U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Reactor Concepts (R-100)

USNRC HRTD Training Course

HRTD
Human Resources
Training & Development

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Nuclear Power for Electrical Generation

Chapter Overview

The purpose of a nuclear power plant is to generate electricity from steam created by nuclear heat. It should not be surprising, then, that a nuclear power plant has many similarities to other electrical generating facilities using steam. Conversely, nuclear power plants have some significant differences from other electrical generation plants.

Throughout this chapter, we will examine: how to generate electricity using steam energy, the most common ways for producing commercial electricity in the United States (U.S.), and the significant ways that nuclear power plants differ from other power plants. We will also explore the differences and similarities between the two types of commercial nuclear plants in the U.S (Pressurized Water Reactors (PWR) and Boiling Water Reactors (BWR)).

Objectives

After completing this chapter, you will be able to:

- Describe the role of nuclear power in generating commercial electrical power in the U.S.
- Describe the process for generating electrical power using steam.
- Identify the most common sources and costs of commercial electrical power generation in the U.S.
- Identify characteristics unique to nuclear power generation.
- List the barriers to the escape of fission products from the fuel.
- Identify the types of nuclear power plants used for commercial electrical power generation in the U.S.

Estimated time to complete this chapter:

35 minutes

Basic Electrical Generation Using Steam

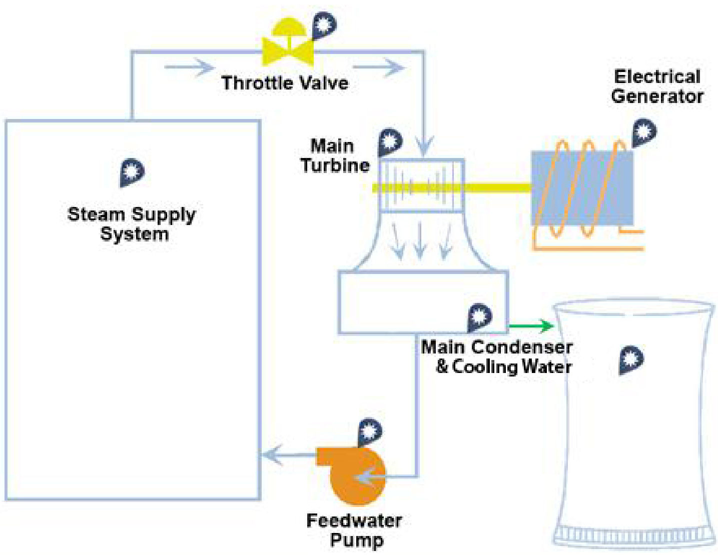
Commercial Generation of Electricity

Commercial generation of electricity requires large-scale production of electricity in an economical and cost-effective manner. As both a primary consumer and producer of electricity, the U.S. has multiple options and methods for producing electricity. By far, the most prevalent is a steam electric power plant.

Components of a Steam Power Plant

First, let's consider the basics of a **steam** power plant. All steam power plants, regardless of fuel type, have the same basic functionality and use many of the same components.

Select the components of a basic steam power plant for an explanation of its function or role:



Steam Supply System

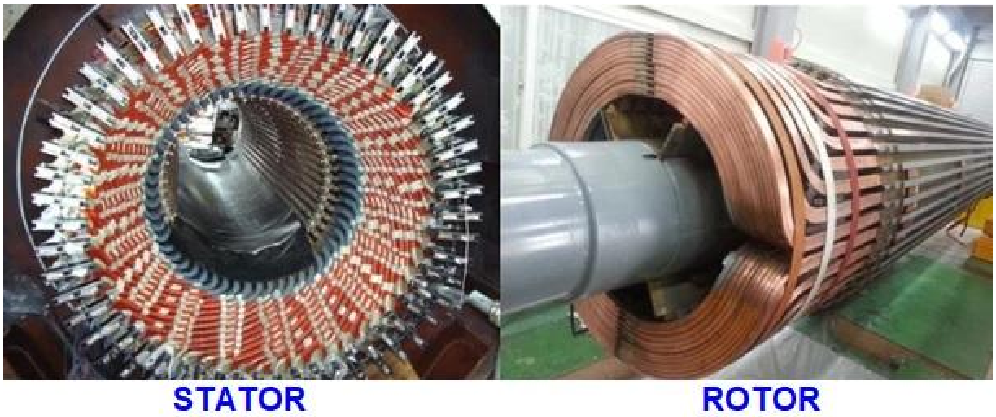
The steam supply system includes the boiler. The boiler provides the heat that produces the steam. All steam electric power plants have a boiler, regardless of the nature of the fuel used to generate the steam.

Electrical Generator

Steam electric power plants can use a number of fuels to create the steam, but all require an electrical generator to produce the electricity. In most large-scale electrical generators, a magnet (rotor) revolves inside a stationary coil of wire (stator). The magnetic lines of flux ‘cut,’ or flow, through the stator and create a flow of electrons inside the wire. This flow of electrons is electricity.

Some mechanical device (wind turbine, water turbine, steam turbine, diesel engine, etc.) must be available to provide the force for turning the rotor.

Here are examples of a rotor and a stator used in commercial power plants.

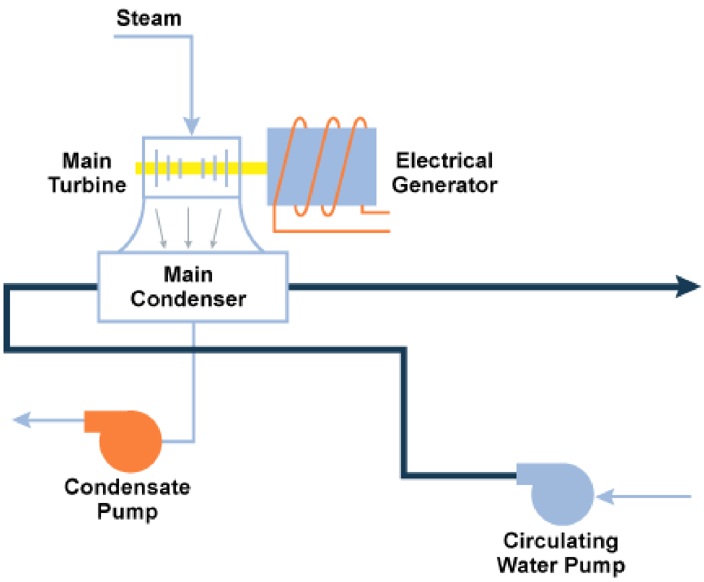


Cooling with Circulating Water

To operate properly, steam plants need a circulating water system to remove excess heat from the steam system. The steam exiting the turbines enters the condenser and its heat is transferred to the environment.

The circulating water system pumps water through thousands of metal tubes in the plant’s condensers. Steam exiting a plant’s turbine is cooled and condensed back into water as it comes into contact with the much-cooler tubes. Since the tubes provide a barrier between the steam and the environment, there is no physical contact between a plant’s steam and the cooling water.

The condenser is maintained at a vacuum since that will increase the amount of energy that the turbine can extract from the steam. Because a condenser operates at a vacuum, any tube leakage in this system will produce an inflow of water into the condenser rather than an outflow of water to the environment.



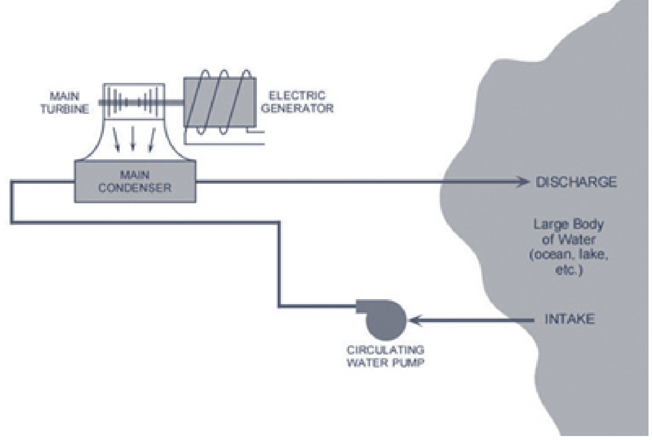
Methods of Cooling with Circulating Water

There are three methods for cooling with circulating water: Cooling using a body of water, Cooling using a forced draft tower, and Cooling using a natural convection tower.

Body of Water

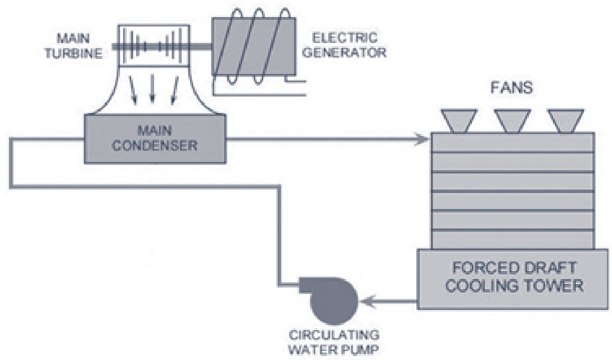
Power plants located on large bodies of water often discharge their circulating water directly back to the source under strict environmental protection regulations. The discharge water temperature and chemical composition are regulated by the states. The expected temperature increase from circulating water inlet to outlet is about 5 to 10 degrees Fahrenheit.

In today's environment, it is likely that no new power plants will utilize this direct method of circulating water back due to the concerns with thermal pollution.



Forced Draft Tower

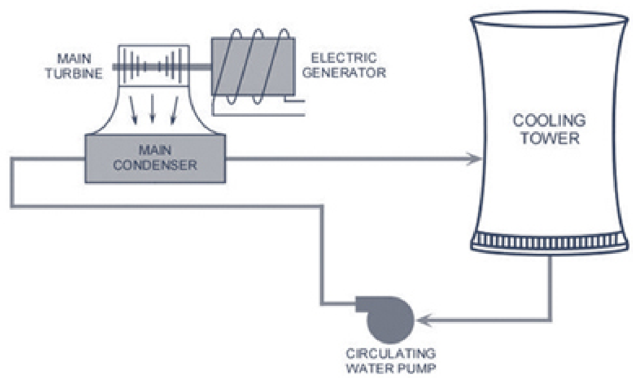
With a forced draft cooling tower, the circulating water is pumped into the top of the tower after passing through the condenser and allowed to splash downward transferring some of its heat to the air. Several large electrical fans, located at the top of the cooling tower, provide forced air circulation for more efficient cooling.



Natural Convection Tower

The tall, hourglass-shaped, natural convection cooling towers do not require fans to transfer the excess heat from the circulating water system into the air. Rather, the natural tendency of hot air to rise removes the excess heat as the circulating water splashes down inside the cooling tower. These towers are typically several hundred feet tall. The "steam" vented from the top of a cooling tower is really lukewarm water vapor.

As the warm, wet air from inside the cooling tower contacts the cooler, dryer air above the cooling tower, the water vapor that cannot be held by the cooler air forms a visible cloud. This is because the colder the air is, the lower its ability to hold water. The released cloud of vapor will only be visible until it is dispersed and absorbed by the air. However, that released water vapor represents a significant 'loss' of volume from the circulating water system. Plants need to make up tens of thousands of gallons of water per hour to accommodate these losses.



Types of Power Plants

Types of Power Plants Used for Large-Scale Power Production in the U.S.

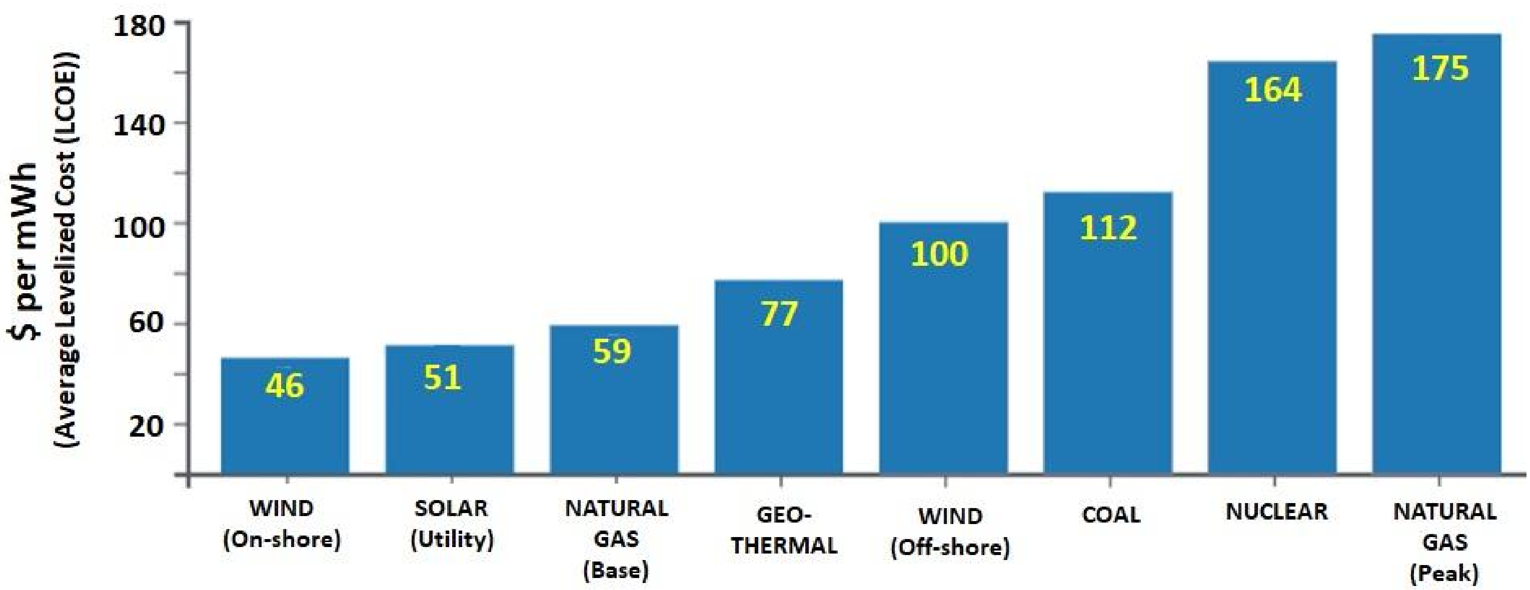
Recall that most of the electrical power in the U.S. is created from steam electric power plants. There are numerous types of steam electric plants, including:

- Coal
- Oil
- Natural gas
- Petroleum
- Nuclear

In addition, a number of alternative sources are being explored for large-scale adoption.

When determining relative costs of power production, a new method known as Levelized Cost Of Energy (LCOE) is used.

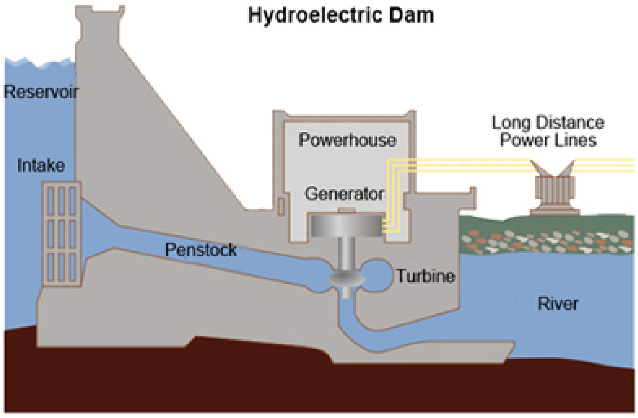
Using LCOE, the cost of electricity production is basically the total generation cost (construction, fuel, maintenance) over the life of the facility versus the amount generated. In the past, such methods as hydroelectric, coal, and nuclear were considered very low cost. However, using LCOE methods, this is no longer true. Factors such as inevitable costs of waste disposal and decommissioning have made the levelized cost of nuclear-generated electricity significantly higher than previously calculated.



Let’s take a look at the various types of power production facilities in the U.S. We’ll start with hydroelectric.

Hydroelectric Power Plants

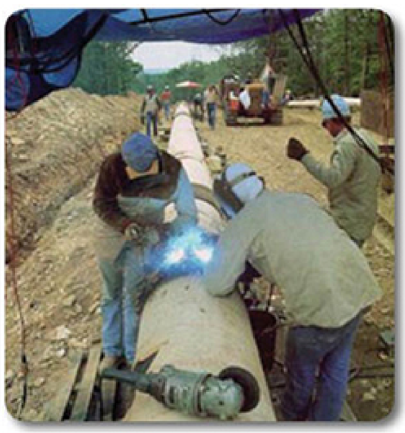
In a hydroelectric power plant, water flows from a higher level to a lower level. The natural, downward flow of the water turns the blades of a water turbine, which causes the rotor of the attached electrical generator to spin and produce electricity. Hydroelectric power has a production cost of approximately $0.005 per kilowatt hour (kwh) (current data through 2017). Basically, once the cost of building the plant is expended, the cost of producing the electricity is incredibly small. However, due to environmental concerns, hydroelectric power generation in the US has dwindled to about 7% of the total 4.1 trillion megawatt hours.



Fossil Fuel Power Plants

Fossil fuel power plants are steam electric power plants. A fossil-fueled power plant burns coal, natural gas, or oil to supply the necessary steam. The **approximate** cost per kwh for coal is $0.12 and gas is $0.06 to $0.18. These costs are highly variable due to the volatility of the fuel cost. (data through 2017). The LCOE of fossil plant electricity has been substantially higher over recent years due to increased costs associated with the environmental costs of carbon reduction and other requirements.

Electric costs can be misleading based on various government subsidies that exist.


Nuclear Power Plants

In a nuclear power plant, many of the components are similar to those in a fossil-fueled plant, except that the steam boiler is replaced by a Nuclear Steam Supply System (NSSS). The NSSS consists of a nuclear reactor and all of the components necessary in supporting the reactor's production of high-pressure steam, which will be used to turn the turbine for the electrical generator. The nuclear core is simply a heat source harnessed to boil water for spinning a steam turbine.

Nuclear plants generally have very high capital costs to build and maintain, but relatively low fuel costs. However, the LCOE for electricity generated through from a nuclear power plant has increased from approximately $0.02 (data from 2009) to a relatively high $0.16 (2017 data).

Alternative Sources

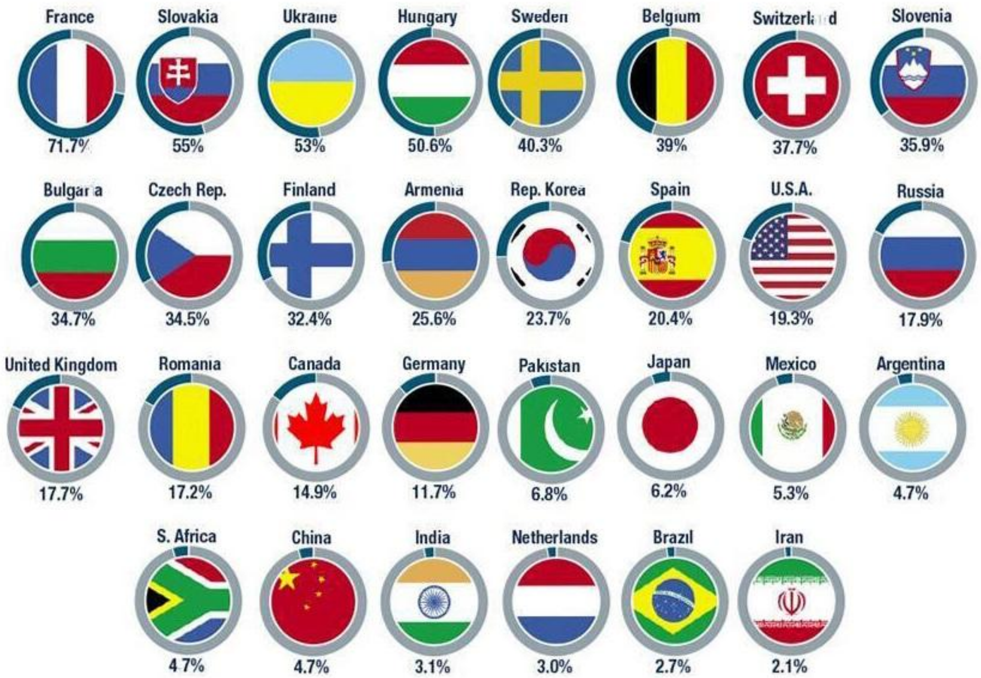
With the increase in environmental concerns, the volatility in the cost of fossil fuels, and concerns regarding dependence on foreign fuel sources, interest is growing in alternative sources for large-scale electrical power generation. The most common and popular include solar and wind farms, at this time, with research into bio-fuels as an alternative for fossil fuel steam plants in its infancy. However, the comparative costs per kwh for solar and wind farms are not competitive with fossil fuels. Solar power costs per kwh are approximately $0.50, while power from wind farms costs approximately $0.05 to $0.08 per kwh (2009 data). The chief concern with these alternative sources is bringing cost down while increasing capacity (how much electricity is generated).

Net Generation by Source

Commercial nuclear power plants continue to generate approximately 20% of the electricity produced in the U.S. The total generation in the US is approximately 4,200 thousand gigawatt-hours.

4.2 TRILLION W-hrs!

For comparison purposes, nuclear generation accounts for the following of the total electrical production in that particular country as shown by the below graphic:



The electricity produced in the U.S. from nuclear power is equivalent to 31% of the world's total nuclear-generated electrical power. This compares with 16% for France, 13% for Japan, 7% for Germany, 5% for Russia, and 4% for South Korea and the United Kingdom.

So, even though only 20% of our power comes from nuclear power, that number represents the largest portion of the global total of power generation by commercial nuclear plants. China continues to build a significant fleet of new nuclear plants as their concern for fossil fuel pollution grows.

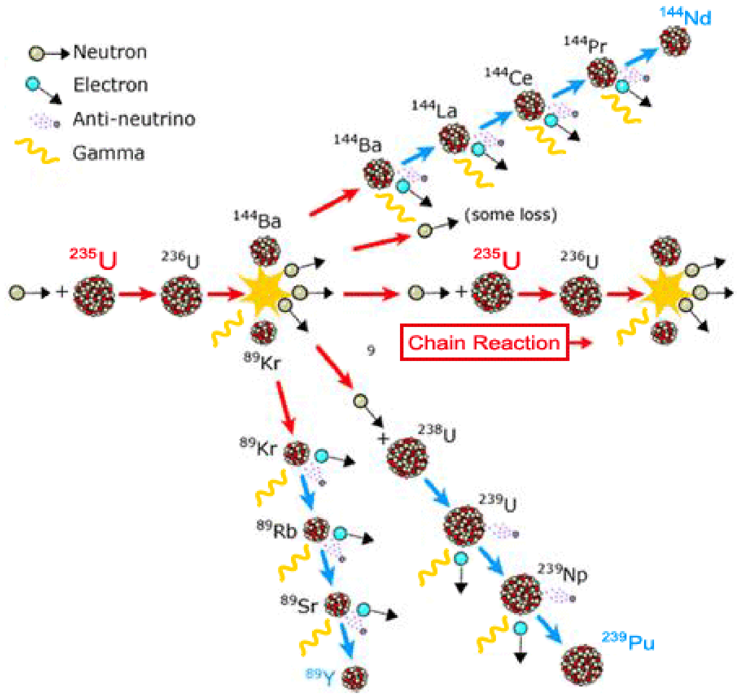
Characteristics Unique to Nuclear Power Generation

Chain Reaction Creates Heat to Create Steam

Like a fossil-fueled plant, a nuclear power plant is a steam power plant and boils water to produce electricity. Unlike a fossil-fueled plant, a nuclear plant's energy does not come from the burning of fuel but from the fission (splitting) of fuel atoms. The bulk of the energy is from the kinetic energy (motion) of the fission products released from the uranium nucleus after it splits.

In short, every time an atom splits in the core, it releases heat energy. That heat energy is used to generate the steam. By creating a chain reaction in the reactor core, heat energy can be controlled, consistent and constant. As a result, the steam generation—and by extension, the generation of electricity—is predictable and controllable.

Chapter 2: Fission Process & Heat Production looks at fission and heat in more detail.



Nuclear Fuel

The most common fuel for the production of electricity by reactor plants in the U.S. is Uranium. The level of enrichment (concentration of U-235) in U.S. power reactors is significantly lower than in Naval reactors and nuclear weapons. Commercial U.S. nuclear power reactors will **not** explode like a nuclear bomb. The NRC regulations have a maximum amount of enrichment in commercial fuel of 5%.

Uranium-235 and Uranium-238

Mined uranium ore contains a very low percentage of the desired isotope (U-235), approximately 0.7% by weight. U-235 is more desirable for fuel because it is easier to cause the U-235 atoms to fission (split) than the much more naturally abundant (99.2%) U-238 atoms.

Uranium Processing

After mining, the ore goes to a uranium recovery facility that increases the percentage of uranium in the ore concentrate, known as 'yellowcake.' This is where NRC's regulatory oversight begins.

The yellowcake is shipped in drums to the only conversion facility in the US near Metropolis, IL. At the conversion facility, the yellowcake is transformed into liquified UF6 (uranium hexafluoride) and shipped to enrichment facilities. At the enrichment facility, the UF6, in gas form, is fed into gas centrifuges and the amount of U-235 is increased (enriched) to up to 5% (NRC regulatory limit) for commercial US light water reactor fuel.

Fuel Pellets

Once the uranium is enriched, it is further processed into fuel pellets. The fuel pellets are then heated to very high temperatures to achieve ceramic properties. All fuel pellets are consistent in size and shape. The consistency allows for greater control over a reactor's processes.

Finally, the fuel pellets are inspected to guarantee their consistency in size, shape, and enrichment.

Reactor Fuel Assemblies and Control Rods

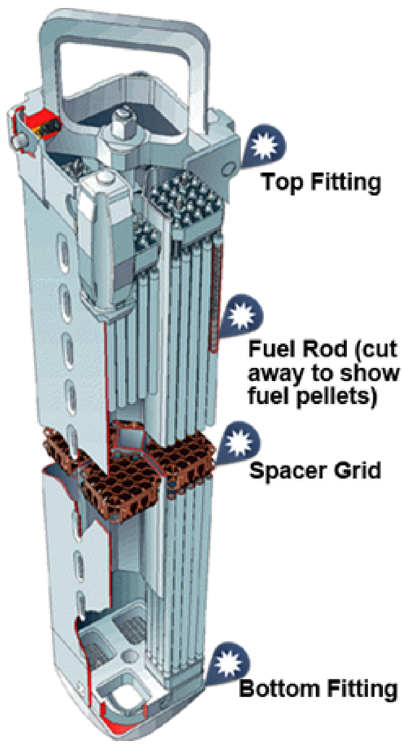
There are slight differences between boiling water (BWR) and pressurized water (PWR) reactor fuel assemblies, but both types of commercial reactor fuel assemblies consist of the same major components. These major components are the fuel rods, the spacer grids, and the upper and lower end fittings.

In all cases, the reactor coolant (water) flows through the reactor fuel assemblies, on the outside of the fuel rods, from bottom to top. As the coolant removes the heat from the fuel, the coolant grows progressively hotter from the bottom to the top of the core.

All reactors also have control rods. Control rods are used to control the chain reactions inside the reactor by absorbing neutrons that normally are absorbed in the fuel to produce fission. Since the control rods absorb neutrons and reduce the chain reaction of fission, they are termed 'poisons.'

A pressurized water reactor (PWR) inserts the control rod 'fingers' within the lattice of the fuel assemblies, while a boiling water reactor (BWR) intersperses the controls rods between the fuel assemblies (outside the lattice). The graphic below is of a typical BWR fuel assembly, but PWR assemblies do not vary appreciably.

Select the component of the fuel assembly for an explanation of its function or role:



Upper and Lower Fittings

The upper and lower end fittings serve as the upper and lower structural elements of the assemblies. The lower fitting (or bottom nozzle) directs the coolant flow to the assembly through several small holes machined into the fitting. There are also holes drilled in the upper fitting to allow the coolant flow to exit the fuel assembly. The upper end fitting also has a connecting point to which the refueling equipment attaches for moving the fuel assembly with a crane.

Barriers to Prevent the Escape of Fission Products

Earlier, we mentioned that the fission processes were a unique characteristic of nuclear reactors. One concern with the fission process is the potential for fission products, which are radioactive and can be extremely long-lived, to escape outside of the fuel assemblies and reactor, and into the environment. To prevent this, there are three barriers to fission product escape. The three barriers that fission products would encounter are:

1. **Fuel Rod Cladding** - Cladding refers to the zirconium alloy (stainless steel) tubes that house the fuel pellets. The zirconium cladding retains most of the fission products. The cladding helps contain the fuel and fission products within the fuel assembly and is highly resistant to corrosion and capable of withstanding extreme heat (usually maintained less than 2000 degrees-F).
2. **Reactor Coolant System (RCS)** - The RCS includes the reactor vessel and attached piping that contains the coolant, usually highly purified water, which flows through the core. The coolant transfers the core’s fuel heat to the feedwater (in a PWR) to produce steam for the turbine. For BWRs, the feedwater IS the reactor coolant. If fission products escape through the fuel rod cladding, they are generally contained within the RCS. U.S. commercial reactors utilize light water, where the bulk of the hydrogen atoms in the water have no neutrons in their nucleus. This is not true for heavy water reactors, which use water with much larger concentrations of hydrogen with a neutron in the nucleus. This isotope of hydrogen is known as deuterium.
3. **Containment Building** - The containment building is the final barrier to the escape of fission products. Containment buildings are designed to prevent the escape of fission products to the environment and general public, should both the fuel rod cladding and the RCS fail.

Types of Nuclear Power Plants

Two Types of Commercial Nuclear Power Plants in the U.S.

There are two basic types of reactor plants being used in the U.S. to commercially produce electricity: the boiling water reactor (BWR) and the pressurized water reactor (PWR).

Both BWR and PWR plants in the U.S. are designed for optimal safety. In other words, they will tend to shut themselves down in accident conditions. As such, regardless of the type, all commercial U.S. reactors require human intervention to keep the reactor **running**.

In the event of various abnormal occurences, the reactor is designed to be automatically shut down. Nuclear plants are also designed with three barriers that prevent fission products from escaping to the environment, thus limiting potential risk to the surrounding environment and general public.

Most nuclear power plants around the world follow the same basic design as the plants in the U.S. but there are some significant differences, particularly in the former Soviet Union. Chapter 12: The Chernobyl Accident discusses some of those differences.

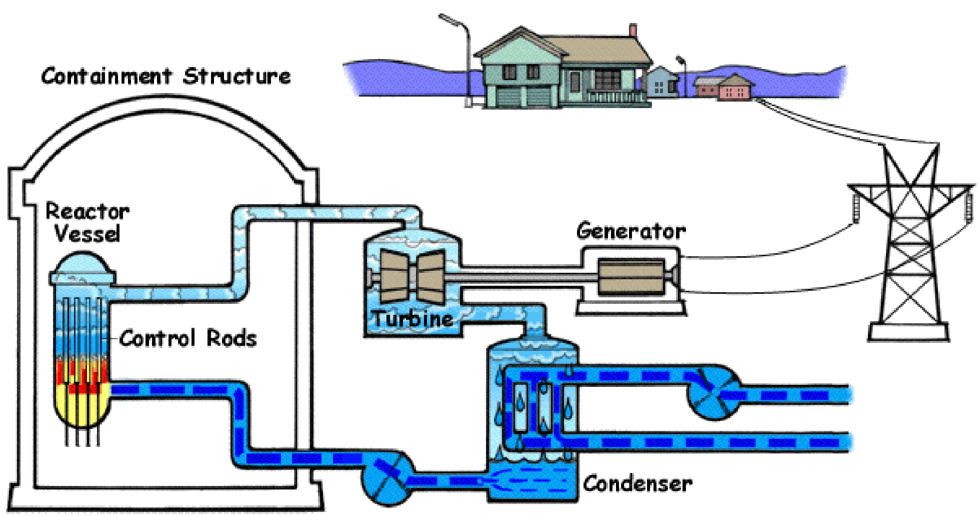
Let's take a look at the basic similarities and differences between a BWR and PWR. We'll start with the BWR.

BWRs

BWRs currently comprise about one-third of the commercial power reactors in the U.S.

Inside the reactor vessel, the reactor pressure is low enough to allow steam formation within the core area. The steam that forms in the core area turns the turbine to generate electricity.

The steam that is used to turn the turbine of a BWR has come in direct contact with the reactor core.



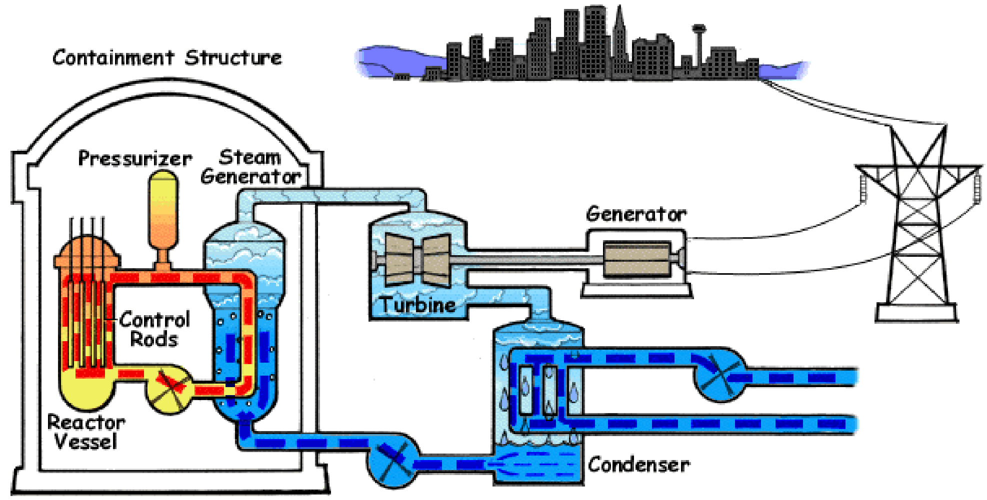
PWRs

PWRs make up about two-thirds of the power reactors in the U.S.

In a PWR, steam is produced in a separate steam generator rather than in the reactor vessel. A pressurizer keeps the coolant water in the reactor vessel under very high pressure, which prevents the coolant from boiling.

The hot coolant flows through tubes in the steam generator, where the heat is transferred to the lower pressure feedwater on the outside of the steam generator tubes. The feedwater turns into steam for turning the turbine.

The steam that turns the turbine of a PWR does not come in direct contact with the reactor core.


Containment Structure
Pressurizer
Steam Generator
Generator
Turbine
Control Rods
Reactor Vessel
Condenser

Summary

Key Points Regarding Basic Electrical Generation Using Steam

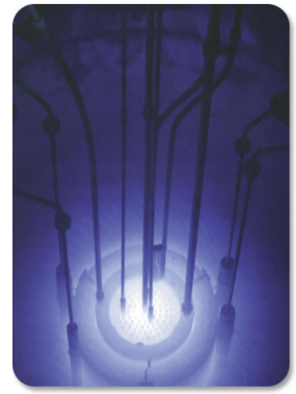
- Steam electric power plants are among the most efficient type of power plant for large-scale commercial production of electricity.
- Basic components of a steam electric plant include:
    - Steam supply system for creating the steam
    - Throttle/control valves for controlling the rate and amount of steam going to the turbine
    - Steam turbine for providing the rotating motion to the electrical generator
    - Electrical generator for creating the electrical charge
    - Main condenser and cooling system for cooling the steam back into liquid water form, prior to returning it to the steam generating system
    - Feedwater pump for pushing the cooled water back to the steam supply system
- An electrical generator includes a rotor and a stator. The rotor is generally a magnet that is centered in the coiled wires of the non-moving stator. The rotor spins within the stator, creating a magnetic field, which in turn generates an electric current in the stator.
- There are three common methods for cooling a steam electric plant with circulating water. They are:
    - Body of water
    - Forced draft cooling towers
    - Natural convection cooling towers
- The most common power plants in the U.S. are:
    - Fossil fuel: including coal, oil, and natural gas
    - Nuclear
    - Hydroelectric
- Alternative sources are being designed for large-scale power production. Alternatives sources include:
    - Solar farms
    - Wind farms
    - Bio-fuels
- Although similar to other steam electric power plants, a nuclear power plant has some unique characteristics. They include:
    - A nuclear power plant's energy comes from the fission (splitting) of fuel atoms (e.g., uranium, plutonium).
    - Uranium-235 is the preferred reactor fuel because it is easier to initiate the fission chain reaction.
    - Significant processing must occur to convert the raw uranium ore to usable fuel.
    - Reactor fuel assemblies have the same components regardless of the type of reactor. The components are the fuel rods, the spacer grids, and the upper and lower end fittings.
- Due to the potential ill effects of radiation on the surrounding environment, U.S. nuclear power plants include three barriers to the escape of fission products. The three barriers are the:
    - Cladding
    - Reactor Coolant System (RCS)
    - Containment Building
- Nuclear power plants in the U.S. are designed for safety. They require the intervention of people to continue operation and in the event of a serious accident, are designed to automatically shut down.
- There are two types of nuclear power plants in the U.S.:
    - Boiling water reactors (BWRs) account for approximately 1/3 of the power plants in the U.S. The reactor coolant pressure is low enough to allow steam formation within the core area. In a BWR, the steam that turns the turbine has direct contact with the reactor core and is thus more radioactive, leading to higher doses for plant workers.
    - Pressurized water reactors (PWRs) account for approximately 2/3 of the power plants in the U.S. A pressurizer keeps the coolant water in the reactor vessel under very high pressure to prevent the coolant from boiling, even at temperatures of more than 600°F. With a PWR, the steam that turns the turbine never comes in direct contact with the reactor core. Rather, the hot primary coolant passes through a separate steam generator

where it gives up its heat to the feedwater and returns to the reactor core. This generally means lower worker doses, but requires more robust containment buildings for the reactor systems.

Fission Process & Heat Production

Chapter Overview

The Uranium atom has a unique ability to split apart and release great quantities of energy. This splitting of the Uranium atom is called fission. Incredibly large numbers of fissions occur every second in order to produce the high temperature, high-pressure steam for the production of electrical power.



This chapter begins by exploring the atomic structure that makes Uranium a useful fuel in harnessing energy within a nuclear reactor. We will continue by examining how a self-sustaining fission reaction is created, maintained, and used to generate large amounts of heat energy. The chapter concludes with a view of the reactor systems designed to support and sustain fission while protecting the environment and the reactor components.

Objectives

After completing this chapter, you will be able to:

- Describe the relationship between fission and heat production.
- Identify the basics of atomic theory.
- Describe the fission reaction.
- Describe how the fission reaction is used to generate heat.
- Describe the role of fission and heat production in nuclear power generation.

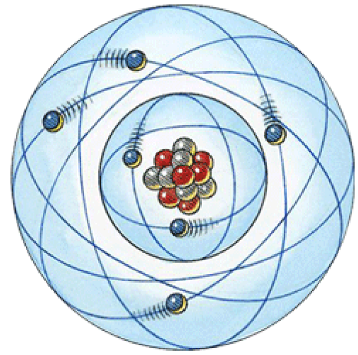
Estimated time to complete this chapter:

45 minutes

Review of Atomic Theory

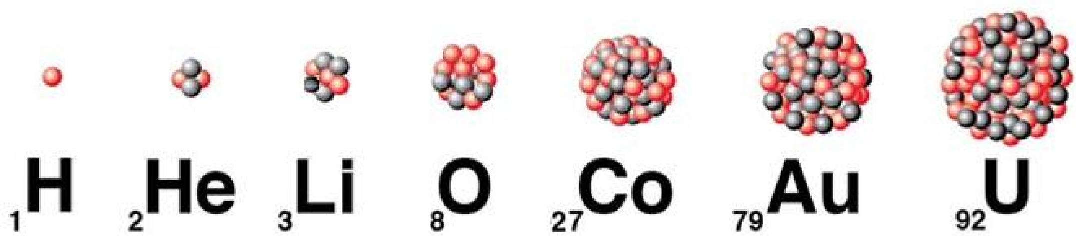
Inside the Atom

The atom is the smallest building block for all manner of substances. An atom is composed of positively charged protons, negatively charged electrons, and neutrons (neutral). Its structure is similar to a model of our solar system: planets orbit around our central sun. In an atom, the electrons orbit around the atom's nucleus of protons and neutrons.



The simplest atom is hydrogen, generally composed of one proton and one electron. Because it has just one proton, its atomic number is 1. A larger atomic number, representing the increasing number of protons, results in a different element (different kind of atom).

Helium, for example, has an atomic number of 2. Two electrons orbit around a nucleus of two protons and two neutrons. Each unique number of protons (positively charged particles in a nucleus) represents a different chemical element.

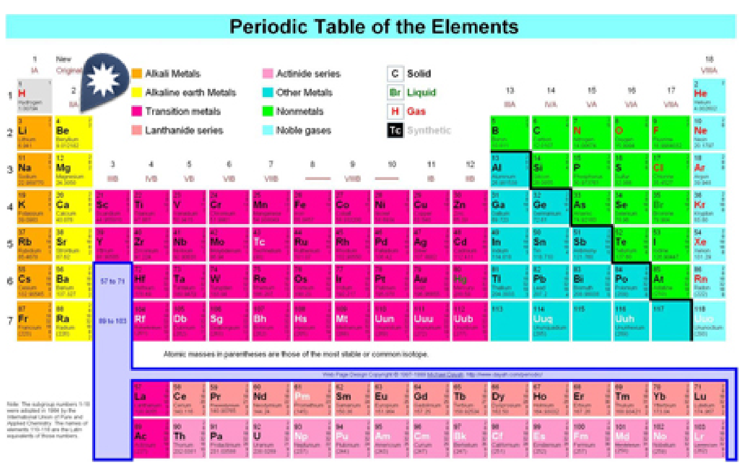


Periodic Table of the Elements

Each element has a chemical symbol. Elements are listed by increasing atomic number and grouped by similar chemical characteristics in the Periodic Table of the Elements.

A group, or family, is a vertical column in the periodic table. Groups are considered the most important method of classifying the elements. In some groups, the elements have very similar properties and exhibit a clear trend in properties down the group. Groups of elements contain the same number of electrons in their outer shell (orbit); this tends to dictate how they react chemically.

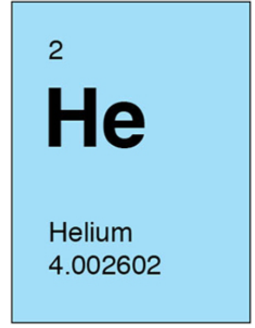
A period is a horizontal row in the periodic table. Although groups are the most common way of classifying elements, there are some areas of the periodic table where the period (horizontal) trends and similarities in properties are more significant than vertical group trends.



When elements are displayed in a visual format, the chemical symbol appears with a number both above and below the symbol. The number below the symbol represents the isotopic or mass number. This number is a combination of the protons and neutrons within the nucleus of the atom. Sometimes, this number is a whole number representing the most common isotopic mass, or a number showing the *average* weight of all the isotopes of that particular element.

The element helium, for example, is represented as He with a **mass number** of approximately 4.

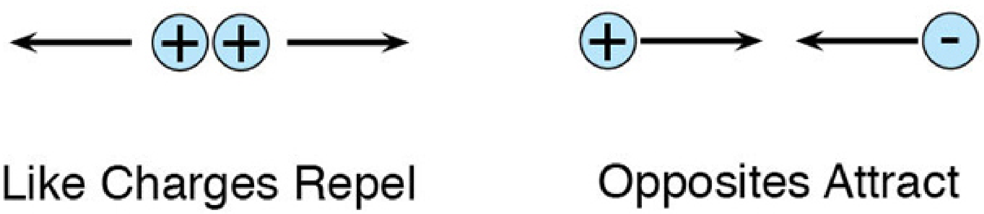
The number above the symbol is the **atomic number**, or just the number of protons. Often, the atomic number for a specific element will be omitted since this number will never change.



Electrostatic Force

Similar to the static electricity you would find in clothes pulled from a residential clothes dryer, electrostatic forces act on the particles within an atom. **Like** charges repel and **opposite** charges attract. In elementary school, you may remember pushing pairs of magnets toward each other; the northern magnetic poles would repel each other, as would the southern poles. However, placing a northern and southern magnetic pole near each other would cause them to stick together (attract).

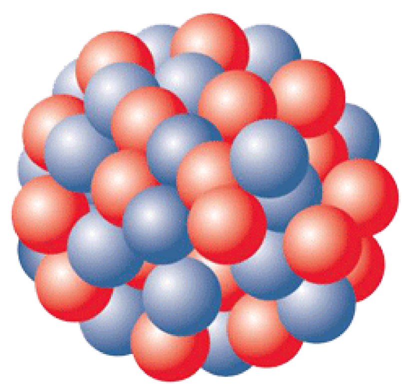
Previously, we explained that all protons within the nucleus of the atom are positively charged. Complex atoms exist with large numbers of protons packed relatively close together. Since **like** charges repel, some type of force must be present to prevent the **like**-charged protons from repelling each other and pushing the nucleus apart.



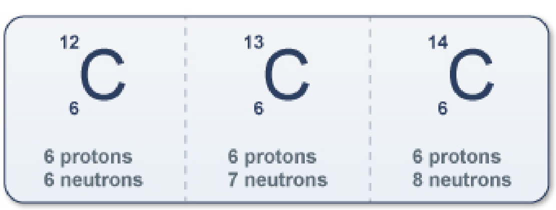
Neutrons

Electrostatic forces between the positively charged protons in a nucleus try to push them apart. When neutrons, with no electrical charge, are part of a nucleus, an attractive—also called binding—nuclear force that is much stronger works between the proton(s) and neutron(s) to overcome the repulsive electrostatic forces. Atoms with more than one proton require neutron(s) to allow the binding force, or nuclear glue, to hold the nuclei together.

All atoms found in nature, except the basic hydrogen atom, have one or more neutrons in their nuclei. The number of neutrons in atoms of the same element can vary, creating different isotopes of that element. Collectively, all elements (and their isotopes) are referred to as nuclides. As the atomic number increases (more protons in the nucleus), even more neutrons are required to overcome the repulsive electrostatic forces. Once the size of the nucleus increases to a certain size, the elements tend to become less stable: more likely to undergo a transformation (radioactivity). The ratio of neutrons-to-protons in an atom is directly related to its stability.



Next, we will explore naturally occurring nuclides.

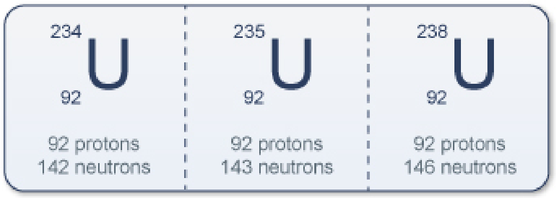


Naturally Occurring Uranium

The element Uranium also has naturally occurring isotopes: U-234, U-235, and U-238.

Power reactors in the United States (U.S.) use uranium as fuel. About 99.3% of all uranium atoms in nature are the isotope U-238; most of the remaining 0.7% of the uranium atoms are U-235. Trace amounts (far less than 1%) of U-234 can also be found. Another isotope, U-233, does not exist naturally but it can be manufactured and used to fuel some types of reactors.

We use U-235 for power reactor fuel, as this simplifies the fission process. Other kinds of isotopes can be induced to undergo fission, but it is generally easier and more practical to use U-235. After the World War II effort to create atomic weapons, the US had a large infrastructure already in place to enrich (i.e., increase % U-235) Uranium.

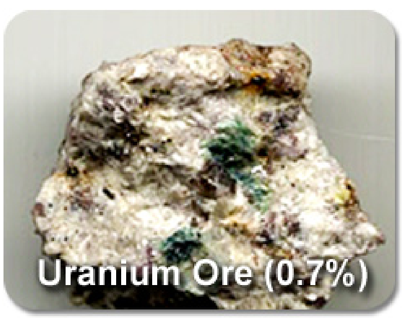


Enrichment

There are facilities that mechanically increase the percentage of U-235 found in naturally occurring uranium (0.7%) to between 3.5 to 5%. This enriched uranium is used to make the fuel pellets in a reactor. This process is called isotopic enrichment.

Naval reactors and nuclear weapons are enriched to much higher levels. The level of enrichment of U.S. power reactors is such that they physically **CANNOT** explode like a nuclear bomb. The NRC limits the maximum enrichment level of Uranium for power reactors to far below that for weapons grade needs.

The graphic shows U-235 fuel pellets that have been enriched from natural ore.



Fission and Heat

We have learned that the naturally occurring isotope U-235 is a desirable fuel for nuclear reactors due to its high probability of absorbing a neutron, which then results in fission. Once the fuel is enriched in U-235, it can more easily sustain a fission reaction releasing great quantities of energy and more neutrons.

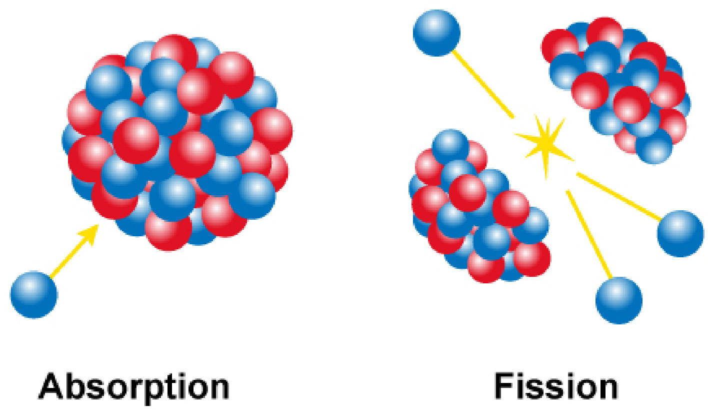
Let's explore the fission reaction in detail.

