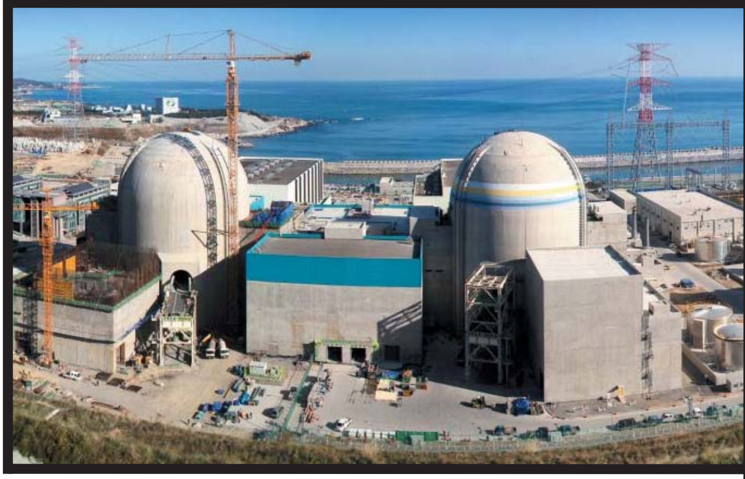
Advanced Large Water Cooled Reactors

A supplement to the IAEA's

Advanced Reactor Information System (ARIS)



September 2015

IAEA
International Atomic Energy Agency
Atoms for Peace

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PREFACE

Despite the adverse ramifications from the accident at the Fukushima Daiichi nuclear power plant, nuclear power is receiving increasing global interest, particularly in Asia. A growing number of countries are considering building nuclear power plants to meet increasing energy needs of their growing economies while decreasing their greenhouse emissions. In his public lecture at the Singapore Energy Market Authority in January 2015, IAEA Director General Yukiya Amano stated that:

> *In the four years since the Fukushima Daiichi nuclear accident in Japan, huge improvements have been made to nuclear safety all over the world, and there has also been significant progress in treating and disposing nuclear waste. Remarkable research is being done on new generations of reactors which will be safer and generate less waste.*

Member states, both those considering their first nuclear power plant and those with notions to expand their existing programmes, are vitally interested in obtaining current information about designs for reactors that are deployable now or in the near term. Fulfilling its mission as stated in the original IAEA Statute (Article III.A.3: “To foster the exchange of scientific and technical information on peaceful uses of atomic energy”) the Nuclear Power Division of the IAEA has regularly issued publications on the status of nuclear reactor technology developments. Over the years the manner of information dissemination has evolved and the latest rendition is offered online in the form of an online database that presents unbiased, detailed design descriptions for currently available nuclear power plants. This Advanced Reactor Information System (http://aris.iaea.org) includes reactors of all sizes and types, from evolutionary nuclear plant designs for near term deployment to innovative concepts still under development.

Hard copy supplements to the ARIS database focusing on Small Modular Reactors (under 700MWe capacity) and on Fast Reactors have already been published:

Advances in Small Modular Reactor (SMR) Technology Developments
https://www.iaea.org/NuclearPower/Downloadable/SMR/files/IAEA_SMR_Booklet_2014.pdf

Status of Innovative Fast Reactor Designs and Concepts
https://www.iaea.org/NuclearPower/Downloadable/FR/booklet-fr-2013.pdf

Water Cooled Reactors have played a significant role in the commercial nuclear industry since its inception and currently account for more than 95% of all operating commercial reactors in the world. Of the 67 nuclear reactors now under construction, 64 are water cooled reactors. Therefore, it seems timely and imperative to offer this booklet, which provides an overview of the status of advanced, large (700 MWe or more) water cooled reactors. The objective is to provide Member States with a brief overview of the large nuclear power plants considered currently deployable. It should be regarded as a complementary publication to the ARIS database itself, as well as to the IAEA guidance document for evaluating nuclear power plants, *Nuclear Reactor Technology Assessment for Near Term Deployment (IAEA NE Series NP-T-1.10).*
http://www-pub.iaea.org/books/IAEABooks/8950/Nuclear-Reactor-Technology-Assessment-for-Near-Term-Deployment

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International Atomic Energy Agency (IAEA)

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

PWR BWR HWR

Table containing important reactor data

Capacity: Gross electric output in MWe and reactor thermal output in MWt
Design Life: Life of non-replaceable components
Temperature: Averaged temperature between core inlet and core outlet
Pressure: Reactor operating pressure
Efficiency: Plant net efficiency
Enrichment: Enrichment of new fuel in equilibrium at core reload

Full name of reactor design

Acronym for the reactor design

Name of design organization

Figures depicting salient aspects of reactor design

Introduction of reactor design with key features summarized

Information about type of fuel and fuel cycle related issues

Illustration of the primary circuit with large components

Information on operation and construction status of design

Number of Units
(as of Sept 2015)

In Operation

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Information on reactor systems that relate to NSSS, safety, fuel and operation

Useful information of general nature

Flag of the country where units are in operation or under construction.
Source: IAEA PRIS database

Unit Names

Evolutionary Power Reactor

AREVA

AREVA, design organization of the EPR™, is a large multinational consortium specializing in nuclear energy with its headquarters in Paris, France. The design has benefited from AREVA's cumulative construction experience of over 100 nuclear reactors in various countries and operational experience of the French and German nuclear industries. This led to the design of an advanced PWR reactor system with highly reliable and diversified safety systems, limited radiological impact and reduced margins for human errors. The EPR has a power output of about 1750MWe, managing to increase fuel utilisation efficiency and decrease radioactive waste products through flexible fuel management strategies, which include the use of MOX fuels. State of the art, four-fold redundancy ensures the availability of the safety related systems. Severe accident mitigation systems, i.e as core catchers and coremelt retentions systems, reduce and delay the impacts of accident scenarios. Currently four EPR reactors are under construction in Finland, France and China, with additional design reviews underway in the US and the UK.

Artist's rendition of EPR Olkiluoto 3, Finland

Nuclear Systems

The EPR nuclear systems are designed according to a conventional 4-loop nuclear plant concept following the French N4 and the German KONVOI models. The primary circuit consists of 4 loops, each containing a steam generator and a reactor coolant pump. The overall pressure of the primary circuit is controlled by a pressurizer connected to one of the primary loops. The increased inventory of the primary circuit, when compared with currently operating conventional PWRs, helps to diminish operational transients and acts as an added safety feature during DBA. Through the addition of neutron reflectors inside the RPV, the life-limiting irradiation damage to the vessel is expected to be minimized and a plant design life of about 60 years is envisioned. Furthermore, the RPV has reduced number of welds and their improved geometry leads to the reduction of maintenance activities and cost during the plant's life time.

EPR Reactor Vessel and Primary Systems (Courtesy of AREVA)

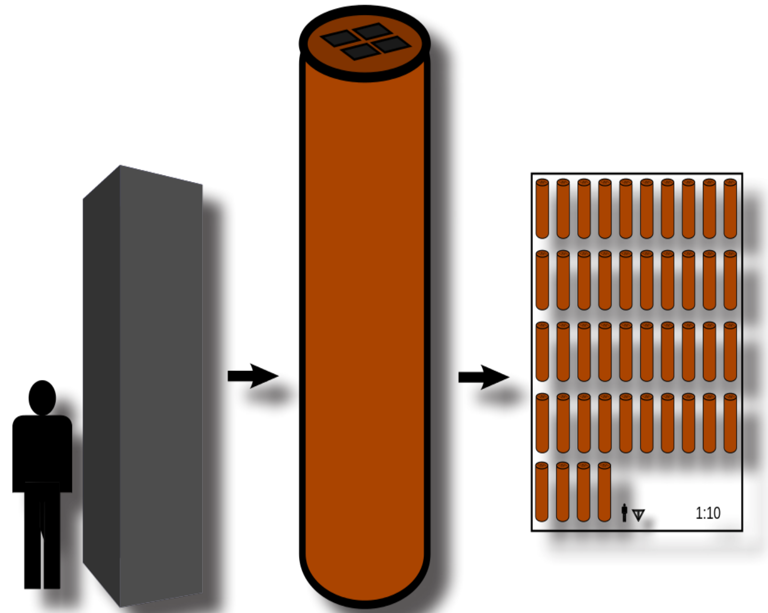
Safety

Safety related systems of the EPR are designed to be mechanically simple and arranged in accordance with the general principles of diversity, redundancy and physical robustness. A set of quadruple redundant systems, which are independent and geographically separated, are installed to protect the nuclear related systems from internal and external hazards. Two of the four ancillary safety buildings, constructed on concrete rafts, are aircraft crash protected. The EPR is designed to withstand seismic incidences with ground acceleration of up to 0.25 g. Considering all the installed safety systems, in concert with defence-in-depth concepts, the CDF for the EPR is calculated to be 10^{-7}. There are several active and passive systems to mitigate the consequences of a very unlikely case of a severe accident, including coolant recirculation, containment spray systems, core catchers, gravity driven systems, double containment and natural circulation. Emergency diesel generators have enough fuel for three days, and backup diesel generators are available, along with emergency DC batteries, for an additional day's power supply to vital nuclear instrumentation, control, and operating systems.

EPR Flamanville 3, France (Courtesy of AREVA)

EPR™

LWR	Coolant Light Water	Moderator Light Water
Capacity 1750 MWe	Temperature 312.6 °C	Fuel LEU
Capacity 4590 MWt	Pressure 15.5 MPa	Enrichment 4.95 %
Design Life 60 Yrs	Efficiency 36 %	Fuel Cycle 24 Mos



Representation of total radioactive waste produced by one EPR reactor during 60 years of operation (Courtesy of AREVA)

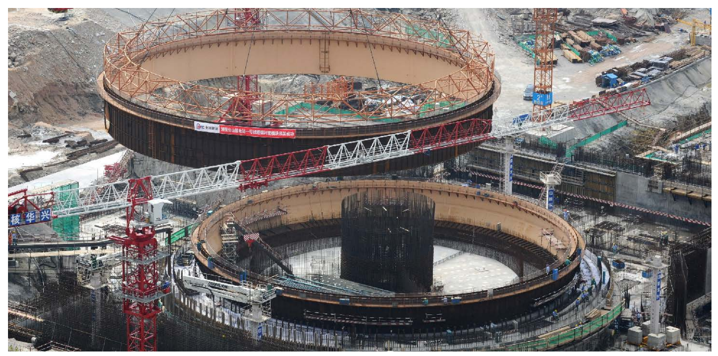
Fuel

Design of the EPR core is characterized by considerable margins for fuel management optimization incorporating conditions flexible for different irradiation cycle lengths and low fuel cycle costs. The reactor can operate with a fuel cycle range between 12 to 24 months using ENU, ERU and MOX fuels, according to the specific needs of the utilities. The larger core and addition of neutron reflectors to reduce neutron leakage generate added fuel savings in the EPR, which can use UO_2 fuel with an enrichment level of up to 5 wt% of U^{235}. Additionally, in order to function as a plutonium burner, PuO2 (up to an enrichment of 12.5%) is an option for the EPR. The larger core size and primary inventory permits a larger power output of around 4600 MWth, which effectively raises the thermal efficiency of the plant. Furthermore the higher fuel burn-up for given enrichment, due to low core power density, lowers the average thermal neutron flux by about 7-15% and the production of long lived actinides is subsequently reduced.

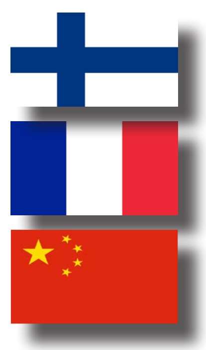
Operation

Through innovative plant design, fuel cycle management and operational strategies the total plant net efficiency is in the order of 37%, while reducing the total produced waste by 26%. To comply with modern complex electricity grid systems, load following capabilities are designed into the EPR, with 60% to 100% load for normal operation mode and a 25-60% load for "less usual" mode.

Due to a design that incorporates good availability/access for maintenance and in-service inspections, the EPR™ has a high availability factor of >92% taking planned and unplanned operation into consideration. Additionally the standardization of the plants systems and components make use of consistent maintenance packages ensuring a reliable process for the evaluation of the systems. The I&C systems are also fully standardised and the computerized operator friend control rooms are designed to minimized human error factors.



