Advanced Large Water Cooled Reactors

A Supplement to: IAEA Advanced Reactors Information System (ARIS) 2020 Edition

IAEA
International Atomic Energy Agency

ADVANCED LARGE
WATER COOLED REACTORS

2020 Edition

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IAEA Advanced Reactors Information System (ARIS)

PREFACE

Nuclear power, as a clean, reliable and affordable energy source plays an important role in the national and global energy mix. Nuclear power can make a vital contribution to meeting climate change targets while delivering the increasingly large quantities of electricity needed for global economic development.

At a high-level UN side event at the United Nations Climate Change Conference in Madrid in December 2019, IAEA Director General Rafael Mariano Grossi stated that:

“Nuclear power provides around one-third of the world’s low-carbon electricity and already plays a significant role in mitigating climate change. Many of our 171 Member States believe that it will be very difficult, if not impossible, to achieve sustainable development and meet global climate goals without significant use of nuclear energy.”

Member States considering or planning to build their first nuclear power plant (NPP) and those with notions to expand their existing nuclear power programmes are interested in having ready access to the most up-to-date information about all available nuclear power reactor designs that are deployable now or in the near future, as well as important development trends. Fulfilling its mission in the IAEA Statute Article III.A.3: “To foster the exchange of scientific and technical information on peaceful uses of atomic energy”, the IAEA has regularly issued publications on the status of technology developments in nuclear power reactors.

The IAEA’s Advanced Reactor Information System (ARIS) – developed to meet this need – is an online database that provides Members States with balanced, comprehensive and up-to-date information on nuclear power reactors of all sizes and types, from advanced evolutionary NPP designs for near term deployment to innovative concepts under early development. Supplements to the ARIS database are published periodically, including recent issues on:

- Advances in Small Modular Reactor (SMR) Technology Developments, 2020;
- Advanced Large Water Cooled Reactors, 2015; and
- Status of Innovative Fast Reactors Designs and Concepts, 2013.

Water Cooled Reactors (WCR) have played a significant role in the commercial nuclear industry since its beginning and currently account for more than 95% of all operating civilian power reactors in the world. In addition, most nuclear power reactors under construction, or planned, are water cooled.

Evolutionary light and heavy water reactor designs are available today. Light water reactors (LWRs) are the most prevalent among the operating NPPs throughout the world, both in numbers and in total generated power. Current LWR technologies have proven to be economical, safe and reliable, and they have a mature infrastructure. Heavy water reactors (HWRs) represent a significant proportion of the world’s reactor installations. They provide improved fuel cycle flexibility and can, for example, burn the spent fuel from LWRs with no major reactor design changes.

This Booklet offers an overview of the status of advanced, large WCRs with a capacity of 700 MWe or more. It includes descriptions of specific large NPPs available for immediate and near-term deployment. It is intended as a complementary publication to the ARIS database and the IAEA guidance document for evaluating the technology for NPPs, Nuclear Reactor Technology Assessment for Near Term Deployment (IAEA Nuclear Energy Series No. NP-T-1.10).

The IAEA officers responsible for this publication were T. Jevremovic and M. Krause of the Division of Nuclear Power.

DISCLAIMER

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LIST OF REACTORS

GUIDE

Each Large, Advance Water Cooled Reactor under construction or deployable in the near future described in a two-page fold:

PWR **BWR** **HWR**

Table with important reactor data

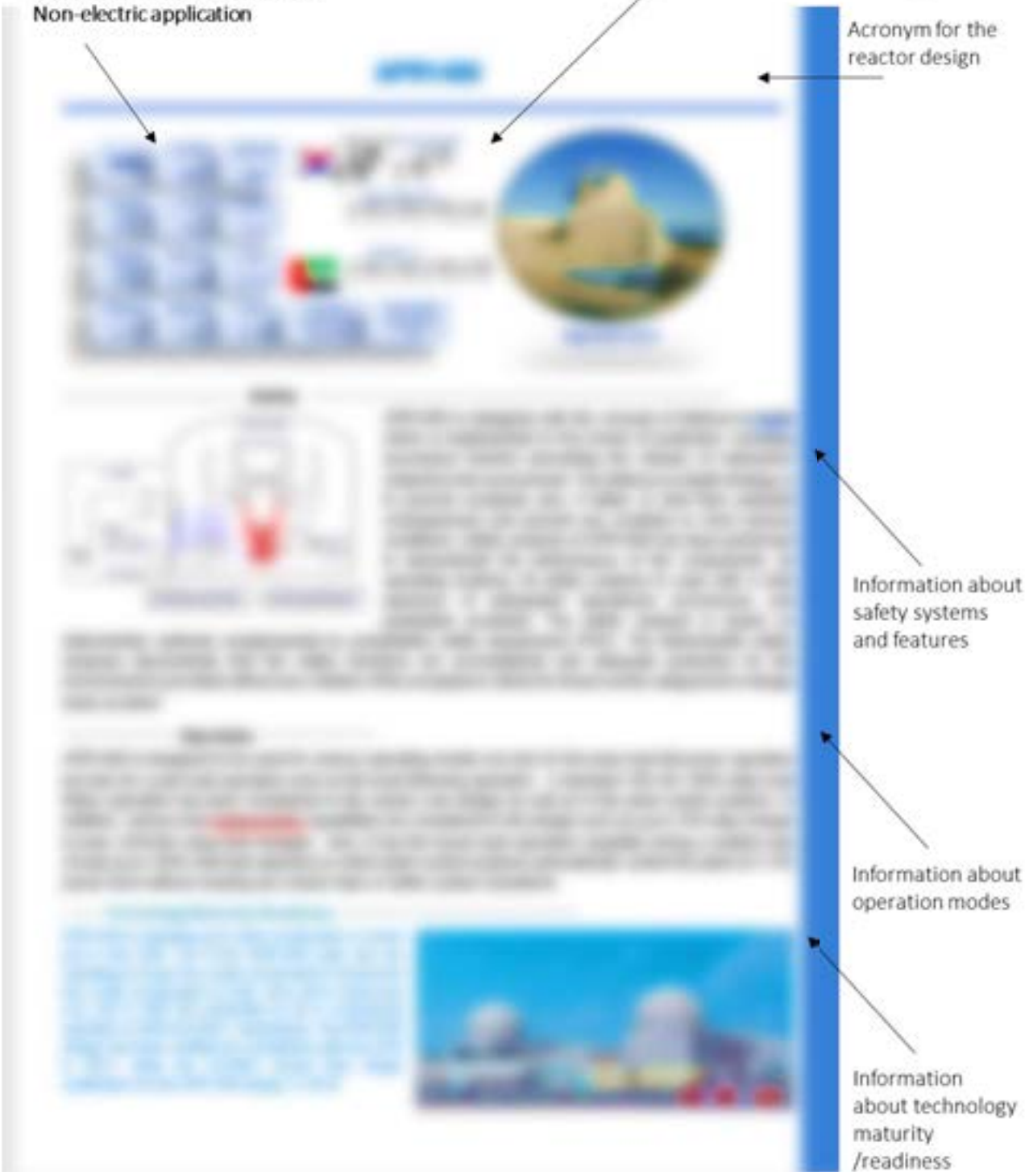
Capacity: gross electric output (MWe) and thermal output (MWth)
Efficiency: plant net efficiency
Coolant: reactor coolant type
Fuel: fuel type
Fuel cycle: refuelling interval in months
Design life: life of non-replaceable components
SSE: seismic design
CDF: core damage frequency
LERF: large early release frequency
Operation: modes of operation
Non-electric application

Information about operation and construction / commissioning status

In operation

Under construction / commissioning

Acronym for the reactor design



Information about safety systems and features

Information about operation modes

Information about technology maturity /readiness

Advanced Passive 1000

The Westinghouse Advanced Passive PWR AP1000 is an 1100 MWe class PWR based closely on the AP600 design. The AP1000 maintains most of the AP600 design configuration, and uses the AP600 components, proven in earlier Westinghouse PWRs or in special test programs, and licensing basis by limiting the changes to the AP600 design to as few as possible. The AP1000 design includes advanced passive safety systems and extensive plant simplifications to enhance the safety, construction, operation, and maintenance of the plant. The plant design utilizes proven technology, which builds on approximately 50 years of operating PWR experience, but with an emphasis on safety features that rely on natural forces. The safety systems use natural driving forces such as pressurized gas, gravity flow, natural circulation flow, and convection. The AP1000 is designed to meet U.S. NRC deterministic safety criteria and probabilistic risk criteria with large margins. The AP1000 design philosophy focuses on extensive plant simplification, smaller plant footprint, and modular construction techniques, all of which lower the complexity of safety systems, minimize construction time and capital costs, while maintaining the high standards required by the industry and regulators. Plant simplification also results in simpler O&M procedures due to fewer components and less building material than previous Westinghouse PWR designs. This reactor model builds on 50+ years of successful construction and operational experience by the US based Westinghouse Electric Company LLC, which was bought by the Japanese company Toshiba in 2005.

Nuclear Systems

The AP100 is a 1,100 MWe class conventional PWR 2-loop design. Each loop consists of a single hot leg, two cold legs, a SG and two reactor coolant pumps that are directly connected to the bottom of each SG. The elimination of a crossover leg between SG and reactor coolant pump reduces pressure drops along the loop and increases the overall efficiency of the plant. A relatively large pressurizer, which controls the pressure and temperature of the primary circuit, is connected to one of the hot leg pipes. The pressurizer helps to increase the operating margins and reduce unwanted reactor trips. Various components inside the RPV such as core shroud and core barrel help to attenuate neutron and gamma fluxes, contributing to longer in-core component design lifetimes.

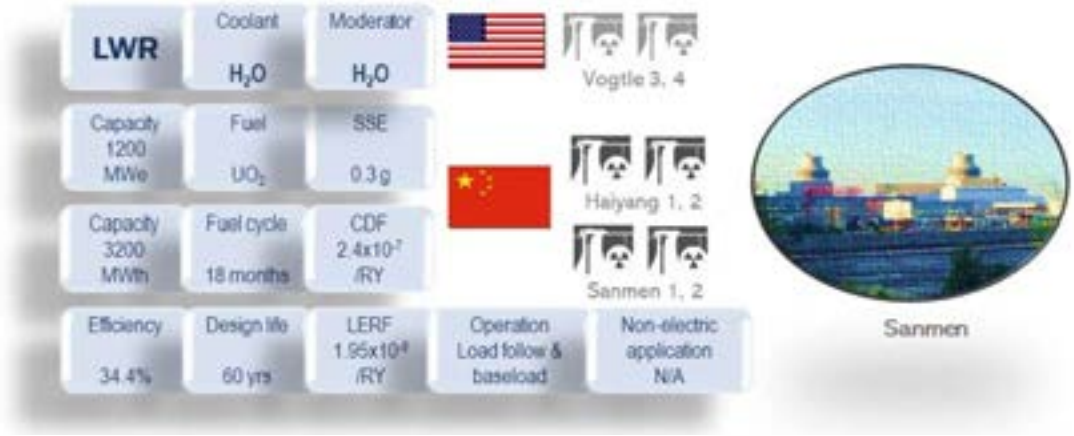
Fuel

The AP1000 is primarily designed to use enriched UO_2 fuel. Following optimal conditions, a refuelling outage can be kept to 17 days. A standard reactor core has fuel rods of varying ^{235}U enrichments 2.3–4.8 %. Integral fuel burnable absorbers contain thin boride surface coating can be used to handle excessive reactivity of fresh fuel and accommodate special utility needs. A fuel assembly is predicted to have an average burnup of 60 GWd/t following good fuel management strategy. Load following capabilities are provided using grey rods rather than changing soluble boron concentration, since this is considered more economical and efficient, and reduces waste products. The standard fuel assemblies are of the Westinghouse ROBUSTTM design, which have a few previously proven design features and have undergone extensive testing in dedicated facilities. The fuel is contained in ZIRLOTM tubing with large gas space to accommodate increased fission gas production for higher burnups.

