PDHonline Course E338 (5 PDH)

Nuclear Power Volume II - Nuclear Power Plants

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2020

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Nuclear Power, Volume II
Nuclear Power Plants

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Cover photograph: Courtesy U.S. Nuclear Regulatory Commission (NRC).

Preface

This is the second in a series of three courses about the nuclear power industry. The series covers the nuclear industry from the physics of nuclear reactions to the types of plants in operation today as well as the potential of the next generation of nuclear power plants that are likely to appear in the first half of the 21st century.

The complete series includes three courses:

1. Volume I – The Nuclear Power Industry
2. Volume II – Nuclear Power Plants
3. Volume III – The Future of Nuclear Power

The first course, *Volume I – The Nuclear Power Industry*, gives a broad overview of the nuclear power industry. This course goes into the details of nuclear reactions and the physics of nuclear power. The prime fuel source, uranium, is covered too.

The second course, *Volume II – Nuclear Power Plants*, reviews the classifications of nuclear power plants and the basic components of a nuclear power plant. The course covers the design and operation of the current generation of nuclear power plants in operation today.

The third course, *Volume III – The Future of Nuclear Power*, gives an overview of the types of plants that are being considered for the next generation of power plants. Some of the designs covered are already operating in experimental stages, some are modifications of current designs, and others are radical new concepts that have not been commercially validated.

It is not necessary to take the courses in sequence. However, for the best comprehensive it is suggested that the courses be taken in the order presented.

Introduction

The world demand for electricity continues to grow. The current electrical energy consumption for the entire planet is approximately 1,517 gigawatts (GW) of continuous power. Approximately 16% of this electric demand is served from 439 nuclear power plants with a capacity of 371 GW. The United States produces the most nuclear energy, with nuclear power providing 19% of the electricity it consumes, while France produces the highest percentage of its electrical energy from nuclear reactors—77% as of 2006. It is quite possible to utilize nuclear power to provide the vast majority of an entire country's need for electricity.

From a *capacity* standpoint, nuclear is only about 10% of the total electrical generation in the U.S. From a consumption – or *energy* - standpoint, nuclear power is responsible for approximately 19% of the electrical energy in the United States. The total generation is approximately 3,800 thousand GWH. For comparison purposes, nuclear generation accounts for the following of the total electrical production in some other countries: 77% in France, 46% in Sweden, 43% in Ukraine, 39% in South Korea, 30% in Germany, and 30% in Japan. There are currently 104 licensed commercial nuclear power plants in the United States.

Unlike the coal and oil plants that supply most of the electrical power in the United States, a nuclear power plant, like the one shown on the right, releases virtually no pollution or greenhouse gases into the Earth's atmosphere, and therefore doesn't contribute to global warming. Although nuclear power plants generate long-lived nuclear waste, this waste arguably poses much less of a threat to the biosphere than greenhouse gases would.

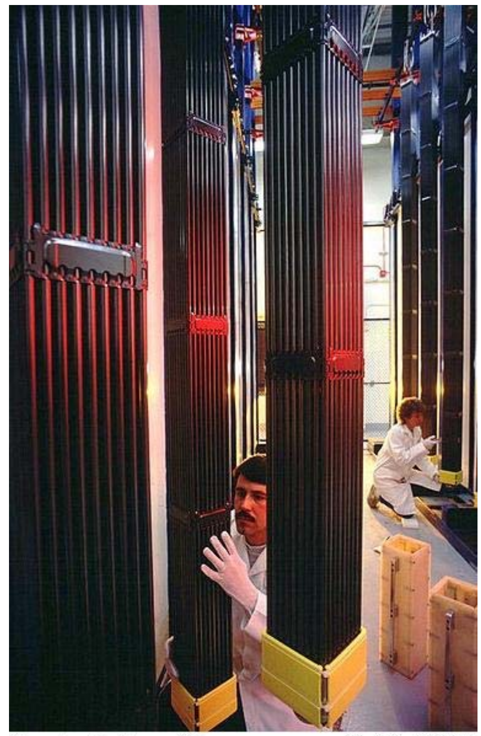

Photo Credit: Nuclear Regulatory Commission

This course is the second in a series about the use of nuclear energy to generate electricity. Volume I in this series included an overview of the nuclear industry and covered the basics of nuclear physics and uranium as a fuel source. While it is not necessary to read Volume I first, it will help understand the concepts and terms presented in Volume II.

Volume II delves into the specific types of nuclear reactors in use around the world today. We will begin with a broad overview of the different types of reactors and how they are classified. Then we will look at each type of nuclear reactor that is currently in commercial operation.

Chapter 1
Types of Nuclear Reactors

A nuclear reactor is a device to initiate, control, and sustain a nuclear chain reaction. This is accomplished by using heat from the nuclear reaction to power steam turbines. In a nuclear reaction, the energy released from continuous fission of the atoms in the fuel as heat is used to make steam. The steam is used to drive the turbines which produce electricity, which is clean and virtually free of greenhouse gas emissions.


Reactor Fuel Assembly Photo Credit: NRC

In this chapter we will look at the components of a nuclear reactor, how nuclear reactors are classified, and provide an overview of the various types of nuclear reactors.

Using a common type of nuclear reactor design, let's review how a reactor works. The uranium reactor fuel is housed in long thin tubes. Many tubes are placed close together and immersed in water. The tubes of uranium emit neutrons and generate heat in the process. The hot water becomes steam, which turns a turbine connected to a generator. The water also cools the uranium tubes and prevents them from getting too hot and melting. The water also slows down the neutrons being emitted from the uranium tubes. This slowing is called *moderating* the nuclear reaction and is necessary to sustain the reaction. The process just described is based on a boiling water reactor. Other reactor types use graphite to moderate the reaction and gas or liquid metals to cool the reaction, but all work on the same basic principle.

Components

The following is an overview of the major components of a nuclear power plant.

1. Fuel

The fuel is used to generate heat. The reactor core generates heat in a number of ways including: The kinetic energy of fission products which is converted to thermal energy when these nuclei collide with nearby atoms, gamma rays produced during fission being absorbed by the reactor causing their energy to be converted into heat, and heat produced by the radioactive decay of fission products and materials that have been activated by neutron absorption.

The fuel is usually pellets of uranium oxide (UO_2) arranged in tubes to form fuel rods. The rods are arranged into fuel assemblies in the reactor core. In a new reactor with new fuel a neutron source is needed to get the reaction going. Typically this is beryllium mixed with polonium, radium or other alpha-emitter. Alpha particles from the decay cause a release of neutrons from the beryllium as it turns to carbon-12. Restarting a reactor with some used fuel may not require this, as there may be enough neutrons to achieve criticality when control rods are removed.

2. Moderator

This is material which slows down the neutrons released from fission so that they cause more fission. It is usually water, but may be heavy water or graphite. Commonly used moderators include regular (light) water, solid graphite, and heavy water (deuterium). Light water reactors compromise about 75% of the moderators, graphite 20%, and heavy water accounts for 5% of the moderators.

3. Control rods

The power output of the reactor is determined by controlling how many neutrons are able to create more fissions. This is done with control rods. Control rods are made with neutron-absorbing material such as cadmium or boron, and are inserted or withdrawn from the core to control the rate of reaction, or to halt it. (Secondary shutdown systems involve adding other neutron absorbers, usually in the primary cooling system.)

Control rods that are made of a nuclear poison are used to absorb neutrons. Absorbing more neutrons in a control rod means that there are fewer neutrons available to cause fission, so pushing the control rod deeper into the reactor will reduce its power output, and extracting the control rod will increase it.

In some reactors, the coolant also acts as a neutron moderator. A moderator increases the power of the reactor by causing the fast neutrons that are released from fission to lose energy and become thermal neutrons. Thermal neutrons are more likely than fast neutrons to cause fission, so more neutron moderation means more power output from the reactors. If the coolant is also a moderator, then temperature changes can affect the density of the coolant/moderator and therefore change power output. A higher temperature coolant would be less dense, and therefore a less effective moderator.

In other reactors the coolant acts as a poison by absorbing neutrons in the same way that the control rods do. In these reactors power output can be increased by heating the coolant, which makes it a less dense poison. Nuclear reactors generally have automatic and manual systems to insert large amounts of poison (often boron in the form of boric acid) into the reactor to shut the fission reaction down if unsafe conditions are detected or anticipated.

4. Coolant

A nuclear reactor coolant — usually water but sometimes a gas or a liquid metal or molten salt — is circulated past the reactor core to absorb the heat that it generates. The heat is carried away from the reactor and is then used to generate steam. Most reactor systems employ a cooling system that is physically separated from the water that will be boiled to produce pressurized steam for the turbines, like the pressurized water reactor. But in some reactors the water for the steam turbines is boiled directly by the reactor core, for example the boiling water reactor.

5. Pressure vessel

The pressure vessel is a robust steel vessel containing the reactor core and moderator/coolant. In some reactors the pressure vessel is actually a series of tubes holding the fuel and conveying the coolant through the moderator.

6. Steam generator

The steam generator is a part of the cooling system where the heat from the reactor is used to make steam for the turbine and functions much like the steam generator of a coal-fired power plant.

7. Containment

The structure around the reactor core which is designed to protect it from outside intrusion and to protect those outside from the effects of radiation in case of any major malfunction inside. It is typically a thick concrete and steel structure. Not all nuclear power plants have a containment vessel.

8. Electric Power Generator

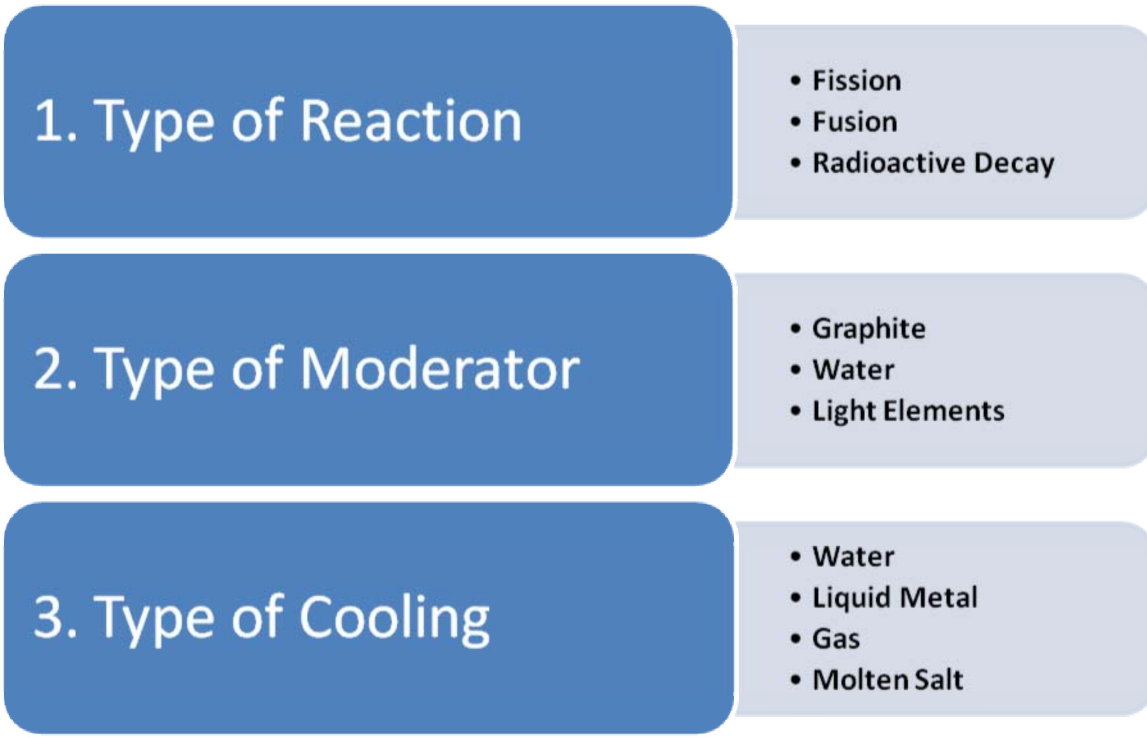
The steam turbine of a nuclear plant it connected by shaft to an electric power generator, which generates electricity by turning a rotor within the generator. Electricity is generated at a low voltage (typically 25 kV or less) and then stepped-up to transmission voltages of 230-500 kV and higher in the substation yard.

Reactor Classifications

Reactors can be classified in a variety of ways. The most common methods of classification are by the type of nuclear reaction, the type of moderator used, or the type of coolant employed. See Figure 1. The following is a brief outline of the typical classifications.

