

Chapter 2

PRESSURIZED WATER REACTORS

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2.1 INTRODUCTION

In a Pressurized Water Reactor (PWR), the coolant is pressurized to about 2,200 psia using a pressurizer and is not allowed to boil. The pressurized water is then pumped to steam generators where steam is produced and then fed to the turbine plant for the production of electricity. The reactor vessel and the associated steam generators, pressurizer, and coolant pumps are enclosed in a containment structure.



Figure1. Ohi power station complex, operated by Kansai Electric Power Inc., Japan.

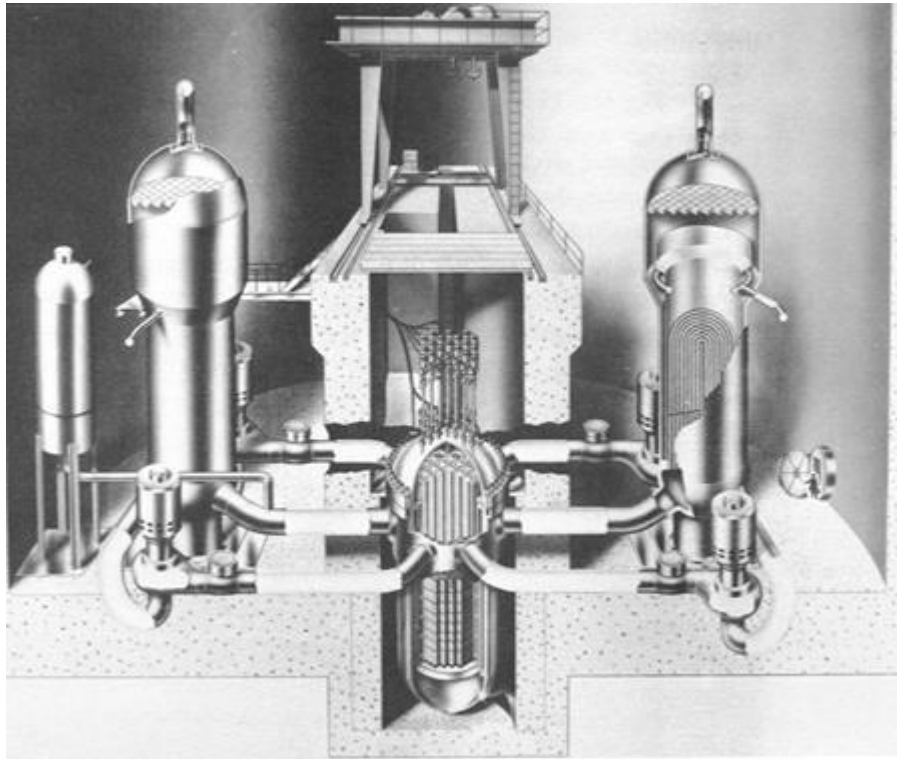


Figure 2. Schematic of a PWR fission reactor showing the core, control rods, coolant pumps and steam generators inside the containment structure. Notice the size of the “standard man” on the right hand side.

Reactors of this type were originally developed for naval propulsion purposes, and then adapted to land-based applications with larger sizes and lower fuel enrichment from the 98 w/o in naval systems to the 3.2 w/o level, which is about four times that of natural uranium at 0.72 percent. Ordinary water has a double function as coolant and moderator.

The cycle efficiency of a PWR is about 32 percent. Its capital cost is low, however, since it operates at a high power density reaching 100 MWth/m^3 .

2.2 POWER LOOP

For reliability considerations a minimum of two and sometimes as many as four steam generators and main coolant pumps are used as shown in Fig. 3. In a typical PWR circuit shown in Fig. 4, water at about 2200 psia or 150 bars is pumped into a pressure vessel containing the reactor core shown. The water flows through an annular region between the reactor vessel and the reactor core and then its flow is distributed by a nozzle system to the core for cooling the fuel elements. The reactor coolant pumps move the coolant to the steam generators, where steam is produced and then fed to the turbine plant. The coolant is condensed in a condenser then fed by the feedwater pumps back to the steam generators. The condenser uses a source of water such as a coolant pond or lake, river, ocean or cooling towers as heat sinks.

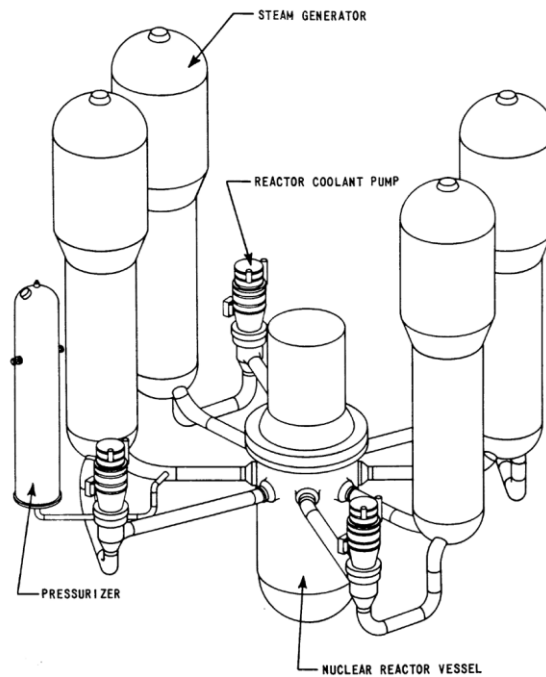


Figure 3. For reliability considerations a minimum of two and sometimes as many as four steam generators and main coolant pumps are used in a PWR.

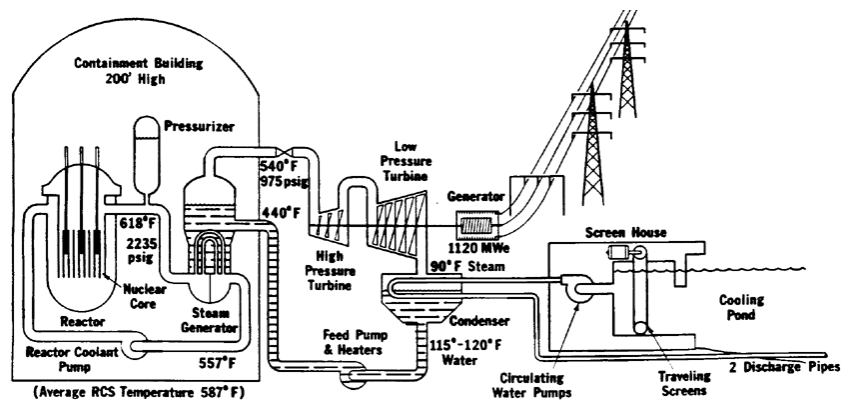


Figure 4. Typical PWR circuit.

2.3 REACTOR VESSEL

High pressure water at a pressure of about 2250 psia enters four inlet nozzles located above the midpoint of the pressure vessel as shown in Figs. 5 and 6.

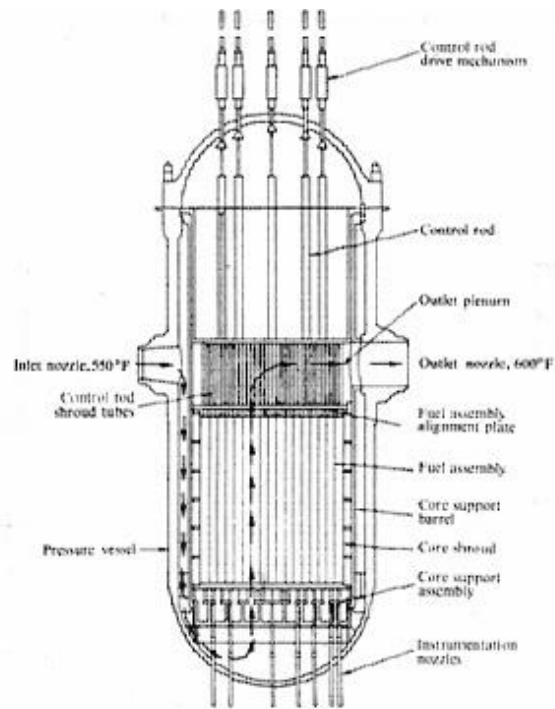


Figure 5. PWR reactor vessel showing the core, control rods, and coolant flow.