Fission of the Uranium Isotope U-235

We will now explore how Uranium-235 is useful as a reactor fuel.

Absorption

U-235 is useful as a reactor fuel because it will readily absorb a thermal (lower energy) neutron to become the highly unstable isotope U-236. Due to its unstable nature, 80% of all U-236 atoms will fission (split).



Fission

The fission of U-236 releases energy (mostly in the form of kinetic energy of the fission products), which is used to heat the reactor coolant water, produce high-pressure steam and, ultimately, electricity.

On the downside, however, the fission products are highly radioactive and can linger for a long time, up to tens of thousands of years. The fission products will decay, which also produces more heat. This is known as decay heat, which must be continually removed even after the reactor is shut down. Even though the decay heat is far less than that produced in an operating reactor, it is sufficient, if not removed by a cooling system, to boil water. If not removed from a shutdown reactor, this boiling can lead to loss of the reactor coolant, and overheating of the fuel.

Supporting the Fission Reaction

U-235 has a relatively high probability of absorbing a thermal (slow) neutron. Therefore, in the reactor, it is desired to slow the neutrons down and then let the U-235 absorb them. This slowing down process is accomplished by the same water used to remove the heat from the fuel. Therefore, the water circulating through the reactor (called the reactor coolant) has two important functions: heat removal and thermalizing, or slowing, neutrons.

Heat Exchange

The water removes the heat from the reactor core to produce the steam used in the turbine. This also prevents the fuel from becoming too hot, which could lead to fuel damage.

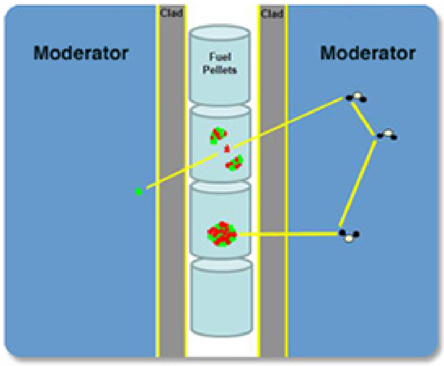
Thermalization

Most neutrons created by fission are born as fast, or high energy, neutrons. The fissioning of U-235 is accomplished with thermal, or slow, neutrons. Therefore, the neutrons must be slowed down, or thermalized. The "slowing down" process is called *thermalization* or moderation.

The Hydrogen in the coolant water is the target of the neutrons and used to control the fission process by slowing the neutrons down and by acting as a reflector to bounce back any high energy (fast) neutrons that try to escape. This conserves the neutron population so that even more fissions may occur. In trying to visualize this slowing (thermalization) process, think of bouncing a billiard ball off of another billiard ball: if the balls are close to the same size, the incoming ball will give up a larger amount of its energy during the collision and slow down more quickly. It the 'target' ball is much larger, the incoming ball will bounce off and recoil closer to its incoming speed and slowing takes longer.

The Hydrogen in light water is mostly the isotope where the nucleus is a lone proton. Since protons and neutrons are about the same size, an incoming neutron that interacts with a like-sized proton will slow more quickly. The faster a neutron is slowed down to thermal levels, the more likely it is to cause a fission in another Uranium atom and the less likely it is for that neutron to 'leak' outside of the core and cause no subsequent fissions.

Neutron Life Cycle



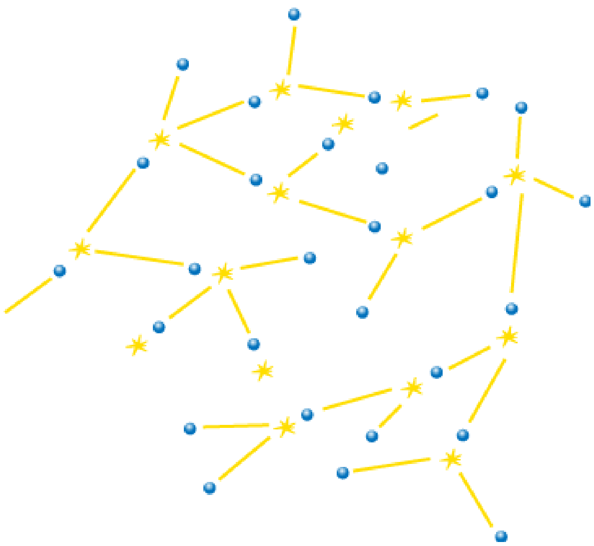
Inside the fuel pellets, U-235 absorbs a neutron, creating an unstable U-236 isotope. Fission of the U-236 isotope releases energy, radiation, and generally two to three new neutrons. The released neutrons are then available to be thermalized and absorbed by other U-235 atoms; thus the fission reaction can continue.

Slower moving neutrons stand a greater chance of being absorbed by U-235. Water inside the reactor slows down a neutron through collisions, increasing the probability of it being absorbed.

The water circulating through the reactor (called the reactor coolant) serves two important functions. First, the water carries the heat from the reactor core to produce the steam used in the turbine preventing the fuel from becoming too hot, potentially leading to fuel damage. Second, the water is used to control the fission process by slowing the neutrons down and acting as a reflector to bounce back any high energy neutrons that try to escape. This conserves the neutrons so that even more fissions may occur.

Imagine AGAIN: On a pool table, a cue ball that strikes another pool ball will transfer more of its energy and slow down faster than if it strikes a bowling ball. We use water to slow the neutrons since the nucleus of the water's hydrogen atoms are protons: essentially the same size as a neutron.

Fission Chain Reaction

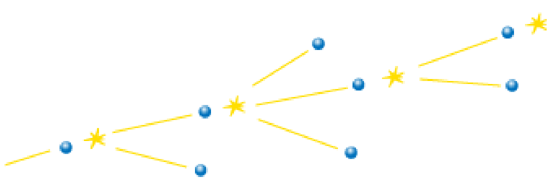


Every fission event releases a tiny amount of energy. Therefore, an incredibly large number of fissions are necessary to produce the high-temperature, high-pressure steam for the production of electrical power. The rate at which the uranium atoms are undergoing fission determines the rate at which heat (and power) is produced.

Since neutrons are necessary to cause the fission event, and since each event of fission releases more neutrons, there is the potential to set up a self-sustaining chain reaction. For this to occur, there must be sufficient material (fuel) capable of undergoing fission, and the material must be arranged so that the neutrons will reach other fuel atoms before escaping (i.e., relatively close together).

Criticality

If conditions in the reactor core are suitable, the chain reaction reaches a self-sustaining state. At this point, for each fission event that occurs a second fission will occur. This point of equilibrium is known as criticality.



If you remember from before, each fission of U-235 produces 2 or 3 new neutrons. If each of these neutrons causes a subsequent fission, you can see that the chain reaction would proceed exponentially and power would become uncontrollable (think Chernobyl). However, most of these neutrons ‘born’ from fission will either leak out of the core area or be absorbed by some material that will not result in a fission (e.g., poison).

Fission in a Nuclear Reactor

Lost Neutrons

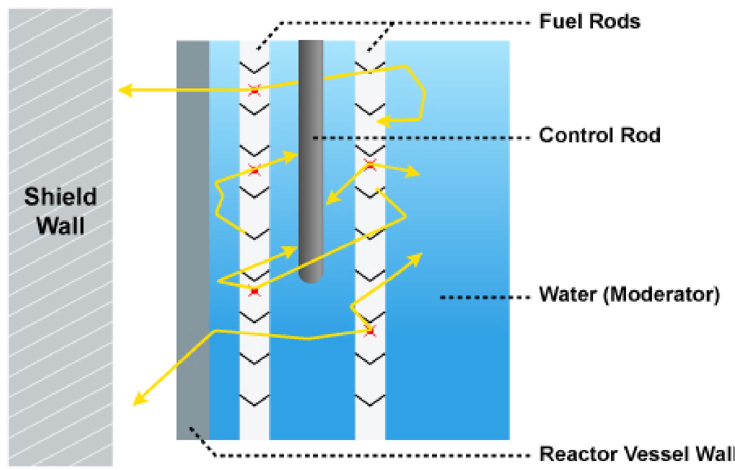
As stated earlier, a single neutron can cause U-235 to fission, thus releasing around 2–3 more neutrons. One can see, then, that all the neutrons produced by the fission process do not end up causing subsequent fissions or the chain reaction could take off uncontrollably (like at Chernobyl). The majority of neutrons produced by the fission process are not absorbed by U-235, causing further fissions; most either leak out of the fuel area or are absorbed by materials in the core that do not fission. Materials that absorb neutrons without causing fission are called neutron poisons.

Neutron Leakage Out of Core

Some of the neutrons released by fission leak out of the reactor core area and are absorbed by the dense concrete shielding around the reactor vessel. In PWRs, the instruments that detect neutrons are placed in this shielding around the reactor vessel, allowing operators to estimate the power being produced by the core by measuring the amount of neutrons leaking out of the core. In BWRs, the production of steam within the core will mask or interfere with external detection of leaked neutrons, so they use in-core detectors to measure the fission rate.

The neutrons that remain within the core area are absorbed by the materials from which the various core components are constructed (U-235, U-238, steel, control rods, etc.). Some of these neutrons absorbed in the various core materials cause fission; however, most are absorbed in non-fissile materials.

The graphic to the right shows the path (in yellow) of neutrons traveling through the core. They bounce off of various materials, especially the Hydrogen in the water/coolant. The neutrons may completely escape the core, be absorbed in fuel creating more fissions, or be absorbed in neutron poisons (e.g., rods) or other materials that do not undergo fission.



Neutron Poisons

Any material that absorbs neutrons and does not fission is a poison to the fission process. The reactor vessel, structural components, and the reactor coolant all absorb neutrons. The new elements formed from the splitting of the large U-235 nucleus, known as fission products, also absorb neutrons. Xenon-135 and Samarium-149 are two important examples of these uranium fission products that readily absorb neutrons. When these poisons absorb neutrons they impact the number of neutrons available to cause fission of uranium, impacting the power produced by the core. These fission product poisons can affect core power without the operator doing anything; therefore, their effects must be monitored closely.

In both BWRs and PWRs operators can insert and withdraw control rods that absorb neutrons, thus affecting power level.

Also, in pressurized water reactors (PWRs), operators routinely add or subtract the element Boron to/from the reactor coolant. This dissolved boron in the reactor coolant will readily absorb neutrons, making it a strong poison. BWRs do not add Boron to the coolant unless it is needed in an emergency to shut down the fission process.

Control Rods

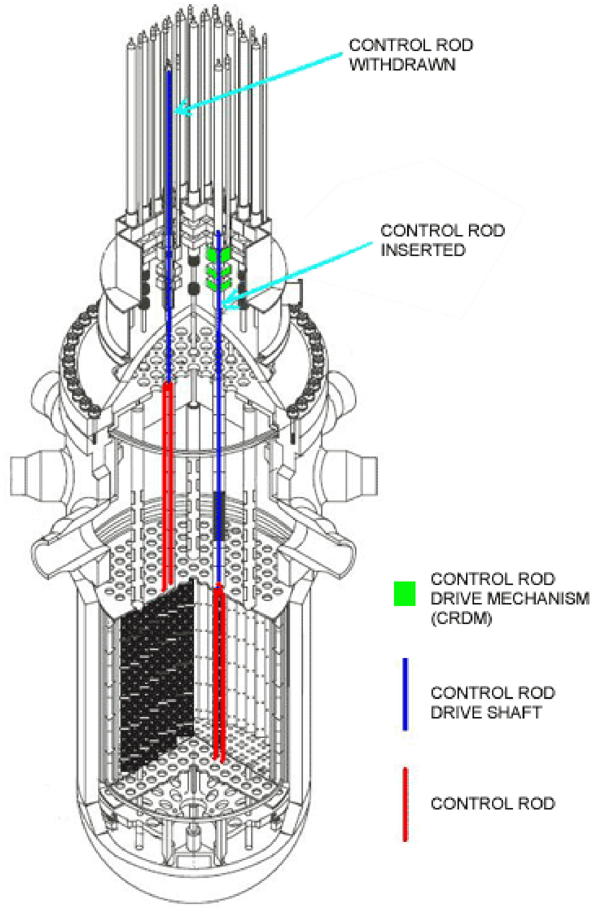
Control rods are concentrated neutron absorbers (poisons) that can be moved into or out of the core to change the rate of fission occurring in the reactor.

Rods IN - Fewer Neutrons - Power Down
Rods OUT - More Neutrons - Power Up

Rod insertion adds neutron poisons to the core area, which makes fewer neutrons available to cause fission. This causes the fission rate to decrease, resulting in a reduction in heat production and power. Pulling the control rods out farther has the opposite effect: it increases available neutrons, causing heat and power to rise.

The graphic to the right shows the control rod arrangement for PWRs. In the BWR chapter, you will see that control rods insert from the bottom, up.

The actual physical construction of control rods (also known as control element assemblies (CEA) or rod cluster control assembly (RCCA) in other designs) varies greatly between different reactor designs. However, their purpose is universal: absorption of neutrons to reduce the fission rate and shut down a critical reactor.



Reactor Coolant Water

Coolant water helps produce a steady rate of reactor power. Water reflects escaping neutrons back into the core and slows neutrons down just enough to facilitate absorption by the U-235 isotope. Despite water's effectiveness as a moderator it also can absorb neutrons, thus causing the removal of neutrons from the fission chain reaction. In addition, absorption of the neutron by the hydrogen nucleus can cause activation of the Hydrogen and result in the formation of Tritium, which is a radioactive isotope. More on activation later in the course.

First, water has some capacity to absorb neutrons—thus acting as a neutron poison. More importantly, the physical characteristics of the water dramatically affect its success as a moderator.

Moderating Effects of Coolant Water

Have you noticed water vapor rising from a hot cup of tea or coffee? As the water is heated it becomes less dense, turns to vapor, and rises up through the cooler surrounding air.

The same temperature-to-density relationship is true for the coolant water inside a reactor. If the reactor coolant temperature increases, the water becomes less dense. This means that the water becomes less effective at slowing and reflecting the neutrons. Conversely, as the coolant temperature decreases, its density increases, improving its moderating potential.

If the only action to occur was a change in the temperature of the moderator, power would also change. This moderator temperature effect is a major factor in the control of the fission process and heat production of the reactor.

Remember this when you review the Chernobyl Accident in Module 11. You will see that this water density change characteristic had a huge effect on that accident.

Voids

There is a point where the temperature of water dramatically affects the fission reaction rate and power production. As the coolant reaches the boiling point, steam bubbles begin to form creating voids in the reactor coolant. A void is an area of very low density water, or steam.

Water at the bottom of the core is far more dense than the water-steam mixture at the top. Therefore, neutron moderation is much better toward the bottom of the core. The effect of voids on moderation, and therefore power, are large in a BWR.

A pressurized water reactor (PWR) avoids the issue of void effect by maintaining pressure in the reactor coolant high enough to prevent boiling. Therefore, the effects of voids on the power production in a PWR are minimal.

Inherent Stability of Light Water Reactors (LWR)

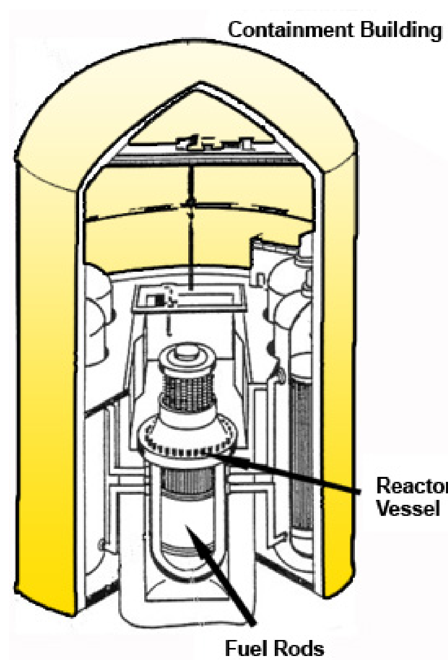
Previously we explained that the coolant temperature/density relationship is an important safety feature of U.S. light water reactors (LWRs). For example, as fission intensifies in a boiling water reactor (BWR), more heat is generated and then transferred to the reactor coolant water. As fission increases, more coolant water boils. This creates voids (steam bubbles), allowing more neutrons to escape from the core or to reduce the slowing down (thermalization) of neutrons that don't escape. Fewer neutrons would then be feeding the fission chain reaction, reducing the power and therefore the amount of heat generated. This is the complete opposite of what occurred in the Chernobyl Disaster (more on that in another chapter).

Coolant water flow can also be increased to lower the temperature and reduce voids. As mentioned previously, the cooler water has a positive moderating effect, thereby increasing reactor power.

Fission Product Decay

As discussed previously, fission products are smaller atoms formed when larger uranium atoms are split during the fission process. Many of these fission products that are neutron absorbers (poisons) must be compensated for by removing some of the controllable poisons. In addition to being neutron poisons these byproducts release heat and radiation, which must be contained or eliminated by the reactor systems.

The fission products are usually highly radioactive. They emit a large amount of radiation, therefore they must be contained within the plant. A system of barriers has been developed to prevent these atoms from escaping into the environment. These barriers are the fuel cladding, the reactor coolant system pressure boundary including the vessel, and the containment.



Fission Product Decay

Decay Heat

Radioactive decay of fission by-products releases a large amount of energy. This energy, called decay heat, can build in the fuel and must be removed by cooling the irradiated fuel even after the reactor fission process is stopped. An excessive buildup of decay heat in the reactor fuel caused the Three Mile Island core melt-down, which will be further explained in Chapter 10.

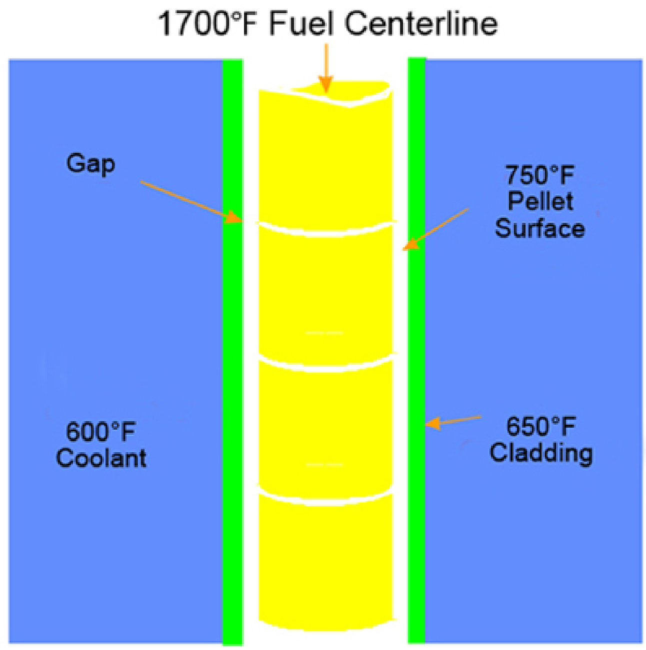
Redundant safety systems are designed into the plant to remove this heat after the plant is shut down. Radiation, decay heat, and fission product barriers will all be discussed in greater detail in subsequent sections of this course.

Fuel Rod Temperatures

When a reactor is operating at full power, under normal conditions, the fuel pellet has an average temperature of about 1400°F. It is about 1700°F at its centerline and 750°F at the pellet surface. Heat is transferred through the gas-filled gap between the fuel pellet and clad, and the outer clad surface is about 650°F.

The melting temperature of the ceramic fuel is approximately 5200°F. The fuel cladding can be damaged by temperatures in excess of 1800°F. Significant clad damage can be expected at sustained temperatures above 2200°F, including a reaction between the clad and water that produces flammable Hydrogen gas.

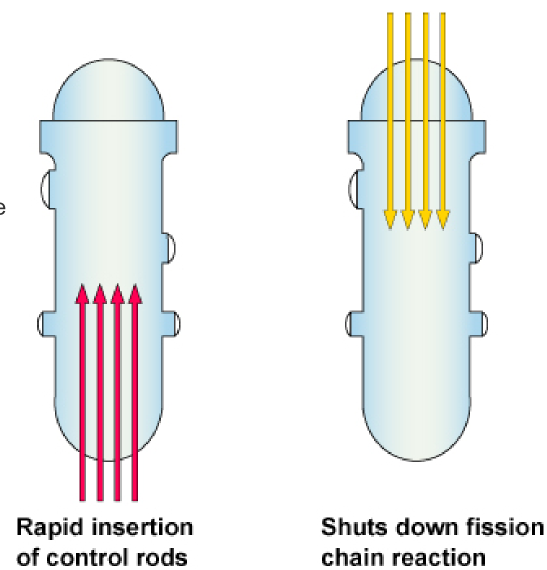
If conditions approach an operating limit, the reactor protection system will rapidly insert the control rods to shut down the fission chain reaction, which removes the majority of heat production. This rapid insertion of rods into the core to shut down the reactor is called a reactor trip or scram.



Reactor Scram

A reactor scram or trip, is the rapid (2 to 4 seconds) insertion of the control rods into the core to stop the fission process. Even though fissions are not completely stopped, the chain reaction is reduced to below a self-sustaining level, causing a significant decrease in reactor power in just a few seconds. As reactor power is reduced, the heat it produces is also reduced. Plant systems can then easily remove potentially dangerous heat levels produced by the decay heat from fission products.

The most immediate method of reducing the power level in the core is to insert all control rods. The graphic at the right shows a BWR with its rods inserting from the bottom of the vessel, and a PWR with its rods inserting from the top of the vessel.



Summary

Key Points of Atomic Theory

- Atoms are composed of positively-charged protons, negatively-charged electrons, and neutrons.
- Elements are listed by increasing atomic number and grouped by similar chemical characteristics in the Periodic Table of the Elements.
- Electrostatic forces act on the particles within an atom. **Like** charges repel and **opposite** charges attract. Neutrons provide the binding nuclear force that holds the nucleus of atoms together.
- Isotopes have a different number of neutrons for each specific element. Since chemical elements can have different numbers of neutrons, the use of mass numbers is necessary to distinguish one isotope from another.
- The element uranium also has naturally occurring isotopes: U-234, U-235, and U-238. U-235 is the preferred fuel for the fission process in commercial reactors.
- Fuel enrichment chemically or mechanically increases the percentage of U-235 found in naturally occurring uranium; enrichment takes it from 0.7% to between 3.5 to 5%. This concentrated uranium-235 is used to make the fuel pellets that power U.S. commercial reactors.

Key Points of Fission and Heat

- U-235 is useful as a reactor fuel because:
  - U-235 will readily absorb a thermal/slow neutron.
  - The fission of U-235 releases large amounts of energy, used to produce high-pressure steam for generation of electricity.
  - Fission also releases from two to three additional neutrons, which causes other fissions and establishes a self-sustaining chain reaction.
- The reactor coolant water has two important functions:
  - The water absorbs the heat from the reactor core to produce the steam and prevents the fuel from becoming too hot.
  - The water moderates the fission process by slowing the neutrons down and acts as a reflector to help bounce back higher energy neutrons that try to escape. This process is also called thermalization.
- The fission chain reaction will reach a self-sustaining state. Fission of U-235 releases neutrons, which sustains subsequent fissions. This point of neutron equilibrium is known as criticality.

Key Points of Fission in a Nuclear Reactor

- Materials that absorb neutrons without causing fission are called neutron poisons.
  - Some of the neutrons leak out of the reactor core and are absorbed by the concrete shielding or other components/materials.
  - Neutrons in the core area will be absorbed by the various core components (U-235, U-238, steel, control rods, etc.).
- Control rods are concentrated neutron absorbers (poisons) that can be moved into or out of the core to change the rate of fissioning.
- The physical characteristics of the reactor coolant water dramatically affect its success as a moderator.
  - As the reactor coolant temperature increases the water becomes less dense, reducing its moderating effectiveness.
  - Conversely as the coolant temperature decreases its density increases, which improves its moderating potential.
  - As the coolant reaches the boiling point, steam bubbles or voids begin to form. A PWR mitigates this problem by maintaining pressure to increase the boiling point, limiting steam production.
- The temperature/density relationship is an important safety feature of U.S. LWRs.
  - As fission increases, more coolant water boils allowing neutrons to escape the chain reaction. This reduces the amount of heat/power generated.
  - Coolant water flow can also be increased to lower the temperature and reduce voids. The cooler water has a moderating effect by increasing reactor power.

- The fission products are usually highly radioactive. A system of barriers has been developed to prevent these atoms from escaping into the environment. These barriers are:
  - Fuel cladding
  - Reactor coolant system pressure boundary
  - Containment
- Radioactive decay of fission byproducts releases a large amount of decay heat, which can build in the fuel pellets or other parts of the barrier systems. Safety systems are designed into the reactor to remove this heat after the plant is shut down.
- Fuel rods and other reactor components can be damaged by excessive heat.
  - Normal Operating Temperatures:
    - Fuel pellet average = 1400°F
    - Clad surface = 750°F
  - Accident Conditions Temperatures
    - Ceramic fuel pellet melting point = 5200°F (decreases with age)
    - Fuel cladding damaged at temperatures above 1800°F
  - Significant fuel clad damage results at sustained fuel clad temperatures above 2200°F, which can release fission products from the fuel rods.
- A reactor scram or trip is the rapid (2 to 4 seconds) insertion of the control rods into the core to stop the fission chain reaction.

Boiling Water Reactor (BWR) Systems

Chapter Overview

Recall that boiling water reactors are one of two types of nuclear power plants in use in the United States (U.S.), and that approximately one-third of the currently active reactors are BWRs. The first commercial nuclear power plant, activated in 1955, was a BWR. Since then, BWRs have evolved as technology and commercial needs have changed. There are currently six different designs of BWRs, five of which are in use in the U.S.

Remember the key differences between a BWR and PWR. With a BWR:

1. The steam that turns the turbine comes in **direct contact** with the reactor core.
2. The water in the reactor core is kept at a relatively low pressure to allow it to boil and produce steam.

In this chapter, we will look at an overview of BWRs; consider the purposes of some of the major systems; and identify components associated with the major systems of a BWR.

Objectives

After completing this chapter, you will be able to:

- Describe BWR systems, to include the purposes of major subsystems and components.
- Explain commercial electric power generation by a BWR nuclear power plant.
- Identify the primary subsystems of a BWR system.
- Identify the purpose of the key subsystems of a BWR system.

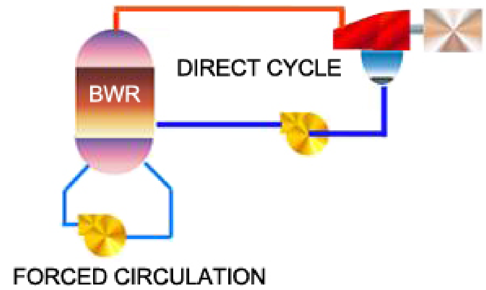
Estimated time to complete this chapter:

30 minutes

Overview of BWR Systems

Evolution of Forced Circulation

As mentioned earlier, BWR designs have evolved over time. In the earliest designs, the flow of water through the core was driven by natural convection. As water was heated and boiled, it rose and incoming colder water replaced it at the bottom of the reactor vessel.



Current BWRs, however, use forced circulation because it allows more direct control of the power in the core. With forced circulation, the amount of power generated by a BWR may be increased by using a mechanical pumping system to force the water through the core.

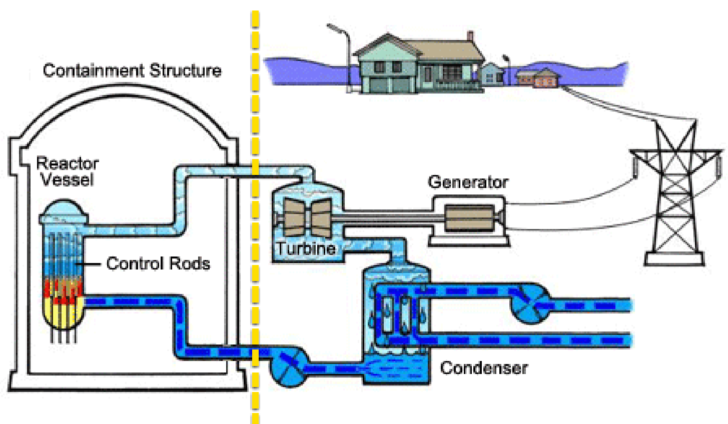
In this design, a portion of the coolant is taken outside of the vessel into recirculation loops, where it is increased in pressure by means of recirculation pumps. Water at increased pressure is pumped from the recirculation loops back into the reactor pressure vessel.

Recall that the number and size of steam voids affects neutron moderation and, therefore, power. In a BWR, increasing forced circulation decreases the steam voids, which causes an increase in power.

Primary and Secondary Systems of a BWR

All nuclear power plants can generally be divided into what are known as primary and secondary systems. With a BWR, however, the true dividing line between the two is not as concrete as with a PWR.

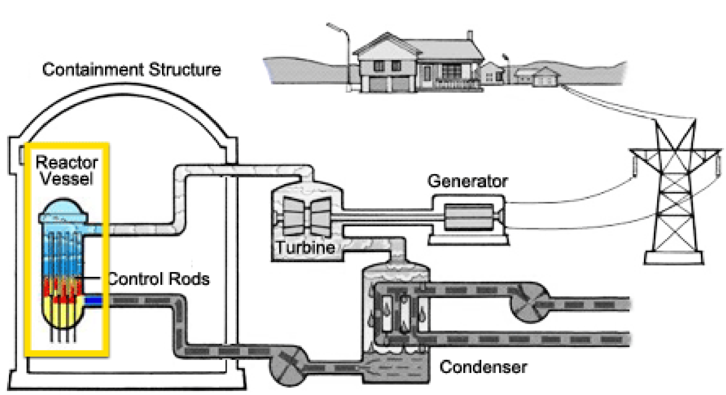
For our purposes, we will refer to BWR primary systems as those being **within** the containment building and BWR secondary systems as those being **outside** of the containment building. Realize that there may be some systems that don't exactly meet this criteria, or that cross the dividing line.



Basic Functionality of a BWR: Steam Production

Steam production occurs within the primary system. Inside the reactor vessel, a steam/water mixture is produced when very pure water (reactor coolant) flows upward through the core absorbing heat. The reactor pressure is low enough to allow this coolant to change to steam within the core area.

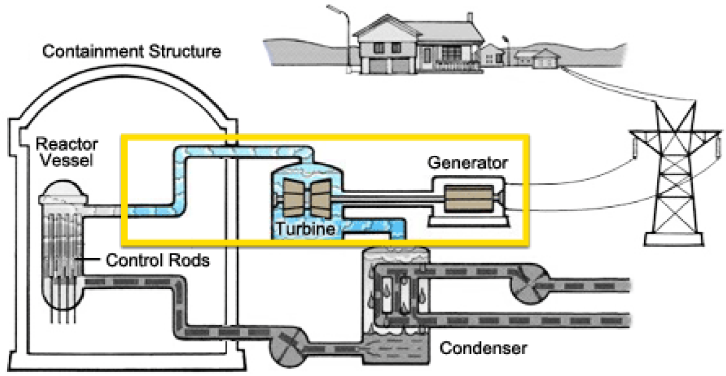
The recirculation and jet pumps allow the operator to vary coolant flow through the core, which changes reactor power by affecting the amount of steam bubbles present.



Basic Functionality of a BWR: Electrical Generation

The steam/water mixture exits the top of the core region (still within the reactor vessel) and enters two stages of moisture separation. Moisture separation occurs within the primary system. Here, the liquid water is removed before the steam is allowed to enter the steam line and the extracted moisture is mixed back with incoming feedwater. This improves plant efficiency by pre-heating the relatively colder incoming feedwater.

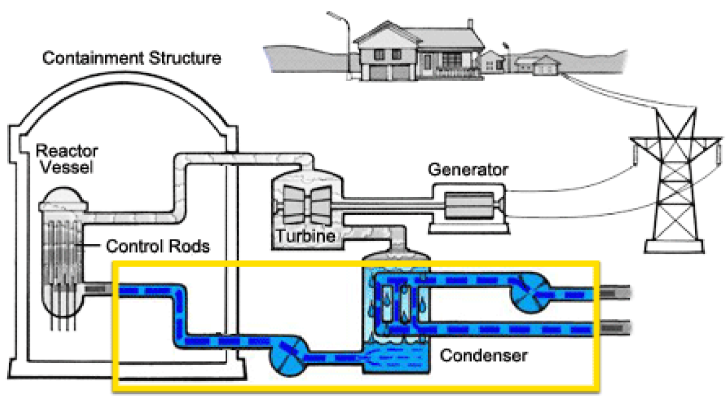
The steam exits the reactor vessel (primary system) through the main steam lines and enters the secondary system. In the secondary system, the steam is directed to the main turbine causing it to spin. The spinning turbine is directly connected to the electrical generator; this motion then generates an electro-magnetic potential (voltage) in the attached electrical generator. The only big difference between the BWR and PWR at this point is that the steam driving the turbine in a BWR has been in direct contact with the core and may have higher levels of radioactivity. Thus, the steam turbines of a BWR must be shielded for dose control during power operations.



Basic Functionality of a BWR: Exhaust Steam

The used steam, exhausted from the turbines, is directed to the condenser where it is cooled and condensed back into water. The resulting water (condensate) is pumped out of the condenser with a series of pumps and back to the reactor vessel, as feedwater, to start the process over again.

As you can see, a BWR is a closed system. The water pumped into the reactor becomes the steam for electrical generation. This steam is then recaptured, condensed back into water, and returned to the reactor vessel again as feedwater.



Functional Systems and Subsystems of a BWR

Basic Functional Systems

All BWR plants have the same basic functional systems. A functional system may include subsystems that are in the primary system, the secondary system, or both. The basic functional systems of a BWR include:

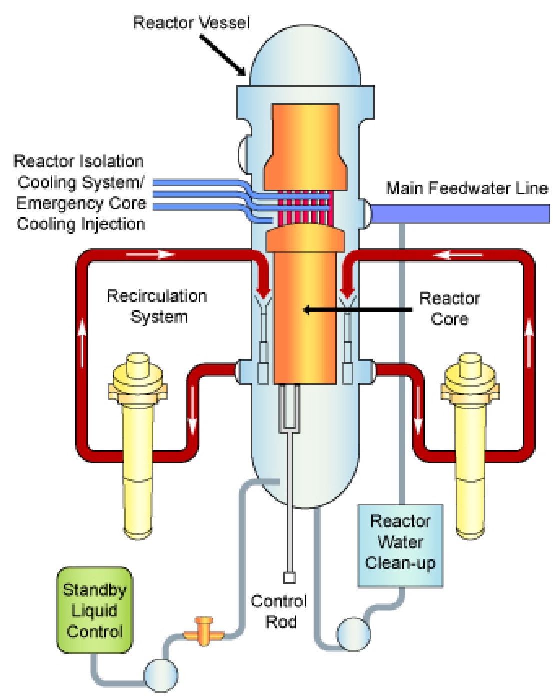
- Reactor coolant system (RCS)
- Emergency core cooling systems (ECCS)
- Containment system (CONT)
- Process instrumentation & control systems (I&C)
- Secondary systems (SEC)

Let's take a look at each of these systems in more detail. We'll start with the RCS. Keep in mind that different designs of BWRs will have slight differences in the layout and types of these systems; however, they all basically operate the same way.

Reactor Coolant System (RCS)

The RCS allows the operators to maintain and control optimal temperature in the reactor core. Subsystems of the RCS include:

- Reactor vessel & core
- Recirculation system
- Reactor core isolation cooling system
- Reactor water cleanup
- Standby liquid control
- Control rods



RCS: Reactor Vessel & Core

The reactor vessel assembly consists of the reactor vessel and its internal components. The functions of the reactor vessel assembly are to:

- House the reactor core
- Support and align the fuel and control rods
- Provide a flow path for circulation of coolant past the fuel

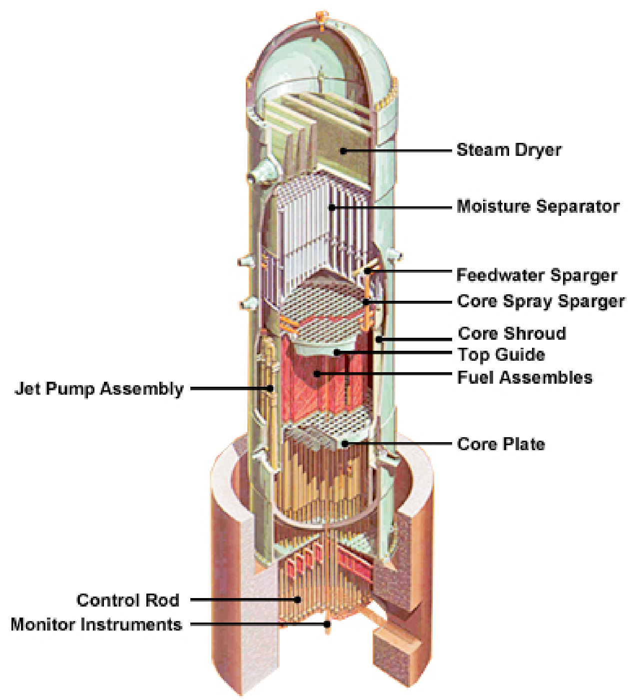
- Remove moisture from the steam exiting the core, and direct the steam to the vessel outlet

The reactor vessel is vertically mounted and consists of a cylindrical shell with an integral rounded bottom head. The top head is also hemispherical and can be removed to facilitate refueling. The head is kept in place by massive nuts and bolts. The entire vessel assembly is supported on its lower, hemispherical head by the circular vessel support skirt, which is mounted to the reactor vessel support pedestal.

RCS: Reactor Vessel and Core Components

The components of the reactor core and vessel are:

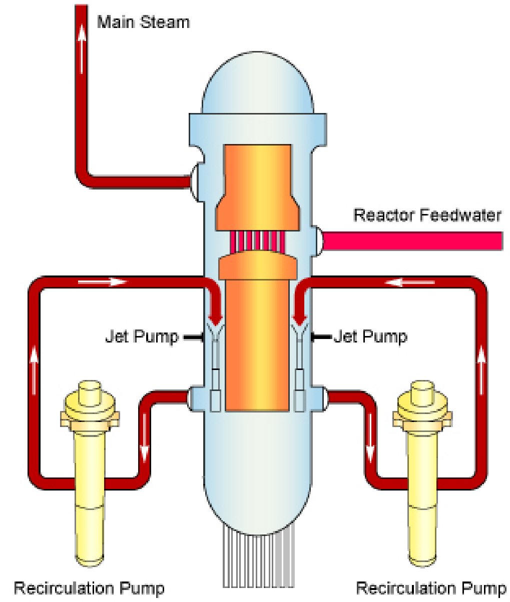
- Steam dryer
- Moisture separator
- Feedwater spargers
- Core spray spargers
- Top guide
- Jet pump assemblies
- Core shroud
- Fuel assemblies
- Core plate
- Control rods
- Monitoring instruments



RCS: Recirculation System

The recirculation system pumps water through the core. Incoming reactor feedwater is mixed with the moisture stripped out of the steam in the vessel moisture separators. A portion of this feed is taken out of the reactor vessel and its pressure is increased by a recirculation pump. The discharge of this pump is re-introduced into the vessel through jet pumps that pick up more feed and push it into the core at higher flow rates than can be achieved with natural circulation alone. The use of recirculation pumps 'feeding' the jet pumps produces a total flow about three times larger than the recirculation pumps, alone, would give.

As the flow through the core is increased, the mass/size of the steam bubbles/voids is reduced. This allows better moderation (slowing or thermalization) of the fission neutrons, which increases reactor power. Operators will vary recirculation flow in order to control reactor power and steam flow to the main turbine.



RCS: Reactor Core Isolation Cooling

The reactor core isolation cooling (RCIC) system supplies high-pressure makeup water to the reactor vessel when the reactor is isolated from the main condenser, and/or when the reactor experiences a loss of the reactor feed pumps.

Basically, following a plant trip, the main steam valves to the turbine will shut, stopping the normal method of removing core heat. The RCIC system will supply steam directly from the reactor to a separate, turbine-driven pump that will add feedwater into the reactor through its normal feed line. This provides a method for removing decay heat and adding coolant to the core.

The capacity of the RCIC system is NOT sufficient to support normal power operations. It is, instead, sized to remove DECAY HEAT. This makes it an important (safety-related) part of the core protective subsystems.

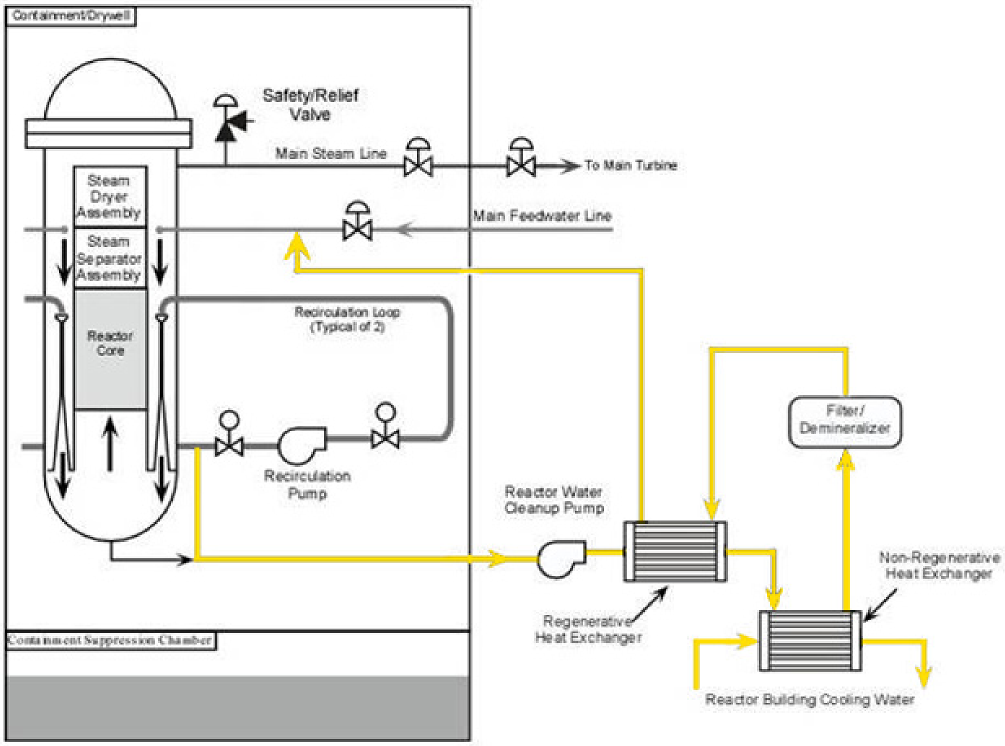
RCS: Reactor Water Cleanup

Since boiling occurs in the core of a BWR, it is important that the reactor coolant stays ULTRA-PURE. The boiling process will result in the concentration of any non-volatile contaminants in the reactor coolant.

The reactor water cleanup (RWCU) system is used to maintain high reactor water quality by removing fission and corrosion products and other soluble and insoluble impurities.

The RWCU pump takes water from the recirculation system and the vessel bottom head. It then pumps the water through heat exchangers to cool the liquid. Cooling the liquid protects the filter and demineralizer resin. The cooled water is sent through the filter and demineralizers for cleanup.

After cleanup, the water is reheated and returned to the reactor vessel via the feedwater piping.



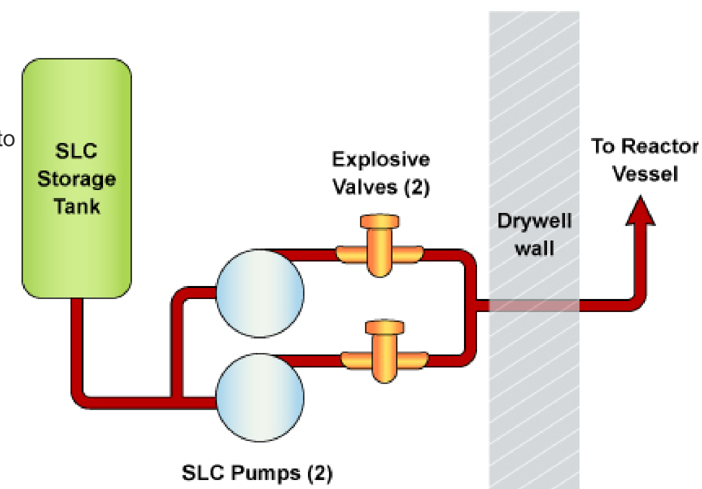
RCS: Standby Liquid Control (SLC)

Normally, BWR plants use pure water as coolant, with no boron added, as in PWRs. When it is necessary to shut down the reactor, all control rods are inserted. This is sufficient to make the core subcritical.

In the event of an emergency, operators have the ability to manually add large quantities of boron (a neutron poison/absorber) to the RCS, using the SLC system, to ensure the core is shut down.

The SLC system consists of:

- A heated storage tank containing boron
- Two positive displacement pumps
- Two explosive (squib) valves
- Piping that directs the boron to the reactor vessel



RCS: Control Rods

In a BWR, except for a few peripheral fuel assemblies, the fuel is arranged in the vessel into control cells. Each control cell consists of a control rod and four fuel assemblies immediately surrounding it.

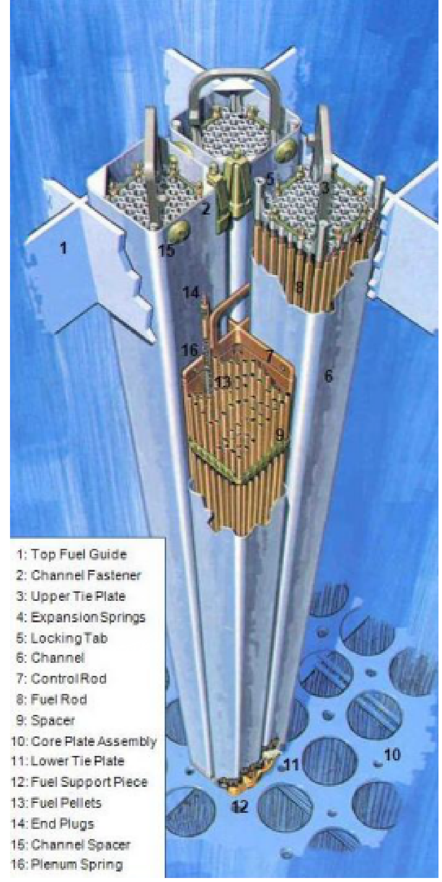
The four fuel assemblies are supported by a fuel support piece. Around the outer edge of the core, the peripheral fuel assemblies that are not in a cell are supported by individual support pieces.

Control Rod Drive System

The control rod drive system makes changes in core reactivity by moving the neutron-absorbing control rods in response to the reactor manual control system (RMCS) signals. It rapidly inserts all control rods to shut down the reactor in response to reactor protection system (RPS) signals or a manual SCRAM as called for by operators.

As discussed in the first module, the BWR control rods are situated externally to the fuel assemblies. Most BWR rods are cruciform-shaped (cross) and slide up between a group of four fuel assemblies, collectively known as a cell.

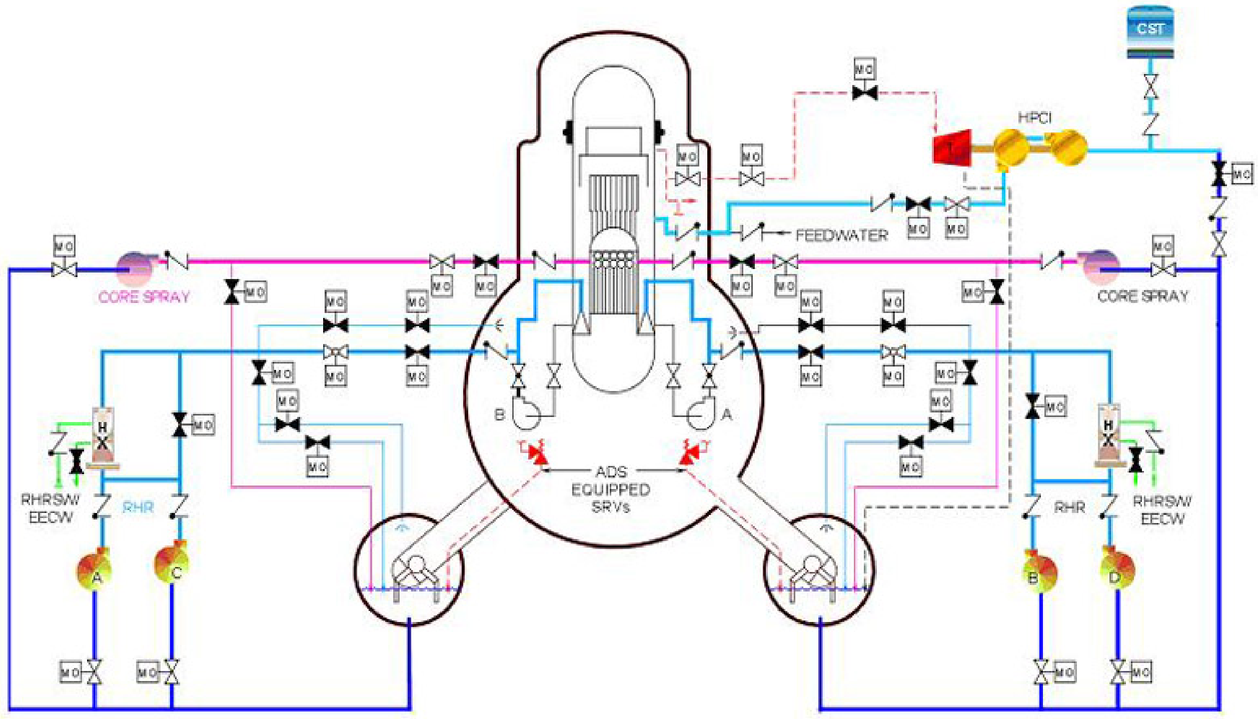
The control rods of a BWR are inserted from the BOTTOM of the core. This is necessary because of the placement of steam drying equipment at the top of the vessel, at the core outlet.



