been constructed and operated in the past decades with different purposes, ranging from testing fast reactor concepts as a whole, to qualifying components, systems and technologies, irradiating fuels and materials, etc.. Five prototype or semi industrial fast reactors were constructed and operated, and a prototype reactor, the Indian PFBR (1250 MW(th)), is currently in advanced phase of construction. Commercial size reactors were operated in France with the Superphenix experience and in the Russian Federation with BN-600, which has already accumulated more than 30 years of operation. In this country the industrial size BN-800 is currently under construction.

There is also a limited experience with heavy liquid metals such as lead (Pb) or lead-bismuth eutectic (LBE), which were proposed and investigated as coolants for fast reactors as early as the 1950s [8]. The first of these systems, a 70 MW(th) land prototype reactor, achieved criticality and started full power operation at IPPE, Russian Federation, in 1959. In total, seven nuclear submarines of the Project 705/705K, equipped with a Pb-Bi cooled 155 MW(th) reactor, were constructed and operated, following an original single submarine with a 73 MW(th) reactor and a different steam supply system. As of today, a significant number of countries are devoting large R&D programmes to develop HLMC technology, which is considered as a valid and promising alternative to sodium as fast reactor coolant.

With regards to early experimental fast reactors, it is worth notice that, although their designs and parameters showed a wide variability, those of the commercial-sized plants are quite similar. Even with the initiation of a new line of development, such as lead and lead-bismuth cooled reactor designs, it is interesting to observe that their parameters are close to those of traditional reactors. It is a further proof that the laws of physics and the principles of good engineering lead to similar optimal solution.

Table 1. Experimental Fast Reactors

Plant	Country	Technology Developer	Nominal Power (MW(th))	Coolant	Status
CEFR	China	CIAE, Beijing Institute of Nuclear Energy	65	Sodium	Under operation
RAPSODIE	France	GAAA	40	Sodium	Shut down
KNK-II	Germany	Interatom	58	Sodium	Shut down
FBTR	India	Indira Gandhi Centre for Atomic Research	40	Sodium	Under Operation
PEC	Italy	ANSALDO/NIRA/ENEA	120	Sodium	Never operated
JOYO	Japan	JNC/Toshiba/Hitachi/ Mitsubishi/Fuji	140	Sodium	Temporarily shut down
BR-5/BR-10	Russian Federation	Ministry for Atomic Energy	8	Sodium	Shut down
BOR-60	Russian Federation	Hydrogen Design Bureau, Podolsk	55	Sodium	Under operation
DFR	United Kingdom	UK Atomic Energy Authority	60	Sodium	Shut down
CLEMENTINE	USA	Los Alamos National Laboratory	0.02	Mercury	Shut down
LAMPRE	USA	Los Alamos National Laboratory	1	Sodium	Shut down
SEFOR	USA	General Electric	20	Sodium	Shut down
EBR-I	USA	Argonne National Laboratory	1.4	Sodium Potassium	Shut down
EBR-II	USA	Argonne National Laboratory	62.5	Sodium	Shut down
FFTF	USA	Westinghouse Advanced Reactors Division	400	Sodium	Shut down

Table 2. Demonstration or Prototype Fast Reactors

Plant	Country	Technology Developer	Nominal Power (MW(th))	Coolant	Status
PHÉNIX	France	CEA and EdF and GAAA	563	Sodium	Shut down
SNR-300	Germany	International Natrium-Brutreaktor-Bau GmbH (INB)	762	Sodium	Construction stopped
PFBR	India	Indira Gandhi Centre for Atomic Research	1250	Sodium	Under construction
MONJU	Japan	PNC/Mitsubishi/Hitachi/ Toshiba/Fuji	714	Sodium	Temporarily shut down
BN-350	Kazakhstan	Machine Building Design Bureau, Nizhny Novgorod	750	Sodium	Under decommissi oning
BN 600	Russian Federation	Machine Building Design Bureau, Nizhny Novgorod	1470	Sodium	Under operation
PFR	United Kingdom	NNC (National Nuclear Corporation)	650	Sodium	Shut down
CRBRP	USA	Westinghouse	975	Sodium	Construction stopped
EFFBR	USA	Atomic Power Development Associates	200	Sodium	Shut down

Table 3. Power Fast Reactors

Plant	Country	Technology Developer	Electrical Power (MW(th))	Coolant	Status
EFR	European Commission	European Consortium	3600	Sodium	Project abandoned
SUPERPHÉNIX	France	NERSA Consortium	3000	Sodium	Shut down
SNR-2	Germany	EDF, ENEL and RWE	3420	Sodium	Project abandoned
DFBR	Japan	JAPC, PNC, JAERI and CRIEPI	1600	Sodium	Project abandoned
BN 800	Russian Federation	Afrikantov OKB Mechanical Engineering	2100	Sodium	Under construction

In France, the construction of the first experimental sodium-cooled reactor, Rapsodie, started in 1962 and it went critical on 28 January 1967 with a nominal capacity of 20 MW(th). The reactor operated until 1983, when it was shut down permanently. The Phénix reactor, after more than 35 years of service, was shut down in 2009. In the course of operation, Phénix served as a technology demonstrator and an R&D support tool for sodium-cooled fast reactors. During the years 1999–2002, Phénix underwent plant renovations and safety upgrades to support the French programme on long-lived radioactive waste management. Overall, the last years of operation (2003–2009) were satisfactory. In 2007, Phénix broke several of its own records (grid connection time, operating cycle length and electrical production run without shutdown). As a final contribution, a series of important end-of-life tests were conducted on Phénix. The results deduced from these tests will serve as an experimental basis for core physics, thermal hydraulics, fuel issues, and for the validation of a wide range of sodium-cooled fast reactor computer codes. Based on the Phenix design, the commercial size sodium fast reactor Superphenix, with a nominal power of 1200 MW(e), went critical in 1985 and was connected to the grid on 14 January 1986. After an operation characterized by several technical problems and administrative hurdles (in its operative life, the reactor generated in total only 8.2 TW·h), the French government decided to abandon the project; the reactor, shut down in 1996 for a routine maintenance, was never restarted.

In India, the Fast Breeder Test Reactor (FBTR) has been in operation since October 1985, producing over 330 GW·h of thermal and 10 GW·h of electrical energy. The core thermal power has been progressively increased from 10.6 MW to 18.6 MW through the introduction of a carbide fuel consisting of more uranium and less plutonium, in an effort to attain the 40 MW(th) design target (for the original MOX fuelled core). The FBTR has been used as an irradiation tool to test fuel concepts for the Indian Pressurized Heavy Water Reactors (PHWRs) and the Prototype Fast Breeder Reactor (PFBR), as well as to conduct physics and engineering experiments to validate codes and systems. Several system upgrades and modifications have taken place and it is estimated that the residual life of this reactor is an additional 12 years at full power.

In Japan, the experimental fast reactor Joyo has been devoted to the development of the sodium-cooled fast reactor. Various irradiation tests were successfully conducted together with the results from post irradiation examination. During a periodical inspection of Joyo in 2007, it was found that there was an obstacle in the in-vessel storage rack. This incident required the replacement of the upper core structure and the retrieval of the irradiation test device for Joyo re-start. Through these restoration works, Joyo has been used for the development of in-vessel inspection and repair techniques in SFRs. The Japanese prototype fast breeder reactor Monju attained the first criticality in April 1994, but was shutdown in December 1995 due to a sodium-leak incident which occurred in the secondary coolant system. After comprehensive reviews on the validity of fast breeder reactor development in Japan as well as the Monju safety, the plant modification work against sodium leakage was carried out in 2005–2007. Additional works aimed toward the Monju restart included verification of the entire plant soundness and improvement of the anti-seismic margin. Monju was restarted in 2010, over 14 years after its shutdown. Presently, the reactor is in shutdown status, and its possible restart will depend on the future Japanese strategy on nuclear power.

In the Russian Federation, the BR-10 reactor was shut down in 2002 after approximately 44 years of operation. As a result of its operation, substantial experience has been gained for fuels, materials, reactor systems and monitoring techniques, as well as a representative test facility for any technology that would later be realized in the subsequent Russian sodium-cooled fast reactors, namely BOR-60, BN-350 and BN-600. As to other applications, the BR-

10 was used as an isotope generator and to provide treatment to cancer patients. At present, the BR-10 reactor is being prepared for decommissioning and will serve as a basis for developing and testing sodium-cooled fast reactor decommissioning technologies. The Russian BOR-60 reactor has been in operation since 1968, serving as a test facility and for supplying power to the grid. The number of unscheduled shutdowns during this period has been minimal, totaling only 1–2 events per year. Experimental tests performed at the BOR-60 facility during this period have focused on irradiation of fuels and materials, as well as testing of sodium, control and safety equipment. Subsequent to a thorough analysis, in December 2009, a second lifetime extension was approved for BOR-60, authorizing the reactor to operate through the end of 2014. The Russian BN-600 reactor has been operating as the third power unit of the Beloyarsk nuclear power plant since 1980, supplying over 112.5 billion kW·h of electricity to the grid. It now holds the distinction of being the largest operating fast reactor in the world and it ranks among the most reliable and stable nuclear power reactors in the Russian Federation. It has operated without steam generator leaks and for sixteen years without any sodium leaks in any of the sodium circuits. Work on lifetime extension began in 2006 and was successfully completed, permitting the BN-600 reactor to operate through the end of March 2020 [9].

The activities of the United States of America on fast reactors started in the 1940s when Enrico Fermi broached the idea of a fast reactor that would breed plutonium in April 1944. Such a reactor, with no moderator and with fully enriched fuel in a compact core, would have required a liquid metal coolant to adequately remove its heat. The idea was promoted by Argonne National Laboratory, and the concept became the Experimental Breeder Reactor (EBR), which, December 20, 1951 became the world's first electricity-generating nuclear power plant. In 1954 it was decided to build an advanced reactor plant at the Argonne National Laboratory. Encouraged by the successful experience with EBR-I, the new breeder reactor EBR-II was a pool type concept. The EBR-II was designed as a power plant for electricity generation and to include an integrated fuel-processing and refabricating facility. The major objective of the facility was to establish the feasibility of a fast power reactor system under operating conditions and to demonstrate the on-site fuel reprocessing and fabrication, a procedure believed to result in favorable fuel-cycle costs for high-power-density FBRs. At the same time, the Detroit Edison Company proposed to build near Detroit a 100 MW(e) class liquid metal-cooled FBR. The reactor was named the Enrico Fermi Fast Breeder Reactor (EFFBR) in recognition of Fermi's many contributions to the development and control of nuclear energy and early interest in FBRs. The EFFBR was the first full-scale semi industrial fast reactor power plant in the US. Another US project was the Fast Flux Test Facility (FFTF), a 400 MW(th) sodium-cooled loop-type fast reactor designed in the 1960s and built in the 1970s. The FFTF was operated by Westinghouse as a facility to develop and test advanced fuels and materials. Early in the design phase the decision was made to not include the capability for producing electricity because the intent was to use it solely for research.

In the United Kingdom, construction of the Dounreay experimental Fast Reactor (DFR) started in 1955 and the criticality was reached on 14 November 1959. The reactor, fuelled with U-Mo and cooled by sodium-potassium eutectic, was designed with a power output of 60 MW(th), and it was intended to develop the technology of fast reactors in the country and to prepare the realization of the Prototype Fast Reactor (PFR). The PFR was a 250 MW(e) pool-type fast breeder reactor, cooled by liquid sodium and MOX fuelled. It achieved criticality in 1974 and began supplying power to the grid in January 1975. It was permanently shut down in 1994.

At present four SFRs are in operation: CEFR in China, FBTR in India, BOR-60 and BN-600 in the Russian Federation. Two SFR, Joyo and Monju in Japan, are in temporarily shut down. Two SFRs are expected to be completed soon: the demonstration reactor PFBR (500 MW(e)) in India and the commercial reactor BN-800 (880 MW(e)) in the Russian Federation are both under advanced construction [10].

There are programmes to develop and implement innovative fast nuclear energy systems in China, France, India, Japan, the Republic of Korea, the Russian Federation, among other countries. In Europe, a strategy and technological pathway for fast reactors includes the development of a sodium cooled fast reactor as a first track, aligned with Europe's prior experience, and two alternative fast reactor technologies to be explored on a longer timescale: lead cooled fast reactors and gas cooled fast reactors. And a number of initiatives, including the Generation IV International Forum (GIF) and the IAEA, are carrying out R&D activities on fast reactor technology. Experts expect that the first Generation IV fast reactor demonstration plants and prototypes will be in operation by 2030 to 2040.

In the following sections of this booklet, brief technical descriptions of innovative fast reactor designs and concepts – generally referred as Generation IV fast neutron systems - are provided for each major fast reactor line, starting with sodium-cooled fast reactors, after that continuing with heavy liquid metal-cooled reactors, gas-cooled reactors, and, finally, molten-salt fast reactors.

SODIUM-COOLED FAST REACTOR DESIGNS

CFR-600 (China Institute of Atomic Energy, China)

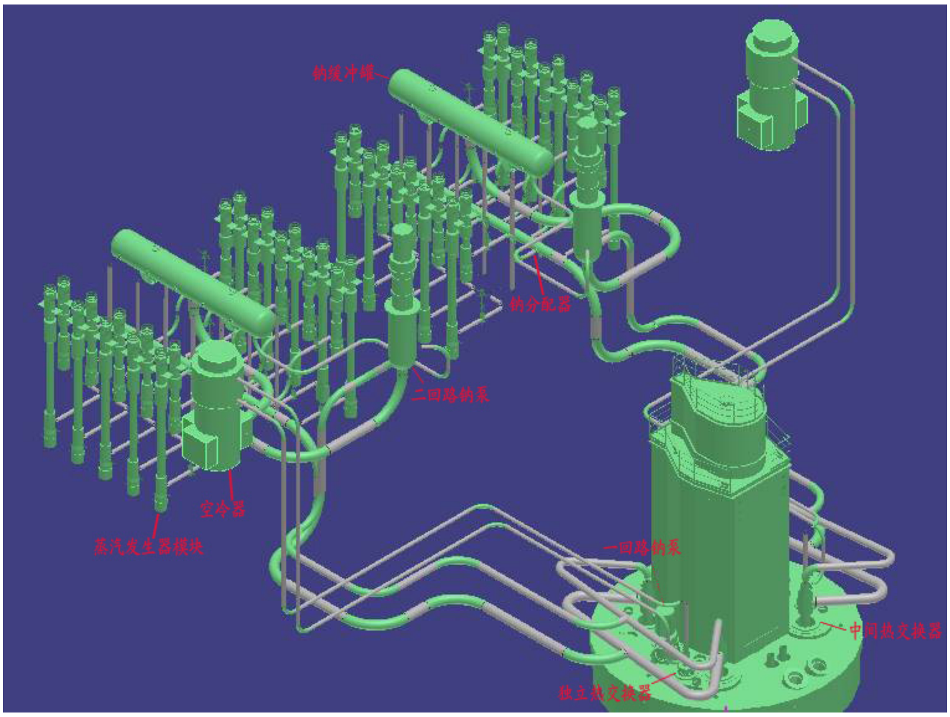


FIG. 5. Schematic representation of CRF-600.

Full name:	*China Fast Reactor 600*
Designer:	*China Institute of Atomic Energy*
Reactor type:	*Sodium -cooled Pool type Reactor*
Electrical capacity:	*600 MW(e)*
Thermal capacity:	*1500 MW(th)*
Coolant	*Sodium*
Primary Circulation	*Forced*
System Pressure:	*0.054MPa*
System Temperature:	*380(inlet)/550(outlet) °C*
Fuel Material:	*UO2(initial stage)/MOX(later stage)*
Emergency safety systems:	*Hybrid active and passive*
Residual heat removal systems:	*Hybrid active and passive*
Design Life:	*60Years*
Design status:	*Concept*
Planned deployment/1st date of completion:	*2023*
New/Distinguishing Features:	*CFR 600 will be designed to demonstrate the breeding ability of fast reactor*

Introduction

CEFR (China Experimental Fast Reactor), which is the first fast reactor of China, has achieved its first criticality at 21st July 2010 and was connected to the grid with 40% rated power on 21st July 2011. Based on the experience of CEFR, CFR-600 (China Fast Reactor-600), the second step of fast reactor development in China, is a sodium cooled pool type fast reactor, with 1500MW thermal power and 600MW electric power. CFR-600 is a prototype reactor and a three circuit design. The primary circuit and the second circuit both apply on the two-loop design. The inlet core temperature is about 380℃, and the outlet core temperature about 550℃. The design objective for thermal efficiency is 40%.

Description of the Nuclear System

CFR-600 is flexible for two fuel types, UO_2 and MOX. UO_2 will be loaded first and then replaced by MOX fuel.

Breeding ability is a significant feature of CFR-600 with 1.2 as the breeding ratio objective.

At the pre-conceptual design stage, research on key parameters related to breeding ratio, including fuel pin diameter, axial blanket design, core layout etc., is carried out in detail. Another important issue is the study of sodium void reactivity effect.

CFR-600 will be divided into three fuel regions. The maximum burn up is about 100MWd/kgHM. Control rods are utilized for reactivity control and reactor shut down. CFR-600 will also be equipped with passive emergency shutdown system.

The instrumentation and control system will be digital system and will be optimized based on CEFR experience.

For the reactor vessel, the main vessel and protection vessel will be used as in the CEFR. Furthermore, containment will be set up for CFR-600.

Description of the Safety Concept

As mentioned above, a passive shutdown system will be used in CFR-600. The hydraulic suspend rod is now under consideration.

To provide defence in depth against core melting caused by beyond design basis accidents, CFR-600 will be equipped with a core catcher which, in case of severe accident, will keep the melting material subcritical and provide long-term cooling.

For the residual heat removal after the accidents, a passive residual heat removal system will be designed and equipped. The specific scheme is being investigated at this stage.

Deployment Status and Planned Schedule

The pre-conceptual design of CFR-600 started in 2012. The latest design status is technological design, which will be finished in 2014. In view of the construction, FCD will be carried out in 2017. The first fuel loading is foreseen in 2023.

