Pressurized Water Reactor Power Plant

This material was, for a purpose to be used in a nuclear education, compiled comprehensively with a caution on appropriateness and neutrality of information, based on references of neutral organizations, suh as NRC, Wikipedia and ATOMICA, and vendors' information especially on advanced reactors. In the end of this material, references are listed.

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Contents

Part 1 Descriptions of PWRs

Chapter 1 PWR Development

1.1. General

A pressurized-water-type nuclear power reactor (PWR) uses light water as the reactor coolant and moderator in the state of high temperature and high pressure not boiling in the reactor core (primary system: reactor coolant system) and sends the high-temperature and high-pressure water to steam generators (primary system) to generate steam with heat exchangers (steam system: secondary coolant system) for a turbine generator to generate electricity.

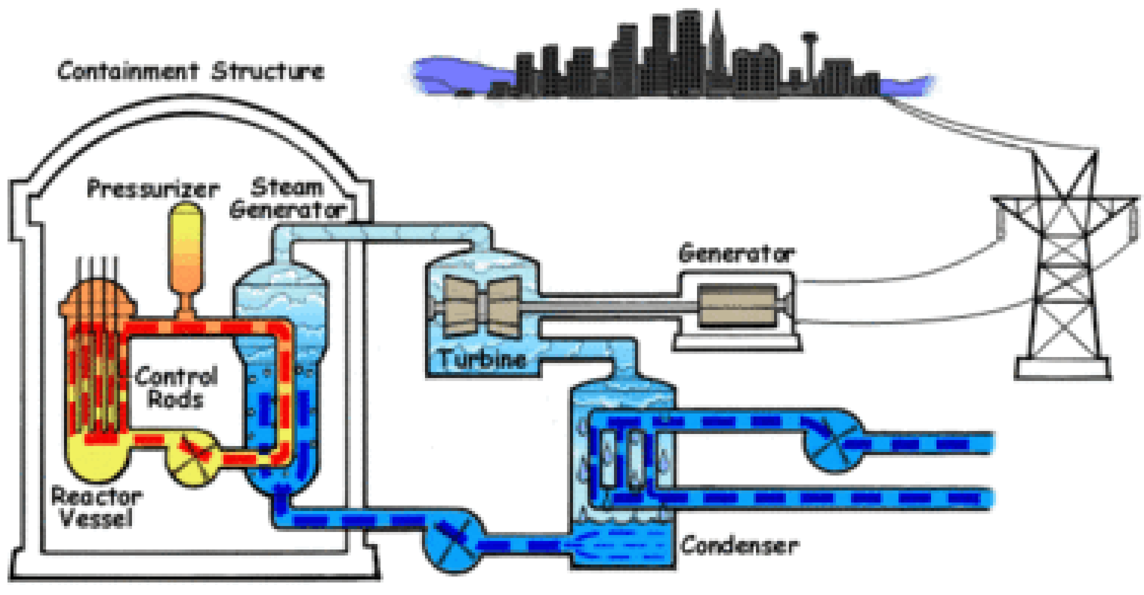


Figure 1 Outline of PWR Power Plant

More details in System Outline of APWR NPP

1.2. PWR Type

Operating PWRs are classified roughly into U.S. / Europe type and Russia type (VVER). The U.S. / Europe type PWRs of WH and CE are similar in use of tetragonal lattice type fuel assemblies and vertically installed steam generators, but different a little in their reactor coolant systems and steam generators. On the other hand, Russian PWR adopts hexagon fuel assemblies and horizontally installed steam generators.

Moreover, as advanced pressurized water reactors with improved safety and economical efficiency, APWRs (made by WH) and System 80+ (made by CE) as large-sized reactors and AP600 (made by WH) as medium size reactors have been developed. The APWR construction is planned in Japan.

Manufacturers / suppliers of PWR power plants are WH, ABB-CE (Asea Brown Boveri Combustion Engineering Nuclear Power) and B&W (Babcock & Wilcox Co.) in U.S. In Europe, Framatome (France), Siemens (KWU) (Siemens Power Generation Group, Germany) and MTM Ministry (Mintyazhmash, old Ministry of Heavy Industries, Russia) are manufacturers / suppliers of PWR power plants, and Mitsubishi Heavy Industries, Ltd. in Japan.

In 1999, 73% of LWRs (light-water type power reactors) under operation (341 units) is PWR in the world, 83% of LWRs under construction (35 units), and 93% of LWRs under planning (28 sets) are PWRs, which shows that the ratio of PWRs is increasing.

1.3 Technical Advance of PWRs

The first PWR used as a power source was for an atomic powered submarine U.S. Nautilus launched in 1954. Westinghouse (WH) cooperated in the submarine PWR development, and started its R&D of the PWR plant based on the project knowledge. After the permission of peaceful use of atomic energy, WH designed and constructed the first PWR nuclear power plant, Shippingport NPS of 100MW output, started operation in 1957. The Yankee Rower NPS started operation in 1961, and PWR plants came to start operation successively in 1965 and afterwards in U.S.

In the meantime, the PWR technology development has advanced and the basic configuration of PWR plants currently used in many countries in the world have been established. Also the plant output has become larger and the types have been standardized accordingly: two loops for 600 MW class, three loops for 900 MW class, and four loops for 1100 to 1300 MW class. In addition to WH, B&W and CE have advanced their PWR technologies. In European countries, such as France and Germany, they started to construct PWRs introducing technologies from WH at the beginning, but those countries developed their technologies according to their experiences obtained. France has standardized and constructed many PWR plants. In addition, Russia (Soviet Union) has developed and constructed many unique PWRs plants (called VVER or WWER). Even though the basic configurations are the same as those of the U.S. types, the equipment shape, plant arrangement etc. differ considerably from them.

1.4 PWR Technologies in Japan

The first PWR unit in Japan is the Unit-1 of the Mihama Power Station of the Kansai Electric Power Co., Inc., which started operation in 1970. PWR plants have been constructed successively after that, and 23 units are in operation as of December 31, 2005. The history of PWR technologies in Japan is shown in Table 1.

Table 1 History of PWR Technologies in Japan

Class / Specifications	300 MW	600 MW Adv. Std.	900 MW		1100 MW		1420MW (APWR)
			WH	Adv. Std.	WH	Adv. Std.	
No. of loops	2	2	3	3	4	4	4
Fuel type (initial)	14 x 14	14 x 14	15 x 15	17 x 17	17 x 17	17 x 17	17 x 17
Core power density (kw/l)	71	83 to 95	92	100	105	105	95
Steam generator	CE type	44, 51, 51M, 51F type	51 type	51M, 51F 52F type	51A type	51F, 52F type	70F-1 type
Coolant pump	63 type	93A (100B) type	93A type	93A, 93A-1 type	93A type	93A-1 type	100A type
Reactor containment	Steel semi-double type	Steel semi-double, Steel double type	Steel semi-double Steel double type	Steel double type	Ice condenser type	PCCV	PCCV
Turbine generator	TC2F44	TC4F40, TC4F44	TC6F40	TC6F40, TC4F52	TC6F44	TC6F44	TC6F52

The PWR plants in Japan are divided into the 1st to 4th generations (different from fuel generations mentioned below) according to the technology development. The 1st generation PWRs are imported plants constructed based on imported designs and technologies. In the order of Mihama Unit-1, Takahama Unit-1 and Ohi Unit-1 and 2, the first unit of two loops, three Loops, and four loops were imported from U.S., and the succeeding plants have been constructed by a domestic supplier. In the meantime, design and construction technologies have been studied and domestic production of equipment have been promoted.

The second generation PWRs, the Genkai Unit-2 of the Kyushu Electric Power Co., Inc. commissioned in 1981, the Ikata Unit-2 of the Shikoku Electric Power Co., Inc. commissioned in 1982 and the later units were constructed with domestic technologies. Based on the experience obtained through construction and operation by then, major equipment, such as reactor internals, steam generators, reactor coolant pumps and turbines, and plant design have been modified for improvement of reliability, safety, and availability factor, reduction of exposure, reduction of annual refueling and maintenance outage period, and reduction of construction period. The reactor coolant pump became to be domestically produced, which means all major equipment became domestic. In the 1975 fiscal year, standardization activities for nuclear power plant improvements started, led by initiative of the Ministry of International Trade and Industry, and design modifications have been promoted. The Sendai Unit-1 of the Kyushu Electric Power Co., Inc. commissioned in 1984 is the first modified standard unit and the succeeding plants are called the Modified Standard Plant.

The third generation PWRs, the Tomari Unit-1 of the Hokkaido Electric Power Co., Inc. commissioned in 1989, the Ohi Unit-3 commissioned in 1991 and the later units, were improved in economical efficiency, operability and maintainability in addition to reliability, safety using the established original technologies. In addition to improvement of steam generators etc., new-type main control panels and digital-type control devices (for Ohi Unit-3 plant and the subsequent ones) have been adopted, and plant arrangement have been modified for improvement in operation and maintenance performance and also for economical efficiency.

From 1982 to 1986, the APWR was developed as the fourth generation PWR. The APWR has incorporated modification designs that have been adopted in the conventional PWRs and those designs aiming at further improvement in safety and reliability, availability factor, radiation exposure and operability, including increase of output and improvement in power generation efficiency. The Tsuruga Units 3 and 4 of the Japan Atomic Power Company Co., Ltd. are planned to adopt this design type. For more, refer to the detail APWR discussion in the later section.

1.5 Improvements in Equipment

(1) Fuel and control rod

The fuel developed for the Yankee Rowe reactor is called the 1st generation PWR fuel (different from the 1st plant generation). The fuel is a segment type and UO_2 pellets of low enrichment are loaded into the stainless steel tube to constitute fuel rods. The fuel assembly consists of nine sub-assemblies of 6x6 array fuel rods with one kind of U-235 enrichment only. The control rod was a cross-section type. From the second-generation fuel, the chemical shim (adjustment of boron concentration in the reactor coolant) came to be used as the reactivity control method together with control rods. Three U-235 fuel enrichments became to be used, which improved in flattening the horizontal power distribution. This fuel was adopted for the San-Onofre reactor. Since the third generation fuel, the RCC (Rod Cluster Control) type control rod, which consists of pin-type control rods that move in specific fuel-rod positions, came to be adopted. From the fourth generation fuel, the material of fuel cladding tube was changed to Zircaloy-4 from stainless steel that had been used up to then. The PWR fuel currently used in Japanese plants is the fourth generation, and the fuel assembly types are 14x14 type, 15x15 type and 17x17 type (refer to Figure 2.) PWR fuel improvements and the history in Japan are summarized in Table 2.

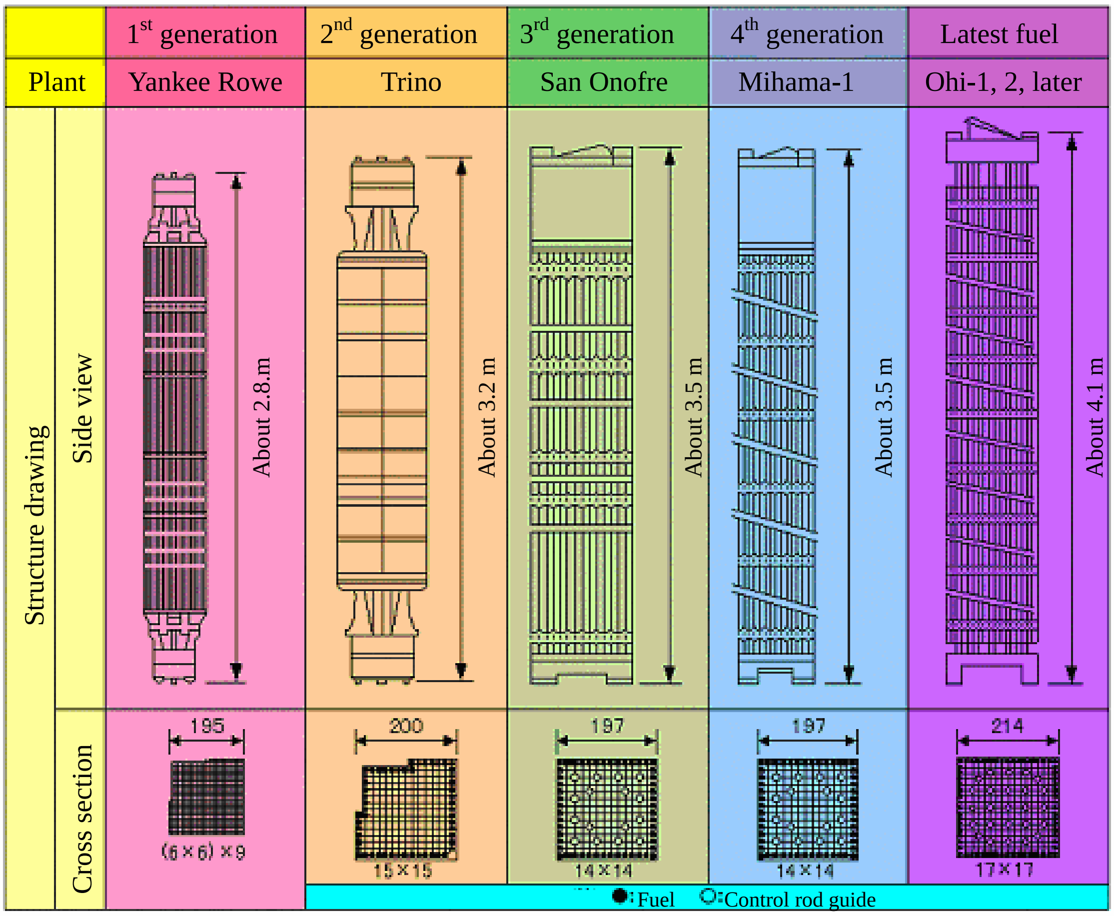


Figure 2 Improvements and History of PWR Fuel Design

Table 2 PWR Fuel Improvements and Its History in Japan

Year	Major technologies	Reactor type / Fuel type			
		Two loops / 14x14	Three loops		Four loops / 17x17
			15x15,	17x17	
1957 Development	• Basic fuel study • Fuel fabrication study • Core design study				
	• High-density pellet • Moisture control	Mihama-1			
	• Cladding tube inspection standard	Mihama-2			
Reliability improvement	• He pressure increase in rods • 15x15 fuel development • 17x17 fuel development				
		Genkai-1	Takahama-1		
1975	• Spacer spring force reduction • Cladding tube thickness control		Takahama-2		
	• Bottom-off design • Increase of spacers	Ikata-1			
	• Spacer corner design and strength improvement		Mihama-3		
	• Quality control				Ohi-1
1980	• Increase of enrichment	Genkai-2			
	• Gadolinia fuel development				Ohi-2
Economical improvement		Ikata-2			
				Semdai-1	
1985	• 48Gwd/t fuel development			Takahama-3	
	• Lower nozzle modification for debris			Takahama-4 Sendai-2	
High performance		Tomari-1			Tsuruga-2
1990	• 55GWd/t fuel development				Ohi-3
	• Development of Zry-4 spacer	Tomari-2			Ohi-4
	• Development APWR fuel				Genkai-3 Genkai-4

(2) Steam generator

The primary cooling circuit that transfers the heat generated at the nuclear reactor to steam generators is called a "loop", and power generation plants have been scaled-up by increasing the capacity and number of the loops. The capacity per loop at the Yankee Rowe reactor was designed to 40MW, but it has been increased to 300,000 to 350MW these days. Since the reliability of steam generators has a large impact on the overall plant reliability, many improvements had been promoted in Japan from the second generation plants to the third generation plants, such as improvement of the downstream water-treatment method, change to corrosion-resistant heat transfer tubes (from Inconel-600 alloy to specially heat treated Inconel-690 alloy), improvement of tube support plates in the hole-shape design and the material (adoption of four-leaves type holes and SUS405 material), reinforcement of the anti-vibration bar of heat transfer tubes.

(3) Safety equipment

(a) Emergency core cooling system

PWRs are provided with emergency core cooling systems (ECCSs) for a loss of coolant accident accompanying a reactor coolant system pipe rupture. The typical ECCS of the current plant is shown in Figure 3. The ECCS consists of high-pressure water injection system, low-pressure water injection system and gas-pressured accumulator tanks to cope with any pipe break ranging from a large break to a small one. Even though basic ECCS configurations of Japanese PWR plants under operation are the same, some system configurations and control systems are improved for reliability. The APWR ECCS is improved for drastic safety improvement by increase of the number of train configuration from two to four, and a design to install the refueling water pit as a water sources in the containment, etc.