Emergency Core Cooling System

The emergency core cooling system (ECCS) provides core cooling under loss-of-coolant-accident (LOCA) conditions to limit fuel cladding damage. The ECCS consists of two high pressure and two low pressure systems. Together, the subsystems of the ECCS allow the operators to maintain and control optimal temperature in the reactor core. Subsystems of the ECCS include:

- High pressure systems
  - High pressure coolant injection (HPCI)
  - Automatic depressurization system (ADS)
- Low pressure systems
  - Low pressure coolant injection (LPCI)
  - Core spray (CS)



Typical ECCSs for BWR/3/4

ECCS: High Pressure Systems

High Pressure Coolant Injection

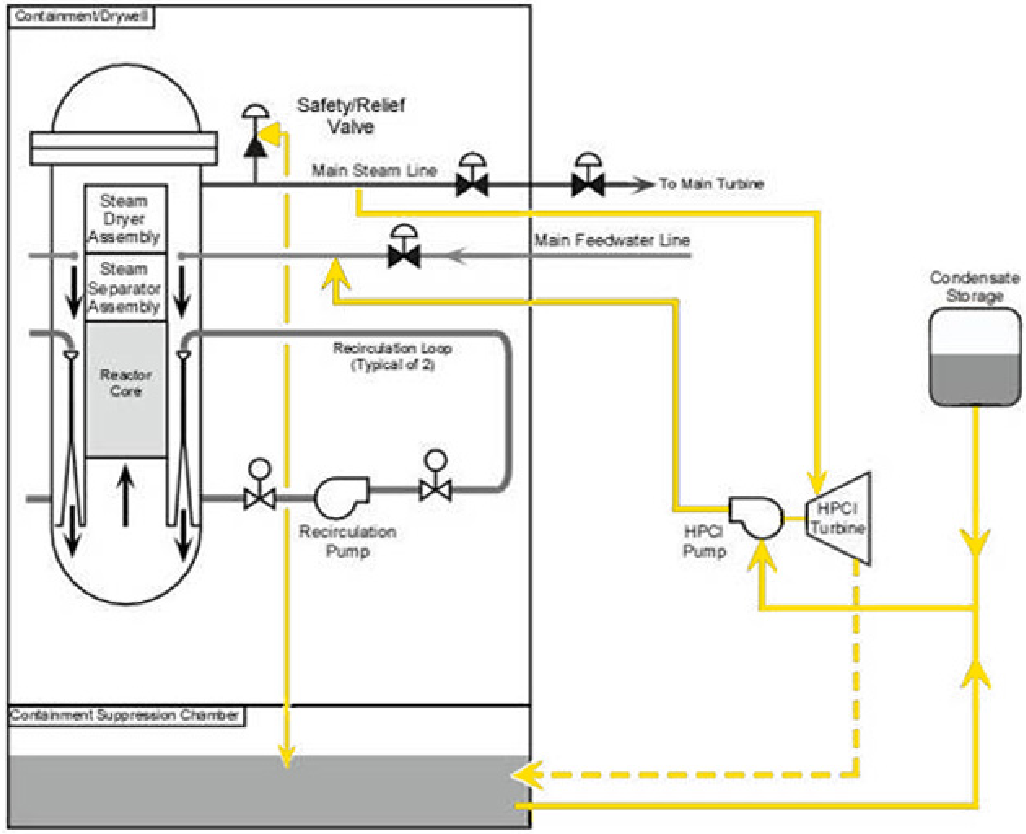
The high pressure coolant injection (HPCI) system is an emergency core cooling system requiring no AC power, plant air, or external cooling water. It is independent, so that a failure of other ECCS systems would not affect its ability to function. The power for running the HPCI pump is steam from the reactor (like the RCIC system), created by the decay heat of the core.

The HPCI system provides make-up water to the reactor vessel for core cooling for relatively small- and medium-sized LOCAs (leaks). It supplies make-up water from a large storage tank to the reactor at pressures above normal down to where the low pressure ECCS systems eventually can inject.

Automatic Depressurization System

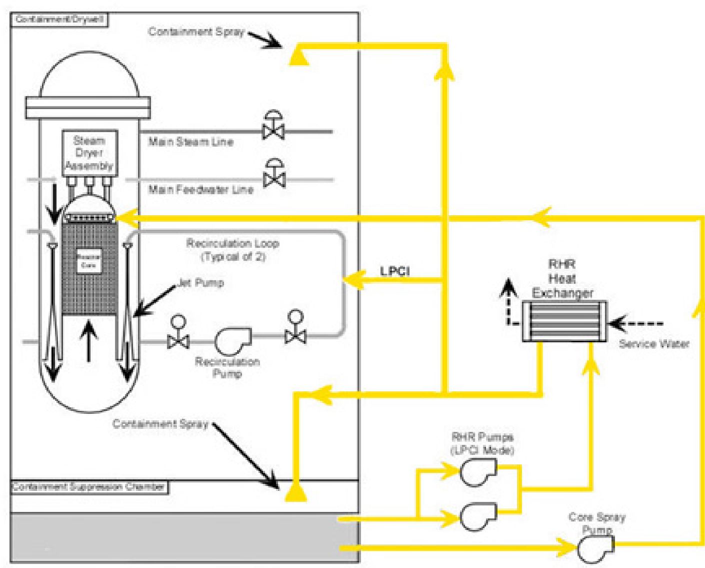
The automatic depressurization system (ADS) consists of a control system that will open select safety relief valves, as required. Opening the valves depressurizes the reactor for small- or intermediate-sized LOCAs in case the high-pressure injection system is not available or cannot sufficiently recover the reactor vessel water level.

Simply put, the ADS releases steam from the RCS so that pressure drops low enough for the low pressure ECCS pumps to inject water into the RCS for cooling.



ECCS: Low Pressure Systems

Low Pressure Coolant Injection/ Residual Heat Removal System



The residual heat removal (RHR) system is a multi-purpose system with several operational modes, each utilizing the same major pieces of equipment. The low pressure coolant injection (LPCI) system is the dominant mode and is the normal valve lineup configuration of the RHR system. The LPCI system is used during normal operations.

The LPCI system provides make-up water to the reactor vessel for core cooling under intermediate and large break LOCA conditions. The LPCI mode operates automatically to restore and, if necessary, maintain the reactor vessel coolant inventory to preclude fuel cladding temperatures in excess of 2200°F. During LPCI operation, the RHR pumps take water from the suppression pool and discharge it to the reactor vessel.

The other mode of the RHR system is for during and after shutdown to maintain an appropriate temperature.

Core Spray

The core spray (CS) system consists of two separate and independent pumping loops, each capable of pumping water from the suppression pool into the reactor vessel. Core cooling is accomplished by spraying water on top of the fuel assemblies.

Containment System

The containment system (CONT) provided for a particular product line is dependent on the vintage of the plant and the cost-benefit analysis performed prior to plant construction. Over the years the CONT evolved for BWRs, resulting in three progressive designs.

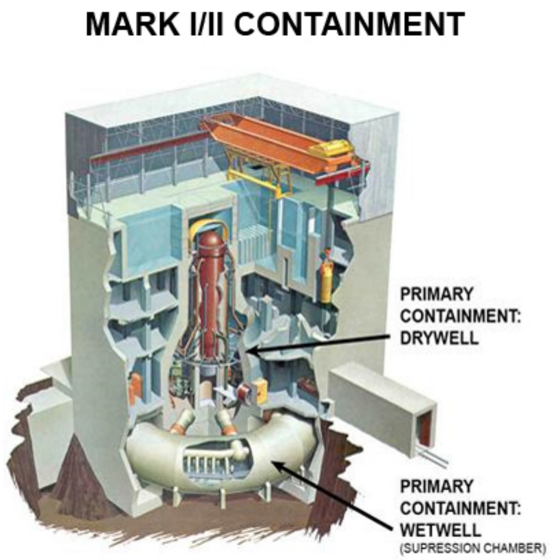
The major containment designs are the Mark I, Mark II, and the Mark III. All three containment designs use the principle of pressure suppression for loss of coolant accidents and include the following subsystems:

- Primary containment
- Secondary containment
- Standby gas treatment

CONT: Primary Containment

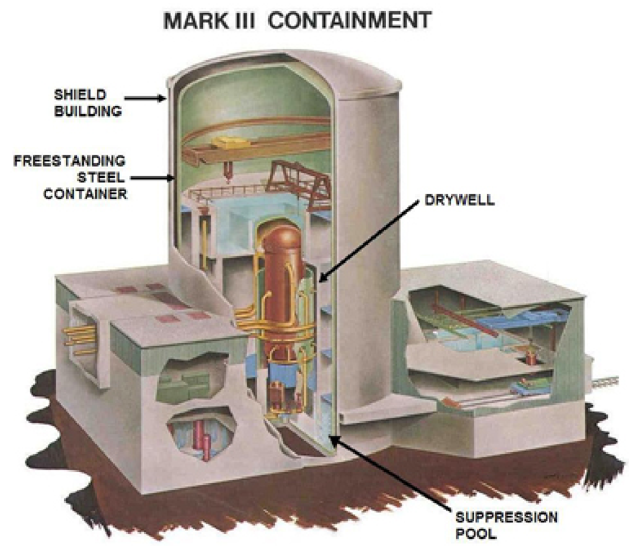
The primary containment is designed to:

- Condense steam and contain fission products released from a loss of coolant accident so that offsite radiation doses specified in 10 CFR 100 are not exceeded
- Provide a heat sink and water source for certain safety-related equipment



Mark I and Mark II

Primary containments for the Mark I and Mark II designs include a drywell and a wetwell. The wetwell functions as a suppression chamber.



Mark III

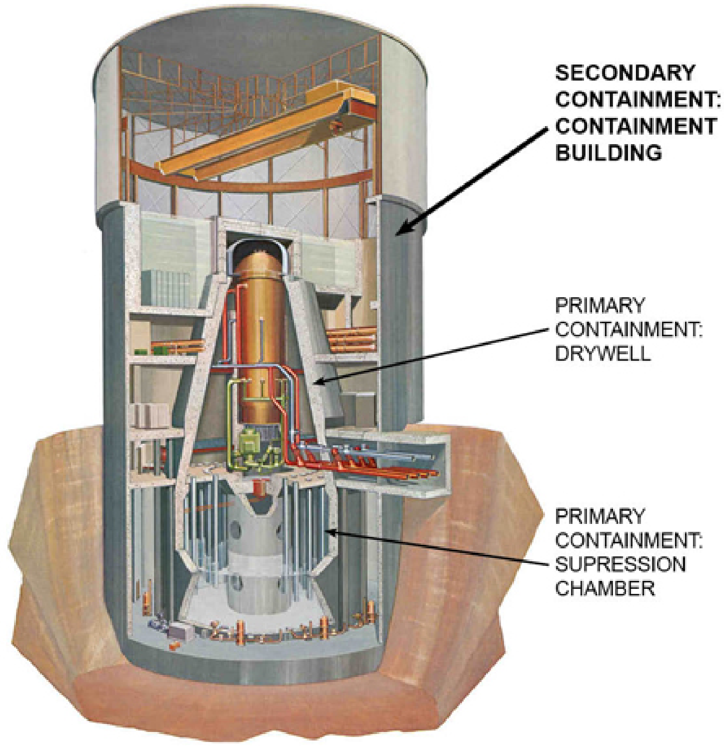
The primary containment for the Mark III design utilizes a suppression pool, versus a wetwell. The function is largely the same: heat energy from the reactor, if not removed by steam to the turbine, can be directed to a large pool of water that can absorb the heat.

CONT: Secondary Containment

Secondary containment is also known as the reactor building. It surrounds the primary and delineates and houses the spent fuel pool and emergency core cooling systems.

The secondary containment of a BWR are not robust buildings like PWR containments. It is generally provided as a lower-pressure barrier to the release of fission products. Both the Chernobyl and Fukushima Disasters showed how easily a BWR secondary containment building can be breached.

All three CONT designs (Mark I, II, and III) include some type of secondary containment.



CONT: Standby Gas Treatment (SGTS)

The SGTS processes exhaust air from the secondary containment during an accident in order to limit the dose rates to the public to within allowable limits. It also maintains a negative pressure in the secondary containment following loss of the normal ventilation system so any leakage is inward, away from the environment. It can also be used to perform containment leak tests and to purge (clean) the primary containment.

The atmosphere inside the Mark I and II PRIMARY containment is made inert with nitrogen gas. This reduces the possibility of an explosion following a loss of reactor coolant. Large amounts of hydrogen can be generated in the reactor vessel if the core becomes uncovered and remains dry for a period of time.

In the Mark III containment design, a leak tight, steel containment vessel surrounds the drywell and suppression chamber to prevent gaseous and particulate fission products from escaping. No nitrogen gas is used.

Process Instrumentation & Control Systems

The process instrumentation and control systems (I&C) allow the operators to effectively monitor the reactor processes and all support processes, and to make corrections to the system when needed.

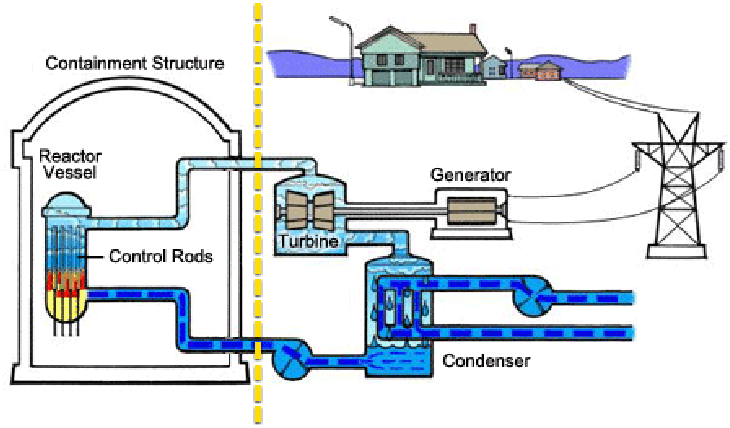
I&C subsystems include:

- ▶ Reactor protection system (RPS)
- ▶ Electro-hydraulic control (EHC)
- ▶ Neutron monitoring
- ▶ Feedwater control

Secondary System

As discussed earlier, the secondary system (SEC) of a nuclear power plant includes those systems that are outside the containment building. In a BWR, the secondary system includes the:

- Main steam system
- Condensate and feedwater system
- Cooling water system



SEC: Main Steam System

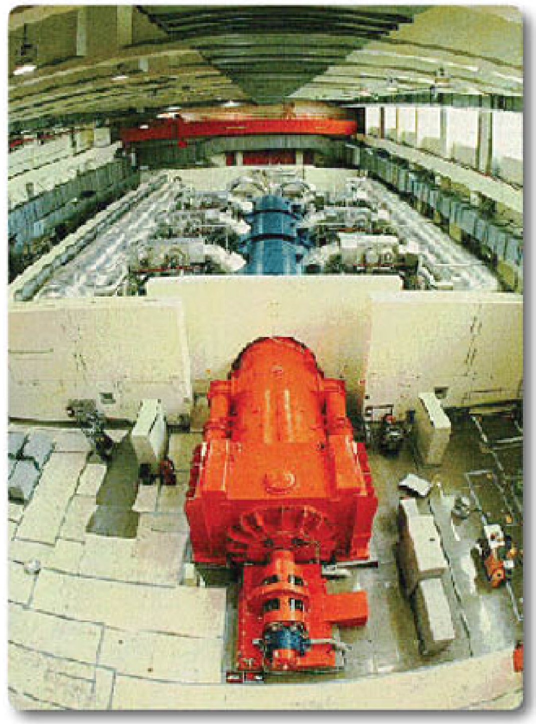
The main steam system directs steam from the reactor vessel to the:

- Turbine generator
- Bypass valves
- Reactor feed pump turbines
- Other selected balance of plant loads

It also directs steam to certain safety-related systems under abnormal conditions and provides overpressure protection for the reactor coolant pressure boundary.

Additionally, the main steam system also provides steam to places that need heating (i.e., feedwater heaters, condenser spargers, and others).

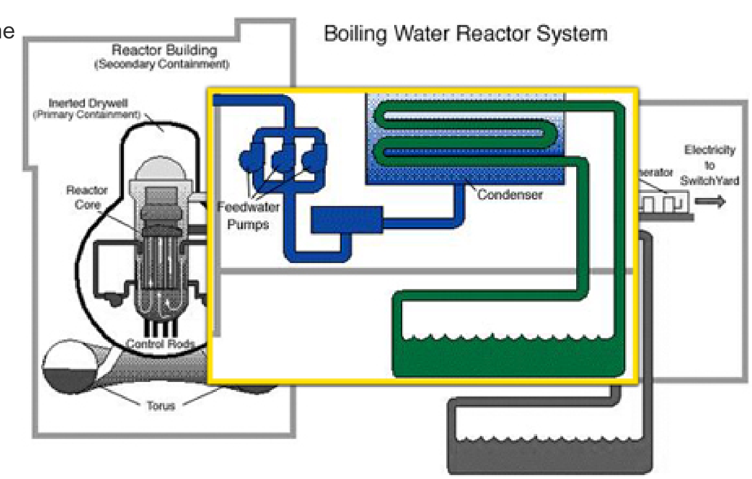
(Note in the graphic to the right: BWR turbine components are SHIELDED for radiation protection. PWR turbines are NOT.)



SEC: Condensate & Feedwater System

The condensate and feedwater (FW) system performs numerous functions. The FW system:

- Condenses turbine exhaust and bypass steam
- Removes impurities
- Heats the feedwater
- Delivers the water back to the reactor vessel at the required rate



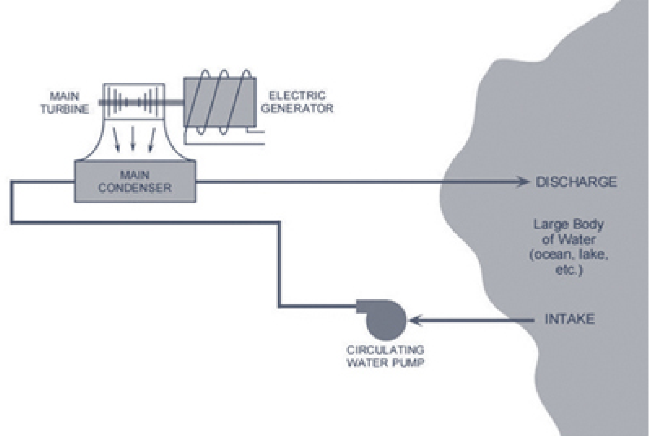
The FW system is integrated into other systems and links them together. The feedwater piping also provides a means to discharge water to the reactor vessel for the:

- Reactor water cleanup (RWCU) system
- Reactor core isolation cooling (RCIC) system
- High pressure coolant injection (HPCI) system

SEC: Cooling Water System

The cooling water system interacts with the FW system, but is not a part of it. Recall that the FW system sends cool water to the reactor vessel and the other primary systems. The cooling water system pulls water from an external source (e.g., a river or ocean, or a cooling tower of some type on the site) to pass through the condenser and cool the steam that has been exhausted from the turbines and other secondary components.

The water in the cooling water system never mixes with the water in the condenser/secondary system. The piping for the cooling water system is separate from the piping of the secondary system. The cooling water condenses the steam back into feedwater in the condenser.



Summary

Key Points

- With a BWR:
    - The steam that turns the turbine comes in **direct contact** with the reactor core.
    - The water in the reactor core is kept at a relatively low pressure to allow it to boil and produce steam.
- Modern BWRs use forced circulation to increase control in maintaining a consistent temperature in the reactor core.
- The primary systems of a BWR are those inside the containment building. The secondary systems of a BWR are those outside of the containment building.
- The three primary functions of the systems in a BWR are:
    - Steam production
    - Electrical generation
    - Steam exhaust
- All BWR plants have the same basic functional systems. The subsystems of basic functional systems vary, based on the design and manufacturer. A functional system may include subsystems that are in the primary system, the secondary system, or both. The basic functional systems of a BWR include:
    - Reactor coolant system (RCS)
    - Emergency core cooling systems (ECCS)
    - Containment system (CONT)
    - Process instrumentation & control systems (I&C)
    - Secondary systems (SEC)
- The reactor coolant system (RCS) allows the operators to maintain and control optimal temperature in the reactor core. Subsystems of the RCS include:
    - Reactor vessel & core
    - Recirculation system
    - Reactor water cleanup
    - Standby liquid control
    - Control rods
- The emergency core cooling system (ECCS) provides core cooling under loss of coolant accident (LOCA) conditions to limit fuel cladding damage. Subsystems of the ECCS include:
    - Reactor core isolation cooling (RCIC)
    - High pressure coolant injection (HPCI)
    - Automatic depressurization system (ADS)
    - Low pressure coolant injection (LPCI)
    - Core spray (CS)
- Over the years, the BWR CONT evolved, resulting in three designs. The major containment designs are the Mark I, Mark II, and Mark III.
- All three containment designs use the principle of pressure suppression for loss of coolant accidents and include the following subsystems:
    - Primary containment
    - Secondary containment
    - Standby gas treatment
- The process instrumentation and control (I&C) systems allow operators to monitor the reactor and support processes and make controlled corrections to the system. I&C subsystems include:
    - Reactor protection system (RPS)
    - Electro-hydraulic control (EHC)
    - Neutron monitoring
    - Feedwater control

- The secondary (SEC) systems are those that are outside of the containment building. In a BWR, the SEC systems includes the:
  - Main steam system
  - Condensate and feedwater system
  - Cooling water system

Pressurized Water Reactor (PWR) Systems

Chapter Overview

The other type of commercial nuclear power plant currently in use in the United States (U.S.) is the pressurized water reactor (PWR). Approximately two-thirds of the operating, commercial nuclear power plants in the U.S. are PWRs. There are currently five major PWR designs in use in the U.S.

Recall the key differences between a PWR and a boiling water reactor (BWR). With a PWR:

1. The steam that turns the turbine does **not** come in direct contact with the reactor core.
2. The water in the reactor core is kept at a very high pressure so that it does **not** boil in the core.

In this chapter, we will look at an overview of PWRs; consider the purposes of some of the major systems; and identify components associated with the major systems of a PWR.

Objectives

After completing this chapter, you will be able to:

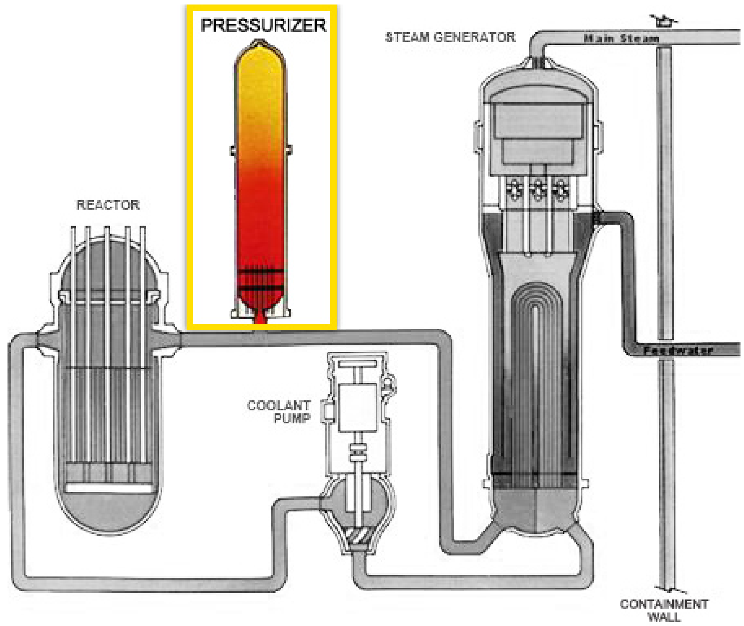
- Describe PWR systems, to include the purposes of major subsystems and components.
- Explain commercial electric power generation by a PWR nuclear power plant.
- Identify the key subsystems of a PWR system.
- Identify the purpose of the key subsystems of a PWR system.

Estimated time to complete this chapter:

40 minutes

Overview of PWRs

Inclusion of a Pressurizer

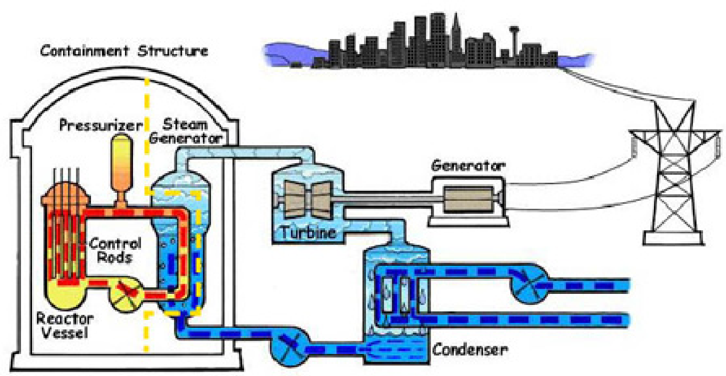


Unlike BWRs, PWRs have not changed much in their basic design. PWRs have always used forced circulation, and include a pressurizer that keeps the coolant under high enough pressure (above 2000 pounds per square inch (PSI) to prevent boiling of coolant anywhere in the Reactor Coolant System (RCS).

The forced circulation is used to transfer the reactor heat from the core to the steam generators (SGs), rather than to control voids in the core. There are virtually no voids in a PWR core due to the high pressure.

The major methodology that operators use to control reactor power in a PWR is by controlling the steam flow out of the SGs. Increased steam flow out of the SG cools off the reactor coolant more; the colder reactor coolant thermalizes/moderates neutrons better, so core power goes up. A simplified mantra in the operation of PWRs is: core power 'follows' steam demand.

Primary and Secondary Systems of a PWR



As with a BWR, a PWR nuclear power plant is also divided into primary and secondary systems. The dividing line between the two is a little more concrete in a PWR than in a BWR.

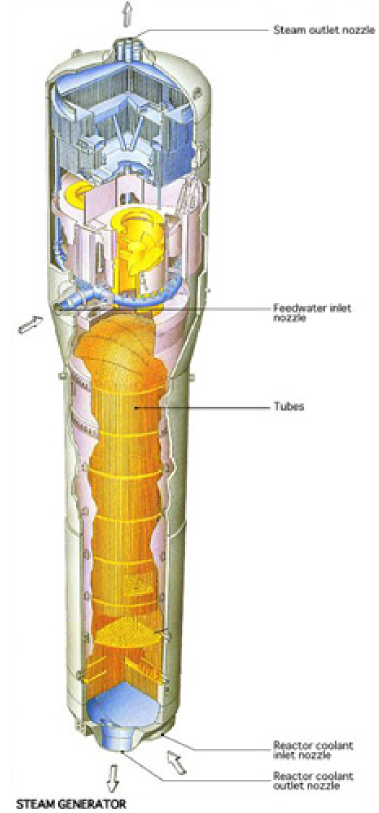
With a PWR, the steam generator (SG) provides the dividing line between the primary and secondary systems. More specifically, the tubes inside the SG constitute the line between primary and secondary. Primary coolant is inside the tubes; secondary water (incoming feedwater mixed with moisture stripped from the steam) is on the outside of the tubes.

The primary systems have direct contact with, or offer direct support to, the core. The secondary system provides support/interaction between the turbine, condenser, and feedwater systems to the SG.

Steam Generator

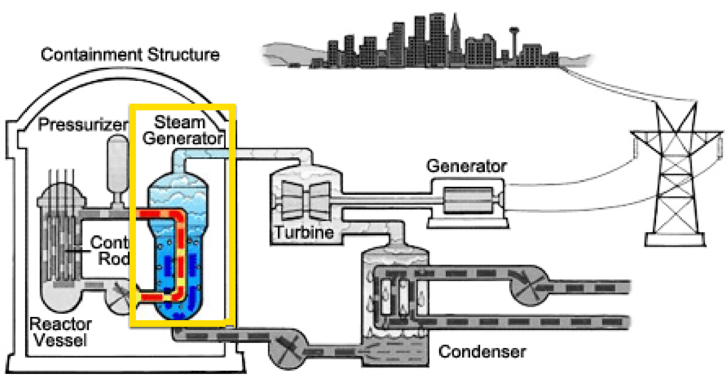
The steam generator (SG) is the heat exchanger used in PWR designs to transfer heat from the primary (reactor coolant) system to the secondary (steam) system. Liquid from both the primary system and the secondary system pass through the steam generator, but they are physically isolated from each other.

The liquid and steam of the primary system is kept INSIDE of the SG tubes. This design permits heat exchange with little or no contamination of the steam OUTSIDE of the SG tubes. This means that secondary system equipment is generally NOT in contact with radioactivity.



Basic Functionality of a PWR: Steam Production

In a PWR, steam is produced in the SG, which is outside of the reactor vessel.



The reactor coolant system (RCS) is kept under very high pressure, more than 2,200 pounds per square inch. This high pressure prevents the primary coolant in the RCS from boiling, even at operating temperatures of more than 600°F. Contrast this with the BWR where reactor/steam pressure is only about 1,000 pounds per square inch, allowing the reactor coolant to boil within the core area at 600°F.

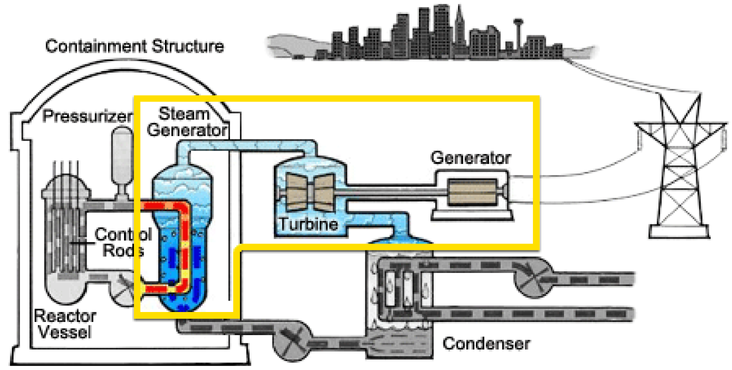
Reactor coolant pumps (RCPs) 'push' the hot coolant from the core through thousands of tubes in the SGs and then back to the core for subsequent reheating. The heat of the reactor coolant inside these tubes (primary side) is transferred to the lower pressure feedwater, which is on the outside (secondary side) of the tubes. Since the SG secondary side pressure is relatively low (1,000 pounds), the feedwater boils to produce steam. You can see that the steam pressure of a PWR is close to that of a BWR; it is the generation of the steam that is different between the two.

Basic Functionality of a PWR: Electrical Generation

Remember that the SG, with the exception of the tubes, and the turbine are part of the secondary system.

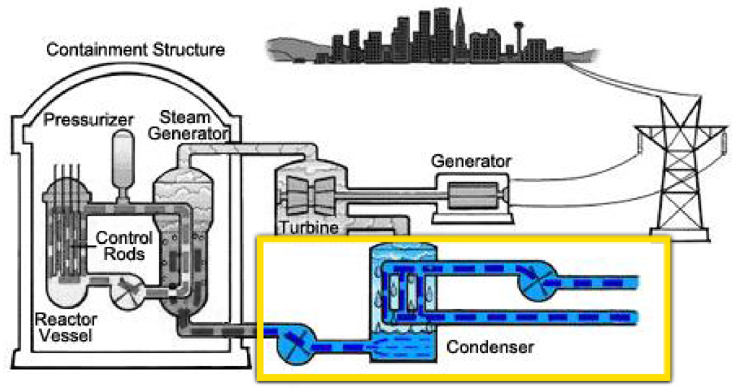
In a PWR, the feedwater of the SG turns into steam to spin the turbine and generate electricity.

Other than the actual location of where the steam is produced, the generation of electricity in a PWR is almost exactly the same as with a BWR. A key difference is that PWR generation does not utilize radioactive steam, so secondary shielding is not necessary.



Basic Functionality of a PWR: Exhaust Steam

The exhaust steam from the turbine goes to the condenser, is cooled, and then returned to the SG as feedwater, to start the cycle over again. Remember, this is different from the BWR where returning feedwater is sent directly back to the reactor core.



The PWR consists of two closed systems—unlike the single, closed system of the BWR. The advantage of this arrangement is that the steam for turning the turbine is never in direct contact with the core, so it is less radioactive. REMEMBER: BWR turbines require shielding but PWR turbines do not.

Functional Systems and Subsystems of a PWR

Basic Functional Systems

Recall that all nuclear power plants have the same basic purpose—to create steam that spins the turbine and creates electricity in the electrical generator. As such, PWR nuclear plants have the same basic functional systems as BWRs. Likewise, the subsystems of those systems vary, based on the design and/or manufacturer (e.g., Westinghouse, Combustion Engineering (CE), Babcock&Wilcox (B&W)). A functional system may include subsystems that are in the primary system, the secondary system, or both. The basic functional systems of a PWR (much like a BWR) include:

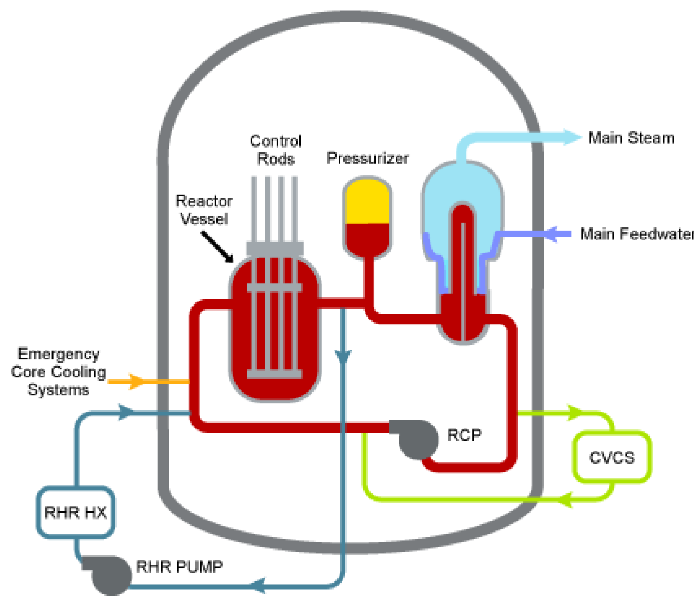
- Reactor Coolant System (RCS)
- Emergency Core Cooling Systems (ECCS)
- Containment System (CONT)
- Process Instrumentation & Control Systems (I&C)
- Secondary Systems (SEC)

Let's take a look at each of these systems in more detail. We'll start with the RCS.

Reactor Coolant System (RCS)

As with a BWR, the RCS allows the operators to maintain and control optimal temperature in the reactor core. Subsystems of the RCS include:

- Reactor Vessel & Core
- Pressurizer (PZR)
- Reactor Coolant Pump (RCP)
- Chemical & Volume Control (CVCS)
- Control Rods and Control Rod Drive System
- Residual Heat Removal (RHR)



RCS: Reactor Vessel and Core

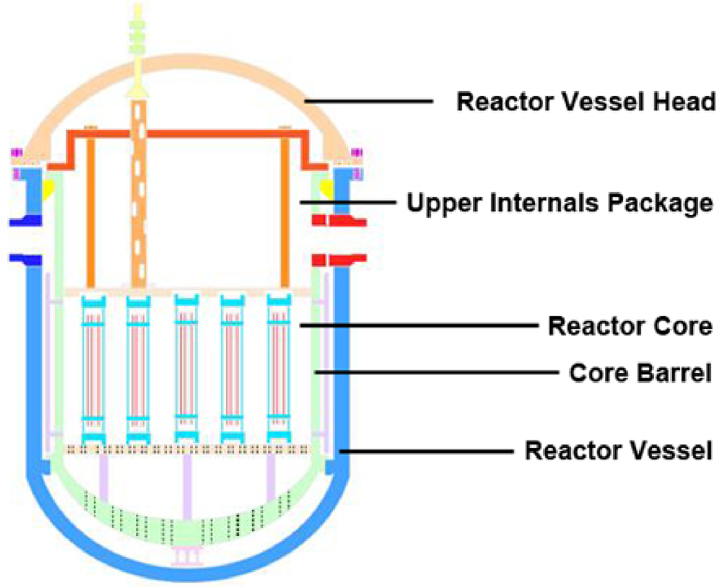
The reactor core, and all the associated support and alignment devices, is housed within the reactor vessel. The major components are:

▶ Reactor vessel and vessel head

▶ Core barrel

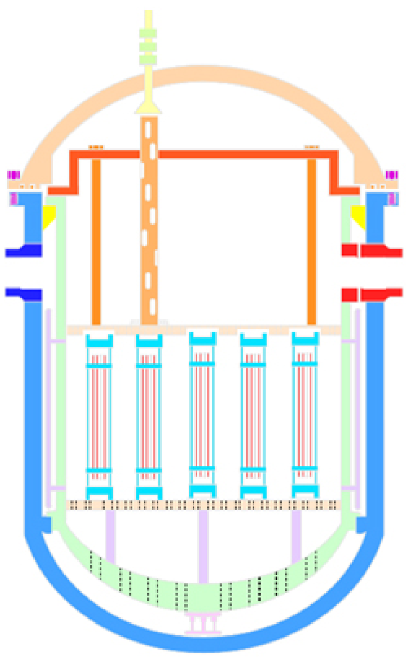
▶ Reactor core

▶ Upper internals package

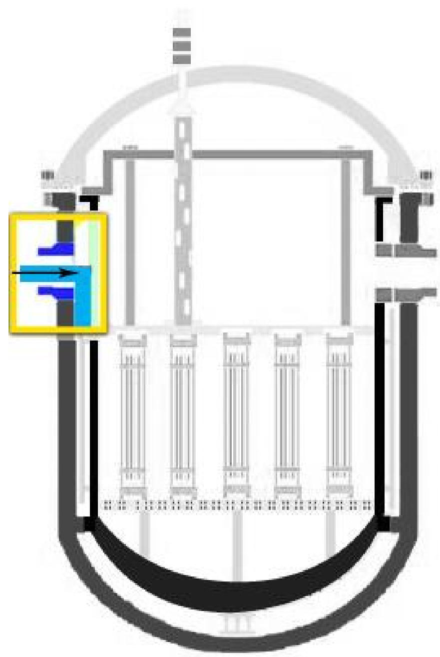


RCS: Reactor Vessel and Core Flow Through

View the images below to learn about the flow path of the reactor coolant.

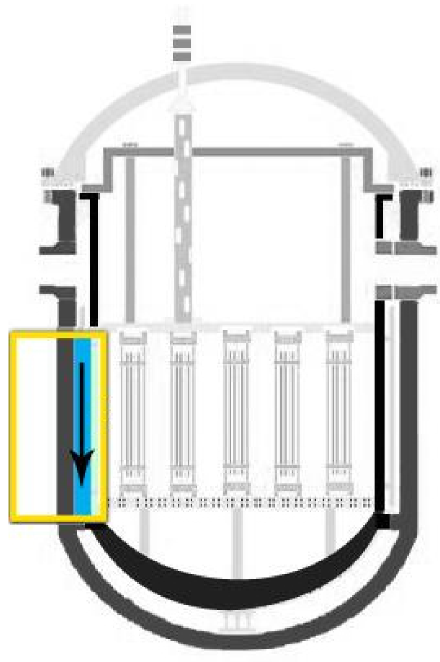


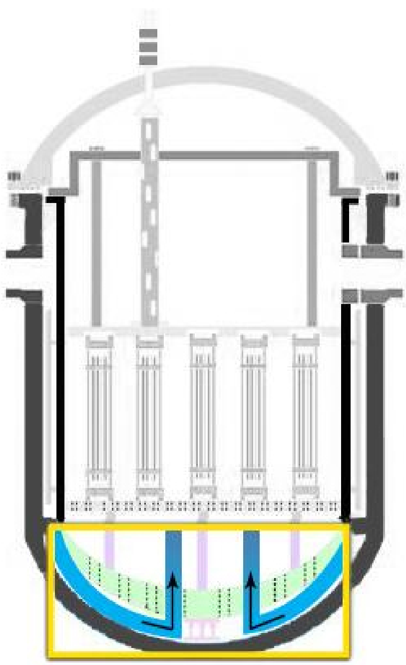
The reactor coolant (water) enters from the cold leg and leaves via the hot leg.



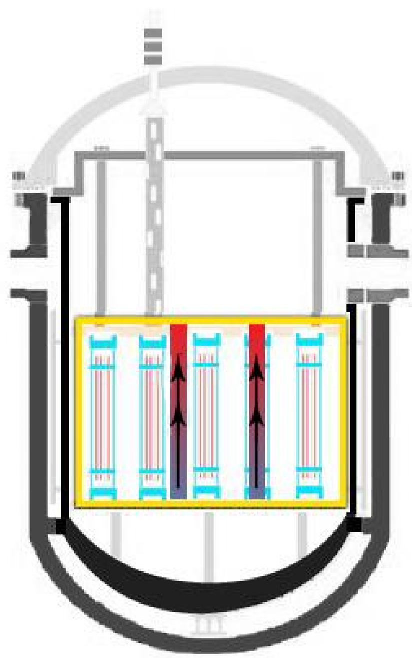
The coolant enters the reactor vessel at the inlet nozzle and hits against the core barrel.

The core barrel forces the water to flow downward in the space between the reactor vessel wall and the core barrel.



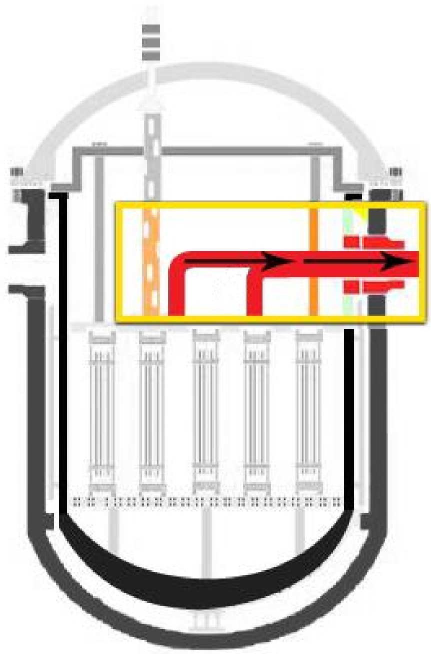


After reaching the bottom of the reactor vessel, the flow is turned upward to pass through the fuel assemblies.

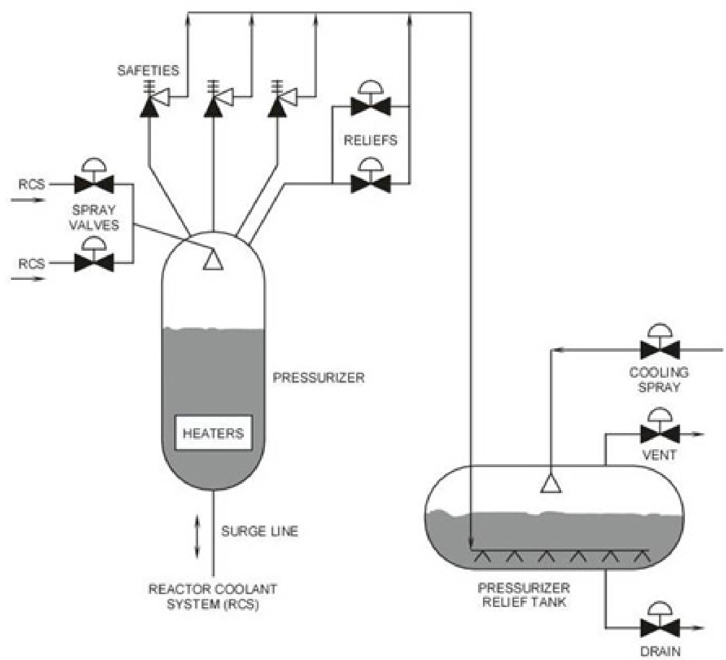


The coolant flows all around and through the fuel assemblies, removing the heat produced by the fission process.

The now hotter water enters the upper internals region, where it is routed out of the outlet nozzle and moves on to the SG.



RCS: Pressurizer (PZR)



The PZR provides a means of controlling the system pressure and reactor coolant water level. Pressure is controlled by the use of electrical heaters, pressurizer spray, power-operated relief valves, and code safety valves. Level is controlled by the charging and letdown system (CVCS).

The PZR operates with about a 50/50 mix of steam and water. Pressure deviation is normally associated with a change in the temperature of the RCS. If the temperature starts to increase, the density of the reactor coolant decreases and the water expands. Since the PZR is connected to the RCS via the surge line, the water expands into the PZR. This causes the steam in the top of the PZR to be compressed and pressure increases. The opposite effect occurs if the RCS temperature decreases. The water becomes denser and shrinks. The level in the PZR decreases, which causes a pressure decrease. For a pressure increase or decrease outside of a programmed band, the PZR components operate to bring pressure back to normal.

For example, if pressure increases above the desired set point, the spray line sprays relatively cold water from the discharge of the RCP (cold leg) into the steam space. The cold water condenses the steam into water and reduces pressure (steam takes up about six times more space than the same mass of water). If pressure continues to increase, the power operated relief valves open and dump steam to the pressurizer relief tank (PRT). If this does not relieve the pressure, the safety valves lift and discharge to the PRT. If pressure starts to decrease, the electrical heaters are energized to boil more water into steam, and therefore increase pressure. If pressure continues to decrease, and reaches a predetermined set point, the reactor protection system trips the reactor.

The PRT contains water with a nitrogen atmosphere. The water condenses any steam discharged by the safety or relief valves. Since the RCS contains hydrogen, the nitrogen atmosphere prevents the hydrogen from creating an explosive mixture.

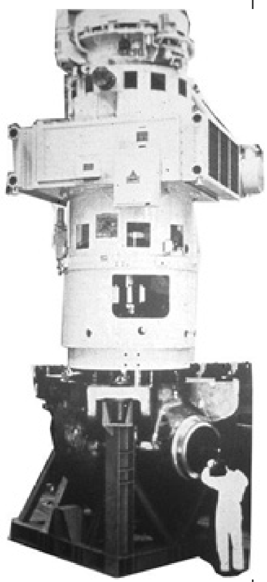
RCS: Reactor Coolant Pumps (RCPs)

PWR RCPs are very large, centrifugal pumps that push the reactor coolant through the core and SG tubes.

The discharge (outlet) of the RCP is sent through an RCS pipe called the cold leg. The coolant enters the reactor vessel and is then directed through the core to pick up heat. The coolant exits the vessel through the hot leg and is sent to the SG tubes. After giving up its heat, the coolant leaves the SG and goes back to the suction (inlet) of the RCP, where the cycle starts over again.

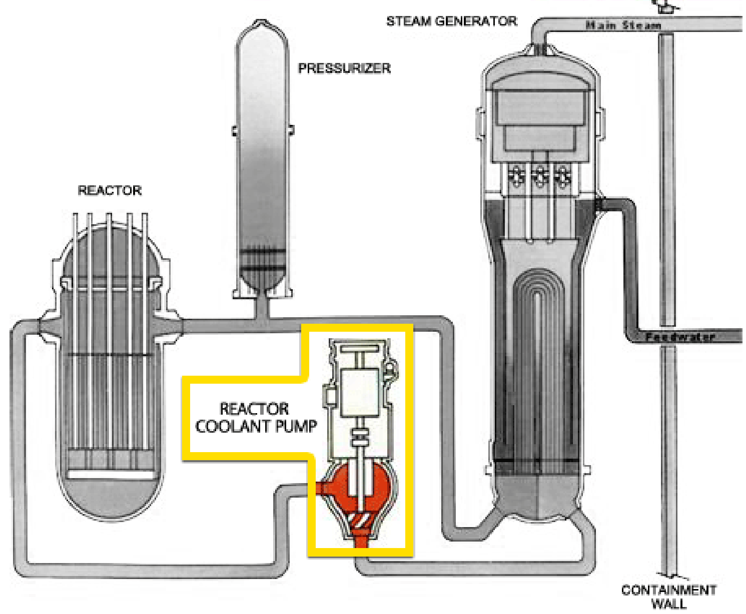
As stated, the RCPs are very large pumps. A typical RCP is characterized as:

- Upwards of 10,000 Horsepower (for comparison, a souped-up Mustang GT might produce 300 to 600 hundred horsepower)
- High voltage (6 to 13 thousand volts)
- Weighs 40 to 50 tons
- Produces around 100,000 gallons per minute of flow



A technician prepping an RCP for installation.