ASTRID (CEA with its industrial partners EDF, AREVA NP, ALSTOM, BOUYGUES, COMEX NUCLEAIRE, TOSHIBA, JACOBS, ROLLS-ROYCE and ASTRIUM Europe, France)

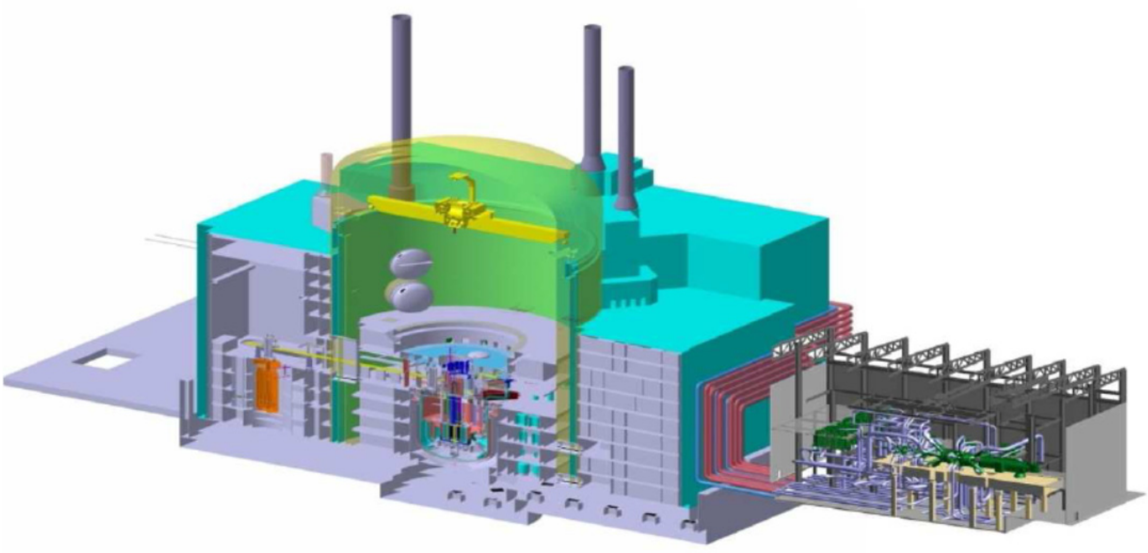


FIG. 6. Schematic view of ASTRID

Full name:	*Advanced Sodium Technological Reactor for Industrial Demonstration*
Designer:	*CEA with its industrial partners (EDF, AREVA NP, ALSTOM,BOUYGUES, COMEX NUCLEAIRE, TOSHIBA, JACOBS, ROLLS-ROYCE and ASTRIUM)*
Reactor type:	*Pool type*
Electrical capacity:	*600 MW(e)*
Thermal capacity:	*1500 MW(th)*
Coolant	*Sodium*
Primary Circulation	*Forced*
System Pressure:	*0.3 MPa*
System Temperature:	*475 °C*
Fuel Material:	*MOX*
Fuel Cycle:	*4 x 360 EFPD*
No. of safety trains:	*3 trains*
Emergency safety systems:	*Hybrid active and passive*
Residual heat removal systems:	*Hybrid active and passive*
Design Life:	*60 Years*
Design status:	*Pre-conceptual design*
Planned deployment/1st date of completion:	*The basic design phase is planned from 2016 to 2019.*
New/Distinguishing Features:	*ASTRID will be designed to pursue the R&D on sodium fast reactors and demonstrate the feasibility of transmutation of minor actinides*

Introduction

The Generation IV International Forum (GIF) was launched to create an international R&D framework capable of boosting research on the most efficient technologies. According to GIF, four main objectives defined to characterise the future reactor systems are sustainability, cost-effectiveness, a high level of safety and reliability, and proliferation resistance and protection against any external hazards. At the same time, the French Act on the sustainable management programme for radioactive materials and waste stipulates the commissioning of a Generation IV reactor by 2020.

The CEA launched the conceptual design of a sodium-cooled fast reactor in 2010, with industrial partners. This project has been called ASTRID which stands for 'Advanced Sodium Technological Reactor for Industrial Demonstration'.

Description of the Nuclear System

During the pre-conceptual phase of the design, several options have been studied.

Regarding the core, the chosen low void effect core retains a number of advantages in terms of longer cycles and fuel residence times. It complies with all the control rod withdrawal criteria, while increasing safety margins for all unprotected-loss-of-flow (ULOF) transients and improving the general design. This core concept involves heterogeneous axial $UPuO_2$ fuel with a thick fertile plate in the inner core and is characterized by an asymmetrical, crucible-shaped core with a sodium plenum above the fissile area. The reactivity effect associated with sodium expansion achieved by design (sodium plenum and heterogeneous fertile plate) is negative in the event of a total loss of primary coolant, and can result in an overall negative void effect if a boiling phase is reached.

Some choices have already been made during the pre-conceptual design phase for the primary and secondary circuits: pool-type reactor with conical '*redan*' (inner vessel) made to allow for extended ISIR access.

In terms of the reactor block, it has been decided to use three primary pumps together with four intermediate heat exchangers. Each intermediate heat exchanger is associated with a secondary sodium loop which includes modular stream generators or sodium-gas system and the chemical volume control system.

Description of the Safety Concept

To provide defence in depth against scenarios such as the melting of the core, the ASTRID reactor will be equipped with a core catcher. It will be designed to recover the entire core keeping the corium in a sub-critical state while ensuring its long-term cooling. As other safety related components, it must be inspectable. At this stage, several options are being investigated in terms of the possible core-catcher technologies, locations (in-vessel or outside the vessel) and attainable performance levels.

The containment will be designed to resist the release of the mechanical energy caused by a hypothetical core accident or large sodium fires, to make sure that no countermeasures are necessary outside the site in the event of an accident.

Deployment Status and Planned Schedule

The pre-conceptual design phase was launched in October 2010 and involves 3 phases. The first was a preparatory phase that ended with an official review which launched the following phase in March 2011. The pre-conceptual design phase aims at analyzing the open options in order to choose the reference design. Finally, the conceptual design started in 2013 and aims at consolidating the project data to obtain a final and consistent conceptual design by late 2015. The basic design phase is planned from 2016 to 2019.

FBR-1 & 2 (IGCAR, India)

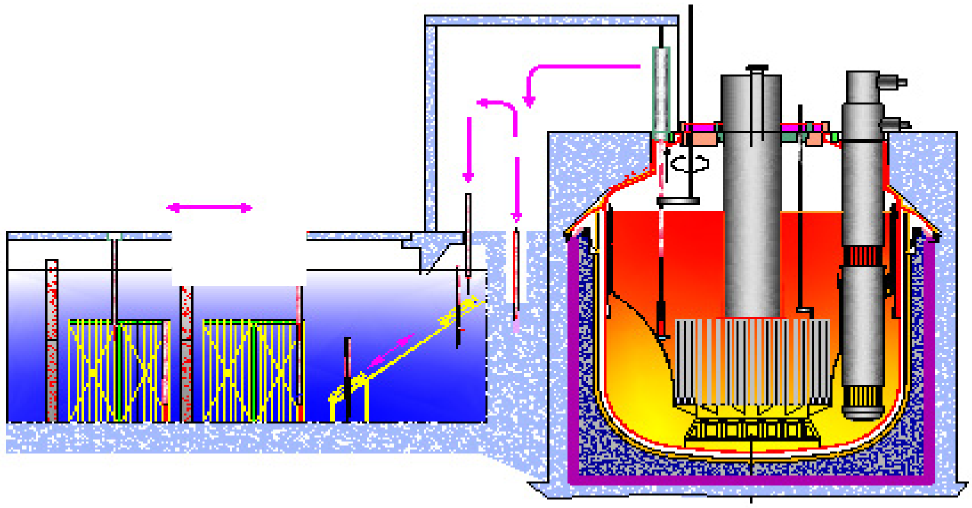


FIG. 7. Schematic representation of FBR-1 & 2

Full name:	*Fast Breeder Reactors 1 & 2*
Designer:	*India Gandhi Centre for Atomic Research*
Reactor type:	*Pool Type*
Electrical capacity:	*500 MW(e)*
Thermal capacity:	*1250 MW(th)*
Coolant	*Sodium*
Primary Circulation	*Forced*
System Pressure:	*0.11 MPa*
System Temperature:	*397 °C / 547°C (Reactor Inlet / Outlet)*
Fuel Material:	*MOX*
Fuel Cycle:	*8 Months*
No. of safety trains:	*6 independent trains (for Decay Heat Removal)*
Emergency safety systems:	*Hybrid Active and Passive*
Residual heat removal systems:	*Hybrid Active and Passive*
Design Life:	*60 Years*
Design status:	*Detailed Design*
Planned deployment/1st date of completion:	*2023 & 2024*
New/Distinguishing Features:	*Incorporates design features towards improved economy and enhanced safety. Innovative reactor assembly design with four primary pipes per sodium pump, shut down systems with advanced safety enhancing features, liquid poison based ultimate shut down system, twin unit design sharing non-safety systems such as the external fuel handling & storage system, core catcher capable of handling whole core melt etc.*

Introduction

The design of 500 MW(e) twin units Fast Breeder Reactors-1&2 (FBR-1&2) is a standardised one for serial construction. It incorporates the feedback experience gained from design, construction and safety review of 500 MW(e) Prototype Fast Breeder Reactor which is under construction presently, towards improved economy and enhanced safety. The reactor assembly design is engineered by incorporating many advanced design features resulting in 25% material savings and favouring manufacture & erection of integrated reactor assembly, as a single unit, thus leading to reduced construction time. Improvements are expected in the shut-down systems, decay heat removal systems and the choice of materials for the permanent reactor assembly components and heat transport system. The layout incorporates a twin unit concept in which non-safety system especially the ex-vessel fuel handling system and fuel storage building are shared. Advanced core structural materials are employed aiming for higher burnup. FBR-1&2 design in general, and the design concepts in particular, would be the standardised options to be adopted for the MOX as well as future metal fuel reactors.

Description of the Nuclear System

The reactor is of pool type with 2 secondary sodium loops. The core consists of 2 fuel enrichment zones enveloped by blankets. Advanced shielding material of Ferro Boron is planned to be used for the bulk core radial shielding leading to reduction in the shielding thickness. Various shells in the primary vessel with optimised annular gaps along with reduced bulk radial shielding leads to a compact reactor block. Grid Plate with an innovative design permits forced flow only through the replaceable core subassemblies and the surrounding outer shielding assemblies are supported on raised spigots at the periphery of the grid plate. Two independent, redundant and diverse shut down systems having enriched B_4C absorber rods are employed, for the purpose of power regulation and safety. In addition, neutron poison systems based on either Li-6 or B_4C granules are deployed as a part of Ultimate Safety System. Primary sodium purification is carried out by in-vessel arrangement. For the roof slab, a new concept of dome shaped structure, which is supported on the reactor vault leading to the top structure being in compression, is expected to offer enhanced seismic safety margin.

Description of the Safety Concept

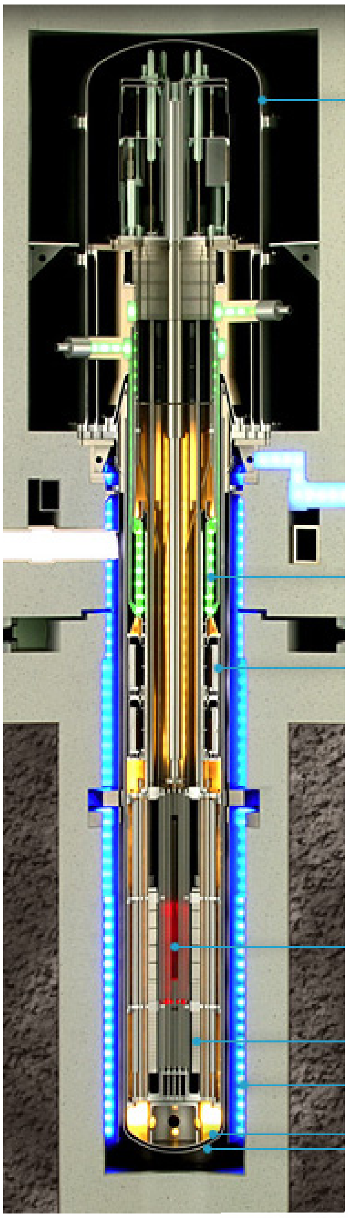
The safety features incorporated in FBR-1&2 are:

- Provision of four primary pipes per sodium pump offer higher safety margin for the Category 4 primary pipe rupture design basis event.
- In-vessel primary sodium purification is adopted to avoid the radioactive sodium being taken outside primary vessel.
- Stroke limiting device is added to the Control & Safety Rod, the primary shutdown system, to guard against inadvertent control rod withdrawal event.
- Temperature sensitive electromagnet is added to the Diverse Safety Rod, the secondary shutdown system, leading to enhanced safety.
- As an ultimate shutdown system, either liquid Li or B_4C granule based system is added to practically eliminate core disruptive accident.
- Three trains each of two diverse concepts (passive, active-passive) for decay heat removal are incorporated. Active-Passive systems are designed for 2/3rd of their capacity by natural circulation
- Innovative multi-layer core catcher to handle whole core melt debris

Deployment Status and Planned Schedule

Detailed design of FBR-1&2 is in progress. Technology development of major reactor assembly components has been completed. Preliminary Safety Analysis Reports are under preparation, Design Safety Criteria are under review by regulatory agency and infrastructure development is in progress at site by the utility. FBR-1&2 are planned to be launched in 2023 and 2024 respectively.

4S (Toshiba, Japan)



Reactor type:	*Liquid metal cooled fast reactor*
Electrical capacity:	*10 MW(e)*
Thermal capacity:	*30 MW(th)*
Coolant:	*Sodium*
Primary circulation:	*Forced circulation*
System pressure:	*Non-pressurized*
Core outlet temperature:	*510°C*
Thermodynamic cycle:	*Indirect Rankine*
Fuel material:	*U–Zr alloy*
Fuel enrichment:	*<20%*
Fuel cycle:	*30 years*
Reactivity control:	*Reflector, Fixed absorber*
Emergency safety systems:	*Hybrid Active and passive*
Residual heat removal systems:	*Hybrid Active and passive*
Design life:	*30 years*
Design status:	*Detailed design*
Seismic design:	*Seismic isolator*
Predicted CDF:	*Max. less than 0.1 in ATWS event*
Planned deployment:	*License activity*
Distinguishing features:	*Passive safety, Negative sodium void reactivity*

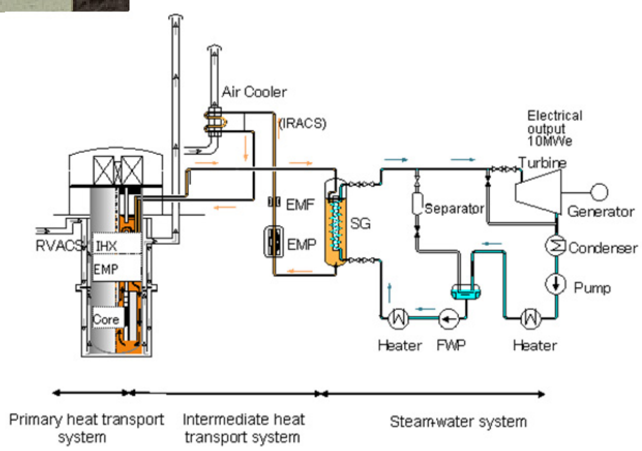


FIG. 8. Simplified plant schematic.

Introduction

The 4S is a sodium cooled reactor without on-site refuelling. Being developed as a distributed energy source for multipurpose applications, the 4S offers two outputs: 30 and 135 MW(th). The 4S is not a breeder reactor since the blanket fuel, usually consisting of depleted uranium located around the core to absorb leakage neutrons from the core to achieve breeding of fissile materials, is not present in its basic design.

The 4S-30MW(th) is a reactor the core of which has a lifetime of approximately thirty years. The movable reflector surrounding the core gradually moves, compensating for the burnup reactivity loss over the thirty year lifetime. The plant electric power can be controlled by the water–steam system, which makes the reactor applicable for a load follow operation mode.

Description of the nuclear systems

The reactor is an integral pool type, as all primary components are installed inside the reactor vessel. Major primary components are intermediate heat exchangers, primary electromagnetic pumps (EMP), moveable reflectors which form a primary reactivity control system, the ultimate shutdown rod, radial shielding assemblies, the core support plate, coolant inlet modules and fuel subassemblies.

Description of the safety concept

To reduce the probability of component failure, the design eliminates active systems and feedback control systems from the reactor side as well as components with rotating parts. There is also limitation of the radioactivity confinement area, since there is no refuelling during the life of the reactor. Other objectives include: the prevention of core damage in accidents, the confinement of radioactive materials, and the prevention of sodium leakage and the mitigation of associated impacts if leakage should occur.

Technical features of the 4S contributing to a high level of proliferation resistance include the use of uranium based fresh fuel with ^{235}U enrichment <20% by weight and a low plutonium content in the spent fuel (<5% by weight). The reprocessing technology available for metal (alloy) fuel, such as U–Zr or U–Pu–Zr, ensures that plutonium is always recovered together with the accompanying minor actinides, which include highly radioactive and radiotoxic nuclides.

Description of the Safety Concept

4S has developed safety feature related to Fukushima Daiichi Accident. The core damage is avoidable without any power supplies because 4S has two passive residual heat removal systems (RHRS) which are reactor vessel auxiliary heat removal system and intermediate auxiliary heat removal system, and other passive safety characteristics, for example negative sodium void reactivity and negative reactivity insertion by thermal expansion of metallic fuel. Reactor building is supported by seismic isolator for earthquake. Also, the building is reinforced and can be protected from massive water invasion by keeping its water-tightness.

Deployment status

Licensing activities for the 4S design initiated with the U.S.NRC in 2007. In pre-application review, four meetings have been held in the past and fourteen technical reports were submitted to NRC. Development for 4S related technology had been performed in the past; full scale test of EMP and electromagnetic flow meter in Toshiba sodium loop facility, development of manufacturing technology of double wall steam generator, and critical experiment in FCA were done, sponsored by the Japanese government. Toshiba is conducting the detailed design and safety analysis for design approval. In parallel, Toshiba continues looking for customers.