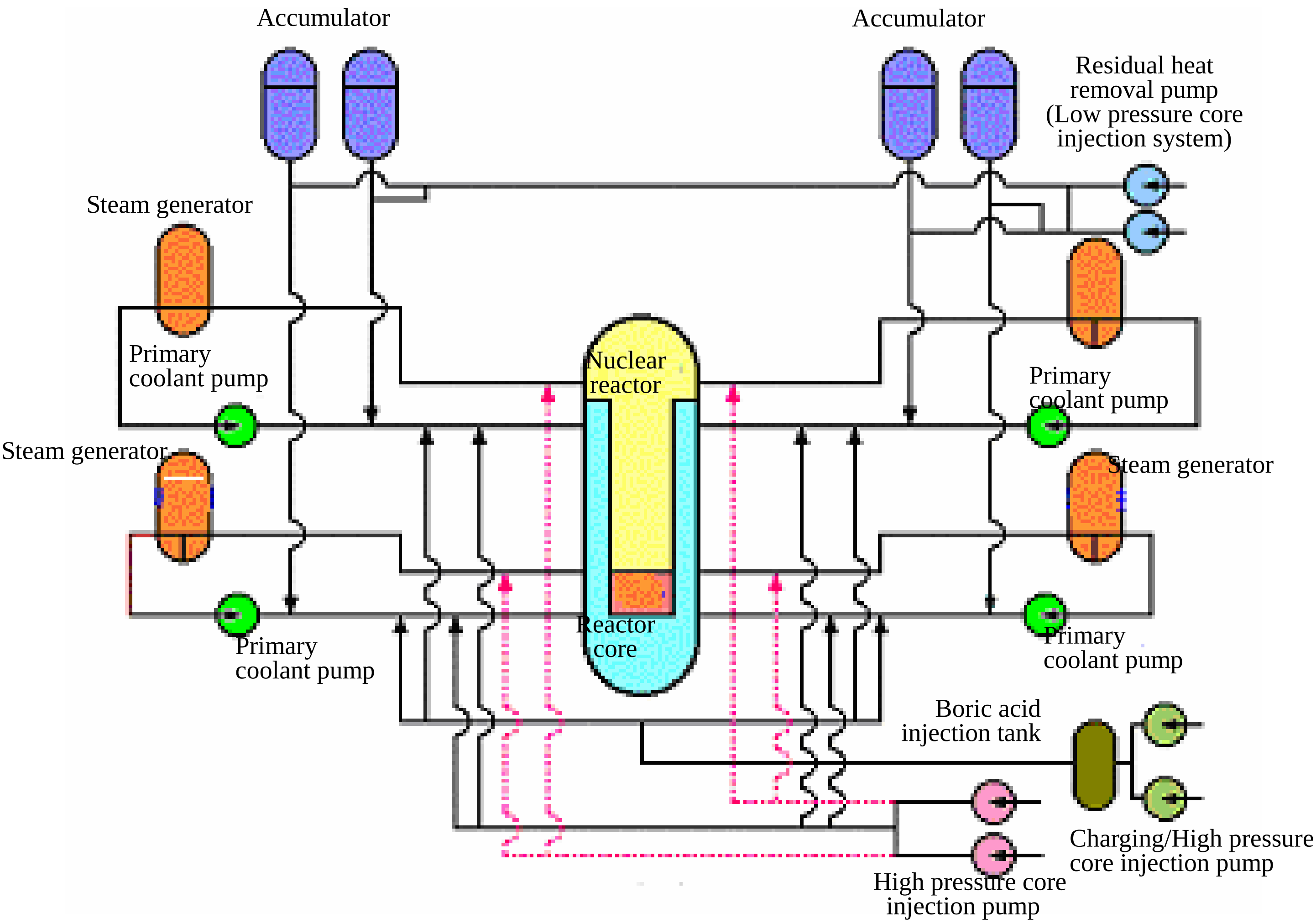
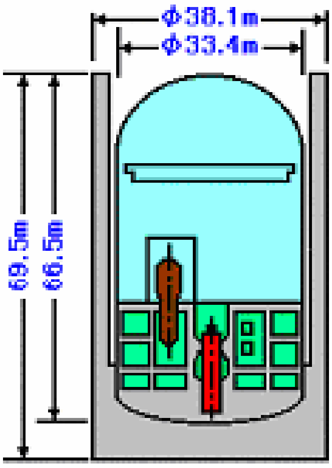


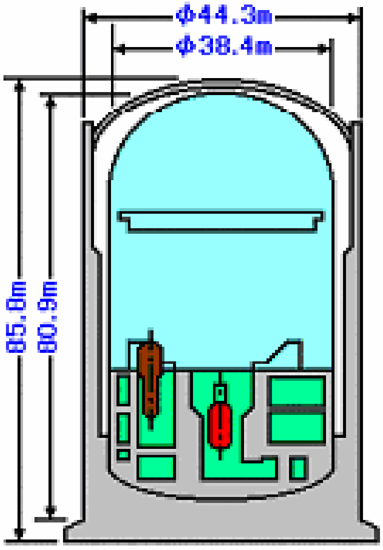
Figure 3 PWR ECCS Design (4 loop plant, Tsuruga-2)

(b) Reactor containment

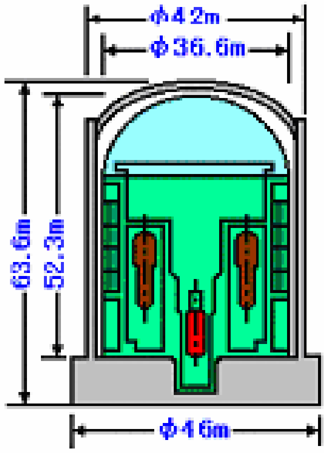
Changes of the reactor containment in Japan are shown in Figure 4. Cylinder-type steel containments were adopted for 2-loop and 3-loop plants. For 4-loop plants, ice condenser type containment vessels were adopted for the Ohi Nuclear Power Station Units 1 and 2 of the Kansai Electric Power Co., Inc., but prestressed concrete containment vessels (PCCV) were adopted for the Tsuruga Unit-2 of the Japan Atomic Power Company Co., Ltd. and subsequent ones. APWRs are planned to adopt PCCVs also. In addition, at the modified standard study, thorough study was made on arrangements within the containment for workability improvement, and the diameter of containment vessel is increased for the design of the Sendai Unit-1 of the Kyushu Electric Power Co., Inc. and subsequent ones.



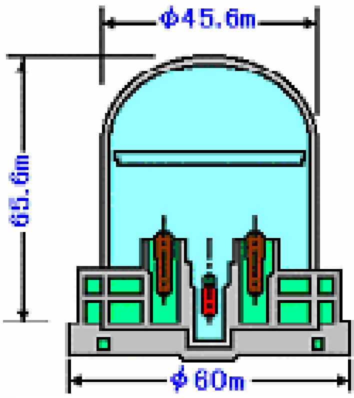
Steel Semi-double Type
(for 2 and 3 loops)
Licensed output: 300MW: Mihama-1
Licensed output: 500 - 600 MW: Mihama-2, etc.
Licensed output: 800MW: Takahama-1 etc.



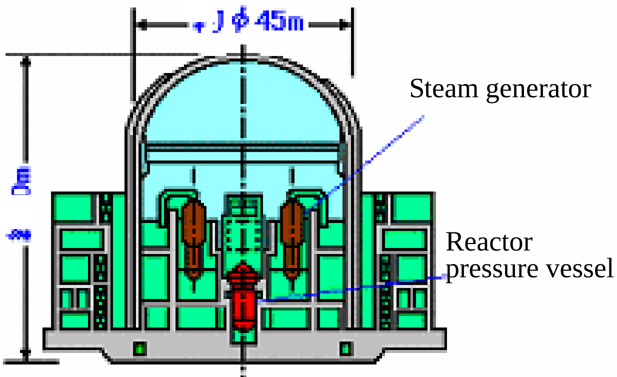
Steel Double Type
(for 2 and 3 loops)
Licensed output: 500 - 600 MW: Genkai-2, Ikata-2, Tomari-1 & 2 etc.
Licensed output: 800 – 900 MW: Mihama-3, Ikata-3 etc.



Ice Condenser Type
(for 4 loops)
Licensed output: 1100 MW: Ohi-1 &2



PCCV
(for 4 loops)
Licensed output: 1100 MW: Tsuruga-2, Genkai-3 &4, Ohi-3 &4



PCCV-APWR Type
(for 4 loops)
Output: 1500 MW (Plan)

Figure 4 History of PWR Containments (Dimensions are for references)

Chapter 2. PWR Technologies

2.1 Basic Configuration of PWRs

The PWR consists of a primary system (reactor system) and a secondary system (steam system) in order to keep radioactive materials in the primary system. The reactor coolant in a reactor vessel (reactor coolant) of the primary system is pressurized so that it circulates with reactor coolant pumps without boiling, and the high-temperature and high-pressure reactor coolant (reactor-vessel outlet temperature: about 325 degree-C, reactor-vessel inlet pressure: about 157 kg/cm^2, at rated power) moves from a reactor core to steam generators (primary side) for effective heat transfer. Within the steam generators, heat exchange occurs at heat transfer tubes transporting the heat from the primary side to the secondary side, and steam (operating temperature: about 277 degree-C and operating pressure: about 62 kg/cm^2, at rated power) is generated. This steam is sent to a turbine to drive a generator, condensed in condensers to water, and sent back to the steam generators (secondary side) with main feedwater pumps (feedwater).

A PWR consists of the primary cooling system, chemical and volume control system, emergency core cooling system, container spray system, residual heat removal system, fuel handling system, waste processing system, turbine-generator system etc.

Following explanations are on a WH (Mitsubishi in Japan) type PWR as an example of PWR.

2.2. Structure of PWRs

(1) PWR pressure vessel and internals

The structure in the reactor vessel of 4-loop PWR of 1,100 MW class is shown in Figure 5. In the reactor pressure vessel, there is a reactor core consisting of fuel assemblies and control rods (control rod clusters) in the center and guide tubes, control rod drive systems, etc. in the upper region of the reactor core, and baffle plates and thermal shields, etc. in the periphery of the reactor core.

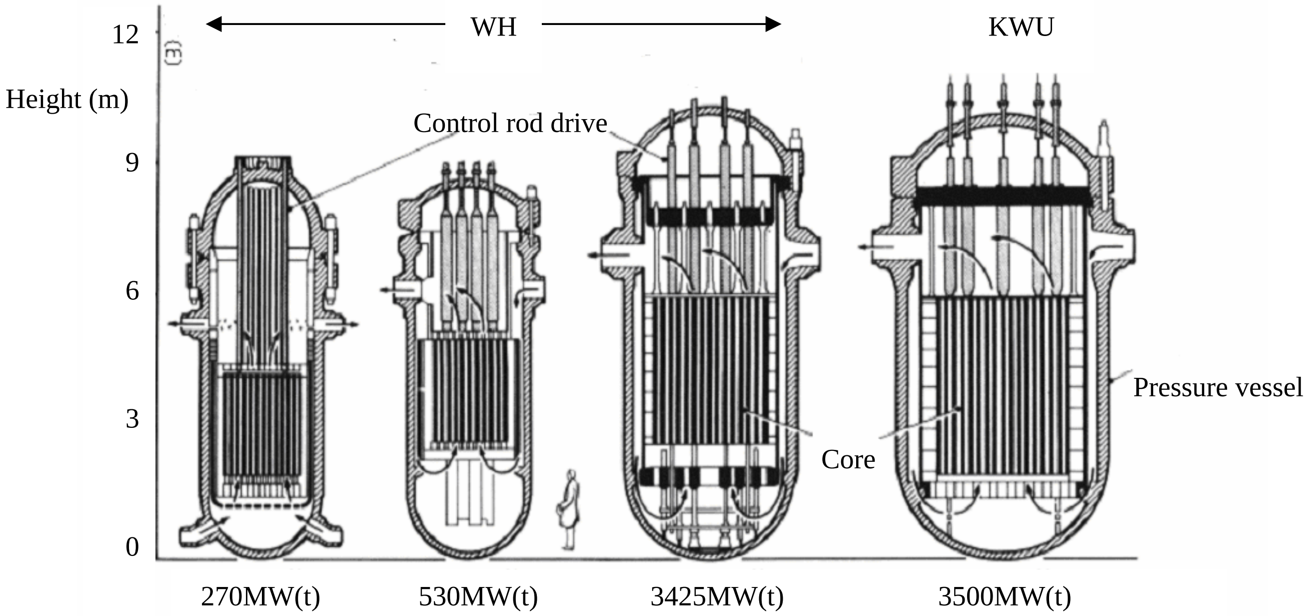


Figure 5 History of PWR Pressure Vessel (WH)

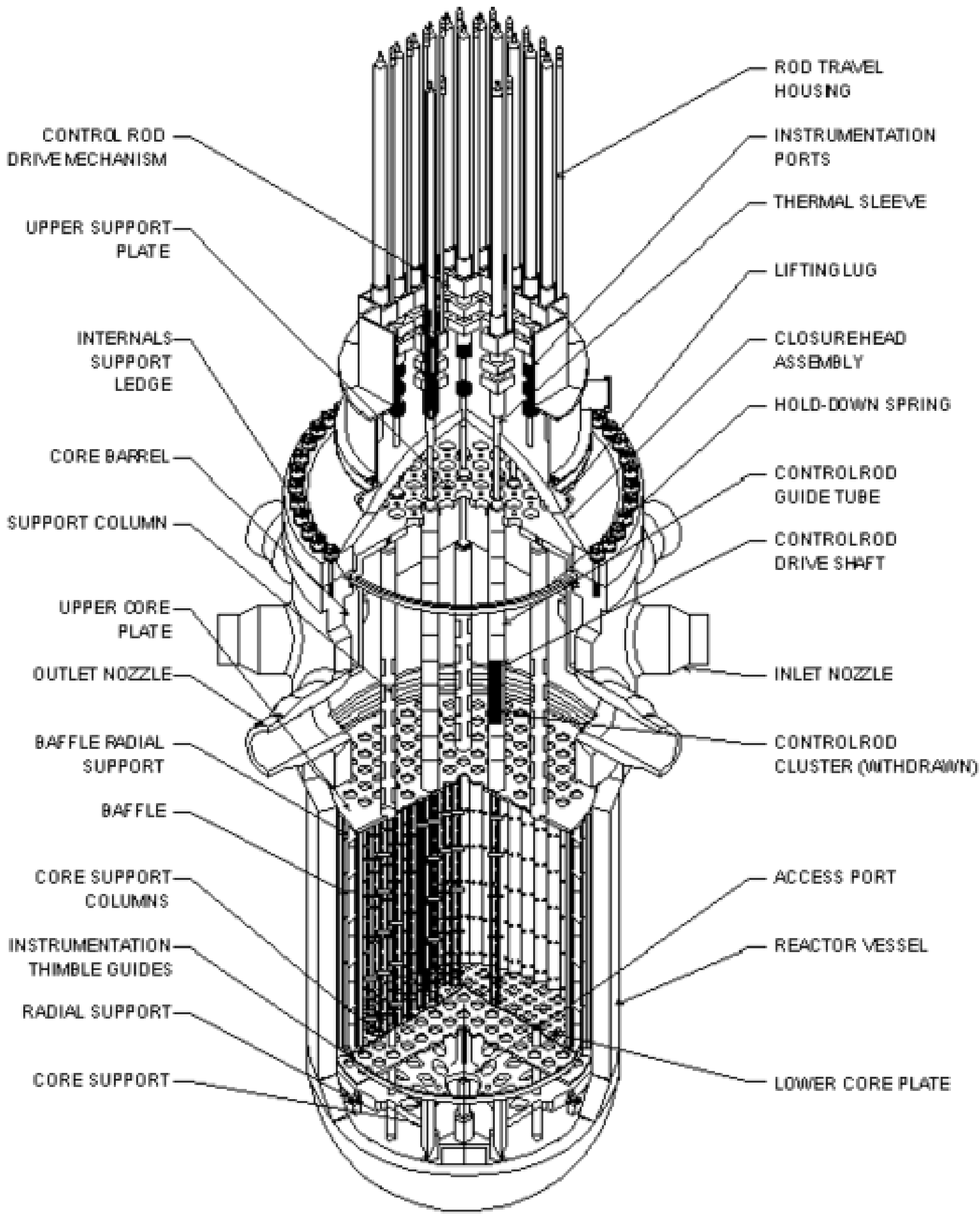


Figure 6 PWR Pressure Vessel Internals

(2) Nuclear fuel

The structural drawing of a fuel assembly is shown in Figure 7. Fuel assembly, for an example, consists of 264 fuel rods arranged in a square array of 17x17, one in-core monitor guide thimble, 24 control rod guide thimbles, 9 support grids, one top nozzle and one bottom nozzle. The fuel rod consists of a Zircaloy-4 cladding tube with sintered pellets of low enrichment uranium dioxide inserted, which is provided with a stainless steel spring in the upper region, pressurized with helium gas, and sealed at both ends by welding Zircaloy-4 end plugs. The fuel rod is designed to have a plenum in the upper region and appropriate gap between pellets and the cladding tube in order to prevent excessive stress on the cladding tube or end-plug welds due to fission gas released from pellets, thermal expansion difference of the cladding tube and pellets and fuel density change with fuel burnup.