Special Design Feature, e.g. Construction Methods

The modular construction method is applied to the containment liner plate and stainless steel liner plate; this method is also applied to the fabrication of equipment such as the reactor internals and the condenser. The deck plate construction method is applied for the construction and the installation of mechanical and electrical equipment. The modularization of reactor internals and the mechanical and structural composite modularization technologies are applied to the construction of Shin-Kori 3 & 4.

AP1000



Safety

A conservatively based design, decades of reactor operations and proven PWR technology are used in the development of the AP1000. Natural forces such as pressurized gas, gravity flow, natural circulation flow, and convection, are utilized to serve safety related functions and ensure that the reactor will safely shut down and remain cooled even with no operator action during a loss of all on-site and off-site AC power. Likewise, active components (pumps, fans or diesel generators) that operate through support systems (AC power, component cooling water, service water or HVAC systems) are excluded from the design, thus requiring no immediate operator action to mitigate postulated DBAs. Previously proven active safety/control systems are used to eliminate unnecessary actuation of the passive safety related systems. In the case of a severe accident, the in-vessel retention mechanism and 72 hour vessel cooling features reduce the risk of PV failure that could result in high energetic corium interactions. The placement of penetration above the reactor core reduces the likelihood of extended core uncovering in LOCA, which are additionally mitigated by the “leak-before-break principle” for primary piping. The AP1000 is designed in accordance with the U.S. NRC deterministic safety and probabilistic risk criteria with large margins, with a safe shutdown earthquake seismic design of 0.3 g, resulting in a low CDF of 10^{-7} /RY.

Operation

Operation and maintenance were important aspects that were considered during the design of the AP1000. A high degree of reliability with low maintenance requirement can result in a high availability factor of ~93% (planned and forced maintenance included). The increased operational margins help reduce unwanted reactors trips, thus predicting a stable and reliable plant operation without compromising on operational safety aspects. Reduced maintenance requirements result in on overall reduction of the operational costs for the entire life of the nuclear plant, lowering the forecasted electricity generation costs. A reduced number of components with dedicated access platforms and lifting devices at key locations facilitate shorter, safer and more reliable maintenance/repair work for radiation workers. The standardization of components across plants with built-in-testing apparatus further reduces the lengthy training required for workers as well as lengthy instrument positioning/calibration for periodic testing purposes. ALARA principles have been enforced to keep worker doses at low levels.

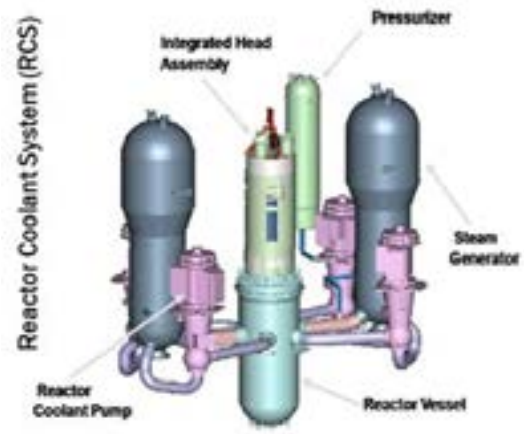
Technology Maturity/Readiness

- Toshiba’s innovative, modular, and parallel construction scheduling to minimize costs and lead-times;
- Designed to accommodate physical protection measures against wide range of natural and security threats;
- Systems and components are improved versions found in currently operating Westinghouse plants;
- In-vessel retention features of corium for DBA;
- MOX fuel capabilities;
- 60 years design life with 93% lifetime availability;
- Reduced radiation exposure, less radioactive waste.

Advanced Power Reactor 1400

Korea Electric Power Company (KEPCO) designed the APR1400, an evolutionary reactor which incorporates a variety of engineering improvements to enhance safety, improve economics, and increase reliability of nuclear electricity generation in the Republic of Korea. This reactor is a 1,400 MWe PWR that utilizes innovative active as well as passive safety systems to provide high performance and safe reactor operating conditions. The design evolved from the OPR1000 with higher safety and seismic resistance features. The philosophy that safety and economics go hand-in-hand resulted in a worldwide deployable design that can be tailored to a variety of utility requirements. The company has drawn from its extensive experience in areas of construction, operations and decommissioning in the nuclear industry to incorporate lessons learned from their previous endeavours. A parallel research and design process allowed the incorporation of results from a series of experimental research projects with the purpose of protecting workers, the public and the environment.

Nuclear Systems

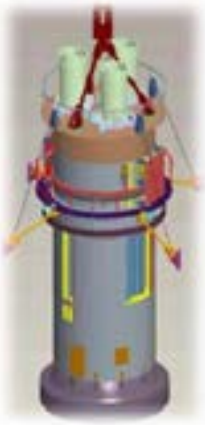


The APR1400 is a PWR with two coolant loops. The RCS circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to a secondary system. The RPV, SGs, reactor coolant pumps, pressurizer, and associated piping are the major components of the RCS. Two parallel heat transfer loops, each containing one SG and two reactor coolant pumps, are connected to the RPV. One pressurizer is connected to one of the RPV hot legs. All RCS components are located inside the reactor containment building.

Fuel

The reactor core is designed to generate 3,983 MWth with an average volumetric power density of 100 W/cm^3. The reactor core consists of 241 fuel assemblies made of fuel rods containing UO_2 fuel; the operating cycle is above 18 months and discharge burnup is ~60 GWd/t; thermal margin is more than 10% to enhance safety and operational performance. The HIPER16TM fuel design has the capability of a batch average discharge burn-up higher than 55 GWd/t and the HIPER16TM design has increased overpower margin in comparison with the previous fuel design (PLUS7TM).

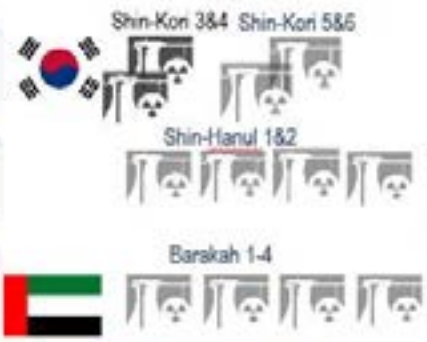
Special Design Feature, e.g. Construction Methods



The modular construction method is applied to the containment liner plate and stainless steel liner plate; this method is also applied to the fabrication of equipment such as the reactor internals and the condenser. The deck plate construction method is applied for the construction and the installation of mechanical and electrical equipment. The modularization of reactor internals and the mechanical and structural composite modularization technologies are applied to the construction of Shin-Kori 3 & 4. The integrated head assembly is a structure to combine and integrate all the RPV closure head area structures into one assembly to reduce the refuelling time for installation and removal of head area components. Also, the integrated head assembly contributes to the reduction of radiation exposures to the maintenance crew since the dissembling and assembling time of the RPV head is reduced.

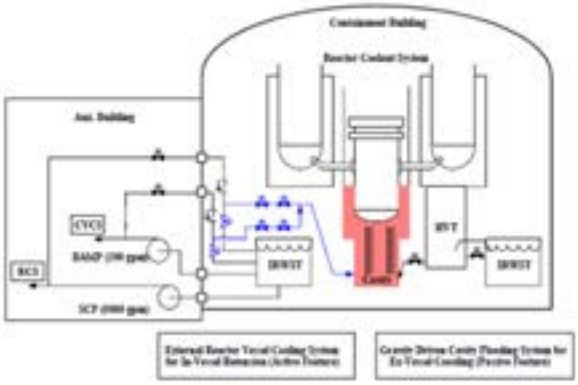
APR1400

LWR	Coolant	Moderator
	H_2O	H_2O
Capacity 1400 MWe	Fuel UO_2	SSE 0.3 g
Capacity 3983 MWth	Fuel cycle 18 months	CDF $<10^{-6}$ /RY
Efficiency 35.1%	Design life 60 yrs	LERF $<10^{-6}$ /RY
	Operation Load follow & baseload	Non-electric application N/A



Shin Kori 3 & 4

Safety



APR1400 is designed with defense in depth (DID) which is implemented to five levels of protection, including successive barriers preventing the release of radioactive material to the environment. The DID strategy is to prevent accidents and, if failed, to limit their potential consequences and prevent any evolution to more serious conditions. Safety analysis of APR1400 has been performed to demonstrate the performance of the components, its operating systems, its safety systems to cope with a wide spectrum of anticipated operational occurrences and postulated accidents. The safety analysis is based on deterministic methods complemented by probabilistic safety assessment. The deterministic safety analyses demonstrate that the safety functions are accomplished and adequate protection for the environment is provided without any violation of the acceptance criteria for DBA events.

Operation

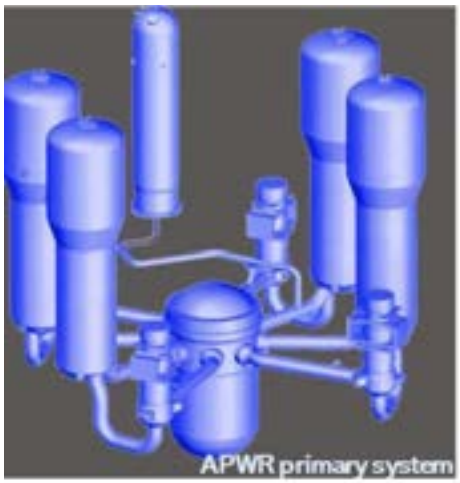
APR1400 is designed to be used for various operating modes not only for the base load full power operation but also for a part load operation such as the load following operation. A standard 100–50–100 % daily load follow operation has been considered in the reactor core design as well as in the plant control systems. In addition, various load maneuvering capabilities are considered in the design such as up to 10 % step change in load, ±5 % / min ramp load changes. Also, it has the house load operation capability during a sudden loss of load up to 100 % (full load rejection) in which plant control systems automatically control the plant at 3–5% power level without causing any reactor trips or safety system actuations.

Technology Maturity/Readiness

APR1400 is operating and under construction in Korea and in the UAE. Out of ten APR1400 units, two are operating in Korea, four under construction in Korea and four under construction in UAE. One unit in Korea and one unit in UAE are scheduled to be in commercial operation in 2020 and 2021, respectively. The APR1400 design has been certified for compliance with the EUR in 2017, while the U.S. NRC issued final design certification for the APR1400 design, in 2019.

Advanced PWR

Mitsubishi and five national electric companies involved in PWR electricity generation, came together in Japan during the 1990s to design and develop an advanced large PWR. Their main vision for the new reactor design was to make electiricity production safer, more reliable and economical. The companies combined their vast operational experience with the design and construction knowledge of Mitsubishi, involved in 24 PWR projects since the 1970s, to build a reliably operating reactor while minimizing the construction period and costs. The APWR was originally designed for domestic use in Japan, but has been diversified to a variety of country/regulatory specific versions: standard APWR, US-APWR, and EU-APWR, each uniquely envisioned by the designers to fit country specific regulatory and utility requirements. Generating capacities range from 1,500 MWe to 1,700 MWe.