Figure 1

Classifications of Reactors



We will look at each of these broad classifications briefly.

1. Classification by Type of Reaction

- Nuclear fission. Most reactors, and all commercial ones, are based on nuclear fission. They generally use uranium and its product plutonium as nuclear fuel, though a thorium fuel cycle is also possible. Fission reactors can be divided roughly into two classes, depending on the energy of the neutrons that sustain the fission chain reaction: Thermal reactors and fast neutron reactors.

  Thermal reactors use slowed or thermal neutrons. Almost all current reactors are of this type. These contain neutron moderator materials that slow neutrons until their kinetic energy approaches the average kinetic energy of the surrounding particles. Thermal neutrons have a far higher probability of fissioning the fissile nuclei uranium-235, plutonium-239, and plutonium-241, and a relatively lower probability of neutron capture by uranium-238 compared to the faster neutrons that originally result from fission, allowing use of low-enriched uranium or even natural uranium fuel. The moderator is often also the coolant, usually water under high pressure to increase the boiling point. These are surrounded by reactor vessel, instrumentation to monitor and control the reactor, radiation shielding, and a containment building.

Fast neutron reactors use fast neutrons to cause fission in their fuel. They do not have a neutron moderator, and use less-moderating coolants. Maintaining a chain reaction requires the fuel to be more highly enriched in fissile material (about 20% or more) due to the relatively lower probability of fission versus capture by U-238. Fast reactors have the potential to produce less waste, but they are more difficult to build and more expensive to operate.

- Nuclear fusion. Fusion power is an experimental technology, generally with hydrogen as fuel.

- Radioactive decay. Examples include radioisotope thermoelectric generators as well as other types of atomic batteries, which generate heat and power by exploiting passive radioactive decay.

2. Classification by Moderator Material

- Graphite moderated reactors – These reactors account for only about 7% of the reactors in use today. The Russian RBMK reactor is an example of a graphite moderated reactor.

- Water moderated reactors – Water moderated reactors fall into two categories: light water moderated reactors and heavy water reactors. Light water reactors use ordinary water to moderate and cool the reactors. When at operating temperature, if the temperature of the water increases, its density drops, and fewer neutrons passing through it are slowed enough to trigger further reactions. That negative feedback stabilizes the reaction rate. Approximately 82% of the reactors in use today use light water for moderation. Heavy water reactors use deuterium-oxide, which is known as “heavy water’, and account for less than 10% of the world’s reactors.

- Light element moderated reactors. These reactors are moderated by lithium or beryllium. Molten salt reactors (MSRs) are moderated by a light elements such as lithium or beryllium, which are constituents of the coolant/fuel matrix salts Lithium Fluoride and Beryllium Fluoride. Liquid metal cooled reactors, such as one whose coolant is a mixture of Lead and Bismuth, may use Beryllium Oxide as a moderator.

3. Classification by coolant

- Water cooled reactor. Virtually all electric generating reactors operating in the United States are water cooled. Most are pressurized water reactors (PWR), and a lesser number are boiling water reactors (BWR).

- Liquid metal cooled reactor. Since water is a moderator, it cannot be used as a coolant in a fast reactor. Liquid metal coolants have included sodium, Sodium-Potassium alloy, lead, and lead-bismuth alloy.

- Gas cooled reactors. GCR's are cooled by a circulating inert gas, such as helium in high-temperature designs. Utilization of the heat varies, depending on the reactor. Some reactors run hot enough that the gas can directly power a gas turbine. Older designs usually run the gas through a heat exchanger to make steam for a steam turbine.

- Molten Salt Reactors. MSR's are cooled by circulating a molten salt, typically a eutectic mixture of fluoride salts, such as Lithium-Fluoride-Beryllium-Fluoride.

The following chapters cover the basic design of each type of commercial nuclear power plant. Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs) will be covered in significant detail since these are the most common reactors in use.

Chapter 2
Pressurized Water Reactor

Reaction: Thermal

Moderator: Water

Coolant: Water

Pressurized Water Reactors (PWR's) are the most common type of nuclear reactor in use today. In a PWR the primary coolant, which is superheated water, is pumped under high pressure to the reactor core, then the heated water transfers thermal energy to a steam generator. Ordinary water is used as both neutron moderators and coolant. In a PWR the water used as moderator and primary coolant is separate from the water used to generate steam and to drive a turbine. In order to efficiently convert the heat produced by the nuclear reaction into electricity, the water that moderates the neutron and cools the fuel elements is contained at pressures 150 times greater than atmospheric pressure. In 2010, there were 268 PWR's in operation.

Pressurized Water Reactor

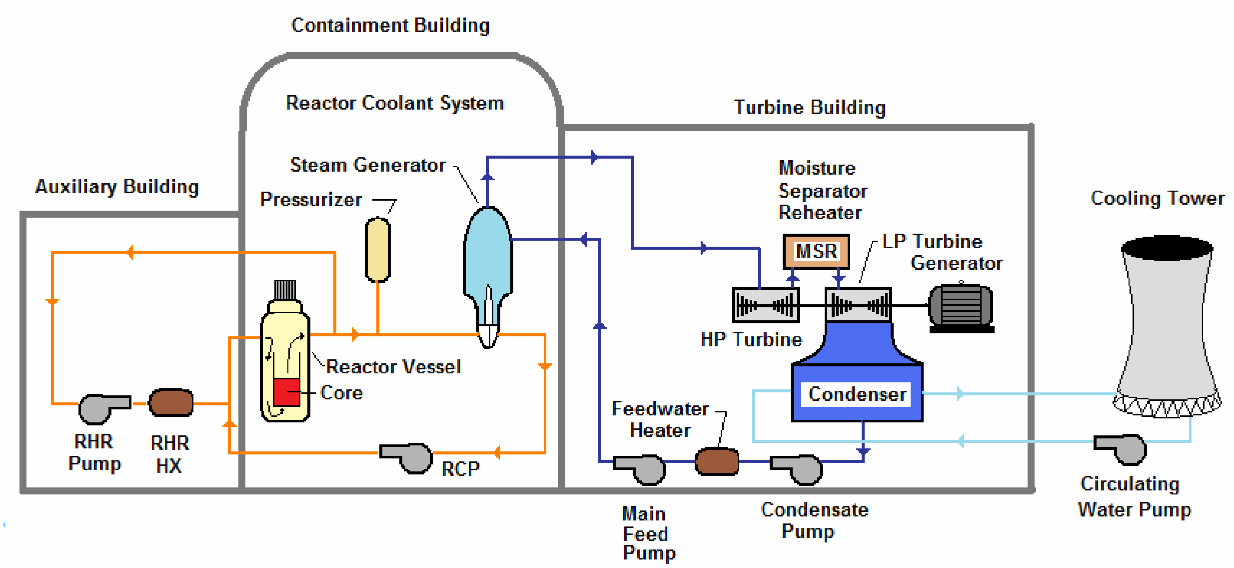


Figure 2

Overview

Refer to Figure 2 for a description of how a PWR plant works. The major structures at a pressurized water reactor plant are: The containment building, which houses the reactor and its high pressure steam generating equipment; the turbine building, which houses the steam turbines, condensers, and the electrical generator; and the auxiliary building, which houses normal and emergency support systems.

There are two major systems utilized to convert the heat generated in the fuel into electrical power. 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 high pressure turbine generator, where it is converted into electricity. The steam then passes through the low pressure turbine and after passing through the low pressure turbine, the steam is routed to the main condenser. Cool water, flowing through the tubes in the condenser, removes excess heat from the steam, which allows the steam to condense. The water is then pumped back to the steam generator for reuse.

The primary system is known as the, *Reactor Coolant System*, and consists of the reactor vessel, the steam generators, the reactor coolant pumps, and a pressurizer. A reactor coolant loop is a reactor coolant pump (RCP), and a steam generator. The primary function of the reactor coolant system is to transfer the heat from the fuel to the steam generators. A second function is to contain any fission products that escape the fuel.

Boron and control rods are used to maintain primary system temperature at the desired point. In order to decrease power, the operator closes turbine inlet valves. This results in less steam being drawn from the steam generators and results in the primary loop increasing in temperature. The higher temperature causes the reactor to fission less and decrease in power. The operator can then add boric acid and/or insert control rods to decrease temperature to the desired point.

Reactivity adjustment to maintain 100% power as the fuel is burned up in most commercial PWRs is normally achieved by varying the concentration of boric acid dissolved in the primary reactor coolant. 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 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 reactor vessel head directly into the fuel bundles, are operated for the following three reasons:

1. To start up the reactor.
2. To shut down the reactor.
3. To accommodate short term transients such as changes to load on the turbine.

The control rods may also be used to compensate for nuclear poison inventory and to compensate for nuclear fuel depletion. However, these effects are more usually accommodated by altering the primary coolant boric acid concentration.

Reactor

The reactor core and all associated support and alignment devices are housed within the *reactor vessel*. The major components are the reactor vessel, the core barrel, the reactor core, and the upper internals package. Figure 3 is a schematic of a PWR reactor vessel.

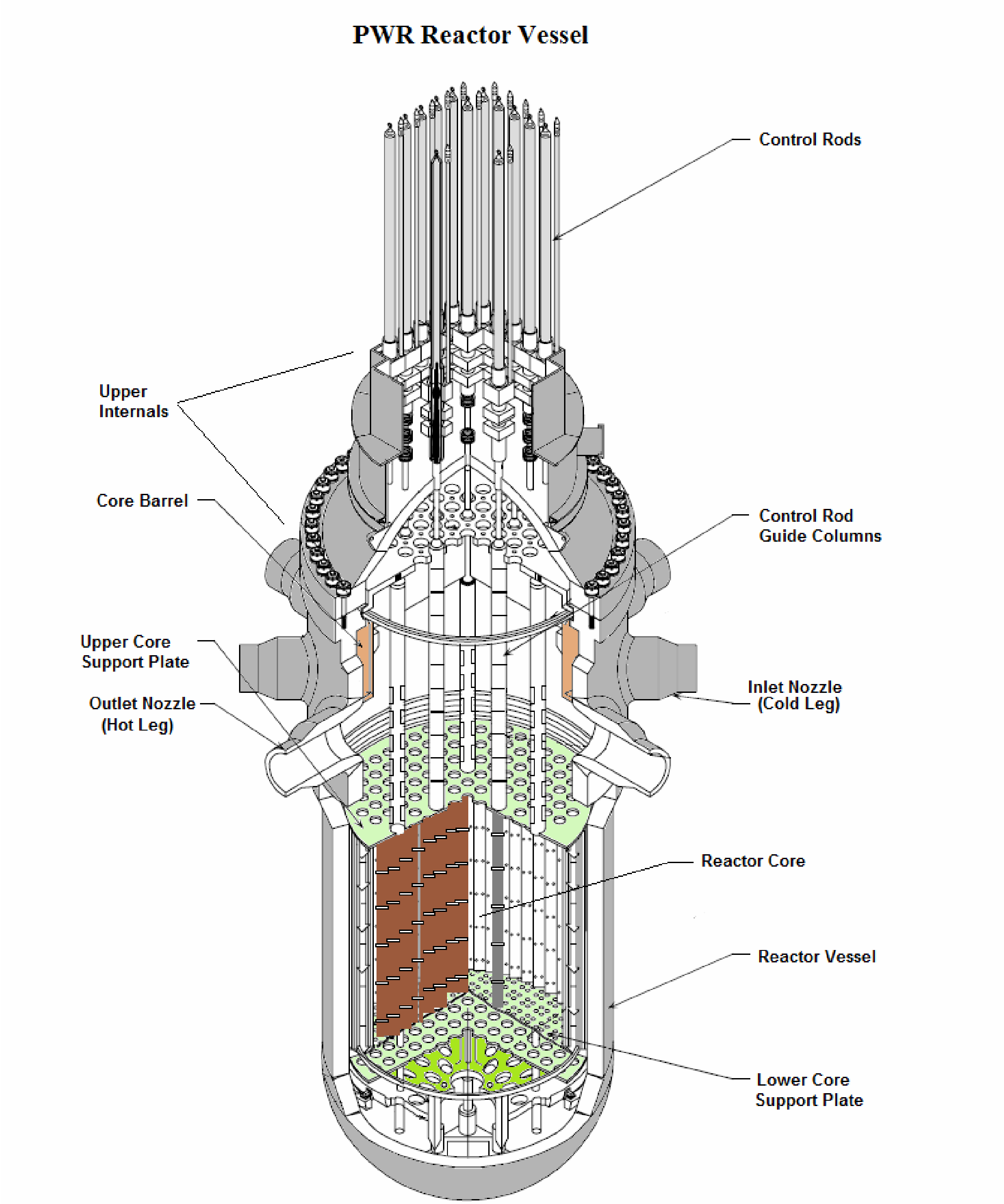


Figure 3

The reactor vessel is a cylindrical vessel with a hemispherical bottom head and a removable hemispherical top head. The top 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 slides down inside of the reactor vessel and houses the fuel. Toward the bottom of the core barrel, there is a lower core support plate on which the fuel assemblies sit. The core barrel and all of the lower internals actually hang inside the reactor vessel from the internals support ledge.