JSFR (JAEA, Japan)

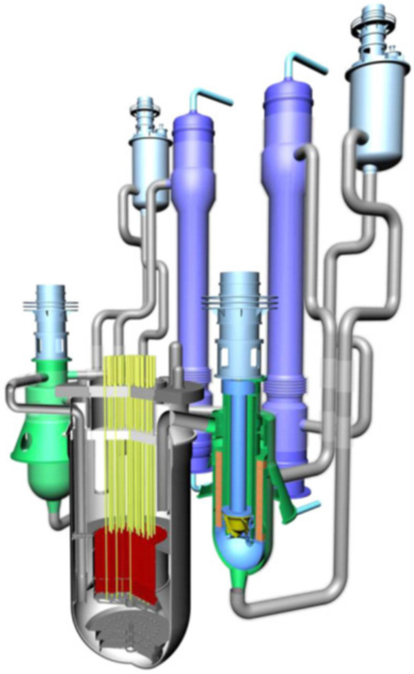


FIG. 9. Schematic Representation of JSFR.

Full name:	*Japan Sodium-cooled Fast Reactor*
Designer:	*Japan Atomic Energy Agency*
Reactor type:	*Sodium-cooled Fast Reactor*
Electrical capacity:	*750 MW(e) (medium scale) 1500 MW(e) (large scale)*
Thermal capacity:	*3530 MW(th)*
Coolant	*Sodium*
Primary Circulation	*Forced*
System Pressure:	*0.15 MPa (gage)*
System Temperature:	*550 °C*
Fuel Material:	*MOX*
Fuel Cycle:	*18-26 Months*
No. of safety trains*:	*3 trains*
Emergency safety systems:	*Primary and backup shutdown systems (2 active)* *Self-actuated shutdown system (SASS) (1 passive)*
Residual heat removal systems:	*DRACS+2PRACS (3 passive)* *AM decay heat removal system (1 or 2 active)*
Design Life:	*60 Years*
Design status:	*Conceptual Design*
Planned deployment/1st date of completion:	-
New/Distinguishing Features:	*Fuel Assembly with Inner Duct Structure (FAIDUS), Self Actuated Shutdown System (SASS), hot vessel, two loop, integrated IHX-pump*

*: Number of safety grade buses

Introduction

Japan Sodium-cooled Fast Reactor (JSFR) is a concept which has potential to achieve sustainable energy production, radioactive waste reduction, safety equal to the future light water reactor (LWR) and economic competitiveness against other future energy sources. In 1999, JAEA and utilities launched the "Feasibility Study on Commercialized Fast Reactor Cycle Systems (FS)" with domestic partners of vendors and universities. The FS targets aim at improved safety and economic competitiveness looking at other energy resources in the future including the future LWR. Those targets are consistent with the goals in Generation IV International forum (GIF).

Description of the Nuclear Systems

JSFR is expected to achieve the development targets of the FaCT (Fast Reactor Cycle Technology Development) project and the Generation IV reactor goals adopting following advanced key technologies:

- High burn-up core with oxide dispersion strengthened steel cladding material
- Safety enhancement with self-actuated shutdown system (SASS) and re-criticality free core
- Compact reactor system adopting a hot vessel and in-vessel fuel handling with a combination of an upper inner structure (slit UIS) with a slit and advanced fuel handling machine (FHM)
- Two-loop cooling system with large diameter piping made of Mod. 9Cr-1Mo steel
- Integrated intermediate heat exchanger (IHX)-pump component
- Reliable steam generator (SG) with double-walled straight tube
- Natural circulation decay heat removal system (DHRS)
- Simplified fuel handling system
- Steel plate reinforced concrete containment vessel (SCCV)
- Advanced seismic isolation system

Those technologies have been evaluated to be suitable for installation to the demonstration JSFR.

Description of the Safety Concept

For safety design, JSFR adopts the defence-in-depth (DiD) principle to the same extent as it has been in LWRs. Securing reactor shutdown, two independent reactor shutdown systems with independent/diversified signals are installed. For the fourth level of DiD, SASS is installed providing passive shutdown capability. The re-criticality free core concept has the great importance to ensure the in-vessel retention scenario against whole core disruptive accidents. The initiating phase energetics due to exceeding the prompt criticality has to be prevented by limiting the sodium void worth and the core height. The possibility of molten fuel compaction has to be eliminated by enhancing the fuel discharge from the core. The effectiveness of fuel assembly with inner duct structure (FAIDUS) has been confirmed by both in-pile and out-of-pile experiments. The JSFR decay heat removal system consists of a combination of one loop of direct reactor auxiliary cooling system (DRACS) and two loops of primary reactor auxiliary cooling system (PRACS) adopting full natural convection system.

Deployment Status and Planned Schedule

Even though the JSFR safety design already took into account measures against severe accident situations and passive safety features such as passive shutdown system and natural convection decay heat removal systems as 2010 design version, the Fukushima Dai-ichi nuclear power plant (1F) accident highlighted the importance of design measures against severe accidents and extreme external events. For further improvement on safety designs based on lessons learned from the 1F accident, Safety Design Criteria (SDC) is under discussion in the GIF framework aiming at global standards for sodium-cooled fast reactor safety designs.

PGSFR (KAERI, Rep. of Korea)

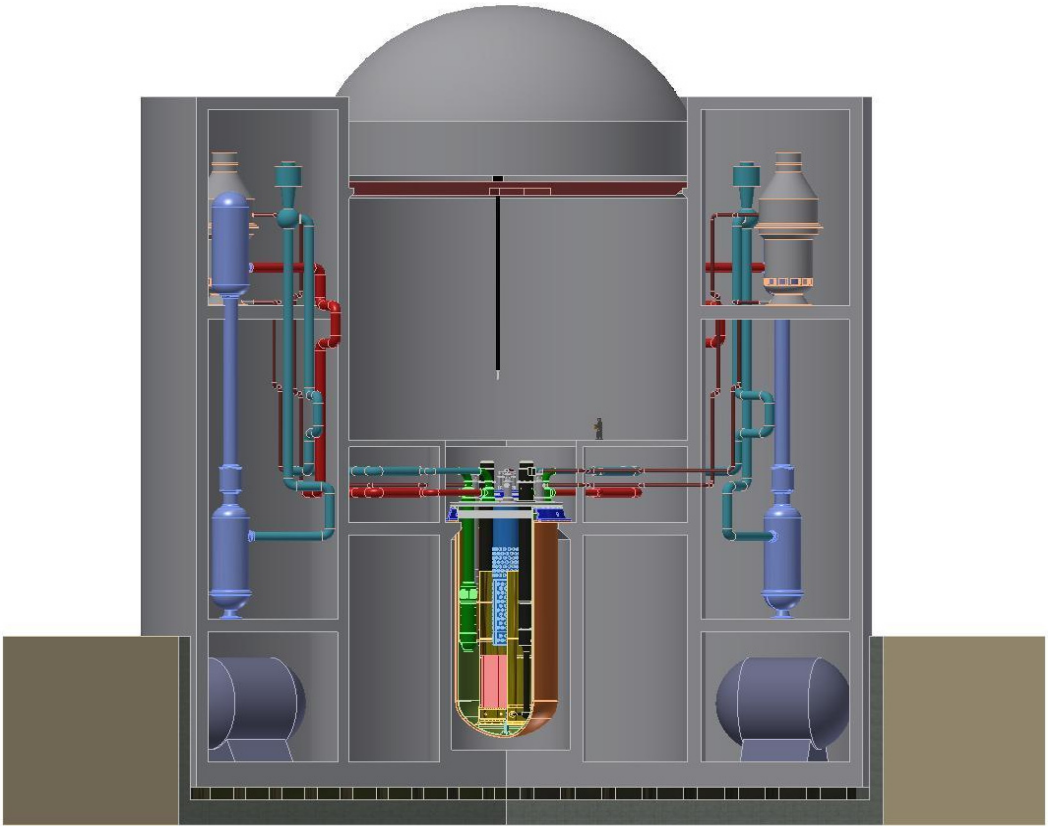


FIG. 10. Schematic view of PGSFR

Full name:	*Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR)*
Designer:	*Korea Atomic Energy Research Institute (KAERI)*
Reactor type:	*Sodium-cooled Reactor*
Electrical capacity:	*150 MW(e)*
Coolant	*Sodium*
Primary Circulation	*Pool*
System Pressure:	*~ 1 bar*
System Temperature:	*390-545 °C*
Fuel Material:	*U-Zr (initial core), U-TRU-Zr (reload core)*
Fuel Cycle:	*~10 Months*
Emergency safety systems:	*Hybrid (Passive and Active)*
Residual heat removal systems:	*Hybrid (Passive and Active)*
Design Life:	*60 Years*
Design status:	*Preliminary Design*
Planned deployment/1st date of completion:	*2028*
New/Distinguishing Features:	*Metal fuel, pool type reactor and RHRS features which accommodate SBO*

Introduction

The objectives of PGSFR are to test and demonstrate the performance of transuranics (TRU)-containing metal fuel required for a commercial SFR, and to demonstrate the TRU transmutation capability of a burner reactor as a part of an advanced fuel cycle system.

Description of the Nuclear System

PGSFR has plant capacity of 150 MW(e) and features a proliferation-resistant core without blankets, metallic fuelled core, pool type PHTS, and two IHTS loops. The core adopts a homogenous configuration in the radial direction that corporate annular rings of inner and outer driver fuel assemblies. All blankets are completely removed in the core so as to exclude production of high quality plutonium. The active core has the height of ~90 cm and a radial equivalent diameter of ~1.6 m. The metallic fuel U-Zr (or U-TRU-Zr) is used as the driver fuel. Each fuel assembly includes 217 fuel pins. All the fuels have a single enrichment of U (or TRU) nuclide. The reactivity control and shutdown system consists of nine control rods assemblies that are used for power control, burnup compensation and reactor shutdown in response to demands from the plant protection control and systems. The heat transport system of PGSFR consists of PHTS and IHTS, steam generation system, and decay hear removal system (DHRS). PHTS mainly delivers the core heat to IHTS and IHTS works as the intermediate system between PHTS, where nuclear heat is generated, and the SGS where the heat is converted to steam. PHTS is a pool type in which all the primary components and primary sodium are located within a reactor vessel. Two mechanical PHTS pumps and four Intermediate Heat Exchangers (IHXs) are immersed in the primary sodium pool. IHTS has two loops, and each loop has two IHXs connected to one steam generator and one IHTS pump. Each steam generator has a thermal capacity of ~200 MW(th). The IHTS sodium flows downward through the shell side while the water/steam goes up through the tube side. The IHTS adopts electro-magnetic pump to simplify installation and to reduce moving parts. The IHX design conditions were established to prevent water/steam and sodium water reaction products from being discharged into reactor vessel. The DHRS, one of the safety design features, is composed of two Passive Decay Heat Removal Systems (PDHRS) and two Active Decay Heat Removal Systems (ADHRS). DHRS is designed to remove the decay heat of the reactor core after a reactor shutdown when the normal heat transport path is unavailable.

For a simple reactor design, the reactor vessel has a uniform thickness of ~5 cm, and there are no penetrations and no attachment to the reactor vessel. The horizontal seismic isolation design is adapted for a reactor island including a reactor building, an auxiliary building, and a wastage/ maintenance building. For the design materials, 9Cr-1Mo-V steel is used for the IHX, DHX, IHTS piping, and steam generator. Other components such as the reactor vessel, reactor internal structures, and reactor head are composed of 316 stainless steel.

Description of the Safety Concept

PGSFR is designed to be safe against severe accidents incurred through earthquakes and tsunamis. DHRS, combination of passive and active decay heat removal systems, has a sufficient capacity to remove the decay heat in all design basis events without operator’s action by incorporating the principles of redundancy and independency. Double reactor vessels and double pipings in IHTS are designed for the prevention of sodium leakage. PGSFR has also a passive reactor shutdown system.

Deployment Status and Planned Schedule

The long-term Advanced SFR R&D plan was updated by KAEC in November 2011 in order to refine the plan. The revised milestones include the specific design of a prototype SFR by 2017, specific design approval by 2020, and construction of a prototype SFR by 2028. The PGSFR has now entered into the preliminary design phase.

BN-1200 (JSC “Afrikantov OKBM”, Russian Federation)

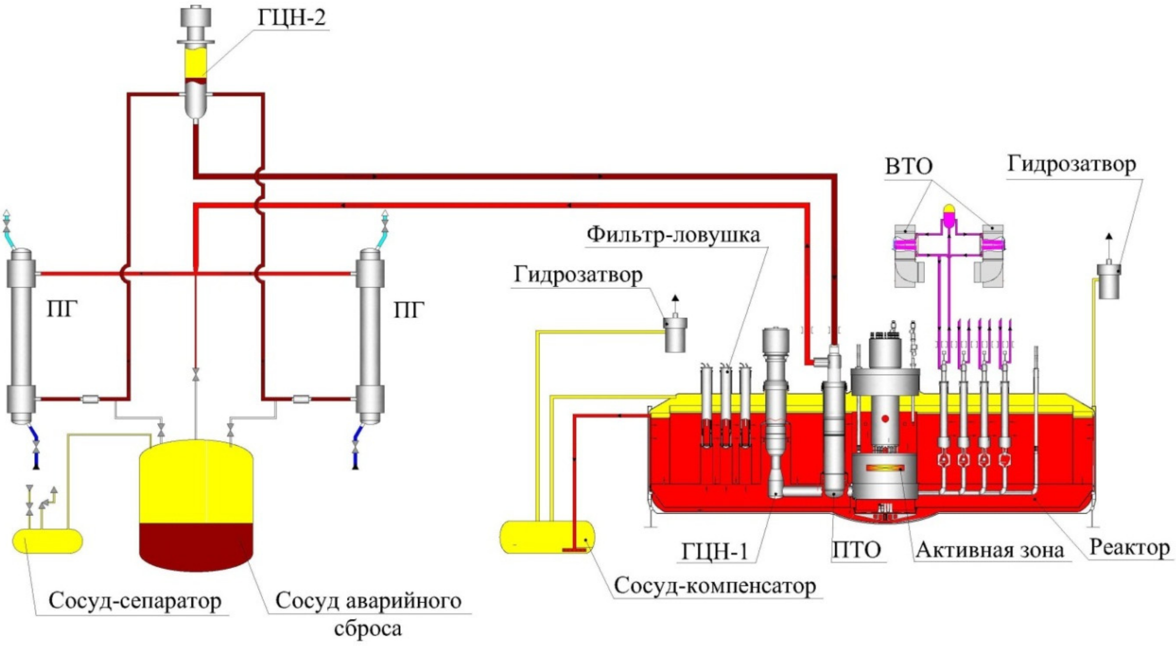


FIG. 11. Schematic view of BN-1200

Full name:	*BN-1200*
Designer:	*JSC “Afrikantov OKBM”*
Reactor type:	*Sodium-cooled Reactor*
Electrical capacity:	*1220 MW(e)*
Thermal capacity:	*2800 MW(th)*
Primary circuit configuration	*Loop*
Primary Circulation	*Forced*
System Pressure:	*0.54 MPa*
System Temperature:	*410 °C*
Fuel Material:	*Nitride or MOX*
Emergency safety systems:	*Hybrid active and passive*
Residual heat removal systems:	*Hybrid active and passive*
Design Life:	*60 Years*
Design status:	*Final design completion in 2015*
Planned deployment/1st date of completion:	*Final decision has not been taken yet*
New/Distinguishing Features:	*Integral layout of primary circuit*

Introduction

BN-1200 is a new sodium-cooled fast reactor for serial construction. The succession of fundamental engineering solutions proved to be positive for BN-600 and BN-800 and has been preserved in the process of BN-1200 reactor development. Monoblock principle has been used in BN-1200 power design, i.e. one reactor and one turbine. The reactor design provides for integral layout of primary circuit equipment (reactor core, MCP-1, IHX, autonomous heat exchangers and cold trap filters). The reactor vessel is supported at its lower part and enclosed in a guard vessel.

Description of the Nuclear System

The reactor core comprises assemblies of different types [fuel subassemblies (FSA), boron shield assemblies and absorber rods]. The central part of the core consists of FSA with fuel of similar enrichment and cells with absorber rods. The adopted tank designed for in-reactor storage reduces residual heat release to the safe level for refueling activities and FSA washing and it was selected taking into account the possible elimination of the spent FSA cask. Rows of assemblies with natural boron carbide are arranged behind in-reactor storage to form additional side shielding of in-reactor equipment. The structure of the core FSA represents a hexagonal wrapper tube with a top nozzle attached to one end and a bottom nozzle to the other one. Inside the hexagonal wrapper tube there are bundles of absorber elements and fuel pins arranged one beneath the another forming a 'sodium' cavity between each other providing Sodium Void Reactivity Effect.

MCP-1 is an impeller submersible pump including a check valve and variable-frequency electric drive with a stepless control of speed.

The reactor plant has three circuits. The primary and secondary circuits are sodium-cooled and the third circuit coolant is water/steam. Each of the three circuits is divided into four parallel flows (loops) equally participating in heat transfer from reactor to turbine plant of power unit. Each loop of the secondary circuit consists of an MCP-2 and a steam generator (SG) connected by pipelines. Temperature transfers at the secondary circuit pipelines are compensated via bellows compensators.

The MCP-2 is a single-stage vertical centrifugal pump with free level of sodium in it. The SG is a block-type once-through heat exchanger consisting of two modules with straight heat exchanging pipes. The SG is fitted with automatic protection system in case of inter-circuit leaks.

Description of the Safety Concept

BN-1200 reactor plant design utilizes a number of new engineering solutions compared to BN-600 and BN-800:

– primary circuit sodium systems and equipment are completely integrated in the reactor tank which eliminates radioactive sodium leaks;

– emergency heat removal system (EHRS) is applied providing natural circulation through all EHRS circuits, including circulation immediately through FSA thereby increasing the level of power output under allowable temperature conditions in the reactor core;

– passive shutdown system comprising hydraulically suspended absorber rods and a system of rods responding to sodium temperature variations in the core is used;

– a special containment is provided in the reactor compartment to confine accidental releases from the reactor under beyond design-basis accidents.

Due to adopted solutions it is expected to achieve better safety performances:

- probability of severe damage (10^{-6}) to the reactor core is an order of magnitude less than the one required by regulatory documentation,
- target criterion has been specified, i.e. the boundary of protective action zone has to coincide with the boundary of production site for severe beyond design-basis accidents with their probability not exceeding 10^{-7} over reactor-year.

Deployment Status and Planned Schedule

Not specified yet.