Due to the pump design and RCS high pressure, the pumps have special seals on their shafts to keep primary water leakage low. These seal packages have special cooling and lubricating systems. Some new PWR plant designs are moving to specially-designed RCPs that will not require seals. The issue with these seals is that they are designed to allow a small amount of leakage, so PWR RCS volumes must be constantly replenished by some system like the CVCS system covered next. In cases where the plant loses all electrical power for a long period (Station Blackout (SBO)), these seals act like small loss of coolant accidents and contribute to the probability of core uncovery and damage.



RCS: Chemical & Volume Control System (CVCS)

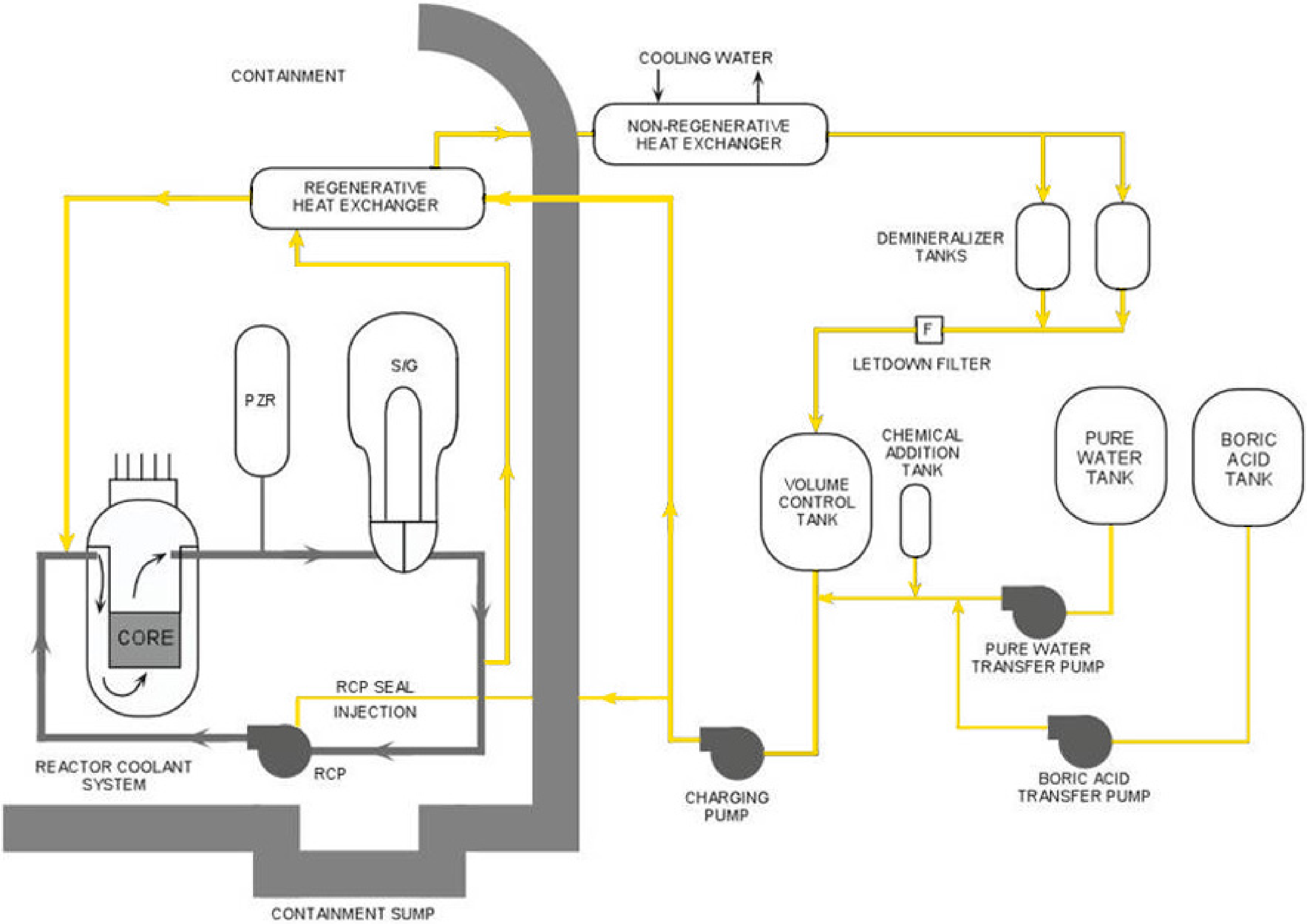
The CVCS is a major support system for the RCS. Some of its functions are:

- Purification of the RCS coolant using filters and ion exchangers/demineralizers
- Control of boron added to the coolant for neutron absorption
- Controlling the volume of water in the RCS, hence controlling the water level in the PZR

Like BWRs, PWRs routinely clean their reactor coolant. The CVCS system allows a small amount (75 gallons per minute) of RCS to be routed out of the primary into the letdown system. This letdown is depressurized and cooled down, then sent to the cleanup system and collected for re-use in a Volume Control Tank (VCT).

Operators can add various chemicals to the coolant to the suction of the charging pumps, and then send the coolant back into the RCS. Chemicals such as hydrogen gas for oxygen scavenging, boric acid for neutron absorption, and lithium hydroxide for pH control are routinely used.

Some of this clean and relatively cool charging water is sent to the RCP seals for cooling and lubrication.

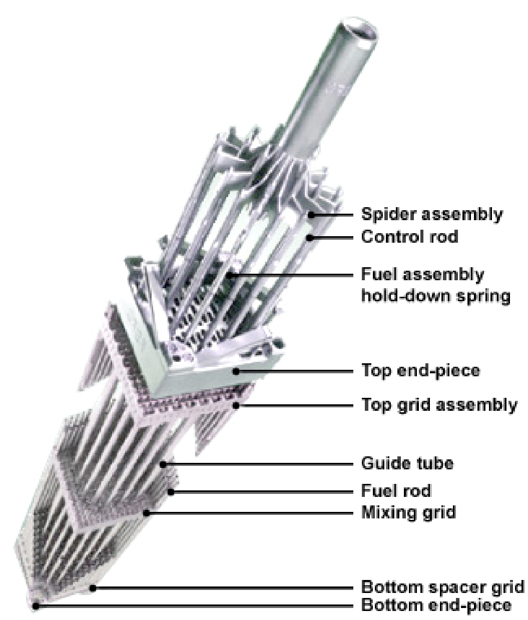


RCS: Control Rods

Recall that control rods are neutron poisons. These strong neutron absorbers are connected to a cluster assembly that can be moved in and out of the core to control the nuclear fission process. PWR control rods consist of a silver-indium-cadmium or a boron carbide mixture sealed inside stainless steel tubes, which are inserted within the fuel assembly lattice (as opposed to external to the fuel assemblies as in BWRs).

In a PWR, the control rod fingers are attached to a spider assembly. The spider provides the connection between the control rod and the control rod extension shaft. The extension shaft extends up through the vessel head where the control rod drive system can move the shaft and rod.

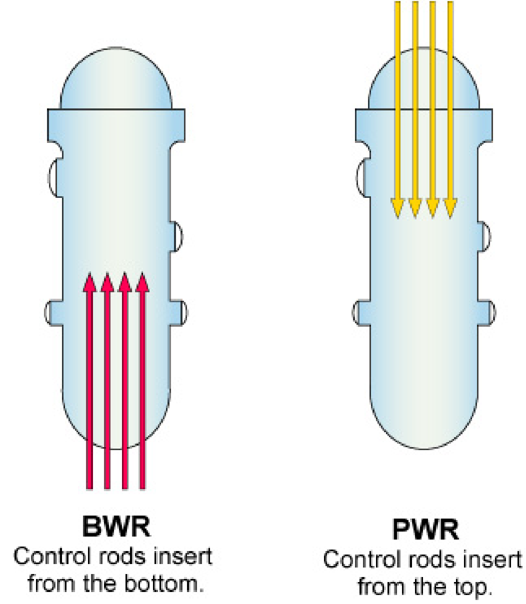
Rod Cluster Control Assembly (RCCA) is the formal name for grouped control rods and the spider. (These are generally Westinghouse terms; Babcock & Wilcox and Combustion Engineering use similar components with different terminologies.)



RCS: Control Rod Drive System

PWR control rod drive systems are very different from those of BWR plants. PWR rod drive systems penetrate the core through the upper reactor vessel head. Since PWR system pressure is so high, these rod drive systems must be inside the pressure boundary of the RCS. This requires a special method to actually move the rods.

Since there is no physical connection between the control rod and its drive system, the PWR rod control system uses magnetism to create the movement. Devices known as grippers are inside the RCS boundary and can be controlled or moved by magnetic fields. The magnetic fields are produced **outside** of the RCS boundary by the rod control system drive. If a plant trip is required, the magnetic fields are de-energized; the grippers let go of the control rod extension shaft; and the rods drop into the core, shutting down the fission process.



RCS: Residual Heat Removal

Just as with BWR plants, PWR fuel will produce significant amounts of decay heat even after PWRs are shut down. This heat must be removed to keep the core safe.

Normally, PWRs use the SGs to cool the core. The operator ‘dumps’ steam from the SG to the condenser via turbine bypass valves, or directly to the environment through atmospheric dump valves.. However, once the RCS cools down far enough, there is insufficient heat to produce much steam.

At this point, operators use the low-pressure, high-flow pumps (RHR) of the ECCS systems to remove the decay heat. Once primary pressure is reduced to less than about 200-300 pounds per square inch, operators realign the low pressure ECCS subsystem (RHR) to remove water from a RCS hot leg; pump it through RHR heat exchangers cooled by safety-related cooling systems; and then return it to the RCS via piping connected to the cold legs.

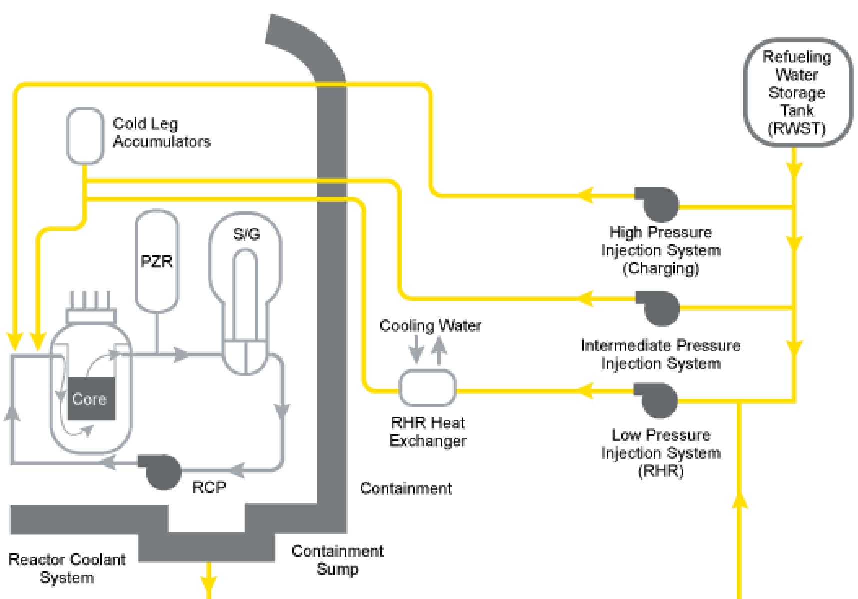
Flow through the core is maintained in its normal direction, and the decay heat can be removed to keep the core cool. The decay heat is ultimately directed out to the environment through the safety related cooling system.

Emergency Core Cooling System (ECCS)

The ECCS of a PWR is much like a BWR in that its main function is to provide cooling to the core in an emergency. However, due to design differences, a PWR must also ensure that the core remains subcritical in case of a steam leak (PWRs rely on boron, in addition to control rods, to stop the fission process in certain accidents). The ECCS subsystems accomplish this by injecting highly- borated water into the core. Recall from an earlier chapter that boron is an extreme poison for the fission process.

PWR ECCS subsystems are generally comprised of:

- High Pressure Coolant Injection (HPCI) Pumps
- Intermediate or Medium Pressure Coolant Injection (MPCI) Pumps
- Low Pressure Coolant Injection (LPCI) Pumps, also known as RHR pumps (dual duty)
- Passive Accumulators (ACC)
- Containment Spray (CS) Pumps

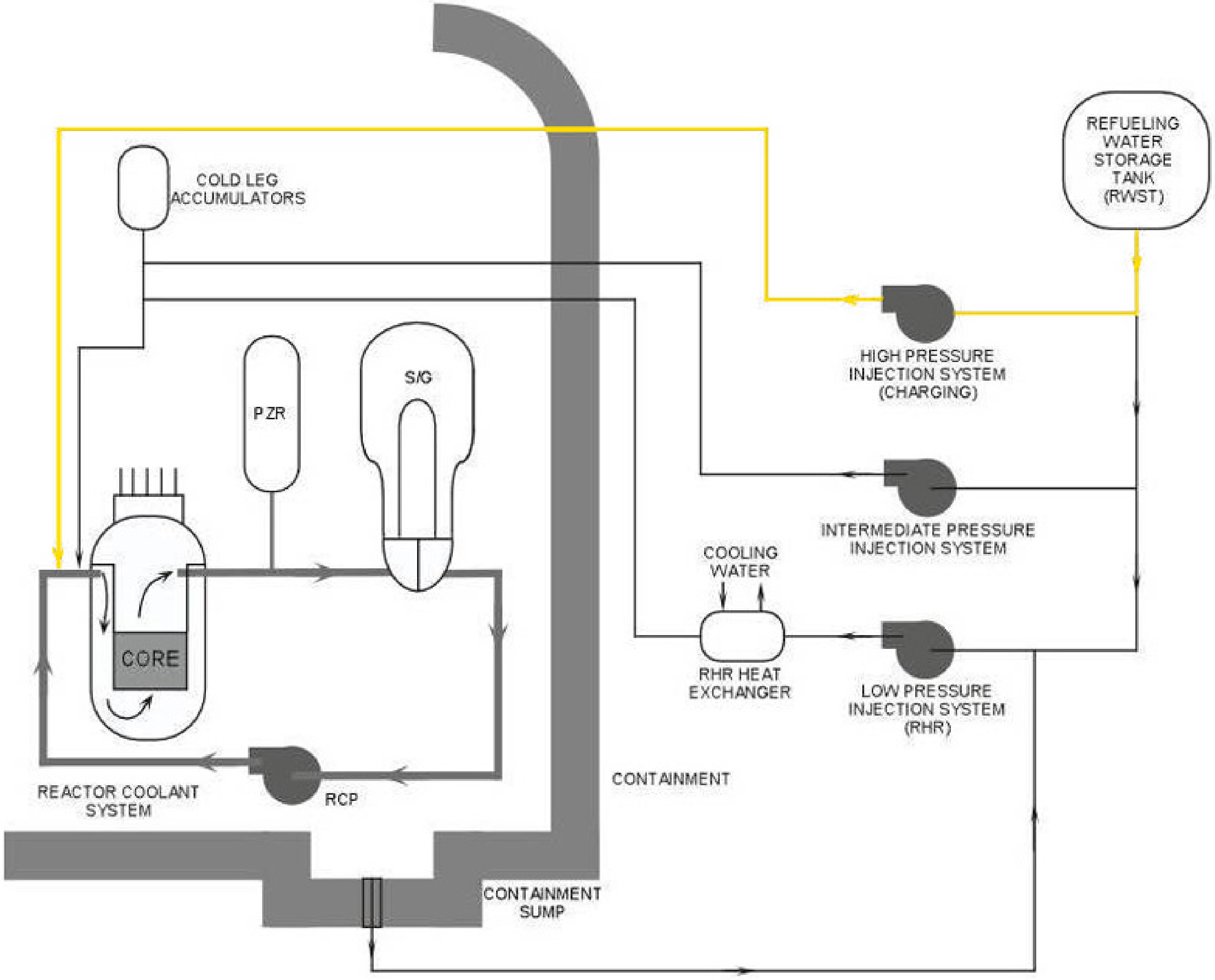


These systems are generally aligned to automatically start in certain emergencies, but they will not inject into the RCS unless pressure drops sufficiently down to their operating point. The exact use of high versus intermediate pressure pumps is design-related and varies from plant to plant.

ECCS: High Pressure Coolant Injection (HPCI)

The HPCI system generally uses the pumps in the chemical and volume control (CVCS) system. After receipt of an emergency actuation signal, the system automatically realigns to take borated water from the refueling water storage tank (RWST) and pump it into the RCS.

The HPCI system provides water to the core during emergencies in which RCS pressure remains relatively high (such as small breaks in the RCS, steam break accidents, and leaks of reactor coolant through an SG tube to the secondary side). Most HPCI systems, though, are lower flow systems.

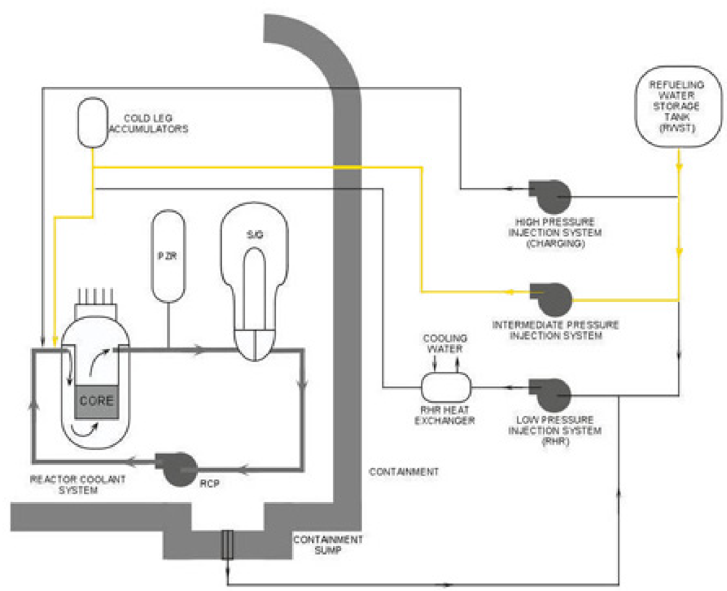


ECCS: Intermediate Pressure Injection System

The intermediate pressure injection system is also designed for emergencies in which the RCS pressure stays relatively high, such as small to intermediate size primary breaks. After an emergency start signal, the pumps take borated water from the RWST and pump it into the RCS.

Because PWR plants operate at high primary pressures, they use pumps to inject water into the RCS in an accident. These pumps require electricity; if electric power is lost, a different method of injecting water is required. A new design of PWR (AP1000) uses passive methods of ECCS that do not require electrical power.

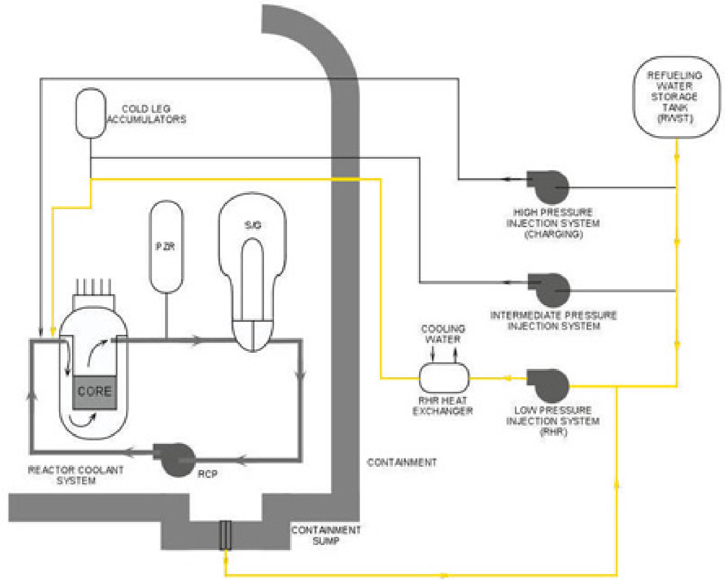
The intermediate injection pumps will produce more flow than the high pressure ones, but they will not pump against as high a pressure.



ECCS: Low Pressure Injection System

The low pressure injection system is commonly referred to as the residual heat removal, or RHR, system. Some designs refer to it as the Low Pressure Injection System (LPSI). During power operations, it will be lined up to automatically inject water from the RWST into the RCS during large breaks that cause a very low RCS pressure.

In addition, the RHR system has a feature that allows it to take water from the containment sump, pump it through the RHR system heat exchanger for cooling, and then send the cooled water back to the reactor for core cooling. This method of cooling would be used when the RWST empties after a large primary system break. It is called the long-term core cooling or recirculation mode.



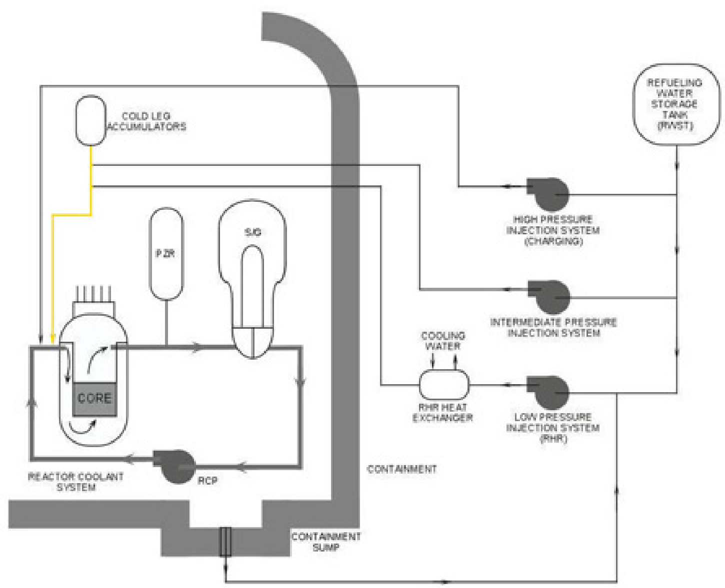
ECCS: Passive Accumulators (ACC)

The cold leg accumulators, also called passive accumulators, do not require electrical power to operate. These tanks provide borated water to the RCS during emergencies in which the RCS pressure drops very rapidly, such as with large primary breaks.

The ACCs contain large amounts of borated water with a pressurized nitrogen gas bubble in the top. If the primary pressure of the RCS drops lower than the nitrogen gas pressure in the accumulators, then the nitrogen forces the borated water out of the tank and into the RCS and core.

Since PWR plants normally operate at high primary pressures, they use electric pumps to inject water into the RCS in an accident. The ACCs provide a PASSIVE method of injecting water if electric power is lost.

Use of ACCs is not a permanent solution to a large primary break. PWR plants have emergency diesel generators for times where electric power is lost. The ACCs allow core cooling during the nominal 10 to 30 seconds it takes for the diesel generators to start and provide power to ECCS pumps.

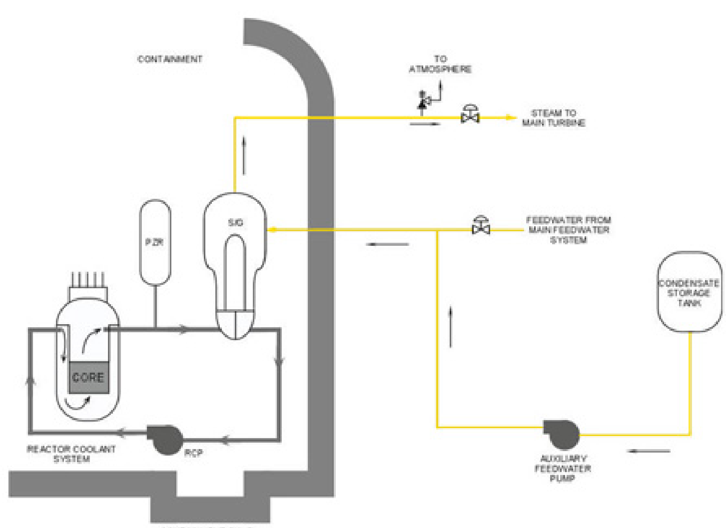


Auxiliary Feedwater (AFW) System

In a PWR, the preferred method of removing reactor core heat is by controlled releasing of steam from the SG. In order to ensure that steam can be produced, there must be a reliable way of getting feedwater into the SG. The AFW system is a safety-related system that can inject clean and cool feedwater into the SGs in an emergency, ensuring a viable method of removing core heat.

While the AFW system is not a true ECCS system, it is the preferred method for the operator to use to add feedwater to the SG in order to remove core heat in an emergency.

Plants usually have a large tank of ultra-pure water available for the AFW system. The AFW system draws water from the tank and adds it to the SG. However, for plants that do not have the safety-related condensate storage tanks, they will fill the SG from the safety related service water system. The operators then send the steam to the condenser for reclamation, or in an emergency, the steam is sent out the roof via atmospheric dump valves, thus removing the decay heat.



Containment System (CONT)

PWR and BWR plants must have a containment building. However, because of the high pressure of the PWR primary and the relatively larger quantity of water in systems inside the containment, the PWR containment building must be stronger than that of a BWR.

PWR containment buildings are generally very large. Sometimes they contain over 1 million cubic feet of internal volume. The walls are reinforced concrete more than 3 feet thick. They are reinforced with metal lattices of 2 to 3 inch rebar. Some PWR containment designs are a little different: with sub-atmospheric pressures (slight vacuum) or with ice condensers inside (steam from a leak contacts ice with boron to reduce heat and pressure).

PWR containments also have tendons running through them. Containment tendons are very large, metal-wire bundles almost 6 inches thick that run through the walls. These tendons are stretched using special hydraulic tensioners so that the tendon squeezes the walls of the containment inward. These pre-tensioned cables allow the containment to withstand much higher internal pressures than one made with only rebar. The graphic shown is the outside of the metal containment LINER of a PWR. The rebar and concrete has not installed yet.

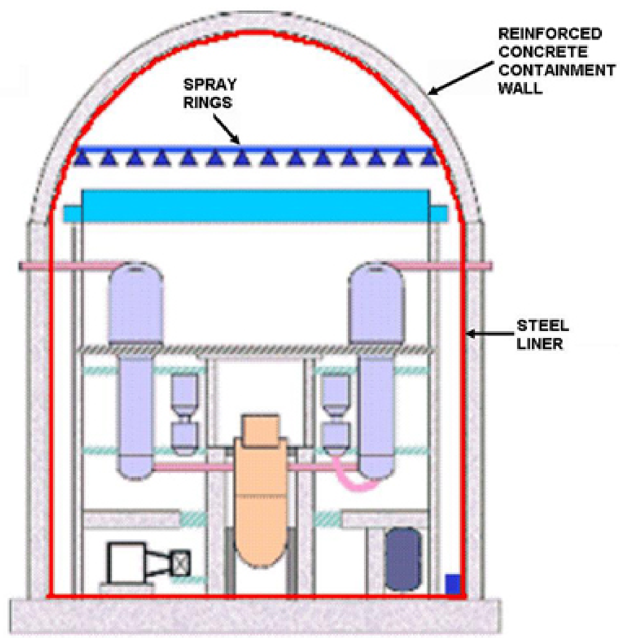
Imagine how much more pressure a can of Coke can withstand if you wrap something around it that is squeezing the walls **inward**.

Containment

PWR containments are designed to withstand the pressures and temperatures that would accompany release of a high energy fluid (i.e., the primary coolant, steam, or feedwater). However, the exposure to high temperature and pressure over a long period of time could degrade the concrete.

If a break occurred in the primary system, the released coolant could contain fission products. If the concrete developed any cracks, the high pressure in the containment would force the radioactive material through the relatively porous concrete containment wall, and into the environment.

To limit leakage following an accident, a steel liner covers the **inside** surface of the containment building. This liner acts as a vapor- and liquid-proof membrane to prevent any gas from escaping through cracks that may develop in the concrete. Some newer design PWR containments have a space, or annulus between this liner and the concrete wall. In this design, the liner provides both the pressure and leakage boundary, while the concrete shell provides protection from various external hazards.



Process Instrumentation & Control (I&C) Systems

PWRs also include extensive process I&C systems. As in a BWR, the process I&C systems allow the operators to effectively monitor the reactor processes and all support processes, and make corrections to the system when needed. Many of the systems are similar to those in the BWR, but there are a few differences.

I&C subsystems include:

▶ Reactor protection system (RPS)

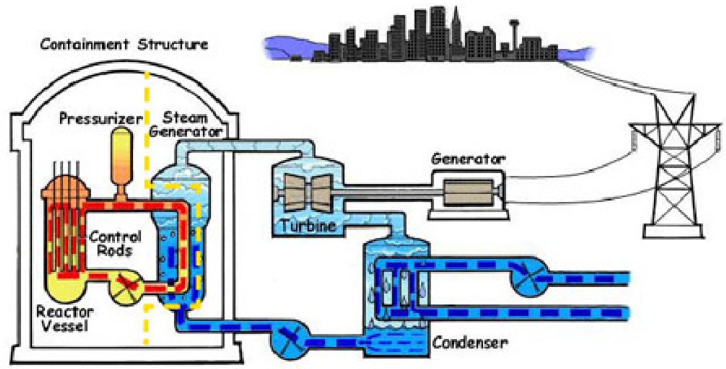
▶ Neutron monitoring

▶ SG water level control

▶ Pressurizer pressure and level control

Secondary Systems (SEC)

Recall that with a PWR, the SG is the dividing line between the primary and secondary systems. Specifically, the tubes inside the SG constitute the dividing line between primary and secondary.

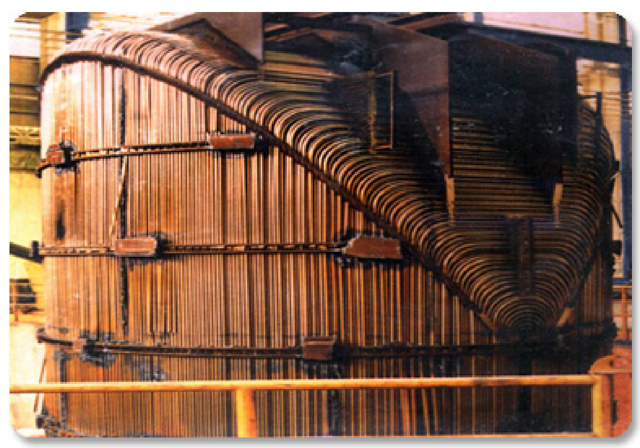


The subsystems in the secondary plant are:

- Steam Generator
- Main Steam
- Turbine
- Condenser

Another important point to remember when discussing the SEC of a PWR is that the steam that turns the turbine does **not** come in direct contact with the reactor core. Therefore, because the steam used throughout the entire SEC does not come into contact with the core it has a lower chance of contaminating the plant.

SEC: Steam Generator (SG)



In a PWR, the steam generator (SG) is the focal point for separating the primary system from the secondary system.

The SG transfers the heat of the reactor core to relatively cool feedwater, allowing it to boil and produce steam for turning the main turbine. As such, the SG is the primary method that operators rely on to remove reactor heat—including decay heat.

Hot coolant exiting the core is directed into thousands of tubes in the SG. This hot coolant gives up its heat across the tube walls to the feedwater on the secondary shell (outside) side of the tubes. The pressure in the shell side is equivalent to that in a BWR, so the feedwater is allowed to boil and become steam.

After giving up its heat, the now-cooler reactor coolant is pumped back into the reactor vessel by RCPs, where it is reheated and the cycle begins again.

There is a moisture separation system in the top area of the SG much like that in the top of a BWR vessel. Steam is able to make the twisting path through the various dryers, but the water cannot and it drains back down to mix with the incoming feedwater.

Most PWR SGs are U-tube types; a limited number are straight-tube, or once-through, types (B&W).

SEC: Main Steam System

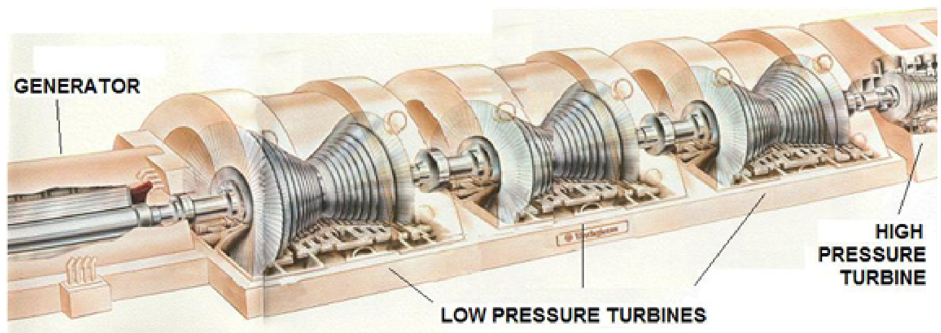
The main steam system of a PWR is almost exactly like that of a BWR. Steam from the SG is directed to the main turbine and to certain other loads (i.e., the feedwater pump turbines, steam dumps, etc.). The steam pipes are generally pretty large and are insulated. Unlike BWRs, PWR turbines are not shielded.

In an emergency, since PWR steam does not come into direct contact with the core, the operator can release or dump steam directly into the environment through valves known as atmospheric dump valves. Normally, the operator prefers to dump steam to the condensers to recover the water and contain any potential radioactivity that might be in the PWR steam.

Main steam can also be used to drive safety-related auxiliary feedwater turbines, providing the operator with an emergency method of pumping feedwater back into the SG. As long as the plant has a way of filling the SG with feed and sending its steam somewhere, the core can be kept cool.

SEC: Turbine

The turbine is a big, fan-like component that is turned by steam produced by the SG. It takes the thermal energy of the steam and converts it to motion in order to drive the electrical generator. Most power plants use a high pressure turbine, driven by main steam, and coupled to two or three low pressure turbines that are driven by dried and re-heated exhaust steam from the high pressure turbine. There is virtually no difference between PWR and BWR turbines, other than shielding concerns with the BWR.



SEC: Condensate System

Except for the obvious difference that BWR feedwater is sent directly to the core while PWR feedwater is not, the other aspects of the feedwater and condensate systems are virtually identical.

Turbine exhaust steam is collected in large shells known as condensers. The steam is on the outside of thousands of tubes that carry cool circulating water. The hot exhaust steam is cooled and converts back into liquid water that collects at the bottom of the condenser. The liquid is referred to as condensate.

The condensate is then pressurized and heated before it is returned to the secondary side of the SG where it is reheated and boiled. Condensate and feedwater pumps are used to increase the pressure so that the condensate can re-enter the SG. Feedwater heaters, using steam collected from various parts of the secondary plant, warm the condensate and feedwater in stages. By the time the feedwater re-enters the SG, it is ready to boil with little additional heat required.

Summary

Key Points about System Basics

- With a PWR
  - The steam that turns the turbine does **not** come in direct contact with the reactor core.
  - The water in the reactor core is kept at a very high pressure so that it does **not** boil in the core.
- The division between the primary systems and the secondary systems in a PWR is more concrete than in a BWR. In a PWR, the tubes inside the SG constitute the dividing line between primary and secondary systems. The primary systems have direct contact with, or offer direct support to, the core. The secondary systems provides support to the turbine, condenser, and feedwater for the steam generator.
- The three primary functions of the systems in a PWR are:
  - Steam production
  - Electrical generation
  - Steam exhaust

Key Points about Subsystems

- PWR plants have the same basic functional systems as BWR plants. The subsystems of those systems vary based on the design and/or manufacturer. A functional system may include subsystems that are in the primary system, the secondary system, or both. The basic functional systems of a PWR include:
  - Reactor Coolant System (RCS)
  - Emergency Core Cooling System (ECCS)
  - Containment System (CONT)
  - Process Instrumentation & Control (I&C) Systems
  - Secondary Systems (SEC)

Key Points about the RCS

- The reactor coolant system (RSC) allows the operators to maintain and control optimal temperature in the reactor core. Subsystems of the RCS include:
  - Reactor Vessel & Core
  - Reactor Coolant Pumps (RCPs)
  - Chemical & Volume Control System (CVCS)
  - Control Rods and Control Rod Drive System

Key Points about the ECCS

- The main function of the ECCS is to provide cooling to the core in an emergency. Unlike a BWR, a PWR must also ensure that the core remains sufficiently subcritical (fission reaction shutdown) in case of a steam leak. The ECCS subsystems accomplish this by injecting highly borated water into the core.

- PWR ECCS subsystems are generally comprised of:
  - High Pressure Coolant Injection (HPCI) Pumps
  - Intermediate or Medium Pressure Coolant Injection (MPCI) Pumps
  - Low Pressure Coolant Injection (LPCI) Pumps, also known as RHR pumps (dual duty)
  - Passive Accumulators (ACC)
  - Containment Spray (CS) Pumps
- These systems automatically start in emergencies, but will not inject into the RCS unless pressure drops to specific, low enough levels.

Key Points about Containment

- The high pressure of the PWR primary system and the relatively large quantity of high-energy water in systems inside the containment require the containment building to be stronger than the containment of a BWR.
- PWR containment structures include:
  - Concrete with an internal lattice of 2-3" rebar
  - Large, tensioned tendons comprised of bundles of metal cables (squeezing the containment walls inward)
  - Steel inner liner for leak-tightness (no appreciable impact on strength of containment)

Key Points about I&C

- PWRs, like BWRs, include extensive process I&C systems that facilitate monitoring and corrections to the system.
- I&C subsystems include:
  - Reactor protection system (RPS)
  - Neutron monitoring
  - SG water level control
  - PZR pressure and level control

Key Points about Secondary Systems (SEC)

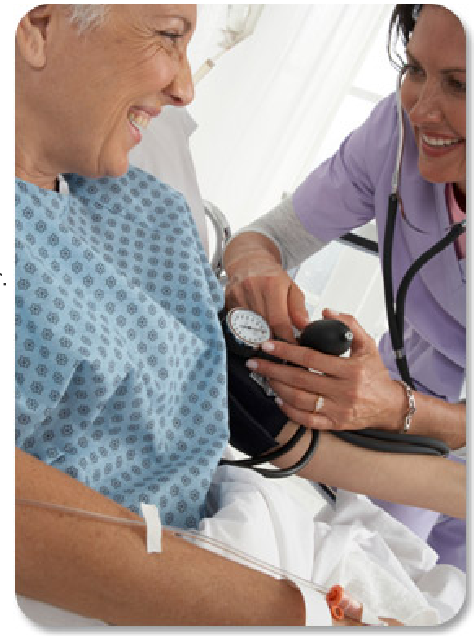
- With a PWR, the SG is the dividing line between the primary and secondary systems. Specifically, the tubes inside the SG constitute the dividing line between primary and secondary systems.
- The key subsystems of the secondary system in a PWR are:
  - SG
  - Main Steam
  - Turbine
  - Condenser

Health Physics & Radiation Protection

Chapter Overview

The effects of radiation on the human body can be both mysterious and frightening to the general public. As an employee of the NRC—no matter your role—family, friends, and members of the public are likely to ask questions related to the possible health impacts of nuclear power energy. This chapter provides an overview of health physics (HP), also commonly termed Radiation Safety, as it relates to the operation of commercial nuclear power plants and other fuel cycle facilities in the United States (U.S.), and how these facilities relate to other sources of radiation that the general population is likely to encounter.

In this chapter, we will consider terms and concepts related to radiation and health; radiation doses and the resulting effects of those doses on human health; precautions that lead to doses that are as low as is reasonably achievable (ALARA); and sources of radiation that the average American is likely to encounter.



Objectives

After completing this chapter, you will be able to:

- Define key terms related to health physics and radiation protection.
- Differentiate between absorbed dose (rad/gray) and dose equivalent (rem/sievert).
- Differentiate between the effects of acute and chronic radiation doses.
- Discuss the ionizing radiation exposure limits by group.
- Identify the three methods of minimizing radiation exposure.
- Identify the sources of ionizing radiation in people.

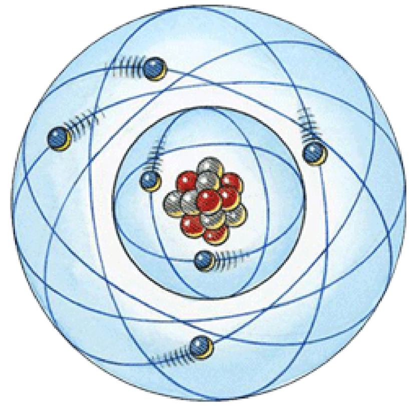
Estimated time to complete this chapter:

60 minutes (1 hour)

Terms for Health Physics

Radioactive Material

Recall our review of atomic theory from Chapter 2: Fission Process & Heat Production. During that discussion, we explored how an atom is either stable or unstable. Unstable atoms can attempt to become stable by emitting energy (photons) and/or particles These emissions are termed as ionizing radiation. Radioactive atoms emit that energy in the form of particulate (has mass) or non-particulate (photons) ionizing radiation. The radiation is called 'ionizing' since the interaction of the radiation with the atoms of exposed substances can have electrons stripped away, leaving behind ions.

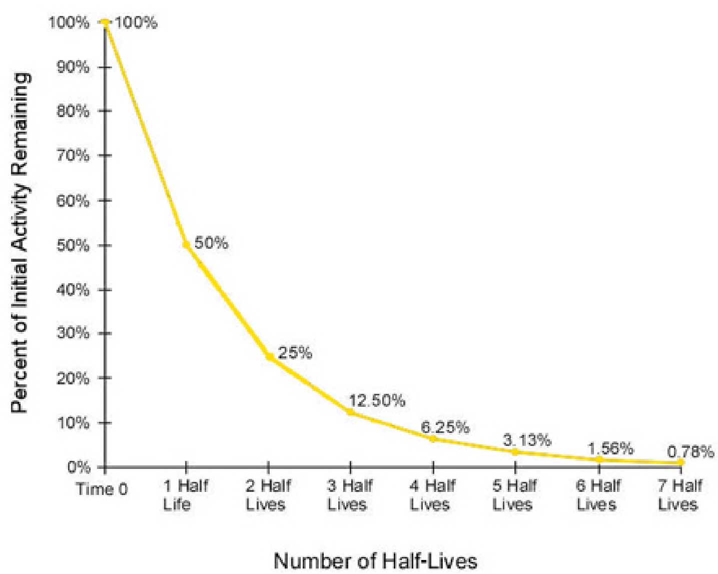


When the particles or photons are being emitted, the unstable atom is undergoing a process called radioactive decay. Each decay event is called a disintegration or transformation.

The particles and/or electromagnetic energy emitted by radioactive material as it decays or disintegrates are called ionizing radiation. This radiation comes in several forms and can cause health effects when it interacts with living tissues. It can also change the physical characteristics of non-living materials (e.g., embrittlement of metals).

Half-Life

Half-life measures a material's rate of nuclear or radioactive decay. The half-life of any radioactive material is the amount of time it takes for one half of the atoms of a radioactive material to decay to a new form. During each half-life, one half of the unstable atoms that started during that half-life period will decay. The 'new' form of the decay product can be radioactive or can be stable (non-radioactive).



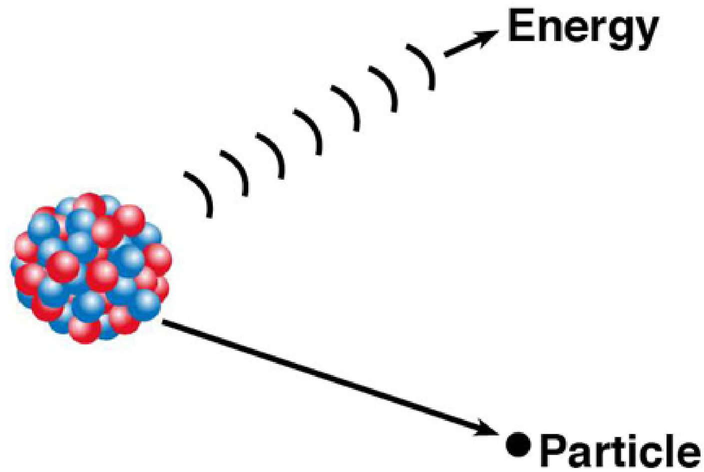
Half-lives range from millionths of a second for highly radioactive fission products to billions of years for long-lived materials, such as naturally occurring uranium. No matter how long or short the half-life is, after about seven half-lives have elapsed, less than 1 percent of the initial activity remains. However, some radionuclides will decay to other radioactive forms, so the overall radioactivity can linger.

Radionuclides have a distinct half-life that does not vary for that particular isotope. For example, Carbon-14 has a half-life of about 5730 years; that half-life will not change.

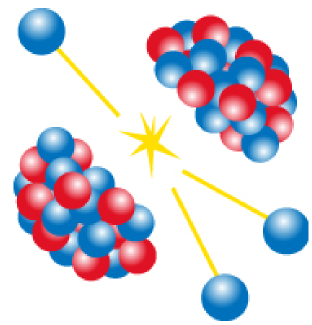
Ionizing Radiation

Ionization is the process of stripping, knocking off, or otherwise removing electrons from their orbital paths, thus creating free electrons and leaving behind charged atoms (ions). The radiation emitted by radioactive material can produce ionizations; this is called ionizing radiation.

The negatively-charged electrons and positively-charged atoms may interact with atoms in other materials to produce chemical changes in the material where the interactions occur. If these chemical changes occur in the cells of our bodies, cellular damage may result. The biological effects of radiation exposure vary widely and will be discussed in a later topic.



Particulate Radiation



As mentioned earlier, radioactive material may emit particles, electromagnetic energy in the form of photons, or both. Particulate radiation is ionizing radiation that has mass.

Particulate radiation can present both an internal and external hazard. Internal and external hazards refer to whether the radiation is created inside or outside of the absorbing body. If you sit next to a radioactive substance giving off gamma rays, the radiation hitting you is EXTERNAL. If you consume or absorb the emitter, the radiation interacting with your body is INTERNAL.

The types of particulate radiation are:

▶ Alpha

▶ Beta

▶ Neutron

Non-Particulate Radiation

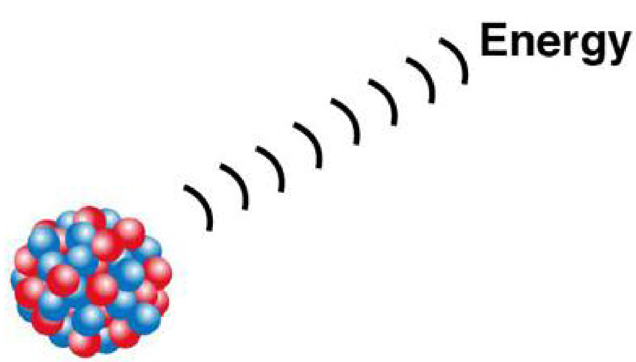
Non-particulate radiation is radiation emitted as electromagnetic energy in the form of photons. Even though photons are considered to exhibit wave-particle duality, they are considered to be without mass.

Non-particulate radiation, like particulate, presents both an INTERNAL and EXTERNAL hazard.

The types of non-particulate radiation are:

▶ Gamma

▶ X-Ray



Traditional/Special Units and the Système Internationale d’Unites (SI)

In the US, we use the **traditional** or **special** system when expressing radiation units. The international community uses the **SI** system. Current NRC documents include both traditional and SI units.

Traditional/Special units:	**SI units:**
• curie	• becquerel
• rad	• gray
• rem	• sievert

When discussing dose, an easy way to remember which system's unit correlates with the other system is:

- Absorbed dose: rad and gray both have an “A”
- Dose Equivalent: rem and sievert both have an “E”

Occasionally you may encounter the unit roentgen. A roentgen is a unit of measurement specifically for the exposure of photons in air. Although this unit is not found in 10 CFR Part 20, it is used in other NRC (e.g., Regulatory Guides) and non-NRC documents. Also, it is a common unit found on many radiation survey instruments used by licensees. Because the roentgen is used for measuring photon exposure, a roentgen is approximately equal to both a rad and a rem.

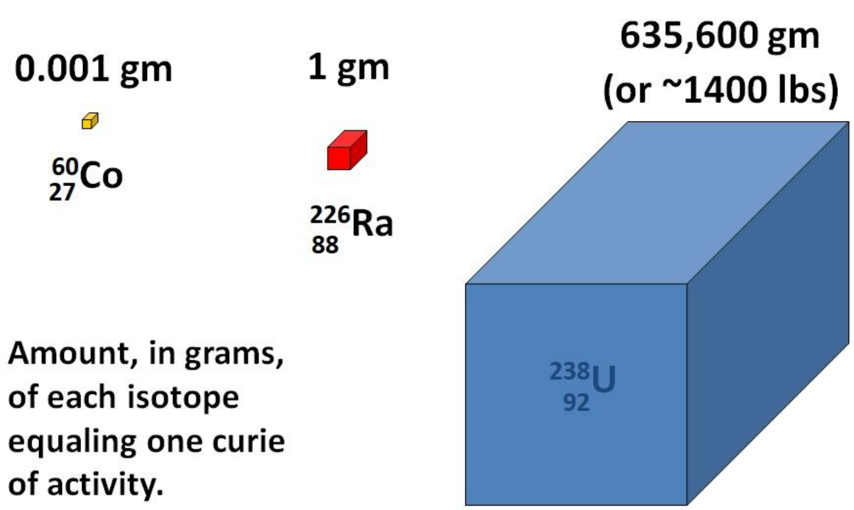
Measuring Radioactivity

Radioactivity is a term that indicates how many radioactive atoms are disintegrating, or decaying, in a time period. It is measured in units of curies (Ci) or becquerels (Bq). USNRC regulations (10 CFR Part 20) now include both units of measure; older records may only include Ci.