The upper internals package sits on top of the fuel. It contains the guide columns to guide the control rods when they are pulled from the fuel. The upper internals package prevents the core from trying to move up during operation due to the force from the coolant flowing through the assemblies.

The light water enters the bottom of the reactor core at about 275°C and is heated as it flows upwards through the reactor core to a temperature of about 300°C. The water remains liquid despite the high temperature due to the high pressure in the primary coolant loop, usually around 2,250 psig. The flow path for the reactor coolant through the reactor vessel is as follows:

- The coolant enters the reactor vessel at the inlet nozzle and hits the core barrel.
- The core barrel forces the water to flow downward in the space between the reactor vessel wall and the core barrel.
- After reaching the bottom of the reactor vessel, the flow is turned upward to pass through the fuel assemblies.
- The coolant flows all around and through the fuel assemblies, removing the heat produced by the fission process.
- The water, which is now quite hot, enters the upper internals region, where it is routed out the outlet nozzle and goes on to the steam generator.

The reactor coolant flows from the reactor to the *steam generator*. Inside of the steam generator, the hot reactor coolant flows inside of the many tubes. The secondary coolant, or feedwater, flows around the outside of the tubes, where it picks up heat from the primary coolant. When the feedwater absorbs sufficient heat, it starts to boil and form steam.

It is important to maintain the moisture content of the steam as low as possible to prevent damage to the turbine blades. In one popular design, the steam/water mixture passes through multiple stages of moisture separation. One stage causes the mixture to spin, which slings the water to the outside. The water is then drained back to be used to make more steam.

The drier steam is routed to the second stage of separation. In this stage, the mixture is forced to make rapid changes in direction. Because of the steam's ability to change direction and the water's inability to change, the steam exits the steam generator, and the water is drained back for reuse.

The *reactor coolant pump* (RCP) provides forced primary coolant flow to remove the heat being generated by the fission process. Even without a pump, there would be natural circulation flow through the reactor. This flow is not sufficient to remove the heat being generated when the reactor is at power. However, natural circulation flow is sufficient for heat removal when the plant is shutdown.

The reactor coolant enters the pump from the outlet of the steam generator and enters the inlet or cold leg side of the reactor vessel. The coolant will then pass through the fuel to collect more heat and is sent back to the steam generators. The reactor coolant pump is a large, air cooled, electric motor capable of delivering approximately 100,000 gallons per minute per pump.

The *pressurizer* is the component in the reactor coolant system which provides a means of controlling the system pressure. The pressurizer operates with a mixture of steam and water in equilibrium. If pressure starts to deviate from the desired value, the various components will actuate to bring pressure back to the normal operating point. The cause of the pressure deviation is normally associated with a change in the temperature of the reactor coolant system. If reactor coolant system temperature starts to increase, the density of the reactor coolant will decrease, and the water will take up more space. Since the pressurizer is connected to the reactor coolant system via the surge line, the water will expand up into the pressurizer. This will cause the steam in the top of the pressurizer to be compressed, and therefore, the pressure to increase. The opposite effect will occur if the reactor coolant system temperature decreases. The water will become denser, and will occupy less space. The level in the pressurizer will decrease, which will cause a pressure decrease. For a pressure increase or decrease, the pressurizer will operate to bring pressure back to normal.

Secondary Systems

During normal operation, the heat produced by the fission process is removed by the reactor coolant and transferred to the secondary coolant in the steam generators. Here, the secondary coolant is boiled into steam and sent to the main turbine.

The major secondary systems of a PWR are the main steam system and the condensate/feedwater system. Since the primary and secondary systems are physically separated from each other, the secondary system contains little or no radioactive material.

The main steam system starts at the outlet of the steam generator. The steam is routed to the high pressure main turbine. After passing through the high pressure turbine, the steam is piped to the moisture separator/reheaters (MSRs). In the MSRs, the steam is dried with moisture separators and reheated using other steam as a heat source. From the MSRs, the steam goes to the low pressure turbines. After passing through the low pressure turbines, the steam goes to the main condenser, which is operated at a vacuum to allow for the greatest removal of energy by the low pressure turbines. The steam is condensed into water by the flow of circulating water through the condenser tubes.

At this point, the *condensate/feedwater system* starts. The condensed steam collects in the hotwell area of the main condenser. The condensate pumps take suction on the hotwell to increase the pressure of the water. The condensate then passes through a cleanup system to remove any impurities in the water. This is necessary because the steam generator acts as a concentrator. If the impurities are not removed, they will be left in the steam generator after the steam forming process, and this could reduce the heat transfer capability of the steam generator and/or damage the steam generator tubes. The condensate then passes through some low pressure feedwater heaters. The temperature of the condensate is increased in the heaters by using steam from the low pressure turbine. The condensate flow then enters the suction of the main feedwater pumps, which increases the pressure of the water high enough to enter the steam generator. The feedwater now passes through a set of high pressure feedwater heaters, which are heated by extraction steam from the high pressure turbine (heating the feedwater helps to increase the efficiency of the plant). The flow rate of the feedwater is controlled as it enters the steam generators.

While not shown in Figure 2, the *chemical and volume control system* (CVCS) is a major support system for the reactor coolant system. Some of the functions of the system are to:

- Purify the reactor coolant system using filters and demineralizers,
- Add and remove boron as necessary, and
- Maintain the level of the pressurizer at the desired setpoint.

In a procedure known as *letdown*, a small amount of water is continuously routed through the CVCS. This provides a continuous cleanup of the reactor coolant system which maintains the purity of the coolant and helps to minimize the amount of radioactive material in the coolant.

This water has been cooled by the heat exchangers and cleaned by the filters and demineralizers. There is also a path to route the letdown flow to the radioactive waste system for processing and/or disposal.

Even after the reactor has been shutdown, there is a significant amount of heat produced by the decay of fission products (decay heat). The amount of heat produced by decay heat is sufficient to cause fuel damage if not removed. Therefore, systems must be designed and installed in the plant to remove the decay from the core and transfer that heat to the environment, even in a shutdown plant condition. Also, if it is desired to perform maintenance on reactor coolant system components, the temperature and pressure of the reactor coolant system must be reduced low enough to allow personnel access to the equipment.

The auxiliary feedwater system and the steam dump system (turbine bypass valves) work together to allow the operators to remove the decay heat from the reactor. The auxiliary feedwater system pumps water from the condensate storage tank to the steam generators. This water is allowed to boil to make steam. The steam can then be dumped to the main condenser through the steam dump valves. The circulating water will then condense the steam and take the heat to the environment. If the steam dump system is not available (for example, no circulating water for the main condenser), the steam can be dumped directly to the atmosphere through the atmospheric relief valves. By using either method of steam removal, the heat is being removed from the reactor coolant system, and the temperature of the reactor coolant system can be reduced to the desired level.

At some point, the decay heat being produced will not be sufficient to generate enough steam in the steam generators to continue the cool down. When the reactor coolant system pressure and temperature have been reduced to within the operational limits, the *residual heat removal* system (RHR) will be used to continue the cool down by removing heat from the core and transferring it to the environment. This is accomplished by routing some of the reactor coolant through the residual heat removal system heat exchanger, which is cooled by the component cooling water system. The heat removed by the component cooling water system is then transferred to the service water system in the component cooling water heat exchanger. The heat picked up by the service water system will be transferred directly to the environment from the service water system. The residual heat removal system can be used to cool the plant down to a low enough temperature that personnel can perform any maintenance functions, including refueling.

Safety Systems

A key safety system in a PWR is the emergency core cooling system. There are two purposes of the *emergency core cooling systems* (ECCS). The first is to provide core cooling to minimize fuel damage following a loss of coolant accident. This is accomplished by the injection of large amounts of cool, borated water into the reactor coolant system. The second is to provide extra neutron poisons to ensure the reactor remains shutdown following the cool down associated with a main steam line rupture, which is accomplished by the use of the same borated water source. This water source is called the refueling water storage tank.

To perform this function of injection of large quantities of borated water, the emergency core cooling systems consist of four separate systems. In order of highest pressure to lowest pressure, these systems are:

- High pressure injection system
- Intermediate pressure injection system
- Cold leg accumulators
- Low pressure injection system

These systems must be able to operate when the normal supply of power is lost to the plant. For this reason, these systems are powered from the plant emergency (diesel generators) power system.

The *high pressure injection system* uses the pumps in the chemical and volume control system. Upon receipt of an emergency actuation signal, the system will automatically realign to take water from the refueling water storage tank and pump it into the reactor coolant system. The high pressure injection system is designed to provide water to the core during emergencies in which reactor coolant system pressure remains relatively high (such as small break in the reactor coolant system, steam break accidents, and leaks of reactor coolant through a steam generator tube to the secondary side).

The *intermediate pressure injection system* is also designed for emergencies in which the primary pressure stays relatively high, such as small to intermediate size primary breaks. Upon an emergency start signal, the pumps will take water from the refueling water storage tank and pump it into the reactor coolant system.

The *cold leg accumulators* do not require electrical power to operate. These tanks contain large amounts of borated water with a pressurized nitrogen gas bubble in the top. If the pressure of the primary system drops below low enough, the nitrogen will force the borated water out of the tank and into the reactor coolant system. These tanks are designed to provide water to the reactor coolant system during emergencies in which the pressure of the primary drops very rapidly, such as large primary breaks.

The *low pressure injection system* (residual heat removal) is designed to inject water from the refueling water storage tank into the reactor coolant system during large breaks, which would cause a very low reactor coolant system pressure. In addition, the residual heat removal system has a feature that allows it to take water from the containment sump, pump it through the residual heat removal system heat exchanger for cooling, and then send the cooled water back to the reactor for core cooling. This is the method of cooling that will be used when the refueling water storage tank goes empty after a large primary system break. This is called the long term core cooling or recirculation mode.

The reactor coolant system is located inside the containment building. The containment building is designed to withstand the pressures and temperatures that would accompany a high energy fluid release into the building, but exposure to high temperature and pressure over a long period of time would tend to degrade the concrete. If a break occurred in the primary system, the coolant that is released into the containment building would contain radioactive material. If the concrete developed any cracks, the high pressure in the containment would tend to force the radioactive material out of the containment and into the environment.

To limit the leakage out of containment following an accident, there is a steel liner that covers the inside surface of the containment building. This liner acts as a vapor proof membrane to prevent any gas from escaping through any cracks that may develop in the concrete.

There are also two systems designed with the purpose of reducing containment temperature and pressure after an accident in the containment building. The fan cooler system circulates the air through heat exchangers to accomplish the cooling. The second system is the containment spray system.

Upon the occurrence of either a secondary break or primary break inside the containment building, the containment atmosphere would become filled with steam. To reduce the pressure and temperature of the building, the containment spray system is automatically started. The containment spray pump will take suction from the refueling water storage tank and pump the water into spray rings located in the upper part of the containment. The water droplets, being cooler than the steam, will remove heat from the steam, which will cause the steam to condense. This will cause a reduction in the pressure of the building and will also reduce the temperature of the containment atmosphere. Like the residual heat removal system, the containment spray system has the capability to take water from the containment sump if the refueling water storage tank goes empty.

Advantages and Disadvantages

The following is a list of the advantages and disadvantages of a pressurized water reactor.

Advantages of a PWR are:

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

Disadvantages of a PWR include:

- The coolant water must be highly pressurized to remain liquid at high temperatures. This requires high strength piping and a heavy pressure vessel.
- The higher pressure can increase the consequences of a loss of coolant accident.
- Pressurized water reactors cannot be refueled while operating.
- The high temperature water coolant with boric acid dissolved in it is corrosive to carbon steel.
- The fuel must be enriched.

Chapter 3
Boiling Water Reactors

In a Boiling Water Reactor (BWR), ordinary light water is used as both a moderator and coolant, like the PWR. However unlike the PWR, in a Boiling Water Reactor there is no separate secondary steam cycle. The water from the reactor is converted into steam and used to directly drive the generator turbine. BWRs are the second most commonly used types of reactors. In 2010, there were 93 units in operation.