MBIR (NIKIET, Russian Federation)

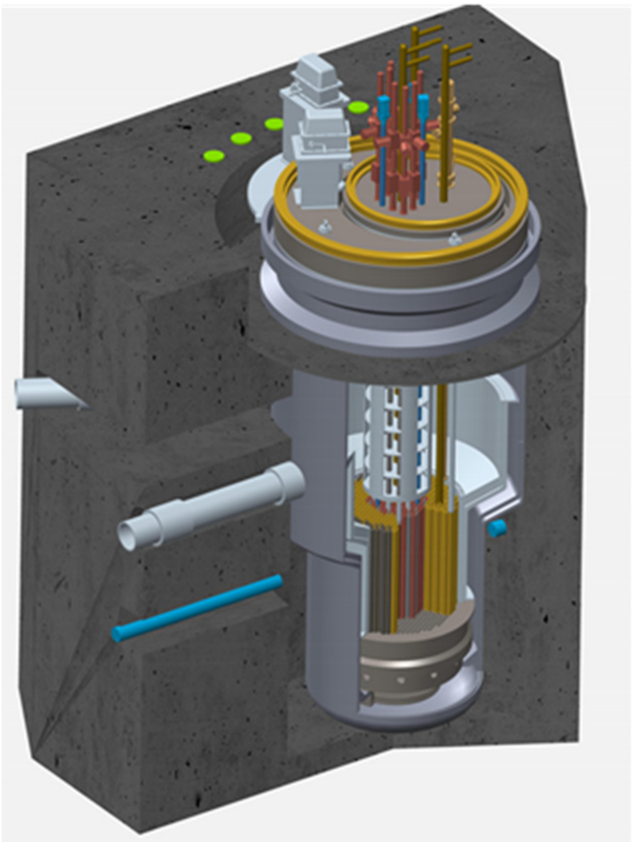


FIG. 12. Schematic view of MBIR

Name:	*Multipurpose fast-neutron research reactor (MBIR)*
General Designer:	*NIKIET*
Reactor type:	*Sodium-cooled fast-neutron reactor*
Electrical capacity:	*up to 60 MW(e)*
Thermal capacity:	*up to 150 MW(th)*
Coolant:	*Sodium*
Primary Circulation:	*Forced*
System Pressure:	*up to 0.6 MPa*
System Temperature:	*330-512 ºC*
Fuel Material:	*MOX*
Fuel Cycle:	*15 weeks*
No. of safety trains:	*2*
Emergency safety systems:	*Active/hybrid*
Design Life:	*50 years*
Planned deployment/1st date of completion:	*2020*
Distinguishing Features:	*Reactor equipped with in-vessel and ex-vessel experimental facilities and devices for a broad range of in-pile research activities and experiments, high fast neutron flux (up to* $5 \cdot 10^{15}$ $cm^{-2} \cdot s^{-1}$*)*

Introduction

The MBIR is a nuclear research facility with a multipurpose sodium-cooled fast-neutron reactor of 150 MW(th), designed for a broad range of in-pile research activities and experiments. The MBIR is developed for various research activities including advanced nuclear fuel and absorber materials, cyclic and emergency modes of operation, studies related to closed fuel cycles, radiation tests of advanced structural materials, study of new and modified liquid-metal coolants, validation of new equipment, production of radioisotopes, and use of neutron beams for medical applications.

Description of the Nuclear Systems

The MBIR includes a number of experimental devices to be accommodated within the core. The maximum in-core neutron flux is not less than $5 \cdot 10^{15}$ $cm^{-2}s^{-1}$. Regarding the facility's purposes, the core should contain three loop channels, independent instrumented experimental channels and more than ten material test and isotope production assemblies, with vertical and experimental channels to be arranged outside the reactor vessel. The core is composed of 94 hexagonal FAs with the width across flats of 72.2 mm and 91 MOX fuel elements of the diameter 6.0 mm. The FA spacing is 75 mm. The refueling interval is not less than 100 days. The core has negative temperature and power reactivity effects. The reactor vessel has the vertical dimension of about 11.6 m and the diameter of 2.5 m in the core accommodation area. The austenitic-steel reactor vessel is enclosed within a safeguard vessel to exclude the sodium loss. The reactor vessel has two inlet and two outlet nozzles. The core components are contained in a barrel intended to separate the coolant flows that enter and leave the reactor, arrange for the reactor vessel and core cooling (through the pressure header), and provide thermal and radiation protection for the barrel and reactor vessel walls. There are 8 CPS members in the core.

There are two reactor heat removal loops. The heat transfer from the reactor core to the ultimate heat sink (a turbine generating up to 60 MW(e)) uses a three-circuit sodium-sodium-water layout.

Description of the Safety Concept

In the event of anticipated operational occurrences and emergences, the maximum values of the MBIR parameters are limited by the inherent safety features. The MBIR is equipped with active safety systems based on a passive principle of action ensuring the regulatory safety requirements are met in full. The actuators for the scram system and the normal operation system have different designs. The primary circuit will include a system for the passive removal of decay heat by natural circulation. The circuit structure basis makes impossible for the primary circuit's radioactive sodium to enter the secondary circuit. The secondary circuit includes a steam generator emergency protection system and discharge tanks in case of emergencies with a sodium-water contact within the steam generator. The design approaches exclude severe damage to the core and release of radioactive contamination to beyond the safety barriers. The fuel melt catcher is located beneath the core inside the reactor vessel. The MBIR implements the single-failure, safe-failure, redundancy, independence, separation and diversity principles; the technical concepts adopted in the automated process control system (APCS) exclude or mitigate operator errors and reduce the operator load. The control interface supports a graphic representation of the holistic picture of the facility status via displays (video walls). The hardware of the integrated control and protection systems is based on modern components.

Deployment Status and Planned Schedule

At present, MBIR is at detailed design stage. A preliminary safety case study and a probabilistic safety analysis are conducted. The completion of the design activities and obtaining the construction license are scheduled for 2014. It is planned that the construction, commissioning and commencement of the experimental works will be completed by 2020.

PRISM (GE-Hitachi, USA)

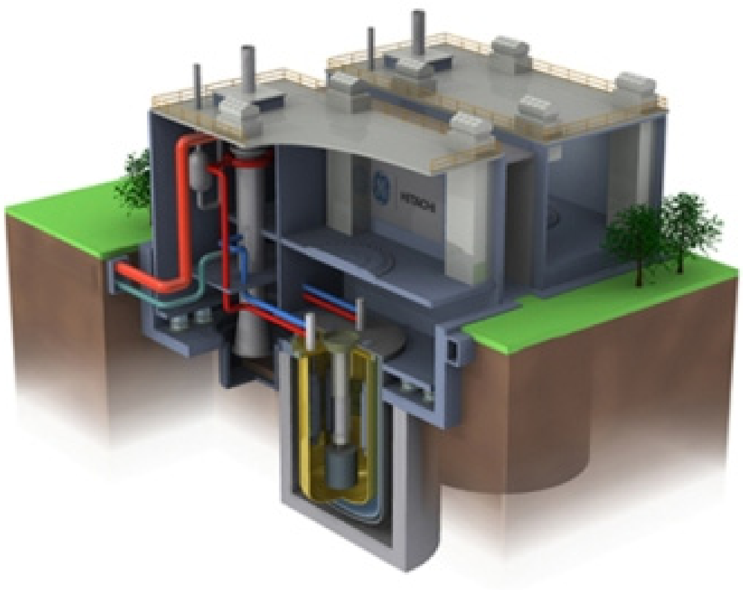


FIG. 6. Schematic View of PRISM

Reactor type:	*Liquid metal cooled fast breeder reactor*
Electrical capacity:	*311 MW(e)*
Thermal capacity:	*840 MW(th)*
Coolant:	*Sodium*
Primary Circulation:	*Forced circulation*
System Pressure:	*Low pressure operation*
Core Outlet Temperature:	*485°C*
Thermodynamic Cycle:	*Indirect Rankine Cycle*
Fuel Material:	*U-Pu-Zr*
Fuel Enrichment:	*26% Pu,10% Zr*
Fuel Cycle:	*18 months*
Reactivity Control:	*Rod insertion*
No. of safety trains:	*N/A*
Emergency safety systems:	*Passive*
RHRS:	*Passive reactor vessel auxiliary cooling system*
Design Life:	*N/A*
Design Status:	*Detailed design*
Seismic Design:	*N/A*
Predicted CDF:	*1E-6/reactor year*
Planned deployment:	*N/A*
Distinguishing Features:	*Underground containment on seismic isolators with a passive air cooling ultimate heat sink; part of the advanced recycling centre for spent nuclear fuel*

Introduction

The PRISM design uses a modular, pool type, liquid sodium cooled reactor. The reactor fuel element design most commonly considered is a metal alloy comprised of uranium, plutonium and zirconium. The reactor employs passive shutdown and decay heat removal features.

Description of the nuclear systems

The PRISM reactor core is designed to meet the following objectives:

- To increase fuel burnup;
- To limit the burnup reactivity swing; and
- To produce thermal power supporting an 18 to 24 month refuelling interval.

The reactor is designed to use a heterogeneous metal alloy core. The core consists of 198 fuel assemblies, 114 reflector assemblies, 66 radial shield assemblies, 10 control and 3 shutdown assemblies. The PRISM fuel and core can be tailored to specific missions ranging from plutonium disposition to the maximization of fuel efficiency.

The primary heat transport system is contained entirely within the reactor vessel. The flow path goes from the hot sodium pool above the reactor core through the intermediate heat exchangers (IHXs), where heat is transferred to the intermediate heat transport system (IHTS); the sodium exits the IHX at its base and enters the cold pool. Four electromagnetic pumps take suction from the cold pool and discharge into the high pressure core inlet plenum. The sodium is then heated as it flows upward through the reactor core and back into the hot pool.

Heat from the IHTS is transferred to a steam generator where superheated steam is produced. This high pressure, high temperature steam drives the turbine-generator to produce electricity.

Description of the safety concept

The passive shutdown characteristics of the reactor core provide diverse and independent means of shutdown in addition to the control rod scram. The passive features comprise several reactivity feedback properties including: the Doppler effect, sodium density and void, axial fuel expansion, radial expansion, bowing, control rod drive line expansion and reactor vessel expansion. The negative feedbacks maintain the reactor in a safe, neutronically stable condition.

The passive Reactor Vessel Auxiliary Cooling System (RVACS) provides primary cooling during all design basis accident conditions and Anticipated Transients without Scram (ATWS). This passive system operates effectively without electricity or operator intervention for an unlimited amount of time. Heat is transferred from the reactor vessel to the containment vessel by thermal radiation and then to the surrounding atmospheric air by natural convection. Redundant decay heat removal is provided by the Auxiliary Cooling System (ACS), which consists of natural circulation of air past the shell side of the steam generator. The combination of systems allows for reduced plant outages for inspections and maintenance.

Deployment Status and Planned Schedule

The PRISM design was initiated in 1981 with support from several U.S. Government development programs. The design incorporates test data and operating experience from previous U.S. sodium reactors such as GE’s Southwest Experimental Fast Oxide Reactor, and Experimental Breeder Reactor II. PRISM was reviewed by the U.S. Nuclear Regulatory Commission from 1987 to 1994 as part of a pre-application licensing review. GEH envisages PRISM most commonly deployed with integral used fuel recycling in areas interested in improving disposal strategies for used nuclear fuel. Also, due to PRISM’s capability to use plutonium bearing fuel, GEH is currently pursuing the use of PRISM to address the UK government’s plans to dispose of plutonium via reuse in power reactors. In 2012, the UK’s Nuclear Decommission Authority contracted GEH to carry out feasibility work in a number of key areas including the proposed commercial structure, the disposability of the fuel, the risk transfer model, the costs, and the ability to license in the UK.

TWR-P (TerraPower, U.S.A.)

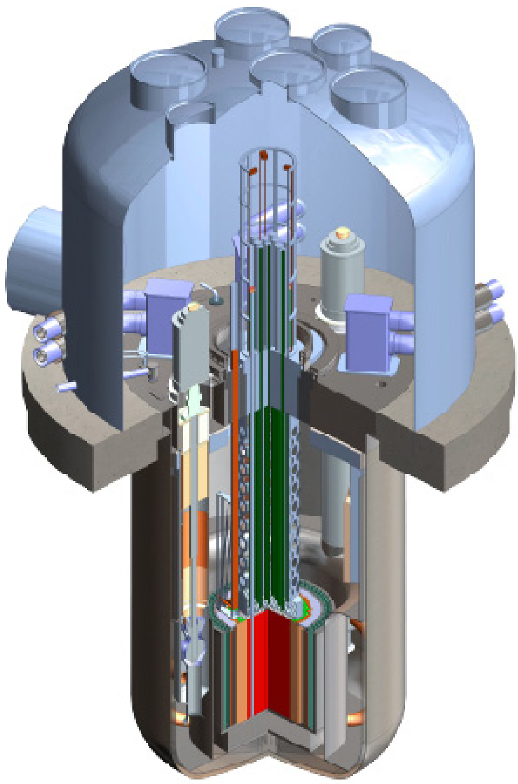


FIG. 7. Schematic View TerraPower's Prototype Reactor

Full name:	*Travelling Wave Reactor-Prototype (TWR-P)*
Designer:	*TerraPower*
Reactor type:	*Pool-type, Sodium-cooled Fast Reactor*
Electrical capacity:	*600 MW(e)*
Thermal capacity:	*1475 MW(th)*
Primary Circulation	*Forced convection using mechanical pumps*
System Pressure:	*0.1 MPa*
System Temperature:	*360/500 °C*
Fuel Material:	*Composed of uranium-zirconium metal alloy fuel slugs with a sodium thermal bond in ferritic-martensitic steel cladding*
Fuel Cycle:	*18-24 Months*
Residual heat removal systems:	*Passive - Direct Reactor Auxiliary Cooling System*
Design Life:	*40 Years*
Design status:	*Conceptual Design*
Planned deployment/1st date of completion:	*Start of Construction by 2018*
New/Distinguishing Features:	*Traveling wave reactor*

Introduction

A TWR is a class of reactors that is designed to operate indefinitely after a start-up period using only natural or depleted uranium. The waves that breed and deeply burn fissile nuclides in-situ travel relative to the fuel and provide the possibility for very long core life (~60 years). This long core life allows for higher fuel utilization, up to about 30 times greater than light water reactors (LWRs).

The breed and burn process is achieved through a "standing wave" by periodic re-shuffling of the fuel in such a manner that higher burn-up assemblies are moved to the periphery of the core and depleted uranium fuel assemblies replace the high burn-up fuel. Excess reactivity in the TWR core increases and control rods are slowly inserted to offset this increase. The re-shuffling is performed during the reactor is shut down each 18 to 24 months. TWR uses advanced metal fuel and advanced ferritic-martensitic steel cladding with capabilities to reach high peak burnups of ~30% and high doses necessary to enable breed and burn operation on depleted uranium. The TWR-P is a prototype of TWR to demonstrate TWR core fuel management, qualify TWR advanced fuel assemblies, and validate core radial expansion reactivity feedback and effectiveness of the core restraint system.

Description of the Nuclear System

TWR-P is a 1475MW(th)/600MW(e) gross liquid sodium cooled, fast neutron spectrum reactor that uses U-10%Zr metallic fuel with HT-9 ferritic-martensitic stainless steel clad. The plant's 4m diameter, 5.5m tall cylindrical core sits near the bottom of a 13.3m diameter, 17.65m tall reactor vessel which is enclosed within a guard vessel. This pool-type configuration has no radial piping penetrations through either vessel so the risk of loss of coolant accidents is eliminated. The large volume of sodium acts as a huge heat sink, so transients are fairly slow, providing operators time to respond to off-normal events.

The TWR-P employs enriched fuel assemblies (driver assemblies) with sealed sodium bonded metallic U-Zr10% fuel pins with 70% smear density, for which data and experience is available. To achieve flux profile and power and temperature gradients similar to those of a TWR and reproduce prototypical power histories of breed and burn assemblies, the core also employs feed fuel assemblies made of depleted uranium. These fuel assemblies have also U-Zr10% metal fuel with 70% smear density.

The core and all primary cooling system components are contained in the reactor vessel with the addition of a number of in-vessel structures. All vessels and structures in direct contact with primary sodium coolant are fabricated from 316H stainless steel. Both vessels and a number of major components are supported from the reactor head, which is supported by a bearing system interfaced to a concrete ledge in the reactor building

Description of the Safety Concept

TWR-P is similar to other pool-type, sodium-cooled fast reactors in that Loss of Coolant Accidents are eliminated from the design bases. The primary design basis accidents are Loss of Flow and Loss of Heat Sink. Transient overpower events are less frequent because interlocks prevent excessive rod withdrawal during operation. The TWR-P plant employs for emergency decay heat removal four Direct Reactor Auxiliary Cooling System (DRACS) loops dissipating heat to air as an ultimate heat sink, each of which is capable of removing enough heat to protect public health and safety. These NaK filled loops are completely passive and require no electricity to activate or operate.

Deployment Status and Planned Schedule

The TWR-P could be constructed between 2018 and 2023. After a period of testing and optimization, commercial plants could potentially be licensed with start up in the late 2020s or early 2030s.

HEAVY LIQUID METAL-COOLED FAST REACTOR DESIGNS

MYRRHA (SCK•CEN, Belgium)

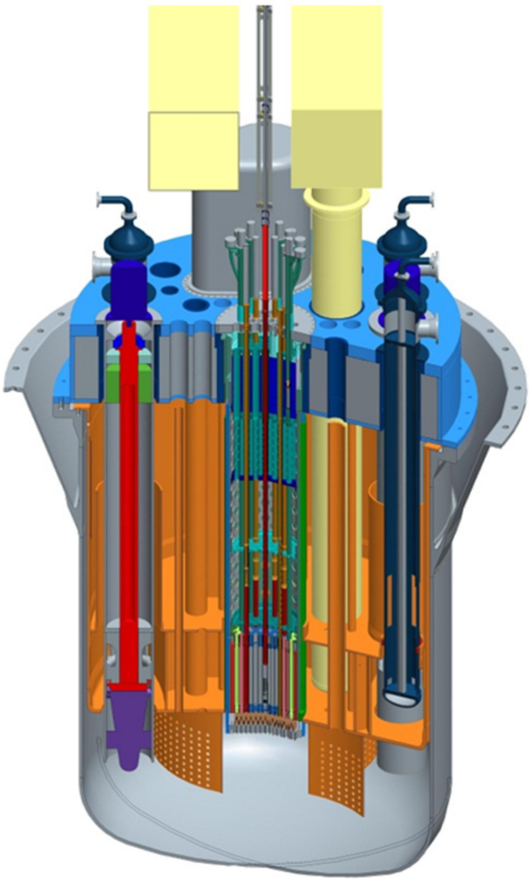


FIG.15. View of the core of MYRRHA

Full name:	*Multi-purpose hYbrid Research Reactor for High-tech Applications (MYRRHA)*
Designer:	*Belgian Nuclear Research Centre (SCK•CEN)*
Reactor type:	*Heavy liquid metal-cooled Reactor*
Thermal capacity:	*100 MW(th)*
Coolant	Heavy liquid metal
Primary Circulation	Natural
System Pressure:	*0 MPa*
System Temperature:	*270-410 °C*
Fuel Material:	*MOX*
Fuel Cycle:	*3 Months*
Emergency safety systems:	*Passive*
Residual heat removal systems:	*Passive*
Design Life:	*40 Years*
Design status:	*Concept*
Planned deployment/1st date of completion:	*2023*
New/Distinguishing Features:	*ADS* *LBE*

Introduction

MYRRHA is an innovative multipurpose research reactor currently being developed at SCK•CEN. MYRRHA is conceived as an accelerator driven system but, with the removal of the spallation target and the insertion of control and safety rods, is also able to operate in critical mode.