Curies

Curies is the unit of measure traditionally used in the US. One curie is the amount of any radioactive material that will decay at a rate of 37 billion disintegrations per second. It is based on the disintegration rate of 1 gram of radium-226 (one of the radionuclides used by Madame Curie in her experimentation).

The amount of material necessary for 1 ci of radioactivity can vary dramatically. For example, 1 ci of cobalt-60 is an amount smaller than a grain of salt, while 1 ci of uranium-238 is more than half a ton of material. So never assume that high amounts of radioactivity requires large amounts of actual material.



Becquerels

Becquerels is the international unit of measure (SI). The unit of measure is equal to one disintegration per second. Since 1 Ci of radioactive materials emits 37 billion disintegrations per second, then 1 Ci is equal to 37 billion Bq.

Measuring Dose

The special unit used for radiation dose measurements in the US is the rem; the SI unit is the sievert. You may also hear about absorbed dose, known as rad.

Absorbed Dose (rad)

The unit rad only quantifies the amount of energy deposited in a given medium. When radiation energy is deposited in living tissue, radiation absorbed dose does not sufficiently describe the potential for biological damage that can result. To accurately describe potential biological damage that can result from radiation interacting in living tissue, the concept of dose equivalent must be used. Dose equivalent is based on the fact that different types of radiation have different potentials for causing damage to tissue, even if each type of radiation enters tissue with the same amount of energy

Dose Equivalent (rem)

The rem is a measure of the potential biological damage that may be caused by ionization radiation interacting with human body tissue. It is a term that measures equivalent dose and equals the biological damage that may be caused by one rad of dose. The rem accounts for the fact that some types of radiation are more biologically damaging than others. That is, the biological damage from one rad deposited by beta radiation is potentially less than that caused by one rad of alpha radiation.

The rem is approximately equal to the dose in rad multiplied by a unit-less quality factor (Q), which allows conversion from absorbed dose to dose equivalent for a particular radiation type.

Betas, gammas, and X-rays have a Q of 1. Neutrons of unknown energy have a Q of 10, while alpha particles have a Q of 20.

Alpha particles, remember, are an internal hazard only. They cannot penetrate very far into the surface of substances, even past the outer layer of your skin.

Comparing the potential for biological damage

The following show how using the quality factor converts rad to rem:

- Non-particulate Radiation
  - Gamma rays: 1 rad = 1 rem
  - X-Rays: 1 rad = 1 rem
- Particulate Radiation
  - Beta particles: 1 rad = 1 rem
  - Neutron particles: 1 rad = 10 rem
  - Alpha particles: 1 rad = 20 rem

Radiation Doses

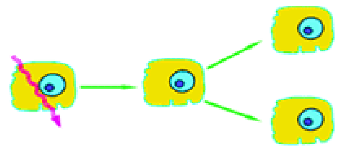
Effects of Radiation on Cells

People tend to think of biological effects in terms of the effect of radiation on living cells. In actuality, ionizing radiation interacts only with atoms within the cells, but this can result in effects on cells.

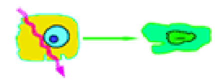
Thus, all biological damage effects begin with the consequence of radiation interactions with the atoms forming the cells. As a result, radiation effects on humans proceed from the lowest to the highest levels. Atoms change molecules, which may affect cells, which in turn may affect tissue, then organs, and finally, the whole body.

Luckily, cells have a tremendous ability to repair damage. As a result, not all radiation effects are irreversible. In many instances, the cells are able to completely repair any damage and function normally.

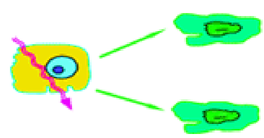
If the damage is severe enough, then the affected cell dies. In other instances, the cell is damaged but is still able to reproduce. However, the daughter cells may be lacking in some critical life-sustaining component and may die.



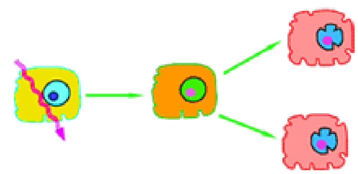
Normal Repair of Damage



Cell Dies from Damage



Daughter Cells Die



No Repair or Non-Identical Repair Before Reproduction

The final possible result of radiation exposure is that the cell is affected (genetic mutation) but does not die. The mutated cell may reproduce and perpetuate the mutation. Most mutations are inconsequential, but some result in tumors that could be benign or malignant.

While radiation is sometimes used in medical treatments, the treatment itself can cause damage to tissues of the body. The picture shows the radiation erythema (skin reddening) suffered by a patient receiving radiation treatment. Skin effects may mirror what we commonly know as sunburn, but can be far more damaging.

Mutations

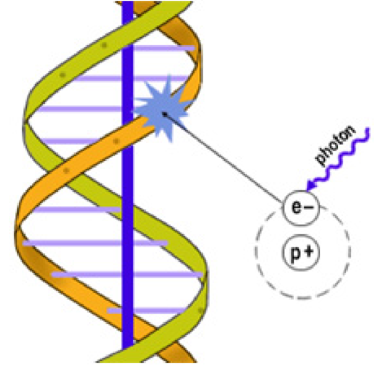
Direct and indirect radiation effects are only associated with DNA and not any other cell component.

A direct radiation effect occurs when the radiation interacts with the atoms of the DNA or other cell component's molecules. A direct effect can inhibit or change the ability of the cell to reproduce and/or survive.

If enough atoms are affected such that the chromosomes do not replicate properly, or if there is significant alteration in the information carried by the DNA molecule, then the cell may be destroyed by direct interference with the DNA that regulates its life-sustaining system.

An indirect radiation effect can occur if the ionization creates or results in a chemical change to the water molecules within the cell due to the formation of free radicals (ions) or other ionization byproducts.

There have not been any observation in humans of mutations caused by ionizing radiation that are not also caused by other mutant effects (like smoking, chemical exposure, spontaneous mutations, etc.). In other words, the Hollywood depiction of Dr. David Banner becoming The Hulk due to exposure to gamma rays is just that: HOLLYWOOD!

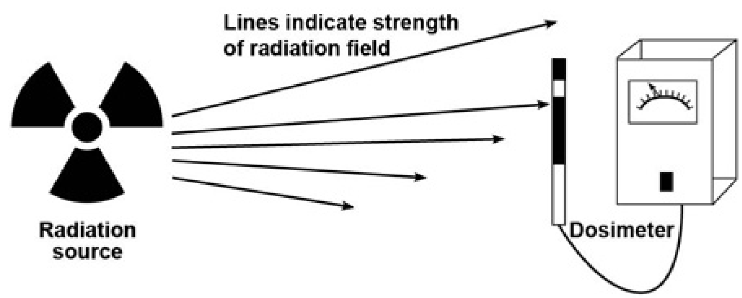


Calculating Dose

Recall that the quality factor (Q) converts the absorbed dose (rad) to the dose equivalent (rem). The table shows the relative damage based on the Q value for the given radiation types. These are average values; neutrons, for example, have a range based on factors such as their energy.

Energy deposition & resulting dose equivalent

Absorbed Dose	Dose Equivalent
1 rad - Gamma	1 rem
1 rad - Beta	1 rem
1 rad - Neutron (unknown energy)	10 rem
1 rad Alpha (internal)	20 rem



Calculating Dose: Dose Rate

The dose rate is the rate at which a person would or did receive a radiation dose or dose equivalent. It is a measure of radiation dose intensity, or strength, in some unit time period. The amount of dose received depends on the:

1. Strength of the radiation field
2. Amount of time exposed to the radiation

Calculating Dose

The dose received is equal to the dose rate multiplied by the length of time spent in that field.

Dose = Dose Rate X Time

For example, if Jane spent 1/2 hour in a location with a dose rate of 50 mrem/hr, Jane's dose would be 25 mrem. Let's look at the calculation.

Dose = Dose Rate x Time

= 50 mrem/hr X 1/2 hour

= 25 mrem

Calculating Dose: Stay Time

Stay time is an important calculation for those working within radiation areas. It is the length of time an individual may remain in a radiation field of some strength before exceeding a pre-defined dose limit (administrative limit).

Stay Time = Dose Limit ÷ Dose Rate

Let's consider Jane again. Jane has a predetermined dose limit of 100 mrem. The dose rate where Jane is working is 50 mrem/hr. How long can Jane work in that location?

Stay Time = Dose Limit ÷ Dose Rate

= 100 mrem ÷ (50mrem/hr)

= 2 hours

Jane's stay time at the current work location is 2 hours, as long as the dose rate doesn't change.

Types of Doses

Radiation doses are typically divided into two categories: acute and chronic. Time and intensity of exposure define the effects of the two categories.

Acute

Acute doses of radiation occur over short periods of time. Low acute doses are typical for many radiation workers in the nuclear industry. There are no known effects, signs or symptoms of low acute doses, but an increased risk of cancer is assumed. Collectively, low acute doses are referred to as chronic doses and will be discussed further in the next section. High acute doses may kill cells or cause irreparable damage to the DNA such that the risk of the cell becoming cancerous is high. High acute doses to a large portion of the body can kill so many cells that it affects the function of some of the body's tissues and organs. If this happens, a whole body response called Acute Radiation Syndrome (ARS) may occur. In the workplace, acute doses that are high enough to cause ARS are always associated with accidents. For example, in 1991, a worker at a Belarus irradiator facility received an acute whole body dose of at least 1100 rad (11 gray) which resulted in ARS and ultimately his death 113 days later despite extensive medical intervention. The cause of the very high acute dose was the worker bypassing the safety features of the irradiator to enter the irradiation chamber while the source was exposed. Victims of radiation poisoning and survivors of the atomic bombs dropped over Hiroshima and Nagasaki all showed the symptoms of ARS as a result of the acute doses they received. High acute doses to patients undergoing therapeutic medical treatments have also caused ARS and death.

Chronic

As mentioned before, occupational workers generally receive what is called chronic doses. Chronic doses can be received daily over the period of a worker's shift, or a worker may perform a task that results in a low acute dose that is added to the worker's chronic dose. These doses are all monitored to ensure they are maintained as low as is reasonably achievable and within regulatory limits. Because chronic doses are low and well regulated, there are no immediate effects on the body's tissues or organs. The only effect is an acceptable increase in the risk of cancer to the worker. If a cancer occurs as a result of the chronic dose, it would not be observed for many years. Note: All occupational radiation workers are required by regulation to be informed of the risks of exposure to ionizing radiation.

Acute Radiation Syndrome

Acute Radiation Syndrome (ARS) occurs when a large, short-term, dose of radiation simultaneously damages a number of vital tissues and organs. ARS symptoms will only be presented if the acute dose is to at least a large portion of the whole body. ARS effects will vary from person to person depending on factors like age, health, and magnitude of dose. ARS has three (3) distinct syndromes which are discussed further below.

After an initial acute exposure of around 25 to 100 rad, the initial signs and symptoms of any ARS syndrome are:

- Nausea
- Vomiting
- Fatigue
- High Temperature
- Blood changes

These symptoms, which mirror those produced by a common viral infection, may be the only outward indication of radiation exposure.

ARS: Manifestation

As the dose increases above 100 RAD, one of the three radiation syndromes begins to manifest itself, depending on the level of the dose.

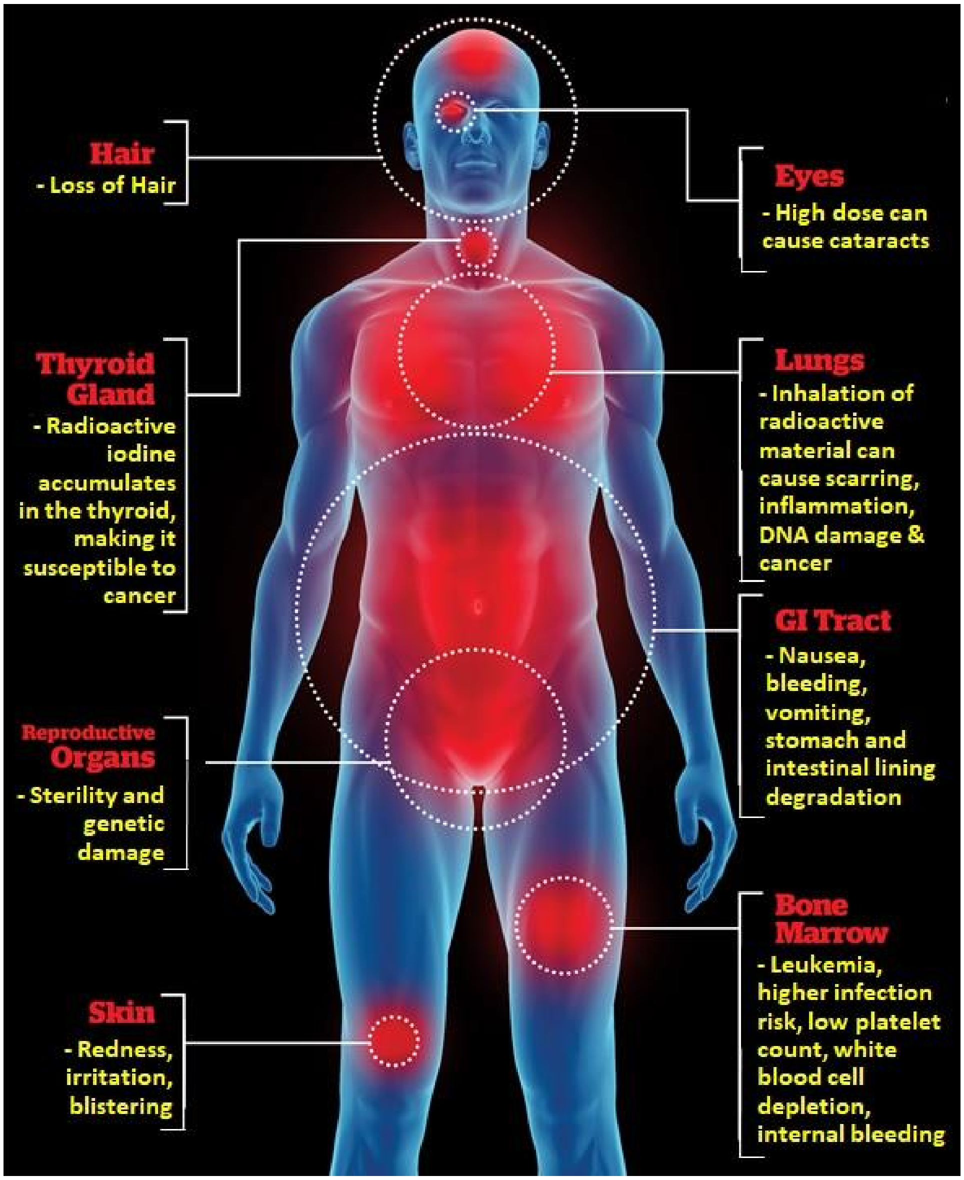
Recall that the three types of ARS are:

▶ Hematopoietic

▶ Gastrointestinal

▶ Central Nervous System

The graphic below shows different parts of the body and potential effects of an acute dose on that body part.



High Dose Effects

Every acute exposure does **not** result in death. If a group of people is exposed to a whole body penetrating radiation dose, the effects in the table might be observed. The information for this table was extracted from NCRP Report No. 98, Guidance on Radiation Received in Space Activities, 1989.

In the table, the threshold values are the doses at which the effect is **first** observed in the **most sensitive** of the individuals exposed. For example, with a dose of 100 RAD, the most sensitive of people exposed will begin to vomit.

The LD 50/60 is the lethal dose at which 50% of those exposed to that dose will die within 60 days.

It is sometimes difficult to understand why some people die while others survive after being exposed to the same radiation dose. The main reasons are the health of the individuals at the time of the exposure and their ability to combat the incidental effects of radiation exposure, such as the increased susceptibility to infections.

Dose in rad and corresponding observable effects for exposure of 1 hour or less

Dose (rad)	Effect Observed
15 – 25	Blood count changes in a group of people
50	Blood count changes in an individual
> 50	Sterility in males
100	Vomiting (threshold)
150	Death (threshold)
320 – 360	LD 50/60 with minimal medical care
480 – 540	LD 50/60 with supportive medical care
1,100	LD 50/60 with intensive medical care and bone marrow transplant

Low Dose Effects

There are two general categories of effects resulting from exposure to low doses of radiation. They are genetic effects and somatic effects. Keep in mind, however, that high doses can cause the same effects.

A GENETIC effect is suffered by the offspring of the individual exposed. The genetic information in the DNA can be altered or lost through many different interactions with chemical and physical (like ionizing radiation) agents. At present, there have been no **observed** genetic effects to human beings from ionizing radiation. One way of looking at this is that the off-spring of groups exposed to larger amounts ionizing radiation (e.g., surviving atomic bomb blast victims) have not appreciably changed in physical terms over subsequent generations. While there may have been higher incidents, for example, of cancers among those off-spring, they remain biologically 'identical' to their ancestors.

A SOMATIC effect is suffered by the individual exposed. Cancer in the exposed individual is an example of a somatic effect; therefore it is also considered a carcinogenic effect.

It is commonly mistaken to consider in-utero (unborn fetus/embryo) effects to be a **genetic** consequence of radiation exposure, because the exposure occurs prior to birth and the effect is seen after birth. However, this is actually a special case of the somatic effect since the embryo/fetus is the individual exposed to the radiation (in addition to the mother). It is the exposure of the embryo/fetus that results in the manifestation of this somatic effect, not the result of DNA changes inherited from the mother. This is also termed a teratogenic effect, or an agent or factor which causes malformation of an embryo.

NRC Dose Limits: Members of the Public

The NRC limits the dose that workers at power plants/civilian nuclear facilities can receive as part of their job. In addition, NRC limits the dose that members of the public can receive from exposure to, the handling of, and the use of radioactive or byproduct materials. The limits are:

- Less than 2 millirem in any 1 hour from external radiation sources in any unrestricted area
- Less than 100 millirem in a calendar year from both external and internal sources of radiation in unrestricted and controlled areas

Additionally, the NRC has provided design objectives for power reactor licensees to keep offsite doses as far below the 10 CFR Part 20 limits as is reasonably achievable. Reactor Power Plant Operating Criteria can be found in 10 CFR Part 50.

Permissible dose levels in unrestricted areas during the transport of radioactive material can be found in 10 CFR Part 71.

The Environmental Protection Agency (EPA) also has established various limits for power reactors discharging into local effluent streams. If you are interested in more information on these EPA limits, then look at 40 CFR Part 190 for the specifics.

NRC Dose Limits: Occupational Workers

The NRC exposure limits apply to all NRC licensees' occupational workers and are designed to ensure that:

1. No worker at a nuclear facility receives an acute radiation exposure sufficient to result in deterministic effects (e.g., cataracts).
2. The NRC radiation dose limits in 10 CFR Part 20 were established by the NRC based on the recommendations of the ICRP and the NCRP as endorsed by the Federal radiation protection guidance developed by the EPA. The limits were recommended by the ICRP and NCRP with the objective of ensuring that working in a radiation-related industry was as safe as working in other comparable industries.

No matter the limit number, licensees are also required by 10 CFR Part 20 to keep radiation exposures as low as is reasonably achievable (ALARA).

Degree of exposure and corresponding annual dose limits

Type/Location of Dose	Annual Limit
Whole Body (total effective dose equivalent, TEDE) - internal & external dose	5 rem*
Skin of the Whole Body	50 rem*
Skin of Extremities (shallow dose equivalent, SDE) - includes elbows and arms below the elbows and knees and legs below the knees	50 rem
Any individual organ or tissue other than the lens of the eye	50 rem
Lens of the Eye (lens dose equivalent, LDE)	15 rem
Minors (worker less than 18 years old) - 10% of annual dose limits for adult workers	500 mrem
Declared Pregnant Woman (DPW)	500 mrem**
Embryo/Fetus (during entire pregnancy)	500 mrem**

***The whole body and skin of the whole body includes all of the body except the elbows and arms below the elbows, and knees and legs below the knees.**

****The limit applies over the gestation period (pregnancy) rather than a year.**

Your Dose History

If you are interested in getting a copy of your dose history, you can submit an 'Automated Dose History Request Form' through the REIRS program via the NRC Home page at ***https://www.reirs.com***.

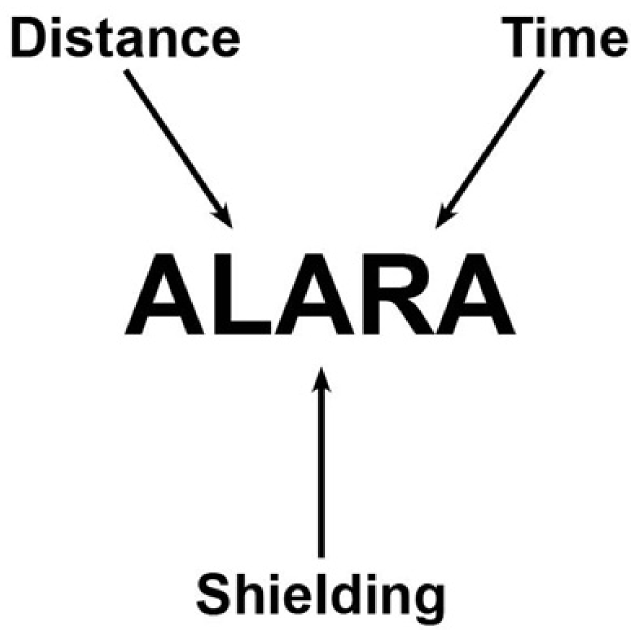
ALARA

Dose and ALARA

Recall that no matter the dose limit established by the NRC, licensees are required to keep radiation exposures AS LOW AS IS REASONABLY ACHIEVABLE (ALARA). The regulation outlines the standards for protection against radiation for nuclear workers. The goal of ALARA is that all exposure be **As Low As Is Reasonably Achievable**.

ALARA Achievement Methods

As defined in 10 CFR 20.1003, ALARA is an acronym for "as low as (is) reasonably achievable," which means making every reasonable effort to maintain exposures to ionizing radiation as far below the dose limits as practical, consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest. 20.1101 also discusses ALARA for members of the public.



Three protective measures are routinely used to reduce the dose from any external source of radiation. The three methods are:

- Minimizing time exposed to the radiation source
- Maximizing distance from the radiation source
- Maximizing shielding from the radiation source

Time and distance can also be applicable for reducing the internal dose from radioactive material. However, once the radioactive material is inside the body, it will continue to cause internal dose until it is either removed via the body's physiological processes or decays to a negligible activity. For some radionuclides and situations, medical interventions can be considered such as the use of chelating agents that can remove a percentage of certain radionuclides from the body or possibly surgical removal of tissue with imbedded radioactive material.

The total dose is the sum of internal and external doses. Total dose should be minimized, since overall risk is proportional to the total dose. In some cases, this may mean accepting a small intake of radioactive material to reduce the external dose.

The important thing is to keep the total dose As Low As Is Reasonably Achievable (ALARA).

ALARA Achievement Method: Time

The dose a person receives from external radiation is directly proportional to the length of time spent in a radiation field. Minimizing the amount of time spent in a radiation field minimizes the dose received. Strategies that can minimize time spent in a radiation field are:

- Plan and rehearse the job under realistic conditions
- Know the exact location of work prior to entering the radiation area
- Ensure all necessary tools are available at the job location
- Establish good communications
- Do not loiter in the area (establish Low Dose Waiting Areas)

Similarly, minimizing the time spent in an area with airborne radioactivity will minimize the internal dose. Since the total intake of radioactive material is proportional to the time spent breathing in the radionuclides, breathing it in for a shorter period of time reduces the eventual internal dose.

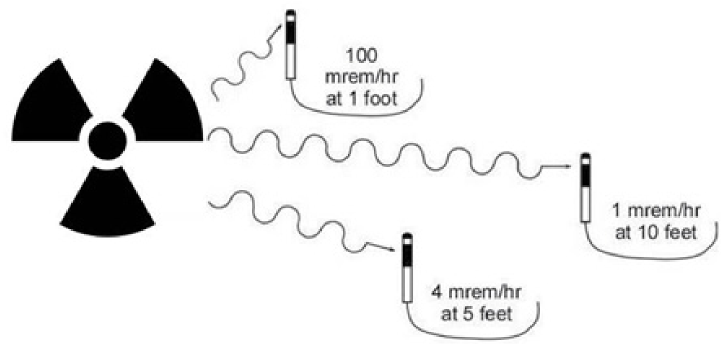
ALARA Achievement Method: Distance

The radiation dose from sources can be significantly reduced by applying the protective measure of distance.

The dose a person receives from a point source of external radiation is inversely proportional to the square of the distance from the source.

The farther you are away, basically, the lower your dose. Methods for increasing distance:

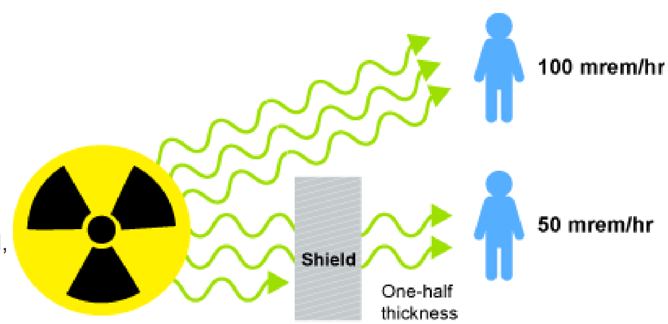
- Using extension tools
- Utilizing remote operating stations
- Staying away from hot spots



Staying as far away as possible from a source of airborne radioactivity minimizes the intake of radioactivity because the activity disperses and becomes less concentrated as it moves away from the point of release.

ALARA Achievement Methods: Shielding

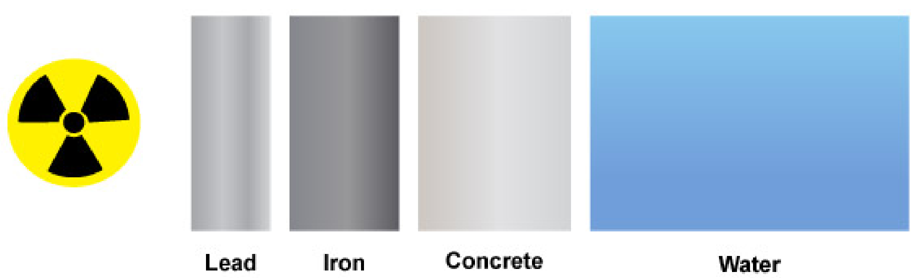
Shielding is one of the most effective means of reducing radiation exposure. A shield simply absorbs the radiation before it can affect anything else, and shielding materials differ in their ability to absorb radiation.



Shielding Materials

Common shielding materials include lead, iron or steel, concrete, and water. In general, to have the same gamma radiation exposure level at the outside of each material, it takes more iron than lead, more concrete than iron, and more water than concrete.

The exact amount of material required depends upon the energy of the radiation (e.g., gamma ray) that is being shielded against.



Shielding Types

By locating the shielding as close as possible to the source, dose rates can be reduced in a large area, and thus reduce the dose to many workers.

The two major types of shielding at the plant are:

▶ Installed shielding

▶ Temporary shielding

Contamination

Contamination generally refers to a quantity of radioactive material in a location where it is neither intended nor desired. Radioactive contamination can be fission products that have escaped the system or structure that would normally contain them, or activated corrosion products that do the same thing. Radioactive contamination can be wet or dry, fixed or removable, and settled or airborne. Since radioactive contamination is radioactive material, ionizing radiation is emitted by the contamination.

Contaminated Areas

A contaminated area is an area that contains radioactive contamination. Some examples of contaminated areas that require periodic access would be the primary side of the steam generator for a pressurized water reactor and the main turbine for a boiling water reactor.

Protective Measures

Protective measures are used to prevent, detect, and/or contain radioactive contamination. Since radioactive contamination can be inhaled and/or ingested, the protective measures are considered to be methods of protection against both internal and external doses.

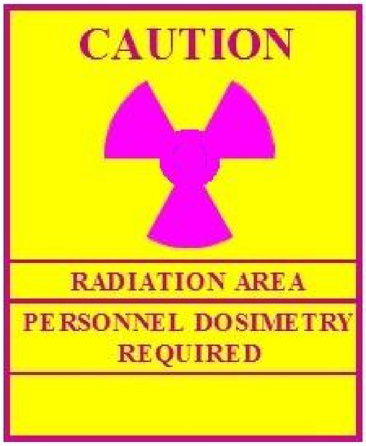
Common protective measures include:

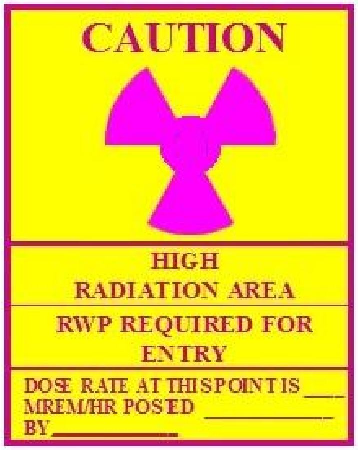
- Utilize containments
- Maintain access control
- Conduct frequent surveys
- Utilize protective clothing
- Wear respiratory protection
- Practice good housekeeping
- Conduct follow-up bioassays
- Minimize radioactive leakage

Signs and Labels

Radiation signs and labels are required to be used to warn people of radiation areas, contaminated areas, and locations where radioactive material is found. The use of these signs is governed by 10 CFR 20.

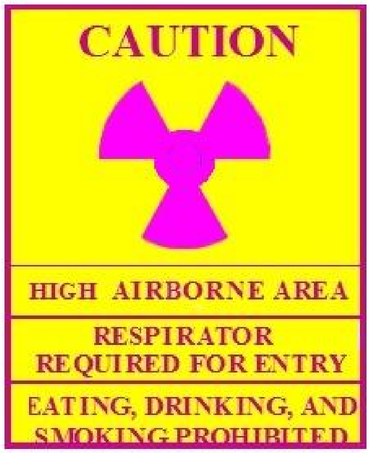
Signs in the US are magenta (purple) on a yellow background; caution signs in the US may have either magenta or black on the yellow background. The international symbol for radioactive material and radiation is a magenta (or black) three-bladed design on a yellow background.

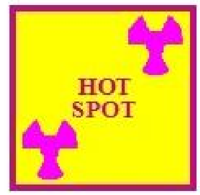

CAUTION
RADIATION AREA
PERSONNEL DOSIMETRY REQUIRED


CAUTION
HIGH RADIATION AREA
RWP REQUIRED FOR ENTRY
DOSE RATE AT THIS POINT IS ____ MREM/HR POSTED ____________ BY____________


GRAVE DANGER
VERY HIGH RADIATION AREA


CAUTION
CONTAMINATED AREA
PROTECTIVE CLOTHING REQUIRED FOR ENTRY
EATING, DRINKING, AND SMOKING PROHIBITED


CAUTION
HIGH AIRBORNE AREA
RESPIRATOR REQUIRED FOR ENTRY
EATING, DRINKING, AND SMOKING PROHIBITED


HOT SPOT


CAUTION
RADIOACTIVE MATERIAL

Sources of Radiation Exposure

Background Radiation

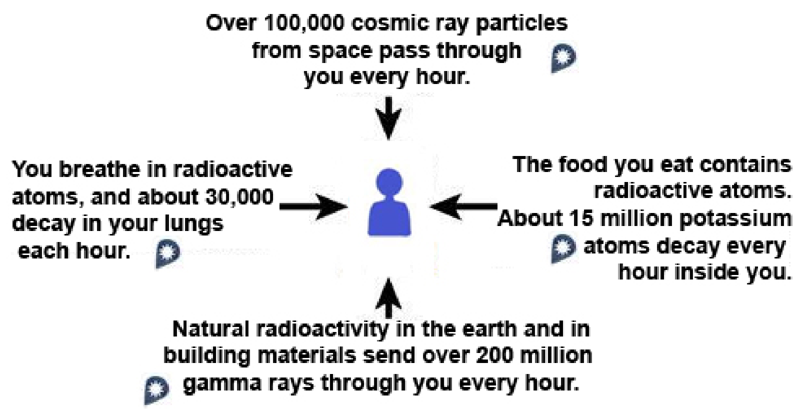
Background radiation is radiation that people are exposed to as part of living on the Earth; it does not include occupational radiation from working in the nuclear field. It is important to remember that radioactivity is a naturally-occurring process, as well as a human-made one.

Natural background radiation comes from:

- Cosmic sources
- Naturally-occurring radioactive materials
- Internal radiation

Human-made background radiation comes from:

- Atmospheric fallout from the testing of nuclear weapons
- Building and other consumer materials
- Medical procedures/diagnostics



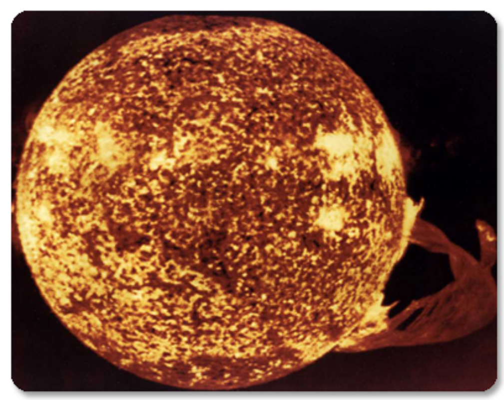
Over 100,000 cosmic ray particles from space pass through you every hour.

The two strongest sources of natural background radiation are cosmic radiation and terrestrial sources of naturally occurring radioactive materials.

Cosmic Radiation

The Earth is constantly bombarded by radiation from space, similar to a steady drizzle of rain. Charged particles from the Sun and stars interact with the Earth's atmosphere and magnetic field to produce a shower of radiation. The dose from cosmic radiation varies in different parts of the world due to differences in elevation and to the effects of the Earth's magnetic field.

The atmosphere serves as a shield reducing the amount of cosmic radiation reaching the Earth. At higher altitude, there is less shielding so the level of cosmic radiation is higher.



Terrestrial Radiation

Radioactive material is also found throughout nature. It is found in the soil, water, and vegetation. Low levels of uranium and thorium (and their decay products) are found everywhere. People regularly ingest low levels of radioactive materials with food and water. Other materials, such as radon, can be inhaled. This leads to all living creatures on Earth having some level of natural radioactive material in their bodies, which results in dose to themselves and others around them.

The dose from terrestrial sources varies in different parts of the world. Locations with higher concentrations of uranium and thorium in the soil have higher dose levels. Living near some granite rock formations can result in exposures of 25 to 100 mrem/year. Working in buildings made of stone can also contribute to higher background doses.

The major radionuclides of concern for terrestrial radiation are uranium and thorium and their decay products (e.g., radium and radon).

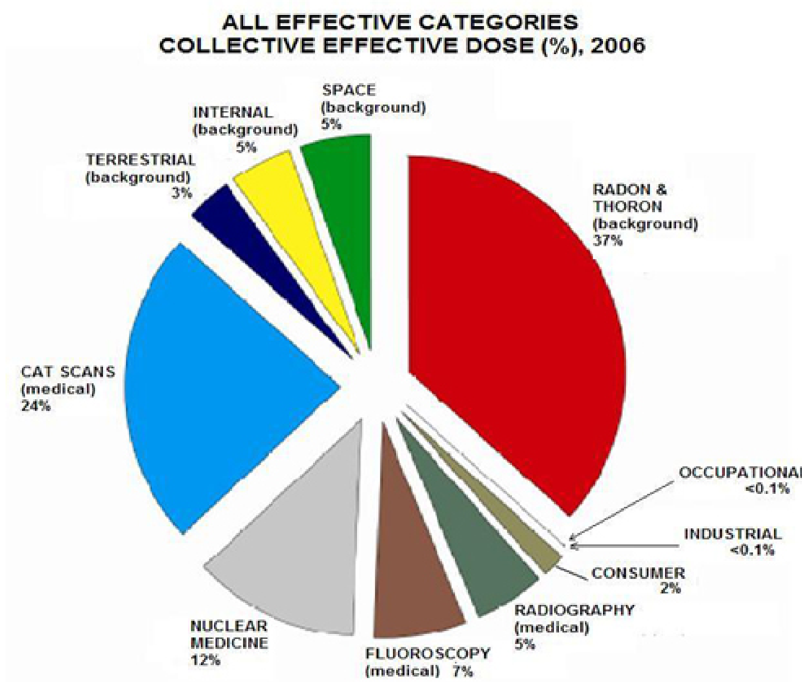
In addition, certain foods consumed regularly can contribute to your dose. Bananas contain radioactive potassium (K-40); consumption of large quantities of bananas, while not dangerous for dose, will provide easily detectable increases in the radionuclides detected in your tissues.

Medical Doses

In recent years, a significant change to the average U.S. dose is the large increase in ionizing radiation received from medical treatments. Up until the 1990's, the average American received approximately 1 mrem per day from natural sources (about 360 mrem per year).

Recent studies show the average American's dose from natural/background radiation is almost 2 mrem per day (approximately 620 mrem per year). The great majority of this increase is from medical and diagnostic exposure (X-rays, CAT scans, etc.).

The annual dose to the average American as of 2006 is about 50% natural background and 50% man-made (mostly medical). If a person doesn't receive any medical exposure, their background dose levels will more closely resemble the previously established 1 mrem per day range.



Nuclear Power Plants

Now that we've looked at some natural or human-made sources of ionizing radiation, let's consider sources associated with nuclear power plants.

Nuclear workers are the individuals most likely to receive a dose of radiation from a nuclear power plant. The sources of ionizing radiation at nuclear power plants include:

- Enriched uranium fuel decay (very low levels)
- Fission process
- Fission product decay
- Activation product decay

In addition, both nuclear workers and members of the public can be exposed to radiation from the nuclear fuel cycle, which includes the entire sequence from mining and milling of uranium to the actual production of power at a nuclear plant. This would primarily be uranium and its daughter products, but other fuel cycle radionuclides can create exposures.

Nuclear Fuel Decay

Uranium-238 (about 96% of the fuel) and Uranium-235 (the remaining 4%) are naturally radioactive and decay by the emission of alpha particles and gamma rays into daughter products. Beta particles are also emitted in the fuel as the daughter products decay toward a stable isotope of lead.

Since the fuel is sealed in airtight fuel rods, there will be no alpha or beta radiation emitted from the fuel rods unless there is some fuel rod damage.

Even though the new fuel bundles are closely inspected and stored prior to use, the natural decay process of the fuel is not a major contributor to a worker's dose at the power plants. This is because the radiation levels associated with new fuel that has not been operated in the reactor core are very low, approximately 1 to 2 mrem/hour on contact.

Fission Process and Fission Products

During the fission process, uranium atoms are split and produce fission products. High-energy gamma rays and high-speed neutrons are released during and immediately following the fission process. Since neutrons and gamma rays can travel long distances in air, very high radiation levels are present in the vicinity of the reactor vessel during power operation.

The fission process is not a major contributor to a worker's dose at the power plants. This is because the fission process is occurring in the reactor core, which is contained in the reactor vessel. The reactor vessel is surrounded by a biological shield wall inside the containment, and workers are not normally allowed around the reactor vessel during operation.

The fission products are radioactive and most of them will decay rapidly. However, several have very long half-lives and decay slowly. By design, the decay of the fission products generally occurs within the reactor vessel, and is not a significant contributor to the radiation dose of workers.

The most likely time that workers could be exposed to the fission product radiation is during refueling of the core and operations around the spent fuel pool. However, refueling and spent fuel moves are performed under water to limit the radiation dose the workers receive.

Since a fission product release could seriously jeopardize public health and safety and the environment, fission product barriers are part of every power reactor design. Recall from Chapter 1 that the fission product barriers are the fuel rod cladding, the reactor coolant system (RCS), and the containment building.

Neutron Activation

Impurities in the RCS and the reactor coolant water absorb some of the neutrons produced during the fission process. This neutron absorption can change a stable isotope into an unstable (radioactive) isotope. Even the atoms of the WATER molecules in the coolant can be activated: hydrogen to tritium and oxygen to a radioactive nitrogen isotope, N-16. This process is called activation and the radioactive isotopes formed are called activation products.

Activation products are located in the RCS and are easily transported to any support system that connects to the RCS. Activation products are the source of most radioactive contamination at nuclear power plants and are also the source of most occupational radiation exposure at the plants.

Deposits of the activation products or any other impurities on RCS surfaces are called crud. Prior to going into a refueling outage, some plants add a chemical to the RCS to force the crud off the surfaces, and then use the reactor water cleanup system to remove the material from the coolant. This reduces the risk of exposure to workers during the outage.

The table shows some of the fission products produced by fission or the activation products produced by neutron absorption. The first five isotopes are fission products and the remaining four are activation products. These materials are of interest because of their:

- Relatively long half-life
- Relatively large abundance in the reactor

Fission products with their radiation type and half-life

Radioisotope	Radiation	Half-life
Krypton-85	Beta/Gamma	10 years
Strontium-90	Beta	29 years
Iodine-131	Beta/Gamma	8 days

- Ability to chemically interact in biological systems

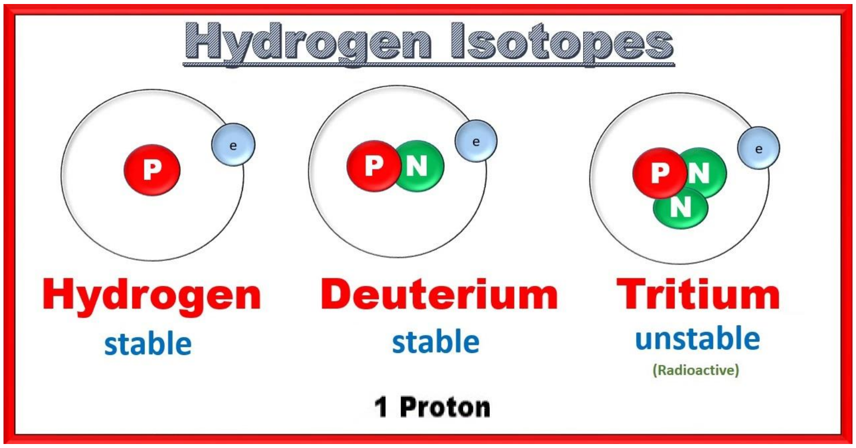
Also of interest are the aforementioned activation products from water. The two most significant are tritium and nitrogen-16. Tritium is a hydrogen isotope from activation of a hydrogen atom in water. Nitrogen-16 is from the activation of oxygen in the water molecule.

Radioisotope	Radiation	Half-life
Cesium-137	Beta/Gamma	30 years
Carbon-14	Beta	5730 years
Zinc-65	Beta/Gamma	244 days
Cobalt-60	Beta/Gamma	5 years
Iron-59	Beta/Gamma	45 days
Tritium (H-3)	Beta	12 years

Tritium

Tritium is both an activation product from water and, to a far lesser degree, a fission product. It is an environmental concern because it can leak from the plant, enter the food chain, and can replace the hydrogen in the water that makes up cells, resulting in an internal dose. Specific release limits on tritium can be found in 10 CFR 20.

Because tritium is actually a hydrogen atom that is part of the water molecule it is difficult or impossible for the radioactive waste handling systems to remove it from the water processed for release from the plant.



Nitrogen-16

Of extreme importance is the isotope Nitrogen-16 (N-16). This isotope has a very short half-life (about seven seconds), but emits a highly energetic gamma ray.

N-16 is formed when an oxygen-16 atom absorbs a neutron and decays. Since every molecule of water has an oxygen atom, there is a large amount of N-16 produced in the moderator (water) flowing through the core.

N-16 is a major concern for shielding due to the high energy of the gamma ray emitted. Also, any system (BWR Main Steam system) that contains primary coolant and exits containment must be of concern. One method of minimizing the radiation from N-16 is to allow the flow of coolant to circulate in a loop for a time period that permits the N-16 to decay, or by slowing down the flow to allow the decay. About a 1 minute delay is sufficient.

N-16 Activation & Decay

$$^{1}n + {}^{16}O \rightarrow {}^{1}p + {}^{16}N \quad \text{(activation reaction)}$$

$$^{16}N \rightarrow {}^{16}O + \beta + \gamma \quad \text{(radioactive decay)}$$

$$t_{1/2} = 7.1 \text{ sec.} \qquad \gamma = 5 \text{ MeV (very strong)}$$

Summary

Key Points

- Half-life measures the rate of a material's nuclear decay. It is the amount of time it takes for half of a radioactive material to decay.
- Ionizing radiation is radiation that produces ionizations. Ionization is the process of removing electrons from their orbital paths. This creates free electrons and charged nuclei. Ionizing radiation may lead to chemical reactions within living cells.
    - Particulate radiation is ionizing radiation where the ionization process is initiated by particles. Particulate radiation includes alpha, beta, and neutrons.
    - Non-particulate radiation is ionizing radiation where the ionization process is initiated by electromagnetic energy (photons). Non-particulate radiation includes gamma rays and x-rays.
- Special units are the traditional methods of measurement in the U.S. SI units are used in many other parts of the world. NRC documents now include both special unit and their SI equivalents.
    - Curie is the traditional measure of the level of radioactivity. Becquerel is the SI unit.
    - Rad is the traditional measure of the amount of radiation absorbed by any material, including human tissue. Gray is the corresponding SI unit.
    - Rem is the traditional measure of the amount of dose equivalent (includes the amount of biological damage). Sievert is the corresponding SI unit.

- Dose limits are established by the NRC and published in 10 CFR 20. Doses in the US are measured in rads and rems.
    - The rad is the unit for absorbed dose and refers to how much radiation the tissue has absorbed. The SI unit is the gray. One rad of radiation is always 1 rad, regardless of the type of radiation.
    - The rem is the unit for dose equivalent and refers to how much potential biological damage could occur. The SI unit is the sievert. One rad of radiation does not always equate to one rem of potential biological damage, since rem takes into account the various types of radiation.
- There are a number of equations used in managing dose.
    - Rem = rad X Quality factor (Q)
    - Dose = Dose Rate X Time of exposure
    - Stay Time = Dose Limit ÷ Dose Rate
- There are two types of doses: acute and chronic
    - Acute doses are characterized by short (less than an hour) exposure to ionizing radiation. High doses tend to kill or severely damage cells in the human body, and may result in a form of Acute Radiation Syndrome (ARS). A person may not die from a high acute dose, but will have a higher potential for developing cancer.
    - Chronic doses are characterized by long-term exposure to lower levels of ionizing radiation. They tend to damage or mutate cells rather than kill them, and higher doses increases the risk of cancer.

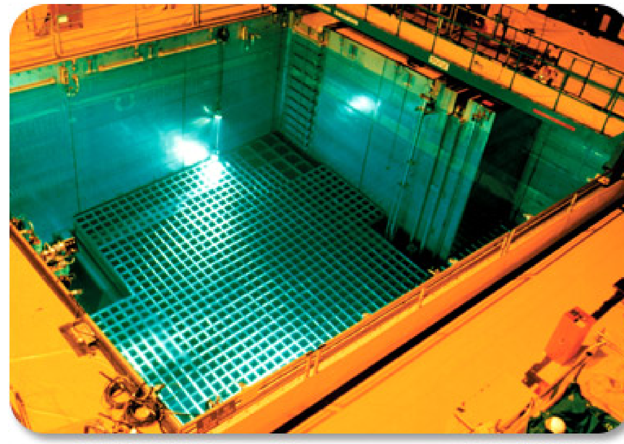
- ALARA is the commonly used acronym for **AS LOW AS IS REASONABLY ACHIEVABLE**. ALARA standards are published in 10 CFR 20.
- ALARA is the guiding principle regarding exposure and doses, regardless of the established dose limits. In other words, even if the dose limit is 5 rem per year, the goal of ALARA is to keep the yearly dose under 5 rem and as close to ZERO as possible.
- There are three main strategies for ALARA doses. They are:
    - Minimize time exposed to radiation
    - Maximize the distance from the radiation source
    - Maximize the shielding from the source
- Contamination and contaminated areas require protective measures to maintain ALARA doses.
- Appropriate signs should be displayed prominently in areas where radiation is known to exist. US radiation signs are required to use a magenta (purple) radiation symbol and text against a yellow background. With international placards, the color combination can be magenta or red on yellow.

- The primary types of background radiation are cosmic and terrestrial. Low dose rates of cosmic radiation constantly bombard the Earth from the sun and the stars, and the dose rates decrease the farther through the atmosphere the radiation travels. Low dose rates of terrestrial radiation are present in the soil, water, and vegetation. People regularly ingest and inhale low levels of radioactive material that gives them chronic background dose.
- The general populace is also exposed to low dose rates of radiation due to the nuclear fuel cycle.
- Nuclear workers are potentially exposed to additional radiation over the background that they and the general public receive. Sources of exposure to nuclear workers include:

- Nuclear fuel decay
- Fission process
- Fission products
- Activation products

Radioactive Waste Management

Chapter Overview



Radioactive waste—also referred to as radwaste—is solid, liquid, and gaseous material from nuclear operations that is radioactive, or that becomes radioactive, and is no longer needed at the plant. This section will discuss the sources, handling, and ultimate disposal of radioactive waste generated by nuclear power plant operation.