The water flows downward through the annular space enclosed by the reactor vessel and a thermal shield, and then upward through the fuel assemblies. Water exits from the fuel assemblies to the upper plenum, and then down around the inside of the upper pressure vessel and out of the four outlet nozzles shown in Fig. 6.

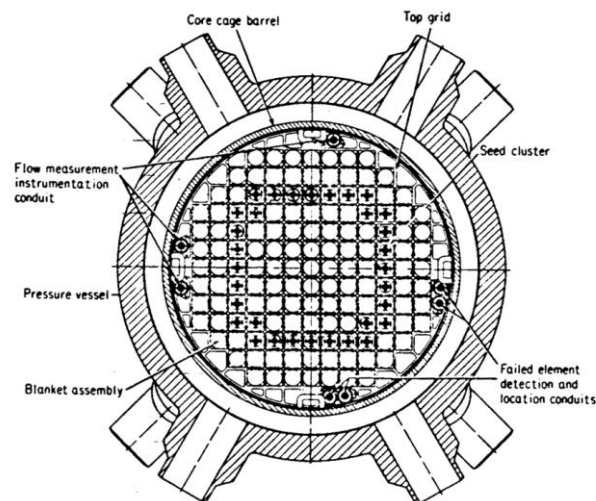


Figure 6. Cutout view of PWR reactor pressure vessel showing the core, fuel channels, control rods, and instrumentation.

The core contains instrumentation channels and grid spacing for the fuel elements. Some elements have spider shaped conduits through them for the control rods.

A core support shield and a core barrel support the core. The latter hangs from the seal ledge near the closure head. Some gamma rays attenuation is provided by the core barrel to supplement the shielding provided by the thermal shields, a non-load bearing component. The thermal shields function is to reduce gamma rays heating in the reactor vessel wall. They also contribute to neutron shielding of the reactor vessel to prevent its embrittlement through radiation damage by neutron irradiation.

Pressure vessels are forged from a 600 ton steel ingot. Steel scrap is heated in an electric furnace to 2,000 degrees Celsius or 3,600 degrees Fahrenheit. Then each of five giant ladles is filled with 120 tons of the orange hot molten metal. Argon gas is injected to eliminate impurities, and manganese, chromium and nickel are added to make the steel harder.

The mixture is poured into a blackened casing to form ingots 4.2 meters wide in the rough shape of a cylinder. Five times over three weeks, the ingots are pressed, reheated and re-pressed under 15,000 tons applied by a machine that rotates them gradually, making the floor tremble as it works.

The heavy forging is needed to make the steel uniformly strong by aligning the crystal lattices of atoms that make up the metal, known as the grain. In a casting, they would be randomly jumbled. The process is partly art and partly science similar to the making of Samurai swords. This single ingot technique is used by the Japan Steel Works (JSW) Company in Japan.

Another alternative is to turn back the technological clock and weld together two smaller forgings that are then welded using explosive techniques. That technique was used over the past 40 years in the USA and France and is still applied in China.

To prevent corrosion, the interior of the 600 ton and 12 inch thick vessel is lined with a stainless steel cladding.

2.4 REACTOR CORE AND FUEL

The core shown in Figs. 5 and 6 has a diameter of about 12 feet. It consists of about 200 fuel assemblies with a square cross section approximately 8.5 inches on the side. A fuel assembly is shown in Fig. 7.

In a typical fuel assembly, the fuel rods are arranged in a 17 by 17 or 16 by 16 square array. About one third of the fuel assemblies contain control rods which are incorporated into them by control rod guide tubes which replace the fuel rods. Other rods within the fuel assemblies contain no fuel and are filled instead by in core instrumentation and burnable poisons such as boron which extend the core life.

The fuel consists of uranium dioxide UO_2 pellets with an enrichment ranging between 2.1 to 3.1 percent in U^{235} . These pellets are 0.32 inch in diameter and 0.6 inch in length. The pellets are enclosed in a Zircalloy-4 cladding tube with a wall thickness of 0.025 inch. Zircalloy is used because of its low neutron absorption helping the neutron economy in the PWR. The fabrication process leaves a radial clearance between the cladding and the fuel pellets of about 0.003 inch.

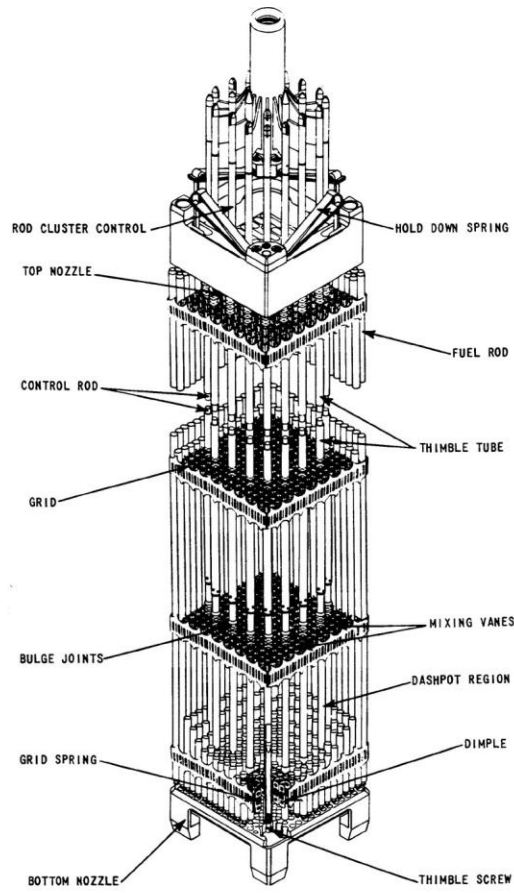


Figure 7. PWR fuel assembly with control rod drive.

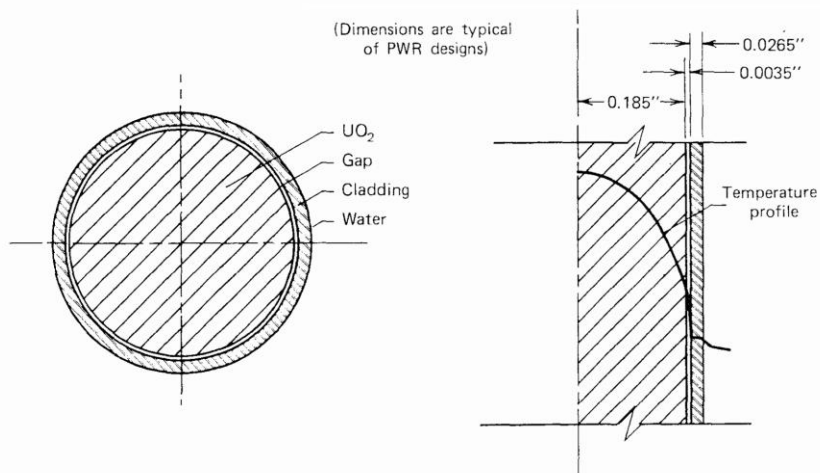


Figure 8. Temperature distribution in UO₂ fuel element and cladding.

2.5 OPERATIONAL CHARACTERISTICS

Table 1 shows the typical operational characteristics of a PWR, and Fig. 9 shows a typical PWR flow diagram.

Table 1. Operational Characteristics of a typical PWR.