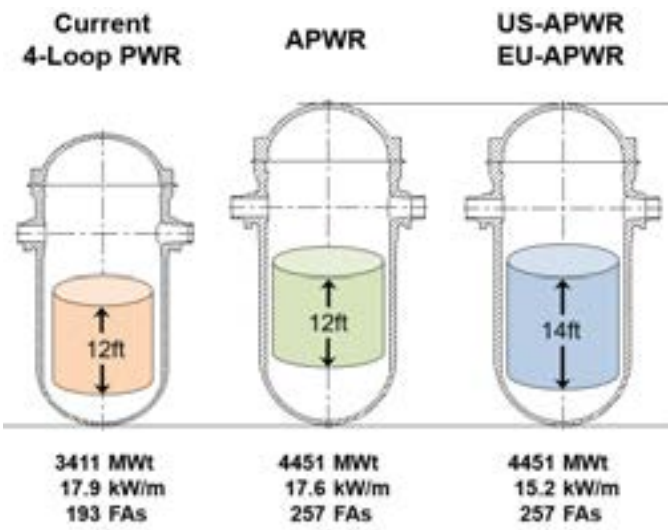


Nuclear Systems

The APWR nuclear systems are based on the existing fleet of PWRs in Japan but with increased core capacity and modernized digital safety related systems. The primary system is a traditional 4-loop PWR design with 4 SGs and 4 reactor coolant pumps. Each loop consists of one hot leg, one cold leg and one crossover leg. A single pressurizer controls the pressure of the primary circuit which has a core thermal capacity of 4,451 MWth. Neutron reflectors are placed around the core for neutron economy as well as reducing vessel fluence to ensure a design life of 60 years. Innovative SGs with increased heat transfer area, high performance moisture separators and alloy TT690 tubes for greater corrosion resistance are employed. All the major nuclear and safety systems are designed withstand 0.3–0.8g of seismic acceleration and internal hazards, such as fires, aircraft crash and flooding.

Fuel

The standard fuel design for the APWR is same as that of currently operating Japanese PWRs, which has demonstrated high reliability through significant irradiation experience in Japan. It consists of sintered UO_2 pellet with low ^{235}U enriched uranium (max 5 %). The EU/US-APWR version has a slighter longer fuel assembly to achieve lower power densities, and a 24 month fuel cycle to enhance fuel economy with larger safety margin. The anticipated burnup average is 55 GWd/t with a maximum of 62 GWd/t. It is also possible to use one third MOX cores without any major modifications to the nuclear systems.



Special Design Feature, e.g. Construction Methods

Reduction in construction cost is achieved through the effect of economies of scale by making the plant output larger by ~40 % compared with the existing Japanese 4-loop plant, and by simplifying engineered safety features, such as the safety system with four sub-systems, adopting advanced accumulator and in-containment RWSP as well as by reducing the amount of cables. Shortening of the construction schedule is being studied by adopting super heavy-duty cranes, increasing the number of large piping modules and components.

APWR

Safety

The APWR uses a mix of active, passive and inherently safe systems for normal operation and emergency situations. The three different versions of the APWR have different safety, diversity and redundancy features that are specific to regulatory or commercial requirements. All versions of the APWR come with a 4-train mechanical ECCS with at least two independent electrical trains. The function of the low pressure injection system and accumulator have been combined into the passive advanced accumulator to simplify the system configuration. The containment vessel air recirculation system, hydrogen ignitors, passive autocatalytic recombiners, filtered containment vents, and alternate containment vent sprays are designed to mitigate the effects of severe accidents or containment over pressurization. For ex-vessel accidents, flooding of reactor cavity is possible to aid corium cooling. Plain and borated water are stored in multipurpose water storage tanks for use during emergency situations.

Operation

The extensive operational experience of Japanese reactors has provided vital information on improvements need in evolutionary reactors with safety and high reliability as top priorities while not negating the economic factors. Through economies of scale the APWR is predicted to produce electricity for about a third less cost than current Japanese plants and has a predicted availability of over 90 %. Load-following operation is designed into the APWR, thus giving the plant the ability to operate in a range of 15%-100 % of full power when required to comply with home and foreign demands. The instrumentation and control systems of the plant are fully digital with vital information about the state of the plant displayed in an easily viewed manner to reduce interpretation and communication errors. Operators' workload and errors are thus reduced, potentially resulting in more reliable operation of the plant. Remote handling equipment in sensitive radiation areas are likely to reduce operational radiation in these controlled areas for radiation workers.

Technology Maturity/Readiness

		APWR	US-APWR	EU-APWR
Electric output		1,500~1,600 MWe	1,700 MWe	1,700 MWe
Fuel assemblies		257 x 3.7 m	257 x 4.2 m	257 x 4.2 m
SG heat transfer area		6,500 m^2	8,500 m^2	8,500 m^2
Safety systems	Electrical	2 / 4 trains	4 trains	4 trains
	Mechanical	4 trains	4 trains	4 trains
Emergency power		Diesel generator	Gas turbine generator	Gas turbine generator
Alternate AC power		Diesel generator	Gas turbine generator	Gas turbine generator

ATMEA1

ATMEA1 is an efficient and safe advanced mid-range PWR designed and developed jointly by EDF, Framatome and Mitsubishi Heavy Industries (MHI) with the goal to market and deploy worldwidely. Using innovative and proven nuclear technologies from the well-developed nuclear power sectors in France and Japan, ATMEA1 accords with compliance across a broad set of regulatory and commercial requirements worldwide. This design incorporates toplevel safety systems, high thermal efficiency and a flexible fuel cycle while simplification reduces projected capital and construction costs. Additionally a simplified effective design and standardizations of components forecast reductions in operational and maintenance costs over the 60 years of design life. Higher plant efficiencies, innovative fuel management strategies and effective surveillance programmes lead to less waste production and help to minimize the impact on the environment.

Nuclear Systems

Primary configuration of ATMEA1

The primary system of the ATMEA1 is mix between previous EDF, Framatome and MHI designs (Japanese PWRs, KONVOI and N4) and their new evolutionary 4-loop designs (APWR and EPR). The ATMEA1 uses a 3-loop primary system with a hot, a cold and a crossover leg in each loop. A SG is dedicated to each loop and one single pressurizer attached to one of the hot legs controls the pressure of the entire primary circuit. The reactor produces around 3,300 MWth with a projected net electricity production of 1,200 MWe. Heavy neutron reflectors are placed around the core to improve neutron economy and fuel efficiency as well as to reduce irradiation to the vessel ensuring the intended 60 years of design life. All the major nuclear and safety systems are designed to withstand 0.3g of ground seismic acceleration and external and internal hazards, such as fires, aircraft crash and flooding. The design of the components and safety systems reduce CDF to less than 10^{-6} /RY.

Fuel

Nuclear safety, fuel management capabilities and fuel economy were significant factors that went into the design concept of the ATMEA1. The reactor provides a flexible 12–24 month fuel cycle using a low enriched core (<5 % ^{235}U) of UO_2. Burnable poisons in the form of gadolinium pellets in the fuel are used to control reactivity when fresh fuel is loaded into the core. Load following capabilities (25–100 %) are part of the standard core design, with the possibility to use cores consisting of up to one third MOX fuel. With a few minor design modifications, the ATMEA1 can support a 100% MOX core for future plutonium burning purposes. Using heavy neutron reflectors with smart fuel management is expected to allow 10 % less fuel consumption and radioactive waste generation per MW produced when compared to current PWRs. To ease the fuel handling and refuelling, the fuel pool storage area is located outside the reactor building. This combined with innovative high speed refuelling machine, reduces the planned duration of normal refuelling outages to about 16 days, for an anticipated overall availability factor of around 92 %.

Reactor pressure vessel and core catcher of ATMEA1

Special Design Feature, e.g. Construction Methods

- Shortening of construction period by adopting the latest proven construction technologies such as progressive pre-installed steel frame with deck plates, super heavy-duty crane, modular containment liner and pre-installation of equipment;
- SAMS: core catcher, cooling systems and hydrogen recombiners for long-term containment integrity;
- Provisions for off site emergency equipment utilization.

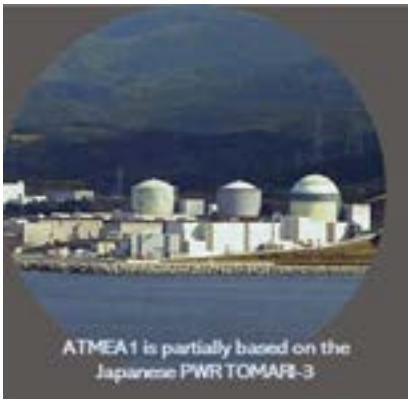
ATMEA1

Safety

Safety systems of the ATMEA1 are designed to meet current national and international sets of regulatory requirements including the U.S. NRC, France's ASN, Japan's NRA and Euratom. The safety and design of the ATMEA1 is based on deterministic analyses of DID aided by probabilistic analyses. The design considers various additional site-specific safety requirements such as seismic resistance, flooding and tsunamis. Safety functions are based on active operator initiated active systems with passive backup systems. All the essential systems assigned to protection and safety of the reactor have full, triple redundancy (and one additional safety train for support systems) with physically separated trains. Diversity and redundancy are designed into the plant in order to provide two heat sinks, autonomy of 30 days, and access to diverse water sources on site, among other things. MHI advanced accumulators are used as passive systems combining low pressure injection. In the case of an ex-vessel severe accident, a dedicated core catcher system with corium retention and cooling systems are available to mitigate the accident progression as well as to minimize the radiological release. Passive hydrogen control provides continuous service, and a pre-stressed concrete containment vessel with steel liner helps protect against large commercial airplane crash.

Operation

Digital instrumentation and control systems are used extensively in the operation of the ATMEA1 to improve the human-system interface with the goal to reduce the likelihood of operator errors. By encompassing the sources of human error in the design, testing and maintenance of the I&C systems, the physical demands on the operators are lowered and permissible response times for critical operator actions are increased. Additionally, top mounted in-core instrumentation systems on the vessel always give valuable information about the operations of the core. A remote shutdown station exists when the main control room becomes inaccessible. Design and configuration of components have been placed to ensure easy access for future maintenance and surveillance programs. They have been designed to eliminate certain phenomena, such as stress corrosion cracking or fatigue cracking, by removing certain structures from high neutron fluence areas. Shielding in the reactor building allows extensive online maintenance capabilities which help to reduce unnecessary plant unavailability. The average collective doses exposure over the plant lifetime is set to less than 0.5 man-Sv/y.


ATMEA1 is partially based on the Japanese PWR TOMARI-3

Technology Maturity/Readiness

- Currently undergoing licensing in various countries;
- Adaptability to different grid requirements (50 or 60Hz);
- Mid-range power output makes this reactor adaptable to countries with smaller or less developed electric grids.

Evolutionary Power Reactor

The EPR is an evolutionary design (Gen III+) developed by EDF and Framatome jointly with Siemens and German utilities. It combines and improves the best features of the latest reactors operating in France (1,450 MWe N4) and in Germany (1,300 MWe Konvoi). The EPR is designed to satisfy the demands of electrical utilities for a new generation of NPPs according to EUR, being at the same time an evolutionary and innovative design with increased level of safety and performance. The EPR benefits from the long experience of EDF and its partners building and operating more than 100 PWRs, which represents over 2,000 reactor-year of operation and embarks on all post-Fukushima design requirements. The EPR is designed with a power output of ~1,650 MWe and 60 years of operation with an average annual design availability factor of ~91 %. The EPR meets the highest safety requirements of the new generation of NPPs, significantly reducing potential radioactive releases in accidents including core melt, global CDF with no necessity of protective measures (no evacuation, no sheltering), and large early releases thus allowing very limited protective measures in area and time for accidents with core melt (no permanent relocation, limited sheltering).

