PRESSURIZED WATER REACTORS

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Keywords: Pressurized Water Reactors, Reactor Core, Fuel Elements, Control Rods, Steam Generators

Contents

Summary

Pressurized Water Reactors were developed in the United States, as well as in Europe and the Soviet Union. Since most European development was as a result of licensing agreements with the United States, there is much commonality between the respective designs. Those developed in the Soviet Union were, however, different and with less rigorous safety provisions.

The principle of the pressurized water reactor is that the coolant circuit containing light water is pressurized to prevent boiling at the prevailing temperatures. The hot pressurized coolant is then used to generate steam in a secondary circuit at a lower pressure. The saturated steam so produced in the steam generators is fed directly to the steam turbines which are designed to operate with wet steam. Steam generators are of different designs with one type producing slightly superheated steam.

The fuel elements consist of many fuel rods extending the full height of the core and arranged in a square array. Guide tubes replace certain rods in alternate fuel elements. Neutron absorbing control rods are inserted into the guide tubes to regulate the heat production in the reactor. The coolant flows upward between the fuel rods removing the heat generated from the fission reaction. The coolant also serves as the moderator to reduce the energy of the neutrons so enhancing their capture in the fuel. Since light water does absorb some neutrons, slightly enriched fuel is required.

The reactor has to be shut down about once a year for refueling. However, only one third of the fuel is replaced each year and some moved within the reactor to ensure optimum fuel utilization.

Generally, pressurized water reactors, with a relatively high power density in the reactor core, are compact and competitive in capital and operating costs. They are by far the most common type of reactors used for large scale commercial power generation.

1. Introduction

1.1 General Information

The Pressurized Water Reactor, commonly known by its acronym PWR, is currently the most common type of reactor in commercial use for power production. These reactors were originally designed by Westinghouse Bettis Atomic Power Laboratory for military ship application and then by the Westinghouse Nuclear Power Division for commercial application. The first commercial PWR plant in the United States was Shippingport, which was located near Pittsburgh, Pennsylvania. Shippingport was a 60 MWe reactor that operated from 1957 to 1982 and later converted to the prototype LWBR. Following the 175 MWe Yankee-Rowe unit in 1961 and another in Italy, Westinghouse built reactors on a modular design (with from one to four steam generator loops) throughout the United States and under various licensing agreements with many countries including France, Germany, Japan, Sweden, and Spain. Moreover, Westinghouse was the vendor for the 1000 MWe Sizewall reactor, the first unit of a planned series of PWRs in the United Kingdom. Figure 1 shows a simplified diagram of a typical PWR.