Characteristic	Value
Thermal Power output	3,800 MWth
System pressure	2,250 psia
Fuel enrichment	1.9/2.4/2.9
Coolant flow	1.59×10^8 lbs/hr
Inlet temperature	565 °F
Outlet temperature	622.4 °F
Maximum fuel temperature	3,420 °F
Average linear heat rate	5.34 kW/ft
Maximum linear heat rate	12.51 kW/ft
Average heat flux	206,000 BTU/(hr.ft ²)
Maximum heat flux	550,000 BTU/(hr.ft ²)
Minimum Departure from Nucleate Boiling ratio (DNBR)	1.3
Active height	150 inches
Equivalent active diameter	142.9 inches
Height to diameter ratio	1.05
Active core volume	1413 ft ³
Average core power density	2,690 kW/ft ³
Fuel weight	103,000 kgs
Specific power	36.9 kW/kg U
Burnup	33,000 Mwdays/MTU
Conversion ratio	0.5
Number of fuel assemblies	241
Fuel element array	16x16
Assembly dimensions	8 in X 8 in
Number of fuel rods per assembly	236
Total number of fuel rods	56,876
Fuel element pitch	0.504 in
Fuel element outer diameter	0.382 in
Pitch to diameter ratio	1.33
Cladding thickness	0.025 in
Fuel pellet diameter	0.325 in
Pellet to clad gap size	0.0035 in

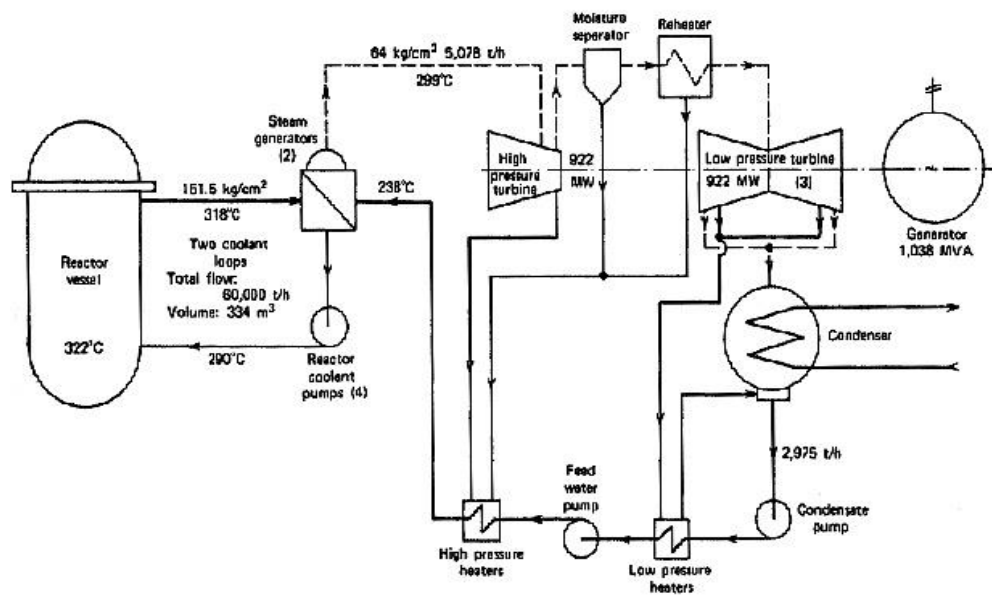


Figure 9. Typical PWR flow diagram.

2.6 PRESSURIZER

A typical pressurizer with its heaters, quench nozzle and pressure relief valves is shown in Fig. 10. Its function is to establish and maintain the reactor's coolant system pressure during steady state operation and transients. It also doubles as a surge container and water reserve to accommodate the the primary coolant volume changes during operation.

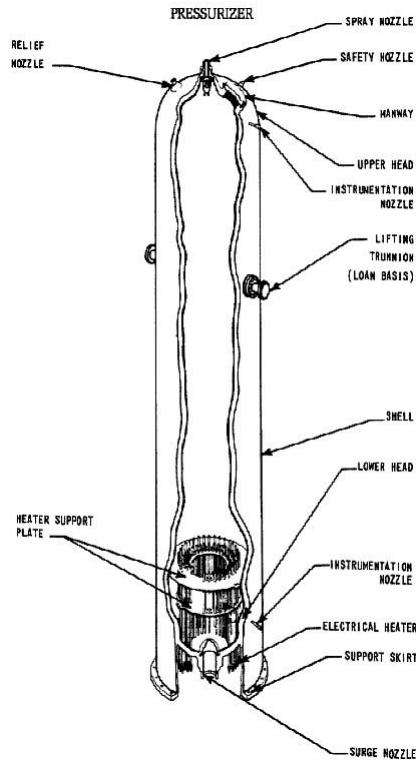


Figure 10. Typical pressurizer showing heaters, quench nozzle and pressure relief valves.

The pressurizer consists of a vertical cylindrical pressure vessel with a bottom surge line inlet connected to the reactor's coolant piping. The pressurizer contains removable electric heaters in its bottom section that is normally immersed in water. The upper section is equipped with a water spray nozzle in a region that is normally filled with a steam bubble. If the pressure in the primary system decreases, some water in the pressurizer flashes into steam and maintains the primary system's pressure. The operator then can activate the electrical heaters until the required system pressure is attained. When the system pressure increases, some steam can be condensed by the activation of the water spray from the reactor coolant line. This action would reduce the pressure back to the normal operating level.

The installed heater power is about 1,800 kWth. The pressurizer has 6 safety and 2 relief valves at its top. The maximum spray rate is 800 gpm and the continuous spray rate 1 gpm.

An important function of the pressurizer is to provide relief from any sudden over-pressurization of the primary system. The safety valves automatically open at a preset overpressure. To reduce the frequency of operation of the safety valves two electromagnetically controlled relief valves are available for manual control to the operator.

2.7 STEAM GENERATORS

There exist two types of steam generators used with PWR designs. The U-tube design is shown in Fig. 11 and the once-through design is shown in Fig. 12.

The function of the steam generators is to transfer the heat from the primary coolant to the secondary feedwater to generate steam for the turbine generator set. Steam generators for the PWR design are shell and tube heat exchangers with high-pressure primary water passing through the tube side and lower pressure secondary feedwater and steam passing through the shell side. In addition to generating steam, steam generators double as a barrier between the primary coolant and the secondary coolant.

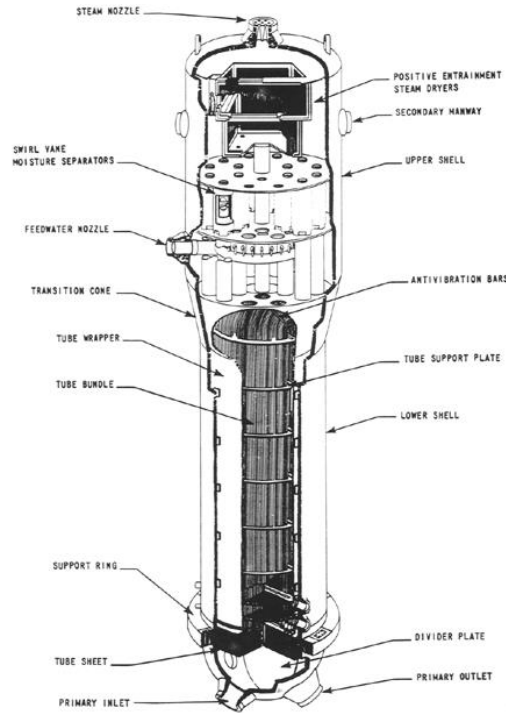


Figure 11. A vertical U-tube steam generator.