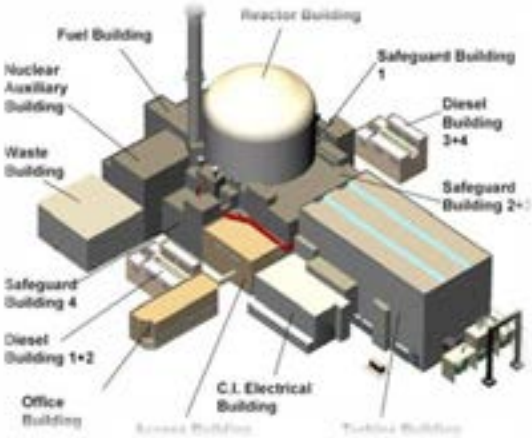
Nuclear Systems

The primary circuit consists of 4-loops, each containing a SG and a reactor coolant pump, and a pressurizer on one loop. The main components are designed to deliver a high core power (4,590 MWth) at high steam pressure level (7.8 MPa) for economic efficiency. The increased water inventory (large pressurizer and large SG) provides operating and safety margins. The RPV has a reduced number of welds which benefits to safety and maintenance, a top-mounted in-core instrumentation to suppress vessel penetrations below core elevation. A metallic heavy reflector provides fuel savings and lower RPV fluence. Main coolant lines and main steam lines are designed according to the break preclusion concept, allowing removal of the anti-whip devices which reduces cost and facilitates inspection.

Fuel

The EPR large core design (241 fuel assemblies, 17x17, 4.2 m) provides fuel management optimization and added safety margins. It allows flexibility for different irradiation cycle lengths (12–24 months), including UO_2 and MOX fuels with lower fuel cycle costs. The higher fuel burnup for given enrichment, due to low core power density, lowers the average thermal neutron flux by 7–15 %, and the production of long lived actinides is subsequently reduced. In-core neutron detectors enhance the core protection by direct monitoring.

Special Design Feature, e.g. Construction Methods



The overall plant layout and the design of civil structures provide a robust containment against radioactive releases and protection of the safety features against external and internal hazards. The reactor building, the four safety buildings, and the fuel building are located on a unique common raft, with the reactor building in the center. The reactor building, the fuel building and two of the four safety buildings are designed against air plane crash. Each safety building houses all equipment belonging to a safety division (mechanical / electrical / I&C) located in different floors with different areas per floor, which ensures a robust physical separation. The reactor building is made of a double wall concrete containment with an internal metallic liner.

EPR

Safety

Four independent safety divisions provide 4 x 100 % redundancy, which allows preventive maintenance during operation and offers high reliability. The DID principle is reinforced by implementing an adequate level of redundancy / independency / diversity / physical separation through the successive defence lines. Common-cause-failure events are considered to prevent core melt (WENRA level 3b–IAEA level 4a) through a deterministic approach, complemented by a probabilistic approach. A robust core melt mitigation is provided (WENRA/IAEA level 4): RCS depressurization system to eliminate high pressure CM, core catcher located below and aside the RPV pit to perform corium retention and cooling with passive flooding from the IRWST, dedicated containment heat removal system, passive hydrogen catalytic recombiners etc. Active and passive safety systems are combined to make optimal use of their respective advantages (good knowledge of performance and reliability of active equipment, operation of passive equipment minimizing support functions). Two digital I&C platforms (one for safety, the other for safety/operation) house all the I&C functions, with mutual back-up and an additional conventional platform is possible if required by the regulatory body. In the event of a loss of off-site power, emergency diesel generators have a fuel reserve for three days (extendable), and backup diesel generators are available.

Operation

Utilities involved in the EPR basic design provided feedback and operational needs (radioprotection optimisation and total generated waste reduced by 26 %). The total plant net efficiency is ~37 % thanks to innovative plant design, fuel cycle management and operational strategies. Power manoeuvrability is an intrinsic part of the EPR design which has significant flexibility capabilities such as fast load following and frequency control adjustments. The EPR has a high availability factor of >91 %. Additionally, the standardization of the plant's systems and components make use of consistent maintenance packages. The I&C systems are also fully standardised and the computerized control rooms are user friendly to minimize the human error factors.

Technology Maturity/Readiness

The robustness of the EPR design has been demonstrated through the analyses performed in the frame of post-Fukushima European stress tests, confirming the soundness of the adopted safety features. The EPR has already been licensed by four different safety authorities, showing that the design is able to meet a large variety of safety and regulatory requirements. The EPR Taishan 1&2 have been in commercial operation in China since 2018. EPR Olkiluoto 3 in Finland and EPR Flamanville 3 in France are close to fuel loading. EPR Hinkley Point 1&2 in the UK are under construction with two additional units under development in Sizewell. Six units are under development in Jaitapur in India. A fleet renewal programme is also under discussion with the French Government for a fleet of six EPRs including some design evolutions following EDF specific requests and optimization targets.

Hualong One

The Hualong One, now officially called the HPR1000 by the Chinese government, is the result of China National Nuclear Corperation (CNNC) and CGNPC merging their designs as suggested by the Chinese National Energy Administration. The Hualong uses systems from CNNC's ACP1000 and CGNPC's ACPR1000. Both reactors were conventional 3-loop PWRs, but ACP1000 core design was finally adopted to be placed in the Hualong One. The design incorporates the latest safety systems following internationally accepted standards, including backup passive safety systems, severe accidents' mitigation systems and enhanced seismic protection. Future reactors will be deployed by both companies separately, maintaining much of their supply chains, but each version will have slight differences concerning the safety systems.

Nuclear Systems

The reactor coolant system configuration is that of a conventional PWR 3-loop design. The sizing of the HPR1000 RPV, SG (especially secondary side) and pressurizer increases in the respective volumes compared to current 3-loop PWR designs. The free water volume between the level of the reactor coolant lines and the top of the active core is increased in order to improve the mitigation of LOCA (smaller breaks) by prolonging the period until the core begins uncovering or by minimizing the core uncover depth. The increased pressurizer water and steam volume, with associated pressure and level scaling, reduces the number of load cycles on relevant systems and components. The RPV high point vent system is designed to remove non-condensable gases from the RPV head during accident conditions.

Fuel

The HPR1000 reactor core is loaded with 177 fuel assemblies containing UO_2 fuel, ensuring sufficient thermal margin while increasing power output. The advanced fuel assembly of CF3 with independent property right of CNNC is adopted. With innovative zircaloy cladding material and design of grid, nozzle and guide tube, CF3 is applicable for long refueling cycle. The core design provides an operating fuel cycle length of 18 months with low leakage loading patterns and has the flexibility of extending the fuel cycle length. In addition, there are three independent means employed for the core reactivity and power distribution control such as the burnable absorber of gadolinium (Gd_2O_3) poisons, rod cluster control assemblies and soluble boron absorber.

Special Design Feature, e.g. Construction Methods

3rd generation PWR
with the latest nuclear safety standards and codes

Design features
177 fuel assemblies, single unit layout, double shell containment

3 active and passive systems
Residual heat removal from secondary side, containment heat removal, cavity injection and cooling

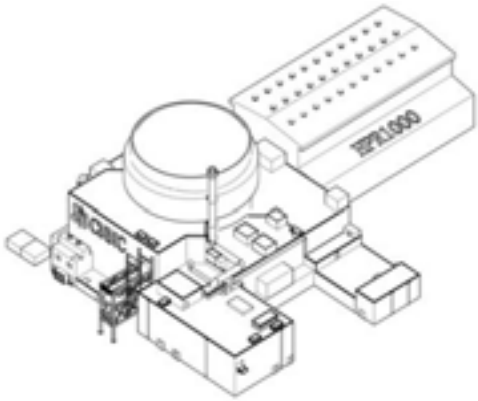
3 enhanced protection design features against
Earthquakes, commercial aircraft crash and plant emergency

HPR1000

LWR	Coolant H_2O	Moderator H_2O		
Capacity 1180 MWe	Fuel UO_2	SSE 0.3 g		
Capacity 3160 MWth	Fuel cycle 18 months	CDF $<10^{-7}$/RY		
Efficiency 36.7%	Design life 60 yrs	LERF $<10^{-8}$/RY	Operation Load follow & baseload	Non-electric application N/A

FUQING 5, 6

Safety



The application of active and passive safety design philosophy is innovation for HPR1000. The design inherits the mature and reliable active technology and introduces passive system as the backup for active system in case of loss of AC power. Both. the active and passive safety features are employed to guarantee the safety functions of emergency core cooling, residual heat removal, in-vessel retention of molten core, and containment heat removal. Extensive tests were performed to verify that the passive features will achieve the functions as designed. Probabilistic safety assessment results predict a CDF of less than 10^{-6}/RY and a LERF of less than 10^{-7}/RY. Comprehensive prevention and mitigation measures against various severe accident scenarios, including high pressure molten corium ejection, hydrogen detonation, basement melt-through, long term containment overpressure and SBO, are incorporated into the HPR1000 design. The design of nuclear island buildings adopts high seismic input standard by increasing the ground acceleration to 0.3 g.

Operation

The HPR 1000 operational performance and economic goals are in line with the requirements contained in utility requirements documents, e.g. plant availability, design lifetime, refueling cycle. Most of main equipment are proven and can be manufactured in China, which provides a more economic and convenient equipment supply chain. Advanced features such as but not limited to leakage-before-break technology and integrated RPV head structure reduce the cost and period of construction, maintenance and refueling.

Technology Maturity/Readiness

HPR1000 is under construction in China.

Water Water Energetic Reactor

There are two families of AES-2006 designs which are tailored to different national regulatory standards. The VVER1200/V392M version was developed by Moscow Atomenergoproekt on the basis of the AES-92 design. The reactors are being built at Novovoronezh II, Russia. The other family of AES-2006 (VVER-1200/491) designs was developed by Saint Petersburg Atomenergoproekt on the basis of the AES-91 originally developed for Tianwan, China. VVER-1200/491 plants are under construction in Russia at Leningrad II and Baltic. In anticipation of the increasing trend of electricity consumption around the world and the benefits from the economics of scale, a large PWR was designed with an electric output of 1,560 MWe. The new design complies with modern safety regulation, codes and standards as well as with the Russian and Euroatom regulatory standards. Main aspects of the design are high competitive ability on the global energy markets while maintaining high safety and operational reliability through the operational feedback from earlier projects. Major design features include extensive use of passive safety systems, tolerance to human errors, design lifetime of 50–60 years and large operational flexibility. Maximum fuel burnup is 70 GWd/t with a fuel cycle duration ranging between 12 and 24 months.

Nuclear Systems

In 2006 Rosatom finalized the designs for the VVER1200/AES2006, their most advanced deployable PWR design. These plants are designed to utilize previously proven technologies from the VVER1000 series to lower capital and construction costs, while maintaining high safety standards using active and passive safety systems. The AES2006 is designed according to the Russian Regulatory Standards and is also in compliance with the EUR. It is predicted to have a CDF of 10^{-6} /RY. Primary systems are very similar to the VVER1000 series, consisting of a 4-loop design. The 1,170 MWe VVER1200 reactors are expected to have an efficiency of 34 %, an availability factor of over 90 % and an expected service life of 60 years. Refuelling outages can be set between 12–18 months with an average 60–70 GWd/t. Modular construction techniques have optimized predicted construction times to ~54 months. In addition to the safety feature of the AES91/92 the AES-2006 has passive HRS for the containment and SGs in the unlikely case of a beyond design basis accident allowing independence and safe cooling for 72 hours without external power supply.