Description of the Nuclear System

The driver of an ADS like MYRRHA is the accelerator, which provides high energy protons to the spallation target, which in turn creates primary neutrons to feed the subcritical core. The MYRRHA accelerator is a linear accelerator able to provide a beam with capacity of 600 MeV and an average beam current of 3.2 mA.

At the present state of design, the core consists of MOX fuel pins. Thirty seven positions can be occupied by In-Pile Test Sections (IPS) or by the spallation target (the central one of the core in sub-critical configuration) or by control and shutdown rods (in the core critical configuration). This gives a large flexibility in the choice of the more suitable position (neutron flux) for each experiment.

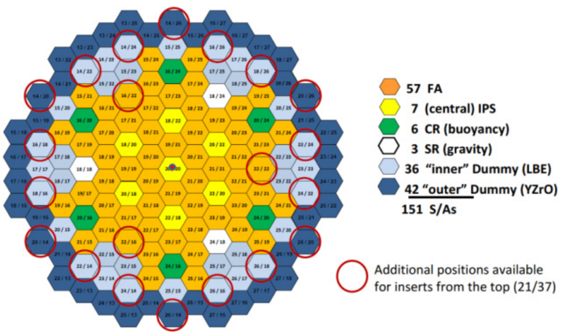


FIG. 16. MYRRHA core

MYRRHA is a pool-type ADS, which means that the reactor vessel houses all the primary systems. The vessel is closed by the reactor cover which supports all the in-vessel components. A diaphragm inside the vessel separates the hot and cold LBE plenums, supports the in-vessel fuel storage and provides a pressure separation. The room directly above the core is occupied by IPS penetrations and the beam tube, thus fuel handling to (un)load the core must be done from underneath. Two in-vessel fuel handling machines are used to fulfil this function.

The primary, secondary and tertiary cooling systems have been designed to evacuate a maximum core power of 110 MW(th). The primary cooling system consists of two pumps and four primary heat exchangers (PHX). These pumps deliver the LBE to the core. The secondary cooling system is a water cooling system, providing pressurized water to the PHX. The tertiary cooling system is an air cooling system.

Description of the Safety Concept

In case of loss of the primary flow, the beam must be shut off in subcritical case and the shutdown rods inserted in critical mode. The primary, secondary and tertiary cooling system have been designed to remove the decay heat using natural convection. Ultimate decay heat removal is achieved by the reactor vessel cooling system, also by natural convection.

In the unlikely event of a reactor vessel breach, the reactor pit will fulfil the function of secondary containment to keep the LBE in place.

Deployment Status and Planned Schedule

The MYRRHA design has now entered into the Front End Engineering Phase, covering the period 2012-2014. At the end of this phase, the purpose is to have progressed in such a way in the design of the facility that the specifications for the different procurement packages of the facility can be written, to have adequately addressed the remaining outstanding R&D issues, to have obtained a positive licensability statement by the Belgian safety authorities and to have formed the international members' consortium for MYRRHA. MYRRHA will be operational around 2023.

CLEAR-I (INEST, China)

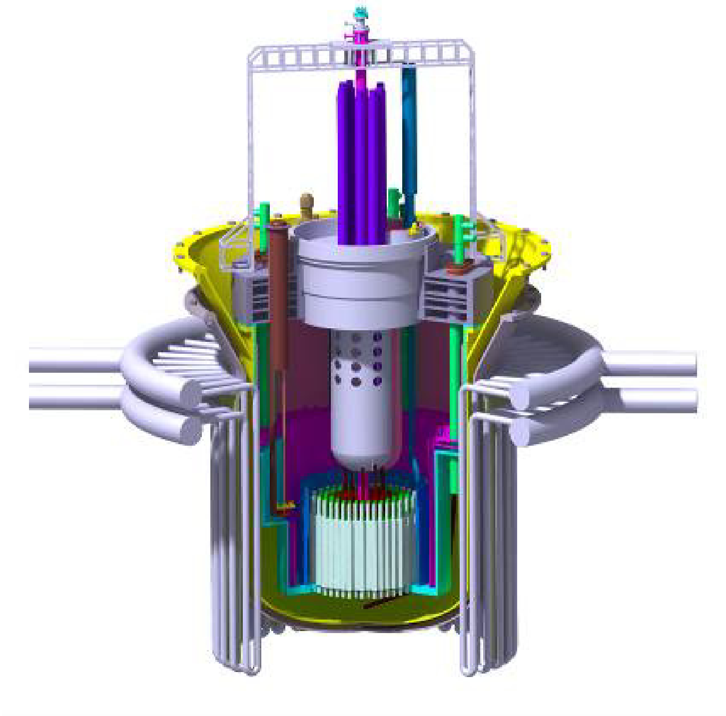


FIG. 17. View of the core of CLEAR-I

Full name:	*China LEAd-based Research Reactor*
Designer:	*Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences*
Reactor type:	*Heavy Liquid metal-cooled Reactor*
Thermal capacity:	*10 MW(th)*
Coolant	*Lead Bismuth Eutectic*
Primary Circulation	*Natural*
System Pressure:	*0.05 MPa*
System Temperature:	*260°C -390 °C*
Fuel Material:	UO_2
Fuel Cycle:	*120 Months/10 Years*
Emergency safety systems:	*Passive*
Residual heat removal systems:	*Passive*
Design Life:	*30 Years*
Design status:	*Preliminary Engineering Design*
Planned deployment/1st date of completion:	*2023*
New/Distinguishing Features:	*Dual-mode operation: Lead-bismuth cooled reactor and ADS*

Introduction

The Chinese Academy of Sciences (CAS) launched an engineering project to develop ADS system and lead-based fast reactors named China LEAd-based Reactor (CLEAR) series.

For the first phase, a 10 MW(th) lead-bismuth cooled research reactor (CLEAR-I) will be developed by the Institute of Nuclear Energy Safety Technology (INEST).

CLEAR-I was designed to be operated in two modes: critical operation for heavy liquid metal fast reactor technology test and accelerator driven subcritical operation for ADS system.

Description of the Nuclear System

In the reference parameters design of CLEAR-I, pool-type configuration is selected, where the reactor vessel is placed inside a safety vessel. The thermal power is 10MW and no electric power is generated. Lead-bismuth (LBE) is adopted as primary coolant and UO_2 with ^{235}U enrichment 19.75% is chosen as the first loading fuel. MOX is the alternative fuel. Hexagonal wrapped fuel assemblies are used in a hexagonal lattice core, which cladding material is SS316Ti and structure material is SS316L. In critical operation mode, the core consists of 76 fuel assemblies, 60 reflector assemblies and 48 shielding assemblies, while in sub-critical operation mode, a part of fuel assemblies will be replaced with reflector assemblies and the K_{eff} will be 0.98.

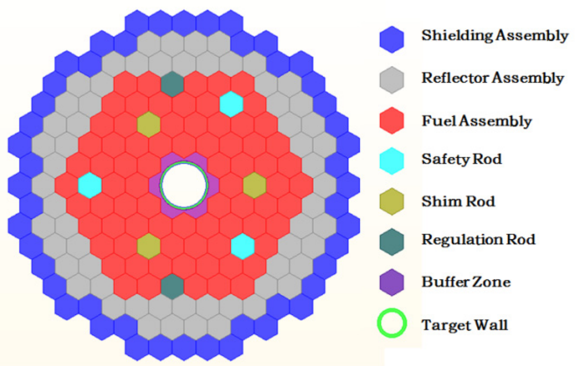


FIG.18. Scheme of CLEAR-I core

Description of the Safety Concept

Considering the characteristic of low power density of reactor and large expansion coefficient of lead-bismuth, the primary cooling system is designed to rely on natural circulation entirely, which improves safety characteristics of the reactor. The operation temperature and coolant velocity is designed relatively low, which can reduce the corrosion-erosion effect effectively. A Reactor Vessel Air Cooling System (RVACS) is incorporated in CLEAR-I to remove the decay heat in emergency. The RVACS is designed to be consisted of tens tubes installed outside the reactor guard vessel to cool the reactor by thermal radiation. The heat removal by the RVACS is continued by means of natural circulation of air. Double-wall containment system concept is used in the design. Main radioactive nuclides are contained and shielded by the internal containing chambers with ventilation system. The integrity of the structure is protected by the external airtight building.

Deployment Status and Planned Schedule

The conceptual designs of CLEAR-I have been performed, preliminary engineering design are underway. Multi-functional large-scaled lead-bismuth experimental loops KYLIN-II for material test, thermal-hydraulics test and safety test will be constructed before the end of 2013.

The design of pre-research platform of key reactor components, including control rod drive system, refuelling mechanism and the reactor simulator, has been accomplished; construction will be carried out in 2013 .The High Intense D-T Neutron Generator HINEG for neutron experiment and codes' validation will be finished constructing by INEST around 2014.

The analysis of typical accidents and the compilation of the Preliminary Safety Analysis Report are in progress. CLEAR-I will be operational around 2023.

ALFRED (Ansaldo Nucleare, Europe/Italy)

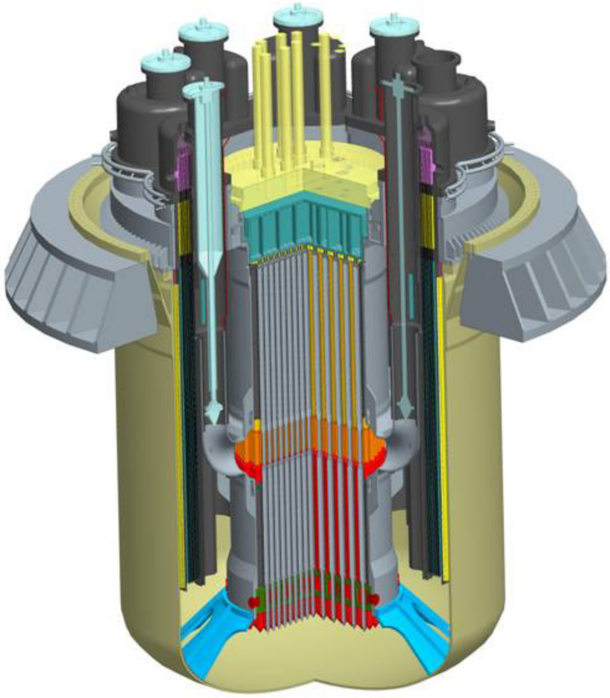


FIG. 19. ALFRED primary system.

Full name:	*Advanced Lead Fast Reactor European Demonstrator*
Designer:	*Ansaldo Nucleare*
Reactor type:	*Lead-cooled Fast Reactor*
Electrical capacity:	*125MW(e)*
Thermal capacity:	*300MW(th)*
Primary Coolant	*Lead*
Primary Circulation	*Forced*
Primary System Pressure:	*< 0.1MPa*
Primary System Temperature:	*400÷480 °C*
Secondary Coolant:	*Water/Superheated-Steam*
Secondary System Pressure:	*18 MPa*
Superheated Steam Temperature:	*450 °C*
Fuel Material:	*MOX*
Fuel Cycle/Residence time:	*12 Months / 5 Years*
No. of safety trains:	*4 trains*
Emergency safety systems:	*No injection safety systems are needed*
Residual heat removal systems:	*2 DHR systems, 4 loops each - Passive*
Design Life:	*40 Years*
Design status:	*Conceptual Design*
Planned deployment/1st date of completion:	*2025*
New/Distinguishing Features:	*Demonstrator reactor for European LFR*

Introduction

The Advanced Lead Fast Reactor European Demonstrator (ALFRED) conceptual design has been carried out in the frame of the EU FP7 LEADER (Lead-cooled European Advanced Demonstration Reactor) Project. ALFRED is a 300 MW(th) pool system developed to demonstrate the viability of the European LFR (ELFR) technology for use in the future commercial power plant. ALFRED design is as close as possible to the reference ELFR using proven and already available technical solutions to proceed to construction in the short term.

Description of the Nuclear System

ALFRED primary system configuration is a pool-type with all components removable. It presents a simple flow path and a low pressure drop of the primary coolant allowing an efficient natural circulation. The primary coolant leaving the core flows upward through the Primary Pump (PP) vertical duct and then, through the Steam Generator (SG) inlet holes, flows downwards in the SG shell and feeds the cold plenum back to the core. The volume between the primary coolant free levels and the reactor roof is filled with Argon. The Reactor Vessel (RV) is cylindrical with a torispherical bottom head and it is anchored to the reactor cavity from the top. A cone frustum, welded to the bottom head, provides radial restraint of the Inner Vessel (IV). A steel liner covering the reactor pit constitutes the Safety vessel (SV). The volume between the RV and SV is such that, in case of RV leak, the primary coolant still covers the SG inlet window maintaining the natural circulation flow path. The IV has the main functions of core support and hot/cold plena separation. It is fixed to the cover by bolts and it is radially restrained at bottom. The Core Support plate is mechanically connected to the IV with pins for easy removal/replacement. The core, with a total power of 300 MW(th), is constituted by 171 wrapped hexagonal Fuel Assemblies (FAs), 12 Control Rods (CRs) and 4 Safety Rods (SRs), surrounded by 108 Dummy Elements. Hollow pellets of MOX fuel with maximum Plutonium enrichment of 30% are used. Each FA extends to the cover gas plenum, is long about 8 m and it consists of 127 fuel pins restrained by spacer grids. A Tungsten ballast in the upper part of the FA is used to overcome buoyancy forces. The SG and PP are integrated into a single vertical unit. Eight SG/PP units are located in the annular space between the IV and the RV walls. SGs are bayonet type with double walls. The double wall gap is filled with Helium and high thermal conductivity particles. The double wall concept prevents water/lead interaction in case of break of one tube wall and, moreover, a tube wall break event can be easily detected by the continuous monitoring of the Helium gap pressure.

Description of the Safety Concept

ALFRED is equipped with two diverse, redundant and separate shutdown systems: the first system, which performs also the control function, is made of absorber rods passively inserted by buoyancy from the bottom of the core; the second is made of absorbers rods passively inserted by a pneumatic system (using depressurization) from the top of the core. The Decay Heat Removal (DHR) system consists of two passive, redundant and independent systems, each one composed of four Isolation Condenser systems (ICs) connected to four SGs secondary side. Three out of four ICs are sufficient to remove the decay heat power. Both systems are passive, with actively actuated valves, equipped with redundant and locally stored energy sources. 2D seismic isolators are installed below the reactor building to cut the horizontal seismic loads.

Deployment Status and Planned Schedule

ALFRED reached the conceptual design maturity level in 2013. ALFRED is an essential step in the evolution of the LFR technology. The Road Map for the realization of the industrial scale First-Of-A-Kind European LFR (the ELFR) requires the realization of the ALFRED demonstrator around 2025.

ELFR (Ansaldo Nucleare, Europe/Italy)

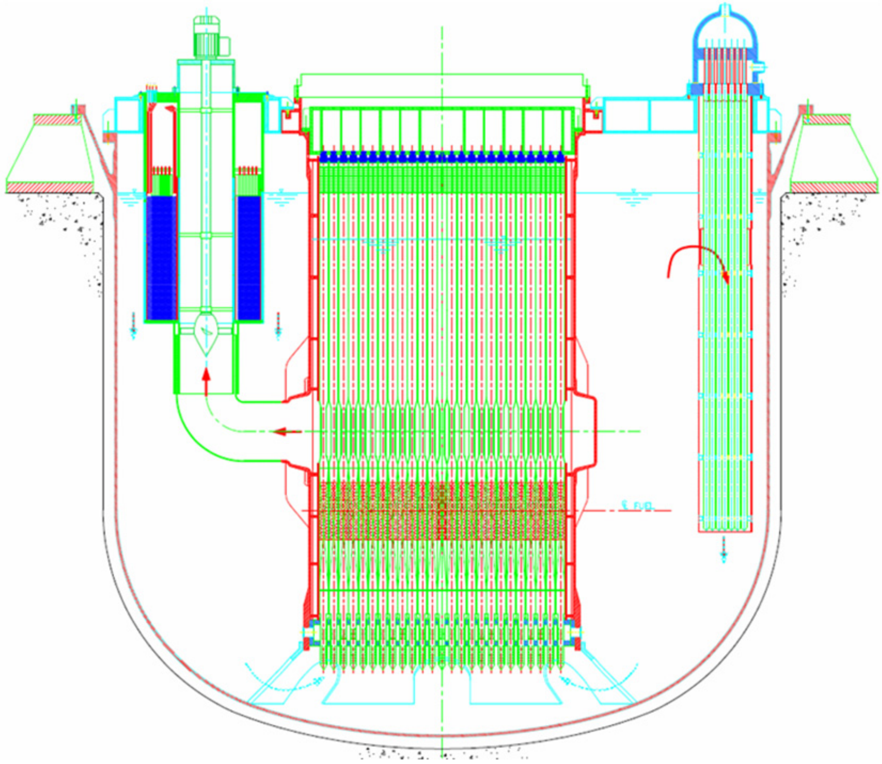


FIG. 20. ELFR primary system.