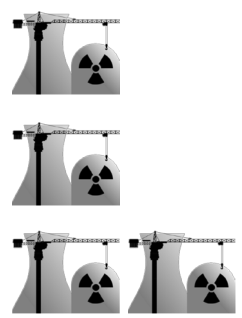
Construction of EPR Taishan, China (Courtesy of AREVA)



Olkiluoto 3

Flamanville 3

Taishan 1, 2



China's Advanced PWRs

China has the largest number of reactor units under construction in the world due to their large scale investment into the industry to develop a sustainable energy mix. As of 2015, it has 26 commercial nuclear power reactors in operation as well 24 new reactor units under construction at various sites across the country. More reactor units are in various phases of planning. The Chinese reactor portfolio includes a variety of reactor types that are available on the world market, including CANDU6-, EPR-, AP1000- and VVER-units. The country has also successfully developed domestic PWR designs through the construction experience and technology transfer from companies such as CANDU Energy Inc., Westinghouse and AREVA. The most important domestic Chinese PWR designs include the CPR-1000, ACPR-1000, CAP-1400 and Hualong One, also known as HPR1000.

Artist's rendition of CAP1400

CPR-1000

CNNC's CPR-1000 ('Improved Chinese PWR') design, with an electric output of 1000MWe, is based on the 900MWe 3- loop French M310 plants and make the largest proportion of current reactors under constructions. Even though China has a nearly complete domestic supply chain for the CPR-1000, the intellectual property rights are retained by AREVA and thus restricts its use to domestic market. The first CPR1000 was connected to the Chinese grid in 2010, but the granting of new construction licences have been suspended in favour of other advanced domestic designs.

LWR	Coolant: Light Water	Moderator: Light Water
Capacity: 1030 MWe	Temperature: 311 °C	Fuel: LEU
Capacity: 3000 MWt	Pressure: 15.5 MPa	Enrichment: <5 %
Design Life: 60 Yrs	Efficiency: 32.9 %	Fuel Cycle: 18 Mos

Fangchenggang 1, 2
Ningde 1, 2, 3, 4
Ling Ao 3, 4

Hongyanhe 1, 2, 3, 4
Yangjiang 3, 4

Fangjiashan 1, 2
Fuqing 1, 2, 3, 4
Yangjiang 1, 2

ACPR-1000

The ACPR-1000 (Advanced Chinese PWR) was designed by the China Guangdong Nuclear Power Corporation based on the CPR-1000 (CGNPC) with full Chinese intellectual property rights. The design focuses on the safe performance of the reactor while improving the economic efficiency. This enhanced version of the 3-loop CPR-1000 has higher seismic standards, features a double containment and a reactor core catcher for SA mitigation purposes. The reactors is designed with SAMS including such in-vessel retention and spray systems to for containment heat removal. Furthermore the ACPR-1000 meets major post-Fukushima safety requirements.

LWR	Coolant: Light Water	Moderator: Light Water
Capacity: 1150 MWe	Temperature: 311 °C	Fuel: LEU
Capacity: 3500 MWt	Pressure: 15.6 MPa	Enrichment: <5 %
Design Life: 60 Yrs	Efficiency: 33.0 %	Fuel Cycle: 18 Mos

Yangjiang 5, 6
Hongyanhe 5

Hualong One (HPR1000)

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 design as suggested by the Chinese National Energy Administration. The Hualong uses systems from CNNC's ACP-1000 and CGNPC's ACPR-1000. Both reactors were conventional 3-loop PWRs, but ACP-1000 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, SA 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.

LWR	Coolant: Light Water	Moderator: Light Water
Capacity: 1150 MWe	Temperature: 301 °C	Fuel: LEU
Capacity: 3050 MWt	Pressure: 15.7 Mpa	Enrichment: <5 %
Design Life: 60 Yrs	Efficiency: 36.6 %	Fuel Cycle: 18 Mos

Fuqing 5

CAP-1400

The CAP-1400 is a large 1400MWe PWR design developed through cooperation between China's State Nuclear Power Technology Corporation (SNPTC) and Westinghouse. The design is based on the AP1000 reactor which utilize passive safety systems and the simplified systems design philosophy to increase safety and operational flexibility. The reactor safety designs have been improved after the Fukushima Daiichi accident to accommodate enhanced seismic design and enhanced response capacity under BDB type events. The CAP-1400 possesses MOX fuel loading capabilities.

The valuable lessons learned from the construction of AP1000 units in China have further helped to reduce issues faced during the construction process. Modularization and advanced construction techniques have helped minimize delays during construction. The first CAP-1400 will be constructed at Shidaowan site in Shandong Province, where site preparations have been ongoing since early 2014.

LWR	Coolant: Light Water	Moderator: Light Water
Capacity: 1500 MWe	Temperature: 304 °C	Fuel: LEU
Capacity: 4058 MWt	Pressure: 15.5 MPa	Enrichment: 4.95 %
Design Life: 60 Yrs	Efficiency: 34.5 %	Fuel Cycle: 18 Mos


Construction of CPR-1000 at Fangchenggang, China


Construction of CPR-1000 and ACPR-1000 at Yangjiang, China


Site preparation for CAP-1400 Shidaowan, Shandong Province, China

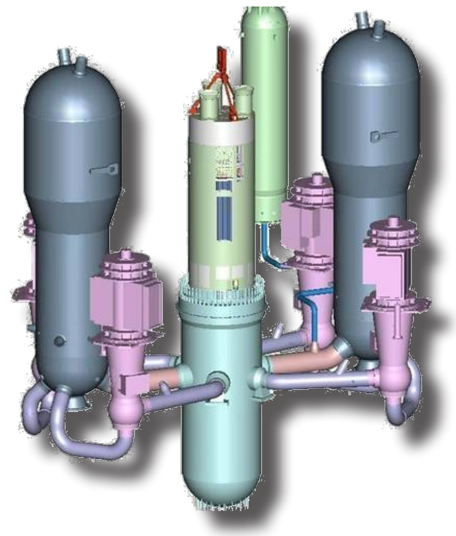
Advanced Power Reactor 1400

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 South Korea. This reactor is a 1400 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 US System 80+ with higher safety and seismic resistance features. KEPCO's 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.

Construction of Shin-Hanul 1, Republic of Korea (Courtesy of KEPCO)

Nuclear Systems

The APR1400's overall nuclear systems are very similar to the Korean OPR1000 type reactors, which have a well proven operating history of over a decade in South Korea. Additional features have been added to enhance safety margins and alleviate transients or operational occurrences. The primary system has two coolant loops, each consisting of one large steam generator, a single hot leg and two cold legs. The reactor coolant pumps are mounted on each of the cold legs, with a single pressurizer connected to one of the hot legs. The primary circuit is arranged in a relatively symmetrical manner. For natural recirculation purposes the SGs are placed at higher elevation. The increased size and coolant inventory of the pressurizer and steam generators have resulted in higher design margins.



Primary systems of AP1400

Safety

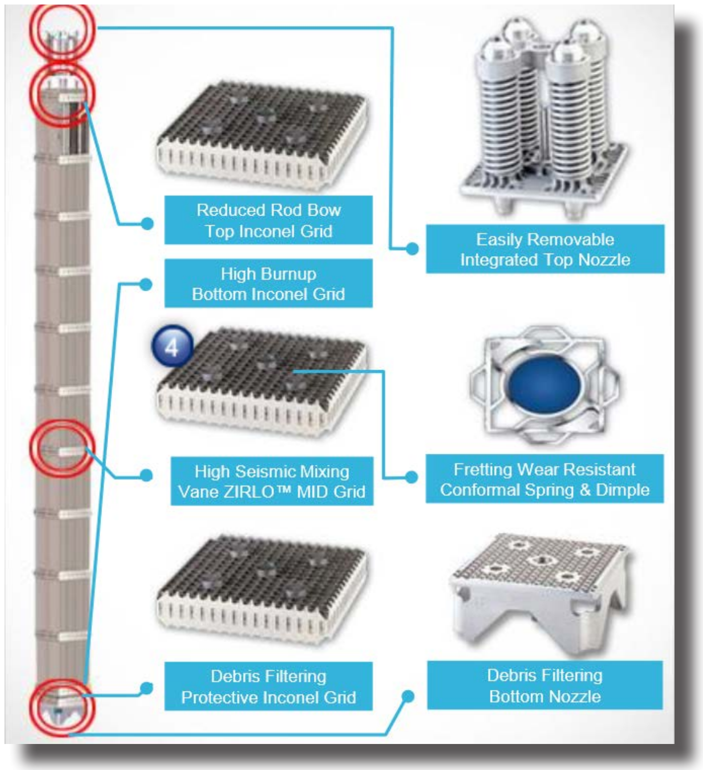
Active, passive and inherent safety features were combined to satisfy the design of safe operational and accident mitigation systems for the APR1400. A core damage frequency of 10^{-5} and a seismic design of 0.3g are quoted by the reactor designers. In the unlikely case of an accident there are several severe event mitigation systems that which are designed to prevent core melt and radiation releases.

The Direct Vessel Injection (DVI) supplies the core with emergency cooling water from dedicated water tanks in the case of a LOCA. Pressure buildup in the containment resulting from transients can be relieved through the Safety Depressurization System (SDS) of the primary without releasing significant amount of radioactivity to the environment. In the case of a severe accident with core melt, the vessel cooling and cavity flooding systems aid in-vessel retention of the corium. The Safety Injection is simpler than the previous OPR1000 SIS designs and has four trains of mechanical equipment with 2 independent electrical trains for redundancy purposes. Fluidic Devices (FD) are installed in the SIS, enabling a passive flow control mechanism for flowrate control during LBLOCAs. Hydrogen Mitigation Systems are installed to control the hydrogen buildup inside the containment during a sever accident.

Construction at Shin Kori-3 Republic of Korea

APR1400

LWR	Coolant Light Water	Moderator Light Water
Capacity 1400 MWe	Temperature 307.2 °C	Fuel LEU
Capacity 3983 MWt	Pressure 15.5 MPa	Enrichment 4.09 %
Design Life 60 Yrs	Efficiency 35.1 %	Fuel Cycle 18 Mos



APR1400 fuel assembly structure (KEPCO)

Fuel

Initially, the APR1400 fuel and core design was specifically tailored to the high load-following demands of the South Korean grid, but it has been adapted for diverse fuel management strategies for the global export business. The standard fuel for the core is UO_2 with an average fuel cycle length of 18 months. Burnable poisons in the form of gadolinium can be used as part of an improved fuel management strategy. The average BU of the reactor is 55GWd/t but can reach a maximum of 60GWd/t under optimal conditions. The APR1400 uses Westinghouse PLUS7 fuel assemblies which have demonstrated enhanced thermal hydraulic, nuclear performance and structural integrity during operation in other PWR reactors. The 10% thermal margin for the fuel enhances safety and operational performance of the plant. Operation of the reactor with a core consisting of one third MOX fuel is also possible with some minor modifications.

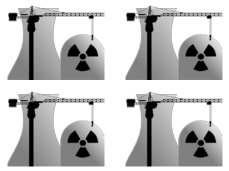
AP1400 Main Control Room (KEPCO)

Operation

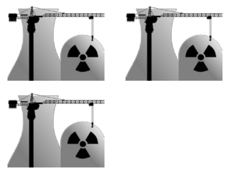
The APR1400 is designed to accommodate utility needs for daily load-following purposes. The large volume of the primary and secondary components alleviates the progression and consequences of operational transients and a reduced reactor trip mechanism lowers the likelihood of non-safety related reactor trips. These mechanisms predict relative higher plant availability, than currently operating PWRs, without endangering the safe operation of the plant. The plant availability for an APR1400 is projected to be around 90%. An Integral Head Assembly combines a number of separate components used for refuelling in conventional PWRs into a single structure. This and the installation of in-core instrumentation cable trays help to reduce refuelling outage lengths as well as radiation exposure of workers during refuelling related activities. In addition, a one-piece reactor head reduces the need for in-service inspections over the lifetime of the reactor.