Reaction: Thermal

Moderator: Water

Coolant: Water

The BWR uses demineralized water – or light water - as a coolant and neutron moderator. Heat is produced by nuclear fission in the reactor core, and this causes the cooling water to boil, producing steam. The steam is directly used to drive a turbine, after which it is cooled in a condenser and converted back to liquid water. This water is then returned to the reactor core, completing the loop. The cooling water is maintained at about 1100 psi, so that it boils in the core at about 285 °C.

Overview

See Figure 4 for an overview of a boiling water reactor power plant. This figure will be used to discuss the major components of a BWR.

Boiling Water Reactors

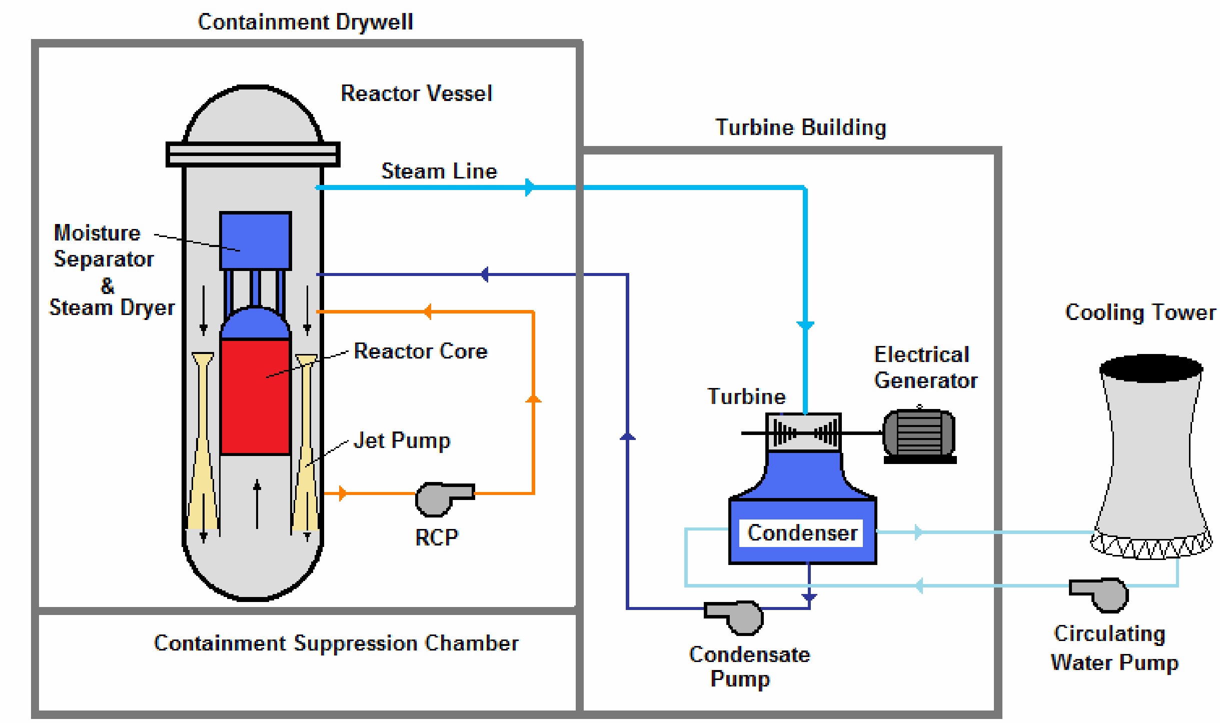


Figure 4

Inside the boiling water reactor vessel, a steam-water mixture is produced when very pure water moves upward through the core absorbing heat. Reactor power is controlled via two methods: by inserting or withdrawing control rods and by changing the water flow through the reactor core.

Positioning (withdrawing or inserting) control rods is the normal method for controlling power when starting up a BWR. As control rods are withdrawn, neutron absorption decreases in the control material and increases in the fuel, so reactor power increases. As control rods are inserted, neutron absorption increases in the control material and decreases in the fuel, so reactor power decreases.

Changing the flow of water through the core is the normal method for controlling power when operating. Power may be varied from approximately 30% to 100% of rated power by changing the reactor recirculation system flow by varying the speed of the recirculation pumps. As flow of water through the core is increased, steam bubbles, or voids, are more quickly removed from the core, the amount of liquid water in the core increases, neutron moderation increases, more neutrons are slowed down to be absorbed by the fuel, and reactor power increases. As flow of water through the core is decreased, steam voids remain longer in the core, the amount of liquid water in the core decreases, neutron moderation decreases, fewer neutrons are slowed down to be absorbed by the fuel, and reactor power decreases.

The major difference in the operation of a BWR from other nuclear reactor systems is the steam void formation in the core. The steam-water mixture leaves the top of the core and enters the two stages of moisture separation, where water droplets are removed before the steam is allowed to enter the steam line. The steam line, in turn, directs the steam to the main turbine causing it to turn the turbine and the attached electrical generator. The unused steam is exhausted to the condenser where it is condensed into water. The resulting water is pumped out of the condenser with a series of pumps and back to the reactor vessel. The recirculation pumps and jet pumps allow the operator to vary coolant flow through the core and change reactor power.

Steam produced in the reactor core passes through steam separators and dryer plates above the core and then directly to the turbine, which is part of the reactor circuit. Because the water around the core of a reactor is always contaminated with traces of radioactive particles, the turbine must be shielded during normal operation, and radiological protection must be provided during maintenance. The increased cost related to operation and maintenance of a BWR tends to balance the savings due to the simpler design and greater thermal efficiency of a BWR when compared with a PWR. Most of the radioactivity in the water is very short-lived, so the turbine hall can be entered soon after the reactor is shut down.

Steam exiting from the turbine flows into condensers located underneath the low pressure turbines where the steam is cooled and returned to the liquid state (condensate). The condensate is then pumped through feedwater heaters that raise its temperature. Feedwater from the feedwater heaters enters the reactor pressure vessel through nozzles high on the vessel, well above the top of the nuclear fuel assemblies but below the water level.

The entering feedwater combines with water exiting the water separators. The feedwater cools the saturated water from the steam separators. This water now flows down into either jet pumps or internal recirculation pumps that provide additional pumping power. The water now makes a 180 degree turn and moves up through the lower core plate into the nuclear core where the fuel elements heat the water.

The heating from the core creates a thermal head that assists the recirculation pumps in re-circulating the water inside of the reactor pressure vessel (RPV). A BWR can be designed with no recirculation pumps and rely entirely on the thermal head to re-circulate the water. The forced recirculation head from the recirculation pumps is very useful in controlling power, however. The thermal power level is easily varied by simply increasing or decreasing the forced recirculation flow through the recirculation pumps.

The two phase fluid (water and steam) above the core enters the riser area, which is the upper region contained inside of the shroud. The height of this region may be increased to increase the thermal natural recirculation pumping head. At the top of the riser area is the water separator. By swirling the two phase flow in cyclone separators, the steam is separated and rises upwards towards the steam dryer while the water remains behind and flows horizontally out and down. Then it combines with the feedwater flow and the cycle repeats.

The saturated steam that rises above the separator is dried. The steam then exits the RPV through the main steam lines and goes to the turbine.

Reactor Components

The reactor vessel assembly consists of the reactor vessel and its internal components, including the core support structures, core shroud, moisture removal equipment, and jet pump assemblies. The purposes of the reactor vessel assembly are to:

- House the reactor core
- Serve as part of the reactor coolant pressure boundary
- Support and align the fuel and control rods
- Provide a flow path for circulation of coolant past the fuel
- Remove moisture from the steam exiting the core
- Provide a re-floodable volume for a loss of coolant accident.

The reactor vessel is vertically mounted within the drywell and consists of a cylindrical shell with an integral rounded bottom head. The top head is also rounded in shape and is removable to facilitate refueling operations. The vessel assembly is supported by the vessel support skirt which is mounted to the reactor vessel support pedestal.

The internal components of the reactor vessel are supported from the bottom head and/or vessel wall. The reactor core is made up of fuel assemblies and control rods. The components making up the remainder of the reactor vessel internals are the jet pump assemblies, steam separators, steam dryers, feedwater spargers, and core spray spargers. The jet pump assemblies are located in the region between the core shroud and the vessel wall, submerged in water. The jet pump assemblies are arranged in two semicircular groups, with each group being supplied by a separate recirculation pump.

The worst case loss of coolant accident, with respect to core cooling, is a recirculation line break. In this event, reactor water level decreases rapidly, uncovering the core. However, several emergency core cooling systems automatically provide makeup water to the nuclear core within the shroud, providing core cooling.

Each control cell consists of a control rod and four fuel assemblies that surround it. Unlike the pressurized water reactor fuel assemblies, the boiling water reactor fuel bundle is enclosed in a fuel channel to direct coolant up through the fuel assembly and act as a bearing surface for the control rod. In addition, the fuel channel protects the fuel during refueling operations. The power of the core is regulated by movement of bottom entry control rods.

The *reactor water cleanup system* maintains a high reactor water quality by removing fission products, corrosion products, and other soluble and insoluble impurities. The reactor water cleanup pump takes water from the recirculation system and the vessel bottom head and pumps the water through heat exchangers to cool the flow. The water is then sent through filter/demineralizers for cleanup. After cleanup, the water is returned to the reactor vessel via the feedwater piping.

Heat is removed during normal power operation by generating steam in the reactor vessel and then using that steam to generate electrical energy. When the reactor is shutdown, the core will still continue to generate decay heat. The heat is removed by bypassing the turbine and dumping the steam directly to the condenser. Water is pumped from the reactor recirculation loop through a heat exchanger and back to the reactor via the recirculation loop. The recirculation loop is used to limit the number of penetrations into the reactor vessel.

The *reactor core isolation cooling* (RCIC) system is designed to remove the residual heat of the fuel from the reactor once it has been shut down. It injects approximately 600 GPM into the reactor core for this purpose, at high pressure. The RCIC provides makeup water to the reactor vessel for core cooling when the main stream lines are isolated and the normal supply of water to the reactor vessel is lost. The RCIC system consists of a turbine-driven pump, piping, and valves necessary to deliver water to the reactor vessel at operating conditions.

The RCIC turbine is driven by steam supplied by the main steam lines. The turbine exhaust is routed to the suppression pool. The turbine-driven pump supplies makeup water from the condensate storage tank, with an alternate supply from the suppression pool, to the reactor vessel via the feedwater piping. Initiation of the system is accomplished automatically on low water level in the reactor vessel or manually by the operator.

The *standby liquid control system* injects a neutron poison (boron) into the reactor vessel to shutdown the chain reaction, independent of the control rods, and maintains the reactor shutdown as the plant is cooled to maintenance temperatures. The standby liquid control system consists of a heated storage tank, two positive displacement pumps, two explosive valves, and the piping necessary to inject the neutron absorbing solution into the reactor vessel. The standby liquid control system is manually initiated and provides the operator with a relatively slow method of achieving reactor shutdown conditions.

The BWR reactor core continues to produce heat from radioactive decay after the fission reactions have stopped, making a core damage incident possible in the event that all safety systems have failed and the core does not receive coolant. A boiling water reactor has a negative void coefficient, that is, the neutron output of the reactor decreases as the proportion of steam to liquid water increases inside the reactor. Unlike a pressurized water reactor which contains no steam in the reactor core, a sudden increase in BWR steam pressure from the reactor will result in a sudden decrease in the proportion of steam to liquid water inside the reactor. The increased ratio of water to steam will lead to increased neutron moderation, which in turn will cause an increase in the power output of the reactor. This type of event is referred to as a "pressure transient".

The BWR is specifically designed to respond to pressure transients, having a "pressure suppression" type of design which vents overpressure using safety relief valves to below the surface of a pool of liquid water within the containment, known as the "wetwell" or "torus". In addition, the reactor will have already have rapidly shut down before the transient affects the reactor pressure vessel.

Safety Systems

The *Reactor Protection System* (RPS) is a designed to automatically, rapidly, and completely shut down the Nuclear Steam Supply System if some event occurs that could result in the reactor entering an unsafe operating condition. In addition, the RPS can automatically spin up the Emergency Core Cooling System (ECCS) if needed.