In the U-tube steam generator design, the hot leg primary coolant is pumped by the main reactor coolant pumps shown in Fig. 13 into the steam generator inlet nozzle and flows upwards in the U-tube bundle and then downward to exit from the outlet nozzle. The feed water enters the side of the steam generator and flows upward through an economizer region. The feedwater continues upward gaining energy and flows past the tube bundle where the water level is maintained above the tubes and below the moisture separators. The generated wet steam passes through swirl vane moisture separators where most of the entrained water is removed. The entrained water from the moisture separators flows downward through a downcomer and mixes with the incoming feedwater. The steam leaving the swirl vane moisture separators enters steam dryers where the moisture content is reduced to less than 0.25 percent.

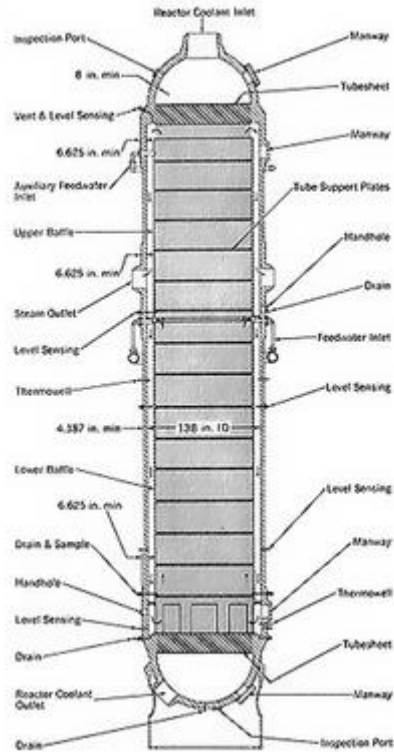


Figure 12. Once-through steam generator design.

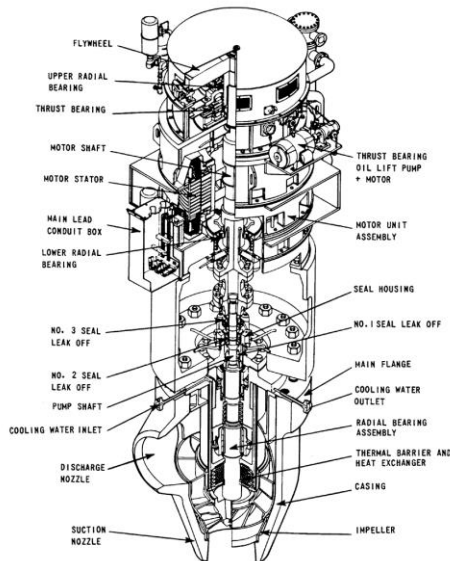


Figure 13. Reactor coolant pump.

In the once-through steam generator design the hot leg primary coolant enters from the top of the steam generator through a single inlet nozzle and flows downward inside the tubes and discharging through two outlet nozzles at the bottom. The feedwater enters the side of the steam

generator and flows downward along the shell side until it reaches the tube sheet. The flow then moves upward in a countercurrent manner to the primary coolant. As the feedwater flows upward it is converted into steam and is even superheated by 50 degrees F. The superheated steam flow at the top of the bundle flows outward toward the shell and then downward in the annular region formed by the shell and the upper baffle to the outlet nozzles.

2.8 RUSSIAN DESIGNS, VODO VODYANOI ENERGETCHESKY REACTOR, VVER

Figure 14 shows the VVER-1000 (Vodo Vodyanoi Energetichesky Reactor) PWR Russian reactor design. The figure shows the upper block (1), the control drive gear (2), the control rod guide and protection tubes (3), the reactor pressure vessel (4), the reactor shaft (5), the reactor core (6), and the ducts for ionization chambers and their cabling (7).

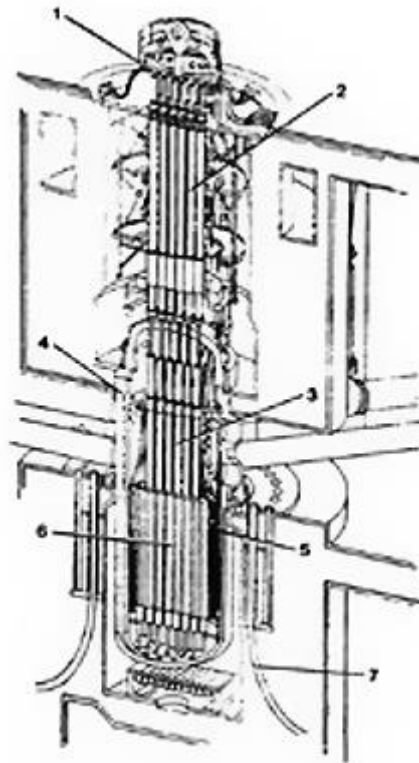


Figure 14. The VVER-1000 Russian PWR design. Upper block: 1, control drive gear: 2, control rod guide and protection tubes: 3, reactor pressure vessel: 4, reactor shaft: 5, reactor core: 6, and the ducts for ionization chambers and their cabling: 7.

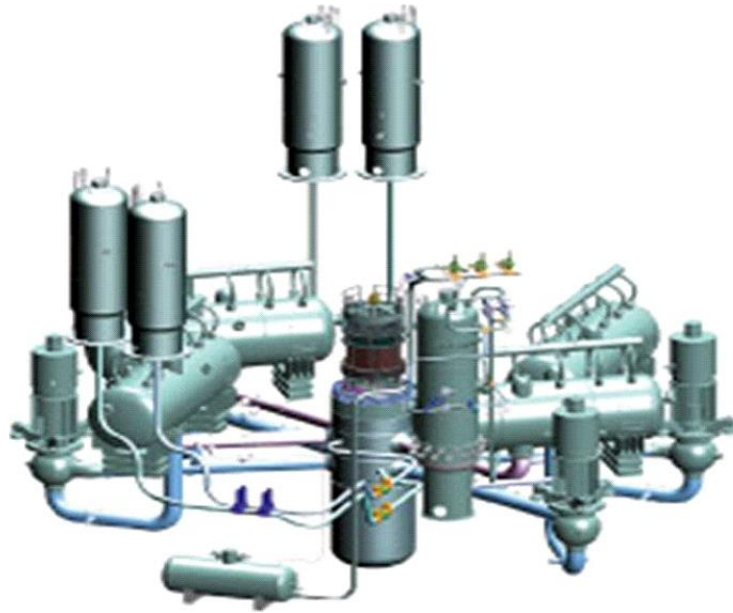


Figure 15. AES 2006 Russian PWR design.

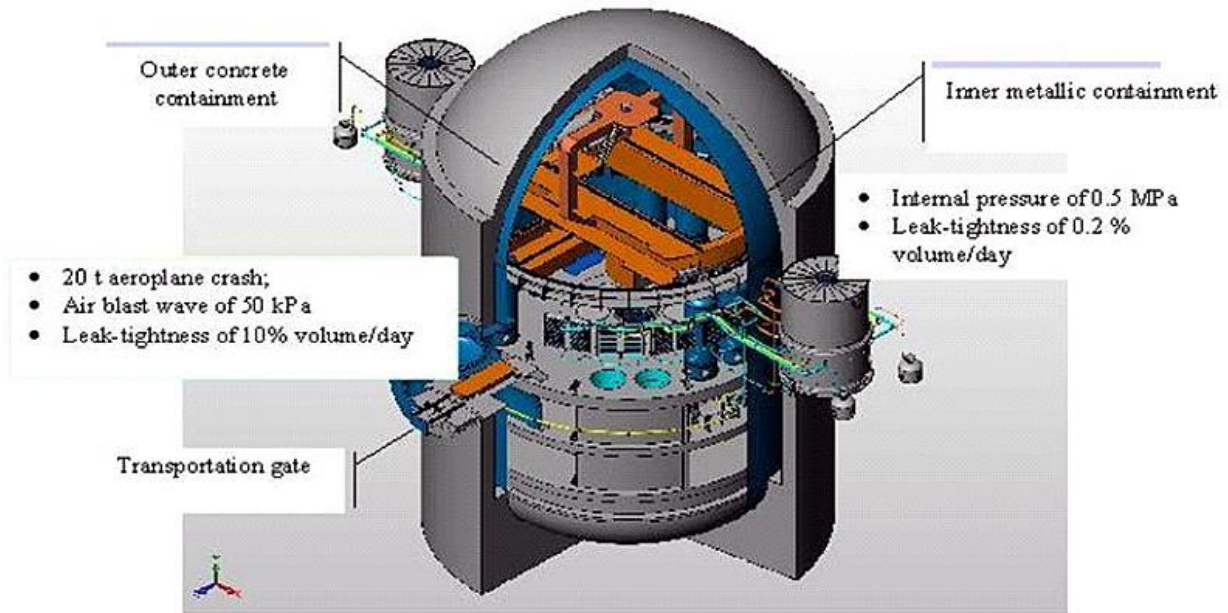


Figure 16. VBER 300 containment structure technical specifications.