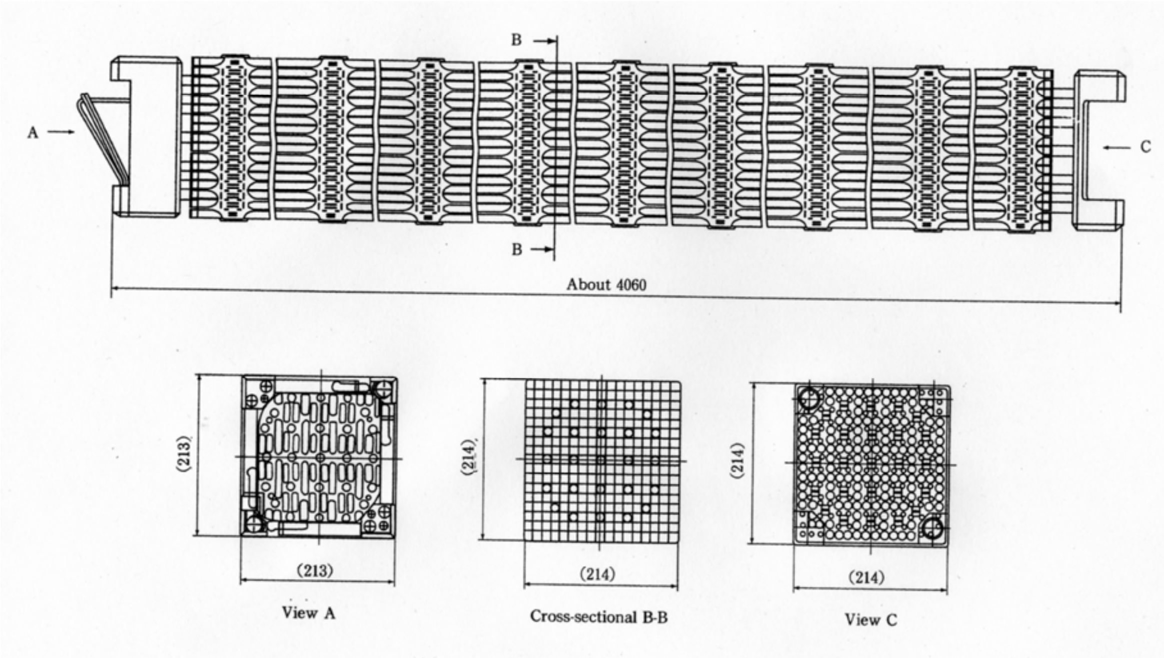


Figure 7 PWR 17x17 Fuel

(3) Control rod and its drive mechanism

The control rod cluster system with a bundle of 24 neutron absorbing rods in the upper region is adopted. As the neutron absorbing material, the Ag-In-Cd alloy is used. An example of the control rod drive system is shown in Figure 8. As the control rod drive system, there are a latch-type magnetic jack drive system and a motor drive screw system. Figure 8 shows a latch-type magnetic jack drive system, which drives the control rod by repeating magnetization and demagnetization of three kinds of actuation coils sequentially. As the reactivity control method in addition to the control rod cluster for short-term reactivity control, chemical shim by boric acid concentration adjustment of boric acid solution and burnable poison (neutron absorber) rod are used for long-term reactivity control.

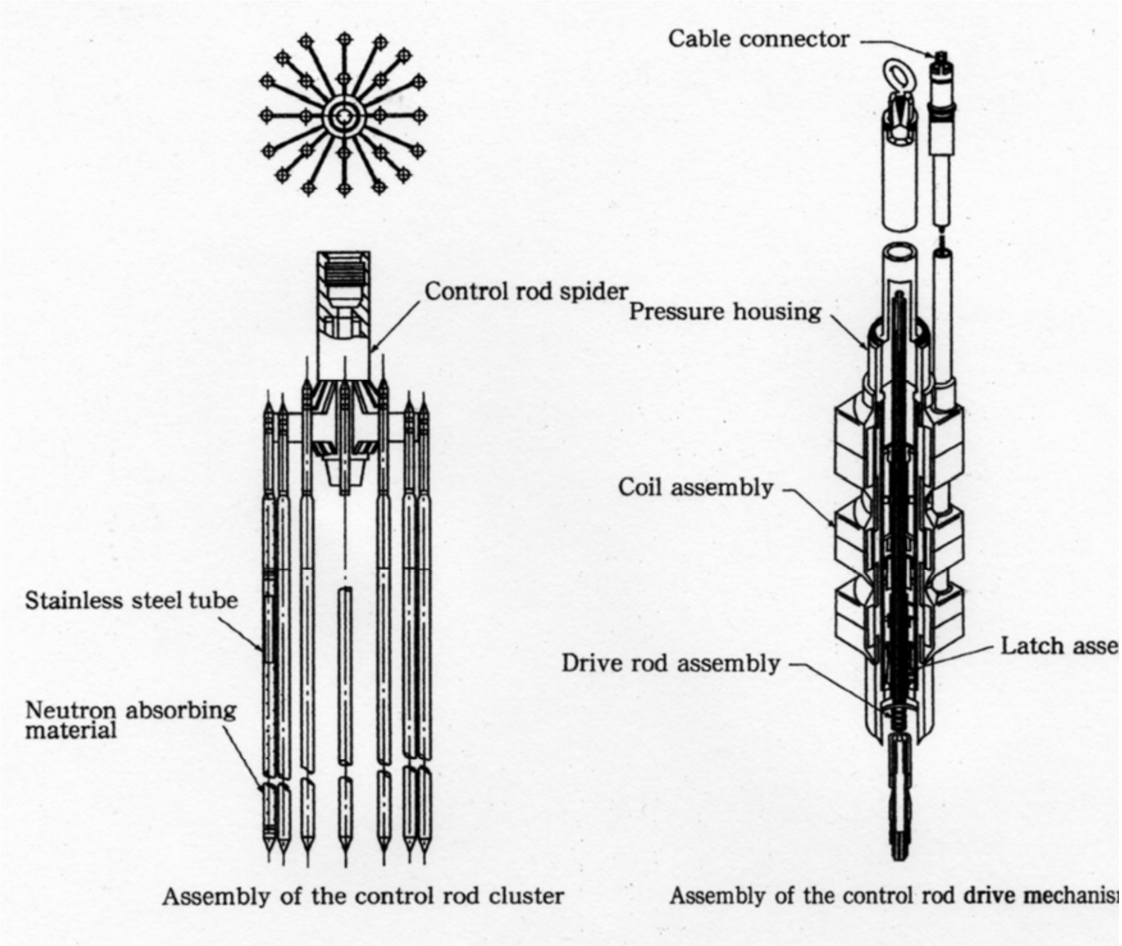


Figure 8 PWR Control Rod and Drive Mechanism

When an anomaly occurs or could occur in a nuclear reactor, all coils of the control rod drive system are de-energized, and then, the control rods drop rapidly into the reactor core by gravity, which leads to automatic shutdown of the nuclear reactor (it is reactor scram, but for PWRs, it is called reactor trip).

2.3 Primary Cooling System

(Source: Nuclear Power Generation Guide, 1999 Edition, Page 335, Edited by Nuclear Power Division, Public Utilities Department, Agency for Natural Resources and Energy, Ministry of International Trade and Industry, Published by Denryoku Shinpou Sha, October 1999)

The PWR cooling system is provided in order to circulate the reactor coolant heated with the thermal energy generated in a reactor core and to generate high-energy steam by conducting heat exchange at a steam generator to drive a turbine, and has the following functions;

(1) To circulate the reactor coolant heated in a reactor core, to transfer heat to a secondary

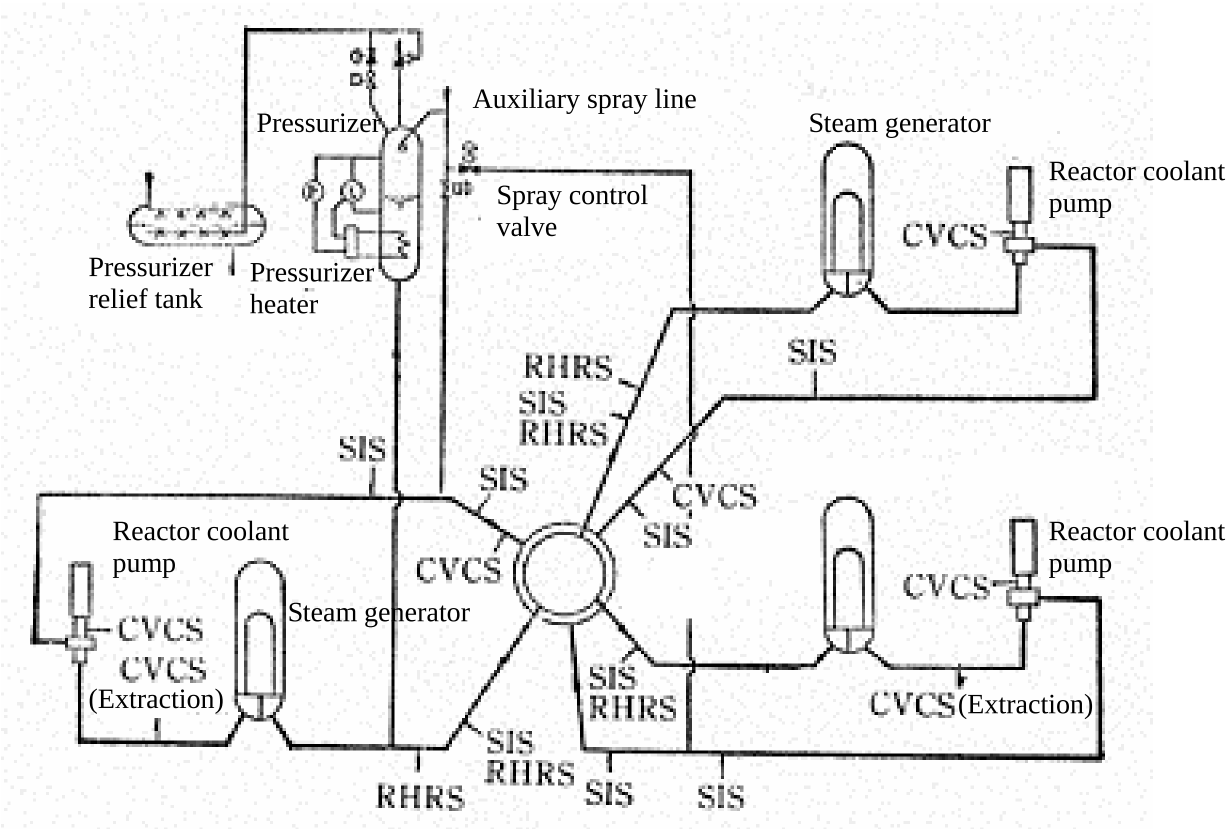
system with a steam generator, and to generate high-temperature and high-pressure steam to drive a turbine,

(2) To provide adequate core cooling in order not to cause any core damage during reactor operation,

(3) To be a coolant pressure boundary that is a barrier to prevent leakage of radioactive materials in the reactor coolant to the outside,

(4) To be moderator and reflector in addition to a core cooling function, and to contain the reactor coolant that has a role of solvent of the boron neutron absorber, and

(5) To control the primary system pressure constant with a pressurizer.



CVCS: Chemical and volume control system
SIS: Safety injection system
RHRS: Decay heat removal system

Figure 9 System Diagram of the Primary Cooling System (3 loops)

The major equipments that consist of the primary cooling system are described in the following;

(a) Reactor coolant pump

The reactor coolant pump, which is a vertical-type single-stage, circulates the coolant for heat transfer to a steam generator, and flow the coolant back into a nuclear reactor. This pump seal consists of three-seal structures to prevent coolant leakage to the outside. Moreover, a flywheel is provided in the top so as to ensure adequate coolant to a reactor core by extending the coast down of the pump even at a power loss.

(b) Steam generator

The steam generator is a vertical heat exchanger using heat transfer tubes of high-nickel alloy (Inconel), and such a structure to contain a steam-water separator and a moisture separator in the upper region. The coolant flows in from a bottom inlet nozzle, transfers heat to the heat transfer tubes, and flows out from an outlet nozzle. The feed water to the secondary side is supplied through a feedwater piping, flows down between a lower shell and an internal shell, and flows upward in the heat-transfer-tube bundle after changing the direction at a tube sheet, absorbing the heat from the coolant, and a part of the water becomes steam. The mixture of rising steam and water is separated with the steam-water separator, the water circulates as feedwater again, and the steam flows passes the moisture separator and flows out from a top outlet nozzle.

(c) Pressurizer

The pressurizer is provided with a liquid-immersion-type heater at the bottom end, and a spray, safety valves and relief valves in the upper region, and during normal operation, the inside of pressurizer is liquid in the lower one-half and steam in the upper one-half. The pressurizer and the high-temperature leg of the primary cooling system are connected with a surge pipe, which absorbs a pressure surge of the primary cooling system due to load fluctuation. Namely, when pressure of the reactor coolant system rises, the pressurizer pressure is lowered by water-spray operation. And, when pressure of the reactor coolant system goes down, heating by the heater raises the pressure, and the pressurizer acts to maintain the primary-coolant pressure at a rated value.

2.4 Chemical and Volume Control System

The chemical and volume control system extracts some part of the primary coolant from the primary-coolant cold leg and returns it to another primary-coolant cold leg after passing it through a makeup line, and consists of components, piping and valves. The functions of the system are coolant makeup to the primary cooling system, removal of corrosion products and fission products from the coolant, adjustment of the boric acid concentration, and supply of the shaft seal water to the seal area of the reactor coolant pumps etc.

2.5 Emergency Core Cooling System (ECCS)

The outline of the emergency core cooling system (ECCS) is shown in Figure 10. In case of the primary coolant discharge from a reactor core at a pipe break accident of a reactor system, etc., ECCS is provided to prevent a fuel failure. The ECCS consists of the accumulator system, high-pressure core injection system, and low-pressure core injection system.