Fuel

VVER fuel is produced by the Rosatom subsidiary TVEL. The fuel company develops and manufactures fuel for the entire Rosatom VVER fleet including the VVER440, VVER1000 and VVER1200 reactors. Each fuel assembly must be tailored to the dimensions of the respective reactor model and the operational need. Modern fuel assemblies have increased technical and economical characteristics in accordance with new reactor designs and fuel cycles such as fuel cycle length, reactor power output, operational modes and anticipated fuel burnup. VVERs utilize enriched sintered UO_2 fuel pellets with an enrichment of up to 4.95 wt% ^{235}U fitted into fuel rods made from zirconium alloy cladding. These fuel rods are bundled together into hexagonal shaped fuel assemblies. Burnable neutron poisons (gadolinium oxide) can be used to control the excess reactivity of fresh fuel loaded into the reactor. Spacer grids, anti-vibration grids and anti-debris filters are used in the recent design to facilitate functions key to the stability and safety of the fuel, including optimal fluid flow, sufficient heat removal and mechanical stability during operation.

Special Design Feature, e.g. Construction Methods

Several unique concepts and construction materials are incorporated in the VVER designs. Plants containing VVER reactors employ horizontal SGs with relatively thick walled stainless steel SG tubes. SGs have been successfully operating for over 40 years without the need for replacement. During the design of some large components, efforts have been made to ensure transportability of the main equipment by rail. VVER RPVs lack penetrations in the reactor lower head, increasing their integrity in situations where severe core degradation or core melt is envisioned. The shells of the vessel are forged without longitudinal welds, thus enabling less frequent maintenance and surveillance tests on the RPV.

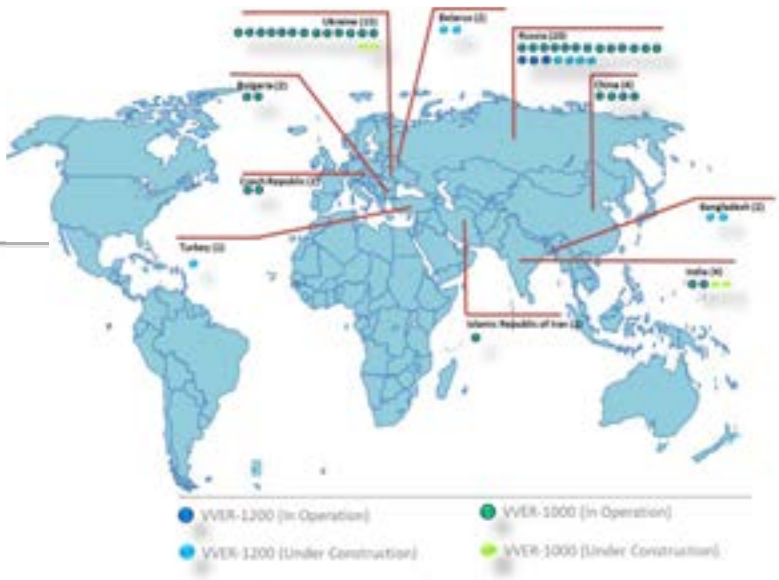
VVER 1200/1500

Safety

VVER units are designed in various sizes and models, but generally they can be distinguished by their electric output and safety systems, that are tailored to national regulation. The VVERs use multiple barriers and DID concepts to minimize the risk of harmful radiological releases to the environment. The distinctive design features that were part of earlier VVER reactor designs are being adopted today across the entire nuclear industry as inherent reactor protection mechanisms. Today's VVERs depend on large coolant volumes with increased secondary feedwater volumes to aid the cooling capability of the core via natural circulation mechanisms. Design features such as the large coolant volume above the core and increased primary inventory act as a damping mechanism for operational transients. These features slow the initiation, progression and releases of fission products in accident scenarios.

Operation

A total of 67 VVER reactors have been constructed in various countries since the 1960s starting with Russia during the Soviet era. Their cumulative operation time exceeds 1,200 reactor years. The experience gained from operating VVERS over the years has benefited Rosatom to improve their designs and increasing operational efficiency. The design lifetime of older VVER models were 30 years, but through continuous improvements in material and operational practices, the new generation of advanced reactor models (VVER1000, VVER1200 and VVER1500 series) are predicted to serve for 50 or 60 years. A number of VVER units have surpassed their intended design life of 30 years and are still under operation. Through implementing of plant life management programmes, operating licences have been renewed/extended in various countries. Several units have gone through safety upgrade processes, fitting the plants with the latest set of requirements mandated by regulators.

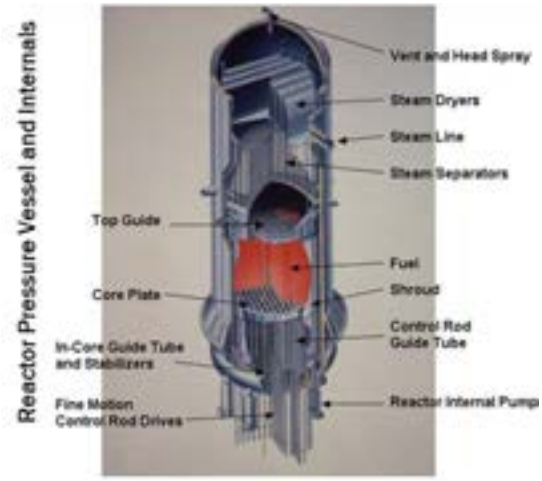


Technology Maturity/Readiness

Advanced Boiling Water Reactor

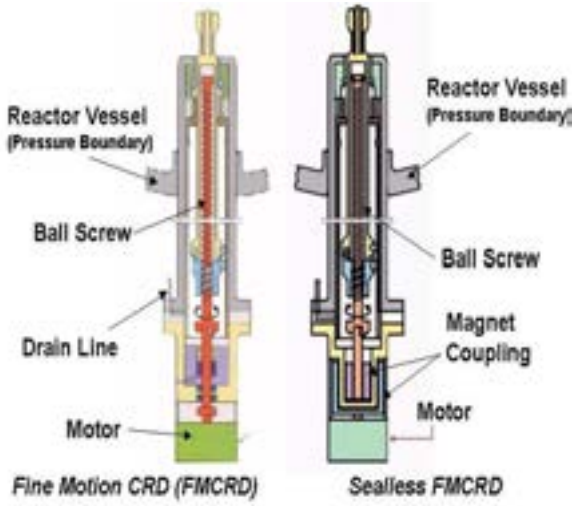
The ABWR design represents a complete design for a nominal 1,350 MWe power plant. The inclusion of such features as reactor internal pumps, FMCRDs, multiplexed digital fiber-optic control systems, and an advanced control room are examples of the type of advancements over previous designs that have been incorporated to meet the ABWR objectives. The ABWR design objectives include: 60 year plant life from full power operating license date, 87% or greater plant availability, less than one unplanned scram per year, 24 month refueling interval, personnel radiation exposure limit of 100 man-rem/year, CDF of less than 10^{-5} /RY, limiting LERF to 10^{-6} /RY, and reduced radwaste generation. The principal design criteria governing the ABWR standard plant encompass two basic categories of requirements: those related to either a power generation function or a safety related function.

Nuclear Systems



The incorporation of reactor internal pumps allows power changes of up to 30% of rated power to be accomplished automatically by recirculation flow control alone, thus providing automatic electrical load following capability for the ABWR without the need to adjust control rod settings. The ABWR FMCRDs are moved electronically in small in-crements during normal operation, allowing precise power management. The FMCRDs are inserted into the core hydraulically during emergency shutdown, with the backup provision for continuous electronic insertion.

Fuel



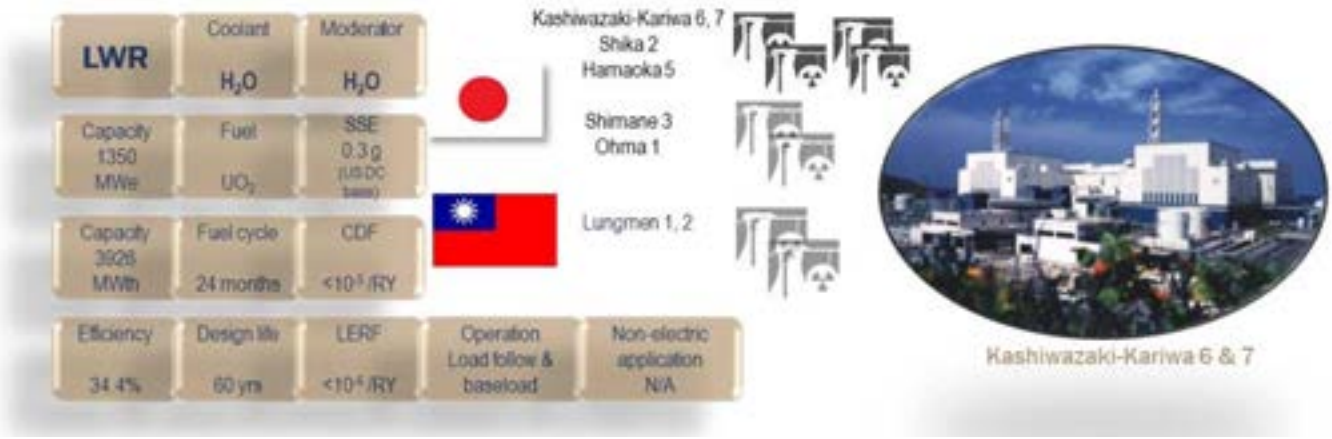
The ABWR uses sintered UO_2 as fuel at an average enrichment of 4 wt% of ^{235}U. Increased fuel utilization, performance and reliability are provided by the GNF2 fuel assemblies used in the ABWR. The fuel assemblies are manufactured to have increased corrosion and debris resistance in the BWR environment capable of operation at 120 % power for up to 24 months. The average burnup is about 50 GWd/t due to the higher fuel mass and high enrichment compared to previous BWR technologies. A lower power density results in improved fuel cycle costs and a greater manoeuvrability for operation.

Special Design Feature, e.g. Construction Methods

Modularization techniques are implemented to reduce costs and improve construction schedules. These techniques are applied to reactor building:

- Building reinforcing bar assemblies;
- Structural steel assemblies;
- Steel liners for the containment and associated water pools; and
- Selected equipment assemblies.

ABWR



Safety

Recognizing the need for continued safety enhancements in plant operation, one goal in designing the ABWR was to reduce CDF by at least an order of magnitude relative to currently operating BWR plants. Essential design features contributing to this are enhancement of the high-pressure ECCS and reactor heat removal functions, including the emergency AC power supply, and the installation of diversified countermeasures against anticipated transients without shut down. The adoption of reactor internal pump eliminates large attached recirculation piping, particularly involving penetrations below the top of the core, and make it possible for a smaller emergency core cooling system network to maintain core coverage during a postulated loss of coolant accident.