If the reactor is at power or ascending to power (i.e. if the reactor is critical; the control rods are withdrawn to the point where the reactor generates more neutrons than it absorbs) there are safety-related contingencies that may arise that necessitate a rapid shutdown of the reactor, which is known as a "SCRAM". The SCRAM is a manually-triggered or automatically-triggered rapid insertion of all control rods into the reactor, which will take the reactor to decay heat power levels within seconds.

While the RPS is designed to prevent contingencies from happening, the ECCS is designed to respond to contingencies if they do happen. The ECCS is a set of interrelated safety systems that are designed to protect the core within the reactor pressure vessel from overheating. These systems accomplish this by maintaining reactor pressure vessel cooling water level, or if that is impossible, by directly flooding the core with coolant.

The *emergency core cooling systems* (ECCS) provide core cooling under loss of coolant accident conditions to limit fuel cladding damage. The emergency core cooling systems consist of two high pressure and two low pressure systems. They are the:

1. High pressure coolant injection system (HPCI)
2. Automatic depressurization system (ADS)
3. Low pressure coolant injection (LPCI)
4. Core spray system (CS)

The manner in which the emergency core cooling systems operate to protect the core is a function of the rate at which reactor coolant inventory is lost from the break in the nuclear system process barrier. The high pressure coolant injection system is designed to operate while the nuclear system is at high pressure.

The core spray system and low pressure coolant injection mode of the residual heat removal system are designed for operation at low pressures. If the break in the nuclear system process barrier is of such a size that the loss of coolant exceeds the capability of the high pressure coolant injection system, reactor pressure decreases at a rate fast enough for the low pressure emergency core cooling systems to commence coolant injection into the reactor vessel in time to cool the core.

The *high pressure coolant injection* (HPCI) system is an independent emergency core cooling system requiring no auxiliary ac power, plant air systems, or external cooling water systems to perform its purpose of providing make up water to the reactor vessel for core cooling under small and intermediate size loss of coolant accidents. The high pressure coolant injection system can supply make up water to the reactor vessel from above rated reactor pressure to a reactor pressure below that at which the low pressure emergency core cooling systems can inject. This system is the first line of defense in the Emergency Core Cooling System. HPCI is designed to inject substantial quantities of water into the reactor while it is at high pressure. HPCI is powered by steam from the reactor, and takes approximately ten seconds to spin up from an initiating signal, and can deliver approximately 5,000 GPM to the core. This is usually enough to keep water levels sufficient to avoid automatic depressurization except in a major contingency, such as a large break in the makeup water line.

The *automatic depressurization system* (ADS) consists of redundant logics capable of opening selected safety relief valves, when required, to provide reactor depressurization for events involving small or intermediate size loss of coolant accidents if the high pressure coolant injection system is not available or cannot recover reactor vessel water level. It is designed to activate in the event that the reactor pressure vessel (RPV) is retaining pressure, but RPV water level cannot be maintained using high pressure cooling alone, and low pressure cooling must be initiated. When ADS fires, it rapidly releases pressure from the RPV in the form of steam through pipes that are piped to below the water level in the suppression pool, which is designed to condense the steam released by ADS or other safety valve activation into water allowing the low pressure cooling systems with extremely large and robust comparative coolant injection capacities to be brought to bear on the reactor core.

The *Low Pressure Core Spray System* (LPCS) is designed to suppress steam generated by a major contingency. The core spray system consists of two separate and independent pumping loops, each capable of pumping water from the suppression pool into the reactor vessel. When activated, the LPCS delivers approximately 12,500 GPM of water in a deluge from the top of the core.

The *Low Pressure Coolant Injection System* (LPCI) is the "heavy artillery" in the ECCS. It is capable of injecting 40,000 GPM of water into the core. Combined with the LPCS to keep steam pressure low, the LPCI can suppress all contingencies by rapidly and completely flooding the core with coolant. Core cooling is accomplished by spraying water on top of the fuel assemblies. The low pressure coolant injection mode of the residual heat removal system provides makeup water to the reactor vessel for core cooling under loss of coolant accident conditions. The residual heat removal system is a multipurpose system with several operational modes, each utilizing the same major pieces of equipment. The low pressure coolant injection mode is the dominant mode and normal valve lineup configuration of the residual heat removal system. The low pressure coolant injection mode operates automatically to restore and, if necessary, maintain the reactor vessel coolant inventory to preclude excessive fuel cladding temperatures. During low pressure coolant injection operation, the residual heat removal pumps take water from the suppression pool and discharge to the reactor vessel.

Containment System

During the evolution of the boiling water reactors, three major types of containments were built. The containment designs are the Mark I, Mark II, and the Mark III. The Mark I and II, designs consist of a drywell and suppression pool whereas the Mark III has a primary containment and a drywell. All three containment designs use the principle of pressure suppression for loss of coolant accidents. The primary containment is designed to condense steam and to contain fission products released from a loss of coolant accident so that allowable offsite radiation doses are not exceeded and to provide a heat sink and water source for certain safety related equipment.

Advantages and Disadvantages

The following is a list of the advantages and disadvantages of a boiling water reactor.

Advantages

- The reactor vessel and associated components operate at a substantially lower pressure than a PWR.
- Operates at a lower nuclear fuel temperature.
- Fewer components due to no steam generators and no pressurizer vessel.
- Lower probability of a rupture causing loss of coolant compared to a PWR.
- Can operate at lower core power density levels using natural circulation without forced flow.

Disadvantages

- Managing consumption of nuclear fuel during operation due to water-steam mixture in the upper part of the core is complex.
- Much larger pressure vessel than for a PWR of similar power, with correspondingly higher cost.
- Control rods are inserted from below the reactor and require energy to operate instead of the fail safe drop down control rods of a PWR.

Chapter 4
Pressurized Heavy Water Reactor

Reaction: Thermal

Moderator: Water (Heavy)

Coolant: Water (Heavy)

Pressurized Heavy Water reactors (PHWR) are similar to PWRs but use water enriched with the deuterium isotope of hydrogen as the moderator and coolant. This is called *heavy water* and the advantage of using heavy water as the moderator is that natural, un-enriched Uranium can be used to drive the nuclear reactor. The heavy water surrounds the fuel assemblies and primary coolant.

The most common form of pressurized heavy water reactor is the CANDU reactor. CANDU is an acronym for a CANada Deuterium Uranium reactor. These reactors are heavy-water (deuterium-oxide) cooled, and moderated, pressurized water reactors. Instead of using a single large pressure vessel as in a PWR, the fuel is contained in hundreds of pressure tubes. These reactors are fueled with natural uranium and are thermal neutron reactor designs. PHWRs can be refueled while at full power, which makes them very efficient in their use of uranium. CANDU reactors use about 30–40% less mined uranium than light-water reactors per unit of electrical energy produced. In 2010, there were 42 CANDU reactors around the world including a few derivative CANDU reactors.

The CANDU reactor is conceptually similar to most light water reactors, although it differs in the details. See Figure 5 for a schematic of a CANDU reactor.

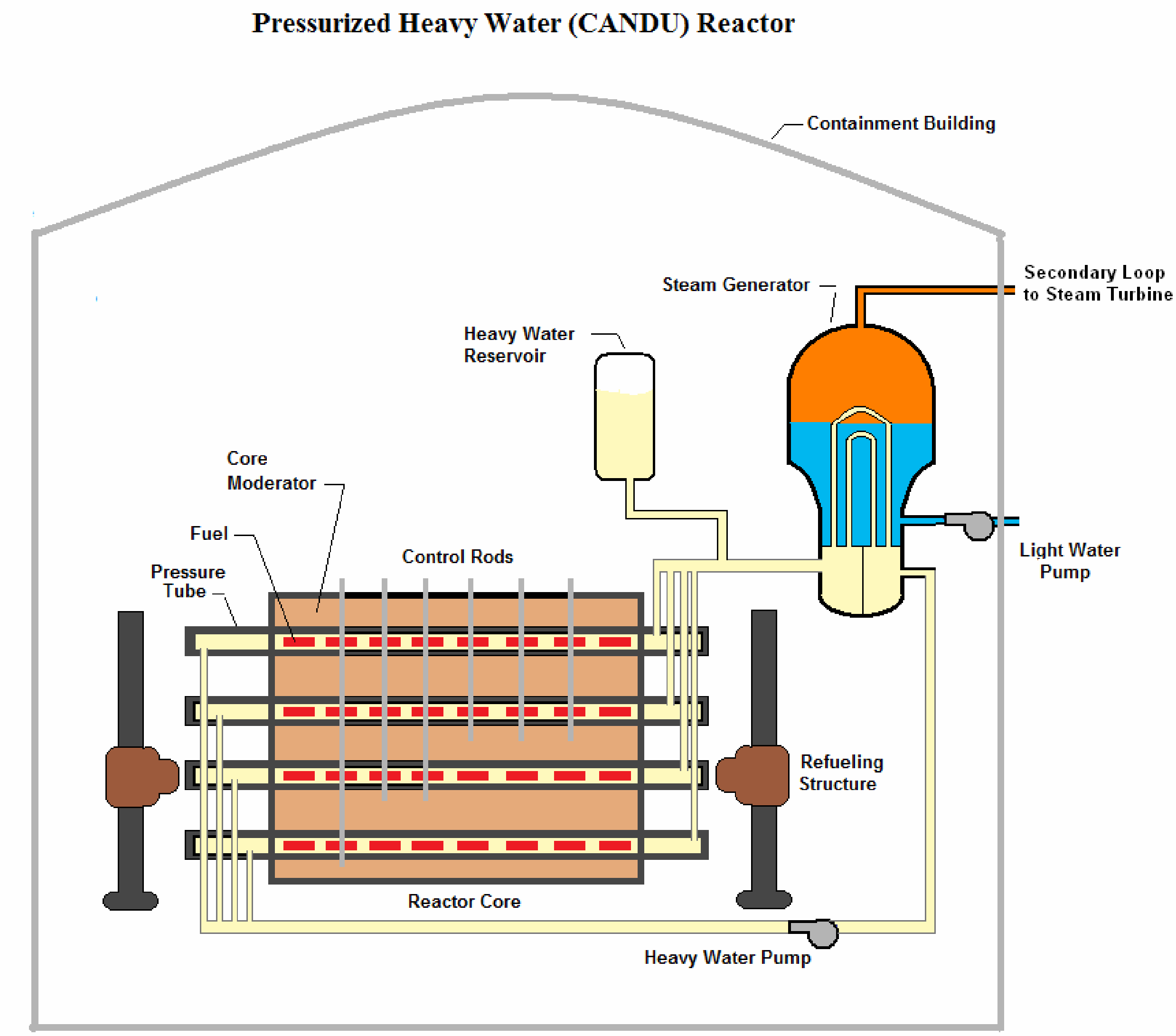


Figure 5

Like other water moderated reactors, fission reactions in the reactor core heat pressurized water in a primary cooling loop. A heat exchanger transfers the heat to a secondary cooling loop, which powers a steam turbine with an electrical generator attached to it. Any excess heat energy in the steam after flowing through the turbine is rejected into the environment in a variety of ways, most typically into a large body of cool water, such as a lake, river, ocean or a cooling tower. More recently built CANDU plants use a discharge-diffuser system that limits the thermal effects in the environment to within natural variations.

Instead of a steel pressure vessel used in most light water reactors, the pressure is contained in small tubes that contain the fuel bundles. These small tubes are easier to fabricate than a large pressure vessel. In order to allow the neutrons to flow freely between the bundles, the tubes are made of a zirconium alloy, which is highly transparent to neutrons. The zircalloy tubes are surrounded by a much larger low-pressure tank known as a calandria, which contains the majority of the moderator.

Qinshan CANDU Plant

Traditional reactor designs using light water as a moderator will absorb too many neutrons to allow a chain reaction to occur in natural uranium due to the low density of active nuclei. By using a heavy water, such as deuterium-oxide as the moderator, potentially all of the neutrons being released can be moderated and used in reactions with the 235U, in which case there is enough 235U in natural uranium to sustain criticality. Deuterium-oxide reacts dynamically with the neutrons in a similar fashion to light water, albeit with less energy transfer. The advantage is that it already has the extra neutron that light water would normally tend to absorb, reducing the absorption rate. Also, the low temperature of the moderator (below the boiling point of water) reduces changes in the neutrons' speeds from collisions with the moving particles of the moderator. The neutrons therefore are easier to keep near the optimum speed to cause fissioning.