Full name:	*European Lead Fast Reactor*
Designer:	*Ansaldo Nucleare*
Reactor type:	*Lead-cooled Fast Reactor*
Electrical capacity:	*630 MW(e)*
Thermal capacity:	*1500 MW(th)*
Primary Coolant	*Lead*
Primary Circulation	*Forced*
Primary System Pressure:	*< 0.1MPa*
Primary System Temperature:	*400÷480 °C*
Secondary Coolant:	*Water/Superheated-Steam*
Secondary System Pressure:	*18 MPa*
Superheated Steam Temperature:	*450 °C*
Fuel Material:	*MOX*
Fuel Cycle /Residence time:	*30 Months /5 Years*
No. of safety trains:	*4 trains*
Emergency safety systems:	*No injection safety systems are needed*
Residual heat removal systems:	*2 DHR systems, 4 loops each - Passive*
Design Life:	*60 Years*
Design status:	*Conceptual Design*
Planned deployment/1st date of completion:	*2040 - 2050*
New/Distinguishing Features:	*First Of A Kind European LFR*

Introduction

The conceptual design of the European Lead Fast Reactor (ELFR) has been developed in the frame of the EU FP7 LEADER (Lead-cooled European Advanced Demonstration Reactor) Project. ELFR is a 1500 MW(th) pool system First Of A Kind (FOAK) of a future industrial size plant. The ELFR design complies with all GEN IV goals. ELFR has a closed fuel cycle (the so called adiabatic core). The nominal power is 1500 MW(th) and, considering the net thermal efficiency of the selected superheated steam cycle, the electric power capacity is about 630 MW(e).

Description of the Nuclear System

The ELFR primary system is a pool-type configuration with all components removable. It presents a simple flow path and low pressure drop of the primary coolant allowing an efficient natural circulation. The primary coolant leaving from the core flows upward through the PP vertical duct, then radially through the SG, then to the cold plenum through perforated double-wall casing and from the cold plenum reaches the inlet of the core. The volume between the primary coolant free levels and the reactor roof is filled with Argon. The RV is cylindrical with a tori spherical bottom head and it is anchored to the reactor cavity from the top. A cone frustum, welded to the bottom head, provides the radial restraint of the Inner Vessel (IV). A steel liner covering the reactor pit constitutes the Safety Vessel (SV). The volume between the RV and SV is such that, in case of RV leak, the primary coolant continues to cover the SG inlet maintaining the flow path. The IV has the main functions of core support and hot/cold plena separation. It is fixed to the cover by bolts and it is radially restrained at the bottom. The Core Support plate is mechanically connected to the IV with pins for easy removal/replacement. The core of a total power of 1500 MW(th), is constituted by 427 hexagonal wrapped Fuel Assemblies (FAs) arranged in two zones of 157 (inner) FAs and 270 (outer) FAs with a different hollow pellet diameter, 12 CRs and 12 Safety Rods (SRs), surrounded by 132 Dummy Elements. Each FA extends above the lead free level and is long about 10 m. The FA consists of 169 fuel pins restrained by spacer grids. A Tungsten ballast in the upper part of the FA is used to overcome buoyancy forces. The ELFR is designed to have a closed fuel cycle (adiabatic core) with a conversion ratio of 1.07. The SG (spiral-type) and PP are integrated into a single vertical unit and placed vertically in the cold pool trough circular penetrations in the reactor roof. Eight SG/PP units are used in ELFR.

Description of the Safety Concept

ELFR is equipped with two diverse, redundant and separate shutdown systems: the first system, that has also the control function, is made of absorber rods passively inserted by buoyancy from the bottom of the core; the second system is made of absorbers rods passively inserted by pneumatic system (using depressurization) from the top of the core. The Decay Heat Removal (DHR) system consists of two diverse, passive, redundant and independent systems: the first system is composed of four Isolation Condenser systems (ICs) connected to four out of eight SGs secondary side; the second system is composed of four ICs connected to four Dip Coolers located in the cold pool. Three out of four ICs of each system are sufficient to remove the decay heat power. Both systems are passive, with actively actuated valves, equipped with redundant and locally stored energy sources. 2D seismic isolators are installed below the reactor building to cut the horizontal seismic loads.

Deployment Status and Planned Schedule

The ELFR is in the conceptual design phase. Demonstration of the LFR technology is on-going and the operation of a FOAK industrial-scale ELFR is foreseen around 2040-2050. The achievement of this target is strictly connected to the realization of the ALFRED demonstrator around 2025.

PEACER (Seoul National University, Republic of Korea)

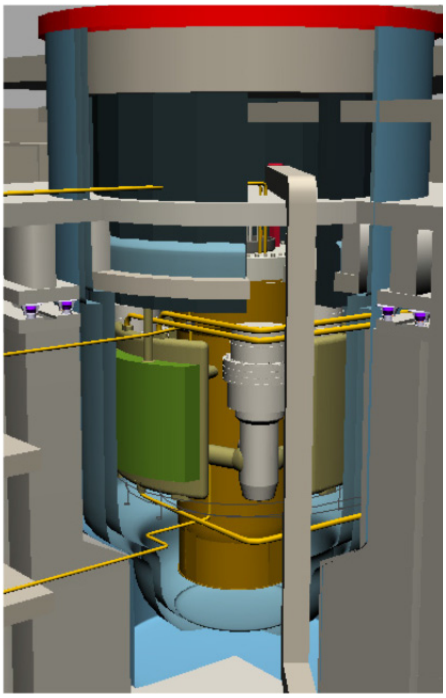


FIG. 21. 3D CAD Drawing of PEACER-300 MW(e)

Full name:	*Proliferation-resistant Environment-friendly Accident-tolerant Continuable and Economical Reactor*
Designer:	*Seoul National University*
Reactor type:	*Lead-bismuth cooled Reactor*
Electrical capacity:	*300 MW(e)*
Thermal capacity:	*850 MW(th)*
Coolant	*Pb-Bi (45-55% Wt.%) eutectic alloy*
Primary Circulation	*Forced*
System Pressure:	*0.1 MPa (primary) 8.0 MPa (secondary)*
System Temperature:	*300°C - 400 °C*
Fuel Material:	*U-TRU-Zr Alloy (58.04-31.07-10.88 Wt. %)*
Fuel Cycle:	*12 Months (Annual reload and in-house pyroprocess)*
No. of safety trains:	*1 train*
Emergency safety systems:	*Passive*
Residual heat removal systems:	*Passive*
Design Life:	*60 Years*
Design status:	*Conceptual Design*
New/Distinguishing Features:	*Reactor Vessel Air Cooling System (RVACS)*

Introduction

A group of researchers at the Seoul National University (SNU) conducted a feasibility study on accelerator-driven nuclear waste transmutation concept, in early 1990's. The Nuclear Materials Laboratory at SNU (SNUMAT) had carried out experimental investigations on Pb-Bi eutectic alloy as an alternative coolant to sodium for use in transmutation systems then under the consideration. Positive experimental verification of its stable physico-chemical properties of LBE enabled SNUMAT to initiate the development of the first dedicated transmutation fast reactor design cooled by LBE, designated in 1997 as Proliferation-resistant, Environment-friendly, Accident-tolerant, Continuable and Economical Reactors (PEACER).

The first PEACER design had a pancake-shaped fast reactor core with metallic fuels containing transuranic (TRU) elements extracted from LWR spent fuels by on-site pyrochemical partitioning processes under through multi-national controls. The upper core region of PEACER has thermalized neutrons for stabilization of long-lived fission products including technetium-99 and iodine-129. With a rated power of 550 MW(e), the holt-leg coolant temperature of PEACER-550 was reduced to 400°C in order to ensure 60 year design life of structural materials including steam generator and 3 year life of fuel cladding material, even with well-known high corrosivity of LBE.

Description of the Nuclear System

The PEACER core design yields a support ratio of two, i.e. the amount of transuranic element (TRU) burnt in the transmutation core being twice the amount of TRU produced from a light water reactor core for the same electricity generation. The high support ratio goal could be achieved by using high diameter-to-height ratio for the PEACER core. Peripheral thermal traps are introduced to produce epithermal neutrons for the transmutation of long living fission products, including Tc-99, I-129, etc..

The second edition of PEACER was designed to have a rated power of 300 MW(e). Each of three primary loops for PEACER-300 is equipped with a once-through steam generator and a centrifugal pump. Reactor inlet and outlet temperatures (300-400°C) are chosen with considerations on materials endurance, transient operability and thermal efficiency. By taking advantage of chemical stability of LBE, the balance of plant with standard superheated Rankine cycle is coupled to the primary coolant system through once-through steam generators that facilitate water flow inside tubes.

Description of the Safety Concept

Accident-tolerance requirement mandated the development of a RVACS. It is expected that RVACS will provide passive cooling capability for decay heat for PEACER-300. An innovative passive water cooling system for the reactor vessel outer wall has been introduced to allow for the system up-scaling from 300 MW(e) to large scale systems with passive safety. As LBE has high density, earthquake can exert severe load on structural components. Therefore seismic isolators are extensively employed for entire plant including pyrochemical partitioning facility. Three dimensional seismic isolators are applied for all the reactor systems, including the containment.

Deployment Status and Planned Schedule

PEACER has not been developed beyond the conceptual design or been built for any commercial system due to inadequate technical confidence. A scaled full-height test loop, HELIOS, has been operational since 2005 to this end. A small modular transmutation reactor named as PASCAR (Proliferation-resistant Accident-tolerant Self-supported Capsular and Assured Reactor) is being developed for demonstrating most design goals of PEACER.

BREST-OD-300 (RDIPE, Russian Federation)

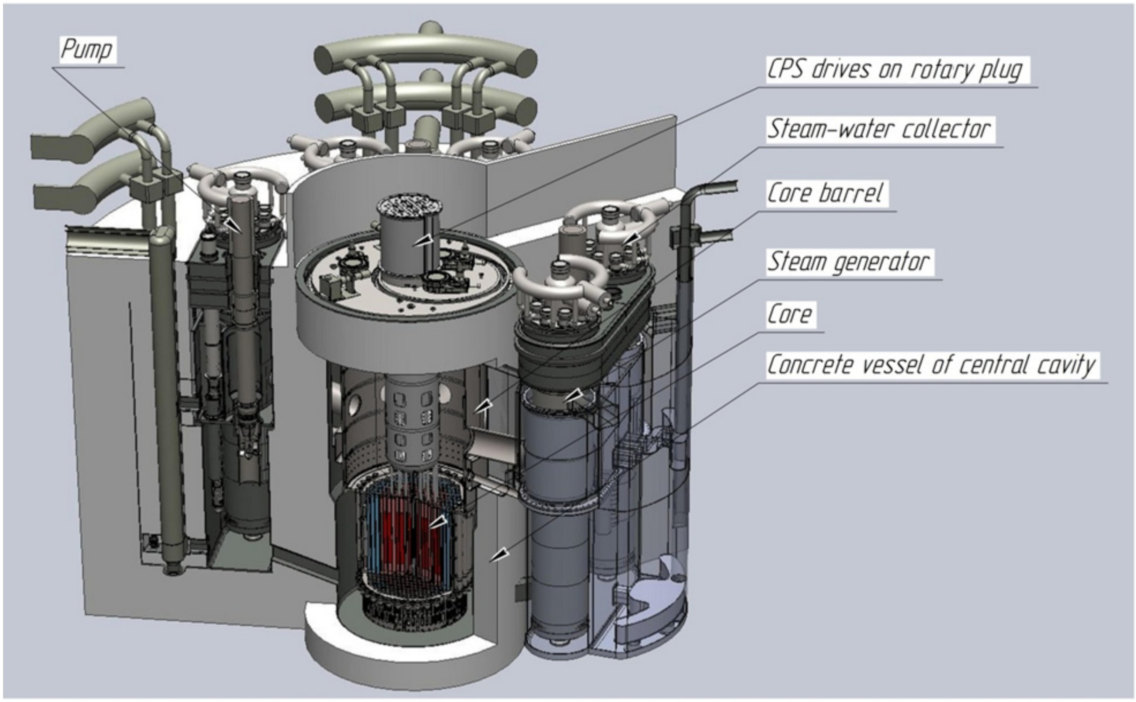


FIG. 22. Schematic representation of BREST-OD-300

Reactor type:	*Liquid metal cooled fast reactor*
Electrical capacity:	*300 MW(e)*
Thermal capacity:	*700 MW(th)*
Coolant:	*Lead*
Primary circulation:	*Forced circulation*
System pressure:	*Low pressure operation*
Core inlet/outlet temperature:	*420 / 540°C*
Thermodynamic cycle:	*Indirect Rankine cycle*
Fuel material:	*PuN–UN*
Fuel enrichment:	*~13,5%*
Fuel cycle:	*5-6 years, partial refuelling – 1/year*
Reactivity control:	*Shim and automatic control rods* ($\Delta\rho \approx 14 \cdot \beta_{eff}$)
Emergency safety systems:	*Emergency protection rods* ($\Delta\rho \approx 6 \cdot \beta_{eff}$)
Residual heat removal systems:	*Passive and time-unlimited direct residual heat removal from the lead circuit system via natural air circulation with heat discharge in the atmosphere*
Design life:	*30 years*
Design status:	*Detailed design*
Seismic design:	*VII–MSK 64*
Planned deployment:	*Unit startup 2020*
Distinguishing features:	*High level of inherent safety due to natural properties of the lead, fuel, core and cooling design*

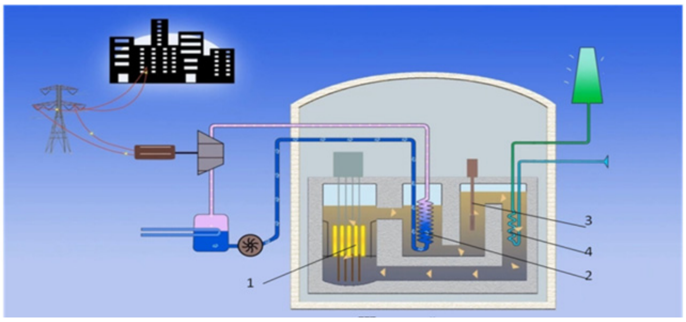


FIG. 23. BREST-OD-300 coolant circulation circuit

1–core; 2–steam generator; 3–reactor coolant pump (RCP); 4–emergency cooling system channel

Introduction

BREST is a lead cooled fast reactor fuelled with uranium plutonium mononitride (PuN–UN) that uses a two circuit heat transport system to deliver heat to a steam turbine.

Description of the nuclear systems

The adopted fuel exhibits high density and high heat conductivity, and is compatible with lead and the fuel cladding of chromium ferritic-martensitic steel. To provide a significant coolant flow area, the level of power removed by natural lead circulation is increased, the coolant preheating temperature is reduced and cooling losses in the damaged FAs are primarily excluded in the case of local flow rate blockage; no core FAs have shrouds. The FA design allows radial coolant overflow in the core that prevents overheating of the damaged FA.

The reactor of a pool-type design has an integral lead circuit accommodated in one central and 4 peripheral cavities of the concrete steel-lined vessel.

The central cavity houses the core barrel together with the side reflector, the CPS rods, the SFA storage and the shell that partitions the hot and the cold lead flows. Four peripheral cavities (according to the loop number) accommodate the SG-RCP units, heat exchangers of the emergency and normal cool-down systems, filters and auxiliary components. The cavities have hydraulic interconnection.

Better radial uniformity of fuel assembly power levels and coolant heating is ensured by regionalized fuelling and lead flow rates by placing assemblies with smaller-diameter rods in central regions, and larger-diameter in peripheral regions.

Description of the safety concept

The use in integral design of the BREST reactor of a high-boiling (~2000 K), radiation-resistant, low-activated lead coolant, which is chemically inert with water and air, does not require high pressure in the primary circuit, and excludes the potentiality of accidents with a loss of coolant and heat removal, fires and explosions in a contact with the environment. The lead coolant properties in combination with a dense, heat-conducting nitride fuel provide conditions for complete plutonium breeding in the core (CBR≥1). That results in a small operating reactivity margin ($\Delta\rho<\beta_{eff}$) and enables power operation without prompt-neutron reactor power excursions.

The safety analysis has shown that none of the considered initial events involving a fast introduction of reactivity up to its full margin and interruption of the forced coolant circulation leads to accidents with fuel damage and inadmissible radioactive or toxic releases, even in the case of a failure of the reactor's active safety systems. Therefore no population evacuation or resettlement is required.

Deployment status

The BREST-OD-300 power unit is designed as a pilot and demonstration unit intended for studying the reactor facility operation in different modes and optimizing all processes and systems that support reactor operation. Furthermore, BREST-OD-300 is also considered the prototype of a fleet of medium sized power reactors.

SVBR-100 (AKME Engineering, Russian Federation)

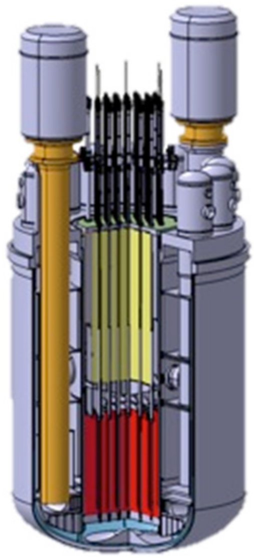


FIG. 24. Schematic View of SVBR-100

Reactor type:	*Liquid metal cooled fast reactor*
Electrical capacity:	*101 MW(e)*
Thermal capacity:	*280 MW(th)*
Coolant:	*Lead–bismuth eutectic alloy*
Primary circulation:	*Forced circulation*
System pressure:	*6.7 MPa*
Core outlet temperature:	*490°C*
Thermodynamic cycle:	*Indirect Rankine cycle*
Fuel material:	UO_2
Fuel enrichment:	*16,5%*
Fuel cycle:	*7–8 years*
Reactivity control:	*Control rod mechanism*
No. of safety trains:	*N/A*
Emergency safety systems:	*Passive and active*
Residual heat removal systems:	*Passive*
Design life:	*60 years*
Design status:	*Complete reactor and power plant design expected by 2014*
Seismic design:	*For structures and pipelines: horizontal PGA=0,12 g (7-point on MSK scale), for reactor module equipment: horizontal PGA=0,25 g (8-point on MSK scale)*
Predicted CDF:	*<1E-7/reactor year*
Distinguishing features:	*Closed nuclear fuel cycle with mixed oxide uranium plutonium fuel; operation in a fuel self-sufficient mode*
	**) figures shown above are targeted parameters for SVBR-100*

Introduction

The SVBR-100 is an innovative, small, modular fast reactor with lead–bismuth coolant (LBC). In the Russian Federation, lead–bismuth cooled reactor technology relies on the unique experience gained thanks to the development and operation of HLMC reactors for submarine propulsion (15 reactors, 80 reactor-years), and SFRs. It has provided knowledge and skills in issues such as: ensuring the corrosion resistance of structural materials, controlling the LBC quality and the mass transfer processes in the reactor circuit, ensuring the radiation safety of the personnel carrying out work with equipment contaminated with the ^{210}Po radionuclide, and multiple LBC freezing and unfreezing in the reactor facility.