Nuclear power plants produce both low- and high-level radioactive waste. The material is either naturally occurring or man-made. Certain kinds of radioactive materials, and the wastes produced from using these materials, are subject to regulatory control by the Federal Government or the various states.

The Nuclear Regulatory Commission (NRC) and some states (known as Agreement States) regulate commercial radioactive waste created from the production of electricity and other, non-military uses of nuclear material.

The graphic to the right shows a 'standard' spent fuel pool where irradiated fuel assemblies are stored until their initial use in the core is complete. Interim storage of spent fuel can be in larger spent fuel pools or dry storage on site. Final storage remains up to the Federal government, with entombment at Yucca Mountain as one option being explored.

Objectives

After completing this chapter, you will be able to:

- Identify sources of radioactive waste produced in nuclear power generation.
- Identify examples of both high- and low-level radioactive waste.
- Understand the methods of processing and disposing of radioactive waste.
- Identify how the NRC classifies and regulates radioactive waste produced by nuclear reactors.

Estimated time to complete this chapter:

25 minutes

Low-level Radioactive Waste

Low-level Radioactive Waste

Radioactive waste—also referred to as radwaste—is solid, liquid, and/or gaseous material from nuclear operations that is radioactive, or becomes radioactive, and is no longer needed at the plant. High-level waste, which we will explore later, includes spent reactor fuel assemblies. Low-level radioactive waste is a secondary product of using other radioactive material.

Throughout this section we will explore sources, processing, and disposal of low-level radioactive waste.

The graphic to the right shows an example of a high-integrity container used to ship and/or store radioactive materials.

Examples

Low-level radioactive waste can be either solid, liquid, and/or gas. Because of the different characteristics of solids, liquids, and gases, each must be processed differently. Here is a list of examples of each type:

Liquid:

- Equipment leak-off points
- Equipment vents and drains
- Floor drain system
- Processing effluents from maintenance (e.g., cleaning of components)

Solid:

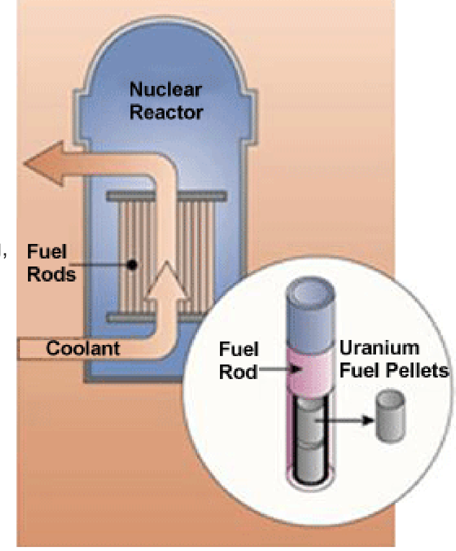
- Contaminated rags, tools, clothing, etc.
- Spent filter cartridges
- Spent demineralizer resins

Gaseous:

- Equipment vents
- Boiling Water Reactor main condenser off-gas
- PWR coolant degassing

Sources

The principal sources of low-level radioactive waste are the reactor coolant water and the components and equipment that come in contact with the coolant. During the normal operation of a nuclear power plant, the coolant water becomes contaminated by activation products (crud) and a very small percentage of fission products that may leak out of the fuel rods.



Source U.S. Nuclear Regulatory Commission

The process of removing radioactive elements from the coolant system, general housekeeping, and maintenance generates much of the low-level radioactive waste. We will look at the generation, processing, and disposal of this low-level radioactive waste.

Generally, the largest contributor to low-level waste is all the extraneous materials associated with operating, maintaining, and cleaning a power reactor such as contaminated anti-C clothing, cleaning materials, broken tools and equipment, and the chemicals and reactor coolant itself.

Chemical and Volume Control System

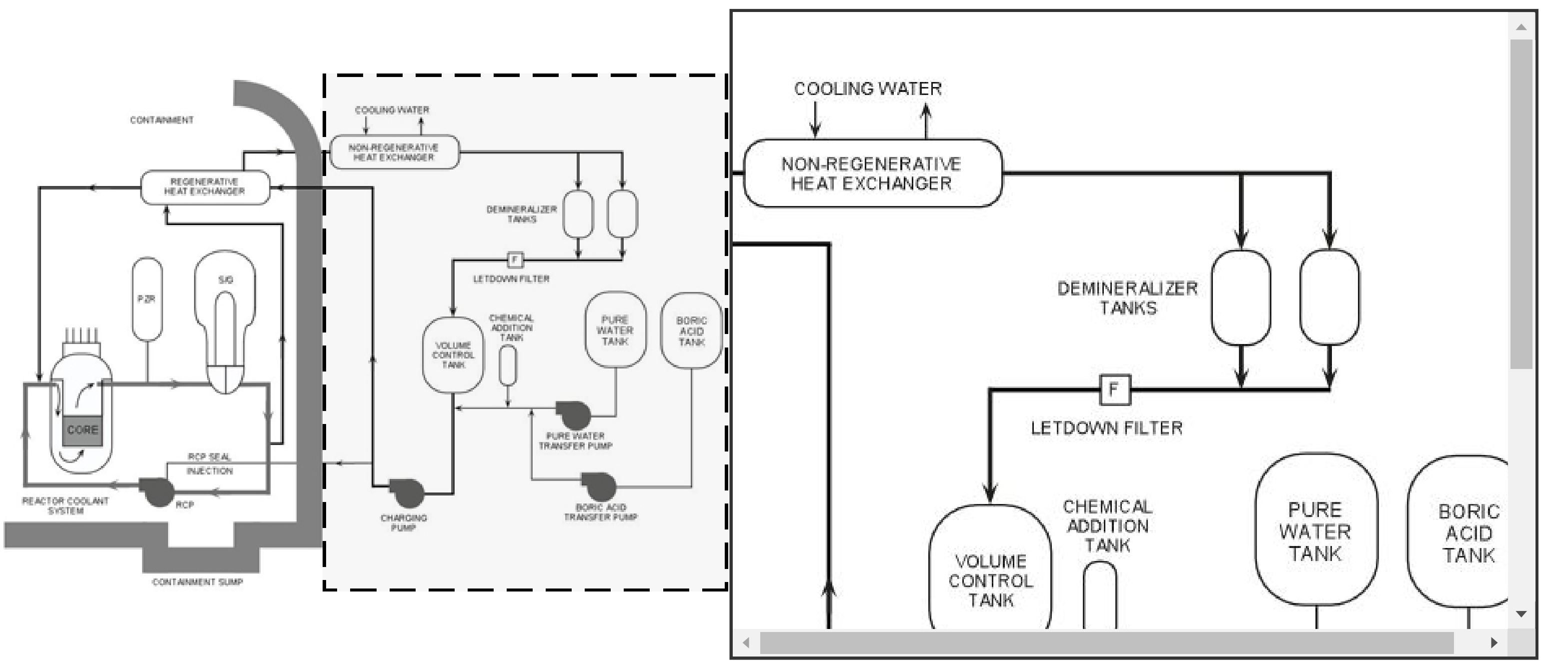
The Chemical and Volume Control System (CVCS) on a Pressurized Water Reactor (PWR) is used to remove the activation and fission products from the reactor coolant.

As the reactor coolant flows through the CVCS, it passes through demineralizers and filters. The resins and filter cartridges become contaminated. After use, the resins and cartridges will be disposed of as solid radioactive waste.

In the PWR volume control tank, the reactor coolant is sprayed into a hydrogen gas bubble. As the water is sprayed, other gases are stripped out of solution. These gases can then be vented to the waste gas system to be processed as gaseous radioactive waste.

If water needs to be removed from the reactor coolant system, there is a flow path that can be lined up to divert the reactor coolant flow from the chemical and volume control system to the liquid radioactive waste system for processing and eventual disposal, or release to the environment after sufficient cleaning/decontamination.

BWR designs include a system called Reactor Water Clean-up (RWCU). This system is similar to the PWR system regarding the continuous removal of small amounts of reactor coolant that is sent through filters and demineralizers for cleaning. As with the PWR system, RWCU water can be sent to waste processing systems for disposal. Below is a simplified graphic of a standard PWR CVCS system.



Cleanups

The CVCS is only one example of how radioactive waste is generated by the operation of a power plant system. Wastes are also generated due to housekeeping and cleanup. For example if a part must be replaced, the tools, clothing, rags, and the used parts can all become contaminated.

Some general housekeeping is also necessary to contain any small leaks or spills during the normal operation of a nuclear reactor. Plant staff routinely capture any material and dispose of it properly. The material and tools (e.g., mops or rags) are now treated as low-level radioactive waste.

Classification

Low-level radioactive waste is classified based on the concentration and type of radionuclides involved. Federal Regulations, Section 10 CFR 61.55, lists the limits on concentrations of specific radioactive materials allowed in each low-level waste class: A, B, or C. Radioactive Waste class depends on two characteristics: dose and stability.

Dose

The first characteristic is the dose that one could receive from the radioactive waste. This dose is dependent on two factors: concentration and the type of radioactive isotope contained in the waste. Thus, something that is in a low concentration, but is highly radioactive, is considered the same as something that is in high concentration, but low in radioactivity.

The regulation noted previously contains a table that gives specific concentrations per cubic meter of waste for the given radionuclide/isotope. Simply put, the lower the concentration or less radioactive the type, the more likely it is Class A. Conversely, the higher the concentration or more long-lived the type, the higher the potential dose and the more likely it will be classified as B or C.

Stability

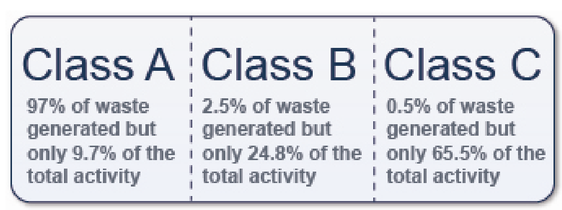
Stability is the ability of the radioactive waste to maintain its gross physical properties and identity over time. To avoid the migration of radionuclides, the radioactive waste is placed in its disposal site and covered. In this way, access to water, which could erode stability, is minimized. Long-term active maintenance and potential exposures to intruders can also be avoided.

Radioactive waste containing the lowest dose can be unstable, similar to ordinary household wastes. If this unstable, low level waste is mixed with the higher activity waste, the lack of stability in the low level waste component could lead to failure of the system, permitting water to penetrate the disposal unit and cause problems with the higher activity waste.

In order to achieve maximum stability, only STABLE Class A waste can be mixed with the other classes. Unstable Class A waste must be segregated at the disposal site.

Class A

As explained previously, Class A waste contains the lowest levels of radioactivity, or radionuclides with relatively short half-lives. If Class A waste does not meet the stability requirements found in 10 CFR 61, it must be segregated from other wastes and disposed of separately at the disposal site.



Class B

Class B contains the next lowest concentration of radioactive materials, and it contains a higher proportion of materials with longer half lives. Class B waste must meet more rigorous requirements on waste form to ensure stability after disposal.

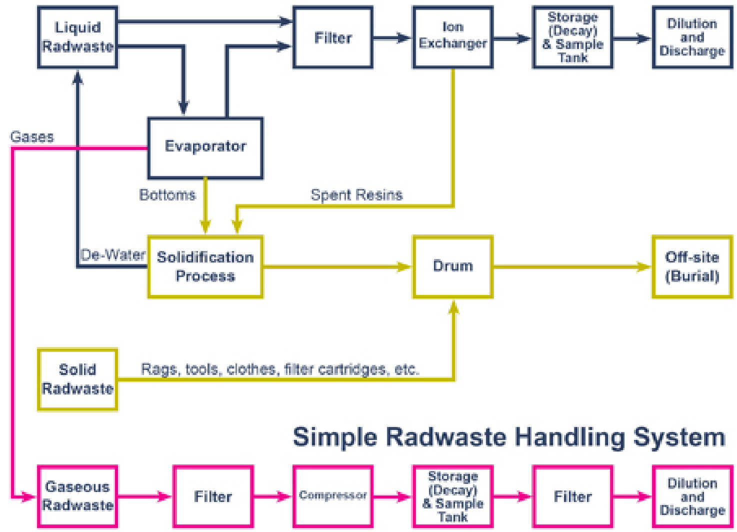
Class C

Class C low-level waste has the highest concentration of radioactive material allowed to be buried in a low-level waste disposal facility. Waste that will not decay to levels that present an acceptable hazard to an intruder within 100 years is designated as Class C waste.

This waste is disposed of at a greater underground depth than the other classes of waste so that subsequent surface activities by an intruder will not disturb the waste. Class C waste must meet all of the Class B requirements and requires additional measures at the disposal facility to protect against inadvertent intrusion.

Handling & Processing

Because of the different characteristics of solids, liquids, and gases each must be processed differently. The waste must also be processed in such a manner as to minimize the risk of public exposure. We will explore how the physical properties of radioactive waste dictate the methods for handling, processing, and disposal.



Liquid Radioactive Waste

Liquids are processed to remove the radioactive impurities. These processes might include:

- Filtering
- Routing through demineralizers
- Boiling off the water (evaporation) and leaving the solid impurities, which are then processed as solid radioactive waste
- Storing the liquid for a time period to allow the radioactive material to decay

After processing the water will be sampled. If samples show the water meets the required standards, the water can be placed in the storage tanks for use in the plant or it can be released to the environment. If the samples show the water does not meet the standards, it will be reprocessed.

Solid Radioactive Waste

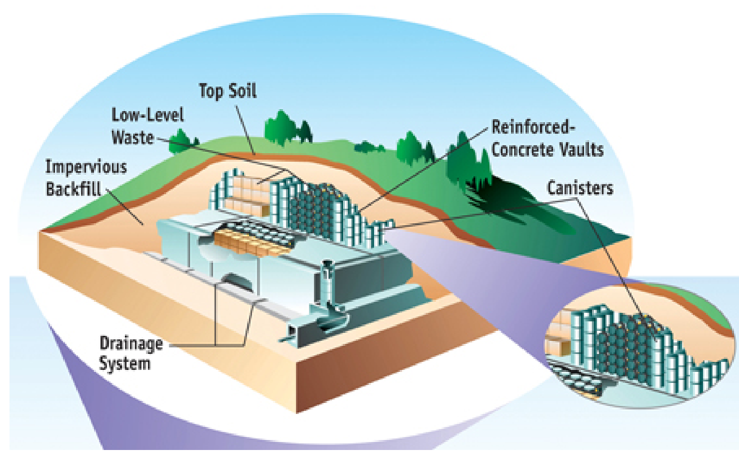
Loose solid radioactive waste (e.g., solids remaining after water has been evaporated) will generally be packaged in high-integrity containers that give the waste package the necessary stability for storage. Solid wastes are packaged as required and shipped to a burial site for disposal. Other loose solid waste, which is commonly referred to as “dry active waste” (e.g., contaminated mop heads, floor coverings, vacuum filters, clothing etc.) is shipped primarily as Class A, so stability requirements don’t have to be met.

Gaseous Radioactive Waste

Gaseous wastes are filtered, compressed to take up less space, and then generally allowed to decay for some period of time (e.g., one month). After the required time has passed, the gases will be sampled and, if the required limits are met, the gases will be released to the atmosphere using a permit. Sometimes the gases will be reused in specific areas of the plant.

Disposal

The proper disposal of radioactive waste will help minimize the dose received by the public. Currently, low-level radioactive waste is all that is accepted for disposal at burial sites.



Low-level waste disposal occurs at commercially operated low-level waste disposal facilities that must be licensed by either NRC or Agreement States. The facilities must be designed, constructed, and operated to meet safety standards. The operator of the facility must also extensively characterize the site on which the facility is located and analyze how the facility will perform for thousands of years into the future.

There are three disposal sites presently operating:

- Barnwell, South Carolina, accepts all low-level waste from the Atlantic Compact only (limited number of states/licensees).
- Hanford, Washington, can accept waste from the Northwest and Rocky Mountain compacts.
- EnergySolutions in Clive, Utah, is authorized to accept only Class A (low-activity/high-volume) waste from all states.

Originally, these three disposal sites were accepting waste from the entire US. The Low-Level Waste Policy Act of 1980 gave each state the responsibility for managing and disposing its own low-level radioactive waste, to be implemented by 1986. This act was amended in 1985 to allow an extension until 1993, and allowed states to enter into compacts with other states to utilize a common disposal site. The Act also divided the United States (U.S.) into these regional low-level waste compacts. Each compact has a host state that will contain the low-level waste disposal site. Some compacts have more than one host state.

High-level Radioactive Waste

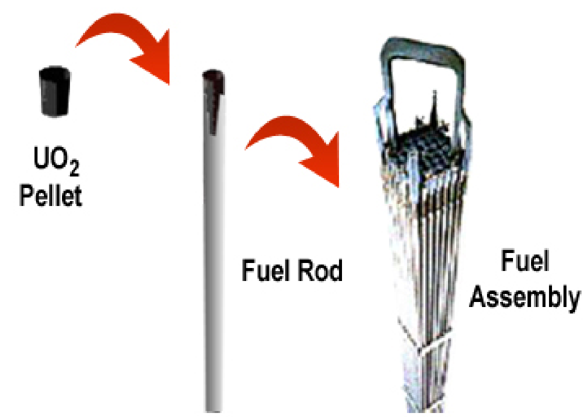
High-level Radioactive Waste

Disposal of high-level radioactive waste is the responsibility of the Department of Energy. The licensing of high-level waste disposal facilities is the responsibility of the USNRC, as specified in 10 CFR Part 60, “Disposal of High-Level Radioactive Waste in Geologic Repositories.”

As with low-level radioactive waste, high-level waste must be processed and disposed of properly. In this section we will explore the current processing and storage of high-level radioactive waste from nuclear reactors, the proposed high-level radioactive waste disposal site at Yucca Mountain, and current temporary storage solutions.

Sources

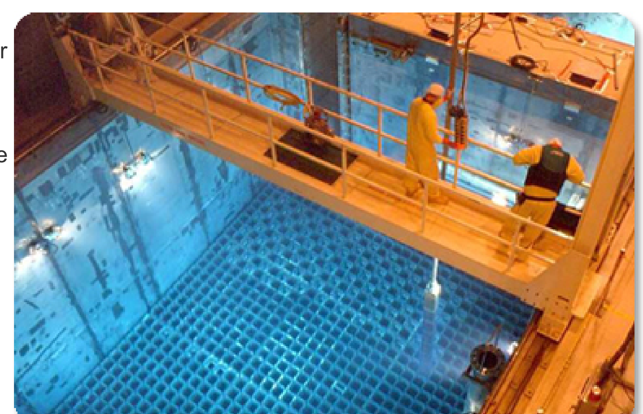
Spent fuel is classified as high-level radioactive waste. This is due to the buildup of very highly radioactive fission products as the fuel is used in the reactor. After a fuel assembly has been used in the reactor core to generate power, there is a large inventory of fission products held inside the cladding of the fuel. Since the reprocessing (recycling) of spent fuel is not done in the U.S. for commercial power plants, the fuel must be disposed of in some safe fashion.



Handling & Processing

Fuel Pool

When the spent fuel is removed from the reactor to be replaced with new fuel, it must be stored for a period of time in the spent fuel pool. The spent fuel must be kept under water due to the heat being generated by the decay of the fission products and to limit the radiation levels in the immediate area.



The spent fuel pools are usually located onsite. However, due to the amount of fuel some power plants have accumulated during their operations, the licensee may also have had to build an Interim Spent Fuel Storage Installation (ISFSI) onsite. These ISFSI facilities store the spent fuel in a dry condition, in robust containers, and can be licensed under the plant's Part 50 license, or separately under Part 72.

Dry Storage

After several years, the heat generated by the decay of the fission products decreases sufficiently to allow the storage of the spent fuel in an air-cooled, dry, above-ground storage facility. These facilities must be designed to remove the heat from the spent fuel and limit radiation exposure to the areas around the facilities. Some licensees have constructed dry cask storage facilities, also known as Independent Spent Fuel Storage Installations (ISFSI). These dry fuel storage facilities must be licensed by the NRC. This licensing can be achieved under the plant's Part 50 license, or under a separate Part 72 license.

Some of these casks have been designed with the idea of using them to ultimately transport the fuel to a permanent facility (e.g., Yucca Mountain) when it is licensed and built. However, the design requirements for these shipping casks need to be approved.

Disposal

Low-level radioactive waste can be dispositioned in several ways. The most common method to disposition low-level waste that is generated at nuclear plants is by sending it for permanent disposal at a licensed burial facility. The proper disposal of radioactive waste at a licensed and monitored burial facility ensures that the dose received by the public is as low as is reasonably achievable. Currently, low-level radioactive waste is all that is accepted for disposal at burial sites in the U.S. (disposition of high-level waste will be discussed later).

Low-level waste disposal at commercially operated burial facilities must be licensed by either NRC or Agreement States. The facilities must be designed, constructed, and operated to meet the safety standards and performance objectives in 10 CFR 61. The operator of the facility must also extensively characterize the site on which the facility is located and analyze how the facility will perform for thousands of years into the future.

The Low-level Radioactive Waste Policy Amendments Act of 1985 ultimately gave the states responsibility for the disposal of their low-level radioactive waste. The Act encouraged the states to enter into compacts that would allow them to dispose of waste at a common disposal facility. Most states have entered into compacts; however, only one new disposal facility has been built since the Act was passed. There are four low-level waste disposal sites presently operating:

- **EnergySolutions Barnwell Operations, located in Barnwell, South Carolina**

Currently, Barnwell accepts waste from the Atlantic compact states (Connecticut, New Jersey, and South Carolina). Barnwell is licensed by the State of South Carolina to dispose of Class A, B, and C waste.

- **U.S. Ecology, located in Richland, Washington**

Richland accepts waste from the Northwest and Rocky Mountain compacts. Richland is licensed by the State of Washington to dispose of Class A, B, and C waste.

- **EnergySolutions Clive Operations, located in Clive, Utah**

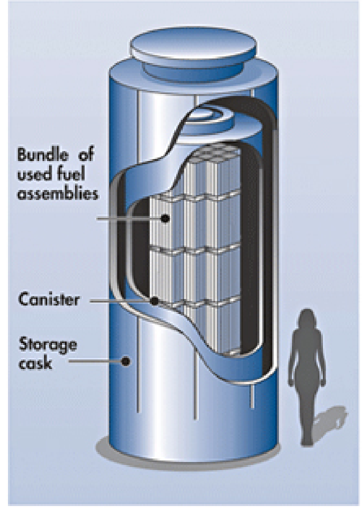
Clive accepts waste from all regions of the United States. Clive is licensed by the State of Utah for Class A waste only.

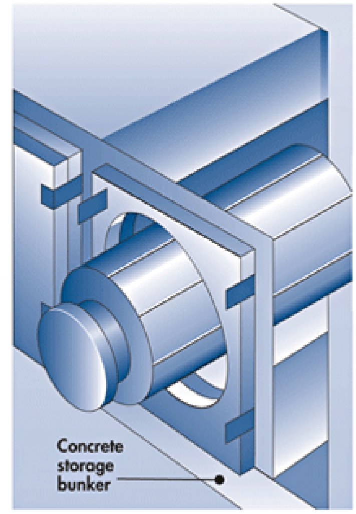
- **Waste Control Specialists (WCS), LLC, located near Andrews, Texas**

WCS accepts waste from the Texas Compact generators and outside generators with permission from the Compact. WCS is licensed by the State of Texas to dispose of Class A, B, and C waste.

Dry Cask Long-term Storage

In response to this, many utilities have extended the licensing and construction of dry fuel storage facilities as a long-term storage solution. These independent spent fuel storage installations (ISFSI) come in different designs, but all utilize natural air flow for cooling. These facilities have separate licenses from those governing the operation of the power plant. The facilities are a temporary fix to the ultimate disposal of spent fuel.

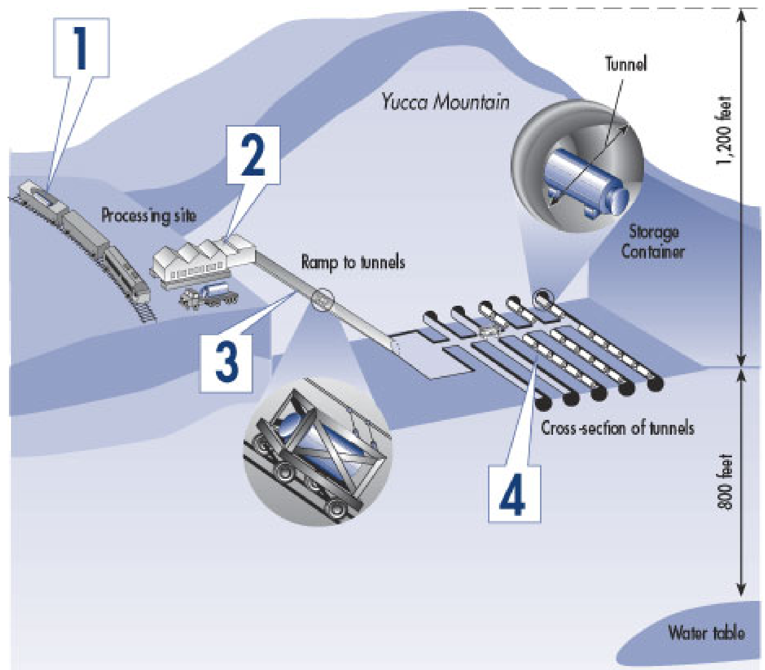




Yucca Mountain

Even though there is not presently a high-level waste repository accepting spent fuel for disposal, the Nuclear Waste Policy Act of 1982, as amended, directed the Department of Energy to site, design, construct, and operate a high-level waste repository.

The proposed site for the high-level repository is Yucca Mountain, Nevada. The site will resemble a mining complex. On the surface will be the waste handling facilities. About 1000 feet below the surface will be the disposal site for the containerized waste. The image to the right illustrates the Yucca Mountain concept.



▶ 1. Canisters of Waste

▶ 2. Preparation for Burial

▶ 3. Ramp to Tunnels

▶ 4. Container Storage

The Environment Protection Agency (EPA) has published its final regulations for the site. They can be found in 40 CFR Part 197, “Environmental Radiation Protection Standards for Yucca Mountain.” The regulations limit the dose to the public to 15 mrem/year from the facility. The regulations also impose an additional groundwater protection dose limit of 4 mrem/year from beta and photon-emitting radionuclides.

The Department of Energy has submitted the application for Yucca Mountain and the licensing process is ongoing.

For more information on Yucca Mountain, the following NRC link may be helpful:

https://www.nrc.gov/waste/hlw-disposal.html

Summary

Key Points of Low-level Radioactive Waste

- Radioactive waste, is solid, liquid, and gaseous material from nuclear operations that is radioactive, or has become radioactive, and is no longer needed at the plant.
- Low-level radioactive waste can be either solid, liquid, or gas. Each type must be processed differently.
  - Gaseous wastes are filtered, compressed, allowed to decay, and then release safely.
  - Loose solid radioactive waste is generally now mixed with a hardener OR de-watered and placed in high-integrity containers to improve stability.
  - Liquid waste is either filtered, demineralized, evaporated, or allowed to decay.
- The principal sources of low-level radioactive waste are the reactor coolant water and the components and equipment that come in contact with the coolant.
  - The CVCS on a PWR is used to remove the activation and fission products from the reactor coolant. The RWCU system in a BWR does the same thing. These processes generate low-level radioactive waste.
  - During routine housekeeping, plant staff routinely capture any material and dispose of it properly.
- Radioactive waste classification depends on two characteristics:
  - **dose** – the concentration and type of radioactive isotope contained in, or emitted from, the waste
  - **stability** - the ability of the radioactive waste to maintain its gross physical properties and identity over time
- Low-level radioactive waste is buried at designated facilities to limit exposure to the public.

Key Points of High-level Radioactive Waste

- Disposal of high-level radioactive waste is the responsibility of the Department of Energy.
- Spent fuel is classified as high-level radioactive waste due to the buildup of highly radioactive fission products.
  - Spent fuel must be stored for a period of time in the spent fuel pool to absorb decay heat and to confine emitted radiation.
  - After several years, the fuel decays sufficiently to allow the storage in an air-cooled, dry, above-ground storage facility.
- High-level radioactive waste is not currently being accepted for burial.
  - In response to this, many utilities have implemented the licensing and construction of dry fuel storage facilities as a long-term storage solution.
- The Nuclear Waste Policy Act of 1982, as amended, directed the Department of Energy to site, design, construct, and operate a high-level waste repository.
  - The proposed site for the high-level repository is Yucca Mountain, Nevada.
  - The Department of Energy has submitted the application for Yucca Mountain and the licensing process is ongoing.

Transportation of Radioactive Material

Chapter Overview

About 3 million packages of radioactive material (RAM) are shipped each year in the United States (U.S.) either by road, rail, air, or water. This represents less than 1 percent of the Nation's yearly hazardous material shipments. Oversight of the safety of commercial RAM shipments is the joint responsibility of the NRC and the U.S. Department of Transportation (DOT).

Generally, the DOT is responsible for regulating safety in transportation of all hazardous materials, including radioactive materials, and the NRC is responsible for regulating safety in receipt, possession, use, and transfer of byproducts, source, and special nuclear materials.

The division of responsibilities between the DOT and NRC are specified in a Memorandum of Understanding (MOU).

The breakdown of responsibilities between DOT and NRC are as follows:

DOT -

1. Regulates shippers and carriers of RAM
2. Regulates Type A and LSA Packaging and Shipments
3. Issues Certificates of Competent Authority for International Shipments

NRC -

1. Approves the design of Type B and Fissile Material Packaging
2. Establishes Transportation Safeguards
3. Investigates Accidents/Incidents
4. Technical Advisor to DOT

The vast majority of these shipments consist of small amounts of RAM used in industry, research, and medicine. The NRC requires such materials to be shipped in accordance with DOT's hazardous materials transportation safety regulations.

The basic principle regarding transport of RAM is to either restrict the type and activity of the contents OR provide accident-proof packaging. This chapter will discuss the NRC's shared oversight of the transportation of RAM in the U.S. with the DOT. DOT follows 49CFR Part 173 and NRC follows 10CFR Part 71 regulations.

Basic Premise

Fundamental to a good understanding of radioactive material transportation safety and packaging requirements is the basic premise that:

Safety in transporting radioactive material primarily depends upon the use of the proper packaging for the type, quantity, and form of the radioactive material (RAM) to be transported. In addition, packaging design is performance oriented, with the packaging integrity being dictated by the hazards of the radioactive content.

Put more simply, proper packaging is the primary means of providing safety, and contents which present higher hazards are to be contained in stronger packaging.

Objectives

At the end of this chapter, you will be able to:

- Describe key regulations and requirements that govern transportation of RAM in the U.S.
- Identify Class 7 as the United Nations (UN) classification for RAM
- Describe the shared responsibility between the NRC and the DOT regarding shipment of RAM
- Differentiate among four types of RAM transport packages

Estimated time to complete this chapter:

45 minutes

United Nations Hazardous Materials Classifications

UN Classifications

All hazardous materials that could potentially be transported are assigned to one of the nine (9) UN Classes. In general, the hazardous materials listed pose an **immediate threat to health and safety.**

Click the different classes below for information about that specific UN Class.

- ▶ Class 1
- ▶ Class 2
- ▶ Class 3
- ▶ Class 4
- ▶ Class 5
- ▶ Class 6
- ▶ Class 7
- ▶ Class 8
- ▶ Class 9

Class 7

All hazardous materials that could potentially be transported are assigned to one of the nine (9) UN classes, as we discussed previously. In general, the hazardous materials listed pose an immediate threat to health and safety. However, for RAM the threat is potentially the less-immediate risk of cancer, although in large enough quantities radiation can pose a more immediate, acute threat.

Since transport accidents cannot be totally prevented, the regulations are primarily designed to:

- Ensure safety in routine handling situations for minimally hazardous material
- Ensure package integrity under all circumstances for highly dangerous materials

These goals are accomplished by focusing on the package and its ability to:

- Effectively contain the material (prevent leaks)
- Effectively control radiation emitted from the package (provide shielding)
- Prevent 'criticality' for fissile material
- Provide adequate dissipation of any heat generated within the package

NRC and DOT Responsibilities

Licensing and Inspection

The NRC conducts about 1,000 transportation safety inspections of fuel, reactor, and materials licensees annually.

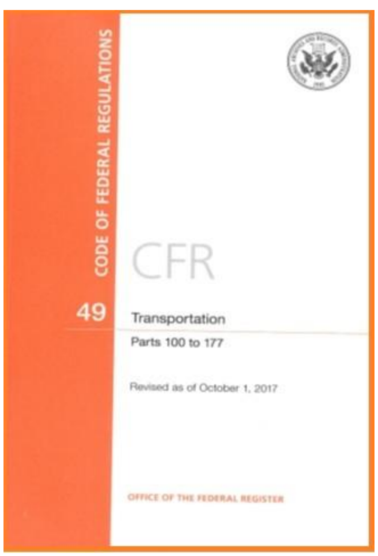
The NRC reviews, evaluates, and certifies approximately 50 to 70 new, renewal, or amended transport package design applications for the transport of nuclear materials annually.

The NRC reviews and evaluates approximately 150 license applications for the import or export of nuclear materials annually.

DOT Reg 49 CFR

The NRC and DOT share responsibility for the control of RAM transport based on a Memorandum of Understanding (MOU).

In general, DOT regulations (49 CFR) are more detailed. They cover all aspects of transportation including packaging, shipper and carrier responsibilities, documentation, and all levels of RAM from exempt quantities to very high levels.

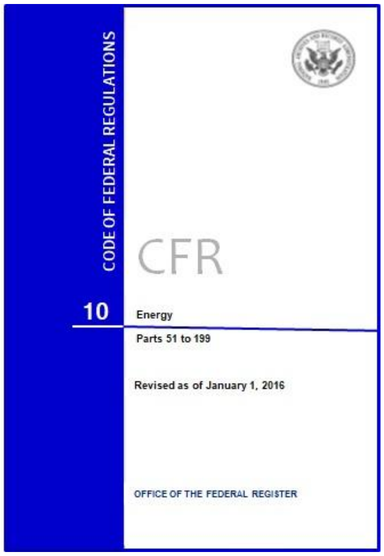


NRC Reg 10 CFR 71

The NRC regulations (10 CFR 71) are primarily concerned with special packaging requirements for higher level quantities. NRC regulation 10 CFR 71.5 requires NRC licensees transporting RAM to comply with DOT regulations.

The NRC regulations (10 CFR 71) establish requirements for packaging, preparation for shipment, and transportation of licensed material; and procedures and standards for NRC approval of packaging and shipping procedures for fissile material and for quantities of material in excess of a Type A quantity (larger quantities of material).

NRC regulation 10 CFR 71.5 requires NRC licensees transporting RAM to comply with the applicable DOT regulations in 49 CFR. If the shipment does not fall within the scope of DOT regulations, the licensee still must comply in part with DOT 49CFR.


CODE OF FEDERAL REGULATIONS
CFR
10
Energy
Parts 51 to 199
Revised as of January 1, 2016
OFFICE OF THE FEDERAL REGISTER

Packages

Transport Packages

Annually, about twenty million packages of all sizes containing RAM are routinely transported worldwide on public roads, railways and ships. These packages are transported using various types of robust and secure containers. There has never been any accident in which a container with **highly** radioactive material has been breached or has leaked.

When transporting hazardous material, packaging is used to store and ship the materials.

Packaging is the container alone; a package is the container AND the material, together (as shown in the graphic below).

We will discuss four of the types of RAM packagings (there are more depending on the type of material being shipped):

- Excepted
- 'Other' (Industrial packaging, Fissile, Uranium Hexafluoride (UF_6))
- Type A
- Type B

Excepted Packaging

Excepted Packaging is designed to survive **normal** conditions of transport. They are generally used for shipping of materials that contain very low levels of radioactivity and/or dose rates. They include radioactive instruments or articles, articles manufactured from natural or depleted uranium or natural thorium, and empty packages.

Excepted packaging can be almost any type of container that meets the basic requirements, with any of the above contents. Excepted packages are for extremely LOW levels of radioactivity with very low hazard. They are excepted from several labeling and documentation requirements, but must at least bear the letters UN and the four digit UN ID number. An example is shown to the right:

'Other' Packaging

- Industrial Packagings (IP) are designed to survive normal conditions of transport (IP-1) and at least the DROP test and stacking test for Type A packaging (IP-2 and IP-3). Industrial packagings (IP) are used for transportation of materials with small amounts of radioacitivity, Low Specific Activity (LSA), or radioactivity on the surface, Surface Contaminated Objects (SCO). Industrial packagings (IP) are usually metal boxes or drums. Examples of LSA material include dry active waste from nuclear power stations, contaminated laundry shipments, and rubble and miscellaneous debris from decommissioning activities. SCO materials include such items as pumps, valves, and piping from nuclear power stations and fuel cycle facilities that are shipped offsite for disposal or repair or perhaps testing. Below are two common types of IP:

- Fissile Packaging for Fissile Material (U-235, U-233, Pu-239, Pu-241) will be transported as either Type A or B packages, and it must ensure that criticality is adequately controlled (i.e., amount and geometry). Below is a graphic of a new fuel (PWR) shipping container. Note that it limits the packaging to only two (2) fuel assemblies with minimum distancing for criticality control, along with restraints for restricting movement.

- Uranium Hexafluoride (UF_6) packaging is generally a cylinder of steel, nickel, or monel, in various diameters ranging from 1 inch to 48 inches. The shape and size of the cylinder will vary depending on the quantity of uranium being shipped and its % by weight of the actual uranium (assay). Below is an example of some UF_6 packaging:

Type A Packaging

Type A packaging is designed to survive normal transportation, handling, and minor accidents. The Type A packaging is used to transport small quantities of radioactive material with higher concentrations of radioactivity than those shipped in industrial packaging but not in amounts that would not result in significant health effects if they were released. There are two (2) levels of Type A packaging that differentiate between material that can, or cannot, be dispersed easily if the container is breached.

Type A packaging may be cardboard boxes, wooden crates, steel drums, or sometimes custom containers. The shipper and carrier must have documentation of the certification of the packages being transported.

Various examples of Type A Packaging:

Type B Packaging

Type B Packaging must be able to survive severe accidents. They are used for the transportation of material with high levels of radioactivity or very high dose rates. Examples of this include spent fuel assemblies

Type B packaging must meet severe accident performance standards that are considerably more rigorous than those required for Type A packaging. They may be shielded metal drums or huge, shielded transport containers.

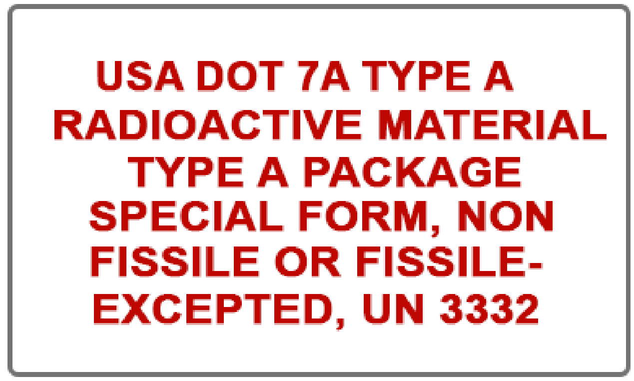
They must either have a Certificate of Compliance (COC) from the NRC or Certificate of Competent Authority (COCA) from the DOT.

The graphic to the right shows a Type B packaging for transport of spent fuel assemblies.

Shipping Requirements

Markings

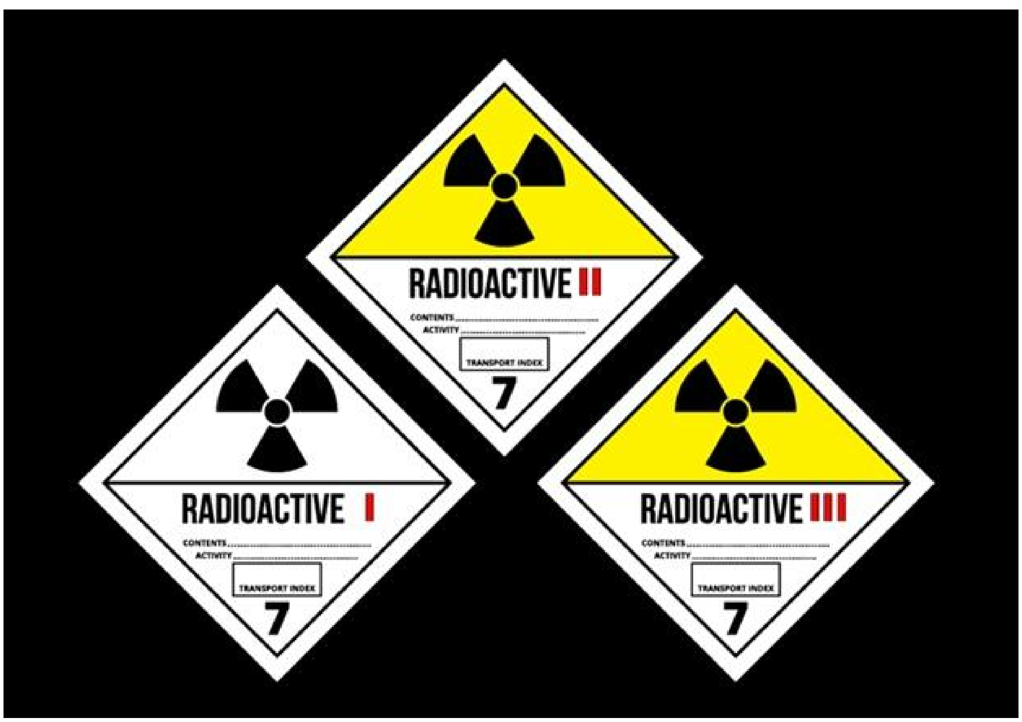
Federal regulations require that shippers meet marking and labeling requirements for packages containing radioactive materials. Markings are designed to provide an explanation of the contents of the package with the proper shipping name, emergency response identification number, and other standard terms and codes.



Labels

Labels identify the contents of the package and the level of radioactivity. There are three different labels depending on the dose rates, one label for empty packaging and another for fissile material. Besides the latter two Class 7, labels are based on the radiation level outside of the package, RADIOACTIVE White-I, RADIOACTIVE Yellow-II, and RADIOACTIVE Yellow-III. Shipments with extremely low levels of radioactivity that do not present a hazard are excepted from labeling requirements.

Although the package required for transporting RAM is based on the activity inside the package, the proper label to apply is primarily determined by the radiation level on contact with the package and the transportation index (TI), which is the radiation level (in mrem/hr) at 1 meter from the package. Both the surface radiation level and the TI will determine the label to be used. Below are examples of the labels for RAM packages.

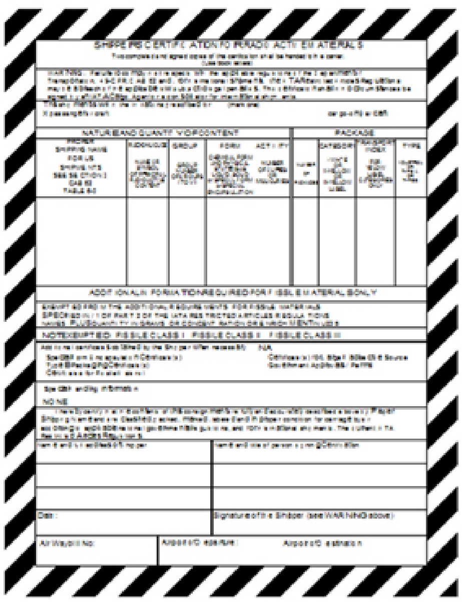


Placards

Placards also play an important role in communicating that hazardous material is present in or on the transport vehicle. Placards look similar to labels but have less information and can be larger in size. Placards are placed on the transport vehicle and the package there needs to be a minimum of four one on every side and end. Placards are specified for hazardous materials in accordance with the tables listed in 49 CFR 172. “Radioactive” placarding is required if any radioactive material package bearing the "RADIOACTIVE YELLOW-III" label and for some cases of LSA or SCO type shipments. In addition, the placard can be used even when it is not required.

Shipping Papers

The only way for anyone to know what is being transported inside a vehicle is by reviewing the shipping papers. These documents, by words and codes, clearly specify what is being transported. They must be readily accessible to the driver and to emergency response personnel if the driver is not available.



Summary

Key Points

- The UN has a system that consists of nine classifications of hazardous materials for transport; Class 7 refers to RAM.
- Class 7 materials pose a potential threat with the non-immediate risk of cancer; and in large enough quantities, radiation can pose an immediate threat.
- NRC and DOT have dual responsibility in the regulation and inspection of the transportation of RAM described in the MOU.
- The transport of RAM has to follow both DOT 49 CFR and NRC 10 CFR 71 requirements.
- Proper packaging ensures material will be contained based on normal conditions and design-based accident conditions.
- Package (complete contents) = Packaging (shipping container) + Radioactive Material (RAM)
- Different types of containers and packages are used for transportation. The following were discussed:
  - Excepted and Industrial Packages are designed to survive normal transportation handling.
  - Type A containers are designed to survive normal transportation handling and minor accidents.
  - Type B containers must be able to survive severe accidents and require a Certificate of Compliance by the NRC for authorized use by an NRC licensee.
- The following are all used to identify the contents of transport and to provide hazardous material information for emergency response:
  - Shipping papers – kept accessible near driver and gives detailed content identification
  - Markings – on package, gives specific information about contents for transportation and emergency workers
  - Labels – on outside of packaging to give indication of radiation levels and content
  - Placards – go on all four sides once package is on the vehicle for transport to give a general idea about hazard (Hazard Class(es))

Refueling Operations

Chapter Overview

A reactor core is designed to operate at its full power output for some period; usually 12, 18, or 24 months. After this time period, known as a fuel cycle, the reactor must undergo refueling operations. During refueling, a portion of the core will be replaced with fresh fuel, and the remaining older fuel will be repositioned. The amount of the core replaced with new fuel during the refueling outage will depend on the projected core lifetime.

This chapter will describe some of the basic activities involved in the refueling process for a 'standard' Westinghouse pressurized water reactor (PWR). Some differences with BWR refueling will be mentioned. The refueling methods and activities of other types of reactors (both CE/B&W PWRs and other generation GE BWRs) are similar in nature.

Objectives

At the end of this chapter, you will be able to:

- Identify the major components and their functions for refueling operations.
- Sequence the basic steps in refueling a nuclear power plant.

Estimated time to complete this chapter:

35 minutes

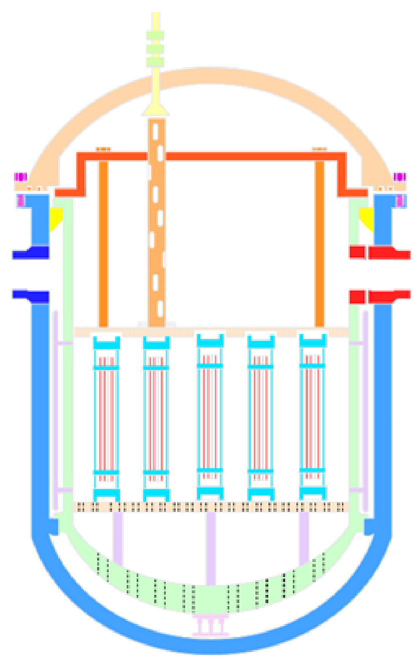
Refueling Components

New Fuel

Long before the actual refueling process takes place, plans are made for the refueling outage. The new fuel is ordered, manpower considerations are taken into account, and any other known maintenance is scheduled during the outage. Some period of time before the refueling is to occur, the new fuel is received onsite. The new fuel will be inspected and then stored in the new fuel storage area.

The new fuel storage area is a dry storage vault in the fuel handling building. Dry storage is sufficient because the fuel has not been used for power production. Therefore, there are insufficient levels of fission products to generate heat and the radiation levels around the fuel are only slightly above background levels.

Vessel Disassembly



When it is time to perform the refueling, the reactor is shutdown and the plant is cooled down. This is necessary to allow access to the vessel area and to allow its disassembly. At this time, the fuel handling equipment will be tested to ensure proper operation.

After the plant is cooled down and depressurized, disassembly of the reactor will commence. First, all cables, ventilation ducts, cable trays, and insulation are removed. Then, a seal must be installed between the reactor vessel and the reactor cavity wall since the cavity over the open vessel will be filled with water. The seal will prevent any water from leaking out of the vessel cavity area. Many plants now leave this seal in place all the time (saves time/money/dose).

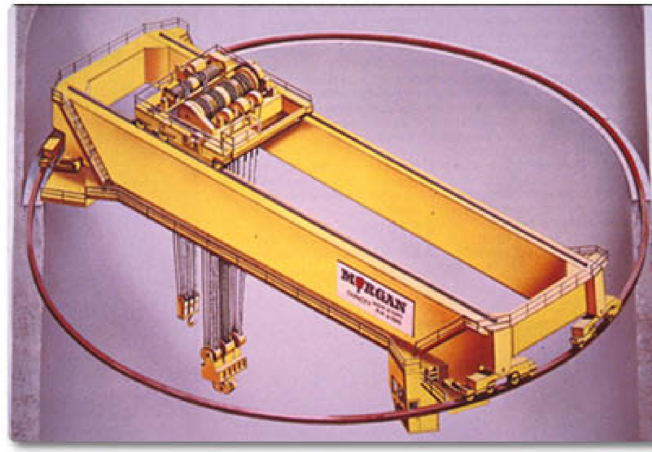
To transfer the fuel back and forth from the containment building/fuel building, the fuel transfer tube is opened up. This is done by removing a blind flange (solid cover) from the end of the transfer tube in the containment building. After the refueling cavity is flooded, a valve is opened in the fuel building to open the transfer path between the spent fuel and the refueling pools.