The use of heavy water moderator is the key to the CANDU system, enabling the use of natural uranium as fuel, which means that it can be operated without expensive uranium enrichment facilities. Additionally, the mechanical arrangement of the CANDU, which places most of the moderator at lower temperatures, is particularly efficient because the resulting thermal neutrons are "more thermal" than in traditional designs, where the moderator normally runs hot. This means that the CANDU is not only able to "burn" natural uranium and other fuels, but tends to do so more effectively as well.

Another unique feature of heavy-water moderation is the greater stability and better coherence of the chain reaction throughout the whole volume of the reactor. This is due to the relatively low binding energy of the deuterium nucleus, leading to some energetic neutrons and gamma rays inside the reactor breaking deuterium nuclei apart and producing extra neutrons. Since gamma rays and neutrons travel for many tens of centimeters through water, an increased rate of chain reaction in any one part of the reactor will produce a wider, less concentrated response from the rest of the reactor, preventing local overheating and allowing various negative feedbacks to stabilize the reaction. This effect tends to even out the power level in the reactor. Neutrons produced directly by fission have a higher average energy than the delayed neutrons emitted in the decay of fission fragments; therefore the prompt neutrons are much more likely to multiply directly.

The large thermal mass of the moderator provides a significant heat sink that acts as an additional safety feature. If a fuel assembly were to overheat and deform within its fuel channel, the resulting change of geometry permits high heat transfer to the cool moderator, thus preventing the breach of the fuel channel, and the possibility of a meltdown. Furthermore, because of the use of natural uranium as fuel, this reactor cannot sustain a chain reaction if its original fuel channel geometry is altered in any significant manner.

As previously mentioned, in a traditional light water reactor design, the entire reactor core is a single large pressure vessel containing the light water, which acts as moderator and coolant, and the fuel arranged in a series of long bundles running the length of the core. To refuel such a reactor, it must be shut down, the pressure dropped, the lid removed, and a significant fraction of the core inventory, such as one-third, replaced in a batch procedure. The CANDU's calandria-based design allows individual fuel bundles to be removed without taking the reactor off-line, improving overall duty cycle or capacity factor.

A CANDU fuel assembly consists of a number of zircalloy tubes containing ceramic pellets of fuel arranged into a cylinder that fits within the fuel channel in the reactor. The most common CANDU reactors use an assembly of either 28 or 37 half-meter-long fuel tubes with 12 such assemblies lying end to end in a fuel channel. Some newer designs use a bundle with 43-tubes and two different pellet sizes.

A number of distributed light-water compartments called *liquid zone controllers* help control the rate of fission. The liquid zone controllers absorb excess neutrons and slow the fission reaction in their regions of the reactor core.

CANDU reactors employ two independent, fast-acting safety shutdown systems. Shutoff rods penetrate the calandria vertically and lower into the core in the case of a safety-system trip. A secondary shutdown system involves injecting a high-pressure solution directly into the low-pressure moderator.

Compared with light water reactors, a heavy water design is neutron rich. This makes the CANDU design suitable for burning a number of alternative nuclear fuels. To date, the fuel to gain the most attention is mixed oxide fuel (MOX). MOX is a mixture of natural uranium and plutonium, such as that extracted from former nuclear weapons. Plutonium can also be extracted from spent nuclear fuel reprocessing. While this consists usually of a mixture of isotopes that is not attractive for use in weapons, it can be used in a MOX formulation reducing the net amount of nuclear waste.

Plutonium isn't the only fissile material in spent nuclear fuel that CANDU reactors can utilize. Because the CANDU reactor was designed to work with natural uranium, CANDU fuel can be manufactured from the used (depleted) uranium found in light water reactor spent fuel. Typically this "Recovered Uranium" has a U-235 enrichment of around 0.9%, and compares favorably with natural uranium which has a U-235 abundance of roughly 0.7%. The recovered uranium that is unusable in a light water reactor is an excellent fuel source for a CANDU reactor. Therefore, a CANDU reactor can extract an additional 30-40% of energy from recovered uranium. CANDU reactors can also breed fuel from natural thorium, if uranium is unavailable.

Advantages and Disadvantages

The following is a list of the advantages and disadvantages of a pressurized heavy water reactor.

The advantages of a CANDU reactor include:

- A wide range fuel types (including natural uranium) can be used.
- The units can be refueled on-line.

The disadvantages of a CANDU reactor are:

- Initial high cost of heavy water. The heavy water required must be more than 99.75% pure and tons are required to fill the calandria and the heat transfer system.
- Since heavy water is less efficient at transferring energy from neutrons, the moderator volume is larger in CANDU reactors compared with light-water designs, making a CANDU reactor core generally larger than a light water reactor of the same power output.
- CANDU tends to have higher capital costs compared with other designs.

Chapter 5
High Power Channel Reactor

Reaction: Thermal

Moderator: Graphite

Coolant: Water

The most notable version of a High Power Channel Reactor is a Russian design known as the Reaktor Bolshoy Moschnosti Kanalniy (RBMK) Reactor. RBMKs are water cooled with a graphite moderator. RBMKs are re-fuelable during operation and employ a pressure tube design. They are very unstable, large, and containment buildings for them are expensive and are not typically used. The famed Chernobyl reactor was an RBMK design. After the Chernobyl accident a series of critical safety flaws were identified with the RBMK design and no new RBMK reactors have been put in service since. The main attraction of the RBMK design is the use of light water and un-enriched uranium. Despite safety improvements in the design, RBMK reactors are still considered one of the most dangerous reactor designs in use and only 16 remain in operation around the world.

Overview

By using light water for cooling and graphite for moderation, it is possible to use natural uranium for fuel. Thus, a large power reactor can be built that does not require enriched uranium or heavy water. RBMK designs of up to 1,500 MW have been built.

Light water is both a neutron moderator and a neutron absorber. This means that not only can it slow down neutrons to velocities in equilibrium with surrounding molecules, but it can also absorb some of them outright. Heavy water is also a good neutron moderator, but does not absorb neutrons as easily.

In RBMKs, light water is used as a coolant and moderation is carried out by graphite. With graphite moderating the neutrons, light water has a lesser effect in slowing down the neutrons, but can still absorb them. This means that the reactor's moderation level has to account for the neutrons absorbed by light water.

See Figure 6 for a diagram of a RBMK reactor.

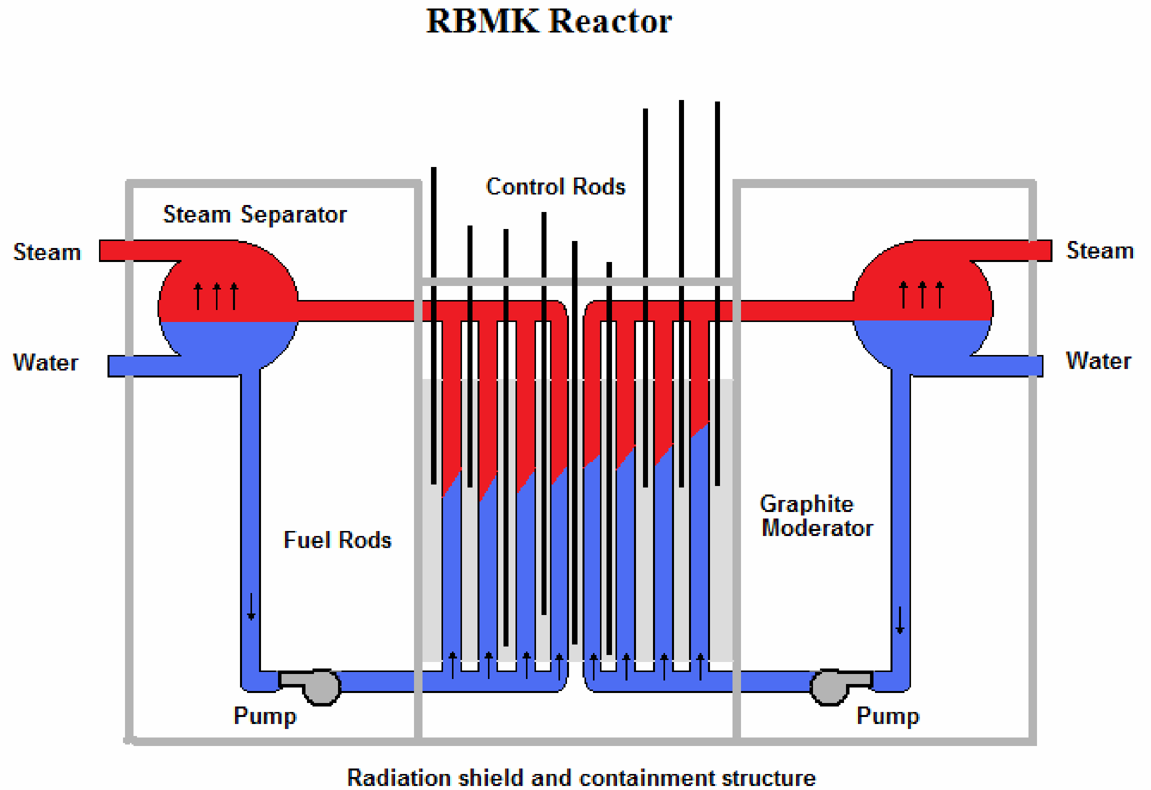


Figure 6

The reactor pit is made of reinforced concrete. It houses the vessel of the reactor, made of a cylindrical steel wall and top and bottom metal plates. The vessel contains the graphite stack and is filled with a helium-nitrogen mixture for providing an inert atmosphere for the graphite and for mediation of heat transfer from the graphite to the coolant channels. The moderator blocks are made of graphite. The blocks are stacked inside the reactor vessel into a cylindrical core. The maximum allowed temperature of the graphite is 730 °C.

The moderator is surrounded by a cylindrical water tank, internally divided into vertical compartments. The water is supplied to the compartments from the bottom and removed from the top; the water can be used for emergency reactor cooling. The tank, a sand layer, and concrete of the reactor pit serve as additional biological shields.

The top of the reactor is covered by the Upper Biological Shield (UBS). It is penetrated by standpipes for fuel and control channel assemblies. The top and bottom are covered with thick steel plates, and additionally joined by structural supports. The space between the plates and pipes is filled with serpentinite, a rock containing significant amount of *bound water*. The structure of the UBS supports the fuel and control channels, the floor above the reactor in the central hall, and the steam-water pipes.

Below the bottom of the reactor core there is the Lower Biological Shield (LBS). It is penetrated by the tubes for the lower ends of the pressure channels and carries the weight of the graphite stack and the coolant inlet piping. A steel structure supports the LBS and transfers the mechanical load to the building.

Bound water is an extremely thin layer of water surrounding mineral surfaces. Water molecules have a strong electrical polarity, meaning that there is a very strong positive charge on one end of the molecule and a strong negative charge on the other. This causes the water molecules to bond to each other and to other charged surfaces, such as soil minerals.

The fuel channels consist of zircalloy pressure tubes led through the channels in the center of the graphite moderator blocks. The top and bottom parts of the tubes are made of stainless steel. The pressure tube is held in the graphite stack channels with two alternating types of split graphite rings; one is in direct contact with the tube and has clearance to the graphite stack, the other one is directly touching the graphite stack and has clearance to the tube; this assembly reduces transfer of mechanical loads caused by neutron-induced swelling, thermal expansion of the blocks, and other factors to the pressure tube, while facilitating heat transfer from the graphite blocks. The tubes are welded to the top and bottom metal plates of the reactor vessel.

About five percent of the core thermal power is in the form of graphite heat and 85% of this heat is removed by the fuel rod coolant channels, via the graphite rings. The rest of the heat is removed by the control rod channel coolant. The gas circulating in the reactor assists with the heat transfer to the coolant channels and plays almost no role in heat removal.

There are 1661 fuel channels and 211 control rod channels in the core of an RBMK reactor design. The fuel assembly is suspended in the fuel channel on a bracket, with a seal plug. The seal plug is designed to facilitate its removal and installation by the remotely controlled refueling machine.

The small clearance between the pressure channel and the graphite block makes the graphite core susceptible to damage. If the pressure channel deforms the deformation or rupture can cause significant pressure loads to the graphite blocks and lead to their damage, and possibly propagate to neighboring channels.

The fuel pellets are made into long barrels of uranium dioxide powder. The material may contain europium-oxide as a burnable nuclear poison to lower the reactivity differences between a new and partially spent fuel assembly. The enrichment level is 2% and the maximum allowable temperature of the fuel pellet is 2,100 °C.


Leningrad RBMK Nuclear Power Plant

The fuel rods are zircalloy tubes. The rods are filled with helium and sealed. Retaining rings help to seat the pellets in the center of the tube and facilitate heat transfer from the pellet to the tube. Each rod contains 3.5 kg of fuel pellets. The maximum allowable temperature of a fuel rod is 600 °C. The fuel assemblies consist of two sets of 18 fuel rods.