- The APR1400 is currently going through design licensing review by the US Nuclear Regulatory Commission.
- In 2007 the Korean government launched designs for the APR+, which will generate 1500MWe. Its overall design is very similar to the APR1400, and envisions a shorter construction time than the APR1400, while accommodating stricter safety standards and a larger core power output. The APR+ also features a double containment and a core-catcher.

Shin-Hanul 1, 2
Shin-Kori 3, 4

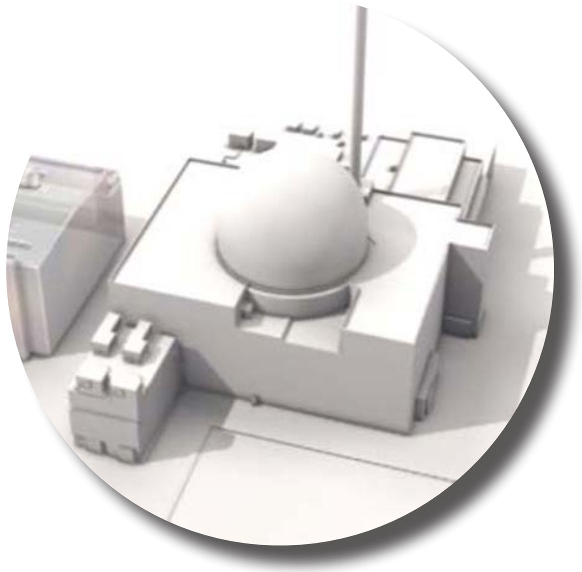


Barakah 1, 2, 3



ATMEA1™

ATMEA™ is a joint venture of AREVA and Mitsubishi Heavy Industries (MHI) with the goal to design, market and deploy an efficient and safe mid-range PWR on the world-market. Using innovative and proven nuclear technologies from the well-developed nuclear power sectors in France and Japan, ATMEA1TM accords with compliance across a broad set of regulatory and commercial requirements worldwide. This design incorporates top-level 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.



ATMEA1 plant layout
(Courtesy of ATMEA)

Nuclear Systems

The primary system of the ATMEA1 is mix between AREVA's and MHI's previous 2-loop primary designs (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 steam generator 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 3150 MWth with a projected net electricity production of 1100 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 0.3 g of ground seismic activity and external and internal hazards, such as fires, aircraft crash and floodings. The design of the components and safety systems reduce core damage frequencies to around 10^{-6} /RY.

Primary Configuration of ATMEA1
(Courtesy of ATMEA)

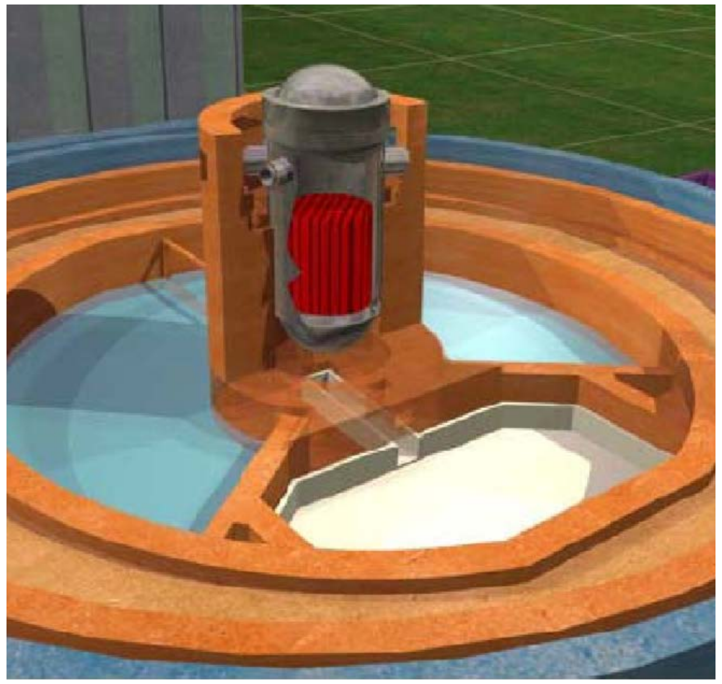
Safety

Safety systems of the ATMEA1TM are designed to meet current national and international sets of regulatory requirements including the US-NRC, France's ASN, Japan's NRA and Euroatom. The safety and design of the ATMEA1 is based on deterministic analyses of defence-in-depth 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 safety injection and LPIS. 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.

ATMEA1 is partly based on the Japanese PWR Tomari 3

ATMEA1™

LWR	Coolant Light Water	Moderator Light Water
Capacity 1150 MWe	Temperature 308.5 °C	Fuel LEU
Capacity 3150 MWt	Pressure 15.5 MPa	Enrichment <5 %
Design Life 60 Yrs	Efficiency 36 %	Fuel Cycle 24 Mos



ATMEA1 Reactor Pressure Vessel and Core Catcher (Courtesy of ATMEA)

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 % U^{235}) 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%.

Operation

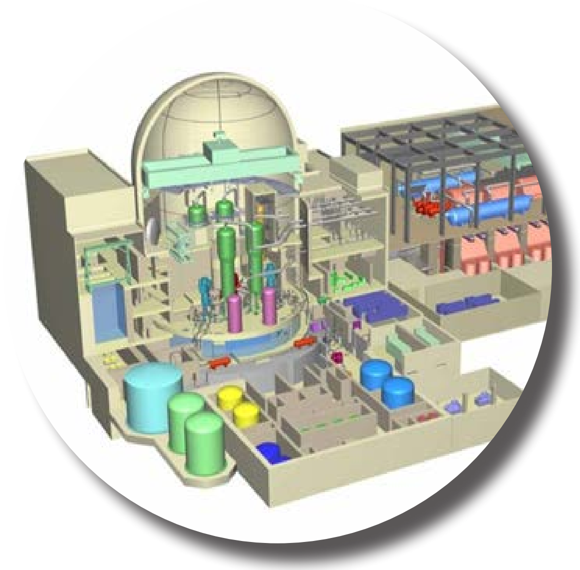
Digital instrumentation and control systems are used extensively in the operation of the ATMEA1 to improve the human-system interface with the goal to reduce the likelihood of operator errors. By encompassing the sources of human error in the design, testing and maintenance of the I&C systems, the physical demands on the operators are lowered and permissible response times for critical operator actions are increased. Additionally top mounted I&C control systems on the vessel give valuable information about the operations of the core at all times. 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. Furthermore they have be 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. Furthermore the maximum collective doses exposure for average radiation work is set to less than 0.5 man-Sv/y.

- Currently undergoing licensing in various countries
- Adaptability to different grid requirements (50 or 60Hz)
- Mid-rage power output make this reactor adaptable to countries with smaller or less developed electric grids
- AREVA and MHI construction expertise and combination of their supply chain
- SAMS: core catcher, cooling systems and hydrogen re-combiners for long-term containment integrity
- Provisions for off site emergency equipment utilization

N4 Type reactors - Chooz B 1, 2 France [1]

Advanced PWR

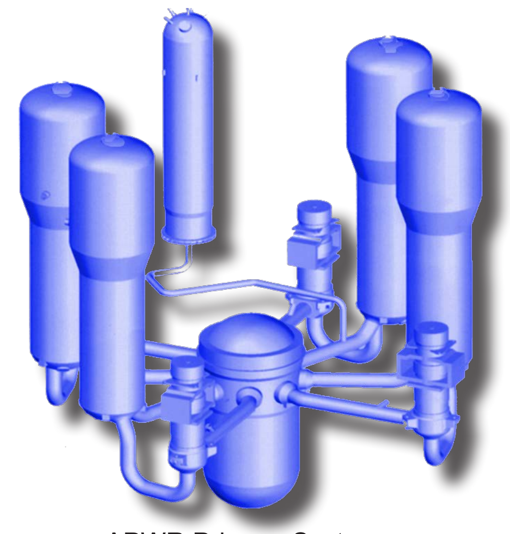
Mitsubishi and five national electric companies involved in PWR electricity generation, came together in Japan during the 1990s to design and develop a large PWR. Their main vision for the new reactor design was to make electricity production safer, more reliable and economical. The companies combined their vast operational experience with the design and construction knowledge of Mitsubishi, involved in 26 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 version. The reactor comes in three version: standard JP-APWR, US-APWR, and EU-APWR, each uniquely envisioned by the designers to fit country-specific regulatory and utility requirements. Generating capacities for this family of large PWRs range from 1538 MWe to 1700 MWe. State- of-the-art technology from the Mitsubishi's own research facilities were used to establish high operational safety margins, provide flexible fuel options and protect the environment by providing diverse and redundant plant protection systems.



APWR plant layout (Courtesy of MHI)

Nuclear Systems

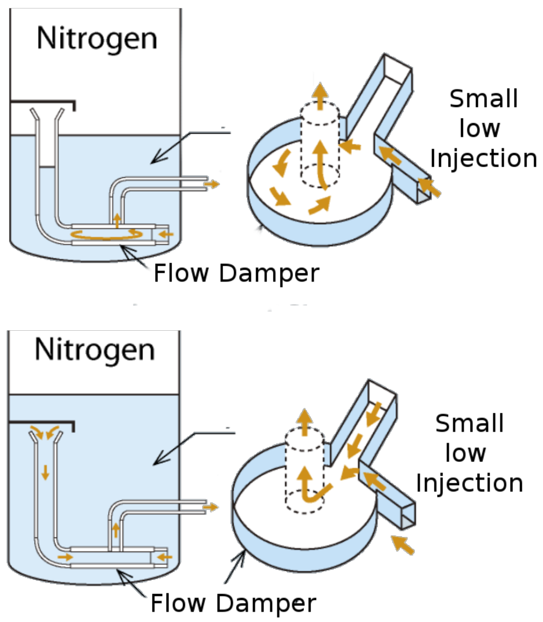
The nuclear systems of the APWR are largely based on the existing fleet of PWRs in Japan but with increased core capacity and modernized digital safety related systems. The primary is a traditional 4-loop PWR design with 4 steam generators and 4 reactor coolant pumps between the loops. Each loop consists of one hot leg, one cold leg and one crossover leg. A single pressurizer controls the pressure of the primary circuit which has a thermal capacity of 4451 MWt. Neutron reflectors are added for neutron economy as well as lowering vessel fluence to endure a design life of 60 years. Innovative steam generators with increased capacity and Inconel 690 tubes for greater corrosion resistance are used in the APWR. Portable pump connections are available for injection of water into the core during severe accident situations and a core damage frequency of 10^{-7} /RY is quoted by the designers. Additionally, nuclear systems are designed withstand ground acceleration limits of 0.3 g.



APWR Primary Systems

Safety

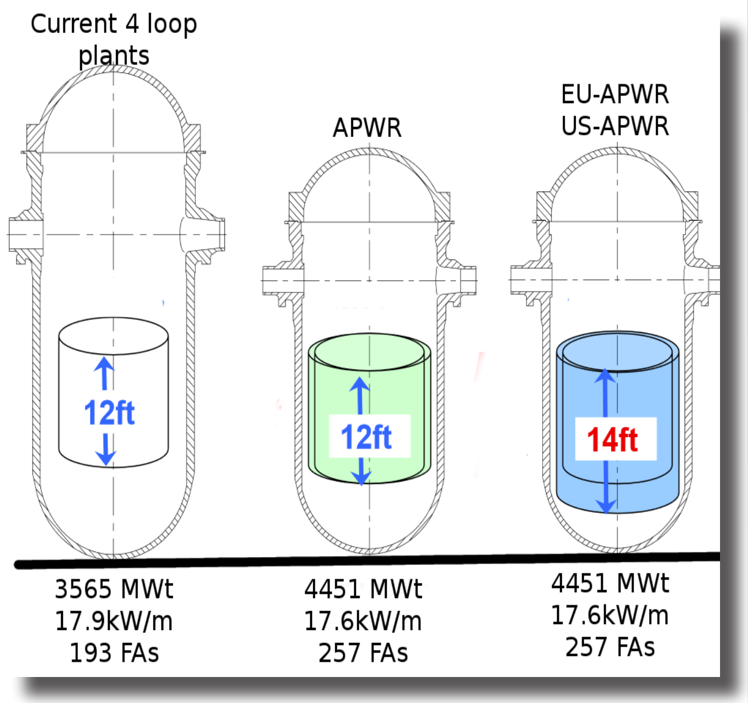
The APWR uses a mix of active, passive and inherently safe systems for normal operation and emergency situations. The three different versions of the APWR have different safety, diversity and redundancy features that are specific to regulatory or commercial requirements. All versions of the APWR comes with a 4 train mechanical Emergency Core Cooling System with at least 2 independent electrical trains. The function of the low pressure injection system and accumulators have been combined into the passive high-performance accumulator tank that serves as a conventional accumulator tank and low-pressure injection tank to streamline the equipment. The Containment Vessel Air Recirculation System, hydrogen ignitors, autocatalytic recombiners, filtered containment vents, and alternate containment vent sprays are designed to mitigate the effects of severe accidents or containment overpressurization. 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.