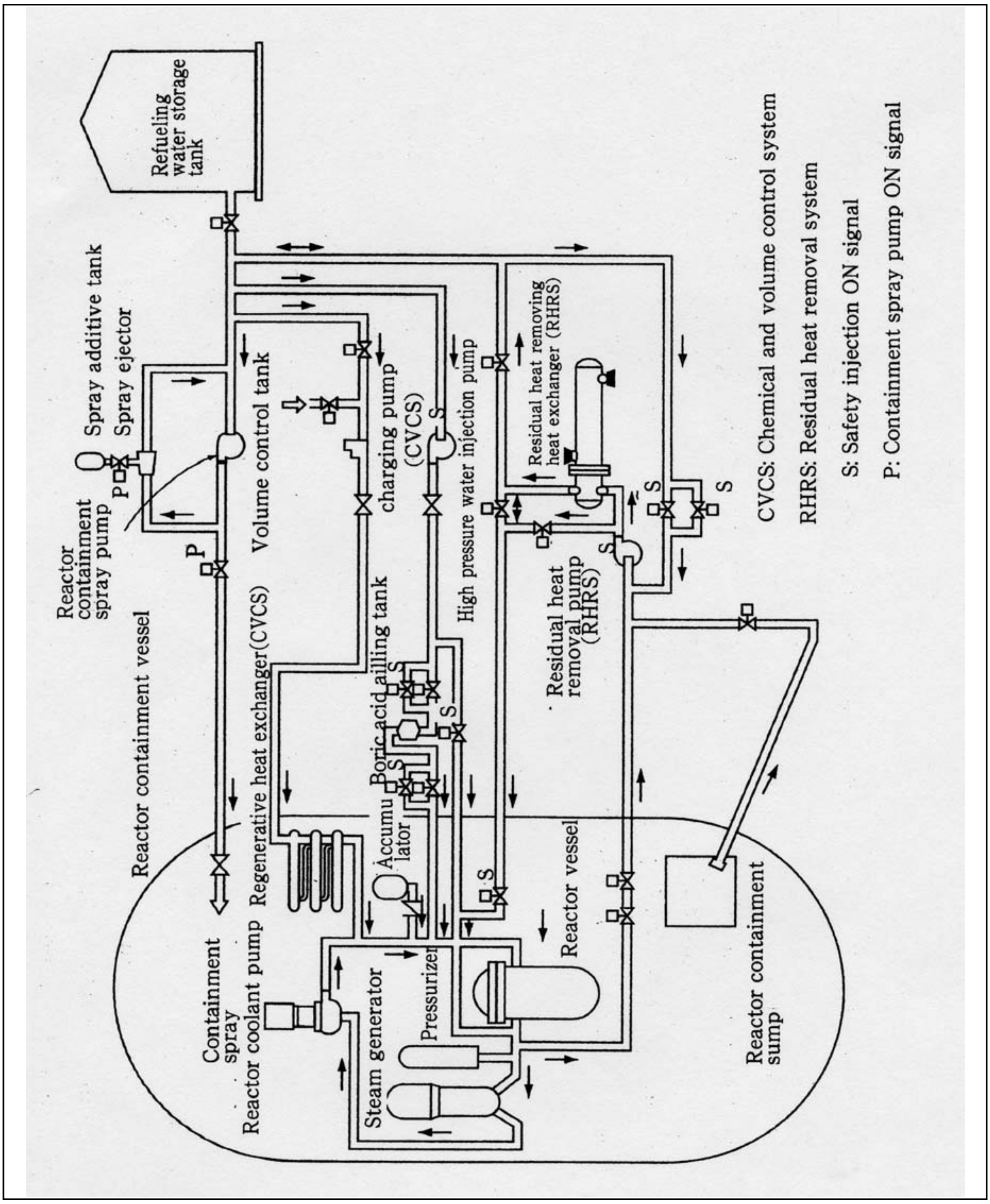


Figure 10 ECCS Network for PWR, 1100 Mwe

[More Details on Safety Design]

The reactor containment is provided so that the reactor coolant would not be released, since the radioactive materials are contained in the high-temperature and high-pressure reactor coolant. The PWR reactor containment contains components of the reactor coolant system including a reactor vessel and the engineered safety features system connected to this. Since the temperature and pressure in a reactor containment rise due to fuel decay heat after an accident, the containment spray system to spray the cooling water from the upper part of the reactor containment is provided for cooling the inside of the reactor containment to reduce the temperature and pressure and to remove radioactive materials, such as iodine, in the air. Also, in order to prevent a release of radioactive materials to the outside the reactor containment, the air leaked to the annulus region of reactor containment (between an internal cylinder and external cylinder) is led to the annulus air cleanup system and processed.

2.6. Containment Spray System

The containment spray system is provided to mix a radioactive-iodine-removal chemical in the borated water of a refueling water storage tank, to spray it from spray nozzles installed in the reactor containment, and to cool down and condensate the steam in the containment to reduce the internal pressure and to reduce iodine concentration in the steam by absorbing the iodine with the spray water droplet at a loss-of-primary-coolant accident etc. Moreover, to provide for the long-time continuous spray, the recirculation line from the reactor-containment sump is provided.

2.7 Residual Heat Removal System

The residual heat removal system has functions to remove the residual heat from the reactor core, to cool down the nuclear reactor during plant shut down and refueling, and to provide low-pressure coolant injection in case of a loss of coolant accident.

2.8 Waste Processing System

The wastes generated in a plant are categorized into liquid, gas and solid, and processed separately.

The liquid waste is, after being collected from each generating source, processed with a boric acid evaporator or a waste evaporator. The processed distilled water is reused as make-up water or released to the outside. The liquid waste condensed with the waste evaporator is stored in a storage facility after being loaded into drums. In addition, as solidification facilities, there are a bituminization facility, plastic-solidification facility, and cement vitrification facility, etc.

The gaseous waste is stored in a gas decay tank connected to each vent-header for a certain time period for attenuating the radioactivity, and is discharged from a stack with monitoring.

2.9 Fuel Handling System

The outline of the fuel handling system is shown in Figure 11.

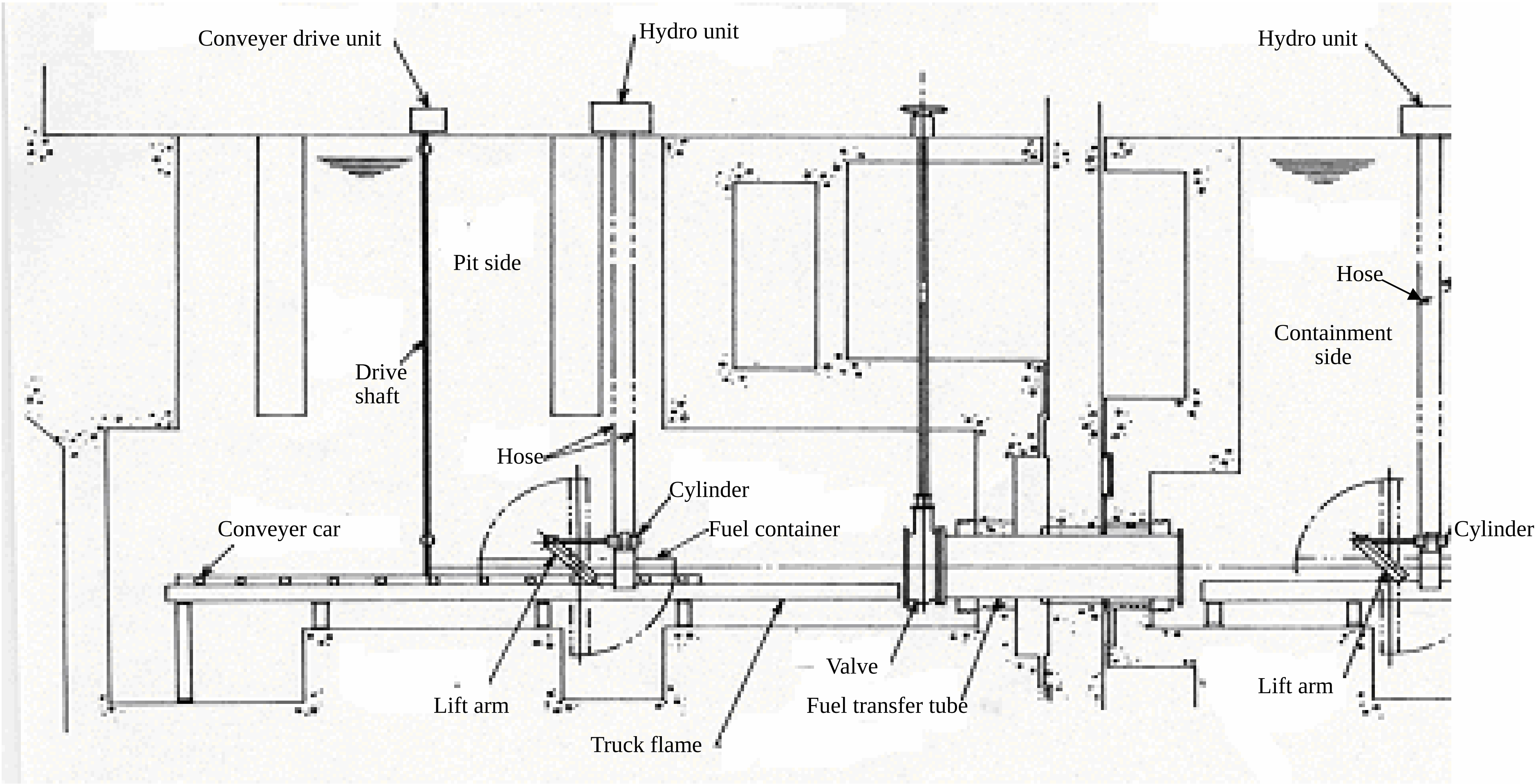


Figure 11 Outline of PWR Fuel Handling System

2.10 Turbine-Generator System

The PWR plant is separated completely into the primary system and the secondary system. Since the steam generated in a steam generator does not contain the radioactivity, the turbine-generator system is essentially the same to that of the conventional thermal power plant. But, as the generated steam is a low-pressure saturated one, it is necessary to use much more steam compared with the conventional thermal power plant and 4 pole turbines (1, 500 or 1,800 rpm) are used due to large turbines.

3. Power Control of PWR

(1) Power control method

As the reactor coolant (primary coolant) of PWR maintains the high-pressure of 157 kg/cm^2 over the entire region of the flow path with the pressurizer, boiling would not occur in the reactor core even if the power fluctuates, and the liquid phase is kept. Therefore, since neutrons are slowed-down sufficiently all over the reactor core, the "control rod cluster" system, which distributes neutron-absorption rods in a fuel assembly, is adopted for power (reactivity) control. In addition to this control rod cluster system, the chemical shim control system is used. The "chemical shim control" means to use the boric acid solution (neutron absorber) in the reactor coolant for reactivity control. By using this chemical shim control, the spatial distribution of the power in a reactor core is flattened and the maximum to average ratio of reactor core power density becomes small. The number of reactor coolant circulation loop (number of steam generators) increases corresponding to the reactor power for PWRs of WH families. For PWR plants, the pressure of reactor coolant is controlled to be constant by operation of heaters and the water spray in the pressurizer, which makes smooth expansion and shrinkage of the reactor coolant accompanying temperature fluctuation.

(2) Load fluctuation and reactor pressure reduction

What is the most important for operation of a nuclear reactor and a turbine/generator is that the reactor power follows the turbine load. PWR plants utilize the load-following characteristics of reactor power depending on the moderator temperature coefficient.

For example, as the turbine load increases resulting in an increase in evaporation (steam flow) in a steam generator, the reactor-coolant inlet temperature decreases and the positive reactivity is added, which increases the nuclear reactor power. In a small range of load fluctuation, such an effect of the moderator temperature coefficient adjusts the reactor power automatically. When the load fluctuation is large, automatic control rod operation comes into play to control the reactor power. Namely, PWR plants use the control system of "turbine master / reactor slave (load priority)."

4. PWRs in the World

Pressurized water reactors (PWRs) are classified roughly into WH (Westinghouse Electric) type, B&W (Babcock & Wilcox) type, CE (Combustion Engineering) type, and VVER type (Russia type) PWRs. For WH-type, B&W-type and CE-type PWRs, even though the principle that those consist of a primary system and a secondary system is the same, but there are differences in some points, such as configurations of primary cooling system and structures of steam generator.

(1) WH PWRs

WH-type PWRs increased the number of steam generators and reactor coolant pumps following the increase in the electrical output (large-scale plant). For 1000 MWe class WH-type PWRs, as shown in Figure 12, the primary cooling system consists of four loops (four steam generators and four primary coolant pumps). The steam generator of WH-type PWRs is shown in Figure 12. The steam generator is designed as a vertical U-tube recirculating system to generate saturated steam. In such a design, Mitsubishi PWR (Japan), Framatome PWR (France) and Siemens (KWU) PWR (Germany) belong to the WH-type PWR.

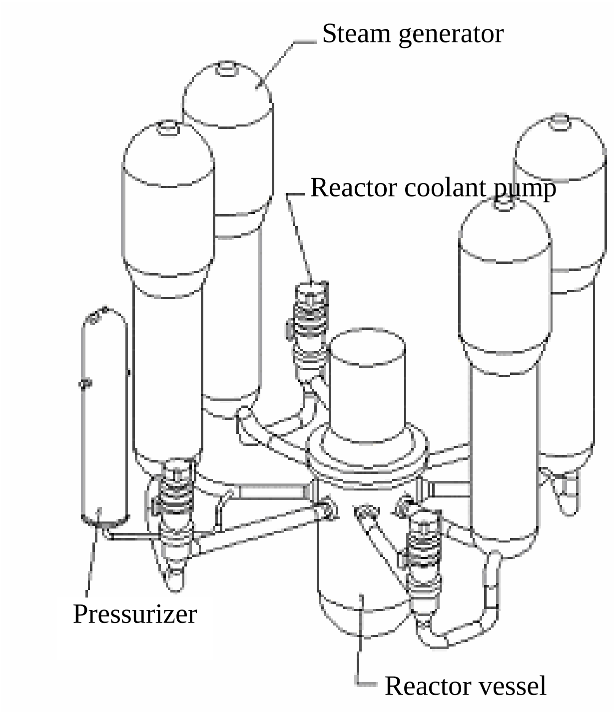


Figure 12 PWR Reactor Cooling System (4 Loops, WH)

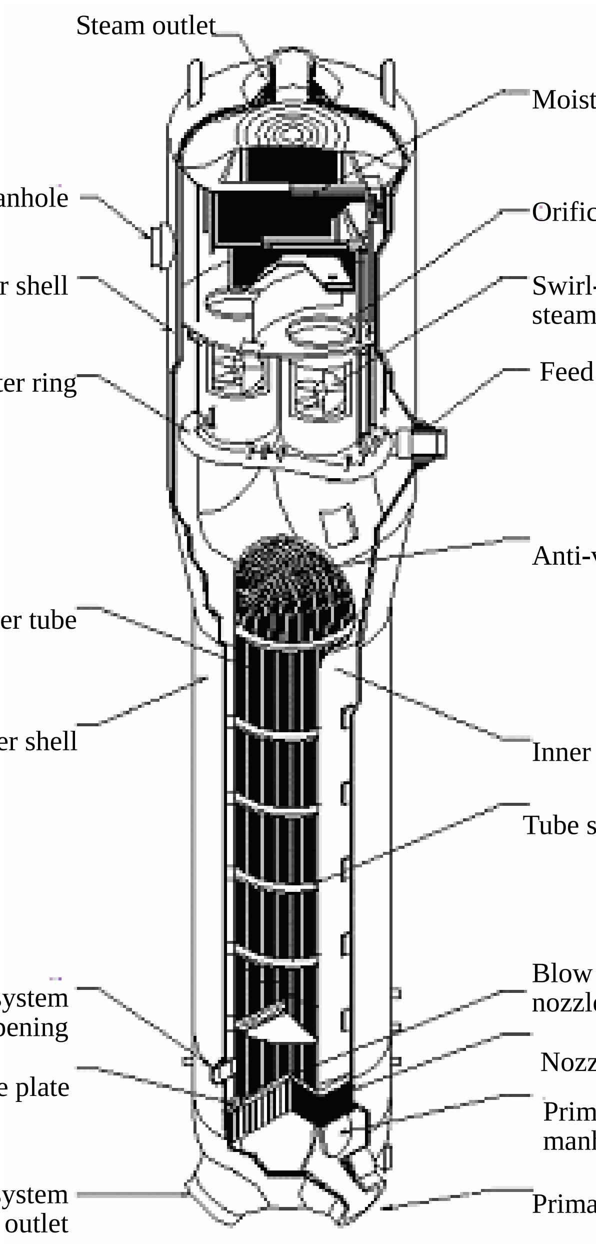
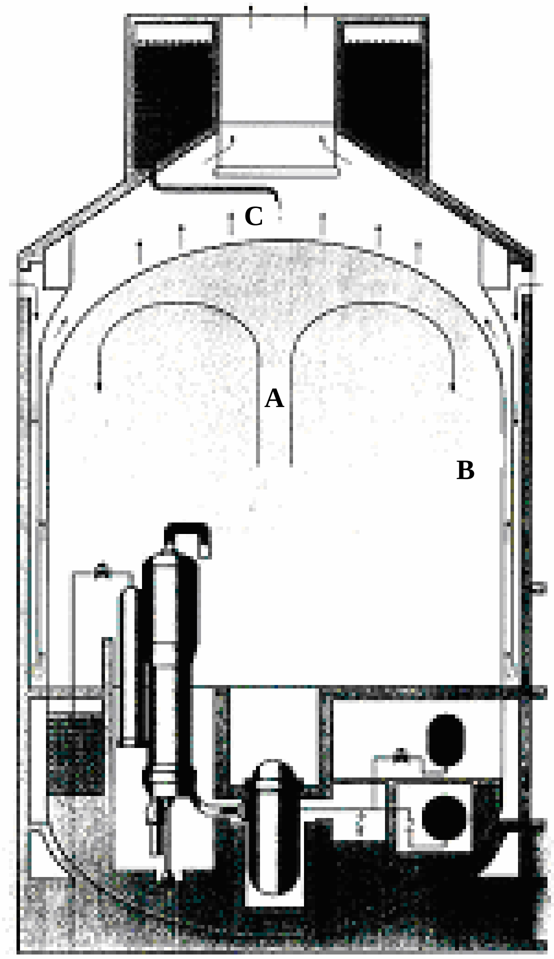


Figure 13 PWR Steam Generator (WH)

The AP600 (600 MWe) is a medium-size reactor adopting a passive-safety design that was developed by WH according to the Advanced Light Water Reactor (ALWR) Program of DOE/EPRI. The reactor is designed to adopt passive cooling (for an example of reactor containment, see Figure 14), and to aim at simplification (reduction of about 50% of valves, about 80% of piping, about 70% of power cables, etc. compared with conventional reactors), modularization, high public safety (no operator action is required for three days after an accidents), easy licensing, and short construction period (three years). As mentioned later, the combination of two steam generators and four primary coolant pumps of the plant is the same configuration as that of the reactor cooling system of B&W-type and CE-type PWRs. The design has been already approved by the U.S. NRC.