The studs that hold the vessel head on the reactor vessel are now removed. Guide studs are installed to provide alignment when moving the head and vessel internals. Except for the guide studs, the remaining stud holes are plugged to protect the threads from the borated water (PWR). The vessel head is moved to a dry storage area inside the containment. To move the head, and other heavy components (e.g., RCP motors), there is an overhead crane installed inside the containment building, called the polar crane.

Polar Crane

High in the containment building is a polar crane strong enough to lift the vessel head (100+ tons). The crane can also move in such a fashion as to reach virtually any part of the containment building for other movable equipment (e.g., reactor coolant pumps, valves, etc.)

Prior to removing the reactor vessel head, operators will cool the plant down to about 100°F and depressurize it to atmospheric pressure. The vessel head is held in place by about 56 large bolts, called studs. Mechanics will utilize hydraulic tensioners to elongate the studs in order to loosen the nuts holding the head in place. The studs and nuts are then each removed as one unit. Upon reassembly at the end of the refueling outage, the tensioners are again used to elongate the bolts so the nuts can be reinstalled with the proper tension on the head.


MORGAN

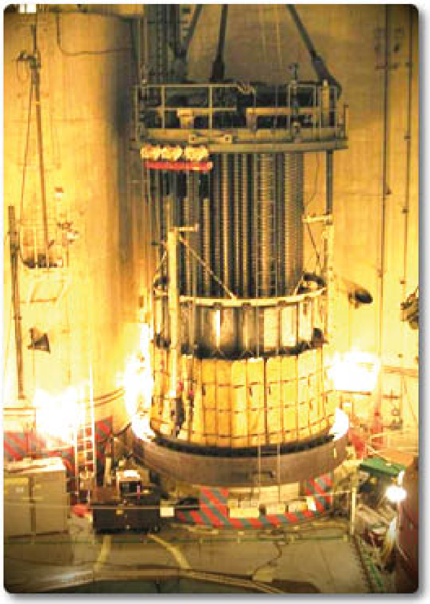
Refueling Process

Lifting The Vessel Head

The reactor vessel head is lifted and set down on a special stand within the containment building. During this time, most personnel are evacuated from the containment building for reasons of industrial and radiological safety (i.e., falling loads and high airborne radioactivity from underneath the head).

When set down, the underside of the vessel head is heavily shielded, but can be inspected by robotic machines looking for weld cracking and corrosion. The outside of the head may be contaminated, but the underside will be highly radioactive due to its proximity to the core.

All other components of the reactor assembly (fuel assemblies, upper guide structure, core shroud, control rods, in-core instruments) inside the vessel will be removed and stored underwater at all times due to their extremely high levels of radioactivity.



Flooding

The flooding of the refueling cavity commences after the vessel head is lifted. The cavity is flooded by pumping water from the refueling water storage tank through the residual heat removal pumps and into the reactor coolant system where it will enter the reactor vessel and overflow into the cavity. The water level will be increased to at least 25 feet above the reactor vessel flange. The water in the refueling pool is an important part of the shielding of personnel from dose coming from vessel and core components.

After the cavity is flooded, the upper core internals are removed. Prior to the internals removal, the control rod drive shafts must be disconnected from the control rods (to prevent pulling the control rods out of the fuel when the upper internals are pulled). The shafts are disconnected and the upper internals package is removed and stored underwater. Fuel movement/shuffling may now commence.

Fuel Transfer

Before a new fuel assembly can be placed into the core, an old assembly must be removed. Once an old assembly has been removed, a new fuel assembly can now be sent to the reactor. Because of the high radioactive levels of spent fuel, all fuel movements are conducted underwater for shielding.

Click below for information about the transferring of spent fuel in the reactor.

▶ Manipulator Crane

▶ Conveyor Car / Fuel Carriage

▶ Upender

▶ Fuel Transfer Tube

▶ Second Upender

▶ Spent Fuel Bridge Crane

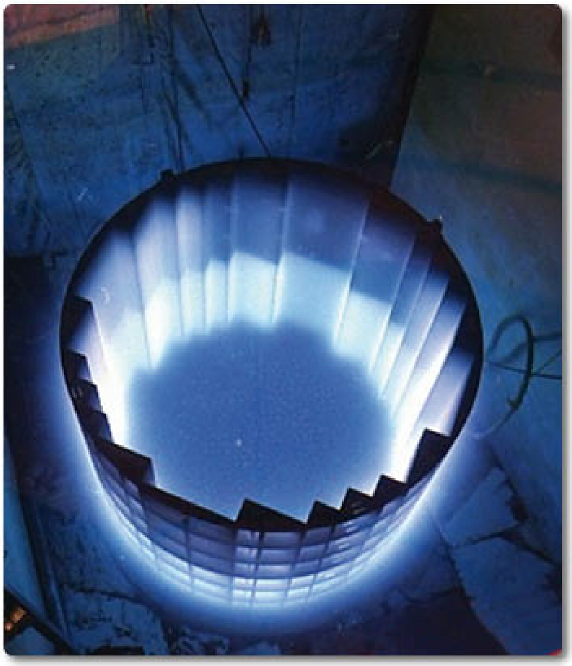
New Fuel Transfer

In order for a new fuel assembly to be transferred to the containment building, it is placed into the spent fuel pool using the fuel building crane and the new fuel elevator. From the spent fuel pool, the spent fuel bridge crane will take the assembly and place it into the conveyor car on the upender. The upender will lower the conveyor car and fuel assembly to the horizontal position. From this position, the new fuel is sent through the fuel transfer tube into the containment building. A similar upender in the containment building will raise the fuel assembly and conveyor car to the vertical position, where the manipulator crane (refueling machine) will pick up the fuel assembly and transfer it to its proper position in the core. The movement of fuel will continue until all fuel assemblies are in their proper location. After the completion of the fuel shuffle, a record is made of all fuel assemblies and their location in the core.

A refueling outage may last a month or more, depending on the extent of other maintenance work. However, the fuel shuffling process usually takes less than 1 week.

Full Core Offload

Some plants perform a full core offload instead of a fuel shuffle. A fuel core offload is where all of the fuel is removed from the reactor vessel and placed in the spent fuel pool. In the spent fuel pool, special tools are used to transfer any fuel assembly inserts (control rods, thimble plugs, etc.) to new fuel assemblies. One major advantage of a full core offload is that certain Technical Specifications that require operable systems do not have to be met, which means that plant personnel can work on many safety systems in parallel.



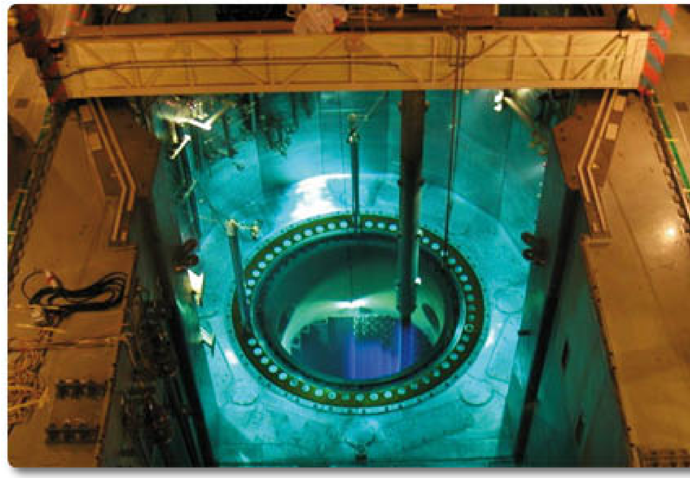
Full offloads would be required if the plant needed to inspect the reactor vessel or core barrel. The picture to the right is a core shroud after it has been removed from the reactor vessel. It is kept underwater due to its high level of radioactivity (note the blue glow from Cherenkov radiation effects).

Reconnection

Once the operators are finished with the fuel shuffle/replacement, the upper internals are replaced over the core. The operators will then reconnect the control rod drive shafts.

The refueling pool will be drained and the reactor vessel head will be replaced and bolted down. The containment area will be decontaminated to the extent possible and all extraneous material and equipment will be removed.

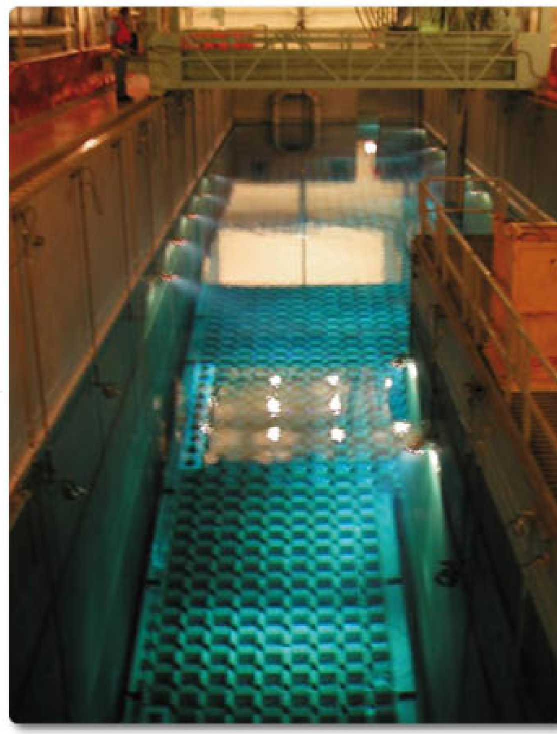
Operators will conduct testing of plant protective equipment and systems to ensure that they can operate effectively for the upcoming fuel cycle. They will heat up and re-pressurize the plant, then take the reactor critical and begin the power operation cycle all over.



Spent (Exhausted) Fuel

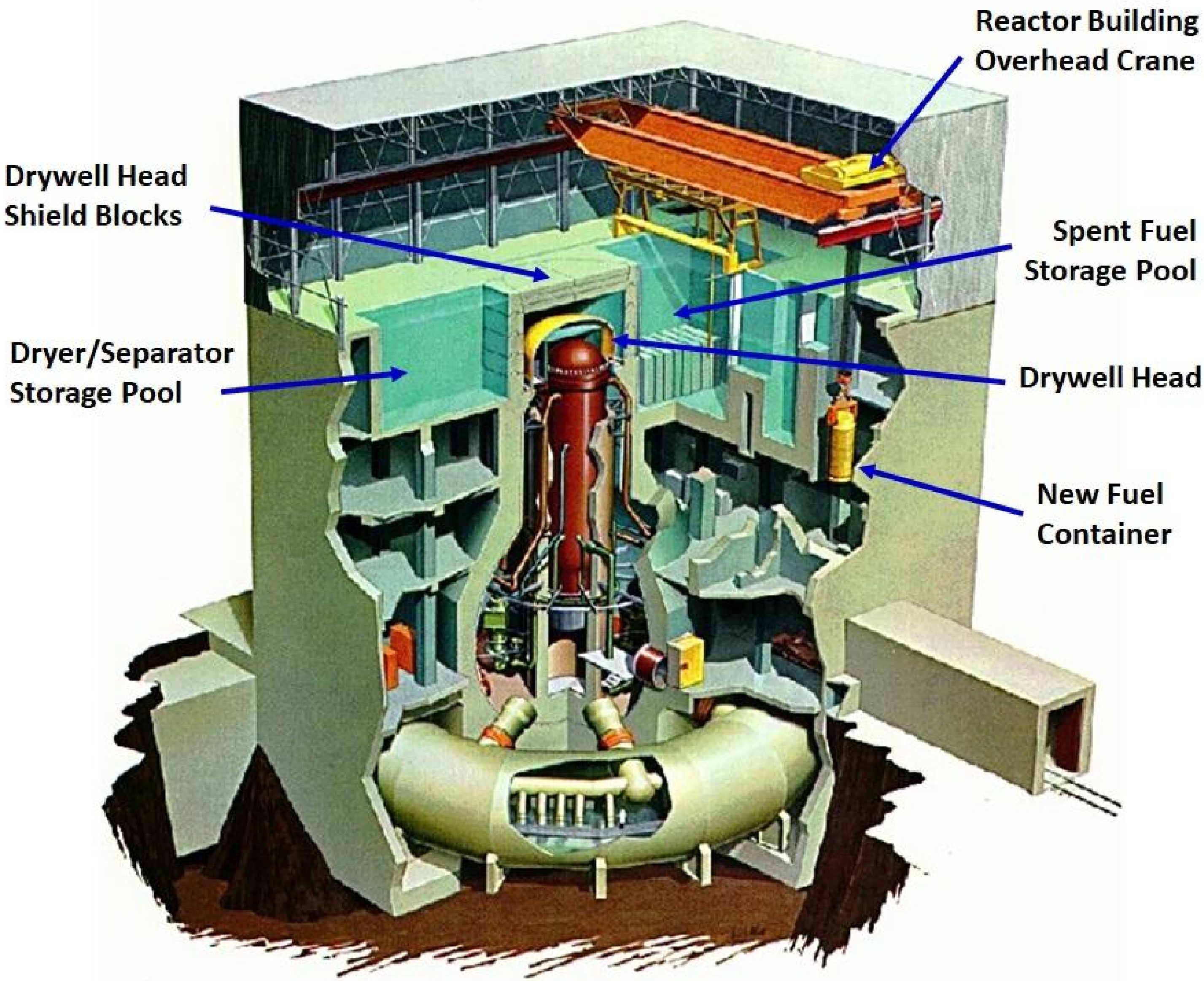
The exhausted fuel that was removed from the core during the outage will be stored in the spent fuel pool. This spent fuel will continue to produce a significant amount of decay heat for many years to come.

Since the spent fuel continues to produce heat, the spent fuel has a cooling system. If cooling to the spent fuel is lost for an appreciable amount of time, it is possible for the pool water to heat up to the point where it will boil. For this reason, the spent fuel pool and its support systems are safety related. They have emergency power, redundant systems, and are seismically qualified.

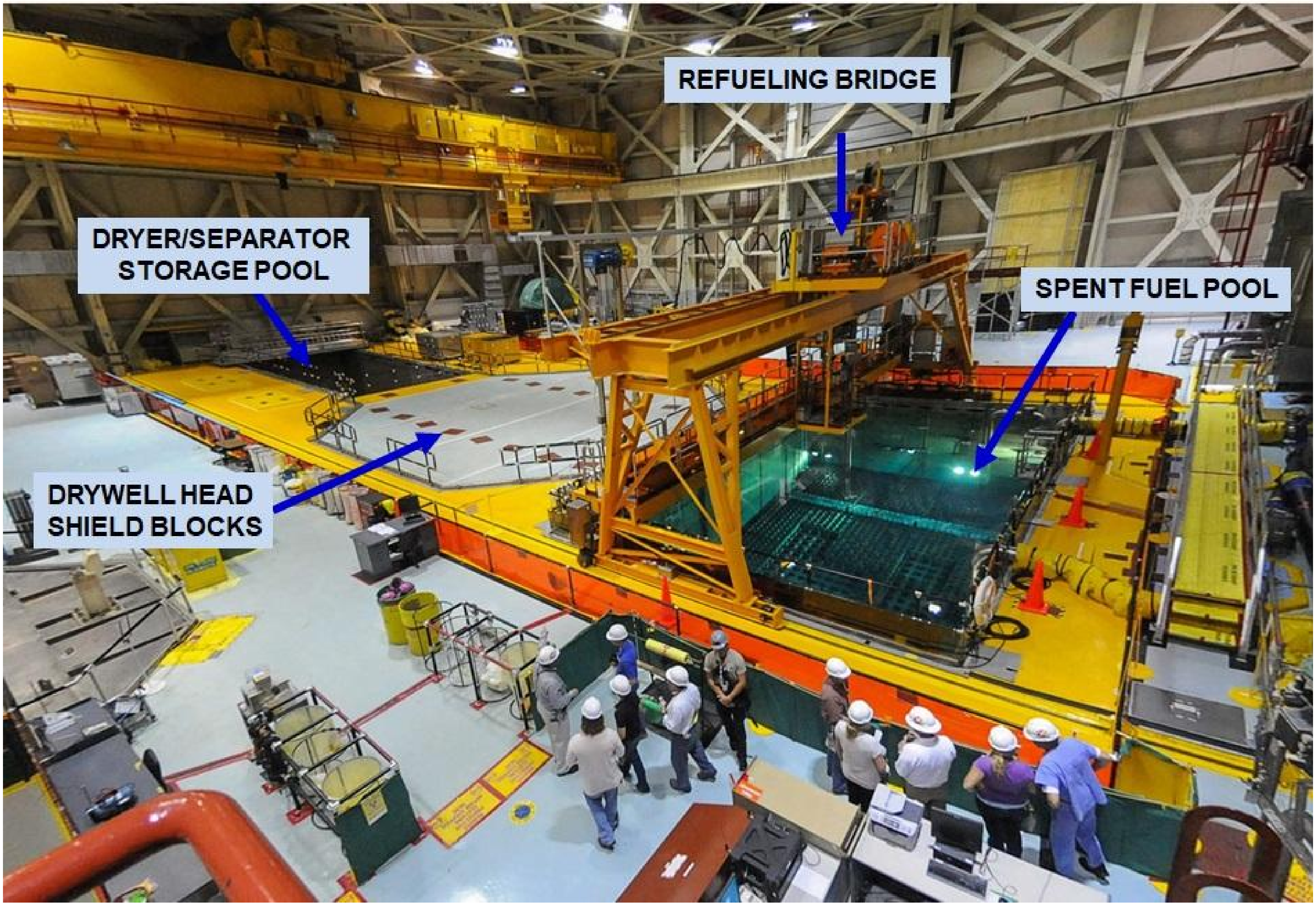


BWR Refueling Differences

The refueling processes in a BWR are not greatly different from that of a standard PWR. One big difference is that the spent fuel pool of the BWR is inside the secondary containment building, immediately next to the refueling pool, except for BWR-6 designs, which are similar to what was described for PWR facilities..



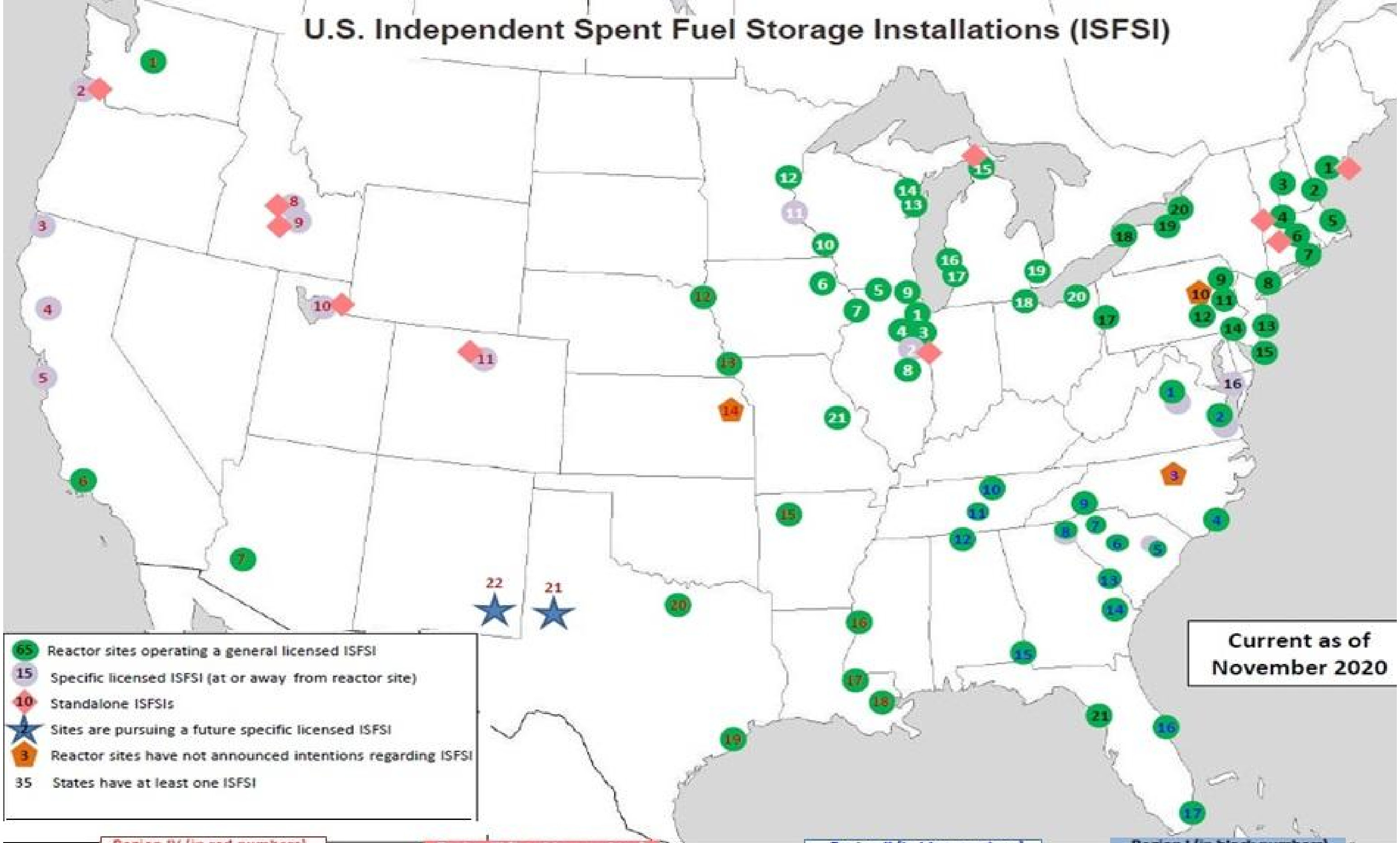
The following picture shows an actual BWR refueling floor preparing for refueling operations. Note that the drywell head shield blocks are still installed over top of the drywell.



Dry Fuel Storage (ISFSI)

Many utilities have begun building dry fuel storage facilities because their spent fuel pools have become full. Spent fuel that has been stored in the Spent Fuel Pool for at least a year can be removed and placed in heavy-duty, stainless steel cylinders, filled with inert gas, that are stored in a dry condition in a separate location at the plant. These storage areas are known as Independent Spent Fuel Storage Installations. These interim storage facilities can be licensed under either the existing Part 50 license for the plant, or a separate Part 72 license, depending on the design. The low levels of decay heat remaining in these dry-stored canisters are removed by natural circulation of air around the containers.

There are approximately 80 separate ISFSI sites in 37 states in the US at present. Until the final repository for high level waste (spent fuel) is built and operational, it is expected that the number of ISFSI sites will increase.



Region IV (in red numbers)

1) Columbia
2) Trojan
3) Humboldt Bay
4) Rancho Seco
5) Diablo Canyon
6) San Onofre
7) Palo Verde
8) DOE TMI-2 Storage
9) DOE Idaho Spent Fuel Facility
10) Private Fuel Storage
11) Ft. Saint Vrain
12) Ft. Calhoun
13) Cooper
14) Wolf Creek
15) Arkansas Nuclear One
16) Grand Gulf
17) River Bend
18) Waterford
19) South Texas Project
20) Comanche Peak
21) Interim Storage Partners
22) Eddy-Lea

Region III (in white numbers)

1) Dresden
2) GE Morris (wet)
3) Braidwood
4) LaSalle
5) Byron
6) Duane Arnold
7) Quad Cities
8) Clinton
9) Zion
10) LaCrosse
11) Prairie Island
12) Monticello
13) Point Beach
14) Kewaunee
15) Big Rock Pt.
16) Palisades
17) Cook
18) Davis Besse
19) Fermi
20) Perry
21) Callaway

Region II (in blue numbers)

1) North Anna
2) Surry
3) Shearon Harris
4) Brunswick
5) Robinson
6) Summer
7) Catawba
8) Oconee
9) McGuire
10) Watts Bar
11) Sequoyah
12) Browns Ferry
13) Vogtle
14) Hatch
15) Farley
16) St. Lucie
17) Turkey Point

Region I (in black numbers)

1) Maine Yankee
2) Seabrook
3) Vermont Yankee
4) Yankee Rowe
5) Pilgrim
6) Haddam Neck
7) Millstone
8) Indian Point
9) Susquehanna
10) Three Mile Island
11) Limerick
12) Peach Bottom
13) Oyster Creek
14) Hope Creek
15) Salem
16) Calvert Cliffs
17) Beaver Valley
18) Ginna
19) Nine Mile Pt.
20) FitzPatrick
21) Crystal River

Summary

Key Points

Key Points

- Before the actual refueling process takes place, plans are made for the refueling outage.
- After the plant is cooled down and depressurized, the reactor is disassembled.
- A polar crane is used to lift the vessel head.
- The vessel head is lifted and set down on a special stand where it is inspected.
- The refueling cavity is flooded to shield personnel from the extreme radioactivity of the core and components inside the vessel.
- The reactor vessel head is lifted and set down on a special stand within the containment.
- The flooding of the refueling cavity commences after the vessel head is lifted. The water level will be increased to a minimum of approximately 25 feet above the reactor vessel flange.
- Spent fuel is moved using the manipulator crane. After the first upender moves the fuel to a horizontal position, it is transferred through the fuel transfer tube. A second upender moves the fuel into a vertical position where the spent fuel bridge crane moves it into position.
- New fuel is transferred to the containment area by placing it into the spent fuel pool using the fuel building crane and the new fuel elevator. From the fuel pool, the spent fuel bridge crane will take the assembly and place it into the conveyor car. On the upender, the fuel is shifted horizontally and transferred through the fuel transfer tube.
- Plants may perform a full core offload instead of a fuel shuffle, depending on work requirements.
- After the fuel shuffle/replacement, the upper internals are replaced over the core and the control rod drive shafts are reconnected.
- The exhausted fuel is stored in the spent fuel pool, where it continues to produce heat.
- When the spent fuel pool was predicted to become full, many licensees opted to go the route of dry fuel storage. They constructed Independent Spent Fuel Storage Installations (ISFSI) to compensate for interim fuel storage until the Federal Government opens the Yucca Mountain (or other) long-term spent fuel storage facility.

Emergency Plan & Event Classifications

Chapter Overview

The NRC requires any applicant for a construction permit of a production or utilization facility to include in their preliminary safety analysis report a discussion of what emergency plans will be in place to manage Emergency Planning and Preparedness. The combined license requires a chapter in their final safety analysis of the plans. These plans are governed by 10CFR50, Appendix E. The emergency plans required for research and test reactors are slightly different.

These production/utilization plant plans are required to contain the following content:

- On-site and off-site organization
- Contacts and arrangements with local, State, and Federal agencies
- Protective measures to be taken within the site boundary and within Emergency Planning Zones (EPZ) to protect health and safety, including notifying the public.
- Emergency Facilities and Equipment for first aid, decontamination, and transport of on-site personnel to off-site treatment facilities.
- Provisions for emergency treatment at off-site facilities for injuries due to licensed activities.
- Training programs for licensee personnel assigned emergency duties, as well as for non-licensee personnel whose assistance during the emergency may be needed.
- A preliminary analysis projecting the time and means to notify State and local governments and the public in the event of an emergency. Power reactors will analyze the time required to evacuate various sectors within the plume pathway of the EPZ.
- An analysis showing facilities, systems, and methods necessary for identifying the degree and scope of the emergency, including real-time meteorological information and dispatch of radiological monitoring teams within the EPZ.

The emergency plan event classifications were established to provide prompt notification to the proper authorities of both minor and major events. Licensees determine which emergency class to declare, based on Emergency Action Levels (EAL) contained within their emergency plan. Depending on the severity of the event, the actions taken could range from notifying the NRC to the staffing of the emergency response facilities and the notifying of local, State, and Federal agencies.

The following chapter will describe the event classifications in ascending order and the purpose of each condition.

Objectives

At the end of this chapter, you will be able to:

- Rank the four emergency classifications in ascending order of severity.
- Describe the four emergency classifications.

Estimated time to complete this chapter:

25 minutes

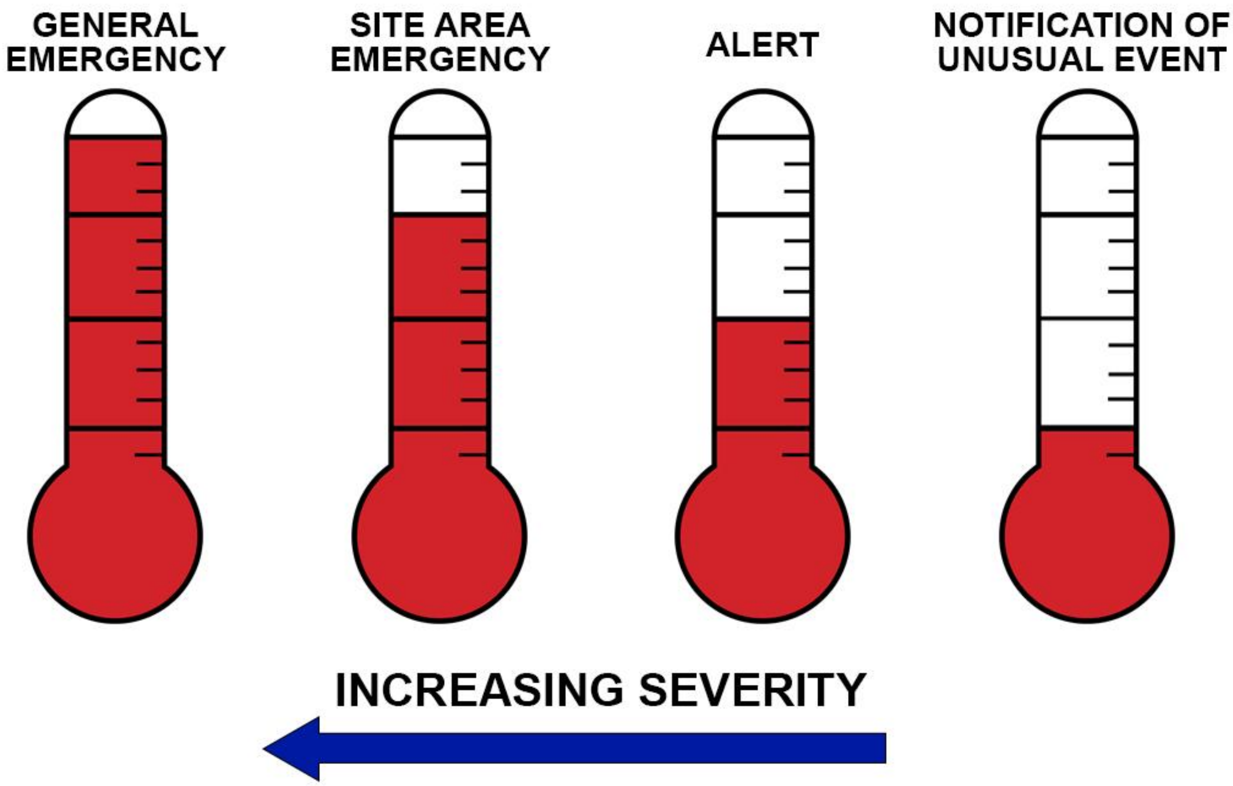
Ranked Emergency Plan Event Classifications

Ranking

An Emergency Classification is a condition that indicates a level of risk to the public. Each operating nuclear power plant is required to include in its emergency plans a standard Emergency Classification and Emergency Action Level (EAL) scheme. An EAL is a pre-determined, site-specific, observable threshold for a plant condition that places the plant in a given emergency classification.

The vast majority of events reported to the NRC are routine in nature and do not require activation of our incident response program. The Emergency Operations Center is continuously manned and is prepared to activate the NRC's incident response plan as needed.

Both nuclear power plants and research & test reactors use the four (4) emergency classifications shown below, indicating the order of severity. Nuclear materials & fuel cycle facilities utilize two classifications: Alert & Site Area Emergency.



Description of Emergency Class

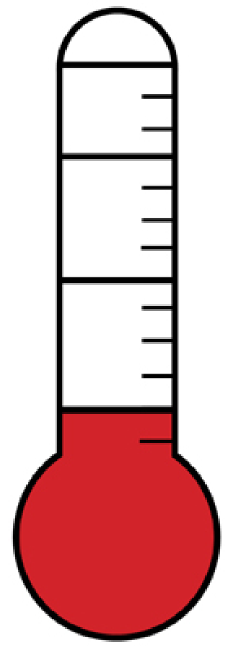
Notification of Unusual Event (NOUE)

Under this category, events are in process or have occurred that indicate potential degradation in the level of safety of the plant. The EAL Guidance has the primary threshold for Unusual Events as operation outside the safety envelope of the plant, as defined in its Technical Specifications (license requirements for operating). The term UNUSUAL EVENT (UE) is often used interchangeably with NOUE.

No release of radioactive material requiring offsite response or monitoring is expected unless further degradation occurs. Some reasons to initiate a NOUE might be an earthquake felt onsite or a security threat.

The reasons for use of this emergency classification are to:

- Assure that the first step in any response later found to be necessary has been carried out
- Bring the operating staff to a state of readiness
- Provide systematic handling of NOUE notification and decision making, including notifying the NRC and local government entities.

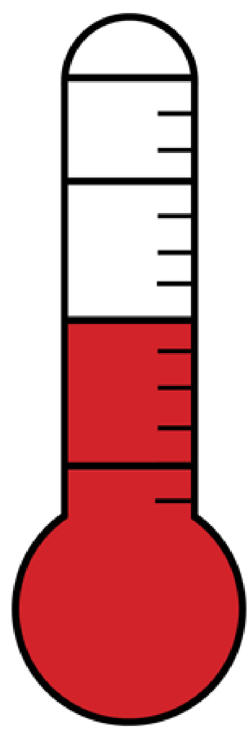


Alert

If an Alert is declared, events are in process or have occurred that involve an actual or potential substantial degradation in the level of safety of the plant, including security events that threaten site personnel or plant equipment. Any releases of radioactive material from the plant are expected to be limited to a small fraction of the Environmental Protection Agency (EPA) protective action guides (PAG). Examples of initial conditions causing an Alert are fuel handling accidents or excessive primary leakage. An Alert generally denotes a condition where one of the three barriers to fission product release is, or may be, compromised (fuel cladding, reactor coolant system, containment).

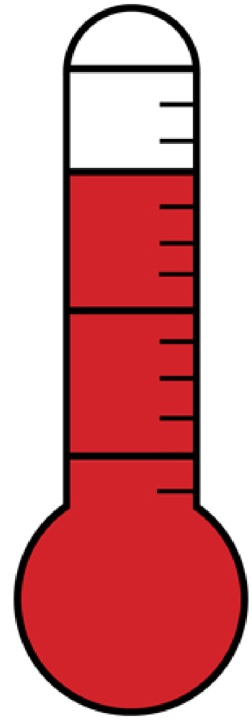
The reasons for use of this emergency classification are to:

- Assure that emergency personnel are readily available to respond if the situation warrants
- Perform confirmatory radiation monitoring if required
- Provide offsite agencies with current information



Site Area Emergency

A Site Area Emergency involves events in process or events that have occurred that result in actual or likely major failures of plant functions needed for protection of the public, including security events involving intentional damage or malicious acts leading to likely failure of safety related equipment. Any releases of radioactive materials are not expected to exceed the EPA protective action guides (PAG) except near the site boundary. Some reasons for a licensee to declare a Site Area Emergency are a station blackout or loss of DC power for greater than 15 minutes in duration.

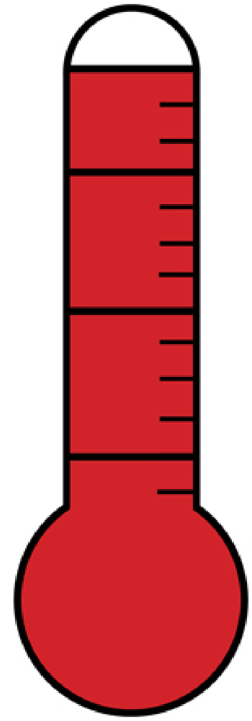


The declaration of a Site Area Emergency will:

- Assure that appropriate response centers are manned
- Assure that monitoring teams are dispatched
- Assure that personnel required for evacuation of near-site areas are available if needed
- Provide consultation with offsite authorities
- Provide updates to the public through offsite authorities

General Emergency

A General Emergency involves actual or imminent substantial core damage or melting of reactor fuel with the potential for loss of containment integrity. Radioactive releases during a General Emergency can reasonably be expected to exceed the EPA protective action guides for more than the immediate site area. Some reasons to initiate a General Emergency are a loss of fuel cladding and reactor coolant system with a high potential for loss of containment, or a loss of coolant accident with failure of the emergency core cooling systems to perform adequately.



The key difference between a General Emergency and Site Area Emergency is whether or not exposure levels to the radioactive release plume will exceed EPA limits. A General Emergency is the only Action Level that will potentially result in evacuation of the public.

The reasons for use of the General Emergency are to:

- Be prepared to initiate predetermined protective action recommendations (PAR) for the public
  - Licensees recommend PARs
  - NRC confirms/refutes reasonableness of recommendation
  - State or local government initiates the PAR
- Provide continuous dose assessment based upon available information
- Initiate additional measures as indicated by actual or potential releases
- Provide consultation with offsite authorities
- Provide updates for the public through offsite authorities

Summary

Key Points

- The four levels utilized by power & test reactor facilities, in ascending order of severity, are: Notification of Unusual Event (NOUE), Alert, Site Area Emergency, and General Emergency. In each case, the licensee is required to notify the NRC Operations Center within 1 hour (60 minutes).
- Within the ECLs are EALs, which are pre-determined, site-specific observable thresholds for placing the plant in an ECL.
- Notification of Unusual Event indicates potential degradation in the level of safety of the plant.
- Alerts involve an actual or potential substantial degradation in the level of safety of the plant.
- Site Area Emergencies result in actual or likely major failures of plant functions needed for protection of the public.
- General Emergencies involve actual or imminent substantial core damage or melting of reactor fuel with the potential for loss of containment integrity. They also may require evacuation of the public from affected areas.

The Three Mile Island Accident

Chapter Overview

On March 24, 1978 the worst accident in the history of United States (U.S.) commercial nuclear power occurred at the Three Mile Island (TMI) Nuclear Power Plant. This watershed event caused many changes in the U.S. nuclear power industry and the NRC. Some of the key changes include:

- Placement of on-site NRC resident inspectors at each operating nuclear power plant
- Improved operator training programs
- Improved main control room panel designs
- New system-based emergency operating procedures (EOPs)

In this chapter, we will examine the events that eventually led to the partial meltdown of the TMI Unit-2 core. We will consider what happened inside this pressurized water reactor (PWR) and how that resulted in the release of fission products to the surrounding environment.

Location of Three Mile Island

TMI is located in south central Pennsylvania. It is approximately 10 miles east of Harrisburg, PA, and 100 miles west of Philadelphia, PA. The power generating station is located on, and named for, Three Mile Island on the Susquehanna River.

Unit 2 is located on the western-most end of the island, while Unit 1 is located on the eastern end. Unit 1 continues to provide power today, while Unit 2 has been shut down and partially dismantled.

Objectives

After completing this chapter, you will be able to:

- Describe the Three Mile Island accident.
- Identify the precipitating event that caused the accident at Three Mile Island Unit 2.
- Sequence the basic events that led to the meltdown of the Three Mile Island Unit 2 core.
- Describe the release of core fission products to the environment.

Estimated time to complete this chapter:

35 minutes

Release of Fission Products

Release of Fission Products in the Containment Building

Fission products (mostly gases) escaped from the damaged reactor core into the RCS. Due to the stuck open PORV, the pressurizer relief tank (PRT) pressure increased to the bursting point of its installed rupture disk.

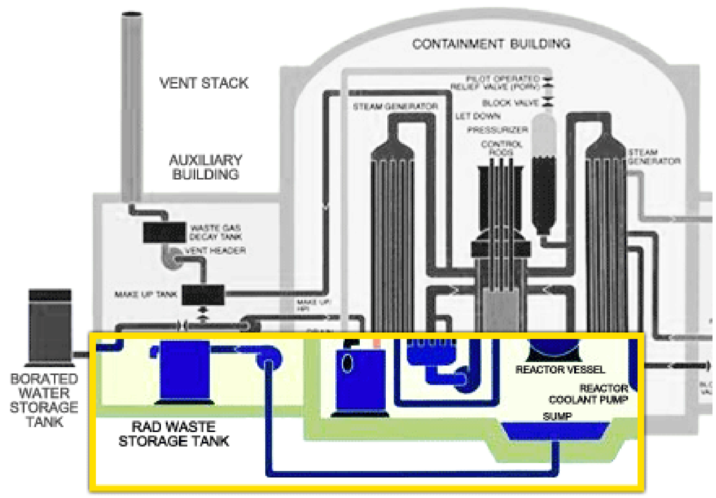
The reactor coolant, containing these fission products, now had a direct release path into the containment building atmosphere. The rupture of the PRT severely contaminated the containment with the fission products released from the fuel cladding.

Again, control panel designs prevented the reactor operators from easily seeing that the PRT had ruptured. These indications were located on a control panel that was located behind, and out of sight, of the main control panel.

Transfer of Fission Products to the Auxiliary Building

The coolant escaping into the containment building was being collected in the containment sump. After reaching a high sump level, the sump pump automatically started and pumped the water to the Auxiliary Building (AB). The fission product gases trapped in the coolant were released into the AB atmosphere and then entered the AB ventilation system.

Due to the high concentration of fission products in the water from the containment sump, as well as the reactor coolant water in the purification system, the AB atmosphere contained a high enough level of radioactive gases that the AB had to be evacuated.

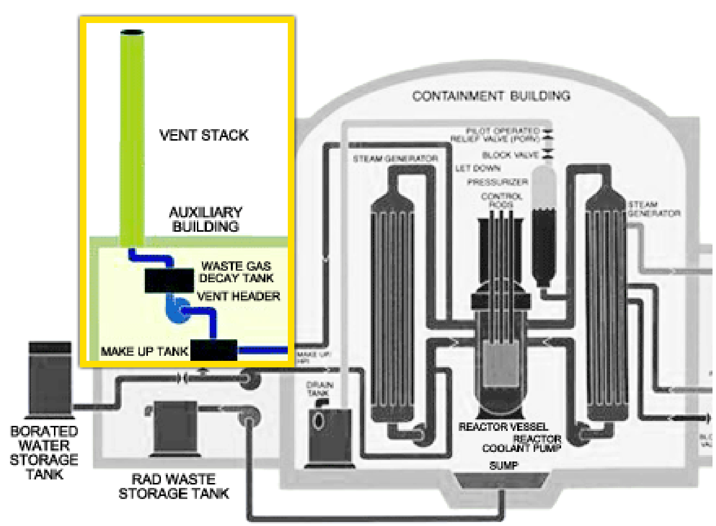


Escape to the Surrounding Environment

The major release path to the outside environment was via the waste gas system. The fission product gases contained in the letdown coolant were being stripped out in the volume control tank, which is then vented to the waste gas system. A vent in the waste gas system allowed the gases to be blown out the ventilation stack via the AB ventilation system.

The radiological consequences for the surrounding area associated with the accident at TMI2 were:

- Maximum projected offsite dose: 100 millirem
- Average dose to population: Approximately 1.4 millirem per person
- Projected additional cancers: 0 to 1

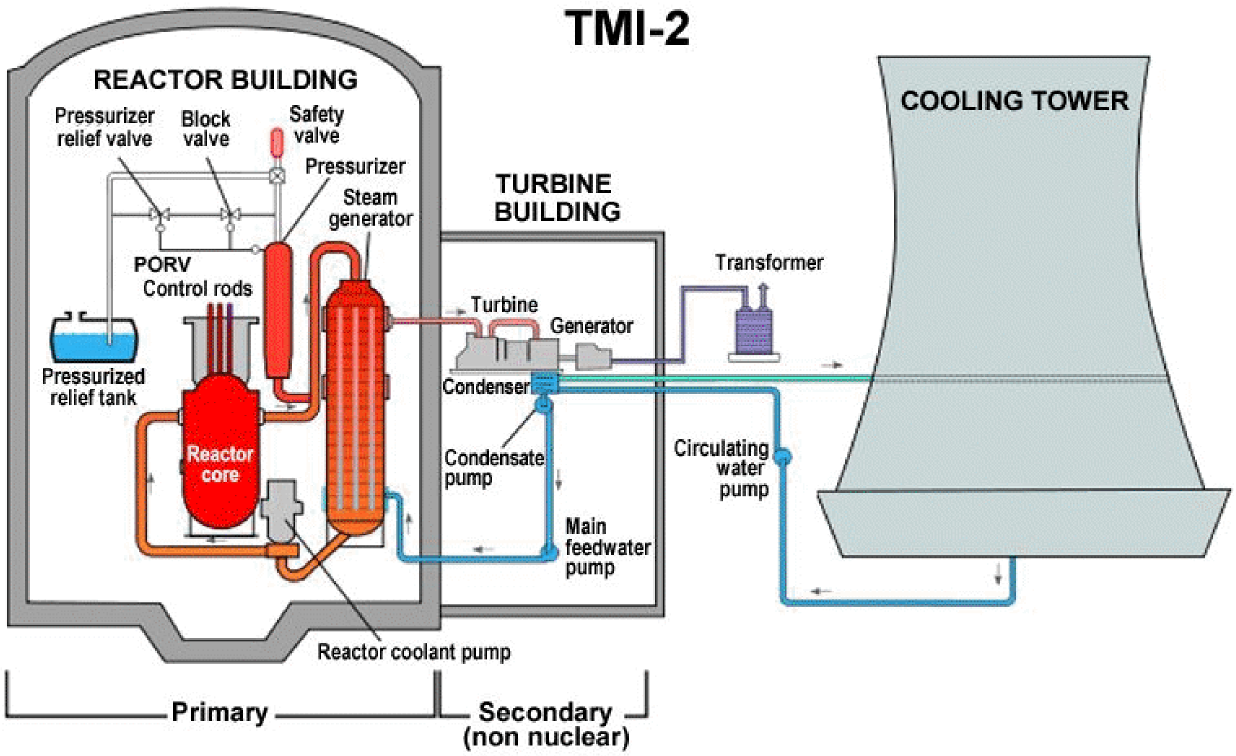


Inside the Reactor System

Design of TMI

TMI is a two-unit PWR of the Babcock & Wilcox design. The reactor coolant system consists of the reactor vessel, two steam generators, four reactor coolant pumps, and the pressurizer. Each unit functions independently and is served by two cooling towers.

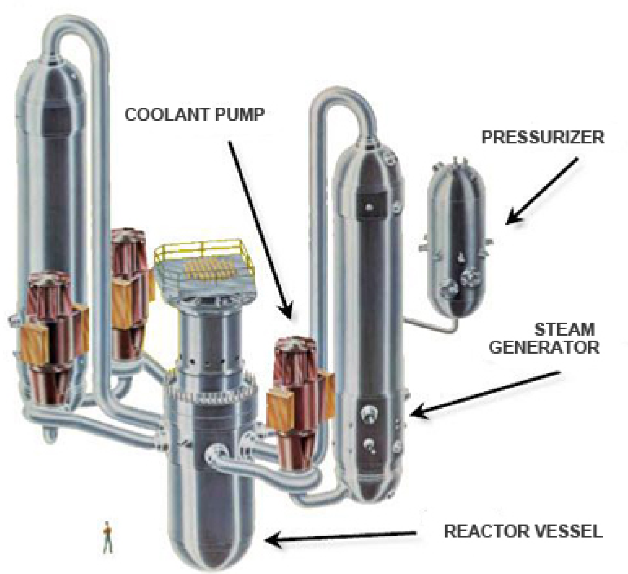
Like all U.S. reactors, TMI was designed for safety and to automatically shut down in the event of loss of coolant accident (LOCA). Unfortunately, the safety measures did not address all of the possible initiating events that might cause a LOCA.



Babcock & Wilcox Design

A Babcock & Wilcox pressurized water reactor has two, once-through steam generators (SG); four reactor coolant pumps; and a pressurizer.

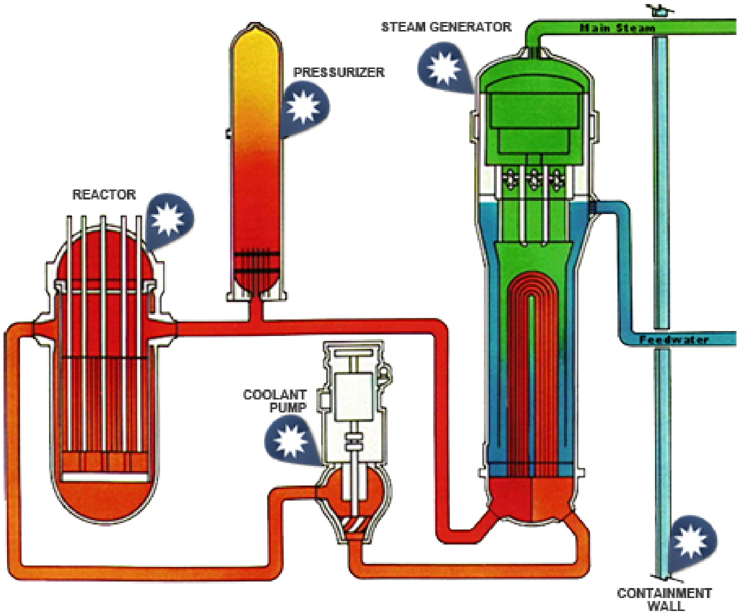
An issue with B&W SGs is that they are more susceptible to loss of feedwater events, than those of Westinghouse and Combustion Engineering SGs.