Advanced accumulator flow mechanism (MHI)

APWR

LWR	Coolant	Moderator
	Light Water	Light Water
Capacity 1540 MWe	Temperature 307 °C	Fuel LEU
Capacity 4466 MWt	Pressure 15.4 MPa	Enrichment 4.5 %
Design Life 60 Yrs	Efficiency 34.4 %	Fuel Cycle 24 Mos



Comparison of core size

Fuel

The standard fuel used in the APWR is the same fuel as in currently operating in Japanese PWRs, thus capitalizing on years of previous operating experience and design improvements. It consists of sintered Uranium Oxide (UO2) pellet with low enriched uranium (max of 5 % of U235). The addition of heavy neutron reflectors in the core reduces the uranium requirements. The EU/US-APWR version has a slighter longer fuel assembly to achieve lower power densities, and a 24 month fuel cycle to enhanced fuel economy with the addition of larger design margin for improved safety. Longer fuel cycles combined with automatic and high speed fuel handling systems shorten the length of refuelling outages. The anticipated average BU is 55 GWd/t with a maximum of 62 GWd/t, but there are plans to increase this value to 70 GWd/t. It is also possible to use one third MOX cores without any major modifications to the nuclear systems.

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 large APWR is predicted to produce electricity for about a third less cost than current Japanese plants and have 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 a easily viewed manner to reduce interpretation and communication errors. Operators' work load 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.

		APWR	US-APWR	EU-APWR
Electric output		1538 MWe	1700 MWe	1700 MWe
Fuel assemblies		257 x 3.65m	257 x 3.65m	257 x 3.65m
SG heat transfer area		6500m²	8500m²	8500m²
Safety sys-tems	Electrical	2 trains	4 trains	4 trains
	Mechanical	4 trains	4 trains	4 trains
Emergency power		Diesel generator	Diesel generator	Gas turbine generator
BDB		-	Alternate AC	ATWS, multiple SGTR, MSLB + SGTR, SBO

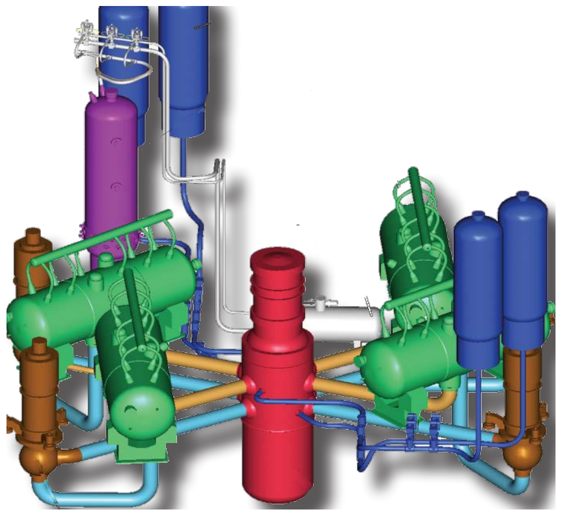
Water Water Energetic Reactor (VVER)

Rosatom subsidiary OKB Gidropress is the designer of a class of Russian Pressurized Water Reactor called VVERs. The plants housing the VVER reactors are created by the design organizations within ROSATOM: Moscow Atom-energoproekt, Saint-Petersburg Atomenergoproekt, and Nizhniy Novgorod Atomenergoproekt. Over 60 VVER units have been constructed by Russian companies across the globe since the mid-1960s, exporting a nuclear technology that has been domestically tested and proven. Modern VVER units are designed to face the current challenges of the nuclear industry by reducing capital costs, shortening construction periods, increasing efficiency and load factors, as well as implementing load-following capabilities. Furthermore the designers try to implement the effective use of VVERs in closed fuel cycles. Currently available large, PWR type models of the VVERs include the 1000 MWe AES-92 and the 1200 MWe AES-2006. Each reactor model can be fitted with components specific to regulatory or utility needs.

Artist's version of Novovoronezh NPP II, Russia, housing a VVER-1200 unit. (Courtesy of Rosatom)

Main Design Features

A number of unique concepts and construction materials are incorporated in the VVER designs. Plants containing VVER reactors employ horizontal steam generators with relatively thick walled steam generator tubes made of stainless steel. Steam generators in Russian plants have been successfully operating for over 40 years without the need for replacement. In addition, these type reactors utilize hexagonal fuel assemblies with a fuel rod cladding of zirconium-niobium alloy. During the design of some large components, efforts have been made to ensure transportability of the main equipment by rail. VVER reactor vessels 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 in less frequent maintenance and surveillance tests on the RPV.



Schematic view of VVER1000 Primary (Courtesy of Rosatom)

Main Safety Features

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 defence-in-depth 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.

Construction of VVER1200 Novovoronezh NPP II, Russia [2]

VVER-1000 (AES-91 & AES-92)

The early VVER-1000 is a 1000MWe PWR design produced during the 1980's. It was one of the first designs to utilize natural forces for passive safety systems and reactor protection. The VVER1000 type reactors were based on the older VVER440 designs, drawing from years of operating history. Proven design features from the smaller plants were scaled up to fit the larger requirements associated with the higher thermal output. The primary systems of these big reactors were based on a conventional 4-loop design with improved horizontal steam generator designs. The use of new cobalt-free materials with lower neutron activation led to significant reduction of exposure rates for radiation workers. Modern VVER1000s are equipped with passive ECCS and HRS for reactor protection.

The VVER1000 reactor is designed to operate at 33.7% efficiency with a 90% availability factor with a design life of 50 years. There are three VVER1000 reactors currently being constructed around the world at two different sites. The first model called the AES91 (VVER-1000/V-428M) complies with the European Safety Standards, as it was originally designed for construction in Finland. Two units are currently under construction in Tianwan, China. The other model is called the AES92 (VVER-1000/V412) was designed for India. Currently there is one AES92 unit in operation in Kudankulam, India and one unit still under construction.

LWR	Coolant Light Water	Moderator Light Water
Capacity 1060 MWe	Temperature 305 °C	Fuel LEU
Capacity 3000 MWt	Pressure 15.7 Mpa	Enrichment 4.55 %
Design Life 60 Yrs	Efficiency 33.7 %	Fuel Cycle 18 Mos

V-412
Kudankulam 1, 2

V-428M
Tianwan 3, 4

VVER-1200 / AES-2006

In 2006 Rosatom finalized the designs for the VVER1200/AES-2006, 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 European Utility Requirements. 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 1170 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 BU of 60-70 GWd/t. Modular construction techniques have optimized predicted construction times to around 54 months. In addition to the safety feature of the AES91/92 the AES-2006 has passive HRS for the containment and steam generators in the unlikely case of BDBA allowing independence and safe cooling for 72 hours without external power supply.

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.

LWR	Coolant Light Water	Moderator Light Water
Capacity 1170 MWe	Temperature 313.5 °C	Fuel LEU
Capacity 3200 MWt	Pressure 16.2 Mpa	Enrichment 4.79 %
Design Life 60 Yrs	Efficiency 33.9 %	Fuel Cycle 18 Mos

V-392M
Novovoronezh II 1, 2

V-491
Leningrad II 1, 2
Baltic 1

VVER

VVER1500

Rosatom's most recent design has benefited greatly from the worldwide successful construction of their VVER1000 and VVER1200 series. 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 1560MWe. 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.

LWR	Coolant Light Water	Moderator Light Water
Capacity 1560 MWe	Temperature 316 °C	Fuel LEU
Capacity 4250 MWt	Pressure 15.7 Mpa	Enrichment 4.92 %
Design Life 50 Yrs	Efficiency 35.7 %	Fuel Cycle 24 Mos

Kudankulam 1, 2 - India
AES92 (VVER-1000/V412)

Novovoronezh II 1 - Russia
AES-2006 (VVER1200/V392M) [2]

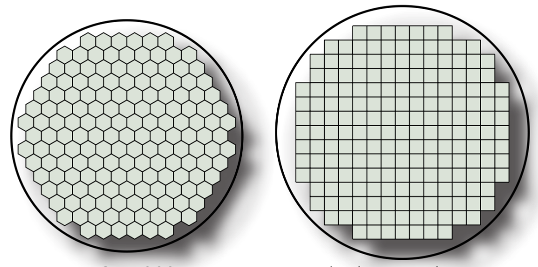
Tianwan 3, 4 - China
AES91 (VVER-1000/V-428M) [3]

	VVER1000	VVER1200	VVER1500
Thermal Output (MWt)	3000	3200	4250
Electric Output (MWe)	1070	1200	1560
Design Life (yrs)	50	60	50
Plant Efficiency	33.7%	33.9%	35.7%
Availability Factor	>90%	>92%	>93%
Construction Time (mos)	46	54	54
Operator Action Time	6 hrs	6 hrs	>6 hrs
Load Following	Yes	Yes	Yes
Occupational Radiation Exposure (Sv/RY)	0.5	0.39	TBD
CDF (/RY)	0.6×10^{-6}	$<10^{-6}$	$<10^{-6}$

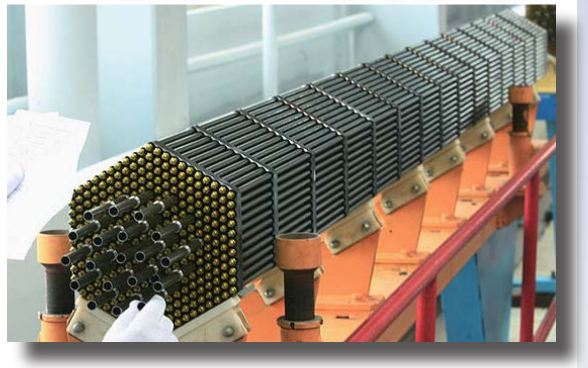
Fuel

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, power output of the reactor, operational modes and anticipated fuel burnup.

VVERs utilize enriched sintered UO2 fuel pellets fitted into fuel rods made from zirconium-alloy cladding. These fuel rods are bundled together into hexagonal-shaped fuel assemblies having an enrichment of up to 4.95 wt% U235. 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 all 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. Slight modifications are introduced into the design of PWR fuel assemblies to facilitate 3, 4, or 5 years of fuel life.



ВВЭР-1000 Westinghouse 4-loop PWR



Hexagonal VVER fuel assemblies. [4] [5]

	VVER1000	VVER1200	VVER1500
Fuel cycle length (mos)	18	12-18	12-24
Avg. reload enrichment (wt%)	4.45%	4.79%	4.92%
Avg. BU (GWd/t)	53	60	57.2
Avg. core height (m)	3.53	3.75	4.2
Core diameter (m)	3.16	3.16	3.85
Avg. core power density (MW/m³)	108	108.5	-
Avg. fuel power density (KW/kgU)	35.8	36.8	-

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 1200 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.