A: Heat transfer to the containment wall by the internal natural circulation and condensation

B: Heat transfer to the atmosphere by outside natural draft

C: Heat removal acceleration by the gravity spray of tank water

Figure 14 AP600 Passive Containment Cooling System

(2) Framatome PWR

The N4 reactor (1,516 MWe) is a twin-type standardized APWR developed by Framatome improving the central control room etc. and increasing reliability and economical efficiency with new knowledge obtained from lessons learned from the TMI plant accident (USA, 1979) Chooz-B1 & B2 and Civaux-1 & 2 are already in commercial operation. The EPR (the Europe pressurized water reactor, 1,500 MWe) is a large-sized reactor currently being developed jointly by France (Framatome) and Germany (Siemens), and the preliminary design has been completed. The plant design is based on the

N4 and Konvoy (Germany-type PWR) and aims at improvement of safety (for a severe accident etc.), simple operation, competitive economical efficiency (60-year lifetime, high-burnup 70 GWd/t), and MOX fuel (used up to 50% of the fuel).

(3) PWRs of other suppliers

The configuration of the reactor cooling system of CE-type PWRs consists of two loops (two steam generators and four primary coolant pumps) as shown in Figure 15. Namely, the reactor coolant coming out of two steam generators is returned to the reactor vessel with four reactor coolant pumps. The steam generator is almost the same type as one of WH-type PWR, which is a vertical reverse U-tube recirculating system to generate saturated steam (Figure 16). Furthermore, System80+ (1,300 MWe) that is modified from Sytem80 is a next-generation large-sized reactor significantly improved in safety and economical efficiency.

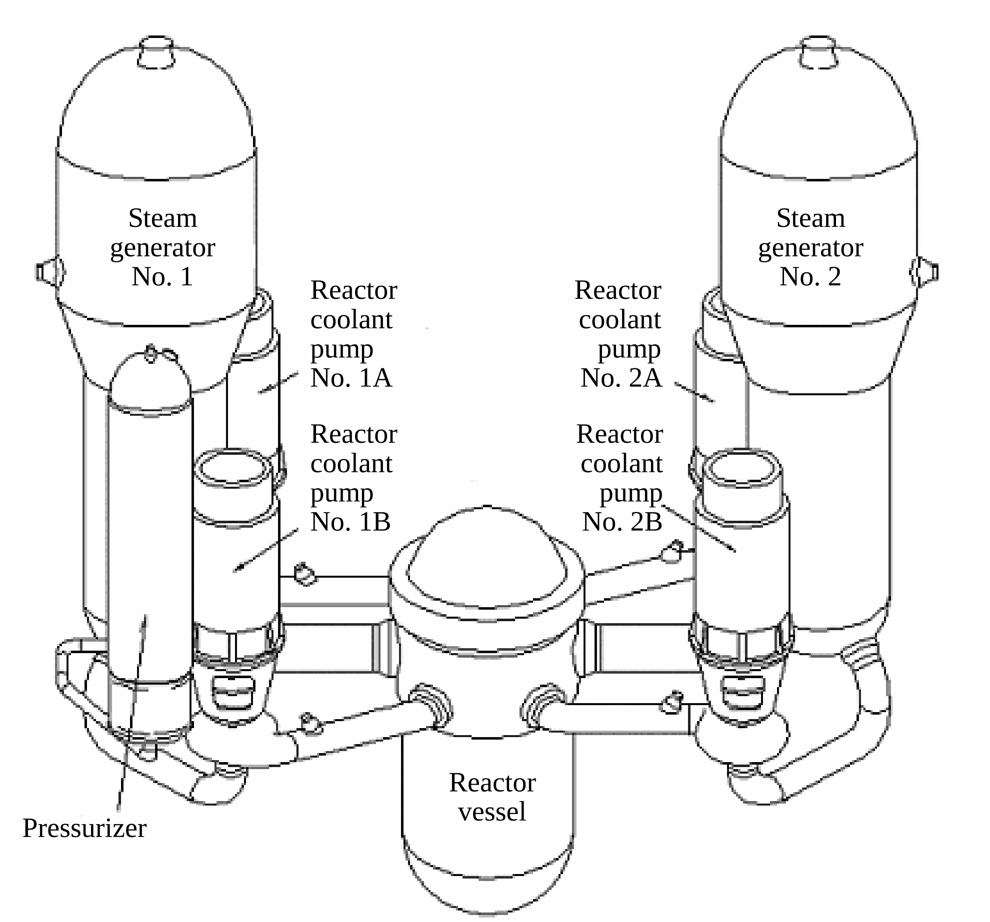


Figure 15 CE PWR Cooling System

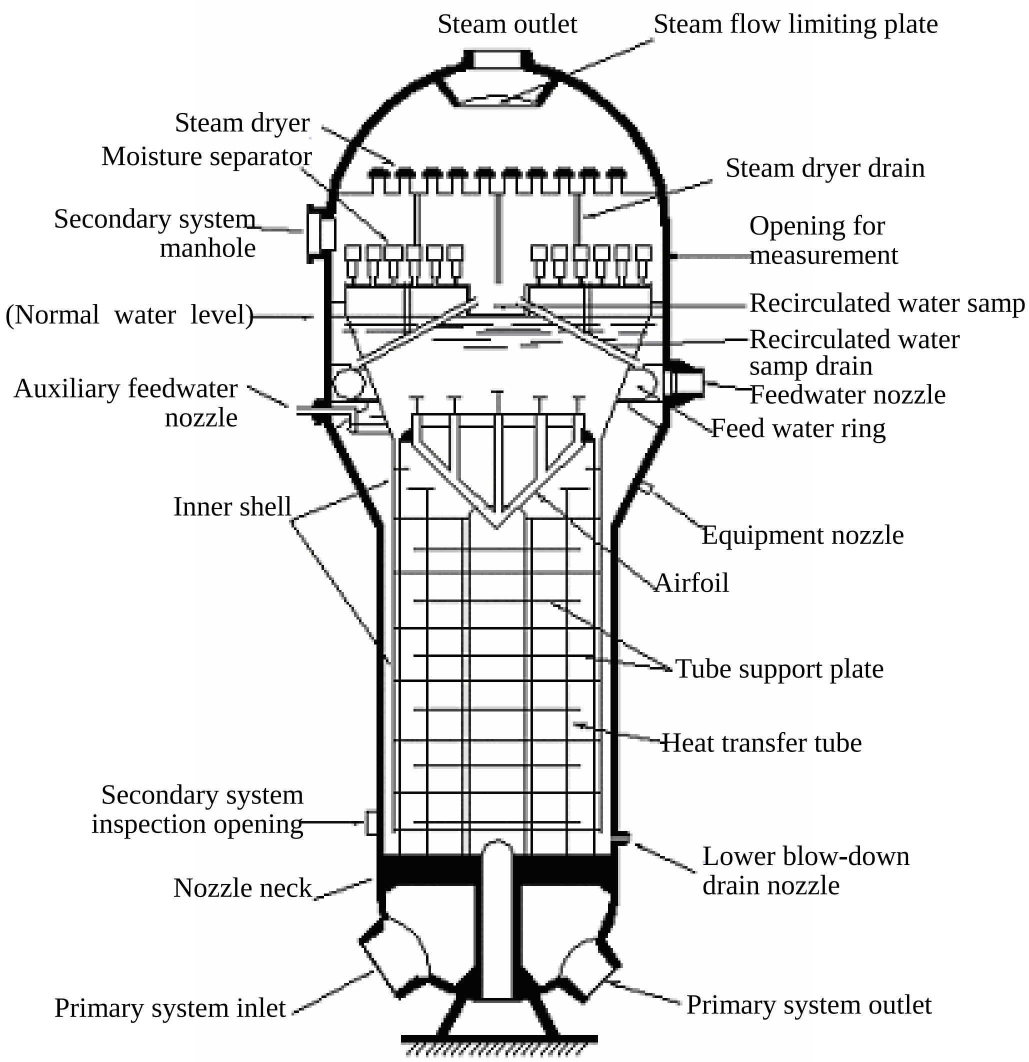


Figure 16 CE PWR Steam Generator

B&W-type PWRs have the same reactor cooling system configuration as CE-type PWRs, which consist of two loops (two steam generators and four reactor coolant pumps). The steam generator is a vertical straight-pipe once-through type to generate superheated steam. The nuclear reactor of TMI power station that caused an accident in 1979 is a B&W-type PWR.

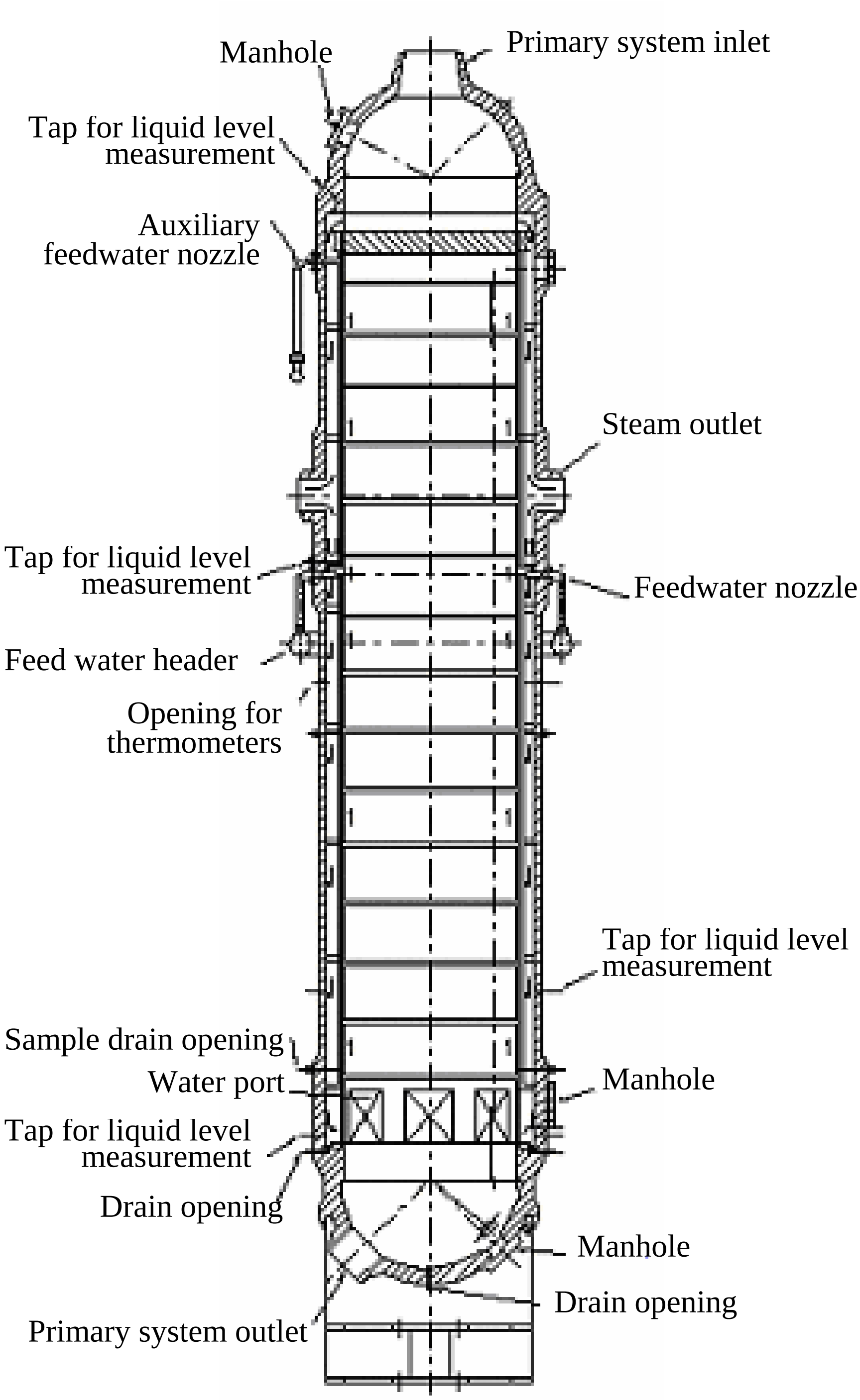


Figure 17 B&W PWR Steam Generator

VVER belongs to the WH-type PWR in terms of its reactor cooling system. The large differences from Western Europe-type PWRs are that the fuel assembly is hexagonal lattice arrangement (tetragonal lattice arrangement for Western Europe-type PWRs) and the steam generator is a horizontal type. Even though the Western Europe side has pointed out the safety issues on old types (VVER-440/V-230 etc.), the VVER-1000 (1000 MWe class) is designed to ensure the same safety level as that of Western Europe-type PWRs.

5. Features of PWR

PWRs (also VVER, Russian design) are generation II nuclear power reactors that use high-pressure ordinary water as coolant and neutron moderator. The primary coolant loop is kept under high pressure to prevent the water from reaching film boiling, hence the name PWR. PWRs are the most common type of power producing reactor and are widely used all over the world. More than 230 of them are in use to generate electric power, and several hundred more for naval propulsion. They were originally designed by the Bettis Atomic Power Laboratory as a nuclear submarine power plant.

A PWR works because the nuclear fuel in the reactor vessel is engaged in a chain reaction, which produces heat, heating the water in the primary coolant loop by thermal conduction through the fuel cladding. The hot water is pumped into a heat exchanger called steam generator, which allows the primary coolant to heat up and boil the secondary coolant. The transfer of heat is accomplished without mixing the two fluids. This is desirable, since the primary coolant is necessarily radioactive. The steam formed in the steam generator is allowed to flow through a steam turbine, and the energy extracted by the turbine is used to drive an electric generator. After passing through the turbine the secondary coolant (water-steam mixture) is cooled down and condensed in a condenser before being fed into the steam generator again. This converts the steam to a liquid so that it can be pumped back into the high pressure steam generator.

Two things are characteristic for the pressurized water reactor (PWR) when compared with other reactor types:

In a PWR, there are two separate coolant loops (primary and secondary), which are both filled with ordinary water (also called light water). A boiling water reactor, by contrast, has only one coolant loop, while more exotic designs such as breeder reactors use substances other than water for the task (e.g. sodium in its liquid state).

The pressure in the primary coolant loop is typically 15-16 Megapascal, which is notably higher than in other nuclear reactors, and nearly twice that of a boiling water reactor (BWR). As an effect of this, only localized boiling occurs and will recondense promptly in the bulk fluid. By contrast, in a boiling water reactor the primary coolant is designed to boil.

Note on Boiling in PWRs

Both PWRs and BWRs are designed such that the water in the core will never reach film boiling during normal operation. In PWRs, however, the water is not required to boil, it just occurs as a consequence of the conditions that the core is operated at. A side effect of localized boiling is that the heat transfer rate to the coolant is increased. This effect is taken advantage of in PWR designs to improve safety margins and efficiency.

5.1. PWR Design

(1) Coolant

Borated water is used as primary coolant in a PWR and flows through the reactor at a temperature of roughly 315 °C (600 °F). The water remains liquid despite the high temperature due to the high pressure in the primary coolant loop (usually around 2200 psig [15 MPa, 150 atm]). The primary coolant loop is used to heat water in a secondary circuit that becomes saturated steam (in most designs 900 psia [6.2 MPa, 60 atm], 275 °C [530 °F]) for use in the steam turbine.

(2) Moderator

Pressurized water reactors, like thermal reactor designs, require the fast fission neutrons in the reactor to be slowed down (a process called moderation) in order to sustain its chain reaction. In PWRs the coolant water is used as a moderator by letting the neutrons undergo multiple collisions with light hydrogen atoms in the water, losing speed in the process. This "moderating" of neutrons will happen more often when the water is more dense (more collisions will occur). The use of water as a moderator is an important safety feature of PWRs, as any increase in temperature causes the water to expand and become less dense; thereby reducing the extent to which neutrons are slowed down and hence reducing the reactivity in the reactor. Therefore, if reactor activity increases beyond normal, the reduced moderation of neutrons will cause the chain reaction to slow down, producing less heat. This property, known as the negative temperature coefficient of reactivity, makes PWRs very stable. In contrast, the RBMK reactor design used at Chernobyl (using graphite instead of water as the moderator) greatly increases heat generation when coolant water temperatures increase, making them very unstable. This flaw in the RBMK reactor design is generally seen as one of several causes of the Chernobyl accident.