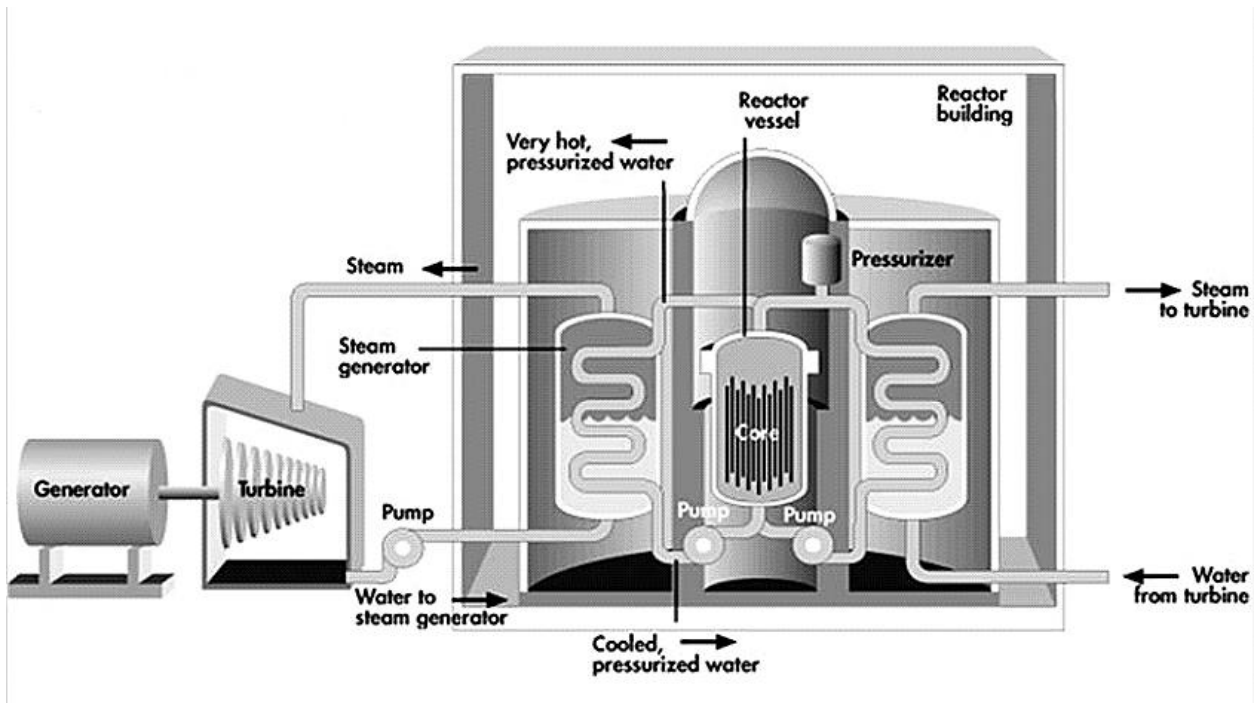


Figure 17. VVER 440 model V230 Russian PWR diagram.

2.9 CONTAINMENT SYSTEM

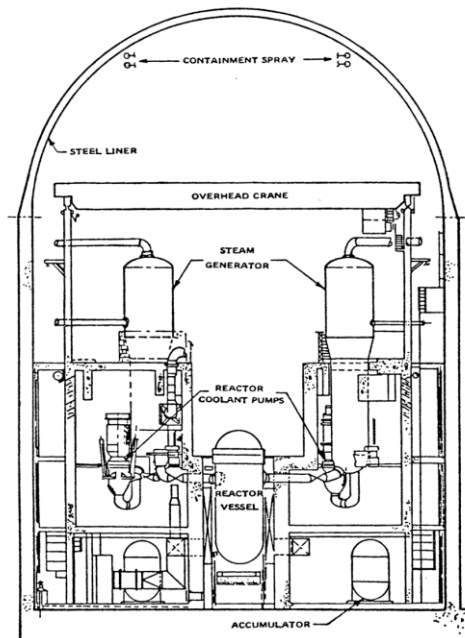


Figure 18. PWR dry containment structure.

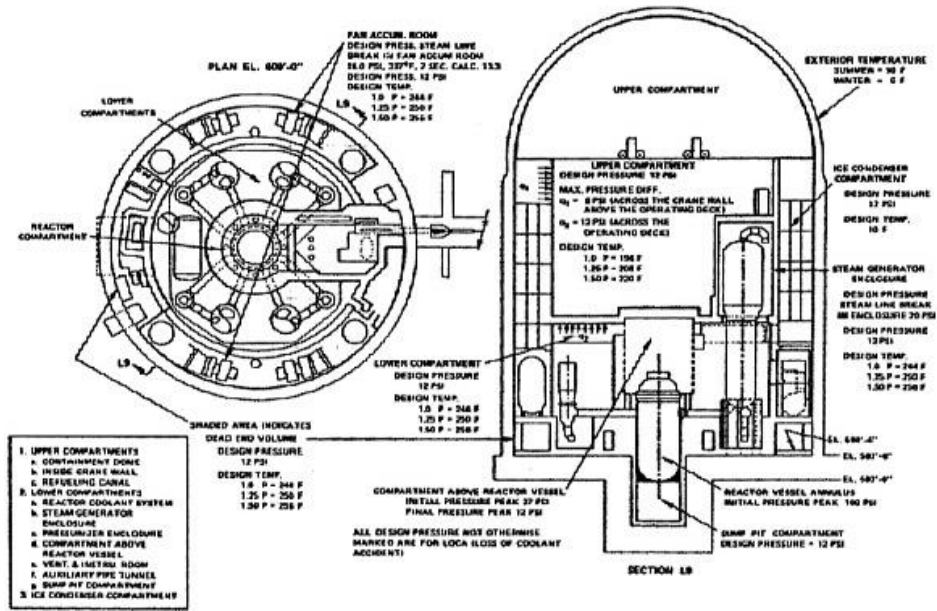


Figure 19. PWR ice condenser containment structure design.

PWR designs are surrounded by a containment system with multiple Engineered Safety Features (ESFs). A dry containment design is shown in Fig. 18, and consists of a steel shell containing multiple safety systems. A concrete biological shield surrounds the steel shell. The biological shield is meant to protect against the outside elements and is not meant as a barrier

against the release of radiation. For instance it is designed to withstand a direct hit by a Boeing-747 aircraft, and 100 miles per hour missile such as a light pole driven by tornado or hurricane winds.

Another containment design is shown in Fig. 19, where an ice condenser is used to quench any release of radioactivity or steam caused maybe by an earthquake event.