Operation

The ABWR design incorporates extensive automation of the operator actions that are required during a normal plant start up, shutdown and power range maneuvers. The automation features are designed for enhanced operability and improved capacity factor, relative to conventional BWRs. The extent of automation ensures that the primary control of plant operations remains with the operators who can intervene at any time. The ABWR control room design provides the capability for a single operator to perform all required control and monitoring functions during normal operations and emergency conditions. One man operation is possible due to implementation of the wide display panel for overall plant monitoring, plant-level automation, system-level automation via sequence master control switches, compact main control console design and implementation of operator guidance functions that display appropriate operating sequences. The role of the operator is to monitor the status of individual systems and the overall plant and the progress of automation sequences, rather than the traditional role of monitoring and controlling individual system equipment. The operating staff organization is based upon having two operators normally stationed at the control console.

Technology Maturity/Readiness

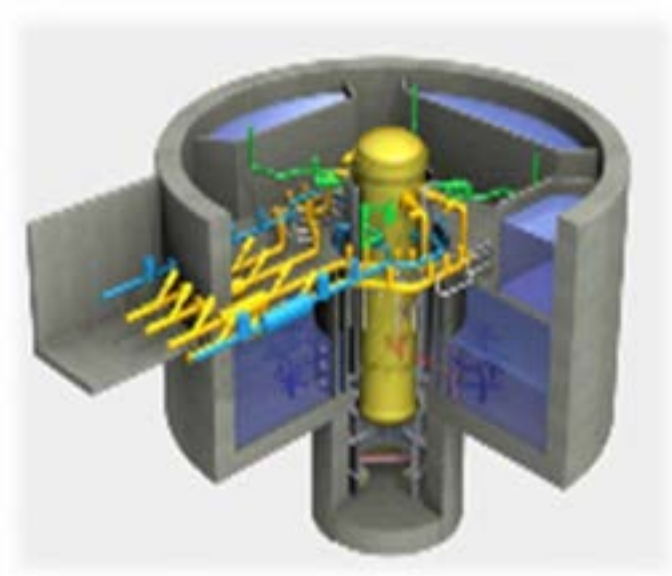
The Japanese version of the ABWR units have established excellent operations records. The goals for enhanced performance and improved operation & maintenance have been satisfied and proven through operational experience. The successful design, licensing, construction, and operation have paved the way for global commercial deployment. The ABWR was designed to improve safety, operation and maintenance practices, economics, radiation exposure and so on by the following factors:

- Design simplification
- Operation flexibility improvement
- Cost reduction of equipment and structures
- Reduction of construction period
- Scope reduction of the maintenance during operation and outages
- Making the maintenance easier and with lower radiation exposure
- Increasing the capacity factor
- Reduction of the power generating cost

Economic Simplified BWR

GE-Hitachi's (GEH's) ESBWR is a 1,560 MWe Gen III+ LWR that combines economy of scale, improvements in safety and design simplification to reliably and efficiently generate carbon free electricity. It is based on 60+ years of GEH's experience in design, construction and fabrication in the nuclear industry. The ESBWR builds upon the Advanced Boiling Water Reactor's (ABWR's) proven technology and achieves greater simplicity by utilizing natural circulation core flow and simple, natural phenomena driven safety systems. The ESBWR has 25 percent fewer pumps, valves and motors compared to active safety system based LWRs. Simplification leads to reduced operation & maintenance costs which, in combination with shorter projected construction time and lower costs, result in an economical power plant. The ESBWR has active non-safety systems to handle operational transients, but all reactor safety systems are passive and do not require AC electrical power or operator actions for cooling for more than seven days. Most systems and components in the ESBWR have proven operating histories from the ABWR. The innovative features have been tested in dedicated facilitates. The ESBWR design has been certified by the U.S. NRC.

Nuclear Systems



The ESBWR has a thermal efficiency of approximately 34% and its NSSS is similar to the ABWR in that it uses FMCRDs, same diameter RPV, same operational pressure, same steam dryer, and four main steam lines connecting the RPV to the turbine. The significant difference is that the ESBWR core flow is completely driven by natural circulation, whereas the flow in the ABWR is driven by reactor internal pumps. ESBWR natural circulation was achieved by increasing the vessel height, adding a partitioned chimney above the reactor core, decreasing active fuel height, and decreasing the hydraulic resistance in the down-comer. The taller RPV with its relatively larger volume of coolant results a very mild response to transient events and enhance the operational flexibility of the reactor.

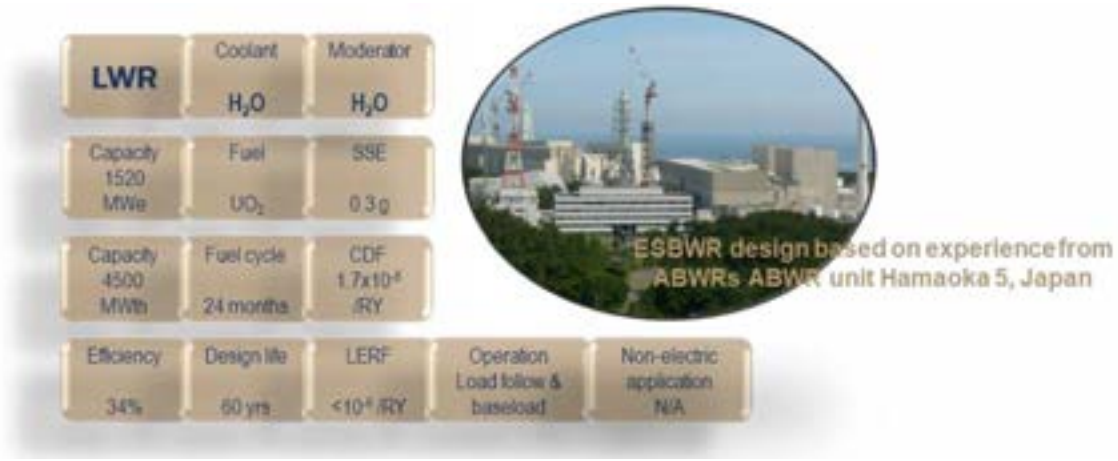
Fuel

The ESBWR uses a shortened version of standard BWR fuel that is used around the world. The shorter fuel enhances natural circulation. The fuel is made up of UO_2 pellets with an average enrichment of ~4.2% of U-235. An average burnup of 55 GWd/t which is within the BWR fleet experience provides flexible fuel cycle lengths of 12 to 24 months. The power density, linear heat generation rate and critical power ratio are within the operating envelope of the BWR fleet.

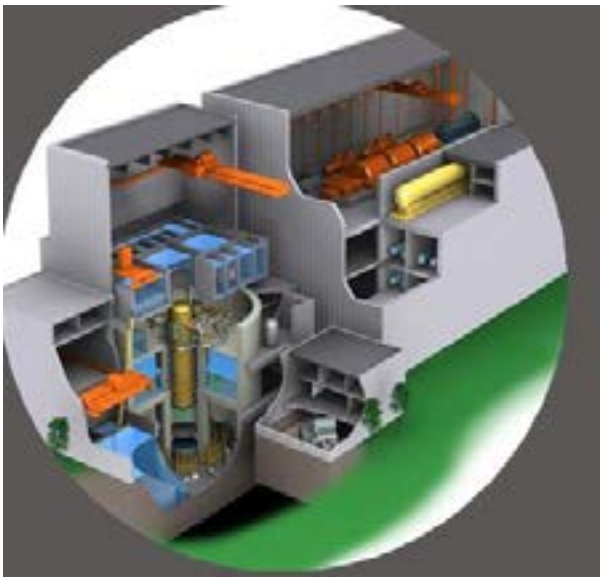
Special Design Feature, e.g. Construction Methods

The ESBWR relies extensively on the lessons learned from operating BWRs. Natural circulation was used in some past BWRs including, most recently, the Dodewaard nuclear power plant in the Netherlands. The safety systems are also driven by natural phenomena. The construction of the ESBWR is based on the proven construction techniques of the ABWR including multi-generational modularization techniques. The simplifications of the ESBWR design and the ABWR construction techniques allow the ESBWR to be constructed in a similar time frame as the ABWR while producing approximately 15% more power.

ESBWR



Safety



The ESBWR internal event core damage frequency is 1.7×10-8/yr which is lower than any other large, light water reactor. Diversity and redundancy were balanced to achieve this low core damage frequency. The isolation condenser system (ICS) provides condensation and gravity driven, safety related decal heat removal in the event of reactor shutdown and isolation such as a station blackout event. LOCAs are mitigated by the Gravity Driven Core Cooling System, Automatic Depressurization System and Passive Containment Cooling System. These systems provide a minimum of seven days of decay heat removal following postulated design basis accidents without the need for operator action or AC power. Additional coping time can be accomplished by adding water to pools at atmospheric pressure via designed-in connections in accordance with post Fukushima lessons learned. In the case of a severe accident, the Passive Containment Cooling System, pressure suppression containment and core catcher are in place to mitigate the progression of the accident and limit radiological consequences.

Operation

The comparatively low plant construction and development costs for the ESBWR with 60 years of design life and an availability factor of over 92% provide an economical way to produce carbon free electricity. Standardization and simplification of the plant systems with a large proven design base can reduce licensing and first-of-a-kind plant costs in many countries. Furthermore, simple passive safety systems with minimal connections and welds decrease the operation & maintenance costs. This will in turn reduce the operational exposure of radiation workers performing maintenance to less than 1 Sv/yr. The fully digitized control room with self-diagnostic features puts fewer demands on plant operators who will be monitoring the automated systems to ensure correct functionality and safe operation of the plant.

Technology Maturity/Readiness

The U.S. NRC approved the design certification of the ESBWR on September 16, 2014 and issued combined construction and operating licenses for Fermi 3 and North Anna 3.

KERENA

AREVA (now Framatome) and major European utilities including E.ON, Vattenfall, EDF and TVO developed an advanced BWR in 2008 called SWR1000 and renamed it to KERENA in 2009. The design and the development of the KERENA reactor aimed at producing an economically competitive plant for the future. KERENA combines years of experience in reactor design and operation with innovative features proven in dedicated research facilities. It is designed with a diverse mix of active and passive safety features to minimize the operational safety risks and mitigate accident scenarios. Operational flexibility and fuel cycle costs are optimized by load following capabilities, improved fuel cycle strategies and less complex maintenance procedures.

Nuclear Systems

The nuclear systems of KERENA are mainly based on the proven designs of Framatome's 1,300 MWe BWR plants with the exception of components for passive safety systems. The core produces 3,370 MWth which is converted to 1,250 MWe by an efficiency of 37 %. The coolant is recirculated by 8 reactor recirculation pumps inside the RPV. Saturated steam generated in the reactor core passes through moister separator reheater before entering the high pressure turbine via three main steam lines. Core thermal power is regulated by the reactivity feedback i.e. by changing the flow rate of the reactor recirculation pumps or by the hydraulic control rod scram system installed at the bottom of the RPV. The reduced active core height and large volume of coolant above the fuel reduce the probability of core uncovering during LOCA type scenarios.

Fuel

ATRIUM 12 fuel assemblies, using UO_2 or MOX fuel, are chosen as the reference fuel design for the KERENA reactor. The new fuel called ATRIUM 12 is modified ATRIUM 10 at low enrichments of UO_2 referring to 4.5–4.7 % of ^{235}U. ATRIUM 12 leads to structurally larger fuel assemblies but to reduced number of fuel assemblies in the reactor core compared to current BWR designs. Reducing the number of fuel assemblies shortens the standard refueling outages to about 11 days. The innovative core design and fuel management strategy promises a higher fuel efficiency and a reduction of long life radioisotope products by ~15 %. Through appropriate fuel management an average discharge burn-up of ~65 GWd/t is expected. The flexible fuel cycle length can be between 12 and 24 months. Additionally, KERENA is capable of using reprocessed uranium and up to 50 % MOX cores without significant design change of the systems or components. The spent fuel assemblies are stored in the spent fuel pool located inside the reactor building providing for residual heat removal and shielding.