(3) Fuel

The PWR fuel bundle is from a pressurized water reactor of the nuclear passenger and cargo ship NS Savannah. Designed and built by the Babcock and Wilcox Company.

The uranium used in PWR fuel is usually enriched several percent in ^{235}U. After enrichment the uranium dioxide (UO_2) powder is fired in a high-temperature, sintering furnace to create hard, ceramic pellets of enriched uranium dioxide. The cylindrical pellets are then put into tubes of a corrosion-resistant zirconium metal alloy (Zircaloy) which are backfilled with helium to aid heat conduction and detect leakages. The finished fuel rods are grouped in fuel assemblies, called fuel bundles, that are then used to build the core of the reactor. As a safety measure PWR designs do not contain enough fissile uranium to sustain a prompt critical chain reaction (i.e, substained only by prompt neutrons). Avoiding prompt criticality is important as a prompt critical chain reaction could very rapidly produce enough energy to damage or even melt the reactor (as is suspected to have occurred during the accident at the Chernobyl plant). A typical PWR has fuel assemblies of 200 to 300 rods each, and a large reactor would have about 150-250 such assemblies with 80-100 tonnes of uranium in all. Generally, the fuel bundles consist of fuel rods bundled 14x14 to 17x17. A PWR produces on the order of 900 to 1500 MWe. PWR fuel bundles are about 4 meters in length.

Refuelings for most commercial PWRs is on an 18-24 month cycle. Approximately one third of the core is replaced each refueling.

(4) Control

Generally, reactor power can be viewed as following steam (turbine) demand due to the reactivity feedback of the temperature change caused by increased or decreased steam flow. Boron and control rods are used to maintain primary system temperature at the desired point. In order to decrease power, the operator throttles shut turbine inlet valves. This would result in less steam being drawn from the steam generators. This results in the primary loop increasing in temperature. The higher temperature causes the reactor to fission less and decrease in power. The operator could then add boric acid and/or insert control rods to decrease temperature to the desired point.

Reactivity adjustments to maintain 100% power as the fuel is burned up in most commercial PWR's is normally controlled by varying the concentration of boric acid dissolved in the primary reactor coolant. The boron readily absorbs neutrons and increasing or decreasing its concentration in the reactor coolant will therefore affect the neutron activity correspondingly. An entire control system involving high pressure pumps (usually called the charging and letdown system) is required to remove water from the high pressure primary loop and re-inject the water back in with differing concentrations of boric acid. The reactor control rods, inserted through the top directly into the fuel bundles, are normally only used for power changes. In contrast, BWRs have no boron in the reactor coolant and control the reactor power by adjusting the reactor coolant flow rate.

5.2 Advantages

- PWRs are very stable due to their tendency to produce less power as temperatures increase, this makes the reactor easier to operate from a stability standpoint.

- PWRs can be operated with a core containing less fissile material than is required for them to go prompt critical. This significantly reduces the chance that the reactor will run out of control and makes PWR designs relatively safe from criticality accidents.

- Because PWRs use enriched uranium as fuel they can use ordinary water as a moderator rather than the much more expensive heavy water.

- PWR turbine cycle loop is separate from the primary loop, so the water in the secondary loop is not contaminated by radioactive materials.

Part 2. Advance Pressurized Water Reactor

6. Next generation designs

Advanced pressurized water reactors are types of nuclear reactors which are improved version of existing PWR types. Examples include Westinghouse's AP600 and AP1000, AREVA's EPR/US-EPR and Mitsubishi's US-APWR.

Westinghouse has obtained approval of two standard plant types; the AP600 and the AP1000, the later of which is expected to see construction before 2010. An EPR is currently being constructed in Finland. Mitsubishi's US-APWR is in the pre-application stage of the NRC's licensing process.

6.1 Design Innovations

In this design the safety systems mainly apply passive protection, which yield such high degree of safety that there is no need for the usual diesel generators, which provide the equipment with power in the case of a loss of electrical supply. They require little intervention, which reduces the chance of human error and other failures. Safety enhancement is also achieved by using modern, reliable devices.

The probability of failures is even more decreased by applying the concept of diversity: several and different type of systems are used and thus the effect of potential intrinsic failures can be avoided.

In a standard PWR design, cooling requires elaborate pumps, while in an advanced pressurized water reactors it can be handled by simple gravity flow with natural convection — cool water enters the bottom of the reactor, which heats it, causing it to rise because warm water is less dense. This process sets up a natural circulation driven only by gravity. Unlike pumps which can fail and are driven by electric power which may not always be available, gravity never stops working. This is what makes this design inherently safer. It is referred to "passive stability," since no active measures by operators or by mechanical or electrical control systems are required.

The design is less expensive to build partly due to the fact that it uses existing and proven technology. The expense is also reduced by rationalizing technology, which means decreasing not only the amount of pipes, wires, and valves necessary, but reducing a number of other components, and therefore reducing cost. Standardization and type-related licencing will also help reduce the time and cost of construction.

6.2 Westinghouse's AP600 and AP1000

Westinghouse has obtained approval of two standard plant types; the AP600 and the AP1000, the later of which is expected to see construction before 2010.

The AP600 is rated at 600 MWe, whereas the AP1000 is rated at 1117 to 1154 MWe, depending on the secondary coolant.

(1) AP600 (WH)

As part of the cooperative U.S. Department of Energy (DOE) Advanced Light Water Reactor (ALWR) Program and the Electric Power Research Institute (EPRI), the Westinghouse developed a simplified, safe, and economic 600-megawatt plant. Designed to satisfy the standards set by DOE and defined in the ALWR Utility Requirements Document (URD), the Westinghouse AP600 is a combination of innovative safety systems that rely on dependable natural forces and field-proven technologies.

The AP600 is very similar to the AP1000. Some of the differences include using a 12-foot fuel assembly in a less dense loading pattern, using smaller steam generators, smaller reactor coolant pumps, and a shorter reactor vessel and containment building. The AP600 has passive safety features characteristic of the AP concept. The projected core damage frequency is nearly 1000 times less than today's NRC requirements, on par with plants currently being considered to construction.

NRC final design certification was received in 1999 but no orders were ever placed. A large reason Westinghouse entered development of the AP1000 was to improve the economies of scale that come with larger MWe plants. The AP1000 was adapted to have a similar footprint but a taller containment and a power output of 1000 MWe or greater.

(2) AP1000 (WH)

The AP1000 is a proposed pressurized water reactor utilizing passive safety features designed and manufactured by Westinghouse Electric Company for nuclear power plants. This is considered a Generation III+ design. Each reactor is designed to generate over 1000-megawatts-electric (1117 to 1154 per Westinghouse). The AP1000 builds on the research for that plant, with the economies of scale of a larger plant reducing the cost per megawatt output.

The Westinghouse AP1000 is a logical extension of its AP600 plant. Design studies have shown that a two-loop configuration could produce over 1000 MWe with minimal changes in the AP600 design. The primary purpose of developing the AP1000 was to retain the AP600 design objectives, design details and licensing basis, while optimizing the power output, thereby reducing the resulting electric generation costs.

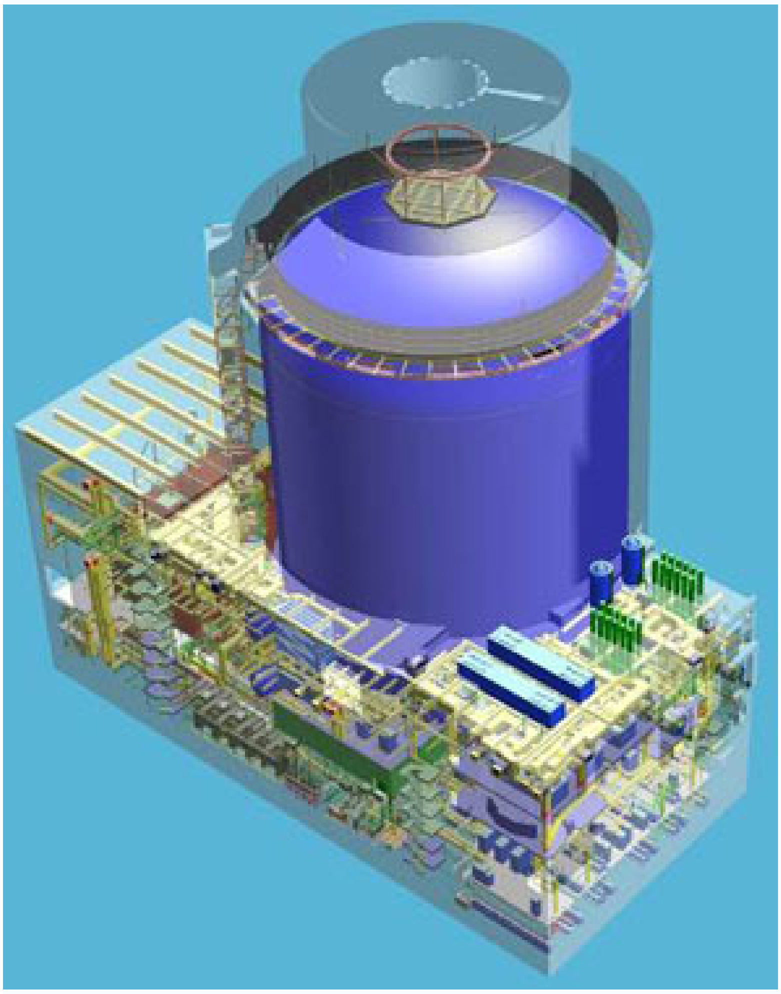


Figure 18 Overview of WH AP1000

(3) Features of AP1000

(a) SIMPLICITY

The safety systems in the AP1000 are passive, relying on things like gravity and natural recirculation instead of on active systems such as pumps. The passive core cooling system (PCCS) is the AP1000's passive analogue to the [emergency core cooling system] used in currently operating reactors. The PCCS is passive because none of the systems are reliant on AC power and the actuation for the safety systems is automatic. The valves required for alignment are usually fail-safe (requiring power to stay in their normal, closed positions) and are always powered by energy stored in batteries, springs, or compressed gas.

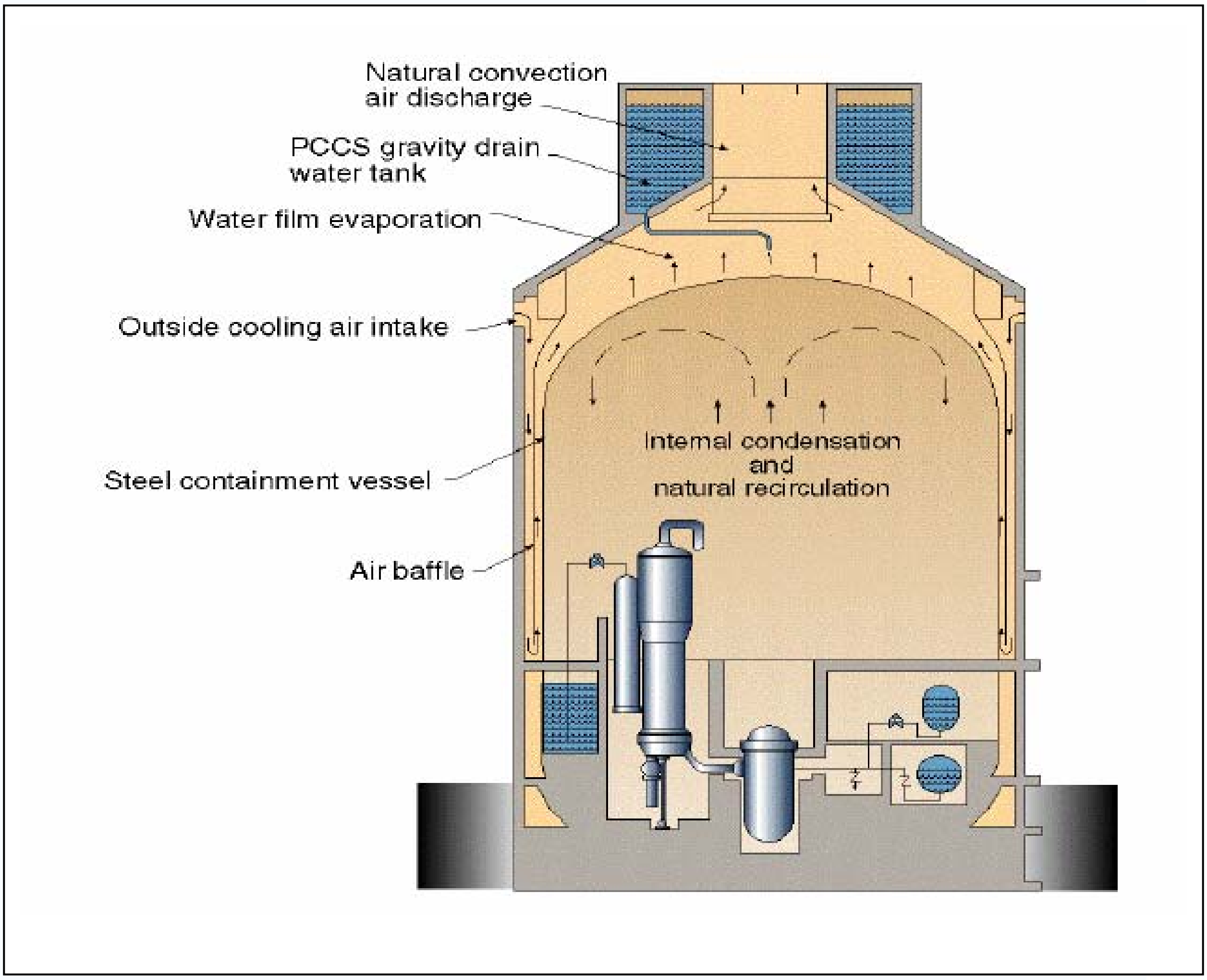


Figure 19 AP1000 Passive Containment Cooling System

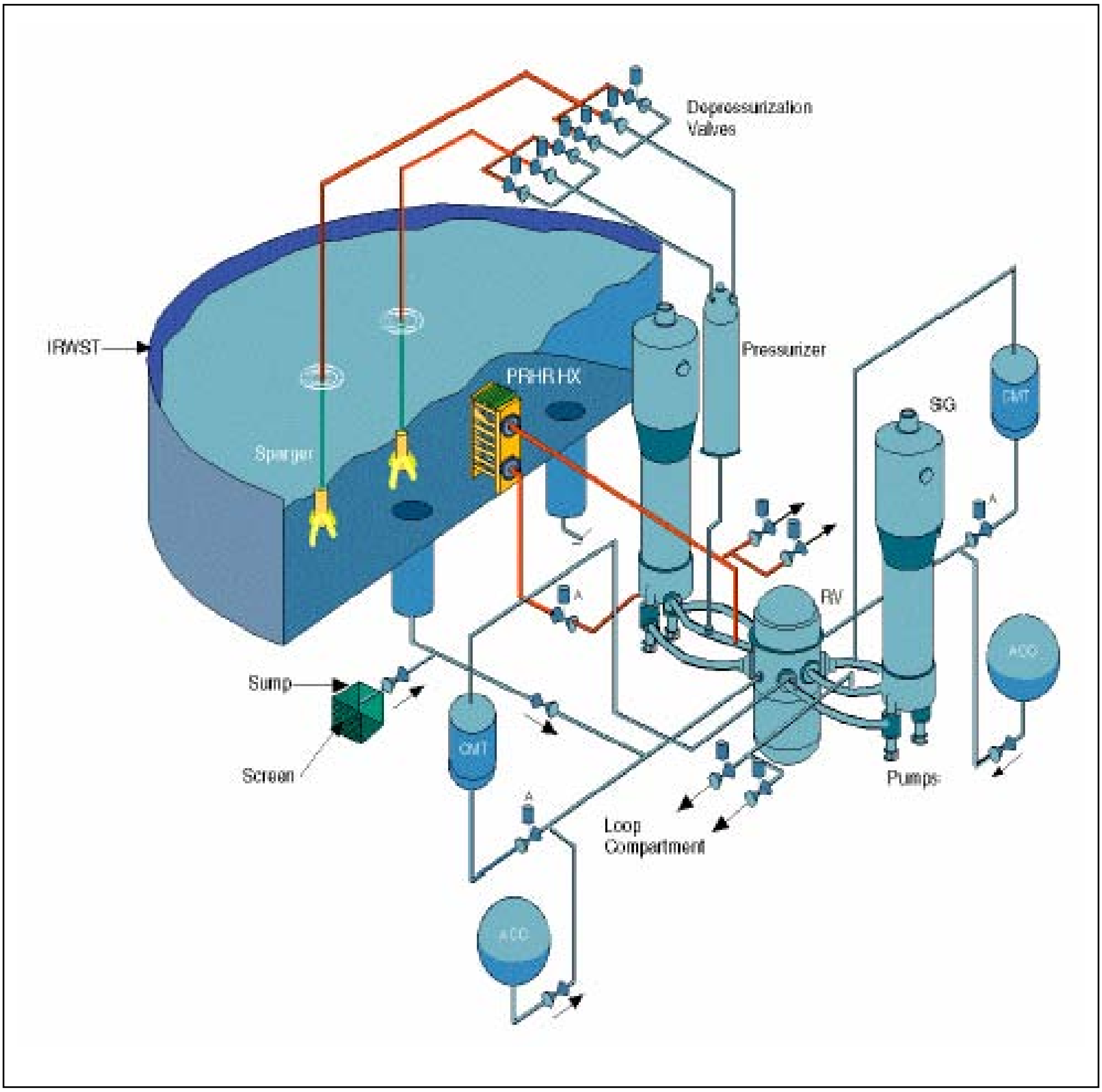


Figure 20 AP1000 Passive Core Cooling System

The passive safety systems are significantly simpler than the traditional PWR safety systems. They do not require the large network of safety support systems needed in typical nuclear plants, such as AC power, HVAC (heating, ventilation & air conditioning), cooling water systems and seismic buildings to house these components. Simplification of plant systems, combined with increased plant operating margins, reduces the actions required by the operator. The AP1000 has 50 percent fewer valves, 83 percent less piping, 87 percent less control cable, 35 percent fewer pumps and 50 percent less seismic building volume than a similarly sized conventional plant. These reductions in equipment and bulk quantities lead to major savings in plant costs and construction schedules.