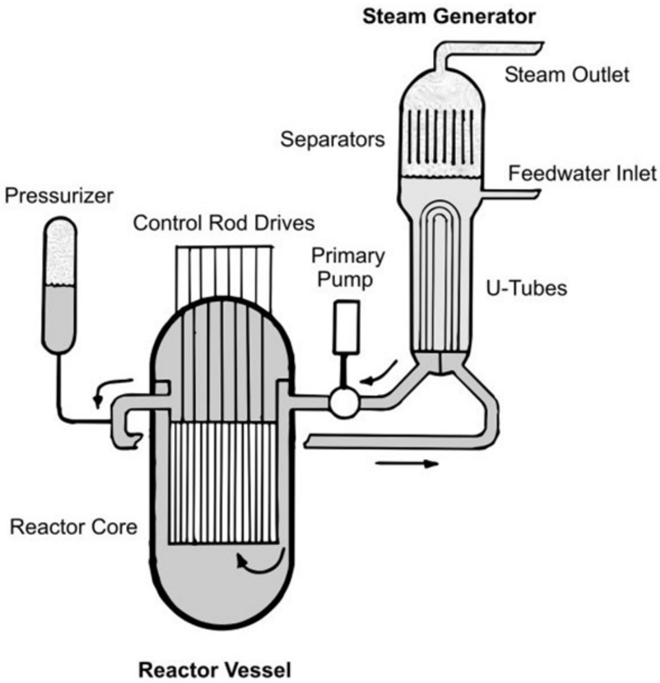

Figure 1: Diagrammatic cross-section of a typical PWR

Other designs have been developed in the United States by Babcock & Wilcox (B&W) and Combustion Engineering (CE), in France by Framatome, in Germany by Siemens, Kraftwerk Union (KWU) and Brown Boveri (BBR), in Japan by Mitsubishi, and in the Soviet Union by Atommash in the late 1980s. Joint ventures have also been used with many other liaisons (especially for advanced “next generation” reactors). In addition to Westinghouse, Asea Brown Boveri-Combustion Engineering (ABB-CE), Framatome, Kraftwerk Union, Siemens, and Mitsubishi have typically built PWRs throughout the world. Babcock & Wilcox (B&W) built a PWR design power plant but used vertical once-through steam generators, rather than the U-tube design used by the rest of the suppliers.

Framatome is now the world leader in PWR design and construction. It was a licensee to Westinghouse until 1984. Many renowned PWR designs have been built by Framotome. The 1450 MWe Chooz B1, an indigenous, state of the art PWR and the first unit in the N4-series, was scheduled for operation in 1992. Chooz B1 and Chooz B2 have been commercially operated since 1996. Framatome exports have gone to the Republic of South Africa, Belgium, South Korea, and the Peoples Republic of China.

The VVER is the Russian version of the commercial PWR developed from nuclear submarine and ice breaker experience. VVER is the abbreviation for Vodo-Vodyannoy Energeticheskiy Reactor which translates as “water-cooled water-moderated reactor”. Soviet-designed nuclear power plants differ from Western nuclear power plants in many respects such as the VVERs use horizontal steam generators while western plants use two, three or four large vertical steam generators. Another main difference is that Soviet-designed reactors do not have a containment-structure (with exception of VVER-1000 design). The initial 265 MWe unit at Novo Voronezh (1964) was followed with a series of 440 MWe and 1000 MWe units, with two of the former (1972 and 1973) and one of the latter (1981) built at the same site. Although the VVER units were intended primarily for domestic and Eastern European use, two 445 MWe units have been in service at Loviisa in Finland since 1977 and 1981, respectively. In 1998, joint stock company Atomstroyexport (ASE) was established in order to promote the export of Russian-made products for nuclear power projects abroad.

Nowadays, the Russian nuclear plant designs have been developed based on the Russian norms and standards due to the international requirements. The latest export projects are two power units (2000 MW total) in China, a nuclear power plant of similar capacity in India and one power unit of 1000 MW in Iran.

Generally, in this chapter, PWR is used in reference to the designs originating in the United States and now widely utilized in Western countries and VVER to those originating in the Soviet Union and utilized within the Eastern Bloc.

2. General Configuration

2.1. Western Pressurized Water Reactor PWR

There are two major flow systems, the *primary system* and the *secondary system*, utilized to transport and convert the heat generated in the fuel into electrical power for industrial and residential use. The primary system transfers the heat from the fuel to the steam generator, where the secondary system begins. The steam formed in the steam generator is transferred by the secondary system to the main turbine generator, where its energy is converted into electricity. After passing through the turbine, the steam is routed to the main condenser. Cool water, passing through the tubes in the condenser from an external source, removes excess heat from the steam, allowing the steam to condense on the outside of the tubes. The condensate is then pumped back to the steam generator for reuse.

The primary system, also called the reactor coolant system, consists of 2, 3, or 4 cooling loops connected to the reactor, each containing a reactor coolant pump and steam generator. The systems built by the three main vendors consist of the same major components, but they are arranged in slightly different ways. For example, Westinghouse has built plants with 2, 3, or 4 loops, depending upon the power output of the plant. The Combustion Engineering plants and the Babcock & Wilcox plants only have two steam generators but they have four reactor coolant pumps.