Special Design Feature, e.g. Construction Methods

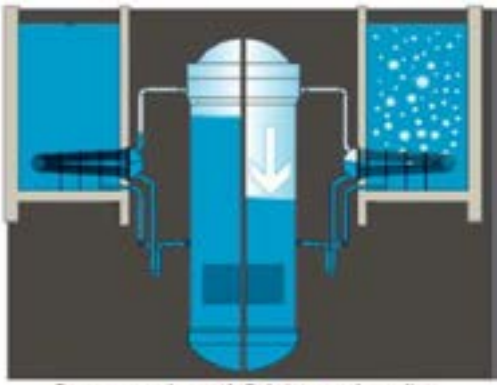
- Designed with input from European BWR Forum, an organization consisting of Europe BWR vendors and utilities;
- 19 BWR Units providing construction and operational experience of 60+ years;
- Minimum construction time through modular construction methods;
- Resistance to airplane crash and seismic vibration;
- Severe accident mitigation systems;
- Load follow operation (40 % and 100 % of its nominal power); and
- Enriched uranium fuel, reprocessed uranium or MOX fuel can be used.

KERENA

LWR	Coolant H_2O	Moderator H_2O
Capacity 1250 MWe	Fuel UO_2	SSE 0.3 g
Capacity 3370 MWth	Fuel cycle 24 months	CDF 10^{-7}/RY
Efficiency 36%	Design life 60 yrs	LERF $3x10^{-8}$/RY
Operation Load follow & baseload	Non-electric application N/A	

Illustration of KERENA plant layout

Safety



Emergency condensers: Left, during normal operation. Right, acting as ECCS when RPV water level drops.

Redundancy and diversity of integral safety systems in KERENA reactor are provided by two active and four passive qualified safety systems. An innovative approach of partially replacing active safety systems with passive safety systems is used to lower the calculated probability of accidents and their radiological consequences due to I&C failure or human error. Active systems employed for reactor protection are manually activated whereas the passive systems act as diversified back-up. In many accident scenarios, passive systems provide an increased time frame for the operator action. Passive components such as passive pressure pulse transmitters monitor the reactor water level and compare it with the secondary pressure build-up to ensure the safe operation of the plant. Safety systems for accident control without overheating / melting of the core include emergency condensers and gravity driven core flooding system. In-vessel retention, drywell flooding, containment cooling and hydrogen recombiners have the safety function confinement by preserving the RPV and containment boundaries and thus preventing radioactive releases to the environment.

Operation

Through streamlined design solutions, low maintenance requirements and the optimized fuel management strategies, the plant availability target is above 92 %. Load following capabilities between 40–100 % of full power and frequency control make KERENA a flexible tool in the energy mix of any utility. The reactor's optimized recycling processes reduce the overall environmental footprint of the plant during operation due to advanced system design and operation strategies. High primary water volume is beneficial to control transients and to avoid unnecessary reactor trips. Through significant simplification of system engineering, standardization and reduced number of components, there are less periodic maintenance procedures. Consequently the plant availability is high, the safety of the workers improved and the occupational dose to radiological workers is minimized.

Technology Maturity/Readiness

A European utility group successfully assessed compliance of the KERENA design with the EUR. The main assessors were EDF (France), TVO (Finland), and the German utilities. In the frame of an application for a decision in principle the Finnish safety authority STUK assessed the KERENA reactor as licensable in Finland. The construction time for the KERENA reactor is reduced as a direct result of the simplification of systems and the incorporation of passive equipment. These two aspects of the design have resulted in less components and have significantly reduced the number of safety-classified components.

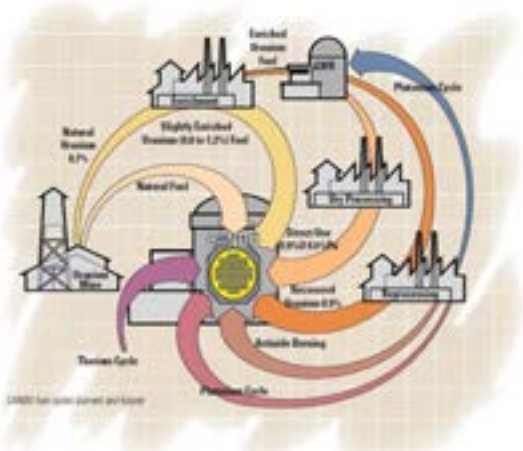
ENHANCED CANDU 6

In the CANDU design, the heavy water moderator circulates through a large tank called the calandria and through the moderator heat exchangers to remove the heat generated during reactor operation. The calandria is penetrated by several horizontal pressure tubes that form channels for the fuel to be cooled by a flow of coolant under high pressure. The heavy water coolant carries the heat to the SGs where the heat is transferred to the water to produce steam. The moderator system is completely independent of the HTS.

Nuclear Systems

The EC6 is a 740 MWe HWR that is moderated and cooled by heavy water. The fuel is contained in fuel bundles that reside in 380 pressure tubes, which are situated in a low pressure, low temperature tank called the calandria and is surrounded by a light-water filled concrete vault. The heat transport system from the CANDU 6 circulates pressurized heavy water in two interconnected "figure-of-eight" loops through 380 horizontal fuel channels, four SGs, four heat transport pumps, and four inlet and outlet headers that connect to the pressure tubes via feeder pipes. It has . It is the one pressurizer. Reactivity control is achieved automatically through liquid zone control units using light water and manually through control rods for large power reductions. An automated, on-power fuel handling and storage system handles fresh fuel loading and spent fuel transfer and storage in an underwater storage bay.

Fuel



The EC6 fuel bundle consists of 37 elements, each approximately 0.5m long containing sintered natural UO_2 pellets, Zircaloy 4 sheath with Canlub (graphite) coating on the inside surface, and two end caps, which are held together with welded end-plates and separated from each other and the pressure tube via appendages. Several fuel bundles are exchanged on a daily basis using the automated online fuel handling system thereby keeping reactor conditions, such as core-average reactivity and burnup, constant. Using NU as fuel permits national fuel cycle independence and technology transfer for localizing fuel manufacture has been achieved successfully in all countries operating CANDU 6 reactors. Several other emerging fuel cycles can also be used, such as recovered or slightly-enriched uranium (0.9–1.2 % enriched ^{235}U from reprocessed commercial LWR fuel), Natural Uranuim Equivalent (blended from used LWR fuel and depleted uranium to obtain 0.7 % NU enrichment) reprocessed high-burnup MOX fuel, or thorium based fuel cycles which can benefit long-term energy security of a nation.

Special Design Feature, e.g. Construction Methods

Using natural uranium as fuel gives the EC6 fuel cycle independence, avoiding the need for enrichment capability or complex fuel transactions. This reactor is designed from experience and feedback gained through the construction and operation of CANDU 6 plants, which have been deployed in five countries. The EC6 incorporates innovative features and new technologies to enhance safety, operation and performance. Key passive safety systems are introduced in addition to the proven systems of the previous CANDU 6 reactors. The EC6 is very flexible with regard to the choice of fuel cycle, allowing different countries to optimize for local fuel availabilities, national priorities, integration with other reactor technologies and grid sizes.

EC6

HWR	Coolant D_2O	Moderator D_2O
Capacity 740 MWe	Fuel UO_2	SSE 0.3 g
Capacity 2084 MWth	Fuel cycle On-line	CDF $<10^{-6}$/RY
Efficiency 35.5%	Design life 60 yrs	LERF $<10^{-7}$/RY
Operation Load follow & baseload	Non-electric application N/A	

Safety

The EC6 plant has been augmented with additional passive, accident resistance and core damage prevention features. Retained features include the two independent passive shutdown systems, each of which is 100 % capable of safely shutting down the reactor and situated in the low pressure moderator region. One system uses spring-assisted, gravity driven shutoff rods while the other injects gadolinium nitrate solution from high pressure tanks into the moderator. ECC provides core refill and cooling by passive accumulator tanks at high pressure and pumps at medium and low pressures. The low pressure moderator serves as a passive heat sink during postulated accident scenarios and the large volume of light water surrounding the calandria provides a second (passive) core heat sink in case of core melt. Elevated water tank located in the upper level of the reactor building provides passive make-up cooling water via gravity feed to the calandria vessel and the calandria vault. The EHRS provides an adequate long-term heat sink following an unavailability of the HRS to the SGs, ECC heat exchangers and the heat transport system through the ECC piping. The EHRS is a modularized system, each module having its own independently powered 100 % capacity pump. These systems and a low flow containment spray system comprise the new severe accident recovery and heat removal system, which operates with the gravity-driven passive water supply in the short term, the EHRS in the medium term, and an independent diesel-powered pump-driven recovery circuit in the long term.

Operation

Based on experience from CANDU 6 reactors and design improvements for efficient operation and ease of maintenance, the EC6 is predicted to have design life of 60 years (with one mid-life refurbishment of certain critical equipment, such as the fuel channels and feeders) and an overall availability factor of 92 %, achieved through online refuelling and periodic short-duration maintenance outages of 30 days per 36 months. Designers expect potential deployment around the world due to EC6's high flexibility in terms of source and manufacturing of fuel The standard design of an EC6 envisions a twin-plant complex, using open-top construction and pre-assembled modules, but a single reactor can be built with no significant changes to the basic design. With its comparatively small electrical output, single units could be deployed to countries with small and medium sized electrical grids. Improved plant operability and maintainability, including reducing worker exposure, are achieved by automation of some standard procedures, improved material and plant chemistry, a number of health monitoring systems, and the ability to return to full power immediately following grid interruption. The advanced control room design and self-diagnostic systems requires a minimum of operator action for all phases of station operation.

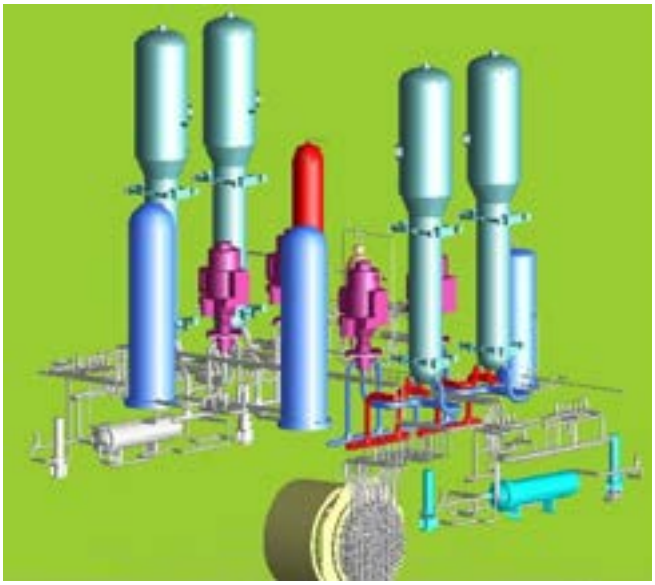
Technology Maturity/Readiness

The EC6 design benefits from the proven principles and characteristics of the CANDU 6 design and the extensive knowledge base of CANDU technology gained over decades of operation. Based on the reference CANDU 6 plant (the Qinshan plant in China), it includes changes driven by safety, licensing, operability, and client requirements. The EC6 design meets all Canadian regulatory requirements, which are aligned with the IAEA recommendations.