(b) Nuclear Steam Supply System (NSSS) & Fuel

The AP1000 NSSS plant configuration consists of two Delta−125 steam generators, each connected to the reactor pressure vessel by a single hot leg and two cold legs. There are four reactor coolant pumps that provide circulation of the reactor coolant for heat removal. A pressurizer is connected to one of the cold leg piping to maintain subcooling in the reactor coolant system (RCS).

The two-loop, 1090 MWe plant retains the same basic design of the AP600. Changes to the design to increase the electricity output have been minimized to allow the direct application of most of the existing design engineering already completed for the AP600. Examples of design features that remain unchanged include the nuclear island footprint and the core diameter.

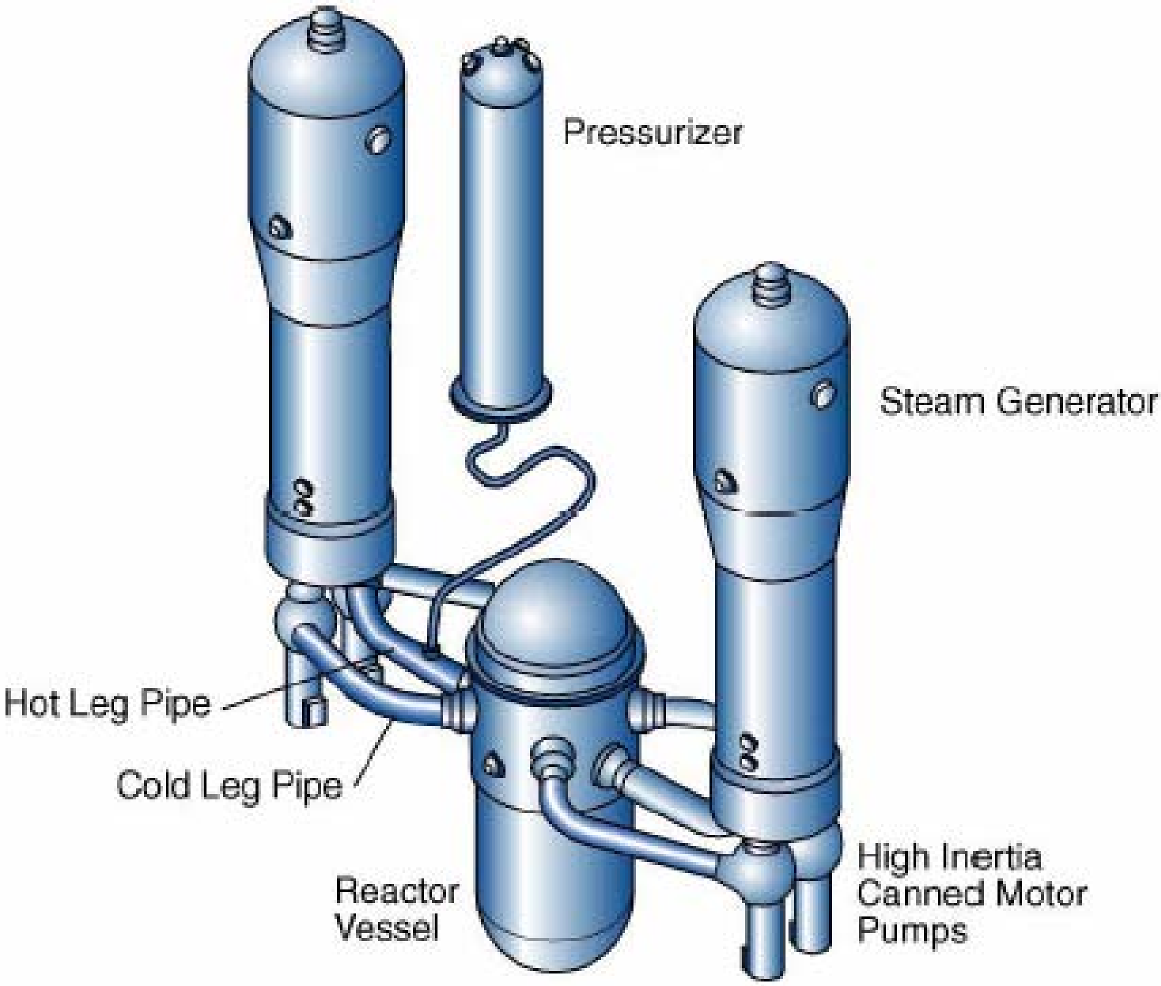


Figure 21 AP1000 Reactor Coolant System

Major component changes incorporated into the AP1000 design include a taller reactor vessel, larger steam generators (Delta–125), a larger pressurizer and slightly taller, canned reactor coolant pumps with higher reactor coolant flows. The designs for these reactor components are based on components that are used in operating PWRs or have been developed and tested for new PWRs. Performance of the passive safety features have been selectively increased, however, these changes have been accomplished with small changes to the AP600 plant design.

The AP1000 fuel design is based on the 17x17 XL (14 foot) design used successfully at plants in the U.S. and Europe. As with AP600, studies have shown that AP1000 can operate with a full core loading of MOX fuel.

(c) Construction

The AP1000 will be manufactured in modules designed for rail or barge shipment. This will allow constructing many modules in parallel, and the plant is designed to have fuel load 36 months after concrete is first poured. This construction period is considerably shorter than earlier generation designs, and if achieved in practice will greatly decrease the overall capital cost of the plant. Such reductions would make the design much more economically competitive against other power sources than previous generation nuclear plants.

(d) Safety

Probabilistic risk assessment was used in the design of the plants. This enabled minimization of risks, and calculation of the overall safety of the plant. (The Nuclear Regulatory Commission is preparing a new safety study - these plants will be orders of magnitude safer than the last study, NUREG-1150.) The AP1000 has a maximum core damage frequency of 5.09×10^{-7} per plant per year.

6.3 AREVA's EPR/US-EPR

The European pressurized reactor (EPR or US-EPR for the United States specific design) is a third generation nuclear fission pressurized water reactor (PWR) design. It has been designed and developed mainly by Framatome (AREVA NP) and Electricité de France (EDF) in France, and Siemens AG in Germany.

As of 2007, two units were under construction, one each in Finland and in France, and two units were planned as part of China's tenth economic plan, to start construction in 2009.

(1) Design

The main design objectives of the EPR design are increased safety while providing enhanced economic competitiveness through evolutionary improvements to previous PWR designs scaled up to an electrical power output of 1600 MWe. The reactor can use 5% enriched uranium oxide or mixed uranium plutonium oxide fuel. This reactor's core can be loaded with 100% MOX fuel, whereas a typical PWR core can loaded with only about 33% MOX fuel.

The EPR design has several active and passive protection measures against accidents:

- Four independent emergency cooling systems, each capable of cooling down the reactor after shutdown.
- Leaktight container around the reactor.
- An extra container and cooling area if a molten core manages to escape the reactor (see containment building).
- Two-layer concrete wall with total thickness 2.6 meters, designed to withstand impact by airplanes.

The EPR has a maximum core damage frequency of 4 x 10^{-7} per plant per year.

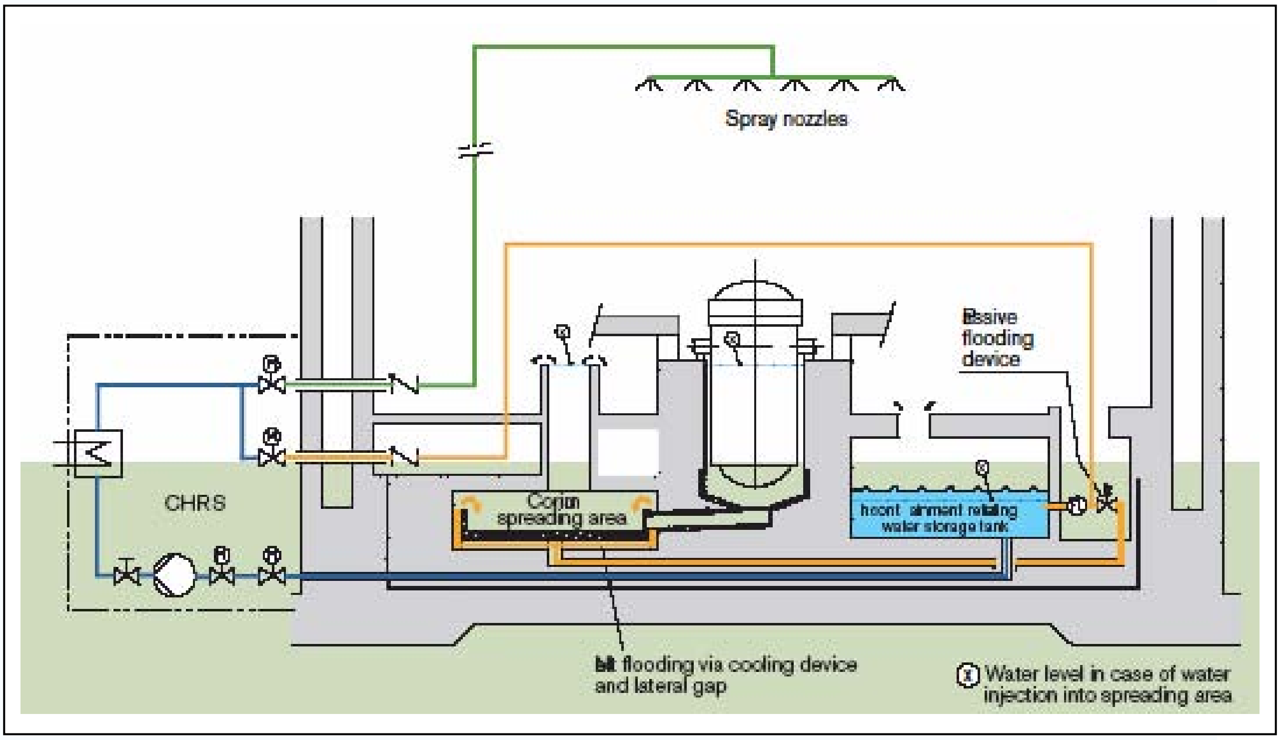


Figure 22 US-EPR Containment Heat Removal system

(2) EPR vs. U.S. EPR

The U.S. EPR is a 1,600 MWe class PWR plant design based on the European pressurized water reactor (EPR) design being built in Olkiluoto, Finland. While the EPR is a standardized global design, it is being marketed and licensed in the U.S. with a different name to reflect its conversion to meet U.S. codes, standards, regulatory requirements, and U.S. cycle frequency and grid voltages. Thus, when referring to the European plant design being built in Finland and scheduled for construction in France, the term “EPR” is used.

Figure 23 Overview of AREVA's ERP

More details in US-EPR Plant Structure (Icon)

"U.S. EPR" refers to the plant design being marketed, certified and licensed in the U.S. The EPR's evolutionary design is based on experience from several thousand reactor-years of operation of light water reactors worldwide, primarily those incorporating the most recent technologies: the N4 and KONVOI reactors currently operating in France and Germany respectively. The EPR design integrates the results of decades of research and development programs, in particular those carried out by the CEA (French Atomic Energy Commission) and the German Karlsruhe research center. Through its N4 and KONVOI affiliation, the EPR benefits from an uninterrupted continuum of evolutionary development and innovation that has supported the development of the PWR since its introduction to the Western marketplace in the mid-1950s.

(3) Technological advance in performance

Offering a significantly enhanced level of safety, the EPR features major innovations, especially in preventing core meltdown and mitigating its potential consequences. The EPR design also benefits from outstanding resistance to external hazards, including military or large commercial airplane crash and earthquakes. The EPR operating and safety systems provide progressive responses commensurate with any abnormal occurrences.

A number of technological advances place the EPR at the forefront of nuclear power plant design. Significant progress has been incorporated into its main features:

- The reactor core and its flexibility of fuel management
- The reactor protection system
- The instrumentation and control (I&C) system, the operator-friendly man-machine interface and fully computerized control room
- The large components such as the reactor pressure vessel and its internal structures, steam generators and primary coolant pumps

These innovations contribute to the high level of performance certainty in terms of efficiency, operability, reliability and, therefore, economic competitiveness. The U.S. EPR is an important part of the solution to ensure the integrity of the world's future energy supply.

Figure 24 US-EPR Primary System

(4) Requirements for new nuclear power plants

The French-German cooperation set up to develop the EPR brought a number of participants together from the start of the project:

- Power plant vendors Framatome and Siemens KWU (whose nuclear activities have since been merged to form Framatome ANP, now AREVA NP)
- EDF (Electricité de France), and the major German utilities now merged to become E.ON, EnBW and RWE Power
- The safety authorities from both countries to harmonize safety regulations

The EPR design takes into account the expectations of utilities as stated by the "European Utility Requirements" (EUR) and the "Utility Requirements Document" (URD) issued by the U.S. Electric

Power Research Institute (EPRI). It complies with the 1993 joint recommendations of the French and German safety authorities.

The technical guidelines covering the EPR design were validated in October 2000 by the French standing group of experts in charge of reactor safety (“Groupe Permanent Réacteurs,” the advisory committee for reactor safety to the French safety authority) supported by German experts.

On behalf of the French government, the French safety authority officially stated on September 28, 2004, that the EPR safety features comply with the safety enhancement objectives established for new nuclear reactors.

On May 4, 2006, the EDF Board of Directors decided to launch the construction of its first EPR unit on the Flamanville site.

(5) Continuity in technology

The N4 and KONVOI reactors are descendants of the earlier Framatome and Siemens KWU generation reactors, which are themselves derivative of those first implemented in the U.S., then refined and expanded upon by Framatome and Siemens KWU. The EPR is the direct descendant of the well proven N4 and KONVOI reactors, resulting in a fully mastered technology that minimizes risks linked to design, licensing, construction and operation – a unique certainty for EPR customers that carries distinct advantages. First, operator expertise acquired through the operation of nuclear power plants using the same technology as the EPR is maintained and its value is increased.