VVER Control Room
Novovoronezh NPP, Russia [6]

Advanced Passive 1000

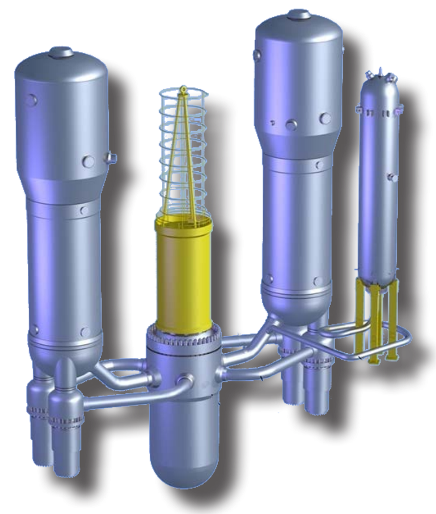
Toshiba - Westinghouse

Toshiba's AP1000 is a 1100 MWe PWR designed with emphasis on passive safety systems. These systems depend on natural forces such as gravity or natural circulation to carry out safety related functions that facilitate the safe operation of the reactor. 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. Also incorporated into the design are unique containment isolation and cooling features to prevent or limit the escape of fission products that may result from potential accidents.

Construction Site Sanmen 1, 2 - China

Nuclear Systems

The AP1000 is a 1100MWe class conventional PWR having a 2-loop design. Each loop consists of a single hot leg, two cold legs, a steam generator and two reactor coolant pumps that are directly connected to the bottom of each steam generator. The elimination of a crossover leg between steam generator 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.



Primary Configuration of AP1000

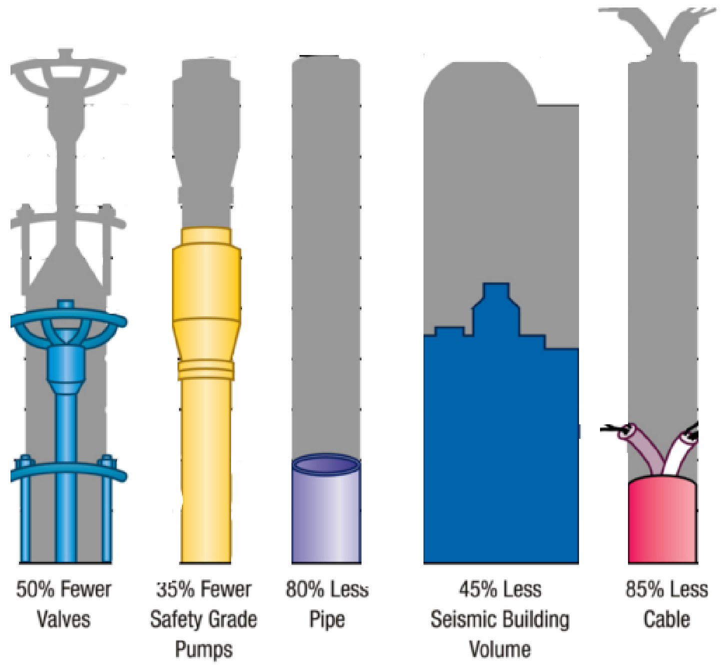
Safety

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 SA scenario, 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. Moreover the placement of penetration above the reactor core reduces the likelihood of extended core uncovering in LOCA-type accidents, which are additionally mitigated by the "leak-before-break principle" for primary piping. The AP1000 is designed in accordance with the US 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 1×10^{-7} /RY.

First Concrete at VC Summer 2 - USA [7]

AP1000™

LWR	Coolant: Light Water	Moderator: Light Water
Capacity: 1200 MWe	Temperature: 303 °C	Fuel: LEU
Capacity: 3200 MWt	Pressure: 15.5 MPa	Enrichment: 4.8 %
Design Life: 60 Yrs	Efficiency: 32 %	Fuel Cycle: 18 Mos



Comparison between AP1000 and existing Westinghouse Plants (Courtesy of Toshiba)

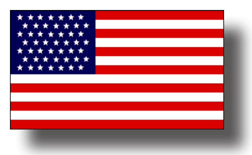
Fuel

The AP1000 is primarily designed to use enriched Uranium Dioxide (UO^2) fuel with an 18- or 16-20-month alternating fuel cycle length. Following optimal conditions, a refuelling outage can be kept to 17 days. A standard reactor core can have fuel rods of varying enrichments from 2.3 to 4.8 % of U235. Integral fuel burnable absorbers (IFBA) 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 BU of 60GWd/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 ROBUST™ design, which have a number of previously proven design features and have undergone extensive testing in dedicated facilities. The fuel is contained in ZIRLO™ tubing with large gas space to accommodate increased fission gas production for higher burnups.

Operation

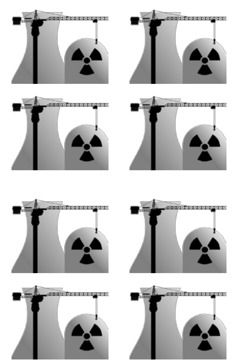
Operation and maintenance were important aspects that were considered during the design of the AP1000. A high degree of reliability with low maintenance requirement can result in a high availability factor of ~93% (planned and forced maintenance included). The increased operational margins help reduce 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.

- 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% life time availability.
- Reduced radiation exposure, less radioactive waste.



VC Summer 2, 3
Vogtle 3, 4

Sanmen 1, 2
Haiyang 1, 2



KERENA™

AREVA and major European utilities including E.ON, Vattenfall, EDF and TVO developed an advanced BWR called SWR1000 in 2008 but renamed to KERENA™ in 2009. Through the cooperation with multinational utility companies, system simplifications and the use of previously proven design concepts, risks associated with foreign licensing and construction can be further reduced. The design and the developing of the plant aimed at producing an economically competitive plant for the future.

KERENA combines years of experience in reactor operation and design 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.

Illustration of KERENA plant layout.

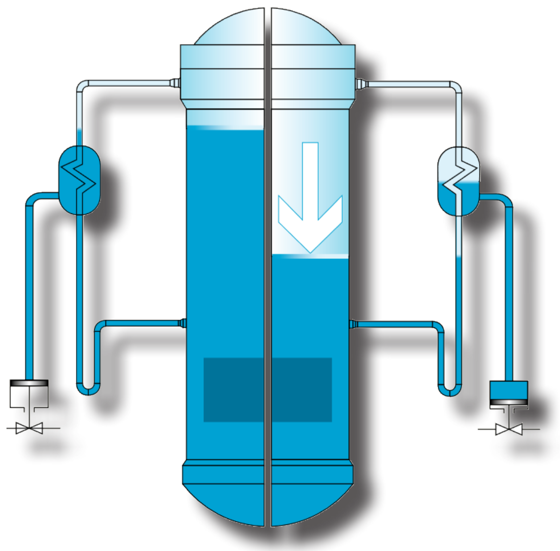
Nuclear Systems

The nuclear systems of KERENA are based largely on the proven designs of AREVA's 1300 MWe BWR plants, with the exception of components belonging to passive safety systems. The core produces 3370 MWt of thermal energy which is converted at 37% efficiency to 1250 MW. The coolant is recirculated by 8 reactor recirculation-pumps inside the RPV. Steam from the core passes through moister separators and steam dryers before entering the high pressure-tubines via three main steam lines. Core thermal power is regulated through reactivity feedback by changing the flow rate of the RR-Pumps or by the hydraulic control rod scram systems through the bottom of the RPV. The reduced active core height and large volume of coolant above the fuel reduce the probability of uncovering the core during LOCA-type scenarios.

KARENA design is based on German BWRs Gundremmingen 2,3, Germany [8]

Safety

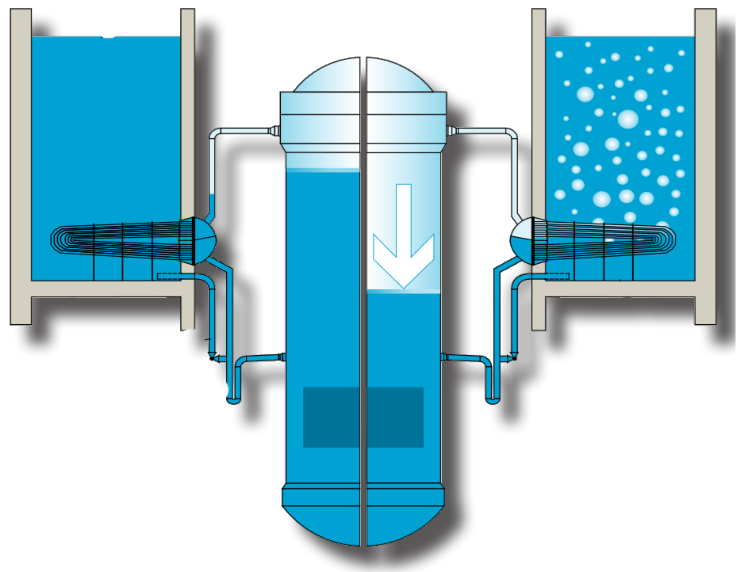
Redundancy and diversity of integral safety systems in the KERENA reactor are provided by two active and four passive qualified safety systems. An innovative approach of partially replacing active safety systems with passive safety systems is used to lower the calculated probabilities of accidents and radiological consequences of accidents due to failure of I&C systems or human errors. Active systems employed for reactor protection are manually activated whereas the passive systems act as backups. In many accident scenarios, passive systems provide an increased time frame for operators to choose intervention procedures. Passive systems such as passive pressure pulse transmitters (PPPT) monitor the reactor water levels and compare with the secondary pressure build-up to ensure operation within safety margins. 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 all help to preserve the RPV and containment boundaries and to prevent the release of harmful radioisotopes to the environment.



PPPTs: Left, normal operation. Right, upon pressure build up in reactor system with Automatic Depressurization (AD) initiated.

KERENA™

BWR	Coolant: Light Water	Moderator: Light Water
Capacity: 1250 MWe	Temperature: 320 °C	Fuel: LEU
Capacity: 3370 MWt	Pressure: 15.5 MPa	Enrichment: 4.5 %
Design Life: 60 Yrs	Efficiency: 36 %	Fuel Cycle: 24 Mos



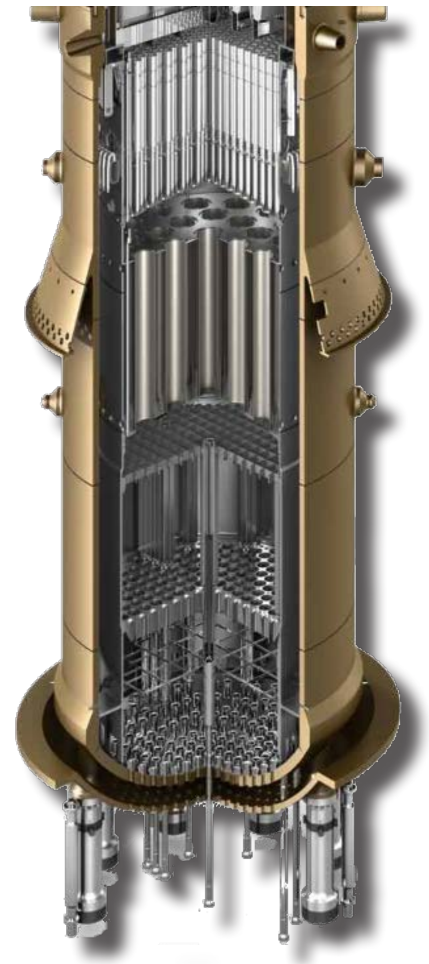
Emergency condensers: Left, during normal operation. Right, acting as ECCS when RPV water levels drops. (AREVA)

Fuel

AREVA plans to use modified ATRIUM™10 for KERENA at low enrichments of UO2 (4.5 - 4.7 % of U235). The new fuel called ATRIUM™12 would result in structurally larger, but fewer number of fuel assemblies in the core than in current BWR designs. Reducing the number of fuel assemblies shortens the reactor's standard postulated refuelling outages to about 11 days. The innovative core design and fuel management strategy promises more fuel efficiency and a reduction of long-life radioisotope products of about 15%. Through appropriate fuel management an average discharge burnup of about 65 GWd/t can be expected. The flexiabel 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 to systems and components. The spent fuel assemblies are stored in the spent fuel pools, which contain residual heat removal systems and shielding, inside the reactor building.