INDIAN PRESSURIZED HWR-700

The 700 MWe Indian Pressurized Heavy Water Reactor (PHWR) design is similar to 540 MWe PHWR units with required modifications to increase the plant rated output by safely utilizing available thermal margins. The design has a more compact layout and includes improved safety features for management of design extension conditions. The design retains the safety features of standardized Indian PHWR 220 MWe and 540 MWe units, which include: two independent, diverse and fast acting shutdown systems; integral calandria–end shield assembly; water filled calandria vault surrounding the calandria; and double containment of reactor building. In addition, 700 MWe PHWR design includes improved safety features, such as, passive decay heat removal system, containment spray system, steel lined primary containment and accident management provisions.

Nuclear Systems



The core comprises 392 coolant channels made of Zr-2.5 % Nb pressure tubes, each of which is kept concentric to its respective calandria tube by four tight-fit garter springs. The inter-space between pressure tube and calandria tube is filled with recirculating carbon dioxide to detect any water leak by monitoring dew point of carbon dioxide. The high pressure, high temperature heavy water coolant in pressure tubes is kept separated from the low pressure, low temperature heavy water moderator. The core produces 2,166 MWth of heat which is converted into a plant output of 700 MWe. Natural uranium fuel in the form of short bundles (0.5 m) resides inside pressure tubes. The design permits boiling to the extent of 2–3 % at pressure tube exit. The calandria is surrounded by light water filled concrete vault. The primary heat transport system has two independent and identical figure of eight loops, which are uniformly distributed in the core by adopting interleaving of feeders to limit void coefficient of reactivity in case of a postulated LOCA. Each loop has two SGs and two circulating pumps. Due to elevation difference between the core and SGs, natural circulation of the coolant permits passive core cooling under shutdown conditions (with no forced circulation). The low temperature moderator in calandria and light water in calandria vault acts as passive safety heat sinks to delay progression of and mitigate design extension conditions.

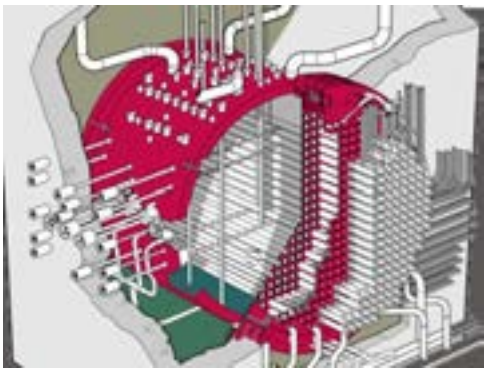
Fuel

The design uses natural uranium fuel, with a Zircaloy-4 cladding, thereby has very little excessive reactivity and during normal operation there is no need for neutron poison inside the fuel or in the moderator. The lack of excess reactivity requires regular online refueling of the reactor, which is carried out by two identical and remotely operated fuelling machines, working in unison, at either ends of the coolant channel. In fuel transfer system, a low pressure light water equipment, mobile transfer machine has been introduced to transfer the new/spent fuel bundles. The short length of fuel bundles helps to mitigate consequences of single bundle failures. Refueling rates can vary according to the previous full core reload. During equilibrium operation stage (about 600 days after fresh core load), approximately two fuel channels are refueled daily. The average discharge burnup using natural uranium is ~7 GWd/t; but it can be increased to 15 GWd/t using fuels with higher fissile content such as slightly enriched uranium and MOX.

Special Design Feature, e.g. Construction Methods

IPHWR-700

Safety



Two fast, independent methods provide total shutdown ability. The first consists of spring-assisted gravity-driven cadmium shutoff rods and the second method employs the injection of gadolinium nitrate into the calandria. Control rods and a liquid zone control system are used for routine power changes. The ECCS is built on 2x100 % redundancy with single failure criterion for each loop and consists of high pressure light water injection, followed by long term coolant water recirculation. Steel lining in the primary containment allows reduced leakage during postulated accident conditions. Independent injection is possible via connections from fire water system diesel driven pumps to the SGs, passive decay heat removal system, calandria, calandria vault and spent fuel storage bay. As a part of accident management, water connections for important systems from outside reactor building, passive catalytic recombination devices for hydrogen management, containment spray system, containment filtered venting system, air cooled DG sets and on-site emergency support center are provided. The design incorporates on-site seismically qualified storage of water and fuel oil for use in normal operation and as part of accident management.

Operation

High flexibility in terms of fuel cycle and manufacturing of fuel make the IPHWR-700 an economical reactor. From a fuel utilization perspective, operating the reactor at close to full power as much as possible would lead to optimal reactor performance. The reactor has an estimated design life of 40 years. The systems are designed so as to keep the radiation exposure to occupational workers and environmental releases to as low as possible during operation. The plant layout and shielding, along with operational practices and following ALARA principle minimizes the collective dose for plant workers. The main control room is designed with ergonomic considerations to ease the sensory impact on operators and thus reduce the probability for human errors affecting the availability, reliability, and safety of the plant. To perform a remote shutdown, decay heat removal and monitoring of critical parameters, a backup control room is provided for use in the event the main control room becomes inaccessible.

Technology Maturity/Readiness

Nuclear Power Corporation of India Ltd. (NPCIL) is a public sector enterprise under the Department of Atomic Energy of India, responsible for design, construction, commissioning and operation of NPPs. NPCIL's most recent design is the 700 MWe PHWR, designed taking into consideration decades of feedback and experience from operation and construction of a fleet of 18 units PHWR of 220 MWe and 540 MWe. It has horizontal pressure tubes, fuelled by natural uranium and using heavy water as both coolant and moderator. A series of 700 MWe PHWR units are planned following successful operation of the 220 MWe and 540 MWe PHWR units. First of the 700 MWe unit is likely to attain its first criticality very soon, and five more units are under construction. Further, Government of India has accorded administrative approval and financial sanctions for 10 more such units, to be implemented in fleet mode.

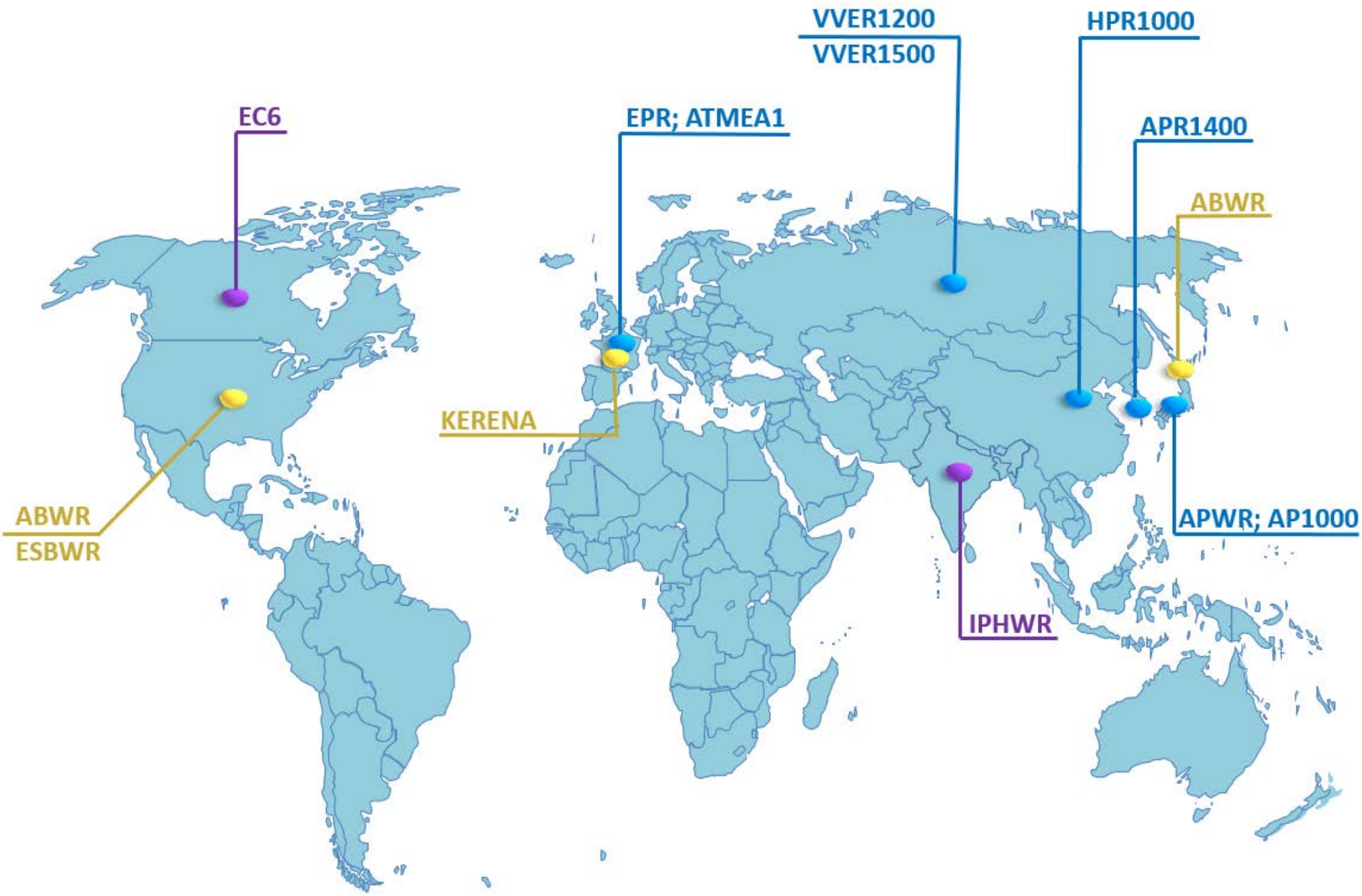

VVER1200
VVER1500
HPR1000
EC6
EPR; ATMEA1
APR1400
ABWR
KERENA
ABWR
ESBWR
APWR; AP1000
IPHWR

IMAGE REFERENCE

The IAEA would like to thank the individuals and organization that have provided contributions and review of the content of this booklet.

COVER PAGE:

(Left to right, top to bottom)

ABWR:	IAEA Imagebank
VVER1000:	IAEA Imagebank
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AP1000:	IAEA Presentation China 2013

Images throughout the Booklet:

All figures in this booklet are either from the 2015 edition of the Advanced Large Water Cooled Reactors booklet, or subsequently directly supplied by the respective reactor technology holder.

ACRONYMS AND ABBREVATIONS

ALARA	As Low As Reasonably Achievable
ARIS	Advanced Reactor Information System
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
DBA	Design Basis Accident
DG	Diesel Generators
DID	Defense In Depth
ECCS	Emergency Core Cooling System
EHRS	Emergency Heat Removal System
EUR	European Utility Requirements
FMCRD	Fine Motion Control Rod Drive
GWd	Giga Watt day
HRS	Heat Removal System
HVAC	Heat, Ventilation, Air Conditioning
I&C	Instrumentation and Control
LERF	Large Early Release Fraction
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MOX	Mixed Oxide
MWe	Mega Watt electric
MWth	Mega Watt thermal
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
O&M	Operation and Maintenance
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RY	Reactor Years
SBO	Station Black-Out
SG	Steam Generator
SSE	Safe Shutdown earthquake

For further information:
Nuclear Power Technology Development Section (NPTDS)
Division of Nuclear Power
IAEA Department of Nuclear Energy

International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria

Telephone: +43 1 2600-22812
Fax: +43 1 2600-29598
Email: Official.Mail@iaea.org
Internet: www.iaea.org

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