The refueling machine is mounted on a gantry crane and remotely controlled. The fuel assemblies can be replaced without shutting down the reactor. When a fuel assembly has to be replaced, the machine is positioned above the fuel channel, mates to it, equalizes pressure within, pulls the rod, and inserts a fresh one. The spent rod is then placed in a cooling pond.

Most of the reactor control rods are inserted from above; a few shortened rods are inserted from below and are used to augment the power distribution control of the core. Additional static boron-based absorbers are inserted into the core when it is loaded with fresh fuel. About 240 absorbers are added during initial core loading. These absorbers are gradually removed with increasing burnup.

The reactor operates in a helium–nitrogen atmosphere. The gas is injected to the stack from the bottom in a low flow rate, and exits from the standpipe of each channel via an individual pipe. The moisture and temperature of the outlet gas is monitored since an increase of them is an indicator of a coolant leak.

The reactor has two independent cooling circuits, each having its own circulating pumps. The cooling water is fed to the reactor through lower water lines to a common pressure header, which is split to distribution headers, each feeding pressure channels through the core, where the feedwater boils. The mixture of steam and water is led by the upper steam lines, one for each pressure channel, from the reactor top to the steam separators, pairs of thick horizontal drums located in side compartments above the reactor top. Steam is taken from the top of the separators by two steam collectors per separator, combined, and led to two generators in the turbine hall, then to condensers, reheated and pumped by the condensate pumps to de-aerators, where remains of gaseous phase and corrosion-inducing gases are removed. The resulting feedwater is led to the steam separators by feedwater pumps and mixed with water from them at their outlets. From the bottom of the steam separators, the feedwater is led by downpipes to the suction headers of the main circulation pumps, and back into the reactor. There is an ion exchange system included in the loop to remove impurities from the feedwater.

The turbine consists of one high-pressure rotor and four low-pressure rotors. Low-pressure separators-preheaters are used to heat steam with fresh steam before being fed to the next stage of the turbine. The uncondensed steam is fed into a condenser, mixed with condensate from the separators, fed by the first-stage condensate pump to a chemical purifier, then by a second-stage condensate pump to four de-aerators where dissolved and entrained gases are removed; de-aerators also serve as storage tanks for feedwater. From the de-aerators the water is pumped through filters and into the bottom parts of the steam separator drums.

The nominal temperature of the cooling water at the inlet of the reactor is about 270 °C and the outlet temperature 284 °C, at a pressure of 1,000 psi. The pressure and the inlet temperature determine the height at which the boiling begins in the reactor; if the coolant temperature is not sufficiently below its boiling point at the system pressure, the boiling starts at the very bottom part of the reactor instead of its higher parts, which makes the reactor very sensitive to the feedwater temperature. If the coolant temperature is too close to its boiling point, cavitation can occur in the pumps and their operation can become erratic or even stop entirely. The feedwater temperature is dependent on the steam production; the steam phase portion is led to the turbines and condensers and returns significantly cooler than the water returning directly from the steam separator. At low reactor power, therefore, the inlet temperature may become dangerously high. The water is kept below the saturation temperature to prevent film boiling and the associated drop in heat transfer rate.

The reactor is tripped in case of too high or low water level in the steam separators, high steam pressure, low feedwater flow, or loss of two main coolant pumps on either side.

The level of water in the steam separators, the percentage of steam in the reactor pressure tubes, the level at which the water begins to boil in the reactor core, the neutron flux and power distribution in the reactor, and the feedwater flow through the core have to be carefully controlled. The level of water in the steam separator is mainly controlled by the feedwater supply, with the de-aerator tanks serving as a water reservoir.

Safety Systems

The RBMK design has several types of safety systems that are used for normal operation and emergency situations. In-core feedback sensors monitor the amount of reactivity during operation; if they detect an increase in reactivity they can automatically insert control rods to reduce power, if they detect a decrease in power they raise controls rods to increase power. If the sensors detect a sharp increase in output they can insert all boron control rods to stop the reaction altogether. There is also a Reactor Protection System. This system is automatically activated when needed or can be manually activated by the operators. RBMK reactors also have a radiation monitoring station that monitors radiation from the plant and the nearby environment. Shielding is provided to absorb radiation produced under both normal operation and emergency situations.

The reactor is equipped with an *emergency core cooling system* (ECCS), consisting of dedicated water reserve tank, hydraulic accumulators, and pumps. ECCS piping is integrated with the normal reactor cooling system. In case of total loss of power, the ECCS pumps are powered by the rotational momentum of the generator rotor until the diesel generators come online. The ECCS has three systems, connected to the coolant system headers. In case of damage, the first ECCS subsystem provides cooling for up to 100 seconds to the damaged half of the coolant circuit, and the other two subsystems then handle long-term cooling of the reactor.

The short-term ECCS subsystem consists of two groups of accumulator tanks, containing water blanketed with nitrogen under pressure, connected by fast-acting valves to the reactor. Each group can supply 50% of the maximum coolant flow to the damaged half of the reactor. The third group is a set of electrical pumps drawing water from the de-aerators. The short-term pumps can be powered by the spin-down of the main generators.

ECCS for long-term cooling of the damaged circuit consists of three pairs of electrical pumps, drawing water from the pressure suppression pools; the water is cooled by the plant service water by means of heat exchangers in the suction lines. Each pair is able to supply half of the maximum coolant flow. ECCS for long-term cooling of the intact circuit consists of three separate pumps drawing water from the condensate storage tanks, each able to supply half of the maximum flow.

Initially, the RBMK design focused solely on accident prevention and mitigation, not on containment of severe accidents. The RBMK design includes a partial containment structure (not a full containment building) for dealing with emergencies. The pipes underneath the reactor are sealed inside leak-tight boxes containing a large amount of water. If these pipes leak or burst, the radioactive material is trapped by the water inside these boxes. However, RBMK reactors were designed to allow fuel rods to be changed without shutting down. This requires large cranes above the core. Because the RBMK reactor is very tall, the cost and difficulty of building a heavy containment structure prevented building of additional emergency containment structure for pipes on top of the reactor.

Advantages and Disadvantages

The following is a list of the advantages and disadvantages of a RBMK reactor.

The advantages of a RBMK reactor include:

- The units can be refueled without shutting down.
- The units can use un-enriched uranium.

The disadvantages of a RBMK reactor include:

- The design has the potential for a runaway reaction.
- The lack of a containment vessel.

Chapter 6
Gas Cooled Reactors

Reaction: Thermal

Moderator: Graphite

Coolant: CO_2 Gas

Gas Cooler Reactors (GCR) are graphite moderated and CO_2 cooled. They have a high thermal efficiency compared with PWRs due to higher operating temperatures. This is a thermal neutron reactor design.

The GCR is a type of nuclear power reactor which was designed and is still in use in the United Kingdom. GCR reactors are also called *Magnox* reactors because the Magnox alloy is used to clad the fuel rods inside the reactor. There are approximately 18 GCR reactors in use around the world, primarily in Great Britain.

Magnox reactors are pressurized, carbon dioxide cooled, graphite moderated reactors using un-enriched uranium as fuel and Magnox alloy as fuel cladding. Control rods are Boron-steel. Early reactors have steel pressure vessels, while later units are made from pre-stressed concrete; some are cylindrical in design, but most are spherical.

Refueling during operation was considered to be an economically essential part of the design for the plants, to maximize reactor availability by eliminating refueling downtime. This is particularly important for GCR's as the un-enriched fuel has a low burnup, requiring more frequent changes of fuel than enriched uranium reactors. However the complicated refueling equipment has proven to be unreliable.

Magnox is an alloy - mainly of magnesium with small amounts of aluminum and other metals - used in cladding un-enriched uranium metal fuel with a non-oxidizing covering to contain fission products. Magnox is short for <u>Mag</u>nesium <u>n</u>on-<u>ox</u>idizing. This material has the advantage of a low neutron capture, but has two major disadvantages:

- It limits the maximum temperature, and hence the thermal efficiency, of the plant.
- It reacts with water, preventing long-term storage of spent fuel under water.

Magnox fuel incorporates cooling fins to provide maximum heat transfer despite low operating temperatures, making it expensive to produce. While the use of uranium metal rather than oxide makes reprocessing more straightforward and cheaper, the need to reprocess fuel a short time after removal from the reactor means that the fission product hazard can be significant. Expensive remote handling facilities are required to address this danger.

The GCR form of nuclear reactor has been superseded by a new design called the Advanced Gas-Cooled reactor or AGR.

The AGR was developed from the Magnox reactor, operating at a higher gas temperature for improved thermal efficiency, requiring stainless steel fuel cladding to withstand the higher temperature. Because the stainless steel fuel cladding has a higher neutron capture than Magnox fuel cans, enriched uranium fuel is needed, with the benefit of higher "burn ups" requiring less frequent refueling.

See Figure 7 for a schematic of an Advanced Gas-Cooled Reactor.

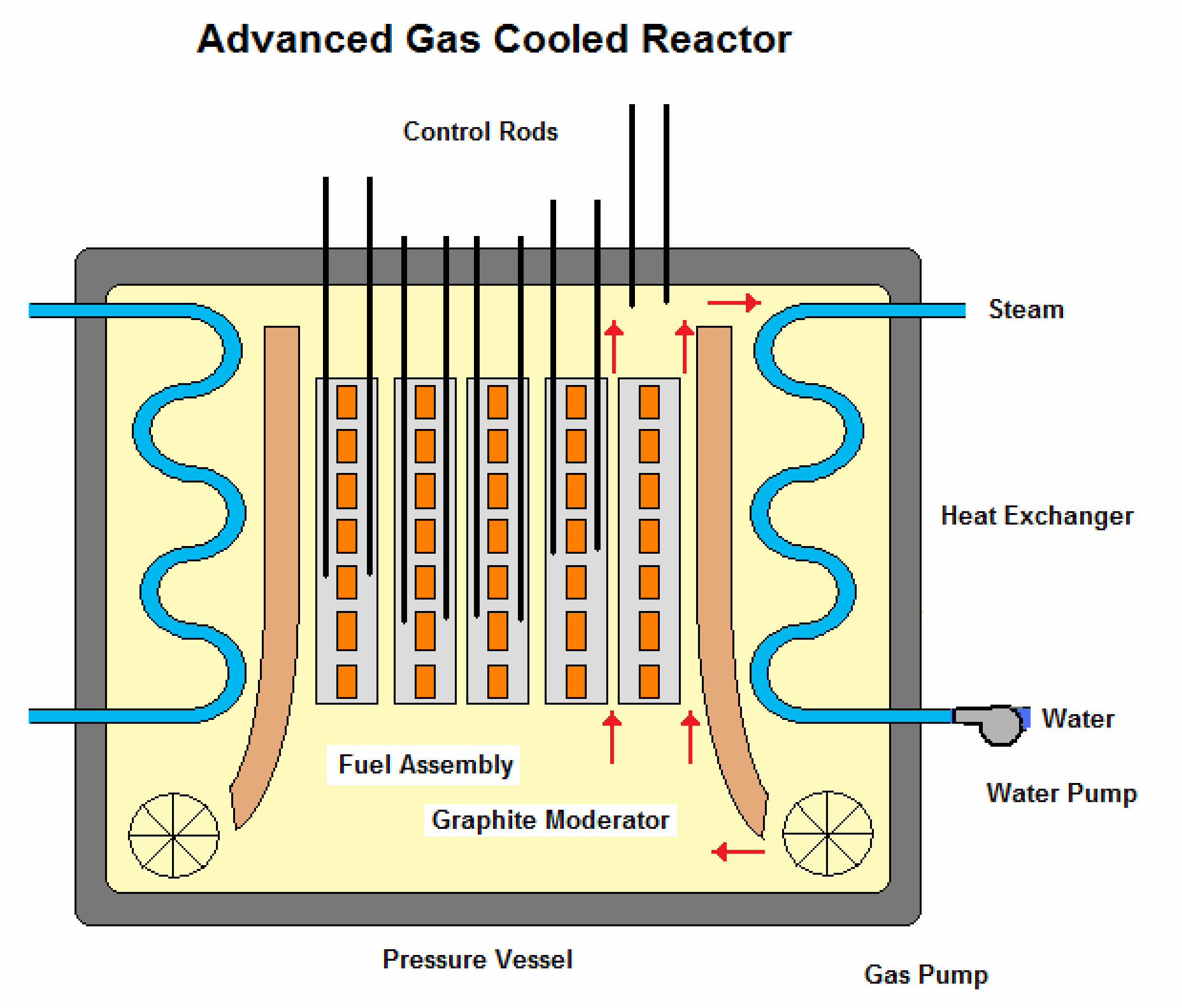


Figure 7

The fuel is uranium dioxide pellets, enriched to 2.5-3.5%, in stainless steel tubes. The carbon dioxide coolant circulates through the core, reaching 640°C and a pressure of around 580 psi, and then passes through the steam generator outside the core but still within the steel lined, reinforced concrete pressure vessel. Control rods penetrate the graphite moderator and a secondary system involves injecting nitrogen into the coolant to hold the reactor down.