Operation

Through streamlined design solutions, low maintenance requirements and optimized fuel management strategies, the plant's availability is trageted to be above 92%. KERENA load following capabilities (between 40-100% of full power) and frequency control make it 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. Large water volume in the primary of reactor helps transient control during operation and avoids unnecessary reactor trips. Thorough significant simplification of system engineering, standardization and reduction of components, there are less periodic maintenance procedure to be carried out. These actions improve the availability of the plant and the safety of the workers as well as minimizing the occupational dose to radiological workers.



Artists rendition of KERENA's RPV and internals design (AREVA)

- 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
- Minimal construction time through modular construction methods
- Resistance to plane crashes and seismic vibration
- Severe Accident Mitigation Systems
- Load following (40% and 100% of its nominal power)
- Enriched Uranium Fuels, Reprocessed Uranium or MOX Fuel can be used

Advanced Boiling Water Reactor - ABWR

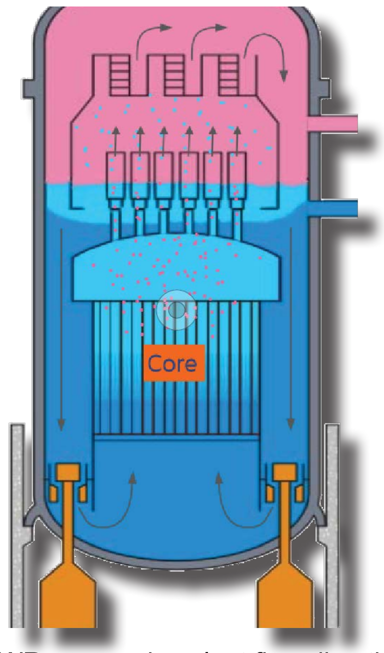
GE - Hitachi | Toshiba

GE-Hitachi's ABWR was designed to be an economically viable method of electric production with higher standards of nuclear safety than current BWRs. The original was completed in 1991 by General Electric and its technical partners, Toshiba and Hitachi Ltd. with important contributions from utilities from Japan and US. This plant, with a design life of 60 years, comes in three different versions tailored to regulator and utility needs of the countries or regions in which they are envisioned to be deployed. Currently available versions are JP-ABWR, US-ABWR and EU-ABWR. The term ABWR now refers collectively to the most advanced version of boiling water reactor designs, whether the parent company is GE-Hitachi, Hitach-GE or Toshiba. Modularization techniques and equipment simplification, without compromising the safety of the plant, enable shorter construction time and reduced up-front capital costs. Load following capabilities and smart fuel managing strategies are intended to help provide a competitive electricity genration option. In 2015 there were 4 operational ABWR's and 4 are under construction.

Kashiwazaki-Kariwa Nuclear Power Station, ABWR. (Kashiwazaki-Kariwa, Japan)

Nuclear Systems

The NSSS of the ABWR uses a number of innovative design improvements that have been developed through monitoring the operation of over 90 BWRs around the world. The ABWR uses 10 reactor internal pumps to circulate 3926 MWt of heat from the fuel through 4 main steam lines to the turbines to produce 1420 MW of electricity at an efficiency of 34.4%. The elimination of recirculation pumps and piping in the primary, especially safety related RPV penetrations below the top of core, has helped to increase the overall safety of the reactor system, which has a CDF of 1. 6 $\times 10^{-7}$. Reactivity control is maintained by a combination of control rods, RIP flow rates and the amount of burnable poisons present in fuel. Simplification of components have made the reinforced concrete containment vessel of the ABWR significantly smaller than current MARK-Type BWR containments. Post-Fukushima adjustments have introduced multiple water injection connection points for portable pumps to the core if ECCS becomes unavailable.



ABWR core and coolant flow directions (Courtesy of GE Hitachi)

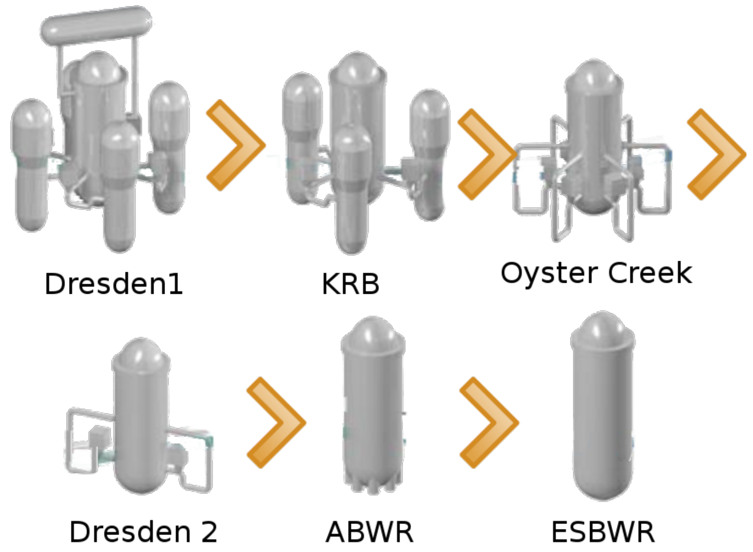
Safety

ABWR uses active, passive and inherent features for reactor protection. Country-specific versions of the plant vary in the type of safety systems, their redundancy/diversity or capacities in accordance to the regulatory requirements. Large volume of coolant in the primary alleviates or delays reactor pressurization during transients. Passive wetwell venting systems and HMS are in place to protect containment integrity during postulated SA. Additional EDG and gas-turbine generators act as backup power for important systems during SBO initiated by extreme external events such as earthquakes, floods or storms. Many systems, such as RHR, dual functions, both for shutdown cooling and for core/containment cooling during postulated LOCAs. ECCS capabilities can also be reduced due to the elimination of recirculation in the primary loop. Three fold diversity and separation for CRDM protect against ATWS type accidents. Post-Fukushima adjustments have resulted in the incorporation of additional external connections to diversify emergency water injection to the core and spent fuel pool. A core catcher is envisioned on the floor of the lower drywell in the RCCV of the next Toshiba variant of ABWR to help mitigate severe accidents involving core melts.

Construction of Lungmen NPP- Taiwan, China

ABWR

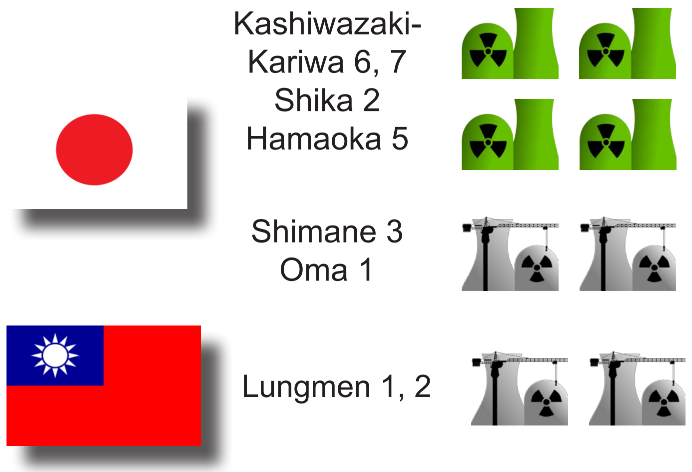
BWR	Coolant: Light Water	Moderator: Light Water
Capacity: 1420 MWe	Temperature: 283 °C	Fuel: LEU
Capacity: 3926 MWt	Pressure: 7.07 MPa	Enrichment: 4 %
Design Life: 60 Yrs	Efficiency: 34.4 %	Fuel Cycle: 24 Mos



Historical evolution of RPV in BWRs

Fuel

The ABWR uses sintered UO2 as fuel at an average enrichment of 4 wt% of U235. 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.



Operation

A 60 year design life is predicted by ABWR designers, accounting for continued functions like startups, shutdowns, automatic responses to load changes and minor transients during normal operation. Primary control of the power plant remains within the control of the operator who monitors the status of individual systems as well as the automation sequences instead of controlling individual equipment.

Load-following capabilities of 70-100% of full power are provided through coolant flow of RIP pumps. This, combined with lower O&M costs through reduction and simplifications of systems, contribute to competitive electricity generation. Having fewer total components also contributes to less maintenance activities, thus resulting in a reduction of occupational exposure for radiation workers.

	JP/US - ABWR	EU - ABWR
Electric output	1420 MWe	1600 MWe
Thermal output	3926 MWt	4300 MWt
Reactor protection systems	3 × 50 % Emergency Core Cooling (2 × HP-Core Flooding, 1 × RCIC, 3 × LP-Core Flooding)	3 × 100 % of Emergency Core Cooling (2 × HP-Core Flooding, 1 × RCIC, 3 × LP-Core Flooding)
Emergency power	3*EDG,CTG	3*EDG, 2*CTG, 4 Divisions
Safety features	Core melt spreading, corium shield, PCV vent	Armored containment, divisional separation, H_2-recombiners, core catcher, passive containment cooling

Economic Simplified BWR

GE-Hitachi's latest design is the ESBWR which combines economy of scale, improvements in safety and design simplification to generate electricity in a reliable fashion. The company uses its 50+ years of experience in design, construction and fabrication in the nuclear industry to utilize natural phenomenon and passive features for a safe operation of the BWR. Reduced O&M costs from simplifications leads to cost reductions which, in combination with shorter projected construction time and lower costs, result in an economical power plant. This design has active non-safety systems to handle operational transients, but all reactor safety systems are passive and thus do not require AC electrical power nor operator actions for cooling for more than seven days. The predecessors of the ESBWR are the ABWR and the SBWR, the latter abandoned in the 1990s due to insufficient power output. Most systems and components in the ESBWR have proven operating histories from the ABWR, but some innovative features have been incorporated and tested by the designers in dedicated facilitates.

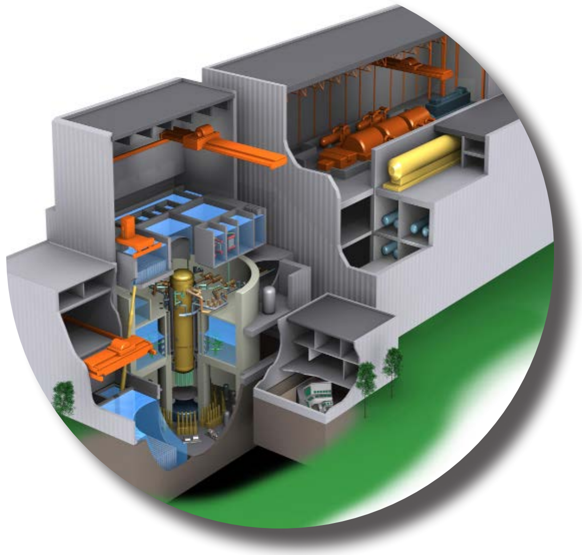


Illustration layout of ESBWR plant layout
(Courtesy of GE Hitachi)

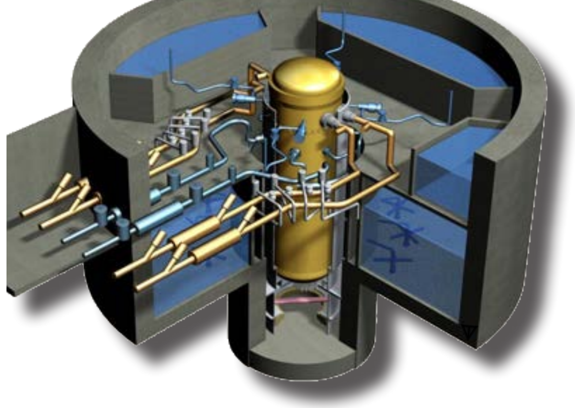
Nuclear Systems

The ESBWR is designed to operate at 35% efficiency, turning 4500 MWt into 1560 MWe. The NSSS of the ESBWR is quite similar to the ABWR which has 4 MSLs to transport the heat from the core to the turbines. The significant difference is that the ESBWR is designed to operate using natural circulation to remove heat out of the core. In order to use natural circulation forces some changes were made to the primary systems, such as an increase in vessel height, installation of a partitioned chimney above the reactor core, a decrease in active fuel height, and a taller, more open down-comer annulus that reduces flow resistance and provides added driving head, pushing water to the bottom of the core. The shorter fuel heights, improved steam separators and tall chimney at the top of the vessel result in reduced pressure drops and higher overall plant efficiency, which translates into greater economy. The taller PV with its relatively larger volume of coolant, combined with the absence of hydraulic instabilities caused by forced circulation, serve to enhance the operational flexibility of the reactor.