Review of Typical PWR Reactor System

Take a moment to review basics of the primary system in a typical PWR. Remember that the primary system includes everything related to the reactor system.

Select the reactor, pressurizer, coolant pump, steam generator, and containment wall for an explanation of each component's role.



Reactor Vessel

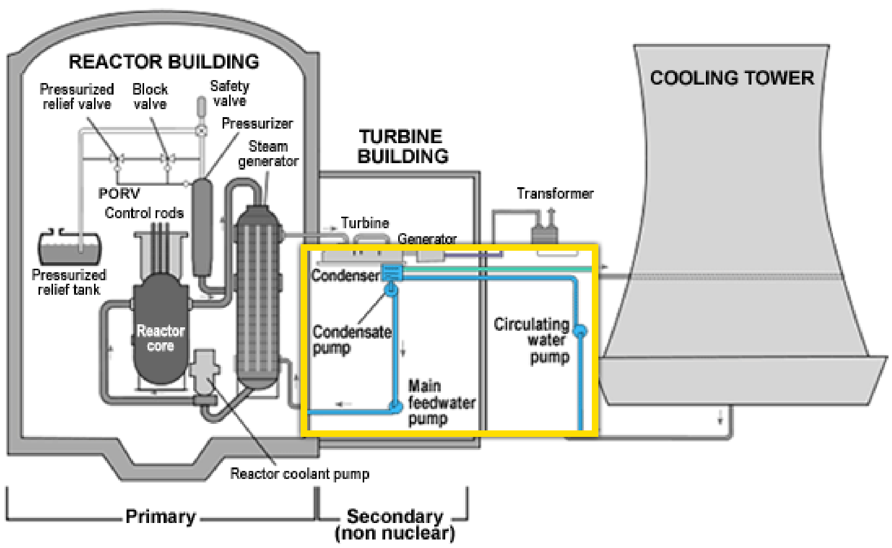
The reactor vessel is a large, forged iron cylinder with a stainless steel interior coating that contains the core, including fuel, control rods, and moderating material (i.e., water in the U.S.).

Loss of Feedwater and Failure of the Auxiliary Feedwater System

The incident began in the secondary (steam) section of the plant when the feedwater pumps malfunctioned. A problem with the condensate polishing (clean-up) system caused a loss of suction to the feedwater pumps. This caused them to trip, interrupting feed flow to the steam generator. As the water level lowered in the steam generator, the reactor automatically tripped (scrammed) and shut down per its design.

Although the reactor successfully shut down, the core's decay heat caused the reactor coolant temperature and pressure to rise. After a trip, this decay heat would be removed from the reactor coolant system by the auxiliary feedwater system providing feed to the steam generators. However, the auxiliary feedwater system was improperly aligned and could not immediately feed the steam generators.

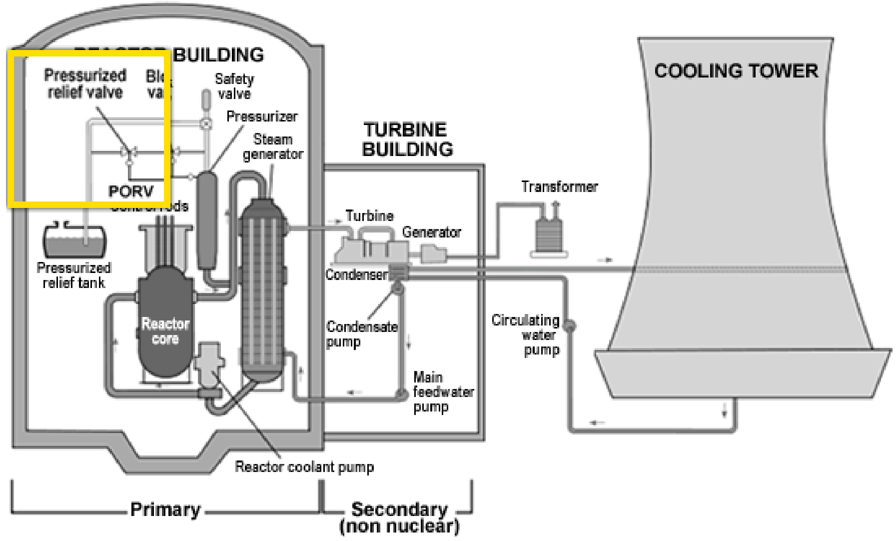
The reactor coolant temperature and pressure continued to rise due to the decay heat not being adequately removed.



The Power Operated Relief Valve (PORV) Opens

In response to the rising primary pressure, the power operated relief valve (PORV) on the pressurizer automatically opened. The PORV is a pressure relief value that opens and closes automatically to help limit the pressure in the pressurizer below that of the code safety valves. The PORV opened, as intended, to prevent the pressure in the pressurizer and reactor coolant system from reaching a dangerously high level.

However, once the PORV failed to close, it created what would be termed a LOCA (loss of coolant accident)...a 'hole' in the RCS boundary.



The PORV Malfunctions

The PORV malfunctioned and stayed stuck in a partially open position. If the PORV had functioned properly, the incident would have been an uncomplicated trip and the plant most likely would have been recovered in an uneventful manner.

However, the stuck-open PORV caused Reactor Coolant System (RCS) pressure to lower while the water level in the pressurizer continued to rise, leading to a gradual loss of reactor coolant pressure in the core. This allowed coolant in the core to start to boil, and the steam ‘bubble’ shifted from the top of the pressurizer to the reactor vessel head.

At this point, since the water in the pressurizer was high, the operators assumed that the core water level was satisfactory. They had no abnormal indications on the control panels alerting them that there was a problem in the core; there was no instrumentation that showed the actual level in the core itself while in this condition. Additionally, the position indicator for the PORV showed the **demand** state (CLOSED) rather than the actual state.

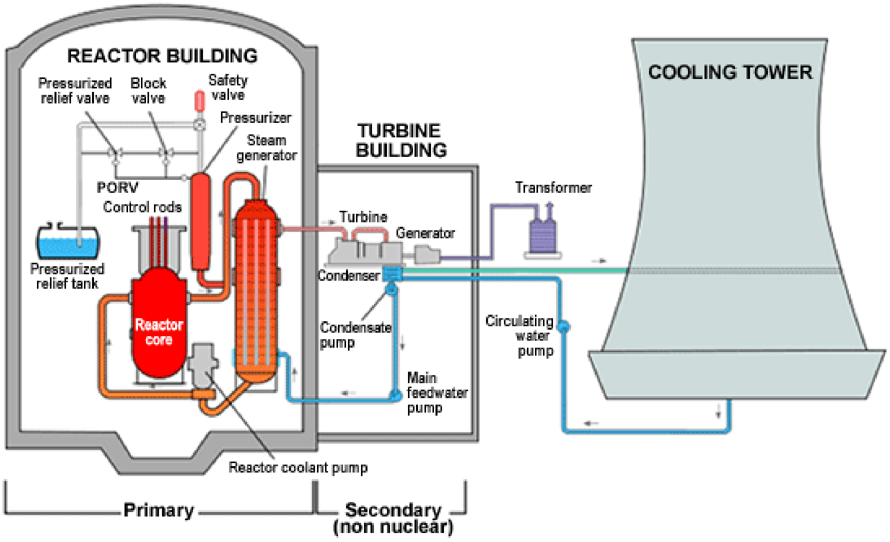
After sensing the loss of reactor coolant pressure, safety equipment automatically started. Since the operators believed the core to be covered, they stopped the safety equipment in order to prevent the pressurizer from overfilling with water. Their training made them especially concerned about a full pressurizer and the pressure control problems that would result from the ‘solid’ condition.

Removal of More Water Leads to a Loss of Coolant Accident

In response to the high water level in the pressurizer, the operators increased the amount of water being removed from the RCS (letdown) via the purification system in an effort to keep the pressurizer from overfilling. The effect of increased letdown, coupled with the stuck-open primary PORV, led to a loss of coolant accident (LOCA) condition.

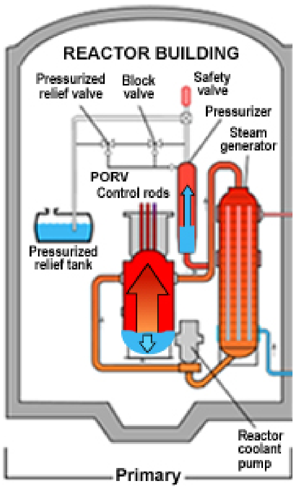
Remember, the amount of pressure in the core is related to the control of temperature in the core. To control pressure in the core, the operators manipulate the ratio of water and steam in the pressurizer as well as its temperature.

Normal Conditions



Under normal conditions, consistent temperature and pressure are controlled throughout the primary system.

LOCA Phase #1

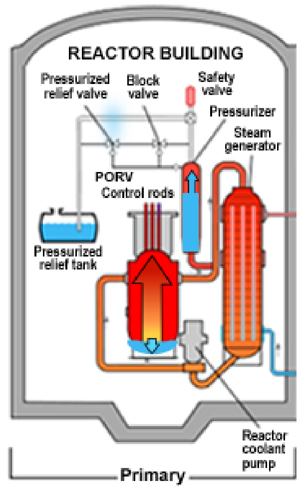


Less steam in the pressurizer due to the PORV malfunction results in more water in the pressurizer.

More water in the pressurizer means less water in the core.

Less water in the core means a higher temperature in the core.

LOCA Phase #2



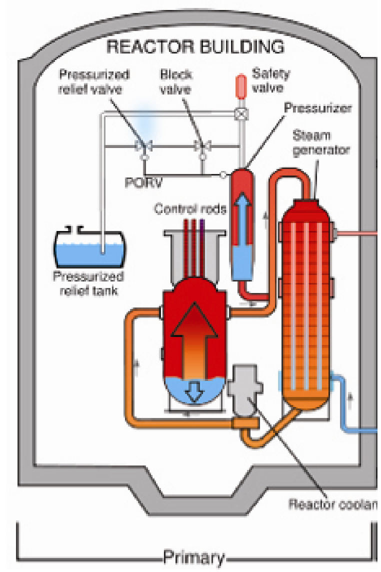
Due to the PORV malfunctioning, steam escapes from the pressurizer so more water flows into it to maintain the pressure. Even more water in the pressurizer results in less water in the core.

The temperature in the core continues to rise as the water decreases in the core.

The Reactor Coolant Boiled in the Core

As the amount of water in the pressurizer increased, and the reactor coolant was diverted to the purification system, the resultant low pressure and high temperature caused the remaining reactor coolant to boil inside the core. In addition, the low coolant pressure caused cavitation and high vibrations in the reactor coolant pumps. The operators were trained to turn off the reactor coolant pumps in this situation, which they did. The concern here was asset protection: replacing RCPs would cost the licensee a large amount of money.

Up until the point that the reactor coolant pumps were secured, enough water and steam was flowing through the core to remove its decay heat. Once secured, the amount of reactor coolant flowing through the core was inadequate to remove the decay heat generated in the fuel and the remaining coolant in the core boiled away.



Overheating Fuel Rods Leads to Core Damage

With little or no cooling available to remove decay heat, the reactor fuel rods started to overheat, crack, and break down. The lack of decay heat removal caused the fuel pellet temperature to rise to the point that the fuel cladding began to melt.

The cladding, which is the first barrier to fission product release, was breached, releasing radioactive fission products into the reactor coolant.

The reactor core experienced a significant amount of fuel melt at this time, reaching almost 50%. A portion of the molten core flowed sideways out to the edges of the core after melting through the stainless steel core support assembly. That material (a mixture of melted fuel, clad, and structural steel known as corium) then flowed down to a lower portion of the reactor vessel, where it cooled and became solid material again. Some of the lower portion of the fuel assemblies—those that had sufficient reactor coolant during the accident—remained intact.

Restarting Reactor Coolant Pumps Causes Severe Core Damage

Later, the operators recognized that the PORV was partially open and isolated it; the reactor coolant pumps were also restarted. Restarting the reactor coolant pumps caused relatively cold water to be pumped onto the now very hot and very brittle fuel rods, causing further severe damage to the core.

Years later, television cameras lowered into the reactor core revealed a large void where the top half of the core had been located. Rubble, including fuel pellets and fuel rod debris, was seen on the top of the lower half of the core.

Click below to view the videos of the core damage.

- ▶ Upper Core Damage
- ▶ Plenum Damage
- ▶ Work Platform
- ▶ Removing Top Debris
- ▶ Core Bore
- ▶ Lower Core Damage
- ▶ Removing Lower Debris

▶ Summary

Summary

Key Points

- TMI was the worst accident in U.S. nuclear power history. It resulted in significant changes to both the U.S. nuclear power industry and the NRC.

- Epidemiological studies analyzing the rate of cancer in and around the area since the accident determined there was not a statistically significant increase in the rate and thus no direct causal connection linking the accident with these cancers has been substantiated

- The accident was caused and exacerbated by a combination of factors, including equipment malfunctions, operator error, and poor instrumentation design.

- The sequence of events inside the reactor system during the accident was:
  - Loss of feedwater due to failure of main (normal) feedwater system, leading to a reactor trip
  - The PORV opens (per design), but malfunctions and does not reclose
  - Loss of coolant through open PORV and increased letdown results in a LOCA
  - LOCA and stopping reactor coolant pumps leads to the core boiling, uncovering, and overheating
  - Overheating fuel pellets lead to damage to the core and cladding failure
  - Restarting reactor coolant pumps makes core damage worse

- The sequence of events that led to the release of fission products to the environment was:
  - Release of fission products in the containment building because of the ruptured PRT
  - Transfer of fission products to the AB through the containment sump
  - Escape from the AB to the outside environment via the waste gas system

The Chernobyl Accident

Chapter Overview

The April 1986 disaster at the Chernobyl nuclear power plant in the Ukraine was the product of a flawed Soviet reactor design coupled with serious mistakes made by the plant operators and staff. It was a direct consequence of Cold War isolation and the resulting lack of any safety culture.

The following chapter will briefly describe the Soviet Union's RBMK-1000 reactor design and discuss the events leading up to, and immediately following, the Chernobyl-Unit 4 accident.

Objectives

At the end of this chapter, you will be able to:

- Differentiate between the Chernobyl design and United States (U.S.) light water reactors (LWRs) that reduces the likelihood of a similar accident in the U.S.
- Sequence the basic events that led to the accident at Chernobyl

Estimated time to complete this chapter:

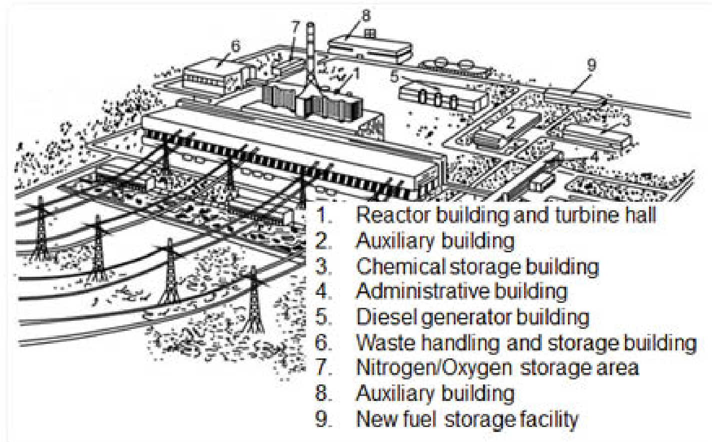
35 minutes

Soviet Reactor Design

RBMK-1000 Schematic

The graphic displays the schematic of a typical layout for an RBMK-1000 site. The major structures are:

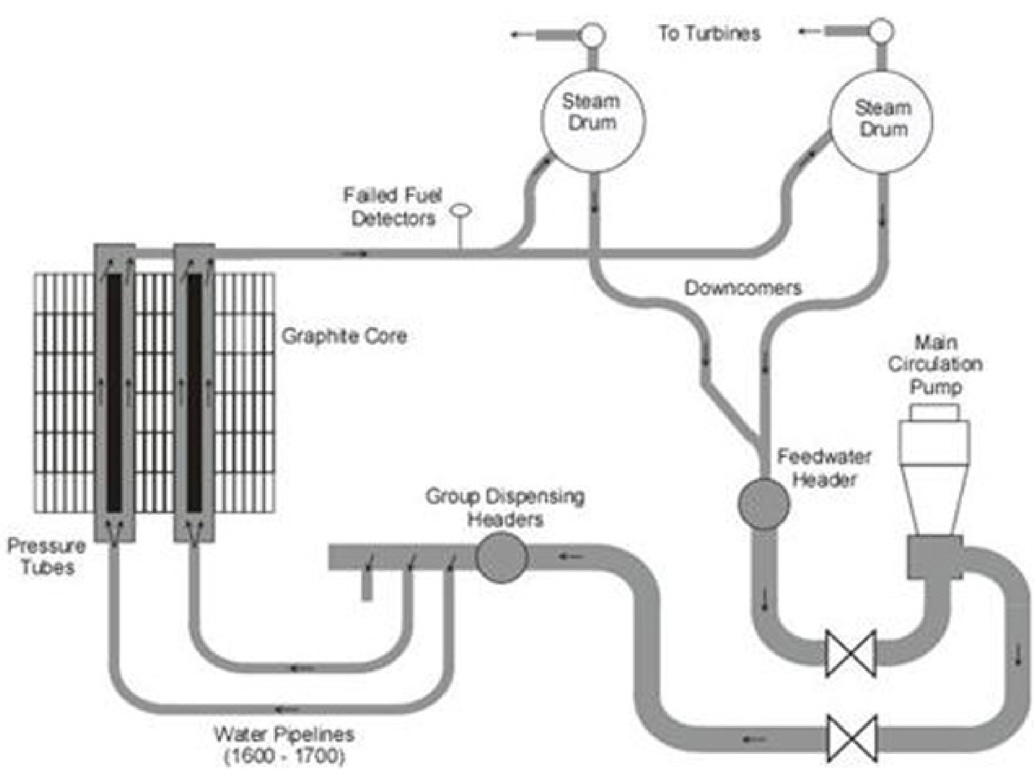
- Reactor building and turbine hall
- Auxiliary buildings
- Chemical storage building
- Administrative building
- Diesel generator building
- Waste handling and storage building
- Nitrogen/Oxygen storage area
- New fuel storage facility



RBMK-1000 Reactor

The RBMK-1000 is a boiling water, pressure tube, graphite-moderated reactor. The cooling medium is water, which is converted into a steam/water mixture as it passes by the fuel rods. The coolant flows inside 1661 sealed pressure tubes. The steam/water mixture is physically separated in the steam drums. The separated steam is routed to the turbine generators and the separated water is combined with the feedwater from the main condenser and pumped back to the reactor via recirculation pumps. The coolant flows inside the pressure tubes from the bottom to the top. The pressure tubes are surrounded by large blocks of graphite. The graphite serves to slow the neutrons down to the energy required to cause fission (acts as the moderator).

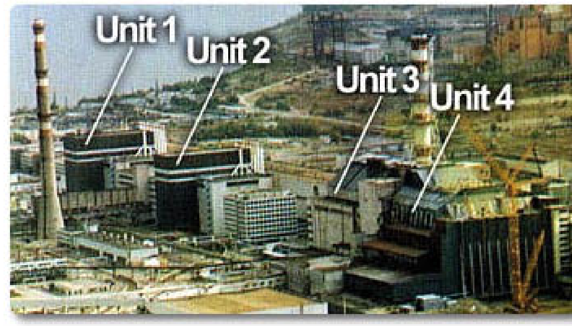
The RBMK-1000 reactor does not have to be shut down to be refueled. Each pressure tube can be individually isolated, opened, and refueled by remotely controlled equipment while the remainder of the pressure tubes continue to operate at power.



Events of Accident

Special Test

At Chernobyl Unit 4, a special test (written by non-nuclear trained electrical engineers) was to be performed to help evaluate a new voltage regulating system. This system would allow plant emergency equipment to be powered by the plant's turbine-driven electrical generator as it slowed following a turbine shutdown. Since the test's authors did not consider the test to be of safety significance, the test procedure was not submitted for the usual safety reviews. The same test had been rejected by the staff of another plant as too risky.



The test called for a reduction in power to 50% (1600 MW), blocking (overriding) the automatic start signal for the plant's emergency core cooling system (ECCS) and connecting four of the reactor's eight recirculation pumps to the turbine generator under test. Normally, the reactor would trip following a turbine trip signal. To prevent this and allow the performance of a second test if needed, the operators took several compensatory measures that overrode/blocked normal safety functions.

Faulty Test Design

Due to a faulty test design (coupled with operator error), the planned power level of 50% was not maintained and power was allowed to drop too low. In an attempt to continue the test, operators tried to raise power to the proper level but a cascading effect ensued within the reactor causing an uncontrollable core power rise and catastrophic steam explosion.

In the aftermath, television crews documented the RBMK-1000 reactor design, the failed Soviet reactor test, and the operator errors that occurred. The end result of the Chernobyl test led to the worst nuclear event seen in modern times.

Below are videos pertaining to different aspects of the accident.

Plant Design

▶ View Transcript

Test Events

▶ View Transcript

Accident Examined

▶ View Transcript

Reactor Explosion

▶ View Transcript

Explosion Aftermath

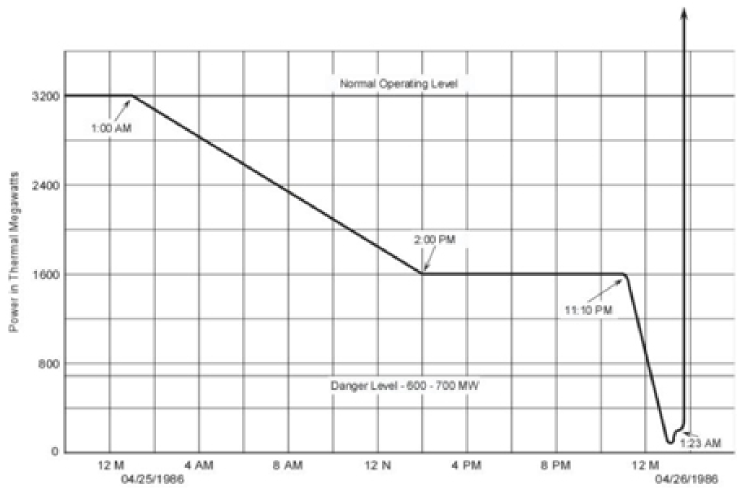
▶ View Transcript

Operator Errors

Due to operator error, the planned power level of 50% was not maintained and power was inadvertently allowed to drop to about 1% (30 Mw). The operators tried to raise power to the proper level by removing nearly all of the control rods from the core, but could only increase power to about 6%.

In an RBMK reactor, extended operation below 20% is not permitted due to reactor instability (steam bubbles in the coolant increase reactor power and an increase in the reactor power creates more steam bubbles, which causes the power to increase even further: a positive void reactivity coefficient). The operators decided to continue the test despite conditions that called for an immediate reactor shutdown.

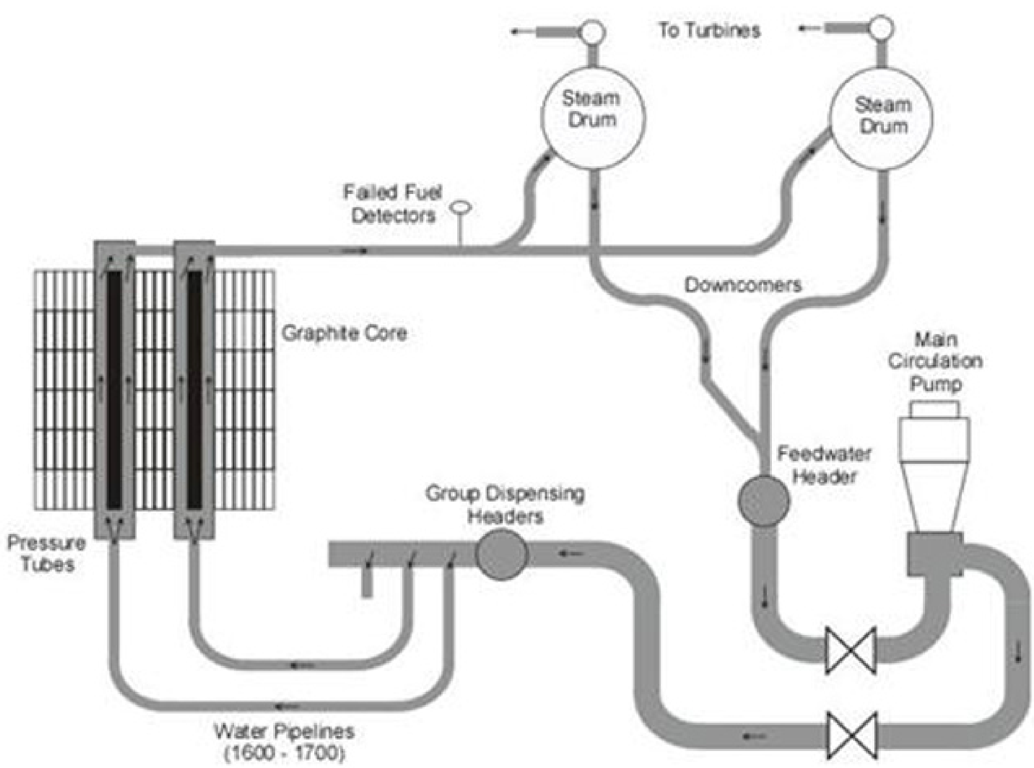
This is the opposite of U.S. light water reactors where a negative void coefficient of reactivity is maintained. In U.S. reactors, if power and/or temperature go up, the moderator (coolant) expands and becomes less dense. This leads to a loss of moderation (thermalization) of neutrons, which causes power to go down. Hence, U.S. light water reactors are more inherently stable.



Test Procedures

In accordance with the test procedure the operators turned on all reactor recirculation pumps. Due to the very low power level at which the plant was operating, the water entering the reactor was mostly from the steam separators and, therefore, was very nearly at its boiling point even before reaching the fuel assemblies. Upon reaching the fuel area it boiled immediately, creating voids that increased power levels.

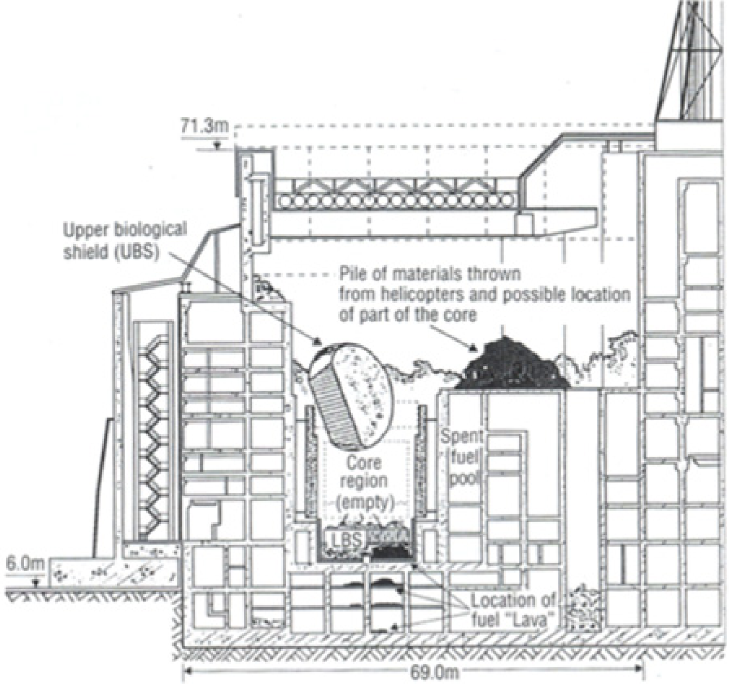
Continuing the test, the operators then tripped the turbine generator and subsequently the principal method of heat removal was isolated. The temperature increased in the fuel channels causing a significant increase in the steam bubble (void) formation. The increased boiling caused reactor power to rise rapidly. Operators tried to manually trip the reactor, but power was increasing far too rapidly. The fuel temperatures increased so much, and so quickly, that the internal fuel rod pressure burst the fuel cladding, releasing extremely hot fuel fragments into the coolant channels.



Reactor Overpower & Steam Explosion

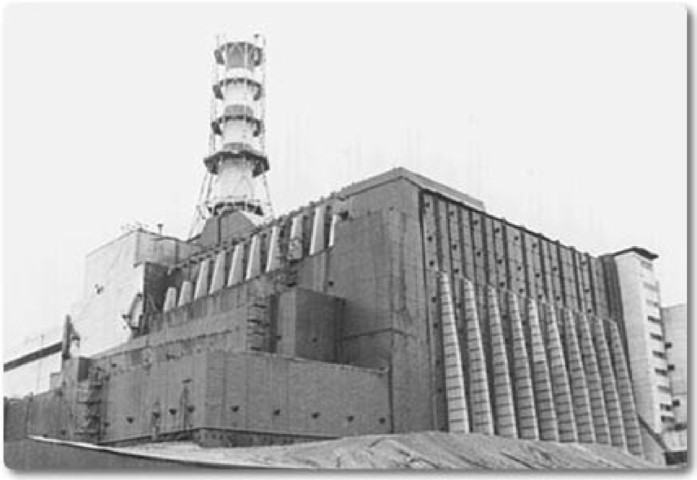
Extremely high pressures resulted from the mixing of the water with the intensely hot uranium fuel. Most, if not all, of the coolant pressure tubes ruptured immediately. The subsequent steam release (steam explosion) was more than sufficient to destroy the area above and around the Unit 4 reactor, and to spread hot pieces of uranium fuel and graphite moderator onto the adjacent buildings. Remember that the RBMK design assumed that the turbine building only needed to be able to contain the rupture of 1 of the 1661 fuel tubes.

A second explosion, occurring about 3 seconds later and much more powerful than the first, caused further damage. About thirty fires were ignited by the materials ejected during the two steam explosions.



Aftermath

Chernobyl unit 4 was enclosed in a large concrete shelter, known as a sarcophagus, which was quickly erected to allow continuing operation of the other reactors at the plant. However, the structure was neither strong nor durable. The International Shelter Implementation Plan in the 1990s involved raising money for remedial work, including removal of the fuel-containing materials. Some major work on the shelter was carried out in 1998 and 1999. Approximately 200 tons of highly radioactive material remains deep within it and this poses a significant environmental hazard until it is better contained.

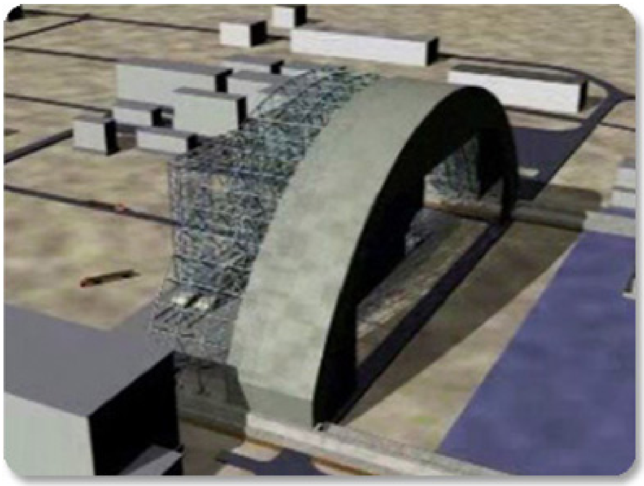


Future Plans

A New Safe Confinement structure was planned to be built by the end of 2011 and then moved into place on rails. It is an 18,000-ton metal arch, 350 feet high, 650 feet long, and spanning 900 feet to cover both Unit 4 and the hastily built 1986 structure.

The Chernobyl New Safe Containment rolled into place in November 2016, which allows for the dismantling of the sarcophagus and for radioactive material to be removed. The containment was expected to replace the existing sarcophagus in 2015. However, delays and a €100 million funding gap caused a year-long delay, before being moved into place in November 2016.

New Safe Containment Building:



Modifications have been made to overcome deficiencies in all the RBMK reactors still in operation. In these reactors, originally, the nuclear chain reaction and power output could increase if cooling water was lost or turned to steam—in contrast to most Western designs. It was this effect that led to the uncontrolled power surge that resulted in the destruction of Chernobyl Unit 4.

The other three reactors at Chernobyl are all, presently, shut down. However, fuel still remains on site and workers are still present to ensure safety of the existing cores. It is estimated that the Ukraine will have the area completely cleared and decontaminated sometime in 2065.

Summary

Key Points

- A special test, later proven faulty, was written to test emergency equipment within the RBMK-1000 reactor at the Chernobyl facility. The test, written by non-nuclear trained engineers, was not considered safety significant.
- Operator errors allowed the power output to drop to 1% during operation of the test. However, the RBMK-1000 was proven unstable if operated at a power output level of 20% or less.
- Test procedures required the recirculating pumps to be turned on while the turbine generators were tripped, removing the cooling capabilities of the circulating water loop.
- The increase in boiling water in the circulating water loop caused reactor power to rise uncontrollably. An attempt to trip the reactor was ineffective.
- The end result of the faulty test and operator errors was a disastrous steam explosion in Unit 4 of the RBMK-1000 reactor at Chernobyl. The reactor explosion spread hot pieces of uranium and graphite onto the adjacent buildings.
- Unit 4 has since been entombed and still poses an environmental hazard.

Plant Decommissioning

Chapter Overview

Decommissioning is defined as the safe removal of a facility from service and the reduction of residual radioactivity (radioactivity in structures, materials, soils, groundwater, and other media at a site resulting from activities under the licensee's control) to a level that permits release of the property for unrestricted use OR under restricted conditions (i.e., termination of the NRC license). The regulations require the completion of decommissioning within 60 years of the permanent cessation of operations.

Decommissioning regulations are contained in 10CFR20 Subpart E, as well as various portions of other 10CFR sections.

This chapter will provide a brief discussion of the decommissioning activities associated with a power reactor facility.

Objectives

At the end of this chapter, you will be able to:

- Identify the activities and their associated time requirements involved in decommissioning a site.
- Differentiate among the three methods of decommissioning.

Estimated time to complete this chapter:

35 minutes

Decommissioning Activities

Permanent Cessation of Operations

Once the licensee has made the decision to permanently cease operations, the NRC must be informed in writing within 30 days. This notification must contain the date on which the power generation operations have ceased or will cease. The licensee must remove the fuel from the reactor and submit a written certification to the NRC confirming its action. There is no time limit specified before the fuel must be removed or the certification must be received by the NRC. Once this certification has been submitted, the licensee is no longer permitted to operate the reactor or to put fuel back into the reactor vessel.

PSDAR

The licensee must submit a post-shutdown decommissioning activities report (PSDAR) to the NRC and the affected state(s) no later than 2 years after the date of permanent cessation of operations.

The PSDAR must:

- Describe the planned decommissioning activities
- Contain a schedule for the accomplishment of significant milestones
- Provide an estimate of the expected cost
- Provide documentation that environmental impacts associated with site-specific decommissioning activities have been considered in previously approved environmental impact statements.

Decommissioning Activities

After receiving a PSDAR, the NRC publishes a notice of receipt, makes the PSDAR available for public review and comment, and holds a public meeting in the vicinity of the plant to discuss the licensee's plans. Following completion of the required submittals—and allowing for a 90-day waiting period after submittal of the PSDAR—the licensee may commence major decommissioning activities.

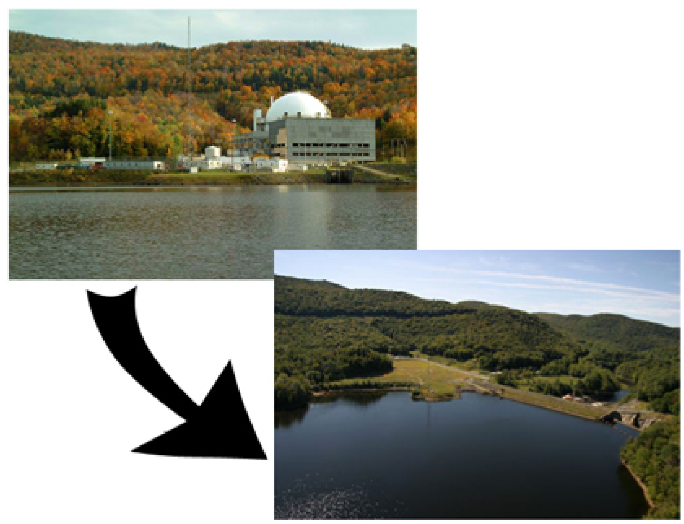
These can include the following:

- Permanent removal of major radioactive components (reactor vessel, steam generators, or other components that are comparably radioactive)
- Permanent changes to the containment structure
- Dismantling components resulting in "greater than Class C" waste

Within 2 years following the date of permanent cessation of operations, the licensee must submit a site-specific cost estimate for the decommissioning project. The licensee is prohibited from using more than 23% of the money that was accumulated during operations for the decommissioning process until the site-specific cost estimate is submitted to the NRC.

Conclusion of Decommissioning Activities

In order to conclude the decommissioning process, the licensee must submit a License Termination Plan to the NRC. This must be submitted at least 2 years before the termination date. It must include the following:



- A site characterization (includes a description of the radiological contamination on the site before any cleanup activities associated with decommissioning took place; a historical description of site operations, spills, and accidents; and a map of remaining contamination levels and contamination locations)
- Identification of remaining dismantlement activities
- Plans for site remediation
- Detailed plans for the final survey of residual contamination on the site
- A description of the end use of the site
- An updated site-specific estimate of remaining decommissioning costs
- A supplement to the environmental report

A licensee may choose to operate an Independent Spent Fuel Storage Installation under either a Part 50 or Part 72 license, and release the remainder of the site under a license termination plan.

Termination Plan

After receiving the license termination plan, the NRC will place a notice of receipt in the Federal Register and will make the plan available to the public for comment. The NRC will schedule a public meeting near the facility to discuss the plan's contents with the public. The NRC will also offer an opportunity for a public hearing on the license amendment associated with the licensee termination plan. If the license termination plan demonstrates that the remainder of decommissioning activities will be performed in accordance with the NRC's regulations, the plan is not detrimental to the health and safety of the public, and it does not have a significant effect on the quality of the environment the Commission will approve the plan by a license amendment (subject to whatever conditions and limitations the NRC deems appropriate and necessary). Once the license amendment is granted, the licensee is authorized to implement the license termination plan.

At the end of the license termination plan process, if the NRC determines that the remaining dismantlement has been performed in accordance with the approved license termination plan, and if the final radiation survey and associated documentation demonstrate that the facility and site are suitable for release, then the Commission will terminate the license or release non-ISFSI portions. At this point the decommissioning process is considered complete.

Methods of Decommissioning

Decommissioning Methods

To decommission a nuclear power plant, the radioactive material on the site must be reduced to levels that would permit removal of the property from the site or termination of the NRC license. This involves removing the spent fuel, dismantling any systems or components containing activation products (such as the reactor vessel and primary systems piping), and cleaning up or dismantling contaminated materials.

The licensee decides how to decontaminate material and the decision is usually based on the amount of contamination, the ease with which it can be removed, and the cost to remove the contamination versus the cost to ship the entire structure or component to a waste-storage site. There are three methods for decommissioning: DECON, SAFSTOR, and ENTOMB.

DECON Method

If the licensee decides on the DECON alternative, the equipment, structures, and portions of the facility and site that contain radioactive contaminants are removed or decontaminated to a level that permits termination of the license shortly after cessation of operations.

The advantages of this method are:

• The facility license is terminated quickly and the facility and site become available for other purposes

• Availability of the operating reactor personnel who are highly knowledgeable about the facility

• Elimination of the need for long-term security, maintenance, and surveillance of the facility; required for the other decommissioning alternatives

• Greater certainty about the availability of low-level waste facilities that would be willing to accept the low-level radioactive waste

• Lower estimated costs compared to the alternative of SAFSTOR; largely as a result of future price escalation because most activities that occur during DECON would also occur during the SAFSTOR period, only at a later date

DECON Disadvantages

The DECON method disadvantages include:

• Higher worker and public doses (because there is less benefit from radioactive decay as would occur in the SAFSTOR option)

• A larger initial commitment of money

• A larger commitment of disposal site space than with the SAFSTOR option

• The potential for complications if spent fuel must remain on the site until a federal repository for spent fuel becomes available

SAFSTOR Method

If the SAFSTOR alternative is chosen, the facility is maintained in a safe, stable condition until it is subsequently decontaminated and dismantled to levels permitting license termination. During SAFSTOR, a facility is left intact but the fuel is removed from the reactor vessel and radioactive liquids are drained from systems and components for processing. Some amount of radioactive decay occurs during the SAFSTOR period, thus reducing the quantity of contaminated and radioactive material that must be disposed of during decontamination and dismantlement. The SAFSTOR option includes active decommissioning at the end of the storage period.

The benefits of this method are:

• A substantial reduction in radioactivity as a result of the radioactive decay that results during the storage period

• A reduction in worker dose (as compared to the DECON) alternative

• A reduction in public exposure because of fewer shipments of radioactive material to the low-level site (as compared to the DECON alternative)

• A reduction in the amount of waste disposal space required (as compared to the DECON alternative)

• Lower cost during the years immediately following permanent cessation of operations

• A storage period compatible with the need to store spent fuel onsite

SAFSTOR

The SAFSTOR method disadavantages are:

• A shortage of personnel familiar with the facility at the time of deferred dismantlement and decontamination

• Site is unavailable for alternate uses during the extended storage period

• Uncertainties regarding the availability and costs of low-level radioactive waste sites in the future

• Continuing need for maintenance, security, and surveillance

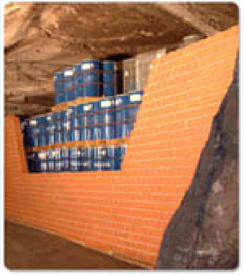
• Higher total cost for the subsequent decontamination and dismantlement period (assuming typical price escalation during the time the facility is stored)

ENTOMB

If the ENTOMB option is chosen radioactive structures, systems, and components are encased in a structurally long-lived substance (e.g., concrete). The entombed structure is appropriately maintained, and continued surveillance is carried out until the radioactivity decays to a level that permits termination of the license.

The benefits of the ENTOMB process relates to the reduced amount of work in encasing the facility in a structurally long-lived substance, thus reducing the worker dose from decontaminating and dismantling the facility. In addition, public exposure from waste transported to the low-level waste site is minimized.

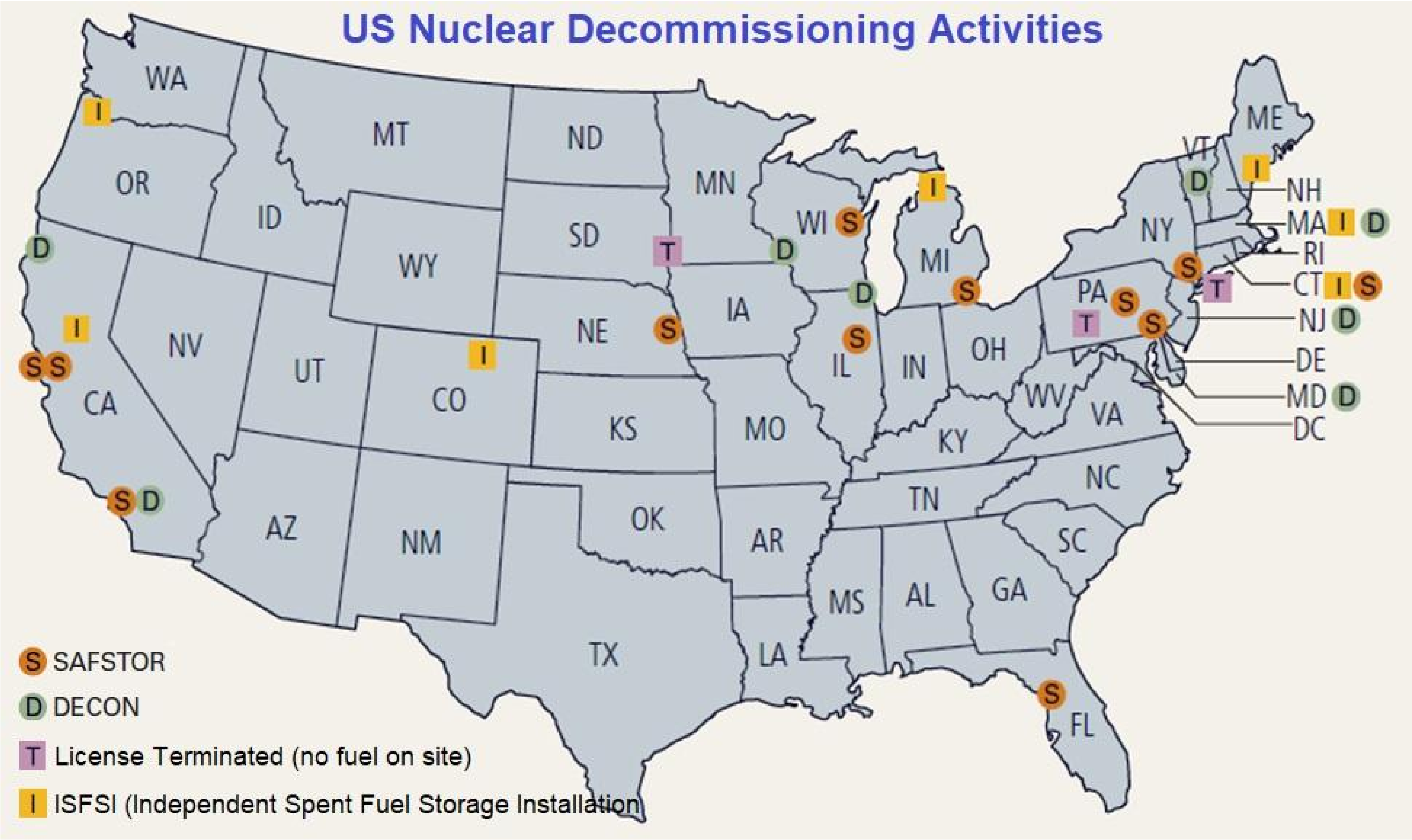
The ENTOMB option may have a relatively low cost. However, because most power reactors will have radionuclides in concentrations exceeding the limits for unrestricted use even after 100 years, ENTOMB is often not feasible. This option might be acceptable if the reactor facilities demonstrate that the radionuclide levels will decay allowing the NRC to grant restricted use of the site. Three small demonstration reactors have been entombed based on this methodology. Currently, no power reactor licensees have proposed the ENTOMB option for any of the power reactors undergoing decommissioning.



Decommission Activities in US

To date (2020), there have been the following decommissioning activities involving US nuclear facilities under the NRC's control:

- 24 reactors in permanent shutdown or decommissioning status
- 7 more reactors plan to cease operations by 2025
- Humboldt Bay, LaCrosse, and Zion 1&2 license terminations expected soon
- 3 research reactors in decommissioning with 2 license terminations expected soon
- 12 Complex Material sites
- 5 Uranium sites in Decommissioning/Remediation
- 28 Mail Tailing Sites in Long-term monitoring



Post License Termination

After the license termination occurs, the NRC issues either an unrestricted or a restricted status for the facility.

Unrestricted use of a facility after license termination means there are no restrictions on how the site may be used. The licensee is free to continue to dismantle any remaining buildings or structures and to use the land or sell the land for any type of application.

Restricted use means that the licensee has demonstrated further reductions in residual radioactivity would result in net public or environmental harm or residual levels are as low as is reasonably achievable, and the licensee made provisions for legally enforceable institutional controls (e.g., restrictions placed in the deed for the property describing what the land can and cannot be used for), which provide reasonable assurance that the radiological criteria set by the NRC will not be exceeded.

Summary

Key Points

- Once the licensee has made the decision to permanently cease operations, the NRC must be informed in writing within 30 days.
- The licensee must submit a post-shutdown decommissioning activities report (PSDAR) no later than 2 years after the date of permanent cessation of operations.
- After the PSDAR is submitted and a 90-day waiting period has passed, the licensee may commence major decommissioning activities.
- Concluding the decommissioning process requires the licensee to submit a License Termination Plan to the NRC.
- After the NRC determines that the remaining dismantlement has been performed in accordance with the approved License Termination Plan, the Commission will terminate the license or release non-ISFSI portions, and the decommissioning process is considered complete.
- There are three alternatives for decommissioning: DECON, SAFSTOR, and ENTOMB.
- If the DECON method is used the equipment, structures, and portions of the facility and site that contain radioactive contaminants are removed or decontaminated to a level that permits termination of the license shortly after cessation of operations.
- If the SAFSTOR method is used the facility is maintained in a safe, stable condition until it is subsequently decontaminated and dismantled to levels permitting license termination.
- If the ENTOMB method is used radioactive structures, systems, and components are encased in a structurally long-lived substance (e.g., concrete).