Figure 2 shows the flow diagram for a three loop PWR with recirculating steam generators as in Figure 1. Boiling, other than minor bubbles arising from subcooled nucleate boiling, is suppressed by pressurization. Pressure is maintained by a pressurizer connected to the reactor coolant system. Pressure is maintained at approximately 15.5 MPa (2250 lbf in^{-2}) through a heater and spray system in the pressurizer. The primary coolant from the reactor is pumped to the steam generator and passes through its tubes. This water at a temperature of about 290°C (554°F) enters the reactor vessel through the inlet nozzles near the top. Water travels downward in the downcomer and then upward through the fuel assemblies. The water picks up heat and the temperature reaches about 310°C to 332°C (590°F to 629°F) as it exits the reactor.

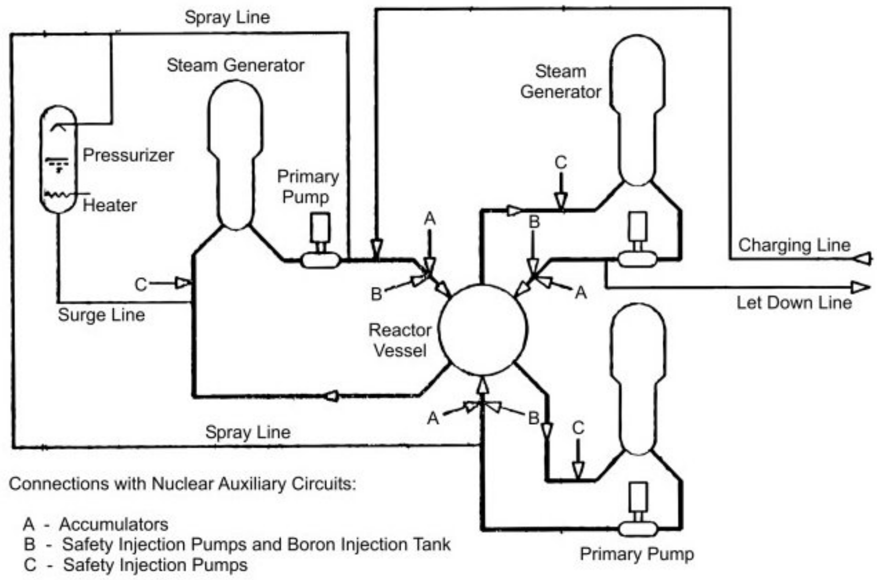

Figure 2: Primary circuit flow diagram

In the secondary system, water is pumped into the steam generators and passes on the outside of the steam generator tubes, where the water is heated and converted to steam. The steam then passes through the main steam lines to the turbine, which is connected to and drives the electrical generator. The steam leaving the turbine condenses in a condenser. The condensate is then pumped by condensate pumps through low pressure feedwater heaters, then by the feedwater pumps through high pressure feedwater heaters and finally back to the steam generators. The condenser is maintained at a vacuum using either vacuum pumps or air ejectors. Cooling of the steam is provided by condenser cooling water pumped through the condenser by circulating water pumps which draw water from the ocean, lake, river, or cooling tower.

2.2. Soviet Pressurized Water Reactor VVER

There are three standard designs of VVER, namely, two different 6-loop 440 MW [440-230 (older) and 440-213 (newer)] designs and a 4-loop 1000 MW design. As in the Western European and United States versions of the PWR, each reactor has primary coolant and secondary steam circuits. The primary coolant passes through the inside of the tubes in the steam generator. The reactor coolant pump circulates the water for cooling the reactor core. The system is pressurized to at least 15 MPa (2200 lbf in^{-2}) by a pressurizer which is connected to one of the reactor coolant loops. Spray valves and heaters in the pressurizer are used to control pressure in the coolant loop and reactor core. A major difference between Western designed PWRs and VVERs is that VVERs have horizontal steam generators. The older VVERs have isolation valves in the reactor coolant loops and accident localization compartments.