Artists rendition of proposed ESBWR
North Anna 3, United States.
(Courtesy of GE Hitachi)

Safety

Some safety systems present in the ESBWR design were previously used as backup systems to active safety. The passive safety systems are designed in a modular system, keeping them independent and physically separated to curtail independent events from affecting the same physical aspects of the plant. Natural circulation eliminates the need for safety grade equipment like pumps and safety DG's to be present inside the containment and thus the size of the containment building can be reduced by about 30%, despite the increase in power output of about 15%. The ESBWR design's core damage frequency of 1.7×10^{-8} per year is considered to be lowest of advanced reactors currently available. The GDCS and ADS act as the plant's Emergency Core Cooling. Isolation condensers are used for HPI and decay heat removal in SBO conditions. LPI by gravity driven cooling systems is connected to three water sources. In the case of a SA the PCCS, Containment- and Core Catcher Cooling System are in place to mitigate the progression of the accident and limit radiological consequences. In accordance with post-Fukushima lessons learned, emergency connections for DG and water makeup to SGs are present as well as HMS.



RPV, Containment Structure and
Safety Systems of the ESBWR
(Courtesy of GE Hitachi)

ESBWR

BWR	Coolant Light Water	Moderator Light Water
Capacity 1520 MWe	Temperature 281.6°C	Fuel LEU
Capacity 4500 MWt	Pressure 7.17 Mpa	Enrichment 4.2 %
Design Life 60 Yrs	Efficiency 34 %	Fuel Cycle 24 Mos

ESBWR design based on experience from ABWRs
ABWR unit Hamaoka - 5, Japan. [9]

Fuel

A key design feature of the ESBWR is its fuel design due to its large core power and innovative flow circulation method. The fuel used is UO2 at an enrichment of 4.2 % of U235. An average BU of 55 GWd/t is predicted with a flexible fuel cycle length of 12-24 months. The fuel assembly design is based on the proven GE14 assemblies, but has shorter active fuel heights to accommodate the use of natural circulation. The Global Fuel Group, a joint venture of GE, Hitachi and Toshiba, are currently working on improving parameters of the ESBWR core and fuel design.

Natural circulation significantly improves key performance parameters of the plant such as fuel efficiency and utilization. There are also a larger number of fuel bundles in the core due to the higher core power and shorter active fuel heights. Larger bypass gaps between the fuel assemblies improve cold shutdown margins and core thermal-hydraulic stabilities, resulting in milder responses from pressure transients.

Operation

The comparatively low plant construction and development costs for the ESBWR with 60 years of design life and an availability factor of over 88% provide an economical way to produce low-carbon electricity. Standardization and simplification of the 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 and reduced maintenance activities decrease the O&M costs. This will in turn reduce the operational exposure of radiation workers performing online maintenance to less than 1 Sv/y. 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 procedure functionality and safe operation of the plan.

	BWR/4	ABWR	ESBWR
Power (MWth | MWE)	3293 | 1098	3926 | 1350	4500 | 1550
Vessel height | diameter (m)	21.9 | 6.4	21.1 | 7.1	27.7 | 7.1
Fuel bundles	764	872	1132
Active fuel height (m)	3.7	3.7	3
Power density (kW/l)	50	51	54
Recirculation pumps	2 (external)	10 (internal)	0
Control rod drive type	Liquid Pressure	Fine Motion	Fine Motion
Safety system pumps	9	18	0
Safety diesel generators	2	3	0
CDF (/RY)	10^{-5}	10^{-7}	10^{-8}

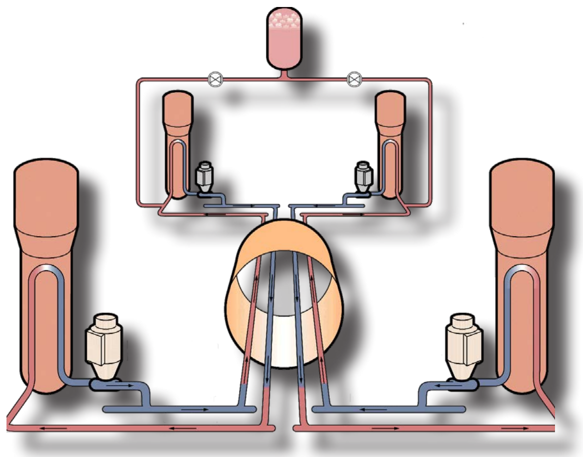
Enhanced CANDU 6

Candu Energy Inc. (previously part of AECL, now a subsidiary of SNC Lavalin, a Canadian engineering company) designs and constructs CANDU reactors. Its most recent design is the Enhanced CANDU 6 (EC6), a 740 MWe heavy water moderated and cooled pressure tube reactor which uses natural uranium as fuel. 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. Furthermore, 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.

Illustration of a Twin EC6 plant layout
(Courtesy of Candu Energy Inc.)

Nuclear Systems

The EC6 is a 740 MWe HWR that is moderated and cooled by heavy water. The fuel is contained in 0.5-m long 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 circulates pressurized heavy water in two interconnected “figure-of-eight” loops through the 380 horizontal fuel channels. It is the same design as in the CANDU 6 reactors, made up of one pressurizer, four steam generators, four heat transport pumps, and four inlet and outlet headers that connect to the pressure tubes via feeder pipes. 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.



Primary components of EC6 HWR

Safety

CANDU 6 reactor plants have 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 EC6’s reactor building provides passive make-up cooling water via gravity feed to the calandria vessel and the calandria vault. The Emergency Heat Removal System is designed to provide an adequate long-term heat sink following an unavailability of the HRS to the steam generators, 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. The above systems and a low-flow containment spray system comprise also the new Severe Accident Recovery and Heat Removal System (SARHRS) in the EC6, 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.

CANDU6 Reactors such as Qinshan NPP, China are predecessors of the EC6
(Courtesy of Candu Energy Inc.)

EC6

HWR	Coolant Heavy Water	Moderator Heavy Water
Capacity 2084 MWe	Temperature 287 °C	Fuel NU
Capacity 740 MWt	Pressure 11.1 MPa	Enrichment 0.7 %
Design Life 60 Yrs	Efficiency 35.5 %	Fuel Cycle Online

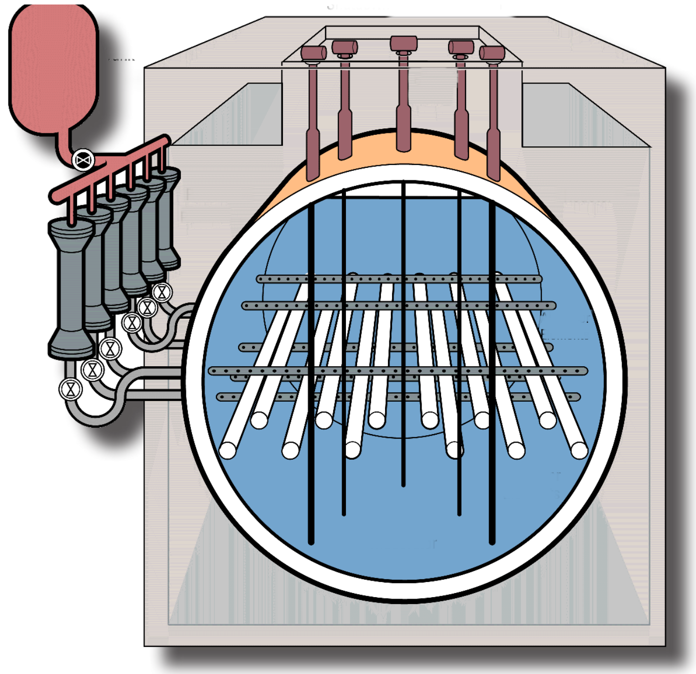


Illustration of vessel and safety shutdown system
(Courtesy of Candu Energy Inc.)

Fuel

The EC6 fuel bundle consists of 37 elements, each approximately 0.5 metres long containing sintered natural uranium oxide (UO2) 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.

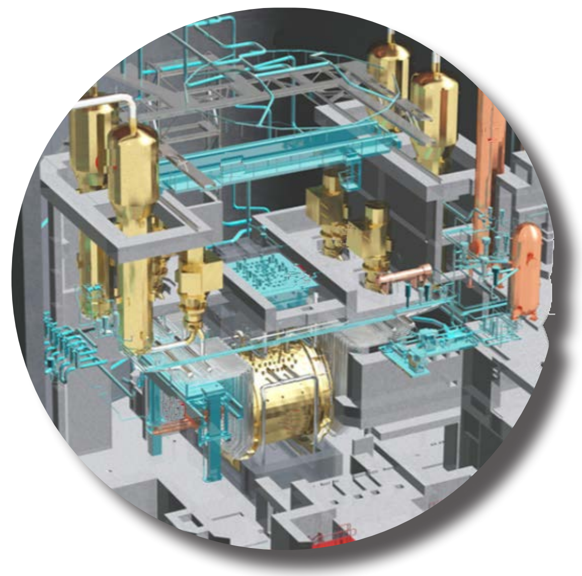
Using NU as fuel permits national fuel cycle independence and technology transfer for localizing fuel manufacture has been achieved successfully in all countries operating CANDU 6 reactors. Several other emerging fuel cycles can also be used, such as Recovered or Slightly-Enriched Uranium (RU/SEU - 0.9 – 1.2% enriched U235 from reprocessed commercial LWR fuel), Natural Uranuim Equivalent (NEU – 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.

37 Element EC6 Fuel Bundle
(Courtesy of Candu Energy Inc.)

Operation

Based on experience from CANDU 6 reactors and design improvements for efficient operation and ease of maintenance, the EC6 reactor 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 once every 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 nuclear power plant 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.



Layout of primary system in CB
(Courtesy of Candu Energy Inc.)

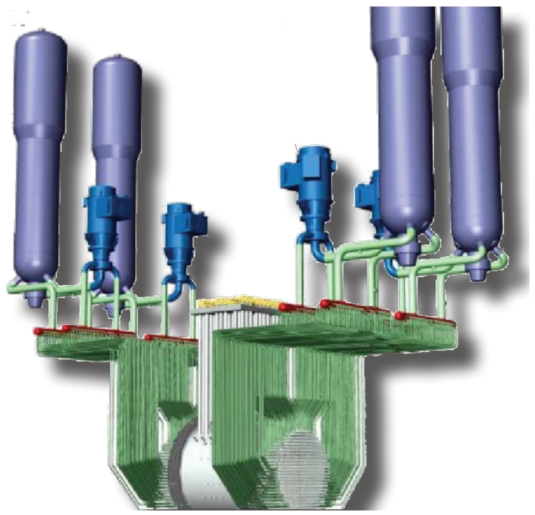
INDIAN PRESSURIZED HWR-700

Nuclear Power Corporation of India Ltd. (NPCIL) is a public sector enterprise under the Department of Atomic Energy of India, responsible for the design, construction, commissioning,and operation of nuclear power plants. Their most recent design is the IPHWR-700, a HWR with horizontal pressure tubes, fuelled by natural uranium and using heavy water as both coolant and moderator. Still under construction it is the largest HWR in a series of Indian PHWR models, following the 220 and 540 MWe version that are currently in operation. Their development began in the early 1980s with the construction of the 220 MWe reactor plant. Increased flexibility of fuel sources and management procedures make the IPHWR-700 an attractive option for India to cope with its growing electricity demand. Its design is fitted with several active and passive safety systems to mitigate the initiation, progression and consequences of postulated accidents.