Fuel cycle option

The SVBR-100's reactor core operates without any partial refuelling. The fresh fuel is loaded as a single cartridge while the spent nuclear fuel is unloaded cassette by cassette. This design has the capability to utilize various fuel cycles. The first stage will be typical uranium oxide fuel, leading to a core breeding ratio (CBR) of 0.84; mixed oxide fuel can also be used, leading to a CBR just near one. Using UO_2 as the starting fuel, the closed fuel cycle can be realized in 15 years. Uranium nitride and uranium-plutonium fuel can also be used to improve safety and fuel cycle characteristics. The SVBR-100 reactor pursues resistance to nuclear fissile material proliferation by using uranium with enrichment below 20%, while using uranium oxide fuel in the initial core. The reactor is designed to operate for eight years without core refuelling.

Reactor coolant system

The entire primary equipment circuit of SVBR-100 is contained within a robust single reactor vessel; LBC valves and pipelines are eliminated. A protective enclosure surrounds the single unit reactor vessel. The heat generated in the core is transferred to a two-circuit heat removal system and a steam generator with a multiple circulation, which represent the secondary coolant system. Natural circulation of the coolant in the reactor is sufficient to passively cool down the reactor and prevent hazardous superheating of the core.

Description of the safety concept

The combination of a fast reactor design, HLMC and integral reactor layout aim to ensure that the SVBR-100 reactor system meets IAEA international safety standards for prevention of severe accidents and inherent safety, according to analysis and studies. The SVBR-100 design pursues a high level of safety with inherent self-protection and passive safety by use of chemically inert LBC and integral arrangement of the primary circuit equipment in a single vessel, operating at approximately atmospheric pressure. The NSSS doesn't include materials releasing hydrogen. This combination of features excludes the loss of coolant accident (LOCA) type accidents, the possibility of chemical explosions and fires, as well as high-pressure radioactive releases like in the Fukushima-Daiichi accident. This will possibly allow the reactor design to exclude many safety systems required for traditional type reactors and to simplify and reduce the cost of the power plant. Safety systems in the reactor facility include fusible locks of additional safety rods to provide passive shutdown, bursting disc membrane to prevent over-pressurization and passive removal of residual heat in the event of a blackout by evaporation of water.

Deployment status and planned schedule

On June, 15th 2006 The Rosatom Scientific and Technical Council has approved the development of the technical design of the experimental industrial power unit based on SVBR-100. A complete reactor and power plant design are expected to be completed by 2014, along with a preliminary safety report. Siting licence works are underway. Construction licence is expected to be obtained in 2014. Meanwhile, the first SVBR-100 training simulator for personnel was developed; education course should start towards the end of 2013. The first trial 100 MW(e) unit planned to be constructed by 2017 at Dimitrovgrad, Ulyanovsk region near the Russian State Atomic Reactor Research Institute.

ELECTRA (KTH, Sweden)

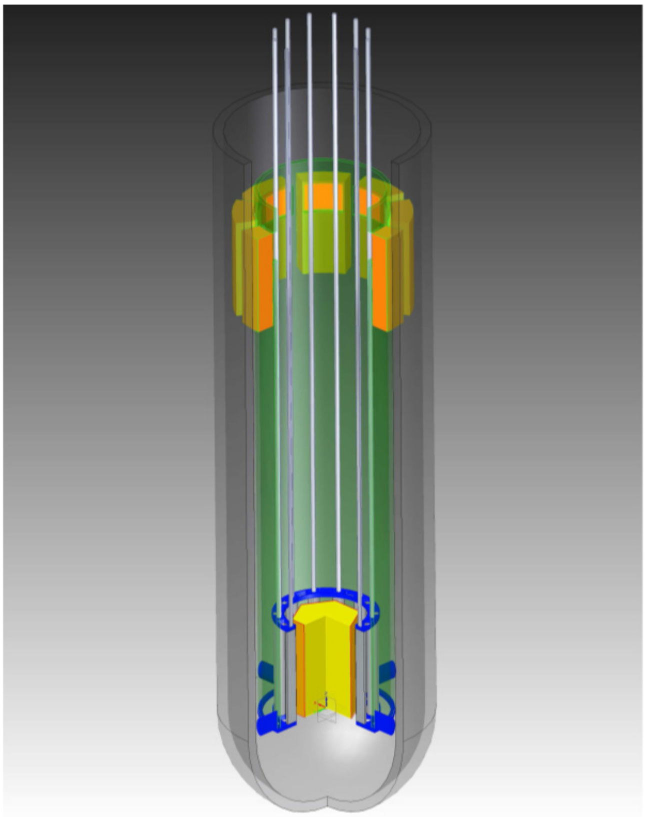


FIG. 25. A model of the primary system of ELECTRA

Full name:	*European Lead Cooled Training Reactor*
Designer:	*Royal Institute of Technology (KTH)*
Reactor type:	*Lead-cooled training Reactor*
Electrical capacity:	*Not foreseen*
Thermal capacity:	*0.5 MW(th)*
Coolant	*Lead*
Primary Circulation	*Natural*
System Pressure:	*< 0.1 MPa (primary), TBD (secondary)*
System Temperature:	*400÷500 °C*
Fuel Material:	*inert matrix (Pu,Zr)N*
Fuel Cycle:	*360 Months (with average availability of 50 %)*
No. of safety trains:	*TBD*
Emergency safety systems:	*Hybrid*
Residual heat removal systems:	*Passive*
Design Life:	*30 Years (with average availability of 50 %)*
Design status:	*Concept*
Planned deployment/1st date of completion:	*Construction is planned to start in 2020 and completion in 2023*
New/Distinguishing Features:	*The use of inert matrix nitride fuel*

Introduction

The European Lead Cooled Training Reactor (ELECTRA) is a low power fast reactor intended for research, education, training and technology demonstration purposes. The reactor concept development is promoted by the Royal Institute of Technology, Uppsala University and Chalmers University of Technology (Sweden) since 2009, and leverages on the R&D performed over the last 15 years on nitride fuels, liquid lead cooling and advanced cladding materials.

Description of the Nuclear System

A very small core size (30×30 cm) is achieved thanks to the use of inert matrix (Pu,Zr)N fuel. The ensuing pressure drop is low enough to permit full heat removal by natural convection of the coolant during normal operation, with a steam generator (SG) elevation of approximately 2 m and a consequent vessel height of approximately 3 m. The high reactivity loss resulting from the use of fertile-free fuel is compensated by the rotation of 12 cylindrical drums with B_4C sections located outside the core.

The fuel for ELECTRA may derive from two different sources. The primary choice is to construct a small recycle facility in direct conjunction to the Swedish intermediate repository CLAB in Oskarshamn. With a capacity of 4 tons spent LWR fuel per year, such a facility could separate the required inventory of plutonium within two years. The second potential source for the fuel consists in 900 kg of separated PuO_2 owned by the Swedish utility OKG, which is presently residing in Sellafield. This material was supposed to be used for producing MOX to be burnt in the Oskarshamn power plant.

Concerning the cladding, a material with a high margin to failure is desired since ELECTRA is intended to be a facility with a special focus on research on fast reactor dynamics,. In addition, the estimated end-of-life dose to the cladding is less than 40 dpa. Therefore, the austenitic 15-15Ti steel is selected as the reference cladding material.

The choice of natural convection for heat removal in ELECTRA avoids the need for developing and qualifying materials for pump impellers. However, flow stability issues may arise. Consequently, the R&D programme for ELECTRA foresees the construction and operation of an electrically heated one-to-one scale mock-up of the reactor.

Description of the Safety Concept

In Sweden, the Swedish Radiation Safety Authority (SSM) is the regulatory body monitoring all activities related to radiation protection and nuclear safety. For ELECTRA, prescription SSMFS 2008:17 (i.e., the Swedish radiation safety authority's regulation concerning the design and construction of nuclear power reactors) is of particular importance. No significant radiotoxic releases (> 1 mSv/year for the general public) are tolerated; hence, a list of anticipated and improbable events has to be dictated, and measures to handle them must be identified. Events of particular importance are those that could threat the integrity of the fuel and cladding, such as:

- change of core geometry during normal operation leading to reactivity and coolability variations;
- Loss Of Coolant Accident (LOCA) or degraded coolability;
- Transient Over Power (TOP) leading to degradation of fuel and cladding tubes;
- loss of active residual heat removal.

Deployment Status and Planned Schedule

Preliminary discussions on the licensing procedure for ELECTRA have been conducted with SSM. The current legal framework is deemed to include sufficient provisions to carry out the licensing process, which includes submitting an environmental impact statement to the Environmental Court and a preliminary safety assessment report (PSAR) to SSM. According to the tentative time-plan for R&D, licensing, construction and operation of ELECTRA and its associated fuel cycle facilities, design and safety analyses will last till 2018, while licensing will take two more years, up to 2020. Pending availability of funding, construction may start in 2020, after which first criticality could be achieved in 2023.

G4M (Gen4 Energy Inc., USA)

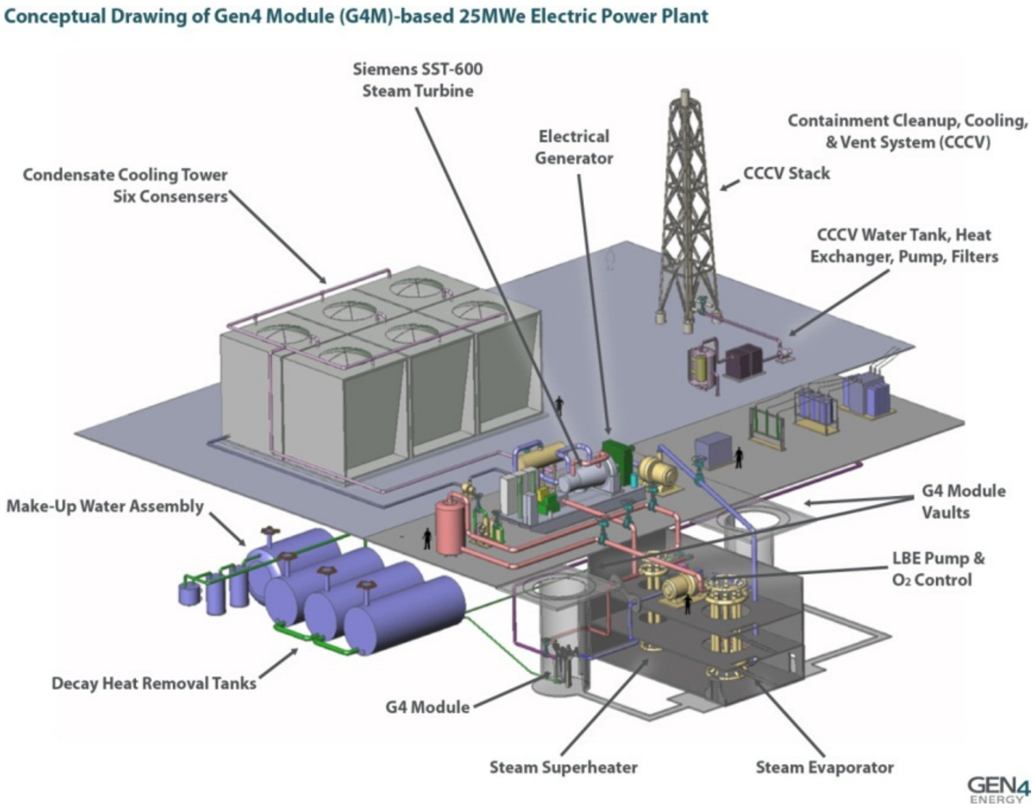


FIG. 26. Depiction of the G4M plant

Reactor type:	*Liquid metal cooled reactor*
Electrical capacity:	*25 MW(e)*
Thermal capacity:	*70 MW(th)*
Coolant:	*Lead–bismuth*
Primary circulation:	*Forced circulation*
System pressure:	*N/A*
Core outlet temperature:	*500°C*
Thermodynamic cycle:	*Indirect Rankine cycle*
Fuel material:	*Uranium nitride*
Fuel enrichment:	*19.75%*
Fuel cycle:	*10 years*
Reactivity control:	*Rod insertion and B_4C ball insertion*
No. of safety trains:	*2*
Emergency safety systems:	*N/A*
Residual heat removal systems:	*Passive*
Design life:	*5–15 (nominal 10) years*
Design status:	*Conceptual design*
Distinguishing features:	*Transportable factory fuelled design*

Introduction

Founded in 2007 as Hyperion Power Generation Inc., Gen4 Energy was formed to develop the Gen4 Module (G4M), first conceived at the Los Alamos National Laboratory (LANL) in New Mexico. Through the commercialization programme at LANL's Technology Transfer Division, Hyperion Power Generation was awarded the exclusive licence to utilize the intellectual property and develop the module.

Description of the nuclear systems

The reactor has been designed to deliver 70 MW of power over a ten year lifetime without refuelling. The materials in the core are uranium nitride fuel, HT-9 as the structural material, lead–bismuth eutectic as the coolant, quartz as the radial reflector, and B_4C rods and pellets for in-core reactivity control. The reactor is approximately 1.5 m in diameter and 2.5 m in height, in which there are 24 subassemblies containing the fuel pins. The pin assembly is filled with liquid LBE to provide a high conductivity thermal bond between the fuel and cladding. The gap in the fuel pins has been sized to preclude fuel clad mechanical interference throughout the core's lifetime. A plenum is located at one end, which serves as both a fission gas plenum and a repository for the LBE inside the pin as the fuel swells with burn-up.

The core coolant is LBE, with a mixed mean exit temperature of 500°C. This temperature limits the cladding temperature, so that maximum cladding creep over the 10 year lifetime of the reactor is less than 1%. During operational shutdown, decay heat is removed from the G4M by two methods. The first method transfers heat from the core by natural circulation of coolants in the primary and secondary loops to the steam generators. The second removes heat by passive vaporization of water from the surface of the secondary containment vessel.

Description of the safety concept

There are two independent, safety grade reactivity control systems in the core: a control rod system comprising 18 B_4C control rods and a reserve shutdown system consisting of a central cavity into which B_4C spheres may be inserted into the core. Both the control rods and the spheres are inserted into dry wells in the core, which are hexagonally shaped thimbles. These thimbles penetrate the reactor vessel and are sealed from the primary coolant. Both systems can independently take the core to long term cold shutdown.

The safety concept of the HPM is driven by a set of design criteria that the designers believe are sufficient to ensure protection of the facility and its surroundings. These criteria are a sealed core, operational simplicity, minimal to no in-core movement, mechanical components and separation of power production and conversion operations.

Deployment status

Gen4 Energy announced in April 2012 that they would not be pursuing the US Department of Energy's small modular reactor licensing support programme because they concluded that "use of well-known Light Water Reactor (LWR) technology of 45 to 300 MW intended for deployment in the USA had a much higher probability of success given the [Funding Opportunity Announcement's] stated maximum of two awards" [11].

GAS-COOLED FAST REACTOR DESIGNS

ALLEGRO (European Atomic Energy Community, Europe)

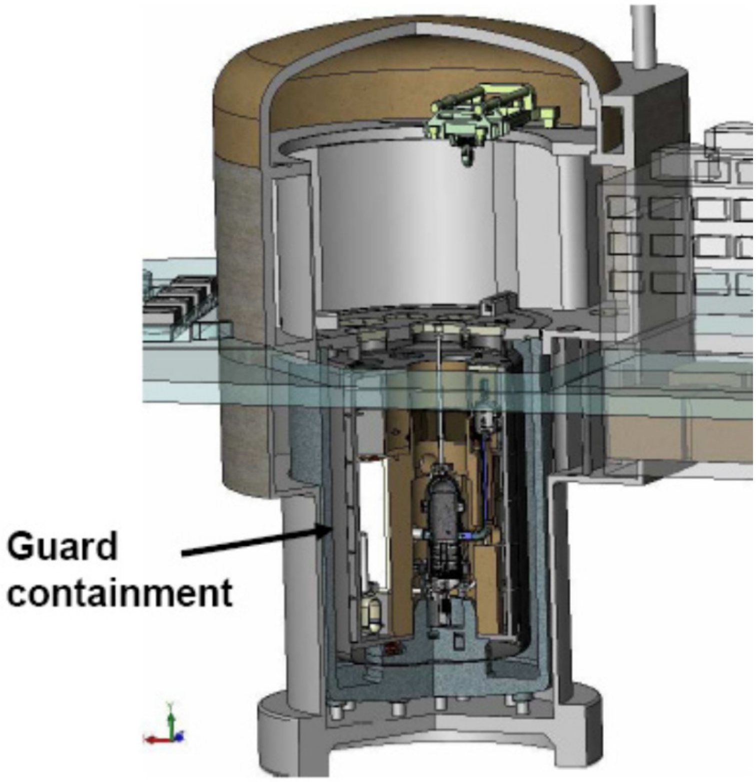


FIG. 27. ALLEGRO reactor building

Full name:	*ALLEGRO*
Designer:	*EURATOM*
Reactor type:	*Gas-cooled Fast Reactor*
Thermal capacity:	*75 MW(th)*
Coolant	*Helium*
Primary Circulation	*Forced/Natural*
System Temperature:	*530 °C*
Fuel Material:	*MOX*
Emergency safety systems:	*Hybrid active and passive*
Residual heat removal systems:	*Hybrid active and passive*
Design status:	*Pre-conceptual Design*
Planned deployment/1st date of completion:	*N/A*
New/Distinguishing Features:	*First step to an electricity generating prototype*

Introduction

On the Gas Fast Reactor (GFR) development roadmap, the Experimental and Technology Demonstration Reactor (ALLEGRO) is the first necessary step towards an electricity generating prototype. It is a low power (~ 75 MW(th)) helium-cooled fast reactor including the main objectives qualification of GFR fuel and sub-assembly concept, demonstration of core operation and control with the appropriate instrumentation, establishment of a first GFR safety reference framework and acquisition of first of a kind GFR operating feedback experience.