Secondary water, passing on the outside of the steam generator tubes, is heated and converted to steam. The steam passes to the turbine as in the PWR. The turbine drives the electrical generator. After being condensed in a condenser by cooling water, the condensate and feedwater are returned to the steam generator as in the PWR.

2.3. Major Differences between Soviet VVERs and Western PWRs

Soviet-designed nuclear power plants differ from Western-designed nuclear power plants in many respects, including plant instrumentation and controls, safety systems and fire protection systems. Although Soviet-designed plants, like Western-style plants, employ the design principle known in the West as "safety in depth", only the VVER-1000 design includes a containment structure as part of that principle.

In the unlikely event of a safety systems failure, plants designed on the "safety in depth" principle rely on a series of physical barriers to prevent the release of radioactive material to the environment. For typical Western-designed plants:

- The first barrier is the nuclear fuel itself, which is in the form of solid ceramic pellets. Most of the radioactive by-products of the fission process remain bound inside the fuel pellets.
- The fuel pellets are sealed in rods, made of special alloy and known as the cladding, to prevent the leakage of gaseous fission products.
- The fuel rods making up the reactor core are inside a large steel pressure vessel which, with the coolant loop pipe work and steam generators, forms a pressure boundary to contain the reactor coolant and any leaked radioactive material.
- At most plants, this vessel which serves to contain any fission products possibly released through the above barriers and associated coolant loop is enclosed in a large, leak-tight shell of steel plate and concrete known as the containment.

Most Soviet-designed reactors employ similar features as mentioned above, but only the VVER-1000 design has a containment structure like that of most nuclear power plants elsewhere in the world. Without this fourth level of protection, radioactive material could escape to the environment in the event of a serious accident involving the rupture of the reactor coolant circuit and failure of the fuel cladding.

At the end of 1994, about 90 commercial nuclear reactors of Soviet design were operating, under construction, and planned in Russia, Ukraine, Lithuania, Bulgaria, the Czech Republic, the Slovak Republic and Hungary. Of these commercial nuclear reactors, 63 were of the VVER type. A two-unit Soviet-designed nuclear plant in Finland was built using the VVER-440 Model 213 basic design, but was upgraded to include a Western instrumentation and control system and a containment structure. Recently, VVER-1000 designed nuclear power plants have been built in China, India and Iran.

3. Core Arrangement

3.1. Reactor Vessel

The major component of the reactor is the reactor vessel, in which is housed the core barrel, the reactor core, the upper internals package and all associated support and alignment devices.