Rajasthan Atomic Power Station, India [10]

NUCLEAR SYSTEMS

In contrast to LWRs, the high pressure, high temperature heavy water coolant of an HWR is kept separated from the low pressure, low temperature heavy-water moderator. The core produces 2166 MWt of heat which is converted into 700 MWe at an efficiency of 32%.Fuel is situated in an integral horizontal cylindrical calandria with 392 horizontal fuel channels, with some channels reserved for instrumentation. The calandria is surrounded by alight-water-filled concrete vault. The heat transport system consists of 2 loops, each with 2 steam generators and 2 circulation pumps. Due to the arrangement of the core with respect to the steam generators,natural circulation forces can be utilized for passive core cooling under shutdown conditions. The low temperature moderator and light water around the core act as passive safety heat sinks to mitigate severe accident progression. The seismic design of the NSSS is supposed to safely handle ground acceleration up to 0.214 g and the CDF has been reduced to less than 10^{-5}/RY.



Primary Configuration of IPHWR-700
(Courtesy of NPCIL)

SAFETY

The IPHWR-700 systems are designed taking into consideration decades of feedback and experience from operation and construction experience gained with the IPHWR-220 and IPHWR-540 series,a total of 18 reactor units. Two independent methods provide total shutdown ability. The first consists of gravity driven shutdown rods of cadmium and the second method employs the injection of a strong neutron poison (gadolinium nitrate)into tubes in the calandria. Normal control rods and a liquid zone control system are used for routine power changes. The ECCS is built on 2 ×100% redundancy with single failure criterion for each loop and consists of high pressure light water injection, which will absorb fast neutrons. In addition, long term coolant water recirculation is maintained to reduce the likelihood of core damage. Part of the severe accident management in the double containment structure consists of the containment spray system and dedicated connection to the clean-up system. In the case of loss of external power, four emergency diesel generators are available as backup power sources. Additionally there are independent injection connections from diesel driven pumps to the steam generators, reactor vessel and SPF.

Hydrogen recombiner test facility for future improvement for HWR
(Courtesy of NPCIL)

IPHWR-700

HWR	Coolant Heavy Water	Moderator Heavy Water
Capacity 700 MWe	Temperature 288 °C	Fuel NU
Capacity 2166 MWt	Pressure 90 Bar	Enrichment 0.7 %
Design Life 60 Yrs	Efficiency 29 %	Fuel Cycle Online

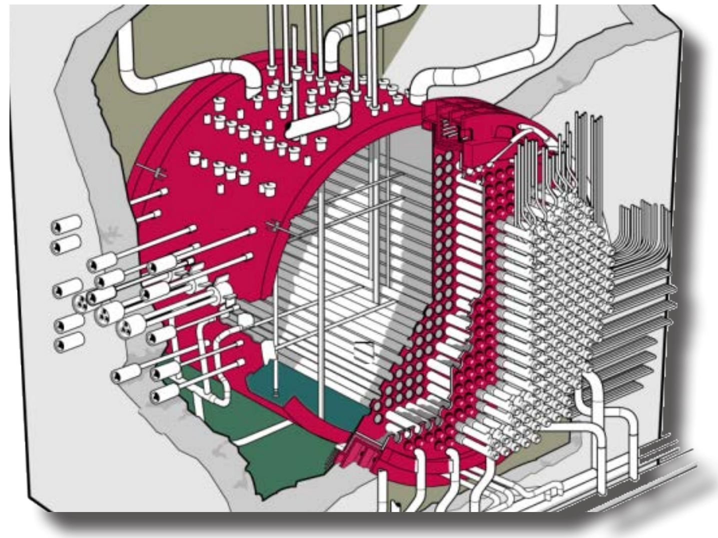
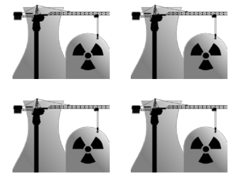


Illustration of Vessel and Safety Shutdown System

Rajasthan 7, 8
Kakrapar 3, 4



FUEL

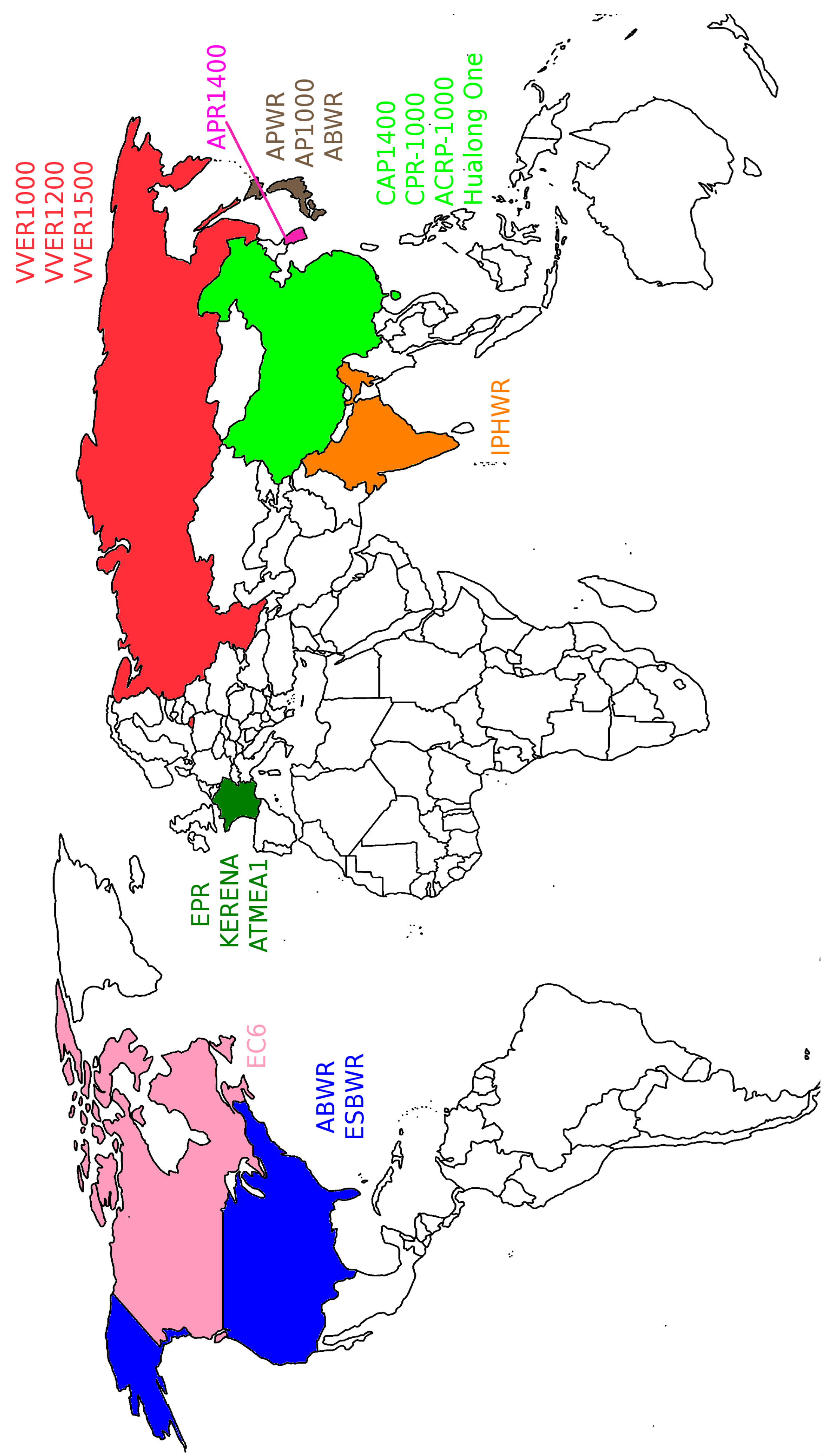
The IPHWR-700 uses natural uranium fuel with a Zircaloy-4 cladding, thus it has very little excessive reactivity and during normal operation there is no need for neutron poison inside the fuel or the moderator. The lack of excess reactivity means that the reactor must be continuously refuelled during operation. Refuelling rates can vary according to the previous full core reload, but in equilibrium operation stages (about 600 days after fresh core load) approximately one fuel channel is reloaded daily. For optimal fuel utilization and to maximize power generation over a long period of time, it is possible to load the core with thorium or depleted uranium bundles, but this complicates fuel management calculations and supply train issues. The average BU using natural uranium is about 7 GWd/t, but can be increased to 15 GWd/t using fuels with higher fissile content such as Slightly Enriched Uranium (SEU), MOX, and ThO2. A maximum BU of nearly 30 GW/t has been achieved in the smaller IPHWR-220 test reactors. The relative short fuel lengths help to mitigate consequences of single bundle failures.

OPERATION

High flexibility in terms of fuel cycle and manufacturing of fuel make the IPHWR-700 an economical reactor without getting into complex uranium transactions. 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 due to material changes from ageing and degradation effects from operation. The design to reduce the radiation exposure to occupational workers and environmental releases during operation. The plant layout and shielding, through regulatory requirements for occupational radiation protection, minimizes the collective doses for plant workers. The main control room is designed 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 or monitor critical parameters, an environmentally secure and radiation hardened backup control room is available, in the event the main control room becomes inaccessible.

	IPHWR-220	IPHWR-500	IPHWR-700
Power (MWth | MWe)	775 | 235	1730 | 540	2166 | 700
Vessel Diameter (m)	6.4	7.1	7.1
Avg BU (GWd/t)	6.7	7.5	7
Active fuel length (m)	5.09	5.94	5.94
Avg linear heat rate (kW/m)	28.6	40.1	50.2
Plant Efficiency (net %)	27.8	28.08	29.08
Steam Generators	4	4	4
CDF (/RY)	10^{-5}	10^{-5}	10^{-5}

Map



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ACRONYMS AND ABBREVIATIONS

Acronyms and abbreviations used in the booklet

AD Automatic Depressurization
ADS Automatic Depressurization System
ALARA As Low As Reasonable Achievable
ARIS Advanced Reactor Information System
ATF Accident Tolerant Fuel
ATWS Anticipated Transient without Scram
BDB Beyond Design Basis
BDBA Beyond Design Basis Accident
BOP Balance of Plant
BU Burn-up
BWR Boiling Water Reactor
CB Containment Building
CDF Core Damage Frequency
CRDM Control Rod Drive Mechanism
CC Core Catcher
CR Control Rods
CS Containment Structure
CV Containment Vessel
CVCV Chemical and Volume Control System
DBA Design Basis Accident
DG Diesel Generators
DID Defense in Depth
DVI Direct Vessel Injection
ECCS Emergency Core Cooling System
EDG Emergency Diesel Generator
ESWS Essential Service Water System
EUR European Utility Requirements
FA Fuel Assembly
FE Fuel Element
GDCS Gravity Driven Cooling System
HEU High Enriched Uranium
HMS Hydrogen Mitigations System
HPI High Pressure Injection
HRS Heat Removal System
HVAC Heat, Ventilation, Air Conditioning
IC Isolation Condenser
I&C Instrumentation and Control
LEU Low Enriched Uranium
LBLOCA Large Break LOCA
LOCA Loss of Coolant Accident
LPI Low Pressure Injection
LWR Light Water Reactor
MOX Mixed Oxide
MSL Main Steam Line
MSLB Main Steam Line Break
MWe Mega Watt electric
MWt Mega Watt thermal
NPP Nuclear Power Plant
NSSS Nuclear Steam Supply System
O&M Operation and Maintenance
PC Primary Containment
PCCS Passive Containment Cooling System
PORV Power Operated Relieve Valve
PP Primary Pump
PS Primary System
PSIS Passive Safety Injection System
PWR Pressurized Water Reactor
RC Reactor Circulation
RCCV Reinforced Concrete Containment Vessel
RCIC Reactor Core Isolation Cooling
RCP Reactor Coolant Pump
RCS Reactor Coolant System
RHR Reactor Heat Removal
RHRS Residual Heat Removal System
RIP Reactor Internal Pumps
RP Reactor Plant
RPV Reactor Pressure Vessel
RS Reactor System
RV Reactor Vessel
RY Reactor Years
SA Sever Accident
SBO Station Black-Out
SBWR Simplified Boiling Water Reactor
SG Steam Generator
SGTR Steam Generator Tube Rupture
SIS Safety Injection System
SLU Slightly Enriched Uranium
SPF Spent Fuel Pool
SS Safety System

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