In the GFR fuel development plan, ALLEGRO is located in between the irradiation of samples in material testing reactors and the full demonstration at a GFR electricity generating prototype scale. Launched in 2005 for 4 years, the Gas Cooled Fast Reactor Specifically Targeted Project of the EURATOM 6th framework program includes pre-conceptual ETDR (previous ALLEGRO name) design and safety studies. The CEA is mainly in charge of core physics, reactor system design and global plant layout. Nexia Solutions is in charge of sub-assembly design, whereas AMEC is in charge of the absorbers and control and instrumentation and NRG of reflector and shielding studies. The GoFastR program, which started in 2010 and will last for three years, includes new partners for ALLEGRO, mainly from central Europe, Czech Republic (RC-Rez) and Hungary (AEKI and University of Budapest).

Description of the Nuclear System

The core design includes a two-step approach: First core (start-up core) using conventional MOX fuel and steel cladding with some experimental GFR fuel sub-assemblies and second core (refractory core) using only GFR reference fuel (carbide fuel with ceramic cladding). The start-up core is composed of 81 MOX fuel S/As with 6 control/shutdown rods (CSD) and 4 diverse shutdown (DSD) rods. After one year of operation, the core will be loaded with 350 experimental GFR S/As. The objective is to irradiate these S/As for about 2000 equivalent full power days with acceptable representative GFR conditions.

The preliminary system design is based on intermediate loops with two primary helium loops able to remove 75 MW without energy conversion. It includes two intermediate heat exchangers compatible with the helium inlet/outlet temperatures of both the start-up and demonstration cores (respectively 260/530°C compatible with the MOX fuel option and 400/850°C). The secondary loops include pressurized water, which avoids the issue of high temperature materials which would have been raised with a gas-gas heat exchanger option. The final heat sink is the atmosphere. An additional circuit reservation has been made to possibly test high temperature processes or components using part of the reactor power (10 MW).

Description of the Safety Concept

Other characteristics are the safety neutronic coefficients of the MOX core. It is noticeable that the void effect remains quite low whereas the Doppler Effect always tends to stabilize the power in case of fuel temperature increase.

Deployment Status and Planned Schedule

In parallel to the pre-conceptual design process, the ALLEGRO R&D plan has been launched to qualify the appropriate calculation tools and the specific helium technology necessary in addition to the VHTR mainstream development. This includes in particular a core physics test program; air and helium tests on reduced size sub-assemblies; system transient analysis codes benchmarking and qualification; high temperature resistant thermal barriers (1250° C for one hour); DHR blowers with quite large specifications (from 3 to 70 bar with constant mass flow rate); specific instrumentation for core thermal monitoring and fuel handling; and specific GFR environment purification systems. The R&D plan is now broadened at the European level through the 7th Framework Program ADRIANA project which aims at identifying the needs and research infrastructures for SFR, GFR and LFR systems.

EM^2 (General Atomics, USA)

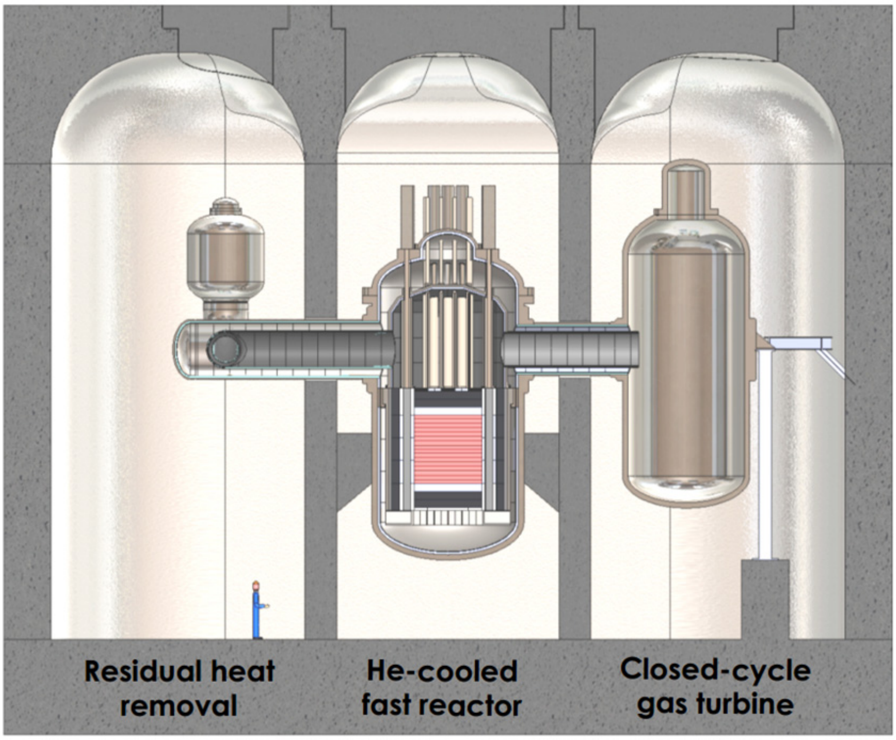


FIG. 28. Schematic View of EM^2

Reactor type:	*High temperature gas cooled fast reactor*
Electrical capacity:	*240 MW(e)*
Thermal capacity:	*500 MW(th)*
Coolant:	*Helium*
Primary Circulation:	*Forced circulation*
System Pressure:	*N/A*
Core Outlet Temperature:	*850°C*
Thermodynamic Cycle:	*Direct Brayton cycle*
Fuel Material:	*Used nuclear fuel*
Fuel Enrichment:	*1% U^{235}, 1% Pu, MA*
Fuel Cycle:	*30 years*
Reactivity Control:	*N/A*
No. of safety trains:	*N/A*
Emergency safety systems:	*N/A*
RHRS:	*Passive*
Design Life:	*30 years*
Design Status:	*Conceptual design*
Seismic Design:	*N/A*
Predicted CDF:	*N/A*
Planned deployment:	*N/A*
Distinguishing Features:	*Helium cooled fast reactor; reduces spent fuel inventories*

Introduction

The EM^2 is designed as a modification of an earlier high temperature, helium cooled reactor. It is an effort to utilize used nuclear fuel without conventional reprocessing.

Description of the Nuclear Systems

The reactor is designed to produce 500 MW(th) and 240 MW(e) based on a closed cycle gas turbine. The EM^2 is a fast reactor design intended to produce electricity, and to burn used nuclear fuel and has a 30 year core without the need for refuelling or reshuffling. The spent fuel cladding is first removed and the fuel is pulverized and processed using the atomics international reduction oxidation (AIROX) dry process to remove fission products. The fuel burned in the reactor is recycled after discharge, processing and refabrication.

The core contains SiC-SiC clad porous UC plates arranged in a SiC-SiC assembly frame making a fuel assembly. There are 21 fuel assemblies creating one layer and 17 layers stacked on top of each other surrounded by first a BeO layer, then a graphite reflector layer, and a B_4C layer all sitting in the core barrel.

In a first generation plant, the fuel consists of

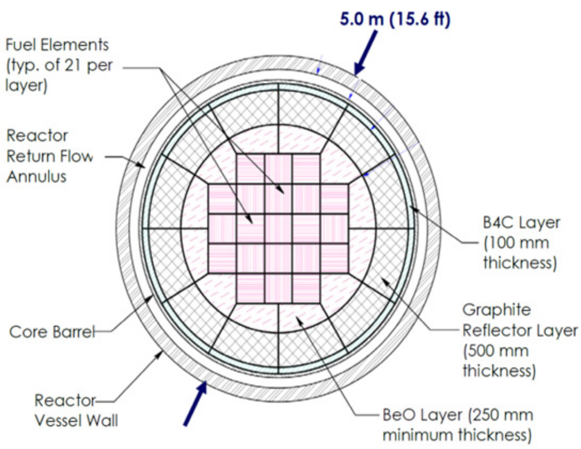


FIG. 8. Core layout

about 22.2 t of LEU starter and about 20.4 t of used nuclear fuel. The used nuclear fuel is roughly 1% ^{235}U, 1% Pu and mixed actinides (MA), and 3% fission products; the rest is ^{238}U. The design organization claims that there is no need for uranium enrichment after the first generation reactor, as the discharge from the preceding generation is used for the

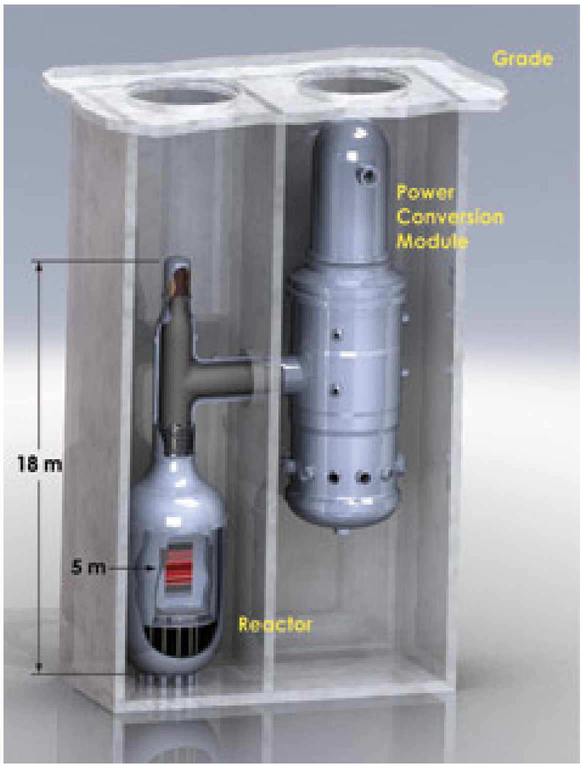


FIG. 30. EM2 reactor and gas turbine

succeeding generation. Out of each discharge, about 38.5 t is used in the succeeding generation while about 4 t of fission products is removed.

Description of Turbine-Generator Systems

Using a gas turbine cycle, the designers claim to achieve 48% efficiency with a core outlet temperature of 850°C. The entire containment is designed to be below grade and sealed for the 30 year core period [11].

MOLTEN-SALT FAST REACTOR DESIGNS

MSFR (CNRS, France)

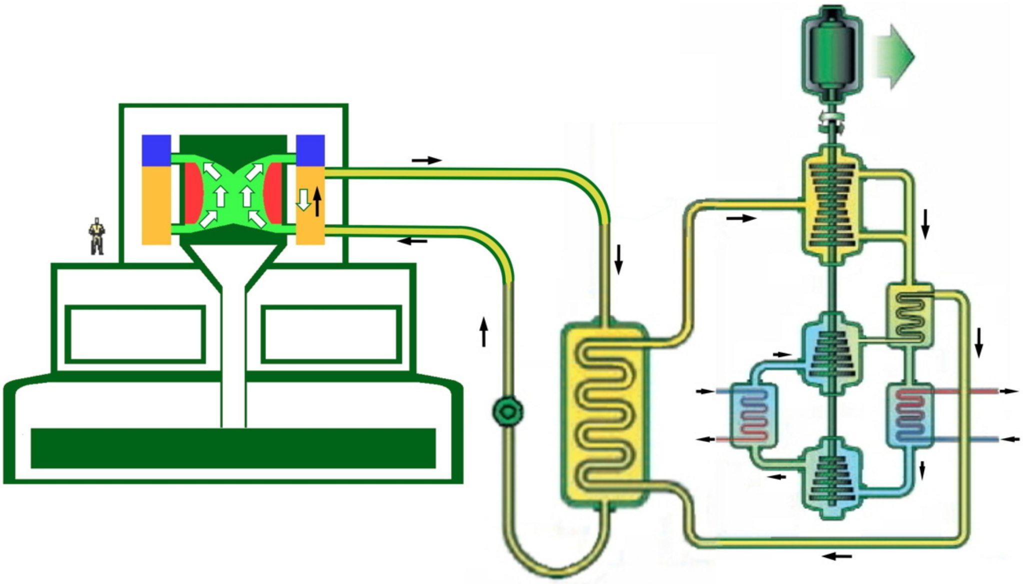


FIG. 31. Depiction of the MSFR system

Full name:	*Molten Salt Fast Reactors (MSFR)*
Designer:	*CNRS (France)*
Reactor type:	*Molten salt reactor*
Electrical capacity:	*1,500 MW(e)*
Thermal capacity:	*3,000 MW(th)*
Coolant	LiF-$(U,Pu)F_3$-ThF_4 (lithium and actinide fluorides)
Primary Circulation	Forced
System Pressure:	*< 10 bar*
System Temperature:	*≅750 °C*
Fuel Material:	*Actinide fluorides containing Th and ^{233}U, ^{235}U and/or Pu*
Fuel Cycle:	*Continuous refuelling*
No. of safety trains:	*No defined yet*
Emergency safety systems:	*Hybrid*
Residual heat removal systems:	*Hybrid*
Design Life:	*>60 Years*
Design status:	*Concept*
Planned deployment/1st date of completion:	*After 2050*
New/Distinguishing Features:	*Fast spectrum molten salt reactor with Thorium fuel cycle. The MSFR combines the generic assets of fast neutron reactors (extended resource utilization, waste minimization) with those associated to a liquid-fuelled reactor (online fuelling and reprocessing, uniform fuel irradiation, passive salt draining system).*

Introduction

Since 2004, the National Centre for Scientific Research (CNRS, France) has focused R&D efforts on the development of a new MSR concept called the Molten Salt Fast Reactor (MSFR) currently supported by the Euratom's EVOL project. As opposed to thermal molten salt reactors, the MSFR does not employ any solid moderator (no graphite lifespan issues) which results in a fast-spectrum breeder reactor with a large negative power coefficient that can be operated in a Thorium fuel cycle. Other advantages of a MSR include homogeneous fuel irradiation and the possibility of fuel reload and processing on-line or in batch mode, without requiring reactor shut-down and involving the transfer of small volumes of fuel. GIF forum selected the MSFR concept in 2008 as one of the GEN IV reference reactors.

Description of the Nuclear System

The reference MSFR is a 3,000 MW(th) reactor with a total fuel salt volume in primary circuit of 18 m^3, operated at a max fuel salt temperature of 750°C. The reactor has three different circuits: the fuel circuit, the intermediate circuit and the power conversion system. The main components of the fuel circuit are the fuel salt which serves as fuel and coolant, the core cavity, the inlet and outlet pipes, the gas injection system, the salt-bubble separators, the fuel heat exchangers and the pumps. The fuel salt is a molten binary fluoride salt with 77.5% of lithium fluoride; the other 22.5% are a mix of heavy nuclei fluorides. As shown in the figure below, the fuel salt flows from bottom to the top of the core cavity. After exiting the core, the fuel salt is fed into 16 groups of pumps and heat exchangers located around the core. Three main components of the core are: the upper and lower axial neutron reflectors and the radial fertile blankets (red in the figure). The reactor blanket (whose purpose is to increase the breeding ratio) is filled with a fertile salt of LiF-ThF4 with an initial composition of 22.5% mole $^{232}ThF_4$. Thanks to the MSFR fast spectrum, the fuel reprocessing unit only extracts a small amount of the fuel salt (order of a few litres per day) for fission product removal and then returned the cleaned fuel salt to the reactor.

Description of the Safety Concept

The MSFR fuel circuit includes a passive salt draining system (by gravity) which can be used for a planned reactor shut down or in case of incidents/accidents leading to an excessive increase of the temperature in the core. Thanks to the online fuelling and reprocessing, the MSFR has a low core reactivity inventory. Moreover, control rods or neutron poisons are not necessary due to the large negative feedback coefficients which allow a reactivity control based on the balance between the power generated in the fuel salt and the power extracted in the heat exchangers. Lastly, adequate reactivity margin during cold shutdown is obtained in the MSFR by draining the fuel salt into the dedicated tanks where the fuel can be passively cooled without returning to criticality (thanks to the tanks volume and geometry). The absence of absorbent rods simplifies the reactor operation and eliminates some accident initiators (e.g. a control rod ejection).

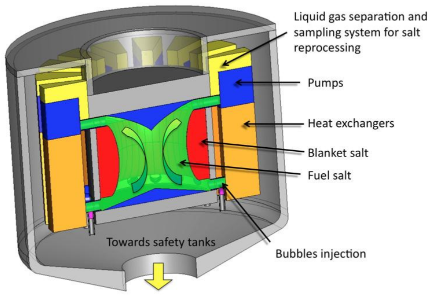


FIG. 32. Schematic MSFR design showing the fluoride-based fuel salt (green) and the fertile blanket salt (red).

Deployment Status and Planned Schedule

The analysis of possible French energy supply scenarios shows that the deployment of the MSFR after 2050 would allow efficient closure of the current fuel cycle (reduce the stockpiles of produced transuranic elements).

ACRONYMS

ADS	Accelerator Driven System
ATWS	Anticipated Transients Without Scram
CBR	Core Breeding Ratio
CDA	Core Disruptive Accident
CPS	Core Protection System
CR	Control Rods
DHR	Decay Heat Removal
DRACS	Direct Reactor Auxiliary Cooling System
EFPD	Effective Full Power Day
EMP	Electromagnetic Pumps
FA	Fuel Assembly
FR	Fast Reactor
GFR	Gas-cooled Fast Reactor
HLMC	Heavy Liquid Metal Coolant
IC	Isolation Condenser
IHTS	Intermediate Heat Transport System
IHX	Intermediate Heat Exchanger
ISIR	In Service Inspection and Repair
IV	Inner Vessel
LBE	Lead-Bismuth Eutectic
LFR	Lead-cooled Fast Reactor
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MCP	Main Circulation Pump

MOX	Mixed OXide
MSFR	Molten Salt Fast Reactor
MSR	Molten Salt Reactor
PHTS	Primary Heat Transport System
PHX	Primary Heat Exchangers
PP	Primary Pump
PRACS	Pool Reactor Auxiliary Cooling System
PSAR	Preliminary Safety Assessment Report
RHRS	Residual Heat Removal Systems
RSS	Reactor Shutdown System
RV	Reactor Vessel
RVACS	Reactor Vessel Air Cooling System
SBO	Station Black Out
SASS	Self-Actuated Shutdown System
SFR	Sodium-cooled Fast Reactor
SG	Steam Generator
SGS	Steam Generators System
SR	Safety Rod
SV	Safety Vessel
TOP	Transient Over Power
VHTR	Very High Temperature Reactor

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