The AGR was designed to have a high thermal efficiency of about 41%, which is better than modern pressurized water reactors which have a typical thermal efficiency of 34%. This is due to the higher coolant outlet temperature practical with gas cooling, compared to about 325 °C for

PWRs. However the reactor core has to be larger for the same power output, and the fuel burnup ratio at discharge is lower so the fuel is used less efficiently, countering the thermal efficiency advantage.

AGRs are designed to be refueled without being shut down first. Though better than GCR's, operational problems with on-line refueling has limited refueling to only part load or when shut down. The AGR design proved to be overly complex and difficult to construct on site.

All AGR power stations currently in operation are configured with two reactors in a single building. Each reactor has a design thermal power output of 1,500 MW.

Advantages and Disadvantages

The following is a list of the advantages and disadvantages of an AGR reactor.

The advantages of an AGR reactor include:

- High thermal efficiency
- Limited ability for on-line refueling

The disadvantages of an AGR reactor are:

- Complex and difficult to construct
- Require frequent re-fueling

Chapter 7
Fast Breeder Reactor

Reaction: Fast Neutron

Moderator: None

Coolant: Liquid Metal

The fast breeder reactor is a fast neutron reactor designed to breed fuel by producing more fissile material than it consumes. All large-scale, commercial breeder reactors have been liquid metal fast breeder reactors (LMFBR) cooled by liquid sodium. At present, there are only a couple of LMFBR reactors in operation.

A LMFBR is cooled by liquid metal, totally un-moderated, and produces more fuel than it consumes. They are said to "breed" fuel, because they produce fissionable fuel during operation because of neutron capture. These reactors can function much like a PWR in terms of efficiency, and do not require much high pressure containment, as the liquid metal does not need to be kept at high pressure, even at very high temperatures. These reactors are fast neutron, not thermal neutron designs.

The LMFBR reactor design can use either lead or sodium as the liquid metal. Using lead as the liquid metal provides excellent radiation shielding, and allows for operation at very high temperatures. Also, lead is mostly transparent to neutrons, so fewer neutrons are lost in the coolant, and the coolant does not become radioactive. Unlike sodium, lead is mostly inert, so there is less risk of explosion or accident, but such large quantities of lead may be problematic from toxicology and disposal points of view. Most LMFBRs are sodium cooled. The sodium is relatively easy to obtain and work with, and it also manages to actually prevent corrosion on the various reactor parts immersed in it. However, sodium explodes violently when exposed to water.

Sodium LMFBR's are either Loop type or Pool type designs.

With a Loop type LMFBR, the primary coolant is circulated through primary heat exchangers external to the reactor tank. Figure 8 is a schematic diagram a Loop type LMFBR.

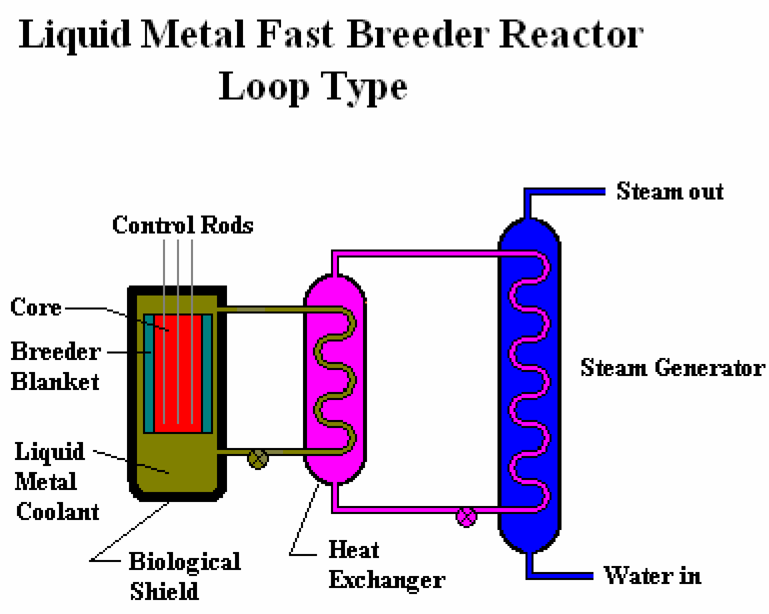


Figure 8

The loop type LMFBR is much smaller than a pool type unit. It can be built in a factory and transported to the power plant site, which reduces construction costs. The loop type has more piping and hence a greater opportunity for loss of containment leaks.

With a Pool type LMFBR the primary heat exchangers and circulators are immersed in the reactor tank. Figure 9 is a schematic of a Pool type of LMFBR.

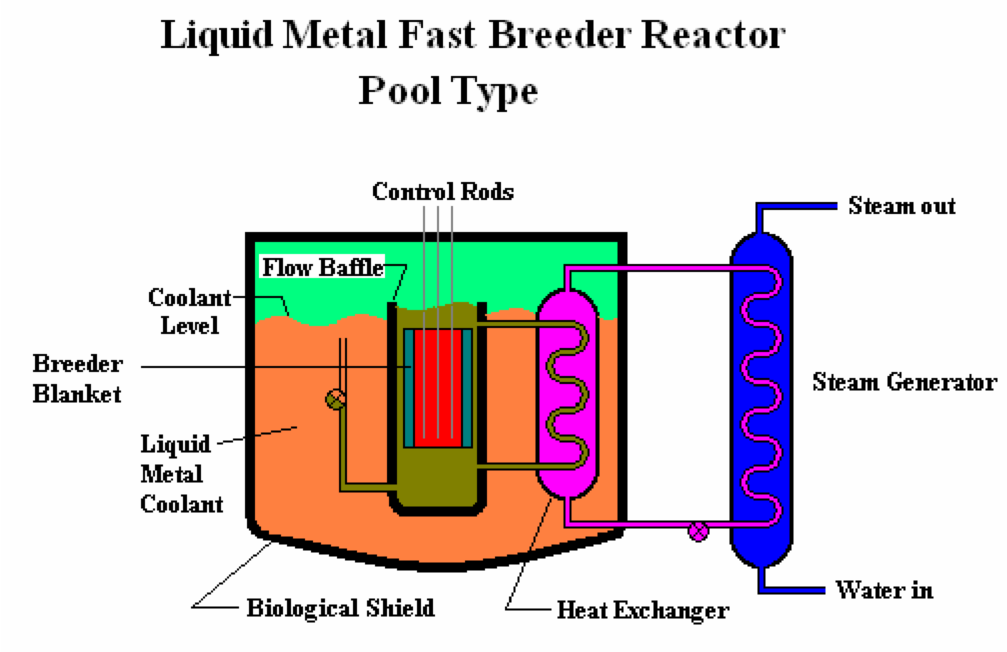


Figure 9

Pool type LMFBR's are simple designs with minimal piping. However, the reactor vessel is large and – unlike the loop type – must be fabricated on-site.

LMFBR's usually use a mixed oxide fuel core of up to 20% plutonium dioxide (PuO_2) and at least 80% uranium dioxide (UO_2). Another fuel option is metal alloys, typically a blend of uranium, plutonium, and zirconium. The plutonium used can be supplied by reprocessing reactor outputs or from dismantled nuclear weapons.

With either design, the reactor core is composed of two parts: core and blanket. Fission takes place in the core and extra neutrons diffusing out from the core are absorbed in a radial blanket surrounding the core. There is also a vertical blanket incorporated into the top and bottom of each fuel rod.

Most LMFBR designs, the reactor core is surrounded in a blanket of tubes containing non-fissile uranium-238 which, by capturing fast neutrons from the reaction in the core, is partially converted to fissile plutonium-239, which can then be reprocessed for use as nuclear fuel.

Since a fast reactor uses a fast spectrum, no moderator is required to thermalize the fast neutrons.

All current fast reactor designs use liquid metal as the primary coolant, to transfer heat from the core to steam used to power the electricity generating turbines.

Water is an undesirable primary coolant for fast reactors because large amounts of water in the core are required to cool the reactor. Since water is a neutron moderator, this slows neutrons to thermal levels and prevents the breeding of uranium-238 into plutonium-239.

Fission of the nuclear fuel in any reactor produces neutron-absorbing fission products, and because of this it is necessary to reprocess the fuel and breeder blanket from a breeder reactor if one is to fully utilize its ability to breed more fuel than it consumes. The most common reprocessing technique is considered a proliferation concern because such reprocessing technologies can be used to extract weapons grade plutonium from a reactor operated on a short re-fuelling cycle.

Breeders can be designed to utilize thorium, which is more abundant than uranium. Currently, there is renewed interest in breeders because they would consume less natural uranium (less than 3% compared to conventional light-water reactors), and generate less waste, for equal amounts of energy, by converting non-fissile isotopes of uranium into nuclear fuel.

A fast breeder reactor can convert Uranium-238 into Plutonium-239 at a rate faster than it consumes its fuel. By repeated recycling of the fuel, it should be possible to exploit 50% of the fuel value of the uranium fuel supply. Therefore, breeders could theoretically increase the energy output of uranium up to 25-times.

Production of fissile material in a reactor occurs by neutron irradiation of fertile material, particularly uranium-238 and thorium-232. In a breeder reactor, these materials are deliberately provided both in the fuel and in a breeder blanket surrounding the core. Production of fissile material takes place to some extent in the fuel of all current commercial nuclear power reactors. Towards the end of its life, a uranium PWR fuel element is producing more power from the fissioning of plutonium than from the remaining uranium-235. Historically, in order to be called a breeder, a reactor must be specifically designed to create more fissile material than it consumes.

One measure of a reactor's performance is the *breeding ratio.* Breeding ratios have ranged from 1.01 for the reactors running on thorium fuel and cooled by conventional light water to over 1.2 for liquid-metal-cooled reactors. Theoretical models of breeders with liquid sodium coolant flowing through tubes inside fuel elements show breeding ratios with an upper limit of 1.8 are possible.

Breeding Ratio is the average number of fissile atoms created per fission event.

In normal operation, most large commercial reactors experience some degree of fuel breeding. It is customary to refer only to machines optimized for this trait as true breeders, but industry trends are pushing breeding ratios steadily higher, thus blurring the distinction. Legacy conventional reactors have breeding ratios of around 0.55, but the next generation of conventional reactors may achieve breeding ratios of up to 0.80.

Advantages and Disadvantages

The advantages of a LMFBR reactor are:

- Produces fuel in the reaction.
- Creates less waste than other types of reactors.

- Can use Thorium as a fuel source.

The disadvantages of a LMFBR include:

- There is a concern about the potential of nuclear proliferation.
- LMFBRs require liquid sodium for cooling.
- The units are expensive to build and complicated to operate.

Summary

Proponents of nuclear energy contend that nuclear power is a sustainable energy source that does not create air pollution, reduces carbon emissions and increases energy security by decreasing dependence on foreign oil. The operational safety record of nuclear plants in the Western world is far better when compared to the other major types of power plants. With the exception of Chernobyl, no radiation-related fatalities ever occurred at any commercial nuclear power plant. Optimists point out that the volume of radioactive waste is very small, and claim it can be stored safely deep underground. Future designs of reactors are promised to eliminate almost all waste.

Critics believe that nuclear power is a potentially dangerous energy source, with decreasing proportion of nuclear energy in production. They claim that radioactive waste cannot be stored safely for long periods of time, that there is a continuing possibility of radioactive contamination by accident or sabotage, and that exporting nuclear technology to other countries might lead to the proliferation of nuclear weapons. The recent slow rate of growth of installed nuclear capacity is said to indicate that nuclear reactors cannot be built fast enough to slow down climate change.

This volume in the series on the nuclear power industry has reviewed the types of plants currently in operation. The final volume in this series, Volume III, looks at the future of nuclear power and the types of plants we are likely to see in the first half of the 21st Century.

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