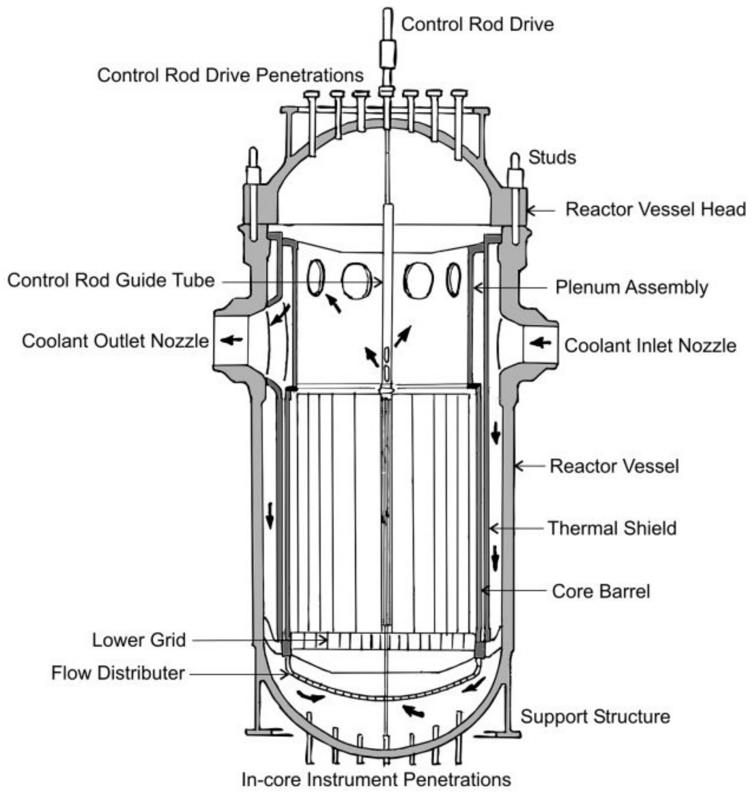

Figure 3: Cross-section of reactor vessel

The reactor vessel, as shown in Figure 3, is a cylindrical vessel with a hemispherical bottom and a removable hemispherical head at the top. The head is removable to allow for the refueling of the reactor. There is one inlet (or cold leg) nozzle and one outlet (or hot leg) nozzle for each reactor coolant system loop. The reactor vessel is constructed of manganese molybdenum steel and all surfaces that come into contact with reactor coolant are clad with stainless steel to increase corrosion resistance.

The core barrel is fitted inside the reactor vessel and surrounds the fuel. It creates an annular channel for the incoming coolant and provides some radiation attenuation and hence protection for the reactor vessel. At the bottom of the core barrel, there is a lower core plate on which the fuel assemblies rest. The core barrel and all of the lower internals actually hang inside the reactor vessel from the internals support ledge. On the outside of the core barrel are irradiation specimen holders in which samples of the material used to manufacture the vessel are placed. At periodic intervals during the life of the reactor vessel, some of these samples are removed and tested to see how the radiation from the fuel has affected the strength of the material.

The upper internals package rests on top of the core. It contains guide columns to guide the control rods when they are pulled from or inserted into the core. The upper internals package also prevents the core from moving up during operation due to the force from the coolant flowing through the assemblies.

The coolant enters the vessel at the inlet nozzles. The core barrel forces the water to flow downward in the annulus between the reactor vessel and the core barrel. On reaching the bottom of the vessel, the flow is turned upward to flow through the fuel assemblies thus removing the heat produced by the fission process. The heated water enters the upper internals region, where it is routed out through the outlet nozzles. The flow is driven by the reactor coolant pumps in the cold leg of each coolant loop after passing through the steam generators. In the steam generators, the heat in the primary system is transferred to the secondary system and the reactor coolant returned to the reactor vessel.

3.2. Moderator and Coolant

All power reactor systems have a moderator which slows down neutrons via scattering interactions. An increase in temperature generally decreases both the density and the effectiveness for slowing down. The PWRs use light water as a moderator. It does, however, absorb neutrons to the extent that slightly enriched uranium fuel must be used to obtain a continuous fission chain reaction. The light water also serves as the coolant and circulates through the reactor. The moderator and coolant characteristics allow for a very compact reactor core. Since the fuel must be lumped and surrounded by both of moderator and coolant, it is formed into long thin vertical rods and arranged into a square array extending across the reactor core which thus takes the shape of a vertical cylinder. The spacing between the fuel rods is determined by the neutron slowing down distance in the light water. This heterogeneous arrangement also promotes effective heat transfer from the fuel to the coolant.

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Biographical Sketch

Janit Pongpuak obtained a B.Sc. (2nd class honors) in Chemical Technology, Chulalongkorn University and a M.S. in Petrochemical Technology from the Petroleum and Petrochemical College, Chulalongkorn University in Thailand. After she finished her Masters degree, she went to Canada to continue her study. She is doing her Ph.D. in Chemical Engineering at the University of New Brunswick. Her research interest is the effect of pressurized water coolant chemistry on oxide formation and corrosion in the primary heat transport system of nuclear reactors.