The containment contains the various circuits for emergency core cooling water injection into the primary system. These include:

1. *The accumulators* which are large vessels containing water under nitrogen gas pressure. They are connected to the primary system by automatic valves, which open if an accident occurs that reduces the primary system pressure below 40 bars.
2. *The High pressure Injection System (HPIS)* allows pumping of water into the system at a high head or pressure of about 100 bars but at low flow rates.
3. *The Low pressure Injection System (LPIS)* allows water to be pumped at a low head or pressure below 30 bars at high flow rates.
4. *A containment spray system* to quench any released steam in the case of an accident.

2.10 ADVANCED PRESSURIZED WATER REACTOR, US-APWR

The new US-APWR (Advanced Pressurized Water Reactor) is a high capacity third generation technology, developed by Mitsubishi Heavy Industries (MHI) from Japan. It has been selected by Dallas, Texas based power company TXU for its next nuclear plant.

TXU plans to file applications for combined construction and operating licenses using the US-APWR technology for 2-6 gigawatts at multiple sites including Comanche Peak site which has two units in operation.

The filings would facilitate commercial operation of the units starting from 2015 to 2020. On Friday, March 9, 2007, TXU formally notified its reactor selection to the USA Nuclear Regulatory Commission (NRC) and launched the preparation of a Combined License (COL) application per 10 CFR Part 52.

The US-APWR is based on technologies for a 1,538 MWe APWR planned for use at the Tsuruga Power Station Units 3 and 4 of the Japan Atomic Power Company. A variety of modifications were added in reflection of the demands of USA utilities for enhanced performance.

Improvements include the world's highest level of thermal efficiency at 39 percent, a 20 percent reduction in plant building volume, 24 months fuel cycle length, and greater economy by increasing the power generation capacity to 1,700 MWe, which is the world largest power class.

Mitsubishi is planning to construct the US-APWR in cooperation with a major engineering and construction company, Washington Group International Inc.

It has established MHI Nuclear Energy Systems Inc. (MNES), a wholly owned subsidiary, in Washington, DC, and has started procedures to submit an application to the NRC for Design Certification of the US-APWR by 2007.

The US-APWR's is advocated as possessing excellent economy, proven safety and reliability, and MHI's comprehensive capability to undertake engineering, fabrication, construction, detailed maintenance and supply of high reliability fuel.

In Japan, MHI has engaged in the construction of 23 Pressurized Water Reactors and is constructing one nuclear power plant.

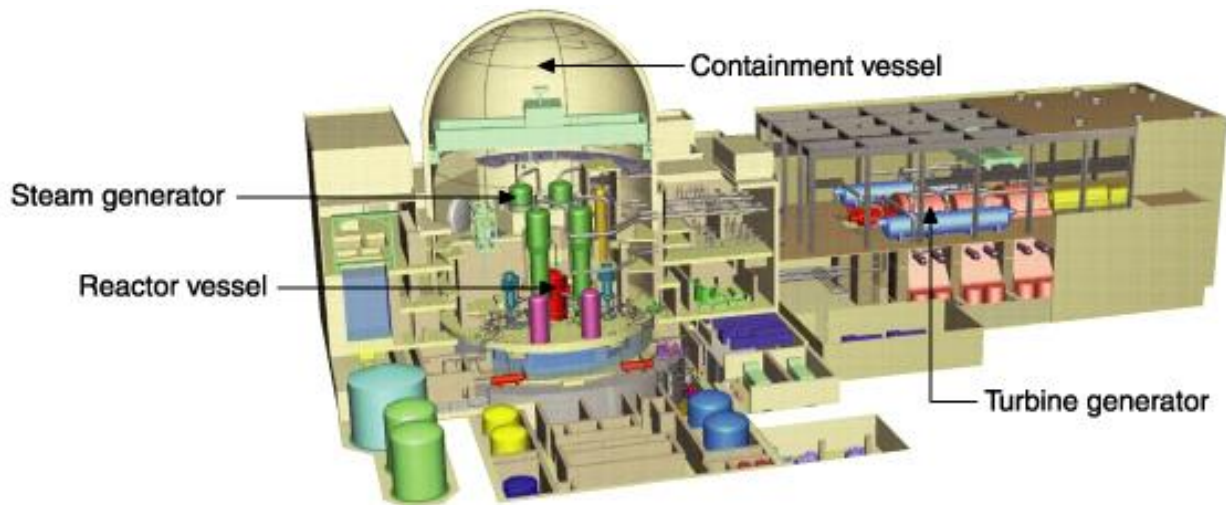


Figure 20. The APWR with a capacity of 1,700 MWe is advocated as the world largest in its class.

Since October 2006, French AREVA and Japanese MHI established an alliance to define the conceptual bases on which a medium capacity future advanced reactor will be developed. They have agreed on the main features of this reactor: an advanced generation 3, pressurized water, 3 loops reactor with a power of around 1100 MWe. It would integrate the latest features already adopted by AREVA and MHI in terms of safety such as resistance to commercial airplane crashes, environment such as reduced spent fuel and waste, and efficiency including the possibility of extended fuel cycles and capacity to use the Mixed Oxide (MOX) PuO_2 and UO_2 fuel cycle.

2.11 ADVANCED PASSIVE AP1000 TOSHIBA-WESTINGHOUSE DESIGN



Figure 21. Conceptual design of the AP600 and AP1000 concepts.

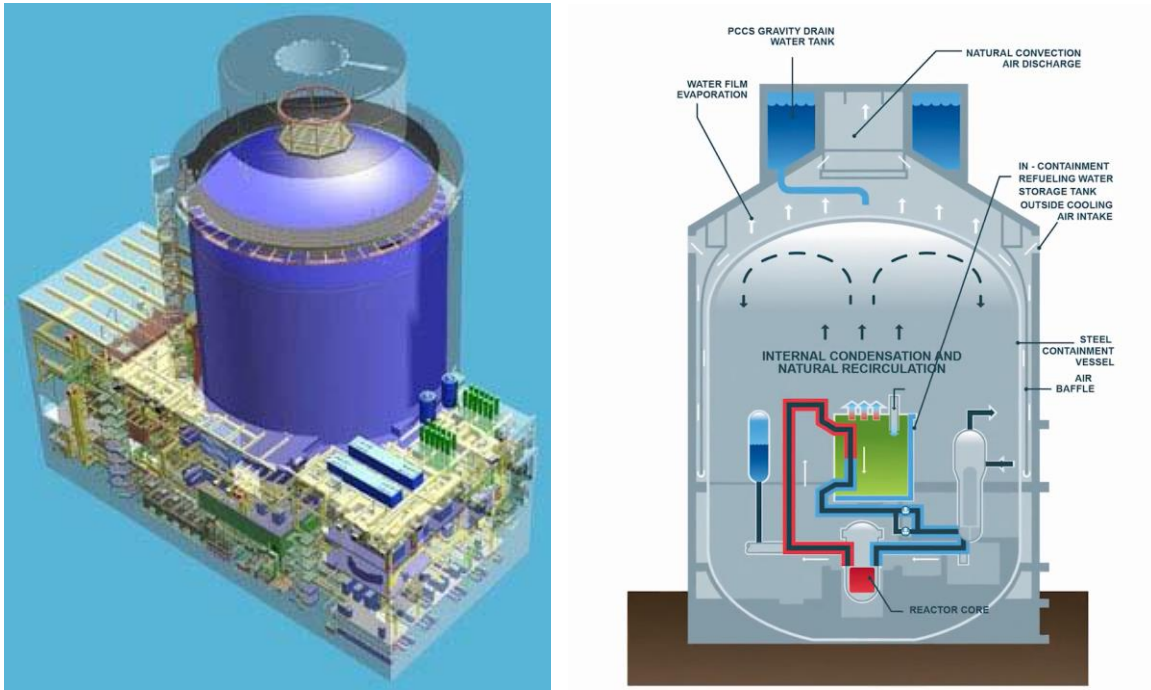


Figure 22. Cutout through the AP1000 PWR design showing the containment structure.