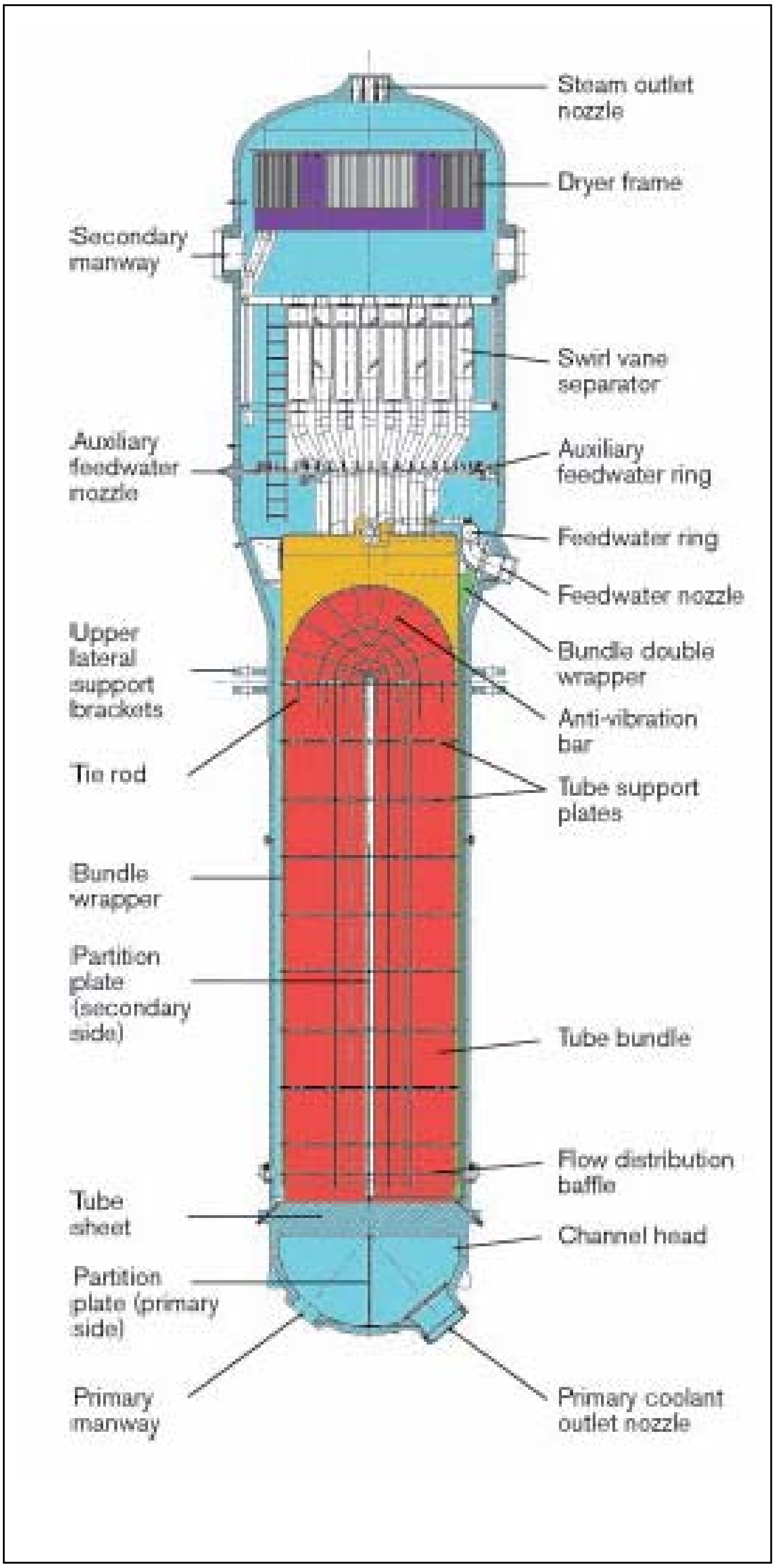


Figure 25 US-EPR Steam Generator

(6) Reduced power generation cost

Thanks to an early focus on economic competitiveness during the design process, the EPR offers significantly reduced power generation costs, estimated to be 10% lower than those of the most modern nuclear units currently in operation. According to the most recent international study, OECD NEA/IEA Projected Costs of Generating Electricity— 2005 Update, in which several countries in Europe chose the EPR as the reference model for their future nuclear programs, the average cost of electricity generated by an EPR would be significantly less than that generated using combined cycle gas turbine (CCGT) technology.

The cost savings amount to around 20% for a gas price of $4- $6/Mbtu and a weighted average capital cost (WACC) of 8% to 9% in real terms.

This high level of competitiveness is achieved through:

- Unit power in the 1,600 MWe range (the highest unit power to date), providing an attractive cost of the installed kWe
- 35% overall efficiency — presently the highest value ever for water reactors
- Shortened construction time incorporating experience feedback and continuous improvement of construction methodology and task sequencing
- A design objective for a 60-year service life enhanced and more flexible fuel utilization
- An average availability factor greater than 94% during the entire service life of the plant, obtained through longer fuel cycles, shorter refueling outages and in-operation maintenance
- Significant advances for sustainable development

Due to its optimized core design and higher overall efficiency compared to reactors in operation today, the U.S. EPR also offers many significant advantages in terms of sustainable development:

- Uses 7% less Uranium/MWh
- 15% reduction of long-lived actinides generation/MWh
- 14% gain on the electricity generation vs. thermal release ratio (compared to 1,000 MWe-class reactors)
- Great flexibility to use MOX (mixed $UO_2^{-}Pu\ O_2$) fuel

(7) Pilot power plant

The Olkiluoto 3 power plant in Finland, initially scheduled to go on line in 2009, will be the first EPR reactor built. The construction will be a joint effort of French AREVA and German Siemens AG through their common subsidiary AREVA NP, for Finnish operator TVO. The power plant will cost about €3.7 billion.

6.4 US-APWR (Mitsubishi)

The US-APWR is a generation III nuclear reactor based on pressurized water reactor technology. The US-APWR was developed by MHI to modify their APWR design to comply with US regulations. It features several design enhancements including a neutron reflector, improved efficiency and improved safety systems. Though its safety systems are improved, the US-APWR would generally not be considered passive.

Plant Parameters:

Electric Power	1,700 MWe
Core Thermal Power	4,451 MWt
Reactor Fuel Assemblies	257
Reactor Fuel	Advanced 17x17, 14 ft.
Active Core Length	4.2 meters
Coolant System Loops	4
Coolant Flow	2.75x104 m3/h/loop
Coolant Pressure	15.5 MPa
Steam Generator Type	90TT-1
Number of Steam Generators	4
Reactor Coolant Pump Type	100A
Number of Reactor Coolant Pumps	4
Reactor Coolant Pump Motor Output	6,000 kW

The US-APWR has several design features to improve plant economics. The core is surrounded by a steel neutron reflector which increases reactivity and saves ~0.1wt% U-235 enrichment. In addition, the US-APWR uses more advanced steam generators (compared to the APWR) which creates dryer steam allowing for the use of higher efficiency (and more delicate) turbines. This leads to a ~10% efficiency increase compared to the APWR.

Several safety improvements are also noteable. The safety systems have enhanced redundancy, utilizing 4 trains each capable of supplying 50% the needed makeup water instead of 2 trains capable of 100%. Also, more reliance is placed on the accumulators which have been redesigned and increased in size. The improvements in this passive system have led to the elimination of the safety injection system, an active system.

6.5 Licensing Status for Advanced Pressurized Water Reactors

(1) U.S.

In January 2006, the Nuclear Regulatory Commission approved the final design certification for the AP1000. This means two things: (1) prospective builders can apply for a combined construction and operating license (COL) before construction starts, whose validity is conditional upon the plant being built as designed, and (2) each AP1000 should be virtually identical.

So far two AP1000s are slated for Cherokee County, South Carolina, and one or two for the Bellefonte Nuclear Generating Station in Alabama.

Licensing status for AP-1000, US-EPR and U.S.-APWR in U.S. is shown in Figure 26.

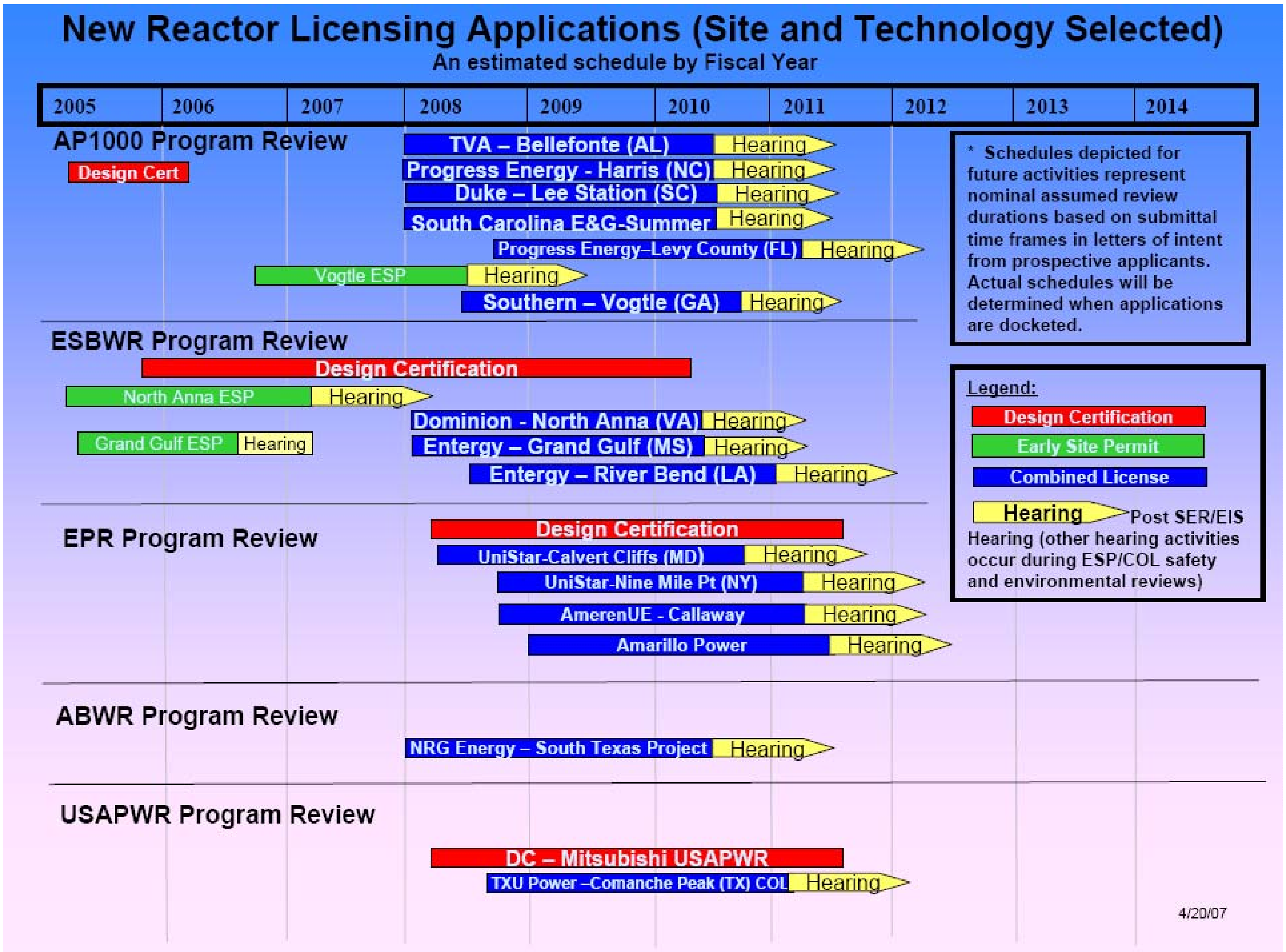


Figure 26 New Reactor Licensing Applications in US

Table 3 Expected New Nuclear Power Plant Applications in U.S.

Expected New Nuclear Power Plant Applications Updated April 27, 2007				
Company	**Design Type**	**Site Under Consideration**	**State**	**Existing Plants**
2007 Applications				
Duke	AP1000	William Lee Nuclear Station (2 units)	SC	N
NuStart Energy	AP1000	Bellefonte (2 units)	AL	N
Progress Energy	AP1000	Harris (2 units)	NC	Y
Dominion	ESBWR	North Anna (1 unit)	VA	Y
NuStart Energy	ESBWR	Grand Gulf (1 unit)	MS	Y
South Carolina Electric & Gas	AP1000	Summer (2 units)	SC	Y
NRG Energy	ABWR	South Texas Project (2 units)	TX	Y
2007 TOTAL NUMBER OF APPLICATIONS =7 TOTAL NUMBER OF UNITS = 12				
2008 Applications				
Progress Energy	AP1000	Levy County (2 units)	FL	N
Southern Nuclear Operating Co.	AP-1000	Vogtle (2 units)	GA	Y
Entergy	ESBWR	River Bend (1 unit)	LA	Y
UNISTAR	EPR	Calvert Cliffs (1 unit)	MD	Y
UNISTAR	EPR	TBD (1 unit)	TBD	UKN
AmerenUE	EPR	Callaway (1 unit)	MO	Y
UNISTAR	EPR	Nine Mile Point (1 unit)	NY	Y
TXU Power	US APWR	Comanche Peak (2 units)	TX	Y
Exelon	TBD	TBD (1 unit)	TBD	UNK
Detroit Edison	TBD	Fermi (1 unit)	MI	Y
Amarillo Power	EPR	Vicinity of Amarillo (2 units)	TX	UKN
2008 TOTAL NUMBER OF APPLICATIONS = 11 TOTAL NUMBER OF UNITS = 15				
2009 Applications				
Florida Power & Light	TBD	TBD (1 unit)	UNK	UNK
2009 TOTAL NUMBER OF APPLICATIONS = 1 TOTAL NUMBER OF UNITS = 1				
2007 – 2009 Total Number of Applications = 19 Total Number of Units = 28				

(2) Europe and China

Finland

In December 2006 TVO announced construction was about 18 months behind schedule so completion was now expected 2010-2011, and there were reports that AREVA was preparing to take a € 500 million charge on its accounts for the delay.

France

As of early 2007, site preparation for a demonstration EPR reactor Flamanville in the Manche département is underway. The site already has two operating reactors, and this reactor will be called Flamanville 3. Electrical output will be 1,600 MW and it is projected to cost 3.3 billion Euros. The following is a condensed timeline for the unit:

- From October 19, 2005 to February 18, 2006 the project was submitted to a national public debate
- On May 4, 2006 the decision was made by the EDF's Board of Directors to continue with the construction.
- Between June 15 and July 31, 2006 the unit was under public enquiry, which rendered a "favorable opinion" on the project.
- Summer 2006 site preparation works began.
- December 2007 construction of the unit itself will begin, construction will last 54 months.
- In 2012 the facility will be comissioned.

United Kingdom

EDF said it will propose, in partnership with AREVA, its European pressurized reactor (EPR) model of nuclear reactors to the UK government.

China

In 2006, there was a bidding in process to build four new EPR reactors to China, and an intent to market EPRs in the U.S. with Constellation Energy. In April of 2006, AREVA SA lost this bid in favour Westinghouse Electric Co. to build four AP1000 reactors, because of its refusal to transfer the expertise and knowledge to China. Nevertheless, in February 2007, AREVA won another deal, worth $ 5 bn, for two other nuclear reactors located in Guangdong, in southern China in spite of sticking to its previous conditions. The local partner for this project is China Guangdong Nuclear Power Co.⊥

References

NRC HP: http://www.nrc.gov/

Wikipedia: http://en.wikipedia.org/wiki/Main_Page

ATOMICA: http://atomica.nucpal.gr.jp/atomica/index.html

Handbook for Thermal and Nuclear Power Engineers, English Edition of the 6th Edition, 2002, Thermal and Nuclear Power Engineering Society of Japan (TENPES)

Behavior of LWR Fuel (4th edition), P 92, Nuclear Safety Research Association (July, 1998)

Nuclear Power Generation Guide, 1999 Edition, Edited by Nuclear Power Division, Public Utilities Department, Agency for Natural Resources and Energy, Ministry of International Trade and Industry, Published by Denryoku Shinpou Sha, October 1999

Outline of a LWR Power Station (Revised edition), Nuclear Safety Research Association (October 1992),

M. Marshal, Development of Nuclear Technologies, Chikuma-shobou, September 1996, p281

I van de Venne, Application of Passive Safety Systems to Large PWRs, Proceedings Vol. II, International Conference on Design and Safety of Advanced Nuclear Plant, Oct. 25 – 29, 1992, Tokyo Japan

WH HP

The AP1000 Reactor Nuclear Renaissance Option, Dr. Regis A. Matzie Senior Vice President and Chief Technology Officer, BNFL & Westinghous, Tulane Engineering Forum, September 26, 2003

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NUCLEAR RENAISSANCE?, A FORMER REGULATOR'S PERSPECTIVE REGARDING THE NRC ROLE AND ACTIVITIES, ASHOK THADANI