Toshiba from Japan and its Westinghouse subsidiary that it acquired from British Nuclear Fuel Ltd. (BNFL) in the USA are committed to the design and development of advanced nuclear reactors that are safe, low-cost, reliable, and environmentally acceptable. Their AP1000 design is meant for near term deployment and their 4S and hydrogen production systems are targeted for future technology development.

The AP1000 is the only Generation III+ nuclear power plant to receive design certification from the United States Nuclear Regulatory Commission. It is an advanced plant that further increases safety through the use of naturally occurring forces such as gravity, natural circulation, and condensation. In the unlikely event of a plant emergency, these safety systems, because of their inherent nature, will automatically activate without the need for human intervention.

In addition to the enhanced safety features, the AP1000 is cost-effective. The plant is composed of modules that are constructed in either factories or shipyards, thereby improving quality while reducing the potential for delays that are associated with field construction.

Even though it is an advanced plant, it is a proven design that is based on the same Westinghouse PWR technology that has supported the nuclear industry over the last 50 years. Toshiba brings to the partnership with Westinghouse a highly efficient and reliable turbine generator design, state-of-the-art construction technologies, and knowledgeable construction management.

Toshiba and Westinghouse are also developing the 4S, a Super Safe, Small, and Simple reactor. The 4S is a 10-50 MWe, passive safety fast spectrum plant that has a 30-year operating life before the need to refuel, also known as a battery reactor.

Future technology development includes the production of hydrogen as a fission energy carrier in nuclear power plants. Toshiba and Westinghouse are developing three kinds of hydrogen production methods:

- a) The dimethyl ether (DME) reforming method for light water reactors.
- b) The high-temperature steam electrolysis (HTE) method for the 4S and fast neutron reactors.
- c) The sulfur iodine (SI) method for high-temperature gas reactors

2.12 EVOLUTIONARY PRESSURIZED WATER REACTOR, EPR AREVA DESIGN

The Evolutionary Pressurized Water Reactor, EPR is a Generation 3+ European Pressurized Water Reactor design with a capacity of 1600 MWe. It features advanced technologies, making it a reactor with the advocated following characteristics:

1. A high level of safety: Extended prevention of the reactor core melt down hypothetical accident and its potential consequences, resistance to external risks such as an aircraft crash or a strong earthquake. The major safety systems comprise four sub-systems or "trains". Each train is capable of performing the entire safety function independently. There is one train in each of the four safeguard buildings, which are separated from each other by the reactor building to prevent simultaneous failure of the trains.
2. Optimized environmental qualities: A 15 percent reduction in the production of long half-life radioactive waste, and increased performance and thermal efficiency.
3. Simplified operation and maintenance conditions: Totally computerized control room, with a more user-friendly human-machine interface.
4. Improved economic competitiveness.

Areva NP is developing two EPR projects in Europe. In Finland construction is underway of an EPR for the Finnish utility TVO (Olkiluoto 3 project). The Finnish EPR will be the first Generation III+ reactor to go into service. In France, Electricité De France (EDF) has reached a decision to build a series of EPR reactors at the Flamanville-3 project site.

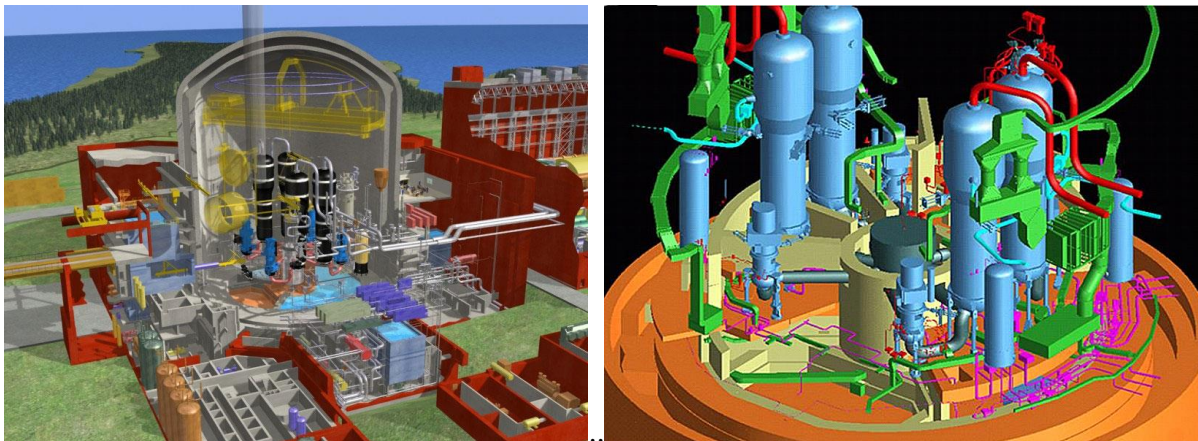


Figure 23. Cutouts through the Aeva EPR pressurized water reactor design showing its double walled containment.

The design certification of the EPR is scheduled to start in early 2008. The EPR is taking part in the Generic Design Assessment launched by the Health and Safety Executive (HSE) in the UK.

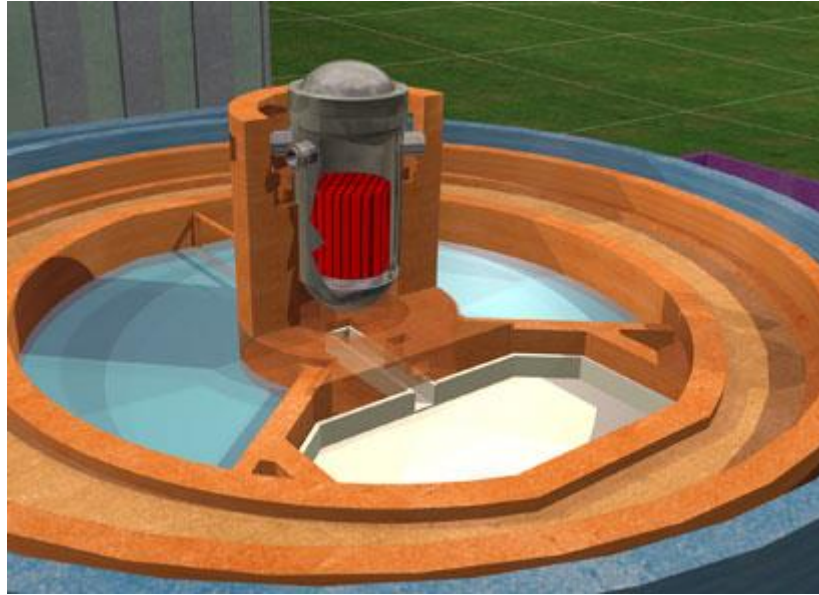


Figure 24. EPR corium retention and auxiliary water storage pools.

The reactor containment building has two walls: an inner prestressed concrete housing internally covered with a metallic liner and an outer reinforced concrete shell. This building houses the reactor coolant system, whose main components are the reactor vessel, the steam generators, the pressurizer and the reactor coolant pumps.

Inside the containment there is a special area where in the unlikely event of core meltdown, any of the molten core escaping from the reactor vessel would be collected, retained and cooled.

The turbine building houses the equipment that turns the steam produced into electricity: the turbine, the alternator and the transformer, which is connected to the grid.

In the event of a power blackout, diesel generators, housed in two separate buildings, supply electricity to the safety functions.

The outer shell covers: the reactor building, the spent fuel building, and two of the four safeguard buildings. The other two safeguard buildings are protected by being in a different location.

The plant is operated from an advanced control room where all operating data are centralized. The control room is located in one of the safeguard buildings, protected by the outer shell. The computerized control room is equipped with state of the art digital technology.

2.13 ADVANCED PRESSURIZED REACTOR 1400, APR1400

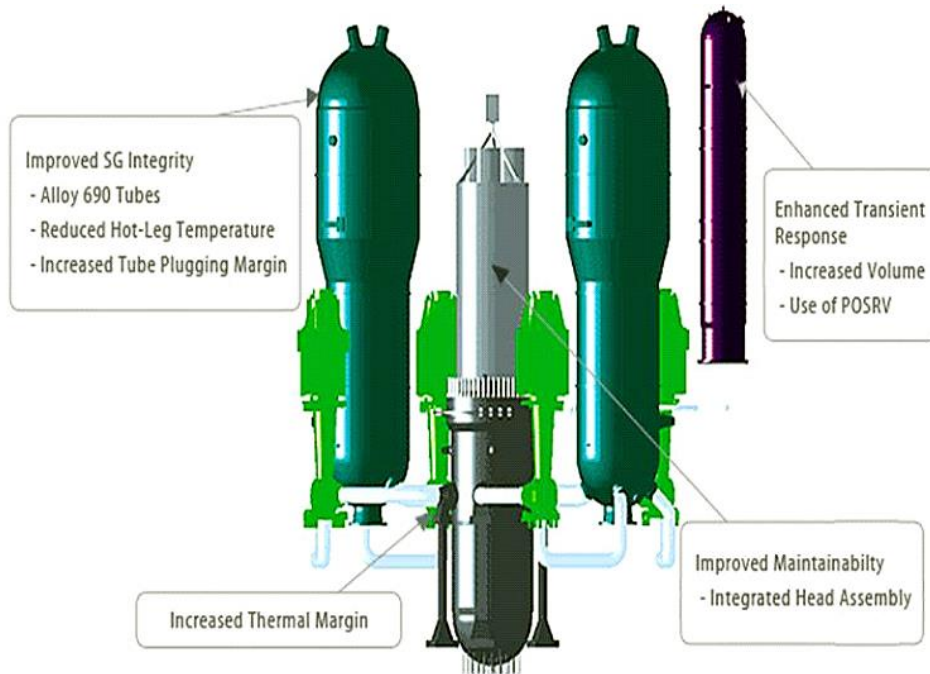


Figure 25. South Korea's AP1400 design.

The APR-1400 Advanced PWR design is a South Korean design that has evolved from the USA System 80+ with enhanced safety and seismic robustness and was earlier known as the Korean Next-Generation Reactor. The design certification by the Korean Institute of Nuclear Safety was obtained in May 2003.

It is rated at 1,455 MWe gross, 1,350-1,400 MWe net, and 3,983 MWth with a 2-loop primary circuit. The first of these reactor is under construction as Shin-Kori-3 and 4.

The fuel has a burnable poison and will attain 55 GWd/t burn-up with a refueling cycle of 18 months. Its design outlet temperature is 324°C.

The capital cost at the end of 2009 was \$2,300 per installed kilowatt, with a 48-month construction time with a plant design life of 60 years.

The design includes a missile shield to defend against both an internal and external missile attack. It includes seismic restraints and improved materials that would prevent damage to the reactor in the event of an earthquake.

The outlet nozzles are placed vertically higher than the inlet nozzles in order to promote natural circulation in the event of a loss of pumping power.

Its low capital cost attracted the attention of the United Arab Emirates, UAE which contracted for 4 units. Jordan is negotiating a similar acquisition.

2.14 ATMEA1 AREVA-MITSUBISHI DESIGN

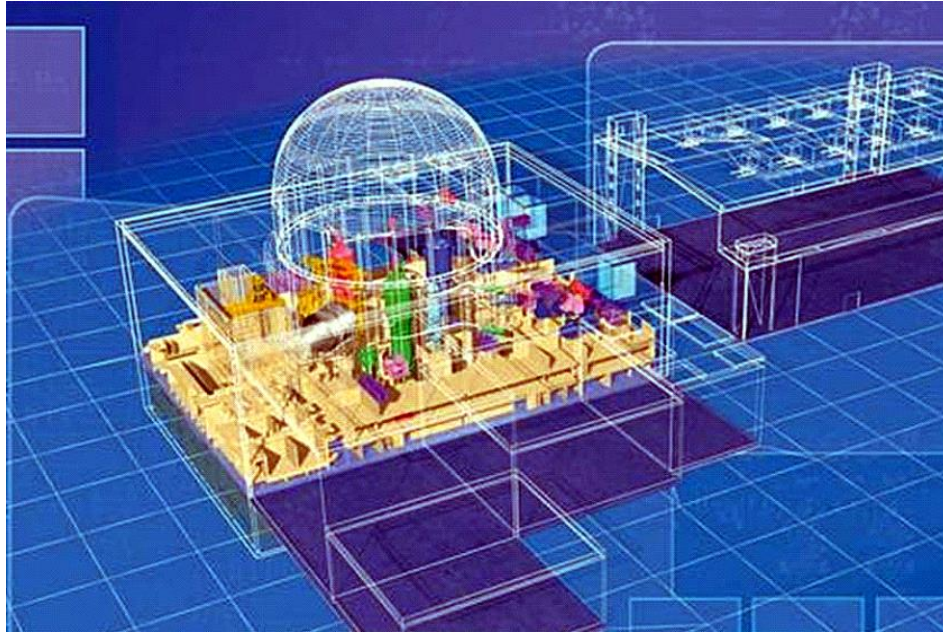


Figure 26. The Atmea1 PWR design based on collaboration between France's Areva and Japan's Mitsubishi.

The Atmea1 is developed as a joint venture of France's Areva NP and Japan's Mitsubishi Heavy Industries to produce an evolutionary 1150 MWe net, 3150 MWth three-loop PWR using the same steam generators as the EPWR concept.

It has extended fuel cycles, 37 percent thermal efficiency, 60-year life, and the capacity to use mixed-oxide fuel only.

Its safety systems provide an optimized balance between passive and active systems, and advanced accumulators.

The probability of core damage and large radiological release is estimated as a factor of 10 lower than a conventional PWR.

In the case of a severe accident, the core catcher will passively collect, retain, and cool the core debris in a specially designed area inside the reactor containment building.

Hydrogen-control technology would prevent hazardous hydrogen accumulation.

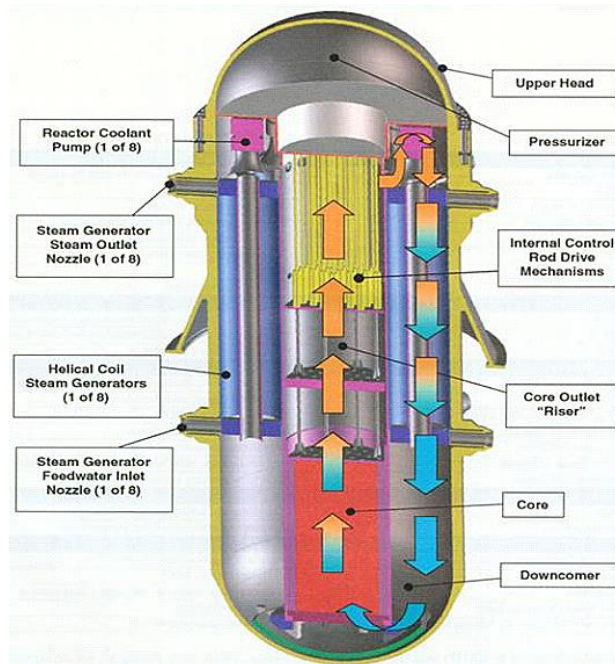


Figure 27. IRIS Integral PWR design contains the reactor core and the steam generators within the same pressure vessel .

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